YANKEE ATOMIC ELECTRIC COMPANY

Telephone (617) 872-8100
TWX 710-380-7619

1671 Worcester Road, Framingham, Massachusetts 01701

2.C.2.1
FYR 85-132

November 22, 1985

United States Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. John A. Zwolinski, Chief
Operating Reactors Branch No. 5
Division of Licensing

References:
(a) License No. DPR-3 (Docket No. 50-29)
(b) Letter, YAEC to NRC, FYR 82-11, "SEP Topic Assessment Completion," dated February 1, 1982
(c) Letter, YAEC to NRC, FYR 82-38, "TMI Item II.D.1, Safety and Relief Valves," dated March 30, 1982
(d) Letter, YAEC to NRC, FYR 82-72, "TMI Item II.D.1, Safety and Relief Valves," dated July 1, 1982
(e) Letter, YAEC to NRC, FYR 82-82, "TMI Item II.D.1, Safety and Relief Valves," dated August 1, 1982
(f) Letter, YAEC to NRC, FYR 82-121, "TMI Item II.D.1, Safety and Relief Valves," dated December 28, 1982
(g) Letter, YAEC to NRC, FYR 83-36, "TMI Item II.D.1, Safety and Relief Valves," dated April 1, 1983
(h) Letter, YAEC to NRC, FYR 84-41, "TMI Item II.D.1, Safety and Relief Valves," dated April 2, 1984
(i) Letter, NRC to YAEC, NYR 85-119, "Request for Additional Information on TMI Action Plan Item II.D.1, Performance Testing of Relief and Safety Valves," dated July 16, 1985
(j) Letter, USNRC to YAEC, NYR 85-119, "Request for Additional Information on TMI Action Plan Item II.D.1, Performance Testing of Relief and Safety Valves," dated July 16, 1985

Subject: TMI Item II.D.1 Safety and Relief Valves

Dear Sir:

Reference (j) requested additional information on the performance testing and design of our Pressurizer Safety and Relief Valves. The attachment to this letter contains the requested information.

We trust this information is satisfactory; however, if you have any questions, please contact us.

Respectfully,

YANKEE ATOMIC ELECTRIC COMPANY

George Papanic Jr.

G. Papanic, Jr.
Senior Project Engineer
Licensing

A046
/1

RDL/glw
Attachment

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

1. The submittal does not include a discussion of consideration of single failures after initiating events. NUREG-0737 requires selection of single failures that produce maximum loads on the safety and relief valves. Include a discussion describing how the single failure considerations are met.

<u>Response</u>:

The safety and relief valve actuating events considered in the NRC submittal covered three separate categories:

1. Overpressurization events included in Safety Analysis Reports (SARs) and core reload analyses.

2. Inadvertent actuation of the High Pressure Safety Injection (HPSI) System during normal hot operating conditions.

3. Low Temperature Overpressurization (LTOP) events.

Concerning the FSAR and core reload analysis events, the most limiting Main Coolant System overpressurization transient is a complete loss of load. This transient is assumed to be initiated via a turbine trip without a direct reactor trip. The Yankee Plant Final Safety Analysis Report, Section 405, provides a description of the current reference analysis for this event.

The loss of load analysis assumed that the Pilot Operated Relief Valve (PORV) failed to open. In addition, several other conservatisms and assumed failures were included in the analysis. These include:

1. The steam generator low level reactor trip was not credited even though it would most likely occur prior to the high MCS pressure reactor trip, and no single failure within the circuitry would preclude a trip signal upon interruption of steam flow from all four steam generators. (Note that the reactor was assumed to trip only on high MCS pressure for the current loss of load analysis.)

2. Neither the condenser nor the Atmospheric Steam Dump System was credited.

3. The effects of pressurizer spray in reducing the primary pressure were neglected (either continuous spray or spray on high MCS pressure).

4. Letdown from the MCS to the Charging and Letdown System was not credited.

5. The heat capacities of the Primary and Secondary Systems' metal were set to essentially zero so as not to provide a heat sink.

6. A 50 psi uncertainty was added to the nominal setpoints of the main steam safety valves.

1. Response (cont.)

7. The setpoints of the two pressurizer code safety valves were increased by 3% over the nominal setpoints in order to support the allowable setpoint tolerance of these valves.

8. A 50 psi uncertainty was added to the nominal setpoint of the high MCS pressure reactor trip.

9. The initial primary pressure was assumed to be at the high end of the pressure cycling band (2,040 psia), plus a 50 psi uncertainty (the assumed initial pressure was 2,090 psia).

As can be seen from the above, the single failure criteria is met and other assumptions in the analysis provide a conservative estimate of the peak pressurizer pressure.

Concerning the inadvertent HPSI event, the August 1, 1982 NRC submittal (Section 3.3, Page 16) states, ". . . the HPSI pump's shutoff head (1,600 psia) is below normal operating pressure. Therefore, this event is of no concern for the YNPS since mass additions to the MCS cannot occur above the HPSI pump shutoff head." This event will not result in the actuation of either a pressurizer relief valve or safety valve.

Concerning the low temperature overpressurization event, the August 1, 1982 NRC submittal states that the most limiting LTOP event is the startup of a single main coolant pump during water solid conditions in the pressurizer. A description of this event can be found in the submittal. LTOP protection is provided by the pressurizer PORV in the low pressure mode with a required setpoint of 500 psig when MCS temperature is less than 380 degrees. Per plant procedure, a steam bubble in the pressurizer is required prior to starting any of the main coolant pumps. Therefore, a procedure violation would be required in order for this event to occur.

The inadvertent startup of a main coolant pump under water solid conditions would result in reverse heat transfer from the associated steam generator, assuming that the steam generator is hotter than the Main Coolant System. The energy addition to the Main Coolant System results in an expansion of the main coolant, thus increasing the primary pressure.

The Shutdown Cooling System relief valves were assumed to fail to operate in this analysis. Further conservative assumptions in the analysis include a main coolant temperature 100^{o} hotter than normal and a main coolant pressure of 400 psia versus 275 psig. The plant Technical Specifications require that the primary system pressure should be below 275 psig until a steam bubble is formed in the pressurizer.

Based on the above, the single failure criteria has been met for the failures that produce maximum loads on the safety and relief valves.

2. Overpressure transients will cause the pressurizer sprays to activate adding moisture to the steam volume. When the safety valves lift or the PORVs are opened they would be passing a steam-water mixture. Provide a discussion on whether this effect was considered in the analysis done to select the transient that produced maximum loads on the discharge piping.

Response:

The effect of water droplets from the pressurizer spray was evaluated in selecting the transients that would produce maximum loads on the discharge piping. The loads on the discharge piping peak prior to the time that the entrained spray reaches the discharge piping. Another way of expressing this phenomena, is the peak loads are caused with the single-phase steam in the safety valve inlet piping, not the subsequent potential spray entrained steam (i.e., the initial pressure wave in the discharge piping caused the peak load). This is also supported by the EPRI/CE safety valve steam/water discharge tests which indicates peak discharge pipe loads occur immediately after the steam actuation, not later during saturated water discharge.

3. The Yankee submittal did not discuss the feedline break event. 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. Discuss the feedline break event providing peak pressure, pressurization rate, temperature, discharge flow rate and expected fluid. Demonstrate safety and PORV functionability for this event, and consideration of this event in the discharge piping analysis.

Response:

The licensing basis for the Yankee plant does not specifically treat a feedwater line break transient. As discussed in the February 1, 1982 NRC submittal concerning SEP topic XV-6, "Feedwater System Pipe Breaks INside and Outside Containment (PWR)", the safety objectives for this transient are contained within the safety objectives of the Main Steam Line break and complete loss of feedwater events. The effects of these transients encompass the feedwater line break event.

