PRESSURIZER SAFETY LINE
PIPING AND SUPPORT EVALUATION
UNDER SAFETY VALVE DISCHARGE LOADING

SOUTHERN NUCLEAR OPERATING COMPANY

J. M. FARLEY UNIT 1 AND UNIT 2

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1.0 INTRODUCTION

1.1 General

The Pressurizer Safety and Relief Valve (PSARV) discharge piping system for pressurized water reactors, located on the top of the pressurizer, provides overpressure protection for the reactor coolant system. A water seal is maintained upstream of each pressurizer safety and relief valve to prevent a steam interface at the valve seat. This water seal minimize the possibility of valve leakage. It is also recognized that, with this system configuration, significant thermal hydraulic loads are generated when the valves are actuated.

Under NUREG 0737[1], Section II.D.1, "Performance Testing of BWR and PWR Relief and Safety Valves," all operating plant licensees and applicants are required to conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents. In addition to the qualification of valves, the functionability and structural integrity of the as-built discharge piping and supports must also be demonstrated on a plant specific basis.

In response to these requirements, a program for the performance testing of PWR safety and relief valves was formulated by EPRI[2]. The primary objective of the Test Program was to provide full scale test data confirming that functionability of the reactor coolant system power operated relief valves and safety valves are capable of performing their design function for expected operating and accident conditions. The second objective of the program was to obtain sufficient piping thermal hydraulic load data to validate models utilized for plant unique analysis of PSARV discharge piping systems. Based on the results of the aforementioned EPRI Safety and Relief Valve Test Program, additional thermal hydraulic analyses were required to adequately define the loads on the piping system due to valve actuation.

1.2 Farley Plant Units 1 and 2

In response to NRC letter of September 29, 1981 and to the requirements of NUREG-0737, Item II.D.1, Alabama Power Company submitted letter responses[3],[4] to NRC on April 1, 1982 and July 1, 1982. In both letters, Alabama Power Company addressed the capability of relief and safety valve and piping issues by committing to complete further analysis of the downstream loads due to valve actuation based on the results of EPRI test program. Subsequent to those submittals, on November 4, 1982, Alabama Power Company informed the NRC that the analysis had been completed[5]. A Westinghouse report was attached to that November 4th letter. It was indicated that the evaluations performed by Westinghouse were based on cold seal discharge which is the design basis for Farley Nuclear Plant. It was concluded that no overstress occurred in piping subsequent to actuation of the power operated relief valves. However, a potential overstressed region in the piping downstream of the safety valves was identified subsequent to safety valve discharge. These results are based on the conservative postulation that all three safety valves simultaneously discharge cold loop seals. In a December 16, 1986 safety evaluation report (SER),[6] the NRC expressed concern regarding the potential impact of the operability of the safety valves due to the over-stress condition in the pipe. The SER postulated that the over-stressed pipe may deform rather than rupture, thus affecting the safety valves overpressure protection capacity.

In response to the SER, Alabama Power Company made a commitment to NRC in a letter dated February 5, 1987 to provide a schedule for resolution of NRC concerns[7]. In a letter, dated September 16, 1988, the NRC was informed of Alabama Power Company's goal of raising the temperature of the water in the loop seal piping[8]. To achieve this goal, modifications to piping insulation were necessary to ensure sufficient heat is conducted to the loop seal water. Upon the completion of these modifications, at-power temperature measurements of the loop seal piping would be made. These modifications, in conjunction with the inspection, test and maintenance procedures were considered to be the resolution of all remaining issues associated with NUREG-0737, Item II.D.1.

In 1991, the above stated temperature measurements were obtained for both units[9]. The modified insulation did serve the intended purpose to raise the temperature of the water in the loop seal piping, and therefore, reduce the severity of the hydraulic shock loads from water slug discharge on the piping and supports. This report is being prepared to discuss the analysis and results from these new loads in the piping and support system.

2.0 ANALYTICAL MODELING AND APPROACHES

2.1 Thermal Hydraulic Modeling

The safety valve discharge loads were calculated for the fluid transient condition that will produce the most severe loading on the piping system. This occurs during a high pressure transient where steam from the pressurizer forces the water in the water seal through the safety valve down the piping system to the relief tank. Forcing functions are normally generated for hot or cold loop seals depending on the temperature in the loop seal. The hot and cold loop seal conditions for Farley plants are consistent with the hot and cold loop seal conditions defined in 1982 EPRI tests. The general arrangement of a safety valve loop seal is shown in Figure 1. Thermal hydraulic analysis for the Farley pressurizer safety valve (PSARV) system were originally analyzed in 1982 for both the hot and cold loop seal conditions. The hydraulic forces generated when the safety valves open are much higher for the cold loop seal condition compared to those forces from the hot loop seal condition. To reduce the loads from cold loop seal condition, modification to piping insulation was necessary to ensure sufficient heat was conducted to the loop seal water. However, due to field installation constraints, the loop seal piping temperatures were not as high as expected. The measured temperature profiles at the three loop seals for the Units 1 and 2 PSARV systems fall between the bounds of hot and cold.

The actual measured loop seal temperatures are tabulated in Table 1. The node notations are based on the Figure 1 convention. It can be seen that the temperatures are higher upstream of the loop seal which is closer to the pressurizer and the temperatures are colder near the inlet of the safety valves.

Thermal Hydraulic Analysis Method for Farley Pressurizer Safety System

Based on the WCAP-10105[10] report "Review of pressurizer safety valve performance as observed in the EPRI Safety and Relief Valve Test Program," (June 1982), the valve opening characteristics are not linear. The valve stem actually lifts partially to let the water seal pass through the valve until the steam behind the water slug reaches the valve stem. Then the valve stem will lift up fully in about .04 second. These valve opening characteristics are consistent with Figure 4-12 of the WCAP-10105 report and the loop seal purge delay curve (Figure 8) for a Crosby 6M6 forged safety valve. The opening characteristics of the Crosby 6M16 safety valves in Farley plants behave similarly with the Crosby 6M6 safety valves. Furthermore, a review of EPRI data, it is confirmed that the pressure increase ramp rate from 2 to 375 psi/sec envelops the ramp rate for Farley.

A nonlinear opening area time history valve characteristics are considered in the latest thermal hydraulic analysis. In addition, an average loop seal temperature of about 200°F, which is below the average Farley loop seal temperature, is used for the loop seal water slug properties. This method use along with programs ITCH and FORFUN is first benchmarked against the previous EPRI test results and good correlations are found (See Figures 3-6). For the Farley plant specific application, the thermal hydraulic forces are generated using the nonlinear valve opening area time history method. The application of this method results in a reduction in the hydraulic thrust forces due to the water slug being more slowly passed through the valve (with 5-10% opening area) before the valve is fully open. The water hammer effect is thus reduced.

