January 18, 1995

Mr. William J. Cahill, Jr.
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, NY 10601

SUBJECT: ADMINISTRATIVE ERROR IN ISSUANCE OF TECHNICAL SPECIFICATIONS AMENDMENT NO. 148, AMENDMENT NO. 156, AND BASES CHANGE DATED DECEMBER 21, 1994 - INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NOS. M85332, M90429, AND M89463)

Dear Mr. Cahill:

Due to administrative errors, several pages transmitted to you via the subject documents were incorrect. However, each error was minor and had no effect on the technical validity of the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs). Details of each error are as follows:

- Amendment No. 148, issued on May 20, 1994, consolidated Section 4.5 such that page 4.5-12 was deleted. However, in the instruction page for Amendment No. 148, page 4.5-12 was not included in the "Remove Pages" column. Enclosed is the correct instruction page that should have been issued with Amendment No. 148.

- Amendment No. 156, issued on December 22, 1994, revised, in part, pages 6-9, 6-10, and 6-11. However, the pages transmitted with Amendment No. 156 did not contain the revision bars in the right hand column. Enclosed are the correct pages 6-9, 6-10, and 6-11.

- By letter dated December 21, 1994, the NRC issued a change to the Bases of TS Section 3.4. However, the Bases pages transmitted by that letter did not reflect a previous change to the TS Section 3.4 Bases which was issued by Amendment No. 151 on October 3, 1994. Enclosed are the correct pages 3.4-4 and 3.4-5.

Please update the IP3 TSs accordingly. If you have any questions, please contact me at (301) 415-1421.

Sincerely,

Original signed by

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

Docket No. 50-286

Enclosures: As stated

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DATED: January 18, 1995

CORRECTION TO AMENDMENT NOS. 148, 151, AND 156 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT UNIT 3

Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
J. Zwolinski, 14/H/3
Michael J. Case
C. Vogan
N. Conicella
OGC
D. Hagan, T-4 A43
G. Hill (2), T-5 C3
C. Grimes, 11/E/22
ACRS (4)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

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

January 18, 1995

Mr. William J. Cahill, Jr.
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, NY 10601

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Please update the IP3 TSs accordingly. If you have any questions, please contact me at (301) 415-1421.

Sincerely,

N. F. Conicella

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

Docket No. 50-286

Enclosures: As stated

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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Mr. Richard L. Patch, Director
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White Plains, NY 10601

ATTACHMENT TO LICENSE AMENDMENT NO. 148

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
4.1-3	4.1-3
Table 4.1-1 (Sheet 2 of 6)	Table 4.1-1 (Sheet 2 of 6)
Table 4.1-3 (Sheet 2 of 2)	Table 4.1-3 (Sheet 2 of 2)
4.4-6	4.4-6
4.5-7	4.5-7
4.5-8	4.5-8
4.5-9	4.5-9
4.5-10	4.5-10
4.5-11	4.5-11
4.5-12	------

CHARTER

6.5.2.2 The conduct of the SRC will be in accordance with a charter approved by the Executive Vice President and Chief Nuclear Officer. The charter will define the SRC's authority and establish the mechanism for carrying out its responsibilities.

MEMBERSHIP

6.5.2.3 The SRC shall be composed of at least six individuals including a Chairman and a Vice Chairman. Members shall be appointed by the Vice President Regulatory Affairs and Special Projects and approved by the Executive Vice President and Chief Nuclear Officer. SRC members and alternates shall have an academic degree in engineering or a physical science, or the equivalent, and shall have a minimum of five years technical experience in one or more areas listed in 6.5.2.1.

ALTERNATES

6.5.2.4 Alternates for the Chairman, Vice Chairman and members may be appointed in writing by the Vice President Regulatory Affairs and Special Projects and approved by the Executive Vice President and Chief Nuclear Officer.

CONSULTANTS

6.5.2.5 Consultants may be used as determined by the SRC Chairman and as provided for in the charter.

MEETING FREQUENCY

6.5.2.6 The SRC shall meet at least once per six months.

QUORUM

6.5.2.7 A quorum shall consist of at least a majority of the appointed individuals (or their alternates) and the Chairman (or the designated alternate). No more than two alternates may participate as SRC voting members at any one time. No more than a minority of the quorum shall have direct line responsibility for the operation of the plant.

REVIEW

6.5.2.8 The SRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications of this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meetings minutes of the Plant Operating Review Committee.

AUDITS

6.5.2.9 Audits of facility activities shall be performed under the cognizance of the SRC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or methods of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B," 10 CFR 50, at least once per 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 12 months.

f. The Facility Security Plan including the Safeguards Contingency Plan and implementing procedures at least once per 12 months.

g. Any other area of facility operation considered appropriate by the SRC or the Executive Vice President and Chief Nuclear Officer.

h. The Facility Fire Protection Program and implementing procedures at least once per two years.

i. A fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

Where:

$Hi\phi$ - Safety Analysis power range high neutron flux setpoint, percent.

Q - Nominal NSSS power rating of the plant (including reactor coolant pump heat) in Mwt (3037 Mwt).

K - Conversion factor, 947.82 $\frac{(Btu/sec)}{Mwt}$

w_s - Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure, including tolerance and accumulation, as appropriate, in lb/sec. (w_s = 150 + 228.61 * (4 - V) lb/sec, where V = Number of inoperable safety valves in the steam line of the most limiting steam generator).

h_{fg} - Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm (608.5 Btu/lbm).

N - Number of loops in plant (4).

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tank is the amount needed for 24 hours at hot shutdown. When the condensate storage supply is exhausted, city water will be used.

The system piping and valves that are governed by Specification 3.4.A.(4) include the two (2) QA Category I, 100% cpacity breather valves installed on the dome of the Condensate Storage Tank (CST). The purpose of these valves is to ensure the CST pressure is within its design limits by providing both pressure relieving and vacuum break capability. Per Specification 3.4.B, if one (1) breather valve is inoperable, it must be returned to operability within 48 hours or the reactor must be shutdown and cooled to below 350°F using normal operating procedures.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove core decay heat after a reactor shutdown.

The limitations placed on turbine-generator electrical output due to conditions of turbine overspeed setpoint, number of operable steam dump lines, and condenser back pressure are established to assure that turbine overspeed (during conditions of loss of plant load) will be within the design overspeed value considered in the turbine missile analysis. [2] In the preparation of Figures 3.4-1 and 3.4-2, the specified number of operable L.P. steam dump lines is shown as one (1) greater than the minimum number required to act during a plant trip. The limitations on electrical output, as indicated in Figures 3.4-1 and 3.4-2, thus consider the required performance of the L.P. Steam Dump System in the event of a single failure for any given number of operable dump lines.