John D. O'Toole
Vice President

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June 15, 1984

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

As requested in your letter of December 15, 1983, enclosed is the additional information regarding performance testing of Relief and Safety Valves (Item II.D.1 of NUREG-0737). This response is being made according to the scope and schedule agreed to in our March 8, 1984 telephone conversation with your staff and EG&G Idaho technical review personnel. We believe that the information provided in the enclosure is responsive to your requirements.

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

Very truly yours,

John D. O'Toole

cc: Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
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ENCLOSURE

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

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2

June, 1984

Question 1

The submittal identifies a steam discharge flow condition as a limiting event for the safety valves at this plant, claiming that liquid discharges through the safety valves are generally not observed. The submittal does not discuss, however, whether single failures after the initiating event were considered that could lead to a water or steam-water discharge through the safety valves. Present a discussion that shows how single failures that could lead to liquid flow were considered in selecting the limiting transient.

Response

The limiting Condition II transient that incurs safety valve actuation is the loss of external load event (FSAR 14.1.8). The analyses assumed an initial core power of 102 percent of rated with no direct reactor trip (on turbine trip). In addition, the pressurizer spray and power-operated relief valves were assumed inoperable. The combined effect from these assumptions produced the greatest (fastest) reactor coolant pressurization rate.

As the peak pressure is observed within a few seconds of transient initiation, single failures within the engineered safeguards systems would have little, or no effect, on the pressurization rate or peak pressure observed.

Question 2

The Westinghouse valve inlet fluid conditions report stated that liquid discharge through both the safety and relief valve is predicted for an FSAR feedline break event. The Westinghouse report gave expected peak pressure and pressurization rates for some plants having a FSAR feedline break analysis. Indian Point 2 was not included in this list of plants having such a FSAR analysis. Nor does the Indian Point 2 plant specific submittal address the FSAR feedline break event. NUREG-0737, however, requires analysis of accidents and occurrences referenced in Regulatory Guide 1.70, Revision 2, and one of the accidents so required is the feedline break. Provide fluid pressure and pressurization rate, fluid temperature, valve flow rate, and the time duration for the feedline break event. Provide assurance that the valves

passed water for a sufficient duration during the tests to cover this event and furnish an analysis which demonstrates safety of the safety/relief valve system for the feedline break event.

Response

The feedline break accident is not part of the Indian Point Unit 2 licensing basis. Nuclear plants such as Indian Point Unit 2, that were licensed prior to issuance of Regulatory Guide 1.70, Rev. 1, were not required to consider the feedline break accident as part of their design basis.

Question 3

Overpressure transients cause the pressurizer sprays to activate which adds moisture to the steam volume. When the safety or relief valves open, they would then pass a steam-water mixture. Show whether this effect was considered in selecting the transient that produces maximum loading on the system.

Response

The effect of a steam-water mixture was considered in the analysis performed for the relief valve piping. However, the limiting cases for piping loads are water solid discharge and nitrogen followed by water (Case B and Case C of the September 15, 1982 submittal) and not steam or steam with entrained water discharge; therefore the effect of water entrainment has been determined to be minimal.

The safety valves will not be challenged when the pressurizer sprays and power operated relief valves function as designed. Expected fluid conditions at the inlet of the safety valves were utilized in the evaluation. These conditions were discussed in the September 15, 1982 submittal.

Question 4

Since Indian Point 2 does not have a high head safety injection system, an extended high pressure injection event was not considered. A failure of the pressurizer level control system, however, could cause an extended high pressure injection from the charging pumps that may lead to water flow through the safety valve. This should be considered a high pressure injection event. Provide details that demonstrate safety of the system for such an event.

Response

As noted Indian Point Unit No. 2 does not have safety grade charging/SI-- thus extended high pressure safety injection cannot lead to water flow through the safety valves (i.e., the IP2 SI shutoff head is about 1500 psi).

The Indian Point Unit 2 CVCS includes three non-safety grade positive displacement charging pumps. Their individual capability is 98 gpm, of which 12 gpm is removed from the RCS by controlled reactor coolant pump (RCP) seal leakage. During normal operations only one pump is operating. An additional pump could be started by operator action but would be in manual mode. The potential for failure of the pressurizer level control which would result in the complete filling of the reactor coolant system (thus the potential of liquid relief through the safety valves) has been evaluated.

