Mailing Address
Alabama Power Company
600 North 18th Street
Post Office Box 2641
Birmingham, Alabama 35291
Telephone 205 783-6090

R. P. McDonald
Senior Vice President
Flintridge Building

Alabama Power
the southern electric system

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Docket Nos. 50-348
50-364

Director, Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. S. A. Varga

Joseph M. Farley Nuclear Plant - Units 1 and 2
NUREG-0737, Item II.D.1

Gentlemen:

By letter dated July 2, 1984, the NRC requested that Alabama Power Company provide additional information regarding performance testing of relief and safety valves (NUREG-0737, Item II.D.1). Attached is Alabama Power Company's response to this request.

If you have any questions, please advise.

Yours truly,

R. P. McDonald

RPM/JAR:grs-D14
Attachment
cc: Mr. L. B. Long
Mr. J. P. O'Reilly
Mr. E. A. Reeves
Mr. W. H. Bradford

A046
1/1

Attachment

Additional Information Regarding NUREG-0737, Item II.D.1

NRC Question 1

The submittal identifies the feedwater line break (FWLB) accident as one which causes liquid water flow through the safety valves. The EPRI tests under similar conditions were performed for only a few seconds. If the plant FWLB accident causes water flow through the valves for a time period greater than that tested provide information that demonstrates that the plant safety valves can perform their pressure relief function and the plant can be safely shutdown.

APCo Response 1

The Farley Nuclear Plant (FNP) pressurizer safety valves were not originally designed for the passage of water. These valves are intended only for the passage of steam; however, their ability to pass water for short periods of time was demonstrated during the EPRI tests. In addition, the FNP FSAR Chapter 15 analyses for FWLB are overly conservative and show that adequate time is available for an operator to take corrective action to prevent water passage through the safety valves and safely shut down the plant.

NRC Question 2

Results from the EPRI tests on the Crosby 3K6 and 6M6 safety valves with loop seal internals and utilizing upstream piping loop seals indicate that blowdowns may exceed the design blowdown of 5 percent, depending on the safety valve ring settings used (see related question 4). The consequences of potentially higher blowdowns are not addressed in the Farley 1 and 2 submittal. Blowdowns in excess of the design blowdown of 5 percent could cause the pressure to be sufficiently decreased such that adequate cooling might not be achieved for decay heat removal. Discuss the consequences of higher blowdowns if blowdowns in excess of 5 percent are expected.

APCo Response 2

The FNP safety valve production testing by the valve manufacturer, Crosby, established ring settings for the safety valves to provide 5% blowdown. These ring settings are utilized at FNP. Valve performance at these ring settings was demonstrated by Crosby to be stable. Blowdowns in excess of 5 percent, although not expected, have been analyzed in conjunction with the Westinghouse Owners Group (WOG) program on safety valves. The analyses utilized valve blowdowns in excess of 10 percent. The results from these analyses showed no adverse effects on plant safety. The peak pressurizer water level calculated remained below the pressurizer inlet piping to the safety and relief valves. Discussion of the WOG program is contained in the Westinghouse report WCAP-10105, "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety and Relief Valve Test Program," dated June 1982. This report was transmitted to the NRC by WOG letter dated August 17, 1982.

NRC Question 3

The Farley Plant Crosby 6M16 safety valves was not tested by EPRI. Results from EPRI tests on the Crosby 3K6 and 6M6 safety valves were used to evaluate performance of the Crosby 6M16 valve of Farley Units 1 and 2. The EPRI test results indicate that the 6M6 valve achieved rated flow for steam flow. Although the submittal states that the 3K6 valve also achieved rated flow, the EPRI test results show that this valve had not receiv d rated flow at 3 percent accumulation for the loop seal tests at certain ring settings. Provide a further evaluation as to whether the test results sufficiently show that the 6M16 valve will pass rated flow. A further consideration is that the safety valve flow rate depends on the specific ring settings used. Demonstrate that the safety valves will pass their rated flow with the plant ring settings.

