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. 174
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 January 28, 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. 174 , 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael L. Boyle

Michael L. Boyle, 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: July 29, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 174

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
4.1-1	4.1-1
4.4-3	4.4-3
4.4-4	4.4-4
4.4-9	4.4-9

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Interval Extension

Unless otherwise noted, each surveillance requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified interval. Excluded from this provision are the following surveillances whose intervals are solely defined by the applicable Technical Specification paragraphs and cannot be extended.

4.4A Integrated Leakage Rate

4.4B Sensitive Leakage Rate

4.13 Steam Generator Tube Inservice Inspection.

Basis

Specification 4.0.1 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are specified to be performed at least once each Refueling Interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed once each Refueling Interval. Likewise, it is not the intent that Refueling Interval surveillances be performed during

3. Frequency

A sensitive leakage rate test shall be performed at a frequency of at least every other refueling but in no case at intervals of greater than 3 years.

C. AIR LOCK TESTS

1. The containment air locks shall be tested at a minimum pressure of 47 psig and at a frequency of every 6 months. The acceptance criteria is included in Specification 4.4.D.2.a.

2. Whenever containment integrity is required, verification shall be made of proper repressurization to at least 47 psig of the double-gasket air lock door seal upon closing an air lock door.

D. CONTAINMENT ISOLATION VALVES

1. Tests and Frequency

a. Isolation valves in Table 4.4-1 shall be tested for operability at every Refueling Interval (R#).

b. Isolation valves in Table 4.4-1 which are pressurized by the Weld Channel and Penetration Pressurization System shall be leakage tested as part of the Weld Channel and Penetration Pressurization System Test at every Refueling Interval (R#).

c. Isolation valves in Table 4.4-1 which are pressurized by the Isolation Valve Seal Water System shall be tested at every Refueling Interval (R#) as part of an overall Isolation Valve Seal Water System Test.

d. Isolation valves in Table 4.4-1 which are not pressurized will be tested at every Refueling Interval (R#).

e. Isolation valves in Table 4.4-1 shall be tested with the medium and at the pressure specified therein.

2. Acceptance Criteria

a. The combined leakage rate for the following shall be less than 0.5 L_a: isolation valves listed in Table 4.4-1 subject to gas or nitrogen pressurization testing, air lock testing as specified in Specification 4.4.C.1, portions of the sensitive leakage rate test described in Specification 4.4.B.1 which pertain to containment penetrations and double-gasketed seals.

b. The leakage rate into containment for the isolation valves sealed with the service water system shall not exceed 0.36 gpm per fan cooler.

c. The leakage rate for the Isolation Valve Seal Water System shall not exceed 14,700 cc/hr.

3. Containment isolation valves may be added to plant systems without prior license amendment to Table 4.4-1 provided that a revision to this table is included in a subsequent license amendment application.

E. CONTAINMENT MODIFICATIONS

Any major modification or replacement of components of the containment performed after the initial pre-operational leakage rate test shall be followed by either an integrated leakage rate test or a local leak detection test and shall meet the appropriate acceptance criteria of Specifications 4.4.A.2, 4.4.B.2, or 4.4.D.2. Modifications or replacements performed directly prior to the conduct of an integrated leakage rate test shall not require a separate test.

The testing of containment isolation valves in Table 4.4-1, either individually or in groups, utilizes the WC & PPS[4] or IVSWS[5] where appropriate and is in accordance with the requirements of Type C tests in Appendix J (issue effective date March 16, 1973) to 10 CFR 50, except for the surveillance frequency. The 25% increase in surveillance frequency allowed (from a maximum of 24 months to a maximum of 30 months) was compensated for by a proportionate increase in the margin between the specified allowable leakage and the maximum allowable leakage (the specified allowable leakage was decreased from 0.6 L_a to 0.5 L_a). The specified test pressures are ≥ the peak calculated accident pressure. Sufficient water is available in the Isolation Valve Seal Water System, Primary Water System, Service Water System, Residual Heat Removal System, and the City Water System to assure a sealing function for at least 30 days. The leakage limit for the Isolation Valve Seal Water System is consistent with the design capacity of the Isolation Valve Seal Water supply tank.

The acceptance criterion of 0.5 L_a for the combined leakage of isolation valves subject to gas or nitrogen pressurization, the air lock, containment penetrations and double-gasketed seals accounts for possible degradation of the containment leakage barriers for a 30 month test interval.

The 350 psig test pressure, achieved either by normal Residual Heat Removal System operation or hydrostatic testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the containment sump return line of 100 psig gives an adequate margin over the highest pressure within the line after a design basis accident. A recirculation system leakage of 2 gal./hr. will limit offsite exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident.

These specifications have been developed using Appendix J (issue effective date March 16, 1973) of 10 CFR 50 and ANSI N45.4-1972 "Leakage Rate Testing of Containment Structures for Nuclear Reactors" (March 16, 1972) for guidance.