EGG-LOFT-5982
Project No. P 394
August 1982

BEST ESTIMATE PREDICTION FOR LOFT EXPERIMENT SERIES L6-8C

K. G. Condie

Idaho National Engineering Laboratory
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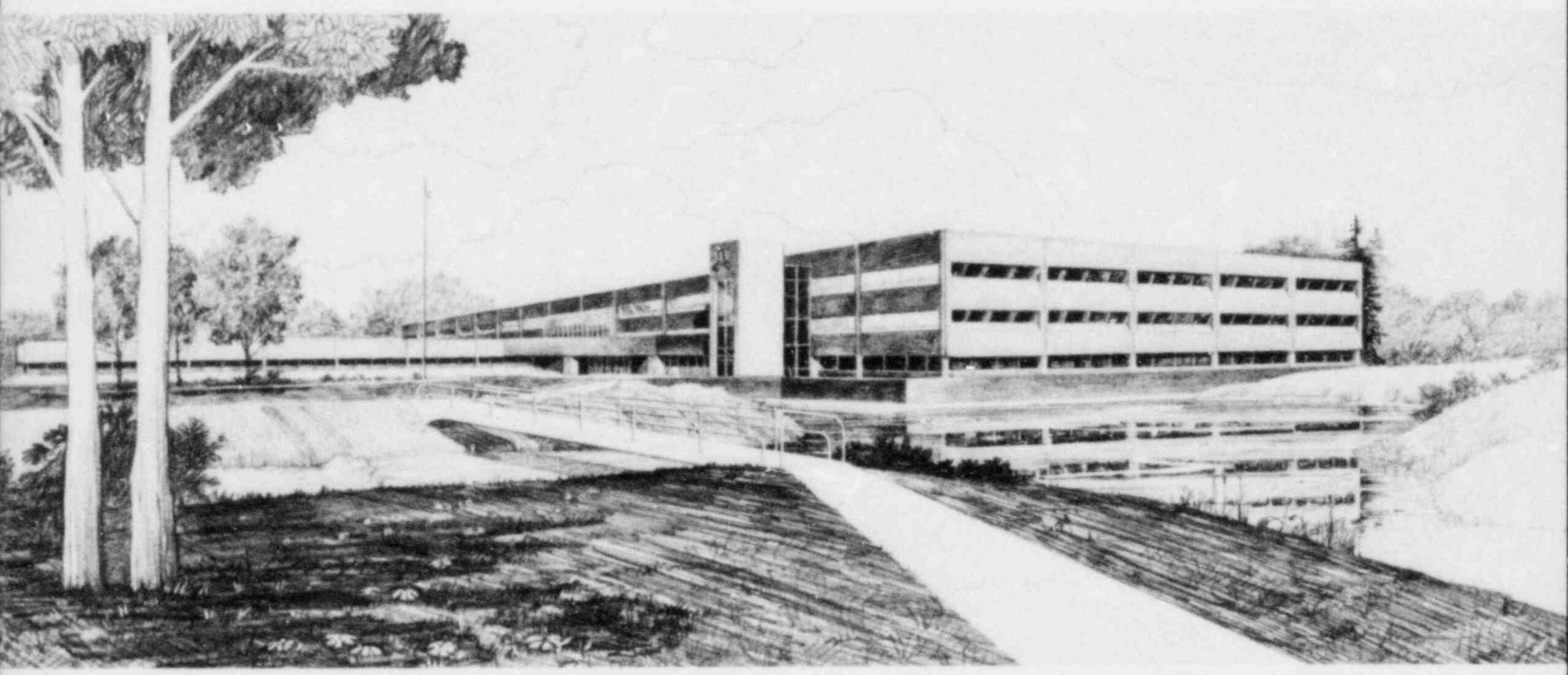

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Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6048

EG&G Idaho

EG&G Idaho, Inc.

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

INTERIM REPORT

Accession No. ________

Report No. EGG-LOFT-5982

Contract Program or Project Title:

LOFT Program Division

Subject of this Document:

BEST ESTIMATE PREDICTION FOR LOFT EXPERIMENT SERIES L6-8C

Type of Document:

Experiment Prediction Analysis Report

Author(s):

K. G. Condie

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.

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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 L6-8C AND FINDS THEM TO BE CONSISTENT WITH ACCEPTED GUIDELINES.

VH Ransom

CODE DEVELOPMENT PROGRAM

TRCharlton/CAAD

CODE ASSESSMENT & APPLICATIONS PROGRAM

JL Perryman/EAD

WATER REACTOR RESEARCH TEST FACILITY PROGRAM

Richard H Smith

THERMAL FUELS BEHAVIOR PROGRAM

ABSTRACT

The preexperiment calculations of the three recovery modes from small break simulations in the LOFT pressurized water reactor (PWR) are documented. Experiments L6-8C-1 and L6-8C-2 simulate a single steam generator tube rupture at nominal PWR pressure and temperature and are designed to investigate recovery procedures of (a) primary coolant pumps "on", with pressure controlled using pressurizer spray, and (b) primary coolant pumps "off", with pressure controlled by venting through the pressurizer power operated relief valve (PORV). Experiment L6-8C-3 simulates the same size break, but to the containment, and is to investigate a proposed pump trip criteria based on sensitivity of reactor coolant pump power to primary system void. The calculations, completed with the RELAP5/MOD1 system code, show that pressure is controlled and pressurizer level recovered with all three recovery modes. Primary and secondary pressure equilibrium is calculated to be achieved within 15 min after the break is initiated. Use of PORV venting is calculated to provide the shortest time to pressure equilibrium and pressurizer level recovery under conditions of constant cooldown rate imposed on the system.

NRC FIN No. A6048--LOFT Experimental Program

SUMMARY

The RELAP5/MOD1 computer code (a transient, one-dimensional, two-fluid system analysis program) was used to simulate the LOFT facility response for the three experiments in Experiment Series L6-8C. Series L6-8C will investigate various recovery procedures from small breaks to the secondary system or to containment.

Experiment L6-8C-1 will combine steam generator operation with pressurizer spray actuation to simulate the mitigation of coolant flow from the primary system to the secondary system without challenging the pressurizer power operated relief valve (PORV). Experiment L6-8C-2 will utilize a procedure similar to that now in use in operating plants in which the pumps will be tripped and the PORV will be utilized to reduce primary system pressure.

The calculations show that both procedures are effective in reducing primary system pressure below secondary pressure thus, mitigating flow from the primary to the secondary system. Pressurizer level is recovered rapidly in both experiments with initiation of scaled high pressure injection flow.

Experiment L6-8C-3 will demonstrate the use of reactor coolant pump current to determine reactor coolant system inventory during a small break. This experiment will have the same size break as C1 and C2 but will simulate a break to containment. The system inventory will be allowed to decrease until a 15% void is established at the pump inlet, at which time, the pumps will be tripped. Approximately 240 s is required to recover primary system inventory and establish pressurizer level after the pump trip and initiation of scaled high pressure injection flow.

CONTENTS

FIGURES

TABLE

BEST ESTIMATE PREDICTION FOR LOFT EXPERIMENT SERIES L6-8C

1. INTRODUCTION

This report contains results for the experiment prediction (EP) analysis performed using the RELAP5/MOD1[1] computer code to simulate the coupled system thermal-hydraulic responses of the Loss-of-Fluid Test (LOFT) Experiment Series L6-8C.[2]

The analysis was performed for the following reasons:

1. To provide experiment planning to assure achieving the experiment objectives

2. To provide operators and technical support personnel event scenarios to assist in conduct of the experiments

3. To provide a best estimate preexperiment calculation to evaluate the RELAP5 code and the LOFT-RELAP5 input model

4. To provide break mass flowrates which will be imposed as boundary conditions on the plant during the course of each of the three experiments.

The LOFT facility, described in Appendix A, is a 50-MW(t) pressurized water 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.

Initial conditions and plant component operation are summarized in Table 1.

TABLE 1. INITIAL CONDITIONS AND COMPONENT OPERATION

	L6-8C-1	L6-8C-2	L6-8C-3
Initial PCS pressure	15.52 MPa 2250 psia	15.52 MPa 2250 psia	15.52 Mpa 2250 psia
Initial PCS temperature	559 K 547°F	559 K 547°F	559 K 547°F
Power (decay only)	275 kW	275 kW	175 kW
Initial pressurizer level	0.508 m 20 in.	0.508 m 20 in.	0.508 m 20 in.
HPIS	One pump full 60 s after pzr level lost-throttled to maintain level after recovery	Same as C-1	One pump full at pump trip throttled to maintain level after pzr level recovery
Break	Area scaled to double offset shear of one tube in large plant steam generator	Same as C-1	Same area as C-1 but break is to containment
Pressurizer sprays	Manually initiated 60 s after pzr level is lost	Not available (pumps off)	Not used
Pumps	On during entire transient	Tripped 60 s after pzr level is lost	Tripped when inlet void = 0.15 restarted when pzr level recovered and subcooling obtained

TABLE 1. (continued)

	L6-8C-1	L6-8C-2	L6-8C-3
Pressurizer level	Will be lost then recovered with HPIS and pressurizer spray	Will be lost then re-covered with HPIS and PORV	Will be lost, then re-covered with HPIS
PORV operation	Not used in this test	Opened 60 s after pzr level lost	Not used in this test
Steam generator operation	Bleed and feed to maintain 55.6 K (100°F) per hour cool-down	Same as C-1	Isolated (simulated turbine trip)

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 each of the three experiments in Experiment Series L6-8C. Section 2 contains a description of the modeling techniques employed in the EP analysis. Section 3 contains discussions of the calculated results for the best estimate calculations. Section 4 presents results of sensitivity calculations made for the L6-8C-3 experiment. Conclusions are given in Section 5. Appendix A presents the experiment objectives and a brief description of each experiment and the LOFT facility. A listing of the code input data for each experiment and a listing of the code updates is provided in Appendix B and is 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 Series L6-8C. The RELAP5 code is a one-dimensional, two-fluid, thermal nonequilibrium reactor transient analysis program. This section describes the specific application for the L6-8C series experiments.

The nodalization used in RELAP5/MOD1 for these EP calculations is based on a standard LOFT nodalization,[b] with changes where necessary, to represent the particular system configuration for Experiment Series L6-8C. The nodalization for L6-8C-1 and L6-8C-2 is shown in Figure 1 and for L6-8C-3 in Figure 2. They are identical except for the secondary side of the steam generator. A complete input data listing for each experiment is supplied in Appendix B.

The following changes to the standard nodalizations were made for this analysis.

1. The simulated steam generator tube rupture flowrate will be achieved in these experiments by controlling the flow from the LOFT system purification line which is connected to the reflood assist bypass lines. The LOFT purification system line is not modeled in this analysis but a break area, scaled to represent

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a. This analysis was performed using RELAP5/MOD1 Cycle 15, a production version of the RELAP5/MOD1 code which is filed under Idaho National Engineering Laboratory Computer Code Configuration Management (CCCM) Archival Number F00835.

b. The standard LOFT input Model Version 130 was used as the basis for the L6-8C series input deck. The model is continually being updated and improved. Complete traceability of each version is maintained in the model and by the LOFT Program Division.

