EG&G Idaho, Inc

FORM EG&G-398
(Rev. 11-79)

INTERIM REPORT

Accession No. __________

Report No. EGG-LOFT-5240

Contract Program or Project Title:

LOFT Program Office

Subject of this Document:

Best Estimate Prediction for LOFT Nuclear Experiment L3-5/L3-5A

Type of Document:

Experiment prediction analysis report

Author(s):

Slawomir M. Modro
Keith G. Condie

Date of Document:

September 1980

Responsible NRC Individual and NRC Office or Division:

G. D. McPherson, Chief, LOFT Research Branch,
Division of Reactor Safety Research, USNRC

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C.
Under DOE Contract No. DE-AC07-76ID01570
NRC FIN No. A6048

INTERIM REPORT

NRC Research and Technical Assistance Report

8010300194

PDR

BEST ESTIMATE PREDICTION FOR LOF NUCLEAR EXPERIMENT L3-5/L3-5A

By:

Slawomir M. Modro
Keith G. Condie

Approved:

J. H. Linebarger
J. H. Linebarger, Manager
Experiment Evaluation Branch

Charles W Solbrig
C. W. Solbrig, Manager
LOFT Program Office

The LOFT Subcommittee of the EG&G Pretest Prediction Consistency Committee has reviewed the RELAP5 model and predicted results for LOFT Small Break Experiment L3-5/L3-5A.

Code Development

Code Assessment

C. P. Fineman

Semiscale

Thermal Fuels

ABSTRACT

A two-fluid, transient, digital computer code (RELAP5) was used to simulate the Loss-of-Fluid Test (LOFT) facility during a small break experiment in which a break occurred in the intact loop cold leg that was equal to 2.5% of the flow area in the primary system piping. Simulation began with the initiation of the break and continued through the depressurization, isolation of the break, and subsequent repressurization. The simulation indicates that core decay energy will be removed from the LOFT primary system by coolant mass leaving the break and/or heat transfer to the secondary system through the steam generator by natural circulation cooling.

NRC FIN No. A6048 - LOFT Experimental Program

SUMMARY

This document contains the prediction of the coupled system thermal-hydraulic response for the Loss-of-Fluid Test (LOFT) system during Loss-of-Coolant Experiment (LOCE) L3-5/L3-5A. LOCE L3-5/L3-5A is the fourth powered experiment to be performed in the LOFT Nuclear Small Break Test Series (Test Series L3). This experiment is divided into two distinct parts designated as L3-5 and L3-5A.

The general programmatic objective of LOCE L3-5/L3-5A, in conjunction with future LOCE L3-6, is to evaluate the system effects of primary coolant pump operation during a small break loss-of-coolant accident transient. This objective will be accomplished during the L3-5 phase of the experiment. A secondary purpose, established with the addition of the L3-5A phase of the experiment, is to evaluate plant recovery by isolating the break and regaining the use of the steam generator as a heat sink. For LOCE L3-5/L3-5A, the break will be located in the intact loop cold leg, with the emergency core coolant injected into the reactor vessel downcomer.

Experiment prediction (EP) analyses provide data for evaluating the EP modeling techniques and specified operating conditions to ensure the experiment will meet its stated objectives without jeopardizing the safe operation of the LOFT facility.

The EP results presented in this report were obtained using the RELAP5 computer code. The predicted results were compared to the RELAP4 calculations, which were made previously to plan the experiment events. In addition, the similarity between the L3-5 phase of the experiment and LOCE L3-1 allowed the predictions to be compared to the results from LOCE L3-1 during the initial depressurization.

The results of this analysis indicate that LOCE L3-5/L3-5A will meet its stated objectives. The transition from the L3-5 phase to the L3-5A phase of the experiment will occur at about 1100 s, when the primary system pressure drops to 2.15 MPa (300 psig) and the break is isolated.

Decay heat will be removed from the core by natural loop circulation through the steam generator and energy removal from the break for the first 528 s of the transient until the primary system pressure drops below the secondary system, eliminating the steam generator as a heat sink. Reflux flow is predicted to occur during this time. From 528 s time until isolation of the break at approximately 1100 s, the decay heat will be effectively removed by coolant flow from the break.

Once the break is isolated and the high-pressure injection system is shut off, the system will repressurize until at 1316 s, when natural circulation will be again established as the primary mode of decay heat removal. During this second period of natural loop circulation, no reflux flow is predicted.

Secondary feed and bleed will be initiated at 3116 s. LOCEs L3-2 and L3-7 have shown secondary feed and bleed to be an effective procedure to reduce primary system pressure while maintaining core cooling. The calculation was terminated just prior to initiation of secondary feed and bleed.

CONTENTS

FIGURES

TABLES

DEFINITIONS

Best estimate-type calculation - a transient simulation using nominal plant values (for example, standard decay heat and nominal relief valve setpoints) with no conservative assumptions made with respect to engineered safety systems.

Natural loop circulation - coolant circulation (flow) in the primary coolant loop caused by density gradients, induced by heat generation in the core and sustained by concomitant heat removal.

Reflux flow - condensation in steam generator primary tubes with concomitant fallback of condensed liquid film into the intact loop hot leg and reactor vessel upper plenum.

BEST ESTIMATE PREDICTION FOR LOFT NUCLEAR EXPERIMENT L3-5/L3-5A

1. INTRODUCTION

As part of the experiment analysis effort performed by the Loss-of-Fluid Test (LOFT) Experimental Program, a best estimate-type experiment prediction (EP) of the thermal-hydraulic response of the LOFT system during an experiment is performed prior to the experiment using computer calculations. These predictions are performed using the best calculational techniques available to LOFT and provide data for:

1. Determining whether a loss-of-coolant experiment (LOCE) will meet its stated objectives

2. Evaluating parameters that affect the safety of the LOFT facility during the intended experiment

3. Determining event times for incorporation into the operating procedure

4. Determining possible instrument range adjustments

5. Evaluating the capability of the modeling techniques employed in EP analyses.

This document describes how the RELAP5 computer code was used to simulate and predict the LOFT system responses and presents predicted results for LOCE L3-5/L3-5A. Sections 1.1 and 1.2 of this introduction discuss the LOCE L3-5/L3-5A objectives and provide a brief description of the experiment and of the LOFT facility. Section 2 contains a description of the modeling techniques employed in the EP analyses. Section 3 contains discussions of the calculated results. Comparisons and conclusions of the analytical results are included in Section 4. References discussed are listed in Section 5. Appendices provide detailed calculational results

(Appendix A), algorithms for generation of the EP data in the data bank (Appendix B), listings of source deck changes (Appendix C), and listings of the code inputs (Appendix D).

1.1 LOCE L3-5/L3-5A Objectives and Description

LOCE L3-5/L3-5A is the fourth powered experiment to be conducted as part of the LOFT Nuclear Small Break Test Series L3. This experiment is divided into two distinct parts designated as L3-5 and L3-5A. The experiment objectives and descriptions for Test Series L3 are discussed in detail in Reference 1. The objectives for LOCE L3-5/L3-5A are given in Section 1.1.1. LOCE L3-5/L3-5A is described in Section 1.1.2.

1.1.1 LOCE L3-5/L3-5A Objectives

The general programmatic objective of LOCE L3-5/L3-5A, in conjunction with the future LOCE L3-6, is to evaluate the system effects of primary coolant pump operation during a small break loss-of-coolant (LOCA) transient. This objective will be accomplished during the L3-5 phase of the experiment. A secondary purpose, established with the addition of the L3-5A phase of the experiment, is to evaluate plant recovery by isolating the break and regaining the use of the steam generator. The specific LOCE L3-5/L3-5A objectives are as follows:

1. Objectives for the L3-5 phase of the experiment are:

    a. To conduct a small break depressurization in the LOFT facility with a 16.19-mm (0.6374-in.) diameter break orifice in the intact loop cold leg between the primary pump and the reactor vessel, with primary coolant pump trip at rupture, with the high-pressure injection system (HPIS) injecting into the reactor vessel downcomer, and with the scaled accumulator isolated from the loop.

b. To measure the primary system coolant inventory and system mass distribution as a function of time during the depressurization using available instrumentation.

2. Objectives for the L3-5A phase of the experiment are:

a. To reestablish the steam generator as a primary system heat sink by isolating the break, allowing the primary system pressure to increase to above the secondary system pressure, and using operator controlled secondary "feed and bleed" cooldown.

b. To obtain flow and density measurements in the intact loop hot leg and fluid temperature difference data associated with the steam generator to investigate the primary coolant loop flow modes and steam generator heat transfer modes following reestablishment of the steam generator as a primary system heat sink.

c. To reestablish primary coolant conditions associated with complete LOFT facility recovery, with the break isolated, and with the LOFT accumulator inactive.

For this experiment, the objectives of the L3-5 phase are considered primary objectives, while those of the L3-5A phase are considered secondary.

1.1.2 LOCE L3-5/L3-5A Description

LOCE L3-5/L3-5A will utilize a new break piping configuration to provide for a small break in the intact loop cold leg rather than the broken loop piping. The break orifice diameter will be 16.19 mm (0.6374 in.) corresponding to a 2.5% break in the primary coolant loop of a large commercial plant. In addition, the HPIS injection point will be into the reactor vessel downcomer for the L3-5 phase. The remaining system configuration will be unchanged from that used for LOCE L3-1.[2]

The initial conditions for LOCE L3-5/L3-5A will be nominally unchanged from those of LOCE L3-1. Primary system pressure will be 14.95 MPa (2156 psig) measured in the hot leg. Primary coolant mass flow will be 478.8 kg/s (3.8×10^6 lbm/hr) at a temperature measured in the cold leg of 556.8 K (542.5°F). The core power level will be 50 MW. These conditions should produce a core temperature difference of about 19 K (34°F).

The following paragraphs provide a descriptive scenario of the LOCE L3-5/L3-5A transient as planned.

Prior to test initiation:

1. The reactor will be operated at 50 MW until a decay heat level corresponding to 40 h of previous full-power operation is reached.

2. The accumulators will be isolated from the primary coolant loop.

3. The required preblowdown conditions will be established in the primary coolant loop.

Initiation of blowdown (L3-5 phase):

4. The transient will be initiated by opening the break isolation valve, scramming the plant and, upon verification of scram, opening the break initiation valves of the new intact loop break piping. The scram signal will also trip off the steam generator main feedwater flow and shut the steam control valve.

5. Also upon verification of scram, the primary coolant pumps will be tripped and will coast down under the influence of the primary loop coolant flow and the installed flywheels until the field breakers trip at approximately 12.5 Hz. Combined primary coolant pump injection of [illegible] ± 0.02 l/s (1.25 ± 0.25 gpm) will be used to cool the pump bearings.

6. A few seconds into the transient, initiated by a 13.15 MPa (1896 psig) hot leg pressure trip, emergency core coolant (ECC) injection from HPIS Pump A to the reactor vessel downcomer will begin at a rate of 0.3 l/s (5 gpm).

7. At 60s after scram, one auxiliary feedwater pump will be started, filling the steam generator secondary at a rate of about 0.50 l/s (8 gpm).

8. At a primary system pressure of 2.15 MPa (300 psig) (chosen to ensure that the primary system achieves a minimum system mass inventory) the mixture level in the upper plenum will be determined, if possible. The break line isolation valve will be closed, and HPIS injection into the reactor vessel downcomer and primary coolant pump injection will be terminated.

Initiation of Recovery (L3-5A phase):

9. After isolating the break, the primary system will be allowed to repressurize until it stabilizes at a value near the steam generator secondary pressure.

10. When the increasing primary system pressure reaches that of the secondary system or at 30 min after scram, whichever comes first, the auxiliary feedwater flow to the steam generator will be terminated.

11. The system will be allowed to stabilize for a minimum of 30 min to observe the primary system cooling mode.

12. "Feed and bleed" cooldown of the secondary will begin after the 30-min period of stabilization at a rate of 40 to 50 K/h (70 to 90^{0}F/hr) and continue until the experiment is terminated.

13. After 15 min of feed-and-bleed operation (∿45 min after the pressure in the primary system reequalizes with that in the secondary system), ECC injection from HPIS Pump A to the reactor vessel downcomer will be restarted to facilitate recovery of the plant. The low-pressure injection system (LPIS), injecting to the intact loop cold leg, will actuate when the hot leg pressure reaches 2.12 MPa (295 psig). The LPIS will start injecting ECC at about 1.6 MPa (220 psig). The experiment will be considered terminated when a 28-K (50^{o}F) subcooled condition is established in the intact loop hot leg and the water level is above the reactor vessel inlet and outlet nozzles.

1.? LOFT Facility Description

The LOFT facility is described in detail in Reference 3. The LOFT instrumentation and major components are shown in Figures 1 through 7. The instrumentation nomenclature is explained in Table 1.

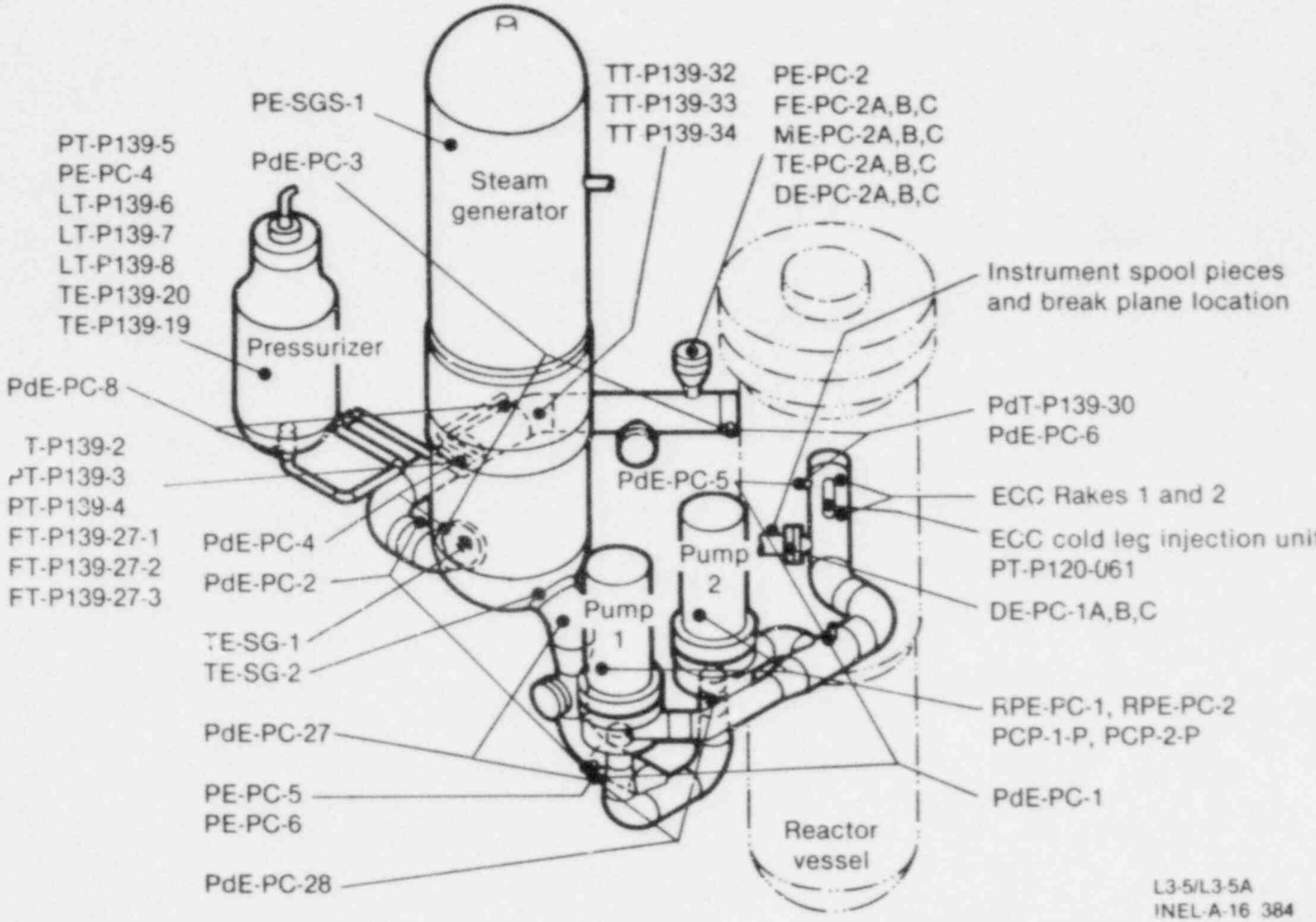


Figure 1. LOFT intact loop thermo-fluid instrumentation.

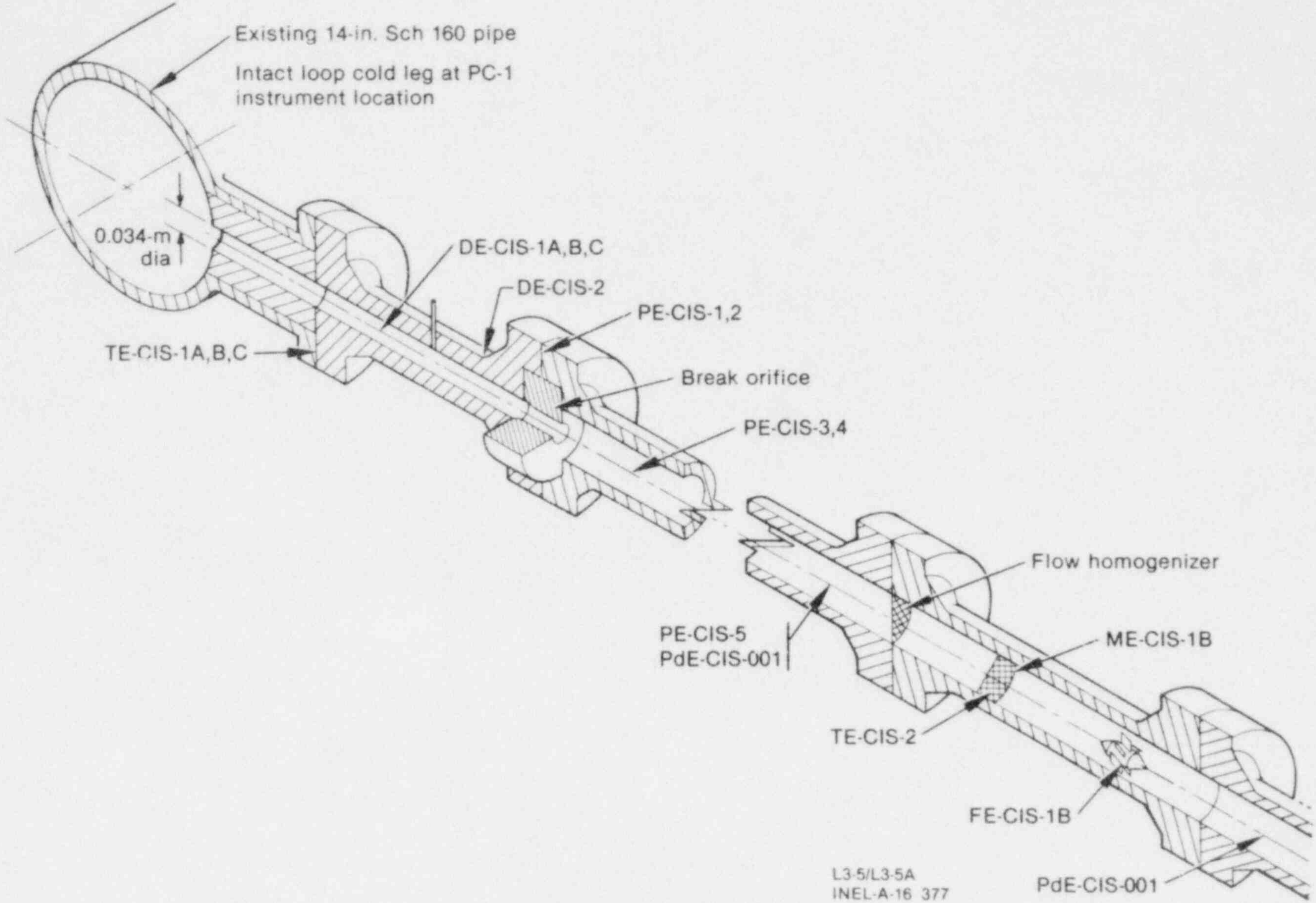


Figure 2. Instrument spool pieces and break orifice for LOCE L3-5/L3-5A.

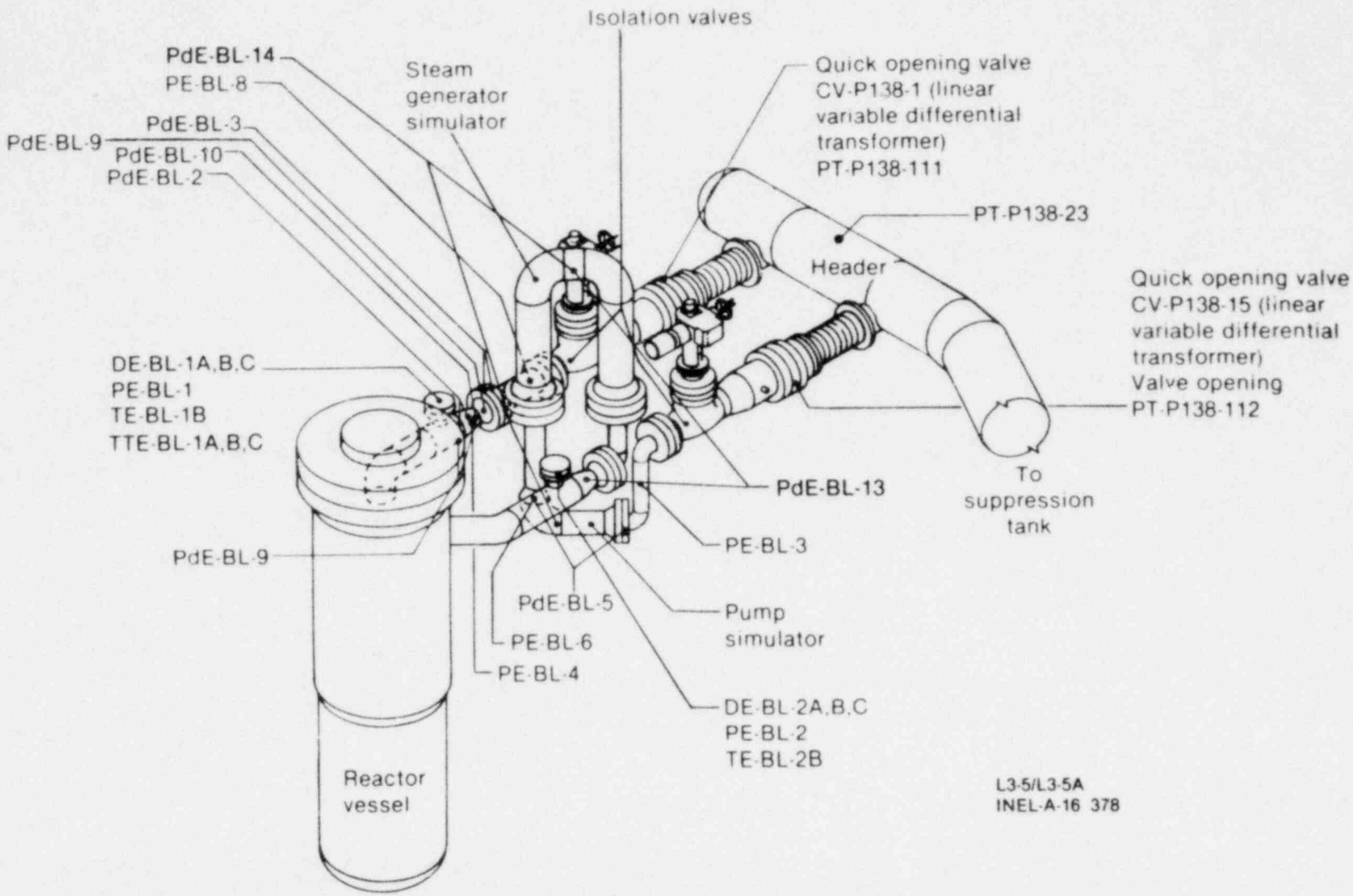


Figure 3. LOFT broken loop thermo-fluid instrumentation.

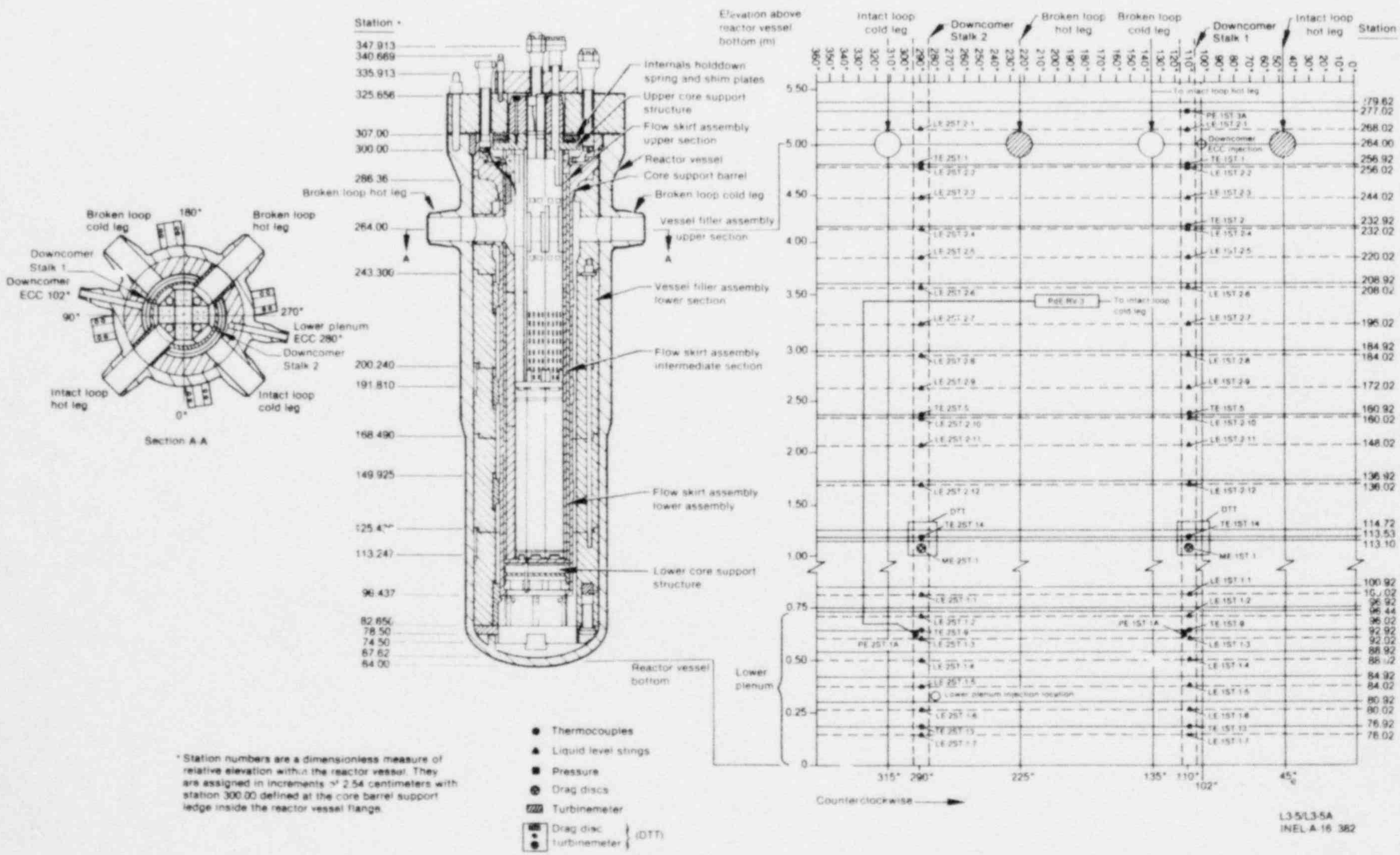


Figure 4. LOFT reactor vessel instrumentation.

* Station numbers are a dimensionless measure of relative elevation within the reactor vessel. They are assigned in increments of 25.4 mm with Station 300.00 defined at the core barrel support ledge inside the reactor vessel flange.

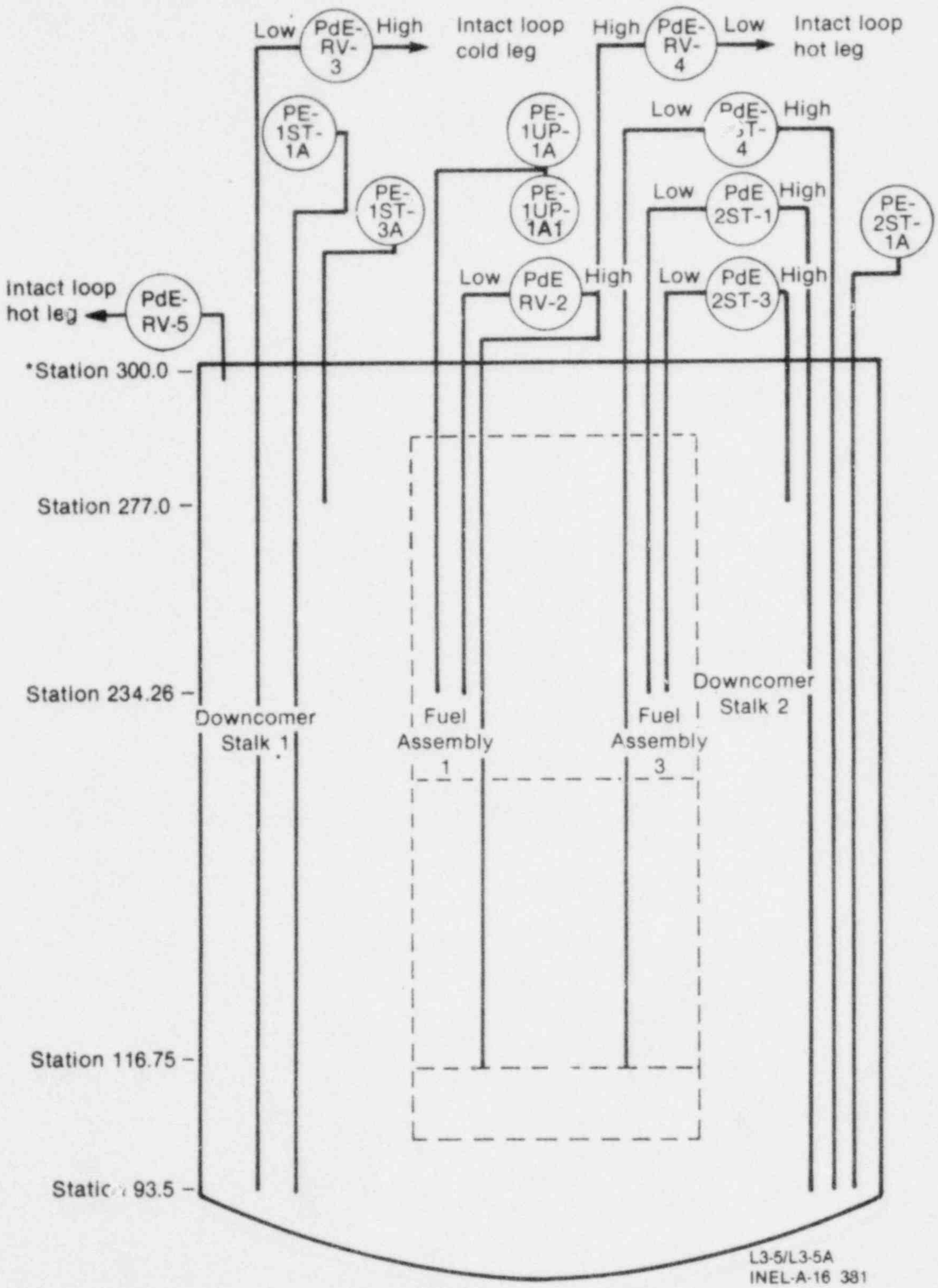


Figure 5. LOFT reactor vessel pressure and differential pressure instrumentation.

