SCE&G

South Carolina Electric & Gas Company
P.O. Box 88
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(803) 345-4041

Dan A. Nauman
Vice President
Nuclear Operations

May 5, 1988

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. J. J. Hayes, Jr.

SUBJECT: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
Relief and Safety Valves
NUREG-0737, II.D.1

Gentlemen:

Please find attached the South Carolina Electric & Gas Company (SCE&G) response to NRC questions dealing with NUREG 0737, item II.D.1. These question responses have been structured to provide the most concise and applicable answers to the NRC specific requests on the original submittal.

If you should have any further questions, please advise.

Very truly yours,

OSBradham for
D. A. Nauman

AMM:DAN/lcd
Attachment

pc: J. G. Connelly, Jr./O. W. Dixon, Jr./T. C. Nichols, Jr.
E. C. Roberts
W. A. Williams, Jr.
J. N. Grace
J. J. Hayes, Jr.
General Managers
C. A. Price
R. B. Clary
W. R. Higgins
R. M. Campbell, Jr.
K. E. Nodland
J. C. Snelson
G. O. Percival
R. L. Prevatte
J. B. Knotts, Jr.
NSRC
RTS (NRR 1591)
File (811.11)

A046
1/1

RESPONSES TO SAFETY EVALUATION QUESTIONS
TMI ACTION NUREG-0737 II.D.1
FOR V. C. SUMMER UNIT 1

EG&G QUESTION NO. 1

The Westinghouse valve inlet fluid conditions report stated that liquid discharge through both the safety and Power Operated Relief Valves (PORVs) is predicted for a FSAR feedline break event. The Westinghouse report gave expected peak pressure, pressurization rate, and fluid temperature range for an FSAR feedline break at the V. C. Summer Plant. The V. C. Summer Plant specific submittal, however, does not address this event. NUREG-0737 requires analysis of accidents and occurrences referenced in Regulatory Guide 1.70, Revision 2, and one of the accidents so required is the feedline break. Therefore, assure that the fluid conditions for this were enveloped in the EPRI tests and that the time period of water relief in the EPRI tests was as long as expected at the plant. Demonstrate operability of the safety valves and PORVs for this event and assure that the feedline break event was considered in analyses of the piping system.

RESPONSE

The issue of feedline break analysis and its relevance to safety valve performance is addressed in WCAP-11677, "Pressurizer Safety Relief Valve Operation for Water Discharge During a Feedwater Line Break."

V. C. Summer was encompassed by the WCAP (see Table 2-1) and it was shown that following the liquid discharge predicted for the feedline break event, the valves would reseat and continue to operate reliably. The WCAP also concluded that the number of cycles the valves would experience are within acceptable limits.

While pressurizer PORVs are conservatively not assumed in the FSAR feedline break analysis or included in WCAP-11677, if the valves should fail during operation, the remotely-operated block valves downstream of the PORVs can be shut by the operator to terminate the flow. In fact, the Emergency Operating Procedures were designed to have the operators perform this action if this scenario is diagnosed.

From a piping design standpoint, loop-seal discharge transients (liquid followed by steam) are more severe, i.e., produce higher loads, than all liquid discharge transients. RELAP5/MOD1 analyses of the V. C. Summer SRV/PORV system for both liquid followed by steam and all liquid discharge reaffirmed this assessment. The FSAR feedline break transient would result in all liquid discharge through the SRV/PORVs. Thus, it can be concluded that a VCS SRV/PORV system piping design based on a loop seal discharge transient (liquid followed by steam) is more conservative than a design based on an all liquid discharge transient. Therefore, the feedline break transient is bounded by the analysis.

EG&G QUESTION NO. 2

Results from the EPRI test on the Crosby safety valves indicate that the test blowdowns exceeded the design value of 5% for both "as installed" and "lowered" ring settings. If the blowdowns expected for the plant (see Question 4) also exceed 5%, the higher blowdowns could cause a rise in pressurizer water level such that water may reach the safety valve inlet line and result in a steam-water flow situation. Also the pressure might be sufficiently decreased such that flashing occurs in the primary loop or the reactor vessel, natural circulation is interrupted, and adequate cooling for decay heat removal is not achieved. Discuss these consequences of higher blowdowns if increased blowdowns are expected.

RESPONSE

The impact on plant safety of excessive pressurizer safety valve blowdowns (up to 14%) was evaluated for V. C. Summer. The results of this evaluation showed no adverse effects on plant safety.

Safety valve blowdowns in excess of that assumed in the V. C. Summer FSAR will have the following effects on the events in which safety valve actuation occurs:

1. Increased pressurizer water level during and following the valve blowdown.

2. Lower pressurizer pressure during and following valve blowdown.

3. Increased inventory through the valve.

The impact of the increased safety valve blowdowns with respect to the above effects were evaluated for the V. C. Summer FSAR events in which the safety valve actuation occurs (i.e., Loss of External Electrical Load, Single Reactor Coolant Pump Locked Rotor, and Major Rupture of the Main Feedwater Pipe).

For the Loss of External Electrical Load event, results from sensitivity analyses performed for a 4-loop plant were used for the evaluation. These analyses investigated the effects of different blowdown rates on the event. Similar results are expected for a 3-loop plant. The results of these analyses showed only marginal increases in pressurizer water volume and the maximum pressurizer water levels were well below the level at which liquid relief would occur. The V. C. Summer FSAR analysis results show that a small increase in pressurizer water volume, due to increased safety valve blowdown, will not result in liquid relief. The sensitivity analyses also showed that peak RCS pressures were unaffected by the increased blowdowns. The increased blowdowns did result in lower pressurizer pressure and increased RCS inventory loss; however, these had no adverse impact on the event and adequate decay heat removal was maintained.

For the Single Reactor Coolant Pump Locked Rotor event, increased safety valve blowdowns have little impact on the event. As analyzed and presented in the V. C. Summer FSAR, the opening and closing of the safety valve occurs over a short time period (less than 3 seconds). As a result, there is little change in either pressurizer level or RCS inventory. Increased safety valve blowdowns would have no impact on peak pressure, peak clad temperature, or DNBR as these occur prior to the closing of the safety valve.

For the Major Rupture of a Main Feedwater Pipe, the current FSAR analysis, as explained in WCAP-11677, will result in three cycles of water discharge (Table 4.4). The concern about potential pressurizer fill as a result of the increased blowdown is meaningless for this event since water relief is already predicted. The overall effect on the transient would be to relieve more mass with each opening. Thus, it is possible that under these conditions only two cycles might result.

The increased blowdown of the safety valves will have no adverse impact on the transient or the ability to mitigate the transient which is provided by the Emergency Feedwater System.

EG&G QUESTION NO. 3

The submittal does not identify the ring settings to be used on the Crosby 6M6 safety valves or what effect these settings have on valve performance in the V. C. Summer installation. Provide the final ring settings selected for the V. C. Summer safety valves. Identify the expected blowdowns corresponding to these plant ring settings and explain how the blowdowns were extrapolated or calculated from test data. Verify that at these ring settings the valves can perform their pressure relief function and the plant can be safely shutdown with the blowdown and fluid conditions occurring at the plant.

RESPONSE

Ring settings for the V. C. Summer safety valves are as follows:

Valve Serial Number	Nozzle Ring	Guide Ring
N56964-01-0079	-18	-250
N56964-01-0078	-18	-225
N56964-01-0077	-18	-250

Please note that the ring setting given above were measured by Crosby from the "highest-locked position," as noted in Crosby procedures and in the EPRI reports "Definitions of Key Terms for Safety Valves." Ring settings reported by EPRI were measured from the "level position."

These ring settings were established by a method which includes a steam operational test on each valve by Crosby. Blowdowns measured during these production tests were equal to or less than 5 percent for all valves. The Crosby 6M6 valve EPRI tests seen with "typical PWR plant settings" had ring settings that were established by the same methods. Therefore, these EPRI tests can be used to show that the V. C. Summer valves can perform their intended function and the plant can be safety shut down.

EG&G QUESTION NO. 4

Results from EPRI tests on the Crosby 6M6 safety valve with loop seal internals show that during some tests the valve attained rate lift and rated flow at 3% accumulation while during other tests it did not. Provide a demonstration that the plant safety valves will pass their rated flow with the ring settings used.

RESPONSE

EPRI report NP-2770-LD, Volume 6, for the Crosby 6M6 safety valve shows in Table 4.4 that in every test for which data was taken, the valve achieved at least the rated flow. This was true at 3% accumulation regardless of the ring settings tested. Since the V. C. Summer valve ring settings were established by the same methods used to establish some of the EPRI test ring settings, the V. C. Summer valves can be expected to also achieve rated flow at 3% accumulation.

EG&G QUESTION NO. 5

During two EPRI hot loop seal-steam tests and one subcooled water test on the 6M6 safety valve, the valve fluttered and chattered upon closure. These tests were terminated by manually opening the valve to stop the chatter. The hot loop seal tests appear to be representative of conditions at the V. C. Summer plant and the liquid flow tests may be representative of a feedline break event (see Question 1). Justify that the valve behavior exhibited in these tests is not indicative of the performance expected for the V. C. Summer valves.

RESPONSE

Based on the temperatures corresponding to the actual V. C. Summer FSAR analysis, WCAP-11677 demonstrated that the Crosby 6M6 pressurizer safety valves would operate reliably. The particular test resulting in "chatter upon opening that stabilized" (Test 931b) was actually experiencing fluttering just prior to popping full open. This conclusion is based on a detailed review of the actual stem position tracings. Fluttering is defined as stem motions below half of the lift while chattering is defined as stem motion equal to the lift. Fluttering does not have any adverse impact on valve performance. Thus, this test did result in a "stable" discharge. Note that the predicted temperature of the water being discharged for V. C. Summer is in excess of 630 degrees-F. This information is also presented on page 10 of WCAP-11677.

The test that discharged subcooled liquid (Test 932) utilized fluid at 463 degrees-F. Since this temperature does not envelope the conditions indicative of the V. C. Summer Feedline Break analysis, it can be neglected. In other words, the test is not representative of the performance expected at V. C. Summer.

EG&G QUESTION NO. 6

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.

RESPONSE

The circuitry is class 1-E and qualifications of the solenoids are documented in the response to NUREG-0588, Revision 4, "Environm,ental Qualification of Safety Related Equipment." Class 1-E circuitry is powered from safety related power systems and is seismically designed.

EG&G QUESTION NO. 7

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 and outlet piping. Make a comparison between the predicted plant moments with the moments applied to the tested valves to demonstrate that the operability of the valves will not be impaired.

Response:

A. SRV qualification

The loads on the inlet side of the SRV's were generated by TES. The loads were transmitted to C. C. Barbier (GAI) (page 7-2), who subsequently passed them on to Crosby Valve & Gage. They were accepted per August 18, 1981 telephone memo of D. T. Klinksiek (GAI) and David Allen (Crosby Valve & Gage Company) (page 7-3). The thermal loads subsequently changed, but were lower than those originally used. Therefore, TES considered the valves as remaining qualified without need to contact Crosby Valve & Gage.

The loads on the outlet side of the SRV's were calculated by G/C. They were transmitted to Westinghouse, letter CGGW-1815 dated 7/15/82 (pages 7-4 through 7-9), which found the loads acceptable, letter CGWG-2628 dated 8/10/82 (page 7-10). Subsequently, new loads were generated on one valve (8010-A) and re-transmitted to Westinghouse, letter CGGW-1824 dated 7/30/82 (pages 7-11 and 7-12). Once again, the new loads were found to be acceptable, letter CGWG-2639 dated 8/13/82 (page 7-13).

B. PORV and block valve qualification

The loads on both the inlets and outlets of the PORV's and block valves were calculated by TES. The loads were qualified and found to be acceptable by TES in accordance with the guidelines set forth in Westinghouse Specification G-677458, and G/C Design Specification DSP-544R-044461-000. The loads are summarized on page 7-14.

C. Comparison of calculated loads to EPRI test loads

A comparison of the loads calculated to be imposed on the SRV outlet flange prior to lift, and the measured loads on the EPRI tested valve are illustrated in a table on page 7-15. Please note that Report EPRI NP-2770-LD, Volume 10 in section 3, part 3.1, page 3-1 states,

"The loads imposed on the safety valves during this test program had no measureable effect on valve operability. The maximum recorded bending moment acting on the safety valve discharge flange is reported for each valve test in Table 3-1. These valves are as-tested bending moments and do not constitute a maximum allowable moment above which the valve will no longer funtion."

Therefore, although the loads calculated to be imposed on the outlet flange are significantly less than the EPRI test loads, operability cannot be assured by this method. Operability was assured by having valve loads acceptable to the valve vendor as was demonstrated in Section A and B above.

To assist with the reviews, the details of the SRV supports have been included and are on pages 7-16 through 7-23. Please note that the SRV is anchored at its base and guided on the discharge to allow only axial movement. The only EPRI PORV moment was 43,000 in.-lb. which is larger than any allowable moment (page 7-14) to which the valves are qualified.

TELEDYNE
ENGINEERING SERVICE
303 BEAR HILL ROAD
WALTHAM MASSACHUSETTS 02254
(617) 890-3350 TWX (710) 324-7508

SCE & G CO.
P. O. Q?5[illegible]
V. C. SUMMER UNIT 1

August 13, 1981
4813-101

Mr. Charles C. Barbier
Gilbert Associates, Inc.
152 Fairbanks Road
Oak Ridge, TN 37830

Subject: Class 1 Analysis of Pressurizer Relief System CROSBY SRV Inlet Loads, V. C. Summer Station, Unit 1

Dear Mr. Barbier:

Per your request in our telephone conversation today please find the loads on the inlet side of the CROSBY SRV at the valve face for loops A, B and C. Please take note that M_x, M_z, F_x, F_z are horizontal and that M_y and F_y are vertical. Also note that the units are lbs for F_x, F_y, F_z and in-lbs for M_x, M_y and M_z. The coordinate system is Global.

	Loop A	Loop B	Loop C
F_x	+ 2,712, - 1,808	+ 5,037, - 5,307	+ 1,693, - 2,007
F_y	+ 12,016, - 8,500	+ 13,097, - 9,925	+ 16,099, - 12,287
F_z	+ 3,717, - 3,705	+ 503, - 441	+ 3,525, - 3,819
M_x	+171,410, -206,374	+ 10,765, - 18,537	+295,596, -245,336
M_y	+ 16,246, - 5,720	+ 11,108, - 3,810	+ 4,095, - 8,125
M_z	+130,711, -103,609	+209,755, -266,043	+103,610, -159,034

If you have any questions or comments, please call me.

Sincerely,
TELEDYNE ENGINEERING SERVICES

Patrick D. Harrison

Patrick D. Harrison, P.E.
Manager, Projects

PDH/ba

cc: D. F. Landers (TES)
K. W. Nettles (SCE&G)
C. A. Price (SCE&G)
D. R. Moore (SCE&G)
R. J. Hoffert (GAI)
A. R. Hoffert (GAI)
P. H. Schmitzer (GAI)
NPCF/Whitaker (SCE&G)
Tim Adams, Westinghouse
R. D. Ciatto (TES)
C. C. Barbier (GAI)
O. W. Dixon (SCE&G)
H. E. Yocom (GAI)
J. F. Bailey (SCE&G)
Carl Rentschler (GAI)
John Palmer, Crosby Valve
T. Matty, Westinghouse
TES Document Control

ATTACHMENT II

REFERENCE 13 GILBERT ASSOCIATES, INC

TELEPHONE AND CONFERENCE MEMORANDUM

DATE: August 18,1981

BY: D.T. Klinksiek

WORK ORDER NO. 044461-000

TELEPHONE CALL [X] CONFERENCE [ ]

WITH: David Allen

COMPANY: Crosby Valve & Gage

SUBJECT: Qualification of RC03

NOTES:

SCE & G CO.
P. O. Q250358-0003
V. C. SUMMER UNIT-1

Mr. Allen called this date to inform me that the review against the piping loads from Teledyne was completed. Crosby has no problems accepting these loads. A later telecon with Charles Barbier indicated that Crosby had been requested to use 700 psi at valve discharge during the review of the Teledyne load. GAI assumes that this pressure was used during the review of the Teledyne loads.

Dave Klinksiek

D.T. KLINKSIEK

Copies To:

P. Schmitzger (GAI)
C. Rentschler
F. Buchanan
D. Kershner
C.C. Barbier (GAI-Oakridge)
K.W. Nettles (SCE&G)
O.W. Dixon (SCE&G)
M.D. Quinton (SCE&G)
R. Ciatto (TES)
P. Harrison (TES)
J. King (TES)
K. Chang (West)
R. Kelly (West)

M

Gilbert/Commonwealth engineers and consultants

GILBERT ASSOCIATES, INC., P. O. Box 1498, Reading, PA 19603 Tel 215-775-2600 Cable Gilasoc Telex 836 431

July 15, 1982

CGGW - 1815

Mr. J. B. Cookinham
Westinghouse Electric Corporation
PWR System Division
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Re: V. C. Summer Nuclear Station Unit 1
GAI. W.O. 04-4461-000
Pressurizer Relief System
File Code: 40.G
Reference: CGGW-1811
Response Code: RR,

Dear Mr. Cookinham:

Attached to this letter are the forces and moments being applied to the outlet flanges of the three (3) safety relief valves 1-8010A, 1-8010B, and 1-8010C from the non-safety portion of the Pressurizer Relief System piping. These values supercede the previous values transmitted via CGGW-1811 on July 8, 1982.

Also attached to this letter are the forces and moments being applied to the Pressurizer Relief Tank Nozzle from the non-safety piping system. These values supercede the previous values transmitted via CGGW-1811 on July 8, 1982.

Please evaluate the loads applied to the safety valves and also the loads applied to the Pressurizer Relief Tank for acceptability and inform Gilbert Associates of the review results.

If you have any questions, please contact me.

DRK:CNR:GJB:cat
Attachments

Very truly yours,

D R Kershner
D. R. Kershner
Piping Engineer

C. N. Rentschler
C. N. Rentschler
As-Built Piping Verification
Task Manager

G. J. Braddick
G. J. Braddick
Project Manager

7-4

cc: J. B. Cookinham (4) w/att.
C. A. Price (2) w/att.
NPCF/Whitaker
V. C. Summer
E. H. Crews, Jr.
D. R. Moore
O. S. Bradham
O. W. Dixon
K. W. Nettles
M. Quinton
R. Faix w/att.
G. J. Braddick (2) w/att.
D. T. Klinksiek w/att.
C. N. Rentschler
M. Z. Lee w/att.
L. Y. Chou w/att.
K. R. Gabel w/att.

2

GILBERT ASSOCIATES, INC. READING, PA.	V. C. SUMMER NUCLEAR STATION	W.O. NUMBER 04-4461-264	FILING CODE (RC01)
PIPING ENGINEERING DEPT. NUMBER 0432	ORINGINATOR: [signature] DATE: 7/15/82	VERIFIER: [signature] 7/15/82	PAGE 001 OF 001

2.10 NOZZLE LOAD SUMMARY

EQUIPMENT : SAFETY RELIEF VALVE TAG NO. : 1-8010A
NOZZLE SIZE : 6" SERVICE : RC JOINT NO. : A23(MEMBER-1400)
REFERENCE : PRESSURIZER SRV OUTLET FLANGE
ORIENTATION : LOCAL (X - AXIAL, Y - VERTICAL)

LOAD CASE (RUN I.D.)	FORCES (LBS)			MOMENTS (FT-LBS)		
	FX	FY	FZ	MX	MY	MZ
DEADWEIGHT (ATGRIIP)	-1.	-567.	2.	-10.	13.	587.
THERMAL (ATGRIIP)	-78.	123.	11.	197.	290.	-1445.
OBE (ATGRIIP)	*3.	*3.	*1.	*1.	*14.	*2.
SSE (ATGRIIP)	*3.	*3.	*1.	*1.	*15.	*2.
BLOWDOWN (ATGRMLS)	*1411.	*7740.	*1045.	*1457.	*1411.	*8262.
RIGID RES. (ATGRIIP)	*69.	*401.	*311.	*163.	*298.	*343.
( )						

NOTE :
1. RUN ATGRIIP DATED 7/14/82
2. RUN ATGRMLS DATED 7/12/82
3. * — POSITIVE AND NEGATIVE VALUES

(M)

GILBERT ASSOCIATES, INC. READING, PA.	V. C. SUMMER NUCLEAR STATION	W.O. NUMBER 04-4461-264	FILING CODE (RC01)
PIPING ENGINEERING DEPT. NUMBER 0432	ORIGINATOR: [signature] DATE: 7/15/82	VERIFIER: D R Kirchner 7/15/82	PAGE 001 OF 001

2.10 NOZZLE LOAD SUMMARY

EQUIPMENT : SAFETY RELIEF VALVE   TAG NO. : 1-8010B
NOZZLE SIZE : 6"   SERVICE : RC   JOINT NO. : A1(MEMBER-1010)
REFERENCE : PRESSURIZER SRV OUTLET FLANGE
ORIENTATION : LOCAL (X - AXIAL, Y - VERTICAL)

LOAD CASE (RUN I.D.)	FORCES (LBS)			MOMENTS (FT-LBS)		
	FX	FY	FZ	MX	MY	MZ
DEADWEIGHT (ATGRIIP)	4.	-503.	4.	-26.	-6.	-468.
THERMAL (ATGRIIP)	-82.	349.	-105.	1718.	383.	916.
OBE (ATGRIIP)	*4.	*2.	*7.	*12.	*6.	*9.
SSE (ATGRIIP)	*6.	*2.	*9.	*16.	*7.	*11.
BLOWDOWN (ATGRMLS)	*1175.	*6380.	*1823.	*1493.	*2157.	*6700.
RIGID RLS. (ATGRIIP)	*244.	*386.	*259.	*155.	*217.	*193.
( )						

NOTE :
1. RUN ATGRIIP DATED 7/14/82
2. RUN ATGRMLS DATED 7/12/82
3. * — POSITIVE AND NEGATIVE VALUES

GILBERT ASSOCIATES, INC. READING, PA.	V. C. SUMMER NUCLEAR STATION	W.O. NUMBER 04-4461-201	FILING CODE (RC01)
PIPING ENGINEERING DEPT. NUMBER 0432	ORIGINATOR: DATE: 7/15/82	VERIFIER: 7/15/82	PAGE 001 OF 001

2.10 NOZZLE LOAD SUMMARY

EQUIPMENT : SAFETY RELIEF VALVE TAG NO. : 1-8010C
NOZZLE SIZE : 6" SERVICE : RC JOINT NO. : A37(MEMBER-1410)
REFERENCE : PRESSURIZER SRV OUTLET FLANGE
ORIENTATION : LOCAL (X - AXIAL, Y - VERTICAL)

LOAD CASE (RUN I.D.)	FORCES (LBS)			MOMENTS (FT-LBS)		
	FX	FY	FZ	MX	MY	MZ
DEADWEIGHT (ATGRIIP)	11.	-626.	-3.	4.	-4.	-538.
THERMAL (ATGRIIP)	-96.	618.	-43.	1450.	11.	-808.
OBE (ATGRIIP)	*3.	*13.	*1.	*2.	*11.	*6.
SSE (ATGRIIP)	*4.	*17.	*1.	*3.	*11.	*8.
BLOWDOWN (ATGRMLS)	*1800.	*13813.	*1963.	*1501.	*2035.	*14317.
RIGID RES. (ATGRIIP)	*413.	*441.	*126.	*100.	*90.	*29.
( )						

NOTE :
1. RUN ATGRIIP DATED 7/14/82
2. RUN ATGRMLS DATED 7/12/82
3. * — POSITIVE AND NEGATIVE VALUES

GILBERT ASSOCIATES, INC. READING, PA.	V. C. SUMMER NUCLEAR STATION	W.O. NUMBER 04-4461-264	FILING CODE (RC01)
PIPING ENGINEERING DEPT. NUMBER 0432	ORINGINATOR: [signature] DATE: 7/15/82	VERIFIER: [signature] 7/15/82	PAGE 001 OF 001

2.10 NOZZLE LOAD SUMMARY

EQUIPMENT : PRES. RELIEF TANK TAG NO. : XTK-5-RC
NOZZLE SIZE : 12" SERVICE : RC JOINT NO. : A55(MEMBER-1980)
REFERENCE : PRESSURIZER RELIEF TANK FLANGE
ORIENTATION : LOCAL (X - AXIAL; Y,Z - SHEAR)

LOAD CASE (RUN I.D.)	FORCES (LBS)			MOMENTS (FT-LBS)		
	FX	FY	FZ	MX	MY	MZ
DEADWEIGHT (ATGRIIP)	218.	8.	-134.	-31.	122.	167.
THERMAL (ATGRMAL)	*2049.	-57.	-5557.	601.	-7723.	3051.
OBE (ATGRIIP)	*806.	*890.	*429.	*1001.	*233.	*3856.
SSE (ATGRIIP)	*983.	*1054.	*501.	*1163.	*277.	*4478.
BLOWDOWN (ATGRMLS)	*23319.	*3006.	*1145.	*7400.	*2165.	*12855.
RIGID RES. (ATGRIIP)	*363.	*314.	*37.	*549.	*136.	*1223.
( )						

NOTE :
1. RUN ATGRIIP DATED 7/14/82
2. RUN ATGRMLS DATED 7/12/82
3. * — POSITIVE AND NEGATIVE VALUES

2

RECEIVED

AUG 13 1982

G. J. BRADDICK

I	A	NAME
		BRADDICK
		LISNEY
		BITTLE
		PAOLINI
		KLINKSIEK
	X	GABEL
		KRAMER
		SETLOCK
		MABON
		REITNAUER
		HARTMAN
		BARNISIN
		NRR

CGWG-2628 ✓

F.S. II-1.21

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

August 10, 1982

AM-SSA-2308
S.O. CGE/145

Mr. G. J. Braddick
Gilbert Associates, Inc.
P.O. Box 14-8
Reading, PA 19603

RSL 8/13/82

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
Pressurizer Safety and Relief Valve System

Dear Mr. Braddick:

Westinghouse has evaluated the pressurizer safety valve outlet flanges and the pressurizer relief tank flange based on the revised calculated loads as transmitted in Gilbert Associates, Inc. letter CGSW-1815, 7/15/82.

Our Systems Structural Analysis Group has determined that the pressurizer relief tank flange loads are acceptable. The applied loads on the three safety valves have been shown to be acceptable by our Pump and Valve Engineering Group.

If there are any questions, please contact me.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

James B. Cookinham

James B. Cookinham, Manager
SCE&G Project

L.C.Smith/jm

cc: G. J. Braddick (GAI), 4L
C. A. Price (SCE&G), 1L
M. B. Whitaker, Jr./NPCF (SCE&G), 1L
E. H. Crews, Jr. (SCE&G), 1L
D. A. Nauman (SCE&G), 1L
O. S. Bradham (SCE&G Site), 1L
H. Radin (SCE&G Site), 1L
R. J. Hoffert (GAI), 1L
Plant Numerical Records System (SCE&G Site), 1L

(Previously Received by Telecopy)

(M)

RC01 2.13.10 1 2

F.C. II-1.21

Gilbert/Commonwealth engineers and consultants

GILBERT ASSOCIATES, INC., P. O. Box 1498, Reading, PA 19603/Tel. 215-775-2600/Cable Gilasoc/Telex 836-431

July 30, 1982

CGGW - 1824

Mr. J. P. Cookinham
Westinghouse Electric Corporation
PWR System Division
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Re: V. C. Summer Nuclear Station Unit 1
GAI W.O. - 04-4461-000
Pressurizer Relief System
File Code: 40.G
Reference: CGGW - 1815
Response Code: RR, 8-10-82

Dear Mr. Cookinham:

Attached to this letter is a revised outlet flange load summary sheet for the Pressurizer Safety Valve 1-8010A. The only changes reflected on the attached load summary sheet as compared to the loadings sent via the letter CGGW-1815 are that the (Fy) and (Mz) values were increased by 20%. These increases were required based upon the completion of the RELAP5 forcing function verification.

These increased blowdown loads for loop 'A' will still be enveloped by the loop 'C' blowdown loads which were used for the previous qualifications of the SRV outlet flanges and the SRV outlet flange studs (Ref. CGGW-1815).

Please evaluate the loads applied to the SRV 1-8010A for acceptability and inform GAI of the review results.

If you have any questions, please contact me.

DRK:CNR:GJB:cat
Attachment

Very truly yours,

D.R. Kershner

D. R. Kershner
Piping Engineer

C. N. Rentschler

C. N. Rentschler
As-Built Piping Verification
Task Manager

J.R Helwig for

G. J. Braddick
Project Manager

cc: J. B. Cookinham (4) w/att.
C. A. Price (2) w/att.
NPCF/Whitaker
V. C. Summer
E. H. Crews, Jr.
D. R. Moore
O. S. Bradham
O. W. Dixon
K. W. Nettles
M. Quinton
R. Faix w/att.
G. J. Braddick (2) w/att.
D. T. Klinksiek w/att.
C. N. Rentschler
M. Z. Lee w/att.
L. Chou w
K. R. Gabel w
F. G. Boutros w
D. R. Kershner w

7-11

GILBERT ASSOCIATES, INC. READING, PA.	V. C. SUMMER NUCLEAR STATION	W.O. NUMBER 04-4461-204	FILING CODE (RC01)
PIPING ENGINEERING DEPT. NUMBER 0432	ORIGINATOR: [signature] DATE: 7/12/82	VERIFIER: D.R. [signature] 7/28/82	PAGE 001 OF 001

2.10 NOZZLE LOAD SUMMARY

EQUIPMENT : SAFETY RELIEF VALVE   TAG NO. : 1-8010A
NOZZLE SIZE : 6"   SERVICE : RC   JOINT NO. : A23(MEMBER-1400)
REFERENCE : PRESSURIZER SRV OUTLET FLANGE
ORIENTATION : LOCAL (X - AXIAL, Y - VERTICAL)

LOAD CASE (RUN I.D.)	FORCES (LBS)			MOMENTS (FT-LBS)		
	FX	FY	FZ	MX	MY	MZ
DEADWEIGHT (ATGRIIP)	-1.	-567.	2.	-10.	13.	587.
THERMAL (ATGRIIP)	-78.	123.	11.	197.	290.	-1445.
OBE (ATGRIIP)	*3.	*3.	*1.	*1.	*14.	*2.
SSE (ATGRIIP)	*3.	*3.	*1.	*1.	*18.	*2.
BLOWDOWN (ATGRMLS)	*1411.	*9288.	*1045.	*1457.	*1411.	*9914.
RIGID RES. (ATGRIIP)	*69.	*401.	*311.	*103.	*298.	*343.
( )						

NOTE :
1. RUN ATGRIIP DATED 7/14/82
2. RUN ATGRMLS DATED 7/12/82
3. * — POSITIVE AND NEGATIVE VALUES
4. FY AND MZ OF BLOWDOWN WERE MULTIPLIED 1.2 CORRECTION FACTOR BASED COMPUTER OUTPUT DATA. (REF. AEA-MEMO)

V.C.S FILE ---- RC01 ... 13.15 PAGE 1 OF 1

CGWG-2639

F.G II - 1.21

RECEIVED

AUG 1 8 1982

G. J. BRADDICK

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Commercial Operations Division

Box 355
Pittsburgh Pennsylvania 15230

August 13, 1982

Mr. G. J. Braddick
Gilbert Associates, Inc.
P. O. Box 1498
525 Lancaster Avenue
Reading, PA 19603

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
Pressurizer Safety Valve Nozzle Loads

Dear Mr. Braddick:

Westinghouse has evaluated the safety valve outlet flange loads forwarded by CGGW-1824 dated July 30, 1982 for safety relief valve #1-8010A.