Figure 1 shows the schematic diagram for the feedwater system at the Yankee plant. A reverse-flow check valve exists in each of the four normal feedwater lines that provide feedwater to the steam generators. Each feedline also contains a motor-operated gate valve to provide for feedwater isolation, and a flow control valve.

There are two possible cases for the feedwater line break event, depending on the break location. If the break occurs upstream of the reverse-flow check valve in any feedline, the transient will be essentially identical to a complete loss of feedwater event. If the break occurs downstream of the reverse-flow check valve, a single steam generator will blow down, discharging its inventory to the containment atmosphere. This will initially cool and depressurize the Main Coolant System, similar to a Main Steam Line break event. This cooldown will be followed by a primary system heatup on decay heat, until the operators terminate the transient by opening the Emergency Atmospheric Steam Dump valves on the intact steam generators.

For both cases of a feedline break event, the peak pressure, pressurization rate, Main Coolant temperature, discharge flow rate, and fluid conditions at the valves are bounded by the complete loss of load analysis. Since a feedwater line break upstream of the reverse-flow check valves is essentially identical to a complete loss of feedwater event, the requested parameter values are bounded by the loss of feedwater analysis which is, in turn, bounded by the complete loss of load analysis.

For a feedwater line break downstream of the reverse-flow check valve, the heatup portion of the transient is less severe than for a complete loss of load event. Therefore, the peak pressure, pressurization rate, primary temperature, and discharge flow rate are bounded by the loss of load analysis. This assumes that the pressurizer safety and relief valves are relieving dry steam. Therefore, it remains only to show that the fluid conditions at the valves will be steam and not water for a feedwater line break downstream of the reverse-flow check valve.

3. Response: (cont.)

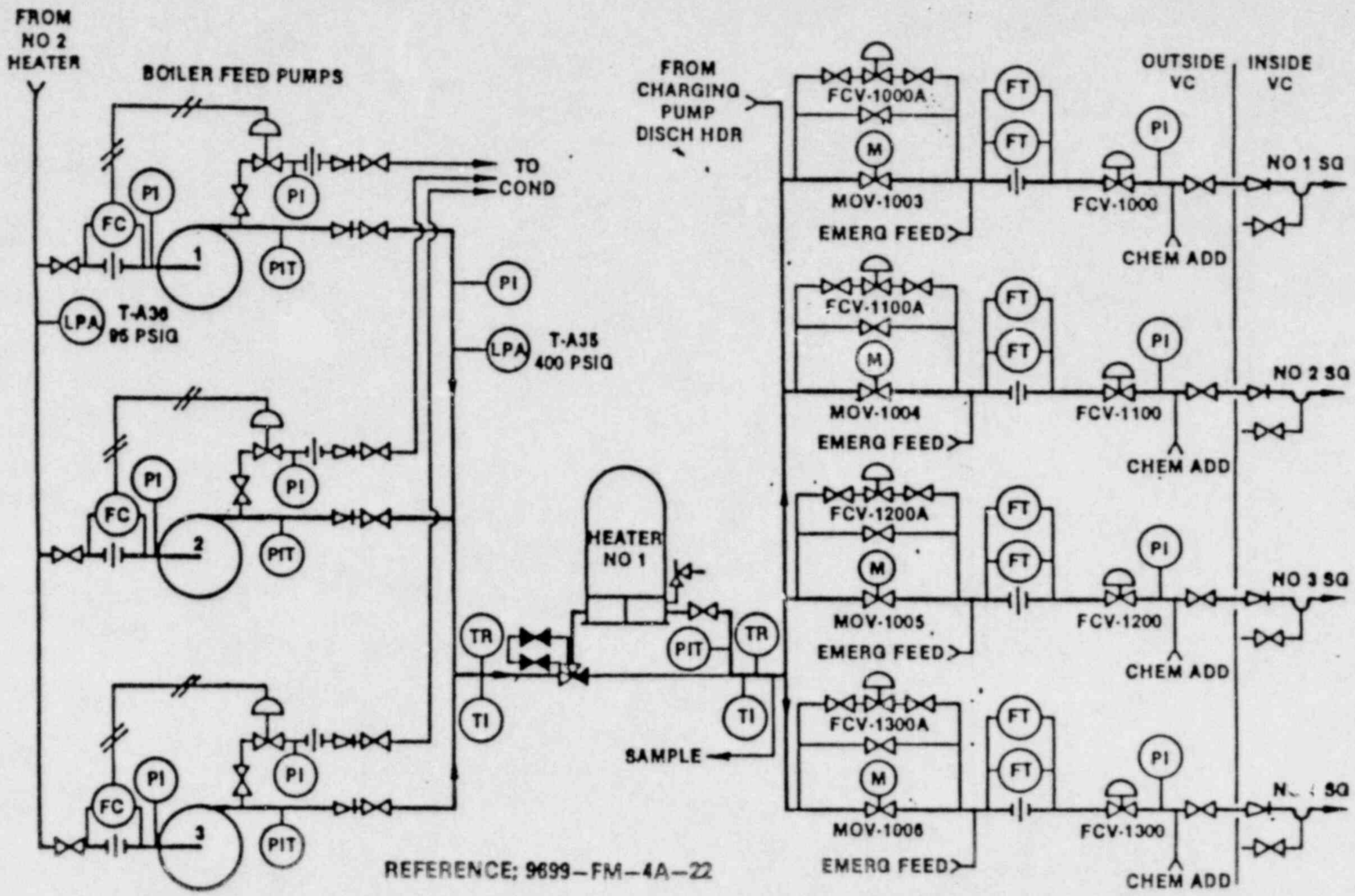
Following a feedwater line break, operator action is assumed to terminate the transient within 15 minutes of its initiation. This is accomplished by opening the Emergency Atmospheric Steam Dump valves on the intact steam generators, which will terminate the primary system heatup. Due to the low shutoff head of the Yankee Safety Injection pumps, there will not be a significant increase in the pressurizer level due to the addition of Safety Injection water to the Main Coolant System following the cooldown portion of the transient. The pressurizer steam volume remaining after the Safety Injection pumps reach their shutoff head will be adequate to accommodate the primary system volume expansion expected until operator action terminates the heatup. Since the pressurizer must be water-solid in order to allow water relief from the safety and relief valves, the fluid conditions at these valves will be dry steam during the heatup portion of a feedwater line break downstream of the reverse-flow check valve.

As shown above, all of the requested parameter values for any feedwater line break case will be bounded by the complete loss of load analysis.

The safety valve and PORV functionability are demonstrated for the feedline breaks since functionality has been demonstrated for a range of fluid conditions that bound the feedline break event.

FIGURE 1

FEEDWATER SYSTEM



REFERENCE; 9699–FM–4A–22

QUESTIONS RELATED TO VALVE OPERABILITY

4. The submittal states that the safety valves are to be replaced by Dresser Model 31719A safety valves. The final safety valve operability report did not provide sufficient detail to allow a complete evaluation of valve operability. Provide additional information that contains:

   (a) The safety valve ring settings in the Dresser safety valves tested in the industry-sponsored program were different from the ring setting in the new Yankee safety valves. Discuss how the ring settings for the Yankee safety valves were determined and the expected performance at these ring settings.

   (b) Provide the expected backpressure on the safety valves and the effects on performance.

   (c) Provide a discussion of the expected blowdown for the 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 and plant safety.

   (d) Verification of valve flow capacity that meets the plant-specific FSAR flow capacity.

   (e) Provide a discussion on the expected stability of the new Yankee 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 acoustic wave component evaluated under steam flow conditions, and then compare to the pressure drops of the EPRI-tested safety valves.

   (f) Provide a copy of the Seismic Qualification Report that contains the structural similarity analysis and operability analysis.

