TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2
DOCKET NO. 50-327/50-328

March 1989

Prepared for the

U.S. Nuclear Regulatory Commission

Washington, D.C. 20555

Under a contract to Parameter, Inc.

CONTENTS

TECHNICAL EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

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 were 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 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 (a) the reactor primary coolant pressure boundary be designed, fabricated, and tested so as to have an extremely low probability of abnormal leakage, (b) 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 (c) 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 and 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 OWNERS' 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. The Tennessee Valley Authority, owner of the Sequoyah Nuclear Plant (SQN) 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 is discussed below.

EPRI developed a plan (Reference 4) for testing PWR safety, relief, and block valves under conditions which bound actual plant operating conditions. EPRI, through the valve manufacturers, identified the valves used in the overpressure protection systems 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).

EPRI contracted with Westinghouse Electric Corp. to produce a report on the inlet fluid conditions for pressurizer safety and relief valves in Westinghouse designed plants (Reference 7). Since Sequoyah Units 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. Additional PORV tests were conducted at the Wyle Laboratories Test Facility located in Norco, California. Safety valves were tested at the Combustion Engineering Company, Kressinger Development Laboratory, located in Windsor, Connecticut. The results for the relief and safety valve tests are reported in Reference 8. The results for the block valves 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 used in PWRs 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 (a) obtain valve capacity data, (b) assess hydraulic and structural effects of associated piping on valve operability, and (c) obtain piping response data that could ultimately be used for verifying analytical piping models.

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

3. PLANT SPECIFIC SUBMITTAL

There has been a long history of hardware modifications at SQN Units 1 and 2. The original design for SQN included water-filled loop seals on both the Power-Operated Relief Valves (PORVs) and the Safety Valves. Reanalysis of the discharge piping loads, after completion of the EPRI safety and relief valve program, indicated that loads caused by the discharge of the loop seal slug were excessive. TVA decided to eliminated the PORV loop seals by re-routing the piping, to drain the safety valve loop seals and to install steam trim in the safety valves. Subsequent to these modifications, high safety valve leakage was detected. To remedy the situation, TVA reestablished the loop seals in the safety valves. Modifications were installed to allow operation with a heated loop seal to reduce the magnitude of the piping loads due to slug discharge. Heat tracing and insulation were installed on the loop seals to maintain elevated fluid temperatures.

These elevated temperatures ensured the flashing of the warm loop seal as it was discharged through the valve. Supports on the discharge piping were added or modified to accommodate the higher-than-steam discharge loads. The history of TVA's involvement in the II.D.1 issue till that point is given in References 11 through 15. This new analysis was reviewed by the NRC and a set of questions was formulated and sent to TVA through Reference 16. TVA responded to these questions through Reference 17. Prior to the resolution of the NRC questions, and in August 1985, SQN units 1 and 2 entered extended shutdowns. Subsequently, and in preparation of the SQN Unit 2 restart, and as a contingency, TVA proceeded with modifications geared towards operating the Unit 2 safety valves with cold loop seals. This contingency was introduced due to the problems associated with the heat tracing control and safety valve leakage. Due to continuing problems and after reevaluating its options TVA decided to operate both units 1 and 2 with drained loop seals (Reference 18 and 19). This configuration is addressed in this TER.

NRC requested additional information through Reference 20 and TVA has responded through Reference 21. The TVA response is considered in this TER.

The response of the overpressure protection system to Anticipated Transient Without Scram (ATWS) and the operation of the system during feed and bleed decay heat removal are not considered in this review.

4. REVIEW AND EVALUATION

4.1 Valves Tested

The SQN, Units 1 and 2 overpressure protection systems are equipped with three (3) safety valves, two (2) PORVs and two (2) PORV block valves. The safety valves are 6-in. Crosby Model HB-BP-86, 6M6, spring loaded valves with steam internals. The design set pressure is 2485 psig and the rated steam capacity is 420,000 lbm/h. The inlet piping to the safety valves are installed with drained loop seals. The PORVs are Target Rock Model 82UU-001 globe valves, which have a nominal set pressure of 2,350 psia and a design flow capacity of 210,000 lbm/h. The PORV block valves are 3-inch Velan Model B10-3054B-13MS gate valves with Limitorque SMB-00-15 operators. There are no loop seals in the piping upstream of the PORVs.