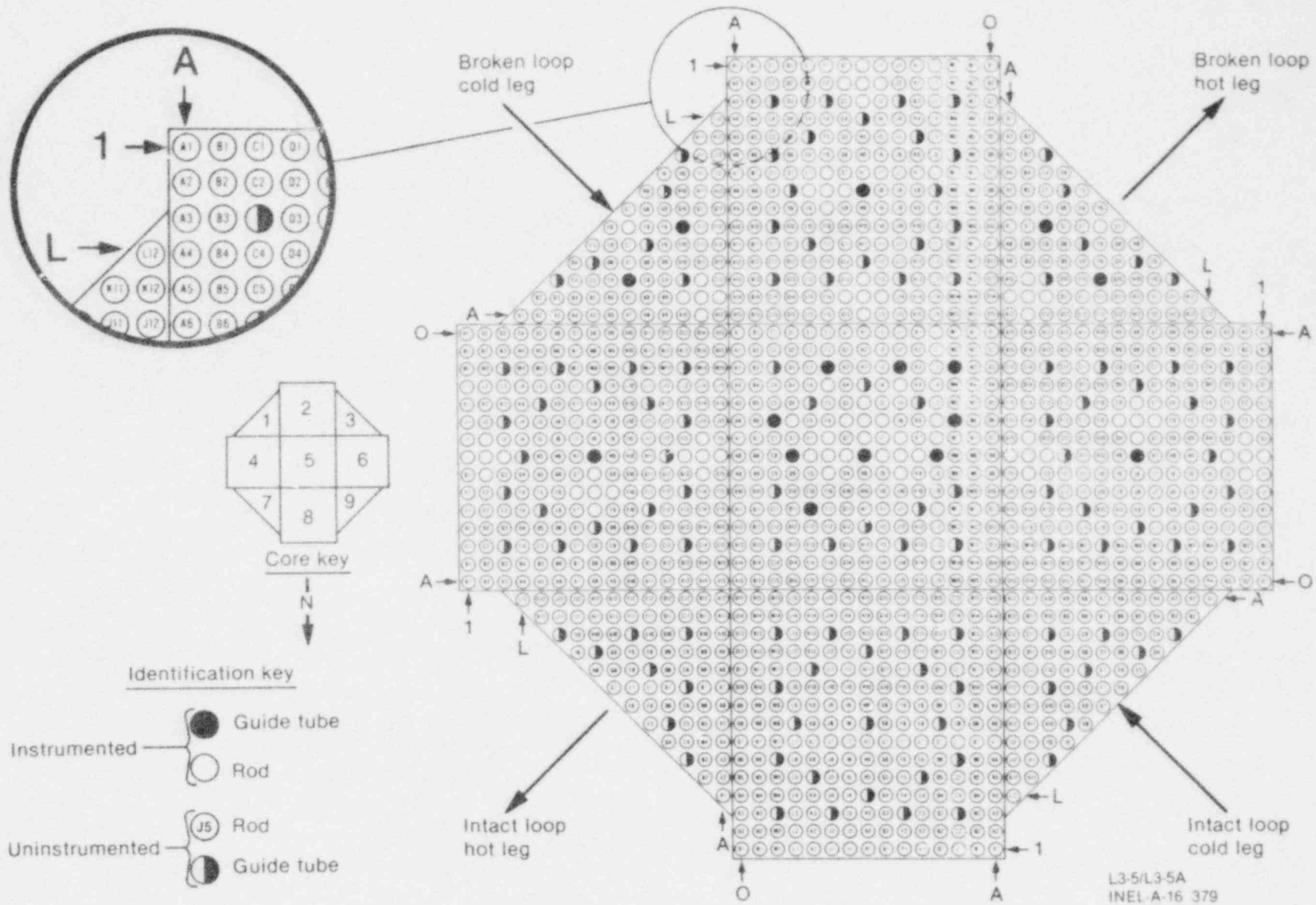
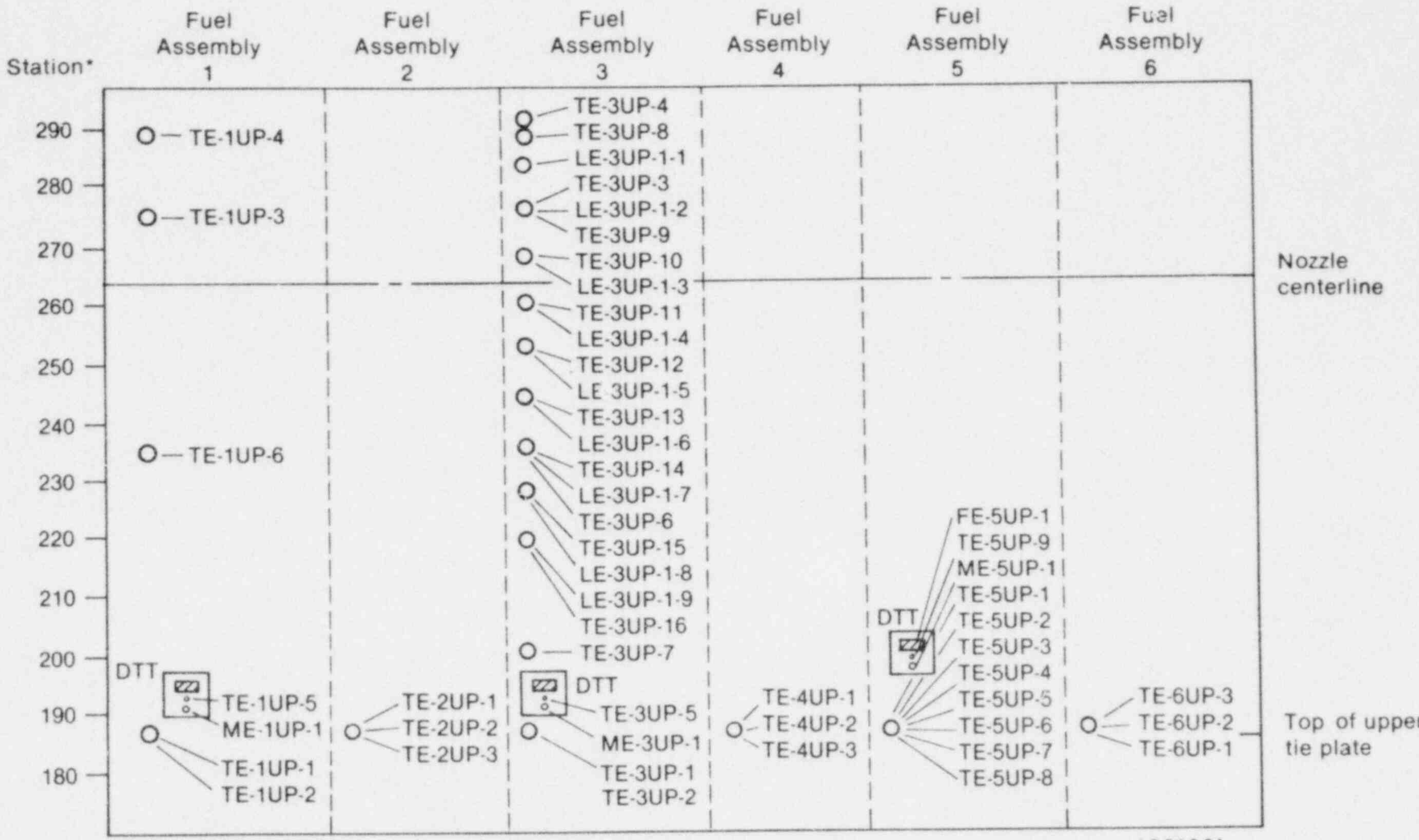


Figure 6. LOFT core map showing position designations.



*Station numbers are a dimensionless measure of relative elevation within the reactor vessel. They are assigned in increments of 25.4 mm with station 300.00 defined at the core barrel support ledge inside the reactor vessel flange.

Figure 7. LOFT reactor vessel upper plenum drag disc-turbine and coolant level transducers and temperature element elevations.

TABLE 1. NOMENCLATURE FOR LOFT INSTRUMENTATION

The de'ignations for the different types of transducers are:

TE	-	Temperature element
TT	-	Temperature transmitter
PE	-	Pressure transducer
PT	-	Pressure transmitter
PdE	-	Differential pressure element
PdT	-	Differential pressure transducer
LE	-	Coolant level transducer
LT	-	Level transmitter
FE	-	Coolant flow transducer
FT	-	Flow transmitter
DiE	-	Displacement transducer
ME	-	Momentum flux transducer
RPE	-	Pump speed transducer
DE	-	Densitometer
LIT	-	Level indicating transmitter
CV	-	Control valve
PCP	-	Pump frequency transducer
TTE	-	Transit time element

The designations for the different systems are:[a]

PC	-	Primary coolant intact loop
BL	-	Broken loop
SG	-	Steam generator
RV	-	Reactor vessel
SV	-	Suppression tank
UP	-	Upper plenum
LP	-	Lower plenum
ST	-	Downcomer stalk

a. For in-core transducers, the system designation is replaced by a fuel assembly number, column and row designations, followed by the elevation (in inch increments from lower grid plate), where applicable.

2. COMPUTER SIMULATION

The RELAP5/MOD"1" computer code[a] was used to simulate the transient thermal-hydraulic responses for the LOFT system during LOCE L3-5/L3-5A. The RELAP5 code uses a two-fluid, thermal nonequilibrium, hydraulic model. The specific application of the code to the LOCE L3-5/L3-5A simulation is discussed in this section.

2.1 Nodalization

The nodalization used for the LOCE L3-5/L3-5A RELAP5 calculation, presented in this section, is based on the standard nodalization presented in Reference 4 with changes where necessary to represent the LOFT system configuration for LOCE L3-5/L3-5A and to reflect experience gained in use of the code. The nodalization scheme is shown on Figure 8. A brief description of each component is given in Table 2.

The following changes were made to the base nodalization[4] for this LOCE L3-5/L3-5A EP analysis:

1. The break location was moved from the broken loop cold leg to the intact loop cold leg and the new break spool piece was modeled.

2. The accumulator was removed from the model.

3. Primary coolant pump injection flows were added to properly account for the system mass inventory.

4. Heat slabs were added to the outside of the steam generator steam dome and downcomer to allow calculation of ambient heat losses.

---

a. This analysis was performed using Cycle 160, an experimental version of the RELAP5/MOD"1" code which is filed under Idaho National Engineering Laboratory Configuration Control Number H011985B.

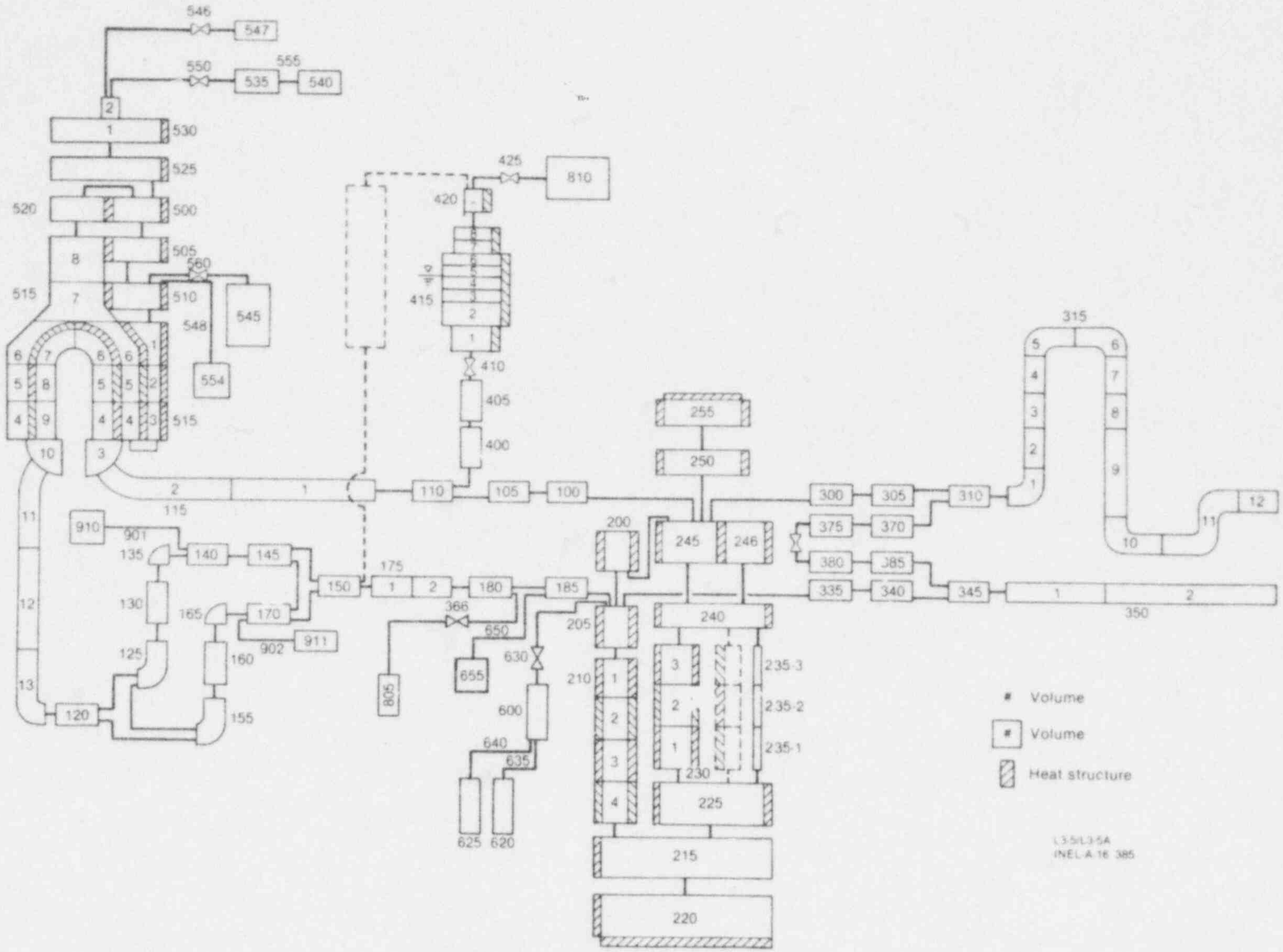


Figure 8. LOFT RELAP5 model schematic diagram.

TABLE 2. DESCRIPTION OF NODALIZATION COMPONENTS

Component Number	Component Type	Volume Number	Number of Junctions	Composition
100	Branch	1	2	Core barrel nozzle, vessel nozzle, and half of flow device.
105	Branch	1	1	Half of flow device, 45-degree elbow, pipe to pressurizer connection.
110	Branch	1	1	Pipe from pressurizer connection and venturi.
115	Pipe	1	12	90-degree elbow, pipe section, and half of reducer.
		2		Half of reducer, 38-degree elbow, and pipe section.
		3		Steam generator inlet plenum.
		4		Vertical steam generator tubes.
		5		Vertical steam generator tubes.
		6		90-degree elbow of steam generator tubes.
		7		90-degree elbow of steam generator tubes.
		8		Vertical steam generator tubes.
		9		Vertical steam generator tubes.
		10		Steam generator outlet plenum.
		11		52-degree elbow and half of reducer.
		12		Half of reducer and pipe section.
		13		90-degree elbow.
120	Branch	1	3	Pipe section and inlet pipe of pump suction tee.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
125	Branch	1	2	Half of pump suction tee, 90-degree elbow, and half of reducer.
130	SNGLVOL	1	0	Half of reducer and primary coolant Pump 1 inlet pipe.
135	Pump	1	2	Primary coolant Pump 1.
140	Branch	1	0	Primary coolant Pump 1 outlet pipe and 45-degree elbow.
145	Branch	1	2	Pipe section, reducer, and half of pump outlet tee.
150	Branch	1	2	Half of pump outlet tee and pipe section.
155	Branch	1	1	Half of pump suction tee, 90-degree elbow, and half of reducer.
160	SNGLVOL	1	0	Half of reducer and primary coolant Pump 2 inlet pipe.
165	Pump	1	2	Primary coolant Pump 2.
170	Branch	1	1	90-degree elbow and inlet of pump outlet tee.
175	Pipe	1	1	90-degree elbow.
		2		Pipe section and 45-degree elbow.
180	Branch	1	1	Pipe section to ECC connection.
185	Branch	1	2	Pipe section from ECC connection, vessel nozzle, and vessel filler.
200	Branch	1	2	Upper part of inlet annulus distributor.
205	Branch	1	1	Lower part of inlet annulus distributor.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
210	Pipe	1	3	Downcomer.
		2		Downcomer.
		3		Downcomer.
		4		Downcomer.
215	Branch	1	3	Upper part of the lower plenum.
220	SNGLVOL	1	0	Lower part of the lower plenum.
225	Branch	1	2	Lower core support structure.
230	Pipe	1	2	Active core lower part.
		2		Active core central part.
		3		Active core upper part.
235	Pipe	1	2	Core bypass.
		2		Core bypass.
		3		Core bypass.
240	Branch	1	2	Upper core support structure.
245	Branch	1	1	Upper flow skirt region.
246	Branch	1	1	Dead end of fuel modules.
250	Branch	1	2	Nozzle region of upper plenum.
255	SNGLVOL	1	0	Upper part of upper plenum.
300	Branch	1	2	Core barrel nozzle and vessel nozzle.
305	Branch	1	1	45-degree elbow and half of reflood assist bypass system (RABS) tee hot leg.
310	Branch	1	2	Half of RABS tee hot leg, pipe section, Flange 1, and half of Orifice XRO-85.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
315	Pipe	1	11	Half of Orifice XRO-85, Flange 11, and 90-degree elbow.
		2		Pipe section and Flange 10.
		3		Flange 9.
		4		Flange 9.
		5		90-degree elbow and half of pipe section.
		6		Half of pipe section and 90-degree elbow.
		7		Flange 8.
		8		Flange 8.
		9		Flange 7, pipe section, and half of reducer.
		10		Half of reducer, 90-degree elbow, and pump simulator Flange 14.
		11		Flange 13, 90-degree elbow, pipe section, 90-degree elbow, Flange 6, Orifice XRO-81, and Flange 4.
		12		Half of isolation valve.
335	Branch	1	2	Vessel filler and vessel nozzle.
340	Branch	1	1	45-degree elbow and half of RABS tee cold leg.
345	Branch	1	2	Half of RABS tee cold leg and flow device.
350	Pipe	1	1	Flange 5.
		2		Orifice XRO-88, Flange 2, pipe section, Flange 3, Orifice XRO-86, Flange 12, and half of isolation valve cold leg.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
366	Valve	--	--	Break orifice.
370	Branch	1	1	RABS hot leg single pipe.
375	SNGLVOL	1	0	RABS hot leg parallel pipes.
380	SNGLVOL	1	0	RABS cold leg parallel pipes.
385	Branch	1	1	RABS cold leg single pipe.
400	Branch	1	2	Pressurizer surge line, primary coolant system side.
405	SNGLVOL	1	0	Pressurizer surge line, pressurizer side.
415	Pipe	1	7	Pressurizer inlet.
		2		Pressurizer vessel water space.
		3		Pressurizer vessel water space.
		4		Pressurizer vessel vapor space.
		5		Pressurizer vessel vapor space.
		6		Pressurizer vessel vapor space.
		7		Pressurizer vessel vapor space.
		8		Pressurizer vessel vapor space.
420	Branch	1	1	Pressurizer outlet.
500	Branch	1	3	Outlet of primary separator (top of volume is at top of shroud).
505	SNGLVOL	1	--	Volume between bottom of Component 500 and top of feed ring.
510	Branch	1	2	Top of volume is feed ring elevation; bottom of volume is at narrow portion of downcomer.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
515	Pipe	1	7	Narrow section of downcomer.
		2		Narrow section of downcomer.
		3		Narrow section of downcomer.
		4		Volume between shroud and tubes.
		5		Volume between shroud and tubes.
		6		Volume between shroud and tubes.
		7		Volume between shroud and tubes.
		8		Lower part of riser.
520	Branch	1	1	Top of riser, inlet to primary separator.
525	Branch	1	1	Bottom of steam dome between primary separator outlet and mist extractor inlet.
530	Pipe	1	1	Top of steam dome between mist extractor and outlet pipe.
		2		Outlet pipe to steam flow control valve.
535	SNGLVOL	1	--	Steam generator outlet pipe between steam flow control valve and air-cooled condenser.
540	TMDPVOL	1	--	Air-cooled condenser.
545	TMDPVOL	1	--	Demineralized water storage tanks.
546	Valve	--	--	Steam control valve bypass valve.
547	TMDPVOL	1	--	Air-cooled condenser.
548	TMDPJUN	--	--	Feedwater valve for feed and bleed.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
550	Valve	--	--	Main steam control valve.
554	TMDPVOL	1	--	Demineralized water storage tanks.
555	SNGLJUN	--	--	Condenser inlet valve.
560	TMDPJUN	--	--	Feedwater valve.
600	Branch	1	--	Pipe between cold leg and low pressure injection pump tee.
620	TMDPVOL	1	--	Borated water storage tank.
625	TMDPVOL	1	--	Borated water storage tank.
630	Valve	--	--	ECC valve.
635	TMDPJUN	--	--	LPIS.
640	TMDPJUN	--	--	HPIS "A".
650	TMDPJUN	--	--	HPIS "B".
655	TMDPVOL	1	--	Borated water storage tank.
805	TMDPVOL	1	--	Suppression tank.
901	TMDPJUN	--	--	Primary coolant Pump 1 injection.
902	TMDPJUN	--	--	Primary coolant Pump 2 injection.
910	TMDPVOL	1	--	Borated water storage tank.
911	TMDPVOL	1	--	Borated water storage tank.

5. The reactor vessel wall heat slabs were modified to allow heat transfer to ambient.

6. The number of volumes in the pressurizer was increased to provide a better calculation for pressurizer level.

7. Heat slabs were added to account for heat transfer from the pressurizer walls to the fluid.

8. Modeling of tees was changed from a two-dimensional branch to a one-dimensional branch. The RELAP5 computer code developers recommended using this modeling change until the momentum interaction term is included for a multiple junction branch.

There are several core bypass flow paths in the LOFT system. These bypass flow paths are shown on Figure 9 with the arrows showing the flow direction at experiment initiation. The pressurizer continuous spray bypass is considered minor and was not simulated in this calculation. The lower plenum to upper plenum bypass and the inlet annulus to upper plenum bypass were modeled with 2 and 3% of full intact loop flow, respectively. The reflood assist bypass valve (RABV) leakage bypass was not quantified prior to the initiation of this analysis; consequently, it is not included.

Examination of data from LOFT Experiments L3-7[5] and L6-5[6] indicated a leakage through the main steam control valve of approximately 0.2 kg/s (3.2 gpm) when the valve was in its fully closed position at a pressure of 6.2 MPa (887 psig). This computer simulation for LOCE L3-5/L3-5A included the steam valve leakage.

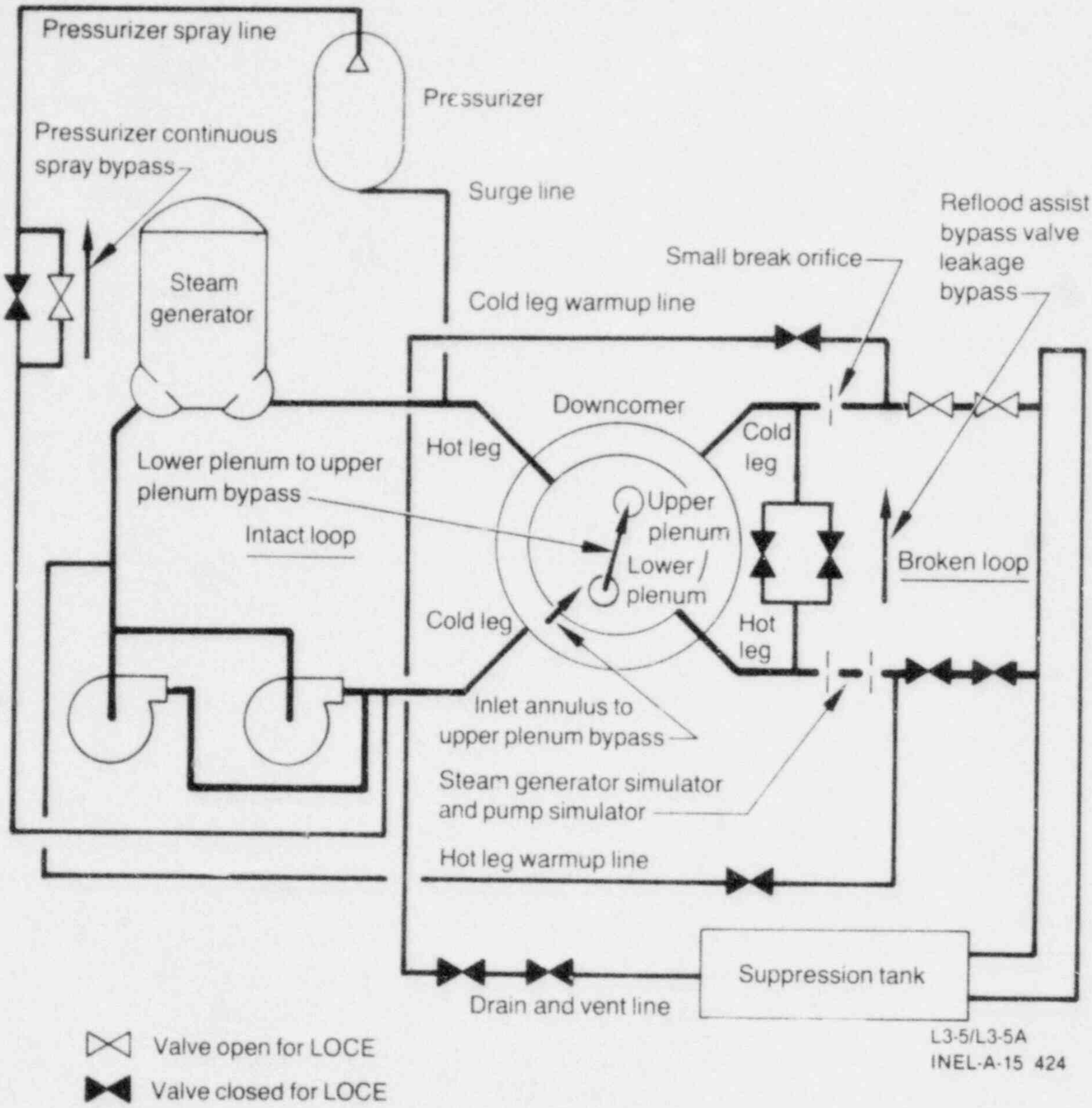


Figure 9. LOFT system schematic showing bypass flow paths.

3. CALCULATIONAL RESULTS

This section gives a general overview of the transient simulation and summarizes the calculational results on natural loop circulation including reflux flow.

3.1 General Overview of Transient Simulation

The LOCE L3-5/L3-5A transient simulation is characterized by the pressure history shown in Figure 10 for the reactor vessel upper plenum. The transient was initiated by reactor scram at time zero. After scram verification at about 2 s, the break initiation valves were actuated, the primary coolant pumps were tripped, the main feedwater pump was turned off, and the main steam control valve started to close. The main steam control valve closes at the rate of 5%/s and was closed at 14 s. Depressurization of the primary system was very rapid for the first approximately 60 s. At 6.5 s, the HPIS was initiated as the primary system pressure dropped to 13.18 MPa (1900 psig). Liquid flashed to vapor as early as 17 s, and by about 60 s, enough vapor had been generated in the reactor vessel upper plenum and intact loop hot leg that the upper plenum controlled system pressure and the depressurization rate decreased. The pressurizer was calculated to be empty of liquid at 65 s. The primary system pressure dropped below secondary pressure at 528 s, eliminating the steam generator as a decay heat removal sink.

Depressurization continued with the break serving as the primary means of heat removal until at 1119 s, when the pressure had dropped to 2.15 MPa (300 psig) and the break was isolated and HPIS flow was terminated as specified. Isolation of the break signaled completion of the L3-5 phase and the initiation of the L3-5A phase.

Primary system pressure began to increase because of the loss of the steam generator as a heat sink until at 1316 s, when the primary pressure became equal to secondary pressure and natural circulation was reestablished. Pressure in the primary and secondary systems continued to rise slowly for the next 30 min of the transient until secondary feed and

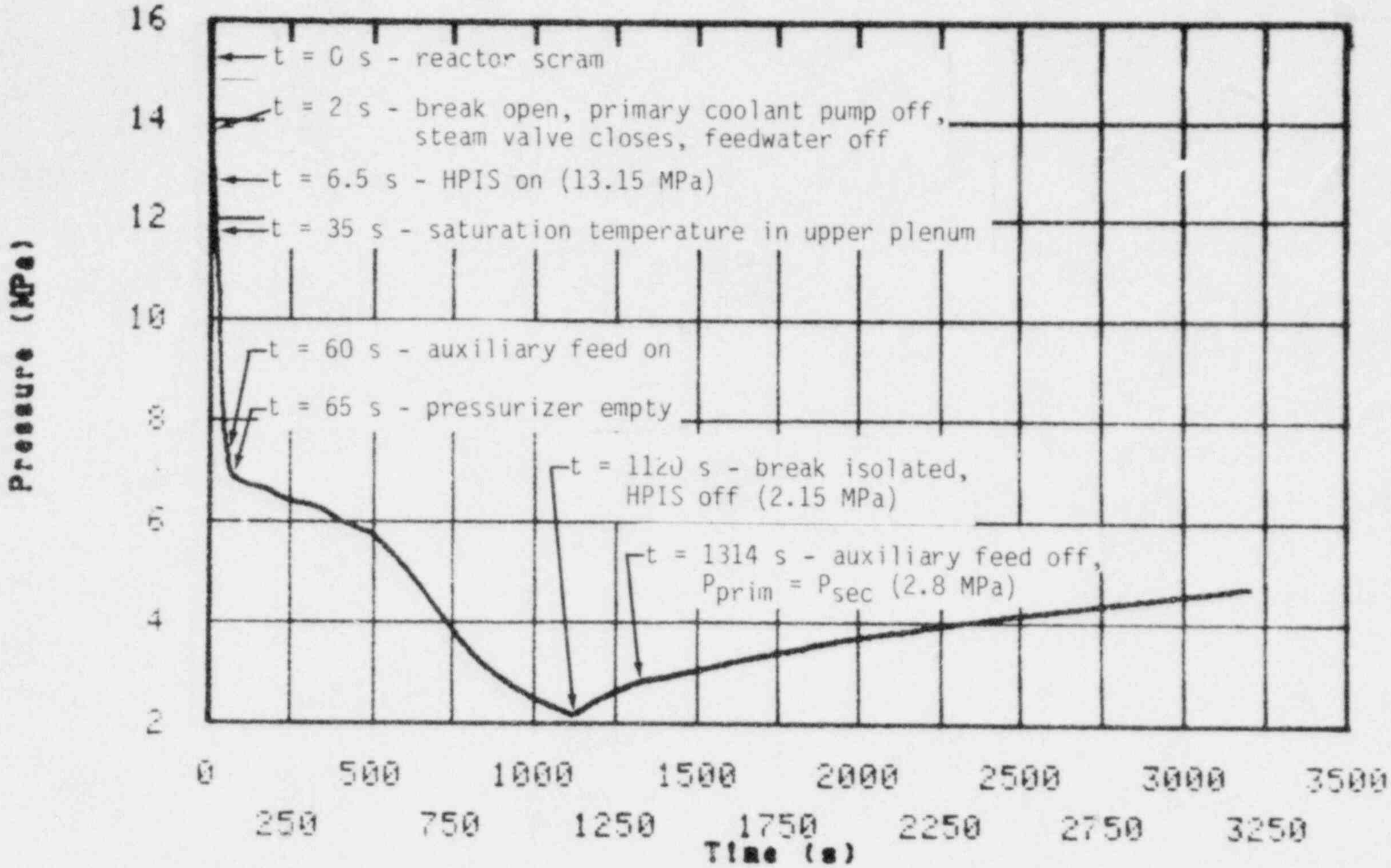


Figure 10. Pressure in reactor vessel upper plenum.

bleed was initiated at 3216 s. The computer simulation was terminated at this time. Previous calculations of secondary system feed and bleed for LOCEs L3-2 and L3-7 showed that primary pressure followed specified secondary pressure; consequently, it was not felt necessary to simulate the feed and bleed operation in LOCE L3-5/L3-5A.

Figure 11 shows the calculated pressure response for the secondary system during the transient. The mass flow out of the break is shown in Figure 12, and the primary system coolant mass inventory is shown in Figure 13. The increase in mass shown after break isolation is due, primarily, to mass error in the code. Figure 14 shows the calculated fuel rod cladding temperature at the core midplane elevation. The cladding surface temperature basically follows the saturation temperature of the fluid throughout the entire transient.

Because of the similarity of the initial portion of the L3-5 phase of this experiment with the previously conducted LOCE L3-1,[2] a comparison of primary pressure was made with this calculation and the data for LOCE L3-1 for about the first 1000 s and is shown in Figure 15. Also shown in this comparison are the results of an earlier RELAP4 calculation for L3-5. These pressure histories are very similar, giving confidence in the RELAP5 predicted response.

3.2 Natural Loop Circulation

Natural loop circulation is a flow mode in which energy is transferred from the nuclear core to the steam generator without the addition of mechanical energy. Energy addition from the core and removal in the steam generator combined with the elevation difference between the source and heat sink will provide the thermal driving head for this flow.

Natural loop circulation can occur for both single- and two-phase conditions, and both single- and two-phase flow can exist in different parts of the loop at the same time. Reflux flow is a special case of two-phase natural loop circulation.

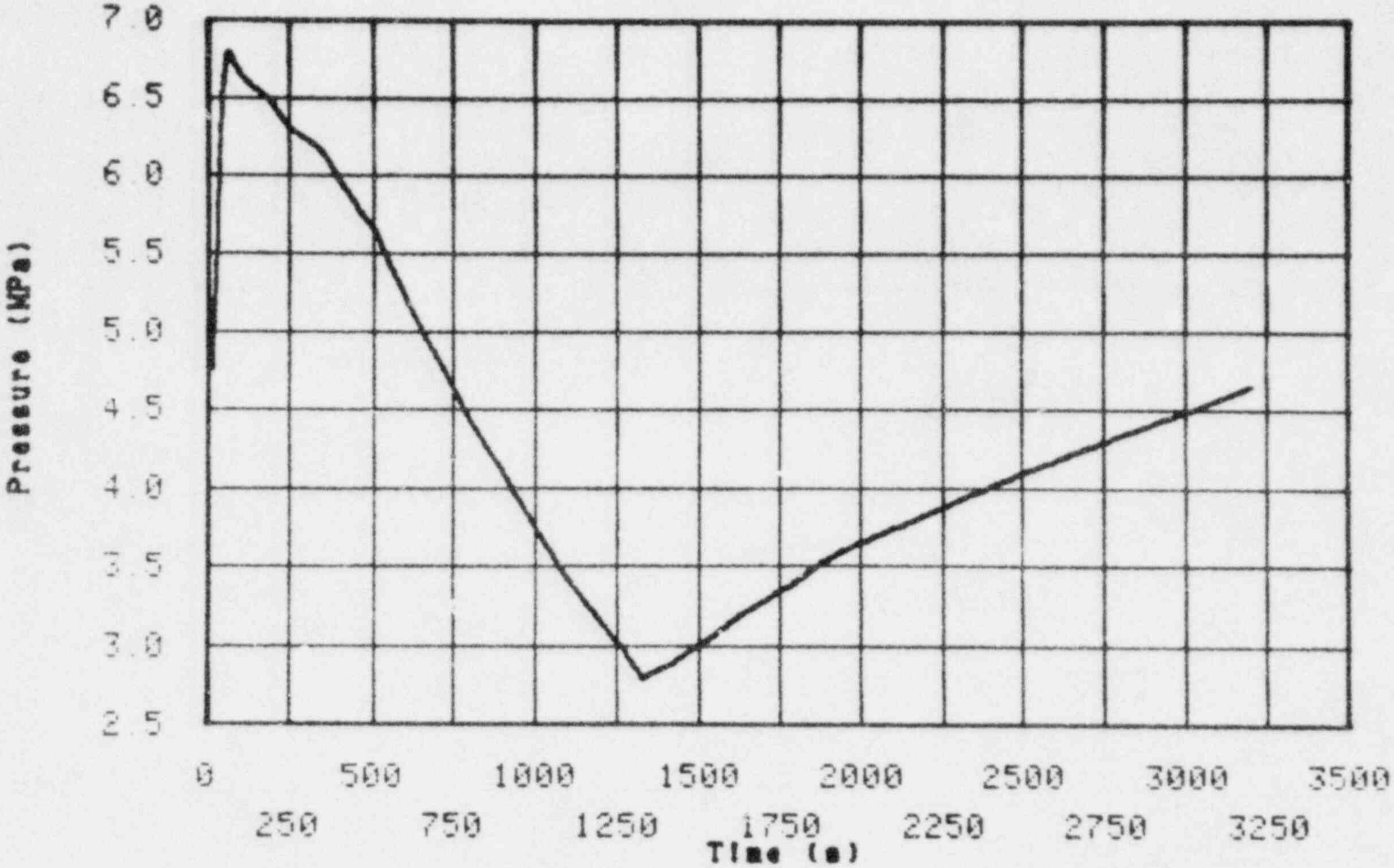


Figure 11. Pressure in steam generator secondary side.

