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John C. Brons
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March 7, 1988
IPN-88-006

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Additional Information Related To NUREG-0737, Item II.D.1 "Pressurizer Safety and Relief Valves (PRZR S/RVs) Testing"

References 1. NYPA letter (J. C. Brons) to NRC (S. A. Varga) on Additional Information re. PRZR S/RVs Testing, dated August 15, 1985 (IPN-85-42).

2. NRC request for information on IP-3's Evaluation of NUREG-0737, Item II.D.1, dated June 6, 1985.

Dear Sir:

In Reference 1, the Authority responded to the NRC's letter (Reference 2) regarding plant specific issues related to NUREG-0737, Item II.D.1. Attachment I to this letter provides information requested by NRC staff during telephone conversations, concerning Reference 1, over the past year.

Should you or your staff have any questions regarding this matter, please contact Mr. P. Kokolakis of my staff.

Very truly yours,

John C. Brons
John C. Brons
Executive Vice President
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Attachments

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ATTACHMENT I TO IPN-88-006

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION-NUREG-0737, ITEM II.D.1 PERFORMANCE TESTING OF PRESSURIZER RELIEF AND SAFETY VALVES

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

Question 1

In response to question 2 of NRC's June 6, 1985 letter regarding qualification of the IP-3's pressurizer relief and safety valves for the fluid conditions resulting from a feedwater line break accident, the Authority's August 15, 1985 submittal (Reference 1) states that a feedwater line break is not part of the licensing basis for the plant. Based on NRC's review of other plant specific responses, it is clear that for many Westinghouse plants the feedwater line break accident is limiting for high pressure liquid discharge from the relief and safety valves. Accordingly, the staff requests the Authority to address this concern.

Response

As indicated in the Authority's August 15, 1985 submittal, a Regulatory Guide 1.70, Revision 2 FSAR-type feedwater line break accident is beyond the original licensing design basis for IP-3 and, therefore, no plant specific analysis is available. The following information is provided to clarify the Authority's position regarding this item.

An analysis of a feedline break using more realistic yet conservative assumptions, has demonstrated that liquid water relief through the PRZR S/RVs is precluded. This analysis 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. These analyses were performed for the Reference Low Pressure ECCS Plant design (1500 psi) and form the bases of the upgraded WOG Emergency Response Guidelines (ERGs) and Operator Training per the requirements of NUREG-0737, Supplement 1. The ERGs and background analyses are applicable to IP-3 and have been used in the development of the plant upgraded Emergency Operating Procedures (EOPs).

Enclosure 1 to this letter provides a typical transient response of the RCS thermal hydraulic behavior for a large double-ended rupture feedline break accident from full power. The analysis includes 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 that of IP-3. The responses to a feedline break for IP-3 under these conditions are therefore expected to be similar to those predicted in the enclosure.

As shown in Figure 9 of the enclosure, 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" condition. Also, Figure 8 of Enclosure 1 shows that neither the PORVs nor SVs are 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 event at IP-3.

Since IP-3 operates with normal pressurizer level at approximately 44% (versus 68% for the reference plant), an even greater margin prevents the pressurizer from approaching a "water-solid" state. At IP-3, the main feedwater piping enters the steam generators through a feedring that is located above the U-tubes. Following a postulated feedwater line break located at the inlet nozzle, steam (rather than water) is initially released from the steam generator. The RCS will undergo a cooldown evolution similar to that generated by a main steam line break transient; as for a steam line break event, no high pressure liquid discharge from the PORVs and safety valves is expected.

Furthermore, it should be noted that recently the WOG has completed a study to provide generic justification of acceptable operation of the pressurizer safety valves under water discharge conditions during a feedwater line break (FLB) event. The EPRI Valve Test Program data was evaluated in conjunction with actual feedline break transient analyses from available member utility FSARs. All the plants addressed in the WOG study for which water relief from the pressurizer is predicted have safety valves which will operate reliably under such conditions. In addition, the study compared plants without an FSAR FLB analyses (such as IP-3), to plants that do have this analysis in their FSAR. The purpose of this comparison was to bound unanalyzed plants with an existing FLB analysis. The study concluded that the unanalyzed plants are of an earlier generation than the plants with FLB analyses that predict water relief, and that the older plants have lower power ratings than the newer plants. The lower ratings would minimize the extent of water relief. In addition, in the event of an FLB, operator actions such as realigning auxiliary feedwater and isolating the faulted loop, can preclude water relief for any Westinghouse plant. This conclusion is consistent with the Authority's position that no water relief is expected following an FLB event at IP-3. The WOG/Westinghouse prepared WCAP on this subject is expected to be issued shortly.

Based on the above discussion, the Authority considers that water relief through the PORVs or safety valves will not occur during a postulated feedwater line break.

Question 2

NUREG-0737, Item II.D.1 requires that the plant-specific PORV control circuitry be qualified for design-basis transients and accidents. Provide information demonstrating this requirement is satisfied.

Response

The FSAR transients have been reviewed to determine if any analyzed transients result in a pressure excursion exceeding the PORV setpoint, thereby challenging the PORVs. Three such transients have been identified; these include the loss of load, locked rotor and loss of normal feedwater events. For the loss of load and locked rotor events, no credit is taken for the pressure-reducing effects afforded by actuation of the PORVs. For the loss of normal feedwater event, the PORVs are assumed to open in order to maximize the insurge into the pressurizer. This is an extreme conservatism intended to minimize the margin available with respect to the acceptance criteria (i.e. no water relief). The analysis demonstrates that even with such extreme conservatisms, no water relief will occur for this event. Similarly, analyses of the loss of load and locked rotor events demonstrate no water relief will occur. In each of these event analyses the pressurizer safety valves are relied upon for pressure mitigation.

For each of these events, steam relieved through the PORVs is quenched in the pressurizer relief tank. Since the relief flow is contained within the discharge piping and pressurizer relief tank, the containment environment remains mild.

The only event that has been suggested as having any potential for creating a harsh environment for the PORVs is the feedwater line break. As noted in response to Question 1, the feedwater line break is beyond the IP-3 design basis. However, using the best-estimate modelling assumptions employed in the analysis contained in Enclosure 1, it can be seen that not only is there no water relief, but the PORVs are never challenged. Accordingly, the PORV control circuitry is qualified for the normal (mild) operating environmental conditions that the PORVs are expected to experience.

Question 3

The NRC staff requested supporting documentation for the computer codes referenced in response to Question 11(b) of the Authority's August 15, 1985 submittal.

Response

The information requested is provided in Enclosure 2 to this attachment.