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OCT 7 1981

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Docket No. 50-286

Mr. George T. Berry, President
and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 38 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated December 30, 1980.

The amendment revises the license and the Technical Specifications to incorporate certain of the TMI-2 Lessons Learned Category "A" requirements.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed By:

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 38 to DPR-64
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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9/28/81

Concur subject to changes

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DATE	8/13/81	9/25/81	9/ /81	9/29/81	10/6/81	9/28/81	

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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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 STATION UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
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 30, 1980, 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 by renumbering paragraph 2.L of Facility Operating License No. DPR-64 to paragraph 3, and by amending paragraph 2.C.(2) and adding paragraphs 2.L, 2.M and 2.N to read as follows:

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 38, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   2.L The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

   1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

   2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

   2.M The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

   1. Training of personnel,

   2. Procedures for monitoring, and

   3. Provisions for maintenance of sampling and analysis equipment.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 7, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1-2	3.1-2
3.1-3	3.1-3
Table 3.5-1	Table 3.5-1
Table 3.5-3 (Sheet 1 and 2)	Table 3.5-3 (Sheet 1, 2 and 3)
Table 3.5-5 (both sheets)	Table 3.5-5 (both sheets)
Table 4.1-1 (Sheets 1, 2 and 3)	Table 4.1-1 (Sheets 1, 2, 3 and 4)
Table 4.1-3	Table 4.1-3
4.8-1	4.8-1
6-1	6-1
Figure 6.2-2	Figure 6.2-2
6-4	6-4
6-5	6-5

2. Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is above the cold shutdown condition except during reactor coolant system hydrostatic tests and/or safety valve settings.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with $\pm$ 1% allowance for error.

3. Pressurizer Heaters

Whenever the reactor is above the hot shutdown condition, the pressurizer shall be operable with at least 150 kw of pressurizer heaters.

a. With less than 150 kw of pressurizer heaters operable, restore the required inoperable heaters within 72 hours or be in at least hot shutdown within an additional 6 hours.

4. Power Operated Relief Valves

Whenever the reactor coolant system is above 400°F, the power operated relief valves (PORVs) shall be operable or their associated block valves closed.

a. If the block valve is closed because of an inoperable PORV, the control power for the block valve must be removed.

b. If the above conditions cannot be satisfied within 1 hour, be in at least hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

5. Power Operated Relief Block Valves

Whenever the reactor coolant system is above 400°F, the motor operated block valves shall be operable or closed.

a. If the block valve is inoperable, the control power is to be removed.

b. If the above conditions cannot be satisfied within 1 hour be in at least hot shutdown within the following 30 hours.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only (1); hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

Each of the pressurizer code safety valves is designed to relieve 420,000 lbs. per hr. of saturated steam at the valve set point.

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load (2) without a direct reactor trip or any other control.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.

References

1) FSAR Section 14.1.6

2) FSAR Section 14.1.8

TABLE 3.5-1

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

No.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMIT
1.	High Containment Pressure (Hi Level)	Safety Injection	$\leq$3.5 psig
2.	High Containment Pressure (Hi-Hi Level)	a. Containment Spray	$\leq$23 psig
		b. Steam Line Isolation	
3.	Pressurizer Low Pressure	Safety Injection	$\geq$1700 psig
4.	High Differential Pressure Between Steam Lines	Safety Injection	$\leq$150 psi
5.	High Steam Flow in 2/4 Steam Lines Coicident with Low T_{avg} of Low Steam Line Pressure	a. Safety Injection	$\leq$40% of full steam flow at zero load
		b. Steam Line Isolation	$\leq$40% of full steam flow at 20% load
			$\leq$110% of full steam flow at full load
			$\geq$540^{o}F T_{avg}
			$\geq$600 psig steam line pressure
6.	Steam Generator Water Level (low-low)	Auxiliary Feedwater	$\geq$5% of narrow range instrument span each steam generator
7.	Undervoltage	Auxiliary Feedwater	$\geq$40% nominal voltage

TABLE 3.5-3 (SHEET 1 of 3)

INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

NO. FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. NUMBER OF OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET*****
1. SAFETY INJECTION					
a. Manual	2	1	1	0	Cold Shutdown
b. High Containment Pressure (Hi Level)	3	2	2	1	Cold Shutdown
c. High Differential Pressure Between Steam Lines	3/steam line	2/steam line	2/steam line	1/steam line	Cold Shutdown
d. Pressurizer Low Pressure	3	2	2	1	Cold Shutdown
e. High Steam Flow in 2/4 Steam Lines Coincident with Low Tavg or Low Steam Line Pressure	2/steam line	1/2 in any 2 steam lines	2 channels in each of 3 steam lines	2	Cold Shutdown or main steam isolation valves closed
	4 Tavg Signals	2	3	2	
	4 Pressure Signals	2	3	2	
f. Pressurizer Low Pressure and (Automatic Unblock)	3	2	2****	1****	Cold Shutdown

TABLE 3.5-3 (SHEET 2 of 3)

		1	2	3	4	5
2.	CONTAINMENT SPRAY					
a.	Manual	2	2	2	0***	Cold Shutdown
b.	High Containment Pressure (Hi Hi Level)	2 sets of 3	2 of 3 in each set	2 per set	1/set	Cold Shutdown

* Permissible to bypass if reactor coolant pressure less than 2000 psig.

*** Must actuate 2 switches simultaneously.

**** The minimum Number of Operable Channels and the Minimum Degree of Redundancy may be reduced to zero if the SI bypass is in the unblocked position.

***** If the condition of Column 3 or 4 cannot be met, the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within 4 hours of the occurrence. If the conditions are not mer within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition, if applicable, within an additional 24 hours.

FUNCTIONAL UNIT	1	2	3	4	5
3. AUXILIARY FEEDWATER					
a. Stm Gen. Water Level-Low-Low					
i. Start Motor Driven Pumps	3/stm. gen	2 in any stm gen.	2 chan. in each stm gen	1	Reduce system temperature such that $T \leq 350^{o}F$
ii. Start Turbine-Driven Pump	3/stm. gen	2/3 in each of two stm. gen.	2 chan. in each stm. gen.	1	$T \leq 350^{o}F$
b. S. I. Start Motor-Driven Pumps	(All safety injection initiating functions and requirements)				
c. Station Blackout Start Turbine-Driven Pump	2	1	1	0	$T \leq 350^{o}F$
d. Trip of Main Feedwater Pumps start Motor-Driven Pumps	2	1	1	0	Hot Shutdown
4. CONTAINMENT VENT AND PURGE					
a. Containment Radioactivity-High	2	1	1	0	Close all containment vent and purge valves
5. LOSS OF POWER					
a. 480 V Bus	2/Bus	1/bus	1/bus	0	Hot Shutdown

TABLE 3.5-5 (Sheet 1 of 2)

TABLE OF INDICATORS AND/OR RECORDERS AVAILABLE TO THE OPERATOR

		1	2	3
	PARAMETER	NO. OF CHANNELS AVAILABLE	MIN. NO. OF CHANNELS REQUIRED**	INDICATOR/ RECORDER**
1)	Containment Pressure	6	1	Indicator
2)	Refueling Water Storage Tank Level	2	1	Indicator
3)	Steam Generator Water Level (Narrow Range)	3/Steam generator	*	Indicator
4)	Steam Generator Water Level (Wide Range)	1/steam generator	*	Recorder
5)	Steam Line Pressure	3/steam	1/steam	Indicator
6)	Pressurizer Water Level	3	2	Indicator/One Channel is Recorded
7)	RHR Recirculation Flow	4	3	Indicator
8)	Reactor Coolant System Pressure (Wide Range)	1	1	Recorder
9)	Cold Leg Temperature (Tc) (Wide Range)	4	1	Recorder
10)	Hot Leg Temperature (Th) (Wide Range)	4	1	Recorder
11)	Containment Sump Level	2	1	Indicator
12)	Recirculation Sump Level	2	1	Indicator
13)	Temperature Sensors in Penetration Area of Primary Auxiliary Building	3	1	Alarm
14)	Temperature Sensors in Auxiliary Boiler Feedwater Pump Building	2	1	Alarm

TABLE 3.5-5 (Sheet 2 of 2)

		1	2	3
15)	Level Sensors in Lower Level of Turbine Building	2	1	Alar
16)	Reactor Coolant System Subcooling Margin Monitor	1	1	Reco
17)	PORV Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	Indi
18)	PORV Position Indicator (Limit Switch)	1/Valve	1/Valve****	Indi and
19)	PORV Block Valve Position Indicator (Limit Switch)	1/Valve***	1/Valve	Indi
20)	Safety Valve Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	Indi
21)	Auxiliary Feedwater Flow Rate	1/Pump	1/Pump	Indi

*One level channel per steam generator (either wide range or narrow range) with at least two wide range channels.

