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John C. Brons
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August 15 , 1985
IPN-85- 42

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 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
Additional Information Related to NUREG-0737, Item II.D.1;
Pressurizer Safety and Relief Valves (PRZR. S/RVs) Testing

References:
1. NRC letter (S.A. Varga) to NYPA (J.C. Brons) dated June 6, 1985 - Request for IP-3 additional information TMI. Action NUREG - 0737 (II.D.1).
2. NYPA letter (J.P. Bayne) to NRC (S.A. Varga) dated December 15, 1982 - Plant specific evaluation of PRZR. S/RVs performance (IPN-82-80).
3. NYPA letter (J.P. Bayne) to NRC (S.A. Varga) dated September 30, 1983 - Results of plant specific evaluation of PRZR. S/RVs piping adequacy (IPN-83-82).

Dear Sir:

Based on the NRC's staff review of the EPRI PWR PRZR. S/RVs Test Program and IP-3 plant specific submittals (Refs. 2 and 3), you requested the Authority (via Ref. 1) to provide additional information and clarification necessary to complete your safety evaluation of the subject item for IP-3. Attachment I of this letter transmits our response to all the items addressed in the enclosure to your June 6, 1985 letter.

This letter also serves to confirm that in accordance with a telephone conversation held on July 15, 1985, between the NRC Project Manager for IP-3 and Authority personnel, plant modifications related to this item will be implemented prior to startup from the next (Cycle 5/6) refueling outage. These modifications will be aimed at relieving discharge piping overstresses and associated support transferred overloads to the appropriate code allowables. Extreme conditions in the PRZR. S/RVs discharge piping are exhibited during safety valve inlet loop seal water discharge following conditions created by applicable postulated design basis accidents described in the IP-3 FSAR.

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As previously stated in Ref. 3, the Authority had planned to perform modifications to the PRZR. S/RVs piping during the ongoing Cycle 4/5 refueling outage (i.e., elevating the safety valve inlet loop seals water temperature by adding piping insulation upstream of the valve, providing numerous new and modified supports and replacement of portions of the discharge piping). However, based on detailed system walkdowns performed after plant shutdown for the purpose of finalizing the installation details, it was concluded that the addition of these supports was impractical due to the creation of undersirable congestion in the discharge piping area. At that time, a redesign effort was undertaken to add insulation boxes upstream of the valves in order to reduce the number and loading capacities of the required additional supports. Subsequently, it has been determined that due to the high congestion in the area on top of the pressurizer, the insulation boxes mounting could not be accommodated without extensive changes to the existing pressurizer top head configuration and adjacent compartment wall.

It should be also noted, as discussed in our response contained in the enclosed Attachment I, that various overconservative assumptions and criteria (e.g., EPRI tests data and Westinghouse WCAP-10105 generic bounding conditions) were employed in the current IP-3 thermal-hydraulic and structural analyses used to establish the PRZR. S/RVs piping adequacy. Accordingly, further reevaluations are being considered to provide for a more realistic yet conservative analytical model reflecting plant specific parameters and IP-3 FSAR transient analyses. In addition, considering the low probability of PRZR. S/RVs actuation, the Authority has determined that there is no significant safety impact on the overall plant continued operation or to the present piping and support configuration contingent upon installation of necessary modifications required by analysis.

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
Senior Vice President
Nuclear Generation

cc: Resident Inspector's Office
Indian Point Unit 3
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ATTACHMENT I

RESPONSE TO NRC 6/6/85 REQUEST
FOR ADDITIONAL INFORMATION
REGARDING
NUREG-0737, ITEM II.D.1
PERFORMANCE TESTING OF PRESSURIZER
SAFETY AND RELIEF VALVES

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

QUESTIONS RELATED TO THE SELECTION OF TRANSIENTS AND VALVE INLET AND DOWNSTREAM CONDITIONS

1. 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. Since the safety valve inlet piping utilizes loop seals, it has been concluded that this condition has been enveloped by the water discharge case for the safety valves. The submittal did not identify if loop seals are used upstream of the PORVs. It was not clear in the submittal if the relief valve piping analysis included the relief opening on water at the expected overpressure and temperature conditions. The piping analysis discussion does not present a description or results of the PORV fluid transient analysis. Indian Point 3 should provide a discussion for one of the following: (1) was the steam-water discharge case considered, or (2) was a solid water discharge case considered, in establishment of maximum loads for the PORV discharge piping.

RESPONSE TO QUESTION 1:

The piping upstream of the IP-3 PORV's does not contain loop seals. The Authority has established the maximum loads for the PORV's discharge piping based on saturated steam discharge through the valves at their set pressure of 2350 psia (Ref. 1, Section 3.0 - Case 1).

The IP-3 FSAR transient analyses demonstrate that water will not be discharged through the PORV's at their design pressure setpoint. However, solid water may pass through the PORV's during cold overpressurization events which could occur during plant heatup or cooldown. The Overpressure Protection System (OPS) will open the PORV's automatically when the reactor coolant system temperature is below 300°F to prevent the pressure from exceeding the Appendix G heatup/cooldown curves. A piping analysis was performed prior to the installation and implementation of the OPS at IP-3. The results of this analysis showed that the loads generated under low temperature solid water discharge (OPS operation range) are significantly lower than those developed by saturated steam discharge through the PORV's at their design setpoint of 2350 psia. Based on the above considerations, only the saturated steam case was considered for the recently completed piping analysis.

2. NUREG-0737 II.D.1 requires that the transients of Regulatory Guide 1.70 Revision 2 be considered. The feedline break is included in these transients. The Westinghouse Valve Inlet Fluid Conditions Report stated that Indian Point 3 was not covered by the feedline break discussion and results section of that report. The submittal stated that the feedline break event is applicable to Indian Point 3.

Provide a discussion of the feedwater line break event and identify the expected peak pressure, pressurization rate, fluid temperature, valve flow rate, and time duration for the event. Assure that the fluid conditions were enveloped in the EPRI tests and that the time period of water relief in the EPRI test was as long as expected at the plant. Demonstrate operability of the safety valves and PORVs for this event and assure that the feedline break event was considered in analyses of the piping system.

RESPONSE TO QUESTION 2:

Liquid discharge is predicted by the Westinghouse WCAP-10105 generic bounding conditions for a feedline break accident. This transient is not analyzed for IP-3 as it is outside of the plant's licensing basis. The conclusions stated in the IP-3 FSAR for the loss of normal feedwater indicate that this does not adversely affect the core, Reactor Coolant System, or Main Steam System since it doesn't result in approach to DNB or overpressurization of primary coolant system, in fact, it does not result in water relief from the pressurizer relief or safety valves, nor does it result in uncovering the tube sheets of the steam generators being supplied with water.

3. The Westinghouse Valve Inlet Fluid Conditions Report identifies Indian Point 3 as one of the plants not being covered by the report with respect to the cold overpressurization event. Indian Point 3 states that the PORVs are used for cold overpressure events. Provide the cold overpressure transient conditions to the PORV and discuss how the inlet fluid conditions were determined.

RESPONSE TO QUESTION 3:

As described above in the response to question 1, the PORV's may be operated during heatup or cooldown through the Overpressure Protection System (OPS). The OPS will open the PORV automatically when the reactor coolant temperature is below 300°F.

The bounding reactor coolant system cold overpressurization events for IP-3 are those identified in the Westinghouse study entitled "Pressure Mitigating Systems Transient Analysis Results" dated July, 1977. These events are: 1) mass addition - inadvertant operation of a single safety injection pump without letdown, and 2) heat addition - start of a reactor coolant pump with the steam generator(s) at an elevated temperature. The OPS utilizes a variable PORV setpoint curve which follows the ASME Appendix G curve. Some typical values taken from the OPS curve are:

70°F	-	465 psia
100°F	-	485 psia
150°F	-	525 psia
200°F	-	615 psia
250°F	-	805 psia
300°F	-	1200 psia

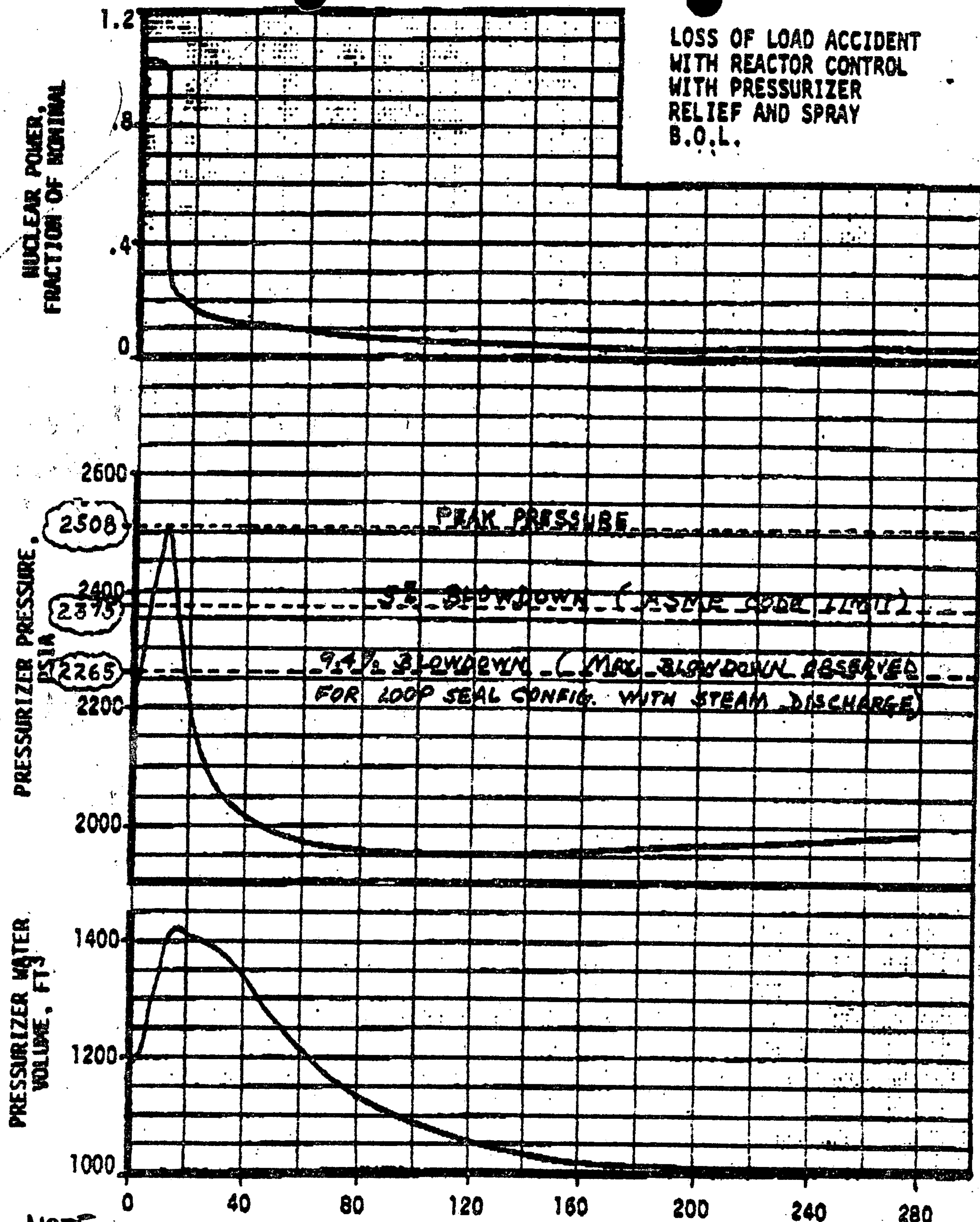
Additional details regarding the OPS operation are contained in the proposed Technical Specifications submitted in our letter dated July 1, 1985 (IPN-85-34).

