May 28, 1987

Docket No. 50-247

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W. Jones	E. Butcher
ACRS (10)	GPA/PA
ARM/LFMB	L. Norholm, Reg I

Mr. Murray Selman
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
New York, New York 10511

Dear Mr. Selman:

The Commission has issued the enclosed Amendment No. 119 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 December 8, 1986 (TAC 64160).

The amendment revises the Technical Specifications to permit operation of Indian Point Unit No. 2 with one or more inoperable Main Steam Line Safety Valves provided that the power range high flux setpoint is reduced to a specified value.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Marylee M. Slosson, Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A, NRR

Enclosures:
1. Amendment No.119 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

*SEE PREVIOUS CONCURRENCE

PDI-1	PDI-1 MS 5/28/87	OGC	PDI-1 RAC
CVogan*	MSlosson*	MKarman*	RCapra
5/11/87	5/26/87	5/14/87	5/28/87

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ARM/LFMB	L. Norholm, Reg I

Mr. Murray Selman
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
New York, New York 10511

Dear Mr. O'Toole:

The Commission has issued the enclosed Amendment No. 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 April 29, 1987 (TAC 64160).

The amendment revises the Technical Specifications to permit operation of Indian Point Unit No. 2 with one or more inoperable Main Steam Line Safety Valves provided that the power range high flux setpoint is reduced to a specified value.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Marylee M. Slosson, Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A, NRR

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2. Safety Evaluation

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See next page

PDI-1 CW	PDI-1 MS	OGC M. Karman	PDI-1
CVogan	MSlosson		RCapra
5/11/87	5/13/87	5/14/87	5/ /87

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Consolidated Edison Company
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Indian Point Nuclear Generating
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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. 119
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 December 8, 1986, 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 119 , 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

Robert A. Capra

Robert A. Capra, Acting Director
Project Directorate I-1
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 28, 1987

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

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.4-1	3.4-1
3.4-1(a)	3.4-1(a)
------	3.4-1(b)
3.4-2	3.4-2
3.4-3	3.4-3

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and City Water System operation is necessary to ensure the capability to remove decay heat from the core.

Specification

A. The reactor shall not be heated above 350°F unless the following conditions are met:

   (1) A minimum ASME code approved steam relieving capability of twenty (20) main steam valves shall be operable (except for testing). With up to three (per steam generator) of the twenty main steam line code-approved safety relief valves inoperable, heat-up above 350°F and power operation is permissible provided either the inoperable valve(s) is restored to operable status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.4-1.

   (2) Three auxiliary feedwater pumps each capable of pumping a minimum of 400 gpm must be operable.

   (3) A minimum of 360,000 gallons of water in the condensate storage tank and a backup supply from the city water supply.

   (4) Required system piping, valves, and instrumentation directly associated with the above components operable.

   (5) The main steam stop valves are operable and capable of closing in five seconds or less.

   (6) The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 μCi/cc.

B. Except as modified by 3.4.C below, if any of the conditions of 3.4.A above cannot be met within 72 hours, the reactor shall be placed in the hot shutdown condition within the next 12 hours and subsequently cooled below 350°F using normal operating procedures.

TABLE 3.4-1

Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4 Loop Operation

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of Rated Thermal Power)
1	85
2	61
3	37

C. If when above 350°F one or both of the series valves (CT-6 and/or CT-64) in the condensate storage tank discharge line is closed, then:

(1) Immediately place the auxiliary feedwater pump controls in the manual mode, and

(2) Within one (1) hour, either valve(s) shall be reopened or the valves from the alternate city water supply shall be opened and the auxiliary feedwater pump controls restored to the automatic mode.

If these requirements cannot be met, then:

(1) maintain the plant in a safe stable mode which minimizes the potential for a reactor trip, and

(2) continue efforts to restore water supply to the auxiliary feedwater system, and

(3) notify the NRC within 24 hours regarding the planned corrective action.

Basis

Reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generators is provided by operation of the turbine cycle feedwater system.

The operability of the twenty main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% Rated Thermal Power coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The total relieving capacity of the twenty main steam safety valves is 15,108,000 lbs/hr which is 129 percent of the total secondary steam flow of 11,669,736 lbs/hr at 100% Rated Thermal Power (2758 Mwt). Startup and/or power operation is allowable with main steam safety valves inoperable within the limitations of Table 3.4-1 on the basis of the reduction in secondary system steam flow and Thermal Power required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following basis:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP= Reduced reactor trip setpoint in percent of RATED THERMAL POWER

V= Maximum number of inoperable safety valves per steam line

(109)= Power Range Neutron Flux-High Trip Setpoint for (4) loop operation

X= Total relieving capacity of all safety valves per steam line (3,777,000 lbs/hr).

Y= Maximum relieving capacity of any one safety valve (823,000 lbs/hr).

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tank is the amount needed for 24 hours at hot standby. When the condensate storage supply is exhausted, city water will be used.

The limit on secondary coolant total iodine activity of I-131 and I-133 is based on a postulated release of secondary coolant equivalent to the contents of four steam generators to the atmosphere due to a net load rejection with loss-of-offsite power. The limiting dose for this case would result from radioactive iodine in the secondary coolant. I-131 and I-133 are the dominant isotopes because of their low MPC's in air and because the other shorter-lived istopes cannot build up to significant concentrations in the secondary coolant under the limits of primary system leak rate and activity. One tenth of the iodine in the secondary coolant is assumed to reach the site boundary making allowance for plate-out and retention in water droplets. The inhalation dose at the site boundary is then as follows:

$$\text{Dose(rem)} = \frac{C \cdot V}{10} \cdot B(t) \cdot x/Q \cdot DCF$$

Where: C = secondary coolant activity ($0.15\ \mu Ci/cc = 0.15\ Ci/m^3$)

V = water volume in four steam generators ($7416\ ft^3 = 210\ m^3$)

B(t) = breathing rate ($3.47 \times 10^{-4}\ m^3/sec$)

$x/Q = 7.5 \times 10^{-4}\ sec/m^3$

DCF = 1.00×10^{6} rem/Ci Iodine (131 and 133) inhaled

The resultant dose is less than 1.0 rem.

Reference

FSAR - Section 10.4 and 14.1.9

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 119 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 December 8, 1986 (Reference 1), Consolidated Edison Company of New York, Inc. proposed changes in the Technical Specifications to permit operation of Indian Point Unit 2 (IP-2) with one or more inoperable main steam line safety valves, provided that the power range high flux setpoint is reduced as specified. A similar amendment request was previously approved (Ref. 2) for IP-2 on a temporary basis when, because of malfunctioning maintenance equipment, it was not possible to fix a leaky valve on each of two steam generators in May of 1986. Technical Specification 3.4.B requires that within 72 hours after discovery that a minimum of twenty (20) ASME Code approved steam-relieving main steam valves are not operable (except for testing), the reactor shall be placed in hot shutdown condition within the next 12 hours. The 72 hour limiting condition of operation was extended (Ref. 2) for a period not to exceed an additional two weeks on a one-time basis by an emergency license amendment. This change required that the high flux trip setpoint be reduced to less than or equal to 78% of rated thermal power. IP-2 now requests a permanent revision to the Technical Specifications to permit operation with inoperable main steam line safety valves with a corresponding reset of the high flux trip setpoints for operation at reduced power.

2.0 REVIEW AND EVALUATION

The IP-2 design provides 5 safety valves per steam generator, for a total of 20 valves. The IP-2 TS 3.4.A(1) requires that a minimum of 20 ASME Code safety valves be operable when above 350°F. If this requirement cannot be met, TS 3.4.B requires the licensee to have all 20 ASME code safety valves operable within 72 hours, or the reactor is to be placed in hot shutdown within the next 12 hours. The licensee received approval (Ref. 2) on a one-time basis to extend, to a maximum of 2 weeks, the time for repairing the inoperable valves (one on each of two steam generators) provided the remaining 18 safety valves are operable and the high flux trip setpoint is reduced to less than or equal to 78% of rated thermal power.

