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Docket Nos.: 50-361
and 50-362

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Gentlemen:

Subject: Safety Evaluation Report for NUREG-0737 Item II.D.1

The enclosed Safety Evaluation Report addresses NUREG-0737 Item II.D.1., Safety and Relief Valve Testing. This issue is also designated Multi-Plant Action (MPA) F-14.

Via the technical assistance of Idaho National Engineering Laboratory (EG&G) we have completed our review of information submitted concerning testing of safety valves for San Onofre 2 and 3. We find the information submitted demonstrates the ability of the reactor coolant system safety valves and associated piping to function under expected operating conditions for design-base transients and accidents as defined under NUREG 0737, Item II.D.1.

This completes our review of this issue for San Onofre 2 and 3.

Sincerely, /S/

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Division of Licensing

Enclosure: As stated

cc: G. Hammer
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SAFETY EVALUATION REPORT

TMI ACTION--NUREG-0737 (II.D.1)

SAFETY VALVE TESTING FOR

SAN ONOFRE NUCLEAR STATION UNITS 2 AND 3

DOCKET NO. 50-0361 AND 50-0362

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 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) 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 system 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 relief and safety valve 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 and 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, safety valves and block valves and associated piping systems. Southern California Edison, the owner of San Onofre Units 2 and 3, 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 overpressure protection would be required (Reference 6).

The utilities participating in the EPRI Safety and Relief Valve Test Program also obtained information regarding the performance of block valves (Reference 7). 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 8).

EPRI contracted with Combustion Engineering Inc. (CE) to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Combustion Engineering designed plants (Reference 9). This report was referenced in the submittal (Reference 10) and treated as a part of the submittal for the purpose of this evaluation.

Several test series were sponsored by EPRI but the test series that is relevant to the San Onofre reactor units 2 and 3 is the series for safety valves which was conducted at the C-E test facility located at the Combustion Engineering, Inc. Kreisinger Development Laboratory, Windsor, Connecticut. The results of these tests are summarized in References 11 and 12. Detailed test results are presented in Reference 13. These reports were referenced in the submittal and were also treated as part of the submittal for the purpose of this evaluation.

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 Southern California Edison (SCE) Company submitted a preliminary safety valve operability report (Reference 14) on April 1, 1982. Their Pressurizer Safety Valve Operability and Safety Valve Discharge Piping Adequacy Report (Reference 10) followed on June 29, 1982. The letter report includes two subreports in the form of appendices. One is a comparison of the San Onofre Units 2 and 3 safety valve inlet piping with the EPRI/C-E valve test inlet piping. The other is a justification for increased San Onofre Units 2 and 3 safety valve blowdown. Requests for, additional information (References 15 and 16) were submitted to SCE by the NRC on June 27, 1983 and January 12, 1984. SCE responded on October 14, 1983 and March 14, 1984 (References 17 and 18).

The submittals, and responses to requests for additional information and relevant EPRI reports were reviewed to evaluate compliance with the requirements of NUREG-0737.

The response of the overpressure protection system to Anticipated Transients Without Scram (ATWS) and the operation of the system during feed and bleed decay heat removal are not considered in this review. Neither the licensee nor the NRC have evaluated the performance of the system for these events.

4. REVIEW AND EVALUATION

4.1 Valves Tested

The San Onofre Units 2 and 3 utilize only two safety valves for overpressure protection and do not use power operated relief valves. Dresser Model 31709NA Safety Valves are used in the San Onofre Units. This valve is included in the list of valves tested in the EPRI/C-E test program. The ring adjustments currently used on the San Onofre Units 2 and 3 valves are the same as was used in 7 of the tests conducted in the EPRI/C-E test program (Reference 11).

During the test series, valve repairs or modifications were made to correct problems related to valve operation. The thickness of the disc holder was reduced and the lower lip of the disc holder was machined to reestablish the clearance between the valve disc and the disc holder. The need for this modification was caused by a new bellows which elongated after the valve was cycled. The licensee stated in Reference 17 that this problem was avoided in the plant valves by the manufacturer cycling the valves prior to shipment.

