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U.S. Nuclear Regulatory Commission
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SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Additional Information Regarding Proposed Changes to Technical Specifications Regarding Setpoint Change of Automatic Reactor Trip on Turbine Trip

REFERENCE: 1. NYPA Letter to NRC, IPN-99-008, "Proposed Changes to Technical Specifications Regarding Setpoint Change of Automatic Reactor Trip on Turbine Trip", dated January 28, 1999.

Dear Sir:

This letter provides additional information regarding the Authority's proposed change (Reference 1) to the Technical Specification setpoint for automatic reactor trip on turbine trip.

Attachment I provides responses to three items identified during a teleconference with the NRC staff.

This letter does not modify the proposed change to the Technical Specifications and does not affect the conclusions of the safety evaluation provided in Reference 1.

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There are no commitments made by the Authority in this letter. If you have any questions, please contact Mr. Ken Peters at (914) 736-8029.

Very truly yours,

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Item #1: Additional information concerning why FSAR Chapter 14 events are not affected by this modification.

The Authority reviewed each of the fourteen accident analyses described in Chapter 14 of the FSAR with respect to the proposed Technical Specification (TS) setpoint change for automatic reactor trip on turbine trip. The proposed change is to increase the setpoint from 10% reactor power to ≤ 50%. Based on this review, the Authority concludes that the proposed change has no effect on the accident analyses. The following sections summarize the results of the review.

1. ***UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL CONDITION***

Event Definition: A rod cluster control assembly (RCCA) bank withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of one or more RCCA banks, resulting in a power excursion. This could occur with the reactor either subcritical, at hot zero power or at power.

Plant Operating Conditions: The analysis assumes that the reactor is initially critical at 10^{-9} fraction of nominal power, which is below the power level expected for any shutdown condition.

Effect of proposed setpoint change: In this scenario, the reactor is not critical and the turbine generator is not on-line. Therefore, this proposed TS change has no affect on this accident scenario.

2. ***UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER***

Event Definition: The design basis for this event is defined as the inadvertent addition of reactivity to the core caused by the withdrawal of an RCCA bank(s) while at power.

Plant Operating Conditions: Cases corresponding to initial power levels of 100, 60 and 10 percent of nominal Rated Thermal Power are analyzed. For all cases analyzed, the results show that (1) integrity of the core is maintained by the reactor protection system as the departure from nucleate boiling (DNBR) remains above the safety analysis limit value, (2) the pressure remains below the accident analysis limit for the primary and secondary systems, and (3) the pressurizer does not fill.

Effect of proposed setpoint change: In this scenario, the reactor trip is provided by the automatic actuation of the first reactor protection signal reached; either overtemperature delta temperature (OTΔT) or power range high neutron flux. Neither of these reactor protective signals are affected by the turbine trip signal. Therefore, this proposed TS change has no affect on this accident scenario.

3. ***ROD ASSEMBLY MISALIGNMENT***

Event Definition: The design basis rod cluster control assembly misalignment event is categorized as an event which could be initiated by one statically misaligned RCCA from normal or allowed RCCA bank position resulting in reactivity and power distribution anomalies.

Plant Operating Conditions: Reactor at full power.

Effect of proposed setpoint change: A reactor trip is provided by the automatic actuation of the RPS by the low pressurizer pressure signal or by the OTΔT trip signal. Neither of these reactor protective signals are affected by the turbine trip signal. Therefore, this proposed TS change has no affect on this accident scenario.

4. ***ROD CLUSTER CONTROL ASSEMBLY (RCCA) DROP***

Event Definition: The dropped RCCA accident is initiated by a single electrical or mechanical failure which causes any number and combination of rods from the same group of a given bank to drop to the bottom of the core.

Plant Operating Condition: The analysis was performed with the plant at full power.

Effect of proposed setpoint change: The reactor trip occurs on OTΔT which is not affected by the reactor trip on turbine trip setpoint. Therefore, this proposed TS change has no affect on this accident scenario.

5. ***CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION***

Event Definition: The inadvertent dilution of the reactor coolant system (RCS) boron concentration due to a chemical and volume control system (CVCS) malfunction or faulty operator action. The limiting scenario considered here is the inadvertent opening of the primary water makeup control valve and failure of the blend system, either by controller or mechanical failure.