   For (a)-(e) either cite EPRI test data or provide other test data (i.e., copy of the Dresser engineering evaluation on the safety valves) supporting the discussions.

   Response:

   (a) The 31719A safety valve is of the same design series (31700) and is geometrically similar to the safety valves tested during the EPRI test program. Similarity in design of the 31700 series safety valves have been certified by the National Board of Boiler and Pressure Vessel Inspectors in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section III. Certification is found in their publication, "Pressure Relief Device Certifications," 1983 Edition, issued January 1, on Pages 164 and 164a. Although the 31719A safety valve was enveloped by the EPRI test program by design similarity, Yankee further verified valve performance by tests rather than extrapolating from the EPRI test data. In accordance with the design requirements, the only media for qualification of the 31719A safety valve was saturated steam at approximately 2500 psig. Because of capacity limitations of the test facility, full

4. <u>Response</u>: (cont.)

scale steam testing was performed at 1500 psig. As was expected, the results of full scale testing at 1500 psig of the 31719A safety valve parallels the EPRI test data at 2500 psig for the larger 31739A and 31709NA safety valves. The design similarity of the 31700 Series valves is is shown by analyzing adjusting ring positions.

The Lower Adjusting Ring. Valve lift is influenced by the position of the lower adjusting ring. If the ring is too low, full lift may not be obtained. During the EPRI tests for safety valve type 31739A, the position of the lower ring was +11 notches above the valve seat; and for safety valve type 31709NA, the position was flush (0) with the valve seat. During the Wyle tests for safety valve type 31719A, full lift was obtained at 0 and +3 positions. During production testing at Dresser, full lift was obtained at +3 notches at 2500 psig without overpressure on all three valves.

The Middle Adjusting Ring. Blowdown is influenced by the position of the middle adjusting ring relative to the valve seat. Based on Wyle test data and EPRI test data, a position of -24 notches was selected for 31719A safety valve to provide a maximum 12% blowdown against a maximum back pressure of 450 psig. The selected position is geometrically equivalent to the EPRI tested valves as indicated in Table 4.1.

The Top Adjusting Ring. The position of the top adjusting helps to eliminate the effects of back pressure causing valve chatter on closing. The recommended position for all valves of 31700 series is one complete rotation of the ring. This position exposes the vent holes in the guide approximately 1/4 of their diameter. There are 31 notches on the top ring in the 31719A safety valve, 48 notches in the 31739A safety valve, and 50 notches in the 31709NA safety valve. During EPRI test, recommended positioning was used (see Table 4.2). During the Wyle test of the 31719A valve, the top ring was adjusted to several different positions, but best performance occurred at the recommended position.

(b) The expected backpressure is 400-450 psig. The full flow, reduced pressure test was conducted with an orifice in the tailpipe which induced this range of backpressure to the test.

(c) The expected blowdown is 12% maximum for these safety valves. Yankee has issued an Overpressure Protection Report to cover this as required by ASME Code, Section III, Subsection NB7200 (1977 Edition with Winter 1978 addenda). The loss of load analysis was redone for a maximum blowdown of 15%. As a result the fluid condition remains saturated steam and the peak pressure within allowable limits.

(d) The FSAR stated that the required capacity for this valve is 125,000 lb/hr at 2485 psig. The valve has a design flow of 125,000 lb/hr at 2485 psig.

4. Response: (cont.)

(e) The most significant technical finding of the EPRI Test Program was that short inlet piping to the safety valves was the preferred method to insure stable blowdown with full lift. The inlet piping at the Yankee plant with a total length of 14-1/16 inches is among the shortest in the industry. This piping did not require change as a result of the test program.

Rather than depend totally on a calculation to insure stable blowdown, Yankee utilized the knowledge gained on the test program and performed full flow testing at a reduced pressure (1500 psi). Stable blowdown was demonstrated in the tests by utilizing the ring settings discussed in (a) above with inlet and discharge piping modeled to duplicate the plant conditions.

(f) A copy of the statement of Certification of Design by Dresser Industries is attached as Attachment A. A copy of the Seismic Qualification Report is available for your review at the Yankee Atomic Electric Company corporate office in Framingham, Massachusetts.

Table 4.1 Relative Middle Adjusting Ring Position

Valve Type	Adjustment per Ring Notch (inches)	Number of Notches Below Seat	Relative Position (inches)
31719A	.0020	24	0.048
31739A	.0015	32	0.048
31709NA	.0012	40	0.048

Table 4.2 Blowdown Comparison

Valve Type	Orifice sq. in.	Test No. (run)	Ring Positions Lower, Middle, Top			% Blowdown	Peak Back Pressure, psig
31719A	0.994	19	+3	-24	-31	11.6	400
31739A	2.545	322	+11	-40	-48	11.1	609
31709NA	4.340	614	0	-40	-48	8.8	354

5. The submittal identifies the Yankee PORV as a Dresser 2-1/2 - 31533 VX. The PORV tested by EPRI was a Dresser Model 31533VX-30-2 with a bore diameter of 1-5/16 in. Discuss the differences between the two valves, also provide the bore diameter for the Yankee PORV. If the plant PORV design and bore diameter are different from the tested PROV bore diameter, discuss the effect on performance due to the different design features and bore diameter and how the EPRI data can be interpolated to verify PORV operability.

<u>Response</u>:

The Yankee plant has one Dresser PORV (Code No. 2-1/2-31533VX-X-2-XNC072). The corresponding bore diameter is 7/8". The EPRI tested PORV used a Dresser Model 2-1/2-31533VX-30-2, with a corresponding bore diameter of 1-5/16". Dresser Industries designed this model such that bore diameter/orifice size affects valve capacity, but not performance, since the valve functions as a result of pressure ratios rather than an absolute valve of pressure.

The Yankee PORV has the -2 intervals which Dresser intended for all PORVs, which previously had the -1 intervals. Among other features, the -2 intervals improved the seat tightness. The -2 intervals were tested during the EPRI tests.

A second difference in the tested PORV from the Yankee PORV is the inclusion in the tested valve of a pilot stem bellows which helps prevent steam leakage past the pilot stem after PORV opening. The absence of the pilot bellows has no effect on PORV operability.

The EPRI test results obtained from the Model 31533VX-30 valve are fully representative of the Model 31533VX valve performance since the valves are essentially identical from a functional standpoint. Consequently, since the EPRI tests on Dresser 2-1/2-31533VX-30-2 demonstrated that the PORV will open and close on demand under high pressure steam conditions, it can be expected that the PORV can perform its function of preventing the primary safety valve from opening during high pressurization transients. It should be noted that there are no Yankee transients or accidents that result in high pressure water discharge from the valve.

6. Dresser Industries transmitted a letter in March 1976 to Metropolitan Edison Co. warning that the PORV block valve should be kept closed when the reactor coolant system pressure is below 1000 psig to avoid damaging the PORV (Model 31533VX-30-1) disk and seat by steam cutting. The EPRI program data indicates that the Dresser PORV was successfully tested on water at pressures in the 500-900 psig range. Steam testing at lower pressures was not performed. The recommendation made by Dresser that the PORV be isolated at pressures lower than 1000 psi would seem to preclude the use of the PORV for low temperature overpressure protection of the reactor vessel. Provide additional information concerning the Dresser recommendation as it applies to Yankee. Explain whether the Dresser recommendation or a modification of it will be followed, or a modification to the PORV will be performed to prevent damage to the disk and seat, or provide additional test detail performed since March 1976 that demonstrates such precautions are unnecessary.

Response:

The Dresser Industries' concern in the March 1976 letter to Metropolitan Edison Company, about the Model 31533VX-30-1 PORV is not applicable to the Yankee PORV. The -2 intervals were designed to resolve the referenced concerns. The Yankee PORV has the -2 intervals as evidenced by the valve code number (2-1/2-31533VX-X-2-XNC072) to prevent damage to the disk and seat during LTOP transients.

