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Docket No. 50-247

Consolidated Edison Company of
New York, Inc.
ATTN: Mr. William J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

Gentlemen:

The Commission has issued the enclosed Amendment No. 40 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment results from your steam generator inspection report dated March 24, 1978, as supplemented by letter dated May 4, 1978, and provides additional requirements for inspection of the steam generators and for steam generator leakage limits as discussed with and agreed to by your staff.

You are requested to submit the details of the next steam generator inspection which you plan for Indian Point Unit No. 2. These details should be submitted no later than 30 days prior to the date you expect the next inspection to commence.

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

Sincerely,

/s/

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 40 to DPR-26
2. Safety Evaluation
3. Notice

cc w/encl:
See next page

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JSaltzman
BHarless
CMiles
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JRBuchanan
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 12, 1978

Docket No. 50-247

Consolidated Edison Company of
New York, Inc.
ATTN: Mr. William J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

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The Commission has issued the enclosed Amendment No. 40 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment results from your steam generator inspection report dated March 24, 1978, as supplemented by letter dated May 4, 1978, and provides additional requirements for inspection of the steam generators and for steam generator leakage limits as discussed with and agreed to by your staff.

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

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 40 to DPR-26
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3. Notice

cc w/encl:
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. 40
License No. DPR-26

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

   A. The facility will operate in conformity with the provisions of the Atomic Energy Act, of 1954, as amended, and the rules and regulations of the Commission;

   B. 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;

   C. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   D. The issuance of this amendment is in accordance with 10 CFR Part 51 of tne 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 paragraphs 2.C.(2) and 2.D of Facility Operating License No. DPR-26 are hereby amended to read as follows:

   "2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 40 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

"2.D Steam Generator Inspections

In order to perform an inspection of all four steam generators, the plant shall be brought to the cold shutdown condition within sixteen equivalent months of operation from May 12, 1978, but in any event, no later than December 1, 1979. For the purpose of this requirement, equivalent operation is defined as operation with a primary coolant temperature greater than 350°F. Nuclear Regulatory Commission approval shall be obtained before resuming power operation following this inspection."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 12, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 40

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove	Insert
3.1-17	3.1-17 & 3.1-17a
3.1-20	3.1-20

Changed areas on the revised pages are shown by marginal lines.

F. LEAKAGE OF REACTOR COOLANT

Specification

1. If leakage of reactor coolant is indicated by the means available such as water inventory balance, monitoring equipment or direct observation, a follow up evaluation of the safety implications shall be initiated as soon as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that either (1) a safety problem does not exist or (2) that the indicated leak cannot be substantiated by direct observation or other indication.

2. If the indicated leakage is substantiated and is evaluated as unsafe or is determined to exceed 10 GPM, reactor shutdown shall be initiated as soon as practicable but no later than within 24 hours after the leak was first detected.

3. Whenever the reactor is shutdown, or a steam generator removed from service, in order to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube, Con Ed shall inform the NRC before any tube is plugged or, if no tube is plugged, before the steam generator is returned to service.

4. Primary to secondary leakage through the steam generator tubes shall be limited to 0.3 gpm per steam generator. With any steam generator tube leakage greater than this limit the reactor shall be brought to the cold shutdown condition within 24 hours.

5. If leakage attributable to the tube denting phenomena from two or more tubes in the steam generators in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24 hours and Nuclear Regulatory Commission aporoval shall be obtained before resuming reactor

operation or if two steam generator tube leaks attributable to the tube denting phenomena are observed after the reactor is in cold shutdown Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

6. The nature of the leak as well as the magnitude of the leak shall be considered in the safety evaluation. If plant shutdown is necessary per specification 2 above, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable. The safety evaluation shall assure that the exposure to offsite personnel to radiation from the primary system coolant activity is within the guidelines of 10 CFR 20. The reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

7. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different principles shall be in operation, with one of the two systems sensitive to radioactivity. The system sensitive to radioactivity may be out-of-service for 48 hours provided two other systems are available.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory balances.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by limitation of steam generator leakage between the reactor coolant system and the secondary coolant system. Leakage in excess of 0.3 gpm for any steam generator will require plant shutdown and the leaking tubes will be located and plugged.

References

FSAR Sections 11.2.3 and 14.2.4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 40 TO OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT UNIT NO. 2

DOCKET NO. 50-247

Background

On May 13, 1977, following the April 1977 steam generator inspections at Indian Point Unit No. 2, the units operating license was amended to allow twelve equivalent months of operation* before the next steam generator inspection. The April 1977 steam generator inspection included eddy current testing and gauging of tubing in steam generators 21 and 22 and support plate flow slot inspections in all four steam generators. A steam generator inspection program for Indian Point Unit No. 2 was also conducted in June and November 1976, which included eddy current testing of tubing and flow slot examination in all four steam generators.

On January 11, 1978, the licensee submitted the details of a steam generator inspection to be performed during the unit's Spring 1978 refueling outage. After receiving the Nuclear Regulatory Commission's (NRC's) comments on the proposed inspection plan and discussing them with the NRC staff, a revised inspection program was agreed upon. Steam generator inspections were conducted in accordance with this program during the refueling outage which began on February 13, 1978.

