May 27, 1987

Docket No. 50-286

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ACRS(10)	OPA LFMB

Mr. John C. Brons
Senior Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

The Commission has issued the enclosed Amendment No. 74 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated October 10, 1986, as supplemented March 4, 1987.

The amendment revises the Technical Specifications to add limiting conditions for operation and surveillance requirements for the reactor trip breakers and reactor trip bypass breakers.

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 Slosson, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 74 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

*SEE PREVIOUS CONCURRENCE

PDI-1	PDI-1	OGC-Beth	PDI-1
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Docket No. 50-286

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Mr. John C. Brons
Senior Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated October 10, 1986, as supplemented March 4, 1987.

The amendment revises the Technical Specifications to add limiting conditions for operation and surveillance requirements for the reactor trip breakers and reactor trip bypass breakers.

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,

Joseph D. Neighbors, Senior Project Manager
PWR Project Directorate #3
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Enclosures:
1. Amendment No. to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

PD#3	PD#3	OGC	PD#3
CVogan	DNeighbors		SVarga
4/14/87	4/ /87	4/ /87	4/ /87

Mr. John C. Brons
Power Authority of the State
of New York

Indian Point Nuclear Generating
Unit No. 3

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated October 10, 1986 as supplemented March 4, 1987, 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 -1 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-64 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 74 , 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, Acting Director
Project Directorate I-1
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 27, 1987

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

ATTACHMENT TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.5-8	3.5-8
Table 3.5-2 (Sheet 2 of 2)	Table 3.5-2 (Sheet 2 of 2)
Table 4.1-1 (Sheet 1 of 5)	Table 4.1-1 (Sheet 1 of 5)
Table 4.1-1 (Sheet 2 of 5)	Table 4.1-1 (Sheet 2 of 5)
Table 4.1-1 (Sheet 3 of 5)	Table 4.1-1 (Sheet 3 of 5)
Table 4.1-1 (Sheet 4 of 5)	Table 4.1-1 (Sheet 4 of 5)
Table 4.1-1 (Sheet 5 of 5)	Table 4.1-1 (Sheet 5 of 5)
Table 4.1-3	Table 4.1-3 (Sheet 1 of 2)
-------	Table 4.1-3 (Sheet 2 of 2)

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 because a one-out-of- two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the Dropped Rod protection from NIS, for the channel being tested; and (b) defeating the ΔT protection CHANNEL SET that is being fed from the NIS channel and (c) defeating the power mismatch section of Tavg control channels when the appropriate NIS channel is being tested. However, the Rod Position System and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

In the event that either the specified Minimum Number of Operable Channels or the Minimum Degree of Redundancy cannot be met, the reactor and the remainder of the plant is placed, utilizing normal operating procedures, in that condition consistent with the loss of protection.

The source range and the intermediate range nuclear instrumentation and the turbine and steam-feedwater flow mismatch trip functions are not required to be operable since they were not used in the transient and safety analysis (FSAR Section 14).

The shunt trip features of the reactor trip and bypass breakers were modified as a result of the Salem ATWS events(4). Operability requirements for the reactor trip breakers and the reactor protection logic relays were added to the reactor protection instrument operating conditions as a result of NRC review of shunt trip modifications at Westinghouse plants(5). Operability is demonstrated when the logic coincidence relays are tested to show they are capable of initiating a reactor trip. Reactor trip breakers are considered operable when tested to show they are capable of being opened: (a) by the undervoltage device and the shunt trip device independent of each other from an automatic trip signal and (b) from the Control Room Flight Panel manual trip during refueling outages. An exception of 72 hours is allowed before a reactor trip breaker is declared inoperable if only one of the diverse trip features (undervoltage or shunt trip) fails to open the breaker when tested.