A Crosby 6M6 safety valve identical to the model installed at the SQN Units 1 and 2 was tested in the EPRI safety valve and PORV testing program. The valve was tested in a long inlet configuration both with loop seal filled and with loop seal drained. Ring settings representative of typical PWR plant ring positions were use in eight tests. The in-plant safety valves have typical ring positions and drained loop seals. The applicable data from the above EPRI tests can be used to demonstrate the operability of the in-plant safety valves.

A PORV identical to the one used in SQN was also tested in the EPRI testing program. The in-plant PORV block valve is equipped with a Limitorque SMB-00-15 motor operator while the valve used in the EPRI tests had a Limitorque SB-00-15 operator. These two operators are essentially the same except that the SB-00-15 operator has a spring pack compensator on the stem nut which makes it more useful for high speed, high temperature service. The PORV and the PORV block valve tested are, therefore, representative of the plant valves.

Based on the above, the valves tested are considered to be representative of the in-plant valves at SQN, Units 1 and 2 and have fulfilled the part of the criteria of Items 1 and 7 identified in Section 1.2 regarding applicability of the test valves.

4.2 Test Conditions

SQN, Units 1 and 2 are 4-loop pressurized water reactors designed by the Westinghouse Electric Corporation. The valve inlet fluid 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 low temperature overpressurization events. The

expected fluid conditions for each of these events and the applicable EPRI tests are discussed in this section.

4.2.1 FSAR Steam Transients

For the SQN PWRs, the limiting events for the FSAR transients resulting in steam discharge through the safety valves only, and through both the safety valves and the PORVs are the loss of load event for maximum pressurizer pressure and locked rotor for maximum pressurization rate transient.

The safety valves are predicted to experience a peak pressurizer pressure of 2555 psia and a pressurization rate of 144 psi/s (Reference 7). The peak back pressure developed at the safety valve outlet is expected to be 438 psia. Two tests with long inlet piping conducted by EPRI on the Crosby 6M6 safety valves are applicable to SQN. These are Tests 929 and 1411. Test 929 was a steam test with a cold loop seal and Test 1411 was a steam test with a drained loop seal. Both tests were performed with typical PWR safety valve ring settings. During Test 929 the peak pressure was 2726 psia, pressurization rate 319 psi/s, and peak backpressure was 710 psia. During Test 1411 the peak pressure was 2664 psia, the pressurization rate was 300 psi/s, and the peak back pressure was 245 psia, the pressurization rate was 300 psi/s, and the peak back pressure was 245 psia. The inlet fluid condition of these tests bound the expected conditions of the FSAR steam discharge transient for the safety valve.

For FSAR transients resulting in steam discharge through both the safety valve and PORV, the maximum pressure predicted for the in-plant valve is 2532 psia and the maximum pressurization rate is 130 psi/s. The above fluid parameters represent the limiting condition for steam discharge through the PORVs for this plant. The Target Rock PORV was subjected to fifteen steam tests in the EPRI testing program. In these tests, the maximum pressure at the valve inlet was 2521 psia which is close to predicted maximum pressure of 2532 psia. The back pressure developed at the outlet of the PORVs is not an important parameter for the evaluation of this type of PORV, because the operation of the air operated PORVs is not sensitive to back pressure (Reference 6). The test inlet conditions for the PORV steam discharge are representative of the in-plant PORV steam discharge transients.

4.2.2 FSAR Liquid Transient

According to Westinghouse analyses of four loop plants, the limiting FSAR transient resulting in liquid discharge through the PORVs and safety valves is the feedline break accident (Reference 7). The PORVs and safety valves are expected to discharge steam and water in succession. However, water discharge through the safety valves and PORVs will not take place until the pressurizer is filled.