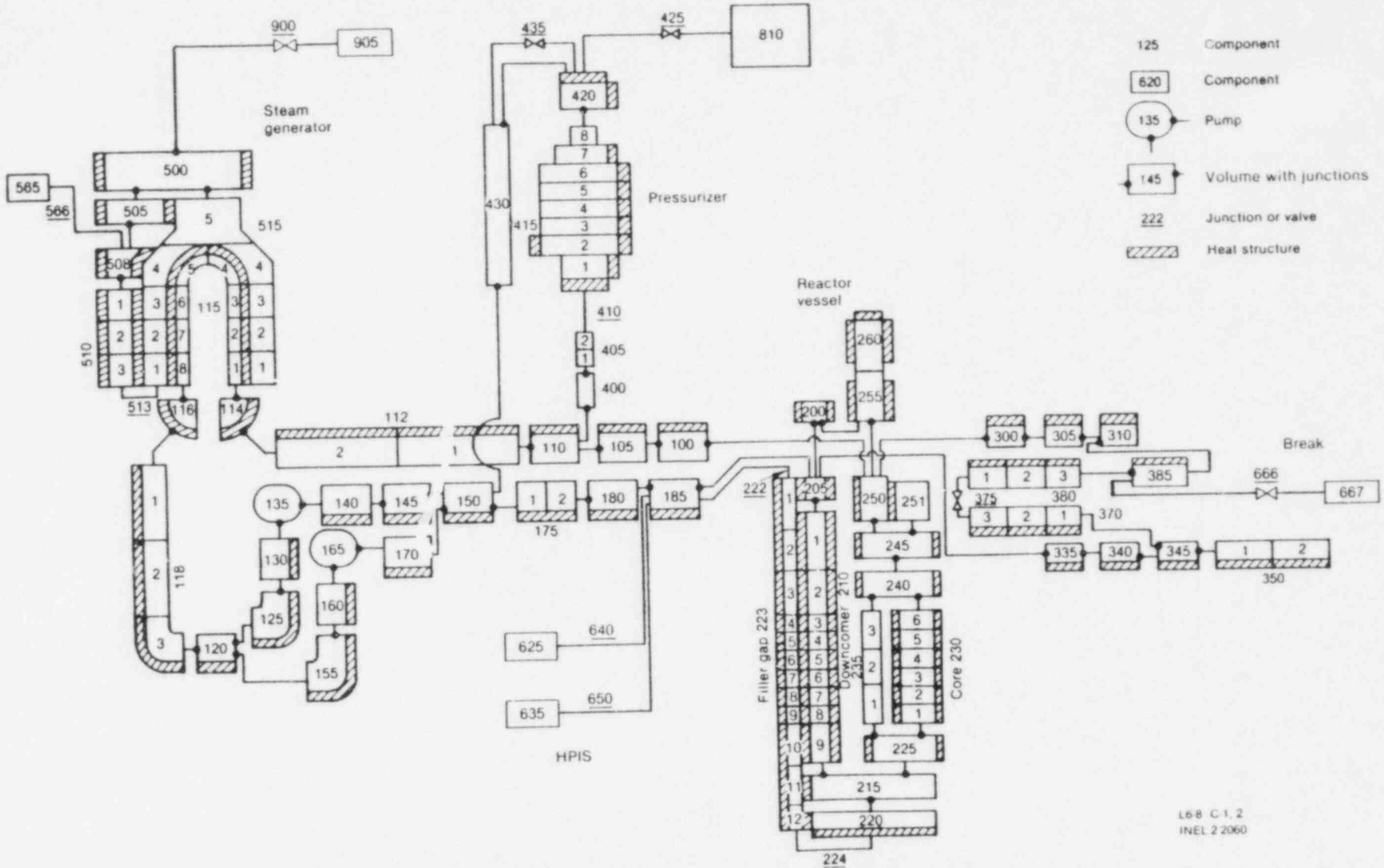

Figure 1. RELAP5 Nodalization for Experiments L6-8C-1 and L6-8C-2

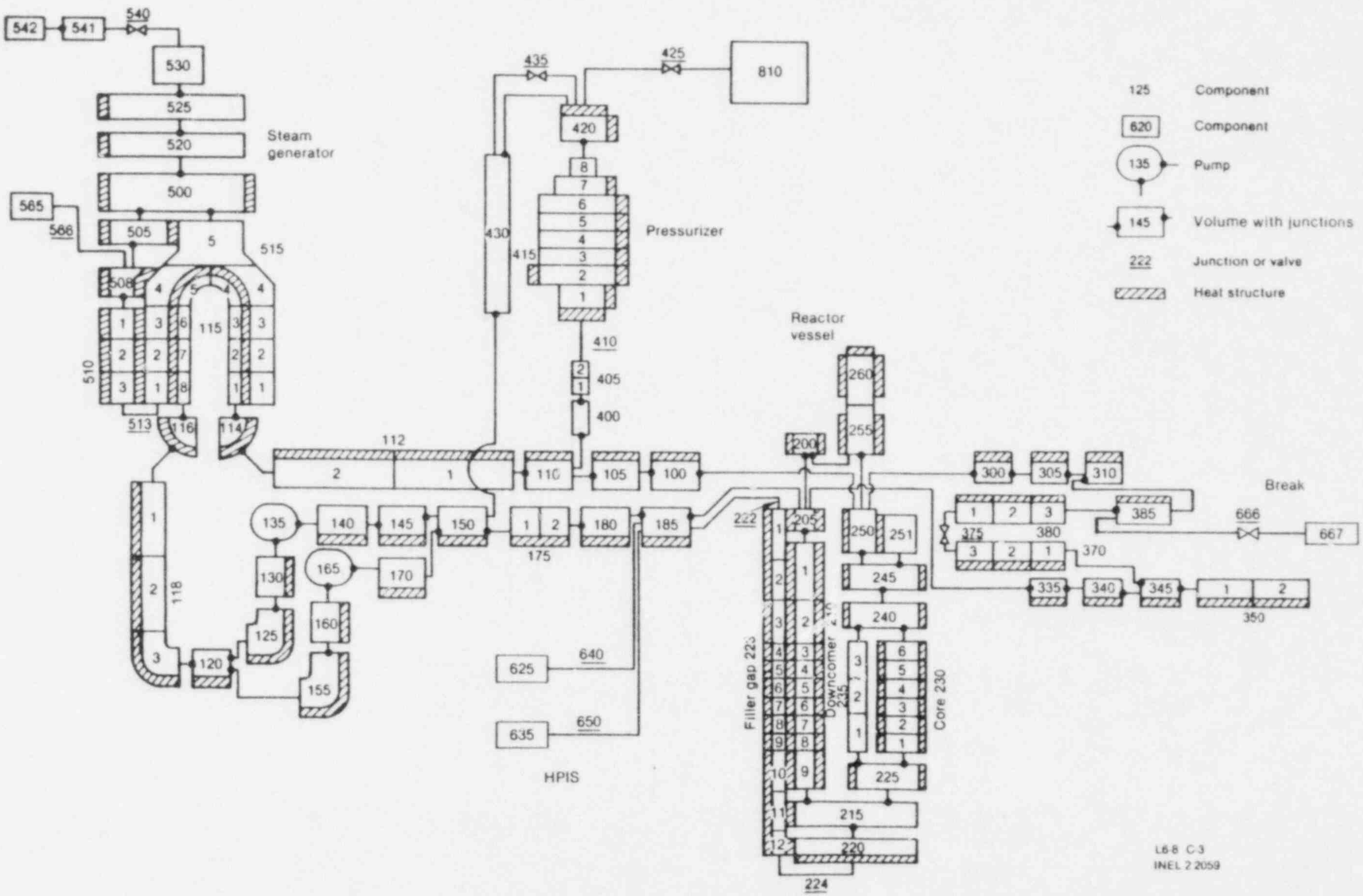

Figure 2. RELAP5 Nodalization for Experiment L6-8C

the area of a double offset shear break in a single tube of a large pressurized water reactor (LPWR), is modeled. A constant back pressure of 6.89 MPa (1000 psi) is used in the model, this is approximately equal to the pressure which would be achieved in an isolated LPWR steam generator. The calculated break flow will then be used as the actual break flow in the conduct of the experiments. In the L6-8C-3 experiment, back pressure is set at containment pressure instead of steam generator pressure. This method will maintain scaling between LOFT and an LPWR witnout requiring the design and fabrication of a scaled blowdown nozzle.

2. For Tests L6-8C-1 and C-2 the specified cool down rate was achieved by attaching a time dependent volume directly to the outlet of the steam separator with a programmed depressurization rate.

3. Two parallel flow paths were used to model the high pressure injection system (HPIS) flow, both of which used a time dependent volume and a time dependent junction. One path represents the HPIS flow as a function of primary system pressure scaled to represent HPIS, intermediate pressure injection and low pressure injection flow for a four-loop Westinghouse PWR. The other path was used when HPIS flow was throttled to maintain a constant pressurizer level and was controlled as a function of breakflow.

4. The broken loop hot leg components downstream of the contraction, mechanical Joint A^3 are removed because that piping will be flanged off for these tests.

5. The broken loop cold leg components downstream of the isolation valve are removed because the valve will remain shut during the tests.

6. The feedwater system is modeled using a time dependent volume and junction.

An initialization run was performed in order to obtain initial conditions specified in the Experiment Definition Document.[2] Time, variable, and logic trips were modified for proper simulation of each experiment scenario.

3. CALCULATIONAL RESULTS

This section contains a general overview of the results for each of the three experiments.

3.1 Experiment L6-8C-1

The initiating event for the transient is the opening of the letdown line valve in the purification system. The desired flow, to simulate a single tube rupture in a LPWR scaled to LOFT, is calculated in this analysis and is shown in Figure 3. This calculated flow will be imposed as a boundary condition during the actual experiment. The flow is choked throughout the transient until just before the primary system pressure drops below the pressure of the broken steam generator, at which time, flow would be back into the primary system terminating release of radioactive steam. Experiment L6-8C-1 is terminated when this pressure equalization is achieved.

The pressurizer level is shown in Figure 4. Since the lower pressure tap in LOFT, used for level indication, is slightly above the bottom of the pressurizer, level indication is defined as being lost when the calculated level is 0.035 m, (1.38 in.). This level is reached 144 s into the transient. Sixty seconds later, as specified in the Experiment Definition Document, both pressurizer spray flow and scaled HPIS flow are initiated resulting in a sudden increase in pressurizer level. When the pressurizer level reaches 0.1 m (3.9 in.) the HPIS flow is reduced to approximately equal break flow, thus maintaining a minimum observable pressurizer level. The pressurizer spray flow and the HPIS flow are shown in Figures 5 and 6, respectively.

