ENCLOSURE

SAFETY EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING FOR
DIABLO CANYON UNITS 1 AND 2
DOCKET NOS. 50-275, 50-323

1. INTRODUCTION

1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant systems. There have been instances of valves opening below set pressure, valves opening above set pressure and valves failing to open or reseat. From these past instances of improper valve performance, it is not known whether they occurred because of a limited qualification of the valve or because of a basic unreliability of the valve design. It is known that the failure of a power-operated relief valve (PORV) to reseat was a significant contributor to the Three Mile Island (TMI-2) sequence of events. These facts led the task force which prepared NUREG-0578 (Reference 1) and, subsequently, NUREG-0737 (Reference 2) to recommend that programs be developed and executed which would reexamine the functional performance capabilities of Pressurized Water Reactor (PWR) safety, relief, and block valves and which would verify the integrity of the piping systems for normal, transient and accident conditions. These programs were deemed necessary to reconfirm that the General Design Criteria 14, 15, and 30 of Appendix A to Part 50 of the Code of Federal Regulations, 10 CFR are indeed satisfied.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require that (1) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have an extremely low probability of abnormal leakage; (2) the reactor coolant system and associated auxiliary, control, and protection systems be

designed with sufficient margin to assure that the design conditions are not exceeded during normal operation or anticipated transient events; and (3) the components which are part of the reactor coolant pressure boundary shall be constructed to the highest quality standards practical.

To reconfirm the integrity of overpressure protection systems and thereby assure that the General Design Criteria are met, the NUREG-0578 position was issued as a requirement in a letter dated September 13, 1979, by the Division of Licensing (DL), Office of Nuclear Reactor Regulation (NRR), to ALL OPERATING NUCLEAR POWER PLANTS. This requirement has since been incorporated as Item II.D.1 of NUREG-0737, Clarification of TMI Action Plan Requirements (Reference 2), which was issued for implementation on October 31, 1980. As stated in the NUREG reports, each pressurized water reactor Licensee or Applicant shall:

1. Conduct testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

2. Determine valve expected operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2.

3. Choose the single failures such that the dynamic forces on the safety relief valves are maximized.

4. Use the highest test pressures predicted by conventional safety analysis procedures.

5. Include in the relief and safety valve qualification program the qualification of the associated control circuitry.

6. Provide test data for Nuclear Regulatory Commission (NRC) staff review and evaluation, including criteria for success or failure of valves tested.

7. Submit a correlation or other evidence to substantiate that the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the Final Safety Analysis Report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must be considered.

8. Qualify the plant specific safety and relief valve piping and supports by comparing to test data and/or performing appropriate analysis.

2. PWR OWNER'S GROUP RELIEF AND SAFETY VALVE PROGRAM

In response to the NUREG requirements previously listed, a group of utilities with PWRs requested the assistance of the Electric Power Research Institute (EPRI) in developing and implementing a generic test program for pressurizer power operated relief valves, safety valves, block valves and associated piping systems. Pacific Gas and Electric Company (PG&E), the owner of Diablo Canyon Units 1 and 2, was one of the utilities sponsoring the EPRI Valve Test Program. The results of the program are contained in a group of reports which were transmitted to the NRC by Reference 3. The applicability of these reports are discussed below.

EPRI developed a plan (Reference 4) for testing PWR safety and relief valves under conditions which bound actual plant operating conditions. EPRI, through the valve manufacturers, identified the valves used in the overpressure protection system of the participating utilities. Representative valves were selected for testing with a sufficient number of the variable characteristics that their testing would adequately demonstrate the performance of the valves used by utilities (Reference 5). EPRI, through the Nuclear Steam Supply System (NSSS) vendors, evaluated the FSARs of the participating utilities and arrived at a test matrix which bounded the plant transients for which over pressure protection would be required (Reference 6).

The utilities that participated in the EPRI Safety and Relief Valve Test Program also obtained information regarding the performance of PORV block valves (Reference 9). A list of valves used or intended for use in participating PWR plants was developed. Seven block valves believed to be representative of the block valves utilized in the PWR plants were selected for testing. Additional tests were performed by Westinghouse Electro-Mechanical Division (WEMD) on valve models they manufacture. (Reference 16).

