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EGG-LOFT-5983
Project No. P 394
August 1982

NRC Research and/or Technical Assistance Rept

BEST ESTIMATE PREDICTION FOR LOFT ANTICIPATED TRANSIENT SLOW AND FAST ROD WITHDRAWAL EXPERIMENT L6-8B

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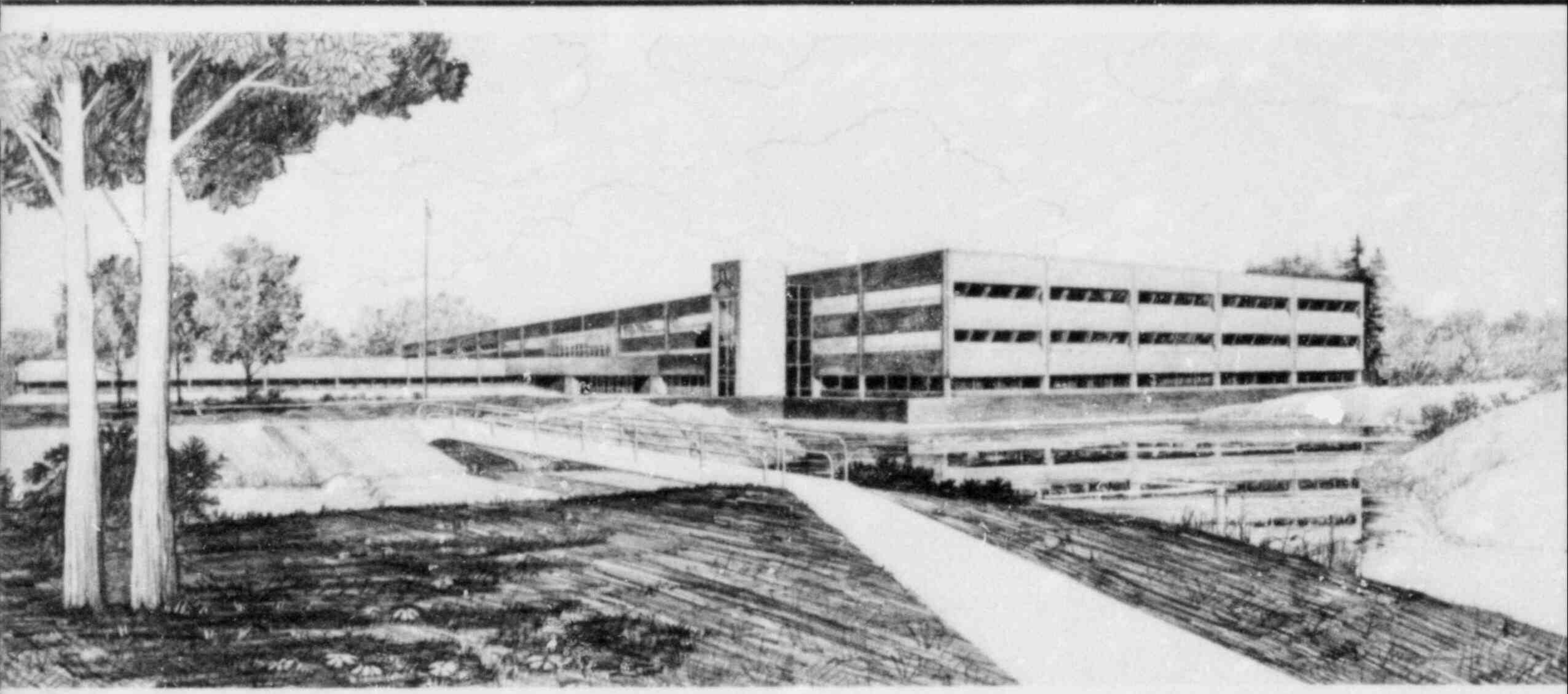

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Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6048

EG&G Idaho

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FORM EG&G-398
(Rev. 11-81)

INTERIM REPORT

Accession No. ____________________

Report No. EGG-LOFT-5983

Contract Program or Project Title:

LOFT Program Division

Subject of this Document:

BEST ESTIMATE PREDICTION FOR LOFT ANTICIPATED TRANSIENT SLOW AND FAST ROD WITHDRAWAL EXPERIMENT L6-8B

Type of Document:

Experiment Prediction Analysis Report

Author(s):

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Date of Document:

August 1982

Responsible NRC/DOE Individual and NRC/DOE Office or Division:

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

EG&G Idaho, Inc
Idaho Falls, Idaho 83415

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THE LOFT SUBCOMMITTEE OF THE EG&G PRETEST PREDICTION CONSISTENCY COMMITTEE HAS REVIEWED THE RELAP5 MODEL AND PREDICTED RESULTS FOR LOFT EXPERIMENT L6-8B AND FINDS THEM TO BE CONSISTENT WITH ACCEPTED GUIDELINES.

VHRansom

CODE DEVELOPMENT PROGRAM

TRCharlton/CAD

CODE ASSESSMENT & APPLICATIONS PROGRAM

JLPerryman/RAD

WATER REACTOR RESEARCH TEST FACILITY PROGRAM

Richard H. Smith

THERMAL FUELS BEHAVIOR PROGRAM

ABSTRACT

The RELAP5 code (a one-dimensional, two-fluid model, reactor transient analysis program with point kinetics) was used to simulate the Loss-of-Fluid Test (LOFT) facility during the L6-8B anticipated transient experiments, which will consist of two control rod withdrawal tests. The L6-8B experiment will simulate the range of expected reactivity insertions resulting from unintended control rod assembly withdrawal events in a commercial four-loop pressurized water reactor. In Test L6-8B-1 all four control rod banks of the LOFT reactor will be withdrawn from the core at a rate of 0.19 cm/s (4.5 in./min) and the reactor is predicted to scram on high pressure. In Test L6-8B-2 the rod withdrawal rate is 1.02 cm/s (24 in./min) and because of the faster reactivity insertion, the reactor is predicted to scram on high power. This report includes the results of the experiment predictions and sensitivity calculations on moderator density and Doppler reactivity feedback coefficients and environmental heat losses. The results indicate that, if conducted as planned, the L6-8B experiments will meet their stated objectives.

SUMMARY

This report documents the Loss-of-Fluid Test (LOFT) Experiment L6-8B pretest calculations of the system thermal-hydraulic response using the RELAP5/MOD1 computer code. The L6-8B experiments are anticipated transient experiments consisting of two control rod withdrawal tests. The objectives of the L6-8B experiments are to obtain plant data on the integral system response, to use the data for evaluating code capabilities, and to evaluate plant automatic recovery methods.

In the L6-8B-1 experiment the four control rods banks of the LOFT reactor will be withdrawn from the initial 1.312 m (51.66 in.) position above the bottom of the core at a rate of 0.19 cm/s (4.5 in./min). This gives an average reactivity insertion of 0.6 ¢/s. In the RELAP5/MOD1 simulation this reactivity insertion is described as a time dependent "scram" table. The inserted positive reactivity results in a reactor power increase. Because the steam generator control valve does not move during the experiment, the primary coolant system (PCS) fluid and fuel heatup result in a negative reactivity feedback. Thus, after 10 s, the total calculated reactivity stabilizes at about 1.4 ¢ positive reactivity, reducing the rate of reactor power increase. Due to PCS fluid heatup, the pressurizer liquid level and PCS pressure increase. The reactor is scrammed at 105.6 s due to the pressure reaching the high pressure setpoint of 15.74 MPa (2282 psia). At the time of the scram, the reactor power is 43.7 MW, the total reactivity is 1.3 ¢, the Doppler feedback effect is -25 ¢ and the moderator feedback effect is -30 ¢.

In the L6-8B-2 experiment the control rods will be located initially at the 1.016-m (40-in.) level above the bottom of the core and will be withdrawn at a rate of 1.02 cm/s (24 in./min), giving an average 5.5 ¢/s reactivity insertion. Due to faster insertion in this test, reactor power increases more rapidly than in the L6-8B-1 test. Because of a relatively long time constant for heat conduction in the fuel and the short time frame

of the test, the negative reactivity feedback effect is Doppler dominated. Reactor scram due to high power occurs at 13.2 s. Calculated total reactivity at scram is 9.3 ¢, Doppler feedback reactivity is -39 ¢ and moderator feedback reactivity is -23 ¢. The PCS hot leg pressure was 15.16 MPa (2198 psia) at the time of the scram in the simulation.

After scram the reactor system parameters were calculated to stabilize for both L6-8B tests within 30 s without any operator action.

Sensitivity calculations showed that a coincident change of ±10% in the values of Doppler and moderator feedback coefficients moves the time of scram by about ±10 s in Test L6-8B-1 and by about ±2 s in Test L6-8B-2. Environmental heat losses are important only in the modeling of the pressurizer whose behavior determines the primary system pressure response during the experiments. Sensitivity calculations indicated that no environmental heat losses and 20 kW heat losses from the pressurizer change the time of scram by about ∓9 s in Test L6-8B-1. There is no effect in Test L6-8B-2 because it is scrammed on high power.

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BEST ESTIMATE PREDICTION FOR LOFT ANTICIPATED TRANSIENT SLOW AND FAST ROD WITHDRAWAL EXPERIMENT L6-8B

1. INTRODUCTION

The report documents the experiment prediction (EP) analysis, using the RELAP5/MOD1 computer code,[1] of the thermal-hydraulic response of the Loss-of-Fluid Test (LOFT) facility during the planned nuclear anticipated transient Experiment L6-8B. This experiment, consisting of two control rod withdrawal tests, will simulate the range of expected reactivity insertions resulting from unintended control rod assembly withdrawal events in a commercial four-loop pressurized water reactor (PWR). 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 similar to a large commercial PWR.