The loads on valve #1-8010A are acceptable.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

R. J. Faix, Project Engineer
South Carolina Electric & Gas Co.

RJF:jt

cc: G. J. Braddick (GAI) 4L
C. A. Price (SCE&G) 1L
H. B. [illegible] Stoker, Jr./NPCF (SCE&G) 1L
[illegible] H. C[illegible]ws, Jr. (SCE&G) 1L
D. A. N[illegible]man (SCE&G) 1L
O. S. [illegible]dham (SCE&G Site) 1L
H. Raw[illegible] (SCE&G Site) 1L
R. J. [illegible]fert (GAI) 1L
Plant [illegible]erical Records System (SCE&G Site) 1L

FILE: 58W		
I	A	NAME
		BRADDICK
		LISNEY
		BITTLE
		PAOLINI
		KLINKSIEK
	X	CABEL
		KRAMER
		SETLOCK
		MARON
		REINHARDT
		HART[illegible]
		BARR[illegible]
Action: [illegible]		
By: [illegible]	Date: [illegible]	

7-13

M

PORV VALVE QUALIFICATION
SUB-SYSTEM RC-01 SUMMARY

CONDITION	VALVE	MB (IN-LB)		MT (IN-LB)		MAX (PSI)		MAX (PSI)	
		Actual	Allow	Actual	Allow	Actual	Allow	Actual	Allow
DESIGN	1-PCV-445A	15573	37139	510	37139	9907	19350	462	11094
	1-PCV-445B	9720	37139	5833	37139	9287	19350	1303	11094
	1-PCV-444B	9824	37139	8535	37139	9644	19350	1881	11094
NORMAL/UPSET-2	1-PCV-445A	33323	37139	14836	37139	18079	19350	3716	11094
	1-PCV-445B	15196	37139	8111	37139	11996	19350	1842	11094
	1-PCV-444B	13607	37139	9928	37139	11693	19350	2191	11094
NORMAL/UPSET-3	1-PCV-445A	14833	26228	2792	26228	10413	13665	865	7835
	1-PCV-445B	12756	26228	3483	26228	9908	13665	895	7835
	1-PCV-444B	8041	26228	6132	26228	9269	13665	1472	7835
FAULTED	1-PCV-445A	-	-	-	-	17037	25800	3969	14706
	1-PCV-445B	-	-	-	-	13211	25800	2485	14706
	1-PCV-444B	-	-	-	-	11887	25800	2117	14706

COMPARISON OF PRESSURIZER SRV OUTLET FLANGE LOADS

Valve	Load Case	Bending M_y	Moment[1],[2] M_z	GAI[1] Design Moment	EPRI[3] Measured Moment
8010A	Thermal	13	587		
	Deadweight	290	-1445		
	Norm. Design	303	-1445	1477	24,895/2074[1]
8010B	Thermal	-6	-468		
	Deadweight	383	916		
	Norm. Design	383	916	993	24,895/2074[1]
8010C	Thermal	-4	-583		
	Deadweight	11	-808		
	Norm. Design	11	-1346	1346	24,895/2074[1]

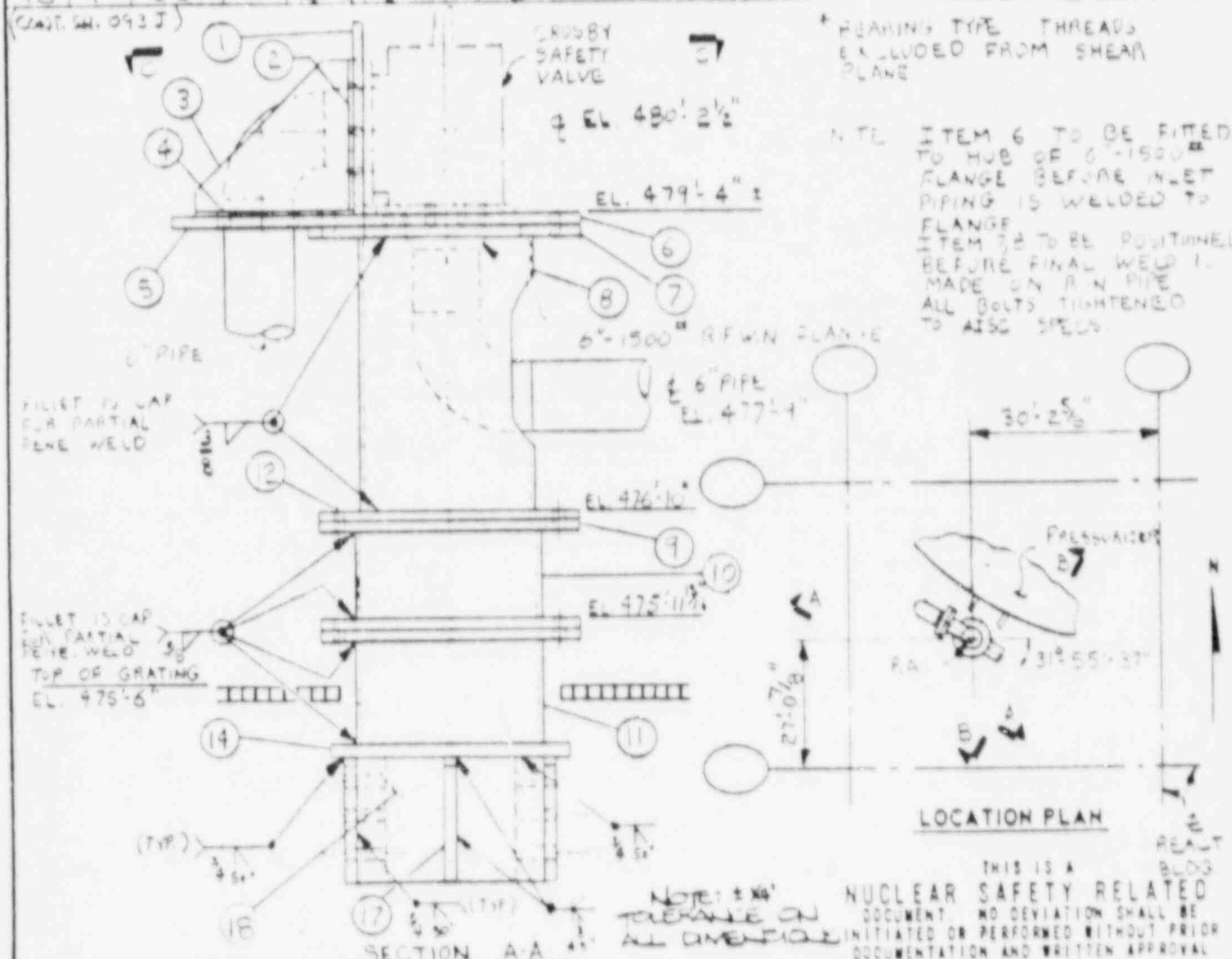
1) Moments are ft.-lbs.

2) The moments shown were transmitted to Westinghouse for evaluation and approval on GAI letters CGGW-1815 (pages 7-4 through 7-9) and CGGW-1824 (pages 7-11 and 7-12). These loadings were deemed acceptable on Westinghouse letters CGWG-2628 (page 10) and CGWG-2639 (page 13).

3) Moment shown is the lowest moment measured as given in Table 3-1 of the EPRI/CE Safety Valve Test Report for the Crosby HB-BP-86 6M6 valve (series 900 and 1400 tests) with inlet loop seal conditions similar to V. C. Summer.

ITEM NO.	NO. REQD.	PART NO.	LCD REV.	SIZE / DESCRIPTION	MATERIAL SPEC.	REF. NOTES
1	1	CS	-	24"x1" PL x 2'-0" LG. (SEE DETAIL SHT 093G)	SA36	2,5
2	2	CS	-	4 3/4"x 1/2" PL x 1'-5 3/4" LG. (SEE DETAIL SHT. 093F)	SA36	5
3	2	CS	-	17 3/4" x 1/2" PL x 1'-5 3/4" LG. (SEE DETAIL SHT. 093G)	SA36	5
4	2	CS	-	4 1/4" x 1/2" PL x 1'-6 1/4" LG (SEE DETAIL SHT 093F)	SA36	5
5	1	CS	-	8"x1" PL x 2'-0" LG W/(2) 1 1/8" ∅ HOLES (SEE SHT. 093D)	SA36	5
6	1	CS	-	24"x1" PL x 3'-2" LG. (SEE DETAIL SHT. 093E)	SA36	5
7	1	CS	-	24"x1" PL x 2'-8" LG. (SEE DETAIL SHT. 093E)	SA36	5
8	1	CS	-	18" ∅ PIPE x 2'-4" LG, SCH XS (SEE DETAIL SHT 093D)	SA106 GR B	5
9	4	CS	-	21"x1" PL x 2'-4" LG. (SEE DETAIL SHT. 093F)	SA36	5
10	1	CS	-	20" ∅ PIPE x 1'-0" LG. SCH XS	SA106 GR B	2,5
11	1	CS	-	20" ∅ PIPE x 1'-0" LG. SCH XS	SA106 GR B	5
12	26	-	-	1" ∅ x 4" LG BOLT W/AHH NUT	SA325 TYPE 1	5
13	10	-	-	1" ∅ 23 1/2" LG. BOLT W/AHH NUT	SA325 TYPE 1	5
14	1	CS	-	22"x 1/2" PL x 2'-2" LG.	SA36	5
15	2	CS	-	15"x 1/2" PL x 1'-10" LG. (SEE DETAIL SHT. 093H)	SA36	5
~~16~~	~~6~~		-	~~1 1/2" ∅ x 7" LG. BOLT W/AHH NUT*~~	~~SA193 GR B7~~	~~5~~
17	1	CS	-	5"x 1/2" PL x 1'-3" LG. (SEE DETAIL SHT 093H)	SA36	5
18	1	CS	-	5"x 1/2" PL x 1'-8" LG	SA36	5

(CONT. SH. 093J)



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SOUTH CAROLINA ELECTRIC & GAS COMPANY

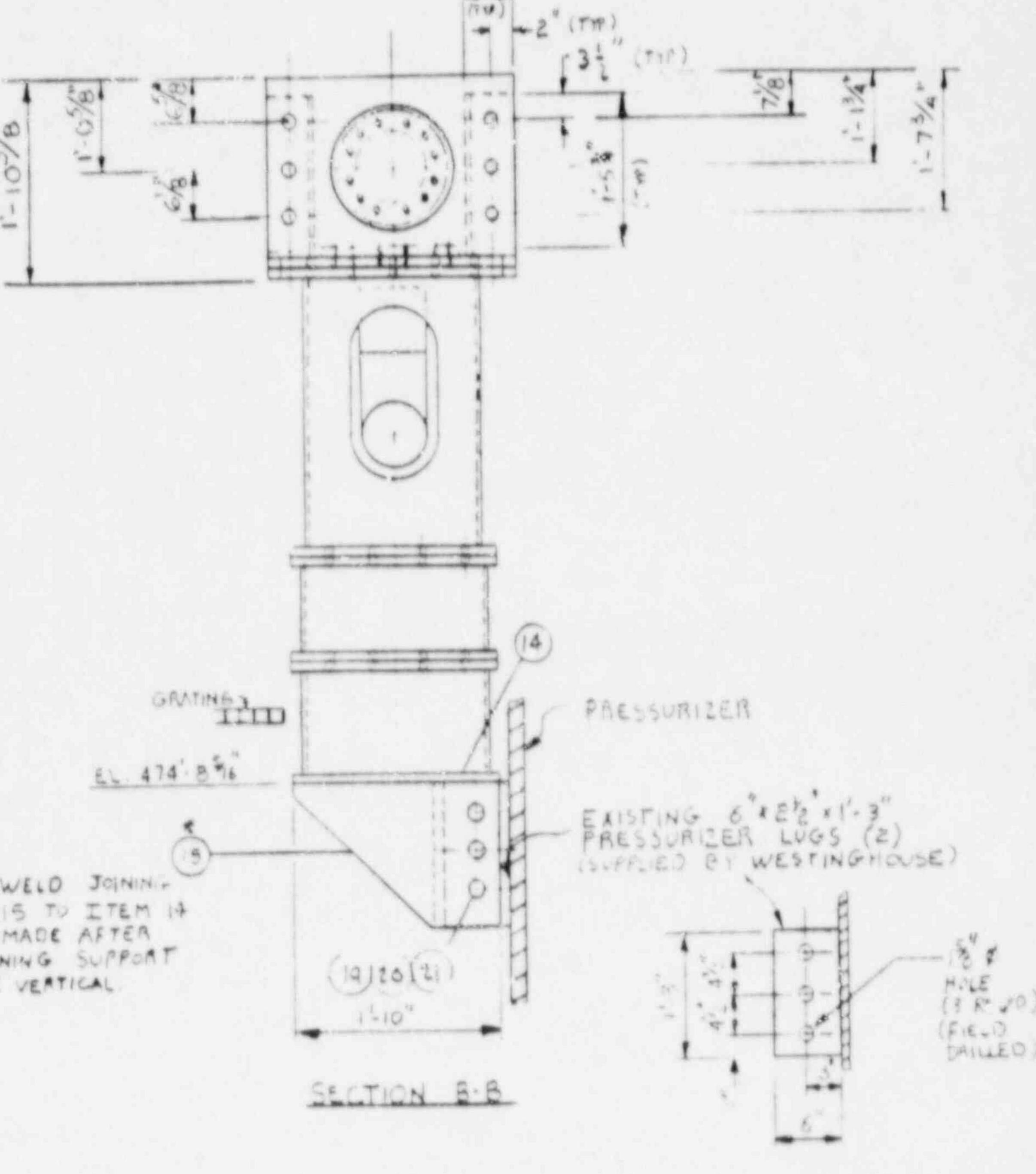
VIRGIL C. SUMMER NUCLEAR STATION UNIT #1

GILBERT ASSOCIATES, INC.
ENGINEERS & CONSULTANTS
READING, PA.

FOR SIGNATURES AND REVISIONS SEE SHEET LETTER A FOR THE PIPE SUPPORT INDICATED BELOW.

DRAWING NUMBER			SHEET	PIPE SUPPORT (DESCRIPTION)	
04	4461	S-321-601	093 B	MK-RCH-093	9

7-16



* FINAL WELD JOINING ITEM 15 TO ITEM 14 TO BE MADE AFTER POSITIONING SUPPORT TO TRUE VERTICAL.

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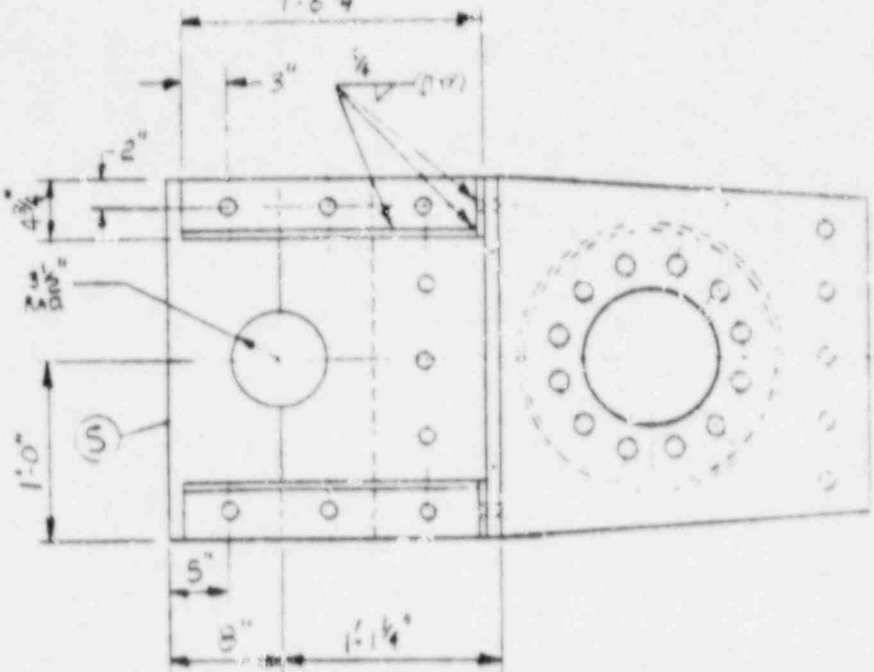
SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION UNIT #1

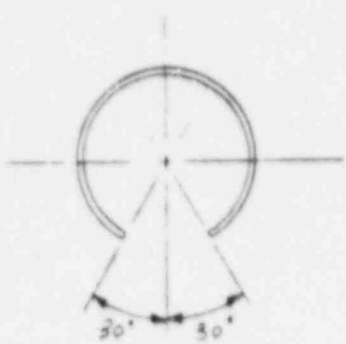
GILBERT ASSOCIATES, INC.
ENGINEERS & CONSULTANTS
READING, PA.

FOR SIGNATURES AND REVISIONS SEE SHEET LETTER A FOR THE PIPE SUPPORT INDICATED BELOW.

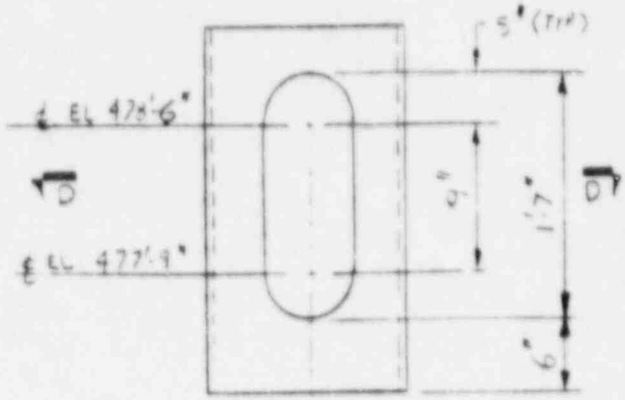
DRAWING NUMBER			SHEET	PIPE SUPPORT	REV.
04	4461	S-321-601	093C	MK-RCH-093	9
			DESCRIPTION		



SECTION C-C



SECTION D-D



ITEM B

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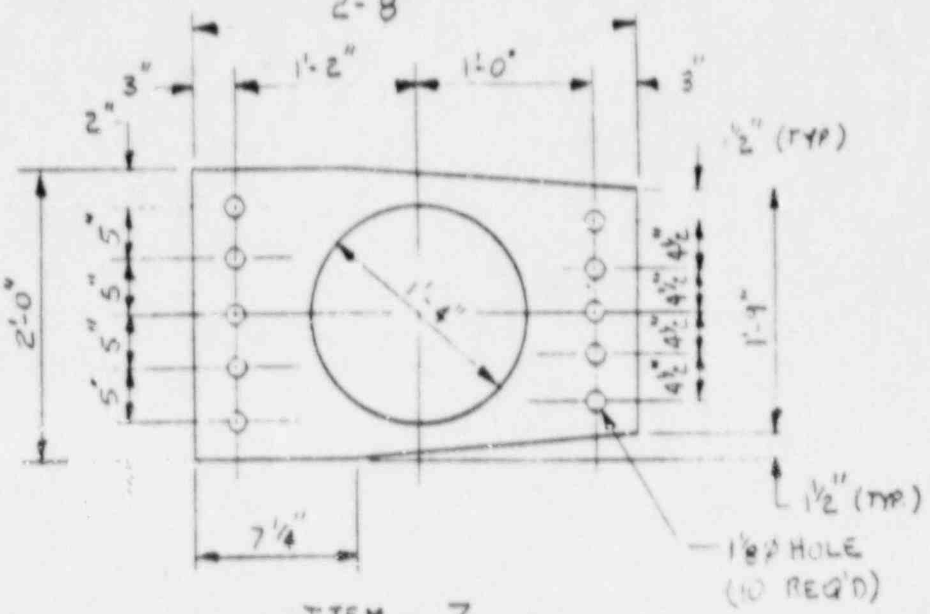
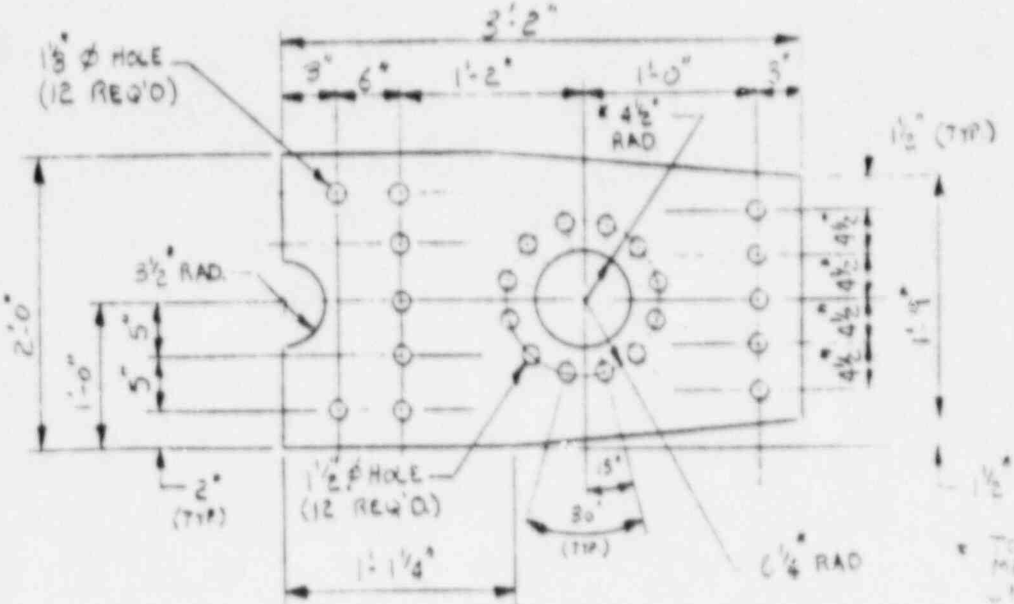
7-18

SOUTH CAROLINA ELECTRIC & GAS COMPANY VIRGIL C. SUMMER NUCLEAR STATION UNIT #1	FOR SIGNATURES AND REVISIONS SEE SHEET LETTER A FOR THE PIPE SUPPORT INDICATED BELOW.
GILBERT ASSOCIATES, INC. ENGINEERS & CONSULTANTS READING, PA.	04 4461 S-321-601 (DRAWING NUMBER) — SHEET 073 D — PIPE SUPPORT MK-RC 3 (DESCRIPT) — REV. 9



ITEM 7



ITEM 6

* TOP PORTION OF HOLE MAY HAVE TO BE CHAMFERED TO FIT OVER FLANGE HUB.

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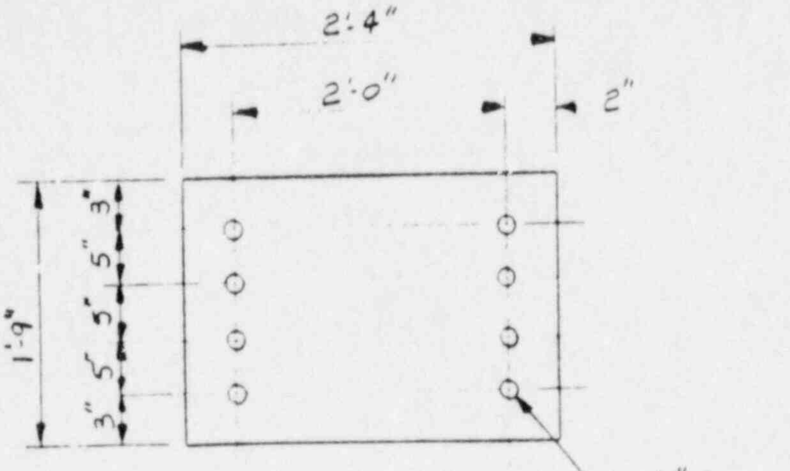
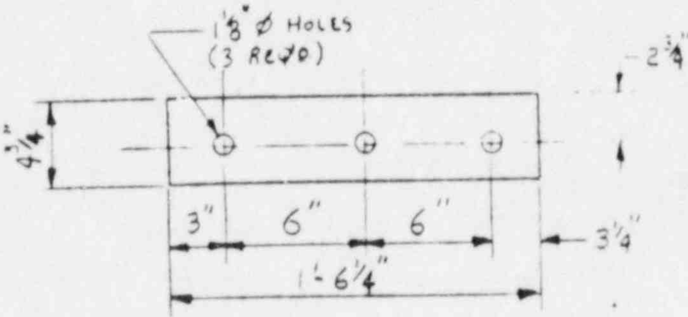
7-19

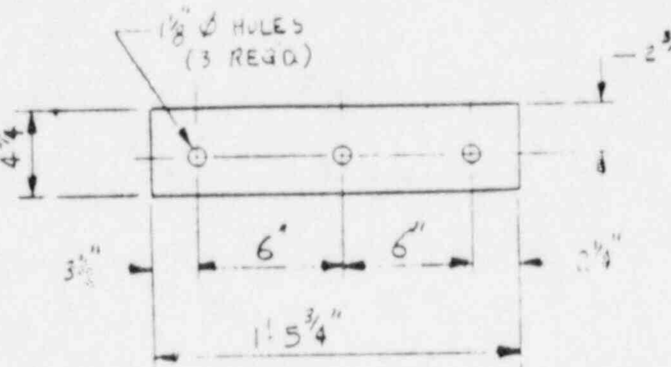
SOUTH CAROLINA ELECTRIC & GAS COMPANY VIRGIL C. SUMMER NUCLEAR STATION UNIT #1	FOR SIGNATURES AND REVISIONS SEE SHEET LETTER A FOR THE PIPE SUPPORT INDICATED BELOW.					
GILBERT ASSOCIATES, INC. ENGINEERS & CONSULTANTS READING, PA.	04	4461	S-321-601	SHEET 093 E	PIPE SUPPORT MK-RCH-093	9
	DRAWING NUMBER			DESCRIPTION		REV.



ITEM 9



ITEM 4



ITEM 2

7-20

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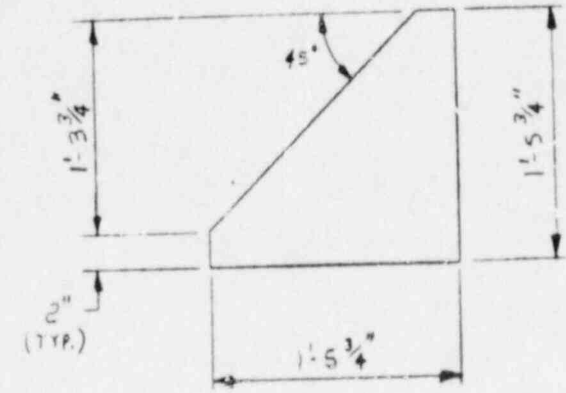
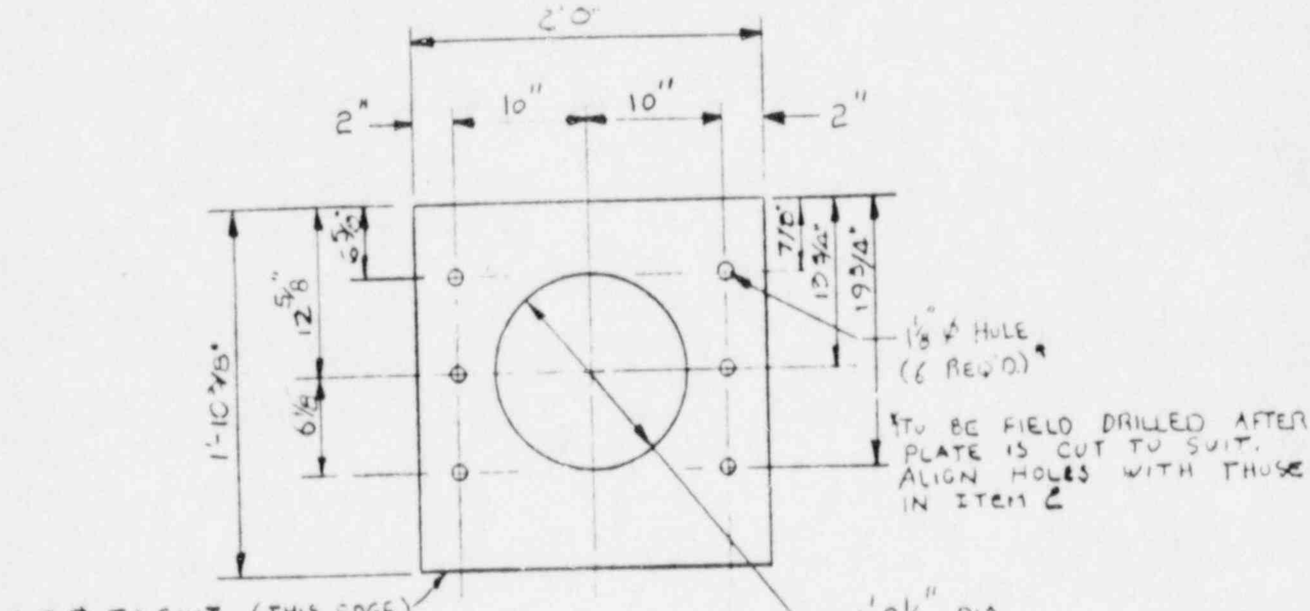
SOUTH CAROLINA ELECTRIC & GAS COMPANY VIRGIL C. SUMMER NUCLEAR STATION UNIT #1	FOR SIGNATURES AND REVISIONS SEE SHEET LETTER A FOR THE PIPE SUPPORT INDICATED BELOW.

GILBERT ASSOCIATES, INC. ENGINEERS & CONSULTANTS READING, PA.	DRAWING NUMBER			DESCRIPTION: SHEET	DESCRIPTION: PIPE SUPPORT	REV
	04	4461	S-321-601	093 F	MK-RCH-093	9



ITEM 3



ITEM 1

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7-21

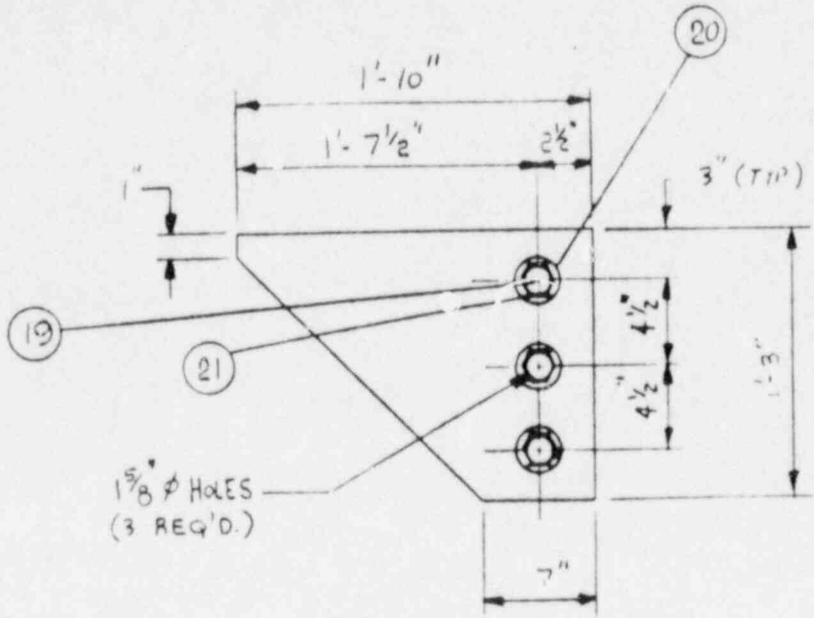
SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION UNIT #1

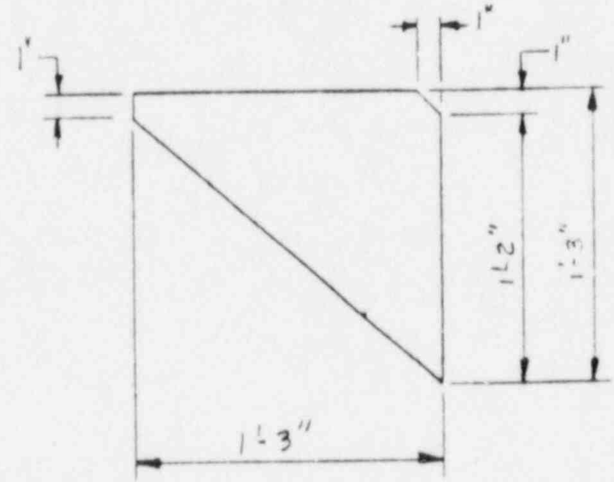
GILBERT ASSOCIATES, INC.
ENGINEERS & CONSULTANTS
READING, PA.

FOR SIGNATURES AND REVISIONS SEE SHEET LETTER A FOR THE PIPE SUPPORT INDICATED BELOW.

DRAWING NUMBER			SHEET	PIPE SUPPORT	REV
04	4461	S-321-601	093 G	MK-RCH-093	9



ITEM 15



ITEM 17

7-22

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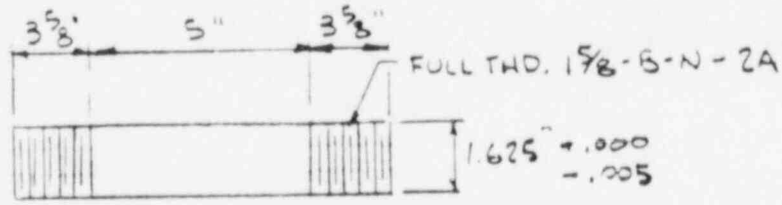
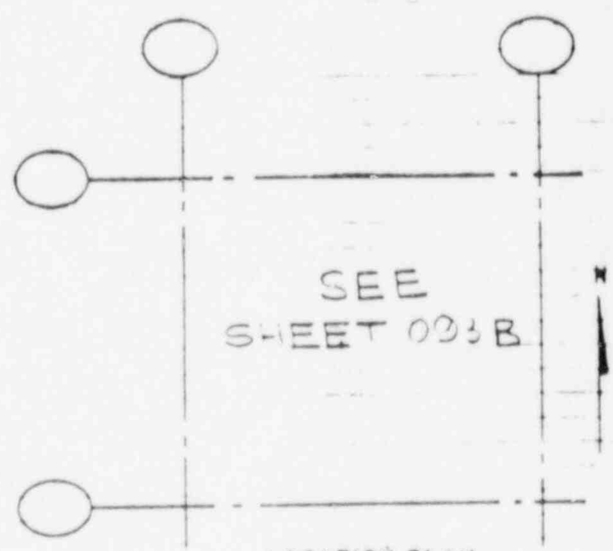
SOUTH CAROLINA ELECTRIC & GAS COMPANY VIRGIL C. SUMMER NUCLEAR STATION UNIT #1	FOR SIGNATURES AND REVISIONS SEE SHEET LETTER A FOR THE PIPE SUPPORT INDICATED BELOW.					
				SHEET	PIPE SUPPORT	
GILBERT ASSOCIATES, INC. ENGINEERS & CONSULTANTS READ[illegible] PA.	04	4461	S-321-601	093H	MK-RCH-093	9
	DRAWING NUMBER			DESCRIPTION		REV

ITEM NO.	NO. REQD.	PART NO.	LCD REV.	SIZE / DESCRIPTION	MATERIAL SPEC.	REF. NOTES
19	6			15/8" Ø STUD (SEE DETAIL SH J)	SA 193 GR B7	1
20	12			HARDENED STEEL WASHERS FOR		
				15/8" STUDS		1
21	12			STANDARD NUTS 15/8"-8 N 2B	SA-194 GR 2H	1



DETAIL ITEM 19



LOCATION PLAN

NOTE: INSTALLATION INSTRUCTIONS ARE SUPPLIED WITH FCR-15668 REV. 3

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7-23

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION UNIT #1

GILBERT ASSOCIATES, INC.
ENGINEERS & CONSULTANTS
READING, PA.