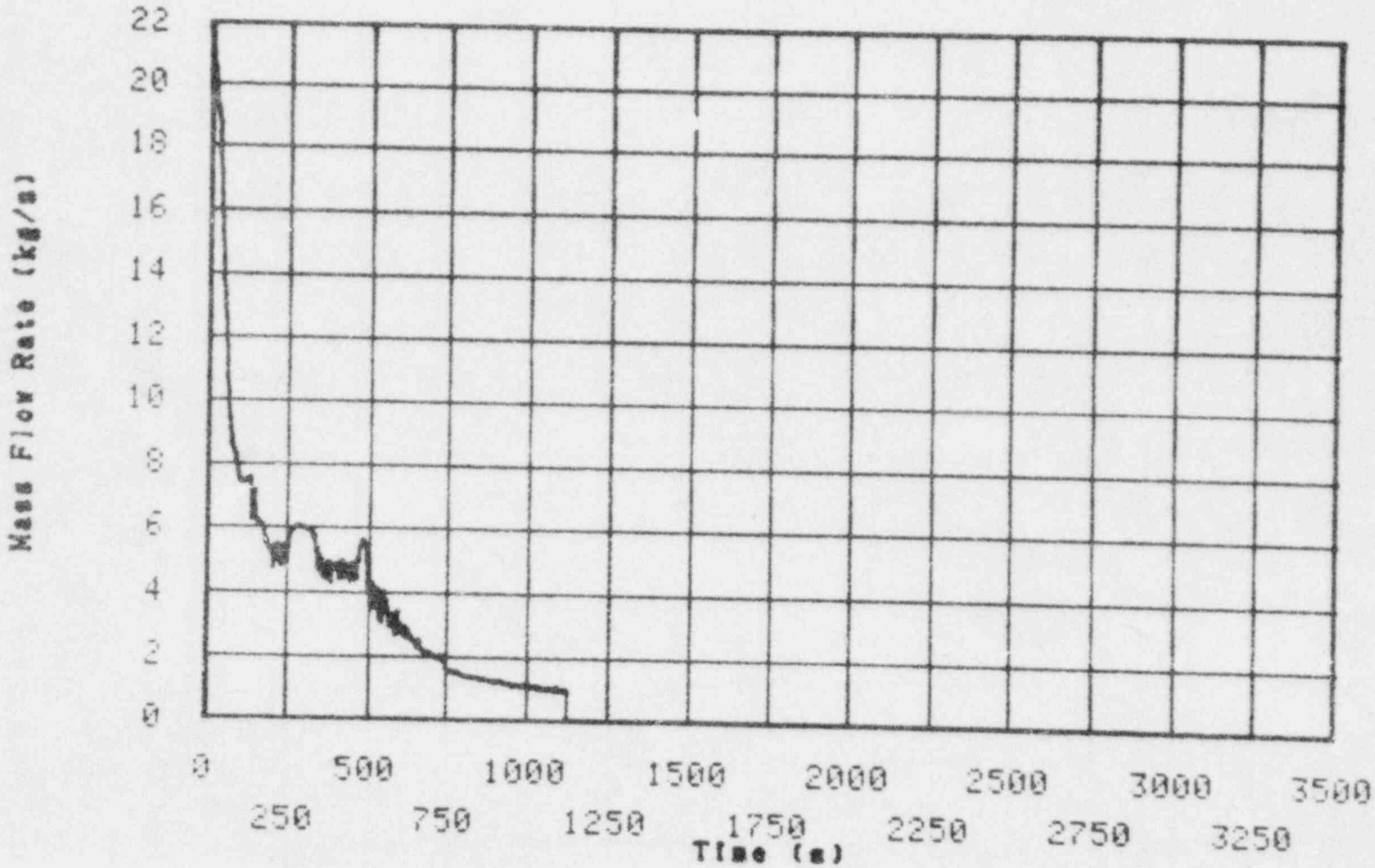


Figure 12. Mass flow at the break location.

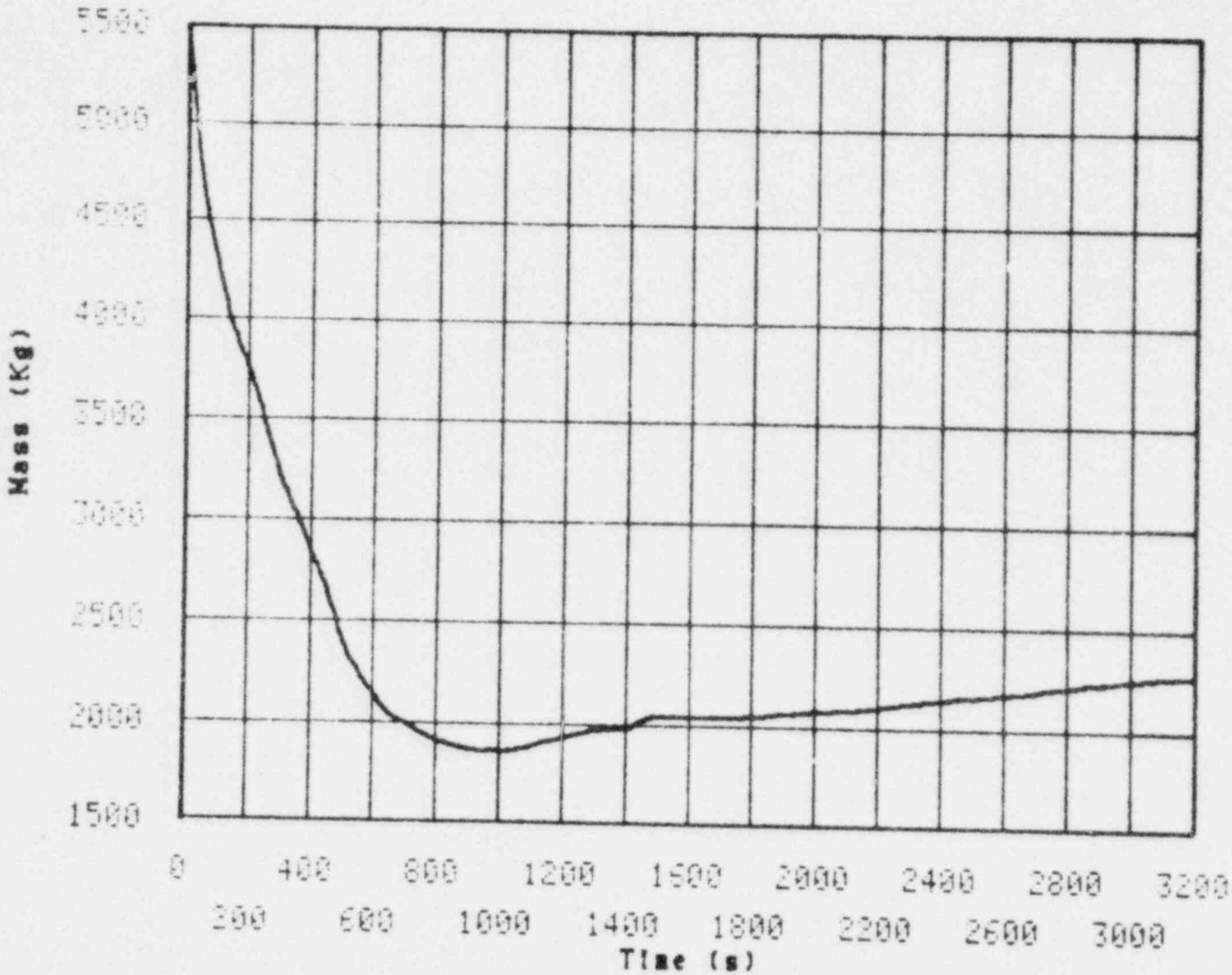


Figure 13. Coolant mass inventory in primary system.

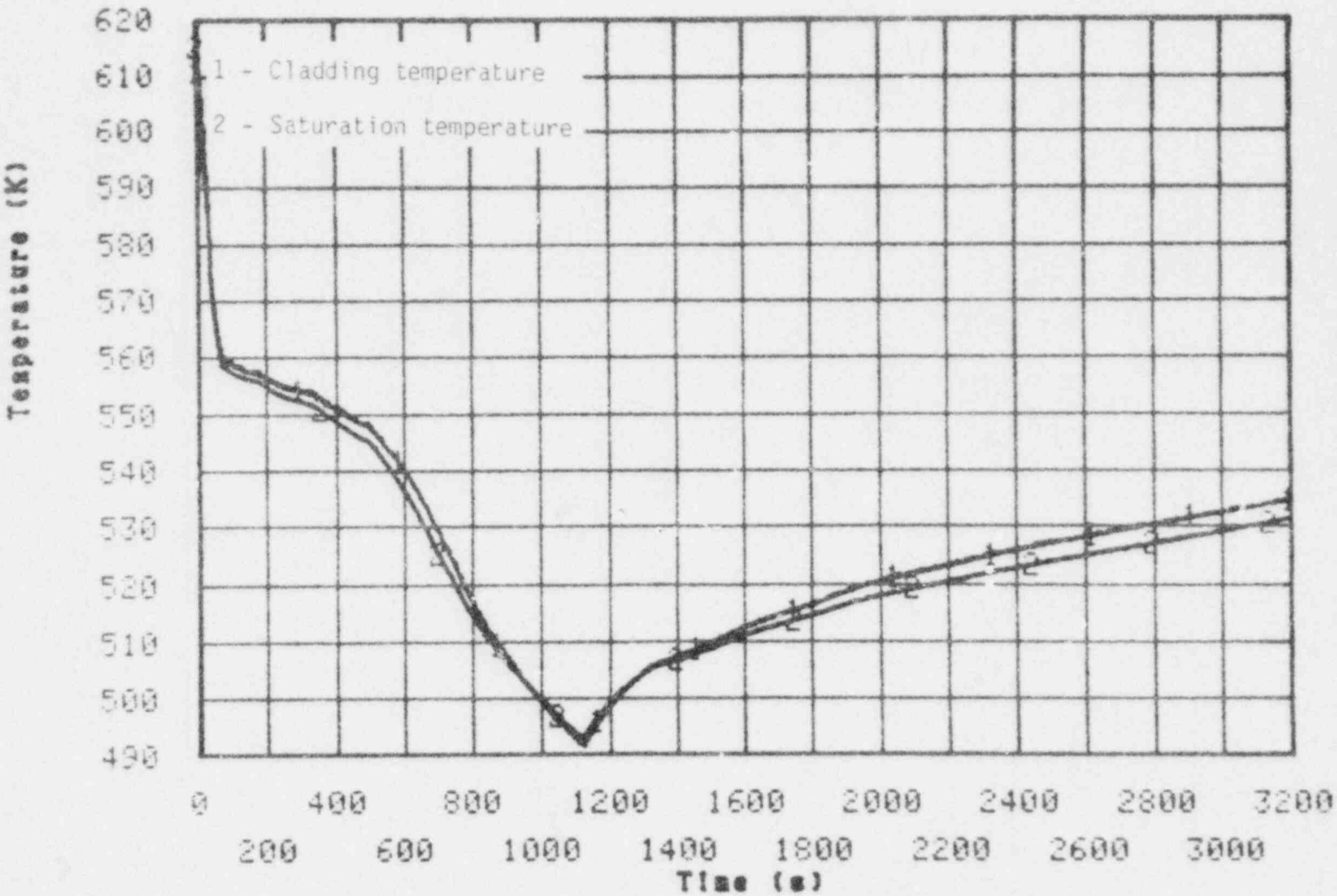


Figure 14. Fuel rod cladding temperature at core midplane.

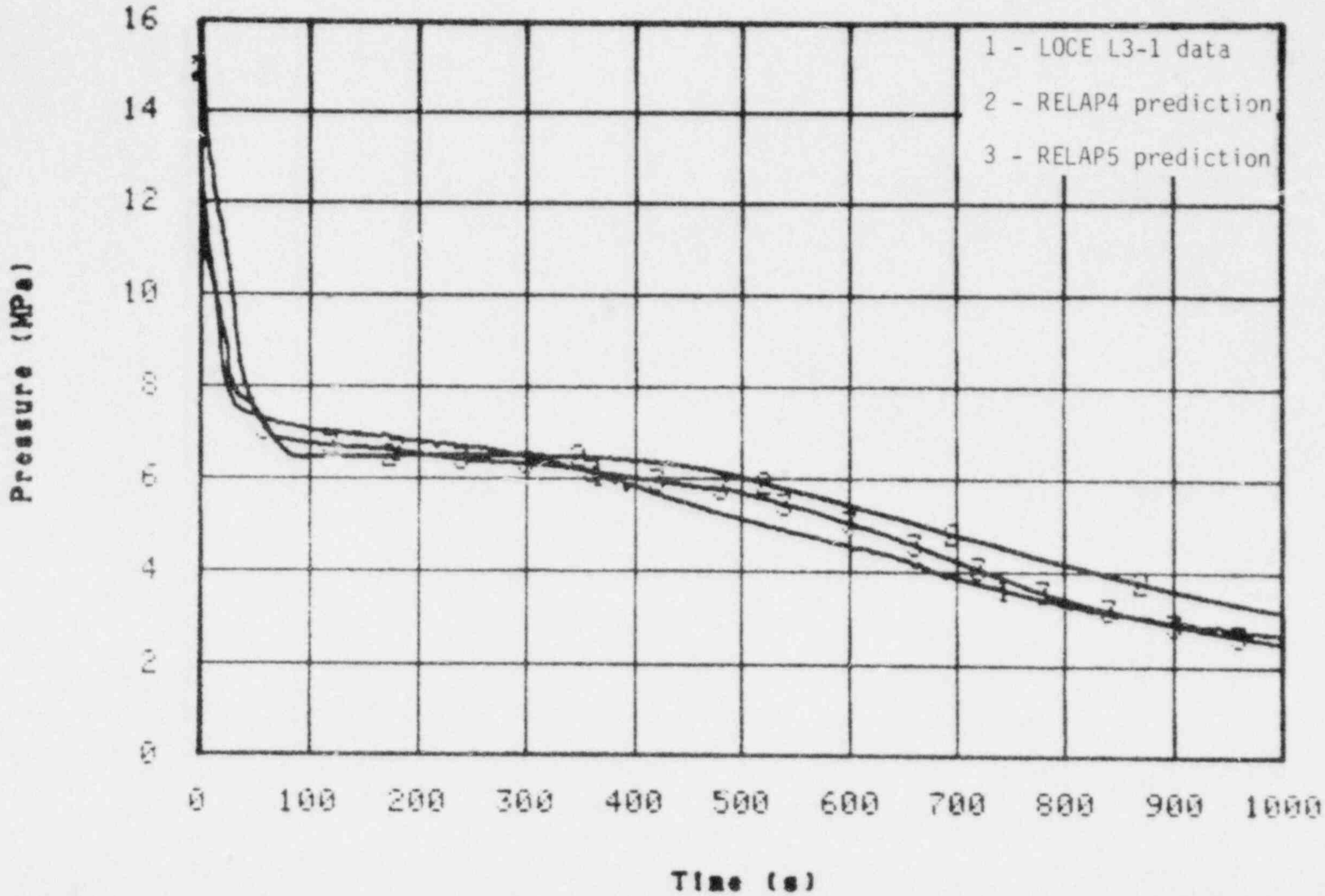


Figure 15. Pressure in reactor vessel upper plenum from RELAP4 and RELAP5 calculations and LOCE L3-1.

In single-phase natural loop circulation, the fluid density gradients are achieved by changes in fluid temperature. Consequently, a significant temperature increase from the inlet to the outlet of the core and a corresponding temperature decrease from the inlet to the outlet of the steam generator will be observed. Heat transfer from the core to the fluid and from the fluid to the steam generator tubes is not particularly efficient and quite dependent on flow velocity, which will be quite low. Consequently, temperature differences of several degrees will exist across the fluid-wall interfaces.

During two-phase natural loop circulation, both the vapor and liquid will be at saturation temperature and the density gradients will be achieved by a change from liquid to vapor in the core and vapor to liquid in the steam generator, giving much larger density gradients than in single phase. Heat transfer from the fuel rods to the coolant will be by nucleate boiling and from the fluid to the steam generator tubes by condensation, both very efficient heat transfer mechanisms resulting in fluid-wall interface temperature differences as low as 1 or 2^{0}. Because of the efficiency of the two-phase natural loop circulation, the temperature difference between the primary and secondary systems will be very small.

In the simplest form of two-phase natural loop circulation, the fluid will be two-phase around the entire loop which will be characterized by no temperature difference from the core inlet to outlet or from the steam generator inlet to outlet. It is possible, especially under transient conditions, for the vapor to superheat at the core outlet or for subcooled liquid to exist at the steam generator outlet. In either case, a temperature difference will then exist across the core and steam generator.

Reflux is a special case of two-phase natural loop circulation that occurs when the two-phase fluid is condensed on the steam generator inlet side and the liquid falls back down the inlet side. If continuous counter-current flow is established in the hot leg with vapor flowing toward the steam generator in the top of the pipe and liquid flowing in the bottom of the pipe toward the reactor vessel, then reflux flow is established. During reflux flow, some vapor may still be condensed on the outlet side of the steam generator, and the liquid flow will continue around the loop.

LOCE L3-5/L3-5A is very interesting because it exibits several of the above modes of natural loop circulation with very smooth transitions between these modes.

Figure 16 shows an overlay of predicted primary system pressure and secondary system pressure for the LOCE L3-5/L3-5A transient. This figure shows the two time periods when natural loop circulation is possible, that is, when the primary system saturation pressure is greater than the secondary system saturation pressure.

Single-phase natural loop circulation occurred very briefly after the pumps stopped at 14 s and before significant saturation occurred in the intact loop hot leg and reactor vessel upper plenum at about 60 s. As the system pressure dropped, more vapor was generated, and two-phase natural loop circulation was established. Figure 17 shows the liquid and vapor velociti[illegible]e intact loop hot leg. At [illegible]ut 140 s, the liquid velocity was nega[illegible], while the vapor velocity rema[illegible]d positive, indicating the establishment of reflux flow. There was a definite cessation of reflux flow at 465 s, just 60 s before natural loop circulation terminated as the primary pressure dropped below the secondary pressure.

Natural loop circulation was again established after 1320 s, but during this period, reflux flow was not indicated by the prediction. Figure 18 shows the liquid and vapor velocities in the inlet side of the steam generator. Note that in this vertical position, negative liquid flow was indicated along with positive vapor flow, but the liquid did not continue to flow into the hot leg.

Figure 19 shows the heat transfer from primary side to secondary side across the steam generator. The negative heat transfer between 500 and 1300 s indicates heat flow from the secondary system to the primary system during that time period, when the primary pressure was lower than the

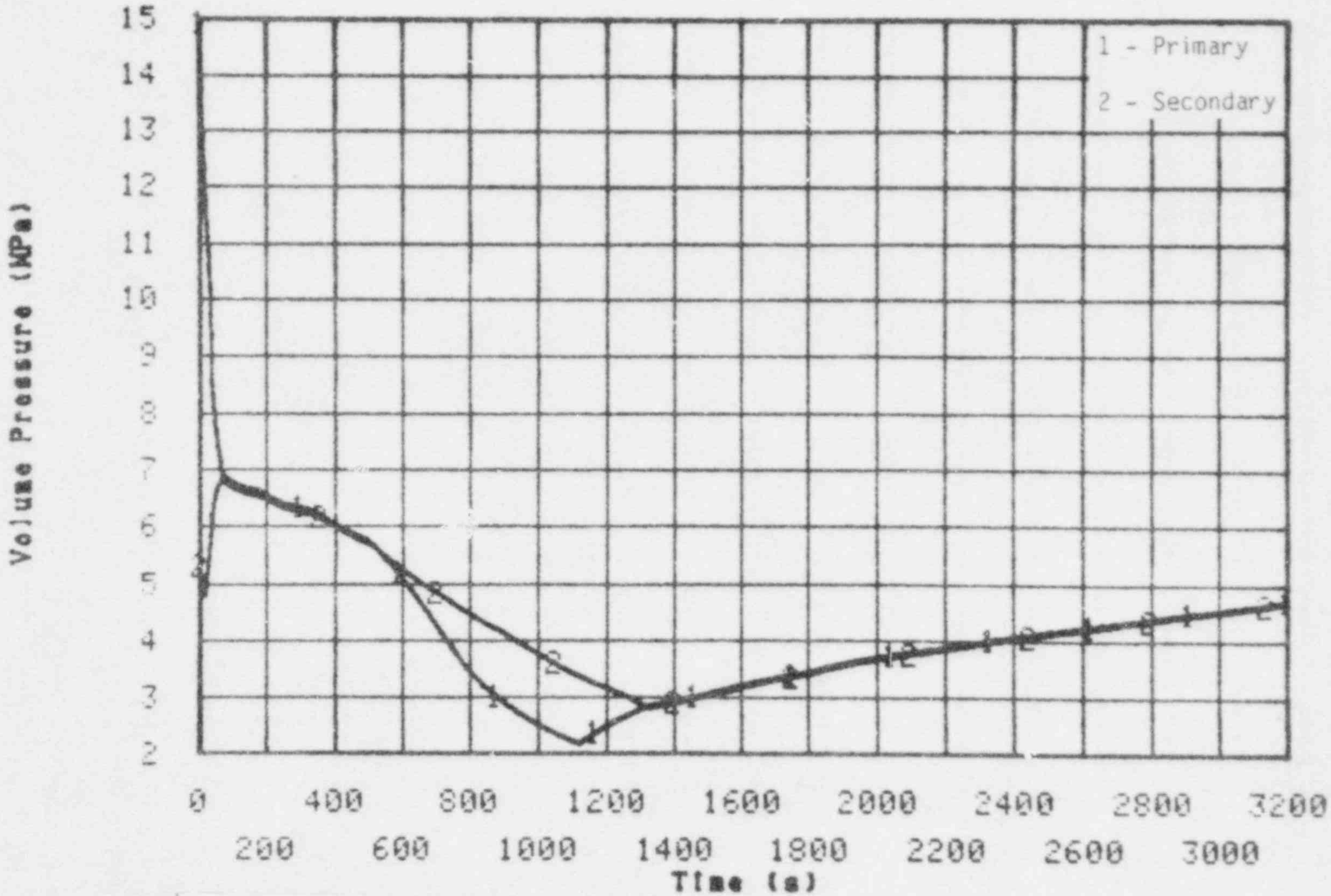


Figure 16. Pressure response in primary and secondary systems.

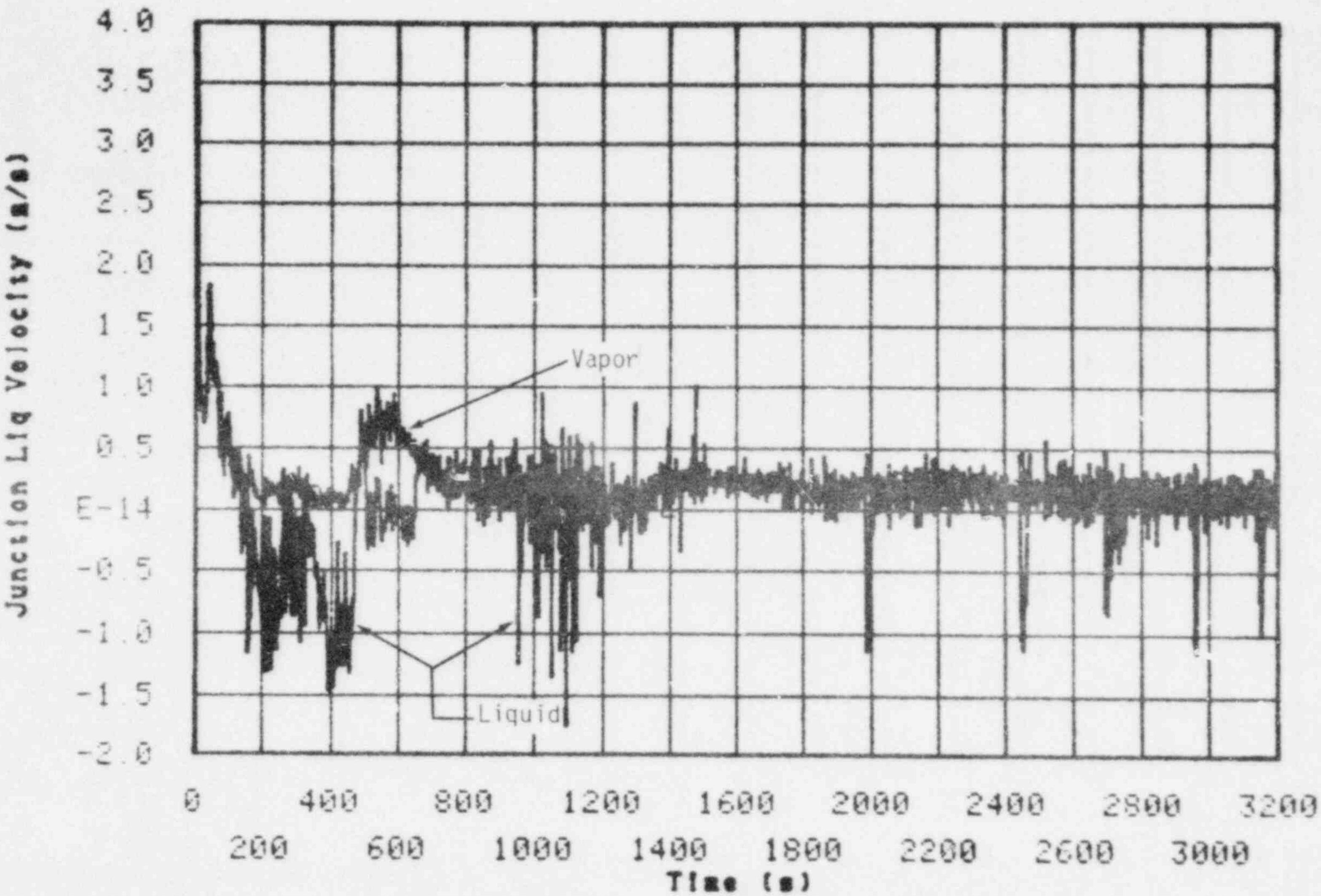


Figure 17. Liquid and vapor velocities in intact loop hot leg.

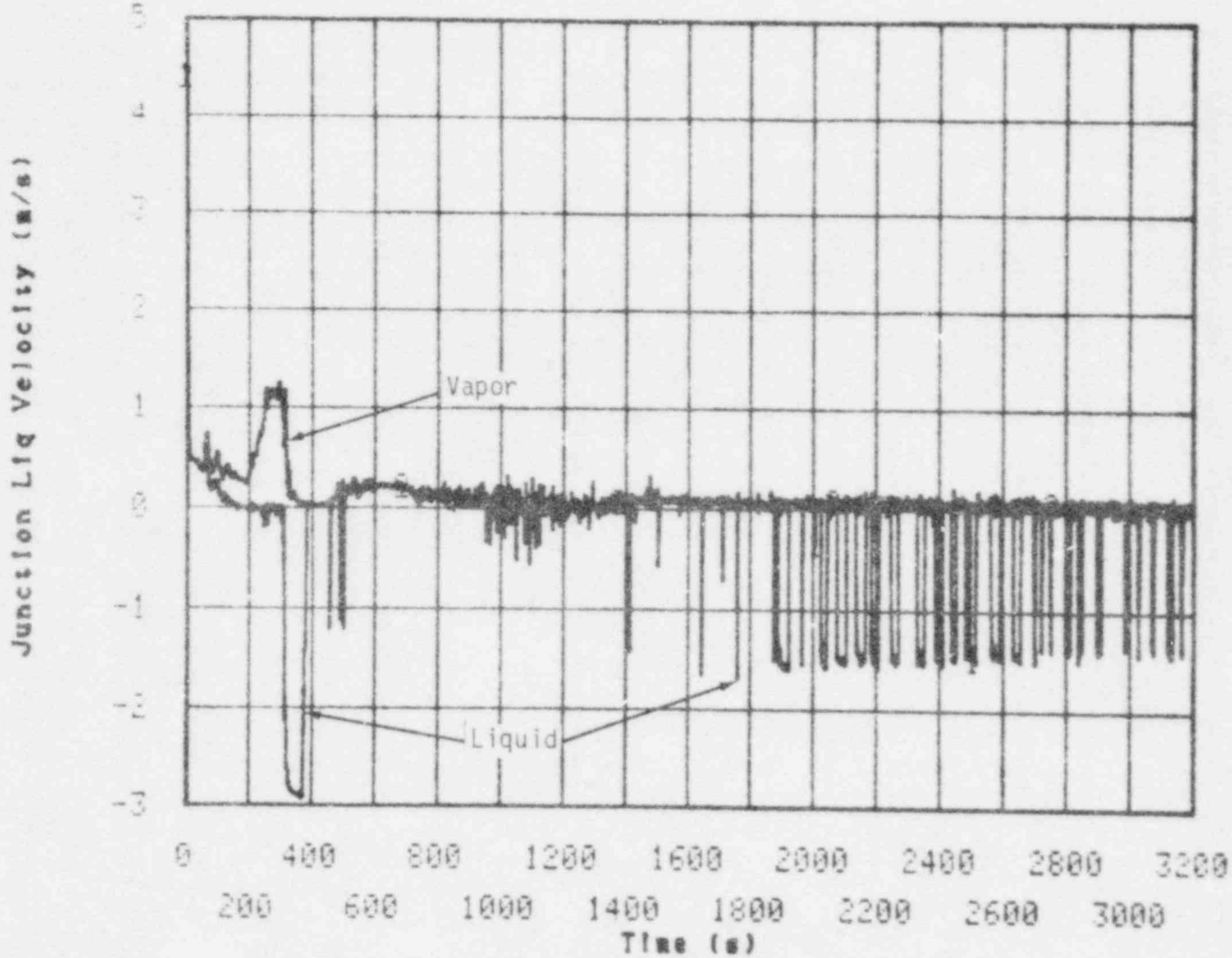


Figure 18. Liquid and vapor velocities in steam generator inlet side.

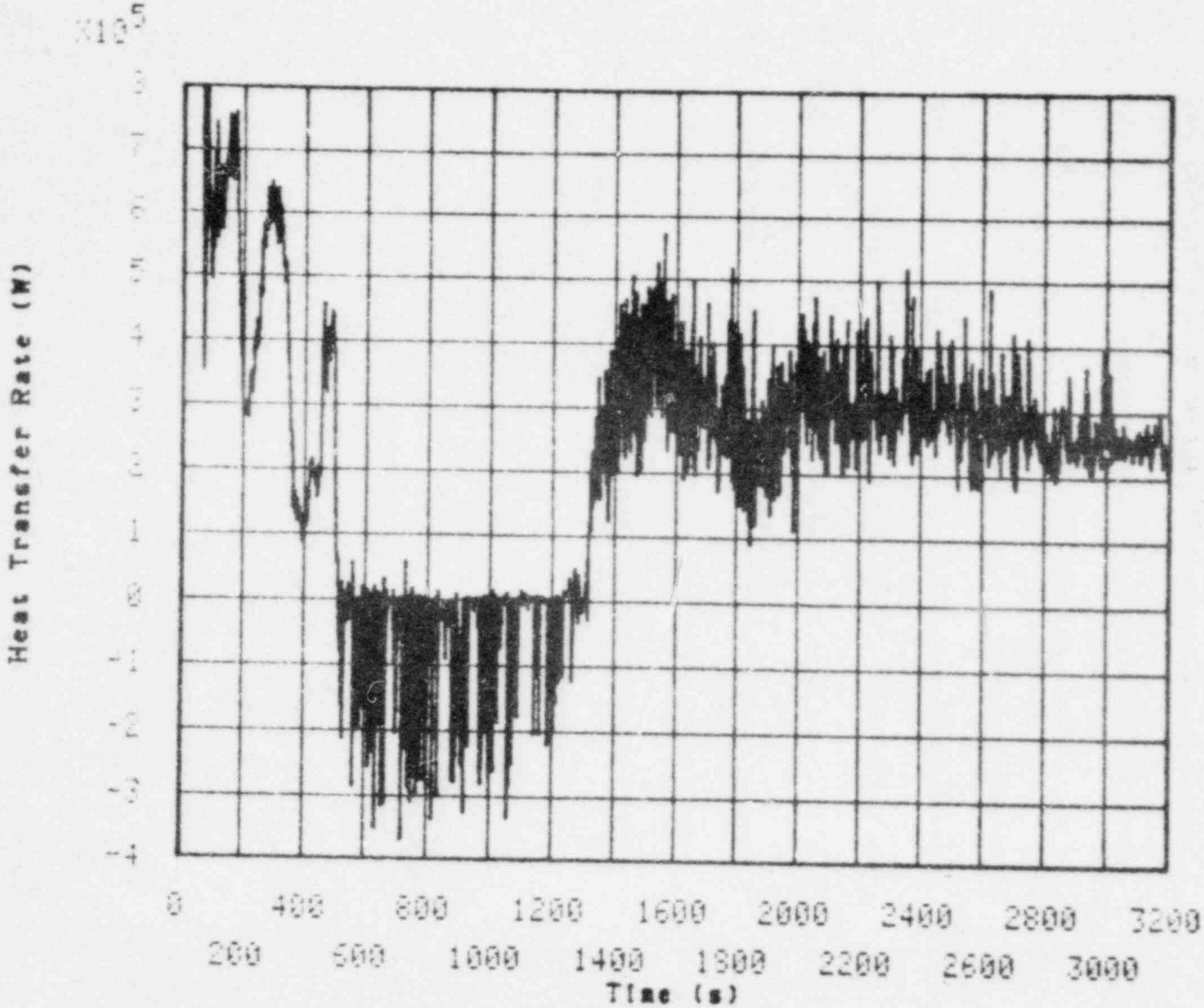


Figure 19. Heat transfer from primary system to secondary system.

secondary pressure. The reactor vessel fluid temperature difference and density difference are shown in Figures 20 and 21, respectively. The same information is shown for the primary side of the steam generator in Figures 22 and 23. Collapsed liquid level in the core is shown in Figure 24. Collapsed liquid is an analytical quantity obtained by assuming the liquid and vapor are completely separated, with the vapor above the liquid. It is useful for calculation of mass inventory, but in reality, there was some vapor below this level and some liquid above. The vapor void fraction for the control volume just below the reactor vessel nozzles is shown in Figure 25. This figure confirms that there will be a very high void two-phase mixture at the hot leg nozzle elevation.

