EG&G Idaho, Inc

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

INTERIM REPORT

Accession No. ____________

Report No. EGG-LOFT-5161

Contract Program or Project Title:

LOFT Program Office

Subject of this Document:

Best Estimate Prediction for LOFT Nuclear Experiments L6-1, L6-2, L6-3, and L6-5

Type of Document:

Experiment prediction analysis report

Author(s):

Clinton D. Keeler

Date of Document:

October 1980

Responsible NRC Individual and NRC Office or Division:

G. D. McPherson, Chief, LOFT Research Branch,
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

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

INTERIM REPORT

BEST ESTIMATE PREDICTIONS FOR LOFT NUCLEAR EXPERIMENTS L6-1, L6-2, L6-3, AND L6-5

By:

Clinton D. Keeler

Approved:

J. H. Linebarger, Manager
Experiment Evaluation Branch

C. W. Solbrig, Manager
LOFT Program Office

The LOFT Subcommittee of the EG&G Pretest Prediction Consistency Committee has reviewed the RETRAN01/MOD2 model and predicted results for LOFT Small Break Experiments L6-1, L6-2, L6-3, and L6-5.

DH Rawson / S. Bohling
Code Development

CB Davis for T R Charlton
Code Assessment

C B Forman
Semiscale

Thermal Fuels

ABSTRACT

Best estimate prediction analyses were performed for the anticipated transient experiments in Loss-of-Fluid Test (LOFT) Test Series L6 using the RETRAN01/MOD2 transient thermal-hydraulic computer code. The code was used to simulate the LOFT facility during anticipated transient Experiments L6-1 (loss of steam load experiment), L6-2 (loss of primary coolant flow experiment), L6-3 (excessive load increase experiment), and L6-5 (loss of feedwater experiment). Simulation included modeling of automatic control components such as feedwater control and steam control valves, pressurizer heaters, and pressurizer spray, in addition to the thermal-hydraulic components in the reactor system. Each analysis simulated the first 200 s of each experiment. The results seem reasonable and indicate that the experiment objectives will be met.

NRC FIN No. A6048 - LOFT Experimental Program

SUMMARY

This document contains the prediction of the coupled system thermal-hydraulic response for the Loss-of-Fluid Test (LOFT) system during anticipated transient Experiments L6-1, L6-2, L6-3, and L6-5. These experiments are the first four experiments in the LOFT Test Series L6 (non-LOCE anticipated transient experiments). The computer predictions were performed using RETRAN01/MOD2.

Experiment L6-1 (loss of steam load experiment) will be initiated by closing the main steam control valve at its maximum rate. Steam generator secondary pressure and temperature will rise due to the loss of steam flow. Primary coolant system pressure will correspondingly rise. The analysis shows the reactor will be scrammed at 12 s after experiment initiation, when the high-pressure scram setpoint of 15.72 MPa (2281 psia) is reached in the primary coolant system. The steam control valve will automatically cycle from 27 to 55 s to relieve steam generator pressure. The analysis was terminated at 200 s.

Experiment L6-2 (loss of primary coolant flow) will be initiated by tripping power to the primary coolant pump motor generator sets, allowing the pumps to coast down under the influence of the flywheel system. The reactor is calculated to scram almost immediately (1.5 s) after experiment initiation when the low-flow scram setpoint [433 kg/s (952.78 lbm/sec)] is reached in the primary coolant system. Concurrent with the scram signal, the feedwater control valve and main steam control valves will begin to close. After these valves close, pressure in the steam generator secondary side will begin to rise. Temperature of the primary coolant will show an average decrease due to scram until 26 s after experiment initiation when it will start to rise. Pressure in the primary coolant system will rise after 26 s due to a pressurizer insurge. The analysis was terminated at 200 s.

Experiment L6-3 (excessive load increase) will be initiated by opening the main steam control valve at its maximum rate. Opening of the valve will cause pressure in the steam generator secondary side to drop and the

steam flow rate to increase. A rise in steam generator downcomer liquid level will cause the feedwater control valve to start closing. At 12 s after experiment initiation heat transfer across the steam generator tubes is calculated to peak. Steam flow rate, a function of both steam control valve area and secondary pressure, is calculated to peak at 17 s. The increased heat transfer through the steam generator will cause the primary coolant system to cooldown and reactor power to rise due to reactivity feedback. Core power is calculated to rise from 37.5 to 44.2 MW. Reactor power is calculated to peak at 17 s, followed by a decrease to around 38.5 MW at 200 s into the transient. The analysis was terminated at 200 s.

Experiment L6-5 (loss of feedwater experiment) will be initiated by tripping the feedwater pump at time zero and closing the feedwater regulating valve. The reactor will be scrammed when steam generator downcomer liquid level drops to 2.82 m (111 in.) above the top of the tube sheet. The reactor scram was calculated to occur at 23 s. Upon scram, the steam control valve will start to close. Closing of this valve will cause pressure to rise in the steam generator secondary side. The steam control valve cycled from 107 to 146 s in the analysis to relieve secondary pressure. From 0 to 23 s, the primary coolant system temperature increased, resulting in an insurge into the pressurizer with the attending pressure rise. Reactor scram caused a drop in primary coolant temperature and a rapid outsurge from the pressurizer. Later in the transient, primary coolant temperature rose again, with an attending insurge into the pressurizer and rise in pressure. The analysis was terminated at 200 s.

It is felt that these four experiments will provide a much-needed data base for developing and assessing anticipated transient codes such as RETRAN01/MOD2 and RELAP5. The experiments will also allow evaluation of modeling techniques.

ACKNOWLEDGMENTS

The author gives recognition to R. D. Hentzen, W. [illegible]. Choe, and G. F. Niederauer of Energy Incorporated, Idaho Falls, Idaho, for their work in the initial set-up of the LOFT RETRAN01/MOD2 Model. Their pretest prediction of LOFT Experiment L6-5 is included in this report.

CONTENTS

FIGURES

TABLES

BEST ESTIMATE PREDICTIONS FOR LOFT NUCLEAR EXPERIMENTS L6-1, L6-2, L6-3, and L6-5

1. INTRODUCTION

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

This document describes how the RETRAN01/MOD2 computer code was used to simulate and predict the LOFT system responses and presents predicted results for Experiments L6-1, L6-2, L6-3, and L6-5 in Test Series L6, which was designed to study anticipated transients in pressurized water reactors (PWR). Sections 1.1 and 1.2 of this introduction discuss the experiment objectives and provide a brief description of the experiments and of the LOFT facility, respectively. Section 2 contains a description of the modeling techniques employed in this EP analysis. 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 configuration control information (Appendix A), detailed calculational results (Appendix B), algorithms for generation of the EP data in the data bank (Appendix C), and listings of the code inputs (Appendix D).

1.1 Experiment Objectives and Descriptions

Test Series L6 was designed to study anticipated transients in which a disturbance to plant equilibrium occurs, resulting in a reactor scram when and if the first safety system setpoint is reached. Reference 1 discusses the experiment objectives for Test Series L6 and describes Experiments L6-1, L6-2, L6-3, and L6-5 in detail. These experiment objectives and descriptions are summarized in the following sections.

1.1.1 Experiment Objectives

The experiment objectives for Test Series L6 are:

1. To provide data required for evaluation of the plant and control systems performance during each of the anticipated transient experiments.

2. To determine the important thermal, hydraulic, operational, and neutronic phenomena during an anticipated transient at the LOFT facility and to identify any unexpected behavior.

3. To provide data to evaluate reactor transient analysis techniques used to analyze anticipated transients.

4. To provide data to assist in analyzing the relationship between behavior in LOFT and in a large PWR during anticipated transients.

5. To determine the effectiveness of instruments normally provided in large PWRs for identifying anticipated transients and monitoring the resulting plant response.

6. To determine what additional information and/or measurements would assist a plant operator in his diagnosis and/or control of an anticipated transient.

7. To continue development and testing of the operational diagnostic and display system (ODDS) by operation of the ODDS during each experiment.

It is expected that the experiments in Test Series L6 will display the same trends as expected in a large PWR, but the magnitudes and response times will be slightly different because of scaling considerations. Piping resistance and total thermal losses are higher for LOFT than would be for a large PWR (scaled). Valve actuation times for feedwater and steam control are slower for LOFT than in a large PWR, and only a single steam generator is used in LOFT. These differences make events initiated by the secondary coolant system in LOFT similar to large PWRs only when uniform disturbances in all steam generators in a large PWR are assumed to occur.

Because of the above differences between LOFT and a large PWR, anticipated transient testing in LOFT will not serve as demonstration tests for large PWRs; however, the codes used to predict transient response can be assessed using LOFT data with a suitable LOFT analytical model. Also, significant LOFT deviations can be identified and evaluated, and modifications can be made if the deviations appear to cause undesireable behavior that would prevent extrapolation to large PWR behavior.

1.1.2 Experiment Descriptions

All experiments in Test Series L6 will be initiated from operating conditions and plant configurations simulating the conditions expected at a typical four-loop PWR should it undergo an anticipated transient. An important assumption made is that commercial power will not be lost during the transient, so all reactor support systems will be available.

All experiments in Test Series L6 will be initiated with the LOFT system operating at an initial maximum linear heat generation rate (MLHGR) of 39.37 kW/m (12 kW/ft), a primary coolant flow rate of 478.8 kg/s (3.8×10^6 lbm/hr), and coolant temperature at the intact loop inlet to the reactor vessel of 552.6 K (535°F).

In preparation for each experiment, the necessary hardware configurations will be established and the plant heated to normal operating temperature using heat from the primary coolant pumps. During the plant heat-up, instrument calibrations and checks will be performed. Anomalous measurements will be identified and corrections made.

After the normal operating temperatures are established in the primary coolant system, reactor criticality will be established and the power will be raised to the required power level (approximately 75% power) and held there until all required initial operating conditions are established and stabilized. The experiments will then be initiated by performing the specified initiating events discussed in the following subsections.

1.1.2.1 Experiment L6-1 Description. Experiment L6-1 will be a loss-of-steam load incident. The steam flow control valve on the steam generator will be closed at its maximum rate (5% stem movement/second) which will stop heat removal from the primary coolant system by the secondary coolant system, causing primary coolant temperature and pressure to increase until the reactor is scrammed. This experiment will simulate a turbine trip with loss of condenser vacuum in a large PWR.

Experiment L6-1 was designed to:

1. Investigate plant response to a transient in which the heat removal capabilities of the secondary system is significantly reduced

2. Evaluate the automatic recovery methods in bringing the plant to a hot standby condition

3. Provide data to evaluate computer code capabilities to predict secondary system initiated events.

1.1.2.2 Experiment L6-2 Description. Experiment L6-2 will be a loss-of-primary-coolant flow incident. The experiment will be initiated by tripping power to the primary coolant pump motor generator sets, allowing the pumps to coast down under the influence of the flywheel system. Reactor scram will be initiated on indication of low primary coolant flow. Upon scram, the steam flow control and feedwater valves will start closing. This experiment will simulate a loss of all forced coolant flow in a large PWR.

Experiment L6-2 was designed to:

1. Investigate plant response to a transient in which forced reactor coolant flow is lost

2. Obtain additional data on the natural loop circulation mode of co ing

3. Evaluate the automatic recovery methods in bringing the plant to a hot standby condition, without operation of the reactor coolant pumps

4. Provide data to assess computer code capabilities to predict primary system initiated events.

1.1.2.3 Experiment L6-3 Description. Experiment L6-3 will be an excessive-load-increase incident. The steam flow control valve will be opened at its maximum rate. Primary coolant temperature will start decreasing, adding positive reactivity. A reactor scram may occur either on indication of high power (approximately 51.5 MW) or low primary coolant pressure. This experiment will simulate an excessive, rapid-power-demand incident at a large PWR.

Experiment L6-3 was designed to:

1. Investigate plant response to a transient in which the heat removal capability of the secondary system is significantly increased

2. Provide continued evaluation on automatic recovery methods

3. Provide data to evaluate code capabilities for secondary system initiated events.

1.1.2.4 Experiment L6-5 Description. Experiment L6-5 will simulate a loss-of-feedwater incident. The experiment will be initiated by tripping the feedwater pump and closing the feedwater regulating valve. Since LOFT does not have a low steam generator water level trip, the reactor will be manually scrammed when the level reaches -0.13 m (-5 in.) [2.82 m (111 in.) above the top of the tube sheet].

Experiment L6-5 was designed to:

1. Investigate plant response to a transient in which the feedwater flow to the secondary system is stopped

2. Provide continued evaluation on automatic recovery methods

3. Provide continued data for assessment of code capabilities to predict secondary system initiated events.

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.

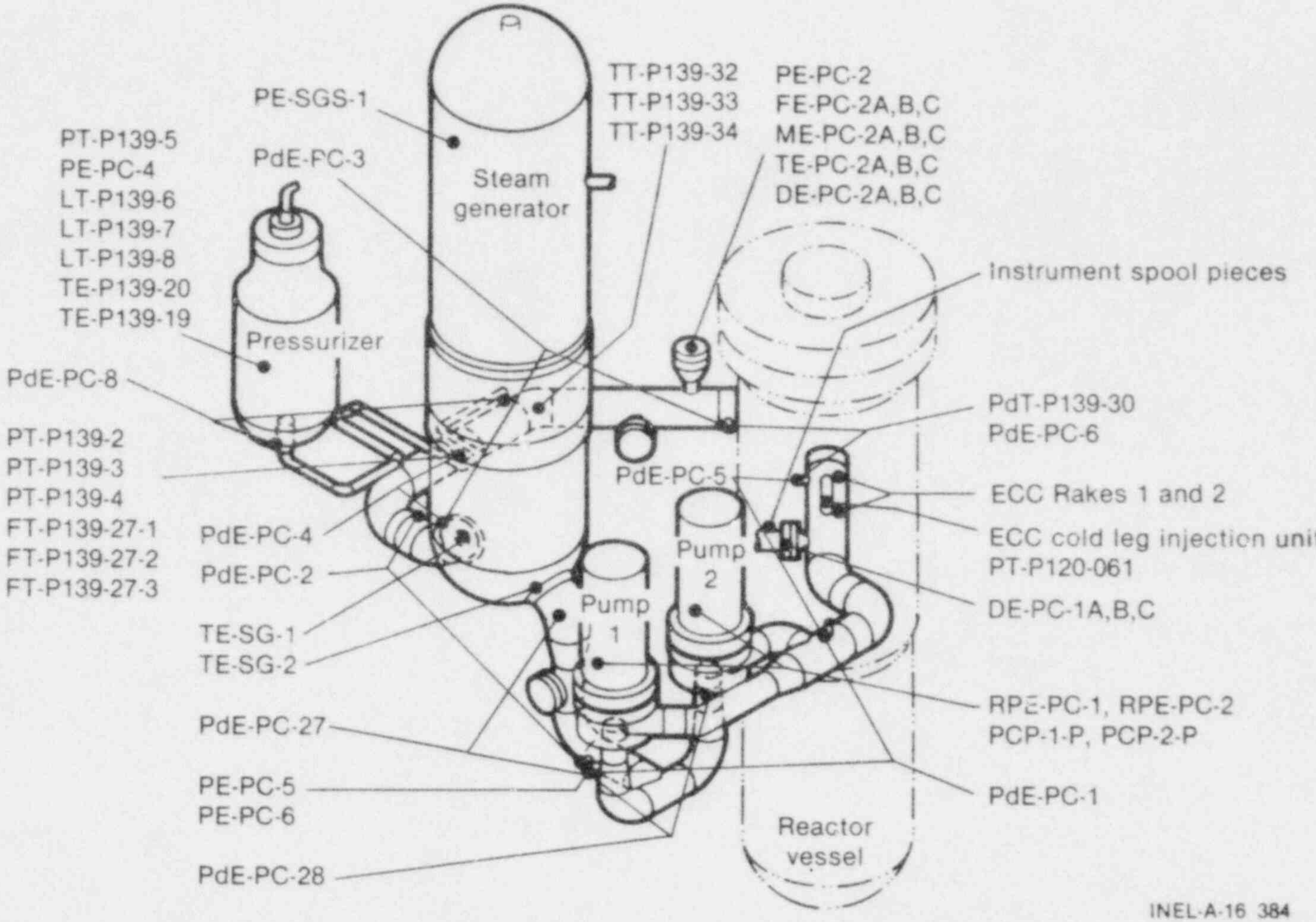


Figure 1. LOFT intact loop thermo-fluid instrumentation.

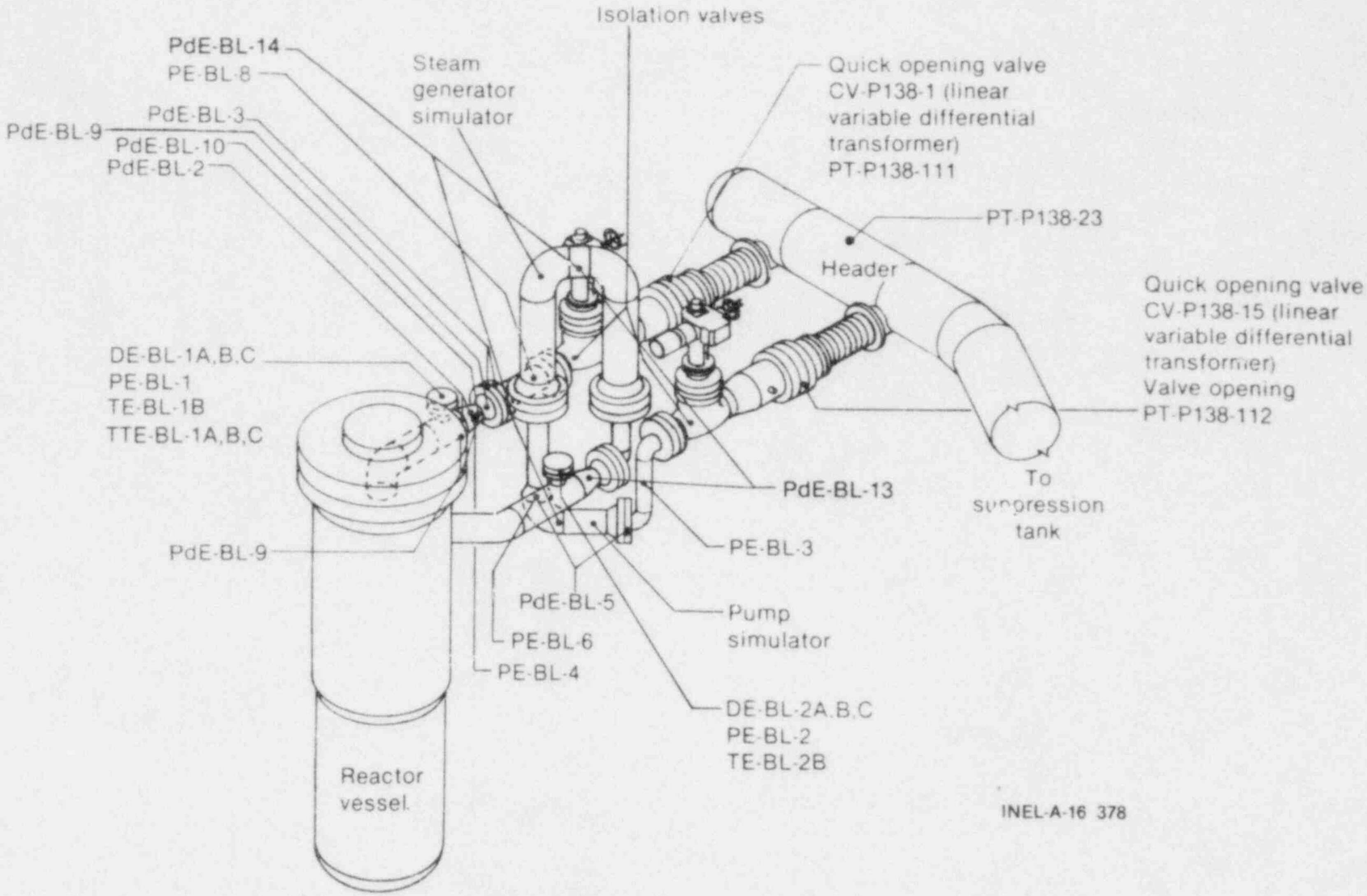


Figure 2. LOFT broken loop thermo-fluid instrumentation.

POOR ORIGINAL

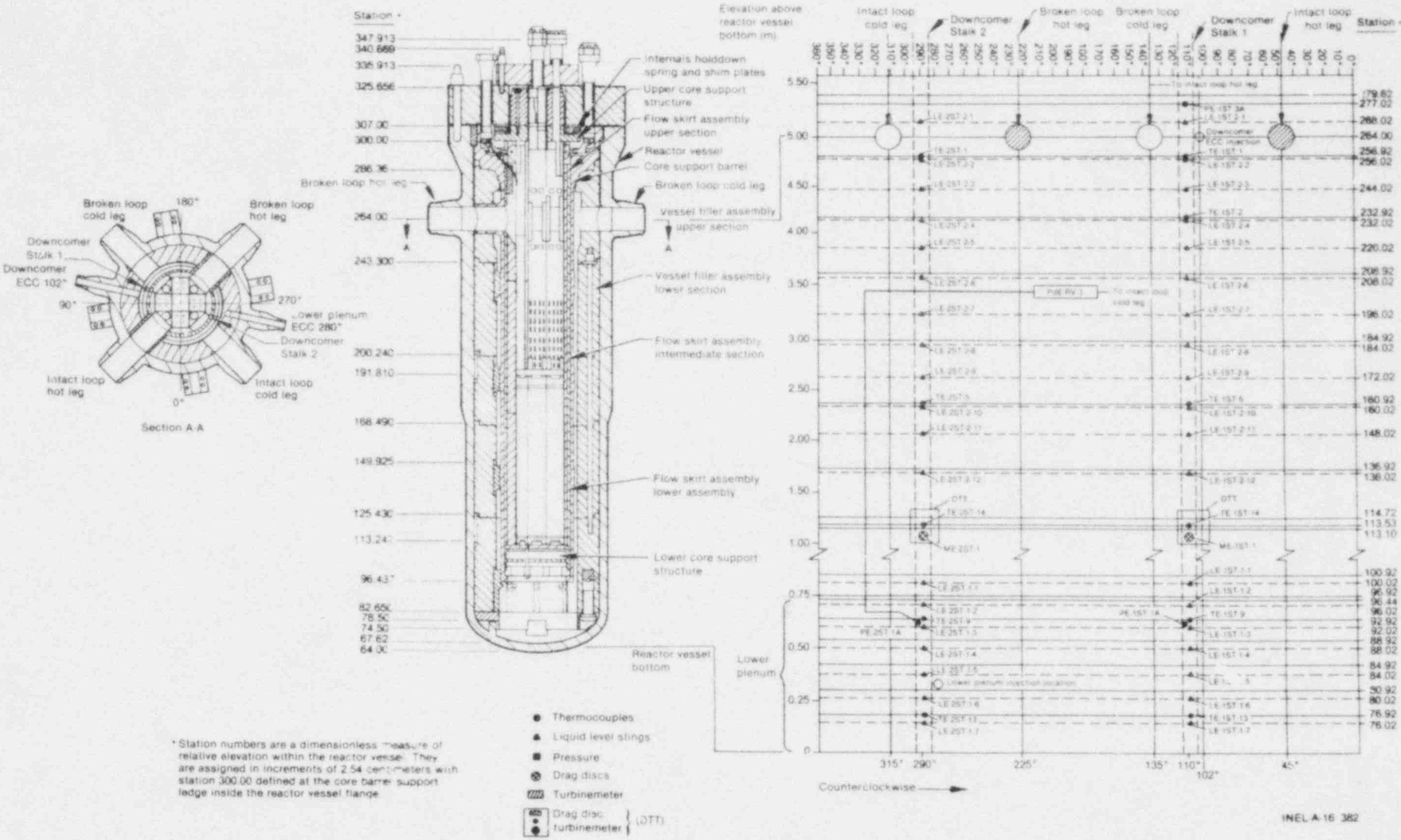


Figure 3. LOFT reactor vessel instrumentation.

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

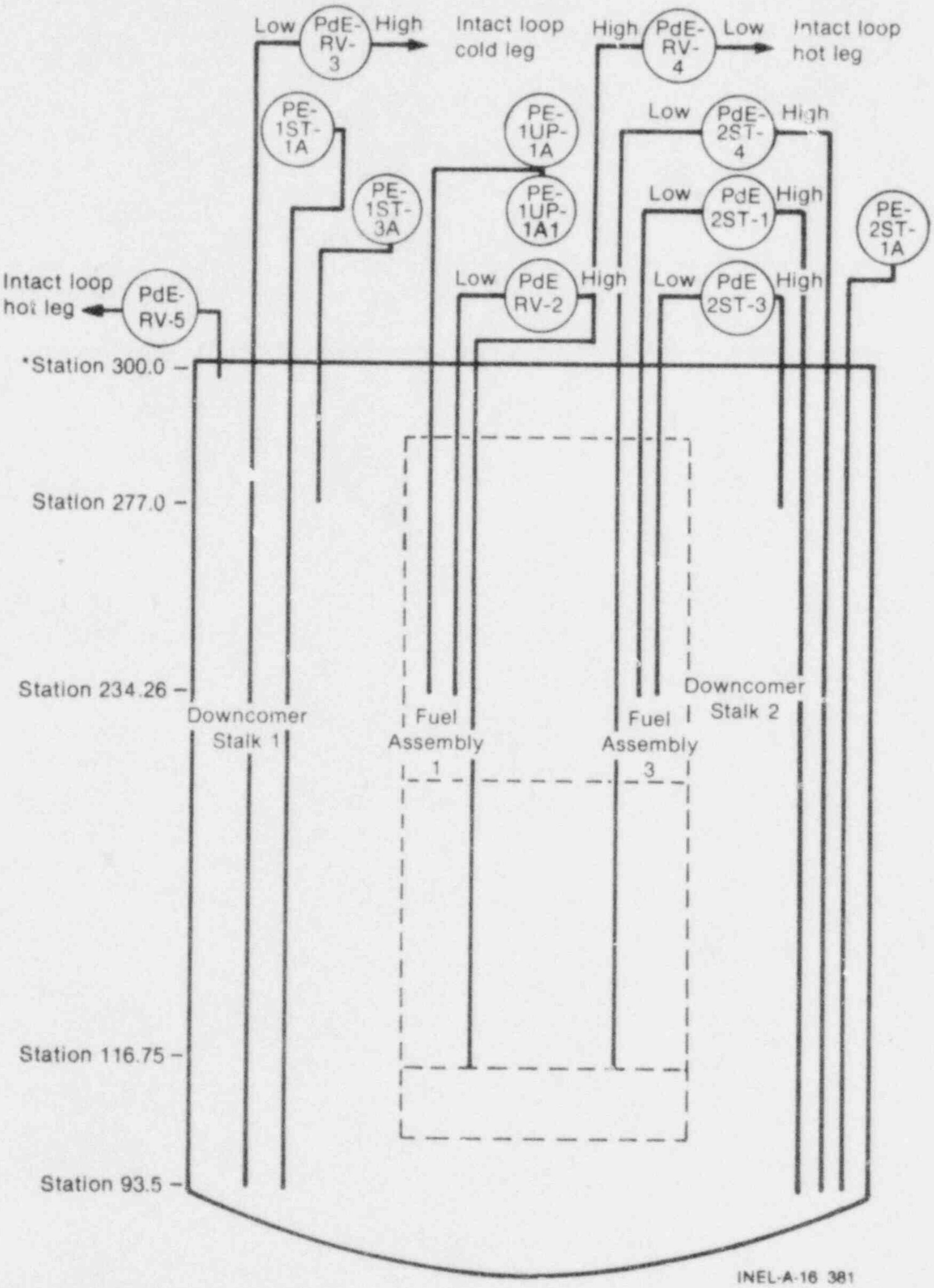


Figure 4. LOFT reactor vessel pressure and differential pressure instrumentation.

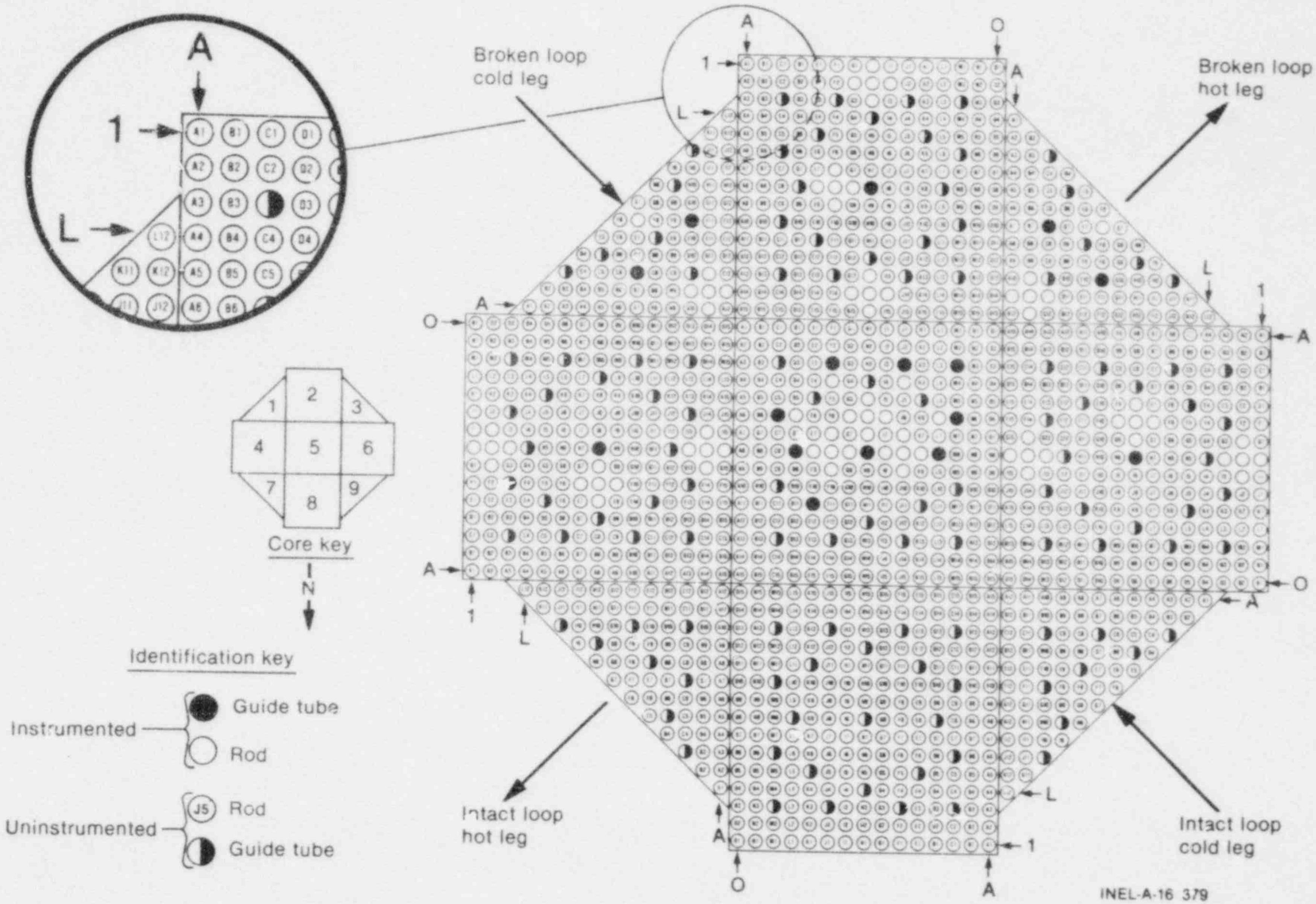
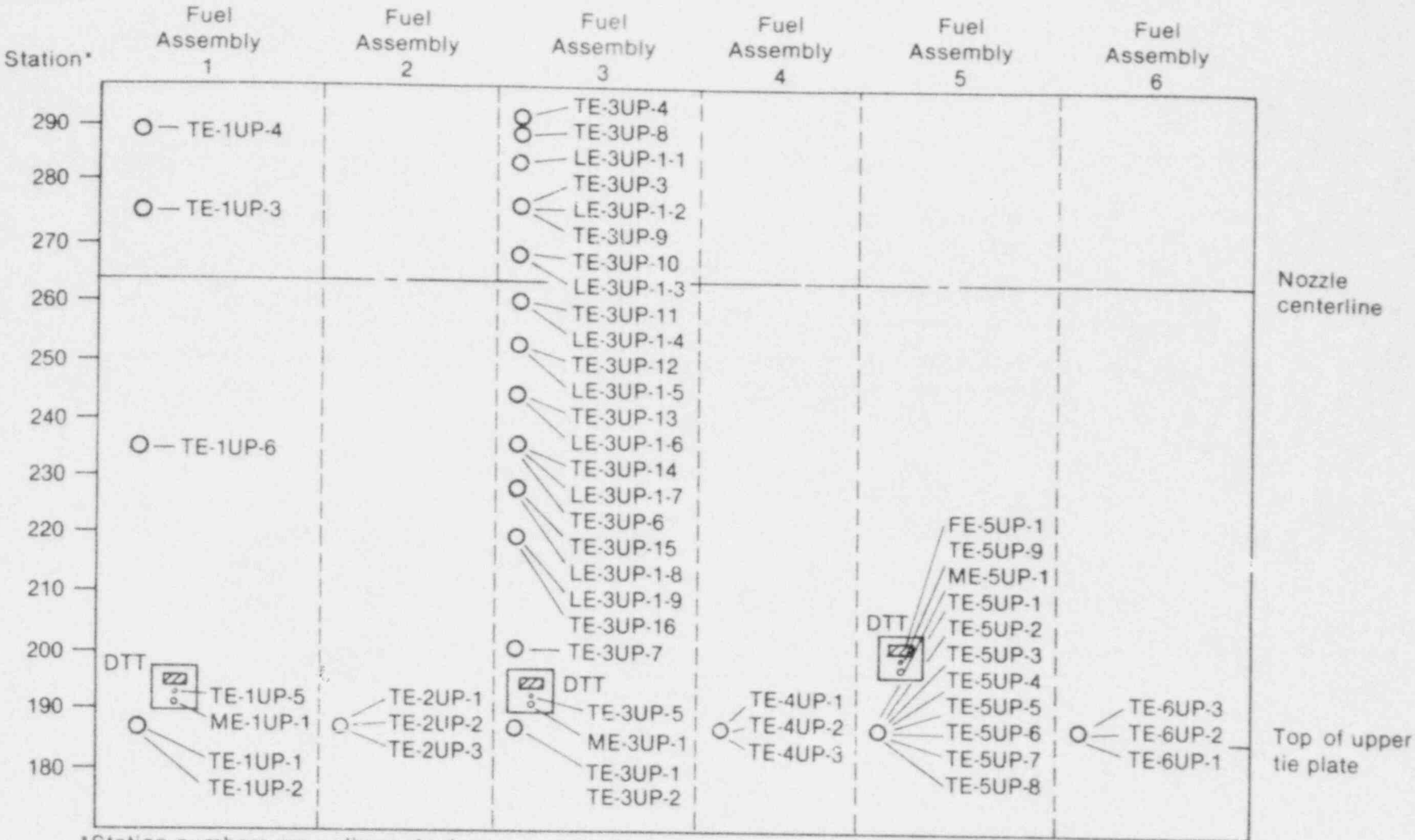


Figure 5. LOFT core map showing position designations.