FOR SIGNATURES AND REVISIONS SEE SHEET LETTER A FOR THE PIPE SUPPORT INDICATED BELOW.

DRAWING NUMBER			SHEET	PIPE SUPPORT	REV
04	4461	S-321-601	093J	MK-RCH-093	9

EG&G QUESTION NO. 8

As part of comparing inlet piping configurations of the plant safety valves and the test valves, a comparison between the two inlet piping pressure drops should be made. Provide a numerical comparison between a calculated plant pressure drop and the test pressure drop. Explain how the plant pressure drop was calculated.

RESPONSE:

Table B-3 of "EPRI PWR SAFETY AND RELIEF VALVE TEST PROGRAM, GUIDE FOR APPLICATION OF VALVE TEST PROGRAM RESULTS TO PLANT-SPECIFIC EVALUATIONS" lists the inlet piping pressure drops for the different valves/piping configurations tested. Per Table B-3, the Crosby 6M6 test inlet piping pressure drop for steam discharge is 263 psi. This value can be compared to the 50 psi inlet piping pressure drop calculated by a plant-specific RELAP5/MOD1 analysis of the V. C. Summer plant for steam flow conditions.

The guidelines of the EPRI plant-specific evaluations report referenced above indicate that: "if the plant pressure difference is less than the test pressure difference, the in-plant valve would be expected to have performance at least as stable as the tested valve." Since the V. C. Summer SRV's valve/piping configuration pressure drop during steam flow discharge is less than 263 psi, the plant safety valves are expected to perform as stable, or better, as the Crosby 6M6 valves tested.

The plant specific analyses performed were not designed to simulate in detail the pressure wave reflections and interactions upstream of the SRV following valve opening. The inlet piping pressure drop criteria discussed above is intended, in part, to evaluate the susceptibility of plant-specific SRV valve/piping configurations to these pressure oscillations. Since the V. C. Summer SRV valve piping configuration meets the inlet piping pressure drop criteria, the present design is deemed acceptable from this standpoint.

EG&G Question NO. 9

The submittal states that backpressures at the safety valves were analyzed for steady state steam dischage from all three safety valves and were shown to be less than 500 psig. It does not, however, identify the expected backpressure for loop seal discharge from the safety valves. Provide this value for expected backpressure and assure that it was enveloped in the EPRI hot loop seal discharge tests.

Response:

The RELAP 5 analysis was conservatively evaluated assuming all three SRV's are activated simultaneously. In addition the flow rates used to develop the flow area for the SRV's were increased 17% over design flow (see response to question no. 13). The resulting transient valve backpressure is given in Figures 7.1 through 7.3 (pages 9-2 through 9-4). In all cases the peak transient backpressure is below 600 psig, and the steady state backpressure is less than 450 psig.

The results of the EPRI/CE test #917 indicate a peak valve backpressure of 600 psig. Therefore, the V. C. Summer SRV transient backpressure is enveloped by the EPRI hot loop seal discharge.

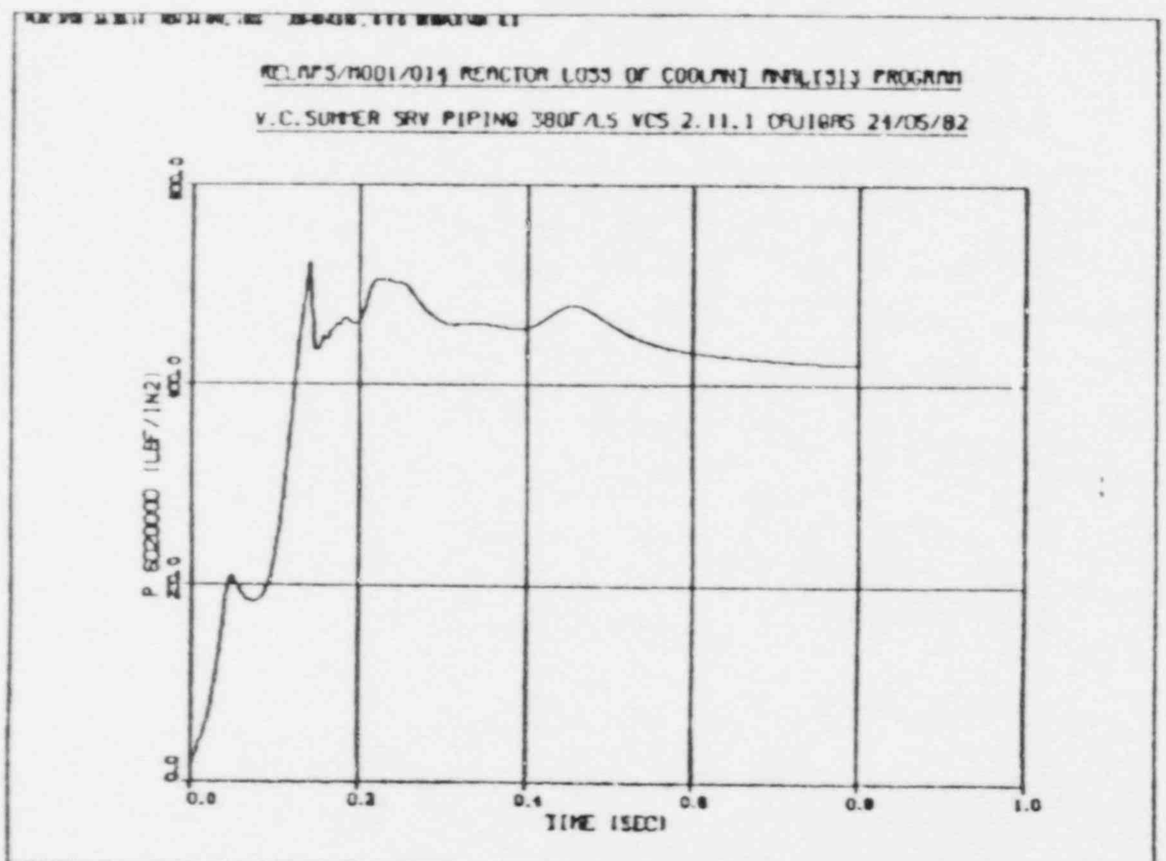

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIBAS 24/05/82
P 6020000 (LBF/IN2)
TIME (SEC)
0.0
0.2
0.4
0.6
0.8
1.0

FIGURE 7.1

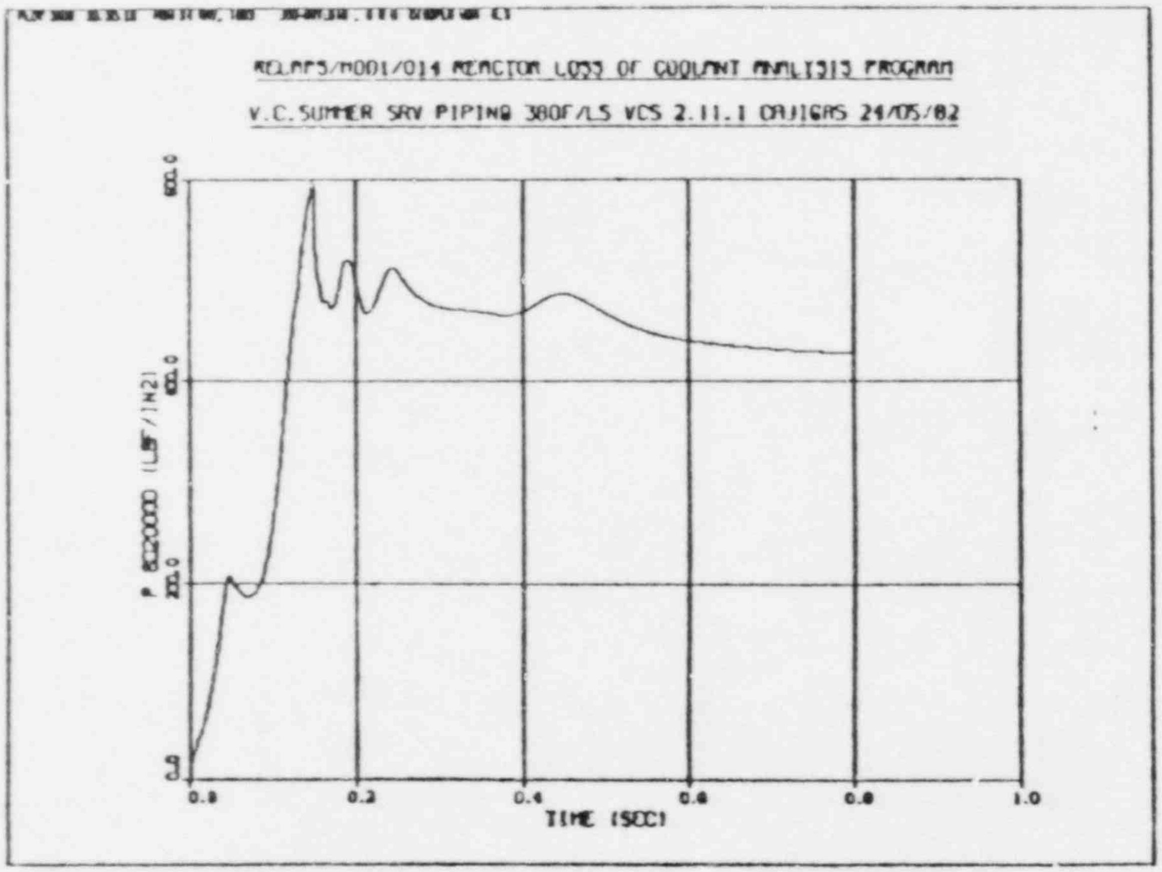

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRJIGAS 24/05/82
P 802000 (LBF/IN2)
TIME (SEC)
0.2
0.4
1.0

FIGURE 7.2

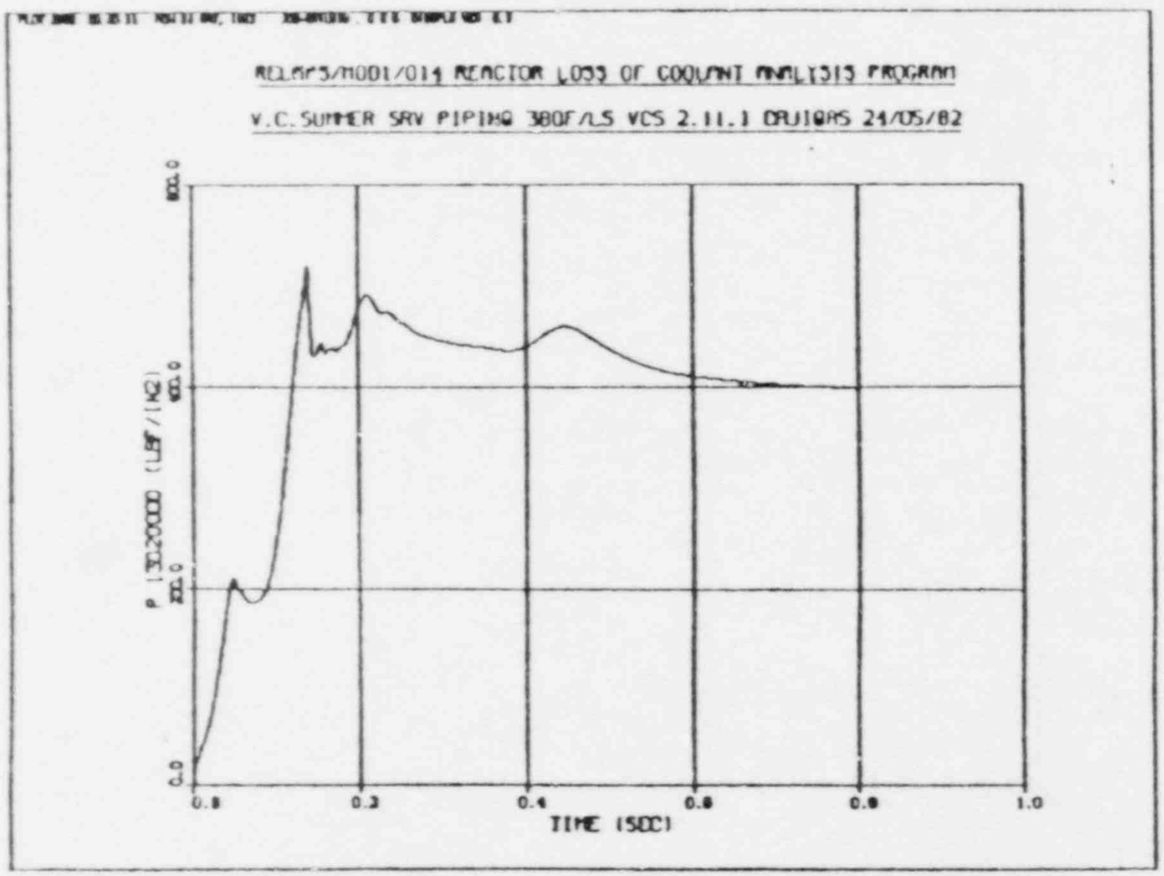

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIQRS 24/05/82
P 1302000 (LBF/IN2)
TIME (SEC)
0.0
200.0
400.0
600.0
0.2
0.4
0.6
0.8
1.0

FIGURE 7.3

EG&G Question No. 10

The submittal does not present details of the thermal hydraulic analysis. Provide a report or other documentation that contains at least the following information: For the analysis involving discharge of saturated steam with a 380°F loop seal through the safety valves, identify parameters used such as peak pressure, pressurization rate, valve opening pop time, and time step. Provide rationale for the values used. Explain how many volumes were used in pipe segments of the thermal hydraulic model. Provide a copy of the computer printout from the RELAP 5 analysis of the loop seal/steam discharge through the safety valves.

RESPONSE:

The RELAP 5 analysis for the discharge of saturated steam with a 380°F loop seal was conservatively evaluated using the following parameters and conditions:

1. Pressurizer

   The highest valve inlet pressure for a pressurizer steam discharge corresponds to a Locked Rotor Transient (Ref. EPRI - 'Valve Inlet Fluid Conditions for Pressurizer Safety and Relief Valves in Westinghouse Designed Plants', March 1982)

   Pmax = 2592 psia
   Pressurizer pressure surge rate = 216 psi/sec.
   Valve opening pressure = 2499.7 psia.

2. Valve opening pop time:

   The SRV pop time was assumed to be 0.040 seconds based on the valve manufacturer's specifications. Based on the EPRI/CE tests for hot loop seal discharge the valve opening time in all cases exceeded .040 seconds.

The shorter the opening time the more conservative the analysis becomes. A shorter opening time allows for the loop-seal water slug to maintain more of its integrity and thus produce higher water-slug induced loads. In addition, the quicker opening time also results in higher initial fluid acceleration, and therefore, higher piping loads.

3. RELAP 5 Time Steps:

The maximum time steps were evaluated using the courant limit.

$$\Delta t = \frac{\Delta X}{V+C}$$

where: Δt = maximum time step
ΔX = minimum nodal length
V = maximum phasic velocity
C = speed of sound

In addition the minimum time step used was 1 x 10-7 seconds.

RELAP 5 Time Steps Information
Requested Time Step: 5.0 x 10-5 sec.
Minimum (allowed) Time Step: 1.0 x 10-7 sec.
Minimum (actual) Time Step Used: 1.95 x 10-7 sec.
Transient Duration: 0.8 sec.
Total Attempted Advancements: 16756
Total Repeated Advancements: 65
Total Successful Advancements: 16691
Total Requested Advancements: 16000

4. The Safety Relief Valve inlet and discharge lines were modeled using a 168 volume and 169 junction RELAP 5 model. Since all the valve setpoints are 2500 psia, the hydraulic forces were evaluated assuming all valves open simultaneously.

The general isometric for the SRV discharge piping is given in

The general isometric for the SRV discharge piping is given in Figure 10-7. The nodal model from the SRV's discharge to the relief tank are given in the attached figures (Calculation page Nos. 12/61 through 15/61, Attachment pages 10-3 through 10-6). In general nodal spacing is determined as follows:

> Near the valve outlet the node size is initially restricted by the geometry of the pipe segment and is typically less than 0.5 feet. As the piping network enters into the main header the nodal lengths are permitted to get larger, typically less than one foot. The main header nodal lengths gradually increase up to approximately four feet.

5. Forcing Function Results

The forcing function results are given in Attachment pages 10-8 through 10-33. A copy of the compute run is available for review at South Carolina Electric and Gas Engineering offices in Jenkinsville, South Carolina.

Gilbert Associates, Inc.
Reading, Pennsylvania

CALCULATION

SUBJECT SRV/PORV FORCING FUNCTIONS
SRV NETWORK - 300°F LOOP SEAL

CISID 2.11.1

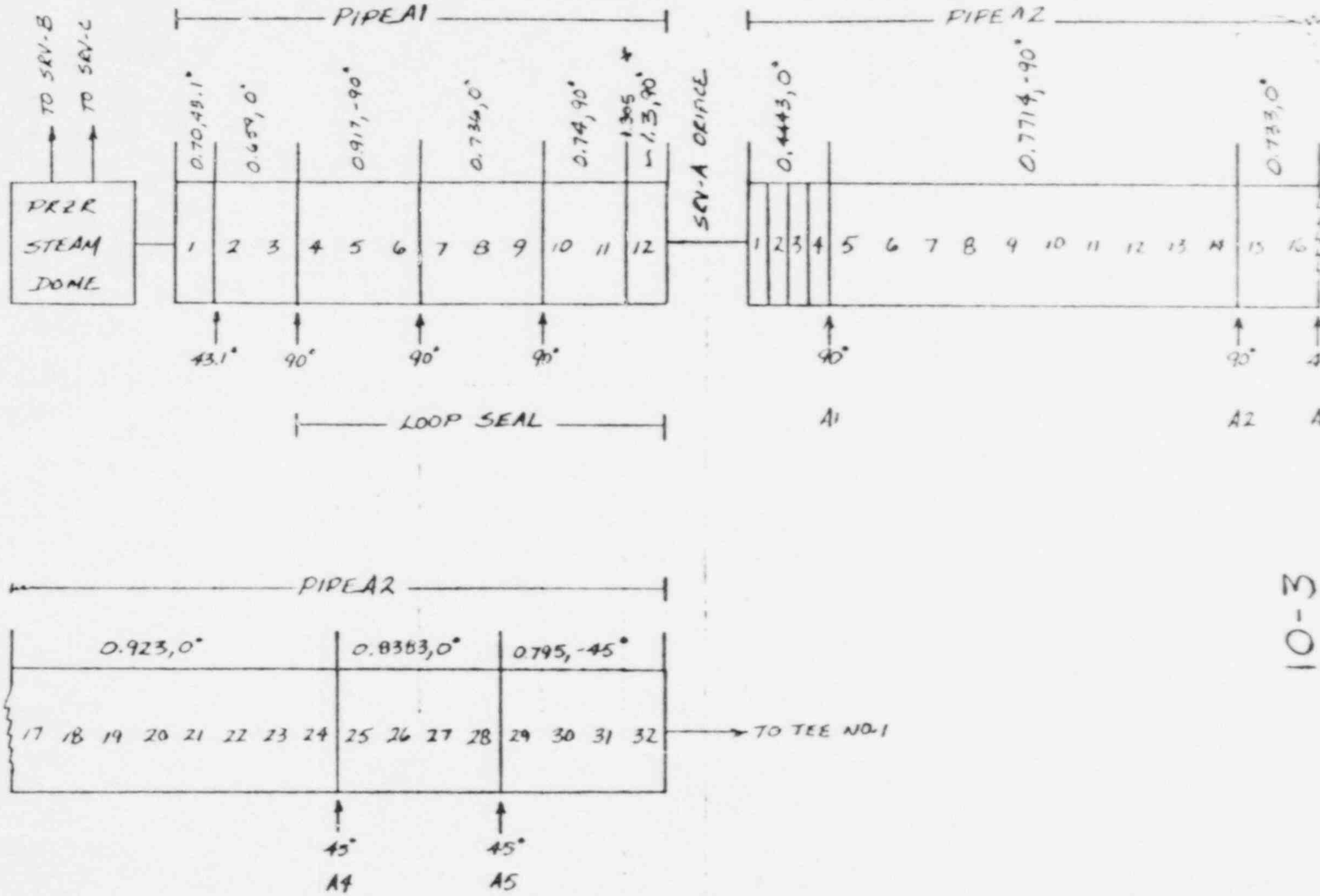
PAGE 12 OF 61 PAGES

REV. 0 1 2 3

MICROFILMED

ORIGINATOR J.M. CASTAGES

DATE 3-2-82



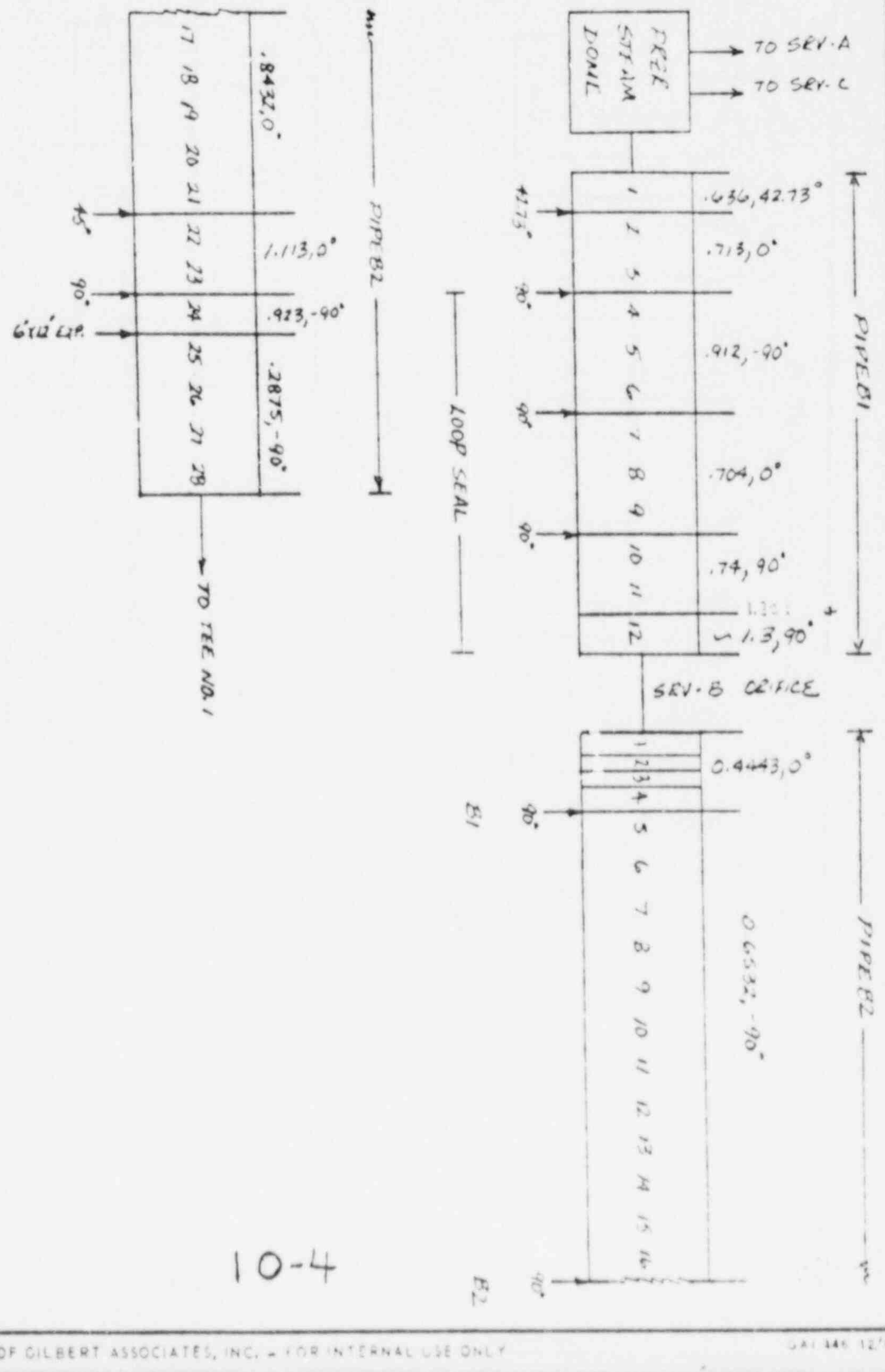
10-3

RELAP5 MODEL: PZR THROUGH SRV-A TO TEE NO. 1

* TO IMPROVE THE OPERATION OF THE RELAP5 ABRUPT AREA CHANGE MODEL USED AT THE VALVE, THE AREA OF THIS VOLUME WAS AVERAGED BETWEEN THE VALVE ORIFICE AREA (.0197) AND THE PIPE AREA (.1469), OR ≈ 0.0833 FT². THE LENGTH OF THE VOLUME IS THEN ADJUSTED TO MAINTAIN THE ORIGINAL VOLUME.

GAI 446 12/81

Gilbert Associates, Inc. Reading, Pennsylvania CALCULATION	SUBJECT SRV/RSRV FORCING FUNCTIONS SRV NETWORK - 280°F LOOP SEAL		CIS:D 2.11.1		PAGE 3 OF 61 PAGES
	REV. 0	1	2	3	
	MICROFILMED				
	ORIGINATOR J.N. CAJIGAS				
	DATE 3-2-82				



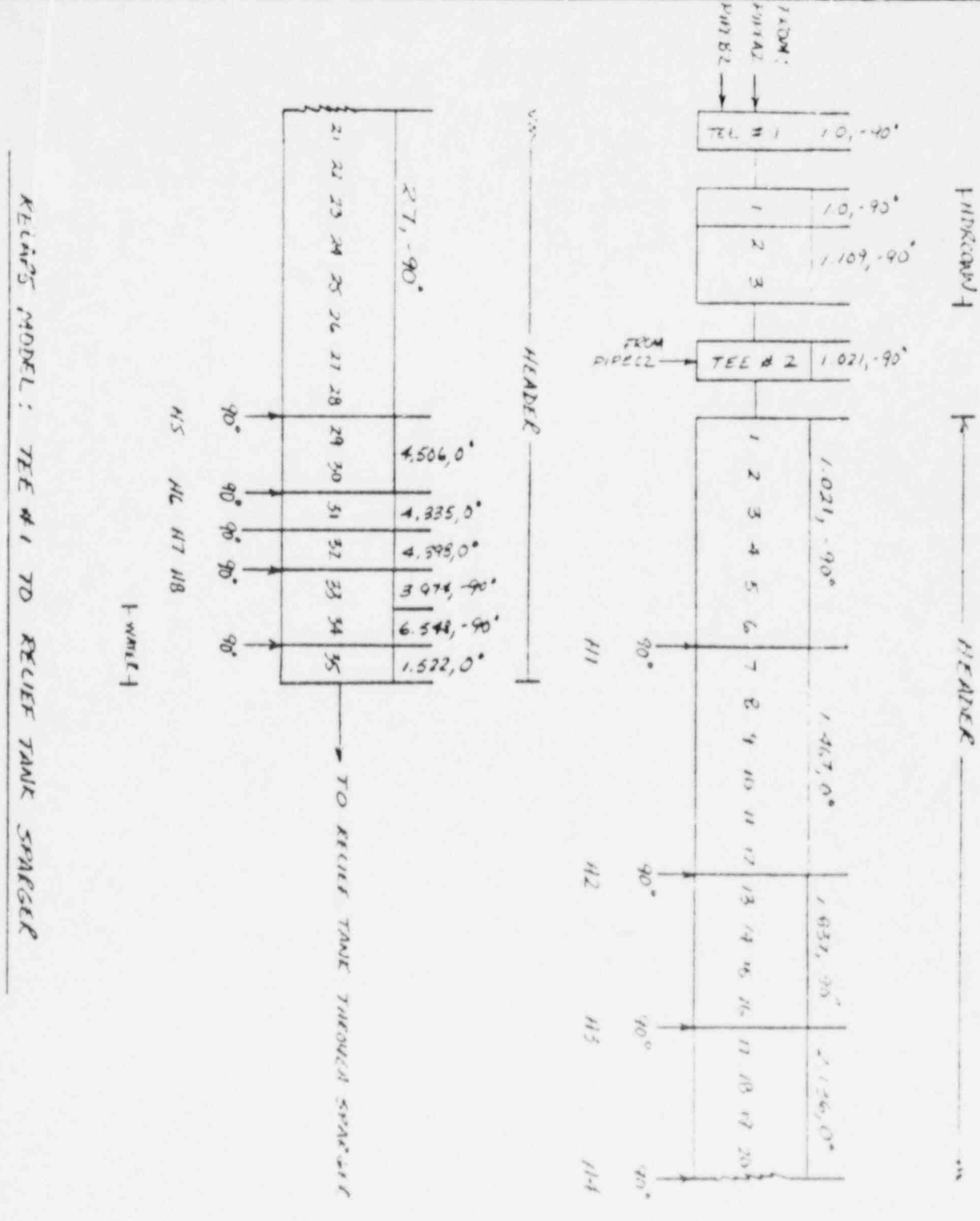
RELAP5 MODEL: FREE THROUGH SRV-B TO TEE NO. 1

* LENGTH MODIFIED TO IMPROVE OPERATION OF RELAP5, SEE PAGE 12

10-4

GAI 448 12/81

Gilbert Associates, Inc. Reading, Pennsylvania CALCULATION	SUBJECT				2 111		PAGE 15 OF 31 PAGES
	REV.	0	1	2	3		
	MICROFILMED						
	ORIGINATOR [illegible]						
	DATE 3-4-82						



10-6

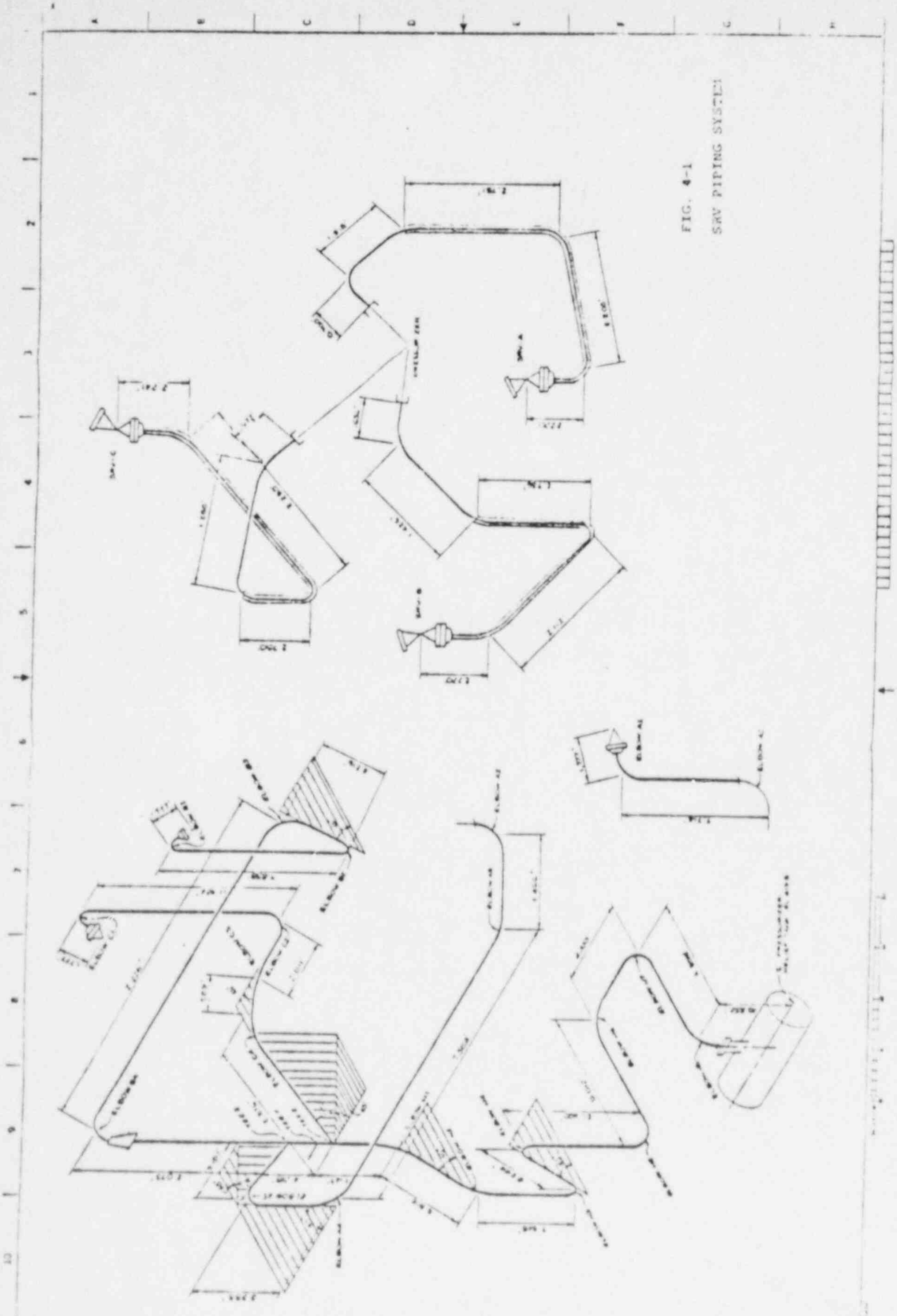


FIG. 4-1
SRV PIPING SYSTEM

TABLE 5-1

SRV NETWORK FORCE DESCRIPTION (SEE FIGURE 4-1)

