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

May 15, 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. M82567)

The Commission has issued the enclosed Amendment No. 117 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 January 8, 1992, as supplemented February 26, 1992. By letter dated May 6, 1992, the staff issued to Indian Point Nuclear Generating Unit No. 3, exemptions from 10 CFR 50.46(a)(1)(i), 10 CFR 50.44(a), and Appendix K to 10 CFR Part 50, which allows use of ZIRLO™ clad fuel.

The amendment revises Technical Specifications Section 5.3 (Reactor) and Section 6.9 (Reporting Requirements) to address the use of ZIRLO™, as well as Zircaloy-4, fuel rod cladding.

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. 117 to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

DF01

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

Indian Point Nuclear Generating
Station Unit No. 3

cc:

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DATED: May 15, 1992

AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT UNIT 3

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
N. Conicella
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (4), P-137
Wanda Jones, P-130A
C. Grimes, 11/F/23
F. Orr, 8E23
ACRS (10)
OPA
OC/LFMB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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. 117
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated January 8, 1992, as supplemented February 26, 1992, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 117, 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 prior to loading ZIRLO™ clad fuel into the reactor.

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: May 15, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
vi	vi
5.3-1	5.3-1
6-17	6-17
6-18	6-18
6-19	6-19
6-20	6-20
6-21	6-21
6-22	6-22

5.3 REACTOR

Applicability

Applies to the reactor core, and reactor coolant system.

Objective

To define those design features which are essential in providing for safe system operations.

A. Reactor Core

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 or ZIRLO™ tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods,[1] except during Cycle 8 operation. For Cycle 8 operation only, fuel assembly T53 will contain two stainless steel filler rods in place of two fuel rods.

2. The average enrichment of the initial core was a nominal 2.8 weight percent of U-235. Three fuel enrichments were used in the initial core. The highest enrichment was a nominal 3.3 weight percent of U-235. [2]

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 4.5 weight percent of U-235.

4. Burnable poison rods were incorporated in the initial core. There were 1434 poison rods in the form of 8, 9, 12, 16, and 20-rod clusters, which are located in vacant rod cluster control guide tubes. [3] The burnable poison rods consist of borosilicate glass clad with stainless steel. [4] Burnable poison rods of an approved design may be used in reload cores for reactivity and/or power distribution control.

2b. T. M. Anderson to K. Kneil (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.
(Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.
(Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

3a. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION," February 1982 (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor.)

3b. WCAP-9561-P-A ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS - SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL," July 1986 (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor.)

3c. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987, (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor.)

3d. WCAP-10054-P-A, "SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE," (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

3e. WCAP-10079-P-A, "NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety limits are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator-Region 1 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification;

a. Sealed source leakage on excess of limits (Specification 3.9)

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Seismic event analysis (Specification 4.10)

d. Inoperable plant vent sampling, main steam line radiation monitoring or effluent monitoring capability (Table 3.5-4, items 5, 6 and 7)

e. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

f. Inoperable fire protection and detection equipment (Specification 3.14)

g. Release of radioactive effluents in excess of limits (Appendix B Specifications 2.3, 2.4, 2.5, 2.6)

h. Inoperable containment high-range radiation monitors (Table 3.5-5, Item 24)

i. Radioactive environmental sampling results in excess of reporting levels (Appendix B Specification 2.7, 2.8, 2.9)

j. Operation of Overpressure Protection System (Specification 3.1.A.8.c)

k. Operation of Toxic Gas Monitoring Systems (Specification 3.3.H.3.)

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspection, repair and replacements of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all source material of record.

i. Records of reactor tests and experiments.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transient cycles.

g. Records of training and qualifications for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the SRC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of secondary water sampling and water quality.

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and records showing that these procedures were followed.

o. Records of service lives of all safety-related hydraulic snubbers including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION AND RESPIRATORY PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure as to maintain exposures as far below the limits specified in 10 CFR Part 20 as reasonably achievable. Pursuant to 10 CFR 20.103, allowance shall be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this plant in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1 of 10 CFR 20.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203 (c) (2) of 10-CFR 20, each high radiation area in which the intensity of radiation is 1000 mrem/hr or less and 100 mrem/hr or greater shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

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* Health Physics Personnel shall be exempt from the RWP issuance requirements for entries into high radiation areas during the performances of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

6.12.2 The requirements of 6.12.1 above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the plant Radiological and Environmental Services Manager or his designee.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 Environmental qualification of electric equipment important to safety shall be in accordance with the provisions of 10 CFR 50.49. Pursuant to 10 CFR 50.49, Section 50.49 (d), the EQ Master List identifies electrical equipment requiring environmental qualification.