Also during the test series the thrust bearing adapter was remachined to prevent the outer surface of the spaces from contacting the inner surface of the adapter by removing a burr. SCE stated that the burr was most likely caused by the numerous set pressure adjustments made to the valve during the test program. This problem is not expected in the plant valves.

It was therefore concluded by SCE that no modifications or repairs were required of the plant valves. It was further stated by SCE that the modification and repair made on the test valve were not required to assure valve operability.

The pressurizer relief system in the San Onofre units 2 and 3 does not include relief or block valves. The only valves relevant to this evaluation are the two Dresser Model 31709NA safety valves. Since the test valve was the same model as the plant valves and tests were conducted with the ring settings used at the plant, the EPRI/C-E tests are considered adequate to meet the requirements of Items 1 and 7 of the criteria (Section 1.2) regarding the selection of the valves tested. Also, based on the SCE response (Reference 17), modifications to the plant valves are not considered necessary.

4.2 Test Conditions

Chapter 15 of the FSAR for the San Onofre Units 2 and 3 addresses the accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2. A feedwater line break (FWLB) is identified as the accident having the greatest potential for driving the pressurizer water level to the safety valve inlet. The FWLB transient is also identified as the accident that bounds the peak pressurizer pressure (Reference 9). The peak pressure, 2670 psia (Reference 17), is based on an assumed valve setpoint of 2525 psia. Reference 17 also identifies loss of offsite electrical power as the single failure that meets the requirements of NUREG-0737.

The SCE adequacy report (Reference 10) is based on the assumption that only steam flow through the valves need be considered. The staff considered the possibility that extended blowdown could produce results that go beyond the analysis of the FSAR and that there could be events that could result in a high water level in the pressurizer or loss of adequate core cooling. Also considered was the effect that the pressurizer spray could have on the steam-only assumption. In particular, consideration was given to the possibility that failure of the pressurizer level controller followed by inaction by the operator could result in water at the safety valve inlet. The investigation indicated that in such an event there is substantial time (approximately 30 minutes per the FSAR) for operator action following failure of the level controller and annunciation of the failure.

The concerns for adequacy of core cooling and verification of steam-only conditions at the valve inlet during extended blowdown were addressed by SCE in Reference 17, where it is stated that the FWLB accident was reanalyzed for extended blowdown (12%) which verified that steam-only conditions will exist at the valve inlet and that adequate core cooling will be maintained. Reference 17 also states that the pressurizer spray will not increase piping loads because the peak load occurs prior to the time when any wet steam due to entrained spray can reach the safety valves. The staff accepts the SCE response to the concerns listed above.

The EPRI/C-E tests of the Dresser 31709NA valve included eight separate steam tests of which two were with the same ring settings as used at the San Onofre Units 2 and 3 plants. Steam-to-water transition and subcooled water tests were also conducted on the valve but are assumed to not be directly applicable to the San Onofre evaluation. The two applicable steam tests reached peak pressure sufficiently close to the predicted peak plant pressure to provide adequate demonstration of the performance of the valves.

Based on the above information, criteria Items 1 through 4 and part of Item 7 (Section 1.2) have been met. These items include the requirements for (1) testing the valves under expected conditions, (2) meeting the requirements of Regulatory Guide 1.70, Rev. 2, (3) identifying the single failure that maximizes dynamic forces, (4) testing to the highest pressure predicted by analysis and (7) supplying correlating evidence relating test condition to expected operating and accident conditions.

4.3 Operability

The San Onofre Units 2 and 3 utilize only two safety valves in a short inlet configuration for overpressure protection of the primary coolant system. Cold overpressure protection against brittle failure is not a function of these valves because protection for these events is provided within the Shutdown Cooling System. Similarly, protection from high pressure injection events is not a requirement of the safety valve because the shutoff head of the high pressure safety injection pump (approximately 1500 psia) is less than the normal operating pressure (2250 psia) and the nominal safety valve setpoint (2500 psia).