EPRI contracted with Westinghouse Corporation to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 7). Since Diablo Canyon 1 and 2 were designed by Westinghouse, this report is relevant to this evaluation.

Several test series were sponsored by EPRI. PORVs and block valves were tested at the Duke Power Company, Marshall Steam Station located in Terrell, North Carolina. Only steam tests were conducted at the Marshall Station. Block valves, therefore, were only tested for full flow, full pressure, steam conditions at Marshall. Water flow tests were performed by WEMD on four valve models they manufacture. Conditions ranged from 60 to 600 gpm and 1500 to 2600 psi differential pressure. Additional PORV tests were conducted at the Wyle Laboratories Test Facility located in Norco, California. Safety valves were also tested at the Combustion Engineering Company, Kressinger Development Laboratory, in Windsor, Connecticut. The results for the relief and safety valve tests are reported in Reference 8. The results for the block valve tests are reported in Reference 9.

The primary objective of the EPRI/C-E Valve Test Program was to test each of the various types of primary system safety valves in pressurized water reactor plant service for the full range of fluid conditions under which they may be required to operate. The conditions selected for test (based on analysis) were limited to steam, subcooled water and steam to water transition. Additional objectives were to (1) obtain valve capacity data, (2) assess hydraulic and structural effects of associated piping on valve operability, and (3) obtain piping response data that could ultimately be used for verifying analytical piping models.

Transmittal of the test results meets the requirement of Item 6 of Section 1.2 to provide test data to the NRC.

3. PLANT SPECIFIC SUBMITTAL

The Pacific Gas and Electric (PG&E) Company submitted a preliminary evaluation of the safety/relief valve qualification for Diablo Canyon Power Plant (DCPP) Units 1 and 2 on March 31, 1982 (Reference 10) and an assessment of block valve qualification on June 30, 1982 (Reference 11). A final evaluation of the pressurizer safety and relief valve piping was transmitted in a summary report on December 13, 1982 (Reference 12). A request for additional information was transmitted to PG&E on November 15, 1983 (Reference 13) to which PG&E responded on January 23, 1984 (Reference 14). A second request for additional information was transmitted to PG&E on September 18, 1984 (Reference 19) to which PG&E responded on March 13, 1985 and March 28, 1985 (Reference 17 and 18).

4. REVIEW AND EVALUATION

4.1 Valves Tested

The Diablo Canyon plants utilize three power operated relief valves (PORVs) and three safety valves in each unit. In addition, each unit employs three PORV block valves. The PORVs are 20,000 series valves manufactured by Masoneilan International. The safety valves are Model HB-BP-86 size 6M6 valves with loop seal internals manufactured by Crosby Valve and Gage Company. Both the safety and relief valves have water seals upstream of the valves. The block valves are Velan Model B10-3054B-13MS gate valves with Limitorque SMB-00-15 motor operators.

The Masoneilan PORV used in Diablo Canyon 1 and 2 is a Model 38-20771, which is one of only two models installed in plants in the U.S. The only difference between the two models is that the valve plug used in the Model 38-20771 has a linear control contour while the other, Model 38-20721, has an equal percentage contour. This difference only affects the flow characteristics of modulating valves and thus has no significant impact on the EPRI tests. The valve used in the EPRI tests, Model 38-20721, is essentially identical to those used in Diablo Canyon 1 and 2.

The safety valve used in Diablo Canyon 1 and 2, which is a Crosby Model HB-BP-86 size 6M6 valve with loop seal internals, was tested in the EPRI program. The valve was tested on a long inlet piping configuration with a loop seal, which represents the Diablo Canyon installation. The results from the EPRI tests can, therefore, be used to demonstrate operability of the safety valve.

The block valve used in Diablo Canyon 1 and 2, which is a Velan Model B10-3054B-13MS, was tested in the EPRI test program. The valve is installed at the plant in a horizontal configuration and the valve was tested in a similar horizontal configuration. The valve is designed for use in either a horizontal or vertical orientation. The plant valve has a Limitorque SMB-00-15 motor operator while the test valve had a Limitorque

SB-00-15 operator. These two operators are essentially identical except that the SB-00-15 has a spring pack compensator on the stem nut which makes it more useful for high speed, high temperature service. The test valve is, therefore, representative of the plant valve.

