UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 16, 1998

Mr. James Knubel
Chief Nuclear Officer
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. MA1641)

Dear Mr. Knubel:

The Commission has issued the enclosed Amendment No. 185 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3 (IP3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated April 16, 1998, as supplemented August 20, 1998. The amendment will extend the surveillance interval for five instrument channels from the current 18 months (at least once in 18 months) to 24 months (at least once in nominally 24 months, not to exceed 30 months). The proposed amendment also revises Section 6 of the TSs to reflect updated analyses.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

George F. Wunder

George F. Wunder, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 185 to DPR-64
2. Safety Evaluation

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ORIGINAL SIGNED BY:

George F. Wunder, Project Manager
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James Knubel
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. 20555-0001

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. 185
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 April 16, 1998, as supplemented August 20, 1998, 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. 185 , 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

S. Singh Bajwa, 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: December 16, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 185

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
Table 4.1-1 (Sheet 1 of 6)	Table 4.1-1 (Sheet 1 of 6)
Table 4.1-1 (Sheet 3 of 6)	Table 4.1-1 (Sheet 3 of 6)
Table 4.1-1 (Sheet 4 of 6)	Table 4.1-1 (Sheet 4 of 6)
Table 4.1-1 (Sheet 5 of 6)	Table 4.1-1 (Sheet 5 of 6)
Table 4.1-1 (Sheet 6 of 6)	Table 4.1-1 (Sheet 6 of 6)
Table 4.1-3 (Sheet 1 of 2)	Table 4.1-3 (Sheet 1 of 2)
4.4-6	4.4-6
4.4-7	4.4-7
4.4-8	4.4-8
4.4-10	4.4-10
4.5-2	4.5-2
4.8-1	4.8-1
4.9-4	4.9-4
4.11-3	4.11-3
6-22	6-22

TABLE 4.1-1 (Sheet 1 of 6)

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 Δ T
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 ## (2)	24M	Q (1)	1) Overtemperature ΔT, overpower ΔT, and low T_{avg} 2) Normal Instrument check interval is once/shift T_{avg} instrument check interval reduced to every 30 minutes when: - T_{avg}-T_{ref} deviation and low T_{avg} alarms are not reset and, - Control banks are above 0 steps
5. Reactor Coolant Flow	S ##	24M	Q	
6. Pressurizer Water Level	S	24M	Q	
7. Pressurizer Pressure	S ##	24M	Q	High and Low

TABLE 4.1-1 (Sheet 3 of 6)

Channel Description	Check	Calibrate	Test	Remarks
e. Main Steam Lines Process Radiation Monitors (R-62A, R-62B, R-62C, and R-62D)	D	24M	Q	
f. Gross Failed Fuel Detectors (R-63A and R-63B)	D	24M	Q	
16. Containment Water Level Monitoring System:				
a. Containment Sump	N.A.	24M	N.A.	Narrow Range, Analog
b. Recirculation Sump	N.A.	24M	N.A.	Narrow Range, Analog
c. Containment Water Level	N.A.	24M	N.A.	Wide Range
17. Accumulator Level and Pressure	S	24M	N.A.	
18. Steam Line Pressure	S	24M	Q	
19. Turbine First Stage Pressure	S	24M	Q	
20a. Reactor Trip Relay Logic	N.A.	N.A.	TM	
20b. ESF Actuation Relay Logic	N.A.	N.A.	TM	
21. Turbine Trip Low Auto Stop Oil Pressure	N.A.	24M	N.A.	
22. DELETED	DELETED	DELETED	DELETED	
23. Temperature Sensor in Auxiliary Boiler Feedwater Pump Building	N.A.	N.A.	18M	
24. Temperature Sensors in Primary Auxiliary Building				
a. Piping Penetration Area	N.A.	N.A.	24M	
b. Mini-Containment Area	N.A.	N.A.	24M	
c. Steam Generator Blowdown Heat Exchanger Room	N.A.	N.A.	24M	

TABLE 4.1-1 (Sheet 4 of 6)