7. In valve operability discussions on cold overpressurization transients, the submittal only identifies conditions for water discharge transients. Although Yankee was not a participant in the EPRI test program, the valve inlet fluid conditions report for Westinghouse designed plants prepared by Westinghouse for EPRI states that 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. To assure that the PORVs operate for all cold overpressure events, discuss the range of fluid conditions expected for the expected types of fluid discharge and identify the test data that demonstrates operability for these cases. Since no low pressure steam tests were performed on the PORVs, confirm that the high pressure steam tests demonstrate operability for the low pressure steam case for both opening and closing of the PORV.

<u>Response</u>:

The PORVs at Yankee are used to protect the reactor vessel from Low Temperature Overpressurization (LTOP) transients during heatup or cooldown. The lowest PORV setpoint required to provide this protection is 500 psig.

Dresser has run a test program to determine the operability of their PORVs at low pressure. The test data indicates that the PORV operates properly at any pressure above 100 psig. Therefore, there are not any operability problems expected during LTOP events.

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

Response:

The control of the PORV is from two separate sources. During power operation, the control is from the Reactor Protection System. Under LTOP conditions, the valve setpoint is at 500 psig with manual operation from the Main Control Board. The Reactor Protection System is designed to the FSAR requirements. The LTOP System meets the requirements of IEEE-323-1974.

Except for the PORV solenoid, the control circuit is outside of the containment and is not exposed to design basis transients and accidents. The PORV solenoid is not qualified for accident environmental conditions. However, failure of the solenoid cannot result in a spurious operation nor can it prevent an open valve from closing.

9. 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. The mounting configuration of the Yankee PORV block valve is vertical. Discuss the effects of the vertical installed block valve configuration on valve operability and reliability.

Response:

The PORV block valve is a 2-inch Pacific Figure No. 1550 (1500 lb) gate valve. The motor operator sizing by the manufacturer provided calculations to determine the required closure force. The force required is a combination of frictions and the force required to overcome the pressure drop across the seat and stem. The calculation is independent of orientation because the components of the total force are independent of orientation. Therefore, the valve can be expected to operate as well vertically as it would horizontally.

10. The Yankee plant PORV block valve and actuator were not tested by EPRI. The Yankee submittal has stated that the plant-specific PORV block valve, a 2-inch Pacific electric motor actuated gate valve, was similar to the Velan valves tested by EPRI. Other valve manufacturers were also represented. Provide additional discussion and detail on how the plant-specific block valve was determined to be similar to the Velan block valves. Also provide a discussion and detail that demonstrates applicability of the Limitorque SMA-00-10 actuator to the EPRI-tested actuators SMB-00-15 and SMB-000-10. Discuss how the EPRI PORV block valve test results were extrapolated to demonstrate Yankee PORV block valve operability over the range of expected operating and accident conditions, and how the requirements of NUREG-0737, Item II.D.1 have been met. Provide a copy of the valve manufacturer's report on the review of the actuator torque requirements.

Response:

The Yankee PORV block valve is a Pacific 2-inch Figure No. 1550 (1,500 lb. ANSI Rating) gate valve. It is very similar to the 3" Velan valve tested at the Marshall station. Both valves are stainless steel, double disc/solid wedge with stellited seats. As such their function and design are similar.

The Marshall steam tests used Limitorque SMB-00-15 and SMB-000-10 motor operated actuators. The function of the actuator is to provide sufficient force to overcome the pressure drop across the valve. The Pacific Valve's sizing letter provides the minimum required thrust (55.9 ft.lbs.) to accomplish closing under a 2500 psig pressure drop. The actuator has been set to provide 10% conservatism to assure closure. Further conservatism is evidenced by the fact that the PORV is set to lift at 2350 psig and reseat at 2300 psig, not the 2500 psig value used in the operator sizing letter.

11. Bending moments are induced on the safety and PORVs during the time they are required to operate because of discharge loads and thermal expansion of the pressurizer tank and inlet piping. Discuss the predicted plant moments and demonstrate that the operability of the valves will not be impaired.

Response:

The final maximum bending moment for the thermal, deadweight and pressure loads is 558 ft-lbs. The combined moment including seismic and transient loads is 1199 ft-lbs. Bending moments were minimized in the Yankee design for operational reasons. These resulting moments are considerably less than the applied EPRI bending moments which were measured near 20,000 ft-lbs with no adverse operability effects.

We further had Dresser Industries analyze the Stone & Webster calculated total structural loads. They found adequate margins for both normal plant operation and the combined thermal, deadload, seismic and discharge transient loads. The maximum combined moment (SRSS, three directions) including all transients analyzed for the SRVs is 1199 ft-lbs.

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

12. The submittal states that a thermal hydraulic analysis of the safety/relief valve piping system has been conducted, but does not present details 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 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 time step, valve flow area, peak pressure and pressurization rate, node spacing, choked flow junctions, valve opening time, the 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 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.

Provide a copy of the contractor's piping thermal hydraulic analysis report.

<u>Response</u>:

(a) Two spring-loaded code safety valves and one solenoid operated relief valve are provided to accommodate the pressure surges which exceed the pressure limiting capacity of the pressurizer Safety and Relief Valve Piping System. These valves are all located such that water cannot condense and form a seal in the inlet lines. Under reduced temperature operation, additional overpressure protection is

12. Response: (cont.)

provided by the low pressure setpoint which is in the range of 485 psig. This setpoint provides Low Temperature Overpressure Protection (LTOP). Four fluid transient cases described as follows are considered:

Case 1: One relief valve open to discharge steam

Case 2: Two safety valves open to discharge steam

Case 3: Two safety valves open to discharge with relief valve already open

Case 4: LTOP relief valve open to discharge water

Maximum loads from the above cases are used for design.

(b) The safety/relief valve discharge loadings (the forcing functions) are developed utilizing SWEC's in-house computer programs STEHAM and WATAIR. Attachment B provides a general description of the computer programs utilized for the development of the forcing functions. These programs were verified by using experimental data or hand calculations. The Qualification Manuals are on file at SWEC and may be reviewed at SWEC offices in Boston.

The STEHAM Program was utilized to develop force time histories for discharge piping due to the steam discharge of each safety/relief valve. Three fluid transient cases, i.e., relief valve opening, safety valve opening with relief valves closed, and safety valve opening with relief valve already open, were analyzed. The two safety valves were conservatively considered to open simultaneously.

The WATAIR Program was utilized to develop the fluid forcing functions for the potential case of water discharge through relief valves (low temperature cold overpressurization LTOP). The program handles only one-dimensional flow networks with a uniform pipe diameter. Based on engineering judgment a conservative flow diameter was established for modeling the piping system while keeping the other parameters such as the piping segment lengths and flow rate as given.

(c) Table 1 identifies the important parameters used in the thermal hydraulic analysis except time steps, node spacing, and choked flow junctions. Time steps used in the STEHAM Program are determined by the node spacing and speed of sound to satisfy the Courant stability criterion. Since the minimum node spacing used was about 1.4 ft., the time steps were in the order of 1 millisecond. The time step used in the WATAIR analysis was about 1 millisecond as well. Choked flow junctions usually occurred at area changes such as at a reducer, valve, or common header. These will be detected by the STEHAM Program, as appropriate.

12. Response: (cont.)

(d) The safety and relief valves are treated as a convergent-divergent nozzle in STEHAM analysis with choked or non-choked flow detection. The relief valve is treated as orifice in WATAIR analysis. The flow rate and valve resistance (discharge or loss coefficients) are reported in Table 1. ASME code derating was considered in the analysis.

(e) The two spring-loaded code safety valves were assumed to have the same set pressure of 2485 psig, such that simultaneous opening is considered for the thermal hydraulic analysis. The actual setpoints are 2485 psig and 2560 psig respectively.