Discussion

The inspection performed included eddy current testing and gauging of tubing in steam generators 23 and 24 and flow slot and support plate examinations in all four steam generators. A standard 700 mil probe was used to perform the eddy current testing. Any tube which restricted passage of the 700 mil probe was probed with successively smaller diameter probes to quantify the magnitude of the restriction. Five different size probes (700 mil, 675 mil, 640 mil, 610 mil and 540 mil) were used. In addition, the tubes immediately adjacent to any tube that did not pass the 610 mil probe were also subjected to eddy current inspection for dents.

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*Equivalent operation for this purpose, is defined as operation with a primary coolant temperature greater than 350°F.

The eddy current inspection included 463 tubes or 14.6 percent of all the unplugged tubes in steam generator 23. All of these tubes were inspected for defects and 444 of the tubes were eddy current inspected for dents. In steam generator 24, 473 tubes, or 15.0 percent of the unplugged tubes were inspected. All of these tubes were eddy current inspected for dents and 283 of the tubes were also inspected for defects. The eddy current inspection included tubes in hard spot areas in rows two and three of the hot leg, all peripheral hard spot areas, and the two rows adjacent to the patch plate seam in steam generators 23 and 24. Row 1 tubes were plugged in all four steam generators when modifications were made to the channel head divider plates during the construction phase prior to plant operation.

The results of the denting inspections revealed an average dent magnitude of 2.5 mils in steam generator 23 and 2.0 mils in steam generator 24. Forty-seven (47) tubes in steam generator 23 and 70 tubes in steam generator 24 did not permit passage of the standard 700 mil eddy current probe. Using successively smaller probes, the size of the restrictions was quantified as shown in the following table.

Steam Generator 23:

Maximum Probe Size Passed by Tube	Number of "Restricted" Tubes Passing This Probe
675 mil	29
640 mil	15
610 mil	1
540 mil	1
<540 mil	1

Steam Generator 24:

Maximum Probe Size Passed by Tube	Number of "Restricted" Tubes Passing This Probe
675 mil	44
640 mil	17
610 mil	3
540 mil	3
<540 mil	3

These tubes were located primarily in the peripheral hard spot areas with no tube along the flow slots restricting passage of a probe smaller than 675 mils. Any tube found during the inspection that did not allow passage of the 610 mil probe was plugged. This resulted in the plugging of two tubes in steam generator 23 and six tubes in steam generator 24. All of the tubes were located in the peripheral hard spot areas.

Inspections of the lowest tube support plate flow slots were performed in all steam generators during the June 1976 steam generators inspection and in steam generators 21 and 22 during the April 1977 inspection. The average increase in flow slot closure during the 13 equivalent months of operation* between June 1976 and March 1978 was approximately 0.08 inches, 0.22 inches, 0.37 inches and 0.16 inches, in steam generators 21, 22, 23 and 24, respectively. The accuracy of the flow slot measurements is $\pm 1/16$ inches (0.0625 inches). Inspections of the intermediate support plates were conducted where possible, in steam generators 21, 22, 23 and 24 during this inspection and in steam generators 21 and 22 during the April 1977 inspection. The rate of average flow slot closure during the ten (10) equivalent months of operation* from April 1977 to March 1978 was 0.06 inches in steam generator 21 and 0.07 inches in steam generator 22. The maximum flow slot closures observed in any support plate was 0.67 inches, 0.92 inches, 1.13 inches, and 0.92 inches in steam generators 21, 22, 23 and 24, respectively.

Examination of the upper tube support plate in steam generator 22 through a specially drilled "hillside port" revealed no discernible change in flow slot widths from the as manufactured conditions.

Examinations of the support plates revealed two cracks in the third flow slot from the manway side in the second tube support plate in steam generator 24. The cracks are in the ligaments between the flow slot and first row tube holes near the center of the flow slot. These cracks had not been previously observed. In addition, it was discovered that a tube support plate is in contact with the wrapper in steam generator 24.

During the inspection, the licensee also removed a section of the lowest tube support plate from steam generator 23. The specimen was removed as part of the chemical cleaning feasibility study currently on-going at Indian Point Unit 2. The support plate section contained the first

*Equivalent operation for this purpose, is defined as operation with a primary coolant temperature greater than 350°F.

two rows of tubes in columns 3 through 13 occupying an area approximately 14 inches by 5 inches. The section was cut out using electrical discharge machining (EDM) and was removed through a 6 inch diameter hand hole below the lowest support plates. In place of the support plate section that was removed, a strongback fitted to the third row of tubes and designed to have the same stiffness as the removed support plate section was installed. Tube cuts were made near the support plate and the tube sheet to ensure no vibrational problems. While the specimen was being removed from the steam generator, parts of the support plate and six of the twenty-two tubes broke loose from the specimen. Measurements of the flow hole elongation between rows 1 and 2 and rows 2 and 3 account for virtually all the hot leg flow slot hourglassing. The specimen is currently being subjected to extensive examinations in the laboratory.