The FSAR was based on the conservative assumption that no mitigating actions were taken and indicated that the pressurizer does not go solid for almost 10 minutes. Reference 7 discussed a more recent analysis contained in Westinghouse Electric Corporation calculation WCAP 10105, Reference 22, which indicates that the saturated liquid discharge is not expected for at least 20 minutes. This 20-minute time period is sufficient for operator actions to mitigate the feedwater line break, which would arrest the pressure rise and prevent a continued surge of water in to the pressurizer even though credit for such actions was not taken in the FSAR analyses. The operator actions include isolation of the faulted steam generator so that the automatic auxiliary feedwater actuation would deliver water to the intact steam generators and isolation of safety injection to mitigate the overpressure event. These actions are described in Emergency Instructions E-0, "Reactor Trip or Safety Injection"; E-1, "Loss of Reactor or Secondary Coolant"; E-2, "Faulted Steam Generator Isolation"; and ES-0.2, "SI Termination".

Although the above actions demonstrate that in the unlikely event of a feedwater line break, operator action will prevent water discharge through the safety valves and PORVs. The following additional assurances are offered by the licensee:

1. The PORV piping and supports have been designed to withstand the loads due to discharge of a water slug followed by steam. These loads are higher than the steam only discharge loads.

2. Although the analyses used to qualify the safety valve piping on Units 1 and 2 are steam discharge cases, the piping and supports can withstand the loads associated with heated loop seal for Unit 1, and cold loop seal discharge for Unit 2. This will be discussed in more detail in the Thermal-Hydraulic Analyses, Section 4.4.

4.2.3 Extended High Pressure Injection Event

The limiting extended high pressure injection transient is a spurious actuation of the safety injection system at power (Reference 7). In this event the safety valves and PORVs are challenged by steam and liquid discharge. The maximum pressure is somewhat smaller than the pressure of the steam discharge condition. The pressurization rate is substantially lower. Therefore, the steam discharge condition is bounded by the FSAR steam discharge condition discussed previously. Subsequent water discharge will not take place until the pressurizer becomes water solid. According to reference 7 this would not occur for at least 20 minutes following the event. This would allow sufficient time for the operating personnel to take appropriate actions to prevent the water discharge since liquid discharge is not expected in this case. The fluid conditions for an extended high pressure injection event are encompassed by the conditions for other steam transients considered previously.

4.2.4 Low Temperature Overpressure Transients

Low temperature overpressure transients only challenge the PORVs. The possibility of this scenario arises from the use of the PORVs in the Cold Overpressure Mitigation System (COMS). During reactor coolant system heat up, the PORVs serve as overpressure protection at low system temperature and pressure. SQN operating procedures call for a steam bubble to be drawn at very low temperature and pressure. The Target Rock PORV was subjected to seven water tests in the EPRI testing program. Two tests cover the low temperature and low pressure conditions. For these tests the conditions at the valve inlet were 715 psia, 447°F and 690 psia, 114°F. Therefore, the inlet fluid conditions of these EPRI tests bound the low temperature overpressure transient conditions for the SQN PORVs. The flow rates arising from COMS actuation are not of sufficient quantity to result in significant loads. As a result, the loads of such a discharge are bounded by the loads caused by the high-pressure steam discharge. Also, previous analyses and qualifications discussed in the response to question 1 show that both units have considerable design margin beyond the high-pressure steam discharge only scenario.

4.2.5 PORV Block Valve Fluid Conditions

The PORV block valves are required to operate over the same range of fluid conditions as the PORVs. In the EPRI tests, the block valve was only tested at full pressure (to 2500 psia) steam conditions. The operability of the block valves under water flow conditions was not directly addressed in the EPRI tests. However, the Westinghouse gate valve closing tests (Reference 9) demonstrated that the required torque to open or close the valve depended almost entirely on the differential pressure across the valve disk and was insensitive to the momentum load. Therefore, the required force is nearly independent of the type of flow (i.e., water or steam).