(f) Thermal hydraulic models for STEHAM and WATAIR are shown in Figures 1 and 2. Control volume modeling technique was not used in the analysis. The branch, node, segment numbers along with forcing function direction are shown on these figures. Piping dimensions are delineated in Table 2.

YANKEE ROWE PSARV PIPING

STEHAM COMPUTER MODEL

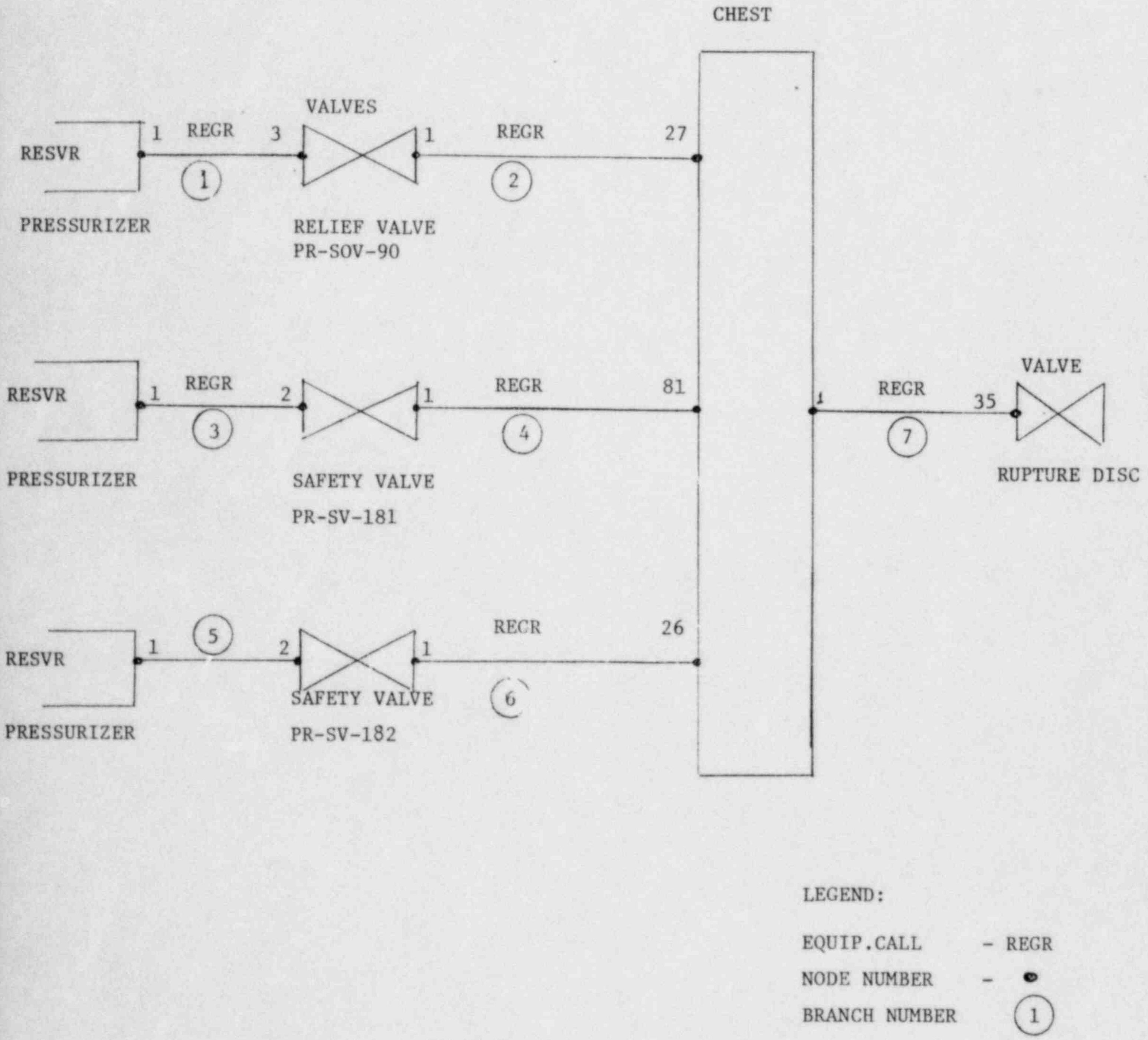
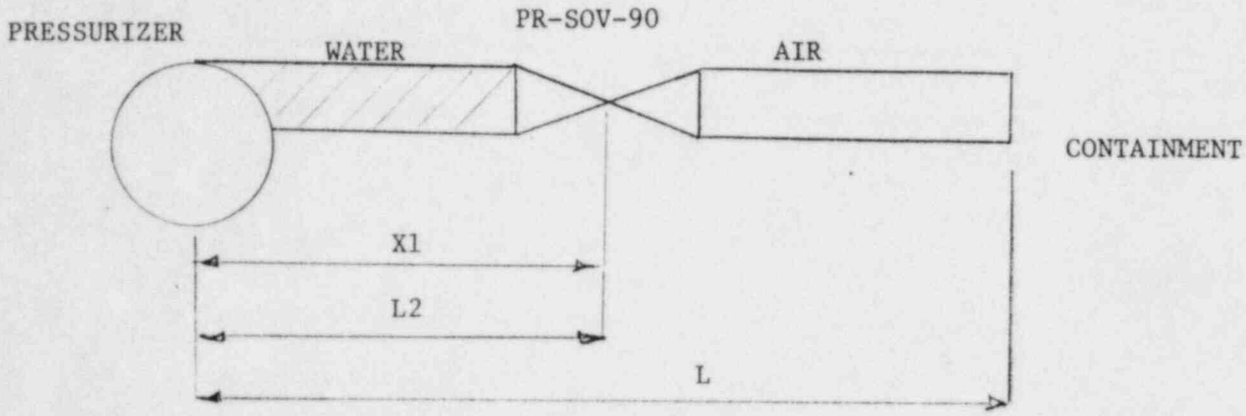


FIGURE 1

WATAIR COMPUTER MODEL AND INPUT DATA



INITIAL LENGTHS

L = Total pipe length = 89.05 ft

L2 = Position of relief valve = 3.73 ft

IX1 = Initial position of water-air interface = 3.73 ft[1]

PIPE PARAMETERS

D = Diameter of pipe = 0.1407 ft

F = Darcy friction factor for pipe = 0.019

KLV = Loss due to valves and fittings upstream of relief valve = 2.12

KLU = Loss due to valves and fittings downstream of relief valve = 3.25

VALVE CHARACTERISTICS

A_{vo} = Valve orifice area = 0.0042 ft^2

T_v = Valve opening time = 0.10 sec

C_{kk} = Valve loss coefficient = 38.

FIGURE 2

TABLE 1 (Question 12)

FLUID CONDITIONS AND CRITICAL PARAMETERS
USED FOR THE ANALYSIS

	Case 1	Case 2	Case 3	Case 4
Transient Case	1 relief valve open discharge steam	2 safety valves open simultaneously discharge steam	2 safety valves open simultaneously discharge steam with 1 relief valve already open	LTOP 2 relief valves water, water fronts combine at common piping
Computer Code Used	STEHAM	STEHAM	STEHAM	WATAIR
Valve Opening Time	0.1 sec (1)	0.015 sec (1)	0.015 sec (1)	0.1 (1)
Flow Rate/Valve	78,400 lb/hr steam at 2400 psig (1)	130,000 lb/hr steam at 2485 psig (1)	130,000 lb/hr steam at 2485 psig (1)	132,000 lb/hr water at 520 psig
Valve Set Pressure	2400 psig (2)	2485 psig (2)	2485 psig (2)	485 psig (2)
Peak Pressure	2490 psig (2)	2490 psig (2)	2490 psig (2)	523.3 psig (2)
Pressurization Rate	34.2 psi/sec (2)	34.2 psi/sec (2)	34.2 psi/sec (2)	13.3 psi/sec (2)
Back Pressure	13.7 psig (3)	13.7 psig (3)	13.7 psig (3)	13.7 psig (3)
Flow Condition	Steam	Steam	Steam	Water

TABLE 1 (Question 12) (Cont)

	Case 1	Case 2	Case 3	Case 4
Discharge Coefficient	1.02	0.987	0.987	-
Valve Loss Coefficient	-	-	-	38

Relief Valve Bore Diameter is 7/8 in.
Safety Valve Area is 0.994
Note: numbers in ( ) is reference provided to SWEC by YAEC.