*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.

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:

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

The designations for the different systems are:[a]

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

---

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

---

2. COMPUTER SIMULATION

The RETRAN01/MOD2 computer code[a,3] was used to simulate the transient thermal-hydraulic responses of the LOFT system during the anticipated transients for Experiments L6-1, L6-2, L6-3, and L6-5. This code was chosen because it is in general use within the nuclear industry for predicting these types of transients. In particular, this code allows the modeling of automatic control components in the LOFT system such as the feedwater valve controller; feedwater valve stem position being a function of steam flow rate and steam generator downcomer liquid level. Details of the LOFT RETRAN model are included in the following sections.

2.1 Input Model Description

This description of the RETRAN01/MOD2 input model used for these EP analyses includes a general model description; details of the model; and descriptions of modeling used for the steam generator, feedwater control system, and main steam control system.

2.1.1 General Model

The general model used for the analyses is shown schematically in Figures 7, 8, and 9. The general model shown has 41 volumes and 49 junctions. Some changes were made to the general model for each specific analysis. These changes will be discussed in detail later.

The general model used two volumes to model the broken loop. Since the flow rate through this loop is expected to be very small, detailed nodalization was considered to be unnecessary. Similarly, no passive heat conductors were modeled in the broken loop. The warm-up lines, which

---

a. RETRAN01/MOD2 (Absolute Executable) Program, Idaho National Engineering Laboratory Configuration Control Number H0009863B.

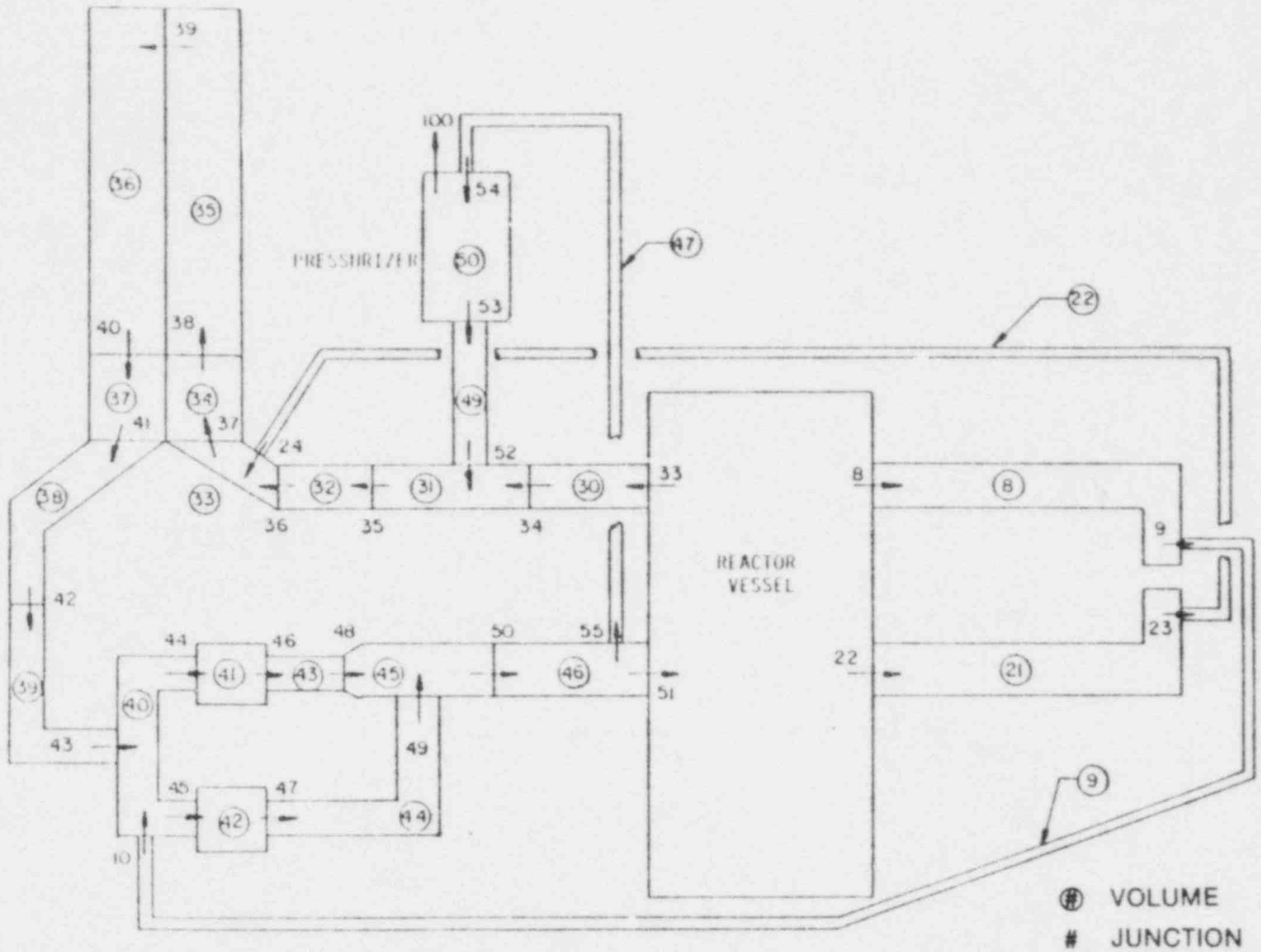


Figure 7. Nodalization of LOFT intact and broken loops.

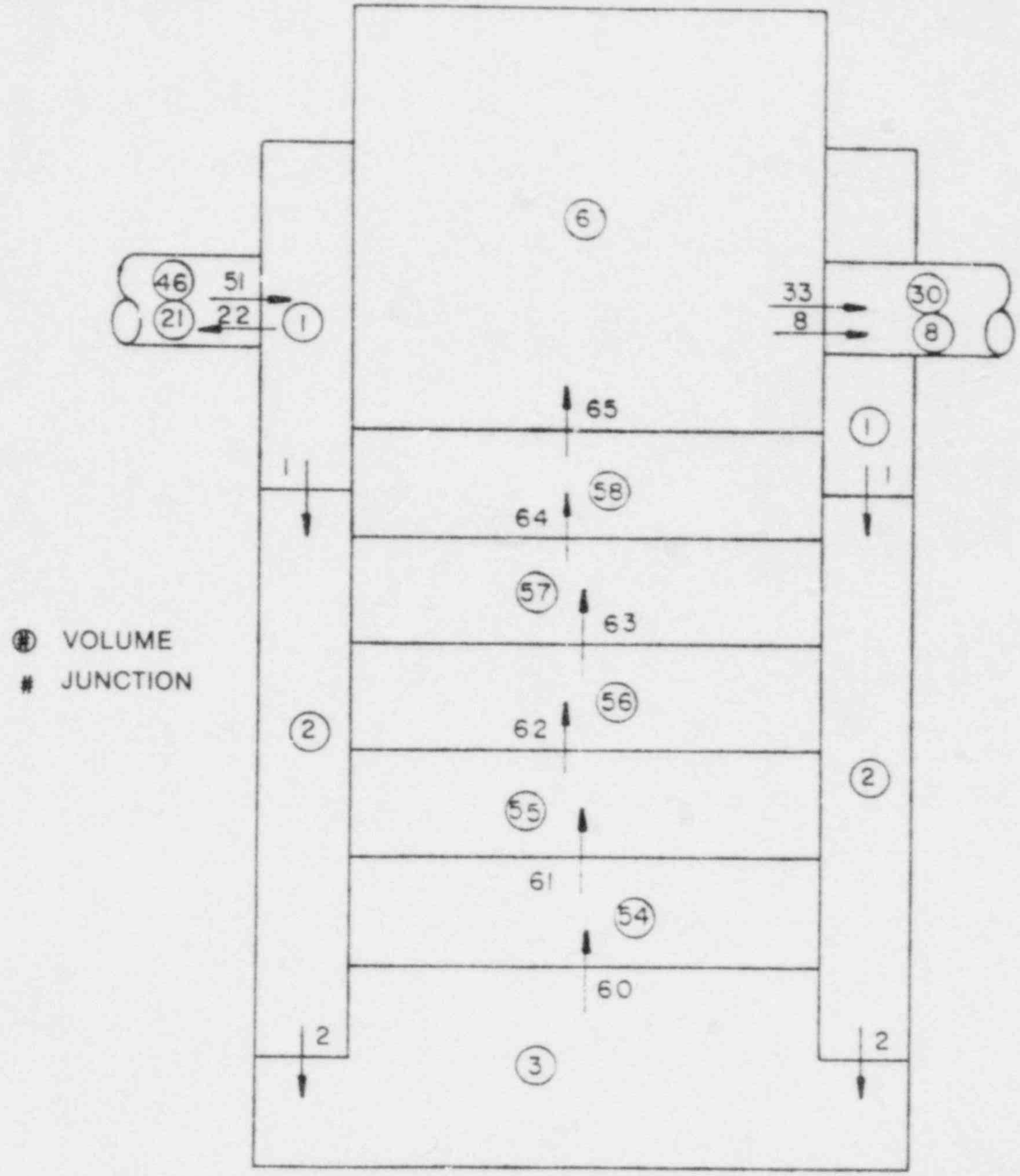


Figure 8. Nodalization of LOFT reactor vessel.

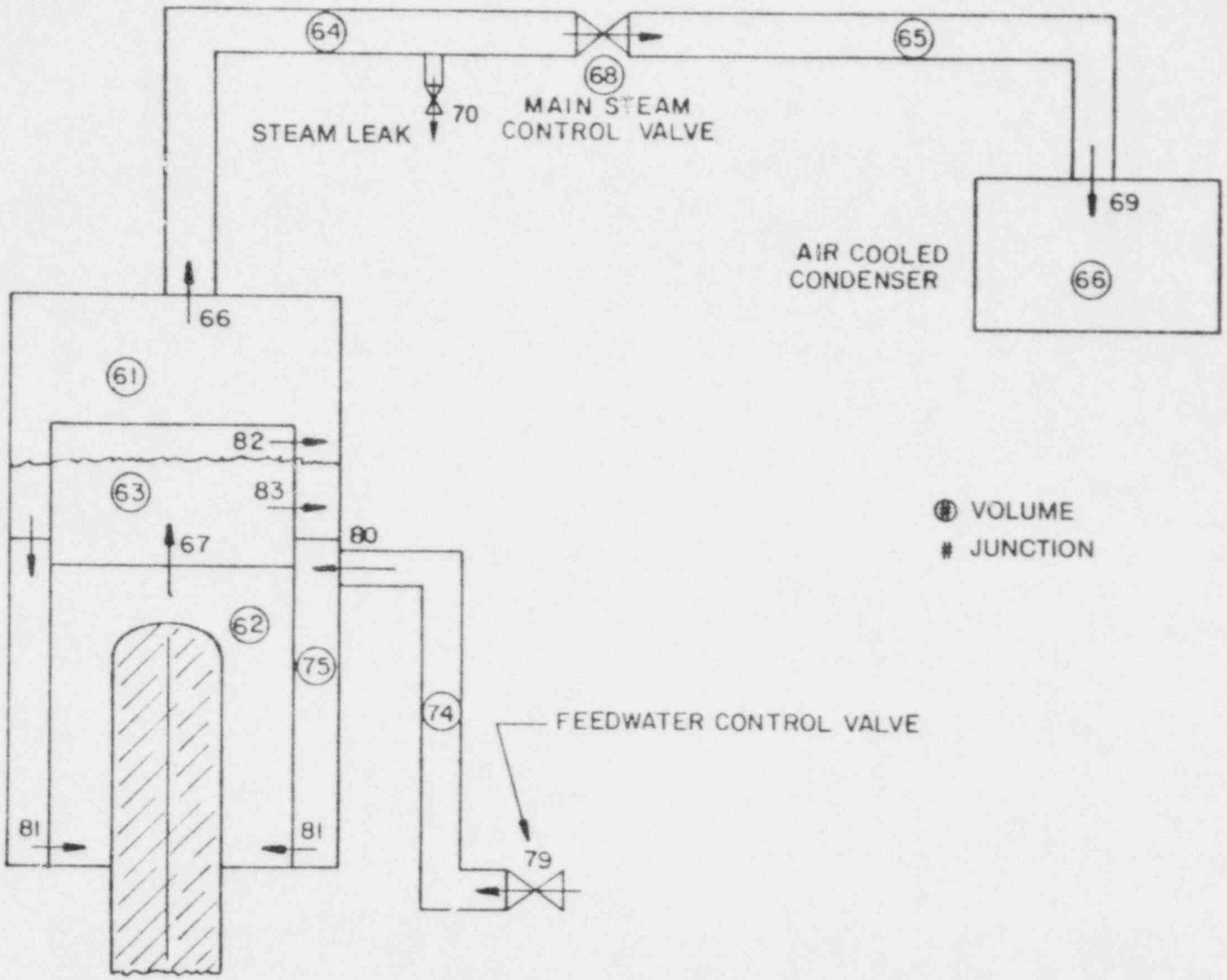


Figure 9. Nodalization of LOFT secondary system.

connect the broken loop to the intact loop, were modeled with Volumes 9 and 22. Although the flows through these lines are small, it was felt that the effect of these flows on certain operational transients might be noticeable. The spray line coming from the intact loop cold leg to the pressurizer was modeled with Volume 47. The pressurizer spray has a major influence on primary system pressure during anticipated transients, thus the inclusion of this in the model was considered to be mandatory. Also, the power-operated relief valve for the pressurizer was included in the model. Two primary coolant system pumps were employed in the model. Two volumes with two corresponding heat conductors were used to model the primary side of the steam generator tubes. These conducted heat to the secondary side, Volume 62. The reactor vessel downcomer was modeled with upper and lower volumes. The reactor core was modeled with a single flow path having five core volumes and five core conductors. The accumulator and low- and high-pressure injection systems were unnecessary for the model because they would not be activated during the experiments.

The secondary coolant system model is shown schematically in Figure 9. This model has both a main steam control valve and a feedwater control valve. Dependent upon which operational transient is analyzed, the operation of these control valve models is crucial to the predicted thermal-hydraulic behavior during the transient. Consequently, considerable effort was expended to obtain RETRAN control system input which would give predicted behavior corresponding to that projected for these valves. The expected pressure in the air-cooled condenser during these opertional transients is considered to be one of the best-known parameters. Thus, the air-cooled condenser was modeled as a time-dependent volume The important secondary system parameters for these experiments are the feedwater and steam flows and the steam generator heat transfer.

The general model for the secondary side of the steam generator has one volume into which heat is transferred from the primary to the secondary system. The model also has one downcomer volume and two separator volumes. Volume 63 has two junctions leading to Volume 61. The upper junction flows steam and the lower junction flows mostly liquid with a small amount of steam bubbles during steady-state operation. Volumes 63 and 61 use different bubble rise models and mixture level elevations for initial input conditions. The recirculation ratios and flow rates which were in the secondary side were obtained from calculations based on previously obtained data. The modeling of the steam generator is discussed in greater detail in Section 2.1.3.

2.1.2 Details of the Model

For the analysis of anticipated transients, such as the experiments in Test Series L6, the controls that adjust primary and secondary system thermal-hydraulic conditions have to be modeled accurately. Thus, reactor scram, operation of valves, pump behavior, reactivity feedback, feedwater flow, steam flow, and pressurizer performance are among the items that must be correctly modeled. The following discussion covers general details of the model. Specific items such as trip setpoints are given in Section 2.2. Reactor scram for the LOFT system can be initiated from a number of different signals. The model includes scrams initiated from the intact loop hot leg setpoints for low pressure, high pressure, high temperature, and low flow. In addition, scrams on high core power and low steam generator liquid level are modeled.

Valves that are important to LOFT operation during the Test Series L6 experiments include the feedwater control valve, the main steam control valve, and the pressurizer power-operated relief valve. The feedwater isolation valve closes more slowly than the feedwater pump coasts down, so it was not modeled. The feedwater control valve and the main steam control valve behaviors were simulated using RETRAN control system models. The development of these models is discussed in Sections 2.1.4 and 2.1.5.

During normal operation, the primary system pressure is controlled by the action of the pressurizer heaters and the pressurizer spray. The LOFT pressurizer has two banks of heaters: the cycling heaters and the backup heaters. The RETRAN model simulates these heaters using heat exchangers with trips, allowing the heaters to cycle on and off appropriately. The LOFT spray valve is always open at least slightly. It allows a spray flow of about 0.0315 L/s (0.5 gpm) in its full-closed position and allows a spray flow of about 1.261 L/s (20 gpm) when the high-pressure signal causes it to open, assuming a primary coolant flow rate of 1.636×10^6 kg/h (3.6×10^6 lbm/hr). This valve was modeled using a control system that varied the valve flow area by a factor of 40, following the trip signal, with initialization at low flow and small area. The LOFT pressurizer is never in a true steady-state condition, because the heater and spray flow contributions do not exactly balance. Therefore, an additional heat exchanger was added to balance the 0.0315 L/s (0.5 gpm) spray flow for steady-state energy balancing in the pressurizer. This small heat contribution was terminated in the analyses shortly after initialization, so that this heater has no effect on the anticipated transient portion of the analyses. The nonequilibrium pressurizer model available in RETRAN was not used for Experiment L6-5, but it was used for the other three anticipated transients.

The primary system pumps will be tripped off in Experiment L6-2. In the analysis for Experiment L6-2 the pump speed was input as a function of time after trip, using a curve obtained from Experiment L3-1 experimental data taken at approximately the same initial operating conditions intended for Experiment L6-2. For the other analyses, the pumps were run at a constant speed, so there were no modeling problems.

The reactor core was modeled using five stacked volumes. The power profile used is given in Section 2.2. The metal of the LOFT system was modeled using passive heat conductors, with the exception of the broken loop, which was not considered to be important. Twenty-six conductors were used in the basic model, including the five core conductors.

The RETRAN01/MOD2 model for Experiments L6-1, L6-2, and L6-3 is slightly different from the model for Experiment L6-5. Experiment L6-5 was performed prior to the issue of this report, but after the pretest prediction for Experiment L6-5 was performed. Analysis of data from Experiment L6-5 gave information useful in improving the RETRAN01/MOD2 modeling. The three areas which differ between the Experiment L6-5 model and the model for the other three experiments are (a) decay heat modeling, (b) leakage through the steam control valve when the valve is closed, and (c) ambient heat losses from the reactor system.

The EP analysis for Experiment L6-5 used the code-default decay heat calculation which assumes an infinite irradiation time. The actual irradiation time prior to Experiment L6-5 initiation was approximately 20 h. Thus, decay heat in the RETRAN EP analysis was too high. This caused the RETRAN predicted pressures in the secondary and primary systems to rise too fast. For the EP analyses for Experiments L6-1, L6-2, and L6-3, best estimate decay heat curves based on operating history were input into the RETRAN code.

The EP analysis for Experiment L6-5 did not model any leakage from the main steam control valve when in the closed position. Analysis of the Experiment L6-5 data showed the leak to be approximately 0.2 kg/s (0.44 lbm/sec) when the valve was closed. This same value of steam leakage for the valve (once closed) was assumed for the EP analyses of Experiments L6-1, L6-2, and L6-3.

No ambient heat losses from the piping and reactor vessel were modeled in the EP analysis for Experiment L6-5. However, in the EP analyses for Experiments L6-1, L6-2, and L6-3, ambient heat losses to the containment environment were modeled as a constant 248-kW heat loss (based on plant operations data) during the first 200 s of the transient.

2.1.3 Steam Generator

The internal geometry of the LOFT steam generator includes a center section inside a shroud, an outer or downcomer region, and an upper steam dome region. The steam generator U-tubes are enclosed in the lower portion of the shroud. Above the U-tubes, the shroud diameter becomes smaller. The steam-liquid mixture is separated by the primary separator and the mist extractor. The internal geometry is shown in Figure 10.

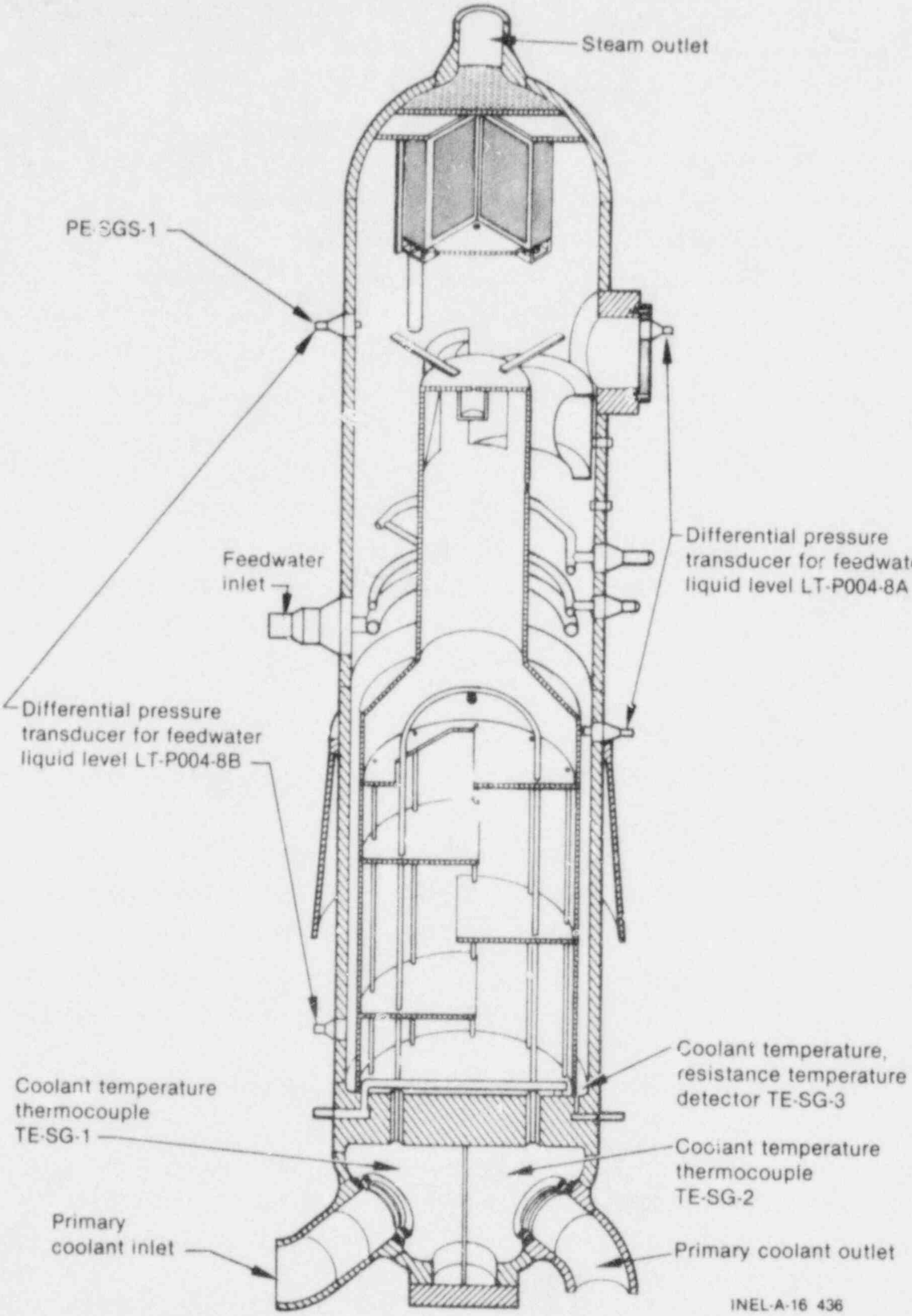


Figure 10. LOFT steam generator.

The important considerations for the steam generator were to obtain correct heat transfer from the primary side to the secondary side, to obtain correct steam flow rates from the steam generator and feedwater flow rates into the steam generator, and to obtain correct recirculation rates and liquid levels within the steam generator secondary side. It is emphasized that fluid conditions are measured in the steam and feedwater lines. The only data obtained from inside the steam generator during an experiment is the downcomer liquid level. A coolant temperature measurement is also available near the bottom of the steam generator downcomer for steady-state conditions. This was used to calculate estimated recirculation rates within the steam generator for different power levels. However, an assumption must be made for the amount of steam carryunder back to the downcomer when calculating the recirculation ratio. If a large steam carryunder does occur, the recirculation ratio would be correspondingly lower.

A large number of different steam generator models were evaluated. Comparisons to data from the level-setpoint-change experiment were used to evaluate the models. Most of these comparisons were made using an eight-volume secondary side model. However, some had to be made using the complete (primary plus secondary) system model, since there was significant interaction between primary and secondary conditions during the experiment.

To adequately model the secondary side of the steam generator, four volumes were considered to be necessary: one volume to represent the region within the shroud that encloses the U-tubes, one volume to represent the region within the shroud above the U-tubes, one volume for the downcomer, and one volume for the steam dome. Since the liquid level measurement is of primary importance, great care was taken to ensure that a change in liquid level would correspond to the correct change in liquid volume. In order to do this, information was obtained on the rate of change of volume with height (above the tube sheet) for the regions inside and outside the shroud.

Several different variations in the model were tried: the recirculation ratio, the steam carryunder, the separation model, junction heights, and mixture levels were among the parameters that were varied. Unfortunately, a single, best model was not found. However, the model chosen did give results that compared well with the level setpoint change data. If more experimental data from locations within the steam generator secondary side were available, perhaps a slightly different model would have been derived. Data from the anticipated transient experiments may show what adjustments should be made to the model.

For the model selected, pressure and enthalpy are input in Volume 61. Volumes 63 and 61 use a separation model and initially have mixture levels. The separation model for Volume 63 has a zero gradient (that is, bubble partial density is contant with elevation in the control volume) and a high separation velocity of 48.46 m/s (159 ft/sec). The separation model for Volume 61 uses a small gradient (0.0626) and a small separation velocity (0.021 ft/sec). These separation velocities are adjusted by RETRAN to obtain steady-state initialization. The gradient in Volume 61 is also adjusted by the code. The adjustments made by RETRAN are dependent upon such factors as the mixture levels and the flow split between Junctions 82 and 83. Fortunately, the comparison to data from the level-setpoint-change experiment showed that results were not greatly dependent upon the separation model parameters used in Volumes 61 and 63. Volume 75 used a separation model with a gradient of 0.8 and a separation velocity of 0.914 m/s (3.0 ft/sec), but it is initially filled with liquid. The recirculation ratios of 9.5 for 50% power and 6.5 for 75% power were obtained from calculations. These calculations were made assuming zero carryunder of steam bubbles back to the downcomer. Heat transfer from the primary to the secondary side of the steam generator was modeled using two heat conductors.

2.1.4 Feedwater Control System

The feedwater control system of LOFT, which is called "Steam Generator Water Level Control System", provides the means for stable control of the fluid inventory in the steam generator.

The basic measure of steam generator inventory is taken as the level of water in the drum/downcomer portion of the unit. The difference between the measured level and a desired, or reference, level provides the basic error signal for the system. The reference water level is programmed to a lower value for low-power levels to accommodate the potentially large level "swell" that would occur in the event the steam control valve (CV-P4-10) is opened rapidly from a closed position. At high power, the level is programmed high in the drum so as to maintain a level above the specified minimum during the level "shrink" following a sudden power reduction (for example, a reactor trip or main steam valve closure). The present operating characteristics are that the programmed level at zero power is 2.921 m (116 in.) above the top of the tube sheet, and a 0.025-m (1-in.) increment in level is equivalent to a 10% increment in power.

A steam flow signal is required by the feedwater control system to obtain a satisfactory transient response with consideration for level swells and shrinks accompanying power changes, and also to reduce the steady-state level error which would occur without it.

To stabilize the control of the liquid level (the "output" of the steam generator in a control sense), a feedback signal representative of the input is needed. In conventional water level controls, the feedback signal is normally feedwater flow, but a feedwater regulating valve position is used for the LOFT control system.

A block diagram of the feedwater control system is shown in Figure 11. Referring to Figure 11, a differential-pressure-type liquid level transmitter is used to generate a current signal proportional to the steam generator water level. This signal is summed with a signal proportional to steam flow and compared to a reference signal equivalent to the desired operating level in the steam generator.

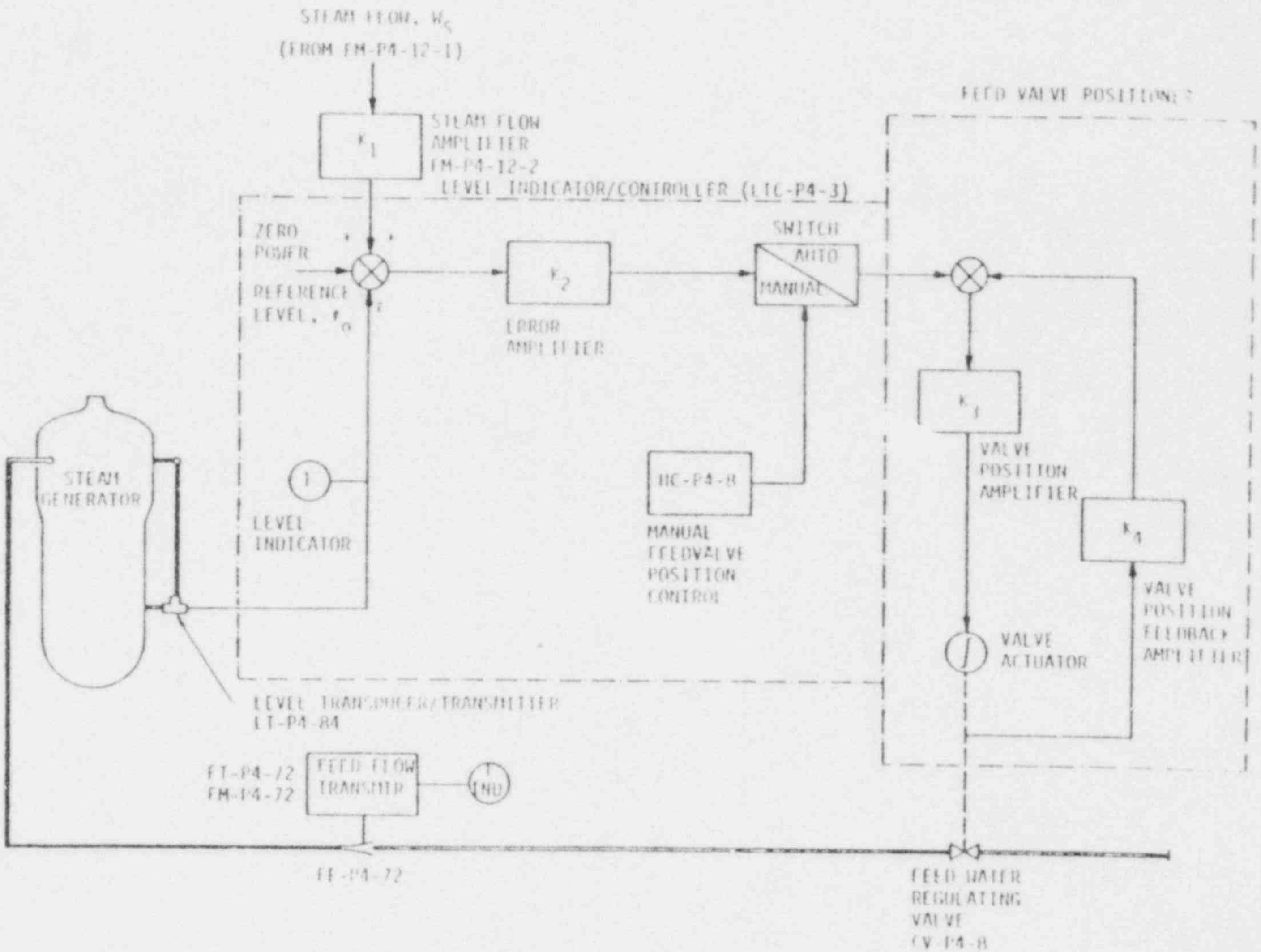


Figure 11. Steam generator feedwater control system.

Figure 12 shows the RETRAN model of the feedwater control system, where

WP** 68 = steam flow (lbm/sec)

LIQL 61 = liquid level (ft)

TIMX 0 = time (sec)

CONS 0 = constant.

The output of the control block "FNG" is the desired operating water level in inches. The output of SUM -2 is the error signal in inches between the reference level and the measured water level, where g_1 = 12.0 for the unit conversion from feet to inches and G_2 = -1.0 for the difference. This error signal is summed with a signal proportional to steam flow by SUM block -3, where g_1 = 0.12 is a conversion factor for changing the steam flow unit, lbm/sec, into the equivalent liquid level in inches. For the LOFT steam generator, 136 363 kg/h (300,000 lbm/hr) of steam flow is equivalent to 0.254 m (10 in.) of liquid level in the steam generator, and the value of 0.12 can be easily calculated.

The output of SUM -4 is the stem position of the feedwater control valve (CV-P4-8), and the value of the output for this control block will depend on the initial stem position of the feedwater control valve. The value of 0.08 for the gain of SUM -4 is based on the assumption that a 1-in. error signal is equivalent to an 8% change of the valve stem position.