Calculated primary system pressure and the back pressure at the break are shown in Figure 7. Flow from the primary to the secondary system is effectively terminated at about 430 s.

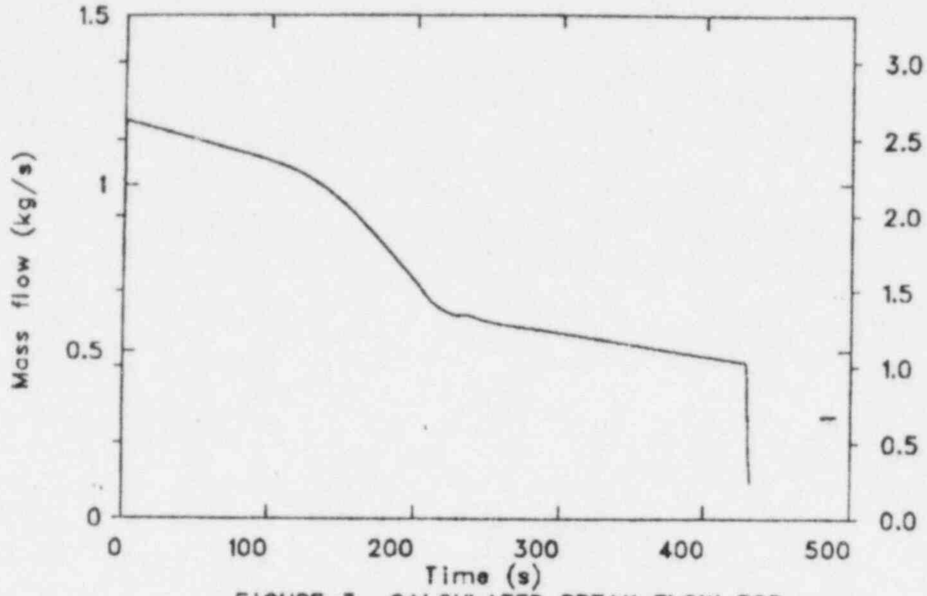

FIGURE 3 CALCULATED BREAK FLOW FOR EXPERIMENT L6-8C-1

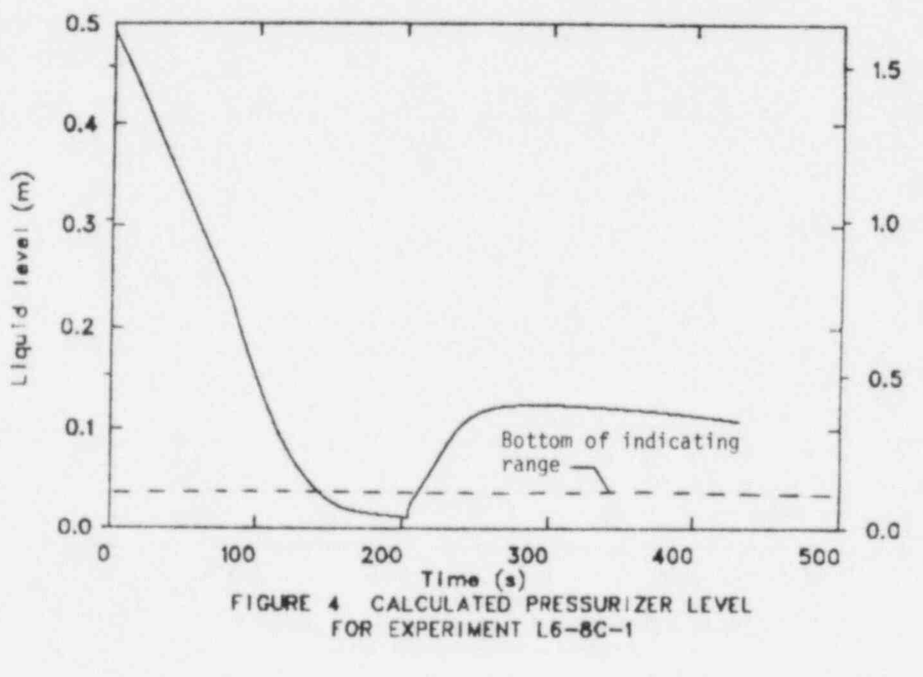

FIGURE 4 CALCULATED PRESSURIZER LEVEL FOR EXPERIMENT L6-8C-1

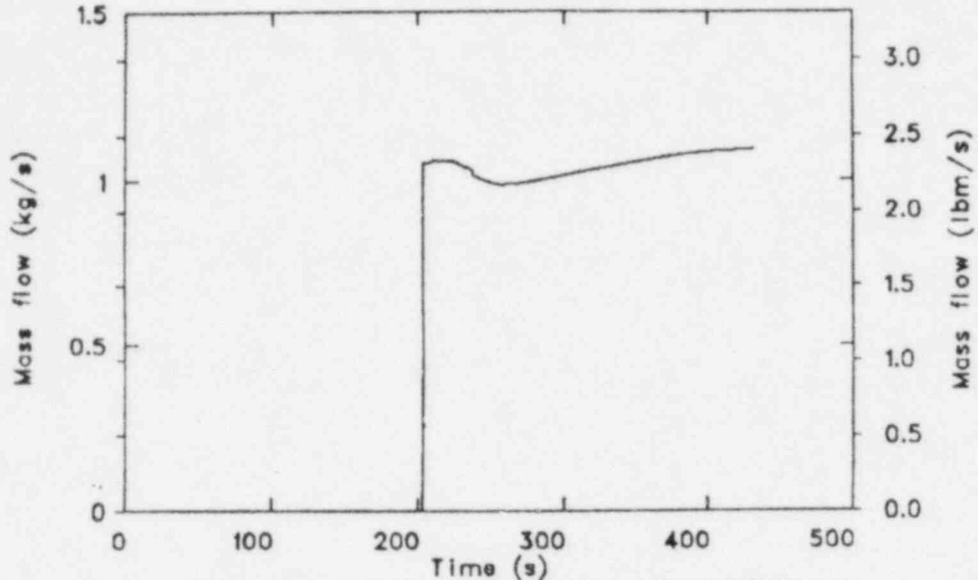

FIGURE 5 CALCULATED PRESSURIZER SPRAY FLOW FOR EXPERIMENT L6-8C-1

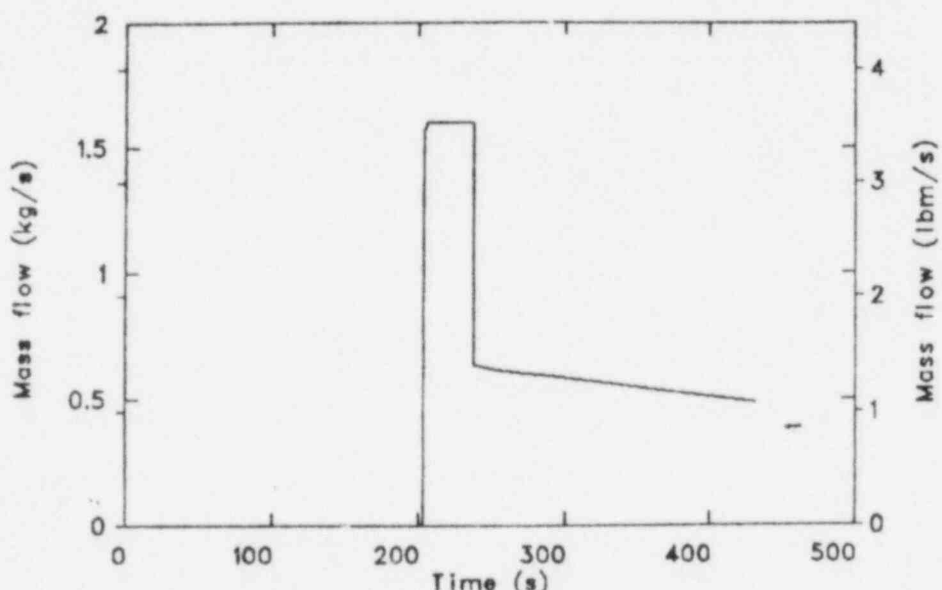

FIGURE 6 CALCULATED HPIS FLOW RATE FOR EXPERIMENT L6-8C-1

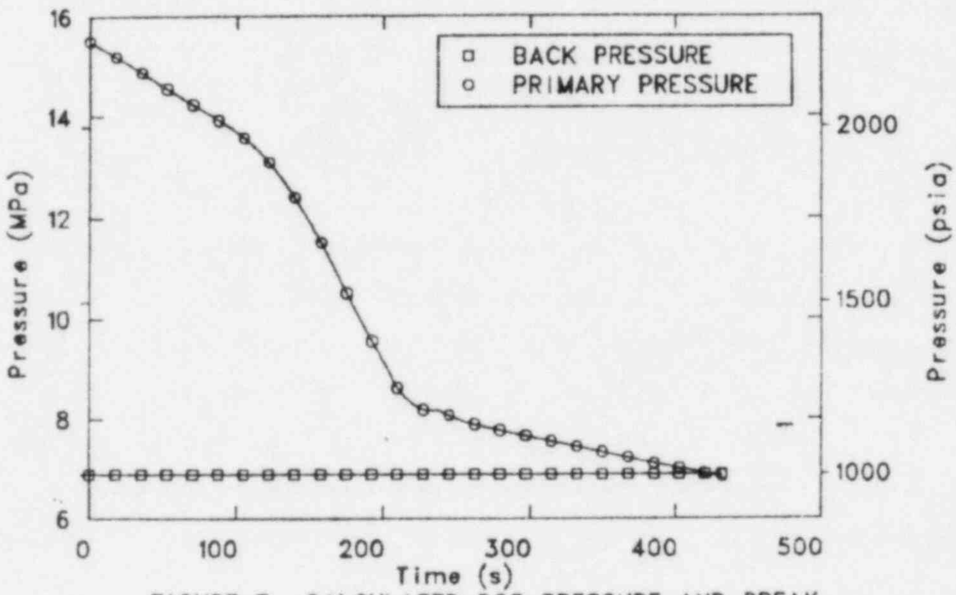

FIGURE 7 CALCULATED PCS PRESSURE AND BREAK BACK PRESSURE FOR EXPERIMENT L6-8C-1

The active steam generator is cooled at a rate of 55.6 K/h (100°F/h) by manually bleeding steam and feeding with auxiliary feedwater. The imposed secondary temperature along with the resulting primary temperature are shown in Figure 8. The sudden drop in primary temperature at 200 s is caused by introduction of the cold HPIS flow into the primary system.

3.2 Experiment L6-8C-2

The first 200 s for Experiment L6-8C-2 are identical to L6-8C-1. The difference after 200 s is in the recovery procedure. Sixty seconds after the pressurizer level is lost the pumps will be tripped, the scaled HPIS flow initiated, and the PORV manually opened. The above procedures result in a rapid depressurization of the primary system and a quick recovery of pressurizer level. Some voiding occurs in the upper head at about 250 s.