Failure of the pressurizer level control is a low probability occurrence. The system is extremely reliable; however, assuming it were to fail the only conditions under which the failure would lead to extended injection are conditions when the pressurizer level automatic control mode is used. The current reactor operations is 23.1% level at no load to 25.2% level at full load (the original plant T_{avg} program, no longer used, resulted in 23.1 to 46.8% level over no load to full load). In addition, in automatic control as noted above, only one charging pump would be under automatic pressurizer level control. The resultant time to fill the pressurizer is very long and credit can be taken for operator action to terminate the event, see table below.

Przr. Level (%)	Time to fill Przr. (min)	(Note
23.1	70.4	(no load level)
25.2	68.5	(full power level, T_{AVG}=549°F)
46.8	49.6	(full power level, T_{AVG}=569.5°F no longer used)
70	29.3	(high level alarm)

Based on the long time available, credit is taken for operator action to terminate the excessive charging flow due to the postulated failure of the pressurizer level control prior to any possibility of filling the pressurizer and passing water through the safety valves (i.e., taking no credit for the relief valves).

Question 5

The Westinghouse valve inlet fluid conditions report lists inlet fluid conditions for reference plants, each of which represents a group of Westinghouse plants. The loss-of-load event and locked rotor event were determined to be the worst case steam flow FSAR transients for maximizing pressure and pressurization rates, respectively. Provide the expected peak pressures and maximum pressurization rates for these two events and compare with those of the reference plant to assure that the expected conditions are enveloped.

Response

The following table compares the expected peak pressures and maximum pressure rates between Indian Point Unit 2 and the reference plant analysis.

Parameter	Limiting Event	IPP	Reference Plant
Peak Pressure (PSIA)	Loss of Load	2550	2555
Maximum Pressure Rate (PSI/Sec)	Locked Rotor (1/4)	110	114

As noted above, the Indian Point Unit 2 parameters are enveloped by the reference plant analysis.

Question 6

In valve operability discussions on low temperature overpressurization transients, the submittal only identifies one pressure and one temperature for a water discharge transient. The relief valves are, however, expected to operate over a range of fluid conditions for cold overpressurization protection. The submittal does later describe three cold overpressurization events that were included in thermal hydraulic analyses that evidently bound all cold overpressurization conditions. To assure that the relief valves operate under all cold overpressurization events, though, discuss the range of fluid conditions expected for the various types of fluid discharge (including the nitrogen bubble case) and identify the test data that demonstrate operability over the range of condition. Confirm that the high pressure steam tests demonstrate valve operability for the low pressure steam case for both opening and closing of the relief valve.

Response

EPRI test conditions were chosen based on expected inlet conditions for the subject valves. Tests were limited but designed to confirm operability over the full range of expected inlet conditions, including cold overpressure events. Results of these tests appear in EPRI Report NP-2670-LD, Volume 8,

dated October 1982. Steam, steam to saturated water, nitrogen to water and water flow tests were conducted, the test parameters and results of which appear in Tables VIII-2 and VIII-3 of the above EPRI report.

Although the steam test was only conducted at the higher pressure, our analysis concludes that satisfactory operation would result at the lower pressures as well. Water tests were successfully conducted at low pressure (675 psia) and low temperature (105 - 442°F), which demonstrates that the valve actuator can provide sufficient force to open and close the valve under the low pressure conditions.

The nitrogen to water test confirms operation of the valve for nitrogen and nitrogen to water discharge for pressures to 1500 psia and temperatures to 290°F. It is noted that this temperature is expected to shift to approximately 350°F over the life of the plant. Although nitrogen to water tests were not conducted above 290°F, steam to water tests were conducted successfully at 2532 psi and 670°F. The successful results of this high temperature steam to water transition test, coupled with the successful nitrogen transition test at 290°F, are considered to bound the higher temperature (350°F) nitrogen to water event.

Saturated water to subcooled water, and steam to saturated water to subcooled water, are additional valve inlet conditions which may occur during cold overpressurization events. Valve operability for these conditions was not specifically demonstrated by the EPRI tests, but saturated steam, subcooled water, and saturated water tests were conducted separately by EPRI. On this basis it is concluded that satisfactory valve performance would result if these conditions were to be tested in sequence.

Question 7

Since arguments presented in the submittal do not preclude liquid flow through the EMOV's, liquid flows through these valves may be possible. The EPRI/Marshall block valve test program did not include tests on the block valves with fluid media other than steam. While the Westinghouse test program did include tests with water, the information presented in Reference 2 of the submittal did not give specific test results. Thus, provide information that conclusively demonstrates that liquid flow through the EMOV's will not occur, or provide specific test results that show the EMOV's operate acceptably on liquid flow.