<u>References</u>:

1) FSAR - Section 7.5
2) FSAR - Section 14.3
3) FSAR - Section 14.2.5
4) GL 83-28 - Item 4.3
5) GL 85-09

TABLE 3.5-2 (Sheet 2 of 2)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Minimum Operable Channels	4 Minimum Degree of Redundancy	5 Operator Action if Conditions of Columns 3 or 4 Cannot Be Met*
9.	Lo Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown
10.	Undervoltage 6.9 KV Bus	1/bus	2	3	2	Maintain hot shutdown
11.	Low Frequency 6.9 KV Bus**	1/bus	2	3	2	Maintain hot shutdown
12.	Turbine Trip:					
	a. Electrical overspeed protection	3	2	2	1	Turbine shutdown (turbine valves closed)
	b. Low auto stop oil pressure	3	2	2	1	Maintain reactor power below 10% of full power
13.	Reactor Trip Breakers***	2	1	2	1	Maintain hot shutdown****
14.	Reactor Protection Relay Logic	2	1	2	1	Maintain hot shutdown****

* Maintain hot shutdown means maintain or proceed to hot shutdown within 4 hours using normal operating procedures, if the unacceptable condition arises during operation.

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

*** A reactor trip breaker is considered inoperable if any of its components fail to meet test specifications. If either the undervoltage or shunt trip device (not both) prevent a breaker from proper operation, then 72 hours are allowed to restore the failed device to operable status before the affected breaker is declared inoperable.

**** Upon proceeding to hot shutdown as a result of an inoperable reactor trip breaker or relay logic, 48 hours are allowed to restore the minimum number of operable channels required by column 3. If minimum operability is not restored after this 48 hour period, rod withdrawal capability shall be defeated within one hour by opening one reactor trip breaker and its associated bypass breaker or isolating power to the Control Rod Drive System.

TABLE 4.1-1 (Sheet 1 of 5)

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)*	M (2)** M (4)	1) Heat balance calibration 2) Bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset 4) Signal to ΔT
2.	Nuclear Intermediate Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Verification of channel response to simulated inputs
3.	Nuclear Source Range	S (1)	N.A.	P (2)	1) Once/shift when in service 2) Verification of channel response to simulated inputs
4.	Reactor Coolant Temperature	S	R	M (1) M (2)	1) Overtemperature - ΔT 2) Overpower - ΔT
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure	S	R	M	High and Low
8.	6.9 KV Voltage & Frequency	N.A.	R	M	Reactor protection circuits only
9.	Analog Rod Position	S	R	M	

Table 4.1-1 (Sheet 2 of 5)

Channel Description	Check	Calibrate	Test	Remarks
10. Steam Generator Level	S	R	M	
11. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
12. Boric Acid Tank Level	S	R	N.A.	Bubbler tube rodded during calibration
13. Refueling Water Storage Tank Level	W	R	N.A.	Low level alarms
14. Containment Pressure	S	R	M	High and High-High
15. Process and Area Radiation Monitoring Systems	D	R	Q	
16. Containment Water Level Monitoring System:				
a. Containment Sump	N.A.	R	N.A.	Narrow Range, Analog
b. Recirculation Sump	N.A.	R	N.A.	Narrow Range, Analog
c. Containment Water Level	N.A.	R	N.A.	Wide Range
17. Accumulator Level and Pressure	S***	R	N.A.	
18. Steam Line Pressure	S	R	M	
19. Turbine First Stage Pressure	S	R	M	
20. Reactor Protection Relay Logic	N.A.	N.A.	TM	
21. Turbine Trip:				
a. Independent Overspeed	N.A.	R	M	Electrical
b. Low Auto Stop Oil Pressure	N.A.	R	N.A.	
22. Boron Injection Tank Return Flow	S	R	N.A.	

