Por

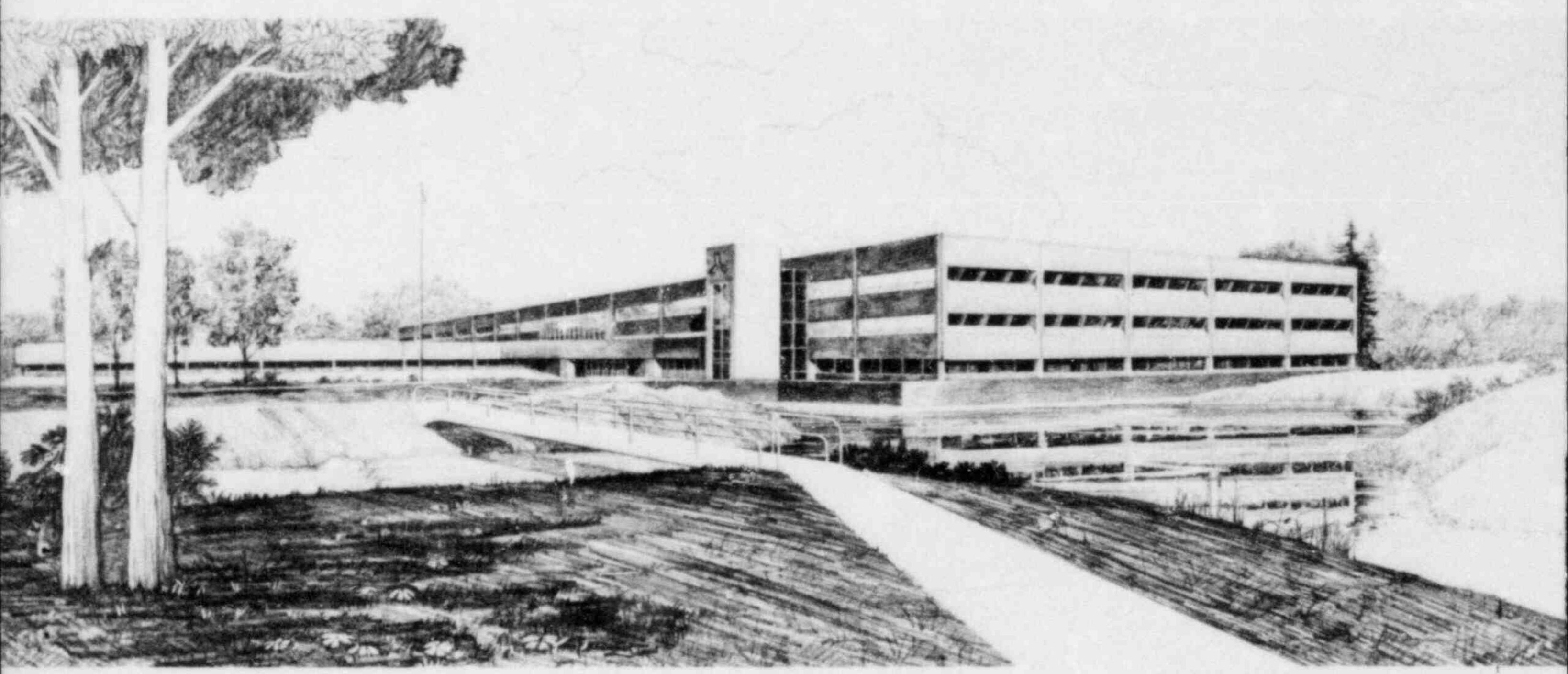
NRC Research and/or Technical Assistance Rept

EGG-LOFT-6095
October 1982

AN EVALUATION OF THE THERMAL-HYDRAULIC OBJECTIVES OF LOFT EXPERIMENT L2-6

E. A. Harvego
S. R. Behling
E. L. Tolman
M. D. Peters

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy



This is an informal report intended for use as a preliminary or working document

Prepared for the
U.S.NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6048

EG&G Idaho

EG&G Idaho, Inc

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

INTERIM REPORT

Accession No. ______

Report No. EGG-LOFT-6095

Contract Program or Project Title:
LOFT Program Division

Subject of this Document:
AN EVALUATION OF THE THERMAL-HYDRAULIC OBJECTIVES OF LOFT EXPERIMENT L2-6

Type of Document:
Experiment Planning Report

Author(s):

E. A. Harvego, S. R. Behling, E. L. Tolman, M. D. Peters

Date of Document:

October 1982

Responsible NRC Individual and NRC Office or Division: G. D. McPherson

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

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

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

INTERIM REPORT

ABSTRACT

This report presents the thermal-hydraulic experiment objectives for Experiment L2-6 to be conducted in the Loss-of-Fluid Test Facility. The results of several previous LOFT experiments are reviewed and the data from other test facilities are assessed to determine the additional need for thermal-hydraulic data which can be obtained by conducting the large break L2-6 experiment. Three blowdown and reflood scenarios in LOFT are investigated by using the RELAP5/MOD1 computer code to identify the initial power level and reflood conditions which will best address the thermal-hydraulic objectives.

SUMMARY

This report documents the results of a study undertaken to help define the specific thermal-hydraulic objectives of Experiment L2-6. The conclusions of this study will be weighed against the conclusions of a concurrent study that provides an independent evaluation of the specific fuel behavior objectives for Experiment L2-6. An integration of these two independent studies will then provide the experiment criteria for Experiment L2-6.

The L2-6 experiment is a 200% large-break LOCA. It is designed to address reactor licensing and safety concerns relating to system thermal-hydraulic response and fuel behavior under inadequate core cooling conditions. Data obtained from this experiment will provide information that can be used to address long term core coolability after fuel rod ballooning and rupture and could potentially address the degree of conservatism in licensing assumptions required in the calculation of thermal-hydraulic conditions during a large break LOCA. The key variables which can strongly influence the thermal-hydraulic behavior and are therefore included in this study are Maximum Linear Heat Generation Rate (MLHGR), the scaled ECC reflood rate following fuel rod ballooning and rupture, and the ECC injection location.

To aid in defining specific thermal-hydraulic objectives for L2-6, the availability and applicability of data from other facilities was assessed to determine the need for additional LOFT data. Also, calculations were performed using the RELAP5/MOD1 computer code for the following conditions (a) MLHGR of 53 kW/m with full scaled ECC injection into the intact loop cold leg, (b) MLHGR of 53 kW/m with scaled HPIS and LPIS injection into the intact loop cold leg, and (c) MLHGR of 26 kW/m with scaled LPIS injection into the lower plenum. The results of these calculations indicate that for either scenario at 53 kW/m fuel ballooning and rupture occur prior to 60 s with a possibility of fuel rod deformation occurring during the initial blowdown phase. For either scenario in which ECC was injected into the cold leg, the low primary system pressure at the time ECC injection is

initiated allows much of the ECC fluid to bypass the core, whereas ECC injection into the lower plenum more quickly refloods the core.

The conclusions drawn from the analysis of the results of this study indicate that all the desired thermal-hydraulic objectives for this test cannot be fully met by specifying a single MLHGR for Experiment L2-6. An initial MLHGR of 53 kW/m is not compatible with the existing end-of-life fuel rod prepressurization of 4.1 MPa and as a result there is a potential for fuel deformation during the initial blowdown phase, which compromises the objective of obtaining representative heat transfer and reflood characteristics. A MLHGR of 26 kW/m will, however, produce a core heatup rate which will maximize the fuel rod ballooning prior to rupture and allow adequate time for plant recovery using scaled ECC flow before unacceptably high fuel temperatures are reached.

The recommendation of this study is to conduct Experiment L2-6 from an initial MLHGR of 26 kW/m, or less, and to use scaled HPIS and LPIS injection into the reactor vessel lower plenum. The sole thermal-hydraulic objective recommended for Experiment L2-6 is to provide experimental data to determine the capability of scaled ECCs to restore and maintain cooling to a core in which the fuel cladding has undergone extensive ballooning and rupture.

CONTENTS

ABSTRACT i

SUMMARY ii

1. INTRODUCTION 1

2. DISCUSSION OF LICENSING ISSUES AND AVAILABLE DATA 4

   2.1 Blowdown Thermal-Hydraulics 4

   2.2 Core Reflood Behavior 12

   2.3 Core Cooling Following Fuel Rod Deformation 15

3. TEST OBJECTIVES AND EXPERIMENT CONDITIONS 20

4. THERMAL-HYDRAULIC PHENOMENA IN EXPERIMENT L2-6 24

   4.1 Results of Scenario 1 with 53 kW/m, Full ECC Into Intact Loop Cold Leg 24

   4.2 Results of Scenario 2 with 53 kW/m, LPIS and HPIS Injection Into Intact Loop Cold Leg 29

   4.3 Results of Scenario 3 with 26 kW/m, LPIS and HPIS Injection Into the Lower Plenum 32

   4.4 Summary of Calculational Results 34

5. CONCLUSIONS AND RECOMMENDATIONS 36

REFERENCES 39

FIGURES

1. Schematic of LOFT F2 center bundle instrumentation 2

2. Measured and predicted peak cladding temperature during the first 20 s in LOFT Experiment L2-3 7

3. Peak cladding temperatures compared to licensing calculations for LOFT Experiment L2-3 8

4. Peak cladding temperature comparison for LOFT Experiments L2-2, L2-3, and L2-5 11

5. Full cladding temperatures at six-axial core elevations (53 kW/m) 26

6. ECC injection flow rate for 53 kW/m case (into cold leg) 28

7. Fuel cladding temperature at six axial core elevations (53 kW)/m without ECC accumulator) 30

8. Average core fluid density (53 kW/m without ECC accumulator) 31

9 Fuel cladding temperature at six axial core elevations (26 kW/m without ECC accumulator, lower plenum injection) 33

10. Average core fluid density (26 kW/m, without ECC accumulator, lower plenum injection) 35

1. INTRODUCTION

Experiment L2-6 is currently scheduled to be the last test conducted in either the Nuclear Regulatory Commission (NRC) LOFT Program or the proposed LOFT Consortium Program. This test will simulate a hypothetical loss-of-coolant accident, caused by a 200% double-ended offset shear break in a cold leg of a four-loop commercial pressurized water reactor (PWR). During the test normal Emergency Core Cooling System (ECCS) injection will be delayed until after deformation and failure of rods in the center fuel assembly have occurred. The center fuel assembly for this test will be pressurized to 4.1 MPa (cold pressure), corresponding to end-of-life fuel conditions in a commercial PWR. Fuel failure will be limited to the center fuel assembly since the peripheral LOFT core fuel assemblies will be unpressurized.

The LOFT facility (described in Reference 1) is a 50 MW (t) nuclear PWR with instrumentation to measure and provide data on the thermal-hydraulic conditions throughout the system. The center fuel assembly for Experiment L2-6 has been specially instrumented to provide detailed information on fuel rod behavior, without influencing the fuel rod balloon and rupture characteristics during the test. The instrumentation arrangement in the pressurized center fuel module is shown in Figure 1. The specified instrumentation will provide detailed measurements of internal cladding temperatures, rod internal pressure, and time of rod failure. In general, the cladding internal thermocouples are located at 0.66 m (26 inches) above the bottom of the core (the expected elevation of cladding deformation). Some thermocouples, however, are also located above and below that elevation to provide an indication of the overall core thermal-hydraulic response throughout the transient.

Both thermal-hydraulic and fuel behavior information will be obtained from LOFT Experiment L2-6 that could be of use in confirming and improving current licensing regulations. However, the thermal-hydraulic and fuel behavior objectives produce conflicting requirements and conditions for

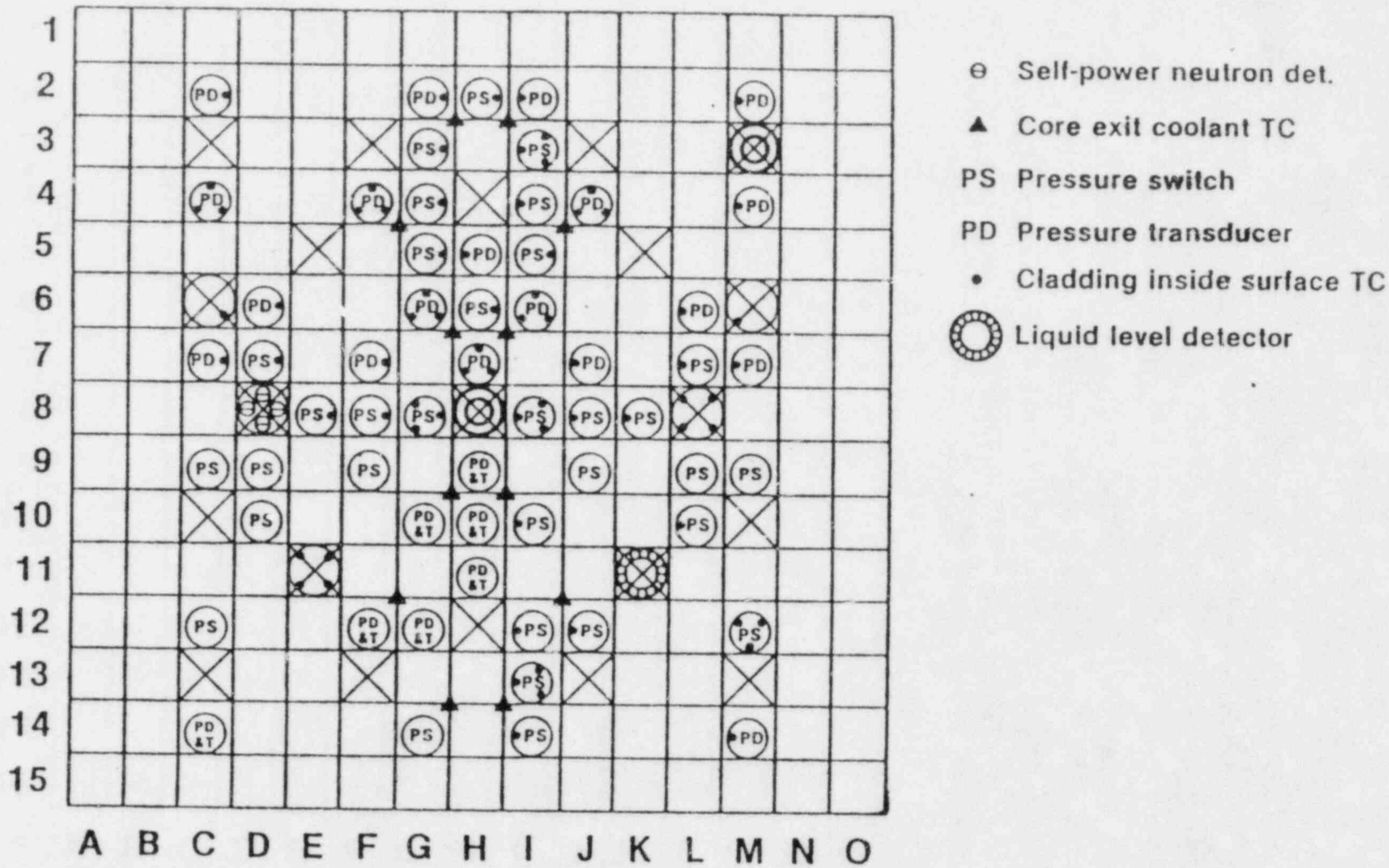


Figure 1. Schematic of LOFT F2 Center Fuel Bundle Instrumentation.

this test. To aid in specifying experiment requirements, two independent studies were undertaken to clearly define the separate fuel behavior and thermal-hydraulic objectives and tests conditions. In the fuel behavior area the main interest is in fuel behavior under inadequate core cooling conditions. To address concerns about long term core cooling after fuel rod ballooning has occurred, LOFT Test L2-6 will provide information from a large bundle in-pile LOCA test to determine cladding ballooning and flow blockage under representative PWR bundle geometry, rod power, and system conditions. The information will then be used to assess our understanding of cladding deformation and rupture, the existing analytical models for LOCA fuel behavior, and the existing out-of-pile data base.

The important thermal-hydraulic information to be obtained from Experiment L2-6 include peak clad temperatures, in-core heat transfer, reflood behavior, and core cooling (following fuel rod deformation). Since no previous large-scale integral tests have been conducted to the point of fuel damage, Experiment L2-6 represents a significant expansion in the experimental data base available for evaluating best estimate code capability and for quantifying conservatism in the Appendix K criteria.

The emphasis of this report will be on evaluating the test objectives and defining the test parameters that will best satisfy the experiment objectives. The specific parameters discussed in this report are (a) initial maximum linear heat generation rate (MLHGR), (b) ECCS injection locations, and (c) scaled core reflood requirements.

Section 2 of this report discusses current Appendix K licensing issues, briefly summarizes the existing data relating to key Appendix K assumptions, and defines additional experimental research needs. Section 3 propose specific thermal-hydraulic objectives and required experiment conditions for Experiment L2-6. Section 4 presents the results of thermal-hydraulic analyses performed for L2-6, and Section 5 presents the conclusions and recommendations of this study.

2. DISCUSSION OF LICENSING ISSUES AND AVAILABLE DATA

To license a nuclear plant, calculations must be performed to demonstrate that acceptance criteria in 10 CFR 50.46 are met. These licensing calculations must be performed using the assumptions and requirement in 10 CFR 50 Appendix K. When Appendix K was written, conservative assumptions were made to account for uncertainties in the calculation of PWR LOCA behavior. Among the primary uncertainties at that time were (a) core peaking factors, (b) metal-water reaction, (c) decay power, (d) system blowdown thermal-hydraulics, (e) core heat transfer during lower plenum refill, (f) core reflood behavior, and (g) core cooling following fuel rod deformation. Research conducted since Appendix K was written has led to a better understanding of margins in conservatisms in core peaking factors and decay power levels. Continued research is also contributing to the experimental data base for assessing other areas of conservatism in Appendix K. Despite the continued research, however, there is still a relatively small amount of data from a large integral facility such as LOFT to address conservatisms in licensing assumptions associated with (a) large break thermal-hydraulics during blowdown, (b) core reflood behavior, and (c) core cooling following fuel rod deformation. Each of these three areas is discussed in more detail below.

2.1 Blowdown Thermal-Hydraulics

To understand the conservatisms in Appendix K requires an understanding of the actual response of commercial plants during a hypothetical large break LOCA. The understanding of actual plant response is obtained through the use of best estimate codes, which are assessed using experimental data from many different facilities. The calculated response using these verified best estimate codes can then be compared with licensing calculations to estimate the conservatisms in Appendix K assumptions. Large break experiments performed in LOFT have played a major role in the assessment of these best estimate codes, particularly during the blowdown phase of a large break LOCA. These tests have helped to define conservatisms in licensing assumptions relating to initial stored energy in nuclear rods, blowdown heat transfer (early core rewet), and ECC bypass.

The blowdown phase of a large break loss-of-coolant accident is characterized by rapid coolant expulsion and primary system depressurization prior to activation of the emergency core cooling systems. During the LOCA blowdown phase, the potential exists for energy transfer from the core to the surrounding coolant since a significant coolant inventory still remains within the reactor vessel. The core heat transfer is controlled largely by the hydraulic behavior in the reactor vessel, particularly in the core region. The prediction of accurate core hydraulic behavior is perhaps the most difficult aspect of system thermal-hydraulic modeling because to predict the small pressure difference across the core region accurately, requires very precise modeling of the overall system response. Analytical experience has shown that most uncertainty in the overall system response is due to uncertainties in the break flow model and primary coolant pump degradation model. Experimental results suggest that 3-dimensional fluid behavior in the downcomer and fluid mixing in the lower plenum also give rise to the uncertainty in predicting the flow through the core region.

Despite the fact that only three large break nuclear experiments have been conducted in the LOFT facility, the data obtained from these tests have contributed significantly to the understanding of important parameters influencing the initial blowdown thermal-hydraulics in a large PWR. During the first LOFT large break experiment, L2-2[2], core stagnation occurred for only a short period, 1-2 s, just following initial departure from nuclear boiling (DNB). By five seconds into the transient, a significant core flow was established and was maintained for several seconds although gradually decreasing from 8 to 12 s as the system coolant was depleted and two-phase degradation of the primary coolant system pumps occurred. The coolant flow at the core inlet was not directly measured but has been estimated from other system measurement to be within the range of 400-1000 kg/m^2.[3,4,5] The effect of this core flow was to rewet the fuel rods and rapidly quench the cladding temperature at all locations within the core. During the quench period, a large portion of the fuel rod stored energy (50 to 60%) was transferred from the fuel rods,[6] thus reducing the initial energy which must be carried off by the ECCS. The rewetting hydraulics were not well predicted prior to the experiment,[7] however, posttest analysis indicates that the measured trend of upward core flow is predicted if the relative influences of break flow and pump behavior are properly matched to the experimentally measured reactor vessel inlet and outlet flows.[4]

The second large-break experiment, L2-3,[8] was nearly identical to L2-2 except for a 50% increase in the core power (MLHGR 40 kW/m) and showed almost exact duplication of the L2-2 blowdown hydraulic behavior. Although the previous L2-2 experiment provided insight into the controlling system hydraulics, the cladding rewets were still not universally predicted for the L2-3 experiment.[9] In general, however, the hydraulic predictions for the L2-3 experiment were closer to the measured reactor vessel flow data, and a cooling trend was predicted during the time of upward core flow from 5 to 10 s. Figure 2 compares the measured and predicted peak cladding temperatures for the first 20 s of the L2-3 experiment.

After the L2-3 experiment, it was generally accepted that the blowdown hydraulics were important in limiting the peak cladding temperature. Comparison with licensing calculations indicate that the conservatisms could be as high as 400 to 500 K as shown in Figure 3. In fact, the LOFT results indicate the blowdown phase to be more important in removing the initial fuel rod stored energy than the final reflood phase.[10]

Following these LOFT tests, extensive separate effects testing was conducted to confirm the LOFT results and better understand the dominant hydraulic behavior influencing the early rewet. The accuracy and selective cooling effects of the LOFT cladding surface thermocouples were extensively evaluated via separate effects experiments in the LOFT Test Support Facility;[5,11] in the COSIMA Facility in Germany under hydraulic conditions simulating the rapid high pressure quench behavior observed in LOFT; and in reflood type experiments conducted in the Halden research reactor (IFA 511 Series),[12] the German REBERA facility, the Swiss NEPTUN

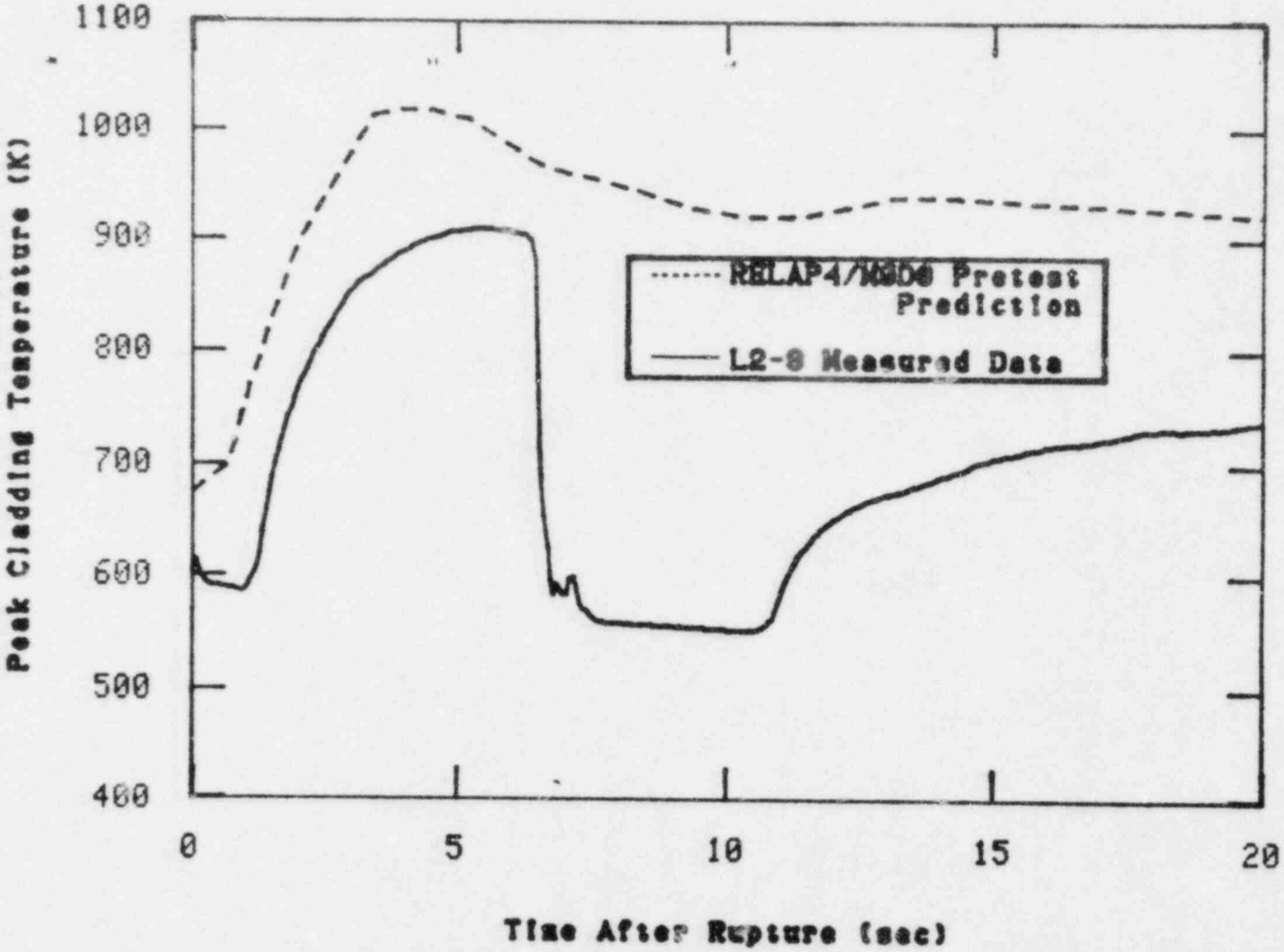


Figure 2. Measured and Predicted Peak Cladding Temperature During the First 20 s. in LOFT Experiment L2-3.

7

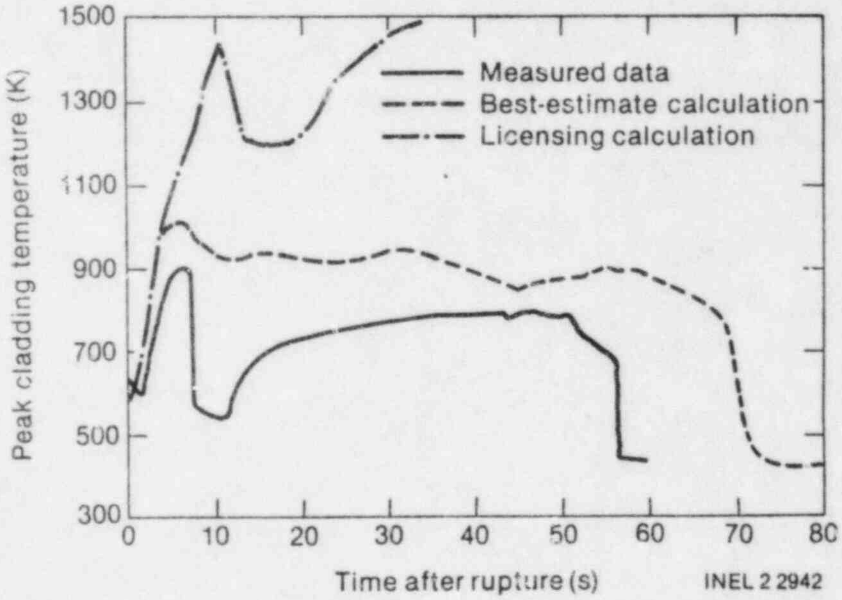


Figure 3. Peak Cladding Temperature Compared to Licensing and Best Estimate Calculations for LOFT Experiment L2-3.

facility, and many others.[13] The results of these experiments, simulating the LOFT blowdown quench hydraulics were that, (a) solid electrical heaters rods did not simulate the rapid nuclear rod quench thermal response,[5,14] and (b) surface thermocouples did not significantly affect the blowdown quench behavior of nuclear rods as observed in the LOFT L2-2 and L2-3 experiments.[5] Experiments conducted in the Power Burst Facility (PBF)[15] also confirmed these finding and showed that nuclear rods can quench within 2 to 3 s from temperatures as high as 1200 K.

To address the question of code capability to predict important hydraulic trends, computer studies were conducted to evaluate if the blowdown quench would be expected for a large PWR. Calculations were performed using both RELAP4/MOD6[16] and RELAP4/MOD7[17] and show that if a commercial four-loop PWR is modeled using the best-estimate assumptions that are required to predict the hydraulic response of the LOFT experiments, the large PWR is also predicted to experience cladding rewet during the blowdown phase of the design basis LOCA. In fact, because of scaling differences between LOFT and the full-size plant (downcomer size, lower plenum volume, and fuel rod length), the blowdown rewet is predicted to be even more dominant for the large plant. These results are significant since current licensing assumptions preclude the return to nucleate boiling (early rewet) until the start of reflood, which is generally predicted to be 40 to 50 s after rupture in licensing calculations.

Since the primary coolant pump operation was the key parameter influencing the LOFT blowdown quench, sensitivity calculations, varying the pump operating conditions during the transients, were conducted to determine if the core flow in a large plant could be stagnated for the entire blowdown period. This study[18] indicated that core stagnation would be difficult and that all four primary system coolant pumps would have to be severely degraded (one broken shaft and the remaining pumps unpowered during the transient). Similar sensitivity calculations for LOFT[19] show that a core flow stagnation could be achieved only if the

LOFT pump flywheel were disabled during the entire transient. These calculations were used to specify the experiment conditions in LOFT Experiment L2-5, which was run to determine if extended core flow stagnation could be achieved during blowdown. The L2-3 and L2-5 experiments were identical except the pump flywheels were disconnected at the beginning of the L2-5 transient, and for L2-3 the pump continued running during the transient. Extended core flow stagnation did occur during L2-5 with resulting higher cladding temperatures. Figure 4 compares the peak cladding temperature response from the L2-2, L2-3, and L2-5 experiments.

While previous LOFT large break LOCA experiments have contributed to our understanding of the blowdown thermal-hydraulic behavior in a commercial plant, several important issues remain to be resolved to complete our understanding of blowdown thermal-hydraulic phenomena in LOFT and to confirm conservatisms in Appendix K licensing assumptions. Specifically, it would be beneficial to conduct a LOFT test (a) to determine if the normal (no assumed pump failure) system rewet response occurs at the higher power levels (46-53 kw/m) at which some commercial plants are currently licensed to operate, (b) to determine if the codes predict reasonable peak cladding temperatures under core flow stagnation conditions at higher powers, (c) to assess the ability of codes to adequately predict larger scale system response for different pump characteristics, and (d) to provide additional data to adequately resolve differences in system response as measured in LOFT, Semiscale, and LOBI.

Since smaller scale nonnuclear experiments (Semiscale and LOBI) have shown significantly different large-break core thermal response than observed in LOFT,[20] it is important to clearly establish the larger-scale nuclear system response in LOFT as a basis for resolving system scaling effects and to provide the basis for extrapolating best-estimate computer codes to commercial-size plants.

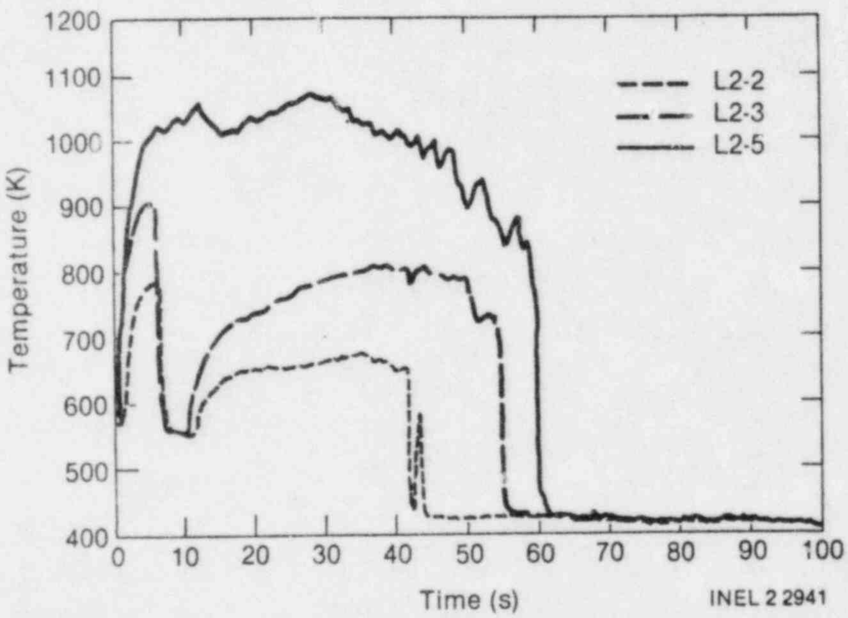


Figure 4. Peak Cladding Temperature Comparison for LOFT Experiments L2-2, L2-3, and L2-5.

2.2 Core Reflood Behavior

Licensing calculations that use Appendix K assumptions always predict the highest fuel rod cladding temperatures during the reflood phase of a large break LOCA. Experimental results from large-break LOCA experiments in LOFT, Semiscale, and other integral facilities, however, have shown that maximum cladding temperatures generally occur during the initial 10 s of large-break LOCA experiments. While these experimental results show the importance of the initial blowdown phase of a large break LOCA, they also point to the need to quantify uncertainties in reflood behavior, since the Appendix K reflood assumptions determine the peak power at which commercial plants are licensed. In addition, a better understanding of the mechanisms influencing reflood behavior in general, will also contribute to a better understanding of core cooling characteristics in blocked fuel rod arrays (discussed in the following section).

An early study by R. W. Shumway and R. T. French[21] attempted to quantify the conservatisms in current licensing assumptions by performing sensitivity calculations on each assumption. The conservatisms were determined by systematically relaxing the assumptions contained in a licensing-type baseline calculation. This study showed that peak cladding temperatures reached during the reflood portion of a design basis LOCA could be as much as 1458 K (810°F) lower than would be obtained in a licensing calculation. The single most influential assumption on cladding temperature during reflood was the restriction on the reflood hydraulics, and the use of 1.2 times the ANS Standard for shutdown decay heat. Major areas identified for further evaluation were liquid entrainment and carryover in the upper plenum, and the three-dimensional effects of hydraulic communication between different regions of the core. The conclusion of the report was that water reactor safety research emphasis should be directed towards a more mechanistic understanding of the factors governing core heat transfer during the reflood phase of a LOCA.

To better understand the heat transfer mechanisms involved in the reflood process, a number of reflood experiments have been conducted in the FLECHT, Semiscale Mod-1, and Semiscale Mod-3 facilities. Tests performed in the Semiscale Mod-1 facility investigated the effects of core inlet flooding rate, initial peak power, pressure, radial power profile, initial cladding temperature, and subcooling of injected ECC on reflood phenomena. Of the parameters investigated, the rod cladding temperature response was most strongly dependent on the core inlet flooding rate, peak rod power, and system pressure. The influence of these parameters on cladding temperatures, in turn, was related to their effect on liquid entrainment during reflood. The higher flooding rates resulted in higher steam flow rates and greater entrainment. The lower system pressures resulted in lower heat transfer rates because of (a) differences in the physical properties of the steam, (b) an increase in the size of the droplets entrained, and (c) an increase in steam super heat, which decreased the temperature difference between the rod surface and the fluid. At higher rod powers, quenching occurred at higher cladding temperatures because the higher powers resulted in a slower quench front progressive, but greater liquid entrainment. The Semiscale experiments demonstrated that the important mechanism influencing core temperatures was the effect of liquid entrainment on core heat transfer during the reflood process.

A study by R. G. Hanson[22] comparing results from FLECHT, Semiscale Mod-1, and Semiscale Mod-3 reflood experiments generally confirmed the results of the Semiscale Mod-1 tests and indicated that differences in reflood behavior for the different facilities were in part, related to factors influencing entrainment and the precursory cooling that occurred prior to quench. A significant finding of this report was that although differences in core length and axial power distributions between the 1.7-m long Semiscale Mod-1 core and the 3.8-m FLECHT and Semiscale Mod-3 cores resulted in different reflood characteristics, the effect of core length on reflood and quenching time was found to be proportional to the ratio of the core heated lengths. Therefore, reflood data from a short core, such as LOFT, could be interpreted with respect to results expected in a full length core.

Another important mechanism influencing reflood that has received a great deal of attention is the effect of grid spacers on both single and two-phase heat transfer during reflood. In single-phase flow, the fluid acceleration and subsequent deceleration downstream of the grid spacer locations cause local increases in heat transfer rates in the downstream region due to the creation of free stream turbulence and the separation and reestablishment of the fluid boundary layers. In dispersed flow heat transfer, the grids can cause enhanced two-phase heat transfer by interacting with the entrained droplets. The dominant grid effects are premature wetting of the grids and droplet breakup. The combination of these effects can cause additional liquid evaporation and desuperheating of steam downstream of the grids, with a resultant improvement in reflood heat transfer. The pronounced effects of the grids on two-phase heat transfer have been observed in the KFK Flooding Experiments with Blocked Arrays (FEBA) forced reflooding experiments and in the FLECHT-SEASET 21 rod bundle experiments. These experiments showed that in two-phase dispersed flow, the grids promoted heat transfer downstream of the grid locations, which then locally depressed temperatures at these axial locations.

Results from recently completed experiments in the National Research Universal (NRU) Reactor, support the above findings with respect to grid spacer effects, and also indicated that nuclear fuel rods tend to reflood faster than the electrical rods used in the separate effects reflood experiments. These results point to the need for additional reflood data, in nuclear fuel bundles, to help characterize the behavior of nuclear rods during reflood.

Results from LOFT Experiment L2-6 will not provide detailed information on reflood heat transfer, or the localized effects of grid spacers on core heat transfer and fluid flow characteristics. Results from this experiment, however, will provide information on the overall reflood behavior of a large nuclear fuel rod array. Since ECC injection will be delayed, the core region should be completely void of water at the start reflood. Because the liquid level measurement devices in the LOFT core region work best from an initially dry condition, the combination of these liquid level devices, cladding temperature measurements in the center and peripheral fuel assemblies, downcomer instrumentation, and flow measurements at the bottom of the core should provide a good indication of major system reflood thermal-hydraulic. Specific information that should be obtained from these measurements are the gravity-feed reflood rate, the hydraulic interactions between the downcomer and core, and the quench front progression following the initiation of reflood. Through posttest analysis, information on overall liquid entrainment characteristics, the potential for steam binding (caused by the evaporation of entrained liquid in the intact loop steam generator), and core heat transfer above the quench front should also be obtained.

Although fuel rod ballooning will occur in L2-6, the ballooning will be limited to the center fuel assembly. Therefore the overall reflood characteristics measured in the peripheral fuel bundles should supply information on the reflood behavior of beginning-of-life fuel bundles that have not ballooned but are adjacent to an end-of-life ballooned fuel bundle. A reflood of this type in LOFT would provide valuable information for evaluating the effects of (1) stored energy, (2) axial power profile (electric rods generally use a "stepped" power profile to simulate the axial power profile in a nuclear rod), and (3) gap conductance on the reflood behavior of a nuclear rod bundle relative to that for an electric rod bundle.

2.3 Core Cooling Following Fuel Rod Deformation

The current Appendix K rule for licensing light water reactors requires conservative assumptions for blocked fuel rod arrays at low reflood rates. Specifically, licensing calculations must be performed assuming steam cooling only if the PWR reflood rate is calculated to be less than 2.54 cm/s, and the effect of flow blockage on both local steam flow and heat transfer must be modeled if cladding balloon and rupture is predicted to occur Therefore, the principal areas of interest in evaluating core cooling following fuel rod deformation are (a) the heat transfer and flow regimes during reflood, (b) the potential for enhanced heat transfer in and downstream of the blockage, and (c) the effect of blockage on the redistribution of flow which could lead to reduced heat transfer in the vicinity of the blockage if the blockage is large enough.

To address these issues, flow blockage heat transfer tests were conducted in the FLECHT-SEASET 21 Rod Bundle Program.[23] The test section consisted of 21 full length (3.048-m heated length) electric fuel rod simulators which were internally heated with 1.66 peak-to-average chopped cosine axial power shape. Tests were conducted in an unblocked bundle (reference case), and in bundles with coplanar and noncoplanar blockages. Axial bundle wide blockages ranged from a minimum of 10% to a maximum of 60%. The blockages we simulated by sleeves attached to the heater rods. The results of the FLECHT-SEASET 21 rod bundle tests indicated that heat transfer improvement occurred for both single and two-phase flows in and downstream of the simulated blockages. The majority of the heat transfer improvement was immediately downstream of the blockage and the improvement diminished as a function of distance from the blockage.

Single-phase experiments conducted at Battelle Pacific Northwest Laboratory (PNL) in an unheated 7 x 7 rod array with standard PWR dimensions, indicated that for a 90% blockage of the center four channels, measured fluid velocities immediately upstream of the blockage were extremely low, and flow reversals were detected immediately downstream of the blockage. The flow recirculation and stagnation zone extended approximately five subchannel hydraulic diameters downstream of the blockage. The results showed that flow redistribution can occur when the subchannel blockages are large and there is an adequate bypass region.

The influence of subchannel blockage size and shape on reflood heat transfer is being investigated in the FEBA program using full-length, 5 x 5 electrically heated rod bundles. These tests are evaluating the combined influences of blockage and grids on reflood heat transfer during typical PWR reflood conditions. Preliminary results from tests with blockages of 62% and 90% indicated improved heat transfer downstream of the 62% blockage because of increased turbulence and droplet dispersion, but reduced heat transfer downstream of the 90% blockage because of flow redistribution around the blockage. The increased turbulence and droplet dispersion caused by grid spacers in the vicinity of the blockage also influenced local heat transfer. A more detailed discussion of the results of these tests and the PNL tests is contained in Reference 24.

Finally, tests were conducted in the NRU reactor to evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length PWR fuel rod bundle during the heatup, reflood, and quench phases of a LOCA. Reflood rates of 2.0 to 27.0 cm/s were performed. Results from these tests showed no observable cooling effect caused by cladding lift off and a decoupling of the cladding from the heat source. (A cooling affect caused by the decoupling of the cladding from the fuel heat source was observed in the FEBA tests). As mentioned earlier, however, there was evidence that the nuclear rods quenched much faster than predicted.

A LOFT test to assess core cooling following fuel rod deformation will produce fuel rod ballooning and rupture in the center fuel module similar to what might be expected to occur in the core of a large commercial PWR. The distribution of cladding deformation within a large PWR core will be primarily controlled by local cladding temperatures and rod internal pressures, and these parameters will, in turn, be controlled by local power, the core thermal-hydraulics, and burnup.

The first cycle (new) fuel is generally loaded around the core periphery which is also the lowest power zone. Fabricated fuel rod pressures are generally around 2.4 MPa (cold) and this pressure probably does not increase substantially during the first year (cycle) of operation. Peak cladding temperatures in the peripheral fuel rods are not expected to exceed 900 K prior to reflood during a hypothetical LOCA. Based on the behavior of the low pressure test rods during the PBF LOC-6 tests the low pressure rods in a PWR core would probably not experience significant deformation or rupture even if cladding temperatures reached 1050 K.

The second and third cycle fuel bundles are generally loaded in a checker board pattern within the central region of a PWR core, and peak cladding temperatures could be between 1000 and 1100 K within this region. Cladding deformation would probably vary substantially between second and third cycle bundles, and possibly even between rods within individual bundles primarily depending upon rod internal pressures which continually increase due to the release of volatile fission products. Cladding deformation of the second cycle rods probably would not be large or uniform between second cycle bundles, with only a few percent of the rods rupturing. Without specific knowledge of rod internal pressure and temperature histories it is not possible to confidently specify the extent or degree of cladding deformation within second cycle bundles. The third cycle rod bundles, however, would probably experience wide spread cladding ballooning and rupture because of increased internal pressures resulting from volatile fission gas release and provided peak cladding temperatures exceeded 1030 K.

Based on the cladding deformation scenario described above, a checker board pattern of subchannel blockage would be expected within a PWR core, with the third cycle bundles experiencing the maximum blockage. These (potentially) highly blocked third cycle bundles would be surrounded by relatively unblocked second cycle bundles. This configuration would, therefore, provide relatively low flow resistance for core flow redistribution around blockages within the third cycle fuel bundles. With the center fuel assembly pressurized and the peripheral fuel assemblies unpressurized in LOFT Experiment L2-6, fuel ballooning in the LOFT core should resemble the checkerboard ballooning pattern described above.

As indicated earlier, the instrumentation available in the LOFT core will not provide detailed information on local heat transfer and fluid flow characteristics in the vicinity of the blockage. However, the instrumentation should indicate the time of fuel balloon and rupture and the overall thermal-hydraulic conditions in the core during and after fuel rod deformation and failure has occurred. The extent of flow blockage, the potential influence of grid spacers on ballooning characteristics, and the affect of the center fuel assembly ballooning on overall core cooling characteristics will be determined through posttest examination of the center fuel assembly as well as the posttest analysis of experimental measurements made during the test.

3. TEST OBJECTIVES AND EXPERIMENT CONDITIONS

LOFT Experiment L2-6 has been proposed as a means to assess conservatisms in current assumptions used in the licensing of commercial PWRs. To aid in the quantification of these conservatisms, two specific thermal-hydraulic test objectives have been proposed for L2-6. These objectives are:

1. Provide experimental data to evaluate the conservatisms in Appendix K assumptions used in the prediction of peak clad temperature.

2. Provide experimental data to determine the capability of scaled ECCS to restore and maintain cooling to a core in which the fuel cladding has undergone extensive swelling and rupture.

Test Objective 1 is intended to confirm that, when Appendix K assumptions are used, calculated maximum fuel element temperatures will provide a conservative prediction of peak clad temperatures measured in the LOFT experiment. Specific conservatism to be addressed in LOFT Test L2-6 are (a) the effect of MLHGR on calculated peak clad temperature, (b) the effect of cladding rewet on peak clad temperature during blowdown, and (c) the effect of ECC bypass assumptions in Appendix K on post-CHF heat transfer.

To meet the second objective, LOFT Experiment L2-6 will address (a) the effect of fuel rod swelling and rupture on calculated flow blockage, (b) the ability to reestablish core cooling after fuel rod ballooning occurs, and (c) the ability to maintain long-term cooling after recovering from an inadequate core cooling situation.

The principal parameters believed to be most important in achieving the above test objectives and which can be controlled are the MLHGR and the scaled reflood rate following fuel rod balloon and rupture.

Three different power levels corresponding to MLHGRs of 53 kW/m, 40 kW/m, and 26 kW/m are considered in this study. To meet the first test objective, a MLHGR of 53 kW/m is desirable because Appendix K requires licensing calculations to be performed with the maximum peaking factors allowed by technical specifications. In addition, a range of power distribution shapes and peaking factors representing power distributions that may occur over the core life time must be studied and the combination which results in the most severe calculated consequences must be evaluated. Although these core peaking factors may never occur, some plants are currently licensed to operate at MLHGRs of 46 to 53 kW/m.

Data have been obtained from previous LOFT large break LOCA experiments at 26 and 40 kw/m. These results showed an early rewet phenomena (discussed earlier) that was not observed in smaller scale electric rod facilities (Semiscale and LOBI). Tests conducted to date have provided data for assessing the ability of codes to predict the rewet phenomena over a range of conditions which included pump operating characteristics, break size, and power level. The range of power levels investigated, however, has been limited to a maximum of 40 kw/m. While the basic system (thermal-hydraulic phenomena are not expected to be different at power levels of 46 to 53 kw/m, the effect of power level on the magnitudes of these phenomena may not be calculated well by the codes. Since the rewet phenomena have been shown to be sensitive to small changes in system behavior, to fully address the first objective, a LOFT test at 53 kw/m is desireable to complete the data base for assessing the codes over the full range of power levels at which plants are licensed.

Although there are some obvious advantages for conducting L2-6 from an initial MLHGR of 53 kW/m, there are also some significant disadvantages. First, a MLHGR of 53 kW/m is not consistent with the 4.1 MPa end-of-life fuel pressurization level already incorporated into the fuel bundle built for the L2-6 experiment. Therefore, although this MLHGR may provide the best demonstration of Appendix K conservatism used in licensing predictions of peak clad temperature during a large break LOCA, the potential for fuel rod ballooning (during the initial decompression phase) and the subsequent effect of post-CHF heat transfer would not be representative of what would occur in a commercial plant. A second reason for considering a lower power level is that results from the fuel behavior study for L2-6[24] indicate that once departure from nucleate boiling (DNB) has occurred and the stored energy in the rod has equilabrated the ensuing temperature rise (when ECC is delayed) should be as slow as practical (corresponding to a MLHGR of 26 kW/m or less) to produce the maximum fuel rod ballooning and flow channel blockage prior to rupture. Therefore, to adequately assess the capability of scaled ECCS to restore and maintain cooling in a core with extensive swelling and rupture (Objective 2) implies a MLHGR much lower than 53 kW/m is desirable.

Conducting Test L2-6 from an initial MLHGR of 40 kW/m was considered. However, two large break experiments, L2-3 and L2-5, have already been conducted at this power level. In addition, 40 kW/m is not compatible with the 4.1 MPa end-of-life fuel pressure nor would it provide a demanding test of Appendix K assumptions. Therefore, it was concluded that conducting L2-6 at an initial MLHGR of 40 kW/m would not adequately fulfill either of the thermal-hydraulic objectives defined above.

The third power level considered for L2-6 was 26 kW/m. This power level is desirable from the typicality standpoint because it is consistent with an end-of-life fuel pressure of 4.1 MPa (cold). As stated earlier, this power level would also produce the maximum fuel rod ballooning prior to rupture and therefore would best meet the requirements of the second objective. This power level, however, would not provide a very meaningful demonstration of the conservatisms in the Appendix K assumptions used to predict peak clad temperatures (Objective 1).

Due to the conflicting requirements described above, no single power level will fully meet both of the L2-6 thermal-hydraulic test objectives. Therefore, the final decision should be made by selecting the overriding objective for this test and defining the power level best suited for that objective. To aid in this decision process, calculations of possible test scenarios for L2-6 were performed at MLHGRs of 53 kW/m and 26 kW/m. Different ECC injection locations and flow rates were also considered in the analysis. The results from these analyses are presented in the following section.

4. THERMAL-HYDRAULIC PHENOMENA IN EXPERIMENT L2-6

Calculations of Experiment L2-6 were performed using the RELAP5/MOD1 computer program[25] in order to evaluate the thermal-hydraulic behavior of the LOFT system during various scenarios. The significant variables that have been discussed previously include MLHGR of 26 kW/m or 53 kW/m; ECC injection rates using accumulators, LPIS and HPIS; or using only LPIS and HPIS; and ECC injection location of either intact loop cold leg or lower plenum. From the possible combinations of these variables, three calculations have been performed. They are:

1. MLHGR of 53 kW/m using full ECC scaled injection into the intact loop cold leg.

2. MLHGR of 53 kW/m using scaled LPIS only with injection into the intact loop cold leg.

3. MLHGR of 26 kW/m using scaled LPIS and HPIS injection into the lower plenum.

These calculations were performed with the same input model and calculational techniques used for the L2-5 Experiment Prediction.[19] A complete discussion of the model as well as an assessment of the capability to accurately simulate the LOFT system is contained in Reference 19. It is felt that based on these assessments, the calculations presented in this section are sufficiently well understood to allow a reasonable decision on the merits of the alternate scenarios.

4.1 Results of Scenario 1: 53 kW/m, Full ECC Into Intact Loop Cold Leg

For the calculation of Experiment L2-6 with of a MLHGR of 53 kW/m and full ECC injected into the intact loop cold leg, several assumptions were used. The major assumptions included:

1. The primary coolant pumps (PCP) were powered until 30 s and were then tripped and assumed to coastdown normally.

2. ECC injection was initiated at 60 s into the transient immediately after fuel rod failure had occurred.

3. Fuel ballooning or rupture were not accounted for in either the hydraulic calculations of core flow or in the calculation of fuel rod surface temperature.

The first assumption is representative of commercial plants and is less likely to result in fuel cladding strain during the initial blowdown phase. The second assumption is based on the results of the FRAP-T calculation for the time of occurrence of fuel failure. The third assumption was required because RELAP5 does not have a channel blockage model.

The cladding surface temperatures at the six axial elevations for the highest power fuel rod are shown in Figure 5. An initial blowdown peak clad temperature of 1230 K occurred about 8 s after rupture. While the peak cladding temperature at this time is well below the maximum allowable peak cladding temperature of 1477 K (2200°F) specified in 10 CFR 50.46, it is in the range where fuel rod deformation could occur if a sufficient pressure differential across the cladding existed. Increased cooling of the high power elevations and an early rewet in the low power regions of the core occurred after the initial peak in cladding temperature. This early partial core rewet is caused by the primary coolant pumps forcing a high density slug of fluid through the core. The phenomena was seen in Experiments L2-2 and L2-3 and would be expected for this scenario. The peak cladding temperature during the second heatup is calculated to be 1280 K (1840°F) at 60 s. At 60 s a precursory cooling trend (cooling prior to quench) is noted throughout the core due to ECC injection. Based on previous comparisons with test data, it is felt that RELAP5/MOD1 calculates more cooling than would actually occur[19] and the peak temperature could be somewhat higher. (Note, that RELAP5/MOD1 does not account for metal-water reaction heat addition nor does it have a radiation heat

