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ABSTRACT

The Loss-of-Fluid Test (LOFT) Program is providing data to evaluate analytical models used to predict the thermal-hydraulic and fuel rod response of a pressurized water reactor (PWR) under loss-of-coolant accident (LOCA) conditions. The fuel rod response for the first nuclear loss-of-coolant experiment (LOCE), LOCE L2-2, is summarized.

INTRODUCTION

The LOFT reactor system[1], shown in Figure 1, has been scaled to simulate the important thermal-hydraulic characteristics of a commercial PWR during a LOCA. Pipe breaks are simulated by rapidly opening blowdown valves. Emergency core coolant (ECC) can be injected at one of several locations. The first nuclear test series (Power Ascension Series), summarized in Table I, consists of several tests in which guillotine breaks in a reactor vessel inlet pipe are simulated.

The Power Ascension Test Series will provide an understanding of the system response and the possibility of resolving major instrumentation problems before performance of the higher power tests, during which fuel rod damage may occur due to higher cladding temperatures.

The LOFT reactor core consists of 1300 fuel rods which are intended to be used for several tests. The fuel rods are of nominal PWR design except for length (1.67 m) and internal prepressurization (0.10 MPa). Approximately 200 cladding surface thermocouples measure the cladding

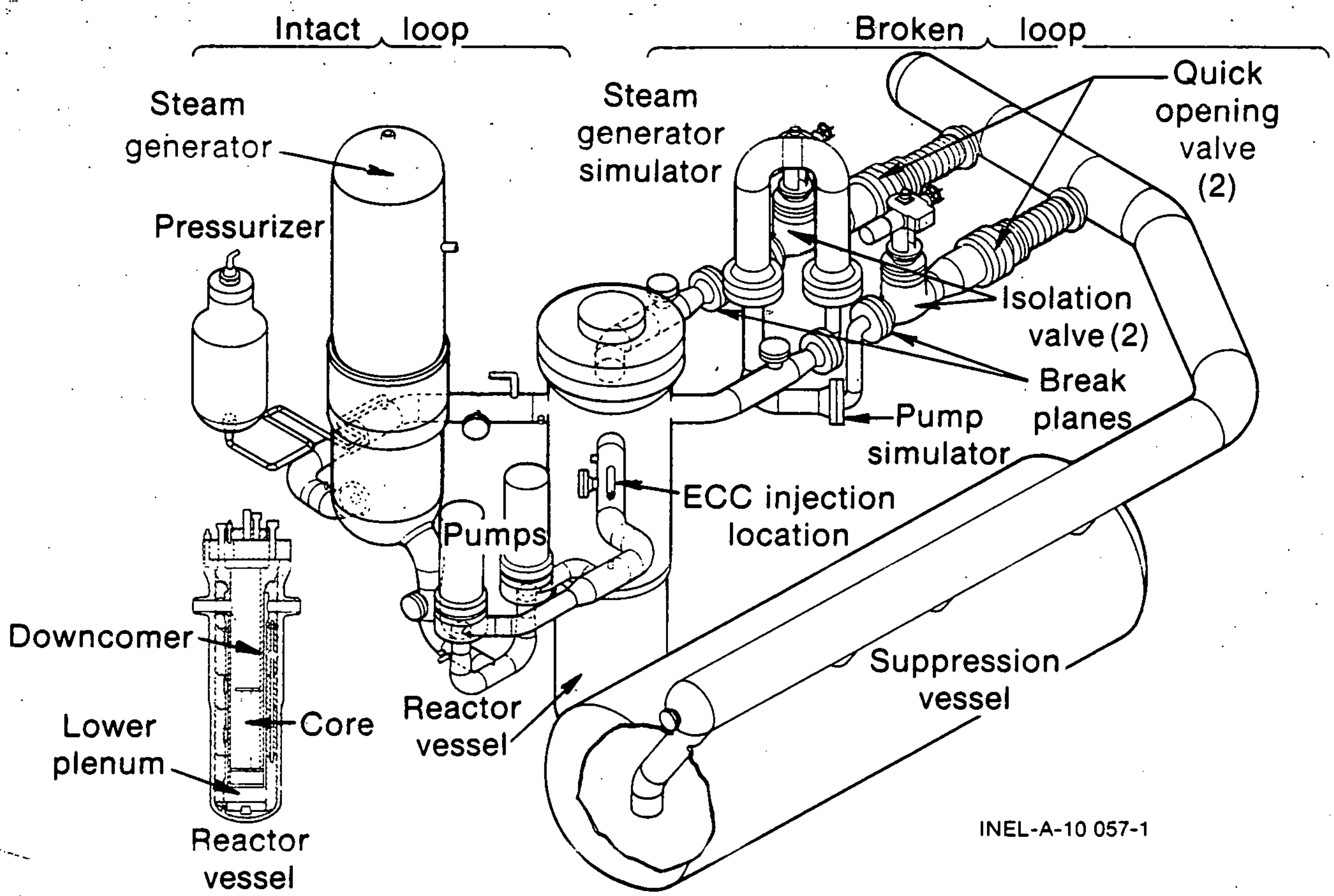

Fig. 1 Major components of LOFT system.

TABLE I

LOFT POWER ASCENSION TEST SERIES

LOCE[a]	Power Level (kW/m)	Central Module Fuel Rod Prepressurization (MPa)	Primary Coolant Pumps	ECC Delay[b]
L2-2	26.3	0.1	On	No
L2-3	39.4	0.1	On	No
L2-4	52.5	0.1	On	No
L2-5	39.4	0.1	Off (locked rotor)	Yes
L2-6	39.4	2.4	On	No

a. All 200%, offset shear break tests.

b. ECC injection into the intact cold leg during all tests.

temperature in six of the nine fuel modules, as shown in Figure 2. The cladding surface temperatures are the only direct measurement of fuel rod response.

