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May 31, 1985
MN-85-97

GDW-85-151

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. James R. Miller, Chief
Operating Reactors Branch No. 3
Division of Licensing

References:
(a) License No. DPR-36 (Docket No. 50-309)
(b) USNRC Letter to MYAPCo dated February 11, 1985 - Request for Additional Information on NUREG-0737, Item II.D.1 - Performance Testing of Relief and Safety Valves
(c) MYAPCo Letter to USNRC dated March 19, 1985 (MN-85-57) Performance Testing of Relief and Safety Valves

Subject: Response to Request for Additional Information on Relief and Safety Valve Testing (NUREG-0737, Item II.D.1)

Gentlemen:

Enclosed are the answers to the questions proposed in Reference (b) above. We believe the answers are full and complete, however, if you need additional information, please feel free to contact us.

Very truly yours,

MAINE YANKEE ATOMIC POWER COMPANY

G. D. Whittier

G. D. Whittier, Manager
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GDW/bjp

Enclosure:

cc: Dr. Thomas E. Murley
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ENCLOSURE

QUESTIONS RELATED TO VALVE OPERABILITY

NRC Question #1

1. The Combustion Engineering Report on operability of PORVs in CE plants indicated that the limiting inlet fluid condition during low temperature pressurization transients is a water discharge event. The CE Inlet Fluid Conditions Report stated that the pressurizer water solid condition and resulting PORV liquid discharge case were chosen for the cold overpressurization event since they gave the most severe pressurization transients. The report further states that a steam bubble can also exist in the pressurizer during low temperature operation, whereby the PORC could lift on steam. No low pressure steam tests were performed by EPRI on the Dresser PORV. Provide verification that the Maine Yankee PORV will operate satisfactorily on low pressure steam.

Maine Yankee Response

The PORVs at Maine 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 250 psig.

Dresser has run a test program to determine the operability of their PORVs at low pressure. Reference 1 documents the results of this testing on a PORV similar to those used at Maine Yankee. The test data indicates that the PORV operates properly at any pressure above 100 psig; therefore, no operability problems are expected during LTOP events.

NRC Question #2

2. The Maine Yankee nuclear plant utilizes a Dresser 31533VX-30 PORV valve. The model number indicates that the valve contains the older obsolete internals. Most plants using this valve have upgraded their valves to the Type 2 internals. The EPRI tests were conducted with the Type 2 internals. The EPRI PWR Safety and Relief Valve Justification Report indicates that as of August 1981, the licensee had not purchased the parts necessary to upgrade their valves to the Type 2 internals. The manufacturer indicated that all plants using this valve are expected to make the modification. Since the EPRI tests were conducted with the Type 2 internals, the licensee should either make the modification or justify that the tests demonstrate acceptable performance of the plant valve.

Maine Yankee Response

The manufacturer of the valves has informed Maine Yankee that the replacement internals have no effect on valve function and operation. Therefore, we have not addressed when and whether or not they should replace them. It appears certain that when the stock replacements for these parts run out, the Type 2 internals will be ordered.

Dresser has redesigned the 31533VX-30 PORV valve internals to allow easier lapping of the seats and discs. They also now offer a higher rate spring which prevents leakage of the valve in the 50 to 100 psig range. At these low pressures, Maine Yankee has not experienced problems with PORV leakage. These internals will undoubtedly make valve servicing much easier, and since they are the only type of replacement parts Dresser now sells, they ultimately will be used in all Dresser PORVs.

NRC Question #3

3. 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.

Maine Yankee Response

The control of the PORVs is from two sources: with the plant at power, the control stems from the Reactor Protection System; under LTOP conditions, the control is described in Reference (2). The Reactor Protection System is designed to the FSAR requirements, which includes IEEE 279-1968. The LTOP System was designed and installed to IEEE 323-1974 and IEEE 344-1975.

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 opening nor can it prevent an open valve from closing.

NRC Question #4

4. The submittal uses a combination of in situ tests and computer analysis to demonstrate acceptable operation of the safety valve. For the in situ testing, the applicability of the test for the modified piping and justifications that the limiting conditions were tested were not provided. For the computer analysis, the details and results of the benchmark of the Program (COUPLE) to the safety valve test and the criteria for selection of the ring settings were not provided. Provide justification that the in situ tests are applicable to the modified piping and that the conditions tested bound the plant limiting conditions. Provide details of the COUPLE Program benchmark including the results of the comparison with the EPRI safety valve tests and provide the criteria that was used to select the ring settings.

