April 9, 1997

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

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M97976)

Dear Mr. Quinn:

The Commission has issued the enclosed Amendment No. 189 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 February 14, 1997, as supplemented by letter dated March 12, 1997.

The amendment revises Technical Specification Section 4.13-2 to allow a one-time extension of the interval for steam generator tube inspection. Specifically, the date for commencement of the steam generator tube inspection is extended from April 14, 1997 to May 2, 1997.

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:

Jefferey F. Harold, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

DF01

Docket No. 50-247

Enclosures: 1. Amendment No. 189 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

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

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Sincerely,

Jeffery F. Harold

Jefferey F. Harold, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-247

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cc w/encls: See next page

Stephen E. Quinn
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station Units 1/2

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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. 189
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, 1997, as supplemented March 12, 1997, 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:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 189, 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 by April 14, 1997.

FOR THE NUCLEAR REGULATORY COMMISSION

SBajwa

S. Singh Bajwa, 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: April 9, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 189

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
4.13-1 thru 4.13-7	4.13-1 thru 4.13-7

4.13 STEAM GENERATOR TUBE INSERVICE SURVEILLANCE

Applicability

Applies to inservice surveillance of the steam generator tubes.

Objective

To assure the continued integrity of the steam generator tubes that are a part of the primary coolant pressure boundary.

Specifications

Steam generator tubes shall be determined operable by the following inspection program and corrective measures.

A. INSPECTION REQUIREMENTS

1. Definitions

a. Imperfection is a deviation from the dimension, finish, or contour required by drawing or specification.

b. Deformation is a deviation from the initial circular cross-section of the tubing. Deformation includes the deviation from the initial circular cross-section known as denting.

c. Degradation means service-induced cracking, wastage, pitting, wear or corrosion (i.e., service-induced imperfections).

d. Degraded Tube is a tube, or sleeved tube, that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.

e. % Degradation is an estimated % of the tube or sleeve wall thickness affected or removed by degradation.

f. Defect is a degradation of such severity that it exceeds the plugging limit. A tube or sleeve containing a defect is defective.

g. Plugging Limit is the degradation depth at or beyond which the tube must be plugged or repaired.

h. Hot-Leg Tube Examination is an examination of the hot-leg side tube length. This shall include the length from the point of entry at the hot-leg tube sheet around the U-bend to the top support of the cold leg.

I. Cold-Leg Tube Examination is an examination of the cold-leg side tube length. This shall include the tube length between the top support of the cold leg and the face of the cold-leg tube sheet.

j. F* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.25 inches and is measured down from the bottom of the roll transition.

k. F* Tube is a tube:

   a) With degradation equal to or greater than 40% below the F* distance, and b) which has no indication of degradation within the F* distance, and c) that remains in service.

l. Sleeving refers to tube repair achieved by laser welded sleeving, as described by Westinghouse Report WCAP-13583 and 13088. Sleeving is used to maintain a tube in service or return a previously plugged tube to service.

2. Extent and Frequency of Examination

   a. Steam generator examinations shall be conducted not less than 12 months nor later than twenty four calendar months after the previous examination.*

   b. Scheduled examinations shall include each of the four steam generators in service.

* Examinations scheduled for 1997 only, shall be conducted during the 1997 Refueling Outage which will commence no later than May 2, 1997. The scheduled examinations will be completed prior to return to service from the 1997 Refueling Outage.

c. Unscheduled steam generator examinations shall be required in the event there is a primary to secondary leak exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steamline or feedwater line break.

d. Unscheduled examinations may include only the steam generator(s) affected by the leak or other occurrence.

e. In case of an unscheduled steam generator examination, the profilometry tensile strain criterion shall be the same as contained in the approved program for the last scheduled steam generator inspection.