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 L6-8B. Section 2 contains a description of the modeling techniques employed in the EP analyses. Sections 3 and 4 contain discussions of the calculated results for slow control rod withdrawal Experiment L6-8B-1 and fast control rod withdrawal Experiment L6-8B-2, respectively. Section 5 presents results of pertinent sensitivity calculations. Conclusions are presented in Section 6. Appendix A provides brief descriptions of Experiment L6-8B and of the LOFT facility. Plots showing the detailed results of the EP are included in Appendix B. A listing of the code input data and updates is provided 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 of the LOFT system during Experiment L6-8B. The point kinetics reactor physics model of RELAP5 was considered to be adequate to simulate the transients because the experiment planning calculations showed that changes in the core axial power shape during the transients were not significant.[2] The code version that was used included special updates (see Appendix C) to output reactivities caused by Doppler and moderator feedback effects as well as the scram reactivity (reactivity insertion) curves.

The nodalization used in RELAP5/MOD1 for this EP calculation is based on the standard nodalization of LOFT,[b] with changes where necessary to represent the LOFT system configuration for the L6-8B experiment. The nodalization is given in Figure 1. A complete input data listing is supplied in Appendix C.

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

1. The steam generator and pump simulator volumes in the broken loop hot leg were removed because they will be flanged off during the tests.

2. The ECCS system was removed from the model because it was not used in the experiments.

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a. This analysis was performed using RELAP5/MOD1 Cycle 15, a production version of the RELAP5/MOD1 code, with updates including improved reactor kinetics minor edits. The code version and updates are filed under Idaho National Engineering Laboratory Computer Code Configuration Management (CCCM) Archival Number F00341.

b. The standard LOFT input model Version 129 is filed under CCCM Archival Number F00763. The model is continually being updated and improved. However, complete traceability of each version is maintained in the model and by the LOFT Program Division.

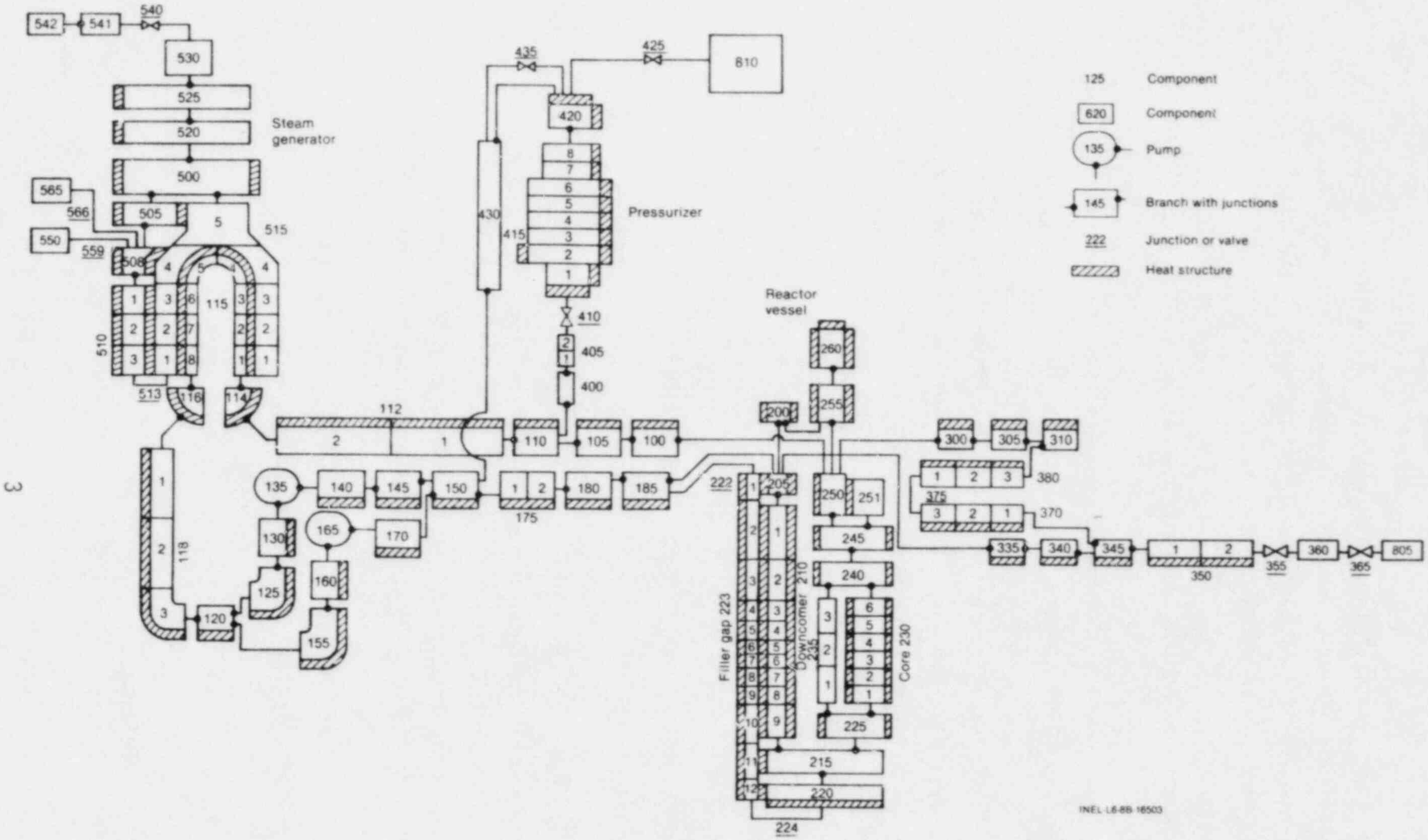

Figure 1. LOFT RELAP5 model schematic diagram for L6-8B experiments.

3. The detailed feedwater flow system was replaced with a simple model containing a time dependent volume and a junction. This was done because of computational difficulties with the detailed feedwater flow system model of the base deck.

Other necessary changes to the base input were:

1. The core axial power shape was given to reflect the experiment specified conditions (Figure 2).

2. The reactivity insertion caused by control rod withdrawals was simulated by time dependent scram curves which were constructed on the basis of control rod worth versus position (Figure 3) and the given rod withdrawal rates.

3. The pressurizer heaters and spray were disabled as specified in Reference 1.

4. Moderator reactivity coefficients were changed to values specifically calculated for the L6-8B tests.

5. The Doppler feedback table used for Test L6-8B-1 (rods at 1.312 m) was a standard table based on control rods at 1.372 m (54 in.) above the bottom of the core. For L6-8B-2 (rods at 1.016 m) this table was modified by multiplying the reactivity values by a factor of 0.86 to take into account the more skewed power profile. This factor was determined from reactor physics calculations. The Doppler feedback was weighted by the peaking factors of the core heat slabs for this case.

Because of different axial power shapes in Experiments L6-8B-1 and L6-8B-2 both tests required a separate initilization run to obtain initial conditions specified in Reference 1.

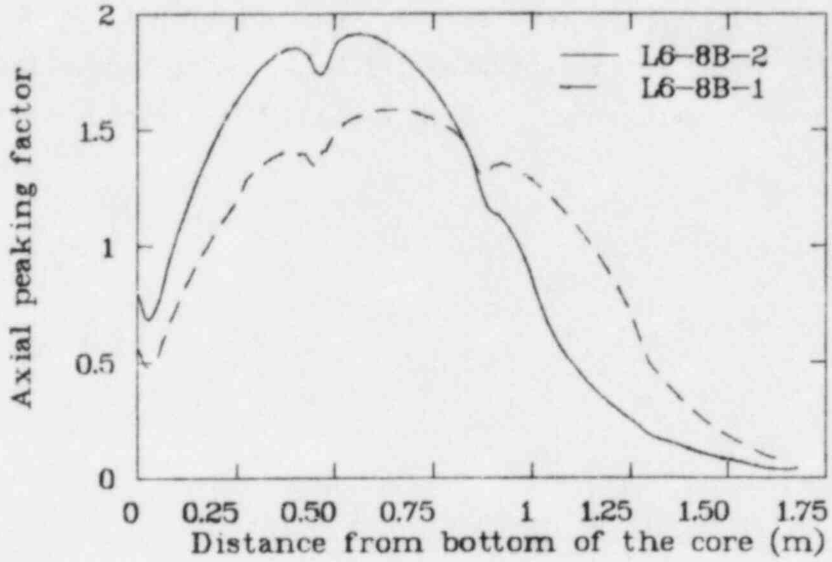

Figure 2. Core axial power distribution at the start of L6-8B experiments.

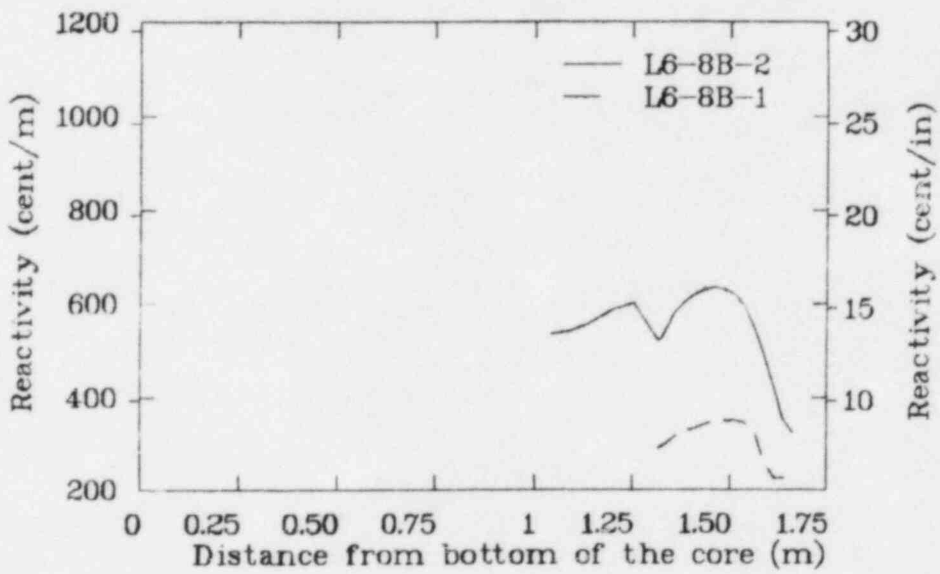

Figure 3. Control rod worth versus axial position in LOFT reactor.