The transfer function of the feedwater valve positioner in Figure 11 is given as

$$\text{Transfer Function} + \frac{K_3/S}{1 + K_3K_4/S} = \frac{1/K_4}{S/K_3K_4 + 1}. \qquad (1)$$

The gains (K_3, K_4) of the actuating loop have been set to convert the input signal to valve travel and to establish the time constant of the valve positioner loop. The value of K_4 has been set to obtain the

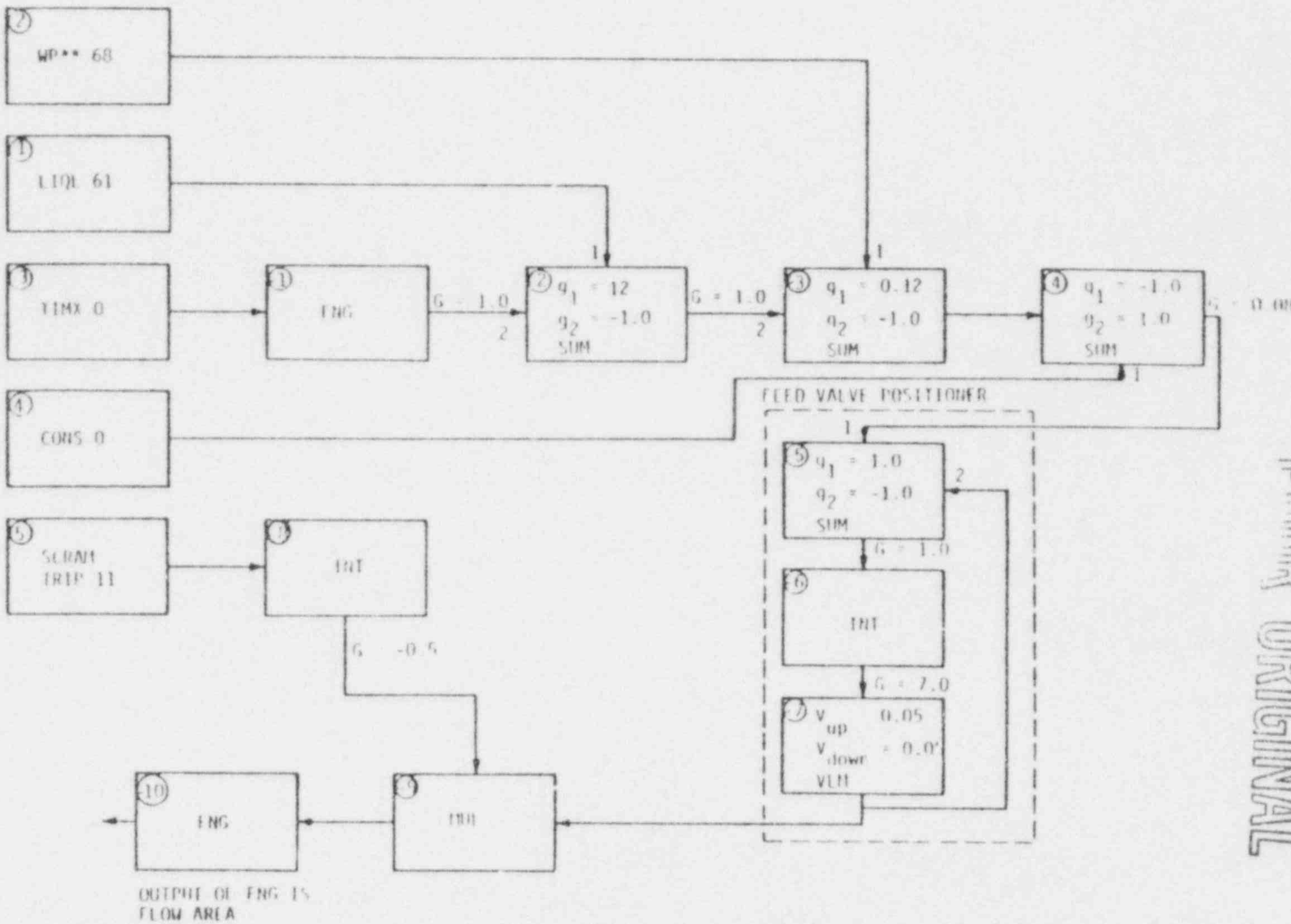


Figure 12. RETRAN model of feedwater control system.

desired ratio of the valve travel to the magnitude of the signal from the level controller. The value of K_3 is set to establish a loop pass band at least a decade higher than the steam generator break frequency (1 Hz versus 0.1 Hz).

The RETRAN input model for K_4 is used with the value of -1.0 at SUM -5 with the assumption that the valve actuator opens 100% for a 100% error signal and the negative sign stands for the negative feedback.

K_3, which is equivalent to the gain of INT -6, is selected for the band width at the valve actuator of not less than 1 Hz, that is,

$$\frac{K_3K_4}{2\pi} > 1 \rightarrow K_3 > 2\pi. \tag{2}$$

Therefore, the closest integer, 7.0, is chosen.

The velocity limiter, VLM -7, is added in the RETRAN input model because the valve actuator cannot move the feedwater valve faster than 5% per second. The control blocks INT -8 and MUL -9 are added to the control system to account for the feedwater pump trip which will occur as soon as the reactor scram is initiated, stopping flow in the feedwater line within 2 s. The control block, FNG -10, changes the valve stem position into the flow area of the valve which is the main hydraulic parameter to calculate flow rate for the valve in RETRAN.

The values used for converting the valve stem position to the valve flow area are given in Table 2. The initial stem positions used for the RETRAN input model are 13.6 and 31% open for 50 and 75% power, respectively.

2.1.5 Main Steam Control System

The steam flow control system is a plant power control system (PPCS), since the amount of heat dissipated in the secondary system will be determined by the steam flow. However, the operator will be provided with

TABLE 2. STEM POSITION VERSUS FLOW AREA FOR FEEDWATER CONTROL

Stem Position (%)	Flow Area (%)
0.0	0.0
2.5	18.94
5.0	24.62
10.0	33.71
13.6	39.17
15.0	41.29
20.0	47.35
25.0	52.65
30.0	56.82
35.0	61.36
40.0	65.15
45.0	68.94
50.0	71.97
55.0	75.00
60.0	78.41
70.0	84.85
80.0	89.77
90.0	95.45
100.0	100.00 (= 0.2 ft^2)

a remotely operable steam control valve (CV-P4-10), a steam flow indicator, and a steam generator pressure indicator to assist in control of steam flow. The steam control valve is assumed to be operated in automatic mode through the transients.

Automatic control of the steam control valve is provided to protect the steam generator from overpressure and to limit primary system cooldown after reactor trip from power operation. If the steam pressure is below 6.34 MPa (920 psig) and a reactor trip signal is present, the steam control valve will begin to close. When pressure reaches 6.41 MPa (930 psig), the valve will stop closing. If the steam pressure equals or exceeds 7.03 MPa (1020 psig), the steam control valve will start opening. When the pressure drops to less than 6.96 MPa (1010 psig), the valve will stop. The valve, therefore, functions in the same manner as a power-operated relief valve, but without venting secondary system water inventory to the containment.

The RETRAN input model for the steam flow control valve is shown in Figure 13, where the input signals are:

1. HI-P trip - signal for high-pressure trip

2. LO-P trip - signal for low-pressure trip

3. SCRAM trip - signal for reactor scram trip.

It should be noted that the part of Figure 13 enclosed by dotted lines is only applicable for Experiment L6-1 (loss-of-load test) in which the initiating event is closure of the steam flow valve. The speed of the valve stem movement is given as 5% per second. There is assumed to be no reactor scram during the period of the initiating event for Experiment L6-1.

The output of SUM -3 in Figure 13 depends on the reactor scram trip. If the reactor is not scrammed, the output of SUM -3 is zero so that no changes occur in the valve flow area. If the reactor is scrammed, the output of SUM -3 can be 0.05 or -0.05, depending on whether the steam generator pressure is above 7.03 MPa (1020 psig) or below 6.34 MPa (920 psig).

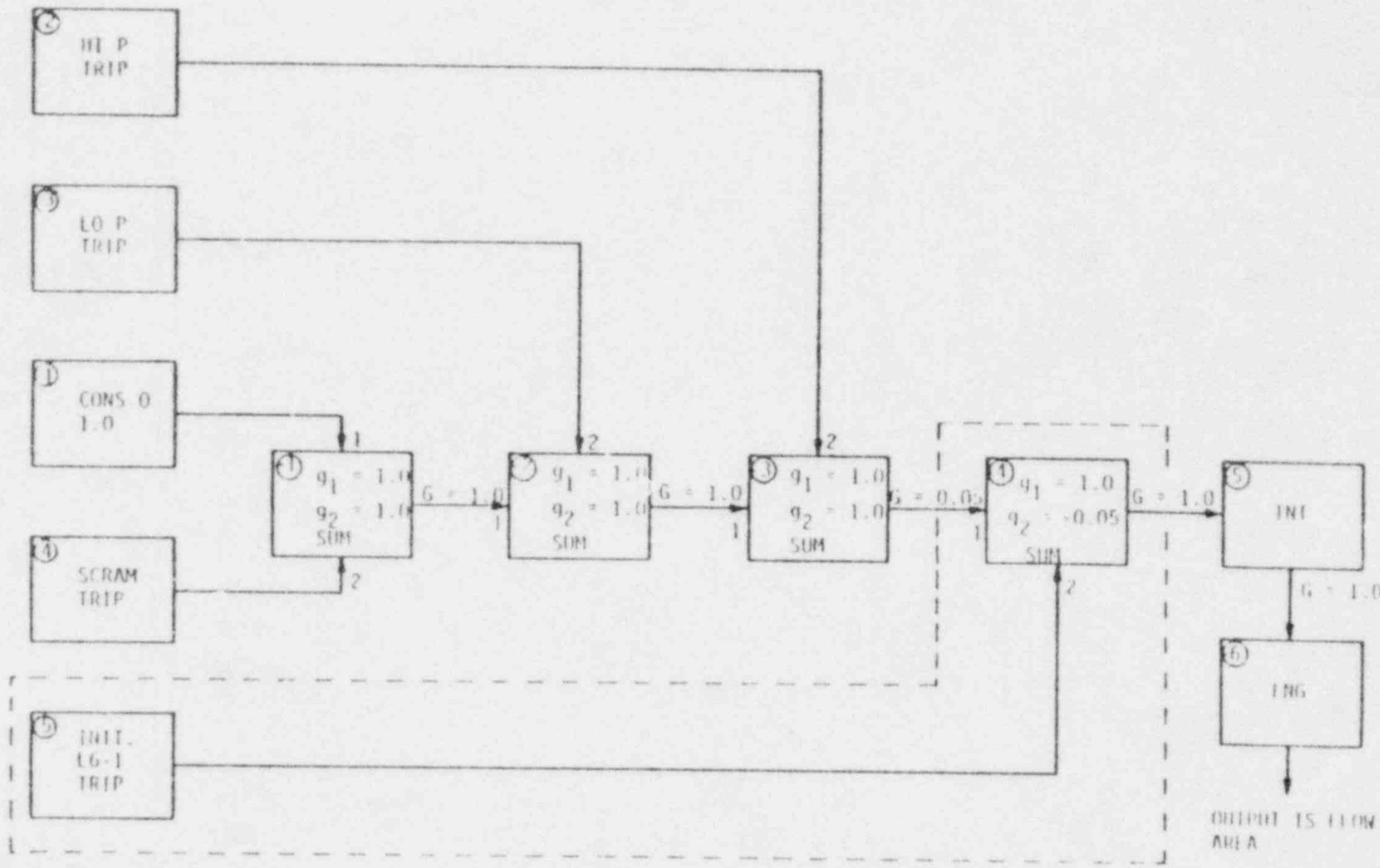


Figure 13. Steam flow valve control system.

The output of INT -5 is the stem position of the steam control valve and FNG -6 (Figure 13) will convert the stem position to the valve flow area according to Table 3. A 100% stem position gives a smaller flow area than the fluid volume areas connected by the steam control valve, even if the multipliers in Table 3 give higher values than 100%.

2.2 Boundary Conditions and Initial Conditions

Boundary conditions and initial conditions specified for Experiments L6-1, L6-2, L6-3, and L6-5[1], and on which these analyses are based, are discussed in the following sections.

2.2.1 Trip Setpoints

The RETRAN model for trip setpoints is given in Table 4. It should be noted that the steam generator low-liquid-level scram is set at 2.82 m (111 in.), but this level scram will be used for Experiment L6-5 (loss of feedwater) only. During normal operation of the LOFT system, the steam flow control valve would close for a 35% mismatch between the feedwater flow and the steam flow of the steam generator secondary side. However, for Experiments L6-1, L6-2, L6-3, and L6-5, the trip logic of 35% mismatch will be disabled from the experimental loop. Therefore, the mismatch trip was not modeled in the RETRAN input for any cases.

2.2.2 Pressurizer

Cycling heaters will come on if the pressurizer pressure drops to 14.84 MPa (2153.5 psia), and trip off if the pressure rises to 15.05 MPa (2183.5 psia). The backup heaters will come on if the pressure drops to 14.81 MPa (2148.5 psia), and trip off if the pressure rises to 14.92 MPa (2163.5 psia). The capacity of the cycling heaters is 36 kW, while that of the backup heaters is 12 kW.

The pressurizer spray valve will be open to 1.26 L/s (20 gpm) for a maximum flow at 15.32 MPa (2222.5 psia) and it will close to 0.0315 L/s (0.5 gpm) for minimum flow at 15.16 MPa (2198.5 psia). These values assume

TABLE 3. STEM POSITION VERSUS FLOW AREA FOR STEAM CONTROL VALVE

Stem Position (%)	Flow Area (%)
0.00	0.00
10.00	14.18
20.00	29.64
30.00	47.36
40.00	65.72
50.00	85.05
57.50	100.00 (= 0.228 ft^2)
60.00	105.67
70.00	127.58
80.00	149.32
90.00	171.39
100.00	193.30

TABLE 4. TRIP SETPOINTS FOR RETRAN MODEL

Signal	Setpoint
Low intact loop hot leg pressure scram	14.19 MPa (2058.5 psia)
High intact loop hot leg temperature scram	583.3 K (590°F)
Low primary coolant system flow scram	433.1 kg/s (952.78 lb/s)
High total reactor power scram	51.5 MW(t) (1.7157 x 10^8 Btu/hr)
High intact loop hot leg pressure scram	15.73 MPa (2281.5 psia)
Low steam generator liquid level scram	2.82 m (111 in.)
Pressurizer power-operated relief valve opens	> 16.70 MPa (2422.5 psia)
Pressurizer power-operated relief valve closes	< 16.56 MPa (2402.5 psia)
Main steam control valve starts opening	> 7.12 MPa (1032.5 psia)
Main steam control valve stops opening (reset)	< 7.05 MPa (1022.5 psia)
Main steam control valve starts closing	< 6.43 MPa (932.5 psia)
Main steam control valve stops closing (reset)	> 6.5 MPa (942.5 psia)
Steam generator main feedwater pump trip	Indirect trip on scram
Pressurizer heaters on	< 14.84 MPa (2153.5 psia)
Pressurizer heaters off	> 15.05 MPa (2183.5 psia)
Pressurizer backup heaters on	< 14.81 MPa (2148.5 psia)
Pressurizer backup heaters off	> 14.92 MPa (2163.5 psia)
Pressurizer spray on	> 15.32 MPa (2222.5 psia)
Pressurizer spray off	< 15.16 MPa (2198.5 psia)

a primary coolant of 1.636 x 10^6 kg/h (3.6 x 10^6 lbm/hr). The RETRAN model uses a control system to change junction area.

A power-operated relief valve was modeled to protect the system from overpressurization and to allow the pressurizer fluid to flash into the pressure suppression system of LOFT. The power-operated relief valve opens at 16.70 MPa (2422.5 psia) and closes at 16.56 MPa (2402.5 psia).

2.2.3 Reactivity Feedback and Power Profile

The feedback of Doppler and void reactivity given in the RETRAN input will be reconstructed such that the feedback reactivity will be zero at time zero.

The power profile used for five core conductors from the bottom to the top of core is:

Core Conductor	Relative Power
1	0.167
2	0.272
3	0.279
4	0.212
5	0.070

These relative power density values were calculated using measured data shown in Figure 14 from Experiment L3-2.

2.2.4 Initial Conditions

The values given in Table 5 are used as the initial conditions for the RETRAN input.

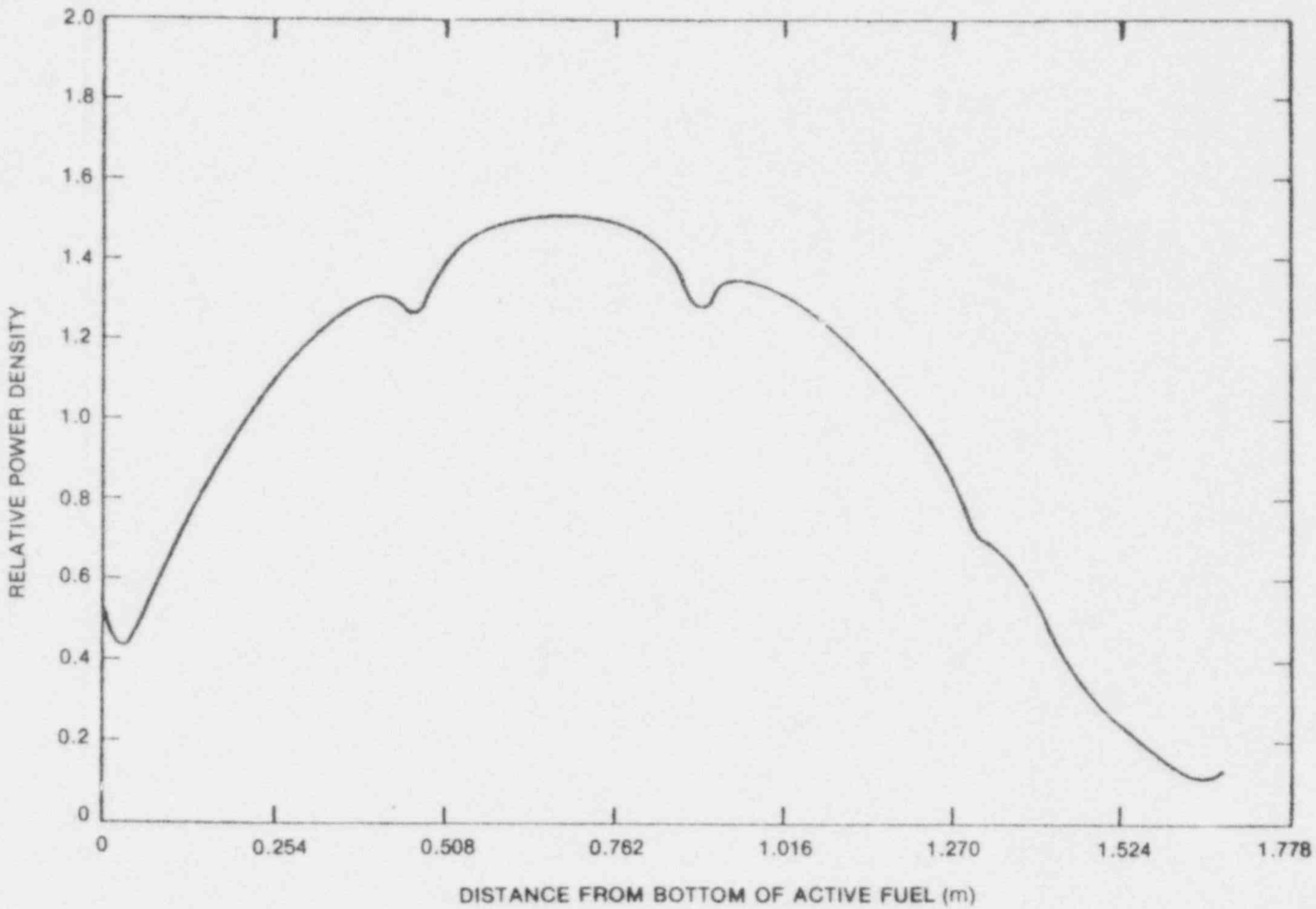


Figure 14. Relative power density for Experiment L3-2.

TABLE 5. SPECIFIED INITIAL CONDITIONS FOR TEST SERIES L6 EXPERIMENTS

Parameter	Specified Value
Power	37.5 MW
Pressure in primary system	14.95 MPa (2168.5 psia)
Flow rate in primary system	479.77 kg/s (1055.5 lb/sec)
Pressure in steam generator	5.44 MPa (788.5 psia)
Flow rate in secondary system	19.84 kg/s (43.65 lb/sec)
Recirculation ratio	6.5
Pressure in condenser	2.04 MPa (296.7 psia)
Feedwater enthalpy	165.52 kJ/kg (385.0 Btu/lb)
Pressure in feedwater system	7.52 MPa (1091.5 psia)
Temperature in intact loop cold leg	552.77 K (535°F,

3. CALCULATIONAL RESULTS

This section presents selected parameter plots that characterize the RETRAN01/MOD2 experiment predictions for Experiments L6-1, L6-2, L6-3, and L6-5.

3.1 Experiment L6-1 Prediction

Experiment L6-1 (loss-of-steam load) will be initiated by closing the main steam control valve at the rate of 5% per second (valve stem travel). The sequence of events that are calculated to occur in this transient is as follows:

Experiment L6-1 Event	Time (s)
Steam control valve starts to close	0.0
Feedwater valve starts to close	11.0
Steam control valve closed	11.5
Scram on high pressure of 15.72 MPa (2281 psi)	12.0
Feedwater flow stops	13.5
Steam control valve starts to open	27.0
Steam control valve closed	55.0

As soon as the steam control valve begins to close, steam flow rate (Figure 15) begins to drop and steam generator secondary pressure (Figure 16) starts to rise. As soon as secondary pressure rises to the pressure threshold for automatic opening of the steam control valve [7.12 MPa (1032.5 psia)] the valve opens to relieve secondary pressure. When the pressure drops to 6.43 MPa (932.5 psia), the valve automatically starts to close.

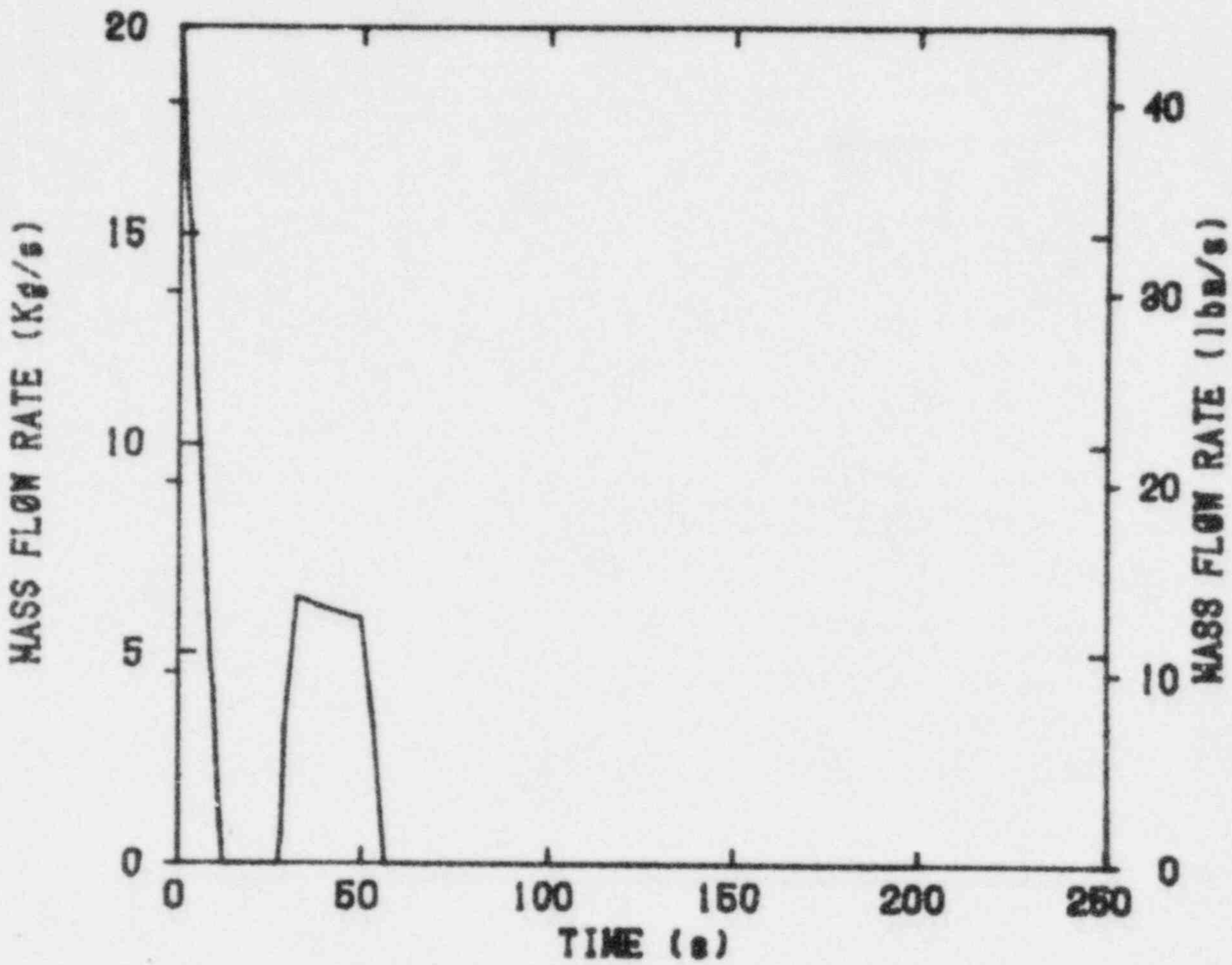


Figure 15. Mass flow rate through steam control valve for Experiment L6-1.

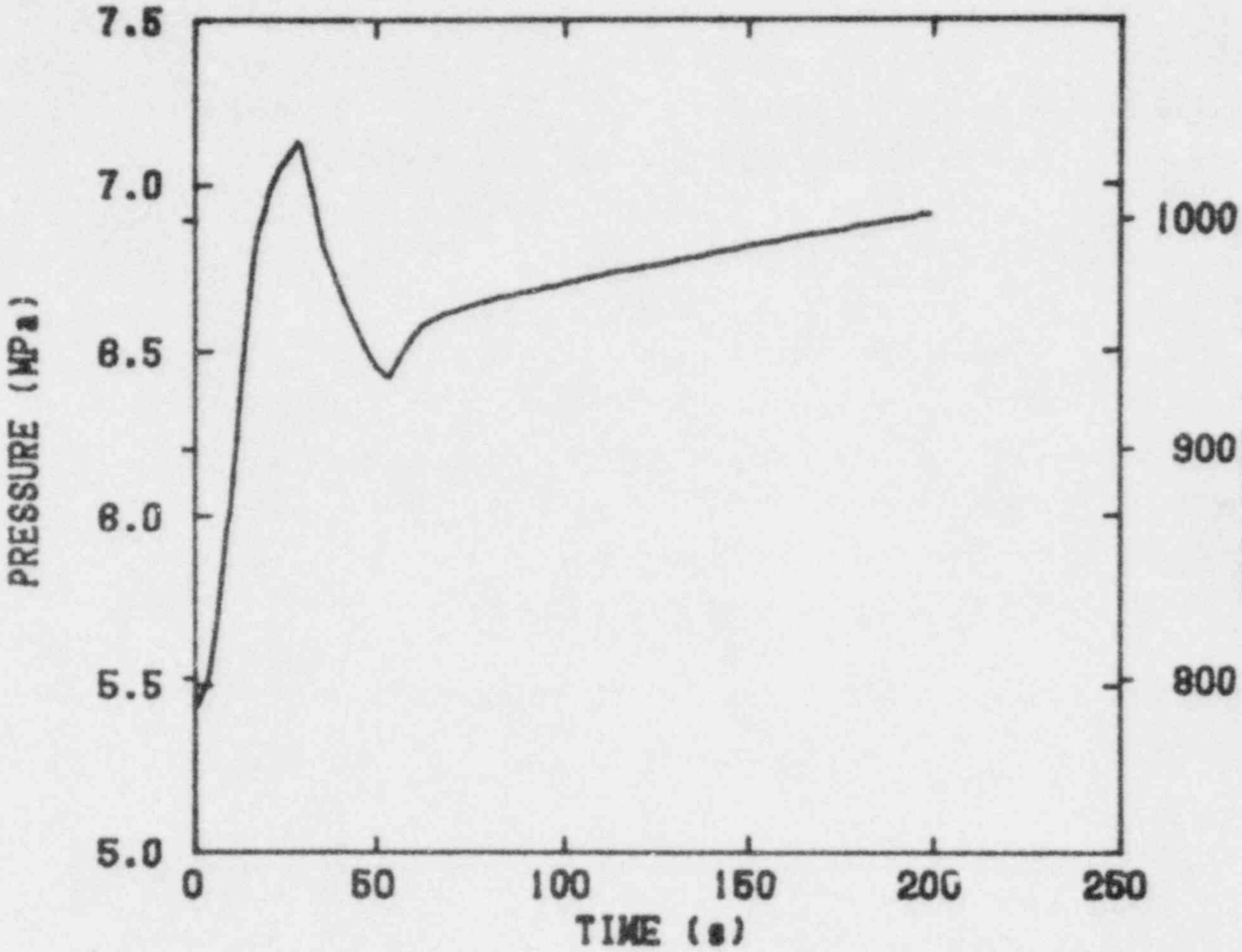


Figure 16. Pressure in steam generator secondary side for Experiment L6-1.

The rise in secondary pressure also has an effect on liquid level in the steam generator downcomer (Figure 17), which in turn has an effect on the feedwater controller. As the pressure increases in the steam generator secondary side, bubbles inside the steam generator shroud tend to collapse, lowering the mixture level inside the shroud. Flow inside the shroud drops as the steam control valve closes, and liquid level in the steam generator downcomer also drops to reach a new equilibrium with the level inside the shroud. Since downcomer liquid level is an input to the feedwater controller, feedwater flow (Figure 18) increases initially, then is shut off when the low steam flow signal into the controller begins to dominate the downcomer liquid level signal.

Figure 19 shows core power versus total heat transfer through the steam generator . As pressure and temperature in the secondary start to rise, heat transfer across the steam generator tubes starts to drop. This reduction in steam generator heat transfer in turn causes temperatures to rise in primary coolant system cold and hot legs, as shown in Figure 20. Hot leg temperature starts to drop when the reactor is scrammed on high pressure at 12 s. Changes in temperature during 27 to 55 s are due to the steam control valve opening, with the attendant heat transfer to the secondary.

Attending this heatup of the primary coolant is an insurge into the pressurizer (Figure 21), followed by an outsurge due to the drop in primary coolant temperature related to the steam control valve opening. The rapid insurge into the pressurizer causes a rapid pressure rise in the pressurizer (Figure 22) and, thus, throughout the primary coolant system (Figure 23). Although pressurizer spray comes on from 10 to 22 s, system pressure rises high enough that the high-pressure scram setpoint [15.72 MPa (2281 psia)] is reached and the reactor scrams at 12 s. Normalized power is shown in Figure 24. Decay heat levels (best-estimate values) used in the analysis assumed 10 h of irradiation time prior to experiment initiation. The code-default assumption of infinite irradiation time was not used. Figure 25 shows cladding temperature at the midplane of an average-powered fuel rod.

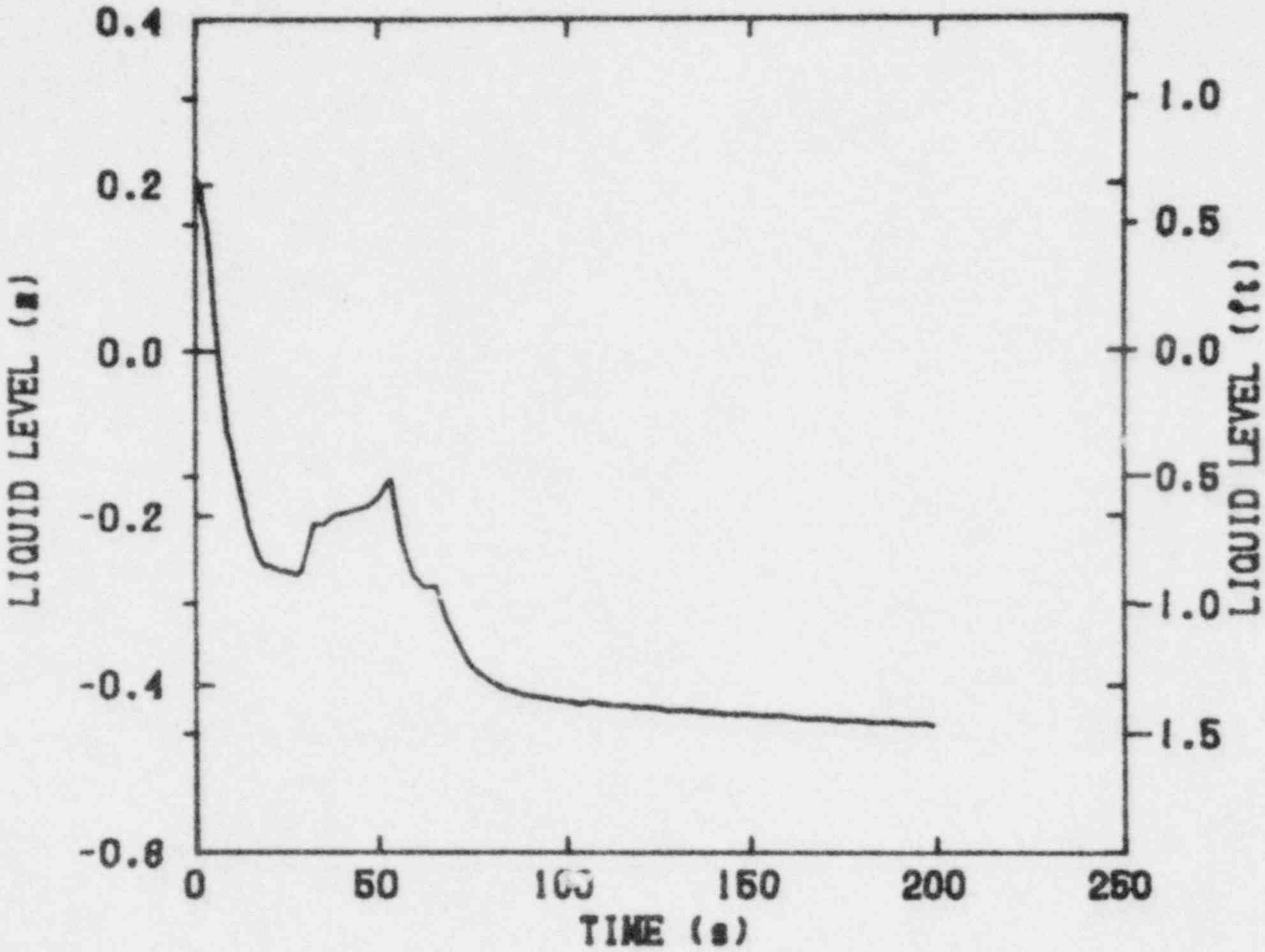


Figure 17. Liquid level in steam generator downcomer for Experiment L6-1.

