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ANALYSIS OF LOFT LOSS-OF-COOLANT EXPERIMENTS
L2-2, L2-3, AND L3-0

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ABSTRACT

A summary of results from Loss-of-Coolant Experiments (LOCE) L2-2, L2-3, and L3-0, conducted in the Loss-of-Fluid Test (LOFT) facility, and conclusions from posttest analyses of the experimental data are presented. LOCEs L2-2 and L2-3 were nuclear large break experiments and were dominated by a core-wide fuel rod cladding rewet, which limited the maximum fuel temperature. Analytical models only conservatively predicted the measured fuel rod temperatures and will require improvements to provide best estimate predictions in this area. Analysis of a large commercial pressurized water reactor (PWR) indicates that the cladding rewet observed in LOFT is also likely to occur in a large PWR, and that, therefore, safety analysis calculations of large loss-of-coolant accidents (LOCA) are more conservative than previously thought. LOCE L3-0 was an isothermal small break (top of pressurizer) experiment and illustrated that the pressurizer fills after the primary system fluid saturates someplace other than the pressurizer itself, that the indicated pressurizer level is higher than the actual level, and that additional model development and assessment work is necessary in order to predict small LOCAs as accurately as large LOCAs.

INTRODUCTION

This discussion describes LOCEs L2-2, L2-3, and L3-0, conducted in the LOFT facility, and presents conclusions reached from analyses of these experiments. LOCEs L2-2 and L2-3 were identical, except for core power level, nuclear large break experiments. LOCE L3-0 was an isothermal small break (top of pressurizer) experiment conducted with the same system and core configuration as LOCEs L2-2 and L2-3.

LOCEs L2-2 and L2-3 are part of the LOFT Power Ascension Test Series (Test Series L2), which included six 200% double-ended cold leg break experiments. LOCEs L2-1 through L2-4 were designed to be identical except for step-wise increases in core power for each experiment. These experiments were planned to provide data for comparison to evaluate the effect of core power on system and core cooling behavior. LOCEs L2-5 and L2-6 were designed to provide data for parametric investigations of the effect of loss of offsite power and prepressurized fuel. Prior to starting Test Series L2, it was decided not to perform LOCE L2-1 because the low core power of LOCE L2-1 was not expected to provide data of significant value over that already obtained from the LOFT isothermal experiments and the "lead rod" tests performed in the Power Burst Facility (PBF).

LOCEs L2-2 and L2-3 had the same specific objectives which were to:

1. Determine core-wide spatial variations of fuel rod cladding thermal response

2. Identify thermal-hydraulic phenomena and determine effects of thermal-hydraulic phemomena on fuel rod cladding thermal response

3. Determine emergency core cooling system (ECCS) performance and core reflood characteristics

4. Determine the integrity of the fuel rod cladding

5. Determine principal variables of temperature, pressure, density, mass flow, and mass inventory as functions of time associated with the core, primary coolant system, and emergency core coolant (ECC) sufficient for comparison with and assessment of code predictions.

LOCE L2-2, performed December 9, 1978, and LOCE L2-3, performed May 12, 1979, met these objectives.

LOCE L3-0 was introduced into the LOFT Experimental Program after the occurrence of the Three Mile Island (TMI) accident and the decision to redirect LOFT to accelerate small break testing. Isothermal small break LOCE L3-0 had the following objectives:

1. Provide data to assess the transient pressure, temperature, and density for comparison with predictions from the RELAP4/MOD6, RELAP4/MOD7, RELAP5, and TRAC-P1A small break computer models
2. Determine the break flow from the available pressurizer pressure and level data
3. Determine if chugging occurs in the suppression tank during the small break blowdown
4. Provide operator training in performing small break experiments.

LOCE L3-0 was conducted on May 31, 1979, and met all of these objectives.

This discussion is divided into three major parts. The first part (a) describes the behavior of LOCEs L2-2 and L2-3, (b) presents the most significant results from posttest analyses of LOCEs L2-2 and L2-3 and from comparisons of the experimental data with computer calculations, and (c) presents results and implications of computer calculations on a commercial-size PWR with the same model used for LOFT. The second part of the discussion describes LOCE L3-0 and presents results from posttest analysis of the experimental data. Conclusions reached from the analyses of LOCEs L2-2, L2-3, and L3-0 are presented as the final part of this discussion.

LOCEs L2-2 AND L2-3

LOCEs L2-2 and L2-3 were 200% double-ended cold leg break experiments with ECC injection into the intact loop cold leg. These experiments were identical except for the core power level and the resultant temperature rise across the core. For this analysis, the data from LOCEs L2-2 and L2-3 were compared with data from nonnuclear isothermal LOCE L1-5 to evaluate the effect of core power on system and emergency core cooling behavior. LOCE L1-5 was identical to LOCEs L2-2 and L2-3 except that the core was at zero power. LOCE L1-5 was used, in some cases, as a baseline and, therefore, is included in the following discussion.

Description of LOCEs L2-2 and L2-3 Behavior

LOCEs L2-2, L2-3, and L1-5 were configured to represent the complete severance of one of the four reactor inlet pipes in a commercial-size four-loop PWR with unobstructed discharge from both sides of the break. Offsite power was assumed available in that the primary coolant pumps were left running at nearly normal speed. ECC was injected from the accumulator and the high-pressure and low-pressure injection systems (HPIS and LPIS) to the intact loop cold leg. The ECC flow rates were adjusted to represent complete loss of ECC injected to the broken loop and failure of one HPIS and LPIS injection train.

The most important plant operating conditions at experiment initiation for LOCEs L2-2, L2-3, and L1-5 are shown in Table 1. Main features of the event sequences for the experiments are listed in Table 2.

The general behavior of the LOFT system during LOCEs L2-2, L2-3, and L1-5 can best be related through consideration of the pressure response of the system shown in Figure 1. On the basis of the pressure response, the experiments may be described in several phases. The first phase, subcooled blowdown, consists of the initial pressure reduction from initial operating pressure to saturation pressure of the reactor outlet fluid. This phase lasted about 50 ms in LOCEs L2-2 and L2-3 and about 80 ms in LOCE L1-5, being shorter in the nuclear experiments due to the lower initial pressure and higher reactor coolant outlet temperature (higher saturation pressure). The subcooled blowdown period is of main interest relative to the structural loading of the system and will not be discussed further here.

TABLE 1. PLANT OPERATING CONDITIONS AT EXPERIMENT INITIATION

	LOCE		
Parameter	L1-5	L2-2	L2-3
Primary system:			
Pressure (MPa)	15.45	15.64	15.06
Temperature (K)	555	570	573
Mass flow (kg/s)	175.1	194.2	199.8
Boron (ppm)	3037	838	679
ECC accumulator:			
Pressure (MPa)	4.17	4.11	4.18
Temperature (K)	304	300	307
Boron (ppm)	3155	3301	3281
Injected volume (m^3)	0.97	1.05	0.96
Reactor core:			
Power [MW(t)]	0	24.9	36.7
Average linear heat generation rate (kW/m)	0	10.9	16.0
Maximum linear heat generation rate (kW/m)	0	26.37	39.4
Coolant temperature rise (K)	0	22.7	32.2

The second phase of the pressure response covers the time from initiation of flashing of coolant in the reactor outlet pipes to the time of flashing in the reactor inlet pipes. This time period (milliseconds) was very short for LOCE L1-5, but longer and roughly the same for L2-2 and L2-3 (∿3.5 s) and quite important to the response of the core, as will be seen subsequently. During this phase, the pressure in LOCE L2-3 was slightly higher than in LOCE L2-2 due to the higher initial temperature of the reactor outlet fluid. This period may also be thought of as the time of subcooled fluid discharge out of the broken loop inlet pipe.

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

	Time after Test Initiation (s)		
Event	LOCE L2-3	LOCE L2-2	LOCE L1-5
Test 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

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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.

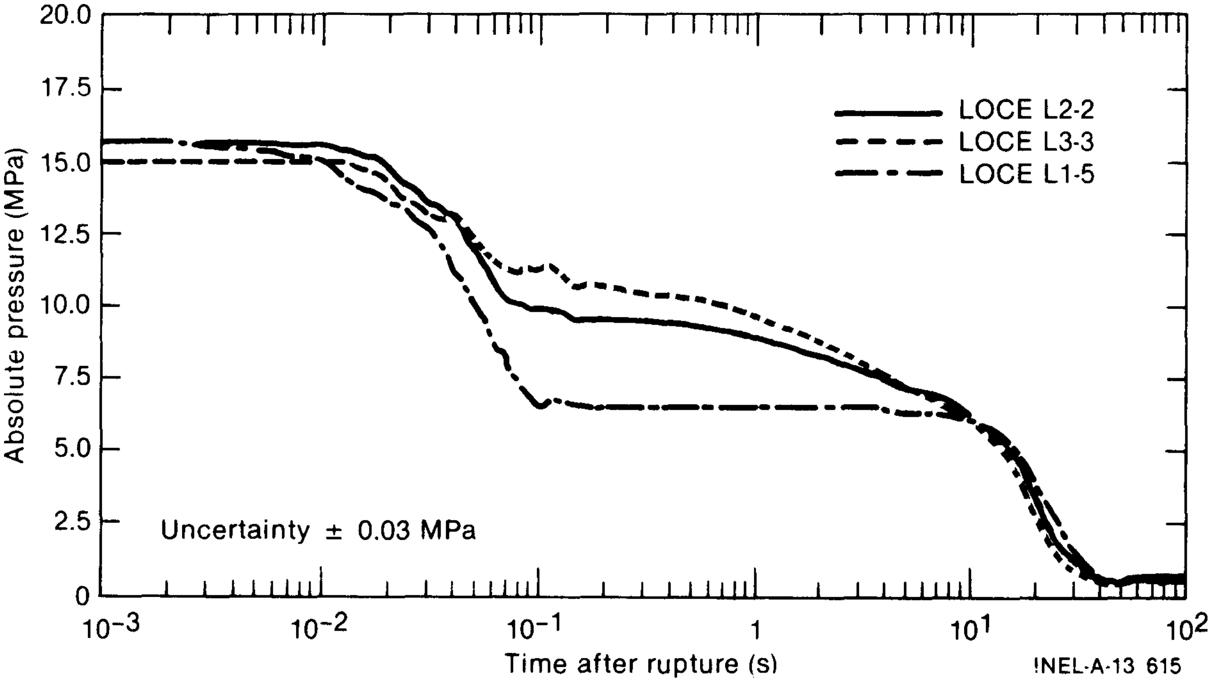

Figure 1. Pressure as a function of time for LOCEs L1-5, L2-2, and L2-3.

During the final phase of the depressurization, saturated blowdown, saturated fluid flowed out of both ends of the broken pipe. This phase may be further subdivided to consider the period of break critical flow discharge and break noncritical flow discharge. The later portions of the saturated blowdown phase are very important because, during this time, ECC was effective in reestablishing a stable thermal condition by cooling the core.

During saturated blowdown, LOCE L2-3 depressurized slightly faster than LOCE L2-2 due principally to a higher break discharge flow in the second phase of the blowdown. Later, the depressurization portion of LOCEs L2-2 and L2-3 were identical.

Analysis of LOCEs L2-2 and L2-3 Behavior

The parameter of most interest during the LOFT nuclear powered large break LOCEs is the fuel rod cladding temperature. Figure 2 shows the measured and predicted response of the fuel rod cladding temperature hot spot during LOCE L2-2. The initial increase in cladding temperature was predicted by the calculation; however, the large decrease, or core rewet, measured during LOCE L2-2 was not predicted by the RELAP4/MOD6[a] cladding temperature calculation.

The measured cladding temperature response of LOCE L2-3 was similar to that observed in LOCE L2-2, as shown in Figures 3, 4, and 5. The following observed differences are attributable to core power.

1. The average time of initial departure from nucleate boiling (DNB) was shorter for LOCE L2-3 than for LOCE L2-2, 1.27 and 1.65 s, respectively

2. The peak cladding temperature was higher for LOCE L2-3 than for LOCE L2-2, 914 and 789 K, respectively.

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[a] The RELAP calculations used in the prediction of LOCEs L2-2 and L2-3 were performed with RELAP4/MOD6, Idaho National Engineering Laboratory Configuration Control Number H001184B.

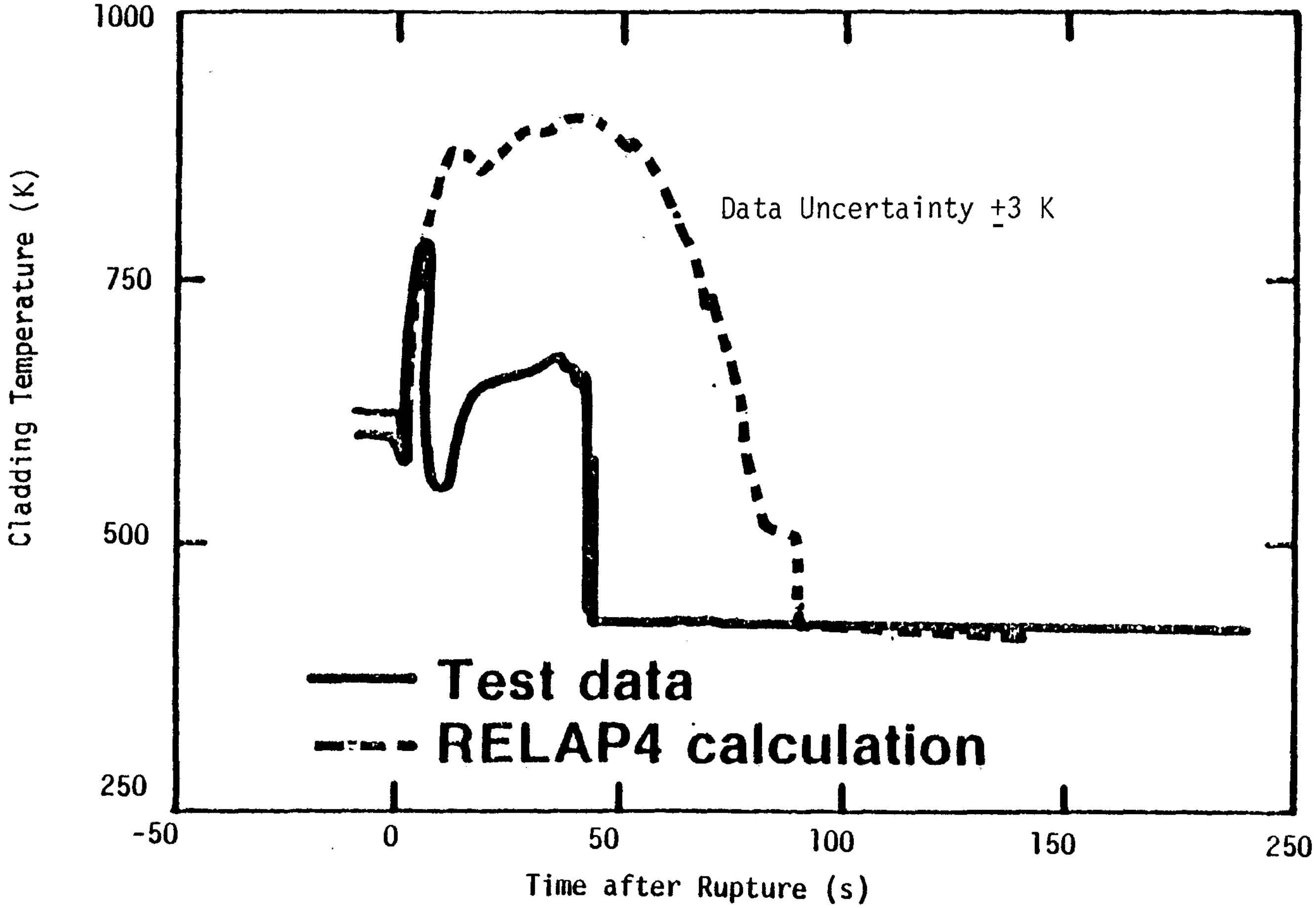


Figure 2. Measured and predicted cladding temperature in center fuel module for LOFT LOCE L2-2.

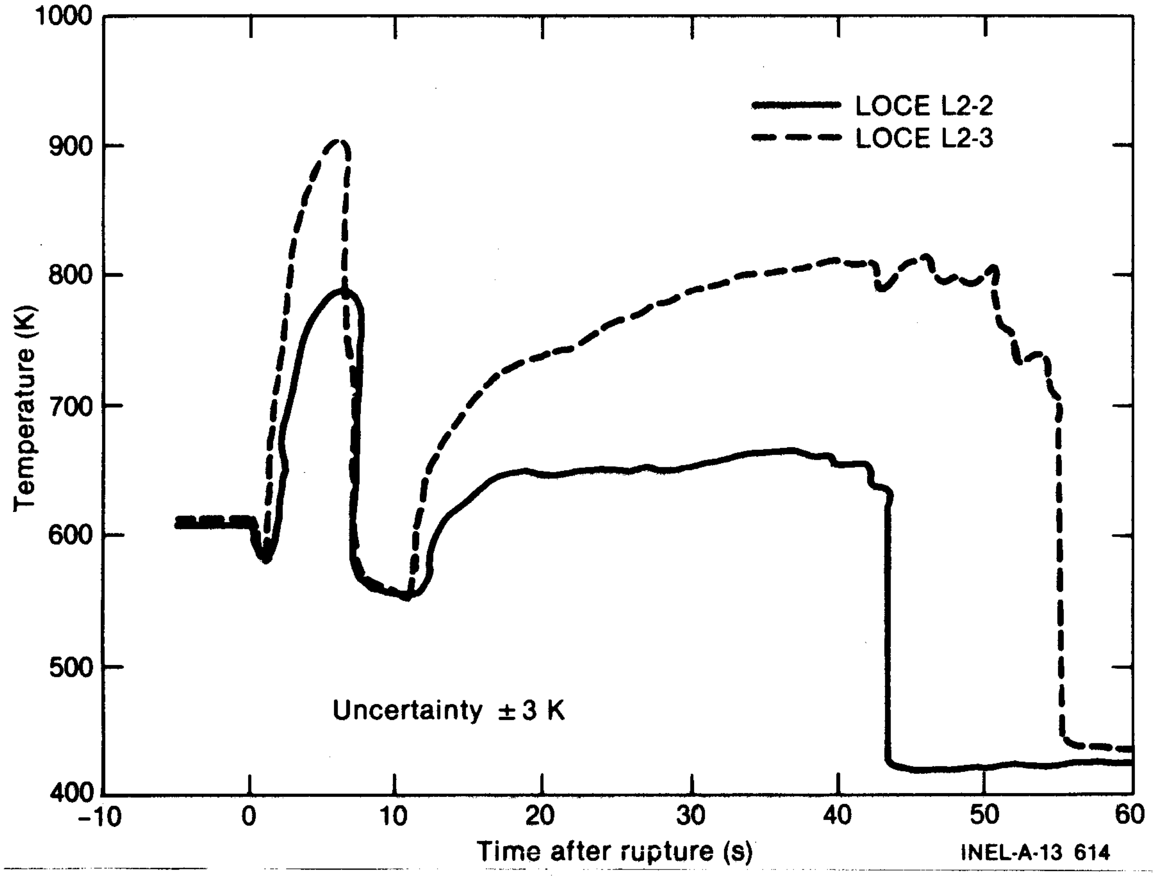


Figure 3. Fuel cladding temperatures for LOCEs L2-2 and L2-3.

