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MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Saul Levine, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER - # 63 LOFT REACTOR SAFETY PROGRAM RESEARCH RESULTS FROM NUCLEAR LOSS-OF-COOLANT EXPERIMENTS L2-2 AND L2-3

1.0 INTRODUCTION

This Research Information Letter transmits the significant results that have been obtained from the LOFT Reactor Safety Research Program from October 1, 1978 through June 1, 1979. During this time, two nuclear loss-of-coolant experiments, L2-2 and L2-3, were conducted successfully.

The LOFT facility (Reference 1) consists of a complete operational pressurized water reactor (PWR) designed to operate over the range of power densities of commercial PWRs and to simulate loss-of-coolant accidents (LOCA). The LOFT system and core configurations are shown in Figures 1 and 2. The LOFT Research Program (References 2, 3) has been developed to provide experimental information relevant to the licensing criteria for large commercial PWRs. The major portion of this program is directed at an improved understanding of the LOCA and the performance of emergency core cooling systems using thermal-hydraulic, core physics, structural and fuel behavior data obtained through loss-of-coolant experiments (LOCEs).

This letter is based on data obtained from the first two nuclear LOCEs conducted in the LOFT facility. The two are part of the L2 experiment series in which the effects of a complete offset shear of a primary coolant pipe are studied. L2-2 was conducted at a maximum linear heat generation rate of 26.4 $\pm$ 2 kW/m which is approximately 2/3 the nominal value of commercial PWRs. L2-3 was conducted at the nominal maximum PWR value of 39.0 $\pm$ 3.0 kW/m. A description of the system configuration and initial conditions for these experiments is contained in Table I.

2.0 SUMMARY

This letter reviews the objectives of the LOFT program and reports experimental and analytical results which address those objectives. The experimental results are from the two nuclear powered full-sized double-ended cold-leg break tests, L2-2 and L2-3. Extrapolation between these test results and those of the zero-powered L1-5 test provide the following conclusions and information which apply to the full range of operating powers equivalent to a maximum linear heat generation rate of zero to 39.4 kW/m (0-12 kW/ft), when there is no loss of offsite power.

The thermal-hydraulic phenomena associated with a full-sized double-ended cold-leg break result in a double reversal of core flow during depressurization. Upon the return to positive core flow, the flow is sufficient to quench the core prior to the initiation of Emergency Core Coolant (ECC) injection.[1] Improvements in code models and nodalization are identified which permit a good prediction of the responsible phenomena.

When ECC from the accumulator begins to enter the cold-leg, local condensation causes a pressure reduction and a consequent flow from the upper plenum into the core. At lower initial powers, this flow is sufficient to quench all fuel which is in film boiling; at higher initial powers, only the upper portion of the core is quenched.

ECC bypass, up until the end of accumulator flow, increases moderately with initial power to a full-power value of 36 percent. Reactor vessel liquid content never decreases below 40 percent of maximum. This is somewhat greater than best-estimate predictions and compares to 100 percent depletion predicted by evaluation model calculations. Since both bypass and minimum inventory affect the time to reflood and consequently the peak clad temperature reached during reflood, these results provide a measure of conservatism in best-estimate and evaluation model calculations.

The largest hydraulic loads associated with a large break occur in a plant initially at hot standby, and decrease with increasing power. The majority of the hydraulic loading energy content is in the frequency range below 40 Hz. Structural design methods, similar to those used for commercial PWRs, are shown to have a large margin of safety.

All four of the best-estimate codes used to predict L2-3 predicted generally higher cladding temperatures than those measured.

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[1] The absence of a significant quench in the Semiscale counterpart tests is being assessed.

Improvements in RELAP4/MOD6 input parameters and system models have led to predictions of system hydraulics and cladding temperature response which agree well with L2-2 and L2-3 measurements. When applied to a commercial PWR, this improved code predicts very similar thermal-hydraulic responses. Scaling differences between LOFT and the commercial plant, such as core flow, core length, and steam generator configuration, are shown to have no significant effect on the conclusions reached.

As a result of the nuclear experiment results obtained thus far and as a result of the recent TMI accident, a new test sequence is being formulated which advances several small break and transient tests to begin this year, and postpones the remaining tests in this large break series until late 1980 and 1981. Information from small break and transient tests will be sent to you via Research Information Letters beginning in early 1980.

3.0 BACKGROUND

3.1 LOFT Program Objectives

The specific LOFT program objectives are:

1. Provide integral system experimental data to the U.S. Nuclear Regulatory Commission (NRC) and the nuclear industry for the assessment and development of analytical methods used to predict:

   a. The transient thermal-hydraulic, mechanical, and nuclear response of the reactor system and primary system components under LOCA and anomalous transient conditions.

   b. The capability of current Emergency Core Cooling System (ECCS) designs to fulfill their intended function.

   c. The margin of conservatism inherent in the capability of current ECCS designs.

   d. The effectiveness of alternate ECCS concepts.

2. Investigate thresholds or unexpected phenomena that could affect the validity of the analytical models used to predict the thermal-hydraulic, mechanical, and nuclear response of the reactor system.

The first LOFT experiment series, designated the L1 series, was nonnuclear in nature and served to (a) evaluate and verify the structural integrity of the LOFT system; (b) provide data for evaluation and development of analytical models used to predict the hydraulics in response to large pipe breaks; and (c) provide operational experience for the simulation of large pipe breaks in PWR systems at powered conditions. Several nuclear test series have been defined, one of which is the L2 series, which is intended to provide integral system experimental data related to the full size double-ended cold-leg break at ascending power levels and with cold-leg ECC injection. This letter is based on the results of two of the L2 series experiments, L2-2 and L2-3, which are the first fully integrated PWR systems loss-of-coolant experiments to be conducted in the world.

3.2 Scaling

The LOFT system is designed to "scale" significant features of a four-loop commercial PWR and to reproducibly simulate typical system transient response to a LOCA. The scaling rationale (Reference 5), applied to LOFT, makes extensive use of principles that have been applied in a wide range of experiments within and beyond the nuclear power industry. The general scaling rules applied in LOFT are as follows:

Fuel linear heat generation rate is full scale. The nuclear fuel design has the same 15 X 15 geometry as commercial reactor fuel. Differences from the commercial fuel are length (1.68m), lower fuel density and fuel pins not prepressurized (an experiment using prepressurized fuel is planned).

When feasible, core power is taken as the basis for scaling of component volumes; that is,

$$\text{LOFT Volume} = \frac{\text{LOFT Power}}{\text{PWR Power}} \times \text{PWR Volume};$$

Flow areas are scaled to provide similar mass fluxes.

The ratio of break area to system volume is set identical to the commercial PWR value under study. Hence, the primary coolant system percentage water inventories vary in the same way with time.

Initial conditions (pressure, temperature, mass flux/core power) are set identical to the commercial PWR values.

LOFT is actually one of three systems that are related by this scaling rationale. The Semiscale facility (Reference 6) is a scale model of LOFT using about the same scale ratios as were used in scaling LOFT to the commercial PWR. The major scaling parameters for LOFT, Semiscale, and the commercial PWR are summarized in Table II.

The design and operation of the LOFT facility ensure that all the significant phenomena occur in approximately the same magnitude and time sequence as would occur in a commercial PWR LOCA. Assessment of the scaling rationale is accomplished by: (a) comparison of LOFT experimental results with results of counterpart experiments conducted in the Semiscale facility, and (b) applying the same modeling techniques to LOFT and the commercial PWRs, with LOFT LOCE initial conditions, and evaluating the comparisons.

3.3 LOFT Program Activities

The LOFT Program consists of both experimental and analytical phases. The experimental phase consists of the planning, preparation, and conduct of the experiments. This phase culminates in the acquisition of data and analysis of results that fulfill the objectives of the LOFT Program. Planning LOFT experiments includes evaluation of Semiscale counterpart experiments and other Semiscale experiments involving parametric variations. The experimental data obtained from L2-2 and L2-3 are contained in References 7 and 8, respectively. Semiscale counterpart experiment data are reported in References 9, 10, and 11.

Concurrent with the experimental program, supporting analysis provides preexperimental prediction and post-experimental analysis for the purpose of developing and refining code models and identifying areas for additional code development. Thermal-hydraulic analysis of L2-2 and L2-3 was carried out principally with the RELAP4/MOD6 (Reference 12) and FRAPT4 (Reference 13) codes by EG&G Idaho, and with the TRAC-P1 (Reference 14) and TRAC-P1A (Reference 15) codes by Los Alamos.

3.4 LOFT Data Uncertainties

In general, the uncertainties in the principal measured variables are as follows:

temperature	± 3 K	1.0%
pressure	± 0.03 MPa	2.2%
differential pressure	± 0.01 MPa	0.2%
density	± 0.03 Mg/m	3.75%
momentum flux	± 12.0 Mg/m.s	20.0%
velocity	± 2.7 m/s	13.5%

Techniques and instruments are well developed for measurements of the first four variables; consequently, these measurements are relatively accurate. However, the fuel cladding temperature measurements can be up to 30 K low during transient conditions due to hydraulic influences on the cladding external thermocouple. The last two variables, momentum flux and velocity, are difficult to measure in two-phase flow conditions. The uncertainties stated for these variables reflect this difficulty and represent the largest uncertainties which occur during low quality fluid conditions. Full-scale flow calibration work, now in progress, is expected to reduce these uncertainties.

4.0 RESULTS

The results obtained from L2-2 and L2-3 represent a significant achievement in the NRC's reactor safety program. After many years of planning, construction, and experimentation, the first experiments have been completed wherein the integrated effects from a loss-of-coolant accident in a fully operational PWR have been evaluated. The experimental data obtained from these first two experiments have many important implications on the early thinking (Reference 16) which formed the basis of the licensing criteria in effect today. The data also have implications on the expectations of thermal-hydraulic phenomena in commercial systems subjected to similar loss-of-coolant accidents and on the safety margins designed into the systems in accordance with the licensing criteria. The experimental data, in conjunction with earlier experiments discussed in RIL #37 (Reference 4), cover the entire operational range of commercial PWRs from hot standby conditions with the reactor shut down to nominal full power operating conditions (v.i.z. maximum linear heat generation rate of 39.4 kW/m). All of the experimental data thus far obtained is indicative of the effects resulting from the largest possible pipe break -- the double ended offset shear of a primary coolant cold-leg pipe. The results of L2-2 and L2-3, and of the nonnuclear test, L1-5, are presented in the succeeding sections along with their relevance to the scaling rationale, expected commercial PWR thermal-hydraulics and aspects believed to be relevant to regulatory licensing criteria.

4.1 LOFT LOCE Thermal-Hydraulics

The thermal-hydraulic transients in experiments L2-2 and L2-3 are quantitatively described by the sequence of events given in Table III and by the summary of phenomena results given in Table IV. The information in Table III and IV, along with the initial conditions defined in Table I, provides a clear description of the loss-of-coolant phenomena resulting from a double ended offset shear in the cold leg of a PWR primary coolant piping loop.