Comparison of Thermal Hydraulic Forces

The thermal hydraulic forces generated by considering time history variable valve opening area are tabulated in Table 2. Since the forces with the 10% initial valve opening area are more conservative than those with the 5% initial valve opening area, they are used to perform the time history structural analysis of the pressurizer safety valve piping system. For the

Farley plant specific safety valves, the actual initial valve opening area is 5% as determined by documented valve characteristics calculations.

Thermal Hydraulic Analysis Computer Programs

The computer program used for the thermal hydraulic analysis is ITCH on Sun workstation.[11c] This program was upgraded several times from original program ITCHVALVE[11a,11b] since 1982 and was renamed to ITCHVENT once on the mainframe computer. The program ITCHVENT was converted to Sun workstation in 1992 (reference 11c). Program ITCHVALVE was benchmarked against the EPRI test data. ITCHVALVE is a 1-D thermal hydraulic code that calculates the time history fluid properties within the PSARV system for the condition when the safety or relief valves open. The thermal hydraulic forces are calculated by another program called FORFUN[11d] considering the momentum changes for the fluid in each element of the piping segment. Mainframe programs FORFUN was also converted to Sun workstation in 1992 (Reference 11d).

2.2 Structural Modeling and Analysis Methods

The structural modeling and analysis of the pressurizer safety valve piping system were performed using the WECAN Computer Code[12]. The piping system was modeled by pipe, elbow, support stiffness elements with both elastic and elastic/plastic capabilities. Consistent mass effect was considered in the analysis. For the analysis of the piping system with combination of deadweight and safety valve thrust discharge loadings, WECAN dynamic transient time history analysis option was chosen. The input time-history was determined by ITCH and FORFUN computer programs and was applied to the piping system structural model.

Figure 7 shows the structural model of the Unit 2 safety line system, which contains three 6-inch safety valves on three lines before meeting a 12-inch common header. The 12-inch common header leading to the pressurizer relief tank is also in the model. Part of the relief line piping was modeled in the structural system to account for the structural system interactions. Structural analyses were made for both Units 1 and 2.

The time-history solution for the dynamic thrust analysis of safety valve discharge with loop seal water slug was obtained from WECAN computer programs using direct integration methods. Since the purpose of this analysis is to determine the elastic behavior of the piping system under the extreme loading of valve thrust, the linear-elastic option of the WECAN program was used. The resulting stress at 8 equally spaced circumferential points of a given cross-section were calculated for a 1.0 second time-history following the three safety valve discharge action simultaneously.

3.0 PIPING COMPONENT AND SUPPORT EVALUATION

3.1 Piping Component Systems Evaluation Criteria

The pressurizer safety and relief valve piping system was originally qualified to its design basis allowables prior to 1980 TMI requirements. The design basis was the requirements of ASME B&PV Code Section III, 1971 edition, including summer 1971 addenda for Class 1 piping and the ANS B31.1-1967 Code with 1971 addenda for the NNS piping. In 1982, Westinghouse performed additional evaluation to address TMI related issues by considering the cold loop seal loads for these piping systems[5]. Criteria used in that analysis was based on the recommendation from piping subcommittee of the PWR PSARV test program and was documented in a WCAP-10105[10]. That criteria was reviewed and accepted by the NRC in a 1986 SER[6].

In this evaluation, the same loading combination and piping evaluation criteria in WCAP-10105 were used with the exception of allowable stress of 2.4 S_h for Emergency Condition for NNS portion of the piping system, as an alternative to the hardship of safety line discharge piping modification in terms of radiation exposure and expenses. This exception has been discussed with the NRC staff (reference 13). The load combination and evaluation criteria are provided in tables 3 and 4.

3.2 Piping Component Evaluation Results

3.2.1 Piping Components

Using elastic analysis techniques, the Class 1 piping (which connects the pressurizer safety line nozzle to the 6" safety valve), were qualified to the allowables listed in Table 3 with the effect of valve thrust under both emergency and faulted conditions. The NNS portions of the piping system area also qualified to meet the allowables listed in Table 4.

3.2.2 Safety Valve Nozzles

One additional means to ensure that the safety valve remains operable after the loop seal water is discharged is to assess the valve nozzle loads with respect to the valve operability limit provided in the equipment specification[14]. For emergency condition, the calculated valve nozzle loads from the combination of deadweight, pressure and valve thrust effects are within the equipment specification allowable. This allowable requires the maximum total valve nozzle stress to be 75% of the yield stress of the nozzle material at temperature. In addition, it further requires that the maximum bending stress be 50% and the maximum torsion stress also be 50% of the yield stress of the nozzle at temperature.

3.3 Support Component Evaluation Results

3.3.1 Loading and Load Combinations

The piping system loading conditions considered for the pipe support evaluation consisted of the valve thrust loadings discussed above in combination with the existing design basis dead weight, normal thermal expansion, transient thermal expansion, and the OBE & SSE seismic event loadings.

Since the pipe supports had previously been qualified for the Normal, Upset, Emergency, and Faulted conditions, the supports were only evaluated for the worst case load combination including the valve thrust loads from the piping system analysis. The loading combination used for support evaluation is:

$$P = DW \pm Thm_{max/min} \pm \sqrt{SSE^2 + Thrst^2}$$

3.3.2 Stress Acceptance Criteria

The purpose of the support evaluation was to demonstrate that the supports retained their integrity for the controlling combined loads as discussed in Section 3.3.1. This was accomplished by generally limiting the actual support member stresses to the allowable stress limits established by the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF and Appendix F, 1974 Edition. The code of record, AISC 7th Ed., does not address the faulted loading combination. ASME Subsection NF was used for this evaluation since it is essentially the same as AISC for the normal and upset conditions, and it provides criteria for the extreme faulted loading combination. In addition, the Subsection NF criteria is consistent with the pipe support criteria utilized by most other nuclear plants.

In accordance with I&C Bulletin 79-02, concrete expansion anchors (CEA) on Class I pipe support base plates were limited to manufacturer's allowables including a Factor of Safety of 4.0. However, for the CEA on NNS Class Pipe Support Base Plates the manufacturer's allowable including a factor of safety of 3.0 was used[13]. The use of this safety factor was discussed with the NRC for this application based on the radiation exposure and expenses associated with an alternative of safety line discharge piping modification.

3.3.3 Results

Class 1 supports -- the results of the pipe support evaluations based on the as-built support data provided to Westinghouse show that all the Unit 1 and Unit 2 pipe support standard Grinnell components,[15] structural members, and base plate element[16],[17] stress levels are within the allowable stress limits of ASME Subsection NF and Appendix F and will maintain their structural integrity and stability for the faulted loading combination provided in

Section 3.3.1. All concrete expansion anchor for class 1 supports have a minimum safety factor of 4.0.