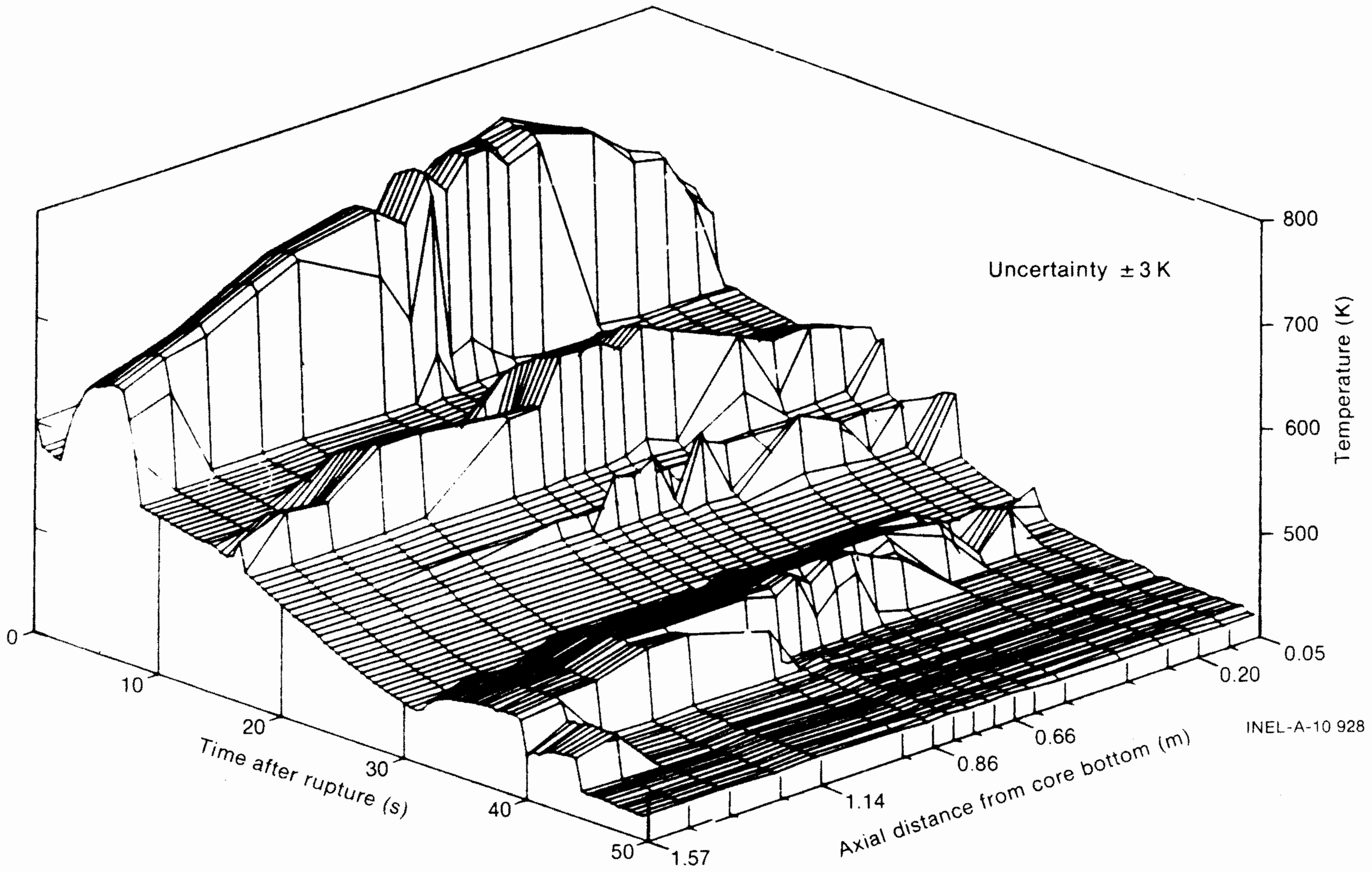


Figure 4. Axial profile of cladding temperature in fuel Module 5 for LOCE L2-2.

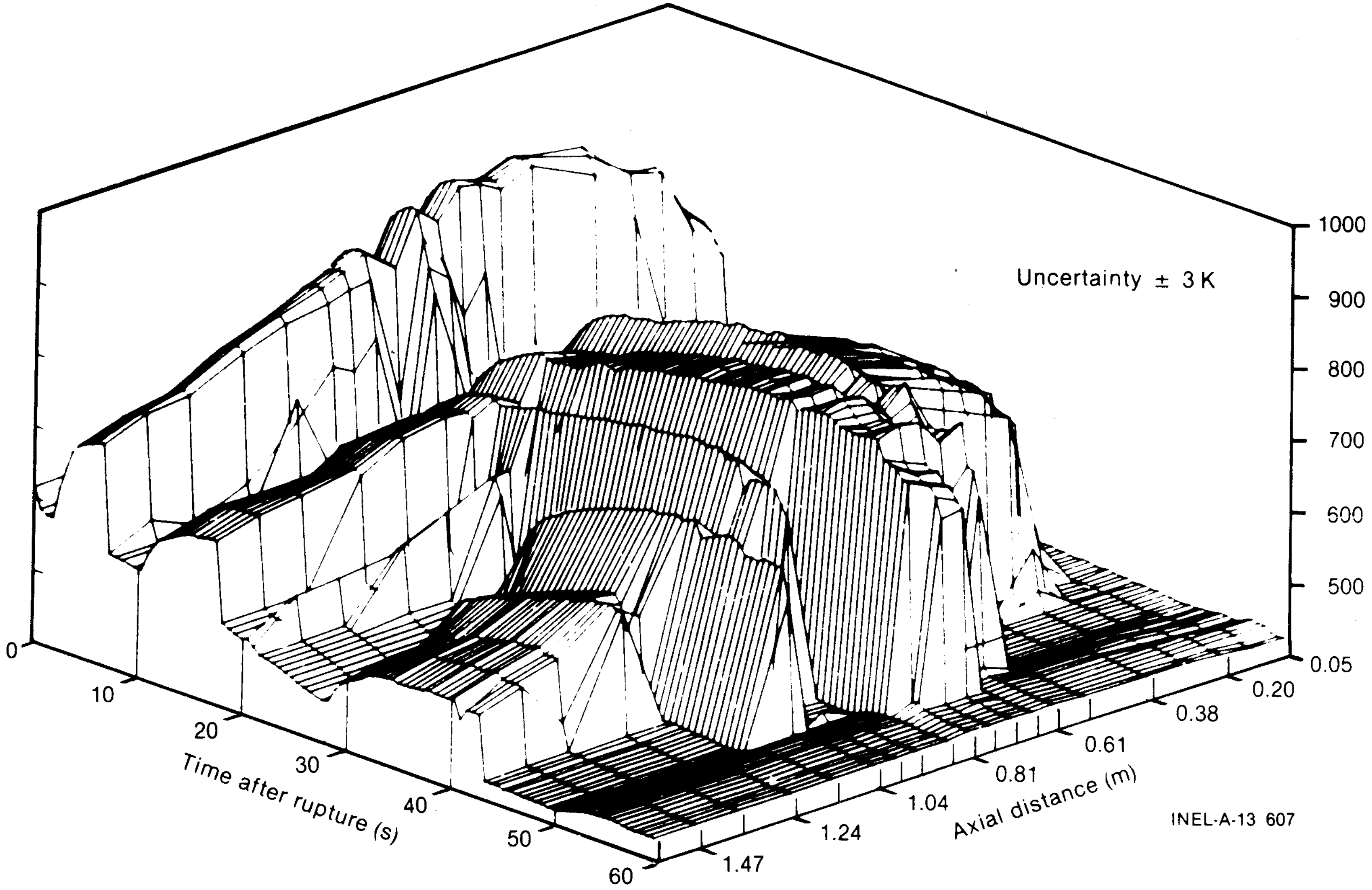


Figure 5. Axial profile of cladding temperature in fuel Module 5 for LOCE L2-3.

3. The pattern of the final rewet, or reflood, in LOCE L2-3 was bottom, top, then middle; whereas, the rewet pattern for LOCE L2-2 was bottom to top.

A revised prediction for LOCE L2-3,[1] based on a computer model that described the rewet of LOCE L2-2,[2] did not predict the rewet observed in LOCE L2-3.

The analysis effort to describe the cause of the disparity between measured and predicted cladding temperature in LOFT LOCE L2-2[3] considered several possible contributors as follows:

1. The thermocouples rewet, but the cladding did not

2. Differences between actual and planned initial conditions

3. The initial fuel rod stored energy

4. The heat transfer correlations in the analytical model

5. The hydraulic calculation.

Investigations into the possibility that the cladding did not rewet concluded that, while additional data on the cladding thermocouple performances are needed, the core did rewet during LOCE L2-2. This conclusion was reached based on data taken in the LOFT nonnuclear isothermal LOCEs, in PBF LOCA tests, from REBECCA tests, detailed thermal analysis, and investigation of other LOFT data. For example, Figure 6 shows the measured fluid temperature and the saturation temperature (determined from pressure) above the LOFT core. The termination of the superheated steam indication at 8 s corresponds to the cladding temperature rewet indication. Additional testing is now being performed at KfK in Germany and at the Idaho National Engineering Laboratory (INEL) to further evaluate the effect of the thermocouples.

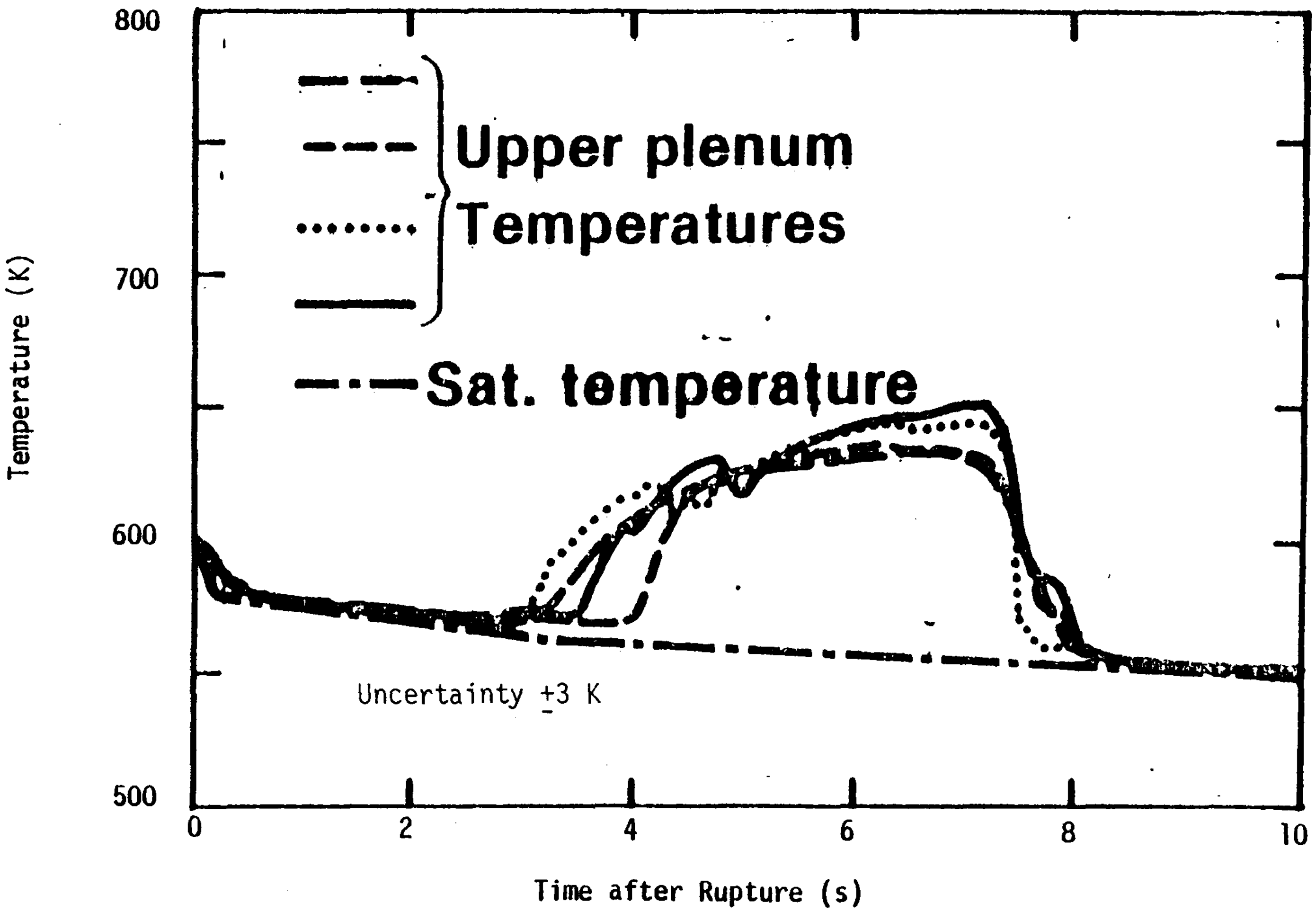


Figure 6. Fluid and saturation temperature above core for LOCE L2-2.

The effect of actual initial conditions on the test was determined by rerunning the prediction model with measured initial conditions. As shown in Figure 7, changing the initial conditions in the calculation did not provide a prediction of rewet.

The stored energy in the fuel rods was correlated to rod experimental and calculated heat-up rates, at times during the transient when fluid cooling effects were negligible. The correlation indicates that the stored energy in the rods is properly calculated. The analysis is currently being refined to include significant parameter uncertainties.

The effect of the heat transfer correlations was evaluated by running analyses with both the specified and measured initial conditions and two different film boiling correlations (new Groeneveld versus Condie-Bengston). Figure 8 indicates that while a better estimate of the peak cladding temperature was obtained with the Condie-Bengston correlation, the rewet was still not predicted.

Evaluation of the system hydraulics indicated generally good comparison between measured and predicted performance, with the exception of the flow in the broken loop cold leg shown in Figure 9. The analytical model for break flow consists of a subcooled discharge model, saturated discharge model, transition quality at which the model changes from subcooled to saturated discharge, and discharge coefficients. Table 3 lists a number of parametric evaluations of these variables. Figure 9 shows how these changes affected break flow, with the last run labeled "Best PT" (posttest) giving the best agreement.

The information in Figure 10 indicates that the changes in critical flow choice did not greatly affect early pressure response, but Figure 11 shows a significant change in core upflow. Figure 12 shows that the Best PT case did predict a cladding rewet, whereas all the others did not. (Note: Figure 12 is not a "cladding hot spot" prediction; therefore, it shows low temperatures.) Thus, the failure to predict the rewet in LOCE L2-2 was due mostly to the core hydraulic calculation.

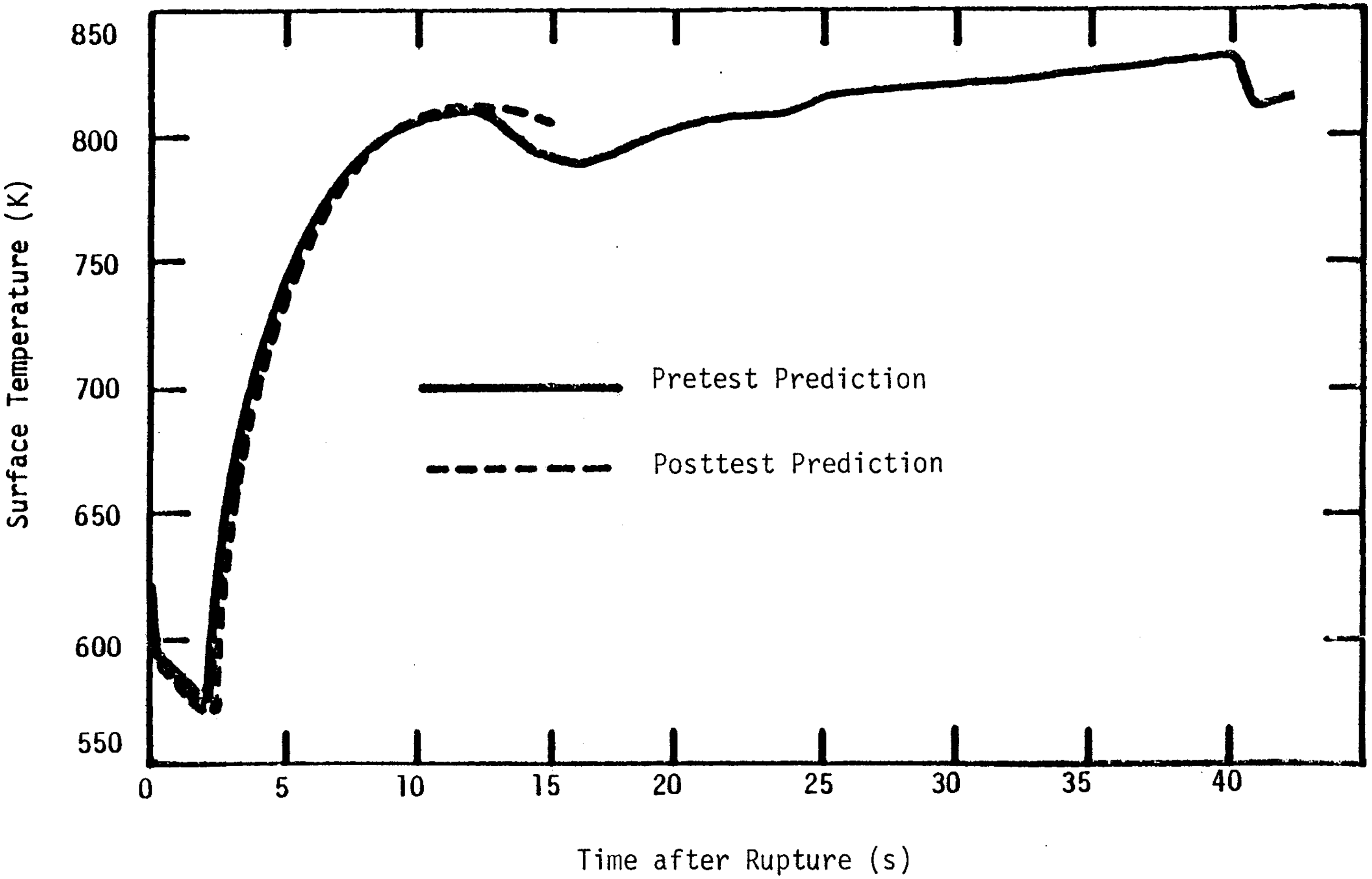
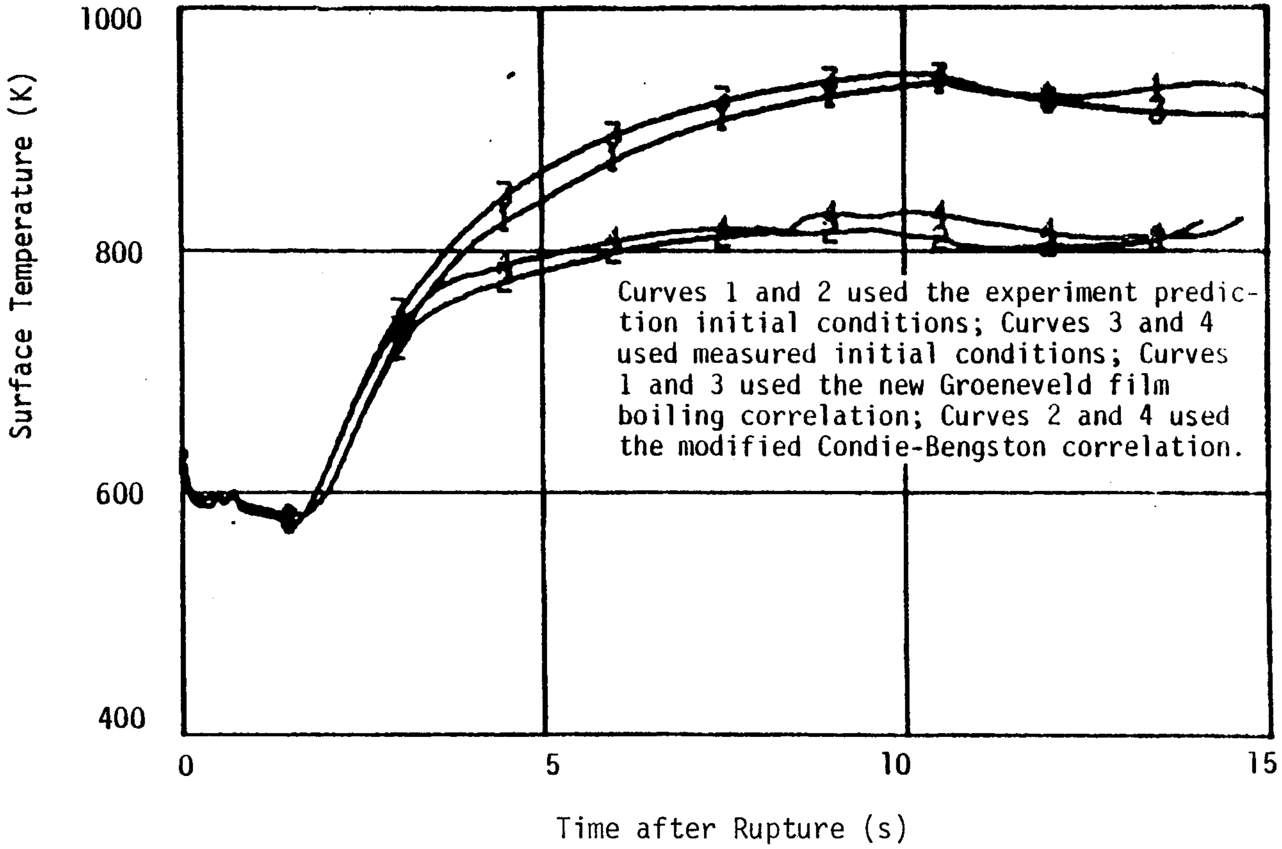


Figure 7. Effect of initial conditions on LOCE L2-2 prediction.