Time, variable, and logic trips were modified for proper simulation of the experiment scenario. The reactor trip setpoint was either hot leg pressure 15.738 MPa (2282 psia) or reactor power 51.5 MW. After scram the secondary side feedwater flow was ramped to 0 in 2 s and the steam control valve started to close.

3. CALCULATIONAL RESULTS FOR SLOW CONTROL ROD WITHDRAWAL EXPERIMENT L6-8B-1

This section contains a general overview of the results of the Experiment L6-8B-1 simulation. In this experiment the control rods will be withdrawn from the core at a rate of 0.19 cm/s (4.5 in./min), which amounts to approximately 0.6 ¢/s reactivity insertion.

The reactivity insertion results in a reactor power increase (Figure 4). Because the steam flow control valve does not move to follow the core power increase, PCS fluid temperatures start to increase as the steam generator heat removal is less than the heat transferred from the fuel to the coolant (Figure 5). PCS heating results in coolant swelling which causes the pressurizer liquid level to rise as the PCS heats up (Figure 6). The rising liquid level compresses the steam space of the pressurizer and the pressure of the whole primary system increases (Figure 7). In the simulation, the high pressure scram setpoint (15.74 MPa) was reached at 106 s. The reactor power was calculated to increase from the initial 37.5 MW to 43.6 MW at the time of the scram.

The total reactivity calculated with reactor kinetics is shown in Figure 8. This curve is the sum of the inserted positive reactivity and the negative feedback effects of Doppler and moderator density. After 43 s moderator density feedback gives more negative reactivity than Doppler as seen in Figure 9.

After scram the reactor power drops to the decay heat level and PCS temperatures start to decrease rapidly (Figure 5). However, when the feedwater and steam control valves are both closed about 10 s after scram, PCS temperatures stabilize with a slowly increasing trend. Similar behavior is seen in the curves for pressurizer pressure and pressurizer liquid level in Figures 6 and 7.

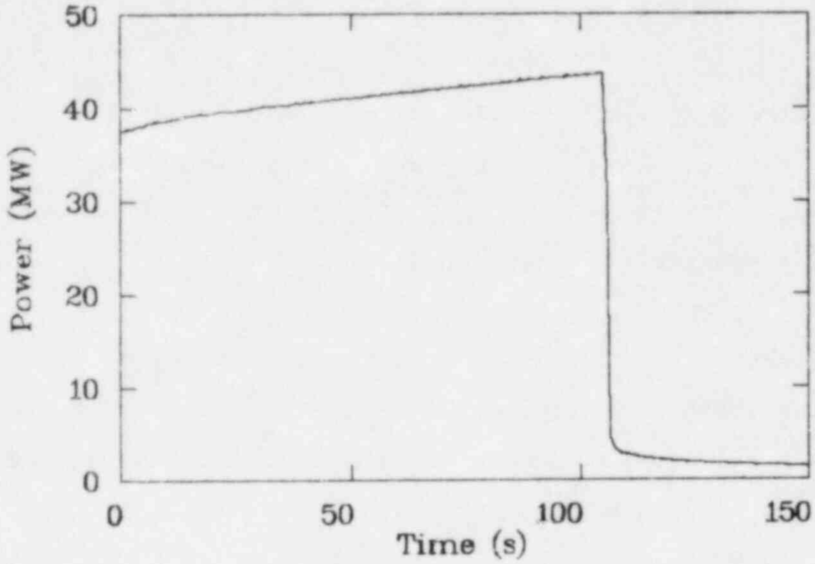

Figure 4. Predicted reactor power during Experiment L6-8B-1.

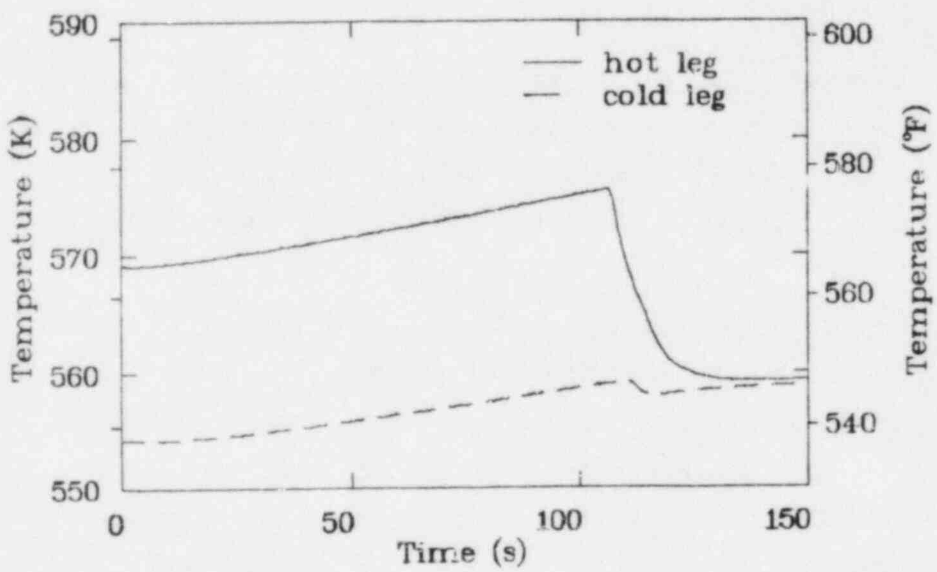

Figure 5. Predicted primary coolant hot and cold leg temperatures during Experiment L6-8B-1.

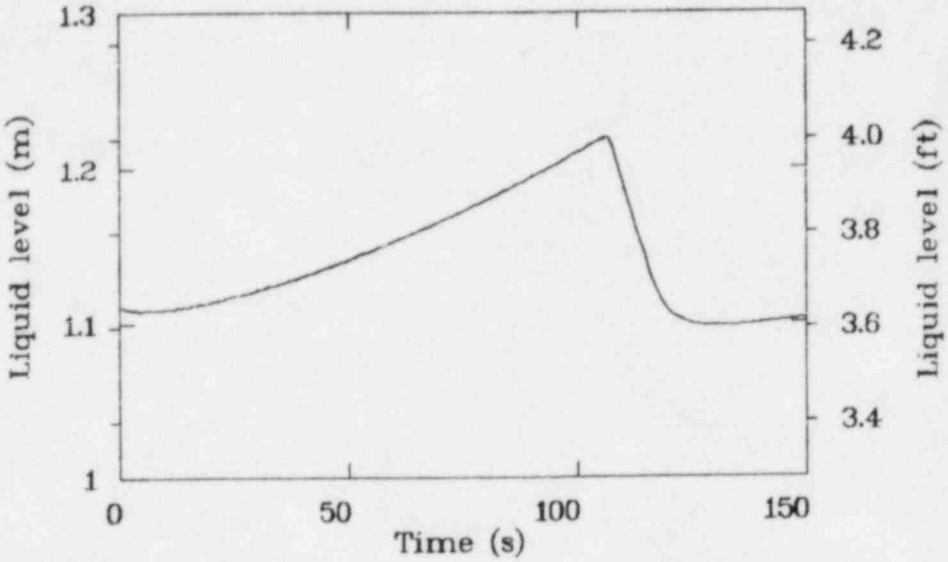

Figure 6. Predicted pressurizer collapsed liquid level during Experiment L6-8B-1.

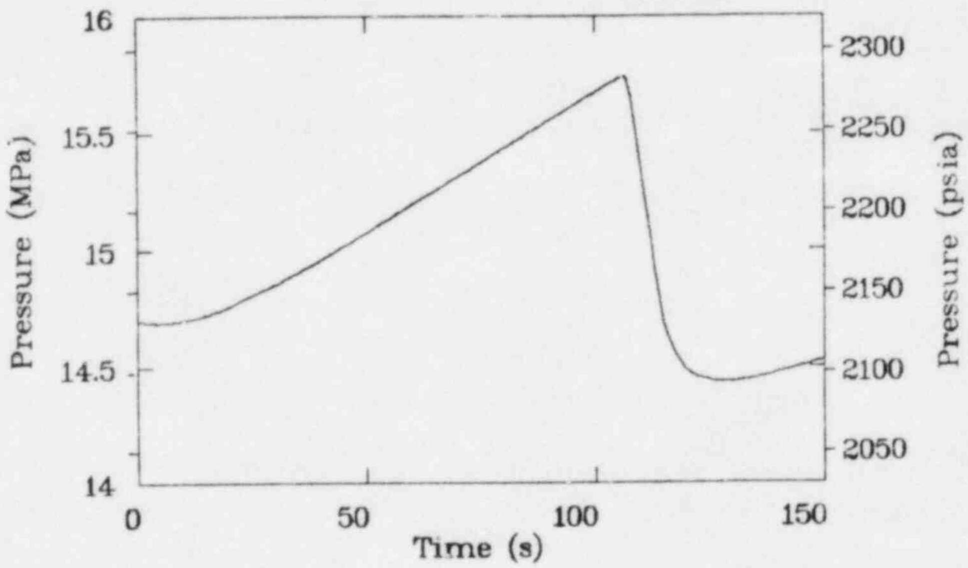

Figure 7. Predicted pressurizer pressure during Experiment L6-8B-1.

