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. 107
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 December 20, 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-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.107 , 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: March 22, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 107

FACILITY OPERATING CENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.5-8	3.5-8
4.1-4	4.1-4
--	4.1-5
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 5 of 5)	Table 4.1-1 (sheet 5 of 5)

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 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 T_{avg} 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

Testing

The minimum testing frequency for those instrument channels connected to the safety system is based on an average unsafe failure rate of 2.5×10^{-6} failure/hrs. per channel. This is based on operating experience at conventional and nuclear plants. An unsafe failure is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

For a specified test interval W and an M out of N redundant system with identical and independent channels having a constant failure rate λ, the average availability A is given by:

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

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 2.5×10^{-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.

Specified surveillance intervals for the Reactor Protection System and Engineered Safety Features have been determined in accordance with WCAP - 10271, Supplement 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and WCAP - 10271, Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System," as approved by the NRC and documented in the SERs (letters to J. J. Sheppard from C. O. Thomas, dated February 21, 1985, and to R. A. Newton from C. E. Rossi, dated February 22, 1989). Surveillance intervals were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The Turbine Steam Stop and Control Valves shall be tested at a frequency determined by the methodology presented in WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency", and in accordance with established NRC acceptance criteria for the probability of a missile ejection incident at IP-3. In no case shall the test interval for these valves exceed one year.

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)*	Q (2)** Q (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	Q (1) Q (2)	1) Overtemperature - ΔT 2) Overpower - ΔT
5. Reactor Coolant Flow	S	R	Q	
6. Pressurizer Water Level	S	R	Q	
7. Pressurizer Pressure	S	R	Q	High and Low
8. 6.9 KV Voltage & Frequency	N.A.	R	Q	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	Q	
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	Q	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	Q	
19. Turbine First Stage Pressure	S	R	Q	
20. Reactor Protection Relay Logic	N.A.	N.A.	TM	
21. Turbine Trip 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 Primary Auxiliary Building				
a. Piping Penetration Area	N.A.	N.A.	R	
b. Mini-Containment Area	N.A.	N.A.	R	
c. Steam Generator Blowdown Heat Exchanger Room	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	Q	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 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 Primary Auxiliary Building				
a. Piping Penetration Area	N.A.	N.A.	R	
b. Mini-Containment Area	N.A.	N.A.	R	
c. Steam Generator Blowdown Heat Exchanger Room	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	Q	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.	

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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
44. Reactor Vessel Level Indication System (RVLIS)	D	R	N.A.	
45. Ambient Temperature Sensors Within the Containment Building	D	R	N.A.	
46. River Water Temperature # (installed)	S	R	N.A.	1) Check against installed instrumentation or another portable device
47. River Water Temperature # (portable)	S (1)	Q (2)	N.A.	2) Calibrate within 30 days prior to use and quarterly thereafter

* By means of the movable incore detector system

** Quarterly 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.

\# These requirements are applicable when specification 3.3.F.5 is in effect only.

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