EG&G Idaho, Inc.

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for H. P. Pearson, Supervisor
Information Processing

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February 1980

BEST ESTIMATE PREDICTION FOR LOFT NUCLEAR EXPERIMENT L3-2

Ernest J. Kee
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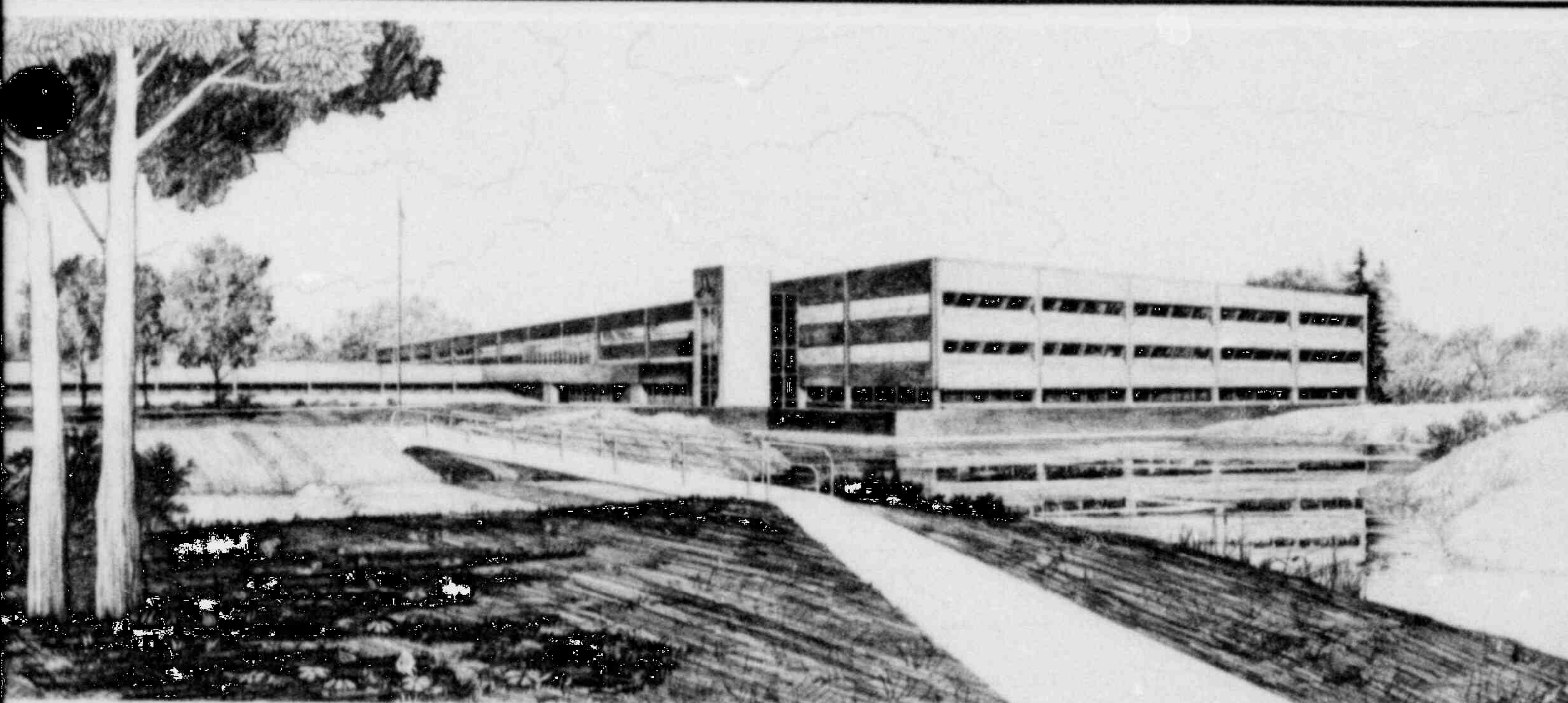
U.S. Department of Energy

Idaho Operations Office • Idaho National Engineering Laboratory



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INTERIM REPORT

BEST ESTIMATE PREDICTION FOR LOFT NUCLEAR

EXPERIMENT L3-2

Approved:

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THE LOFT SUBCOMMITTEE OF THE EG&G PRETEST PREDICTION CONSISTENCY COMMITTEE HAS REVIEWED AND CONCURS THAT THE RESULTS OF THE RELAP4 L3-2 EP ANALYSIS ARE REASONABLE AND CONSISTENT.

S. R. BEHLING, CODE DEVELOPMENT

T. R. CHARLTON, CODE ASSESSMENT

J. L. Steiner

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D. J. VARACALLE, PBF

THE LOFT SUBCOMMITTEE OF THE EG&G PRETEST PREDICTION CONSISTENCY COMMITTEE HAS REVIEWED AND CONCURS WITH THE INPUT MODEL AND CODE VERSION WITH UPDATES USED FOR THIS RELAP5 L3-2 EP ANALYSIS.

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ABSTRACT

Comprehensive analyses using both the RELAP4 and the RELAP5 computer codes were performed to predict the LOFT transient thermal-hydraulic response for nuclear Loss-of-Coolant Experiment L3-2 to be performed in the Loss-of-Fluid Test (LOFT) facility. The LOFT experiment will simulate a small break in one of the cold legs of a large four-loop pressurized water reactor and will be conducted with the LOFT reactor operating at 50 MW. The break in LOCE L3-2 is sized to cause the break flow to be approximately equal to the high-pressure injection system flow at an intermediate pressure of approximately 7.6 MPa.

Based on the RELAP5 analysis it is expected that operation of the protective and emergency core cooling systems will result in relatively stable plant conditions at the end of 1 hour with the reactor core completely covered and being cooled by natural circulation. The steam generator cooldown procedure will be effective after that time to bring the reactor to a long-term cooling condition in a controlled and safe manner.

The RELAP4 results predict that the plant will repressurize after the steam generator tubes void at approximately 2200 s and that natural circulation in the intact loop will terminate. The core will remain completly covered at the end of 1 hour when cooldown is initiated. The effectiveness of the steam generator cooldown procedure has not been verified for the RELAP4 predicted conditions.

NRC FIN No. A6048 - LOFT Experimental Program.

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FIGURES

TABLES

ACRONYMS

ECC	Emergency Core Coolant
EOS	Experiment Operating Specification
EP	Experiment Prediction
HEM	Homogenous Equilibrium Model
HPIS	High-Pressure Injection System
LOCA	Loss-of-Coolant Accident
LOCE	Loss-of-Coolant Experiment
LOFT	Loss-of-Fluid Test
LPIS	Low-Pressure Injection System
PWR	Pressurized Water Reactor
QOBV	Quick-Opening Blowdown Valve

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) L3-2. LOCE L3-2 is the second experiment to be performed in the LOFT Nuclear Small Break Test Series (Test Series L3). The objective of LOCE L3-2 is to examine LOFT's response to a break sized to cause the break flow to be approximately equal to high-pressure injection flow at an intermediate pressure of 7.6 MPa. LOCE L3-2 will represent a 0.15% break in a cold leg of a pressurized water reactor primary coolant pipe.

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. Separate EP analyses have been performed for LOCE L3-2 using the RELAP4 and the RELAP5 computer codes.

The RELAP4 analysis was terminated at 3600 s after experiment initiation. At this time during LOCE L3-2, manual control of the reactor will be initiated to bring the reactor into a cold shutdown mode. The RELAP5 analysis was run beyond the 3600-s time period by modeling a 44.4-K-per-hour cooldown ramp in the steam generator secondary. Accumulator flow was predicted to occur at approximately 7200 s and the analysis was terminated about 400 s later.

Differences in predictions from the two codes were expected because of the inherent differences between the codes, user input limitation between the codes, and slight differences between the two system models. However, both the RELAP4 and the RELAP5 analyses predict the upper plenum will depressurize to saturation pressure at about 450 s and establish a positive natural circulation flow through the intact loop of between 10 and 18 kg/s.

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

The RELAP5 calculation shows the pressure in the primary system to drop slowly during the transient and then stabilize at 7.5 MPa, at which time the high-pressure injection flow is approximately equal to break flow. Steam generator cooldown is initiated at 3600 s which is effective in cooling the primary system.

The RELAP4 calculation shows a faster depressurization in the primary system than does the RELAP5 calculation from 450 to 2500 s, then stabilizes briefly at 6.38 MPa. RELAP4 then predicts that vapor generated in the core will enter the intact loop hot leg from the upper plenum, proceed through the hot leg piping, and eventually blanket the steam generator tubes which effectively stops the intact loop flow at about 3200 s. The loss of steam generator heat removal causes the repressurization of the primary system. At 3600 s when the calculation was terminated, the upper plenum pressure was calculated to be 7.66 MPa. Calculated break flow exceeds high-pressure injection flow throughout the entire calculation. Steam blanketing of the steam generator as calculated by RELAP4 would effectively decouple the primary and secondary systems, possibly making ineffective the cooling of the primary system by cooling the secondary system. Initiation of the plant protection system would be required if pressure continued to increase.

The actual behavior of the primary system during LOCE L3-2 is expected to follow the RELAP4 calculation for pressure response during the first 2000 s, then continue to depressurize slowly as predicted by RELAP5, thus meeting the objective of the test. The RELAP4 and RELAP5 calculations both predict several feet of water will remain above the core throughout the entire LOCE L3-2 transient.

BEST ESTIMATE PREDICTION FOR LOFT NUCLEAR
EXPERIMENT L3-2

1. INTRODUCTION

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) L3-2. LOCE L3-2 is the second experiment to be performed in the LOFT Nuclear Small Break Test Series (Test Series L3). Predicted system responses are furnished to emphasize and clarify significant points of experiment predictions and to illustrate how the test objectives will be met.

Both the RELAP4 and the RELAP5 computer codes were used to calculate thermal-hydraulic behavior in the LOFT system during LOCE L3-2. Descriptions of the RELAP4 and the RELAP5 analytical models are presented. The analytical models used to perform these predictions should be recognized as a best estimate predictive mechanism.

Prior to performing LOCEs in the LOFT facility, best estimate experiment prediction (EP) analyses are performed. These EP analyses provide data for

1. Determining whether a LOCE will meet its stated objectives
2. Evaluating parameters that affect the safety of the LOFT facility during the intended LOCE
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.

The best estimate calculations for LOFT LOCE L3-2 also indirectly serve the function of providing data for the assessment of computer codes. However, true code assessment can only be done under strict modeling guidelines and must be done over a wide data base of experiments.

This document provides a description of the calculational techniques used in performing the experiment prediction for LOCE L3-2. Selected results are presented to illustrate the overall system response and to provide assurance that the experiment objective will be met.

Sections 1.1 and 1.2 of this introduction discuss the LOCE L3-2 objective and provide a brief description of LOCE L3-2 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 L3-2 Objective and Description

LOCE L3-2 is the second 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 objective for LOCE L3-2 is given in Section 1.1.1. LOCE L3-2 is described in Section 1.1.2.

1.1.1 LOCE L3-2 Objective

The objective of LOCE L3-2 is to examine LOFT's response to a small break sized to cause the break flow to be approximately equal to high-pressure injection system (HPIS) flow at an intermediate pressure. The plant pressure is expected to stabilize at about 7.6 MPa.

Items of particular interest during LOCE L3-2 include

1. Establishment and measurement of natural circulation around the intact loop

2. Identification of condensation heat transfer modes within the steam generator

3. Effectiveness of steam generator bleed for primary system cooldown.

1.1.2 LOCE L3-2 Description

LOCE L3-2 will represent a 0.15% 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.

Extensive instrumentation has been installed in the LOFT nuclear core area, the intact loop, the broken loop, and the blowdown suppression tank. All instruments providing data pertinent to evaluation of LOCE L3-2 are connected to the data acquisition system.

1.1.2.1 Initial Experiment Conditions. The following major initial conditions were selected for LOCE L3-2 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 ECC action takes place requiring the 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 L3-2 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 40 h of previous operation. Forty hours of operation provides a decay heat of 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 just upstream of the broken loop cold leg quick-opening blowdown valve (QOBV) will be opened. The blowdown 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 simulating 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.

Scaled amounts of borated liquid will be injected into the primary coolant system cold leg from only one accumulator, one HPIS pump, and one LPIS pump. HPIS flow will start upon receipt of a low system pressure signal of 13.16 MPa. The system is expected to stabilize at about 7.5 MPa at which time 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 per hour by bleeding steam from the steam generator while adding auxiliary feedwater to maintain steam generator water level. Accumulator flow will become available as the pressure decreases below 4.32 MPa. Cooldown will continue by bleeding the steam generator until the primary coolant temperature reaches 366 K. The EOS[1] provides a complete description of the cooldown procedure for all potential system conditions.

1.2 LOFT Facility Description

The LOFT facility is described in detail in Reference 2. The LOFT instrumentation and major components are shown in Figures 1 through 6. The instrumentation nomenclature is explained in Table 1.

2. CALCULATIONAL TECHNIQUES

The calculational techniques used to generate the EP data are discussed in this section. Section 2.1 presents the RELAP4 system model while a similar discussion for RELAP5 is presented in Section 2.2. Since no significant core thermal transient was predicted, no detailed fuel rod calculations were performed.

The modeling approaches presented in this report have been reviewed by the Experiment Prediction Consistency Committee at the Idaho National

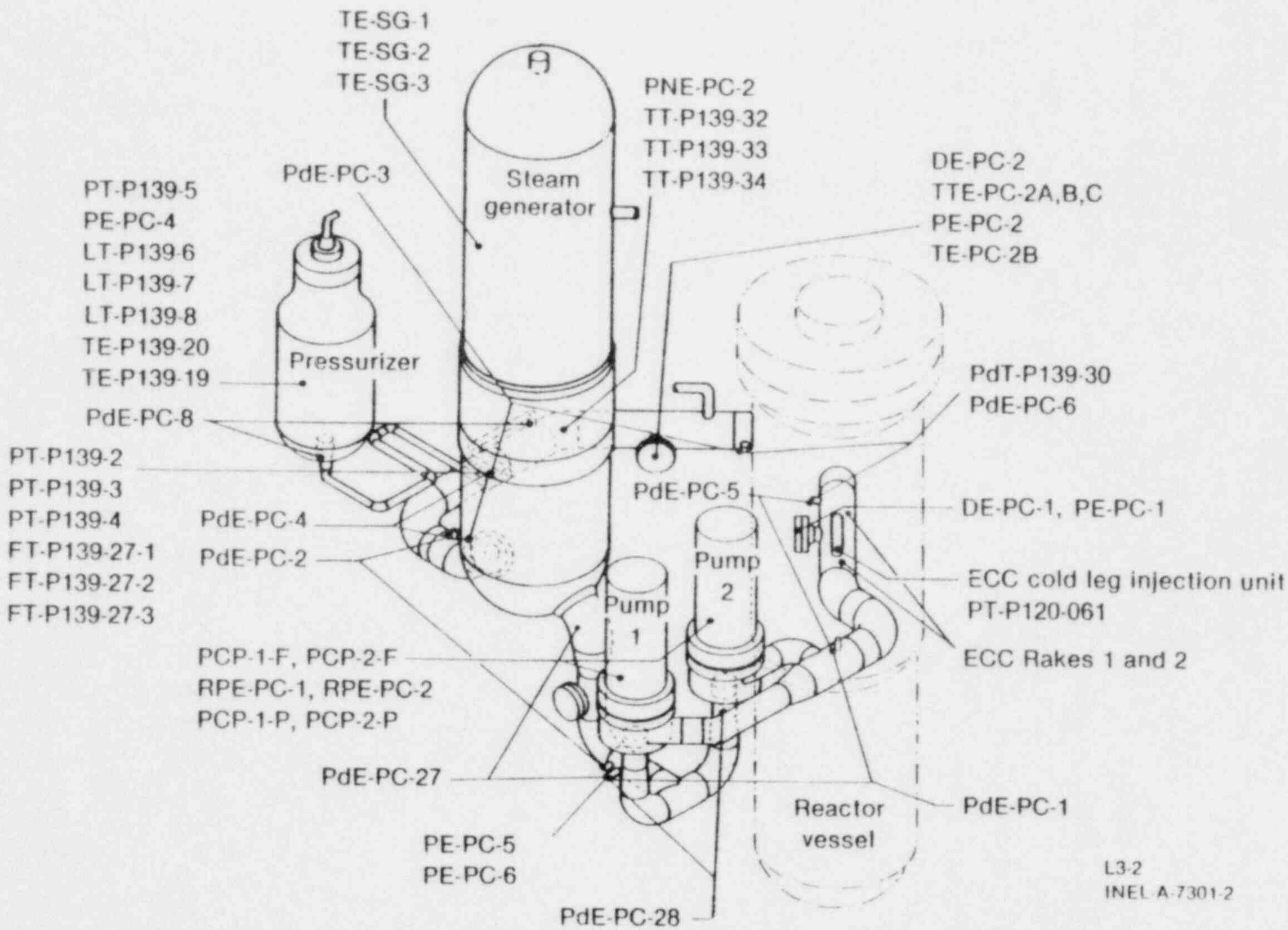


Figure 1. LOFT intact loop thermo-fluid instrumentation.

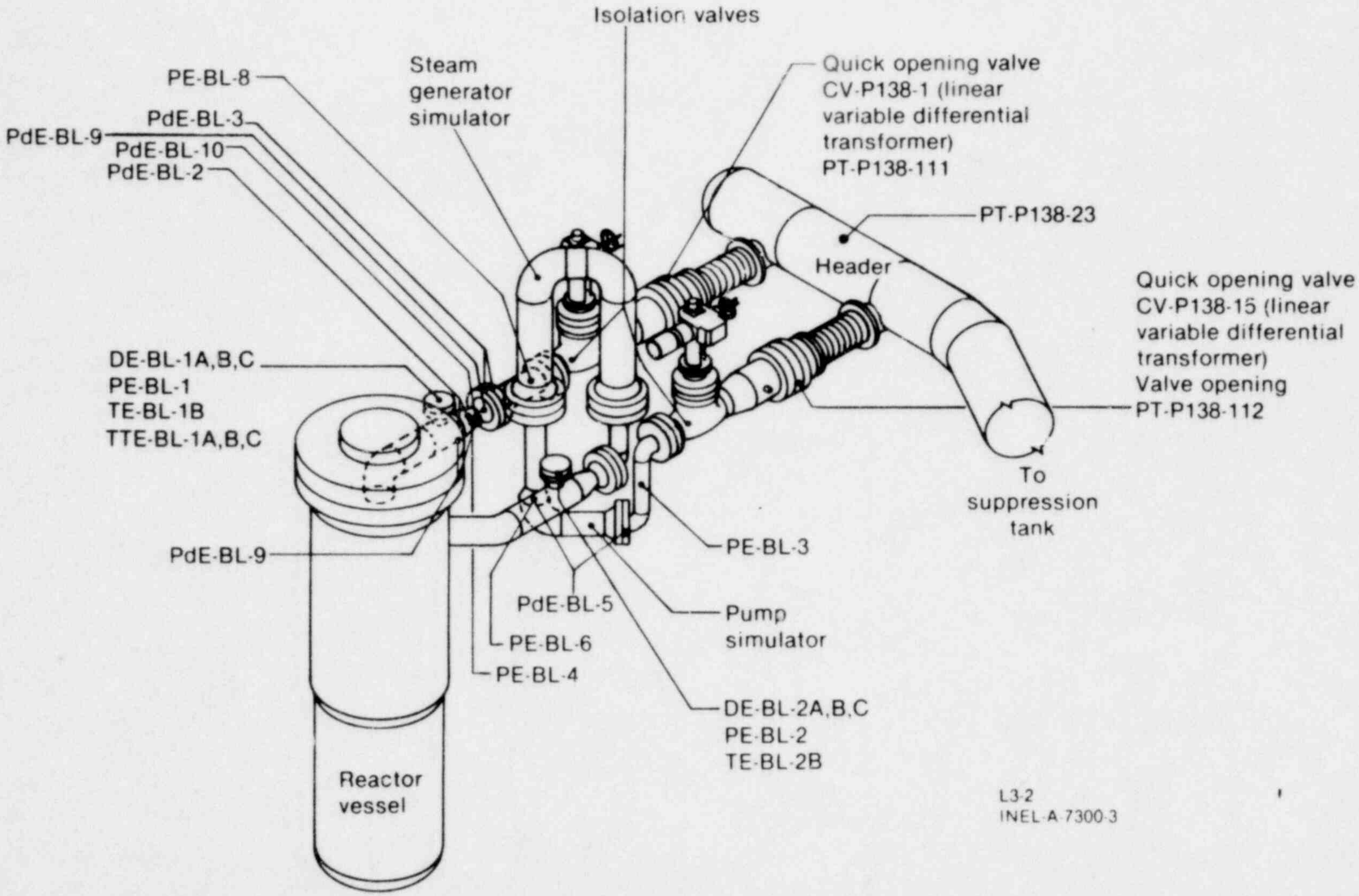
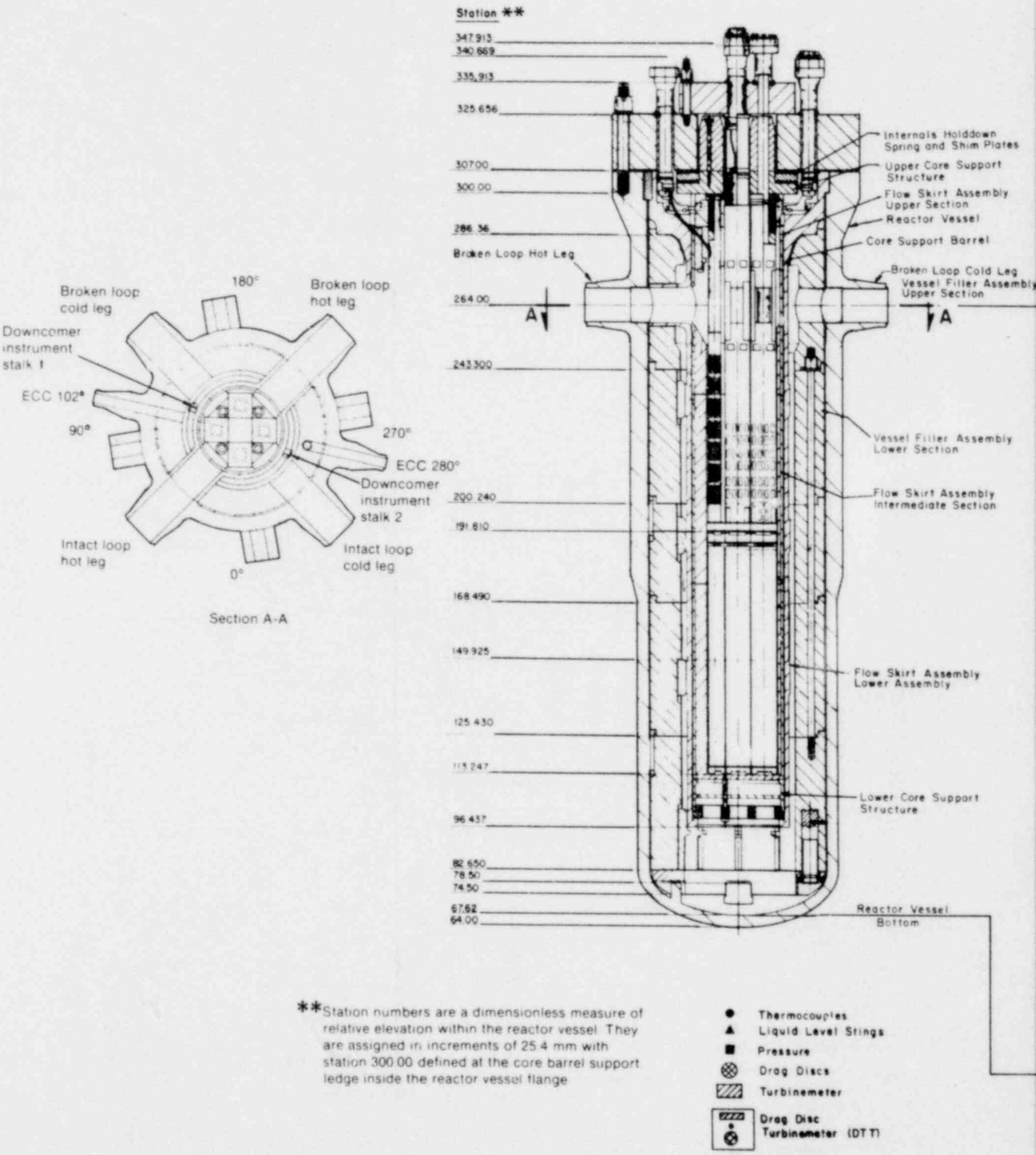


Figure 2. LOFT broken loop thermo-fluid instrumentation.



