

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

March 13, 2000

*Template
NRR-058*

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING
UNIT NO. 3 RE: PLANT OPERATING REVIEW COMMITTEE (PORC) REVIEW
OF FIRE PROTECTION PROGRAM (TAC NO. MA5964)

Dear Mr. Knubel:

The Commission has issued the enclosed Amendment No. 201 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 of September 9, 1996, as supplemented on June 6, 1997, and June 7, 1999.

The amendment revises TS Section 6 to delete requirements for Plant Operating Review Committee review of the fire protection program and implementing procedures and forwards bases pages 4.6-3 and 4.6-4 in accordance with your request dated February 2, 2000, and Table of Contents page ii which was to be issued with amendment 200.

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,

/RA/

George F. Wunder, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 201 DPR-64
2. Safety Evaluation

cc w/encs: See next page

DISTRIBUTION: See next page

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*See previous concurrence

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DATED: March 13, 2000

AMENDMENT NO. 201 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT
UNIT 3

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

March 13, 2000

Mr. James Knubel
Chief Nuclear Officer
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New York
123 Main Street
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cc w/encls: See next page

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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. 201
License No. DPR-64

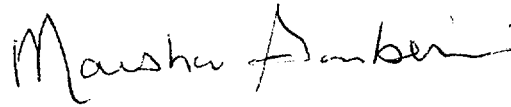
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by the Power Authority of the State of New York (the licensee) dated September 9, 1996, as supplemented on June 6, 1997, and June 7, 1999, complies with the standards and requirements of the Atomic 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. 201 , 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



Marsha Gamberoni, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 13, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 201

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of changes.

Remove Page

ii

6-8

4.6-3

4.6-4

Insert Page

ii

6-8

4.6-3

4.6-4

<u>Section</u>	<u>Title</u>	<u>Page</u>
1.14	E-Average Disintegration Energy	1-5
1.15	Dose Equivalent I-131	1-5
1.16	Reportable Event	1-6
1.17	Core Operating Limits Report	1-6
2	Safety Limits and Limiting Safety System Settings	2.1-1
2.1	Safety Limits, Reactor Core	2.1-1
2.2	Safety Limit, Reactor Coolant System Pressure	2.2-1
2.3	Limiting Safety System Settings, Protective Instrumentation	2.3-1
3	Limiting Conditions for Operation	3.1-1
3.1	Reactor Coolant System	3.1-1
3.1.A	Operational Components	3.1-1
3.1.B	Heatup and Cooldown	3.1-17
3.1.C	Minimum Conditions for Criticality	3.1-25
3.1.D	Primary Coolant Activity	3.1-26
3.1.E	Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration	3.1-29
3.1.F	Leakage of Reactor Coolant	3.1-31
3.1.G	Secondary Coolant Activity	3.1-35
3.1.H	RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits	3.1-36
3.2	Deleted	3.2-1
3.3	Engineered Safety Features	3.3-1
3.3.A	Safety Injection and Residual Heat Removal Systems	3.3-1
3.3.B	Containment Cooling and Iodine Removal Systems	3.3-5a
3.3.C	Isolation Valve Seal Water System (IVSWS)	3.3-7
3.3.D	Weld Channel and Penetration Pressurization System (WC & PPS)	3.3-8
3.3.E	Component Cooling System	3.3-9
3.3.F	Service Water System/Ultimate Heat Sink	3.3-10
3.3.G	Containment Hydrogen Monitoring Systems	3.3-11
3.3.H	Control Room Ventilation System	3.3-11
3.3.I	Electric Hydrogen Recombiner System	3.3-12
3.4	Steam and Power Conversion System	3.4-1

- j. Deleted
- k. Deleted
- l. Review of every unplanned onsite release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Site Executive Officer and to the Safety Review Committee.
- m. Deleted

AUTHORITY

- 6.5.1.7 The Plant Operating Review Committee shall:
- a. Recommend to the Site Executive Officer approval or disapproval of items considered under 6.5.1.6(a) through (e) above.
 - b. Render determinations with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question, as defined in 10 CFR 50.59.
 - c. Provide notification within 24 hours to the Chairman of the SRC and the Chief Nuclear Officer of disagreement between the PORC and the Site Executive Officer; however, the Site Executive Officer shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Plant Operating Review Committee shall maintain minutes of each meeting and copies shall be provided to the Chairman of the SRC and the Chief Nuclear Officer.

6.5.2 SAFETY REVIEW COMMITTEE (SRC)

FUNCTION

- 6.5.2.1 The SRC shall function to provide independent review and audit of designated activities in the areas of:
- a. Nuclear power plant operations
 - b. Nuclear engineering
 - c. Chemistry and radiochemistry
 - d. Metallurgy
 - e. Instrumentation and control

The testing frequency specified will be often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply is continuously monitored. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

Each of the three emergency diesel generators (EDG) consists of an Alco engine coupled to a Westinghouse generator. Any two EDGs can power the minimum safeguards loads. Surveillance testing in accordance with Specification 4.6.A.2 consists of operating each EDG in the range of 1900kW to 1950kW for at least 105 minutes. The EDGs have four capacity ratings as defined below that can be used to assess EDG operability.

- Continuous: Electrical power output capability that can be maintained 24 hours /day, with no time constraint.
- 2000-hour: Electrical power output capability that can be maintained in one continuous run of 2000 hours or in multiple shorter duration runs totaling 2000 hours.
- 2-hour: Electrical power output capability that can be maintained for up to 2 hours in any 24-hour period.
- 1/2 - hour: Electrical power output capability that can be maintained for up to 30 minutes in any 24-hour period.

