EG&G Idaho, Inc.

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Best Estimate Prediction for LOFT Nuclear Experiment L3-7

Type of Document:

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Idaho Falls, Idaho 83415

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

INTERIM REPORT

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EG&G Idaho, Inc.

P. O. Box 1625
Idaho Falls, Idaho 83401

June 16, 1980

Mr. R. E. Tiller, Director
Reactor Operations & Programs Division
Idaho Operations Office - DOE
Idaho Falls, ID 83401

EXPERIMENT PREDICTION FOR LOFT L.CE L3-7 - Kau-137-80

Dear Mr. Tiller:

This letter transmits the Experiment Prediction (EP) for the Loss-of-Fluid Test (LOFT) Loss-of-Coolant Experiment L3-7.

LOCE L3-7 is the fourth experiment to be performed in the LOFT Nuclear Small Break Test Series (Test Series L3). The primary objectives of test L3-7 are to impose a break flow equal to HPIS flow at an intermediate pressure during the transient, to establish conditions for steam generator reflux cooling, to isolate the break and recover the plant to cold shutdown, and to analyze the data obtained to investigate associated phenomena.

The EP analyses provides 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. This EP was performed using RELAP5/MOD0 computer code.

The RELAP5 calculation simulated the thermal-hydraulic response from break initiation to system recovery and refill to a solid plant condition. The calculation indicates the test will achieve its stated objectives.

Very truly yours,

N. C. Kaufman
Director, LOFT

EJK:sem

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Experiment Prediction Report

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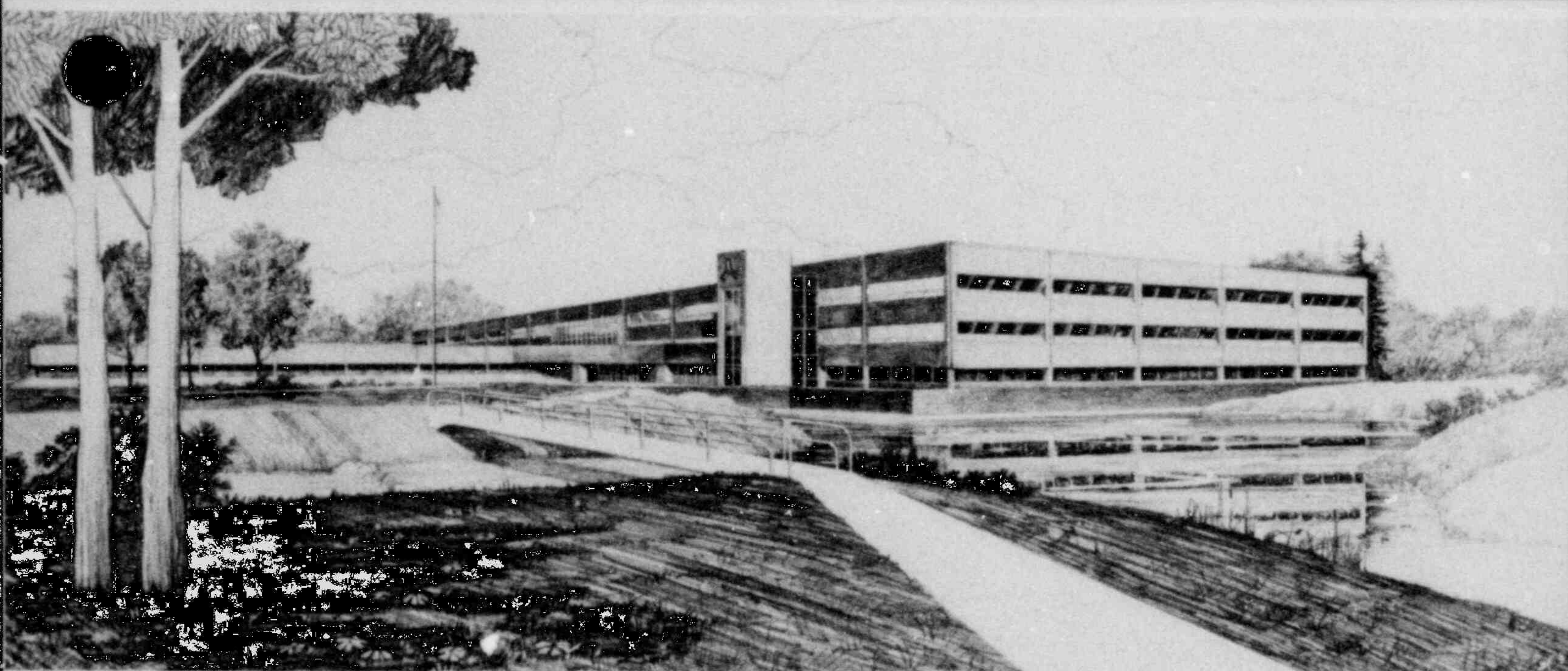
BEST ESTIMATE PREDICTION FOR LOFT NUCLEAR EXPERIMENT L3-7

Ernest J. Kee
Glen A. Taylor
Keith G. Condie

U.S. Department of Energy

Idaho Operations Office • Idaho National Engineering Laboratory



This is an informal report intended for use as a preliminary or working document

NRC Research and Technical Assistance Report

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Under DOE Contract No. DE-AC07-76ID01570
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EG&G Idaho

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Division of Reactor Safety Research, USNRC

> This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final.

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

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U.S. Nuclear Regulatory Commission
Washington, D.C.
Under DOE Contract No. DE-AC07-76ID01570
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INTERIM REPORT

BEST ESTIMATE PREDICTION FOR LOFT NUCLEAR EXPERIMENT L3-7

By:

E. J. Kee

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K. G. Condie

Approved:

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LOFT Program Division

The LOFT Subcommittee of the EG&G Pretest Prediction Consistency Committee has reviewed the RELAP5 model and predicted results for LOFT Small Break Experiment L3-7.

Code Development

C B Davis for T R Charlton

Code Assessment

Semiscale

Thermal Fuels

ABSTRACT

A two-fluid, transient analysis, digital computer code RELAP5/MOD"0" was used to simulate a small break in the cold leg of the Loss-of-Fluid Test (LOFT) facility, a large pressurized water reactor scale model test facility. The entire transient was simulated from break opening to recovery to a liquid-full condition. The simulation indicates that energy generated in the nuclear core and not leaving through the break can be adequately removed in the steam generator.

NRC FIN No. A6048 - LOFT Experimental Program

CONTENTS

FIGURES

TABLES

DEFINITIONS

Best estimate-type calculation - a t[illegible]nt similation 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 - loop circulation (flow) 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 vessel upper plenum.

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) L3-7. LOCE L3-7 is the third experiment to be performed in the LOFT Nuclear Small Break Test Series (Test Series L3). The primary objectives of LOCE L3-7 are to impose a break flow equal to HPIS flow at an intermediate pressure during the transient, to establish conditions for steam generator reflux flow[a], to isolate the break and recover the plant to cold shutdown, and to analyze the data obtained to investigate associated phenomena.

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 RELAP5 computer code was used to perform this EP analysis. Since the initial conditions and the conditions for the first 1800 s of the transient for LOCE L3-7 will be the same as those conditions for LOCE L3-2 (the preceding LOFT small break experiment), results of the RELAP4 EP analysis for LOCE L3-2 are presented for this time period for comparison.

The results of the analysis indicate that natural loop circulation will persist throughout the LOCE L3-7 transient, with some time periods when reflux flow may occur as well. The plant is predicted to be liquid full at about 11 000 s after experiment initiation. Significant events during LOCE L3-7 are calculated to occur at the following times into the transient:

a. The objectives stated here for LOCE L3-7 are as stated in the Experiment Operating Specification, except the term "reflux flow" was changed from "reflux cooling."

1. The reactor will scram and the primary coolant pumps will trip at 34 s

2. The high-pressure injection system (HPIS) will start injecting emergency core coolant (ECC) into the intact loop at 62 s

3. The secondary coolant system auxiliary feed pump will start initial steam generator fill at 94 s

4. The pressurizer will be empty at 400 s

5. The upper plenum fluid will reach saturation temperature at 450 s

6. The HPIS flow rate will be higher than or equal to the flow rate out the break at 1500 s

7. The HPIS will be turned off at 1800 s

8. The secondary coolant system auxiliary feed pumps will trip, ending initial steam generator fill at 1894 s

9. The secondary coolant system steam bleed will start at 3600 s

10. The HPIS will again start injecting ECC at 5400 s

11. The accumulator will start injecting ECC at 6080 s

12. The quick-opening blowdown valve isolation valve will close at 7200 s

13. The primary system fluid will be subcooled at 11 000 s.

1. INTRODUCTION

As part of the experiment analysis effort performed by the Loss-of-Fluid Test (LOFT) 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 the program and provide data for:

1. Determining whether an experiment 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 analysis.

This document describes how the RELAP5 computer code was used to simulate the LOFT system response for Loss-of-Coolant Experiment (LOCE) L3-7. Results from this analysis and, because of the similarity between LOCEs L3-7 and L3-2, the first 1800 s of the LOCE L3-2 RELAP4 analysis are presented to illustrate the overall system response and to provide assurance that the experiment objectives will be met.

Sections 1.1 and 1.2 of this introduction discuss the LOCE L3-7 objectives and provide a brief description of LOCE L3-7 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 L3-7 Objectives and Description

LOCE L3-7 is the third powered experiment to be conducted as part of the LOFT Nuclear Small Break Test Series L3. The experiment objectives and descriptions for Test Series L3 are discussed in detail in Reference 1. The objectives for LOCE L3-7 are given in Section 1.1.1. LOCE L3-7 is described in Section 1.1.2.

1.1.1 LOCE L3-7 Objectives

The primary objectives of LOCE L3-7 are to impose a break flow equal to HPIS flow at an intermediate pressure during the transient, to establish conditions for steam generator reflux flow[a], to isolate the break and recover the plant to cold shutdown, and to analyze the data obtained to investigate associated phenomena.

1.1.2 LOCE L3-7 Description

LOCE L3-7 will represent a 0.16% break in a cold leg of a pressurized water reactor (PWR) primary coolant pipe. The periods of interest are from break initiation to plant depressurization and long-term cooldown. When operator action is required to effect depressurization, a secondary heat removal method recommended by large PWR vendors will be used.

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a. The objectives stated here for LOCE L3-7 are as stated in Reference 1, except the term "reflux flow" was changed from "reflux cooling."

1.1.2.1 Initial Experiment Conditions. The following major initial conditions were selected for LOCE L3-7 to simulate the conditions expected at the start of a small break loss-of-coolant accident (LOCA) in a typical large PWR:

1. The reactor has been operating at steady state 100% power long enough to establish equilibrium fission product concentrations.

2. There has not been a loss of site power coincident with the LOCA which requires the emergency core coolant (ECC) injection to be activated by automatic signals and not be delayed until after the emergency diesel is delivering power.

3. The minimum emergency core cooling system (ECCS) action takes place requiring the high-pressure injection system (HPIS) and the low-pressure injection system (LPIS) flow rates to be scaled to represent only one of the two pumps available for each system. The accumulator volume was scaled to represent the four accumulator tanks available on a typical large PWR.

Initial conditions specified for the LOFT system at the initiation of LOCE L3-7 are: calculated core power - 50 MW, primary system pressure - 14.86 MPa, intact loop cold leg temperature - 556.8 K, and primary coolant flow rate - 478.8 kg/s.

1.1.2.2 Experiment Operation. The reactor will be taken critical and operated at 100% power (50 MW) for a period long enough to establish sufficient fission product concentrations to provide decay heat levels corresponding to 86% of 1-year irradiation time at 100 s after shutdown and 67% of 1-year irradiation time at 1 h after shutdown.

After all of the specified initial conditions have been established, the blowdown system isolation valve on the vessel side of the broken loop cold leg quick-opening blowdown valve (QOBV) will be opened. The experiment will then be initiated by opening the corresponding downstream QOBV.

The control rods will be scrammed by the reactor shutdown system when a low system pressure (14.11 MPa) is indicated. Power to the primary coolant pumps will be tripped when the lights indicating the control rods have reached bottom are lighted on the control panel, approximately 2 s after the scram signal is received. The pumps will then coast down at a rate representative of a typical large PWR.

The blowdown effluent from the primary coolant system will be directed to the pressure suppression tank. Back pressures in the pressure suppression tank are calculated to not have enough influence on small break experiments to require using a programmed back pressure.

Upon receipt of a low system pressure signal of 13.16 MPa, a mass scaled amount of borated liquid will be injected into the primary coolant system cold leg from the HPIS pump. The HPIS will be shut off at 1800 s after experiment initiation. The LOFT system is expected to stabilize at about 7.5 MPa, when the HPIS flow will be nearly equal to the flow out the break. After 1 h of operation, the steam generator will be cooled at the rate of 39 to 50 K/h by bleeding steam from the steam generator while adding auxiliary feedwater to maintain steam generator water level. The HPIS will be turned on at 5400 s into the transient. One accumulator and one LPIS pump will also be available to inject mass scaled amounts of borated water into the primary system when the primary pressure is lower than the pressure maintained by these ECCSs.

At 7200 s into the transient, the cold leg QOBV and isolation valve will be closed. Again the steam generator bleeding operations will be performed. When primary system pressure is less than 2.41 MPa and hot leg subcooling is greater than 27.8 K, purification flow will be initiated through the nonregenerative heat exchanger to cool the plant at < 55.6 K/h when primary coolant temperature is > 449.8 K, and at < 17.8 K/h when primary coolant temperature is < 449.8 K.

The accumulator will be isolated from the primary coolant system when its liquid has been injected but prior to nitrogen from the accumulator entering the system. This will occur when primary system pressure is 1.52 MPa.

Section 2 of this report discusses the RELAP5 simulation of LOCE L3-7. The RELAP4 simulation of LOCE L3-2 is presented in Reference 2. Since no significant core thermal transient was predicted, no detailed fuel rod calculations were performed.

The simulations presented in this report have been reviewed by the Experiment Prediction Consistency Committee at the Idaho National Engineering Laboratory and have been found to be in accordance with current and accepted practices.

1.2 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 6. The instrumentation nomenclature is explained in Table 1.

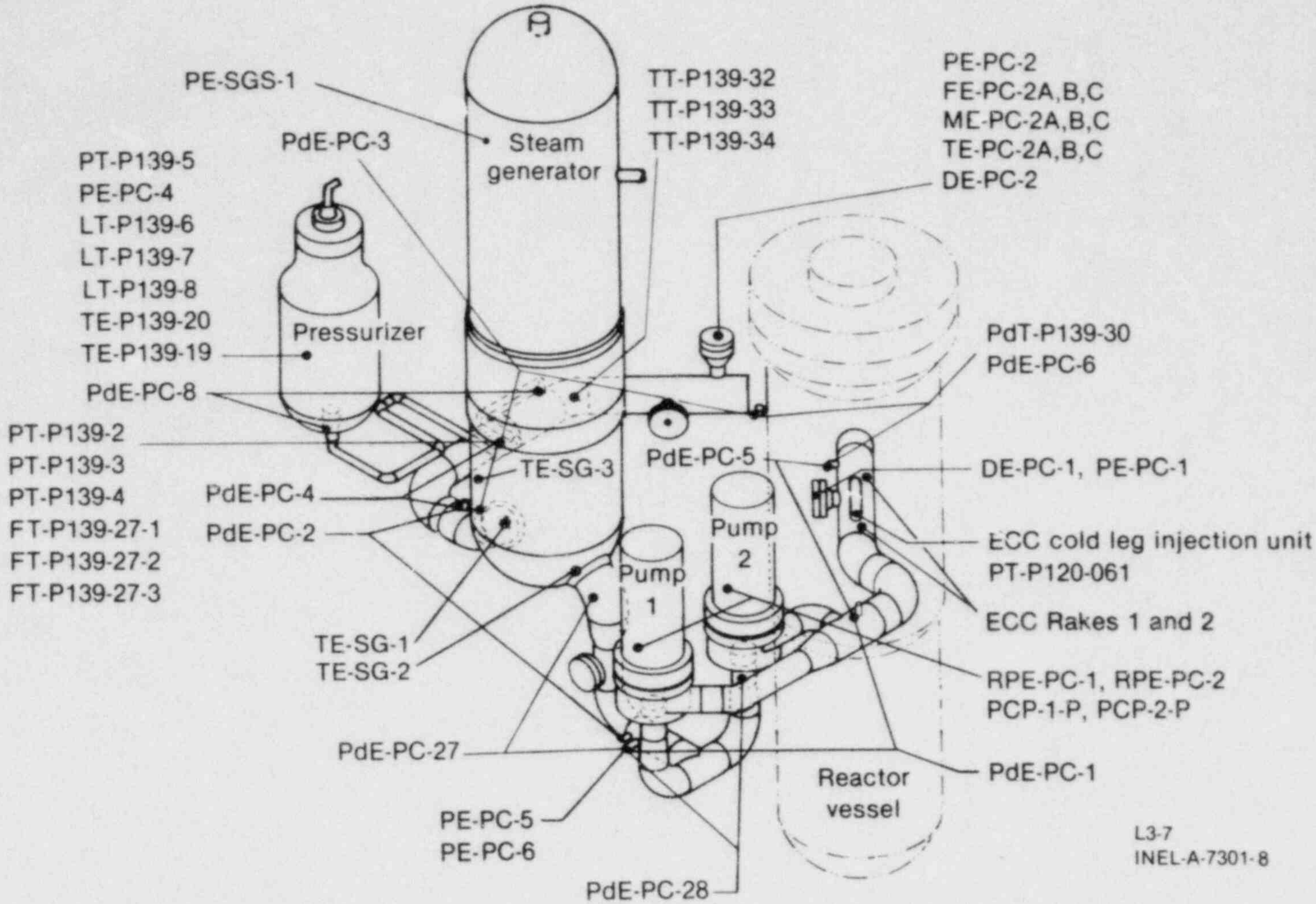
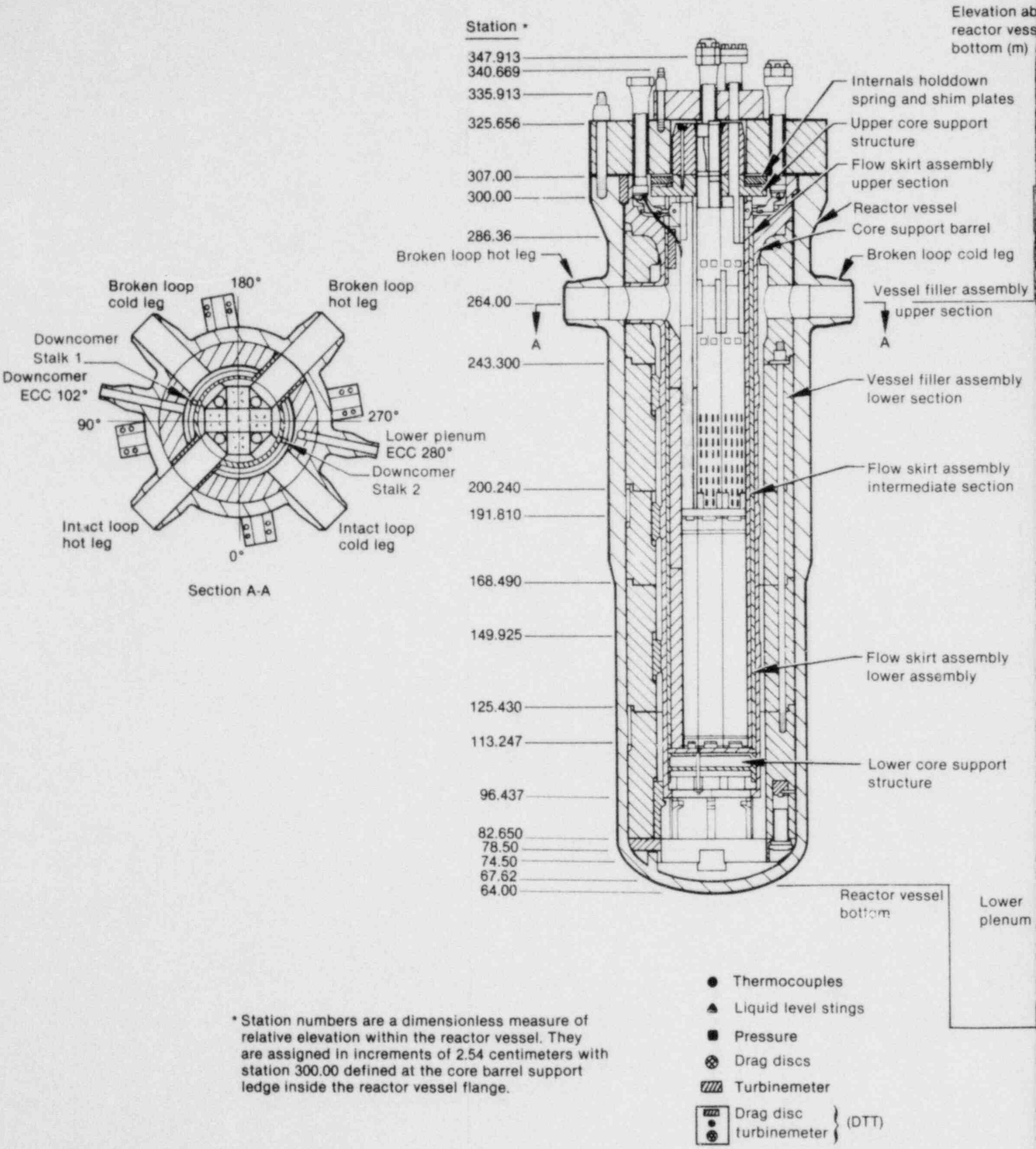
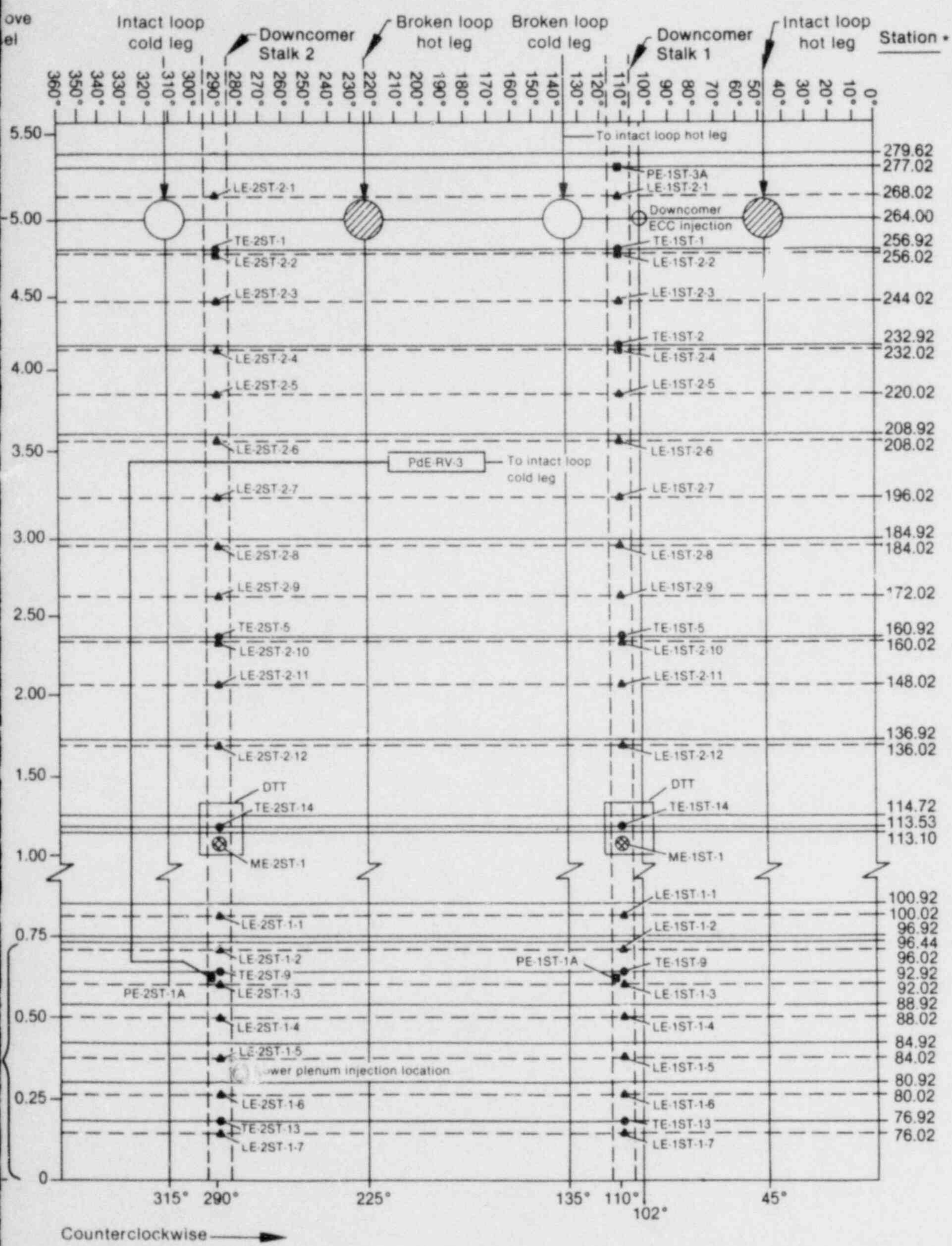