**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 3. LOFT reactor ve

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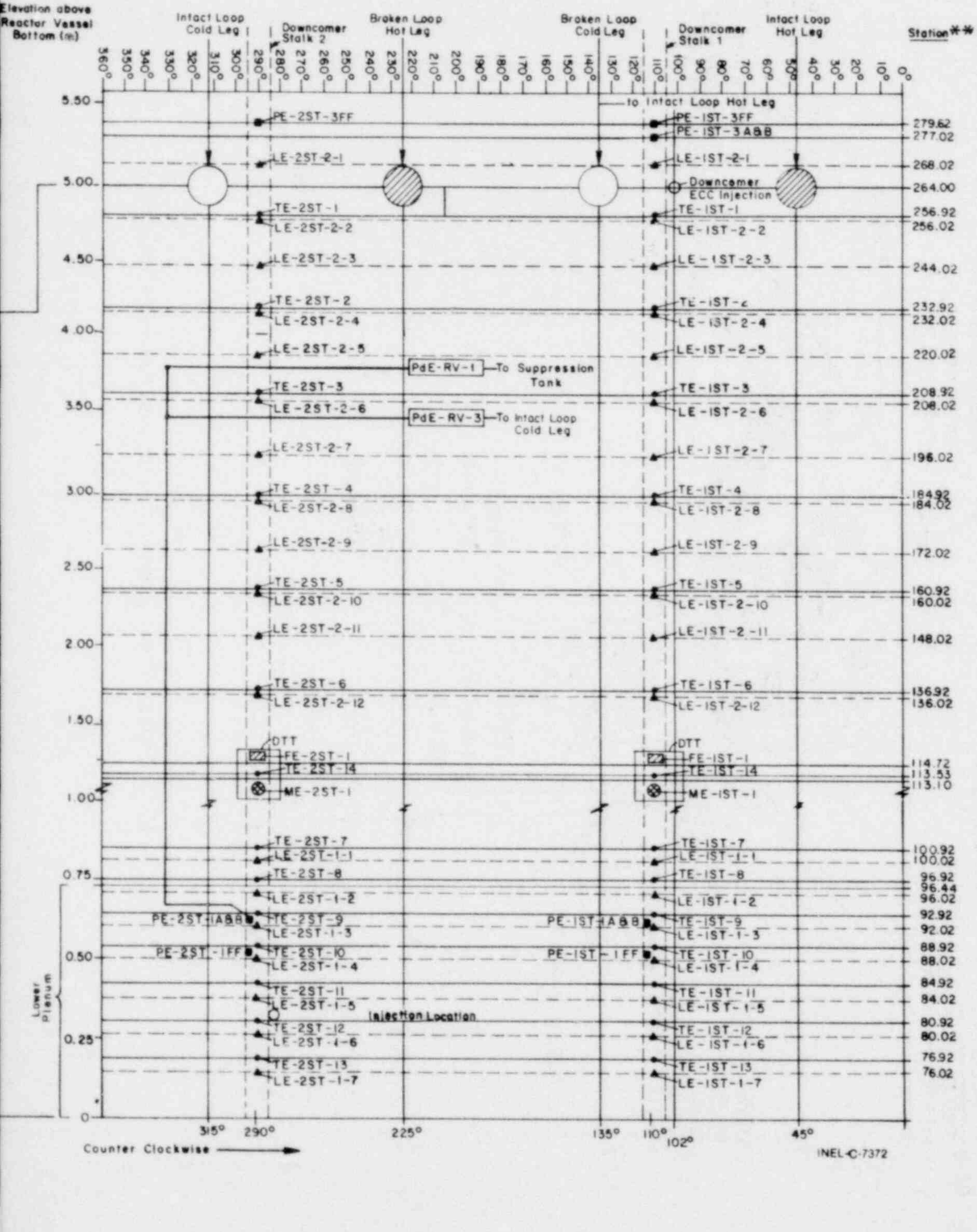

sel instrumentation.

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* 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.

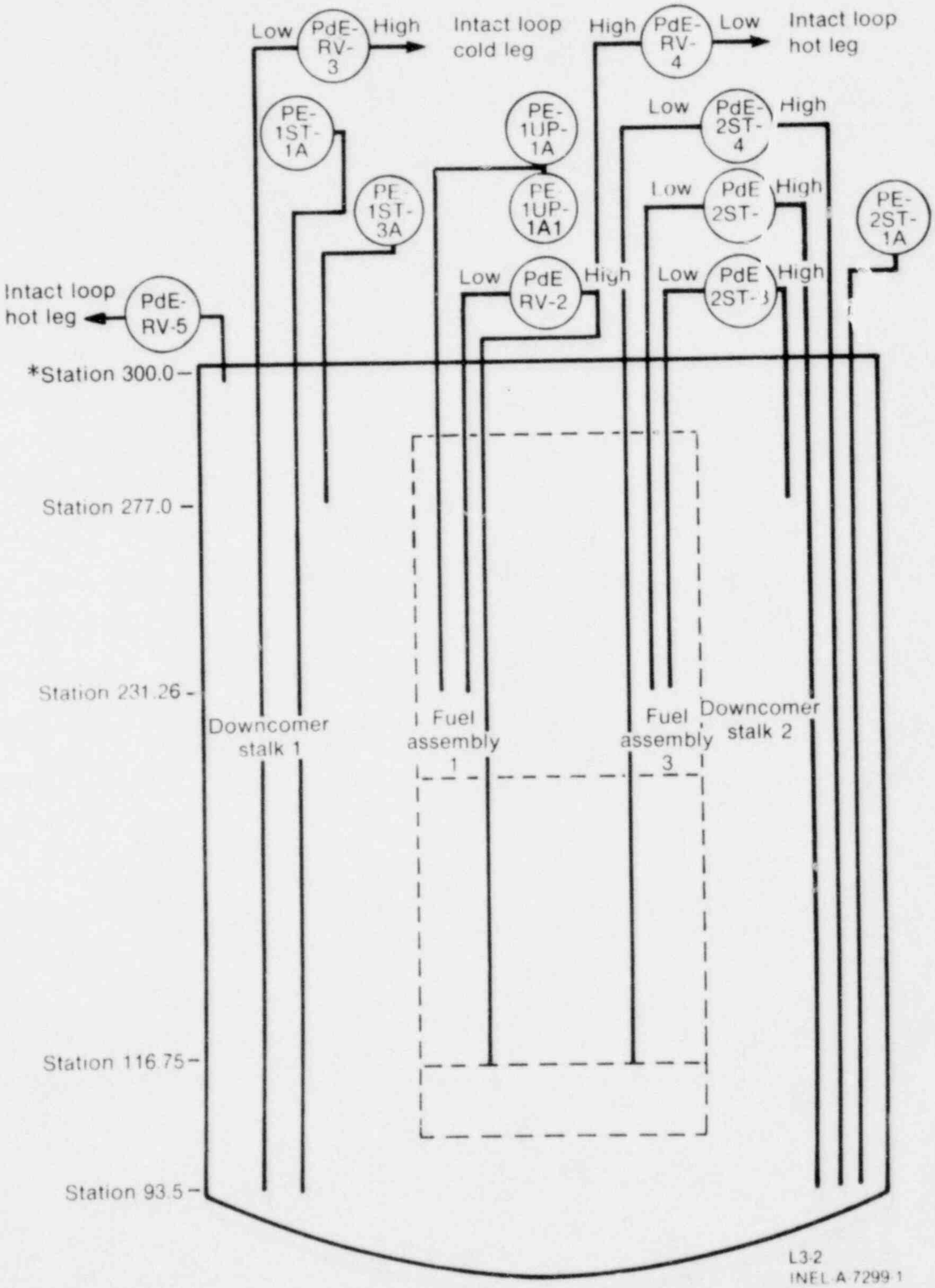


Figure 4. LOFT reactor vessel pressure and differential pressure instrumentation.

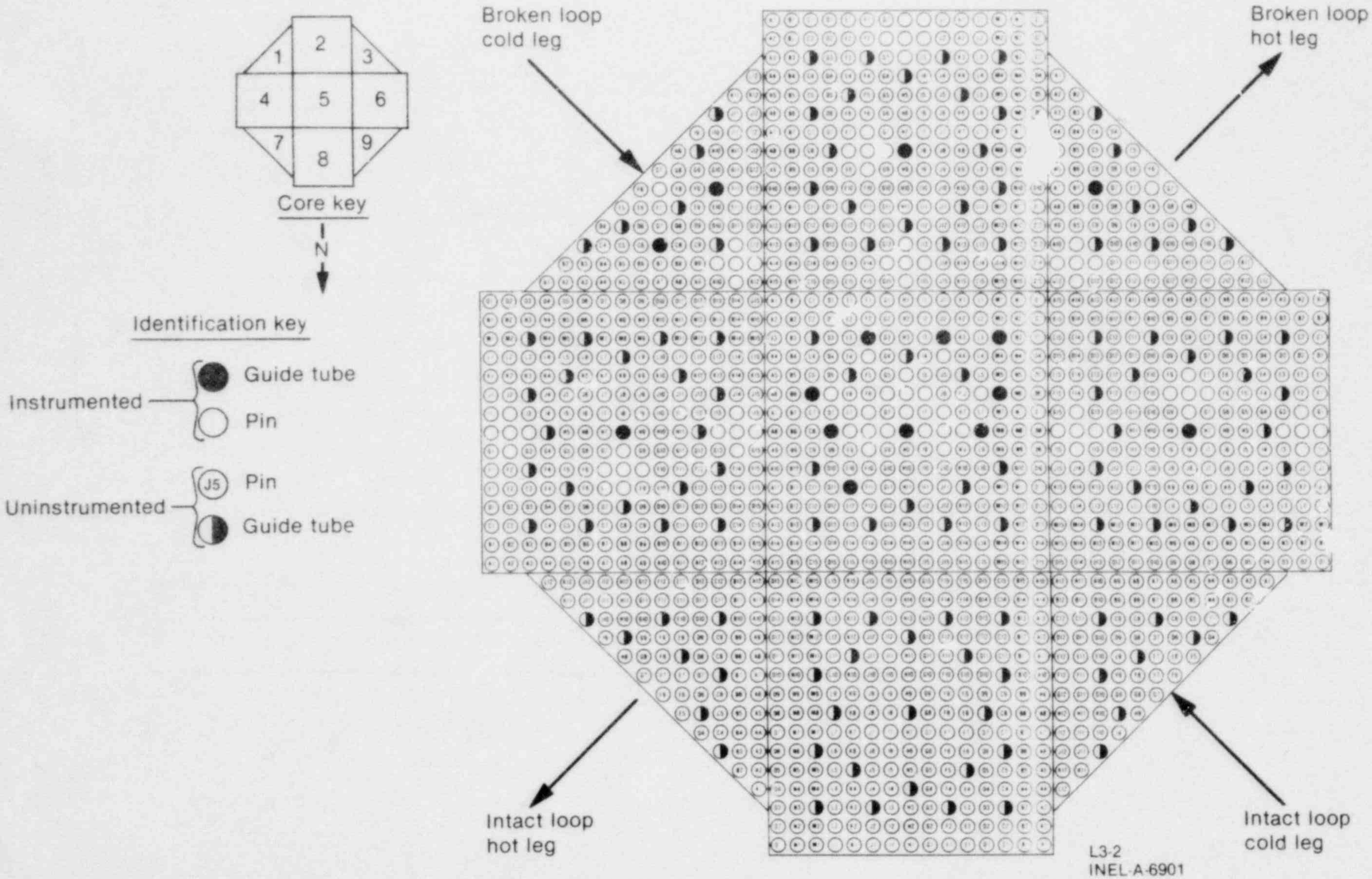
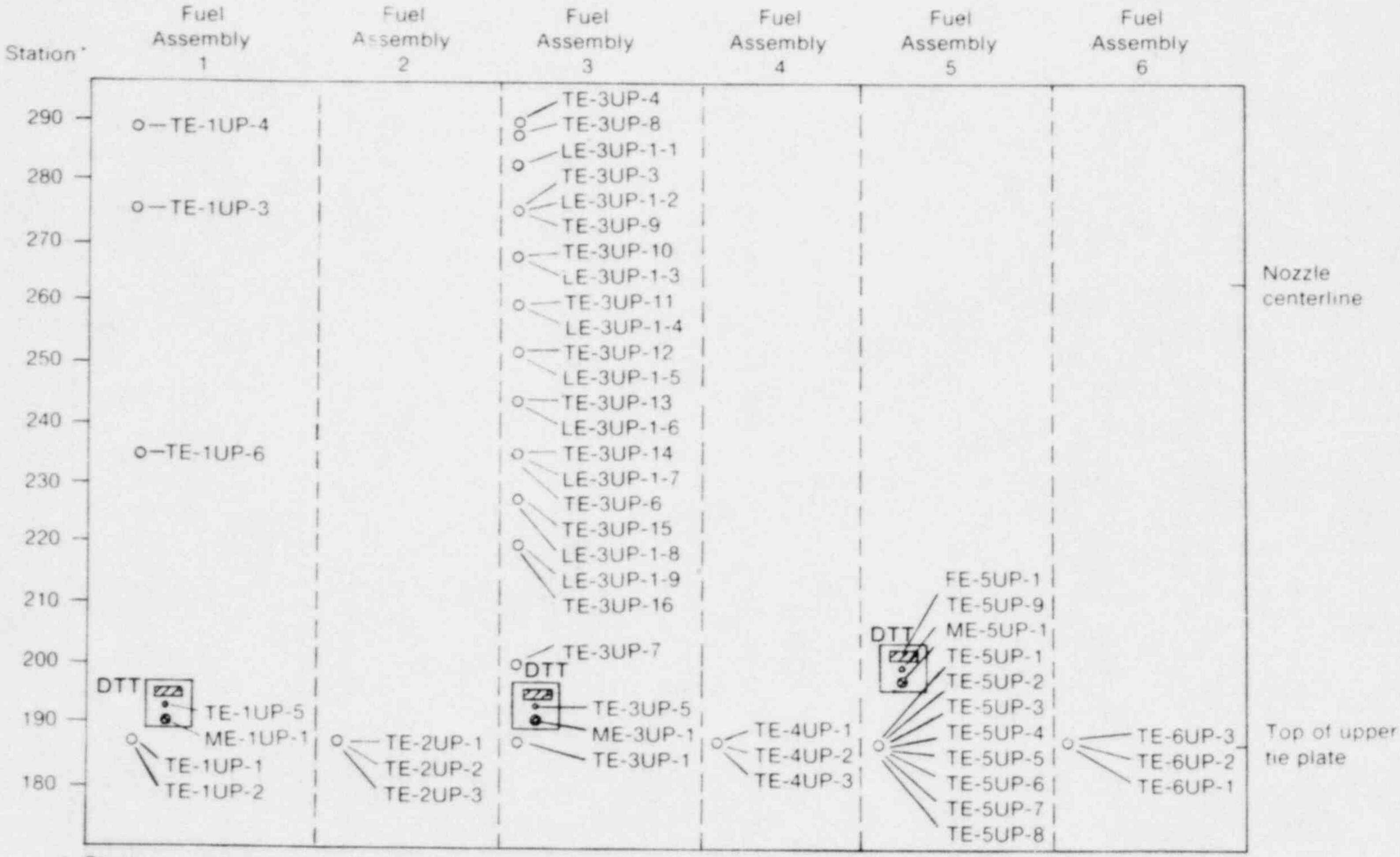


Figure 5. LOFT core map showing position designations.



* 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.

L3-2
INEL-A-7373-1

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 are as follows:

(1)	TE	-	Temperature element
(2)	TT	-	Temperature transmitter
(3)	PE	-	Pressure transducer
(4)	PT	-	Pressure transmitter
(5)	PdE	-	Differential pressure element
(6)	PdT	-	Differential pressure transducer
(7)	LE	-	Coolant level transducer
(8)	LT	-	Level transmitter
(9)	FE	-	Coolant flow transducer
(10)	FT	-	Flow transmitter
(11)	DiE	-	Displacement transducer
(12)	ME	-	Momentum flux transducer
(13)	RPE	-	Pump speed transducer
(14)	DE	-	Densitometer
(15)	LIT	-	Level indicating transmitter
(16)	CV	-	Control valve
(17)	PCP	-	Pump frequency transducer
(18)	TTE	-	Transit time element

The designations for the different systems are as follows:[a]

(1)	PC	-	Primary coolant intact loop
(2)	BL	-	Broken loop
(3)	SG	-	Steam generator
(4)	RV	-	Reactor vessel
(5)	SV	-	Suppression tank
(6)	UP	-	Upper plenum
(7)	LP	-	Lower plenum
(8)	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.

Engineering Laboratory and have been found to be in accordance with current and accepted practices.

2.1 RELAP4 Model

The code used for the EP analysis presented in this section was an experimental version of RELAP4.[a,3] Some subroutines in the RELAP4 code were changed to correct known coding errors and to incorporate the LOFT steam valve control logic into the code.[b] These changes are presented on microfiche in Appendix C.

The RELAP4 LOFT system model used to calculate blowdown during LOCE L3-2 is described in this section. The previously developed RELAP4 model for LOCE L3-1[4] was used in preparing this EP model with minor changes. The RELAP4 blowdown model of the LOFT thermal-hydraulic system is defined with close correspondence to the actual system and with the detail required to provide best estimate experiment predictions for LOFT nuclear LOCE L3-2. For example, junctions in the model generally correspond to changes in flow cross-sectional area and instrument locations. The system model was developed with the objective that the model should have sufficient nodalization so an increase in the number of nodes would not significantly alter analytical results.

A schematic of the LOFT system blowdown model is given in Figure 7. The model consists of 37 control volumes, 44 junctions, and 16 heat slabs. A brief description of each control volume is given in Table 2. The critical flow model specified for the junctions was the Henry-Fauske homogenous equilibrium model (HEM), which used the extended Henry tables in the subcooled region with a transition into the HEM model in the saturated region at 0.08% quality. Multipliers of 1.0 and 1.2 were applied to the saturated Henry and HEM values, respectively. These were

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a. The experimental RELAP4 code used for this analysis was RELAP4/MODG, Version 92, Idaho National Engineering Laboratory Configuration Control Number H00718B. This is an experimental version of RELAP4/MOD7.

b. The new object deck, which includes these changes, was RLP4G92LFT04, Idaho National Engineering Laboratory Configuration Control Number H01168IB.

