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POSTTEST ANALYSIS OF LOFT ANTICIPATED TRANSIENT EXPERIMENTS L6-1, L6-2, L6-3 AND L6-5

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Operated by the U.S. Department of Energy

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ABSTRACT

LOFT anticipated transient Experiments L6-1, L6-2, L6-3 and L6-5 provided useful data for evaluating thermal-hydraulic computer codes. The experiments simulated the following initiating events: Loss of steam load (L6-1); loss of forced primary coolant system flow (L6-2); excessive steam load increase (L6-3); and loss of feedwater (L6-5). Comparison of posttest calculations using the RELAP5 code to the measured data are presented. It is concluded that RELAP5 is able to satisfactorily simulate anticipated transients, provided accurate input models are used. Code and model improvements to address some remaining deficiencies are identified.

SUMMARY

This report presents the results of analyses of four anticipated transient experiments performed in the LOFT facility along with the posttest computer code analyses performed with RELAP5/MOD1.

These experiments were L6-1 (loss of load transient), L6-2 (loss of forced primary coolant system flow), L6-3 (excessive load increase transient), and L6-5 (loss of feedwater transient). They are representative of a wide spectrum of transients which can occur in a commercial pressurized water reactor (PWR). These experiments provide a useful data set against which thermal-hydraulic codes can be evaluated. Part of this evaluation process is documented in this report.

The RELAP5/MOD1 computer code was used for the post-experiment calculations described in this report. The RELAP5/MOD1 posttest calculations used measured initial and boundary conditions and were, in general, able to calculate the transients quite well. Modifications suggested to improve the RELAP5/MOD1 calculations include determining a method to improve the steam generator secondary initial conditions.

The portions of the computer code input model requiring particular care for the analyses were the pressurizer, including spray flow and heaters; the steam generator, including the feedwater and steam flow controllers; and the reactor power, including the effect of power history.

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POSTTEST ANALYSIS OF LOFT ANTICIPATED TRANSIENT EXPERIMENTS L6-1, L6-2, L6-3 and L6-5

1. INTRODUCTION

This report describes the results of analyses of four nuclear anticipated transient experiments conducted by EG&G Idaho, Inc., at the Loss-of-Fluid Test (LOFT) facility at the Idaho National Engineering Laboratory. The experiments, performed in May and October 1980, simulated transients in a typical four-loop commercial pressurized water reactor (PWR). The four experiments were L6-1 (loss of steam load), L6-2 (loss of forced primary coolant system flow), L6-3 (excessive steam load increase), and L6-5 (loss of feedwater). Pre-experiment calculations were performed using the RETRAN01/MOD2 computer code.[1] Comparisons of pre-experiment calculation with measured data have been presented in Reference 2. Post-experiment calculations were performed using the RELAP5/MOD1 (hereafter designated RELAP5) computer code.[3] This report presents the results of these analyses.

The LOFT facility is a 50 MW(t) pressurized water reactor (PWR) designed to operate over the range of power densities and operating conditions representative of those in a commercial PWR, and to simulate loss of coolant accidents (LOCAs) and anticipated transients. A description of the LOFT system configuration for the anticipated transient experiments is contained in Appendix A. A detailed description of the LOFT facility is contained in Reference 4.

Anticipated transients are defined as a class of transients which have a probability of occurring at least once during the lifetime of a commercial PWR. Various anticipated transients are analyzed and documented in the Final Safety Analysis Report (FSAR) issued prior to the licensing of a commercial PWR. The results of the FSAR analyses are used to determine the technical specifications under which the PWR is allowed to operate. The analysis of these transients is generally performed by the reactor vendors using their own proprietary computer codes. By comparing code calculations with LOFT experimental data it is possible to evaluate both modeling techniques and the capability of a specific computer code.

Each of these four experiments was conducted with the reactor initially at conditions typical of commercial PWR nominal operating conditions. The four anticipated transient experiments are described in Section 2.

The pre-experiment calculations performed using RETRAN01/MOD2 are documented in Reference 5. The basis for the experiment planning for these four experiments is detailed in Reference 6. Some experimental data and pre-experiment calculation comparisons are presented in References 7 and 8. Detailed system and measurement data information for all four experiments are documented in References 9 and 10.

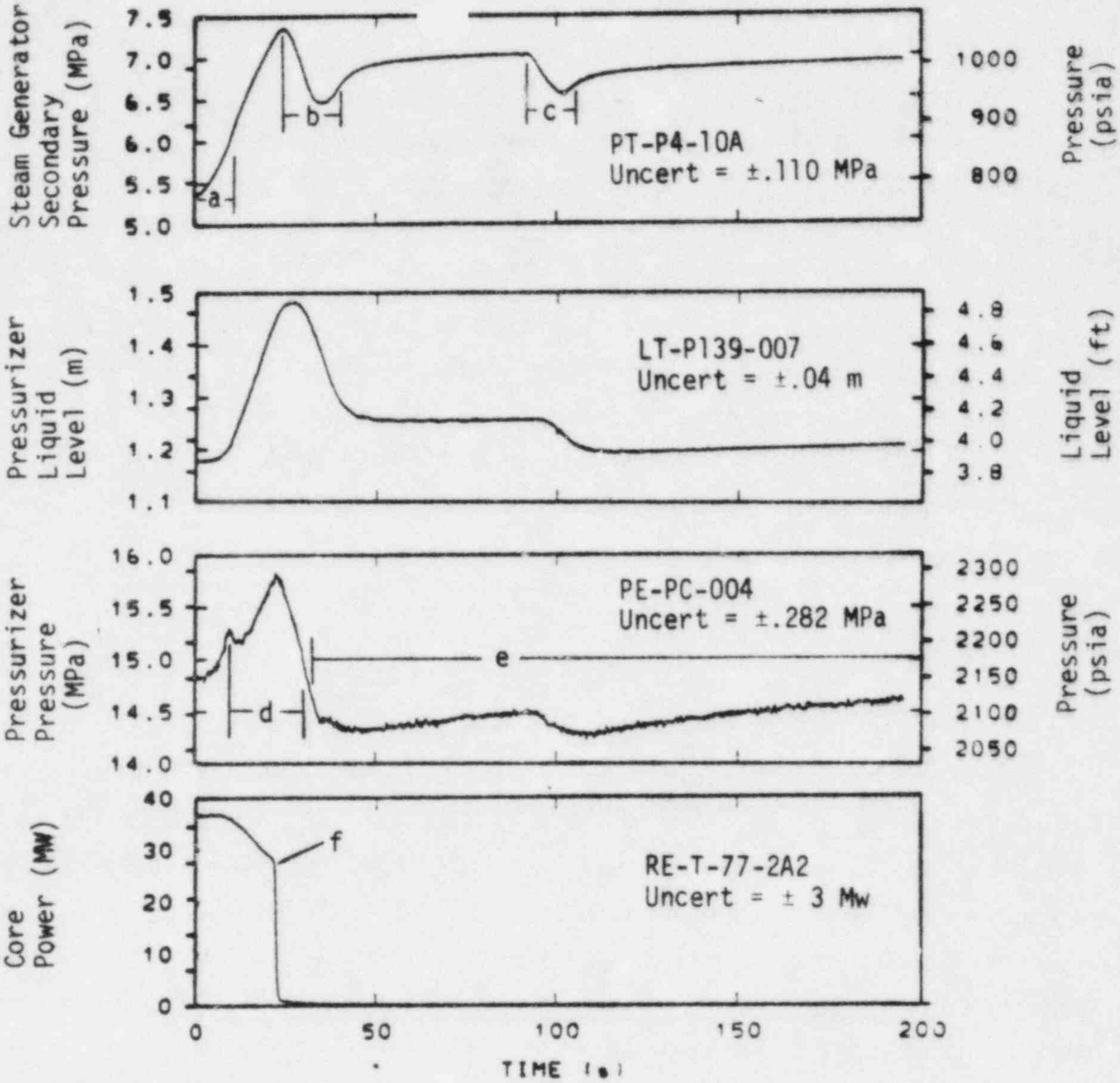
2. EXPERIMENT DESCRIPTIONS

The LOFT Experiments L6-1, L6-2, L6-3, and L6-5 were the first four experiments in the LOFT anticipated transient experiments series (L6 series). Experiment L 5 was performed in May 1980. The other three experiments were performed in October 1980.

In these four experiments similar phenomena were observed. The primary coolant system remained subcooled except in the pressurizer. Changes in core power due to reactivity changes were also observed in all of the experiments. These experiments provided useful data to evaluate various PWR phenomena. In the following sections each transient will be discussed in detail. Initial conditions and sequence of events for each experiment are given in Appendix B.

2.1 Anticipated Transient Experiment L6-1

Experiment L6-1 was initiated by the closure of the main steam control valve (MSCV) simulating the loss of load in a large PWR. As soon as the MSCV started to close, steam generator secondary pressure started to rise (see Figure 1a). The rising secondary pressure and temperature in turn reduced heat transfer across the steam generator (SG) tubes and the average primary coolant system (PCS) temperature began to rise. The rise in PCS temperature had two visible effects: (a) the PCS heatup caused an insurge into the pressurizer due to swelling of the liquid (see pressurizer liquid level, Figure 1), and (b) the PCS heatup caused core power to decrease due to the negative reactivity associated with moderator heatup (see core power, Figure 1). The rapid insurge into the pressurizer caused the pressurizer vapor to be compressed, with the resulting rise in pressurizer pressure (Figure 1). Pressurizer pressure dropped momentarily when the pressurizer spray started injection automatically on a high pressure signal at 9.1 s. The 3 s pressure drop when spray was first initiated is believed to be due to initial desuperheating of the vapor space steam when spray was initiated.



a. MSFCV closes 0 - 11.6 s

b. MSFCV cycles 22.2 - 40.6 s

c. MSFCV cycles 91.2 - 104.4 s

d. Pressurizer spray on 9.1 - 30.4 s

e. Pressurizer backup heaters on 32.5 - 415.4 s

f. Reactor scrammed 21.8 s

Figure 1. LOFT data for loss of load Experiment L6-1.