The chronology of events shows very similar behavior between the two nuclear experiments and also very similar behavior between nuclear and nonnuclear experiments for those events not strictly associated with core power. All conclusions from the nonnuclear series and reported in RIL #37 (Reference 4) were verified in the L2-2 and L2-3 tests.

The cessation of fuel cladding temperature rise and subsequent core-wide return of fuel cladding temperature to fluid saturation temperature within the first 10 seconds of the transients were the dominant events that influenced the similar sequence characteristics which occurred subsequent to the first 10 seconds in both the nuclear and nonnuclear experiments. These thermal phenomena are the result of primary coolant system hydraulic phenomena that are dominant and which control and limit the fuel cladding maximum temperature to well below damage thresholds. As the chronology shows, the cladding returns to fluid saturation before actuation of any of the ECC systems in both L2-2 and L2-3. This demonstrates that for the case of no loss of offsite power the hydraulic phenomenon that causes the core-wide return of cladding temperature to fluid saturation temperature is sufficiently dominant to overcome the thermal driving force for maximum linear heat generation rates up to the nominal PWR value of 39.4 kW/m.

The hydraulic limitation of the fuel cladding thermal response was caused by the reestablishment of a positive core flow of high density fluid during the period from about 2.5 to 6 seconds after break initiation (Reference 17, 18). The resumption of positive core flow occurred once the lower plenum fluid reached saturation, effectively decoupling the broken loop cold leg and core inlet mass flows. Since the primary coolant pumps had not yet degraded significantly, their driving force under these conditions served to reestablish positive core flow. The high density of this reestablished positive core flow was caused by the transition, at about 3.6 seconds from subcooled to saturated flow at the cold leg side of the break. The transition reduced the break flow at this time such that the mass ejected was less than that supplied to the downcomer from the primary coolant pumps. This condition of excess coolant flow into the downstream side of the core lasted until about 6 seconds as shown in Table IV. Consequently, during this period, the reestablished positive core flow provided especially good heat transfer from the core and a core-wide return of fuel cladding temperature to fluid saturation temperature.[2]

Clearly, to predict the conditions described above, a code must permit asymmetric downcomer flows. In addition, nodalization in lumped parameter models becomes important in the handling of the propagation of density and temperature waves in the downcomer. Amplitude reduction through mixing in the modeled volumes can result in underpredicting the amplitudes of thermal-hydraulic phenomena. Split downcomer models and nodal optimization studies can be made using L2-2 and L2-3 results as references in order to minimize this deficiency.

The influence of the primary coolant system hydraulics on the fuel cladding thermal response is shown in three-dimensional Figures 3 and 4. During the first 10 seconds the hydraulic behavior just described resulted in a significant removal of the stored energy in the fuel: 65 percent in the L2-2 case and 64 percent in the L2-3 case. After 10 seconds the remaining stored energy was sufficiently small that subsequent clad temperatures did not rise as high as the initial values, and the course of both experiments proceeded without significant differences in the phenomena and chronology of events.

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[2] A modified RELAP4/MOD6 model which predicts the L2-2 and L2-3 quench also predicts that in the case of L2-3 with a loss of offsite power, the intact loop pump coastdown would not supply sufficient primary coolant to the pressure vessel to result in this early quench. This situation will be studied in the L2-5 test scheduled for late 1980.

Furthermore, peak clad temperatures never exceeded damage thresholds, as shown in Figure 5. This indirect conclusion is corroborated by visual examination of the fuel which shows that the core internals are clean and undamaged after two nuclear LOCEs (Reference 20). Also, water chemistry samples have shown no fuel damage or leakage of fission products. Furthermore, lead rod tests (LLR 1 to 4 and loss-of-coolant test, LOC-11) done in the Power Burst Facility demonstrate that the degree of damage to the fuel is not too severe to prevent reuse, even in the event the fuel was prepressurized.

Figures 3 and 4 also show that all cladding temperatures in the L2-2 case, and the upper level cladding temperatures in the L2-3 case, quench once again at about 18 seconds, very shortly after the accumulator injection began. Accumulator injection caused local condensation and a consequent pressure reduction at the point of injection in the intact loop cold leg. This induced a flow downward through the core from the upper plenum and caused a top down quench of the fuel cladding. In L2-2 this effect progressed over the full core length of 1.68m; whereas, in L2-3 the effect did not penetrate below the 0.85m elevation because of the higher cladding temperature in that region.[3]

4.1.2 ECC Bypass and Reactor Vessel Inventory

The performance of the ECCS was essentially the same in both nuclear and nonnuclear experiments. The ECC bypass increased slightly with increasing core power density. At the end of accumulator flow in the nonnuclear experiment L1-5, the ECC bypass was 30 percent of the total ECC injected up to that time. The ECC bypass increased to 32 percent in L2-2 (initial power 26.4 kW/m) and 36 percent (initial power 39.0 kW/m) in L2-3.[4]

ECCS in conjunction with the hydraulic phenomena in the primary coolant system prevented complete depletion of fluid mass in the reactor vessel during the transients. Calculations of mass inventory in the reactor vessel as a function of time showed that reactor vessel fluid mass does not deplete to less than approximately 40 percent of maximum at any time. The lower plenum mass inventory was not fully depleted at any time during the L2-2 and L2-3 transients. Best estimate calculations using the RELAP4/MOD6 code also show incomplete depletion of lower plenum fluid, but more depletion than the experimental data as shown in Figure 6. Application of Appendix K evaluation model criteria to these test conditions results in calculations of complete depletion of the lower plenum fluid. Thus, Appendix K is demonstrated to be conservative in the calculation of lower plenum refill and start of core reflood.

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[3] Movies are available from the coordination contact for this RIL which show the progression of the quench fronts through the core as described.

[4] These bypass figures are based on the results of the nonnuclear series, which was specially designed to evaluate bypass (Reference 21), modified in accordance with broken loop flow measurements and other experimental differences. An independent measurement of ECC bypass by Kehler (Reference 22) supports the ECC bypass calculations.

The core reflood rate was essentially the same in all experiments principally because of the significant removal of stored energy from the fuel early in the transient. However, quenching of the fuel cladding in the high powered region of the core was delayed with increasing initial power, as shown in Figure 7.[5]

4.1.3 The Effects of Cladding Surface-Mounted Thermocouples on the Results

Many investigators, including a group of 44 specially convened experts (Reference 23) have studied the possibility that the presence of the cladding surface-mounted thermocouples influenced the early cladding quench observed. While the majority agree that the pertinent data (Reference 24) support the conclusion that there was no significant influence, it is felt that current experimental work described below will shed more light on this question.

Regarding the accuracy of the cladding temperature measurement by these thermocouples, experiments in the LOFT test support facility (LTSF), the Power Burst Facility, the heat transfer facility at Columbia University, and the German REBEKA facility have led to the conclusion that the thermocouples do not significantly perturb the cladding surface temperature. The LTSF results indicate that the external thermocouples agree with imbedded thermocouples to within 30 K. Independently, analysis with the FRAP-T4 and other codes has shown, through fuel rod stored energy correlations, that the LOFT external thermocouple measurements are not significantly perturbed by the thermocouple fin effect. Further experimentation in this area is being carried out in these and other facilities to corroborate these conclusions, define their limitations, and better quantify measurement uncertainties.

4.1.4 Hydraulic Loads During Subcooled Blowdown

The three experiments, L1-5, L2-2 and L2-3 cover the range of core ΔT values which occur in PWRs (0, 22.7 and 32.3 K, respectively). Hence these experiments provide information for determining the pressure transient typical of that in PWRs that result from the largest pipe break for initial conditions varying from hot standby to nominal operation. The system pressure data are shown in Figure 8 and the chronology of events in Table III. Subcooled blowdown lasts longest (0.10 seconds) for the hot standby condition test, L1-5. This is because the conditions provide for the largest difference between initial and saturation pressure. However, the subcooled blowdown time also depends on the sonic velocity which decreases as the temperature increases. Thus, as observed in Figure 8, as the temperature increases in the upper plenum and hot leg, the rate of depressurization decreases. Since hydraulic loading of reactor components is reduced both by the reduction in depressurization rate and by the reduced subcooling, the subcooled blowdown loads are less severe as the core ΔT increases.

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For example, Figure 7 shows that in L2-3 the central fuel assembly, module 5, did not quench until about 10 seconds after the flooding level passed.

The data out to 0.2 seconds were taken at a bandwidth of 1000 Hz to ensure measurement of all significant frequencies. Analysis of the subcooled blowdown pressure data reveals negligible energy content at frequencies above 40 Hz. Furthermore, the data indicate that at high temperature and pressures where water density is only 2/3 the maximum lower-temperature value, the attenuation of high frequency pressure waves is very high. The LOFT system was designed using WHAM6 code subcooled blowdown predictions as forcing functions for structural analysis codes. The requirements imposed on the WHAM6 calculations were an isothermal system condition, a very rapid 1.0 millisecond break opening time, and no pressure wave attenuation. Consequently, the system was predicted to remain structurally sound with a large margin of safety. Measurements of strain and acceleration have confirmed this prediction. Since commercial PWR systems have been designed with similar conservatisms, the pressure waves generated by a pipe break in a commercial PWR are not expected to result in failure of structural components or alteration of core geometry.

4.2 LOFT-Semiscale Comparisons

The Semiscale counterpart experiments to L2-2 and L2-3 show the same basic hydraulic phenomena on approximately the same timing sequence (References 9, 10). Where differences occur they are usually attributable to overrides in the scaling rationale (References 25, 26, 27) such as preservation of core and downcomer lengths.

One notable difference in results was the absence of an early quench. The following possible reasons for this difference are being assessed:

1. Differences in core hydraulics;
2. Inherent differences between a Semiscale electrical heater rod and a nuclear rod;
3. Stainless steel cladding as opposed to LOFT zircaloy cladding;
4. Power profile used in Semiscale to simulate the LOFT nuclear core behavior;
5. Pump locked rotor simulation in Semiscale broken loop hot leg as opposed to a free wheeling pump simulation in LOFT;

The first two items are considered to have the most influence. Calculations, using the INVERT code, and the actual hydraulics measured in the LOFT experiments with the same heater rod power profiles resulted in a predicted Semiscale core thermal response that was in much better agreement with the measured LOFT core thermal response. Also, the return to fluid saturation temperature was then calculated to occur. However, similar calculations with the RELAP4/MOD6 code did not show the return to fluid saturation. Analysis is continuing in an attempt to explain this difference.

The performance of the TRAC and RELAP4 codes in predicting the return of the cladding temperature to the fluid saturation value in L2-2 and L2-3 is being studied. Preliminary results show that the cladding thermal response can be calculated by modifying the heat transfer logic in the codes. This can result from a different selection of the transition boiling correlations, the minimum film boiling point, and the critical heat flux correlation. At this time insufficient post-CHF experimental data in the region of low flow and low quality have been assembled on which to base the proper heat transfer logic. However, theoretical work is continuing in this area and, as experimental data continue to be added, the correct heat transfer logic is expected to result.