The electrical output capabilities (EDG load) applicable to these four ratings are as follows:

<u>RATING</u>	<u>EDG LOAD</u>	<u>TIME CONSTRAINT</u>
Continuous	≤ 1750 kW	None
2000-hour	≤ 1950 kW	≤ 2000 hours / calendar year
2-hour	≤ 1950 kW ≤ 1750 kW	≤ 2 hours in a contiguous 24-hour period; AND for the remaining 22 hours. [See NOTE A]
1/2-hour	≤ 2000 kW ≤ 1750 kW	≤ 30 minutes in a contiguous 24-hour period; AND for the remaining 23.5 hours. [See NOTE A]

NOTE A: The loading cycle permitted for the '2-hour' and the '1/2-hour' rating is operation at the overload condition (e.g. > 1750 kW) for the specified time followed by operation at the 'continuous' (e.g. ≤ 1750kW) rating for the remaining time in the 24-hour period. This loading cycle may be repeated each day, as long as back-to-back operation in the overload condition does not occur. The 2000-hour cumulative time constraint also applies to repetitive operation at the overload conditions allowed by the 2-hour and the 1/2-hour ratings.

Operation in excess of 2000 kW, regardless of the duration, is an unanalyzed condition. In such cases, the EDG is assumed to be inoperable and the vendor should be consulted to determine if accelerated or supplemental inspection and/or maintenance is necessary. The EDG can be returned to an operable status following completion of vendor-required inspection and/or maintenance.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails. The periodic equalizing charge will ensure that the ampere-hour capability of the batteries is maintained.

The service and performance discharge test of each battery, together with the visual inspection of the plates, will assure the continued integrity of the batteries. The batteries are of the type that can be visually inspected, and this method of assuring the continued integrity of the battery is proven standard power plant practice.

The battery service test demonstrates the capability of the battery to meet the system design requirements. The Indian Point Unit 3 design duty cycle loads are determined by a LOCA concurrent with a loss of AC power.

The performance discharge test is a test of the constant current capacity of a battery, normally done in the as found condition after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The modified battery performance discharge test is a composite test which addresses both the service test and performance discharge test requirements. It shall consist of a one minute peak load equivalent to that of the service test and a constant discharge current for the remainder of the test which envelopes the next highest load value of the service test. The purpose of the modified performance discharge test is to compare the capacity of the battery against the manufacturer's specified capacity and thereby determine when the battery is approaching the end of its life, as well as to demonstrate capability to meet system design requirements. Every other 24 month operating cycle, the modified performance discharge test may be performed in lieu of the battery service test required by Technical Specification 4.6.B.3.

The station batteries are required for plant operation, and performing the station battery service and performance discharge (or modified performance discharge) test require the reactor to be shutdown.

Reference

FSAR, Section 8.2



UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 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 letter dated September 9, 1996, as supplemented by letters dated June 6, 1997, and June 7, 1999, the Power Authority of the State of New York (the licensee) submitted an amendment to the Technical Specifications (TSs) for the Indian Point Nuclear Generating Unit No. 3 (IP3). The proposed amendment would remove the requirement for the Plant Operating Review Committee (PORC) review of the fire protection program and implementing procedures.

On October 13, 1999, the NRC staff issued Regulatory Issue Summary 99-02 (RIS 99-02) to address the issue of PORC review of fire protection programs. In the RIS, the staff informed all licensees of its policy concerning the removal of this particular TS requirement.

2.0 EVALUATION

Section 6.5.1 of the TS describes the function, membership, and responsibilities for the IP3 PORC. The licensee has proposed to delete TS 6.5.1.6.m, which contains one of the responsibilities of the PORC, and currently reads as follows:

- m. Review the Fire Protection Program and implementing procedures.

Presently, as stated in TS 6.5.1.6.m, the PORC is required to review all changes to the fire protection program and implementing procedures. The licensee believes that this requirement represents an unnecessary administrative burden on the PORC, since the large majority of changes reviewed are not safety-significant. The licensee states that its proposal makes PORC review responsibilities for fire protection issues more consistent with the other responsibilities given in TS 6.5.1.6.

PORC responsibility for review of the fire protection program and procedures was added by Amendment No. 157, which included relocation of fire protection program requirements in accordance with the guidance of Generic Letter (GL) 88-12, "Removal of Fire Protection Requirements from the Technical Specifications." Subsequent to the relocation of the fire protection program requirements, Amendment No. 159 established a new review and approval process for procedures required by TS 6.8, which includes fire protection program procedures. This process requires that procedures be reviewed by qualified individuals, and approved by appropriate plant management. If safety and/or environmental evaluations are required, the PORC is required to review those evaluations. It is the licensee's position that this process provides for appropriate PORC review of changes when safety and/or environmental

evaluations are required. Otherwise, less significant changes which do not impact safety or environmental considerations can be made without imposing an unnecessary administrative burden on the PORC.

There are several mechanisms to ensure that the effectiveness of the fire protection program will be maintained. First, Amendment No. 157 revised the Facility Operating License to contain a license condition which states, in part:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Second, the fire protection program is also included in the IP3 Final Safety Analysis Report, Section 9.6.2, so the provisions of 10 CFR 50.59 apply. This regulation prohibits changes to the facility without prior NRC approval if it is determined that the change constitutes an unreviewed safety question. Finally, the licensee also notes that TS 6.5.2.9 requires the Safety Review Committee to inspect and audit the fire protection program.

Based on the above discussion, the staff has determined that PORC review of all fire protection program and implementing procedure changes is unnecessary, and that safety is assured by the procedure review and approval process, and other license requirements as discussed above. The staff concludes, therefore, that the proposed amendment 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

This amendment changes recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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: G. Wunder

Date: March 13, 2000