4. Results from the EPRI tests on the Crosby safety valves indicate that the test blowdowns exceeded the ASME Code limit of 5% for both the "as installed" and "lowered" ring settings. If the blowdowns expected for Indian Point 3 also exceed 5%, 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 such that adequate cooling might not be achieved for decay heat removal. Discuss these consequences of higher blowdowns if increased blowdowns are expected.

RESPONSE TO QUESTION 4:

Since the EPRI safety valve tests are applicable to the IP-3 valves, there is no reason to expect blowdowns different from the test results. Therefore, blowdowns ranging from 5.1 to 9.4% can be expected for IP-3.

As stated previously in Reference 2 the only FSAR transient which challenges the safety valves is the "Loss of External Electrical Load". A review of this transient has been performed to evaluate the consequences of increased blowdown. It can be seen that the increased blowdown will not cause the pressurizer water level to reach the safety and relief valves inlet piping and will not reduce the pressurizer pressure below the minimum pressure established by the IP-3 FSAR analysis which is sufficient for adequate core cooling. This conclusion is based upon the FSAR transient response curves for the total loss of external load at beginning of life with zero moderator temperature coefficient assuming full credit for the pressurizer spray, PORV's and automatic control rod insertion but no credit for steam dump (see FIGURE 1). As seen from the pressurizer pressure transient curve, the pressure decreased below the minimum safety valve reseating pressure and, therefore, the increased blowdown is enveloped by the existing pressure curve. Furthermore, the water volume would actually decrease due to increased blowdown. The water volume in the pressurizer is reduced approximately 2% based on the additional time the safety valves are open and rated steam flow through the three safety valves.



NOTE
TOTAL PRESSURIZER VOLUME, 1,800 FT3

FIGURE Nº 1

INDIAN POINT 3 FSAR UPDATE

LOSS OF LOAD ACCIDENT WITH REACTOR CONTROL WITH PRESSURIZER RELIEF AND SPRAY B.O.L.

QUESTIONS RELATED TO VALVE OPERABILITY

5. The Indian Point 3 plant utilizes Crosby 6M6 safety valves. The same model safety valve was also tested by EPRI. To allow for a complete evaluation provide information that contains at least:

   (a) EPRI testing of the 6M6 was performed at various ring settings. If the plant current ring settings were not used in the EPRI tests, the results may not be directly applicable to the Indian Point 3 safety valves. Identify the Indian Point 3 safety valve ring settings and discuss the expected performance at these ring settings.

   (b) If the plant specific ring settings were not tested by EPRI, explain how the extrapolations or calculations were made to determine the expected values for flow capacity and blowdown for the plant-specific ring settings and the resulting backpressure.

   (c) Provide a discussion on the expected blowdown for the Indian Point 3 safety valves. If the blowdown is expected to exceed the ASME Code limit of 5%, discuss the effects of the higher blowdowns on safety valve operability.

   (d) Provide a discussion on the stability of the Indian Point 3 safety valves. A method recommended by the EPRI test program to demonstrate valve stability was to calculate the inlet piping pressure drop consisting of a frictional component and an acoustic wave component evaluated under steam flow conditions, and then compare to the pressure drops of the applicable EPRI tested safety valves.

RESPONSE TO QUESTION 5(a):

The IP-3 valve ring settings are 225/18 as referenced from the upper locked position. These ring settings correspond to the EPRI ring setting of (-71/-18). The EPRI ring settings are referenced from (0 notches) a position where the bottom of the upper ring is flush with the bottom of the disc ring. Since the IP-3 ring setting was used in the EPRI tests the results are directly applicable.

RESPONSE TO QUESTION 5(b):

Not applicable.

RESPONSE TO QUESTION 5(c):

Refer to response to question 4. As stated in Reference 2, the EPRI tests for the Crosby 6M6 safety valves demonstrated adequate performance of the valves with increased blowdown.

RESPONSE TO QUESTION 5(d):

Refer to applicable test as listed on attached Table 1 for the IP-3 inlet piping pressure drop.

The observed performance for Test 929 indicated the valve began opening at 2600 psia (4% above set point). The valve simmered for 0.83 seconds while the loop seal discharged then popped at 2717 psia (8% above set point). Total opening time, including loop seal discharge time, was 0.85 seconds. The ramp rate during the test was 319 psi/sec. and the loop seal contained 1.02 $ft.^3$ of water. The observed blowdown was 5.1%.

The required performance from the Westinghouse generic bounding conditions identifies the valve challenged during a loss of load event with a maximum pressurizer pressure of 2555 psia. The bounding ramp rate is 144 psi/sec. for a locked rotor event.

Comparing the observed performance to the required performance and considering differences in the plant-specific configuration, the Authority has determined that the IP-3 safety valves will perform their required function with no system overpressurization.

- o The generic bounding condition ramp rate is less than the tested rate (144 psia/sec. vs 319 psia/sec.) and the lower the ramp rate for a given time dealy, the lower the system pressure peak.

- o The plant-specific seal temperature is higher than the test temperature (130°F vs 90°F). The test data indicates increase seal temperature tend to result in opening pressures closer to the set point. Therefore, we would expect the IP-3 valves to open within its designed 3% accumulation.

- o The as-tested valve showed no signs of failure or excessive wear as a result of the loop seal discharge.

TABLE 1

Safety Valve Test Applicability Assessments

Crosby HB-BP-86 6M6 (Loop Seal Internals)

Ring Setting: (-71/-18;-77/-18)*

	Plant Specific	Test No. 929	Test No. 931a	Test No. 932	Test No. 1406	Test No. 1411	Test No. 1415	Test No. 1419
Test Type		Loop Seal	Loop Seal Transition	Water	Loop Seal	Steam	Loop Seal	Loop Seal
Back Pressure (psia)	587	710	725	650 #	250	245	255	245
Inlet Piping ΔP (PSI) Valve Opening	225	263	263	263	263	263	263	263
Valve Closing	135	181	181	181	181	181	181	181
Inlet Fluid Condition	Sat. Steam (2500 psia) Water (567-572 F)	Steam	Steam/Water	Water (515 F)	Steam	Steam	Steam	Steam
Observed Performance		Flutter and/ or chatter during loop seal discharge and the valve is stable on steam.	Flutter and/ or chatter during loop seal discharge and the valve is stable otherwise.	Chatter	Flutter and/ or chatter during loop seal dis- charge and the valve is stable on steam.	Stable	Stable	Valve chattered during loop seal discharge also during closure
Temp. at Valve Inlet (F)	120	90	117	463	147	Sat.	290	350
Applicability of Test		Yes	Yes	Yes	No	No	No	No
Remarks About Safety Valve Performance (predicted by the test) under plant specific conditions.		The valve is expected to chatter and/or flutter during loop sea dis- charge and be stable during steam discharge	Stable during transition	Test is applicable since the plant specific back pressure. inlet Δ P and subcooling are lower than test. The valve is expected to chatter on water discharge.	Test back pressure is too small to derive any new conclu- sions.	Test back pressure is too small.	Test back pressure is too small. Hot loop seal shows a stabili- zing effect.	Test back pressure is too small. Valve chattered inspite of hot loop seal.

* For tests 929, 931a, and 932, the ring settings were - 71 (relative to bottom of disc ring) for the upper ring and -18 for the lower ring. For tests 1406, 1411, 1415, and 1419, the ring settings were -77 (relative to bottom of disc ring) for the upper ring and -18 for the lower ring. The change in ring settings from -71 to -77 is expected to result in a slight increase in valve stability; however, all tests shown above are still considered "reference tests."

\# Same back pressure orifice was used as in Test 929.

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 TO QUESTION 6:

The electrical components associated with the operation of the PORV's, which are exposed to harsh environments as a result of design basis accidents, were addressed and determined qualified as indicated in our EQ submittal dated December 27, 1984 and the response to "Short Term Requirements of TMI Lessons Learned", Section 2.1.1, dated January 8 1980. Also, the control circuitry for the PORV block valves, including the motor operators, are environmentally qualified.