Response

In a June 1, 1982 letter from R. C. Youngdahl to Mr. H. Denton, several block valve test submittals were made which included an explanation as to why block valve tests beyond the Marshall tests were not considered necessary, as well as an EPRI summary report covering Westinghouse gate valve closure testing. The Westinghouse report, transmitted to the NRC by the Youngdahl submittal, also includes a section on friction testing of stellited seating parts.

Friction testing done by Westinghouse on stellite test specimens (Note the Velan valve has stellite seats) indicates that over the initial 200 cycles of testing, water test specimen friction factors increased from as low as 0.12 until a level of 0.4 to 0.75 is reached. With 550°F steam, the friction factor starts in the 0.5 to 0.6 range (higher than the water tests) and drops to approximately 0.35 over the 200 cycle range. Considering the 21 test cycles completed at Marshall Steam Station, and in view of the above frictional data, the thrust required to cycle the valve during the steam tests would be similar to that if the test medium were water.

Question 8

Results from the EPRI tests on the Crosby safety valves indicate that the test blowdowns exceeded the 5 percent value given in the valve specifications. These increased blowdowns occurred with the valve vendor's recommended ring

settings and with ring settings that were adjusted for a lower blowdown. If the expected plant blowdowns also exceed 5 percent (see related Question 9), the higher blowdowns could cause a rise in pressurizer water level such that water may reach the safety valve inlet line and result in a steam-water flow situation. Also the pressure might be sufficiently decreased that adequate cooling might not be achieved for decay heat removal. Discuss these consequences of higher blowdowns if increased blowdowns are expected.

Response

Blowdowns in excess of 5 percent have been analyzed in conjunction with the Westinghouse Owners Group (WOG) program on safety valves. The analyses utilized valve blowdowns in excess of 10 percent for a 4-loop plant having coolant expansion parameters similar to Indian Point Unit 2 (i.e., reactor coolant volume, core power, pressurizer steam space volume, etc.). The results from these analyses showed no adverse effects on plant safety. The peak pressurizer water level calculated remained below the pressurizer inlet piping to the safety and relief valves. Discussion of the WOG program is contained in the Westinghouse report WCAP-10105, "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program," dated June 1982.

While tests with blowdowns in excess of 5 percent were conducted by EPRI, such blowdowns are not considered necessary for the Indian Point Unit 2 safety valves to ensure adequate performance. Of the 96 steam tests conducted by EPRI, only four resulted in valve chatter occurring during steam flow (an additional three tests experienced flutter which did not increase to chatter). Three of the four chatter cases occurred on closing; these may be affected by blowdown. These three cases involved blowdown values of 5.6 percent (Test 508, Crosby 3K6), 8.0 percent (Test 920, Crosby 6M6), and 9.4 percent (Test 1005, Dresser 31739A). For each of these, the valve performed stably (no chatter) at blowdown values less than that observed for the test resulting in chatter (WCAP-10105, p. 3-11).

See also the response to question 9 below.

Question 9

The submittal does not address the expected blowdown and corresponding valve performance for the Indian Point 2 Crosby 4M6 safety valves at the plant ring settings. The submittal does state that the valve manufacturer (Crosby) and the NSSS supplier (Westinghouse) were reviewing these ring settings. Report the ring settings to be used at the plant. Explain how the expected values for backpressures and blowdown corresponding to these ring settings were extrapolated or calculated from test data, and identify the values for backpressure and blowdown so determined. If the blowdown exceeds the design blowdown of 5 percent, discuss the consequences of the higher blowdown relative to valve behavior.

Response

Ring settings for the Indian Point safety valves were developed by Crosby during production testing of the valves. Blowdowns measured for each valve were equal to or less than 5 percent. Crosby lists in the valve test reports final ring settings as follows:

Valve Ident. No.	Ring Settings	
	N.R.	G.R.
RV-468	-5	-275
RV-466	-5	-250
RV-464	-5	-250

Please note the ring settings indicated above were measured by Crosby from the highest "locked" position as noted in Crosby procedures and in the EPRI reports definition of key terms and parameters and not as measured from the "level position" as reported by EPRI. Also, dynamic backpressures measured during the tests are approximately 14 percent of the setpoint. It is noted, however, that the Crosby valves employ a balanced bellows design and are relatively insensitive to backpressure effects as noted by EPRI on page 5-5 of Report "EPRI NP-2770-LD, Volume 5."

