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FORM EG&G 398
(Rev 11-79)

INTERIM REPORT

Accession No. ____________
Report No. EGG-LOFT-5127

Contract Program or Project Title:

LOFT Experimental Program

Subject of this Document:

A Summary and Assessment of Return to Nucleate Boiling Phenomena During Blowdown Tests Conducted at the INEL

Type of Document:

LOFT Technical Report

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Date of Document:

April 24, 1980

Responsible NRC Individual and NRC Office or Division:

G. D. McPherson, RSR

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

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

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C.
Under DOE Contract No. DE-AC07-76ID01570
NRC FIN No. A 6048

INTERIM REPORT

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Project No. P394

April 24, 1980

A SUMMARY AND ASSESSMENT OF RETURN TO NUCLEATE BOILING PHENOMENA DURING BLOWDOWN TESTS CONDUCTED AT THE INEL

System (00)

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U.S. Department of Energy

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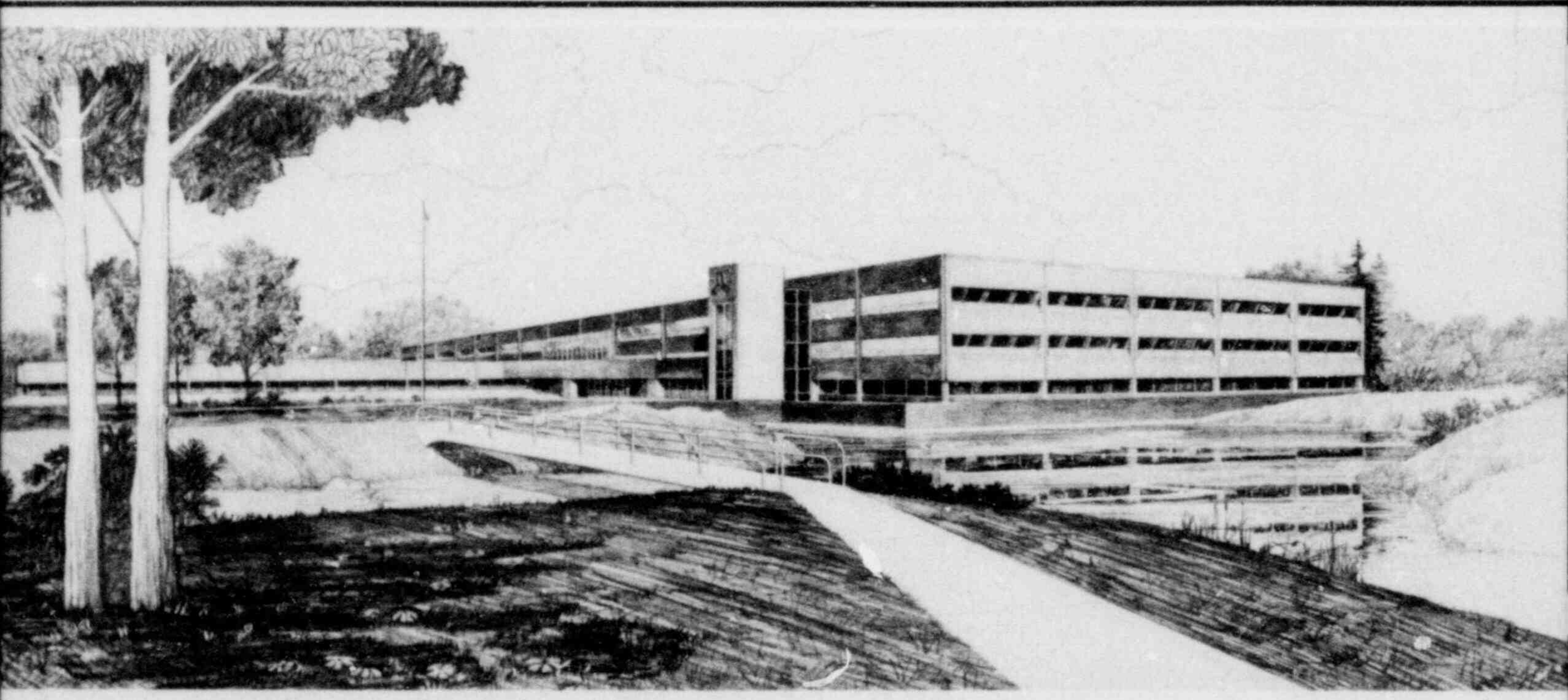

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Prepared for the
U. S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6048

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LOFT TECHNICAL REPORT
LOFT PROGRAM

FORM EG&G-229
(Rev. 12-78)

TITLE

A SUMMARY AND ASSESSMENT OF RETURN TO NUCLEATE BOILING PHENOMENA DURING BLOWDOWN TESTS CONDUCTED AT THE INEL

LTR 20-99

AUTHOR
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435HA001

DATE
RELEASED BY LOFT CDCS
April 24, 1980

LOFT APPROVAL

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This report summarizes the data obtained in Loss-of-Coolant Experiments (LOCEs) at Idaho National Engineering Laboratory (INEL) which demonstrate the presence of cladding rewetting after the critical heat flux has been exceeded as a viable cooling mechanism during the blowdown phase of a LOCE. A brief review of the mechanisms associated with the boiling crisis and rewetting is also provided. The relevance of INEL LOCE rewetting data to nuclear reactor licensing Evaluation Model Requirements is considered, and the conclusion is made that the elimination of rewetting and return to nucleate boiling (RNB) in Evaluation Models represents a definite conservatism. However, further experimental work must be done and analytical models developed which adequately characterize the transition boiling regime which encompases RNB and rewetting during blowdown type conditions.

DISPOSITION OF RECOMMENDATIONS

The results presented herein demonstrate that the current evaluation models are conservative. However, no credit can be taken at this time because more experimental results are required to definitize the phenomena taking place.

GD 4-25-80

ABSTRACT

This report summarizes the data obtained in Loss-of-Coolant Experiments (LOCE) at Idaho National Engineering Laboratory (INEL) which demonstrate the presence of cladding rewetting after the critical heat flux has been exceeded as a viable cooling mechanism during the blowdown phase of a LOCE. A brief review of the mechanisms associated with the boiling crisis and rewetting is also provided. The relevance of INEL LOCE rewetting data to nuclear reactor licensing Evaluation Model Requirements is considered, and the conclusion is made that the elimination of rewetting and return to nucleate boiling (RNB) in Evaluation Models represents a definite conservatism. However, further experimental work must be done and analytical models developed which adequately characterize the transition boiling regime which encompasses RNB and rewetting during blowdown type conditions.

SUMMARY

This document summarizes the data obtained in Loss-of-Coolant Experiments (LOCE) at Idaho National Engineering Laboratory (INEL) which demonstrate that cladding rewetting which may lead to a Return to Nucleate Boiling (RNB) after the critical heat flux is exceeded is a viable cooling mechanism during the depressurization (blowdown) phase of a Loss-of-Coolant Accident (LOCA). The demonstration of rewetting as a viable cooling mechanism during blowdown indicates that the elimination of any allowance for rewetting in the analysis of a LOCA is definitely a conservative approach.

The LOCA has been established as the design basis accident for Light Water Reactors (LWR). As such, the analysis of this accident is outlined in detail in the Code of Federal Regulations (CFR) defining LWR licensing requirements. Specifically, Appendix K to CFR, Title 10, part 50 outlines Evaluation Model Requirements (EMR) which comprise a detailed specification of the necessary features of analytical models that will be used to analyze a LOCA and establish acceptable operational safety limits. Of necessity, EMR contain sets of analytical assumptions which are imposed when experimental data is limited or lacking, and/or adequate analytical models are not available. Specific assumptions in the 10 CFR 50, Appendix K criteria at the present time do not allow the consideration of rewetting cooling mechanisms or RNB during a LOCA blowdown after the critical heat flux has been exceeded and film boiling is predicted because adequate data and analytical models are not available to characterize these phenomena, or, more generally, to characterize the transition boiling regime and minimum film boiling point which encompass such phenomena.

LOCE data demonstrating the existence of rewetting or RNB at INEL have been obtained in the Semiscale, LOFT and PBF LOCE programs[7,8,38]. Semiscale data have shown that rewetting phenomena are affected by local rod power densities and local coolant vapor flow (quality, or void fraction). Individual rod

characteristics played an important part in RNB situations when thermal-hydraulic conditions were on the borderline between rewetting, and not rewetting.

In the LOFT L2-2 Test, rewetting occurred throughout the core during the blowdown after the core had experienced a boiling crisis. The rewetting in L2-2 has been correlated wih hydraulic phenomena associated with the coolant flow dynamics in the intact and broken loops during blowdown. LOFT Test L2-2 and PBF LOC-11A (which experienced rod rewetting due to valve cycling after the boiling crisis) demonstrated that rewetting is a viable cooling mechanism during a LOCE blowdown that is dependent on thermal-hydraulic conditions.

While INEL LOCE data have demonstrated that the elimination of any rewetting considerations in evaluation models after film boiling is predicted is conservative in nature, further research in this area is still required. Adequate analytical models which can characterize the minimum film boiling point and the transition boiling regime, which encompass rewetting and describe the path leading from film to nucleate boiling, must still be developed from a reliable data base. Such models are necessary to predict rewetting or RNB on the basis of calculated local therma' hydraulic conditions.

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I. INTRODUCTION

Safety analyses and reactor licensing in the nuclear power industry rely heavily on analytical predictions to estimate the behavior and establish the criteria for the safe operation of nuclear power plants. The predictions are generally based on available experimental evidence coupled with sets of generally conservative assumptions that are imposed when direct experimental data is limited or lacking, or accurate analytical models are not available.

In the qualification of Light Water Reactors (LWR), the Loss-of-Coolant Accident (LOCA) has been established as the design basis accident[1]. As a result, the understanding of this accident has been the focus of numerous analytical and experimental programs. For licensing purposes, the manner in which power reactor safety calculations are done is regulated in detail, as outlined by the Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for LWR's[2]. These criteria define so-called Evaluation Model Requirements[3], which comprise a detailed specification of the necessary features of the analytical models that will be used to analyze the LOCA and establish acceptable operational safety limits to the public.

The Evaluation Model Requirements specify a number of assumptions that must be made in the analysis of a LOCA. These assumptions can be very restrictive since many design variables are set by LOCA related considerations. Significant research is now underway all over the world to obtain a better understanding of a number of phenomena which would lead to a relaxation of some overly conservative assumptions[4].

One assumption of significant importance relates to rewetting of the hot cladding surface or a return to nucleate boiling (RNB) after the critical heat flux (CHF) is exceeded during the initial depressurization (blowdown) phase of a LOCA. The assumption is made that after CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations again at that axial location during the analysis of the blowdown, even if calculated local fluid and fuel rod surface conditions would apparently justify RNB. Heat transfer assumptions characteristic of rewetting are permitted when justified during the reflood portion of a LOCA.

The problem of characterizing rewetting and RNB is essentially one of adequately describing (in a quantitative manner) the transition boiling regime, which represents the bridge between nucleate boiling and film boiling. The transition boiling regime is bounded by the critical heat flux on one side and the minimum film boiling heat flux on the other side. At the present time, as concluded by Groeneveld and Fung[18] in their review of transition boiling, no generally reliable transition boiling correlation exists, and existing correlations must be judiciously applied only for conditions from which they were derived. The description of transition boiling is further complicated by the possibility of hysteresis, i.e., the path from nucleate boiling to film boiling for a given set of conditions may not be the same as the reverse path from film to transition boiling.

The omission of rewetting during blowdown can result in a distinctly "conservative" calculation because much lower overall heat transfer occurs and subsequently higher peak cladding temperatures are predicted. Since this restriction may unnecessarily restrict the operating limits of a nuclear power plant, some vendors have proposed that a "switching criterion" be used with DNB calculations to permit RNB, but this approach has been rejected on the basis that rate effects in rewetting heat transfer are not sufficiently well understood at this time[5].

The presence of rewetting as a viable cooling mechanism and RNB during blowdown has been demonstrated in several Loss-of-Coolant Experiment (LOCE) test programs[6,7,8,9]. LOCE tests with electrical heated rods have also shown that rods which experience rewetting during blowdown have significantly lower peak cladding temperatures (100-200 K) than rods which do not experience rewetting.

This report summarizes the data obtained at the Idaho National Engineering Laboratory which provide evidence for RNB during loss-of-coolant experiment (LOCE) blowdown tests. The discussion of rewet data at INEL will serve as a basis for the discussion of further testing and analysis required to identify rewet mechanisms and develop analytical models which could be used in alleviating Evaluation Model Requirements conservatisms with respect to the analysis of post CHF blowdown heat transfer in a LOCA.

The second section of the report briefly reviews the analysis of LOCA blowdown heat transfer. A brief summary will be made of a LOCA blowdown scenario, together with the Evaluation Model Requirements applicable to the analysis of the blowdown phase of a LOCA. This section also discusses the physics of the boiling crisis and the rewetting of a hot surface.

The third section of the report discusses RNB phenomena evident in tests at INEL including Semiscale, LOFT L2-2, and PBF LOC-11 data. Data trends and possible RNB mechanisms will be discussed.

In the final section of the report, conclusions are discussed that were reached as a result of the analysis of RNB data. Suggestions for further testing and analysis in the area of rewetting phenomena are also included in this section.

II. LOCA BLOWDOWN HEAT TRANSFER AND THE PHYSICS OF RNB

To place the discussion of RNB and rewetting phenomena at INEL in proper perspective, this section will briefly review the Evaluation Model Requirements pertinent to the analysis of a LOCA blowdown. This will be followed by a discussion of the qualitative mechanisms associated with the physics of rewetting of a hot surface that may be applicable during blowdown.

1. LOCA BLOWDOWN EVALUATION MODEL REQUIREMENTS

The LOCA has been established as the major design basis accident for LWR's. By definition, a LOCA is a postulated accident where the loss of reactor coolant exceeds reactor coolant system's makeup capability[1]. In a PWR system, the most severe LOCA in terms of highest peak cladding surface temperature is considered to be the double-ended guillotine break of one of the primary coolant cold legs.

The sequence of events in the analysis of a LOCA is generally divided into three distinct phases: blowdown, refill, and reflood. Although all are important, only the blowdown will be discussed here, since the rewetting phenomenon to be addressed in this report is related to this portion of the LOCA. During blowdown, the initial depressurization of the reactor occurs and the loss of core flow and increase in coolant quality generally results in a boiling crisis, i.e., the critical heat flux for given thermal hydraulic conditions is exceeded. Once the boiling crisis is reached, the heat transfer from the rod decreases markedly and the cladding temperature rises rapidly. Although by this point in time the reactor has been shutdown by control rod scram and significant void reactivity feedback, several percent of the reactor power (decay heat) and stored energy in the fuel contribute to the rapid temperature rise.