Figure 1. LOFT intact loop thermo-fluid instrumentation.



* Station numbers are a dimensionless measure of relative elevation within the reactor vessel. They are assigned in increments of 2.54 centimeters with station 300.00 defined at the core barrel support ledge inside the reactor vessel flange.

Figure 3. LOFT reactor vessel in



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strumentation.

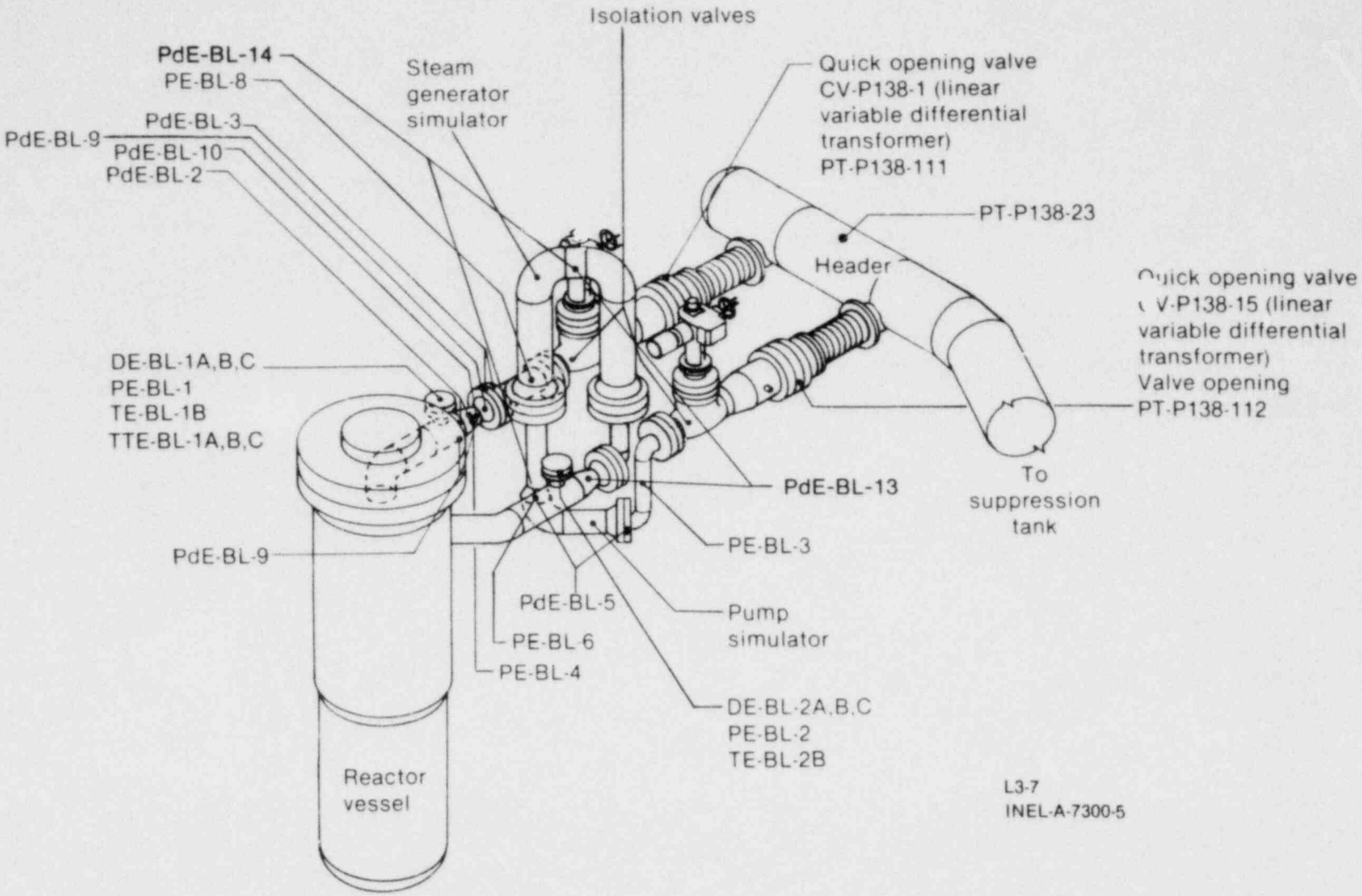


Figure 2. LOFT broken loop thermo-fluid instrumentation.

* 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.

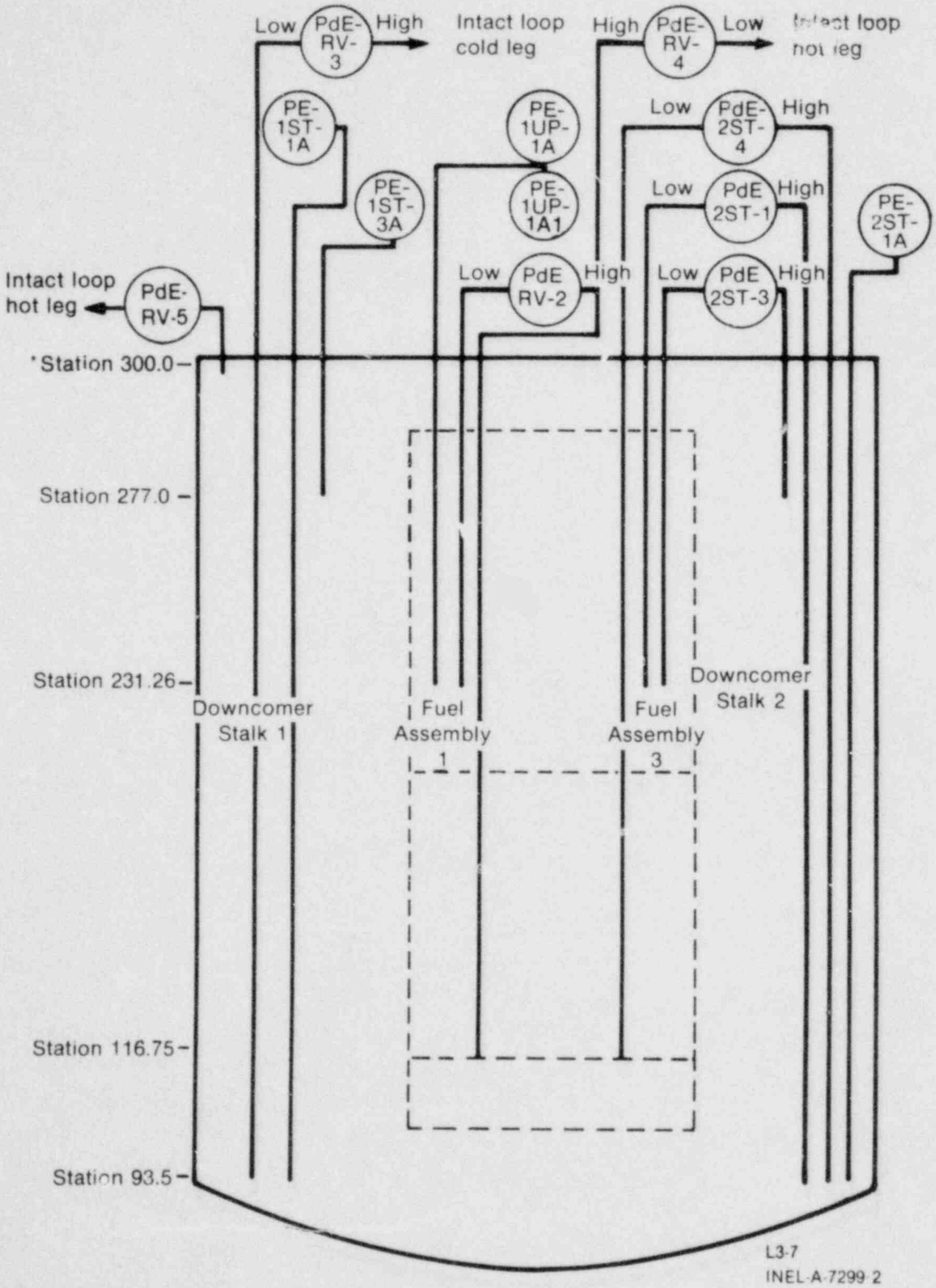


Figure 4. LOFT reactor vessel pressure and differential pressure instrumentation.

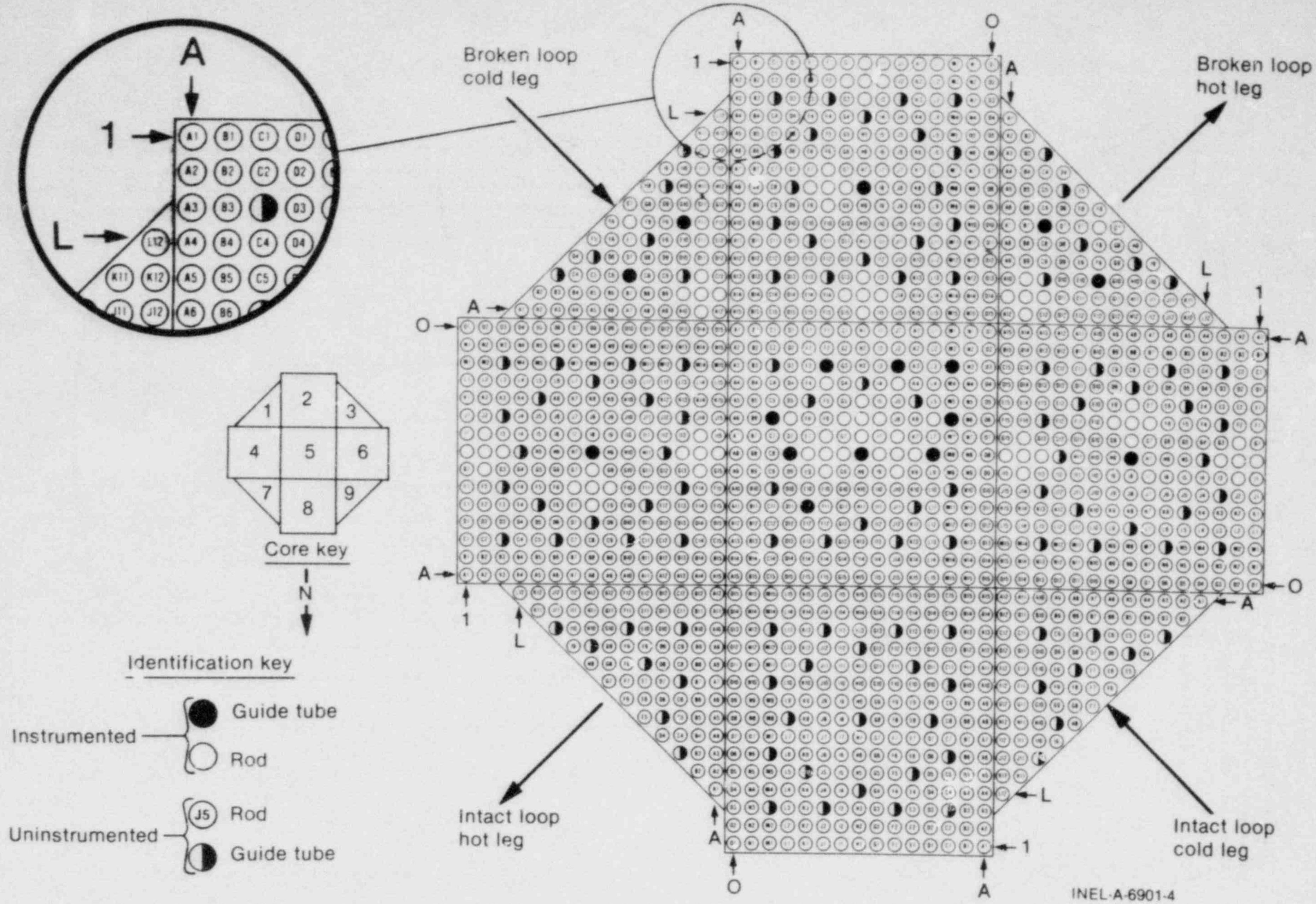
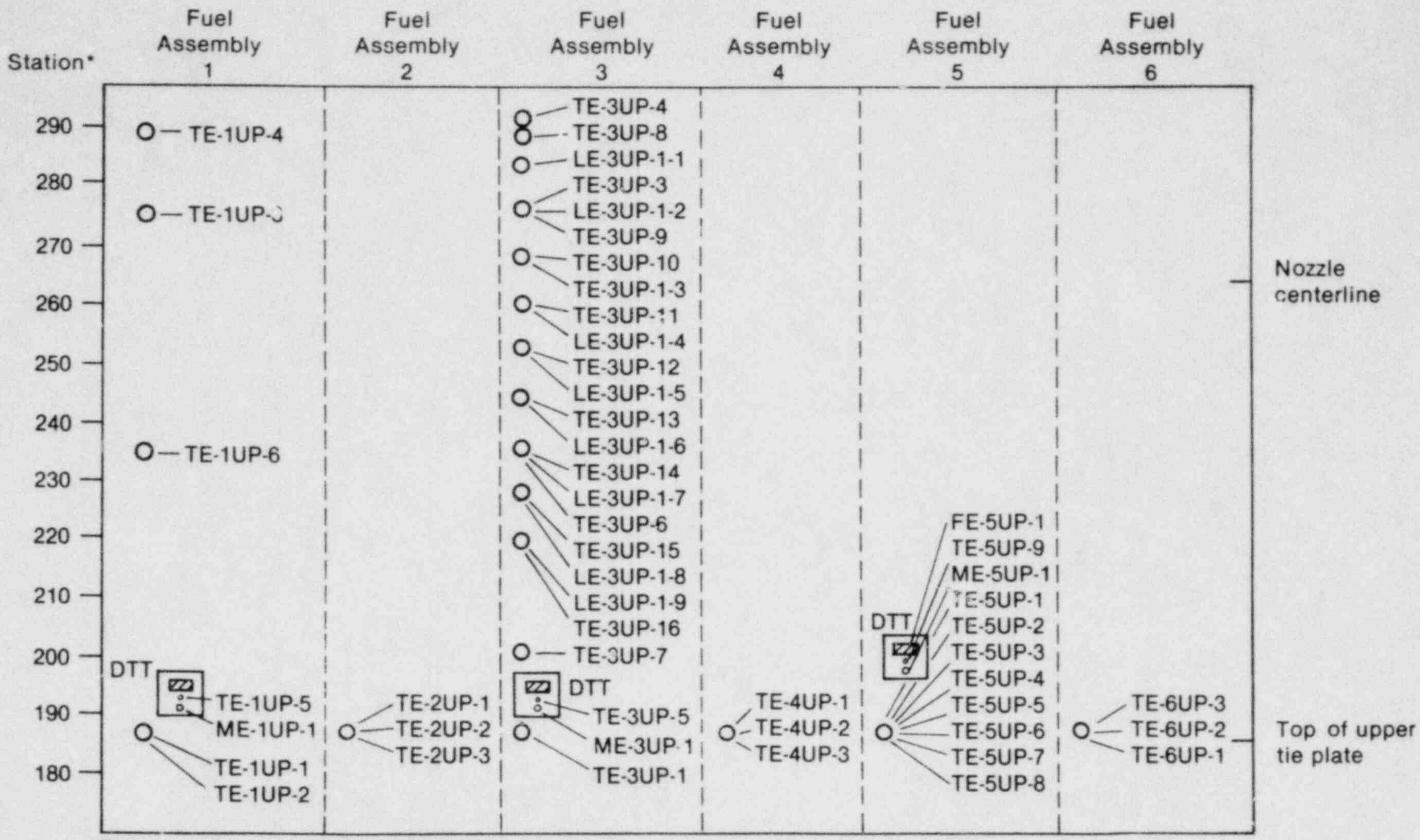


Figure 5. LOFT core map showing position designations.



*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.

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Figure 6. LOFT reactor vessel upper plenum drag disc-turbine and coolant level transducers and temperature element elevations.

TABLE 1. NOMENCLATURE FOR LOFT INSTRUMENTATION

The designations for the different types of transducers:

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:[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.

