UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 29, 1994

Docket No. 50-247

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M88463)

The Commission has issued the enclosed Amendment No. 176 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 (TSs) in response to your application transmitted by letter dated December 10, 1993, as supplemented by letter dated August 11, 1994.

The amendment revises TS Section 5.3.A., "Reactor Core," to allow the use of VANTAGE + fuel with ZIRLO cladding and of fuel with filler rods to permit fuel reconstitution. The amendment also revises the Basis for TS Section 2.1, "Safety Limit: Reactor Core," to more accurately describe the basis of the departure from nucleate boiling correlations and how they are applied to ensure that the design criteria are met.

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,

Francis J. Williams, Jr.

Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 176 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Stephen B. Bram
Consolidated Edision Company
of New York, Inc.

Indian Point Nuclear Generating
Station Units 1/2

cc:

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Regional Administrator, Region I
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475 Allendale Road
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DATED: September 29, 1994

AMENDMENT NO. 176 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
NRC & Local PDRs
PDI-1 Reading
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J. Calvo, 14/A/4
M. Boyle
C. Vogan
F. Williams
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
J. Menning
ACRS (10)
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OC/LFDCB
PD plant-specific file
C. Cowgill, Region I
T. Collins

cc: Plant Service list

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 176
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of of New York, Inc. (the licensee) dated December 10, 1993, and supplemented on August 11, 1994, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 176, 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 startup from the next refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael J. Case

Michael J. Case, Acting 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: September 29, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 176

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	-
5.3-1	5.3-1
5.3-2	5.3-2

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT: REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, and coolant temperature during four-loop and three-loop operation, and reactor coolant flow during four-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specifications

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot region of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters: thermal power, reactor coolant temperature and pressure have been related to DNB through correlations which have been developed to predict the DNB flux and location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB thermal design criterion is that the probability of DNB not occurring on the most limiting rod is at least 95 percent (at a 95 percent confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in operating parameter, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analyses limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show the loci of points of thermal power Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. These curves are based on a peak nuclear hot channel factor as stated in the Core Operating Limits Report (COLR) and a 1.55 cosine axial power shape.

5.3 REACTOR

Applicability

Applies to the reactor core, reactor coolant system, and emergency core cooling systems.

Objective

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

A. REACTOR CORE

1. The core shall contain 193 fuel assemblies. Each fuel assembly shall consist of 204 Zircaloy-4 or ZIRLO clad fuel rods. Limited substitutions of Zircaloy-4, ZIRLO, or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, may be used. Fuel assembly configurations shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or cycle-specific reload analyses to comply with all fuel safety design basis. Each fuel rod shall have a nominal active fuel length of 144 inches. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

2. Deleted

3. The enrichment of reload fuel will be no more than 5.0 weight percent U-235 and will be stored in accordance with Technical Specification 5.4.

4. Deleted

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with stainless steel (1).

B. REACTOR COOLANT SYSTEM

1. The design of the reactor coolant system complies with the code requirements (2). Design values for system temperature and pressure are 650°F and 2485 psig, respectively.

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no loss of function.

3. The nominal liquid volume of the reactor coolant system, at rated operating conditions, and with 0% Steam Generator tube plugging is 11,350 cubic feet.

References

(1) UFSAR Section 3.2

(2) UFSAR Table 4.1-9

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 176 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

1.0 INTRODUCTION

By letter dated December 10, 1993, as supplemented by letter dated August 11, 1994, the Consolidated Edison Company of New York (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 Technical Specifications (TSs). The requested changes would revise TS Section 5.3.A., "Reactor Core," to allow the use of VANTAGE + fuel with ZIRLO cladding and of fuel with filler rods to permit fuel reconstitution. The amendment would also revise the Basis for TS Section 2.1, "Safety Limit: Reactor Core," to more accurately describe the basis of the departure from nucleate boiling (DNB) correlations and how they are applied to ensure that the design criteria are met. The August 11, 1994, submittal provided a revised TS page to incorporate a change resulting from the issuance of Amendment No. 173 and also provided a change in the wording of the Basis. It did not change the initial proposed no significant hazards consideration and was not outside the scope of the original Federal Register notice.