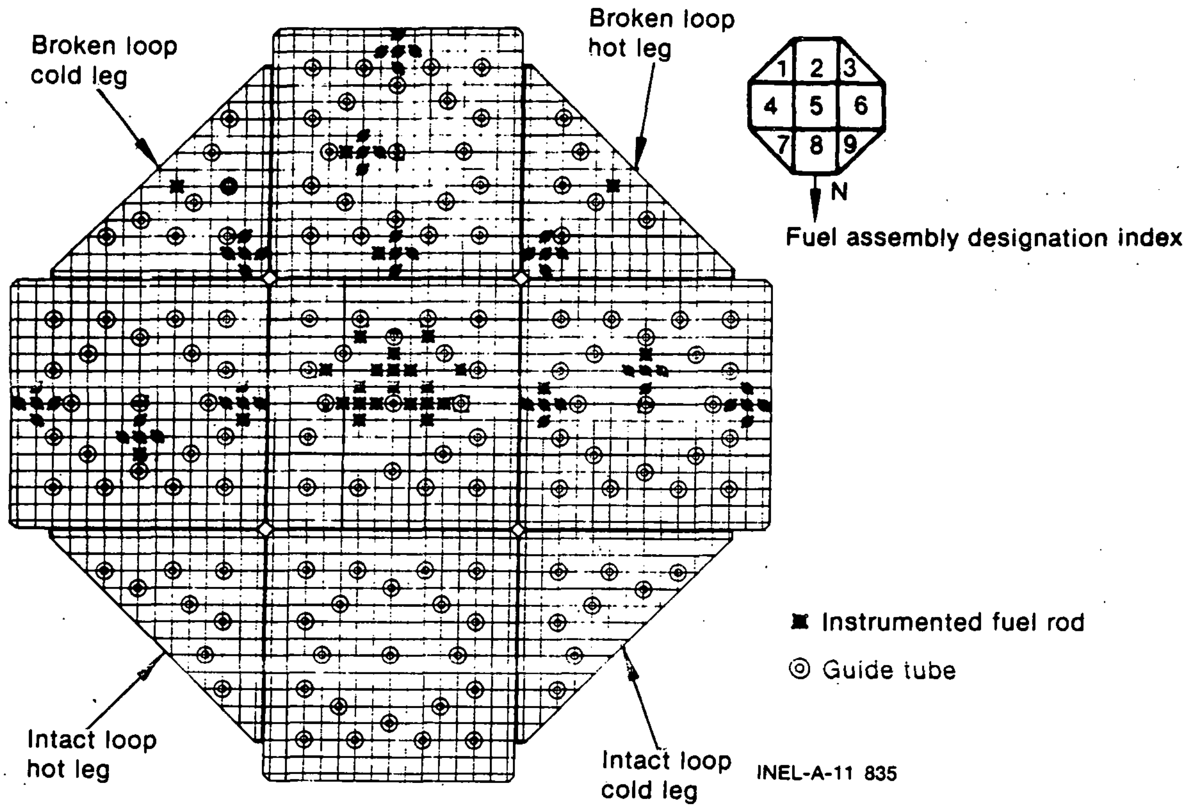

Fig. 2 LOFT core configuration and instrumentation.

FUEL ROD RESPONSE DURING LOCE L2-2

The first nuclear loss-of-coolant experiment, LOCE L2-2, was conducted in December 1978. Table II summarizes the steady state conditions prior to the LOCE transient.

The fuel rod cladding surface temperatures for LOCE L2-2 are illustrated in Figures 3 and 4. Figure 3 shows the axial distribution of measured fuel rod cladding surface temperatures obtained from a five-rod cluster in fuel Module 5. Figure 4 shows the radial distribution of measured fuel rod cladding surface temperatures at an elevation of 0.66 m above the core bottom in fuel Modules 4, 5, and 6.

TABLE II

STEADY STATE CONDITIONS FOR LOCE L2-2

Primary coolant mass flow rate	194.2 kg/s
System pressure	15.64 MPa
Cold leg temperature	558.0 K
Hot leg temperature	580.0 K
Core power	24.88 MW
Maximum linear heat generation rate	26.4 kW/m

Generally uniform fuel rod response was observed in the center fuel module. Cladding heatup [departure from nucleate boiling (DNB)] occurred between 0.55 and 2.0 seconds[a]. Those thermocouples which indicated this DNB are indicated in Figure 5. A rapid cladding temperature rise occurred until 2.5 seconds when it terminated. Between 6.5 to 8.5 seconds, all thermocouples indicated a rewet. The rewet was observed to progress from the bottom to the top of the fuel rods. Peak cladding temperatures during the first DNB approached 800 K.

The cladding remained quenched for 4 to 8 seconds. Secondary, relatively minor cladding heatup and rewets occurred from 15 to 35 seconds. At 30 seconds, a small heatup occurred throughout the center fuel module. Injection of ECC water quenched this heatup within 45 seconds.

As indicated in Figures 3 and 4, uniform fuel rod response did not occur during the test. In the center fuel modules, all thermocouple locations, which encompass linear power ratings from 2.1 to 26.9 kW/m, experienced the early DNB. Locations of higher power ratings, up to 15 kW/m, in the peripheral fuel modules did not exhibit the early heatup. Nonuniform core fluid conditions are thought to be responsible for the differences in fuel response. One explanation centers around the

a. All times are with respect to opening of the blowdown valves unless otherwise noted.

