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BEST ESTIMATE PREDICTION FOR LOFT NUCLEAR ATWS EXPERIMENT L9-3

W. H. Grush
Y. Koizumi

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INTERIM REPORT

BEST ESTIMATE PREDICTION FOR LOFT
NUCLEAR ATWS EXPERIMENT L9-3

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THE LOFT SUBCOMMITTEE OF THE EG&G PRETEST PREDICTION CONSISTENCY COMMITTEE HAS REVIEWED THE RELAP5 MODEL AND PREDICTED RESULTS FOR LOFT EXPERIMENT L9-3 AND FINDS THEM TO BE CONSISTENT WITH ACCEPTED GUIDELINES.

[illegible]

CODE DEVELOPMENT PROGRAM

A C Peterson for T R Charlton

CODE ASSESSMENT AND
APPLICATIONS PROGRAM

Craig M Kullberg for J. L. Rayman

WATER REACTOR RESEARCH
TEST FACILITY PROGRAM

R. H. Smith

THERMAL FUELS BEHAVIOR PROGRAM

ABSTRACT

The RELAP5 code (a transient, one-dimensional, two-fluid model, reactor loss-of-coolant analysis program) was used to simulate the Loss-of-Fluid Test (LOFT) facility response during nuclear Experiment L9-3 which will be an anticipated transient without scram (ATWS) experiment. Experiment L9-3 will simulate the loss of normal feedwater with subsequent failure to scram in a commercial pressurized water reactor (PWR). The report includes the results of a nominal best estimate calculation and sensitivity calculations which investigate the sensitivity of Experiment L9-3 to various parameters. The results indicate that, if conducted as planned, Experiment L9-3 will meet its objectives. During the L9-3 experiment, the reactor pressure is expected to reach a maximum of 18.6 MPa (2700 psia) before decreasing to below ECC injection during the recovery phase. The peak pressure for the nominal experiment prediction calculation was 17.9 MPa (2600 psia) due to an anticipated under prediction of the transient reactor power. The reactor control rods will be inhibited from inserting throughout the experiment, however, no fuel damage is expected.

NRC FIN No. A6048 - LOFT Experimental Program

SUMMARY

This document reports the Loss-of-Fluid Test (LOFT) Experiment L9-3 pretest calculation of the system thermal-hydraulic response using the RELAP5/MOD1 computer code. Experiment L9-3 will be the first anticipated transient without scram (ATWS) simulation to be performed in an actual nuclear reactor. The experiment will be initiated by a loss of normal feedwater to the steam generator. The reactor scram which noramlly would occur will be inhibited. Results of the nominal experiment prediction calculation indicate that the test will meet its stated objectives. Sensitivity calculations were performed which indicate that the peak primary system pressure is very sensitive to allowable deviations from the specified initial conditions and to uncertainties in other input parameters. These sensitivity studies indicate that one or more of these variables or uncertainties could cause the pressure to exceed the setpoint of the plant safety valves, thereby aborting the experiment.

ACKNOWLEDGMENTS

The authors gratefully acknowledge the technical advise and assistance of R. J. Beelman and S. R. Behling.

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BEST ESTIMATE PREDICTION FOR LOFT
NUCLEAR ATWS EXPERIMENT L9-3

1. INTRODUCTION

This report contains results from the experiment prediction (EP) analysis performed using the RELAP5/MOD1 computer code to simulate the coupled system thermal-hydraulic responses of Loss-of-Fluid Test (LOFT) Experiment L9-3. This experiment will simulate a loss of feedwater anticipated transient without scram (ATWS) in a commercial pressurized water reactor (PWR). It will be the first ATWS experiment ever performed using a nuclear reactor. The purpose of this report is to document an analysis that provides a basis for evaluating the best known modeling techniques by supplying a prediction of the experiment. In addition, the EP may be used as a basis to judge whether the experiment will meet its stated objectives (Appendix A).

The LOFT facility (Appendix A) is a 50-MW(t) pressurized water (nuclear) reactor (PWR) with instrumentation to measure and provide data on the thermal-hydraulic conditions throughout the system. The steady-state operation of the LOFT system is typical of a large commercial PWR.

The specified nominal initial conditions are:[1] A reactor power of 49.5 MW, an average primary temperature of 569.3 K (565°F), a core delta-T ($T_{hot} - T_{cold}$) of 21.1 K (38 F), a pressurizer pressure of 14.95 MPa (2170 psia), a pressurizer level of 1.168 m (46 inches), the control rods withdrawn 1.37 m (54 inches), and a steam generator level of 3.2 m (126 inches) above the top of the tube sheet.

This report describes how the RELAP5 computer code was used to simulate and predict the LOFT system thermal-hydraulic responses and presents predicted results for Experiment L9-3. Section 2 contains a description of the modeling techniques employed in the EP analyses. Section 3 contains discussions of the calculated results for a nominal best estimate calculation. Section 4 presents results of pertinent sensitivity calculations. Conclusions are presented in Section 5. Appendix A provides brief descriptions of Experiment L9-3 and of the LOFT facility. A listing of the code input data and updates is provided in Appendix B and plots showing the detailed results of the EP are included in Appendix C. Appendices B and C are found on microfiche on the report back cover.

2. COMPUTER SIMULATION

The RELAP5/MOD1 computer code[a] was used to simulate the transient thermal-hydraulic responses for the LOFT system during Experiment L9-3. The RELAP5 code is a one-dimensional, two-fluid, thermal nonequilibrium reactor transient analysis program. The specific application of the code to the Experiment L9-3 simulation is discussed in this section.

The nodalization used in RELAP5/MOD1 for this EP calculation is based on a standard LOFT nodalization[b] with changes where necessary to represent the particular system configuration for Experiment L9-3. The nodalization is shown in Figure 1. A complete input data listing is supplied in Appendix B.

Since a previous LOFT Experiment, L9-1[2,3] was initiated by a loss of feedwater to the steam generator, the L9-3 transient is expected to be nearly identical to L9-1 until the time of reactor scram. For that reason improvements both in the modeling techniques and in the RELAP5 code which were precipitated by the L9-1 experiment and posttest analysis have been incorporated into this analysis.

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a. This analysis was performed using RELAP5/MOD1 Cycle 12, a production version of the RELAP5/MOD1 code which is filed under Idaho National Engineering Laoratory Computer Code Configuration Management (CCCM) Archival Number F00341.

b. The standard LOFT input model version 117 was used as the basis for the L9-3 input deck. The model is continually being updated and improved. However, complete tracibility of each version is maintained in the model and by the LOFT program division.

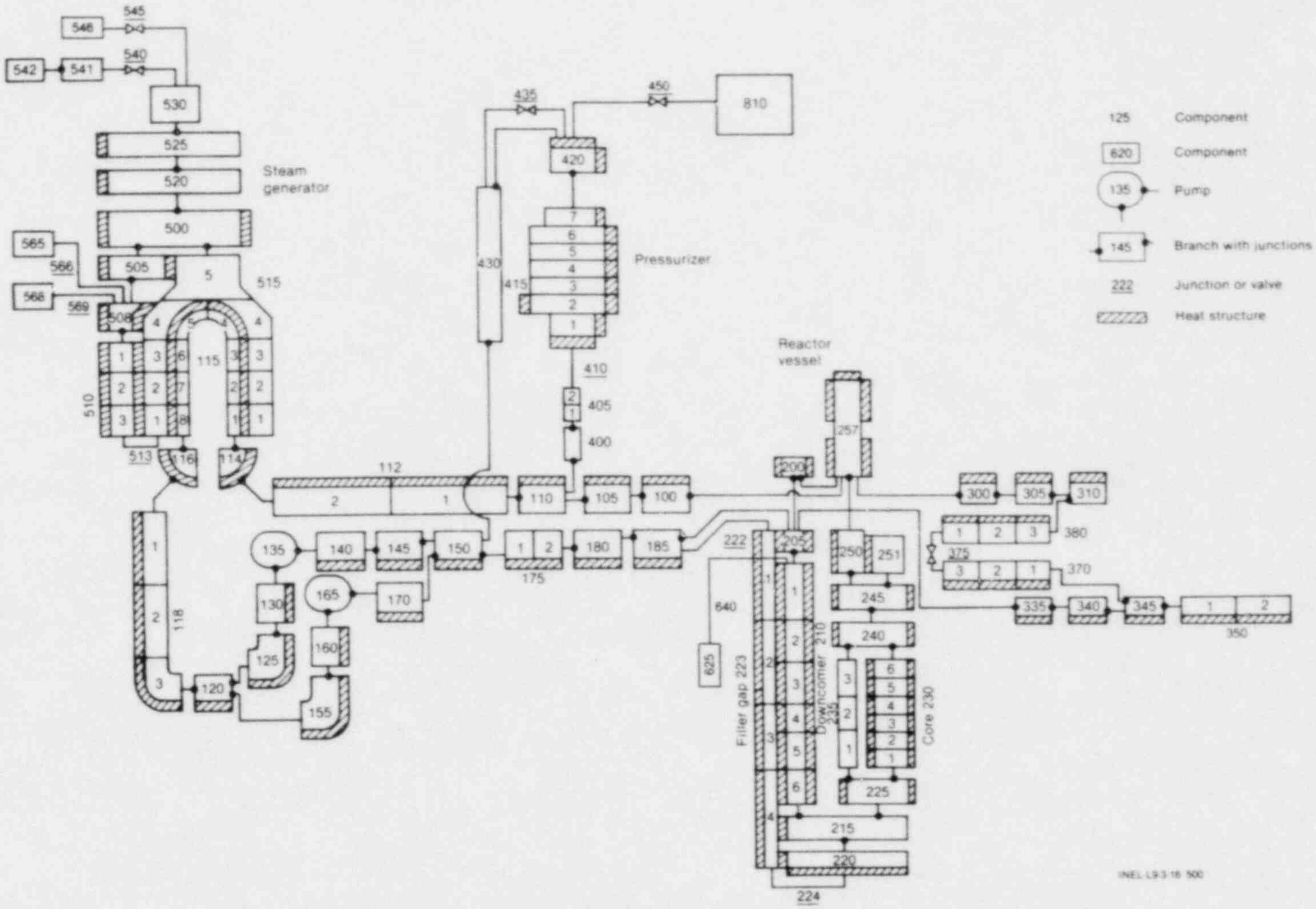

Figure 1. LOFT RELAP5 model nodalization diagram for L9-3

The most significant change in the modeling technique for the L9-3 prediction concerns the modeling of heat structures, particularly the filler blocks within the reactor vessel. The hydraulic passage between the filler blocks and the reactor vessel, known as the filler gap, is now explicitly modeled so that both surfaces of these massive structures are available for heat transfer. During the L9-1 experiment the filler blocks acted as a significant heat sink. Particular attention was also given to accurately modeling the heat losses from the LOFT system to the environment since this is expected to affect the later portions of the L9-3 transient.

The major RELAP5 code improvement resulting from the L9-1 experiment is a linking of the heat transfer calculation to the interfacial mass transfer model. The improvement, which is currently being incorporated as an update to the code, forces the mass transfer model to produce liquid in a superheated volume when the heat transfer routines are calculating the condensation of vapor at a subcooled wall. This phenomena was observed in Experiment L9-1 to be a dominant mechanism for removing heat from the steam generator secondary after dryout had occurred. The update also forces the mass transfer model to produce vapor in a subcooled volume when the heat transfer routines are calculating boiling heat transfer, however, this effect is expected to be of lesser importance in the L9-3 transient. A complete listing of this and other code updates used for this analysis can be found in Appendix B.

The following changes to the standard nodalization were made for this analysis:

1. The test PORV, safety relief valve is modeled as a single valve component which can move at a constant rate between the shut position, a PORV position, and a combination PORV plus safety relief valve position.

2. The broken loop hot leg components downstream of the contraction, mechanical joint A (Ref. 4), are removed because that piping will be flanged off for this test.

3. The broken loop cold leg components downstream of the isolation valve are removed because the valve will remain shut throughout the test.

4. The high pressure injection system is routed into the downcomer as it will be for the test. Low pressure injection and the accumulator are removed because they will not be used.

5. A nominal fluid leakage from the primary coolant system is included in the model. The leakage rate of 1.577×10^{-5} m^3/s (0.25 gpm) at 15 MPa (2177 psia) increases with pressure above a threshold value of 12 MPa (1742 psia).

6. The reactor vessel upper plenum region above the nozzles is combined into a single control volume and the junction connections are changed to route the primary coolant loop flow through this volume. This change is necessary to provide the mixng in the upper plenum which is known to occur while the primary coolant pumps are running in LOFT.