Table 4.1-1 (Sheet 3 of 5)

Channel Description	Check	Calibrate	Test	Remarks
23. Temperature Sensor in Auxiliary Boiler Feedwater Pump Building	N.A.	N.A.	R	
24. Temperature Sensors in Penetration Area of Primary Auxiliary Building	N.A.	N.A.	R	
25. Level Sensors in Turbine Building	N.A.	N.A.	R	
26. Volume Control Tank Level	N.A.	R	N.A.	
27. Boric Acid Makeup Flow Channel	N.A.	R	N.A.	
28. Auxiliary Feedwater:				
a. Steam Generator Level	S	R	M	Low-Low
b. Undervoltage	N.A.	R	R	
c. Main Feedwater Pump Trip	N.A.	N.A.	R	
29. Reactor Coolant System Subcooling Margin Monitor	D	R	N.A.	
30. PORV Position Indicator	N.A.	R	R	Limit Switch
31. PORV Position Indicator	D	R	R	Acoustic Monitor
32. Safety Valve Position Indicator	D	R	R	Acoustic Monitor
33. Auxiliary Feedwater Flow Rate	N.A.	R.	N.A.	

Table 4.1-1 (Sheet 4 of 5)

Channel Description	Check	Calibrate	Test	Remarks
34. Plant Effluent Radio-iodine/ Particulate Sampling	N.A.	N.A.	R	Sample line common with monitor R-13
35. Loss of Power				
a. 480v Bus Undervoltage Relay	N.A.	R	M	
b. 480v Bus Degraded Voltage Relay	N.A.	R	M	
c. 480v Safeguards Bus Undervoltage Alarm	N.A.	R	M	
36. Main Steam Line Radiation Monitors	D	R	Q	R-62A, B, C, D
37. Containment Hydrogen Monitors	D	Q	M	
38. Wide Range Plant Vent Monitor	D	R	Q	R-27
39. High Range Containment Radiation Monitors	D	R	Q	R-25, R-26
40. Core Exit Thermocouples	D	N.A.	N.A.	
41. Overpressure Protection System (OPS)	D	R	R	
42. Reactor Trip Breakers	N.A.	N.A.	TM(1)	1) Independent operation of undervoltage and shunt trip attachments
			R(2)	2) Independent operation of undervoltage and shunt trip from Control Room manual push-button

Table 4.1-1 (Sheet 5 of 5)

	Channel Description	Check	Calibrate	Test	Remarks
43.	Reactor Trip Bypass Breakers	N.A.	N.A.	(1)	1) Manual shunt trip prior to each use
				R(2)	2) Independent operation of undervoltage and shunt trip from Control Room manual push-button
				R(3)	3) Automatic undervoltage trip

* By means of the movable incore detector system

** Monthly when reactor power is below the setpoint and prior to each startup if not done previous month.

*** If either an accumulator level or pressure instrument channel is declared inoperable, the remaining level or pressure channel must be verified operable by interconnecting and equalizing (pressure and/or level wise) a minimum of two accumulators and crosschecking the instrumentation.

S - Each shift
P - Prior to each startup if not done previous week
NA- Not applicable
D - Daily
TM- At least every two months on a staggered test basis (i.e., one train per month)
W - Weekly
M - Monthly
Q - Quarterly
R - Each refueling outage

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS

		Check	Frequency
1.	Control Rods	Rod drop times of all control rods	R
2.	Control Rods	Partial movement of all control rods	Every 2 weeks during reactor critical operations
3.	Pressurizer Safety Valves	Set Point	R
4.	Main Steam Safety Valves	Set Point	R
5.	Containment Isolation System	Automatic actuation	R
6.	Refueling System Interlocks	Functioning	R (Prior to movement of core components)
7.	Primary System Leakage	Evaluate	5 days/week
8.	Diesel Generators Nos. 31, 32, & 33 Fuel Supply	Fuel Inventory	Weekly
9.	Turbine Steam Stop Control Valves	Closure	Monthly
10.	L. P. Steam Dump System (6 lines)	Closure	Monthly
11.	Service Water System	Each pump starts and operates for 15 minutes (unless already operating)	Monthly
12.	City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	R
13.	RHR Valves 730 and 731	Automatic isolation and interlock action	R*
14.	PORV Block Valve	Operability through 1 complete cycle of full travel	R
15.	PORV Valves	Operability	R
16.	Reactor Vessel Head Vents	Operability	R

R Each Refueling Outage

* If not done during the previous 18 months, the check will be performed next time the plant is cooled down.