Question 10

Results from tests on the Crosby 3K6 and 6N8 safety valves were used in the plant specific submittal to bound the Crosby 4M6 valves of Indian Point 2. However, flow rate data were not taken for the Crosby 6N8 safety valve. Additionally, a flow rate determination depends on the specific ring settings used at the plant. Thus, provide a demonstration that the plant safety valves will pass their rated flow.

Response

The 4M6 Crosby safety valve was not tested as part of the EPRI test program. Results of the 3K6 and 6N8 EPRI tests are to be used to bound the 4M6 valve. Furthermore, flow measurements were made only during the 3K6 EPRI valve tests. A review of EPRI Tables 4-3 and 4-4 in Volume 5 of EPRI Report NP-27770-LD reveal that for steam tests of the 3K6 valve where blowdown was measured to be less than 10 percent, flow rates of 119-122 percent of rated flow (at 3 percent accumulation) were reported. (Please note rated flow for safety valves is to be measured at 3 percent accumulation).

The EPRI tables indicate that lower than rated flows occurred at blowdowns greater than 15 percent. Since the Crosby production test results indicate ring settings that result in less than 5 percent blowdown, rated flow can be expected for the Indian Point 2 valves.

Question 11

The submittal states that thermal expansion of the pressurizer and piping was considered in calculating loads on the safety and relief valve piping and on the pressurizer nozzles. This thermal expansion would also, however, induce loading on the inlet flange of a safety or relief valve at the time the valve is required to lift. Evaluate the effect that this loading may have on valve operability.

Response

All pertinent combinations of thermal conditions were evaluated in the design analysis. The assumption that a power operated relief valve would open prior to the opening of any safety valve was made. The maximum force and moment in each direction were determined for each valve type.

The mechanical loads utilized were due to the absolute addition of deadweight, thrust and seismic loadings. Again, the maximum force and moment in each direction were determined for each valve type.

The primary stress evaluation was conducted consistent with the guidelines of Section III of the 1967 ASME Boiler and Pressure Vessel Code. That is (a) the average primary membrane stress intensity must be below S_m at temperature and (b) the primary membrane plus bending stress intensity must be below 1.5 S_m at temperature.

Each maximum component of the thermal loadings was absolutely added to each mechanical loading component. Section III of the 1967 ASME Boiler and Pressure Vessel Code requires that the sum of primary and secondary stresses be less than 3 S_m. The design evaluation performed substantiated this.

In summary, the stresses in the safety valve flanges and relief valve nozzles have been shown to meet code requirements.

Operability of the Crosby safety valves was demonstrated during the EPRI test program. Each valve type opened and closed on demand with no evidence of sticking or malfunction. As the test conditions bound installed conditions, no adverse effect on IPP valves is anticipated.

Question 12

The PORV's of Indian Point 2 were qualified on the basis of similarity to an EPRI test valve. The Indian Point 2 valve, however, has a different valve

cage from that of the EPRI test PORV. The IP2 valve is a Copes-Vulcan 3 in. globe valve with 316 SS stellite plug and Haynes #25 cage while the EPRI test valve was a similar Copes-Vulcan valve with a 316 SS stellite plug and 17-4 PH cage. This difference in cage material is not addressed in the submittal or associated documents. Provide an assessment on whether the plant combination of valve and cage is satisfactory.

Response

Please refer to Table II-4 (Pg. 20) of the EPRI "PWR Safety and Relief Valve Test Program Valve Selection/Justification Report" dated March 1982. This Table identifies the selected test valve, a 3-inch CVI valve with 316 SS plug and 17-4PH cage, represents the 3-inch CVI valve with 316 SS plug and Haynes #25 cage.

The Haynes #25 Alloy used for the cage in the Indian Point Unit 2 valve is a stellite material. It is tougher than the 17-4 pH stainless steel used for the cage in the test valve. The Haynes #25 cage when used in conjunction with the stellited plug provides superior galling resistance due the the high material hardness (approx. 38 Rc) of the guiding surfaces. The Indian Point Unit 2 plant combination is the superior selection and perfectly satisfactory for the service.