FORCE NO.	FROM	TO
1	SRV-A	Elbow A1
2	Elbow A1	Elbow A2
3	Elbow A2	Elbow A3
4	Elbow A3	Elbow A4
5	Elbow A4	Elbow A5
6	Elbow A5	Tee #1
7	SRV-B	Elbow B1
8	Elbow B1	Elbow B2
9	Elbow B2	Elbow B3
10	Elbow B3	Elbow B4
11	Elbow B4	Tee #1
12	SRV-C	Elbow C1
13	Elbow C1	Elbow C2
14	Elbow C2	Elbow C3
15	Elbow C3	Elbow C4
16	Elbow C4	Tee #2
17	Tee #1	Tee #2
18	Tee #2	Elbow H1
19	Elbow H1	Elbow H2
20	Elbow H2	Elbow H3
21	Elbow H3	Elbow H4
22	Elbow H4	Elbow H5
23	Elbow H5	Elbow H6
24	Elbow H6	Elbow H7
25	Elbow H7	Elbow H8

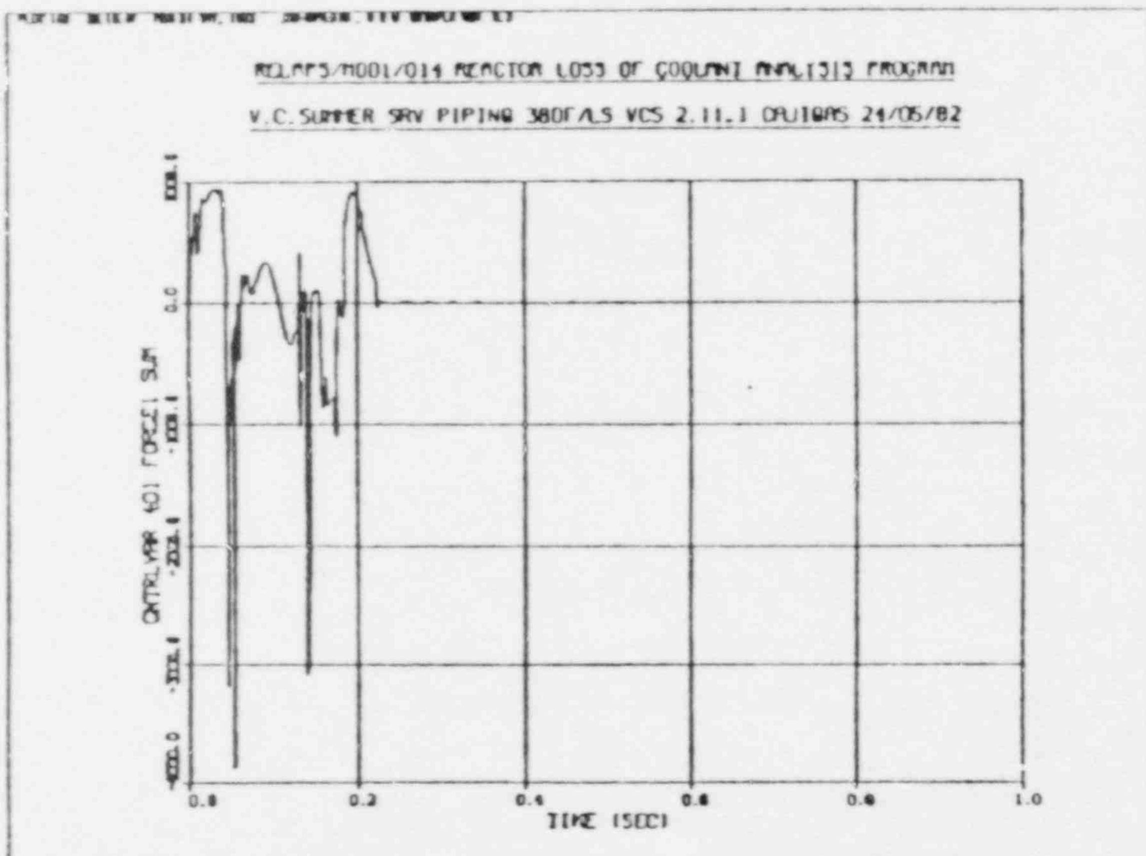

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C. SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUTGRS 24/05/82
CNTRLVAR 401 FORCE1 SUM
TIME (SEC)

FIGURE 5.1

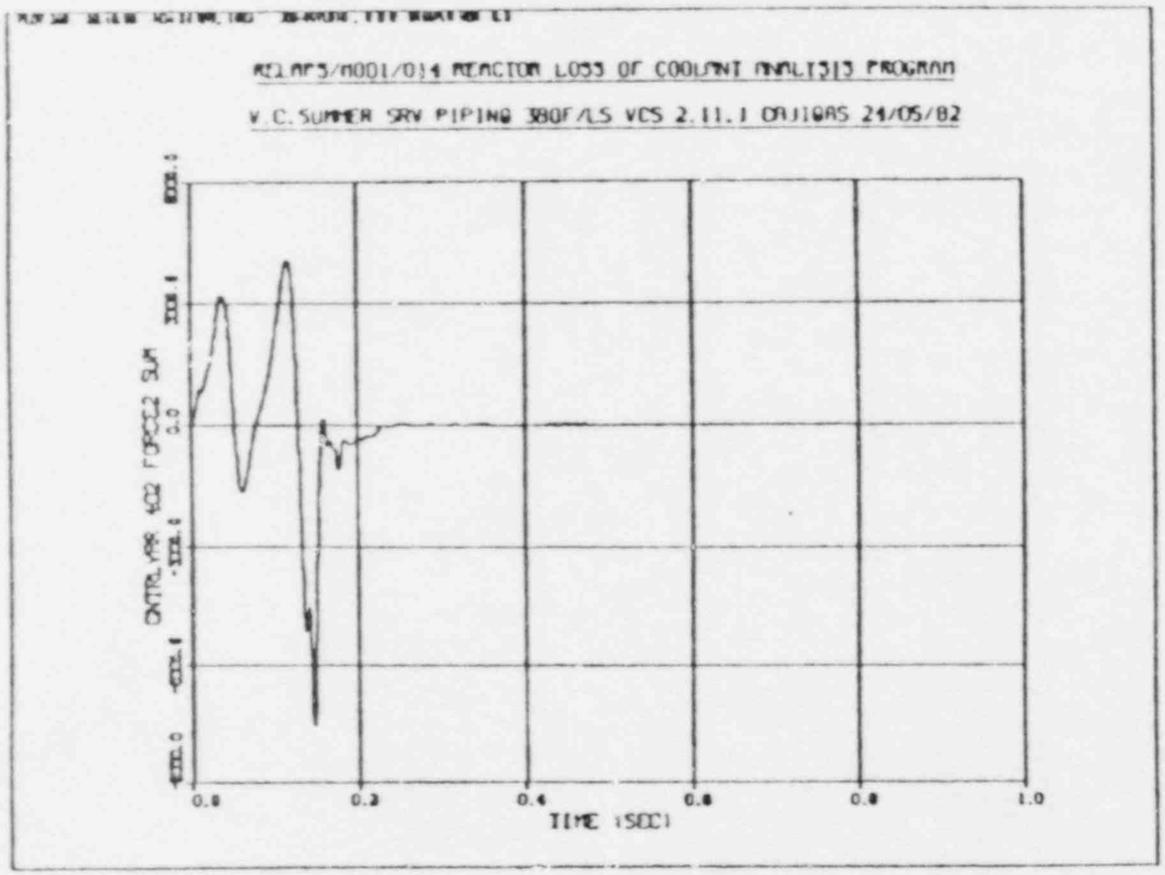
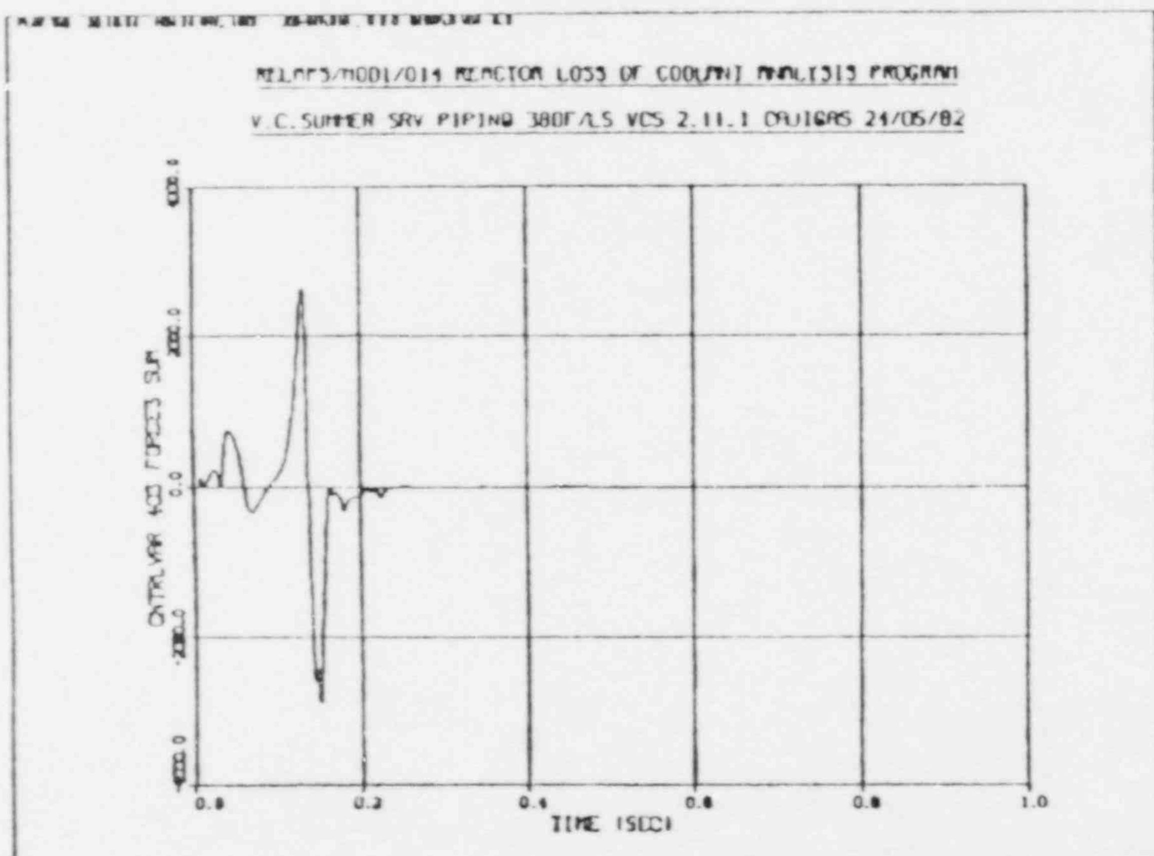


FIGURE 5.2

10-10



10-11

FIGURE 5.3

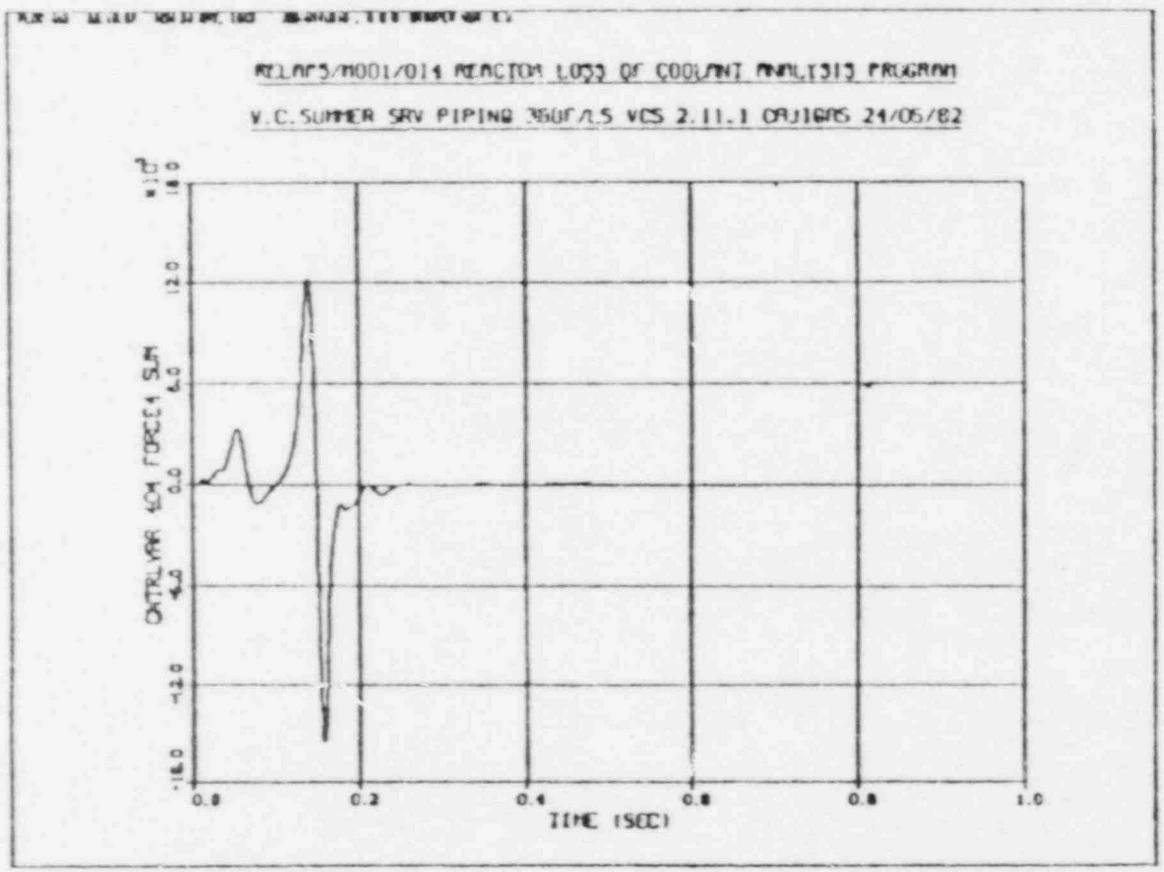
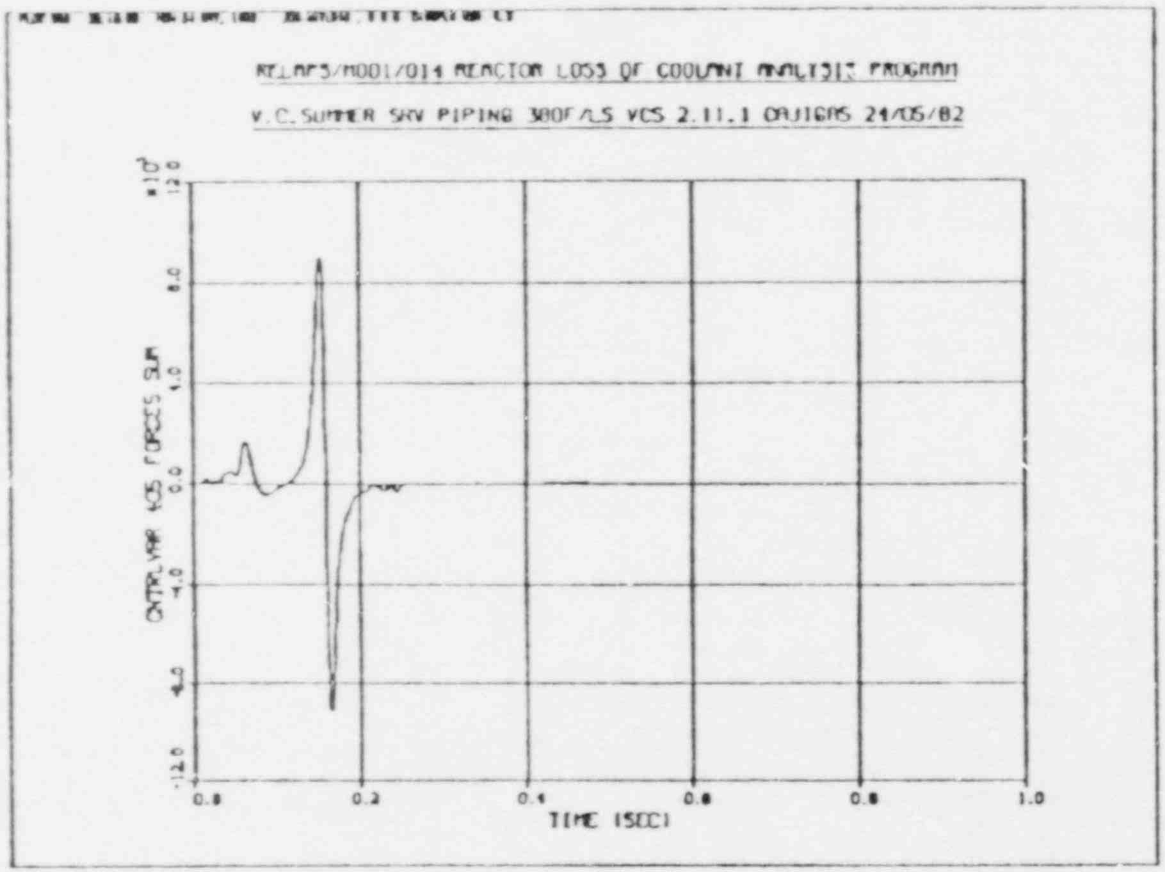

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 360F/1.5 VCS 2.11.1 CRJ1GRS 24/05/82
CNTRLVAR 404 FORCE14 SUM
TIME (SEC)

FIGURE 5.4


RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIGRS 24/05/82
CNTRLVAR 405 FORCES SUM
×10²
12.0
8.0
4.0
0.0
-4.0
-4.0
-12.0
0.0
0.2
0.4
0.6
0.8
1.0
TIME (SEC)

FIGURE

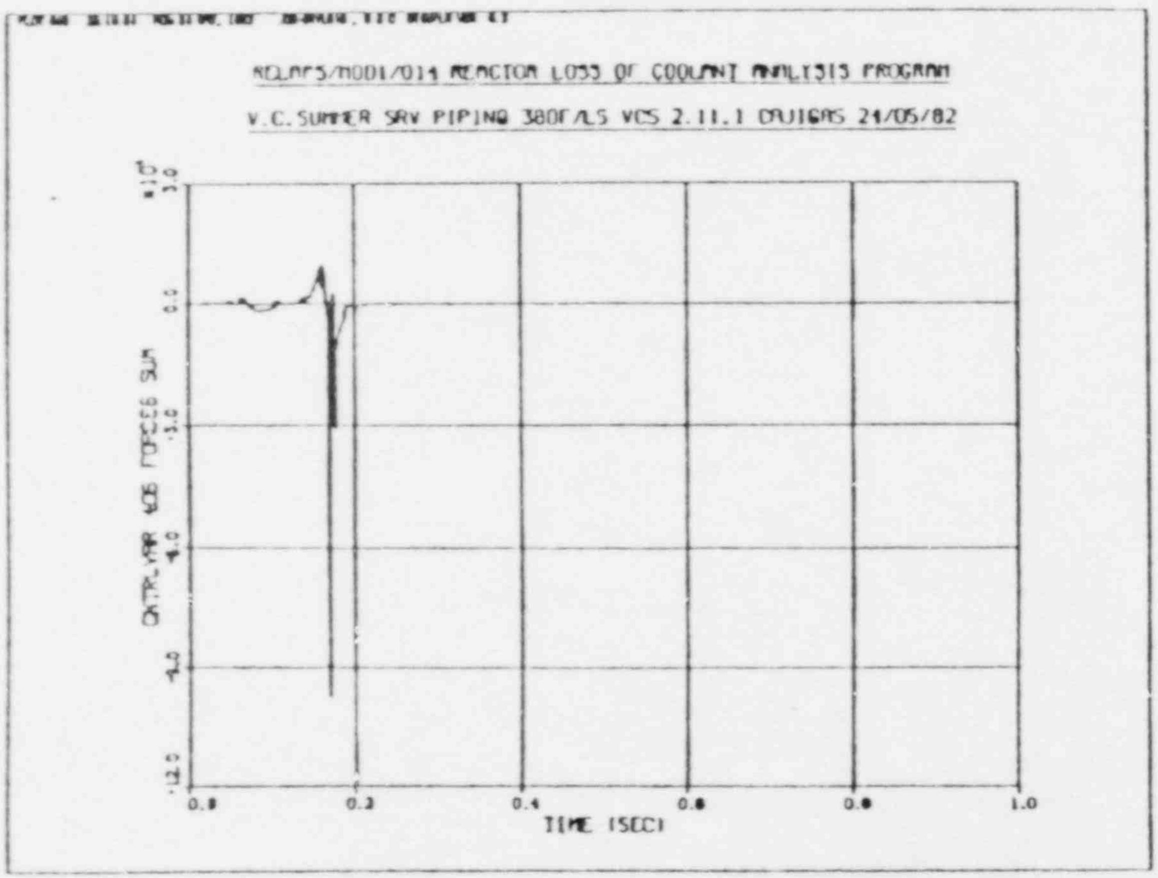

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIGAS 24/05/82
TIME (SEC)
0.0
0.2
0.4
1.0

FIGURE 5.6

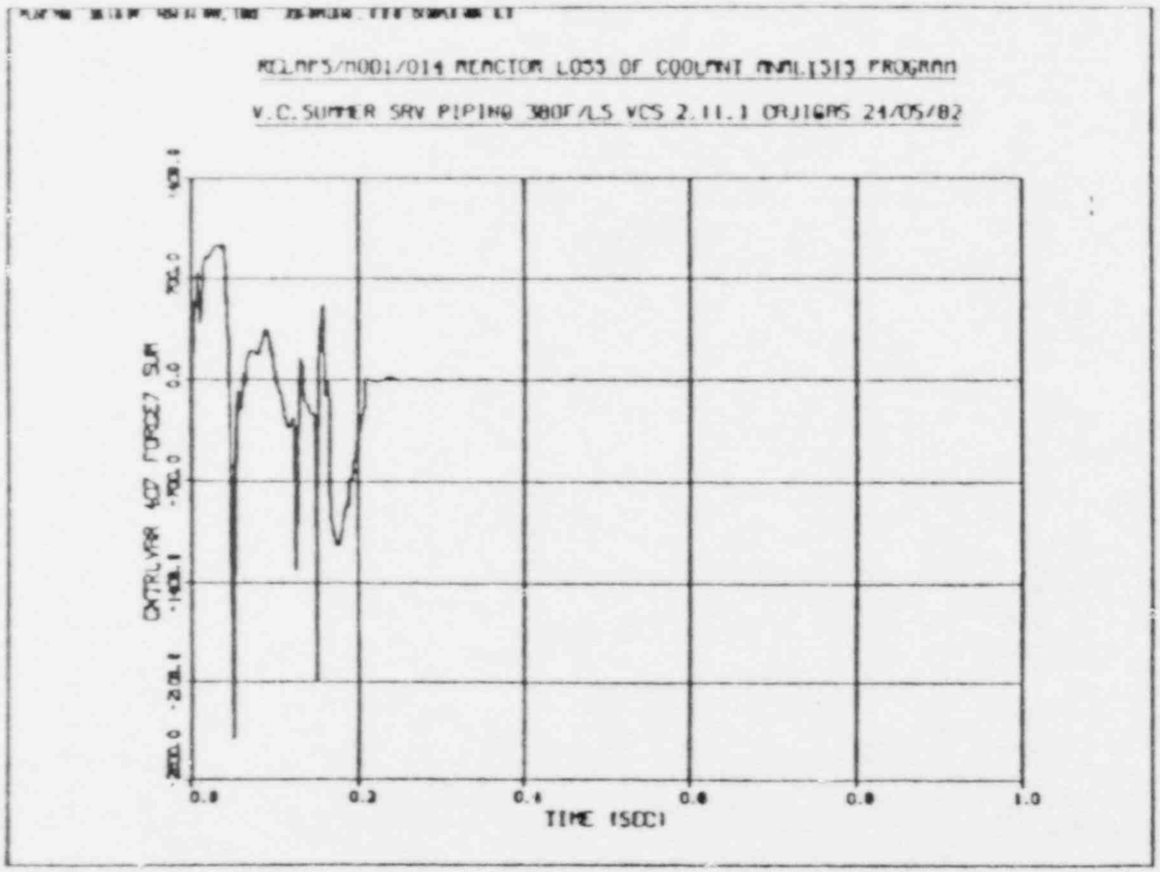
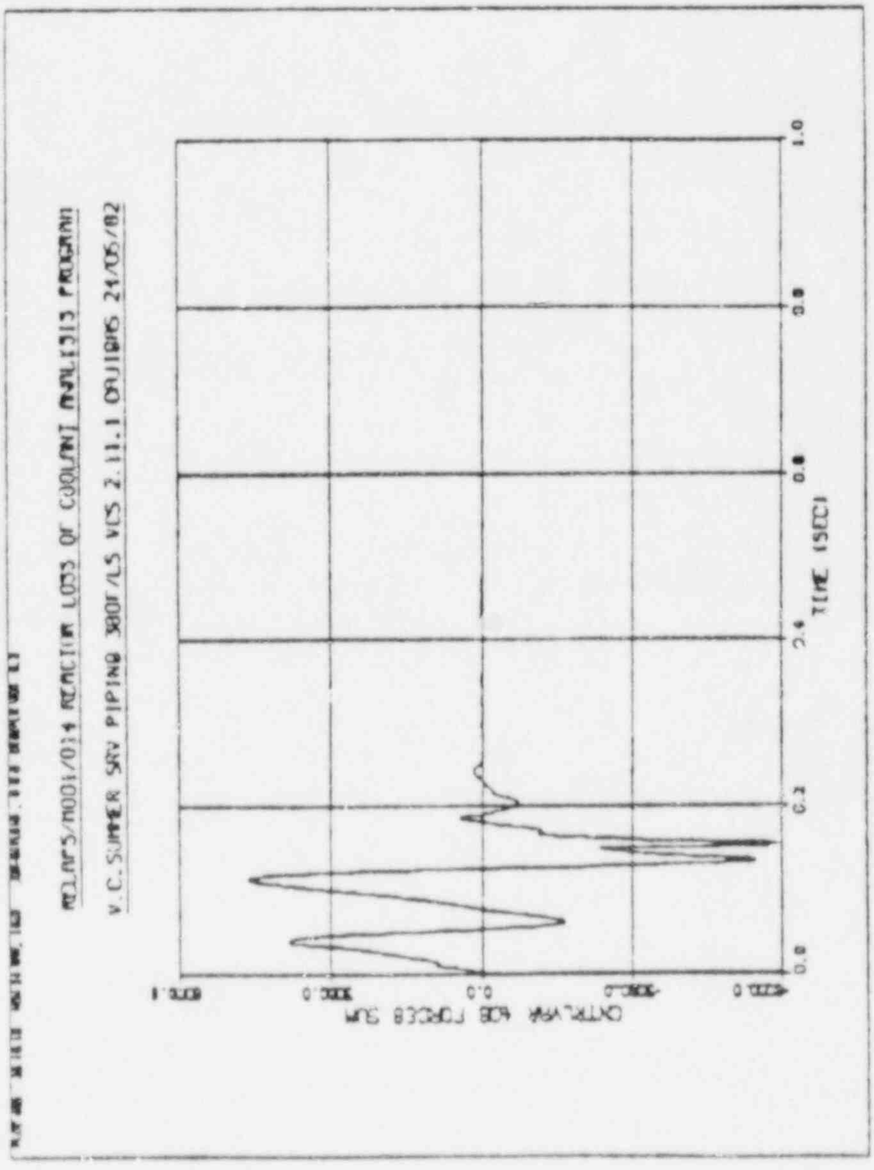

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRJIGAS 24/05/82
CNTRLVAR 407 FORCE7 SUM
0.0
0.2
0.4
0.6
0.8
1.0
TIME (SEC)

FIGURE 5.7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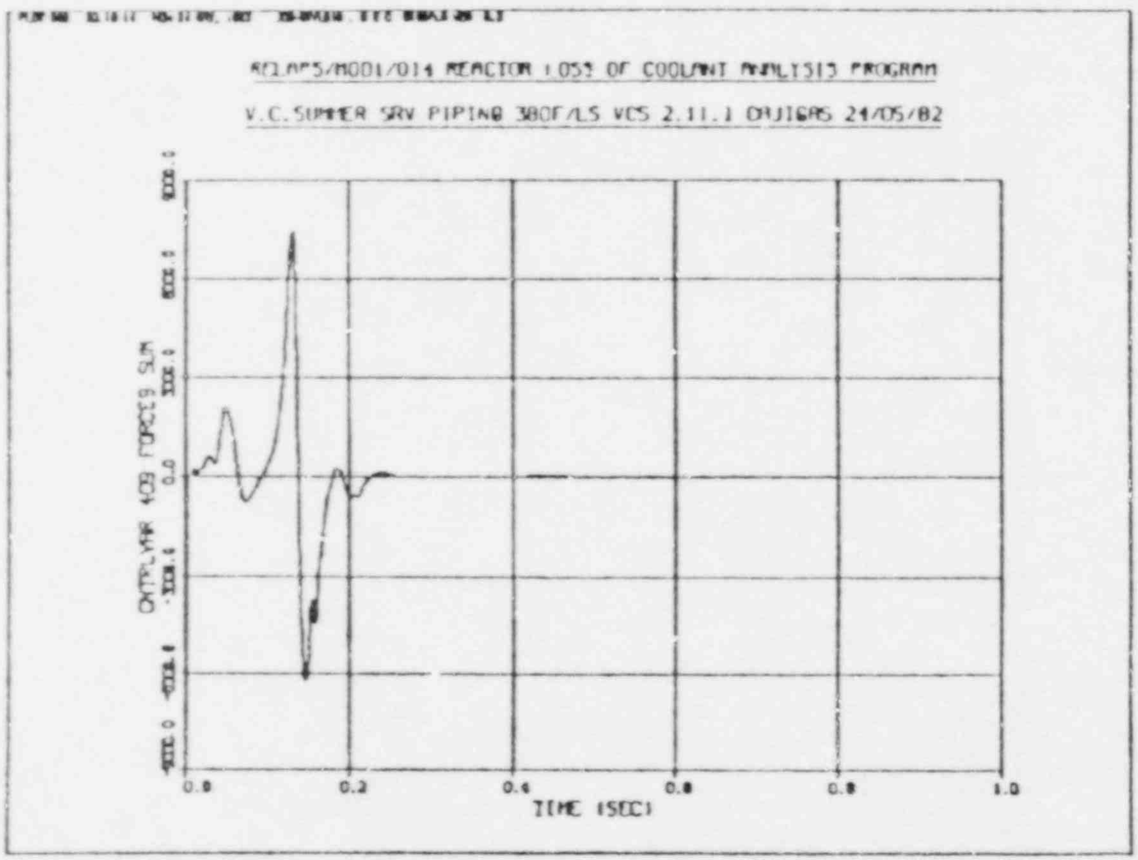
V.C.SUMMER SRV PIPING 300F/LS VCS 2.11.1
CNTRLVAR 408 FORCES SUM
TIME (SEC)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V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIGRS 24/05/82
CNTRLVAR 409 FORCES SUM
0.0
0.2
0.4
0.6
0.8
1.0
TIME (SEC)

