April 18, 1986

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
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Dear Mr. Brons:

The Commission has issued the enclosed Amendment No. 65 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 applications transmitted by letter dated February 17, 1984 and October 4, 1985, as supplemented February 12, 1986.

The amendment revises the Technical Specifications to incorporate requirements related to NUREG-0737 and Generic Letter 83-37 dated November 1, 1983.

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,

/s/JDNeighbors

Joseph D. Neighbors, Senior Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A

Enclosures:
1. Amendment No. 65 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

OFC	:PAD#3	:PAD#3	:OELD	:D/PAD#3	:	:	:
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P PDR

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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. 65
License No. DPR-64

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

   A. The applications for amendment by Power Authority of the State of New York (the licensee) dated February 17 and October 4, 1984, as supplemented February 12, 1986, comply 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 applications, 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. 65 , 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

for Steven A. Varga, Director
PWR Project Directorate #3
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 65

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
————	3.1-2a
3.1-3a	3.1-3a
3.5-2	3.5-2
Table 3.5-4 (Sheet 2 of 2)	Table 3.5-4 (Sheet 2 of 2)
Table 3.5-5 (Sheet 1 of 2)	Table 3.5-5 (Sheet 1 of 3)
Table 3.5-5 (Sheet 2 of 2)	Table 3.5-5 (Sheet 2 of 3)
————	Table 3.5-5 (Sheet 3 of 3)
Table 4.1-1 (Sheet 1 of 4)	Table 4.1-1 (Sheet 1 of 5)
Table 4.1-1 (Sheet 2 of 4)	Table 4.1-1 (Sheet 2 of 5)
Table 4.1-1 (Sheet 3 of 4)	Table 4.1-1 (Sheet 3 of 5)
Table 4.1-1 (Sheet 4 of 4)	Table 4.1-1 (Sheet 4 of 5)
————	Table 4.1-1 (Sheet 5 of 5)
Table 4.1-3	Table 4.1-3
6-5	6-5
6-13a	6-13a
6-16	6-16
6-17	6-17

7. REACTOR VESSEL HEAD VENTS

Whenever the reactor coolant system is above 350°F, two reactor vessel head vent paths consisting of two valves in series with power available from emergency buses shall be OPERABLE.

a. If one of the above reactor vessel head vent paths is inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path. Restore the inoperable vent path to operable status within 90 days, or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

b. With both reactor vessel head vent paths inoperable restore one vent path to operable status within 7 days or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

T_{cold} of 546.9^{O}F at a flow of 323,600 gpm. Restricting maximum Tavg to 576^{O}F (indicated) at all power levels will preserve the steady-state DNB margins assured in WCAP-10704.

Reactor vessel head vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November, 1980.

References

1) FSAR Section 14.1.6

2) FSAR Section 14.1.8

3.5.4 In the event of instrumentation channel failure permitted by specification 3.5.2, the Minimum Degree of Redundancy listed in Tables 3.5-2 through 3.5-4 may be reduced by one, but to not less than zero, and the Minimum Number of Operable Channels listed in these tables may be reduced by one, but not to less than one (except as noted in Table 3.5-3) for a period of 8 hours while instrument channels are tested. The failed channel may be blocked to prevent an unnecessary reactor trip during this time. . In the case of three loop operation, the out-of-service channel is permitted to be bypassed during the test period.

3.5.5 The low pressurizer pressure safety injection trip shall be unblocked when the pressurizer pressure is $\geq$ 2000 psig.

3.5.6 At least one source range and one intermediate range nuclear instrument channel shall be operable prior to reactor start-up.

3.5.7 When the reactor is not in the cold shutdown condition, the instrumentation requirements as stated in Table 3.5-5 shall be met.