NNS supports -- all Unit 1 and Unit 2 NNS pipe supports satisfied the ASME Subsection NF and Appendix F faulted stress criteria. Therefore, all the NNS pipe supports will maintain their structural integrity for the specified loading combination. Most expansion anchors have safety factor greater than 4.0. Table 7 provides a summary of only those NNS class pipe supports which have concrete expansion anchors with safety factor less than 4.0 but greater than 3.0 in their qualification[18]. Therefore, they are acceptable per reference [13].

4.0 SUMMARY AND CONCLUSIONS

The purpose of the analysis and evaluation described in this report is to address the concerns identified in the NRC Safety Evaluation Report (SER) published in 1986 concerning NUREG-0737, Item II.D.1[6]. As discussed in the introduction of this report, subsequent to the NRC SER, a variable loop seal condition in the pressurizer safety lines has been achieved due to the modification of insulation on the pipe. As a result of the new insulation, water temperatures in the loop seal were increased. Considering the valve opening characteristics and the increased loop seal temperatures due to insulation, new thermal hydraulic loads were generated. The method in this report reflects a more realistic yet conservative valve discharge condition.

With these time history thermal hydraulic thrust loads applied simultaneously to the three safety line system, the elastic responses of the system meet all the stress allowables listed in Table 3 for Class 1 piping and Table 4 for NNS piping.

In addition to the pipe, all pipe supports and their structural embedments were evaluated to their faulted stress limits and found to be acceptable.

5.0 REFERENCES

1. NUREG-0737, "Clarification of TMI Action Plan Requirements," NRC, November, 1980.

2. "Application of RECARS/MODI for Calculation of Safety and Relief Valve Discharge Piping Hydrodynamic Loads," EPRI NP-2479, Final Report, Dec. 1982.

3. Letter, F. L. Clayton, Jr., to S. A. Varga, NRC, "Response to NUREG-0737/NUREG-0660 TMI Action Plan Requirements," April 1, 1982.

4. Letter, F. L. Clayton, Jr., to S. A. Varga, NRC, "Joseph M. Farley Nuclear Plant - Units 1 and 2 NUREG-0737, Item II.D.1 Response," July 1, 1982.

5. Letter, F. L. Clayton, Jr., to S. A. Varga, NRC, "Joseph M. Farley Nuclear Plant, Units 1 and 2 NUREG-0737, Item II.D.1," November 4, 1982.

6. Letter, E. A., Reeves, NRC to R. P. McDonald, Alabama Power Co., "Completion of Review of Item II.D.1 NUREG-0737 Safety and Relief Valve Testing for Joseph M. Farley Nuclear Plant Unit Nos. 1 and 2," December 16, 1986.

7. Letter, R. P. McDonald to L. S. Rubenstein, NRC, "Joseph M. Farley Nuclear Plant - Units 1 and 2 Completion of NUREG Item II.D.1 Review," February 5, 1987.

8. Letter, W. G. Hairston, III to NRC, "Joseph M. Farley Nuclear Plant, Units 1 and 2 NUREG-0737, Item II.D.1, Review Completion Schedule Update," September 16, 1988.

9. Letter, J. E. Garlington to J. A. Knochel, Westinghouse, Activity Code 52073, ES-91-2073, "Pressurizer Loop Seal Analysis," November 27, 1991.

10. Westinghouse Report WCAP-10105, "Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety Valve and Relief Valve Test Program", June, 1982.

11. (a) W Report WCAP-9924, "ITCHVALVE Code Description and Verification", M. A. Berger & K. S. Howe, July, 1982.

(b) Westinghouse Internal Letter SE&PT-CSE-465 1/31/90, "Release of Thermal Hydraulic and Related Computer Codes (ITCHVALVE, ITCHVENT, FORFUN, KJTRPLT2) for Production Use on the Cray X-MP."

(c) Westinghouse Program ITCH Sun Workstation Version 1.0 Configured 9/7/92.

(d) Westinghouse Program FORFUN Sun Workstation Version 1.0 Configured 10/30/92.

12. W Computer Program WECAN, WECAN/PLUS User's Manual, Dec. 1, 1990, First Edition, Westinghouse Electric Corp. Pittsburgh, PA.

13. NRC letter, Byron, L. Siegel, "Meeting Summary Related to TMI Issue II.D.1 (tag Nos. M84666 and M84667)," Joseph M. Farley Nuclear Plant, Units 1 and 2, May 9, 1994.

14. Westinghouse Equipment Specification #952445, Rev. 1, March 31, 1977.

15. Letter from Frank Birch, Grinnell Corp. to J. Himler, Westinghouse, #FB1V/1940D, "Farley Units 1 and 2 Project," April 3, 1992.

16. Bechtel Letter AP-19877, 4/3/92, "Pressurizer Loop Seal Analysis," Unit 2.

17. Bechtel Letter AP-19975, 4/30/92, "Pressurizer Loop Seal Analysis," Unit 1.

18. Bechtel Letter REA-94-0528, 7/25/94, "Evaluation of Revised Footprint Loads for Farley PSARV," Bechtel File A-88. In reply refer to W-2735.

TABLE 1

LOOP SEAL TEMPERATURE DATA
(DEG. F)

Farley Unit 1				Farley Unit 2			
LOC	LOOP A	LOOP B	LOOP C	LOC	LOOP A	LOOP B	LOOP C
1	653	653	653	1	653	653	653
2	560	552	536	2	548	529	557
3	502	470	432	3	483	440	495
4	457	424	350	4	465	450	467
5	457	424	350	5	465	450	467
6	404	343	289	6	416	410	397
7	404	343	289	7	416	410	397
8	283	228	210	8	290	290	276
9	160	144	141	9	151	148	154

TABLE 2

FORCE COMPARISON FOR DIFFERENT LOOP SEAL CASES

FORCE	5% Initial Valve Opening Area	10% Initial Valve Opening Area
1	165	224
2	207	277
3	187	243
4	599	710
5	169	181
6	1038	1164
7	195	1727
8	323	1164
9	9342	12681
10	2565	4397
11	12640	18912
12	5526	8290
13	6618	9592
14	26593	33399
15	5554	6228
16	165	231
17	207	285
18	187	251
19	599	740
20	169	195
21	1038	1249
22	335	2592
23	8901	12591
24	3907	6108
25	165	231
26	207	285
27	187	251
28	599	739
29	169	185
30	1038	1244
31	303	2482
32	5098	8586
33	2450	4142
34	1387	2216
35	1753	2269
36	3776	5585

TABLE 3

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER AND RELIEF VALVE PIPING - UPSTREAM OF VALVES CLASS 1 PIPING

Plant/System Operating Condition	Load Combination	Piping Allowable Stress Intensity
Normal	N	$1.5\ S_m$
Upset	N + OBE	$1.5\ S_m$
Upset	$N + SOT_U$	$1.5\ S_m$
Upset	$N + OBE + SOT_U$	$1.8\ S_m/1.5\ S_y^{(3)}$
Emergency	$N + SOT_E$	$2.25\ S_m/1.8\ S_{y(3)}$
Faulted	$N + SSE + SOT_F$	$3.0\ S_m$

NOTES: (1) See Table 5 for definitions of load abbreviations

(2) Use SRSS for combining dynamic load responses.