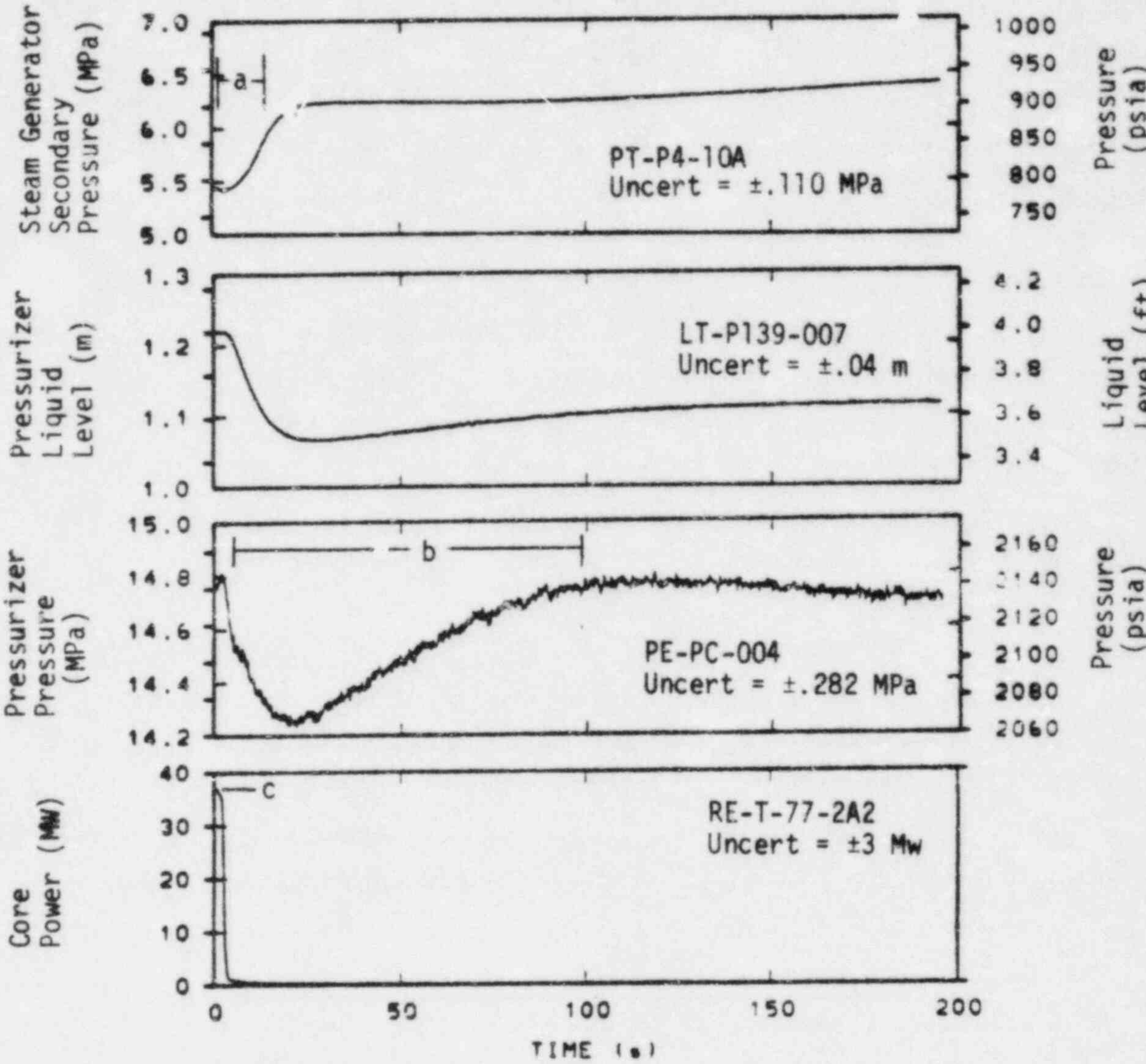
Pressure continued to increase until the high pressure scram setpoint was reached at 21.8 s. Average PCS temperature continued to rise for 4.7 s after the scram signal due to the stored energy of the fuel. Because the pressurizer spray was on, pressure in the pressurizer started to drop as soon as the insurge started to decrease (4.5 s before the insurge into the pressurizer ended).

The outsurge from the pressurizer following scram was as rapid as the insurge had been. This was caused by the drop in PCS average temperature following scram and the opening of the main steam flow control valve (MSCV). The drop in PCS average temperature resulted in the shrinking of the fluid. The main steam flow control valve opened on high secondary pressure 0.4 s after scram. The MSCV cycled once more later in the transient to relieve secondary pressure.

Following the rapid outsurge caused by scram and the steam valve opening, PCS average temperature started a gradual increase as decay heat addition exceeded SG heat removal. This rise in average temperature caused a gradual insurge into a pressurizer with the attending gradual rise in PCS pressure. After 200 s, control of the plant was returned to the operators for recovery to a hot standby condition.

2.2 Anticipated Transient Experiment L6-2

Experiment L6-2 was initiated by tripping of the power to both primary coolant pumps, allowing them to coastdown, simulating a loss of flow in a large PWR. Two seconds after the pumps were tripped the reactor was scrammed automatically upon receipt of a low-PCS flow rate signal. Upon scram, both feedwater and steam control valves started to close. Feedwater flow was zero within 2 s and steam flow was zero 11.4 s after the scram signal was sent. The pumps had coasted down by 50 s. Four data channel plots which characterize plant performance during the experiment are shown in Figure 2.



Power tripped to primary coolant pumps 0. s

a. MSFCV closes 1.8 - 13.4 s

b. Pressurizer backup heaters on 6 - 97.2 s

c. Reactor scrammed 2.0 s

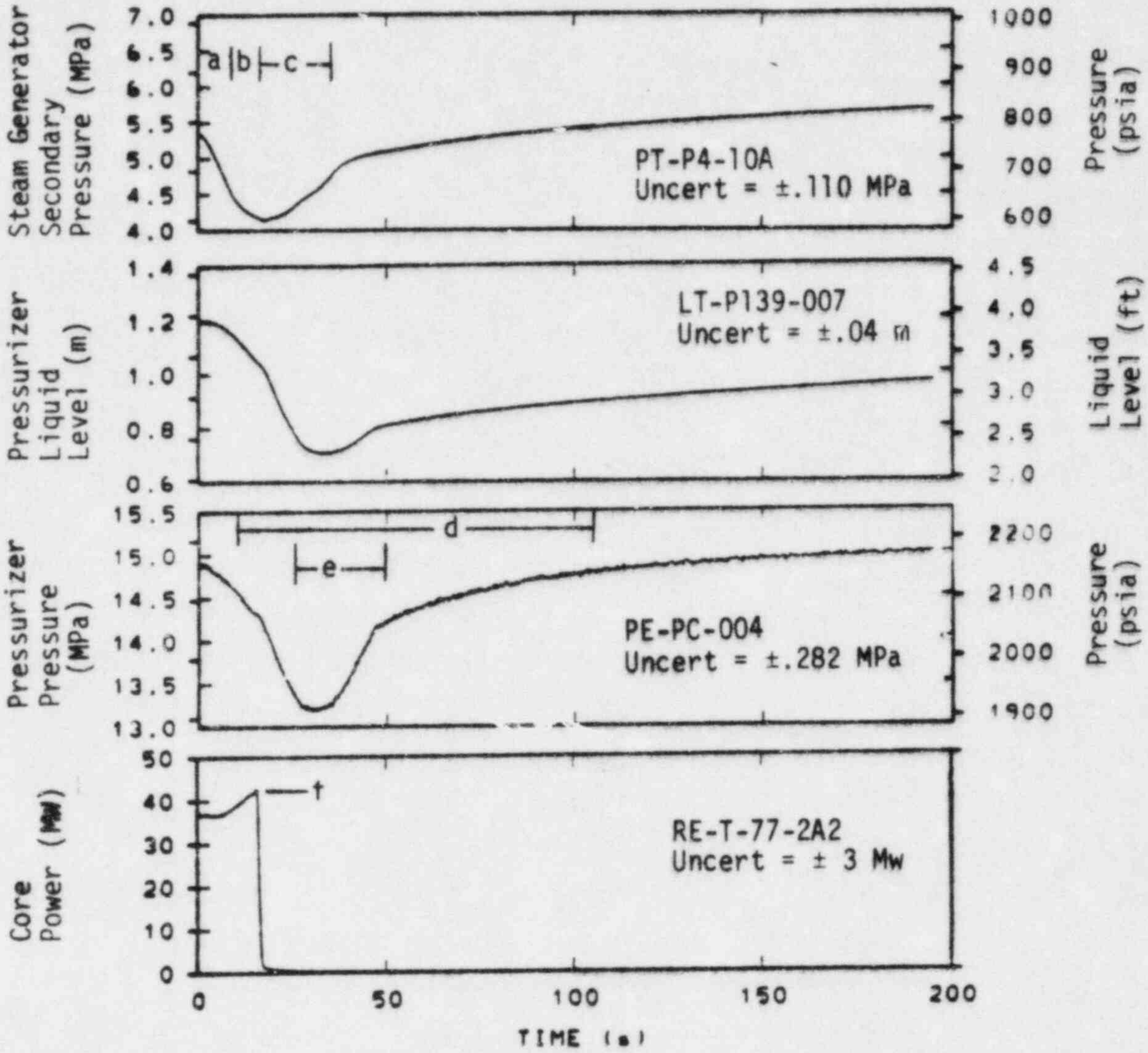
Figure 2. LOFT data for loss of forced PCS flow Experiment L6-2.

As soon as the main steam control valve started to close, pressure in the SG secondary started to rise (Figure 2). From 0 to 29.0 s heat transfer across the SG tubes exceeded core heat transfer to the fluid and PCS average temperature dropped. A good indication of PCS average temperature is pressurizer liquid level (Figure 2); insurges and outsurges from the pressurizer being a function of PCS heatup/cooldown. The initial rapid drop in average PCS temperature caused an outsurge from the pressurizer, with the attending pressure drop shown in Figure 2. The pressurizer backup heaters came on automatically on low pressure at 6.0 s and remained on for 91.2 s.

After 29 s core heat addition exceeded SG heat removal and PCS average temperature started to increase, causing an insurge into the pressurizer. However, there was an increase in pressurizer pressure only during part of this insurge. When the pressurizer pressure increased enough to shut off the backup heaters, pressure in the pressurizer started to decrease gradually. The decrease in pressure is thought to be due to condensation occurring at the liquid-vapor interface in the pressurizer and on the walls of the pressurizer.