TABLE 2

STEAM PIPING DIMENSION

Branch	Segment No.	Segment Length (Ft)	Total Length (Ft)	N Nodes	Nodal Length Total/N-1 (Ft)
1	1	1.25	3.73	3	1.865
	2	2.48			
2	1	1.48	36.94	27	1.42
	2	1.50			
	3	7.25			
	4	3.56			
	5	4.75			
	6	7.32			
	7	2.58			
	8	8.50			
3	1	.75	2.14	2	2.14
	2	1.39			
4	1	2.04	30.30	21	1.52
	2	12.36			
	3	6.83			
	4	7.07			
	5	2.0			
5	1	.75	2.14	2	2.14
	2	1.39			
6	1	3.88	38.43	26	1.54
	2	3.41			
	3	13.35			
	4	8.58			
	5	6.57			
	6	2.71			
7	1	33.18	48.38	35	1.423
	2	10.12			
	3	5.08			

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

13. The submittal states that a structural analysis of the safety/PORV valve piping system has been conducted, but does not present details 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 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) Identification of important parameters used in the structural analysis and rationale for their selection. These include lumped mass spacing, time step in the applied fluid forces, solution time step, damping and cut off frequencies (if applied).

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

(e) 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.

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

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

Provide a copy of the contractor's piping structural analysis report.

Response:

(a) The structural analysis of the upstream and downstream piping due to safety and relief valve discharge was performed using SWEC's NUPIPE-SW computer program which performed an elastic time-history evaluation of the three-dimensional piping systems.

The NUPIPE computer code was orginally developed by Nuclear Service Corporation (NSC), a division of Quadrex Corporation, and has been available for public use through Control Data Corporation's (CDC) Cybernet services since late 1976. Non-exclusive rights to use of

13. <u>Response</u>: (cont.)

this program, along with verification and qualification documentation was purchased by Stone & Webster Engineering Corporation (SWEC) in 1973. This program was given a name NUPIPE-SW to differentiate the SWEC version of the program from the NSC version.

Since acquisition of the program, numerous updates and improvements in the program have been made resulting in new versions and levels of the computer program. Each new version and level of the computer program is verified by comparison to previous versions and levels and/or to NRC benchmark problems from NUREG-CR 1677, Reference 1. The verification of results are documented in qualification manual filed at the SWEC computer library and are available for inspection at the offices of SWEC in Boston.

Further, as part of verification requirements stated in IE Bulletin No. 79-07 (Reference 2), NRC has independently verified the NUPIPE-SW computer program to be acceptable (Reference 3).

The time-history dynamic analysis portion of the NUPIPE-SW Program has not been modified since the acquisition of the program and, therefore, qualification of the time-history dynamic analysis method is covered under NSC qualification/verification document. The method of qualification was by comparison to ASME Benchmark Problem No. 5 from Reference 4.

(b) The methodology used to calculate the forcing functions is described in the response to Question 12. The thermal-hydraulic forces developed for closed pipe segments (e.g., bounded by two elbows) are wave or net forces. These are time-dependent segment forces which are applied to the dynamic model so that the line of action is along representative piping segments. The points of application of these forces are preferably located at the beginning or ending point of each pipe segment and away from support nodes. Nodal application of forcing functions is done recognizing that these forces tend to excite axial modes of vibration.

"Blowdown Forces" or nodal forces are considered for open-ended piping which will experience a steady-state type force at the exit point. This load can be applied as a static force, and is in addition to the time-dependent segment force analysis.

(c) The parameters for the time history analysis are selected to ensure sufficient accuracy and dynamic stability of the solution to the dynamic analysis of piping for fluid transient loadings. Typically, the model will contain at least three mass points between restraints active in the same direction. The piping geometry and large number of supports on the pressurizer safety and relief valve piping typically results in many closely spaced mass points along straight runs of pipe.

13. <u>Response</u>: (cont.)

The cut-off frequency and mode are selected by a review of the piping geometry and system response characteristics recognizing the fact that the typical modes of excitation in this analysis are the higher frequency axial modes. The total analysis time and integration time steps for the analysis are selected based on a review of the input forcing function and to ensure a stable solution. The total analysis time is selected to allow sufficient time for the piping system to respond to the input forces and for the responses to begin to damp out.

The following analysis parameters were used for the bounding case of steam discharge through the relief valves:

Forcing Function Time Step	0.001 sec.
NUPIPE-SW Analysis Time Step	0.001 sec.
Cut-off Frequency	400 Hz
Damping	1%

(d) The dynamic model is a lumped mass three-dimensional representation of the actual installation. Supports as well as piping are modeled in their true orientation which can either be coincident with global axes or skewed. Supports as well as connection to the pressurizer and relief tank are modeled as elastic springs in the NUPIPE-SW piping model.

The valves are input in the analysis as a rigid member and the weight of the vale as well as the weight of the actuator are input as concentrated weights located at the center of gravity of the valve and the actuator.

(e) The piping stress analysis for the Yankee Rowe pressurizer safety and relief valve piping was performed in accordance with the ANSI B31.1 Code for Power Piping, 1977 Edition (Reference 5). The load combination and allowable stress limits for the safety and relief valves discharge condition are consistent with ASME Section III, 1977 Edition.

$P + DL \leq 1.0\ S_h$

$P + DL + OCC \leq 1.2\ S_h$

$P + DL + SRSS\ [(OCC,\ SSE\ (I)] \leq K\ S_h$

$THERMAL \leq S_A$

where

P = Longitudinal pressure stress

DL = Deadload stress

13. Response: (cont.)

SSEI = Seismic Stress due to Safe Shutdown Earthquake Inertia

OCC = Occasional load

S_c = Basic material allowable stress at room temperature

S_h = Basic material allowable stress at maximum operating temperature

S_A = $(1.25\ S_c + 0.25\ S_h)$

K = 1.8 UPSTREAM OF VALVE
= 2.4 DOWNSTREAM OF VALVE

The occasional load in the above equations are the maximum of:

a) Low temperature cold overpressurization (potential case of water discharge through relief valves).

b) Steam discharge through relief valve

c) Steam discharge through safety valve

d) Steam discharge through safety valve with relief valve already open

The occasional load is combined by the square root sum of the squares technique with the SSEI earthquake.

Pipe supports were analyzed and/or designed by Teledyne using the American Institute of Steel Construction (AISC) 7th Edition. A 1/3 increase in allowable stress was used for the seismic loading condition. Concrete expansion anchors were designed using a factor of safety of 4.0.

(f) Table 1 shows the maximum stress level in the piping system. The results indicate that the maximum stress level in the piping system meet the code allowable limits for all loading conditions.

(g) Figure 1 shows the mathematical model utilized for the structural analysis of the piping system. The mathematical model shows the lumped mass point locations (identified by △) and pipe sizes. The fluid forces are applied at all elbows but not indicated on the figure.