The licensee plans to utilize Westinghouse 15 X 15 VANTAGE + fuel. The VANTAGE + fuel uses ZIRLO as its cladding material rather than Zircaloy-4. The NRC staff documented its acceptance of the use of VANTAGE + fuel in a letter from A. Thadani (NRC) to S. Tritch (Westinghouse) dated July 1, 1991. The staff's approval was limited to a rod-average burnup of 60 MWd/kgM and did not include related loss-of-coolant accident (LOCA) analyses methods, which were to be addressed in a separate evaluation report. The licensee has proposed changes to TS Section 5.3.A. to allow the use of VANTAGE + fuel with ZIRLO cladding and to the Basis for TS Section 2.1, "Safety Limit: Reactor Core," to more accurately describe the basis of the DNB correlations and how they are applied to ensure that the design criteria are met.

The licensee has also proposed changes to TS Section 5.3.A. to allow the use of fuel with filler rods to permit fuel reconstitution. Fuel assembly reconstitution involves replacing leaking or damaged fuel rods with filler rods of either ZIRLO, stainless steel, or zirconium alloy. This permits the continued use of fuel assemblies that would otherwise be discharged from the core.

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P PDR

Westinghouse Electric Corporation evaluated the use of reconstituted fuel assemblies as documented in Topical Report WCAP-13060-P.A., "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology." NRC staff approval of this evaluation was documented in a letter from A. Thadani (NRC) to S. Tritch (Westinghouse) dated March 30, 1993. The licensee's submittal dated December 10, 1993, stated that the methodology described in this Topical Report or other approved methodologies will be used for each cycle where reconstituted fuel assemblies are used.

On February 1, 1990, the NRC issued Generic Letter (GL) 90-02 to provide licensees with flexibility in repairing fuel assemblies containing damaged and leaking fuel rods. GL 90-02 included model TSs that permitted licensees to substitute Zircaloy-4 fuel rods with dummy rods or vacant water spaces if the substitution is justified by cycle-specific reload analyses using an NRC-approved methodology. However, the model TSs also provided for fuel configurations which unfortunately were beyond the scope of application for any currently NRC-approved methodologies. On July 31, 1992, the NRC issued Supplement 1 to GL 90-02 to clarify the limitations on applying current NRC-approved analytical methods used in the reconstituted fuel and to revise the previous model TSs to be consistent with realistic reconstitution configurations. The licensee's submittal dated December 10, 1993, proposed changes that are consistent with the model TSs in Supplement 1 to GL 90-02.

2.0 EVALUATION

The licensee has proposed that TS Section 5.3.A.1. be revised to read as follows:

> The core shall contain 193 fuel assemblies. Each fuel assembly shall consist of 204 Zircaloy-4 or ZIRLO clad fuel rods. Limited substitutions of Zircaloy-4, ZIRLO, or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, may be used. Fuel assembly configurations shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or cycle-specific reload analyses to comply with all fuel safety design basis. Each fuel rod shall have a nominal active fuel length of 144 inches. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

The staff finds that the proposed changes to TS 5.3.A.1. are acceptable since they are consistent with the staff's previous approval of the use of VANTAGE + fuel with ZIRLO cladding and with the model TSs in Supplement 1 to GL 90-02. In addition, use of the methodology in the approved report ensures that core configurations are determined consistent with applicable limits in the safety analyses.

The licensee has also proposed that changes be made to TS Sections 5.3.A.5. and 5.3.B.1. and to the References for Section 5.3 to reflect the redesignation of references. The staff finds these changes to be acceptable since they are administrative and required for consistency with the changes to TS Section 5.3.A.1.

The licensee has proposed that the Basis for TS Section 2.1 be changed to more accurately describe the basis of the DNB correlations and how they are applied to ensure that the design criteria are met. The staff has no objections to these Basis changes since they are descriptive and will more accurately describe the DNB methodology used.

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 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 (59 FR 10003). 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.

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:
J. Menning

Date: September 29, 1994

September 29, 1994

Docket No. 50-247

Mr. Stephen B. Bram
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Dear Mr. Bram:

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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 <u>Federal</u> <u>Register</u> notice.

Sincerely,
Original signed by:
Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 176 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

PDI-1:LA	PDI-1:PM	PDI-1:PM	NRR/SRXB	OGC	PDI-1:D
CVogan	JMenning:avl	FWilliams	~~JCollins~~ R Jones	U Young	~~MBoyle~~ MCase
8/19/94	8/19/94	8/19/94	8/29/94	9/23/94	9/21/94

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