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

September 8, '99

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: REACTOR TRIP ON TURBINE TRIP (TAC NO. MA4696)

Dear Mr. Knubel:

The Commission has issued the enclosed Amendment No. 192 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 application transmitted by letter dated January 28, 1999, as supplemented by letter dated May 4, 1999. The amendment changes the reactor trip on turbine trip from at or above 10 percent rated power to at or above the P-8 setpoint.

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, Section 1
Project Directorate 1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation



Docket No. 50-286

Enclosures: 1. Amendment No. 192 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

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Docket No. 50-286

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cc w/encls: See next page

Indian Point Nuclear Generating Station
Unit No. 3

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Gerald C. Goldstein
Assistant General Counsel
Power Authority of the State
of New York
1633 Broadway
New York, NY 10019

Mr. Eugene W. Zeltmann, President
and Chief Operating Officer
Power Authority of the State
of New York
99 Washington Ave., Suite # 2005
Albany, NY 12210-2820

Mr. Robert J. Barrett
Site Executive Officer
Indian Point 3 Nuclear Power Plant
P.O. Box 215
Buchanan, NY 10511

Ms. Charlene D. Faison
Director Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Mr. F. William Valentino, President
New York State Energy, Research,
and Development Authority
Corporate Plaza West
286 Washington Ave. Extension
Albany, NY 12203-6399

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Resident Inspector
Indian Point 3 Nuclear Power Plant
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Mr. John McCann
Nuclear Safety and Licensing
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, NY 10511

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. Richard L. Patch, Director
Quality Assurance
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Mr. Paul Eddy
New York State Dept. of
Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

Mr. Harry P. Salmon, Jr.
Vice President - Engineering
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Mr. Eric Beckjord
2909 29th St. NW
Washington, DC 20008-3416

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. 192
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 January 28, 1999, as supplemented by letter dated May 4, 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. 192 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. Implementation shall consist of the relocation of the Technical Specification requirements to the final safety analysis report.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, 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: September 8, 1999

ATTACHMENT TO LICENSE AMENDMENT NO.192

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 marginal lines indicating the areas of change.

| Remove Pages | Insert Pages |
|---|---|
| 2.3-4 | 2.3-4 |
| 2.3-6 | 2.3-6 |
| 2.3-7 | 2.3-7 |
| Table 3.5-2 (Sheet 2 of 3) | Table 3.5-2 (Sheet 2 of 3) |
| Table 4.1-1 (Sheet 3 of 6) | Table 4.1-1 (Sheet 3 of 6) |

C. Other reactor trips

(1) High pressurizer water level - $\leq$ 92% of span.

(2) Low-low steam generator water level - $\geq$ 5% of narrow range instrument span.

(3) Anticipatory reactor trip upon turbine trip - $\geq$50% of rated power.

2. Protective instrumentation settings for reactor trip interlocks shall satisfy the following conditions:

A. The reactor trips on low pressurizer pressure, high pressurizer level and low reactor coolant flow for two or more loops shall be unblocked when:

(1) Power range nuclear flux $\geq$ 10% of rate power, or

(2) Turbine first stage pressure $\geq$ 10% of equivalent full load.

B. The single loop loss of flow reactor trip and the reactor trip on turbine trip shall be unblocked when the power range nuclear instrumentation indicates $\geq$50% of rated power (P-8).

<u>Basis</u>

The high flux reactor trip provides redundant protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis. [1]

The power range nuclear flux reactor trip high setpoint protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point in the accident analysis. [2] [3]

protection channel are set in accordance with the measured delta-T and $T_{avg}$ at rated power existing in the loop from which the process inputs for a particular protection channel are supplied. This is done to account for loop to loop differences in delta-T and $T_{avg}$ which may exist as a result of asymmetric steam generator tube plugging.