7. According to the Westinghouse Valve Inlet Fluid Conditions Report for the cold overpressure transients in some Westinghouse plants, the PORVs are expected to operate over a range of steam, steam-water, and water conditions because of the potential presence of a steam bubble in the pressurizer and water solid operations. For the range of fluid conditions expected for cold overpressure events at Indian Point 3, identify the test data that demonstrates operability for these cases. Since no low pressure steam tests were performed for the PORVs, confirm that the high pressure steam tests demonstrate operability for the low pressure steam case for both the opening and closing of the PORVs.

RESPONSE TO QUESTION 7:

The observed valve performance (EPRI Marshall and Wyle Phase III tests) indicated that the relief valve opened and closed on demand for a full range of inlet pressures. These inlet pressures envelope both the Westinghouse generic bounding conditions, the IP-3 FSAR and OPS parameters for the plant-specific applications.

Specifically the EPRI Wyle Phase III tests 76-CV-316-2W and 74-CV-316-5W approximate the upper and lower bound OPS set points.

	Test 76-CV-316-2W	Plant upper bound
Pressure psia	2535	1200
Temperature ^{O}F	647	300

	Test 74-CV-316-5W	Plant lower bound
Pressure psia	675	465
Temperature ^{O}F	105	70

In addition to solid water operation the PORVs may operate to relief low pressure steam should cold overpressurization events occur with a steam bubble in the pressurizer. Although there were no low pressure steam tests performed for the PORVs, valve performance is enveloped by the high pressure steam results.

Based on the above discussion it is concluded that the tested conditions are representative of expected conditions for cold overpressurization events.

8. Bending moments are induced on the safety valves and PORVs during the time they are required to operate because of discharge loads and thermal expansion of the pressurizer tank and inlet piping. Make a comparison between the predicted plant moments with the moments applied to the tested valves to demonstrate that the operability of the valves will not be impaired.

RESPONSE TO QUESTION 8:

The calculated maximum bending moment at the safety discharge flange is 150,189 inch-pounds compared to the maximum moment developed in the applicable EPRI test at 179,250 inch-pounds for the Crosby 6M6 safety valve. For the Copes-Vulcan relief valves the calculated maximum bending moment at the discharge flange is 37,900 inch-pounds compared to EPRI test 64-CV-174-2S value of 43,000 inch-pounds. The IP-3 calculated bending moments are based upon implementation of the modification as described below in the response to question 12 (e).

9. The Westinghouse inlet fluid conditions report stated that liquid flow could exist through the PORV for the FSAR feedline break event and the extended high pressure injection event. Liquid PORV flow is also predicted for the cold overpressurization event. These same flow conditions will also exist for the Block Valve. The EPRI/Marshall Block Valve Report did not test the block valves with fluid media other than steam. The Westinghouse Gate Valve Closure Testing Program did include tests with water; however the information presented in the report did not provide specific test results. Since it is conceivable that the EMOV could be expected to operate with liquid flows, discuss EMOV block valve operability with expected liquid flow conditions and provide specific test data.

RESPONSE TO QUESTION 9:

As discussed in the response to question 2 above, the feedline break event is outside the IP-3's original licensing basis. The IP-3 safety injection pump head is less than both the PORV's and safety valves setpoint pressure, therefore, the extended high pressure injection event is not applicable to our plant. The PORV block valves may discharge liquid flow during cold overpressurization events. Although there are no plant specific test data, it is the Authority's position that the EPRI high pressure steam test results envelop the low pressure/low temperature solid water operation of these valves. This is based on the fact that the stem thrust is a function of valve seat area and differential pressure and is independent of the fluid media. Therefore, the high pressure steam would produce a higher stem thrust and require a higher motor torque than lower pressure water. Since the EPRI/Marshall block valve tests showed that the valve opened and closed on demand during higher pressure steam testing, it is expected that the valve will operate as required during low pressure solid water operations.

10. The PORV block valves tested at the Marshall steam station were tested only in horizontal piping runs with the PORV block valve stems in the vertical upright position. Discuss the mounting configuration of the Indian Point 3 PORV block valve. If the mounting configuration is other than horizontal with the valve stem vertically upright, discuss the effects of the installed block valve configuration on valve operability and reliability. Also the submittal identifies the Indian Point 3 block valve actuator as a SMB-00-5 whereas the EPRI Block Valve Information Report identifies the Indian Point 3 Block Valve actuator as a SMB-00-10. Provide information to clarify.

RESPONSE TO QUESTION 10:

The IP-3 block valve configuration is horizontal with vertical stems. The valve actuator is SMB-00-05.

The Marshall Electric Motor Operator Valve (Block Valve) Test Report, tested the Velan gate valve (drawing 88425/B) equipped with a Limitorque SMB-00-15 ft.-lb. motor operator.

The plant specific valve is a Velan gate valve (drawing 88405/1). The tested and plant specific valves are identical except that the tested valve used a forged wedge (gate) while the plant specific valve has a cast wedge (gate).

The Limitorque operators differ between the tested and plant specific valves (15 ft.-lb. vs. 5 ft.-lb.). The operator output required is a function of closing speed. The tested valve with the 15 ft.-lb. operator had a closing time of 10 seconds while the plant specific valve with a 5 ft.-lb. operator has a closing time of 60 seconds. For the plant specific valve, Limitorque calculated a total stem thrust required of 6967 lb_f, based on a cycle time of 60 second and a pressure differential of 2515 psia. The SMB-00-05 ft.-lb. operator produces a total stem thrust of 8000 lb_f. Therefore, the operator is adequately sized for the plant specific cycle time.

Considering the valve bodies are identical and that the smaller operator is more than adequate at the slower plant specific speed, the Block Valve test can be used to demonstrate plant specific operability.

QUESTIONS RELATED TO THE THERMAL HYDRAULIC ANALYSIS OF THE INLET AND DISCHARGE PIPING

11. The submittal states that a thermal hydraulic analysis of the safety/relief valve piping system has not been completed, but does present preliminary results of the analysis. To allow for a complete evaluation of the methods used and the results obtained from the thermal hydraulic analysis, provide a discussion on the thermal hydraulic analysis that contain at least the following information:

(a) Evidence that the analysis was performed on the fluid transient cases producing the maximum loading on the safety-PORV piping system. The cases should bound all steam, steam to water, and water flow transient conditions for the safety and PORV valves.

(b) A detailed description of the methods used to perform this analysis. This includes a description of methods used to generate fluid pressures and momenta over time and methods used to calculate resulting fluid forces on the system. Identify the computer programs used for the analysis and how these programs were verified.

(c) Identification of important parameters used in the thermal hydraulic analysis and rationale for their selection. These include peak pressure and pressurization rate, valve opening time, and fluid conditions at valve opening.

(d) An explanation of the method used to treat valve resistances in the analysis. Report the valve flow rates that correspond to the resistances used. Because the ASME Code requires derating of the safety valves to 90% of actual flow capacity, the safety valve analysis should be based on flows equal to 111% of the valve flow rating, unless another flow rate can be justified. Provide information explaining how derating of the safety valves was handled and describe methods used to establish flow rates for the safety valves and PORVs in the analysis.

(e) A discussion of the sequence of opening of the safety valves that was used to produce worst case loading conditions.

(f) A sketch of the thermal hydraulic model showing the size and number of fluid control volumes.

(g) A copy of the thermal analysis report.

REFERENCES

(QUESTIONS 11 and 12 only)

1) EPRI PWR Safety and Relief Valve Test Procedure Guide for application of valve test program results to plant specific evaluations. (July, 1982).

2) Mainstream-EKS, Relap 5/MOD 1, Code Manual, volume 1 and 2. Boeing Computer Services Company.
Code: "Repas 5/MOD 1, SRV 300"
Nuclear Program L.B.

3) Mainstream - EKS, Force Reference Manual and Access Guide, Boeing Computer Services Company.
Code: "Force, EECCL, Ver. 4.01"
Nuclear Program L.B.

4) Application of Relap 5/MOD 1 for Evaluation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads (April 1982)

5) Review of Pressurizer Safety Valve Performance as observed in EPRI Safety and Relief Valve Test Program (WCAP - 10105) (June 1982)

6) EPRI PWR Safety and Relief Valve Test Program Test Condition Justification Report (April 1982)

7) STARDYNE User Manual, Sept. 1979
Boeing Computer Services Co.
Code: STARDYNE - 3, DYNREG"
Decol/82AM Level

RESPONSE TO QUESTION 11:

11 (a) The safety and relief valve piping adequacy was evaluated using the general guidelines provided in Reference 1. The first evaluation is for the post loop-seal discharge period and is based on the sequential actuation of PORV's and SRV's at their respective set pressures. A second post loopseal discharge case with the PORV block valve closed and the SRV's only opening at their set pressures, was also evaluated. The piping and support evaluation is based on selecting the maximum of the peak load predicted for each case with some modification.

The pressurization rates, peak pressures, valve opening times and flowrates are given below in response to questions 11 (c) and (d). The ramp rates and peak pressures represent the maximum values developed by Westinghouse for generic bounding conditions. The opening times represent the mimimum opening times as determined in the EPRI test program for these valves. The flowrates represent the maximum measured flowrates from the EPRI test data, adjusted for the inlet pressure drop with considerations given to ASME valve deratings.

(b) The transient pressures, temperatures and flow rates were computed using the "Relap 5/Mod 1", cycle 14 computer code. A post processor code, "FORCE" was used to determine the forces during the transient. Both codes were verified and provided by Boeing Computer Services Company. The detailed description is in References 2 and 3.

(c) Parameters used in the thermal hydraulic analysis. The valve inlet conditions are from References 4 and 5. They represent Westinghouse calculated generic conditions for a typical four loop plant and as such envelope IP-3 conditions.

c-1 Pressurizer:

For the sequential actuation of S/R valves

Pressurization Rate	=	130 psi/sec.
Peak Pressure	=	2532 psia

For the safety valve actuation only.

Pressurization Rate	=	144 psi/sec.
Peak Pressure	=	2555 psia

c-2 Safety Valve: - Crosby HB-BP-86-6M6

Valve Opening Pressure	=	2500 psia
Opening Time	=	0.01 sec.

c-3 Relief Valve - Copes Vulcan 316/stellite

Valve Opening Pressure	=	2350 psia
Main Disc Opening Time		0.39 sec.