3.5.8 A minimum of two channels of containment pressure must be operable when T_{avg} is greater than $350^{\circ}F$.

TABLE 3.5-4 (Sheet 2 of 2)

	1	2	3	4	5
3. FEEDWATER LINE ISOLATION a. Safety Injection	See Item No. 1 of Table 3.5-3				
4. CONTAINMENT VENT AND PURGE					
a. Containment Radioactivity High (R11 and R12 monitor)	2	1	1	0	close all containment vent and purge valves when above cold shutdown
5. PLANT EFFLUENT RADIOIODINE/ PARTICULATE SAMPLING (sample line common with monitor R13)	1	NA	1	0	(see note 3)
6. Main Steam Line Radiation Monitors	1/line	NA	1/line	0	(see note 3)
7. Wide Range Plant Vent Monitor (R27)	1	NA	1	0	(see note 3)

NOTES

1. If the conditions of Columns 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 met 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.

2. Main steam isolation valves may be closed in lieu of going to cold shutdown if the circuitry associated with closing the valves is the only portion inoperable.

3. If the plant vent sampling capability, the wide-range vent monitor or the main steam line radiation monitors is/are: determined to be inoperable when the reactor is above the cold shutdown condition, then restore the sampling/monitoring capability within 72 hours or:

   a) Initiate a pre-planned alternate sampling/monitoring capability as soon as practical, but no later than 72 hours after identification of the failures. If the capability is not restored to operable status within 7 days, then,
   b) Submit a Special Report to the NRC pursuant to Technical Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system.

TABLE 3.5-5 (SHEET 1 of 3)

TABLE OF INDICATORS AND/OR RECORDERS AVAILABLE TO THE OPERATOR

PARAMETER	1 NO. OF CHANNELS AVAILABLE	2 MIN. NO. OF CHANNELS REQUIRED**	3 INDICATOR/ RECORDER**
1) Containment Pressure	6	2	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 line	1/steam line	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 Water Level (Narrow Range, Analog)+	2	1	INDICATOR/ RECORDER
12) Recirculation Sump Water Level (Narrow Range, Analog)+	2	1	INDICATOR/ RECORDER
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 3)

		1	2	3
15)	Level Sensors in Lower Level of Turbine Building	2	1	ALARM
16)	Reactor Coolant System Subcooling Margin Monitor	1	1	RECORDER
17)	PORV Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	INDICATOR
18)	PORV Position Indicator (Limit Switch)	1/Valve	1/Valve****	INDICATOR AND ALARM
19)	PORV Block Valve Position Indicator (Limit Switch)	1/Valve***	1/Valve	INDICATOR
20)	Safety Valve Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	INDICATOR
21)	Auxiliary Feedwater Flow Rate	1/Pump	1/Pump	INDICATOR
22)	Containment Water Level (Wide Range)	2	1	INDICATOR/ RECORDER
23)	Containment Hydrogen Monitor	2	1	INDICATOR/ RECORDER
24)	High-Range Containment***** Radiation Monitors (R25 R26)	2	1	ALARM
25)	Core Exit Thermocouples	4/quadrant	2/quadrant	INDICATOR

* 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 7 days and/or the recorder to be inoperable for up to 14 days.

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

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

***** If the high-range containment radiation monitor is determined to be inoperable when the reactor is above the cold shutdown condition, then restore the monitoring capability within 7 days, and

TABLE 3.5-5 (Sheet 3 of 3)

a) Initiate an alternate monitoring capability as soon as practical, but no later than 72 hours after identification of the failure of the monitor. If the monitor is not restored to operable status within 7 days, then,

b) Submit a Special Report to the NRC pursuant to Technical Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperabiblity and the plans and schedule for restoring the system.

\+ If both narrow range analog monitor channels are determined to be inoperable, at least one channel will be restored to operable status within 30 days or the plant will be brought to hot shutdown within the next 12 hours.

With the exception of the High Range Containment Radiation Monitors, 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 hours period.

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 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 simiulated 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 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.	(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 Water Level Monitoring System				
	(a) Containment Sump (Narrow Range, Analog)	N.A.	R	N.A.	
	(b) Recirculation Sump (Narrow Range, Analog)	N.A.	R	N.A.	
	(c) Containment Water Level (Wide Range)	N.A.	R	N.A.	
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 crosschecking the instrumentation.