The low flow reactor trip protects the core against DNB in the event of a loss of one or two reactor coolant pumps. The undervoltage reactor trip protects the core against DNB in the event of a loss of two or more reactor coolant pumps. The setpoints specified are consistent with the values used in the accident analysis.[8] The low frequency reactor coolant pump trip also protects against a decrease in flow. The specified setpoint assures a reactor trip signal by opening the reactor coolant pump breaker before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1600 $ft^3$ of water (39.75 ft. above the lower instrument tap) corresponds to 92% of span. The specified setpoint allows margin for instrument error and transient level overshoot beyond their trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against postulated loss of feedwater accidents. This specified setpoint assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the Auxiliary Feedwater System.[9]

Specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed setpoints at which these trips are unblocked assures their availability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above the P-8 setpoint for four-loop operation, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below the applicable safety limit during normal operational transients.

A turbine trip causes a direct reactor trip, when operating at or above the P-8 setpoint (P-8 will be set at a monitored Reactor Thermal Power of less than or equal to 50%; an allowance for instrument uncertainty has been incorporated in the supporting analyses). This anticipatory trip will operate in advance of the pressurizer high pressure reactor trip to reduce the peak Reactor Coolant System pressure. No credit was taken in the accident analyses for operation of this trip.[10] A turbine trip without a direct reactor trip below 50% of rated power is bounded by the Loss of Load/Turbine Trip at full power[11], and the Loss of Flow at full power[8] analyses.

The steam-feedwater flow mismatch trip does not appear in the specification as this setting is not used in the transient and accident analysis (FSAR Section 14).

(1) FSAR 14.1.1

(2) FSAR 14.1.2

(3) FSAR Table 14.1-1

(4) FSAR 14.3.1

(5) FSAR 14.1.2

(6) FSAR 7.2

(7) FSAR 3.2.1

(8) FSAR 14.1.6

(9) FSAR 14.1.9

(10) Generic Letter 82-16, II.K.3.12 (NUREG-0737)

(11) FSAR 14.1.8

TABLE 3.5-2 (Sheet 2 of 3)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

| No. | FUNCTIONAL UNIT | 1<br>NO. OF CHANNELS | 2<br>NO. OF CHANNELS TO TRIP | 3<br>MIN. NUMBER OF OPERABLE CHANNELS | 4<br>MIN. DEGREE OF REDUNDANCY | 5<br>OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET* |
|---|---|---|---|---|---|---|
| 9. | Lo Lo Steam Generator Water Level | 3/loop | 2/loop | 2/loop | 1/loop | Maintain hot shutdown |
| 10. | Undervoltage 6.9 KV Bus | 1/bus | 2 | 3 | 2 | Maintain hot shutdown |
| 11. | Low Frequency 6.9 KV Bus** | 1/bus | 2 | 3 | 2 | Maintain hot shutdown |
| 12. | Turbine Trip: Low auto stop oil pressure(Power $\geq$P-8) | 3 | 2 | 2 | 1 | Maintain reactor power below 50% of full power |
| 13. | Reactor Trip Breakers*** | 2 | 1 | 2 | 1 | Maintain hot shutdown**** |
| 14. | Reactor Protection Relay Logic | 2 | 1 | 2 | 1 | Maintain hot shutdown**** |

* Maintain hot shutdown means maintain or proceed to hot shutdown within 4 hours using normal operating procedures, if the unacceptable condition arises during operation.

** 2/4 trips all four reactor coolant pumps.

*** A reactor trip breaker is considered inoperable if any of its components fail to meet test specifications. If either the undervoltage or shunt trip device (not both) prevent a breaker from proper operation, then 72 hours are allowed to restore the failed device to operable status before the affected breaker is declared inoperable.