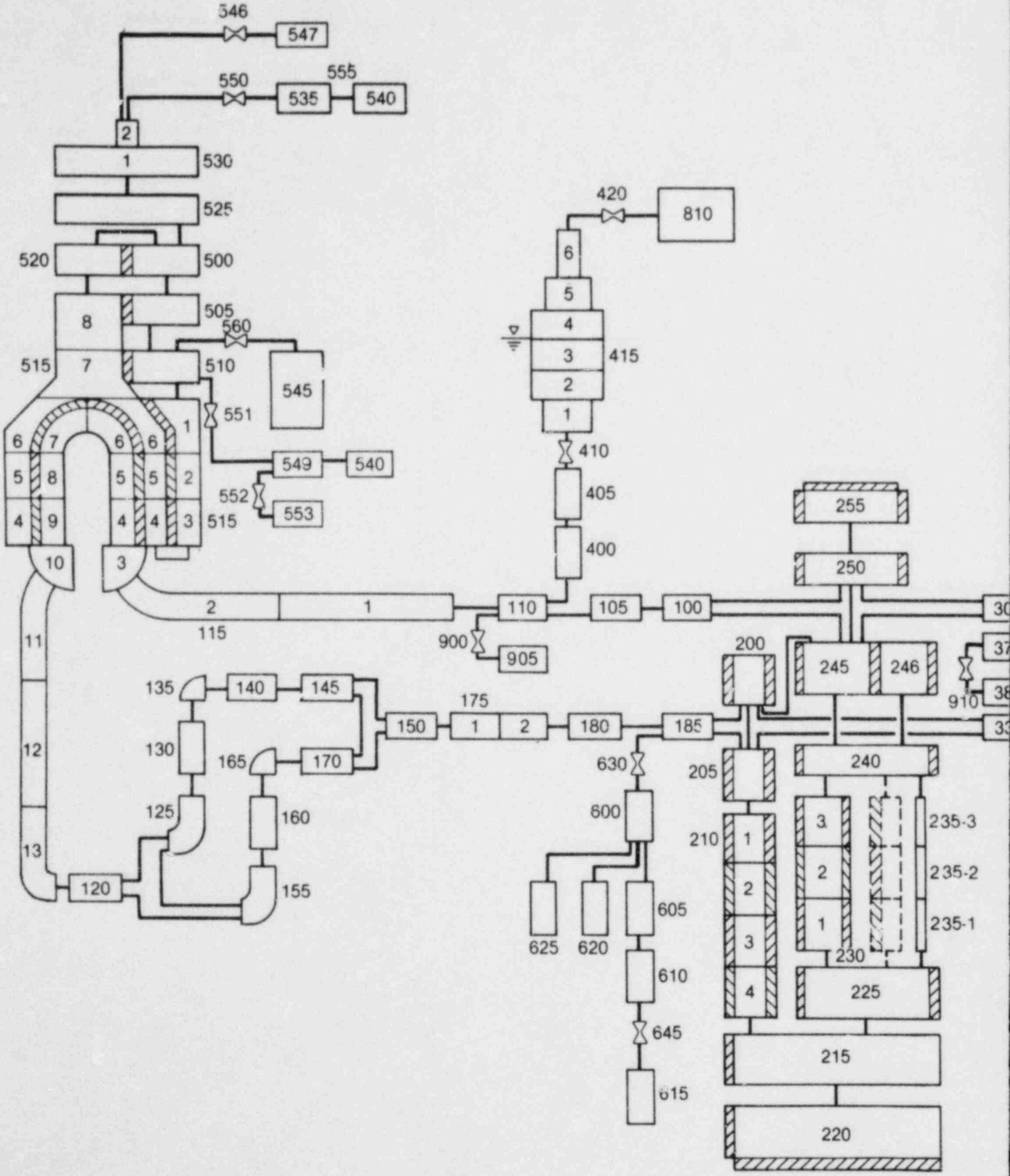
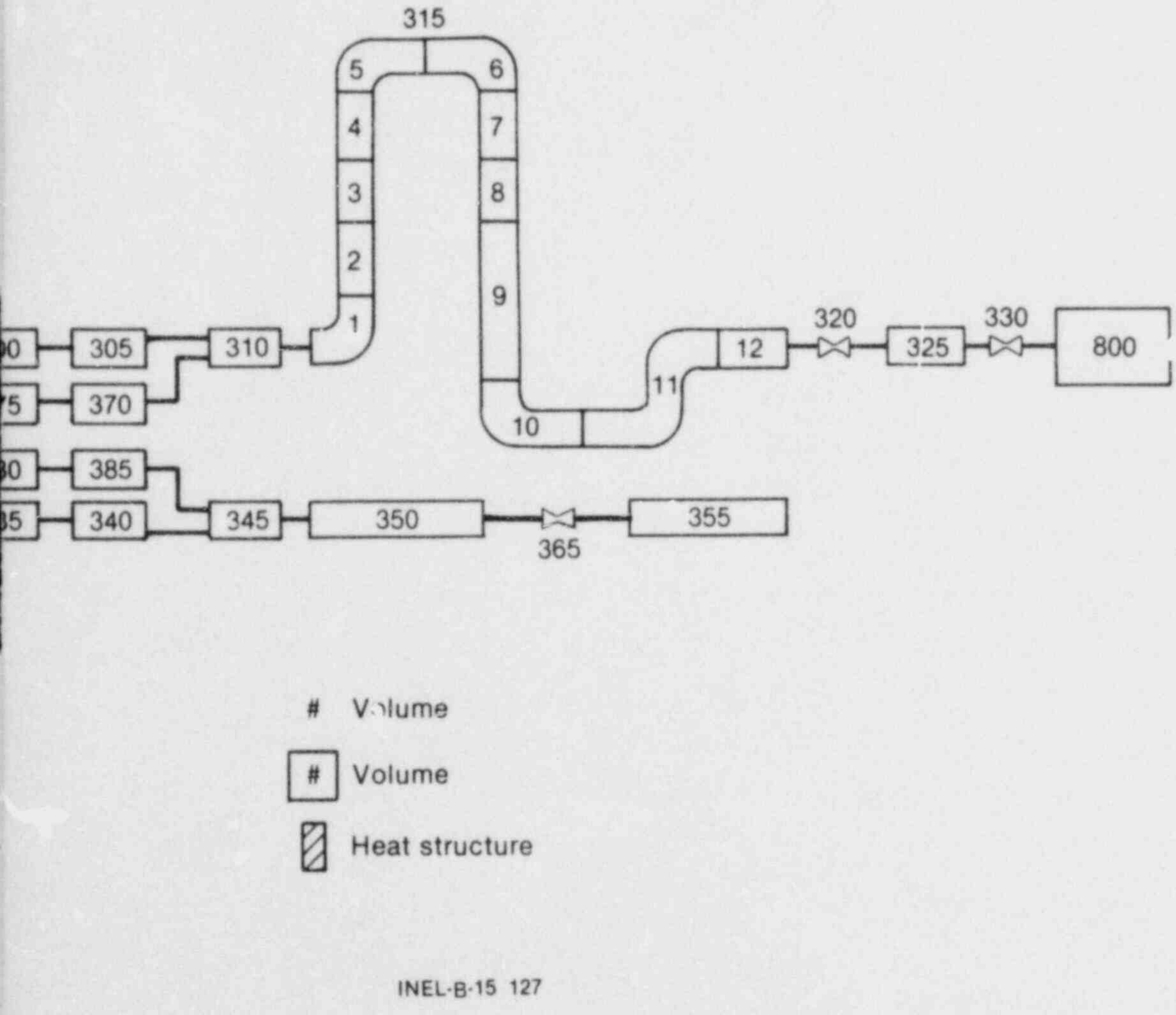


Figure 7. LOFT RELAP5 model s



chematic diagram.

2. COMPUTER SIMULATION

The RELAP5/MOD"0" computer code[a] was used to simulate the transient thermal-hydraulic response for the LOFT system during LOCE L3-7. The first 1800 s of data from the EP analysis for LOCE L3-2 that are used for comparison in this EP analysis were obtained using the RELAP4/MODG computer code.[b] The RELAP5 code uses a two-fluid, thermal nonequilibrium hydraulic model. The specific application of the code to the LOCE L3-7 simulation is discussed in this section.

2.1 Nodalization

The nodalization used for the LOCF L3-7 RELAP5 calculation is presented in this section. This nodalization is a proposed standard nodalization for LOFT suitable for use in RELAP5. It uses more nodes than the RELAP5 calculation for LOCE L3-2 and is based on the most recent LOFT description data. The nodalization scheme is shown on Figure 7. A brief description of each node is given in Table 2.

There are several bypass flow paths in the LOFT system. These bypass flow paths are shown on Figure 8 with the arrows showing the flow direction at test initiation. The pressurizer continuous spray bypass is considered minor and was not simulated in this calculation. The other three, lower plenum to upper plenum bypass, inlet annulus to upper plenum bypass, and reflood assist bypass valve (RABV) leakage bypass, were simulated. In the current simulation, the inlet annulus to upper plenum bypass represents 2% of the full intact loop flow and the RABV leakage, 3% of full flow. The

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a. RELAP5/MOD"0" is filed under Idaho National Engineering Laboratory Configuration Control Number H012381B.

b. RELAP4/MODG, an experimental version of RELAP4/MOD7, is filed under Idaho National Engineering Laboratory Configuration Control Number H00718B; the new object deck, which included changes required for LOCE L3-2, was RLP4G92LFT04 and is filed under Idaho National Engineering Laboratory Configuration Control Number H01168IB.

TABLE 2. DESCRIPTION OF NODALIZATION COMPONENTS

Component Number	Component Type	Volume Number	Number of Junctions	Composition
100	Branch	1	3	Core barrel nozzle, vessel nozzle, and half of flow device.
105	Branch	1	1	Half of flow device, 45-degree elbow, and pipe to pressurizer connection.
110	3ranch	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.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
120	Branch	1	3	Pipe section and inlet pipe of pump suction tee.
125	Branch	1	2	Half of pump suction tee, 90-degree elbow, and half of reducer.
130	SNGLVOL	1	0	Half of reducer and Pump 1 inlet pipe.
135	Pump	1	2	Primary coolant Pump 1.
140	SNGLVOL	1	0	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 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	3	Pipe section from ECC connection, vessel nozzle, and vessel filler.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
200	Branch	1	2	Upper part of inlet annulus distributor.
205	Branch	1	1	Lower part of inlet annulus distributor.
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	3	Core barrel nozzle and vessel nozzle.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
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.
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.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
325	SNGLVOL	1	0	Half of isolation valve and half of QOBV.
335	Branch	1	3	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.
360	SNGLVOL	1	0	Half of isolation valve cold leg and half of QOBV cold leg.
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 system side.
405	SNGLVOL	1	0	Pressurizer surge line, pressurizer side.
415	Pipe	1		Pressurizer inlet.
		2		Pressurizer vessel water space.
		3		Pressurizer vessel water space.
		4		Pressurizer vessel vapor space.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
		5		Pressurizer vessel vapor space.
		6		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.
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.

TABLE 2. (continued)

Component Number	Component Type	Volume Number	Number of Junctions	Composition
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.
600	Branch	1	--	Pipe between cold leg and low-pressure injection pump tee.
605	Branch	1	--	Half of piping between low pressure injection pump tee and accumulator.
610	SNGLVOL	1	--	Pipe between accumulator outlet and Component 605.
615	SNGLVOL	1	--	Accumulator.
620	TMDPVOL	1	--	Borated water storage tank.
625	TMDPVOL	1	--	Borated water storage tank.

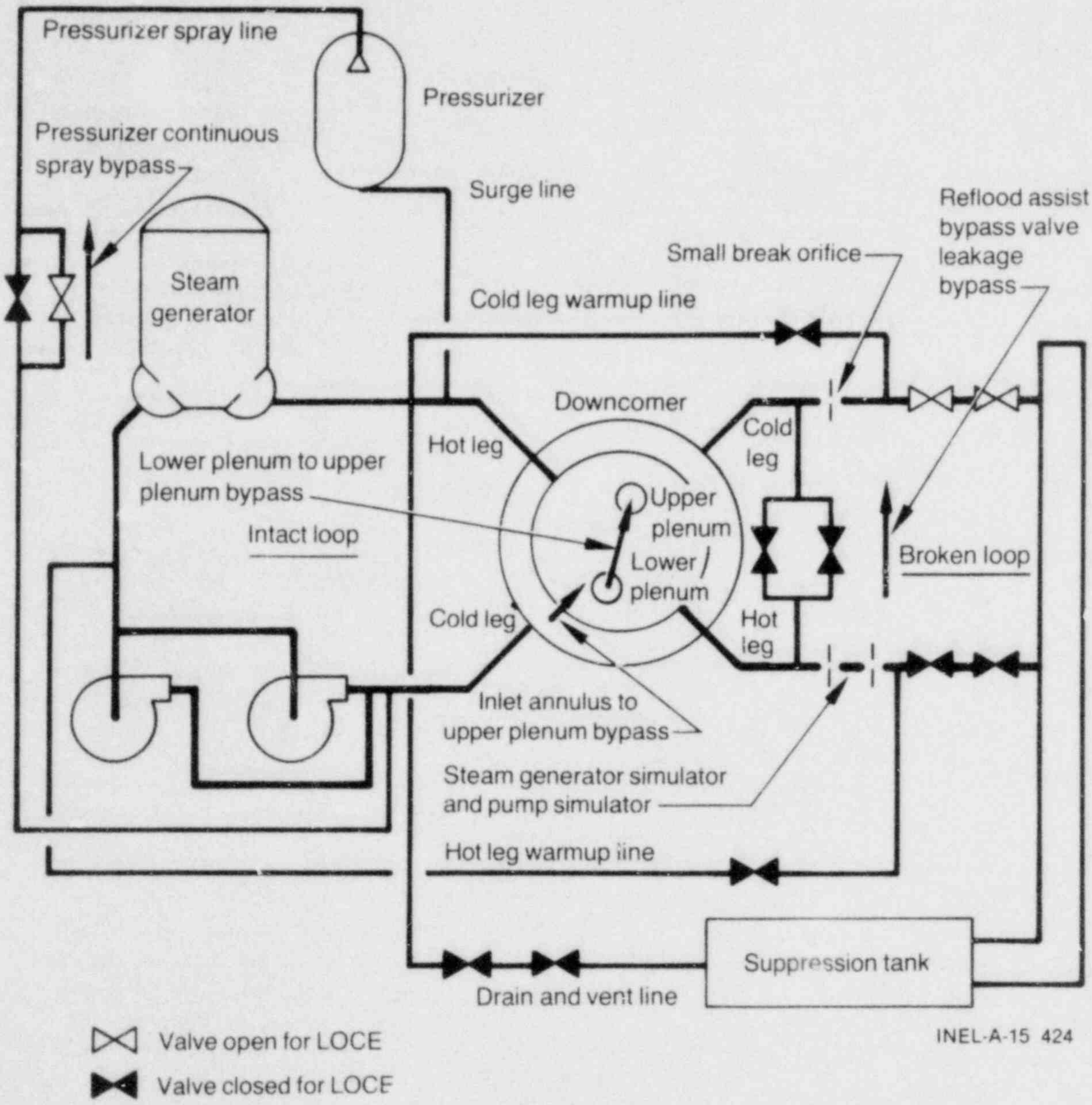

Figure 8. LOFT system schematic showing bypass flow paths.

lower plenum to upper plenum bypass was simulated as having 5% of full intact loop flow at experiment initiation. The total core bypass, therefore, is 10% of full flow which is the bypass calculated to exist in previous LOCEs L2-3, L3-1, and L3-2.

2.2 Special Models

Special models were installed in the RELAP5 code for this calculation by updating[a] the MOD"0" version of the code. These models are discussed below.

2.2.1 Separator

The primary separator and mist extractor in the steam generator were modeled by modifying the donor formulation of the convective terms for Component 500 in Figure 7. The separator was assumed to be 100% efficient when in normal operation. An efficiency factor could be included in this model for abnormal operation, such as, flooded; however, since this condition is not expected to occur in the LOCE L3-7 transient, it was not included in this analysis.

2.2.2 Motor-Operated Valve

The steam flow control valve in the steam generator outlet piping was designed to operate as a pressure regulator. Since this pressure regulating function is expected to be used during LOCE L3-7, a model was developed to simulate a constant speed, motor-driven valve. The flow area in the valve was assumed to change linearly as the stem position changed and the driver was assumed to have no inertia.

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a. The update input deck, which includes these changes, is filed under Idaho National Engineering Laboratory Configuration Control Number H006685B.

2.2.3 Accumulator

An experimental accumulator model was used for this calculation to represent ECCS Accumulator A. The accumulator gas space is assumed to be nitrogen filed and act as an ideal gas. Tank wall-to-gas and water-to-gas heat transfer was assumed, allowing for a polytropic-type gas expansion.

2.2.4 Pump

The HPIS Pump A is a variable speed, positive displacement-type pump with flow controlled by the operator as a function of primary system hot leg pressure for LOCE L3-7. This pump was modeled by explicitly varying flow from Component 620 as a function of pressure in Component 100 according to the operator's schedule.

2.2.5 Heat Transfer

The current version of RELAP5/MOD"0" uses a Dittus-Boelter-type correlation for heat transfer from the fluid to the wall. Because LOCE L3-7 is intended to achieve the reflux flow mode of decay heat removal, a condensation heat transfer model was installed in the code for the heat transfer calculation of single-phase steam or two-phase fluid to the wall at wall surface temperatures less than gas-phase saturation temperature. The saturated nucleate boiling in the shell side of the steam generator was changed to use Thom's correlation because the tubes are in cross flow.

3. CALCULATIONAL RESULTS

This section gives a general overview of the transient simulation and summarizes the calculational results on natural loop circulation, reflux flow operation, and the steam generator feed and bleed operation.

3.1 General

The LOCE L3-7 transient simulation was characterized by the upper plenum pressure history shown on Figure 9. The experiment will be initiated by opening the cold leg QOBV. As the pressurizer begins to empty, the system pressure will decrease until the scram pressure setpoint is reached at about 34 s. The resulting transient will cause pressure to decrease even more rapidly, and the HPIS will trip on at about 62 s due to low primary system pressure. The break flow will be larger than the HPIS flow, allowing further depressurization and eventual pressurizer emptying at about 400 s. As the pressurizer empties and is no longer the component with the highest pressure in the primary system, the fluid in the reactor vessel upper plenum and hot leg will be no longer subcooled, thus the fluid will begin boiling and can be said to be controlling system pressure. At 1800 s into the transient, HPIS will be turned off by the operator and the boiling in the hot leg will have progressed to the steam generator tubing. The steam flow control valve operation will be reflected in the primary system pressure fluctuations as a result of the two-phase mixture appearing in the tubes.

At 3600 s, a 44.5-K/h steam generator cooldown will be initiated by opening the steam flow control valve bypass valve. After 5400 s, the operator will restart HPIS flow, and the primary system will begin refilling while pressure will continue to decrease. By 6080 s, pressure will have decreased sufficiently to allow accumulator flow to start. At 7000 s into the transient, the liquid in the primary loop will begin to subcool and the pressure will cease to drop rapidly, even though the steam generator cooldown will be continuing. At 7200 s, the break will be

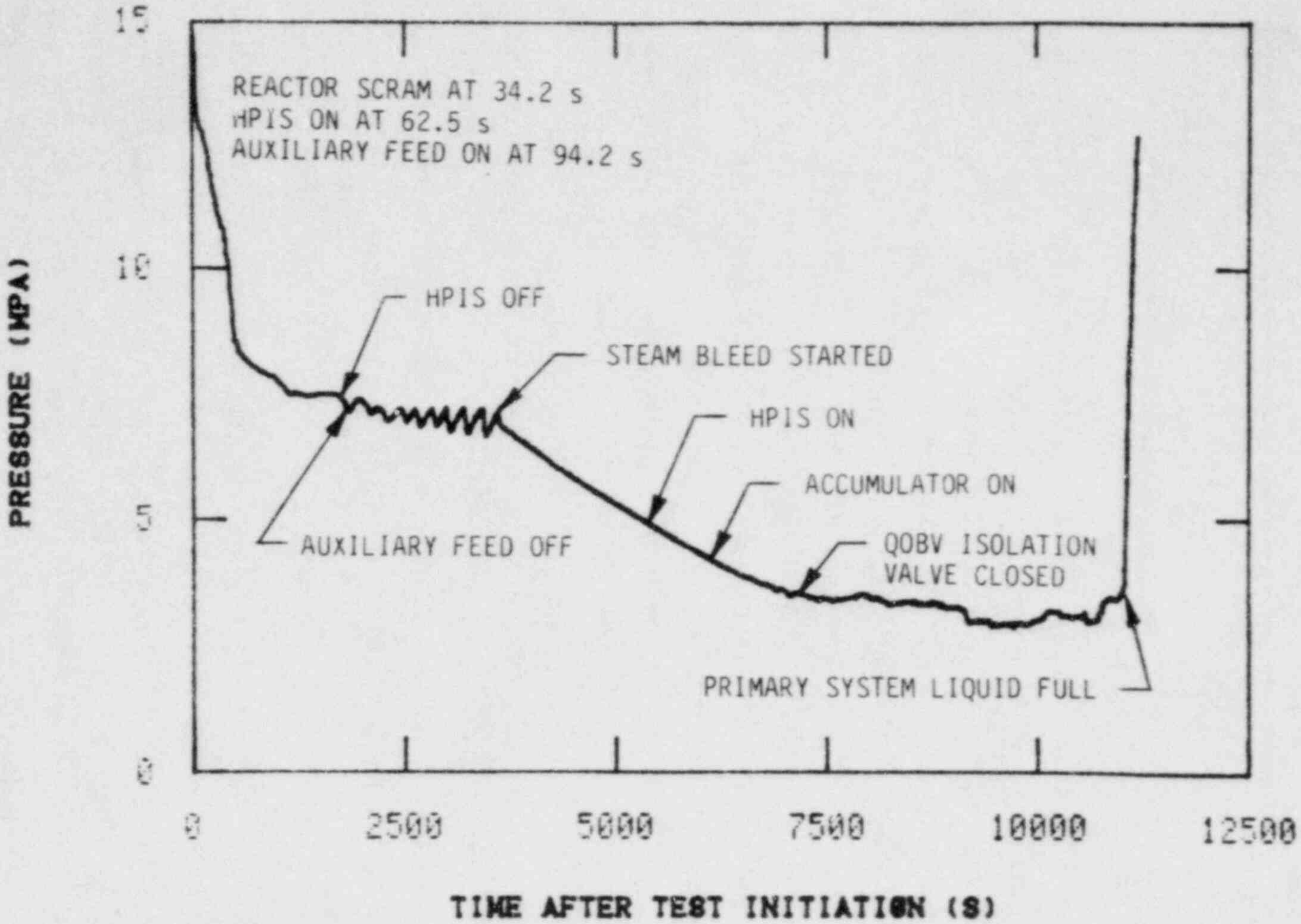


Figure 9. Pressure in reactor vessel upper plenum.

isolated and the system will rapidly refill. Shortly after 11 000 s, the system will become liquid full and pressure control can be effected by the operator.