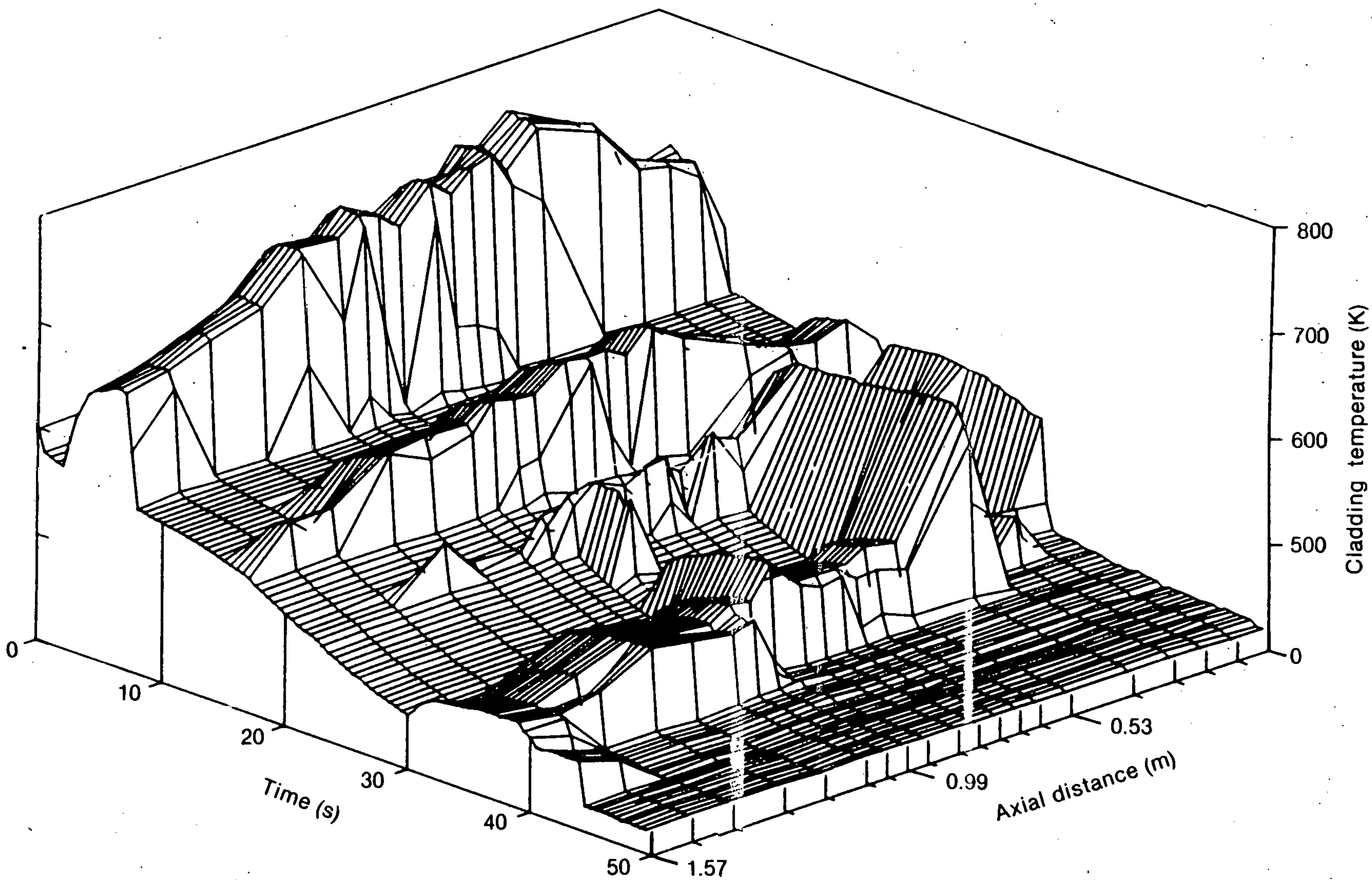

Fig. 3 Axially distributed cladding temperatures on fuel Module 5, five-rod cluster.

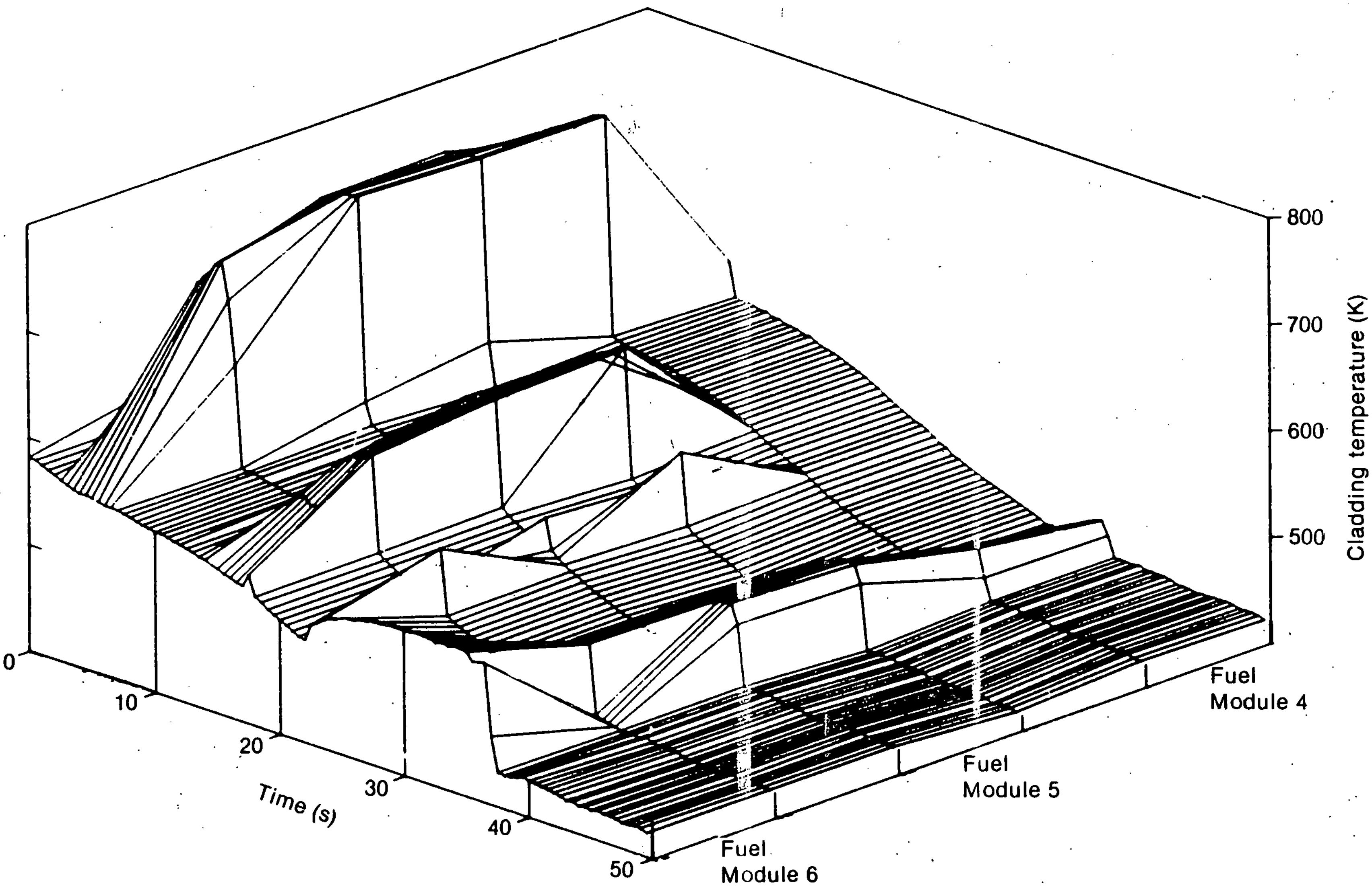

Fig. 4 Radially distributed cladding temperatures at 0.66 m in fuel Modules 4, 5, and 6.