7. The main and auxiliary feedwater systems are modeled using time dependent volumes and junctions.

8. The main steam bypass valve is explicitly modeled.

9. A nominal leakage through the steam flow control valve is accomplished by establishing a minimum area of 4.419×10^{-6} m^2 (6.85×10^{-3} $inches^2$). This value will approximate the leakage observed during experiment L3-7.

10. The moderator density reactivity feedback table was taken from the Reference 5 table for 500 ppm boron and multiplied by 1.0375 to compensate for an anticipated initial boron concentration of 650 ppm.

11. The expected initial boron concentration in the primary coolant system is included in the model as is the specified time dependent boron concentration in the high pressure injection flow. However, reactivity feedback effects due to changes in boron concentration are not explicitly modeled. A dummy reactivity insertion table is invoked during the recovery phase to partially compensate for the reactivity insertion caused by the cooling of the primary coolant system.

An initialization run was performed in order to obtain initial conditions specified in the Experiment Definition Document (Reference 1). Time, variable and logic trips were modified for proper simulation of the experiment scenario. Main events and trips are listed in Table 1.

3. CALCULATIONAL RESULTS

This section contains a general overview of the results of the Experiment L9-3 simulation. For convenience in this discussion, the transient is divided into three phases: the initiation phase during which the primary pressure challenges the test safety valve and the peak pressure is attained; the stablization phase during which the test PORV cycles to maintain the plant pressure; and finally the recovery phase during which the operators take action to begin recovery of the plant.

The initiating event for Experiment L9-3 is a termination of feedwater flow to the LOFT steam generator. Previous LOFT Experiment L9-1[2,3] used the same initiating event. The initial conditions for the two tests are nearly identical with the exception of (a) a slightly higher initial boron concentration for L9-3 (about 100 ppm higher than for L9-1), and (b) a higher pressurizer level (about 0.5 m higher than for L9-1) due to scaling differences. The fundamental difference between the tests is that the reactor scram, which occurred at 65.4 s in L9-1, will be inhibited for

TABLE 1. L9-3 EXPERIMENT TRIP SETPOINTS

Instrument Location	Setpoint	Action
Pressurizer Pressure (before 600 s)	>16.20 MPa (2350 psia)	Test PORV opens
Pressurizer Pressure (before 600 s)	<16.00 MPa (2320 psia)	Test PORV closes
Pressurizer Pressure (before 600 s)	>16.24 MPa (2500 psia)	Test safety valve opens
Pressurizer Pressure (before 600 s)	<16.46 MPa (2388 psia)	Test safety valve closes
Pressurizer Pressure (after 600 s)	>15.5 MPa (2250 psia)	Test PORV opens
Pressurizer Pressure (after 600 s)	<15.0 MPa (2177 psia)	Test PORV closes
Steam Generator Secondary	<4.13 MPa (600 psia)	Main steam valve closes
Steam Generator Secondary (before 600 s)	>6.63 MPa (963 psia)	Main steam bypass valve opens
Steam Generator Secondary (before 600 s)	<6.63 MPa (963 psia)	Main steam bypass valve closes
Steam Generator Secondary (after 600 s)	>4.48 MPa (650 psia)	Main steam bypass valve opens
Steam Generator Secondary (before 600 s)	<4.13 MPa (600 psia)	Main steam bypass valve closes
Primary Coolant Temperature (after 600 s)	>588 K (599°F)	Auxiliary feed on at 16 gpm
Primary Coolant Temperature (after 600 s)	<583 K (590°F)	Auxiliary feed off

Experiment L9-3. Therefore, the first 65 s of the two transients are expected to be nearly identical. In many of the figures which follow the first 65 s of L9-1 test data is shown with the L9-3 experiment prediction calculation.

The nominal experiment prediction calculation which is discussed in the remainder of this section reaches a peak pressure of 17.9 MPa (2600 psia) which is 0.7 MPa (100 psia) below the target pressure for the test. The reason is that a different calculation (reported in detail in Section 4.1) was used to size the test valve. That calculation used the measured reactor power from the first 65 seconds of Experiment L9-1 as a boundary condition, rather than the code calculated reactor power. This was done because the L9-3 power is expected to be nearly identical to the L9-1 measured power. The anticipated 100 ppm difference in initial boron concentration is expected to cause a 2.5% change in the moderator coefficient, however, the difference in the transient reactor power during the 0-65 s interval due to the boron concentration change is expected to be small. Since it was necessary to use the best available information to size the test valve, the test valve area was chosen such that the calculation which used the L9-1 measured reactor power as a boundary condition would reach the target peak pressure of 18.6 MPa (2700 psia).

3.1 Initiation Phase (0-130 s)

Since the L9-3 transient is initiated by terminating feedwater flow to the steam generator, there is very little immediate effect on the primary coolant system. As shown in Figure 2, the predicted collapsed liquid level in the steam generator downcomer drops for 50 s (to nearly one-third of its initial value) before the predicted primary coolant cold leg temperature begins to rise sharply. This increase in cold leg temperature is followed closely by a corresponding decrease in the predicted reactor power due to the moderator density decrease. Figure 3 shows that the negative reactivity feedback due to moderator density is partially offset by a positive Doppler feedback as the fuel cools. The net reactivity feedback

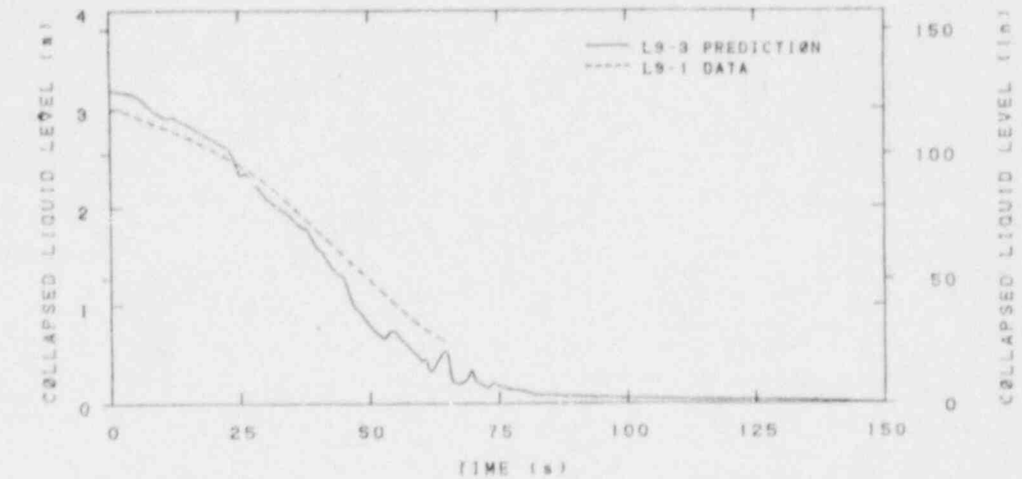

Figure 2a. Steam generator downcomer collapsed liquid level (0-150 s)

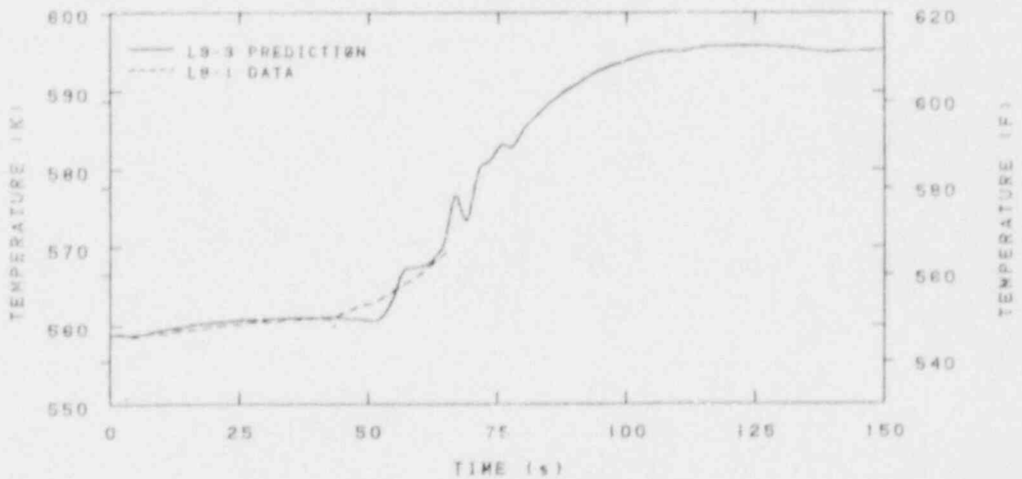

Figure 2b. Intact loop cold leg temperature (0-150 s)

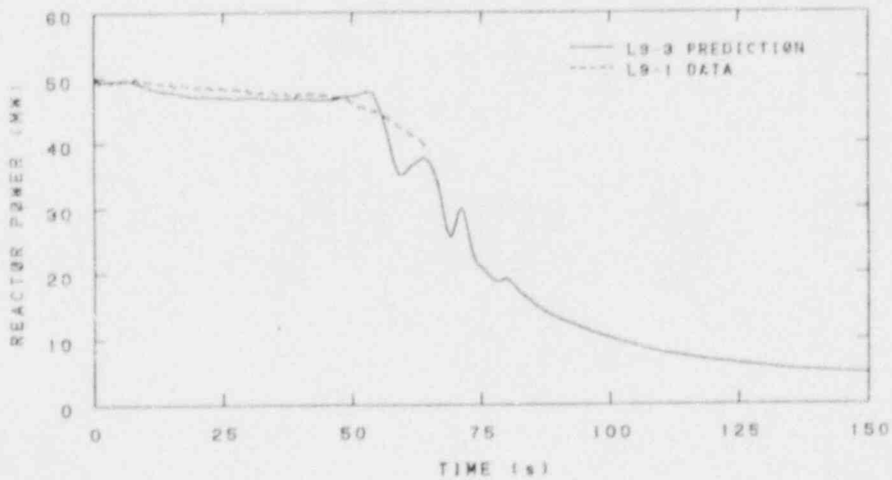

Figure 2c. Reactor power (0-150 s)

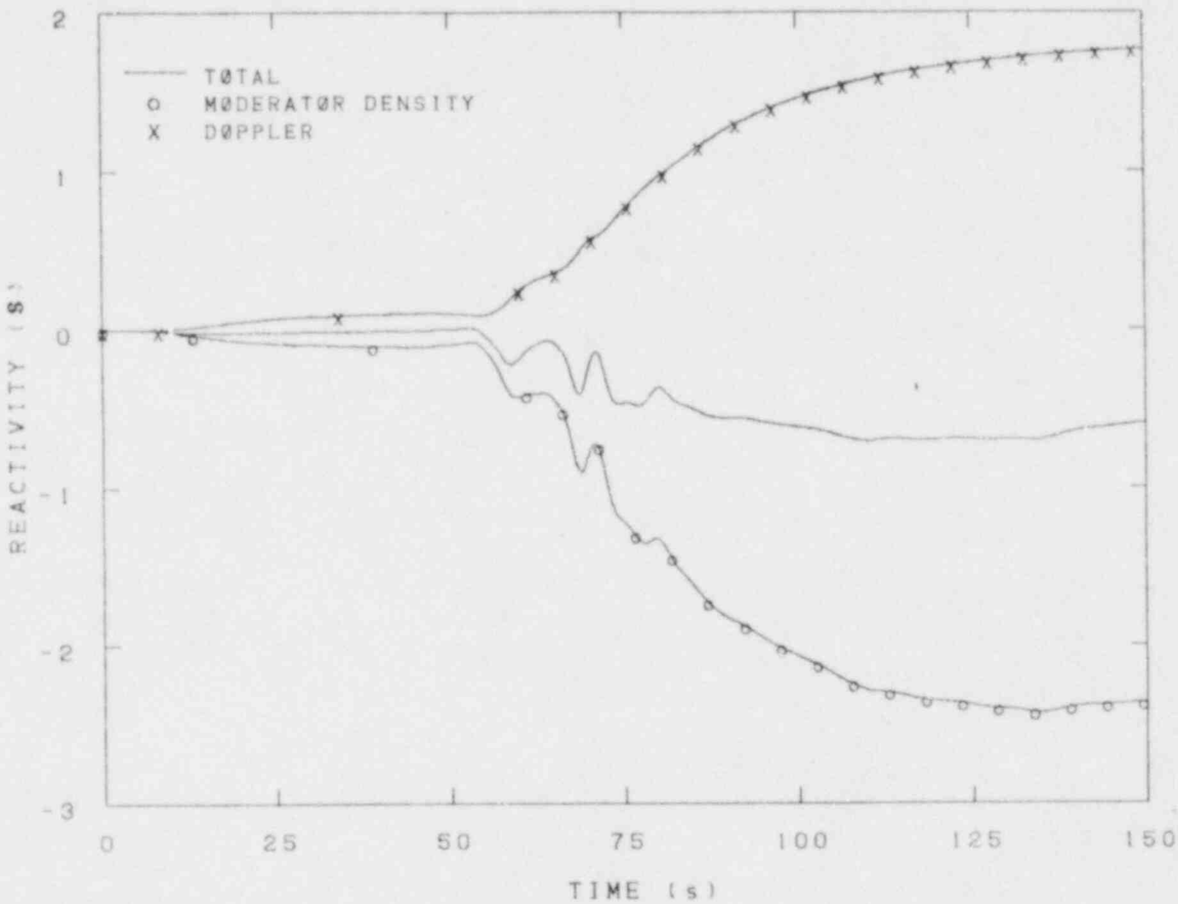

Figure 3. Moderator density, Doppler, and total reactivity (0-150 s)

however is negative. The decrease in moderator density is due only to the temperature rise and is not caused by voiding. No voiding in the primary coolant system is calculated to occur at any time during the transient except in the pressurizer.