4.3 Commercial PWR Considerations

4.3.1 Best-Estimate and WREM Predictions

Four independent pretest best-estimate predictions were made of the L2-3 tests, using the Combustion Engineering BE code package (Reference 28), EPRI's RETRAN (Reference 29), INEL's FRAP-S3/RELAP4MOD6 package (Reference 30) and LASL's TRAC-PIA (Reference 31). A brief comparison is given in (Reference 32) and eventually a detailed comparison will be issued under the US Standard Problem program. All four predictions indicated a cladding temperature turnover at the time of the first quench; however, only the CE predictions showed a quench at the high-powered location, and this was not core wide. Thus, all best-estimate codes predicted generally higher cladding temperatures than measured.

A WREM (Water Reactor Evaluation Model) prediction was made for L2-3 by DSS. Thus, L2-3 provides a basis for the evaluation of the margin of conservatism in the WREM code package. Analyses are being performed within the Standard Problem program and under the guidance of DSS to qualify various conservatisms contained in WREM.

4.3.2. Application to the Zion PWR

Since L2-2 took place, several relevant improvements were made to the INEL computer code package. The hydraulics had not been predicted correctly for L2-2 principally because of a lack of understanding needed to define the values of certain input parameters such as the critical flow transition quality and multipliers for the Henry-Fauske and homogeneous equilibrium models (Reference 33). Post-experiment analyses have led to significant improvements such that best-estimate predictions of system hydraulics now agree very well with L2-2 and L2-3 hydraulics (References 34, 35, 36). Also, improved understanding of the heat transfer surface used in RELAP4/MOD6, now permit predictions of the early cladding quench measured in L2-2 and L2-3; although, as discussed above, a better understanding is still being sought.

The improved LOFT modeling techniques have been used in RELAP4/MOD6 and applied to the ZION commercial PWR. Predictions of system thermal hydraulics were made using L2-2 and L2-3 initial conditions. The mass flowrate results were divided by the ZION-LOFT volume ratio for comparison to the LOFT results. Comparisons of the most significant thermal-hydraulic phenomena in LOFT and ZION are shown in Figure 9 for L2-2 initial conditions, and in Figure 10 for L2-3 initial conditions. Corresponding comparisons of LOFT predictions with LOFT data are shown in Figure 11 for L2-3. Comparison of corresponding curves in Figures 10 and 11 generally show a strong similarity, and the only area where LOFT data and ZION predictions differ appreciably is in the core thermal response. The prediction shows that the cladding in the hot region of the ZION core does not return to fluid saturation conditions within the first 10 seconds of the transient with L2-3 initial conditions. However, as shown in Figure 12, the cooler regions of the ZION core approach fluid saturation temperature within the first 10 seconds. Temperature turnover does occur throughout the ZION core, and the maximum cladding temperature is reached within the first 10 seconds.

One important parameter involved in the LOFT-ZION comparison is core flow. Two separate scaling considerations cause the LOFT core flow to be different from that in ZION. Initially, the intact loop, which represents three intact loops, must carry the flow of four loops because the broken loop is quiescent prior to the break initiation. However, to obtain the ZION core Δ T over a core length of 1.68 m, the LOFT core flow must be 1.68/3.66 or 46 percent of the ZION core flow. The combination of these two effects results in a LOFT core flow which is 61 percent lower than the ZION value, and hence conservative, by comparison, during the initial part of the test. An indication of this conservatism is shown by comparing the LOFT results with the LOFT predictions (Figure 11 where the prediction exceeds the measurement) and then comparing the LOFT measurement with the ZION prediction (Figures 9 and 10, where the measurement exceeds the ZION prediction).

The question of the different core lengths involved has been addressed in the Semiscale facility. Experimental results shown in Figure 13 indicate that core length does not have a significant effect on peak cladding temperature (Reference 37). The ZION steam generator design was varied as shown in Table V to change the physical properties affecting heat transfer. The largest change was to use the LOFT steam generator tube design. The results of the changes to the ZION steam generators on the system hydraulics were negligible. The core thermal response showed a negligible change in peak cladding temperature except for the case involving the LOFT steam generator tube design in which the peak cladding temperature showed a decrease of 50 K. However, as discussed above, the LOFT predictions show higher peak cladding temperature than LOFT data which in turn show a higher peak cladding temperature than ZION predictions. We conclude that the combined effect of the various scaling discrepancies is to produce a conservative indication of PWR core thermal response. The details of the ZION calculations are being prepared for publication.

5.0 RECOMMENDATIONS

The results of the LOFT experiments described in this letter are applicable to large cold-leg breaks in the primary piping coolant loops of PWRs. They are recommended for use by NRR in its interpretation and application of LOCA ECCS evaluation model criteria and related codes.

6.0 FUTURE PROGRAM

The tests remaining in the large break power ascension series are designed to:

(i) L2-4: evaluate the core thermal response to an additional power increase (i.e., to 52.5 kW/m, equivalent to 133 percent normal full power);

(ii) L2-5: repeat the L2-3 test, but with a simulated loss of offsite power (primary pump coastdown, delayed HPIS and LPIS); and

(iii) L2-6: examine the effect and behavior of prepressurized fuel.

The results of the two nuclear tests reported here, and Semiscale counterpart tests, have been used in predicting the results of the remaining tests; these are summarized in Table VI. Differences from L2-3 results are not expected to be significant except in the case of L2-5 where the cladding temperature is expected to reach a maximum at a much later time and reflood several seconds later than any other test in this series. The importance of performing the above tests which remain in the power ascension series lies in the extension of the range of conditions over which our conclusions and codes can be demonstrated to be applicable.

Because of the TMI accident, the focus of the LOFT program has shifted to the small break experiments. The existing testing sequence is, therefore, revised to reflect this shift. A new experimental sequence has tentatively been defined, as shown in Table VII, which advances the small break experiments such that they begin this year. Planning for these small break tests is based on input from your staff (Reference 38), vendor recommendations (Reference 39) and review group advice (Reference 40). Information on small break tests should be available early in 1980.

7.0 COORDINATION CONTACT

For coordination of any further evaluation of these results and for discussion and future experiments, contact Dr. G. Donald McPherson LOFT Program Manager, RES, Telephone 427-4437.