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 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 5)

	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.	
34.	Plant Effluent Radiodine/ Particulate Sampling (Sample line common with monitor R 13)	N.A.	N.A.	R	
35.	* Loss of Power				
	a. 480 v Bus Undervoltage Relay	N.A.	R	M	
	b. 480 v Bus Degraded Voltage Relay	N.A.	R	M	
	c. 480 v Safeguards Bus Under Voltage Alarm	N.A.	R	M	
36.	Main Steam Line Radiation Monitors	D	R	Q	
37.	Containment Hydrogen Monitors	D	Q	M	
38.	Wide-Range Plant Vent Monitor (R27)	D	R	Q	

	Channel Description	Check	Calibrate	Test	Remarks
39.	High-Range Containment Radiation Monitoring (R25, R26)	D	R	Q	
40.	Core Exit Thermocouples	D	N.A.	N.A.	

* To be effective after completion of all required modiciations.

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

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	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 Valves	Operablility 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.

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 of ANSI N18.1-1971 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.

6.4.3 A training program for use of the post-accident sampling system shall be maintained to ensure that the plant has the capability to obtain and analyze reactor coolant and containment atmosphere samples under post-accident conditions.

6.4.4 A training program shall be maintained to ensure that the plant has the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluent during and following an accident.

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 adivse the Resident Manager on all matters related to nuclear safety and all matters which could adversely change the plants environmental impact.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Process Control Program implementation.

g. Offsite Dose Calculation Manual implementation.

h. Post-accident sampling and analysis and maintenance of required equipment.

i. Collection and analysis or measurement of post-accident radioactive iodine and particulates in plant gaseous effluents and maintenance of required equipment.

6.8.2 Temporary changes to procedures above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PORC and approved by the Resident Manager within 14 days of implementation.

6.8.3 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Resident Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator - Region 1, unless otherwise noted.

ANNUAL REPORTS

6.9.1.6 A summary of any challenges to the pressurizer power-operated relief valve and (or) safety valves shall be submitted to the Regional Administrator of Region 1 on an annual basis.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator - Region 1 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification;

a. Sealed source leakage on excess of limits (Specification 3.9)

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Primary coolant activity in excess of limits (Specification 3.1.D)

d. Seismic event analysis (Specification 4.10)

e. Inoperable fire protection and detection equipment (Specification 3.14)

f. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

g. Inoperable plant vent sampling, main steam line radiation monitoring or effluent monitoring capability (Table 3.5-4, items 5, 6 and 7)

h. Release of radioactive effluents in excess of limits (Appendix B Specifications 2.3, 2.4, 2.5, 2.6)

i. Inoperable containment high-range radiation monitors (Table 3.5-5, Item 24).

j. Radioactive environmental sampling results in excess of reporting levels (Appendix B Specifications 2.7, 2.8, 2.9)

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspec- tions, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all source material of record.

i. Records of reactor tests and experiments.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 65 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

I. INTRODUCTION

On November 1, 1983, a letter was sent by the Director, Division of Licensing to "All Pressurized Water Reactor Licensees." This Generic Letter (83-37) provided staff guidance on the content of the Technical Specifications associated with certain items in NUREG-0737. On February 17, 1984, New York Power Authority (the licensee) filed a response to the generic letter for Indian Point 3 Nuclear Generating Unit No. 3. On October 4, 1985, the licensee filed an amended response. The following report provides the staff evaluation of the NYPA submittals and makes recommendations for resolving the remaining issues. Supplemental information was provided on February 12, 1986.

II. DISCUSSION AND EVALUATION

The licensee was requested to provide Technical Specifications for several different systems. Each of these proposals is discussed and evaluated in an individual subsection below:

A. Reactor Coolant System Vents (II.B.1)

The generic letter stated:

"At least one reactor coolant system vent path (consisting of at least two valves in series which are powered from emergency buses) shall be operable and closed at all times (except for cold shutdown and refueling) at each of the following locations:

a. Reactor Vessel Head
b. Pressurizer Steam Space
c. Reactor Coolant System High Point"

A typical Technical Specification for reactor coolant system events was provided. For the plants using a power operated relief valve (PORV) as a reactor coolant system vent, the block valve was not required to be closed as long as the PORV was operable.