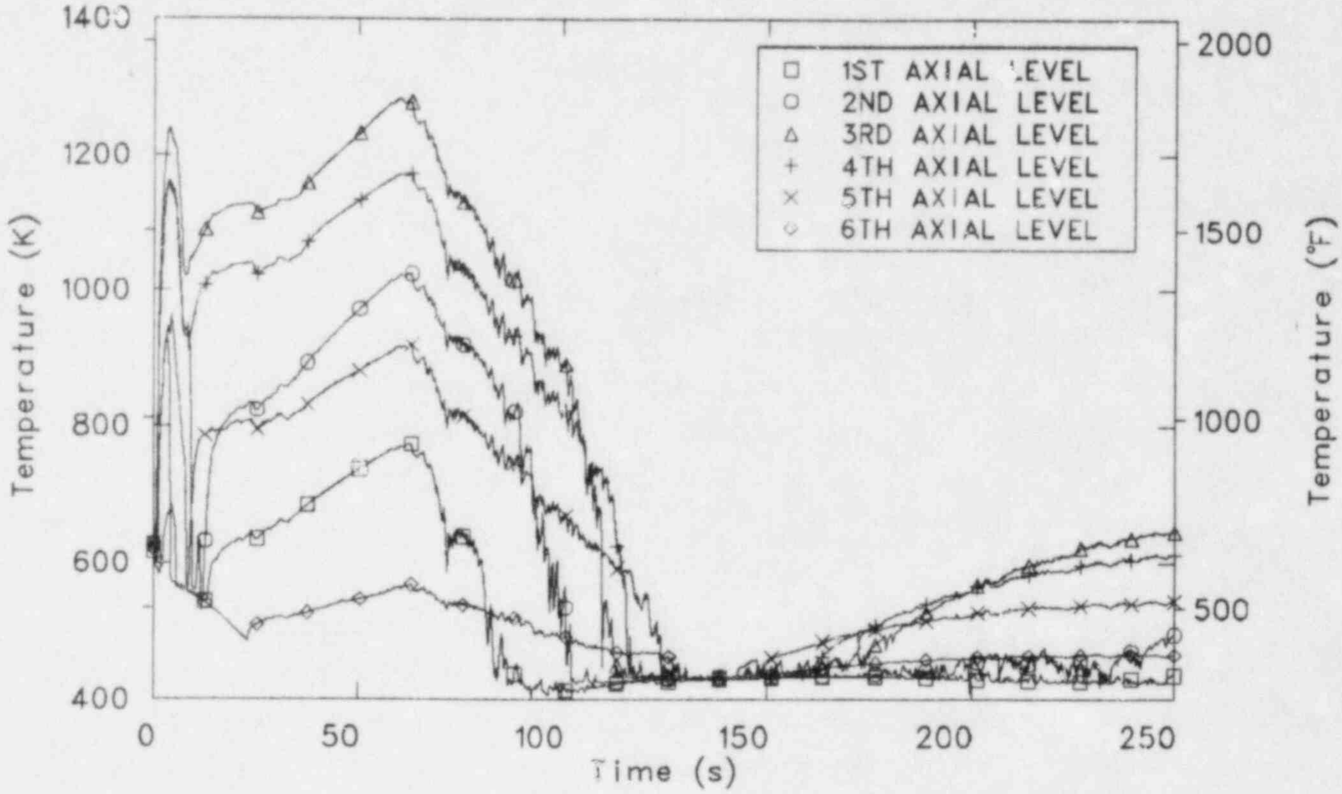


Figure 5. Fuel Cladding Temperatures at Six Axial Core Elevations (52 Kw/m).

transfer model.) Prior to 60 s, the heatup rate is about 5 K/s (9°F/s) which is higher than the .5 to 2 K/s heatup rate required to produce the maximum ballooning and core flow blockage.[24] A quench front proceeds up the core with the entire core quenched by 130 s. The quench front propogation velocity is approximately 0.02 m/s (0.8 in/s) until the peak power region is quenched and approximately 0.05 m/s (2 in/s) afterward.

After 150 s a second heatup is calculated to occur. This second heatup is not actually expected and is felt to be due to excessive entrainment calculated by RELAP5/MOD1. After 150 s, RELAP5/MOD1 calculates that most of the liquid entering the core is carried through the core and out the broken loop hot leg. The rest of the liquid in the core is calculated to boil off allowing the second heatup. The total ECC mass flow is presented in Figure 6. When the accumulator, which is pressurized to 4.1 MPa (600 psia), is allowed to inject at 60 s, the primary system pressure is only 0.3 MPa (44 psia). This large pressure differential causes a much more rapid injection than is typical and also causes the injected flow to be split between the vessel side of the injection point and the primary pump side.

Results from these calculations indicate that at a MLHGR of 53 kW/m with the pumps running hydraulic conditions in the core during the initial blowdown period are at the threshold of conditions required to produce early full core rewet. The results from a test at 53 kW/m would, therefore, contribute to the assessment of the ability of best-estimate codes to predict early rewet over the full range of operating conditions. In addition, results from this test would contribute to the data base for assessing conservatisms in Appendix K assumptions which preclude the return to nucleate boiling (early rewet) prior to the end-of-blowdown. The initial blowdown peak clad temperature calculated for this test (1230 K), however, is close to the temperature at which cladding deformation could occur. If clad ballooning were to occur, it would not be representative of fuel rod behavior in a commercial PWR because fuel rods capable of operating at power levels close to 53 kW/m (beginning of life) would be operating with internal fuel pressures much lower than the 4.1 MPa (cold) pressure of LOFT rods in the center fuel assembly.

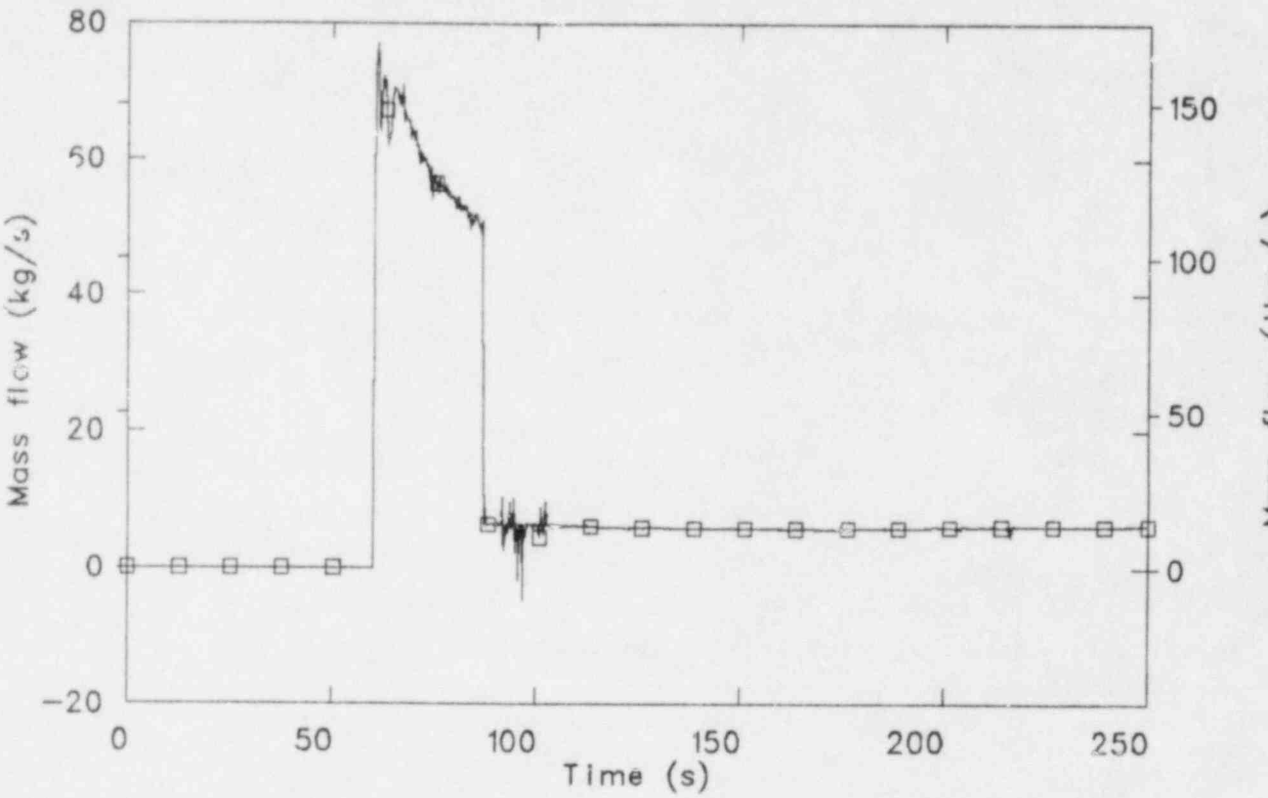


Figure 6. ECC Injection Flow Rate for 52 Kw/m Case (into Cold Leg).

As indicated in the calculations, the use of scaled accumulator injection resulted in atypical reflood behavior because of the large pressure differential between the accumulator tank and the primary system at the start of ECC injection. Therefore, a second calculation was performed assuming only scaled HPIS and LPIS injection into the intact loop cold leg.

4.2 Results of Scenario 2: 53 kW/m, LPIS and HPIS Injection Into Intact Loop Cold Leg

The same assumptions used in the calculations described in the previous section were used for Scenario 2 and the calculations are identical until the start of ECC injection at 60 s. The calculated high power fuel rod cladding surface temperatures for the six axial core elevations are plotted in Figure 7. After the beginning of ECC injection at 60 s, a slight decrease in the cladding heatup rate can be seen. After 80 s, a cooling of the cladding surface at all axial elevations is predicted to occur.

The time when the cladding temperatures begin to decrease is the same as when an increase in average fluid density occurs near the core high power region (Figure 8). Previous test results have shown that the interphasic mass transfer model in RELAP5/MOD1 is inaccurate and forces calculated vapor temperatures to be only slightly different than saturation if any liquid is present.

A more realistic formulation would allow substantial vapor superheat even if a small amount of liquid was present. The lower vapor temperature calculated by RELAP5 increases the heat transfer from the rod surface and promotes the cooling of the cladding. The actual temperature response after 80 s would probably be different, and would likely continue to increase for a significant length of time.

At 340 s the lowest axial elevation indicates a fuel cladding quench. None of the other elevations experience a quench by the time the calculation was terminated at 500 s. Excessive entrainment is also seen in

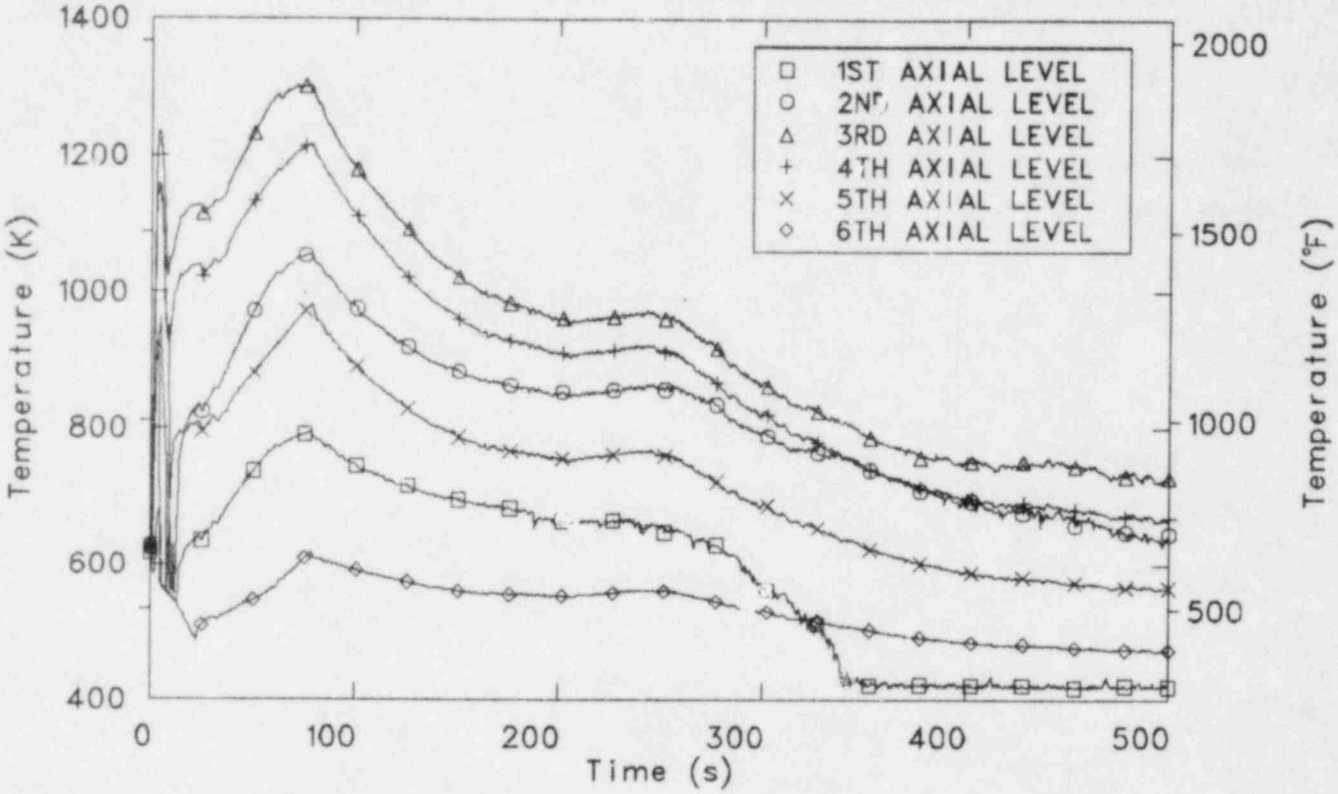


Figure 7. Fuel Cladding Temperatures at Six Axial Core Elevations (52 Kw/m without ECC Accumulator).

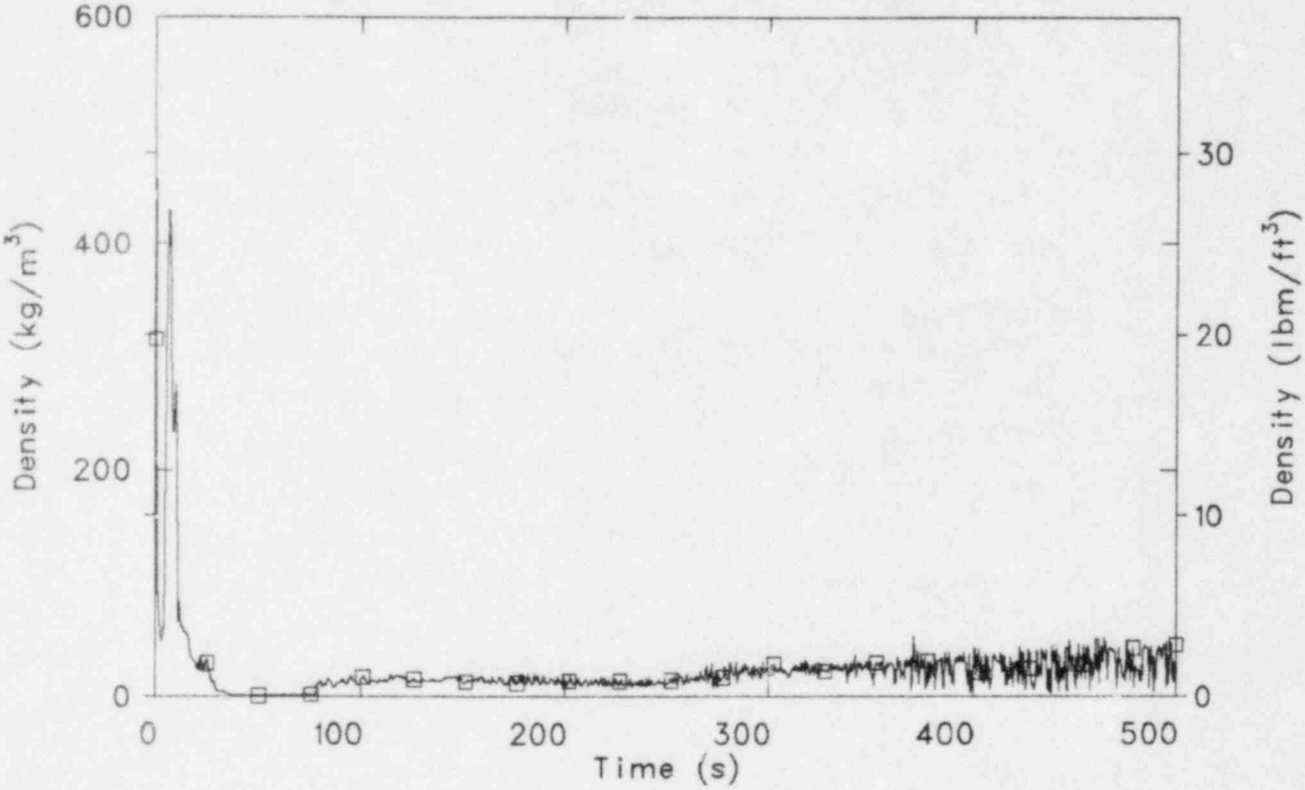


Figure 8. Average Core Fluid Density (52 Kw/m without ECC Accumulator).

this calculation which slows the core reflood. In addition, two-thirds of the fluid injected into the cold leg flows away from the vessel toward the PCPs, which causes a much lower core reflood than would occur if injection beg n at a typical time and pressure.

To produce more realistic reflood characteristics, a third calculation was performed in which scaled HPIS and LPIS were injected into the lower plenum. In addition, the calculation was performed using a MLHGR of 26 kW/m. As mentioned earlier, a MLHGR of 26 kW/m results in a more realistic end-of-life heatup rate to fuel failure and is expected to produce the maximum fuel rod deformation prior to failure.[24]

4.3 Results of Scenario 3: 26 kW/m, LPIS and HPIS Injection Into The Lower Plenum

For the calculation of the scenario using the MLHGR at 26 kW/m and LPIS and HPIS injection into the lower plenum, the assumptions of the previous calculations were used except that the ECC injection initiated when the maximum calculated peak cladding temperature reached 1200 K (1700°F). The resulting cladding surface temperatures are shown in Figure 9.

Because of the lower power level, early rewet over the entire core is observed. The maximum heatup rate after 20 s is approximately 3 K/s (5.4°F/s). The temperature increase is slowed following ECC initiation. The core is calculated to be completely quenched by 415 s with a quench front propogation velocity of approximately 0.02 m/s (1.7 in/s).

Again, the calculated entrainment is felt to be excessive and the actual quench front velocity would probably be somewhat greater.

For this scenario a much larger percentage of the injected flow actually penetrates the core. The average fluid density near the core hot spot is shown in Figure 10, indicating that portion of the core is full of liquid by 390 s.

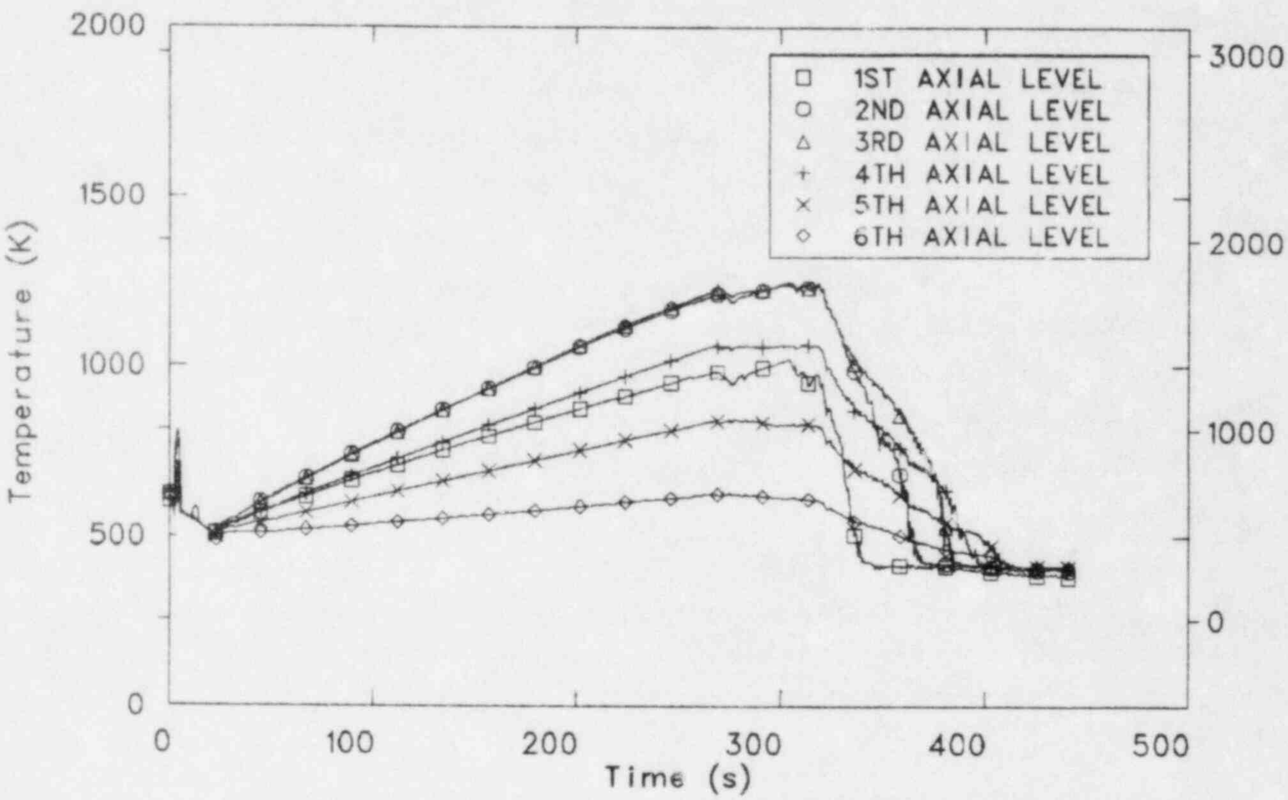


Figure 9. Fuel Cladding Temperatures at Six Axial Core Elevations (26 Kw/m without ECC Accumulator, Lower Plenum Injection).

4.4 Summary of Calculation Results

The descriptions in the previous sections are the results of calculations performed with the RELAP5/MOD1 computer program. Although there are known deficiencies in the RELAP5/MOD1 code, several general conclusions can be made:

1. For a MLHGR of 53 kW/m, fuel ballooning and burst would occur at approximately 60 s. There is also a possibility that fuel rod deformation could occur during the initial blowdown. A MLHGR of 26 kW/m would lead to the fuel failure at 270 s.

2. Injection of ECC fluid into the intact loop cold leg when the primary coolant system pressure is low causes much of the fluid to bypass from the reactor core. Injection of fluid into the lower plenum more quickly refloods the core.

3. For LPIS injection into the intact loop cold leg a significant potential exists for an extended period of core heatup because, as indicated above, at low system pressures, much of the injected ECC bypasses the reactor core.

4. Injection into the lower plenum improves the control of the experiment and produces a more typical reflood rate.

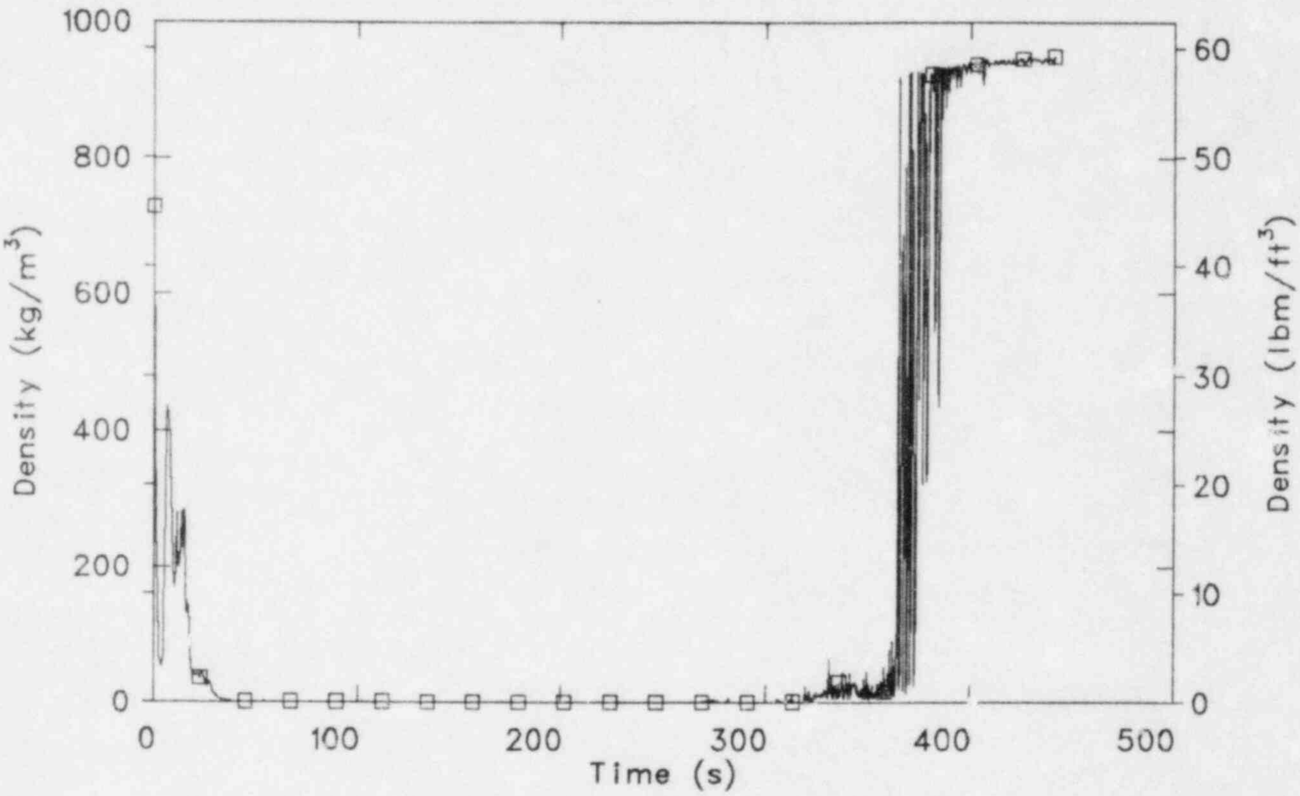


Figure 10. Average Core Fluid Density (26 Kw/m without ECC Accumulator, Lower Plenum Injection).

5. CONCLUSIONS AND RECOMMENDATIONS

Results from this study indicate that both thermal-hydraulic objectives defined in Section 3 of this report cannot be fully met by specifying a single MLHGR for LOFT Experiment L2-6. Conducting L2-6 at 53 kW/m would provide the most direct test of licensing calculations used to predict peak clad temperatures at the MLHGR currently used to license some plants. Results from a test at 53 kW/m would, therefore, provide a good data base for assessing conservatisms in Appendix K assumptions. However, the end-of-life fuel rod pressure selected for L2-6 (4.1 MPa) is not compatible with the beginning of life MLHGR of 53 kW/m required to fulfill the first test objective. For a cold fuel pressure of 4.1 MPa and a MLHGR of 53 kW/m, calculations indicate that there is a potential for fuel rod ballooning during the initial blowdown phase of the experiment. Should this occur, subsequent fuel behavior, and core reflood and heat transfer characteristics may not be representative of worst case cladding deformation which requires much slower heatup rates. Therefore, Objective 2 in Section 3 of this report would not be met.

From a code assessment standpoint, the principal reason for conducting L2-6 at 53 kW/m would be to determine whether the codes can predict the core hydraulic behavior that either caused or prevented early rewet in previous LOFT tests. Although a test at 53 kW/m would provide data over the full range of powers (previous LOFT large break tests were conducted at 26 kW/m and 40 kW/m), a substantial amount of data has already been obtained for assessing codes at different power levels, at different initial conditions, and for different pump operating modes. Therefore, the need for additional data (at a MLHGR of 53 kW/m) for best estimate code assessment purposes must be weighted against the potential for compromising the Appendix K objectives of maximum cladding deformation and flow blockage.

Conducting L2-6 from an initial MLHGR of 26 kW/m, or less, would not provide a good demonstration of conservatisms in Appendix K assumptions (Objective 1 in Section 3) since this MLHGR is considerably below that used in licensing calculations. Since a previous LOFT large break experiment

(L2-2) has already been conducted at 26 kW/m, the principle value of conducting the initial blowdown phase of L2-6 from an initial MLHGR of 26 kW/m, or less, would be to demonstrate the repeatability of LOFT large break experimental results.

To fulfill the second test objective, results from the fuel behavior study[24] and this report indicate that Experiment L2-6 must be conducted from an initial MLHGR of 26 kW/m or less. This MLHGR will produce a core heatup rate of approximately 3 K/s or less in the latter stages of the test, which will maximize fuel rod ballooning prior to rupture. A test at 26 kW/m will therefore provide the most demanding conditions for assessing the capability of scaled ECCS to restore and maintain cooling to the core following fuel rod balloon and rupture.

From a plant control standpoint, conducting L2-6 from an initial MLHGR of 26 kW/m should result in fuel rod failure between 1000 and 1100 K.[24] Fuel failure at this temperature should allow adequate time to recover the plant using scaled ECCS before temperatures reach the point where metal-water reactions will occur (~1255 K). Calculations at 53 kW/m[3] indicate that temperatures may reach 1100 to 1200 K before fuel rod failure occurs. At 53 kW/m, the higher heatup rate (relative to the 26 kW/m case) and the higher expected fuel failure temperature increases the likelihood that the LOFT experiment control system (LECS) will be required to recover the plant. If this occurs, considerably more than the scaled ECC will be injected into the LOFT vessel, and the thermal-hydraulic objectives of this test would not be met.

Based on the above discussion the recommendation of this study is that LOFT Experiment L2-6 be conducted from an initial MLHGR of 26 kW/m or less. Since a MLHGR of 26 kW/m will not meet the requirements of Objective 1 in Section 3, this objective should be eliminated. Therefore, the sole thermal-hydraulics objective of Experiment L2-6 should be:

> To provide experimental data to determine the capability of scaled ECCS to restore and maintain cooling to a core in which the fuel cladding has undergone extensive swelling and rupture.

Finally, to provide the most meaningful evaluation of ECC injection capabilities, scaled HPIS and LPIS injection should be used to produce the minimum realistic reflood rate. Core area scaling should be used to properly characterize the quench front progression. In addition, it would be desirable to correct the scaled reflood rate by the ratio of the LOFT-to-commercial plant core lengths so that the time frame in which the hot regions of the two different length cores quench will be equivalent. To simplify the conduct of the experiment, ECC injection should be into the lower plenum.

REFERENCES

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

2. McCormick-Marger, M., Experimental Data Report for LOFT Power Ascention Test L2-2, NUREG/CR-0492, February 1979.

3. J. C. Linn, Posttest Analysis of LOFT Loss-of-Coolant Experiment L2-3, EGG-LOFT-5075, March 1980.

4. Grush, W., et al., Results of Predictions for Scaled Nuclear Large Break Loss-of-Coolant Experiment, ASME Publication 80-WA-HT-49.

5. Gottula, R. C., Effects of Cladding Surface Thermocouples and Electrical Heater Rod Designs on Quench Behavior, NUREG/CR 2691, November 1982.

6. E. Tolman, et al., Fuel Rod Response for LOFT Test L2-2, EGG Report LTR 20-97, April 1979.

7. W. Grush, et al., Best Estimate Experiment Predictions for LOFT Nuclear Experiment L2-2, L2-3, and L2-4, EGG Report LOFT-TR-101, November 1978.

8. P. Prassinos, et al., Experiment Data Report for LOFT Power Ascention Experiment L2-3, NUREG/CR 0792, July 1979.

9. E. Kee, W. Grush, Best Estimate Prediction for LOFT Nuclear Experiment L2-3, EGG Report EP-L2-3, April 1979.

10. E. Tolman, E. Copryell, Review of LOFT Cladding Temperature Response for L2-2 and L2-3: Recommendations for Improved LOFT Fuel Rod Measurements, EGG-LOFT-5244, October 1980.

11. Gottula, R., et al., The Effect of Cladding Surface Thermocouples on the Quench Behavior of an Electrical Heater Rod, EGG Report LO-00-80-115, March 1980.

12. Kellpe, S., Observatioh on Halden IFA-511.3 Test Data and Results, EGG-LOFT-5520, January 1982.

13. L. S. Tong, "Minutes of the NRC Denver Workshop on Rewet Phenomena (April 11, 12, 1980)" NRC Letter, May 15, 1979.

14. S. Aksan, Evaluation of Analytical Capability to Predict Cladding Quench, EGG-LOFT-5555, August 1982.

15. R. W. Garner, PBF Thermocouple Effects Tests, Test Series TC-4 Quick Look Report, EGG-TFBP-5465, August 1981

16. L. Winters, Large Break Transient Calculations in a Commercial PWR and LOFT Prototypicality Assessment, EGG-LOFT-5093, April 1980.

17. T. deBoer, RELAP4/MOD7 Commercial PWR Large Break Transient Analysis Compared with LOFT Data, EGG-LOFT-5401, June 1981.

18. P. Demmie, Primary Coolant Pump Effects on Core Thermal Response During Large Break LOCA Transients in a Commercial Pressurized Water Reactor, EGG-LOFT-5505, July 1981.

19. P. Demmie, et al., Best Estimate Predictions for LOFT Nuclear Experiment L2-5, EGG-LOFT-586 9, May 1982.

20. W. Driskell, A Comparison of Thermal-Hydraulic Response of Three Large Break Test Facilities, EGG-LOFT 5685, November 1982.

21. R. W. Shumway and R. T. French, "Effect of Licenscing Assumption on Calculated PWR LOCA Core Thermal Response and Recommendations for Reactor Safety Research," Proceedings of Topical Meeting on Thermal Reactor Safety, CONF-770708, Vol. V, pg. 1, July 1977.

22. R. G. Hanson, Comparison of Thermal-Hydraulic Response in the Semiscale, THTF, and FLECHT Facilities, NUREG/CR-207, EGG-2093, December 1981.

23. M. J. Loftus and L. E. Hochreiter, "Flow Blockage Heat Transfer in the FLECHT-SEASET 21-Rod Bundle Tests," Joint NRC/ANS Meeting on Basic Thermal-Hydraulic Mechanisms in LWR Analysis, September 14-15, 1982.

24. J. M. Broughton, P. E. MacDonald, D. L. Hagrman, Fuel Behavior Experiment Requirements for LOFT Test L2-6, EGG-TFBP-6077, October 1982.

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