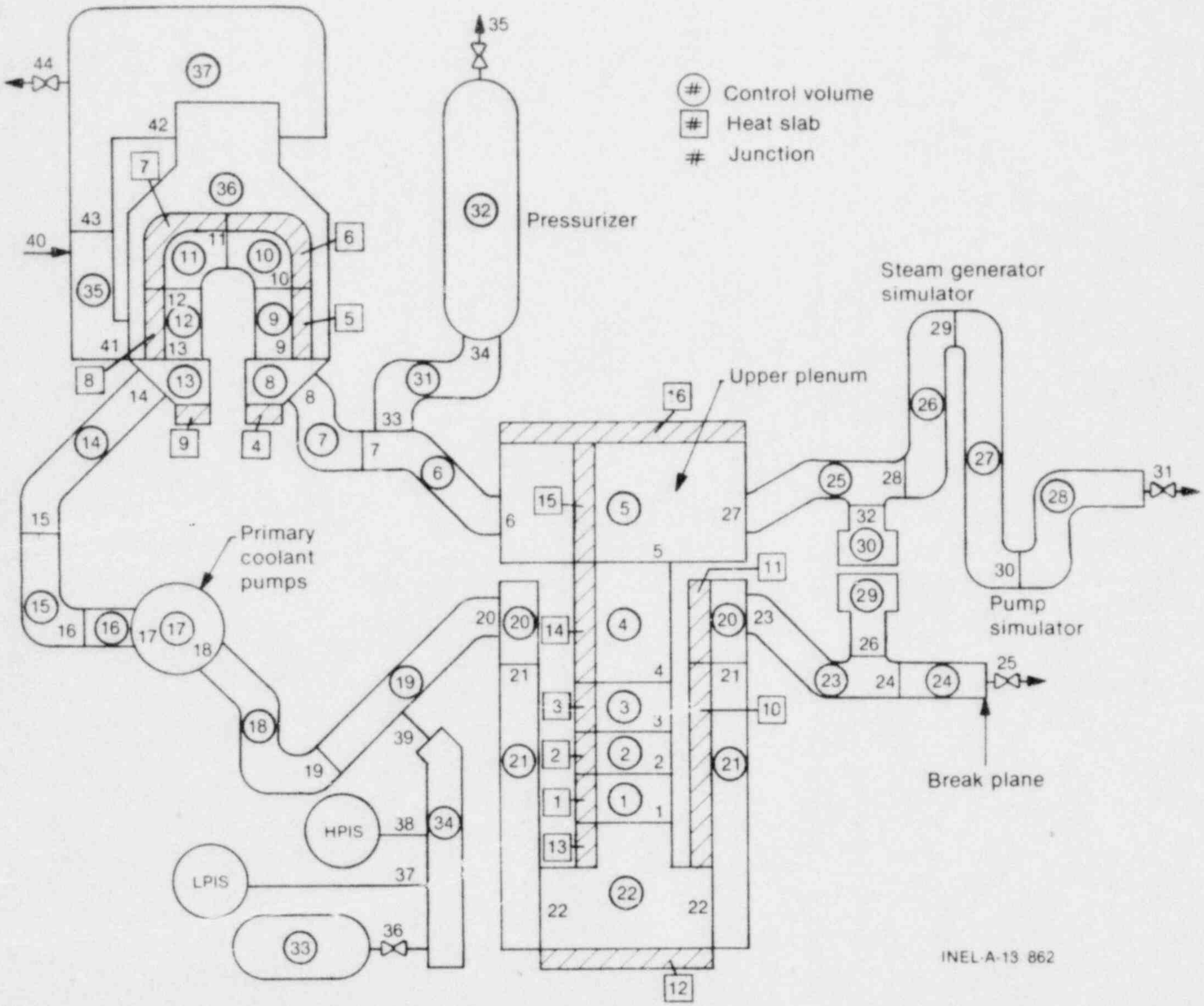

Figure 7. LOFT RELAP4 model schematic diagram.

TABLE 2. RELAP4 BLOWDOWN SYSTEM MODEL DESCRIPTION

Control Volume	Description
1, 2, and 3	Nuclear core
4 and 5	Upper plenum
6 and 7	Intact loop hot leg
8 and 13	Steam generator inlet plenum and outlet plenum
9 and 12	Straight sections of steam generator tubes
10 and 11	Curved sections of steam generator tubes
14	Steam generator outlet piping to the 16-to-14-in. (0.40-to-0.35-m OD) contraction
15	14-in. (0.35-m OD) piping leading to the tee preceding the coolant pumps
16	Piping from tee to primary coolant pumps
17	Primary coolant pumps
18 and 19	Intact loop cold leg
20	Upper annular region of the vessel inlet region
21	Downcomer region of the reactor vessel
22	Lower plenum
23 and 24	Broken loop cold leg
25, 26, 27, and 28	Broken loop hot leg
29 and 30	Reflood assist bypass piping
31	Pressurizer surge line
32	Pressurizer
33	ECC accumulator
34	ECC injection line
35	Steam generator secondary downcomer
36	Steam generator secondary shroud region
37	Steam generator secondary steam dome

the best available values based on data from separate blowdown experiments for an orifice of the size installed for LOCE L3-2. Model assumptions and considerations specific to LOCE L3-2 are discussed in the following paragraphs, along with changes from the LOCE L3-1 model. Model representations of the major components of the LOFT system are also discussed in the paragraphs following. An input listing and a zero time output listing are provided on microfiche in Appendix D.

Due to the small size orifice and long duration (3600 s) of LOCE L3-2, several additions to the RELAP4 model were considered but were not included: heat losses to environment, heat capacity effects of all primary piping, and core bypass. Heat losses to the environment were neglected since the total energy due to decay heat, over the 1-h period, is large compared to the total energy loss to the environment.

Heat capacities of the pump, hot and cold leg piping, and the pressurizer were not added to the model because the effects of the reactor vessel and steam generator plena heat capacities were already included, and more such heat slabs are not expected to have much additional effect. Although LOCE L3-2 may be as slow as the time constant of heat transfer to the piping walls, no large temperature differences between the water and piping are expected, minimizing the driving force for such heat transfer.

A core bypass path was not incorporated into the RELAP4 model because previous experimental comparisons and preliminary claculations indicated that it had little effect and was not required in the RELAP4 model. The presence of a bypass path does not alter the calculation during normal flow or natural circulation, but could be important during no-flow conditions. Since break flow in LOCE L3-2 is expected to remain liquid, a path to allow steam to travel directly from the upper plenum to the broken loop cold leg was not expected to be needed. There are potential flow paths around the core in LOFT, but the magnitudes of the leakage flows through them are uncertain.

Besides changes made due to the differences between LOCEs L3-1 and L3-2, the RELAP4 model was changed by making the volumes in the broken loop cold leg homogeneous because the break flow during LOCE L3-2 is expected to remain liquid. In the intact loop hot leg, allowance was made for flow stagnation by specifying the "vertical slip" model at all of the junctions there. This allows circulation of steam moving from the upper plenum to the steam generator while liquid flows in the reverse direction. However, RELAP4 does not have a special reflux heat transfer model for condensing steam in the steam generator tubes to model this condition completely.

The pressurizer and pressurizer surge line and the section of the hot leg into which the pressurizer empties were modeled with three nodes. Control Volumes 31 and 32 represented the pressurizer surge line and the pressurizer, respectively. The pressurizer model included a steam head above the initially saturated liquid contained in the pressurizer. The Wilson bubble rise model was used in the pressurizer volume, but tripped to homogeneous when the level dropped to 0.03 m. A two-phase multiplier was applied to the single-phase form loss coefficient at the pressurizer surge line junctions to account for two-phase effects in the bends of the pressurizer surge line.

Both the primary and secondary sides of the LOFT steam generator were represented in the model. The primary side model included control Volumes 8 and 13 representing the steam generator inlet and outlet plenums (with Wilson bubble rise), and Volumes 9 through 12 representing the tube bundle. The steam generator secondary side was represented by three control volumes and five junctions. Volumes 35, 36, and 37 represent the downcomer, shroud, and steam dome regions, respectively. Volume 35 was homogeneous and bubble rise was specified in Volume 36 and complete separation in Volume 37. For LOCE L3-2, the steam generator will be operated in a manner specified by the Experiment Operating Specification (EOS).[1] Therefore, the secondary side model included a feedwater inlet and a steam outlet junction. The time-dependent mass flow rate for the feedwater junction was specified in the input data. Coincident with the reactor scram, the steam flow valve will be controlled as a power-operated relief valve for the secondary system. The control logic and valve characteristics based on actual valve performance were included in the model. Heat conduction from the primary coolant to the secondary coolant was by means of four heat slabs, one for each tube bundle control volume. The natural convection heat transfer option in RELAP4 was used for the heat slabs connected to the steam generator secondary side fluid volume. A heat slab was also included on each steam generator plenum volume.

The pump suction piping was modeled with control Volumes 14, 15, and 16. Wilson bubble rise in the volumes with vertical slip at the vertical junctions and the vertically-stacked-volumes option were used in this region to adequately calculate the loop seal blowout phenomenon.

The primary coolant pumps were represented by control Volume 17. Both single- and two-phase pump operating performance was described by the pump model. The LOFT pump model has been described in previous EP documents.[4,5]

The LOFT core was modeled with three axially stacked control volumes (Volumes 1, 2, and 3). The inlet annulus and downcomer were each modeled as a single control volume. Wilson bubble rise was used in all reactor vessel control volumes, and vertical slip was used at all vertically oriented junctions. The vertically stacked volume option was used both for the two-volume inlet annulus-downcomer stack and for the six-volume upper plenum-core-lower plenum stack. One heat slab in each of Volumes 1, 2, and 3 was used to model the reactor core. Seven additional heat slabs were used to model structures within the reactor vessel as shown in Figure 7.

The model of the ECC systems includes the accumulator injection system, the LPIS, and the HPIS. For LOCE L3-2, the ECC will be injected directly into the intact loop cold leg through a control volume which models the flow line connected to the accumulator, HPIS, and LPIS. The LPIS and HPIS were represented by fill junctions (Junctions 37 and 38, respectively) for which flow rates were determined by tables in the RELAP4 input, which describes flow as a function of pump discharge pressure. Tabular data for LOCE L3-2 were taken from the EOS.[1] (Note that the LPIS and accumulator were retained in the model although they should not be activated during the first hour of LOCE L3-2.) The accumulator was modeled by control Volume 33, which used the complete phase separation bubble rise model of RELAP4. The nitrogen gas present in the LOFT accumulator was modeled as an air head with a polytropic expansion model (PV^n = constant with n = 1.030 to model near-isothermal behavior). The accumulator injection line resistance was calculated from single-phase test data.

Since the broken loop hot leg QOBV will remain closed for LOCE L3-2, detailed modeling of the steam generator and pump simulators was not necessary. Four control volumes (Volumes 25, 26, 27, and 28) for the broken loop hot leg and one volume (Volume 30) for the hot leg reflood assist bypass piping were required to adequately model draining of the broken loop hot leg. Wilson bubble rise was specified in Volumes 26 and 27.

The broken loop cold leg was modeled using two control volumes (Volumes 23 and 24), and one volume (Volume 29) was used for the cold leg reflood assist bypass piping. Volumes 23 and 24 are homogeneous, and Volume 29 tripped to homogeneous just prior to emptying of liquid. The vertically-stacked-volume option was used with Volumes 29 and 23 with vertical slip at Junction 26 to adequately model draining of the reflood assist bypass piping.

Since the back pressure from the containment is expected to have little effect on a small break transient, the blowdown suppression system was not explicitly included in this RELAP4 model. A leak junction with constant back pressure was used to model the break location.

2.2 RELAP5 Model

This experiment prediction for LOCE L3-2 represents the first time that the RELAP5 code has been used successfully for a formal prediction. For this reason it seems appropriate here to describe briefly the RELAP5 code and some of its advantages. This code description is followed by a description of the nodalization used for the LOCE L3-2 prediction.

2.2.1 RELAP5 Code

RELAP5 is a computationally-efficient, two-fluid, nonequilibrium, user-oriented code. Simulation of the LOFT integral test system required little user time in setup and debugging of the data required to describe the details of the LOFT system. The RELAP5 code is described in Reference 6, which states:

> The hydrodynamic calculation is primarily organized around volumes and junctions and to a lesser extent around components. Components are organized collections of volumes and junctions and are defined for either input convenience or to specify specialized processing. The physical space over which the hydrodynamic behavior is being simulated is subdivided into volumes. The continuity and energy equations are approximated by finite difference approximations to the volume and surface integrals of these equations over each volume. A junction is the connection of one volume to another and is associated with the momentum equations. Finite difference approximations to the line integral of the stream tube form of the momentum equations are used.

The thermal calculation is organized around heat structures. Different heat structures attached to the same hydrodynamic component are identified by a geometry number. A heat structure can simulate a conductor consisting of laminations having different thermal properties. Temperatures and heat transfer rates are computed from the one-dimensional form of the transient heat conduction equation.

RELAP5 offers several advantages at the user level. Virtually no decisions are made at this level (code input data) with relation to the form of the equations mentioned above. Additionally the computational efficiency means that a single input deck can be generated for a given facility which can be used to simulate a wide variety of transients in that facility. These features allow evaluation of the code on its own merits using a standard input data set that has undergone a high level of quality assurance.

2.2.2 RELAP5 Nodalization

The nodalization used for the LOCE L3-2 RELAP5[a] calculation is described in this section. This nodalization is similar to the

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a. The version of the code used for this calculation was RELAP5/MOD"0". The source deck and update input data deck are stored under Idaho National Engineering Laboratory Configuration Control Numbers H005785B and H005985B, respectively.

nodalization used for the RELAP4 blowdown calculation of LOFT LOCE L2-3. In areas where significant elevation differences exist, the RELAP5 nodalization was increased to define steep density gradients. The RELAP5 nodalization also includes simulation of the potential bypass flow paths between the reactor vessel inlet annulus and upper plenum. The nodalization scheme is shown in Figure 8. A brief description of each node is given in Table 3.

Special treatment was required in certain components as described below. The primary separator and mist extractor of the steam generator are modeled by modifying the donor formulation of the convective terms for Component 10. The steam flow control valve is assumed to have a linear area change with stem position and a zero inertia constant speed driver. The RELAP5 valve subroutines required modification to model this type of valve. The sophisticated trip logic in RELAP5 allows simulation of the valve controller. The out flow is sent to Component 16, which simulates the air-cooled condenser where the pressure is given. The feed flow is input as a function of time by Component 17.

The ECCS is represented by Components 168, 500, and 505. Component 168 uses the accumulator model described in the RELAP5 manual[7] and models LOFT Accumulator A. The LPIS and HPIS pump models, Components 505 and 500, respectively, required modification to the TMDPJUN subroutine in RELAP5. The flow provided by these components is assumed to be known as a function of downstream pressure.

The orifice at the break plane is modeled by Component 365, a valve. The valve area is the same area as the drilled section of the orifice.

Heat conduction between the primary and secondary sides of the steam generator is through heat Structure 5-2, the steam generator tubes. The reactor pressure vessel, filler blocks, core filler, upper and lower core support structures, and core also use heat conductors. The system is modeled with no heat loss to the surroundings.

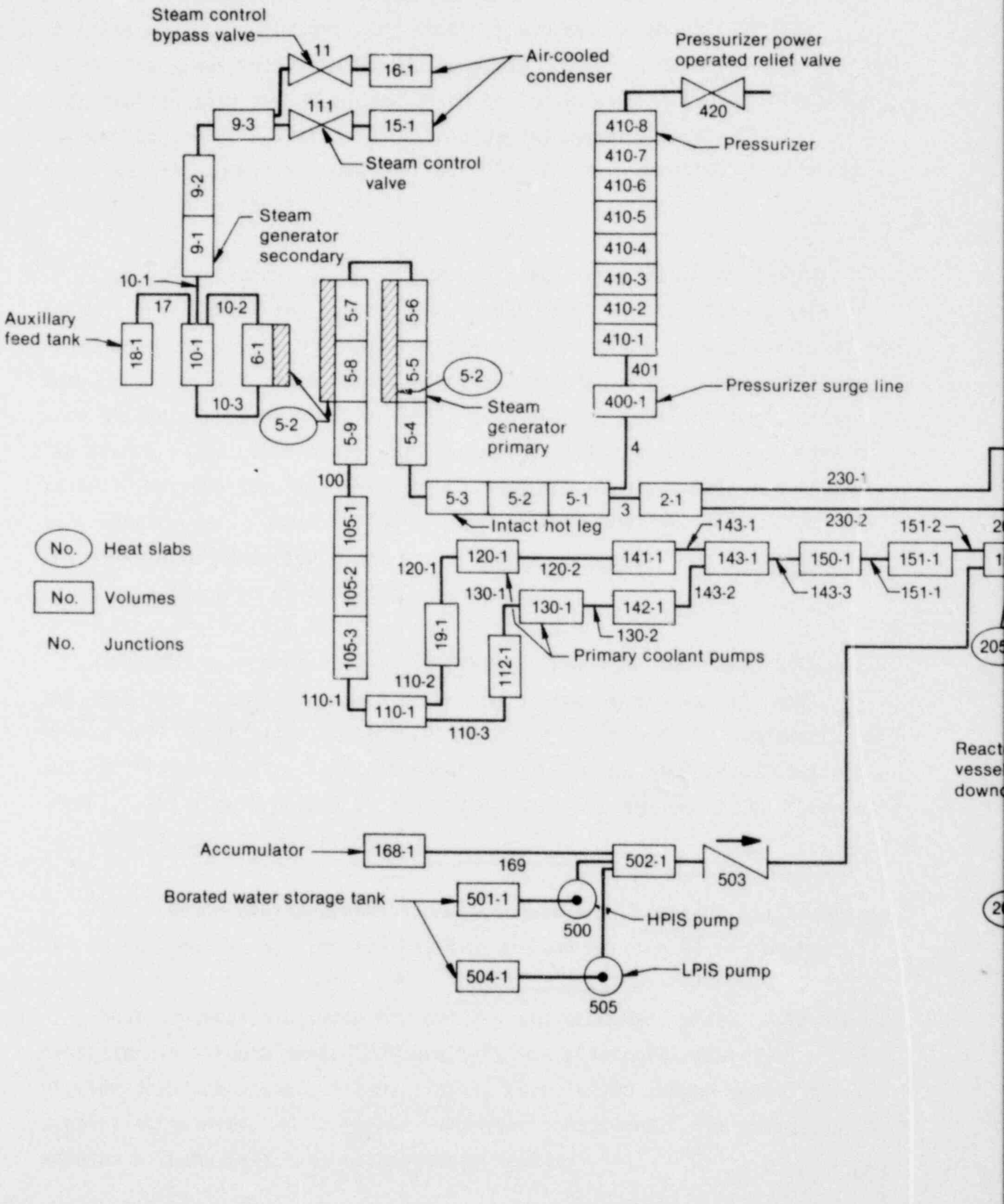
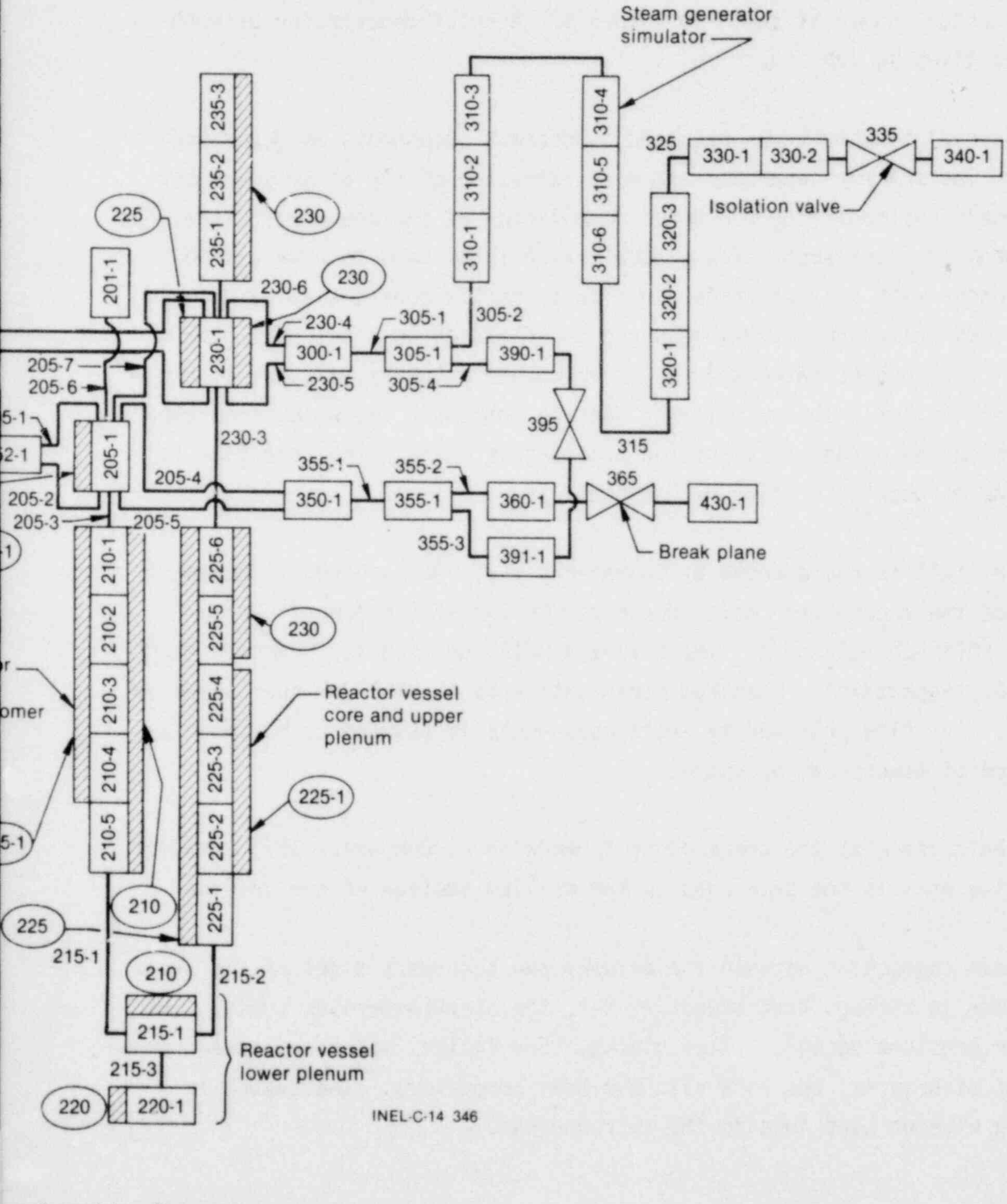


Figure 8. LOFT RELAP5 mode



schematic diagram.