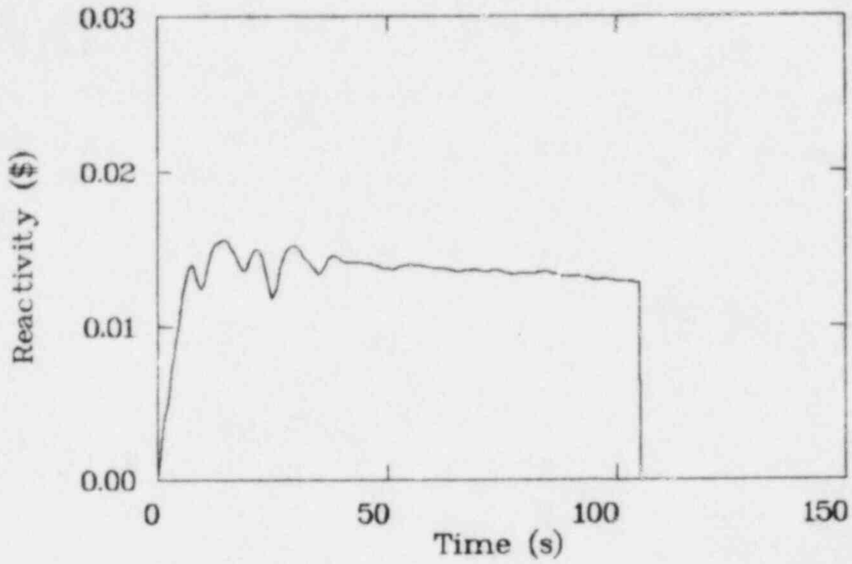

Figure 8. Predicted total reactivity during Experiment L6-8B-1.

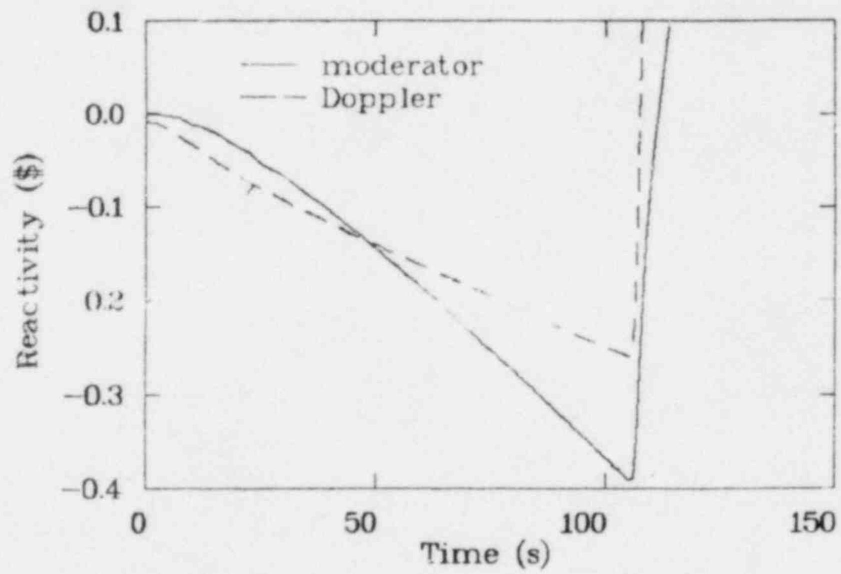

Figure 9. Predicted moderator density and Doppler feedback reactivities during Experiment L6-8B-1.

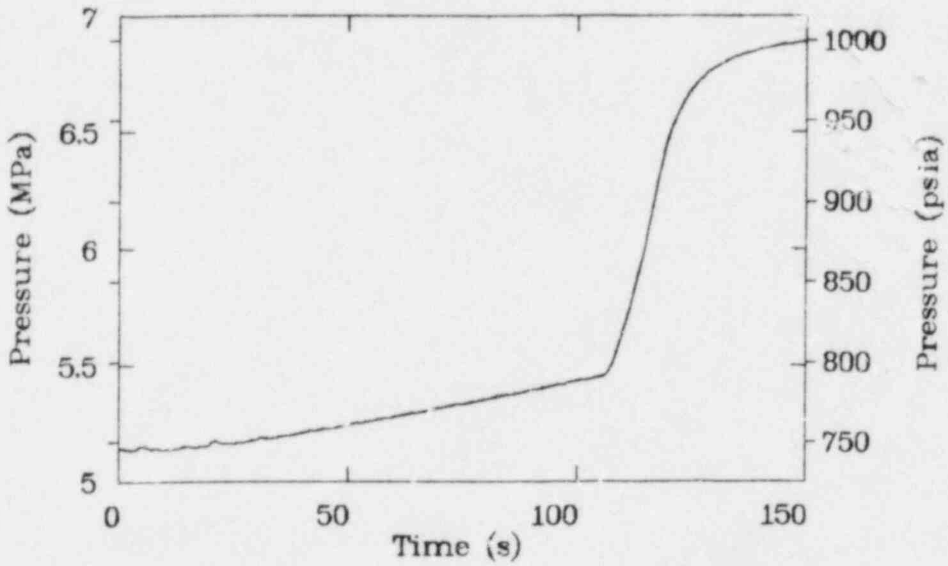

Figure 10. Predicted steam generator secondary side pressure during Experiment L6-8B-1.

The calculated steam generator secondary side pressure is shown in Figure 10. During the time of reactivity insertion, secondary side pressure is increasing as more heat is transferred from the PCS to the secondary. After scram the pressure increases due to closure of the main steam control valve.

4. CALCULATIONAL RESULTS FOR FAST CONTROL ROD WITHDRAWAL EXPERIMENT L6-8B-2

This section contains a general overview of the results of the Experiment L6-8B-2 simulation. In this experiment the control rods will be withdrawn from the core at a speed of 1.02 cm/s (24 in./min) which represents approximately 5.5 ¢/s reactivity insertion.

The calculational results of L6-8B-2 generally exhibit the same phenomena as observed in the L6-8B-1 simulation with a shortened time scale. Due to faster reactivity insertion the reactor power increases more rapidly and the high power scram setpoint is reached at 13.2 s (Figure 11). Because of a relatively large time constant for heat conduction in the fuel rods, PCS fluid temperature and pressure do not increase as much as in L6-8B-2 before the high power scram setpoint is reached (Figures 12 and 13). The PCS hot leg pressure is 15.16 MPa (2198 psia) at the time of scram. Initial reactor power was 37.5 MW and the reactor was scrammed when the calculated power reached 51.5 MW.

The calculated total reactivity is shown in Figure 14. The dominant negative reactive feedback effect is Doppler reactivity as can be seen in Figure 15. The predicted steam generator secondary side pressure and pressurizer liquid level are shown in Figures 16 and 17.

After scram the reactor system is recovered in the same way as in the simulation of Experiment L6-8B-1.

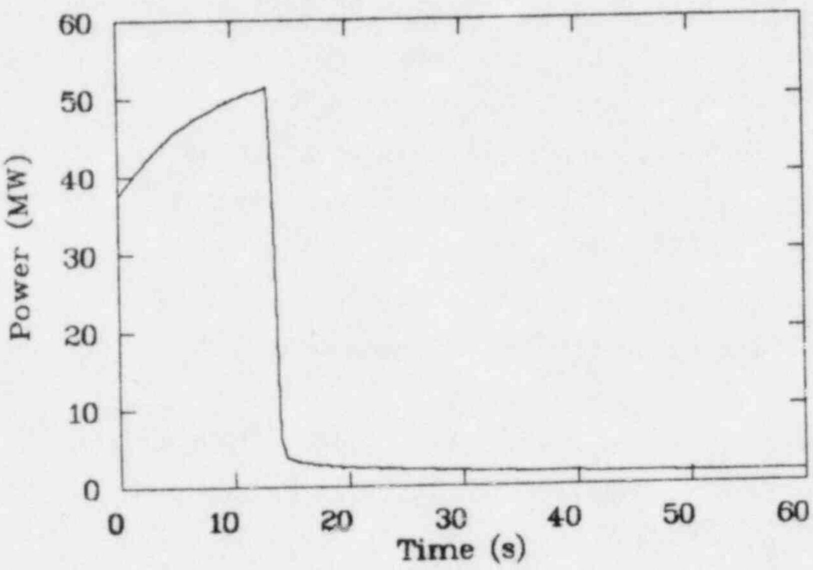

Figure 11. Predicted reactor power during Experiment L6-8B-2.

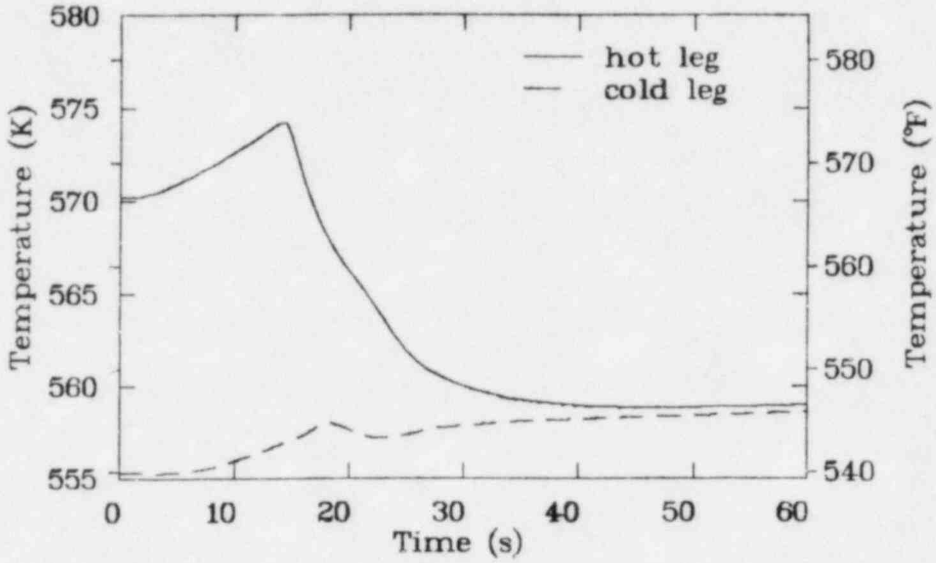

Figure 12. Predicted primary coolant hot and cold leg temperatures during Experiment L6-8B2.