3. Basic Sample Selection and Examination

a. At least 12% of the tubes in each steam generator to be examined shall be subjected to a hot-leg examination.

b. At least 25% of the tubes inspected in Specification 4.13.A.3.a above shall be subjected to a cold-leg examination.

c. Tubes selected for examination shall include, but not be limited to, tubes in areas of the tube bundle in which degradation has been reported, either at Indian Point 2 in prior examinations, or at other utilities with similar steam generators.

d. Examination for deformation ("dents") shall be either by eddy current or by profilometry.

e. Examination for degradation other than deformation shall be by eddy current techniques, using a 700-mil diameter probe. If the 700-mil diameter probe cannot pass through the tube, a 610-mil diameter probe shall be used. For examination of the U-bends and cold-legs of tubes in rows 2 through 5, a 540-mil diameter probe may be used, provided it is justified by profilometry measurement within the tensile strain criterion.

f. In addition to the minimum sample size as determined by Table 4.13-1, all F* tubes shall be inspected within the pertinent tubesheet region. The results of F* tube inspections are not to be utilized as a basis for additional inspections per Table 4.13-1.

4. Additional Examination Criteria

1. Degradation Not Caused by Denting

a. If 5% of more of the tubes examined in a steam generator exhibit degradation or if any of the tubes examined in a steam generator are defective, additional examinations shall be required as specified in Table 4.13-1.

b. Tubes for additional examination shall be selected from the affected area of the tube array and the examination may be limited to that region of the tube where degradation or defective tube(s) were detected.

c. The second and third sample inspections in Table 4.13-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

2. Degradation Caused by Denting

a. Additional examinations, for degradation caused by denting, shall be performed as described in the most recent steam generator examination program approved by the NRC.

B. ACCEPTANCE CRITERIA AND CORRECTIVE ACTION

1. Tubes shall be considered acceptable for continued service if:

a. depth of degradation is less than:

- 40% of the tube wall thickness, or
- 23% of the sleeve wall thickness

AND

b. the tube will permit passage of a 0.540" diameter probe and the strain in the tube wall (if measured) is less than the tensile strain criterion as specified in the approved examination program, or the tube will permit passage of a 0.610" diameter probe in the absence of strain measurement.

c. the tube is an F* tube and meets a. and b. above the F* region.

2. Tubes or sleeves that are not considered acceptable for continued service shall be plugged or repaired.

C. REPORTS AND REVIEW AND APPROVAL OF RESULTS

1. The proposed steam generator examination program shall be submitted for NRC staff review and concurrence at least 60 days prior to each scheduled examination.

2. The results of each steam generator examination shall be submitted to NRC within 45 days after the completion of the examination. A significant increase in the rate of denting or significant change in steam generator condition shall be reportable immediately.

3. An evaluation which addresses the long term integrity of small radius U-bends beyond row 1 shall be submitted within 60 days of any finding of significant hour-glassing (closure) of the upper support plate flow slots.

4. Restart after the scheduled steam generator examination need not be subject to NRC approval.

Basis

Inservice examination of steam generator tubing is essential if there is evidence of mechanical damage or progressive deterioration in order to assure continued integrity of the tubing. Inservice examination of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

An essentially 100% tube examination was performed on each tube in each steam generator by eddy current techniques prior to service in order to establish a baseline condition for the tubing. No significant baseline imperfections were identified. In addition, prior to the discontinuance of phosphate treatment and the institution of all-volatile treatment (AVT), a baseline inspection was conducted in March, 1975 before the resumption of power operation.

Wastage-type defects are unlikely with the all-volatile treatment (AVT) of secondary coolant; however, even if this type of defect occurs, the steam generator tube examination will identify tubes with significant degradation from this effect.

The results of steam generator tube burst and collapse tests have demonstrated that tubes having wall thickness of not less than 0.025 inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents. An allowance of 10% for tube degradation that may occur between inservice tube examinations added to the 40% degradation depth provided in the acceptance criteria provides an adequate margin to assure that tubes considered acceptable for continued operation will not have a minimum tube wall thickness of less than the acceptable 50% of normal tube wall thickness (i.e. 0.025 inch) during the service life-time of the tubes. Steam generator tube examinations of other operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050 inch wall thickness.