Maine Yankee Response

The in situ testing of safety relief valve operation conducted during the original plant hot functional testing demonstrated stable blowdown of these valves in the original piping configuration. These tests were not repeated for the modified piping configuration.

The shortened inlet piping in the modified configuration guarantees improved stability over that which was previously tested. The degree of that improvement is demonstrated by the COUPLE Program analysis results which compared with the Maine Yankee test results is obviously conservative. Thus, the modified piping configuration has improved the potential for stable blowdown (the system has never been challenged except in test) from better than borderline stability to a very stable system with a deeper blowdown capability.

The Dresser safety relief valves in use at Maine Yankee are Model No. 31709KA. These valves use three rings: middle, lower, and upper to modify the pressure at which the valves "pop" open, the degree of lift throughout the blowdown, and the pressure at which they finally close. The COUPLE code uses the characteristics of the middle and the lower ring positions and the associated fluid dynamics to compute the valve performance. Several runs of the couple model of those valves with different ring settings were performed in order to identify the configuration which yielded the desired valve performance. The code does not model the dynamics of the valve as controlled by the upper ring, thus, Dresser's recommended position of this ring is used.

The benchmarking process for the COUPLE code was done by Continuum Dynamics during the EPRI testing on an EPRI contract. It included the code development and extensive benchmarking to the EPRI test program results. That activity is reported in EPRI Report No. NP-3493, entitled, "Coupled Valve Dynamic Model for Steam Two-Phased and Subcooled Discharge", Project Manager Avtar Singh.

NRC Question #5

5. The April 4, 1983 submittal states that the operability of the valves is demonstrated by confirming that the calculated piping loads at the valve inlets are conservative when compared to the EPRI test program loads. Thermal expansion of the pressurizer causing displacement of the piping nozzles and thermal expansion of the piping from the nozzles to the valves can contribute to the bending moment induced in the valve body. The submittal does not make clear what loads were considered in calculating the bending moments applied to the plant safety valves and PORVs. Provide additional discussion comparing the measured moment on the tested valves to the calculated induced moments from all effects including those described above on the plant-specific valves. Verify that the bending moments would have no adverse effect on the operability of the plant valves.

Maine Yankee Response

The major consideration in achieving a piping configuration which produces stable blowdown of the SRVs was to limit bending moments across these valves for seat leakage considerations during normal plant operations. This limits bending due to thermal, pressure, and deadload to 1500 ft-lbs, the manufacturers recommended operating limit. Our final analyzed bending moment for these, under normal operating loads, is 1300 ft-lbs.

For further assurance that our total structural loads sustained by these valves (as identified by Stone and Webster's analysis) would have no adverse effect, we had Dresser analyze them. They found adequate margins for both normal plant operation and the combined thermal, deadload, seismic, and blowdown. This was done for the SRVs and PORVs.

The maximum combined moment (SRSS three directions) including all transients analyzed for the SRVs was 4071 ft-lbs and for PORVs 1534 ft-lbs. The comparable bending moment for the EPRI tests were on the order of 20,000 ft-lbs. in which no failures were noted.

NRC Question #6

6. The PORV block valves are 2 1/2" solid wedge gate valves manufactured by Anchor Darling, equipped with Limitorque SMB-00-10 Motor operators. The actual model number of the block valve is not given in the submittal. The only Anchor Darling block valve tested in the EPRI-Marshall tests was a 3" double disc gate valve with a Rotork operator. The submittal states that the block valves in the Maine Yankee plant are similar to the Velan 3" gate valves tested in the EPRI-Marshall test program. The Marshall tests did include tests on a 3" Velan model B10-3054B-13MS gate valve with a Limitorque SMB-000-10 operator. The test results for the Velan valve appear to be satisfactory. The April 4, 1983 submittal states that the block valve test performed in 1981 is reported in the June 30, 1982 submittal. However, there are no test results reported in the June 30, 1982 submittal. It contains only the reference to the EPRI tests on the Velan valve.

   Provide additional information comparing the design of the Maine Yankee block valves with that of the Velan valve. Provide evidence that the force necessary to operate the Maine Yankee block valve is less than that required to operate the Velan valve.

Maine Yankee Response

As reported in the June 30, 1982 transmittal (Appendix A, Page 3), the block valves at Maine Yankee are Anchor Darling 2 1/2" wedge gate valves which were originally equipped with SMB-00-10 operators (2 1/2-1500^{0} Gate Valve, Cast CF8M, BW Ends, Bolted Bonnet).