Channel Description	Check	Calibrate	Test	Remarks
25. Level Sensors in Turbine Building	N.A.	N.A.	24M	
26. Volume Control Tank Level	N.A.	24M	N.A.	
27. Boric Acid Makeup Flow Channel	N.A.	24M	N.A.	
28. Auxiliary Feedwater:				
a. Steam Generator Level	S	24M	Q	Low-Low
b. Undervoltage	N.A.	24M	24M	
c. Main Feedwater Pump Trip	N.A.	N.A.	24M	
29. Reactor Coolant System Subcooling Margin Monitor	D	24M	N.A.	
30. PORV Position Indicator	N.A.	N.A.	24M	Limit Switch
31. PORV Position Indicator	D	24M	24M	Acoustic Monitor
32. Safety Valve Position Indicator	D	24M	24M	Acoustic Monitor
33. Auxiliary Feedwater Flow Rate	N.A.	18M	N.A.	
34. Plant Effluent Radioiodine/ Particulate Sampling	N.A.	N.A.	18M	Sample line common with monitor R-13
35. Loss of Power				
a. 480v Bus Undervoltage Relay	N.A.	24M	M	
b. 480v Bus Degraded Voltage Relay	N.A.	18M	M	
c. 480v Safeguards Bus Undervoltage Alarm	N.A.	24M	M	
36. Containment Hydrogen Monitors	D	Q	M	

TABLE 4.1-1 (Sheet 5 of 6)

Channel Description	Check	Calibrate	Test	Remarks
37. Core Exit Thermocouples	D	24M	N.A.	
38. Overpressure Protection System (OPS)	D	18M (1)	24M	1) Calibration frequency for OPS sensors (RCS pressure and temperature) is 24 months
39. Reactor Trip Breakers	N.A.	N.A.	TM(1)	1) Independent operation of under-voltage and shunt trip attachments
			24M(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
40. Reactor Trip Bypass Breakers	N.A.	N.A.	(1)	1) Manual shunt trip prior to each use
			24M(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
			24M(3)	3) Automatic undervoltage trip
41. Reactor Vessel Level Indication System (RVLIS)	D	24M	N.A.	
42. Ambient Temperature Sensors Within the Containment Building	D	24M	N.A.	
43. River Water Temperature # (installed)	S	18M	N.A.	
44. River Water Temperature # (portable)	S (1)	Q (2)	N.A.	1) Check against installed instrumentation or another portable device
				2) Calibrate within 30 days prior to use and quarterly thereafter
45. Steam Line Flow	S	24M	Q	Engineered Safety Features circuits only

Table Notation

* 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.

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

\#\# The "each shift" frequency also requires verification that the DNB parameters (Reactor Coolant Temperature, Reactor Coolant Flow, and Pressurizer Pressure) are within the limits of Technical Specification 3.1.H.

S - Each Shift (i.e.,at least once per 12 hours)
W - Weekly
P - Prior to each startup if not done previous week
M - Monthly
NA - Not Applicable
Q - Quarterly
D - Daily
18M - At least once per 18 months
TM - At least every two months on a staggered test basis (i.e., one train per month)
24M - At least once per 24 months
6M - At least once per 6 months

TABLE 4.1-3 (Sheet 1 of 2)

FREQUENCIES FOR EQUIPMENT TESTS		
	Check	Frequency
1. Control Rods	Rod drop times of all control rods	24M
2. Control Rods	Movement of at least 10 steps in any one direction of all control rods	Every 31 days during reactor critical operations
3. Pressurizer Safety Valves	Set Point	24M
4. Main Steam Safety Valves	Set Point	24M
5. Containment Isolation System	Automatic actuation	24M
6. Refueling System Interlocks	Functioning	Each refueling, 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 And Control Valves	Closure	Not to exceed 6 months**
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)	Quarterly
12. City Water Connections to Charging Pumps and Boric Acid Piping	Temporary connections available and valves operable	24M

** The turbine steam stop and control valves shall be tested at a frequency determined by the methodology presented in WCAP-11525, "Probablistic Evaluation of Reduction in Turbine Valve Test Frequency," as updated by Westinghouse Report, WOG-TVTF-93-17, "Update of BB-95/96 Turbine Valve Failure Rates and Effect on Destructive Overspeed Probabilities." The maximum test interval for these valves shall not exceed six months. Surveillance interval extension as per Technical Specification 1.12 is not applicable to the maximum test interval.

I. Residual Heat Removal System

1. Test

a. (1) The portion of the Residual Heat Removal System that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 350 psig at the interval specified below.

(2) The piping between the residual heat removal pumps suctions and the containment isolation valves in the residual heat removal pump suction line from the containment sump shall be hydrostatically tested at no less than 100 psig at the interval specified below.

b. Visual inspection shall be made for excessive leakage during these tests from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

2. Acceptance Criterion

The maximum allowable leakage from the Residual Heat Removal System components located outside of the containment shall not exceed two gallons per hour.