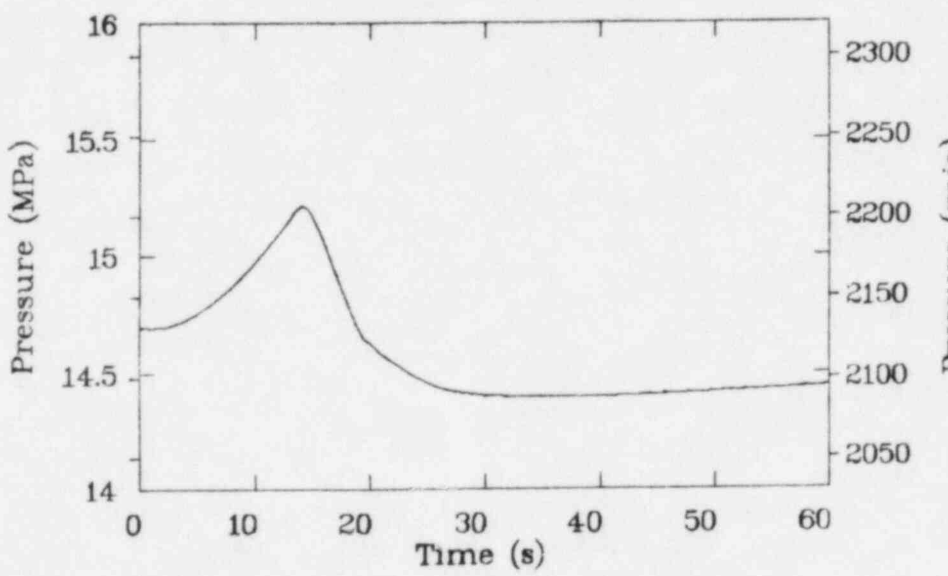

Figure 13. Predicted pressurizer pressure during Experiment L6-8B-2.

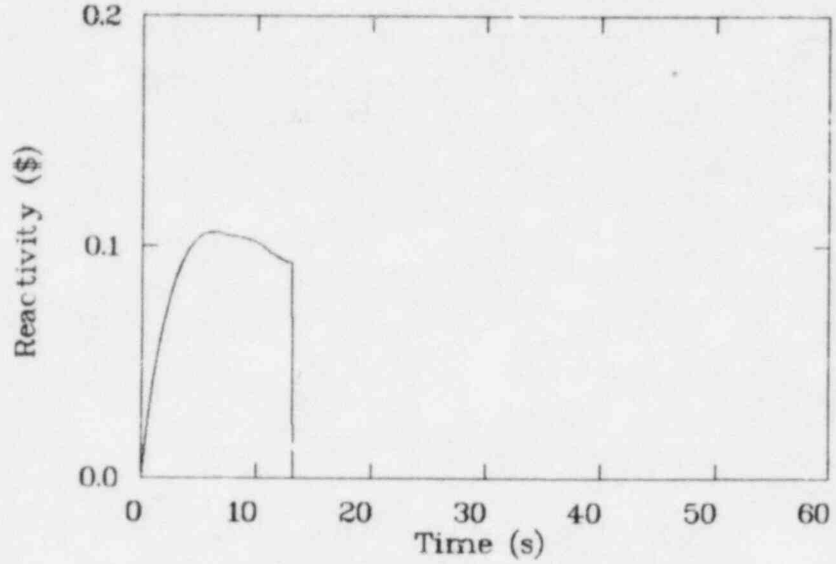

Figure 14. Predicted total reactivity during Experiment L6-8B-2.

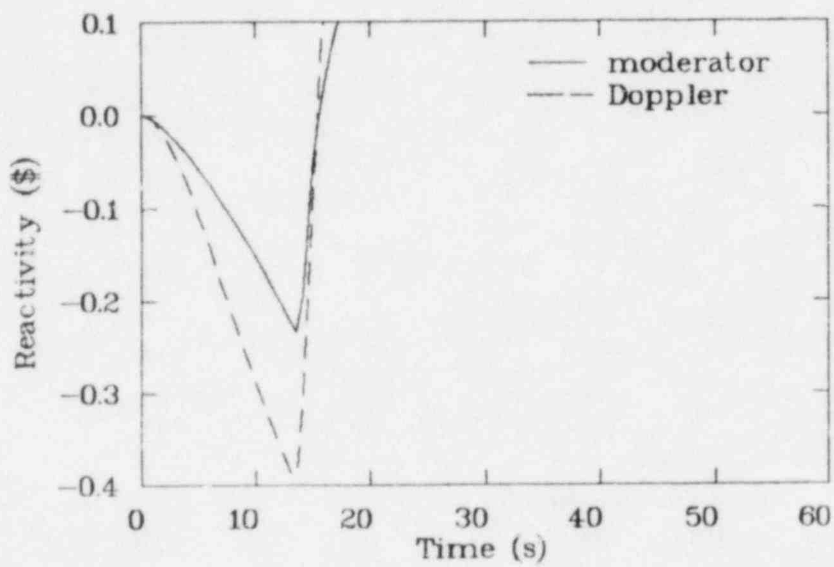

Figure 15. Predicted moderator density and Doppler feedback reactivities during Experiment L6-8B-2.

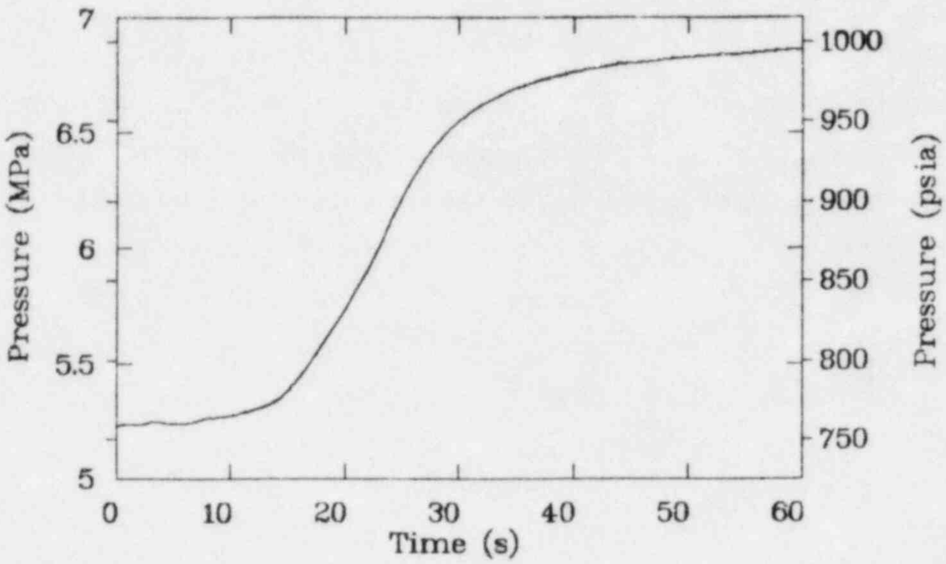

Figure 16. Predicted steam generator secondary side pressure during Experiment L6-8B-2.

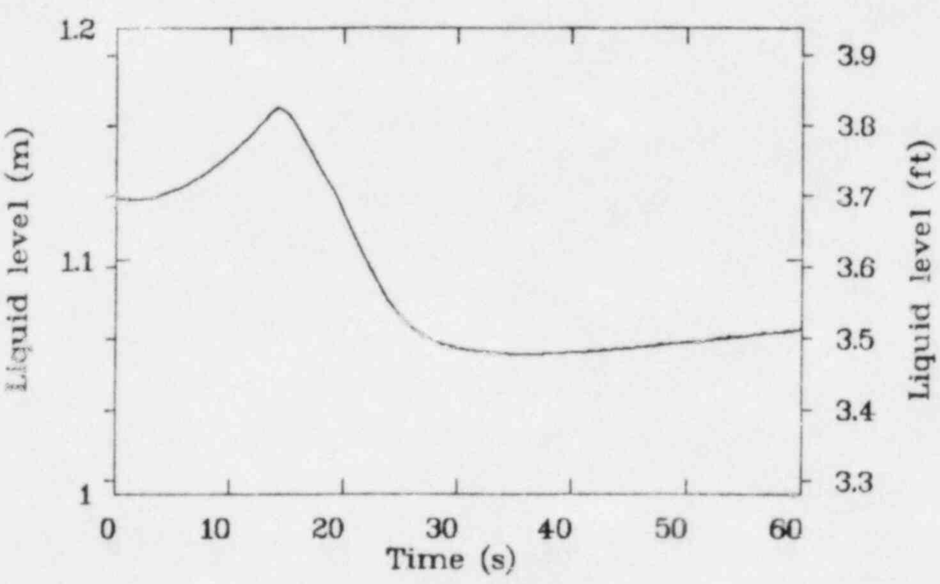

Figure 17. Predicted pressurizer liquid level during Experiment L6-8B-2.

