LICENSE AUTHORITY FILE COPY DO NOT REMOVE

April 10, 1989

Docket No. 50-247

P O S T E D

50-247
INDIAN PT 2
AMENDMENT NO. 137
TO DPR-26

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

WJones	EButcher
ACRS (10)	GPA/PA
ARM/LFMB	PSwetland, RI

Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 55358)

The Commission has issued the enclosed Amendment No. 137 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 August 18, 1986 and supplemented by your letter dated January 25, 1989.

The amendment revises and adds new requirements to Technical Specification Tables 3.5-2 and 4.1-1 requiring the operability and surveillance testing of the reactor trip breakers shunt trip attachment.

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

Sincerely,

ORIGINAL SIGNED BY

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 137 to DPR-26
2. Safety Evaluation

cc: w/enclosures
See next page

See n ecessary changes to Amend 3/23

OFC	:PDI-I	:PDI-I	:OGC	:PDI-I	:	:	:
NAME	:CVogan	:DBrinkman:vr	:RBachmann	:RCapra	:	:	:
DATE	:3/14/89	:3/15/89	:3/23/89	:4/10/89	:	:	:

OFFICIAL RECORD COPY

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

Indian Point Nuclear Generating
Station 1/2

cc:

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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. 137
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 August 18, 1986 and supplemented January 25, 1989, 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) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.137 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 10, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 137

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
Table 3.5-2 (2 of 3)	Table 3.5-2 (2 of 3)
Table 3.5-2 (3 of 3)	Table 3.5-2 (3 of 3)
Table 4.1-1 (page 2)	Table 4.1-1 (page 2)
	Table 4.1-1 (new page 6 of 6)

TABLE 3.5-2 (Continued)

		1	2	3	4	5
10.	Low Flow Loop ≥ 75% F.P.	3/loop	2/loop(any loop)	2/operable loop	1/operable loop	Maintain hot shutdown
	Low Flow Two Loops 10-75% F.P.	3/loop	2/loop(any two loops)	2/operable loop	1/operable loop	
11.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown
12.	Undervoltage 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown
13.	Low frequency 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown***
14.	Quadrant power tilt monitors	2	NA	1	0	Log individual upper and lower ion chamber currents once/shift and after load change >10%
15.	Turbine trip (overspeed protection)*****	3	2	2	1	Maintain hot shutdown
16.	Control Rod Protection****	3	2	2	1	During RCS cooldown, manually open reactor trip breakers prior to T_{cold} decreasing below 350°F. Maintain reactor trip breakers open during RCS cooldown when T_{cold} is less than 350°F.
17.	Turbine Trip ≥ 35% F.P. A. Low Auto Stop Oil Pressure	3	2	2	1	Maintain reactor power below 35% F.P.
18.	Reactor Trip Logic	2	1	2#	1#	Be in Hot Shutdown within the next six hours.

TABLE 3.5-2 (Continued)

	1	2	3	4	5
19. Reactor Trip Breakers	2	1	2#	1#	With either diverse trip feature inoperable, or the breaker incapable of tripping for any other reason, restore it to operable condition or, be in hot shutdown within the next six hours and open both reactor trip breakers. The breaker shall not be bypassed except for the time required for performing maintenance and/or testing to restore it to operability.

F.P. = Rated Power

* If two of four power channels greater than 10% F.P., channels are not required.

** If one of two intermediate range channels greater than 10^{-10} amps, channels are not required.

*** 2/4 trips all four reactor coolant pumps.

**** Required only when control rods are positioned in core locations containing LOPAR fuel.

***** This will provide a turbine trip at all power levels and a reactor trip when greater than or equal to 35% F.P..

\# A reactor trip breaker and/or associated logic channel may be bypassed for maintenance or surveillance testing for up to eight hours provided the redundant reactor trip breaker and/or associated logic channel is operable.

TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
22.	Accumulator Level and Pressure	S	R	N.A.	
23.	Steam Line Pressure	S	R	M	
24.	Turbine First Stage Pressure	S	R	M	
25.	Reactor Trip Logic Channel Testing	N.A.	N.A.	M#	
26.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
27.	Turbine Trip a. Low Auto Stop Oil Pressure	N.A.	R	N.A.	
28.	Control Rod Protection (for use with LOPAR fuel)	N.A.	R	*	
29.	Loss of Power a. 480v Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	R	
	b. 480v Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	R	
	c. 480v Emergency Bus Undervoltage (Alarm)	N.A.	R	M	
30.	Auxiliary Feedwater: a. Steam Generator Water Level (Low-Low)	S	R	R	

*Within 31 days prior to entering a condition in which the Control Rod Protection System is required to be operable unless the reactor trip breakers are manually opened during RCS cooldown prior to T_{cold} decreasing below 350°F and the breakers are maintained opened during RCS cooldown when T_{cold} is less than 350°F.

Amendment No. 137,

TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
42.	Manual Reactor Trip	N.A.	N.A.	R	Includes: 1) Independent verification of reactor trip and bypass breakers undervoltage trip circuit operability up to and including matrix contacts of RT-11/RT-12 from both manual trip initiating devices, 2) independent verification of reactor trip and bypass breaker shunt trip circuit operability through trip actuating devices from both manual trip initiating devices.
43.	Reactor Trip Breaker	N.A.	N.A.	M#	Includes independent verification of undervoltage and shunt trip attachment operability.
44.	Reactor Trip Bypass Breaker	N.A.	N.A.	M#	Includes: 1) Automatic undervoltage trip, 2) Manual shunt trip from either the logic test panel or locally at the switchgear prior to placing breaker into service.

Each train shall be tested at least every 62 days on a staggered test basis (i.e., one train per month).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 137 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 August 18, 1986, Consolidated Edison Company (licensee) proposed changes and additions to the Indian Point 2 Technical Specifications which would require the operability and surveillance testing of the shunt trip attachments of the reactor trip breakers as required by Generic Letter 83-28, Item 4.3. The original required changes were clarified and modified by Generic Letter 85-09. The licensee's proposal was reviewed and additional changes to make the Technical Specifications conform more fully to the requirements of Generic Letter 85-09 were made during a meeting with the licensee held on December 12, 1988. These changes were incorporated by the licensee's submittal of January 25, 1989 and are reviewed in this SER.

2.0 EVALUATION

The licensee's Technical Specifications differ in many respects from the Standard Westinghouse Technical Specifications which were used as the basis for the requirements of Generic Letter 85-09. Consequently, the changes proposed in the August 18, 1986 letter addressing the operability and surveillance test requirements for the shunt trip attachment of the reactor trip breakers did not fully meet the requirements of Generic Letter 85-09. In a meeting with the staff on December 12, 1988, the licensee agreed that the originally proposed allowed out-of-service time (AOT) of 48 hours with one train inoperable would be changed to six hours which is in accordance with Generic Letter 85-09 recommendations. The licensee's submittal of January 25, 1989 contained this change.

The modified Technical Specification changes now provide for testing the manual reactor trip switch functions independently for the shunt and undervoltage trip attachments of the reactor trip breakers and the reactor trip switch contacts and wiring at each refueling outage. The action statements for Items 18 (Reactor Trip Logic), 19 (Reactor Trip Breakers), and a note to Table 3.5-2 were changed to conform more closely to the guidance of Generic Letter 85-09. The previous Technical Specification changes contained in the licensee's submittal of August 18, 1986 which addressed changes to

Item 25 and added Items 42, 43, and 44 to Table 4.1-1 were left in place. The licensee's test procedure PT-M14A was revised to reflect the installation of the automatic shunt trip modification. In addition, the licensee proposed increasing the time a reactor trip breaker or its associated logic channel could be bypassed for maintenance or surveillance testing from the two hours specified in Generic Letter 85-09 to eight hours. This increased AOT is required to perform the required surveillance test since Indian Point Unit 2 uses relay protection logic rather than solid state protection logic as was envisioned when the requirements of Generic Letter 85-09 were developed. We agree that relay protection logic requires this additional amount of time to perform the required testing.

3.0 CONCLUSION

We find that the proposed changes to the Indian Point Unit 2 Technical Specifications presented in the licensee's submittals and agreed to in the December 12, 1988 meeting meet the requirements of both Generic Letter 83-28, Item 4.3 and Generic Letter 85-09 and are, therefore, acceptable.

ENVIRONMENTAL CONSIDERATION

The amendment changes 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 surveillance requirements. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, 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 Sec 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 this amendment.

CONCLUSION

We have 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: April 10, 1989

PRINCIPAL CONTRIBUTOR:

D. Lasher