APCo Response 3

According to the EPRI Valve Selection/Justification Report NP-2292-LD, the 6M6 test valve was selected by Crosby to be representative of the M1 orifice safety valve design (6M16 valve) installed at Farley. As noted in Table 4.4 of EPRI Report NP-2770-LD, Volume-6, the Crosby 6M6 test valve achieved rated flow for each of the tests reported at 3 percent accumulation regardless of the ring setting used in the test. A review of EPRI Tables 4-3 and 4-4 in Volume 5 of EPRI report NP-2770 LD reveals that for steam tests of the 3K6 valve where blowdown was measured to be less than 10 percent, flow rates of 119-122 percent of rated flow at 3 percent accumulation were reported. The EPRI tables indicate that lower than rated flows occurred at blowdowns greater than 15 percent. As stated in the response to NRC question 2, the Crosby production testings have determined that the FNP ring settings will provide blowdowns of 5 percent for the safety valves. This is within the range of both the 3K6 and 6M6 tests where rated flow was achieved; therefore, rated flow can be expected for the FNP 6M16 valves. EPRI Reports NP-2292-LD and NP-2770-LD have been transmitted to the NRC.

NRC Question 4

The submittal indicates that the EPRI tests on the Crosby safety valves were conducted at varying ring settings. The submittal does not clarify, though, whether any of these ring settings correspond with those used at the Farley plant. If the plant current ring settings were not used in the EPRI test, the test results may not be directly applicable to the Farley plant valves. The submittal did not state if either the valve manufacturer (Crosby) or the NSSS supplier (Westinghouse) were reviewing the Farley plant ring settings. Identify the ring settings to be used in the Farley plant safety valves. Identify the equivalent ring settings for the Crosby 3K6 and 6M6 test valves. If the plant-specific ring settings were not tested in the EPRI

NRC Question 4 (Continued)

program, explain how the expected values for blowdown and resulting back pressure, corresponding to the plant-specific ring settings, were extrapolated or calculated from the EPRI test data. Identify these values for backpressure and blowdown so determined. Evaluate and discuss the effects of the expected backpressures and blowdowns on valve behavior.

APCo Response 4

The FNP 6M16 Crosby safety valve ring settings are as follows:

Valve Identification No.	Ring Settings	
	Nozzle Ring	Guide Ring
N56963-01-000-1	- 18	- 275
N56963-01-000-2	- 18	- 285
N56963-01-000-3	- 18	- 275
N56963-01-000-4	- 18	- 220
N56963-01-000-5	- 18	- 250
N56963-01-CJ0-6	- 18	- 270
N56963-01-010 (spare)	- 18	- 230

It should be noted that the ring settings indicated above were measured by Crosby (Response to Question 2) from the highest "locked" position as noted in Crosby procedures and definition of key terms and parameters in the EPRI reports but not as measured from the "level position" as those tested ring settings reported by EPRI.

Ring settings for the FNP 6M16 safety valves were developed by Crosby during production testing of the valves; however, based on the above reporting difference, equivalent ring settings for the Crosby 3K6 and 6M6 valves can not be provided. Blowdowns measured in the Crosby production testing for each valve were equal to or less than 5 percent. The dynamic backpressures measured during the EPRI tests were approximately 9 - 29 percent of the setpoint; however, Crosby valves, including the 6M16 and the 6M6 and 3K6, employ a balanced bellows design that is relatively insensitive to backpressure effects as noted by EPRI on page S-5 of EPRI Report NP-2770-LD, Volume 5. The blowdown and backpressure effects, therefore, will not impact valve functionability.

NRC Question 5

Bending moments are induced on the safety valves and PORV's during the time they are required to operate because of discharge loads and thermal expansion of the pressurizer tank and inlet piping. Make a comparison of the predicted plant-specific valve bending moments to the tested valve bending moments to demonstrate that the operability of the valve will not be impaired.

APCo Response 5

The bending moments induced on the safety valves tested by EPRI exceeded the bending moments predicted for the FNP safety valves. The maximum moment tested for the 6M6 valve was during test 908 and was 298.75 in-Kips . The largest moment predicted at the safety valve outlet for FNP is 146.95 in-Kips, thus demonstrating functionability for the FNP safety valves.

A bending moment of 43.0 in-Kips was induced in the inlet to the Copes-Vulcan PORV test valve per test EPRI 64-CV-174-2S. EPRI states this test bending moment for the Copes-Vulcan 17-4/PH plug and cage test valve can also be used for the Copes-Vulcan 316 stellite plug and 17-4/PH cage test valve. This is the test valve which is most similar to the FNP valve. The bending moments predicted for the FNP PORVs are within the test value when deadweight, thermal and relief valve thrust is considered. The predicted moments are greater, however, when safety valve thrusts loads are considered. The predicted value is approximately 56.62 in-Kips in this case. This is not to say that the valve would not function at this load but that the predicted bending moment is in excess of that tested. A review of the EPRI data plots reveals, however, that the bending moment measured at the relief valve outlet during test 64-CV-174-2S was approximately 58.0 in-Kips. This was in excess of the FNP predicted value of 56.62 in-Kips, thus demonstrating functionability for the FNP PORV.