6.13.2 Complete and auditable records which describe the environmental qualification method used, for all electrical equipment identified in the EQ Master List, in sufficient detail to document the degree of compliance with the appropriate requirements of 10 CFR 50.49 shall be available and maintained at a central location. Such records shall be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 117 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 January 8, 1992, as supplemented February 26, 1992, 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 TS Section 5.3 (Reactor) and Section 6.9 (Reporting Requirements) to address the use of ZIRLO™, as well as Zircaloy-4, fuel rod cladding. ZIRLO™ is a zirconium alloy that the Westinghouse Electric Corporation uses as cladding for its Vantage+ fuel. The February 26, 1992, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The letter dated January 8, 1992, had as an attachment, WCAP-13170, "Safety Assessment for the Indian Point Unit 3 Fuel Assemblies with ZIRLO™ Clad Fuel Rods." This safety assessment, which was performed by Westinghouse, contained the results of an IP3 large break loss-of-coolant accident (LOCA) analysis, and a small break LOCA evaluation performed for a plant similar to IP3. The letter dated February 26, 1992, had as an attachment, a revised safety assessment (WCAP-13170, Revision 1). The revised safety assessment included a small break LOCA analysis which was plant specific for IP3.

2.0 EVALUATION

2.1 Evaluation Criteria

The analyses in WCAP-13170, Revision 1, address the requirements of 10 CFR 50.46 using methodologies complying with 10 CFR Part 50, Appendix K. Application of these criteria to cores containing ZIRLO™-clad fuel was approved in a staff safety evaluation (SE) of October 9, 1991. Exemptions are also identified as needed for application of zircaloy criteria to ZIRLO™-clad fuel. By letter dated May 6, 1992, the staff issued to IP3, exemptions from 10 CFR 50.46(a)(1)(i), 10 CFR 50.44(a), and Appendix K to 10 CFR Part 50, which allows use of ZIRLO™ clad fuel.

2.2 Analysis Methodology

The large break LOCA analyses for IP3 were performed using the approved Westinghouse 1981 Large Break Emergency Core Cooling System (ECCS) Evaluation Model (EM) with BART/BASH. The small break LOCA analyses were performed using the approved Westinghouse NOTRUMP Small Break ECCS Evaluation Model. Application of these models to ZIRLO™-clad fuel was discussed and approved in the October 9, 1991, staff SE.

In applying the models, overall system thermal-hydraulic behavior was calculated assuming the core fueled with zircaloy-clad fuel, and the hot rod fuel temperature calculation using the LOCBART (large break) and LOCTA-IV (small break) codes was performed assuming ZIRLO™-clad fuel. Differences between this application and assuming ZIRLO™-clad fuel for all calculations are insignificant due to the similarity of ZIRLO™ and zircaloy properties as noted in staff SE of July 1, 1991, and October 9, 1991, and the dominant influence of geometry and overall system thermal-hydraulics on calculations for VANTAGE-5 and VANTAGE+ fuel loading combinations. This is consistent with the October 9, 1991, SE finding that a mixed core penalty need not be applied to any mixed core combination of VANTAGE-5 and VANTAGE+ fuel assemblies, if both types of fuel have the same design features. The staff finds this application of the ECCS EMs acceptable.