The licensee responded in the October 4 submittal, and further supported February 12, 1986, by proposing a Technical Specification that states:

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"Reactor Vessel Head Vents

Whenever the reactor coolant system is above 350°F, two reactor vessel head paths consisting of two valves in series with power available from emergency buses shall be OPERABLE.

a. If one of the above reactor vessel head vent paths is inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path. Restore the inoperable vent path to inoperable status within 90 days, or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours.

b. With both reactor vessel head vent paths inoperable restore one vent path to operable status within 7 days or be in hot shutdown within 6 hours and be below 350°F within the following 30 hours."

"With regard to the Reactor Coolant System Vents, IP-3 will use two vent paths; the Reactor Head Vent System (two vent paths, each consisting of two valves in series) and the Pressurizer PORVs. The NRC has reviewed IP-3's system and issued their acceptable Safety Evaluation on September 19, 1983."

As a result of the staff's audit of the material cited, we consider this issue to have been resolved acceptably.

B. Post-Accident Sampling (TMI II.B.3)

The generic letter stated:

"Licensees should ensure that their plant has the capability to obtain and analyze reactor coolant and containment atmosphere samples under accident conditions. An administrative program should be established, implemented and maintained to ensure this capability. The program should include:

a. training of personnel
b. procedures of sampling and analysis, and
c. provisions for maintenance of sampling and analysis equipment

It is acceptable to the staff, if the licensee elects to reference this program in the administrative controls section of the Technical Specifications and include a detailed description of the program in the plant operations manuals. A copy of the program should be easily available to the operating staff during accident and transient conditions."

The licensee responded in the February 17 submittal by stating:

"Technical Specification Sections 6.4 (Training) and 6.8 (Procedures) have been modified to reflect the administrative controls specified in the Generic Letter."

The staff's audit of the proposed changes to Sections 6.4 and 6.8 indicate that the licensee is correct and, therefore, we consider this issue to be resolved.

C. Long Term Auxiliary Feedwater System Evaluation (TMI II.E.1.1)

The generic letter stated:

"The objective of this item is to improve the reliability and performance of the auxiliary feedwater (AFW) system. Technical Specifications depend on the results of the licensee's evaluation and staff review of each plant. The limiting conditions of operation (LCO) and surveillance requirements for the AFW system should be similar to safety-related systems. Typical generic Technical Specifications are provided in Enclosure 3. These specifications are for a plant which has three auxiliary feedwater pumps. Plant-specific Technical Specifications could be established by using the generic Technical Specifications for the AFW system."

The licensee responded in the February 17 submittal by stating:

"As stated in the Authority's letter dated June 1, 1983, IPN 83-49, the Limiting Conditions for Operation (LCO) and the Surveillance Requirements for the Auxiliary Feedwater (AFWS) System are defined in Technical Specifications Sections 4.3.1 and 4.8, respectively. Both the LCO and the Surveillance Requirements for the AFWS System are similar to those for other safety-related systems and the requirements of the Generic Letter are thereby fulfilled. NRC acceptance of the current Indian Point 3 AFWS Technical Specifications has been previously documented in Mr. S. A. Varga's letter dated August 10, 1982 and no additional Technical Specification changes are required."

As a result of the staff's audit of the subject letter (accession number 8209100291), the staff agrees.

D. Noble Gas Effluent Monitors (TMI II.F.1.1)

The generic letter stated:

"Noble gas effluent monitors provide information, during and following an accident, which are considered helpful to the operator in accessing the plant condition. It is desired that these monitors be operable at all times during plant operation, but they are not required for safe shutdown of the plant. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability in a reasonable period of time. Considering the importance of the availability of the equipment and possible delays involved in administrative controls, seven days is considered to be the appropriate time period to restore the operability of the monitor. An alternate method for monitoring the effluent should be initiated as soon as practical, but no later than 72 hours after the identification of the failure of the monitor. If the monitor is not restored to operable conditions within seven days after the failure, a special report should be submitted to the NRC within 14 days following the event, outlining the cause of the inoperability, actions taken and the planned schedule for restoring the system to operable status."