NRC Question 6

During an EPRI loop seal steam-to-water transition test on the Crosby 3K6 valve, the valve fluttered and chattered when the transition to water occurred. The test was terminated after the valve was manually opened to stop chattering. The Crosby 6M6 valve exhibited similar behavior on two loop seal steam tests and one subcooled water test. Again, these tests were terminated after the valve was manually opened to stop chatter. Justify that the valve behavior exhibited in these tests is not indicative of the performance expected for the Farley Unit 1 and 2 valves.

APCo Response 6

Testing conducted by EPRI was designed to envelop numerous plant-specific conditions of piping arrangement, inlet fluid conditions and valve types. Because worse case conditions were tested, adjustments had to be made to the test valves to obtain stable performance during the tests. This sometimes resulted in ring settings outside the normal "as shipped" range and blowdowns in excess of 5 percent. As stated in the response to question 4, ring settings for the FNP valves were established by Crosby during production tests and were reported to result in blowdowns of 5 percent. The test results, obtained with ring settings different from those established by Crosby, may not be directly applicable to the FNP valves and consideration should be given to the effects of these adjustments.

The 3K6 steam-to-water transition test was conducted using ring settings that resulted in blowdowns in excess of 15 percent and is therefore not considered representative of the ring settings specified for the Farley valves which result in 5 percent blowdown and stable valve performance.

The two 6M6 loop seal steam tests 920 and 1419 which ended in valve chatter were repeats of two other successful steam tests 917 and 1415. Close examination of these tests reveals that for each test the safety valve opened on demand, relieved the test system pressure transient for approximately 20 seconds, reduced the initial pressure by 4-5 percent as required and then closed. Upon closing the valve reopened only on tests 920 and 1419 and chattered. Valve inlet pressure plots show that pressure oscillations at the valve inlet which vary from the magnitude of the set pressure to the blowdown pressure, drove the valve to chatter. These pressure oscillations are acoustic in nature and are due to the long loop seal piping length tested by EPRI. As the FNP loop seals are shorter than those tested by EPRI and the acoustic pressure drop is less than that for the test configuration, more stable results can be expected at FNP.

NRC Question 7

The adequacy of the thermal hydraulic analysis could not be verified since sufficient detail is not presented in the submittal. Provide a thermal hydraulic report which discusses the detailed descriptions of the methods and computer programs used to perform this analysis. Identify these programs and how these programs were verified. The thermal hydraulic report should include a description of the methods used to generate fluid pressures and moments over time and the methods used to calculate the resulting fluid forces on the

NRC Question 7 (continued)

system. Also identify important parameters used in the thermal hydraulic analysis and discuss rationale for their selection. These include time-step, valve flow area, peak pressure and pressurization rate, choked flow junctions, node spacing, valve opening time, and fluid conditions upstream and downstream of the safety valve at the time the valve pops open.

APCo Response 7

The thermal hydraulic report requested was provided in Alabama Power Company's November 4, 1982 letter to the NRC. In this report, the adequacy of the thermal-hydraulic analyses can be verified by the comparison of the generic Westinghouse analytical results to the EPRI test results for thermal hydraulic loadings in safety valve discharge piping for EPRI tests 908 and 917. In the Westinghouse analytical model, node spacing and time-step size were selected on the basis of stable solutions of the characteristic equations and matching of test data. The safety valve full open flow area of 0.022 ft^2 was used in the Westinghouse model. This area is slightly smaller than the Crosby M-orifice area of 0.025 ft^2 for the tested valve, but resulted in a good analytical match of the tested fully open valve flow rate. Appropriate water temperatures were used. All pertinent data, including friction factors, loss factors and flow areas were based upon representative calculations and the system layout. Modeling of the water was conducted with the water seal upstream of the valves prior to transient initiation. At time equal to zero, the transient was initiated and the slug position was analytically calculated during and subsequent to valve opening.