Furthermore, according to the friction tests done by Westinghouse on a satellite coated specimen, the friction coefficient between satellite surfaces is approximately the same for steam and water tests. In some instances the friction force in water media is lower than in steam. The Velan block valves have satellite coated disks and seats. The force required to overcome disk friction in steam is essentially equal to the force required in water. Therefore, the steam tests are adequate to demonstrate the operability of the block valves for expected water conditions.

The test sequences and analyses described above demonstrate that the test conditions bound the conditions for the plant values. They verify that Items 2 and 4 of Section 1.2 were met, in that the conditions for the operational occurrences were determined and the highest predicted pressures were chosen for testing. The part of Item 7, which requires showing the test conditions are equivalent to conditions found in the FSAR, is also met.

4.3 Operability

4.3.1 Safety Valves

As discussed in Section 4.2, the representative EPRI tests for the steam discharge conditions for the SQN safety valves are the cold loop seal test (929) and the drained loop seal test (1411) on the Crosby 6M6 safety valve. In both tests, the test valve opened within 5% of the design set pressure and closed with less than 8.2% blowdown. The valve performance was stable during both tests. Rated flow was exceeded at 3% accumulation and the maximum valve position was 93% of rated lift during Test 929 and 92% during Test 1411. Blowdown during Test 1411 was in excess of the 5% value specified by the valve manufacturer and the ASME Code. The concern is that the increased blowdown of the safety valves might lower the reactor coolant system pressure to such an extent that adequate core cooling cannot be maintained. Westinghouse performed a generic analysis, Reference 23, on the effects of increased blowdown and concluded that no adverse effects occurred on plant safety in that the reactor core remained covered.

The pressure drop through the inlet piping was calculated for the SQN safety valve and the 6M6 test valve. The SQN valve pressure drop of 255 psi is less than the 6M6 long inlet loop seal configuration pressure drop of 263 psi. The inlet piping to the SQN, Units 1 and 2 safety valves were equipped with loop seal internals. It was later decided to drain the loop seals in order to alleviate a stress problem in the discharge piping.

A comparison has been made by the licensee between the predicted plant moments and the moments applied to the tested safety valve. The maximum predicted plant moments for any of the safety valves is a factor of at least 1.75 less than the moments applied to the tested valves. Therefore, it is concluded that the operability of the valves will not be impaired.

Based on the above arguments the operability of the SQN, Units 1 and 2 safety valves has been adequately demonstrated.

Administrative Controls for Minimizing Leakage through Safety Valves

Technical Specification (TS) surveillance requirement (SR) 4.4.6.2.1.d requires performance of a reactor coolant system water inventory balance at least once every 72 hours. Surveillance Instruction (SI) 137.2, "Reactor Coolant System Water Inventory," is the procedure used to comply with this TS SR. If the limit on the identified leakage as set forth in the plant TS is exceeded, the plant personnel are required to reduce the leakage rate to less than the limit within 4 hours or have the plant in HOT STANDBY within 6 hours and in COLD SHUTDOWN within 30 hours.

SI-112, "Testing and Setting of Setpoint of Pressurizer Safety Valves," covers testing of the pressurizer safety valves to satisfy SRs 4.4.2 (applicable for modes 4 and 5), 4.4.3.1 (applicable for modes 1, 2, 3), and 4.0.5. This instruction also fulfills the requirements for in-service performance testing of nuclear power plant pressure relief devices in accordance with American National Standards Institute/American Society of Mechanical Engineers OM-1-1981. The instruction includes leak rate testing of the valves after completion of the setpoint testing. The acceptance criteria require the leak rate to be less than or equal to 15 bubbles nitrogen per minute measured in accordance with the vendor's test procedure. The valves may be refurbished to bring the valve performance to within the required limits. The SI is required to be performed on at least one of the installed pressurizer safety valves in any 24-month period.