The time to reach the boiling crisis, and any rewetting or return to nucleate boiling (RNB) of the fuel rods after that time are of paramount importance, since these factors significantly effect the fuel rod stored energy and subsequent fuel rod temperatures as the coolant inventory is depleted and dryout of the core occurs.

Appendix K to 10 CFR 50[3] specifies the Evaluation Model Requirements for the analysis of blowdowns. These requirements define the characteristics of models and correlations that are acceptable and define the conservatisms applicable in the analysis.

The criteria of particular importance in the discussion of rewetting and RNB are those defined in Sections I.C.4 and I.C.5 of Appendix K. These sections define the correlations appropriate for the prediction of CHF and for post CHF heat transfer. The conservatisms in these sections that limit a consideration of RNB are defined such that "after CHF is predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown, even if the calculated local fluid and surface conditions would apparently justify the re-establishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA."

In addition, limitations are placed on the use of transition boiling correlations, these being that transition boiling will not be used "during the blowdown after the temperature differences between the clad and the saturated fluid first exceeds 300^{o}F," and "transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300^{o}F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions."

These conservative limitations have been established because of uncertainties that exist in the time factors influencing the rewetting of a hot surface during blowdown[5].

2. BOILING CRISIS AND THE REWETTING OF A HOT SURFACE

The purpose of this section is to examine some of the qualitative mechanisms which can lead to the boiling crisis and provide for the rewetting of a hot surface afte· the critical heat flux is exceeded. Information discussed here will serve as a basis for postulating possible mechanisms associated with the boiling crisis and rewetting experienced in INEL LOCE testing.

2.1 The Boiling Crisis

As discussed by Hsu [10], the boiling crisis occurs when the heating surface dramatically rises in temperature because of a sharp reduction in ability to transfer heat from the surface. This phenomenon has also been referred to as burnout, and the heat flux associated with the boiling crisis has been called the critical heat flux or burnout heat flux.

While a boiling crisis can occur in both pool boiling and boiling two-phase flow, the only mechanism associated with the boiling crisis in pool boiling is the nucleate boiling transition called departure from nucleate boiling (DNB). A number of different mechanisms can be associated with the boiling crisis in two-phase flow however. Specifically, in the two-phase flow the following mechanisms may lead to a boiling crisis:

(1) Development of a dry patch under coalescing bubbles (similar to the DNB condition in pool boiling)

(2) Liquid film dryout at the end of annular flow

(3) Dryout of the thin film surrounding a cylindrical bubble in slug flow

(4) Bubble nucleation in annular flow.

Smith and Griffith[11] in their discussion of CHF mechanisms concluded that for a given pressure, geometry and mass velocity there are two fundamental processes which interact to produce the boiling crisis mechanisms listed above. The first is a heat flux controlled process which dominates at low qualities. In this process, adequate liquid may exist for cooling but at sufficiently high heat fluxes and surface temperatures the vapor generation at the surface may be at a rate that retards surface wetting by the liquid. The vapor may be produced in preferred nucleation sites on the heated surface, or may, at sufficiently high temperatures, result from density fluctuations within the liquid, which is referred to as heterogeneous nucleation. Under most situations encountered with water, heterogeneous nucleation may be discounted as a mechanism for significant vapor generation because of the high liquid superheats required. However, Henry[12] has postulated that the rapid depressurization in a LOCA, coupled with preferred nucleation site deactivation during steady state operation of a reactor prior to a LOCA, can result in heterogeneous nucleation at a rate that rapidly creates a boiling crisis. A heat flux dominated process with the associated hydrodynamic instabilities would lead to the boiling crisis mechanism #1 listed previously.

The second fundamental process affecting CHF mechanisms is controlled by the vapor flow rate. At sufficiently high vapor flow rates, an annular flow regime exists with a vapor core and liquid film on the solid surface. The interaction of the liquid film with the vapor core can be described by the equation given by Hsu[10] as

$$\frac{dW_{fL}}{DdL} = \frac{-q}{H_{fg}} + E_D - E_N - E_S$$

where W_{fL} is the mass flow rate of the liquid film, $-q/H_{fg}$ the evaporative depletion rate, E_D the droplet deposition, and $-E_N$ and $-E_S$ the entrainment rates due to nucleate boiling spray and to interfacial shear, respectively. Under appropriate conditions, the liquid lost to the vapor through evaporation and entrainment exceeds the deposition rate to the liquid and a "dryout" of the liquid film occurs, thereby precipitating a boiling crisis (mechanism #2 listed previously). The interaction of this process with the heat flux controlled hydrodynamic instability process discussed previously results in the boiling crisis mechanisms such as #4 listed previously, with bubble generation creating a boiling crisis in the liquid annular flow film.

Extensive reviews of boiling crisis correlations and models for two-phase flow are given by Hsu[10], Tong[13], and Collier[14]. Smith and Griffith[11] have concluded that boiling crisis correlations derived from steady state experiments, when properly related to transient local conditions, can adequately predict boiling crisis phenomena during flow reversal transients.

2.2 Rewetting of a Hot Dry Surface

Once the boiling crisis has occurred and the critical heat flux has been exceeded, the surface temperatures rise rapidly, since the change in heat transfer mechanism is such that the superheat required in nucleate boiling heat transfer may need to become significantly higher in film and transition boiling to transfer the same heat flux. A crucial issue in the analysis of a LOCA is the determination of the criteria required for a hot dry surface with hundreds of degrees superheat to rewet and return to nucleate boiling (RNB).

The rewetting of a hot surface governed by a complex interaction between surface heat flux, surface temperature, and coolant state conditions. A description of rewetting requires adequately characterizing the transition boiling regime, a "twilight" zone between the film and nucleate boiling regime. The problem of this characterization is further compounded since it has not been clearly demonstrated that the path leading from nucleate to film boiling (via the boiling crisis), is the same as the reverse path from film to nucleate boiling, i.e., the possibility for hysterisis in transition boiling cannot be ruled out.

The transition boiling regime is bounded by the critical heat flux and the minimum film boiling heat flux, with the minimum film boiling temperature defined as the surface temperature at the minimum film boiling heat flux. Photographic studies have demonstrated that intermittent solid-liquid contact exists during transition boiling[15], which has led to the description by Berenson[15] of the transition boiling heat transfer mechanism as "a combination of unstable nucleate boiling and unstable film boiling existing at any given location on a heating surface. The variation in heat transfer rate with temperature is primarily a result of change in the fraction of time each boiling regime exists at a given location." Most of the heat transferred during transition boiling result from liquid-solid contact, and such factors as droplet size, droplet impact velocity, impact angle, and surface roughness can influence the magnitude of heat transferred[17]. The liquid not only transfers heat, but contributes to the forced convection heat transfer to the vapor by agitating the vapor boundary layer[17].

Groeneveld and Fung[18], with an update by Fung[19], have provided a summary of transition boiling experiments and predictive methods currently available in the literature. The basic conclusion of their review is that no generally reliable transition boiling correlation presently exists, and existing correlations must be judiciously applied only for conditions from which they were derived.

Rewetting of a hot surface is assumed to occur at the minimum film boiling temperature, although there is no agreement in the literature whether liquid-solid contact exists near the minimum film boiling point[17]. As discussed by Groeneveld and Fung[18], some investigators believed that the vapor film, which may be in violent motion, may persist below the minimum film boiling superheat, while other investigators believed that liquid-solid contact will occur at the minimum film boiling superheat.

Rewetting of a hot surface can commence once the vapor forces precluding liquid-solid contact have been overcome. The following two basic mechanisms have been postulated for solid-liquid separation by a vapor film[18]:

(1) Thermodynamically controlled separation - This mechanism assumes that liquid is instantaneously vaporized when in contact with the solid because the maximum liquid superheat of the liquid has been exceeded. The maximum liquid superheat is a thermodynamic property of a fluid that is a function of pressure and fluid material properties.

Gunnerson and Cronenberg[20] suggest a method for calculating the thermodynamically controlled minimum film boiling temperature (and incipient rewetting) by the use of a contact temperature. The contact temperature is calculated by assuming that at the instant of contact both solid and liquid act as semi-infinite solids, which results in the following expression

$$T_c = \frac{Tw\,(k/\sqrt{\alpha})w + T_L\,(k/\sqrt{\alpha})_L}{(k/\sqrt{\alpha})w + (k/\sqrt{\alpha})_L}$$

where: T_c = contact temperature

Tw = temperature of the wall just prior to solid-liquid contact

T_L = temperature of the liquid just prior to solid-liquid contact

k = thermal conductivity

α = thermal diffusivity.

Rewetting is therefore possible whenever

$$T_{SAT} < T_C < T_{MAX,s}$$

where: T_{SAT} = liquid saturation temperature

$T_{MAX,s}$ = maximum allowable liquid superheat.

The maximum allowable liquid superheat, $T_{MAX,s}$, can be calculated by using different approaches[21] involving the stability of the equilibrium liquid state, equilibrium between the liquid state and suspended vapor nuclei, and the kinetic theory of vapor nucleus formation. Gunnerson and Cronenberg[20] give a relatively simple equation for $T_{MAX,s}$ based on work by Leinhold for non-metals as

$$\frac{T_{MAX,s} - T_{SAT}}{T_{crit.}} = \left[1 - \frac{T_{SAT}}{T_{crit.}}\right] - 0.0905 \left[1 - \frac{T_{SAT}}{T_{crit.}}\right]^8$$

(2) Hydrodynamically controlled separation - This mechanism assumes that the forces developed by vapor formation at the solid surface are greater than the forces directing the liquid towards the heated surface. Berenson's[22] correlation for minimum film boiling temperature and heat flux is based on this approach, with modifications by Henry[23] to account for solid surface and liquid material properties.

As discussed by Groeneveld and Fung[18], during fast transients where insufficient time is available to fully develop hydrodynamic forces, rewetting will be predominantly thermodynamically controlled, while in low pressure, low flow situations when sufficient time is available rewetting will be primarily hydrodynamically controlled. Henry[24] has also shown that for low pressures, hydrodynamic conditions dominate, but at higher pressures (> 200 kPa), the thermodynamically controlled mechanism is most important.

While no adequate transition boiling correlation presently exists, the identification of minimum film boiling point provides at least an initial approximation for incipient rewetting of a hot surface. Once rewetting has occurred locally, assuming sufficient liquid inventory is available, a rewetting front can propagate at a rate that is not only related to fluid conditions, but which can be strongly controlled by axial conduction[25]. Axial conduction becomes significant as a result of severe temperature gradients that exist at the solid surface between areas that have rewet and areas that have not rewet.

III. RNB PHENOMENA AT INEL

The data base which serves as a source for evidence of RNB in tests at INEL at the present time is comprised of loss-of-coolant experiments (LOCE) performed in the Semiscale[26] (electrically heated rods), Loss-of-Flow Test (LOFT) reactor[27] (nuclear core) and Power Burst Facility (PBF)[28] (individual fuel rod behavior experiments) testing programs. Because of its much larger data base, the Semiscale data will be discussed first to establish qualitative data trends, followed by a discussion of the LOFT and PBF LOCE's.

1. SEMISCALE LOCE DATA

Semiscale is a volume scaled model of the LOFT nuclear reactor. It consists of an active coolant loop, a cold leg and hot leg break path, and a core of 40 electrically heated rods with a 1.52 m active heated length. Each of the core heater rods are instrumented with thermocouples at various axial and azimuthal locations sandwiched between cladding annuli. Several series of blowdowns have been conducted with this loop to evaluate blowdown heat transfer and ECC injection schemes as well as LOFT test series counterpart experiments.

The occurrence of RNB during blowdown in Semiscale tests has been of major interest, and the subject has been discussed and analyzed in various reports discussing test results[29,30]. During Semiscale tests it was found that some rods rewet and others, although nominally identical, did not during blowdown. There is no Semiscale Test where all the rods rewet during the blowdown. In the following sections, a brief description of a representative Semiscale blowdown will be presented, followed by a discussion of RNB data trends evident in all Semiscale blowdowns, and finally possible RNB mechanisms will be discussed.

1.1 Description of a Semiscale Blowdown

The Semiscale Mod-1 blowdown heat transfer tests serve as good examples which demonstrate the general thermal-hydraulic characteristics of the Semiscale facility during blowdown, with evidence of RNB phenomena. These tests have also had extensive posttest analysis to examine thermal-hydraulic characteristics.

For the purposes of discussion, Test S-02-9[31] was chosen as an example of a blowdown test with RNB behavior. This test was used by Snider as a basis for analysis to shed a quantitative light on the blowdown thermal-hydraulic phenomena[32].

Test S-02-9 was intended to simulate a 200% cold leg break LOCA (without reflood). This test had a flat core radial power profile with 37 powered heater rods. Figure 1 illustrates that core geometry of this test and the location of the thermocouples. The initial conditions for this test are shown in Table I. Figure 2 illustrates the axial power profile for the Semiscale heater rod and Figure 3 illustrates the construction of the heater.