The FNP-specific thermal-hydraulic analysis was conducted based upon the same approach as used to develop the generic Westinghouse analysis. Node spacing and time-step size utilized were consistent with values utilized in the above mentioned comparison. Selected valve flow areas were based upon actual valve data with appropriate margins applied to account for flow rate uncertainties. Analyses performed assumed a 100 percent linear safety valve opening time (0.040 seconds) with the pressurizer conditions held at initial values. The safety valves were assumed to open at a pressure of 2575 psia (valve set pressure plus accumulation) with the pressurizer pressure held constant at this same value for the entire transient. All pertinent data, including friction factors, loss factors and flow areas were based upon representative calculations and the system layout. Modeling of the water slug from a temperature profile, initial location and movement post-transient initiation point of view was consistent with the comparison study. Choked flow was checked internally and automatically every time-step to ensure the proper formulation was applied at every flow path.

APCo Response 7 (continued)

As discussed in the November 4, 1982 thermal-hydraulic report, the computer code ITCHVALVE was utilized to perform the transient hydraulic analysis for the system. This program utilizes the Method of Characteristics approach to generate fluid parameters as a function of time. A discussion of this Method of Characteristics solution technique is presented in the following articles:

1. A. C. Spencer and S. Nakamura, "Implicit Characteristic Method for One-Dimenstional Fluid Flow", ANS Transaction, Volume 17, P.247, November 1973.

2. S. Nakamura, M. A. Berger and A. C. Spencer, "Implicit Characteristic Method for One Dimensional Fluid Flow", Proceedings of the Conference on Computational Methods in Nuclear Engineering, Conference 75040, National Technical Information Service, Springfield, VA, 1975.

A. C. Spencer is a full-time Westinghouse employee who was and is directly involved in the development of the ITCHVALVE Computer Program.

Once the time-history fluid properties were available, the properties were utilized in determining the forcing functions. Unbalanced forces were calculated for each straight segment of pipe from the pressurizer to the relief tank. A discussion of the methodology for generating the thermal hydraulic forcing functions and a comparison of analytically determined hydraulic force results to test data is presented in the following article:

L. C. Smith and K. S. Howe, "Comparison of EPRI Safety Valve Test Data with Analytically Determined Hydraulic Results", The International Conference on Structural Mechanics in Reactor Technology, Chicago, Illinois, August 22-28, 1983, Volume F, 2/6, pp. 89-96.

L. C. Smith and K. S. Howe are full-time Westinghouse employees who were and are involved in pressurizer safety and relief valve and thermal hydraulic issues. Because of the proprietary nature of the ITCHVALVE computer program, a descriptive report has not been supplied; however, the above three referenced articles and the attached report discuss the verification of this computer program.

NRC Question 8

The submittal does not describe the method used of treating valve resistance in the analysis and does not report flow rates corresponding to the resistance used. The ASME Code requires derating of the safety valves to 90 percent of expected flow capacity to obtain the ASME rated flow capacity. The EPRI safety valve data indicated that steam flow rates in excess of rated flows

NRC Question 8 (continued)

are attainable. Therefore, the piping analysis should be based on a flow rating equal to 111 percent of the safety valve's rated flow unless another flow rate can be justified. Provide further explanation on how derating of the safety valves was handled and the methods used to establish flow rates for the safety valves and PORVs in the thermal hydraulic analysis.

APCo Response 8

Safety and relief valves are modeled as two-way junctions. The pressure drop across the valve, provided the system is sub-cooled, is given by:

$$\text{delta } P = pC_d \, v^2$$

Where delta P = pressure drop
p = fluid density
C_d = discharge coefficient = f(Cv)
v = velocity through the valve

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

The expected steam flow rate through the FNP Copes-Vulcan PORVs is 210,000 lb/hr at approximately 2350 psia. Values of 236,000 and 232,000 lb/hr, respectively at 2140 and 2120 psia were observed in the EPRI/Marshall Tests (EPRI report NP-2144-LD, EPRI-Marshall Power - Operated Relief Valve Interim Test Data Report"), February 1982. This report has been transmitted to the NRC. To account for all uncertainties and tolerances in the valve flow rate, the valve flow area was adjusted accordingly. The minimum analytically calculated steam flow through each of the two PORVs is greater than 291,000 lb/hr (full open valves). This is a flow of 139 percent of rated which is greater than the derating value of 111 percent (90 percent derated value). The analysis assumed a 100 percent linear PORV valve opening in 1.00 second. Full open times, based upon tests, averaged 1.48 seconds with a minimum value of 1.40 seconds for opening on steam.