Temperature indicators and acoustic monitors are provided on the pressurizer safety valve discharge piping to aid the operators in determining which, if any, pressurizer safety valve is leaking. These instruments annunciate in the control room. However, no operator action is required unless there is an indication of excessive loss of reactor coolant system inventory. Instrument indications can be used to determine if detected leakage is from a pressurizer safety valve and which one. Indications can also be used to select pressurizer safety valves for planned maintenance. The instruments are verified to be operational periodically according to SIs.

Administrative Controls for Draining the Safety Valve Loop Seal

Administrative procedures require all valves in the drain lines on the inlet loops of the pressurizer safety valves to be locked open. System operating instruction 68.1 for the reactor coolant system requires double verification of the valves' position and locking before entering mode 4 operation. Locking open of the loop seal drain valves allows any condensate in the pressurizer safety valve inlet piping to drain continuously back to the pressurizer through a tap located near the bottom of the pressurizer.

4.3.2 Power Operated Relief Valve

The EPRI tests applicable to the SQN PORVs indicated that the test valve opened on demand in all twenty three (23) and closed on demand in twenty two (22) out of the twenty three tests. The valve did not close on demand for the water simulation test at the full pressure of 2505 psia and a temperature of 113°F. This is the water seal simulation test with a temperature in the accumulator of 656°F. This condition is not expected to occur in SQN.

According to the Licensee, a comparison was made between the predicted plant moments and the moments applied to the tested PORVs. The maximum predicted PORV moment is approximately a factor of 1.8 lower than the moments applied to the tested valves. Therefore, it is concluded that the operability of the valves will not be impaired.

4.3.3 PORV Block Valves

The Velan PORV block valve was subjected to 21 cycles of steam tests against full flow in the EPRI testing program. Steam pressure upstream of the block valve varied from 2440 psia to 2500 psia during the opening cycles and between 2340 and 2410 psia during the closing cycles. The stroke times of the test valve were 9.7 s to 9.9 s, which are within the required stroke time of 10.0 s. Tests for water flow for the Velan block valve were not performed in the EPRI test program. As explained in Section 4.2.5 of this report, the valve behavior under water flow condition is expected to be similar to that of the full pressure steam tests. Thus, the operability of the valves for liquid flow condition was indirectly demonstrated.

It is therefore concluded that the in-plant block valve is able to open and close successfully in a pressure range consistent with the operational requirements of the associated PORV. The tests also showed that the PORV flow was not limited by the block valve. Therefore, the operability of the SQN block valves is adequately demonstrated.

4.3.4 Electric Control Circuits

NUREG-0737, Item II.D.1, required environmental qualification (EQ) of the associated control circuitry as part of the safety and relief valve qualification task. The NRC staff has agreed, however, that meeting the licensing requirements of 10 CFR 50.49 for this circuitry is satisfactory and that specific testing per the NUREG-0737 requirement is not required. The safety valves at SQN are Crosby 6M6 pressure relief valves, which do not have any control circuitry, and therefore are not included in the 10 CFR 50.49 EQ program. The SQN PORV's are Target Rock Model 82UU-001 power operated relief valves. These valves and the Velan PORV block valves are included in the 10 CFR 50.49 program. They are qualified to perform their required active safety functions in the containment environments that result from LOCA's, main steam line breaks, feedwater line breaks, CVCS line breaks, and RHR line breaks. Documentation of this data is included in the SQN CATEGORY AND OPERATING TIMES listing, the SQN 10 CFR 50.49 list, and various SQN EQ binders. Therefore, it is concluded that the technical requirements of 10 CFR 50.49, regarding these valves have been met.

The above discussion and test results demonstrate that the SQN safety and relief valves operated satisfactorily for all expected operating and accident conditions. Therefore, the part of Item 1 of Section 1.2 of this report, which requires conducting tests to qualify the valves, Item 5, which requires qualification of the associated control circuits, and the part of Item 7, which requires that the effect of discharge piping on operability be considered, have all been met.