**Columns 2 and 3 may be modified to allow the instrument channels to be inoperable for up to 7 days and/or the recorder to be inoperable for up to 14 days.

***Except at times when valve operator control circuit is de-energized.

****Except when the respective block valve is closed.

If the minimum number of channels required are not restored to meet the above requirements within the time periods specified, then:

1. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

2. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

3. In either case, if the requirements of Columns 2 and 3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

TABLE 4.1-1 (Sheet 1 of 4)

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 off-set 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) (2)	1) Overtemperature - ΔT 2) Overpower - ΔT
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure(High and Low)	S	R	M	
8.	6.9 Kv Voltage & Frequency	N.A.	R	M	Reactor protection circuits only
9.	Analog Rod Position	S	R	M	

* 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.

TABLE 4.1-1 (Sheet 2 of 4)

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. (a) Containment Pressure	S	R	M	High
(b) Containment Pressure	S	R	M	High High
15. Process and Area Radiation Monitoring Systems	D	R	Q	
16. Containment and Recirculation Sump Level	N.A.	N.A.	R	
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. Logic Channel testing	N.A.	N.A.	M	
21. Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
22. Boron Injection Tank Return Flow	S	R	N.A.	

*** 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 cross-checking the instrumentation.

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
23. Temperature Sensors 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 Make-Up Flow Channel	N.A.	R	N.A.	
28. Auxiliary Feedwater:				
a. Steam Generator Water Level (Low-Low)	S	R	M	
b. Undervoltage	N.A.	R	R	
c. Trip of Main Feedwater Pumps	N.A.	N.A.	R	

TABLE 4.1-1 (SHEET 4 of 4)

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
29. Reactor Coolant System Subcooling Margin Monitor	D	R	N.A.	
30. PORV Position Indicator (Limit Switch)	N.A.	R	R	
31. PORV Position Indicator (Acoustic Monitor)	D	R	R	
32. Safety Valve Position Indicator (Acoustic Monitor)	D	R	R	
33. Auxiliary Feedwater Flow Rate	N.A.	R	N.A.	

S - Each Shift
D - Daily
W - Weekly
M - Monthly
P - Prior to each startup if not done previous week
Q - Quarterly
R - Each Refueling Outage
NA - Not Applicable

TABLE 4.1-3

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	Prior to each refueling outage
7.	Primary System Leakage	Evaluate	5 days/week
8.	Diesel 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.	Block Valve	Operability through 1 complete cycle of full travel	R
15.	PORV Valves	Operability	R

R Each Refueling Outage

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

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater System.

Objective

To verify the operability of the Auxiliary Feedwater System and its ability to respond properly when required.

Specification

1.a. Each auxiliary feedwater pump will be started manually from the control room at monthly intervals with full flow established to the steam generators once every refueling.

b. The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

c. Backup supply valves from the city water system will be tested once every refueling.

2. Acceptance levels of performance shall be that the pumps start, reach their required developed head and operate for at least fifteen minutes.

3. At least once every refueling outage,

a. Verify that the recirculation valve will actuate to its correct position.

b. Verify that each auxiliary feedwater pump will start as designated automatically upon receipt of each auxiliary feedwater actuation test signal.

Basis

The testing of the auxiliary feedwater pumps will verify their operability. The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Resident Manager shall be responsible for overall facility operation. During periods when the Resident Manager is unavailable, the Superintendent of Power will assume his responsibilities. In the event both are unavailable, the Resident Manager may delegate this responsibility to other qualified supervisory personnel.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for plant management and technical support shall be as shown on Figure 6.2-1.

PLANT STAFF

6.2.2 The Plant organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one Licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. ALL CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A Fire Brigade of at least five members shall be maintained on site at all times. This excludes four members of the minimum shift crew necessary for safe shutdown of the plant and any personnel required for other essential functions during a fire emergency. During periods of cold shutdown the Fire Brigade will exclude two members of the minimum shift crew.

g. One shift Technical Advisor shall be on site at all times during which the reactor is above the cold shutdown condition.