Figure 4 compares the predicted reactor power to the predicted primary to secondary heat transfer rate. While the primary to secondary heat transfer rate drops rather sharply, the reactor power decreases much more slowly. This results in a significant period during which more energy is being added to the primary coolant than is being removed. Figure 5 shows the net energy balance on the primary coolant which includes energy added by the core, energy removed to the steam generator, and heat lost to piping and structures but neglects energy added by the primary coolant pumps and removed via the test valve and leakage.

After 50 s the primary coolant begins to absorb a significant amount of energy. The resultant heatup and volumetric expansion of the primary coolant causes the predicted pressurizer level and primary system pressure to increase as shown in Figure 6. (The difference in the initial pressurizer levels in L9-1 and L9-3 is due to different scaling decisions.) The rate of volumetric expansion of the primary coolant is so great that neither pressurizer spray nor the test PORV opening has an appreciable affect on the pressurization rate. Finally, when the test valve opens to the combined PORV, safety relief valve (SRV) position, the volumetric steam relief flow rate stops the system pressure rise. The pressure then drops to the SRV reset pressure and the valve closes to the PORV position. The pressure begins to rise again almost immediately and the valve cycles between the combined PORV, SRV position and the PORV position until the pressurizer becomes liquid filled. Since the relief flow changes from vapor to liquid, the relief mass flow increases as shown in Figure 7, however the volumetric flowrate decreases substantially. The reduced volumetric flow through the test valve results in a continued primary system pressure rise until the volumetric relief flow equals the rate of volumetric expansion of the primary coolant. The peak pressure of 12.9 MPa (2598 psia) occurs at 97 s.

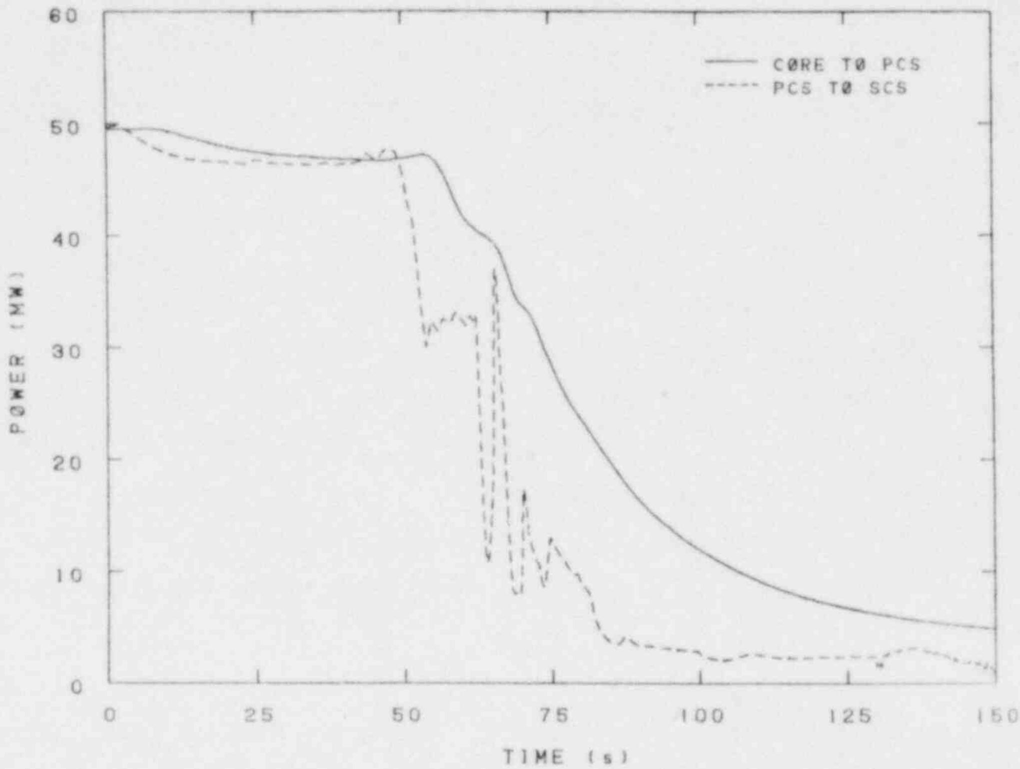

Figure 4. Comparison of power transferred from the core to the primary coolant system to the power transferred from the primary coolant system to the secondary coolant system (0-150 s)

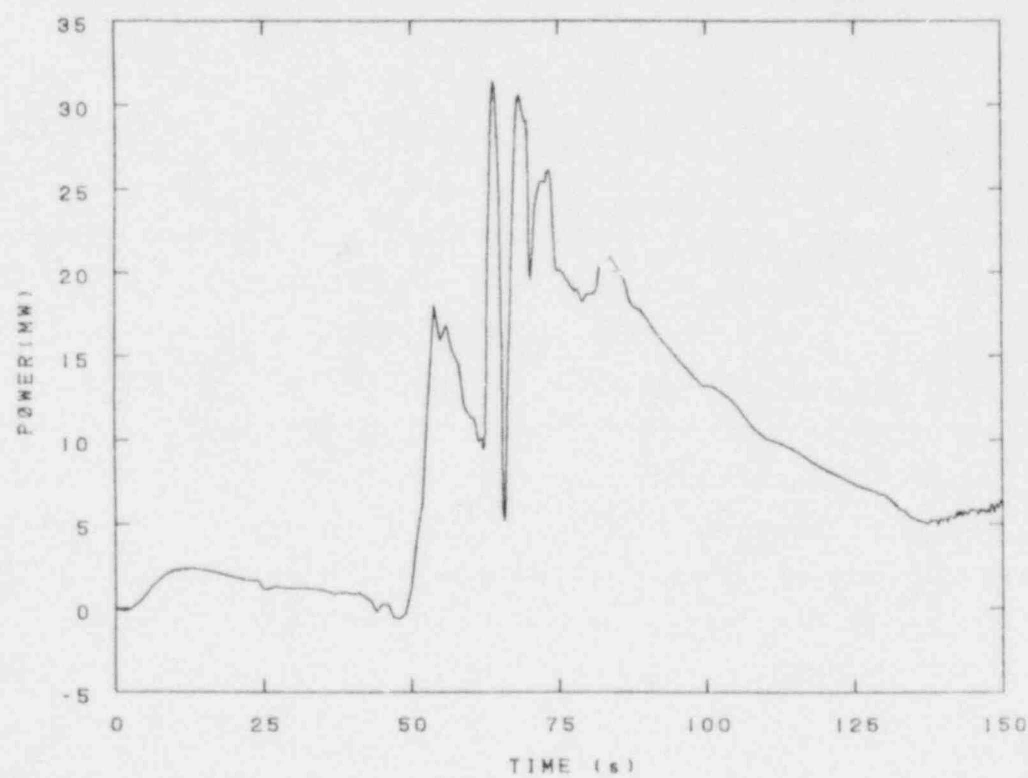

Figure 5. Net power deposited in the primary coolant (0-150 s)

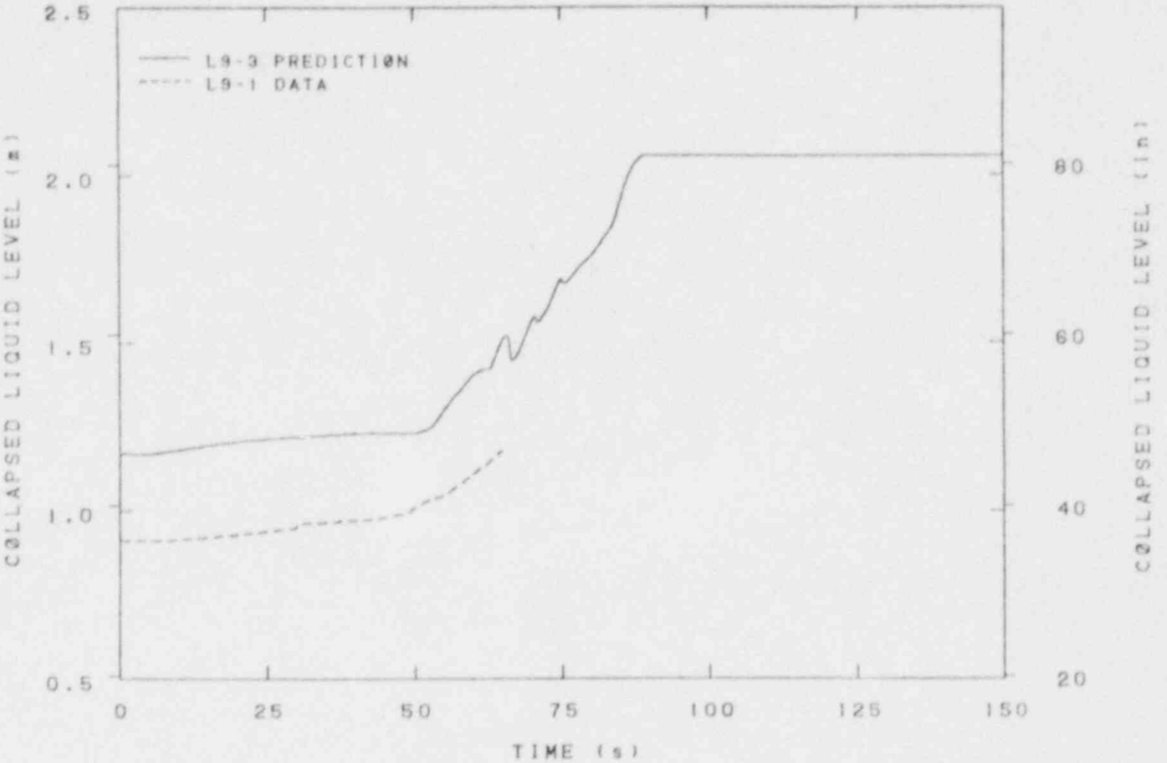

Figure 6a. Pressurizer collapsed liquid level (0-150 s)