TABLE 3. NODALIZATION DESCRIPTION FOR RELAP5 LOCE L3-2 PREDICTION

Component	Description
2	Intact loop hot leg nozzle
5	Intact loop hot leg steam generator plenums and tubes
6	Steam generator boiler and riser
9	Steam generator steam dome and outlet piping
10	Steam generator downcomer
11	Steam flow-control valve bypass valve
15 and 16	Air-cooled condenser
17	Steam generator auxiliary feedwater
18	Air-cooled condenser hot well, auxiliary feed supply tank
105	Pump suction piping
110	Pump suction tee
019	No. 1 pump inlet
111	Steam flow control valve
112	No. 2 pump inlet
120	No. 1 pump
130	No. 2 pump
141	No. 1 pump outlet
142	No. 2 pump outlet
143	Pump discharge tee
150, 151, and 152	Cold leg piping
168	ECC accumulator
201, 205, and 210	Reactor vessel downcomer
215, 220, and 225-1	Reactor vessel lower plenum
225-2, -3, and -4	Reactor core
225-5 and -6, 230, and 235-1 and -3	Reactor vessel upper plenum
300 and 305	Broken loop hot leg
310	Steam generator simulator
320	Pump simulator
330	Pump simulator outlet

TABLE 3. (continued)

Component	Description
340	Containment
350, 355, and 360	Broken loop cold leg
365	Orifice at broken loop break plane
430 and 431	Containment
390 and 391	Reflood assist piping
400	Pressurizer surge line
410	Pressurizer
500	HPIS pump
501 and 504	Borated water storage tank
502	ECC piping to cold leg
505	LPIS pump

3. COMPUTATIONAL RESULTS

This section presents the results of the experiment predictions made using both the RELAP4 and RELAP5 computer codes. A separate brief scenario for each of the predictions is presented in Sections 3.1 and 3.2. A discussion of the differences between the two calculations is presented in Section 4. The RELAP4 calculation was terminated 1 h after the initiation of the transient. At 1 h into the transient calculation, manual depressurization of the secondary system was started. The RELAP5 calculation continued through this manual operation period until the accumulator flow started.

3.1 RELAP4 Results

The transient is initiated by opening the cold leg QOBV. For the first 52.6 s after experiment initiation, the pressure is predicted to decrease 1.41 MPa from the initial value, causing a reactor scram to initiate as shown in Figure 9. For the next 10 s, the steam generator removes more energy from the primary system than the reactor core adds, as the steam valve closes, while the reactor drops to decay heat levels resulting in a net stored energy loss in the primary loop. The resulting density increase shown in Figure 10 in the primary coolant adds to the demand on the pressurizer, resulting in the liquid level in the pressurizer dropping and a high depressurization rate. Around 60 s, the steam flow control valve shuts completely, slowing the rapid pressure decline in the primary system. Pressure in the secondary rises abruptly. The coolant pumps coast down in 25 s. At about 100 s, the HPIS is initiated by low pressure in the hot leg, but at about 380 s (see Figure 11) the pressurizer empties and the primary system rapidly goes to saturation pressure corresponding to the fluid temperature in the intact loop hot leg (see Figure 12). The steam flow control valve is predicted to start opening at about 250 s. There is a small natural circulation flow of about 5% in the intact loop (see Figure 13). The steam valve is predicted to stay closed in the period between 1000 and 2000 s. At 1900 s, auxiliary feedwater flow is shut off, causing a transient in the steam generator which reduces pressure. Changes in secondary pressure can be observed as density changes in the intact loop cold leg piping. The subcooled, auxiliary feedwater had remained in the steam generator downcomer, but now it mixes and reaches equilibrium with the fluid in the shroud surrounding the steam generator tubes, causing the sudden drop in the secondary pressure and temperature. This low temperature promotes an increased flow in the primary loop.

The steam generator pressure response is shown in Figure 9 and indicates that for the first 3200 s heat transfer will be from the primary loop into the steam generator with the steam generator pressure being controlled by the steam flow control valve at its open and shut setpoints.

The quality in the reactor vessel upper plenum is shown in Figure 14. Voiding has been contained to the upper plenum and the core is not predicted to uncover, thus the thermal response shown in Figures 15 through 18 is calculated not to be severe.

As the system stabilizes at about 6.2 MPa during the period from 2000 to 3000 s, vapor develops inside the upper parts of the steam generator tubes and by 3000 s the tubes become steam filled, impeding the loop circulation, and flow in the primary loop drops off.

The remainder of the transient shows the progression of the steam bubble, which formed in the steam generator tubes, as it enlarges to fill the entire hot leg. Colder fluid drains to the loop seal below the pump, where mixing with subcooled HPIS water further enhances this stagnation effect. Heat input from the core and lack of any significant heat transfer out of the steam generator (due to their voided fluid condition and lack of net loop circulation) cause the primary to begin to repressurize. These conditions persist to the end of the calculation, and they would remain stable until the steam pressure became large enough to lift the slug of cold water out of the loop seal, since there is no other path from the upper plenum to the break location in this model.

Note that the HPIS flow is always less than the break flow in this calculation (see Figure 19). This is due to the larger discharge coefficient than previously used in planning calculations and the system repressurization at 3300 s. The objective of LOCE L3-2 to determine the LOFT response to the type of small break where HPIS flow approximately equals break flow will not be met according to these calculations.

Figures 9 through 23 show system responses calculated with RELAP4. The following is a brief explanation of the figures:

1. Figure 9 shows the primary and secondary pressure responses discussed in the preceding paragraphs.

2. Figure 10 shows the density in the reactor vessel downcomer and lower plenum. It is rising at 3000 s into the transient as the cold leg traps the subcooled HPIS water.

3. Figure 11 shows the water level in the pressurizer, which drains as the primary reaches saturation.

4. Figure 12 shows pressure and saturation pressure in the hot leg. The saturation pressure forms the lower bound which maintains the primary pressure around 6.9 MPa after 400 s.

5. Figure 13 shows flow in the cold leg holding steady during the auxiliary feed period, increasing for a time during the secondary transient, then dropping to zero as the steam generator tubes void.

6. Figure 14 shows quality in the reactor vessel above the core. Voiding has been contained to the upper plenum.

7. Figures 15 through 18 depict the temperature transient in the upper third of the core. The core remains covered, and the transient is not severe since the core remains immersed in low quality water.

8. Figure 19 shows the break flow and the HPIS flow. Break flow follows the pressure response since there is no back pressure. Break flow is always choked. HPIS flow mirrors the pressure response. Repressurization of the system after 3300 s is not due to HPIS flow since it never exceeds the break flow.

9. Figure 20 shows quality in the steam generator tubes. The upper parts void first, followed by the lower regions as the steam bubble progresses lower.

10. Figure 21 shows temperature and saturation temperature in the steam generator downcomer. When feed flow is turned off at 1900 s, the downcomer fluid mixes and reaches equilibrium.

11. Figure 22 shows the densities in the three regions of the steam generator secondary. The subcooled fluid in the downcomer mixes with the shroud region and fluid flows up into the steam dome.

12. Figure 23 shows the water level in the steam generator downcomer which shrinks while auxiliary feedwater is being supplied but holds an equalibrium level later.

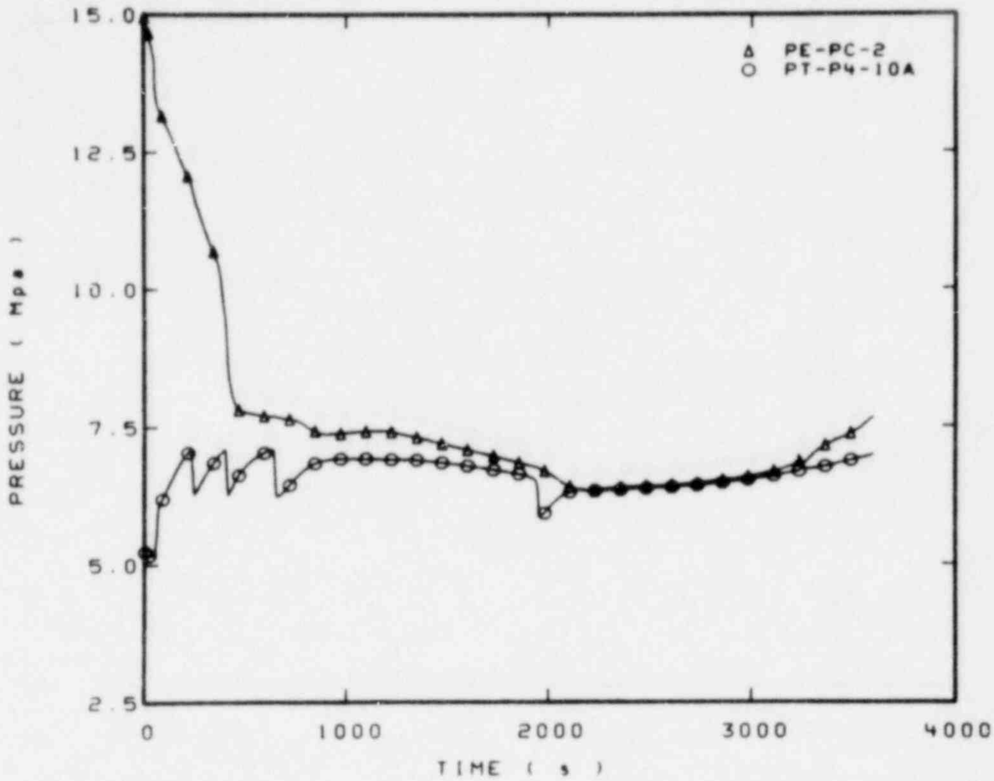


Figure 9. RELAP4 predicted primary and secondary pressure.

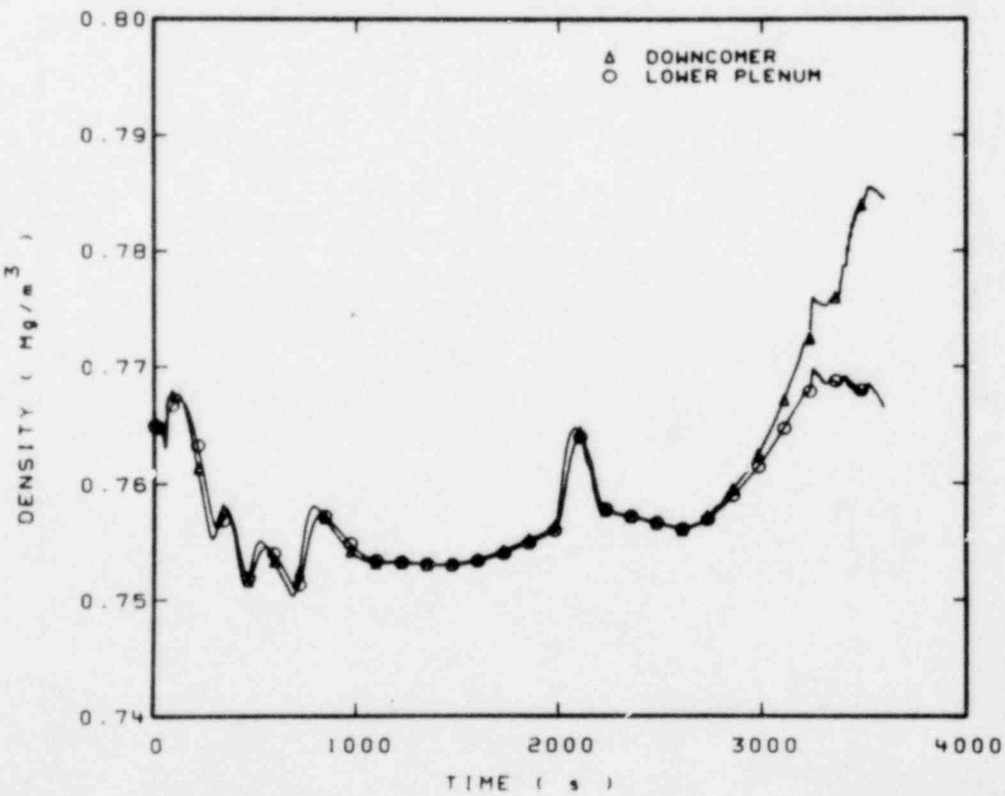


Figure 10. RELAP4 predicted reactor vessel downcomer and lower plenum density distribution.

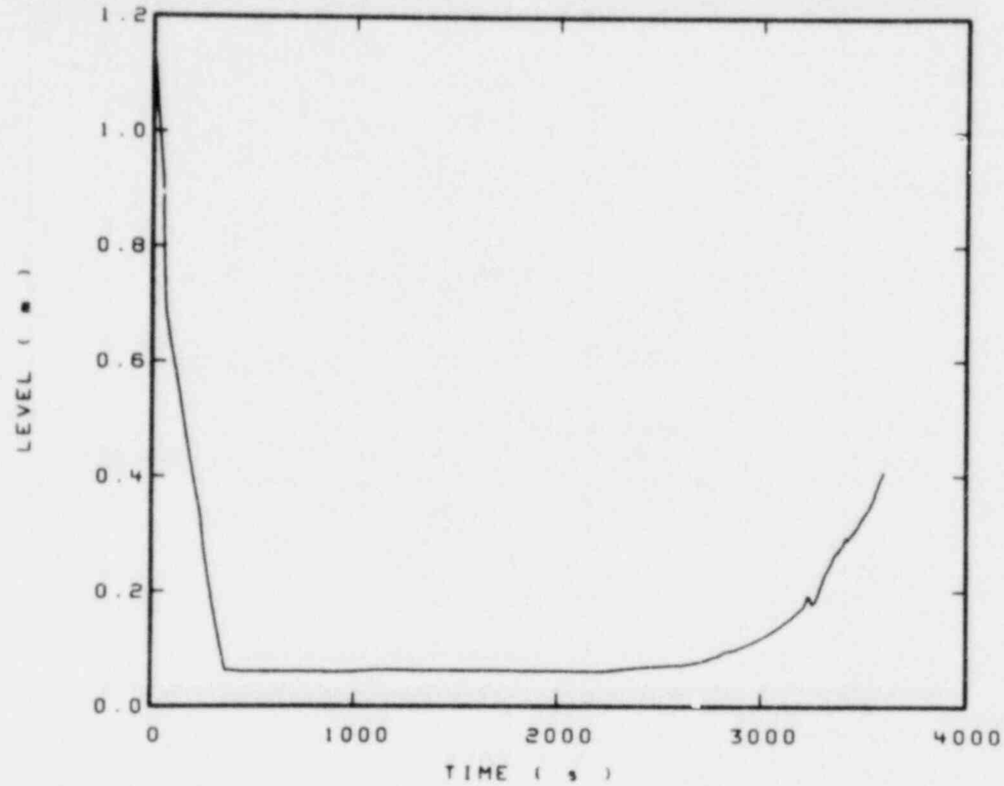


Figure 11. RELAP4 predicted pressurizer mixture level.

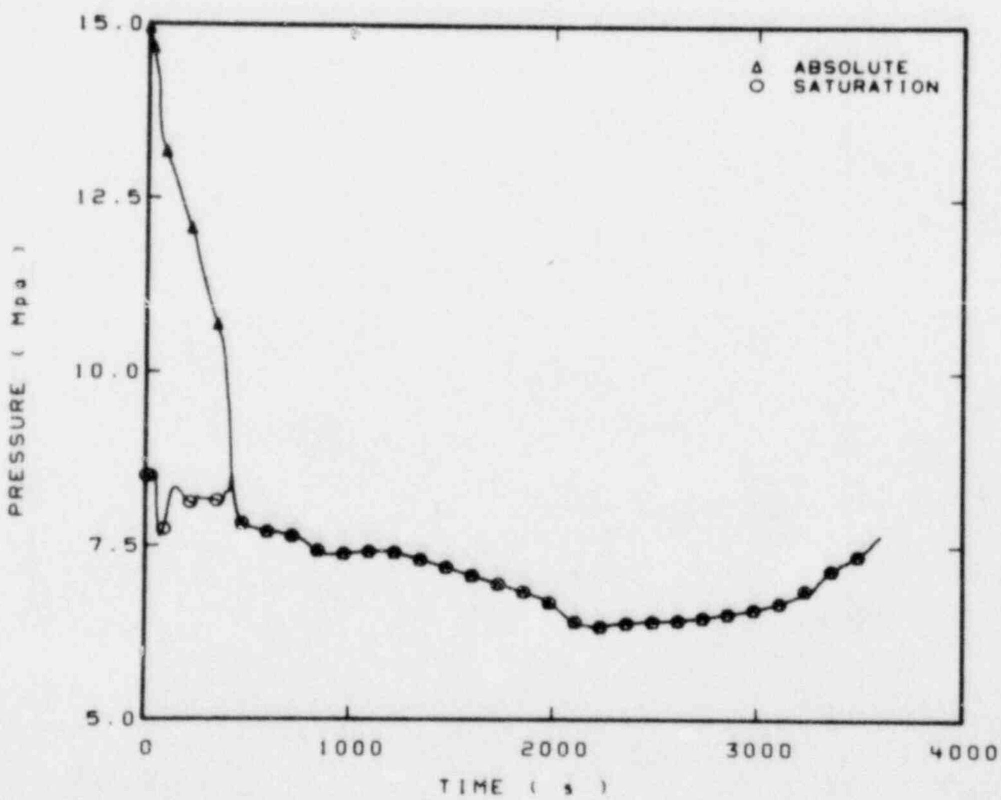


Figure 12. RELAP4 predicted intact loop hot leg pressure and saturation pressure.

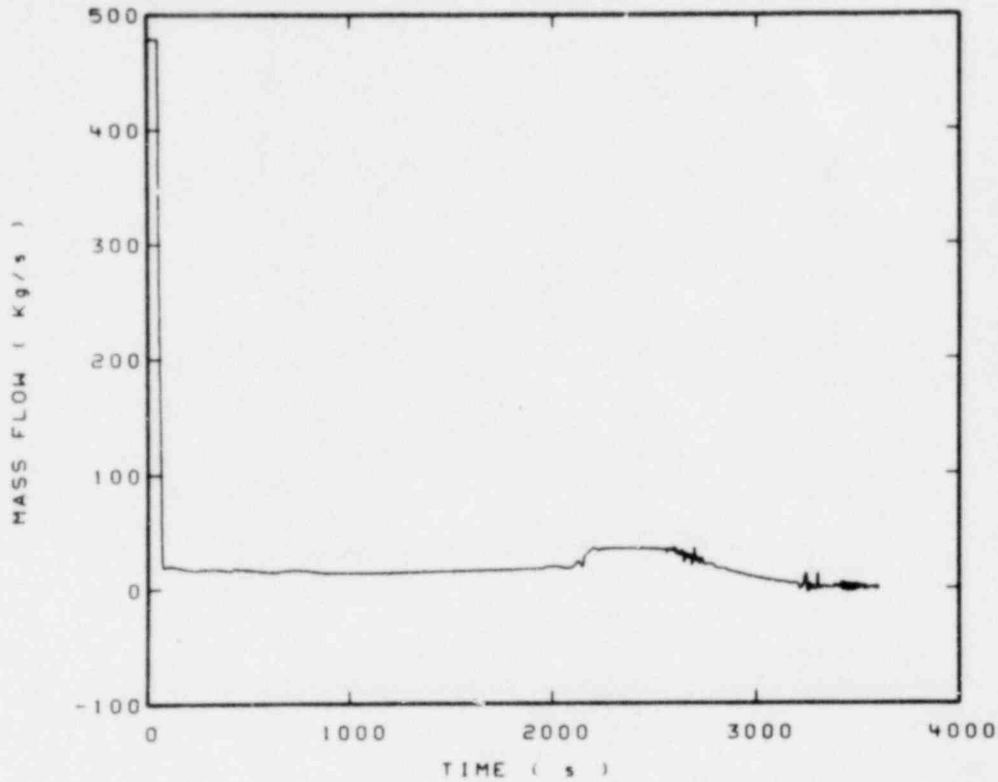


Figure 13. RELAP4 predicted intact loop flow.

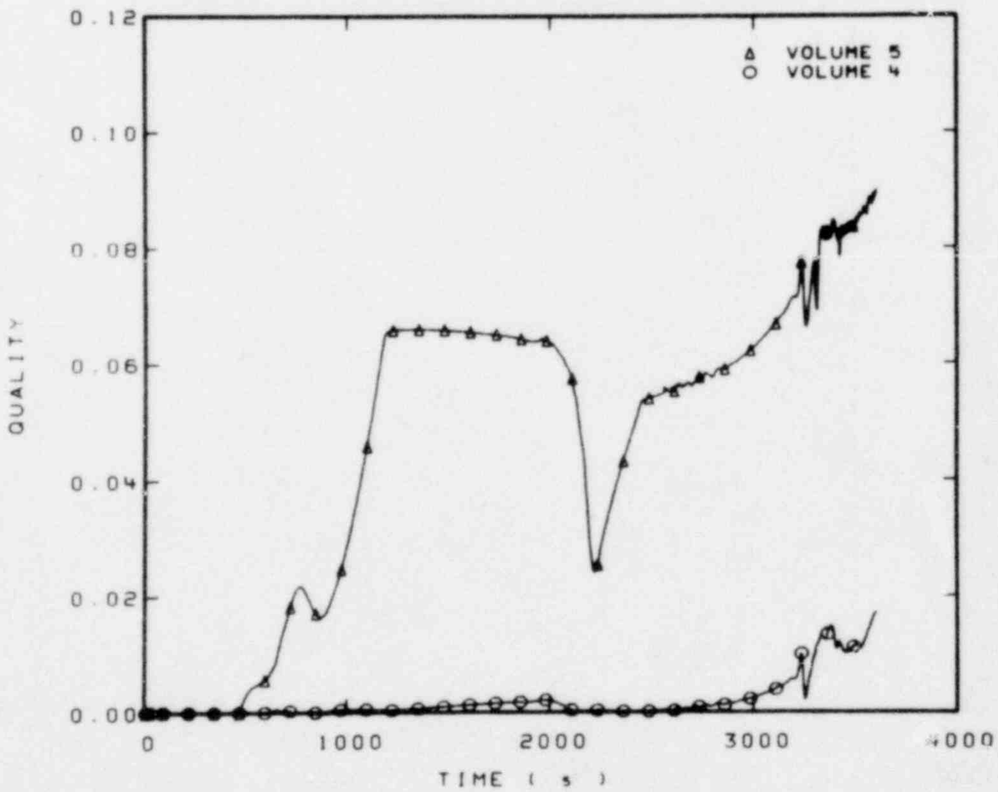


Figure 14. RELAP4 predicted core and upper plenum quality distribution.