The break flow is shown on Figure 10, indicating the magnitude of the break. Voids appearing upstream of the break orifice caused the variations seen in the 2500- to 3600-s time interval.

3.2 Natural Circulation

Natural loop circulation is a flow mode by which energy is transferred from the nuclear core to the steam generator in a PWR. Energy addition from the core and removal in the steam generator combined with the elevation difference between them will provide the thermal driving head for this flow. Figure 11 shows the total hydrostatic head in the intact loop. This positive head will drive the flow, see Figure 12, through the primary loop.

3.3 Reflux Flow

Energy can be removed from the core of a PWR by evaporating liquid in the core region, allowing it to rise to the steam generator and be condensed. When the condensate returns by counter-current flow to the core via the hot leg, the system is said to be in the reflux flow mode of decay heat removal. The code, as currently configured, cannot predict reflux in the horizontal hot leg. As can be seen on Figure 13, however, low liquid velocities are predicted to occur in the hot leg during the 3000- to 4000-s and 5500- to 6000-s time periods. This could indicate that reflux could possibly occur during this transient; however, the major energy removal mechanism appears to be 2ϕ natural loop circulation in this simulation. Figure 14 shows the fluid densities in the steam generator inlet and outlet. These densities indicate that the two-phase mixture leaving the core will be condensed in the steam generator.

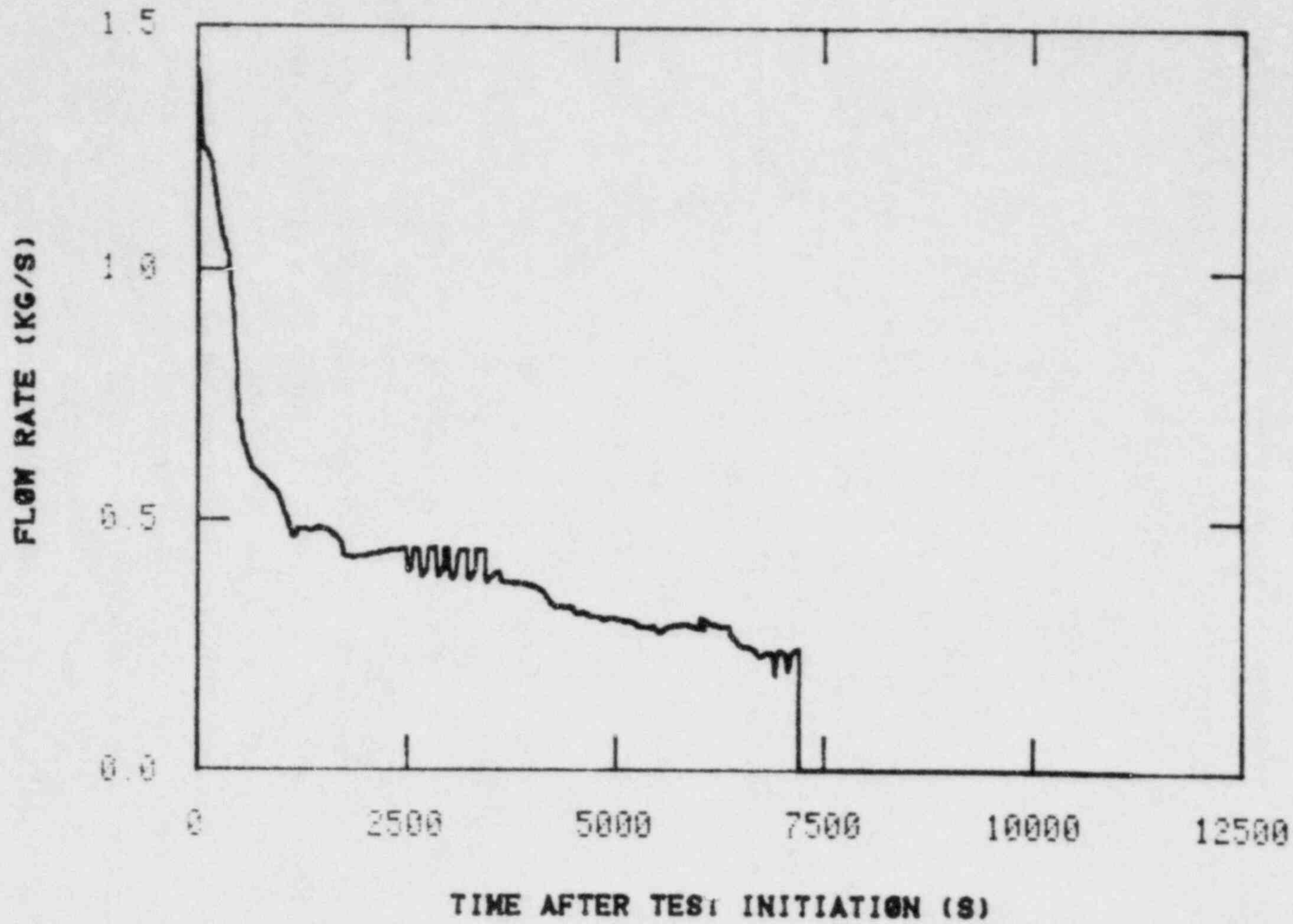


Figure 10. Mass flow rate at the break location.

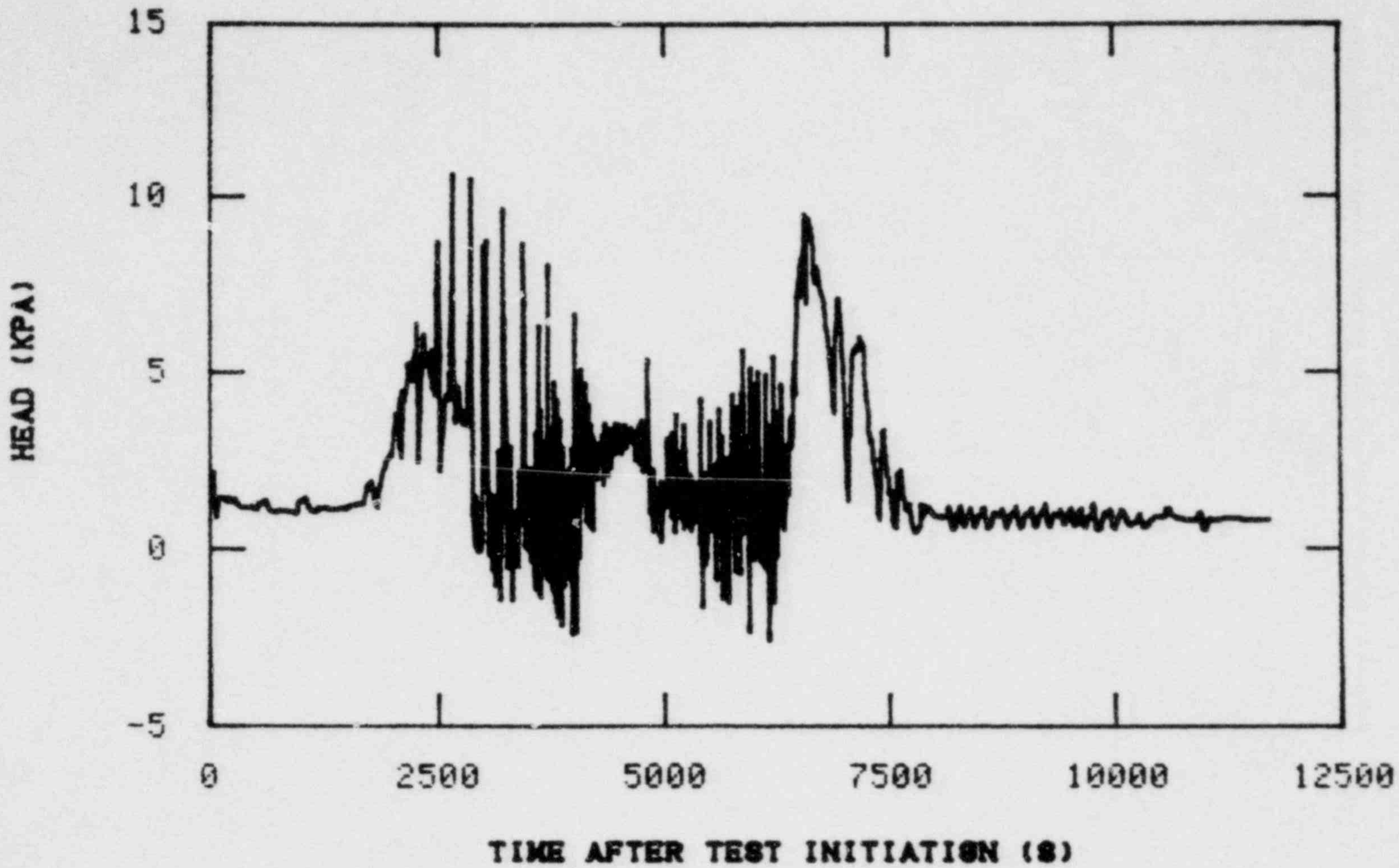


Figure 11. Total hydrostatic head in intact loop.

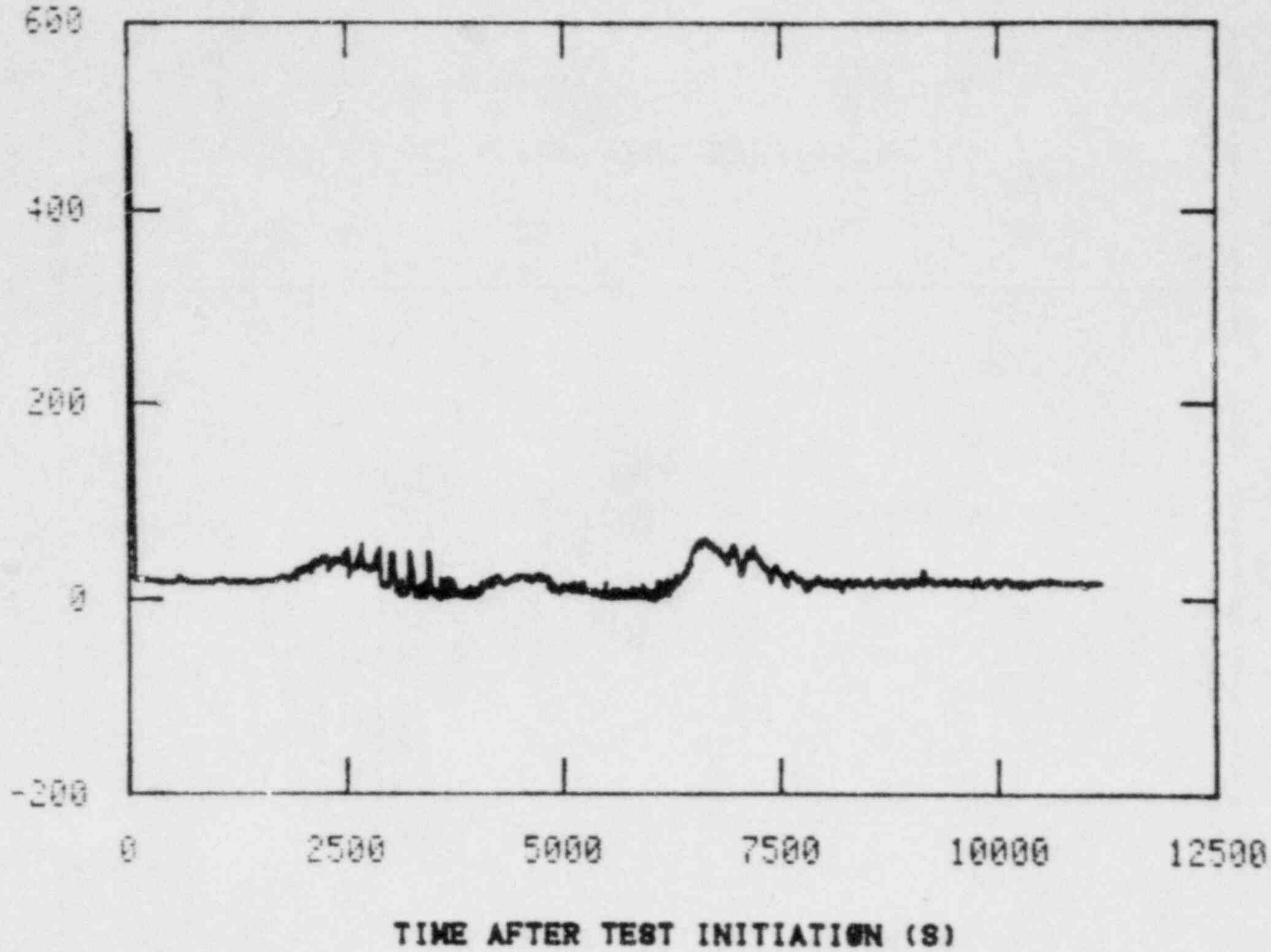


Figure 12. Flow rate in intact loop hot leg.

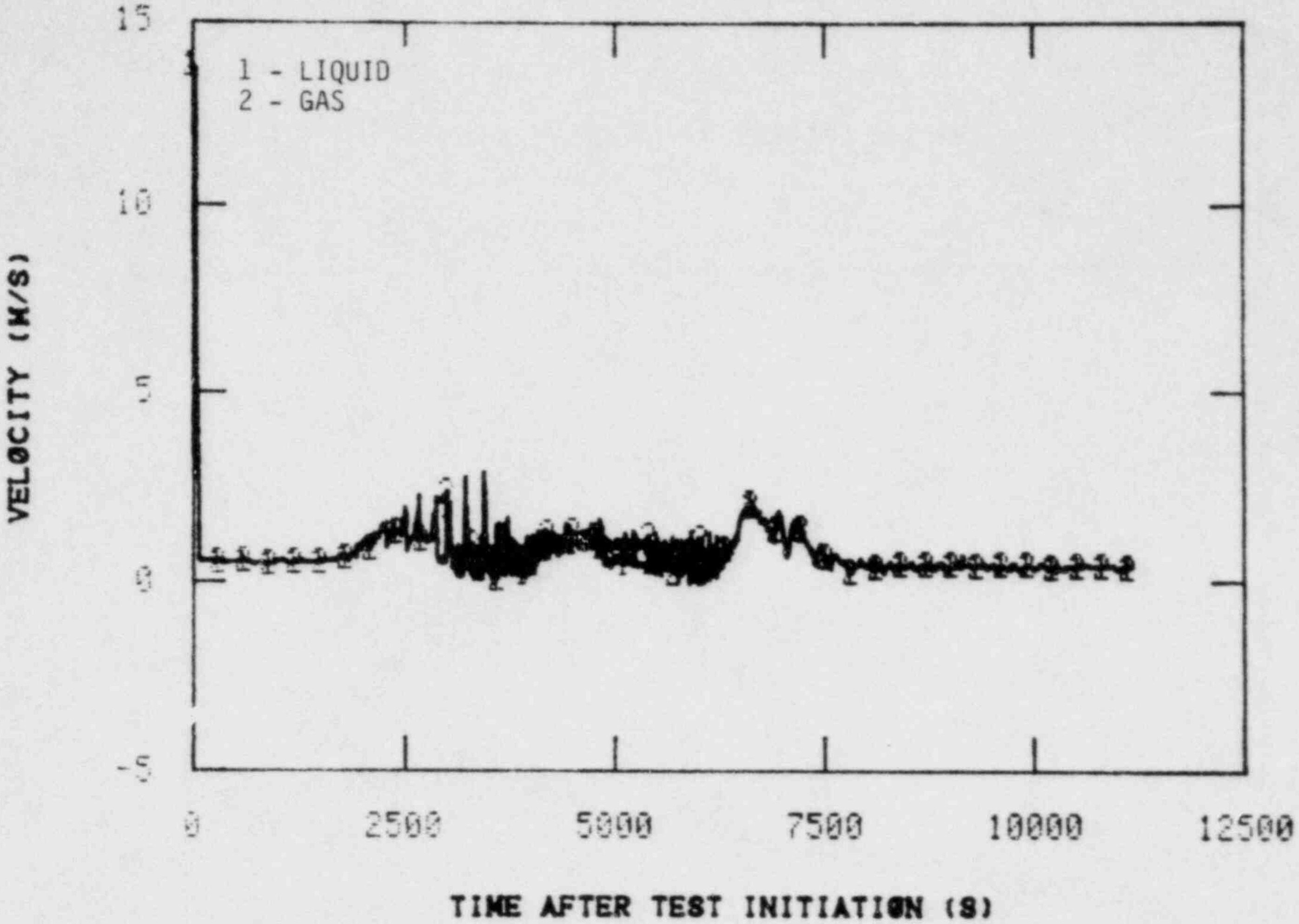


Figure 13. Phasic velocities in intact loop hot leg at Measurement Station PC-2.

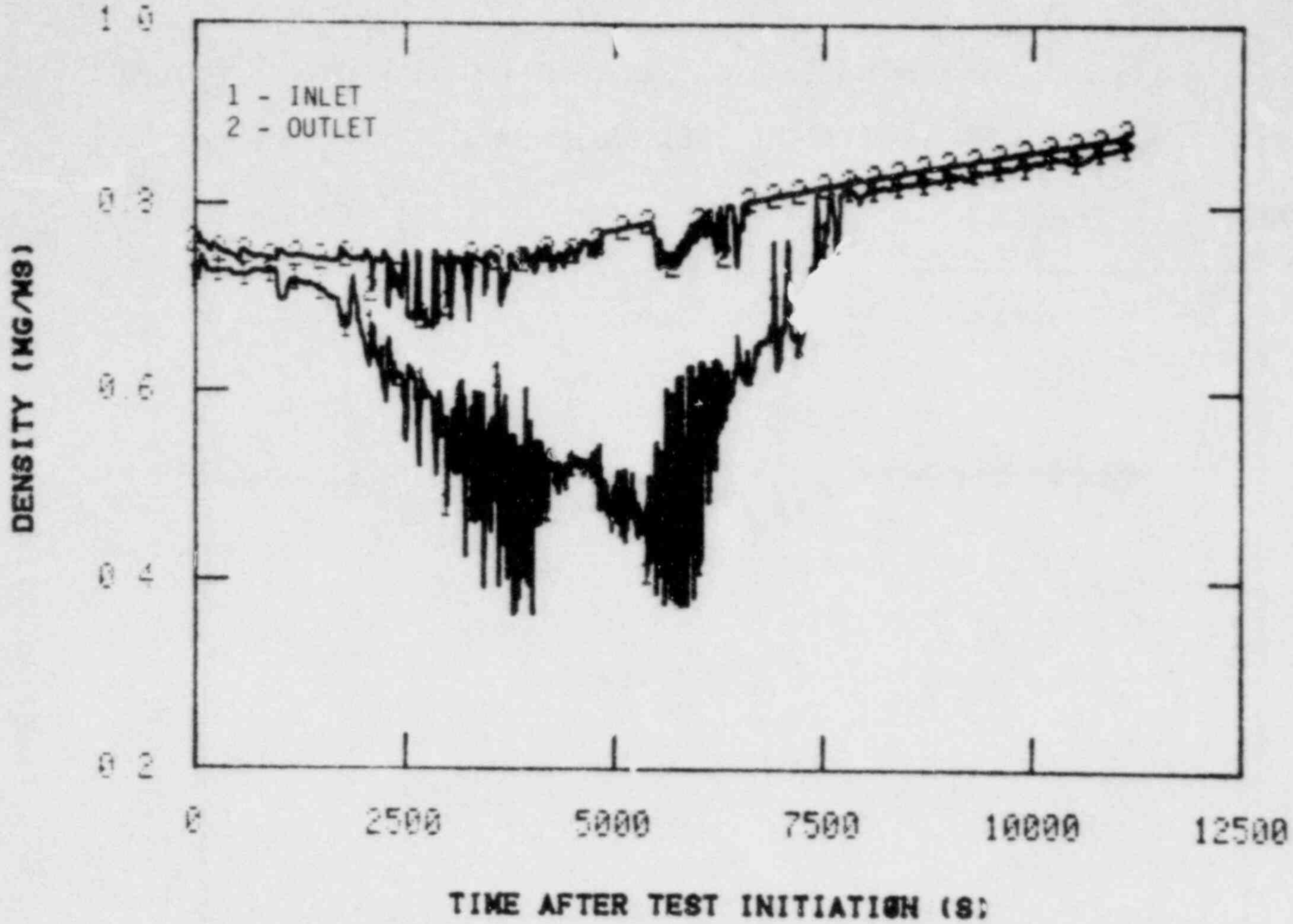


Figure 14. Densities in steam generator inlet and outlet.

3.4 Steam Generator Feed and Bleed Operation

To proceed more rapidly towards a cold shutdown condition, steam will be removed from the steam generator secondary at a rate which will produce a prescribed cooldown rate in the steam generator secondary. If heat transfer between the primary and secondary sides of the steam generator is sufficient, the primary coolant system will cool down more rapidly. As can be seen on Figure 15, the primary cooldown rate (starting at 3600 s) is approximately 45 K/h, which is the same cooldown rate imposed in the steam generator secondary.