(3) The smaller of the given allowable is to be used.

TABLE 4

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY AND RELIEF VALVE PIPING - DOWNSTREAM OF VALVES
NNS PIPING

Plant/System Operating Condition	Load Combination	Piping Allowable Stress
Normal	N	1.0 S_h
Upset	N + OBE	1.2 S_h
Upset	N + SOT_U	1.2 S_h
Upset	N + OBE + SOT_U	1.8 S_h
Emergency	N + SOT_E	2.4 S_h*
Faulted	N + SSE + SOT_F	2.4 S_h

NOTES: (1) See Table 5 for definitions of load abbreviations

(2) Use SRSS for combining dynamic load responses.

*see reference [13]

TABLE 5

DEFINITIONS OF LOAD ABBREVIATIONS

N	=	Sustained loads during normal plant operation
SOT	=	System operating transient
SOT_U	=	Relief valve discharge transient
SOT_E	=	Safety valve discharge transient
SOT_F	=	Max (SOT_U; SOT_E); or transition flow
OBE	=	Operating basis earthquake
SSE	=	Safe shutdown earthquake
S_h	=	Basic material allowable stress at maximum (hot) temperature
S_m	=	Allowable design stress intensity
S_y	=	Yield strength value

TABLE 6

FARLEY UNITS 1 AND 2
SAFETY LINE PIPE STRESS AND STRAIN SUMMARY
FOR EMERGENCY CONDITION

Node Point	Piping Components	Code Maximum Stress (ksi)	Allowable Stress (ksi)
1290*	Butt weld at valve end nozzle	15.1	18.8
1460*	Long radius elbow	34.2	36.45
100**	Branch connection	32.9	44.67
690**	Reducer	25.1+	44.67
1490**	Welded attachment at support R120***	54.97***	55.42

* ASME Class 1 piping, upstream of safety valves
** ASME NNS piping, downstream of safety valves
*** Based on ASME Code Case N-318 allowable
\+ Stress index based on ANSI B31.1-1967, including 1971 Addenda

TABLE 7

FARLEY NUCLEAR PLANT - TMI ACTION NUREG-0737, II.D.1
UNITS 1 AND 2 PSARV LINE PIPE SUPPORTS EVALUATED UNDER REA 94-0528
ANCHOR BOLT DATA FOR SUPPORTS WITH FACTOR OF SAFETY F.S. <4

Unit	Serial No.	Support Mark No.	Total No. of Bolts	No. of Bolts w/ F.S. ≥4	No. of Bolts w/ F.S. <4	Actual F.S.		Types of Bolts with F.S. <4
						Bolt #	F.S.	
1	1	RC-R61	4	2	2	#3 #4	3.57 3.57	#3 and 4 3/4 ϕ HILTI KWIK
2	1	2RC-131X	5	3	2	#2 #5	3.77 3.20	#2 and #5 1/2" ϕ HILTI KWIK