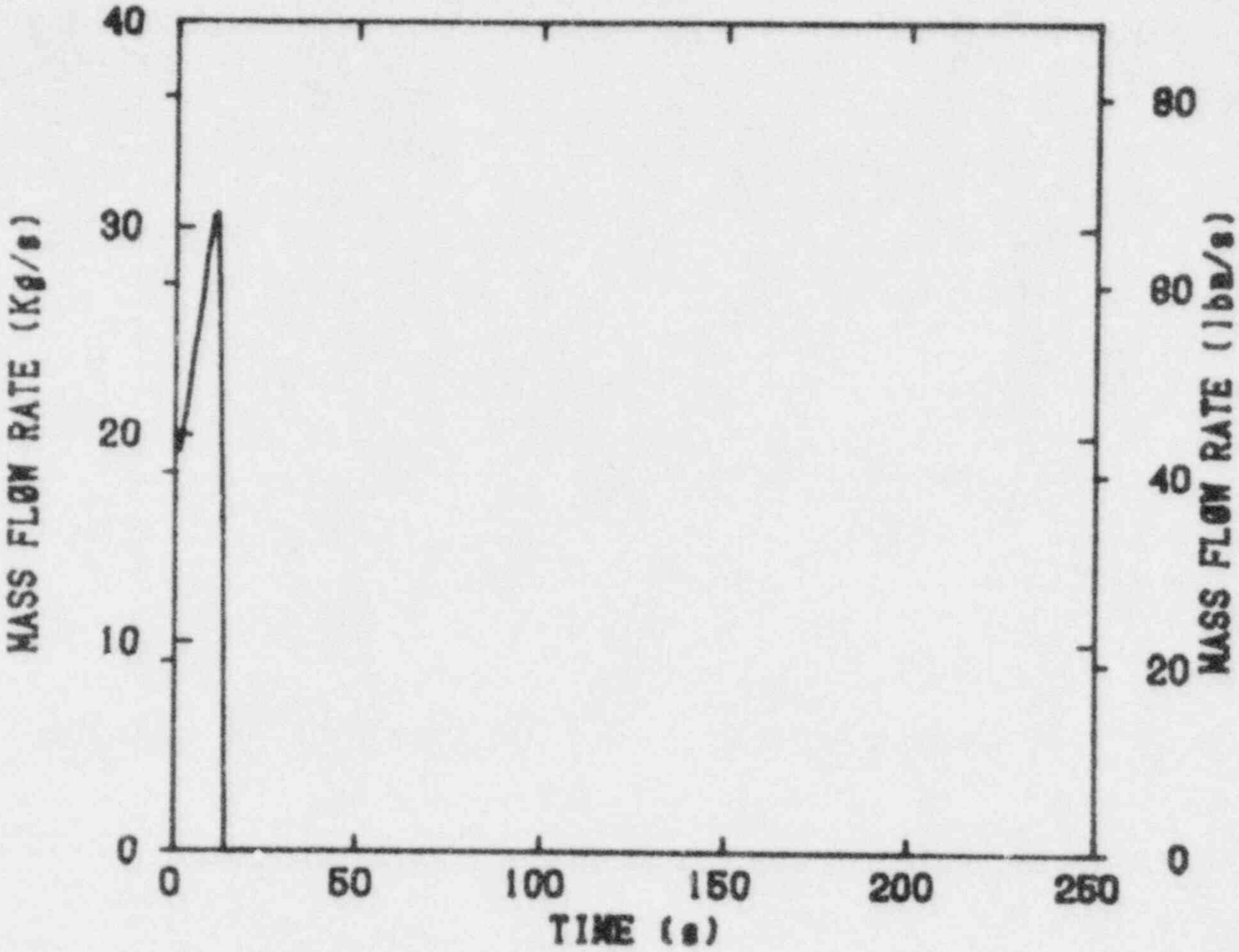


Figure 18. Mass flow rate of steam generator feedwater for Experiment L6-1.

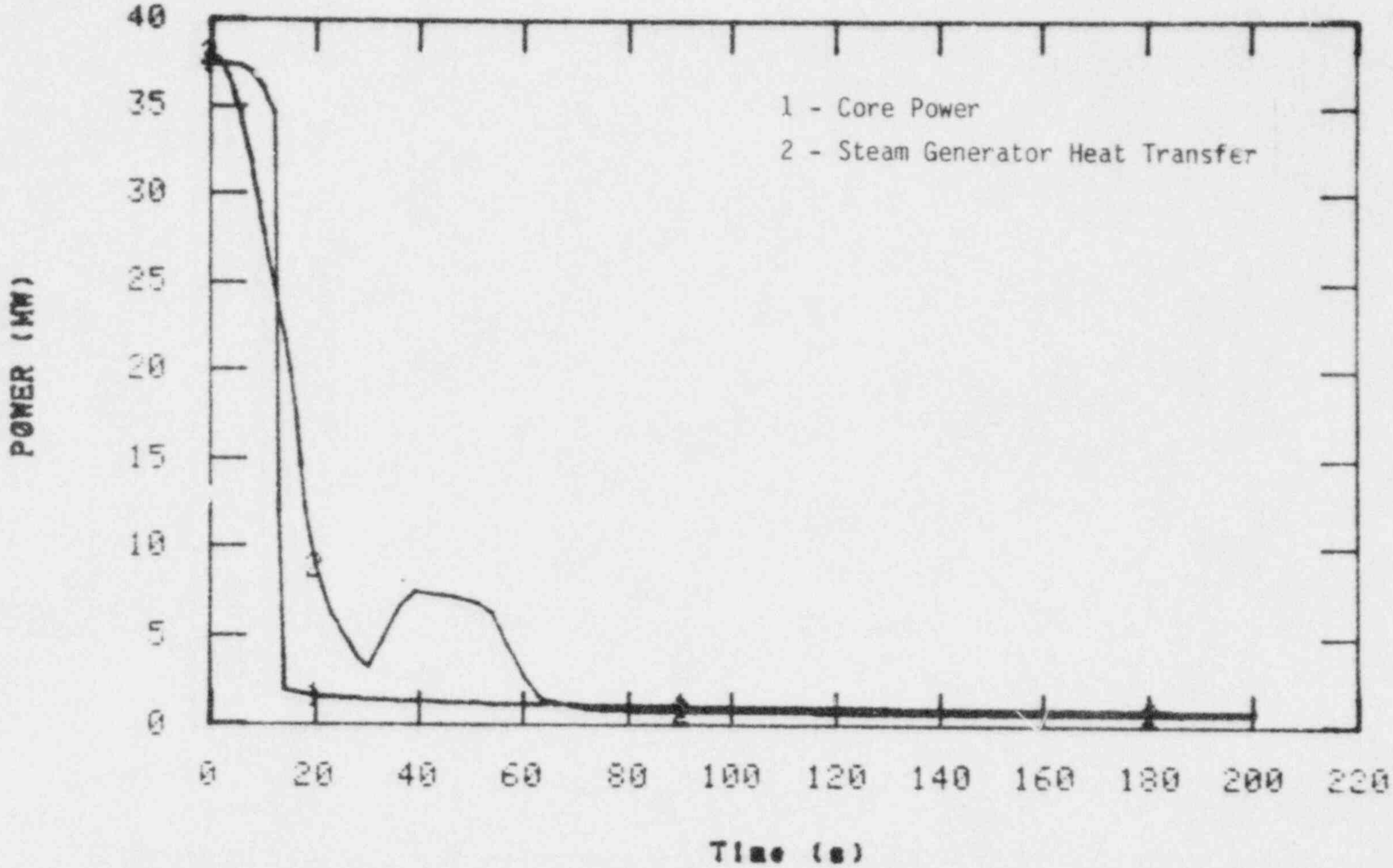


Figure 19. Core power and heat transfer through the steam generator for Experiment L6-1.

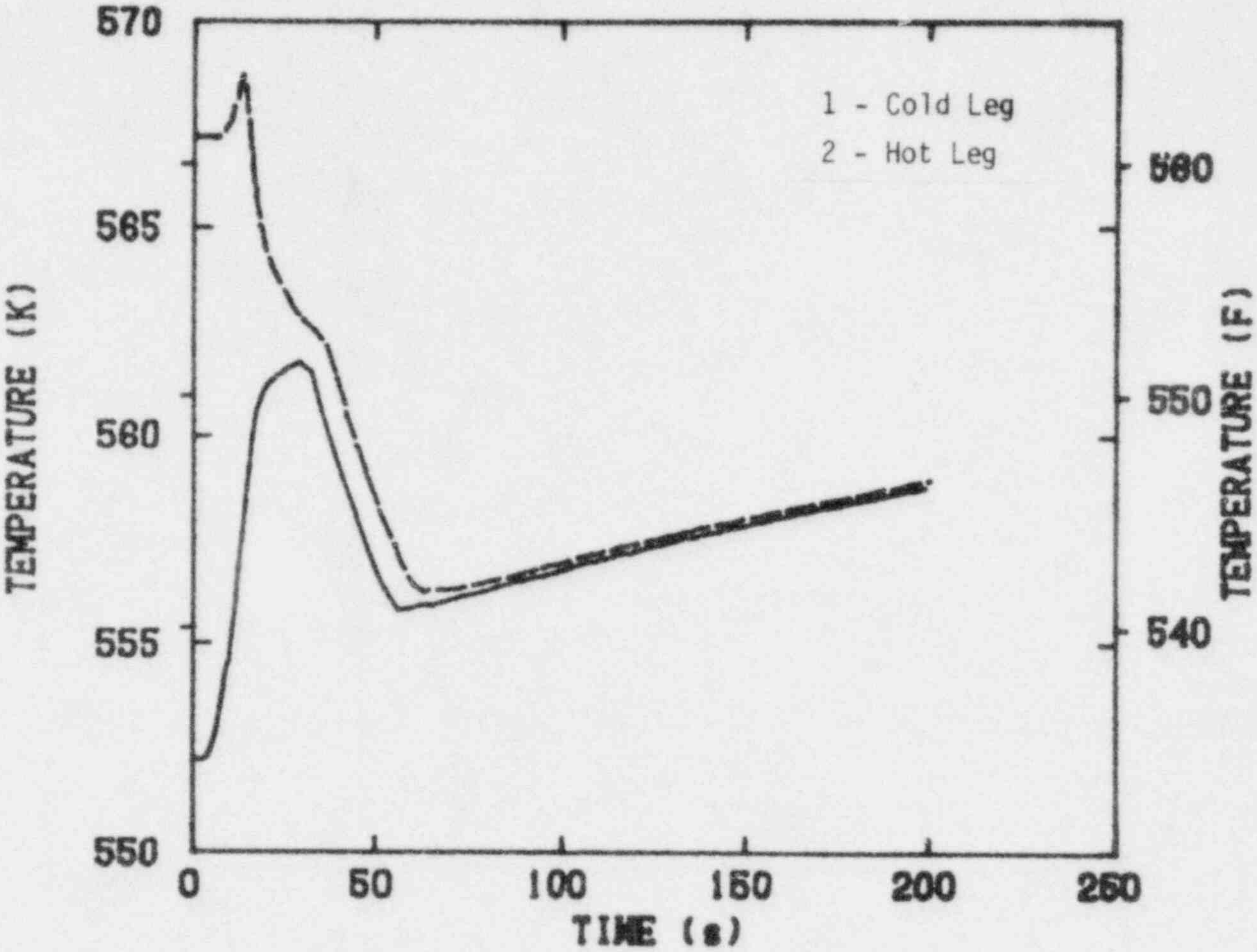


Figure 20. Coolant temperature in intact loop cold and hot legs for Experiment L6-1.

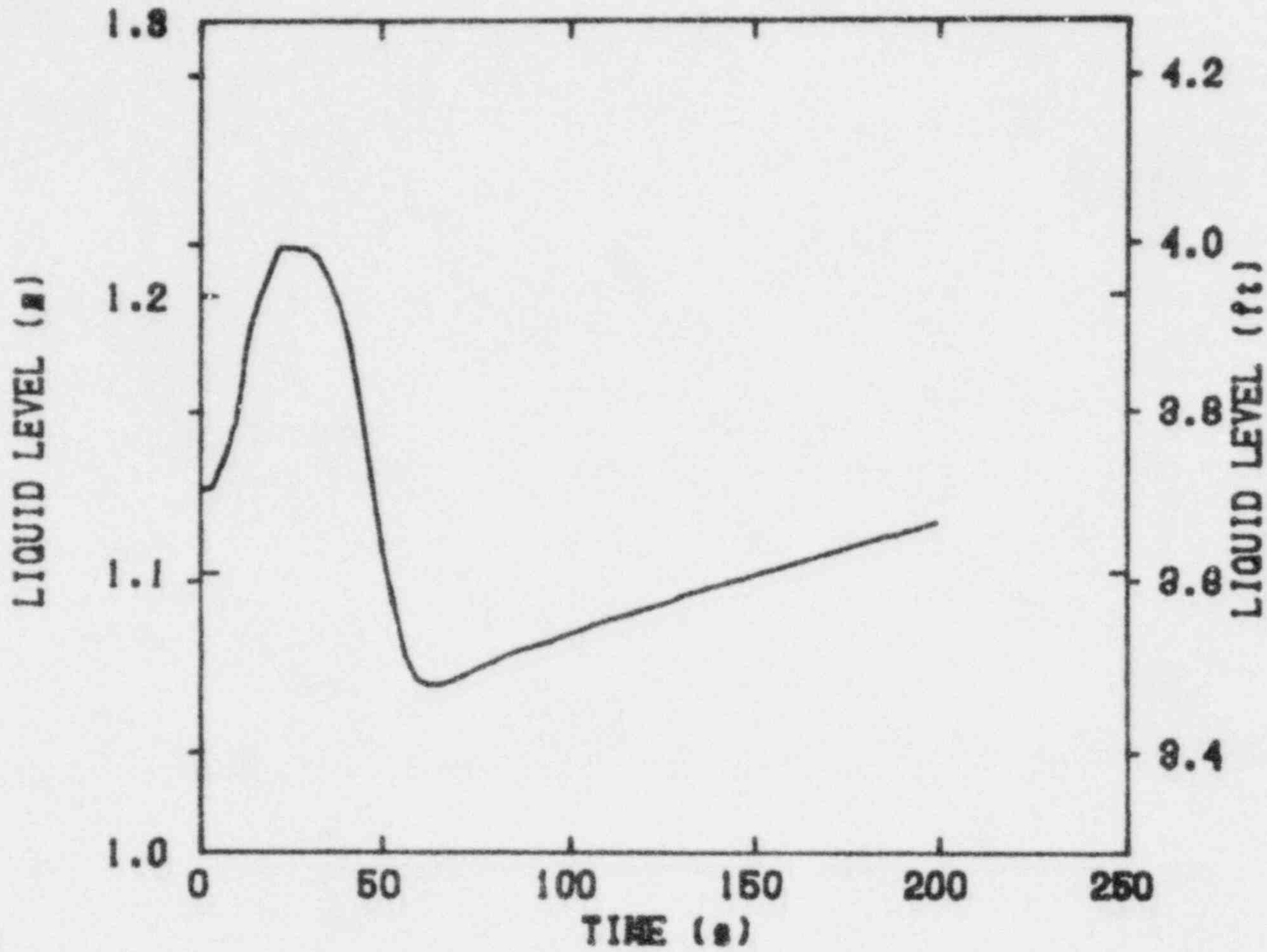


Figure 21. Liquid level in pressurizer for Experiment L6-1.

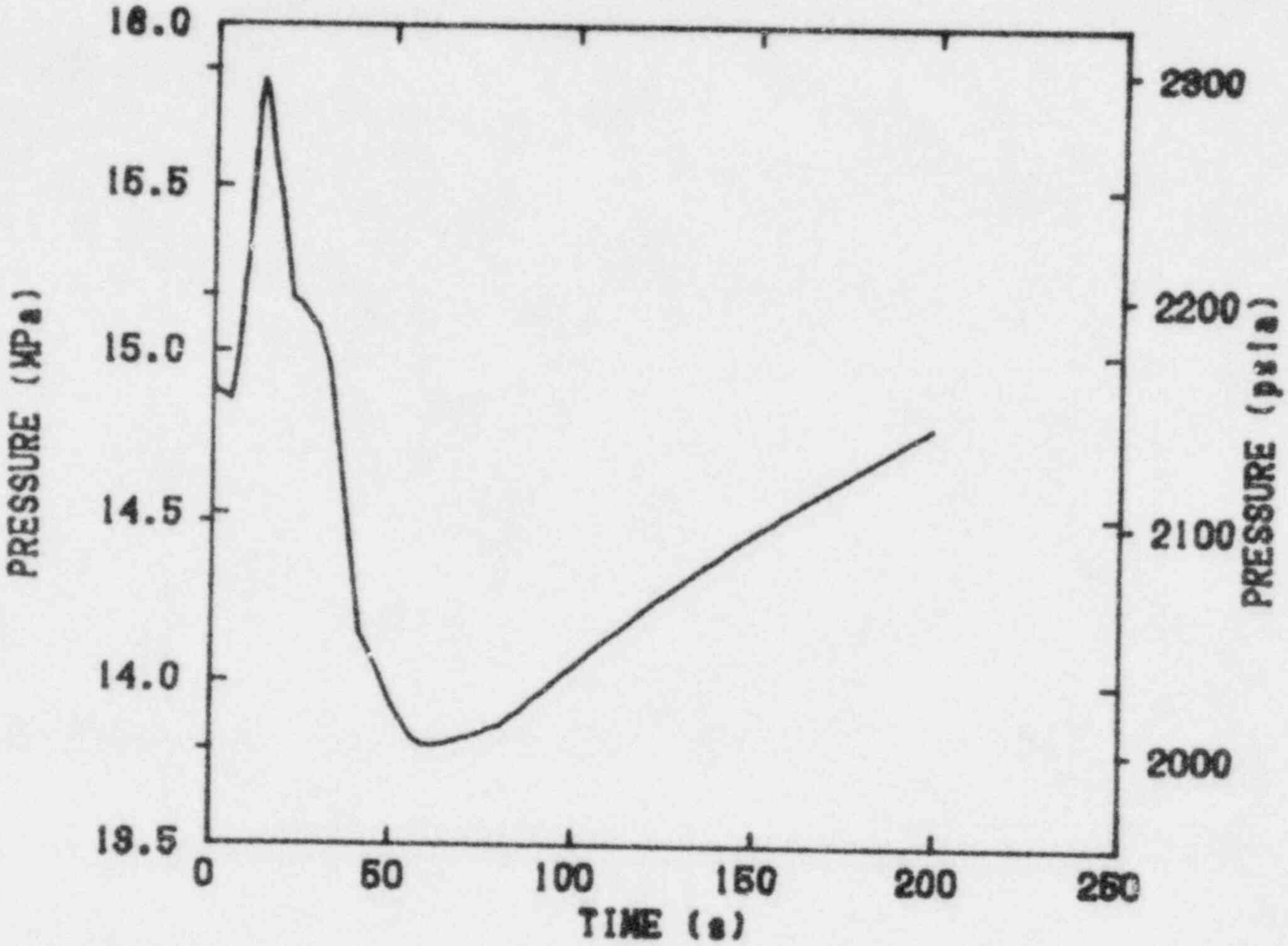


Figure 22. Pressure in pressurizer for Experiment L6-1.

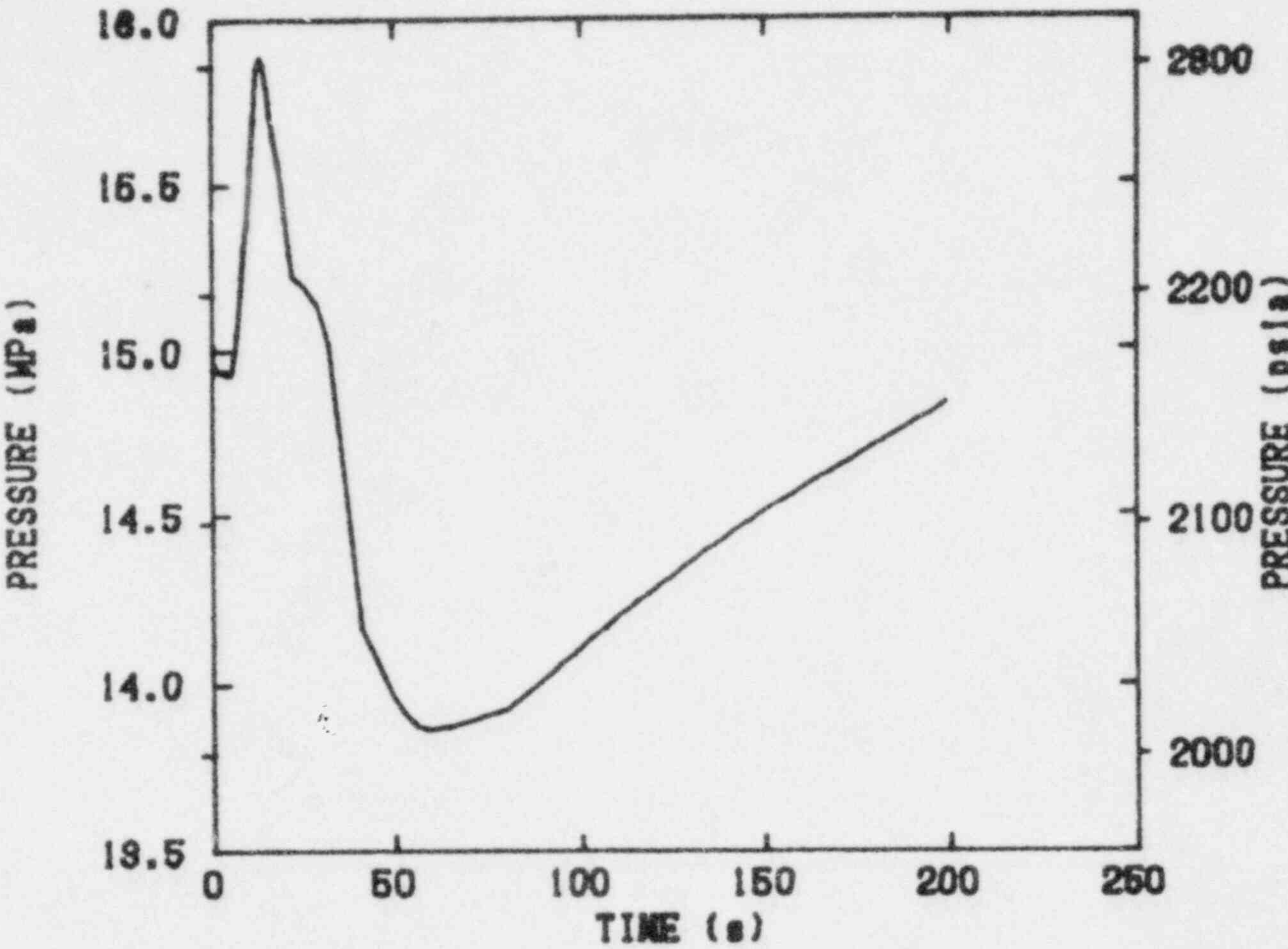


Figure 23. Pressure in reactor vessel upper plenum for Experiment L6-1.

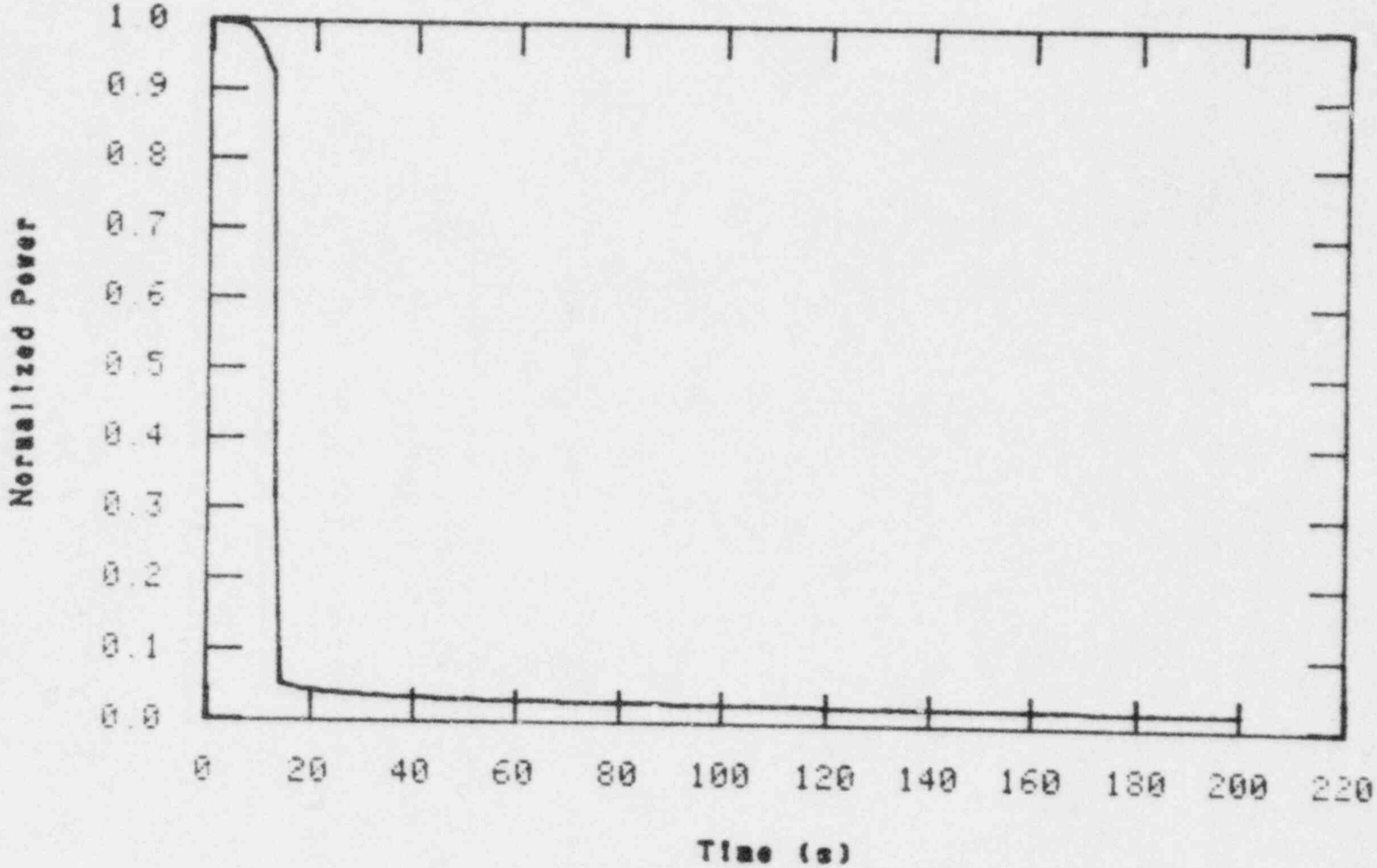


Figure 24. Normalized reactor power for Experiment L6-1.

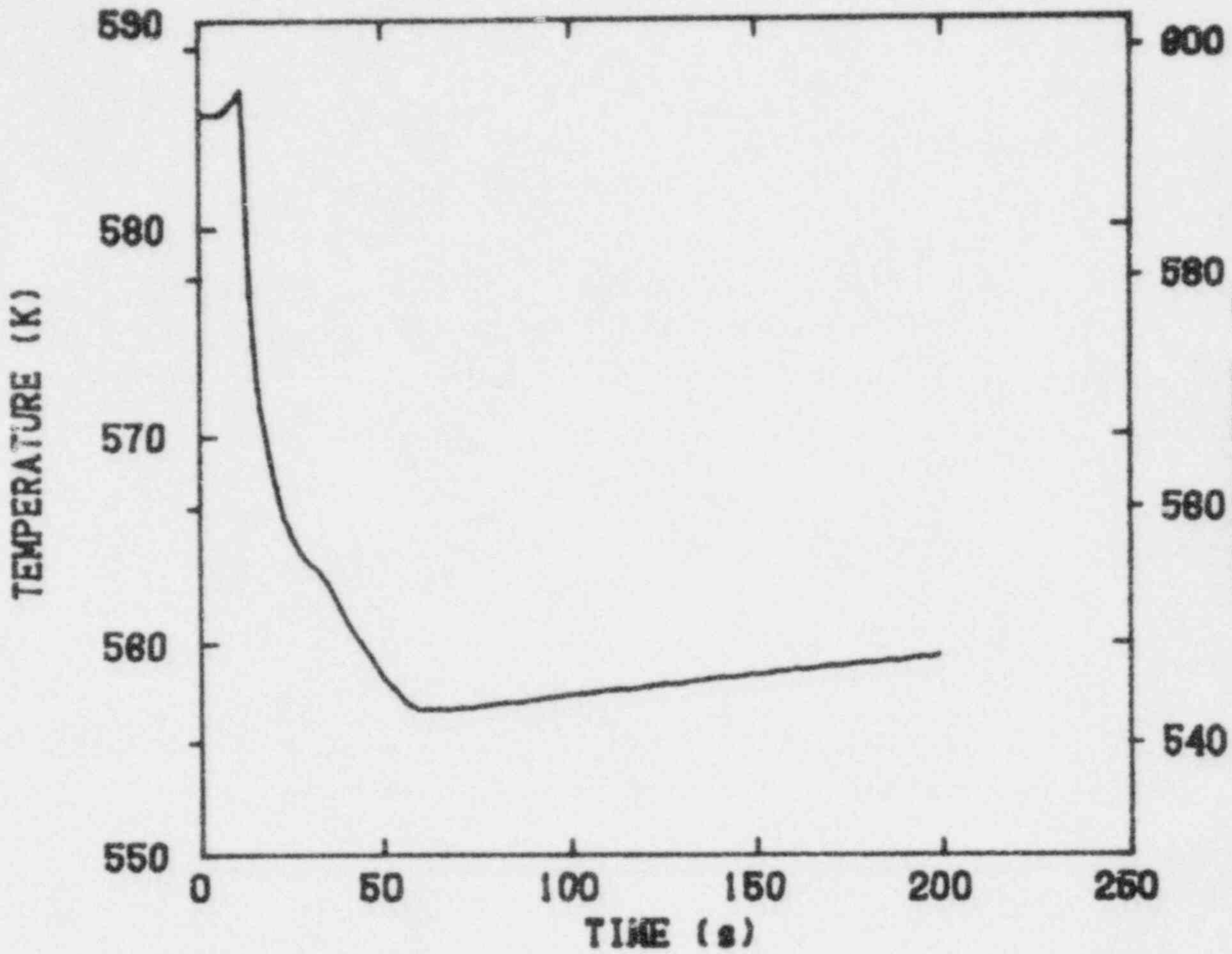


Figure 25. Cladding temperature at midplane of average-powered fuel rod for Experiment L6-1.

After the reactor scrammed and the steam control valve cycled once, average primary coolant temperature started to rise again, gradually, due to decay heat. This rise in temperature caused a gradual insurge into the pressurizer with an attendant pressure rise. The pressure rise after 60 s is probably estimated higher in the analysis than what will occur during the experiment due to the use of the nonequilibrium pressurizer model, a model which tends to thermally isolate the liquid and vapor regions in the pressurizer.

3.2 Experiment L6-2 Prediction

Experiment L6-2 (loss of primary coolant flow) will be initiated by tripping power to the primary coolant pump motor generator sets, allowing the pumps to coast down under the influence of the flywheel system. The sequence of events calculated to occur in this transient is as follows:

Experiment L6-2 Event	Time (s)
Pumps tripped	0.0
Scram on low flow [433 kg/s (952.78 lbm/sec)] and start of feedwater valve and steam control valve closures	1.5
Feedwater flow stops	3.5
Steam control valve closed	13.0

Primary coolant system flow starts to drop as soon as the pumps start to coast down. At 1.5 s, the low flow scram setpoint in the primary coolant system is reached and the reactor is scrammed. Figure 26 shows the primary coolant system flow rate. Decay heat levels (Figure 27) used in this analysis assumed 15 h of irradiation prior to experiment initiation.

Figures 28 and 29 show feedwater and steam flow rates, respectively. The valves controlling these flows start to close coincident with the reactor scram signal. As soon as the steam control valve starts to close,

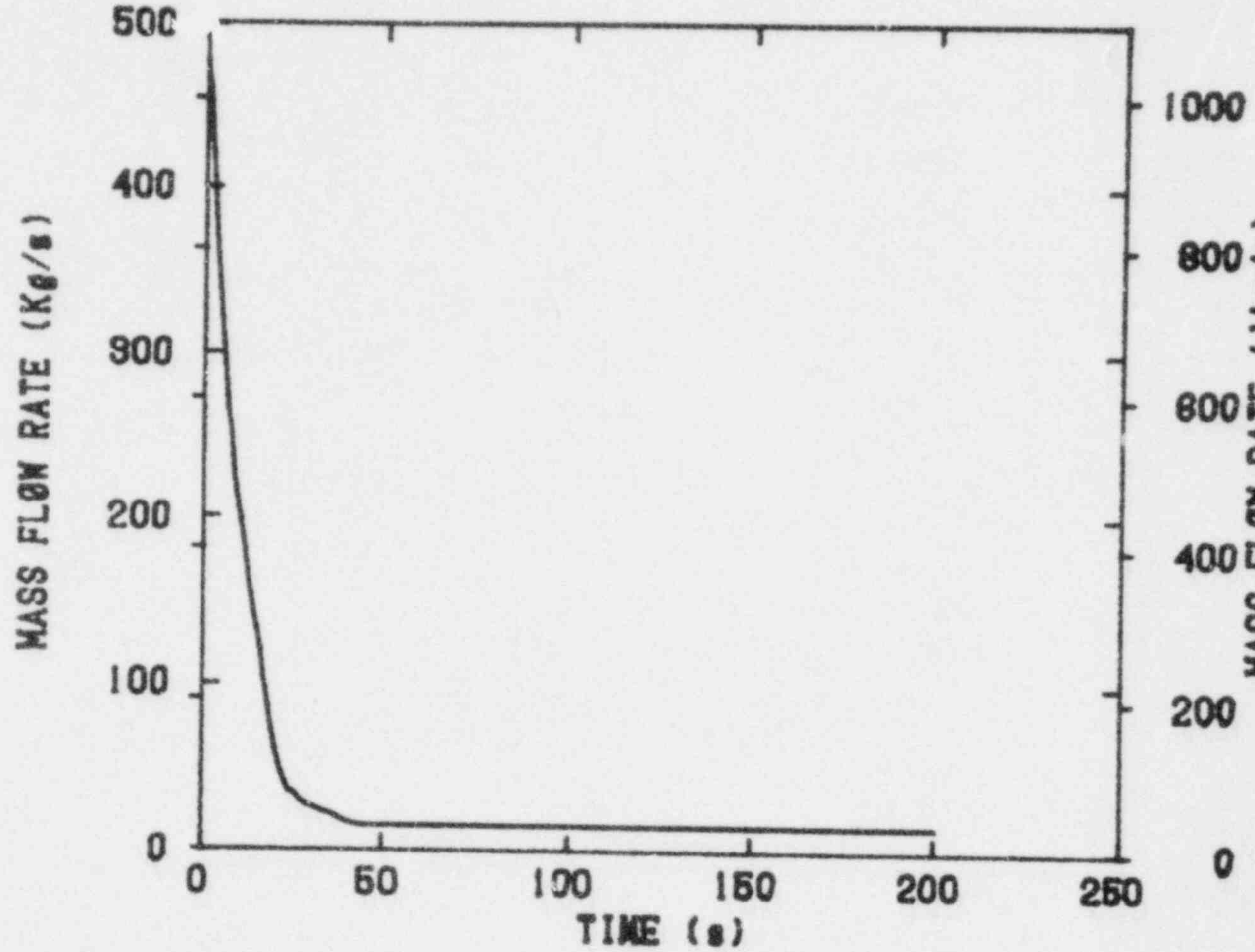


Figure 26. Mass flow rate in primary coolant system for Experiment L6-2.