5. SENSITIVITY CALCULATIONS

Sensitivity calculations were performed to investigate the effects of uncertainties in reactivity feedback coefficients and environmental heat losses. The largest uncertainty in the experiment predictions was estimated to be associated with the values of reactivity feedback coefficients. The effect of environmental heat losses were studied because of their relatively large uncertainty and to clarify their importance in the modeling of reactivity insertion transients.

5.1 Reactivity Feedback Coefficients

The uncertainty associated with the values of reactivity feedback coefficients was estimated to be ±10%. In the rod withdrawal transients both the Doppler feedback and moderator feedback effects give negative reactivity. Thus to maximize the effect of uncertainty in the values of these coefficients both coefficients have to be changed in the same direction. Two sensitivity calculations were performed for both L6-8B-1 and L6-8B-2 experiments. In the one calculation the Doppler and moderator coefficients were increased by 10%, in the other the coefficients were decreased by 10%.

The calculational results obtained in sensitivity calculations are summarized in Table 1. When feedback coefficients are increased, more negative reactivity is supplied and the reactor power decreases. Reduced reactor power results in a slower PCS heatup and a slower pressure increase. Thus in the slow rod withdrawal case, where the reactor is scrammed on high pressure, scram time is later. The opposite occurs when the feedback coefficients are decreased. A more positive reactivity results in higher reactor power, faster primary system pressure rise, and earlier scram time.

In the fast rod withdrawal case the change of the reactor power caused by the change in reactivity is the source of different scram time. When coefficients are increased, scram time is later, whereas scram is earlier in the calculations with decreased coefficients. In sensitivity calculations for both slow and fast rod withdrawals the changes in total inserted reactivity were less than 10% compared to the base case values.

TABLE 1. SUMMARY OF L6-8B SENSITIVITY CALCULATIONS

Case	Effect on Scram Time (s) L6-8B-1	Effect on Scram Time (s) L6-8B-2
Doppler and moderator feedback coefficients +10%	+10	+1.5
Doppler and moderator feedback coefficients -10%	-9	-2
No environmental heat losses	-8	0
20 kW heat losses from the pressurizer	+9	0

5.2 Environmental Heat Losses

The environmental heat loss distribution at the start of the base case L6-8B predictions was: 174 kW heat losses from PCS (except the pressurizer), 6 kW heat losses from the pressurizer, and 10 kW heat losses from the secondary side of the steam generator. This is considered to be the best estimate heat loss distribution of the LOFT reactor system at this time.

To study the effect of heat loss modeling on the results of the L6-8B simulations two other bounding cases were calculated. One calculation was with no environmental heat losses, the other with the heat loss distribution: 161 kW from PCS, 20 kW from the pressurizer and 59 kW from the steam generator secondary side. Sensitivity calculations were performed for both the L6-8B-1 and L6-8B-2 transients.

The only major parameter of the LOFT reactor system which changed significantly in the calculated sensitivity cases was the primary system pressure. This indicates the importance of the pressurizer heat loss modeling in the analysis because the primary system pressure is determined by the pressurizer behavior. Changing heat losses in other parts of the flow system had only a minor effect on the overall system behavior because the reactor power and reactivity did not change compared to the base cases.

In L6-8B-1, where the reactor is scrammed on a high pressure setpoint, scram times changed when the rate of pressure increase was different depending on the magnitude of the pressurizer environmental heat losses. The results of the heat loss sensitivity study are summarized in Table 1.

6. CONCLUSIONS

The RELAP5 calculations performed for Experiment L6-8B appear consistent with the phenomena expected to occur. The slow rod withdrawal, Test L6-8B-1, will scram on high PCS pressure and the fast rod withdrawal, Test L6-8B-2, will scram on high power as desired. In the L6-8B-2 experiment the dominant negative reactivity feedback effect is Doppler. In L6-8B-1 the moderator density feedback effect is more important after 43 s in the experiment. After scram the reactor system conditions are predicted to stabilize without any operator action.

Sensitivity calculations showed that ±10 percent uncertainty in Doppler feedback and moderator density feedback coefficients results in about ±10 s difference in the scram time for L6-8B-1 experiment and about ±2 s difference for L6-8B-2 experiment, respectively. For these transients modeling of the environmental heat losses is important only for the pressurizer, whose behavior determines the PCS pressure.

7. REFERENCES

1. D. J. Varacalle, Jr., LOFT Experimental Definition Document L6-8 Anticipated Transient Test Series, EGG-LOFT-5734, 1982.

2. V. H. Ransom et al., RELAP5/MOD1 Code Manual EGG-2070, November 1980.

APPENDIX A
EXPERIMENT L6-8B AND LOFT FACILITY DESCRIPTIONS

APPENDIX A

EXPERIMENT L6-8B AND LOFT FACILITY DESCRIPTIONS

Experiment L6-8B is an anticipated transient experiment simulating uncontrolled control rod withdrawal events in a large pressurized water reactor. Control rod withdrawal results in a reactivity insertion and an increase in reactor power. Depending on the speed of the reactivity insertion the reactor will be scrammed due to high power or due to high pressure.

Experiment L6-8B will involve the withdrawal of all four control rod assemblies of the LOFT reactor. Two tests will be conducted. Test L6-8B-1 will utilize an average reactivity insertion of 0.6 ¢/s which will be achieved with a control rod withdrawal rate of 0.19 cm/s (45 in./min). In Test L6-8B-1 the reactor is expectd to be scrammed due to reaching the high pressure setpoint which will be set to 15.738 MPa (2282 psia) for these experiments. Test L6-8B-2 will utilize an average reactivity insertion of 5.5 ¢/s corresponding a withdrawal rate of 1.02 cm/s (24 in./min). In Test L6-8B-2 the reactor is expected to be scrammed due to reaching the high power setpoint which will be set to 51.5 MW.

The tests will be initiated from 37.5 MW reactor power. The primary coolant pumps will remain on during the experiments. Pressurizer heaters and spray will be disabled. Control rod initial postions are at 1.312 m (51.66 in.) above the bottom of the core in Experiment L6-8B-1 and at 1.016 m (40 in.) in Experiment L6-8B-2.

1. EXPERIMENT OBJECTIVES

The L6 test series was developed to study anticipated transients. Data from the tests will be utilized in evaluating the computer codes and analytical techniques used to predict anticipated transients.

The L6-8B test specific objectives are:

1. Obtain plant response data from a transient caused by the withdrawal of all control rod assemblies

2. Provide data on the integral system response and reactor kinetics required to evaluate code capabilities

3. Provide data to evaluate plant automatic recovery methods

2. LOFT FACILITY DESCRIPTION

The LOFT facility is described in detail in Reference A-1. The LOFT instrumentation and major components are shown in Figures A-1 through A-5. The instrumentation nomenclature is explained in Table A-1.

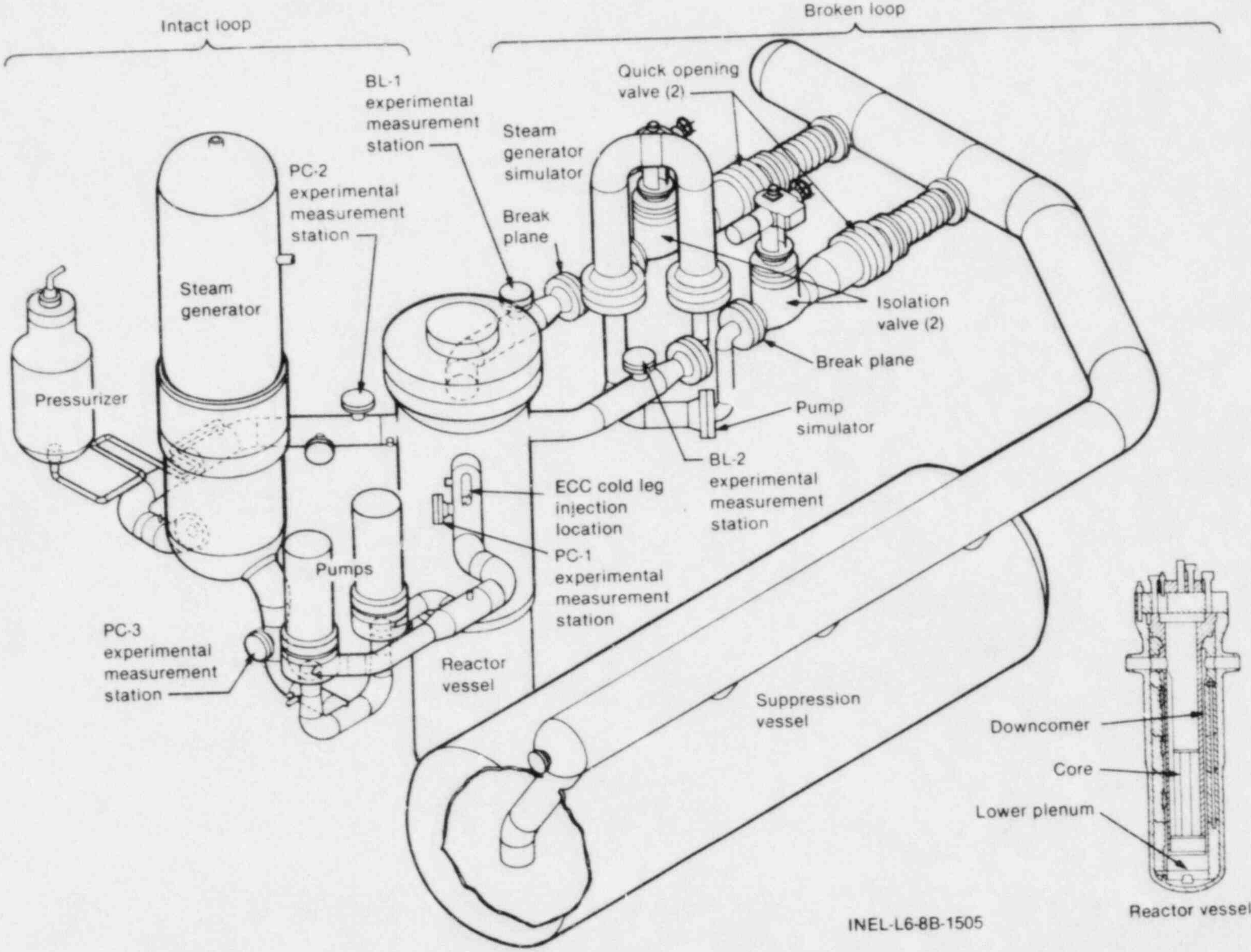

Figure A-1. Axonometric projection of LOFT system.