Based on the above, the valves tested are considered to be applicable to the in-plant valves at Diablo Canyon 1 and 2, and to have fulfilled the criteria of Items 1 and 7 of Section 1.2 regarding applicability of the test valves.

4.2 Test Conditions

The valve inlet fluids conditions that bound the overpressure transients for Westinghouse-designed PWR plants are identified in Reference 7. The transients considered in this report include FSAR, extended high pressure injection, and cold overpressurization events.

For FSAR transients resulting in steam discharge through the safety valves, the pressurizer experiences a peak pressure of 2555 psia (loss-of-load transient) and a maximum pressurization rate of 144 psia/sec (locked rotor transient). The expected plant backpressure for steam flow is less than 400 psig. The Crosby HB-BP-86 6M6 safety valve was subjected to eleven loop seal-steam tests with a long inlet configuration in the EPRI testing program. Of these tests, four were applicable to the Diablo Canyon safety valves since the ring settings in these four tests (-71,-18) were representative of the plant ring settings. In these tests the peak tank pressures reached 2760 psia. The pressurization rates ranged from 319 to 360 psi/sec and the peak backpressures ranged from 245 to 710 psia. The test inlet fluid conditions for the loop seal-steam tests are representative of the expected conditions for FSAR transients resulting in steam discharge for the safety valves.

For the FSAR transients resulting in steam discharge, the PORVs will open at a pressure somewhat above the opening setpoint of 2350 psia. The maximum pressurizer pressure is 2532 psia (loss-of-load) and maximum pressurization rate is 130 psi/sec (locked-rotor) when the safety and

relief valves actuate. The Masoneilan PORV was subjected to thirteen steam tests and one water seal simulation test in the EPRI test program. In the steam tests the maximum pressure at valve opening ranged from 2455 to 2510 psia. The water seal test was conducted at an initial pressure of 2640 psia. The test fluid conditions in the steam and water seal tests on the PORVs are representative of FSAR transients.

The Diablo Canyon FSAR indicates that there is potential liquid flow through the safety valves and PORVs for the feedwater line break accident. The FSAR analysis for feedwater line break, however, was based on very conservative assumptions. PG&E and Westinghouse will perform a revised analysis (per Reference 18) of this event using the LOFTRAN code and the 1979 ANS decay heat model. Westinghouse has performed sensitivity analyses of the feedline break event with offsite power available on a 4-loop plant similar to DCPP (Reference 18). In these analyses, Westinghouse also used the LOFTRAN Code and the 1979 ANS decay heat model and determined that the pressurizer did not become water solid and that liquid relief through the valves did not occur. Based on these results, it is expected that the revised analysis on DCPP will demonstrate that the safety valves and PORVs of DCPP will not pass water in a feedline break event. The NRC staff has concluded that this demonstration will satisfy the requirements of NUREG-0737, Item II.D.1.

The limiting extended High Pressure Injection (HPI) event is a spurious activation of the safety injection system at power. In this event, the safety valves and PORVs would open on steam and no liquid discharge would be observed until the pressurizer became water solid. According to Reference 7, this would not occur until at least 20 minutes into the event, which allows ample time for operator action. Therefore, the potential for liquid discharge for extended HPI events can be disregarded. The steam flow conditions for the limiting extended HPI event as presented in Table 5-3 of Reference 7 are enveloped by those previously presented for the loss-of-load and locked rotor transients. Thus, the fluid conditions for extended HPI events are encompassed by the conditions considered for other transients.

The cold overpressurization events challenge only the PORVs since these are used to mitigate such transients. The fluid conditions for these events can vary between steam and subcooled water because of administrative requirements for maintaining a steam bubble in the pressurizer during low temperature operations. The range of potential fluid conditions for cold overpressure events are presented in Reference 7. In addition to the high pressure water, steam, and transition tests previously mentioned, the relief valve was subjected to four low pressure water tests. The pressures in these tests ranged from 590 to 652 psia while the valve inlet temperatures ranged from 113° to 450°F. These test conditions together with the test conditions in the high pressure tests sufficiently encompass the range of expected fluid conditions for cold overpressure events in this Westinghouse 4-loop PWR.