Plant Operating Condition: The analysis is performed for an inadvertent dilution of the RCS for power operation (mode 1), startup (hot zero power) and refueling modes of plant operation.

Effect of proposed setpoint change: In Mode 1, the power and temperature rise will cause the reactor to reach the OTΔT trip setpoint resulting in a reactor trip. The reactor trip occurs at OTΔT which is not affected by the reactor trip on turbine trip setpoint. Therefore, this proposed TS change has no affect on this accident scenario.

6. ***LOSS OF REACTOR COOLANT FLOW***

A) Partial and Complete Loss of Reactor Coolant Flow

Event Definition: The loss of flow incident can result from a mechanical or electrical failure in a reactor coolant pump (RCP), or from a fault in the power supply of these pumps.

Plant Operating Condition: The plant is assumed to be operating at full power. Bounding analyses are performed at full power since this is the most conservative in terms of potential consequences, specifically a more limiting minimum DNBR.

Effect of proposed setpoint change: The reactor trip is initiated by low primary coolant loop flow, RCP undervoltage, RCP underfrequency, or low reactor coolant flow. This trip is not affected by the reactor trip on turbine trip setpoint. Therefore, this proposed TS change has no affect on this accident scenario.

B) Reactor Coolant Pump Shaft Seizure (locked rotor)

Event Description: The design basis reactor coolant pump shaft seizure event is defined as an instantaneous seizure of a single RCP rotor which results in a rapid reduction in reactor coolant loop flow from full power. The locked rotor event is based on a hypothesized mechanical interference with the RCP impeller.

Plant Operating Condition: The locked rotor case is based on assuming that the plant is operating at maximum reactor coolant pressure, average temperature and power at the thermal design flow rate when the event occurs.

Effect of proposed setpoint change: The reactor trip is initiated by low primary coolant loop flow. This trip is not affected by the reactor trip on turbine trip setpoint. Therefore, this proposed TS change has no affect on this accident scenario.

7. ***STARTUP OF AN INACTIVE REACTOR COOLANT LOOP***

Event Definition: The analysis conservatively assumes initial conditions representative of this event with three (3) coolant loops operating.

Plant Operating Condition: A power level equal to 77% of hot full power.

Effect of proposed setpoint change: The reactor trip is assumed to initiate at 118% of reactor power (high power range monitors). The nuclear power increase causes the trip to occur as a result of the increase in core flow which causes a decrease in core average temperature before the core inlet temperature begins to decrease. Therefore, this proposed TS change has no affect on this accident scenario.

8. ***LOSS OF EXTERNAL ELECTRICAL LOAD***

Event Definition: The loss of external electrical load and/or turbine trip event is defined as a complete loss of steam load from full power without a direct reactor trip, or a turbine trip without a direct reactor trip.

Plant Operating Condition: The evaluation is performed for a complete loss of steam load from full power with no credit taken for the direct reactor trip on turbine trip.

Effect of proposed setpoint change: The evaluation was performed for a complete loss of steam load from full power with no credit taken for direct reactor trip on turbine trip. The reactor trip is initiated by the OTΔT, high pressurizer pressure, high pressurizer water level, or low-low steam generator water level signals. Therefore, this proposed TS change has no affect on this accident scenario.

9. ***LOSS OF NORMAL FEEDWATER***

Event Definition: The design basis loss of normal feedwater event is defined as a reduction in the capability of the secondary system to remove heat generated in the reactor core.

Plant Operating Condition: A complete loss of main feedwater flow is assumed to occur from 102% of Rated Thermal Power without a direct reactor trip from the loss of offsite power.

Effect of proposed setpoint change: Reactor trip is initiated from any of the following RPS trip signals: low-low steam generator water level signal, OTΔT, high pressurizer pressure, high pressurizer water level and RCP undervoltage. These trips are not affected by the setpoint change. Therefore, this proposed TS change has no affect on this accident scenario.