Question 13

The adequacy of the thermal hydraulic analysis could not be verified since sufficient detail is not presented in the submittal. Identify parameters such as time-step, valve flow area, pressure ramp rate, choked flow junction, and node spacing and discuss the rationale for their selection. Provide more detailed information on the ITCHVALVE program so that methodology for generating fluid parameters can be better evaluated. To allow for a complete evaluation of the thermal hydraulic analysis, provide a computer printout

containing input and output from a key problem such as the loss-of-load accident case for opening of the safety valves and the N2 bubble case (Case C) for opening of the PORVs.

Response

The adequacy of the thermal-hydraulic analyses can be verified by the comparison of analytical and test results for thermal-hydraulic loadings in safety valve discharge piping for EPRI tests 908 and 917 presented in "the submittal". In that evaluation, node spacing and time-step size were selected on the basis of stable solutions of the characteristic equations and matching of test data. The safety valve full open flow area of 0.022 ft^2 was used in the model. This area is slightly smaller than the Crosby M-orifice area of 0.025 ft^2 for the tested valve, but results in a good analytical match of the tested fully open valve flow rate. Appropriate water temperatures were used. All pertinent data, including friction factors, loss factors and flow areas were based upon representative calculations and the system layout.

The Indian Point Unit 2 Plant Specific thermal-hydraulic analysis was conducted based upon the same approach as used for the comparison to test data. Node spacing and time-step size were utilized consistent with values utilized in the comparison. Valve flow areas were selected based upon actual valve data with appropriate margins applied to account for flow rate uncertainties. Analyses performed assumed a 100 percent linear valve opening time with the pressurizer conditions held at initial values. All pertinent data, including friction factors, loss factors and flow areas were based upon representative calculations and the system layout. Choked flow is checked internally and automatically every time-step to ensure the proper formulation is applied at every flow path.

Because of the proprietary nature and extensive detail involved in the computer program, input and output data and more detailed methodology information are not supplied at this time. This information is available for additional NRC review if required.

Question 14

There is no discussion as to whether multiple valve actuations were considered in the thermal hydraulic analysis. The maximum loading on the piping typically occurs under a multiple valve actuation condition during which the valves open in sequence. The experience of EG/G Idaho indicates that the maximum loading occurs when the sequence of opening is such that the initial pressure waves from opening of the safety valves reach the common header downstream simultaneously. Additionally, if a PORV is discharging with flow in the common header, the piping loads could be significantly affected. Provide justification that sequential opening of the valves under multiple valve actuation conditions was considered.

Response

Two valve opening cases with steam discharge were considered in the evaluation: the three safety valves opening simultaneously and discharging without PORV flow and the two PORVs opening simultaneously and discharging without safety valve flow. Additionally water discharge cases through the PORVs with and without a nitrogen bubble were evaluated with no flow through the safety valves. No common header initial flow was assumed.

The three safety valves are identical and have the same set pressure (± 1 percent). It was, therefore, assumed for the analysis that all three safety valves open simultaneously without PORV flow. Because of similarity, the two PORVs were also assumed to open simultaneously without safety valve flow.

Maximum common header (area of piping common to both safety and relief valve discharge piping) forces theoretically could be expected when valve sequencing is such that the initial pressure waves from valve opening reach a common downstream junction simultaneously. Based upon engineering judgment:

1. The simultaneous opening of the safety valves results in practically simultaneous peak loads at the safety valves common branch point. The peak forces occur within approximately .02 seconds of each other. As a result, no significant impact in the common header region, due to safety

valve discharge, is expected, if the valve sequencing is adjusted such that the peaks of the initial pressure waves reach a common downstream header point simultaneously.

2. The total lengths of effective piping between each valve outlet and the common junction point are not exactly the same. The likelihood of the valve phasing being such to compensate for the different lengths is very small; therefore, the peaks of the initial pressure waves from valve opening, either safety or relief, would not reach a common downstream junction at exactly the same time.

Question 15

The momentum equation used for calculating the fluid forces for the structural analysis contains an acceleration term. This term commonly introduces spurious spikes into the calculated fluid forces for water discharge transients. These spikes can be avoided by using a large time step in the force calculations, but this may result in losing the peak force values. Thus, discuss the accuracy of the force calculation results for water discharge transients, such as the cold overpressurization event.

Response

A finite time-step that resulted in stable results was utilized for both the data comparison and the plant specific thermal-hydraulic evaluation. A discussion of the method of characteristics solution technique is presented in the following articles:

1. A. C. Spencer and S. Nakamura, "Implicit Characteristic Method for One-Dimensional Fluid Flow", ANS Transaction, Volume 17, P. 247, November, 1973.