The block valves are required to operate over a range of fluid conditions (steam, steam-to-water, water) similar to those of the relief valves. The block valves were, however, only tested under full pressure steam conditions (to 2500 psia). In friction testing done by Westinghouse on stellite coated specimens (the Velan valve has stellite seats), water specimen friction factors increased from as low as 0.12 to as high as 0.75 over the initial 200 cycles of testing. With 550°F steam, the friction factor dropped from a high of 0.6 to 0.35 over 200 cycles. These test results show that in the span of 21 test cycles (as applied in tests at the Marshall steam station) the thrust required to cycle the valve during steam tests would be similar to that required if the test medium were water. In addition, the required torque to open or close the valve depends almost entirely on the differential pressure across the valve disk and is rather insensitive to the momentum loading and, therefore, is nearly the same for water or steam and nearly independent of the flow. The full pressure steam tests, therefore, are adequate to demonstrate operability of the valve for low pressure steam and the required water conditions.

The presentation above, demonstrating that the test conditions bounded the conditions for the plant valves verifies that Items 2 and 4 of Section 1.2 have been met, in that conditions for the operational occurrences have been determined and the highest predicted pressures were chosen for the tests. The presentation also verifies that the portion of Item 7 which requires showing that the test conditions are equivalent to those prescribed in the FSAR has been met.

4.3 Operability

As discussed in the previous section, the safety valves and PORVs are required to operate over a range of full pressure steam, steam to water transition, and subcooled water fluid conditions. The valves were tested for the range of required conditions in the EPRI test program. The block valves are also required to operate for steam and liquid flow conditions. These valves were subjected to full pressure steam tests, the results of which apply also to liquid flow, as discussed in Section 4.2.

In all applicable loop seal-steam tests (where representative ring settings were used), the safety valve opened, fluttered or chattered in a partial lift position during loop seal discharge, then popped open on steam and closed with 6.2 to 9.4% blowdown. In all of these tests the valve reached rated steam flow at 3% accumulation, though without necessarily reaching full rated lift. Thus, in all applicable cases, the valve performed its safety function of opening, relieving pressure, and closing. In one test, however, the valve reopened after closure and chattered, and the test was terminated after the valve was manually opened to stop the chattering. This result does not constitute a valve instability problem at DCPP since this test was a repeat of a previous test where the valve operated successfully, and the operator may well have opened the valve and terminated the test before the valve had opportunity to close on its own. Further, the EPRI valve inlet pressure trace showed that pressure oscillations at the valve inlet, which are acoustic in nature, caused the valve to chatter. The shorter valve inlet piping configuration at DCPP (about 9 feet) than was used in the test (approximately 14 feet) is expected to reduce the magnitude of such oscillations and lead to more stable valve behavior. Thus, the test results obtained from the EPRI tests on the Crosby 6M6 safety valve adequately demonstrate that the safety valve will perform its safety function for the DCPP fluid conditions.

Bending moments as high as 286,800 in-lb were induced on the discharge flange of the Crosby 6M6 test valve, which had no adverse effect on valve performance. Since this applied moment exceeds the maximum estimated

bending moment of 91,400 in.-lb for the DCPP valve, the performance of the plant valve is expected to also be unaffected by bending moments imposed during discharge transients. The value of the predicted maximum bending moment at the outlet flange of the Diablo Canyon safety valve was not provided by the licensee. An estimate was made based on the piping stress data in the Westinghouse stress report (Reference 12). Based on the estimate, the maximum bending moment at the outlet of the in-plant safety valve will be less than 91,400 in.-lb. This bending moment was calculated using the maximum piping stress predicted by Westinghouse for a faulted condition at a pipe section immediately upstream of the safety valve (Node 84). The corresponding bending moment, which produced this stress, was calculated using the equation for the longitudinal stresses due to occassional loads from the governing Code, ANSI B31.1-1973 Summer Addenda. The size for the inlet pipe is 6 in. Schedule 160. The maximum operating pressure was used in the equation instead of the design pressure. For the calculation of the bending moment from a given pipe stress, as in this case, it is conservative to use a lower internal pressure (i.e., the operating pressure). The bending moment thus calculated represents an upper bound on the bending moment at the inlet flange of the safety valve. Although the bending moment at the discharge flange cannot be calculated with the available data, it is apparent that its value cannot exceed the moment at the inlet, because the maximum stress at the outlet section is lower than that at the inlet section and the outlet pipe (6 in. Sch 80S) is weaker than the inlet pipe.