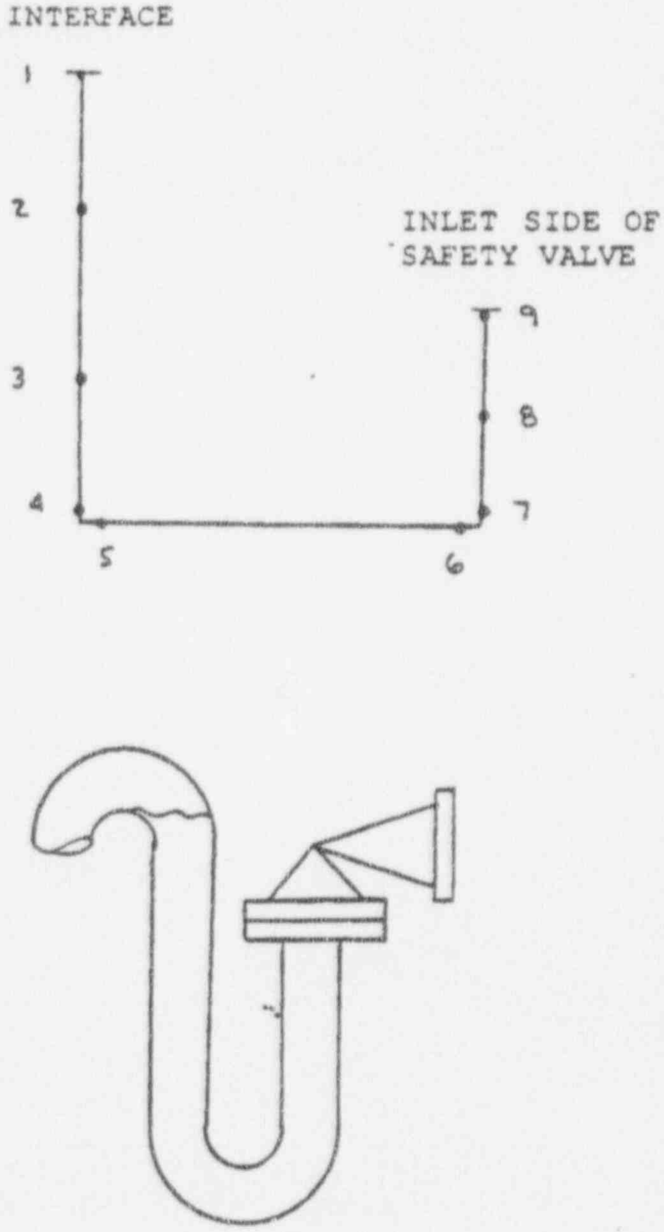

FIGURE 1
GENERAL LOOP SEAL
PERMANENT NODE LOCATIONS

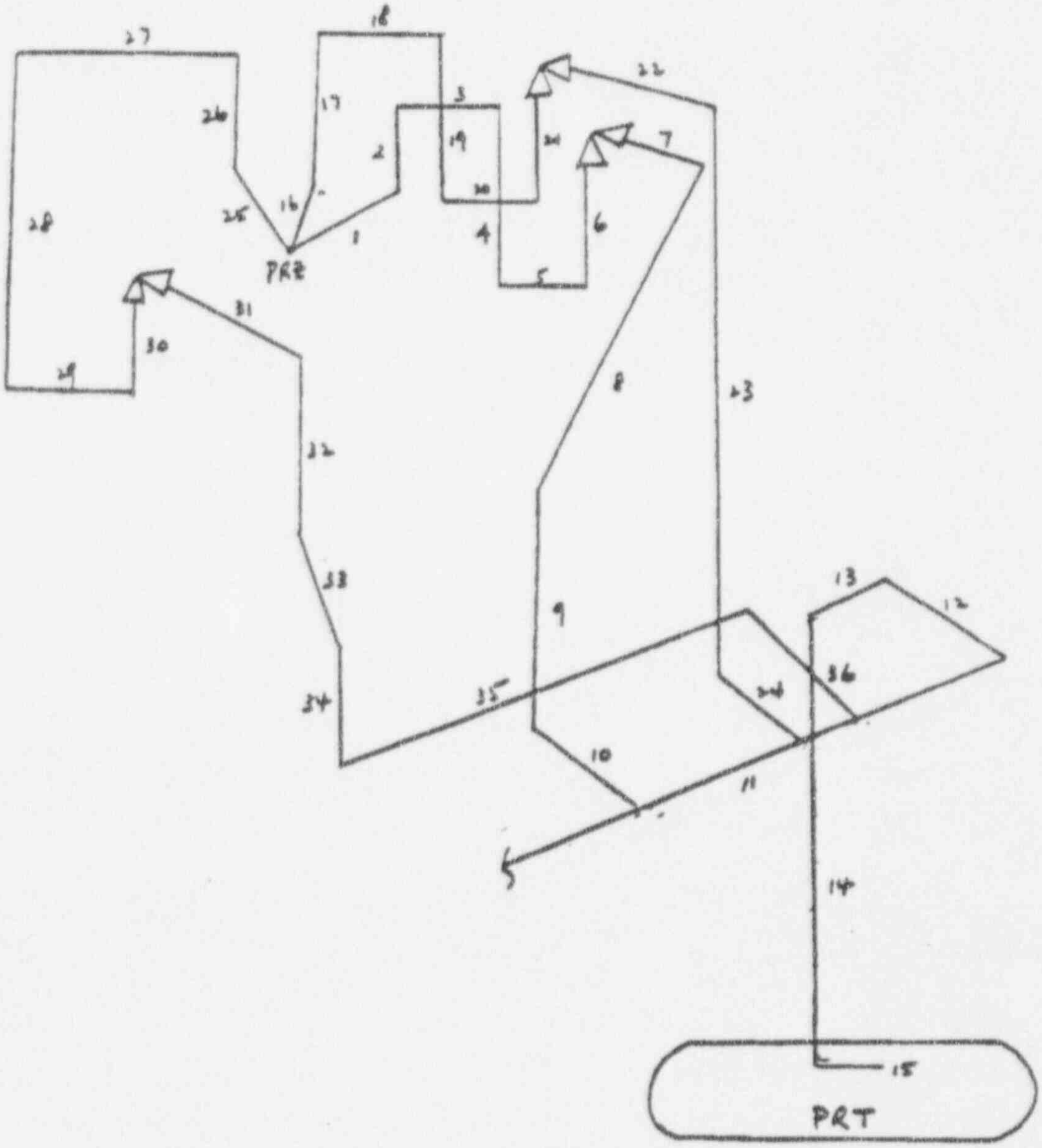

FIGURE 2
FARLEY PSARV SYSTEM HYDRAULIC FORCE LOCATIONS
(UNITS 1 AND 2)

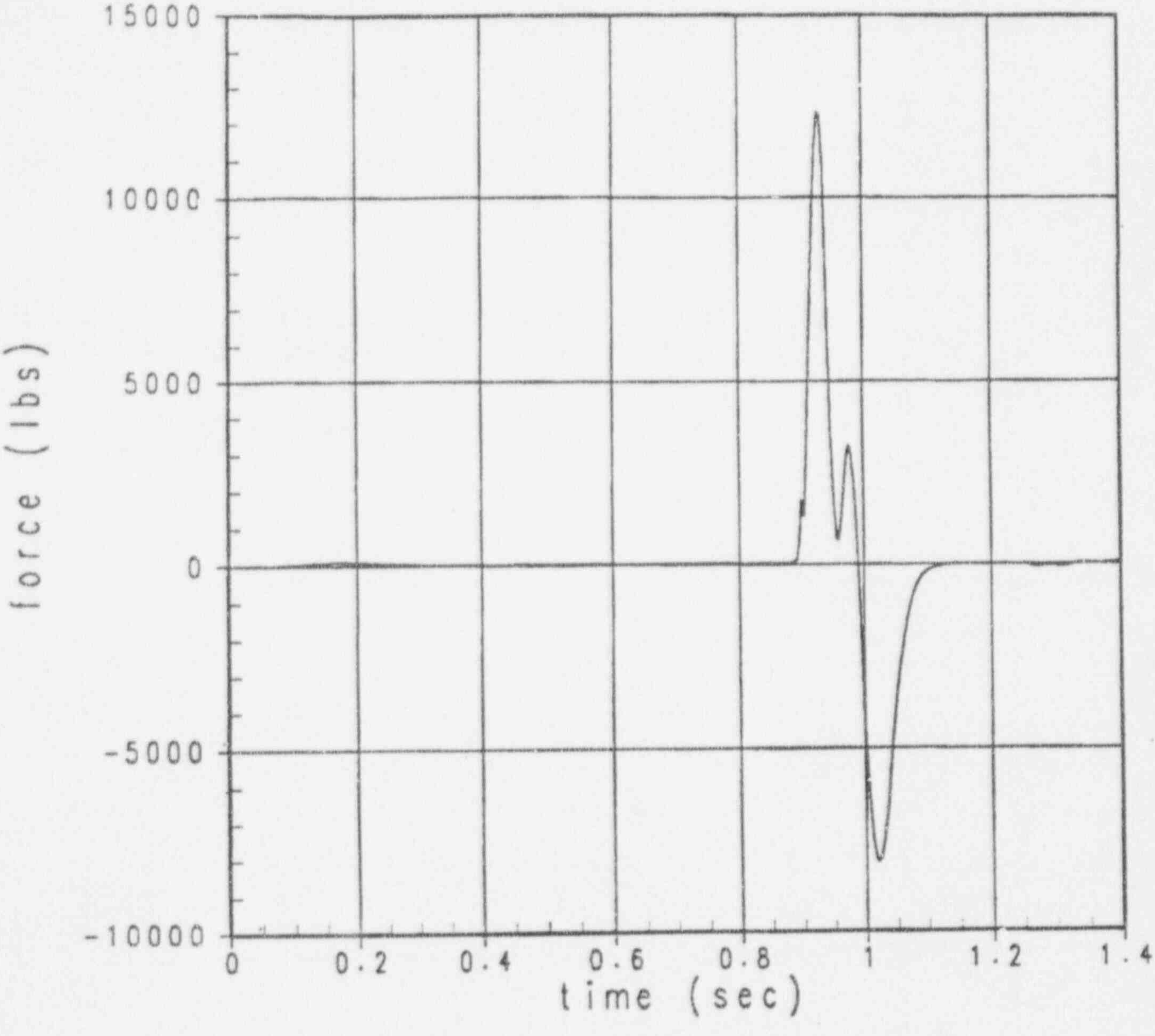

FIGURE 3
THERMAL HYDRAULIC FORCE TIME HISTORY FOR WE32/WE33
ON EPRI TEST 917 MODEL USING ACTUAL VALVE
OPENING CHARACTERISTICS