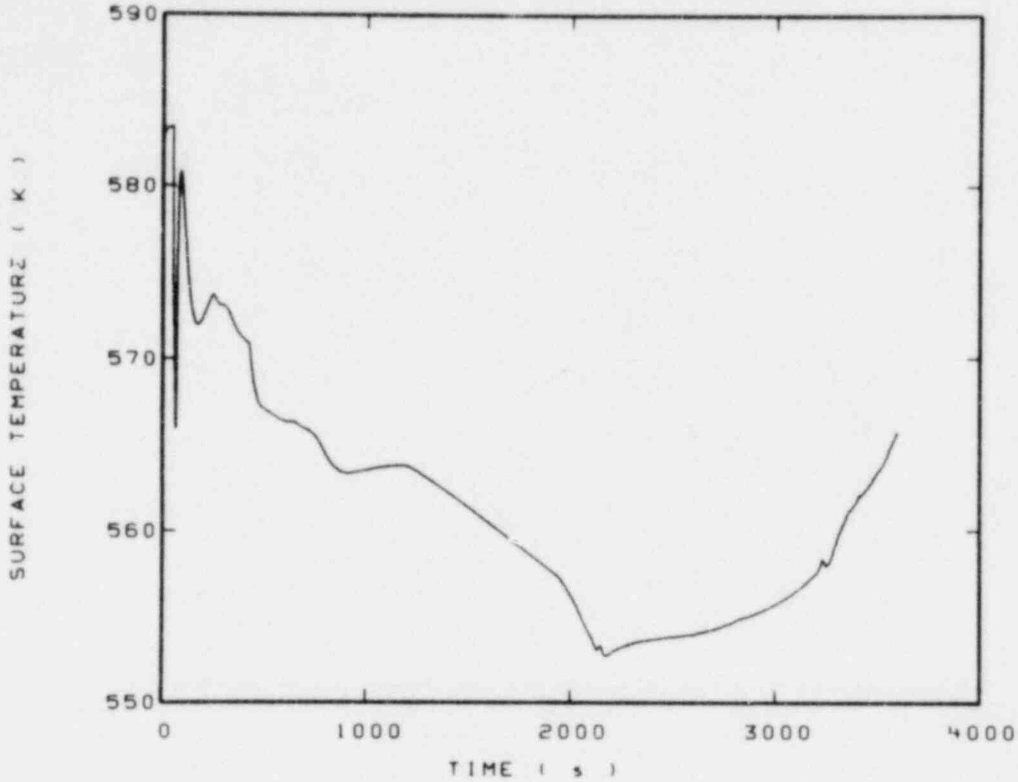


Figure 15. RELAP4 predicted cladding temperature for upper third of core.

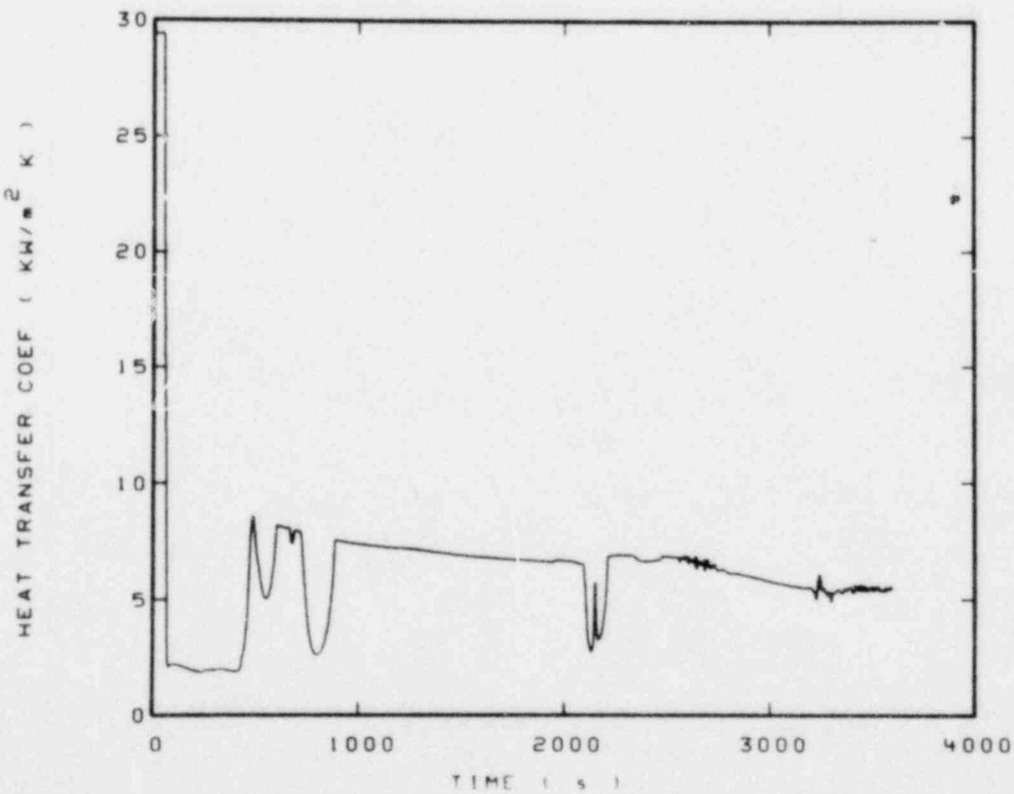


Figure 16. RELAP4 predicted heat transfer coefficient for upper third of core.

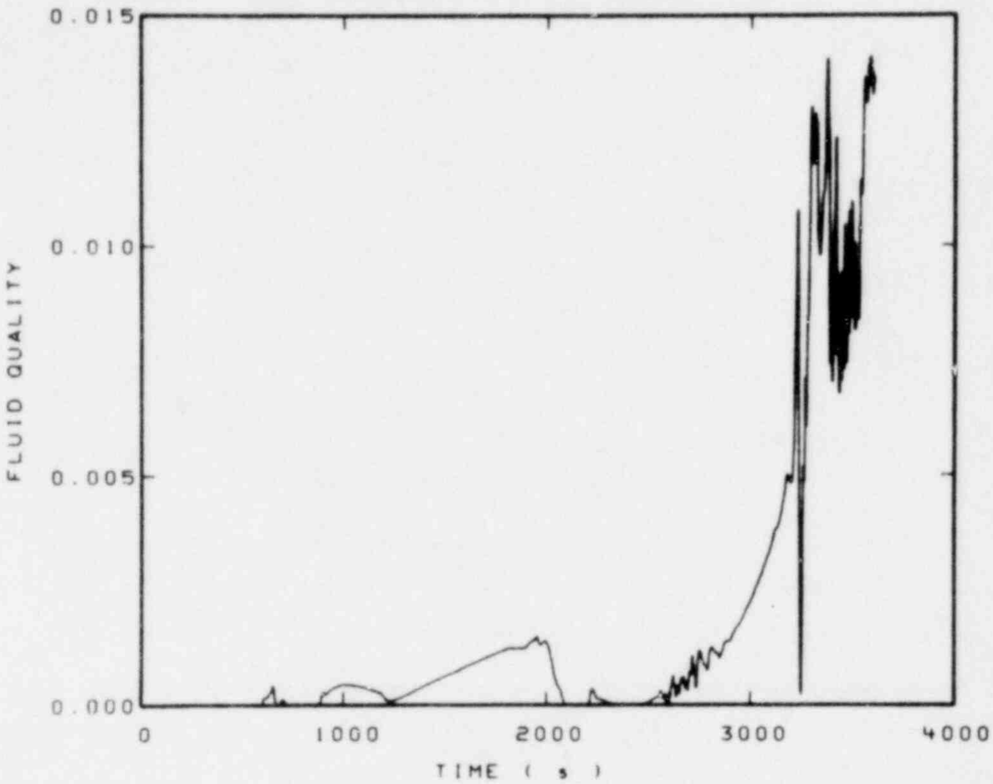


Figure 17. RELAP4 predicted fluid quality in upper third of core.

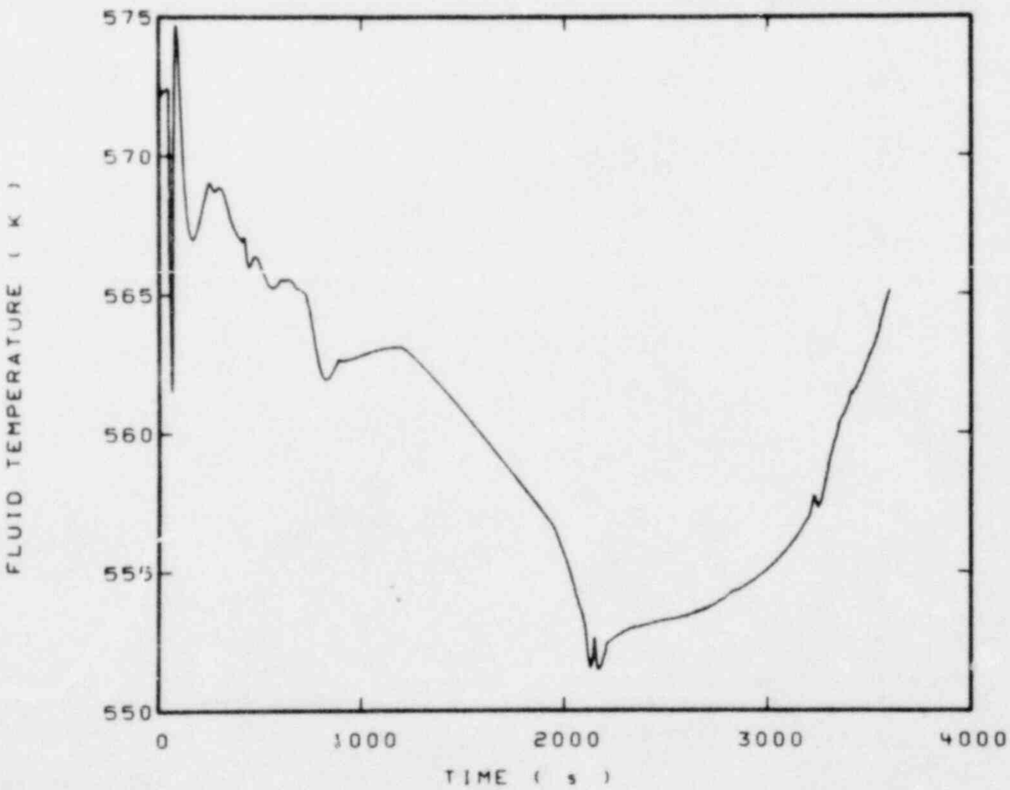


Figure 18. RELAP4 predicted fluid temperature in upper third of core.

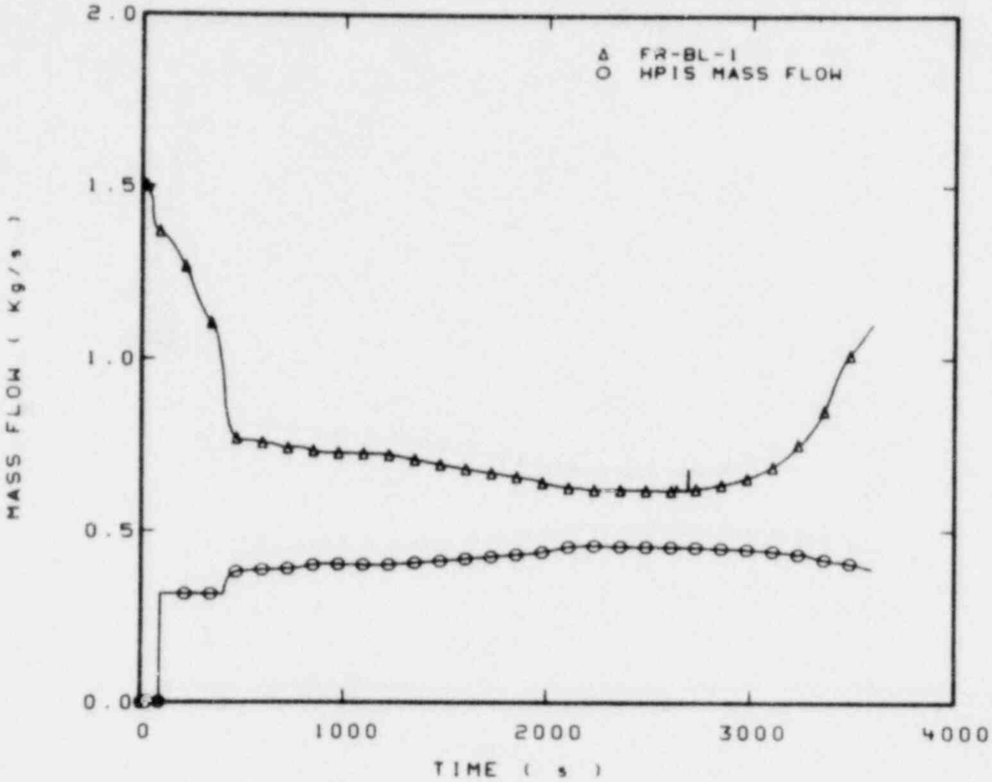


Figure 19. RELAP4 predicted break flow and HPIS flow.

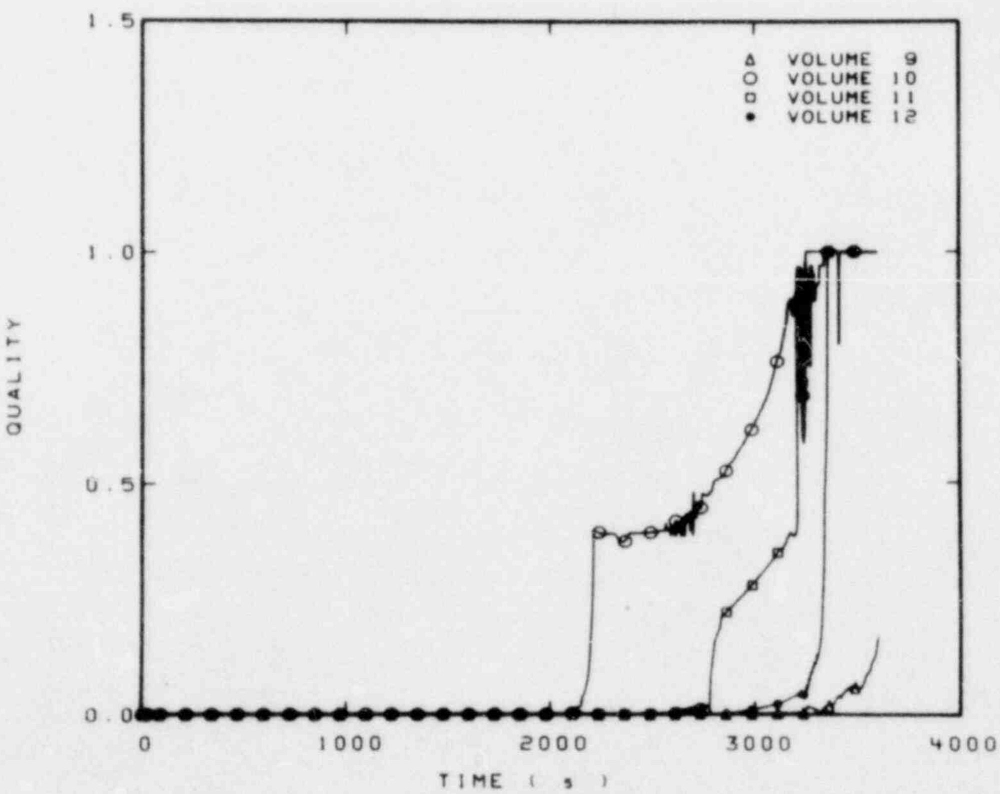


Figure 20. RELAP4 predicted steam generator primary quality distribution.

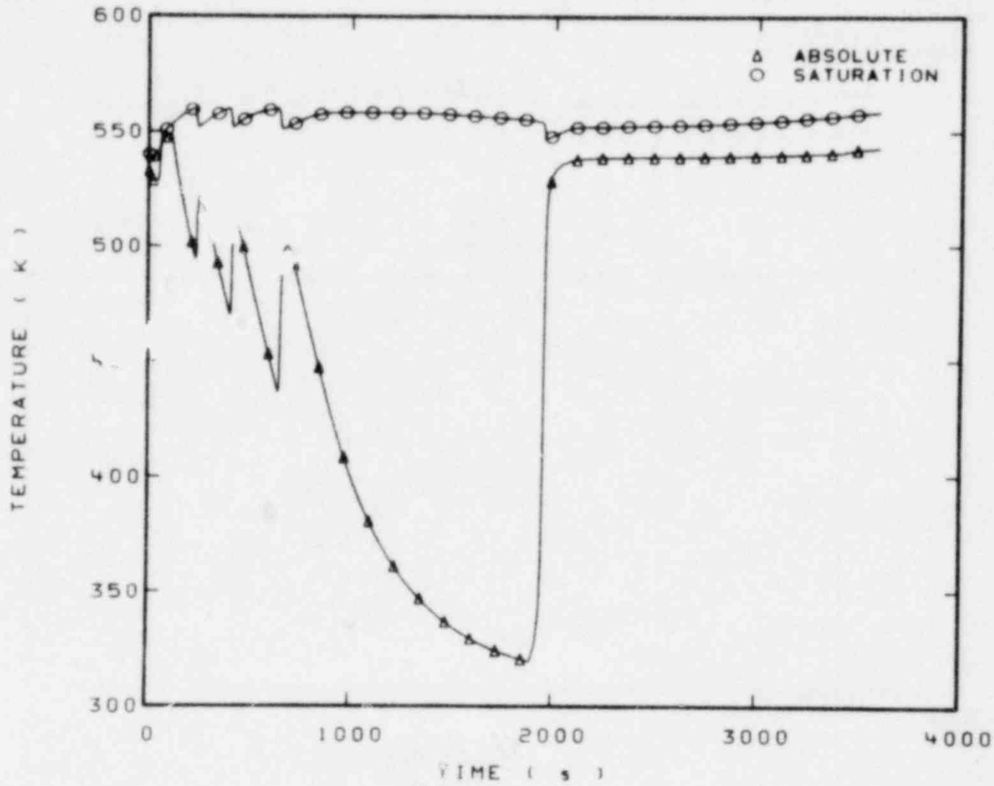


Figure 21. RELAP4 predicted steam generator downcomer fluid temperature and saturation temperature.

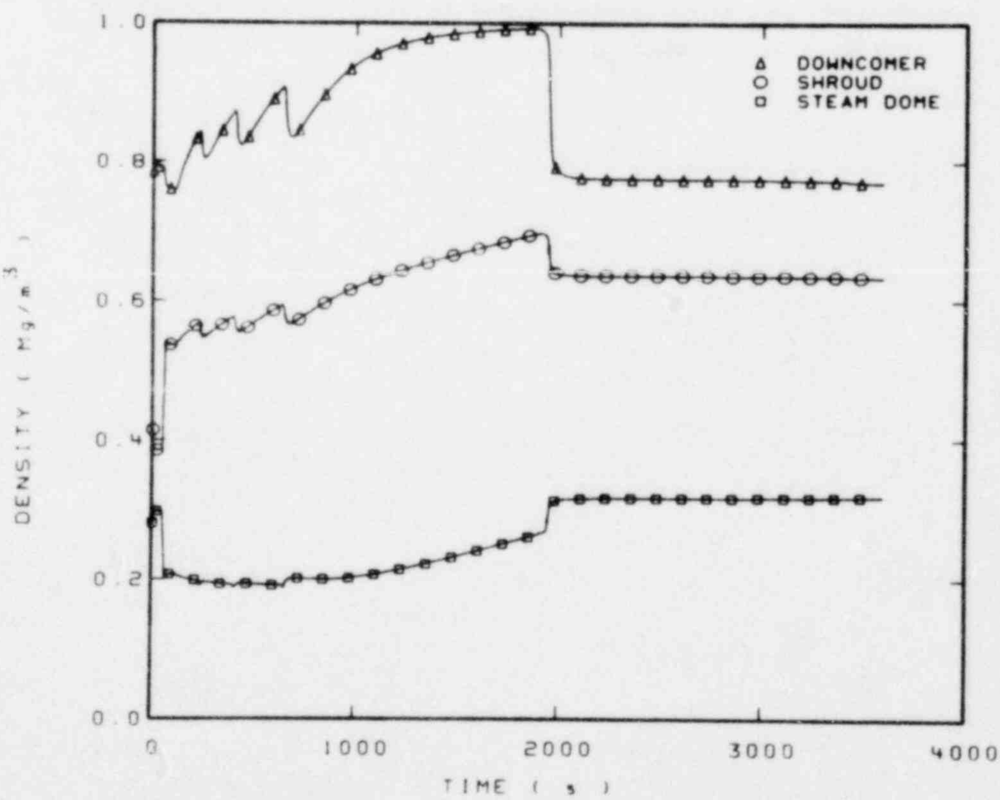


Figure 22. RELAP4 predicted steam generator secondary density distribution.