During the actual EPRI/C-E steam tests the Dresser Model 31709NA valve with the same ring settings as the plant valves did not always achieve full rated lift at 3% accumulation. However, the valve exceeded rated flow in all the steam tests. Some steam-to-water transition and water tests were also conducted on the same valve and ring settings, although these are assumed to have no direct significance to the operation of the San Onofre plants. The valve performed satisfactorily in a steam-to-water transition test and some water tests but in one 400°F water test the valve performed erratically.

The EPRI/C-E tests showed that the Dresser 31709NA valve blowdown was dependent on discharge backpressure. The SCE adequacy report states that a backpressure of 427 psig was calculated for two safety valves discharging as compared to 245 psig for a single valve. The calculated backpressure for simultaneous lifting of two safety valves exceeded the maximum backpressure for the tests conducted with the ring settings of the plant valves. It was also noted that a linear plot of the limited test data was used to extrapolate to higher backpressure and corresponding smaller blowdowns. Small blowdowns are a concern because with shorter blowdown the probability for unstable valve operation is increased.

In Reference 17, SCE provided a justification for the linear extrapolation of decreasing blowdowns vs. backpressure for the range of pressures expected based on the ASME Paper 82-WA/NE-9. The response states that the maximum expected pressure drop at the valve inlet will remain low enough to assure stable operation for a minimum predicted blowdown of 3.5%.

SCE also provided a comparison of the measured flow characteristics of the valves with those used in the FSAR analyses to demonstrate that overpressure transients will be limited to 110% of the design pressure. Curves were presented in Reference 17 that compared the trajectory of valve opening for the tests vs. the description in the FSAR. The test valve is reported to open to full open at approximately the setpoint pressure rather than opening to only 70% as was assumed in the FSAR analysis. A review of the test data indicates that the test valve did not always go to full open at the setpoint (i.e. test 615 had 83% lift at 3% accumulation). The valve, however, always went to greater than 100% rated flow and did always go to 100% rated lift at 6% accumulation. It is therefore concluded that the valves will pass sufficient flow to limit pressure to 110% of design.

The inlet piping to the San Onofre Units 2 and 3 safety valves was compared to the inlet piping used in the EPRI/C-E tests setup in the appendix to Section 8 of the SCE submittal (Reference 10). The conclusion presented is that the plant inlet pipe configuration provides a lower pressure drop upon valve actuation than the test configuration.

The requirement (Item 5, Section 1.2) for qualifying the associated control circuitry on the safety valves of the San Onofre Units 2 and 3 is not a concern since the Dresser valves are direct acting without control circuitry. Any other circuits associated with the valves such as position indication are subject to review for other requirements and have not been considered in this evaluation.

Based on the above information the requirements of Items 5, 6, and parts of 7 of Section 1.2 have been met. These items include the requirements for (5) qualification of the associated control circuitry, (6) criteria for the success or failure of the valves tested, and (7) the test program that demonstrates the functionability of the as-installed safety valves.

4.4 Piping and Support Evaluation

4.4.1 Safety Valve Inlet Configuration

The submittal does not include any stress analysis of the inlet piping. However, SCE states in Reference 17 that the inlet piping is part of the ASME Class 1 piping that was required to meet the requirements of paragraph NB 3650 of the ASME Section III Code, 1974 Edition. The loading combinations considered were those specified in the San Onofre project design specifications which are consistent with the load combinations considered in the FSAR. The safety valve discharge loads were considered in the design condition and were combined with the loads from thermal expansion and movement of the pressurizer nozzle in the fatigue analysis. The documentation of the analysis is subject to the 10CRF50, Appendix B quality assurance program. SCE further states that the forces and moments predicted to act on the valves in the San Onofre Units 2 and 3 piping analysis are less than those measured during the test program.