TABLE 1

STRESS LEVEL - MAXIMUM

EQUATION	POINT NO.	CALCULATED STRESS PSI	ALLOWABLE STRESS PSI
$P + DL^{(1)} < 1.0\ S_h$	305	8,353	15,900
$P + DL + OCC. < 1.2\ S_h$	490	18,219	19,080
$P + DL + SRSS\ [(OCC^{(1)}, SSE\ (I)] < K^{(2)} S_h$	305	19,558	28,620
$THERMAL < S_A$	530	10,110	27,350

NOTES:

(1) OCCASSIONAL = MAXIMUM OF

(a) LTOP THROUGH RELIEF VALVES

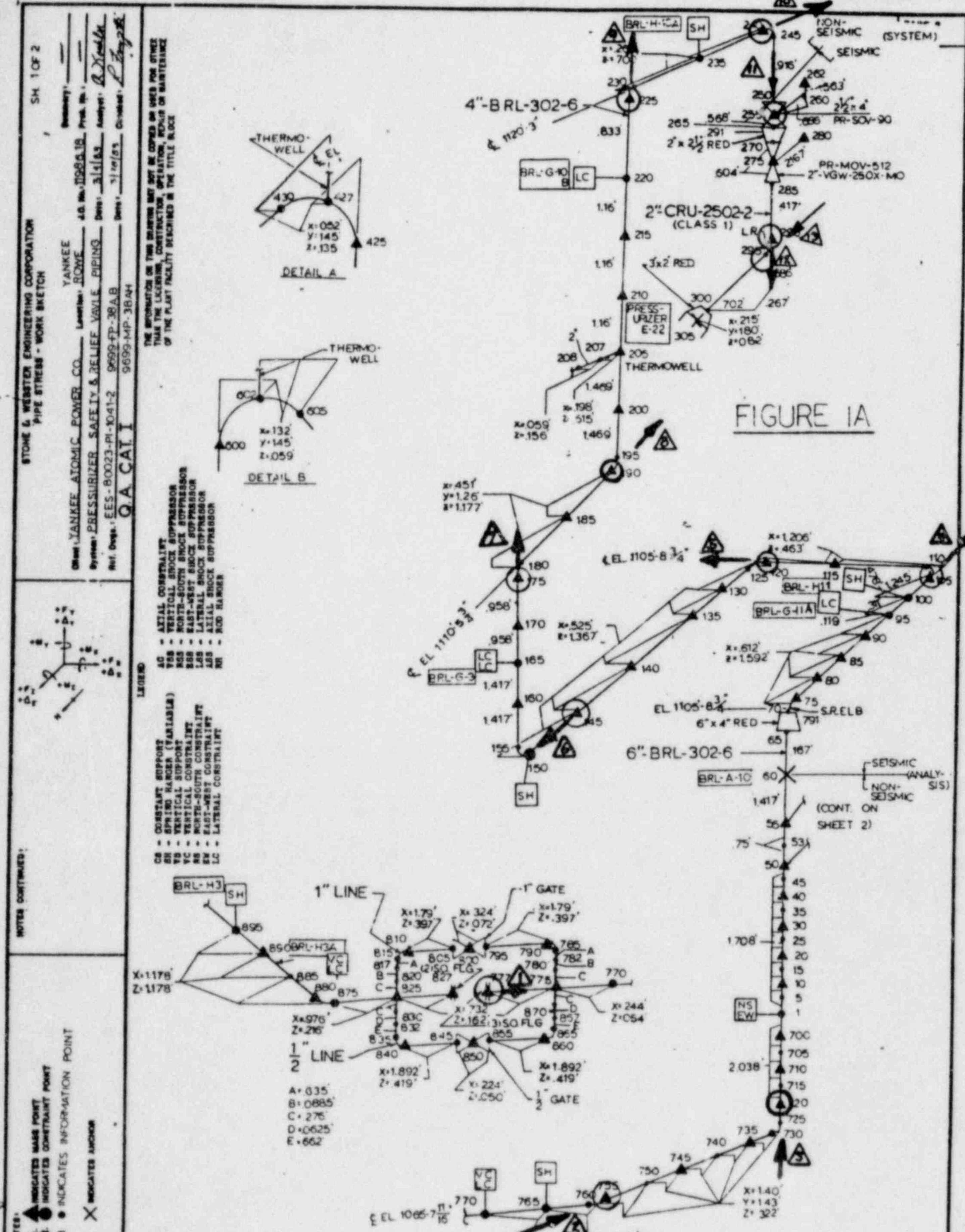
(b) STEAM DISCHARGE - SAFETY VALVE

(c) STEAM DISCHARGE - RELIEF VALVE

(d) STEAM DISCHARGE THROUGH RELIEF VALVE FOLLOWED BY SAFETY VALVES

(2) K = 1.8 UPSTREAM OF VALVE
= 2.4 DOWNSTREAM OF VALVE

(3) The pipe stresses and support loads for the relief valve are significantly smaller for the LTOP water discharge case when compared to the high pressure steam case.



STONE & WEBSTER ENGINEERING CORPORATION
PIPE STRESS - WORK SKETCH
SH. 1 OF 2
Client: YANKEE ATOMIC POWER CO.
Location: ROWE
System: PRESSURIZER SAFETY & RELIEF VALVE PIPING
Ref. Dwgs.: EES-800023-PI-1041-2 9699-FP-38A,B 9699-MP-38AH
Q.A. CAT. I
THE INFORMATION ON THIS DRAWING MAY NOT BE COPIED OR USED FOR OTHER THAN THE LICENSING, CONSTRUCTION, OPERATION, REPAIR OR MAINTENANCE OF THE PLANT FACILITY DESCRIBED IN THE TITLE BLOCK
LEGEND
CS : CONSTANT SUPPORT
SH : SPRING HANGER (VARIABLE)
VS : VERTICAL SUPPORT
VC : VERTICAL CONSTRAINT
NS : NORTH-SOUTH CONSTRAINT
EW : EAST-WEST CONSTRAINT
LC : LATERAL CONSTRAINT
AC : AXIAL CONSTRAINT
VSS : VERTICAL SHOCK SUPPRESSOR
NSS : NORTH-SOUTH SHOCK SUPPRESSOR
EWS : EAST-WEST SHOCK SUPPRESSOR
LSS : LATERAL SHOCK SUPPRESSOR
ASS : AXIAL SHOCK SUPPRESSOR
RH : ROD HANGER
NOTES:
INDICATES MASS POINT
INDICATES CONSTRAINT POINT
INDICATES INFORMATION POINT
INDICATES ANCHOR
NOTES CONTINUED:
4"-BRL-302-6
6"-BRL-302-6
2"-CRU-2502-2 (CLASS 1)
NON-SEISMIC
SEISMIC
(SYSTEM)
PR-SOV-90
PR-MOV-512
2"-VGW-250X-MO
2" x 2½ RED
3"x2" RED
6"x4" RED
S.R.ELB
PRESSURIZER E-22
THERMOWELL
DETAIL A
DETAIL B
SEISMIC (ANALYSIS)
(CONT. ON SHEET 2)
1" LINE
1" GATE
½" LINE
½" GATE
BRL-H-13A
BRL-G-10
BRL-G-3
BRL-H11
BRL-G-11A
BRL-A-10
BRL-H3
BRL-H3A

FIGURE 1A

NOTES:
1. ▲ INDICATES MASS POINT
2. ● INDICATES CONSTRAINT POINT
3. • INDICATES INFORMATION POINT
4. X INDICATES ANCHOR

NOTES CONTINUED:

STONE & WEBSTER ENGINEERING CORPORATION
PIPE STRESS - WORK SKETCH

SH. 2 OF 2

Client: YANKEE ATOMIC POWER CO.	Location: YANKEE ROWE	J.O. No.: 11986.18	Prob. No.: —
System: PRESSURIZER SAFETY & RELIEF VALVE PIPING	Date: 3/4/85	Analyst: [signature]	Summary: —
Ref. Dwgs.: LES-80023-PI-1041-2 9699-FP-38A,B 9699-MP-3RAH	Date: 3/10/85	Checked: [signature]	