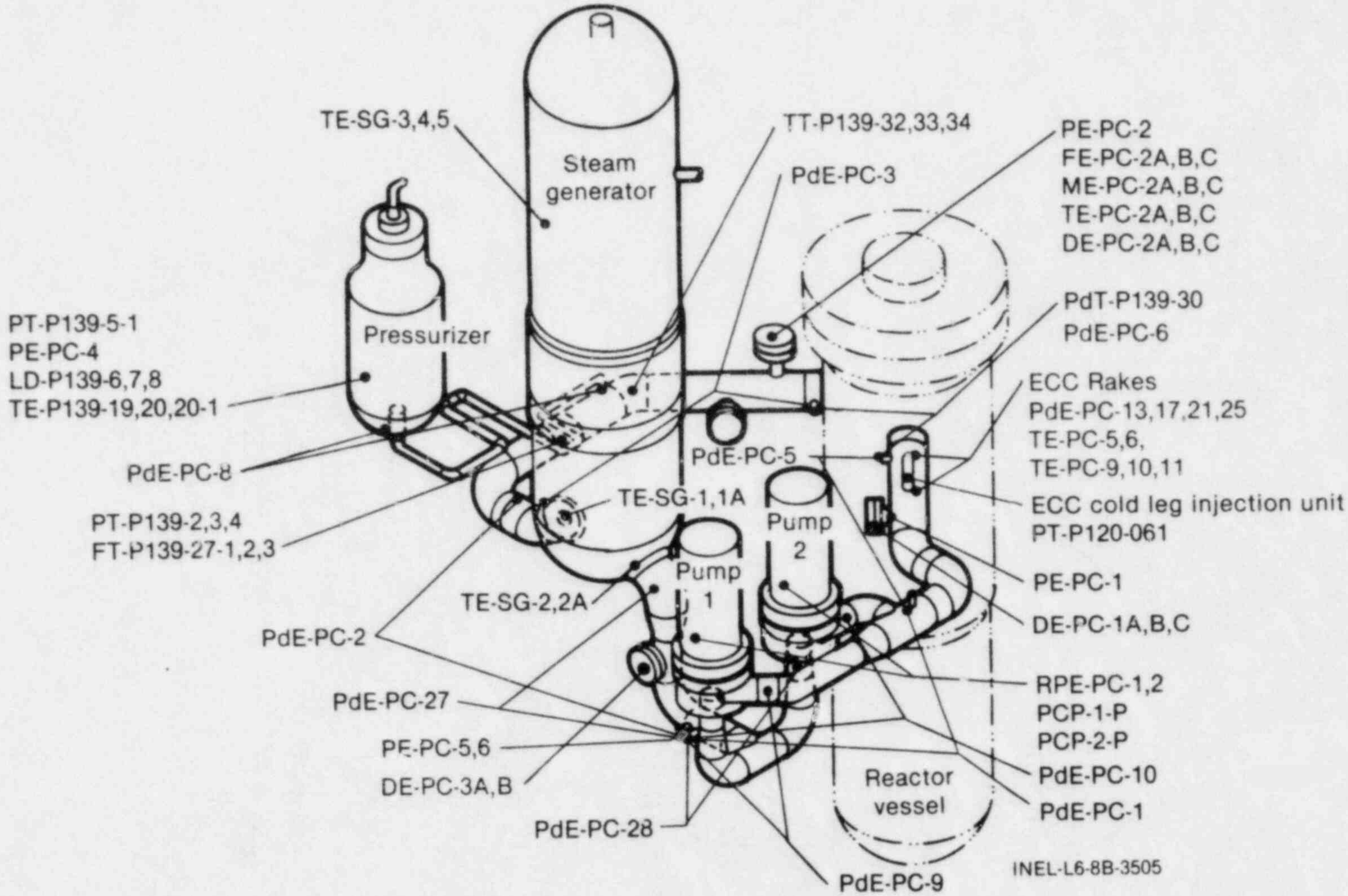

Figure A-2. LOFT intact loop thermo-fluid instrumentation.

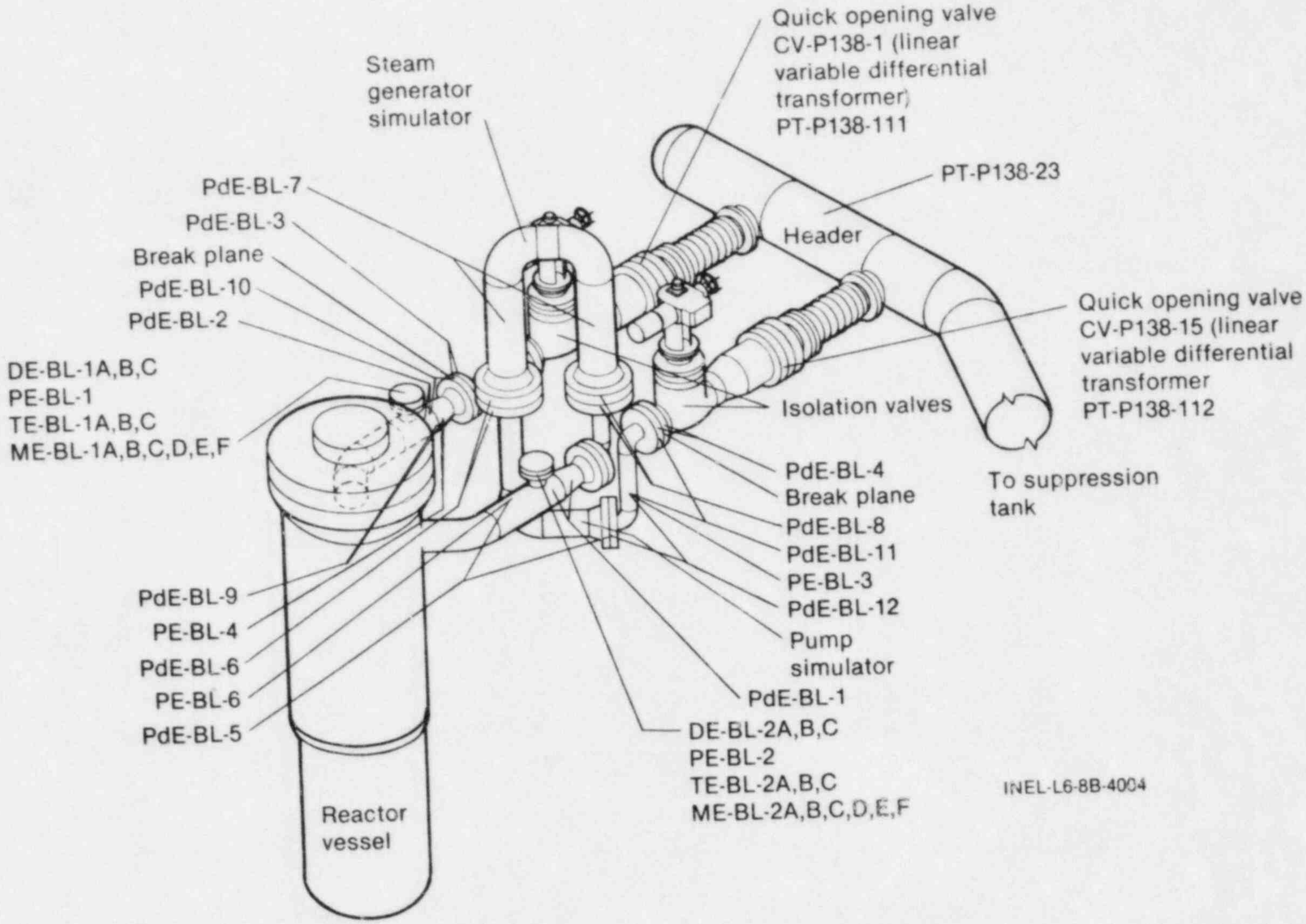

Figure A-3. LOFT broken loop thermo-fluid instrumentation.