2.3 Anticipated Transient Experiment L6-3

Experiment L6-3 was initiated by opening the main steam control valve to its full open position while the plant was at power, simulating an excessive load increase in a large PWR. Plant parameters which characterize the system response during the experiment are shown in Figure 3. As soon as the steam control valve started opening steam generator secondary pressure started to decrease rapidly. Decreasing SG pressure and temperature in turn caused increased SG heat transfer and cooldown of the PCS. The cooldown resulted in an outsurge from the pressurizer and a decrease in pressurizer pressure as well as a reactor power increase. The reactor scrammed automatically on a low PCS pressure signal at 15.6 s. The decrease in core power after scram increased the rate of PCS cooldown. After scram, pressurizer pressure decreased to the high pressure injection system (HPIS) setpoint and injection started automatically. HPIS flow remained on for 23.6 s.



a. MSFCV opens 0-8.2 s

b. MSFCV completely open 8.2 - 17.8 s

c. MSFCV closes 17.8 - 36.2 s

d. Pressurizer backup heaters on 10.2 - 105.4 s

e. HPIS on 26.4 - 50. s

f. Reactor scrammed 15.6 s

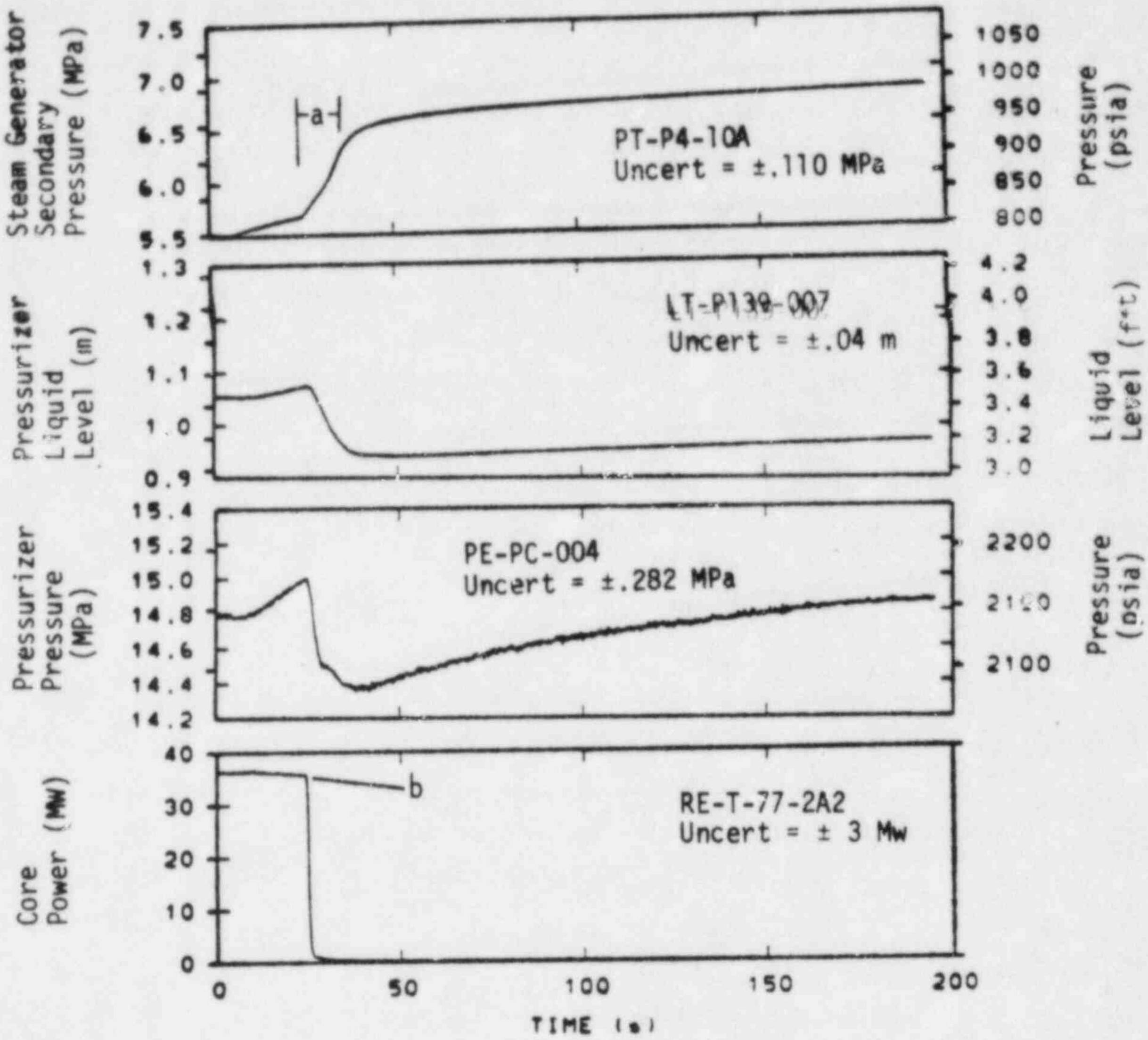
Figure 3. LOFT data for excessive load increase Experiment L6-3.

Upon scram the main steam control valve started to close, taking 18.4 s to close from its full open position. For this experiment the feedwater controller was left in automatic mode, and feedwater flow increased at the start of the experiment in response to increased steam flow. Upon receipt of the scram signal feedwater flow was terminated. Once the steam control valve started to close at 17.8 s (2.2 s delay after scram), SG secondary pressure started to rise. Even though secondary pressure started to rise at 17.8 s the PCS continued to cool down for another 19.7 s before core power exceeded the SG heat removal. After HPIS flow terminated, the transient was characterized by a gradual PCS heatup and rise in pressurizer liquid level and pressure. By 200 s PCS pressure was nearly equal to the initial value.

2.4 Anticipated Transient Experiment L6-5

The L6-5 experiment was initiated by tripping power to the main feedwater pump. By 2 s feedwater flow was zero. No auxiliary feedwater was injected during the experiment. System parameters characterizing system response during the experiment are shown in Figure 4. As soon as feedwater was lost, pressure and temperature in the SG secondary started to rise (Figure 4). The reactor was scrammed at 23.8 s on low SG liquid level. Upon receipt of the scram signal, the main steam control valve started to close, and was completely closed within 11.6 s. As soon as the main steam control valve started to close the SG pressure rise became more rapid.

Heatup of the PCS fluid and the resulting pressurizer insurge occurred until the time of scram. Immediately following scram SG heat transfer exceeded core heat transfer and average PCS temperature dropped, resulting in an outsurge from the pressurizer. By 26 s after scram core decay heat level exceeded SG heat transfer and PCS average temperature started to rise again resulting in a pressurizer insurge and pressure rise.



Feedwater pump tripped 0. s

a. MSFCV closes 25.0 - 35.4 s

b. Reactor scrammed 23.8 s

Figure 4. LOFT data for loss of feedwater Experiment L6-5.

3. RESULTS OF COMPUTER ANALYSES

The ability to calculate primary system pressure during these anticipated transients is a good indicator of the adequacy of the code and input model because pressurizer pressure (PCS pressure) is a function of insurges, outsurges, condensation due to pressurizer spray, action of pressurizer heaters, etc. The insurges and outsurges, in turn, are determined by primary to secondary heat transfer and core heat transfer. The correct calculation of these two heat transfer rates requires a correct calculation of:

1. Core power changes due to moderator and fuel temperature feedback

2. Time of scram

3. Decay heat levels determined by power history

4. Fuel stored energy

5. PCS flowrate

6. Secondary pressure and temperature, which depend on a correct calculation of feedwater flow rate (feedwater controller in automatic mode), steam flow rate (steam valve open/closes to initiate two experiments), and initial SG secondary fluid inventory.

This list is not comprehensive, but gives some indication of the complexity involved in the code calculations.

The posttest analyses documented in this report were performed with the RELAP5/MOD1[3] computer code.[a] Scope and schedule limitations

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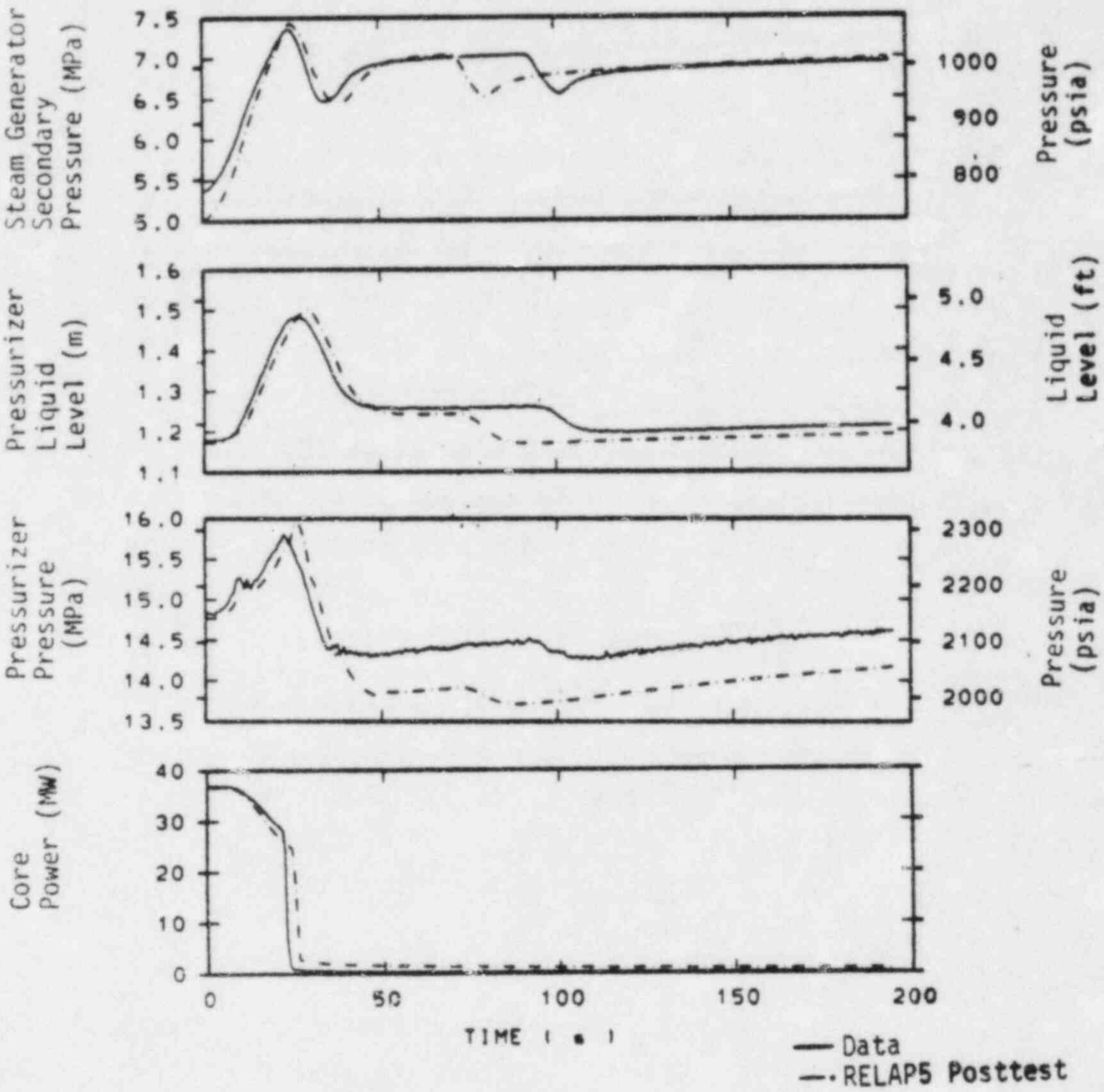
a. This analysis was performed using RELAP5/MOD1 Cycle 13, a production version of the RELAP5/MOD1 code which is filed under Idaho National Engineering Laboratory Computer Code Configuration Management Archival Number F00341.

required the use of only one computer code for posttest calculations. The RELAP5/MOD1 computer code is based on a one-dimensional, two fluid, non-equilibrium hydrodynamic model. The major simplification of the two-phase flow physics assumes the least massive phase to be at saturation, which would be the case if that phase were just appearing or disappearing from a pure phase condition. This assumption allows the use of one energy equation. The energy equation contains a source term which couples the hydrodynamic model to the heat structure conduction model by a convective heat transfer formulation. The code contains special process models for choking, abrupt area changes, branching, pumps, accumulators, valves, annuli, and steam separators.