TABLE I

TEST S-02-9 INITIAL CONDITIONS

Break	200% cold leg break. (a 200% break is a full size double-ended offset shear)	
Core Power	1.56 MW	
Core inlet temperature	557 K	(542°F)
Core temperature rise	38 K	(68°F)
System pressure	15.53 MPa	(2253 psia)
Peak power density	38.9 kW/m	(11.84 kW/ft)

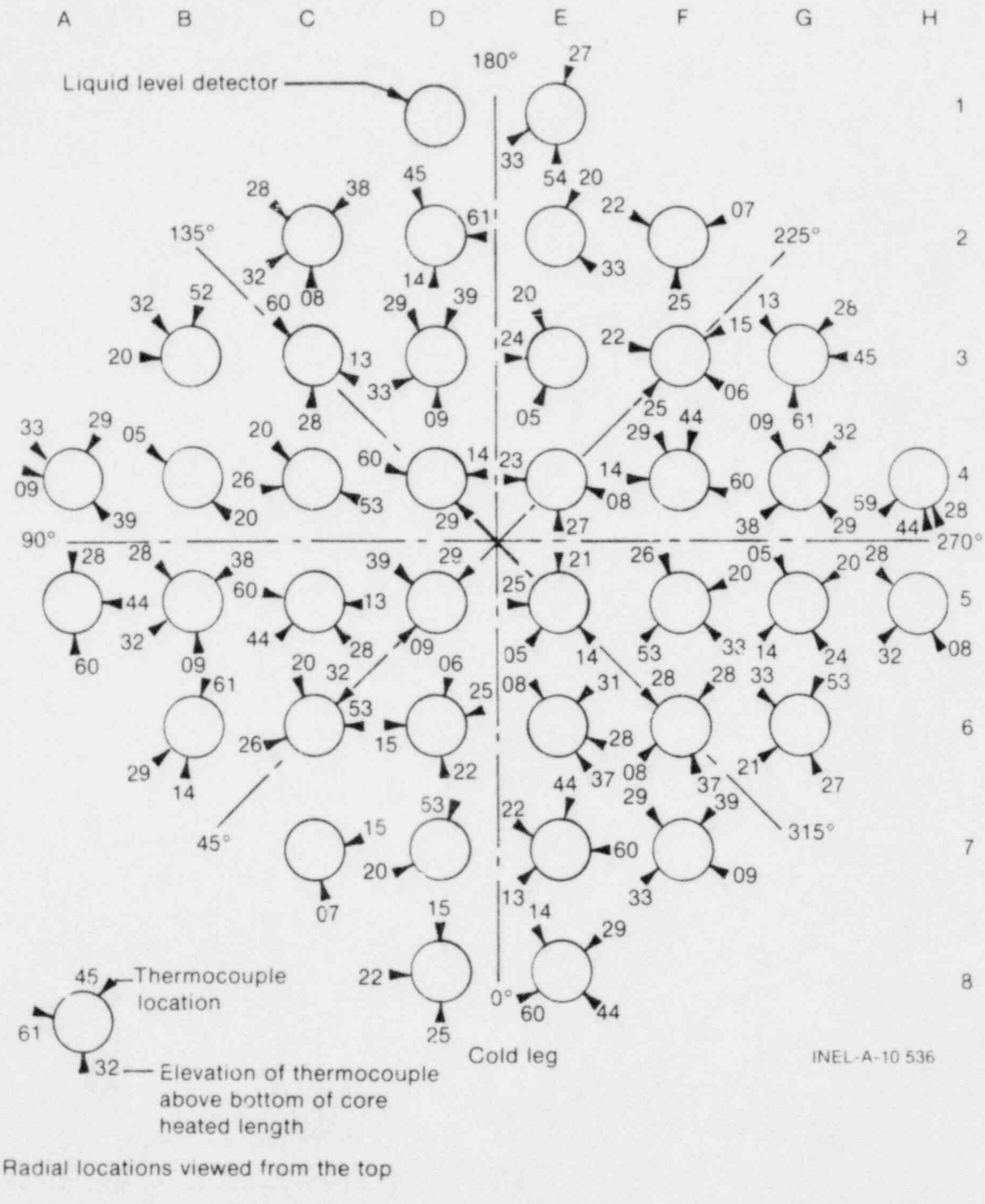

Fig. 1 Semiscale Mod-1 Core Geometry

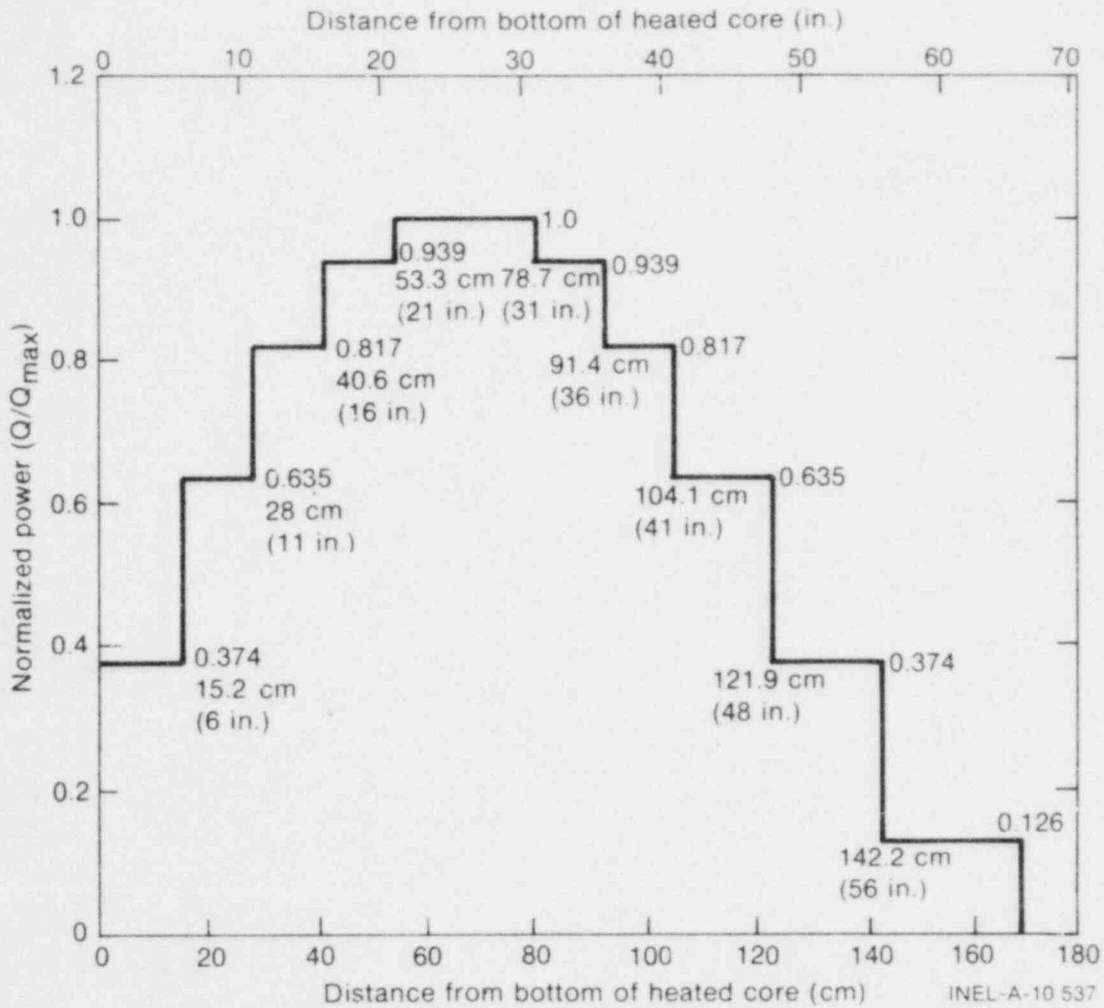

Fig. 2 Semiscale electrical heater rod axial power profile.

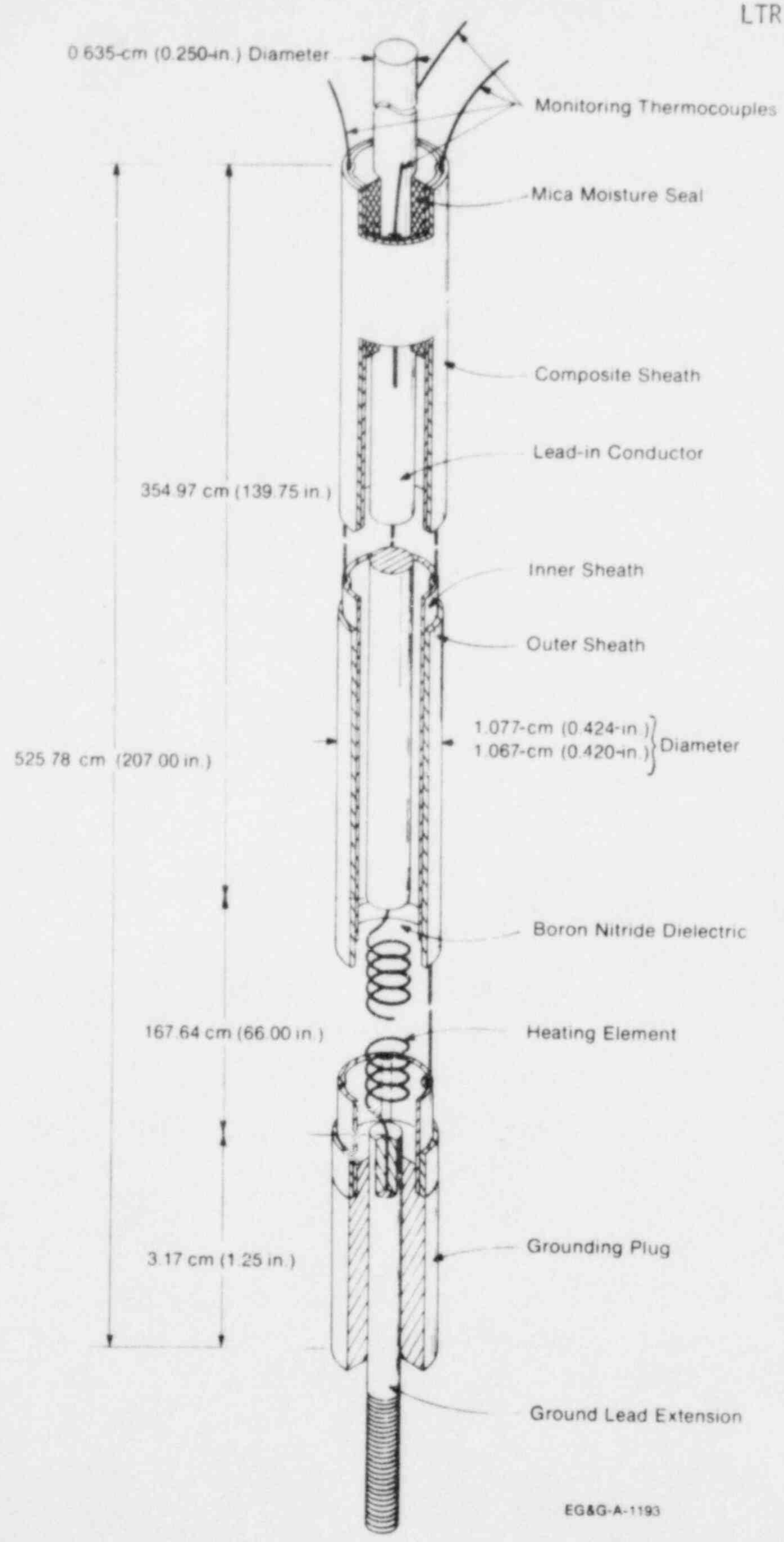

Fig. 3 Semiscale MOD-1 Electric Heater Rod.

The significant thermal-hydraulic features of this test are established in the first few seconds of the blowdown. The mass flux and quality for various axial elevations is shown in Figure 4. The core power for the test is shown in Figure 5.

The cladding surface temperature histories as measured by cladding thermocouples at the peak rod power axial elevation (74 cm from the bottom of the heated length) are shown in Figure 6. Of particular interest are the wide ranges of variation between the cladding temperatures seen by the rods. Those rods which saw an early (> 1 second) departure from nucleate boiling (DNB) have very similar cladding surface temperature histories, while those rods which saw an early DNB and then rewet (RNB), or a delayed (> 1 second) DNB had a wide variety of cladding temperature histories and significantly reduced peak cladding temperatures. Even more interesting is the fact that adjacent rods (for example D4 and D5) had significantly different cladding temperature histories, with D4 experiencing RNB and D5 experiencing no RNB.

The results of this test clearly demonstrate the occurrence of RNB during the blowdown portion of the LOCE, and the effects RNB can have on peak cladding temperature. A careful comparison of the cladding temperature histories with Figures 3 and 4 shows that the conditions associated with RNB result from the hot leg flow reversal and subsequent reduction in coolant quality along the rod after the initial cold leg break and flow stagnation. The problem becomes one of identifying the data trends and experimental conclusions that can be made with regard to why one rod rewet and another did not rewet at the same axial level.

To better understand the RNB behavior, the following areas were analyzed for data trends:

(1) Axial distribution of RNB,

(2) Radial distribution of RNB,

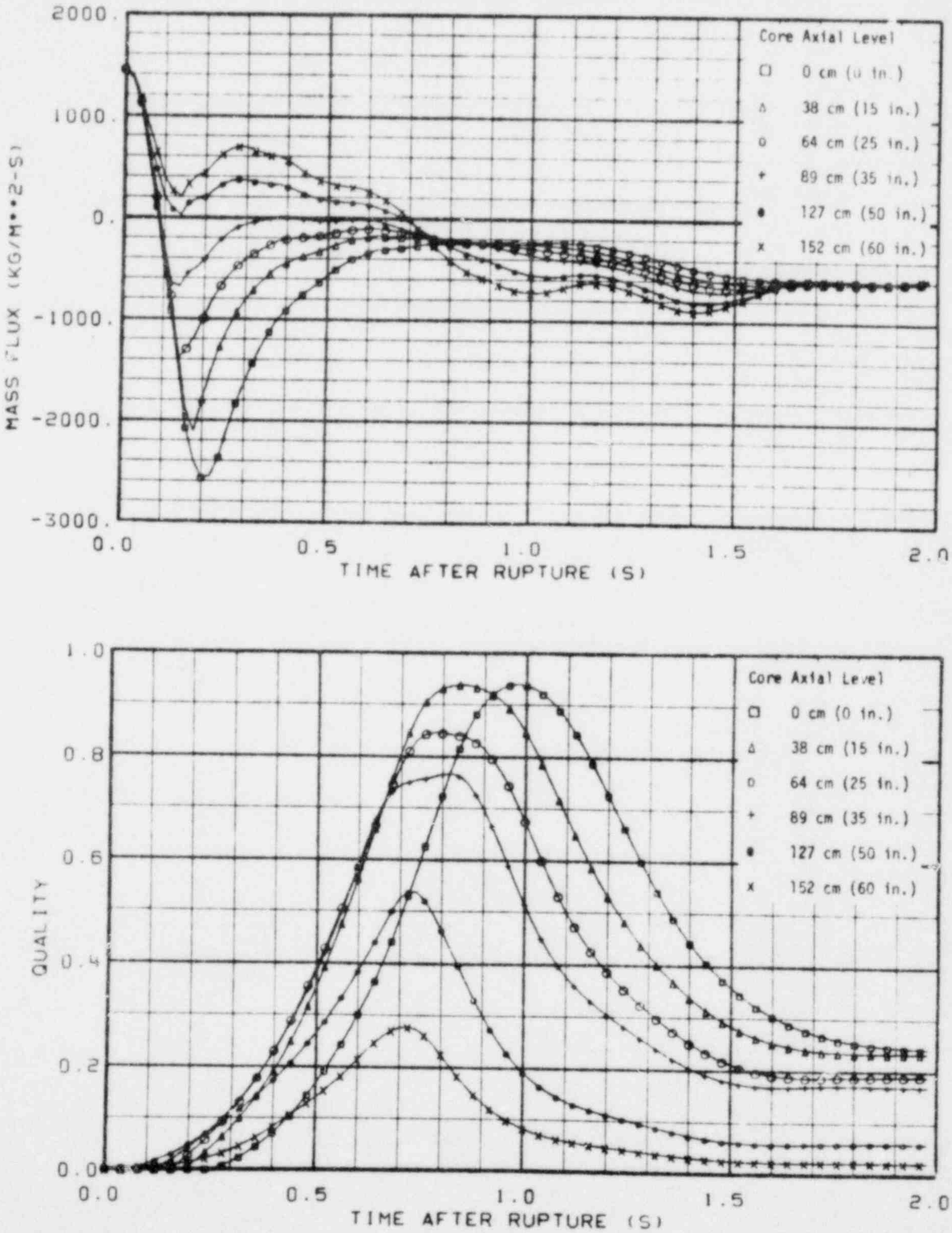

Fig. 4 Mass Flux and Coolant Quality for Semiscale Test S-02-9 Calculated by COBRA-IV.