Curves 1 and 2 used the experiment prediction initial conditions; Curves 3 and 4 used measured initial conditions; Curves 1 and 3 used the new Groeneveld film boiling correlation; Curves 2 and 4 used the modified Condie-Bengston correlation.

Figure 8. Cladding temperature on highest-powered fuel rod showing sensitivity to both film boiling correlation choice and initial conditions for LOCE L2-2.

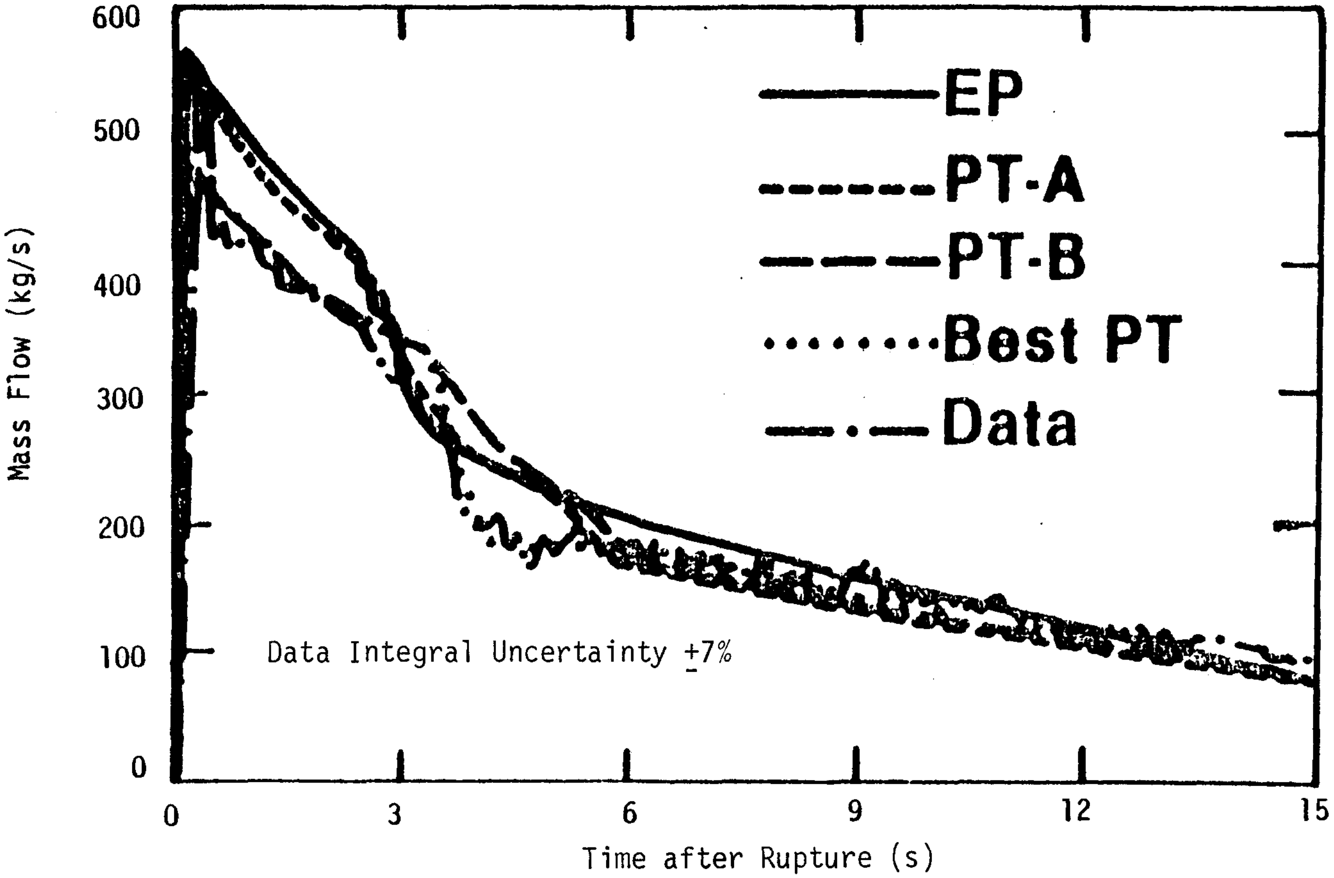


Figure 9. Broken loop cold leg flow for various critical flow choices for LOCE L2-2.

TABLE 3. CRITICAL FLOW SENSITIVITIES

Run	Multipliers		
	Henry-Fauske (Subcooled)	Homogeneous Equilibrium (Saturated)	Transition Quality
Experiment prediction	1.0	1.0	0.02
PT-A[a]	1.0	0.848	0.02
PT-B[a]	0.848	0.848	0.02
Best PT[b]	0.848	0.848	0.0025

[a] PT-A and PT-B are posttest break flow sensitivity calculations.

[b] Best PT is the posttest calculation which best predicted the LOCE L2-2 results.

The relatively abrupt transition between subcooled and saturated discharge at the break was traced to the progression of hot fluid from the core to the break, as shown in Figure 13. This type of behavior is difficult to calculate with a discrete model of the system, but using a very low value of transition quality allows for the abrupt transition even though the transmission of the temperature increase and consequent fluid density decrease are damped by the discrete model.

The net result of the broken loop cold leg mass flow transition is to enhance positive core flow and force fluid down the downcomer and into the core. This can be seen in Figure 14, which compares the measured intact and broken loop cold leg mass flows for LOCE L2-2. The overlap between 4 and 6 s after rupture represents the fluid forced down the downcomer and through the core. The data in Figure 15 indicate that the same phenomena occurred in LOCE L2-3.

As previously mentioned, a reprediction of LOCE L2-3 was made with the Best PT model that did predict the rewet for LOCE L2-2. Figure 16 shows the comparison between the original RELAP4/MOD6 prediction[3] (RELAP A) and the revised model[2] (RELAP B). Although the revised model

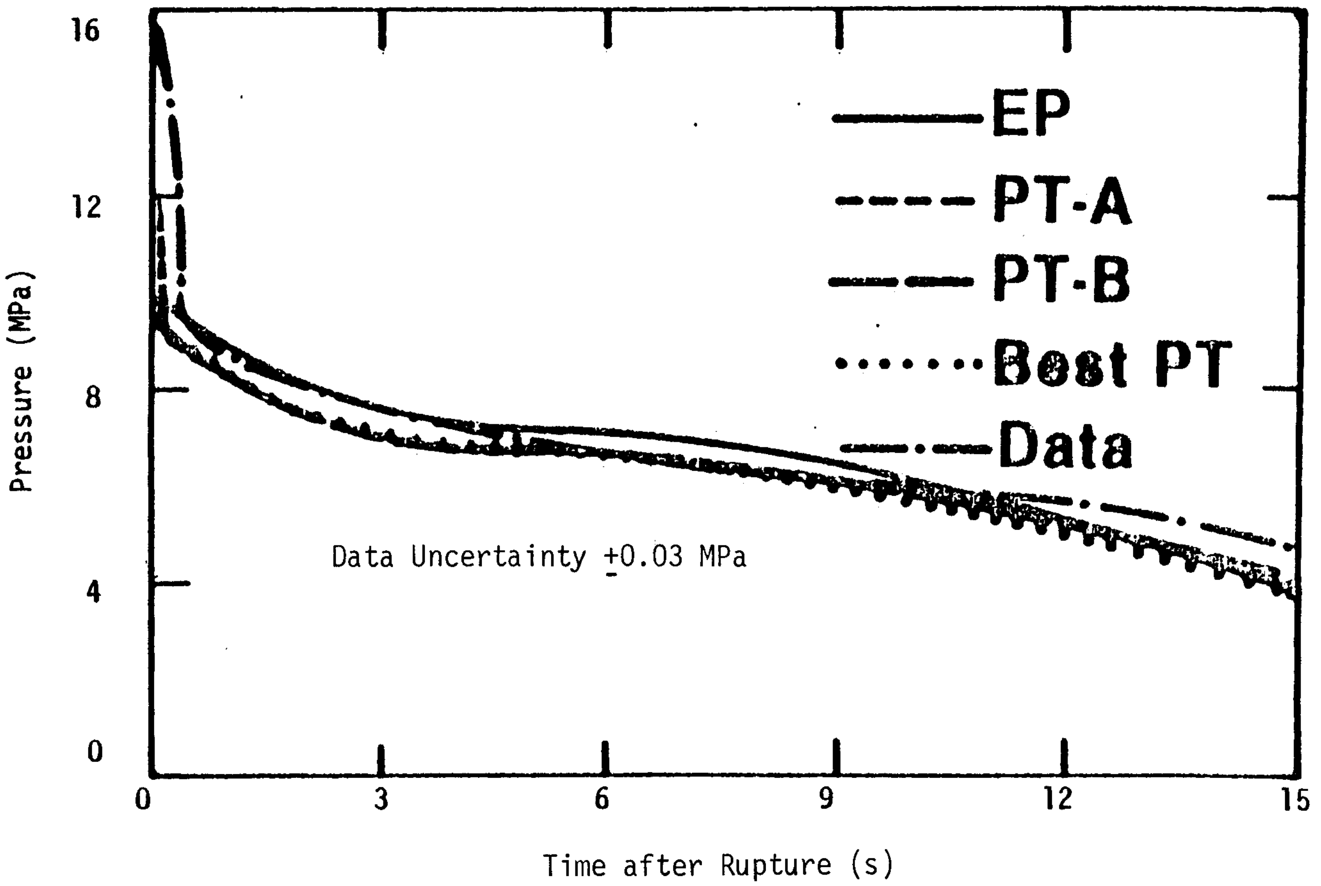


Figure 10. Posttest analysis of system depressurization for various critical flow choices for LOCE L2-2.

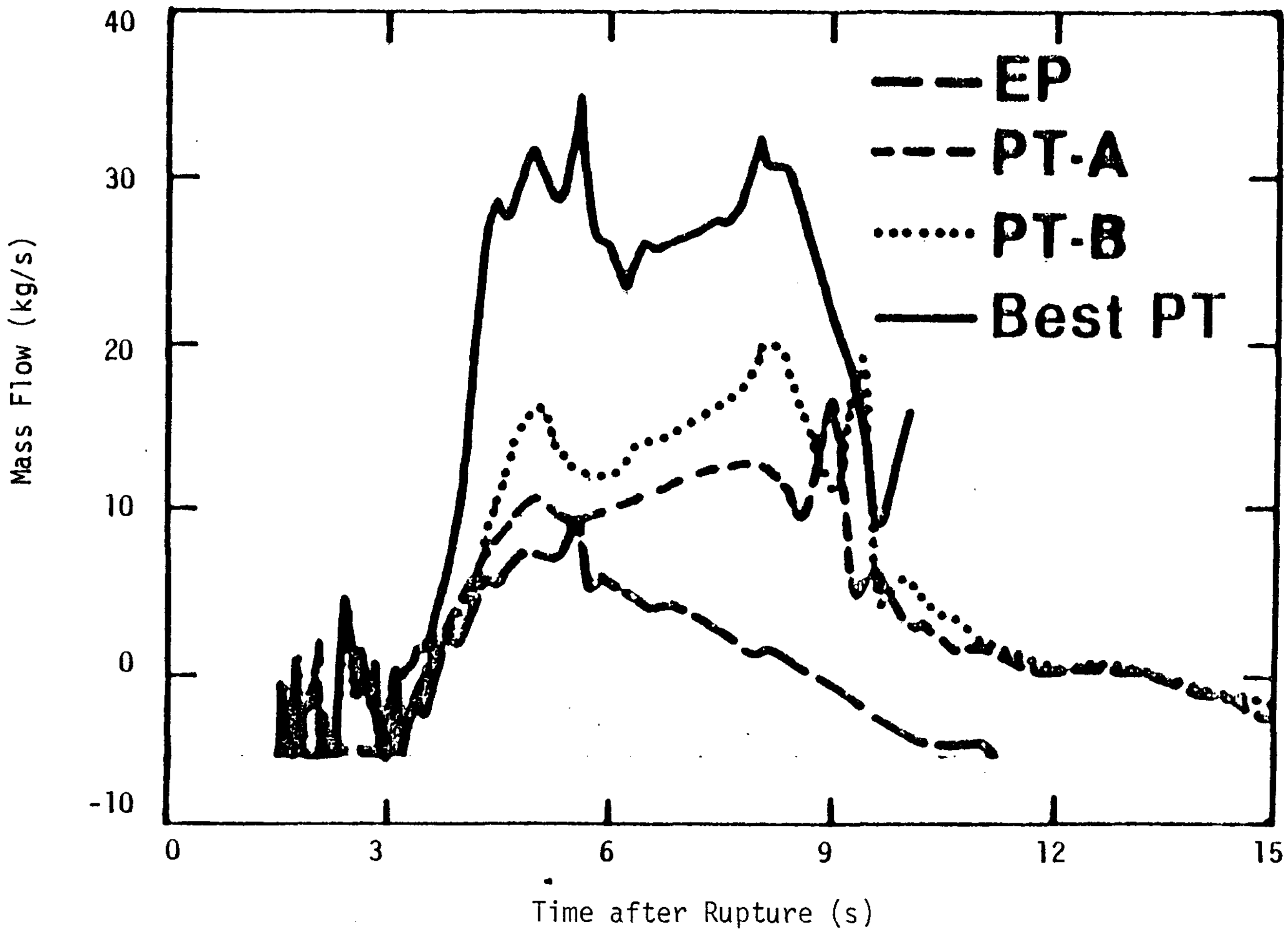


Figure 11. Posttest analysis of core inlet upflow for various critical flow choices for LOCE L2-2.

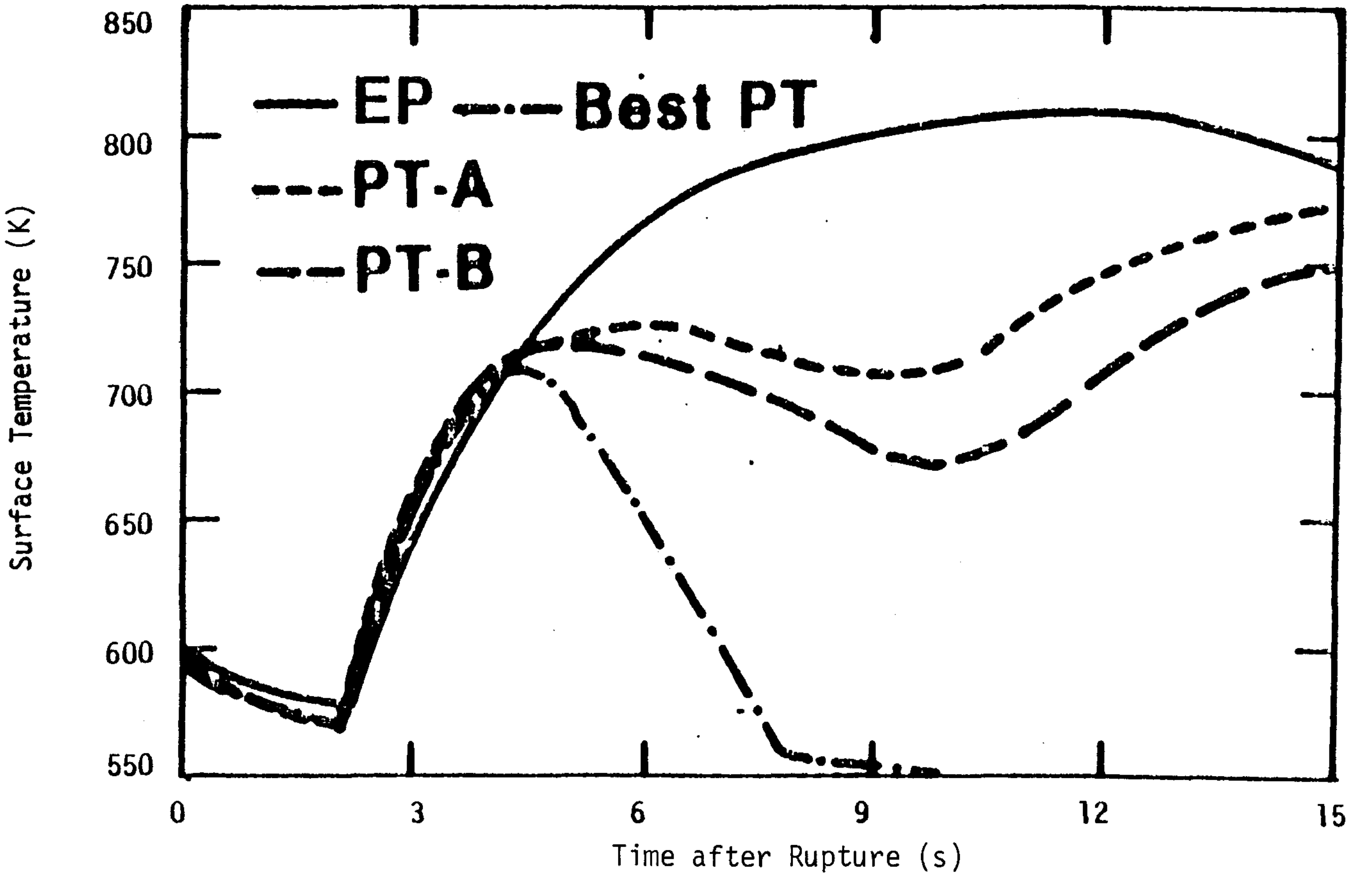


Figure 12. Posttest analysis of system model cladding temperature for various critical flow choices for LOCE L2-2.

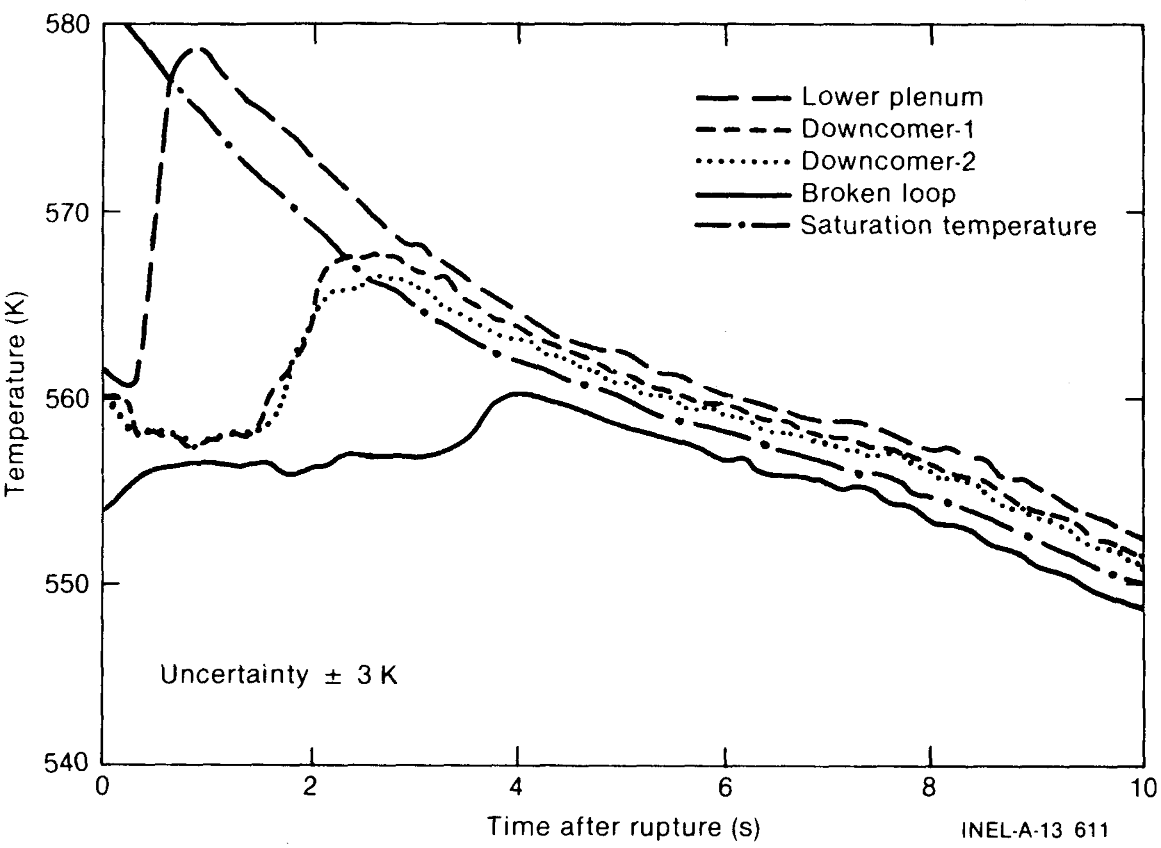


Figure 13. Fluid temperatures at various locations for LOCE L2-2.