As stated above, observed blowdowns ranged from 6.2 to 9.4%, which exceed the design value of 5%. Thus, it must be demonstrated that these extended blowdowns will not impact plant safety. The revised analysis that PG&E and Westinghouse will perform on the feedline break event (see Section 4.2) will include the effects of an extended blowdown, on the order of 10%. Based on sensitivity analyses and other evaluations of a 4-loop plant (Reference 17, pp. 3 and 4) in which Westinghouse investigated the effects of different blowdowns, it is not expected that the extended blowdowns will increase pressurizer water volume to a level at which liquid relief would occur or that inventory loss from the Reactor Coolant System

would result in core uncovery. Thus, the analysis should demonstrate that the blowdowns will not impact plant safety. If, however, the analysis does not satisfactorily confirm that the extended blowdowns pose no safety hazard, then PG&E will have to resolve this concern about blowdowns in some other manner.

For all tests on the PORV, the valve opened and closed on demand. The valve opening time was sensitive to the air supply system pressure and to the size of the tubing used in the air supply system. Both the pressure and tubing size had to be increased during tests to reduce opening times to reasonable values. The Diablo Canyon design of this system accounts for this difficulty. The actuator housing air connection was bored and retapped to 3/8-in. NPT. The solenoid valve is wall mounted as close to the valve as possible and is connected with a 1/2-in. I.D. Synflex hose. This assembly is connected to a 1-in. NPT pressure regulator which is fed directly from a 1-in. supply header. Pressure to the valve is maintained at a minimum pressure of 55 psig. This combination of high pressure, short tubing, and larger line size will assure response times of less than 2 seconds from full closed to full open, and 1 second from full open to full closed. Based on valve performance during the EPRI tests and modifications made to the valve air supply system, the PORVs have been adequately demonstrated to operate under expected fluid transient conditions.

A bending moment of 35,600 in-lb was induced on the discharge flange of the test PORV, which had no adverse effect on valve performance. Since this applied moment exceeds the maximum estimated bending moment of 29,000 in.-lb for the DCPP valve, the performance of the plant valve is expected to also be unaffected by bending moments imposed during discharge transients. The value of the predicted maximum bending moment at the outlet flange of the Diablo Canyon PORV was not provided by the licensee. An estimate was made based on the piping stress data given in the Westinghouse stress report (Reference 12). Based on the estimate, the maximum bending moment at the outlet of the inplant PORV will be less than

29,000 in.-lb. This bending moment was calculated using the maximum piping stress predicted by Westinghouse for a faulted condition at a piping section upstream of the PORV (Node 4010). The technique used for making this estimate is the same as that described on Page 12 for the bending moment on the safety valve.

The Velan block valve with Limitorque SB-00-15 operator having a motor rated at 15 ft-lb of torque was subjected to 21 cycles during evaluation tests with steam pressures up to 2500 psig. The Velan valve fully opened and closed for all 21 test cycles. Thus, the block valves have been demonstrated to operate under expected fluid transient conditions.