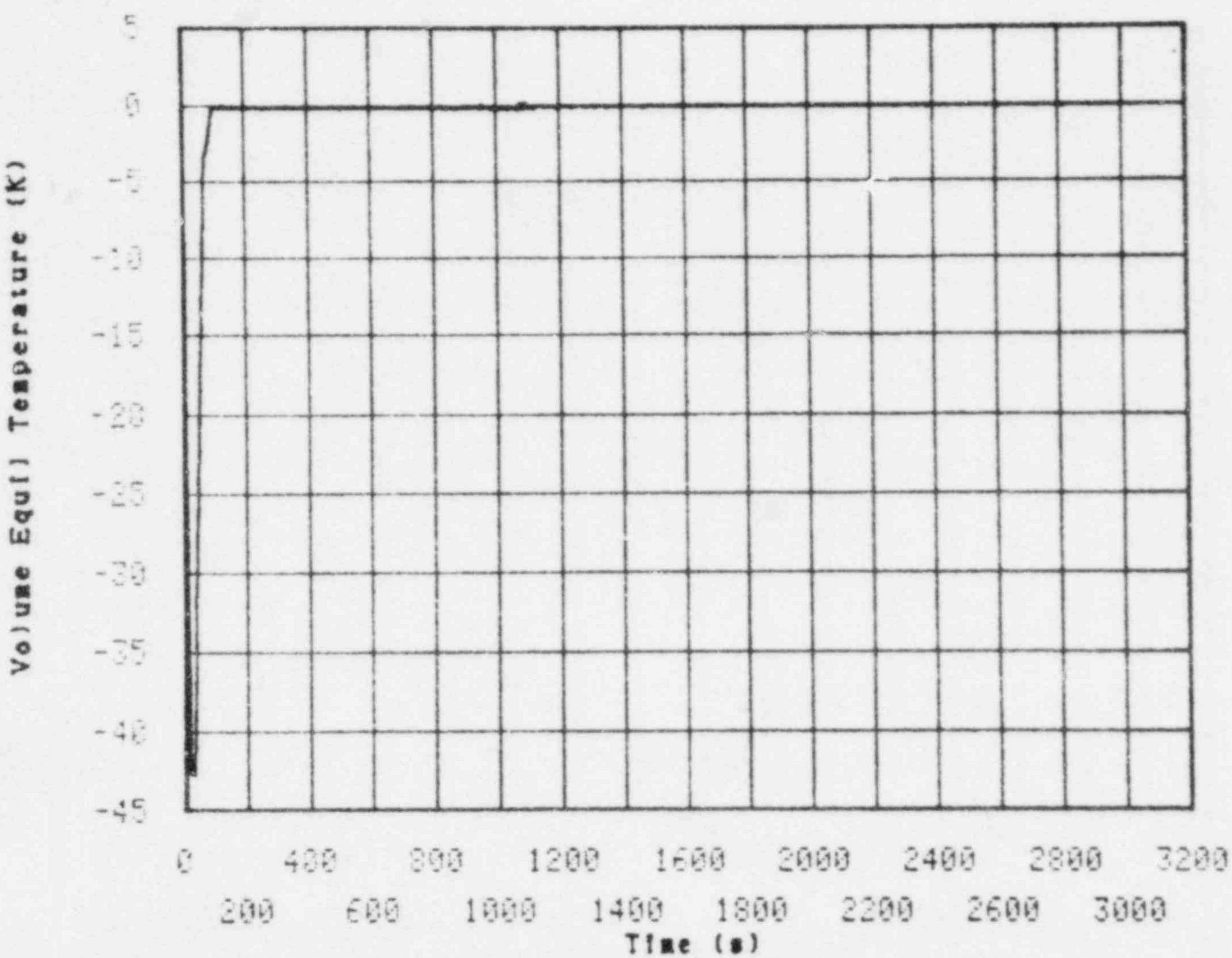


Figure 20. Fluid temperature difference in reactor vessel.

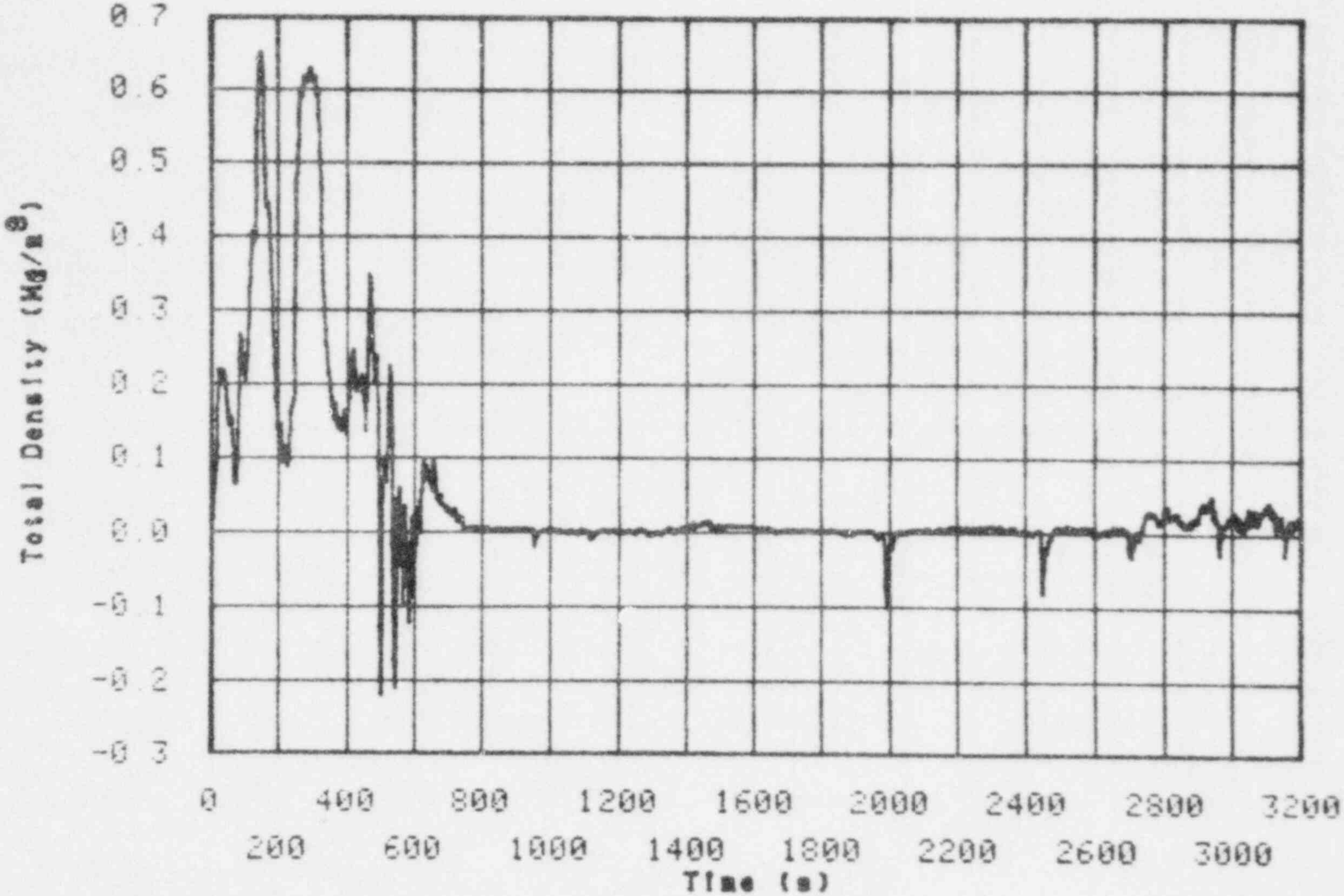


Figure 21. Fluid density difference in reactor vessel.

POOR ORIGINAL

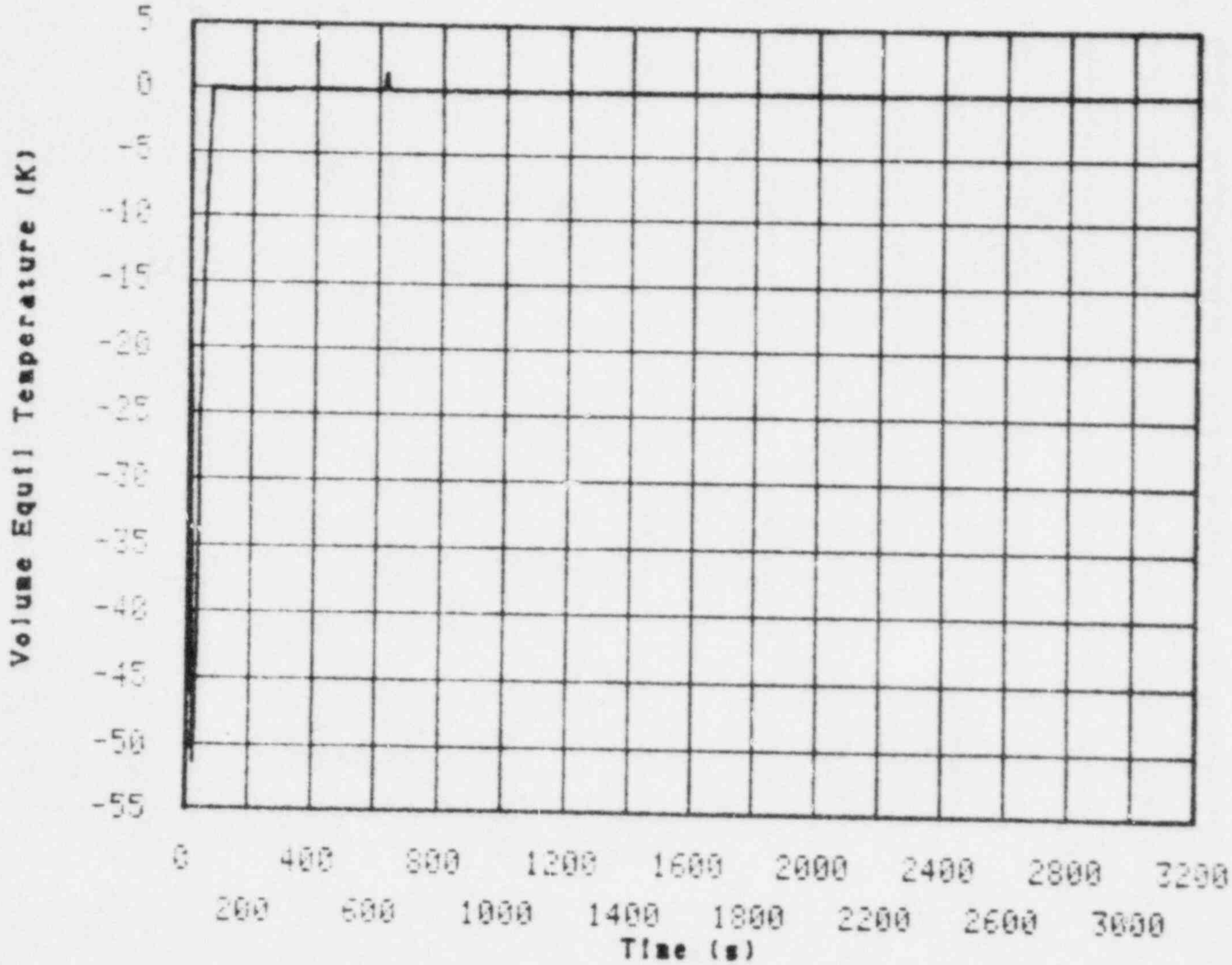


Figure 22. Fluid temperature difference in steam generator primary side.

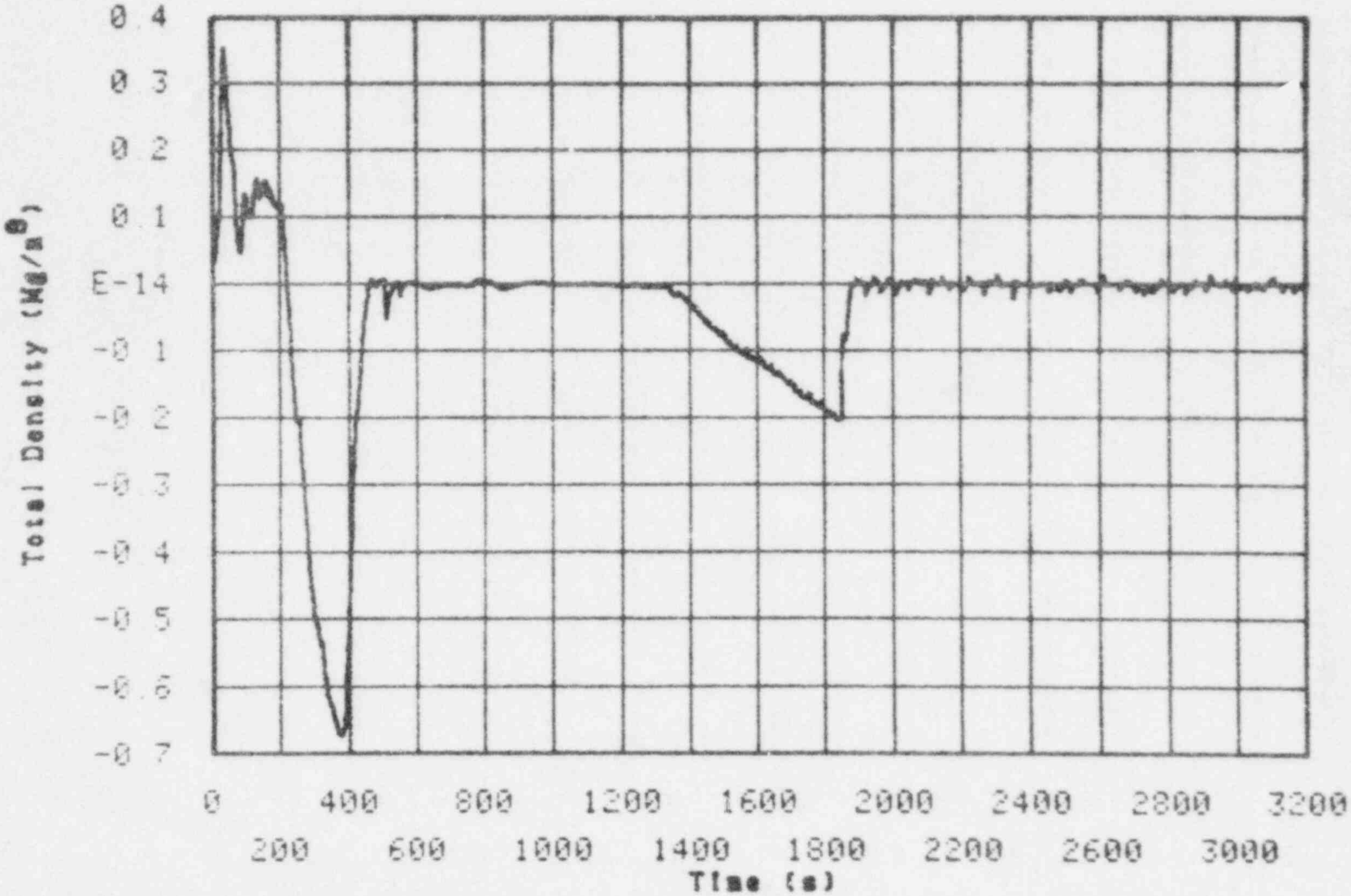


Figure 23. Fluid density difference in steam generator primary side.

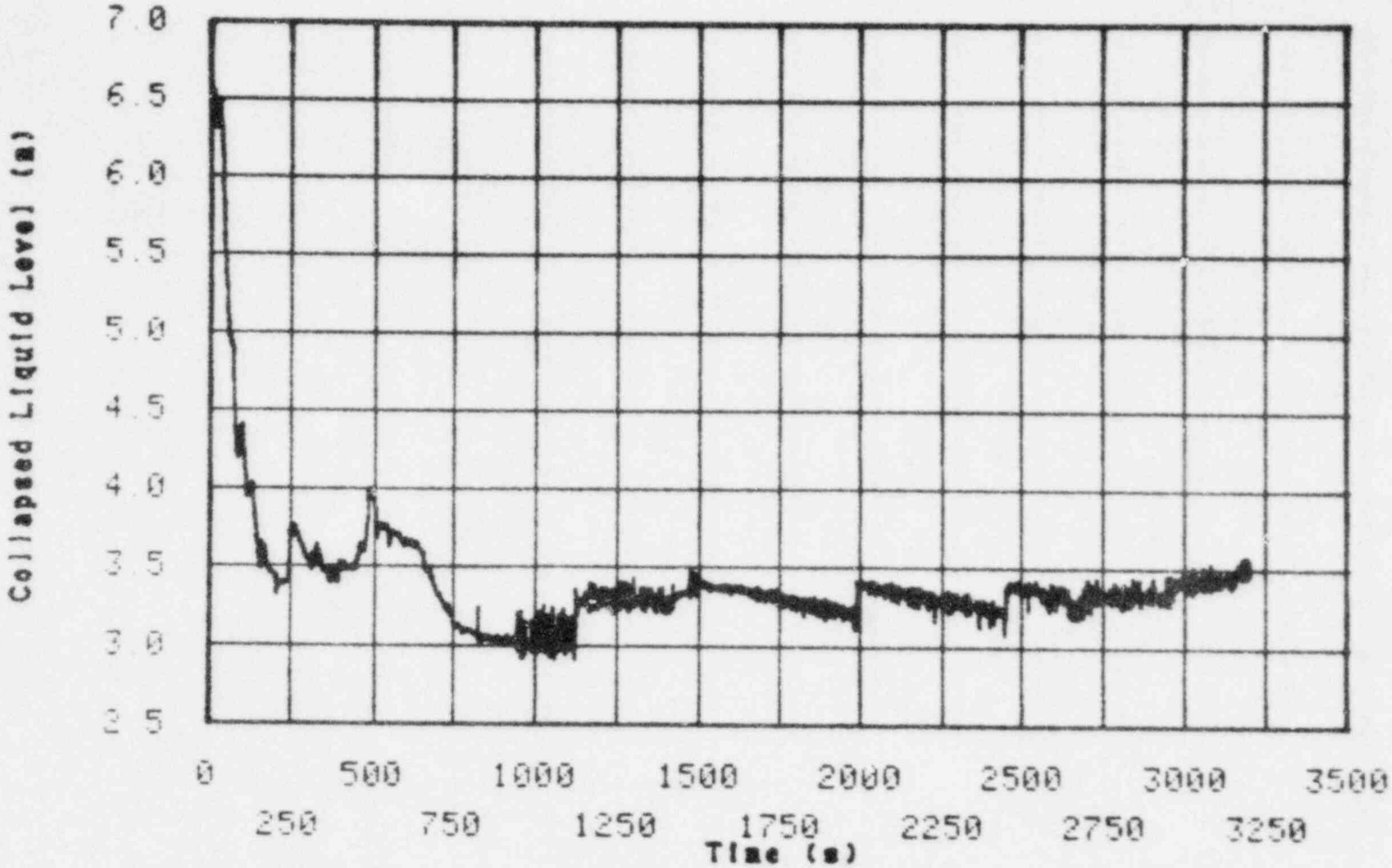


Figure 24. Collapsed liquid level in reactor vessel.

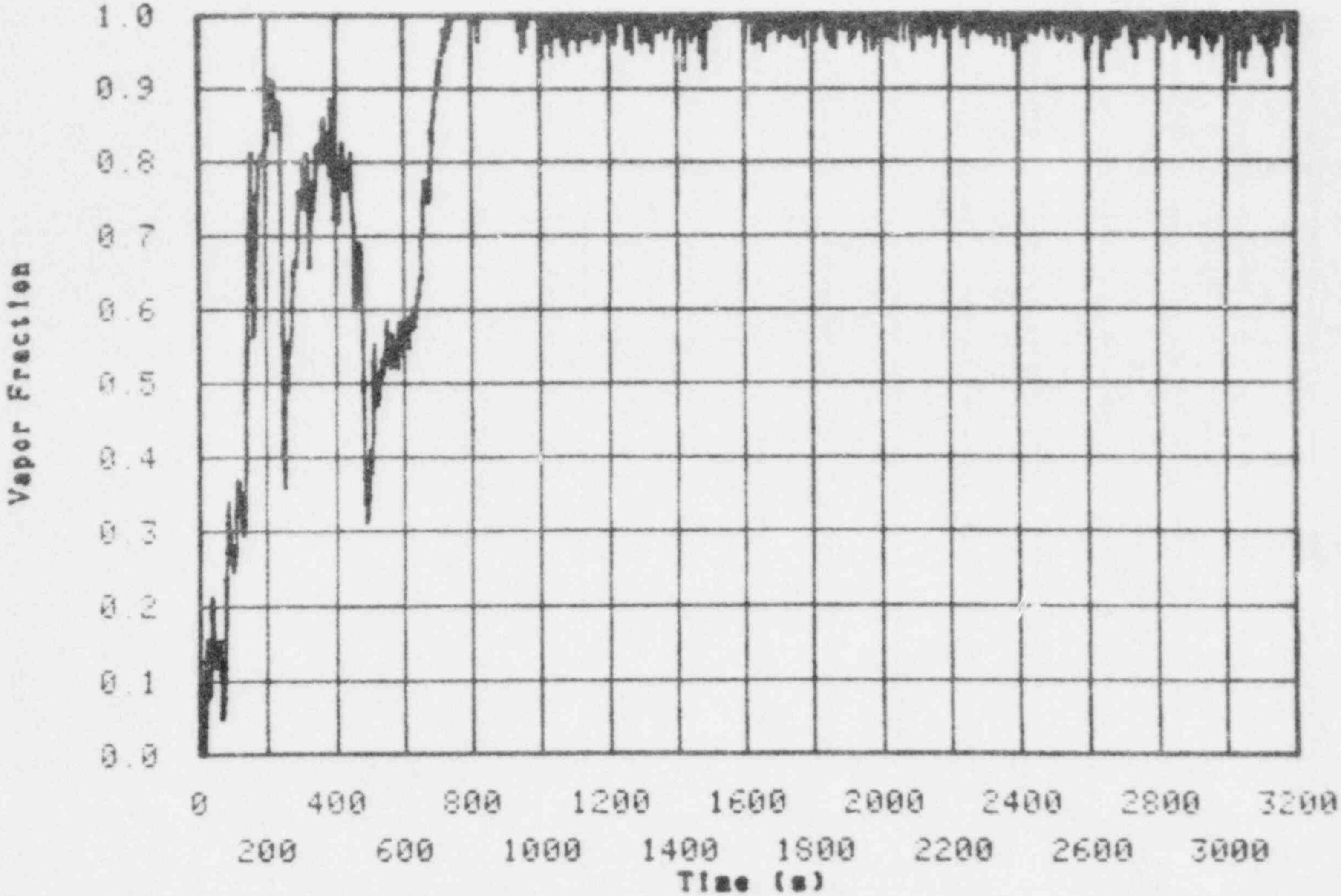


Figure 25. Void fraction in reactor vessel upper plenum.

4. CONCLUSIONS

The RELAP5 calculations indicate that LOCE L3-5/L3-5A will successfully meet its objectives. Several natural loop circulation modes will be demonstrated and they will be effective in removal of decay heat. Transition from mode to mode will be smooth. The core will remain covered, with the fuel cladding temperature predicted to follow saturation temperature throughout the entire transient.

5. REFERENCES

1. R. S. Semken, LOFT Experiment Operating Specification, Small Break Tests Series L3, Nuclear Test L3-5/L3-5A, NE L3 Series, EOS, Rev. 0, September 1980.

2. P. D. Bayless, J. B. Marlow, R. H. Averill, Experiment Data Report for LOFT Nuclear Small Break Experiment L3-1, NUREG/CR-1145, EGG-2007, January 1980.

3. D. L. Reeder, LOFT System and Test Description (5.5-ft Nuclear Core 1 LOCEs). NUREG/CR-0247, TREE-1208, July 1978.

4. E. J. Kee, P. J. Schally, L. Winters, Base Input for LOFT RELAP5 Calculations, EGG-LOFT-5199, July 1980.

5. D. L. Gillas and J. M. Carpenter, Experiment Data Report for LOFT Nuclear Small Break Experiment L3-7, NUREG/CR-1570, EGG-2049, August 1980.

6. B. D. Stitt and J. M. Carpenter, Experiment Data Report for LOFT Anticipated Transient Experiment L6-5, NUREG/CR-1520, EGG-2045, July 1980.

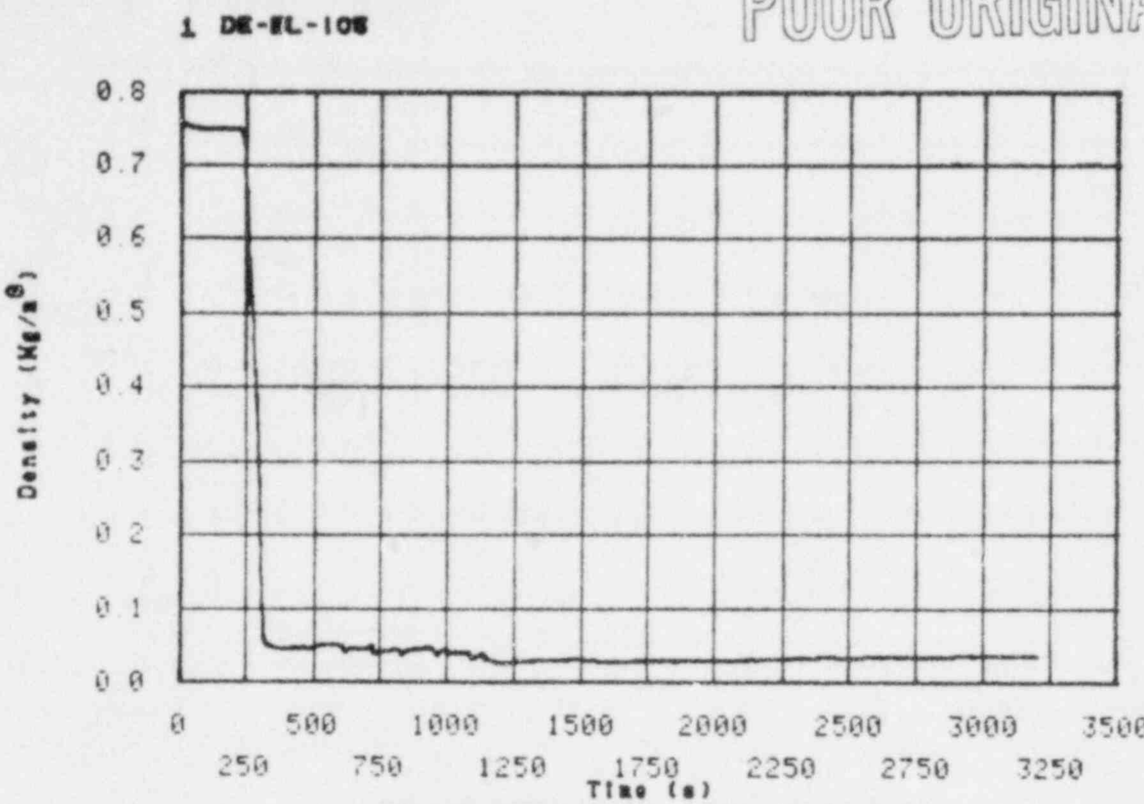
APPENDIX A

DETAILED TEST PREDICTION DATA FOR LOFT LOCE L3-5/L3-5A

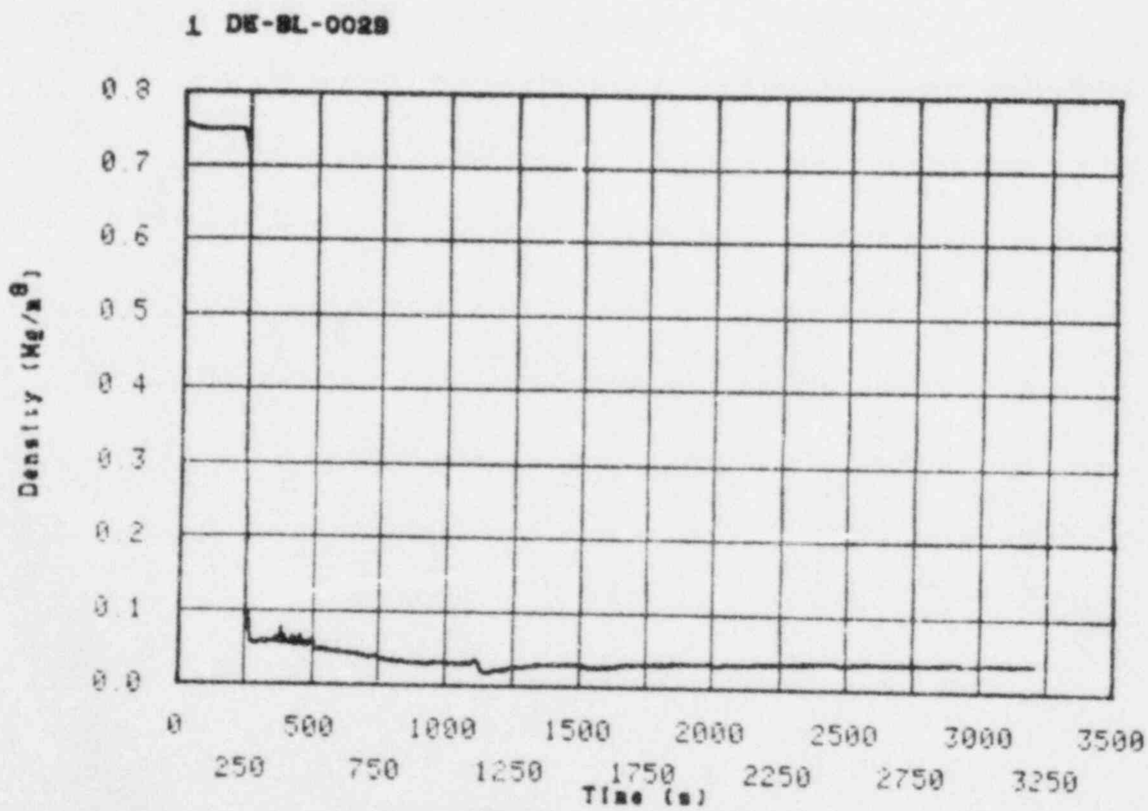
APPENDIX A

DETAILED TEST PREDICTION DATA FOR LOFT LOCE L3-5/L3-5A

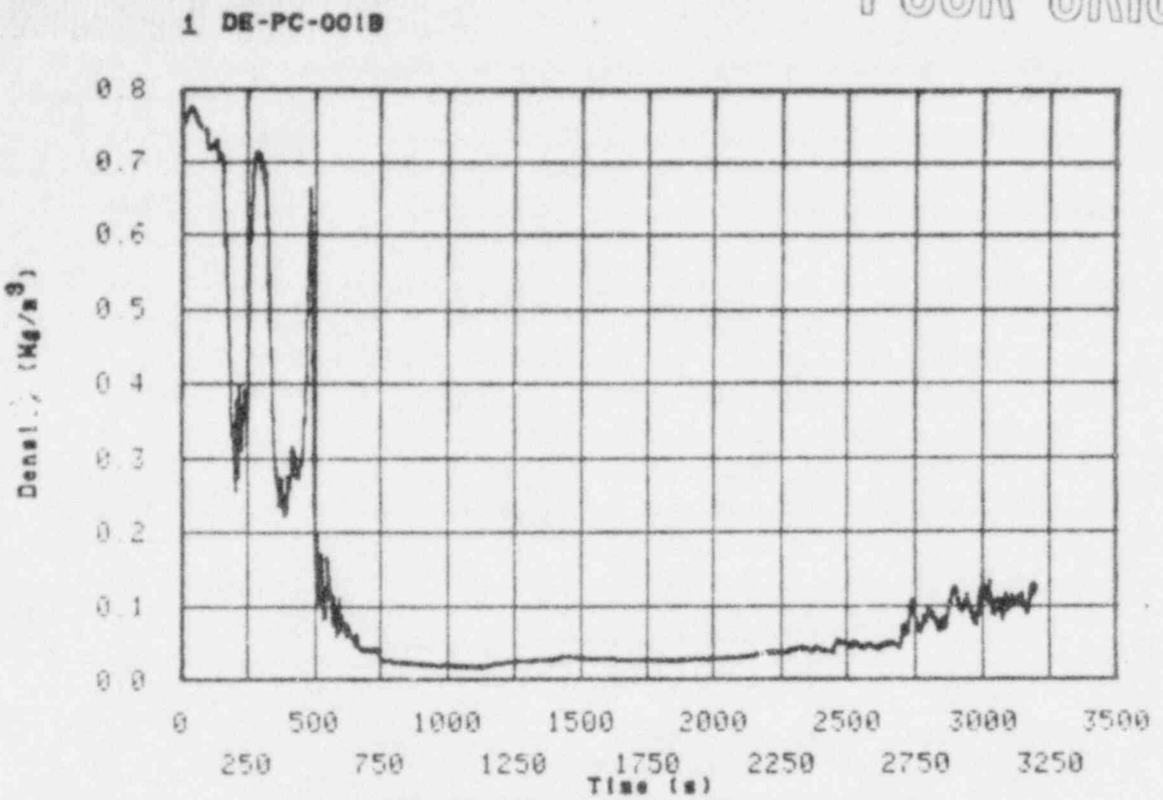
Detailed test prediction data for Loss-of-Coolant Experiment (LOCE) L3-5/L3-5A are provided in Figures A-1 through A-49 in this appendix. These figures are computer plots of the variables calculated for LOCE L3-5/L3-5A using RELAP5, and these data have been transmitted to the LOFT Data Bank for future comparison with experiment results. The calculated variables and figure numbers are as follows:



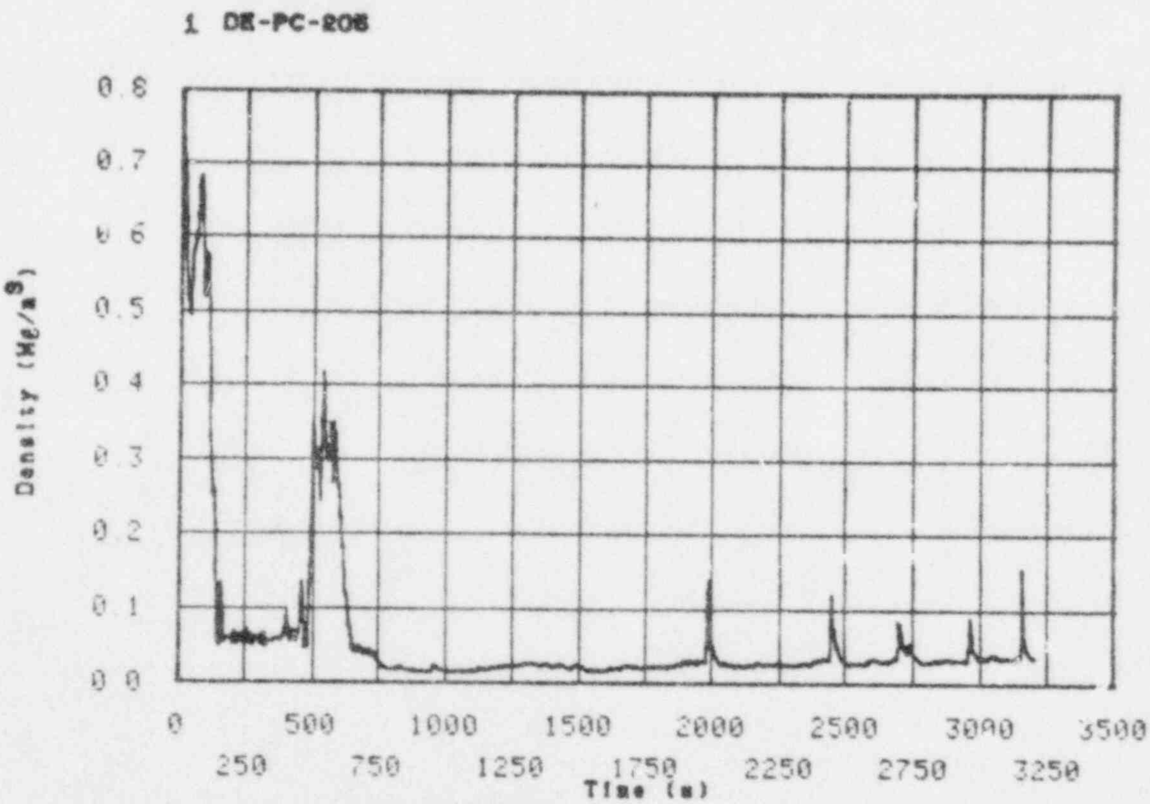
RELAP5 PREDICTION LOFT L8-5/L8-5A
Figure A-1. Ave. Density - BLCL.