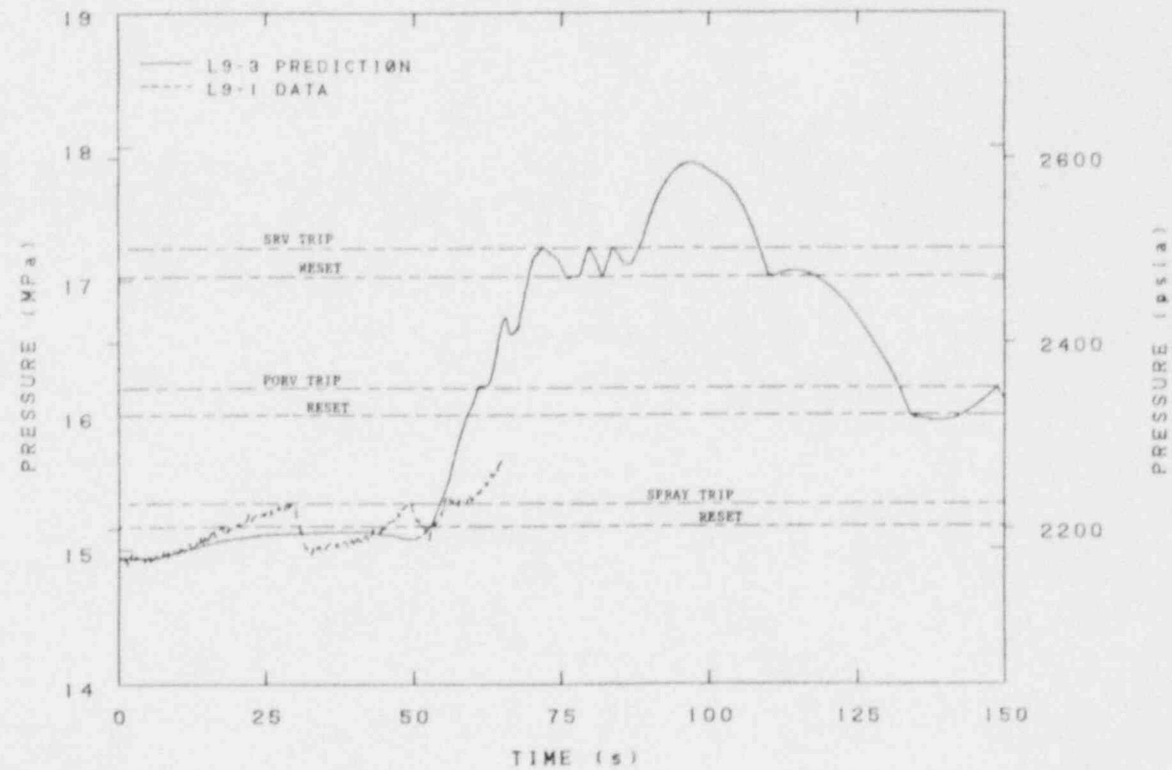

Figure 6b. Primary system pressure (0-150 s)

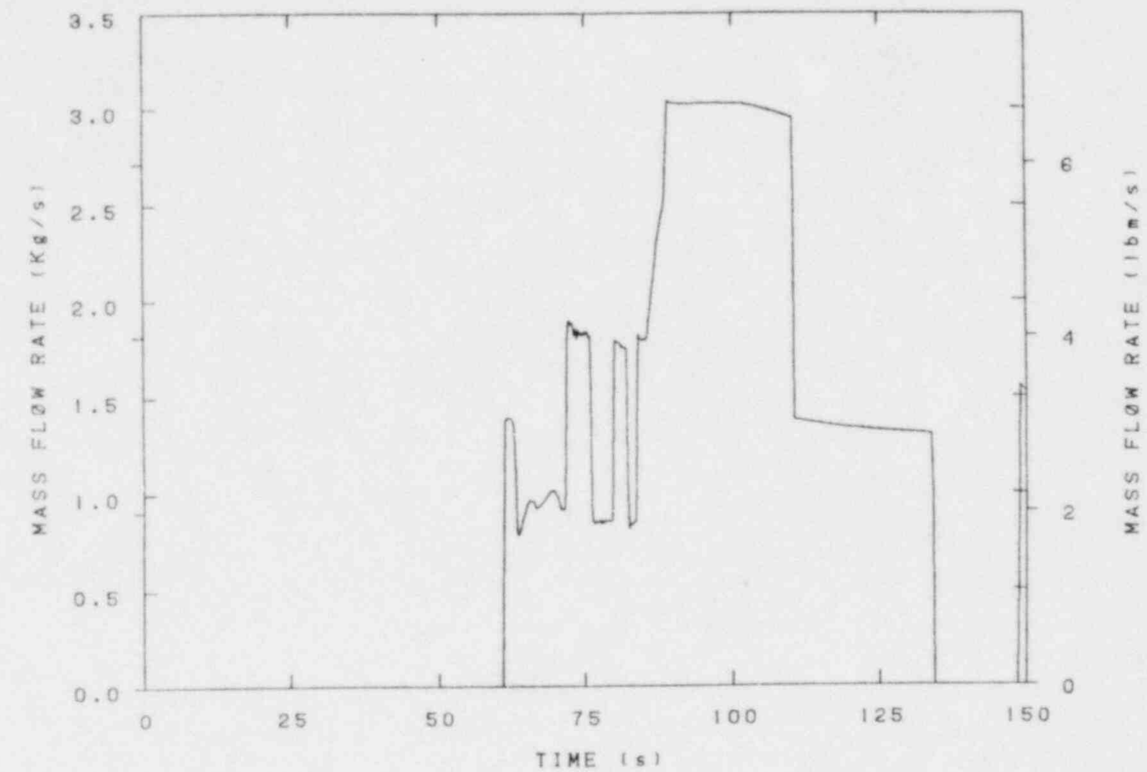

Figure 7a. Test valve mass flow rate (0-150 s)

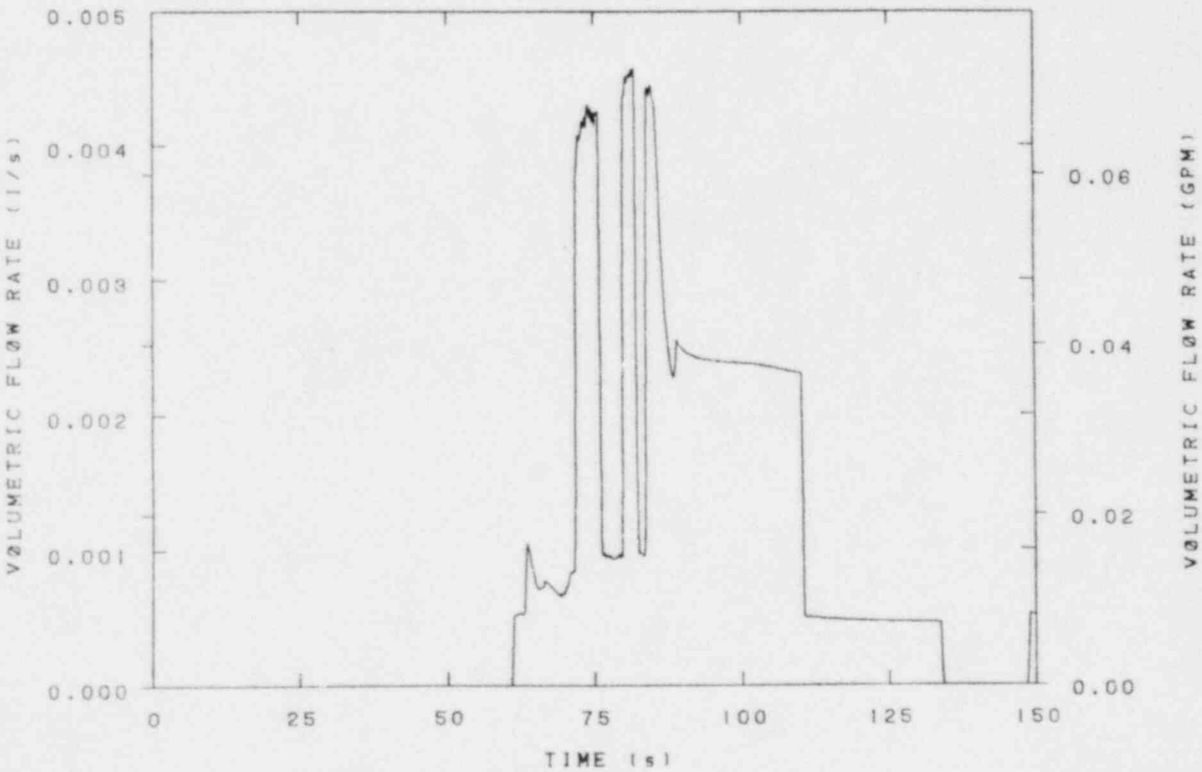

Figure 7b. Test valve volumetric flow rate (0-150 s)

As the reactor power decreases, the heatup rate diminishes as does the volumetric expansion rate which allows the primary pressure to decrease. By 110 seconds, the volumetric expansion rate has decreased below the valve capacity allowing the primary system pressure to fall below the SRV reset pressure.

3.2 Stabilization Phase (130-600 s)

As the transient enters the stabilization phase, the primary pressure continues to decrease until the PORV reset pressure is reached at about 140 s into the transient. After the valve closes, the primary pressure begins to increase due to the energy imbalance which still exists in the primary coolant system as shown in Figure 8. When the PORV trip pressure is reached the valve opens. The relief capacity of the test PORV is then more than adequate so the pressure decreases and the valve closes. As shown in Figure 9, the cycling of the test PORV continues (with increasingly longer pressurization times) with corresponding decreases in the primary system mass until the end of the stabilization phase.

3.3 Recovery Phase (after 600 s)

The recovery phase begins 600 s after the transient was initiated. During this phase the reactor operators will perform specific actions intended to bring the reactor toward a safe, shutdown condition. The defined actions are: (1) latch open the test PORV to reduce the primary system pressure, (2) turn on auxiliary feedwater to the steam generator to begin cooling the primary coolant system, and (3) initiate high pressure injection of borated water into the reactor to prevent a recriticality when the moderator temperature decreases.

The predicted primary pressure and temperature and pressurizer level response during the recovery phase is shown in Figure 10. The pressure drops sharply when recovery starts and continues to decrease after the PORV is closed at 15.0 MPa (2177 psia). This indicates that much of the depressurization is caused by the volumetric contraction of the primary fluid as the system cools down. As the cooldown and depressurization

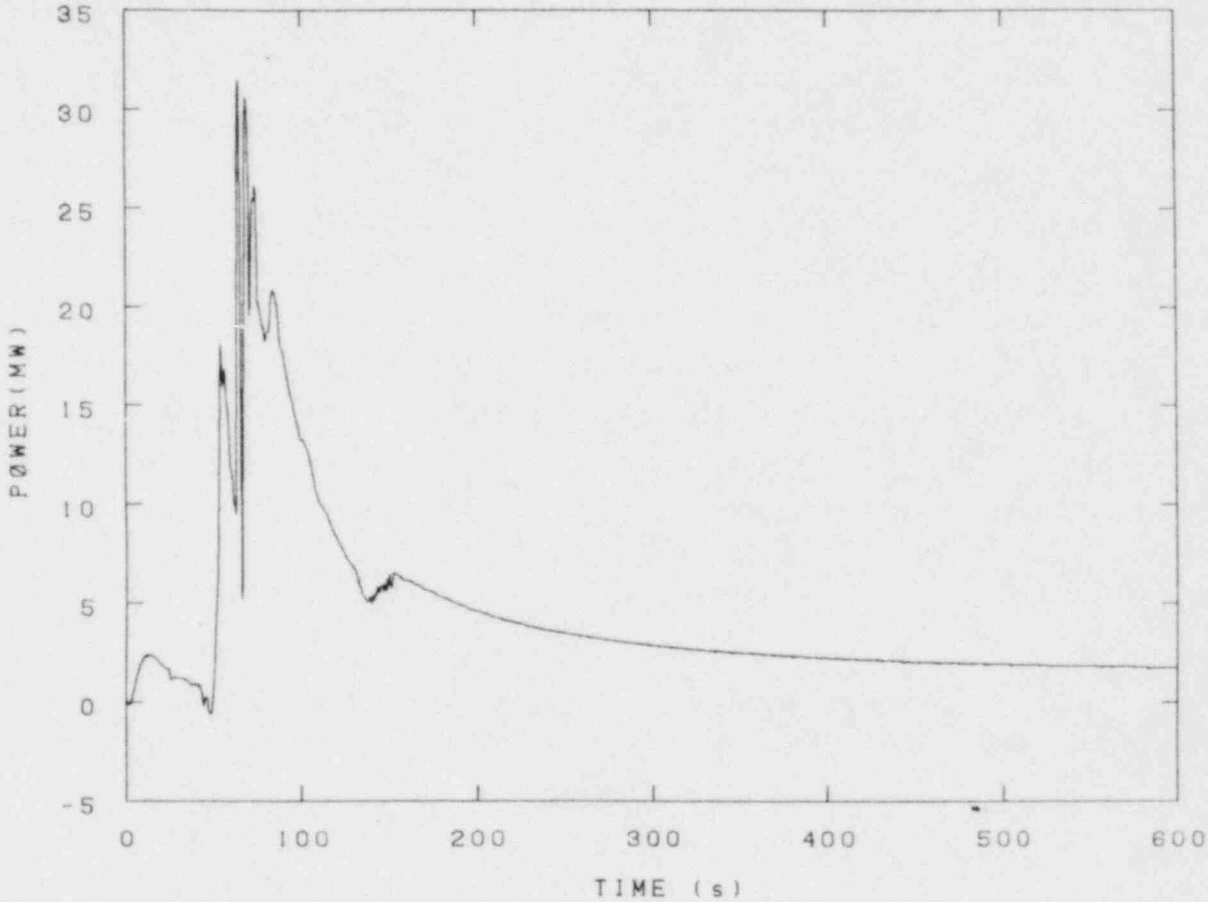

Figure 8. Net power deposited in the primary coolant (0-600 s)