Q.A. CAT. I

THE INFORMATION ON THIS DRAWING MAY NOT BE COPIED OR USED FOR OTHER THAN THE LICENSING, CONSTRUCTION, OPERATION, REPAIR OR MAINTENANCE OF THE PLANT FACILITY DESCRIBED IN THE TITLE BLOCK

LEGEND

CS - CONSTANT SUPPORT	AC - AXIAL CONSTRAINT
SH - SPRING HANGER (VARIABLE)	VSS - VERTICAL SHOCK SUPPRESSOR
VS - VERTICAL SUPPORT	NSS - NORTH-SOUTH SHOCK SUPPRESSOR
VC - VERTICAL CONSTRAINT	ESS - EAST-WEST SHOCK SUPPRESSOR
NS - NORTH-SOUTH CONSTRAINT	LSS - LATERAL SHOCK SUPPRESSOR
EW - EAST-WEST CONSTRAINT	ASS - AXIAL SHOCK SUPPRESSOR
LC - LATERAL CONSTRAINT	RH - ROD HANGER

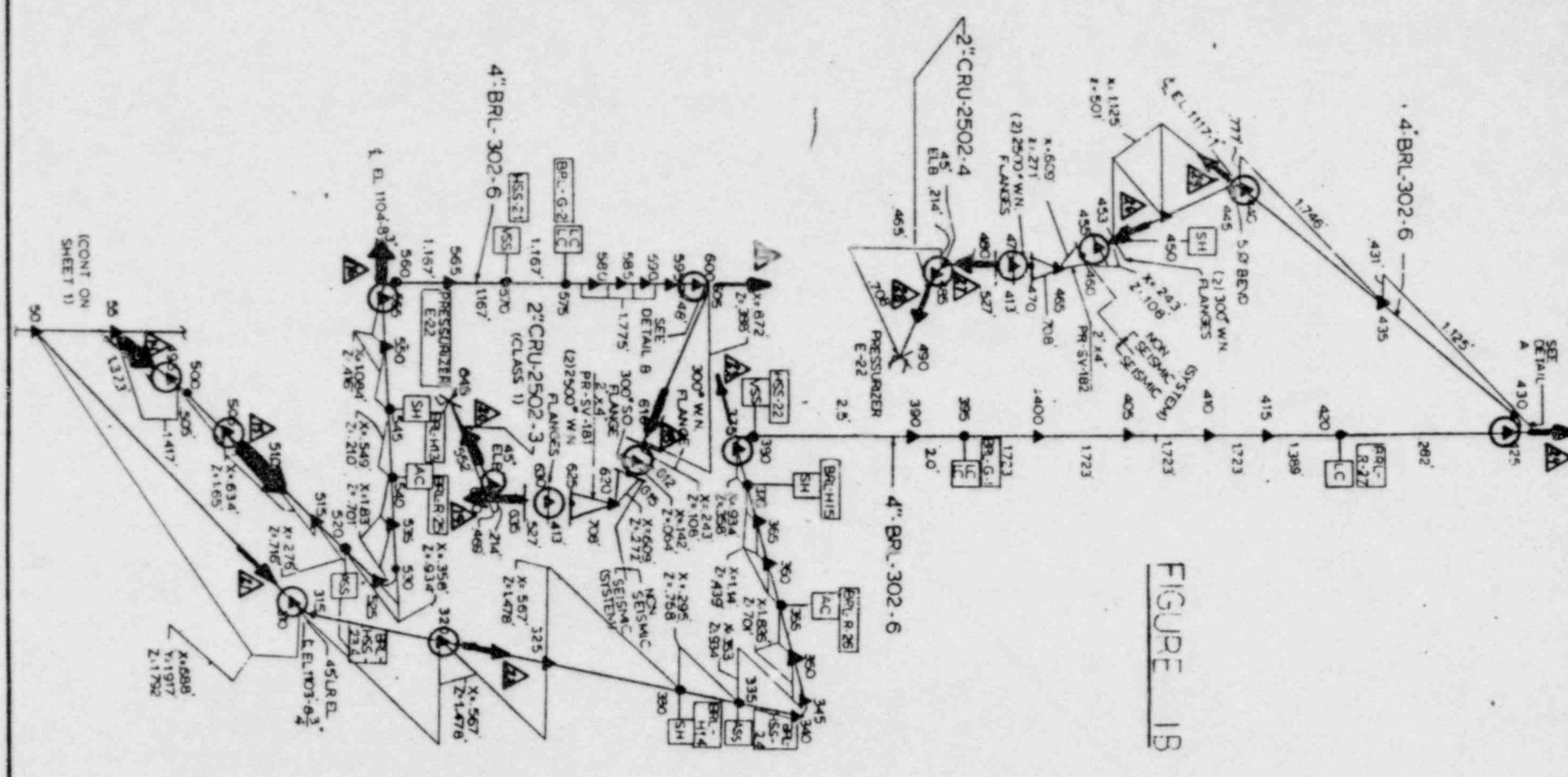


FIGURE 1B

ATTACHMENT B TO QUESTION 12(b)

STEHAM

STEHAM is a computer program which is used to determine the steamhammer transients of piping systems. This program uses the method of characteristics with finite difference approximation both in space and in time. It calculates the one-dimensional transient flow responses and the flow-induced forcing functions in a piping system caused by rapid operational changes of piping components, such as the actuation of a stop valve or the safety/relief valve. Unbalanced fluid forces on the pipe segments are computed by integrating the rate of change of the fluid momentum within that control volume. Flow characteristics of piping components are mathematically formulated as boundary conditions in the program.

These components include the flow control valve, the stop valve, the safety/relief valve, the steam manifold, the steam reservoir, and the discharge pipe with water slug. STEHAM also considers frictional effects.

Program input consists of program control data, the flow network representation of the piping system, piping data, initial conditions, and time dependent flow characteristics of the piping components. The component models already exist in the program, but the user must supply the specific operational characteristics for each of the models used.

Program output consists of time values of flow pressure, density, velocity, nodal forces for all nodes, and segment forces for all segments of the flow network at each time increment. The dynamic forcing functions are then stored on tape for direct input to NUPIPE (ME-110) Program.

WATAIR

The purpose of WATAIR is to determine forcing functions induced on piping systems by a hydraulic transient with trapped air (such as water discharging into empty piping) for subsequent input to piping dynamic analysis.

The analysis uses rigid column theory to calculate fluid accelerations and velocities, the first law of thermodynamics and the ideal gas law to calculate air properties, including pressure, and the control volume theory to calculate the unbalanced fluid forces on the pipe segments.

The input to the program consists of dimensions of the piping system, frictional coefficients of the pipes, valves and fittings, valve characteristics, and the operating conditions of the flow network.

The output consists of inertial segment forces, positions and pressure of the air pocket, velocities, and accelerations of the water slugs at indicated time increments for subsequent input to NUPIPE-SW for piping dynamic analysis.

REFERENCES

1. Stone & Webster Engineering Corporation Computer Program NUPIPE-SW.

2. U.S. NRC NUREG/CR-1677 "Piping Benchmark Problems, Dynamic Analysis Uniform Support Motion Response Spectrum Method" dated August 1980.

3. NRC I&E Bulletin 79-07 "Seismic Stress Analysis of Safety-Related Piping" dated April 4, 1979.

4. NRC Letter to SWEC "Benchmark Verification of the S&W Computer Codes PSTRESS/SHOCK 3 and NUPIPE-SW" (copy attached).

5. USAS B31.1 Code for Pressure Piping, 1977 Edition.