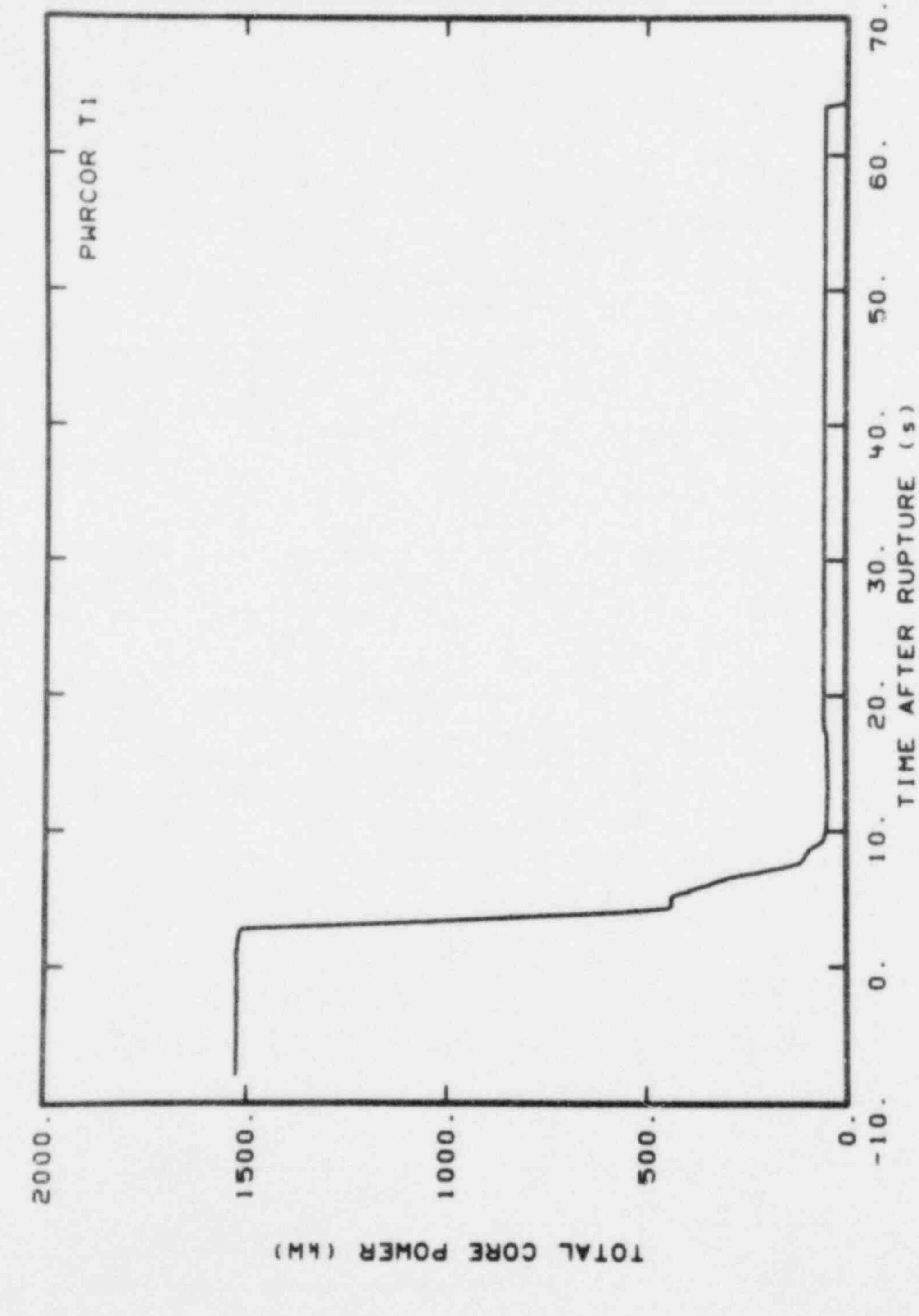

Fig. 5 Core power during Semiscale Test S-02-9.

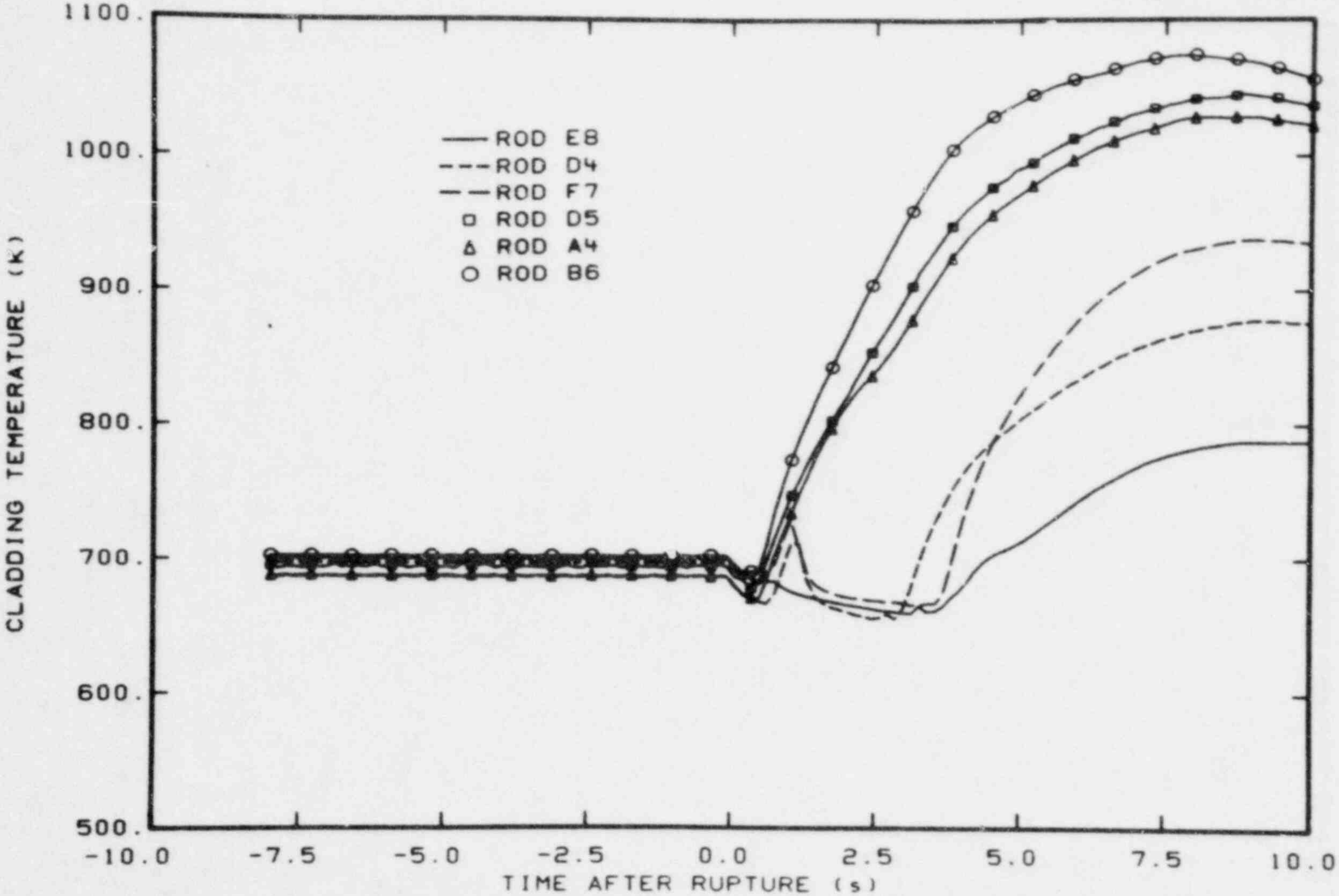

Fig. 6 Cladding surface temperature histories for Semiscale Test S-02-9 at the 74-cm axial core elevation.

(3) Effect of unpowered rods or cold walls on RNB,

(4) Effects of break nozzle geometry on RNB,

(5) Individual rod characteristics,

(6) Repeatability of test results.

1.2 Axial Distribution of RNB

The cold leg break blowdown simulations of the Semiscale testing program were found to have a definite RNB axial distribution. For example, the rewet distribution for Test S-02-7[33] (similar to Test S-02-9) is shown in Figure 7. From this figure, it can be seen that rewets occurred predominantly in the upper part of the core. Some axial zones where rewets occurred are bounded by areas where no rewets occurred.

The observed RNB axial dependence can probably be attributed to the interaction of local rod power denisty variations (surface heat flux) and the ratio of vapor to liquid flow rate (void fraction and quality). The dependency on vapor flow rate can be observed by examining the conditions at the peak rod power step (53- to 79-cm) as shown in Figures 8 and 9. Figure 8, which shows the surface temperature histories bounding the peak power step, demonstrates that the temperatures were fairly uniform over this portion of the rod prior to rewetting. However, the quality gradient over the same section, shown in Figure 9 at 1.18-s, shows a quality gradient of about 10% over the same section. The increase in vapor flow along this power step (as manifested by the quality gradient) was apparently large enough to preclude any rewetting on the lower portion of the rod while enough liquid was still available for rewetting on the upper portion of the rod. At a lower power step, however, further downstream, some rewetting occurred demonstrating that the vapor flow was not large enough to preclude rewetting at a lower power.

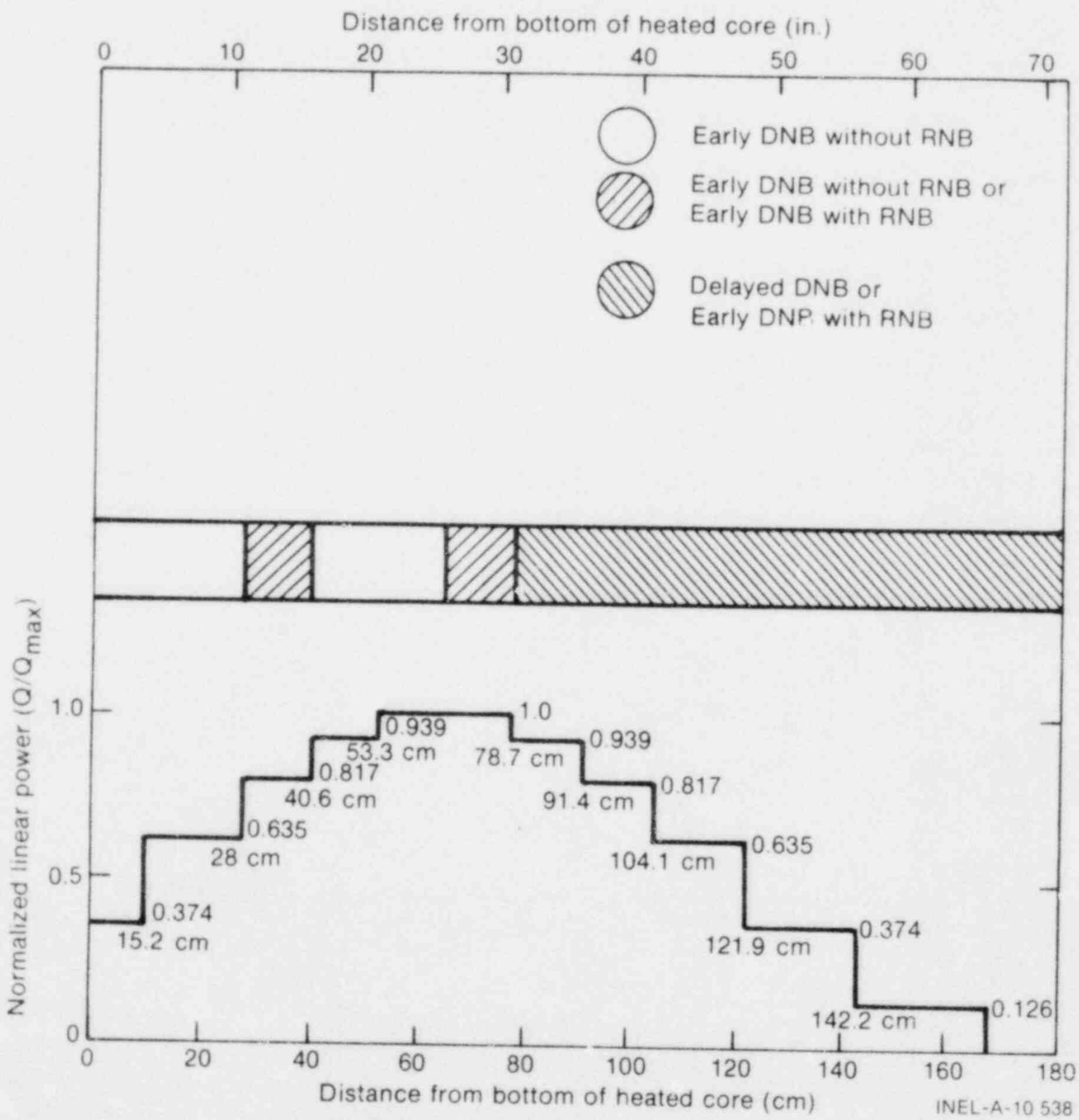

Fig. 7 Core axial rewet distribution during blowdown for Semiscale Test S-02-7.

