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

Docket No. 50-247

July 29, 1991

See Correction letter of 10/9/91

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. 77930)

The Commission has issued the enclosed Amendment No. 154 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 October 31, 1990.

The amendment revises Technical Specifications Sections 3.5 and 4.1 and associated Bases to allow routine analog channel testing in a bypassed condition instead of a tripped condition and to increase the surveillance intervals for the Reactor Protection System and the Engineered Safety Features analog channel tests from monthly to quarterly.

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. 154 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

DFO1 1/1

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

Indian Point Nuclear Generating Station
Units 1/2

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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.154
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 October 31, 1990, 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.154 , 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, 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: July 29, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 154

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.5-1	3.5-1
3.5-2	3.5-2
3.5-3	3.5-3
4.1-5	4.1-5
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)

3.5 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to plant instrumentation systems.

Objectives

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specifications

3.5.1 When the plant is not in the cold shutdown condition, the Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.5-1.

3.5.2 For instrumentation channels, plant operation at rated power shall be permitted to continue in accordance with Tables 3.5-2 through 3.5-4. No more than one channel of a particular protection channel set shall be tested at the same time. By definition, an instrumentation channel failure shall not be regarded as a channel being tested.

3.5.3 In the event the number of channels of a particular function in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Column 5 of Tables 3.5-2 through 3.5-4. For on-line testing of instruments with installed bypass capability, the required minimum degree of redundancy may be reduced by one to permit testing of a channel in bypass.

3.5.4 In the event of sub-system instrumentation channel failure, Tables 3.5-2 through 3.5-4 need not be observed during the short period of time the operable sub-system channels are tested where the failed channel must be blocked to prevent unnecessary reactor trip.

3.5.5 The cover plate on the rear of the safeguards panel in the control room shall not be removed without authorization from the Watch Supervisor.

3.5.6 When the reactor coolant system is above 350°F, the instrumentation requirements as stated in Table 3.5-5 shall be met.

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features[1,4].

Safety Injection System Actuation

Protection against a loss-of-coolant or steam break accident is brought about by automatic actuation of the Safety Injection System, which provides emergency cooling and reduction of reactivity.

The loss-of-coolant accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safety Features have been designed to sense the effects of the loss-of-coolant accident by detecting low pressure and generator signals actuating the SIS active phase.

The SIS active phase is also actuated by a high containment pressure signal (Hi-Level) brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure signal actuation of the SIS and also adds diversity to protection against loss of coolant.

4. The steamline high differential pressure limit is set well below the differential pressure expected in the event of a large steamline-break accident as shown in the safety analysis[3].

5. The high steamline flow limit is set at approximately 40% of the full steam flow at 0% to 20% load. Between 20% and 100% (full) load, the trip setpoint is ramped linearly with respect to first stage turbine pressure from 40% of the full steam flow to 110% of the full steam flow. These setpoints will initiate safety injection in the case of a large steamline-break accident. Coincident low T_{avg} setting limit for SIS and steamline isolation initiation is set below its hot shutdown value. The coincident steamline pressure setting limit is set below the full load operating pressure. The safety analyses show that these settings provide protection in the event of a large steamline break[3].

Instrument Operating Conditions

During plant operation, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. A channel bistable may also be placed in a bypassed mode; e.g., a two-out-of-three circuit becomes a two-out-of-two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode

$$A = \frac{W - Q\left(\frac{W}{N-M+2}\right)}{W} \doteq 1 - \frac{N!}{(N-M+2)!\,(M-1)!}\,(\lambda W)^{N-M+1}$$

where A is defined as the fraction of time during which the system is functional, and Q is the probability of failure of such a system during a time interval W.

For a 2-out-of-3 system A = 0.9999708, assuming a channel failure rate, λ, equal to $2.5 \times 10^{-6}\ hr^{-1}$ and a test interval, W, equal to 2160 hrs.

This average availability of the 2-out-of-3 system is high, hence the test interval of one quarter is acceptable.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for quarterly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

Table 4.1-1

<u>Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels</u>

	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 and Area Radiation Monitoring Systems	D	R	M	
20.	Boric Acid Make-up Flow Channel	N.A.	R	N.A.	

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
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*	
22.	Accumulator Level and 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#	
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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 154 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 October 31, 1990, 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 (TS). The requested changes to Section 3.5 and to the Bases applicable to Section 3.5 would allow routine analog channel testing in a bypassed condition instead of a tripped condition. The requested changes to Section 4.1 and to the Bases applicable to Section 4.1 would allow extension of surveillance intervals for the Reactor Protection System and the Engineered Safety Features analog channel tests from monthly to quarterly.