The nominal steam flow rating for the Crosby Safety Valves is 345,000 lb/hr. As with the PORVs, to ensure that adequate margin existed in the valve flow rate to account for all uncertainties and tolerances, the analytically calculated steam flow was checked prior to finalizing this phase of the overall effort. The flow used in the analysis (478,600 lb/hr) was 139 percent of rated which is greater than the derating value of 111 percent (90 percent derated value). The safety valves were presumed to open fully in 0.040 seconds. This is based upon an effective linear opening time.

NRC Question 9

The adequacy of the structural analysis could not be verified since sufficient detail is not presented in the submittal. The submittal does state that the dynamic solution was obtained using a modified predictor-correction integration technique and normal mode theory. Provide a structural analysis report describing in greater detail this solution technique and the computer program used to perform the analysis. Identify the program(s) and how the program(s) was verified. Identify important parameters used in the structural analysis and the rationale for their selection. These include lumped mass spacing, solution time step, damping, and cutoff frequencies (if applica:'e). Also describe the methods used to model the connections to the pressurizer and relief tanks, and the safety valve bonnet assemblies and relief valve actuators.

APCo Response 9

As noted in Alabama Power Company's November 4, 1982 submittal, the major structural analyses programs utilized in the static and dynamic analyses were described in WCAP-8252 which was reviewed and approved by the NRC by letter dated April 7, 1981. A discussion of the methodology utilized in performing a safety valve discharge structural analysis and a comparison of analytical results to structural test results is presented in the following article:

L. C. Smith and T. M. Adams, "Comparison of Analytically Determined Structural Solutions with EPRI Safety Valve Test Results", 4th National Congress on Pressure Vessel and Piping Technology, Portland, Oregon, June 19 - 24, 1983 PVP-Volume 74, pp. 193 - 199.

Following is a discussion of key parameters used in the structural analyses of the thermal hydraulic events performed for FNP:

1. Damping: A conservative system damping of 2 percent was utilized. This is much lower than the actual expected value and is below the 10 percent damping used in the structual comparison to EPRI Tests 908 and 917.

2. Lumping: Lumped mass spacing was determined to ensure that all appropriate mode shapes were accurately represented.

3. Supports: The structural supports were modeled in sufficient detail to analytically represent the system. The shock suppressors and struts were modeled by inputing a stiffness in series with the piping. A linear overall system analyses was conducted.

APCo Response 9 (continued)

4. Time-Step: The integration time step is internally determined within the structural program and is based upon convergence criteria that results in stable solutions. The largest time-step ever used could be 0.0001 seconds. The time step is automatically adjusted so that the relative error of each modal coefficient is at least less than 10^{-2}.

5. Cutoff Frequency: The cutoff frequency utilized in both the relief valve and safety valve discharge cases was approximately 500 Hz.

A three mass pressurizer model was included in the overall system model. The pressurizer nozzles and pipe connections were represented with appropriate pipe properties. Intensification at the nozzle to pipe welds was included. The downstream piping terminated at the relief tank inlet flange where the model was anchored.

The safety valve bonnet assemblies and the relief valve actuators were modeled as extended masses, displaced from the pipe centerline. The valves' weight and center of gravity were selected from the valve drawings. The stem properties (diameter and thickness) were selected to represent the valve frequency.

Because of the proprietary nature of the computer program, input and output data has not been supplied; however, the above article and the November 4, 1982 submittal discuss the verification of the computer program.

NRC Question 10

The submittal does not describe the methods used to apply the fluid forces to structural model. Since the forces acting on a typical pipe segment are composed of a net, or "wave", force and opposing "blowdown" forces, describe the methods used in applying both types of forces to the model.

APCo Response 10

Program FORFUN was utilized to calculate the unbalanced wave forces for each segment of piping. The time history hydraulic forces determined by FORFUN were then applied to the appropriate piping system lump mass points. The axial extension from the balancing forces (opposing "blowdown" forces) on each end of the structural segment was considered in the FORFUN evaluation; however, this effect was determined to be negligible relative to the net unbalanced forces. Referring to structural analyses comparisons to test results for EPRI Tests 908 and 917, maximum support and pipe loads compared well with test results. Good comparisons of the maximum displacement values downstream of the safety valve were also seen. See answers to NRC Questions 7 and 9 for additional discussions of structural analysis comparisons.