Saul Levine, Director
Office of Nuclear Regulatory Research

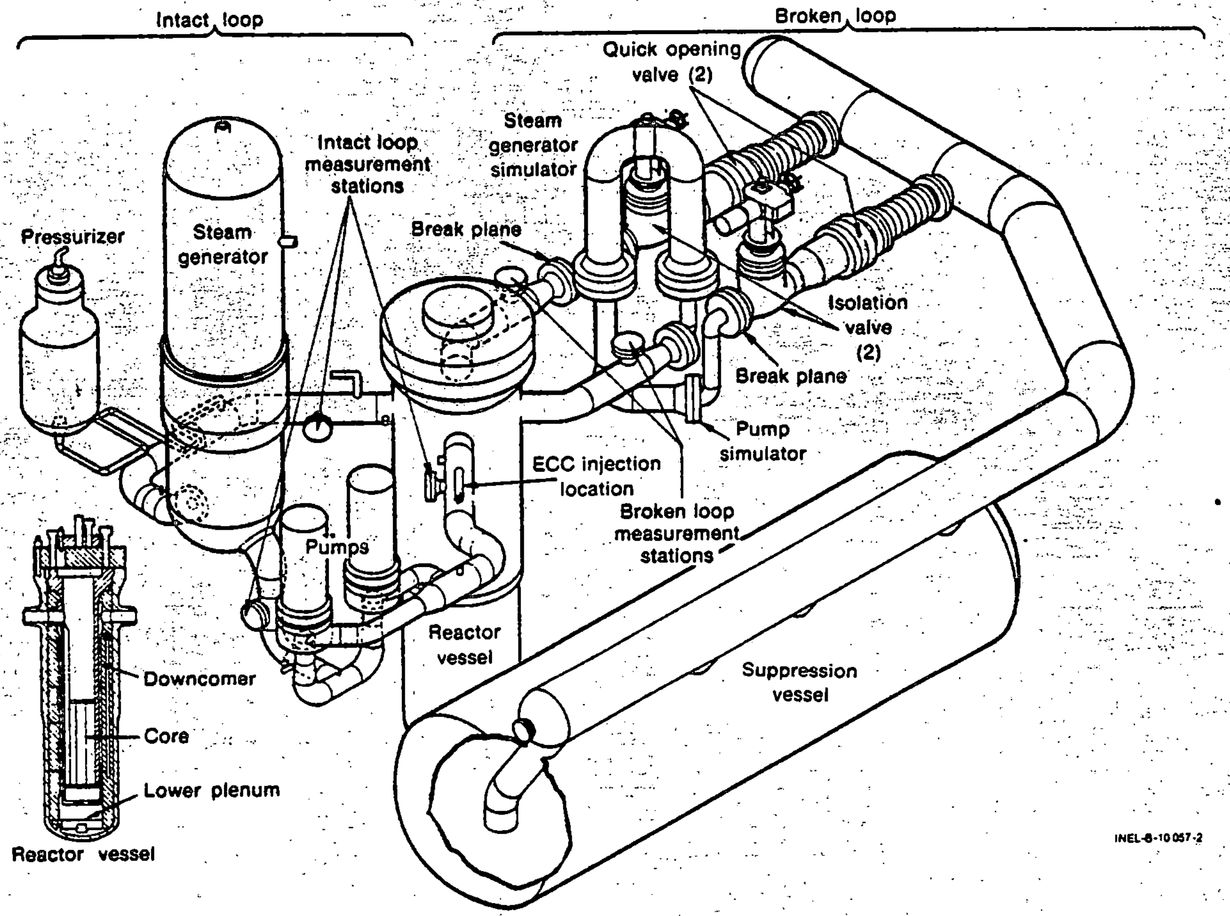


Figure 1. LOFT System Configuration

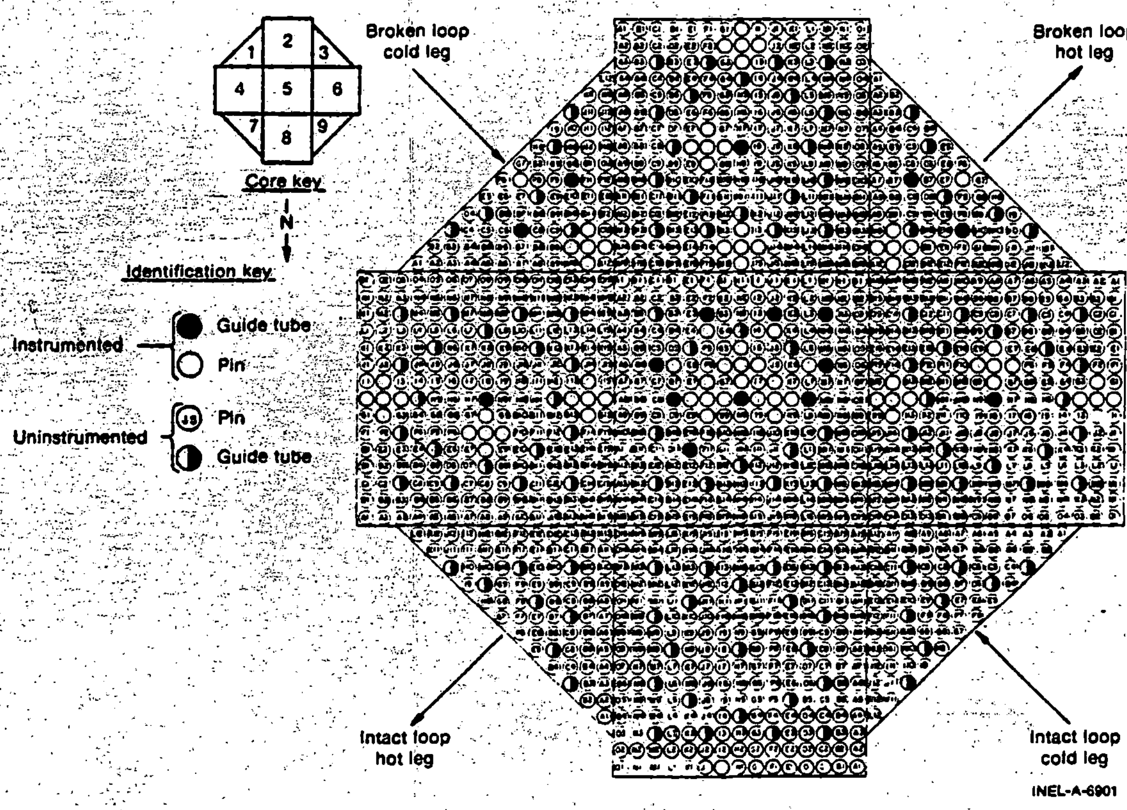


Figure 2. LOFT Core Configuration

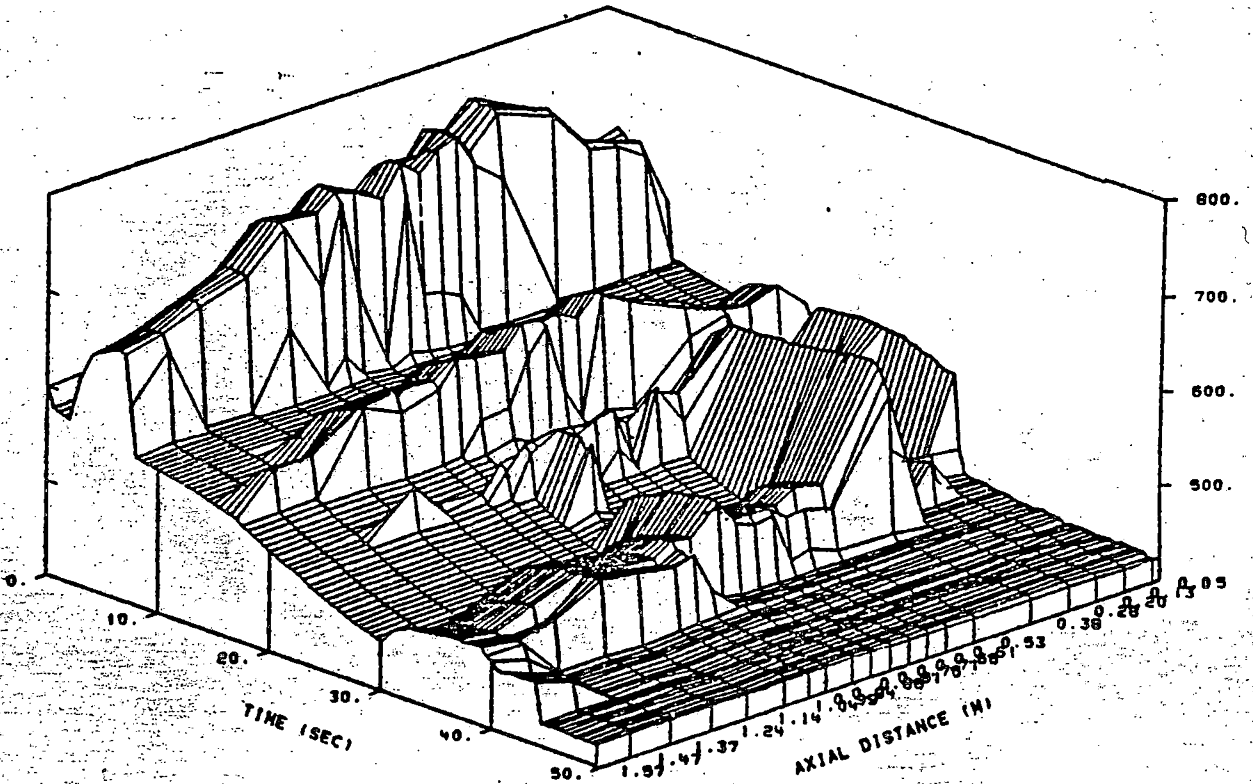


Figure 3. LOCE L2-2 Axial Profile of Cladding Temperature in Fuel Module 5.

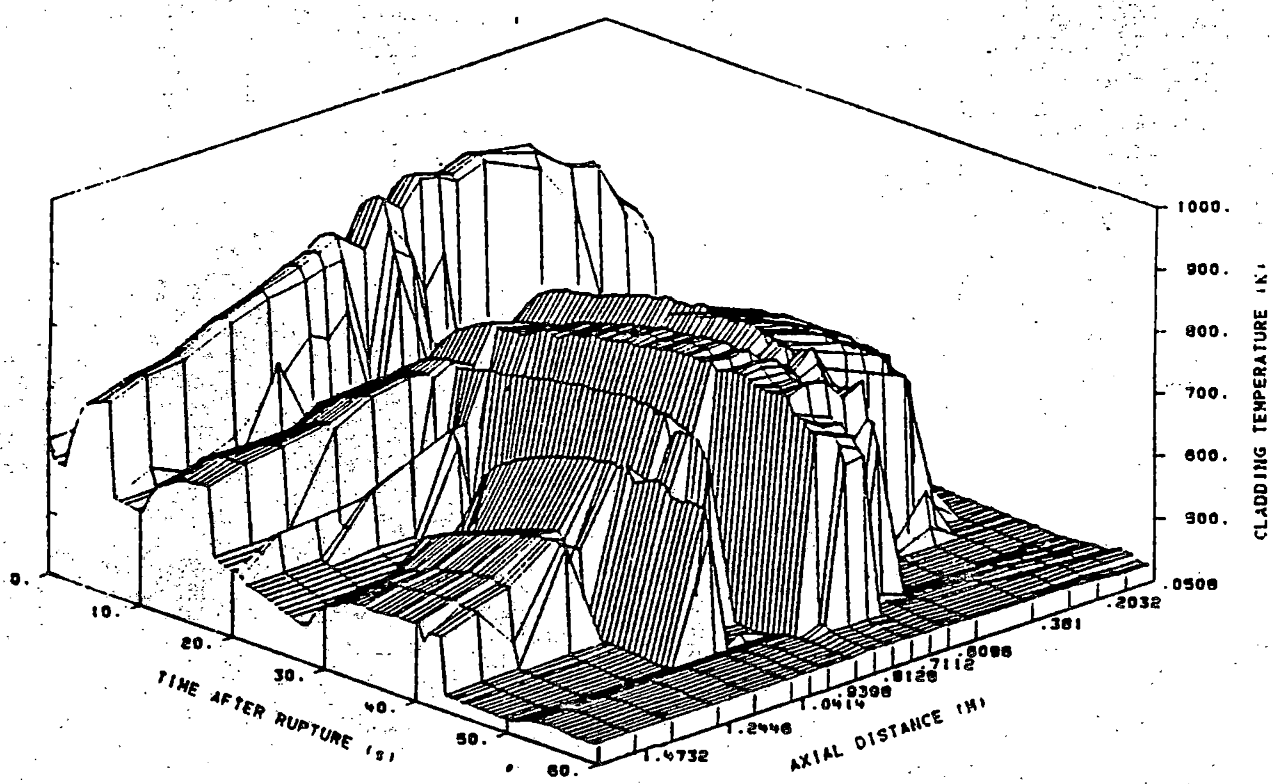


Figure 4. LOCE L2-3 Axial Profile of Cladding Temperature in Fuel Module 5.

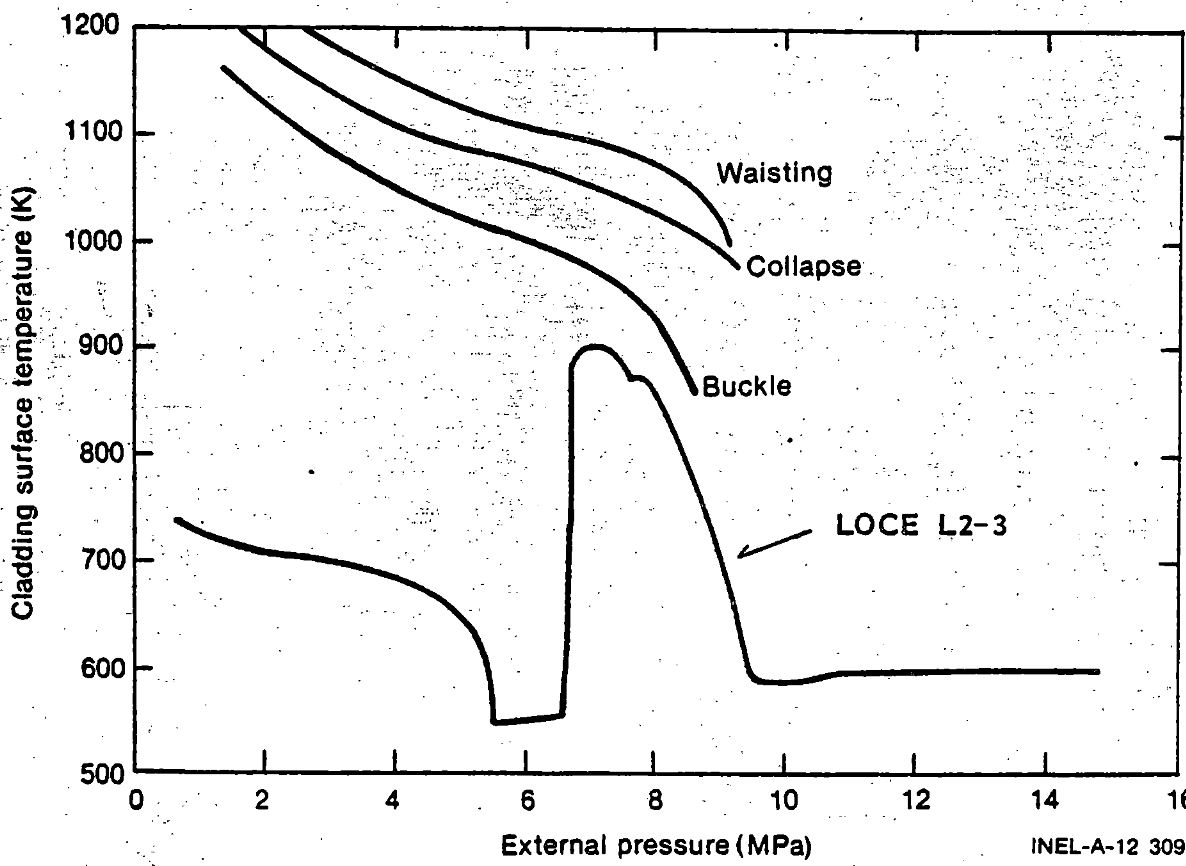


Figure 5. Temperature-Pressure History of Fuel Rod Experiencing Peak Cladding Temperature During LOCE L2-3 Compared with Modes of Cladding Deformation (Ref. 19).

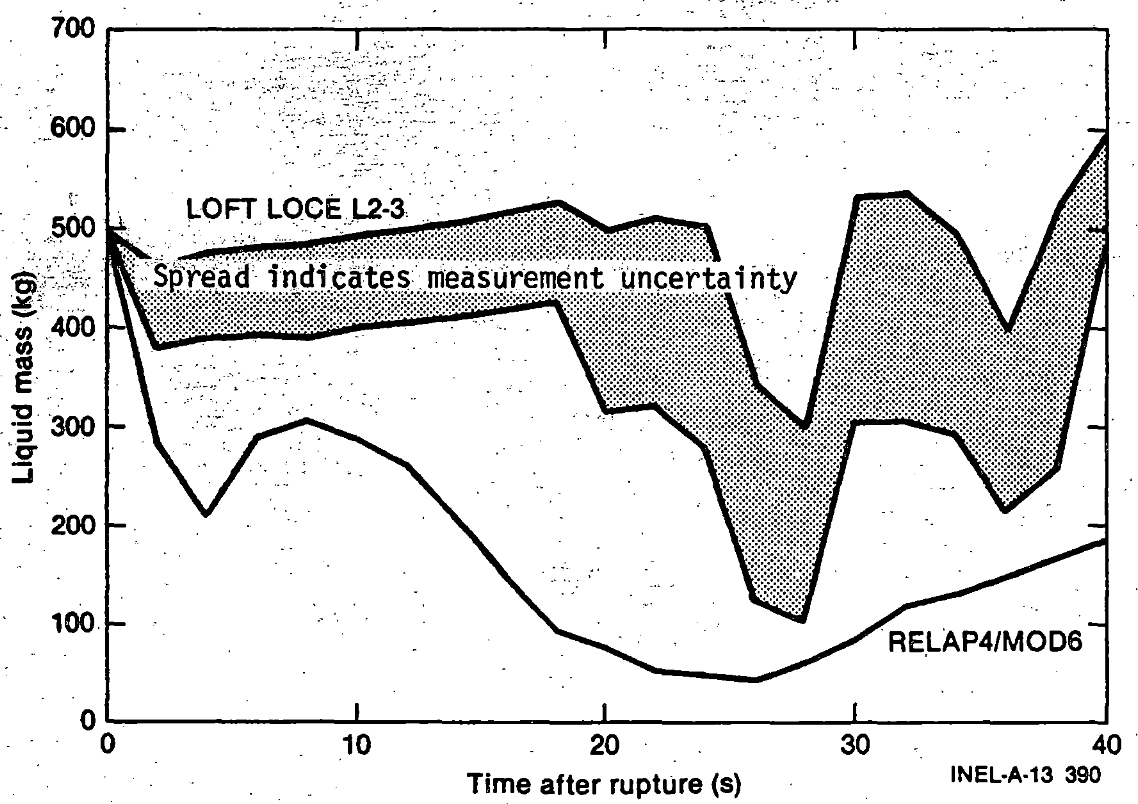


Figure 6. LOFT Data and LOFT Best Estimate Prediction of Lower Plenum Mass Inventory in LOCE L2-3

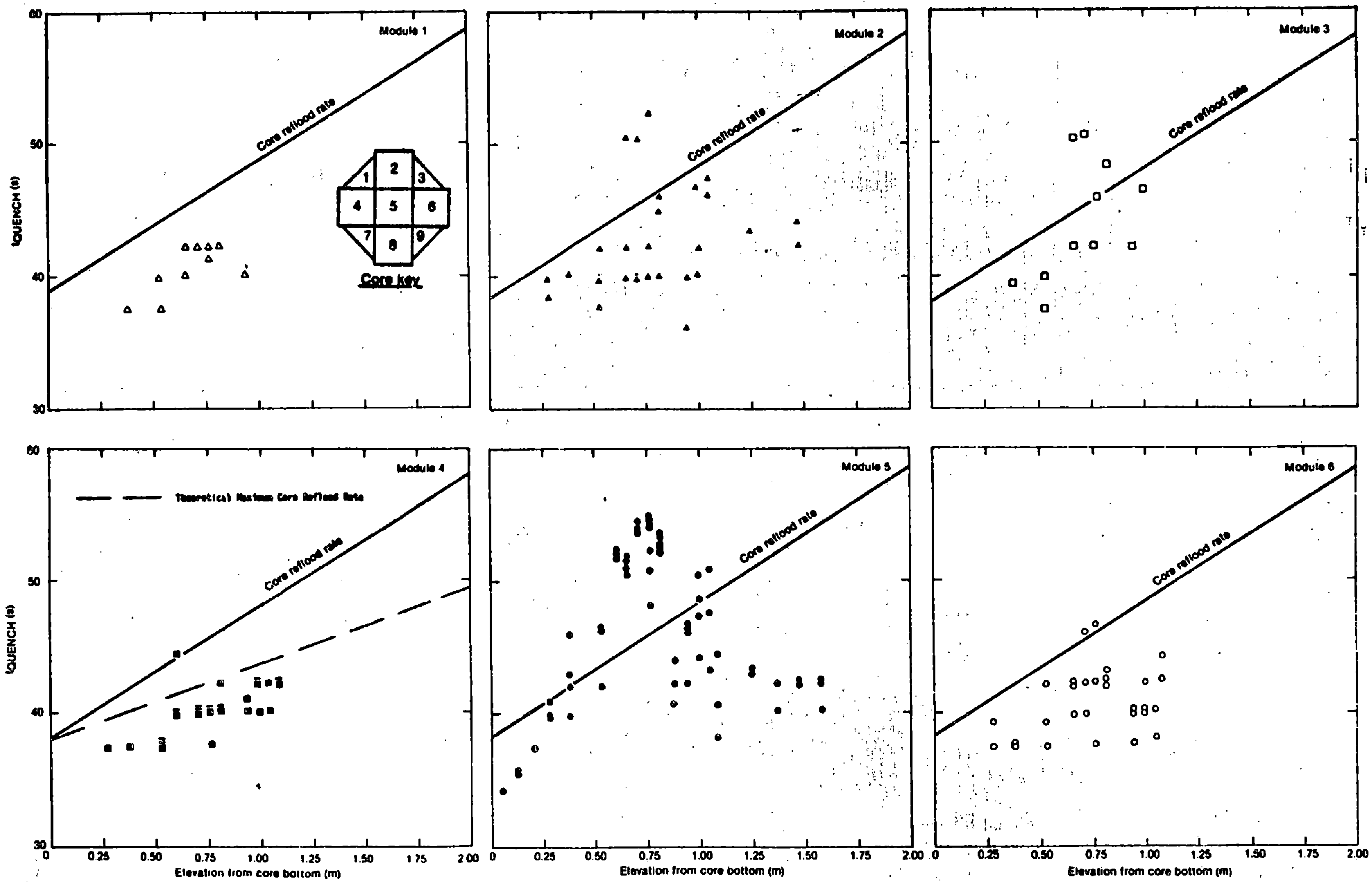


Figure 7. Fuel Cladding Quench in the LOFT Core Compared with the Reflood Rate in LOCE L2-3.