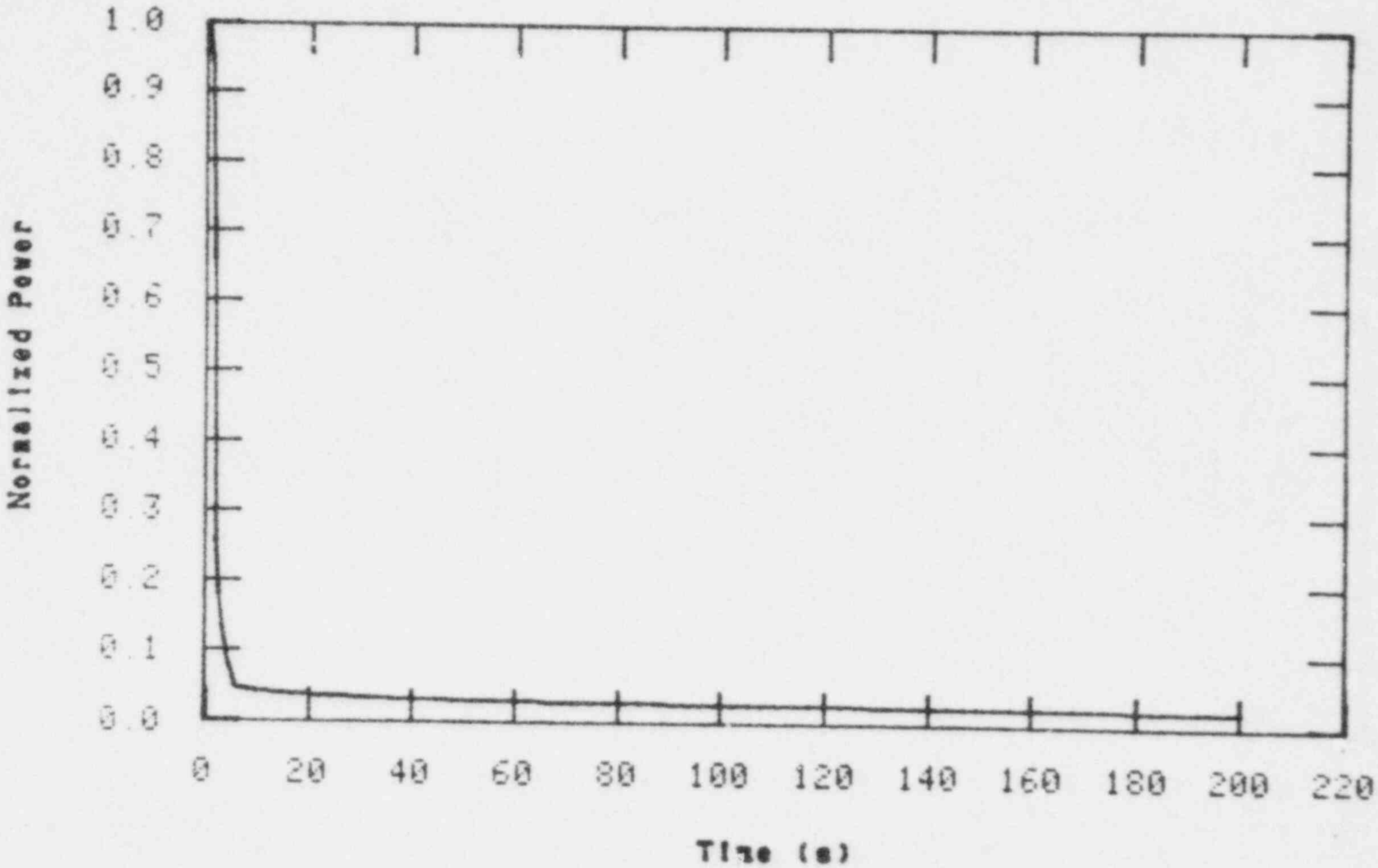


Figure 27. Normalized reactor power for Experiment L6-2.

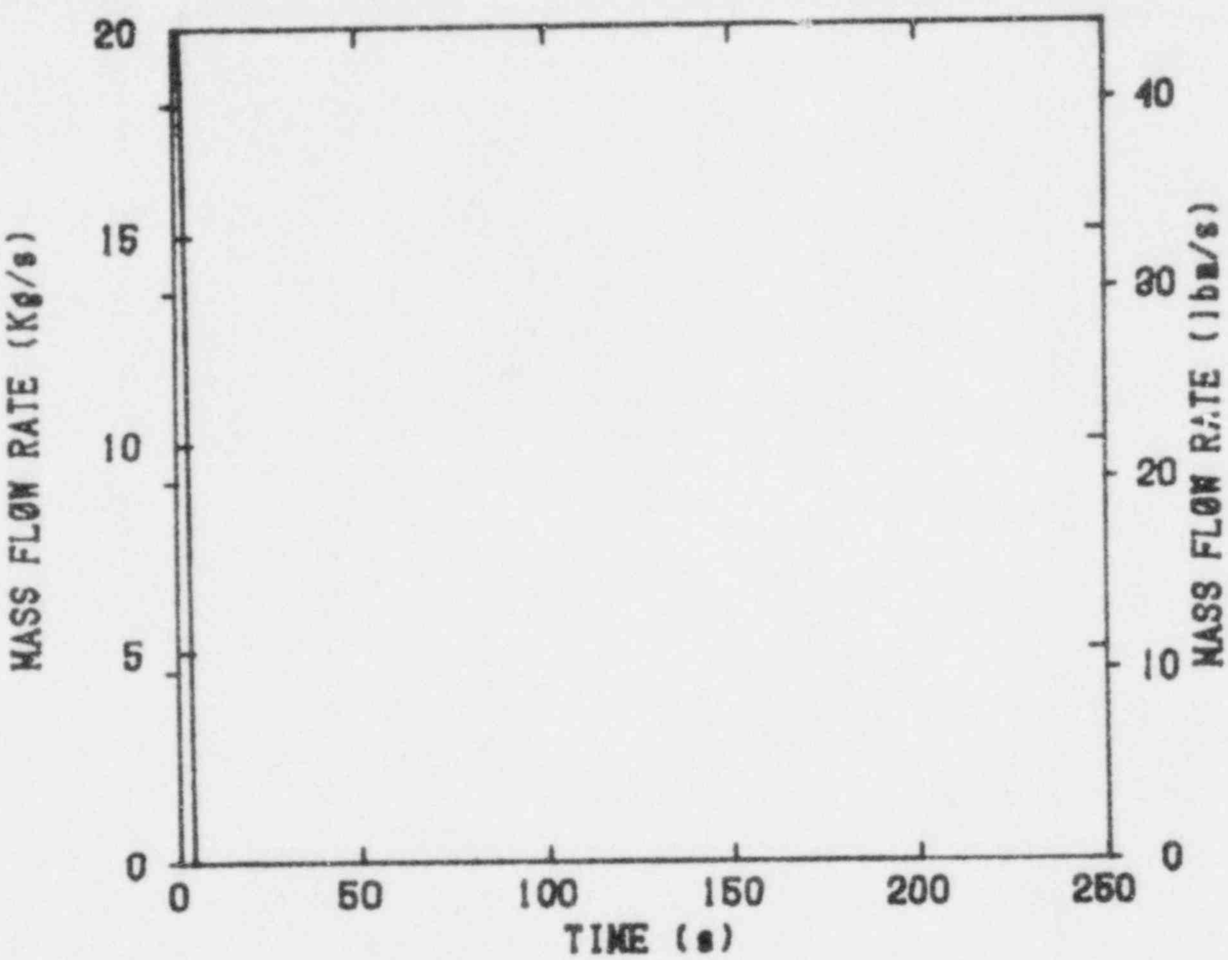


Figure 28. Mass flow rate of steam generator feedwater for Experiment L6-2.

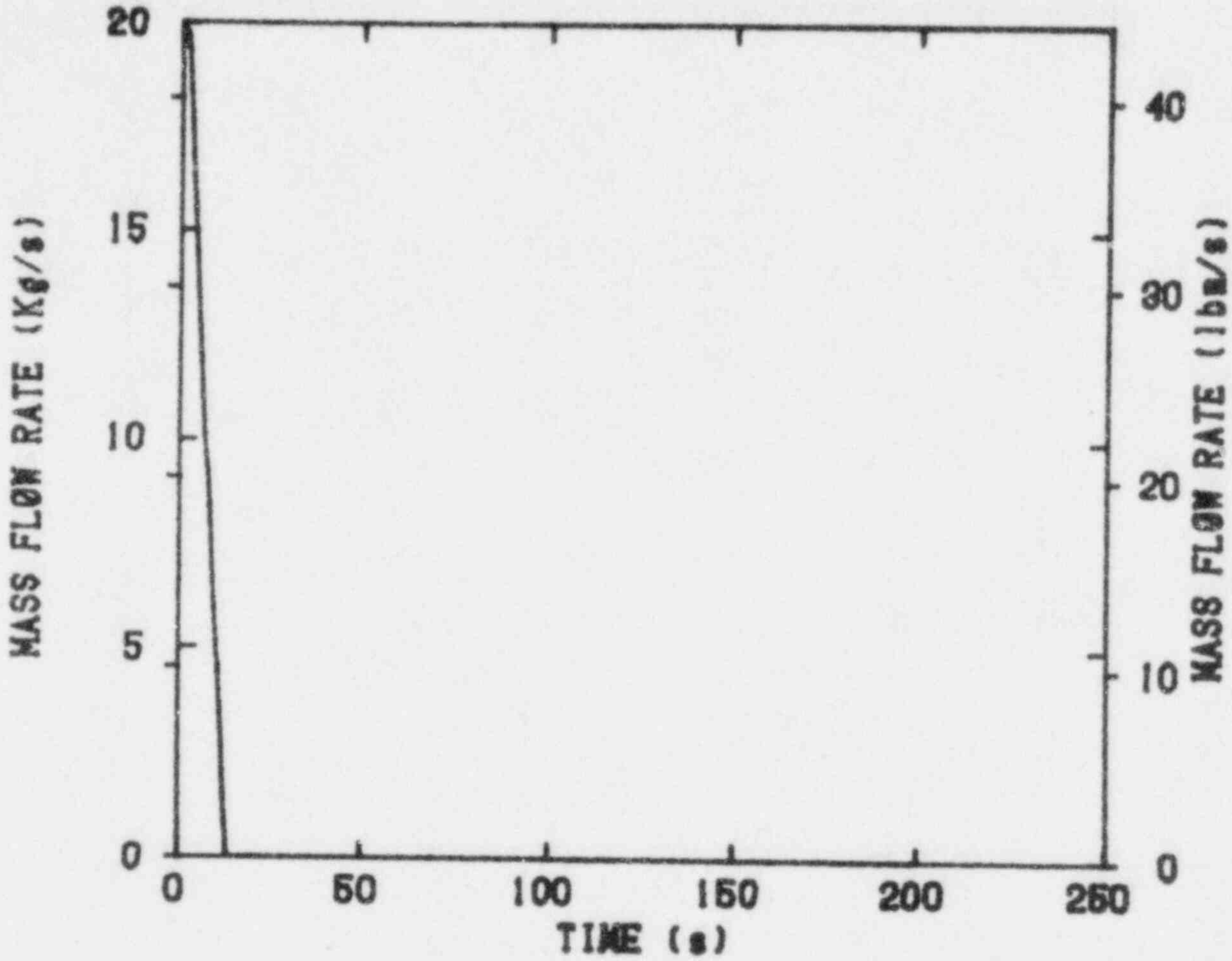


Figure 29. Mass flow rate through steam control valve for Experiment L6-2.

pressure in the secondary side of the steam generator (Figure 30) starts to rise. Since the reactor has scrammed and decay heat levels are relatively low, secondary pressure does not get high enough for the steam control valve to automatically open during the first 200 s of the experiment, which is the duratiom of this analysis. As secondary pressure rises, the mixture level inside the steam generator shroud drops, with a corresponding drop in steam generator downcomer liquid level (Figure 31). In this experiment, feedwater flow will be shut off at this time, so changes in downcomer liquid level have no effect on feedwater valve position.

Fluid temperatures in the primary coolant (Figure 32) drop due to scram at 1.5 s. Average primary coolant temperature doesn't start to rise until approximately 26 s, when total heat transfer across the steam generator tubes drops below decay heat levels (Figure 33). The effect of the initial drop and later rise in average primary coolant temperature may be seen in the behavior of the pressurizer liquid level (Figure 34); that is, the sudden drop in primary coolant temperature following scram causes a rapid outsurge, and the slower rise in temperature later causes a slower insurge into the pressurizer. These fluctuations in pressurizer liquid level cause changes in both pressurizer and upper plenum pressures (Figures 35 and 36, respectively). The pressure rise after 26 s is considered to be too high due to the nonequilibrium pressurizer model, as mentioned previously. The model tends to predict faster pressure rises attending slow insurges into the pressurizer than experimental data show.

Cladding temperature at the midplane of an average-powered fuel rod is shown in Figure 37. The cladding temperature rises somewhat after 20 s due to a reduction in surface heat transfer coefficient attending the loss of forced flow through the core.

3.3 Experiment L6-3 Prediction

Experiment L6-3 (excessive load increase) will be initiated by opening the main steam control valve at the rate of 5% per second (valve stem travel). The sequence of events calculated to occur in this transient is as follows:

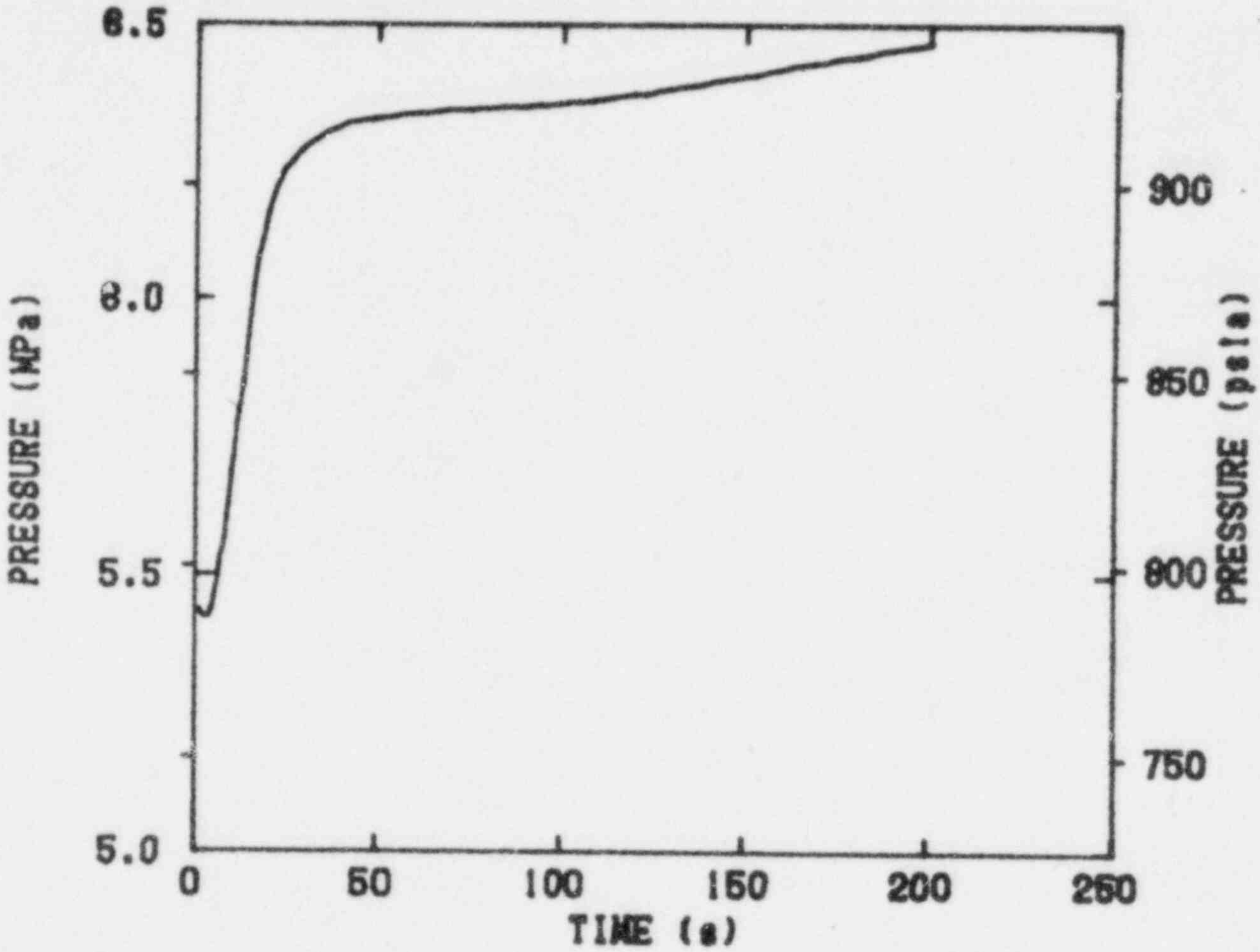


Figure 30. Pressure in steam generator secondary side for Experiment L6-2.

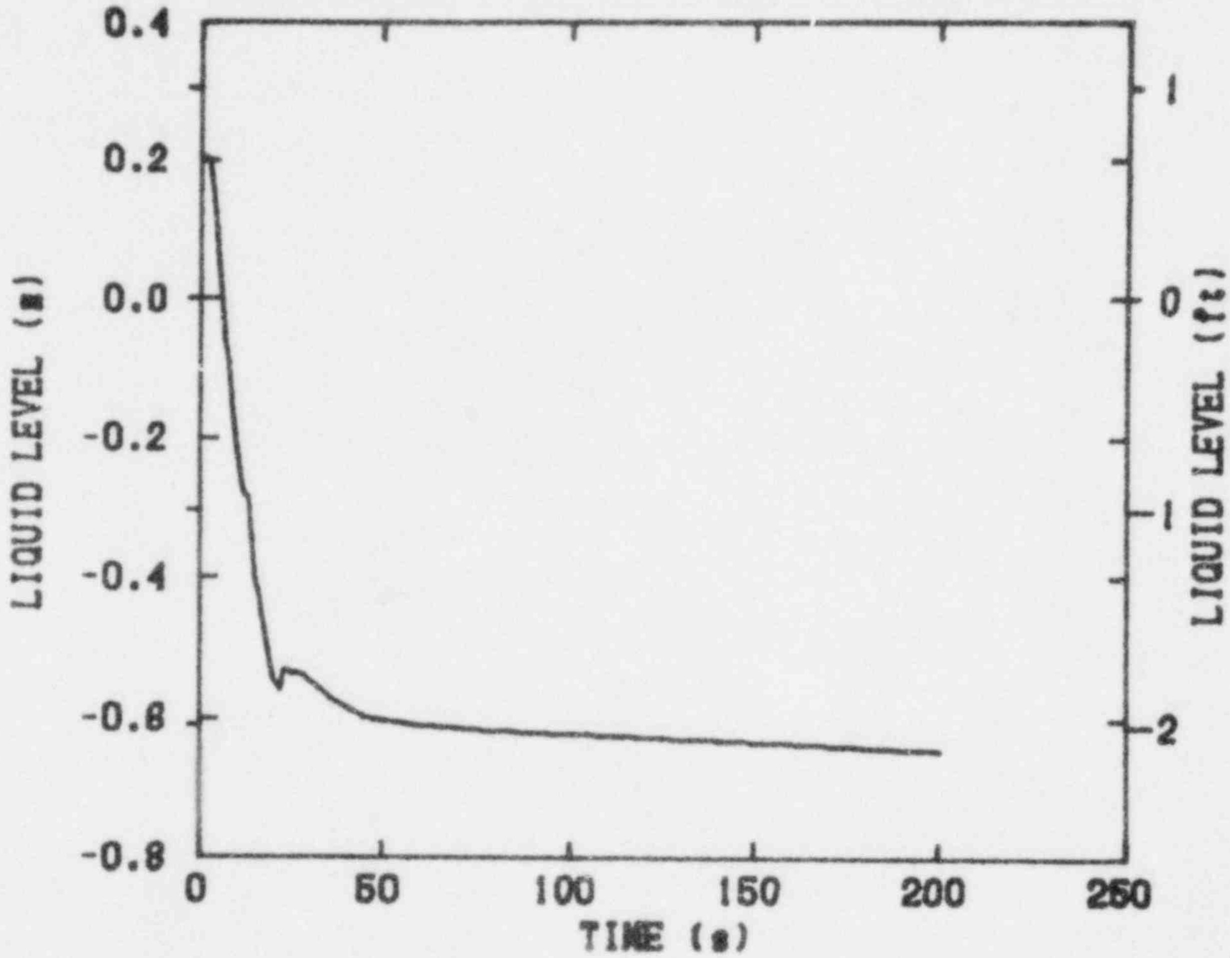


Figure 31. Liquid level in steam generator downcomer for Experiment L6-2.

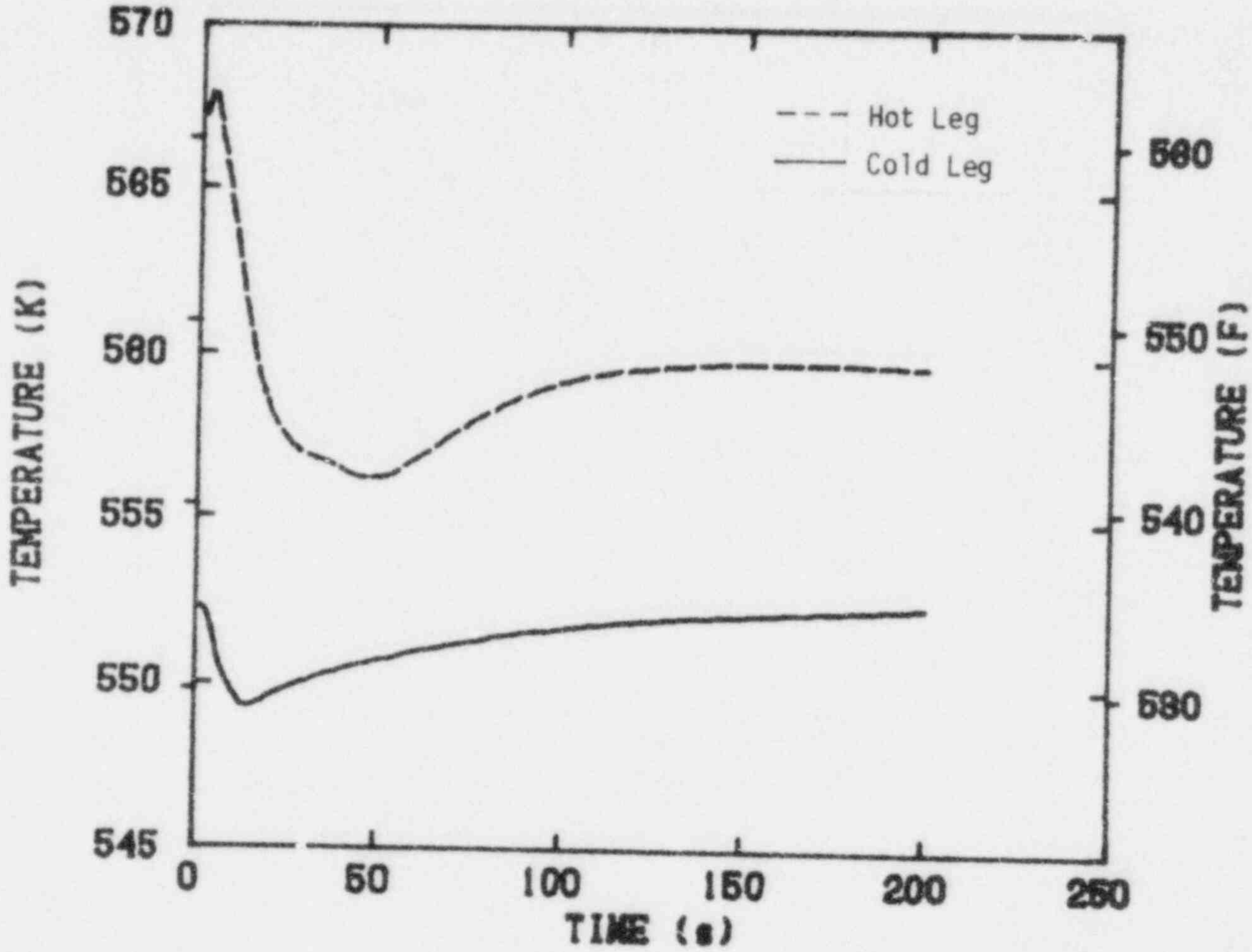


Figure 32. Coolant temperature in intact loop cold and hot legs for Experiment L6-2.

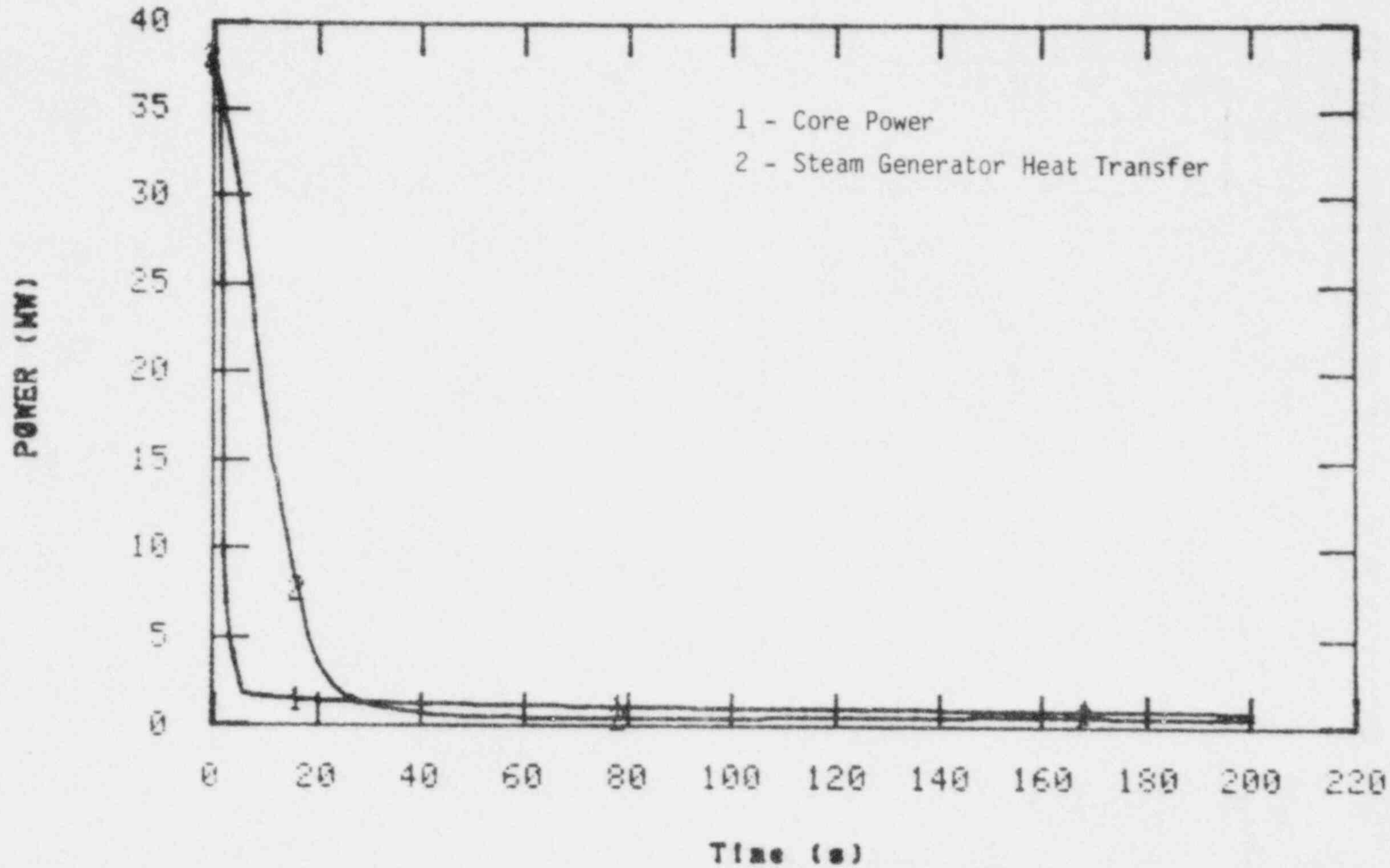


Figure 33. Core power and heat transfer through the steam generator for Experiment L6-2.

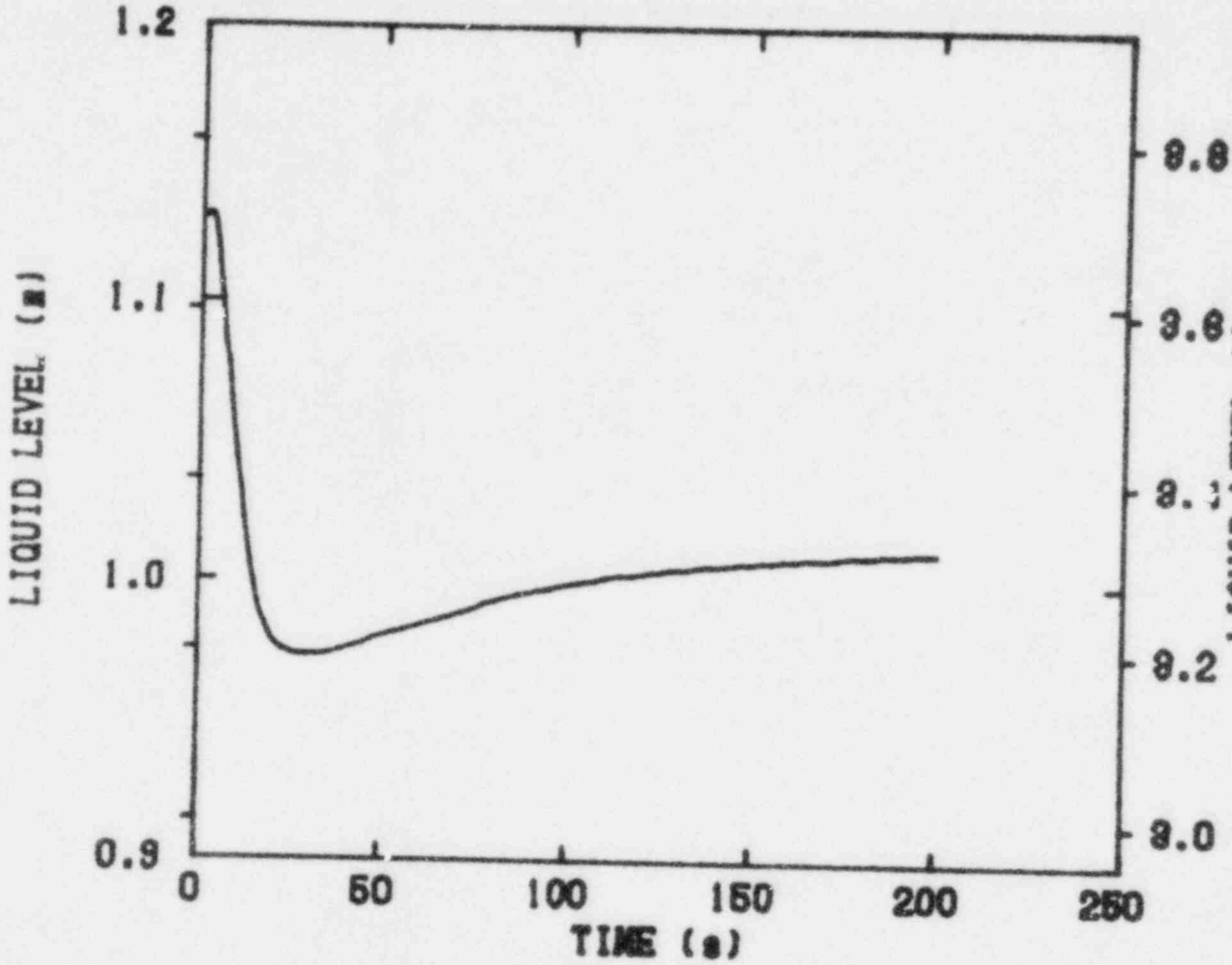


Figure 34. Liquid level in pressurizer for Experiment L6-2.