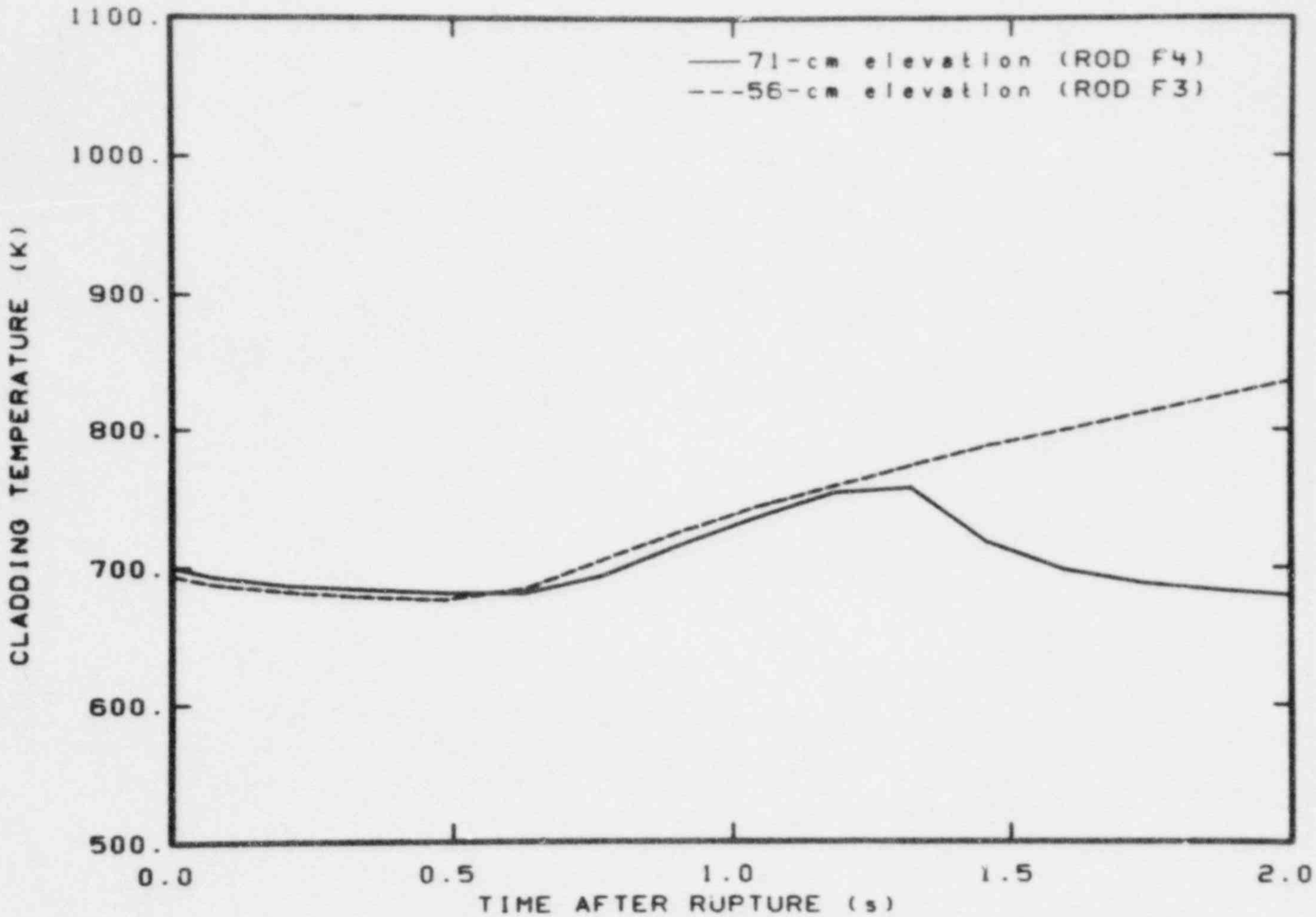

Fig. 8 Surface temperatures bounding the peak power step for Semiscale Test S-02-9.

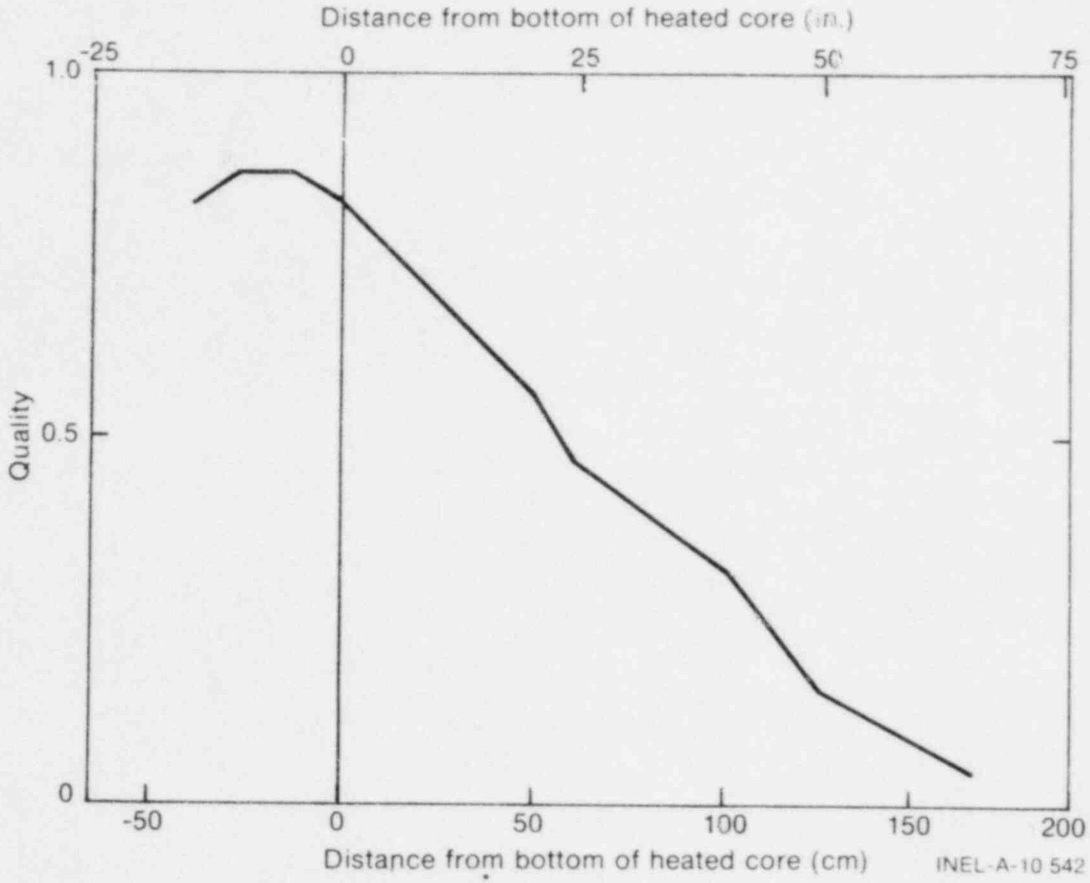

Fig. 9 Coolant quality gradient calculated by COBRA-IV at 1-18-s after rupture for Semiscale Test S-02-9.

The apparent dependency of RNB on vapor flow and local power density (surface heat flux) suggests that inherent differences from rod to rod that result in different local power densities may be the reason that some rods rewet and others did not at the same axial elevation during the same test. This possibility will be discussed later in the section on the possible effects of inherent rod characteristics on rewetting.

1.3 Radial Distribution of RNB

The radial distribution for observed rewetting in Test S-02-7 (similar to Test S-02-9) is shown in Figure 10[29]. This figure illustrates the pattern of rewet phenomena during a test with a flat core radial power profile.

The analysis of various rod groupings have been fruitless in establishing any kind of radial trend or pattern in RNB phenomena[29] that would be evident due to repeatable thermal-hydraulic behavior. However, this may be the result of several interacting local factors, such as inherent heater rod nonuniformities (see Section 1.6), cold wall effects, thermocouple distributions, etc., interacting to mask a thermal-hydraulic radial trend in data that would be evident if, for example, all the rods were identical with no unique characteristics distinguishing one rod from another.

1.4 Effect of Unpowered Test Rods

The effect of unpowered rods on the DNB-RNB behavior of adjacent rods was tested during the Semiscale Mod-1 Integral Blowdown-Reflood tests[26]. In this series of tests, Test S-04-5[34] was conducted with all 40 rods in the core powered, whereas Test S-04-6[34] was conducted with four of the 40 rods unpowered. The location of the four unpowered rods is shown in Figure 11. The unpowered rods would be analogous to control rod guide tubes in a PWR. All powered rods

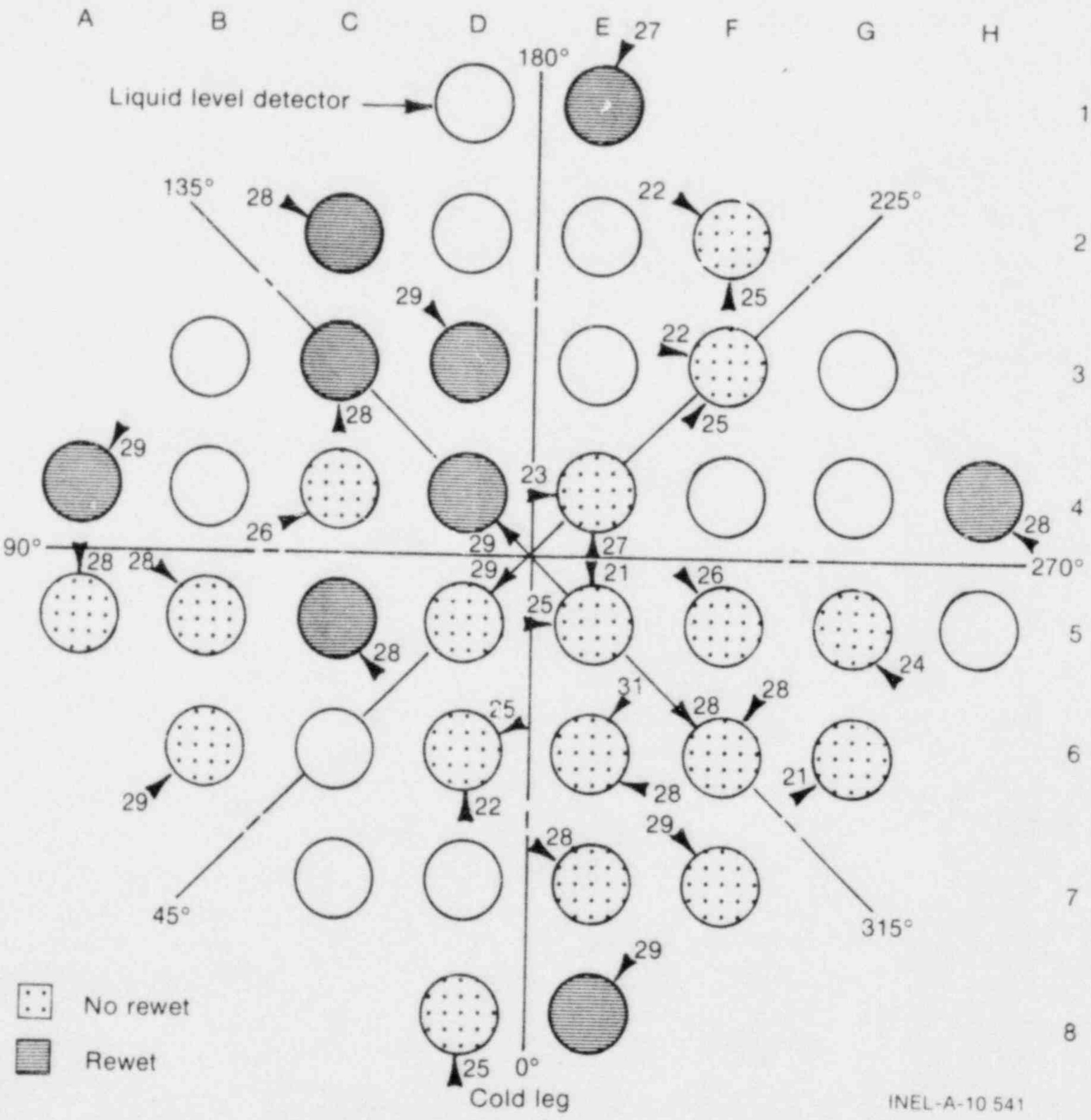

Fig. 10 Core radial rewet distribution during blowdown on the peak power plateau (53 to 79-cm) for Semiscale Test S-02-7.

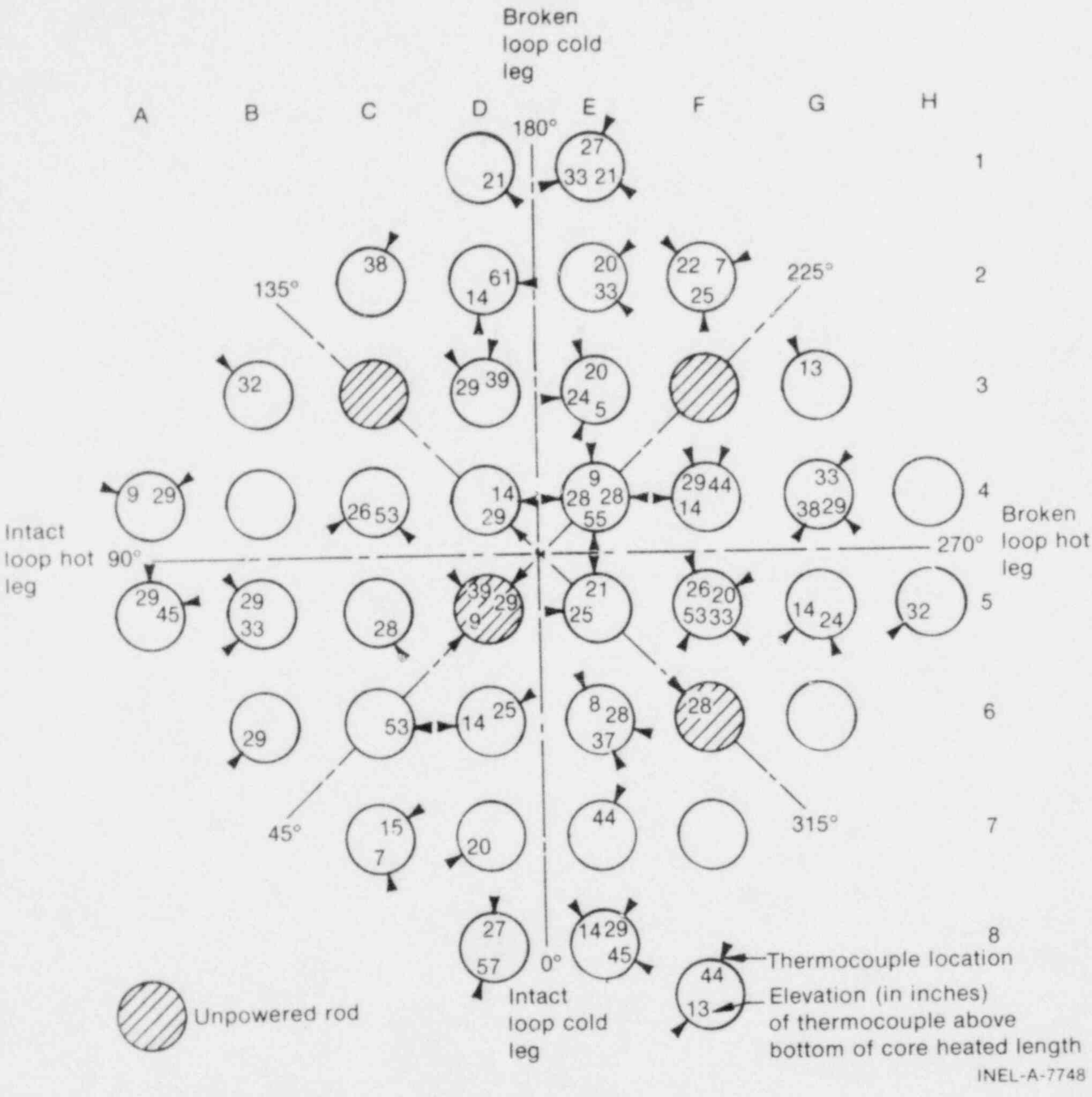

Fig. 11 Semiscale Mod-1 core geometry for Semiscale Test S-04-6.

were operated at an initial peak power density of 37.73 kW/m, with the exception of the high power rods (D4, E4, and E5) in Test S-04-6 which were operated at 39.70 kW/m. All other conditions were essentially the same for both tests.

Figure 12 compares the results observed in both tests for the RNB-DNB behavior. In general, the results show a change in DNB behavior for rods adjacent to unpowered rods, but little change in DNB behavior for rods not adjacent to unpowered rods. In Test S-04-6, for the rods adjacent to unpowered rods, 28 of the 34 thermocouple locations between the 0- and 84-cm elevations exhibited a change from early DNB to early DNB with rewet or delayed DNB; or from early DNB with rewet to delayed DNB when compared with Test S-04-5[30].

Typical changes in cladding temperature histories as a result of the change in DNB behavior due to unpowered rods are shown in Figures 13, 14, and 15. Figure 13 illustrates a change on rod D6 at the 63.5-cm location from early DNB to delayed DNB, and Figure 14 illustrates a change on rod D2 at the 36-cm location from early DNB to early DNB with rewetting. Figure 15 illustrates that the DNB and cladding temperature history for rod B6, which was not adjacent to an unpowered rod is essentially unchanged from Test S-04-5 to Test S-04-6.

1.5 Effects of Break Nozzle Geometry

In preparation for the LOFT counterpart tests, the break leg nozzle geometry of the Semiscale facility was altered to better simulate the LOFT reactor blowdown leg nozzle. The new nozzle design was a geometrically scaled version of the nozzle used in the LOFT test facility and differed significantly from the converging-diverging nozzle (Henry nozzle) used in previous Semiscale tests[28].

To test the effect of nozzle geometry on the system response, a baseline integral blowdown-reflood test was repeated (Test S-04-6) as

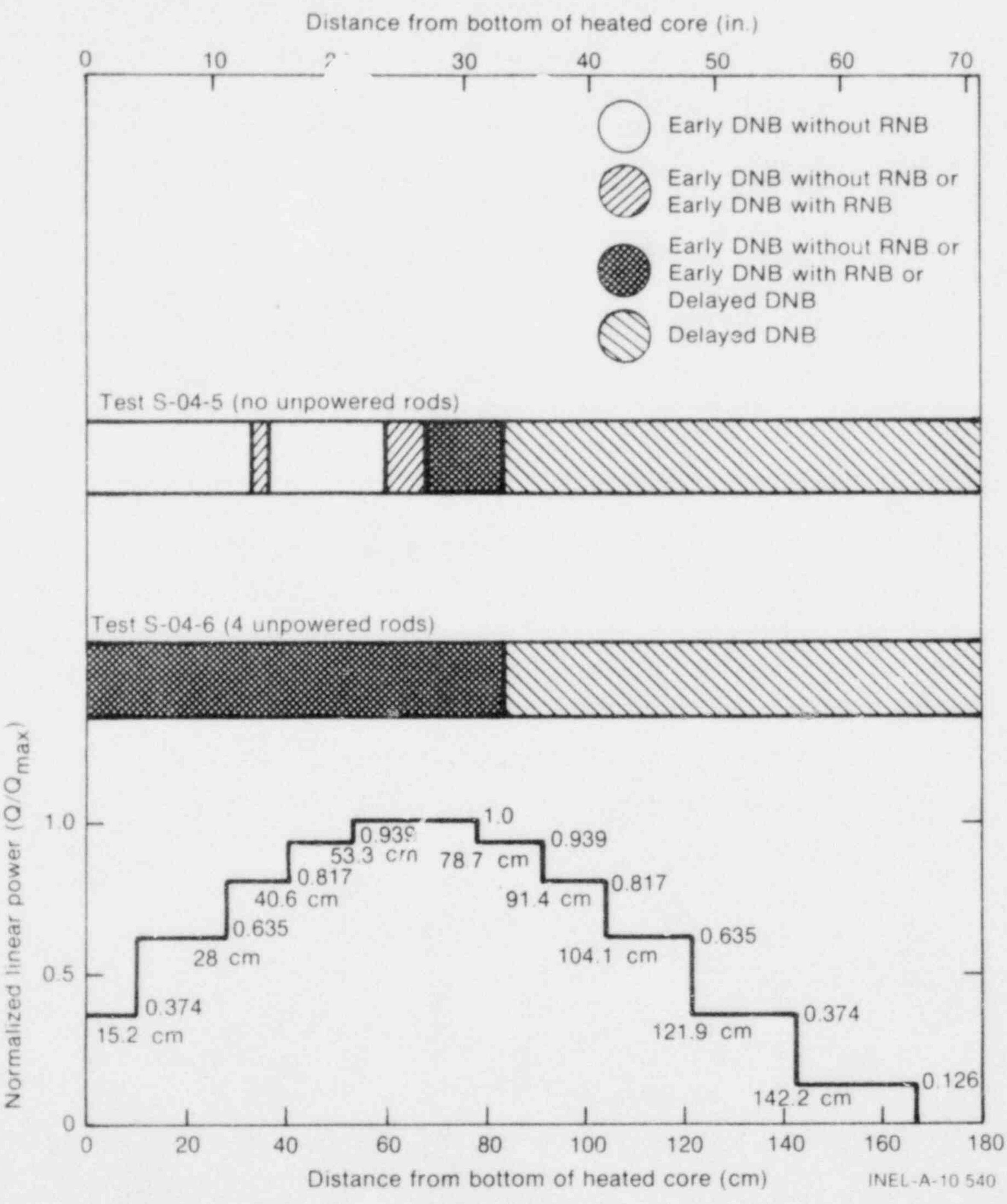

Fig. 12 Core axial rewet distribution during blowdown for Semiscale Tests S-04-5 and S-04-6.

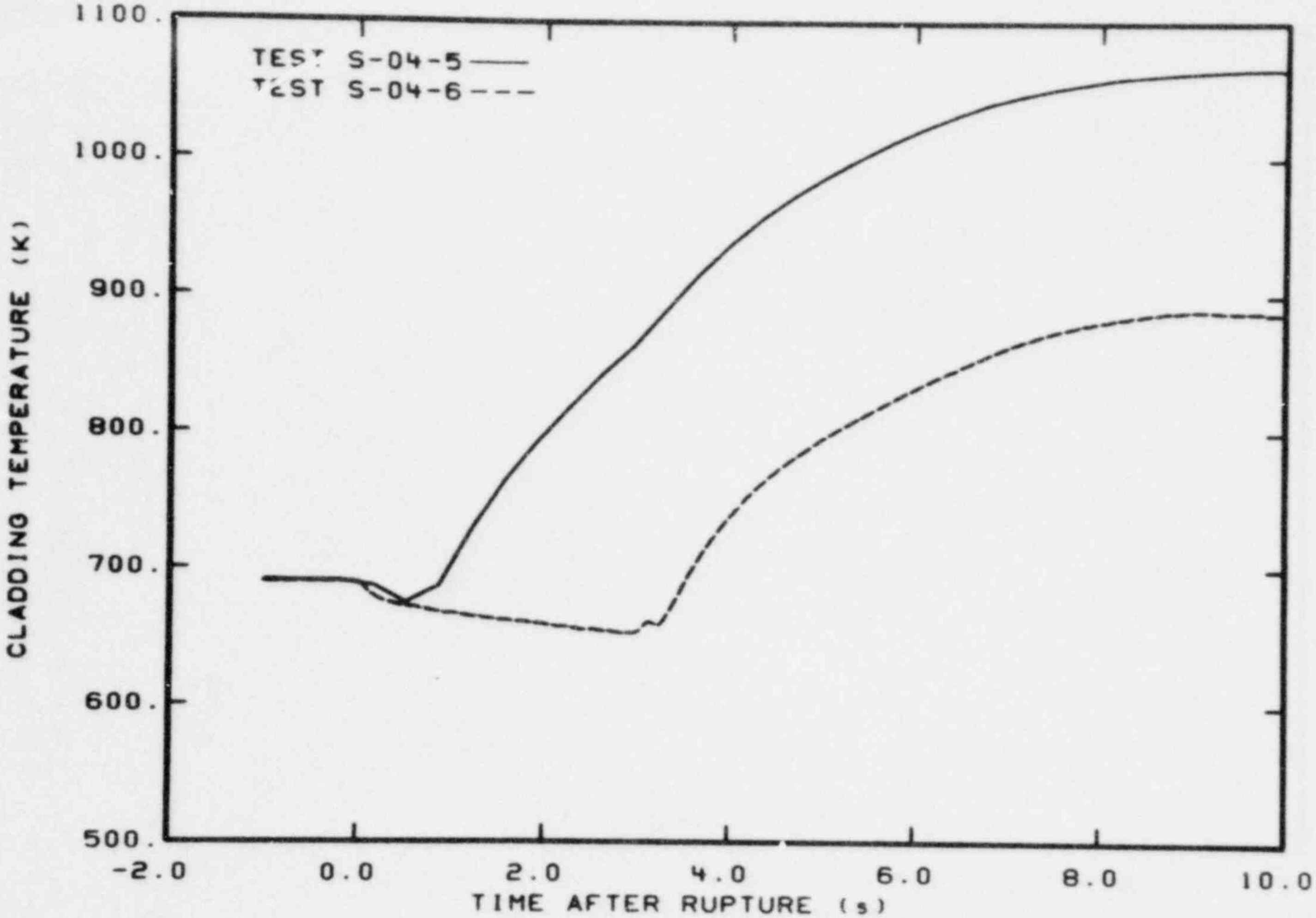

Fig. 13 Rod D6 cladding surface temperature histories at the 63.5 cm elevation for Semiscale Tests S-04-5 and S-04-6.

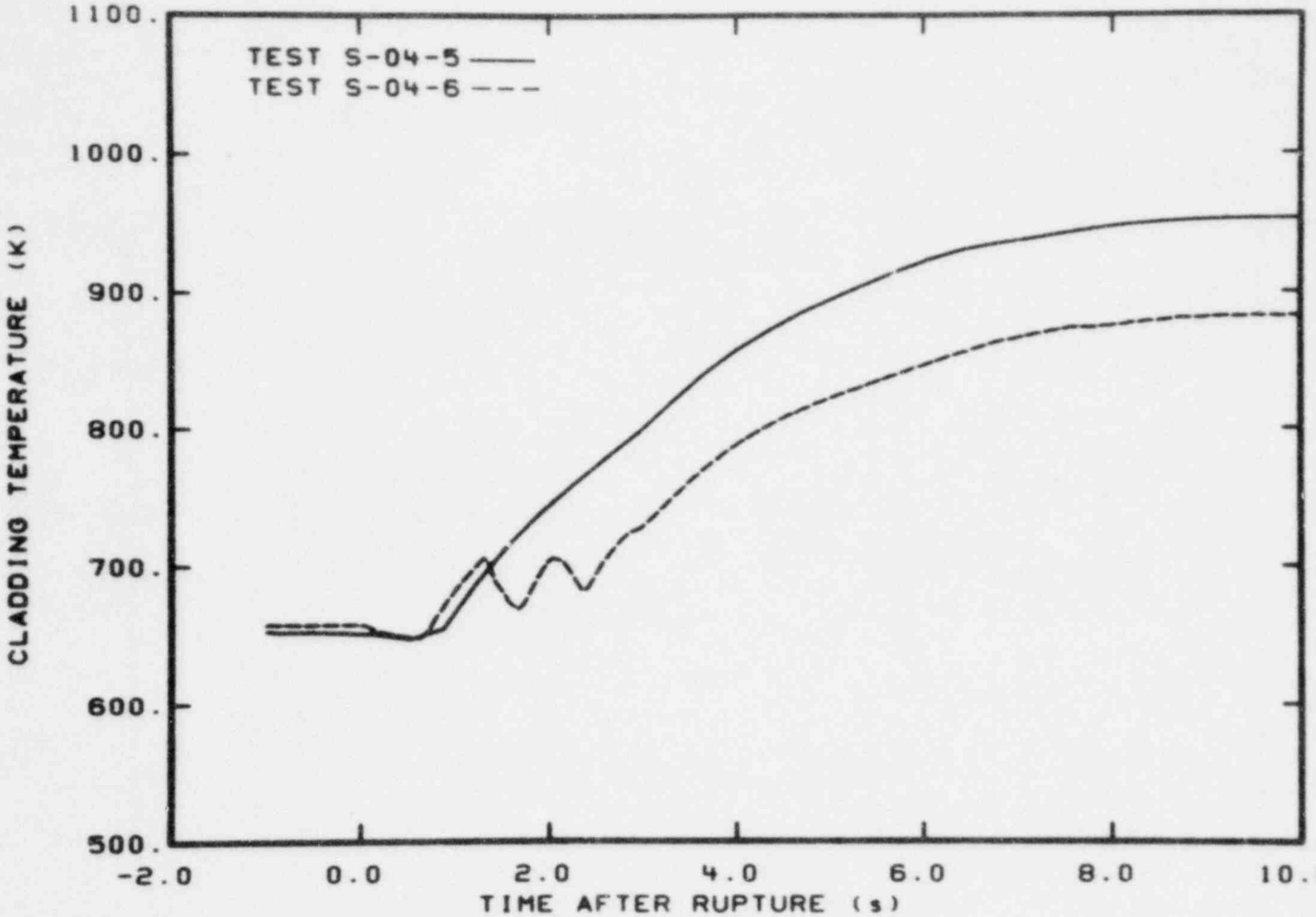

Fig. 14 Rod D2 cladding surface temperature histories at the 36-cm elevation for Semiscale Tests S-04-5 and S-04-6.

