ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO ANTICIPATORY REACTOR TRIP ON TURBINE TRIP

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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. ~~26~~

Amendment No. 38, 65

	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.

ATTACHMENT II

SAFETY EVALUATION

RELATED TO

ANTICIPATORY REACTOR TRIP ON TURBINE TRIP

TECHNICAL SPECIFICATION CHANGES

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

Section I - Description of Change

The proposed changes to page 2.3-4 adds anticipatory reactor trip upon turbine trip to a list of other reactor trips. It adds anticipatory reactor trip upon turbine trip above or equal to 10% of rated power to the list of protective instrumentation settings for reactor trip interlocks. In addition, a change to this page discusses blocking of the reactor trip on turbine trip to permit turbine overspeed surveillance testing.

The proposed change to page 2.3-6 adds the bases for including this trip in the list of the protective instrumentation settings for reactor trip interlocks.

The proposed change to page 2.3-7 includes a reference to Generic Letter 82-16, II.K.3.12 which requested Anticipatory Reactor Trip on Turbine Trip Technical Specifications.

The proposed change to Table 3.5-2 (Sheet 2 of 2) includes Turbine Trip for Low Auto Stop Oil Pressure above or equal to 10% of full power in the list for reactor trip instrumentation limiting operating conditions.

The proposed change to 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.

Section II - Evaluation of Change

The purpose of this proposed change is to include an anticipatory reactor trip upon turbine trip in the Indian Point 3 Technical Specifications. NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," describes analyses performed by Westinghouse which demonstrates that the anticipatory reactor trip on turbine trip will reduce the ensuing Reactor Coolant System pressure transients.

The anticipatory reactor trip on turbine trip may be blocked during turbine overspeed surveillance testing in order to ensure safe operation of the turbine overspeed trip.

The Authority has previously confirmed that Indian Point 3 has an anticipatory reactor trip on turbine trip (IPN-80-117, dated December 30, 1980). This change merely adds additional surveillance to an existing condition.

Section III - No Significant Hazards Evaluation

The purpose of these changes is to incorporate those Technical Specifications related to an anticipatory reactor trip upon turbine trip.

In accordance with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based upon the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response

The proposed Technical Specification changes reflect existing plant instrumentation configuration. This configuration is described in the Final Safety Analysis Report (FSAR). Plant system operation or functions will not be affected and new systems or equipment are not being introduced.

Therefore, the Technical Specification changes will not involve any significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response

A new or different kind of accident cannot be created by the proposed changes. As discussed in NUREG-0611, analyses performed by Westinghouse have demonstrated that the anticipatory reactor trip on turbine trip will reduce the ensuing Reactor Coolant System pressure transients. This proves to be beneficial to the safe shutdown of the plant during certain overpressure transients. The changes incorporate an existing instrument trip condition into the Technical Specifications. The function of systems and equip- ment have not been impacted by the changes.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response

The proposed changes add an anticipatory reactor trip upon turbine trip. The incorporation of an existing instrumentation trip condition into the Technical Specifications provides additional assurance that Generic Letter 82-16, Item II.K.3.12 will be satisfied. Therefore, these changes do not involve any significant reduction in a margin of safety.

In the April 6, 1983 Federal Register, Vol. 48, No. 67, Page 14870, the NRC published a list of examples of amendments that are not likely to involve a significant hazards concern. Example (ii) of that list applies to the addition of the anticipatory reactor trip upon turbine trip and states:

> "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement."

Section IV - Impact of Change

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of these changes: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI- References

a) IP-3 FSAR
b) IP-3 SER
c) NUREG-0737
d) Generic Letter 82-16