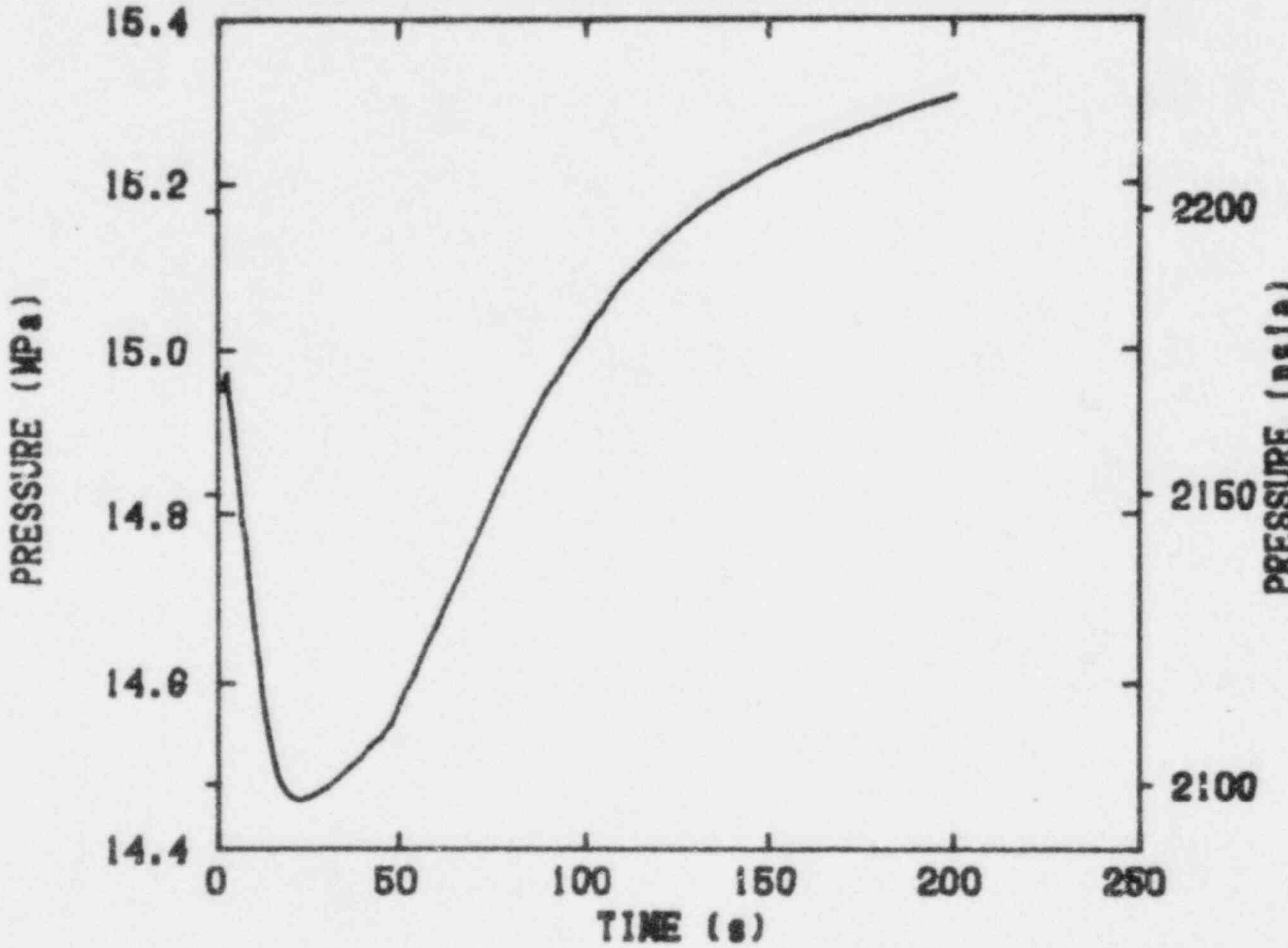


Figure 35. Pressure in pressurizer for Experiment L6-2.

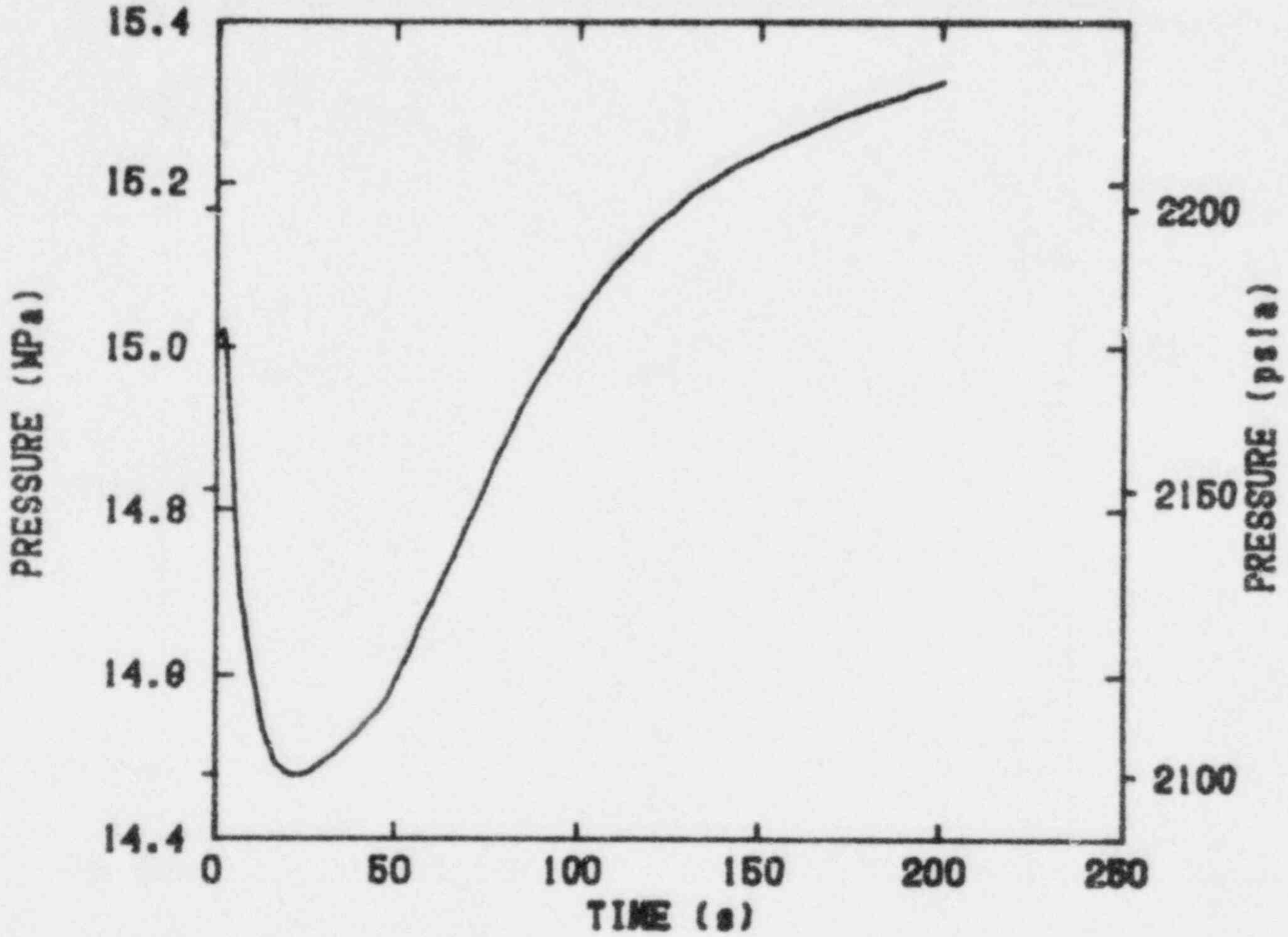


Figure 36. Pressure in reactor vessel upper plenum for Experiment L6-2.

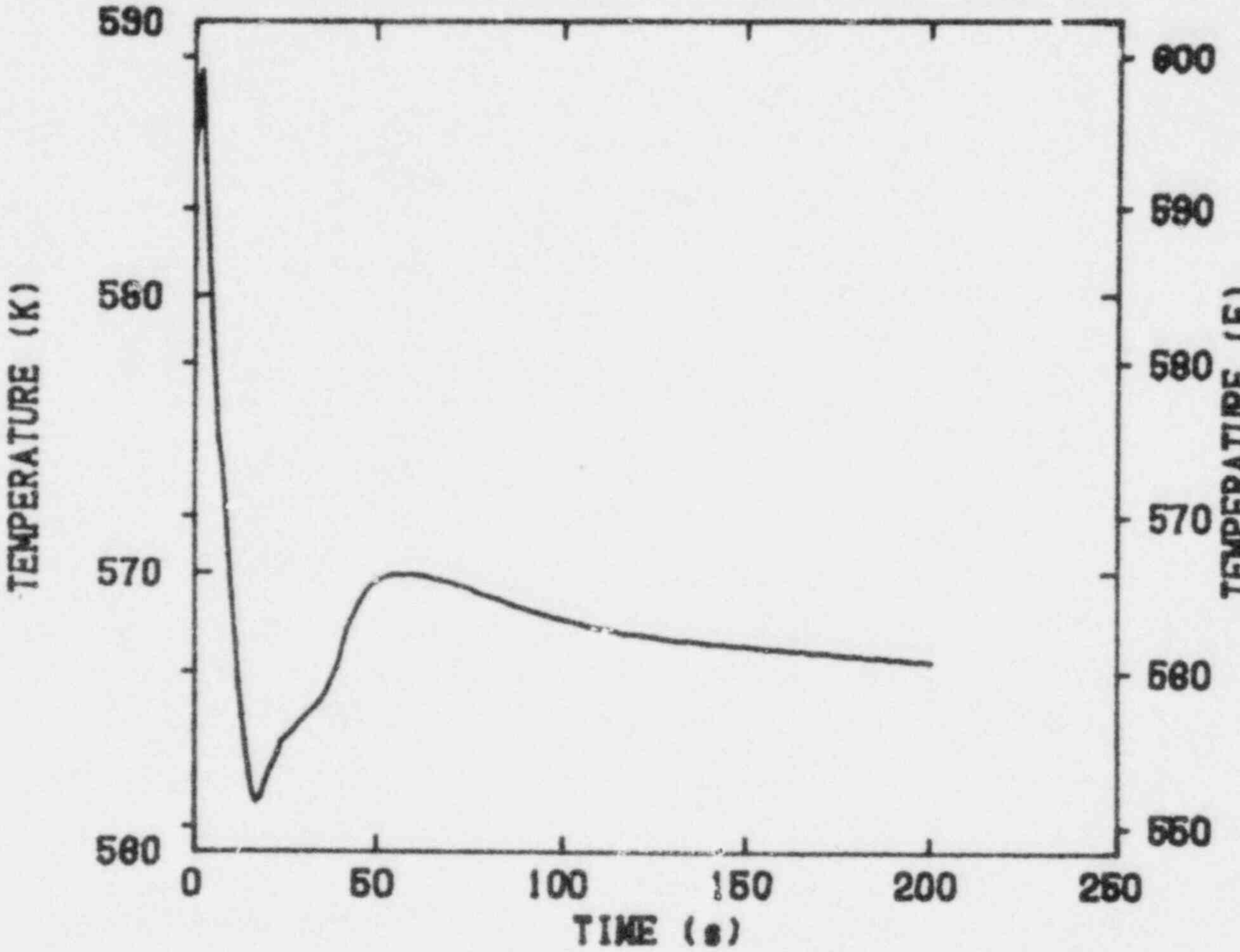


Figure 37. Cladding temperature at midplane of average-powered fuel rod for Experiment L6-2.

Experiment L6-3 Event	Time (s)
Steam control valve starts to ~~close~~ open	0.0
Feedwater valve starts to close	2.0
Pressurizer heaters on	7.7 to 51
Steam valve completely open	8.5
Steam generator heat transfer peaks	12.0
Feedwater valve starts opening	13.0
Steam flow rate peaks	17.0
Core power peaks	17.0
Pressurizer spray on	90 to 96

As the steam control valve opens, steam flow increases as shown in Figure 38. The momentary drop in steam flow rate from 8.5 to 13 s seems to be related to the shutoff of feedwater during that time, and will likely not occur during the experiment. By 8.5 s, the steam control valve is fully open, but steam flow does not peak until 17 s; steam flow rate being a function of both secondary pressure and valve flow area.

During the first 40 s of the transient, competing phenomena determine the state of the secondary system and, consequently, the primary system also. The steam flow rate increases with increasing steam valve flow area, but decreases due to dropping secondary pressure (Figure 39). Dropping pressure and temperature in the steam generator secondary side increase the temperature differential (ΔT) across the tubes, increasing heat transfer across the tubes. The increased heat transfer tends to moderate the pressure drop on the secondary side. Conditions on the secondary side are further complicated by changes in feedwater flow. The feedwater valve controller will be in automatic mode during the transient. As the steam control valve opens and pressure starts to fall, flashing inside the shroud

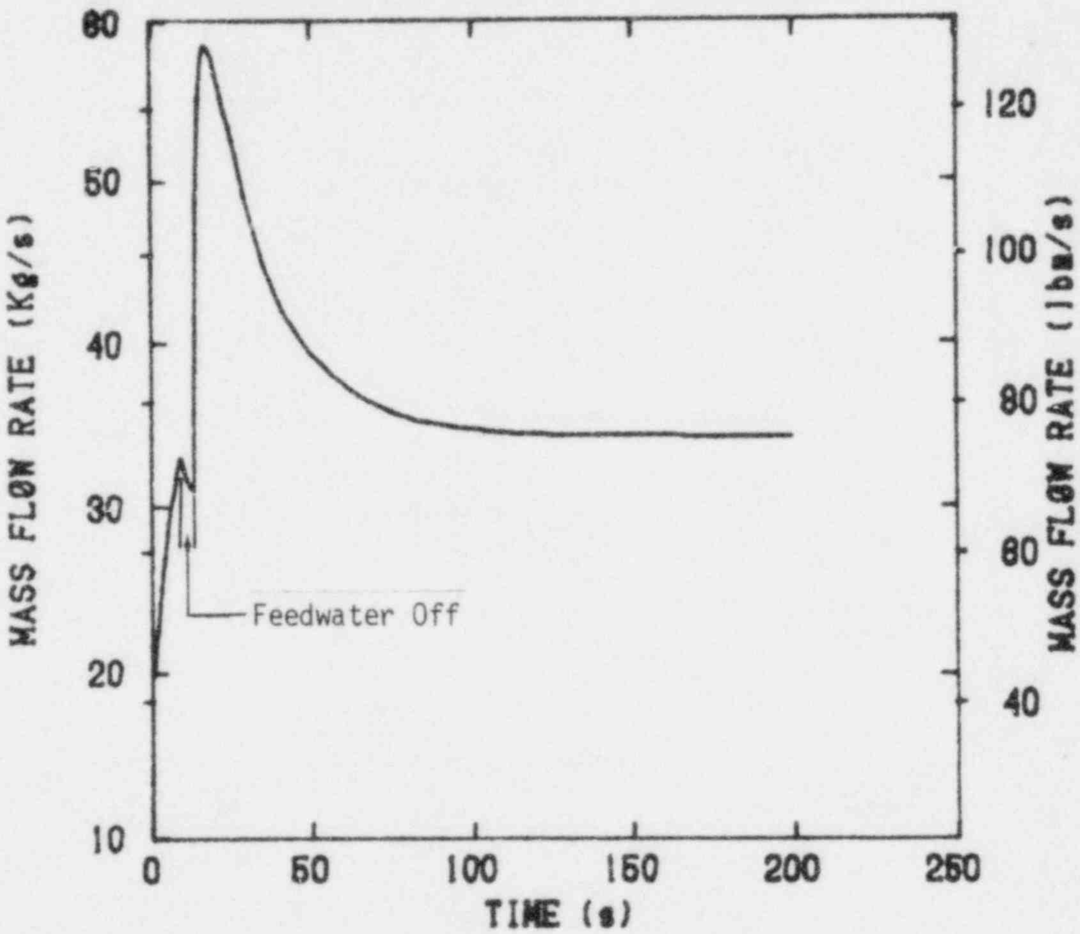


Figure 38. Mass flow rate through steam control valve for Experiment L6-3.

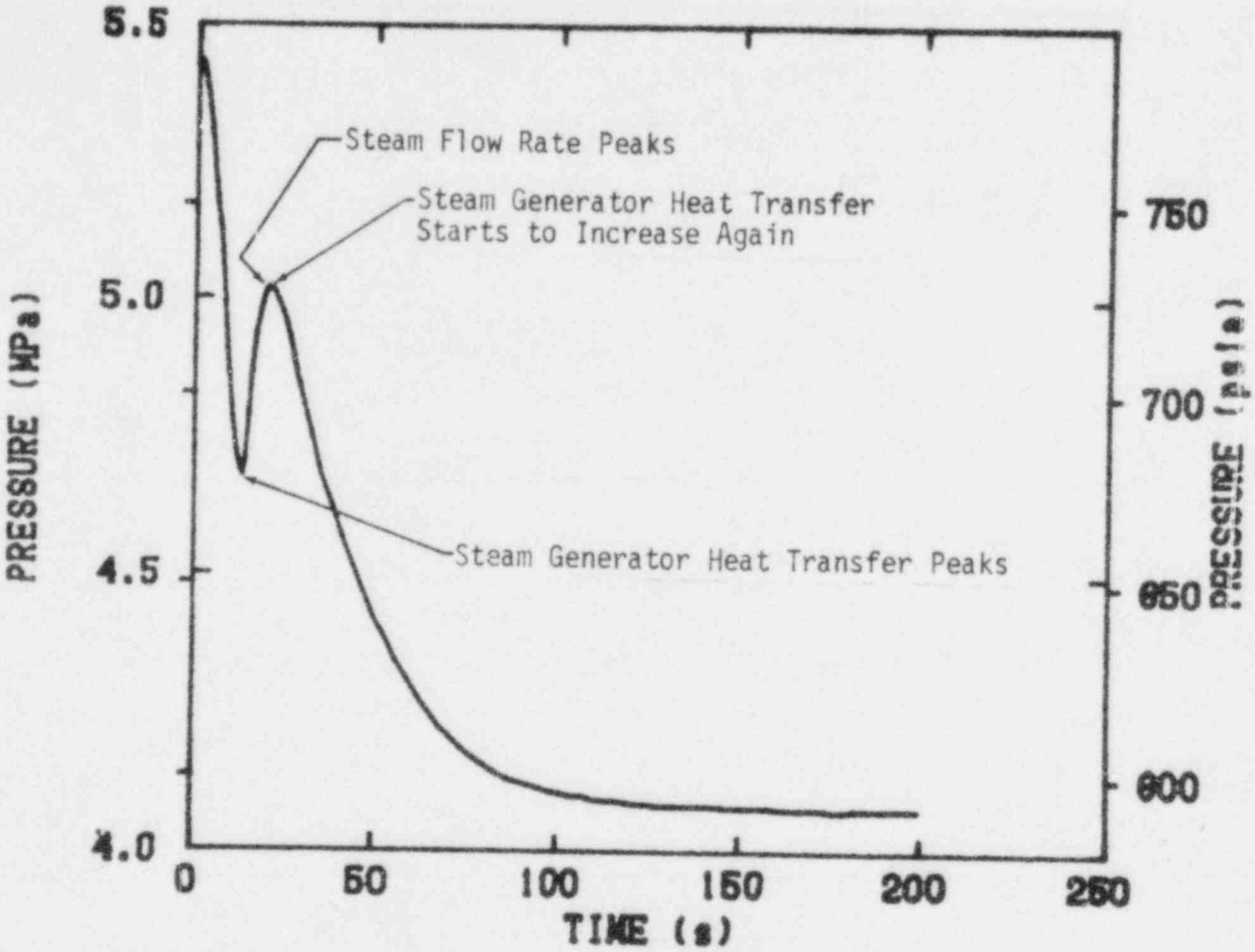


Figure 39. Pressure in steam generator secondary side for Experiment L6-3.

will displace some liquid in the steam generator downcomer, and downcomer liquid level will rise initially (Figure 40). This is essentially the same phenomenon observed in an up-power maneuver. The rise in downcomer liquid level, in turn, will signal the feedwater controller to shut off feedwater flow (Figure 41). The loss of feedwater aids the depletion of secondary fluid mass. Downcomer liquid level eventually drops, and main feedwater comes on automatically at 13 s. Even though feedwater is on after 13 s, secondary side liquid mass inventory (Figure 42) continues to drop until approximately 100 s.

Secondary conditions during this transient are not a function of any one dominant phenomenon. The inflection points in the curve of secondary pressure (Figure 39), for instance, coincide with the inflection points of total steam generator heat transfer (Curve 2 of Figure 43). Steam generator heat transfer, in turn, is a function of all the phenomena mentioned in the previous paragraph.

The action of the feedwater valve is important in this experiment. If the feedwater flow rate was to increase at the start of the transient, instead of decrease as shown in this analysis, the results of the transient would be much different. An initial increase of feedwater flow would increase the temperature differential across the steam generator tubes, and give a more severe cooldown transient. In such a case, there would possibly be enough density reactivity feedback to raise core power to the high-power scram setpoint. This analysis does not predict a scram during the first 200 s of the transient.

Figure 43 shows that steam generator heat transfer is higher than core heat transfer from 0 to 17 s. During this time, fluid temperatures in the primary coolant are dropping (Figure 44). It is this drop in temperature that causes reactor power to rise as a result of fluid density feedback.

As average primary coolant fluid temperature drops, initially there is an attendant outsurge from the pressurizer (Figure 45). A later increase in temperature causes a more-gradual insurge. The pressure changes resulting from these fluctuations in pressurizer liquid level may be seen

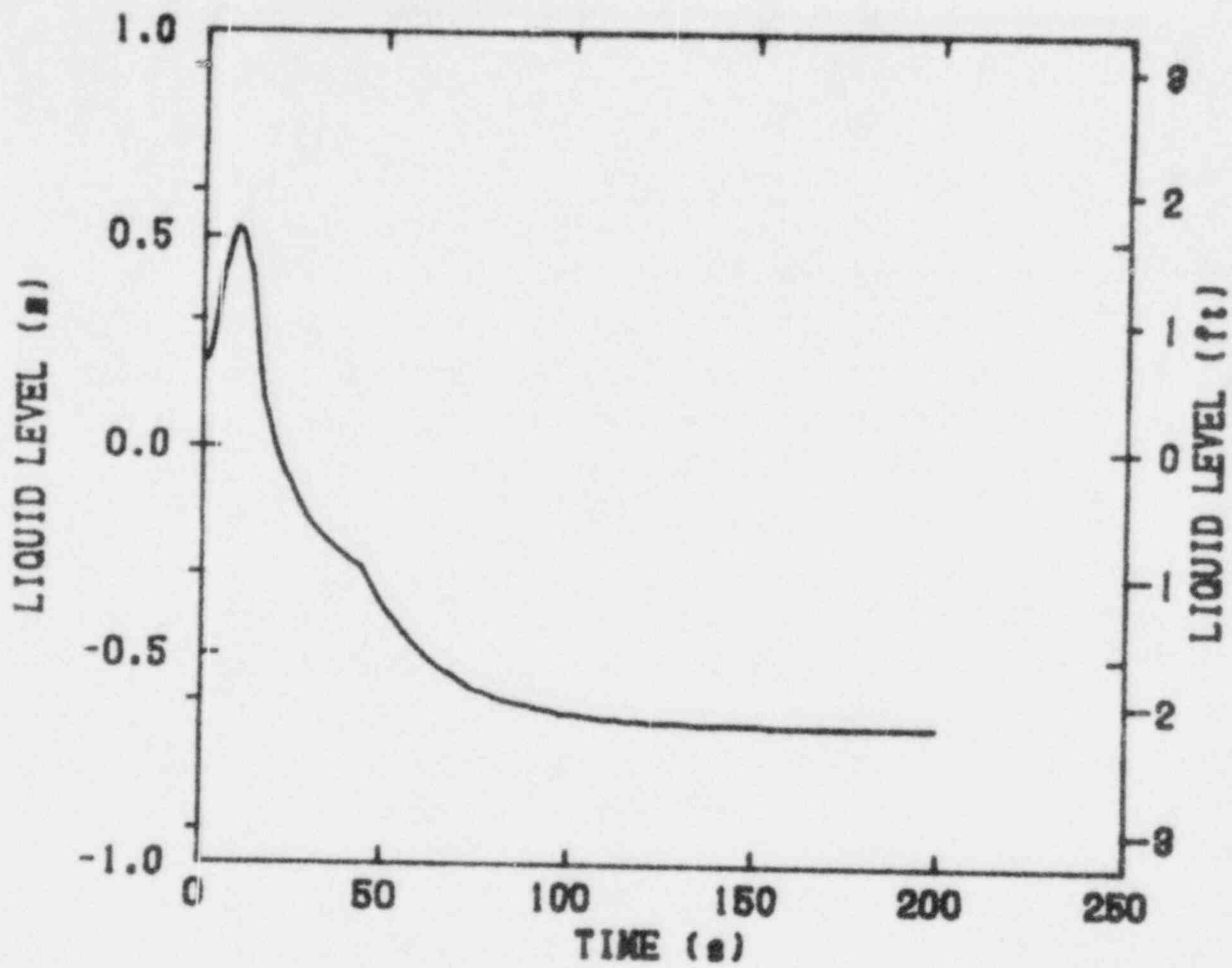


Figure 40. Liquid level in steam generator downcomer for Experiment L6-3.

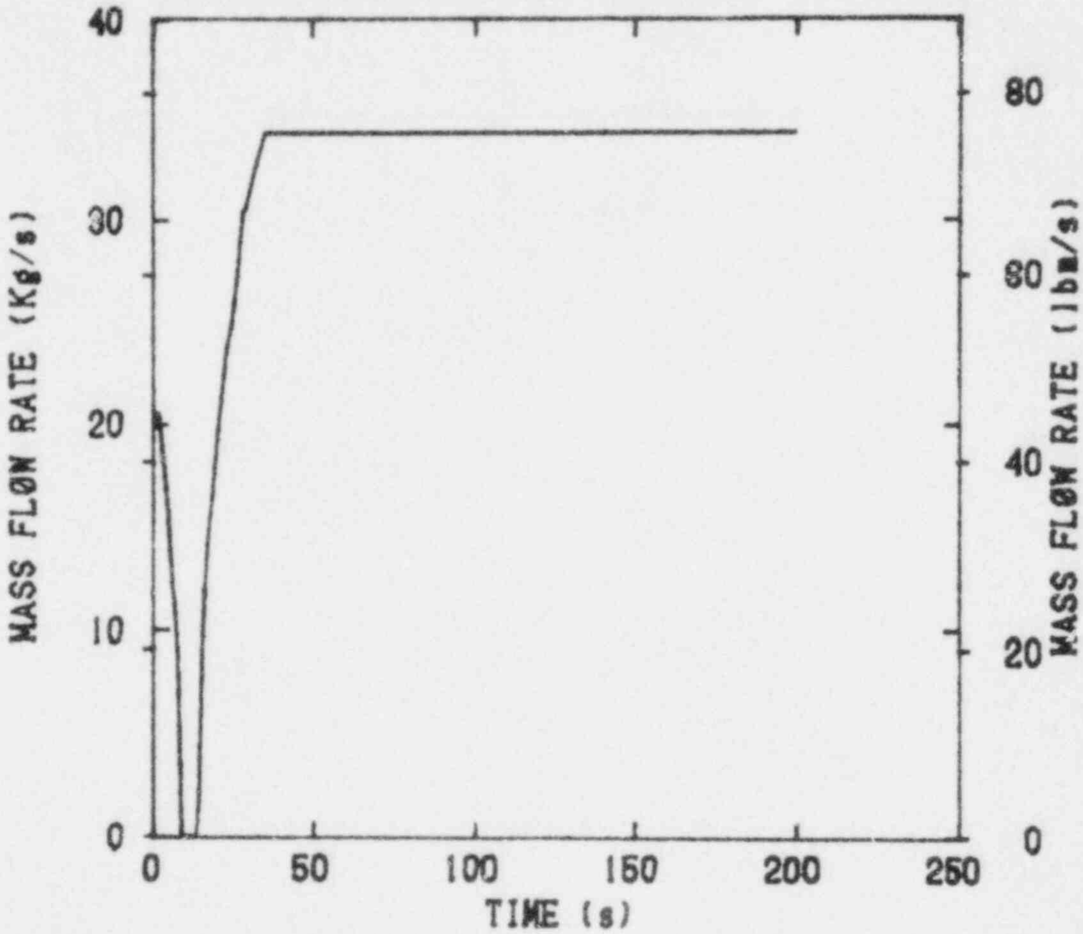


Figure 41. Mass flow rate of steam generator feedwater for Experiment L6-3.

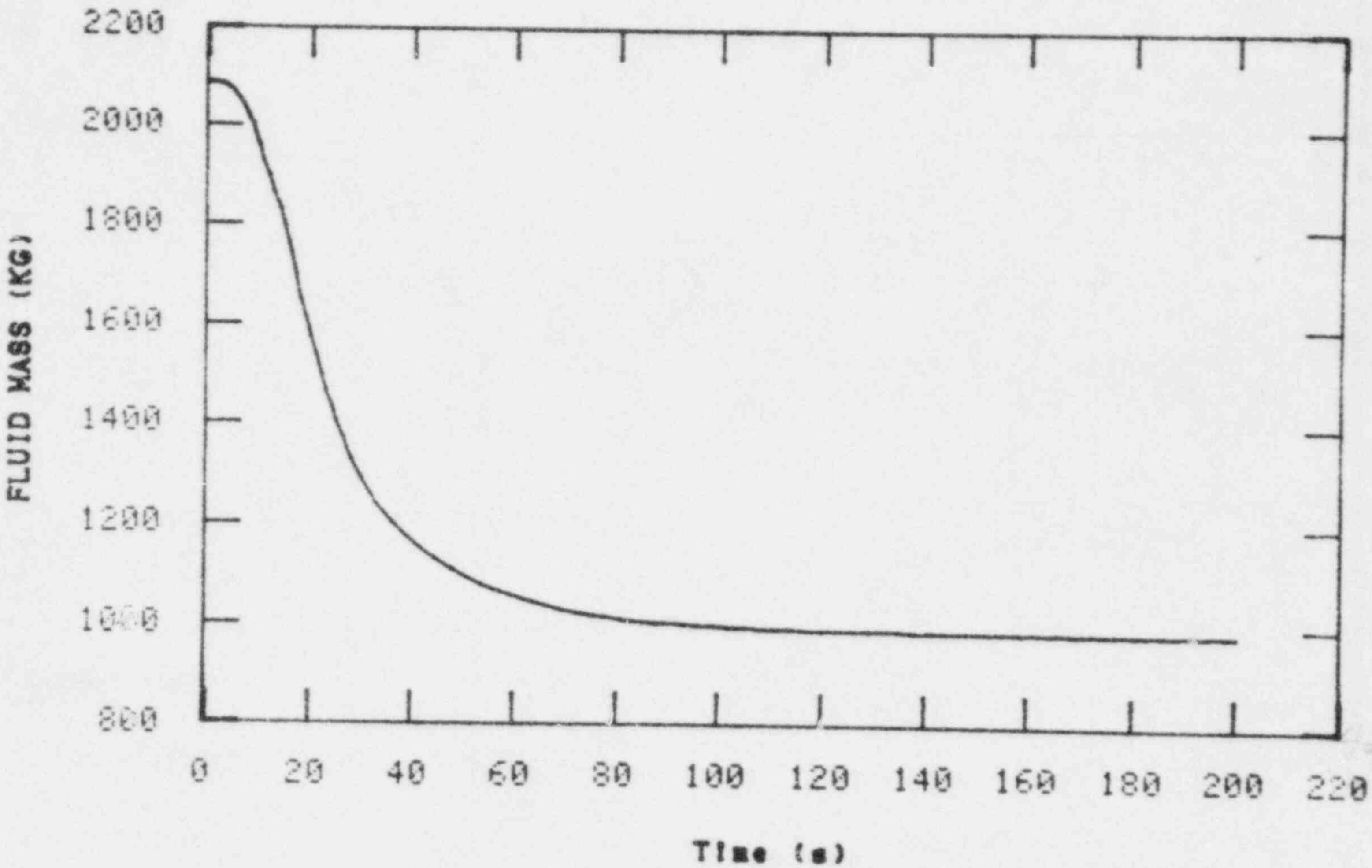


Figure 42. Liquid mass inventory in steam generator secondary side for Experiment L6-3.

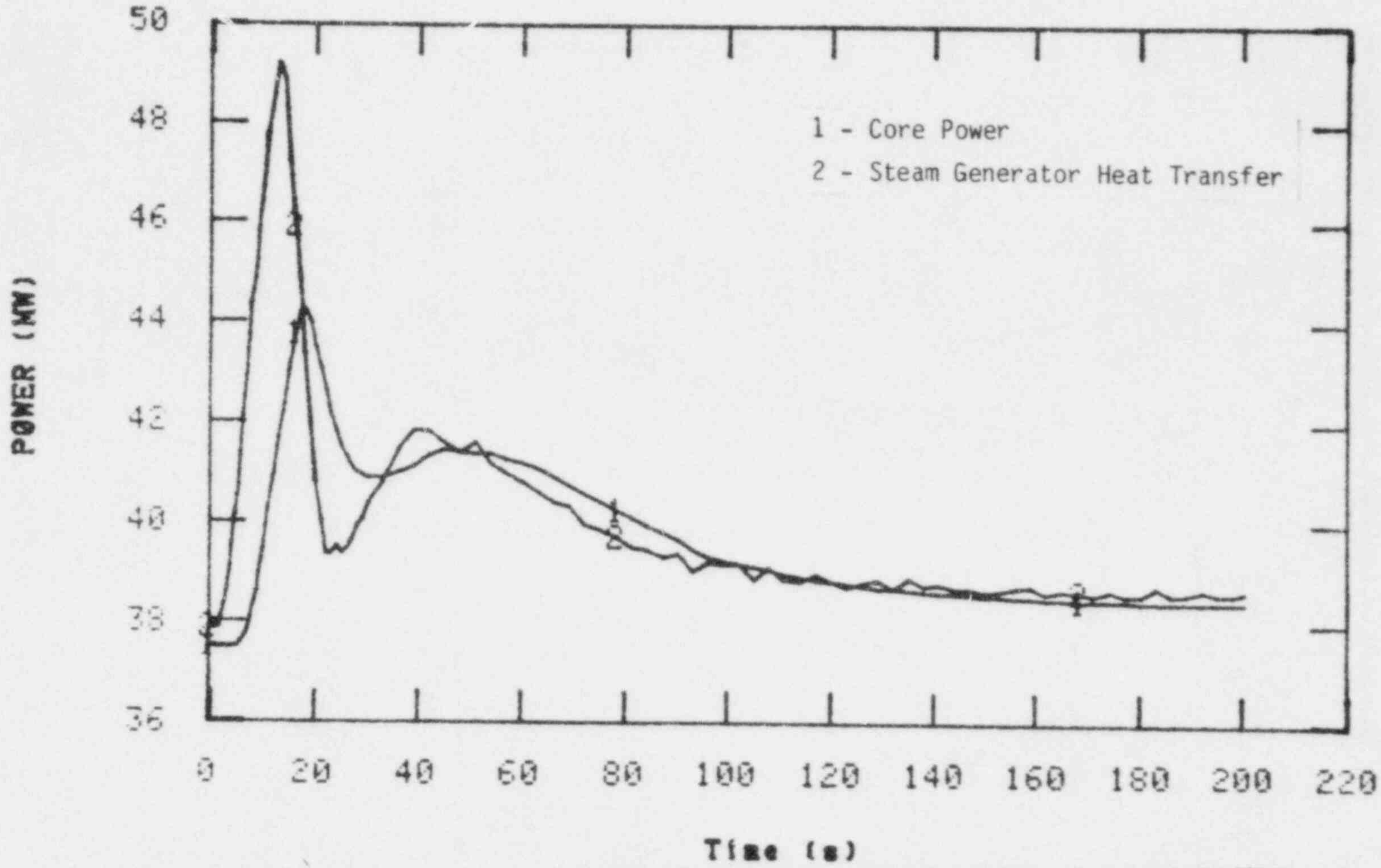


Figure 43. Core power and heat transfer through the steam generator for Experiment L6-3.