While these valves are similar in operation to those used in the EPRI Test Program, the capability of the valve/operator combination in use at Maine Yankee, or anywhere else, can be verified only by tests of that combination in the specific configuration. Therefore, the following test program was used for this purpose.

Following the successful Marshall Steam Station tests, Maine Yankee, upon consultation with the valve manufacturer (Anchor Darling), modified the Limitorque Operators to increase the maximum closing torques delivered by the motor operators. After that modification and prior to start-up following a refueling, an in situ test of the block valve closure with full flow was performed on July 9, 1981. Initial RCS conditions included pressurizer pressure at 2248 psig, and pressurizer level at 64%. The test was run by opening the block valve with the PORV open. Once full flow was verified, the block valve was closed. The valve stroked opened and closed as designed. Closure time was less than 14 seconds. The valve closed leak tight as verified by subsequent measurements.

During the 1984 Refueling Outage, the modified Limitorque Operators were replaced by environmentally qualified SMB-00-15 operators with identical gearing to those used in the 1981 test to provide the same closure force as the tested configuration.

The successful results of both the EPRI tests on similar valves and the successful in situ tests at Maine Yankee clearly demonstrate the operability of the PORV block valves as installed at Maine Yankee.

NRC Question #7

7. The April 4, 1983, submittal states that the safety/relief valve discharge loadings (forcing function) were developed utilizing Stone & Webster's in-house computer programs, STEHAM and WATAIR. It also states that the STEHAM program was benchmarked to RELAP5/MOD1 and compared to EPRI test data. To allow for an evaluation of the analysis, identify the important parameters and the rationale for their selection. This should include a description of the method used to generate fluid pressures and moments over time, and how the program calculates resulting fluid forces on the system. Fluid conditions assumed for the analysis should be provided such as peak pressure, pressurization data, back pressure, temperature, fluid range, and number and type of valves actuated.

   Because the ASME Code requires derating of the safety valves to 90% of expected flow capacity, the safety valve analysis should be based on 111% of flow rating, unless otherwise justified. Information should be provided explaining how the derating of the safety valves was handled and the method used to establish flow rates for the safety valves and PORVs in the analysis.

Maine Yankee Response

The safety/relief valve discharge loadings (the forcing function) are developed utilizing SWEC's in-house computer programs, STEHAM, WATAIR, and WATSLUG. Attachment A provides a general description of the computer programs utilized for the development of the forcing function.

Table 1 shows thermal-hydraulic transient cases and associated important parameters considered in the analysis.

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 considering the time delay between relief and safety valve opening, were analyzed. The two relief valves were considered to open simultaneously, while the three safety valves were considered to open sequentially in a time interval consistent with pressurizer rise rate and the set pressure of the individual valves.

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 network with uniform diameter. Since the piping model involves different pipe sizes, a conservative flow diameter was established by modeling the piping system with the largest diameter while keeping the other parameters as given, such as the piping segment lengths, flow rate and other operating parameters. The water discharge from each relief valve was combined at the common entry point to maximize forces in the discharge piping as applicable.

The unusual transition case, from saturated steam discharge to continuous subcooled water discharge through the relief valves, due to extended high pressure safety injection event (HPSI), was also considered.

The subcooled water flashes through the relief valves and a two-phase flow is formed downstream of the relief valves. The fluid forces upstream of the relief valves were developed using the WATSLUG program even though there is a continuous water column formed in the upstream piping. The fluid forces, due to two-phase flow on the downstream of relief valve piping, were developed utilizing the STEHAM computer program by modeling the water/steam mixture as an equivalent ideal gas.

The relief valve bore diameter of 1-5/16 in. was obtained from the manufacturer. The relief valves' opening times for the various fluid inlet conditions and water discharge flow rates for LTOP and HPSI were derived from the EPRI/Wyle PORV test data (see Table III-3).

The safety valve area of 1.84 in.^{2} and valve opening time of 15 milliseconds were obtained from the manufacturer/EPRI test data.

The pressurizer transients considered, namely: loss of load, LTOP, and extended HPSI will generate bounding loads for safety/relief valve actuation.

The flow rates for SRVs and PORVs were established using the equation from ASME B&PV Code 1977. Section I, Part PG-69.1.2, uprated by 111% and using the coefficient of discharge established by Dresser Industries' tests. Since we were establishing flow rates for the piping flow and stress analysis, no valve derating was done.