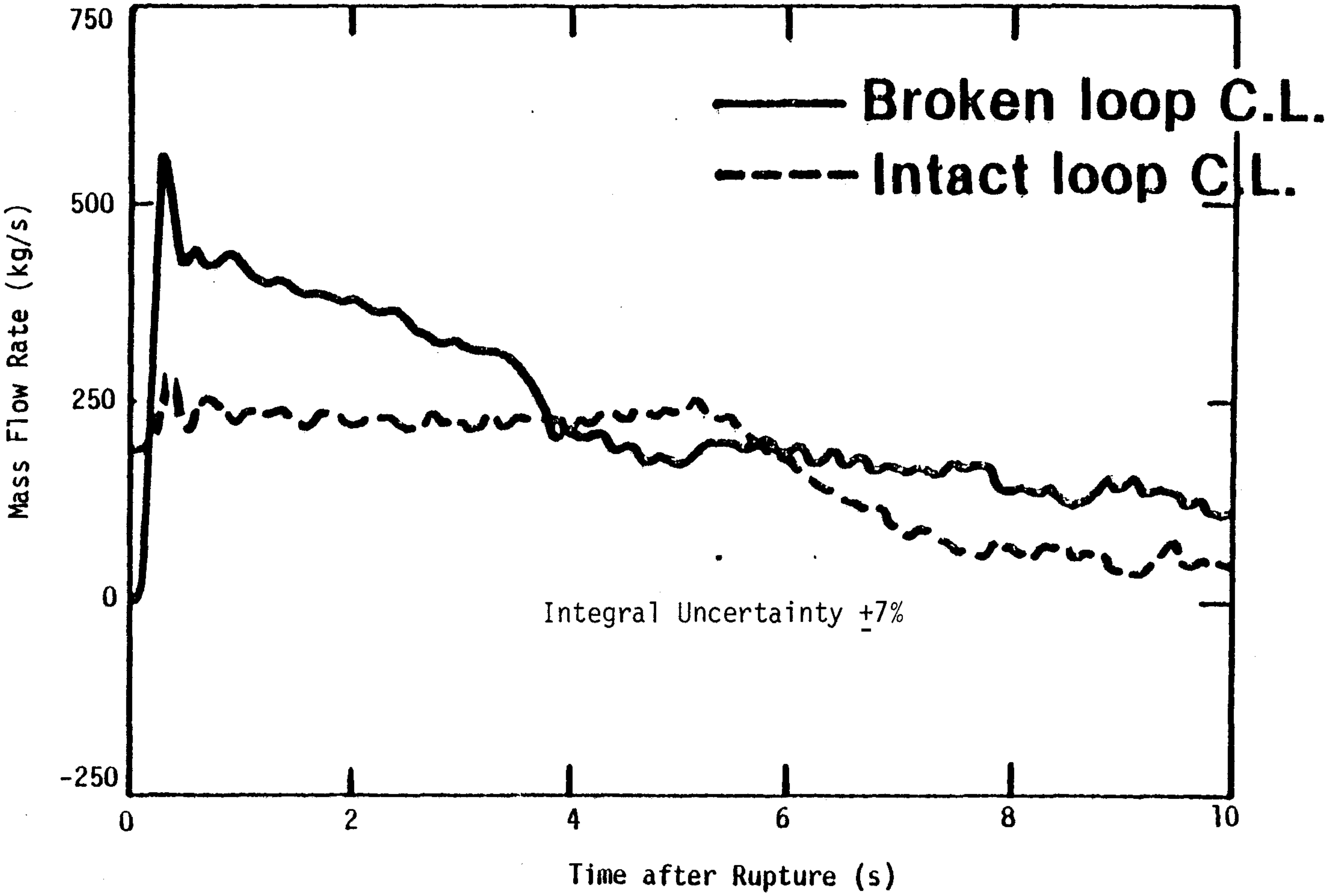


Figure 14. Measured mass flow rates for LOCE L2-2.

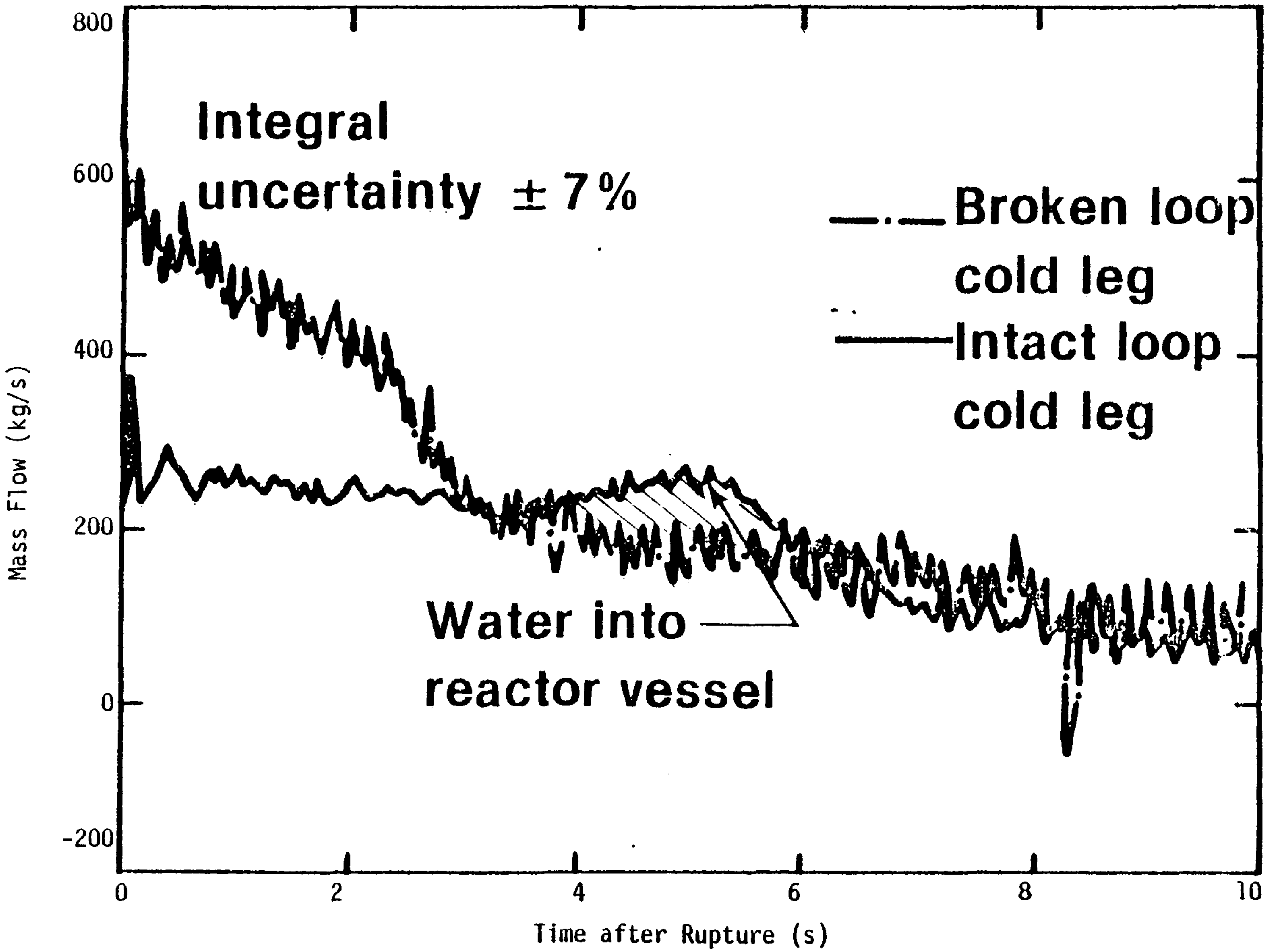


Figure 15. Measured cold leg mass flows for LOCE L2-3.

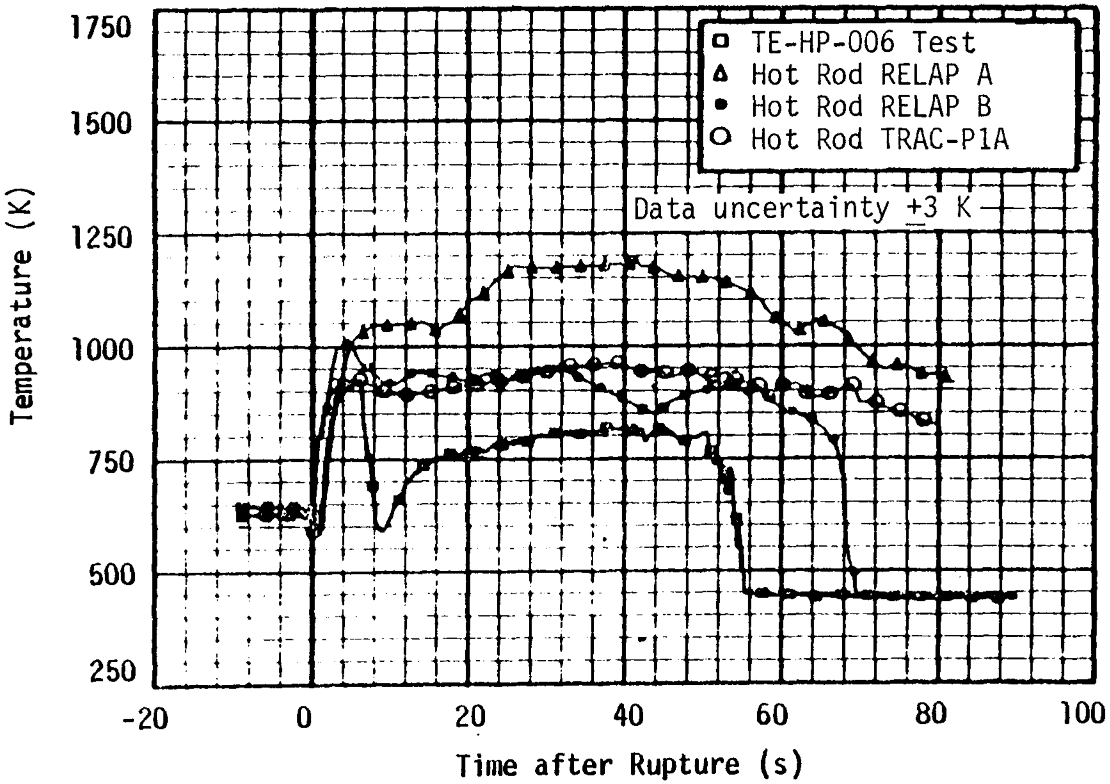


Figure 16. Comparison of measured and calculated hot rod cladding temperatures for LOCE L2-3.

did not predict rewet, the peak cladding temperature was better predicted. Also, review of the system hydraulics compared with the revised prediction showed excellent agreement. Figure 16 includes the cladding temperature prediction made with TRAC-P1A. This prediction does not show the rewet either. An alternate TRAC-P1A prediction with a revised rewet criteria, however, did predict the rewet. Posttest analysis of LOCE L2-3 is continuing, but it now appears that additional work on the heat transfer models is required in order to obtain best estimate calculations of cladding temperature in large PWRs.

Relationship to a Large PWR

In order to evaluate the relationship of the LOFT results to large commercial PWRs, a RELAP4/MOD6 analysis of a large PWR was made using the same model options as applied in the LOFT LOCE L2-2 Best PT model and the initial conditions for LOCEs L2-2 and L2-3. That is, the large PWR in both cases was run at a power sufficient to reproduce the core fluid temperature rise (core ΔT) forecast for LOCEs L2-2 and L2-3.

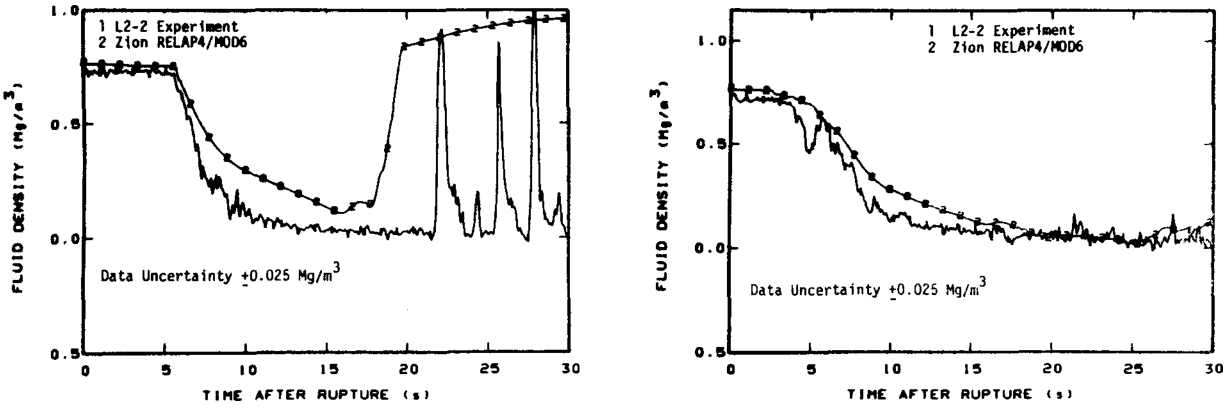
Figures 17 and 18 are comparisons of the best estimate predictions for the large PWR with the LOFT data for LOCEs L2-2 and L2-3, respectively. The excellent agreement shown implies the following:

1. The LOFT system is a good model of expected large PWR behavior (that is, scaling is accurate) during a large LOCA

2. The large PWR LOCA consequences are much less than previously thought

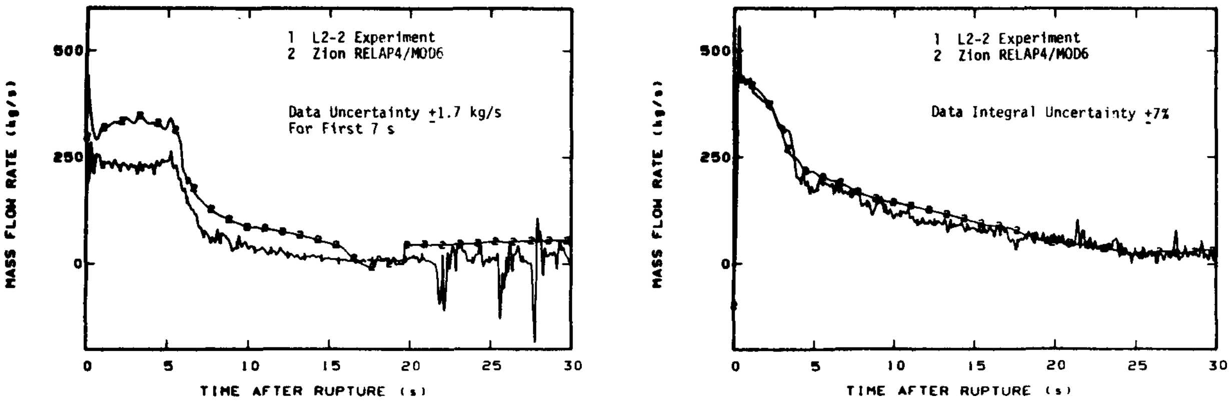
3. The peak cladding temperatures that would occur in the large PWR are probably less than occurred in LOFT, since the LOFT prediction with the same model yields higher cladding temperatures.

This information has been transmitted to the licensing portion of the NRC and is currently under evaluation.



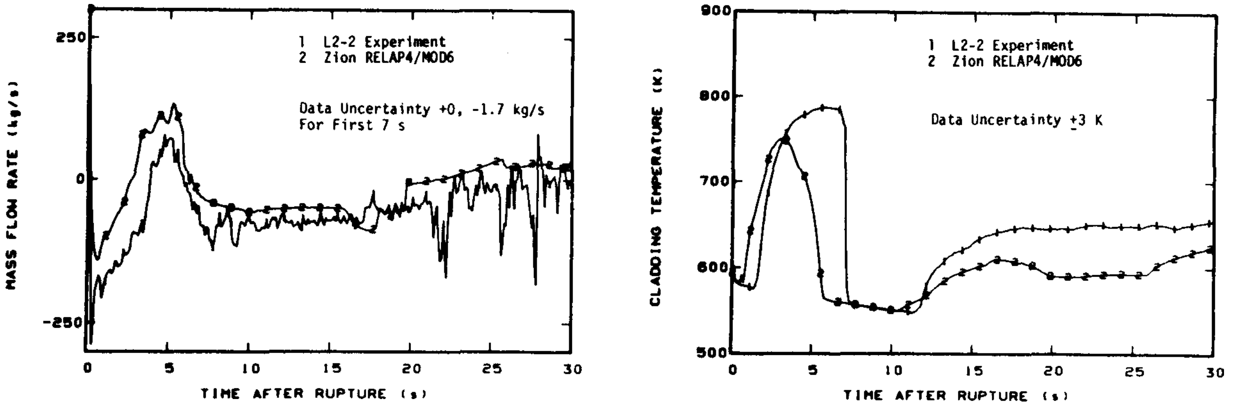
INTACT LOOP COLD LEG DENSITY

BROKEN LOOP COLD LEG DENSITY



INTACT LOOP COLD LEG MASS FLOWRATE

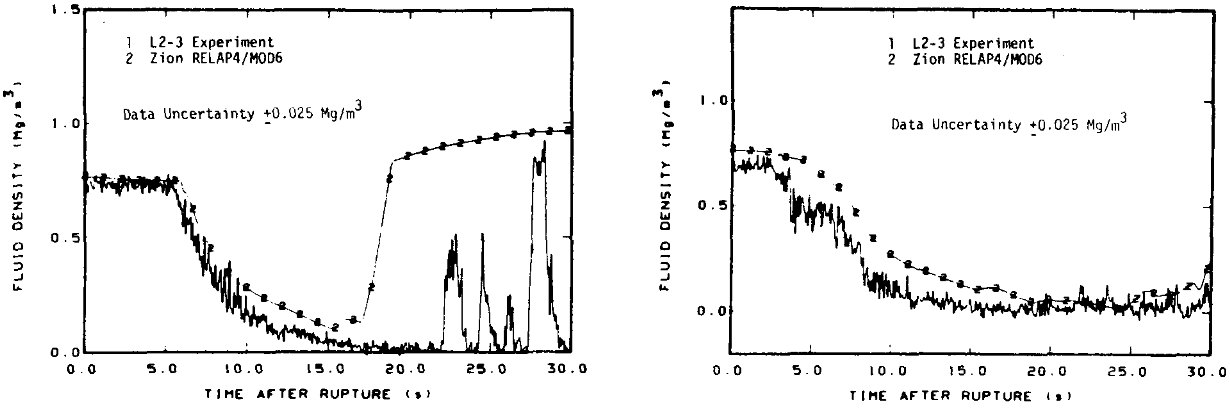
BROKEN LOOP COLD LEG MASS FLOWRATE



DIFFERENCE BETWEEN INTACT LOOP AND BROKEN LOOP COLD LEG MASS FLOWRATES

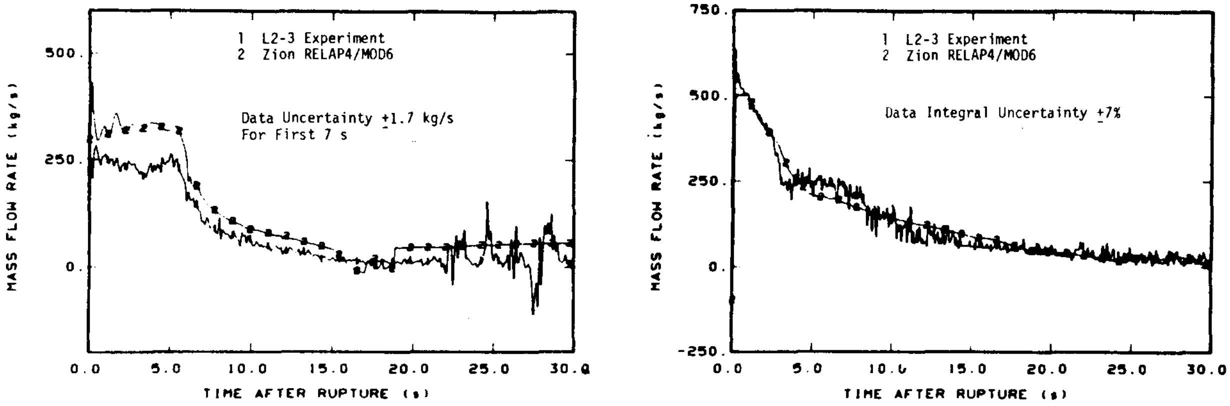
CLADDING TEMPERATURE IN THE PEAK POWER REGION

Figure 17. LOFT data and Zion prediction comparisons for LOCE L2-3 initial conditions.