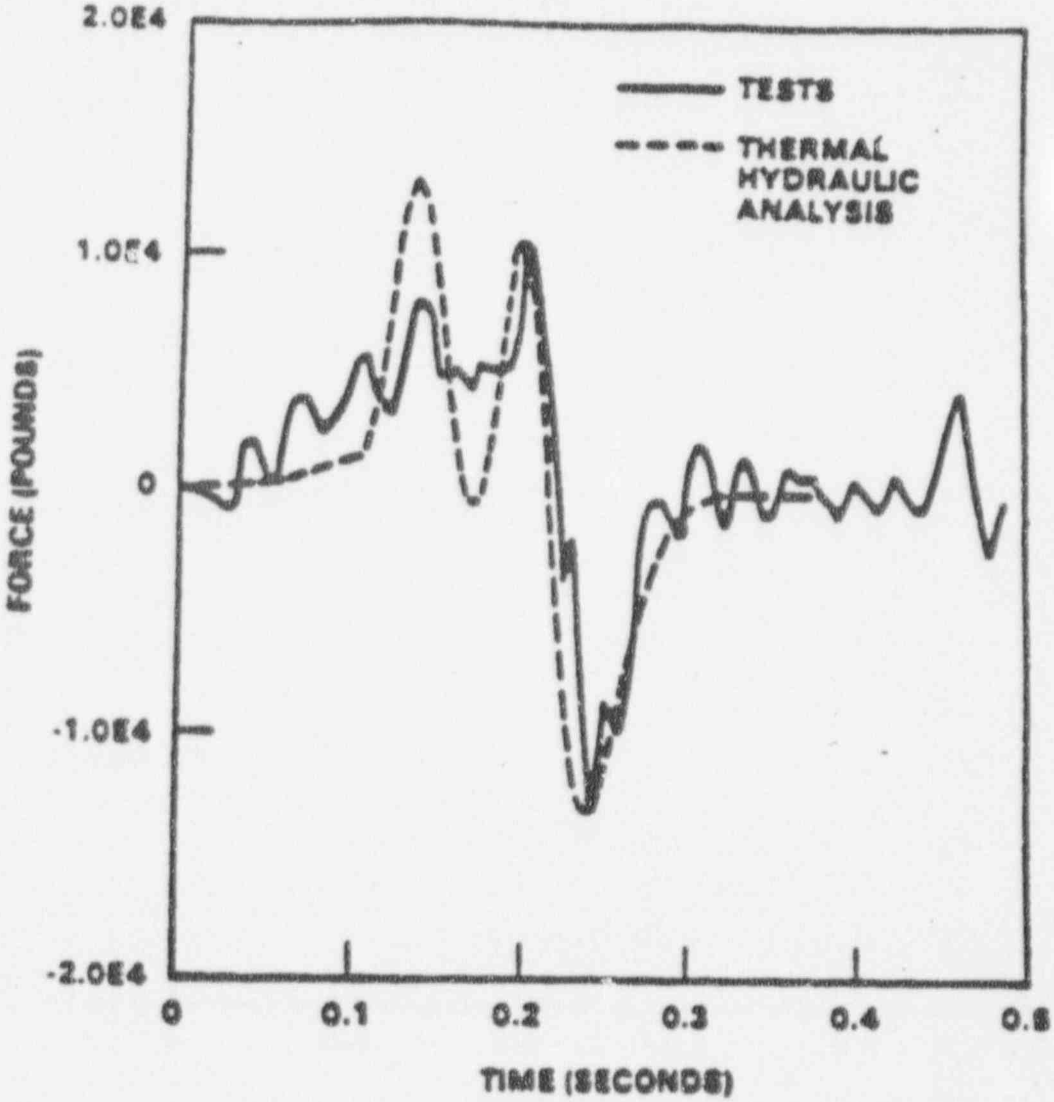

FIGURE 4
COMPARISON OF THE EPRI FORCE TIME-HISTORY FOR
WE32 AND WE33 FROM TEST 917 WITH THE
THERMAL HYDRAULIC ANALYSIS PREDICTED FORCE TIME HISTORY