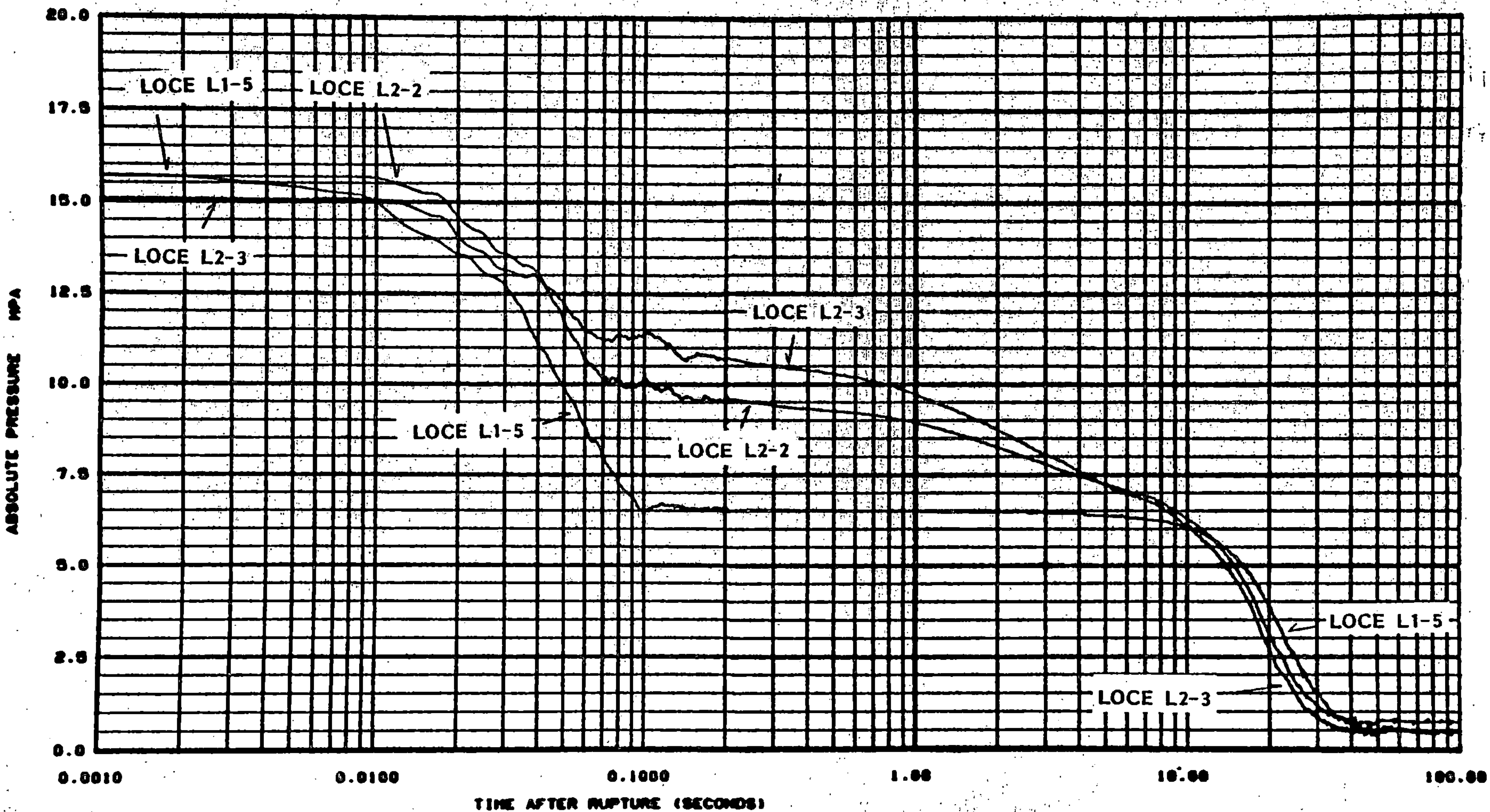


Figure 8. Pressure as a Function of Time for LOCEs L1-5, L2-2, and L2-3

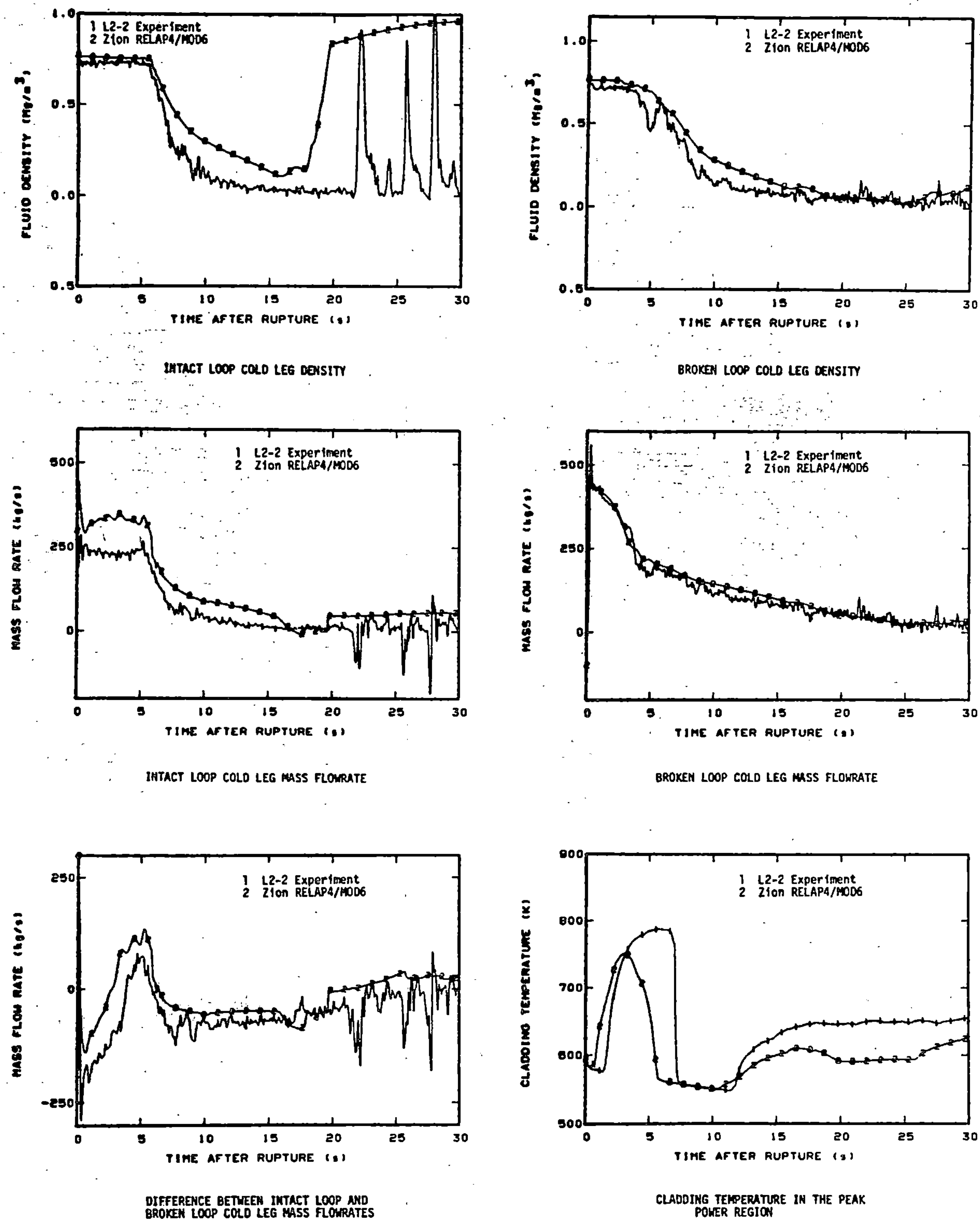
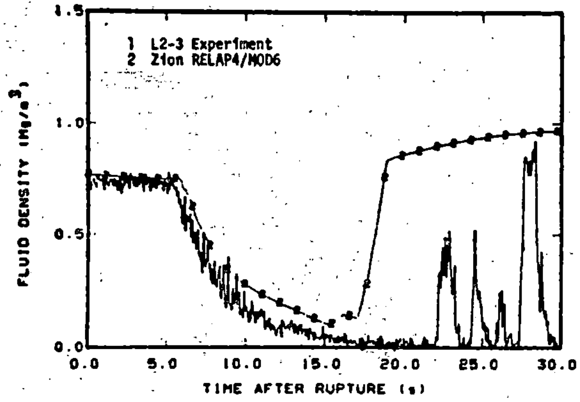
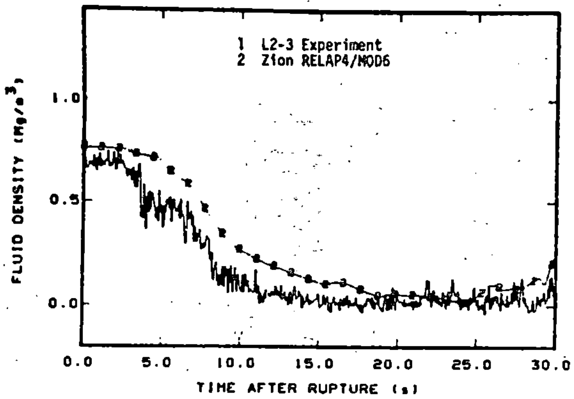


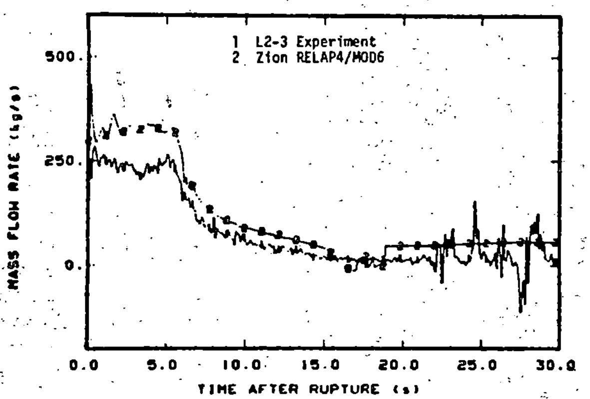
Figure 9. LOFT Data and Zion Prediction Comparisons for LOCE L2-2 Initial Conditions.



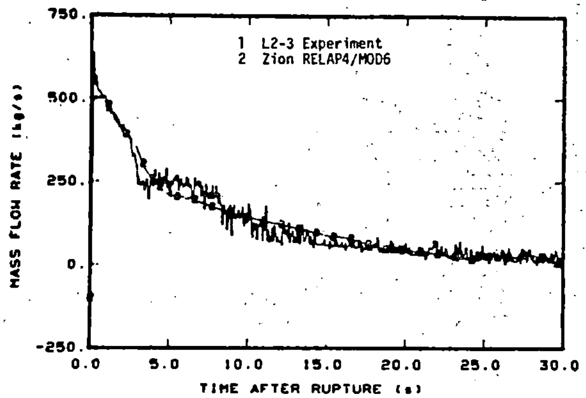
INTACT LOOP COLD LEG DENSITY



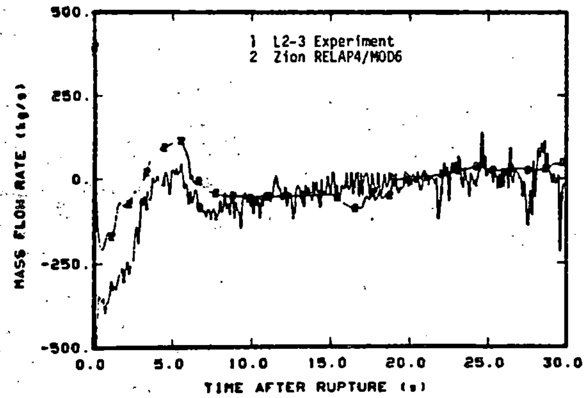
BROKEN LOOP COLD LEG DENSITY



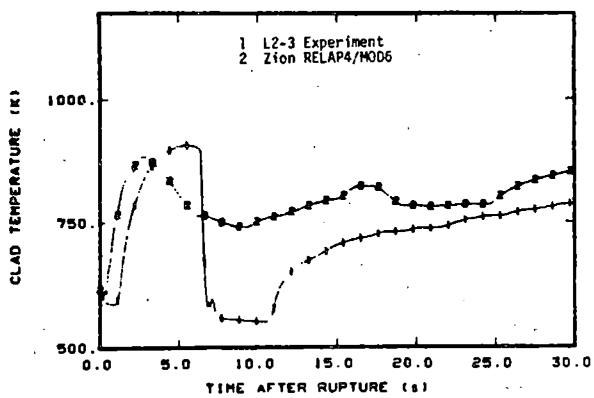
INTACT LOOP COLD LEG MASS FLOWRATE



BROKEN LOOP COLD LEG MASS FLOWRATE

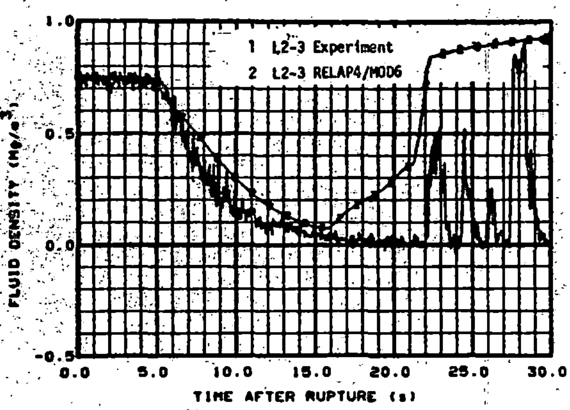


DIFFERENCE BETWEEN INTACT LOOP AND BROKEN LOOP COLD LEG MASS FLOWRATES

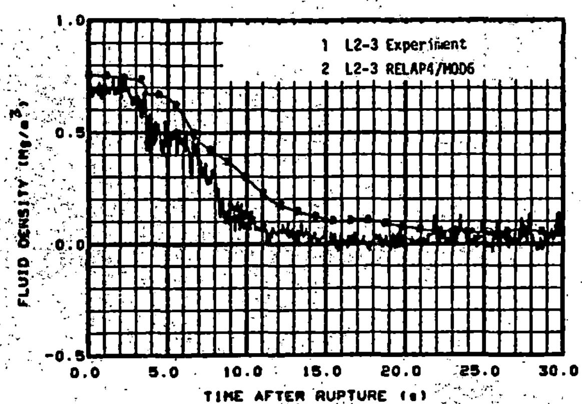


CLADDING TEMPERATURE IN THE PEAK POWER REGION

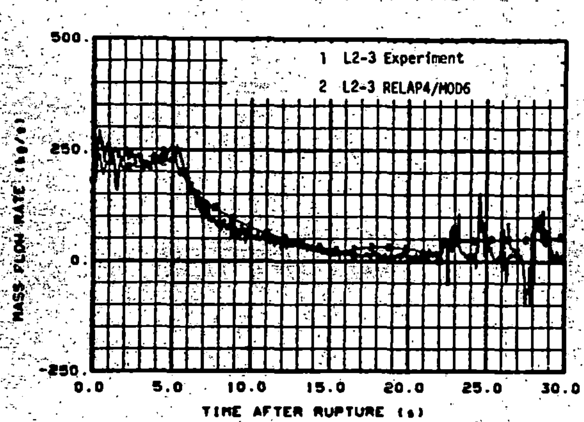
Figure 10. LOFT Data and Zion Prediction Comparisons for LOCE L2-3 Initial Conditions.