NUREG-0737 Item II.D.1 states that, in addition to the PORVs, their associated control circuitry shall also be qualified for design basis accidents and transients. The instruments and controls of two PORVs and of all three block valves of the Diablo Canyon plants meet the safety-grade design criteria of General Design Criteria (GDC) 1, 2, 4, 12, 13, 19, 20, 21, 22, 23, 24 and 29 and have been qualified for the Hosgri seismic event. Remote control circuitry, however, is not provided for the third PORV. According to findings of the Atomic Safety and Licensing Board in the DCPP Full Power Decision, the PORV's and their associated block valves and instrumentation and controls are not required (with one exception) by the criteria in Section III.C of Appendix A to 10 CFR Part 100 to be qualified as safety-grade. The exception is that of protection from low-temperature overpressurization, which is adequately provided for by two safety-grade PORV systems. Thus, the third PORV is not needed to perform a safety-grade function. Further, the Board found that if a PORV malfunctioned and failed to close and its block valve failed to close and isolate the PORV, the capability of the Emergency Core Coolant System would be sufficient to permit safe shutdown of the reactor without the core being uncovered and damaged. Therefore, the PORV control circuitry at DCPP is acceptable.

The presentation above, demonstrating that the valves operated satisfactorily, verifies that the portion of Item 1 of Section 1.2 that requires conducting tests to qualify the valves and that part of Item 7 requiring that the effect of discharge piping on operability be considered have been met.

4.4 Piping and Support Evaluation

In the piping and support evaluation, the safety/relief valve piping between the pressurizer nozzles and pressurization relief tank was analyzed for requirements of the ANSI B31.1-1967 Code for Class A piping and B31.7-1969 with 1970 addenda for Class B and C piping. Stress equations were taken from ANSI B31.1-1973 Summer Addenda since the earlier code did not address all loading cases. The piping was analyzed for thermal expansion, pressure, weight, earthquake, plant operational thermal and pressure transients, and safety valve and relief valve discharges. The load combinations and acceptance criteria were based on EPRI piping subcommittee recommendations as presented in Reference 15. The service stress limits used were more conservative than those of ASME Code Section III Division 1 Class 2, which are considered adequate to demonstrate that the discharge piping would not deform in a manner that would significantly restrict the discharge flow of the safety valves or PORVs.

The safety and relief valve discharge loads were calculated for the fluid transient condition that will produce the most severe loading on the piping system. This occurs during a high pressure steam transient where steam from the pressurizer forces the water in the water seal through the safety or relief valve down the piping system to the relief tank. Forcing functions were generated for both hot and cold loop seals and were found to be highest in magnitude for the cold loop seal case. The computer code ITCHVALVE was used to perform the thermal hydraulic analysis for this transient. The unbalanced fluid forces for each straight segment of piping were calculated using the program FORFUN. The capability of these two programs to calculate accurate fluid force time histories for loop seal

discharges was demonstrated by comparing calculated force histories using ITCHVALVE and FORFUN with force histories generated from tests conducted by EPRI at the Combustion Engineering facility (Reference 12). The comparisons showed that the magnitudes of the forces in the calculated force-time histories generally agree well with those of the test time histories. Additionally, the calculated and test histories had similar shapes, though (as would be expected) the calculated histories did not always reflect all of the fluctuations evident in the test histories. Since favorable comparisons were obtained between the calculated and test results, this is considered to be an adequate verification of the capability of the ITCHVALVE and FORFUN programs.

The code ITCHVALVE uses the Method of Characteristics approach to generate fluid parameters as a function of time. The code FORFUN calculates fluid forces from the fluid parameters using the momentum balance equation. To perform the hydraulic analysis, thermal hydraulic models of the safety and relief valve piping from the pressurizer to the relief tank were developed. The analysis was conducted using the same approach as was used in the comparison analysis discussed above, where a good comparison between calculated and test data was achieved. The node spacing and time step size were selected to be consistent with the values used in the comparison study. The time step used in that analysis yielded stable force-time histories. The water slug was modeled with a temperature profile consistent with that of the comparison study and was assumed to be upstream of the valve prior to transient initiation. For the hot loop seal case, the loop seal insulation was designed to assure that the temperature distribution of the seal was consistent with that assumed in the analysis. Additionally, the licensee stated that confirmatory measurements of temperature at the end of the loop seal would be made prior to criticality. At time zero in the analysis, the transient was initiated and the loop seal water slug position was analytically calculated during and subsequent to valve opening.