FIGURE 5.9

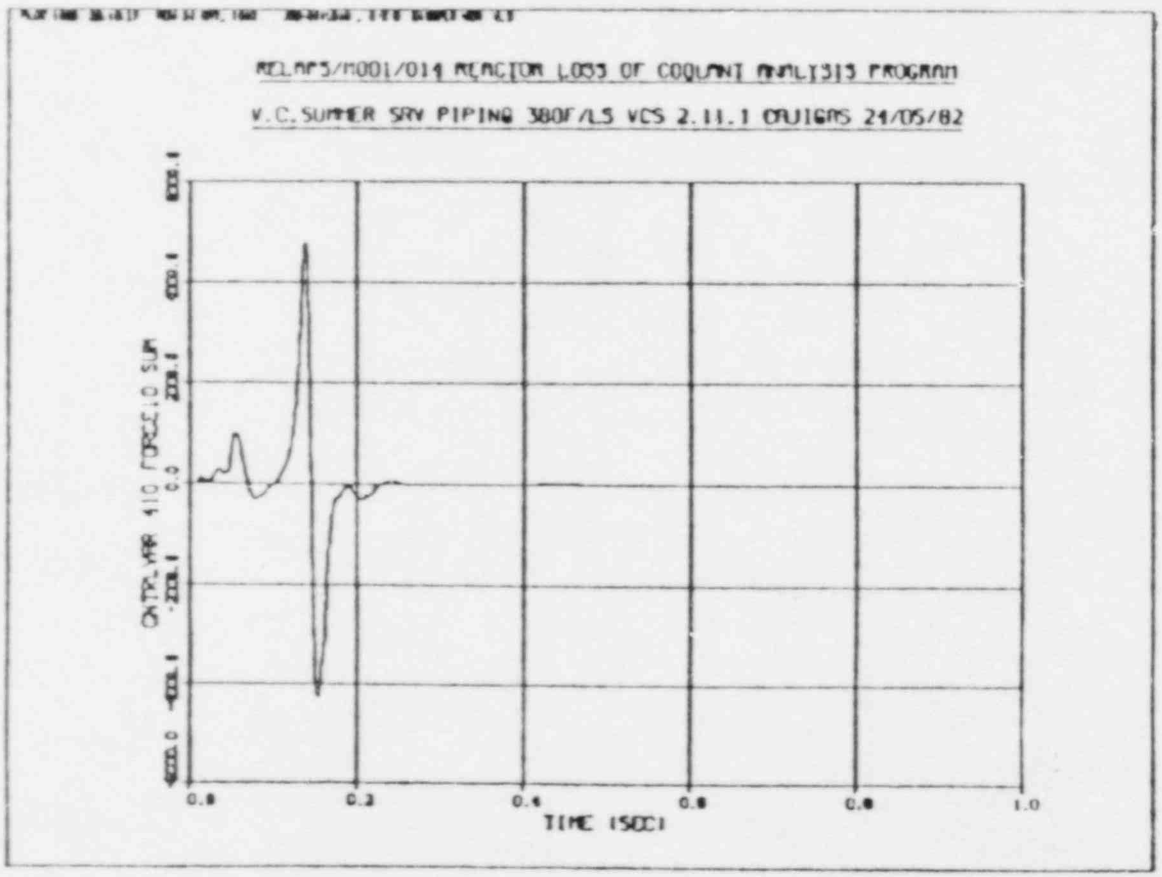

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIGAS 24/05/82
CNTRLVAR 410 FORCE10 SUM
TIME (SEC)

FIGURE 5.10

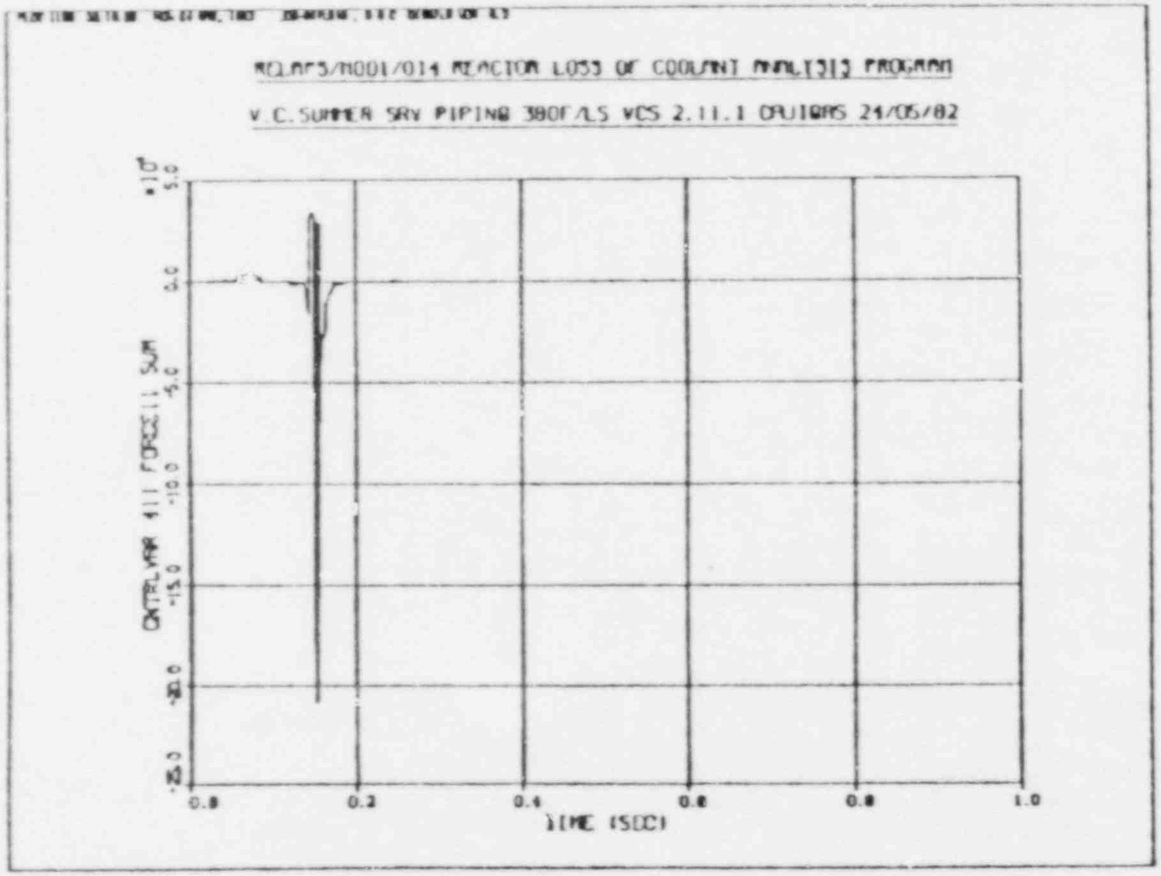

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIQRS 24/05/82
CNTRLVAR 411 FORCE11 SUM
*10
5.0
0.0
-5.0
-10.0
-15.0
-20.0
-25.0
0.0
0.2
0.4
1.0
TIME (SEC)

FIGURE 5.11

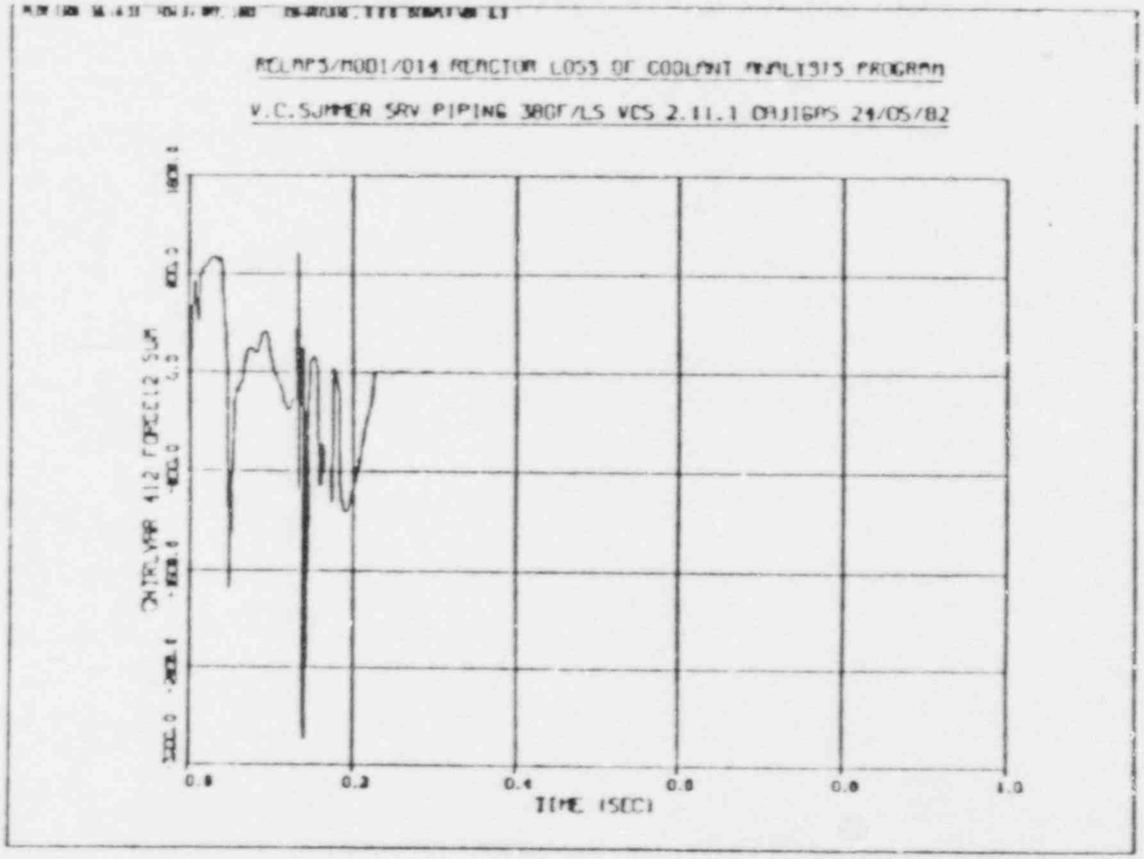
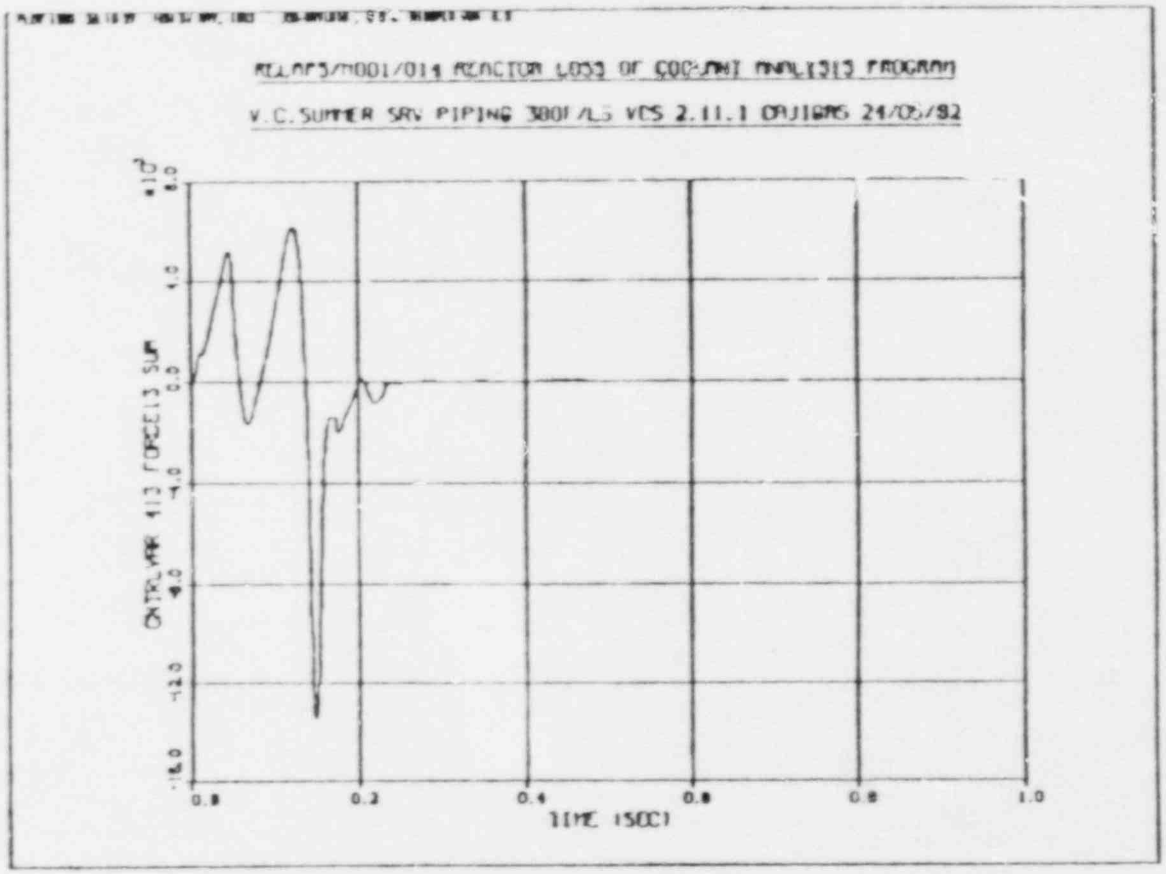
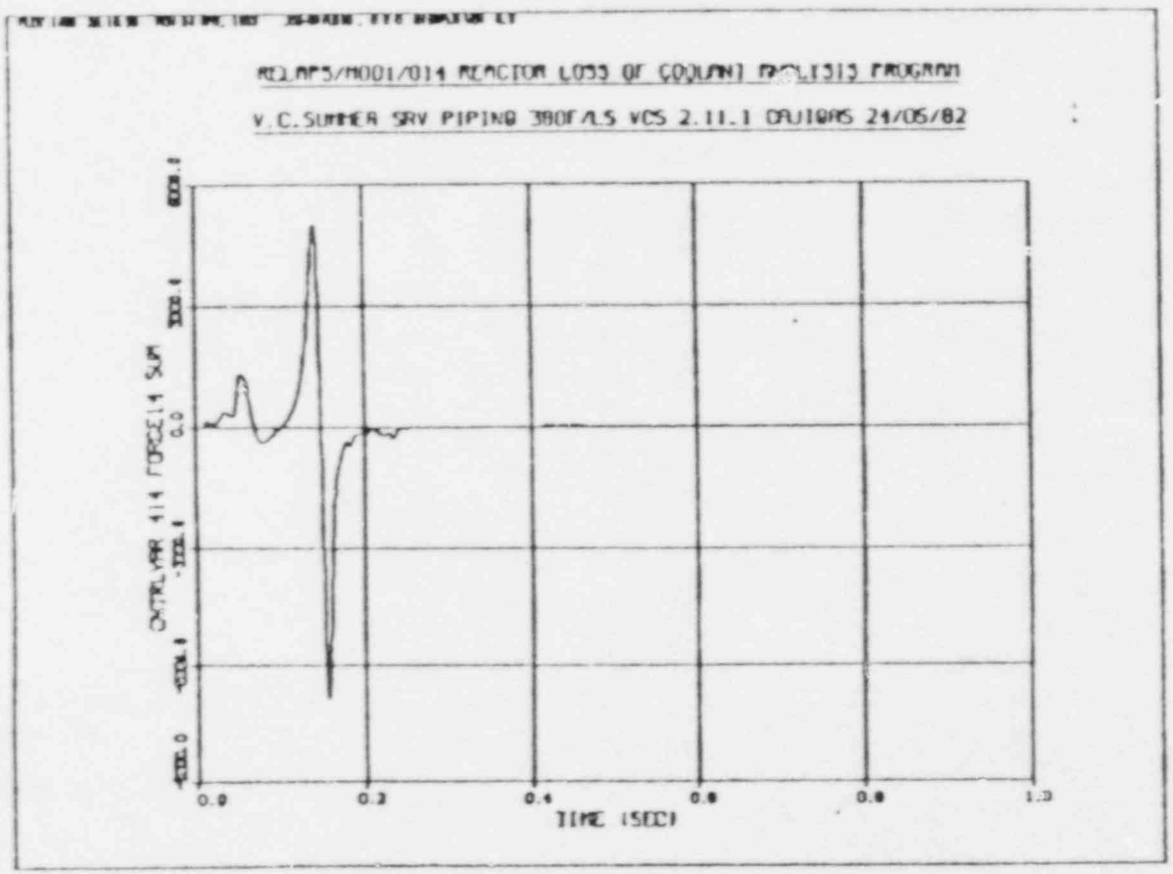

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRJIGPS 24/05/82
CNTRLVAR 412 FORCE12 SUM
0.0
0.2
0.4
1.0
TIME (SEC)

FIGURE 5.1



RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1
24/05/82
CNTRLVAR 413 FORCE13 SUM
TIME (SEC)
0.0
0.2
0.4
1.0

FIGURE 5.13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V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIBAS 24/05/82
CNTRLVAR 414 FORCE14 SUM
TIME (SEC)
0.0
0.2
0.4
1.0

FIGURE 5.15

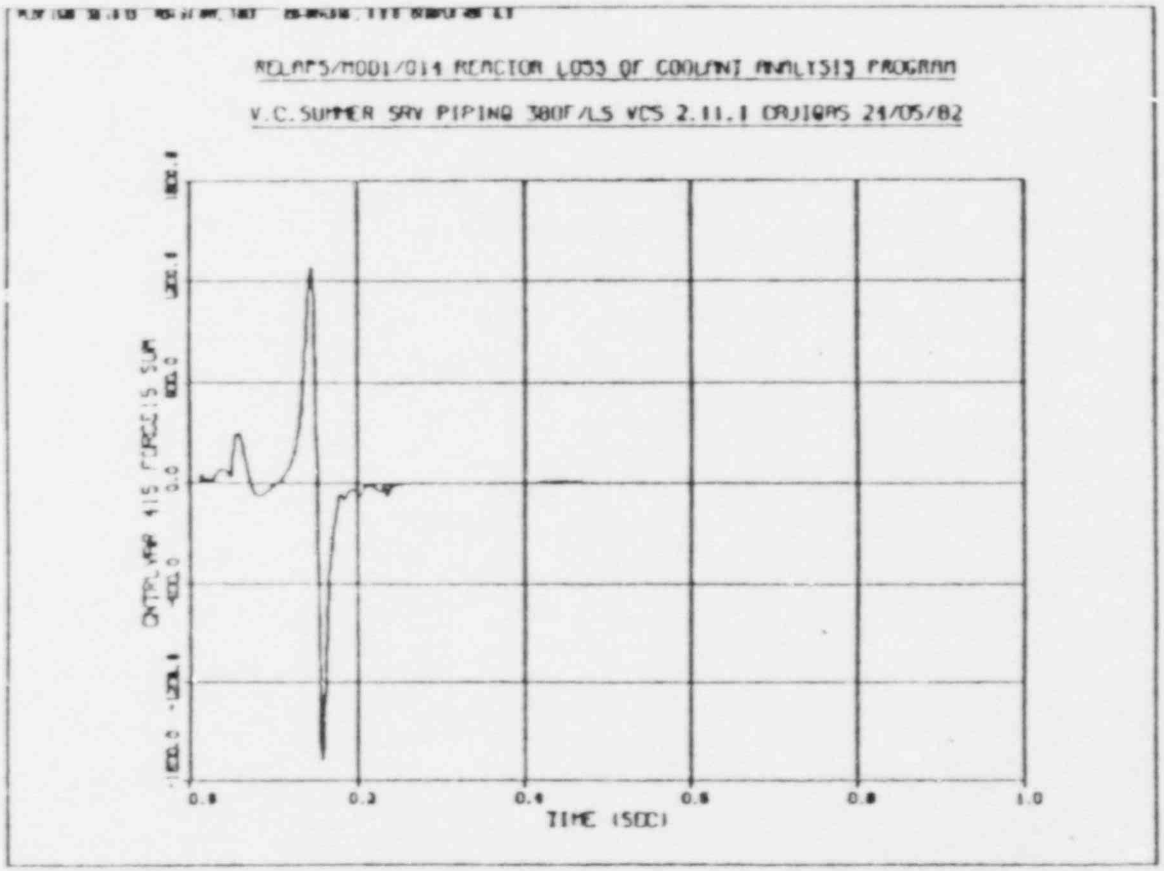

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRJIGRS 21/05/82
CNTRLVAR 415 FORCES SUM
0.0
TIME (SEC)
0.2
0.4
0.6
0.8
1.0

FIGURE 5.1

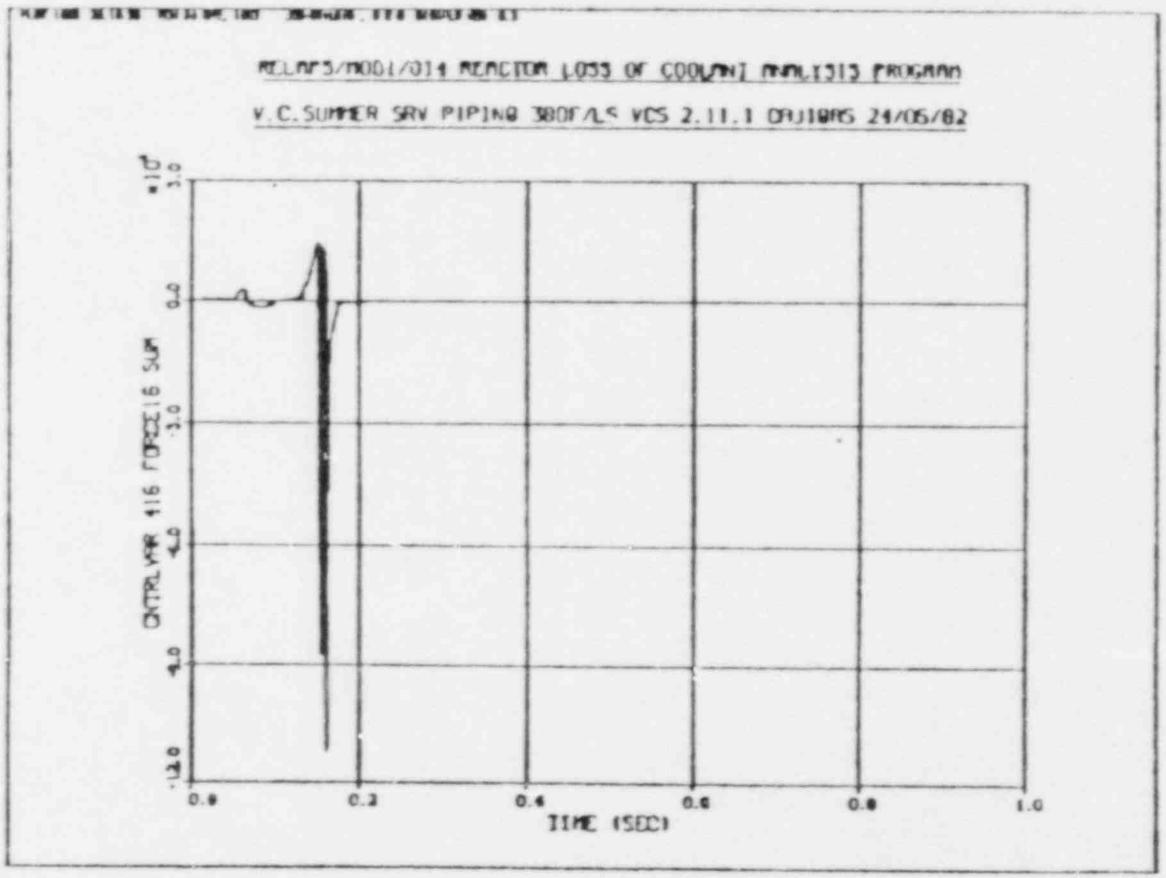
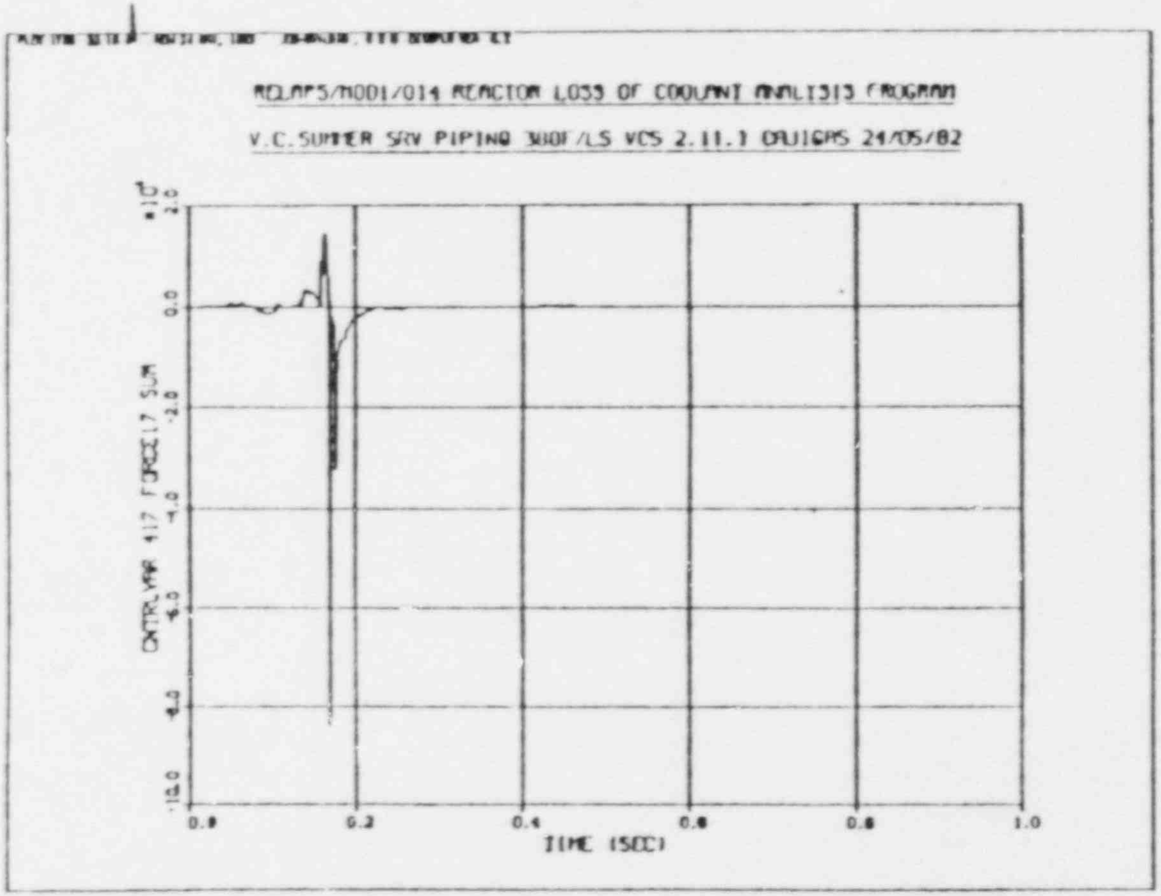
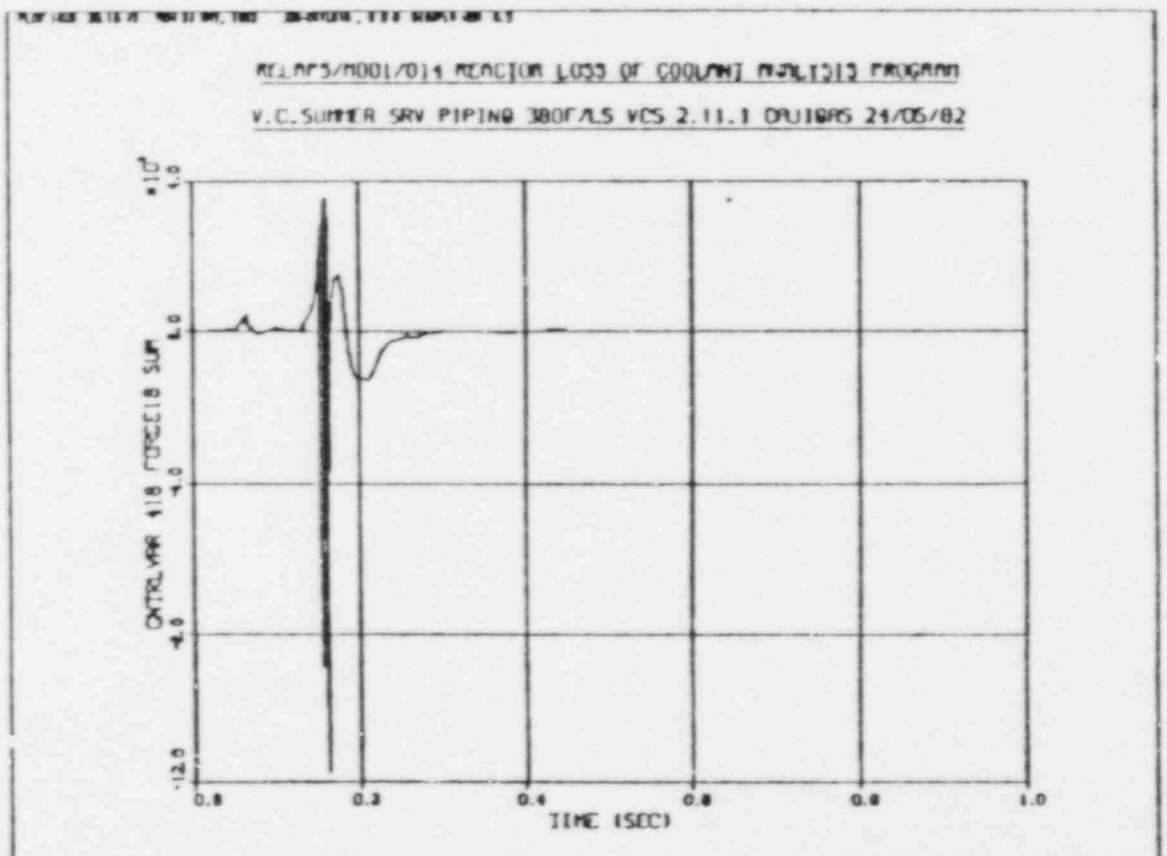

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRJ1BRS 24/05/82
CNTRLVAR 416 FORCE16 SUM
TIME (SEC)
0.0
0.2
0.4
0.6
0.8
1.0

FIGURE 5.16



RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIGAS 24/05/82
*10
CNTRLVAR 417 FORCE17 SUM
2.0
0.0
-2.0
-4.0
-6.0
-8.0
-10.0
0.0
0.2
0.4
0.6
0.8
1.0
TIME (SEC)

FIGURE 5.17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CNTRLVAR 418 FORCE18 SUM
TIME (SEC)

FIGURE 5.18

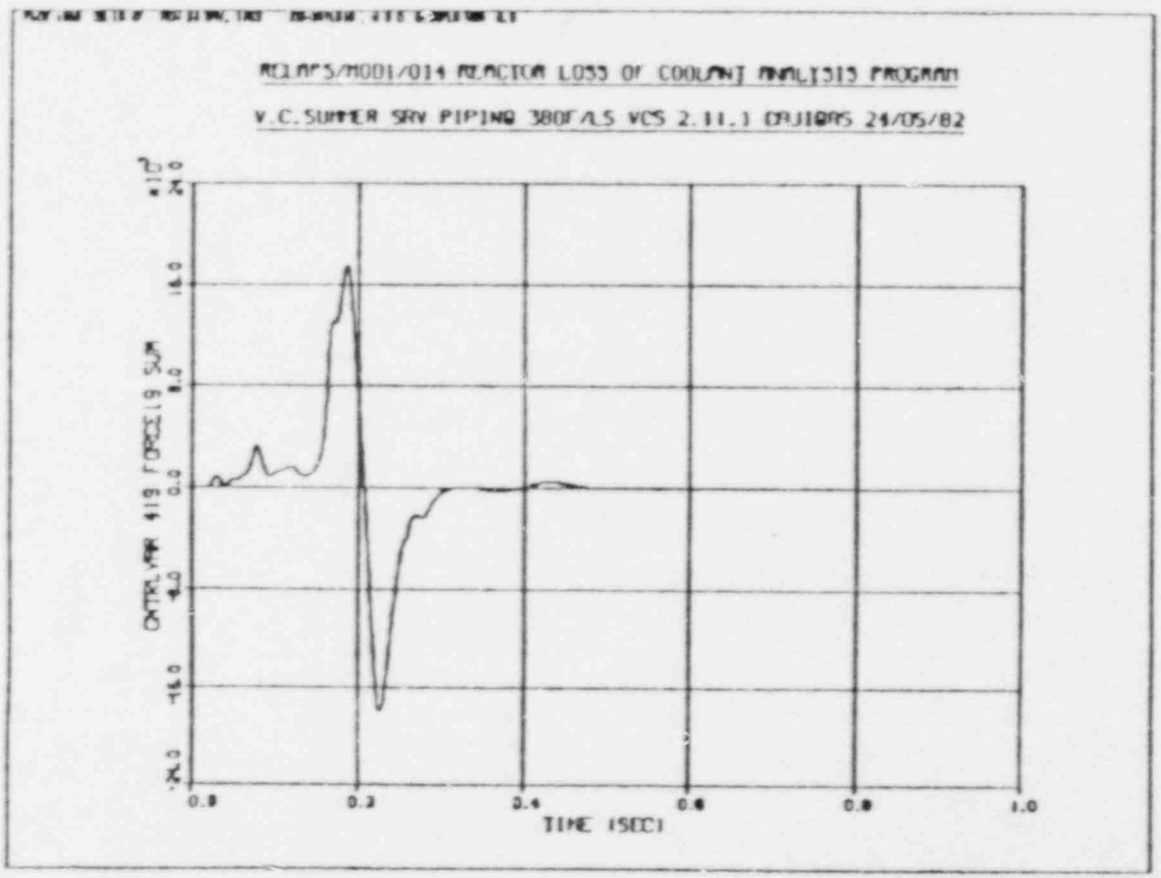

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRJ1ORS 24/05/82
CNTRLVAR 419 FORCE19 SUM
*10²
24.0
16.0
8.0
0.0
-8.0
-16.0
-24.0
0.0
0.2
0.4
0.6
0.8
1.0
TIME (SEC)