Figure 9 shows the break flow calculated for Experiment L6-8C-2 which will be used in the conduct of the experiment. The oscillations at 240 s are caused by voiding in the system. The resulting pressurizer level is shown in Figure 10. The flow from the PORV and the HPIS flow are shown in Figures 11 and 12. Because the pumps were tripped, pressurizer spray was not available. As shown in Figure 13 where the primary system pressure and the break back pressure are given, equilibrium pressure conditions were achieved at about 240 s.

3.3 Experiment L6-8C-3

The break size in Experiment L6-8C-3 was identical to C1 and C2 however, the back pressure used in the model was lower because this experiment is intended to simulate a very small break to containment. The lower back pressure had no effect on the flow as compared to C1 and C2 when the primary pressure was above about 6.8 MPa (1000 psi) because the flow was choked. The break flow is shown in Figure 14. The cooldown imposed on the secondary, in Experiments C1 and C2, was not used during this experiment in order to obtain saturation conditions and system voiding as

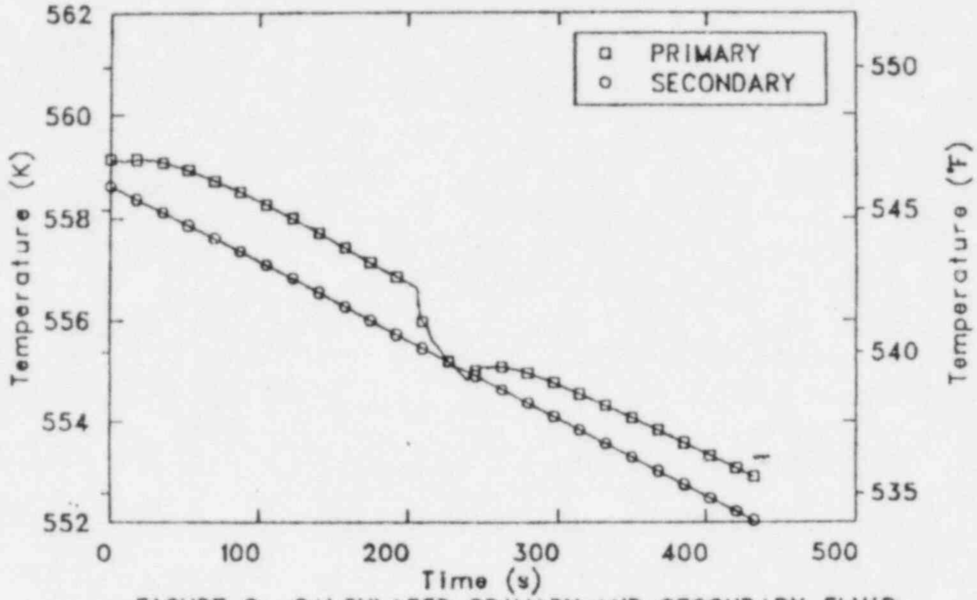

FIGURE 8 CALCULATED PRIMARY AND SECONDARY FLUID TEMPERATURES FOR EXPERIMENT L6-8C-1

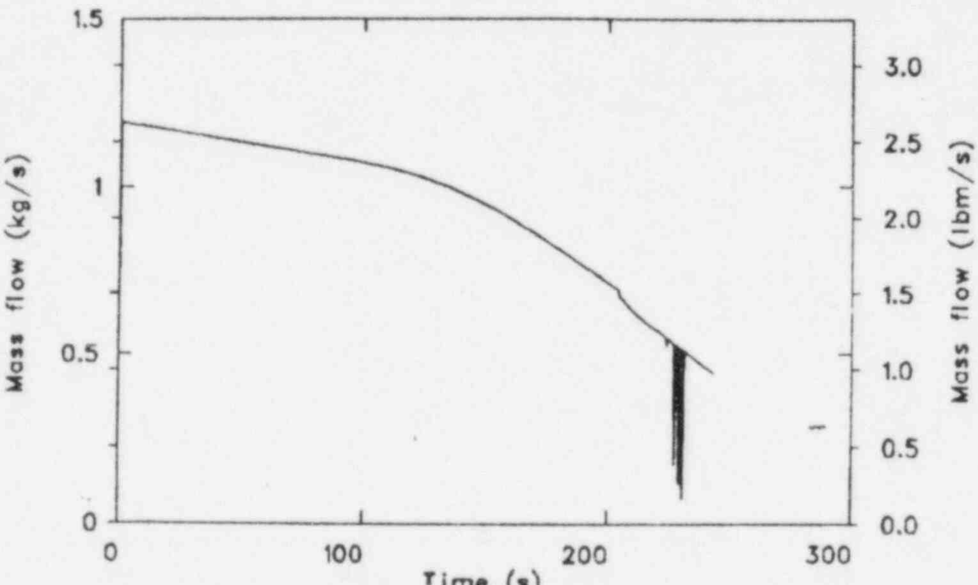

FIGURE 9 CALCULATED BREAK FLOW FOR EXPERIMENT L6-8C-2

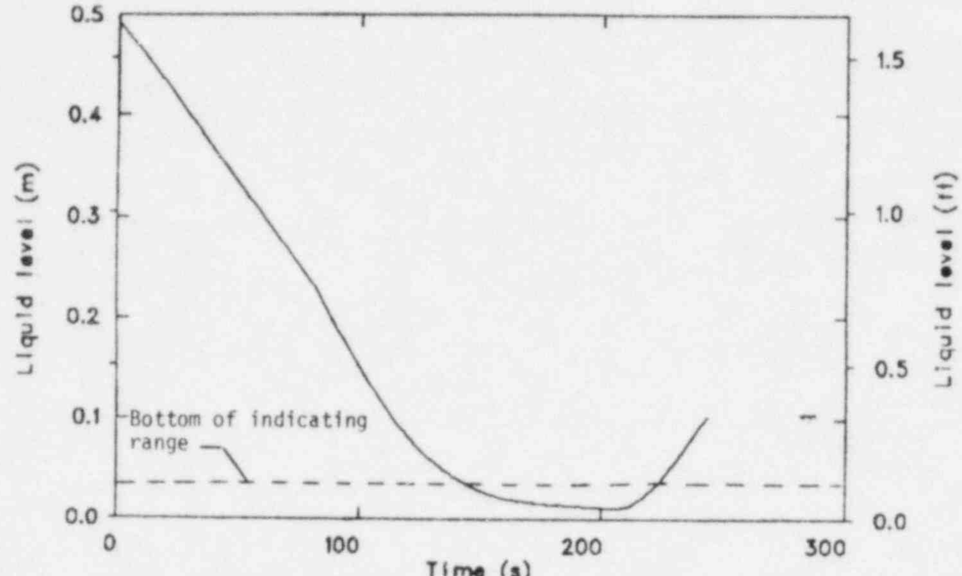

FIGURE 10 CALCULATED PRESSURIZER LEVEL FOR EXPERIMENT L6-8C-2

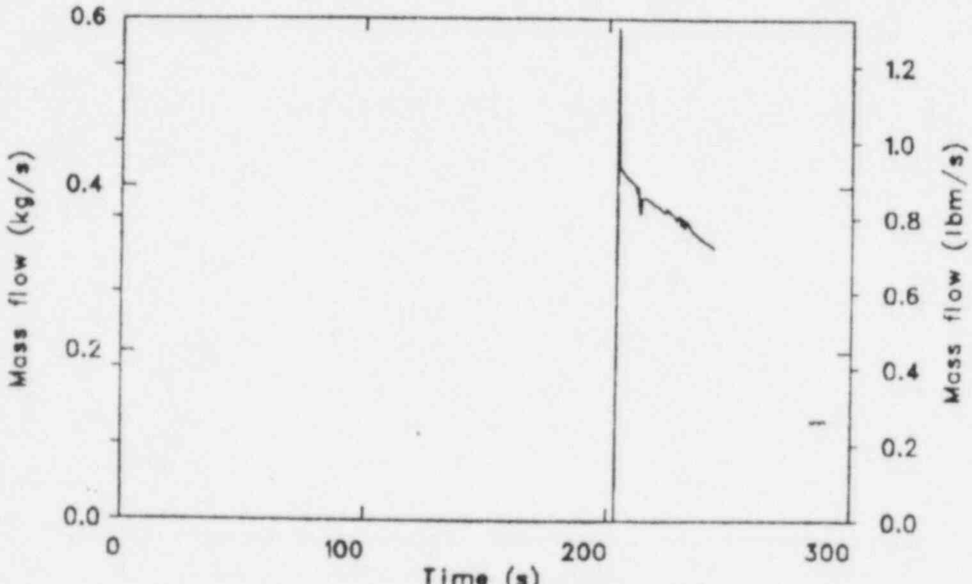

FIGURE 11 CALCULATED PORV FLOW RATE FOR EXPERIMENT L6-8C-2

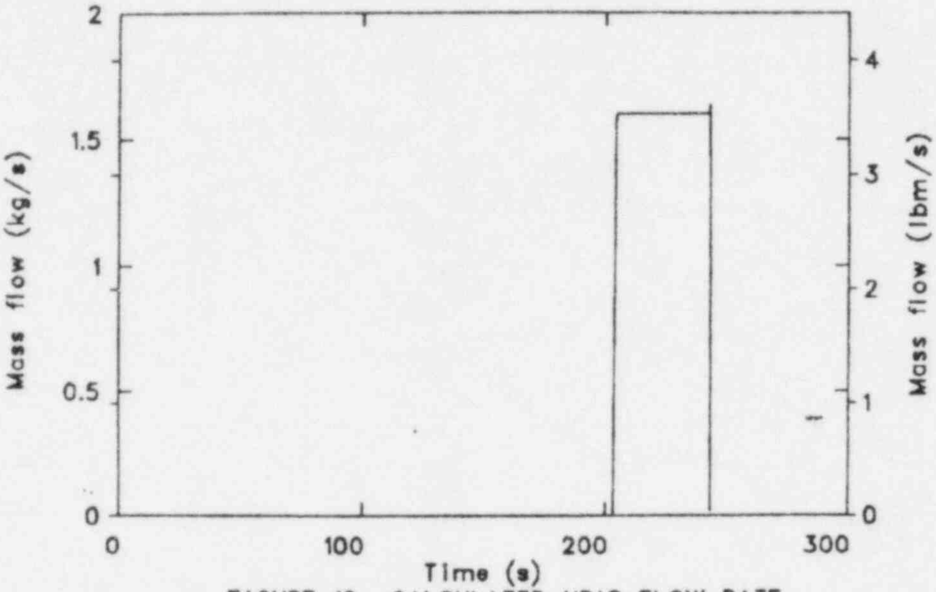

FIGURE 12 CALCULATED HPIS FLOW RATE
FOR EXPERIMENT L6-8C-2

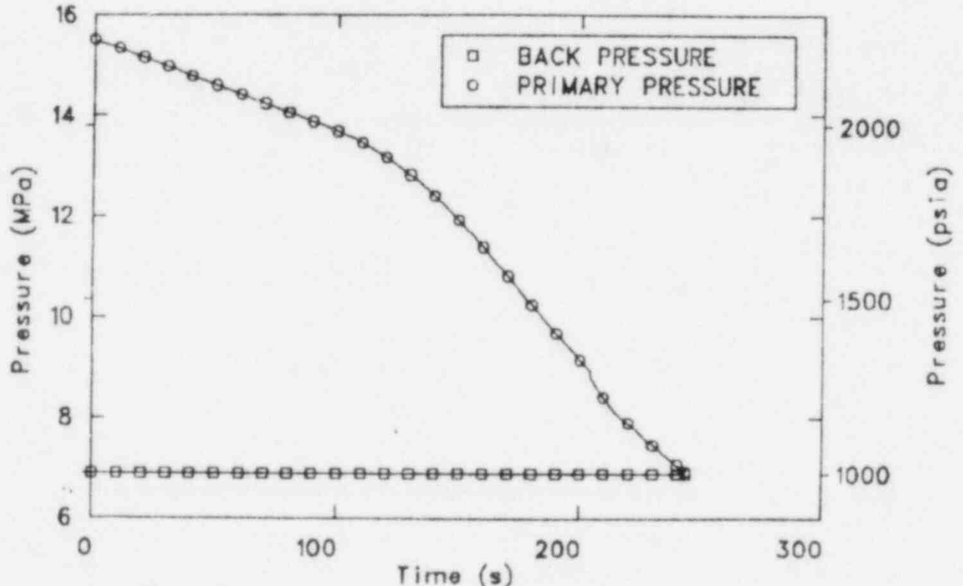

FIGURE 13 CALCULATED PCS PRESSURE AND BREAK
FOR EXPERIMENT L6-8C-2

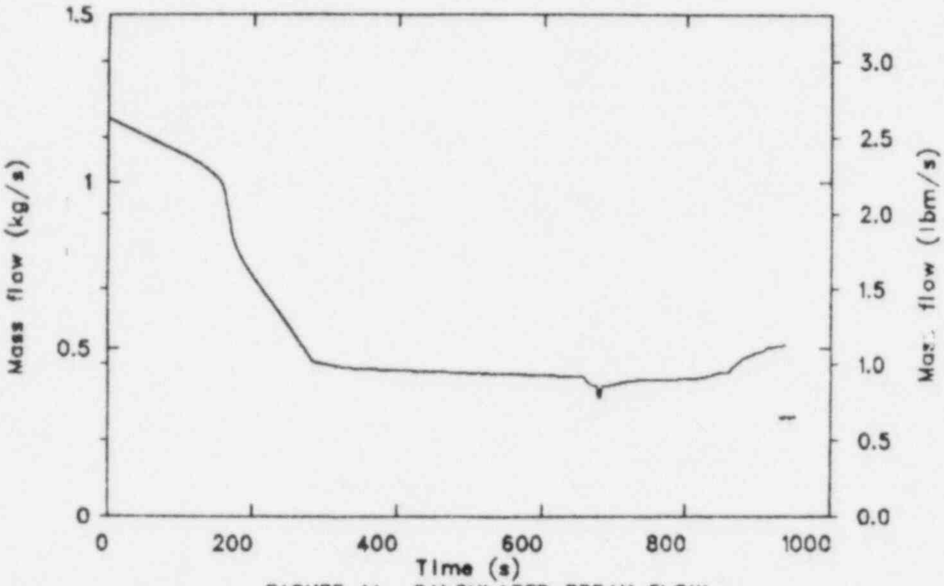

FIGURE 14 CALCULATED BREAK FLOW FOR EXPERIMENT L6-8C-3

soon as possible. The steam generator steam flow control valve cycled to relieve high pressure as shown in Figure 15. The pressurizer level is shown in Figure 16. Pressurizer level is lost at about 162 s and voiding at the pump inlet (Figure 17) begins at about 280 s. The pump inlet void reaches the prescribed 15% at about 655 s.

When the pump inlet void reached 15% the primary coolant pumps were tripped and the scaled HPIS was initiated as shown in Figure 18. It took approximately 240 s to refill the system until pressurizer level was recovered. The pumps were turned on again at 903 s when the pressurizer level recovered to 0.1 m (3.9 in.) and a hot leg subcooling of at least 5.6 K (10°F) was achieved. As shown in Figure 19 adequate subcooling was achieved at about 860 s. The experiment is concluded with reestablishment of primary system flow.