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

Valve flow areas were based on actual flow data with margins applied to account for flow rate uncertainties. The rated steam flow through the Masoneilan PORV's is 210,000 lb/hr at 2350 psia. Values of 228,600 and 230,400 lb/hr were observed in EPRI/Wyle tests at pressures of 2745 and 2780 psia, respectively. To account for uncertainties in valve flow rates, the flow area was conservatively adjusted. The minimum analytically calculated steam flow for each PORV was greater than 275,000 lb/hr, which is 131 percent of rated. The analysis assumed a linear full PORV opening in 1.0 second. Full opening times in the tests averaged 2.77 seconds with a minimum value of 1.64 seconds for opening on steam. The nominal flow rating for the Crosby safety valves is 420,000 lb/hr at 2500 psia and 3% accumulation. A flow rate of 460,000 lb/hr at 2700 psia was observed in tests. To account for uncertainties in the valve flow rate, a flow of 119% of rated was used in the analysis. The valves were assumed to fully open in a linear manner in 0.04 s. This opening time produced favorable comparisons with test data in the comparison study previously mentioned. The conservative valve flow areas used in the analysis acceptably account for 10% ASME derating and potential error in the flow rate.

Two valve opening cases were addressed in the analysis. One was a simultaneous opening of the three safety valves, the other a simultaneous opening of the three relief valves. This approach is reasonable since the three safety valves are identical and have the same set pressure. Likewise, the relief valves are identical and have an equal set pressure. Maximum forces in the common header region of the piping system could theoretically be expected when the opening sequencing is such that the initial pressure waves from valve opening reach the common junction downstream simultaneously. This event is unlikely, however, because the valves would be required to open at times perfectly spaced to compensate for differing pipe lengths leading to the common junction. More

importantly, the common region is sufficiently isolated from the valves and pressurizer by a significant amount of piping and dynamic supports between the valve outlets and common junction that valve operability and nozzle loading would not be significantly affected. Thus, the assumption of simultaneous valve openings is acceptable.

The structural analysis was performed using the WESTDYN computer program. The piping deflection solution for static loads was obtained by the transfer matrix method. In this method, transfer matrices for all pipe sections were defined, then an overall stiffness matrix was developed. The seismic analysis was performed by modifying the model used in the static analysis to include mass characteristics of the piping and equipment, then using the response spectrum method. The analysis for safety/relief valve discharge loading was performed using essentially the same model as was used for the seismic analysis. Time-history forces determined by FORFUN were applied at piping system lumped mass points. The dynamic solution was obtained by using a modified-predictor-corrector-integration technique and normal mode theory. The time-history piping displacement response was determined with the FIXFM3 program. Input to this program consisted of natural frequencies and normal modes that were determined with the WESTDYN program and applied forces from FORFUN. Results from the FIXFM3 analysis were then used as input to WESTDYN2 to determine internal forces and deflections at the ends of each pipe element. These results were in turn used as input to POSDYN2 to determine maximum forces, moments and displacements at the ends of pipe elements and maximum loads for pipe supports.

The programs used in the structural analysis have previously been reviewed and approved by the NRC (letter dated April 7, 1981 from R. L. Tedesco to T. M. Anderson). In the analysis a system damping of 2 percent of critical was assumed, which is in accord with the damping specified by U.S. Regulatory Guide 1.61. Lumped masses were spaced to ensure that all appropriate mode shapes were accurately represented. Supports were modeled with linear stiffness elements. The integration time-step was internally determined within the structural program based on

convergence criteria for stable solutions. The largest time-step that can be so selected is 0.0001 seconds, which is sufficiently small for safety/relief valve discharges.

Potential axial extension caused by balancing forces at the ends of pipe segments was evaluated in the analysis. The effect of these forces was determined to be negligible relative to the unbalanced forces for this application. In structural analysis comparisons with test results, maximum support and pipe loads compared well between analysis and test results. Maximum displacement values downstream of the safety valve also compared well.

The safety valve bonnet assemblies and relief valve actuators were modeled as extended masses, displaced from the pipe centerline. The valve stem diameter and thickness were represented in the model to obtain the appropriate frequencies.