Table 4.1-3 (Sheet 2 Of 2)

FREQUENCIES FOR EQUIPMENT TESTS

		CHECK	FREQUENCY
17.	Reactor Trip Breakers	a. Independent operation of undervoltage and shunt trip attachments	Bimonthly*
		b. Independent undervoltage and shunt trip tests from manual trip	R
18.	Reactor Trip Bypass Breakers	a. Manual shunt trip	Prior to each use
		b. Independent undervoltage and shunt trip tests from manual trip	R
		c. Automatic undervoltage trip	R

R Each Refueling Outage

* Once every two months on a staggered test basis

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

INTRODUCTION

By letters dated October 10, 1986 and March 4, 1987, the Power Authority of the State of New York (the licensee) requested an amendment to Facility Operating License DPR-64 for the Indian Point Nuclear Generating Unit No. 3. These proposed changes would revise the Technical Specifications to add limiting conditions for operation and surveillance requirements for the reactor trip and reactor trip bypass breakers.

BACKGROUND

Generic Letter 85-09 specifies Technical Specification changes applicable to the Reactor Trip System Instrumentation and Surveillance. Generic Letter 85-09 concluded that Technical Specification changes should be proposed by licensees to explicitly require independent testing of the undervoltage and shunt trip attachments of the reactor trip breakers during power operation, testing of bypass breakers prior to use, and independent testing of the control room manual switch contacts and wiring during each refueling outage.

EVALUATION

The licensee proposed the following changes:

Table 3.5.2, Functional Unit 13 (Reactor Trip Breakers), Functional Unit 14 (Reactor Protection Relay Logic).

Table 4.1-1, Channel Description 20 (Reactor Protection Relay Logic), Channel Description 42 (Reactor Trip Breakers), Channel Description 43 (Reactor Trip Bypass Breakers).

We have compared the licensee's proposed Technical Specification changes with those of Generic Letter 85-09 and find that they are consistent with the requirements of Generic Letter 85-09.

The Table 3.5-2 proposed changes pertain to limiting conditions for operation and associated action statements if the minimum operability conditions are not satisfied. The action statements would require going to hot shutdown within 4 hours and subsequent opening of the reactor trip breakers within 48 hours if the inoperable condition persisted. If only one of the diverse trip features of a reactor trip breaker became inoperable, the action statement would permit 72 hours before proceeding to hot shutdown within an additional 4 hours.

Although Generic Letter 85-09 specifies an allowable 48 hours when one of the diverse trip features is inoperable, we find that the extension of time to 72 hours would have a negligible impact on reliability, since the other trip feature would have been found to be operable. We therefore find the 72 hours to be acceptable.

The Table 4.1-1 proposed modifications would require verification of operability of the manual trip circuits of the reactor trip breakers (RTBs) and bypass breakers during refueling outages, including independent verification of the undervoltage and shunt trip circuitry. The Table 4.1-1 proposed modifications would also require on-line testing of the RTBs, including independent testing of the undervoltage and shunt trip functions, and on-line testing of the automatic trip logic channels for each train, at least every two months on a staggered test basis. They also provide for testing of the operability of the reactor trip bypass breaker prior to placing the breaker in service.

The licensee also proposes an addition to the bases section, editorial changes involving format changes, moving and removing footnotes and relocating items on the pages. We find these changes add to the overall clarity of the specifications and are acceptable.

SUMMARY

We have reviewed the Indian Point 3 Technical Specification changes proposed by the licensee by letters dated October 10, 1986 and March 4, 1987, including the action statements and footnotes, and find they are consistent with those of Generic Letter 85-09. We find that the proposed changes enhance plant reliability and safety, and therefore find them to be acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 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: May 27, 1987

PRINCIPAL CONTRIBUTOR:

Argil Toalston