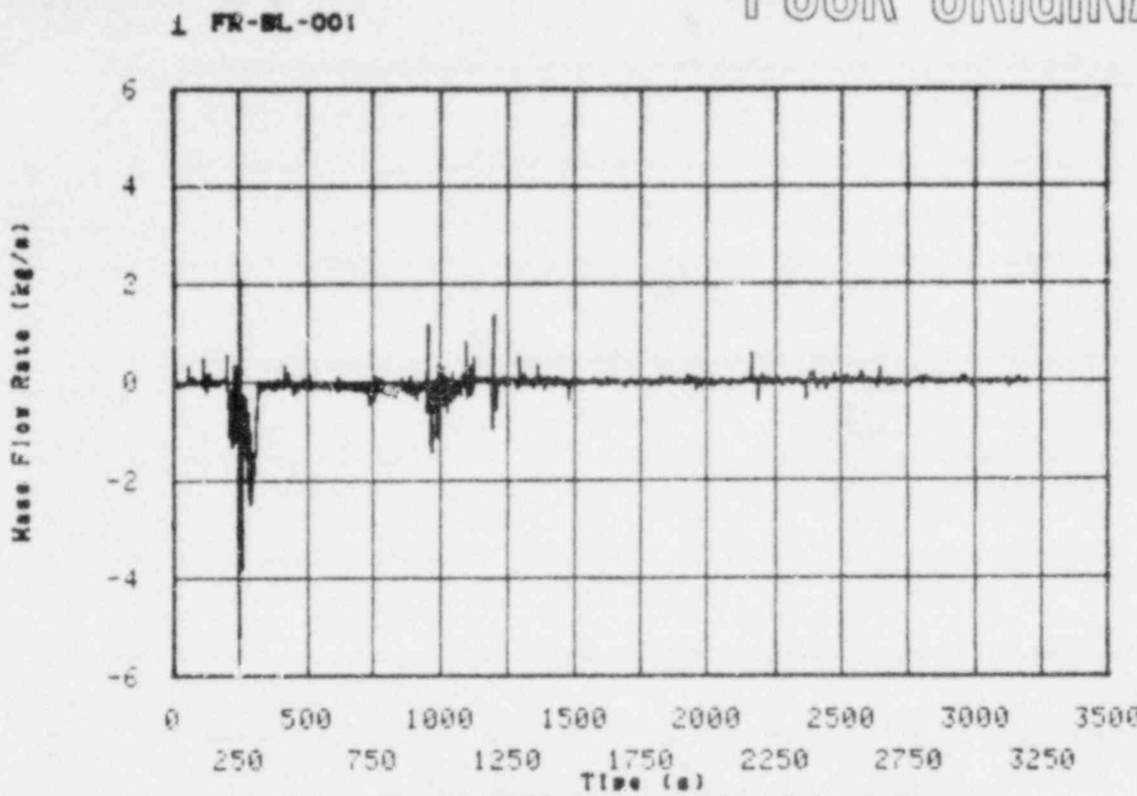
RELAP5 PREDICTION LOFT L8-5/L8-5A
Figure A-2. Ave. Density - BLHL.



RELAP5 PREDICTION LOFT L9-5/L9-5A
Figure A-3. Ave. Density - ILCL.

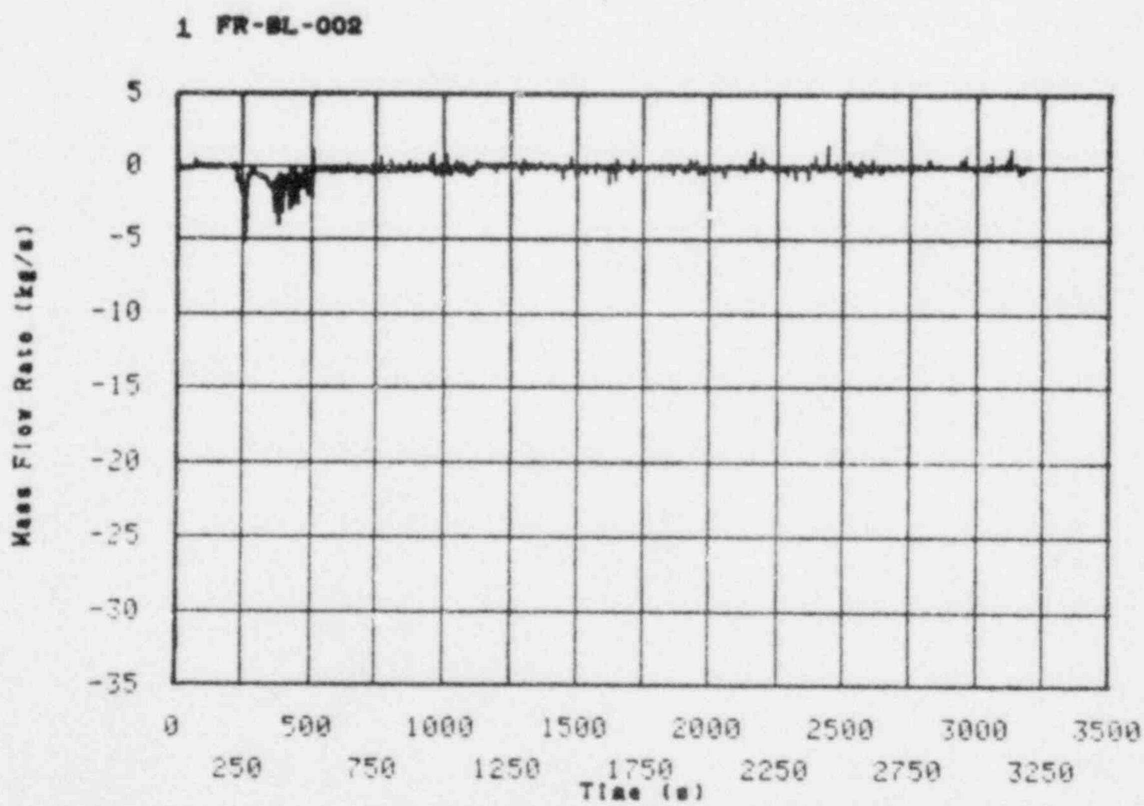


RELAP5 PREDICTION LOFT L9-5/L9-5A
Figure A-4. Ave Density - ILHL.



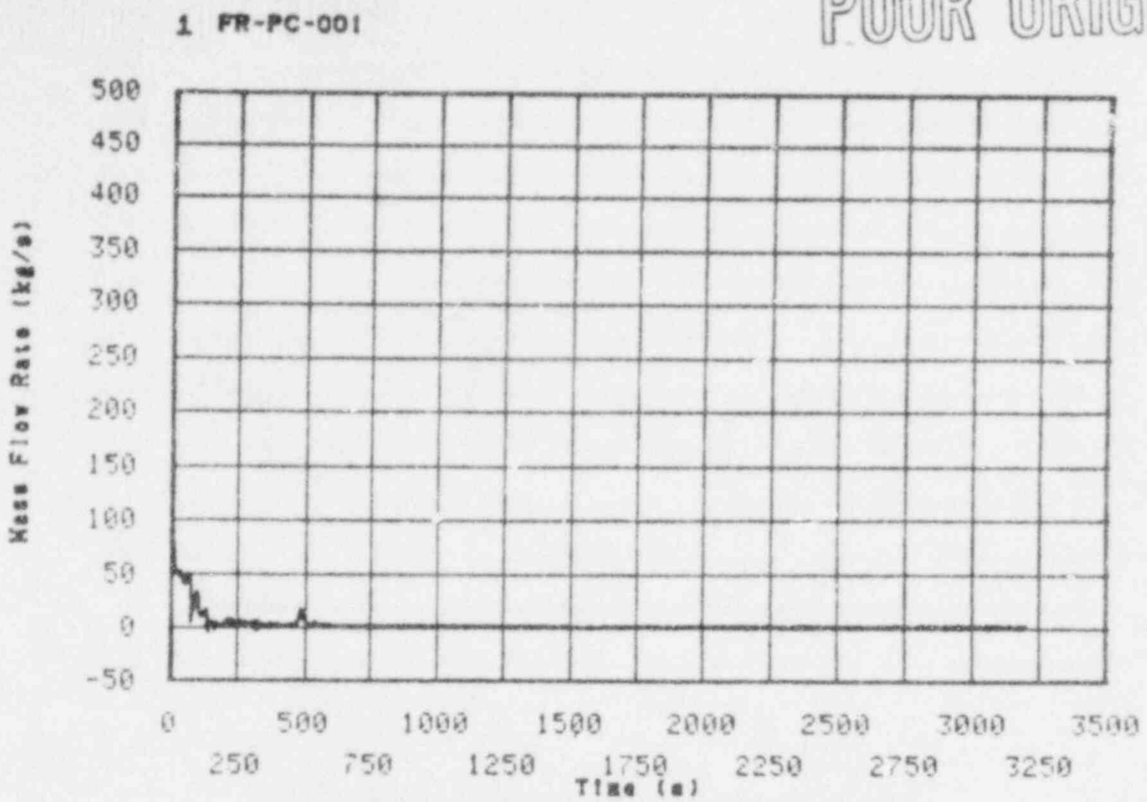
RELAP5 PREDICTI(N LØFT L8-5/L8-5A

Fig. A-5 Mass Flow Rate - BLCL

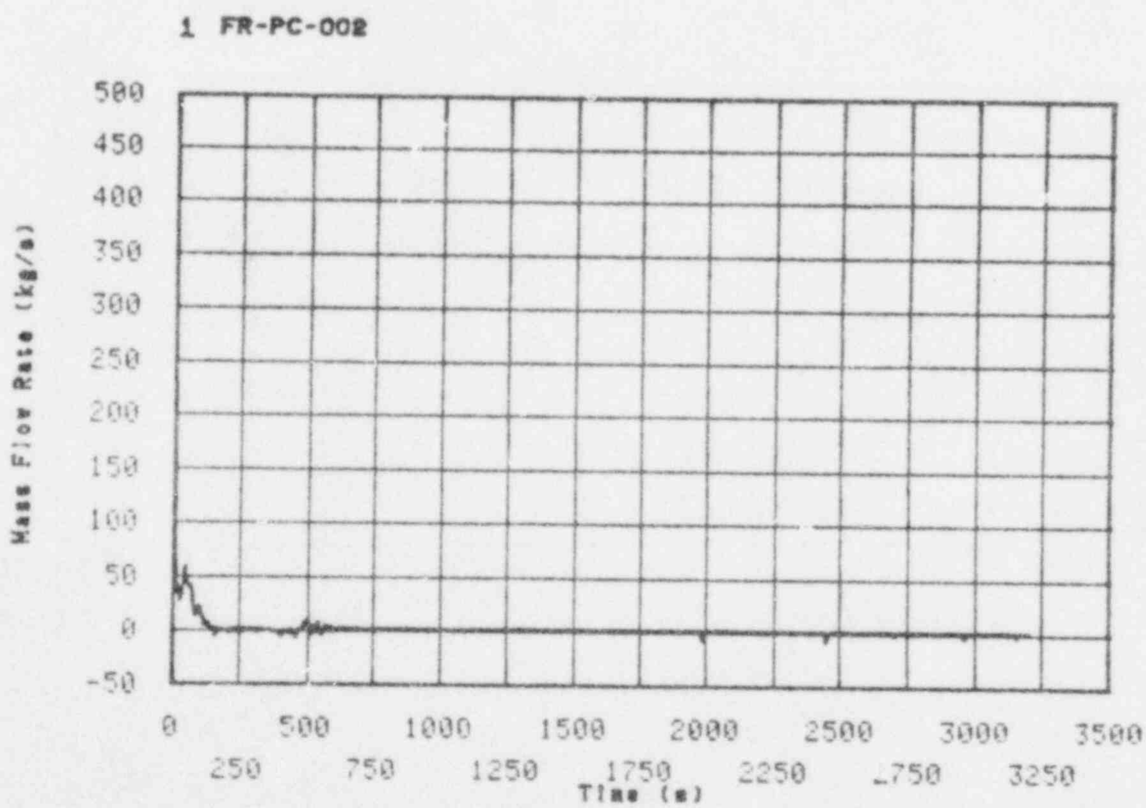


RELAP5 PREDIC.IØN LØFT L8-5/L8-5A

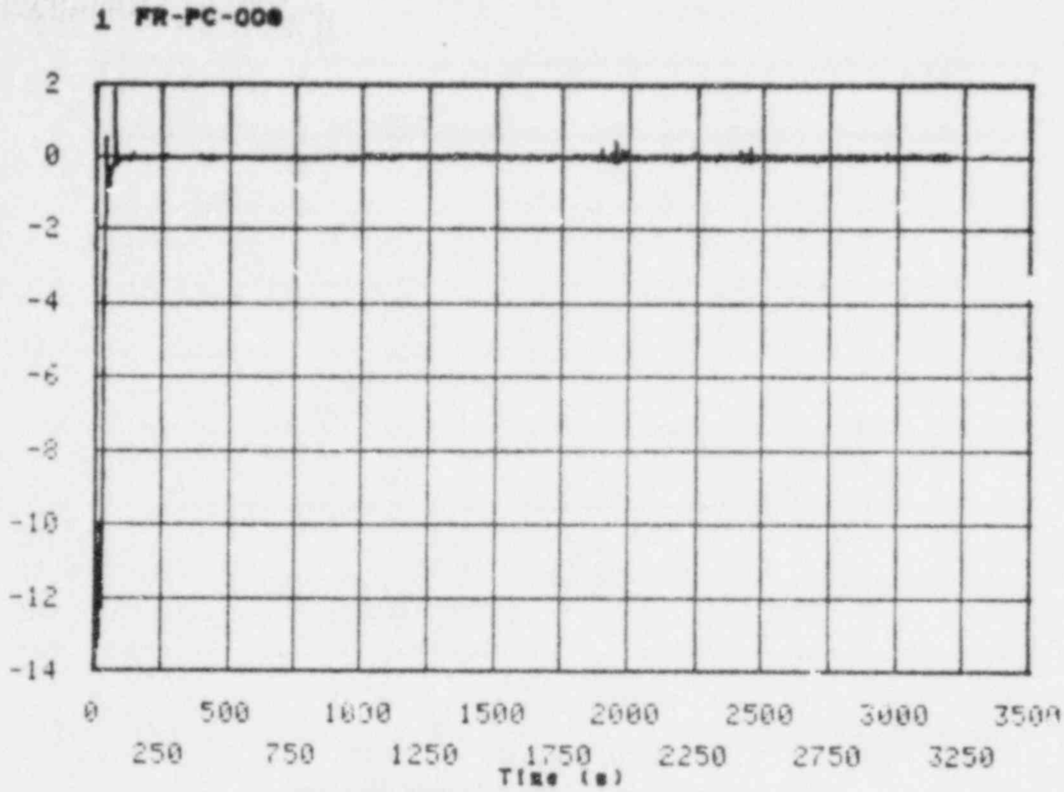
Fig. A-6 Mass Flow Rate - BLHL



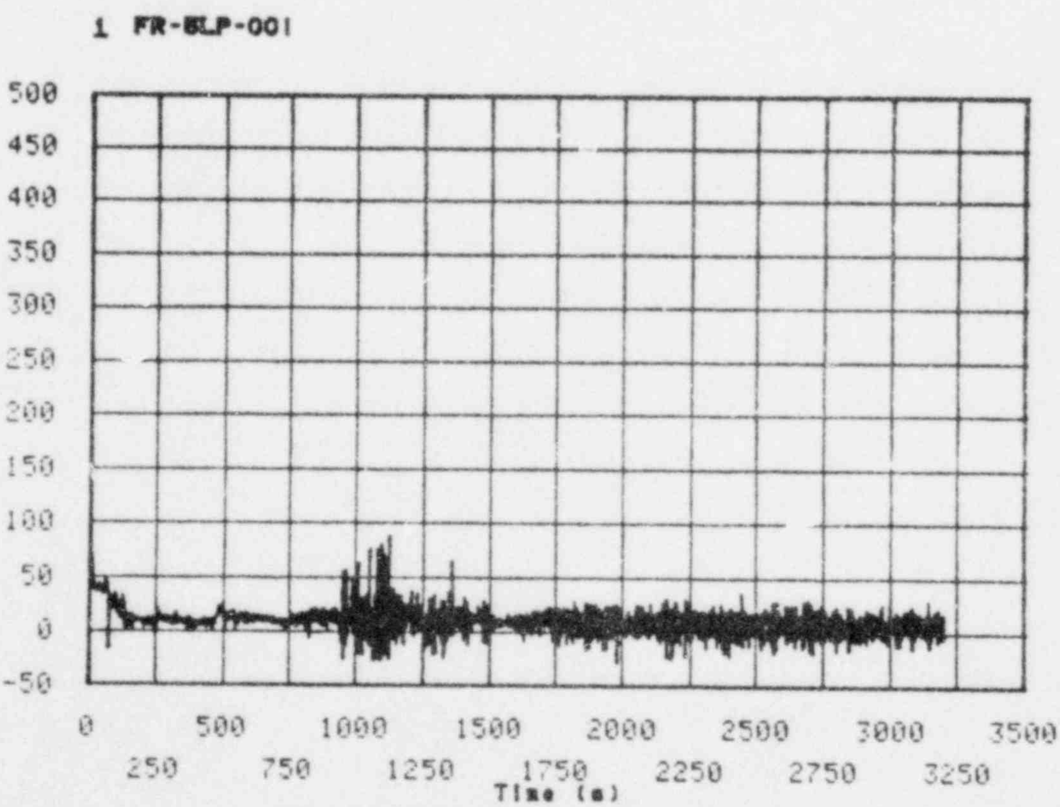
RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig. A-7 Mass Flow Rate - ILCL



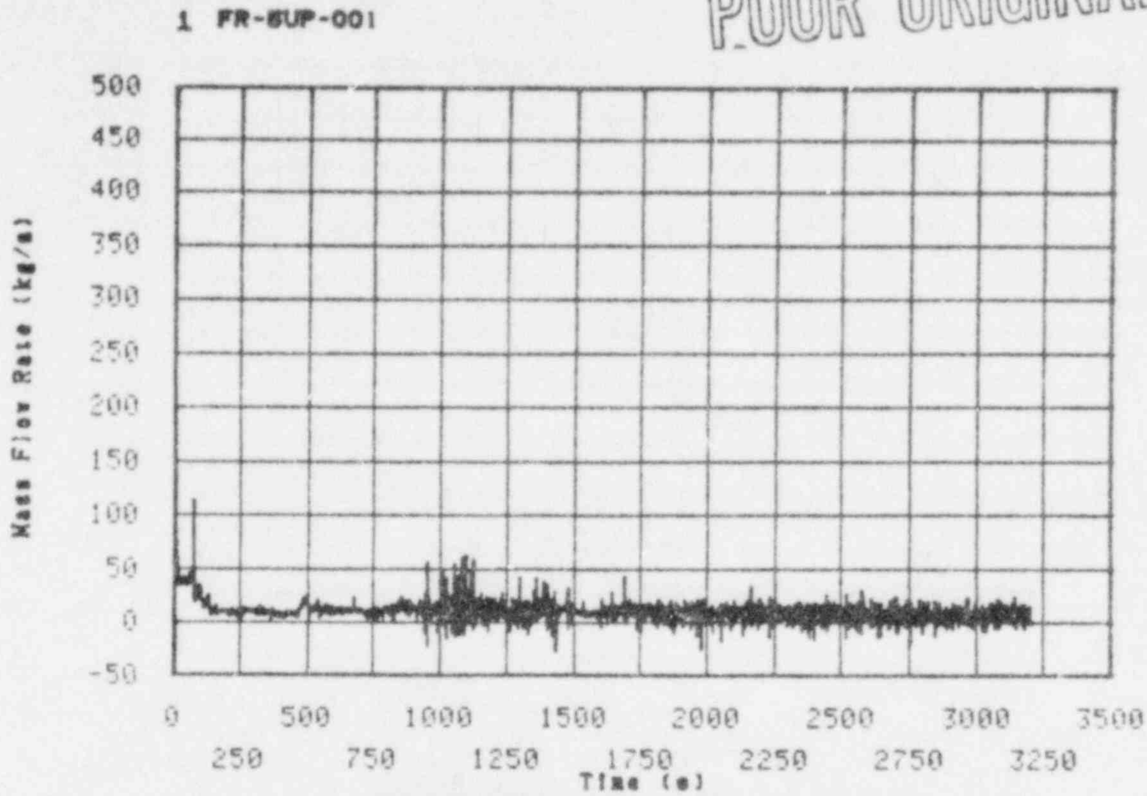
RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig. A-8 Mass Flow Rate - ILHL