RELAP5 uses a one-dimensional finite difference or discrete formulation for calculating both fluid flow and heat conduction. In the code, fluid paths are divided into a series of control volumes connected by junctions. Mass and energy conservation relations are used over the control volumes. The vector quantities such as liquid and vapor velocity are evaluated at the junctions or boundaries of the control volumes. This discrete model is used to build up flow paths representative of the physical flow paths. Metal masses are modeled as heat conductors which are divided axially into heat structures according to the hydraulic nodalization. Heat structures are then radially subdivided into mesh intervals according to the resolution required for the solution. The model uses a generalized one dimensional conduction solution. The boundaries of the conductors can be adiabatic or coupled through convective heat transfer correlations to control volumes.

3.1 Experiment L6-1 Posttest Calculation

RELAP5 calculated parameters for Experiment L6-1 (loss of steam load) are compared to measured data in Figure 5. There is an initial offset between the measured steam generator pressure and the RELAP5 calculation. Two factors contributing to the offset are the heat transfer correlations and the discrete volume modeling requirement. The RELAP5 heat transfer correlations are developed for flow in channels, while the flow in the



a. MSFCV closes 0 - 11.6 s

b. MSFCV cycles 22.2 - 40.6 s

c. MSFCV cycles 91.2 - 104.4 s

d. Pressurizer spray on 9.1 - 30.4 s

e. Pressurizer backup heaters on 32.5 - 415.4 s

f. Reactor scrammed 21.8 s

Figure 5. LOFT data and posttest calculation for loss of load Experiment L6-1.

secondary more closely resembles flow perpendicular to the tube bundle. The heat transfer on the secondary is thus higher than calculated and therefore the secondary pressure is also higher than calculated.

In RELAP5, the energy transferred to or from the fluid is transferred simultaneously throughout the control volume. That is, the state of the fluid in the volume is calculated as if all the energy transfer occurred at the volume entrance. Therefore, the primary fluid temperature in the steam generator tubes is, on the average, calculated to be less than physically occurs, resulting in a lower corresponding secondary temperature. The inaccuracy caused by this assumption decreases as the number of volumes increases. Therefore, it is recommended that a perpendicular flow bundle heat transfer model be incorporated in RELAP5 or a realistic method of enhancing the present models be incorporated. It is also recommended a study be performed that determines the optimum number of volumes in the primary tubes.

In the experiment the main steam control valve cycled twice to automatically relieve steam generator secondary pressure (Figure 5). The first valve cycle was quite well calculated by the code, but the second cycle was calculated to occur too early.

The next curve in Figure 5 shows pressurizer liquid level. As the main steam control valve closed to initiate the transient, pressure and temperature in the steam generator secondary started to rise. This temperature rise reduced the temperature difference across the steam generator tubes and hence heat transfer from primary to secondary. This caused an increase in the average primary system temperature and was attended by a rapid insurge into the pressurizer. As soon as scram occurred, core power dropped below the steam generator heat transfer rate and the primary coolant system started to cool down. This cooldown is evidenced by an outsurge from the pressurizer. A later outsurge was associated with the automatic cycling of the main steam control valve. In general, the RELAP5 calculation of pressurizer liquid level was quite good.

The major difference between calculation and data was due to the early cycling of the steam control valve in the RELAP5 calculation. Other differences between the data and the calculated liquid level are due to differences in scram time and hence net energy of the primary coolant system.

The third curve of Figure 5 presents pressurizer pressure. There is an initial offset between the data and the calculation due to a difference between the published initial conditions (Reference 5), which were used in the calculation, and the data channel plotted in this report. The rise in pressurizer pressure attending the initial insurge into the pressurizer was momentarily halted in both the experiment and calculation when the pressurizer spray automatically started at ~10 s. This pressure decrease upon initiation of pressurizer spray is felt to be the product of two conditions: (a) the presence of water in the spray line which had cooled due to environmental losses in the spray line, and (b) the presence of superheated steam in the vapor dome of the pressurizer at the time of spray initiation. As the sudden insurge into the pressurizer occurs at the start of the transient the column of water in the pressurizer acts as a piston, compressing the vapor in the top of the pressurizer. This rapid compression can cause the vapor to superheat. The introduction of subcooled spray into a superheated environment would cause the vapor to desuperheat, with the attending pressure decrease. Due to the slow response of the temperature probe in the pressurizer vapor space the amount of superheat during this initial compression cannot be confirmed. In the calculation the depressurization during initial spray operation could be achieved only by introducing water into the spray line volume which was 100 K colder than the source of the spray water, the intact loop cold leg.

The calculated peak pressurizer pressure was greater than the measured values partially because of initial condition offset and partially because the calculated time of scram was 2.8 s later than in the data. These extra 2.8 s of full power operation resulted in a net system energy gain in the calculation compared to the data (evidenced in the longer insurge into the pressurizer). Following scram, average PCS temperature started to decrease with a resulting pressurizer outsurge and expansion of the pressurizer vapor space causing a decrease in pressurizer pressure. In both the experiment and the calculation, cycling and backup heaters in the pressurizer came on automatically and raised the pressurizer pressure.

Even though pressurizer liquid level was quite well calculated by the code until ~75 s, the code calculated a low pressurizer pressure from ~40 s on. This difference is believed to be caused by deficiencies in calculating subcooled voiding and environmental losses. During this time period the RELAP5 heat transfer mode on the outside of the pressurizer heaters is subcooled nucleate boiling. In this mode RELAP5 does not model the void creation in the hydrodynamic volume containing the heater and the possible void collapse in upper control volumes. This leads to an underprediction of total void and consequently vapor pressure in the pressurizer.

The environmental heat losses from the outside pressurizer walls are too great in the RELAP5 calculation. The steady state heat losses of the RELAP5 model in the pressurizer were 16 kW in contrast to 6 kW given as a best estimate value for pressurizer heat losses in Reference 11, leading to a low calculated pressurizer pressure. The difference between the data and the calculation in the time of later pressure drops is due to differences in time when the main steam control valve cycled, causing a cooldown of the PCS.

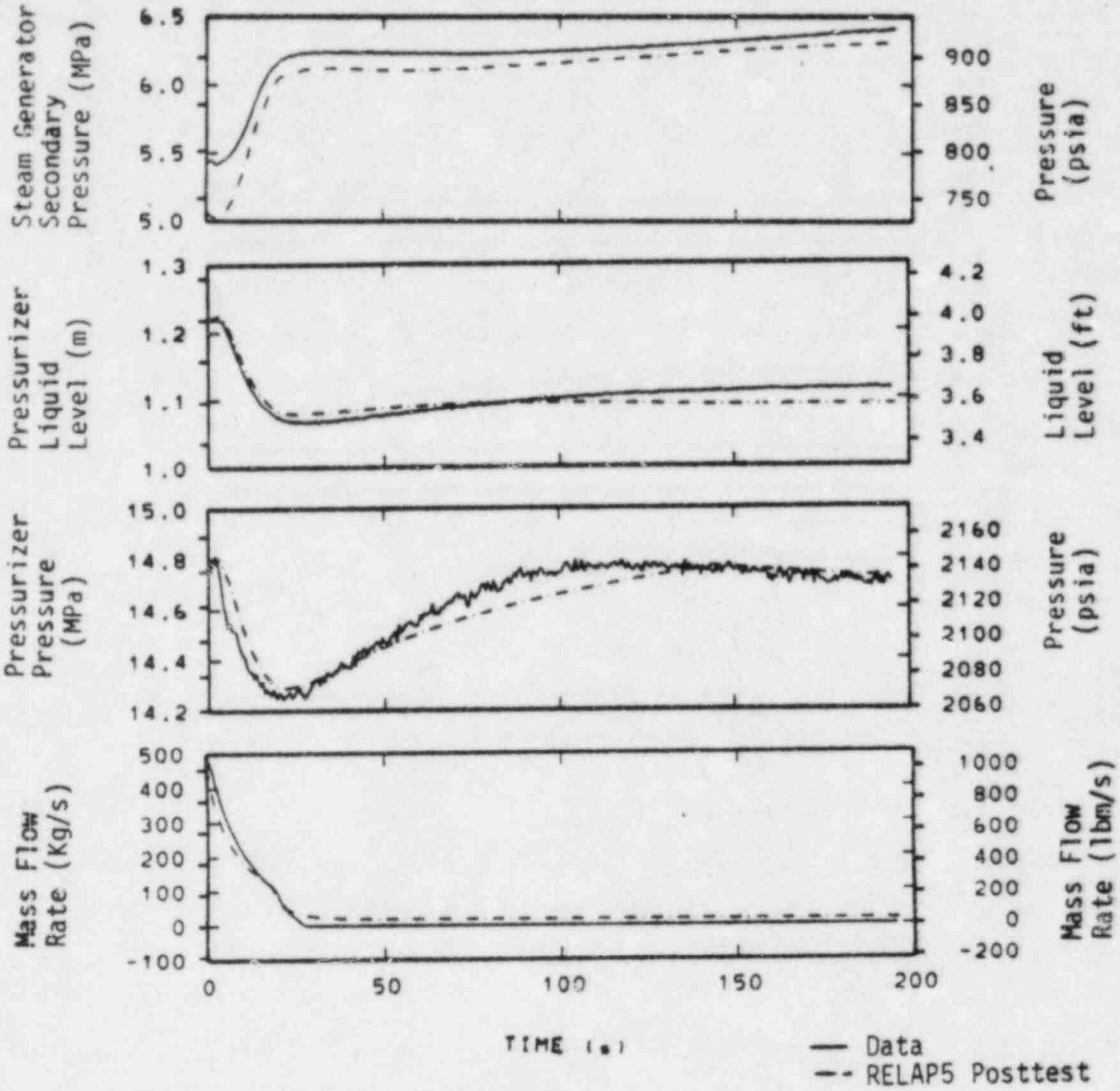
The fourth set of curves in Figure 5 shows core power. Both the calculated and measured core power decrease at the start of the transient due to the negative reactivity caused by the heatup of PCS coolant. The late time of scram on high hot leg pressure in the calculation was a result of the initial pressure in the PCS being low, and differences in calculated pressurizer pressure during the initial insurge. The data curve of core power in Figure 5 is only ranged for normal power operation. After scram the data are not qualified and should not be used for comparison purposes.