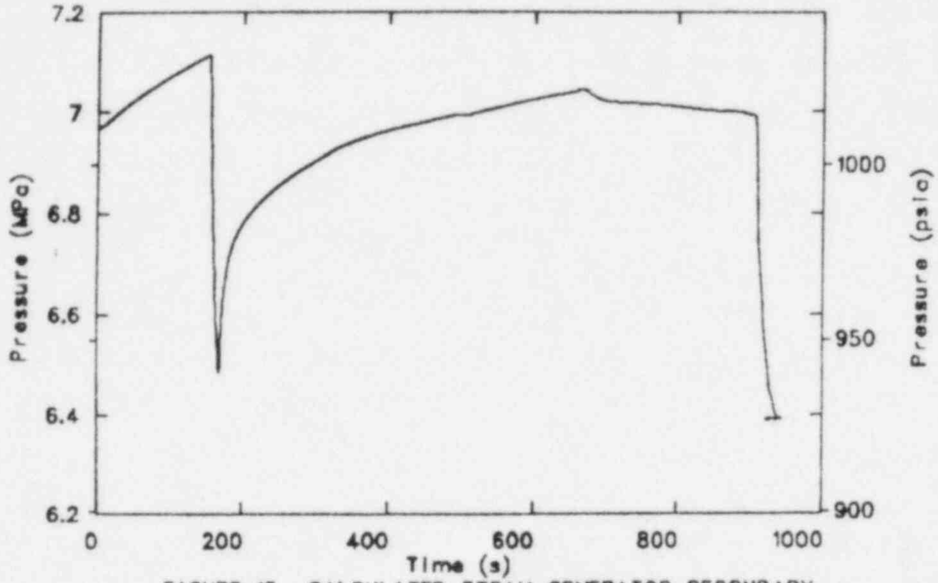

FIGURE 15 CALCULATED STEAM GENERATOR SECONDARY PRESSURE FOR EXPERIMENT L6-8C-3

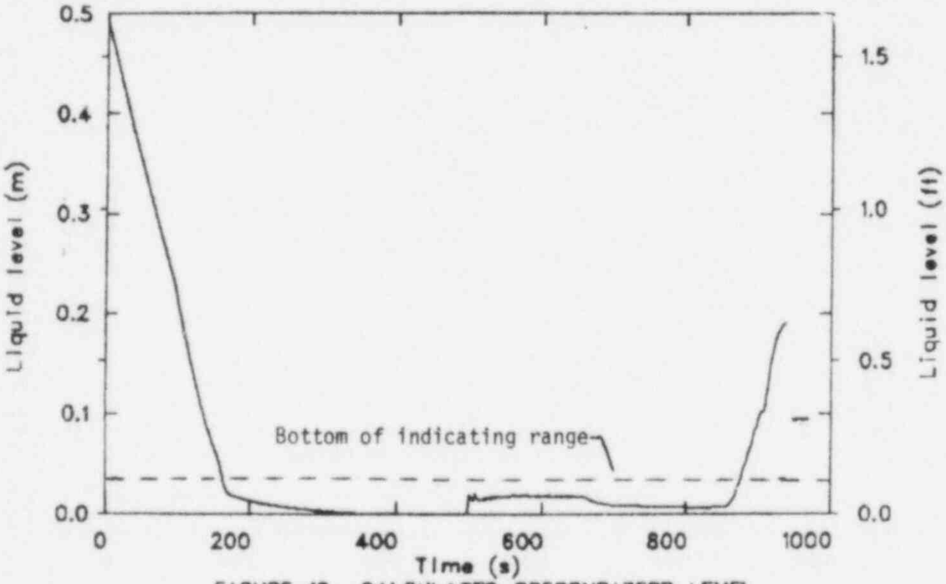

FIGURE 16 CALCULATED PRESSURIZERR LEVEL FOR EXPERIMENT L6-8C-3

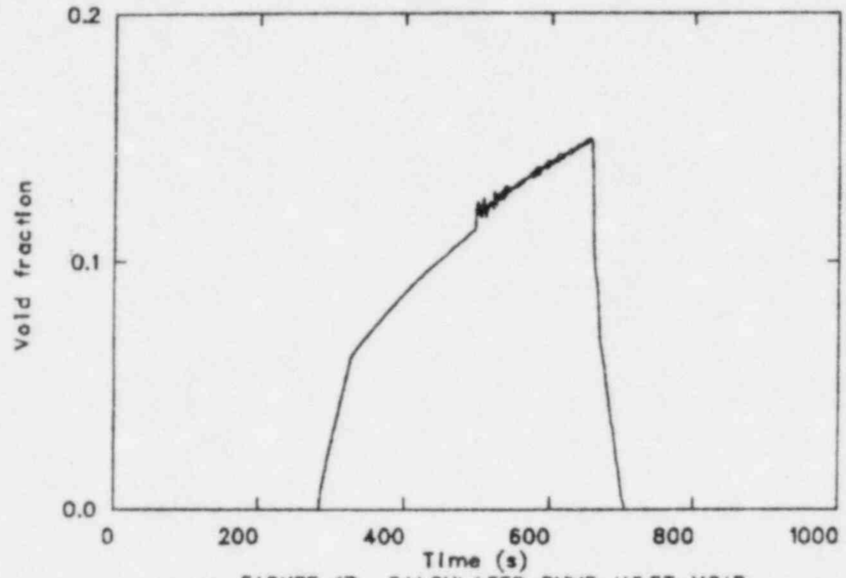

FIGURE 17 CALCULATED PUMP INLET VOID FRACTION FOR EXPERIMENT L6-8C-3

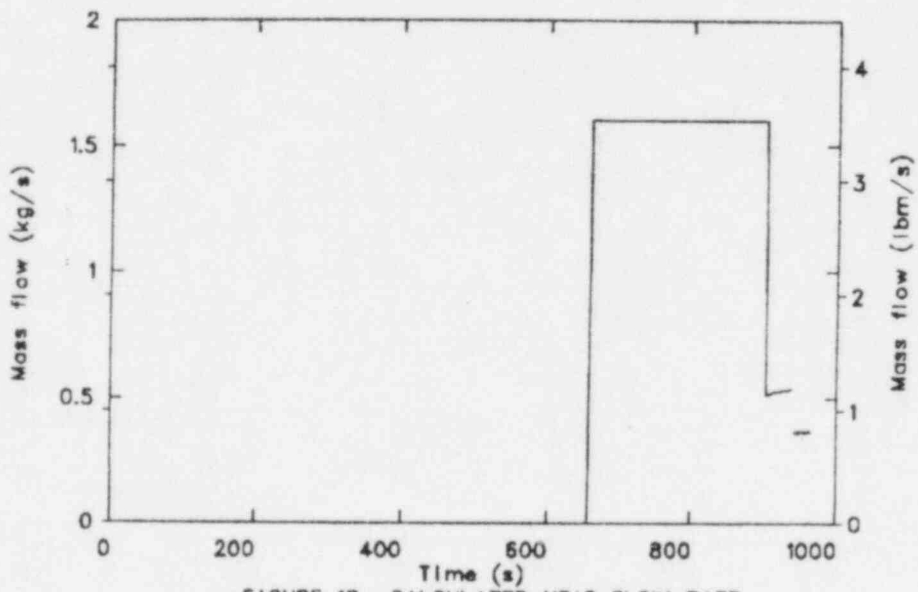

FIGURE 18 CALCULATED HPIS FLOW RATE FOR EXPERIMENT L6-8C-3

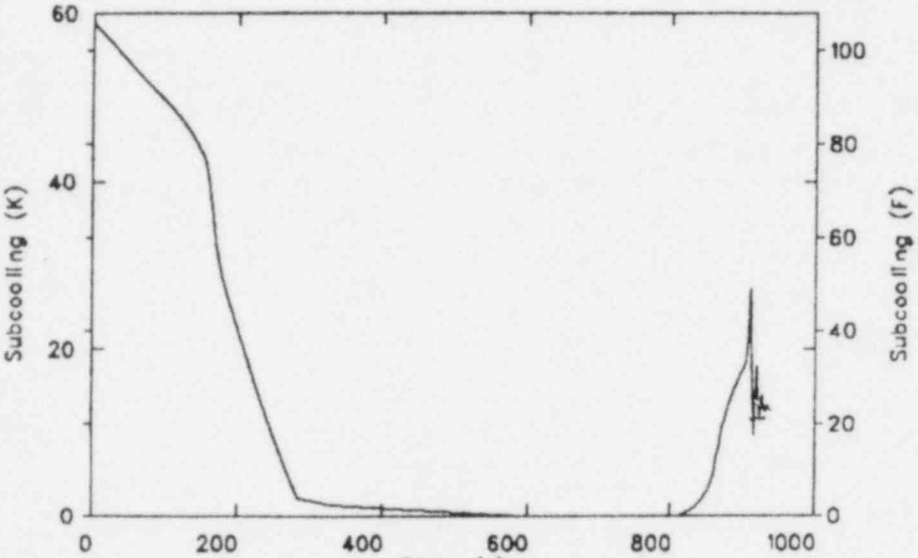

FIGURE 19 CALCULATED HOT LEG SUBCOOLING FOR EXPERIMENT L6-8C-3

4. SENSITIVITY CALCULATIONS

There are areas of uncertainty which affect the ability to accurately predict the actual experiment. All three of these experiments will be run at decay power levels which are of the same order as the heat losses in the plant. The ability to achieve the desired decay heat levels and the high uncertainty on the magnitude and location of the LOFT heat losses may affect the duration of the individual transients. The affect of these uncertainties in Experiments L6-8C-1 and C-2 will be minimized because of the prescribed cooldown rate which is imposed on the secondary. The duration of Experiment L6-8C-3 could vary from the prediction depending on the actual decay power and heat losses.

In order to define the effects of decay heat on Experiment L6-8C-3 an additional calculation was made in which the decay heat was increased from 175 kW, which was used in the base case, to 275 kW. The resultant primary system pressure response is shown in Figure 20. The higher decay heat caused the pressure to drop at a slightly slower rate than in the base case which delayed the initial void formation at the pump inlet as shown in Figure 21; however, the 15% void condition was achieved sooner in the high decay heat case. The sudden drop in void at 550 s for the high decay heat case is caused by the cycling of the main steam control valve. The steam valve cycled twice for the high decay heat case but only once for the base case (Figure 22). The pressurizer level response is basically identical for the two cases (Figure 23).

As previously noted, the break flow from these calculations will be used as a boundary condition during the conduct of each of the experiments. Since there is some uncertainty in the ability to measure and control the break flow, a sensitivity study was made to define the effects of flow uncertainty on the time required to empty the pressurizer. Using the L6-8C-3 base case model, two runs were made in which the break area was reduced by 10% and increased by 10% from the base case. The pressurizer level response is shown in Figure 24. The 10% change in break area resulted in an approximate 10 s change in the time that pressurizer level indication was lost. Because the first 150 s of each of these three experiments is basically the same the sensitivity applies to Experiments L6-8C-1 and L6-8C-2 as well.