4.4.2 Safety Valve Discharge Piping

Part C of Reference 10, supplemented by References 17 and 18, addresses the adequacy of the safety valve discharge piping. The hydraulic analysis was performed using RELAP4. RELAP4 was shown to be adequate for prediction of discharge piping hydrodynamic loads for a steam discharge by comparison with RELAP5. RELAP5 was previously shown to be a suitable tool for the prediction of discharge loads (Reference 19). In the San Onofre model the key parameters of time steps and choked flow modes were acceptable but only two control volumes were included in the first horizontal leg downstream of the safety valve, which is much less than the ten control volumes recommended by Reference 19. The length of the control volumes in the San Onofre model, however, are within the range of the 1.0 foot used in the model of Reference 15 and SCE provided data to verify that this noding would provide adequate results (Reference 18). ANSYR was used as the interface computer code to couple RELAP4 with the structural analysis code, ANSYS. The ANSYR code uses the derivative of the mass velocity to compute the wave force, which is acceptable for a steam only discharge.

The structural analysis was performed using the computer program ANSYS which is a structural program with wide use in industry for problems of this type. The key parameters of node spacing and damping were acceptable and the method of reducing the degree of freedom described in Reference 10, supplemented by Reference 17 and Reference 18, was considered acceptable. The loads from valve discharge were considered as Service Level C and only primary stresses were considered with a stress limit of $1.8S_h$. With the original supports in place, one support was shown to be overloaded. The analyses showed that with the support removed the loads were distributed such that the piping stresses and support loads were within the required limits. The designation of safety valve discharge as Service Level C with a stress limit of $1.8S_h$ is considered acceptable since it will assure that the piping will not deform in a way that would restrict flow.

Two loadings were considered for the safety valve discharge piping. One was the seismic load alone. The other was the worst of three selected events which were: (1) both valves opening simultaneously, (2) one valve opening and flow reaching steady state followed by second valve opening, and (3) the second valve opening when the first valve reaches half-way. The evaluation of seismic loading and the loading from safety valves lifting as separate events was previously considered by the NRC during the FSAR review and was judged to be acceptable for this generation of CE plants. The acceptance was based on the premise of the licensee that the lifting of a safety is a rare event and the probability of the peak load from the safety valve lift occurring simultaneously with the peak load from an earthquake is extremely small. Although all possible combinations of relief valve opening were not considered, the three selected are considered sufficiently representative such that near maximum load has been computed and the loading combination used are considered acceptable.

5. EVALUATION SUMMARY

The Licensee for the San Onofre Units 2 and 3 has provided an acceptable response to the requirements of NUREG-0737, and thereby, reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met. The rationale for this conclusion is given below.

The licensee participated in the development and execution of an acceptable Relief and Safety Valve Test Program designed 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. The subsequent tests were successfully completed under operating conditions which by analysis bounded the most probable maximum forces expected from anticipated design basis events. The generic test results and piping analyses showed that the valves tested functioned correctly and safely for all relevant steam discharge events specified in the test program and that the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the licensee justifications indicated direct applicability of the tested valve to the performances of the in-plant valves and systems intended to be covered by the test program.

Thus, the requirements of Item II.D.1 of NUREG 0737 have been met (Items 1-8 in Paragraph 1.2) and, thereby demonstrate by testing and analysis, that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14) and that the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) have been designed with sufficient margin such 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 has been constructed in accordance with high quality standards (General Design Criterion 30).

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2. Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.

3. Letter, D. P. Hoffman, Consumer Power Co. to H. Denton, NRC, Transmittal of PWR Safety and Relief Valve Test Program Reports, September 30 1982.

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5. EPRI PWR Safety and Relief Valve Test Program Valve Selection/Justification Report, EPRI NP-2292, December 1982.

6. EPRI PWR Safety and Relief Valve Test Program Test Condition Justification Report, EPRI NP-2460, December 1982.

7. EPRI Marshall Electric Motor Operated Block Valve, EPRI NP-2514-LD, July 1982.

8. EPRI Summary Report: Westinghouse Gate Valve Closure Testing Program, Engineering Memorandum 5683, Revision 1, March 31, 1982.

9. Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Combustion Engineering-Designed Plants, EPRI NP-2318, December 1982.

10. Transmittal Letter, K. P. Baskin, SCE to F. Miraglia, NRC, Pressurizer Safety Valve Operability and Safety Valve Discharge Piping Adequacy Report for San Onofre Units 2 and 3, June 29, 1982.