4.4 Thermal Hydraulic Analyses

This evaluation covers the piping upstream and downstream of the safety valves and the PORVs extending from the pressurizer nozzle to the pressurizer relief tank. The same analysis method and valve discharge conditions were used for Units 1 and 2. The calculation of the time histories of hydraulic forces due to valve discharge, are discussed below.

Pressurizer fluid conditions were selected for use in the thermal hydraulic analysis such that the calculated pipe discharge forces would bound the forces for any of the FSAR, HPI, and cold overpressurization events, including the single failure that would maximize the forces on the valve.

The thermal hydraulic analyses were performed using the RELAP5/MOD1 computer code, Reference 24. RELAP5 calculates the thermal hydraulic properties of the fluid as a function of time in each control volume and at each junction of the piping model. The RELAP5 results are then used as input to a post processor, REPIPE, reference 25, to calculate the force histories acting on the piping system. RELAP5 is widely used in the industry and was shown to be an adequate tool for predicting piping discharge loads, Reference 26. Both the RELAP5 and REPIPE computer codes used in the thermal hydraulic analysis are part of the CDC CYBERNET system. It is concluded that verification packages for both these programs are correct and complete.

TVA's computer program, FORCE, was used to modify the REPIPE output format to a format acceptable by TVA's stress analysis program. The FORCE program does not perform any analysis of the REPIPE output nor does it smooth, time average, or modify the force histories except to combine force histories at different nodes on a straight pipe segment. In addition FORCE provides plotting and printing of results. Since FORCE is a post-processing program which performs no further manipulation of the force histories, the analyst provides quality assurance of the process by spot checking the peak forces against the disk file output, the plot output, and the REPIPE output tables which were combined by the FORCE program.

The results of the review of both Safety Valve and PORV analyses, to the extent of the generation of dynamic loads, are given in the following sections. The use of these loads for the design of pipes and pipe supports is the subject of an evaluation performed by the NRC staff.

4.4.1 Safety Valves

Steam Discharge

Several different scenarios were considered during the development of the forcing functions for the PORV relief and safety valve events. Forcing functions were developed for the safety valve discharge with and without PORV actuation in an effort to determine the worst case. The thermal hydraulic

analysis of steam discharge for simultaneous pressurizer safety valve actuation without prior actuation of the PORVs (and no water-filled loop seal) is case 6 of the analysis of record, TVA calculation TI-ANL-96, revision 2. No stress analysis was performed for this case. Loads were also generated and stress analysis performed for the case where the PORVs discharge before the safety valves' lifting, case 5 of the above calculation. A comparison of the results for these two cases indicated very little difference in the magnitude of loads resulting from the discharge of the valves. These differences in turn were either in positive or negative direction, that is, higher or lower loads. No definite conclusion could be made on these trends.

Some unit specific information on the design of safety valve piping and supports, follows:

Unit 1

Following the cold loop seal modifications for the establishment of the heated loop seal, the Unit 1 pressurizer safety and relief valve piping and supports were designed for a much more severe load case of a heated water loop seal discharge in comparison to steam-only loads. According to the Licensee, the pipe support modifications required for this heated loop seal condition are installed. Coincident with the determination from the plant operating staff to operate the plant with the steam-trimmed safety valves, case 5, and the associated stress analysis were reestablished as the analyses on record.

Unit 2

Following the cold loop seal modifications for the establishment of the heated loop seal and the problems associated with heat tracking and valve leakage, the piping supports for the unit 2 pressurizer safety and relief valve piping were designed for the extreme case of a cold water loop seal discharge. The pipe support modifications for this extreme cold loop seal condition have been installed, and the piping satisfied the FSAR stress limits for the safety and relief valve discharge condition. When a decision was made to operate the plant with a drained loop and steam trimmed safety and relief valves, case 5, the associated stress analysis was re-established for the analyses on record.

