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New York Power Authority

J. Phillip Bayne
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IPN-83-82
September 30, 1983

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

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

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Pressurizer Safety and Relief Valves -
Results of Plant Specific Piping Adequacy
NUREG-0737, Item II.D.1.A

Dear Sir,

The enclosed attachment to this letter provides Supplement 1 to the plant-specific evaluation of the EPRI Safety and Relief Valve Performance Testing which was submitted on December 15, 1982 (IPN-82-80).

This Supplement details the Authority's evaluation regarding the adequacy of the inlet and discharge piping for the pressurizer safety and relief valves. Submittal of this information is being made in accordance with the commitment made in our letter dated June 8, 1983 (IPN-83-55). We believe that this letter along with the attachment now completes all NUREG-0737, Item II.D.1 submittal requirements.

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

Very truly yours,

J. P. Bayne
Executive Vice President
Nuclear Generation

cc: attached

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cc: Resident Inspector's Office
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ATTACHMENT

SUPPLEMENT 1 TO

PLANT SPECIFIC SUBMITTAL

SAFETY AND RELIEF VALVE PERFORMANCE

IN RESPONSE TO

NUREG-0737, ITEM II.D.1

"CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS"

NEW YORK POWER AUTHORITY

INDIAN POINT 3

9/30/83

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APPENDICES

A	9321-F-25453-10	Containment Building Primary Coolant Pressurizer Safety Relief Valve Piping, Sheet No. 1
B	9321-F-25463-8	Containment Building Primary Coolant Pressurizer Safety Relief Valve Piping, Sheet No. 2

3.0 RESULTS OF PLANT SPECIFIC PIPING ADEQUACY

The safety and relief valve piping adequacy was evaluated using the general guidelines provided in Reference 1. The evaluation considered the following two cases:

Case 1: Sequential actuation of the power operated relief valves (PORV's) and safety valves (SV's) at their respective set pressures with pressurizer pressure increasing at a rate of 130 psi/sec and no insulation around the loop seals at the inlets to the SV's.

Case 2: Block valves upstream of the PORV's closed and, only SV's open at their set pressures with pressurizer pressure increasing at a rate of 144 psi/sec and loop seals at the inlets to the SV's wrapped with 3" of calcium silicate. (Note: The volumes of the loop seals in the analysis were reduced to include only liquid at temperatures below 220°F. A graph of loop seal liquid temperature versus loop seal length is provided in Figure 7.)

The transient pressures, temperatures and flow rates were computed using the RELAP-5, MOD 1, cycle 14 computer code. A post processor code (TULIPS) was used to determine the forces during the transient. The presures developed in the SV inlet piping during the loop seal discharge period have not been calculated using the RELAP code, since no viable methods for modelling the valves have been established by the EPRI PWR Safety and Relief Valve Test (S/RV) Program. However, these pressures in the inlet piping during the loop seal discharge have been evaluated by comparison with the Westinghouse generic observations presented in Reference 8.

3.1 Peak Pressure in SV Inlet Piping During Loop Seal Discharge

The Westinghouse evaluation of the EPRI PWR S/RV Test Program (Reference 8) calculates peak pressures in the SV inlet piping during the loop seal discharge period that are in good agreement with observed test data. In their evaluation, Westinghouse performed analyses to determine the variation of peak inlet pressure with loop seal length. The results of these analyses show that the peak pressure for the plant-specific loop seal length (uninsulated) is below the maximum permissible pressure for the plant specific inlet piping for Level C Service as defined in Tables 4-5 and 4-6 of Reference 8.

3.2 Discharge Piping Stresses and Support Loads

The calculated pipe stresses and support transferred loads indicate that the allowable stresses for the discharge piping are exceeded and the support elements are overloaded.

However, the thermal-hydraulic analysis which provides the input transient forces for the structural piping analysis was based on various conservative assumptions such as:

1. The pressurizer ramp rates were taken from the W generic bounding conditions for a RCP locked rotor event. The actual plant specific ramp rates, as shown in section 14.1.6 of the FSAR, are much less. Furthermore, the locked rotor event does not challenge the safety valves.

2. The only FSAR transient which challenges the safety valves is the loss of external electrical load accident coupled with a failure of the steam dump valves to open at BOL. The ramp rate for this event is less than the plant specific ramp rate for the "locked rotor" event.

4.0 Conclusions

The Indian Point 3 safety and relief valves were evaluated in accordance with the guidelines provided by EPRI Reference 1.