FIGURE 5.19

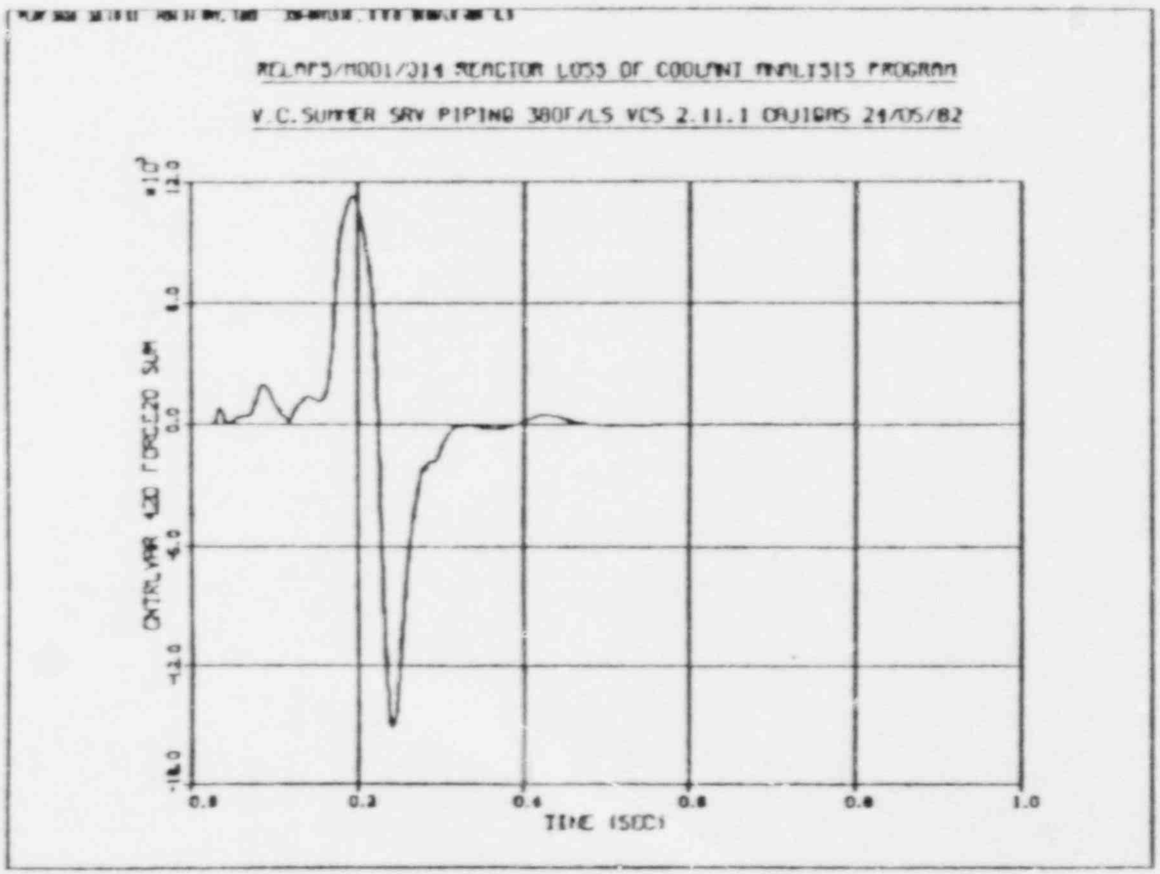

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIBAS 21/05/82
CNTRLVAR 420 FORCE20 SUM
$*10^2$
12.0
6.0
0.0
-6.0
-12.0
-18.0
0.0
0.2
0.4
0.6
0.8
1.0
TIME (SEC)

FIGURE 5.20

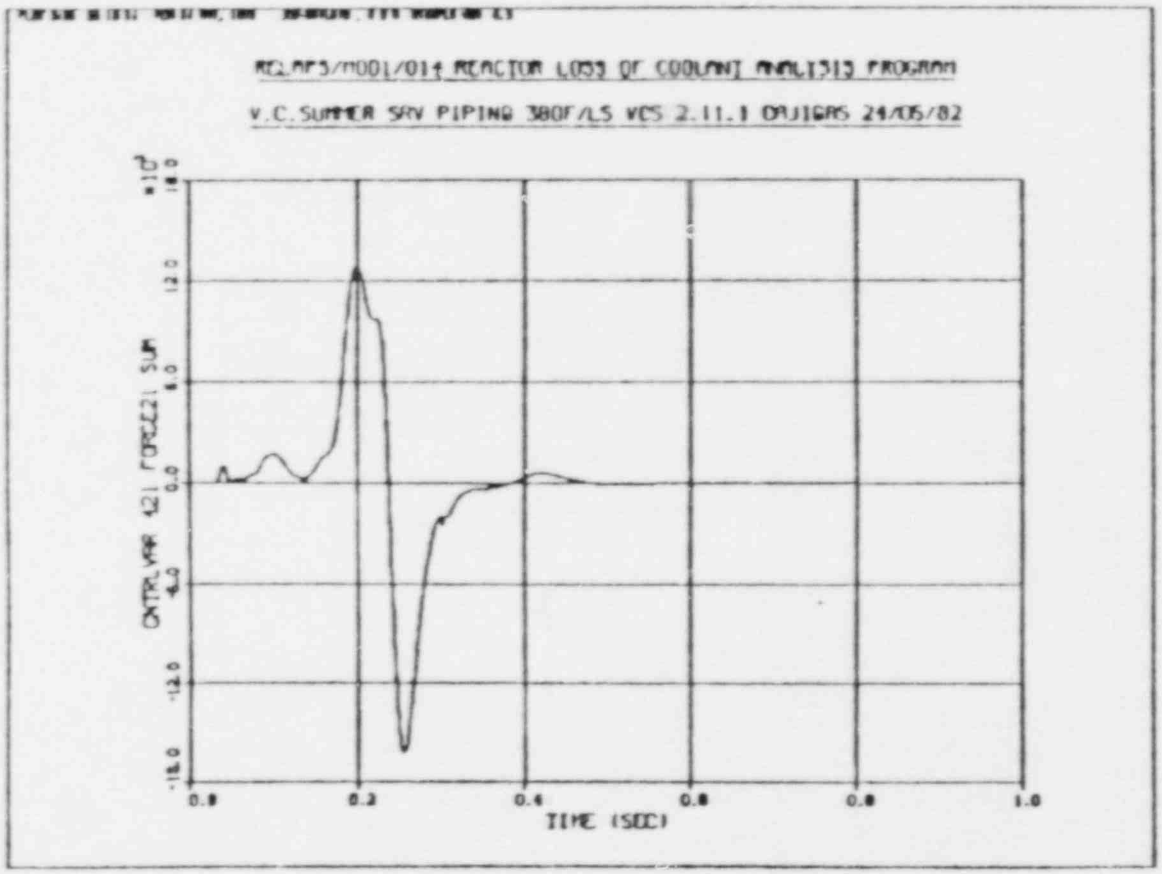

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIGRS 24/06/82
CNTRLVAR 421 FORCE21 SUM
*10^3
18.0
12.0
6.0
0.0
-6.0
-12.0
-18.0
0.0
0.2
0.4
1.0
TIME (SEC)

FIGURE 5.21

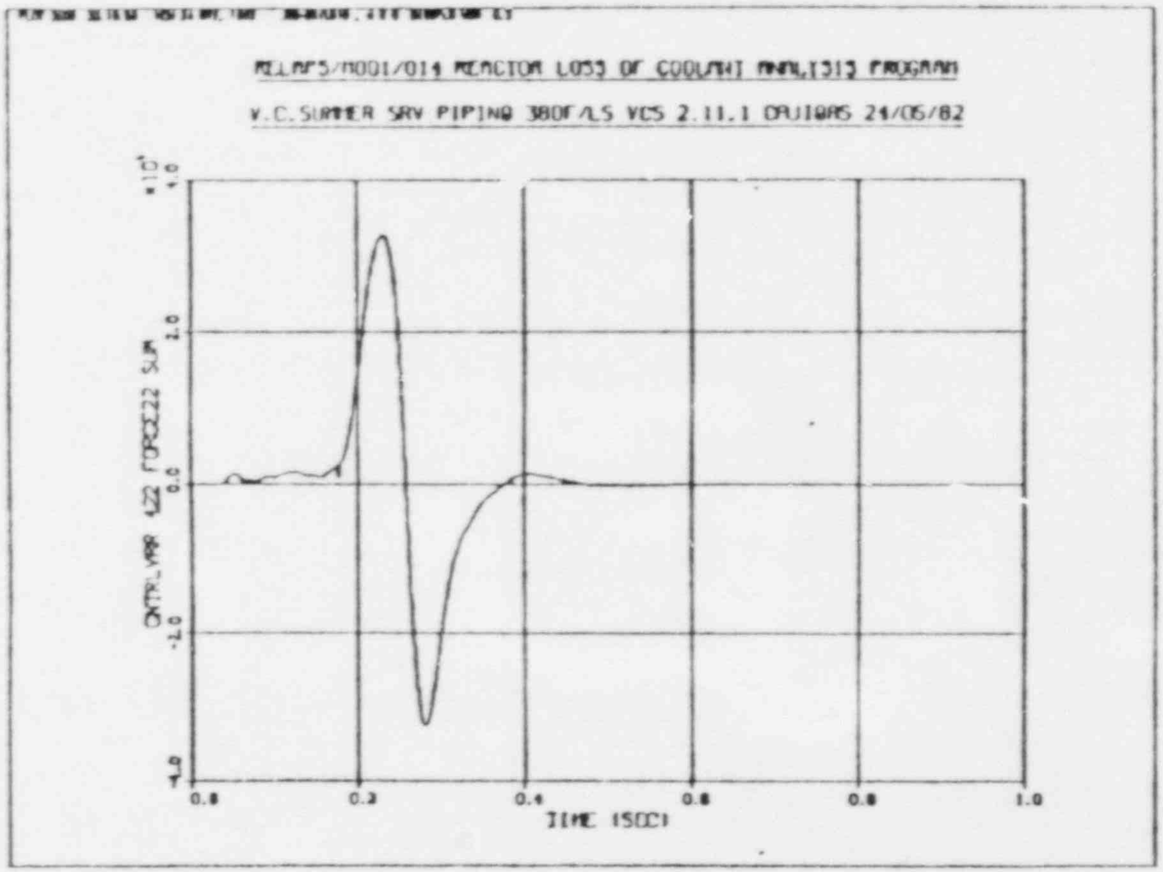

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRUIQAS 24/05/82
CNTRLVAR 422 FORCE22 SUM
*10'
4.0
2.0
0.0
-2.0
-4.0
0.0
0.2
0.4
0.6
0.8
1.0
TIME (SEC)

FIGURE 5.22

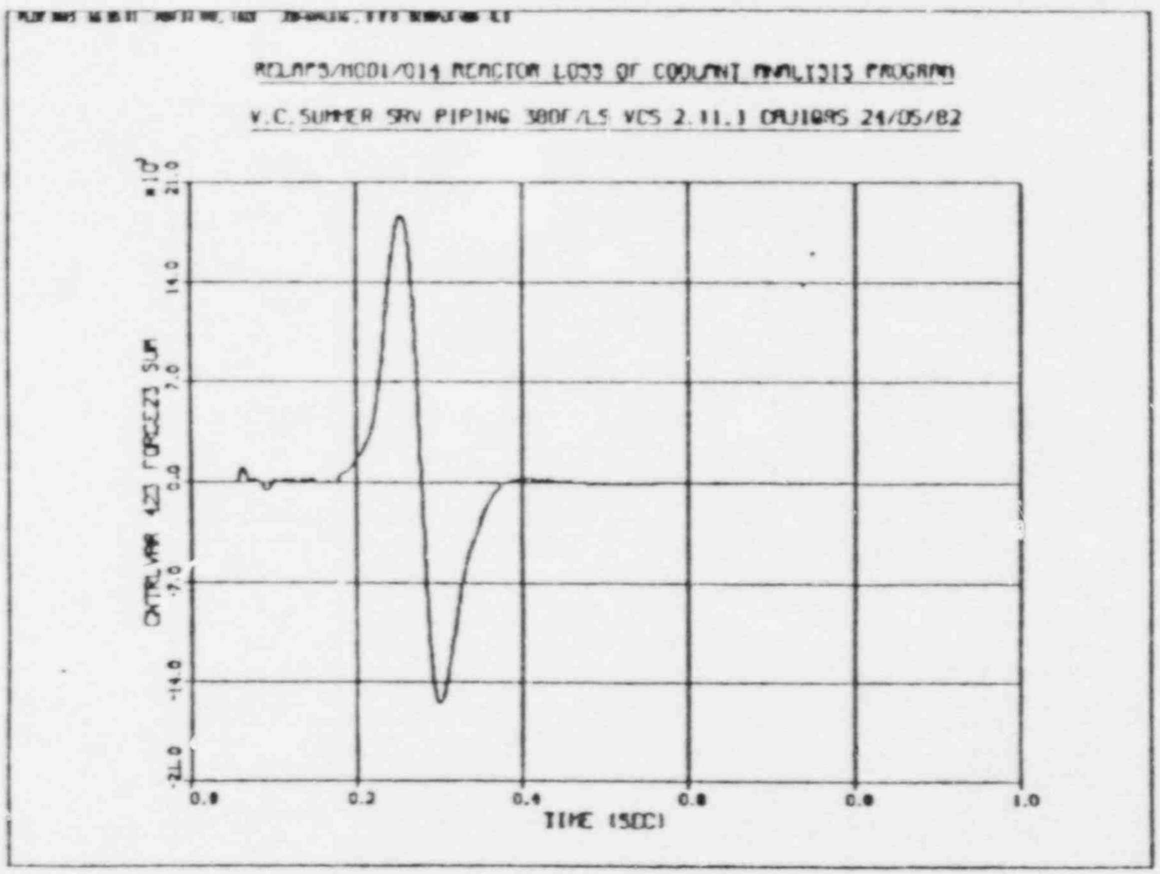

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C. SUMMER SRV PIPING 380F/LS VCS 2.11.1 DAJ10RS 24/05/82
CNTRLVAR 423 FORCE23 SUM
TIME (SEC)
21.0
14.0
7.0
0.0
-7.0
-14.0
-21.0
0.0
0.2
0.4
1.0

FIGURE 5.23

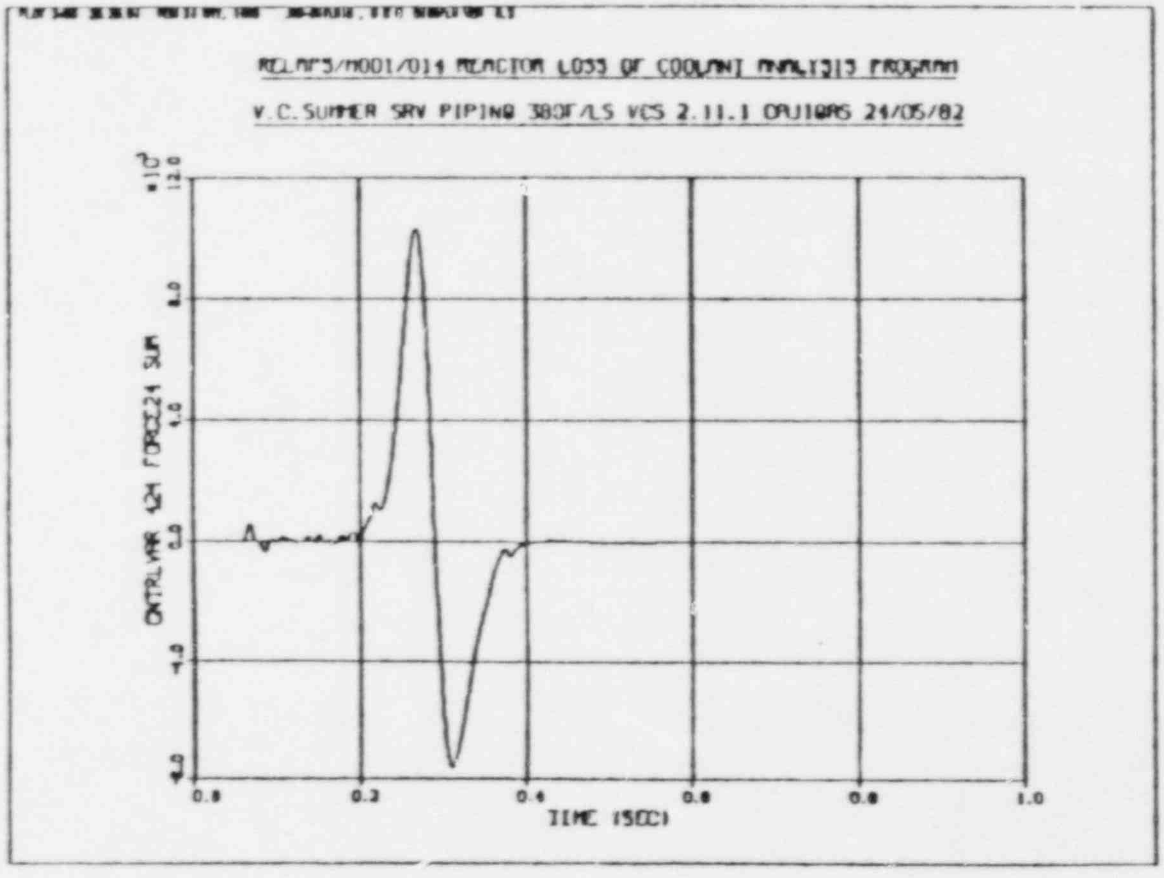

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CRU10RS 24/05/82
CNTRLVAR 424 FORCE24 SUM
*10²
12.0
8.0
4.0
0.0
-4.0
-8.0
0.0
0.2
0.4
0.6
0.8
1.0
TIME (SEC)

FIGURE 5.24

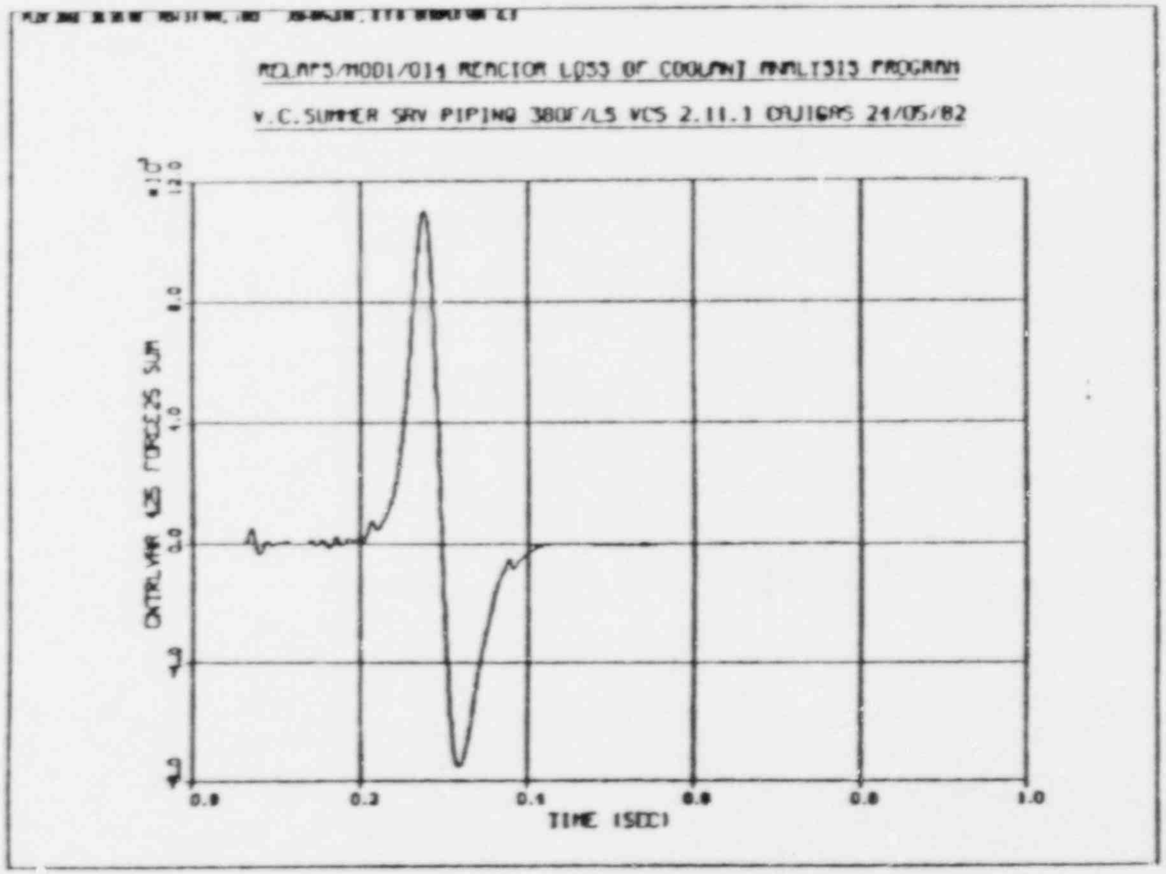

RELAP5/MOD1/014 REACTOR LOSS OF COOLANT ANALYSIS PROGRAM
V.C.SUMMER SRV PIPING 380F/LS VCS 2.11.1 CFUIGAS 24/05/82
CNTRLVAR 425 FORCE25 SUM
TIME (SEC)

FIGURE 5.25

EG&G Question No. 11

The submittal presents the loop seal temperature distribution that was used as input to the RELAP 5 analysis, but does not explain how the simmering of the loop seal water through the safety valve was simulated in the RELAP 5 calculations. Explain how the valve flow area was varied in the analysis as water passed through the valve and how long the simmering process lasted before the valve popped open. Specify the resulting water flow rate and explain why this was deemed to be appropriate.

Response:

The SRV's were ramped open linearly in .040 seconds. No simmering was accounted for. Since the loop seal water temperature is 380°F the water flow through the valve was determined by the RELAP 5 model. The maximum water flow through the valve (i.e., before loop seal clearing) is approximately 437 lbm/sec @ 0.1 seconds, well after the valve is fully open.

The model used for valve opening is considered conservative since the simmering was not taken into account and the valve was fully open in .040 seconds. Simmering would prolong the valve opening time and allow more of the loop seal water downstream of the valve to flash, reducing loads.

EG&G Question No. 12

The submittal states that the thermal hydraulic analysis was performed using RELAP 5/MOD1 and that the RELAP 5 control system was used to calculate the fluid forces. Identify the methodology used to calculate forces from RELAP 5/MOD1 and provide additional verification that the methodology produces accurate force histories for similar problems.

Response:

The methodology used in calculating forces is attached (Attachment pages 12-2 through 12-12); in addition the same methodology is used in RELAP 5 - FORCE where additional comparisons are given for the EPRI/CE test Nos. 1411 and 908 (Ref. RELAP 5-FORCE Verification Manual - UCCEL, 1984).

The above information along with that previously submitted verifies the methodology used in determining the SRV discharge piping fluid forces.

Appendix A

FLUID TRANSIENT INDUCED FORCES CALCULATION

A.1 GENERAL DISCUSSION

Calculation of the pipe forcing functions requires modelling of the piping system and analysis of the applicable transients with the RELAP5/MOD1[1] program. While the code evaluates the time dependent thermofluid conditions in the piping, the transient induced forces are not directly evaluated. These forces can, however, be calculated from the results of the thermofluid calculations as derived below.

A.2 THEORY

The force on a piping system can be evaluated by the following equation[2,3] for homogeneous one-dimensional flow:

$$F = (P + \rho V|V|)_1 A_1 - (P + \rho V|V|)_2 A_2 + \frac{\partial}{\partial t} \int_x \rho V A dx \qquad \text{A-1}$$

(pipe force) (net pressure - momentum force) + (acceleration force)

where A = pipe flow area

F = force on piping

p = pressure

ρ = density

t = time

V = fluid velocity

1,2 = pipe section indices

as depicted in Figure A-1.

Since the fluid conditions are constant within any control volume, the acceleration force term above may be approximated as:

$$\frac{\partial}{\partial t} \int_x \rho VA dx = \frac{d}{dt} (m\, L_{cv}) = L_{cv} \frac{dm}{dt} \qquad \text{A-2}$$

where L_{cv} = control volume length

m = control volume mass flow rate

Therefore, the total force equation for a constant area control volume becomes:

$$F = (P + \rho V|V|)_1 A - (P + \rho V|V|)_2 A + L_{cv} \frac{dm}{dt} \qquad \text{A-3}$$

Extending the above expression to a two-fluid case as used by RELAP5/MOD1:

$$\begin{aligned} F = & \Big[(P + \alpha_\ell \rho_\ell V_\ell |V_\ell| + \alpha_v \rho_v V_v |V_v|)_1 \\ & - (P + \alpha_\ell \rho_\ell V_\ell |V_\ell| + \alpha_v \rho_v V_v |V_v|)_2 \Big] A \\ & + L_{cv} \frac{d}{dt} (m_\ell + m_v) \end{aligned} \qquad \text{A-4}$$

where α = void fraction

l = liquid phase

v = vapor phase

If more than one control volume exists, the pressure-momentum terms cancel out except at an open end of the pipe as shown in Figure A-2. Therefore, the force for a straight run of pipe with more than one control volume becomes:

$$F = (P - P_\infty + \alpha_\ell \rho_\ell V_\ell |V_\ell| + \alpha_v \rho_v V_v |V_v|)_1 A + \sum_i \Big[L_{cv} \frac{d}{dt} (m_\ell + m_v) \Big]_i$$

where P_∞ = environmental pressure

If the pipe does not have any open ends, as in the case of an operational transient, the force term is simply:

$$F = \sum_i \left[L_{cv} \frac{d}{dt} (m_l + m_v) \right]_i \qquad \text{A-6}$$

A.3 APPLICATION AND VERIFICATION OF THEORY

Gilbert Associates, Inc. has developed a RELAP5/MOD1 processor for the evaluation of equations A-5 and A-6. This methodology has been verified by analyzing the steam line rupture given in Figure 3-3 which has also been analyzed by Moody[2], Strong and Baschiere[3], and Burke and Webb[4]. Figures A-4, A-5, and A-6 compare the results of the analysis using RELAP5/MOD1 with answers reported previously by Burke and Webb[4]. The magnitude and timing of the forces compare favorably. The small differences seen are probably due to the inclusion of friction in the RELAP5/MOD1 model, which was not included in the other calculations.