RELAP5 PREDICTION LOFT LS-5/LS-5A
Fig. A-9 Pressurizer Surge Line

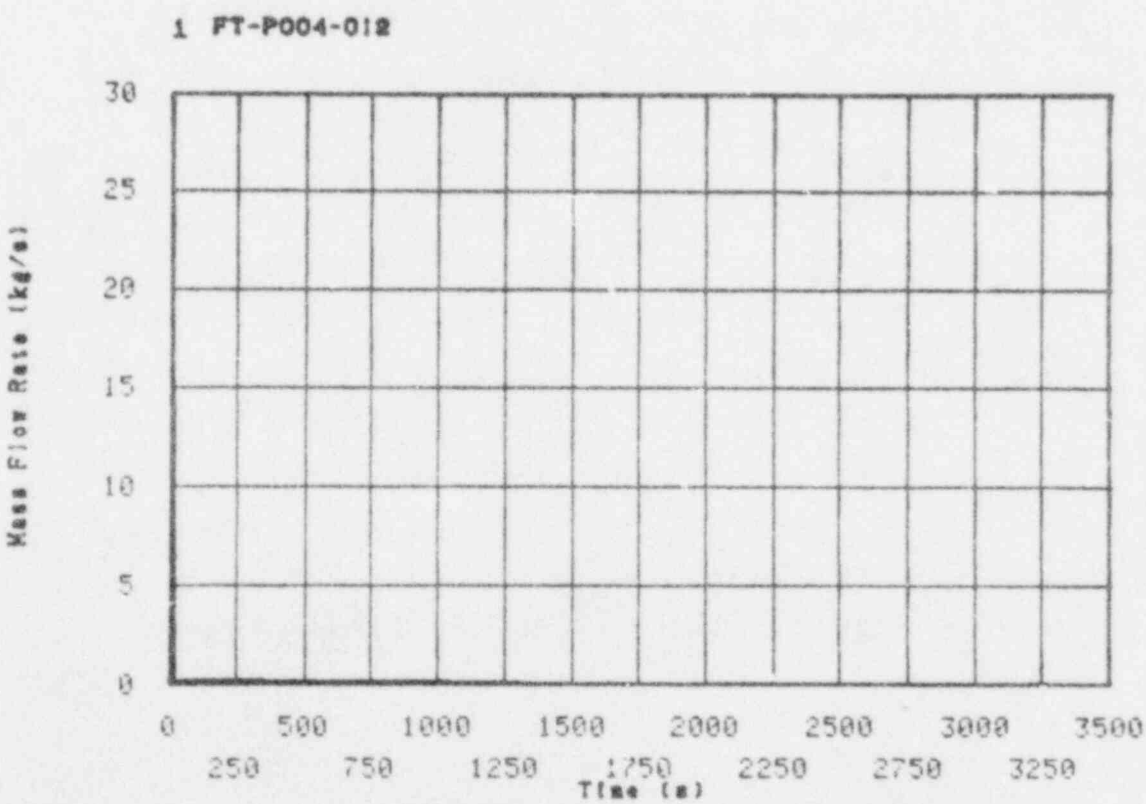


RELAP5 PREDICTION LOFT LS-5/LS-5A
Fig. A-10 Core Inlet



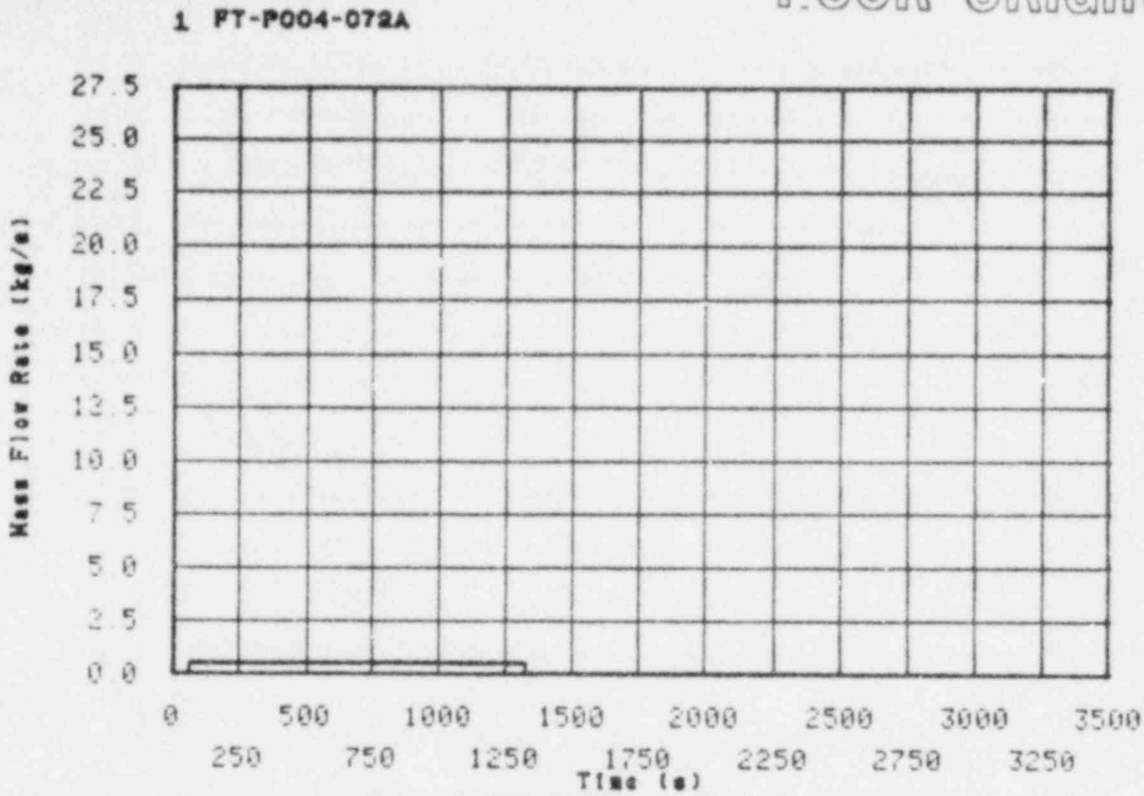
RELAP5 PREDICTION LOFT LS-5/LS-5A

Fig A-11 Core Outlet



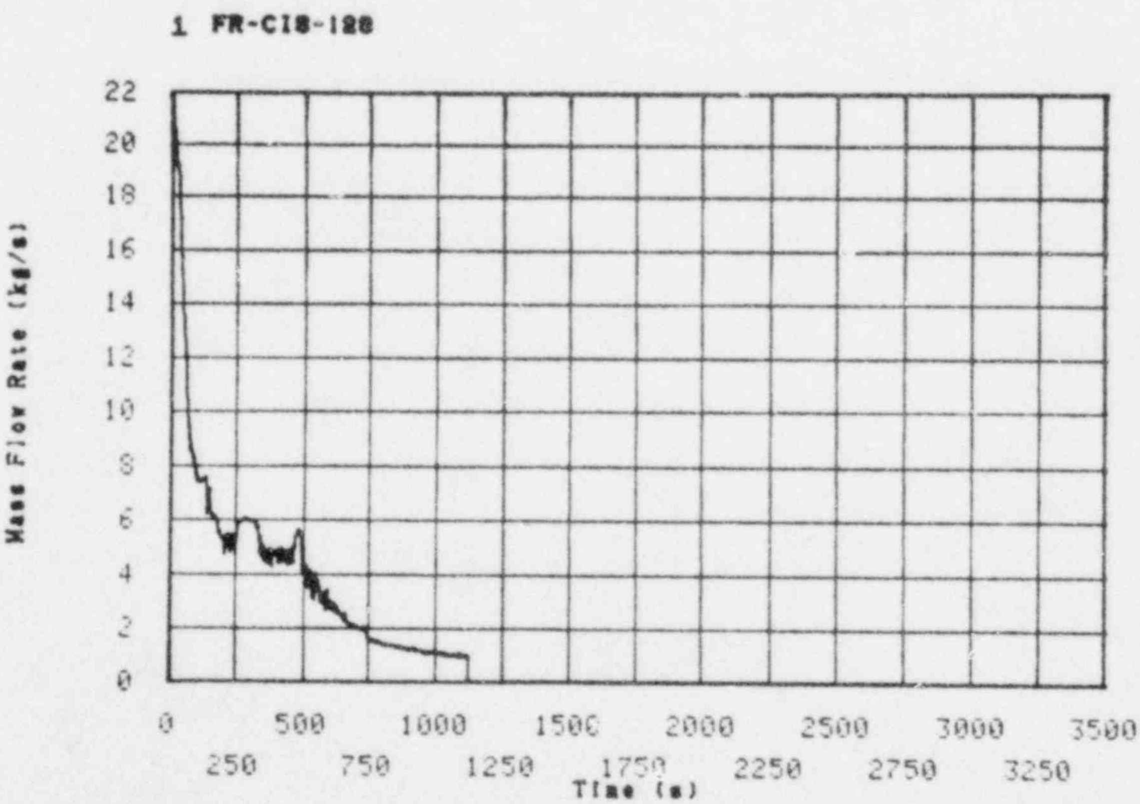
RELAP5 PREDICTION LOFT LS-5/LS-5A

Fig. A-12 Steam Flow



RELAP5 PREDICTION LOFT LS-5/LS-5A

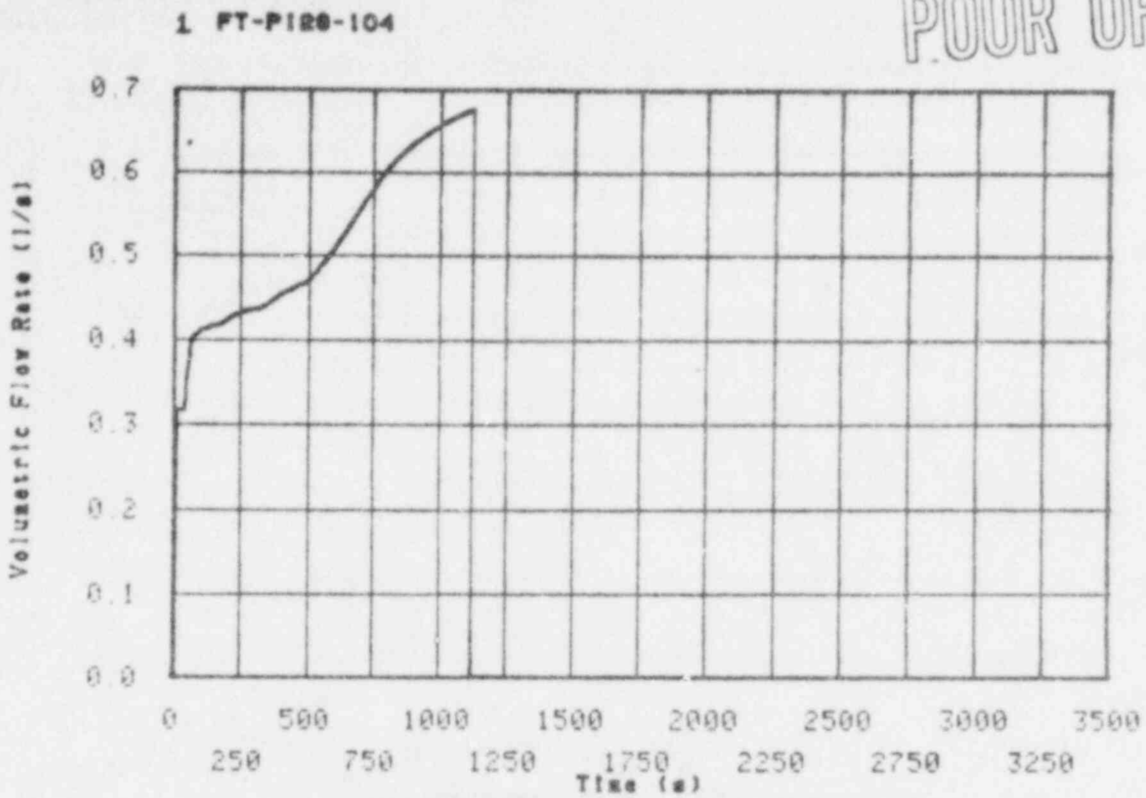
Fig. A-13 Feed Water Flow



RELAP5 PREDICTION LOFT LS-5/LS-5A

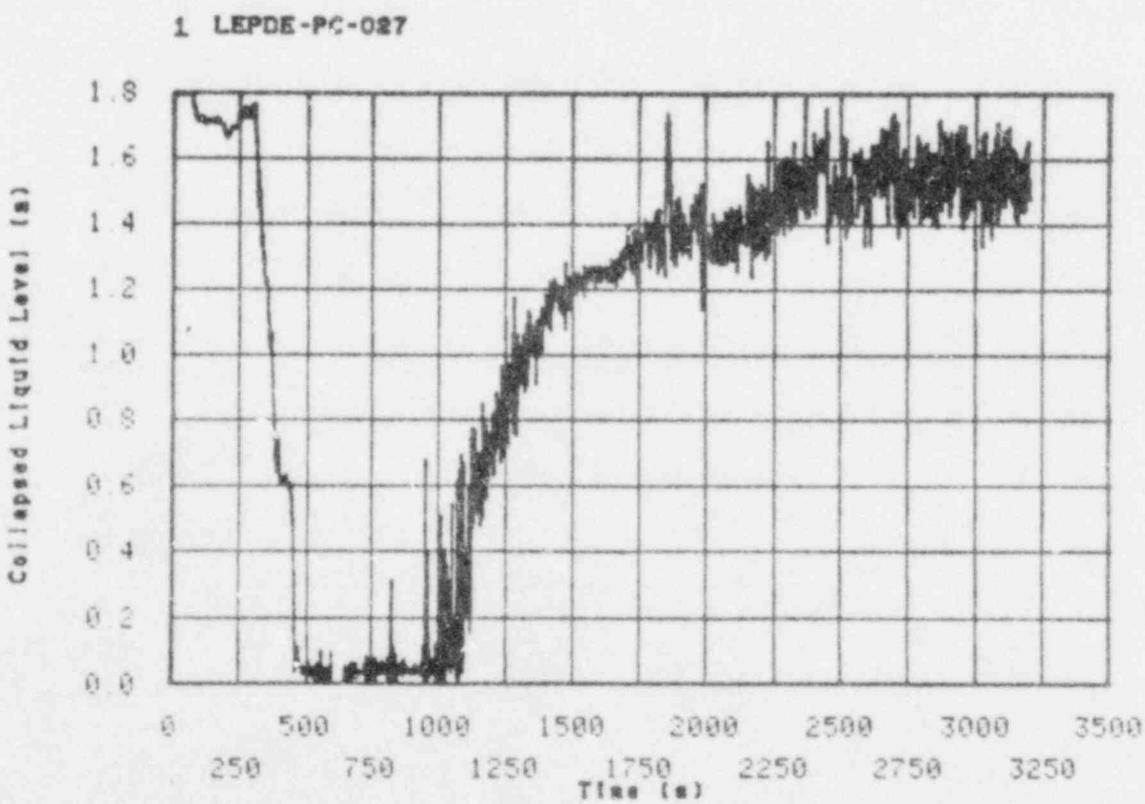
Fig. A-14 Break

POOR ORIGINAL



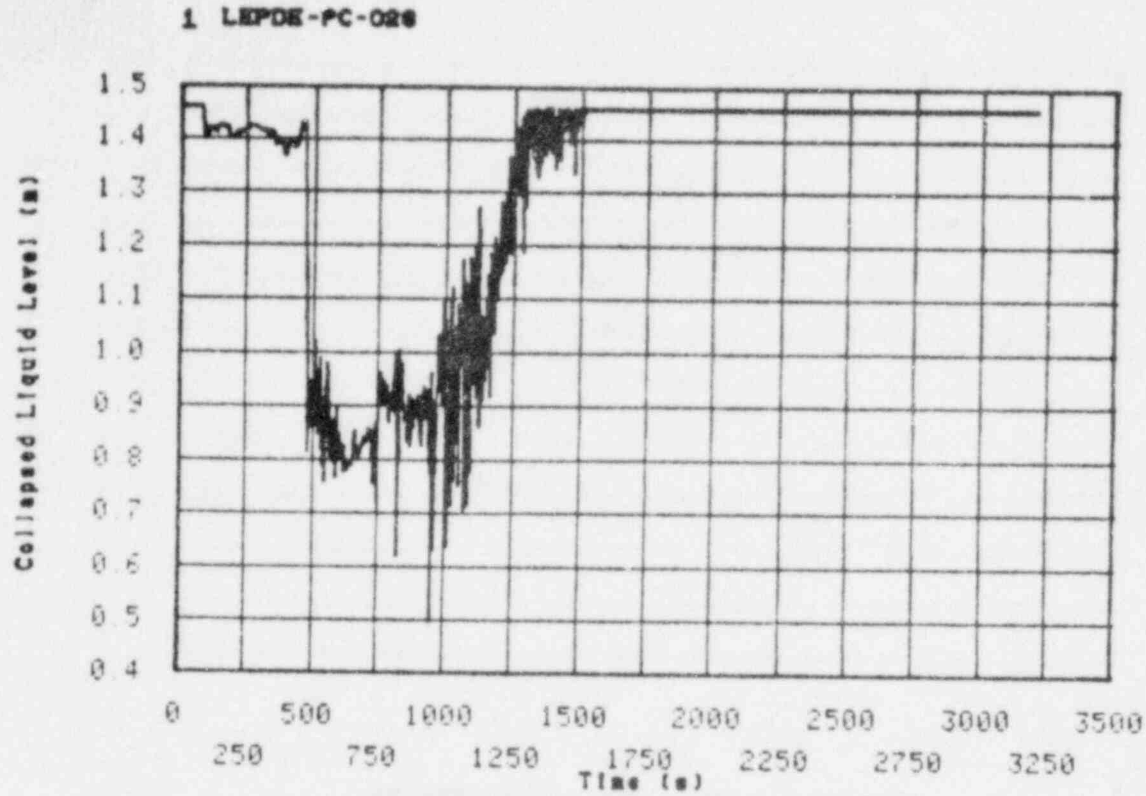
RELAP5 PREDICTION LOFT LS-5/LS-5A

Fig. A-15 HPIS

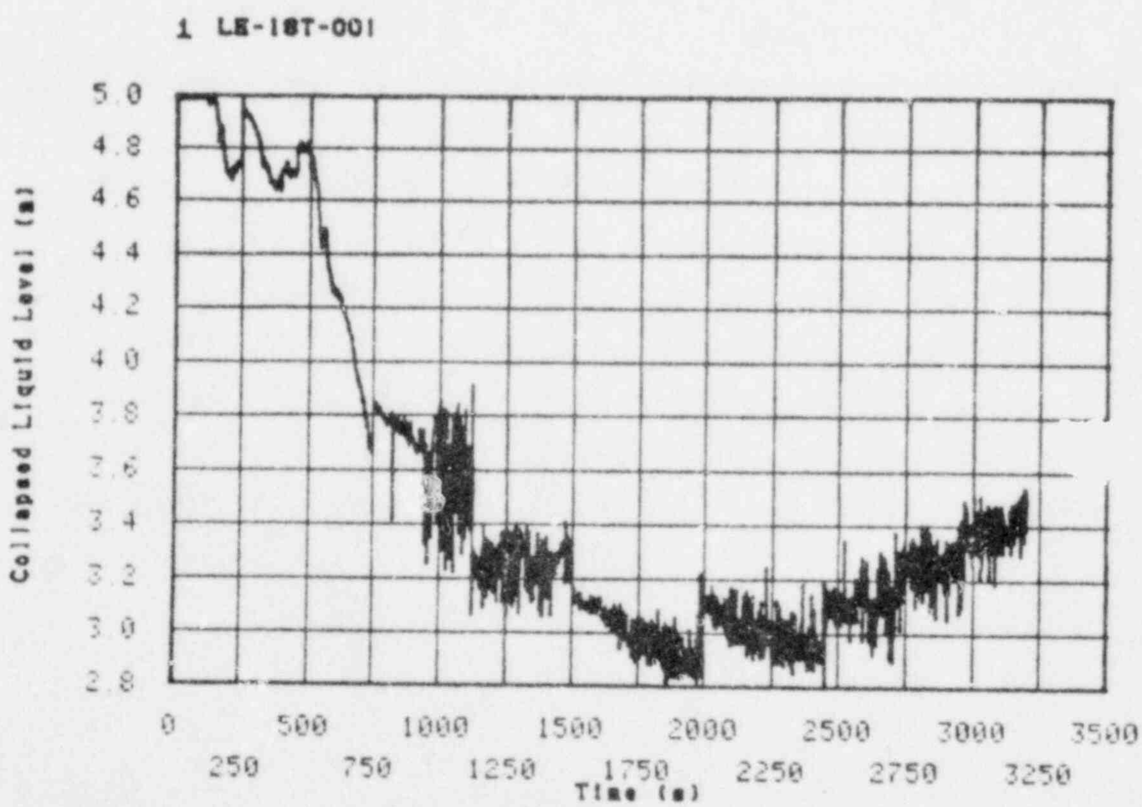


RELAP5 PREDICTION LOFT LS-5/LS-5A

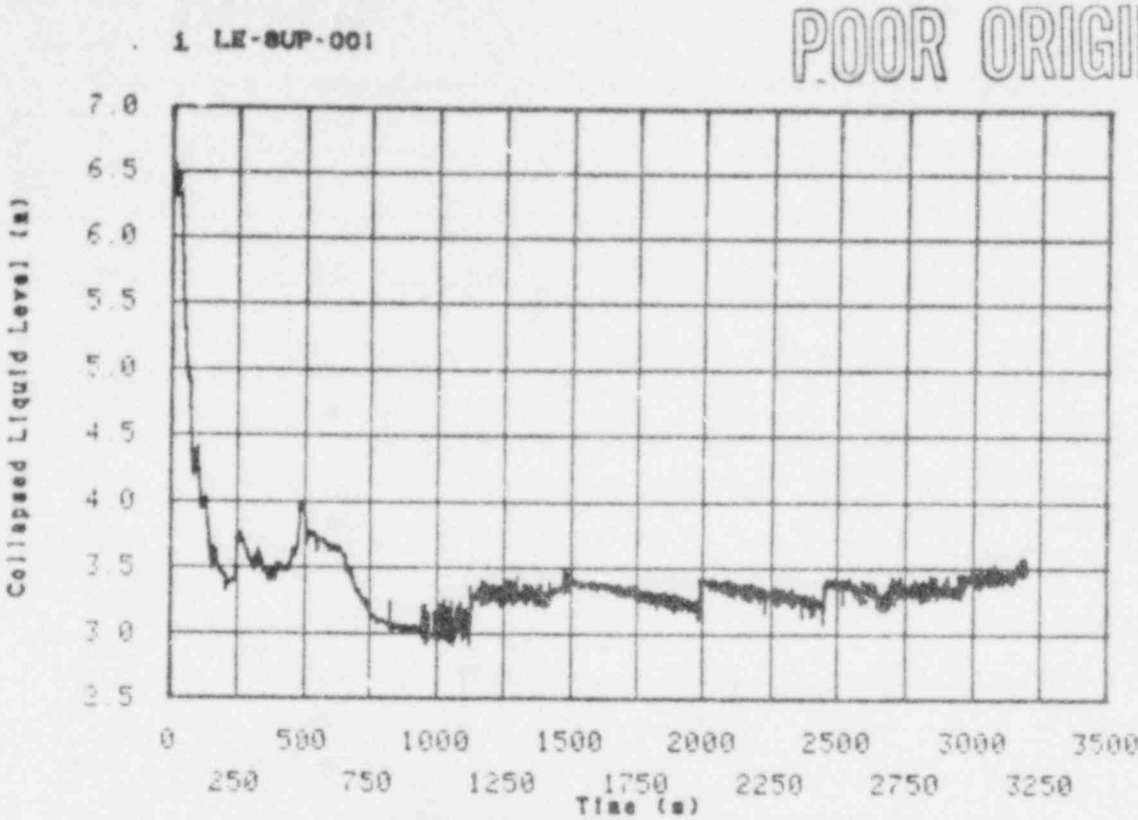
Fig. A-16 Loop Seal Inlet



RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig. A-17 Loop Seal Outlet

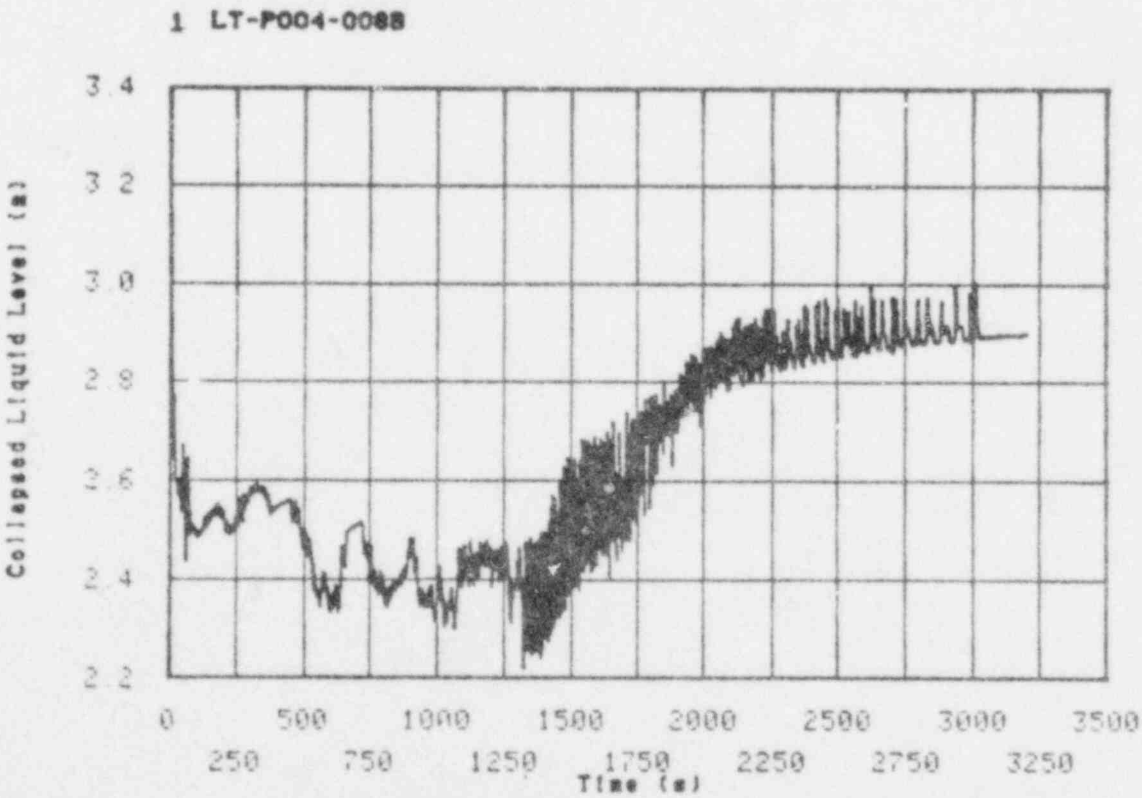


RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig. A-18 R. V. Downcomer



RELAP5 PREDICTION LOFT L8-5/L8-5A

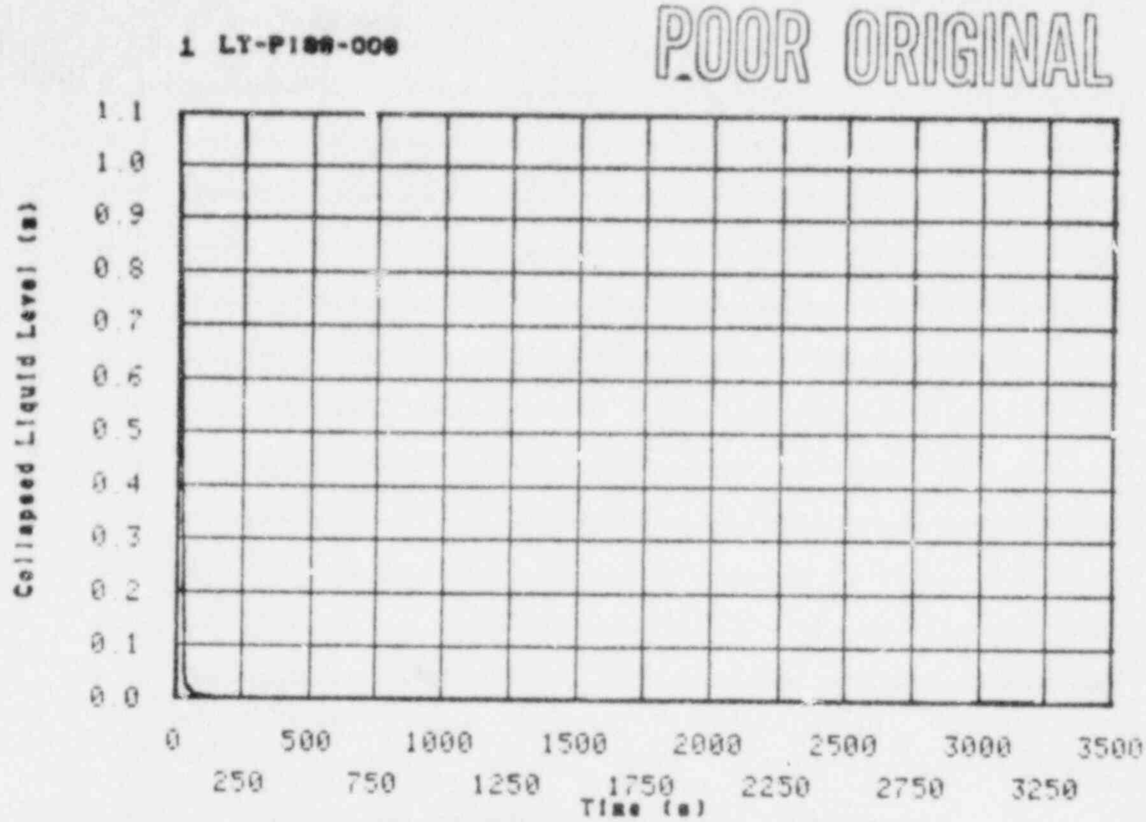
Fig. A-19 Reactor Vessel Core



RELAP5 PREDICTION LOFT L8-5/L8-5A

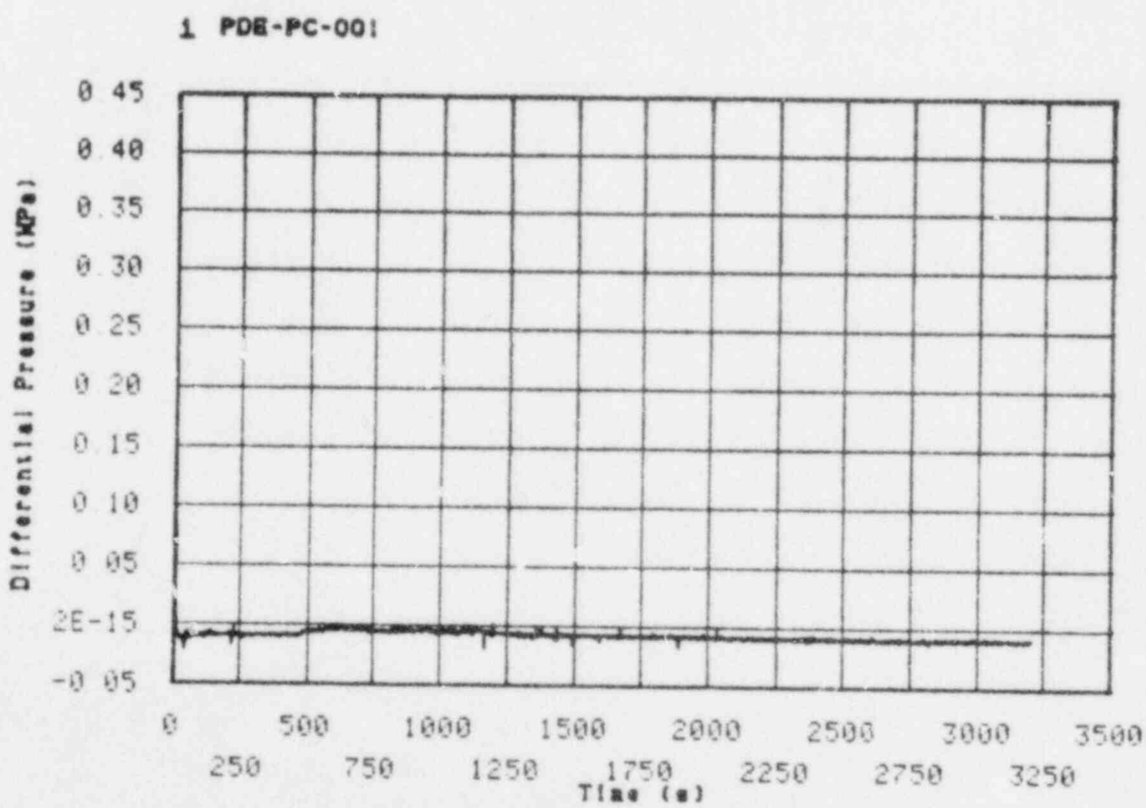
Fig. A-20 S/G Secondary

POOR ORIGINAL



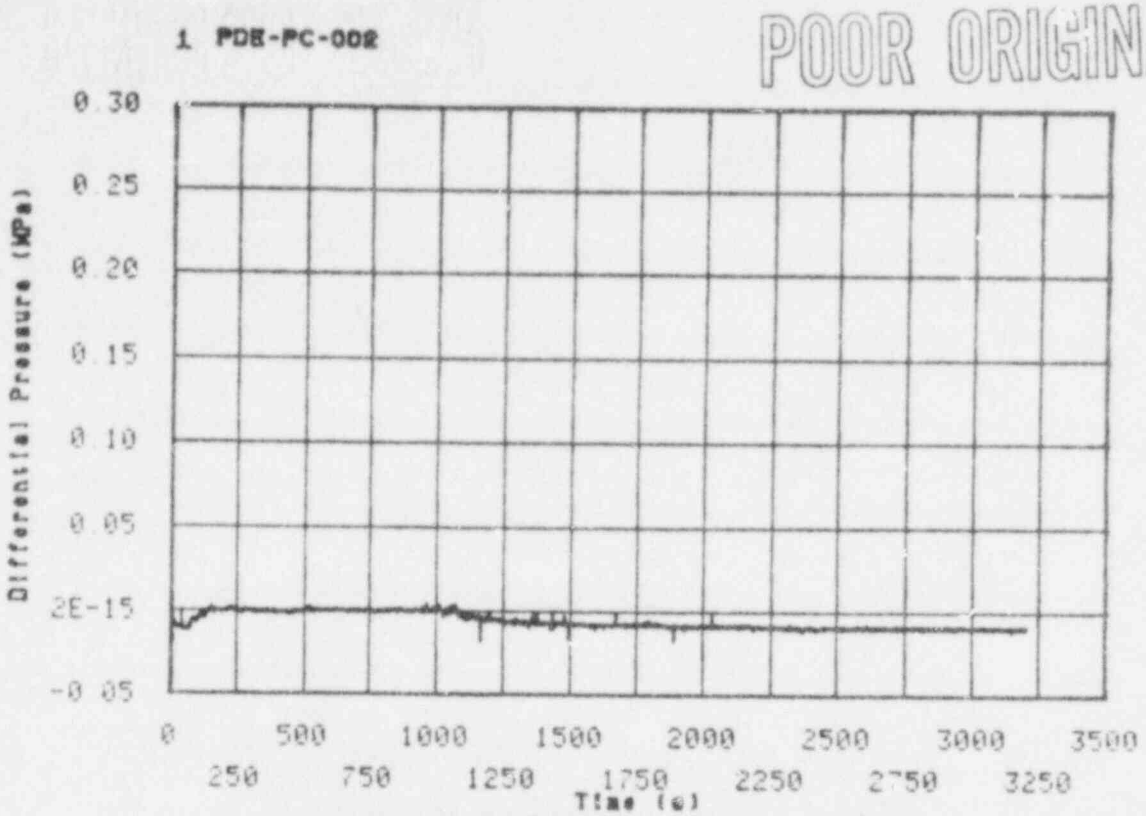
RELAP5 PREDICTION LOFT L9-5/L9-5A

Fig. A-21 Pressurizer

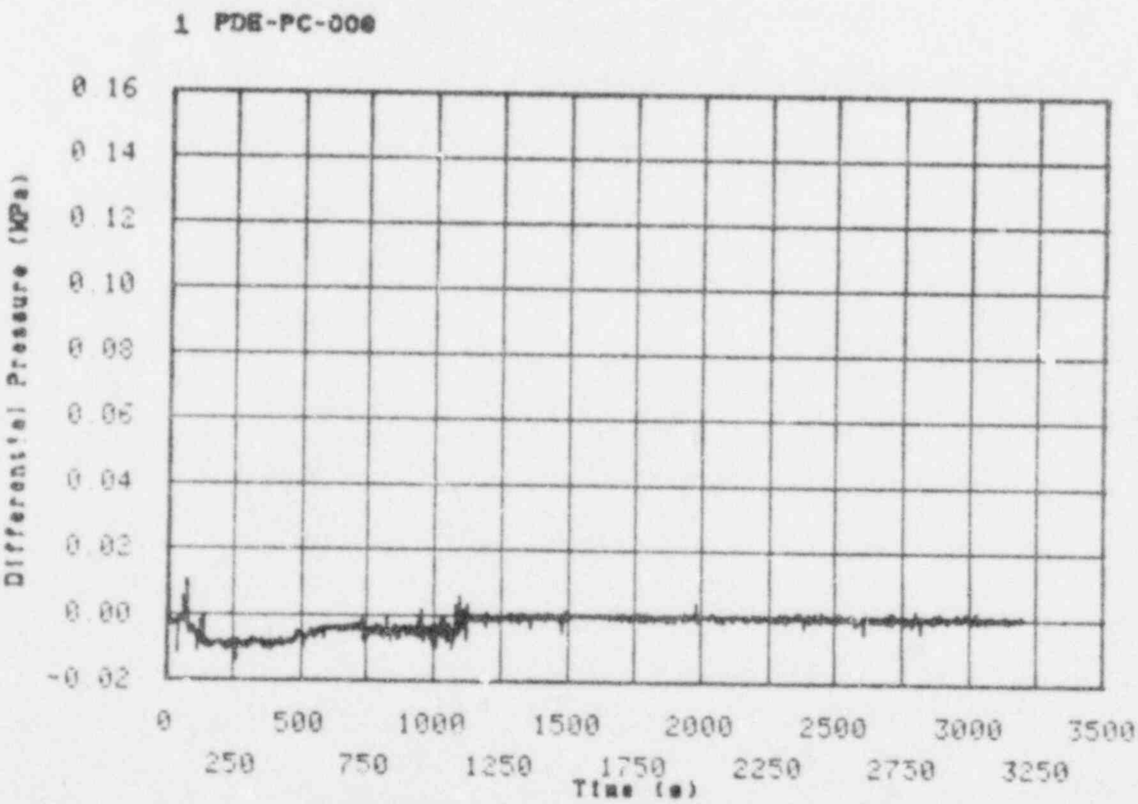


RELAP5 PREDICTION LOFT L9-5/L9-5A

Fig. A-22 Primary coolant pump



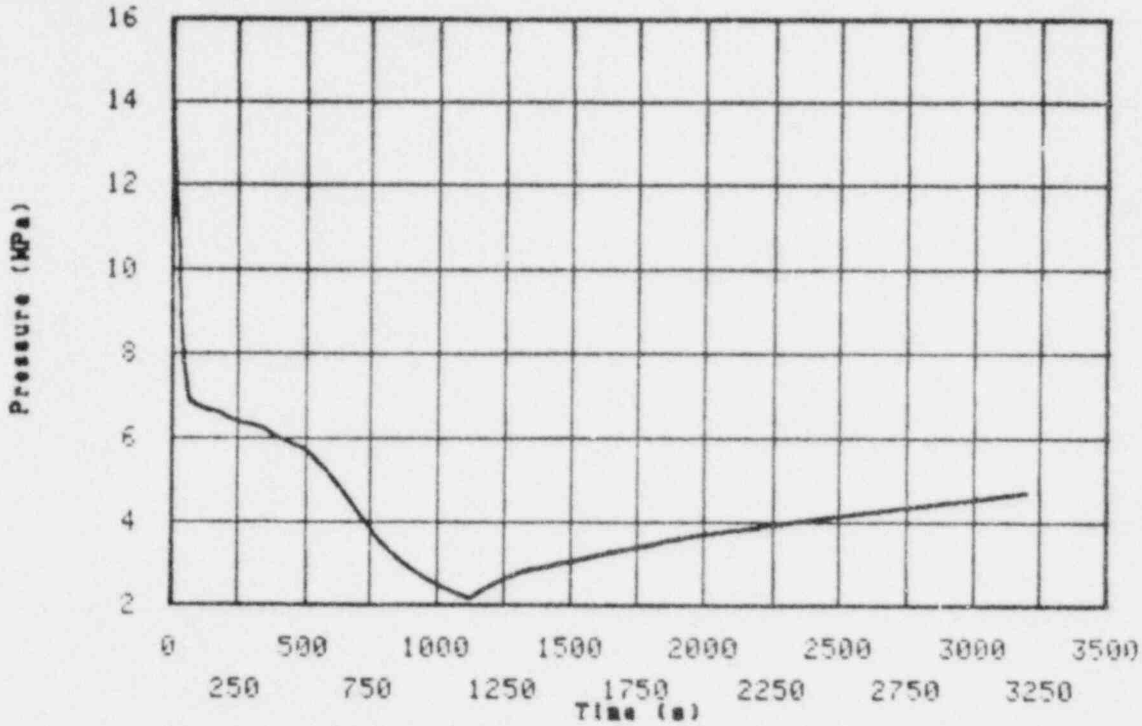
RELAP5 PREDICTION LOFT LS-5/LS-5A
Fig. A-23 Steam Generator