In the February 17 response, licensee stated:

"Technical Specification Table 3.5-4, Table 4.1-1 and Section 6.9.2 (Special Reports) have been modified to reflect the recommendations of the Generic Letter for the post-accident effluent monitor on the unit vent. The main steam safety valves/atmospheric dump valves monitors' will be installed during the cycle 4/5 outage. The Authority will prepare and submit proposed Technical Specifications for the main steam effluent monitors no later than two months prior to start of the cycle 4/5 outage."

In the revised response of October 4 the licensee submitted a requirement for a minimum of one plant effluent radioactive/particulate sampler and one main steam line radiation monitor per steam line during power operations. The proposed LCO states:

"If the plant vent sampling capability or the main steam line radiation monitors is/are: determined to be inoperable when the reactor is above the cold shutdown condition, then restore the sampling/monitoring capability within 72 hours or:

a. Initiate a pre-planned alternate sampling/monitoring capability as soon as practical, but no later than 72 hours after identification of the failures. If the capability is not restored to operable status within 7 days, then,

b. Submit a Special Report to the NRC pursuant to Technical Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system."

The staff's audit of Table 3.5-4, Table 4.1-1, and Section 6.9.2 confirms the licensee response and, therefore, we consider this issue to be complete.

E. Sampling and Analysis of Plant Effluents (TMI II.F.1.2)

The generic letter stated:

"Each operating nuclear power reactor should have the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. An administrative program should be established, implemented and maintained to ensure this capability. The program should include:

a. training of personnel
b. procedures for sampling and analysis, and
c. provisions for maintenance of sampling and analysis equipment

It is acceptable to the staff, if the licensee elects to reference this program in the administrative controls section of the Technical Specifications and include a detailed description of the program in readily available to the operating staff during accident and transient conditions."

In the February 17 response, the licensee stated:

"Technical Specification Sections 6.4 (Training) and 6.8 (Procedures) have been modified to reflect the Administrative Controls specified in the Generic Letter. Operability requirements for this item were proposed in the Authority's letter dated February 1, 1982, IPN-82-10, and were subsequently incorporated in the Indian Point 3 Technical Specifications via Amendment No. 44 to the Facility Operating License."

As a result of the staff's audit of the material cited, we consider this issue to have been resolved acceptably.

F. Containment High-Range Radiation Monitor (TMI II.F.1.3)

The generic letter required that:

"A minimum of two in-containment radiation level monitors with a maximum range of 10^8 rad/hr (10^7 R/hr for photon only) should be operable at all times except for cold shutdown and refueling outages. In case of failure of the monitor, appropriate actions should be taken to restore its operational capability as soon as possible. If the monitor is not restored to operable condition within seven days after the failure, a special report should be submitted to the NRC within 14 days following the event, outlining the cause of inoperability, actions taken and the planned schedule for restoring the equipment to operable status.

Typical surveillance requirements are shown in Enclosure 3. The setpoint for the high radiation level alarm should be determined such that spurious alarms will be precluded. Note that the acceptable calibration techniques for these monitors are discussed in NUREG-0737."

In the February 17 response, the licensee stated:

"Technical Specification Table 3.5-5, Table 4.1-1, and Section 6.9.2 have been modified to reflect recommendations of the Generic Letter for the High-Range Containment Radiation Monitors."

Neither alarm setpoint nor a methodology for avoiding spurious alarms is specified. However, since no automatic action is initiated by this equipment, the staff finds that the proposed changes are acceptable.

G. Containment Pressure Monitor (TMI II.F.1.4)

The generic letter stated that:

"Containment pressure should be continuously indicated in the control room of each operating reactor during Power Operation, Startup and Hot Standby modes of operation. Two channels should be operable at all times when the reactor is operating in any of the above mentioned modes. Technical Specifications for these monitors should be included with other accident monitoring instrumentationin the present Technical Specifications. Limiting conditions for operating (including the required Actions) for the containment pressure monitor should be similar to other accident monitoring instrumentation included in the present Technical Specifications."