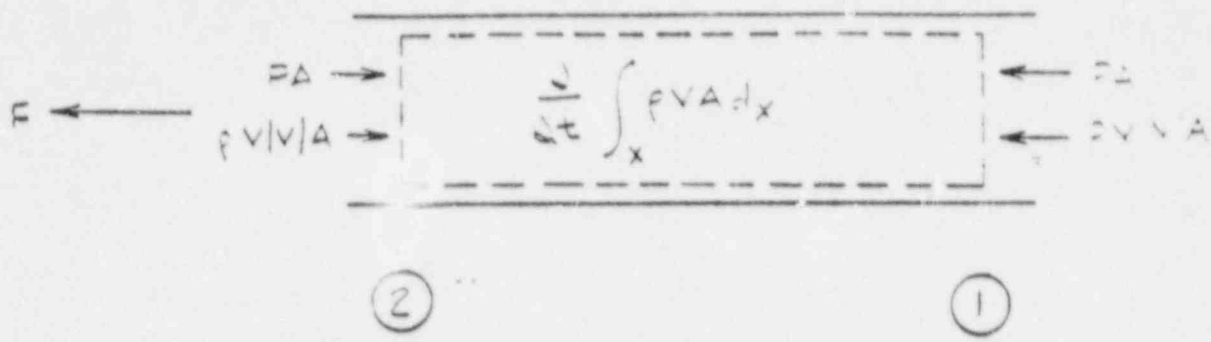


FIGURE A-1
PIPE FORCE ELEMENTS FOR SINGLE CONT. VOLUMES

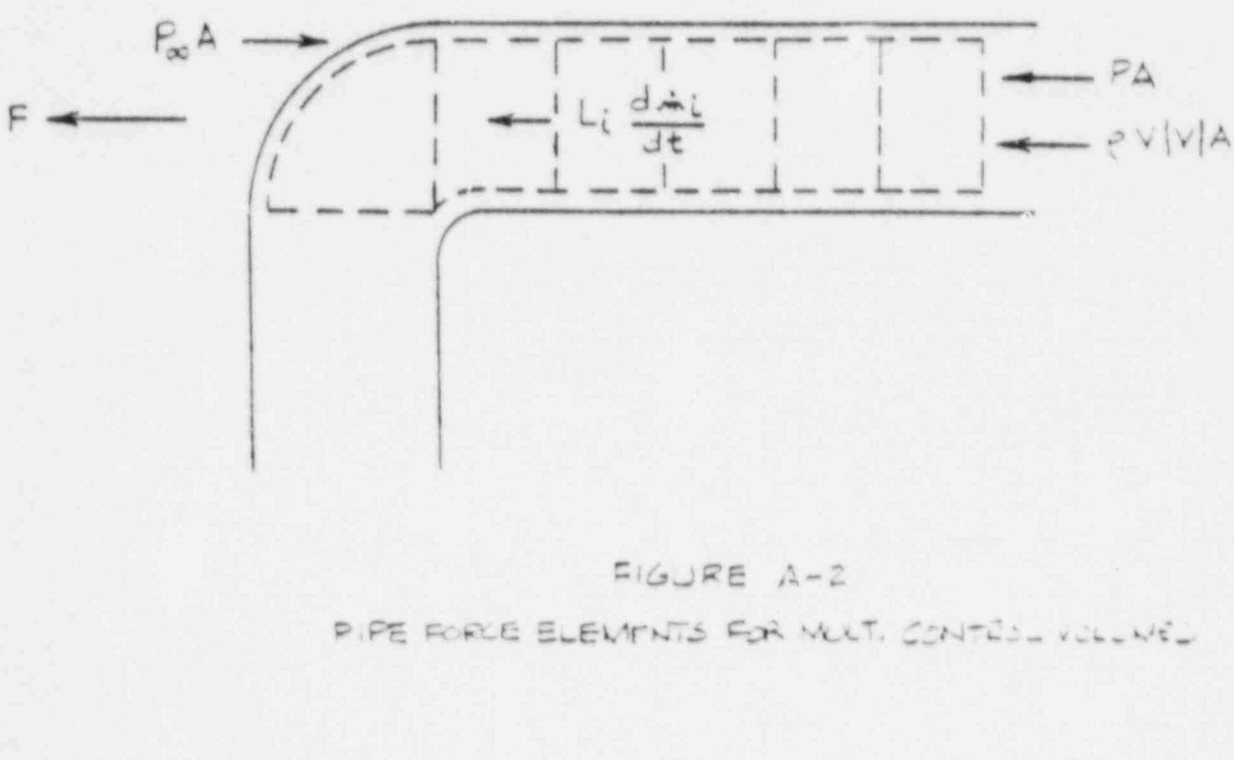


FIGURE A-2
PIPE FORCE ELEMENTS FOR MULT. CONTROL VOLUMES

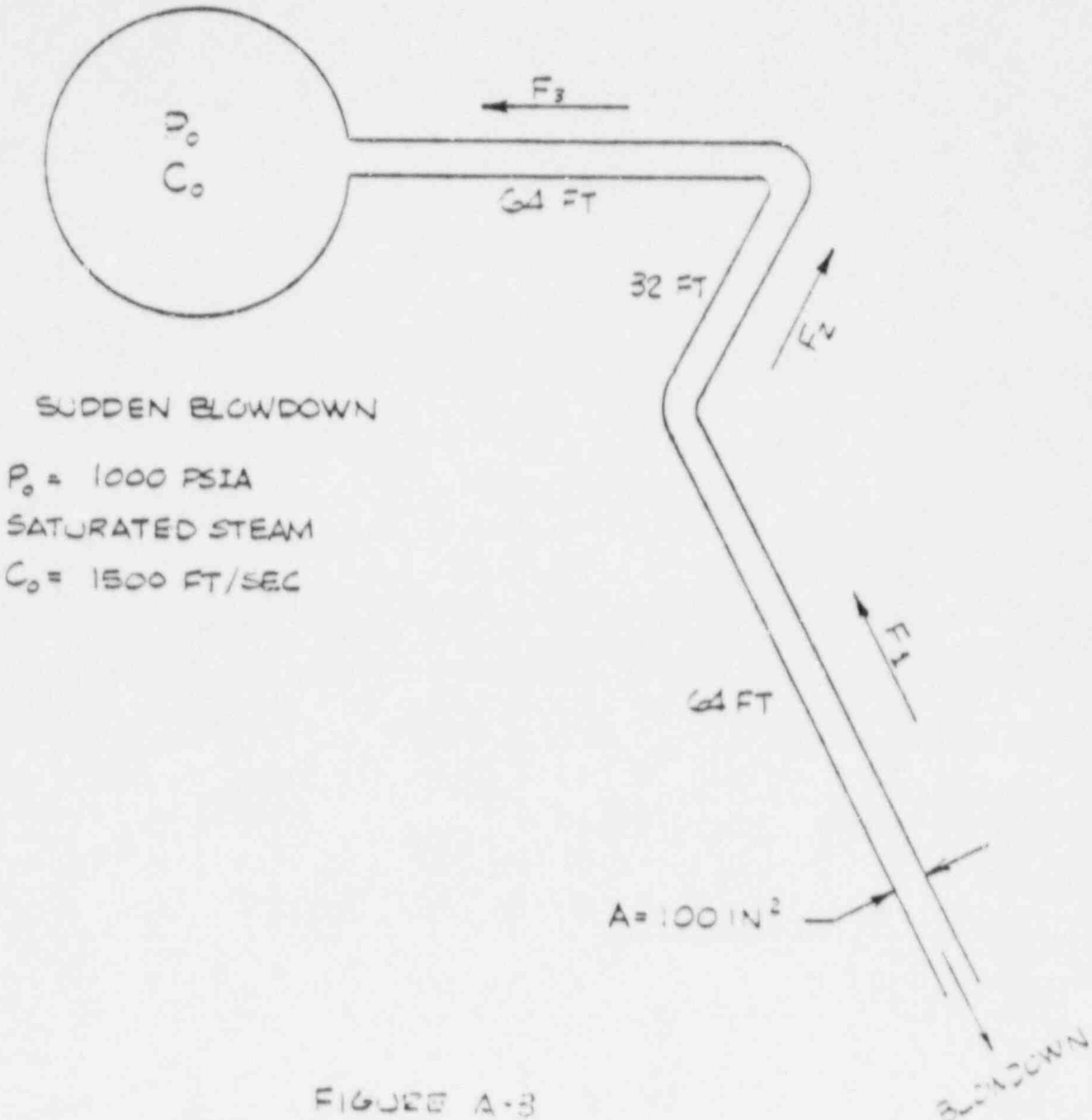


FIGURE A-8
FORCING FUNCTION PROCESSOR
VERIFICATION PROBLEM

FIG [illegible]

FORCING FUNCTIONS PROCESSOR VERIFICATION PROB.: FORCE 1

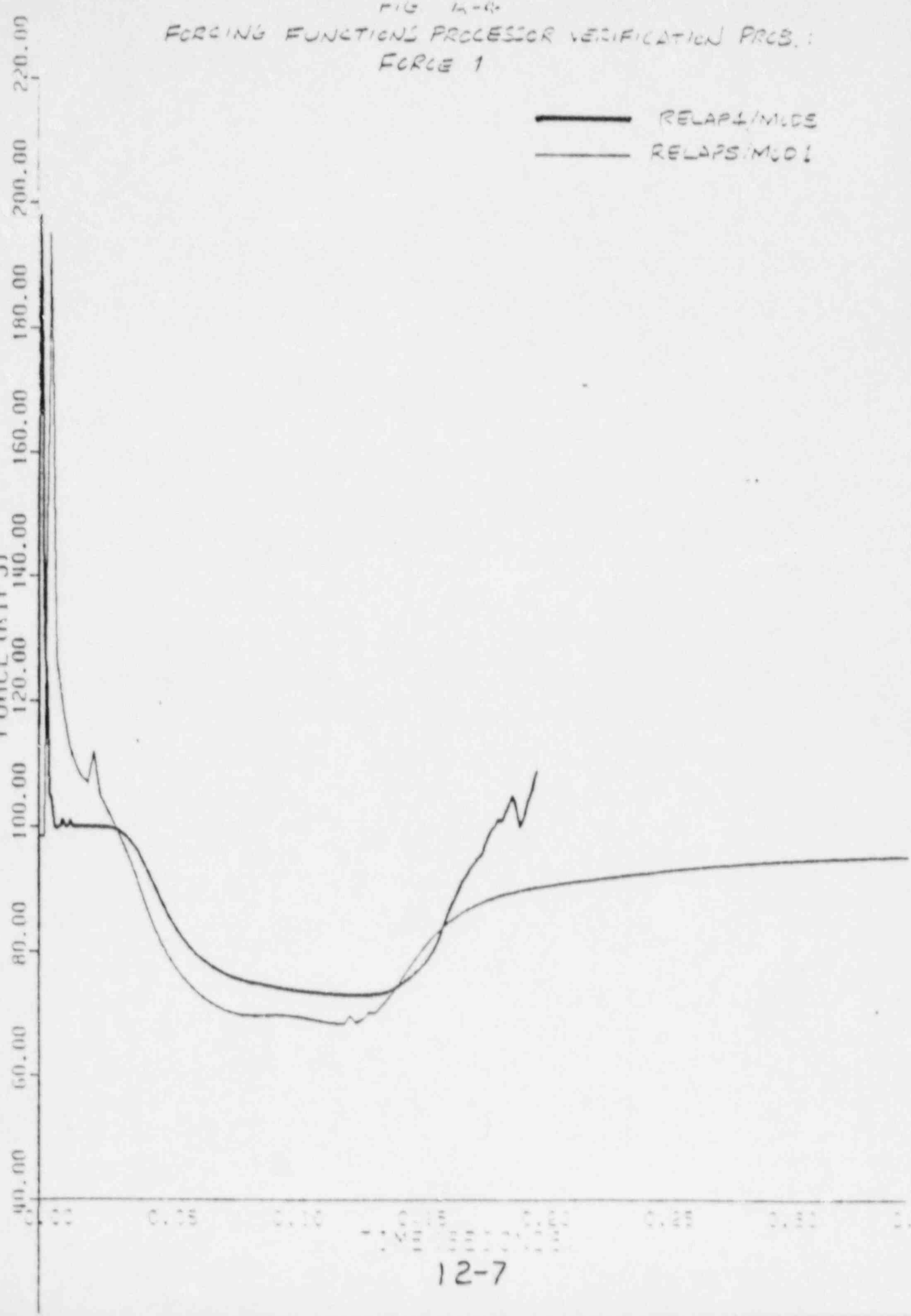


FIG. A-5

FORCING FUNCTIONS PROCESSOR VERIFICATION PROB.:

FORCE 2

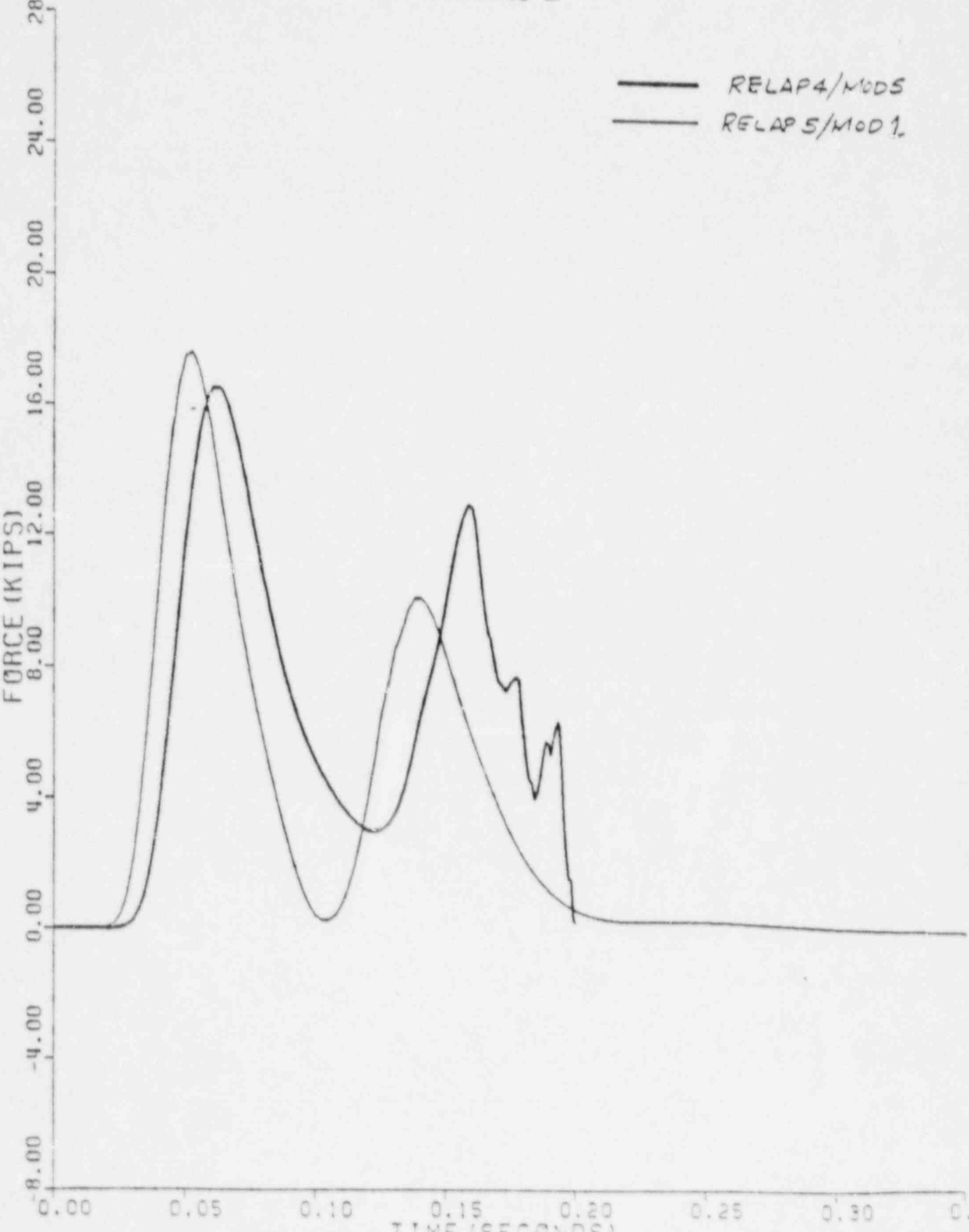
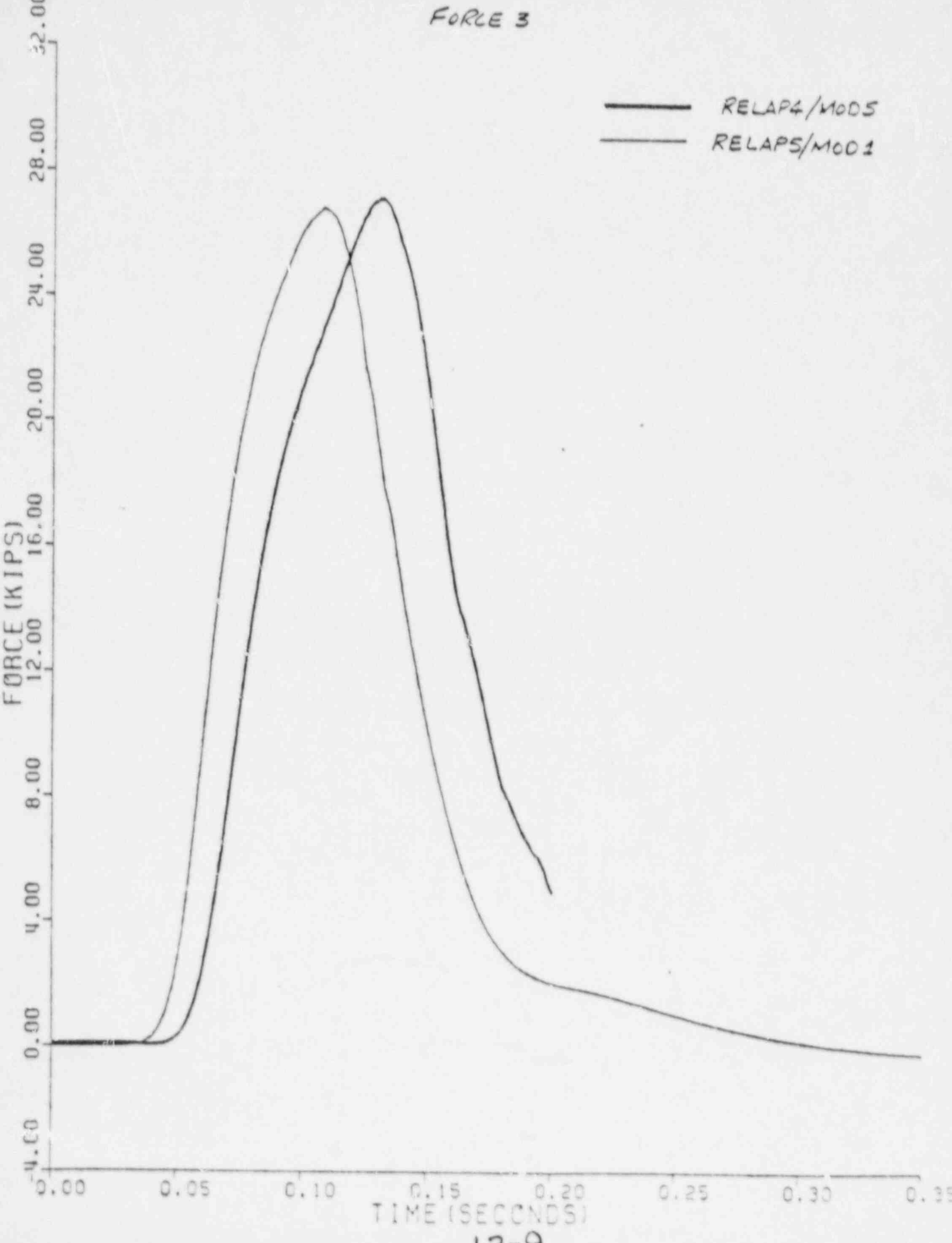


FIG A-6
FORCING FUNCTIONS PROCESSOR VERIFICATION PROB.:
FORCE 3



Appendix A

REFERENCES

1. V. H. Ransom, et.al., "RELAP5/MOD1 Code Manual," Vols. 1 and 2, EG&G Idaho, NUREG/CR-1826, March 1981.

2. F. J. Moody, "Fluid Reaction and Impingement Loads," Conference on Structural Design of Nuclear Plant Facilities, ASCE, Chicago, 1973.

3. B. R. Strong, Jr., and R. J. Baschiere, "Pipe Rupture and Steam/Water Hammer Design Loads for Dynamic Analysis of Piping Systems," Nuclear Engineering and Design, Vol. 45, 1978, p. 419-428.

4. V. R. Burke, and S. W. Webb, "RELAP4/THRUST Computer Code Manual," March 1980.

PLOT 200 14:09:11 MON 27 FEB, 1984 ... DISSPLA VER 8.2

RELAP5-FORCE VERSION 14 - GILBERT ASSOCIATES INC. NOV. 1983

EPRI/CE SRV TEST NO. 908 J.M.CAJIGAS R5F/1-14 27/02/84

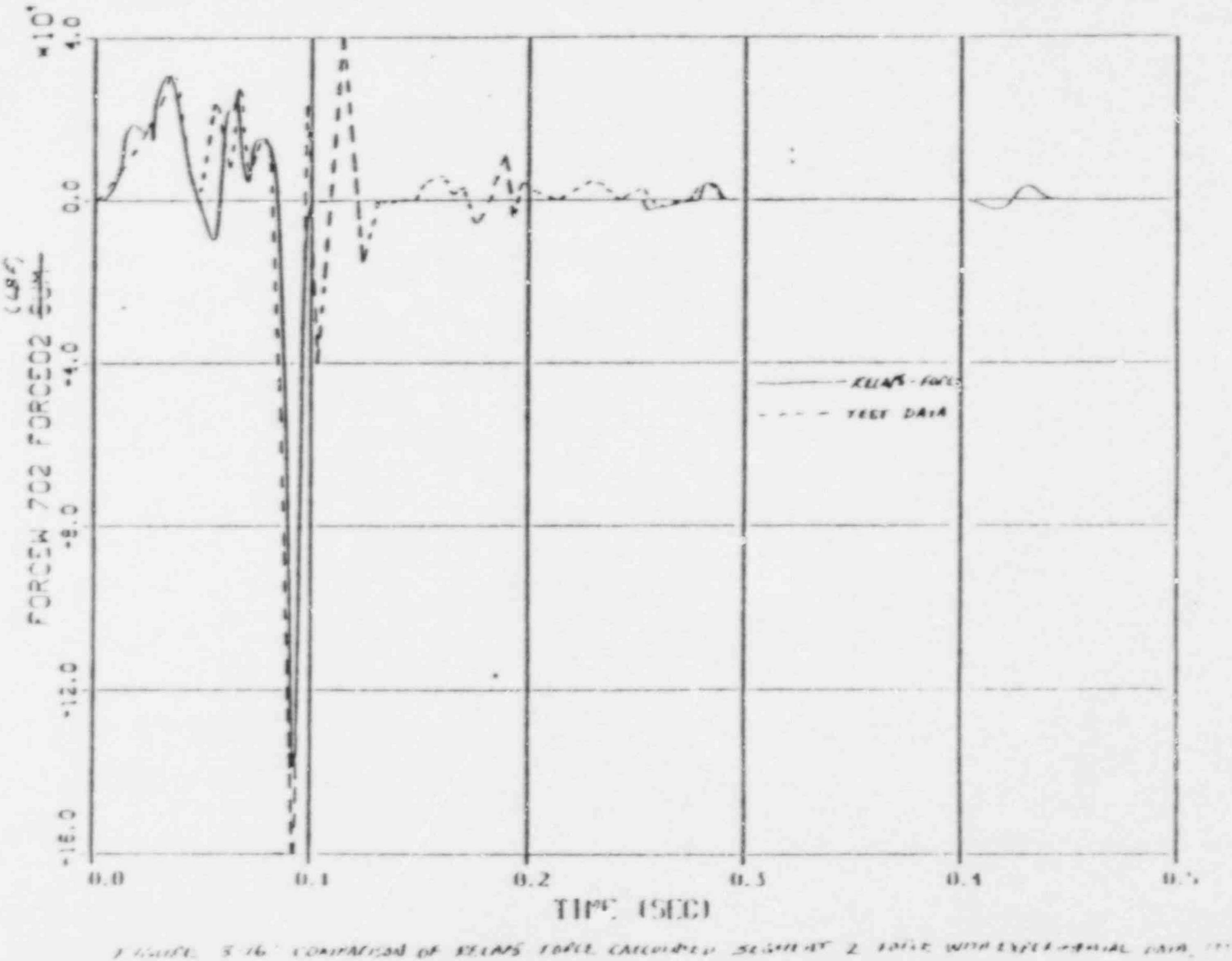
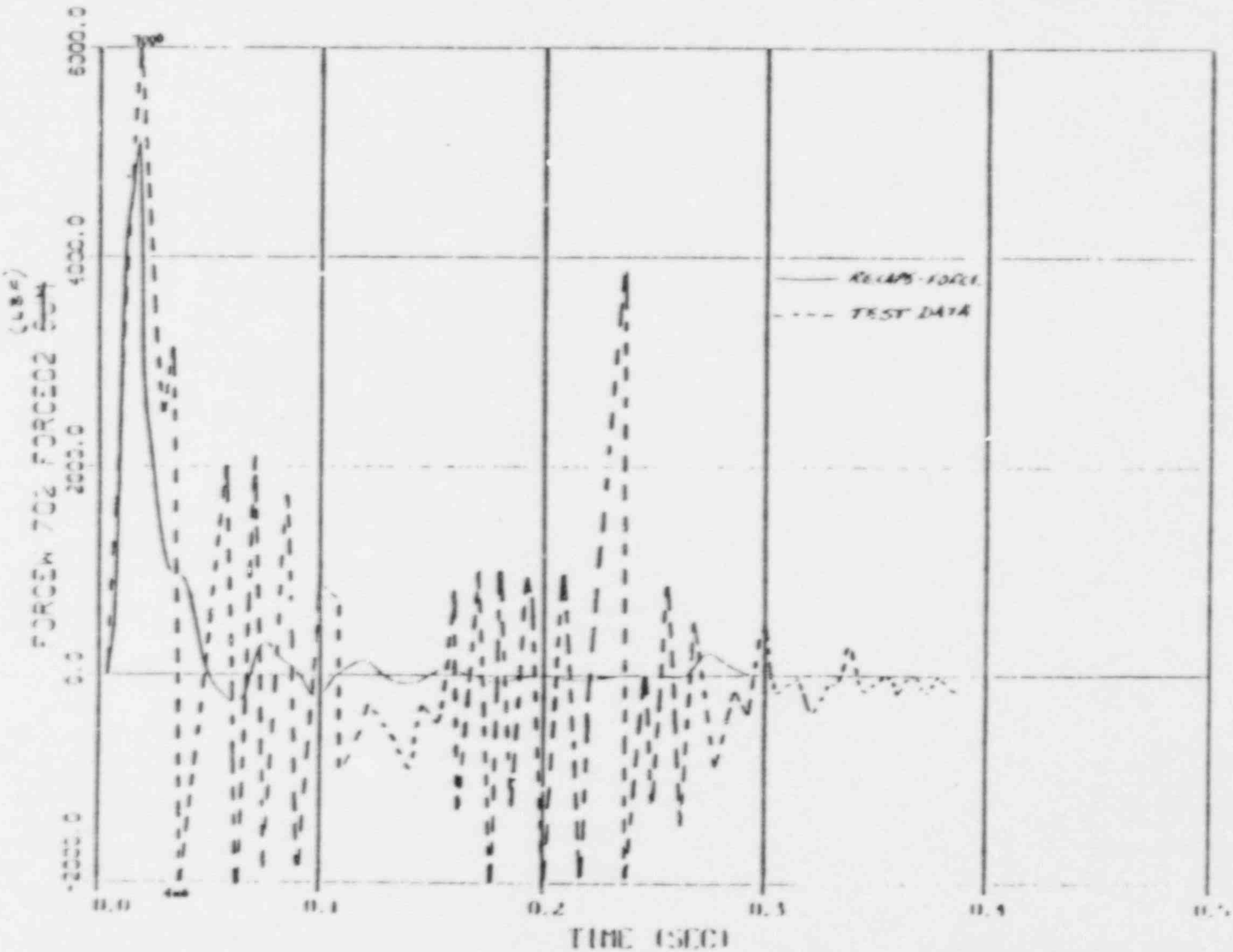


FIGURE 5-16 COMPARISON OF RELAP5 FORCE CALCULATED SEGMENT 2 FORCE WITH EXPERIMENTAL DATA, TEST 908

RELAP5-FORCE VERSION 14 - GILBERT ASSOCIATES INC. NOV. 1983

EPRI/CE SRV TEST NO 1411 J.M.CAJIGAS RSF/1-14 27/02/84



EG&G Question No. 13

Report the flow rates through the safety valves and PORV's that were assumed in the thermal hydraulic analysis. Because the ASME Code requires derating of the safety valves to 90% of actual flow capacity, the safety valve analysis should be based on a flow rate of at least 111% of the flow rating of the valve, unless another flow rate can he justified. Provide information explaining how derating of the safety valves was handled.

Response:

The derating of the SRV's and PORV's are handled as follows:

Safety Relief Valves

Design Flow Rate = 420,000 lbm/hr @ 2499.7 psig

To allow for code derating and 5% margin the flow used to determine maximum valve flow area is $\frac{420{,}000 \text{ lbm/hr}}{0.9} \times 1.05 = 490{,}000$ lbm/hr

This is a 17% increase over valve rated flow. A preliminary RELAP 5 run was made in order to determine the valve flow area required in order to achieve this flow (0.0197 ft^2).

Power Operated Relief Valves

Design Flow Rate = 210,000 lbm/hr @ 2364.7 psig.

Similar to the SRV's a 17% margin was applied. A preliminary RELAP 5 run was made in order to determine the maximum area to achieve this flow (0.0134 ft^2).

EG&G Question No. 14

The submittal does not present details of the structural analysis. Provide a report or other documentation that contains at least the following information: For the analysis involving discharge of saturated steam with a 380·F loop seal through the safety valves, identify a) the time step used in the forcing function time histories, b) the time step used in the integration solution, c) damping values used, d) the cutoff frequency if modal superposition was used, e) and the spacing between lumped masses in the structural model, f) provide rationale for the values used, g) explain how the connections to the pressurizer and relief tanks were treated in the structural model, h) identify the manufacturer and model numbers of snubbers used to support the safety valve piping (down to the relief tank) and specify the stiffnesses used in the model to represent the snubbers, i) provide a copy of the computer printout from the TPIPE and TMRPIPE analyses of the loop seal/steam discharge through the safety valves, j) also, provide clear, readable as-built drawings of the piping configuration from the pressurizer to the relief valve showing dimensions, pipe sizes and locations of pipe supports and snubbers.

Response:

Item a)

The time step used in the forcing function time history analysis is 0.0005 seconds. (See response to Question #10 for further discussion).

Item b)

The direct integration solution time step used for the GAI analysis is 0.001 seconds. This is considered to be sufficient since it allows the piping response to adequately account for a dynamic frequency of up to approximately 250 Hz. It is estimated that one cycle can adequately be modelled by four points allowing up to 250 cycles to be modelled in a one second time period.

For the TES portion of the analysis, a time step of 0.004 seconds was used which was considered to adequately model the dynamic characteristics of the RC03 piping since no dynamic effects of the blowdown could pass beyond the anchor point at the inlet to the SRV's.

Item c)

The damping valves used to formulate the damping matrix 'C' were $\alpha = \beta = 0.0$ for the GAI portion of the analysis. This yields a damping matrix of zero which is both conservative and allows for more refined future analyses if required. The damping values used in the TES portion of the analysis were $\alpha = 1.106$ and $\beta = 0.0000707$.

Item d)

The cutoff frequency is not applicable to the analyses under consideration since direct integration was used rather than modal superposition.

Item e)

The spacing between lumped masses is as indicated on the piping isometric drawings listed under Item J. In the TPIPE computer code, used for the GAI analyses, all node points are mass points so that reasonable mass point spacing is assured by appropriate and reasonable modelling techniques.

Item f)

Rationale for the above items (a - e) is described above.

Item g)

The flexibility of the pressurizer relief tank was incorporated into the piping analysis of RC01 by means of spring constants. The pressurizer nozzles (three SRV's and one PORV) were all modelled as rigid anchors.

Item h)

Pages 14-3 through 14-7 summarize support data for all supports in RC01/RC03. The manufacturer and model numbers of the snubbers are provided along with other pertinent information. The snubbers were considered rigid in the dynamic analysis of the system.

Item i)

The requested computer output is identified as follows:

Run ID	Date	Description
ATGRMLS	7/12/82	RC01 SRV Blowdown

This copy is available for review at South Carolina Electric & Gas Engineering offices in Columbia, South Carolina.