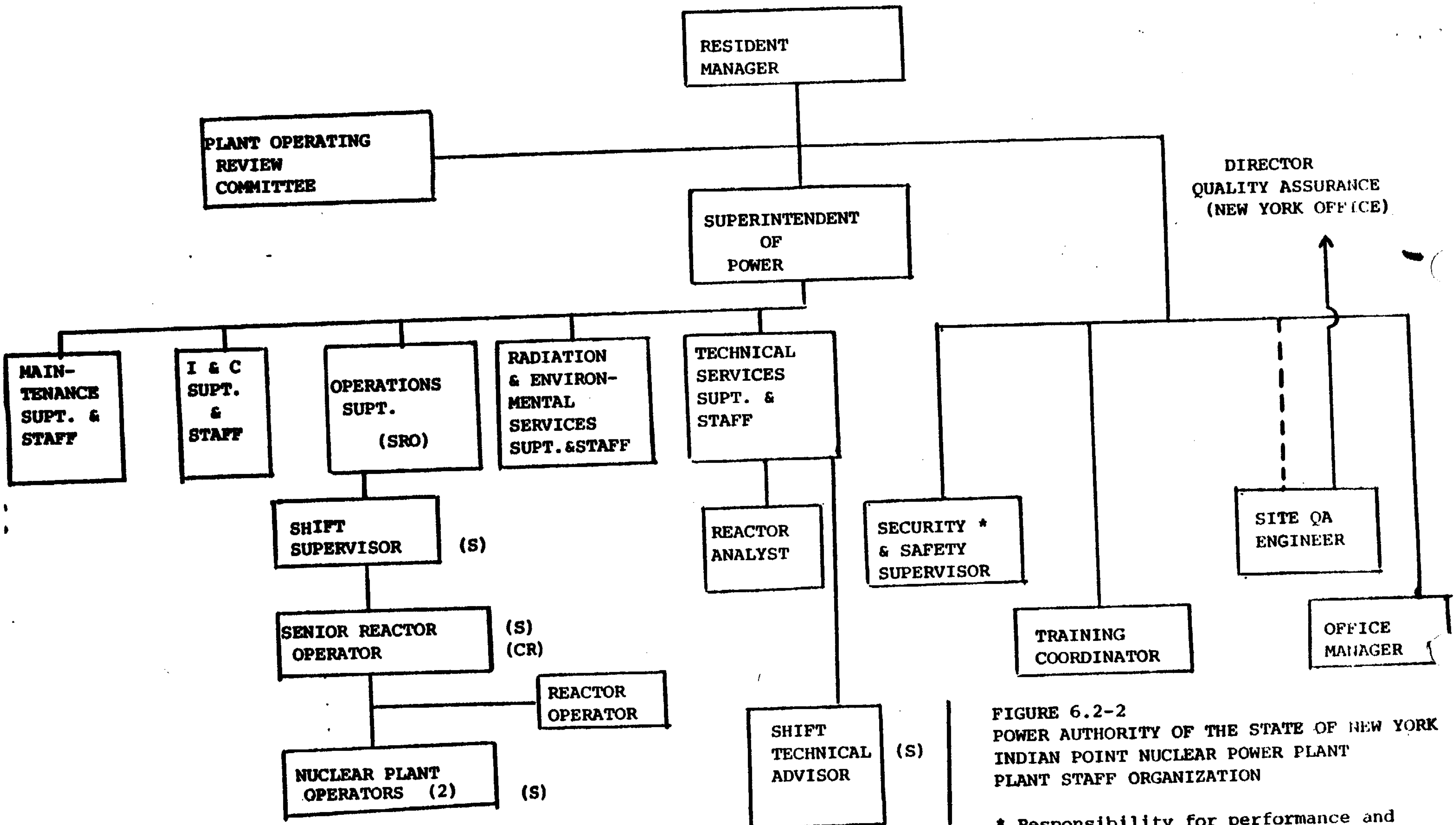

FIGURE 6.2-2
POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT NUCLEAR POWER PLANT
PLANT STAFF ORGANIZATION

* Responsibility for performance and monitoring of the fire protection program.

(S) CONTINUOUS COVERAGE
(CR) CONTROL ROOM
(SRO) SENIOR REACTOR OPERATOR

Amendment No. ~~5~~, ~~12~~ 38

Table 6.2-1

Minimum Shift Crew Composition *

License Category	During Operations Involving Core Alterations	During Cold Shutdown or Refueling Periods	At All Other Times
Senior Operator License	2**	1	1
Operator License	1	1	2
Non-Licensed	(As Required)	1	2
Shift Technical Advisor	None Required	None Required	1

* Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of this Table.