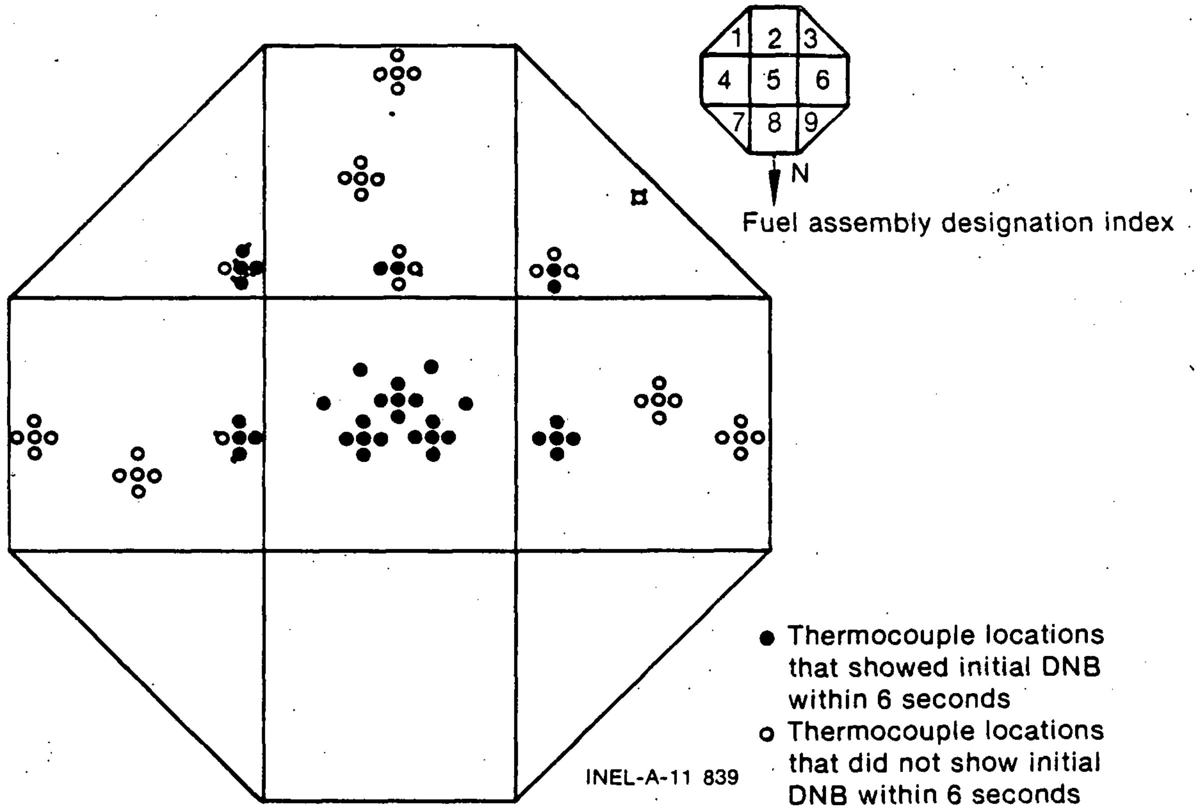

Fig. 5 Core thermocouples which indicated DNB within 6 seconds after experiment initiation.

possibility of flow diversion from the center fuel module and those high powered fuel rods immediately adjacent to the center fuel module into the outer region of the peripheral fuel module. The flow generated to the diversion was sufficient to preclude DNB at the outer fuel rods.

Comparison of fuel rod temperatures in fuel Modules 4 and 6 (see Figures 4 and 6) reveals nonuniform behavior. Flow diversion from the central region of the core is not the only cause of nonuniform fuel rod temperature response. The geometry of the core must be a source of non-uniform, three-dimensional fluid conditions.

Local nonuniform fluid behavior within the central fuel module was observed, as indicated in Figure 7 which shows the cladding temperature response of two fuel rods at 0.76 m from the core bottom. These fuel rods have identical power generation rates and are at the same geometrical location in the center fuel module. The difference in the temperature response after 9 seconds is attributed to differences in local thermal-hydraulic conditions.

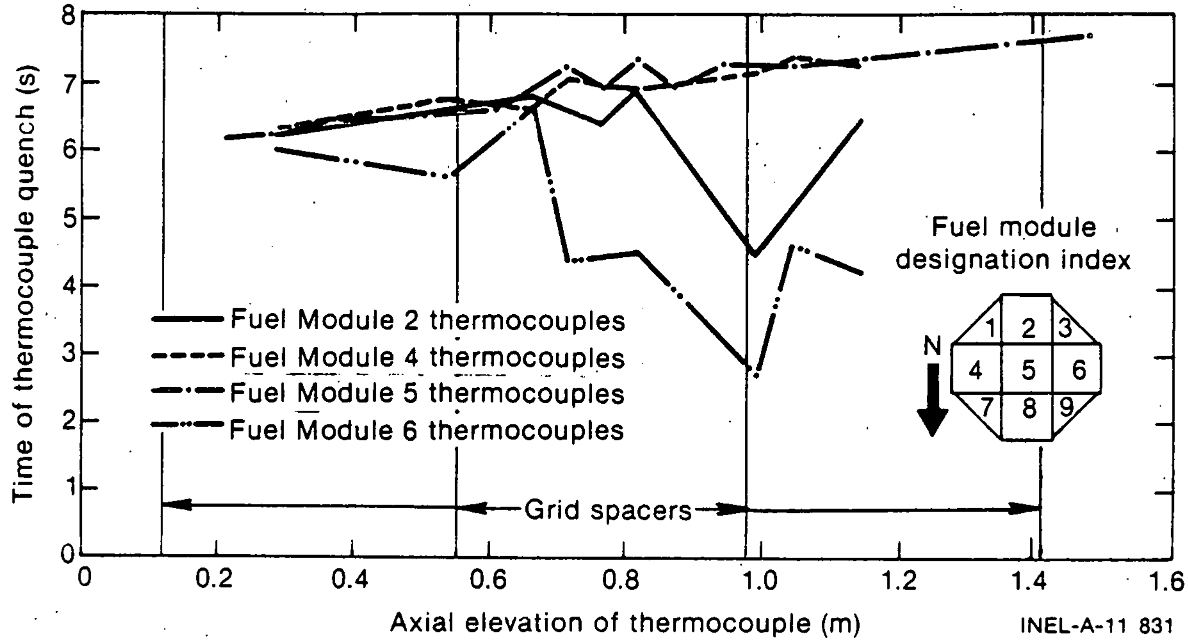

Fig. 6 Square fuel module quench times as a function of elevation for the first rewet during LOCE L2-2.

