October 6, 1986

Docket No. 50-286

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L. Harmon	ACRS(10)
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B. Grimes	LFMB

Mr. John C. Brons
Senior Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

The Commission has issued the enclosed Amendment No. 68 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 April 30, 1986.

The amendment revised the Technical Specifications to add anticipatory reactor trip upon turbine trip to a list of other reactor trips.

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

Sincerely,

Original signed by:

Joseph D. Neighbors, Senior Project Manager
PWR Project Directorate #3
Division of PWR Licensing-A, NRR

Enclosures:
1. Amendment No. 68 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

* see previous concurrence

PD#3*	PD#3	PD#3	OGC
CVogan	DNeighbors:mak	SVarga	M. Karman
9/ /86	9/24/86	10/6/86	9/24/86

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9/24/86	9/ /86	9/ /86

Mr. John C. Brons
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

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. 68
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 30, 1986, 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. 68 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Director
PWR Project Directorate #3
Division of PWR Licensing-A, NRR

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 6, 1986

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

ATTACHMENT TO LICENSE AMENDMENT NO.68

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

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 2)	Table 3.5-2 (sheet 2 of 2)
Table 4.1-1 (sheet 2 of 5)	Table 4.1-1 (sheet 2 of 5)

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.

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

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

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

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

The reactor trip on turbine trip may be blocked at power levels $\geq$ 10% during turbine overspeed surveillance testing.

B. The single loop loss of flow reactor trip may be bypassed when the power range nuclear instrumentation indicates $\leq$ 50% of rated power. The single loop loss of flow reactor trip may be bypassed below 75% of rated power only after the overtemperature ΔT trip setpoint has been adjusted to the three-loop operation value given in 2.3.1(B)4 above. The single loop loss-of-flow trip setpoint is hereafter referred to as P-8.

Basis

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 set point protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis. (2) (3)

The constants delta-T_0 and T' for each overtemperature and overpower protection channel are set in accordance with the indicated delta-T and Tavg 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 Tavg 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 set points 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 set point 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 set point 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 set point 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 set points 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 design limit during normal operational transients.

A turbine trip causes a direct reactor trip, when operating at or above 10% power. 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)

The turbine and steam-feedwater flow mismatch trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14).

References

(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)

Table 3.5-2 (Sheet 2 of 2)

		1	2	3	4	5
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
12.	Turbine trip a. Electrical overspeed protection	3	2	2	1	Turbine shut-down (turbine valves closed)
	b. Low auto stop oil pressure	3	2	2	1	Maintain reactor power below 10% of full power

* 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.

Amendment No. ~~38~~, ~~65~~ 68

Table 4.1-1 (Sheet 2 of 5)

	Channel Description	Check	Calibrate	Test	Remarks
10.	Steam Generator Level	S	R	M	
11.	Residual Heat Removal Pump Flow	N.A.	R	N.A.	
12.	Boric Acid Tank Level	S	R	N.A.	Bubbler tube rodded during calibration
13.	Refueling Water Storage Tank Level	W	R	N.A.	Low level alarms
14.	(a) Containment Pressure	S	R	M	High
	(b) Containment Pressure	S	R	M	High High
15.	Process and Area Radiation Monitoring Systems	D	R	Q	
16.	Containment Water Level Monitoring System				
	(a) Containment Sump (Narrow Range, Analog)	N.A.	R	N.A.	
	(b) Recirculation Sump (Narrow Range, Analog)	N.A.	R	N.A.	
	(c) Containment Water Level (Wide Range)	N.A.	R	N.A.	
17.	Accumulator Level and Pressure	S***	R	N.A.	
18.	Steam Line Pressure	S	R	M	
19.	Turbine First Stage Pressure	S	R	M	
20.	Logic Channel Testing	N.A.	N.A.	M	
21.	Turbine Trip				
	a. Independent Overspeed Protection (Electrical)	N.A.	R	M	
	b. Low Auto Stop Oil Pressure	N.A.	R	N.A.	
22.	Boron Injection Tank Return Flow	S	R	N.A.	

*** If either an accumulator level or pressure instrument channel is declared inoperable, the remaining level or pressure channel must be verified operable by interconnecting and equalizing (Pressure and/or level wise) a minimum of two accumulators and crosschecking the instrumentation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 68 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

INTRODUCTION

By letter dated April 30, 1986, the Power Authority of the State of New York (the licensee) requested amendment to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. This amendment would list the existing anticipatory reactor trip on turbine trip in the Technical Specifications (TS).

DISCUSSION AND EVALUATION

As a result of the accident at the Three Mile Island Nuclear Unit No. 2, the Commission developed a comprehensive and integrated plan to improve safety at commercial power reactors (NUREG-0660, NRC Action Plan Developed as a Result of TMI-2 Accident, May 1980). Among the action items required by the plan was the confirmation, by all owners of Westinghouse designed reactors, of the existence of an anticipatory reactor trip upon turbine trip.

As discussed in NUREG-0611 "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants," analyses conducted by Westinghouse demonstrated that the anticipatory reactor trip upon turbine trip prevents unnecessary challenges to the power operated relief valve (PORV). Thus, this trip assists in circumventing the adverse impact of the potential failure of the PORV to completely reseat following actuation. NUREG-0660 directed licensees of all Westinghouse reactors, where this trip was found to be unavailable, to provide a conceptual design and evaluation for the trip's installation.

In response to this directive, by letter dated June 16, 1980, the licensee confirmed the availability of the desired reactor trip for Indian Point No. 3 (IP-3). However, the licensee did not submit a proposal for modifying the unit's TS to include this trip among the reactor trips comprising the unit's reactor protection system. (The need for TS changes was not explicitly addressed in NUREG-0660).

Publication of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980, provided, among other things, scheduling, implementation and clarification of technical positions for action items addressed in NUREG-0660. Among the items addressed, was the need for technical specifications to reflect the existence of the reactor trip upon turbine trip and to provide limiting conditions for operation, bases, and surveillance requirements for the trip (Item II.K.3.12)

In compliance with the NUREG-0737 directive and in response to NRC letter dated December 11, 1985, the licensee has submitted by letter dated April 30, 1986, the following proposed Technical Specification (TS) changes:

1. TS page 2.3-4: statements adding (a) the anticipatory reactor trip upon turbine trip to the list of identified reactor trips, (b) instrumentation setting for the reactor trip interlock, and (c) criterion for blocking the trip.

2. TS page 2.3-6: statement adding the bases for including the trip in the lists of protective instrumentation settings for reactor trip interlocks.

3. TS page 2.3-7: lists, as a reference to TS section 2.3, Generic Letter 82-16, section II.K.3.12 which directed the change to the TS to include the reactor trip upon turbine trip.

4. TS Table 3.5-2 (sheet 2 of 2): includes turbine trip for low auto stop oil pressure above or equal to 10 percent of full power in the list for reactor trip instrumentation limiting operating conditions.

5. TS Table 4.1-1 (sheet 2 of 5): adds turbine trip for low auto stop oil pressure to this table of minimum frequencies for checks, calibrations and tests of instrument channels.

The changes proposed by the licensee identify the availability of an additional reactor trip which is already a part of the existing reactor protection system, i.e., no modification to the reactor protection system's logic or hardware is required. An item by item review of the proposed changes surfaced no technical concern, in fact, the proposed Technical Specification changes comply with the recommendation contained in NUREG-0660 and are in accord with NUREG-0452, Rev. 5, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors." The staff therefore finds the proposed changes acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 this amendment.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 6, 1986

PRINCIPAL CONTRIBUTOR:

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