**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 (Power $\geq$ P-8) | 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 | |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.192TO 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 January 28, 1999, as supplemented by letter dated May 4, 1999, the Power Authority of the State of New York (the licensee) proposed changes to the Technical Specifications (TSs) for the Indian Point Nuclear Generating Unit No. 3 (IP3). The changes would increase the power level required for the reactor trip following a turbine trip signal. The current TS requires a reactor trip when a turbine trip signal occurs and the reactor power is equal to or greater than 10 percent of the rated power (RP). It also requires that the automatic reactor trip on a turbine trip signal be blocked when the reactor power decreases below 10 percent RP. The proposed TS will change the requirement so that the anticipatory reactor trip on turbine trip shall be unblocked at 50 percent RP (P-8 setpoint). The changes allow the licensee to implement an interlock system at IP3 to block reactor trips on turbine trips for reactor power levels below 50 percent RP. Because most of the turbine trips occur at low power levels, the proposed TS would decrease unnecessary challenge to the reactor protection system and increase plant availability. The proposed TSs 2.3.1.C(3), 2.3.2.B, item 12 of Table 3.5-2, item 21 of Table 4.1-1 specify the increased power level required for reactor trips on a turbine trip signal.

The May 4, 1999, letter provided additional information that did not change the staff's proposed finding of no significant hazards considerations.

## 2.0 SAFETY EVALUATION

### 2.1 Loss-of-coolant accident (LOCA) and Transient Analyses

The licensee identified the limiting cases for each event category discussed in the safety analysis sections of the Final Safety Analysis Report (FSAR) and evaluated the effect of TS changes on the LOCA and the transient analysis for each limiting case. For the LOCA analysis, the licensee considered the following cases: (1) large and small LOCAs, (2) post-LOCA long-term core cooling subcriticality, and (3) post-LOCA long-term core cooling minimum flow and hot-leg switchover to prevent boron precipitation. Since the proposed TS changes do not affect the normal plant operating parameters, the safeguards systems actuation or accident mitigation capabilities important to a LOCA, or the assumptions used in the LOCA-related accidents analysis, they will not affect the results of the LOCA analyses in the current FSAR.

For the Non-LOCA Transient Analyses, the licensee considered the following events:

1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition
2. Uncontrolled Control Rod Assembly Withdrawal at Power
3. Rod Assembly Misalignment
4. Rod Cluster Control Assembly Drop
5. Chemical and Volume Control System Malfunction
6. Loss of Reactor Coolant Flow
7. Startup of an Inactive Reactor Coolant Loop
8. Loss of External Load/Turbine Trip
9. Loss of Normal Feedwater
10. Excessive Heat Removal due to Feedwater System Malfunctions
11. Excessive Load Increase Incident
12. Loss of all AC Power to the Station Auxiliaries
13. Startup Accidents without Reactor Coolant Pump Operation
14. Startup Accident with a Full Pressurizer

From its analysis, the licensee identified the only event that would be affected by the TS changes was the loss-of-load/turbine trip (LOL/TT) event because no other events required the reactor trip on a turbine trip signal for event mitigation. With the proposed TS, the LOL/TT event without a subsequent reactor trip on a turbine trip signal could occur at or below 50 percent power. Because the initial power is low and the reactor protection system activator (on such signals as high pressurizer pressure, high pressurizer water level, low-low steam generator water level or overtemperature delta signals) will be available to trip the reactor, the LOL/TT event without a reactor trip on a turbine trip is bounded by the current analysis of record for the LOL/TT event, and the loss of flow event from full power conditions with respect to the peak RCS pressure and the minimum departure from nucleate boiling ratio (DNBR), respectively. Therefore, the proposed TS changes do not affect the results of the transient analyses in the current FSAR.

2.2 TMI Action Item II.K.3.10 Analysis

The staff position for TMI action Item II.K.3.10 in NUREG-0737 states that:

> "The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power-operated relief valve (PORV) is substantially unaffected by the modification."

To address the staff concerns of TMI Action Item II.K.3.10 satisfactorily, the licensee analyzed the turbine trip without a reactor trip transient (TTWORT) initiated from 50 percent power, identified by the licensee as the most limiting case affected by the proposed TS. The results of the analysis quantified the effect of the proposed TS on the potential of a small-break LOCA resulting from stuck-open PORVs.