11. EPRI PWR Safety and Relief Test Program Safety and Relief Valve Test Report, EPRI NP-2628-SR, December 1982.

12. EPRI/C-E Safety Valve Test Report, Volume 1 of 10 Summary, July 1982.

13. EPRI/C-E Safety Valve Test Report Volume 4 of 10, Test Results for Dresser Safety Valves, Model 31709NA, July 1982.

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15. Letter, G. W. Knighton, NRC to R. Dietch, SCE, Request for Additional Information-NUREG 0737, Item II D.1-Performance Testing of Relief and Safety Valves, San Onofre 2 and 3, June 27, 1983.

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17. Letter, M. O. Medford, SCE to G. W. Knighton, NRC, Response to NRC Questions on Pressurizer Safety Valves, San Onofre Nuclear Generation Station Units 2 and 3, October 14, 1983.

18. Letter, M. O. Medford, SCE to G. W. Knighton, NRC, Response to Questions on Pressurizer Safety Valves and Discharge Piping San Onofre Nuclear Generation Station Units 2 and 3, March 14, 1984.

19. Application of RELAP5/MOD1 for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads, EPRI-2479, December 1982.

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SAFETY EVALUATION REPORT

TMI ACTION--NUREG-0737 (II.D.1)

SAFETY VALVE TESTING FOR

SAN ONOFRE NUCLEAR STATION UNITS 2 AND 3

DOCKET NO. 50-0361 AND 50-0362

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 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) 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 system 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 relief and safety valve 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 and 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, safety valves and block valves and associated piping systems. Southern California Edison, the owner of San Onofre Units 2 and 3, 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 overpressure protection would be required (Reference 6).

The utilities participating in the EPRI Safety and Relief Valve Test Program also obtained information regarding the performance of block valves (Reference 7). 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 8).

EPRI contracted with Combustion Engineering Inc. (CE) to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Combustion Engineering designed plants (Reference 9). This report was referenced in the submittal (Reference 10) and treated as a part of the submittal for the purpose of this evaluation.

Several test series were sponsored by EPRI but the test series that is relevant to the San Onofre reactor units 2 and 3 is the series for safety valves which was conducted at the C-E test facility located at the Combustion Engineering, Inc. Kreisinger Development Laboratory, Windsor, Connecticut. The results of these tests are summarized in References 11 and 12. Detailed test results are presented in Reference 13. These reports were referenced in the submittal and were also treated as part of the submittal for the purpose of this evaluation.

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 Southern California Edison (SCE) Company submitted a preliminary safety valve operability report (Reference 14) on April 1, 1982. Their Pressurizer Safety Valve Operability and Safety Valve Discharge Piping Adequacy Report (Reference 10) followed on June 29, 1982. The letter report includes two subreports in the form of appendices. One is a comparison of the San Onofre Units 2 and 3 safety valve inlet piping with the EPRI/C-E valve test inlet piping. The other is a justification for increased San Onofre Units 2 and 3 safety valve blowdown. Requests for additional information (References 15 and 16) were submitted to SCE by the NRC on June 27, 1983 and January 12, 1984. SCE responded on October 14, 1983 and March 14, 1984 (References 17 and 18).

The submittals, and responses to requests for additional information and relevant EPRI reports were reviewed to evaluate compliance with the requirements of NUREG-0737.

The response of the overpressure protection system to Anticipated Transients Without Scram (ATWS) and the operation of the system during feed and bleed decay heat removal are not considered in this review. Neither the licensee nor the NRC have evaluated the performance of the system for these events.

4. REVIEW AND EVALUATION

4.1 Valves Tested

The San Onofre Units 2 and 3 utilize only two safety valves for overpressure protection and do not use power operated relief valves. Dresser Model 31709NA Safety Valves are used in the San Onofre Units. This valve is included in the list of valves tested in the EPRI/C-E test program. The ring adjustments currently used on the San Onofre Units 2 and 3 valves are the same as was used in 7 of the tests conducted in the EPRI/C-E test program (Reference 11).