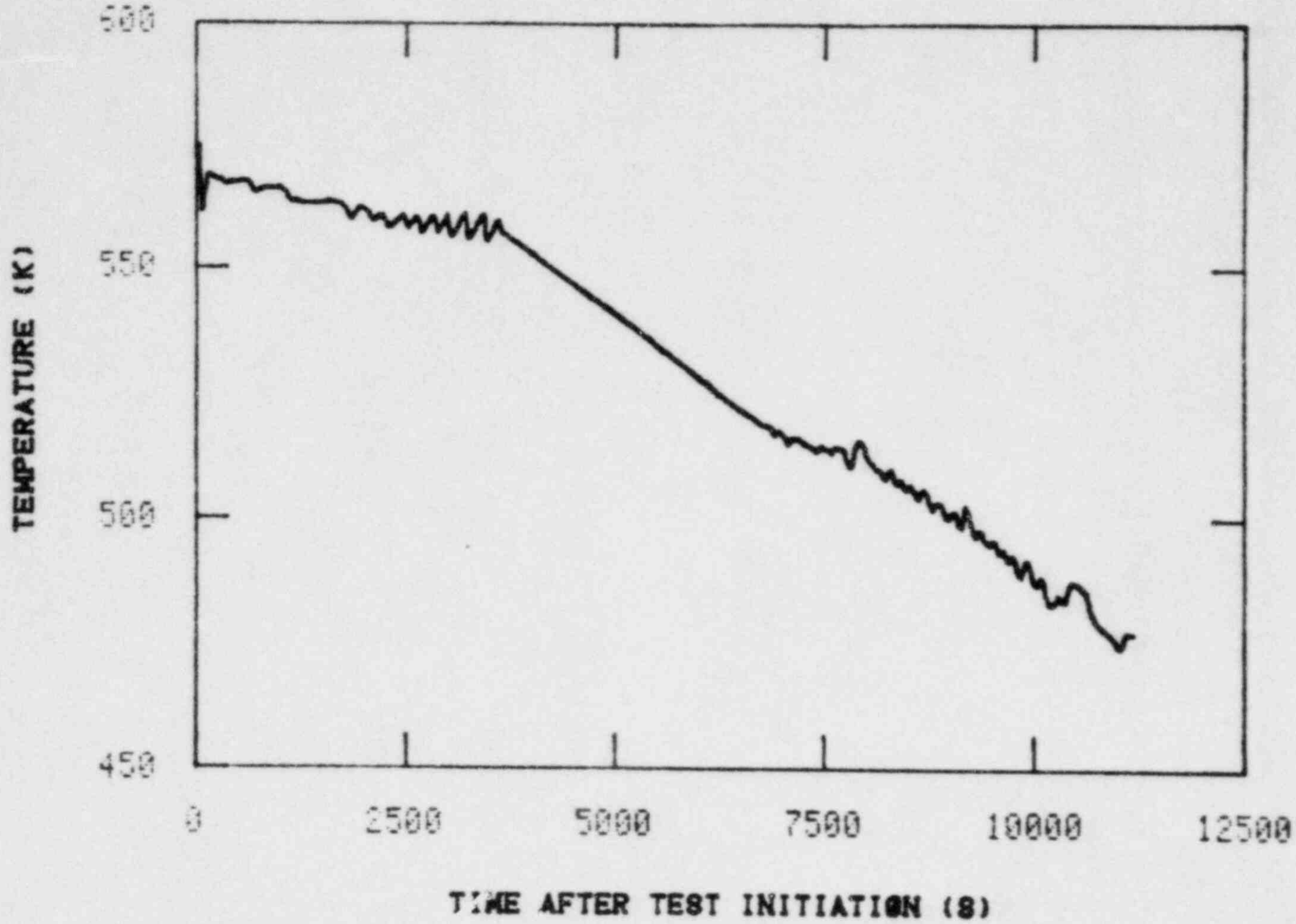


Figure 15. Liquid temperature in intact loop hot leg.

4. CONCLUSIONS

The primary system is predicted to be refilled to a solid condition at about 11 000 s after experiment initiation. Natural loop circulation is predicted to be sustained throughout the transient and be the major decay heat removal mechanism. The system could possibly enter the reflux flow mode during the 3000- to 4000-s and 5500- to 6000-s time periods.
LOCE L3-7 is expected to achieve its stated objectives based on the results of this prediction analysis.

5. REFERENCES

1. R. J. Beelman, LOFT Experiment Operating Specification Small Break Test Series L3 Nuclear Test L3-7, NE L3 Series, EOS L3-7, Rev. 0, May 19, 1980.

2. E. J. Kee et al., Best Estimate Prediction for LOFT Nuclear Experiment L3-2, EGG-LOFT-5089, February 1980.

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

APPENDIX A

DETAILED TEST PREDICTION DATA FOR LOFT LOCE L3-7

APPENDIX A

DETAILED TEST PREDICTION DATA FOR LOFT LOCE L3-7

Detailed test prediction data for Loss-of-Coolant Experiment (LOCE) L3-7 are provided in Figures A-1 through A-37 in this appendix. These figures are computer plots of the variables calculated for LOCE L3-7 using RELAP5. The RELAP4 calculations from the experiment prediction for the first 1800 s of LOCE L3-2 are included on each plot for comparison. The calculated variables and figure numbers are as follows:

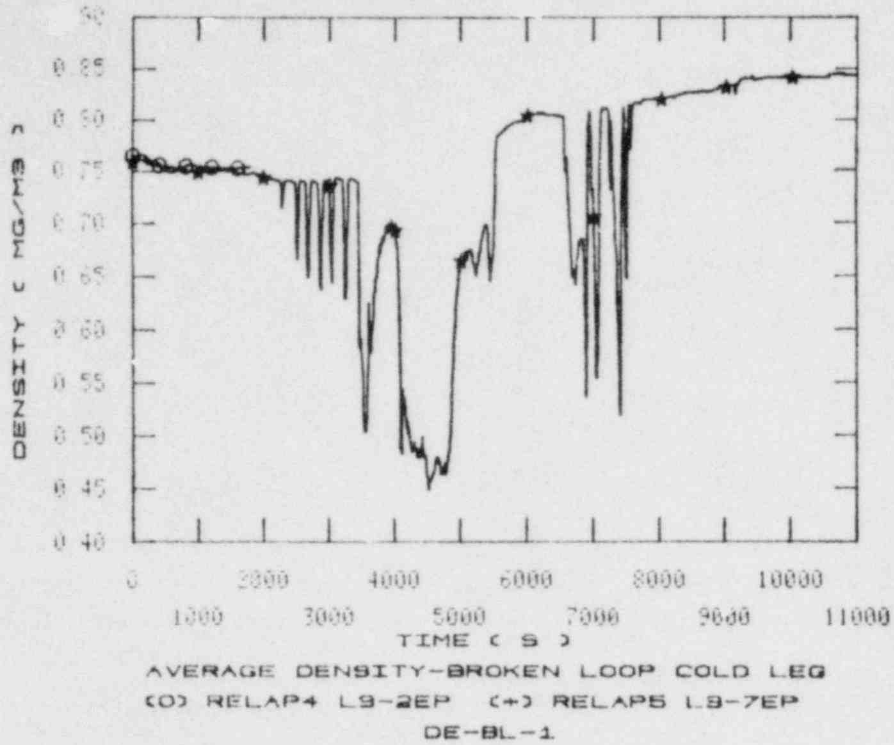


Figure A-1.

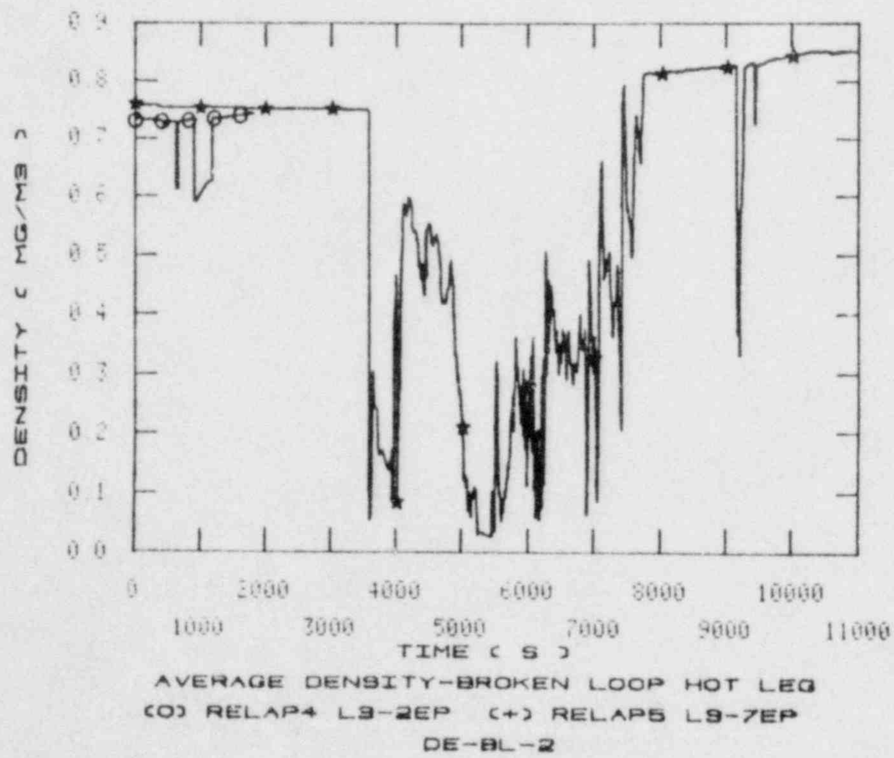
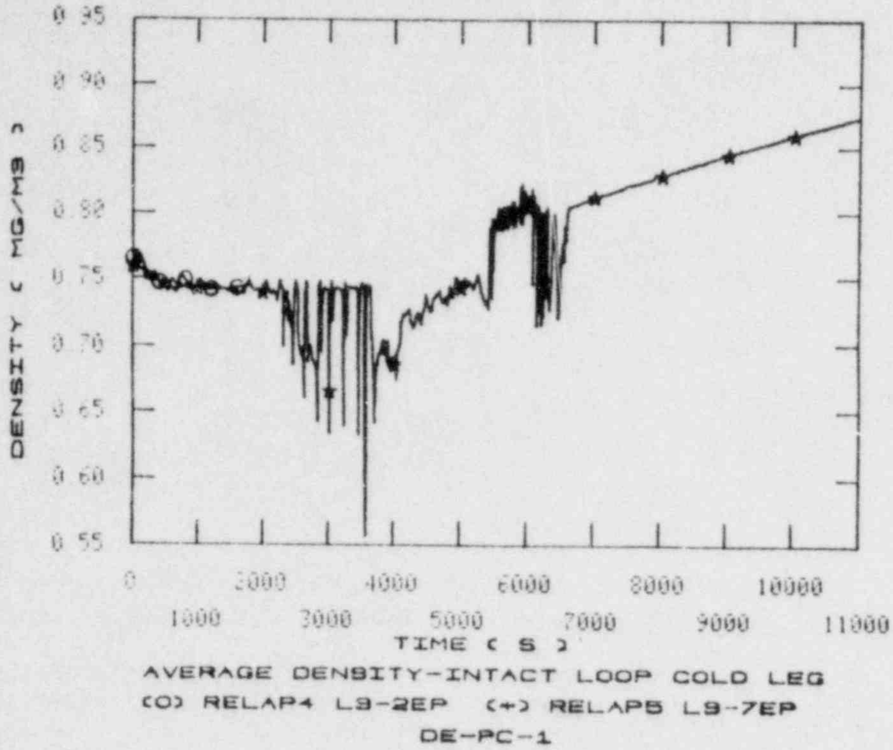


Figure A-2.



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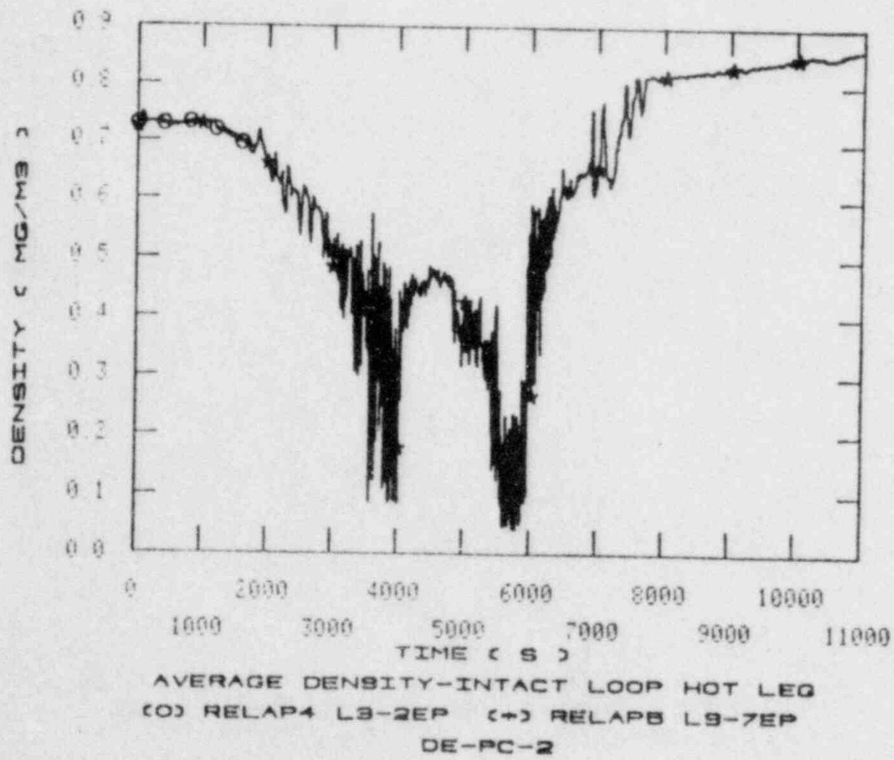

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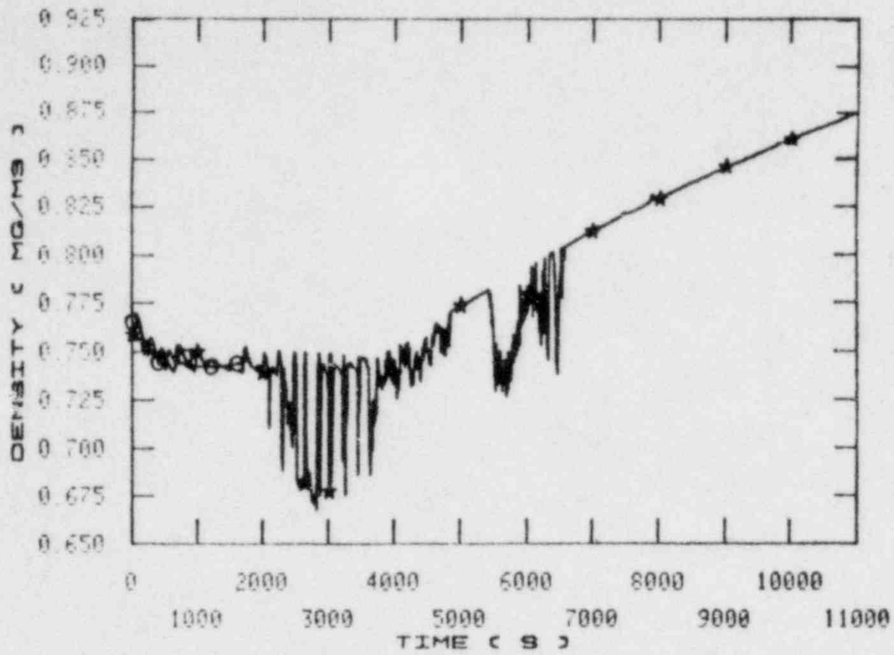
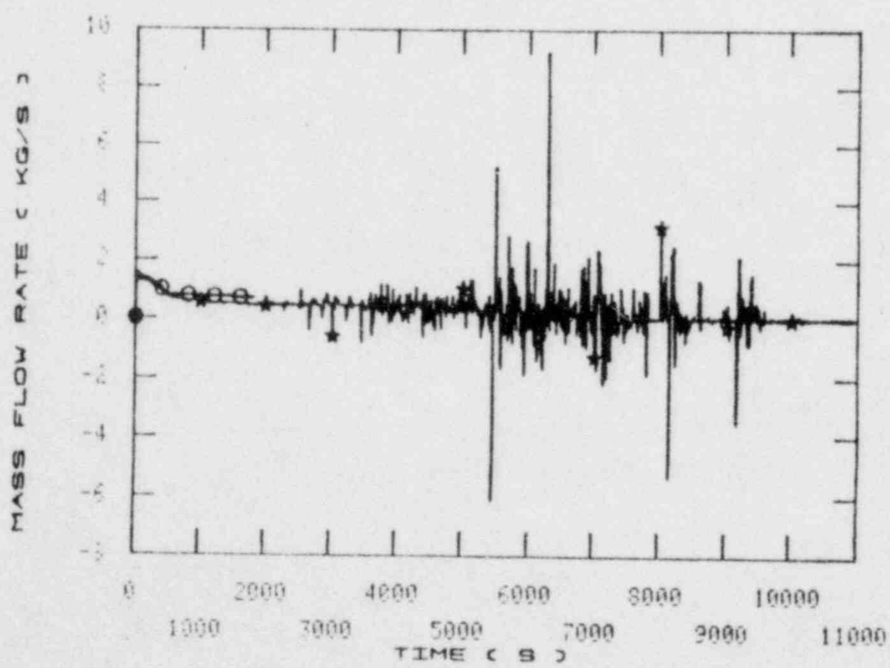


AVERAGE DENSITY - INTACT LOOP SG OUTLET
(O) RELAP4 L9-2EP (+) RELAP5 L9-7EP
DE-PC-9

Figure A-5.



MASS FLOW RATE AT STATION BL-1
(O) RELAP4 L9-2EP (+) RELAP5 L9-7EP
FR-BL-1

Figure A-6.

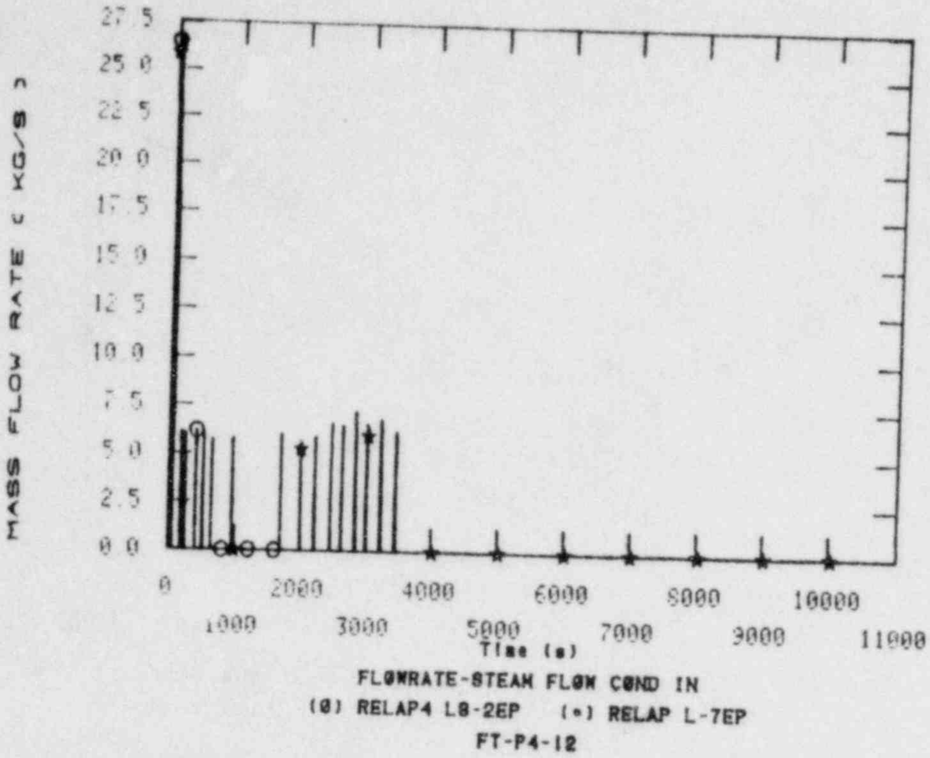


Figure A-7.

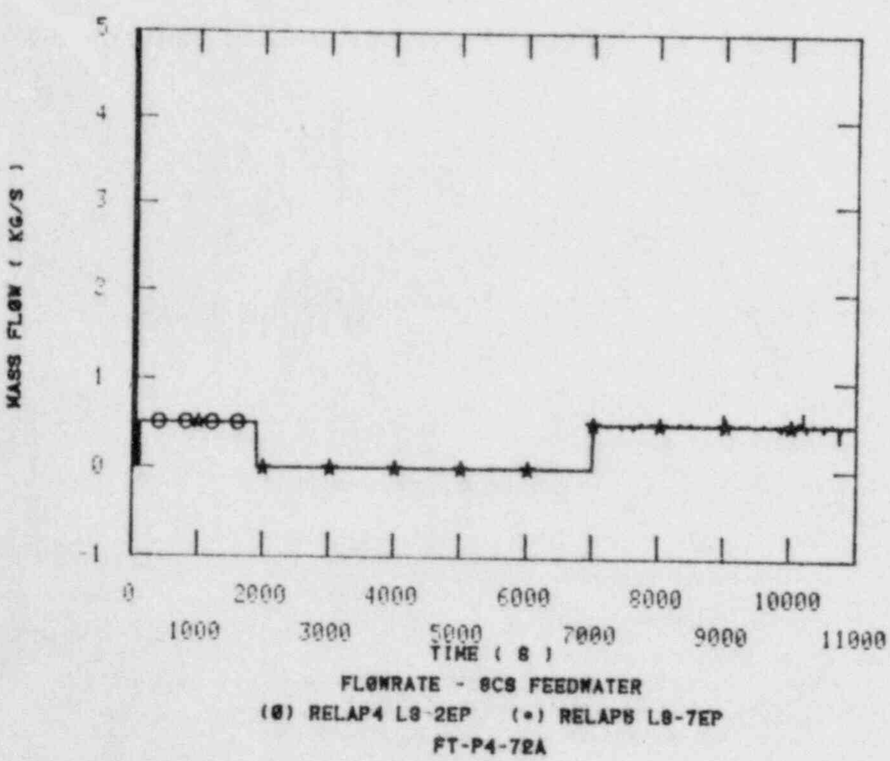


Figure A-8.