3.2 Experiment L6-2 Posttest Calculation

Experiment L6-2 (loss of forced PCS flow) measured data are compared to selected calculated values in Figure 6. The top curve shows steam generator secondary pressure. As soon as the main steam control valve started to close, steam generator secondary pressure and temperature started to rise. The initial offset between the data and posttest calculation curves is due to the same factors defined in Experiment L6-1. An estimated leakage rate of 0.2 kg/s through the steam valve was modeled in the calculation.

Even though temperature in the steam generator secondary rose at the start of the transient (reducing primary to secondary heat transfer) heat transfer across the steam generator tubes exceeded decay heat production (reactor scram at 2.0 s) until ~20 s. This resulted in a net cooldown of the PCS fluid and an outsurge from the pressurizer (see pressurizer level, Figure 6). After 20 s decay heat production exceeded steam generator heat transfer and a gradual heatup of the PCS fluid occurred with an attending gradual insurge into the pressurizer. During this experiment there were no charging or letdown flows thus pressurizer liquid level was a good indicator of average PCS temperature.

Both calculated and measured pressurizer pressure are shown in the third set of plots in Figure 6. It is interesting to note the effect of the pressurizer backup heaters during the experiment. Even though a slow insurge occurred after 20 s, pressurizer pressure rose only as long as the backup heaters were on. When the backup heaters turned off automatically at 97.2 s pressurizer pressure started to decrease even though an insurge (and compression of the pressurizer vapor space) was occurring. This turnaround of pressure is also seen in the RELAP5 calculation, even though it occurs later in the transient. The decrease in pressure is thought to be due to the condensation occurring at the liquid-vapor interface in the pressurizer, and on the walls of the pressurizer. The difference in pressurizer pressure between data and calculation from 50 to 130 s is believed to be due mainly to the difference in insurge rate during this time period.



Power tripped to primary coolant pumps 0.s

a. MSFCV closes 1.8 - 13.4 s

b. Pressurizer backup heaters on 6 - 97.2 s

c. Reactor scrammed 2.0 s

Figure 6. LOFT data and posttest calculation for loss of forced PCS flow Experiment L6-2.

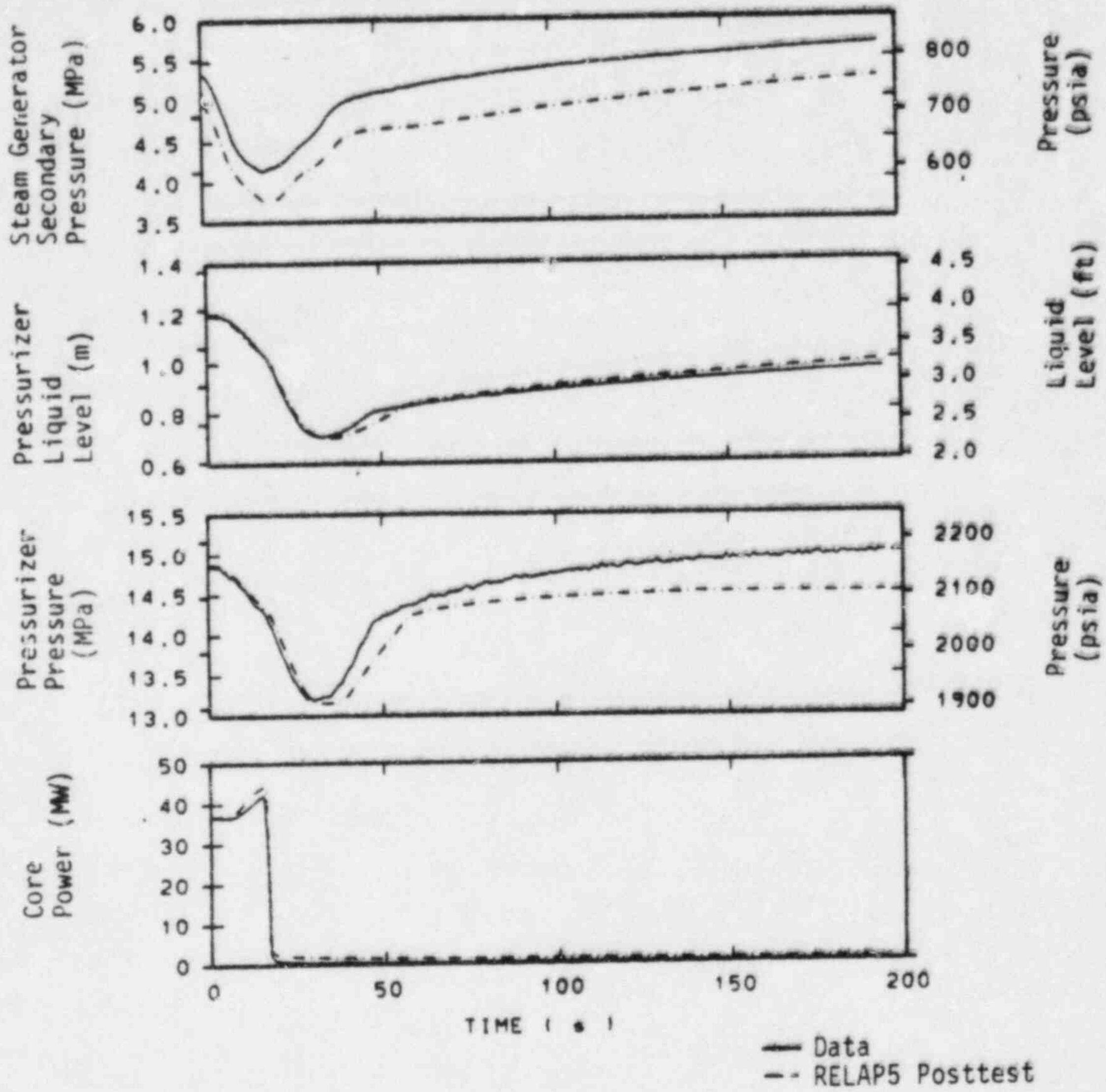
The last set of curves shown in Figure 6 show predicted and measured PCS flowrate. In this experiment natural circulation of PCS fluid was established after the primary coolant pumps coasted down at about 30 s. Since the data channel shown in that plot is ranged only for high flow rates, the curve is valid for comparison purposes only up to 30 s, the start of natural circulation.

Assuming steady state conditions, PCS natural circulation flow rate can be easily calculated from the known core decay power and temperature difference across the core. The flow rate calculated with this method is in agreement with the results obtained by RELAP5.

3.3 Experiment L6-3 Posttest Calculation

Experiment L6-3 (excessive load increase) was initiated by opening the main steam control valve to its full open position at a rate of 5%/s. Figure 7 shows comparison between measured data and RELAP5 calculated values for four parameters of interest in this transient. The reasons for the initial offset between data and the posttest calculation of steam generator secondary pressure are the same as discussed earlier for Experiment L6-1. As the steam valve opened, pressure and temperature in the steam generator secondary began to decrease, causing an increase in primary to secondary heat transfer. This, in turn, caused a cooldown of the PCS fluid and a rapid outsurge from the pressurizer. This outsurge and drop in pressurizer pressure were quite well calculated. At 15.6 s the reactor scrammed on low PCS pressure. Upon scram, the main steam control valve started to close, with an attending rise in secondary pressure and a rise in average PCS fluid temperature as evidenced by an insurge into the pressurizer.

The rapid decrease in PCS pressure at the beginning of the transient was compounded by the effect of the scram. Once scram occurred steam generator energy removal far exceeded core energy addition to the PCS fluid and the PCS cooldown rate accelerated. This additional drop in PCS pressure brought the primary system pressure low enough for high pressure



a. MSFCV opens 0-8.2 s

b. MSFCV completely open 8.2 - 17.8 s

c. MSFCV closes 17.8 - 36.2 s

d. Pressurizer backup heaters on 10.2 - 105.4 s

e. HPIS on 26.4 - 50. s

f. Reactor scrammed 15.6 s

Figure 7. LOFT data and posttest calculation for excessive load increase Experiment L6-3.

injection system (HPIS) to start automatically. This injection lasted from 26.4 to 50 s. During HPIS injection the pressurizer level variation is a combined effect due to the increased PCS fluid volume and the change of the primary fluid average temperature.

From 30 to 200 s the calculated average PCS fluid temperature was 6 K lower than measured in the experiment. Thus the good agreement between calculated and measured pressurizer liquid level is accidental. This is at least partly due to a larger amount of injected HPIS water in the calculation.

