

FEB 10 1981

Docket No. 50-247

Mr. John O'Toole
Assistant Vice President
Nuclear Affairs and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 67 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 application transmitted by letter dated February 3, 1981.

The amendment revises the reactor vessel surveillance specimen removal program in the Technical Specifications.

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

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:

1. Amendment No. 67 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page



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no legal objection

OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR#1:DL	OELD	
SURNAME	CParrish	LOlshan;ds	Svarga	TNovak	JW Moore	W. Hazelton
DATE	2/6/81	2/6/81	2/7/81	2/9/81	2/10/81	



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

Docket

February 10, 1981

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Assistant Vice President
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Sincerely,

A handwritten signature in black ink, appearing to read "Steven A. Varga".

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:

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2. Safety Evaluation
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cc: w/enclosures
See next page

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

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. 67
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 February 3, 1981, 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 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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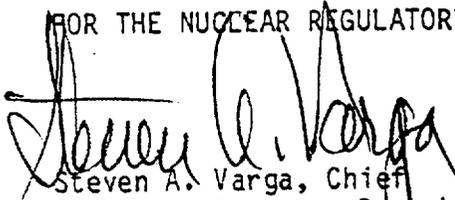
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 67, 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


Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 10, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 67

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages

3.1-4

3.1-8(a)

4.2-15

4.2-27

Insert Pages

3.1-4

3.1-8(a)

4.2-15

4.2-27

B. HEATUP AND COOLDOWN

Specifications

1. The reactor coolant temperature and pressure and system heatup and cooldown rates averaged over one hour (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 for the service period up to 5 effective full-power years. The heatup or cooldown rate shall not exceed 100°F/hr.
 - a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.
 - b. Figure 3.1-1 and Figure 3.1-2 define limits to assure prevention on non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.
2. The limit lines shown in Figure 3.1-1 and Figure 3.1-2 shall be recalculated periodically using methods discussed in WCAP-7924A and results of surveillance specimen testing as covered in WCAP-7323 (7) and as specified in Specification 4.2. The order of specimen removal may be modified based on the results of testing of previously removed specimens. The NRC will be notified in writing as to any deviations from the recommended removal schedule no later than 6 months prior to scheduled specimen removal.
3. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.
4. The pressurizer heatup and cooldown rates averaged over one hour shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.
5. Reactor Coolant System integrity tests shall be performed in accordance with Section 4.3 of the Technical Specifications.

follows that the ΔT induced during cooldown results in a calculated higher allowable K_{IR} for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature, and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculations are provided in WCAP-7924A⁽⁴⁾.

Pressurizer Limits

Although the pressurizer operates at temperature ranges above those for which there is reason for concern about brittle fracture, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition and associated Code Addenda through the Summer 1966 Addendum.

References

- (1) Indian Point Unit No. 2 FSAR, Section 4.1.5
- (2) ASME Boiler & Pressure Vessel Code, Section III, Summer 1965, N-415.
- (3) Indian Point Unit No. 3 FSAR, Section 4.2.5
- (4) WCAP-7924A, "Basis for Heatup and Cooldown Limit Curves", W.S. Hazelton, S.L. Anderson, S.E. Yanichko, April 1975.
- (5) ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, Appendix G.
- (6) ASTM E185-79, Surveillance Tests on Structural Materials in Nuclear Reactors.
- (7) WCAP-7323, "Consolidated Edison Company, Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program", S.E. Yanichko, May 1969.
- (8) Final Report - SWRI Project 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T", E.B. Norris, June 30, 1977.

ITEM 6.6 (CATEGORY K-1) - Integrally-Welded Supports

There are no integrally-welded supports on the valves subject to this examination.

ITEM 6.7 (CATEGORY K-2) - Supports and Hangers

The supports and hangers of the valves subject to this examination shall be visually examined in accordance with Section XI of the code, as shown in Table 4.2-1.

G. Miscellaneous Inspections

ITEM 7.1 - Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods. The examinations scheduled are shown in Table 4.2-1.

ITEM 7.2 - Materials Irradiation Surveillance Specimens

The reactor vessel surveillance program* includes six specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation testing of specimens. The specimens will be removed and examined at the following intervals.

Capsule 1 End of Cycle 1 operation.
Capsule 2 End of Cycle 2 operation.
Capsule 3 End of Cycle 5 operation.
Capsule 4 End of Cycle 8 operation.
Capsule 5 End of Cycle 16 operation.
Capsule 6 Spare

* Refer to FSAR section 4.5, WCAP-7323, and Indian Point Unit No. 2 Application for Amendment to Operating License," sworn to on February 3, 1981.

TABLE 4.2-1 (sheet 11 of 11)

<u>Item No.</u>	<u>Examination Category</u>	<u>Components and Parts to be Examined</u>	<u>Method</u>	<u>Extent of Examination (Percent in 10 Year Interval)</u>	<u>Remarks</u>
6.4	G-1	Pressure-retaining bolting		Not applicable	
6.5	G-2	Pressure-retaining bolting	V	100%	Exception is taken for valves which are not accessible.
6.6	K-1	Integrally-welded supports		Not applicable	
6.7	K-2	Supports and hangers	V	100%	Exception is taken for supports and hangers which are not accessible.
MISCELLANEOUS INSPECTIONS					
7.1		Primary pump flywheel	V & UT	See Remarks	The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods.
7.2		Irradiation Specimen Schedule	Tensile and Charpy V Notch (Wedge Open Loading)	See Remarks	Refer to Examination G (Miscellaneous Inspections) Item 7.2 in the bases of this Specification 4.2.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

Introduction

By letter dated February 3, 1981, Consolidated Edison Company of New York (the licensee) proposed changes to the reactor vessel surveillance specimen removal program in the Technical Specifications for Indian Point Unit No. 2. The licensee, in a letter dated September 24, 1980, committed to send in this application for amendment. The proposed change had been discussed and agreed to by the NRC prior to the September 24, 1980 letter.

Evaluation

The licensee's review of the current version of ASTM E 185-79 indicated that a revision to the original surveillance capsule removal schedule for Indian Point Unit No. 2 would provide more useful irradiation effects data for the reactor vessel. The proposed specimen removal schedule will satisfy the criteria of Table 1 of ASTM E 185-79; and we find it acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the 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 impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

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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 decrease in a safety margin, the amendment 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.

Date: February 10, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSIONDOCKET NO. 50-247CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 67 to Facility Operating License No. DPR-26, issued to the 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 reactor vessel surveillance specimen removal program in the Technical Specifications.

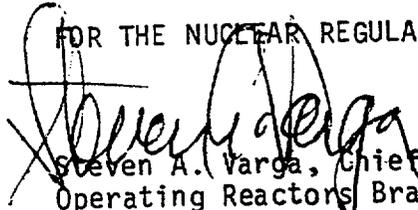
The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter 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 February 3, 1981, (2) Amendment No. 67 to License No. DPR-26, and (3) the Commission's related Safety Evaluation. 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) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 10 day of February, 1981

FOR THE NUCLEAR REGULATORY COMMISSION


Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing