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

December 28, 1993

Docket No. 50-247

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

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M86409)

The Commission has issued the enclosed Amendment No. 167 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 May 5, 1993, as revised by letter dated August 27, 1993, and supplemented by letter dated December 21, 1993.

The amendment revises surveillance intervals for the Containment Pressure Channels, the Steam Pressure Channels, and the Reactor Coolant Temperature Channels to accommodate a 24-month refueling cycle. These revisions are being made in accordance with the guidance provided by Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle."

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,

Francis J. Williams, Jr.

Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 167 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Stephen B. Bram
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station Units 1/2

cc:

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, New York 10511

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Mr. Charles W. Jackson
Manager of Nuclear Safety and
Licensing
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, New York 10511

Mr. Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Mr. Peter Kokolakis, Director
Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Mr. Walter Stein
Secretary - NFSC
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

DATED: December 28, 1993

AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
F. Williams
OGC
D. Hagan, 3302 MNBB
J. Ganiere, 8/H/3
G. Hill (2), P1-22
C. Grimes, 11/F/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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. 167
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 May 5, 1993, as revised by letter dated August 27, 1993, and supplemented by letter dated December 21, 1993, 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. 167, 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

Brian C. McCabe for

Robert A. Capra, 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: December 28, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 167

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.3-2	2.3-2
Table 3.5-1 (page 1 of 1)	Table 3.5-1 (page 1 of 1)
Table 4.1-1 (page 1 of 7)	Table 4.1-1 (page 1 of 7)
Table 4.1-1 (page 2 of 7)	Table 4.1-1 (page 2 of 7)
Table 4.1-1 (page 3 of 7)	Table 4.1-1 (page 3 of 7)

P = Pressurizer pressure, psig

P' = 2235 psig

$K_1 \leq 1.22$

$K_2 = 0.022$

$K_3 = 0.00095$

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -36% and +7%, $f(\Delta I) = 0$, where q_t and q_b are percent rated power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total power in percent of rated power;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -36%, the ΔT trip setpoint shall be automatically reduced by 2.14% of its value at rated power; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT trip setpoint shall be automatically reduced by 2.15% of its value at rated power.

(5) Overpower ΔT:

$$\Delta T \leq \Delta T_o \left[K_4 - K_5 \frac{dT}{dt} - K_6 (T - T^*)\right]$$

where:

ΔT = Measured ΔT by hot and cold leg RTDs, $^{\circ}F$

$\Delta T_o \leq$ Indicated ΔT at rated power, $^{\circ}F$

T = Average temperature, $^{\circ}F$

Table 3.5-1

Engineered Safety Features Initiation Instrument Setting Limits

No.	Functional Unit	Channel	Setting Limits
1.	High Containment Pressure (Hi Level)	Safety Injection	≤ 2.0 psig
2.	High Containment Pressure (Hi-Hi Level)	a. Containment Spray b. Steam Line Isolation	≤ 24 psig
3.	Pressurizer Low Pressure	Safety Injection	≥ 1833 psig
4.	High Differential Pressure Between Steam Lines	Safety Injection	≤ 155 psi
5.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	a. Safety Injection b. Steam Line Isolation	≤ 40% of full steam flow at zero load ≤ 40% of full steam flow at 20% load ≤ 110% of full steam flow at full load ≥ 540°F T_{avg} ≥ 525 psig steam line pressure
6.	Steam Generator Water Level (Low-Low)	Auxiliary Feedwater	≥ 7% of narrow range instrument span each steam generator
7.	Station Blackout (Undervoltage)	Auxiliary Feedwater	≥ 40% nominal voltage
8a.	480V Emergency Bus Undervoltage (Loss of Voltage)	------	220V + 100V, -20V 3 sec ± 1 sec
8b.	480V Emergency Bus Undervoltage (Degraded Voltage)	------	403V ± 5V 180 sec ± 30 sec

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S	D (1) M* (3)	Q (2)	1) Heat balance calibration 2) Signal to delta T; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset.
2.	Nuclear Intermediate Range	S (1)	N.A.	S/U**(2)	1) Once/shift when in service Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	S (1)	N.A.	S/U**(2)	1) Once/shift when in service 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	S	R#	Q (1)	1) Overtemperature - delta T 2) Overpower - delta T
5.	Reactor Coolant Flow	S	R#	Q	
6.	Pressurizer Water Level	S	R#	Q	
7.	Pressurizer Pressure (High & Low)	S	R#	Q	
8.	6.9 kV Voltage & Frequency	N.A.	R	Q	Reactor Protection circuits only
9.	Analog Rod Position	S	R	M	

* By means of the movable incore detector system.

** Prior to each reactor startup if not done previous week.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S	N.A.	N.A.	With analog rod position
11.	Steam Generator Level	S	R#	Q	
12.	Charging Flow	N.A.	R#	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R#	N.A.	
14.	Boric Acid Tank Level	W	R	N.A.	Bubbler tube rodded during calibration
15.	Refueling Water Storage Tank Level	W	R	N.A.	
16.	DELETED				
17.	Volume Control Tank Level	N.A.	R	N.A.	
18a.	Containment Pressure	D	R#	Q	Wide Range
18b.	Containment Pressure	S	R#	Q	Narrow Range
18c.	Containment Pressure (PT-3300, PT-3301)	M	R#	N.A.	High Range
19.	Process Radiation Monitoring System	D	R#	M	
19a.	Area Radiation Monitoring System	D	R#	M	
19b.	Area Radiation Monitoring System (VC)	D	R#	M	

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
20.	Boric Acid Make-up Flow Channel	N.A.	R	N.A.	
21a.	Containment Sump and Recirculation Sump Level (Discrete)	S	R#	R#	Discrete Level Indication Systems.
21b.	Containment Sump, Recirculation Sump and Reactor Cavity Level (Continuous)	S	R#	R#	Continuous Level Indication Systems.
21c.	Reactor Cavity Level Alarm	N.A.	R#	R#	Level Alarm System
21d.	Containment Sump Discharge Flow	S	R	M	Flow Monitor
21e.	Containment Fan Cooler Condensate Flow	S	R#	M*	
22a.	Accumulator Level	S	R#	N.A.	
22b.	Accumulator Pressure	S	R#	N.A.	
23.	Steam Line Pressure	S	R#	Q	
24.	Turbine First Stage Pressure	S	R#	Q	
25.	Reactor Trip Logic Channel Testing	N.A.	N.A.	M[1]	
26.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R#	M	

* Monthly visual inspection of condensate weirs only.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167 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 May 5, 1993, as revised by letter dated August 27, 1993, and supplemented by letter dated December 21, 1993, the Consolidated Edison Company of New York (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 Technical Specifications (TSs). The requested changes are a follow-up to License Amendment No. 159, issued on December 10, 1992, which changed the TS Section 1.0, Definitions, to accommodate a 24-month fuel cycle and which extended test intervals for specific surveillance tests. The requested changes in this proposal would extend the surveillance intervals to 24 months for the Containment Pressure Channels, the Steam Pressure Channels, and the Reactor Coolant Temperature Channels. The changes requested by the licensee are related to a 24-month fuel cycle and are in accordance with Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle." The December 21, 1993, letter provided a replacement TS page (Table 4.1-1, page 2 of 7) since it had been revised by issuance of Amendment No. 166 on December 16, 1993. It did not change the initial proposed no significant hazards consideration and was not outside the scope of the original Federal Register notice.

2.0 EVALUATION

Improved reactor fuels allow licensees to consider an increase in the duration of the fuel cycle for their facilities. A longer fuel cycle increases the time interval between refueling outages and the performance of TS surveillance requirements. GL 91-04 provides guidance to support the development of TS revisions to allow 24-month surveillance intervals and includes requirements to evaluate the effect on safety for an increase in surveillance testing and calibration intervals to accommodate a 24-month fuel cycle.

The licensee evaluation should conclude that the net effect on safety is small, that historical plant maintenance and surveillance data support the proposed extended surveillance interval, and that the assumptions in the plant licensing basis are still bounding with the incorporation of a 24-month surveillance interval. The staff also determined that a licensee should

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address the issue of instrumentation errors/setpoint methodology assumptions when proposing an extended instrumentation calibration interval. Specifically, the licensee must evaluate the effects of an increased calibration interval on instrument uncertainties, equipment qualification, and vendor maintenance requirements to ensure that an extended surveillance interval does not result in exceeding the assumptions stated in the safety analysis.

To support the proposed changes the licensee reviewed instrument calibration data from applicable surveillances and maintenance records and recorded the historical as-left and as-found drift information. The licensee confirmed that instrument drift has not, except on rare occasions, exceeded acceptable results and that the historical data does indicate any problems that would preclude an increase in the intervals for instrument calibration. The licensee's description of the methodology and assumptions used to determine the rate of instrument drift with time was approved by the staff as documented in Amendment No. 159 to Facility Operating License No. DPR-26.

The licensee statistically evaluated the past drift data to determine a projected 30-month drift value. The projected 30-month drift value was used as input to determine the Channel Statistical Allowance using the NRC-approved Westinghouse setpoint methodology. This evaluation included, along with instrument drift, the determination of all other channel uncertainties, including sensor, rack, measurement and test equipment, and process effects for normal environmental conditions. The licensee evaluated the resulting channel uncertainties to determine if they supported the current TSs and safety analysis limits. For some instrument channels it was determined that the uncertainties exceeded those that could be supported by TSs setpoints and safety analysis setpoints and, as a result, setpoint changes were proposed. Each of the proposed changes is evaluated below.

2.1 Calibration of Steam Pressure Channels

The licensee has proposed to extend the calibration interval for the steam pressure channels from 18 to 24 months. The steam pressure channels initiate a safety injection signal on high differential pressure between steamlines. In addition, the steam pressure channels provide a safety injection signal and steamline isolation signal on high steam flow in 2-out-of-4 steamlines coincident with low steamline pressure.

The licensee reviewed historical test data from February 1986 (1985 date in submittal was a typographical error per telecon with licensee on 12/14/93) to the present and statistically evaluated this data to determine a projected 30-month drift value with a 95 percent probability at a 95 percent confidence level. The results of the channel statistical allowance calculations indicated that the channel uncertainties exceed those which can be supported by the existing high differential steam pressure TS setpoint. To accommodate the increased channel uncertainties, the licensee has proposed to increase the high differential steam pressure safety analysis limit from 215 psi to 270 psi and the TS setpoint from 150 psi to 155 psi. The channel uncertainties do not exceed those which can be supported by the high steam flow coincident with low steam pressure TS setpoint; however, for increased operating flexibility, the licensee has proposed to reduce the safety analysis limit and TS limit for the high steam flow coincident with low steam pressure trip from 445 psig to 400 psig and from 600 psig to 525 psig, respectively.

The licensee has stated that these setpoint changes only affect non-loss-of-coolant accident (LOCA) safety analyses and do not affect any LOCA safety analyses. For the non-LOCA safety analyses, only the steam pipe rupture event is affected. The licensee evaluated the main steamline break cases for this event taking into account the new setpoints, and it was determined that there is no adverse effect on main steamline break core response or resulting mass and energy release inside containment.

Based on the information provided by the licensee, the staff finds that the proposed change to extend the steam pressure channels calibration interval to 24 months is acceptable. Also, the proposed changes to increase the high steamline differential pressure setpoint, and to decrease the high steam flow coincident with low steamline pressure setpoint do not have adverse effects on the response to a steam pipe rupture event. Therefore, this setpoint change is considered acceptable.

2.2 Calibration of Reactor Coolant Temperature Channels

The licensee has proposed to extend the calibration interval for the reactor coolant temperature channels from 18 to 24 months. The narrow range temperature channels initiate a reactor trip on overtemperature delta-temperature with a nominal setpoint based on K1 less than or equal to 1.25 or on overpower delta-temperature with a nominal setpoint based on K4 less than or equal to 1.074.

The licensee reviewed historical test data from February 1986 to the present and statistically evaluated this data to determine a projected 30-month drift value with a 95 percent probability at a 95 percent confidence level. The results of the channel statistical allowance calculations indicated that the channel uncertainties exceeded those which can be supported by the K1 factor in the existing overtemperature delta-temperature TS setpoint and by the safety analysis limit. As a result, the licensee has proposed to decrease the K1 factor from 1.25 to 1.22 to accommodate the projected channel uncertainty over a 30 month operating cycle.

Based on the information provided by the licensee, the staff finds the proposed change to extend the reactor coolant temperature channels calibration interval to 24 months is acceptable. Also, the proposed change to decrease the K1 factor does not have an adverse effect on safety. Therefore, this setpoint change is considered acceptable.

2.3 Calibration of Containment Pressure Channels

The licensee has proposed to extend the calibration interval for the containment pressure channels from 18 to 24 months. The containment pressure channels initiate a safety injection signal at a trip setpoint of less than or equal to 2.0 psig and containment spray and steamline isolation at a trip setpoint of less than or equal to 30 psig.

The licensee reviewed historical test data from February 1986 to the present and statistically evaluated this data to determine a projected 30-month drift value with a 95 percent probability at a 95 percent confidence level. The results of the channel statistical allowance calculations indicated that the channel uncertainties exceeded those which can be supported by the existing high containment pressure TS setpoint or by the safety analysis limits. As a result, the licensee has proposed to increase the safety analysis limit from 2.0 psig to 7.3 psig in order to maintain the existing TS setpoint of 2.0 psig and to accommodate the projected channel uncertainty over a 30-month operating cycle. For the high-high containment pressure, the licensee proposed to decrease the TS setpoint from 30 psig to 24 psig in order to maintain the existing safety analysis limit of 30 psig. Reducing this TS setpoint is a conservative change and is, therefore, acceptable.

The licensee evaluated the effects of the increased safety analysis limit for the high containment pressure channel on the analysis assumptions and results for LOCA related accident analyses and the containment integrity analysis. The evaluation demonstrated that the peak calculated containment pressure will be less than the containment design and integrated leak test value of 47 psig as identified in WCAP-12237, "Containment Integrity Analysis for Indian Point Unit 2 - December 1989." Therefore, the design basis analysis remains valid, and margin is maintained between the peak calculated containment pressure and the design pressure. The LOCA analyses are demonstrated to be unaffected by the proposed increase in the containment high pressure setpoint.

Based on the information provided by the licensee, the staff finds that the proposed change to extend the containment pressure channels calibration interval to 24 months is acceptable.

Based on the above the staff finds the proposed TS changes to increase the calibration surveillance interval from 18 to 24 months (30 months with grace period) for reactor coolant temperature channels, steam pressure channels, and containment pressure channels as proposed in the licensee submittal to be acceptable. The licensee has evaluated the proposed changes in accordance with the guidance contained in GL 91-04. Their evaluation has concluded that the net effect on safety is small, that historical plant maintenance and surveillance data support the proposed extended surveillance intervals, and that the assumptions in the plant licensing basis are still bounding with the incorporation of a 24-month surveillance interval. In order to accommodate a 30-month surveillance interval, the licensee proposed to change TSs setpoints.

The channel statistical allowance calculations in conjunction with the setpoint changes have shown that sufficient margin exists between the analytical limits and the corresponding TSs trip setpoints and that the assumptions of the safety analysis are not violated. Therefore, the staff finds the proposed setpoint changes to be 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 amendment changes surveillance requirements. The NRC 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 52981). Accordingly, the 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 or 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. Ganiere

Date: December 28, 1993

December 28, 1993

Docket No. 50-247

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

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M86409)

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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:

Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 167 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

Distribution:
See attached sheet

*See previous concurrence

LA:PDI-1	PM:PDI-1	OGC*	D:PDI-1		
CVogan	FWilliams:smm	RBachmann	RACapra		
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