In summary, the calculations of record do not include stress analysis for forcing functions for case 6. However, prior actuation of the PORVs does not significantly affect the loads resulting from the safety and relief valve discharge. Further, the piping and supports for both Units have been previously qualified for higher transient loads than are generated by the steam discharge case.

For the current trim configuration discharge analysis, the loss-of-load (maximum pressure) and the locked rotor (maximum pressurization rate) events were chosen as the limiting conditions that would generate the highest piping loads. The three safety valves were assumed to open simultaneously. This approach is reasonable because the three safety valves are identical and have

the same set pressure. Maximum forces in the common header could theoretically occur when the valve opening reaches the common header junction downstream simultaneously. This event is unlikely, however, because the valves would be required to open at times perfectly spaced to compensate for differing piping lengths leading to the common junction. Thus, the assumption of simultaneous valve opening is acceptable.

In the RELAP5 analysis of the steam discharge condition, the safety valves were assumed to activate at a set pressure of 2500 psia. The pressurization rate was assumed to be 150 psi/sec.

The above values are comparable to the predicted peak pressure of 2555 psia and pressurization rate of 144 psi/sec given in Reference 7. The valve opening time was assumed to be 10% faster than the fastest pop-opening time of 8.0 milliseconds observed in the EPRI tests. The analysis used a safety valve flow area of 0.025 ft^2 and a valve discharge coefficient of 0.84.

These modelling assumptions resulted in a steady state flow rate of approximately 500,000 lbm/hr which is 20% higher than the rated flow of 420,000 lbm/hr. Therefore, the ASME Code requirement for 90% derating of the safety valve was accounted for in the analysis. Other key input parameters and assumptions made in the thermal hydraulic analysis including RELAP5 nodalization scheme and time steps as well as REPIPE nodalization scheme were reviewed and found acceptable.

Steam to Liquid Transition: FW Line Break

As discussed in Section 4.2.2, in the unlikely event of a FW line break, operation action will prevent water discharge through the safety valves. Therefore, the piping was not analyzed for the steam to liquid transition. However, the licensee has offered an additional assurance by designing the piping for the much higher loads resulting from a heated loop seal (Unit 1) and a cold loop seal (Unit 2).

4.4.2 PORVs

Steam Discharge

The pressurizer PORVs and block valves are configured on a horizontal run of piping elevated above the pressurizer. As such, there is no location amenable for the collection water to form or sustain a loop seal. As such, blowdowns through the PORVs are limited to steam blowdowns, except in the unlikely event of a transition to liquid relief. The fluid transient forces resulting from simultaneous actuation of the PORVs are provided in case 8 of TVA Calculation TI-ANL-96, Revision 2. Case 8 is similar to, but bounded by, case 5, which assumed a water slug upstream of the PORVs. The resulting loads from case 5 were considered in the piping analysis. In the RELAP5 analysis the PORVs were assumed to activate at a set pressure of 2265 psia and temperature of 650°F. The PORV flow area was determined based upon the requirement of steady state

flow rate of 210,000 lbm/hr. The opening time was taken to be 60 milliseconds which is a realistic estimate concurred by the valve manufacturer. For the case of record, case 5 of TI-ANL-96, Revision 2, a loopseal was assumed upstream of the PORV. This resulted in substantially higher loads than the realistic steam only discharge care. These higher (conservative) loads were used in the design of the PORV piping and supports.

Cold Water Discharge - Low Temperature Overpressure Transient

As discussed in Section 4.2.4, the possibility of this scenario arises from the use of the PORVs in the COMS.

The flow rates arising from COMS actuation are not of sufficient quantity to result in significant fluids transients loads. As a result, the loads because of such a discharge are bounded by the loads caused by the high-pressure steam discharge.

Moreover, as noted above, the piping and supports have been designed for loop seal discharge followed by high-pressure steam discharge and as such are conservative.