The opening times selected are based EPRI test program results in References 5 and 6.

c-4 Fluid Conditions

Water loop seals are at inlets of safety valves.

The temperature of water at the inlet is 260^{o}F assuming insulation around the loop seals.

11 (d) The safety valve is rated for the steam flow of 420,000 lbs/hr at 2500 psia. The flow rate used in the analysis is 480,240 lbs/hr. This represents the maximum measured flow rate from the EPRI test data adjusted for the inlet pressure drop.

The capacity of relief valve is rated for 179,000 lbs/hr at 2350 psia. The flow from the EPRI test data adjusted for inlet pressure drop are 245,000 lbs/hr at valve 456 and 246,960 lbs/hr at valve 455 c.

11 (e) The effect of the sequence of opening of the safety valves was studied.

The safety and relief valve piping adequacy was evaluated using the general guidelines provided in Reference 1. The piping stresses and support loads are the peak values of the two transient cases being evaluated as described in 11 (a). Furthermore, the thermal hydraulic analysis which provides the input transient forces for the structural analysis was based on various conservative assmuptions such as:

(1) The pressurizer ramp rates were taken from Westinghouse generic bounding conditions for RCP locked rotor event. The actual plant specific ramp rate, as shown in section 14.1.6 of the FSAR, is 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.

11 (f) Isometrics of the thermal hydraulic model are shown in Figure 1 through 7 attached. The model consists of 292 volumes and 310 junctions. The average length of volume is less than one foot.

11 (g) A summary of the thermal anslysis report is reflected in the above responses. Detailed calculations and computer output are available at the NYPA White Plains office for review.

NewYorkPower Authority

Project IP3 (83-0111) Page 14 of 147

Subject PRESSURIZER S/R V DISCHARGE PIPING LOAD Calc. Date 6/12/85 19

Computed by HYC

Checked by

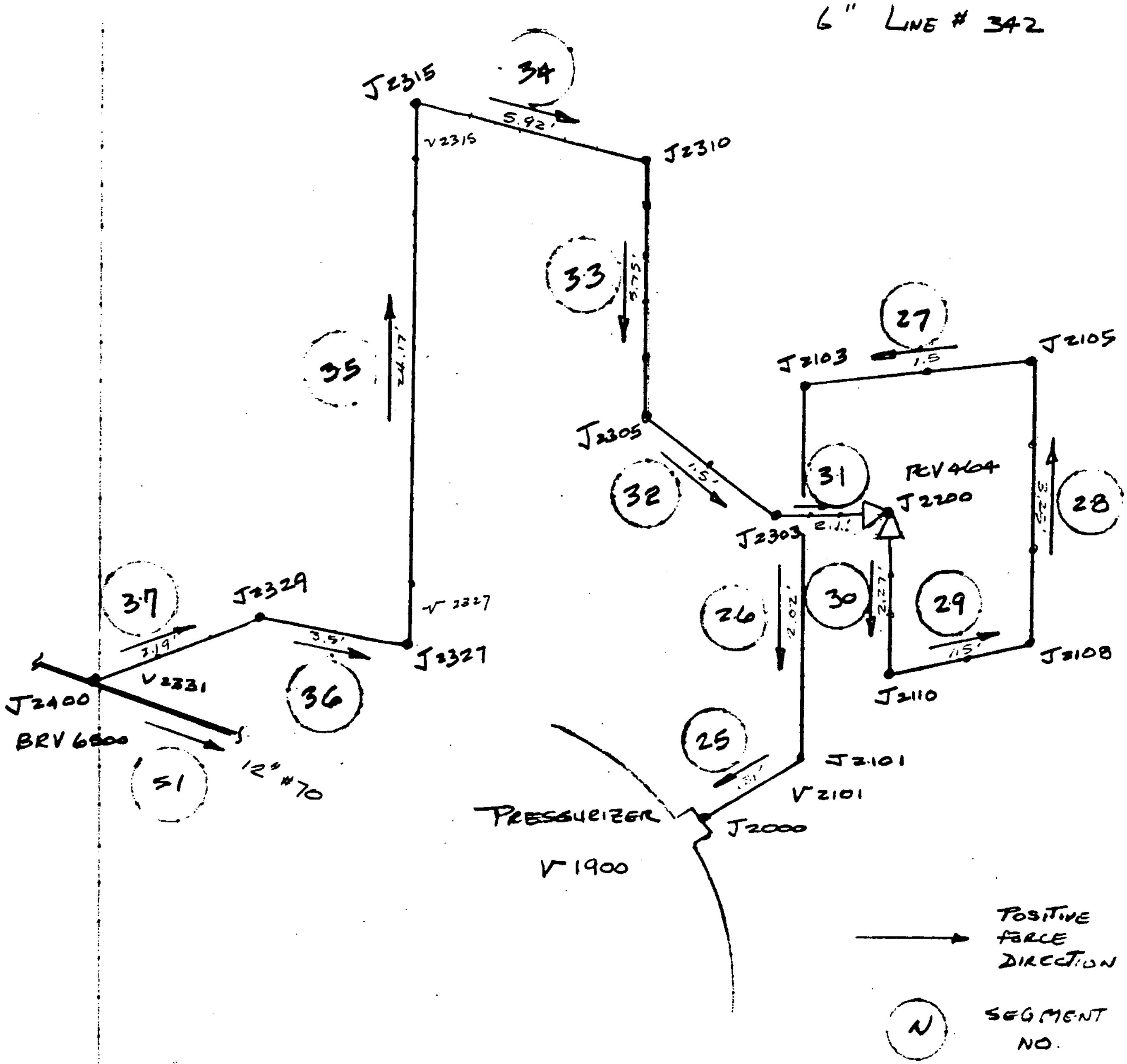


FIG. 1 ISOMETRIC OF LINE 342

V VOLUME

J : JUNCTION

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Project JP3 (83-0..)
Page 15 of 147

Subject PRESSURIZER S/RV DISCHARGE PIPING LOAD Calc.
Date 6/12/85 19

Computed by HYC

Checked by

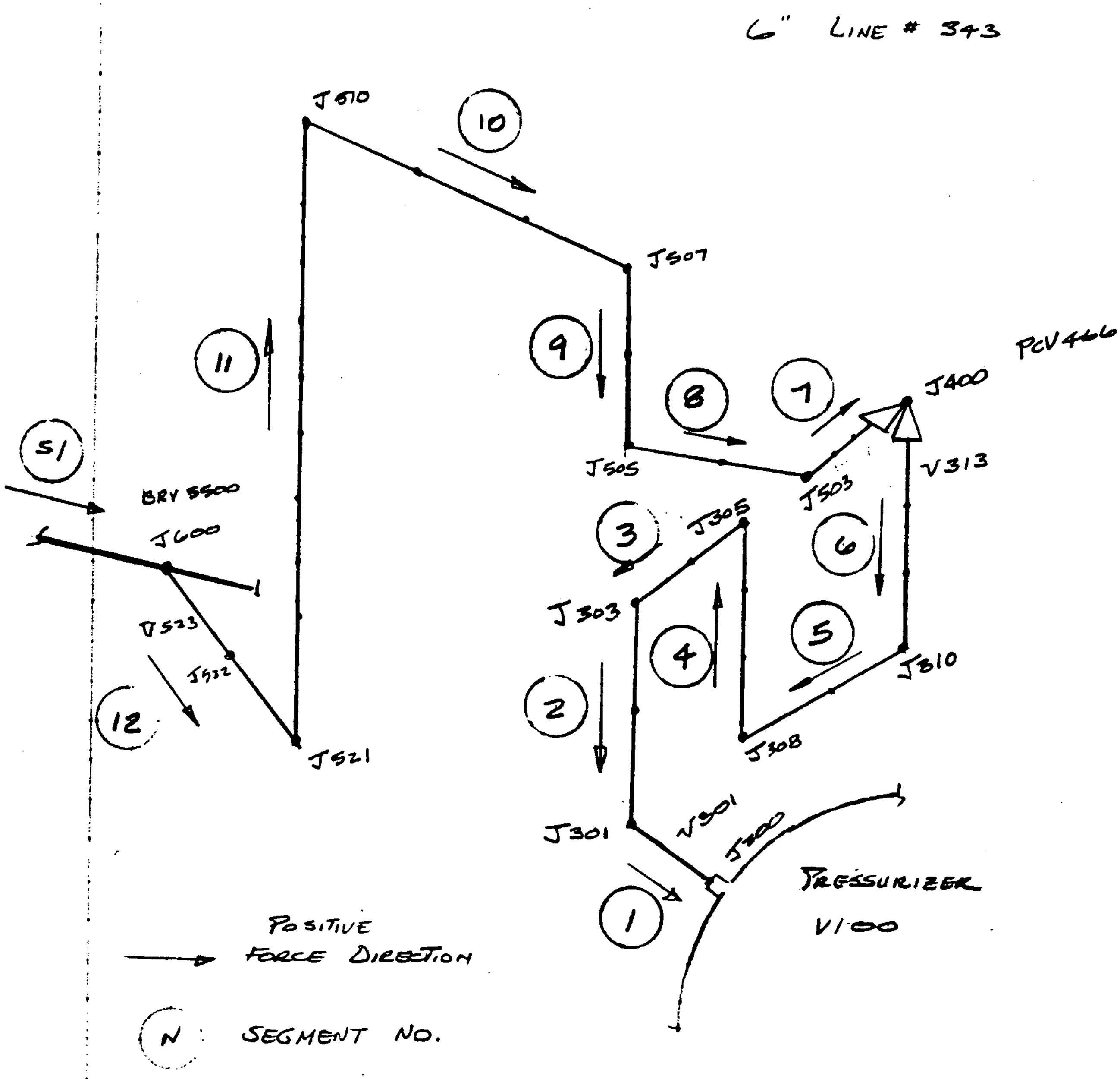

FIG. 2 ISOMETRIC OF LINE # 343

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Project IP3 (83-0...) Page 16 of 147
Subject PRESSURIZER S/R V DISCHARGE PIPING LOAD CALC. Date 6/12/85 19
Computed by HYC
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6" LINE # 344

