POWER AUTHORITY OF THE STATE OF NEW YORK

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December 21, 1979
IPN-79-100

Director, Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Low Power Physics Testing

Dear Sir:

On or about December 26, 1979, the Authority plans to perform a hydrostatic test of the reactor coolant system at its Indian Point 3 facility followed by low power physics testing ($\leq$ 5% power). The current schedule for the return to service of the Indian Point 3 facility from the on-going refueling outage is prior to February 1, 1980. The Authority is scheduling these tests early in order to prevent the completion of these tests from becoming a critical path or delaying item for the return to service.

The Authority presently has an acceptable Appendix K ECCS analysis based on the NRC approved Westinghouse October 1975 evaluation model with the appropriate correction for the Zr-H_2O reaction. This analysis was performed based on 0% steam generator tube plugging. The Authority has plugged up to 3.65% of the tubes in the IP-3 steam generators during the current refueling outage. The Authority has requested the Westinghouse Electric Corporation to provide a new Appendix K ECCS analysis based on the NRC approved February 1978 evaluation model considering up to 4% steam generator tube plugging. This new analysis will be available to the Authority by the end of December 1979. This reanalysis will not reduce the current FQ^T limit of 2.17 and will not require Technical Specification changes for a return to service.

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The Authority has investigated with Westinghouse the necessity of having the new Appendix K analysis completed prior to the start of the low power physics tests. The Authority has determined that ample margins exist in the present analyses to the 10 CFR §50.46 limits for up to 5% power. In a phone conservation on November 9, 1979 between Mr. Olshan and Ms. Chatterton of your staff and Messrs. Wilverding, Grochowski, Khan, Deem, Iyer and Goyette of the Authority's staff and Messrs. Johnson and Nixdorf of Westinghouse, the above described situation was presented to your staff together with the Authority's justification for its conclusion. These reasons are as follows:

1. The primary determinant of the LOCA ECCS result is the peak kw/ft in the core allowed by the Technical Specifications.

2. At 5% power, the peak kw/ft in the core is less than 11.5% of the peak kw/ft in the core at 100% power (based on current 100% power maximum allowable $F_Q^T = 2.17$).

3. At 5% power, the fuel average temperature (stored energy) in the hottest rod is reduced by more than 1300°F.

4. Utilizing the LOCA ECCS sensitivity to tube plugging established for Indian Point 2, one would expect a maximum of F_Q^T reduction of approximately 0.06 for as much as 6% tube plugging. (One expects no change to Indian Point 3 allowable F_Q^T even when considering tube plugging since the analysis with the February, 1978 ECCS evaluation model should compensate for the effect of tube plugging).

   If we apply four times this reduction to Indian Point 3, or a reduction in allowable F_Q^T of 0.25, as an extremely conservative estimation of potential result of the new analysis, the peak kw/ft in the core at 5% power is still only 13% of that allowed at an F_Q^T of 1.92 (= 2.17 - 0.25).

These considerations demonstrate the ability of the Indian Point 3 facility to operate up to 5% power for completion of the low physics tests prior to completion of the revised ECCS analysis. It was concluded by your staff also that such operation was acceptable. This situation has been reviewed by the Authority's Plant Operating Review Committee and Safety Review Committee. The safety committees have determined that the low power physics testing (a) does 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) does not increase the probability for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis

Report; (c) does not reduce the margin of safety as defined in the basis for any Technical Specification, and (d) does not constitute an unreviewed safety question pursuant to 10 CFR §50.59.

Very truly yours,

Paul J. Early
Assistant Chief Engineer-Projects