5. EVALUATION SUMMARY

TVA participated in the development and execution of an acceptable relief and safety valve test program designed to qualify the operability of prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The generic test results and piping analyses showed that the valves tested functioned correctly and safely for all relevant events specified in the test program. Analysis and review of the test results and the Licensee's justifications indicated direct applicability of the prototypical valve and valve performances to the in-plant valves and systems intended to be covered by the generic test program. The tests were successfully completed under operating conditions, which by analysis bounded the most probable maximum forces expected from anticipated design basis events.

This report provides the results of the technical review of these programs to the extent of the generation of the dynamic loads. The use of these loads for the design of pipes and pipe supports is the subject of another technical evaluation performed by the NRC staff. This review finds the responses of the licensee, covering the above defined scope of work, to be technically acceptable.

6. REFERENCES

1. TMI Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, July 1979.

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

3. D.P. Hoffman, Consumers Power Company, letter to H. Denton, NRC, "Transmittal of PWR Safety and Relief Valve Test Program Reports," September 30, 1982.

4. EPRI Plan for Performance Testing of PWR Safety and Relief Valves, July 1980.

5. EPRI PWR Safety and Relief Valve Test Program Valve Selection/Justification Report, EPRI NP-2292, January 1983.

6. EPRI PWR Safety and Relief Valve Test Program Test Condition Justification Report, EPRI NP-2460, January 1983.

7. Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Westinghouse-Designed Plants, EPRI NP-2296, January 1983.

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

9. R.C. Youngdahl, Consumers Power Company, letter to H. Denton, NRC, "Submittal of PWR Valve Data Package," June 1, 1982.

10. EPRI PWR Safety and Relief Valve Test Program Guide for Application of Valve Test Program Results to Plant-Specific Evaluations, Revision 2, Interim Report, July 1982.

11. L.M. Mills, TVA, letter to E. Adensam, NRC dated April 1, 1982. Response to D.G. Eisenhut's letter of September 29, 1981, requesting submittal of documentation on the relief and safety valves by April 1, 1982.

12. L.M. Mills, TVA, letter to E. Adensam, NRC dated June 30, 1982. Supplementary response to D.G. Eisenhut's letter of September 29, 1981 requesting submittal of documentation on the relief and safety valves by April 1, 1982.

13. L.M. Mills, TVA, letter to E. Adensam, NRC dated July 9, 1982. Response to D. G. Eisenhut's letter of May 5, 1982.

14. L.M. Mills, TVA, letter to E. Adensam, NRC dated January 7, 1983. Submittal of supplemental information to the first two letters.

15. L.M. Mills, TVA letter to E. Adensam, NRC dated May 30, 1984. Revision of the above supplemental information.

16. E. Adensam, NRC, letter to H.G. Parris dated March 25, 1985. Request for additional information.

17. J.A. Domer, TVA letter to E. Adensam, NRC dated July 10, 1985. Response to the above letter.

18. R. Gridley, TVA letter to NRC dated June 24, 1988, "Sequoyah Nuclear Plant (SQN) - Performance Testing of Reactor Relief and Safety Valves." Addressed only Unit 2.

19. R. Gridley, TVA letter to NRC dated August 1, 1988. Update to the above letter for Unit 1.

20. S. Black, NRC, letter to S.A. White dated August 18, 1988, "Request for Additional Information for TMI Action Item, II.D.1, Performance Testing of Reactor Relief and Safety Valves for Sequoyah, Units 1 and 2 (TAC 44621, 51421)."

21. R. Gridley, TVA, letter to NRC dated October 27, 1988. Response to the above request.

22. Westinghouse Electric Company, Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program, WCAP-10105, June 1982.

23. Safety Valve Contingency Analysis in Support of the EPRI Safety/Relief Valve Testing Program - Volume 3: Westinghouse Systems, EPRI NP-2047-LD, October 1981.

24. RELAP5/MOD1 Code Manual, Volume 1: System Models and Numerical Methods, Volume 2: Users Guide and Input Requirements. EG&G Idaho, Cycle 14, NUREG/CR-1826, EGG-2070 Draft, Revision September 1981.

25. REPIPE, Application Reference Manual, Cybernet Services CDC, July 1980.

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