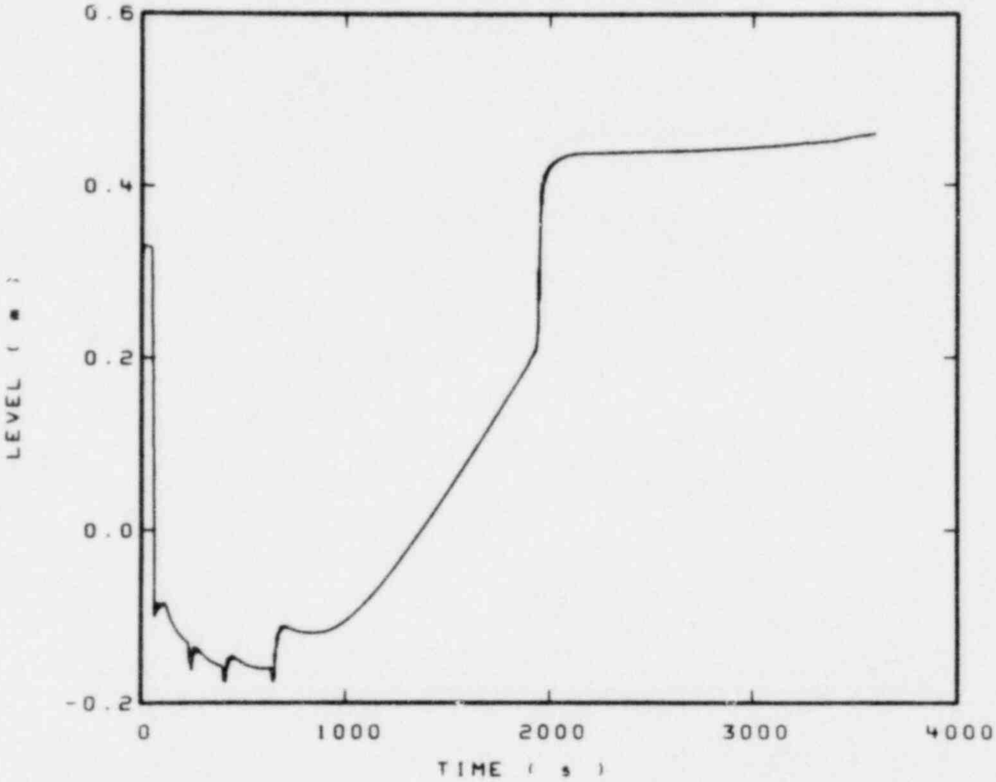


Figure 23. RELAP4 predicted steam generator downcomer liquid level.

3.2 RELAP5 Results

As in the RELAP4 calculation, the transient is initiated by opening the cold leg QOBV. For the first 94 s after experiment initiation, the pressure is predicted to decrease 1.41 MPa from the initial value, causing a reactor scram to initiate as shown in Figure 24. During the next 13 s the steam generator removes more energy from the primary system than the reactor core adds, resulting in a net stored energy loss in the primary loop. The resulting density increase shown in Figure 25 in the primary coolant places a further demand on the pressurizer, resulting in the high depressurization rate after 94 s. At 107 s, the steam flow control valve shuts completely, mitigating the rapid pressure decline in the primary system. At 127 s, HPIS is initiated by low pressure in the hot leg, but at about 400 s (see Figure 26) the pressurizer empties and the primary system rapidly goes to saturation pressure corresponding to the fluid temperature in the intact loop hot leg. The steam flow control valve is predicted to start opening at about 150 s. A small flow in the intact loop (see Figures 27 and 28) carries thermal waves, generated by the valve opening, throughout the system. These waves are of large enough amplitude to be measured at the PE-PC-1 location, see Figure 29. The steam valve is predicted to stay closed in the period between 1000 and 2000 s. The steam valve is predicted to start opening again at about 2000 s, reducing pressure to about 7.2 MPa. At this point, HPIS flow is about equal to break flow as shown in Figures 30 and 31.

Figure 32 shows the liquid level in the steam generator downcomer. The trends in these data indicate that the physical phenomena occurring in the steam generator during power transients are being modeled correctly by the RELAP5 code; however, the magnitude of the shrink and swell transients are not expected to be as large as calculated. Additionally, the manometer-type oscillations indicated after steam valve closure are not expected to occur. The LOFT RELAP5 input data set is being reviewed to obtain more realistic losses, flow areas, and volumes in the natural circulation reflux loop of the steam generator secondary. The steam generator pressure response is shown in Figure 33 and indicates that for the first 3600 s heat transfer will be from the primary loop into the steam generator with the steam generator pressure being controlled by the steam flow control valve at its open and shut setpoints.

After 1 h, steam is removed from the steam generator by opening the steam flow control valve bypass valve in such a manner to cause a cooldown in the steam generator secondary of 44.4 K per hour as shown on Figure 34. This energy removal causes a cooldown in the primary loop of 42.5 K per hour shown in Figure 35. The RELAP5 calculations therefore indicate that the steam generator cooldown will be effective in cooling down the primary system. After 1.1 h of cooldown, the auxiliary feed pump is turned on to fill the steam generator secondary. The addition of this cold water causes a cooldown rate in the steam generator greater than 44.4 K per hour. The steam flow control valve bypass valve is therefore shut whenever the cooldown exceeds 44.4 K per hour.

The liquid level in the reactor vessel upper plenum is shown in Figure 36. Since the top of the active core is at about 2.9 m, the core is not predicted to uncover, thus the cladding surface temperature response shown in Figure 37 is calculated to be benign.

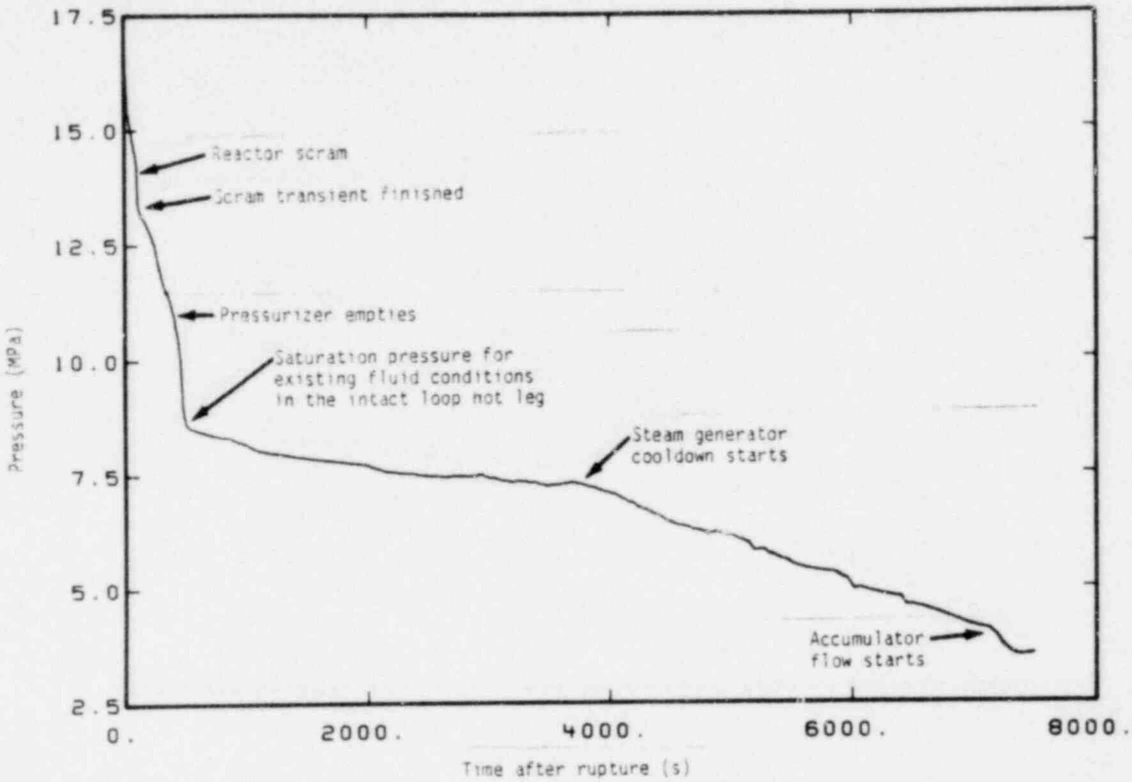


Figure 24. RELAP5 predicted upper plenum pressure response.

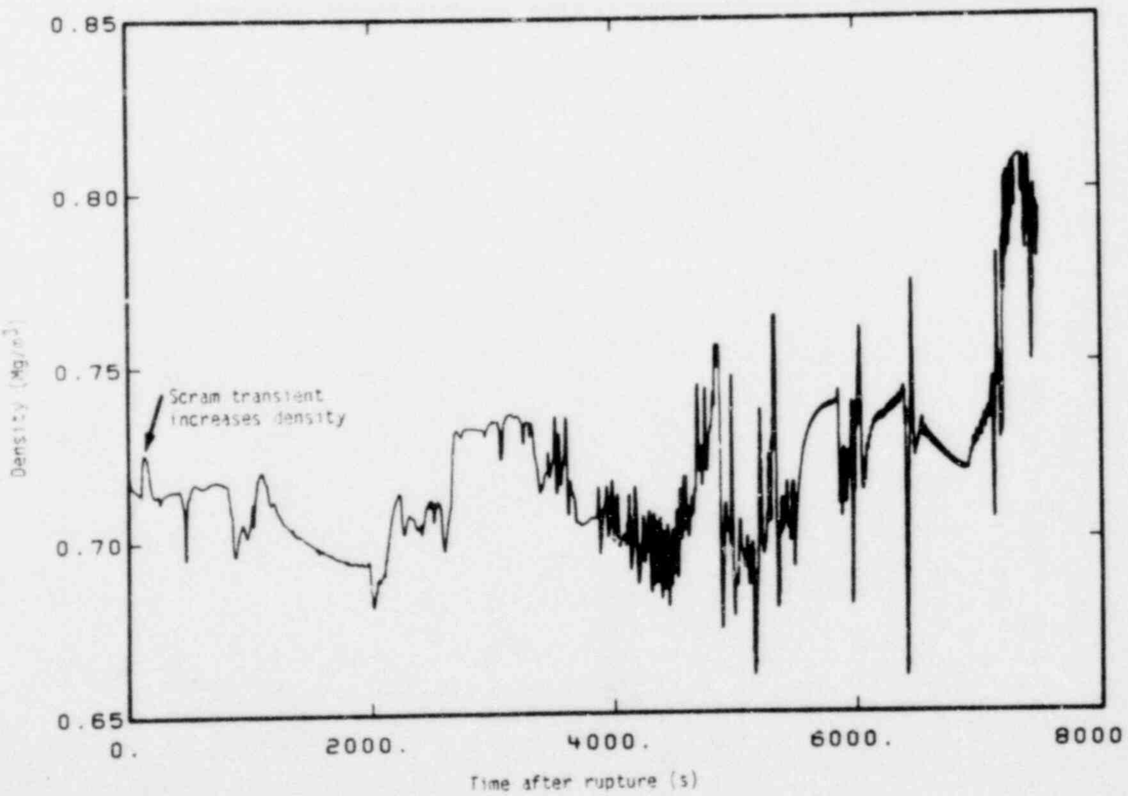


Figure 25. RELAP5 predicted intact loop hot leg density.

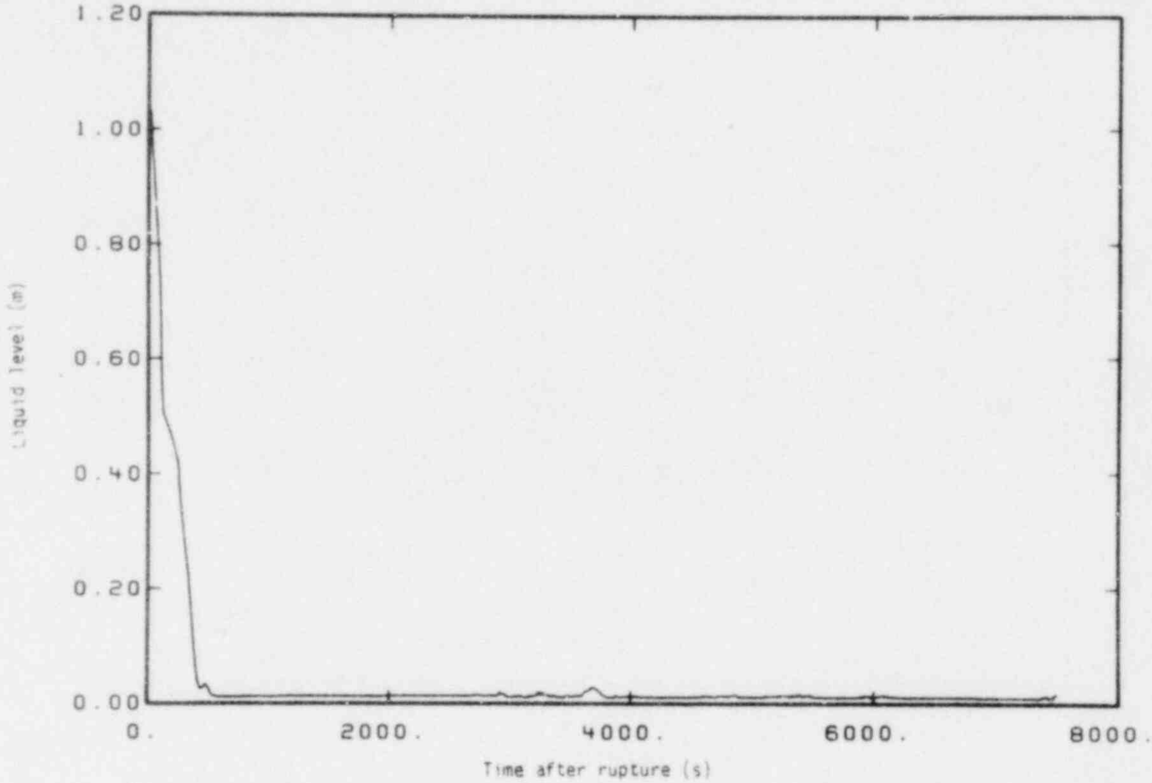


Figure 26. RELAP5 predicted pressurizer liquid level.

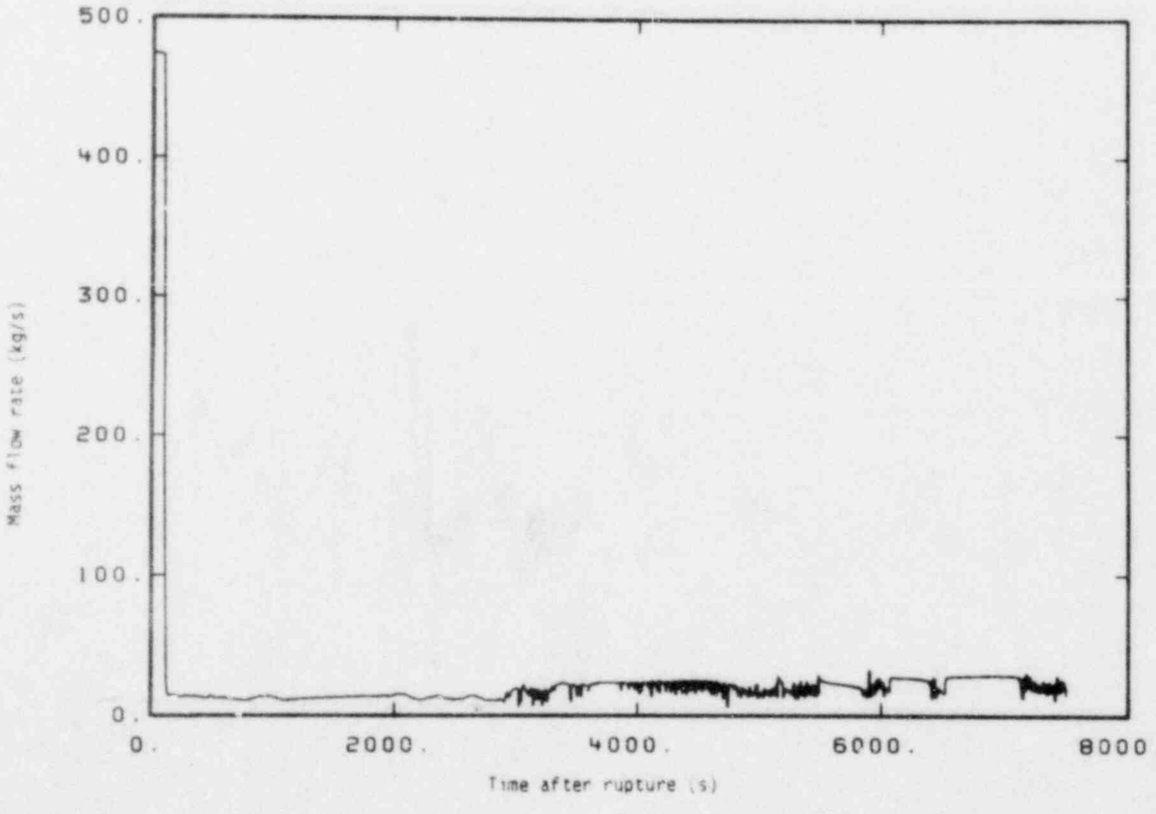


Figure 27. RELAP5 predicted intact loop cold leg mass flow.

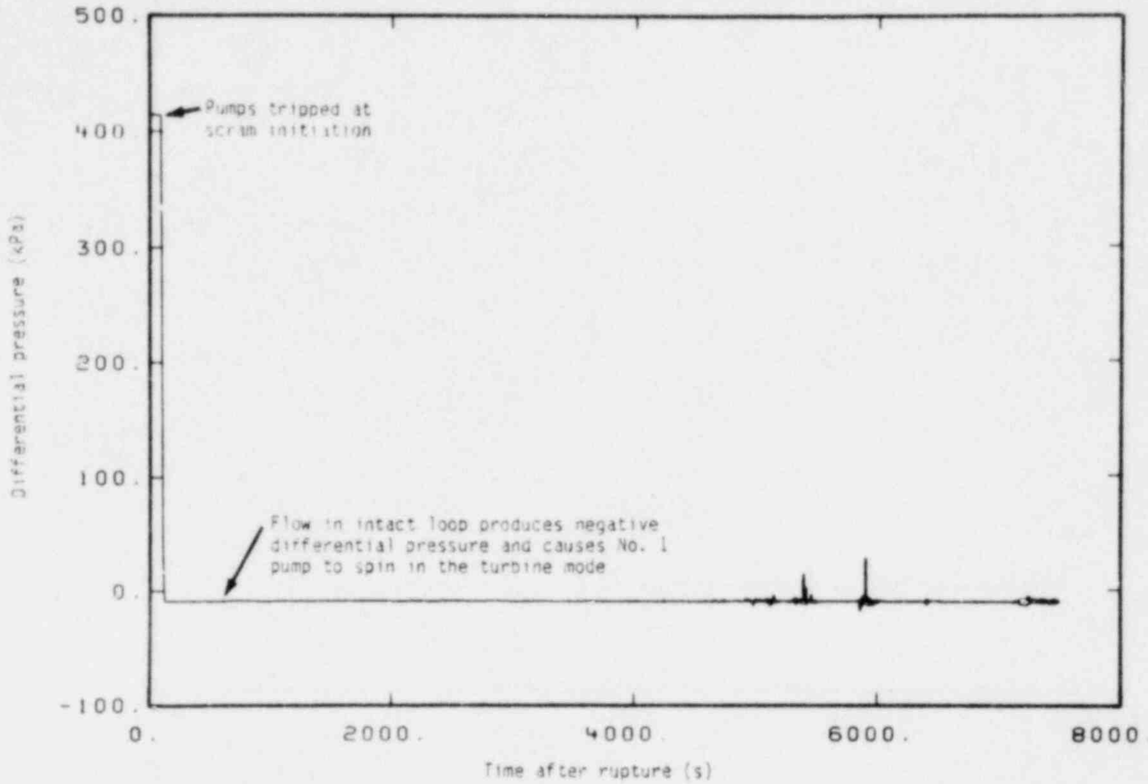


Figure 28. RELAP5 predicted pump differential pressure.

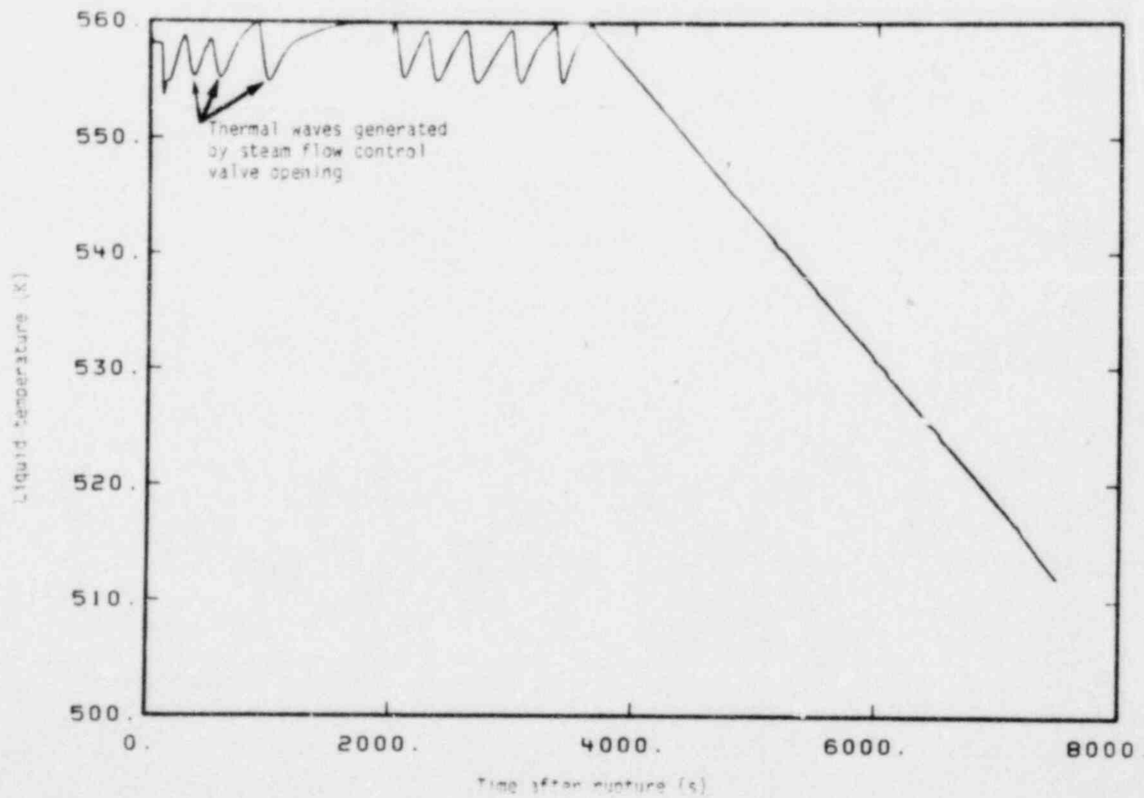


Figure 29. RELAP5 predicted intact loop cold leg fluid temperature.