During the test series, valve repairs or modifications were made to correct problems related to valve operation. The thickness of the disc holder was reduced and the lower lip of the disc holder was machined to reestablish the clearance between the valve disc and the disc holder. The need for this modification was caused by a new bellows which elongated after the valve was cycled. The licensee stated in Reference 17 that this problem was avoided in the plant valves by the manufacturer cycling the valves prior to shipment.

Also during the test series the thrust bearing adapter was remachined to prevent the outer surface of the spaces from contacting the inner surface of the adapter by removing a burr. SCE stated that the burr was most likely caused by the numerous set pressure adjustments made to the valve during the test program. This problem is not expected in the plant valves.

It was therefore concluded by SCE that no modifications or repairs were required of the plant valves. It was further stated by SCE that the modification and repair made on the test valve were not required to assure valve operability.

The pressurizer relief system in the San Onofre units 2 and 3 does not include relief or block valves. The only valves relevant to this evaluation are the two Dresser Model 31709NA safety valves. Since the test valve was the same model as the plant valves and tests were conducted with the ring settings used at the plant, the EPRI/C-E tests are considered adequate to meet the requirements of Items 1 and 7 of the criteria (Section 1.2) regarding the selection of the valves tested. Also, based on the SCE response (Reference 17), modifications to the plant valves are not considered necessary.

4.2 Test Conditions

Chapter 15 of the FSAR for the San Onofre Units 2 and 3 addresses the accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2. A feedwater line break (FWLB) is identified as the accident having the greatest potential for driving the pressurizer water level to the safety valve inlet. The FWLB transient is also identified as the accident that bounds the peak pressurizer pressure (Reference 9). The peak pressure, 2670 psia (Reference 17), is based on an assumed valve setpoint of 2525 psia. Reference 17 also identifies loss of offsite electrical power as the single failure that meets the requirements of NUREG-0737.

The SCE adequacy report (Reference 10) is based on the assumption that only steam flow through the valves need be considered. The staff considered the possibility that extended blowdown could produce results that go beyond the analysis of the FSAR and that there could be events that could result in a high water level in the pressurizer or loss of adequate core cooling. Also considered was the effect that the pressurizer spray could have on the steam-only assumption. In particular, consideration was given to the possibility that failure of the pressurizer level controller followed by inaction by the operator could result in water at the safety valve inlet. The investigation indicated that in such an event there is substantial time (approximately 30 minutes per the FSAR) for operator action following failure of the level controller and annunciation of the failure.

The concerns for adequacy of core cooling and verification of steam-only conditions at the valve inlet during extended blowdown were addressed by SCE in Reference 17, where it is stated that the FWLB accident was reanalyzed for extended blowdown (12%) which verified that steam-only conditions will exist at the valve inlet and that adequate core cooling will be maintained. Reference 17 also states that the pressurizer spray will not increase piping loads because the peak load occurs prior to the time when any wet steam due to entrained spray can reach the safety valves. The staff accepts the SCE response to the concerns listed above.

The EPRI/C-E tests of the Dresser 31709NA valve included eight separate steam tests of which two were with the same ring settings as used at the San Onofre Units 2 and 3 plants. Steam-to-water transition and subcooled water tests were also conducted on the valve but are assumed to not be directly applicable to the San Onofre evaluation. The two applicable steam tests reached peak pressure sufficiently close to the predicted peak plant pressure to provide adequate demonstration of the performance of the valves.

Based on the above information, criteria Items 1 through 4 and part of Item 7 (Section 1.2) have been met. These items include the requirements for (1) testing the valves under expected conditions, (2) meeting the requirements of Regulatory Guide 1.70, Rev. 2, (3) identifying the single failure that maximizes dynamic forces, (4) testing to the highest pressure predicted by analysis and (7) supplying correlating evidence relating test condition to expected operating and accident conditions.

4.3 Operability

The San Onofre Units 2 and 3 utilize only two safety valves in a short inlet configuration for overpressure protection of the primary coolant system. Cold overpressure protection against brittle failure is not a function of these valves because protection for these events is provided within the Shutdown Cooling System. Similarly, protection from high pressure injection events is not a requirement of the safety valve because the shutoff head of the high pressure safety injection pump (approximately 1500 psia) is less than the normal operating pressure (2250 psia) and the nominal safety valve setpoint (2500 psia).