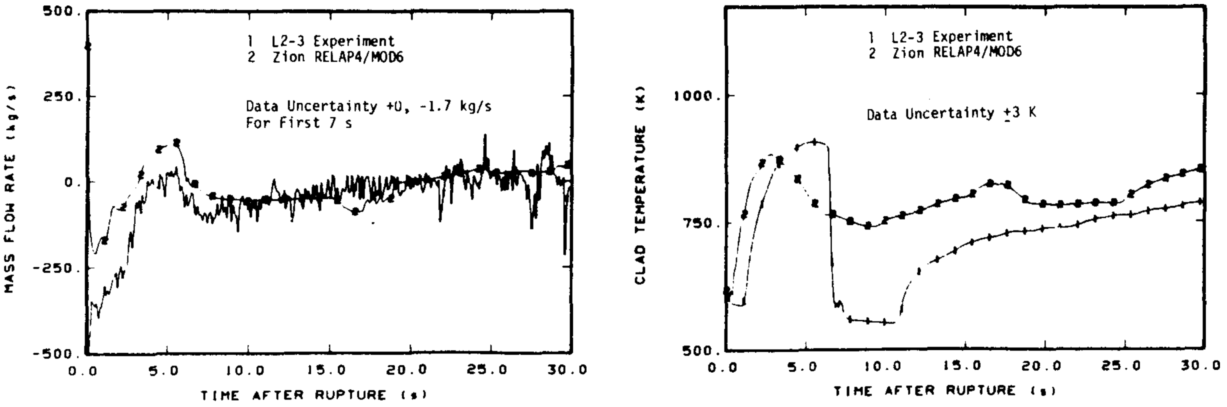
INTACT LOOP COLD LEG DENSITY

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 18. LOFT data and Zion prediction comparisons for LOCE L2-3 initial conditions.

LOCE L3-0

LOCE L3-0 was conducted with the same system and core configuration as LOCEs L2-2 and L2-3. Only minor instrumentation changes were made in order to record data for a longer time. LOCE L3-0 was an isothermal experiment with negligible nuclear decay heat power. The primary coolant pumps were tripped at the initiation of the transient. No ECC was used in LOCE L3-0. Initial conditions are given in Table 4. The experiment was initiated by opening the power operated relief valve (PORV), rather than the quick-opening blowdown valves. The experiment was terminated by opening the quick-opening valves after the pressure had reduced below normal accumulator pressure, and the LPIS was used to refill the system. The sequence of events for LOCE L3-0 is given in Table 5.

TABLE 4. LOCE L3-0 INITIAL CONDITIONS

Parameter	Measured Value
Primary system mass flow (kg/s)	201
Pressurizer pressure (MPa)	14.7
Primary system fluid temperature (K)	557
Decay heat level (kW)	4.2
Steam generator pressure (MPa)	6.8

The data from LOCE L3-0 were not processed for 3 weeks following the experiment in order to complete computer calculations of experiment performance without the benefit of the experimental data. Since the flow characteristics of the PORV were relatively uncertain, it was necessary to use data from the early portion of the transient to determine a valve coefficient for steam flow. The valve steam flow characteristics and experiment initial conditions data were incorporated into the computer

TABLE 5. CHRONOLOGY OF EVENTS FOR LOCE L3-0

Event	Time After LOCE Initiation (s)
LOCE initiated	0
PSMG[a] power tripped	11
PCP[b] coastdown completed	15
Pressurizer reached minimum indication	48
Primary system reached saturation pressure	48
Pressurizer indicated full	73
Pressurizer returned to indicating range	1420
Broken loop isolation valves opened	2416
Quick-opening blowdown valves opened	2460
End of saturation blowdown	2490
LPIS initiated	2535

a. PSMG - primary system motor generator.

b. PCP - primary coolant pump.

calculations performed with the RELAP4/MOD6[a], RELAP4/MOD7[b], RELAP5[c], and TRAC-P1A[d] computer codes. (Only 87 s of transient time were calculated with RELAP4/MOD7.)

The measured and calculated pressure response are shown in Figure 19. While all of the computer models accurately calculated the steam depressurization phase, none of the models were accurate for the long-term

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[a] RELAP4/MOD6, Idaho National Engineering Laboratory Configuration Control Number H007561B.

[b] RELAP4/MOD7, Idaho National Engineering Laboratory Configuration Control Number H013382B.

[c] RELAP5, Update Cycle 102, Idaho National Engineering Laboratory.

[d] TRAC-P1A, Los Alamos Scientific Laboratory.

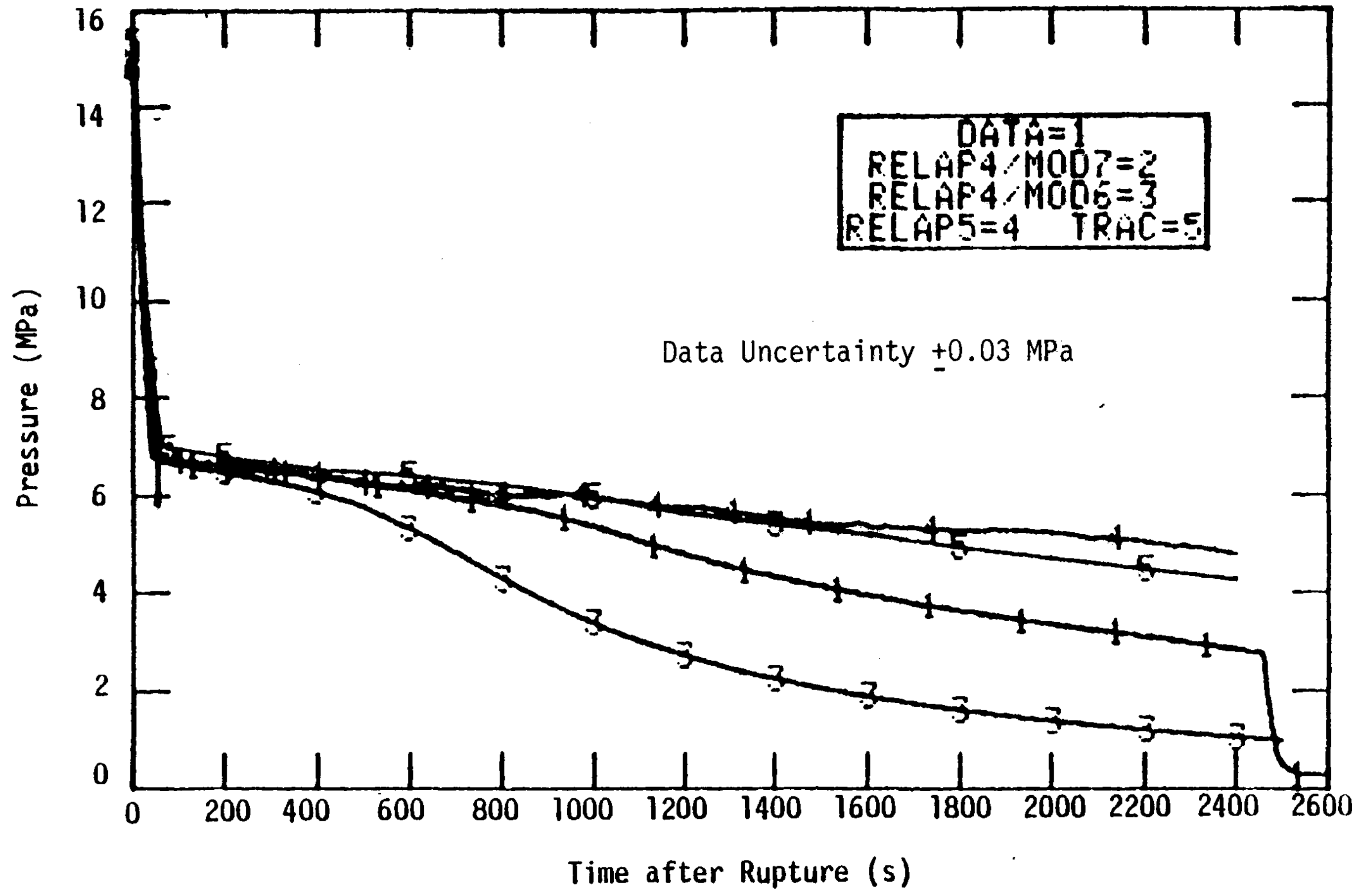


Figure 19. Calculated and measured primary system pressure for LOCE L3-0.

pressure response where saturated liquid and/or two-phase flow was passing through the PORV. The valve characteristics in two-phase flow and/or liquid discharge are unknown, so this result is not surprising.

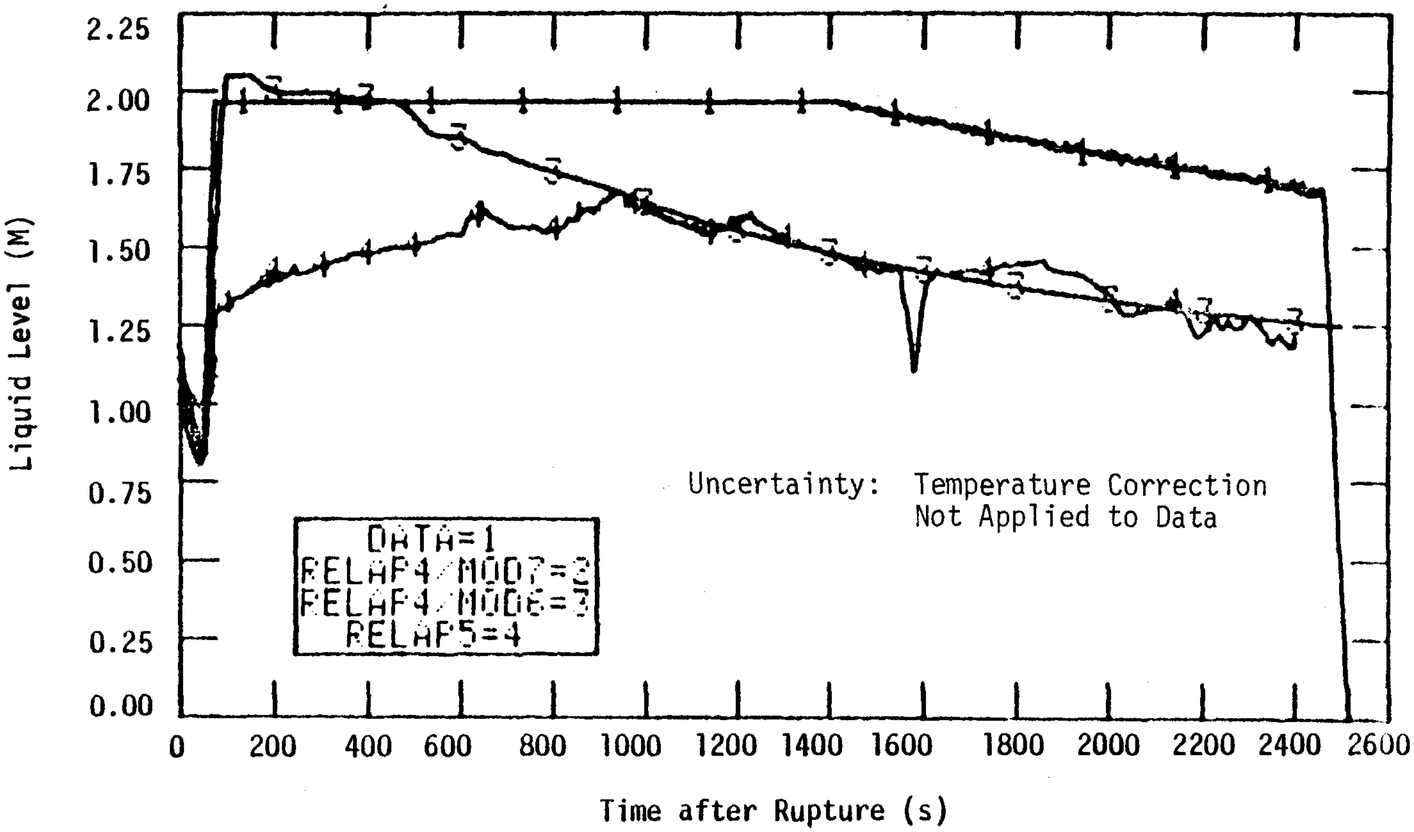
Figure 20 exhibits the measured and predicted pressurizer liquid level response. The data show the pressurizer rapidly filled after the system pressure reached system saturation and remained full until 1400 s. The indicated level shown was not compensated for fluid temperature and, thus, does not assure that the pressurizer was completely full for the whole time it was indicated full. In fact, correcting the indicated level for temperature makes it almost match the RELAP4/MOD6 calculation. While all the calculations showed the initial decrease and rapid increase in pressurizer level, neither TRAC-P1A (not shown) nor RELAP5 calculated a full pressurizer; whereas, RELAP4/MOD6 and RELAP4/MOD7 calculations did.

The measured density in the intact loop hot leg near the top of the pipe is shown in Figure 21. These data indicate stratified flow in the pipe with steam at the top. The surge line, which connects the pressurizer to the intact loop hot leg, connects to the top of the hot leg pipe. The LOFT surge line does not contain a loop seal; that is, the pipe from the hot leg to the pressurizer first rises, is horizontal, and then rises again. The pressurizer may have remained full after the time that steam was known to exist in the intact loop hot leg. However, this conclusion cannot be confirmed with measured data from LOCE L3-0.

CONCLUSIONS

The main conclusions reached from analysis of the LOFT LOCEs L2-2 and L2-3 are as follows:

1. The behavior of LOCEs L2-2 and L2-3 was dominated by the occurrence of a core-wide early rewet of the fuel rod cladding. This rewet occurred from a rereversal of core flow caused by a critical flow transition at the broken loop cold leg piping.



Uncertainty: Temperature Correction Not Applied to Data

Figure 20. Calculated and measured liquid level in pressurizer for LOCE L3-0.

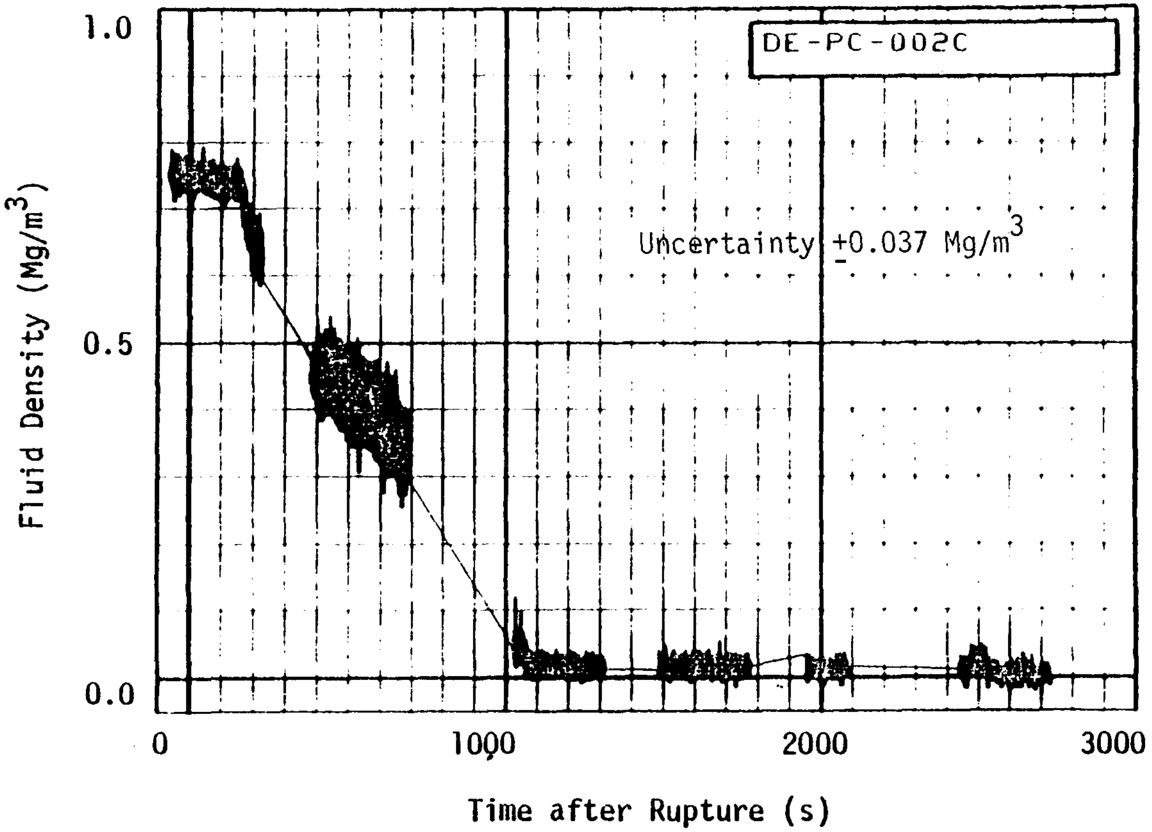


Figure 21. Fluid density in intact loop hot leg (chordal density) for LOCE L3-0.

2. Analytical model predictions of LOCEs L2-2 and L2-3 provided a good prediction of system hydraulics but did not predict the core-wide rewet. The model "deficiencies" identified as needing further work in order to improve the best estimate predictive capability (the original models are conservative) are as follows:

    a. Critical flow in the transition region from subcooled discharge to saturated discharge

    b. Wave transport of density variations, particularly as they affect discharge flow and rewet

    c. High pressure (≈6 MPa), low quality (0.1 to 0.3), low flow film boiling, and rewet heat transfer.

3. Analysis of a commercial PWR with the models that best predict LOFT behavior indicate the same expected response as observed in LOFT which means that LOFT is scaled correctly to represent commercial PWR behavior. Therefore, commercial PWRs are likely to experience the same cladding rewet in the event of a large LOCA which implies there is significantly more conservatism in the reactor safety analysis requirements of 10 CFR 50.46, Appendix K than previously envisioned.

The main conclusions reached to date from analysis of LOCE L3-0 are as follows:

1. During a LOCA caused by a break at the top of the pressurizer, for example, a PORV stuck open, the pressurizer starts filling when the primary system fluid saturates someplace other than the pressurizer itself. The pressurizer fills and may remain full even though steam exists in the hot leg piping and there is no loop seal in the pressurizer surge line.

2. The pressurizer level indication indicates a higher level than occurs in the pressurizer, as the indicated level is not compensated for fluid temperature.

3. The RELAP4/MOD6 model of LOCE L3-0 small break predicted the trends of the observed behavior, but more testing and modeling experience are needed to make the predictive capability for the small break as good as for the large break.

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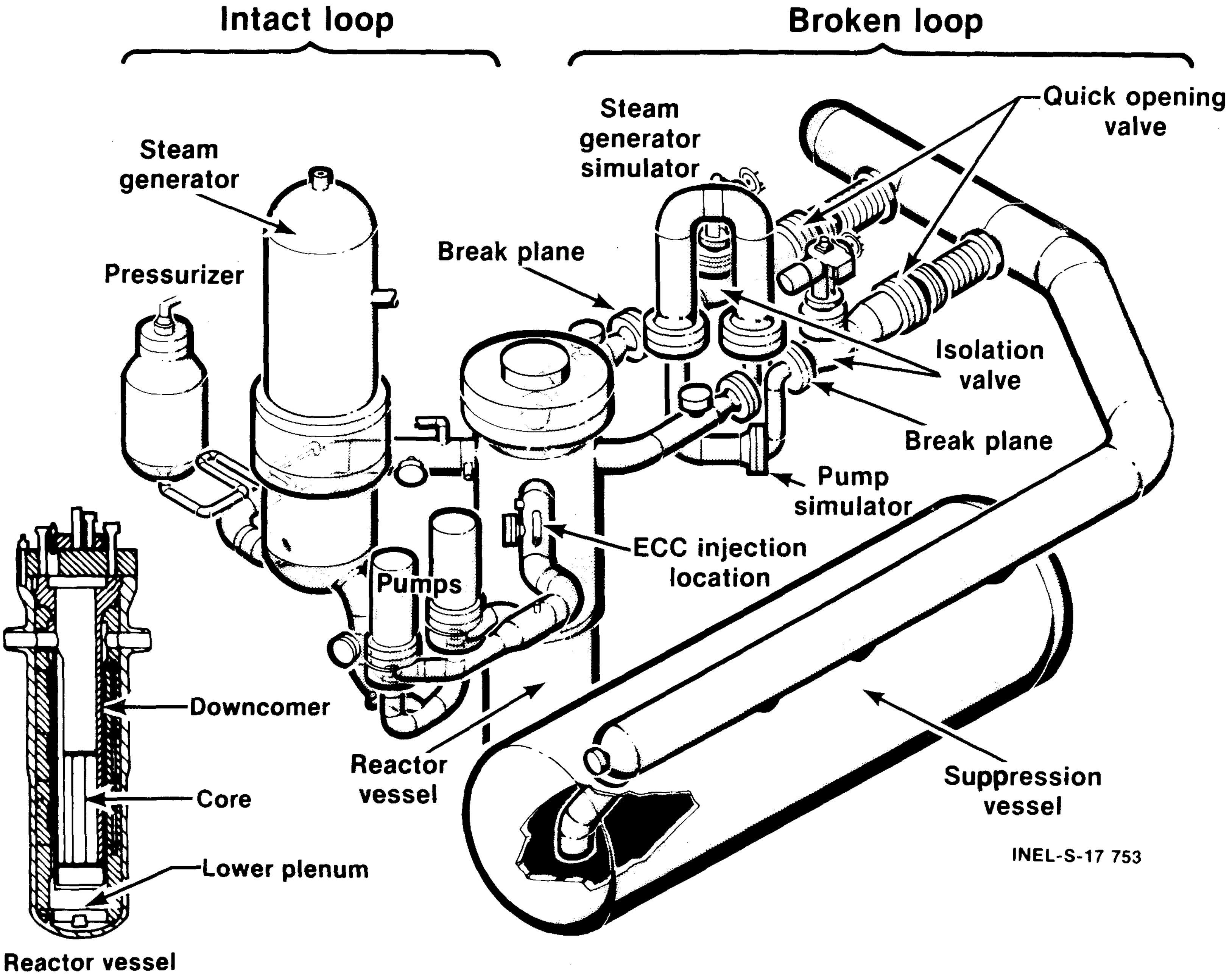
Results of LOFT Loss-of-Coolant Experiments

Presented by
J.H. Linebarger

EG&G Idaho, Inc.

LOFT System Configuration



LOCE s Completed Since October 1978

Designation	Type	Date
L2-2	Nuclear, 200% DECL	Dec. 9, 1978
L2-3	Nuclear, 200% DECL	May 12, 1979
L3-0	Isothermal, Small Break, PORV	May 31, 1979

Contents

- L2-2/L2-3 Results and Analysis
- Blowdown Prototypic Study
- L3-0 Results and Analysis
- Summary Conclusions

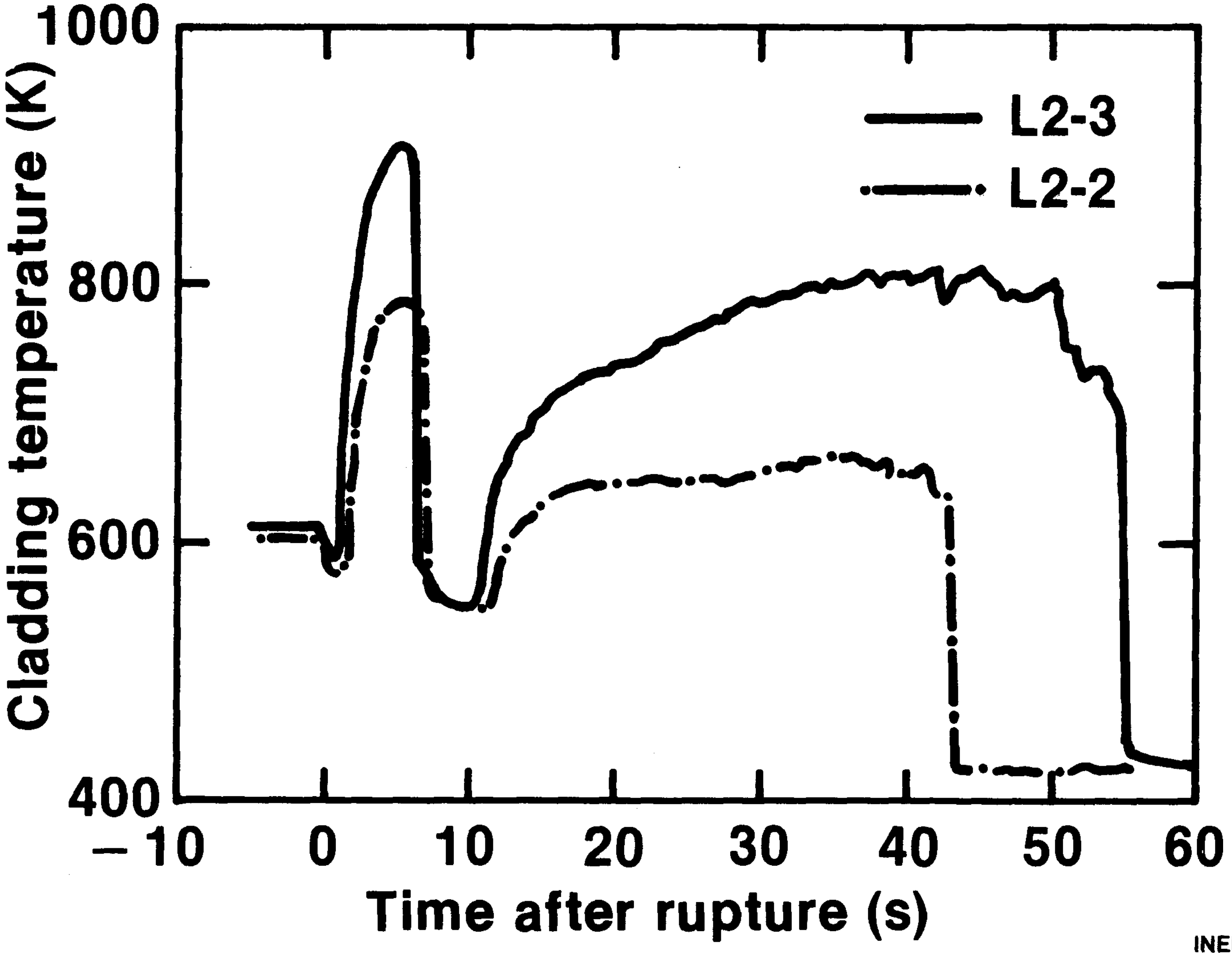
L2-2/L2-3 Initial Conditions

	LOCE L2-2	LOCE L2-3
MLHGR (kW/m)	26.4	39.4
Power (MW)	24.9	36.7
Mass flow (kg/s)	194.2	199.8
ΔT (K)	22.7	32.2
P (MPa)	15.64	15.06

L2-3 Replica of L2-2

- Blowdown dominated
- Core thermal response tightly coupled to hydraulics
- Same event sequence

L2-2/L2-3 Cladding Temperature Comparison



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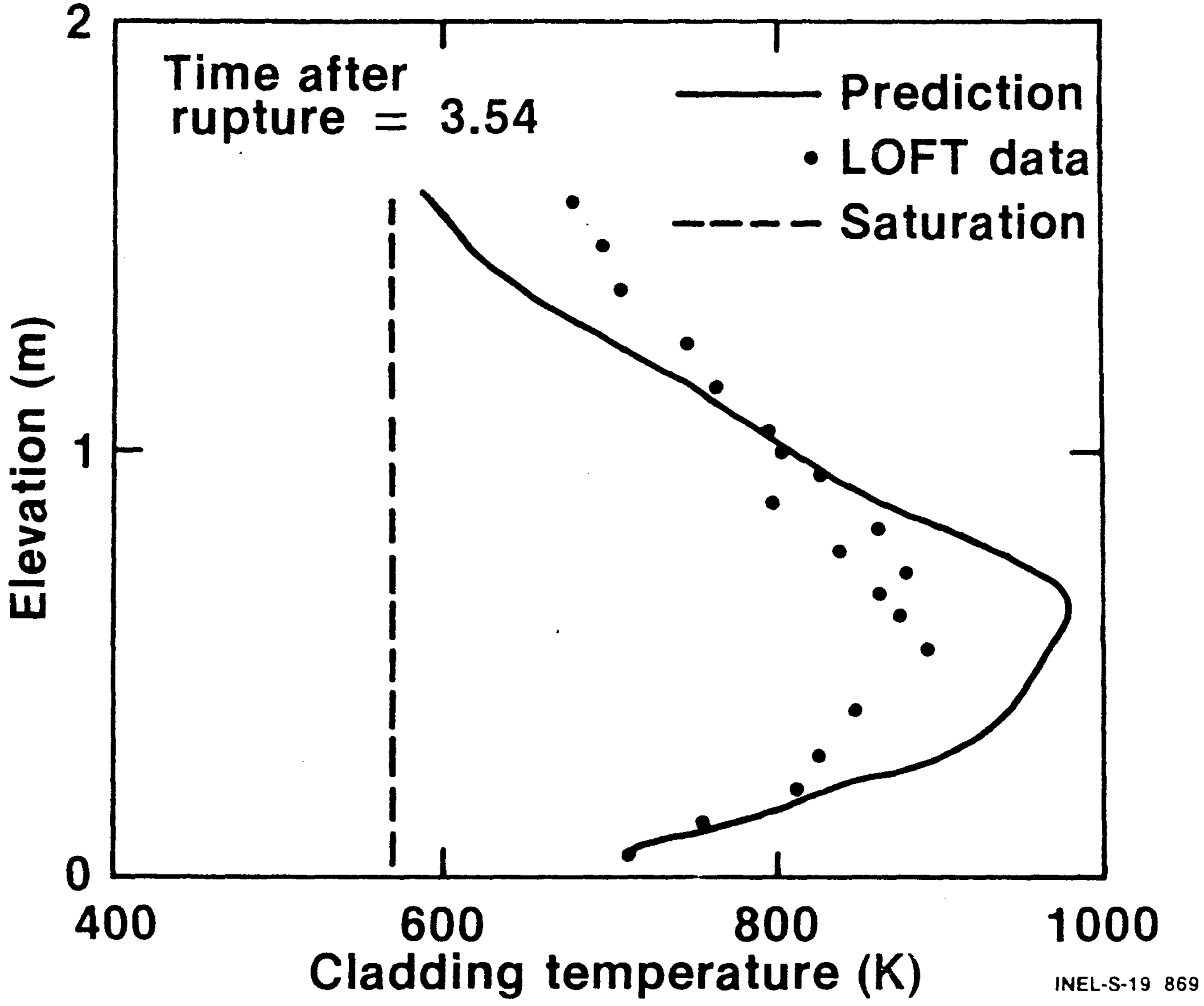
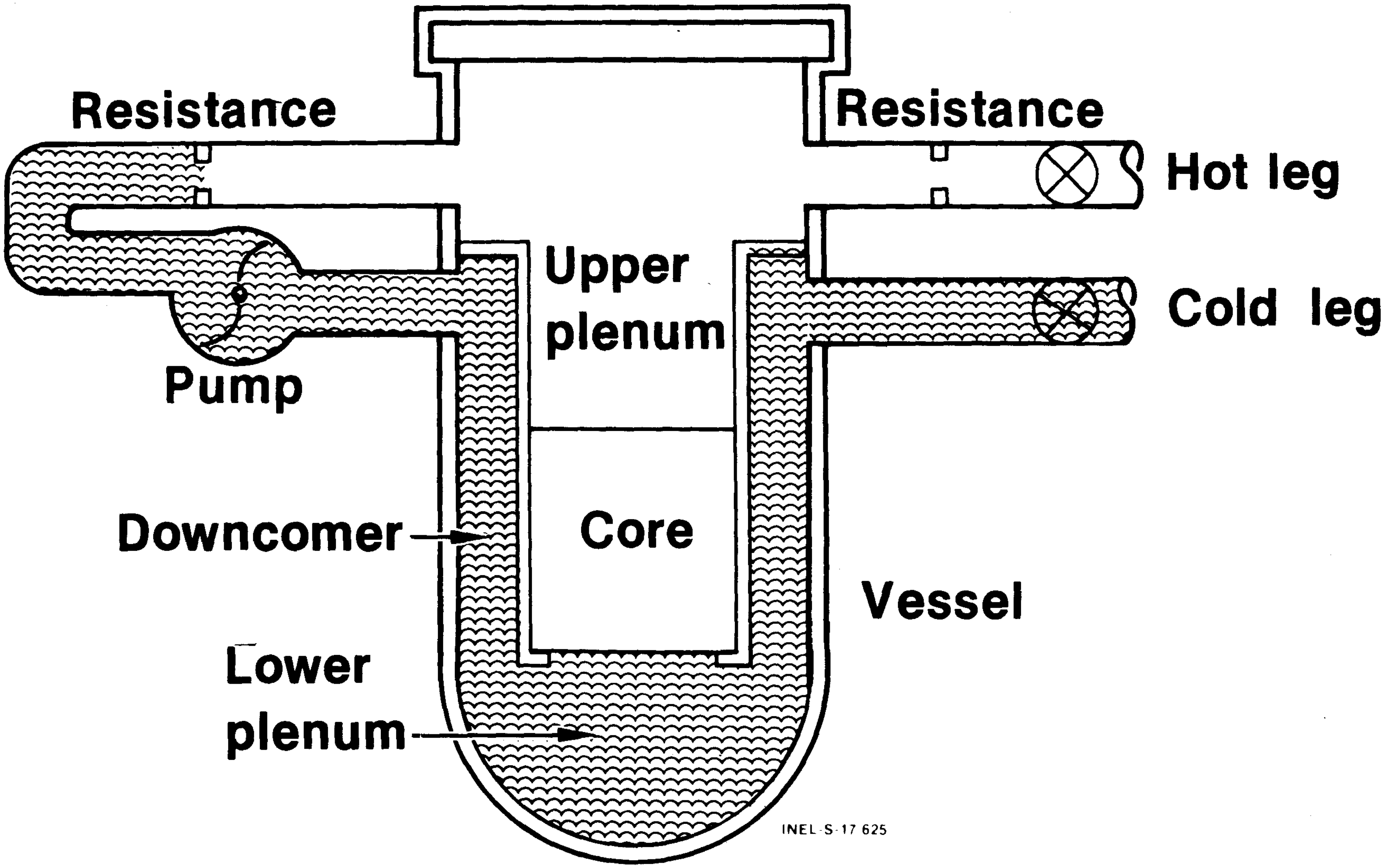
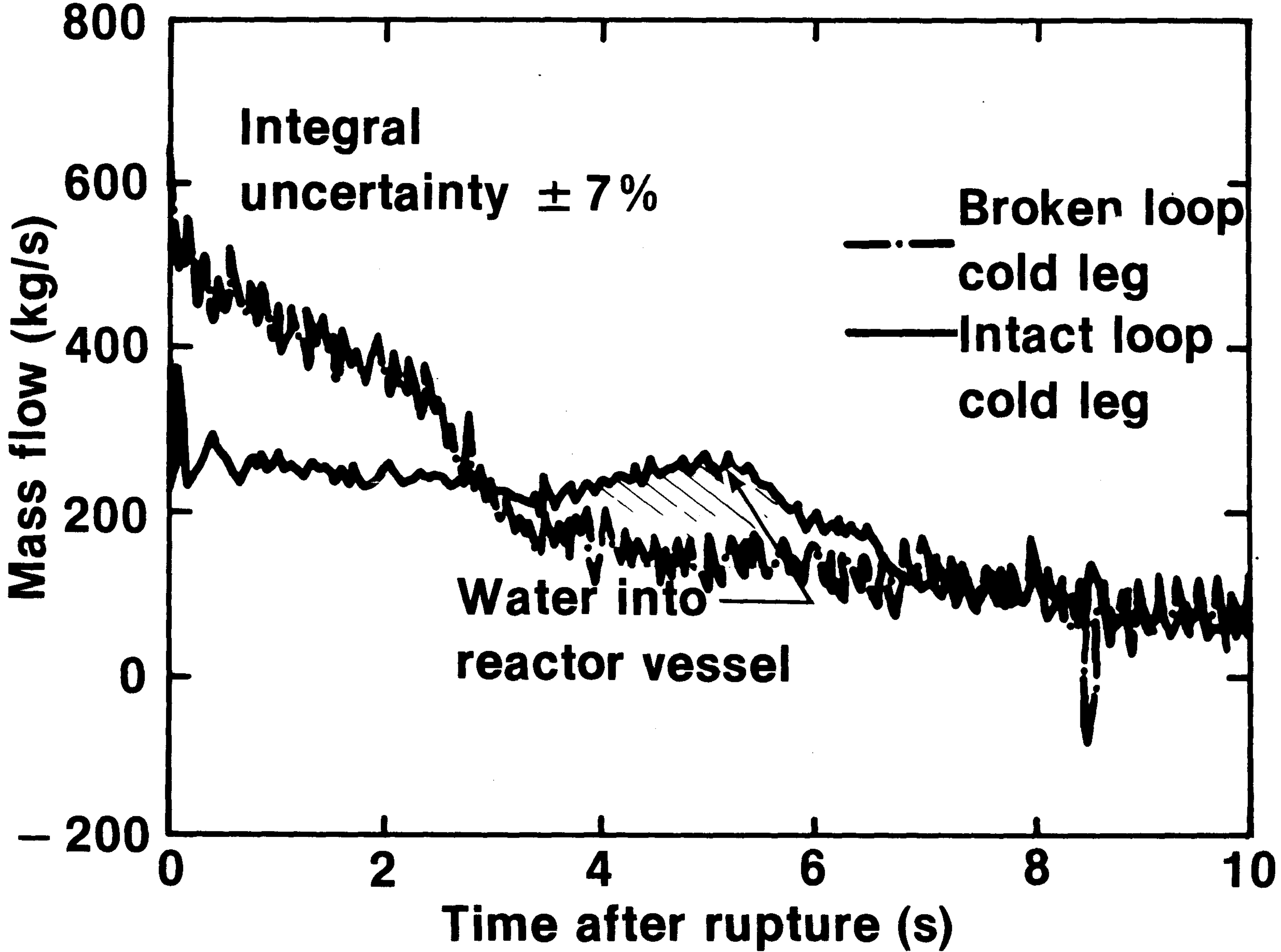
Thermocouples Clustered About Rod 5J8