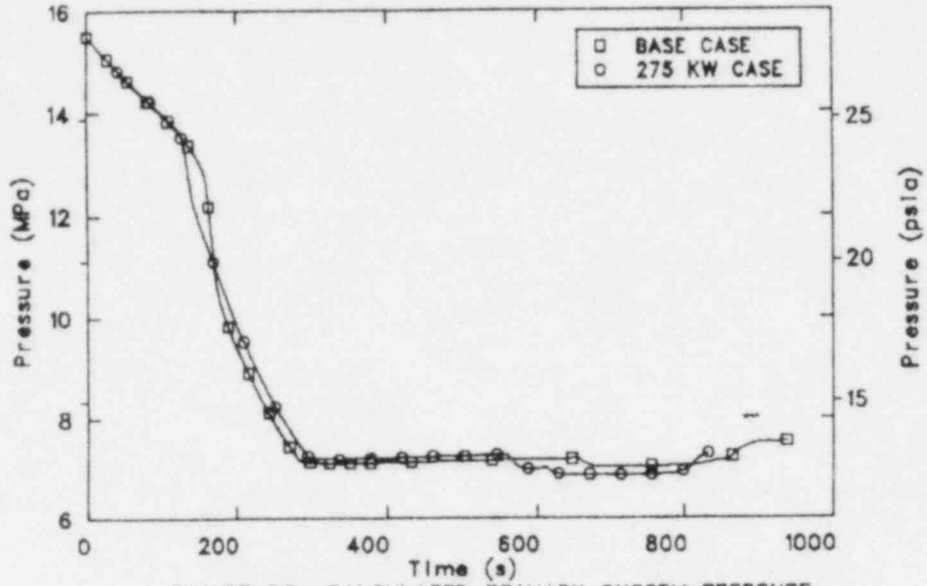

FIGURE 20 CALCULATED PRIMARY SYSTEM PRESSURE RESPONSE FOR DECAY HEAT SENSITIVITY FOR EXPERIMENT L6-8C-3

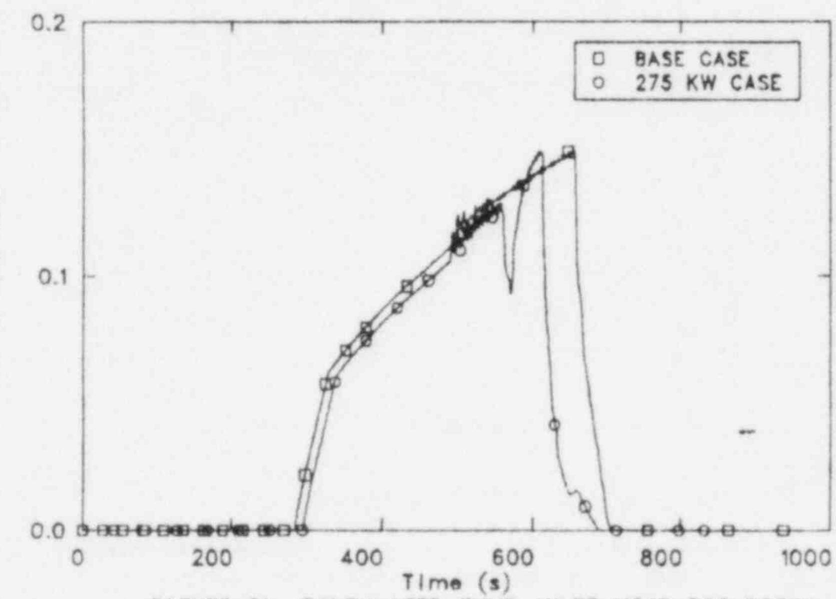

FIGURE 21 CALCULATED PUMP INLET VOID FOR DECAY HEAT SENSITIVITY FOR EXPERIMENT L6-8C-3

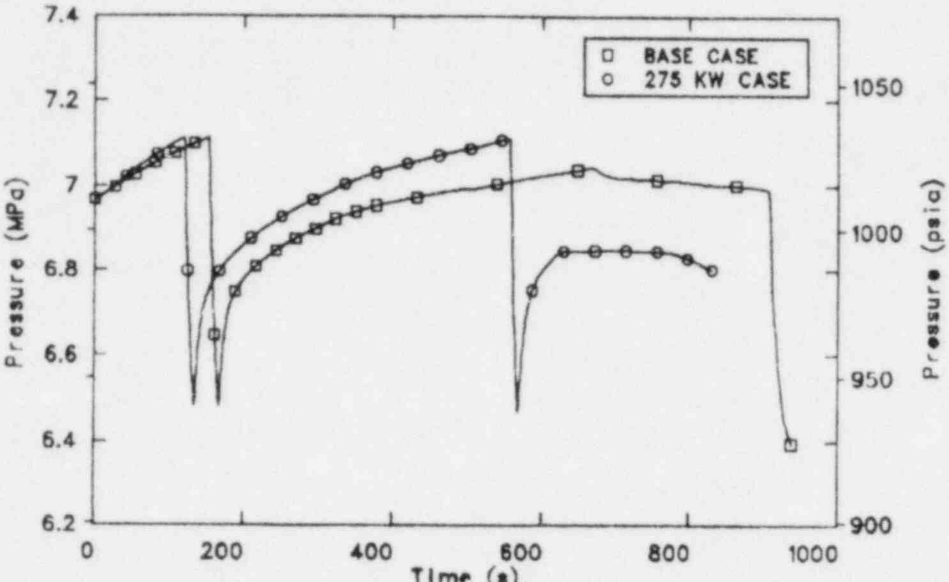

FIGURE 22 CALCULATED SECONDARY SYSTEM PRESSURE RESPONSE FOR DECAY HEAT SENSITIVITY FOR EXPERIMENT L6-8C-3

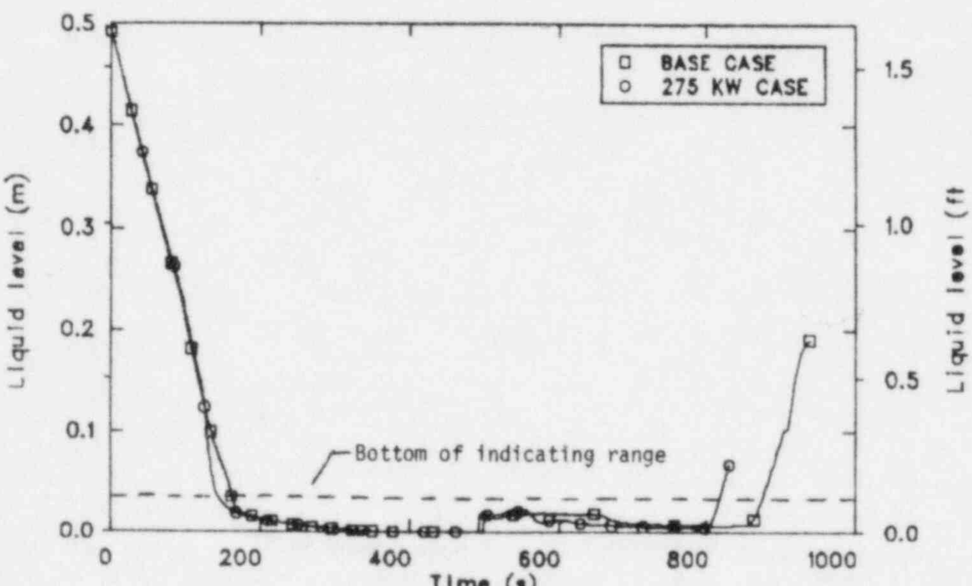

FIGURE 23 CALCULATED PRESSURIZER LEVEL RESPONSE FOR DECAY HEAT SENSITIVITY FOR EXPERIMENT L6-8C-3

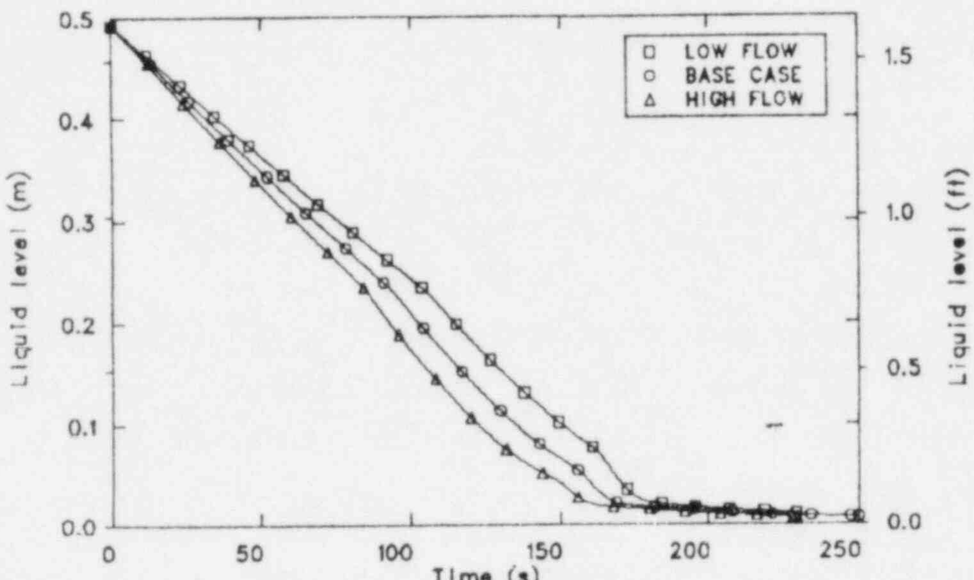

FIGURE 24 CALCULATED PRESSURIZER LEVEL RESPONSE FOR BREAK FLOW SENSITIVITY FOR EXPERIMENT L6-8C-3

5. CONCLUSIONS

The analysis described in the previous sections presents the best-estimate calculations for the three small break experiments in LOFT Series L6-8C. The calculations show that for the two steam generator tube rupture Experiments (L6-8C-1 and L6-8C-2) both recovery procedures are effective in reducing primary system pressure, thus mitigating the flow from the primary system to the secondary system. As expected, for the same steam generator energy removal rate, Experiment L6-8C-2, in which the PORV was utilized, was more effective in reducing the primary system pressure. However, this procedure resulted in the loss of about 15 Kg (33 lbm) of mass out of the PORV and introduced the potential for additional mass loss should the PORV stick open.

Figure 25 shows the integrated break flow for Experiments L6-8C-1 and L6-8C-2 over the time required to reach pressure equilibrium. Approximately 106 Kg (233 lbm) more mass was lost from the break during Experiment L6-8C-1 than in L6-8C-2. The amount of mass lost in each experiment, however, is dependent on the imposed steam generator cooldown rate which determines the total amount of energy transferred from the primary and secondary systems.

Experiment L6-8C-3 was designed to demonstrate the use of pump power to indicate system void and inventory. This analysis shows that the test objectives (Appendix A) can be met if the experiment is carried out as planned.

The break flow rates calculated in this analysis for each of the three experiments can be used as a boundary condition during the actual conduct of the experiments which will provide the proper break flow scaling to an LPWR.

The calculations show that the plant, at the conclusion of each of the experiments is in a stable and safe condition with the envelope of conditions within the intermediate shutdown (Mode 3) range as defined by the LOFT Technical Specifications.

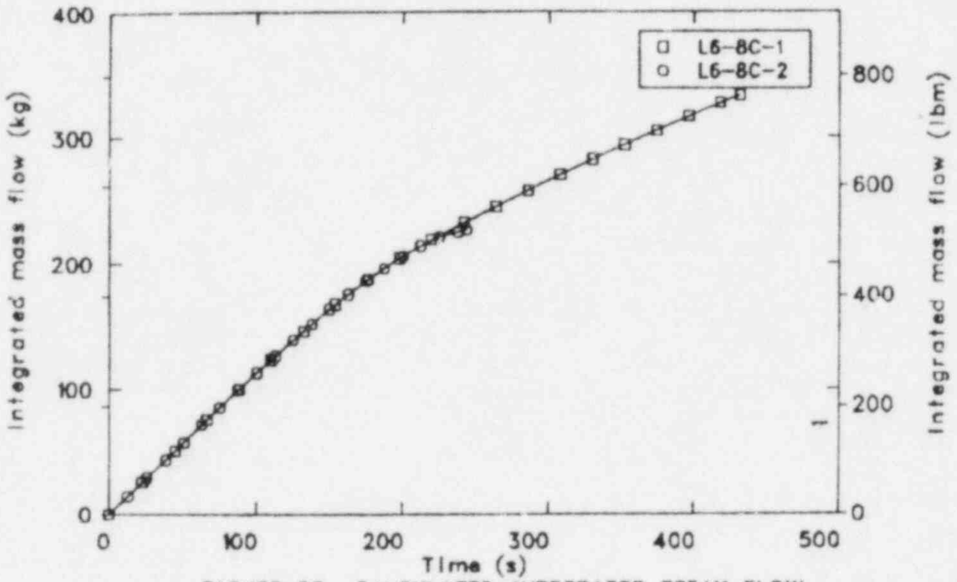

FIGURE 25 CALCULATED INTEGRATED BREAK FLOW RATE FOR EXPERIMENTS L6-8C-1 AND L6-8C-2

6. REFERENCES

1. V. H. Ransom et al., RELAP5/MOD1 Code Manual Volumes 1 and 2 (Draft), NUREG/CR-1826 EGG-2070, November 1980.

2. D. J. Varacalle, Jr., LOFT Experiment Definition Document--L6-8 Anticipated Transient Series, EGG-LOFT-5734, July 1982.

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

APPENDIX A

EXPERIMENT SERIES L6-8C AND LOFT FACILITY DESCRIPTIONS

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EXPERIMENT SERIES L6-8C AND LOFT FACILITY DESCRIPTIONS

The small break recovery procedure, Experiments L6-8C, will apply to breaks to containment as well as breaks to the secondary system. The concerns and actions of a reactor operator are different for these two events.

A steam generator tube rupture is one of the most common sources of accidental radioactivity release for a commercial PWR. The justification for these experiments (L6-8C-1, -2) in LOFT is to provide industry with confidence in procedures that can mitigate this event. The tests will simulate the required operator actions, following a steam generator tube rupture and loss of pressurizer level to bring the primary system pressure to below that of the steam generator with the ruptured tube, and thus mitigate the release of radioactivity to the environment.