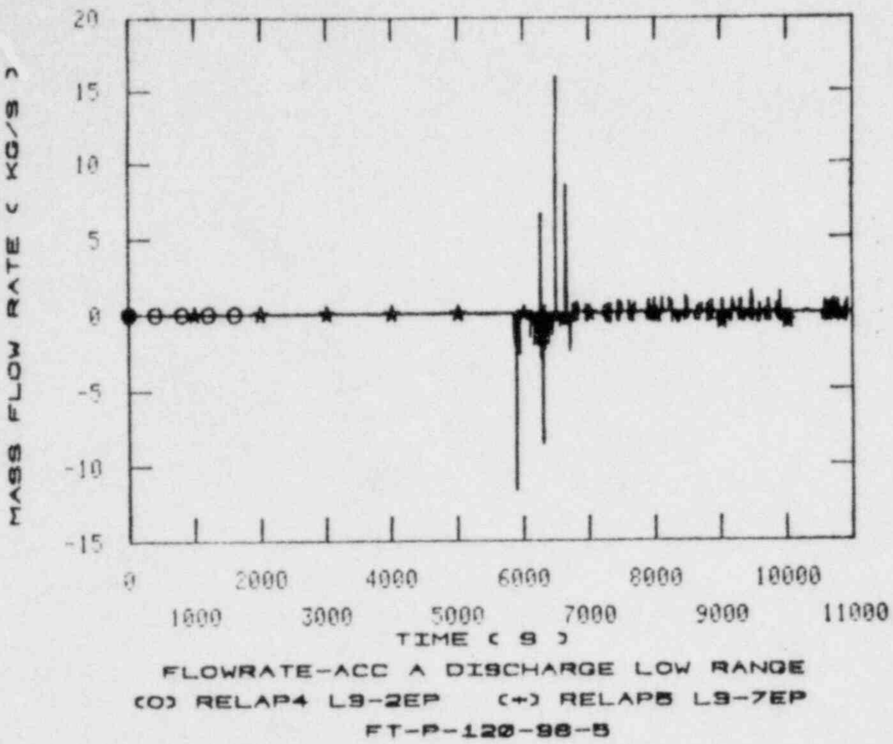


Figure A-9.

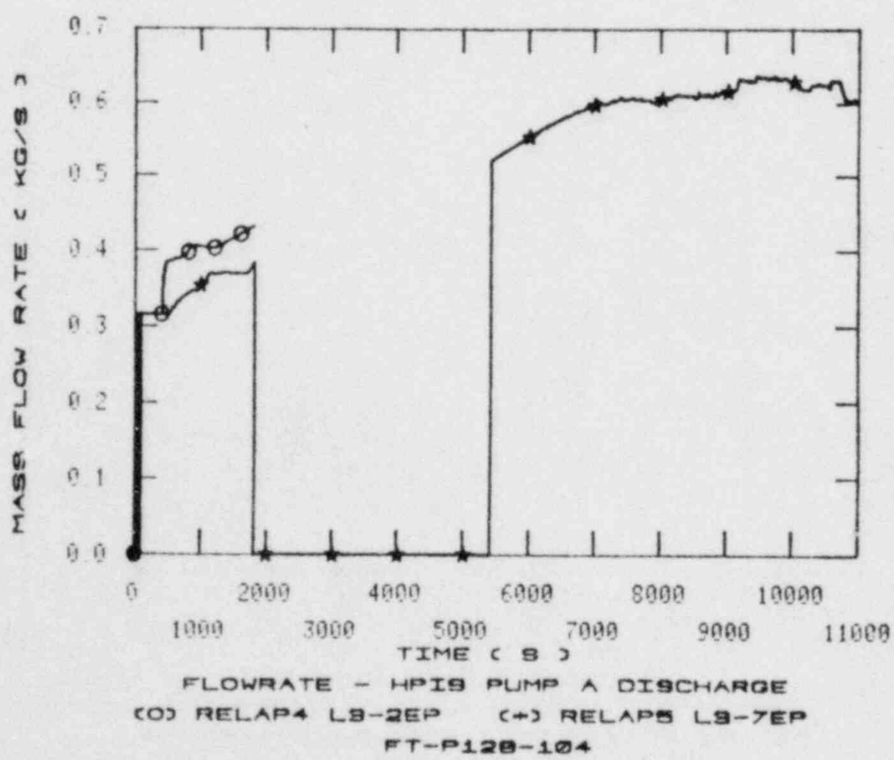


Figure A-10.

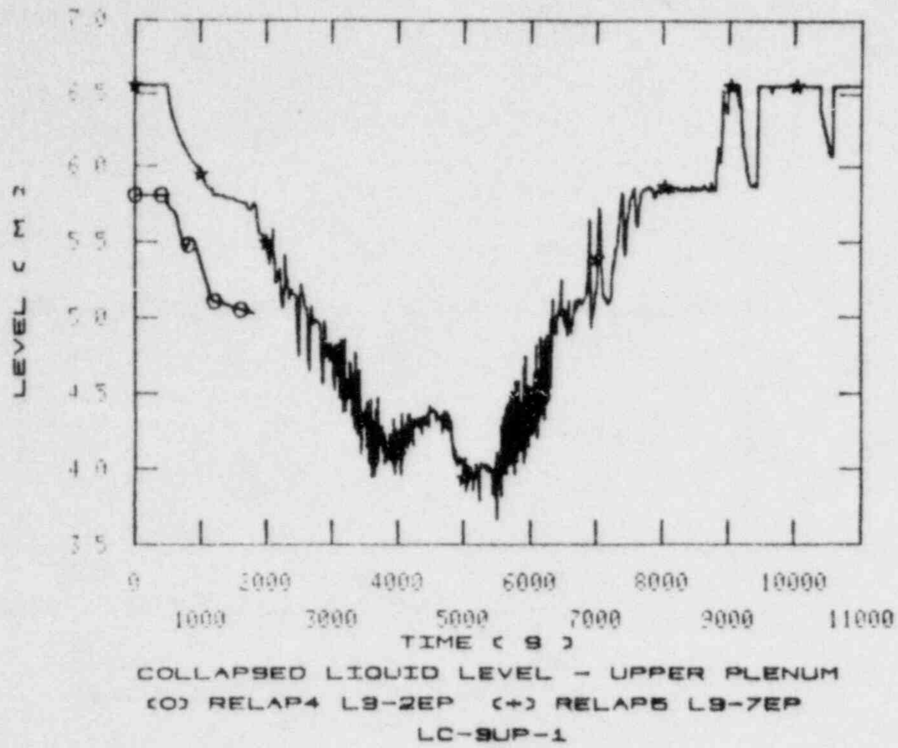
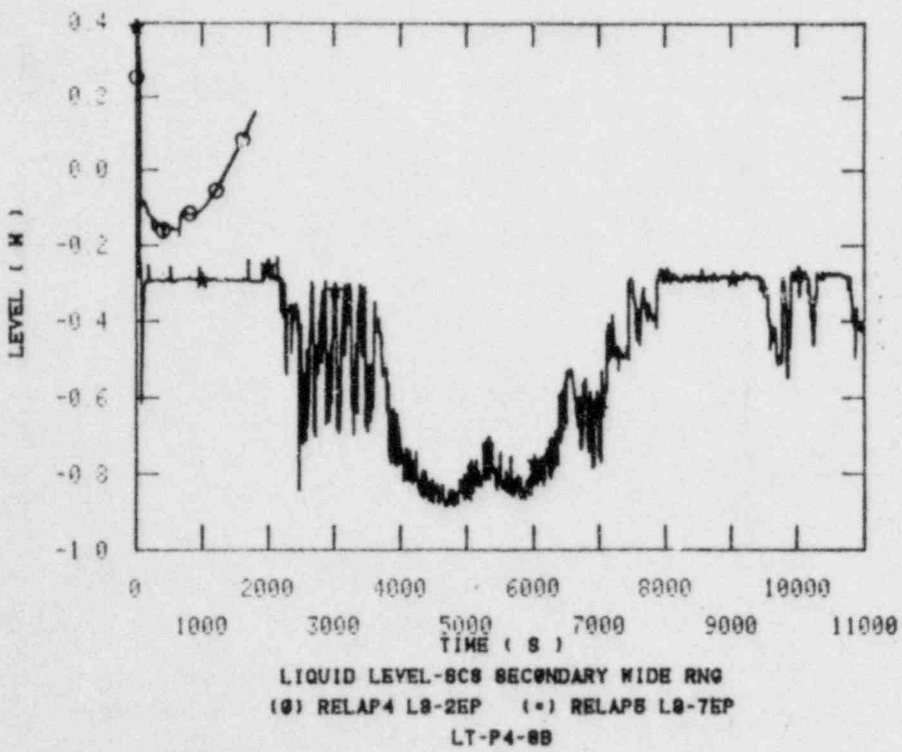


Figure A-11.



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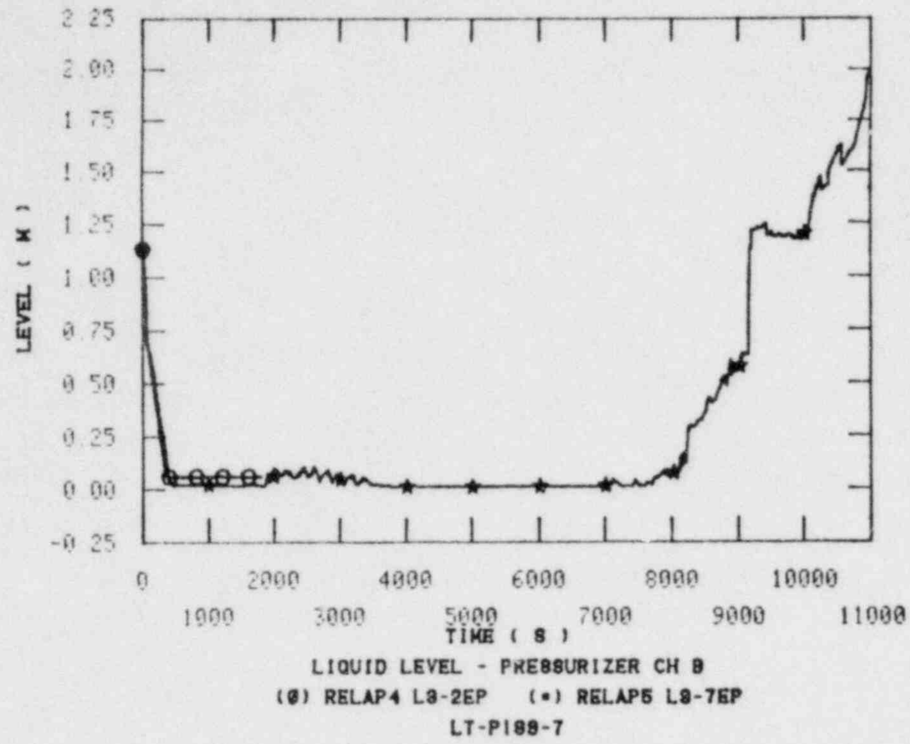


Figure A-13.

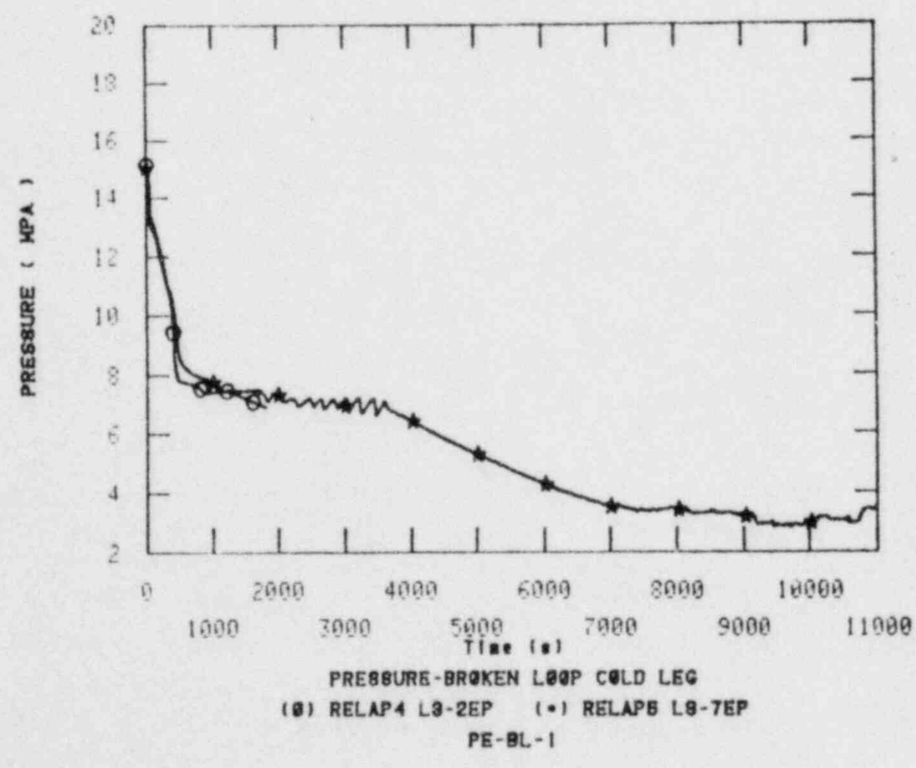
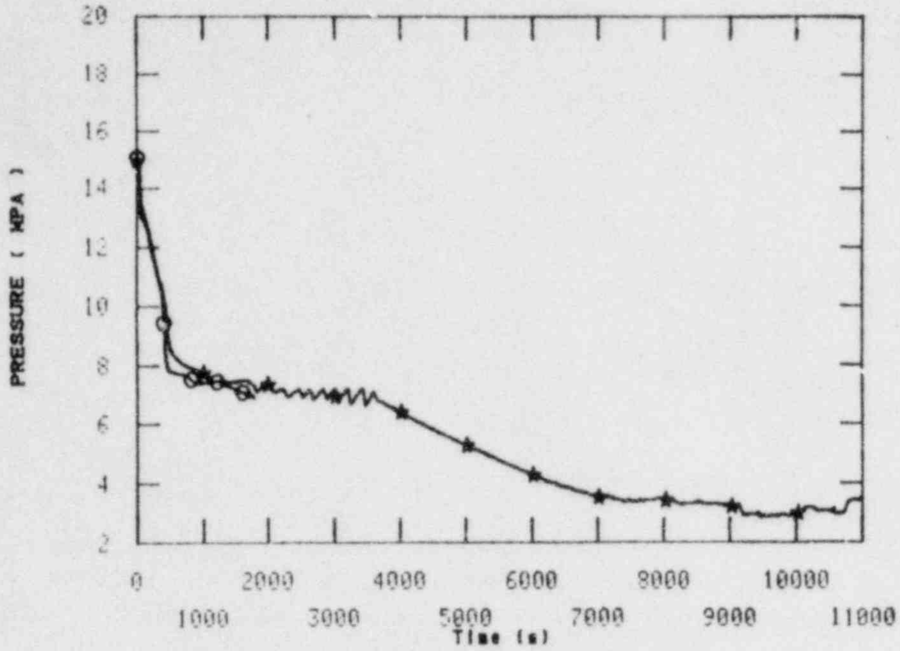
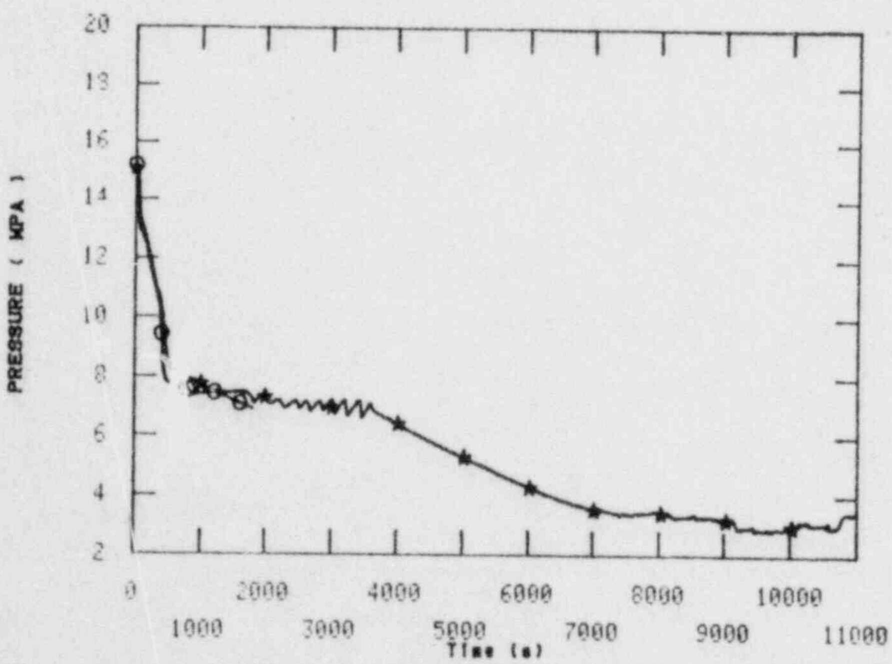


Figure A-14.



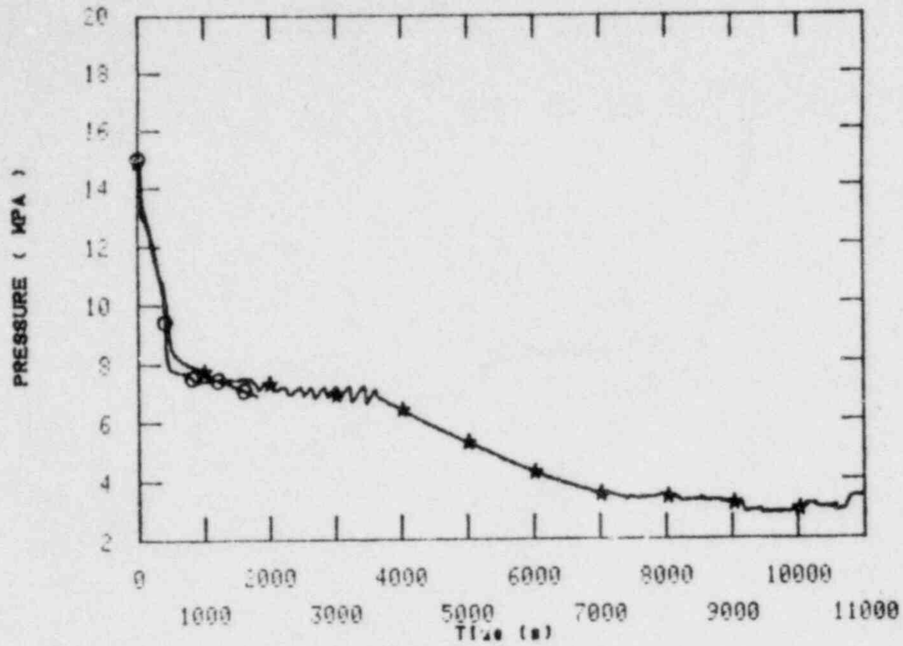
PRESSURE-BRØKEN LØØP HØT LEG
(Ø) RELAP4 L8-2EP (*) RELAP5 L8-7EP
PE-BL-2

Figure A-15.



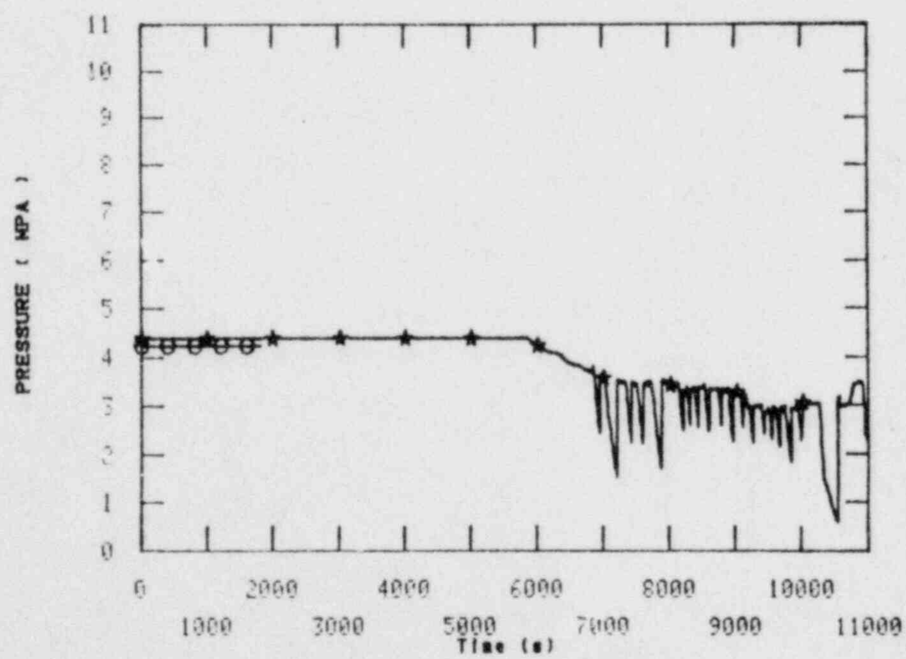
PRESSURE-INTACT LØØP CØLD LEG
(Ø) RELAP4 L8-2EP (*) RELAP5 L8-7EP
PE-PC-1

Figure A-16.