3. Corrective Action

Repairs or isolation shall be made as required to maintain leakage within the acceptance criterion.

4. Test Frequency

Tests of the Residual Heat Removal System shall be conducted at least once per 24 months.

Basis

The containment is designed for a pressure of 47 psig. [1] While the reactor is operating, the internal environment of the containment will be air at essentially atmospheric pressure and an average maximum temperature of approximately 130°F. The Design Basis Accidents (DBA) that represent a challenge to the containment structure are a Loss of Coolant Accident (LOCA) and a Main Steam Line Break (MSLB). The limiting calculated peak containment pressure of 42.40 psig is a result of the MSLB [7], which is less than the stated design pressure of 47 psig. In addition, DBA analyses demonstrate that the calculated peak containment temperature will remain less than the Equipment Qualification (EQ) envelope temperature of 290 degrees F.

The containment structure is designed to contain, within established leakage limits, radioactive material that may be released from the reactor core following a DBA. The containment was designed with an allowable leakage rate of 0.1 weight percent of containment air per day. This leakage rate, used to evaluate offsite doses resulting from DBAs is defined in 10CFR 50 Appendix J as L_a; the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing performed in accordance with the program required by Technical Specification 6.14. The minimum test pressure of 42.42 psig used for this program is based on analyses performed to support an increase of the ultimate heat sink temperature, [4] as incorporated by Technical Specification Amendment 98. The minimum test pressure, 42.42 psig, bounds the current limiting DBA pressure, 42.40 psig.

Prior to initial operation, the containment was strength-tested at 54 psig and was leak-tested. The acceptance criterion for this pre-operational leakage rate test was established as 0.075 W/o (.75 L_a) per 24 hours at 40.6 psig and 263°F, which were the peak accident pressure and temperature conditions at that time. This leakage rate is consistent with the construction of the containment, [2] which is equipped with a Weld Channel and Penetration Pressurization System for continuously pressurizing the containment penetrations and the channels over certain containment liner welds. These channels were independently leak-tested during construction.

The safety analysis has been performed on the basis of a leakage rate of 0.10 W/o per day for 24 hours. With this leakage rate and with minimum containment engineered safeguards operating, the public exposure would be well below 10CFR100 values in the event of the design basis accident. [3]

Maintaining the containment operable requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in surveillance requirement 4.4.D does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test is required to be <0.6 L_a for combined Type B and C leakage, and < 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. Surveillance requirement frequencies are as required by the Containment Leakage Rate Testing Program. Thus, Specification 1.12 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

The Weld Channel and Containment Penetration Pressurization System (WCCPPS)[5] is in service continuously to monitor leakage from potential leak paths such as the containment personnel lock seals and weld channels, containment penetrations, containment liner weld channels, double-gasketed seals and spaces between certain containment isolation valves and personnel door locks. A leak would be expected to build up slowly and would, therefore, be noted before design limits are exceeded. Remedial action can be taken before the limit is reached. The sensitive leakage rate test of the WCCPPS demonstrates that pressurized containment penetrations and liner inner weld seams are within a leakage acceptance criteria that will allow the air receivers and the standby source of gas pressure, nitrogen cylinders, to provide a 24 hour supply of gas to the system. The WCCPPS is not credited for limiting containment isolation valve leakage and the sensitivity test is not used for demonstrating compliance with containment isolation valve leakage criteria. The frequency of the sensitive leakage test reflects an extension of 25 percent from the 24 month refueling cycle and, therefore, Specification 1.12 (which allows Frequency extensions) does not apply[10].

Maintaining containment air locks operable requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. The surveillance requirement reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program. Thus, Specification 1.12 (which allows Frequency extensions) does not apply. During normal plant operation, containment personnel lock door seals are continuously pressurized after each closure by the WCCPPS. Whenever containment integrity is required, verification is made that seals repressurize properly upon closure of an air lock door. The verification meets the intent of the 10 CFR 50 Appendix J requirements.[8]

The maximum permissible inleakage rate from the containment isolation valves sealed with service water for the full 12-month period of post accident recirculation without flooding the internal recirculation pumps is 0.36 gpm per fan cooler.