2. S. Nakamura, M. A. Berger and A. C. Spencer, "Implicit Characteristic Method for One-Dimensional Fluid Flow", Proceedings of the Conference on Computational Methods in Nuclear Engineering, Conf. -75040, National Technical Information Service, Springfield, VA. 1975.

A. C. Spencer is a permanent Westinghouse employee who was and is directly involved in the development of the ITCHVALVE Computer Program.

Modifying the time-steps could change the results; however, the time-step was selected on the basis of stable results and showed good comparison to test results.

Question 16

The submittal identifies the initial conditions for the safety and relief valve thermal-hydraulic analyses. However, it does not describe the method used for treating valve resistance in the analyses and does not report flow rates corresponding to the resistances used. Because the ASME Code requires derating of the safety valves to 90 percent of actual flow capacity, the safety valve analysis should be based on a flow rating equal to 111 percent of the flow rate stamped on the valve, unless another flow rate can be justified. Provide further information explaining how derating of the safety valves was handled and describing methods used to establish flow rates for the safety valves and PORVs in the thermal hydraulic analyses.

Response

Safety and relief valves are modeled as two-way junctions. The pressure drop across the valve provided the system is sub-cooled is given by:

$$\Delta P = \rho C_D v^2$$

Where ΔP = pressure drop

C_D = discharge coefficient = f(Cv)

ρ = fluid density

v = velocity through the valve

In the case of choking at the valve, the velocity at the valve orifice area is set at the sonic velocity. Upstream and downstream boundary conditions are iteratively set to conserve mass and energy. Choked flow is internally checked to ensure the proper formulation is applied.

The expected steam flow rate through the Copes-Vulcan PORVs, the valves utilized at Indian Point Unit No. 2, is 210,000 lb/hr at approximately 2350 psia. Values of 236,000 and 232,000 lb/hr, respectively at 2140 and 2120 psia were observed in EPRI/Marshall Tests (EPRI report NP-2144-LD, "EPRI-Marshall Power-Operated Relief Valve Interim Test Data Report"), February 1982. To account for all uncertainties and tolerances in the valve flow rate, the valve flow area was adjusted accordingly. The minimum analytically calculated steam flow of each of the two PORVs is greater than 271,000 lb/hr (full open valves). This is a flow of 129 percent of rated. The analysis assumed a 100 percent linear PORV valve opening in 0.40 seconds. Full open times, based upon tests, averaged 1.48 seconds with a minimum value of 1.40 seconds for opening on steam.

The expected water flow rate through the PORVs is 1.0×10^6 lb/hr at 1600 psi and 150°F. The minimum analytically calculated water flow of each of the two PORVs is greater than 1.2×10^6 lb/hr. This is based upon the valves being fully open.

The nominal steam flow rating for the Crosby Safety Valves at 2500 psia +3 percent accumulation is 420,000 lb/hr. A value of 460,000 lb/hr was observed in both EPRI tests 908 and 917. This flow, however, was based upon upstream conditions of 2700 psia. As with the PORVs, to ensure that adequate margin existed in the valve flow rate to account for all uncertainties and tolerances, the analytically calculated steam flow was checked prior to finalizing this phase of the overall effort. The flow used in the analysis (467,000 lb/hr) was 111 percent of rated. The safety valves were presumed to open fully in 0.020 seconds. This is based upon an effective linear opening time.

Question 17

The inlet piping pressure drops for the Crosby 3K6 and Crosby 6N8 EPRI test valves were compared to the calculated Indian Point 2 Crosby 4M6 inlet piping pressure drop. It is not clear though whether the Crosby 3K6 and 6N8 pressure drops were measured or were calculated according to the EPRI/C-E test configurations and the equations of Appendix I of the plant specific

submittal, or were determined by some other means. Explain the method for determining these pressure drops. If they were measured, make reference to the appropriate test data.

Response

The Crosby 3K6 and 6N8 pressure drops were obtained from reference (8) of the submittal, "EPRI PWR Safety and Relief Valve Test Program Guide for Application of Valve Test Program Results to Plant - Specific Evaluations" Interim Report, March 1982. EPRI states in reference (8) that these pressure drops represent the calculated transient upstream pressure drop.