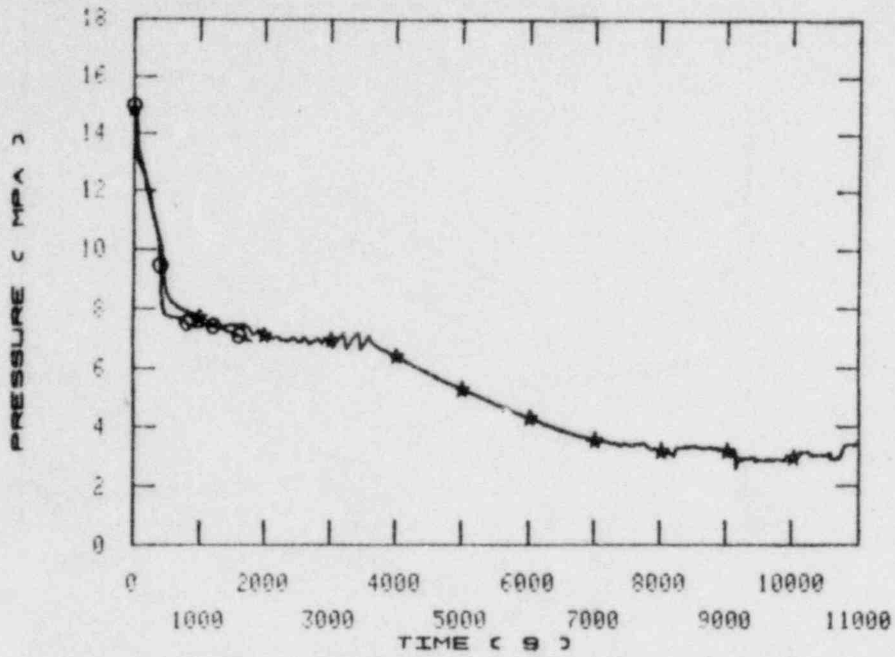
PRESSURE-INTACT LOOP HOT LEG
(0) RELAP4 L8-2EP (*) RELAP5 L8-7EP
PE-PC-2

Figure A-17.



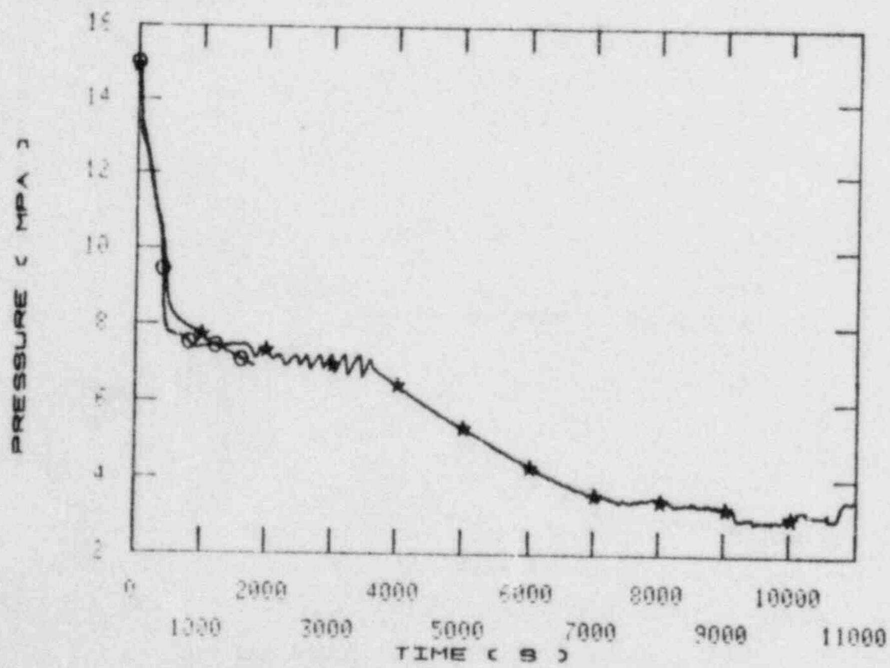
PRESSURE-ECCS ACCUMULATOR A
(0) RELAP4 L8-2EP (*) RELAP5 L8-7EP
PT-P120-43

Figure A-18.



PRESSURE - INTACT LOOP PRESSURIZER
(O) RELAP4 L3-2EP (+) RELAP5 L3-7EP
PE-PC-4

Figure A-19.



PRESSURE - FA1 AB UPPER END BOX HI R
(O) RELAP4 L3-2EP (+) RELAP5 L3-7EP
PE-1UP-1A

Figure A-20.

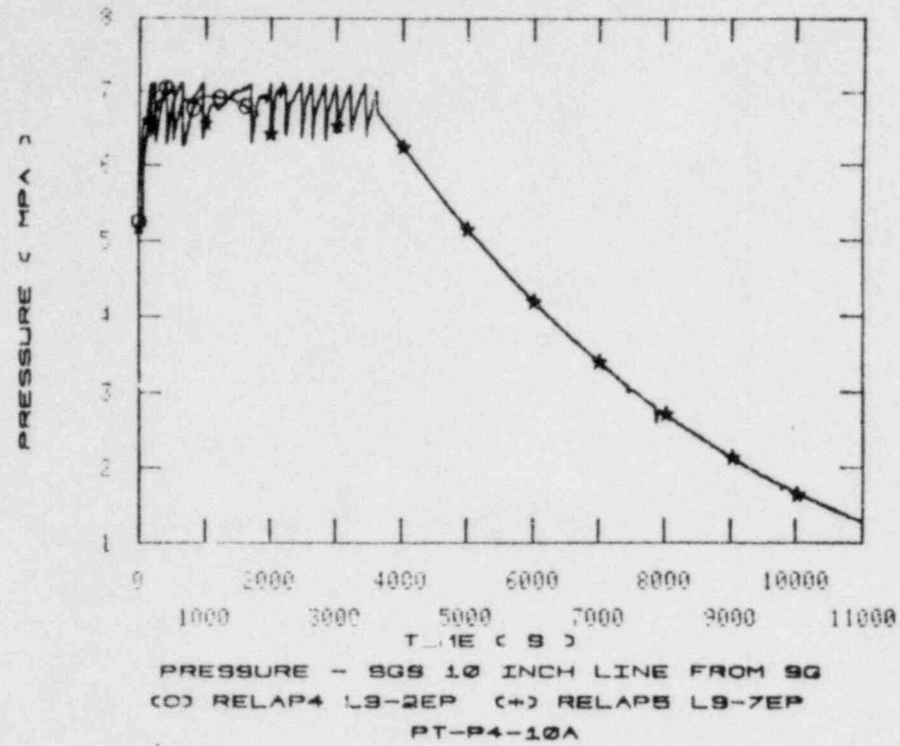
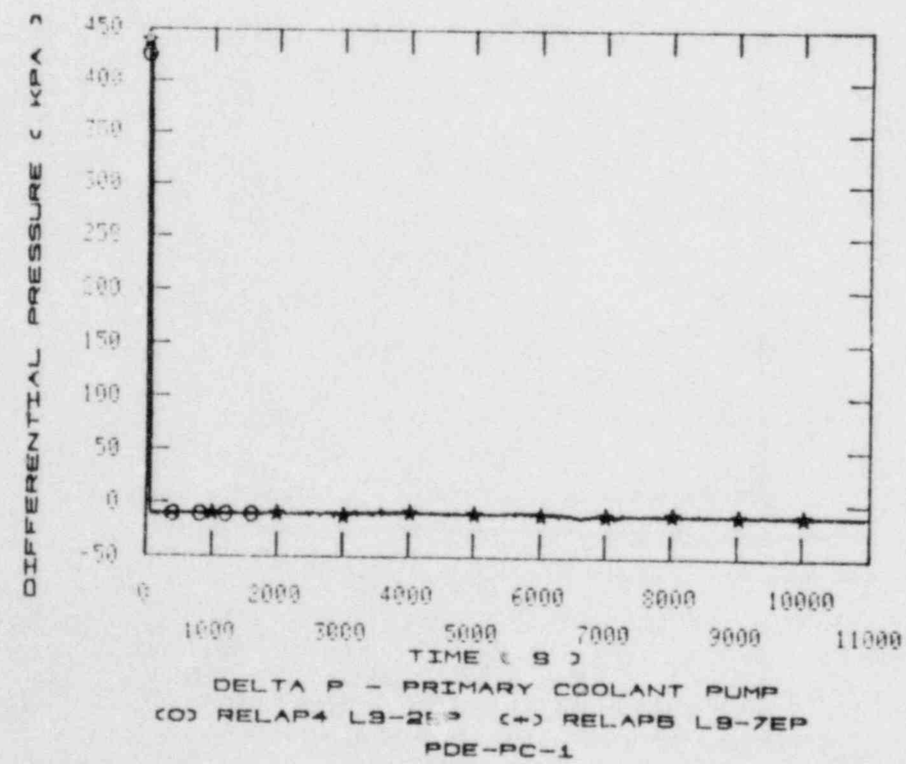


Figure A-21.



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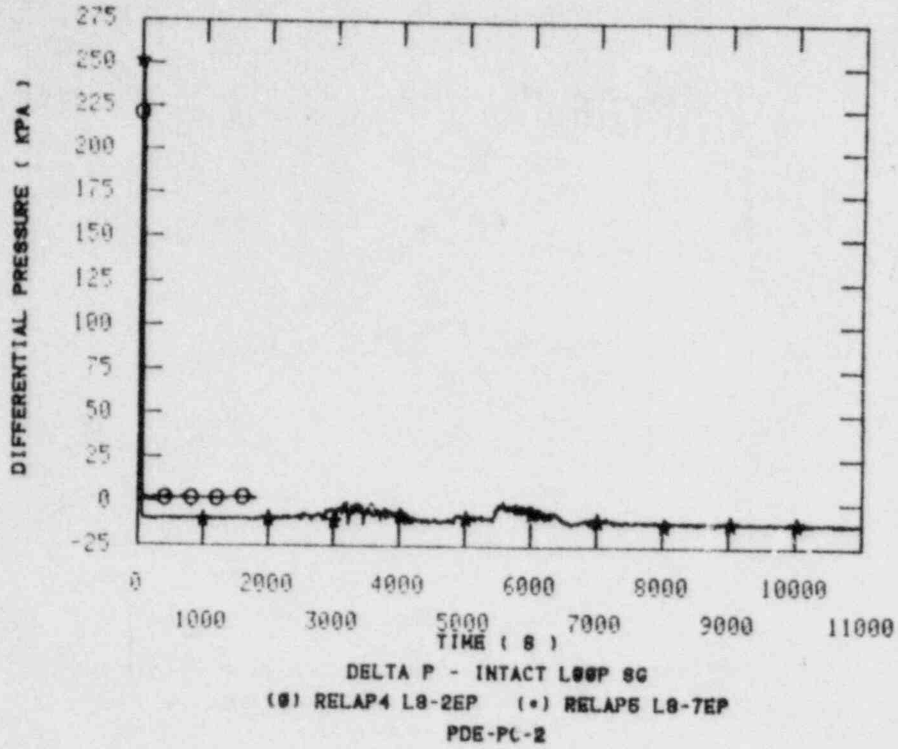

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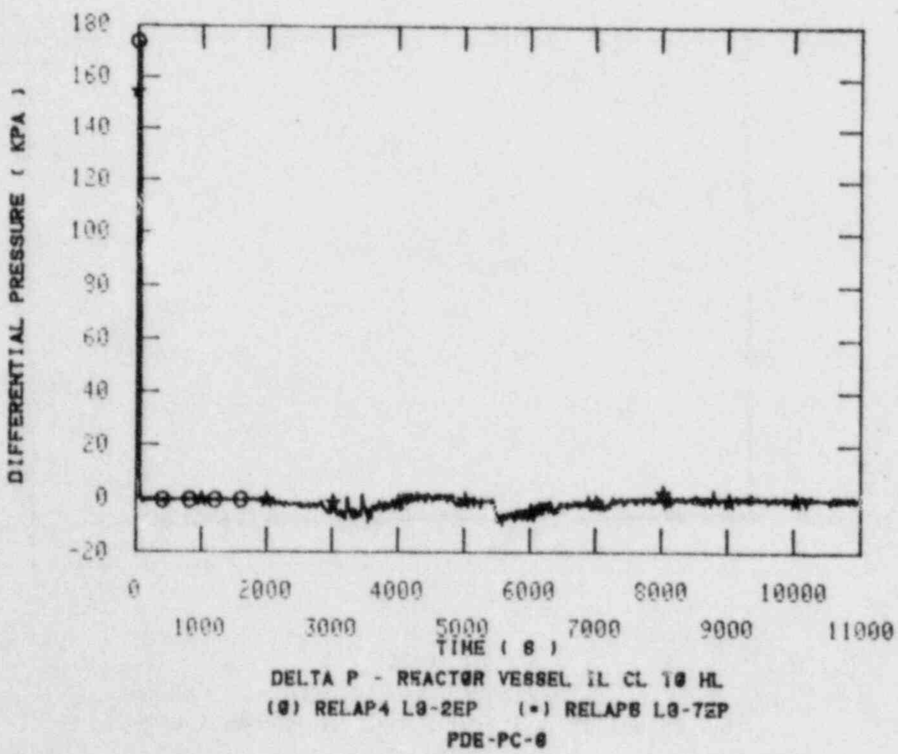

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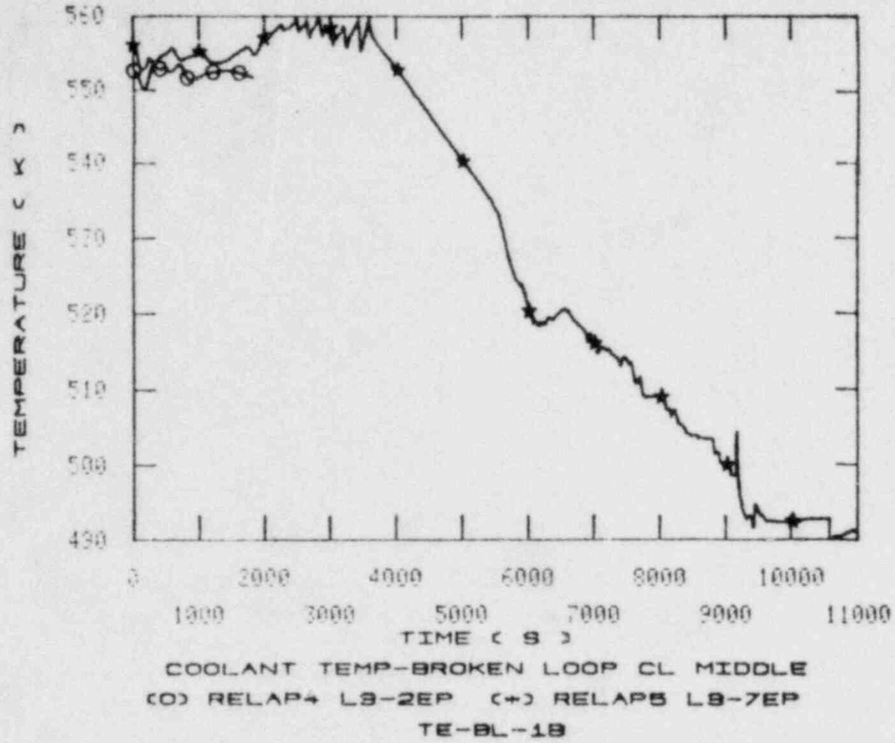

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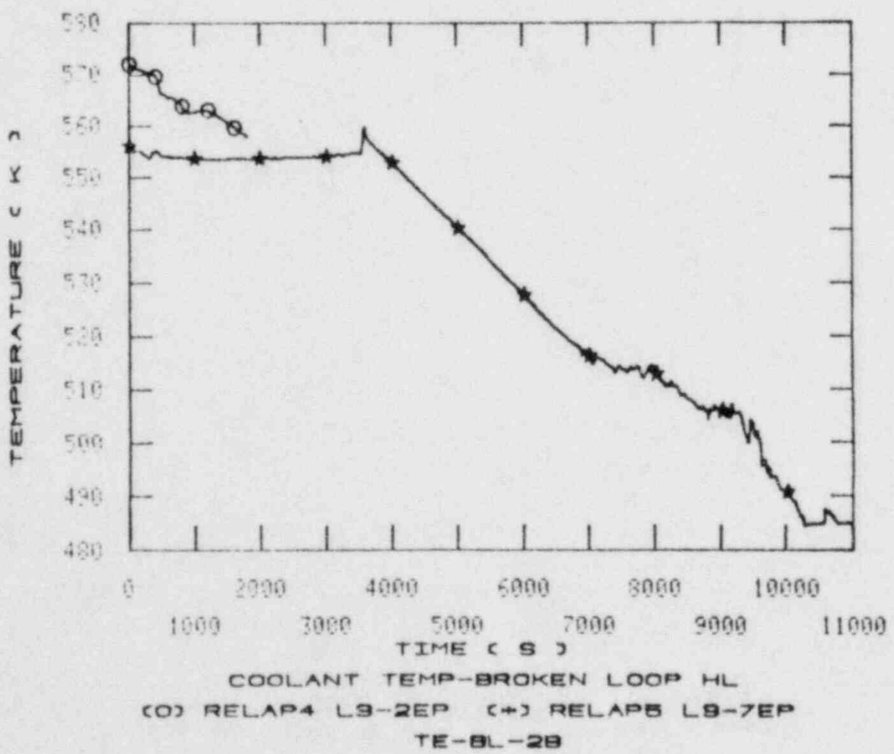

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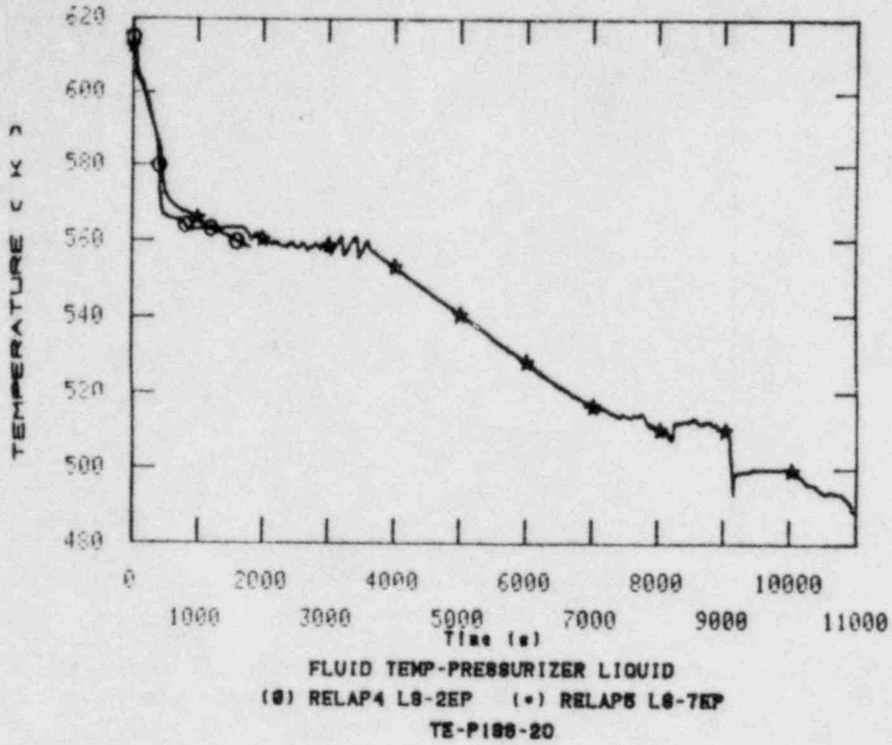