REFERENCES

(1) FSAR - Section 5
(2) FSAR - Section 5.1.7
(3) FSAR - 14.3.5
(4) WCAP - 12269 Rev. 1, "Containment Margin Improvement Analysis for IP-3 Unit 3"
(5) FSAR - Section 6.6
(6) FSAR - Section 6.5
(7) Nuclear Safety Evaluation 98-3-013-MULT, "Integrated Safety Evaluation of 24-Month Cycle Instrument Channel Uncertainties," Revision 0, dated March 3, 1998.
(8) SECL-96-103, Indian Point Unit 3 Safety Evaluation of 24-Month Fuel Cycle Phase I Instrument Channel Uncertainties, June 1996
(9) Indian Point 3 Safety Evaluation Report, Supplement 2, December 1975.
(10) NRC Safety Evaluation Related to Amendment 129 to Operating License DPR-64.

2. <u>Containment Spray System</u>

a. System tests shall be performed at least once per 24 months. The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

b. The spray nozzles shall be checked for proper functioning at least every five years.

c. The tests will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

3. <u>Containment Hydrogen Monitoring Systems</u>

a. Containment hydrogen monitoring system tests shall be performed at intervals no greater than six months. The tests shall include drawing a sample from the fan cooler units.

b. The above tests will be considered satisfactory if visual observations and control panel indication indicate that all components have operated satisfactorily.

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 on a staggered test basis (i.e., one pump per month, so that each pump is tested once during a 3 month period) with full flow established to the steam generators at least once per 24 months.

   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 at least once per 24 months.

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 per 24 months,

   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 an 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.

4. Interval of Inspection

a. The first inservice inspection of steam generators should be performed after six effective full power months but not later than completion of the first refueling outage.

b. Subsequent inservice inspections should be not less than 12 or more than 24 calendar months after the previous inspection.

c. If the results of two consecutive inspections, not including the preservice inspection, all fall into the C-1 category, the frequency of inspection may be extended to 40-month intervals. Also, if it can be demonstrated through two consecutive inspections that previously observed degradation has not continued and no additional degradation has occurred, a 40-month inspection interval may be initiated.

B. Corrective Measures

All leaking tubes and defective tubes should be: (1) plugged, or (2) repaired.

C. Reports

1. Following each inservice inspection of steam generator tubes, the number of tubes plugged and repaired in each steam generator shall be reported to the Commission within 15 days.

2. The complete results of the steam generator tube inservice inspection shall be reported in writing on an annual basis for the period in which the inspection was completed per Specification 6.9.2. This report shall include:

   a. Number and extent of tubes inspected.

   b. Location and percent of wall-thickness penetration for each indication of an imperfection.

   c. Identification of the tubes plugged and the tubes repaired.

2. Visual inspection shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundations or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for the particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.11.B.5. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable via functional testing for the purpose of establishing the next visual inspection period. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

B. Functional Testing

1. At least once per 24 months during plant shutdown, a representative sample of 10% of all the safety-related hydraulic snubbers shall be functionally tested for operability, either in place or on a bench test. For each snubber that does not meet the requirement of 4.11.B.5, an additional 10% of the total installed of that type of hydraulic snubber shall be functionally tested. This additional testing will continue until no failures are found or until all snubbers of the same type have been functionally tested. The representative sample shall include each size and type of snubber in use in the plant.

2. The representative sample selected for functional testing should include the various configurations, operating environments, sizes and capacities of snubbers. At least 25% or the maximum possible if less than 25%, of the snubbers in the representative sample should include snubbers from the following three categories:

   a. The first snubber away from each reactor vessel nozzle.

6.12.2 The requirements of 6.12.1 above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the plant Radiological and Environmental Services Manager or his designee.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 Environmental qualification of electric equipment important to safety shall be in accordance with the provisions of 10 CFR 50.49. Pursuant to 10 CFR 50.49, Section 50.49 (d), the EQ Master List identifies electrical equipment requiring environmental qualification.

6.13.2 Complete and auditable records which describe the environmental qualification method used, for all electrical equipment identified in the EQ Master List, in sufficient detail to document the degree of compliance with the appropriate requirements of 10 CFR 50.49 shall be available and maintained at a central location. Such records shall be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995" as modified by the following exception:

a. ANS 56.8 - 1994, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

The peak calculated containment internal pressure, P_a, for the limiting design basis accident is 42.40 psig. The minimum test pressure is 42.42 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

Leakage acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and C tests and $\leq 0.75 L_a$ for Type A tests;

b. Air lock testing acceptance criteria are :

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

   2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

c. Isolation valves sealed with the service water system leakage rate into containment acceptance criterion is ≤ 0.36 gpm per fan cooler unit

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 185 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