In the February 17 response, the licensee stated:

"Technical Specification Section 3.5 and Table 3.5-5 have been modified to require that at least two containment pressure channels be operable when Tavg is greater than 350°F."

In addition to verification of this statement, the staff audit of Table 3.5-5 indicated that the requirements for this equipment is similar to other accident monitoring equipment. Therefore, the staff finds that the proposed changes are acceptable.

H. Containment Water Level Monitor (TMI II.F.1.5)

The generic letter stated that:

A continuous indication of containment water level should be provided in the control room of each reactor during Power Operation, Startup and Hot Standby modes of operation. At least one channel for narrow range and two channels for wide range instruments should be operable at all times when the reactor is operating in any of the above modes.

Narrow range instruments should cover the range from the bottom to the top of the containment sump. Wide range instruments should cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon (or less if justified) capacity.

Technical Specifications for containment water level monitors should be included with other accident monitoring instrumentation in the present Technical Specifications. LCOs (including the required Actions) for wide range monitors should be similar to other accident monitoring instrumentation included in the present Technical Specifications. LCOs for narrow range monitor should include the requirement that the inoperable channel will be restored to operable status within 30 days or the plant should be brought to at least a hot standby condition within the next six hours. If both monitors are inoperable, at least one monitor should be restored to operable status within 72 hours or the plant should be brought to at least hot standby condition within the next six hours."

The licensee, in the October 4 response, provided a Technical Specification change to Table 3.5-5 that requires a minimum of one channel of narrow range and one channel of wide range containment level at all times. The limiting conditions for operation (LCO) permit both channels to be inoperable for seven days. Furthermore, both narrow range channels can be out of service for 30 days or the plant should be brought to hot shutdown within the next six hours. If these conditions are not satisfied within an additional 48 hours, the reactor shall be brought to cold shutdown.

The staff's analysis of this Specification shows that the requirements are similar to that recommended in the Generic Letter, except that the LCO proposed provides for action times that are twice as large. The staff has found that this difference is not significant and we conclude that the proposed change is, therefore, acceptable. The licensee's February 12, 1986 letter further supports this conclusion.

I. Containment Hydrogen Monitor (II.F.1.6)

The Generic Letter stated:

"Two independent containment hydrogen monitors should be operable at all times when the reactor is operating in Power Operation or Startup modes. LCO for these monitors should include the requirement that with one hydrogen monitor inoperable, the monitor should be restored to operable status within 30 days or the plant should be brought to at least a hot standby condition within the next six hours. If both monitors are inoperable, at least one monitor should be restored to operable status within 72 hours or the plant should be brought to at least hot standby condition within the next six hours." The licensee responded in the October 4 submittal by referencing proposed Technical Specification Table 3.5-5. As noted in Section II.H, the proposed Technical Specification is more liberal than that recommended by the staff. As noted in Section II.G above, the staff does not consider the additional time significant when compared to the likelihood of a DBA (7 days out of 10,000 years).

J. Instrumentation for Determination of Inadequate Core Cooling (TMI II.F.2) and Control Room Habitability Requirements (TMI III.D.3.4)

The licensee notified the staff that design and implementation of these Action Plan items has not been completed. Accordingly, these items will be the subject of future licensing actions.

III. SUMMARY

The following subsections describe those issues that the staff considers to have been satisfactorily addressed by the licensee:

A. Reactor Coolant System Vents (II.B.1)
B. Post-Accident Sampling (II.B.3)
C. Long-Term Auxiliary Feedwater System Evaluation (II.E.1.1)
D. Noble Gas Effluent Monitor (II.F.1.1)
E. Sampling and Analysis of Plant Effluents (II.F.1.2)
F. Containment High-Range Radiation Monitors (II.F.1.3)
G. Containment Pressure Monitor (II.F.1.4)
H. Containment Water Level Monitor (II.F.1.5)
I. Containment Hydrogen Monitor (II.F.1.6)

IV. 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.

V. 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 18, 1986

PRINCIPAL CONTRIBUTOR:

R. Scholl