NRC Question 11

According to results of EPRI tests, high frequency pressure oscillations of 170 - 260 Hz typically occur in the piping upstream of the safety valve while loop seal water passes through the valve. The submittal refers to an evaluation of this phenomenon that is documented in the Westinghouse report WCAP 10105 and states that the acoustic pressures occurring prior to and during the safety valve discharge are below the maximum permissible pressure. The study discussed in the Westinghouse report determined the maximum permissible pressure for the inlet piping and established the maximum allowable bending moments for Level C Service Condition in the inlet piping based on the maximum transient pressure measured or calculated. While the internal pressures are lower than the maximum permissible pressure, the pressure oscillations could potentially excite high frequency vibration modes in the piping, creating bending moments in the inlet-piping that should be combined with moments from other appropriate mechanical loads. Provide one of the following: (1) a comparison of the allowable bending moments established in WCAP 10105 for Level C Service Conditions with the bending moments induced in the plant piping by dynamic motion and other mechanical loads or (2) justification for other alternate allowable bending moments with a similar comparison with moments induced in the plant piping.

APCo Response 11

The FNP piping system response (including the safety valve loop seal region) is due to frequencies less than 100 Hz. The frequency of the forces and moments in the 170-260 Hz range potentially induced by acoustic pressure oscillations is significantly greater than this frequency. The upper limit of significant frequency content for similar piping systems is much less than this 170-260 Hz range. Industry data indicates that only frequencies of 100 Hz or less are meaningful. The EPRI test data confirms this. Consequently no significant bending moment during the pressure oscillation phase of the transient will occur.

In the November 4, 1982 Alabama Power Company submittal, pressure stresses based upon a design pressure of 2485 psig were included with the bending moments resulting from the safety valve discharge piping loads. Because of the time phasing of the pressure oscillation (during water slug discharge through the safety valve) and the discharge piping loads (subsequent to water slug discharge through the valve), this pressure term and moment term were not added as they do not occur coincidentally. A comparison of the intensified bending moments from the stress evaluation and the allowable moment presented in WCAP-10105 shows that all values are below the allowable values. Specifically, the maximum allowable moment from Table 4-7 of WCAP-10105 for 6 inch schedule 160 piping for an internal pressure of 5000 psi is 516 in-kips. The FNP bending moments for water slug discharge for the components listed in Table 6-11 of the submittal at nodes 3020, 3030, 4110 and 4080 respectively are 333.1, 303.1, 239.1 and 89.9 in-kips.

NRC Question 12

The submittal indicates that the thermal hydraulic loads were recalculated subsequent to the EPRI tests to reflect results of the tests. A letter from F. L. Clayton, Jr. to S. A. Varga, dated November 4, 1982, states that the new loads caused overstresses in the discharge piping to the safety valves. The letter contends that the pressurizer, pressurizer nozzles, valve inlet piping, and operability of the safety valves would not be affected by a rupture in the discharge piping. The submittal does not, however, present any specific results of the analysis to support this contention. The November 4th letter mentions an attached report that evidently contains stress results but this report was not actually included in the submittal. So that the effects of the overstresses identified in the analysis can be better evaluated, provide the mentioned report and other reports that contain specific results of the stress analysis and that evaluate the consequences of overstresses for the specific piping location that are overstressed. Specifically, provide a report that demonstrates that the overstress in the discharge piping will not impair the ability of the safety valves to operate and will not deform the piping in a manner that will restrict flow.

APCo Response 12

Contrary to the statement above, the referenced report was provided to the NRC with the November 4, 1982 submittal. Conversations with the Farley Nuclear Plant NRC Project Manager have confirmed that this report was received by the NRC.

NRC Question 13

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

APCo Response 13

The electrical components associated with the operation of the PORV, which are exposed to harsh environments as a result of design basis accidents, were reviewed and considered qualified as stated in Alabama Power Company's two letters to the NRC dated June 23, 1982. These components were listed in Section 3 of the June 23, 1982 submittals; however, the cables on the outside of the containment located in the electrical penetration room were not listed.

APCo Response 13 (continued)

During the Regulatory Guide 1.97 compliance review, the electrical penetration rooms, which are exposed to the post-accident radiation, were considered to be harsh areas. In consideration of this, two additional cables were added to the list of FNP equipment requiring environmental qualification. These cables are considered qualified for FNP.