Evaluation

Indian Point Unit 2 is one of the six lead PWR facilities that were identified to have suffered steam generator tube denting and that have been under close monitoring by the NRC staff following the September 15, 1976 tube failure occurrence at Surry Unit 2. The inspection just completed during the current outage is the third program implemented for this unit. A discussion on the technical background and safety evaluation of the denting related phenomenon were made in a SER attached to the licensee Amendment No. 30 for the Indian Point Unit 2 dated May 13, 1977. The background information contained in the May 13, 1977 SER remains valid and is incorporated in this Safety Evaluation by reference. The information discussed above represents an update on the condition of steam generators at Indian Point Unit 2.

The licensee has conducted an adequate inspection to establish the present condition of the steam generators. The results of this inspection show that the degree of denting and the rate of flow slot closure are small. Steam generator tube operating experience at Indian Point Unit No. 2 has been good. Only three non-dent related tube leaks have occurred in the unit since it began operation. Based on this and experience at other Westinghouse units, with similar water chemistry and operating histories, the plugging of all tubes which do not allow passage of a 610 mil probe provides adequate assurance that tubes dented to the extent observed at Indian Point Unit 2 and subject to the slow rate of denting, indicated by the rate of flow slot closure, will not likely be susceptable to stress corrosion cracking during the next period of operation.

Nevertheless, the inspection results also indicate that active corrosion of the carbon steel tube support plate is continuing, but at a slower rate in comparison with other units that have been under close monitoring. In the unlikely event that steam generator tube leakage does occur a stringent primary to secondary leakage rate limit (0.3 gpm) will ensure timely corrective actions and will limit the maximum through-wall crack to a length which will not cause tube failure during postulated accidents. The limit on the frequency of tube leaks will require reevaluation of the Indian Point Unit 2 steam generators and provide timely response in the event that an accelerated rate of tube denting occurs. Since the inner row tubes have been plugged in all four steam generators and there is no indication of flow slot hourglassing in the upper support plates, there is no concern for the possibility of stress corrosion cracking in the small radius inner U-bend. With respect to the support plate cracks in steam generator 24, Westinghouse analyses have shown that extensive tube vibration and wear may result only when lateral support is lost at three or more tube support plates. Since the cracking of support plates at Indian Point 2 is limited and has not even approached conditions discussed above and the rate of flow slot closure is small, no excessive tube vibration resulting from cracked support plates is expected to occur during the next period of operation. Furthermore, concern over tube damage due to vibration is lessened because as mentioned above, all the row one tubes, including those in the cracked support plate areas are plugged.

Removal of the support plate and tube specimen from steam generator 24 suggests the existence of cracking in the support plates between the row one and row three tubes. The potential for cracking in interior support plate areas has been addressed previously and removal of the support plate specimen has proven the hypothesis that inplane compressive forces will hold cracked portions of the support plate in position and that they will continue to perform their intended function.

Conclusion

We have reviewed the licensee's inspection reports dated March 24, 1978, and May 4, 1978, and we have met with the licensee on March 22, 1978, and May 5, 1978. Based on the above discussion and evaluation, we conclude that Indian Point Unit 2 may be returned to power for a period of sixteen equivalent power months of operation but not to exceed December 1, 1979, before the next steam generator inspection without undue risk of steam generator deterioration. For the purpose of this requirement, equivalent operation is defined as operation with a primary coolant temeprature greater than 350°F. In addition, we are requiring the following measures:

1. In the event that a primary to secondary leakage exceeds 0.30 gpm in any steam generator, the reactor shall be brought to cold shutdown condition within 24 hours. The leaking tube(s) shall be evaluated and plugged and the NRC shall be informed before resuming power operation.

2. If leakage attributable to tube denting from two or more tubes in the steam generators in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24 hours and NRC approval shall be obtained before resuming reactor operation or if two steam generator tube leaks attributable to tube denting are observed after the reactor is in cold shutdown NRC approval shall be obtained before resuming reactor operation.

3. The proposed program for the next steam generator inspection shall be submitted for NRC review and comment at least 30 days prior to the implementation of the inspection program.

4. The next steam generator inspection, performed within 16 equivalent full power months of operation, will include gauging and eddy current testing for denting in all four steam generators.

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 5.15(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.

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 signifcant 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.

Dated: May 12 ,1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED 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. 40 to Facility Operating License No. DPR-26, issued to the Consolidated Edison Company of New York, Inc. (the licensee), 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 its date of issuance.

The amendment requires an inspection of steam generators on or before December 1, 1979. The Technical Specifications for the facility has also been revised to establish new steam generator leakage limits.

The Commission has made appropriate findings as required by the Atomic Energy Act of 1954, as amended, 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 licensee's submittal dated March 24, 1978, as supplemented by letter dated May 4, 1978, (2) Amendment No. 40 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, NW., Washington, D.C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601. 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 Operating Reactors.

Dated at Bethesda, Maryland, this 12th day of May 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors