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AN EVALUATION OF THE THERMAL-HYDRAULIC AND RADIOLOGICAL EFFECTS OF REMOVING OR BYPASSING THE LOFT STEAM GENERATOR AND PUMP SIMULATORS FOR EXPERIMENT L2-6

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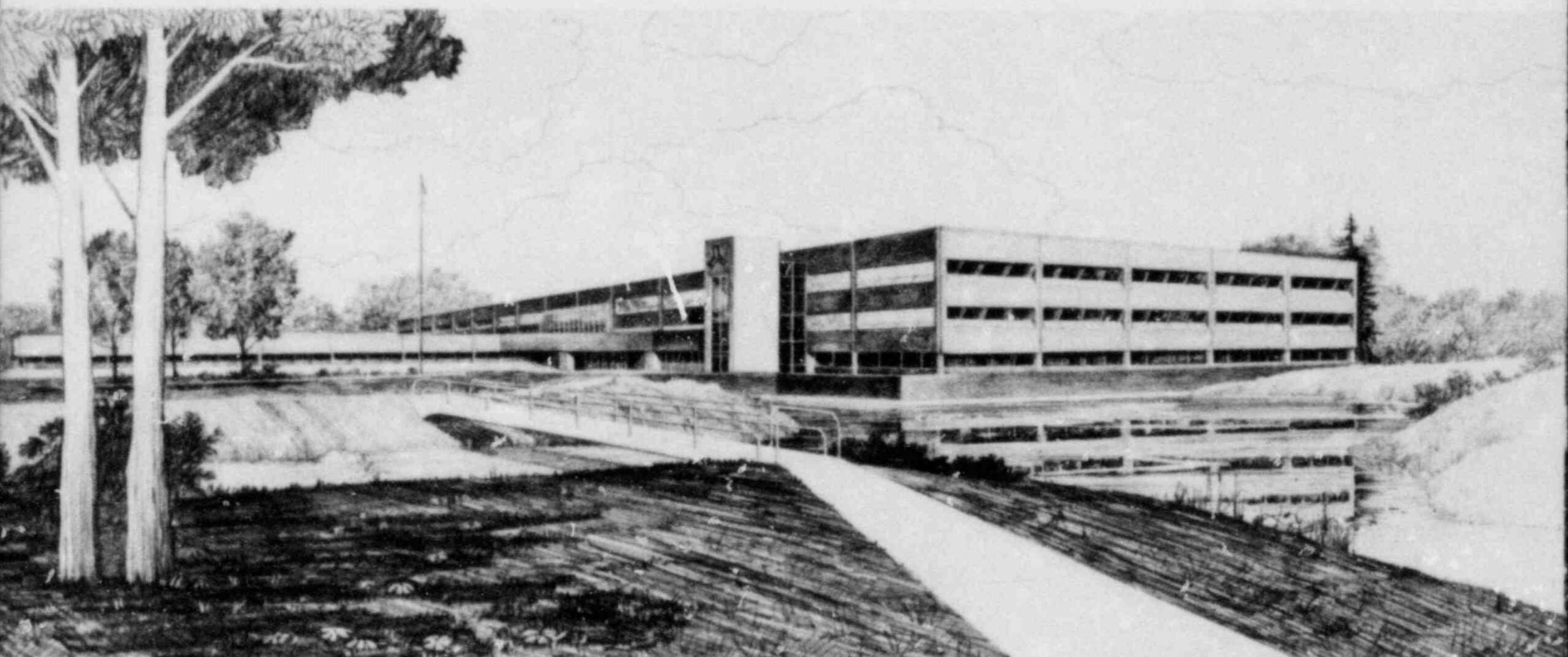

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ABSTRACT

During Loss-of-Fluid Test (LOFT) Experiment L2-6, nuclear fuel rod failure will occur and the transport of fuel particles from the reactor vessel is possible. Since the broken loop steam generator and pump simulators are potential traps for these particles, the possibility of removing or bypassing them for this experiment is being considered. Calculations of the thermal-hydraulic responses of the LOFT facility with and without these simulators indicate that minimal thermal-hydraulic impact is expected in any large-break experiment by replacing the simulators with a spool piece containing a single orifice. Estimates of fuel particle distributions after fuel rod failure indicate that sufficient fuel could be deposited in the simulators to warrant their removal for Experiment L2-6. A calculation of the thermal-hydraulic response of the LOFT facility during a large-break experiment with the reflood assist bypass valves opened and the broken loop hot leg isolation valve closed prior to fuel rod failure and reflood indicates that minimal thermal-hydraulic impact is expected in large-break experiments by utilizing these valve maneuvers.

SUMMARY

This report documents thermal-hydraulic and fuel particle distribution calculations which were performed to elucidate the issue of removing or bypassing the broken loop steam generator and pump simulators in the Loss-of-Fluid Test (LOFT) facility for Experiment L2-6. Experiment L2-6 is planned to simulate a large-break, loss-of-coolant accident in a commercial pressurized water reactor with inhibited emergency core coolant (ECC) injection. During this experiment, nuclear fuel rod failure will occur and the transport of fuel particles from the reactor vessel is possible. Since the LOFT broken loop simulators are potential traps for these particles, the possibility of removing or bypassing them for Experiment L2-6 is being considered.

Calculations of large-break, loss-of-coolant experiments (LOCEs) provided the calculated thermal-hydraulic responses of the LOFT facility with and without the simulators during blowdown and reflood. Since final parameters for Experiment L2-6 are not yet specified, these calculations were performed with various initial reactor powers and ECC injection initiation times, magnitudes, and locations. The results of these calculations indicate that minimal changes in the system thermal-hydraulic behavior are expected in any large-break experiment by replacing the simulators with a spool piece containing a single orifice.

The calculations also provided the liquid and vapor velocities in the upper plenum, intact loop hot leg, and broken loop hot leg required for estimating the distribution of fuel particles during and after reflood. The results of the distribution calculations indicate that sufficient fuel could be deposited in the simulators to warrant their removal for Experiment L2-6. Furthermore, these results indicate that lower plenum ECC injection should be used to minimize the transport of fuel particles into the intact loop steam generator.

A calculation was also performed to determine the thermal-hydraulic response of the LOFT facility during a large-break LOCE with the reflood assist bypass valves opened and the broken loop hot leg (BLHL) isolation valve closed prior to fuel rod failure and reflood. These valve maneuvers mitigate the problem of fuel particle transport into the broken loop simulators by providing a flow path from the BLHL into the broken loop cold leg which bypasses the simulators. The results of this calculation indicate that minimal thermal-hydraulic impact is expected in large-break experiments with slow core heatups by utilizing these valve maneuvers. Therefore, since these maneuvers also mitigate the problem of fuel particle transport into the broken loop steam generator and pump simulators during reflood and eliminate the cost and schedule impact of modifiying the BLHL, utilizing these maneuvers for Experiment L2-6 is recommended.

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AN EVALUATION OF THE THERMAL-HYDRAULIC AND RADIOLOGICAL EFFECTS OF REMOVING OR BYPASSING THE LOFT STEAM GENERATOR AND PUMP SIMULATORS FOR EXPERIMENT L2-6

1. INTRODUCTION

Experiment L2-6, which will be conducted in the Loss-of-Fluid Test (LOFT) facility, is planned to simulate a large-break, loss-of-coolant accident (LOCA) in a commercial pressurized water reactor (PWR) with inhibited emergency core coolant (ECC) injection. The ECC injection will be delayed until the cladding on nuclear fuel rods in the center module balloon and fail. Fuel particles that escape from the ruptured fuel rods could be transported to locations outside the reactor vessel during reflood.

Fuel particles deposited in the blowdown suppression tank (BST) present no special difficulties for postexperiment plant operations since the BST is adequately shielded. However, fuel particles trapped in the broken loop steam generator and pump simulators do present difficulties since the simulators must be shielded for postexperiment plant operations and eventually decontaminated.

The LOFT broken loop pump and steam generator simulators originated from a design compromise and were necessary to ensure scaling validity using the volumetric scaling approach. The major design compromise which resulted in the inclusion of the simulators was to combine three loops of the four-loop reference PWR into a single operating loop and to design a broken loop to hydraulically simulate the fourth loop of the reference PWR in which the break occurred. The LOFT broken loop was designed to maintain volume scaling, volume distribution, and flow resistances through the use of pump and steam generator simulators. In a LOCA situation, mass flow would be from the reactor vessel and other loops, through the broken loop pump and steam generator and out the break for either a cold leg or a hot leg break LOCA. The influence of the steam generator and pump in the broken loop would be principally hydraulic. Heat transfer in the steam generator in the loop in which the break occurred would not have a significant effect on the transient in a LOCA situation.

The broken loop steam generator simulator is a large, U-shaped piping arrangement containing seven orifice plates on each side of the "U". Each orifice plate has 85, 0.696-in. diamater holes distributed across it. The broken loop pump simulator also contains orifice plates to provide a hydraulic resistance typical of an inactive primary coolant pump. These orifice plates provide potential traps for fuel particles expelled from the reactor vessel into the broken loop hot leg (BLHL) during an experiment. For this reason, the possibility of replacing or bypassing the simulators for Experiment L2-6 is being considered.

Information needed to facilitate a decision on the removal of the steam generator and pump simulators for Experiment L2-6 must include the following:

1. The calculated thermal-hydraulic responses of the LOFT facility with and without the simulators during the blowdown and reflood phases of Experiment L2-6.

2. Estimates of the amount of fuel particles present in the simulators after Experiment L2-6 and the anticipated radiation field levels.

The results of thermal-hydraulic calculations indicate if the thermal-hydraulic response of the LOFT facility with the modified BLHL is expected to be similar to the response with the unmodified BLHL during blowdown and reflood. The system response without the simulators must be typical for Experiment L2-6 to be a credible test. Knowledge of the amount of fuel deposited in the simulators indicates if anticipated postexperiment radiation fields are sufficiently intense to seriously impact center fuel module removal or other postexperiment plant operations.

Several calculations of large-break, loss-of-coolant experiments (LOCEs) were performed to determine the calculated thermal-hydraulic responses of the LOFT facility with and without the broken loop steam generator and pump simulators during blowdown and reflood. Some results of these calculations were used to estimate the distribution of fuel particles during and after reflood. Section 2 discusses these thermal-hydraulic and fuel particle distribution calculations, while Sections 3 and 4 summarize their results.

Since there are cost and schedule impacts associated with modifying the BLHL for Experiment L2-6, a calculation was performed to determine the thermal-hydraulic response of the LOFT facility during a LOCE with the reflood assist bypass valves (RABVs) opened and the BLHL isolation valve closed prior to fuel rod failure and reflood. This calculation and its results are discussed in Section 5. These results are particularly important since a LOCE utilizing these valve maneuvers mitigates the potential problem of fuel particle transport into the broken loop simulators by providing a flow path from the BLHL to the broken loop cold leg (BLCL) which bypasses the simulators and eliminates the cost and schedule impact of modifying the BLHL for Experiment L2-6.

Conclusions obtained from and recommendations based on the results of the calculations are given in Section 6. These recommendations are concerned with the issue of the broken loop steam generator and pump simulators' removal and the conduct of Experiment L2-6.

2. CALCULATIONS

Calculations were performed to determine the calculated thermal-hydraulic responses of the LOFT facility with and without the broken loop steam generator and pump simulators during the blowdown and reflood phases of several LOCEs. Liquid and vapor velocities and mass flows in the upper plenum, intact loop hot leg (ILHL), and BLHL obtained from these calculations were then used for estimating the distribution of fuel particles during and after reflood.

Since final parameters for Experiment L2-6 are not yet specified, these calculations were performed with various initial reactor powers (24.88, 37, and 49.8 MW corresponding to 8, 12, and 16 kW/ft maximum linear heat generation rate), ECC injection times (normal and delayed), ECC injection magnitudes (from low pressure injection system only to full ECC injection with low and high pressure injection systems and accumulator), and ECC injection locations (lower plenum and intact loop cold leg). These calculations include:

1. Experiment L2-5 (12 kW/ft).[1]

2. A large-break LOCE at 8 kW/ft with delayed low and high pressure injection system (LPIS and HPIS) injection into the lower plenum.

3. A large-break LOCE at 16 kW/ft with delayed accumulator and LPIS injection into the intact loop cold leg (ILCL).

4. A large-break LOCE at 16 kW/ft with delayed LPIS injection into the ILCL.

Calculations of Experiment L2-5 were included to compare the calculated thermal-hydraulic responses with and without the simulators for a completed experiment whose experiment predicition calculation[2] agreed well with its data.

All computer calculations used the same version of the RELAP5/MOD1 computer code and the same code modifications as used for the Experiment L2-5 prediction calculation (see Reference 2). The physical models present in this code were adequate to calculate this experiment well. However, since RELAP5/MOD1 does not model the behavior of fuel rods under steady-state or transient conditions, the effects of flow blockage which result from ballooning fuel rods could not be included in the present calculations.

For the calculations without the simulators, the BLHL was replaced by piping exactly the same as the BLCL piping but containing a single break orifice whose area is 43% of the area of the BLCL break orifice. This area ratio is approximately the ratio of the maximum BLHL to BLCL mass flows measured during previously conducted LOFT large-break experiments.

Under critical flow conditions, the smaller BLHL orifice is expected to produce a flow distribution between the BLHL and BLCL representative of the distribution with the simulators intact. During reflood, however, flow through the BLHL orifice will not be choked. As a result, differences in the BLHL hydraulic resistances with and without the simulators may result in different reflood behavior. Since core coolability during reflood following fuel rod failure is an important aspect of Experiment L2-6, this portion of the transient is particularly important.

Table 1 lists pertinent ECC system parameters. Although the primary coolant pumps are unpowered after one second during the calculations of Experiment L2-5, they are powered for the first 30 s of the remaining calculations.

The fuel particle distribution calculations were based on liquid and vapor velocities in the upper plenum, ILHL, and BLHL which were obtained from RELAP5/MOD1 calculations with the unmodified BLHL. Upper plenum velocities indicate whether or not fuel particles can be transported from the reactor vessel into the ILHL or BLHL. If fuel particles are expelled

TABLE 1. EMERGENCY CORE COOLANT SYSTEM PARAMETERS

LOCE	Accumulator	High Pressure Injection System	Low Pressure Injection System	Injection Location
1	Normal initiation	Delayed until 22 s	Delayed until 3[illegible] s	Intact loop cold leg
2	Inactive	Delayed until cladding reaches 1200 K (1706°F)	Delayed until cladding reaches 1200 K (1706°F)	Lower plenum
3	Delayed until 60 s	Inactive	Delayed until 60 s	Intact loop cold leg
4	Inactive	Inactive	Delayed until 60 s	Intact loop cold leg

from the reactor vessel, the ILHL and BLHL mass flows determine their distribution in the ILHL and BLHL. All fuel particles entering the BLHL are assumed to be trapped by the steam generator and pump simulators. More details on the calculation methodology are found in Section 4. Estimates of the fuel particle distribution after experiment completion were made for the 8 kW/ft LOCE and the 16 kW/ft, high-ECC LOCE.

3. RESULTS OF THERMAL-HYDRAULIC CALCULATIONS

This section contains a brief overview of the results of the thermal-hydraulic calculations discussed in the previous section. Sections 3.1 through 3.4 summarize the results of the four LOCE scenarios with and without the broken loop steam generator and pump simulators.

3.1 Experiment L2-5

Calculation of the first 80 s of Experiment L2-5 was performed with the steam generator and pump simulators replaced by a spool piece containing a single orifice. The results of this calculation were compared with those of the Experiment L2-5 prediction calculation to determine if significant differences in system thermal-hydraulic response result from this modification to the BLHL. The calculation of similar thermal-hydraulic responses for this experiment with and without the simulators would support the decision to remove the simulators for Experiment L2-6. For this reason, calculations were first performed for Experiment L2-5 even though the scenario for Experiment L2-6 will be different.

Figure 1 compares calculated fuel rod cladding temperatures at the highest powered section of the core for calculations with and without the simulators. The calculated core thermal responses are in excellent agreement at this location. Such agreement is important since the thermal behavior of the nuclear fuel rods is the greatest concern during large-break LOCEs.

Excellent agreement was also calculated for system (upper plenum) pressure, as shown in Figure 2.

Calculated mass flows in the BLHL are compared in Figure 3. Although some differences exist prior to 15 s, the integrated mass flows are about the same at 15 s. Some differences are expected since the configurations

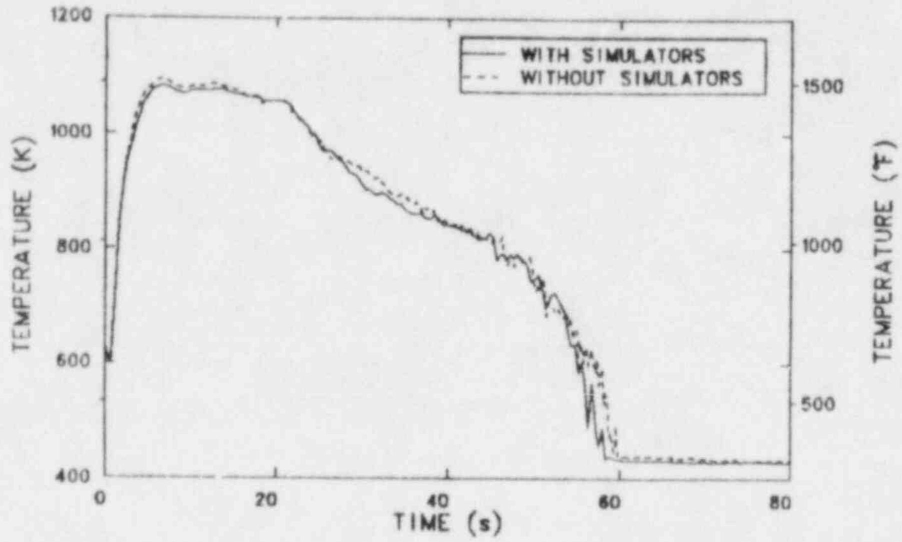

Figure 1. Cladding temperatures for L2-5 calculations.

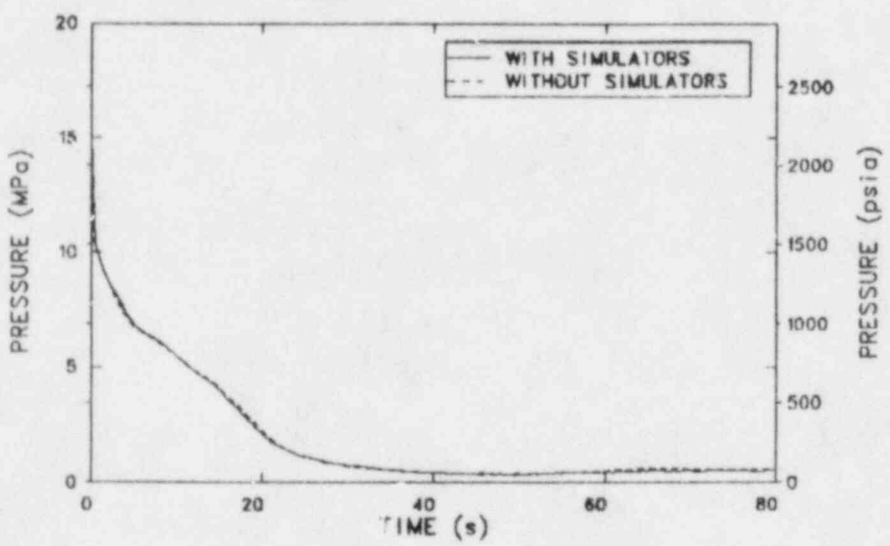

Figure 2. System pressures for L2-5 calculations.

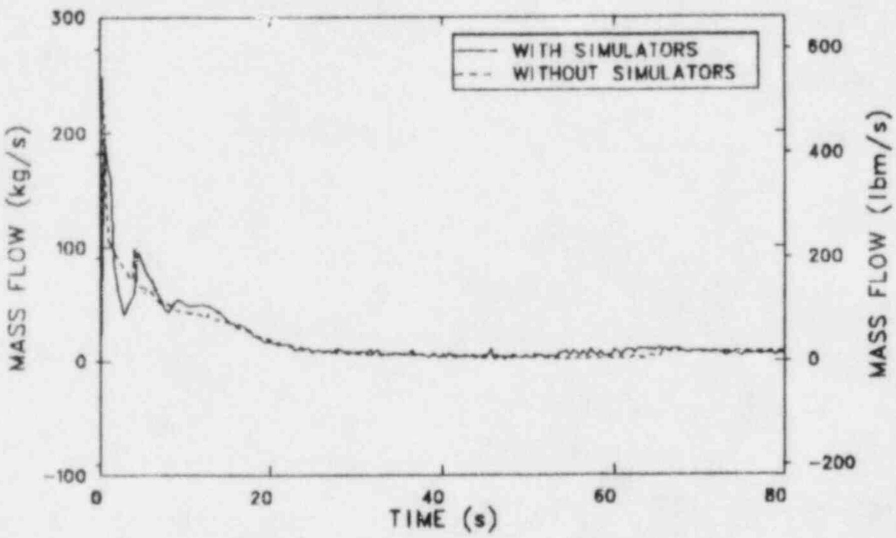

Figure 3. Broken loop hot leg mass flows for L2-5 calculations.

and volumes of the BLHL with and without the simulators differ. With the simulators present, there are significant pressure losses associated with two-phase friction and form losses. With the simulators removed and replaced by a spool piece containing a single orifice, such losses are small, but are compensated for by a reduction in choked flow at the reduced size break orifice during blowdown. In spite of these differences, the agreement in mass flows is reasonable prior to 15 s and good for the remainder of the calculation.

Figure 4 shows good agreement for calculated mass flows in the BLCL during blowdown (first 24.2 s). Agreement is also good for the remainder of the calculation except for some large increases that are calculated to occur at different times. These increases appear to be associated with bypass of liquid from the accumulator, which injects between 14.3 and 46.8 s.

Similar good agreements are shown in Figures 5, 6, and 7 for calculated mass flows in the ILHL, ILCL, and core center module, respectively. Particularly important is the agreement in the core center module during blowdown since this agreement shows that the small differences in BLHL mass flows are not propagated into the core.

Therefore, Figures 1 through 7 indicate that minimal thermal-hydraulic impact is calculated in Experiment L2-5 by replacing the steam generator and pump simulators with a spool piece containing an orifice whose area is 43% of the BLCL break orifice.

3.2 Large-Break LOCE at 8 kW/ft

A possible scenario for Experiment L2-6 is a large-break LOCE with (a) initial power corresponding to a maximum linear heat generation rate (MLHGR) of 8 kW/ft, (b) ECC injection delayed until a number of nuclear fuel rods in the center module balloon and fail, and (c) HPIS and LPIS injection into the lower plenum. A MLHGR of 8 kW/ft is appropriate for fuel rods pressurized to 4.137 MPa (600 psia), the internal pressure to

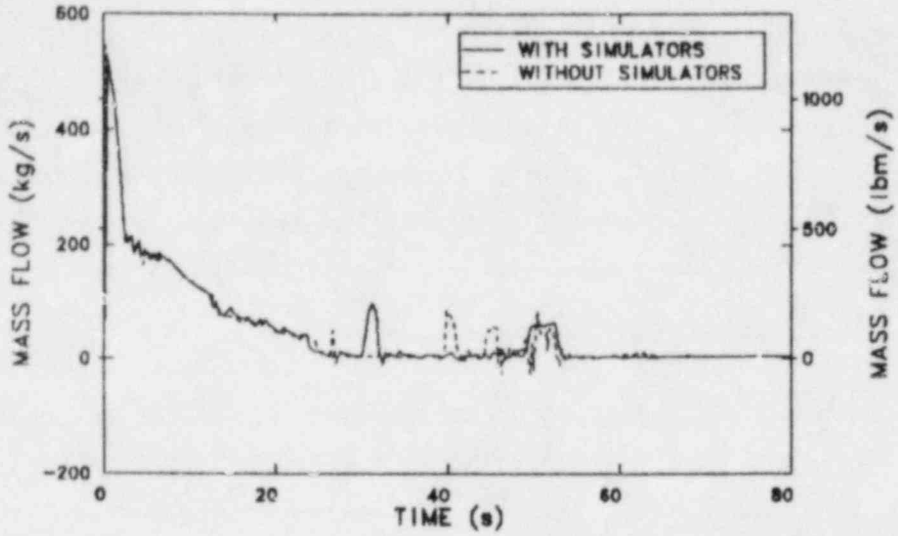

Figure 4. Broken loop cold leg mass flows for L2-5 calculations.

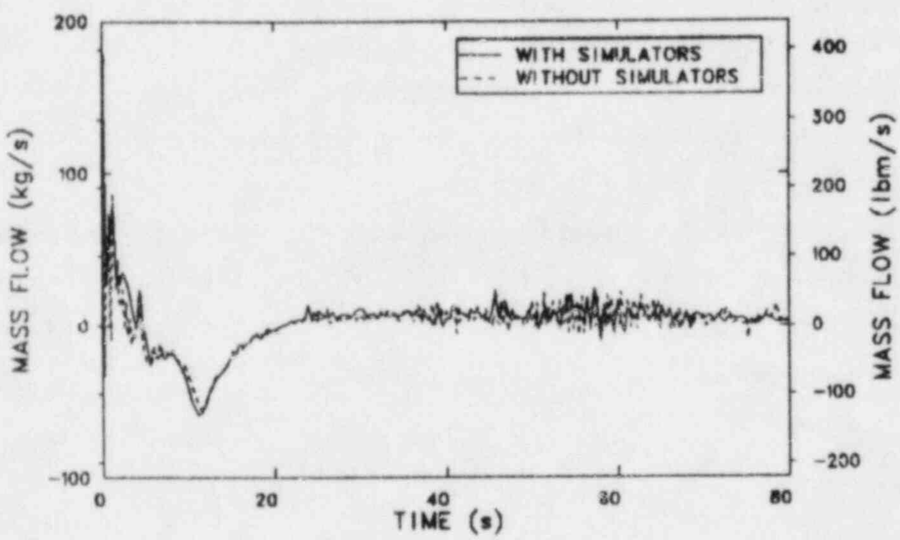

Figure 5. Intact loop hot leg mass flows for L2-5 calculations.

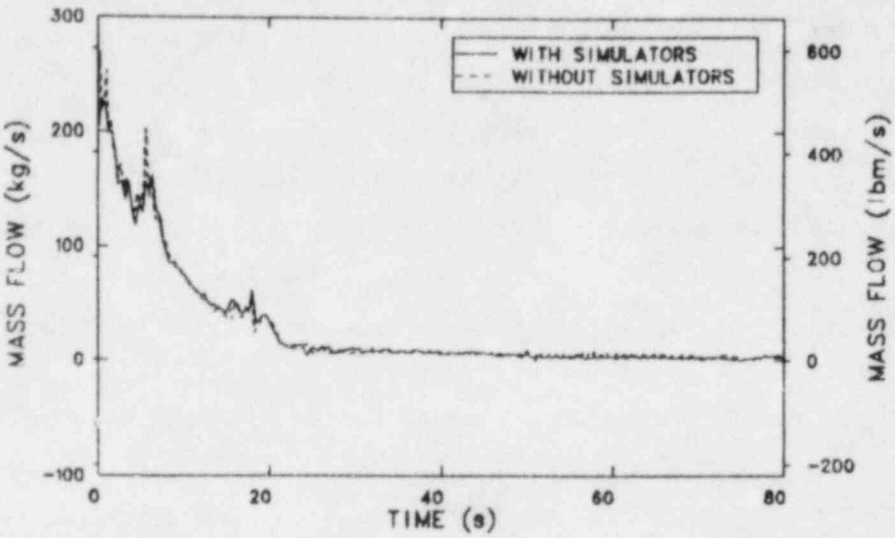

Figure 6. Intact loop cold leg mass flows for L2-5 calculations.

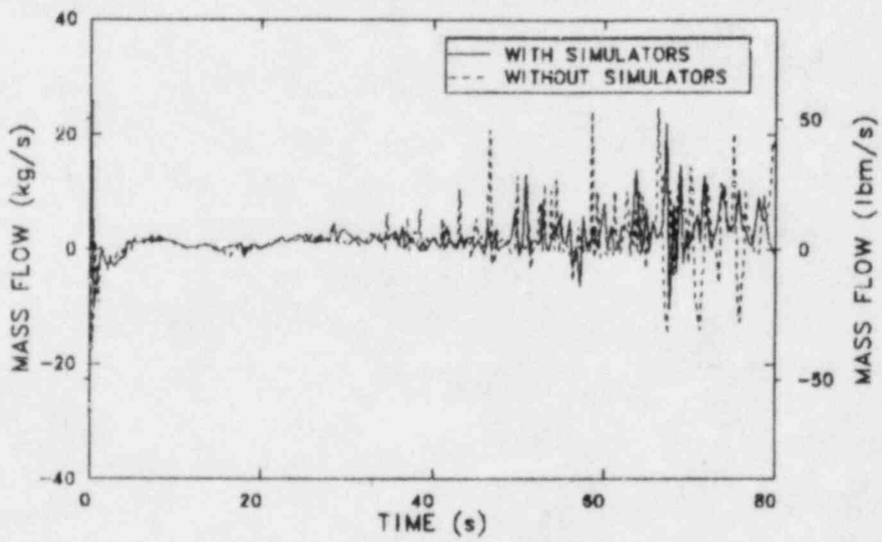

Figure 7. Core center module mass flows for L2-5 calculations.

be used for fuel rods in the center module for Experiment L2-6. The delay of ECC until fuel rods are deformed is a principal objective of the experiment. In the 8 kW/ft calculations, ECC injection was initiated when the cladding temperature of the average fuel rod in the center module reached 1200 K (1706°F). FRAP-T6[3] calculations indicate that fuel rod failure occurs at approximately this temperature for rods pressurized to 4.137 MPa (600 psia). Using this initiation criterion, ECC injection commenced at 268 s during the calculation with the simulators and at 256 s during the calculations without the simulators. ECC liquid was injected into the lower plenum to achieve a more controlled reflood by eliminating ECC bypass associated with injection into the ILCL. Only the HPIS and LPIS were used to provide a low reflood rate typical of those rates calculated by conservative models. Furthermore, these systems were adequate to reflood the core during the second heatup in Experiment L2-5.[4]

Figure 8 compares calculated fuel rod cladding temperatures for the 8 kW/ft calculations with and without the simulators. The minor differences in these temperatures result from the second heatup starting earlier during the calculation without the simulators (12.8 s without simulators vs 23 s with the simulators). The second heatup during the calculation with the simulators was delayed because the core did not dry out as rapidly after the early rewet. The second heatup during the calculation without the simulators occurs at about the same time as the second heatup occurred during Experiment L2-2,[5] which is an identical transient prior to accumulator injection. Therefore, the influence of removing the simulators falls within the limits of the computer code's capability to calculate the experiment.

No similar disparity appears in Figure 9, which compares calculated system pressures for the 8 kW/ft calculations. The agreement shown in this figure is excellent.

3.3 Large-Break LOCE at 16 kW/ft with High-Flow ECC

Since the initial reactor power to be used for Experiment L2-6 is currently being evaluated, several LOCE calculations were performed with a

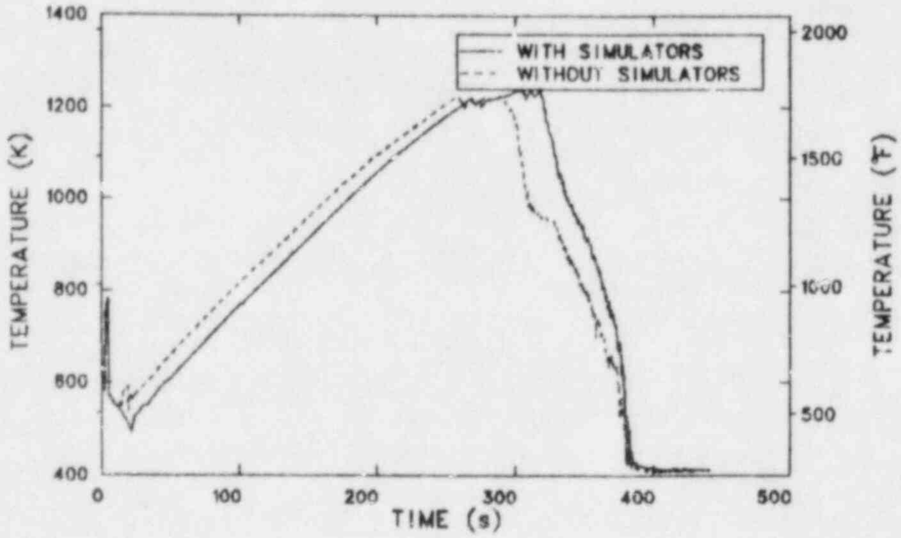

Figure 8. Cladding temperatures for 8 kW/ft calculations.

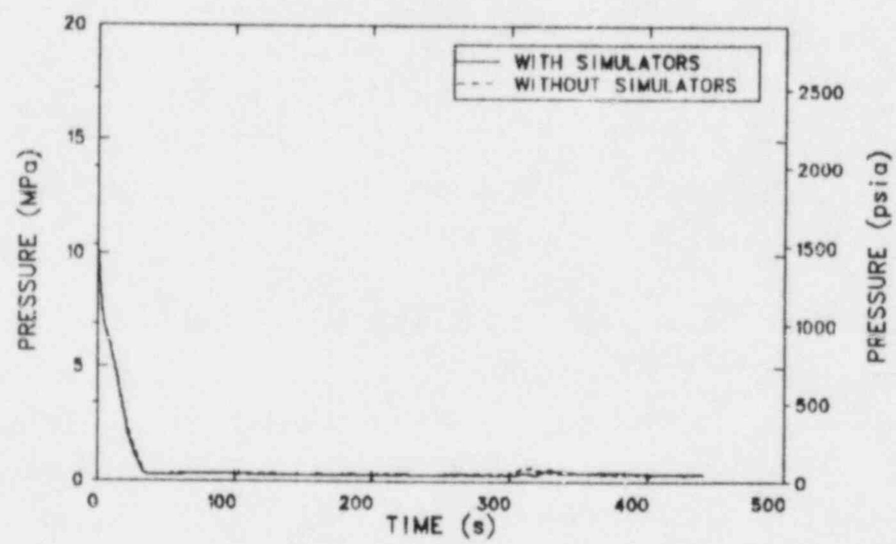

Figure 9. System pressures for 8 kW/ft calculations.

MLHGR of 16 kW/ft. For the LOCE discussed in this section, accumulator and LPIS flows into the ILCL were initiated at 60 s, which is about 10 s after the time FRAP-T6 calculated fuel rod failure to occur. In these calculations, the accumulator emptied at about 87.2 s.

Calculated fuel rod cladding temperatures are compared in Figure 10 for the 16 kW/ft, high-flow ECC calculations with and without simulators. These cladding temperatures agree well. The main differences are the times at which the second heatup began after the quenches at about 125 s. The second heatups are caused by core dryouts which result from the entrainment of liquid in the core by steam venting to the BST. The dryouts occur at different times since the BLHL hydraulic resistances are different with and without the broken loop simulators and, hence, lead to different entrainment rates. Their occurrence indicates that LPIS injection alone into the ILCL may be insufficient to maintain adequate core cooling.

Excellent agreement in calculated system pressures is shown in Figure 11.

3.4 Large-Break LOCE at 16 kW/ft with Low-Flow ECC

For the large-break LOCE at 16 kW/ft discussed in this section, only LPIS flow into the ILCL was initiated at 60 s.

Figure 12 compares calculated fuel rod cladding temperatures for the 16 kW/ft, low-flow ECC calculations with and without the steam generator and pump simulators. For this LOCE, significantly different cladding temperature behaviors are calculated. After LPIS initiation, the cladding temperature increase is arrested and the cladding is slowly cooled by fluid reaching the core during the calculation with the simulators. However, during the calculation without the simulators, this temperature increase is also arrested but the cladding temperatures remain elevated until the cladding is quenched at about 350 s.

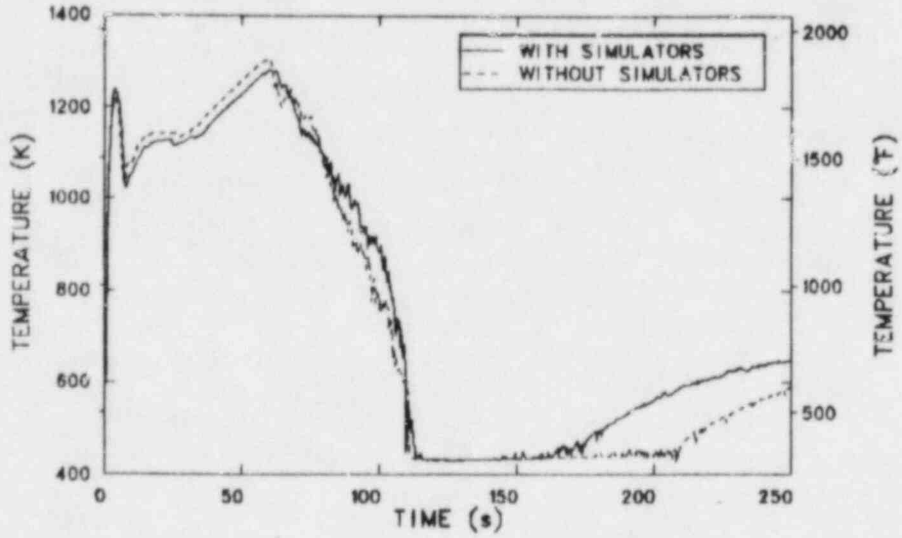

Figure 10. Cladding temperatures for 16 kW/ft, high-ECC calculations.

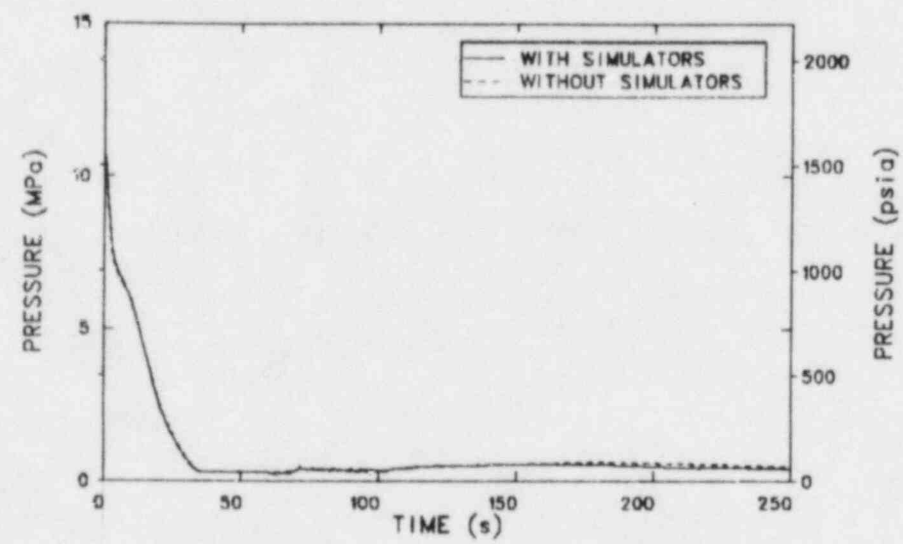

Figure 11. System pressures for 16 kW/ft, high-ECC calculations.

The behavior of the cladding temperatures between 100 and 300 s during the calculation without the simulators suggests that little LPIS liquid reached the core during this time interval. This conjecture is substantiated by Figure 13, which compares calculated mass flows in the core center module. Mass flows during the calculation without the simulators are close to zero or negative between 100 and 300 s.

The reason there is insufficient LPIS liquid reaching the core during this calculation can be ascertained from Figures 14 and 15. Figure 14 compares calculated densities in the ECC injection volume in the ILCL; Figure 15 compares them in a volume upstream of the injection point in the ILCL. Considerably lower densities are shown in Figure 14 for the calculation without the simulators, whereas the opposite is the case in Figure 15. Therefore, without the simulators a large fraction of the LPIS liquid injected into the ILCL flows through the intact loop away from the reactor vessel toward the BLHL prior to the initiation of the cladding quench, a large enough fraction that the cladding temperatures remain elevated. During reflood, the different BLHL hydraulic resistances with and without the simulators result in different ECC flow splits for low-magnitude ECC injection into the ILCL.

Although there is significant disagreement between calculated cladding temperatures, the calculated system pressures agree well, as shown by Figure 16.

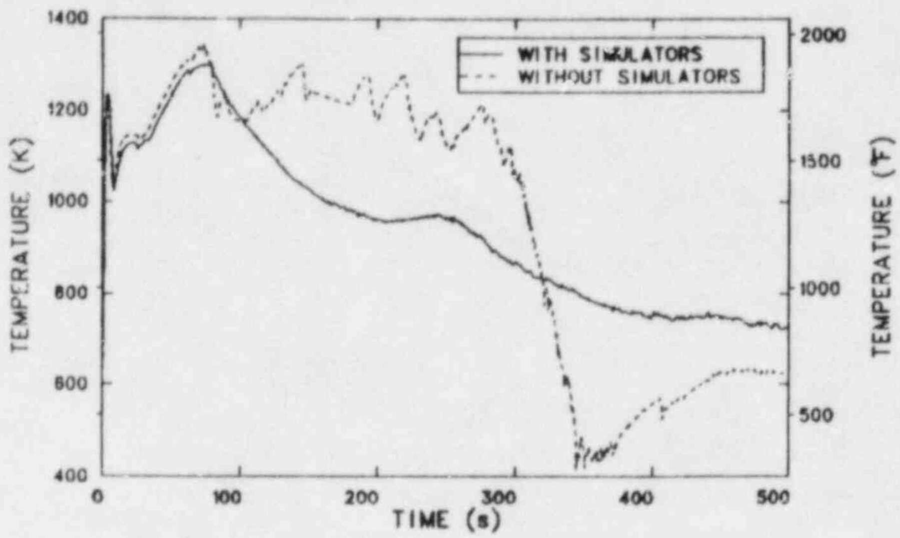

Figure 12. Cladding temperatures for 16 kW/ft, low-ECC calculations

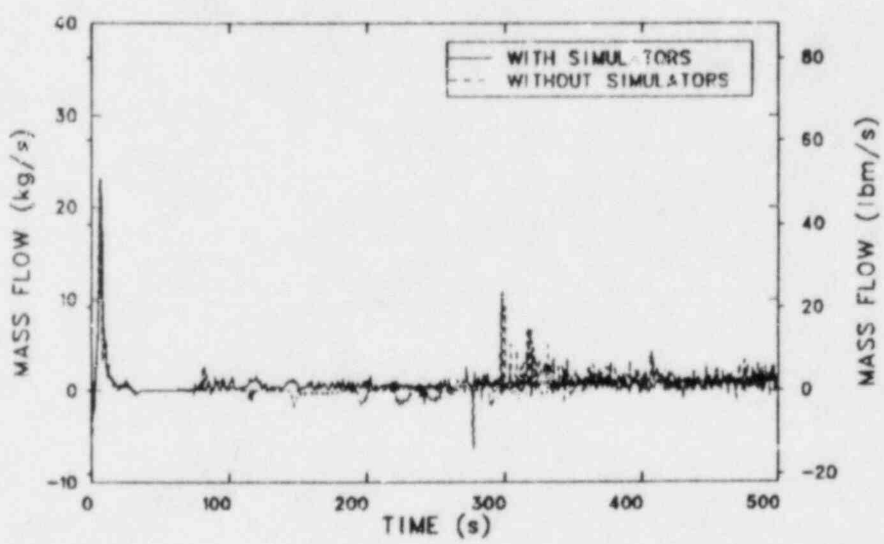

Figure 13. Core center module mass flows for 16 kW/ft, low-ECC calculations.

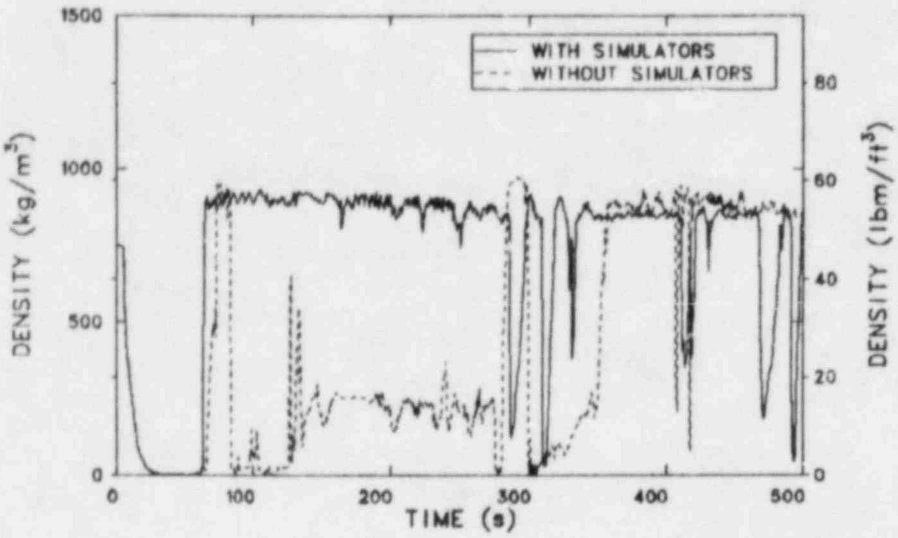

Figure 14. ECC injection volume densities for 16 kW/ft, low-ECC calculations.

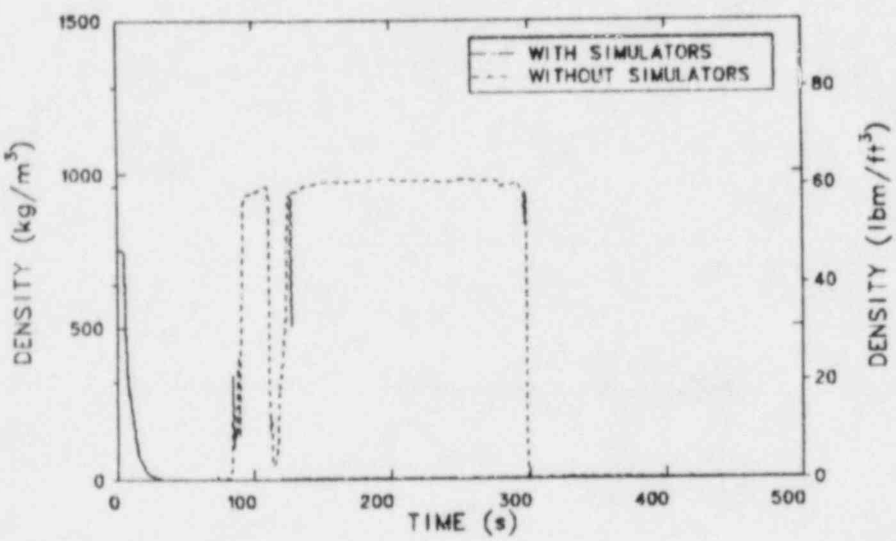

Figure 15. Densities upstream of ECC injection volume for 16 kW/ft, low-ECC calculations.

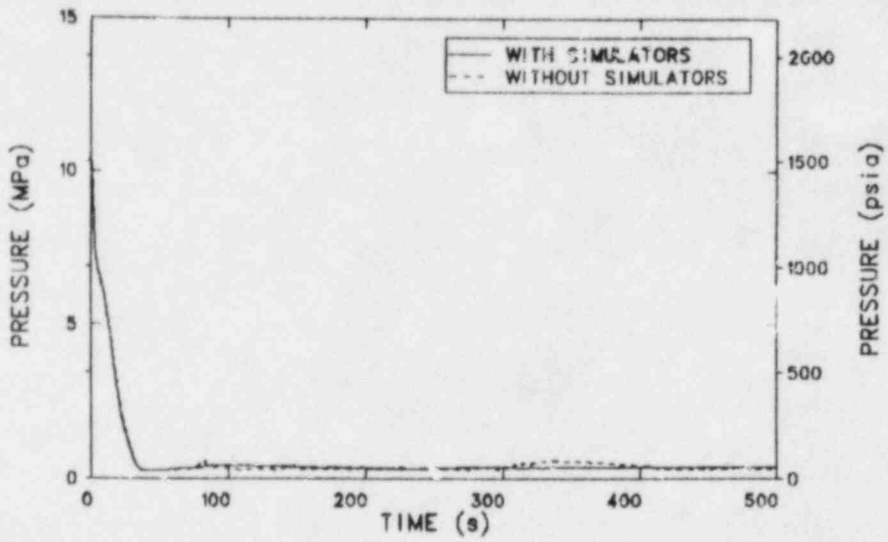

Figure 16. System pressures for 16 kW/ft, low-ECC calculations.

4. RESULTS OF FUEL PARTICLE DISTRIBUTION CALCULATIONS

In order to evaluate the postexperiment decontamination problems that may occur by allowing the broken loop steam generator and pump simulators to remain in the LOFT facility for Experiment L2-6, it is necessary to first evaluate the transport characteristics of the fuel particles that may be released from the ruptured rods. The present test scenario calls for a large percentage of the rods (about 70%) in the center fuel assembly to fail. Two pellets of fuel are assumed to be released from each ruptured rod, resulting in a total mass of 3191 grams. Power Burst Facility (PBF) data indicates that the released fuel will assume the particle size distribution shown in Figure 17, 40 μm being a lower bound and 2500 μm being an upper bound for particle size.

In addition to the particle size distribution, it is necessary to determine the minimum velocity necessary to move a nominally sized fuel particle into the primary coolant system (PCS) piping. This velocity can be estimated by determining the terminal velocity of each size particle under consideration. The terminal velocity of a particle through a fluid occurs when the drag on the particle is equal to its submerged weight. Assuming a spherical object, this relationship can be expressed as,

$$\frac{1}{6} \pi d_s^3 (\rho_s - \rho) g = C_D \frac{\pi}{4} d_s^2 \rho \frac{V_f^2}{2}$$

or

$$V_f^2 = \frac{4}{3} \frac{1}{C_D} g d_s \frac{\rho_s - \rho}{\rho} \qquad (1)$$

where

V_f = terminal velocity

C_D = drag coefficient

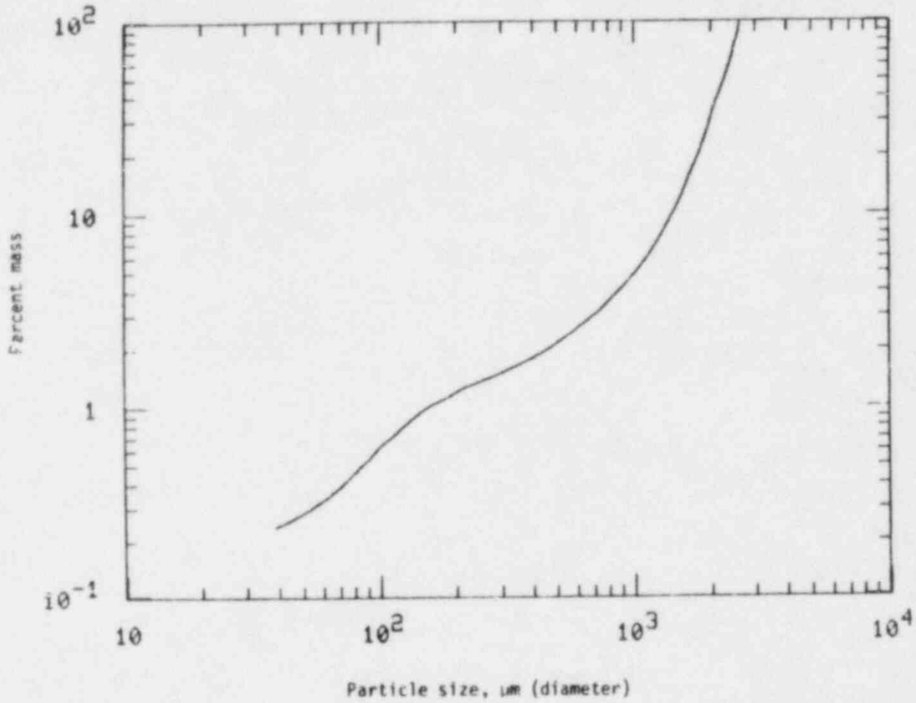

Figure 17. Released fuel particle size distribution.

d_s = particle diameter

ρ_s = particle density

ρ = fluid density

g = acceleration due to gravity = 9.8 m/s^2.

Using the relationship for the Reynolds number (Re)

$$Re = \frac{V_f d_s}{\nu}$$

where

ν = kinematic viscosity,

Equation 1 yields

$$V_f = \sqrt[3]{\frac{4}{3} \frac{1}{C_D} g \, Re \, \nu \, \frac{\rho_s - \rho}{\rho}} \qquad (2)$$

This expression is valid for a single sphere falling in an infinite fluid (i.e., no surface effects from the container). For a number of particles uniformally dispersed in a fluid, the terminal velocity will be less than that for a single particle and will decrease as the particle concentration increases. The terminal velocity was first evaluated at bulk fluid conditions of 294 K, 0.1 MPa (70°F, 14.7 psia). Substituting water properties into Equation (2), along with the density of uranium dioxide, 10.21 Mg/m^3, yields

$$V_f = \sqrt[3]{4.154 \times 10^{-3} \frac{Re}{C_D}}$$

Substituting the relationship of C_D to Reynolds number for a sphere (Figure 18) the terminal velocity can be determined as a function of particle size. This technique was used and applied to saturated steam at pressures ranging from 0.138 to 13.79 MPa (20 to 2000 psia) to approximate the fuel particle transport characteristics of steam during the initial quench.

These results use the following assumptions:

1. Vertical particle motion. Horizontal transport requires a higher flow rate which depends on the size of the particles, their relative location, their shape, their cohesiveness, and their concentration, as well as the velocity profile and turbulance of the medium. The flow rate required to transport particles horizontally must be determined experimentally and may be somewhat higher.

2. Spherical particle shape. Nonspherical particles have higher drag coefficients than spherical particles and are more likely to be transported.

3. Noncohesive particles. Particles which are attached and move together will have a higher drag coefficient than a single spherical particle of the same mass. Such particles are more likely to be transported than a single spherical particle of identical mass.

4. No fluid boundary layer effects. The present correlation relates a velocity to a transported particle size. Boundary layers create a velocity profile across a given cross-sectional flow area and a nonsymmetric velocity profile across a particle flow (unless the particle is on the centerline). A nonsymmetrical velocity profile causes a lateral motion of a particle which could inhibit its transport.

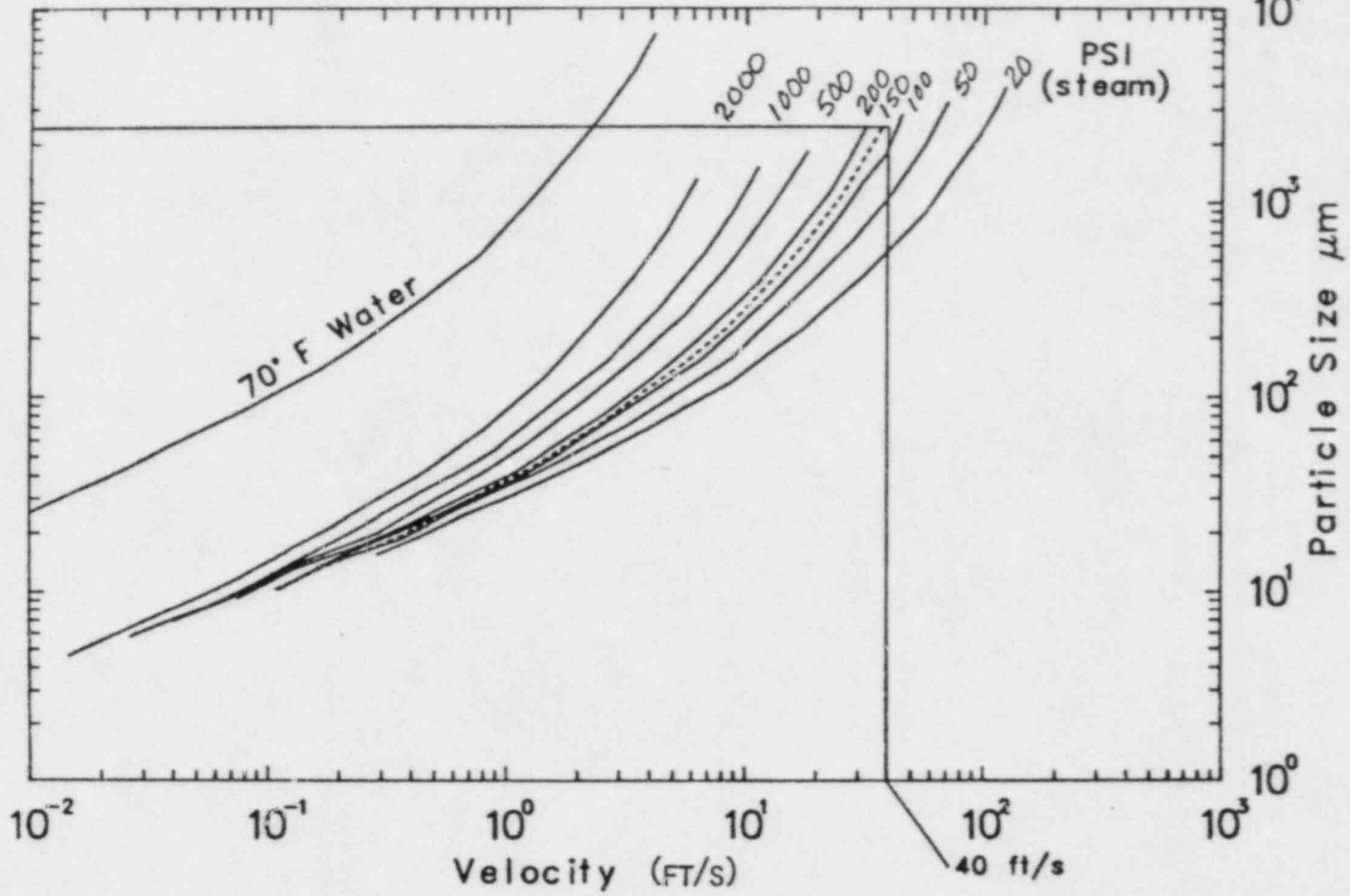

Figure 18. Required steam velocity necessary for particle carryover.

The results of the above technique are presented in Figure 18 and are estimates of the velocity necessary to move a particle. Due to the assumptions necessary to develop this result, these estimates should be considered to be nonconservative. To estimate the fuel particle transport characteristics during Experiment L2-6, the above results were used for two different scenarios assuming the simulators remain in the flow path. The two scenarios considered are:

1. 16 kW/ft MLHGR with scaled LPIS and accumulator injection into the ILCL, and

2. 8 kW/ft MLHGR with scaled LPIS and HPIS injection into the lower plenum.

Velocities in the vessel and around the loops were obtained from the thermal-hydraulic calculations discussed in Section 3. For the 16 kw/ft case, the highest velocities occurred between 60 and 110 s. This time interval corresponds to the time between the initiation of ECC injection and the final core quench (Figure 10). Figure 19 shows calculated velocities in the primary system during the time interval (60 to 110 s) following fuel failure. Since the upper plenum flow area is larger than the flow area in the core region, velocities in the upper plenum are lower and, therefore, represent the limiting case for particle carryover into the hot legs. For a velocity (12.2 m/s, 40 ft/s) and a primary system pressure (1.03 MPa, 150 psia), representative of the conditions in the reactor vessel between 60 and 110 s, particle sizes less than or equal to 2500 μm can be carried into the hot legs. Extrapolating this particle size to the percentage of total mass available (Figure 18), 100% of the available fuel can be transported into the primary system piping with an upper plenum steam velocity of 12.2 m/s (40 ft/s). For the 16 kW/ft case, RELAP5/MOD1 calculated velocity spikes of very short duration in the upper plenum and core regions that also exceeded the 12.2 m/s (40 ft/s) required to transport all the available fuel.

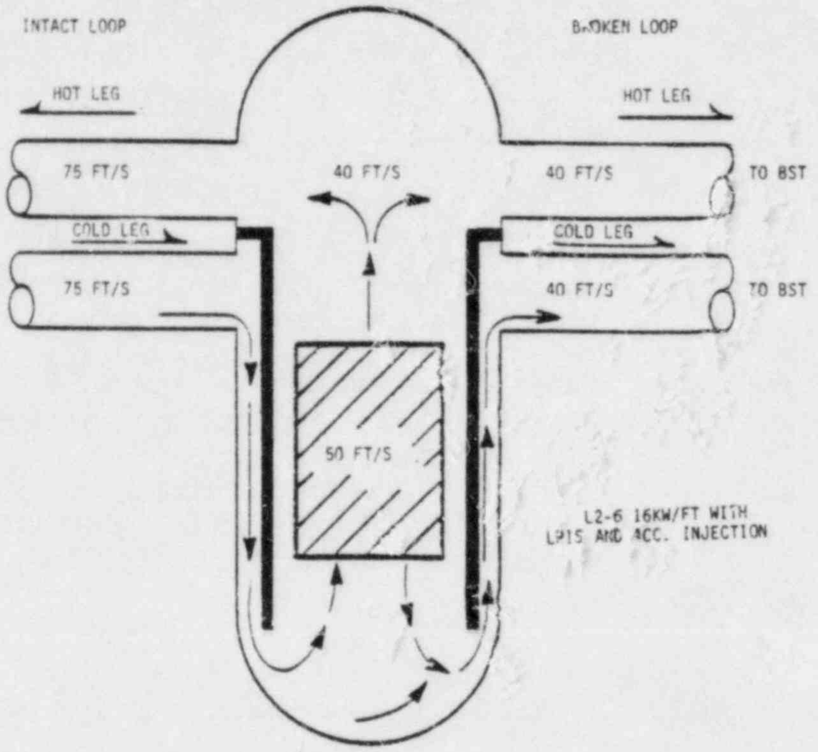

Figure 19. L2-6 anticipated steam velocities from 60 to 110 s.

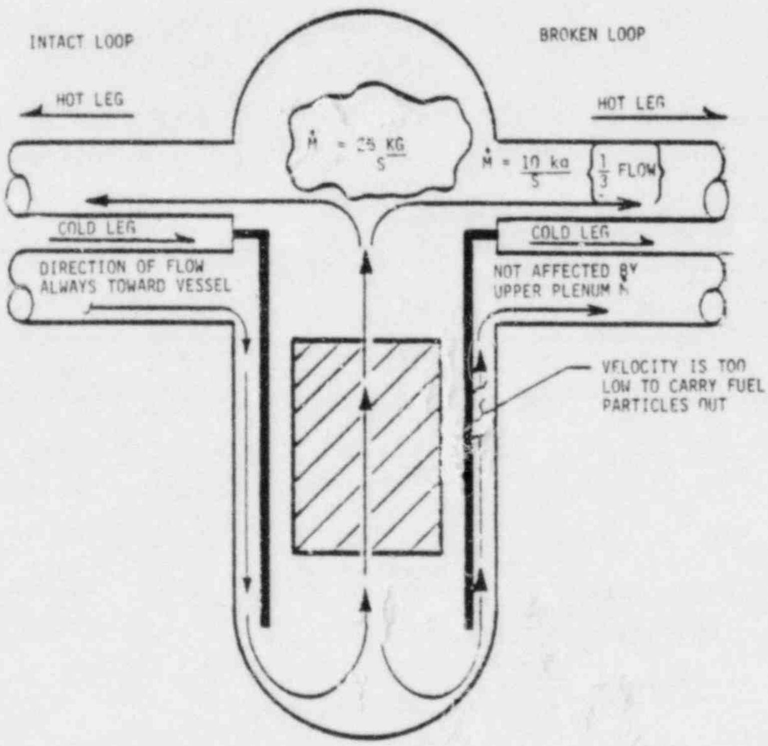

Figure 20. L2-6 anticipated mass flow split from 60 to 110 s.

Assuming that all the available fuel reaches the upper plenum, a flow split must be determined. The mass flows shown in Figure 20 were calculated for the 16 kW/ft case using the RELAP5/MOD1 code. The mass flow distribution shown in this figure is representative only during the high velocity period (60 to 110 s) following fuel failure. During this period the mass flow to the upper plenum is approximately 25 kg/s (55 lbm/s). The mass flow then splits into 15 kg/s (33 lbm/s) toward the ILHL and 10 kg/s (22 lbm/s) towards the BLHL. Thus, approximately two-thirds of the available fluid mass and consequently two-thirds of the available fuel particles are expected to travel into the ILHL toward the steam generator. Velocities decrease to less than 3.048 m/s (10 ft/s) in the steam generator inlet plenum enabling the entrained particles to fall out. The other one-third of the available fuel particles are carried into the BLHL.

The first large obstruction encountered in the BLHL is the steam generator simulator. As discussed earlier, the steam generator simulator is a large U-tube piping arrangement containing seven baffle plates on each side of the "U". Each baffle plate has 85, 0.6815 in. diameter holes distributed evenly across it. This arrangement (Figure 21) makes the simulator impassible for any appreciable amount of fuel particles with the possible exception of small amounts of silt which may be present.

The ILCL and BLCL are not considered to be potential flow paths for the transport of fission products from the vessel region. During the period following fuel failure, the ILCL flow is directed towards the vessel, which prevents fuel particle transport away from the reactor vessel. The BLCL flow is towards the BST; steam must travel upwards through the downcomer region before leaving the vessel into the BLCL. The velocities that are calculated in the downcomer region are low enough that the carryover is minimal.

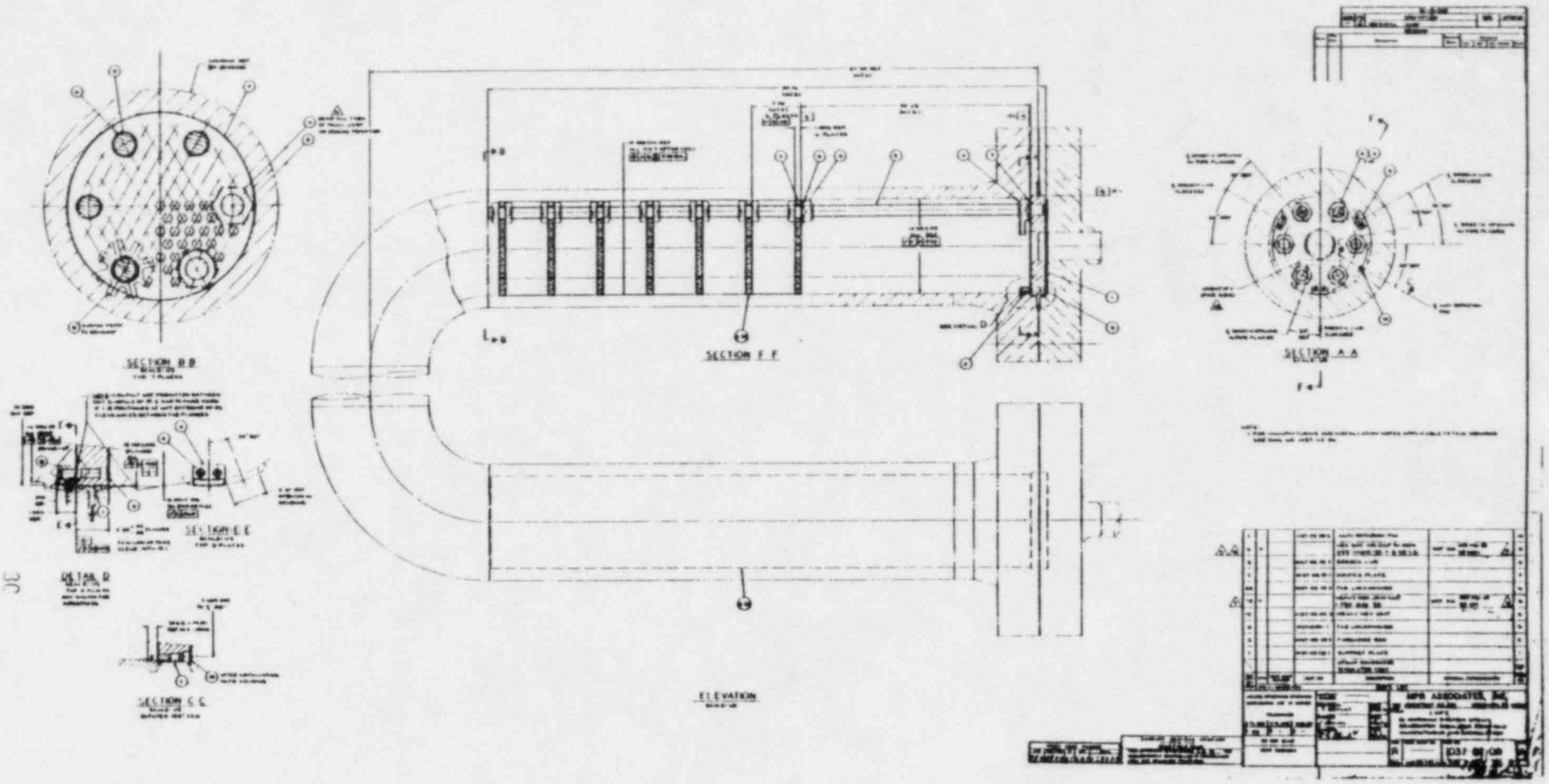

Figure 21. LOFT steam generator simulator.

The assumptions and conservatisms used to obtain the above results should be noted. The velocities used in the calculations were obtained from RELAP5/MOD1 code calculations. The steam curves (Figure 18) are nonconservative as previously mentioned. Furthermore, the fuel particles entrained in the steam must follow a tortuous path to reach the upper plenum. During this migration, it is not known, nor can it be predicted, what percentage of fuel particles will de-entrain or become lodged in obstructions. There are many other unknowns which may also mitigate or enhance the transport of fuel particles into the PCS, such as (a) cladding rupture size, (b) rod ballooning, (c) flow channeling, and (d) mass flow reversals in the lower plenum region. In view of these uncertainties, it is felt that a conservative approach should be taken and all of the available fuel should be assumed to be removed from the core and transported into the PCS piping.

The post experiment cleanup scenario includes removal of the center fuel bundle 60 to 90 days after the experiment. Therefore, it is necessary to estimate the radiation fields at the reactor vessel head where people will be working. It is necessary to consider the unshielded radiation fields from each component individually.

The following assumptions were made to estimate the radiation fields at the reactor vessel head:

1. One-third of all released fuel particles are transported to and remain in the broken loop simulators following Experiment L2-6.

2. Two-thirds of all released fuel particles are transported to and remain in the steam generator inlet plenum following Experiment L2-6.

3. The fuel particles are considered a point source 3.048 m (10 ft) from the reactor vessel head.

4. There is no shielding or self absorption by the individual components or fuel.

5. The fuel particles in the broken loop simulators and intact loopsteam generator are the only sources of radiation in the containment.

6. No volatile fission products are contained within either component after 60 days.

The unshielded radiation field (R_2) at the reactor vessel head 60 days after Experiment L2-6, which results from the fuel particles remaining in the broken loop simulators, is calculated below.

$$R_2 = \frac{f\ G_2 Q}{(d_2)^2} \quad = 9.2\ R/hr$$

where

f = normalized radiation field at 0.3048 m (1 ft) after 60 days = 1.7×10^{-3}

G_2 = mass of source = 1/3 x 3191 g = 1064 g

d_2 = distance to reactor vessel head from source = 3.048 m (10 ft).

Q = radiation field from 2500μm diameter particle 150 s after shutdown = $5.066 \times 10^2\ R{\cdot}ft^2/g{\cdot}hr$.

The unshielded radiation field (R_3) at the reactor vessel head 60 days after Experiment L2-6, which results from the fuel particles trapped in the intact loop steam generator, is calculated below.

$$R_3 = \frac{f\ G_3 Q}{(d_3)^2} = 18.3\ \text{R/hr}$$

where

G_3 = mass of source = 2/3 x 3191 g = 2127 g

d_3 = distance to reactor vessel head from source = 3.048 m (10 ft).

Since the broken loop simulators and the intact loop steam generator are equidistant from the reactor vessel head, both sources will contribute equally to the exposure field in that area. To reduce the exposure field to the presently acceptable limit of 10 mR/hr, both source terms must be reduced to 5 mR/hr, which will require 0.094 and 0.088 m (3.72 and 3.45 in.) of solid lead shielding on the steam generator and simulators, respectively. This result was derived from the standard shielding equation,

$$A = A_o B e^{-\mu x},$$

where

x = shielding thickness in cm

A = radiation field with shielding in R/hr

A_o = radiation field without shielding in R/hr

B = buildup factor = 3.5

μ = linear attenuation factor = 1 cm^{-1}.

Assuming the average gamma ray energy equals 800 Kev, the following calcualtions yield the required lead shielding thickness:

Shielding Required for Intact Loop Steam Generator

$$A = 0.005 = 18.3\ (3.5)\ e^{-x}$$

$$x = -\ln \frac{0.005}{18.3\ (3.5)} = 9.45 \text{ cm } (3.72 \text{ in.})$$

Shielding Required for Steam Generator and Pump Simulators

$$A = 0.005 = 9.2\ (3.5)\ e^{-x}$$

$$x = -\ln \frac{0.005}{9.2\ (3.5)} = 8.77 \text{ cm } (3.45 \text{ in.})$$

Since this estimate might seem unrealistically conservative, the effects of reducing the amount of fuel transported into the PCS to 6% of the released fuel were considered. The effect on the required shielding would be minimal (Table 2). The intact loop steam generator would require 0.066 m (2.61 in.), while the simulators would require 0.059 m (2.34 in.) of solid lead. Supporting this amount of lead around the simulators is not practical. The configuration of the simulators would require a shielded box 2.54 m (100 in.) by 2.54 m (100 in.) by 0.813 m (32 in.) as a minimum. Rough estimates indicate that if a 0.102 m (4 in.) lead thickness is used, approximately 24.2 Mg (26.65 tons) of lead would have to be supported in an area which is 6.1 m (20 ft) above the containment floor and an area of the containment building would require rigorous seismic qualification of the support structure. If only 0.051 m (2 in.) of solid lead shielding is required, the overall weight would be reduced in half to approximately

TABLE 2. RADIATION FIELDS AND SHIELDING REQUIRED AS A FUNCTION OF FUEL TRANSPORTED

% Fuel Transported to Primary System	Source in Grams		Unshielded Field in R/hr		Required Shielding in Inches of Solid Lead	
	Simulators	Steam Generator	Simulators	Steam Generator	Simulators	Steam Generator
100	1064	2127	9.16	18.32	3.45	3.72
66	702	1404	6.04	12.09	3.28	3.56
50	532	1063	4.58	9.15	3.17	3.45
33	351	702	3.02	6.05	3.01	3.28
6	64	127	.55	1.09	2.34	2.61

11.8 Mg (13 tons), which is still impractical if not impossible to support. However, if the simulators were replaced by a spool piece and orifice and the same amount of fuel is assumed to be trapped by the break orifice, 0.102 m (4 in.) of solid lead shielding would be required. However, the physical configuration of the spool piece is a straight pipe and the shielding weight required to be supported would be reduced to approximately 0.91 Mg (1 ton).

At 8 kW/ft with scaled LPIS and HPIS into the lower plenum, the RELAP5/MOD1 code calculations indicate that the mass flow split is reversed while the velocities are the same as during the 16 kW/ft calculation with cold leg injection. Thus two-thirds of the total released mass is deposited in the broken loop simulators while only one-third is transported into the inlet plenum of the intact loop steam generator. Therefore, with lower plenum injection the intact loop steam generator shielding problem is reduced while the steam generator simulator shielding problem is increased.

Therefore, it is recommended that the broken loop steam generator and pump simulators be removed or bypassed for Experiment L2-6 for any experiment scenario being considered. Furthermore, it is recommended that lower plenum ECC injection be used rather than ILCL injection for reflood to minimize the transport of fuel particles to the intact loop steam generator during reflood.

5. CALCULATION OF LOCE WITH VALVE MANEUVERS PRIOR TO REFLOOD

This section discusses a calculation to determine the thermal-hydraulic response of the LOFT facility during a LOCE with the RABVs opened and the BLHL isolation valve closed prior to fuel rod failure and reflood. The calculation was executed by restarting at 110 s the previously discussed 8 kW/ft LOCE calculation with the broken loop simulators and by utilizing the indicated valve maneuvers.

At 120 s the RABVs were opened linearly over an interval of 21.75 s; at 145 s the BLHL isolation valve was closed linearly over an interval of 62.1 s. HPIS and LPIS injection into the lower plenum were again initiated when the cladding temperature of the average fuel rod in the center module reached 1200 K (1706°F).

Figure 22 compares calculated fuel rod cladding temperatures for the 8 kW/ft calculations with and without the valve maneuvers. These cladding temperatures agree well. However, a slightly lower heatup rate is calculated after the isolation valve commences closing. The lower heatup rate is caused by the slightly different core hydraulic conditions created by the different hydraulic resistance in the new flow path to the BST. Choking cannot occur in the new flow path during reflood since the pressure differentials across the RABVs are small during reflood and the piping from the BLHL to the BLCL has a large flow area. The main consequence of the lower heatup rate is a delay of 10 s to about 278 s for ECC injection initiation and corresponding delay for the fuel rod quench.

Although minor differences were calculated for cladding temperatures, no similar differences are shown in Figure 23 for calculated system pressures.

The advantages of an experiment scenario which utilizes the above valve maneuvers are to mitigate the problem of fuel particle transport into the broken loop simulators and to eliminate the cost and schedule impact of

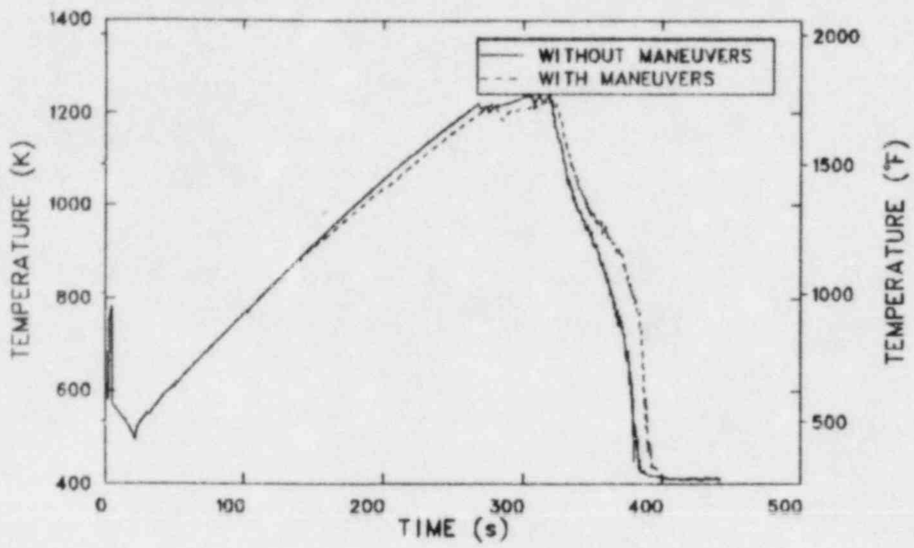

Figure 22. Cladding temperatures for 8 kW/ft calculations with and without valve maneuvers.

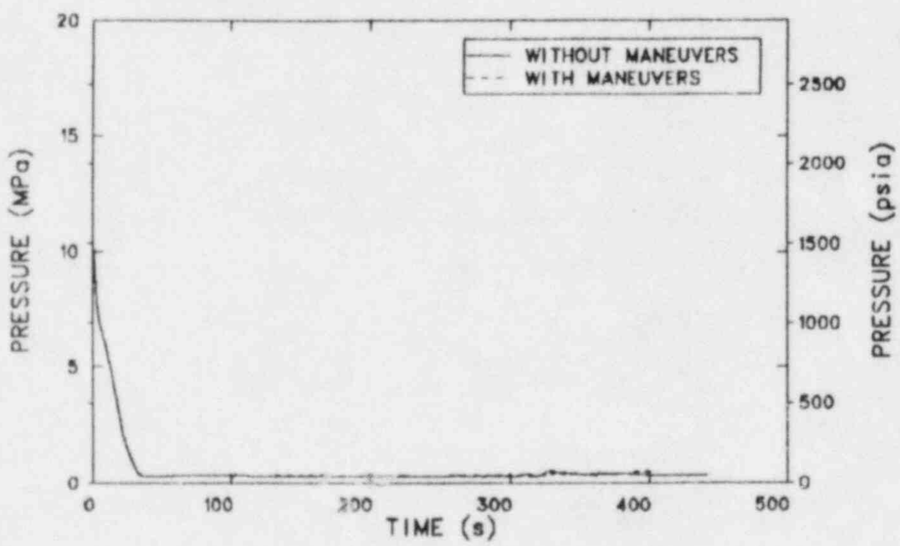

Figure 23. System pressures for 8 kW/ft calculations with and without valve maneuvers.

modifying the BLHL. Assuming that the amount of fuel particles deposited in the RABVs is insignificant, the shielding required for this experiment scenario is the same as discussed in Section 4 for an experiment with the broken loop simulators replaced by a spool piece containing a single orifice. Only the location of the shielding changes from the BLHL to the BLCL.

6. CONCLUSIONS AND RECOMMENDATIONS

Calculations were performed to determine the thermal-hydraulic responses of the LOFT facility during large-break LOCEs with and without the broken loop steam generator and pump simulators. The results of these calculations indicate that similar thermal-hydraulic responses of the LOFT facility are expected with and without the simulators during the blowdown and reflood phases of large-break experiments. The slight differences that do exist are insignificant, except for the reflood initiated by only the LPIS into the ILCL. If for Experiment L2-6 the simulators are removed and only the LPIS is used, it is recommended that this injection be into the lower plenum. Other than the injection location for low-flow ECC injection, the thermal-hydraulic impact of replacing the simulators with a spool piece containing a single orifice is minimal.

The fuel particle distribution calculations indicate that unshielded radiation field levels in excess of 101 R/hr at 0.9144 m (3 ft) after 60 days are anticipated from the fuel particles deposited in the broken loop steam generator and pump simulators during reflood. Since the location and configuration of the simulators create difficult shielding problems, removal of the simulators is warranted for Experiment L2-6. Furthermore, to minimize fuel particle transport to the intact loop steam generator during reflood, it is recommended that lower plenum ECC injection be used.

A calculation was performed to determine the thermal-hydraulic response of the LOFT facility during a large-break LOCE with the RABVs opened and the BLHL isolation valve closed prior to fuel rod failure and reflood. The results of this calculation indicate that similar thermal-hydraulic responses are expected during reflood with and without these valve maneuvers utilized. These valve maneuvers (a) mitigate the potential problem of fuel particle transport into the broken loop steam generator and pump simulators, (b) eliminate the cost and schedule impact of modifying the BLHL, and (c) have minimal thermal-hydraulic impact on reflood. Therefore, utilizing these maneuvers for Experiment L2-6 is recommended.

7. REFERENCES

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