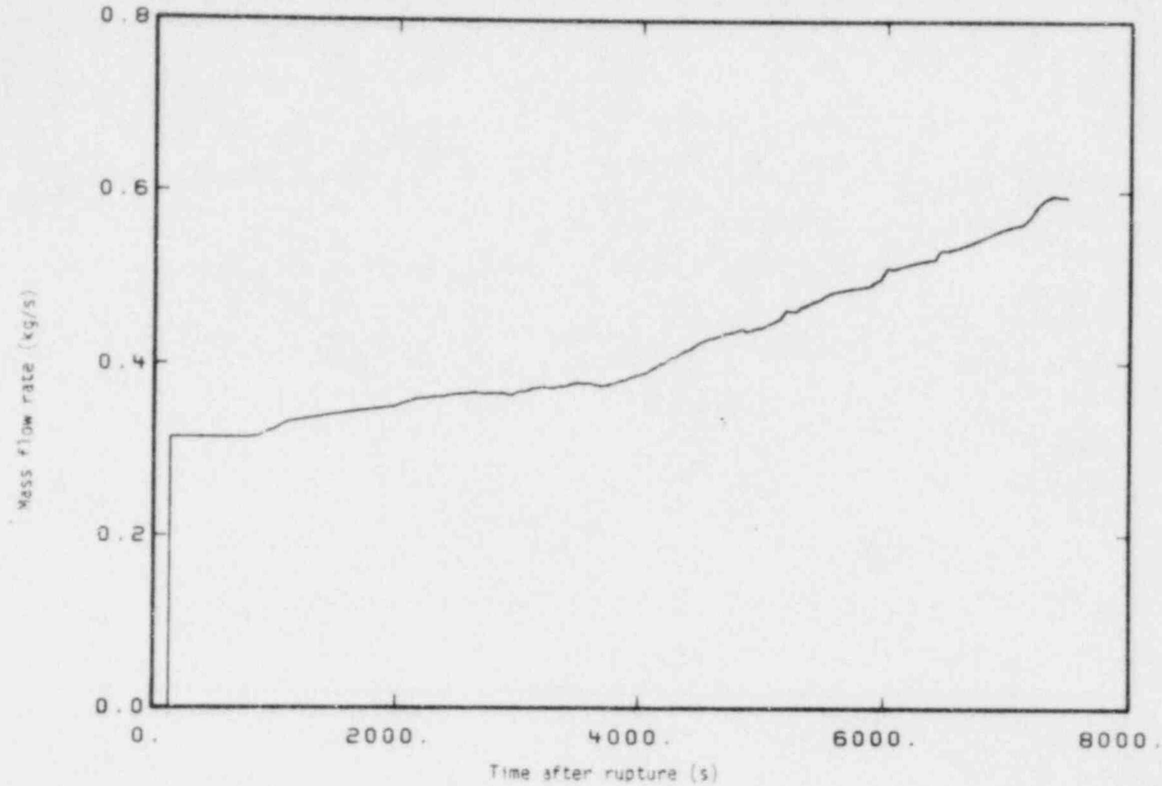


Figure 30. RELAP5 predicted HPIS flow.

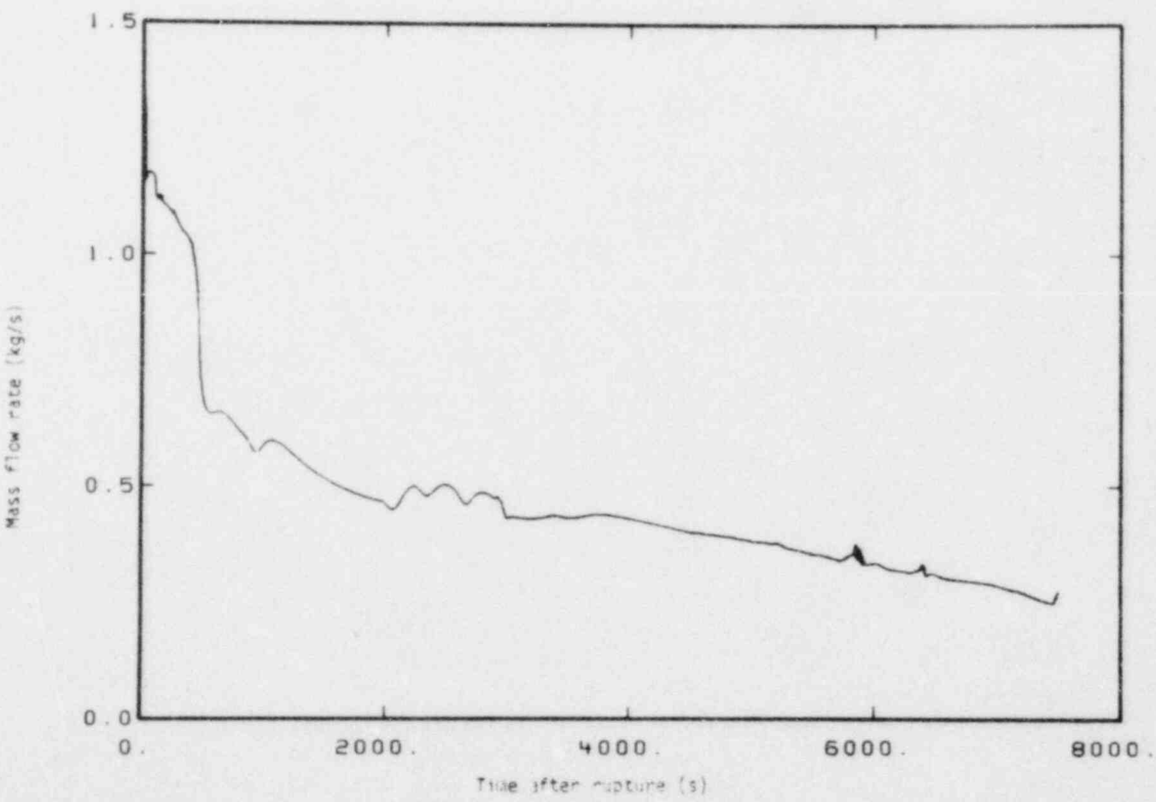


Figure 31. RELAP5 predicted break flow.

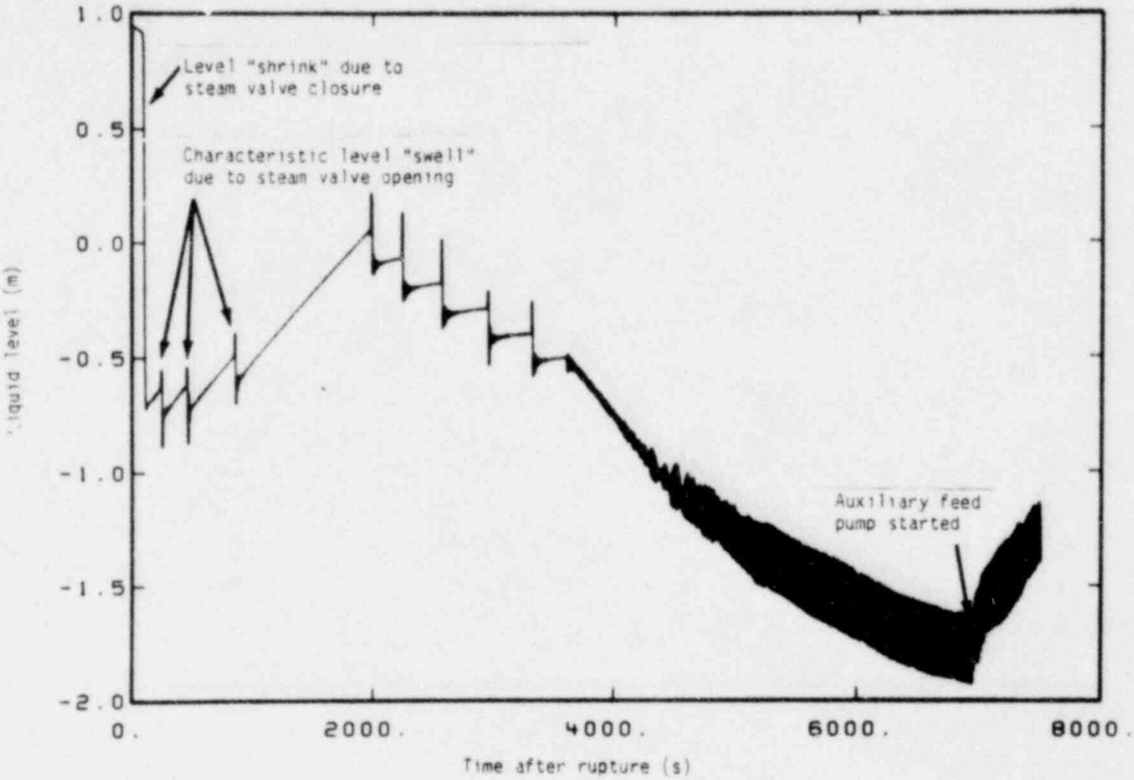


Figure 32. RELAP5 predicted steam generator downcomer liquid level.

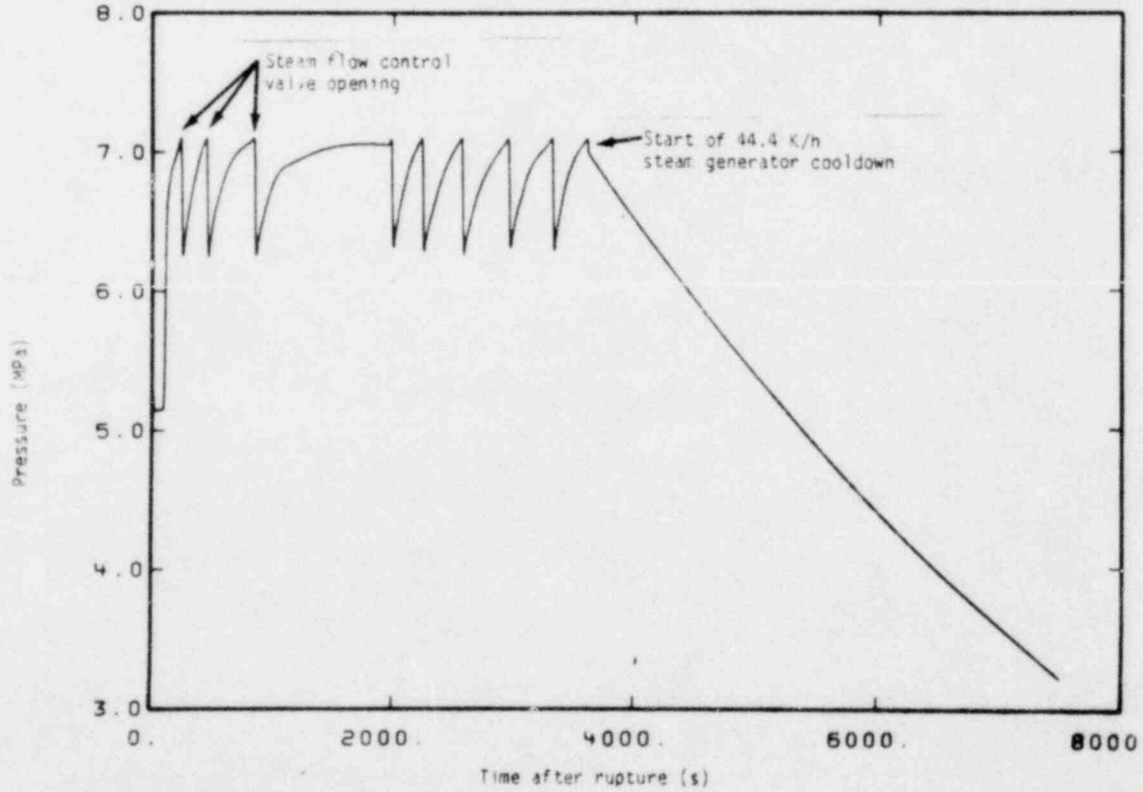


Figure 33. RELAP5 predicted steam generator secondary pressure.

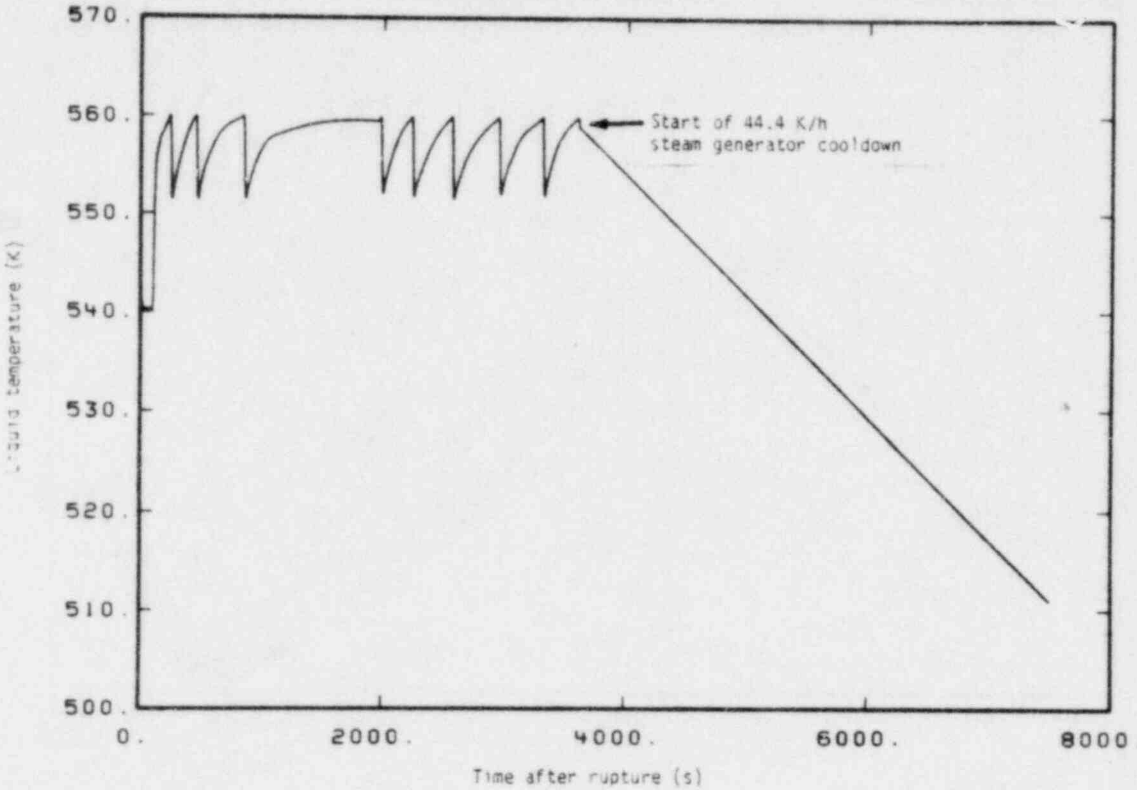


Figure 34. RELAP5 predicted steam generator secondary fluid temperature.

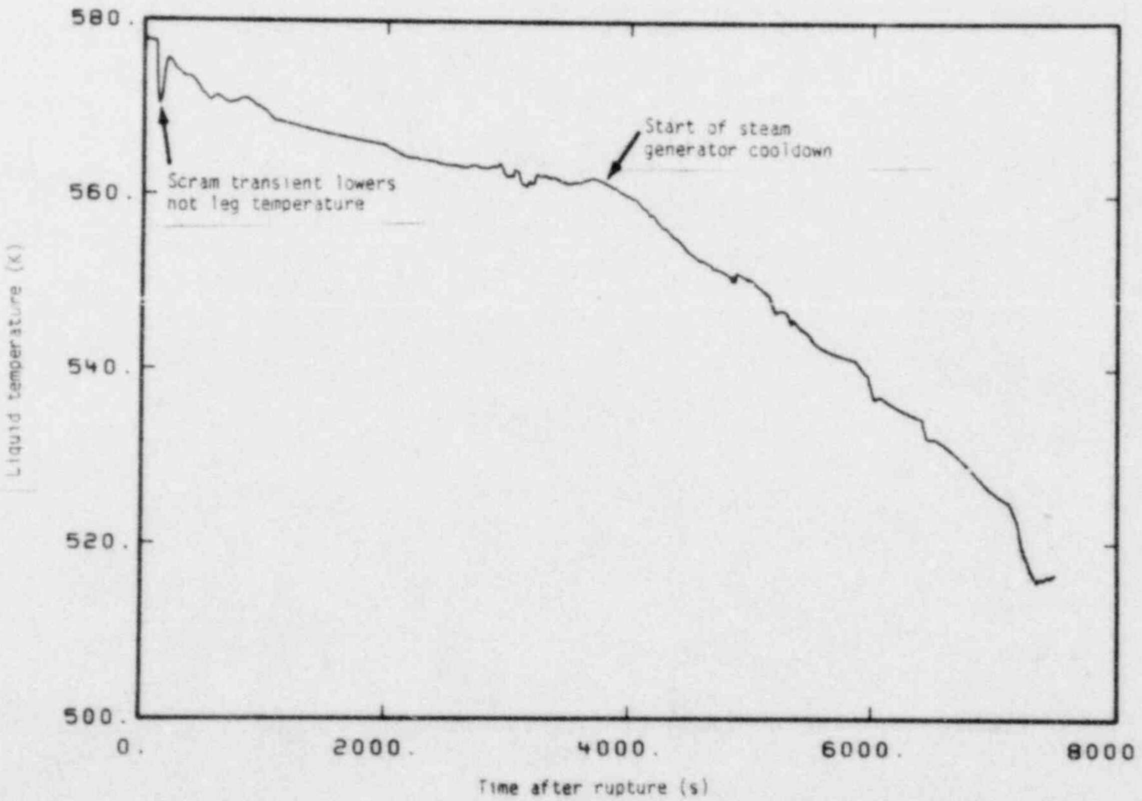


Figure 35. RELAP5 predicted intact loop hot leg fluid temperature.

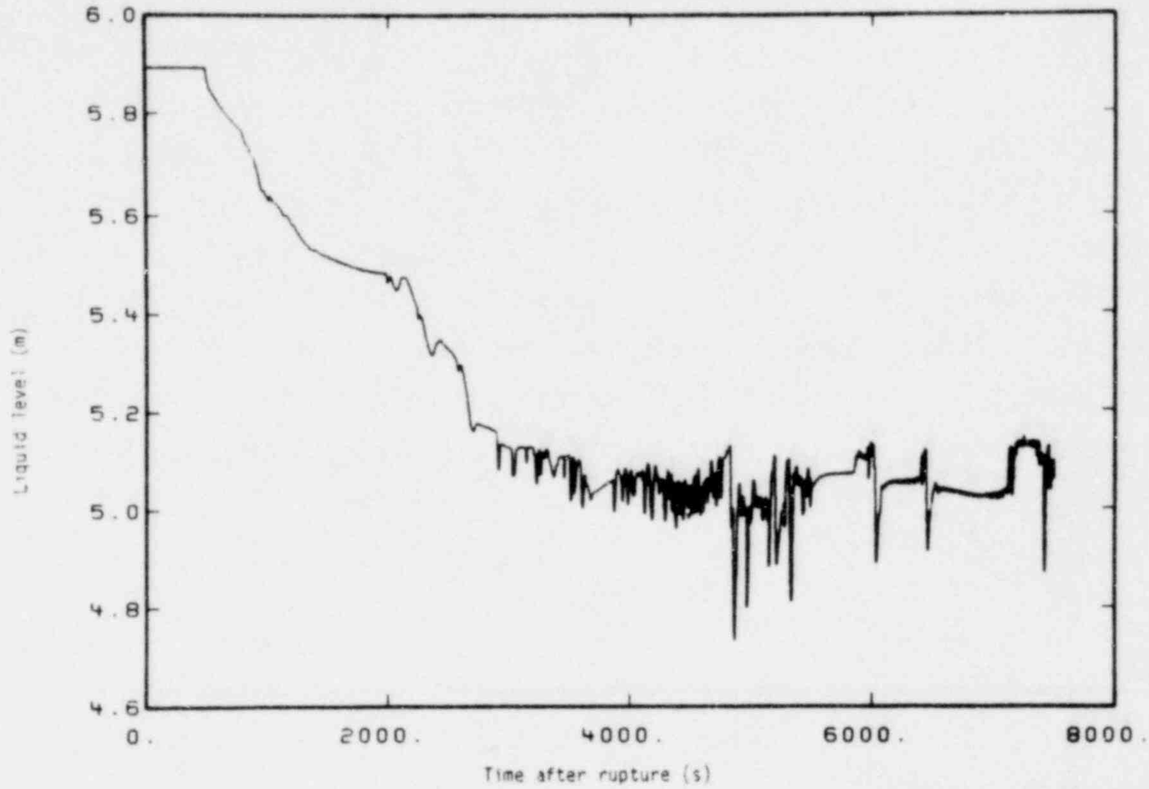


Figure 36. RELAP5 predicted reactor vessel collapsed liquid level.