Question 18

The submittal does not provide some of the details needed to make an assessment of the structural analysis. Describe methods used to model supports and give the spacing used between lumped masses. Identify, describe, and reference the computer program used to perform the structural analysis. To allow for a complete evaluation of the structural analysis provide a diagram of the structural model together with the computer printout of input and output from the loss-of-load accident case for opening of the safety valves (unless analysis shows another accident case to be more severe for pipe loading), and the N2 bubble case (Case C) for opening of the PORVs.

Response

In the design analyses, the three-dimensional piping system model which includes the effect of supports, valves and equipment was represented by an ordered set of data which numerically describes the physical system. All piping and piping components were assumed to behave in a linear elastic manner.

Model for Dynamic Seismic Analysis

The piping system was modeled as a lumped multi-mass system. The model boundaries were at the pressurizer relief tank nozzle and the pressurizer vessel base support. Rigid and snubber restraints were included in the analytical model as members with equivalent stiffness matrices.

Model for Static Analysis

The static analysis for the original design were performed with the aid of the MEL-ZIP Computer Code. The model boundaries were at the pressurizer safety and relief valve nozzles and at the pressurizer relief tank nozzle. During 1978, modifications were performed to the pressurizer safety relief line and revised analyses were performed using the ADLPIPE Computer Code.

The boundaries for the mathematical model used in the revised static analysis were:

- The 4" diameter pressurizer nozzle for Line No. 70.
- The junction of the 6" diameter line to the 12" diameter header.

The second boundary was assumed to be an anchor with appropriate boundary displacements. The static model was utilized for deadweight, thermal and transient analysis.

Figures 18-1 and 18-2, the stress isometrics, illustrate the structural model.

Load Analysis

The following describes the revised analysis performed for each load case, and the analytical assumptions:

P - Internal Pressure: Internal pressure stresses were calculated using the B31.1.0 - 1967 Code and the applicable design pressure.

D - Deadweight: The deadweight analysis was performed with the aid of the ADLPIPE Computer Code (ADLPIPE, Static and Dynamic Pipe Design and Stress Analysis Computer Code, CDC User Information Manual Publication No. 86602800, Revision E, dated August 1, 1977) using uniform piping weight for empty, uninsulated or insulated pipe as applicable. Concentrated weight of the valves was included in the analysis.

Deadweight analysis was performed only for the normal plant operating condition. Deadweight loading during the heat-up mode of operation was not considered since the effect of the additional water weight was found to be negligible.

OBE - Seismic: The dynamic seismic analysis was performed with the aid of the ADLPIPE Computer Code in accordance with the applicable seismic criteria. Two computer runs were performed. The first addressed the north - south (or X direction) Amplified Response Spectra (ARS) and vertical (or Y direction) ARS. The second addressed the east-west (or Z direction) ARS and the vertical (or Y direction) ARS.

Modal responses in each run were combined using the square-root of the sum of the squares method.

The largest values from the two runs were compared with the applicable limits.

T - Thermal Expansion and Anchor Motion: The analysis was performed with the aid of the ADLPIPE Computer Code. Eight thermal cases were analyzed to comply with the different modes of operation of the safety and relief valves.

The boundary displacement conditions (translational and rotational motions) at the pressurizer vessel nozzle and the connection of the 6" diameter pipe with the 12" diameter header were used.

The ambient temperature was assumed to be 120°F. Applicable coefficients of thermal expansion and modulus of elasticity were utilized.

TR - Transient: The static analysis approach was discussed in the September 15, 1982 submittal.

Input and output data are not supplied at this time. However, if so desired, this information could be reviewed at a later date.

Question 19

The piping structural response to fluid transient loading was solved for by performing a static analysis. In this analysis the hydraulic loads were amplified by a dynamic load factor (DLF) of 2 to determine forces applied to

the static model. Due to the oscillatory nature of some of the force-time histories, as can be seen in Figure 6.1 of the September 1982 submittal, dynamic amplification in piping response may, however, exceed 2. For this type of loading, the dynamic amplification in piping response depends on the frequency content of the force-time histories, the natural frequencies of the structure, and the damping in the structure. Justify that a DLF of 2 will result in conservative stresses in the piping and supports for the fluid transient loads.

Response

The piping structural response to fluid transient loadings was performed statically. In the design analyses of the relief line, the hydraulic forces were amplified by an appropriate dynamic load factor. The value was calculated by using the overall fundamental frequency of the piping system and the range of frequencies associated with the transients under investigation. The characteristics of the transient loadings discussed in the September 15, 1982 submittal are consistent with those from the design analysis. For the safety line analysis, the forces were amplified by a Dynamic Load Factor (DLF) = (2.0). Considering the frequency content of the forcing functions discussed in the September 1982 submittal and the overall system frequency a DLF = 2.0 is a conservative value to utilize in the static analysis.

