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Docket No. 50-247

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

REGULATORY DOCKET FILE COPY

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment, based on results from your steam generator inspection report dated August 23, 1979, as supplemented by letter dated August 31, 1979 authorizes operation for up to 16 additional equivalent months before the next inspection of all steam generators.

No later than 30 days prior to the date you expect the next inspection to commence, you are requested to submit your detailed plan for that inspection.

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

Sincerely,

Original Signed By

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

Enclosures:
1. Amendment No. 58 to DPR-26
2. Safety Evaluation
3. Notice of Issuance

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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. 58
License No. DPR-26

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

   A. The facility will operate in conformity with the application, the provisions of the Act, 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; and

   D. 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, Facility Operating License No. DPR-26 is amended by changing paragraph 2.D to read as follows:

"2.D <u>Steam Generator Inspections</u>

The plant shall be brought to the cold shutdown condition within sixteen equivalent months of operation from August 31, 1979, but in any event, no later than May 1, 1981. For the purpose of this requirement, equivalent operation is defined as operation with a reactor coolant temperature greater than 350°F. An inspection of all four steam generators shall be performed and 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

Date of Issuance: September 7, 1979

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 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

Background

On May 12, 1978, following the March 1978 steam generator inspections at Indian Point Unit 2, the unit's operating license was amended to allow sixteen (16) equivalent full power months of operation before the next steam generator inspection. The March 1978 steam generator inspection 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. Prior steam generator inspections conducted in April 1977, included eddy current testing in steam generators 21 and 22 and support plate flow slot examinations in all four generators. In June and November 1976, eddy current testing of tubing and flow slot examinations were performed in four steam generators.

By letter dated April 13, 1979, the licensee submitted details of planned steam generator inspections to be performed during a Summer 1979 refueling outage. This inspection plan, with minor revisions, was approved by the Nuclear Regulatory Commission staff and the Summer 1979 inspections were conducted in accordance with that plan. The results are discussed below.

Eddy Current Gauging and Probing Results

Eddy current testing and gauging of hot leg (inlet side) tubing and flow slot and support plate examinations were completed in all four steam generators. A standard 700 mil probe was used to perform the eddy current testing. Any tube that did not permit passage of this probe was tested with successively smaller probes until the size of the restriction was quantified. 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 inspected for dents by eddy current testing.

In steam generator 21 (SG-21) the eddy current inspections were performed in 440 tubes (13.8% of the unplugged tubes). Similarly 389 tubes were inspected (12.3%) in SG-22, 289 tubes (9.1%) in SG-23, and 401 tube (12.8%) in SG-24. In each steam generator, the eddy current inspection included tubes in hardspot areas in rows 2 and 3 of the hot leg, peripheral hard spot areas, and tubes adjacent to the patch plate seam. Tubes in row 1 were not inspected because all the tubes in this row in all four steam generators were plugged during the unit's construction phase when modifications were made to the water box divider plates.

The average and maximum dent magnitudes in each steam generator are given in the table below, with comparison data from prior inspections. A comparison of these measurements taken over a period of years, indicates no significant increase in magnitude or any other obvious trend.

Dent Measurements (mils)(Average/Max)

Steam Generator	July 79	March 78	April 77	June 76
21	3.1/13		3.0/20	4.2/15
22	2.4/9		2.7/10	2.7/9
23	2.7/15	2.6/9		4.3/12
24	2.3/9	2.2/9		3.4/10

Of the tubes inspected, 112 tubes in SG-21, 76 tubes in SG-22, 68 tubes in SG-23, and 107 tubes in SG-24 did not permit passage of the 700 mil probe. Using successively smaller probes, the size of the restrictions were quantified as shown below. Comparative data from the March 1978 and April 1977 inspections are included.

No. of Tubes Not Passing Successively Smaller Probes
July 1979

Steam Generator	No. of Tubes Inspected	700 mils	675 mils	640 mils	610 mils	540 mils
21	440	112	39	10	3	0
22	389	76	26	13	9	0
23	289	68	31	9	6	0
24	401	107	53	7	5	1

March 1978

Steam Generator	No. of Tubes Inspected	700 mils	675 mils	640 mils	610 mils	540 mils
23	463	47	18	3	2	1
24	473	70	26	9	6	3

April 1977

Steam Generator	No. of Tubes Inspected	700 mils	675 mils	640 mils	610 mils	540 mils
21	400*	8	3	**	0	0
22	400*	3	3	**	0	0

* approximately
** data not available

During the Summer 1979 inspections, all tubes that did not permit the passage of the 510 mil probe were plugged. Three tubes were plugged in SG-21, eleven in SG-22, six in SG-23, and six in SG-24 (one inadvertently).

Tube Support Plate and Flow Slot Examination

Inspection of the lower support plates in all four steam generators indicated increased "hourglassing" of the flow slots since the March 1978 inspection. The average flow slot reduction measured in all four steam generators in March 1978 was about 7/16 inches. This went to an average of about 5/8 inches in July 1979. The maximum flow slot reduction observed in any support plate during the July 1979 inspection was 1-3/8 inches in SG-22.

Examination of the upper tube support plate in SG-22 and SG-23 revealed no discernable change in the flow slot widths from the as-manufactured conditions.

During the March 1978 inspection, two cracks were observed in the third flow slot from the manway side in the second tube support plate of SG-24. These cracks are in the ligaments between the flow slot and the first row tube holes near the center of the flow slot. In addition, there was one crack in SG-23 in the second flow slot from the nozzle side in the third support plate. These cracks had not been observed prior to the March 1978 inspection (which was the first inservice inspection of these areas).

Examination of these cracks during the July 1979 inspection indicated that they have opened slightly since March 1978. In addition a crack was observed in the third flow slot from the manway side in SG-22. This lot was not previously examined.

The wrapper to tube support plate annulus in SG-24 had not changed since the previous examination of this area conducted during the March 1978 inspection when the wrapper was first observed to be in contact with the support plate. No evidence of bulging or distortion of the wrapper was found during the Summer 1979 inspection.

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 fourth inspection 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 license Amendment No. 30 for the Indian Point Unit 2 dated May 13, 1977. The background information contained in that 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 compared to that observed in other PWRs which have experienced tube leakage due to denting. Steam generator tube operating experience at Indian Point Unit 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 susceptible 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 the 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 responses 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 generators 22, 23, and 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 Unit 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 another sixteen equivalent months of operation. Furthermore, concern over tube damage due to vibration is lessened because, as mentioned above, all the row tubes including those in the cracked support plate areas are plugged.

Summary

We have reviewed the licensee's August 23, 1979 and August 31, 1979 submittals regarding the July 1979 inspection of the Indian Point Unit 2 steam generators. Based upon the above discussion and evaluation, we conclude that Indian Point Unit 2 may be returned to power for a period of sixteen (16) equivalent full power months of operation, but not to extend beyond May 1, 1981 before the next steam generator inspection without undue risk of steam generator degradation and with a high degree of assurance that the public health and safety will not be endangered. For the purpose of this report, equivalent operation is defined as operation with a reactor coolant temperature greater than 350°F. In addition, the following requirements should be maintained.

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

2. The next steam generator inspections shall include gauging and eddy current testing for denting in all four steam generators.

3. Nuclear Regulatory Commission approval should, as before, be obtained prior to resuming power operation following the next inservice inspection of the 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 §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.

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: September 7, 1979

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. 58 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 requires an inspection of steam generators on or before May 1, 1981.

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.

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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 August 23, 1979 as supplemented by letter dated August 31, 1979, (2) Amendment No. 58 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 Operating Reactors.

Dated at Bethesda, Maryland, this 7th day of September, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

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