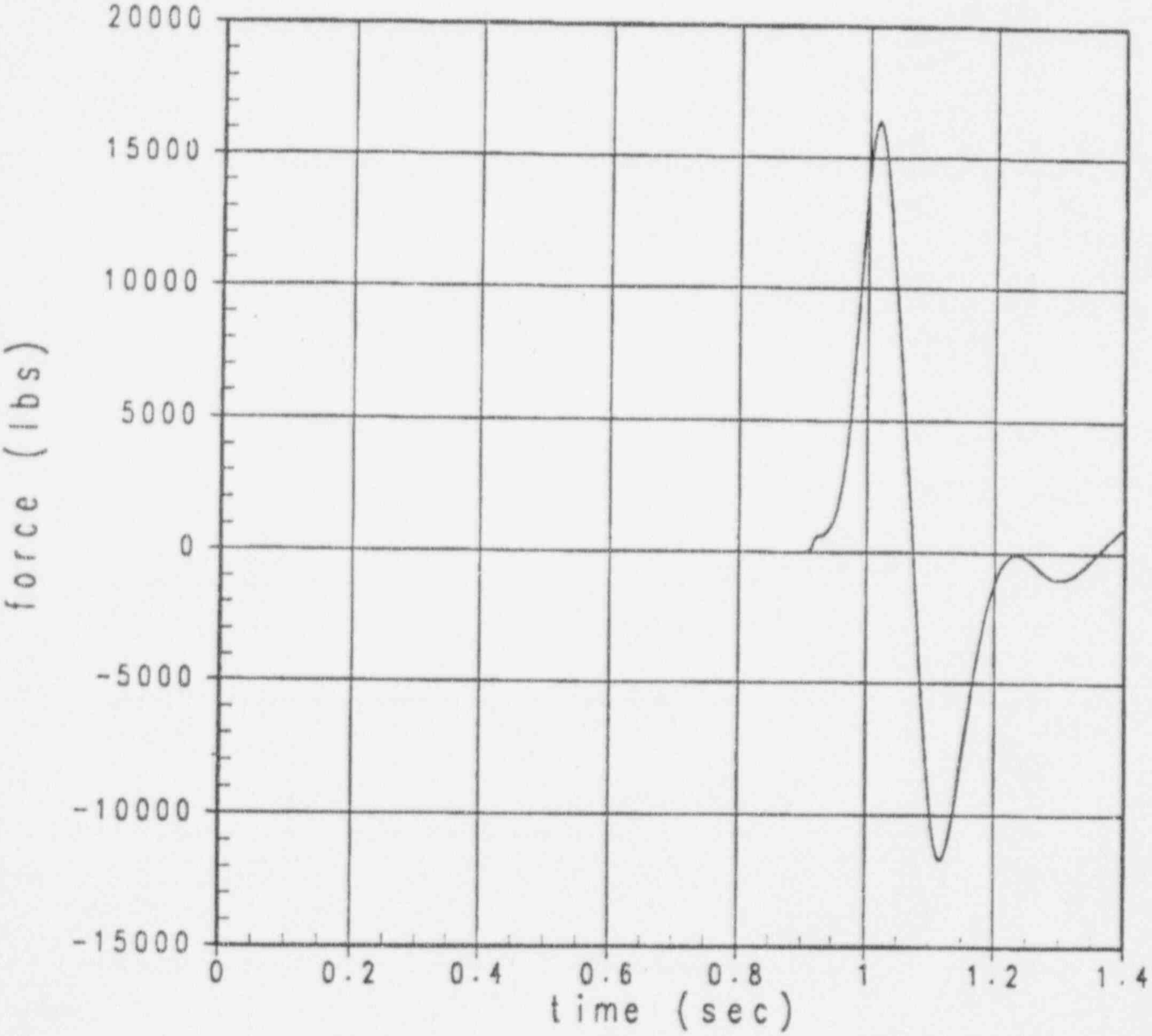

FIGURE 5
THERMAL HYDRAULIC FORCE TIME HISTORY FOR WE30/WE31
ON EPRI TEST 917 MODEL USING ACTUAL VALVE
OPENING CHARACTERISTICS

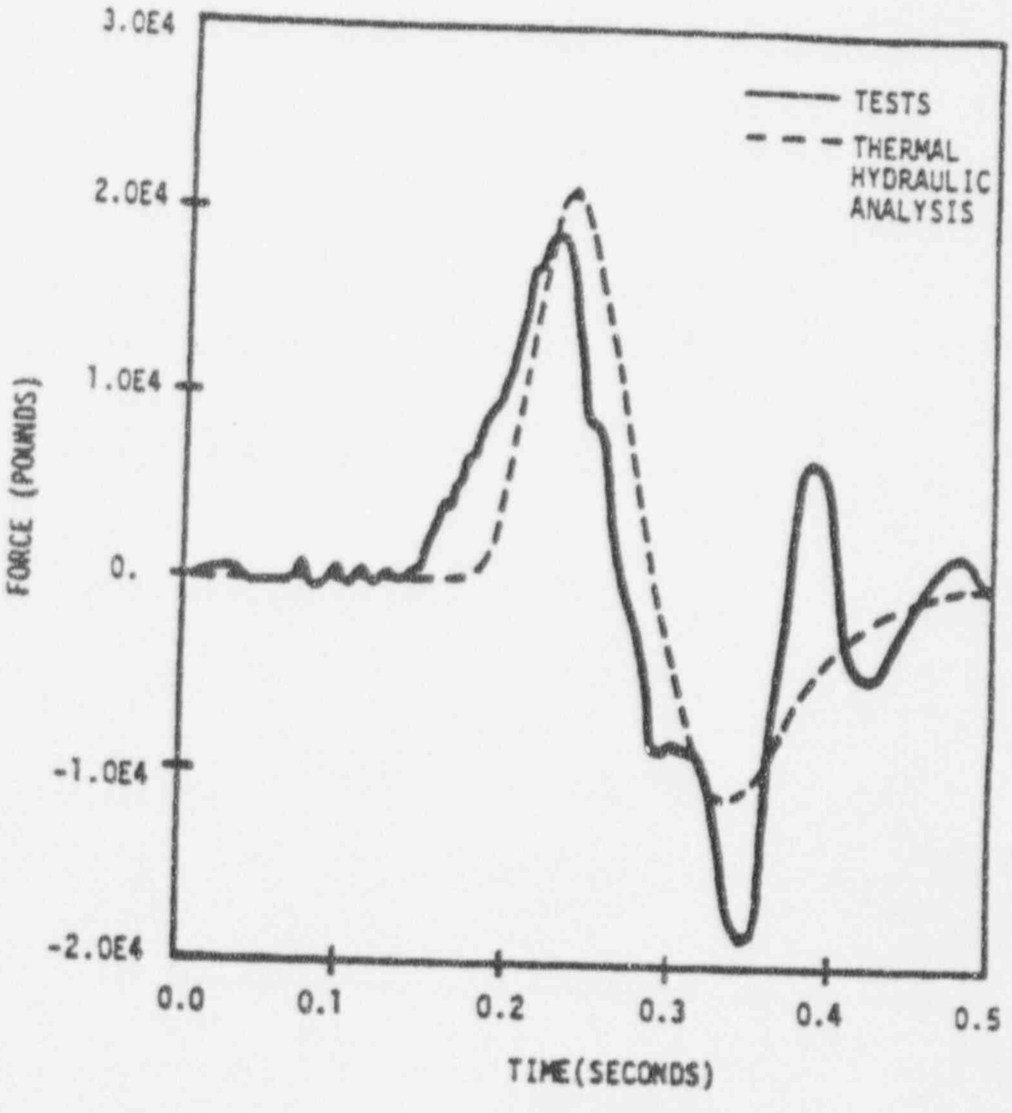

FIGURE 6
COMPARISON OF THE EPRI FORCE TIME-HISTORY FOR
WE30 AND WE30 FROM TEST 917 WITH THE
THERMAL HYDRAULIC ANALYSIS PREDICTED FORCE TIME HISTORY