Item j)

See isometric drawings listed below:

Isometric Dwg. No.	Sheet
C-314-601	3
C-314-601	1
C-314-601	2
C-314-601	31
C-314-601	32

OVERSIZE DOCUMENT PAGE PULLED

SEE APERTURE CARDS

NUMBER OF OVERSIZE PAGES FILMED ON APERTURE CARDS 5

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Support Mark No.	Isometric No. Teledyne Analysis/ GAI Analysis	Analysis Code	Analysis Node No. Teledyne/GAI	Pipe Size	P.A. Elev.	Support Type	Snubber Size, IF Applicable	Support Class	Member/Stress Ratio	Source of Allowable
RCH-034	C-314-601-1/31	RC 01	149/H149	6	482	Spring	-	1	3x3x3/8 /0.35	NF
RCH-035	C-314-601-1/31	RC 01	145/H145	6	482	Spring	-	1	3x3x3/8 (Item 3)/0.35	NF
RCH-041	C-314-601-1/31	RC 01	1461/S146	6	482	Snubber	PSA-1.5	1	Stresses Negligible	-
RCH-042	C-314-601-1/31	RC 01	1941/S194	6	482	Snubber	PSA-1.5	1	Stresses Negligible	-
RCH-043	C-314-601-1/31	RC 01	1501/S150	6	482	Snubber	PSA-1.5	1	Stresses Negligible	-
RCH-044	C-314-601-1/31	RC 01	1951/S195	6	482	Snubber	PSA-1.5	1	Stresses Negligible	-
RCH-045	C-314-601-1/31	RC 01	1421/S142	6	480	Snubber	PSA-1.5	1	Snubber (Item1) /0.15	LCD Sheets
RCH-046	C-314-601-1/31	RC 01	1931/S193	6	480	Snubber	PSA-1.5	1	Snubber (Item 1) /0.14	LCD Sheets
RCH-047	C-314-601-1/31	RC 01	1495/S187	3	477	Snubber	PSA-1.5	1	Snubber (Item 1) /0.66	LCD Sheets
RCH-048	C-314-601-1/31	RC 01	1492/S188	3	477	Snubber	PSA-1.5	1	Snubber (Item 1) /0.83	LCD Sheets
RCH-049	C-314-601-1/31	RC 01	92/R92	6	477	Rigid	-	1	Weld (Item 1 to 3) /0.70	NF
RCH-050	C-314-601-1/31	RC 01	962/TR99	6	477	Rigid	-	1	Hilti-Kwik Bolts (Item 7) /0.87	Mfr. Catalog and IE 79-02
RCH-051	C-314-601-1/31	RC 01	192/S192	3	477	Snubber	PSA-6	1	Stresses Negligible	-
RCH-052	C-314-601-1/31	RC 01	8301/S84	3	477	Snubber	PSA-6	1	4x3x3/8 (Item 3) /0.54	NF
RCH-053	C-314-601-1/31	RC 01	8101/S831	3	477	Snubber	PSA-6	1	W4x13 (Item 3) /0.43	NF
RCH-4015	C-314-601-1/31	RC 01	1609/G117	Vlv. PCV-444B	477	Two Snubbers	PSA-1.5	1	C4x7.25 (Item 1) /0.55	NF
RCH-4017	C-314-601-1/31	RC 01	1209/G121	Vlv. 8000 B	477	Two Snubbers	PSA-1.5	1	Stresses Negligible	-
RCH-4018	C-314-601-1/31	RC 01	1371/V138	Vlv. 8000 C	479	Two Snubber	PSA-.65	1	C4x7.25 (Item 2) /0.69	NF

Note: Stress Ratios listed as "Stresses Neglibible" are less than 10% of their Allowables

Support Mark No.	Isometric No. Teledyne Analysis/ GAI Analysis	Analysis Code	Analysis Node No. Teledyne/GAI	Pipe Size	P.A. Elev.	Support Type	Snubber Size, IF Applicable	Support Class	Member/Stress Ratio	Source of Allowable
RCH-4019	C-314-601 1/31	RC 01	1331/V134	Vlv. 445 B	479	Two Snubbers	PSA-1.5	1	Stresses Negligible	-
RCH-4020	C-314-601-1/31	RC 01	8703/87	3	476	Snubber	PSA-1.5	1	Snubber (Item 2) /0.85	LCD Sheets
RCH-036	C-314-601-1/31	RC 01	1282/H128	6	478	Spring	-	NNS	Welded Beam Attch. (Item 4)/.71	LCD Sheets
RCH-037	C-314-601-1/31	RC 01	66/H66	6	472	Spring	-	NNS	Welded Beam Attch. (Item 2)/.31	LCD Sheets
RCH-038	C-314-601-1/31	RC 01	62/H62	6	472	Spring	-	NNS	Welded Beam Attch. (Item 2)/.35	LCD Sheets
RCH-039	C-314-601-1/31	RC 01	7701/H771	3	477	Spring	-	NNS	Spring Can (Item 5)/.92	LCD Sheets
RCH-040	C-314-601-1/31	RC 01	107/H107	6	472	Spring	-	NNS	Welded Beam Attch. (Item 1) /0.25	LCD Sheets
RCH-054	C-314-601-1/31	RC 01	114/R114	3	477	Rigid	-	NNS	3" x 3/4" BAR x 0'-11 1/2" (Item 1) /0.43	NF
RCH-055	C-314-601-1/31	RC 01	113/S190	3	476	Snubber	PSA-6	NNS	W4x13 (Item 4)/0.54	NF
RCH-056	C-314-601-1/31	RC 01	113/S190	3	476	Snubber	PSA-6	NNS	Hilti-Kwik Bolts/0.85	Mfr. Catalog and IE 79-02
RCH-057	C-314-601-1/31	RC 01	1082/S108	6	472	Snubber	PSA-15	NNS	Hilti-Kwik Bolts/0.87	Mfr. Catalog and IE 79-02
RCH-058	C-314-601-1/31	RC 01	1891/S108	6	472	Snubber	PSA-15	NNS	Hilti-Kwik Bolts/0.39	Mfr. Catalog and IE 79-02
RCH-059	C-314-601-1/31	RC 01	1061/S106	6	472	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-060	C-314-601-1/31	RC 01	1831/S101	6	472	Snubber	PSA-6	NNS	Stresses Negligible	-
RCH-061	C-314-601-1/31	RC 01	1822/S182	6	472	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-062	C-314-601-1/31	RC 01	76/S76	6	476	Snubber	PSA-15	NNS	Snubber (Item 1)/0.78	LCD Sheets
RCH-063	C-314-601-1/31	RC 01	1314/S131	3	479	Snubber	PSA-6	NNS	Stresses Negligible	-
RCH-064	C-314-601-1/31	RC 01	1312/S131	3	478	Snubber	PSA-6	NNS	Stresses Negligible	-

Note: Stress Ratios listed as "Stresses Neglibible" are less than [illegible]% of their Allowables

Support Mark No.	Isometric No. Teledyne Analysis/ GAI Analysis	Analysis Code	Analysis Node No. Teldyne/GAI	Pipe Size	P.A. Elev.	Support Type	Snubber Size, IF Applicable	Support Class	Member/Stress Ratio	Source of Allowable
RCH-065	C-314-601-1/31	RC 01	1273/S126	6	477	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-066	C-314-601-1/31	RC 01	1031/S103	6	472	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-067	C-314-601-1/31	RC 01	6701/S67	6	472	Snubber	PSA-6	NNS	Weld (Item 2 to Item 3)/0.50	NF
RCH-068	C-314-601-1/31	RC 01	65/S64	6	472	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-069	C-314-601-1/31	RC 01	6301/S63	6	472	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-071	C-314-601-1/31	RC 01	6101/S61	6	472	Snubber	PSA-6	NNS	Snubber (Item 1)/0.62	LCD Sheets
RCH-391	C-314-601-1/31	RC 01	9721/S71	6	474	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-392	C-314-601-1/31	RC 01	9171/S171	6	471	Snubber	PSA-15	NNS	Hilti-Kwik Bolts for Item 5/0.83	Mfr. Catalog and IE 79-02
RCH-397	C-314-601-1/31	RC 01	103/S103	6	472	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-398	C-314-601-1/31	RC 01	63/S63	6	472	Snubber	PSA-15	NNS	Hilti-Kwik Bolts (Item 7)/0.63	Mfr. Catalog and IE 79-02
RCH-4000	C-314-601-1/31	RC 01	1003/S111	6	475	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-4001	C-314-601-1/31	RC 01	1072/S189	6	472	Snubber	PSA-6	NNS	Stresses Negligible	-
RCH-4002	C-314-601-1/31	RC 01	5602/S571	6	470	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-4003	C-314-601-1/31	RC 01	1271/S126	6	476	Snubber	PSA-15	NNS	Stresses Negligible	-
RCH-032	C-314-601-2/32	RC 01	701/H45	6	472	Spring	-	NNS	Welding Lug (Item 2)/0.27	LCD Sheets
RCH-033	C-314-601-2/32	RC 01	1601/H163	6	472	Spring	-	NNS	Weldless Eye Nut (Item 8)/0.32	LCD Sheets
RCH-070	C-314-601-2/32	RC 01	1621/S162	6	472	Snubber	PSA-6	NNS	Snubber (Item 1)0.58	LCD Sheets
RCH-085	C-314-601-2/32	RC 01	7025/S160	6	472	Snubber	PSA-15	NNS	Hilti-Kwik Bolts (Item 4)/0.73	Mfr. Catalog and IE 79-02
RCH-086	C-314-601-2/32	RC 01	3004/S159	6	473	Snubber	PSA-6	NNS	Snubber (Item 1)/0.14	LCD Sheets

Note: Stress Ratios listed as "Stresses Neglibible" are less than 10% of their Allowables

Support Mark No.	Isometric No. Teledyne Analysis/ GAI Analysis	Analysis Code	Analysis Node No. Teledyne/GAI	Pipe Size	P.A. Elev.	Support Type	Snubber Size, IF Applicable	Support Class	Member/Stress Ratio	Source of Allowable
RCH-087	C-314-601-2/32	RC 01	1672/S167	6	467	Snubber	PSA-6	NNS	Snubber (Item 1)/0.74	LCD Sheets
RCH-088	C-314-601-2/32	RC 01	1674/S167	6	467	Snubber	PSA-6	NNS	Snubber (Item 1)/0.92	LCD Sheets
RCH-089	C-314-601/2/32	RC 01	1612/S161	6	472	Snubber	PSA-15	NNS	Hilti-Kwik Bolts (Item 4)/0.98	Mfr. Catalog and IE 79-02
RCH-090	C-314-601-2/32	RC 01	1603/S164	6	472	Snubber	PSA-15	NNS	Attch. Pad (Item 3)/0.81	NF
RCH-091	C-314-601-2/32	RC 01	2001/S165	6	473	Snubber	PSA-6	NNS	Snubber (Item 1)/0.43	LCD Sheets
RCH-092	C-314-601-2/32	RC 01	7722/S196	6	474	Snubber	PSA-6	NNS	Snubber (Item 2)/0.44	LCD Sheets
RCH-093	C-314-601-2,3/32	RC 01,03	23/A23	6	480	Anchor	-	1	Weld of Item 8/0.82	NF
RCH-094	C-314-601-2,3/32	RC 01,03	1/A1	6	480	Anchor	-	1	Weld of Item 8/0.82	NF
RCH-095	C-314-601-2,3/32	RC 01,03	37/A37	6	480	Anchor	-	1	Weld of Item 8/0.82	NF
RCH-393	C-314-601-2/32	RC 01	704/S199	6	472	Snubber	PSA-15	NNS	Hilti-Kwik Bolts (Item 5)/0.84	Mfr. Catalog and IE 79-02
RCH-394	C-314-601-2/32	RC 01	9198/S198	6	473	Snubber	PSA-15	NNS	Hilti-Kwik Bolts (Item 8)/0.77	Mfr. Catalog and IE 79-02
RCH-028	C-314-601-2/32	RC 01	173/H173	12	458	Spring	-	NNS	Spring Can/.64	LCD Sheets
RCH-029	C-314-601-2/32	RC 01	43/H44	12	450	Spring	-	NNS	Welded Attch. (Item 5)/0.61	LCD Sheets
RCH-030	C-314-601-2/32	RC 01	181/R181	12	428	Spring	-	NNS	Welded Lug (Item 1)/0.56	LCD Sheets
RCH-031	C-314-601-2/32	RC 01	169/H169	12	465	Spring	-	NNS	Beam Attch. (Item 8)/0.97	LCD Sheets
RCH-072	C-314-601-2/32	RC 01	1802/S181	12	428	Snubber	PSA-15	NNS	Trunion (Item 4)/0.81	NF
RCH-073	C-314-601-2/32	RC 01	1793/S180	12	428	Snubber	PSA-6	NNS	Snubber (Item 1)/0.84	LCD Sheets

Note: Stress Ratios listed as "Stresses Neglibible" are less than 10% of their Allowables

Support Mark No.	Isometric No. Teledyne Analysis/ GAI Analysis	Analysis Code	Analysis Node No. Teledyne/GAI	Pipe Size	P.A. Elev.	Support Type	Snubber Size, IF Applicable	Support Class	Member/Stress Ratio	Source of Allowable
RCH-074	C-314-601-2/32	RC01	1791/S179	12	428	Snubber	PSA-15	NNS	Trunion (Item 4)/0.95	NF
RCH-075	C-314-601-2/32	RC01	1783/178	12	430	Snubber	PSA-50	NNS	Hilti-Kwik Bolts (Item 4)/0.85	Mfr. Catalog and IE 79-02
RCH-076	C-314-601-2/32	RC01	1781/178	12	430	Snubber	PSA-50	NNS	Hilti-Kwik (Item 4)/0.98	Mfr. Catalog and IE 79-02
RCH-077	C-314-601-2/32	RC01	1861/S186	12	439	Snubber	PSA-15	NNS	Weld Item 1 to plate /0.74	NF
RCH-078	C-314-601-2/32	RC01	1771/S177	12	448	Snubber	PSA-50	NNS	Hilti-Kwik Bolts (Item 4)/0.79	Mfr. Catalog and IE 79-02
RCH-079	C-314-601-2/32	RC01	1761/S176	12	450	Snubber	PSA-50	NNS	Hilti-Kwik Bolts (Item 11)/0.83	Mfr. Catalog and IE 79-02
RCH-080	C-314-601-2/32	RC01	1751/S175	12	452	Snubber	PSA-50	NNS	Hilti-Kwik Bolts (Item 3)/0.50	Mfr. Catalog and IE 79-02
RCH-081	C-314-601-2/32	RC01	1741/S174	12	453	Snubber	PSA-50	NNS	Hilti-Kwik Bolts (Item 4)/0.49	Mfr. Catalog and IE 79-02
RCH-082	C-314-601-2/32	RC01	1721/S172	12	458	Snubber	PSA-50	NNS	Hilti-Kwik Bolts (Item 3)/0.66	Mfr. Catalog and IE 79-02
RCH-083	C-314-601-2/32	RC01	170/S170	12	462	Snubber	PSA-15	NNS	Snubber (Item 1)/0.72	LCD Sheets
RCH-084	C-314-601-2/32	RC01	1301/S13	12	469	Snubber	PSA-50	NNS	Hilti-Kwik Bolts (Item 8)/0.86	Mfr. Catalog and IE 79-02
RCH-390	C-314-601-2/32	RC01	40/S40	12	456	Two Snubbers	PSA-6	NNS	Hilti-Kwik Bolts (Item 8)/0.85	Mfr. Catalog and IE 79-02
RCH-395	C-314-601-2/32	RC01	1801/S181	12	427	Snubber	PSA-6	NNS	Stresses Negligible	-
RCH-396	C-314-601-2/32	RC-01	NA/R53	12	427	Rigid	-	NNS	Hilti-Kwik Bolt (Item 1)/0.63	Mfr. Catalog and IE 79-02
RCH-399	C-314-601-2/32	RC-01	13/13	12	464	Snubber	PSA-50	NNS	Hilti-Kwik Bolt (Item 2)/0.81	Mfr. Catalog and IE 79-02

Note: Stress Ratios listed as "Stresses Neglibible" are less than 10% of their Allowables

EG&G Question No. 15

The submittal states that the structural analysis on the piping system was performed using the TPIPE and TMRPIPE computer codes. It further states that these programs have had application on numerous projects in the nuclear industry. Provide verification that these programs have produced accurate results for problems similar to a valve actuation in the safety valve/PORV piping system. Explain whether the dynamic piping response was obtained using the direct integration, modal superposition, or other solution technique.

Response:

1. TPIPE:

   The TPIPE computer code has an extensive testing program to execute all logical options designed into the code. Twenty-eight example problems or benchmarks were developed to test all the options or combination of options available.(1)

   Three independent and widely accepted computer programs, PIPESD, PISOL, and SAPIV, were employed to prove the accuracy of the TPIPE results. Each benchmark that considered static or dynamic analysis was modeled and executed with one of the three programs. The ensuing results were then compared with the corresponding TPIPE results to confirm accuracy.

   A benchmark problem involving time history (direct integration) analysis of time varying nodal loads similar to those resulting from a valve actuation in the safety valve/PORV piping is run as part of the TPIPE testing program.

   (See response to Question #14 for a description of the methods used in obtaining the dynamic piping response).

   Later program revisions are verified by comparison with previously verified benchmark outputs.

2. TMRPIPE:

   A letter from TES which illustrates the NRC acceptance of TMRSAP is attached (Attachment pages 15-2 and 15-3). TMRSAP is the analytical sub-program of TMRPIPE which performs the stress calculations.

(1) Reference: 'TPIPE Verification Manual', PMP Systems Engineering, Inc., 500 Sansome Street, San Francisco, California, Revision 1, October 1977.

TELEDYNE
ENGINEERING SERVICES

130 SECOND AVENUE
WALTHAM MASSACHUSETTS 02254
(617) 890-3350 TWX 710-324-7508

May 14, 1986
865-051

Mr. Alfred Hoffert
Gilbert Associates, Incorporated
P. O. Box 1498
Reading, PA 19603

Reference: **NRC Inspection Report and Docket No. 99900513/85-01**
Dated July 11, 1985

Dear Mr. Hoffert:

In response to your request that we provide documentation of TMRSAP verification, the following is offered.

On January 7-11, 1985, NRC personnel conducted the Referenced Inspection at our facility in Waltham, Massachusetts. The purpose of this inspection was to review our Quality Assurance Program in the areas of computer code verification, computer code error handling procedures, and pipe support design calculations.

Regarding computer program verification, the following is an excerpt from the Report:

"The development and verification of the computer program TMRSAP, which is used by TES in the design of safety-related items was reviewed during this inspection. Technical Engineering Procedures TEP-1-005, Application Computer Program Development, was reviewed and utilized throughout the inspection of TMRSAP.

The computer code TMRSAP, which was developed internally by TES, is used for static and dynamic analysis of linear piping systems. It employs a finite element solution technique with a library consisting of curved and straight pipe elements, and a boundary element for simulation of pipe restraints. TMRSAP provides capability for analysis of such static loading as deadweight, thermal, and pressure elongation loadings. Capabilities for dynamic analysis include response spectrum and time history (both modal and direct) analysis. Solution methods include Gaussian elimination for static solutions, and determinant search or subspace iteration for the modal dynamic solutions. Direct integration is performed with the Wilson-θ-method.

15-2

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TES verified TMRSAP by a comparison of the output of 22 verification problem solutions with either the results of hand calculations or the output of other computer codes (STARDYNE, EPIPE, ANSYS, and ADLPIPE). During this inspection all verification problems were reviewed. Although the verification of this code was done according to a general design control procedure (Section 3.0 of the TES Quality Assurance Manual), it was found to meet the requirements of the latest TES procedure controlling computer code verification (TEP-1-005), with one exception. The exception was that the source code listing and computation outputs were not included in the verification manual. However, the computation output includes a source code listing that was clearly identified in the verification manual and was readily available at the TES office. No violations or nonconformances were identified during this part of the inspection."

Please call if you have any questions regarding the foregoing.

Very truly yours,

TELEDYNE ENGINEERING SERVICES

R. H. Berks

Richard H. Berks
Principal Engineer

RHB:slg

EG&G Question No. 16

The submittal states that pressure oscillations in the safety valve inlet piping were reported by EPRI for some fluid conditions and upstream piping configurations. According to EPRI results these oscillations commonly occurred during passage of loop seal water and were in the 170-260 H_z frequency range. The submittal states that the oscillations have been evaluated and that stresses are within code allowable for the V. C. Summer plant. It is not clear though whether this evaluation reflects the fact that the pressure oscillations could excite high frequency vibration modes in the piping causing significant bending moments in the inlet piping. Show that the bending moments caused by this dynamic response do not exceed the allowable bending moment. Provide the referenced report "Pressure Oscillations in Safety Valve Inlet Piping", EPRI, March 17, 1982.

Response:

Based on the attached letter "Pressure Oscillations in Safety Valve Inlet Piping" (pages 16-2 and 16-3), pressure oscillations observed during EPRI testing of safety relief valves (SRV) resulted in a peak pressure of 5000 psia in the inlet piping. This peak pressure was evaluated in the SRV piping by considering it to occur coincident with the specified emergency and faulted (Level C and D) bending moments. For the V. C. Summer configuration, no blowdown piping loads are transmitted to the safety relief piping upstream of the valve due to the anchor located at the valve inlet flange. No additional moment loadings due to high frequency vibration modes induced by the pressure oscillations were considered in the analysis. For the emergency condition, the maximum primary stress intensity of 11949 psi occurs at the analysis point 25 (elbow in loop C), well below the allowable of 32000 psi. For the faulted condition, the maximum primary stress intensity of 12489 psi occurs at analysis point 25 (elbow in loop A), again well below the faulted allowable of 48240 psi. The above information is in the attached calculation on pages 16-4 through 16-8.

ELECTRIC POWER RESEARCH INSTITUTE

EPRI

March 17, 1982.

TO: UTILITY TECHNICAL CONTACTS

SUBJECT: PRESSURE OSCILLATIONS IN SAFETY VALVE INLET PIPING

As has been noted in previous communications, spring-loaded safety valve tests at Combustion Engineering with loop seals and certain liquid inlet conditions resulted in high amplitude pressure oscillations. These oscillations were observed just upstream of the valve and appear to have been caused by water-hammer induced by valve flutter or chatter. This letter transmits a final data package on this subject and also summarized analyses performed by EPRI contractors.

On two previous occasions EPRI has transmitted data packages on this subject. The enclosed package contains more information and is considered to be the final version.

In order to better understand the test data, EPRI contractors have performed several analyses. In the 1400 series tests, the inlet pipe was instrumented with axial and transverse strain gages as well as pressure transducers. S. Levy incorporated and Combustion Engineering have analyzed this data from run 1406 (a 6M6 cold loop seal test) to determine the internal pressure required to produce the measured strain. Based on the hoop strain data, both contractors have determined that the peak internal pressure at the strain gage location is about 5000 psia. The S. Levy results show a peak to peak oscillation of 4750 psi about a mean of 2625 psia. Examination of the axial strain data gives a higher pressure (the S. Levy result is 6000 psia) but both S. Levy and CE agree that this data could be complicated by axial motion of the valve-piping assembly. For this same run, pressure transducers PT105 and PT12 showed peak pressure of 7000 and 8600 respectively. Continuum Dynamics and Combustion Engineering have investigated possible signal amplification in the pressure sensing lines. Both have concluded that the sensing lines are likely to amplify the signals: i.e., the amplitude of the pressure oscillation at the transducer is significantly larger than the pressure in the inlet pipe. This means that the data for PT12 and PT105 cannot be interpreted as the actual pipe pressure. It is our best judgment at this time that the actual peak pressure at the strain gage location is 5000 psia which is based on the

16-2

Headquarters 3412 Hillview Avenue Post Office Box 10412 Palo Alto CA 94303 (415) 855-2000

hoop strain measurement. However, there may be an axial pressure variation along the pipe axis and it is possible that the pressure is higher at other locations. Continuum Dynamics is evaluating this axial variation and we will notify you of the results.

If there are questions, please call.

Sincerely,

A.J. Whe

Anthony J. Wheeler
Project Manager
Safety and Analysis Department

AJW/11
Enc.

5.0 Pressure Oscillation Effects:

The effect of the pressure oscillations occuring during safety relief valve operation is evaluated by considering the peak pressure of approximately 5000 psia occuring in the piping upstream of the SRV (RC03 piping).

6.0 Consideration of Level C Service Limits:

6.1 Permissible Pressure

When Level C service limits are specified, the permissible pressure shall not exceed the pressure Pa, calculated in accordance with equation (3) of NB-3641.1, by more than 50%.

EQ(3) $$Pa = \frac{2\ Sm\ t}{(Do - 2y\ t)}$$

Where: Size = 6" NPS

Schedule = 160

Material = SA-376 Type 304

Pa = Allowable working pressure of pipe

Sm = Maximum allowable stress intensity for the material at design temperature.

(16080 psi at 680°F)

t = Specified wall thickness minus any allowance

= 0.718 in.nom.X 0.875 = 0.628"

(Manufacturer's tolerance of approx. 12½%)

D. = Outside diameter = 6.625"

y = o.4

16-4

RC01 2.12.18 12 22.

Gilbert Associates, Inc. Reading, Pennsylvania CALCULATION	SUBJECT EVALUATION OF PRESSURE OSCILLATIONS IN RC03 CLASS 1 PIPING		V. C. SUMMER		PAGE 5 OF 15 PAGES
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	MICROFILMED		M		
	ORIGINATOR				
	DATE		11/5/8		

Thus: $Pa = \frac{2\ (16080)\ (.628)}{6.625 - 2X0.4X0.628} = 3300$ psi

Allowable Pressure is 150% of Pa

$= (1.5)\ (3300) = 4950$ psi

This is considered to be within acceptable limits because of the following:

(1) The 5000 psia pressure oscillation is an approximate result which based on experimental strain gage results.

(2) The calculated permissible pressure is within 1% of the pressure oscillation and is well within engineering tolerance. 2

(3) The actual working pressure within the pressurizer during SRV operation is much lower than the 5000 psia oscillation. In addition, the pressure oscillations have only been identified near the valve itself.

(4) The allowable stress is evaluated at a design temperature of $680^{o}F$.

(5) The actual stresses calculated for both the emergency and faulted conditions are well below the allowable limits.

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Gilbert Associates, Inc. Reading, Pennsylvania CALCULATION	SUBJECT EVALUATION OF PRESSURE OSCILLATIONS IN RC03 CLASS 1 PIPING		CLSID V. C. SUMMER		PAGE 6 OF 15 PAGES
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	ORIGINATOR				
	DATE				

6.2 Analysis of Piping Components

Under the emergency loading condition for which the level C stress limits are specified, the coincident pressure and moments resulting in maximum calculated stresses are evaluated as follows. The allowable stress to be used for this condition is 2.25Sm but not greater than 1.8Sy, thus:

EQ(9) $$B_1 \frac{PDo}{2t} + B_2 \frac{Do\ Mi}{2I} < \text{The lesser of } 2.25\ Sm \text{ or } 1.8\ Sy$$

Where the quantities are defined as:

B_1, B_2 = Primary stress indices

P = Peak pressure (psi)

Mi = Resultant deadweight moment (in-lb)

I = Moment of inertia (in^4)

As per the EPRI load combinations, the peak pressure oscillation of 5000 psia is considered with the sustained deadweight moments obtained from Ref. (4). Table I lists these moments for all component points in each of the three SRV loops. In order to evaluate the above equation, the maximum moments from each of the three loops are enveloped and the combined stresses calculated.

 GAI 446 12/81

6.3 Results

The results of this evaluation are presented in Table II. The maximum primary stress intensity for the RC03 piping is 11949 psi occuring at point 25 (elbow of loop C). This stress is well below the allowable stress of 1.8Sy = 32000 psi evaluated at 680°F. (2.25Sm = 36180 psi)

7.0 Faulted Conditions (Level D Service Limits):

The faulted condition load combinations are presented in Table 1 of Appendix A. Safe shutdown earthquake seismic moments equal to 1.5 X OBE are used for the faulted condition evaluation and are obtained from Ref. (4). There are no additional mechanical loadings due to design basis accident/LOCA or main steam line break specified for this system. Table III lists combined deadweight plus seismic moments for all component points in each of the three SRV loops. The same equation (9) is evaluated with the peak pressure oscillation of 5000 psia and the enveloped moments of each of the three loops. /2\

7.1 Results:

The results of this evaluation are presented in Table IV. The maximum primary stress intensity for the RC03 piping is 12489 psi occuring at point 25 (elbow of loop A). This stress is well below the faulted allowable of 3.0Sm = 48240 psi evaluated at 680°F.

8.0 Conclusions:

The piping components meet both emergency and faulted stress limits under the application of the 5000 psia pressure oscillation in the RC03 piping system. 2

EG&G Question No 17

The submittal presents the load combinations that were considered in the piping analysis. The combinations listed consider all those that are recommended in the report EPRI PWR Safety and Relief Valve Test Program Guide for Application of Valve Test Program Results to Plant-Specific Evaluations except for an upset condition in the Class 1 piping in which PORV discharge transient, OBE, and normal loads are combined. Provide justification for not considering this load combination in the analysis.

Response:

During the development of the ASME, Section III, Class NB, Design Specification for the Class 1 pipe, July 1981, the load combinations for the Upset Primary condition were given as:

Design Pressure + Deadweight + OBE

Pressure during VTC + Deadweight + VTC*

The preliminary EPRI issue of March, 1981 had these loading combinations from valve lift.

There are certain plant transients which are postulated by the NSSS to result in PORV lifts. These plant transients are also postulated by the NSSS to be induced by an OBE. Since the VTC loads are of a very short duration, less than 0.4 seconds, and the OBE loads are of longer duration but multi-frequency, it was judged not to combine these loads for the UPSET Primary Case but to qualify the piping to sustain the loading as given in the Design Specification

Note that the Design Specification requires for the Fatigue Condition, that Normal Transients, or Upset Transients, or Test Transients + Weight + OBE + VTC loads be combined and equations 10, 11, 12, 13, and 14 of the ASME Code be satisfied. Therefore, the combination of OBE and VTC is considered in the analysis and more conservatively than the EPRI loading combinations on UPSET conditions.

* Valve Thrust Conditions from valve lift

EG&G Question No. 18

The submittal states that piping stresses and support loads in both the upstream and downstream portions of the safety valve piping system are acceptable. It also states that piping stresses and support loads in the PORV system are acceptable. Provide a numerical comparison between the calculated and allowable stresses for the piping and supports for these systems to verify this conclusion. Also, identify the codes and standards from which the allowable piping stresses and support loads were obtained.

Response:

A. Piping downstream of PORV's and SRV's (Non-Safety)

Attached (see page 18-3) is a copy of the stress summary from computer output ATGRNGS (7/12/82), RC01 Post Processor. This post processor combines stresses from the appropriate load cases to form the proper stress summary for each plant condition. It then scans the stress ratios [(Actual Stress)/(Allowable Stress)] of every node within a load case and prints the maximum stress ratio for each load case. No nodes exceed the allowable ratio of 1.00.

B. Piping Upstream of PORV's and SRV's (Class 1)

TES reports TR-4813-22 and TR-4813-24 address the stresses in the Class 1 portion of piping upstream of the PORV's and SRV's respectively. Pages 18-4 through 18-6 summarize the stresses and allowable stresses from these reports. These reports conclude that all stresses are acceptable.

In the course of performing the calculations for these Stress Reports it became evident that two critical piping components required more detailed analysis. These two components were the 3" Sch. 166, Pipe-to-Valve Tapered Transition Joints, and the 6" - 1500# Welding Neck Raised Face Flanges of the Pressurizer Safety Relief Valves. The in-depth evaluations of these components are addressed in TES Report TR-4813-23 and TR-4813-25 respectively.

The subject components are qualified to the applicable design codes and code cases as follows:

- o Tapered Transition Joint - largest total fatigue usage factor is 0.880 where the allowable is 1.0.
- o Weld Neck Flange - largest total fatigue usage factor is 0.448 where the allowable is 1.0.

C. Applicable Codes and Standards

The conventional portion of the piping utilized upgraded analysis rules and was qualified as per Subsection NC of the 1971 ASME code up to and including the Summer 1973 addenda.

The Class 1 portion of piping was qualified as per Subsection NB of the 1977 ASME code through Summer 1979 Addenda inclusive, and ASME code case N-196-1.

D. Supports

For a comparison of the support loads to allowables, see response to Question No. 14, pages 14-3 through 14-6.

GILBERT ASSO. INC.   V.C.SUMMER NUCLEAR STATION #1 PWR/SRV PIPING
READING, PA. USA.   RC01/EPRI FROM S/RV TO PRES. REL. TANK
########## EPRI/RELAP5/AEA SRV(LS380)+PORV(PV902) ######
NPRV-T450/LS-SRV-T380, FROM S.V. AND PORV TO PRES. REL. TAN

TPIPE VERSION
RUN DATE- 82/
PAGE N

A S M E   C O D E   C L A S S   2   S T R E S S   S U M M A R Y

MEMBER NAME	NODAL NAME	EQN NO.	CODE STRESS	ALLOWABLE STRESS	STRESS RATIO	DESCRIPTION
3520 *	PR56 *	8	9461.	15900.	.60	MAX STRESS
3520 *	PR56 *	8	9461.	15900.	.60	MAX STRESS RATIO
3290 *	CENTR*	10	13495.	27475.	.49	MAX STRESS
3290 *	CENTR*	10	13495.	27475.	.49	MAX STRESS RATIO
3290 *	CENTR*	11	17492.	43375.	.40	MAX STRESS
3290 *	CENTR*	11	17492.	43375.	.40	MAX STRESS RATIO
1580 *	24 *	9U	15498.	19080.	.81	MAX STRESS
1580 *	24 *	9U	15498.	19080.	.81	MAX STRESS RATIO
1540 *	27 *	9E	20946.	28620.	.73	MAX STRESS
1540 *	27 *	9E	20946.	28620.	.73	MAX STRESS RATIO
1540 *	27 *	9F	20952.	38160.	.55	MAX STRESS
1540 *	27 *	9F	20952.	38160.	.55	MAX STRESS RATIO
2010 *	CENTR*	PR	18280.	37244.	.49	MAX STRESS
2010 *	CENTR*	PR	18280.	37244.	.49	MAX STRESS RATIO
1540 *	27 *	AV	24442.	600000.	.04	MAX STRESS
1540 *	27 *	AV	24442.	600000.	.04	MAX STRESS RATIO

18-3

TOTAL NUMBER OF PIPE MEMBERS WITH NODAL POINTS GREATER THAN A THRESHOLD STRESS RATIO OF 1.000

EQUATION 8..... 0
EQUATION 10..... 0
EQUATION 11..... 0
EQUATION 9U..... 0
EQUATION 9E..... 0
EQUATION 9F..... 0

PIPE RUPTURE.... 0
ACTIVE VALVE.... 0

CODE CLASS I MAXIMUM STRESS SUMMARY
SUBSYSTEM RC-03

POINT NO	POINT ID	EQUATION	CODE STRESS (PSI)	ALLOWABLE STRESS (PSI)	LOAD SET	COMMENTS
25	Elbow (Loop B)	Equation 9 (Seismic)	6460	24120 (1.5 Sm)	N/A	
10	Nozzle (Loop B)	Equation 9 (Blowdown)	5767	24120 (1.5 Sm)	N/A	
10	Nozzle (Loop B)	Equation 10 (Sn)	32511	48564 (3.0 Sm)	0 - 4	
-	-	Equation 12 (Se)	N/A	N/A	N/A	
-	-	Equation 13	N/A	N/A	N/A	
75	Sockolet Conn. (Loop A)	Usage Factor U = 0.015	N/A	Usage Factor U = 1.0	0-4	

NOTES:

(1) No emergency condition specified
(2) Faulted condition enveloped by design condition evaluations (Equation 9)

CODE CLASS I MAXIMUM STRESS SUMMARY
SUBSYSTEM RC-01

POINT NO	POINT ID	EQUATION	CODE STRESS (PSI)	ALLOWABLE STRESS (PSI)	LOAD SET	COMMENTS
124	Red.	Equation 9 (Seismic)	18796	24120 (1.5 Sm)	N/A	
83	Elbow	Equation 9 (Blowdown)	18434	24120 (1.5 Sm)	N/A	
83	Elbow	Equation 10 (Sn)	51436	54120 (3.0 Sm*)	5 - 6	* Sm is the average value for the load set pair
-	-	Equation 12 (Se)	N/A	N/A	N/A	
-	-	Equation 13	N/A	N/A	N/A	
82	GBW	Usage Factor U = 0.193		Usage Factor U = 1.0	5 - 6	

NOTES:

(1) No emergency condition specified
(2) Faulted condition enveloped by design condition evaluations (Equation 9)

CODE CLASS I MAXIMUM STRESS SUMMARY (cont'd)
SUBSYSTEM RC-03

Equations 10, 11, and 12 stresses for Branch Connection Analyzed as per Rev. 4

POINT NO	POINT ID	EQUATION	CODE STRESS (PSI)	ALLOWABLE STRESS (PSI)	LOAD SET	COMMENTS
922	BR. Conn.	Equation 10 (Sn)	49999**	48564 (3.0 Sm)	0 - 5	** Denotes requirements of equation 10, primary plus secondary stress intensity range (NB-3653.1), have not been met; but the requirements of thermal stress ratchet (NB-3653.7), and equations 12 and 13 (NB-3653.6) have been met
922	BR. Conn.	Equation 12 (Se)	1520	48564 (3.0 Sm)	0 - 5	
922	BR. Conn.	Equation 13	34947	48564 (3.0 Sm)	0 - 5	