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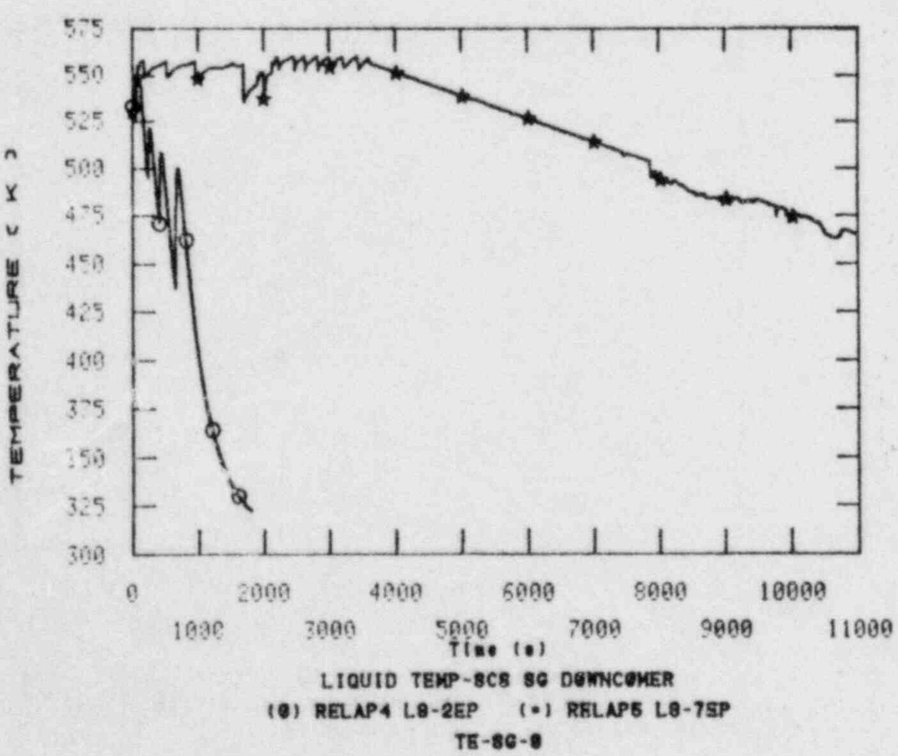

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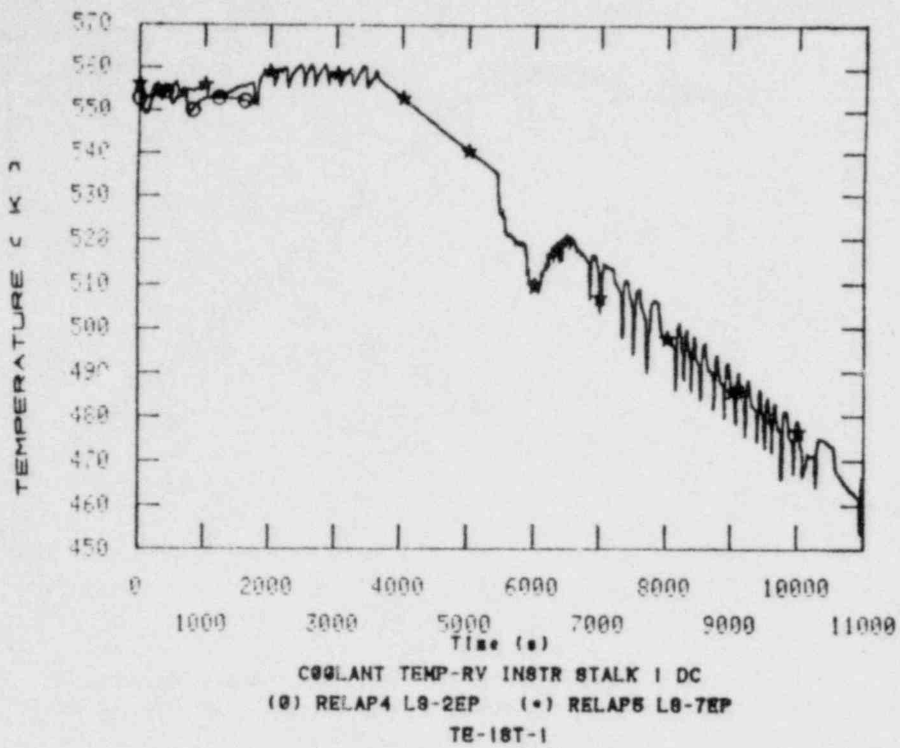

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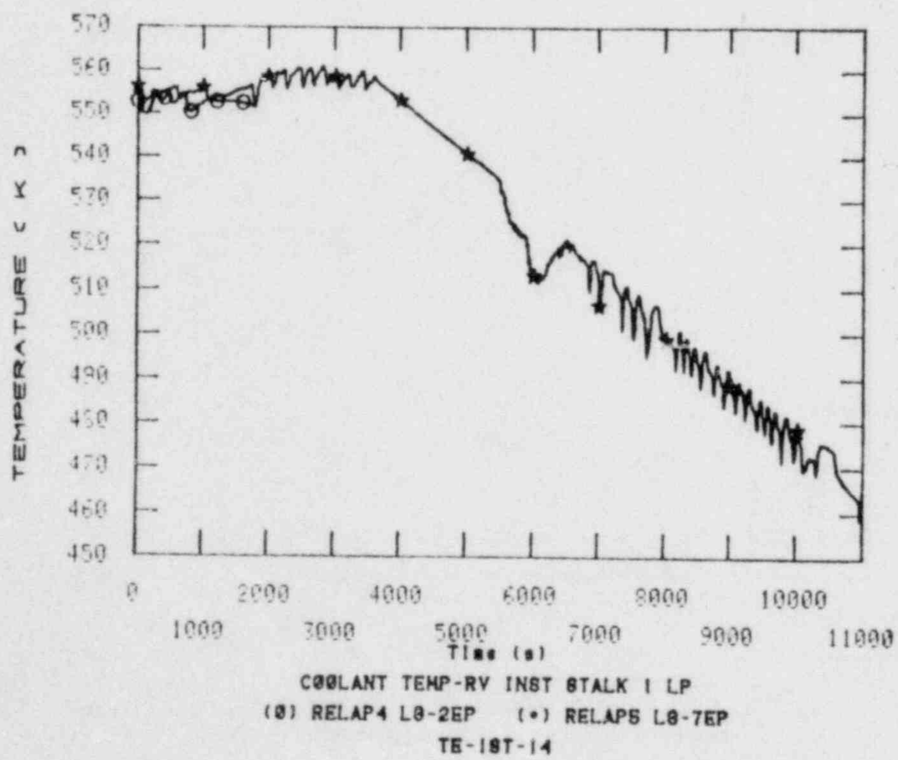

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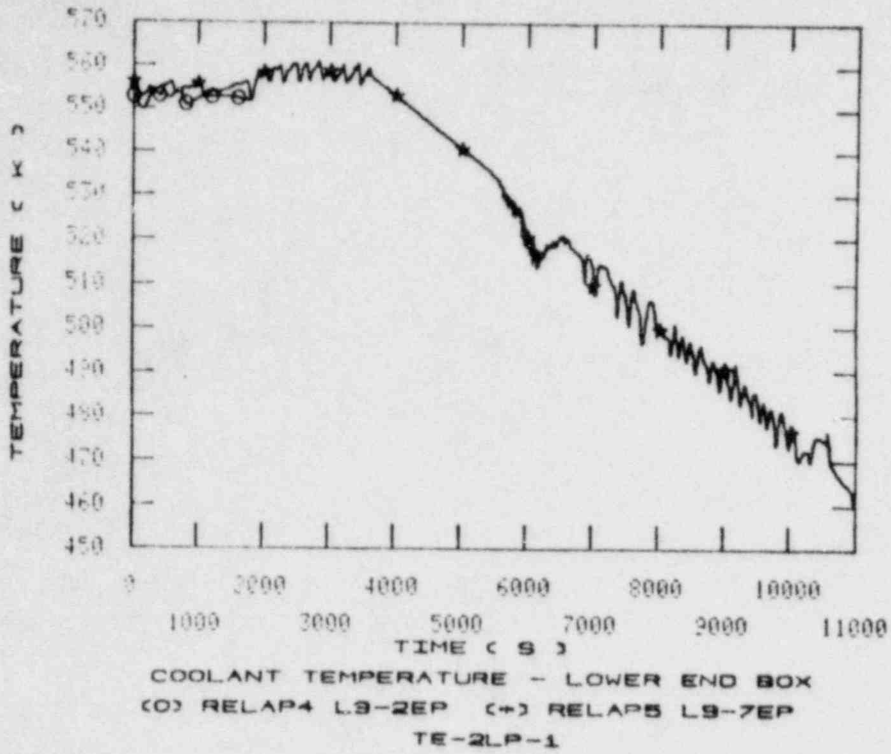

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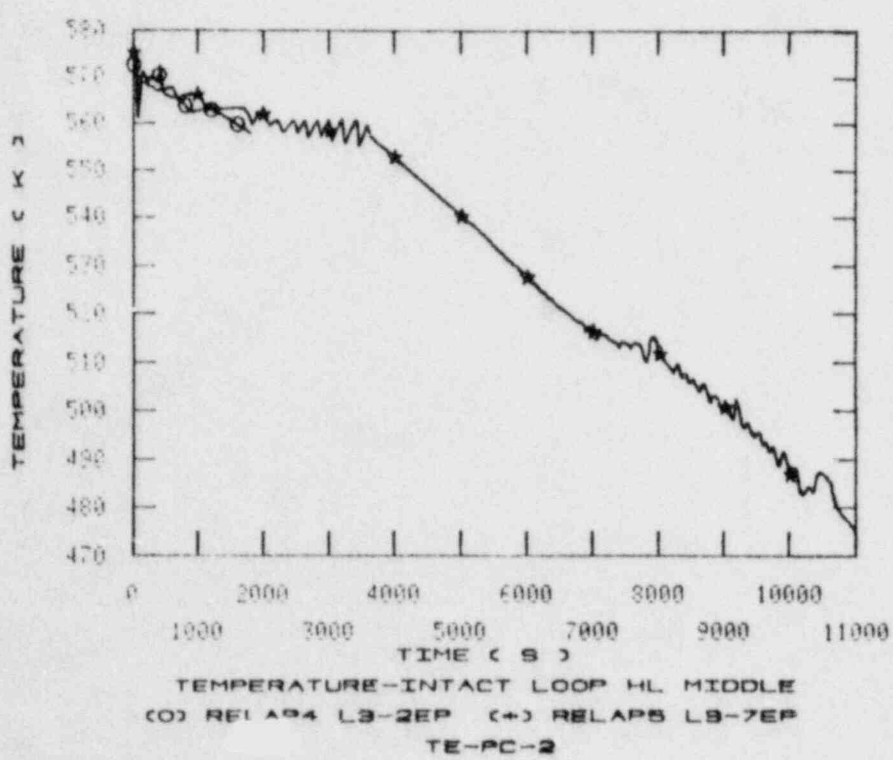

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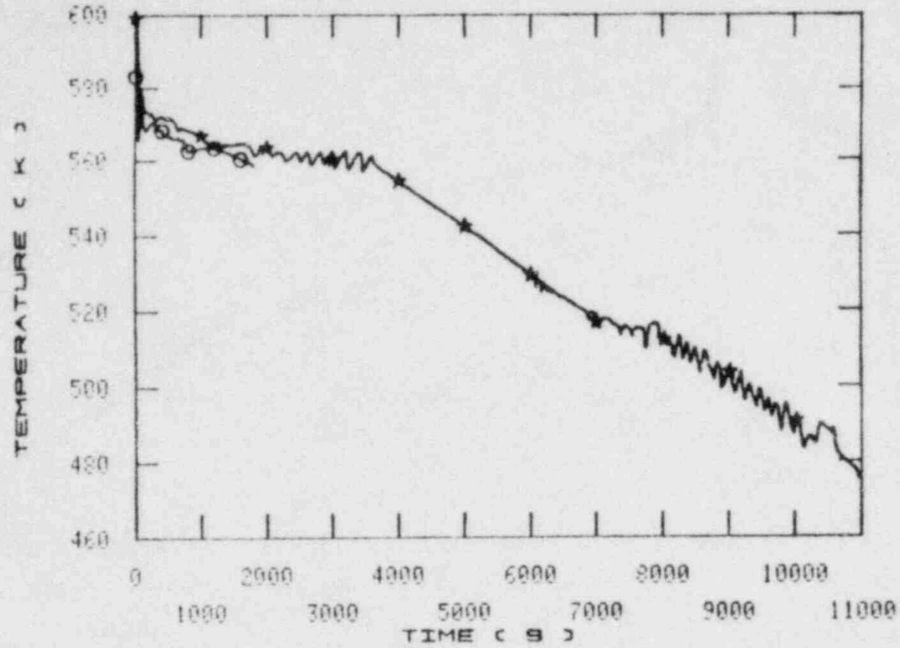
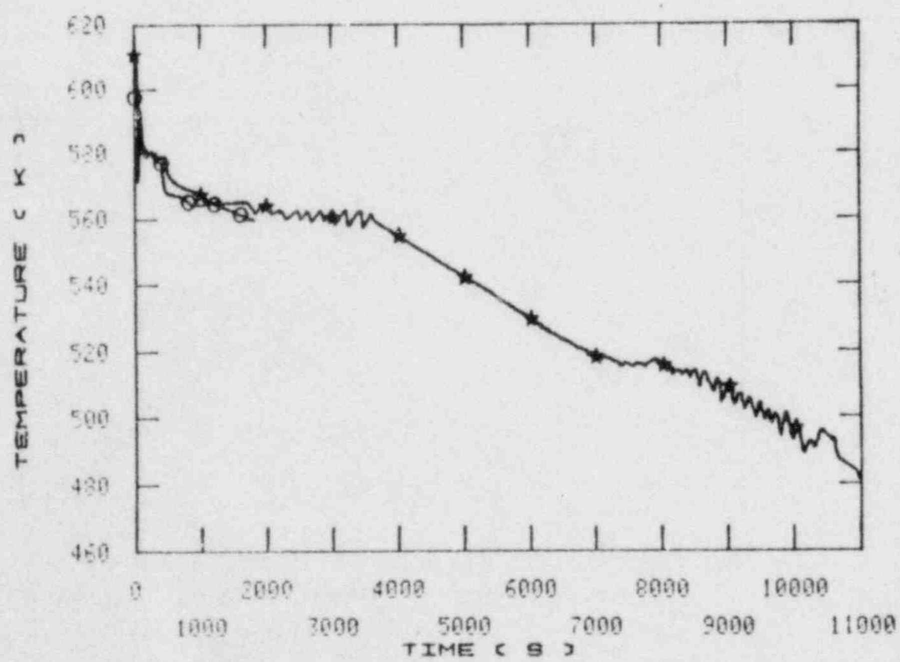


TEMP-CLADDING/FA 2 PIN G14 11 IN.
(O) RELAP4 L9-2EP (+) RELAP5 L9-7EP
TE-2G14-11

Figure A-33.



TEMP-CLADDING/FA2 PIN G14 90 IN.
(O) RELAP4 L9-2EP (+) RELAP5 L9-7EP
TE-2G14-90

Figure A-34.

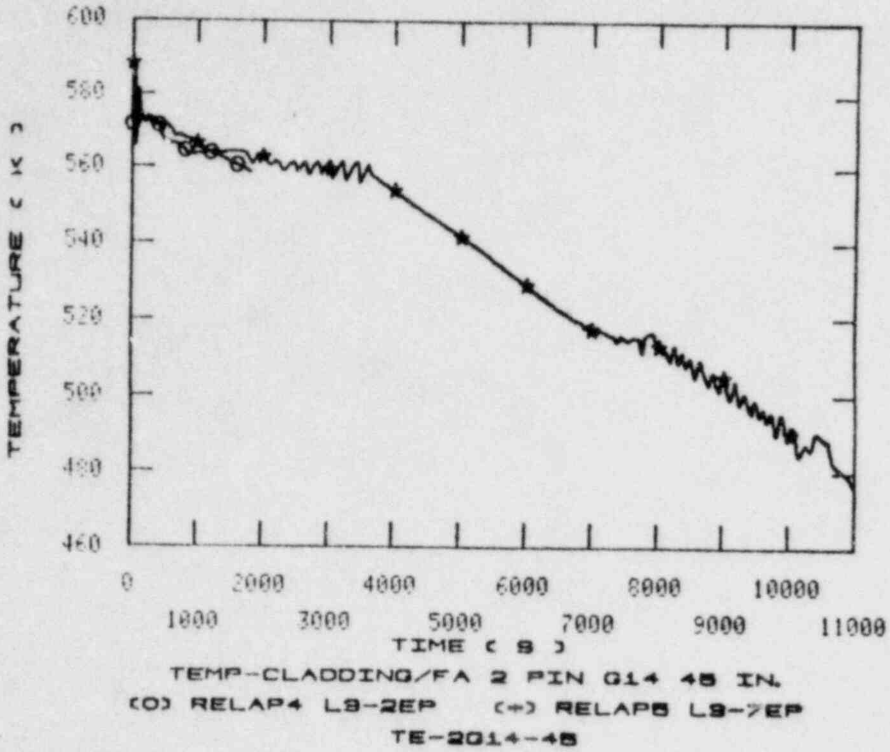


Figure A-35.

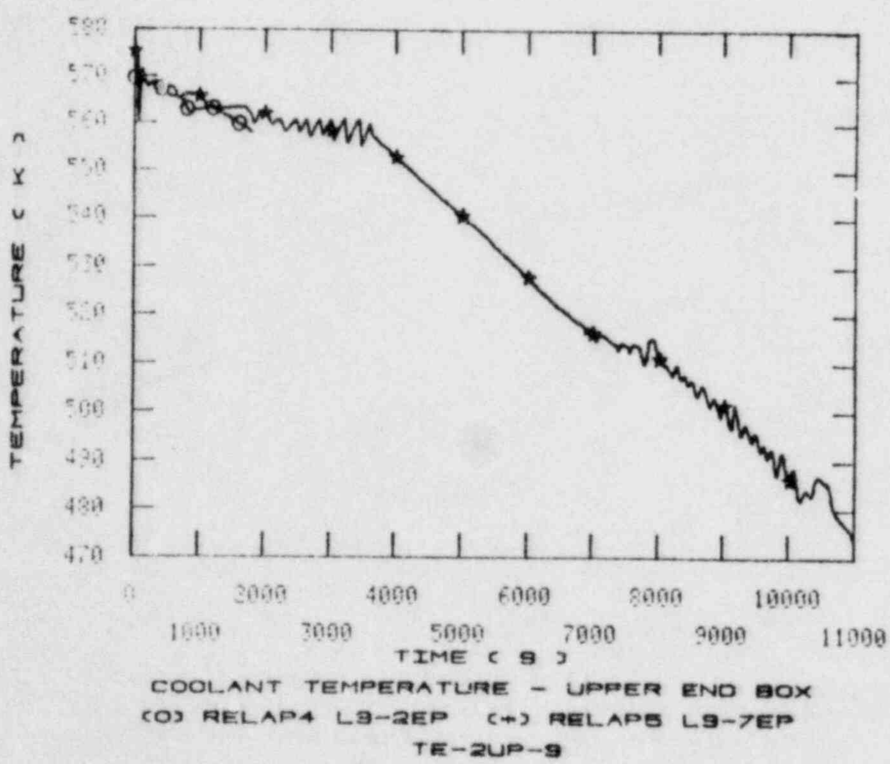


Figure A-36.

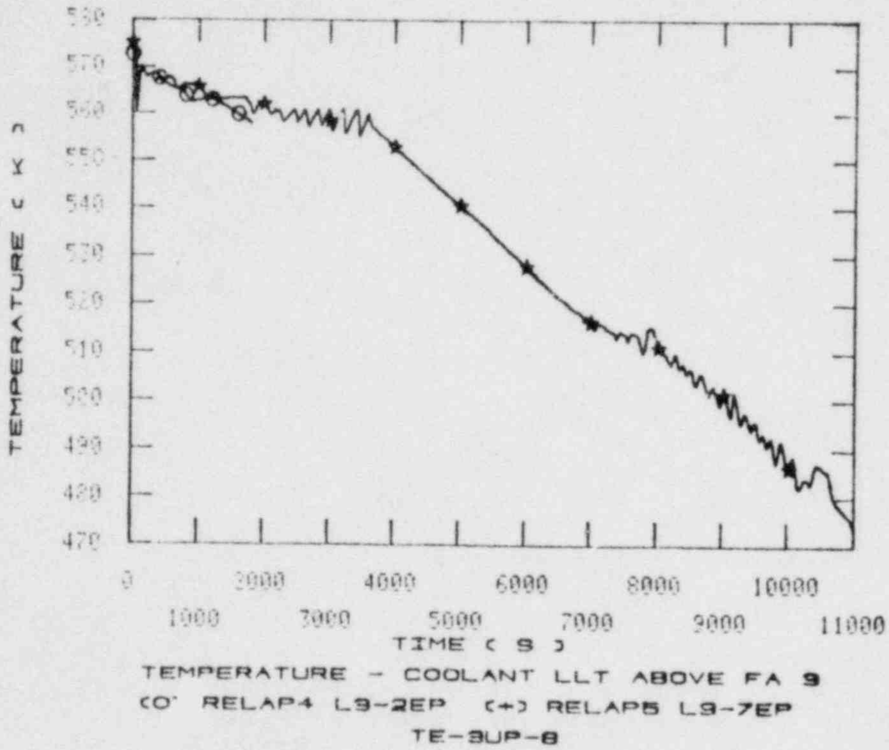

TEMPERATURE (K)
580
570
560
550
540
530
520
510
500
490
480
470
0
2000
4000
6000
8000
10000
1000
3000
5000
7000
9000
11000
TIME (S)
TEMPERATURE - COOLANT LLT ABOVE FA 9
(O) RELAP4 L9-2EP (+) RELAP5 L9-7EP
TE-9UP-8

Figure A-37.

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"0" source deck is stored under Configuration Control Number H01238IB.

2. The RELAP5/MOD"0" update input data deck is stored under Configuration Control Number H006685B.

3. The RELAP5 input deck is stored under Configuration Control Number H006885B.

APPENDIX D

RELAP5 INPUT DATA

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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.

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