Question 20

The program FORFUN was used to calculate an unbalanced wave force for each bounded segment of straight piping. These wave forces were applied to the model in the structural analysis. Application of only a wave force on a pipe segment, however, ignores the axial extension that is caused by opposing blowdown forces at the ends of the pipe segment. This axial extension induces bending moments on pipe segments adjacent to the bounded segment in question. Explain how this was accounted for in the analysis.

Response

For all cases evaluated, program FORFUN was utilized to calculate the unbalanced wave forces for each bounded segment of straight piping. This

evaluation was conducted subsequent to the release of the EPRI Test Data. A comparison of the loads was found to be in good agreement with the design analysis loads.

The axial extension from the balancing forces on each end of the structural segment was considered in the FORFUN evaluation. However, this effect was determined to be negligible relative to the net unbalanced forces. Referring to structural analyses comparisons to test results for Tests 908 and 917, maximum support and pipe loads compared well with test results. Good comparisons of the maximum displacement values downstream of the safety valve were also seen.

Question 21

The submittal lists load combinations that were developed to evaluate the safety and relief valve piping. It is not clear whether the fluid transient loads included in these combinations pertain to a worst case transient such as a multiple safety valve actuation. If a worst case transient is not included, then a load combination corresponding to an emergency condition (per Westinghouse report WCAP-10105), which includes a multiple safety valve case, should be added for the piping upstream of the valves. In addition, a load combination that combines a safe shutdown earthquake with a worst case safety valve actuation condition (as a faulted condition in WCAP-10105) should be included for the upstream piping. Present an assessment of the piping system that includes these load combinations for the upstream piping.

Response

The fluid transient loads in the design analyses result from the worst case transients for both the safety and relief lines.

The relief valve inlet piping is essentially isolated from safety valve actuation transients, consequently the most severe loading case utilized was for relief valve actuation. The combination P + /DWT/ + /OBE/ + /TR/, where

P = Pressure stress

DWT = Deadweight stress

OBE = Seismic stress

TR = Stress from most severe thermal hydraulic transient, was shown to be less than upset limits. A load combination involving a safe shutdown earthquake (SSE) with a worst case valve actuation transient was not however explicitly considered in the evaluation. For Indian Point Unit No. 2 the SSE = (1.5) x OBE. It follows that since P + /DWT/ + /OBE/ + /TR/ is less than upset limits that P + /DWT/ + /SSE/ + /TR/ is less than faulted limits.

The multiple safety valve actuation case was considered in the evaluation of the safety valve inlet piping. Emergency limits were shown to be met for P + /DWT/ + /OBE/ + /TR/. As with the relief line, a load combination involving a safe shutdown earthquake with a worst case valve actuation transient was not explicitly included in the design evaluation. By inspection of the results (and by applying a factor 1.5 to the OBE stresses) the combination P + /DWT/ + /SSE/ + /TR/ meets faulted limits.

Question 22

The thermal hydraulic and structural analyses were performed prior to the EPRI tests. Results from a revised analysis performed after the EPRI tests indicate that the revised forces for two relief valve discharge cases [the water solid cold overpressurization case (Case B) and the nitrogen bubble case (Case C)] were greater than the original design forces. This was especially true for the nitrogen bubble case, where measured forces were significantly greater than the design forces. The submittal contends that this will probably not cause an overstress problem because the applied loads are of very short duration. It further states that the support adequacy for the nitrogen bubble case is being evaluated. Include in this evaluation consideration for Case B as well as Case C and evaluate stresses in the piping as well as the supports. Submit the results of the final piping and support adequacy analysis and describe modifications if any.

Response

No structural analyses subsequent to the release of EPRI test data were performed for Case B and Case C. A comparison of the new applied loadings to the design basis loadings was made. Results for Case B, water solid discharge

through the relief valves, show that the loads are comparable to the design loads. The support load capacity will not be exceeded and stress levels will be below the allowable values. The results for relief valve discharge of water with a nitrogen bubble (Case C) showed higher loads than the applied design basis loads. However, the loads in excess of the design loads were further evaluated and determined to be of short duration (less than 20 milliseconds). On this basis, it was concluded that detailed structural analysis is not necessary to verify support adequacy.