The analysis of the safety/relief valve piping system for a cold loop seal-safety valve discharge revealed an overstressed region upstream of, into and along the common header at the safety valve discharge line branch point. Thus, it was decided to make a system modification, which entails insulating the piping between the pressurizer and the safety valves. This significantly reduces the magnitude of the thermal hydraulic forces. The analysis of the piping system with a hot safety valve loop seal yielded acceptable stress results throughout the system. Additionally, a simultaneous discharge of all the relief valves was found to be acceptable. In the hot loop seal-safety valve discharge case, the loop seal temperature was assumed to vary between 300°F at the valve inlet to 655°F eight feet upstream. This temperature distribution was based on an analysis of the loop seal to assure that the assumed temperature at the end of the loop seal was accurate. Additionally, the licensee stated that measurements at the end of the loop seal would be made prior to criticality to confirm the assumed temperature.

The seismic design verification included analyses of the system for "Design," "Double Design," and "Hosgri" earthquakes. The results of these analyses indicated that support additions and modifications were required to qualify the system. These modifications have been implemented in the field for Units 1 and 2. In addition, the licensee stated that the preliminary seismic response spectra as used in the analysis would be checked for envelopment of the final seismic spectra. With the implementation of field modifications and verification of appropriate response spectra, the seismic analysis is acceptable.

According to results of EPRI tests performed on the Crosby 6M6 safety valve, high frequency pressure oscillations of 170-260 Hz occurred in the piping upstream of the safety valve as loop seal water passed through the valve. This raises a concern that these oscillations could potentially excite high frequency vibration modes in the inlet piping that could contribute to higher bending moments in the piping. This phenomenon was not accounted for in the structural analysis of the system. The piping between the pressurizer and safety valves in the EPRI tests, however, was composed of 8-in. Schedule 160 and 6-in. Schedule XX while that at DCPP is 6-in. Schedule 160. Since the test piping did not sustain any discernable damage during pressure oscillations occurring in the tests, it is expected that the plant piping also would not incur damage during similar oscillations. Thus, a specific analysis for these pressure oscillations is not necessary for this plant.

With the modifications made to the safety/relief valve piping system, the calculated stresses for all applicable loadings and load combinations meet acceptance criteria. Calculated stresses come nearest to allowable stresses in the piping upstream of the relief valve under an upset condition (normal sustained loads plus "Design" earthquake plus relief valve discharge transient). The maximum calculated stress was 17.16 ksi versus an allowable stress of 19.08 ksi.

The discussion above demonstrating that a bounding loading case has been chosen for the piping evaluation verifies that Item 3 of Section 1.2 has been met. The analysis of the piping and support system verifies that Item 8 has been met provided that the assumed hot loop seal temperature distribution is confirmed by measurement and that appropriate modifications to the system are made.

5. EVALUATION SUMMARY

The licensee for Diablo Canyon 1 and 2 has provided an acceptable response to the requirements of NUREG-0737, reconfirming that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met with regard to the safety valves and PORVs. The rationale for this conclusion is given below.

The licensee developed an acceptable relief and safety valve test program to qualify the operability of the prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. This conclusion is based on an acceptable demonstration that a feedline break event will not result in liquid discharge through the safety valves/PORVs. The subsequent tests were successfully completed under operating conditions which by analysis bound the most probable maximum forces expected from anticipated design basis events. The test results showed that the valves tested functioned correctly and safely for all steam and water discharge events specified in the test program that are applicable to Diablo Canyon Units 1 and 2 and that the pressure boundary component design criteria were not exceeded. Analysis and review of both the test results and the licensee justifications indicated the direct applicability of prototypical valves and piping performance to the in-plant valves and piping intended to be covered by the test program. The plant-specific piping also has been shown by analysis to be acceptable.

Thus, the requirements of Item II.D.1 of NUREG-0737 have been met (Items 1-8 in Paragraph 1.2) and, thereby, ensure that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14). In addition, the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) have been designed with a sufficient margin so that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15).

Further, the prototypical tests and the successful performance of the valves and associated components demonstrated that this equipment was constructed in accordance with high quality standards, meeting General Design Criterion No. 30.

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