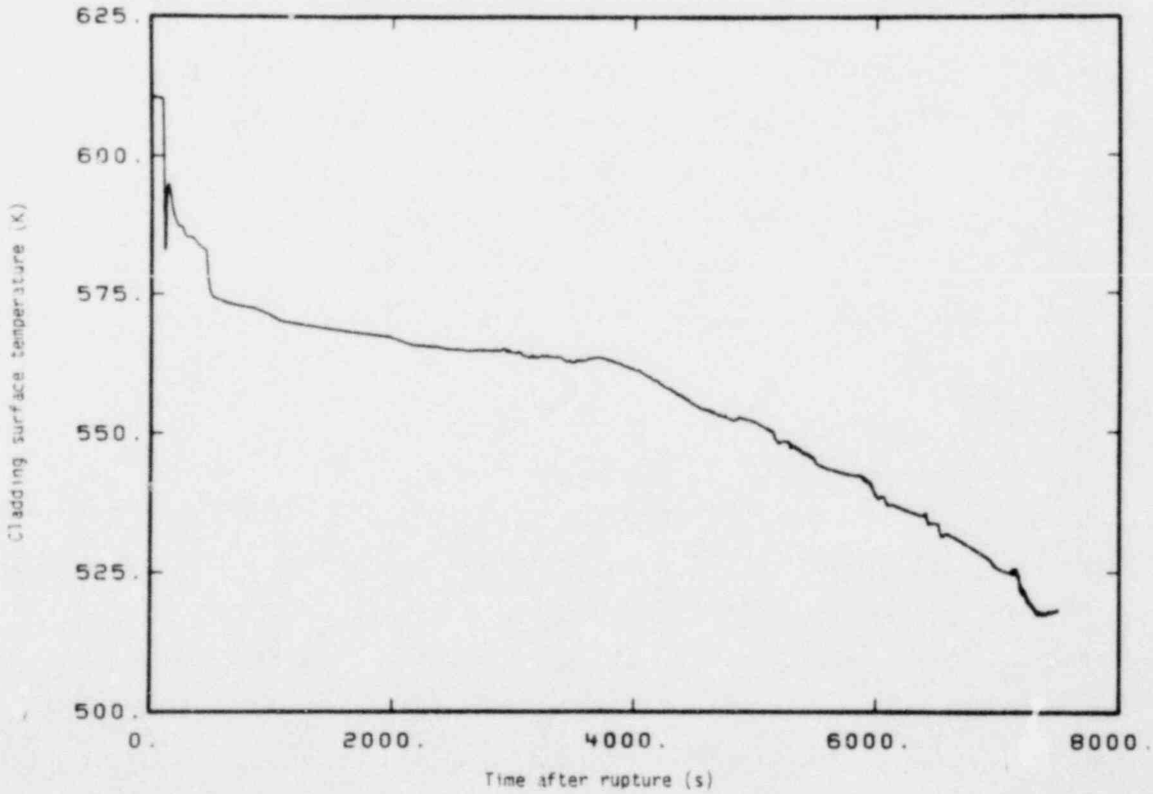


Figure 37. RELAP5 predicted cladding surface temperature.

4. RESULTS COMPARISONS AND CONCLUSIONS

This section addresses the major differences between the RELAP4 and the RELAP5 analyses and indicates the most probable sequence of events which are expected to occur during LOCE L3-2.

The primary system pressure responses for the two analyses are shown in Figure 38. The figure shows that the RELAP4 calculation depressurizes faster than the RELAP5 calculation and drops to a lower pressure before the system saturates. The repressurization calculated by RELAP4 to occur after 2500 s is also evident. The causes of these two areas of disagreement are discussed below.

The steeper initial depressurization predicted by RELAP4 is caused by a larger break flow being predicted by RELAP4 as shown in Figure 39. RELAP4 allows the user to select critical break flow models to be used for different break flow hydraulic conditions and also to specify break flow multipliers for these models. User input for RELAP5 is limited to a geometric description of the break orifice. Extensive testing of the break orifice configurations for Test Series L3 at the LOFT Test Support Facility and at Wylie Laboratories has provided data which were used to select the RELAP4 break flow models and appropriate multipliers. It is expected that the LOCE L3-2 break flow is more properly represented by the RELAP4 calculation. Both the RELAP4 and the RELAP5 calculations indicate that the break flow will be subcooled throughout the entire transient which reduces the number of break flow models required to describe the break flow.

The repressurization predicted in the RELAP4 calculation results from the vapor generated in the core being carried into the intact loop hot leg and blanketing the steam generator as discussed in Section 3. Figure 40 shows the intact loop hot leg densities from both the RELAP4 and RELAP5 calculations. The passage of the low density steam into the primary piping, as calculated by RELAP4, is obvious from this figure. Figure 41 shows both calculated densities in the cold leg piping. The sudden jump in density shown by the RELAP4 calculation reflects the density of the

HPIS flow indicating stoppage of primary coolant flow. The loop seal effectively separates the steam from the liquid.

The differences between the RELAP4 and RELAP5 responses during this period are attributed to the 5% core bypass flow modeled in RELAP5 which is not modeled in RELAP4. With the bypass included, the vapor generated in the core has an additional flow path apart from the intact loop piping from the upper plenum to the break. The exact bypass flow is not known nor is the effect of the bypass completely understood. Additional computer runs using both the RELAP4 and RELAP5 codes to understand the bypass effects are being performed but are not yet available.

The effectiveness of the manual steam generator feed and bleed depends on the system condition at 3600 s. If the system depressurizes as shown in the RELAP5 calculation, then the feed and bleed will be very effective in depressurizing the primary system. If the steam generator primary is steam bound as shown in the RELAP4 calculations, the primary and secondary are effectively decoupled and the feed and bleed procedure will not reduce the primary system pressure until the vapor in the steam generator can be condensed and natural circulation reestablished.

It is expected that the actual transient sequence will follow that predicted by RELAP4 during the first 2000 s, but will continue to depressurize as predicted by RELAP5. Based on this expected sequence, the stated objective of LOCE L3-2 will be met as follows:

1. The system will depressurize to about 7.5 MPa and stabilize with HPIS flow and break flow about equal.

2. The primary system can be effectively and safely cooled by the bleed and feed procedure in the secondary side.

3. The core will remain completely covered throughout the transient.

4. Natural circulation will be established in the intact loop at a flow rate of about 15 kg/s.

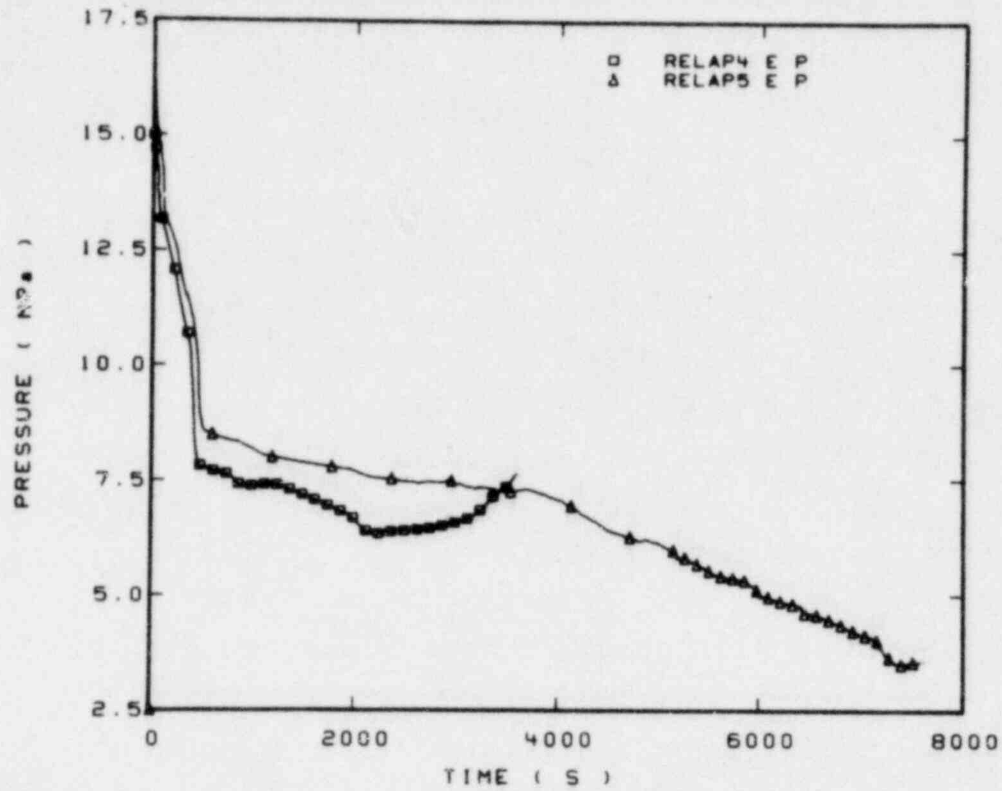


Figure 38. Comparison of RELAP4 and RELAP5 predicted upper plenum pressure.

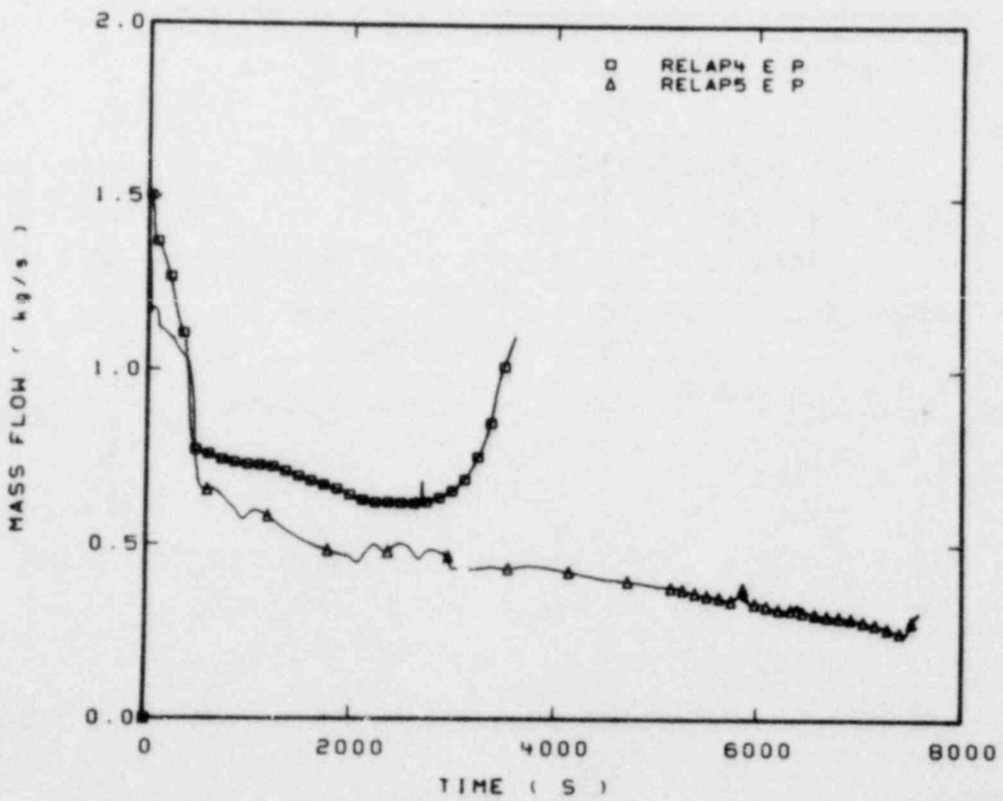


Figure 39. Comparison of RELAP4 and RELAP5 predicted break mass flow.

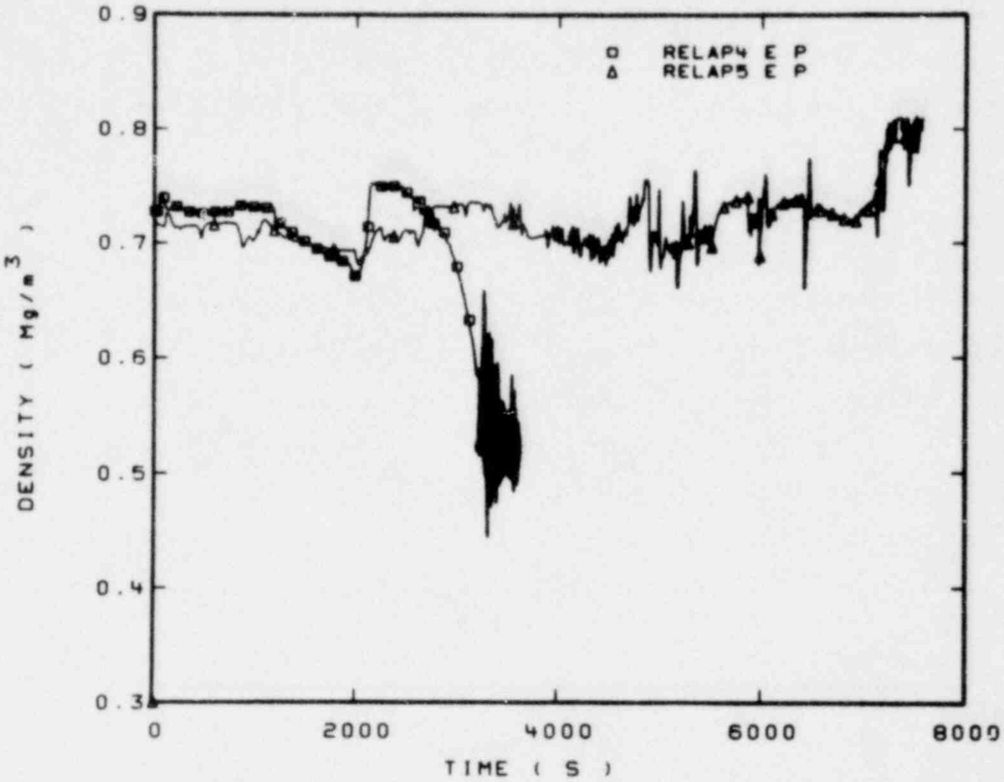


Figure 40. Comparison of RELAP4 and RELAP5 predicted intact loop hot leg density.

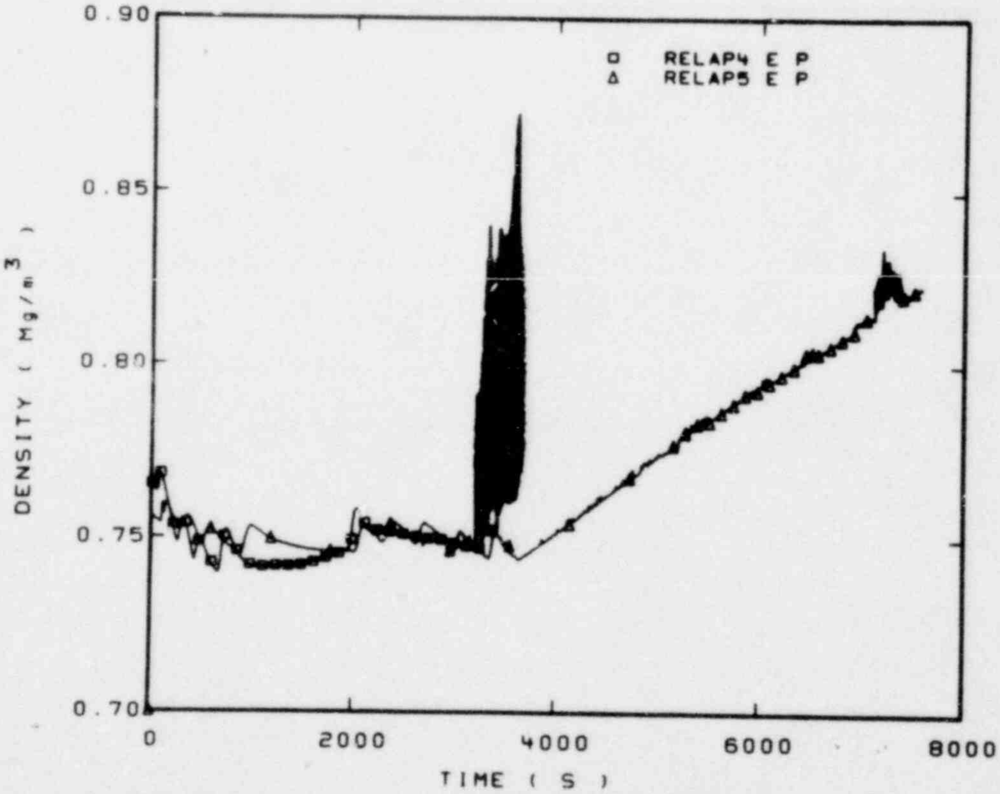


Figure 41. Comparison of RELAP4 and RELAP5 predicted intact loop cold leg density.

Data from the experiment prediction calculations have been made available to LOFT Facility personnel so that the sequence and time of major events can be incorporated into the LOCE L3-2 operating procedures and instrument ranges can be adjusted as necessary.

Data from the experiment prediction calculations have been reduced and incorporated into the LOFT Data Base to facilitate comparison with the experimental data when it becomes available. Through these comparisons, the capability of the modeling techniques can be evaluated.

5. REFERENCES

1. H. J. Welland, LOFT Experiment Operating Specification, "Small Break Test Series L3," NE L3 Series, Rev. 2, February 1980.

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

3. G. W. Johnsen et al., RELAP4/MOD7 (Version 2) User's Manual, CDAP-TR-78-036, August 1978.

4. W. H. Grush and M. S. Shinko, Best Estimate Prediction for LOFT Nuclear Experiment L3-1, EGG-LOFT-5033, November 1979.

5. E. J. Kee and W. H. Grush, Best Estimate Prediction for LOFT Nuclear Experiment L2-3, EP-L2-3, April 1979.

6. V. H. Ransom, et al, RELAP5/MOD"0" Code Description, CDAP-TR-057, May 1979.

APPENDIX A

DETAILED TEST PREDICTION DATA FOR LOFT LOCE L3-2

APPENDIX A

DETAILED TEST PREDICTION DATA FOR LOFT LOCE L3-2

This appendix provides detailed prediction data for LOCE L3-2. The data plots showing parameters listed in Table A-1 are presented on microfiche in the pouch attached on the inside of the report back cover. The microfiche are identified as APPENDIX A -- PREDICTION DATA, and the plots appear in the order they are presented in Table A-1.

TABLE A-1. DETAILED TEST PREDICTION DATA

Parameter	Title
Average Density	
DE-BL-1	AVERAGE DENSITY - BROKEN LOOP CL
DE-BL-2	AVERAGE DENSITY - BROKEN LOOP HL
DE-PC-1	AVERAGE DENSITY - INTACT LOOP CL
DE-PC-2	AVERAGE DENSITY - INTACT LOOP HL
DE-PC-3	AVERAGE DENSITY - INTACT LOOP SG OUT
Mass Flow Rate	
FR-BL-1	MASS FLOW - AT STATION BL-1
FT-P4-12	MASS FLOW - STEAM
FT-P4-22A	MASS FLOW - FEEDWATER
Volumetric Flow Rate	
FT-P120-36-5	VOLUMETRIC FLOW - ACCUMULATOR
FT-P120-85	VOLUMETRIC FLOW - LPIS
FT-P128-104	VOLUMETRIC FLOW - HPIS
Collapsed Liquid Level	
LC-3UP-1	COLLAPSED LIQUID LEVEL - UPPER PLENUM
Mixture Level	
LE-3UP-1[a]	LIQUID LEVEL - UPPER PLENUM
LE-3F10[a]	COOLANT LEVEL - FUEL ASSY 3 LOC F10
LT-P4-8B	LIQUID LEVEL - SCS SG SECONDARY
LT-P139-7	LIQUID LEVEL - PRESSURIZER CH B

TABLE A-1. (continued)

Parameter	Title
Differential Pressure	
PdE-PC-1	DELTA P - PRIMARY COOLANT PUMP
PdE-PC-2	DELTA P - INTACT LOOP SG
PdE-PC-6	DELTA P - REACTOR VESSEL IL CL TO HL
Pressure	
PE-BL-1	PRESSURE - BROKEN LOOP COLD LEG
PE-BL-2	PRESSURE - BROKEN LOOP HOT LEG
PE-PC-1	PRESSURE - INTACT LOOP COLD LEG
PE-PC-2	PRESSURE - INTACT LOOP HOT LEG
PE-PC-4	PRESSURE - INTACT LOOP PRESSURIZER
PE-1UP-1A	PRESSURE - UPPER END BOX
PT-P4-10A	PRESSURE - SCS 10 INCH LINE FROM SG
PT-P120-43	PRESSURE - ECCS ACCUMULATOR A
Pump Speed	
RPE-PC-1[a]	PUMP SPEED - PRIMARY COOLANT PUMP 1
Temperature	
TE-BL-1	COOLANT TEMP - BROKEN LOOP CL
TE-BL-2	COOLANT TEMP - BROKEN LOOP HL
TE-PC-2	COOLANT TEMP - INTACT LOOP HL
TE-SG-3	COOLANT TEMP - SGS DOWNCOMER
TE-P139-20	COOLANT TEMP - PRESSURIZER LIQUID
TE-1ST-4	COOLANT TEMP - RV INSTR STALK 1 DC
TE-1ST-14	COOLANT TEMP - RV INSTR STALK 1 DC
TE-2LP-1	COOLANT TEMP - FA2 LOWER END BOX
TE-2UP-1	COOLANT TEMP - UPPER END BOX
TE-3UP-8	COOLANT TEMP - FA3 AT LLT
TE-2G14-11	CLADDING TEMP - FUEL ASSEMBLY 2
TE-2G14-30	CLADDING TEMP - FUEL ASSEMBLY 2
TE-2G14-45	CLADDING TEMP - FUEL ASSEMBLY 2

a. RELAP4 only.

APPENDIX B

UNITS CONVERSION OF RELAP4 AND RELAP5 DATA

APPENDIX B

UNITS CONVERSION OF RELAP4 AND RELAP5 DATA

This appendix describes in detail how the data output from the RELAP4 and RELAP5 computer codes is 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

RELAP4 AND RELAP5 UPDATE INPUT DATA

APPENDIX C

RELAP4 AND RELAP5 UPDATE INPUT DATA

A listing of the input data for updating RELAP4 and for 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 original and updated RELAP4 object decks and for the RELAP5 source deck and update input data deck used in this prediction analysis are as follows:

1. The RELAP4/MODG, Version 92 object deck is stored under Configuration Control Number H00718B.

2. The updated object deck (file name RLP4G92LFT04) is stored under Configuration Control Number H01168IB.

3. The RELAP4 preload program is stored under Configuration Control Number H01037IB.

4. The RELAP4 input deck is stored under Configuration Control Number H006185B, PFN L32GONA, Cycle=1, ID = MST.

5. The RELAP5/MOD"0" source deck is stored under Configuration Control Number H005785B.

6. The RELAP5/MOD"0" update input data deck is stored under Configuration Control Number H005985B.

7. The RELAP5 input deck is stored under Configuration Control Number H005985B.

APPENDIX D

RELAP4 AND RELAP5 INPUT MODELS

APPENDIX D

RELAP4 AND RELAP5 INPUT MODELS

The input and time zero edit listing for the RELAP4 and RELAP5 models are provided on microfiche in the pouch on the inside of the report back cover.

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