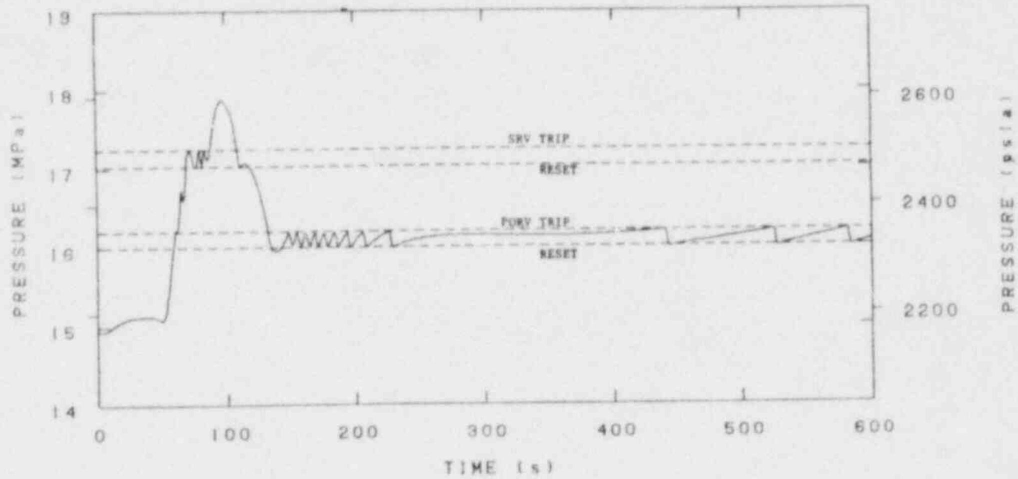

Figure 9a. Primary system pressure (0-600 s)

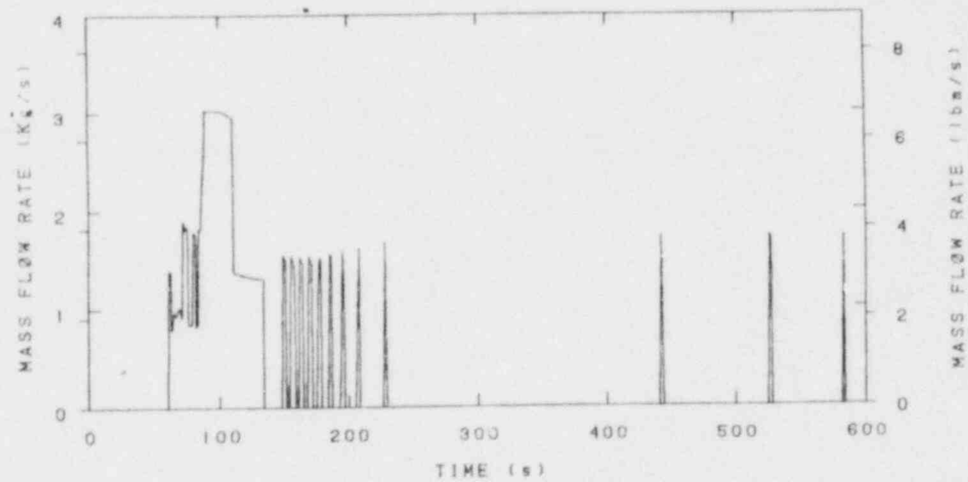

Figure 9b. Test valve mass flow rate (0-600 s)

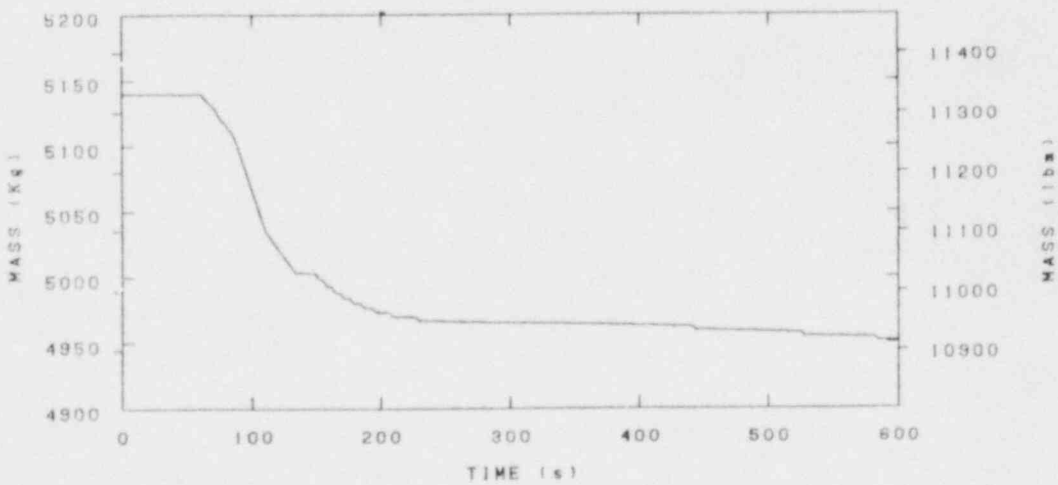

Figure 9c. Primary system total fluid mass (0-600 s)

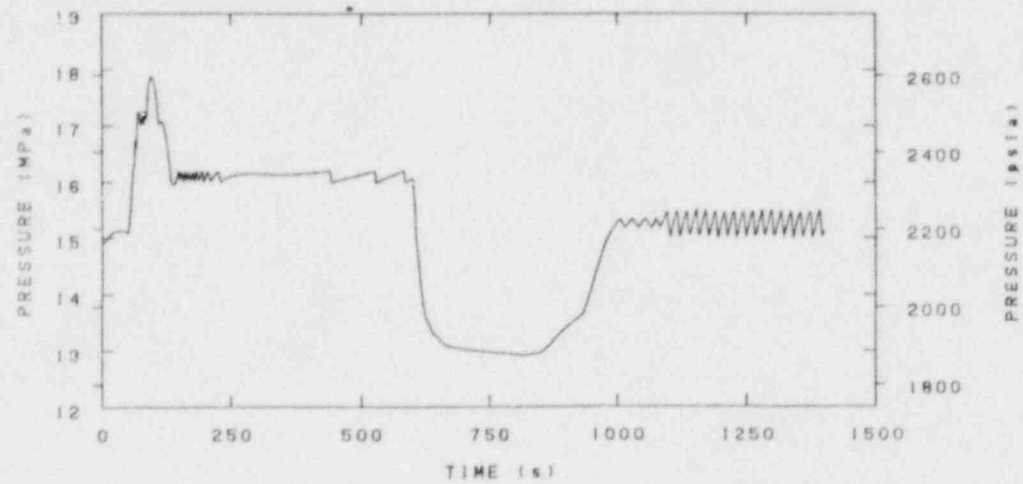

Figure 10a. Primary system pressure (0-1400 s)

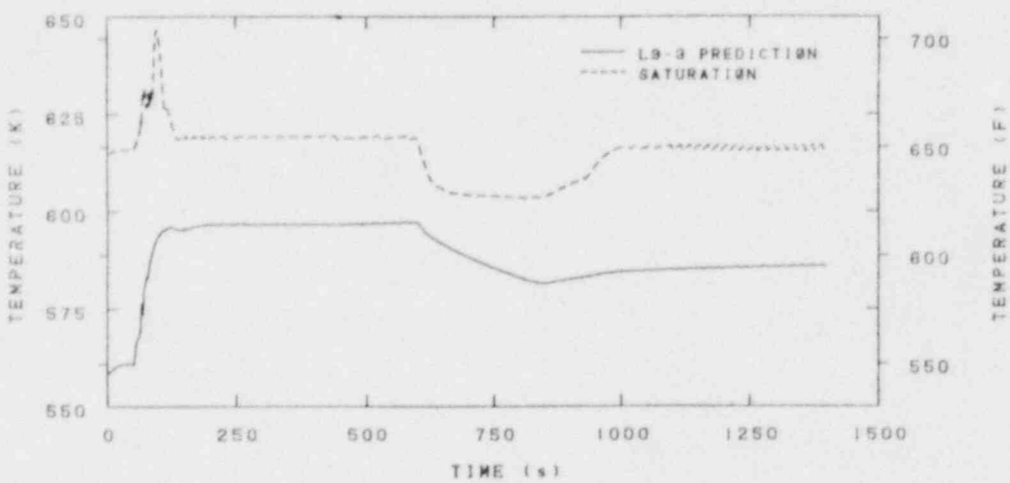

Figure 10b. Priamry system temperature and saturation temperature (0-1400 s)

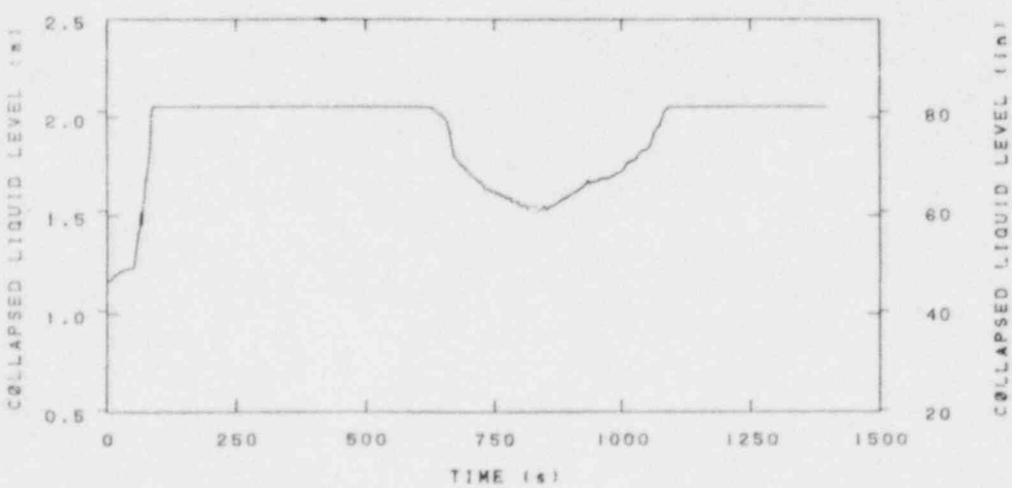

Figure 10c. Pressurizer collapsed liquid level (0-1400 s)

continue, a vapor bubble forms in the pressurizer and the level drops to within the indicating range of the instrumentation. As indicated in Figure 10b, the primary coolant loop remains subcooled throughout the recovery phase.

At 809 s, the primary coolant temperature reaches 583 K (590°F) and auxiliary feedwater is turned off, causing the cooldown to stop. The HPIS, combined with a slight volumetric expansion of the primary coolant due to a slow heatup, causes the pressurizer level to increase as does the primary system pressure. When the pressure recovers to 15.5 MPa (2250 psia) the PORV is used to control the pressure to below that value. The recovery phase ends with the plant cooling slowly with auxiliary feedwater off. The primary mode of decay heat removal is feed via HPIS and intermittent bleed via the PORV.

Since the control rods will remain withdrawn through the recovery phase, the injection of boron into the primary system is necessary to prevent a recriticality. Figure 11 shows the calculated boron concentration in the primary coolant system during recovery, however the current version of the RELAP5 code does not calculate the reactivity feedback due to changes in boron concentration. Since recriticality is neither planned nor expected during recovery, its detection will result in a termination of the experiment.

4. SENSITIVITY CALCULATION RESULTS

In any calculation of this type there are uncertainties associated with many of the input parameters. In this test the parameter of most concern is the primary system pressure, since too high a pressure will cause the test to abort by lifting the plant safety valves and too low a pressure may compromise the experiment objectives. The sensitivity calculations reported in this section show that the peak pressure is particularly sensitive to the transient reactor power, the test valve flowrate, the initial pressurizer level, the moderator density coefficient and the initial steam generator level. Table 2 summarizes the variation in peak pressure with these parameters. The following paragraphs discuss the sensitivity calculations performed and the results.