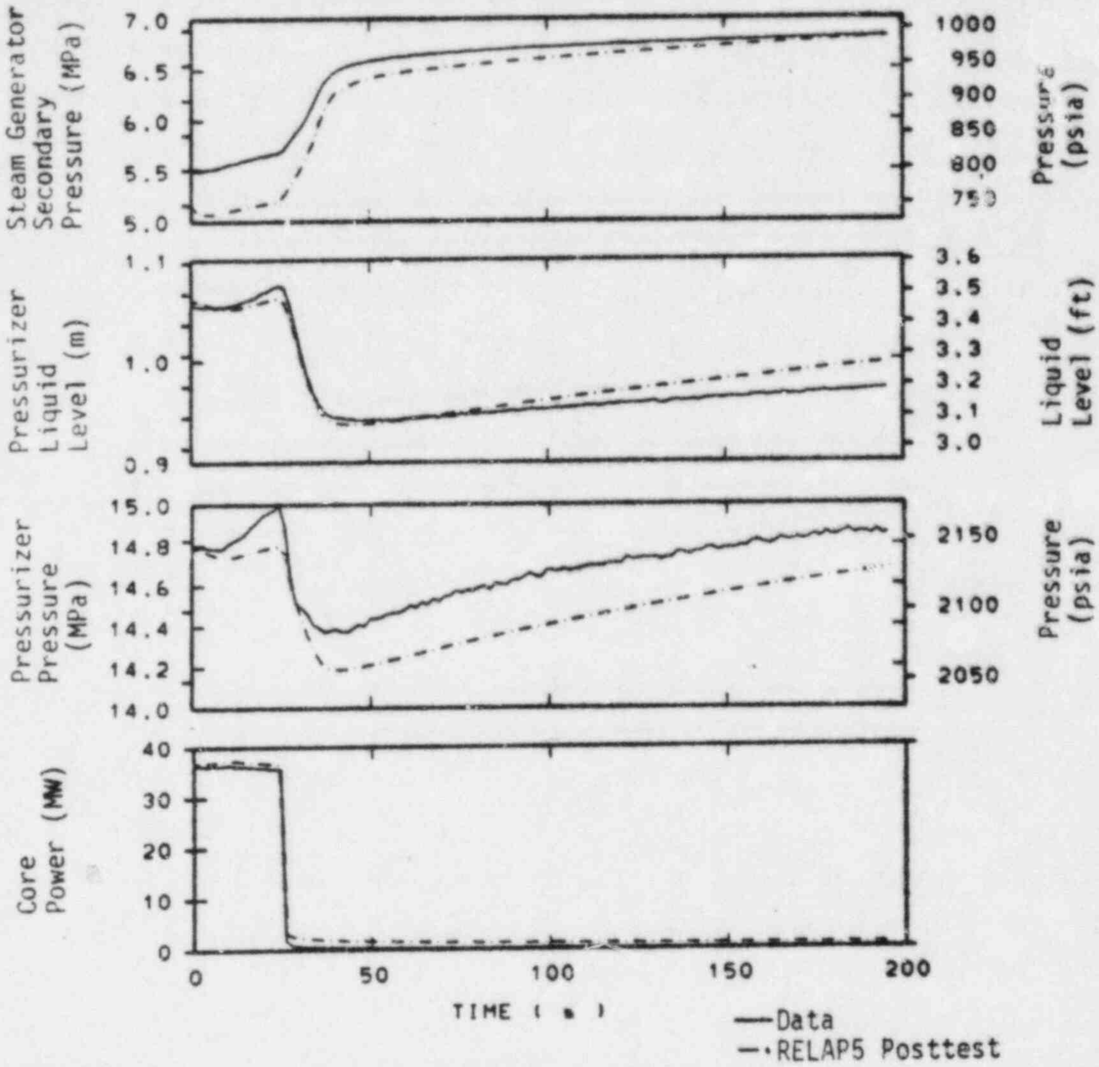
The calculated secondary side pressure is low throughout the transient. The difference between calculated and measured saturation temperature on the secondary side is about 6 K, which is the same as the temperature underprediction on the primary side after 30 s in the transient. In the calculation it was necessary to increase the full open area of the steam valve by a factor of 1.34 in order to achieve the measured steam flow through the valve. Even though this increase resulted in a good agreement with the initial cooldown of the primary system it could have been a cause for the later overcooling of PCS.

The underprediction of pressurizer pressure from 50 s is due at least partially to an overcalculation of environmental heat losses from the pressurizer.

The last set of curves in Figure 7 shows the increase in reactor power at the start of the transient due to the positive reactivity insertion attending the cooldown of the PCS fluid.

3.4 Experiment L6-5 Posttest Calculation

Experiment L6-5 (loss of feedwater) measured data are compared to selected calculated variables in Figure 8. The initial offset of steam generator secondary side pressure is due to factors discussed earlier. Before reactor scram at 23.8 s the calculated steam generator secondary side pressure parallels the data. After scram when the MSCV is closed the



Feedwater pump tripped 0. s

a. MSFCV closes 25.0 - 35.4 s

b. Reactor scrammed 23.8 s

Figure 8. LOFT data and posttest calculation for loss of feedwater Experiment L6-5.

calculated secondary side pressure approaches the data curve. A 0.255 kg/s leak through the valve was modeled in the calculation. After the scram the calculated primary system average fluid temperature increased too rapidly, resulting in an overprediction in the pressurizer liquid level after 60 s in the transient.

The calculated pressurizer pressure was lower than the measured data throughout the transient. In the first part of the transient this is partly due to an underprediction of the liquid level in the pressurizer. The other reason for the initial low pressure is the presence of a small amount of saturated liquid in the pressurizer vapor space at the start of the calculation. During vapor space compression, when the liquid level in the pressurizer rises, heat transfer at the vapor-liquid interface prevents the pressurizer pressure from increasing. After the rapid outsurge from the pressurizer due to the reactor scram, the pressurizer backup and cycling heaters cannot restore the pressure in the calculation as fast as in the experiment for the same reason as discussed in the L6-1 posttest calculation.

Both the calculated and measured reactor power remained nearly constant before the reactor scram. The time of reactor scram was set as an input value for the calculation.

4. CONCLUSIONS AND RECOMMENDATIONS

The anticipated transient experiments described in the report provided much useful information. The phenomena that occurred during these experiments should be similar to phenomena that would occur in similar transients in a commercial PWR. The data measured during these experiments provides a basis for evaluating computer codes.

The RELAP5 calculated results were generally in good agreement with the experimental data. It has been found that much care is required to provide accurate input data, initial conditions, and boundary conditions. Because of the complex controls on the system, a detailed model was required to calculate most of the phenomena. It is recommended that a detailed, well checked model and a best estimate computer code be used to calculate operational transients in order to allow the realistic planning of subsequent operator actions.

The RELAP5 computer code is able to calculate the anticipated transients discussed in this report quite well. Most of the phenomena occurring in the pressurizer were well calculated with the exception of heater operation in subcooled liquid. It is felt RELAP5 should be modified so that voids are produced during subcooled nucleate boiling. A method of achieving more accurate steam generator initial conditions should be developed by improving the heat transfer modeling across the bundles.

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APPENDIX A

DESCRIPTION OF THE LOFT FACILITY

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DESCRIPTION OF THE LOFT FACILITY

The LOFT facility has been designed to simulate the major components and system responses of a commercial PWR during a LOCA or anticipated transient. The experimental assembly comprises five major subsystems which have been instrumented such that system variables can be measured and recorded during each experiment. The subsystems include: the reactor vessel, the intact loop, the broken loop, the blowdown suppression system, and the emergency core cooling system (ECCS). The LOFT major components are shown in Figure A-1, and the LOFT piping configuration is shown in Figure A-2.

The LOFT reactor vessel, which simulates the reactor vessel of a commercial PWR, has an annular downcomer, a lower plenum, lower core support plates, a nuclear core, and an upper plenum. The downcomer is connected to the cold legs of the intact and broken loops and contains two instrument stalks. The upper plenum is connected to the hot legs of the intact and broken loops. The core contains 1300 unpressurized nuclear fuel rods arranged in five square (15 x 15 assemblies) and four triangular (corner) fuel modules, shown in Figure A-3. The center assembly is highly instrumented. Two of the corner and one of the square fuel modules are not instrumented. The fuel rods have an active heated length of 1.67 m and an outside diameter of 10.72 mm.

The fuel consists of UO_2 sintered pellets with an average enrichment of 4.0 wt% fissile uranium (^{235}U) and with a density that is 93% of theoretical density. Fuel pellet diameter and length are 9.29 and 15.24 mm, respectively. Both ends of the pellets are dished with the total dish volume equal to 2% of the pellet volume. Cladding material is Zircaloy-4. Cladding inside and outside diameters are 9.48 and 10.72 mm, respectively.

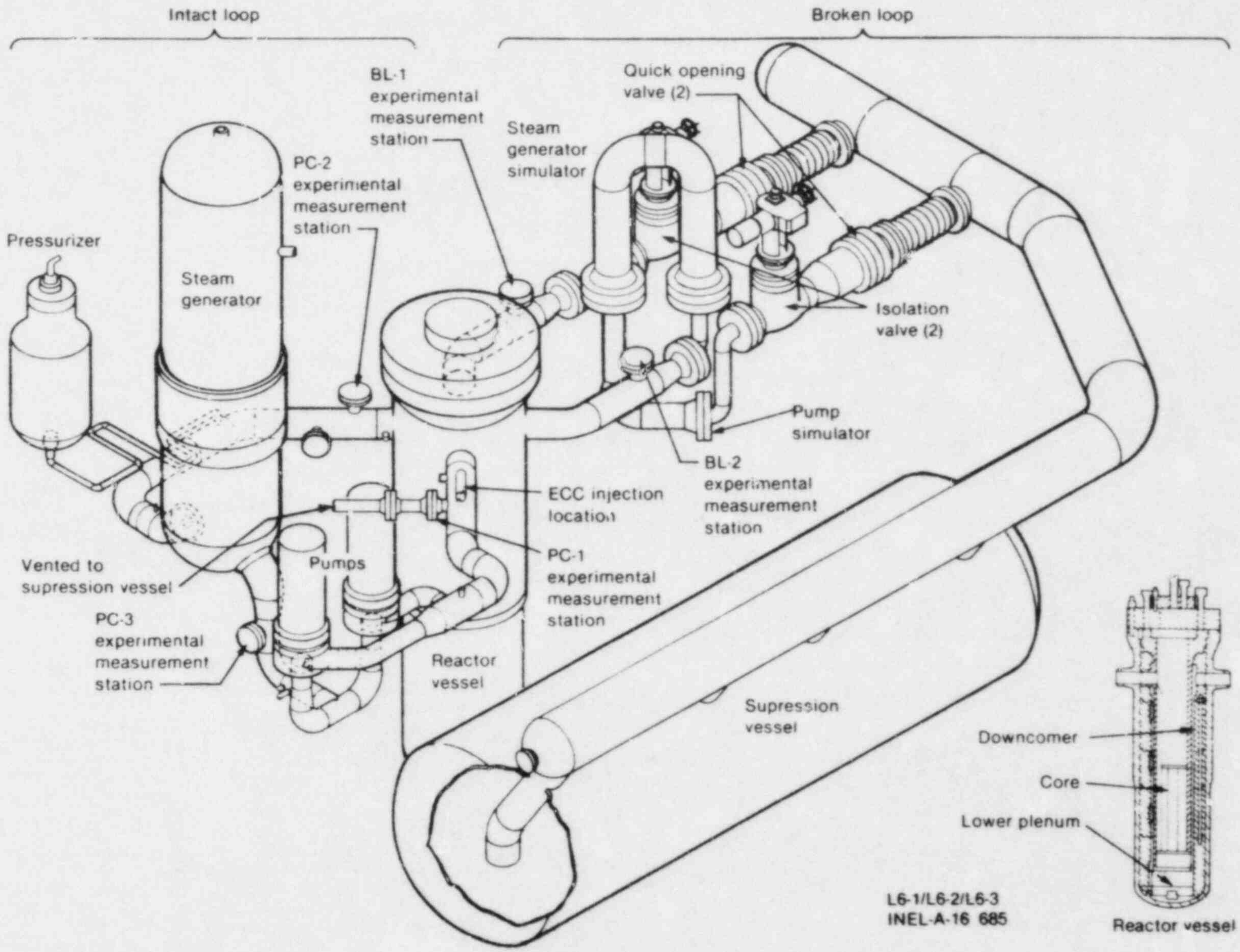

Figure A-1. LOFT major components.