Experiment L6-8C-1 will demonstrate an alternative primary system recovery technique (pumps on) in a steam generator tube rupture event such as that which occurred in the GINNA event. In this experiment and L6-8C-2, the effect of the broken steam generator back pressure and a double-ended tube rupture will be simulated but not the isolation of the broken steam generator and the consequent heat transfer effects. Experiment L6-8C-1 will involve steam generator operation, combined with pressurizer spray actuation, to simulate the mitigation of radioactivity release from the secondary system to the atmosphere. Experiment L6-8C-2 will simulate the procedure normally used in commercial PWRs to recover from a steam generator tube rupture event. The pumps will be tripped on low pressure, and the PORV will be utilized to reduce the system pressure.

Experiment L6-8C-3 will demonstrate the use of reactor coolant pump motor power or current to determine reactor coolant system inventory during a small break. The system inventory will be allowed to decrease in the experiment until the pump trip criteria described in Reference A-1 is met, and the pumps are tripped. System inventory would then be increased until the pump restart criteria was met, and the pumps restarted.

1. Experiment Objectives

The L6 experiment series was developed to study anticipated transients. Data from the experiments will be utilized in evaluating the computer codes and analytical techniques used to predict anticipated transients, including anticipated transients without scram (ATWS).

The intent of the LOFT experiments is to provide data which, through the verification of computer codes and analytical models, will contribute to an understanding of symptoms, events and plant conditions leading to emergency or off-normal situations. In addition, the LOFT experiments will provide information on the ability of reactor trip systems, engineered safety features, and manually initiated systems to perform their intended functions.

In order to address issues relating to the identification and recovery from anticipated transient events, the following major programmatic objectives have been defined:

1. Assist NRC in evaluating reactor transient analysis techniques used in reactor licensing by applying the same techniques to transients performed in the LOFT facility

2. Demonstrate that LOFT results can be related to larger PWRs by providing data that can be compared to data obtained from commercial plants (traceability)

3. Provide data for evaluating commercial plant instrumentation and control system response characteristics over a range of transients which could occur in a commercial plant.

To support the above programmatic objectives, experiment specific objectives have been defined for each of the L6-8 experiments. The experiment specific objectives for these experiments are defined as those which can be evaluated shortly after the conduct of the experiment.

1. Experiment L6-8C-1:

   a. Evaluate a primary coolant system recovery technique, for a primary system to secondary system break which avoids a challenge to the PORV.

   b. Determine whether the proposed procedure will enhance plant control.

   c. Evaluate the effectiveness of the proposed operator pump current display system in ascertaining primary coolant system inventory.

2. Experiment L6-8C-2:

   a. Provide a base comparison experiment by employing procedures which use the PORV to mitigate a steam generator tube rupture.

   b. Determine the effectiveness of the proposed operator pump current display system in ascertaining primary coolant system inventory.

3. Experiment L6-8C-3:

   a. Evaluate the the use of reactor coolant pump motor power or current to determine reactor coolant system inventory during a small break to containment.

   b. Evaluate the effectiveness of a small break recovery procedure in which the PCPs continue running until a system void fraction of 0.15 is reached.

c. Determine the effectiveness of the proposed operator pump current display system in ascertaining the primary coolant system inventory.