Both cases with a single failure and multiple failures were performed. For the multiple failure considerations, the licensee identified three control systems that would affect the results of the TTWORT transient. The control systems are the steam dump control system, the pressurizer pressure control system and the rod control system. The steam dump system consists of 12 valves arranged in four banks with three valves in each bank. Each bank of the steam dump valves has 25 percent of the total dump capacity of 40 percent nominal steam flow at nominal steam pressure. A failure of one bank of steam dump valves was assumed to be single failure in the steam dump control system. In the pressure control system, the types of failure assumed either a reduction in spray flow capacity or a complete failure to deliver spray flow. A 50 percent reduction in the spray flow (one spray valve stuck closed) was assumed to be a single failure in the pressure control system. The failure assumed in the rod control system was a failure of power mismatch channel. The power mismatch channel signal is to provide a fast feed-forward signal to the rod control system during a rapid change in a turbine load. With the assumption of a failure of a power mismatch channel, the control rod is controlled by the $T_{avg}$ error signal, resulting a slower response and a longer delay time to drive the rods into the core following the turbine trip. The licensee analyzed 14 cases with various combinations of control system failures (from a single failure up to four failures in the control systems) and presented the results.

In the analyses, the licensee made the conservative assumptions that maximized the calculated pressurizer pressures. As a result, they maximized the calculated potential to challenge the PORVs to open. The assumptions were the following:

- Transients were initiated from 52 percent power. This assumption is conservative since 50 percent power is the maximum proposed value that would permit a turbine trip without actuating a reactor trip and transients initiated from a higher core power would be more severe with respect to predicting the peak pressurizer pressure.

- Best-estimate beginning-of-life reactivity parameters for the IP3 cycle 10 core were used. This assumption involved use of the minimum moderator reactivity feedback and thus, resulted in a minimum decrease in nuclear power due to the initial increase in primary coolant temperature during the transient.

- No steam generator tube plugging (SGTP) was assumed. A zero percent SGTP level results in a higher peak pressurizer pressure because of the lower initial RCS average temperature and the consequential smaller steam dump demand signal.

- LOFTRAN was used to perform the system response. The use of LOFTRAN is conservative because comparison studies have shown that LOFTRAN overpredicts pressurizer pressure during pressure-increase transients.

The results of the analyses showed that (1) for cases with all control systems available, the calculated pressurizer pressure did not reach the point to actuate pressurizer PORVs and (2) for cases with a single failure and multiple failures in the control systems, the pressurizer PORVs were actuated for nine cases (cases 3, 5, 6, 7, 9, 11, 12, 13 and 14 in Table 2 of reference 2.) As shown in reference 3, the highest pressurizer water level of about 53 percent of the span was calculated for case 7, the TTWORT event with a failure of all banks of steam dump valves. For all the analyzed cases, the calculated water levels are below the location where the PORVs are seated and the PORVs, when actuated, release steam only. The calculated steam conditions in the top region of the pressurizer assure that the PORVs will reclose during transients since the PORVs are designed to open and close in a steam environment. Therefore, the staff concludes that the licensee's analyses provide reasonable assurance that the likelihood of a small-break LOCA resulting from stuck-open PORVs is not affected by the proposed TSs and thus, the licensee has addressed the staff concerns of TMI Action Item II.K.3.10 satisfactorily.

Based on its review, the staff finds that the licensee's analysis demonstrates adequately that the proposed TSs do not change the results of the transient and the LOCA analyses in the current safety analysis report. The licensee has also satisfactorily addressed the staff's concerns of TMI action Item II.K.3.10 by showing that the proposed TSs do not affect the likelihood of small-break LOCAs resulting from stuck-open pressurizer PORVs. Therefore, the staff concludes that the proposed TSs 2.3.1.C.(3), 2.3.2.B, Table 3.5-2, item 12, Table 4.1-1, item 21 and associated bases with respect to the increased power level required for the reactor trip on a turbine trip are acceptable. The removal of the last sentence in TS 2.3.2.A regarding overspeed testing is consistent with the proposed TS 2.3.2.B and is also 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. 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 (64 FR 19563). 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. Sun

Date: September 8, 1999