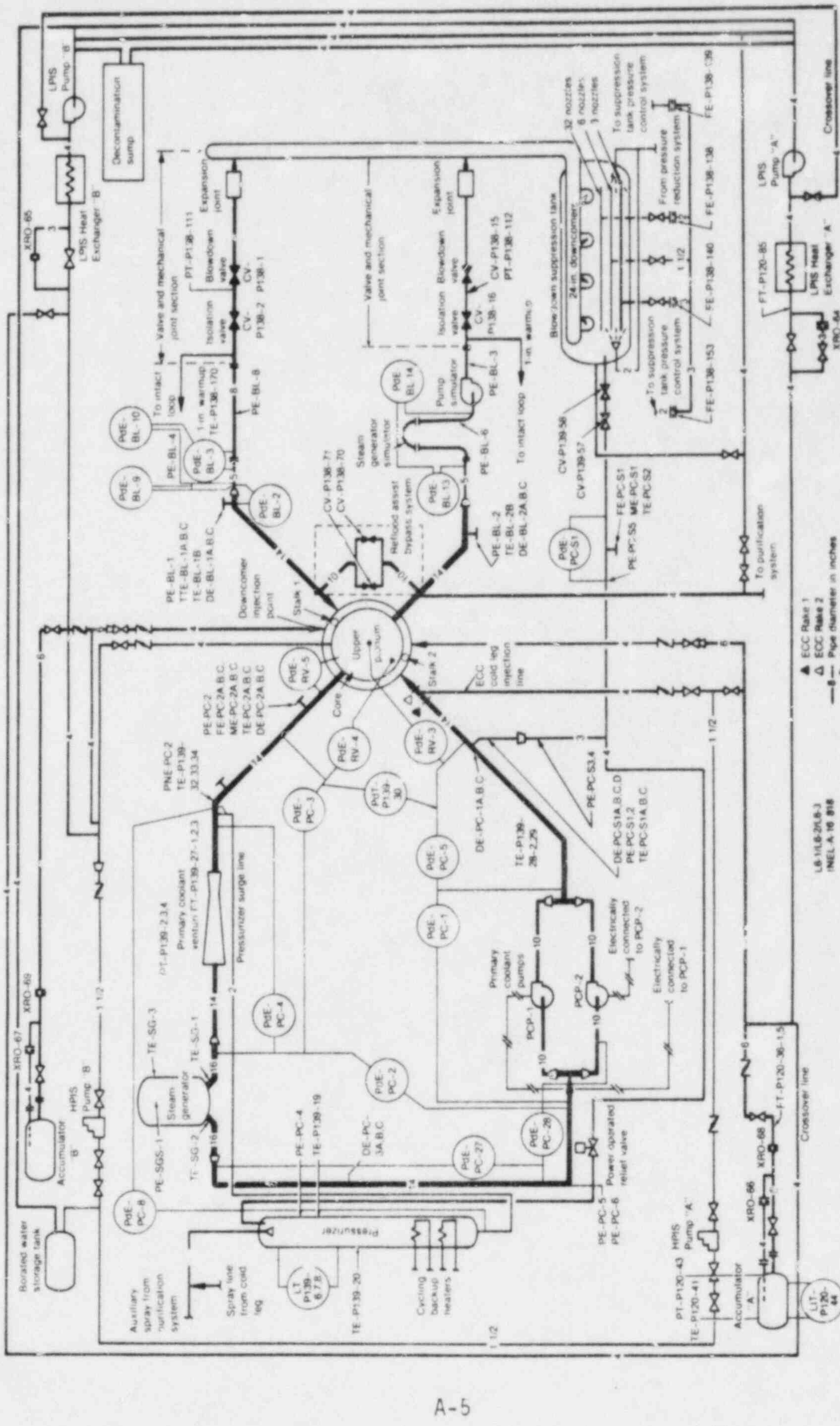

Figure A-2. LOFT piping configuration.

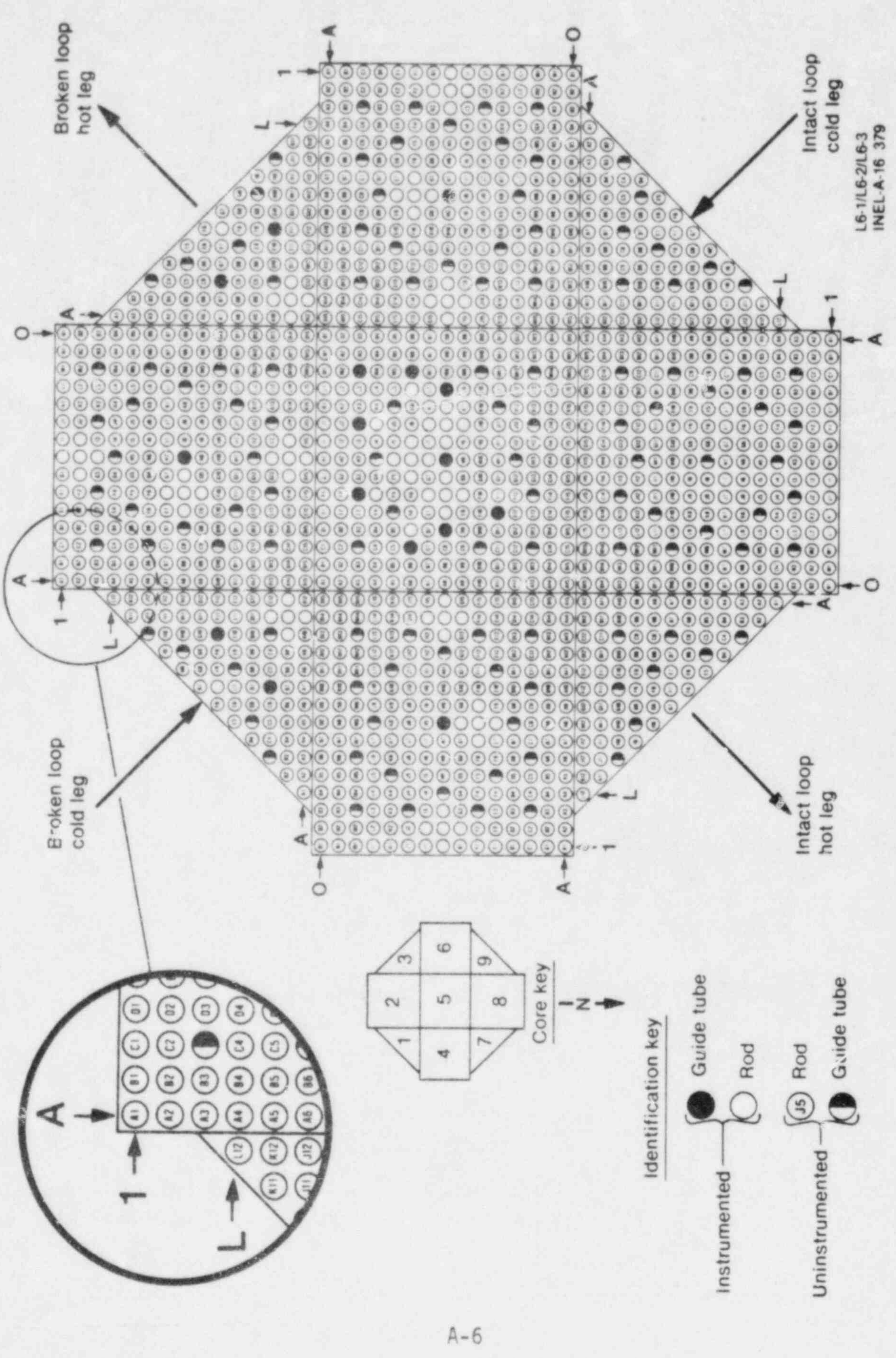

Figure A-3. LOFT core map showing position designations.

The intact loop simulates the three unbroken loops of a commercial, four-loop PWR and contains a steam generator, two primary coolant pumps in parallel, a pressurizer, a venturi flowmeter, and connecting piping.

The broken loop consists of a hot leg and a cold leg that are connected to the reactor vessel and the blowdown suppression tank (BST) header. Each leg consists of a break plane orifice, a quick-opening blowdown valve (QOBV), a recirculation line, an isolation valve, and steam generator and pump hydraulic resistance simulators (in the hot leg) with connecting piping. The recirculation lines establish a small flow from the broken loop to the intact loop to maintain approximately equal loop temperatures. The QOBVs remained closed during the anticipated transient experiments.

The LOFT ECCS simulates the ECCS of a commercial PWR. Of the four anticipated transient experiments discussed in this document, L6-3 (excessive load increase) was the only one in which the ECCS injected water into the PCS. This injection from the high-pressure injection system (HPIS) was initiated automatically upon the receipt of a low-pressure signal.

The LOFT system is designed to scale significant features of a four-loop commercial PWR and to reproducibly simulate typical system transient responses to LOCAs and anticipated transients from non-LOCA origins. The scaling rationale is based on maintaining a power-to-volume ratio in LOFT approximately equal to that in a four-loop commercial plant. The LOFT design preserves the dominant PWR features and, together with the operational range of the plant, ensures that most significant phenomena associated with off-normal condition initiating events are preserved.

There are, however, several hardware differences as well as automatic control differences between LOFT and commercial PWRs which should be kept in mind when examining the LOFT anticipated transient experiments results. Even though anticipated transients are basically generic in character, particular system parameters as a function of time will differ from plant to plant due to differences in scram setpoints, power/volume ratios, valve opening and closing times, automatic control logic, etc. These differences necessitate the use o. the computer codes, which account for the differences through plant model input. Thus, the thermal-hydraulic computer codes are the connecting link between LOFT data and commercial PWR calculations.

APPENDIX B

INITIAL CONDITIONS AND SEQUENCE OF EVENTS

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INITIAL CONDITIONS AND SEQUENCE OF EVENTS

Initial conditions measured at the initiation of operational transient experiments L6-1, L6-2, L6-3 and L6-5 are listed in Table B-1. Major events that occurred during Experiments L6-1, L6-2, L6-3 and L6-5 are sequentially listed in Tables B-2, B-3, B-4 and B-5, respectively.