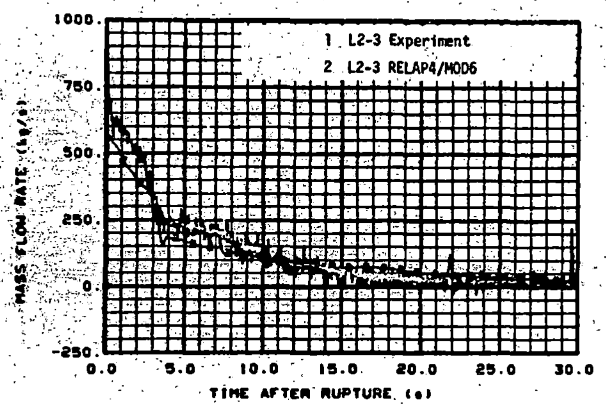
INTACT LOOP COLD LEG DENSITY



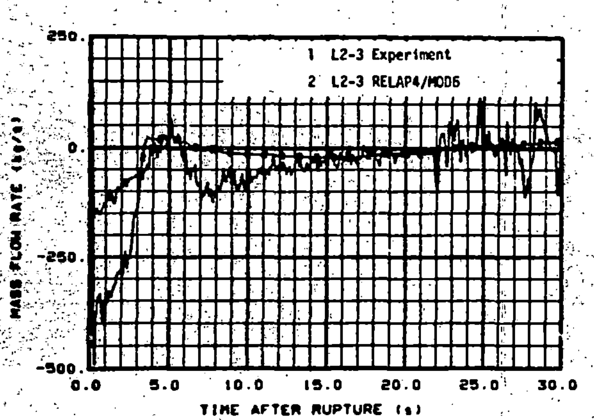
BROKEN LOOP COLD LEG DENSITY



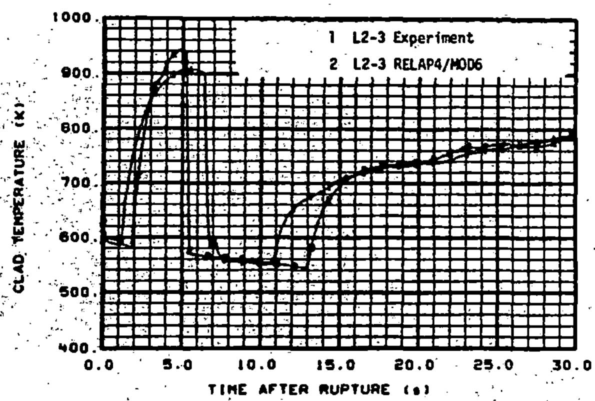
INTACT LOOP COLD LEG MASS FLOWRATE



BROKEN LOOP COLD LEG MASS FLOWRATE



DIFFERENCE BETWEEN INTACT LOOP AND
BROKEN LOOP COLD LEG MASS FLOWRATES



CLADDING TEMPERATURE IN THE PEAK POWER
REGION

Figure 11. LOFT Data and LOFT Prediction Comparisons for LOCE L2-3

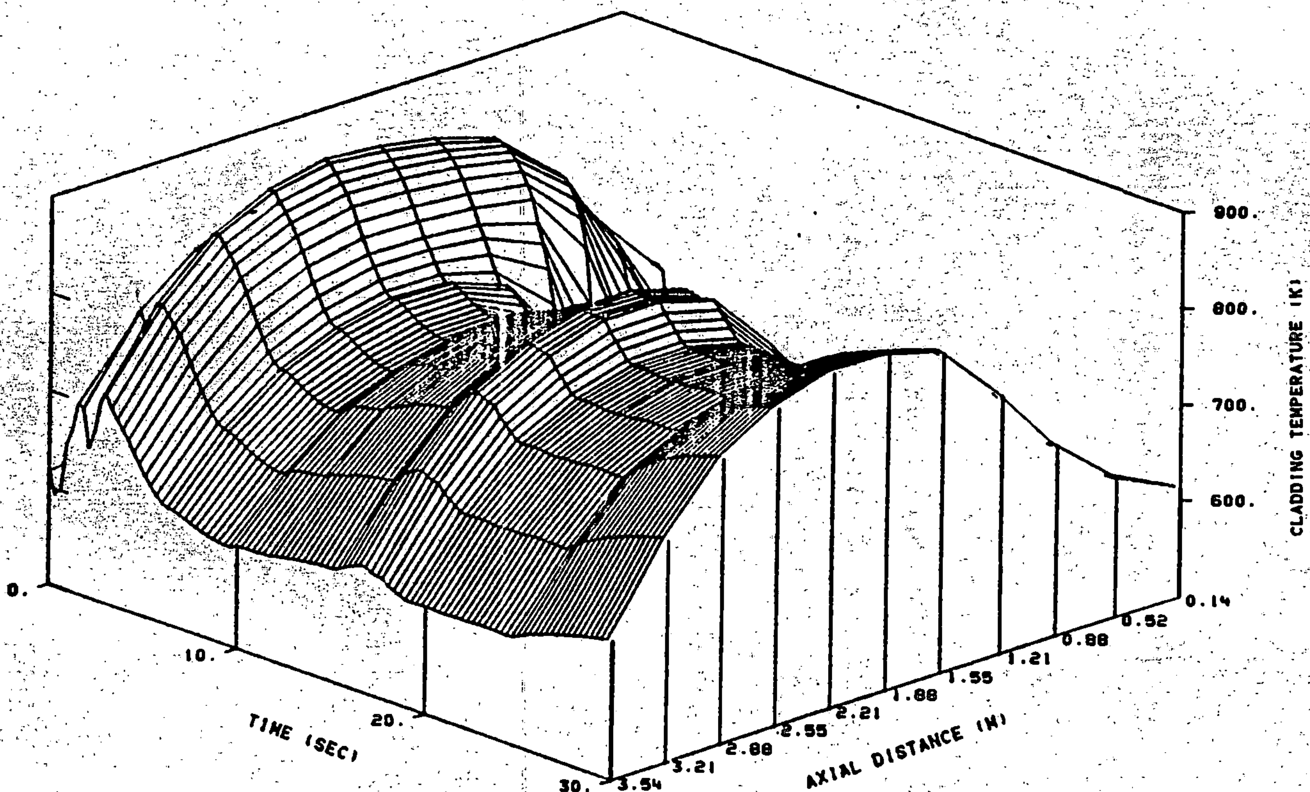


Figure 12. Zion Predicted Axial Profile of Cladding Temperature

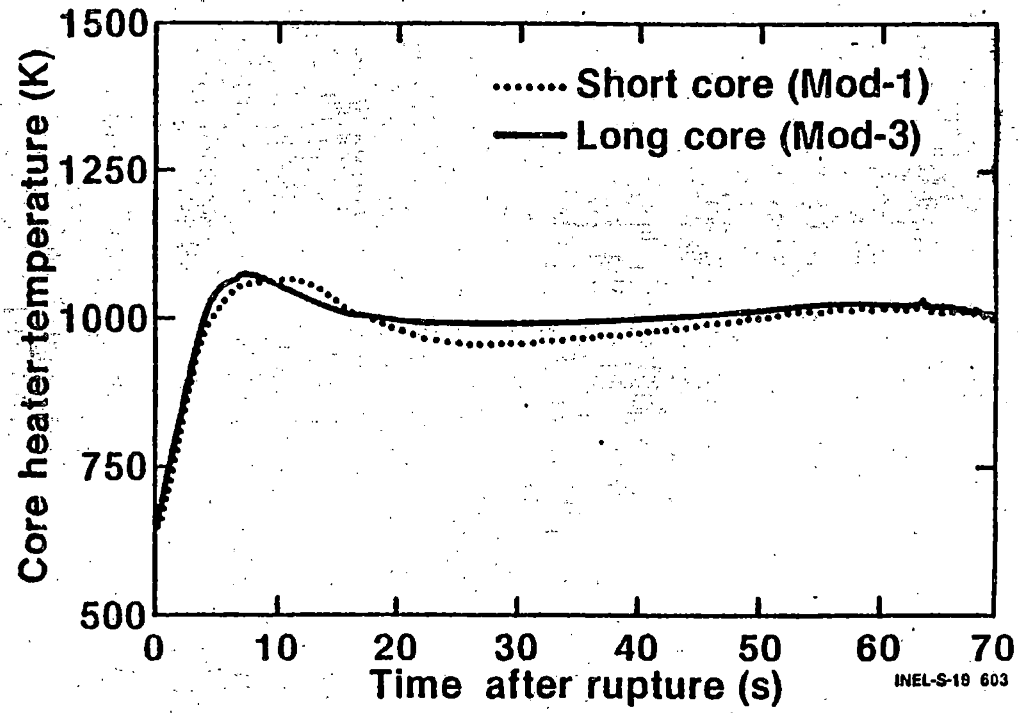


Figure 13. Measured Rod Temperatures at Midplane of Semiscale Core

TABLE I
SYSTEM CONFIGURATION AND INITIAL CONDITIONS
FOR NUCLEAR LOCES L2-2 and L2-3

Parameter	LOCE L2-2	LOCE L2-3
Pipe break:		
Location	cold leg	cold leg
Size	200%	200%
Opening time (ms)	17	17.2
Primary system pump operation	Powered to T_0 + 200 s	Power to T_0 + 200 s
Broken loop pump simulator*	Operating pump K = 9.95	Operating pump K = 9.95
Intact loop resistance	Low resistance K = 131.7	Low resistance K = 131.7
ECCSs	HPIS, LPIS, and accumulator	HPIS, LPIS, and accumulator
ECC injection location	Intact loop cold leg	Intact loop cold leg
ECC actuation mode:		
Accumulator	Pressure	Pressure
LPIS	Pressure-level	Pressure-level
HPIS	Pressure-level	Pressure-level
Steam generator secondary:		
Pressure (MPa)	6.35 ± 0.08	6.18 ± 0.08
Flow rate (kg/s)	12.76 ± 0.40	19.5 ± 0.4
Primary system:		
Pressure (MPa)	15.64 ± 0.03	15.06 ± 0.03
Temperature (K):		
Hot leg	580.4 ± 3	592.9 ± 1.8
Cold leg	557.7 ± 3	560.7 ± 1.8
Core power (MW)	24.9 ± 1.0	36.0 ± 1.0
MLHGR (kW/m)	26.4 ± 2	39.0 ± 3.0
Mass flow (kg/s)	194.2 ± 16	199 ± 6.3
Boration (ppm)	838 ± 4	679 ± 4
ECCS accumulator:		
Pressure (MPa)	4.11 ± 0.05	4.18 ± 0.05
Temperature (K)	300.8 ± 3	307.8 ± 3
Boration (ppm)	3301 ± 19	3281 ± 17
Injected volume (m^3)	1.68 ± 0.03	1.71 ± 0.03
Gas volume (m^3)	1.05 ± 0.03	0.96 ± 0.03

* Darcy K factor based on 0.016 m^2 flow area.