The plant conditons used to demonstrate the valve adequacy were later modified for the determination of piping and support adequacy. The changes and the impact on the initial valve justifications are summarized as follows:

The piping/support evaluation considered both sequential actuation of the PORV's and SV's and the actuation of the SV's only with the PORV's block valves closed. The determination of back pressure in evaluating valve performance was based on the sequential actuation of the PORV's and SV's only. This sequential actuation case yields the maximum back pressure. Therefore, the valve evaluation need not consider the PORV's blocked case.

4.1 Valves

The Indian Point 3 safety, relief and block valves, as installed, can be expected to perform their required function for the FSAR transient and OPS conditions with no resulting system overpressures in excess of the code allowables.

4.2 Piping

The Authority is currently in the process of evaluating various options to relieve the discharge piping overstresses and support overloads. We plan to implement any necessary modifications by the end of our cycle 4/5 refueling outage.

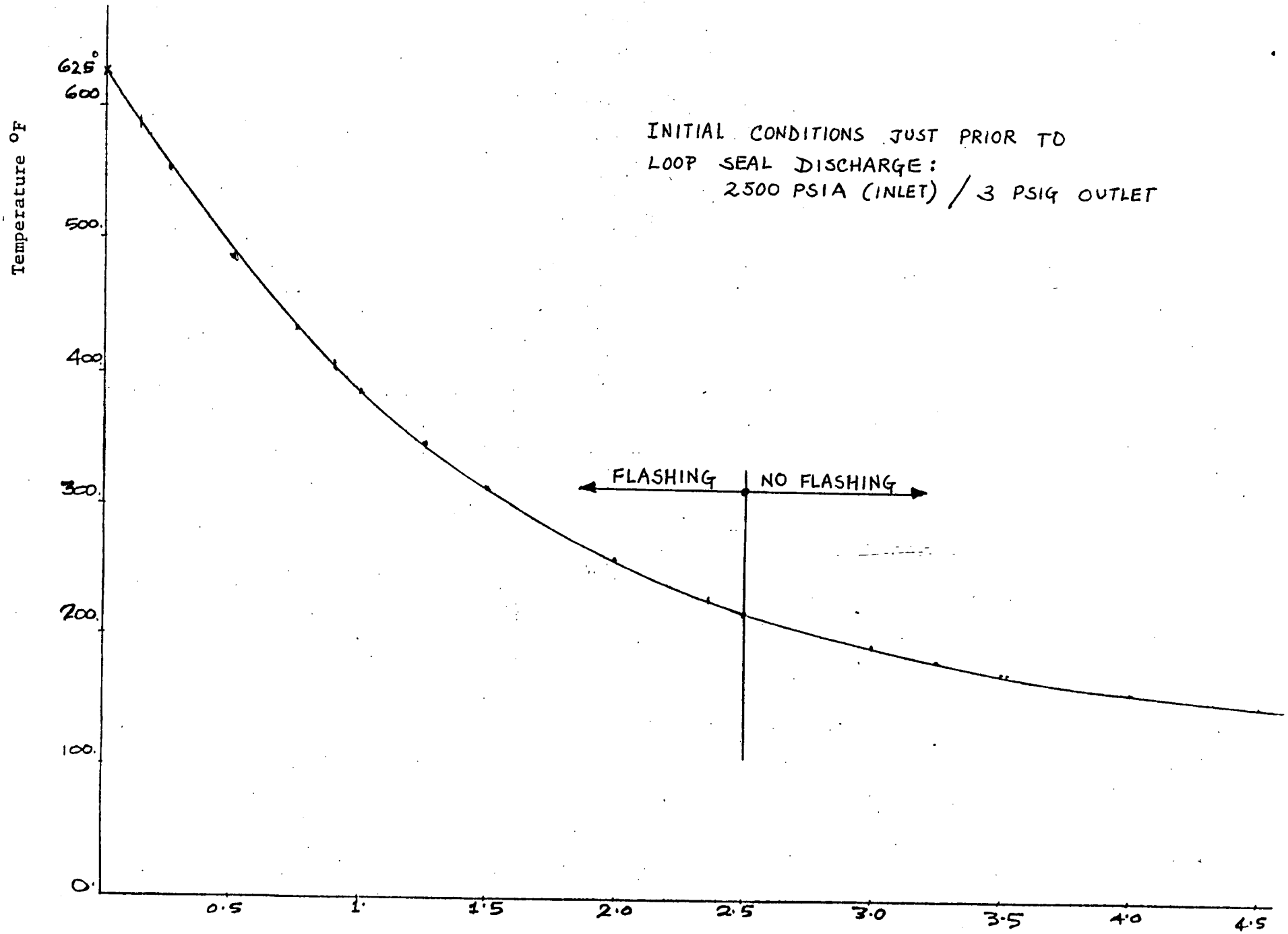

Fig. 7 Loopseal Temperature Distribution