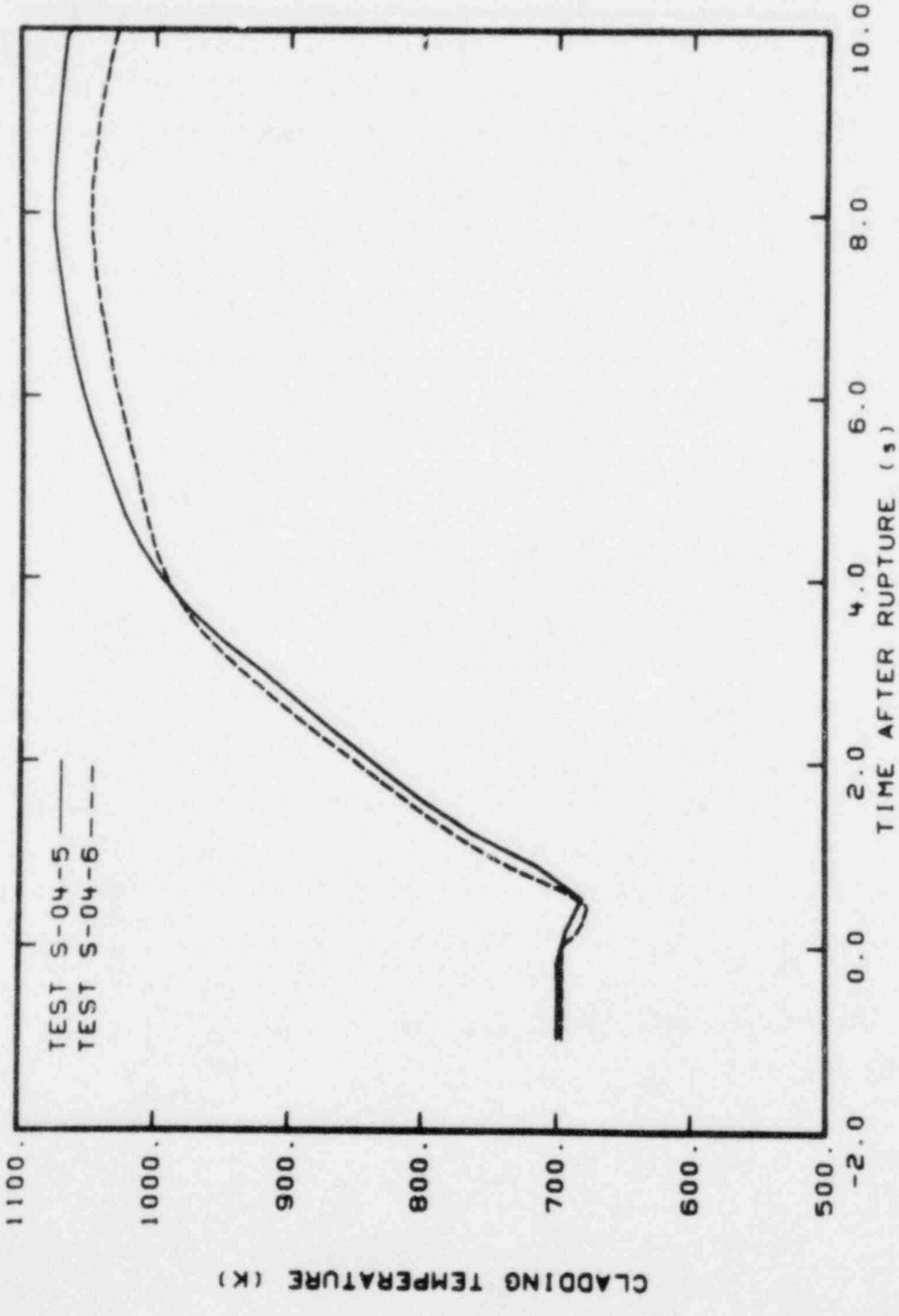

Fig. 15 Rod B6 cladding surface temperature histories at the 74-cm elevation for Tests S-04-5 and S-04-6.

Test S-06-5[35]. Any changes in the system response could then be analyzed on the basis of the difference in nozzle geometry.

A comparison of the break flows for the two tests is shown in Figure 16. From this figure, it can be seen that during the subcooled depressurization, the break flow rate for the LOFT counterpart nozzle was considerably lower than that for the Henry nozzle. In addition, because the break flow rate for the LOFT counterpart nozzle was lower, the fluid just upstream from the nozzle did not reach saturation until about 0.5 seconds later than the corresponding condition for the Henry nozzle, thereby resulting in a slightly later transition from a relatively high subcooled flow rate to a lower saturated flow rate.

The lower break flow rate in the early stages of blowdown with the LOFT counterpart nozzle resulted in a core inlet flow reversal which was smaller than the corresponding flow for the Henry nozzle. The smaller flow reversal caused a shift to earlier DNB in the central and lower portion of the core, with subsequently higher peak cladding temperatures.

A comparison of the DNB-RNB behavior for the two nozzle tests and also Test S-04-5 (discussed previously) is shown in Figure 17. This figure shows that the nozzle geometry change negated any of the RNB and delayed DNB behavior observed when the unpowered rods were included in the test. The change in flow behavior was enough to suppress previous RNB behavior, reiterating that RNB is a strong function of the coolant conditions.

1.6 Individual Rod Characteristics

The differences in individual rod characteristics are manifested in comparisons shown in Figures 18 and 19. Figure 18 is a comparison of typical steady state cladding temperatures at all the thermocouple elevations prior to testing. Differences between the thermocouple readings at the same axial elevations are within 30 K and seem to imply that differences exist in the characteristics of the individual rods.

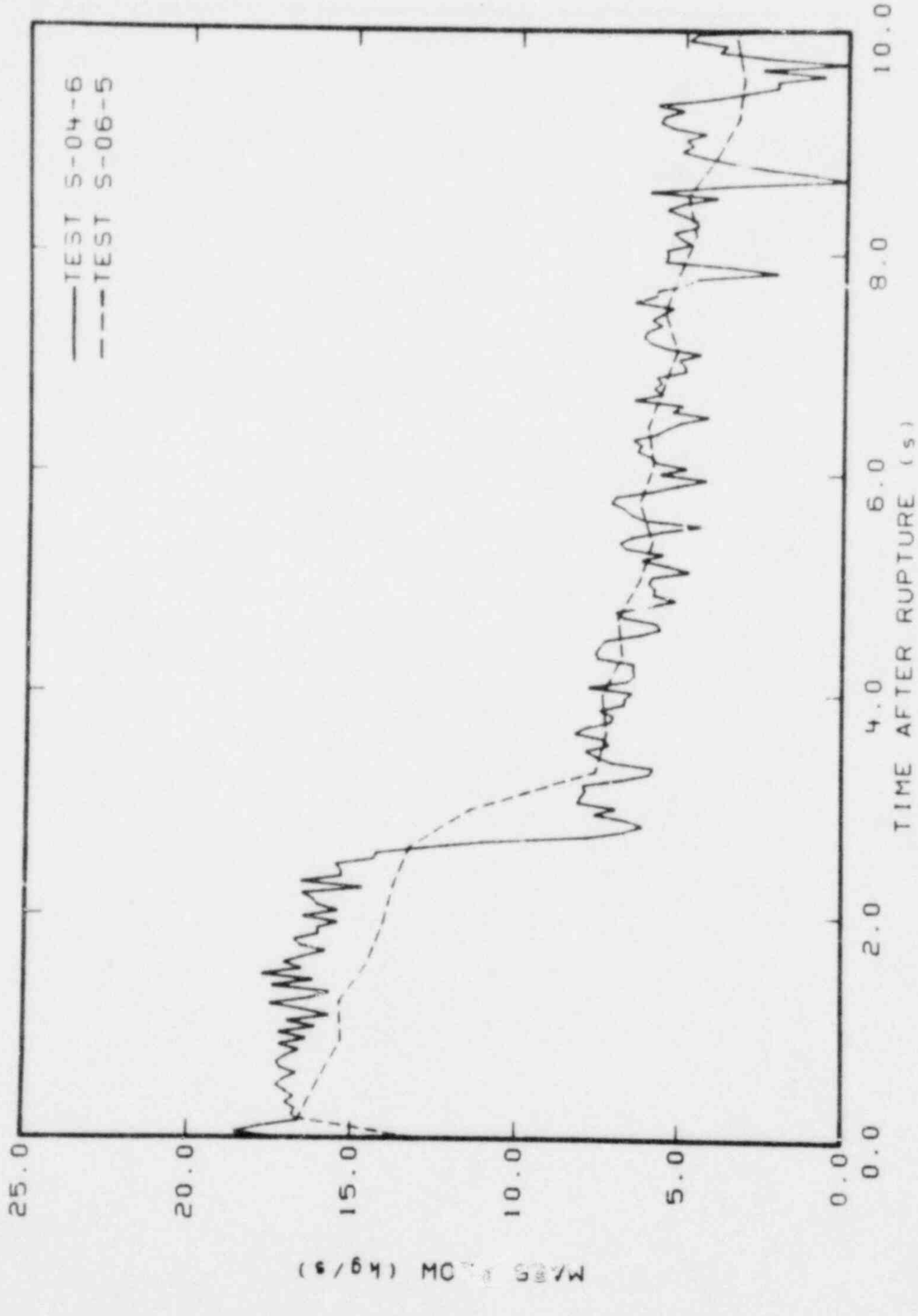

Fig. 16 Nozzle break flow rate histories for Semiscale Tests S-04-6 and S-06-5.

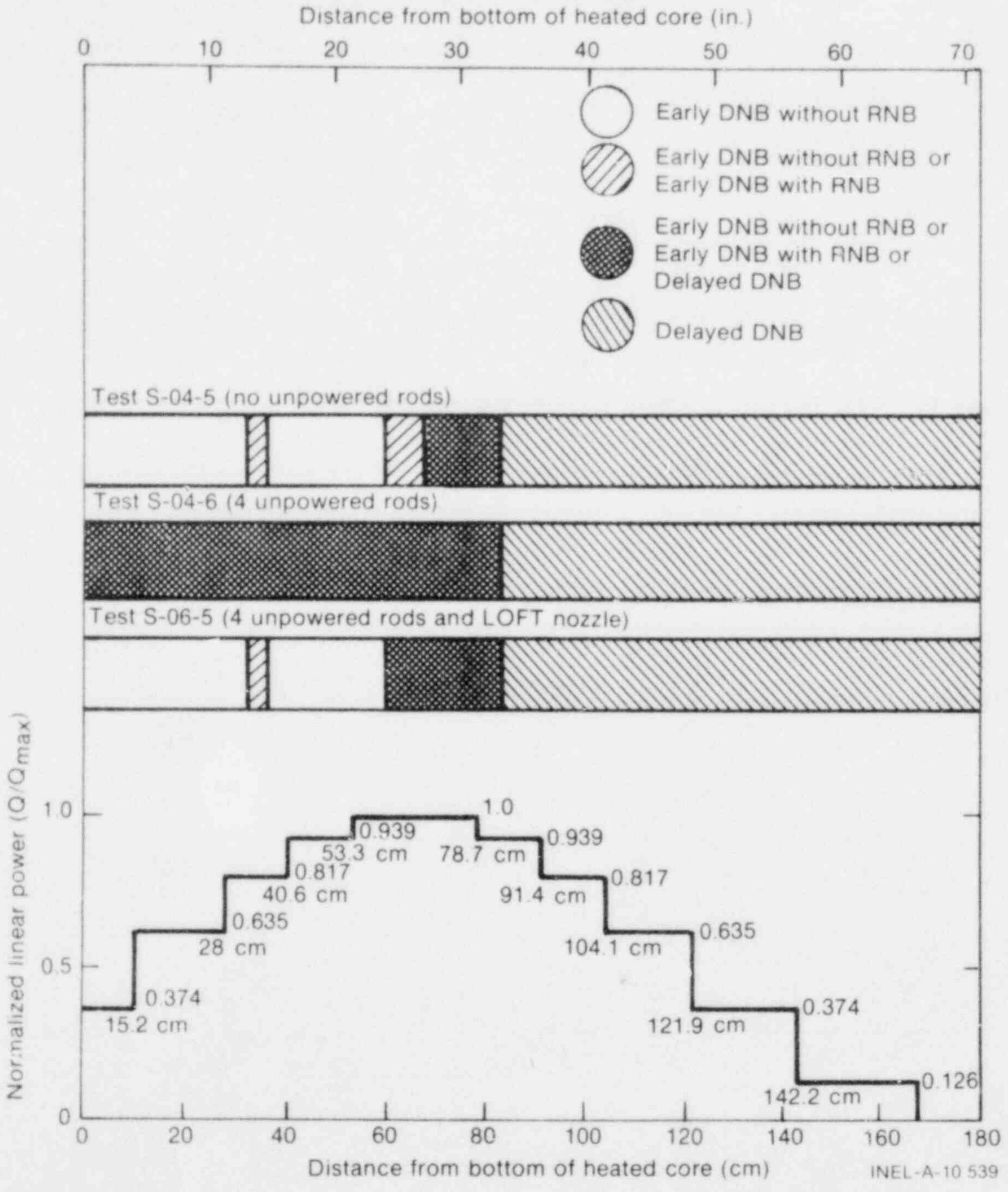

Fig. 17 Core axial rewet distribution during blowdown for Semiscale Tests S-04-5, S-04-6, and S-06-5.

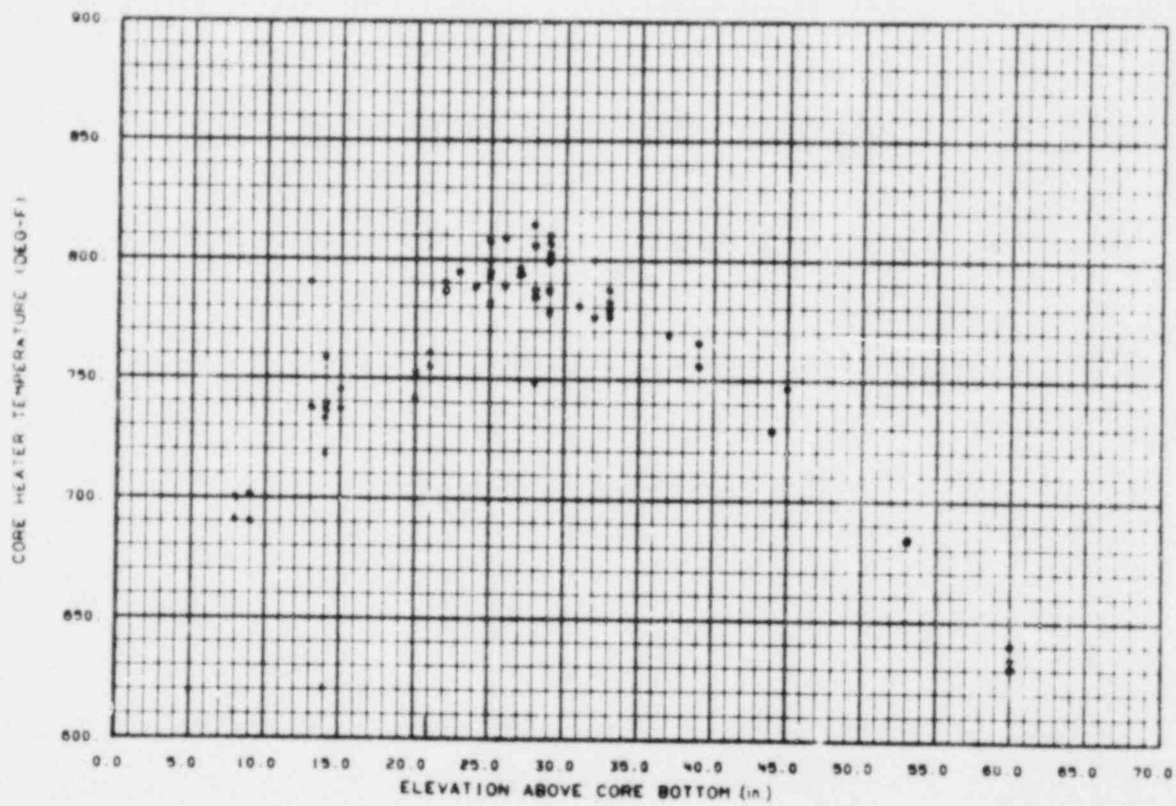

Fig. 18 Semiscale Steady State Axial Cladding Temperatures.