TABLE 1 (Question 7)

FLUID CONDITIONS AND CRITICAL PARAMETERS
ASSUMED FOR THE ANALYSIS

	Case 1	Case 2	Case 3	Case 4	Case 5	Notes
Transient Case	2 relief valves open discharge steam	3 safety valves open in sequence discharge steam	3 safety valves open in sequence discharge steam with 2 relief valves already open	LTOP 2 relief valves open discharge water, water fronts combine at common piping	HPSI 2 relief valves open discharge water, flashing occurs at valves two-phase flow downstream	
Computer Code Used	STEHAM	STEHAM	STEHAM	WATAIR	WATSLUG STEHAM	
Valve Opening Time	0.1 sec (3)	0.015 sec (3)	0.015 sec (3)	0.07 (7)	0.06 (8)	
Flow Rate/Valve	156,420 lb/hr steam at 2350 psig (4)	228,000 lb/hr steam at 2535 psig (6)	228,000 lb/hr steam at 2535 psig (6)	388,000 lb/hr water (8)	545,000 lb/hr water (9)	
Valve Set Pressure	2385 psig (3)	1st valve - 2485 psig 2nd valve - 2510 psig 3rd valve - 2535 psig (3, 6)	1st valve - 2485 psig 2nd valve - 2510 psig 3rd valve - 2535 psig (3, 6)	485 psig (4)	2400 psig (9)	
Peak Pressure at Valve Inlet	2464.3 psig (5)	2574.3 psig (7)	2574.3 psig (7)	549.3 psig (7)	2400 psig (7)	
Pressurization Rate	25 psi/sec (5)	63.1 psi/sec (7)	63.1 psi/sec (7)	12.5 psi/sec (7)	0.0 psi/sec (7)	
Back Pressure	2 psig	2 psig	56.3 psig	2 psig	2 psig	
Temperature	670°F	670°F	670°F	470°F	568°F (7)	

Relief Valve Bore Diameter is 1-5/16 in.
Safety Valve Area is 1.84 in.^{2}
All steam flow rates were derated by dividing by the derate factor of 0.9
Note: Numbers in ( ) are references provided to SWEC by YAEC.

ATTACHMENT A (TO QUESTION 7 RESPONSE)

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-SW (ME-110) program.

WATSLUG

General Description

The purpose of WATSLUG is to determine forcing functions on piping systems during water slug discharge events for subsequent input to piping dynamic analysis.

The analysis is based upon rigid body motion of the generally subcooled water slug and ideal gas representations of the steam or air using rigid column theory to facilitate tracking the several water-steam or water-air interfaces. The driving force is the steam pressure between the valve and the slug, less friction and other losses, and back pressure. Density changes due to possible local flashing of the water slug are considered. Having recourse to the control volume theory, the subsequent segment force calculation is carried out.

The input consists of complete piping system geometry, pipe dimensions, valve flow characteristics, valve opening time, detail upstream steam conditions, and initial downstream steam or air conditions, while the output contains forcing functions for each piping segment based upon flow velocities, pressures, and densities during the water slug discharge event. Forces are written on tape for direct input to NUPIPE-SW.

WATAIR DESCRIPTION

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.

NRC Question #8

8. The April 4, 1983 submittal indicates that blowdowns of the safety valves are expected to exceed the 5% design value. A minimum blowdown of 13.5% for all valves up to a maximum of 17% for one valve is reported in the submittal.

The December 30, 1982 submittal contains a computer analysis indicating that the pressurizer level remains well below the valve nozzles for a loss-of-load event followed by 20% blowdown; thus, a steam water flow through the safety valves is not expected. However, the submittal does not address the adequacy of decay heat removal for the higher blowdown. Discuss the effects of the reduced pressures on the adequacy of decay heat removal.

Maine Yankee Response

The results of a CE analysis which predicts the behavior of the pressurizer safety valves at Maine Yankee during a loss-of-load transient are presented in Figure 15 of Reference 10. A blowdown of 20% was calculated with valve closure at 2,000 psia. In this pressure range, decay heat would be removed via the steam generators under forced convection, or, in the event of a loss of ac power, natural circulation conditions.

However, during full-power operation, the peak RCS temperature of approximately 600^{o}F occurs in the reactor vessel upper head region. This corresponds to a saturation pressure of 1,543 psia. Since the upper head region is relatively isolated from transient temperature changes during forced convection, and nearly stagnant during natural circulation, the fluid temperature is expected to remain constant throughout an overpressure transient. Depressurizing the RCS to 2,000 psia following a safety valve challenge will not result in void formation; therefore, no degradation in decay heat removal will occur.