2.0 EVALUATION

2.1 Extension of the surveillance intervals for channel operational tests of the RPS and ESFS instrumentation from 1 month to quarterly.

In justifying the request for extending the RPS and ESFS instrumentation surveillance intervals, the licensee states that the specified surveillance intervals have been determined in accordance with WCAP-10271 (Reference 1), WCAP-10271, Supplement 1 (Reference 2), and WCAP-10271, Supplement 2, Revision 1 (Reference 3), which were approved by the NRC (References 4 and 5). As stated in the licensee's submittal, several conditions were imposed by the NRC to allow use of WCAP-10271 for amending technical specifications. The licensee's responses to these conditions are discussed in this section.

a. The licensee must implement procedures to identify common cause failures and to test other channels that may be affected by the common cause.

   The licensee has committed to modify their procedures prior to the institution of quarterly testing, to require an evaluation for common cause failure should any RPS or ESFS channel fail during its quarterly test. Additional testing of other channels in the function will be performed if a determination is made that a plausible common cause exists. The staff finds this commitment to be acceptable.

b. The instrument setpoint methodology must include sufficient adjustments to offset the drift anticipated as a result of less frequent surveillances.

The licensee based their justification for the extended surveillance intervals on the results of an evaluation of Indian Point 2 plant instrument drift data. The licensee examined a representative sample of "as found" and "as left" RPS and ESFS test data that were gathered over a period of 9 months. The data was analyzed by grouping it into approximately 3 month intervals during which no adjustments were made to the bistables. The licensee concludes, that for quarterly surveillance testing, the setpoint drift is expected to be bounded by the bistable setpoint tolerances specified in the analog channel test procedure (the licensee will retain their analyses for possible future NRC staff audit). The licensee's conclusions are acceptable to the staff.

c. The licensee shall confirm the applicability of the generic WCAP-10271 analyses to the Indian Point 2 plant.

Indian Point 2 does not have a completely installed bypass capability and has not adopted the Westinghouse Standard Technical Specifications. Nevertheless, the licensee concurs with the Westinghouse studies (References 1, 2, and 3) and the proposed TS changes. The licensee does not concur with the suggested increase (to 6 hours) in the time an inoperable channel may remain untripped. Current plant Abnormal Operating Instructions (AOIs) require the operator to place failed instruments in the tripped condition as part of and prior to completion of the instrument failure procedure which is more restrictive than the 6 hour time limit analyzed in WCAP-10271. Therefore, the licensee is not requesting an increase in the time an inoperable channel may remain untripped. The NRC staff finds the licensee's conclusion acceptable. Additionally, the NRC staff notes that the Indian Point 2 TS allows testing only one channel at a time which is more conservative that the Westinghouse analog fault tree analysis which assumes that more than one channel will be tested at a time.

2.2 Analog Channel Testing In Bypass Mode

The licensee requested NRC staff concurrence allowing routine analog channel testing in a bypassed condition. The licensee states the plant does not have full bypass testing capability but it intends to make hardware changes to provide bypass testing capability. The licensee commits to the routine testing with channels in bypass after the plant modifications are made to provide the bypass hardware. The NRC staff concurs that testing in bypass is acceptable once the bypass testing hardware is installed. This avoids the lifting of leads or installation of jumpers to perform this function.

3.0 SUMMARY

The NRC staff accepts the licensee's justification for extending the monthly surveillance intervals to a quarterly frequency. The NRC staff finds that routine analog channel testing with the channel in a bypassed condition instead of a tripped condition is acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to 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 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 (55 FR 51176). 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.

6.0 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.

7.0 REFERENCES

1) WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Time for the Reactor Protection Instrumentation System," January 1983.

2) WCAP-10271, Supplement 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," July 1983.

3) WCAP-10271, Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System," March 1987.

4) Letter from Mr. C. O. Thomas (NRC) to Mr. J. J. Sheppard (WOG), dated February 21, 1985, enclosing NRC Safety Evaluation for WCAP-10271 including Supplement 1.

5) Letter from Mr. C. E. Rossi (NRC) to Mr. R. A. Newton (WOG-WEPC), dated February 22, 1989, enclosing NRC Safety Evaluation for WCAP-10271 Supplement 2, and WCAP-10271 Supplement 2, Revision 1.

Principal Contributor:
F. Williams

Date: July 29, 1991

July 29, 1991

Docket No. 50-247

Mr. Stephen B. Bram
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Dear Mr. Bram:

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. 77930)

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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. 154 to DPR-26
2. Safety Evaluation

cc w/enclosures:
See next page

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