1.0 INTRODUCTION

By letters dated April 16, 1998, as supplemented August 20, 1998, The Power Authority of the State of New York also known as New York Power Authority, the licensee for Indian Point Nuclear Power Plant, Unit 3 (IP3), proposed license amendments to change the Technical Specifications (TSs) for IP3. The proposed TS modifications will extend the surveillance interval for five instrument channels from the current 18 months (at least once in 18 months) to 24 months (at least once in nominally 24 months, not to exceed 30 months). The five instrument channels are;

(a) Pressurizer Water Level (Channel Number 6),
(b) Accumulator Level and Pressure (Channel Number 17),
(c) Reactor Coolant System Subcooling Margin Monitor (Channel Number 29),
(d) Core Exit Thermocouples (Channel Number 37), and
(e) Reactor Vessel Level Indication System (Channel Number 41).

This application for amendment also proposes administrative changes to delete various notes from the TS that granted one-time extensions of surveillances for nine instrument channels. These one-time extensions supported the schedule for refueling outage 9, which has since been completed.

This application for amendment further proposes to identify the required surveillance for the core exit thermocouples as a calibration instead of a test and makes changes to section 6 of the TS to reflect updated analyses.

2.0 EVALUATION

2.1 Change the Surveillance Intervals for Five Instrument Channels

Beginning with cycle 9, which started in August 1992, IP3 began operating on 24-month cycles instead of the previous 18-month cycles. This operation has resulted in a mismatch between the 24-month period allowed for the fuel cycle and the 18-month interval allowed by the current TS for calibration of certain instrumentation channels that were supposed to be calibrated during each refueling outage. The licensee in order to avoid either a separate surveillance outage or an extended mid-cycle outage, proposed a TS change to extend the surveillance interval for these instrument channels to 24 months.

The licensee conducted an instrument drift analysis using the Westinghouse Drift Evaluation Methodology. This methodology consists of a statistical analysis of data obtained during past calibrations. The instrument drift is established using a graded approach whereby the probability and confidence of the evaluation (95%/95% or 95%/75%) is selected depending on the safety significance of the function. The drift data is examined for time dependence. Linear regression is used to arrive at instrumentation drift for a 24-month fuel cycle plus 25%, or a maximum allowable interval of 30 months.

The licensee's application contain a summary of the results of their analyses. The staff has reviewed the licensee's application for amendment and finds that it meets the requirements of Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle." The staff finds the proposed TS changes acceptable.

2.2 Deletion of Various Notes

The staff has reviewed the licensee's proposed administrative changes, which delete various notes that granted one-time extensions of certain tests and surveillances. These one-time extensions supported the schedule for refueling outage 9, which was completed in 1997. The staff concludes that the deletion of these notes is acceptable because the periods covered by these notes have expired and no longer apply.

2.3 Change of "Test" to "Calibrate" for the Core Exit Thermocouples

The existing 18-month surveillance requirement for the core exit thermocouples is characterized in TS Table 4.1-1 as a "test." In addition to changing the surveillance frequency from 18 months to 24 months, this application for amendment proposes to identify the required surveillance as a "calibration" instead of a "test," which is also consistent with the Standard Technical Specifications. Even though the thermocouple sensors are not subject to a calibration adjustment, the signal-processing electronics are exercised when the calibration procedures are performed. On the basis of the foregoing, the staff concludes that the proposed transfer of the surveillance requirement from the "Test" column to the "Calibration" column for Channel Number 37, "Core Exit Thermocouples," in TS Table 4.1-1 is acceptable.

Based on our review, we have determined that the licensee has complied with the applicable provisions of GL 91-04 and conclude that the proposed TS revision of the interval for the calibration of the subject five instrument channels from the current 18 months to 24 months is acceptable. We further conclude that the proposed deletion of various notes that granted one-time extensions of certain tests and surveillances, and the proposed transfer of the surveillance requirement from the "Test" column to the "Calibration" column for the core exit thermocouples are also acceptable.

2.4 Change to Section 6.14

The proposed change to Section 6.14 of the TS consists of replacing the statement

> "The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.39 psig."

with the statement

> "The peak calculated containment internal pressure, P_a, for the limiting design basis accident is 42.40 psig."

This change reflects updated safety analyses that show that the calculated containment pressure following a main steam line break accident will be 42.40 psig. The peak calculated containment internal pressure is bounded by the containment design internal pressure of 47 psig. and the minimum test pressure of 42.42 psig; therefore, this change is acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 56256). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: December 16, 1998