September 5, 1996

Mr. William J. Cahill, Jr.
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. M95867)

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 168 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 in response to your application transmitted by letter dated June 21, 1996, as supplemented August 19, 1996, and August 21, 1996. The amendment extends the surveillance interval for certain instruments from 18 to 24 months.

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,

ORIGINAL SIGNED BY:

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. 168 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

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DATE	08/23/96		08/23/96		08/ /96		08/ /96		09/04/96

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DATE: September 5, 1996

ISSUANCE OF AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-28

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

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Docket No. 50-286

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cc w/encls: See next page

William J. Cahill, Jr.
Power Authority of the State
of New York

Indian Point Nuclear Generating
Station Unit No. 3

cc:

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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. 168
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 June 21, 1996, as supplemented August 19, 1996, and August 21, 1996, 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. 168 , 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

Alexander W Dromerick for

Jocelyn A. Mitchell, Acting 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: September 5, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 168

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 6 of 6	Table 4.1-1 Sheet 6 of 6
4.4-7	4.4-7
4.4-10	4.4-10

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	18M	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	18M***	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	18M	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 6 of 6)

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.

*** This surveillance requirement may be extended on a one time basis to no later than April 26, 1997.

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

S - Each Shift
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

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 limiting peak containment temperature, based on LOCA containment response, is 261.5°F. [7] The peak containment pressure, also based on LOCA containment response, is approximately 42.39 psig. [7] [4] The acceptance criteria of specification 4.4.A.2. was changed by amendment 98 to reflect analysis [4] done for the ultimate heat sink temperature increase. The acceptance criteria of 42.42 psig is conservative with respect to the current peak pressure of approximately 42.39.

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_t) 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. [1]

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, the containment isolation valves are to be closed in the normal manner and without preliminary exercising or adjustments.

These specifications have been developed using Appendix J (issue effective date March 16, 1973) of 10CFR50 (with the surveillance frequency exception noted previously) and ANSI N45.4-1972 "Leakage Rate Testing of Containment structures for Nuclear Reactors" (March 16, 1972) for guidance.

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) SECL-92-131, Indian Point Unit 3 High Head Safety Injection Flow Changes Safety Evaluation, June 1992
(8) SECL-96-103, Indian Point Unit 3 Safety Evaluation of 24-Month Fuel Cycle Phase I Instrument Channel Uncertainties, June 1996

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 168 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 June 21, 1996, August 19, 1996, and August 21, 1996, the Power Authority of the State of New York, the licensee for Indian Point 3 Nuclear Power Plant (IP3), requested NRC's approval to implement amendments to its Operating License, DPR-64, by incorporating modifications to the Technical Specifications (TSs). The proposed TS modifications will extend, on a one-time basis, the nominal surveillance-interval for accumulator pressure and accumulator level from 18 months to 24 months; it will also permanently extend from 18 months to 24 months the surveillance interval on volume control tank (VCT) level. The August 19, 1996, and August 21, 1996, submittals fell within the scope of and did not change the staff's initial proposed finding of no significant hazards considerations.

2.0 EVALUATION

Starting with cycle nine which began in August 1992, IP3 began operating on 24 month cycles. This has resulted in a mismatch between TS required refueling-outage-frequency and frequency-of-calibration of instrumentation channels which were supposed to be calibrated during each refueling outage. Therefore, to avoid either a separate surveillance outage or an extended mid-cycle outage, the licensee has proposed a TS revision which extends instrument channel surveillance calibration intervals from 18 months to 24 months. In their submittal, the licensee stated that their evaluation of the impact of this extension has addressed all applicable factors including the instrument's past performance and its effect on safety system functions; the results of loop accuracy and setpoint calculations; and the effect on IP3 emergency operating procedures (EOPs), accident analysis, and the capability for safe shutdown of the plant.

The request for the proposed modification is based on guidance provided by the staff in a Generic Letter (GL) 91-04. GL 91-04 provides guidance on how licensees should evaluate the effects of an extension to a 24 month surveillance interval on the safety of the plant and on the safety significance of the effect of such an extension. The licensee has performed a detailed engineering analyses of the affected systems and instrument-loops to establish the basis for a 30 month (24 months + 25% additional surveillance frequency allowance) calibration frequency, to verify that the surveillance interval extensions have a small effect on plant safety, and to verify that the extended frequency of surveillance would not invalidate any assumptions in the plant licensing basis.