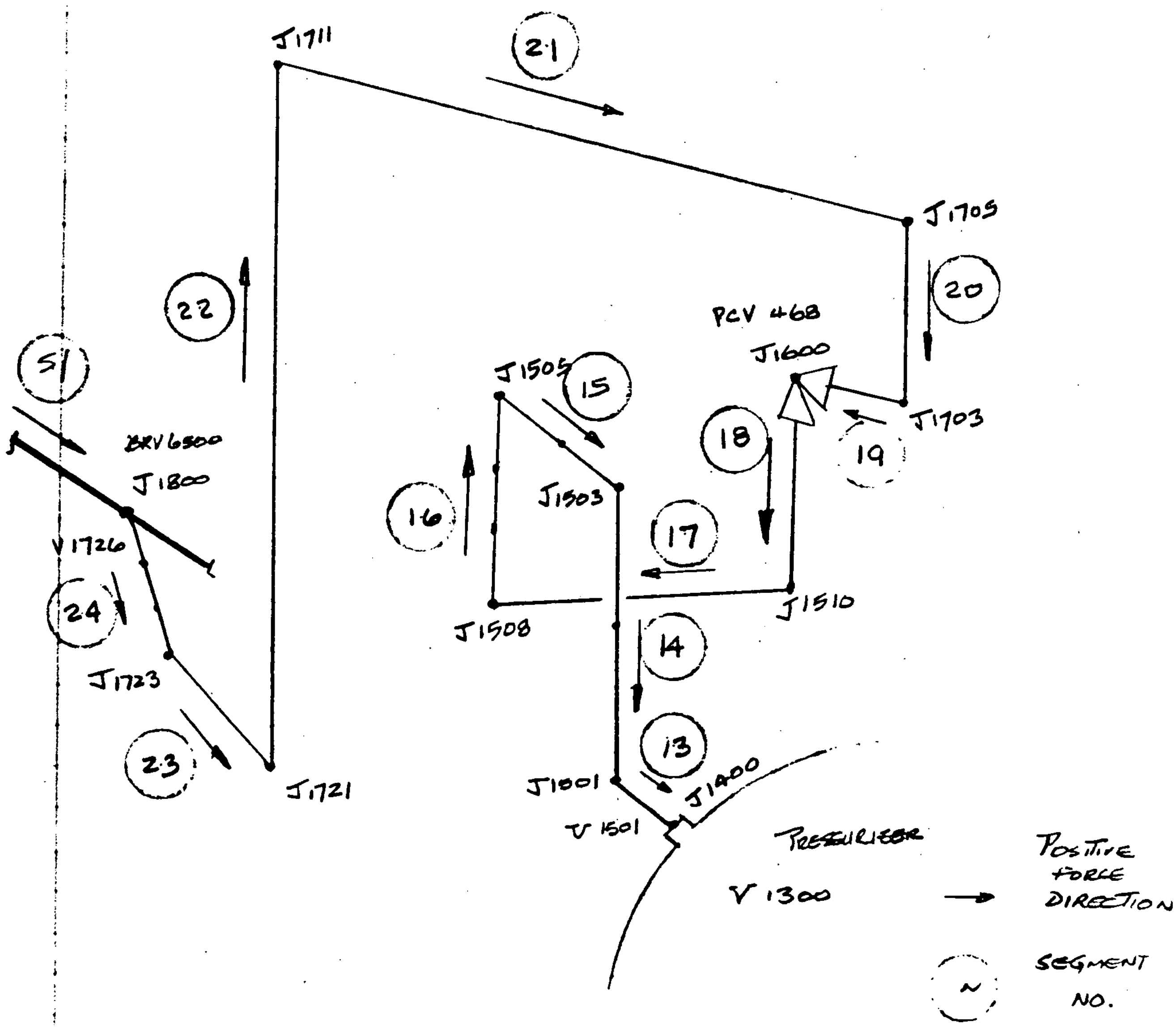

FIG. 3 ISOMETRIC OF LINE # 344

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Project IP3 (8-0''') Page 17 of 147

Subject PRESSURIZER S/R V DISCHARGE PIPING LOAD CALC. Date 6/12/85 19

Computed by MYC

Checked by

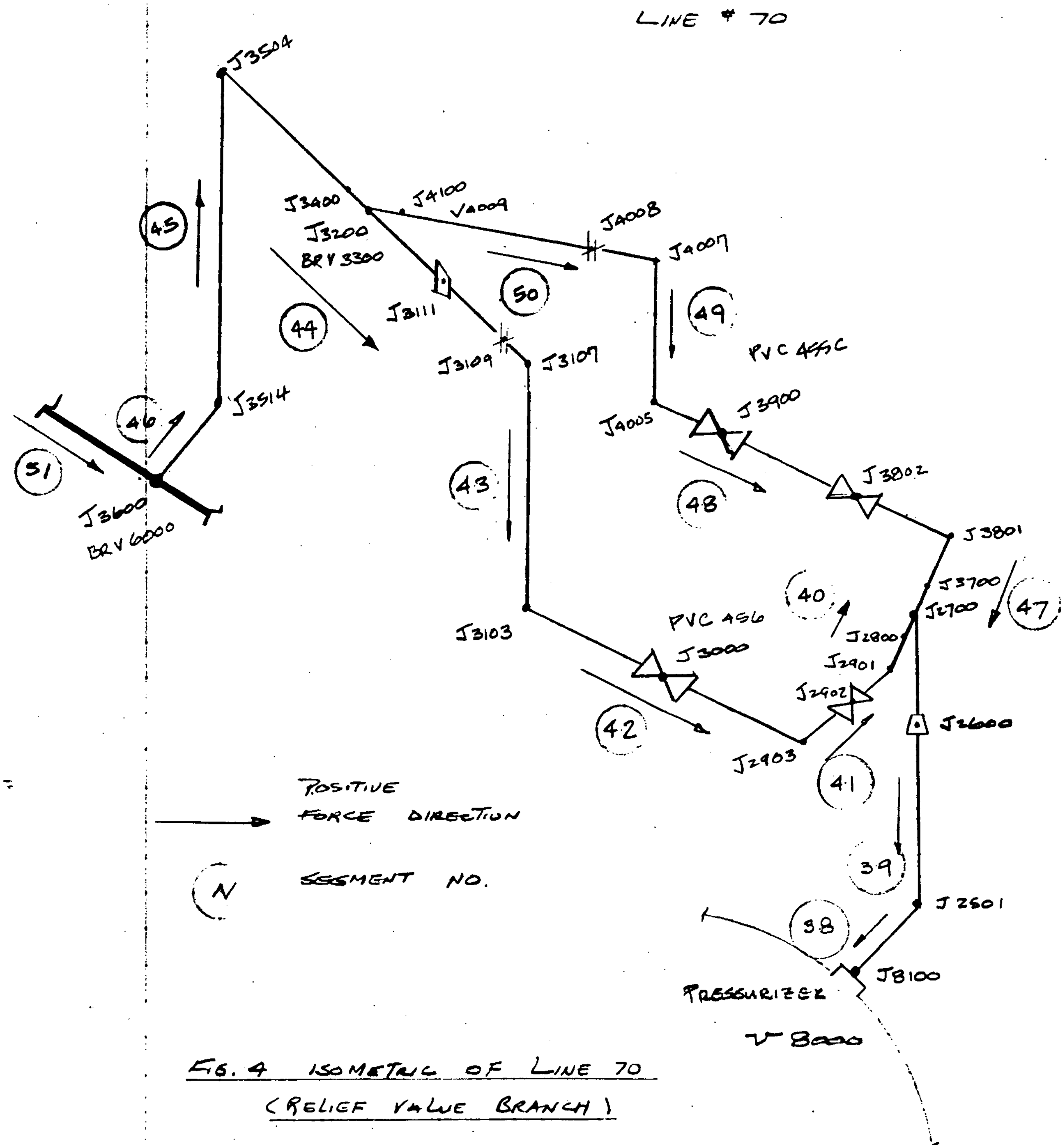

FIG. 4 ISOMETRIC OF LINE 70
(RELIEF VALVE BRANCH)

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Project IP3 (83-011)
Subject PRESSURIZER S/R V DISCHARGE PIPING LOAD CALC.