TABLE II

LOFT - SEMISCALE - LPWR SCALING PARAMETERS

	Semiscale	LOFT	LPWR
Volumes			
Total PCS (m^3)	0.23	7.80	347
Reactor Vessel (% of PCS)	37	34	38
Intact Loop (% of PCS)	44	47	51
Broken Loop (% of PCS)	19	19	11
Power (MW)	1.6	50	3400
Length of Active Core (m)	1.67 and 3.66	1.67	3.66
Ratios			
Volume/Power (m^3/MW)	0.14	0.16	0.10
Break Area/PCS Volume (m^{-1})	0.0026	0.0026	0.0026
PWR Volume/Volume	1530	44	1

TABLE III

CHRONOLOGY OF EVENTS FOR NUCLEAR LOCE L2-2 AND
L2-3 WITH NONNUCLEAR LOCE L1-5 COMPARATIVE VALUES

	Time After LOCE Initiation (s)		
Event	LOCE L2-3	LOCE L2-2	LOCE L1-5
LOCE initiated	0	0	0
Subcooled blowdown ended [a]	0.06	0.07	0.1
Reactor scram signal received at control room	0.103	0.085	0.087
Earliest departure of cladding temperature from fluid saturation temperature ($T_{clad} > T_{sat}$)	0.96	1.0	25.6
Control rods completely inserted	1.683	1.725	1.85
Subcooled break flow ended[b]	3.0	3.8	0.1
Maximum cladding temperature attained	4.95	5.8	steady state value at time 0
Earliest core-wide return of cladding temperature to fluid saturation temperature	8.5	8.0	48
HPIS injection initiated	14	12	13
Pressurizer emptied	14	15	14
Accumulator injection initiated	16	18	19
LPIS injection initiated	29	29	34
Lower plenum filled with liquid	35	35	37
Saturated blowdown ended	40	44	47
Accumulator liquid flow ended	45	49	54
Core volume reflooded	55	55	59

a. End of subcooled blowdown is defined as the occurrence of the first phase transition in the system other than at the pipe break location.

b. End of subcooled break flow is defined as the completion of subcooled fluid discharge from the break (hot and cold legs) in the broken loop.

TABLE IV

SUMMARY OF LOFT NUCLEAR LOCE RESULTS

Experiment Results	LOCE L2-3	LOCE L2-2
Times for cladding temperature to exceed fluid saturation temperature (s)		
minimum in hot region	0.94	1.00
maximum in hot region	1.84	2.30
Peak cladding temperature (K)	914 ± 3	789 ± 3
Core reflood rate (m/s)	0.10 ± 0.02	0.12 ± 0.02
Minimum mass/volume in reactor vessel (kg/m^3)	431 ± 75	468 ± 75
Accumulator flow duration (s)	29	31
Maximum accumulator flow/system volume ($kg/s/m^3$)	6.42 ± 0.45	7.71 ± 0.45
Accumulator polytropic gas constant	1.25 ± 0.02	1.22 ± 0.02
Cladding quench time/core reflood time	<1 for all measurements	<1 for all measurements
ECC bypass at end of accumulator flow (% of total ECC injected)	36 ± 4	32 ± 3
First 10 s of the transient		
Duration of primary pump pressure differential (s)	0 to 9	0 to 8
Mass flow rate/system volume		
Initial value intact loop cold leg ($kg/s/m^3$)	25.6 ± 2.0	24.9 ± 2.0
Maximum value (t>0) broken loop cold leg ($kg/s/m^3$)	96.2 ± 14.4	60.8 ± 9.1
Time interval ($\dot{m}_{BLCL} > \dot{m}_{ILCL}$)* (s)	to 3.65	to 3.60
Time interval ($\dot{m}_{ILCL} > \dot{m}_{BLCL}$)* (s)	3.65 to 5.71	3.60 to 6.16
Integral $\dot{m}_{BLCL}$ (kg/m^3)	323.6 ± 22.7	254.7 ± 17.8
Integral $\dot{m}_{ILCL}$ (kg/m^3)	231.6 ± 16.2	215.0 ± 15.0
Difference in $\dot{m}$ integrals (kg/m^3)	92.0 ± 27.9	39.7 ± 23.3
Stored energy removed (% of nuclear heat source energy)	≈ 64	≈ 65

*

BLCL broken loop cold leg
ILCL intact loop cold leg.

TABLE V

ZION STEAM GENERATOR PARAMETRIC VARIATIONS
FOR THERMAL-HYDRAULIC EFFECTS ANALYSIS

Steam Generator Parametric Variation	Physical Change
Tube material changed from Inconel to SS 316 Tube geometry unchanged	Thermal conductivity decreased: at 478 K 10.6% at 589 K 13.9% Heat capacity increased slightly at 478 K negligible at 589 K 1.3 %
Tube geometry changed: ID increased from 19.7 mm to 22.2 mm OD increased from 22.7 mm to 25.7 mm Number of tubes decreased from 3250 to 2430 Tube material unchanged (Inconel)	Heat transfer area reduced: Primary side 13.4% Secondary side 13.2%
LOFT tube geometry used: 10.2 mm ID 12.7 mm OD Number of tubes 12050 Tube material unchanged (Inconel)	Heat transfer area increased: primary side 92.8% secondary side 112.4%

TABLE VI

EXPECTED RESULTS OF REMAINING
L2 LOCES

Experiment	Variations in initial conditions or system configuration relative to LOCE L2-3	Expected differences in results relative to LOCE L2-3
L2-4	Power 33% greater Mass flow 25-30% greater. Core fluid temperature differential remains unchanged.	Same hydraulic phenomena. Similar fuel cladding temperature transient with a peak value of 1100 K occurring at 5 s and core wide return to fluid saturation by 9 s. Subsequent clad temperatures lower than peak value during blowdown. Core wide cladding quench by ECC by 60 s (5 s later than L2-3).
L2-5	Pumps tripped at experiment initiation. HPIS and LPIS delayed. All initial conditions are unchanged.	Initial cladding temperature response will be similar with the clad temperature at 5 s possibly up to 30 K higher. However, there will not be a core wide return of cladding temperature to fluid saturation temperature in the hot region. The cladding temperature will reduce 100 K by 7 s followed by a gradual increase to the peak value of 950-1050 K by 35 s. ECC quench of the cladding will be complete by 65 s.
L2-6	All conditions same as in L2-5 except pressurized fuel is used.	Thermal response is expected to be the same as in L2-5.

TABLE VII

LOFT FY 80 TEST SEQUENCE AND TARGET DATES

TEST	TARGET DATE*	POWER LEVEL (MW) MW	kW/M	kW/FT	COMMENTS
L3-1	11-14-79	50	52.5	16	SMALL BREAK COLD LEG. BREAK FLOW GREATER THAN HIGH PRESSURE SAFETY INJECTION FLOW.
L3-2	01-16-80	50	52.5	16	SMALL BREAK COLD LEG. HIGH PRESSURE SAFETY INJECTION FLOW GREATER THAN BREAK FLOW.
L3-5	03-07-80	0	0	0	SMALL BREAK COLD LEG, PRIMARY COOLANT PUMPS OFF.
L3-6	03-21-80	0	0	0	SMALL BREAK COLD LEG, PRIMARY COOLANT PUMPS ON.
L6-1	05-09-80	37	39.4	12	OPERATIONAL TRANSIENT, LOSS OF STEAM LOAD.
L3-4	05-16-80	50	52.5	16	SMALL BREAK, PRESSURIZER RELIEF VALVE.
L6-2	07-01-80	37	39.4	12	OPERATIONAL TRANSIENT, LOSS OF PRIMARY COOLANT FLOW.
L3-3	07-08-80	50	52.5	16	SMALL BREAK COLD LEG. HIGH PRESSURE SAFETY INJECTION FLOW EQUAL TO BREAK FLOW.
L6-3	09-15-80	37	39.4	12	OPERATIONAL TRANSIENT, EXCESSIVE LOAD INCREASE.
L2-5	09-22-80	37	39.4	12	LARGE DOUBLE-ENDED COLD-LEG BREAK AS L2-3 BUT WITH LOSS OF OFFSITE POWER.
L5-1	01-81	37	39.4	12	INTERMEDIATE BREAK UNSPECIFIED AT THIS DATE.
L6-4	02-81	37	39.4	12	OPERATIONAL TRANSIENT: ROD WITHDRAWAL
L5-2	02-81	37	39.4	12	INTERMEDIATE BREAK UNSPECIFIED AT THIS DATE.
L6-5	04-81	37	39.4	12	OPERATIONAL TRANSIENT: LOSS OF FEEDWATER
L2-4	05-81	49	51.5	16	16 kW/FT 200% DECL.
L6-6	01-82	37	39.4	12	OPERATIONAL TRANSIENT: UNCONTROLLED BORON DILUTION.
L2-6	01-82	37	39.4	12	200% DECL WITH PREPRESSURIZED FUEL
L7-1	05-82	37	39.4	12	STEAM GENERATOR TUBE RUPTURE & LOCA.
L7-2	07-82	37	39.4	12	STEAM GENERATOR TUBE RUPTURE & LOCA.
L3-7	10-82	50	52.5	16	SMALL BREAK UNSPECIFIED AT THIS DATE.
L3-8	12-82	50	52.5	16	SMALL BREAK UNSPECIFIED AT THIS DATE.

*TARGET DATES ASSUME NO SIGNIFICANT PROBLEMS NOR PROGRAM CHANGES

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7.0 COORDINATION CONTACT

For coordination of any further evaluation of these results and for discussion and future experiments, contact Dr. G. Donald McPherson LOFT Program Manager, RES, Telephone 427-4437.

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