In GL 91-04, the NRC staff discussed seven issues pertaining to increasing the interval of instrument surveillance and identified specific actions that licensees should take to address each of these issues. The seven issues are related to collection of current/historical drift data and methodology to determine the projected 30 month drift with high confidence, revisiting instrument uncertainty/setpoint calculations to verify that all revised setpoints and drift values are acceptable for safe operation/safe shutdown/EOPs and do not invalidate licensing assumptions, and establishing an ongoing drift-monitoring-program to verify that the actual observed drifts are within their projected values. To address these issues, the licensee has evaluated instrument drift, determined instrument-loop uncertainties, updated setpoint calculations, and established an instrument-drift monitoring program at IP3.

In their submittal, the licensee stated that an assessment of instrument drift was performed using as-found and as-left calibration data from a minimum of the past four 18-month calibrations. Westinghouse setpoint methodology using statistical analyses was employed for the assessment and extrapolation of drift associated with the 24-month operating cycle. The licensee stated that this Westinghouse methodology for assessment of drift has been previously reviewed and approved by the staff.

The following steps for drift-assessment were described in the licensee's submittal dated June 21, 1996.

- As-left/as-found data from past calibrations was converted into percentage-span-drift-values and was reviewed for mechanistic errors including obvious data recording errors, identifiable measurement and test equipment (M&TE) problems, and transmitters that were declared failed. The licensee stated that in addition to the identification of data that was flawed by mechanistic causes, statistical outlier techniques were applied on a limited basis to remove suspect data sets in the case when a large number of points in a set were determined to be flawed.

- Distribution of the data was examined and the sample data was extrapolated to the population using descriptive statistics and tolerance factors resulting in drift allowances at specified probability/confidence levels. The drift was established using a graded approach, whereby the probability and confidence level of an evaluation was varied in accordance with the safety significance of the function. This approach resulted in drift evaluations being performed from a 95/95 to a 75/75 probability/confidence level.

- The drift data was examined for the presence of time dependence using a combination of statistical and visual checks.

To answer questions raised by the staff in a meeting on August 15, 1996, the licensee provided a submittal dated August 21, 1996, in which they provided additional descriptions relating to information in their initial submittal. The staff reviewed the above methodology and drift assessment approach and finds it consistent with GL 91-04 guidance and, therefore, to be acceptable.

Using a graded approach based on the combinations of probability and confidence, the results of the drift assessment was implemented in the following three categories.

1. For those functions that provide reactor protection system/engineered safety feature actuation system (RPS/ESFAS) automatic actuation or critical control used to establish initial conditions for accident analysis, the drift evaluation was based on a 95% probability at 95% confidence level (95/95).

   <u>For Pressurizer Pressure monitoring instruments</u>, the 30-month drift was established using a 95/95 confidence level bases because these instruments provide input to the RPS and ESFAS and also provide input for critical accident analysis assumptions. An evaluation of the historical data indicated that drift was not time dependent.

2. For those functions that are used for indication in order to take EOP actions or initiate important nuclear steam supply system (NSSS) control, the drift evaluation was based on a 75% probability at a 75% confidence level (75/75).

   <u>For Accumulator Pressure and Accumulator Level instruments</u>, a 30-month drift was established using a 75/75 confidence level bases. The staff expressed concern regarding the use of a 75/75 probability/confidence level for the uncertainty assumed for these two functions. Because of this concern, the licensee provided an additional submittal dated August 19, 1996, removed their request for a permanent surveillance-interval extension for the instruments for these two functions and substituted it with a request for a one-time change to a surveillance interval of 24 months not to exceed 30 months. The licensee committed that after the 24- month surveillance is complete, the supporting calculations will be revised, as necessary, to reflect the 18-month surveillance interval unless a request for a permanent 24 month-surveillance interval has been approved prior to the end of the next refueling outage. The licensee performed an analysis using drift calculated on a 95/75 confidence level bases. This analysis indicated that for the accumulator pressure and accumulator level functions, all of the acceptance criteria of the safety analyses will be met with only insignificant changes to the margins. The staff has reviewed the justification for a one time surveillance interval extension for these two functions and concludes that appropriate justification is provided based on their safety significance, and the request is, therefore, 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 (61 FR 20853). 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.

Principal Contributor: S.V. Athavale

Date: September 5, 1996