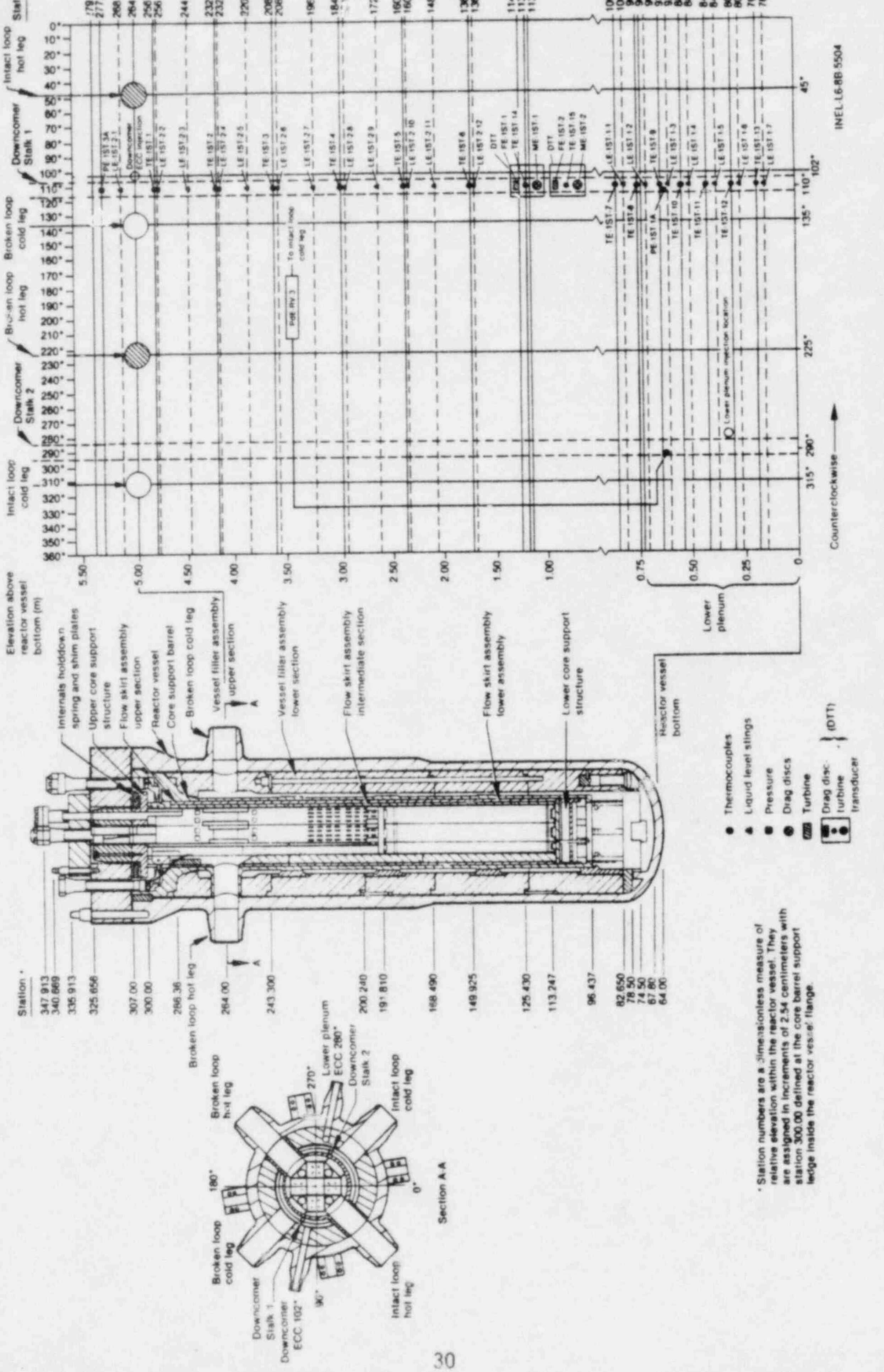

Figure A-4. LOFT reactor vessel instrumentation.

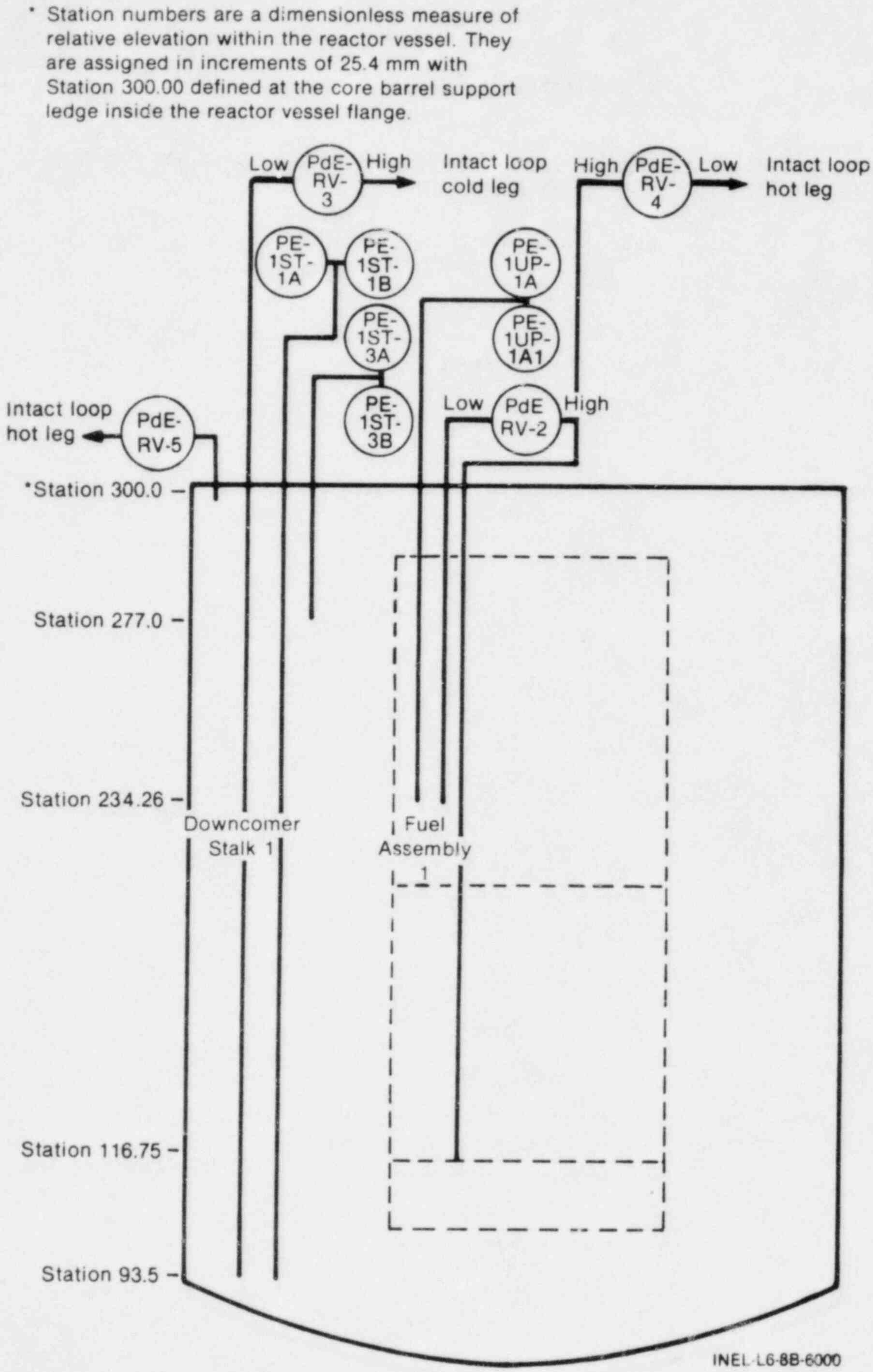

Figure A-5. LOFT reactor vessel pressure and differential pressure instrumentation.

3. REFERENCE

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

TABLE A-1. NOMENCLATURE FOR LOFT INSTRUMENTATION[a]

The designations for the different types of transducers are:

Transducers		Designation
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]

Systems		
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.

APPENDIX B
DETAILED TEST PREDICTION DATA FOR EXPERIMENTS L6-8B-1 AND L6-8B-2

APPENDIX B

DETAILED TEST PREDICTION DATA FOR EXPERIMENTS L6-8B-1 AND L6-8B-2

Detailed test prediction data for Experiment L6-8B are provided on two microfiche on the inside of the report back cover. The titles of the microfiche are "L6-8B EPD, Appendix B, L6-8B-1 predictions" and "L6-8B EPD, Appendix B, L6-8B-2 predictions." The figures on the microfiche 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 (same figure numbers for both tests):

Figure B-1. Reactor power

Figure B-2. Total reactivity feedback

Figure B-3. Doppler reactivity feedback

Figure B-4. Moderator density feedback

Figure B-5. Mass flow rate--intact loop hot leg

Figure B-6. Mass flow rate--steam line

Figure B-7. Mass flow rate--feed system

Figure B-8. Collapsed liquid level--pressurizer

Figure B-9. Collapsed liquid level--steam generator secondary

Figure B-10. Pressure--intact loop hot leg

Figure B-11. Pressure--pressurizer

Figure B-12. Pressure--steam generator steam dome

Figure B-13. Fluid temperature--pressurizer liquid

Figure B-14. Fluid temperature--pressurizer vapor

Figure B-15. Fuel centerline temperature--elevation from 0.558 m to 0.8382 m above the bottom of the core

Figure B-16. Fuel centerline temperature--elevation from 0.8382 m to 1.1176 m above the bottom of the core

APPENDIX C
INPUT DATA AND TIME ZERO EDITS FOR L6-8B-1 AND L6-8B-2 PREDICTIONS

APPENDIX C

INPUT DATA AND TIME ZERO EDITS FOR L6-8B-1 AND L6-8B-2 PREDICTIONS

The RELAP5 input data listing for the L6-8B experiment predictions and the RELAP5 time zero edits are on microfiche in the pouch on the inside of the report back cover. Also included on the microfiche is a complete listing of updates to Cycle 15 of RELAP5, which were used for this analysis. The titles of the microfiche are "L6-8B Experiment Prediction Report, Appendix C-1," containing input data and zero time edit for L6-8B-1 and "L6-8B Experiment Prediction Report, Appendix C-2," containing input data and zero time edit for L6-8B-2 as well as the code update listing.