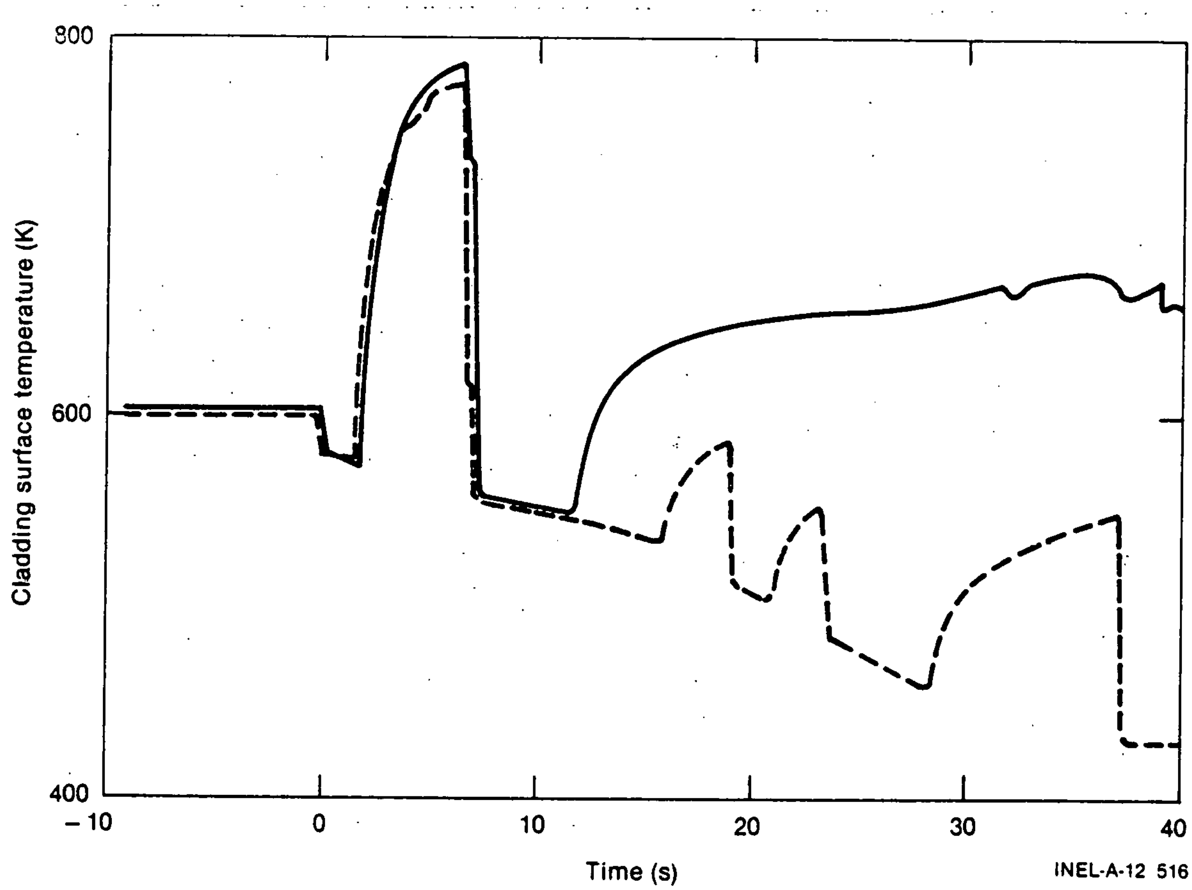

Fig. 7 Comparison of temperature response of two fuel rods with identical power generation and geometrical location in the core.

For times between 12 and 20 seconds, the thermocouple which indicated sustained secondary DNB shows a geometrical pattern toward the broken loop hot leg, as shown in Figure 8. This pattern is similar to the dryout behavior during the time between 25 and 35 seconds for the zero power LOCE L1-5, performed in a previous nonnuclear test series in LOFT.

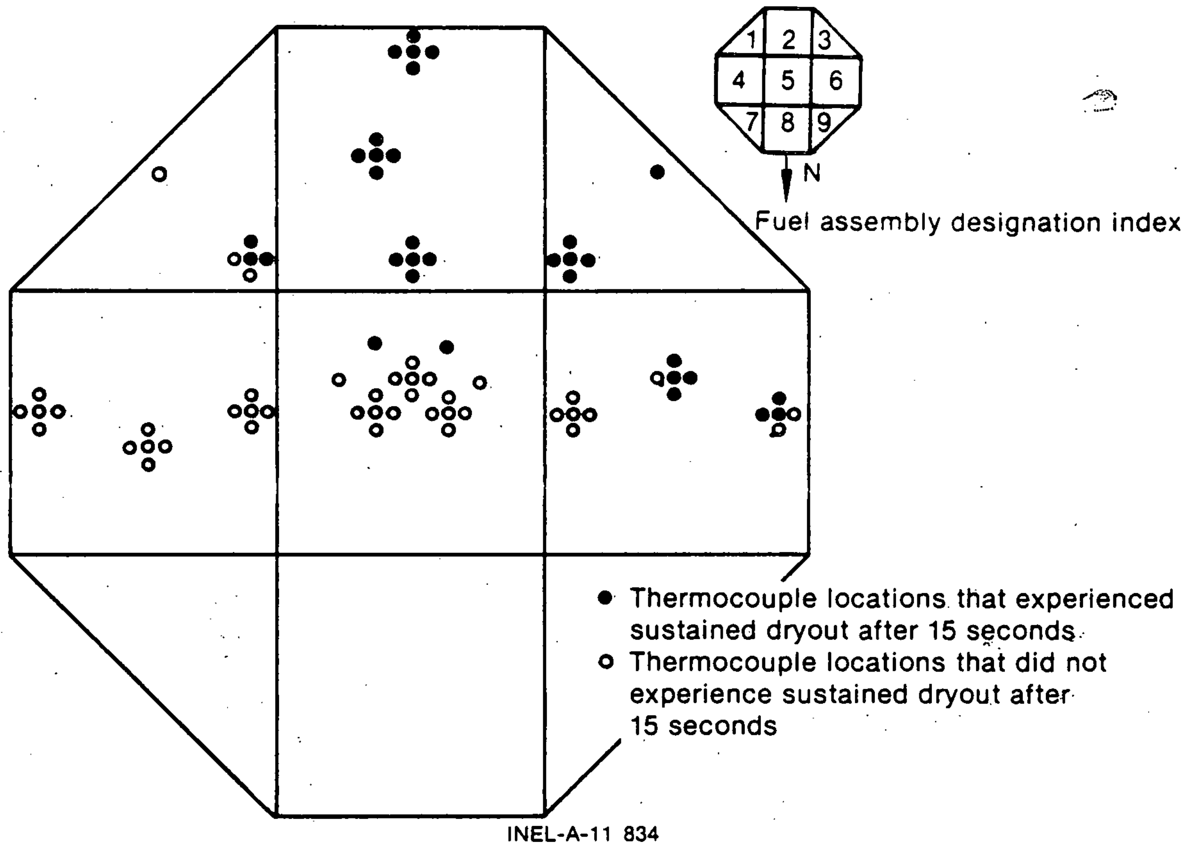

Fig. 8 Intrumented fuel rods which experienced sustained dryout after 15 seconds following experiment initiation.