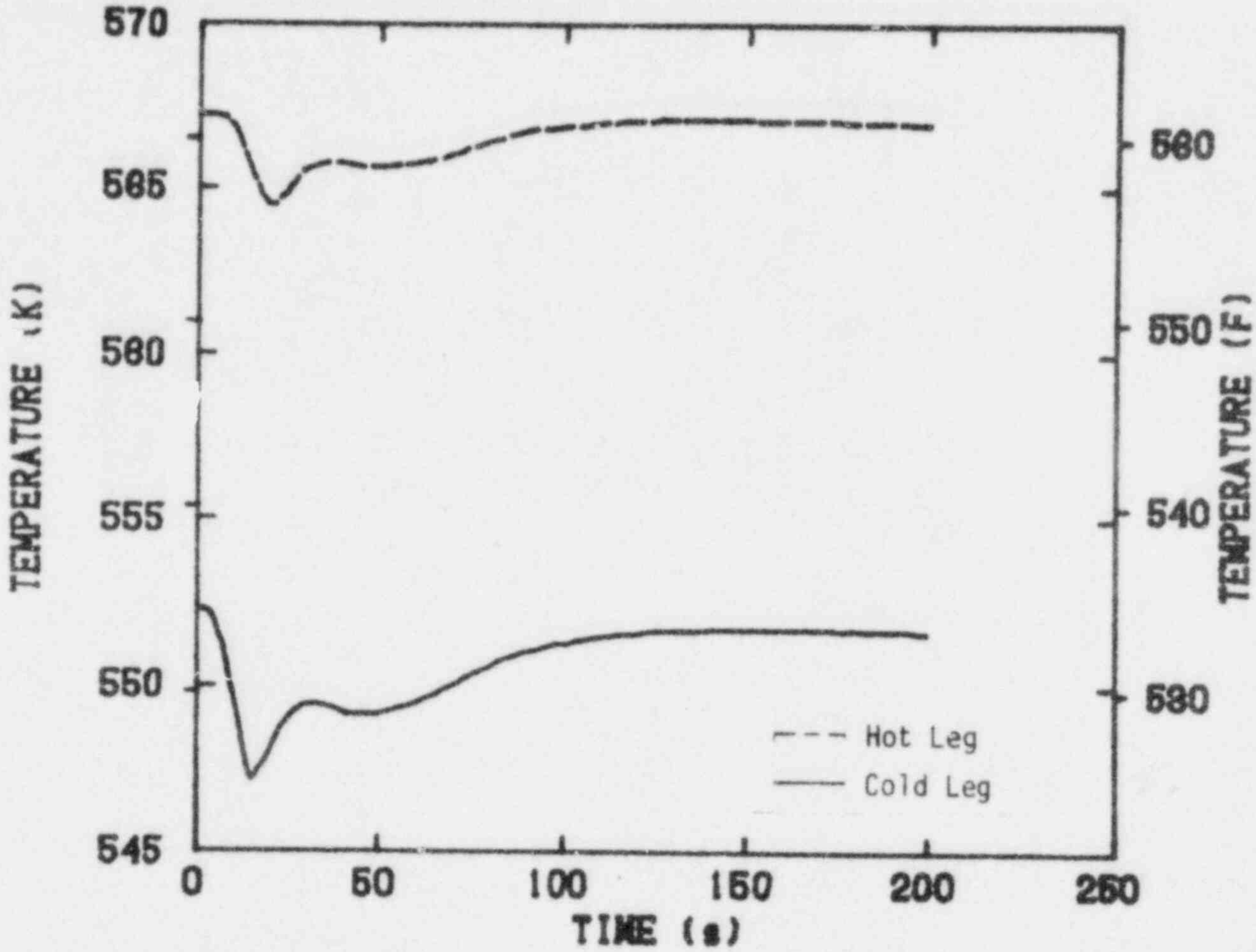


Figure 44. Coolant temperature in intact loop cold and hot legs for Experiment L6-3.

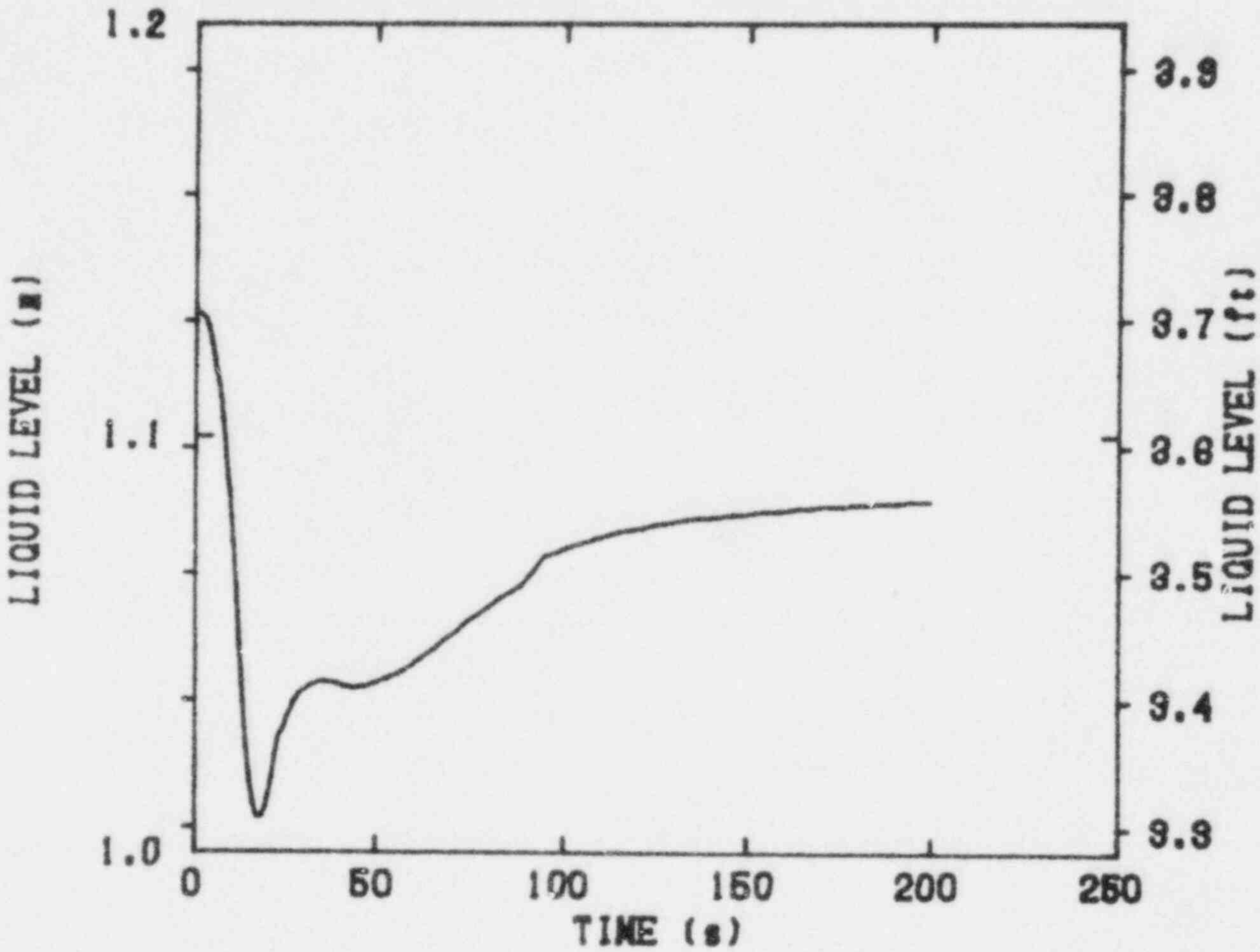


Figure 45. Liquid level in pressurizer for Experiment L6-3.

in Figures 46 and 47. The pressure increases after 20 s are most likely predicted high in this analysis due to the use of the nonequilibrium pressurizer model. This model tends to predict too rapid pressure increases attending slow pressurizer insurges. The pressurizer spray is calculated to come on from 90 to 96 s due to the high primary coolant pressure. Figure 48 shows normalized reactor power.

3.4 Experiment L6-5 Prediction

Experiment L6-5 will simulate a loss-of-feedwater anticipated transient. The experiment will be initiated by tripping the feedwater pump at time zero and closing the feedwater regulating valve. Since LOFT does not have a low steam generator water level trip, the reactor will be manually scrammed when the level drops to 2.82 m (111 in.) above the top of the steam generator tube sheet. The sequence of events which are calculated to occur in this transient is as follows:

Experiment L6-5 Event	Time (s)
Loss of feedwater	0
Reactor scram	23
Steam control valve begins to close	24
Steam control valve closes	39
Steam control valve opens	107
Steam control valve closes	146
Experiment terminated	≅200

The experiment will be terminated by operator intervention when the make-up water is turned on.

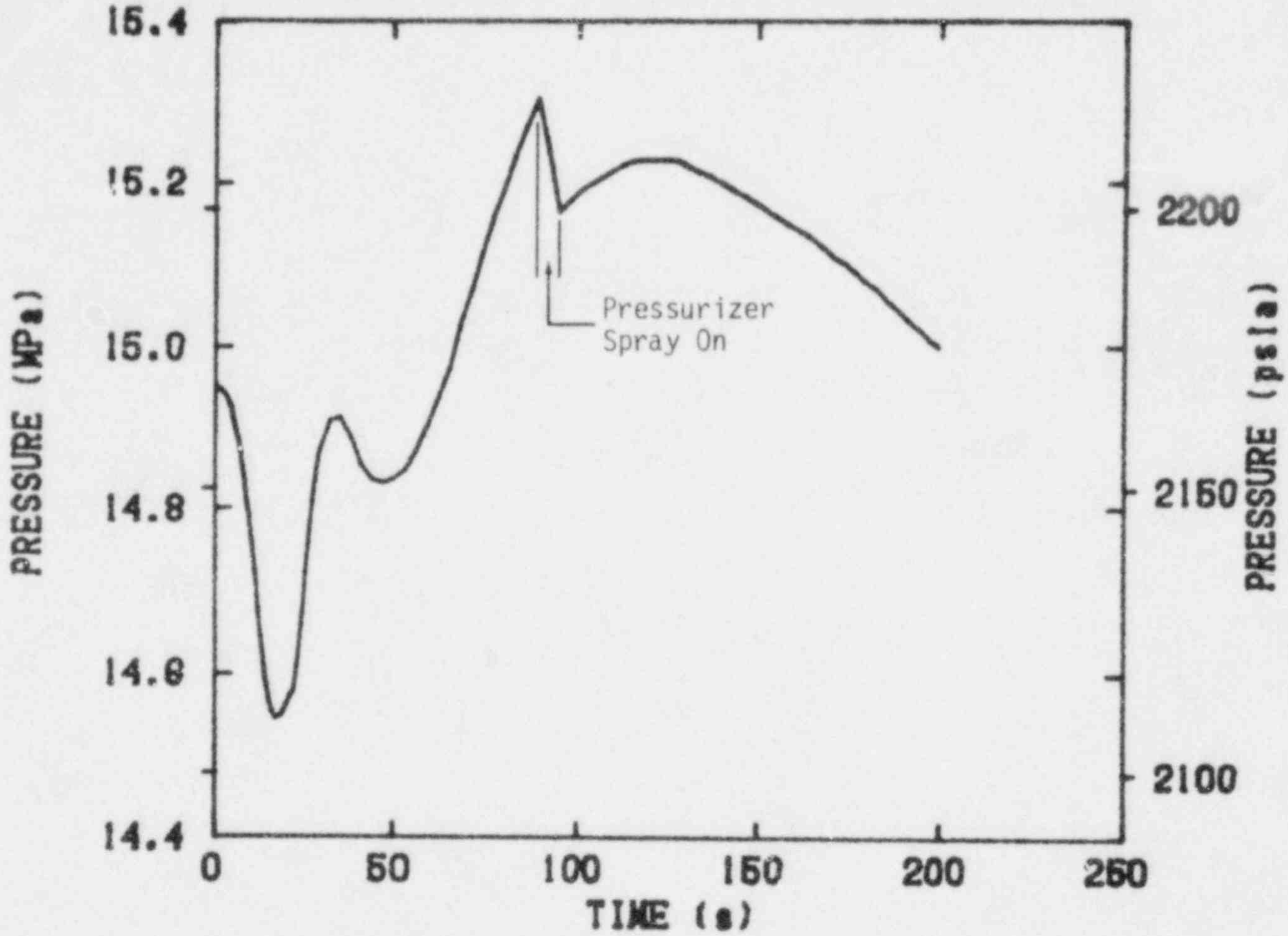


Figure 46. Pressure in pressurizer for Experiment L6-3.

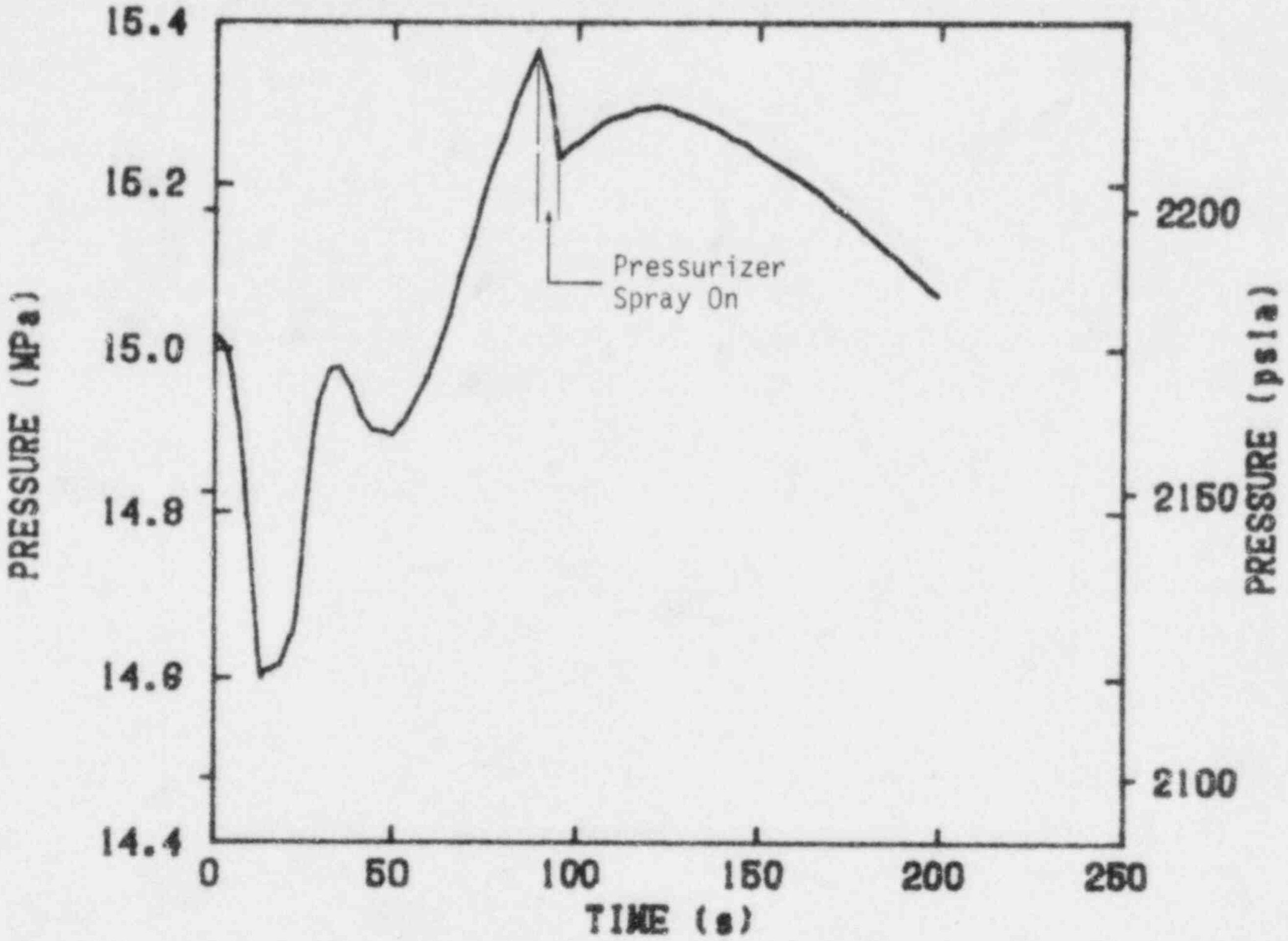


Figure 47. Pressure in reactor vessel upper plenum for Experiment L6-3.

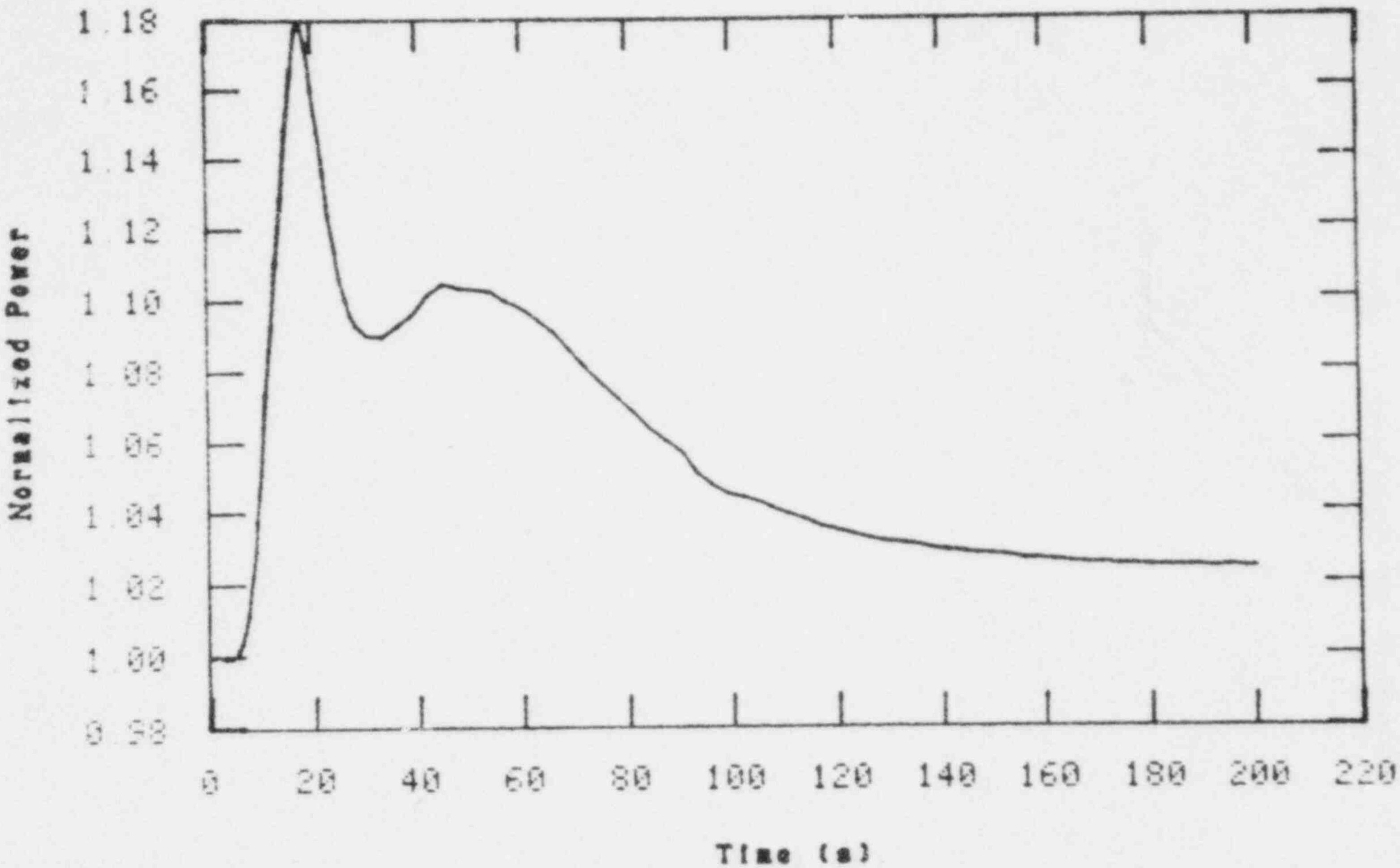


Figure 48. Normalized reactor power for Experiment L6-3.

The major uncertainty in the predictions is in the behavior of the steam control valve. The RETRAN prediction for Experiment L6-5 shows that the steam control valve will open at 107 s. Previous experience in other LOFT experiments has shown that this valve has a tendency to leak when closed. If this leakage could have been quantified and modeled in the RETRAN analysis, the steam control valve may not have opened in the analysis. It is expected that the valve will not open during the experiment. A brief discussion is included to qualitatively describe the plant behavior if significant valve leakage occurs.

The steam control valve on LOFT is analogous to the steam bypass valves in a large PWR. The setpoints and leak characteristics for these valves will vary from plant to plant. Therefore, exact characterization of the behavior of the valve is probably unnecessary for plant safety and typicality considerations. Elimination or quantification of the valve leakage is essential for code assessment purposes.

Figure 49 shows steam generator downcomer liquid level. Zero elevation corresponds to 2.95 m (116 in.) above the top of the tube sheet. Liquid level drops due to loss of feedwater and passes the 2.82-m (111-in.) level at 23 s. The main steam valve starts to close as the reactor scrams and is completely closed at 38 s, at which time the water level stops dropping. The rise in liquid level starting at 107 s is due to the opening of the steam control valve at that time. Significant leakage through the steam valve will result in a lower secondary liquid level after 38 s. If the steam valve does not open, the sudden level increase predicted at 110 s and subsequent decrease will not occur.

Pressure in the steam generator secondary (Figure 50) is closely coupled with flow through the steam control valve (Figure 51). The steam control valve starts closing at 24 s and is completely closed by 38 s. The steam control valve opens again from 107 to 146 s. Inflections in the steam generator secondary pressure correspond to changes in steam control valve position as well as to reactor scram. Figure 52 shows the steam control valve stem position. Figure 53 shows steam generator secondary

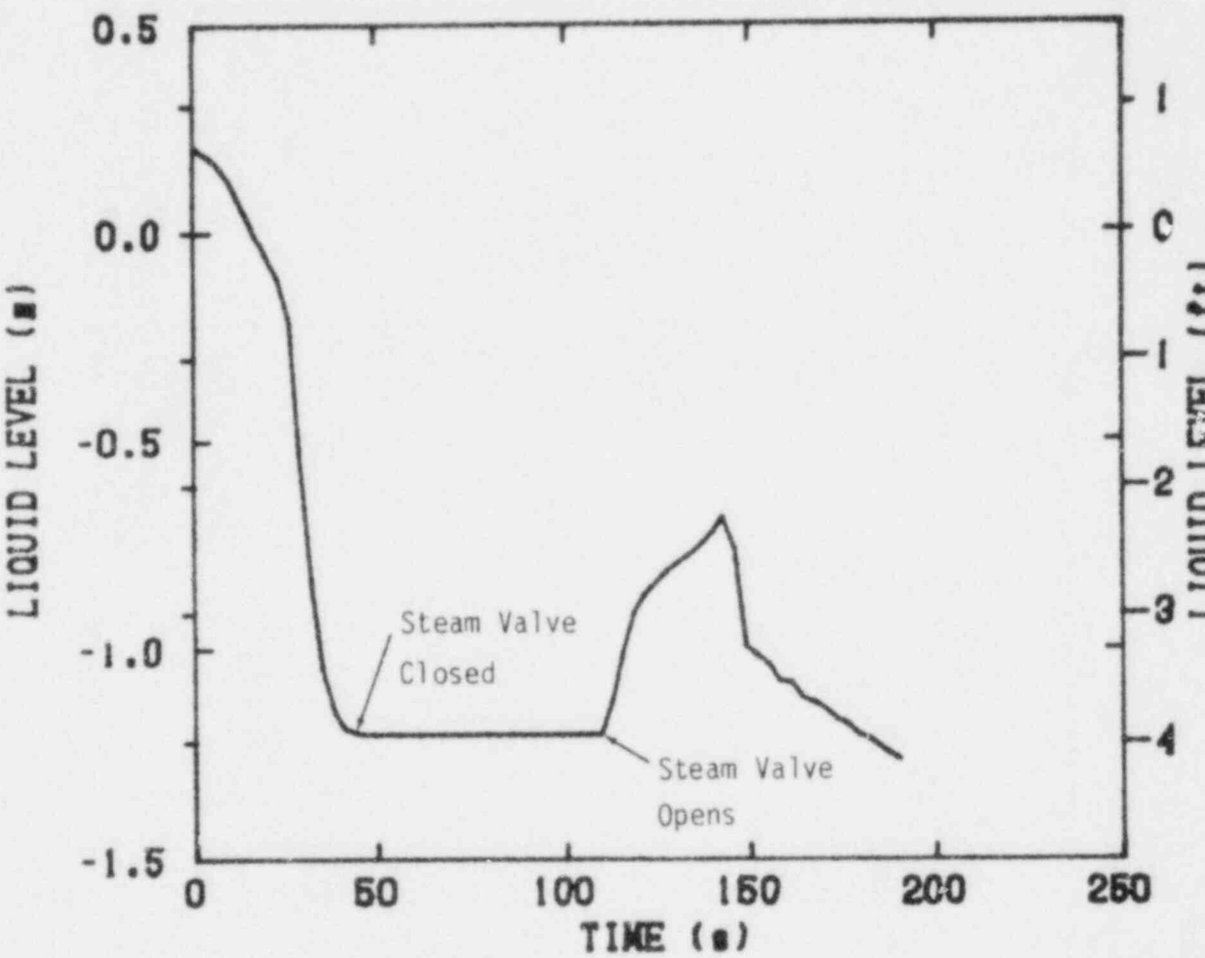


Figure 49. Liquid level in steam generator downcomer for Experiment L6-5.

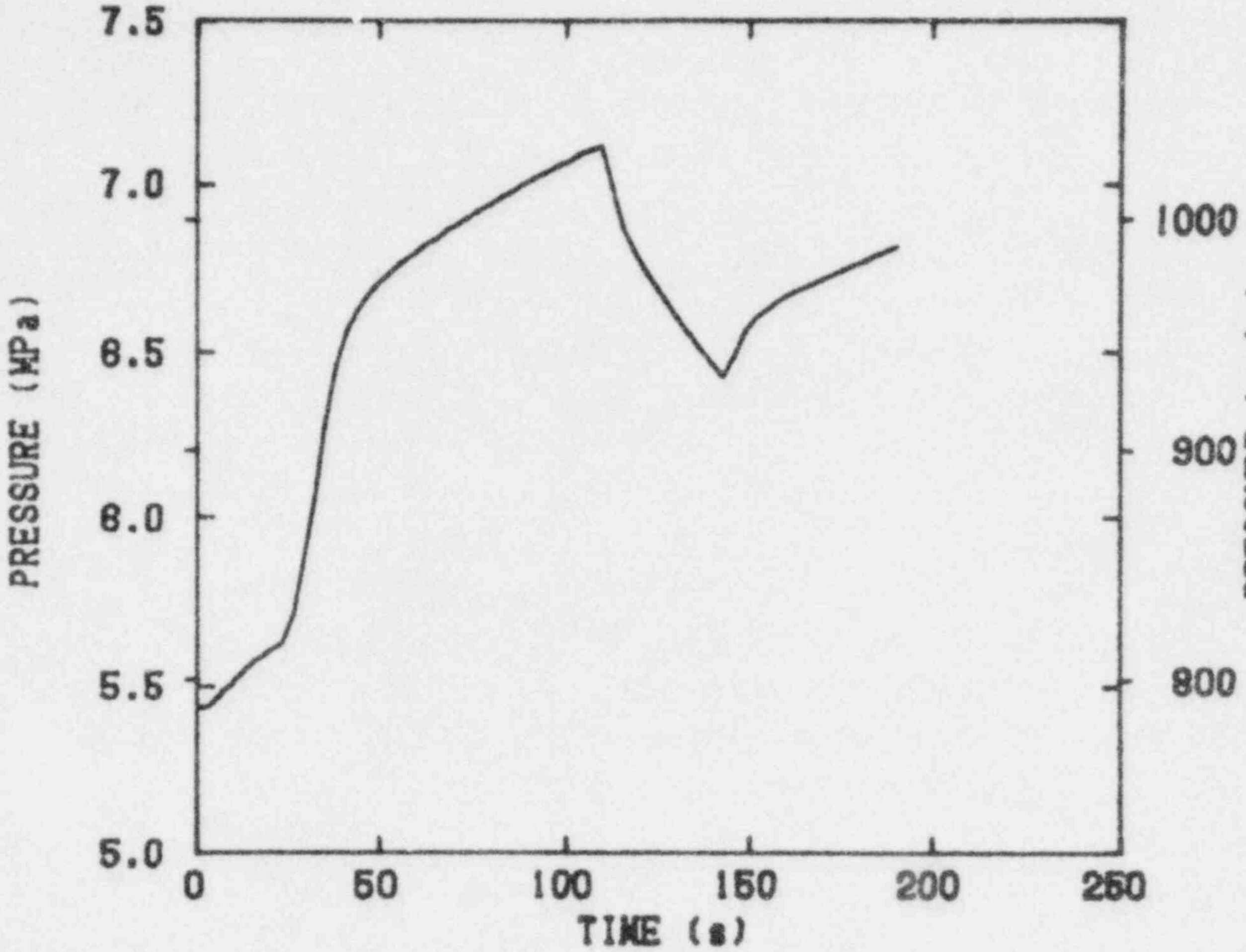


Figure 50. Pressure in steam generator secondary side for Experiment L6-5.

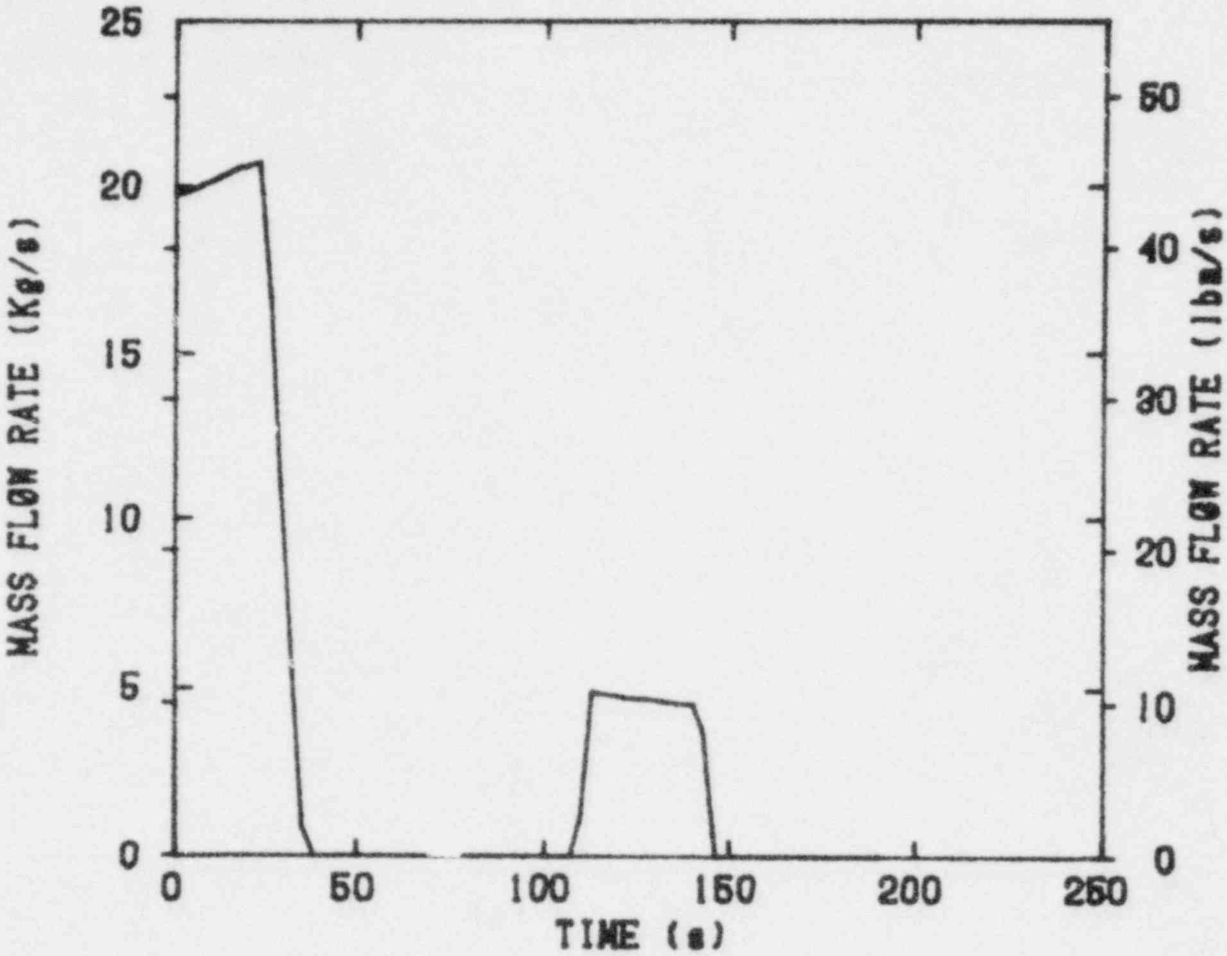


Figure 51. Mass flow rate through steam control valve for Experiment L6-5.