Simplified Primary Coolant System



L2-3 Cold Leg Mass Flows



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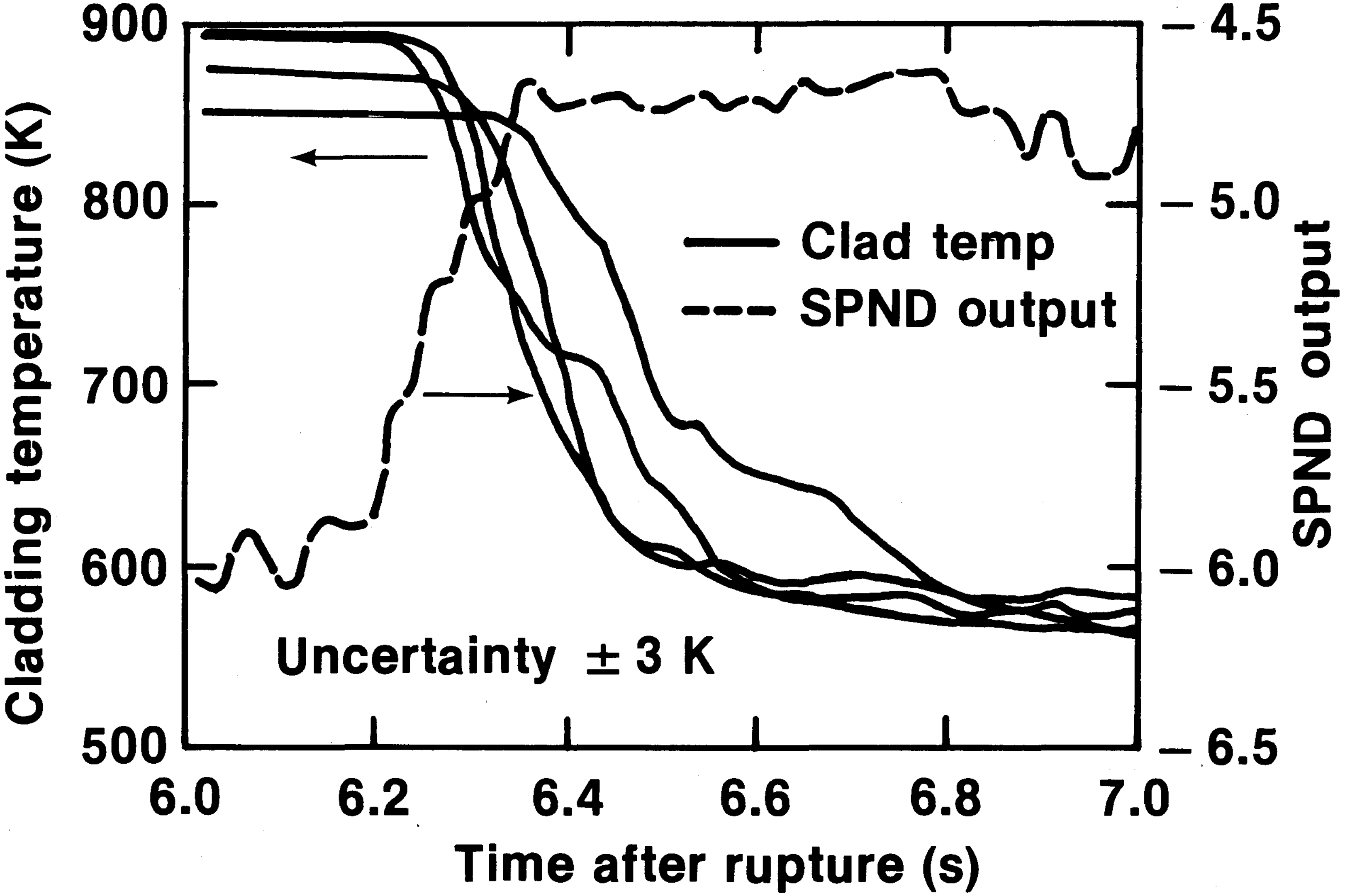
L2-3 Cladding Temperature vs SPND Output



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Conclusions

L2-2/L2-3 Results:

- LOCES - blowdown dominated and self-limiting.
- During blowdown — thermal tightly coupled to hydraulics
- During blowdown — phenomena relative magnitudes/timing consistent with initial power/core ΔT.

Conclusions (cont'd)

L2-2/L2-3 results: (cont'd)

- ECC fluid delivery not significantly different from L1-5

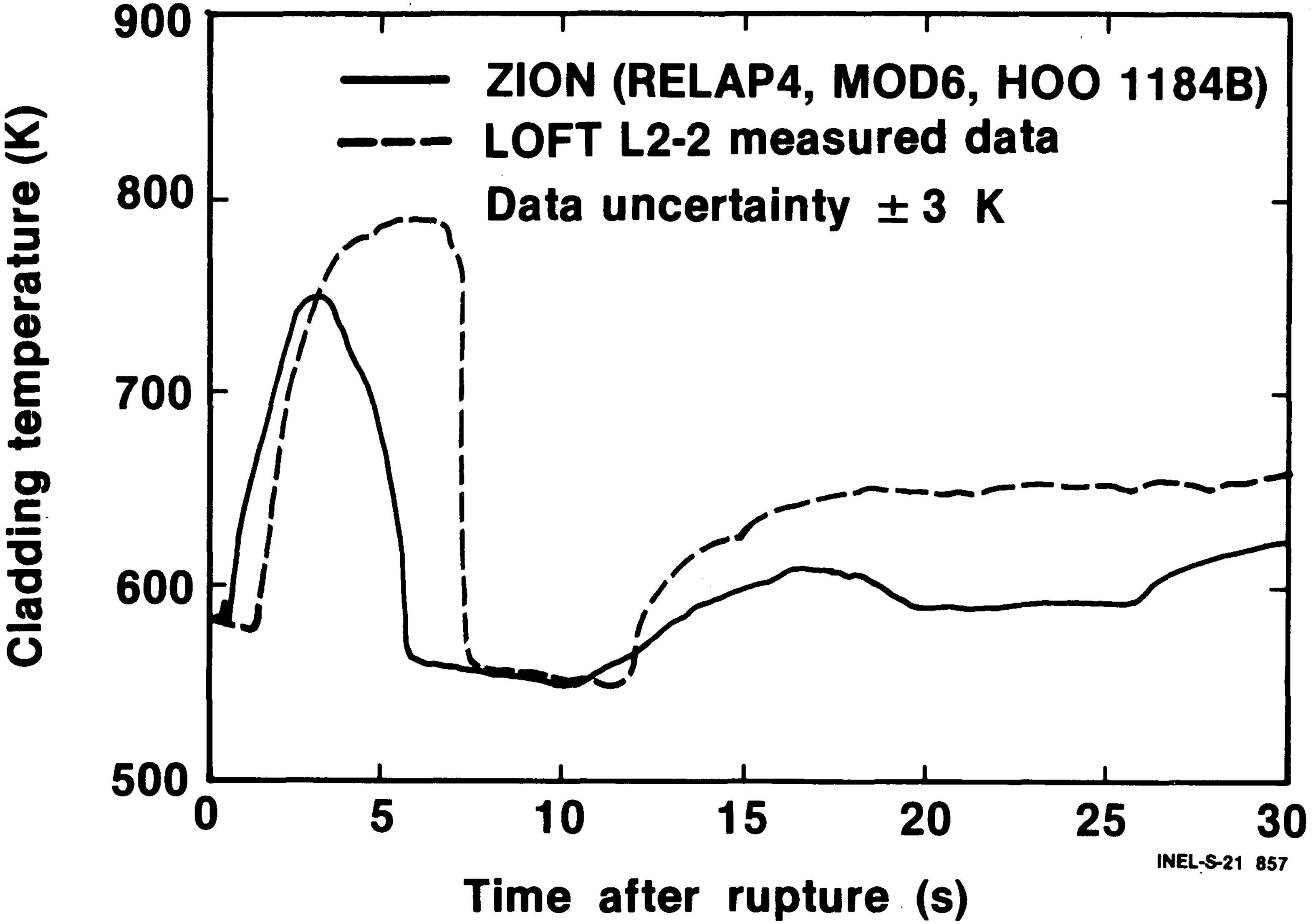
- Core rewetting during L1-5 and L2-2 virtually identical. However transient time increased 22% during L2-3

- Best-estimate calculations conservative and developmental areas are known

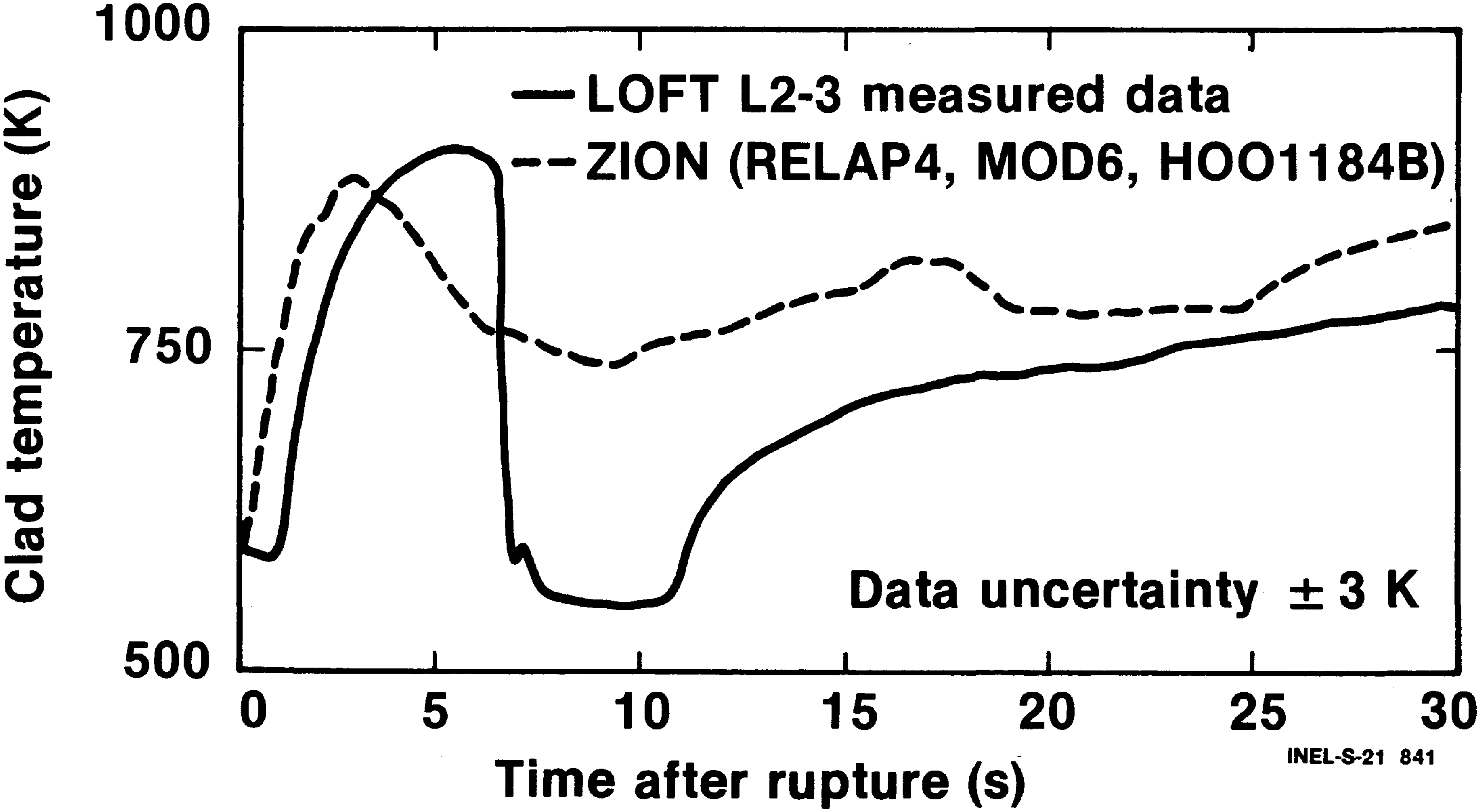
Initial Conditions

	L2-2	ZION	L2-3	ZION
Mass flow (kg/s)	194.20	18395	199.80	18395
ΔT (K)	22.70	23.90	32.20	35.80
Power (MW)	24.88	2296.60	36.70	3540.0
MLHGR (kW/m)	26.40	25.60	39.40	39.40
Tc (K)	557.70	549.80	560.70	549.80
P (MPa)	15.64	15.42	15.06	15.43

Fuel Cladding Temperature Comparison



Fuel Cladding Temperature Comparison (L2-3)



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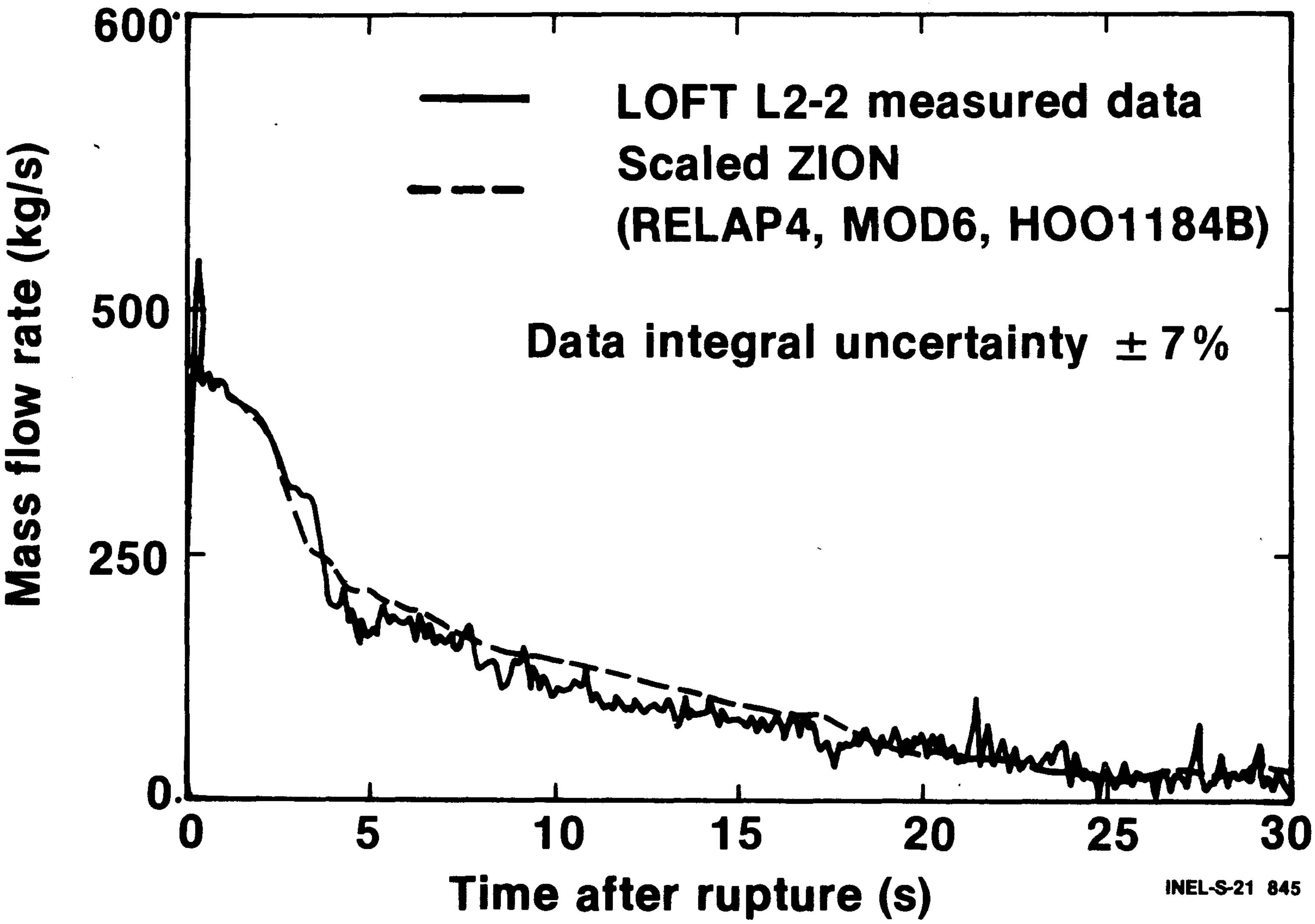
Comparison Criteria

- LOCE results
- Initial conditions
- Difference in scale

Broken Loop Cold Leg Mass Flow Comparison (L2-2)



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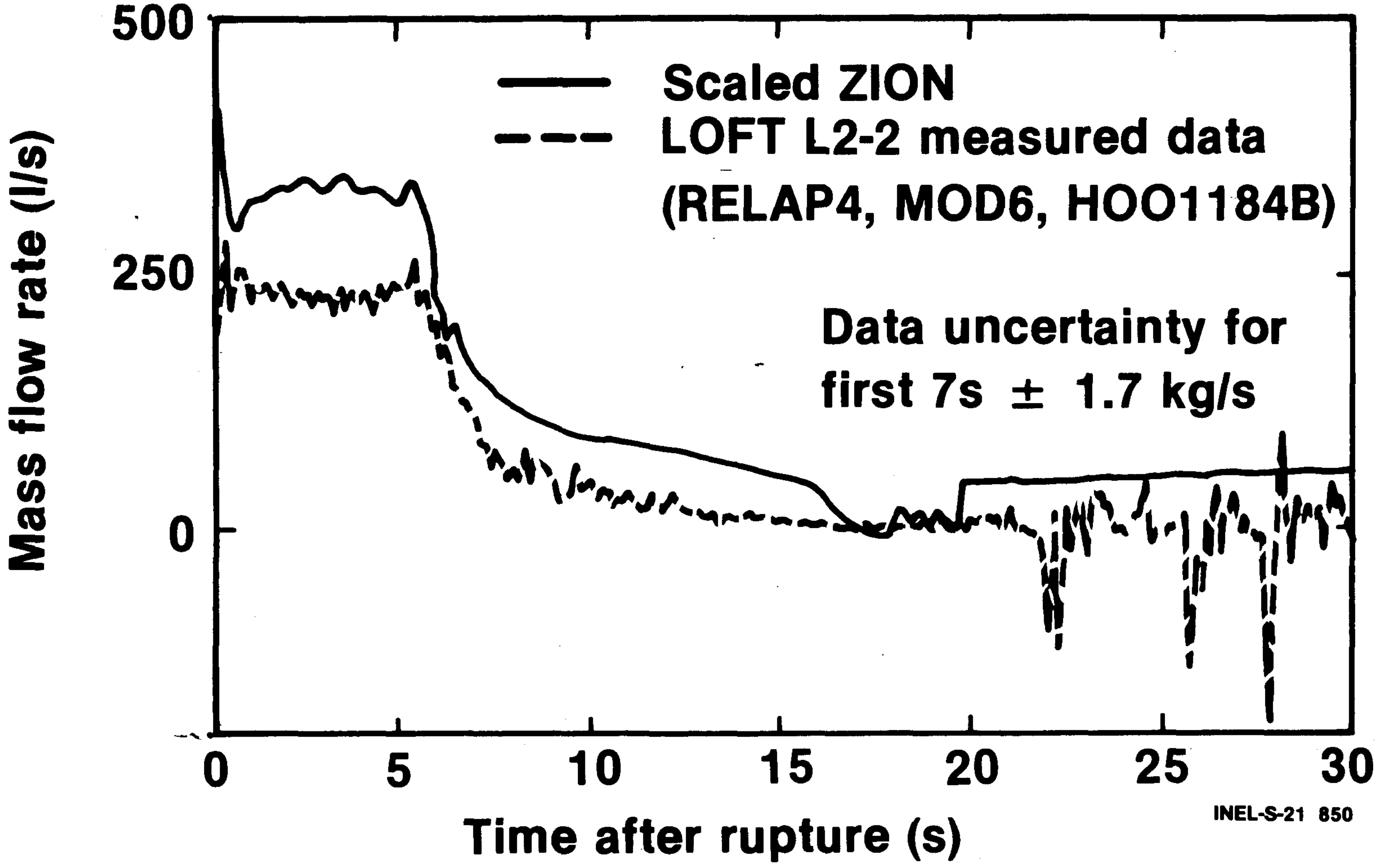
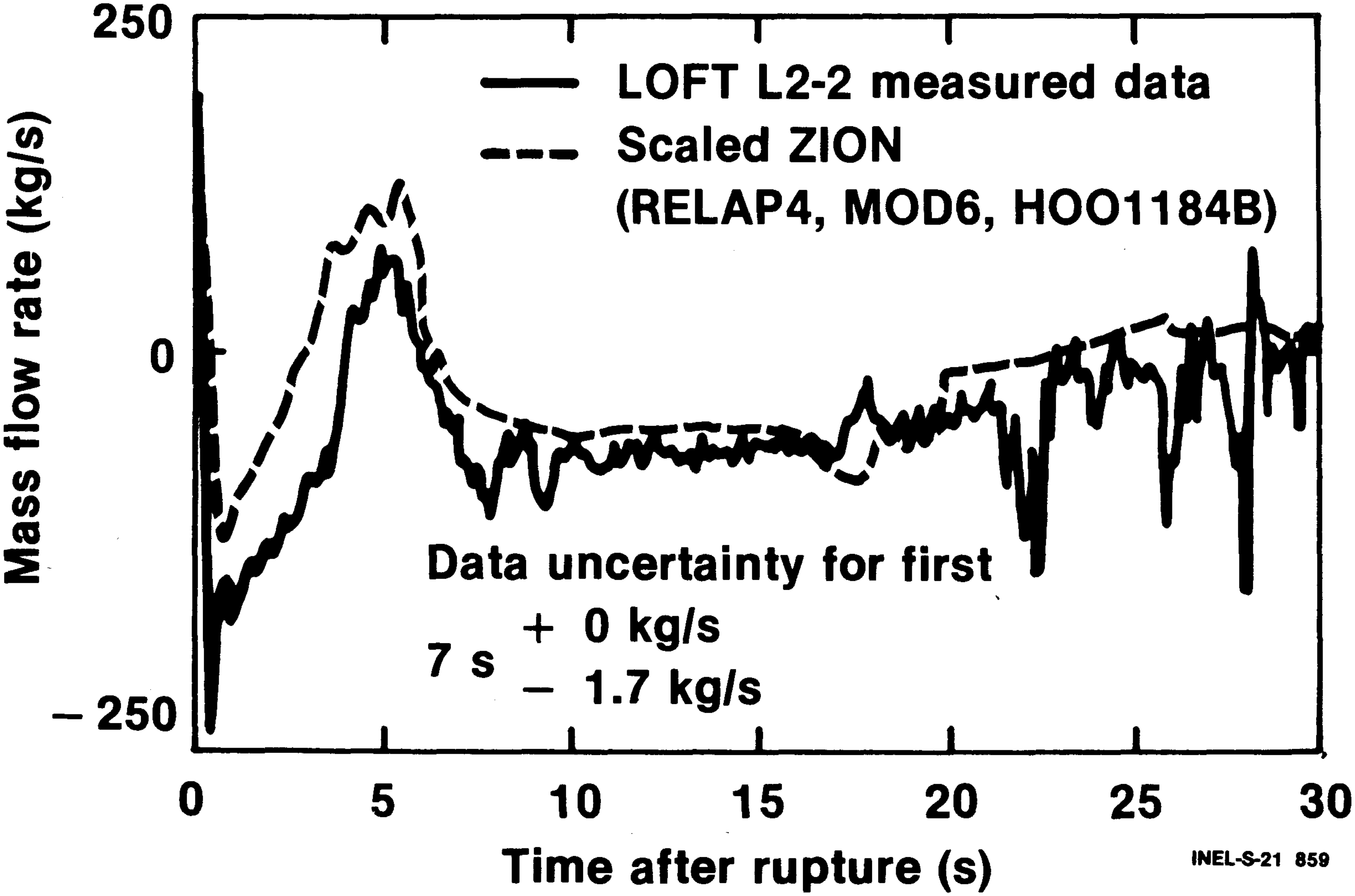
Intact Loop(s) Cold Leg Mass Flow Comparison (L2-2)