The cladding surface temperature seems strongly coupled to core thermal-hydraulics. Figure 9 shows the momentum flux measurements in the reactor vessel upper plenum. A flow stagnation occurred very early in the test. Coincident with this stagnation was the occurrence of DNB between 0.5 and 7.5 seconds in fuel Module 5. Figure 10 shows the broken loop and intact loop cold leg mass flow rates.

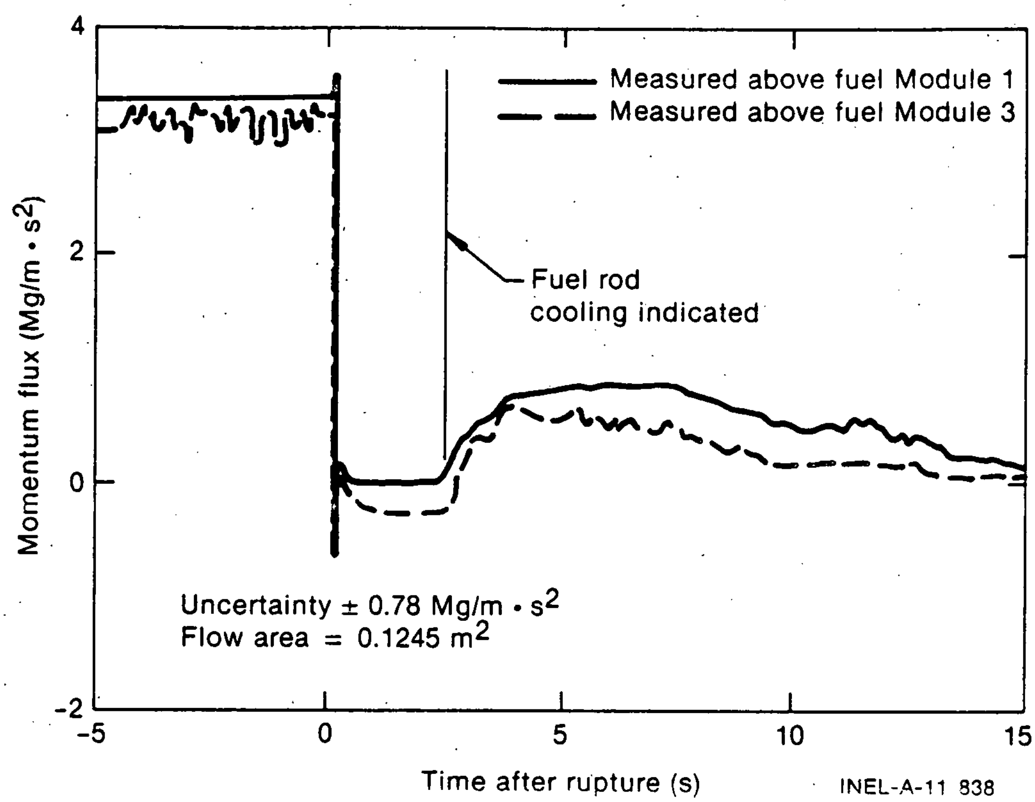

Fig. 9 Momentum flux in the reactor vessel upper plenum for LOCE L2-2.

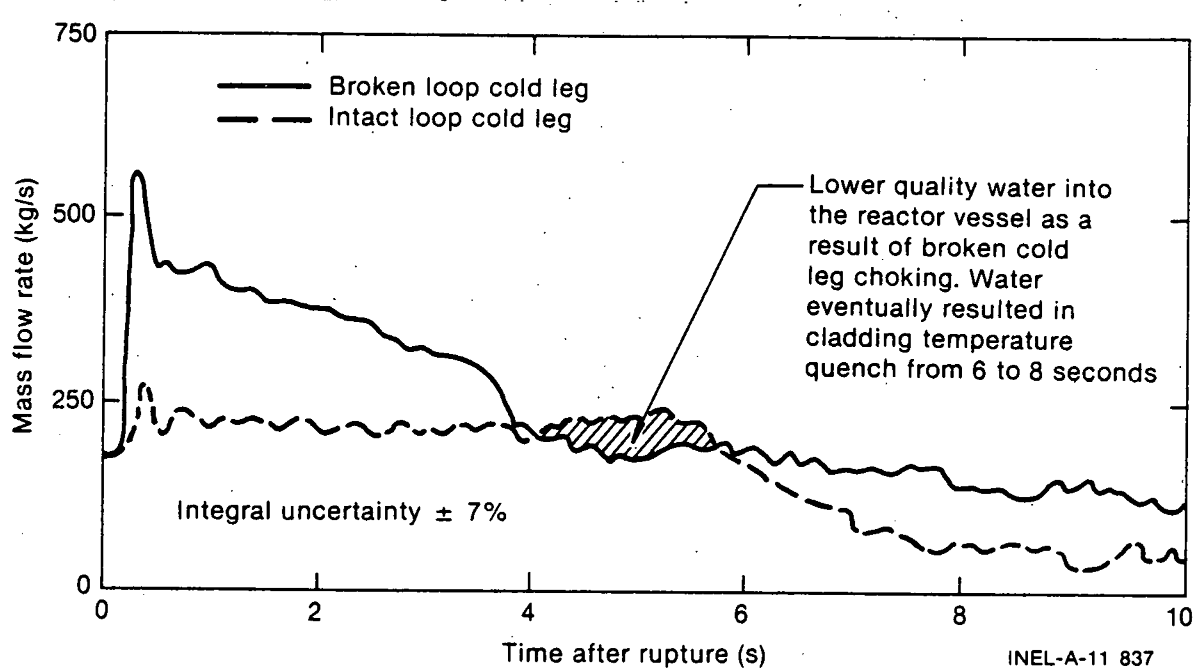

Fig. 10 Comparison of intact loop cold leg and broken loop cold leg mass flows during LOCE L2-2.

Due to the reduced critical flow out of the broken loop cold leg at 4 seconds, the intact loop cold leg supplied more fluid to the downcomer, and eventually to the reactor vessel, than could be removed from the broken loop. The net mass accumulation induced a positive core flow. The arrest of the cladding temperature rise and quench coincided with the influx of fluid from the intact loop cold leg.

Later in the test, momentum flux measurements showed uniformly stagnant flow in the core, which led to the cladding heatup at about 35 seconds. The arrival of ECC water into the core quenched this heatup.

The cladding surface heat transfer coefficients calculated by inverse heat conduction methods from the measured cladding temperatures indicated convective or nucleate boiling heat transfer during the rewet period from 6 to 12 seconds. FRAP-T4[a] calculations using these heat transfer coefficients indicated that the fuel rod stored energy would decrease by nearly 40% during the initial quench (rewet) period, and, by 15 seconds, nearly 80% of the initial stored energy has been transferred from the fuel as shown in Figure 11.