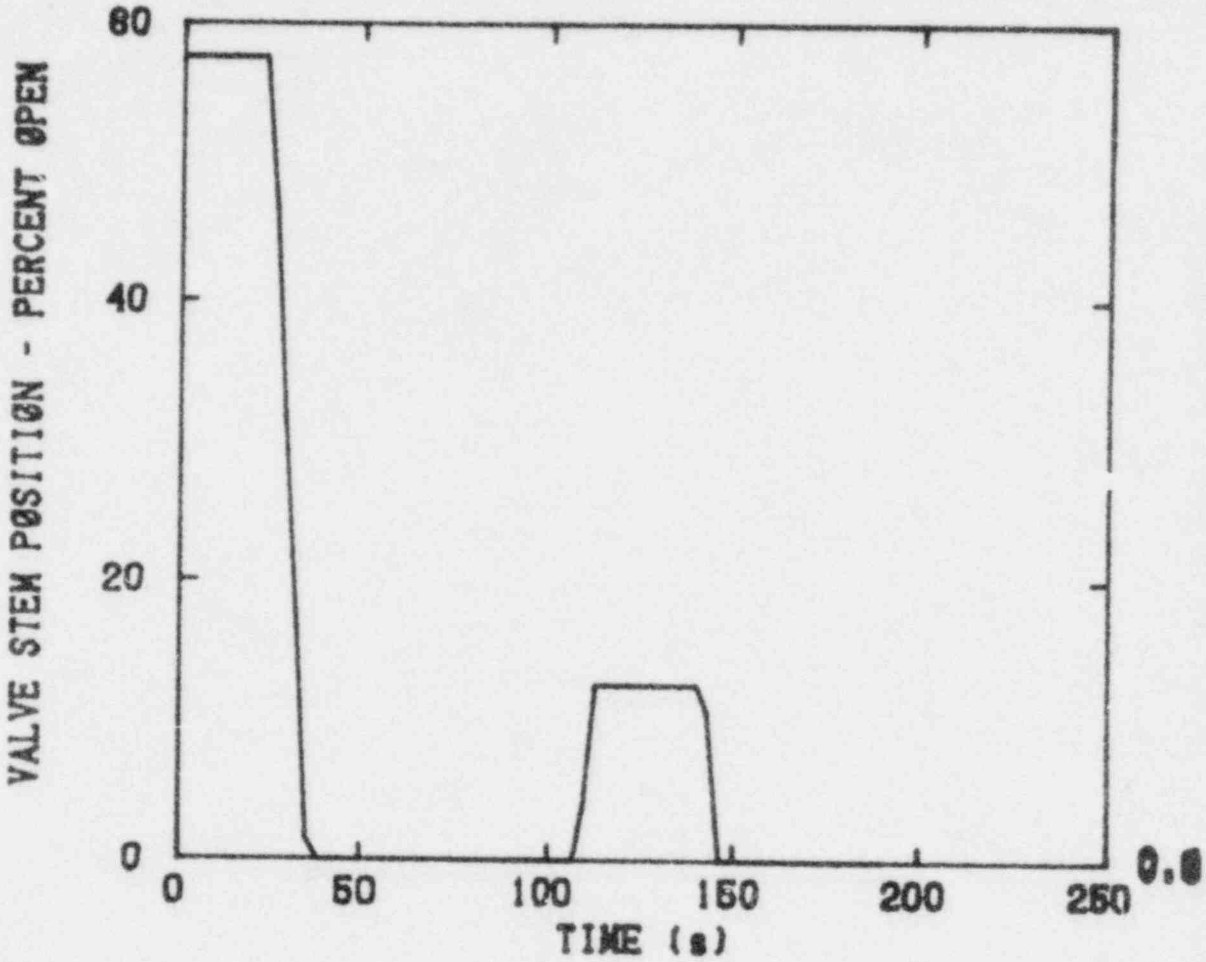


Figure 52. Stem position for steam control valve for Experiment L6-5.

fluid temperature. Steam leakage through the steam control valve after 38 s will cause the secondary pressure and temperature to be lower than shown in Figures 20 and 53, with no sudden drop at 107 s.

Scram occurs at 23 s due to low liquid level in the steam generator secondary. Figure 54 shows an overlay plot of total core heat transfer to fluid with total steam generator heat transfer. Figure 54 shows that steam generator heat transfer starts to drop as soon as feedwater is lost. Reactor power also drops slightly before scram due to reactivity feedback. The rise in steam generator heat transfer at approximately 110 s is due to the opening of the steam control valve. Steam leakage through the steam control valve should have little effect on the steam generator heat transfer. The sudden increase in steam generator heat transfer at 107 s will not occur if the steam control valve does not open.

Pressure in the reactor vessel (Figure 55) and in the pressurizer (Figure 56) increases initially due to the reduction in steam generator heat transfer caused by the loss of feedwater. Once the reactor scrams at 23 s, primary coolant system pressure starts to decrease. Inflections in the pressurizer pressure curve correspond to changes in steam generator and core heat transfer shown in Figure 54. Figure 54 shows that steam generator heat transfer lags behind core heat transfer during the time immediately after scram. During this time period (≅ 23 to 38 s), the steam generator takes out more heat from the primary coolant than the core is adding, and system pressure drops. At approximately 38 s, the two curves in Figure 54, cross and from then until 107 s, core heat transfer is greater than steam generator heat transfer. During this time period, the primary coolant heats up approximately 6 K and there is an insurge into the pressurizer (Figure 57). Primary system pressure drops again at 107 s due to the opening of the steam control valve. Leakage through the steam control valve would tend to lower primary coolant pressure (Figures 55 and 56) somewhat from approximately 38 to 107 s, and the sudden drop in pressure at 107 s will not occur.

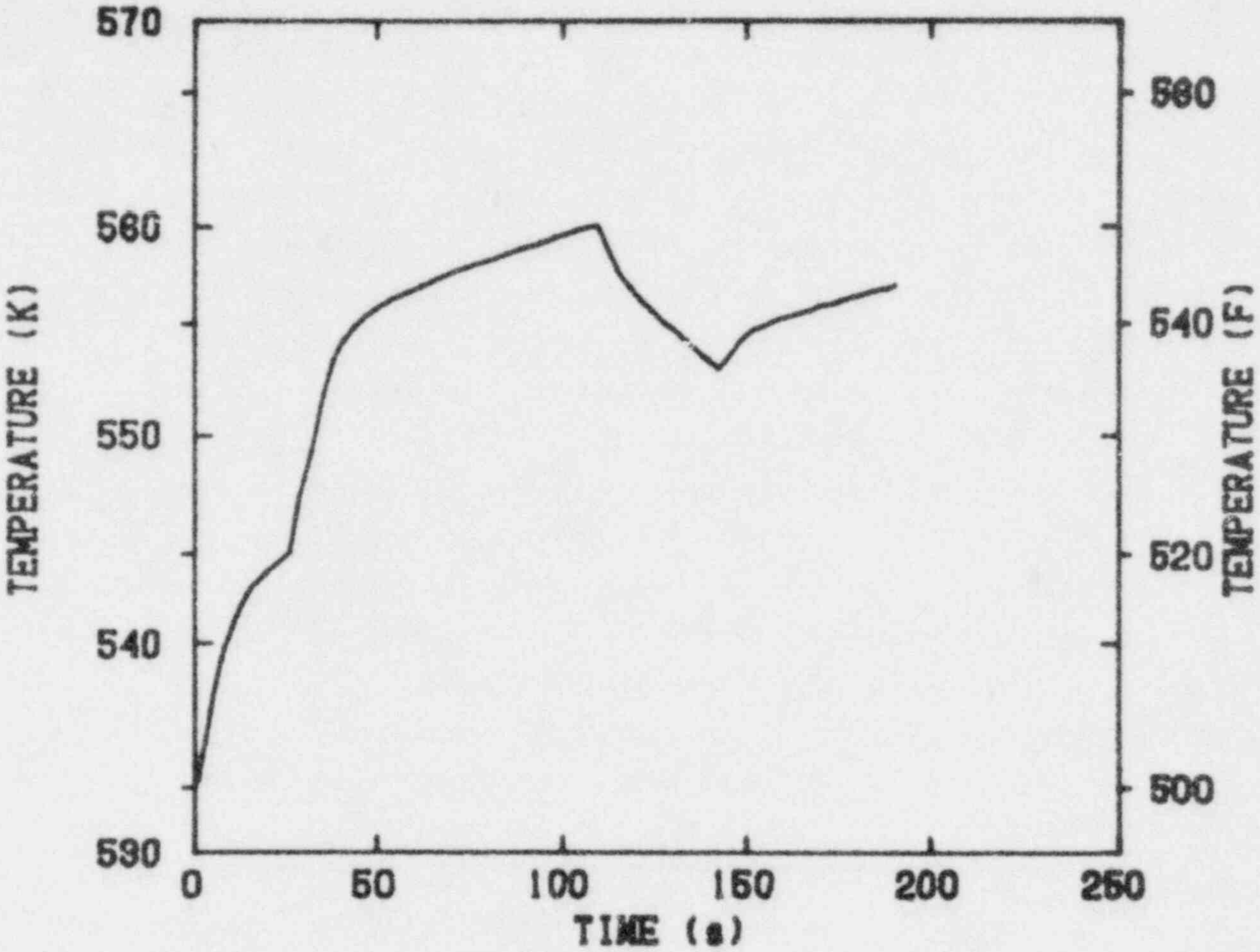


Figure 53. Liquid temperature in steam generator secondary side for Experiment L6-5.

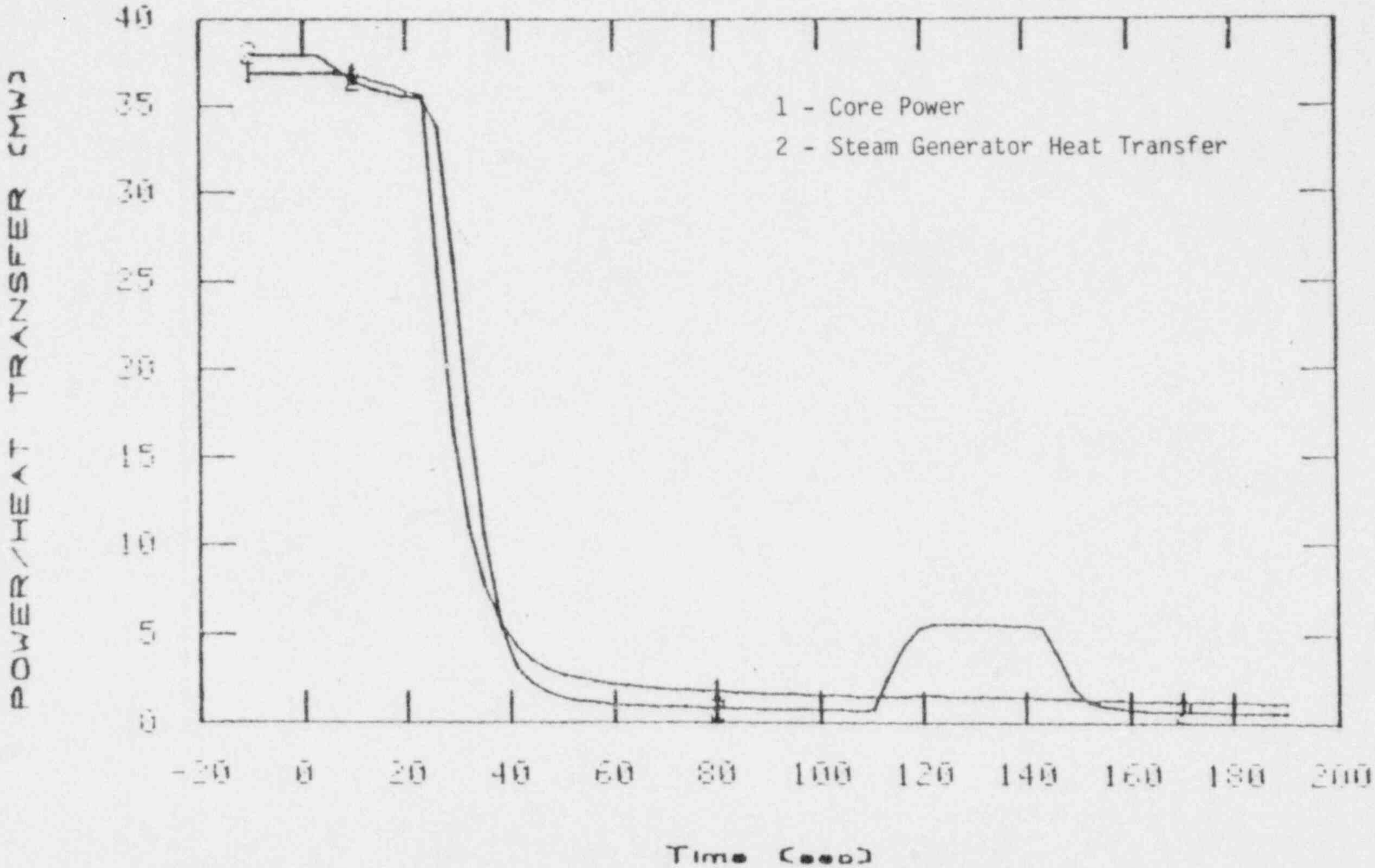


Figure 54. Core power and heat transfer through the steam generator for Experiment L6-5.

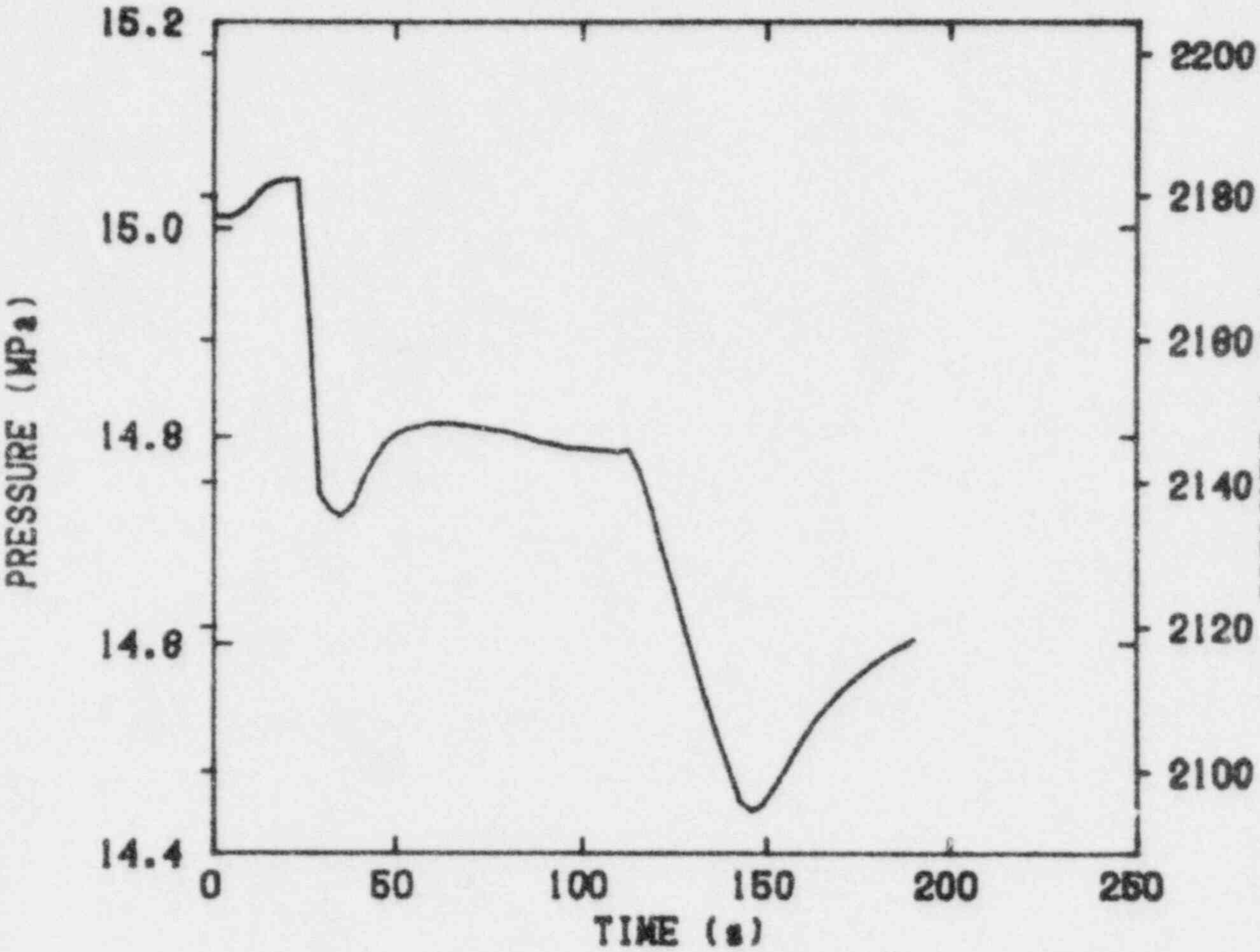


Figure 55. Pressure in reactor vessel upper plenum for Experiment L6-5.

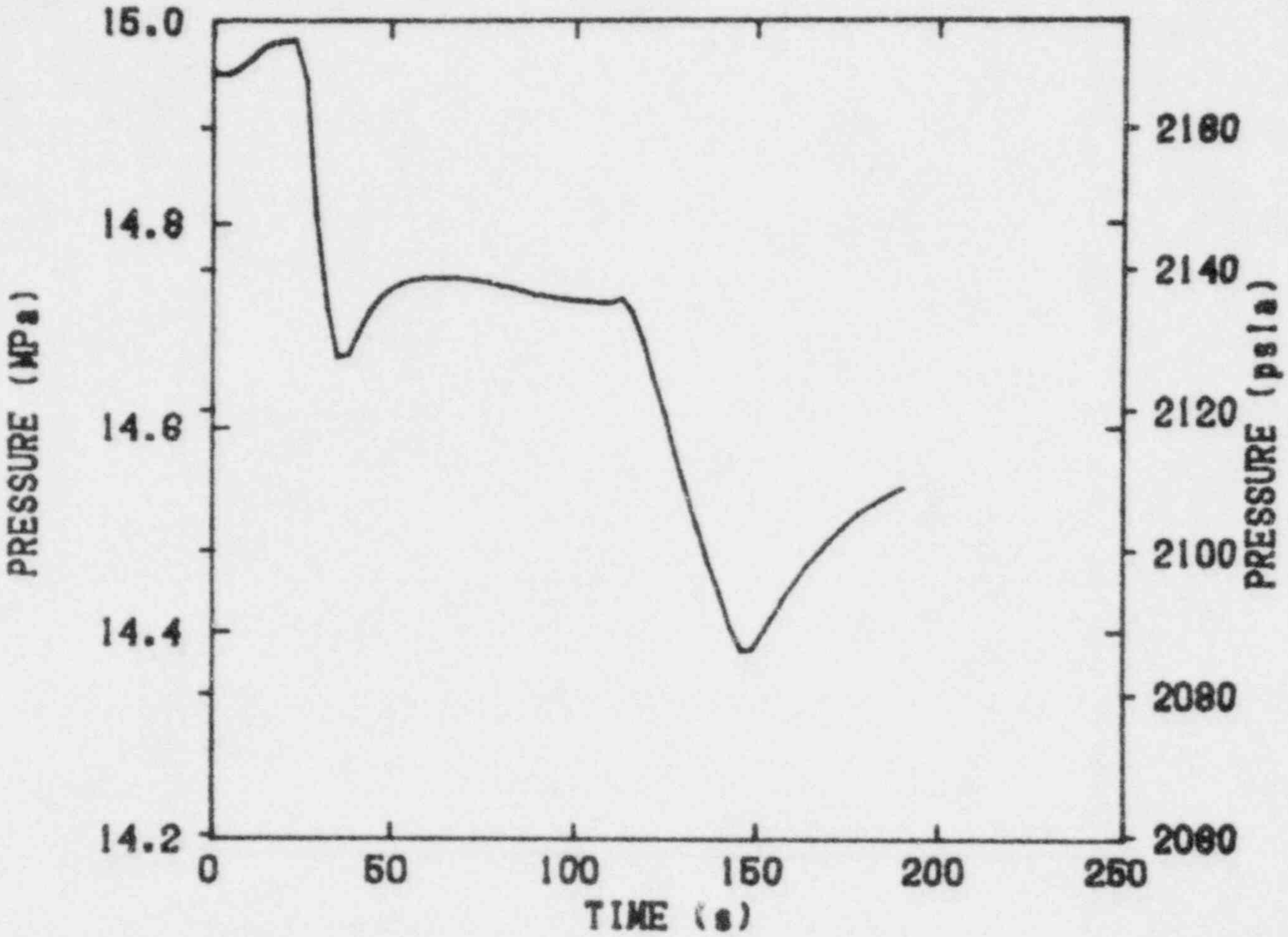


Figure 56. Pressure in pressurizer for Experiment L6-5.

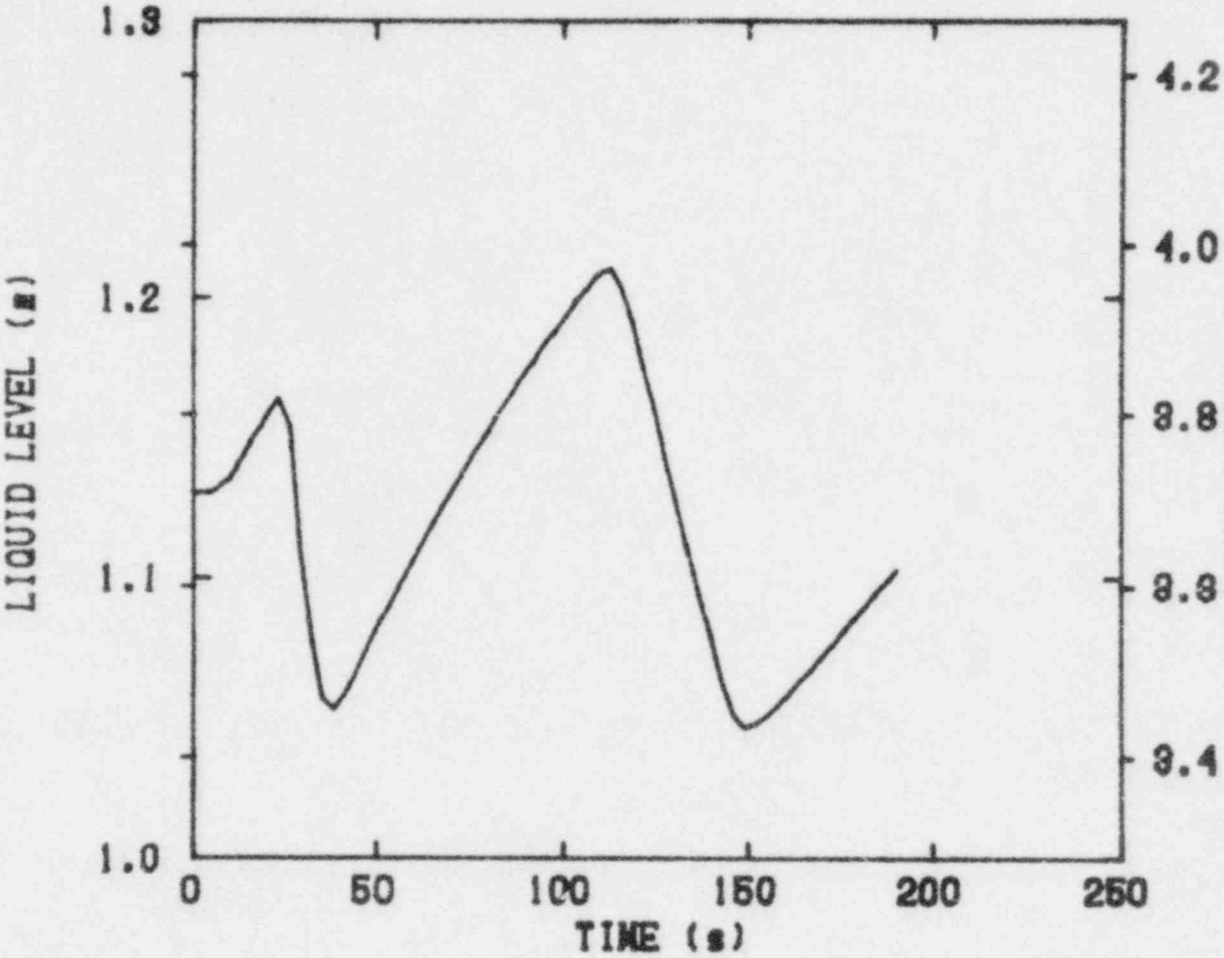


Figure 57. Liquid level in pressurizer for Experiment L6-5.

Figure 58 shows fluid temperatures in the intact loop hot and cold legs. Changes in reactor power and secondary pressure cause inflections in these curves. Figure 59 shows midplane cladding temperature of an average-powered fuel rod. The cladding temperature follows primary system fluid temperature very closely. If the steam control valve does not open as predicted, the inflections in cladding temperature at 110 and 150 s will not occur.

The RETRAN analysis was terminated at approximately 200 s, the time at which the experiment will be terminated.

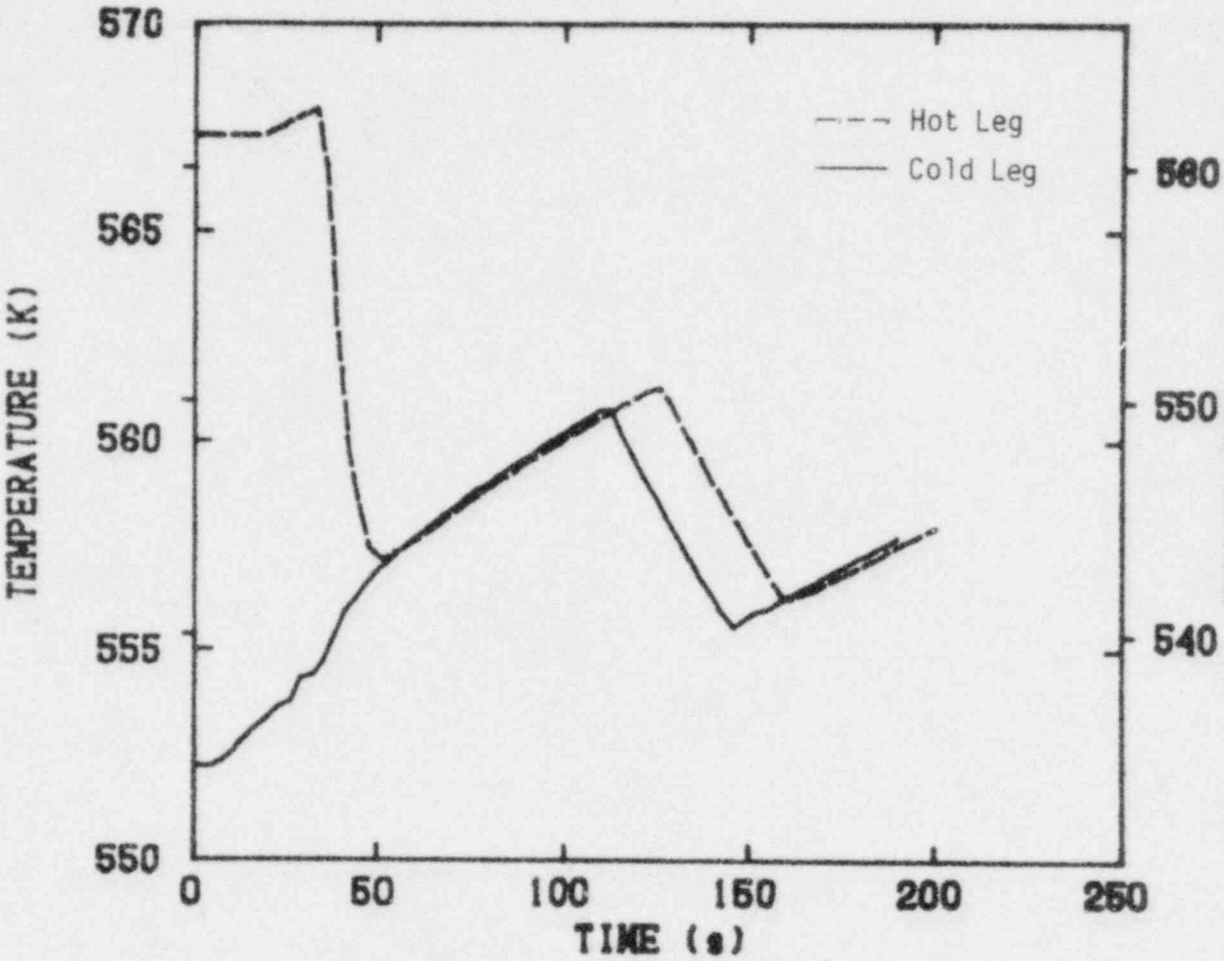


Figure 58. Coolant temperature in intact loop cold and hot legs for Experiment L6-5.

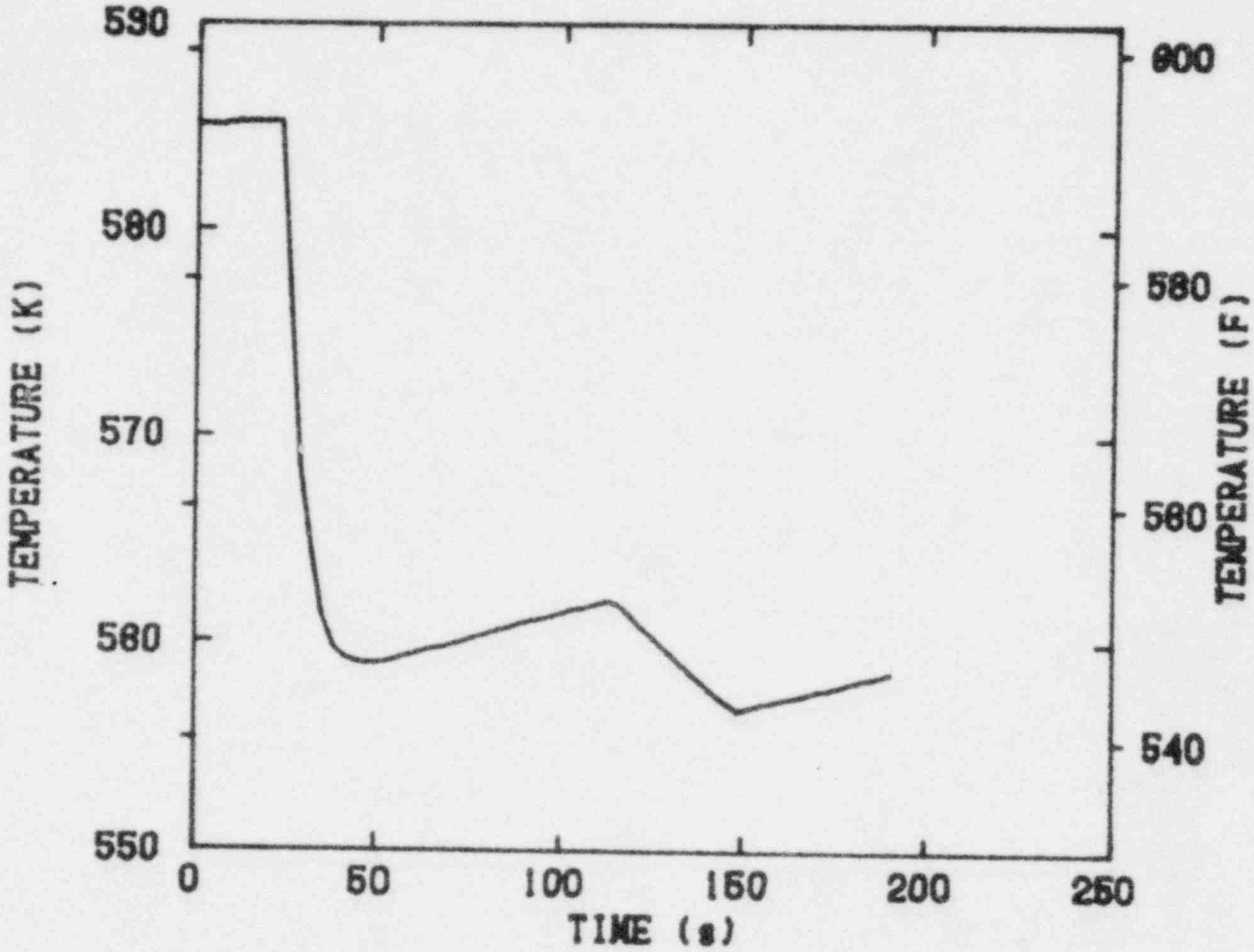


Figure 59. Cladding temperature at midplane of average-powered fuel rod for Experiment L6-5.

4. CONCLUSION

The RETRAN01/MOD2 calculations indicate that each of these anticipated transient experiments will meet their objectives.

5. REFERENCES

1. R. P. Jordan, LOFT Experiment Operating Specification, Non-LOCE Baseline Test Series L6, NE L6 Series, EOS, Rev. 1, September 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. RETRAN - A Program for One-Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI CCM-5, prepared by Energy Incorporated, December 1978.

APPENDIX A

CONFIGURATION CONTROL INFORMATION

APPENDIX A

CONFIGURATION CONTROL INFORMATION

The Idaho National Engineering Laboratory configuration control numbers for the RETRAN01/MOD2 input decks, output tapes, and RETRAN01/MOD2 program are as follows:

1. Experiments L6-1, L6-2, L6-3, and L6-5 input listings - H012385B
2. Experiment L6-1 output tape - H012085B
3. Experiment L6-2 output tape - H012185B
4. Experiment L6-3 output tape - H012285B
5. Experiment L6-5 output tape - H001186B
6. RETRAN01/MOD2 (Absolute Executable) Program - H000986B

APPENDIX B

DETAILED TEST PREDICTION DATA FOR LOFT TEST SERIES L6 EXPERIMENTS

APPENDIX B

DETAILED TEST PREDICTION DATA FOR LOFT TEST SERIES L6 EXPERIMENTS

This appendix provides detailed prediction data for Loss-of-Fluid Test (LOFT) Experiments L6-1, L6-2, L6-3, and L6-5. The data plots showing parameters listed in Table B-1 are presented on microfiche in the pouch attached on the inside of the report back cover. The microfiche are identified as APPENDIX B, and the plots appear in the order they are presented in Table B-1.

TABLE B-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-P4-72A	MASS FLOW--FEEDWATER
Mixture Level	
LT-P4-8B	LIQUID LEVEL--SCS SG SECONDARY
LT-P139-7	LIQUID LEVEL--PRESSURIZER CH B
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

TABLE B-1. (continued)

Parameter	Title
Pressure (continued)	
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
Pump Speed	
RPE-PC-1	PUMP SPEED--PRIMARY COOLANT PUMP 1
Temperature	
TE-BL-1	COOLANT TEMP--BROKEN LOOP CL
TE-BL-2	COOLANT TEMP--BROKEN LOOP HL
TE-P139-29	COOLANT TEMP--INTACT LOOP CL
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-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
Control Valve Stem Position	
CV-P4-8	STEM POSITION--FEEDWATER CONTROL VALVE
DV-P4-10	STEM POSITION--MAIN STEAM CONTROL VALVE

APPENDIX C

UNITS CONVERSION OF RETRAN DATA

APPENDIX C

UNITS CONVERSION OF RETRAN DATA

This appendix describes in detail how the data output from the RETRAN computer code 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 that is utilized in making the prediction.

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

APPENDIX D

RETRAN INPUT LISTINGS

APPENDIX D

RETRAN INPUT LISTINGS

The input listings for the RETRAN models for Loss-of-Fluid Test (LOFT) Experiments L6-1, L6-2, L6-3, and L6-5 are provided on microfiche in the pouch attached on the inside of the report back cover.

DOCUMENT/ PAGE PULLED

ANO. 8011050541

NO. OF PAGES 2 Fiche

REASON

- ☑ PAGE ILLEGIBLE
  - ☐ HARD COPY FILED AT: PDR CF
    OTHER ______
  - ☐ BETTER COPY REQUESTED ON ___/___/___
- ☐ PAGE TOO LARGE TO FILM
  - ☐ HARD COPY FILED AT: PDR CF
    OTHER ______
  - ☐ FILMED ON APERTURE CARD NO ______