Docket No. 50-247

March 7, 1984

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ORB#1 Rdg
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OELD
LHarmon
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WJones
ACRS (10)
RDiggs

Mr. John D. O'Toole
Vice President
Nuclear Engineering 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. 88 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 in response to your application transmitted by letter dated February 14, 1983.

The amendment modifies the Technical Specifications by relocating the requirements of the reactor vessel surveillance program from the Miscellaneous Inspections of Section 4.2 to the Reactor Coolant System Limiting Conditions for Operation in Section 3.1.B. This is an administrative change to the Technical Specifications.

The staff is currently reviewing a separate Technical Specification change proposed on inservice inspection requirements. If approved, this change would delete most of the surveillance requirements from Section 4.2. The requirements for reactor vessel surveillance (item 7.2 in Section 4.2) will however, remain in the Technical Specifications. The licensee has proposed to separate out these requirements and relocate them in Section 3.1.B. The staff agrees that the reactor vessel surveillance program logically fits into Section 3.1.B. This change in the Technical Specifications is acceptable to the staff since no change to the reactor vessel surveillance requirements is being made.

We have determined that the amendment does not authorize a change in effluent types of 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.

We have 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, and (2) 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.

MAR 7 1984

The February 14, 1983 submittal from the licensee contained requests for Technical Specification changes dealing with several other issues. This amendment addresses only the reactor vessel surveillance program issue. The other change request issues will be the subject of separate licensing actions.

The Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

Original signed by
Steven A. Varga

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

Enclosure:
1. Amendment No. 88 to DPR-26

cc: w/enclosure
See next page

*See other white for concurrences

ORB#1:DL*	ORB#1:DL	C-ORB#1:DL*	OELD*	AD:OR:DL
CParrish	RPedersen;ps	SVarga		GLainas
2/14/84 3/1/84	~~2/ /84~~ 3/1/84	2/14/84	2/24/84	3/ /84

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Sincerely,

Roger L. Pedersen, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosure:
1. Amendment No. to DPR-26

cc: w/enclosure
See next page

ORB#1:DL	ORB#1:DL	C-ORB#1:DL	OELD	AD:OR:DL
CParrish	RPedersen;ps	SVarga	M KARMAN	GLainas
2/14/84	2/14/84	2/ /84	2/21/84	2/ /84

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

March 7, 1984

Docket No. 50-247

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

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

Enclosure:
1. Amendment No. 88 to DPR-26

cc: w/enclosure
See next page

~~Mr. John D. O'Toole~~
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of New York, Inc.

Indian Point Station, Unit ~~1~~
Indian Point Nuclear Generating Unit 2

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

CONSOLIDATED EDISION COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
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 14, 1983, 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 th 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. 88 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 7, 1984

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
	3.1-4a
3.1-6	3.1-6
3.1-8a	3.1-8a
4.2-15	4.2-15
4.2-27	4.2-27

6. The reactor vessel surveillance program** includes six specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation tensile and charpy V notch (wedge open loading) 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.

An approximation of the maximum integrated fast neutron ($E > 1$ Mev) exposure is given by Figure 2-4 of WCAP 7924A[4]. Exposure of the Indian Point Unit No. 2 vessel will be less than that indicated by this figure.

The actual shift in RT_{NDT} will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. These samples are evaluated according to ASTM E185.[6] To compensate for any increase in the RT_{NDT} caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown, in accordance with the requirements of the ASME Boiler & Pressure Vessel Code, 1974 Edition, Section III, Appendix G, and the calculation methods described in WCAP-7924A[4].

The first reactor vessel material surveillance capsule was removed during the 1976 refueling outage. That capsule was tested by Southwest Research Institute (SWRI) and the results were evaluated and reported.[8][9] The second surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by SWRI and the results have been evaluated and reported.[10] Based on the SWRI evaluation, heatup and cooldown curves (Figures 3.1-1 and 3.1-2) were developed for up to seven (7) effective full power years (EFPYs) of reactor operation.

The maximum shift in RT_{NDT} after 7 EFPYs of operation is projected to be 130°F at the 1/4T and 65°F at the 3/4T vessel wall locations, per Plate B2002-3 the controlling plate. The initial value of RT_{NDT} for the IP2 reactor vessel was 60°F based on Plates B2002-1 and B2002-3 as shown in Table 3.1-1. The heatup and cooldown curves for 7 EFPYs have been computed on the basis of the RT_{NDT} of Plate B2002-3 because it is anticipated that the RT_{NDT} of the reactor vessel beltline material will be highest for Plate B2002-3 at least through that time period.

Heatup and Cooldown Curves

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non Mandatory Appendix G in Section III 1974 Edition of the ASME Boiler and Pressure Vessel Code and discussed in detail in WCAP-7924A.[4]

The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the 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 No. 02-4531 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, June 30, 1977.

(9) Supplement to Final Report - SWRI Project No. 02-4531- "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule T," E.B. Norris, December 1980.

(10) Final Report - SWRI Project No. 02-5212 - "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule Y," E.B. Norris, November 1980.

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.

TABLE 4.2-1 (sheet 11 of 11)

Item No.	Examination Category	Components and Parts to be Examined	Method	Extent of Examination (Percent in 10 Year Interval)	Remarks
6.4	G-1	Pressure-retaining bolting		Not applicable	
6.5	G-2	Pressure-retaining bolting	V	100%	Exception is taken for valv(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 fl(wheel shall be examined by ultrasonic methods.