Large differences between calculated and measured cladding temperature are shown in Figure 12. These differences bring to issue the accuracy and potentially selective cooling effects of the LOFT surface thermocouples.

Out-of-pile tests to evaluate the accuracy of these thermocouples by comparison with measurements from small embedded thermocouples near the cladding surface have shown the accuracy of the surface thermocouples to be within ±20 K for blowdown and reflood LOCE conditions. Comparison of measured ECC core reflooding rates from LOCE L2-2 with a similar cold leg break, zero-power experiment (LOCE L1-5) shows no measurable difference. Figure 13 shows these results together with the RELAP4/MOD6[a] prediction of the core reflood rate, in which no cladding rewet was

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a. FRAP-T4, Version 9/14. Idaho National Engineering Laboratory Configuration Control Number H00389IB.

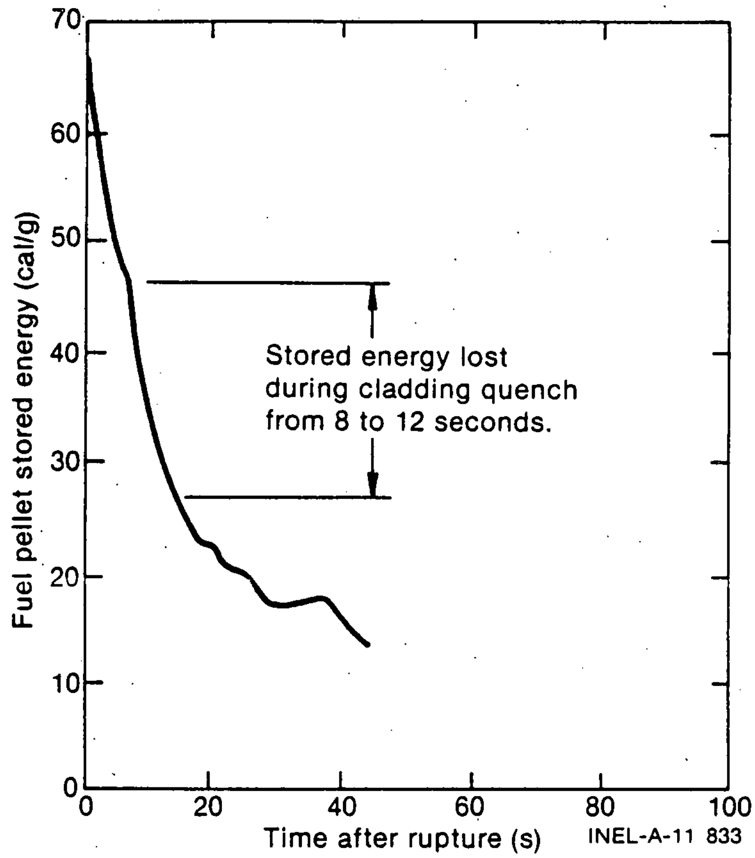

Fig. 11 Fuel pellet stored energy as a function of time after rupture.

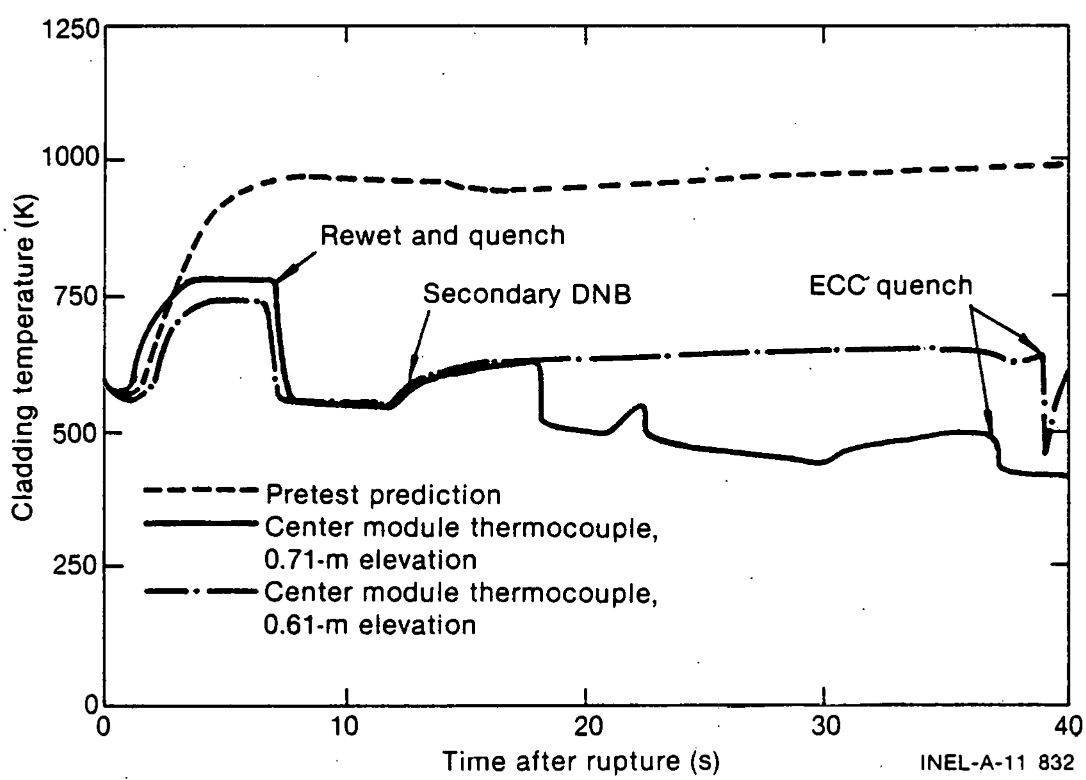

Fig. 12 Comparison of calculated and measured fuel surface temperature for LOCE L2-2.