RELAP5 PREDICTION LOFT LS-5/LS-5A
Fig. A-24 Reactor Vessel

POOR ORIGINAL

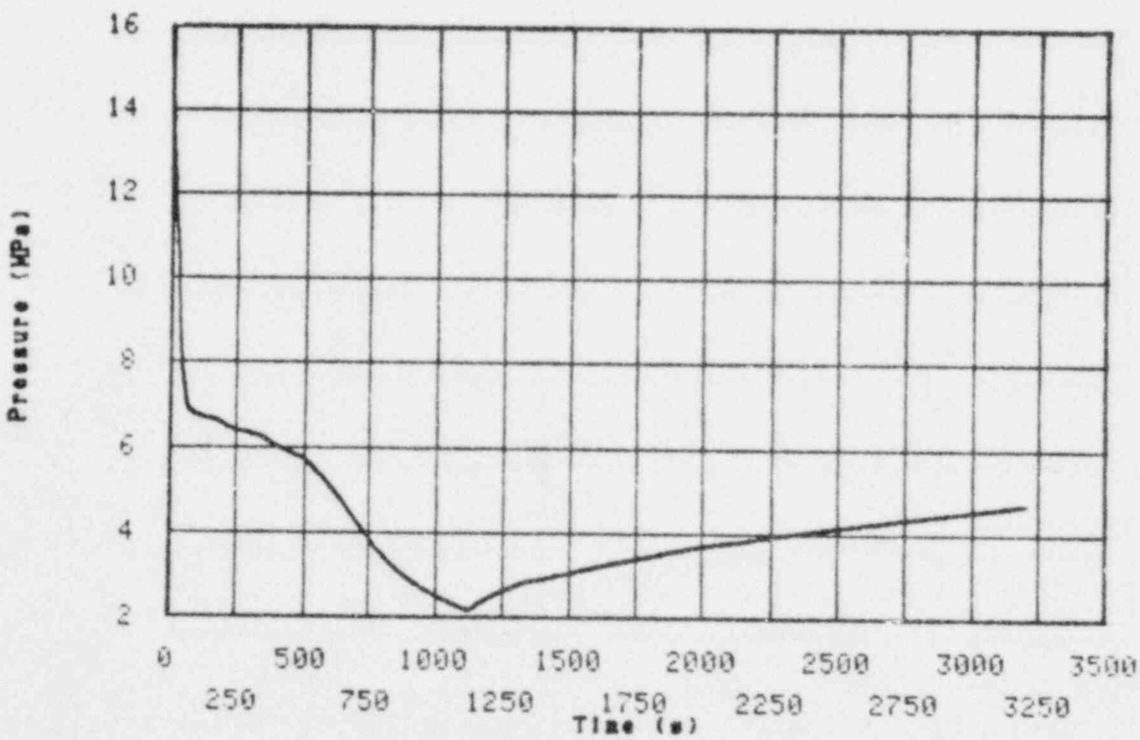
1 PE-BL-001



RELAP5 PREDICTION LOFT L9-5/L9-5A

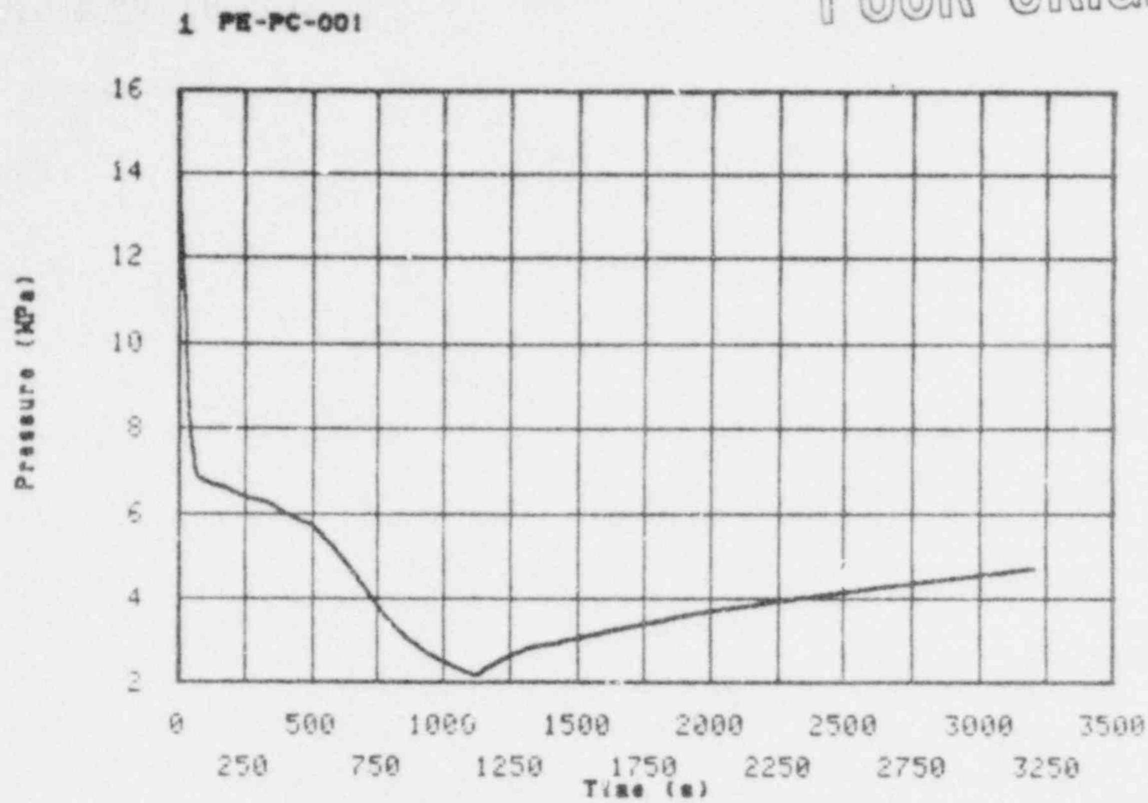
Fig. A-25 Pressure - BLCL

1 PE-BL-002

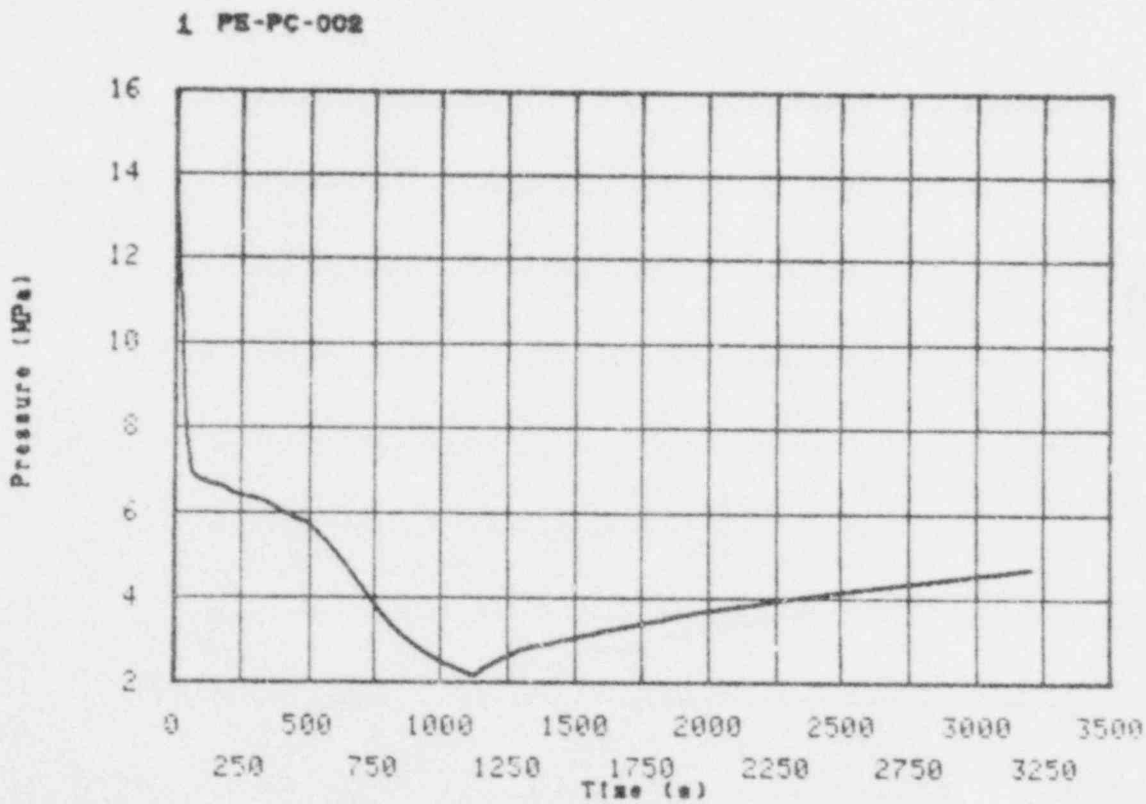


RELAP5 PREDICTION LOFT L9-5/L9-5A

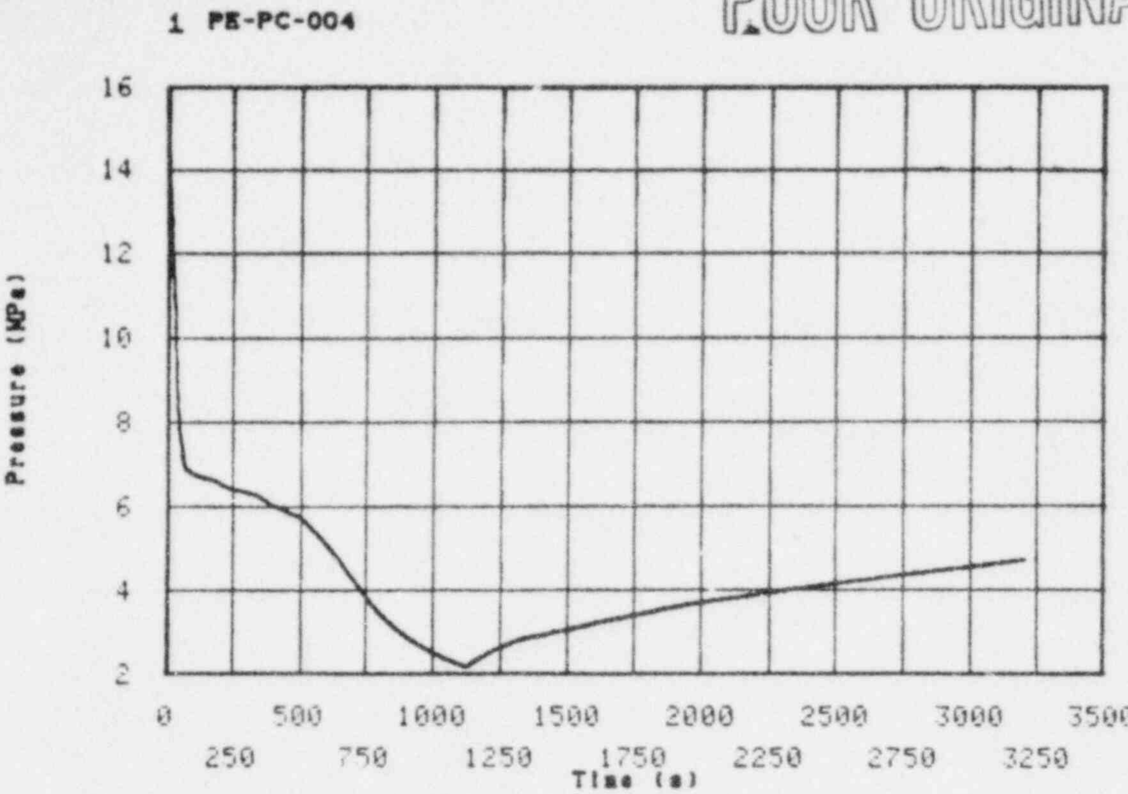
Fig. A-26 Pressure - BLHL



RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig. A-27 Pressure - ILCL

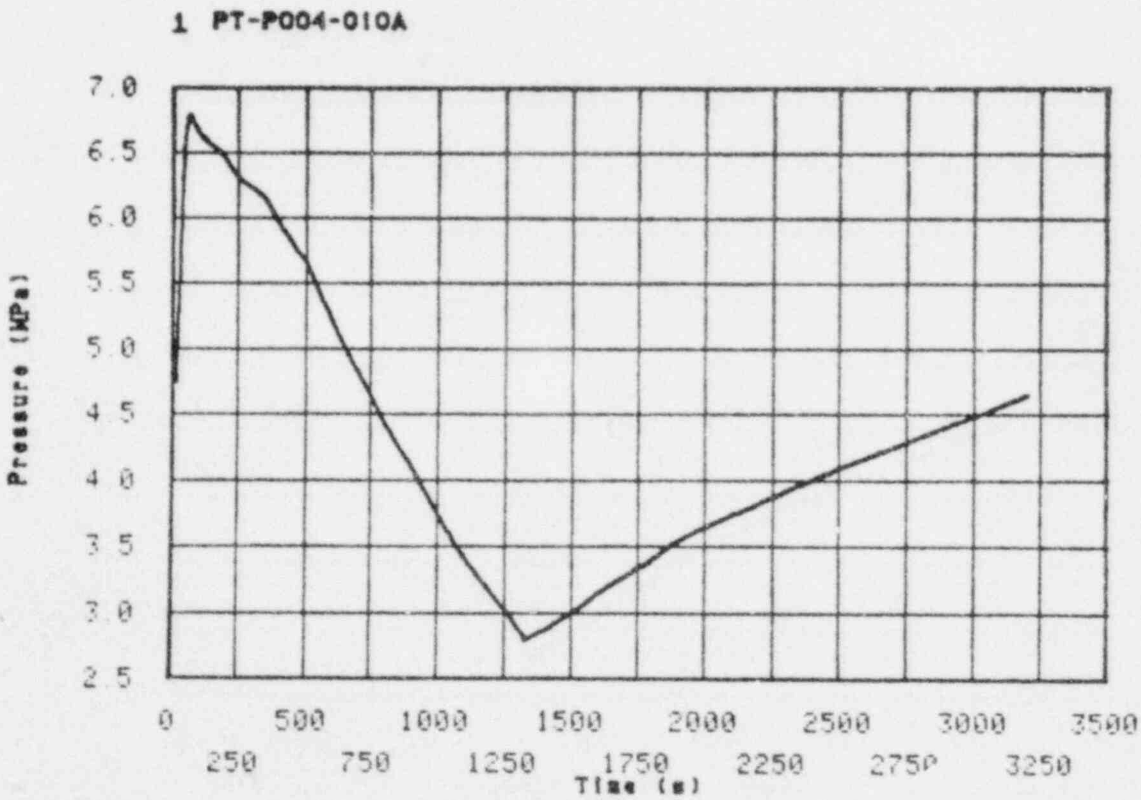


RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig. A-28 Pressure - ILHL



RELAP5 PREDICTION LOFT L8-5/L8-5A

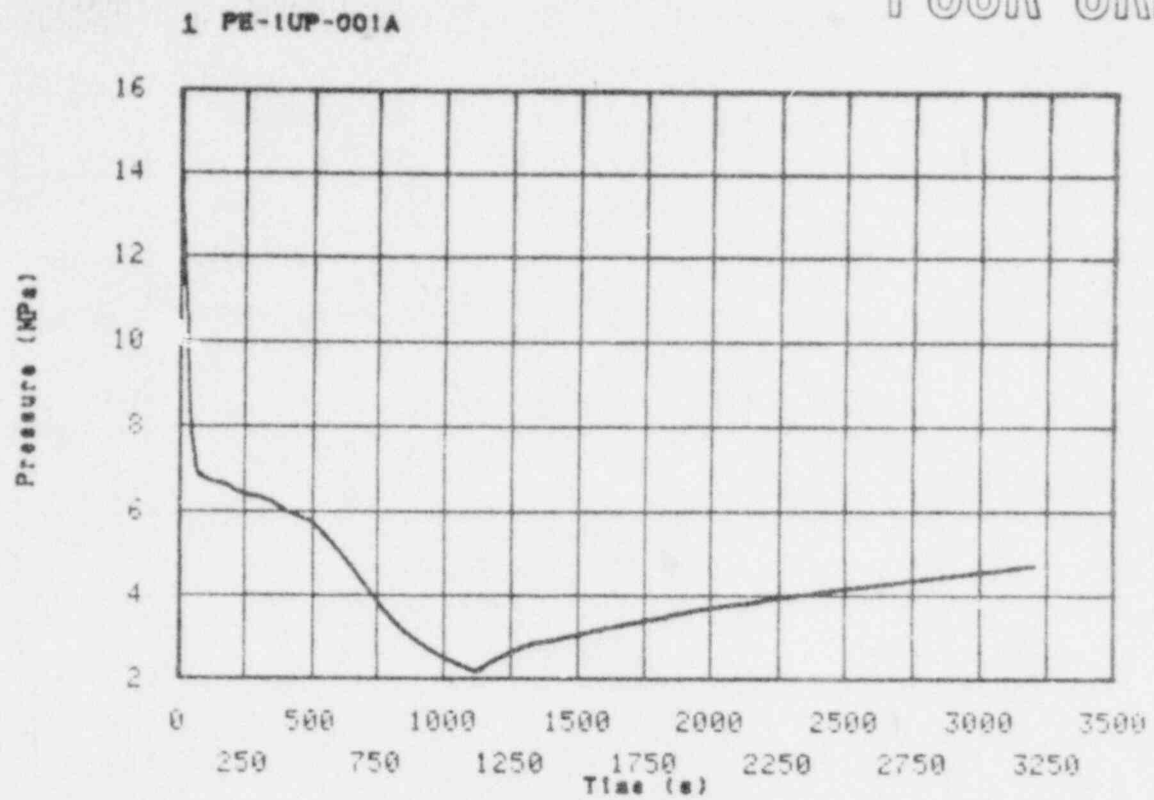
Fig. A-29 Pressure - Pressurizer



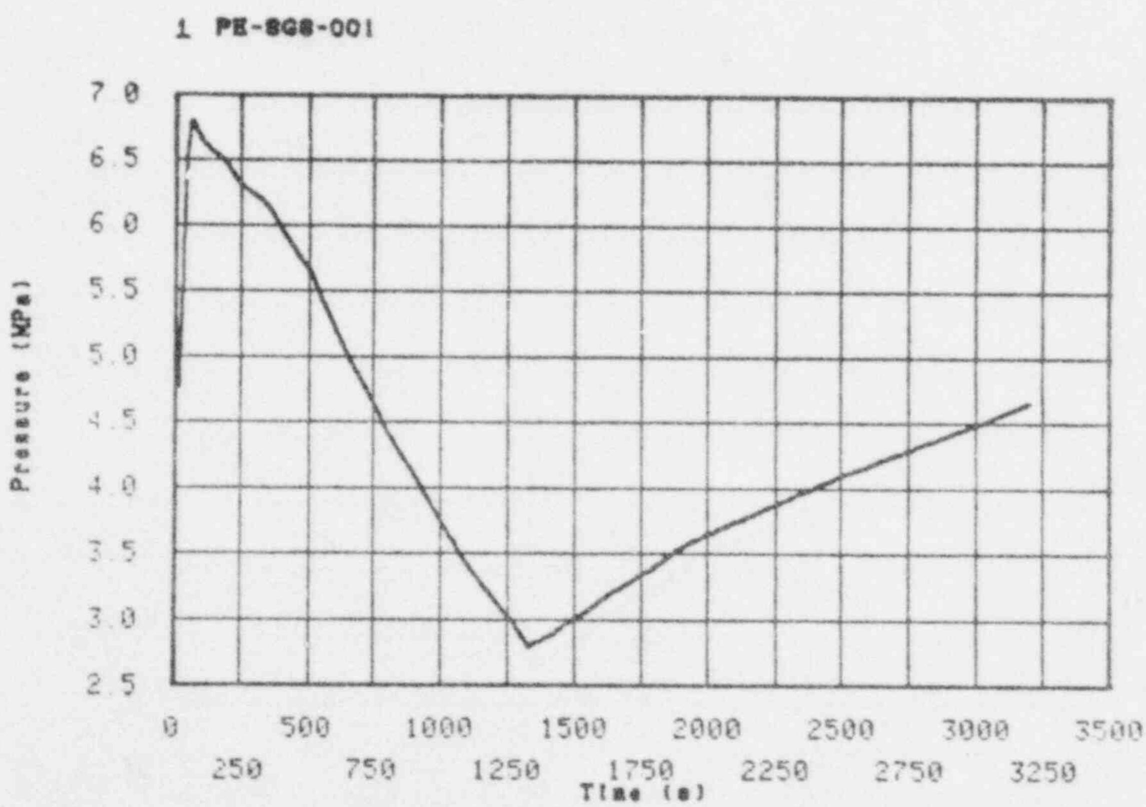
RELAP5 PREDICTION LOFT L8-5/L8-5A

Fig A-30 Pressure - Steam Line

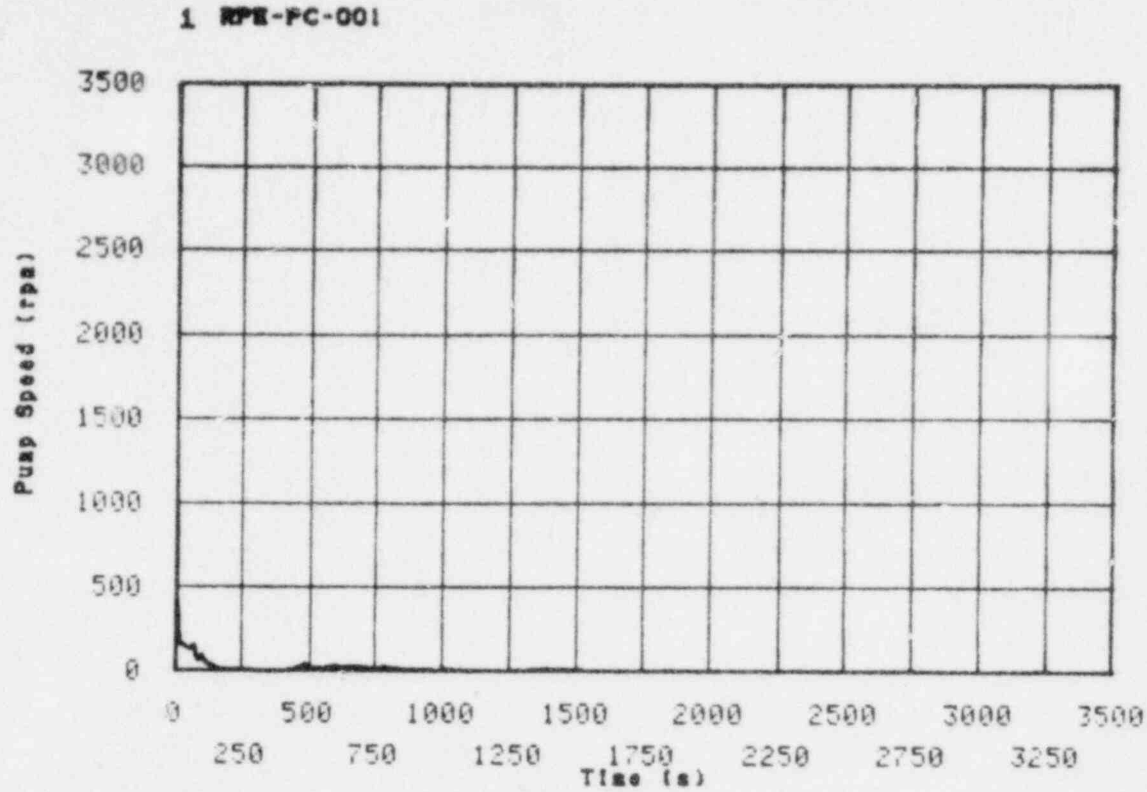
POOR ORIGINAL



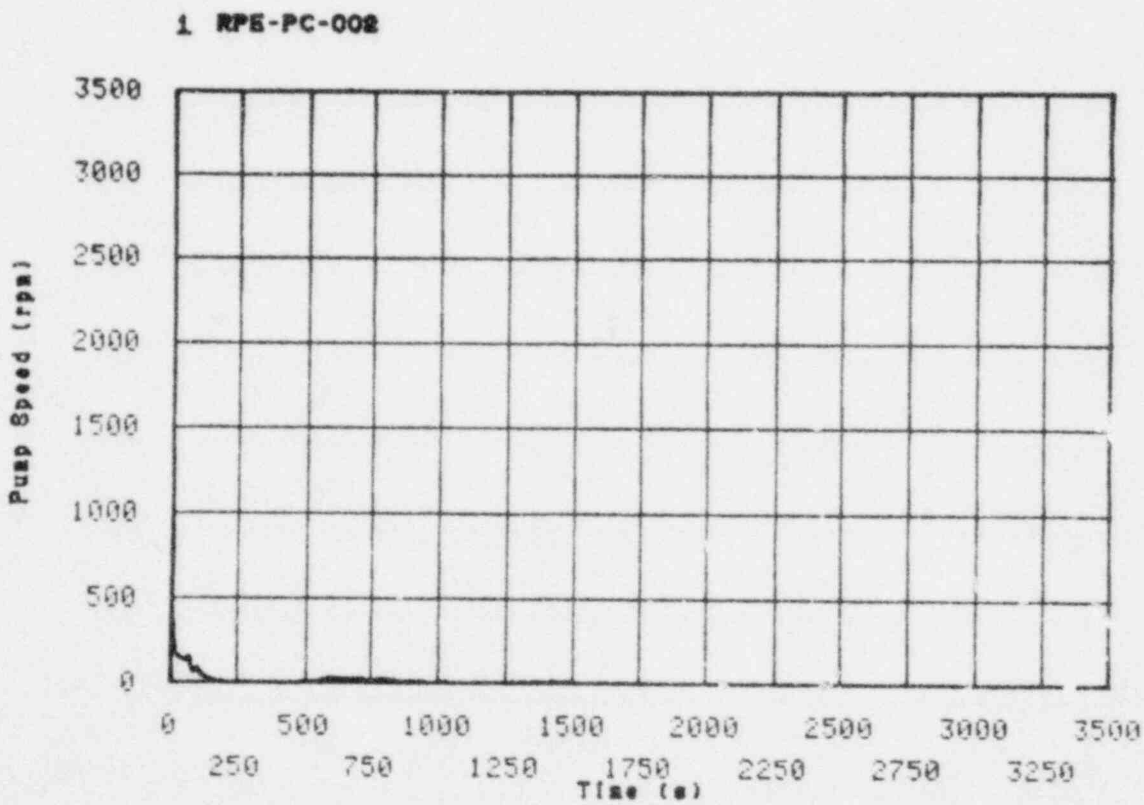
RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig. A-81 Pressure-Upper Plenum



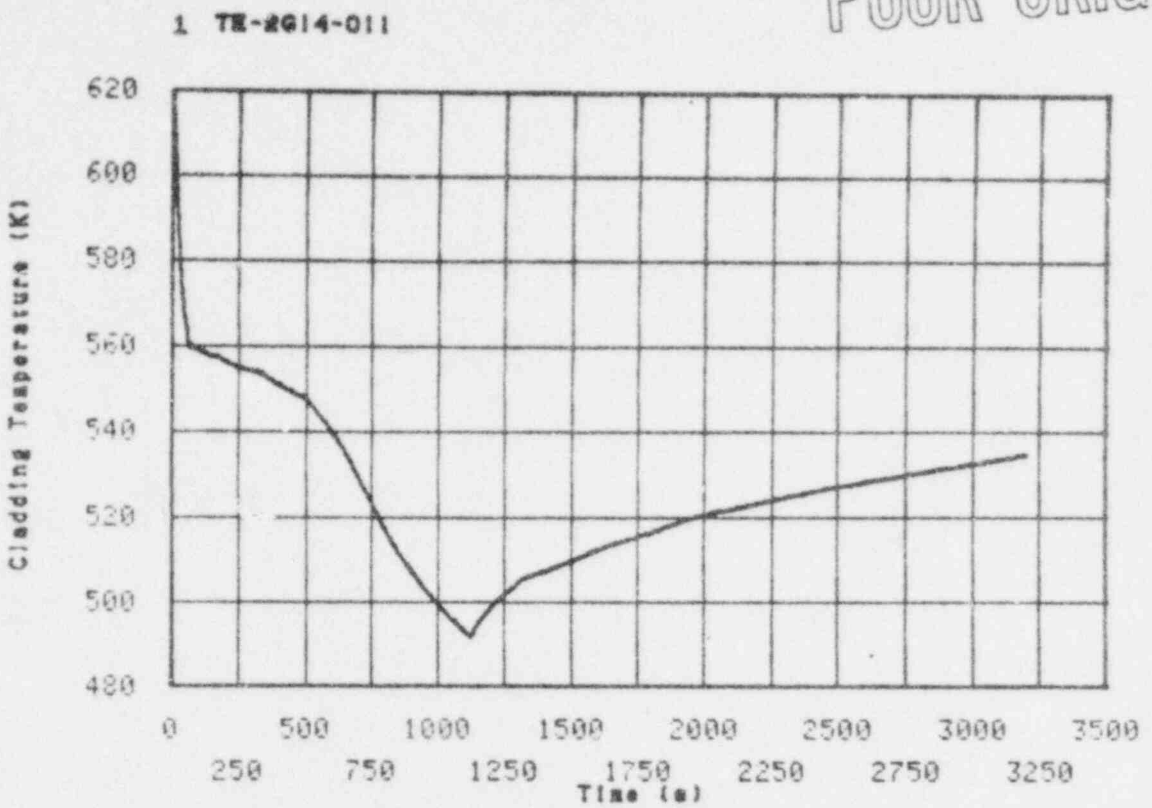
RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-82 Pressure-S/G Steam dome



RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-83 Primary Coolant Pump 1

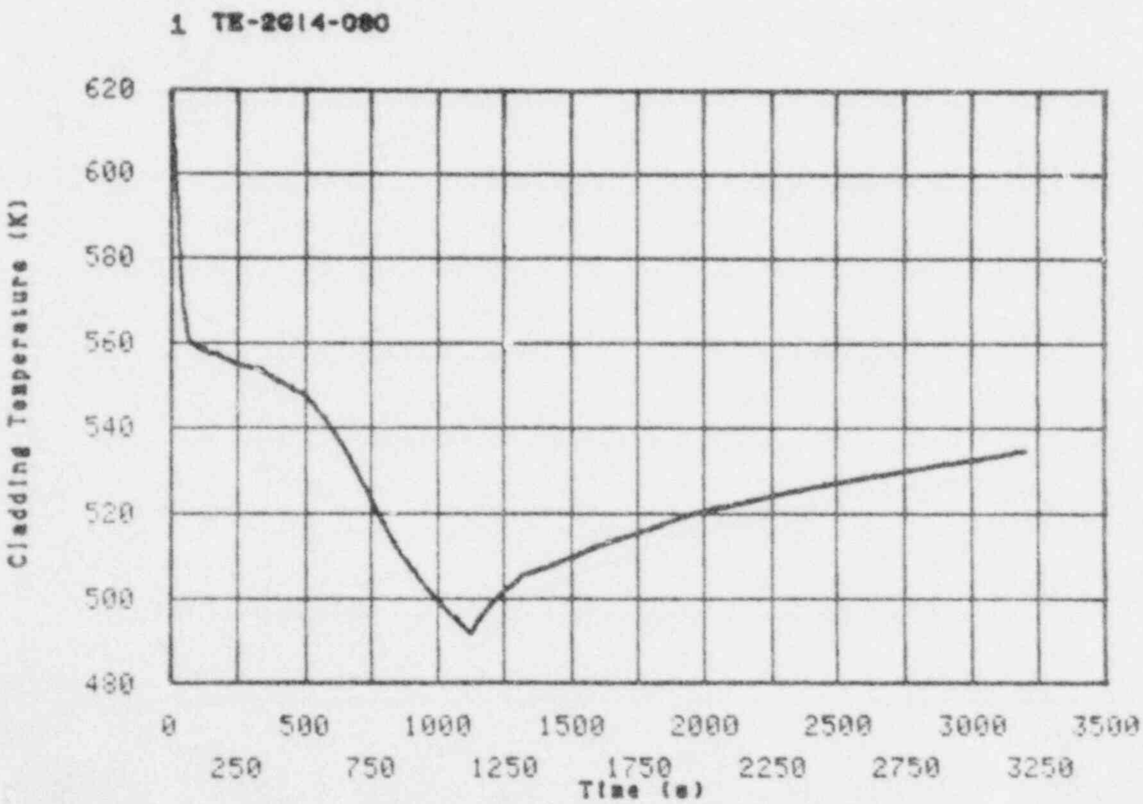


RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-84 Primary Coolant Pump 2



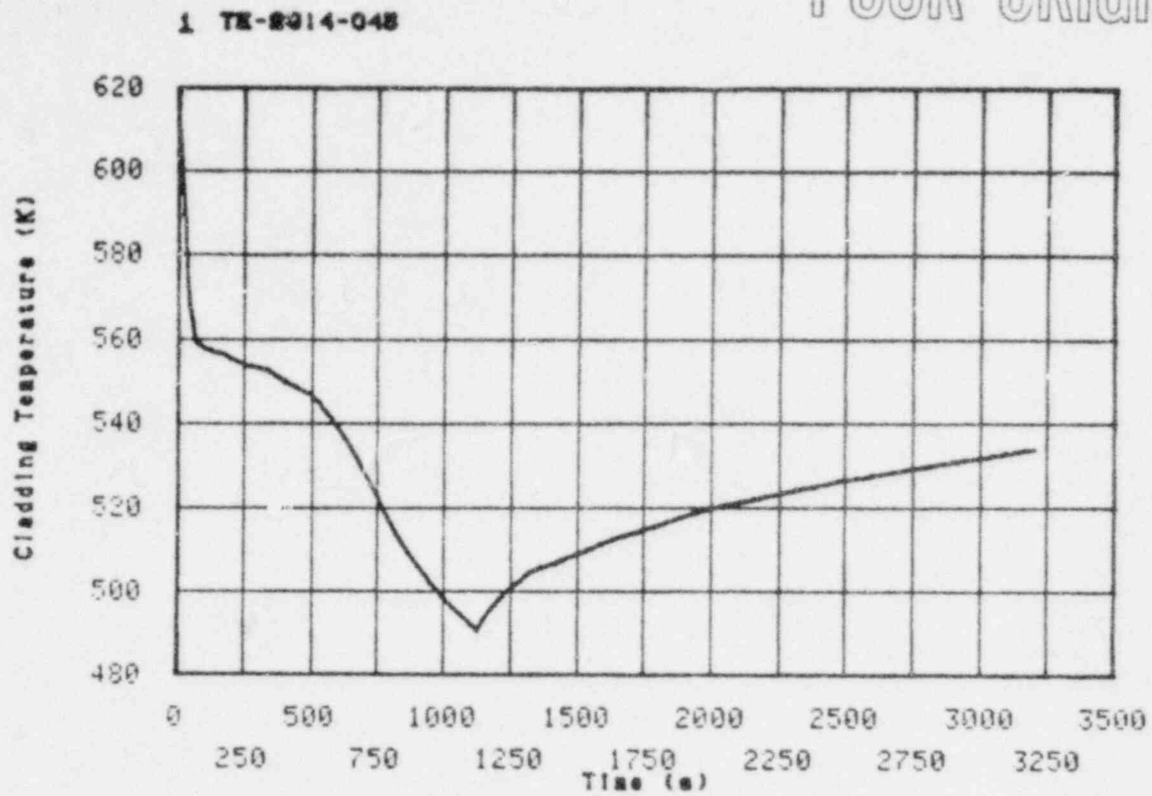
RELAP5 PREDICTION LOFT L8-5/L8-5A

Fig A-85 Lower Third of Core

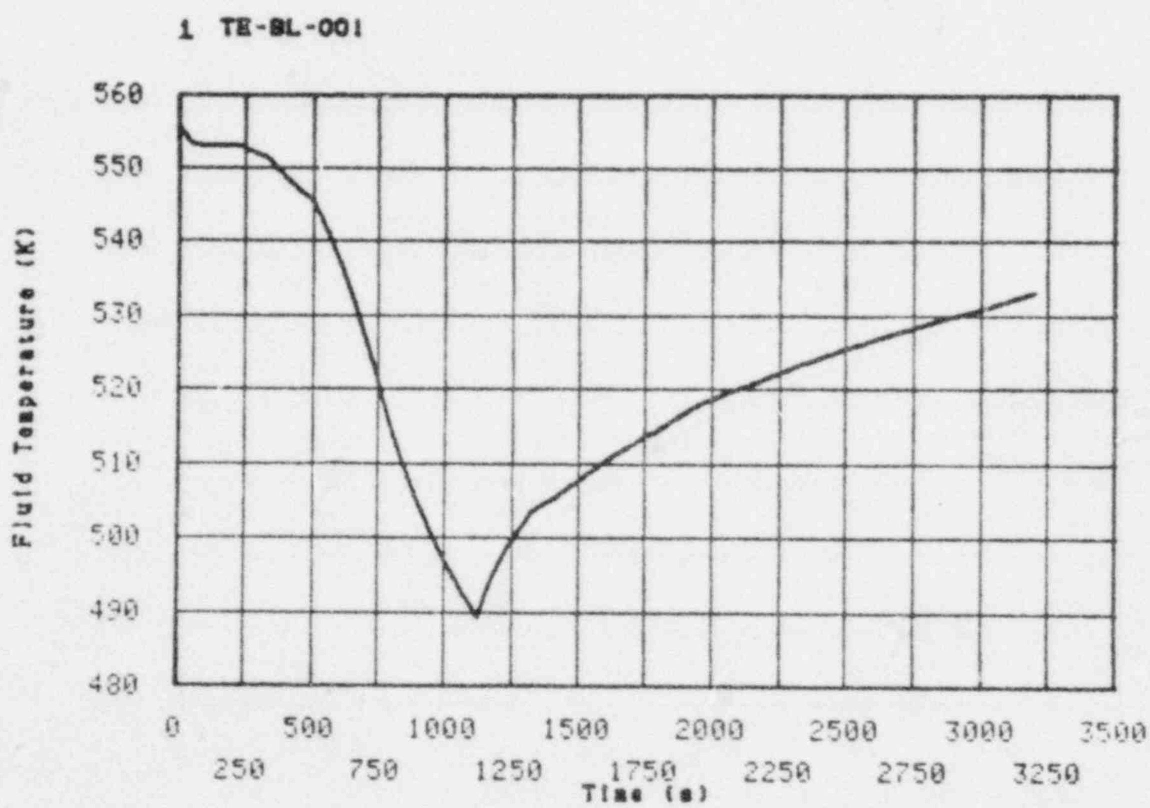


RELAP5 PREDICTION LOFT L8-5/L8-5A

Fig A-86 Middle Third of Core

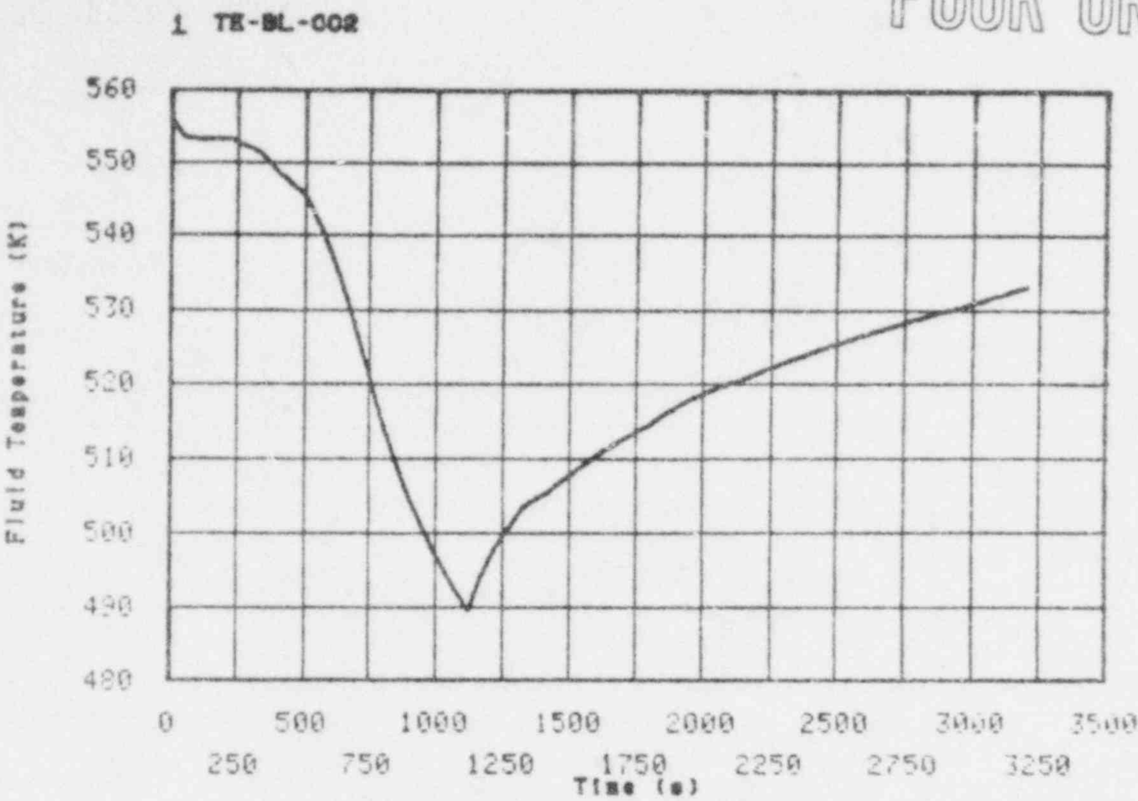


RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-87 Upper Third of Core

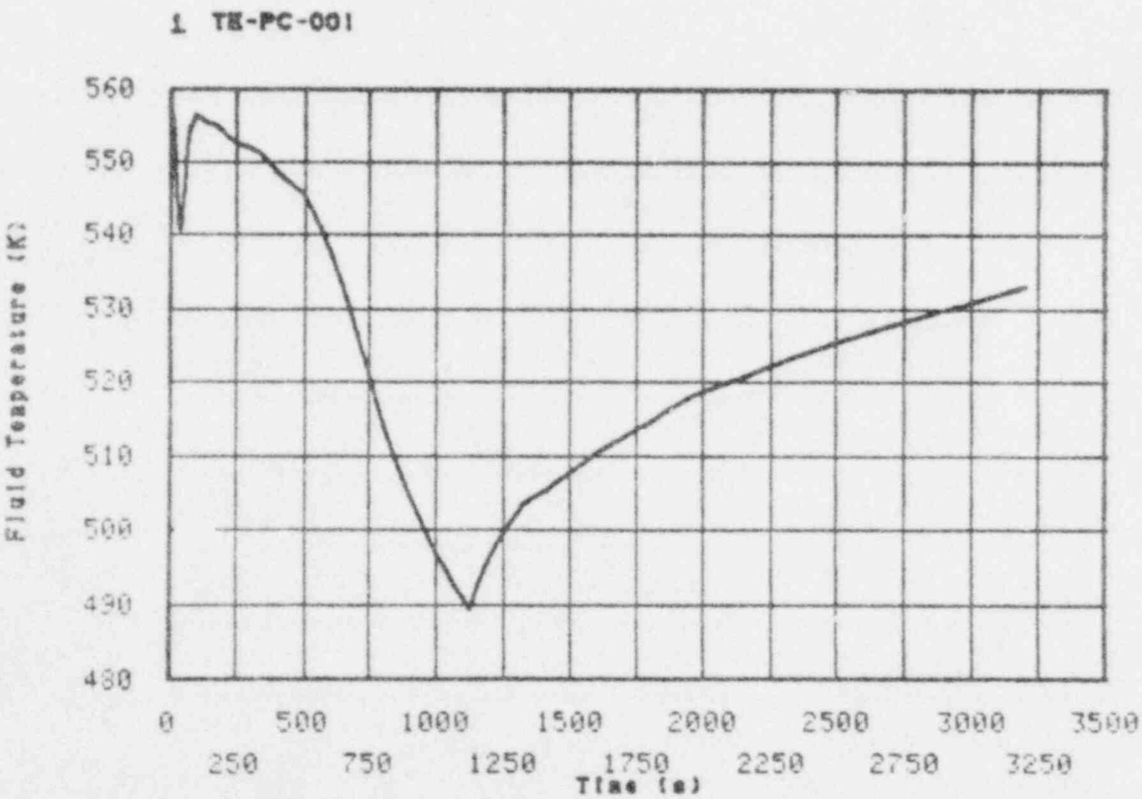


RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-88 Broken Loop Cold Leg

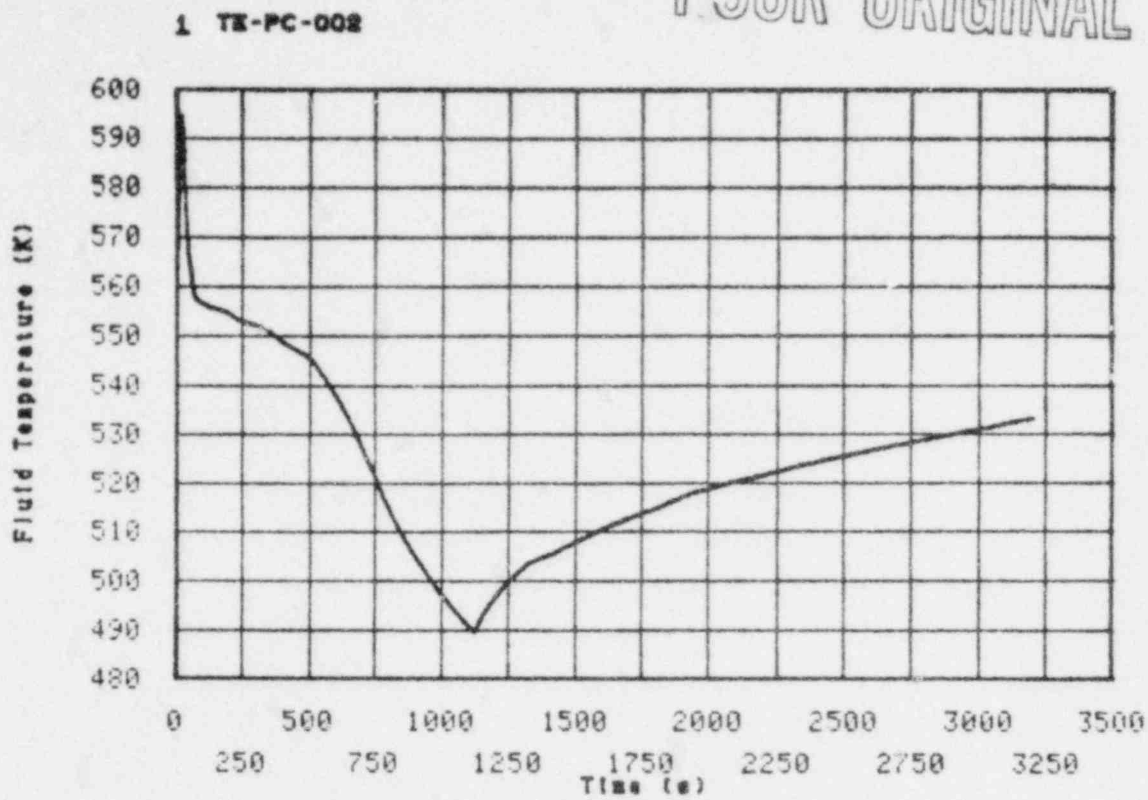
POOR ORIGINAL



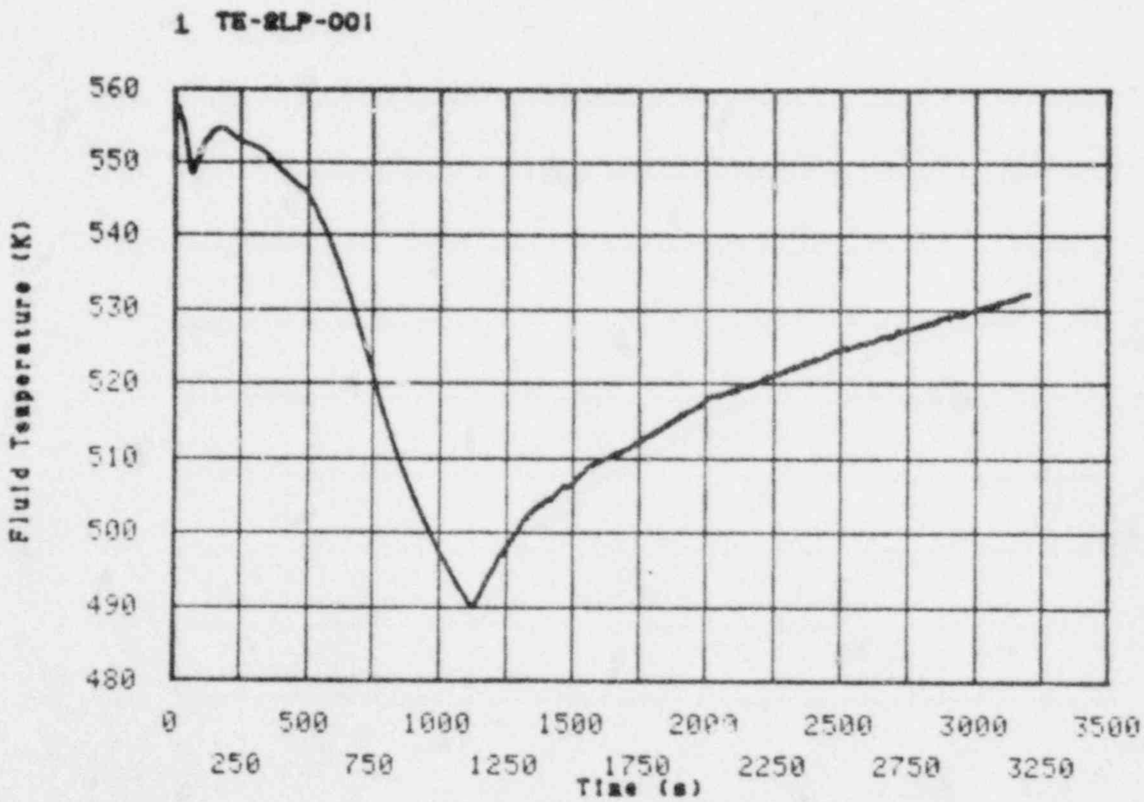
RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-39 Broken Loop Hot Leg



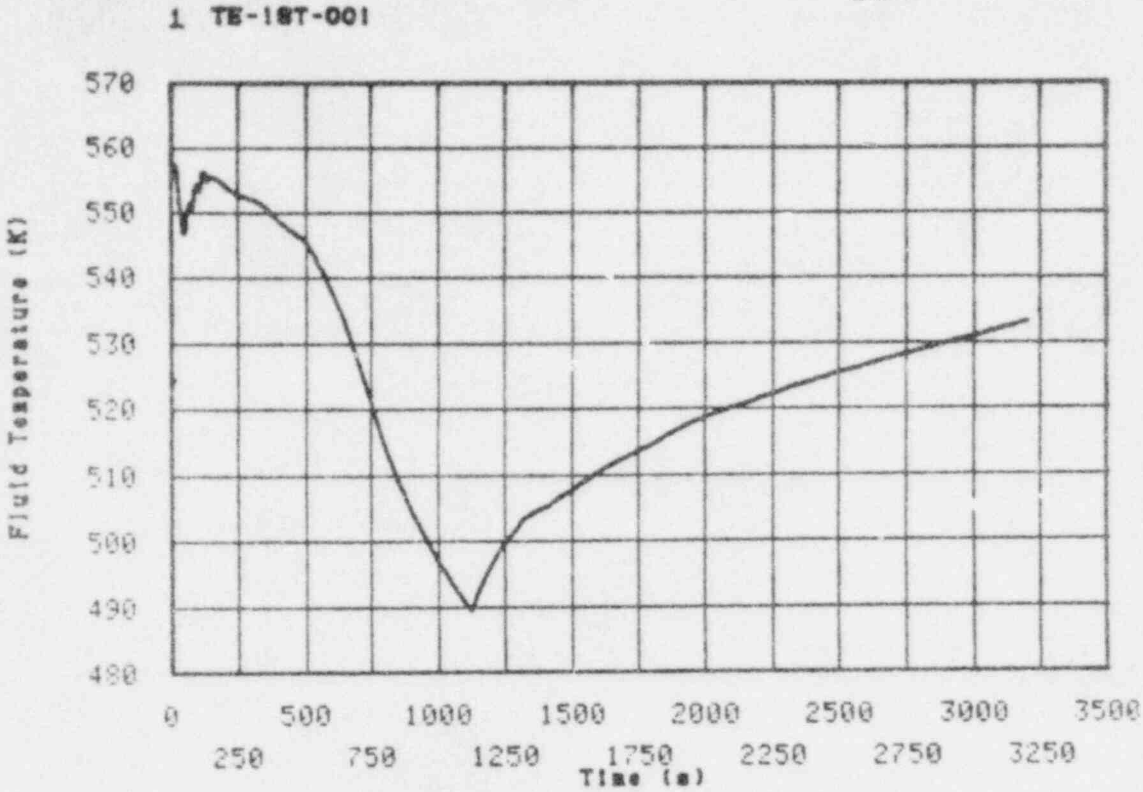
RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-40 Intact Loop Cold Leg



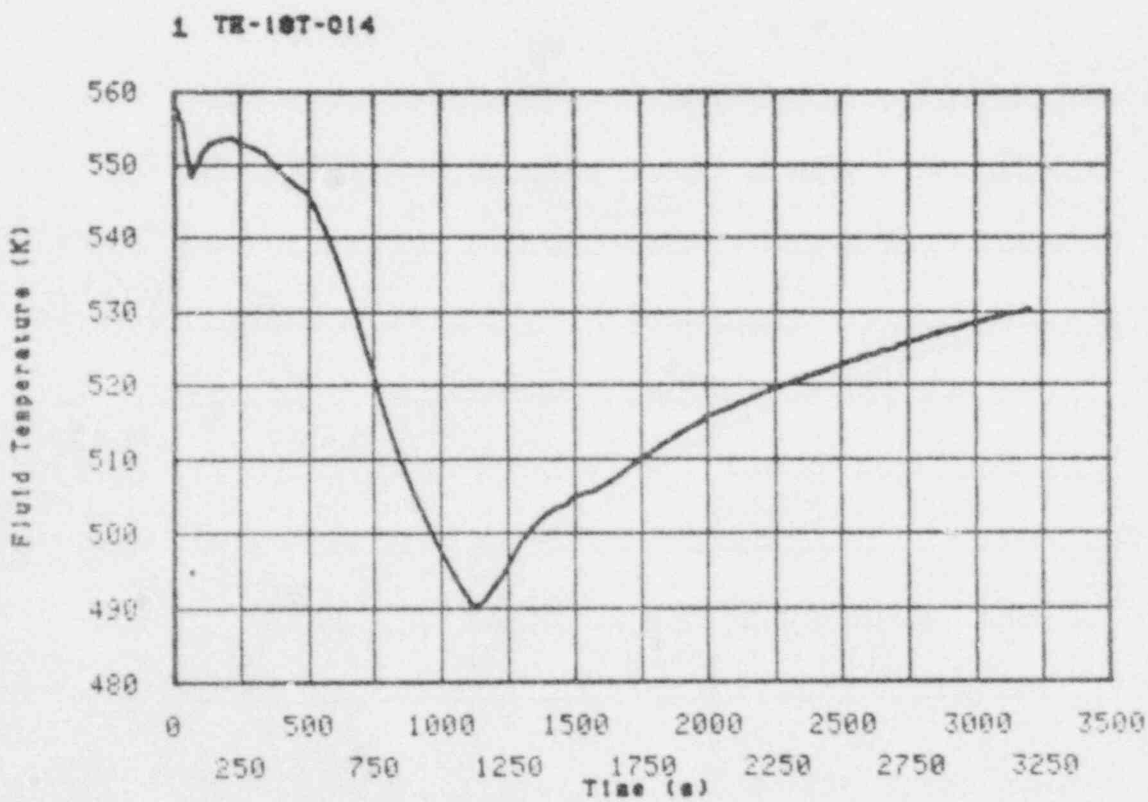
RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-41 Intact Loop Hot Leg



RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-42 Lower Plenum

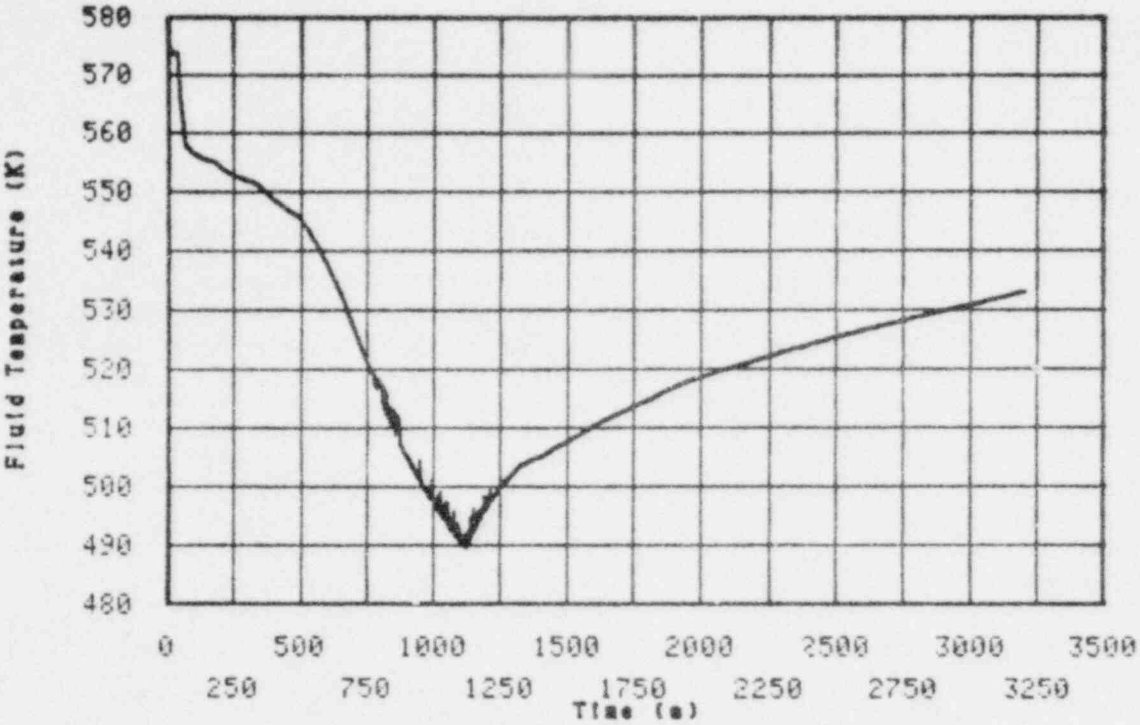


RELAP5 PREDICTION LOFT L3-5/L3-5A
Fig A-43 Inlet Annulus



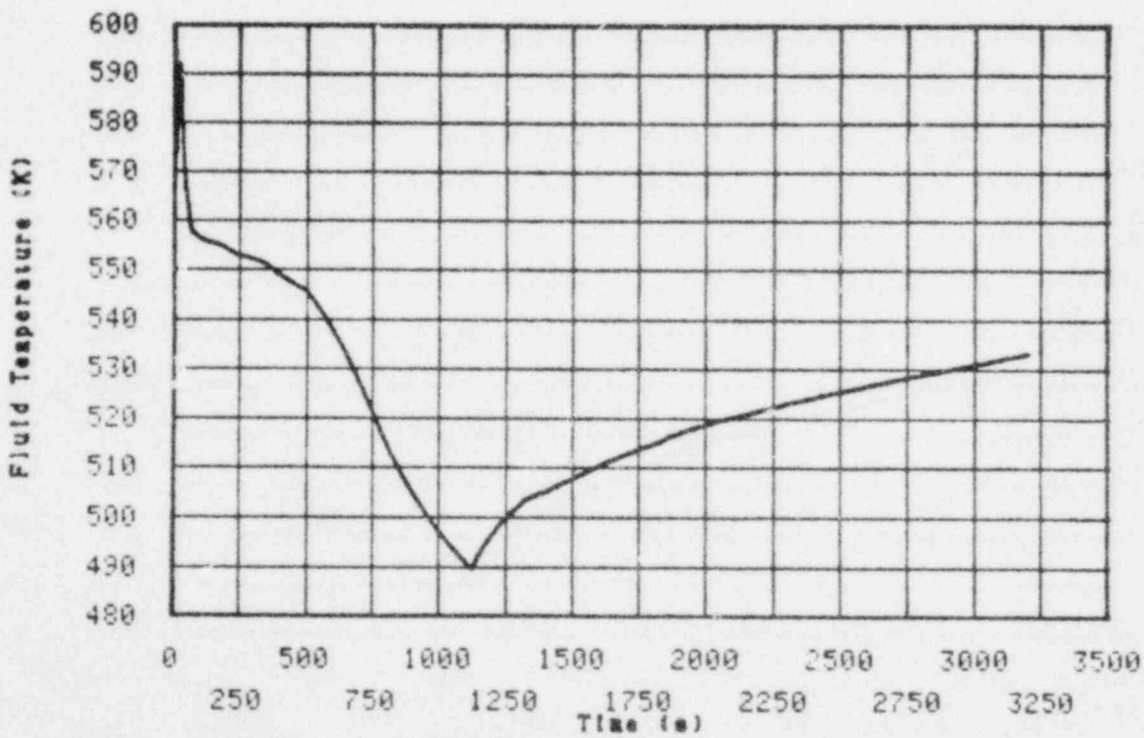
RELAP5 PREDICTION LOFT L3-5/L3-5A
Fig A-44 R. V. Downcomer

1 TE-5UP-008

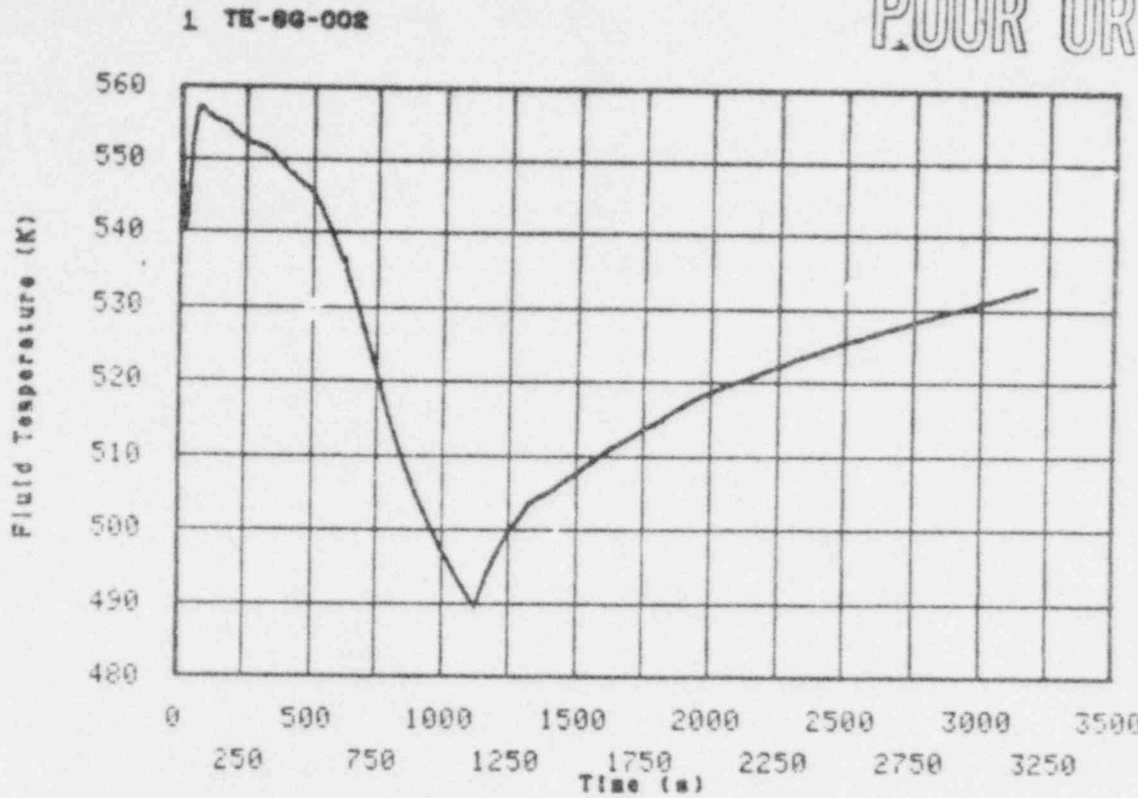


RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-45 Upper Plenum

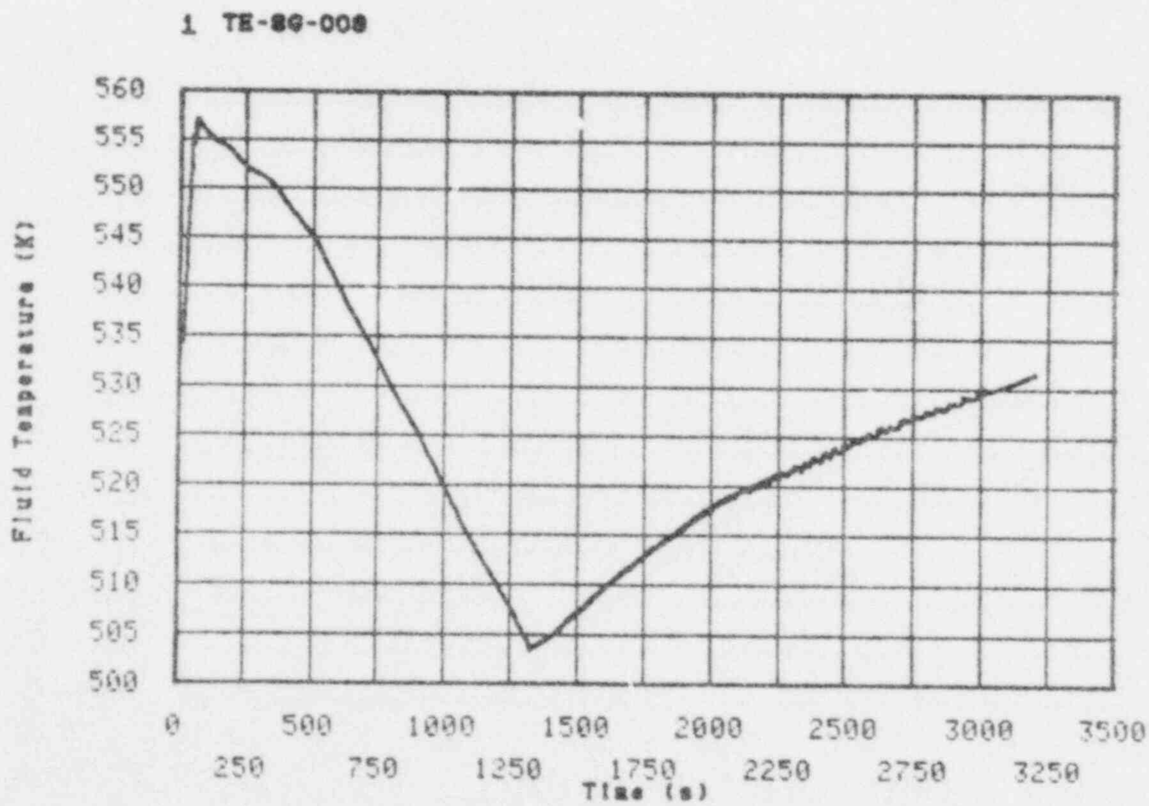
1 TE-SG-001



RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-48 Steam Generator Inlet

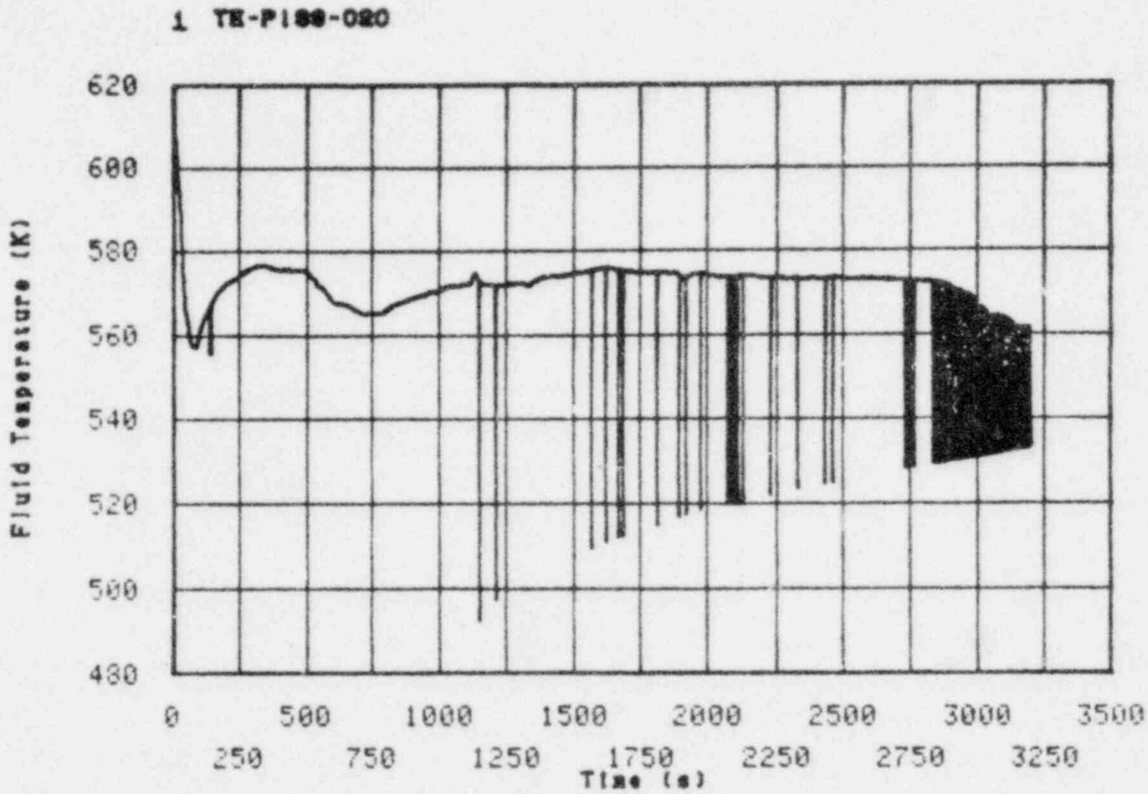


RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-47 Steam Generator Outlet



RELAP5 PREDICTION LOFT L8-5/L8-5A
Fig A-48 S/G Downcomer

POOR ORIGINAL



RELAP5 PREDICTION LOFT L9-5/L9-5A
Fig A-48 Pressurizer Liquid

APPENDIX B

UNITS CONVERSION OF RELAP5 DATA

APPENDIX B

UNITS CONVERSION OF RELAP5 DATA

This appendix describes in detail how the data output from the RELAP5 computer code are converted to an SI units prediction for a specific instrument. This allows the reader to associate the predicted SI units data to the computer code model which is utilized in making the prediction.

The algorithms that are used to calculate the predictions are provided on microfiche in the pouch on the inside of the report back cover.

APPENDIX C

RELAP5 UPDATE INPUT DATA

APPENDIX C

RELAP5 UPDATE INPUT DATA

A listing of the input data for updating RELAP5 is provided on microfiche in the pouch on the inside of the report back cover. The Idaho National Engineering Laboratory configuration control numbers for the RELAP5 source deck and update input data deck used in this prediction analysis are as follows:

1. The RELAP5/MOD"1" source deck is stored under Configuration Control Number H011985B.

2. The RELAP5/MOD"1" update input data deck is stored under Configuration Control Number H011785B.

3. The RELAP5 input deck is stored under Configuration Control Number H011885B.

APPENDIX D

RELAP5 INPUT DATA

APPENDIX D

RELAP5 INPUT DATA

The input deck listing for the RELAP5 model is provided on microfiche in the pouch on the inside of the report back cover.

DOCUMENT/ PAGE PULLED

ANO. 8010300194

NO. OF PAGES 1 microfiche

REASON

- [x] PAGE ILLEGIBLE
  - [x] HARD COPY FILED AT: PDR CF
    OTHER ______
  - [ ] BETTER COPY REQUESTED ON __/__/__
- [ ] PAGE TOO LARGE TO FILM
  - [ ] HARD COPY FILED AT: PDR CF
    OTHER ______
  - [ ] FILMED ON APERTURE CARD NO ______