2.3 Sensitivity Analyses, Spectrum Studies, and Analysis Assumptions

The licensee referenced previous sensitivity analyses, spectrum studies, and limiting analysis assumptions for the VANTAGE-5 fuel resident in the IP3 core as applicable to VANTAGE+ fuel. This reference is consistent with the findings of the October 9, 1991, SE for VANTAGE-5 and VANTAGE+ fuels of like features. The licensee compared the fuel design features discussed in the October 1991 SE for the IP3 VANTAGE-5 and VANTAGE+ fuels to demonstrate the acceptability of reference to the previous analyses. The IP3 VANTAGE-5 and VANTAGE+ fuels are alike in that both have 6-inch natural uranium blankets, optimized fuel rods, reconstitutable top nozzles (RTNs), and debris filter bottom nozzles (DFBNs). Neither fuel type has intermediate flow mixer grids (IFMs). The VANTAGE+ fuel and most of the resident fuel (VANTAGE-5) feature integral fuel burnable absorber (IFBA); however, some of the resident fuel does not have this feature. Therefore, the licensee has considered the fuels both with and without IFBA, and has determined that non-IFBA fuel is limiting. The limiting case analysis assumes non-IFBA fuel. The staff concludes that the licensee has treated both types of fuel alike with respect to the identified fuel features and, therefore, reference to previous VANTAGE-5 sensitivity analyses, spectrum studies, and limiting analysis assumption determinations is acceptable.

Based on burnup sensitivity studies, beginning-of-life (BOL) fuel conditions were assumed for the limiting analysis. The analysis also assumed loss of offsite power and a single failure of one low pressure injection pump. The analysis also assumed a chopped cosine axial power shape. No penalty was assumed for the power shape assumption because the Westinghouse Power Shape Sensitivity Model (PSSM) will be implemented at IP3.

The analyses postulate a double-ended cold leg guillotine (DECLG) rupture with a coefficient of discharge (Cd) of 0.4 as the worst large break LOCA, based on previous analyses. The plant was assumed to be operating with an enthalpy rise hot channel factor (F-delta-h) of 1.62 and a total peaking factor (Fq) of 2.32. The licensee's submittal provided results of analyses of the limiting large break LOCA for both VANTAGE-5 and VANTAGE+ fuels. The calculated peak cladding temperature (PCT) for VANTAGE-5 fuel was 1891 °F, and the PCT for VANTAGE+ fuel is slightly more limiting at 1894 °F.

The submittal also reported the results of several small break LOCA analyses for both VANTAGE-5 and VANTAGE+ fuels. The limiting small break LOCA PCT was 1470 °F for a 6-inch break at BOL with VANTAGE-5 fuel. This is about 400 °F below the large break LOCA PCT spectrum. Therefore, small breaks are not limiting.

2.4 Limiting Case Results

The licensee submitted the results of an analysis of the IP3 limiting case LOCA event assuming operation at 102 percent of the IP3 licensed core power (3025 MWt), and other assumptions and inputs as discussed above. For this case, a DECLG large break LOCA with a Cd of 0.4 and VANTAGE+ fuel, the calculated PCT was 1894 °F, the maximum local metal/water reaction was 4.65 percent, and the total core-wide metal/water reaction was less than 1 percent. These results are within the criteria specified in 10 CFR 50.46(b)(1 through 3) of 2200 °F, 17 percent, and 1 percent, respectively. The results assure that the core will remain amenable to cooling as required by 10 CFR 50.46(b)(4), and the IP3 ECCS design as approved assures continued conformance with the long-term cooling requirement of 10 CFR 50.46(b)(5).

2.5 Technical Specifications Changes

The licensee's submittal contains proposed technical specifications (TS) changes associated with the use of VANTAGE+ fuel. These are:

1. TS Section 5.3.A.1, - add: "or ZIRLO™," i.e., "... in Zircaloy-4 or ZIRLO™ tubing..."

2. TS Section 6.9.1.6 (References) - insert:

"3d. WCAP-10054-P-A, 'SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE,' (W Proprietary)
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor)."

"3e. WCAP-10079-P-A, 'NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE,' (W Proprietary).
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"3f. WCAP-12610, 'VANTAGE+ Fuel Assembly Report,' (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor)."

The staff has reviewed and analyzed the data, methodology, and conclusions of the licensee regarding use of VANTAGE+ fuel and finds they are acceptable. Therefore, the staff finds the proposed TS changes acceptable.

3.0 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.

4.0 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 amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. 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 (57 FR 6041). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). 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.

5.0 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.

Principal Contributor:
F. Orr

Date: May 15, 1992

May 15, 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. M82567)

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The amendment revises Technical Specifications Section 5.3 (Reactor) and Section 6.9 (Reporting Requirements) to address the use of ZIRLO™™, as well as Zircaloy-4, fuel rod cladding.

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. 117 to DPR-64
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

OFFICE	LA:PDI-1	PM:PDI-1	*OGC	D:PDI-1	
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DATE	5/6/92	5/15/92	04/23/92	5/15/92	/ /

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