10. ***EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS***

Event Definition: This event is defined as an increase in feedwater flow to the steam generator. The increase in feedwater flow will result in an increase in the heat transfer rate from primary to secondary in the steam generators and a consequential reduction in primary system temperature and pressure. The transient responses for an excessive feedwater flow event were analyzed for four cases: two cases at hot full power (one case with automatic rod control and one without) and two cases at hot zero power (one case with automatic rod control and one without).

Plant Operating Condition: Cases are analyzed at power levels corresponding to zero and full load. Feedwater isolation and turbine trip occur on the high-high steam generator water level signal. Since no primary side reactor trip setpoint is reached for the full power cases, reactor trip occurs on turbine trip. For the zero-power cases, there is no reactor trip actuated.

Effect of proposed setpoint change: For full power conditions, feedwater isolation and turbine trip (with subsequent reactor trip signal on turbine trip) occur on the high-high steam generator water level signal. For the zero-power cases, there is no reactor trip actuated. Therefore, this proposed TS change has no affect on this accident scenario.

11. ***EXCESSIVE LOAD INCREASE INCIDENT***

Event Definition: An excessive load increase event is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand.

Plant Operating Condition: The reactor control system is designed to accommodate a 10% step-load increase or a 5% per minute ramp load increase in the range of 15 to 100% power. Any loading rate in excess of these values may cause a reactor trip by the reactor protection system (RPS).

Effect of proposed setpoint change: Although the RPS is assumed to be operable, the reactor trip in not encountered in this analysis. Therefore, this proposed TS change has no affect on this accident scenario.

12. ***LOSS OF ALL AC POWER TO THE STATION AUXILIARIES***

Event Definition: A complete loss of non-emergency AC power may result in the loss of all power to the plant auxiliaries; i.e., the RCPs, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of onsite non-emergency AC distribution system.

Plant Operating Condition: The plant is initially operating at 102% of rated thermal power.

Effect of proposed setpoint change: The reactor will trip due to: (1) turbine trip (2) upon reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coast down and decrease in secondary heat removal; or (3) due to loss of power to the control rod drive mechanisms as a result of the loss of power to the plant. Therefore, this proposed TS change has no affect on this accident scenario.

13. ***STARTUP ACCIDENTS WITHOUT REACTOR COOLANT PUMP OPERATION***

As noted in the Technical Specifications under "Limiting Conditions for Operation," the reactor is not permitted to be critical above 2% rated power unless at least two reactor coolant pumps are in operation except for low power tests and natural circulation tests. These tests are conducted under carefully approved procedures and supervision for the purpose of insuring control of core power and control of any reactivity insertion.

Therefore, this proposed TS change has no affect on this scenario.

14. ***STARTUP ACCIDENT WITH A FULL PRESSURIZER***

The Technical Specifications under "Minimum Conditions for Criticality" state that: "The reactor shall be maintained subcritical by at least 1% δk/k until normal water level is established in the pressurizer." In view of this restriction, the reactor will not be solid when criticality is achieved.

Therefore, this proposed TS change has no affect on this scenario.

Item #2: Peak pressurizer water level for failure mode cases where the pressurizer power operated relief valves (PORVs) are actuated.

The pressurizer PORVs were actuated for failure mode cases 3, 5, 6, 7, 9, 11, 12, 13 and 14 (See "P-8 Permissive (Deletion of Reactor Trip on Turbine Trip) Setpoint Analysis for Indian Point Unit 3", November 1998, attachment to INT-99-203). Among these cases, the highest peak pressurizer level of 952 ft^3 (including the pressurizer surge volume), or about 53% of span, was calculated for Case 7. The peak water level occurred at 53 seconds.

When actuated, the pressurizer PORVs relieve steam only and there was no water discharge through the PORVs.

Item #3: Identification of the codes used in the analysis.

The LOFTRAN code, as documented in WCAP-7878, was used for this analysis.

REFERENCES

1. Indian Point 3 Nuclear Power Plant Final Safety Analysis Report

2. Westinghouse Letter INT-99-219 from Mr. Robert R. Laubham to New York Power Authority, "Transmittal of Responses to NRC RAIs Regarding Deletion of Reactor Trip on Turbine Trip Below P-8 (SECL-97-140, Rev. 2)", dated March 22, 1999.