John D. O'Toole
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June 14, 1985

Re: Indian Point Unit No. 2
Docket No. 50-247

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

ATTN: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Dear Mr. Varga:

The attachment to this letter provides our response to Items 1 and 6 of the enclosure to your April 24, 1985 letter regarding unresolved questions on TMI Action NUREG-0737 II.D.1 for Indian Point Unit No. 2. With respect to the remaining questions, Items 2, 3, 4 and 5, we are planning to submit the requested information by September 30, 1985.

Should you or your staff have any questions, please contact us.

Very truly yours,

John D. O'Toole

cc: Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, New York 10511

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ATTACHMENT A

RESPONSE TO NRC 4/24/85 REQUEST
FOR ADDITIONAL INFORMATION
REGARDING
NUREG-0737, ITEM II.D.1
PERFORMANCE TESTING OF RELIEF AND SAFETY VALVES

Question 1

In response to Question 2 regarding qualification of the Indian Point 2 safety valves for the fluid conditions resulting from a feedwater line break accident, the June 15, 1984 submittal states that the feedwater line break accident is not a part of the Indian Point 2 licensing basis since the plant was licensed prior to issuance of Regulatory Guide 1.70 Rev. 2. This response is not acceptable. Item II.D.1 in NUREG-0737 specifically requires that PORVs and safety valves be qualified for fluid conditions resulting from transients and accidents referenced in Regulatory Guide 1.70 Rev. 2. The feedwater line break accident is specifically defined in Regulatory Guide 1.70 Rev. 2. Additionally, from the staff review of other plant specific responses to Item II.D.1, it is clear that for many Westinghouse plants the feedwater line break accident is the limiting case for providing high pressure liquid to the safety valves, a fluid for which they were not specifically designed originally. This is exactly the type of concern that NUREG-0737, II.D.1 was established to address. In accordance with the requirements of the NUREG, we require that information be provided to demonstrate that the PORVs and safety valves will function as required to assist in safe shutdown of the plant and will not experience any degradation that would inhibit safe plant shutdown if exposed to the feedwater line break accident.

Response

Since the feedline break accident is not part of the Indian Point Unit 2 (IP2) licensing basis a rigorous FSAR type analysis is not available and may not be required to demonstrate valve adequacy during a feedwater line break accident. In developing the assumptions for a FSAR type feedwater line break analysis, most plants use extreme conservatism to provide a

non-realistic conservative basis to which the auxiliary feedwater system is shown to be adequately designed. The intended results of using these conservative assumptions is to maximize fluid loss and minimize heat removal through the safety valves, thereby maximizing the RCS heatup, decreasing margin to core uncovery, and increasing the design basis heat removal requirement of the auxiliary feedwater system. Listed below are some of the conservative assumptions used in present day licensing for a typical FSAR type analysis of a PWR feedwater line break:

1. No credit is taken for the high pressurizer pressure reactor trip.
2. Main feedwater flow to all steam generators is assumed to be lost at the time the break occurs (all main feedwater spills out through the break).
3. The worst possible break area is assumed.
4. A conservative feedline break discharge quality is assumed prior to the time the reactor trip occurs.
5. The Auxiliary Feedwater System is assumed to supply a minimal flowrate.
6. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.
7. No credit is taken for charging or letdown.
8. Conservative core residual heat generation is assumed based upon long term operation at the initial power level preceding the trip.
9. No credit is taken for operator actions in the first 10 minutes of the accident.

With the above extreme assumptions, the results of a most typical FSAR analysis would certainly indicate that the limiting case for providing high pressure liquid challenge to the safety valves is the main feedline break accident, the basis of NRC,s Question No. 1.

However, a more representative analysis for feedline break using more realistic yet conservative assumptions with respect to the above considerations would demonstrate that liquid water relief through the safety valves is precluded and that the results of the EPRI testing bound the IP2 PORV & SV design. Such analyses has been performed by Westinghouse for the Westinghouse Owners Group (WOG) for the purpose of describing the thermal hydraulic behavior of the RCS for feedline and steamline breaks, not for auxiliary feedwater system design. These analyses were performed for the Reference Low Pressure ECCS Plant design (1500 psi) that forms the basis of the upgraded WOG Emergency Response Guidelines (ERG) and Operator Training per the requirements of NUREG-0737, Supplement 1. These ERGs and background analyses are applicable to IP2 and have been used in the development of our upgraded Emergency Operating Procedures. Thus, in response to this question we submit Attachment B to this letter which provides typical transient responses of the RCS thermal hydraulic behavior for a large double-ended rupture feedline break accident, from full power, with best estimate modeling assumptions with regard to physical properties, initial conditions and the response of available control systems. The analysis

models a low pressure 1500 psi safety injection system, which is of similar design to the IP2 Safety Injection System. Although the analysis results presented in Attachment B are based on the generic Reference Low Pressure Plant design, the transient trends and response are similar for all PWR type plants including IP2. Thus, the response to a feedline break for IP2 under these conditions will be similar to those predicted in Attachment B. As shown in Figure 9 of Attachment B, the water level in the pressurizer rapidly drops from approximately the normal Reference Plant level of 68% to approximately 42% following a reactor and turbine trip. The water level then recovers to approximately 47% one minute thereafter (two minutes into the accident) and then remains constant. At no time during the transient does the pressurizer approach a "water-solid" state. Also, Figure 8 of Attachment B shows that the PORV or SV is never challenged. It is, therefore, concluded that under these conditions, it is unlikely that liquid discharge through the Pressurizer safety valves and PORVs will ever occur during a feedline break accident at IP2. Furthermore, since IP2 operates with normal pressurizer level at approximately 23% even more margin exists to approaching a "water-solid" state.

Question 6

NUREG-0737 Item II.D.1 requires that the plant-specific PORV control circuitry be qualified for design-basis transients and accidents. Please provide information which demonstrates that this requirement has been fulfilled.

Response

The electrical components associated with the operation of the PORV, which are exposed to harsh environments as a result of design basis accidents, were addressed and determined qualified as indicated in our submittal to the NRC dated October 5, 1984 and the NRC's response in their SER dated December 7, 1984. Also, it should be noted that the design-basis of IP2 requires that the PORVs be isolated since no credit is taken in the accident analyses for their relief capabilities. As such, the PORV Block Valves are qualified to isolate the PORVs in a harsh environment as well as the PORV solenoids being capable of dumping air or N_2 from the PORV operators. This isolation capability is included in the IP2's Technical Specifications.