NRC Question #9

9. The April 4, 1983, submittal states that stress analysis of the pressurizer safety and relief valve piping system was performed utilizing NUPIPE-SW computer code. Explain how the program was verified for this particular application. Identify the multi-valve opening sequence used to produce the worst case loading on the piping and supports. Identify the load combinations considered in the analysis and the allowable stress limits used. Load combinations and acceptance criteria were recommended in the EPRI PWR Safety and Relief Valve Test Program Guide for application of Valve Test Progress Results. If other load combinations and criteria are appropriate, the rationale for their selection should be provided.

Maine Yankee Responses

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 originally 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. Nonexclusive right to use of 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 4), 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 6.

The methodology used to calculate the forcing functions due to safety/relief valve discharge loadings is described in the response to Question 7. The forcing functions developed due to various fluid conditions (described in response to Question 7) were then applied to the appropriate piping segments and a time-history modal superposition analysis was performed using NUPIPE-SW. The dynamic responses obtained for the various conditions were compared and the maximum responses due to

any fluid condition was selected and included in the load combination with the corresponding earthquake, pressure, and dead weight loads to ensure worst case loading on the piping and supports.

The piping stress analysis for the Maine Yankee pressurizer safety and relief valve piping was performed in accordance with the ANSI B31.1 Code for Power Piping, 1977 Edition (Reference 7). The load combinations and allowable stress limits for the safety and relief valves discharge condition are:

P + DL _ 1.0 S_h

P + DL + SRSS [(OCC, OBE (I)] _ 1.2 S_h

P + DL SRSS [(OCC, SSE (I)] _ 1.8 S_h

P + DL + SRSS [EXT. HPSI, OBE (I)] _ 1.2 S_h

P + DL + SRSS [EXT. HPSI, SSE (I)] _ 1.8 S_h

THERMAL _ 1.0 S_A

Where:

OCC	=	Loads due to relief valve discharge
P	=	Longitudinal pressure stress
DL	=	Deadload stress
OBEI	=	Seismic stress due to Operating Basis Earthquake Inertia
SSEI	=	Seismic stress due to Safe Shutdown Earthquake Inertia
S_c	=	Basic material allowable stress at room temperature
S_h	=	Basic material allowable stress at maximum operating temperature
S_a	=	$(1.25\ S_c + 0.25\ S_h)$
EXT. HPSI	=	Extended HPSI (high pressure safety injection) through relief valve

The occasional load in the above equations is 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.

NRC Question #10

10. The April 4, 1983, submittal states that an analysis to qualify supports is in progress. The results of the analysis should be provided. Any required modifications indicated by the analysis should be identified along with a schedule for implementation.

Maine Yankee Response

Twenty-seven supports were re-analyzed as a result of the safety valve piping modification. Of these twenty-seven supports, a total of nineteen required some modification. The modifications resulted from the revised piping geometry, the newly calculated support loads (deadload, thermal, seismic, and valve discharge), and the resolution of IE Bulletin 79-02 concerns.

All required support modifications were installed during the 1984 Refueling Outage.

The results of the support analysis are available at the Engineering Office of Yankee Nuclear Services Division in Framingham, Massachusetts.

References:

1. Dresser Industries Testing Summary, SV-203F, "31533 VX-30 Electromatic Relief Valve", R. S. Huffman, dated april 17, 1984.

2. MYAPCo Letter to NRC dated September 17, 1984 "Low Temperature Overpressurization Protection Additional Information",(MN-84-156).

3. "S&W Design Basis and Input Documents, Safety nd Relief Valve Analysis, Maine Yankee Atomic Power Company", Letter No. SLYA-7, dated February 25, 1983.

4. YAEC Letter, M. M. Allison to G. Choquette, SEG 96/83, W.O. 5999, 1MS-MY-G.1.13.5, DCC-MY-S&W-83-01, dated March 1, 1983.

5. EPRI Test V102-20, Phase B Extension Table 5-5.

6. Telecopy of YAEC P.O. 102868, M. M. Allison to G. Choquette, dated September 28, 1983.

7. Memorandum, "Report on Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves, Maine Yankee Atomic Power Station", P. A. Bergeron to W. G. Jones, dated July 28, 1982, Tag 82-26, Table 4.1.

8. "EPRI/Wyle Power-Operated Relief Valve Phase III Test Report", Volume 3, Summary of Phase III Testing of the Dresser Relief Valve, EPRI NP-2670-LD, Table III-3 and III-10.

9. Telephone Memorandum, W. Jones and D. A. VanDuyne, dated March 22, 1983.

10. MYAPCC Letter to NRC, dated December 30, 1982, "Evaluation of Safety and Relief Valve Operation", MN-82-255.