Data uncertainty for first 7s ± 1.7 kg/s

INEL-S-21 850

Cold Leg Mass Flow Balance
(L2-2 Intact-Broken)



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Conclusions (cont'd)

Blowdown prototypic study:

For the tests to run to date the LOFT results, relative to a commercial pressurizer water reactor, conservatively scale the dominant hydraulic phenomena during blowdown resulting in a realistic, if not conservative, indication of fuel cladding temperatures.

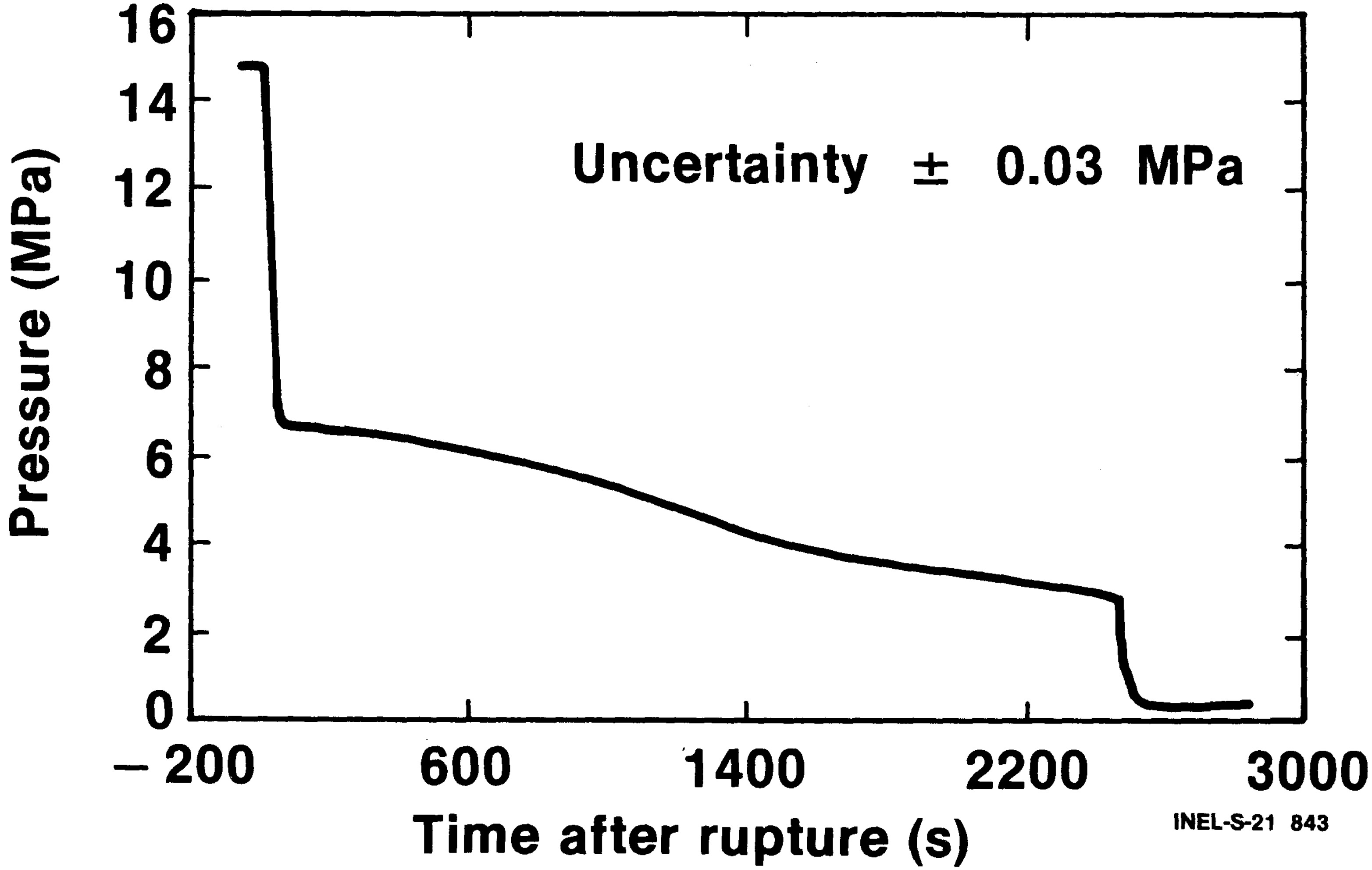
L3-0 Initial Conditions

P (MPa)	14.74 ± 0.07
Mass flow (kg/s)	201.0 ± 6.3
T_H (K)	556.7 ± 3.0
T_C (K)	559.7 ± 3.0

L3-0 Transient Conditions

Pumps tripped (s)	0 ± 0.5
Decay heat (kW)	4.2 ± 1.0
Ambient heat loss, pumps off (kW)	140.0 ± 20.0
Auto. accumulator injection control	Disabled
HPIS injection	Logic unsatisfied

L3-0 Primary System Pressure



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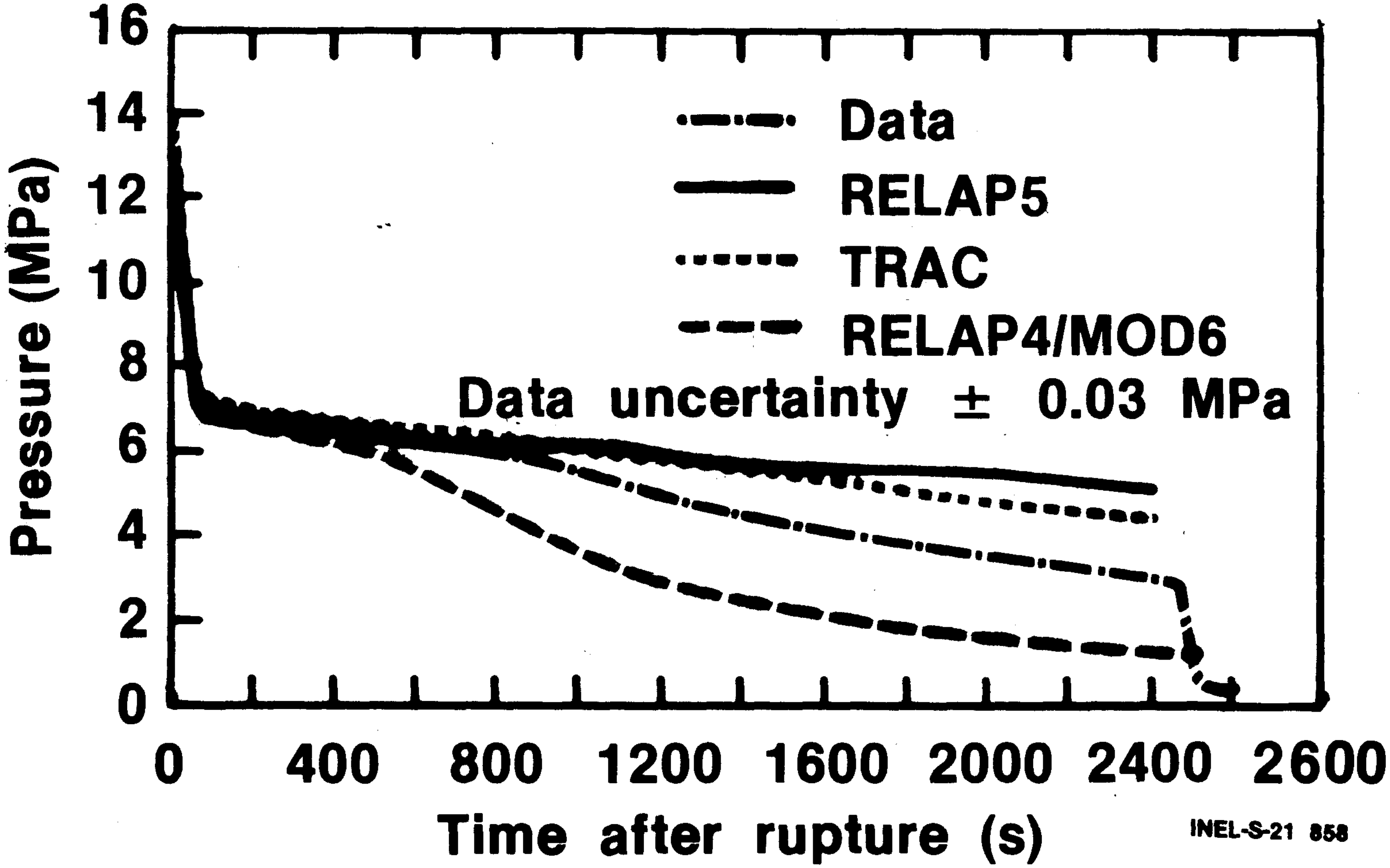
L3-0 Predicted and Measured Primary System Pressure



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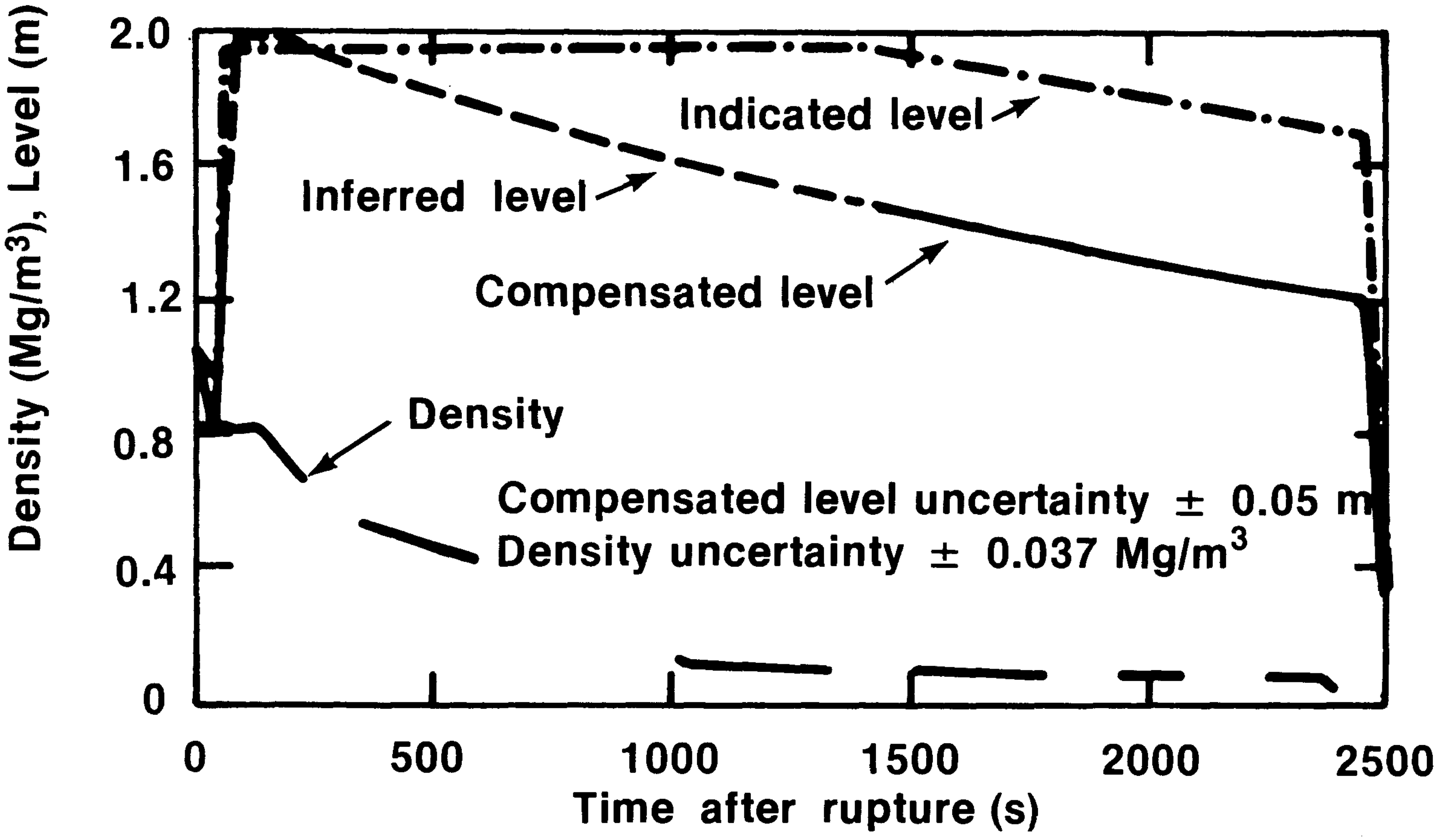
L3-0 Pressurizer Level Comparison and Intact Loop Hot Leg Fluid Density



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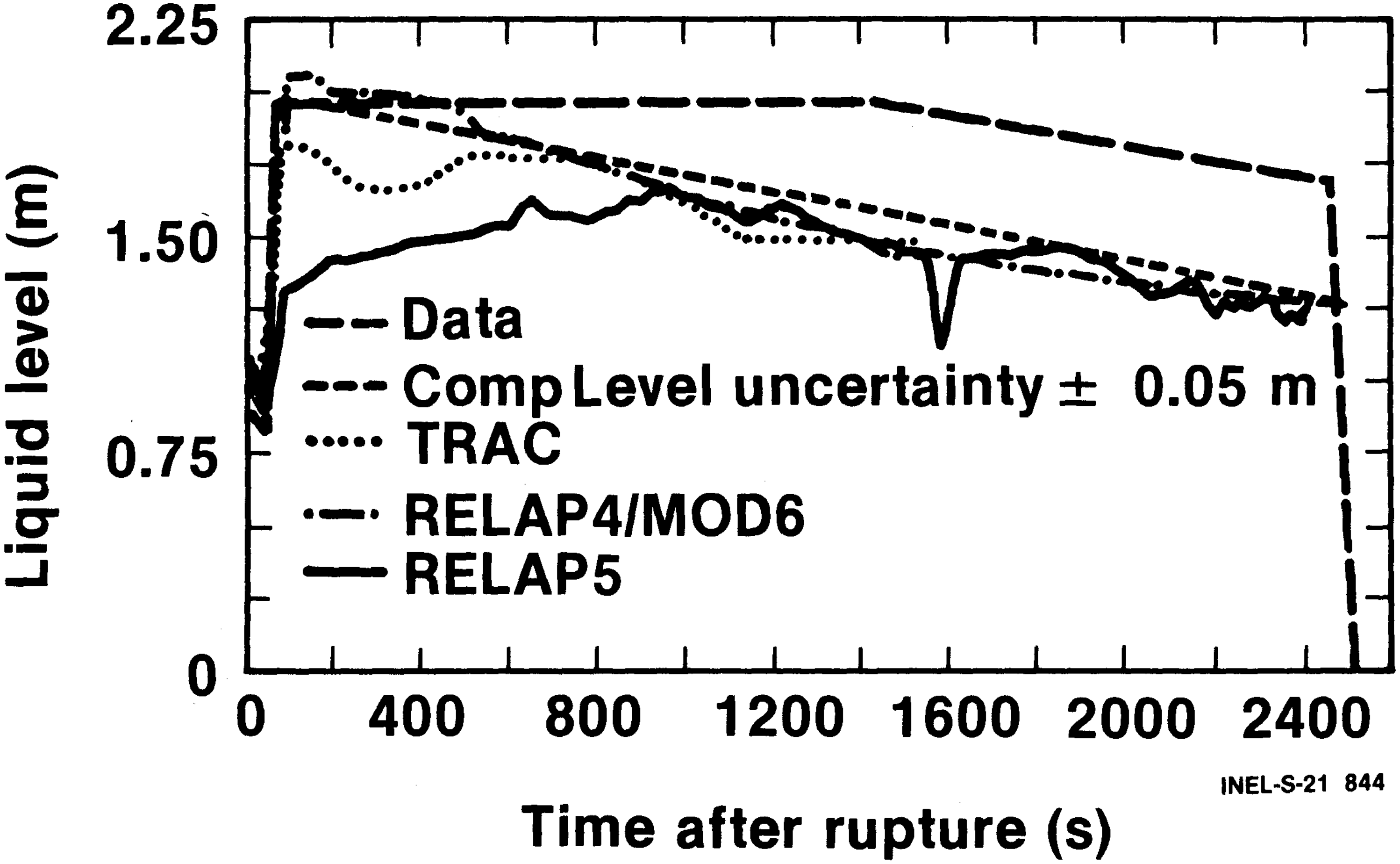
L3-0 Predicted and Measured Pressurizer Liquid Level



INEL-S-21 844

Conclusions (cont'd)

L3-0 test results:

- Pressurizer filled after vapor generated elsewhere in system
- Pressurizer fluid did not return to operational level
- Pressurizer indicated level uncompensated for fluid temperature -indicated high
- Calculations predicted data trends but experience needed to upgrade capability

Summary Conclusions

- Large break LOCEs self-limiting
- Self-limiting mechanisms understood
- Large break cladding temperatures are prototypic
- Large break calculations conservative — developmental areas known
- Data for small break code development
- Small break code development not as advanced as large breaks