2. LOFT FACILITY DESCRIPTION

The LOFT facility is described in detail in Reference A-2. The LOFT instrumentation and major components of interest for this experiment series are shown in Figures A-1 and A-2.

3. REFERENCES

A-1. D. J. Varacalle Jr., LOFT Experiment Definition Document--L6-8 Anticipated Transient Series, EGG-LOFT-5734, July 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.

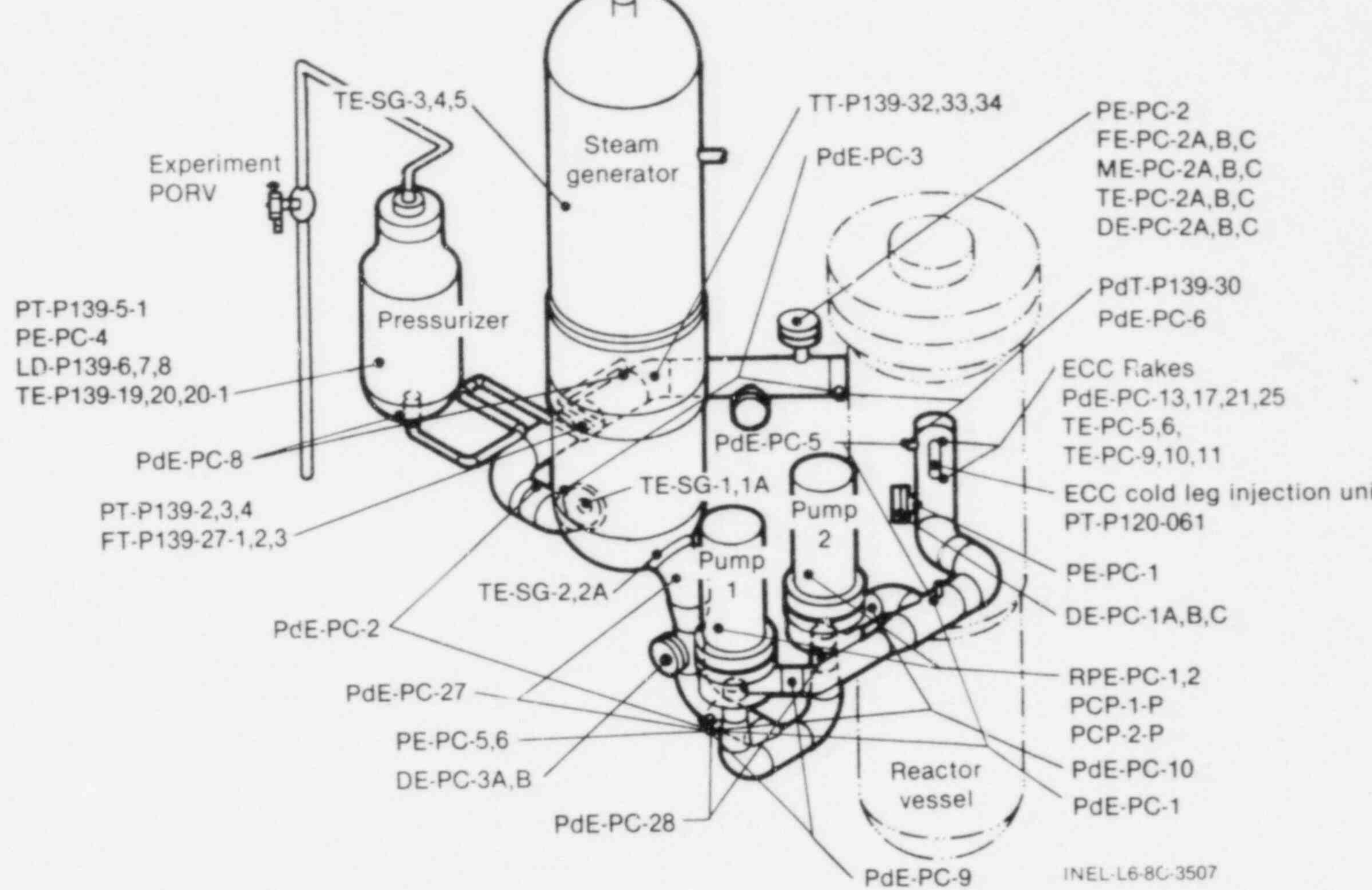

Figure A-1. LOFT intact loop components showing thermo-fluid instrumentation

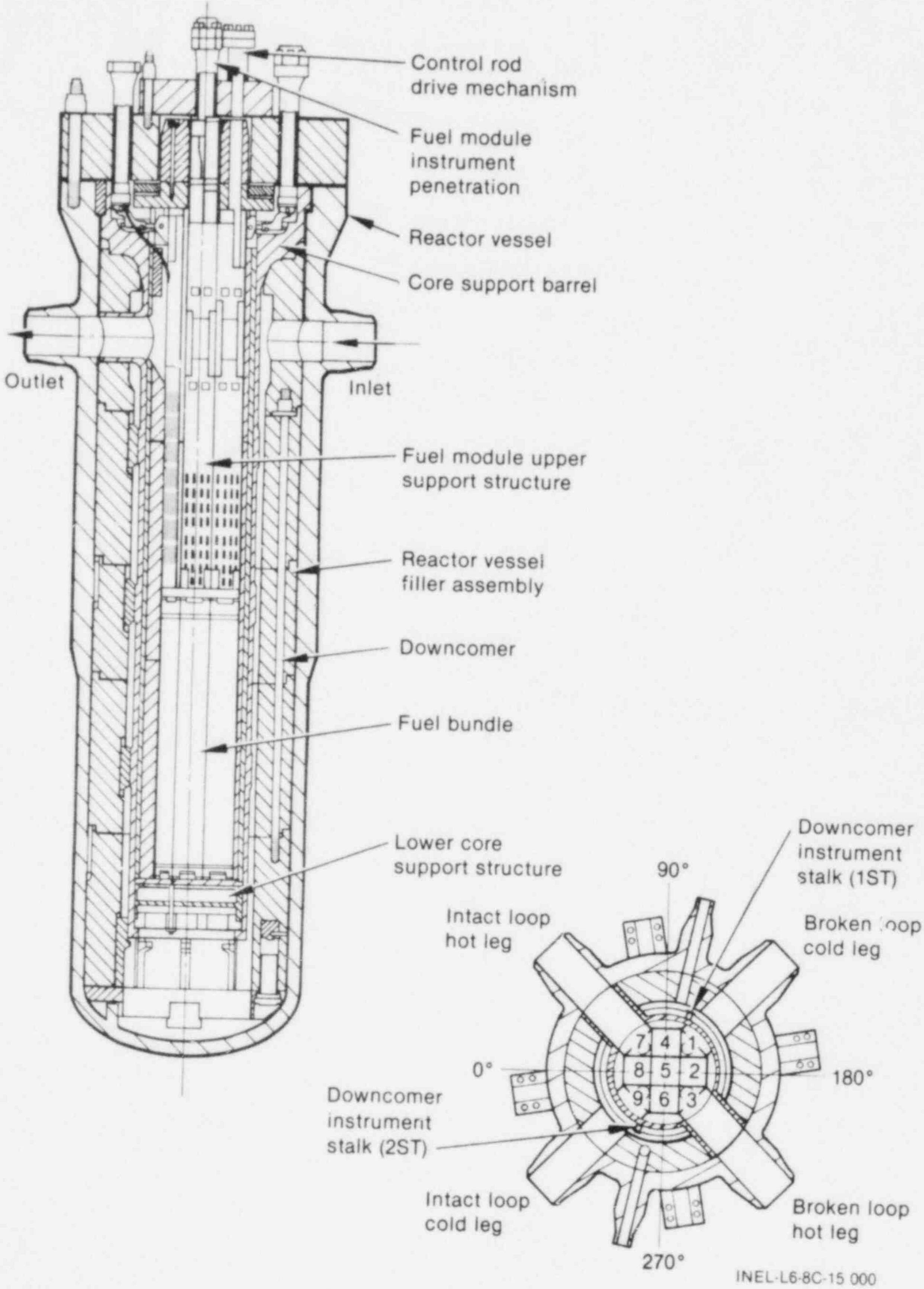

Figure A-2. LOFT reactor vessel

APPENDIX B
CODE UPDATES AND INPUT LISTINGS

APPENDIX B
CODE UPDATES AND INPUT LISTINGS

The following listings are included on the microfiche in the pouch on the inside of the report back cover.

1. RELAP5 code updates to Cycle 15

2. Input deck for L6-8C-1

3. Input deck for L6-8C-2

4. Input deck for L6-8C-3.

NRC Research and/or Technical Assistance Rept PDR

EG&G Idaho, Inc.

P.O. BOX 1625, IDAHO FALLS, IDAHO 83415

August 13, 1982

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

BEST ESTIMATE PREDICTION FOR LOSS-OF-FLUID TEST (LOFT) SMALL BREAK RECOVERY EXPERIMENT SERIES L6-8C - LPL-180-82

Dear Mr. Tiller:

The attached report is the prediction for the LOFT Small Break Recovery Experiment Series L6-8C. Experiment Series L6-8C consists of three separate experiments. Experiment L6-8C-1 simulates a recovery procedure from a small break between the primary coolant system and the secondary side of one steam generator in a four-loop commercial pressurized water reactor (PWR) using available steam generator energy removal capacity and pressurizer sprays to reduce primary system pressure below the effected steam generator secondary pressure. Experiment L6-8C-2 simulates a different recovery procedure from the same small break with the same steam generator energy removal but with the pressurizer power operated relief valve (PORV) opened to further reduce primary system pressure. Experiment L6-8C-3 simulates a recovery procedure from a small break to containment in which the electrical power to the primary coolant pumps will be monitored to provide an indication of primary coolant system mass inventory. These experiments will provide operational data for specific recovery procedures in the LOFT reactor and will provide a basis for investigating accident management procedures in commercial PWRs.

The RELAP5/MOD1 computer code was used to predict these three experiments. The predictions indicate Experiment L6-8C-1 will be complete at 430 s when the primary pressure reaches 6.9 MPa (1000 psia), which is the assumed back pressure for the effected steam generator. In Experiment L6-8C-2, with the PORV opened, the experiment is calculated to be complete at 240 s. In Experiment L6-8C-3, the average system void fraction is calculated to reach a value of 0.15 and the primary coolant pumps turned off at 655 s. Experiment L6-8C-3 is calculated to be complete at 903 s when the pressurizer level recovers to the indicating range and the primary coolant pumps are restarted. Calculations have been performed to ascertain the sensitivity of the results to uncertainties in decay heat and flow out to break.

The results of the calculations indicate that the experiments, if conducted as planned, will meet the stated objectives.

Very truly yours,

L. P. Leach

L. P. Leach
Manager, LOFT Department

KGC:seb

Attachment:
As Stated

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