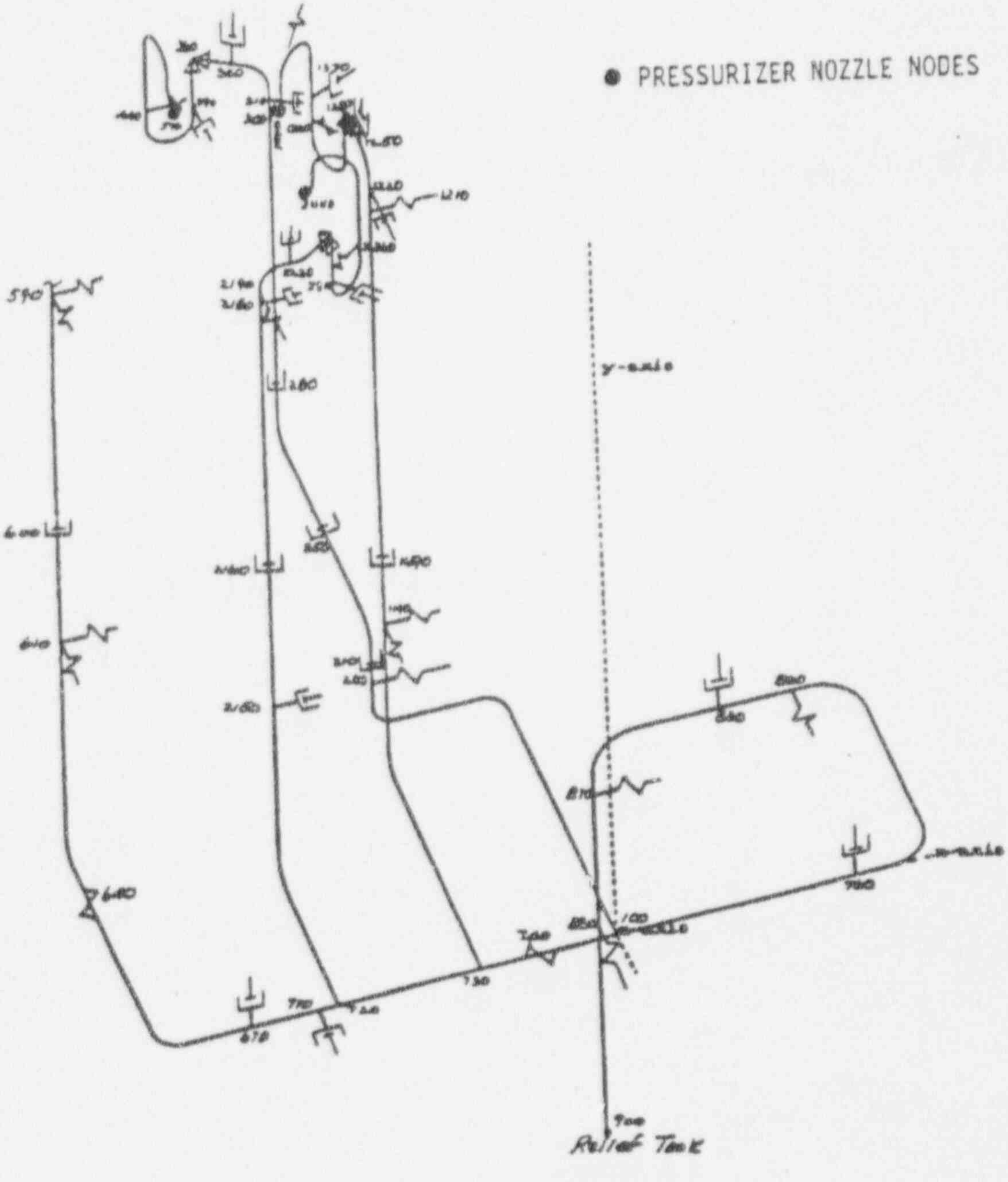

FIGURE 7
PRESSURIZER SAFETY LINE SYSTEM STRUCTURAL MODEL

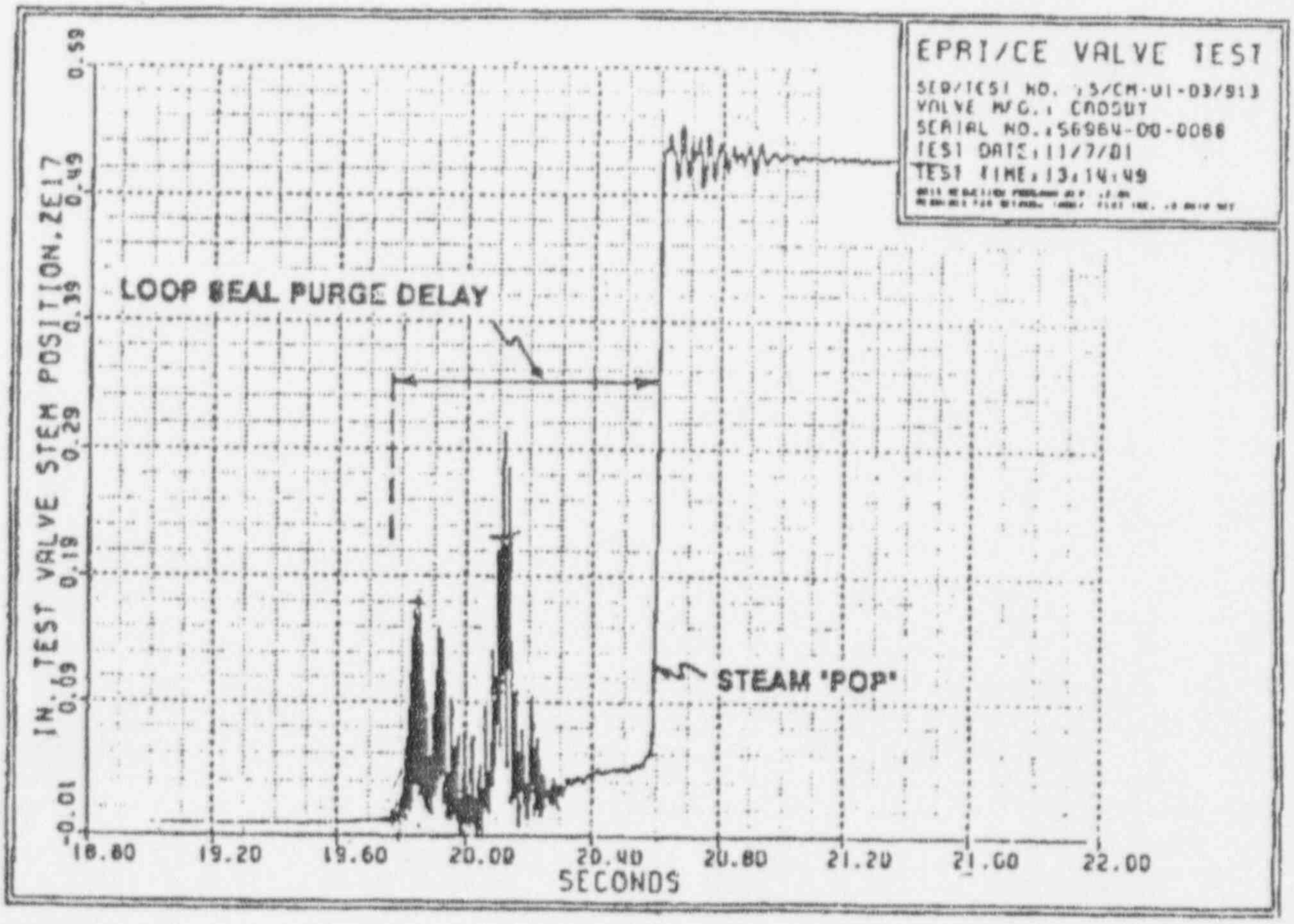

FIGURE 8
EPRI LIFT VS. TIME FOR WATER TEST (6M6 FORGED)