Date 6/12/83
Computed by HYC
Checked by

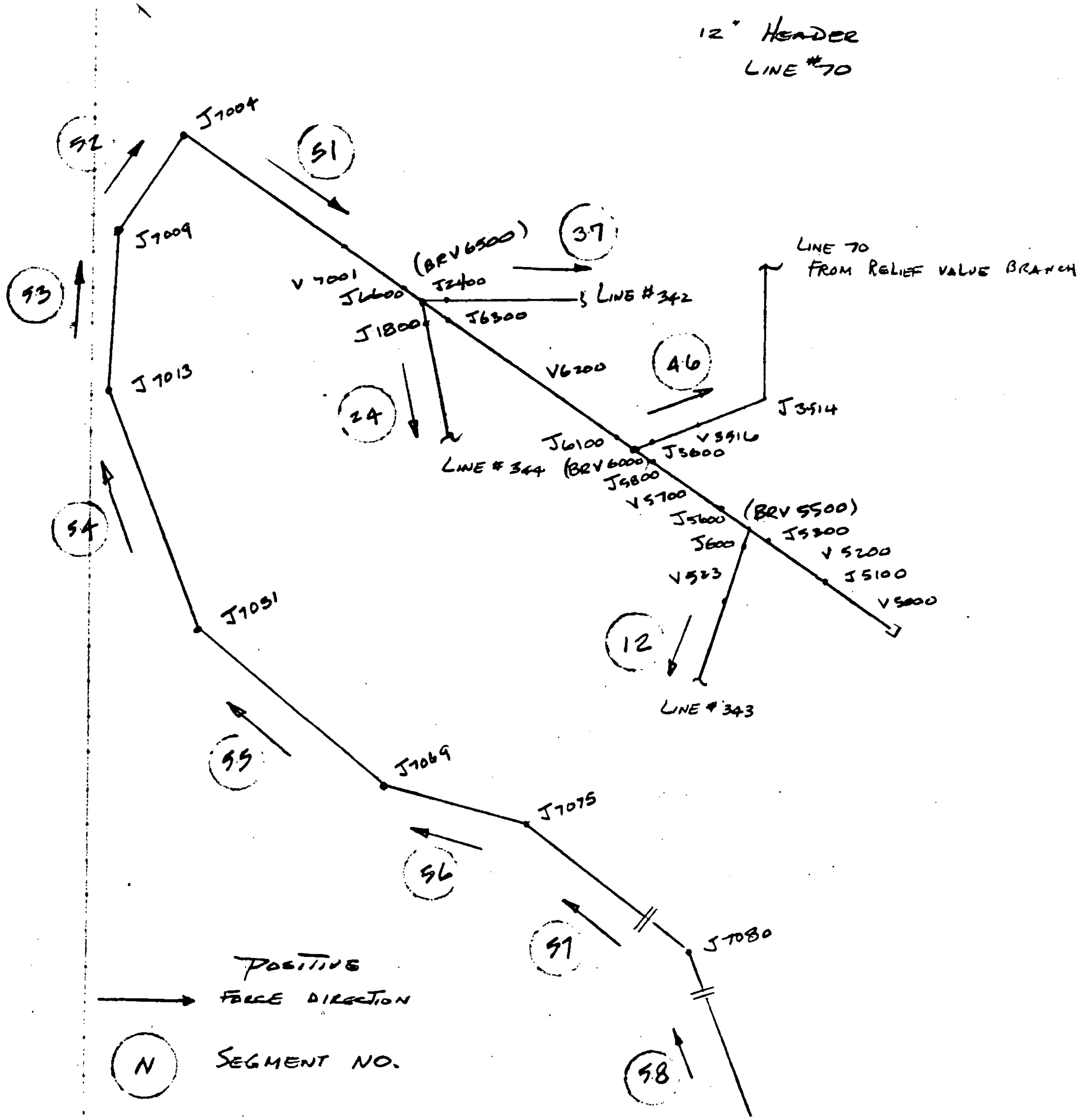

FIG. 5 ISOMETRIC OF HEADER (LINE 70)

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Project IP3 (87-0111) Page 19 of 147
Subject PRESSURIZER S/R V DISCHARGE Date 6/12/85 19
PIPING LOAD CALC. Computed by HYC
Checked by

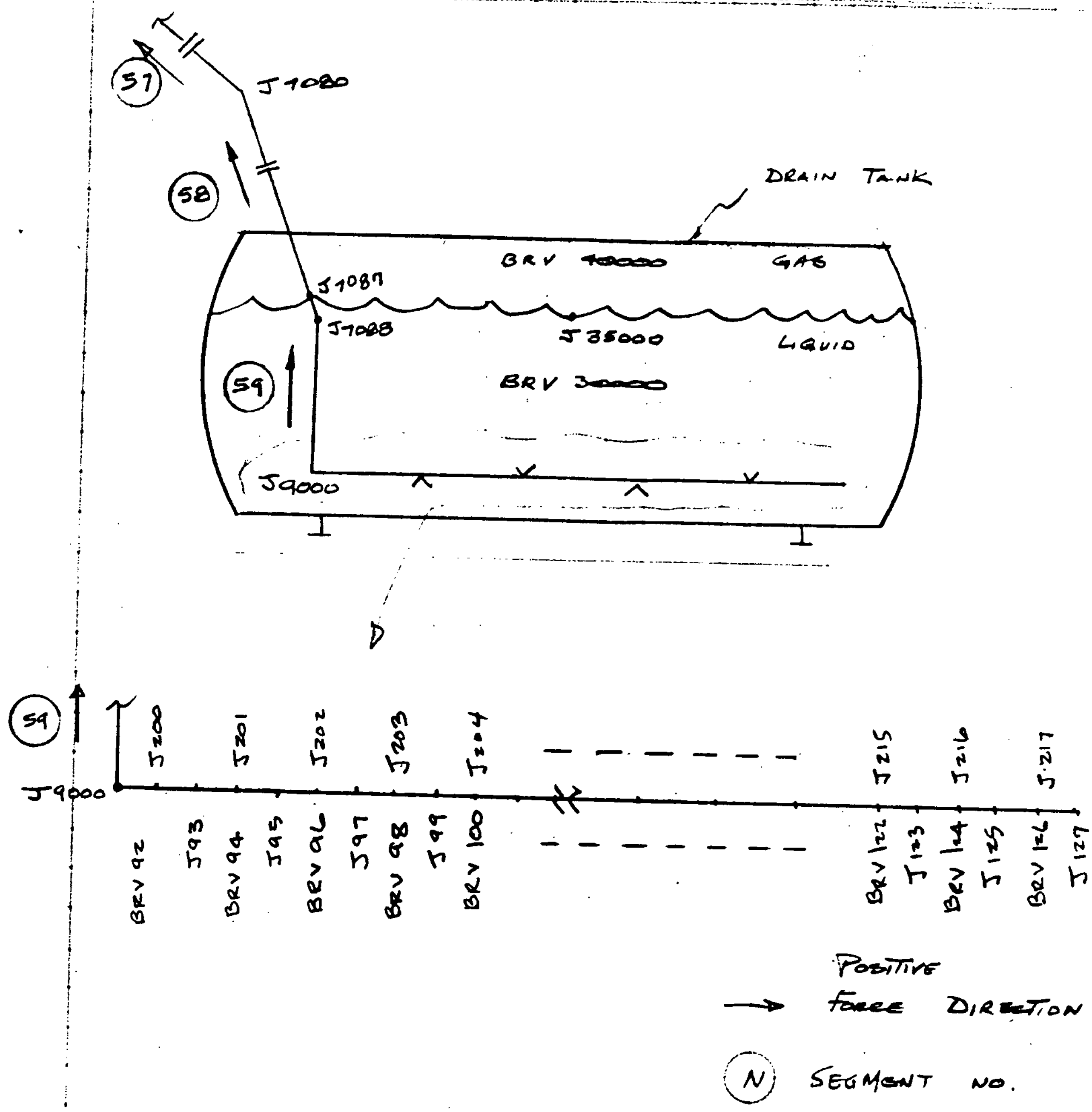

Fig. 6 DRAIN TANK

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Project IP3 (83-0...) Page 20 of 147

Subject PRESSURIZER S/RV DISCHARGE P.PWG LOAD CALC. Date 6/12/85 19

Computed by HYC

Checked by

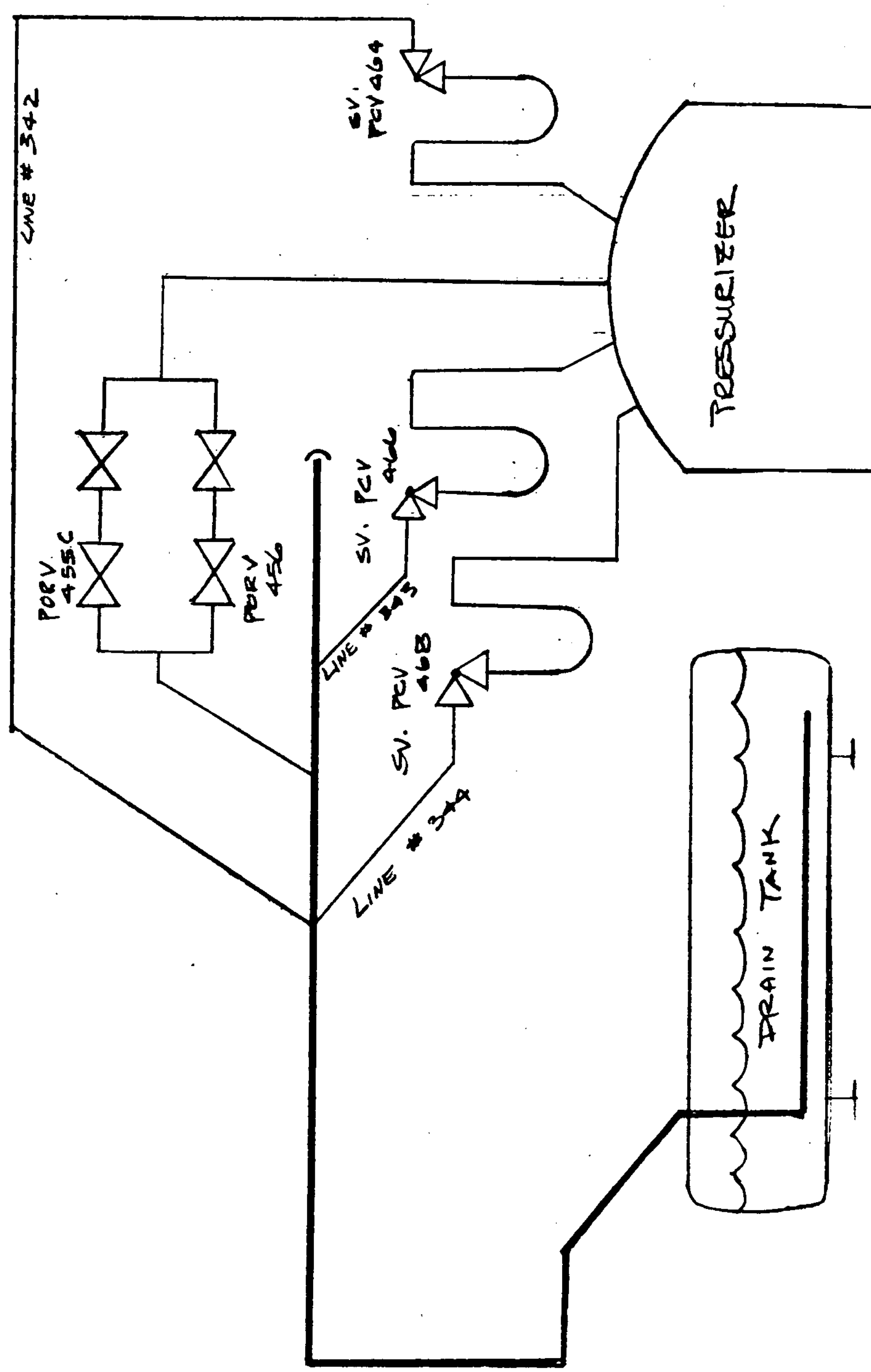

FIG. 7 PRESSURIZER S/RV DISCHARGE SYSTEM SCHEMATIC

QUESTIONS RELATED TO THE STRUCTURAL ANALYSIS OF THE INLET AND DISCHARGE PIPING

12. The submittal states that a structural analysis of the safety/PORV valve piping system has not been completed, but does present preliminary results of the analysis. To allow for a complete evaluation of the methods used and results obtained from the structural analysis, please provide reports containing at least the following information:

    (a) A detailed description of the methods used to perform the analysis. Identify the computer programs used for the analysis and how these programs were verified.