Table B-1. Measured initial conditions for Experiments L6-1, L6-2, L6-3 and L6-5

Parameter	Experiment L6-1	Experiment L6-2	Experiment L6-3	Experiment L6-5
Primary Coolant System				
Mass flow (kg/s)	478.5 ± 6.3	465.9 ± 6.3	479.3 ± 6.3	479.4 ± 6.3
Hot leg pressure (MPa)	14.78 ± 0.11	14.78 ± 0.11	14.87 ± 0.11	14.79 ± 0.25
Cold leg temperature (K)	552.8 ± 1.2	553.5 ± 1.2	552.6 ± 1.2	554.7 ± 3
Hot leg temperature (K)	567.5 ± 1.8	568.5 ± 2.8	567.3 ± 1.8	568.2 ± 0.5
Reactor Vessel				
Power level (MW)	36.9 ± 0.9	37.2 ± 0.9	36.9 ± 0.9	36.7 ± 1
Maximum linear heat generation rate (kW/m)	39.2 ± 3.0	39.7 ± 3.0	39.2 ± 3.0	39.6 ± 2
Control rod position (above full-in position) (m)	1.371 ± 0.010	1.371 ± 0.010	1.371 ± 0.010	1.372 ± 0.01
Pressurizer				
Steam volume (m^3)	0.30 ± 0.04	0.28 ± 0.04	0.29 ± 0.04	0.36 ± 0.02
Liquid volume (m^3)	0.63 ± 0.04	0.65 ± 0.04	0.64 ± 0.04	0.57 ± 0.02
Liquid temperature (K)	614.3 ± 0.8	614.5 ± 0.8	615.3 ± 0.8	614.1 ± 1.3
Pressure (MPa)	14.78 ± 0.20	14.78 ± 0.20	14.87 ± 0.20	14.79 ± 0.25
Liquid level (m)	1.18 ± 0.07	1.22 ± 0.07	1.20 ± 0.07	1.06 ± 0.04
Broken Loop				
Cold leg temperature near reactor vessel (K)	551.1 ± 2.5	551.7 ± 2.5	550.7 ± 2.5	554.2 ± 2.5
Hot leg temperature near reactor vessel (K)	557.0 ± 2.5	557.8 ± 2.5	556.9 ± 2.5	558.8 ± 2.5
Steam Generator Secondary Side				
Liquid level (m)[a]	0.233 ± 0.034	0.189 ± 0.030	0.170 ± 0.030	0.27 ± .06
Liquid temperature (K)	541.7 ± 0.8	541.1 ± 0.8	541.2 ± 0.8	554.0 ± 0.2
Pressure (MPa)	5.37 ± 0.06	5.44 ± 0.06	5.34 ± 0.06	5.58 ± 0.012
Mass flow (kg/s)	20.1 ± 0.6	20.[illegible] ± 0.6	20.7 ± 0.6	20.6 ± 0.4

a. The liquid level is defined as 0.0 at 2.95 mm above the top of the tube sheet.

Table B-2. Sequence of events for Experiment L6-1

Event	Time After Experiment Initiation (s)
MSCV closing initiated	0
Pressurizer backup heater off	6.1 ± 0.1
Pressurizer spray on	9.1 ± 0.1
MSCV closed	11.6 ± 0.2
Reactor scrammed	21.8 ± 0.2
Maximum PCS pressure reached	22.0 ± 0.2
MSCV opened	22.2 ± 0.2
Pressurizer spray off	30.4 ± 0.1
Pressurizer backup heaters on	32.5 ± 0.1
MSCV closed	40.6 ± 0.2
MSCV opened	91.2 ± 0.2
MSCV closed	104.4 ± 0.2
Code calculations terminated	200.0 ± 0.0

Table B-3. Sequence of events for Experiment L6-2

Event	Time After Experiment Initiation (s)
Primary coolant pumps tripped	0
MSCV closing initiated	1.8 ± 0.2
Reactor scrammed	2.0 ± 0.2
Pressurizer backup heaters on	6.0 ± 0.1
MSCV closed	13.4 ± 0.2
PCP coastdown complete	18.25 ± 0.25
Natural circulation established	23 ± 3
Pressurizer backup heaters off	97.2 ± 0.1
Code calculations terminated	200.0 ± 0.0

Table B-4. Sequence of events for Experiment L6-3

Event	Time after Experiment Initiation (s)
MSCV started to open	0
Feedwater flow increased	1.4 ± 0.2
Pressurizer backup heaters on	10.2 ± 0.1
Maximum reactor power reached during experiment	15.6 ± 0.2
Reactor scrammed	15.6 ± 0.2
Feedwater flow terminated	16.6 ± 0.2
MSCV started to close	17.8 ± 0.2
HPIS Pump A on	26.4 ± 0.2
HPIS Pump B on	26.6 ± 0.2
Minimum PCS pressure reached	26.8 ± 0.2
MSCV closed	36.2 ± 0.2
HPIS Pump A off	48.6 ± 0.2
HPIS Pump B off	50 ± 2
Pressurizer backup heaters off	105.4 ± 0.1
Pressurizer cycling heaters off	154.9 ± 0.1
Code calculations terminated	200.0 ± 0.0

Table B-5. Sequence of events for Experiment L6-5

Event	Time After Experiment Initiation (s)
Experiment initiated	0
Main feedwater pump shut down	0.128 ± 0.1
Reactor scrammed	23.7 ± 0.1
Control rods on bottom	25.8 ± 0.1
Main feedwater isolation valve (CV-P004-73) closed	27.6 ± 0.1
MSCV (CV-P004-10) closed	35.4 ± 0.2
Code calculations terminated	200.0 ± 0.0

APPENDIX C

RELAP5 INPUT MODEL

APPENDIX C

RELAP5 INPUT MODEL

The nodalization used in the RELAP5 calculations was based on the LOFT base input deck documented in Reference C-1. The schematic of this model is shown in Figure C-1. Changes to this nodalization were made on the basis of experience gained in earlier LOFT analyses. The basic differences between the current nodalization and the base model nodalization are as follows:

1. The number of volumes in the pressurizer was increased to nine to provide a better calculation of phenomena in the pressurizer.

2. Pressurizer cycling and backup heaters and pressurizer wall heat structures were included in the model.

3. The reactor vessel filler gap volume was added to the model.

4. Environmental heat losses to the containment were modeled via modeling the metal mass of the reactor vessel, intact and broken loop piping and steam generator secondary side volumes.

5. The feedwater line with its check and control valves was included in the model.

6. The pressurizer spray line was modeled.

The listings of the RELAP5 input decks are contained on microfiche on the report back cover. The input data decks for each transient differed due to experiment-specific boundary conditions and initiating events. The initial conditions used in the RELAP5 analyses match as closely as possible the actual experiment initial conditions.

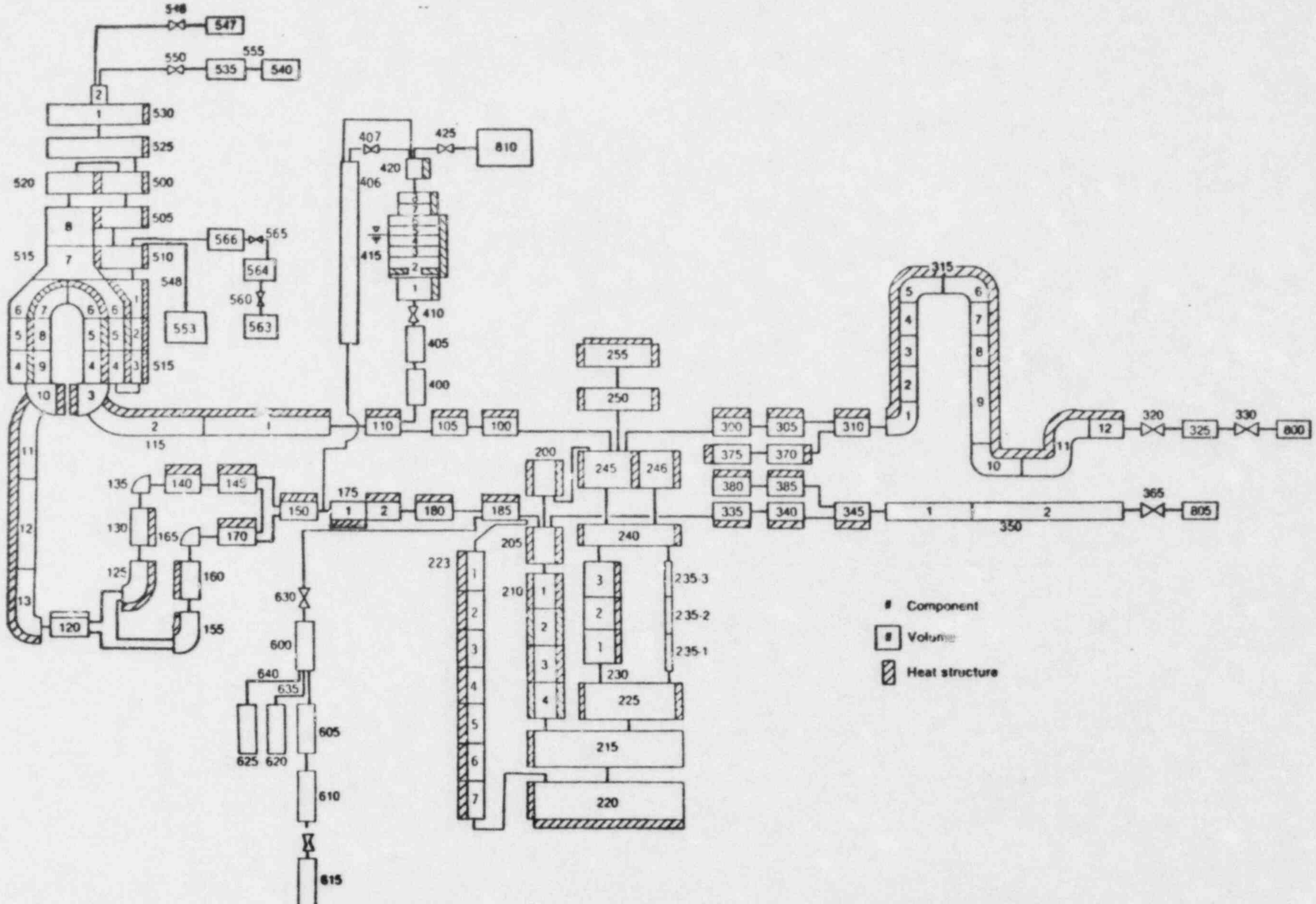

Figure C-1. RELAP5 nodalization schematic.

The power history prior to experiment initiation was input as a power histogram. This gives a realistic calculation of decay heat levels after scram.

REFERENCE

C-1 E. J. Kee, et al., Base Input for LOFT RELAP5 Calculations, EGG-LOFT-5199, July 1980.