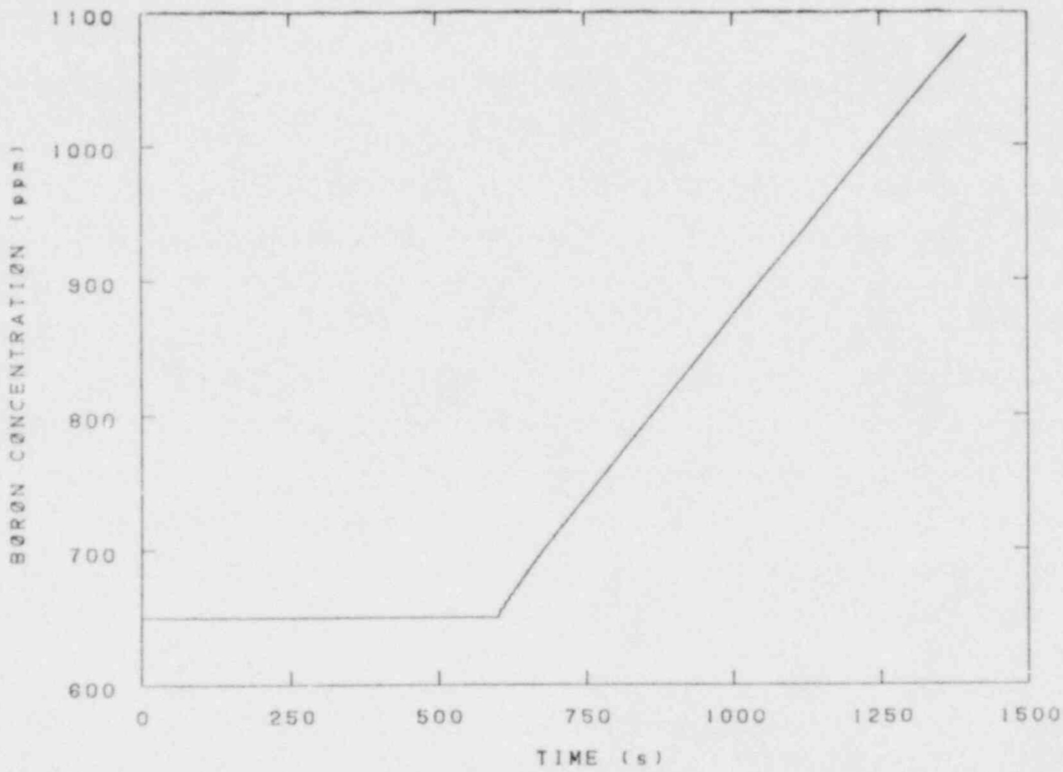

Figure 11. Boron concentration in primary coolant (0-1400 s)

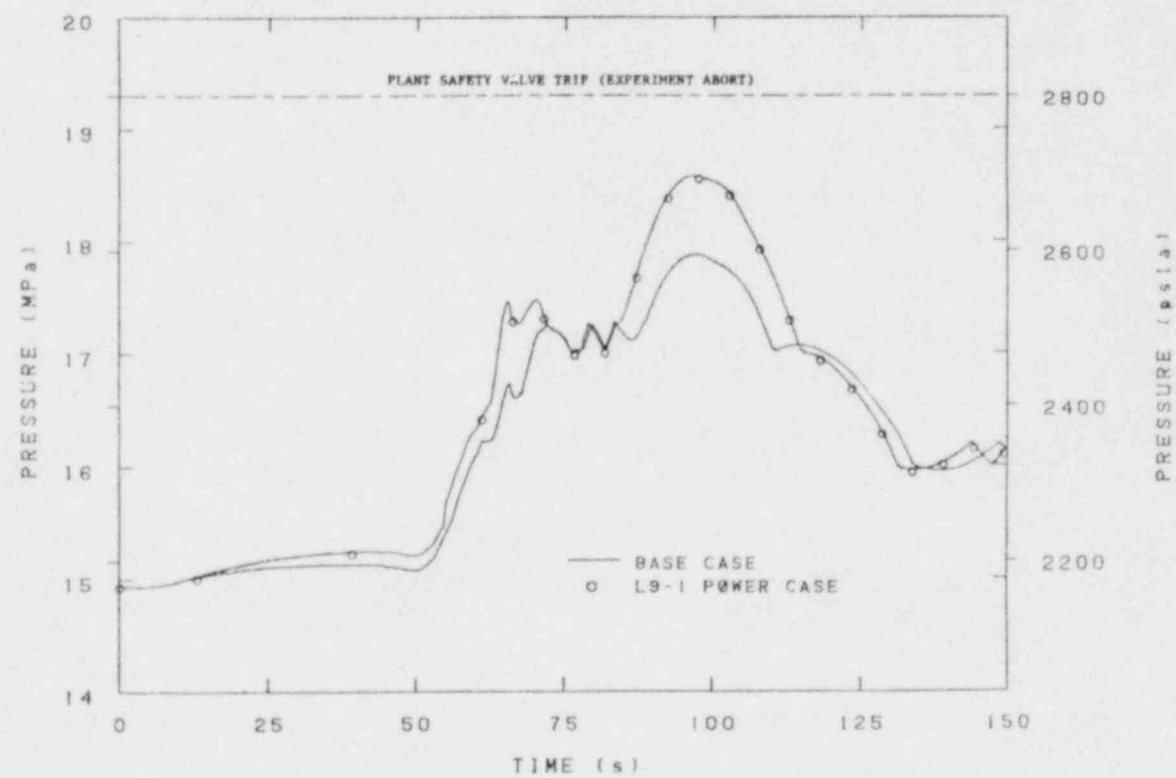

Figure 12. Sensitivity of primary system pressure to reactor power

TABLE 2. SUMMARY OF SENSITIVITY CALCULATION RESULTS

Sensitivity Parameter	Range of Variation from Nominal	Effect on Peak Primary Pressure	
Reactor power	Time dependent from L9-1	+0.696	MPa (+101 psia)
Test valve flowrate	+10%	-0.397	MPa (-58 psi)
	-10%	+0.371	MPa (+54 psi)
Pressurizer level	+.05 m (2 inches)	+1.209	MPa (+176 psi)
	-.05 m (2 inches)	-0.174	MPa (-25 psi)
Moderator density coefficient	*1.08	-0.591	MPa (-86 psi)
	*.92	+0.997	MPa (+145 psi)
Steam generator level	+.05 m (2 inches)*	+0.274 MPa	(+40 psi)
	-.05 m (2 inches)	+0.659 MPa	(+96 psi)
	-.15 m (6 inches)*	+0.523 MPa	(+76 psi)

* The results for these cases may have been influenced by the inability of the code to achieve a true steady-state for these initial conditions.

4.1 Transient Reactor Power Sensitivity

A comparison of the L9-1 measured reactor power before scram to the L9-3 predicted reactor power was shown in Figure 2c. Since the only difference in reactor power between L9-1 and L9-3 would be due to the small (100 ppm) difference in boron concentration, the reactor power in L9-3 is expected to be nearly identical to the L9-1 power until the time of scram. Therefore, a calculation was performed to determine the effect on primary system pressure of the inaccuracy of the reactor power calculation (when compared with the L9-1 measured power). For this case the L9-1 measured reactor power until 65 s and the L9-3 calculated power after 65 s were input as a time dependent power table. Figure 12 compares the primary system pressure for this calculation to the L9-3 prediction. The peak pressure for this sensitivity calculation exactly equals the target pressure for the experiment of 18.6 MPa because this calculation (rather than the nominal experiment prediction calculation) was used to set the test valve flow area. Since the effect of the reactor power could otherwise overshadow the effects of the other sensitivities, this time dependent L9-1 reactor power case will be used as the basis for the other sensitivities unless otherwise stated.

4.2 Test Valve Flowrate Sensitivity

The calculation of critical flow through relief valves is generally considered to be an area of high uncertainty. To determine the effect of errors in the calculated critical flow, two sensitivity calculations were performed in which the areas of both the PORV and the combination positions were varied by ±10%. Figure 13 compares the primary system pressures for these calculations to that of the time dependent reactor power base case. As expected, the higher relief flow rate results in a lower peak power.

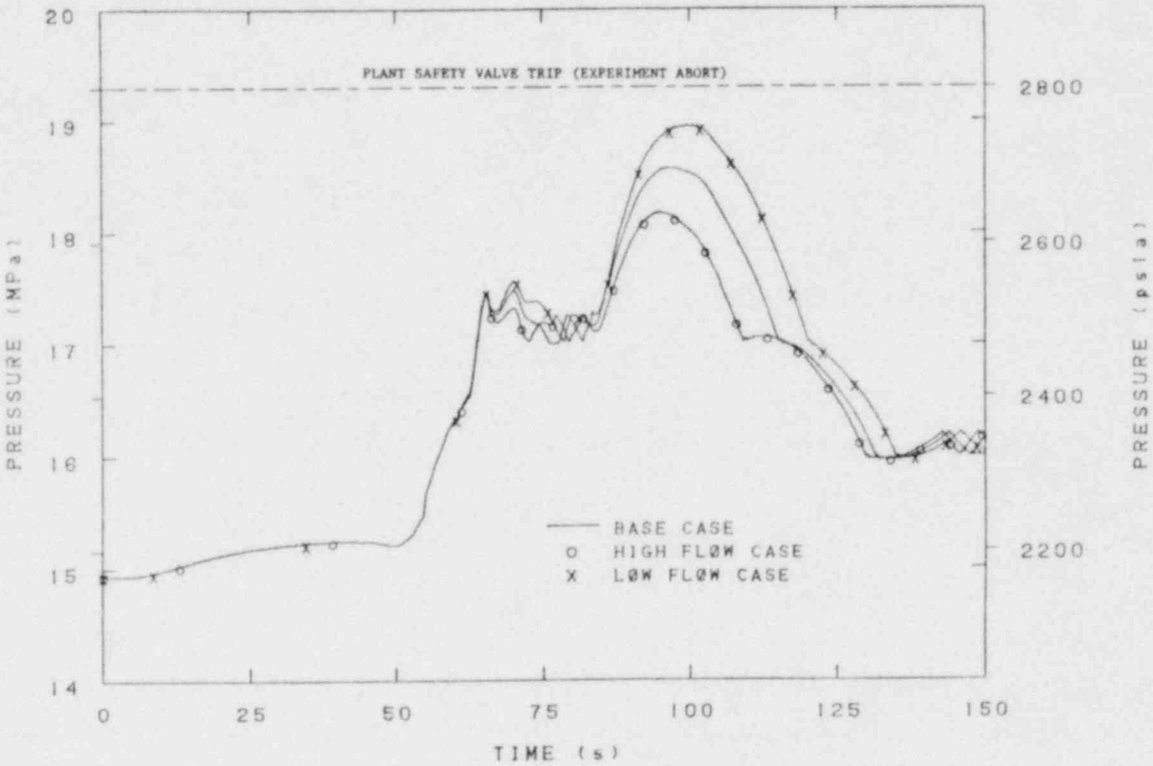

Figure 13. Sensitivity of primary system pressure to test valve flowrate

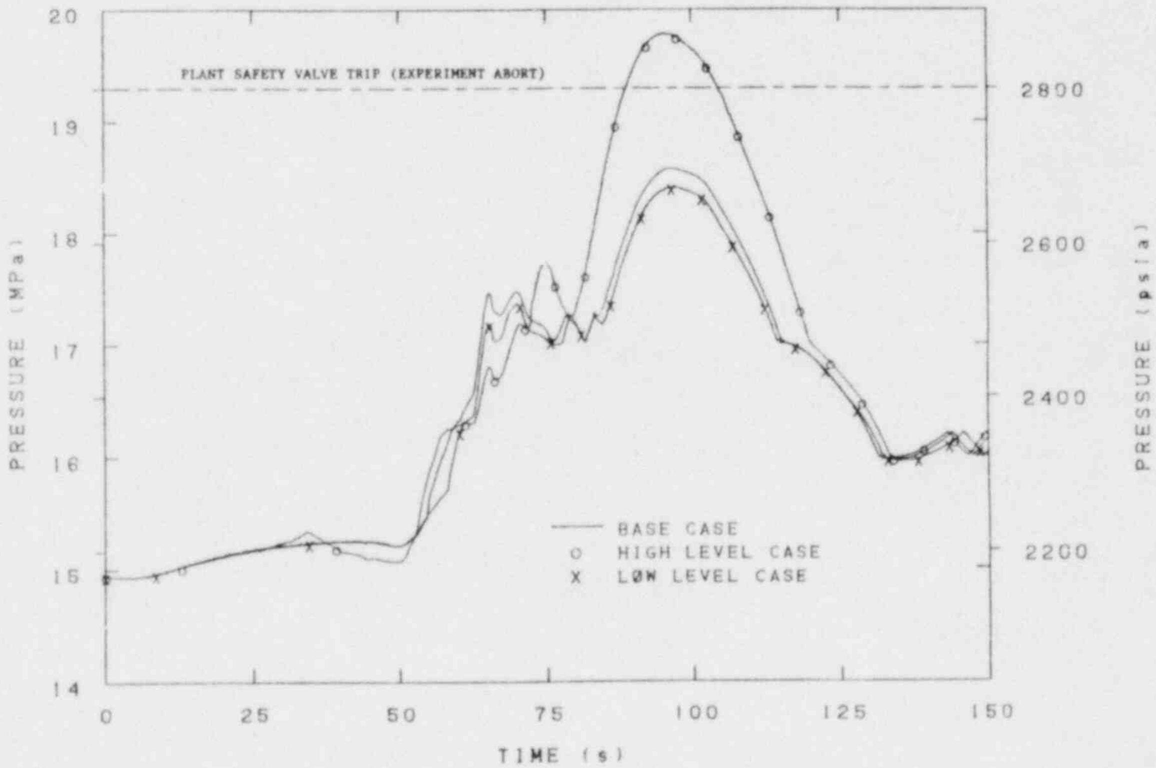

Figure 14. Sensitivity of primary system pressure to initial pressurizer level

4.3 Initial Pressurizer Level Sensitivity

The specification for the initial pressurizer level is 1.168 ± 0.51 m (46 ± 2 inches). The 1.168 m (46 inch) value was used in the nominal experiment prediction. To determine the effect of the initial level being at the top or bottom of the acceptable band, two sensitivity calculations were performed. Figure 14 compares the primary system pressures for these calculations to that of the time dependent reactor power base case. The higher pressurizer level results in a higher peak primary system pressure which would exceed the plant safety valve setting, causing a test abort.