The Standard Technical Specification (STS) for Westinghouse PWRs, NUREG-0452, Rev. 2, permits operation with one or more inoperable safety valves if the power range high flux setpoint is reduced to 87% of rated thermal power with one inoperable safety valve on any operating steam generator. The STS states that a minimum of two operable safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable thermal power restriction as shown in Table 3.4-1.

IP-2 has made plant-specific calculations for their proposed Table 3.4-1 using the STS formula (provided in the IP-2 Bases, page 3.4-2) as shown below to obtain the reactor trip setpoint reductions. This is for use when up to three safety valves on any operating steam generator are inoperable, leaving at least two operable.

$$SP = \frac{(X)-(Y)(V)}{X} \times (109) \qquad (1)$$

Where:

SP = Reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = Maximum number of inoperable safety valves per steam line

(109) = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

X = Total relieving capacity of all safety valves per steam line (3,777,000 lbs/hr).

Y = Maximum relieving capacity of any one safety valve (823,000 lbs/hr).

Formula (1) was used to derive the reduced reactor trip setpoint values shown in the proposed Table 3.4-1 below.

TABLE 3.4-1

Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4 Loop Operation

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of Rated Thermal Power)
1	85
2	61
3	37

The staff has found that the proposed high flux trip setpoint values for IP-2 are conservative relative to the Westinghouse STS values and also that the values were rounded down conservatively from the calculations using formula (1) above. The staff finds the proposed high flux setpoints shown in proposed Table 3.4-1, to be acceptable for one to three safety valves inoperable on any steam generator.

3.0 TECHNICAL SPECIFICATION CHANGES

The changes to the Technical Specifications for Indian Point Unit 2 as a result of allowing operation at reduced power with one to three main steam valves inoperable involve pages 3.4-1, 3.4-1(a), 3.4-1(b), 3.4-2, and 3.4-3. These changes are discussed below.

Page 3.4-1

This page has the following sentence added to Specification A. (1) - "With up to three (per steam generator) of the twenty main steam line code-approved safety relief valves inoperable, heat-up above 350°F and power operation is permissible provided either the inoperable valve(s) is restored to operable status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.4-1."

Also, an asterisk is removed after "72 hours" in the sentence below for Specification A.(B).

B. Except as modified by 3.4.C below, if any of the conditions of 3.4.A above cannot be met within 72 hours*, the reactor shall be placed in the hot shutdown condition within the next 12 hours and subsequently cooled below 350°F using normal operating procedures.

The asterisk referred to the one time basis (which is now removed) when an additional 2 week period was allowed for making the two faulted main steam safety valves operable as explained in the introduction.

These changes are acceptable for the reasons explained above in the Review and Evaluation Section.

Page 3.4-1(a)

This new page provides Table 3.4-1, "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4 Loop Operation". The values shown in this table were compared with the Westinghouse STS and also the new plant-specific values were recalculated using the Westinghouse STS formula. We find this table to be acceptable as further explained above in the Review and Evaluation Section.

Page 3.4-1(b)

This page has not been changed except to editorially change the page number from 3.4-1(a) to 3.4-1(b) because of the new Table 3.4-1. Therefore, we find it acceptable.

Page 3.4-2

This page has new material inserted pertaining to the Westinghouse STS method for calculating the reactor trip setpoints as explained for formula (1) in the Review and Evaluation Section. We have found the formula derivation to be in agreement with the STS and acceptable.

Page 3.4-3

This page contains more information, because of runover from page 3.4-2, than previously included. However, it is only editional in nature. Therefore, we find it acceptable.

4.0 SUMMARY

We conclude that the changes made to the IP-2 Technical Specifications are acceptable because: 1) the changes are in accordance with the Westinghouse STS, including Table 3.4-1 and the method for calculating the reactor trip setpoints, 2) the values shown in Table 3.4-1 are conservative, and 3) the editorial changes are correct.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff 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; 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.

Dated: May 28, 1987

Principal Contributor:

H. Balukjian

REFERENCES

1. Letter from M. Selman, Consolidated Edison Company of New York, Inc., to S. A. Varga, NRC, dated December 8, 1986.

2. Letter from M. Slosson, NRC, to John D. O'Toole, Consolidated Edison Company of New York, Inc., dated June 11, 1986.