** Includes individual with SRO license supervising fuel movement as per Section 6.2.2e.

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 Each member of the plant staff shown in Fig. 6.2-2 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Radiation and Environmental Services Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the training coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 or ANSI N18.1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Coordinator and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976 with the exception of the training program schedule.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATING REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The Plant Operating Review Committee shall function to advise the Resident Manager on all matters related to nuclear safety and all matters which could adversely change the plants environmental impact.

Amendment No. ~~5~~, ~~10~~, ~~11~~, ~~12~~ 38

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

I. INTRODUCTION

By letter dated December 30, 1980, the Power Authority of the State of New York (the licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-64 for Indian Point 3. The changes involve the incorporation of certain of the TMI-2 Lessons Learned Category "A" requirements. The licensee's request is in direct response to the NRC staff's letter dated July 2, 1980.

II. BACKGROUND INFORMATION

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter dated February 21, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TS to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable. The licensee's application is in direct response to our request. Each of the issues identified by the NRC staff and the licensee's response is discussed in the Evaluation below.

III. EVALUATION

2.1.1 Emergency Power Supply Requirements

The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in a post-accident situation. Adequate emergency power supplies add assurance of post-accident functioning of these components. The licensee has the requisite emergency power supplies.

The licensee has proposed actions in the event of component inoperability. We have reviewed these proposed TSs and find that the emergency power supplies are reasonably ensured for post-accident functioning of the subject components and are thus acceptable.

2.1.3.a Direct Indication of (of Flow) Valve Position

The licensee has provided a direct indication of power-operated relief valve (PORV) and safety valve position in the control room. These indications are a diagnostic aid for the plant operator and provide no automatic action. The licensee has provided TSs with a daily channel check and an 18-month channel calibration requirement; thus, the TSs are acceptable and they meet our July 2, 1980 model TS criteria.

2.1.3.b Instrumentation for Inadequate Core Cooling

The licensee has installed a reactor coolant system subcooling margin monitor. The licensee submitted TSs with daily channel check and an 18-month channel calibration requirement and actions to be taken in the event of component inoperability. We conclude the TSs are acceptable as they meet our July 2, 1980 model TS criteria.

2.1.7.a Auto Initiation of Auxiliary Feedwater Systems

The plant has provisions for the automatic initiation of auxiliary (emergency) feedwater flow on loss of normal feedwater flow. The TSs submitted by the licensee list the appropriate components, describe the tests and provide for proper test frequency. The TSs contain appropriate actions in the event of component inoperability; therefore, we conclude that the TSs are acceptable.

2.1.7.b Auxiliary (Emergency) Feedwater Flow Indication

The licensee has auxiliary (emergency) feedwater flow indication that meets our testability and vital power requirements. We reviewed this system in our Safety Evaluation dated February 21, 1980. The licensee has proposed a TS with an 18-month channel calibration requirement. We find this TS acceptable.

2.2.1.b Shift Technical Advisor (STA)

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The licensee's application would add one STA to each shift, except for cold shutdowns and refuelings, to perform the function of accident assessment. This clarifies the requirement in the February 11, 1980 Confirmatory Order that the STA be "on shift."

The individual performing this function will have at least a bachelor's degree or equivalent in a scientific or engineering discipline with special training in plant design, and response and analysis of the plant for transients and accidents. Part of the STA duties are related to operating experience review function. Based on our review, we find the licensee's submittal satisfies our requirements and is acceptable.

EVALUATION TO SUPPORT LICENSE CONDITIONS

2.1.4 Integrity of Systems Outside Containment

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition would require the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems, which are located outside reactor containment. By letter dated December 30, 1980, the licensee agreed to adopt such a license condition; accordingly we have included this condition in the license.

2.1.8.c Iodine Monitoring

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to iodine monitoring. Such a condition would require the licensee to effect a program which would ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions. By letter dated December 30, 1980, the licensee agreed to adopt such a license condition; accordingly. we have included this condition in the license.

IV. ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

V. CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because that amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: October 7, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-286

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 38 to Facility Operating License No. DPR-64, issued to the Power Authority of the State of New York (the licensee), which revised the license and the Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 3 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment revises the license and the Technical Specifications to incorporate certain of the TMI-2 Lessons Learned Category "A" requirements.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated December 30, 1980, (2) Amendment No. 38 to License No. DPR-64, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Thehesda, Maryland, this 7th day of October 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

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