    (b) A description of the method used to apply the fluid forces to the structural model. Since the forces acting on a typical pipe segment are composed of a net, or "wave," force and opposing "blowdown" forces, describe the methods for handling both types of forces.

    (c) A description of methods used to model supports, the pressurizer and relief tank connections, and the safety valve bonnet assemblies and PORV actuator.

    (d) An identification of the load combinations performed in the analysis together with the allowable stress limits. Differentiate between load combinations used in the piping upstream and downstream of the valve. Explain the mathematical methods used to perform the load combinations, and identify the governing codes and standards used to determine piping and support adequacy.

    (e) An evaluation of the results of the structural analysis, including identification of overstressed locations and a description of modifications if any.

(f) A sketch of the structural model showing lumped mass locations, pipe sizes, and application points of fluid forces.

(g) A copy of the structural analysis report.

Response to question 12:

12 (a) The direct time integration technique is used to compute the response of multiple-degree-of-freedom piping system subjected to unbalanced segment forces. The computer code STARDYNE-DYNRE6 is used to perform the analysis. The code was verified as provided by Boeing Computer Service Company. The detail is described in Reference 7.

12 (b) Fluid forces generated from the thermal hydraulic analysis provide for the un-balanced segment forces acting on the pipe segment. The segment forces are applied to their corresponding piping segments to evaluate the piping stresses and support loads dynamically.

12 (c) The support stiffness is evaluated by the standard component stiffness combined with its back-up structure in series. Pressurizer connections are assumed as anchors on the piping. An effective stiffness is used to model the connection between the pipe and relief tank. The valve assemblies are modeled with the valve weight at its center of gravity.

12 (d) The load combinations for the piping stresses and support loads and the governing codes and standards are listed in Tables 1 and 2. The upstream and down-stream tripings of valves are applied with the same criteria.

12 (e) With a modification, the piping stresses and

pipe loads are all within acceptable allowables. The modification includes eight new snubbers added and five Tee sections strengthened with pads.

It should be noted that:

(1) Stress due to "SOTU" are assumed to be negligible.

(2) The effects of MS/FWPB, DBPB and LOCA are assumed to be negligible on the subject system, based on the following physical characteristics of the plant design:

(a) The reactor and pressurizer vessels are independently supported and physically separated by concrete encasements.

(b) The 14 inch surge piping interconnecting subject vessels have relatively low stiffness characteristics as compared to the vessels and their supporting structures. This results in negligible transmissibility of reactions due to pipe breaks.

(c) Primary piping systems such as Main Steam and Feedwater have an independent scheme of whip restraint hardware which is provided to prevent a pipe rupture from damaging structures or other components important to safety.

12 (f) A sketch of the structural model is shown in Figure 8.

12 (g) A summary of the structural analysis report is reflected in the above responses. Detailed calculations and computer output are available at the NYPA White Plains office for review.

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Project .... Page 5 of ......

Subject .... Date .... 19......

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TABLE 1

LOAD COMBINATION AND STRESS LIMITS FOR PIPING

OPERATION CONDITION	LOAD COMBINATION	STRESS LIMITS	CODE & STANDARD
NORMAL	N *	S_h	FSAR TABLE 16.1-2,
UPSET	N ± OBE ± SOT_U	$1.2 S_h$	EPRI W.CAP-10105 TABLE 4-5 AND
EMERGENCY	N ± SOT_E	$1.8 S_h$	ASME SECTION III NC/ND-3652
FAULTED	N ± DBE ± SOT_F + MS/FWPB (OR DBPB)	$2.4 S_h$	
	N ± DBE ± SOT_F + LOCA		
THERMAL ANALYSIS	THERMAL EXPANSION	S_A	ASNI B31.1
	N + THERMAL EXPANSION	$S_h + S_A$	

* SEE DEFINATIONS OF LOAD ABBREVIATIONS

NewYorkPower
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Project

Subject

Page 6 of

Date 19

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TABLE 2

LOAD COMBINATIONS AND STRESS LIMITS
FOR SUPPORT LOADS

OPERATION CONDITION	LOAD COMBINATION	ALLOWABLE	
		SNUBBER *	STRUCTURE
NORMAL	N + THERMAL	1.0 CAPACITY	FSAR TABLE 16.1-2 · AISC CODE APPLIED.
UPSET	N + THERMAL ± OBE ± SOT_U	1.0 CAPACITY	
EMERGENCY	N + THERMAL ± SOT_E	1.33 CAPACITY	
FAULTED	N + THERMAL ± DBE + SOT_F + MS/FWPB (OR DRPB)	1.5 CAPACITY	
	N + THEMAL ± DBE + SOT_F + LOCA		

* PROVIDED BY THE MANUFACTURE

DEFINITIONS OF LOAD ABBREVIATIONS

N	:	Deadweight plus pressure
SOT	:	System operation transient
SOTU	:	Relief Valve Discharge Transient
SOTE	:	Safety Valve Discharge Transient
SOTF	:	Max. (SOTU, SOTE)
OBE	:	Operating Basis Earthquake
DBE	:	Design Basis Earthquake
MS/FWPB	:	Main Steam or Feedwater Pipe Break
DBPB	:	Design Basis Pipe Break
LOCA	:	Loss of Coolant Accident

LINE 70 PLUS

LINES 342 TO 344 FROM FIRST ANCHOR TO LINE 70

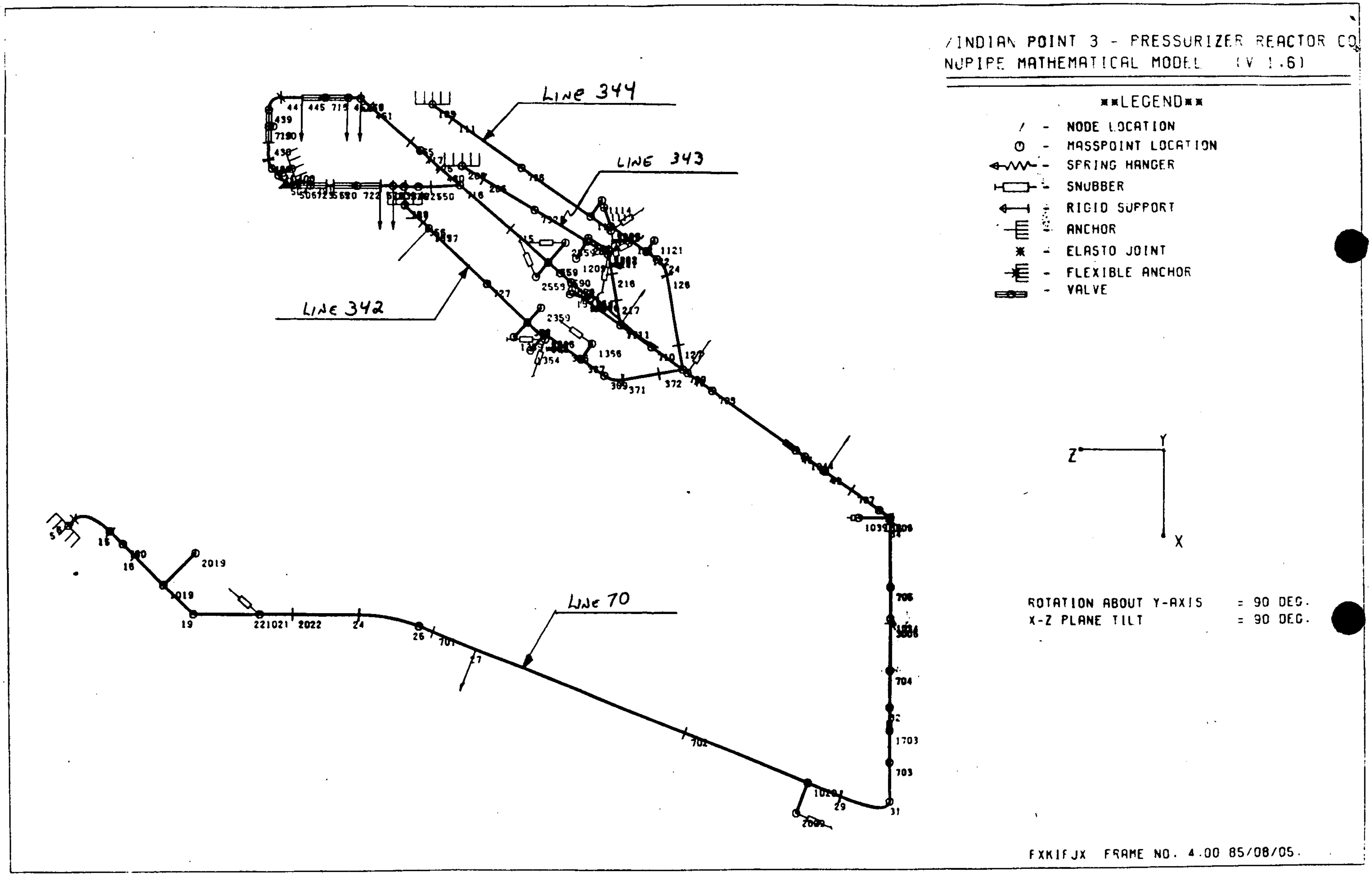

FIGURE 8(a)