Examination of samples of tubes and support plates removed from steam generators have revealed that "denting" is caused by the accretion of steel corrosion products in the tube/support plate annuli. As these corrosion products are more voluminous than the support plate material from which they are derived, a compressive force is exerted on the tubes in the plane of the support plates, resulting in deformation of the tubes. If the deformation results in an ovalization of the tubes, the resulting strain is low and there is no risk of development of stress corrosion cracking in the tubes. However, if the deformation results in an irregular tube shape, the resulting strain may be high enough for the tube to become susceptible to stress corrosion cracking inservice, and it should be preventively repaired. Beginning with the steam generator examination to be conducted during the Cycle 5/6 Refueling Outage, the tensile strain criterion for profilometry shall be 25%. The 25% strain criterion is based on a review of data currently available from operating steam generators, and will be revised as necessary as more experience is gained with the evaluation of this measurement. In the future, this criterion may be revised, either higher or lower, based on steam generator examination results. The profilometry criterion to be used for any steam generator examination shall be established in the most recent program approved by NRC.

A first report on the R&D work leading to the development of profilometry, entitled "Profilometry of Steam Generator Tubes" dated August, 1980, was forwarded to the NRC by Con Edison. Additional R&D work has improved the accuracy of the profilometer and the calculation of strain in a deformed tube.

Before the development of profilometry, a minor diameter of 0.610" was established as the criterion for continuing a tube inservice. This criterion was used successfully for several years at Indian Point Unit 2 and at other plants, and appears to be sufficiently conservative so that it can be continued in the absence of more accurate strain determination by means of profilometry.

A sound roll expansion throughout the F* distance provides a tube to tubesheet interface that ensures the requirements of Regulatory Guide 1.121 are met regardless of the severity of any tube degradation below the F* distance. The F* distance of 1.25 inches is comprised of 1.01 inches of sound roll that ensures tube integrity requirements are met plus 0.24 inches which allows for eddy current measurement uncertainty. The testing and analysis supporting the F* distance is documented in B&W Nuclear Technologies Qualification Report No. BAW-10195P.

Testing performed as documented in BAW-10195 P demonstrates the maximum postulated leakage under accident conditions for repair of 100% of the tube ends using the F* criteria is well below the allowable leakage limits for Indian Point 2 steam generators. If, in the future, steam generator tubes are allowed to remain in service by the use of F* and, in addition, other tube acceptance criteria, then the aggregate maximum postulated accident leakage must be below the allowable leakage limits for Indian Point 2 steam generators.

This program for inservice inspection of steam generator tubes exceeds the requirements of Regulatory Guide 1.83, Revision 1, dated July 1975.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 189 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 February 14, 1997, as supplemented by letter dated March 12, 1997, Consolidated Edison Company of New York, Inc. (the licensee) submitted a proposed amendment to the Technical Specifications (TSs) for Indian Point Unit 2 (IP2). The licensee requested to extend the interval for steam generator tube inspection specified in TS 4.13.A.2.a. The licensee proposed to begin the tube inspection on May 2, 1997, and complete the inspection before restart from the scheduled refueling outage. The March 12, 1997, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration.

2.0 EVALUATION

The proposed amendment would permit a one-time extension of the current steam generator tube inservice inspection cycle. The licensee proposed to attach the following footnote to TS 4.13.A.2.a: "Examination scheduled for 1997 only, shall be conducted during the 1997 Refueling Outage which will commence no later than May 2, 1997. The scheduled examinations will be completed prior to return to service from the 1997 Refueling Outage."

Technical Specification 4.13.A.2.a requires that "steam generator tube examinations shall be conducted not less than 12 months nor later than twenty-four calendar months after the previous examination." The previous steam generator tube examination was completed on April 14, 1995, after operating for 598 effective full-power days at full power. On the basis of the above requirement, the steam generator tube inspection for the 1997 outage must be completed no later than April 14, 1997.

Although the steam generator inspection was completed on April 14, 1995, the unit did not restart until May 2, 1995. In addition, IP2 was shut down in early 1997 for a 49-day maintenance outage. When the reactor is shut down and the reactor coolant system is at a reduced temperature, the steam generators are not subject to the conditions that lead to tube degradation. Therefore, that the actual number of days that the steam generators will be allowed to be subjected to an environment conducive to tube degradation is not being increased. Since no additional steam generator service time is involved, the staff finds the proposed one time TS change acceptable.

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2.1 Conclusion

Based on the review of information submitted, the staff concludes that the licensee's proposed one-time extension of the interval for steam generator tube inspection is 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 amendments change 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 dtermined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 9816). Accordingly, the amendments meet 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 of 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. Tsao

Date: April 9, 1997

DATED: April 9, 1997

AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
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