4.4 Moderator Density Feedback Sensitivity

The LOFT sytem pressure and temperature during L9-3 is beyond the range of the available reactor kinetics data. Therefore, there is some uncertainty in the extrapolation of the data into the range of interest. For Experiment L9-3, the moderator density feedback coefficient is likely to be important because it is responsible for the reactor shutdown. To determine the effect of variations in the moderator density feedback, two sensitivity calculations were performed in which the coefficients were multiplied by 1.08 and 0.92, respectively. The base case for this sensitivity study was a calculation in which the test valve area was adjusted to achieve a peak pressure of 18.6 MPa with the nominal moderator density feedback coefficients. Figure 15 compares the primary system pressures for the three calculations. Since the higher (more negative) coefficients cause a more rapid shutdown of the fission process, that case results in a lower peak pressure. The less negative coefficients, however, result in a peak pressure slightly above the plant safety valve setpoint and could result in a test abort.

4.5 Steam Generator Initial Level Sensitivity

Another initial condition variation which could noticably affect the transient is the steam generator initial level. To determine its effect on

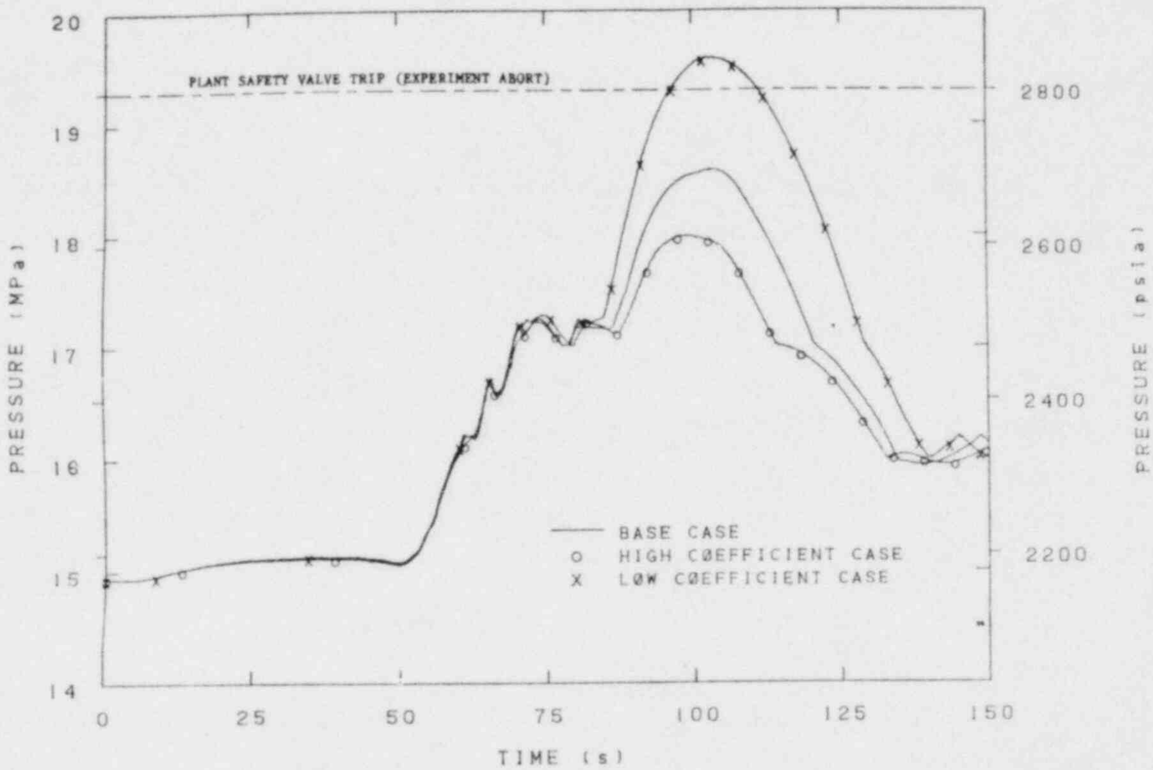

Figure 15. Sensitivity of primary system pressure to moderator density reactivity feedback

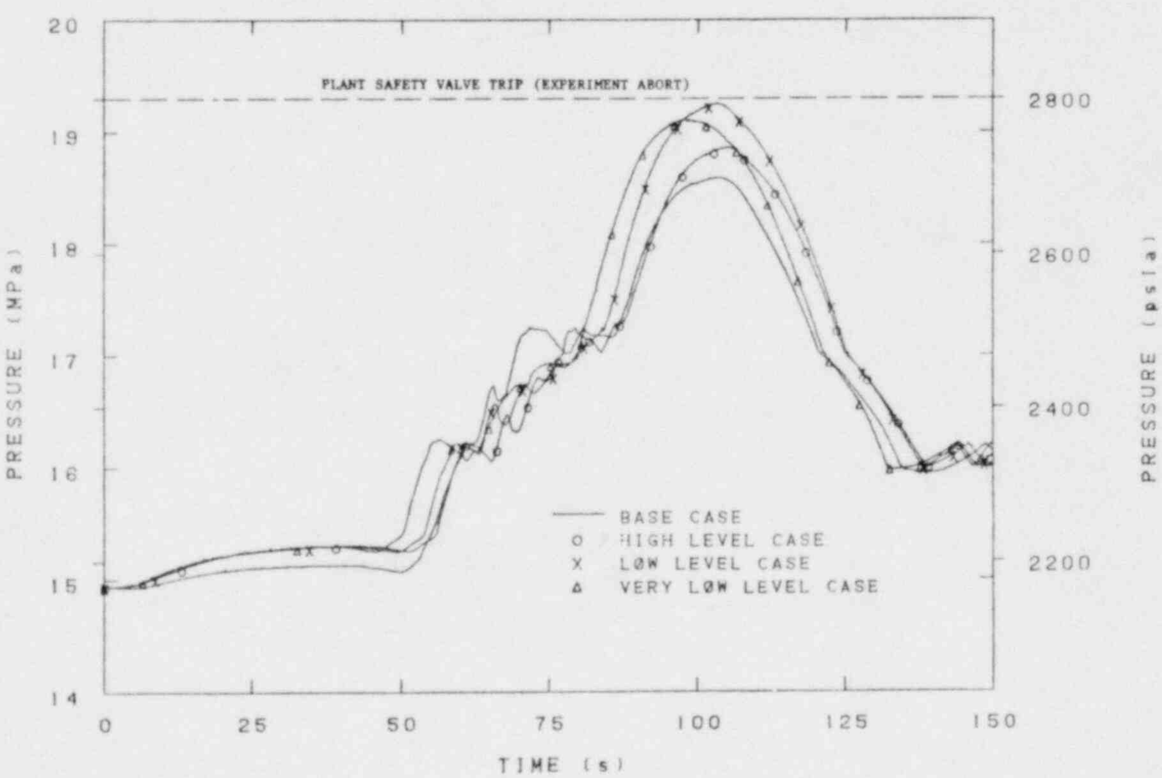

Figure 16. Sensitivity of primary system pressure to initial steam generator level

the calculated transient, three senstivity calculations were performed. For two of these the initial level was in error by a nominal ±0.05 m (±2 inches), and for the third case it was 0.15 m (6 inches) low. Considerable difficulty was encountered in the steady-state calculations which provided the initial conditions for these cases. Two of the three, in fact, failed to achieve a true steady-state. The degree to which this problem may have affected the results is not known. The base case for this sensitivity study was the same as for the moderator density coefficient sensitivity in Section 4.4. Figure 16 shows the effect of steam generator initial level on the primary system pressure calculation. The peak pressure was higher than the base case pressure for all three initial levels. The reasons for this unexpected result are being investigated.

5. CONCLUSIONS

The results of the nominal experiment prediction calculation and the sensitivity studies indicate there is a high probability LOFT Experiment L9-3 will meet its objectives. The sensitivity studies reveal that the peak primary system pressure is particularly sensitive to a number of parameters. Since the plant safety valves will lift, aborting the test, if the primary system pressure exceeds the target pressure (18.6 MPa or 2700 psia) by more tha 0.69 MPa or 100 psi, there is a significant chance that one or more off-nominal conditions will cause the test to abort. On the other hand, several off-nominal conditions could effectively cancel one another allowing the test to run to its normal termination.

L9-3 is expected to provide data on the response of the LOFT reactor system to a loss of feedwater ATWS. Barring an abnormal termination due to high pressure, the recovery phase of the experiment will demonstrate a method of bringing the LOFT nuclear reactor to a safe, shut-down condition.

REFERENCES

1. P. Kuan, LOFT Experiment Definition Document Anticipated Transient Test Series Nuclear Test L9-3, EGG-LOFT-5732, March 1982.

2. J. P. Adams, Quick-Look Report on LOFT Nuclear Experiment L9-1/L3-3, EGG-LOFT-5430, April 1981.

3. M. L. McCormick-Barger, et.al., Experiment Data Report for LOFT Anticipated Transient with Multiple Failures Experiment L9-1 and Small Break Experiment L3-3, NUREG/CR-2119, EGG-2101, June 1981.

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

5. A. J. Scott, "Temperature Coefficients and Boron Worth for LOFT I at Beginning of Life", LTR-111-52, August 31, 1973.

APPENDIX A

EXPERIMENT L9-3 AND LOFT FACILITY DESCRIPTIONS

APPENDIX A

EXPERIMENT L9-3 AND LOFT FACILITY DESCRIPTIONS

The L9-3 Experiment will simulate a loss of feedwater without scram transient in a commercial pressurized water reactor.

Anticipated Transients Without Scram (ATWS) for light water reactors has been a long-standing unresolved safety issue of the U.S. Nuclear Regulatory Commission (NRC). The significance of ATWS in reactor safety is that some ATWS events can result in high system pressures which can potentially lead to fuel damage and the potential release of a large amount of radioactivity into the environment.

In evaluating ATWS accidents the NRC lists ten initiating events for pressurized water reactors (PWRs), which are expected to occur one or more times during the life of a nuclear power unit.[3] These events can be classified into four categories, i.e., (a) reactivity related accidents (rod withdrawal, boron dilution, inactive primary loop startup, load increase, excessive cooldown), (b) degradation of reactor heat transfer (loss of primary flow, loss of electrical load, loss of normal electrical power), (c) degradation of reactor heat sink (loss of normal feedwater), and (d) primary system depressurization caused by accidental opening of a pressurizer relief valve. The L9-3 experiment is intended to simulate the important physical conditions following a loss of feedwater without scram transient hypothesized for future commercial PWRs conforming to the acceptance criteria proposed by the NRC.[1]

Upon loss of feedwater to the steam generators in a PWR power plant, the heat transfer from the primary to the secondary system is degraded with the decrease in steam generator secondary inventory. Normally the reactor will trip (insert control rods to shut down the reactor, or scram) on a signal of low feedwater flow or low steam generator level. In the unlikely event that a scram does not occur, the steam generator secondary will soon boil dry and most of the heat produced by the reactor core will be dissipated in the primary fluid, raising its temperature. The expansion of the primary fluid associated with its temperature rise at first compresses the vapor space of the pressurizer, forcing the relief and safety valves to open. Subsequently the pressurizer will be filled with liquid water and the system pressure will continue to rise to a maximum when the volumetric relief flow rate equals the volumetric expansion rate of the primary fluid at constant pressure. It is this maximum pressure that constitutes one of the main safety concerns of ATWS events.