LINE 344

FROM PRESSURIZER TO FIRST ANCHOR

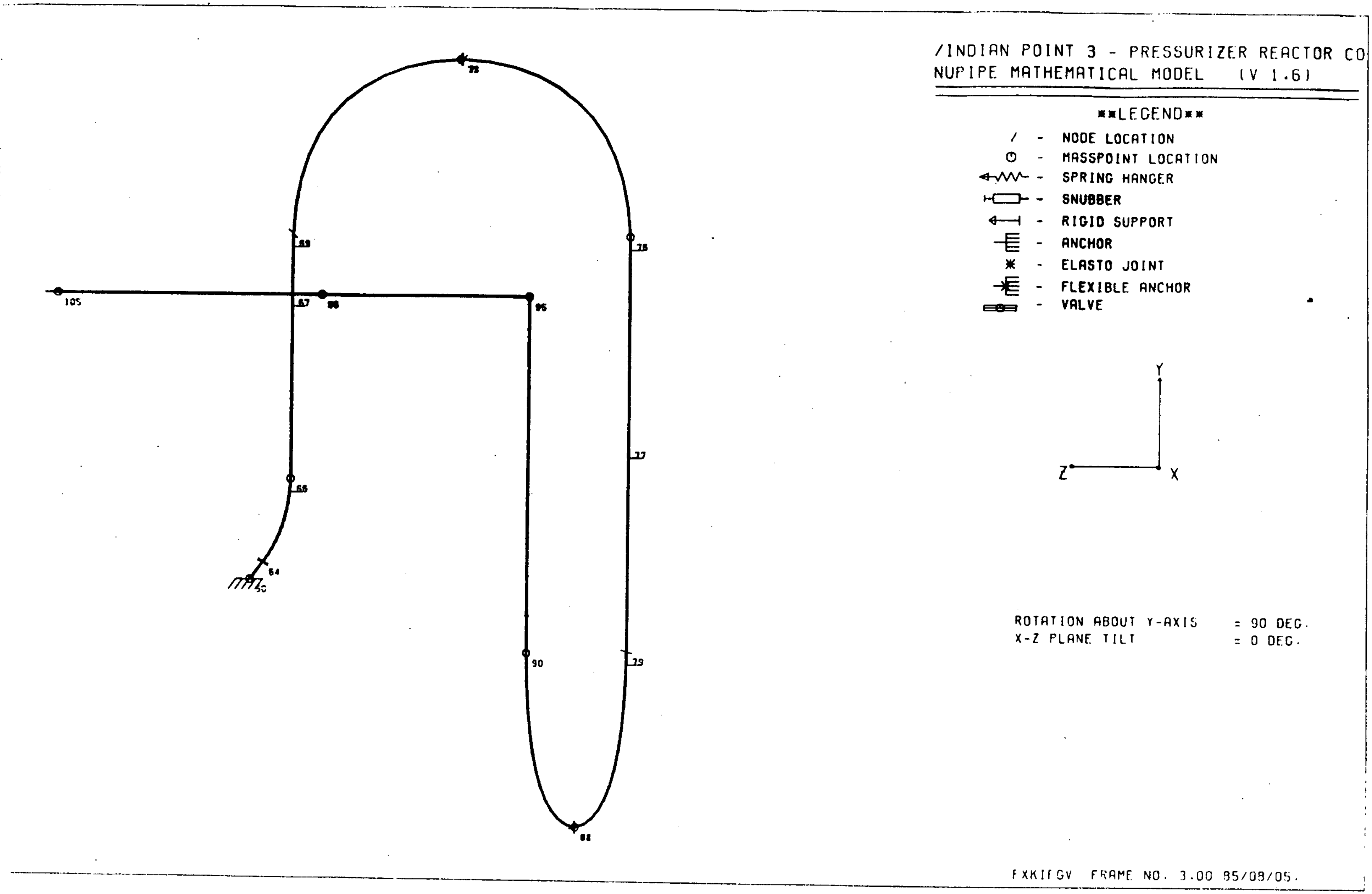

FIGURE 8 (b)

LINE 342
FROM PRESSURIZER TO FIRST ANCHOR
(RESULTS ARE VALID FOR LINE 343 ALSO)

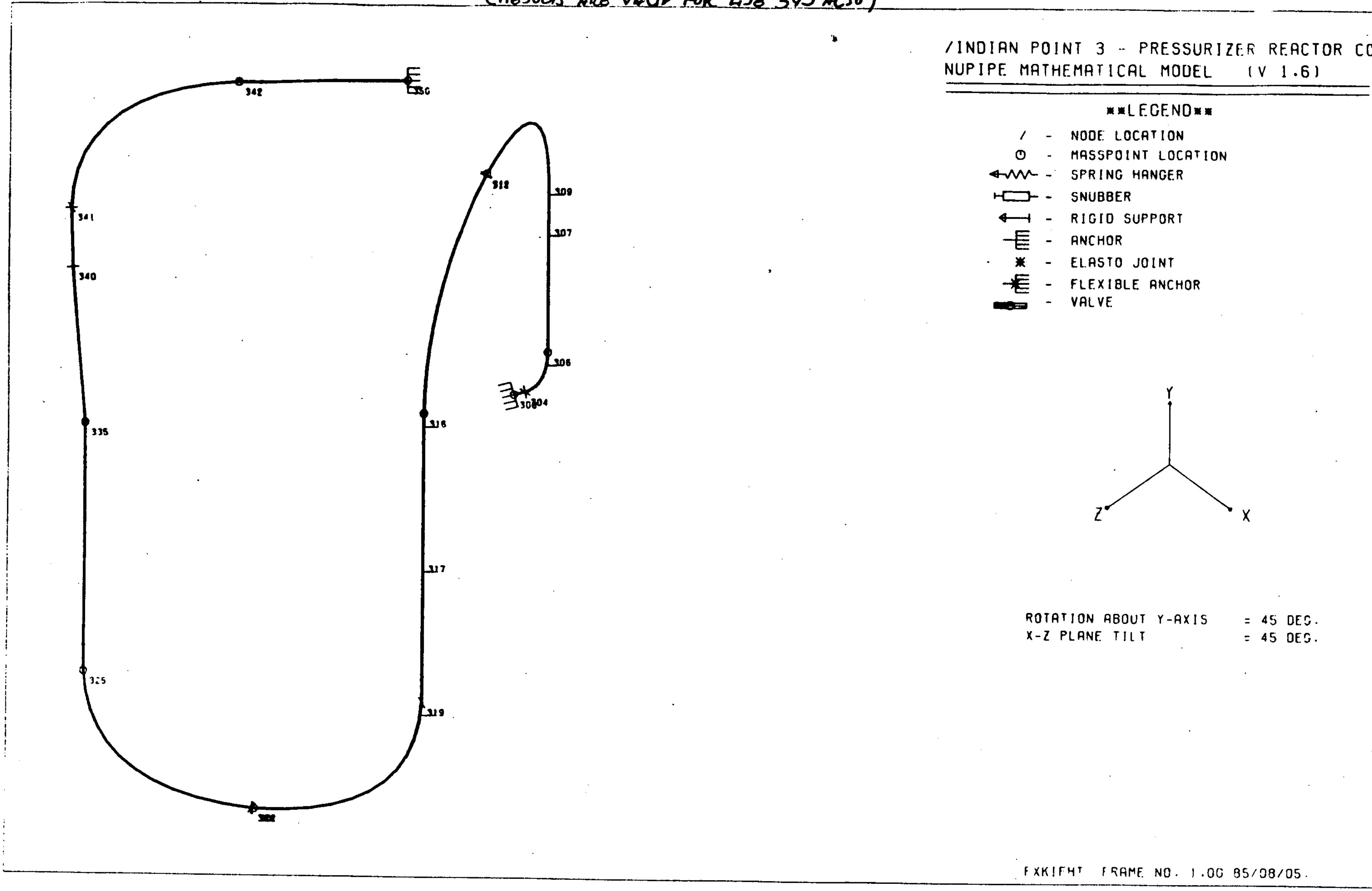

FIGURE 8(c)

13. According to results of EPRI tests, high frequency pressure oscillations of 170-260 Hz typically occur in the piping upstream of the safety valve while loop seal water passes through the valve. An evaluation of this phenomenon is documented in the Westinghouse report WCAP 10105 and states that the acoustic pressures occurring prior to and during safety valve discharge are below the maximum permissible pressure. The study discussed in the Westinghouse report determined the maximum permissible pressure for the inlet piping and established the maximum allowable bending moments for Level C Service Condition in the inlet piping based on the maximum transient pressure measured or calculated. Provide the peak pressures expected at Indian Point 3 and a comparison to the pressure allowed in WCAP 10105. The pressure oscillations could potentially excite high frequency vibration modes in the piping, creating bending moments in the inlet piping that should be combined with moments from other appropriate mechanical loads. Provide one of the following: (1) a comparison of the expected peak pressures and bending moments with the allowable values reported in the WCAP report or (2) justification for other alternate allowable pressure and bending moments with a similar comparison with peak pressures and moments induced in the plant piping.

Response to question 13:

Applying the Westinghouse report WCAP 10105 for the Indian Point 3 plant-specific loop-seal length, provides a peak pressure of 4200 psia. The permissible pressure for the safety valve inlet pipe size of 6" sch. 120 is 5460 psia for Level C service limits. The safety valve loop seal discharge peak pressure meets the limits. The calculated maximum bending moment at the safety discharge flange is 150,189 inch-pounds, compared to the maximum moment developed in the applicable EPRI test at 179,250 inch-pounds. For the primary stress intensity limit, the evaluated internal pressure and maximum moment (B_2 M_I) are 4200 psia and 306 in-kips compared to the allowable values 5000 psia and 386 in-kips, respectively, from Table 4-7 of the WCAP report.