During the actual EPRI/C-E steam tests the Dresser Model 31709NA valve with the same ring settings as the plant valves did not always achieve full rated lift at 3% accumulation. However, the valve exceeded rated flow in all the steam tests. Some steam-to-water transition and water tests were also conducted on the same valve and ring settings, although these are assumed to have no direct significance to the operation of the San Onofre plants. The valve performed satisfactorily in a steam-to-water transition test and some water tests but in one 400°F water test the valve performed erratically.

The EPRI/C-E tests showed that the Dresser 31709NA valve blowdown was dependent on discharge backpressure. The SCE adequacy report states that a backpressure of 427 psig was calculated for two safety valves discharging as compared to 245 psig for a single valve. The calculated backpressure for simultaneous lifting of two safety valves exceeded the maximum backpressure for the tests conducted with the ring settings of the plant valves. It was also noted that a linear plot of the limited test data was used to extrapolate to higher backpressure and corresponding smaller blowdowns. Small blowdowns are a concern because with shorter blowdown the probability for unstable valve operation is increased.

In Reference 17, SCE provided a justification for the linear extrapolation of decreasing blowdowns vs. backpressure for the range of pressures expected based on the ASME Paper 82-WA/NE-9. The response states that the maximum expected pressure drop at the valve inlet will remain low enough to assure stable operation for a minimum predicted blowdown of 3.5%.

SCE also provided a comparison of the measured flow characteristics of the valves with those used in the FSAR analyses to demonstrate that overpressure transients will be limited to 110% of the design pressure. Curves were presented in Reference 17 that compared the trajectory of valve opening for the tests vs. the description in the FSAR. The test valve is reported to open to full open at approximately the setpoint pressure rather than opening to only 70% as was assumed in the FSAR analysis. A review of the test data indicates that the test valve did not always go to full open at the setpoint (i.e. test 615 had 83% lift at 3% accumulation). The valve, however, always went to greater than 100% rated flow and did always go to 100% rated lift at 6% accumulation. It is therefore concluded that the valves will pass sufficient flow to limit pressure to 110% of design.

The inlet piping to the San Onofre Units 2 and 3 safety valves was compared to the inlet piping used in the EPRI/C-E tests setup in the appendix to Section B of the SCE submittal (Reference 10). The conclusion presented is that the plant inlet pipe configuration provides a lower pressure drop upon valve actuation than the test configuration.

The requirement (Item 5, Section 1.2) for qualifying the associated control circuitry on the safety valves of the San Onofre Units 2 and 3 is not a concern since the Dresser valves are direct acting without control circuitry. Any other circuits associated with the valves such as position indication are subject to review for other requirements and have not been considered in this evaluation.

Based on the above information the requirements of Items 5, 6, and parts of 7 of Section 1.2 have been met. These items include the requirements for (5) qualification of the associated control circuitry, (6) criteria for the success or failure of the valves tested, and (7) the test program that demonstrates the functionability of the as-installed safety valves.

4.4 Piping and Support Evaluation

4.4.1 Safety Valve Inlet Configuration

The submittal does not include any stress analysis of the inlet piping. However, SCE states in Reference 17 that the inlet piping is part of the ASME Class 1 piping that was required to meet the requirements of paragraph NB 3650 of the ASME Section III Code, 1974 Edition. The loading combinations considered were those specified in the San Onofre project design specifications which are consistent with the load combinations considered in the FSAR. The safety valve discharge loads were considered in the design condition and were combined with the loads from thermal expansion and movement of the pressurizer nozzle in the fatigue analysis. The documentation of the analysis is subject to the 10CRF50, Appendix B quality assurance program. SCE further states that the forces and moments predicted to act on the valves in the San Onofre Units 2 and 3 piping analysis are less than those measured during the test program.