Another major concern of a loss of feedwater without scram accident is the long-term shutdown capabilities of PWR systems after the initial peak pressure has passed. The L9-3 experiment will explore a way to depressurize the primary system by timely latching open the PORV and by using the auxiliary feedwater system for additional heat removal such that high concentration boron solution can be injected into the system to permanently shut down the reactor. This will be at least a first step in bringing the reactor to a stable cold shutdown condition after a loss of feedwater without scram accident.

1. EXPERIMENT OBJECTIVES

The intent of the L9-3 experiment is to identify and evaluate LOFT system thermal-hydraulic response characteristics during a loss of feedwater without scram experiment. Programmatic and specific test objectives are provided in the listing below (Reference A-1)

Programmatic Objectives:

1. Provide experimental data for benchmarking PWR vendors' ATWS computer codes as required by the NRC proposed ATWS rule (USNRC SECY-80-409).

2. Evaluate alternate methods of achieving long term shutdown (without the insertion of control rods) following an ATWS event, to address concerns defined in the proposed NRC staff rule (Federal Register Vol. 46 No. 226).

Test Objectives:

1. To achieve a maximum primary system pressure that is several measuring standard errors above the code safety valve opening pressure setpoint but below 110% of the setpoint pressure.

2. To determine the transient reactor power by using available neutron flux instrumentation and measured core thermal-hydraulic parameters to assess the applicability of the point kinetics model used in predicting transient reactor power.

3. To determine the steam generator secondary dryout behavior and its effect on the primary system response characteristics.

4. To determine the two-phase and subcooled flow characteristics of the experimental pressurizer PORV and safety valve at high pressures, ≥17 MPa (2500 psia).

2. LOFT FACILITY DESCRIPTION

The LOFT facility is described in detail in Reference A-2. The LOFT instrumentation and major components of interest for Experiment L9-3 are shown in Figures A-1 and A-2.

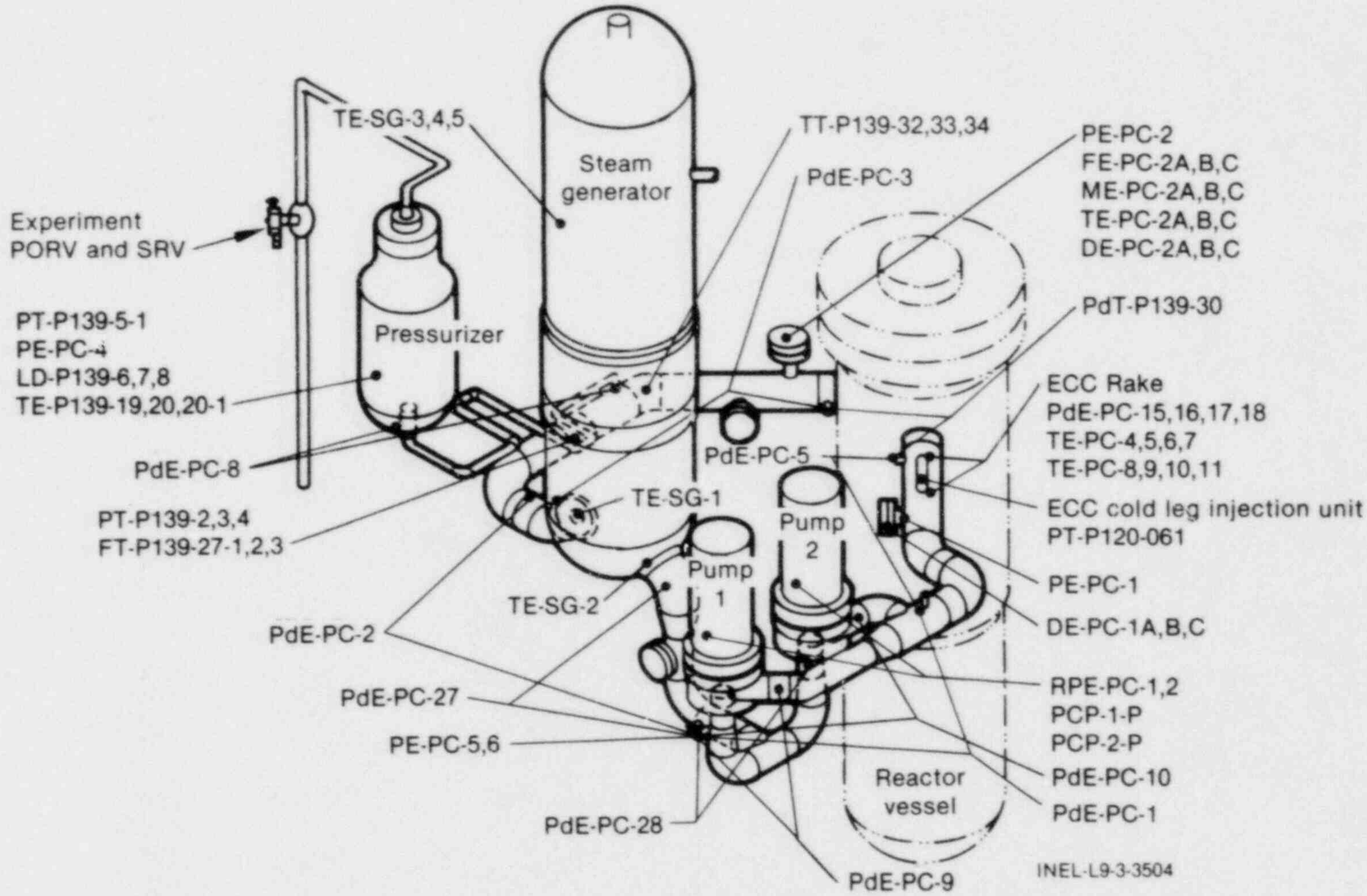

Figure A-1. LOFT intact loop components showing thermo-fluid instrumentation

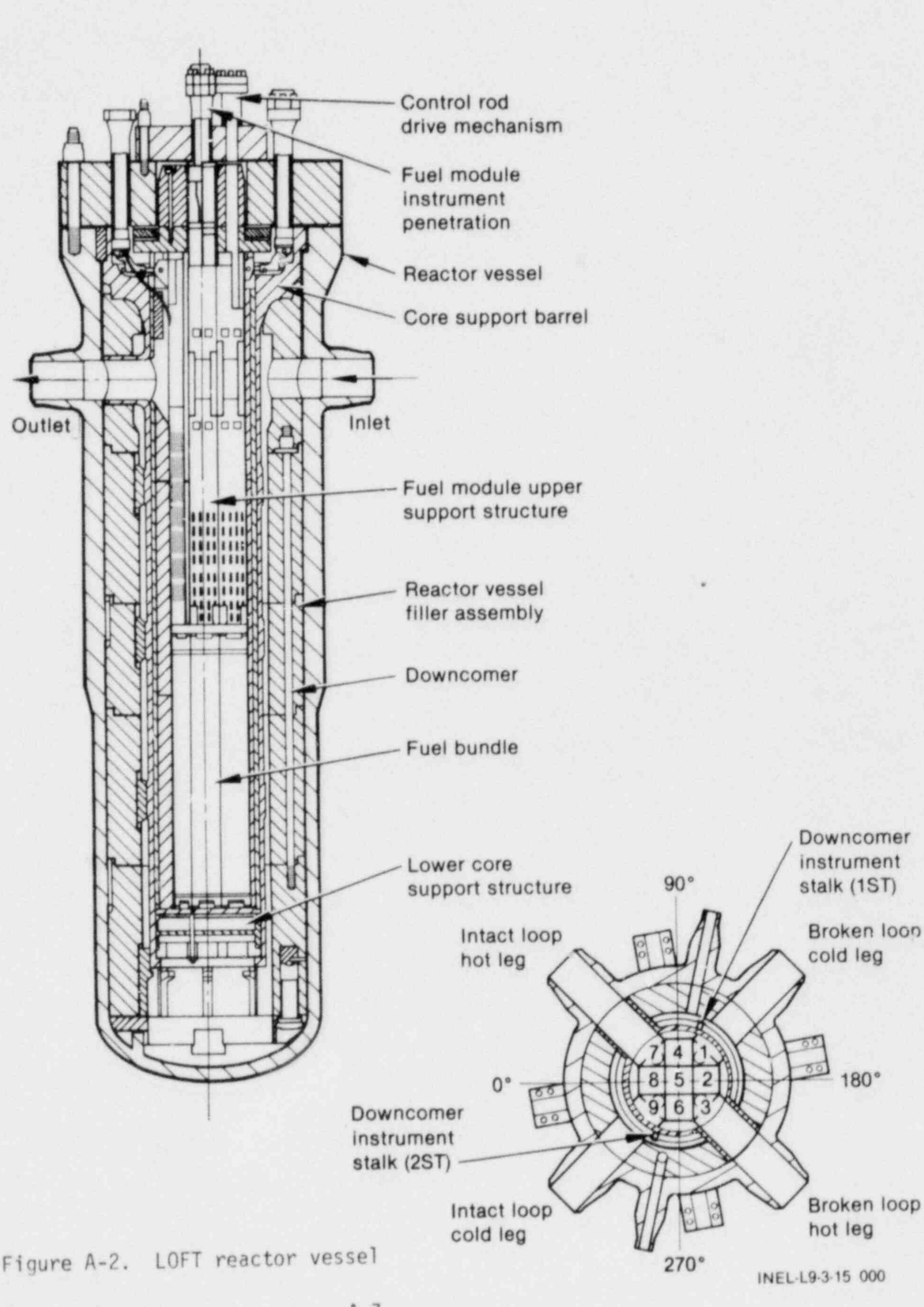

Figure A-2. LOFT reactor vessel

3. REFERENCES

A-1. P. Kuan, LOFT Experiment Definition Document Anticipated Transient Test Series Nuclear Test L9-3, EGG-LOFT-5732, March 1982.

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

APPENDIX B

RELAP5 INPUT DATA, TIME ZERO EDIT, AND CODE UPDATES FOR L9-3 EXPERIMENT PREDICTION NOMINAL CALCULATION

APPENDIX B

RELAP5 INPUT DATA, TIME ZERO EDIT, AND CODE UPDATES FOR L9-3 EXPERIMENT PREDICTION NOMINAL CALCULATION

The RELAP5 input deck listing for the L9-3 experiment prediction nominal calculation and the RELAP5 time zero edit are on microfiche in the pouch on the inside of the report back cover. Also included on the microfiche is a complete listing of the updates to cycle 12 of RELAP5 which were used for this analysis. The title of the microfiche is "L9-3 EP Appendix B, RELAP5 Input Data, Time Zero Edit, and Code Updates".

APPENDIX C

DETAILED TEST PREDICTION DATA FOR

EXPERIMENT L9-3 NOMINAL CALCULATION

APPENDIX C

DETAILED TEST PREDICTION DATA FOR L9-3 EXPERIMENT PREDICTION NOMINAL CALCULATION

Detailed test prediction data for Experiment L9-3 are provided on microfiche in the pouch on the inside of the report back cover. The title of the microfiche is "L9-3 EP Appendix C, Detailed Test Prediction Data". These figures are computer plots of selected variables calculated using RELAP5. These data have been transmitted to the LOFT Data Bank for future comparison with experiment results. The calculated variables and figure numbers are as follows:

Figure C-1. Average density - intact loop cold leg.

Figure C-2. Average density - intact loop hot leg.

Figure C-3. Mass flow rate - intact loop hot leg.

Figure C-4. Mass flow rate - pressuriz[illegible]ge line.

Figure C-5. Mass flow rate - steam line.

Figure C-6. Mass flow rate - steam bypass.

Figure C-7. Mass flow rate - main feedwater.

Figure C-8. Mass flow rate - auxiliary feedwater.

Figure C-9. Mass flow rate - ECC.

Figure C-10. Mass flow rate - test valve.

Figure C-11. Volumetric flow rate - test valve.

Figure C-12. Collapsed liquid level - steam generator.

Figure C-13. Collapsed liquid level - pressurizer

Figure C-14. Pressure - intact loop hot leg.

Figure C-15. Pressure - pressurizer.

Figure C-16. Pressure - steam generator steam dome.

Figure C-17. Fluid temperature - intact loop cold leg.

Figure C-18. Fluid temperature - intact loop hot leg.