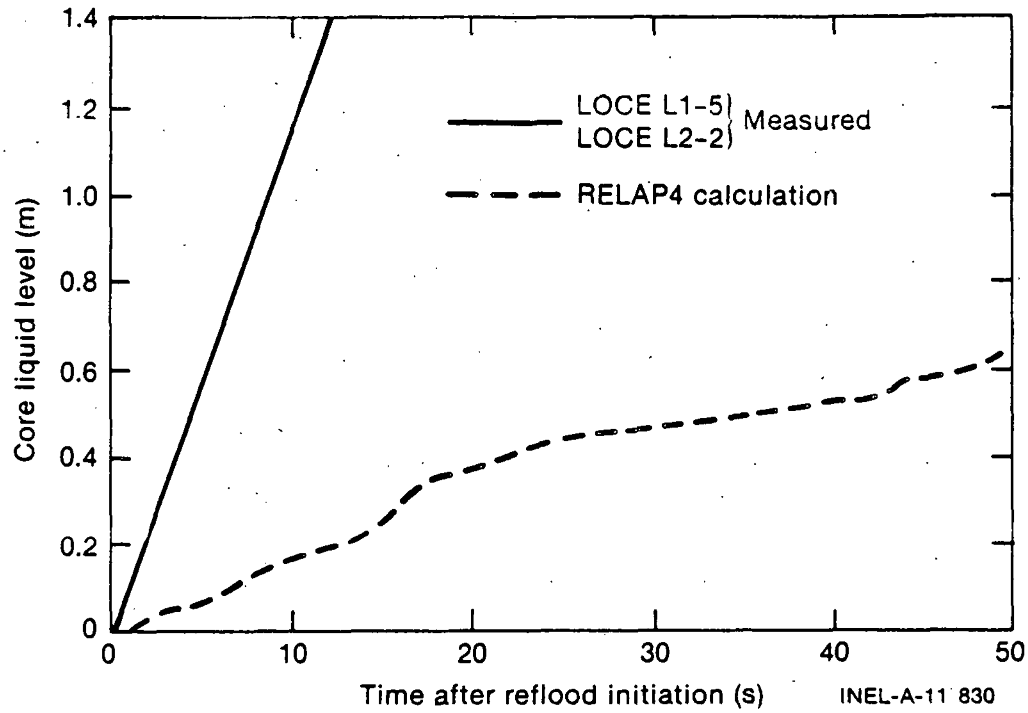

Fig. 13 Comparison of core reflood during LOCE L2-2 with LOCE L1-5 (isothermal LOCE) and pretest RELAP4 calculations.

predicted, and, therefore, the beginning of core reflood. The test results imply no significant energy was retained in the fuel at the initiation of ECC core reflood. The absence of stored energy could only occur if all the rods in the core rewet and quenched as indicated by the thermocouple data. Thus, the inference is that the uninstrumented rods behaved similarly to the rods with cladding thermocouples.

Radiation measurements in the primary system coolant before the transient and in the blowdown suppression tank after the transient indicated that no fuel rod failures occurred during the test. The measured cladding temperatures were not high enough to result in damaging cladding deformation (collapse) during the test. Figure 14 presents a comparison of the cladding temperature and mechanical loadings during the test with out-of-pile cladding collapse data[2]. The comparison indicates that

a. RELAP4/MOD6, Idaho National Engineering Laboratory Configuration Control Number H00608IB.

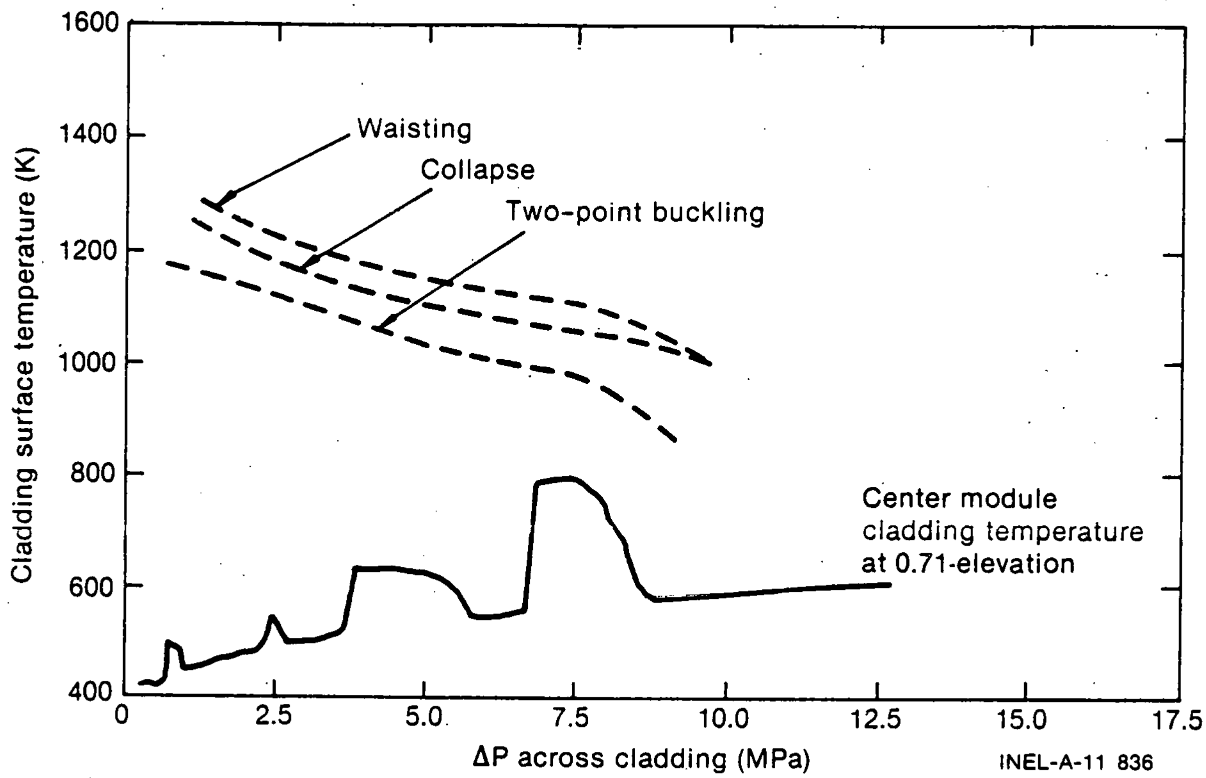

Fig. 14 Modes of cladding deformation as a function of temperature and pressure.

the cladding was within its elastic limits and that plastic cladding collapse, which may result in damaging fuel-cladding interaction during power preconditioning for subsequent tests, will not occur.

SUMMARY

The first LOFT nuclear experiment (LOCE L2-2) showed that cladding temperatures are a strong function of local core coolant flow and that rewetting of a nuclear fuel rod during the blowdown phase of a LOCE is possible and is an important cooling mechanism. Three-dimensional thermal-hydraulics within the core region were implied by the measured cladding temperatures during the test. The data suggest cladding surface thermocouples did not significantly affect rod behavior. On the basis of the measured cladding temperature and mechanical loadings, cladding deformation that would preclude using the rods for subsequent tests did not occur.

REFERENCES

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

2. C. S. Olsen, Zircaloy Cladding Collapse Under Off-Normal Temperature and Pressure Conditions, TREE-NUREG-1239 (April 1978).