4.4.2 Safety Valve Discharge Piping

Part C of Reference 10, supplemented by References 17 and 18, addresses the adequacy of the safety valve discharge piping. The hydraulic analysis was performed using RELAP4. RELAP4 was shown to be adequate for prediction of discharge piping hydrodynamic loads for a steam discharge by comparison with RELAP5. RELAP5 was previously shown to be a suitable tool for the prediction of discharge loads (Reference 19). In the San Onofre model the key parameters of time steps and choked flow modes were acceptable but only two control volumes were included in the first horizontal leg downstream of the safety valve, which is much less than the ten control volumes recommended by Reference 19. The length of the control volumes in the San Onofre model, however, are within the range of the 1.0 foot used in the model of Reference 15 and SCE provided data to verify that this noding would provide adequate results (Reference 18). ANSYR was used as the interface computer code to couple RELAP4 with the structural analysis code, ANSYS. The ANSYR code uses the derivative of the mass velocity to compute the wave force, which is acceptable for a steam only discharge.

The structural analysis was performed using the computer program ANSYS which is a structural program with wide use in industry for problems of this type. The key parameters of node spacing and damping were acceptable and the method of reducing the degree of freedom described in Reference 10, supplemented by Reference 17 and Reference 18, was considered acceptable. The loads from valve discharge were considered as Service Level C and only primary stresses were considered with a stress limit of $1.8S_h$. With the original supports in place, one support was shown to be overloaded. The analyses showed that with the support removed the loads were distributed such that the piping stresses and support loads were within the required limits. The designation of safety valve discharge as Service Level C with a stress limit of $1.8S_h$ is considered acceptable since it will assure that the piping will not deform in a way that would restrict flow.

Two loadings were considered for the safety valve discharge piping. One was the seismic load alone. The other was the worst of three selected events which were: (1) both valves opening simultaneously, (2) one valve opening and flow reaching steady state followed by second valve opening, and (3) the second valve opening when the first valve reaches half-way. The evaluation of seismic loading and the loading from safety valves lifting as separate events was previously considered by the NRC during the FSAR review and was judged to be acceptable for this generation of CE plants. The acceptance was based on the premise of the licensee that the lifting of a safety is a rare event and the probability of the peak load from the safety valve lift occurring simultaneously with the peak load from an earthquake is extremely small. Although all possible combinations of relief valve opening were not considered, the three selected are considered sufficiently representative such that near maximum load has been computed and the loading combination used are considered acceptable.

5. EVALUATION SUMMARY

The Licensee for the San Onofre Units 2 and 3 has provided an acceptable response to the requirements of NUREG-0737, and thereby, reconfirmed that the General Design Criteria 14, 15, and 30 of Appendix A to 10 CFR 50 have been met. The rationale for this conclusion is given below.

The licensee participated in the development and execution of an acceptable Relief and Safety Valve Test Program designed 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. The subsequent tests were successfully completed under operating conditions which by analysis bounded the most probable maximum forces expected from anticipated design basis events. The generic test results and piping analyses showed that the valves tested functioned correctly and safely for all relevant steam discharge events specified in the test program and that the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the licensee justifications indicated direct applicability of the tested valve to the performances of the in-plant valves and systems intended to be covered by the test program.

Thus, the requirements of Item II.D.1 of NUREG 0737 have been met (Items 1-8 in Paragraph 1.2) and, thereby demonstrate by testing and analysis, that the reactor primary coolant pressure boundary will have a low probability of abnormal leakage (General Design Criterion No. 14) and that the reactor primary coolant pressure boundary and its associated components (piping, valves, and supports) have been designed with sufficient margin such 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 has been constructed in accordance with high quality standards (General Design Criterion 30).

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6. EPRI PWR Safety and Relief Valve Test Program Test Condition Justification Report, EPRI NP-2460, December 1982.

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12. EPRI/C-E Safety Valve Test Report, Volume 1 of 10 Summary, July 1982.

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18. Letter, M. O. Medford, SCE to G. W. Knighton, NRC, Response to Questions on Pressurizer Safety Valves and Discharge Piping San Onofre Nuclear Generation Station Units 2 and 3, March 14, 1984.

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