Figure 19 is a comparison of cladding temperature histories at the 73.7-cm elevation for rods D4 and D5 during the blowdown heat transfer Test S-02-9. Reference to Figure 1 shows that these two thermocouple locations face the same flow channel and would be expected to respond in essentially the same manner. However, Figure 19 clearly shows that during the blowdown, one rod rewet at that axial elevation and the other did not. The implication is evident that individual characteristics in the rod result in a different response.

Possible sources for the differences in individual rods may be:

(1) Local power density variations due to asymmetries in heater coil density or distance between the coils and the inside cladding surface (this may introduce azimuthal variations around the same rod as well as different power densities between different rods at the same axial elevation),

(2) Complex two dimensional anomalies in thermocouple contact resistance, contact resistance between insulator and cladding, or contact resistance between the cladding annuli,

(3) Azimuthal location of the thermocouple in relation to the heater rod coils, as shown in Figure 20,

(4) Anomalies in individual rod material thermal properties.

There is a distinct possibility that two or more of the above anomalies may be interacting to perturb the thermal response of a heater rod.

To evaluate some of the differences in the heater rods, power pulse and dry core heatup tests were conducted with the Semiscale core[17]. The power pulse tests correlated a change in thermocouple temperature for a given step change in heater rod power. Any

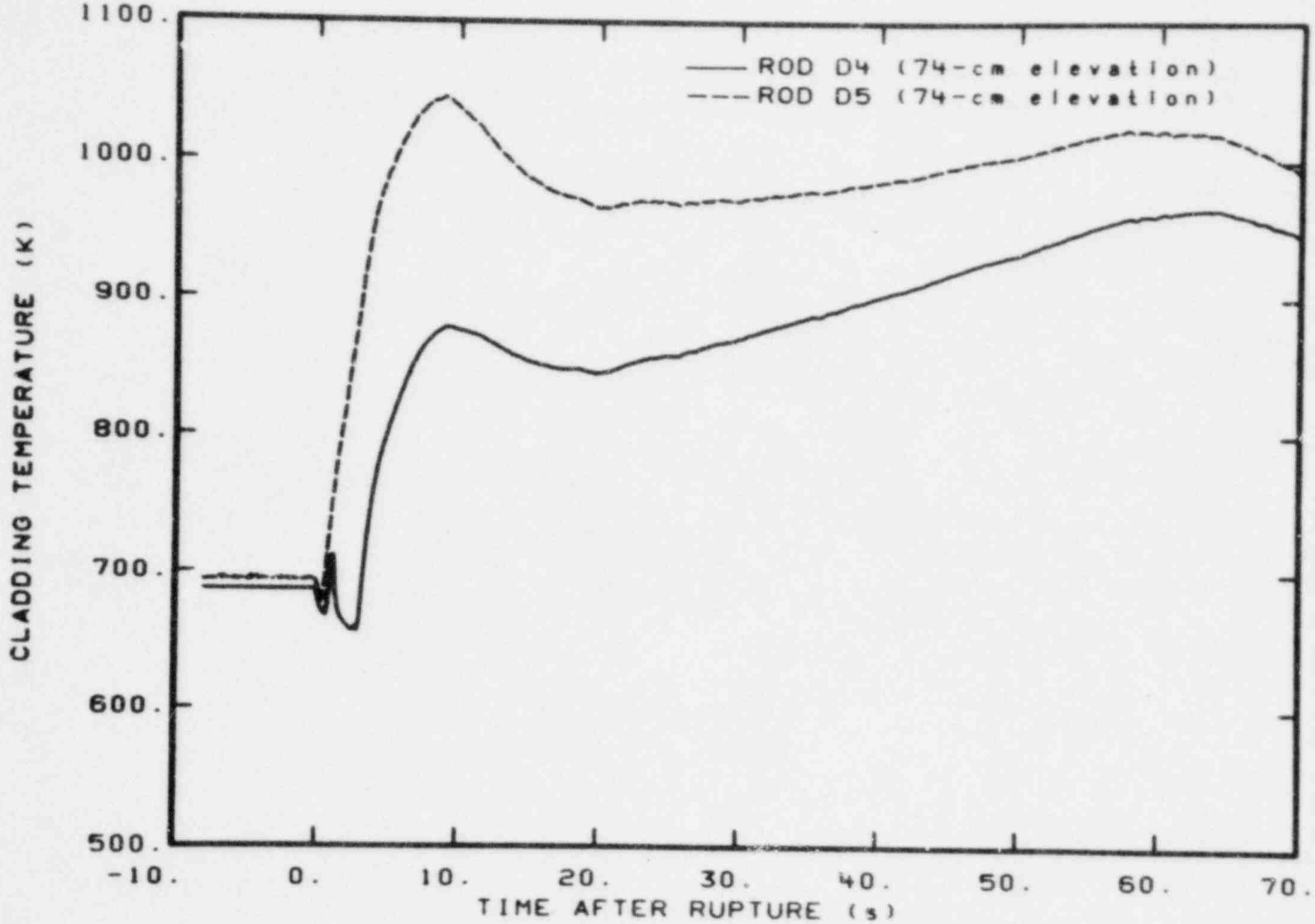

Fig. 19 Cladding surface temperature histories for rods D4 and D5, Semiscale Test S-02-9.

Fig. 20 Cross section of a Semiscale electrical heater rod.

differences in local power density or thermal response from rod to rod would be expected to result in different temperatures with higher local power densities or thermal response producing larger ΔT's. Although the temperatures varied from rod to rod, there was found to be no consistent relation between the magnitude of a rod's power pulse ΔT and whether it rewet during blowdown or not. In addition, this test could give no indication of possible azimuthal asymmetries associated with the surface heat flux and cladding temperature of a rod.

The dry core heatup tests consisted of heating up the core in air with no coolant. Since these tests resulted in a rod heat up that was essentially adiabatic, local power densities could be calculated as a function of the thermocouple temperature gradient with respect to time. Once again differences between rod local power densities were found, but no correlation was evident between these differences and whether a rod rewet or not. Here again however no information could be deduced concerning azimuthal asymmetries on the same rod.

Although no correlation could be found between differences in rod local power densities and differences in rod RNB behavior, it is not conclusive that no such relation exists. The interaction of some other heater rod anomalies may mask or dominate the power density effect, such as complex two-dimensional effects creating azimuthal surface heat flux variations that would dominate variations between the axial power density on different rods. Another factor that could dominate the power density effect may be a cannister or cold wall effect (discussed in Section II) promoting rewet on rods that otherwise have a relatively higher local power density.

The presence of inherent rod characteristics on the rewet behavior of the heater rods was also evident in the Semiscale reflood tests. For example, the rewet behavior of rods D4 and D5 are compared in Figure 21 for the reflood Tests S-03-1, S-03-2, S-03-3, and S-03-4[36]. Here D4 is seen to clearly rewet sooner during the reflood than D5, a behavior consistent with the RNB behavior observed

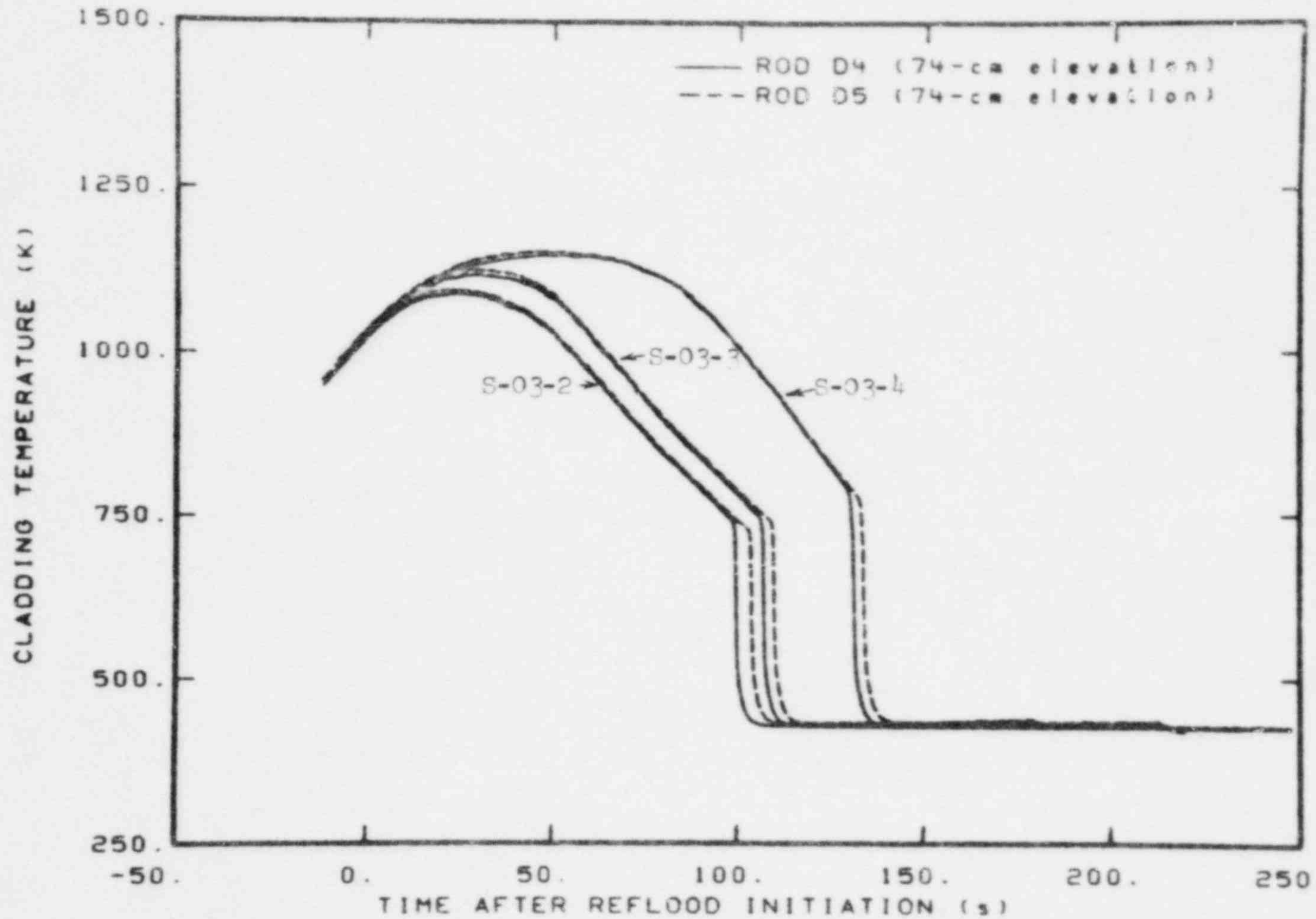

Fig. 21 Cladding surface temperature histories for rods D4 and D5, Semiscale reflood tests S-03-2, S-03-3, and S-03-4.

with these two rods. This also demonstrates that at least some factors influencing the rewetting of a rod during blowdown (RNB) are the same as those influencing the rewetting of a rod during reflood.

1.7 RNB Repeatability

The occurrence of RNB has a strong influence on the peak cladding temperature experienced by a rod during a LOCE. Because of this, it is important to determine if this phenomenon is repeatable on a given rod from test to test (assuming the test conditions are similar) or if it is random in nature, varying from rod to rod in different tests. Semiscale blowdown heat transfer Tests S-02-7, S-02-9, and S-02-9A were three tests conducted in a similar manner and were thus shown to demonstrate the repeatability of the test results.

The rewet behavior for Tests S-02-7, S-02-9, and S-02-9A for the thermocouples in the peak power region is shown in Table II. These results are typical, and a statistical analysis of these data has shown that the probability of these events being random is less than 0.02[29]. These results, coupled with the results of other tests demonstrate that the core thermal response and RNB is a very repeatable phenomenon.

TABLE II

INDICATIONS OF REWETTING AT ROD HOT SPOTS

Thermocouple	Test S-02-7	Test S-02-9A	Test S-02-9
TH-E5-21			
TH-G6-21			
TH-F2-22			
TH-F3-22			
TH-E4-23			
TH-F2-25			
TH-E5-25			
TH-G5-25			
TH-D6-25			
TH-C4-26			
TH-F5-26			
TH-E1-27	Rewet	Rewet	Rewet
TH-E4-27			
TH-C2-28	Rewet	Rewet	Rewet
TH-C3-28	Rewet		
TH-C5-28	Rewet		Rewet
TH-E6-28			
TH-F6-28J			
TH-F6-28P			
TH-D3-29	Rewet	Rewet	Rewet
TH-D4-29	Rewet		
TH-D4-29	Rewet	Rewet	Rewet
TH-A5-29			
TH-B5-29			
TH-D5-29			
TH-B6-29			
TH-E7-29		Rewet	Rewet
TH-F7-29		Rewet	Rewet
TH-E8-29	Rewet	Rewet	Rewet
TH-E6-31	Rewet	Rewet	Rewet

2. LOFT TEST L2-2

The Loss-Of-Flow Test (LOFT) reactor is a volume scaled model of a pressurized water reactor (PWR)[27]. An illustration of the LOFT reactor system configuration is shown in Figure 22. As shown in this figure, the LOFT reactor consists of an active coolant loop, a cold leg and hot leg simulated break path, and a 1.68-m active fuel length nuclear core. Several series of LOCE are planned for this reactor to evaluate LOCA and ECC injection phenomena for a variety of initial reactor operating conditions.

LOFT Test L2-2 is the first test in a series of LOCE's to be performed in the LOFT nuclear reactor. A representation of the core configuration illustrating instrument locations is shown in Figure 23 and Figure 24 gives a more detailed description of fuel rod axial thermocouple locations.

The initial conditions for Test L2-2 are shown in Table III. A preliminary evaluation and summary of test results are presented in the experiment quick look report[37] and extensive analysis of test results is presently in progress. The discussion of L2-2 results in this document will be confined to examining the rewetting which occurred in the core during the blowdown phase of the LOCE and pertinent thermal hydraulic material relevant to identifying possible rewetting mechanisms.

The cladding temperature history for fuel rod F8 in Fuel Assembly 5 at the 26-inch (66-cm) elevation (axial hot spot) is shown in Figure 25. This figure illustrates the general cladding surface temperature response for the center module fuel rods at the peak axial power during L2-2.

A sequence of axial temperature profiles illustrating the initial boiling crisis for center module rods is shown in Figure 26. This sequence is for the period 0.0 to 5.0 seconds after rupture and

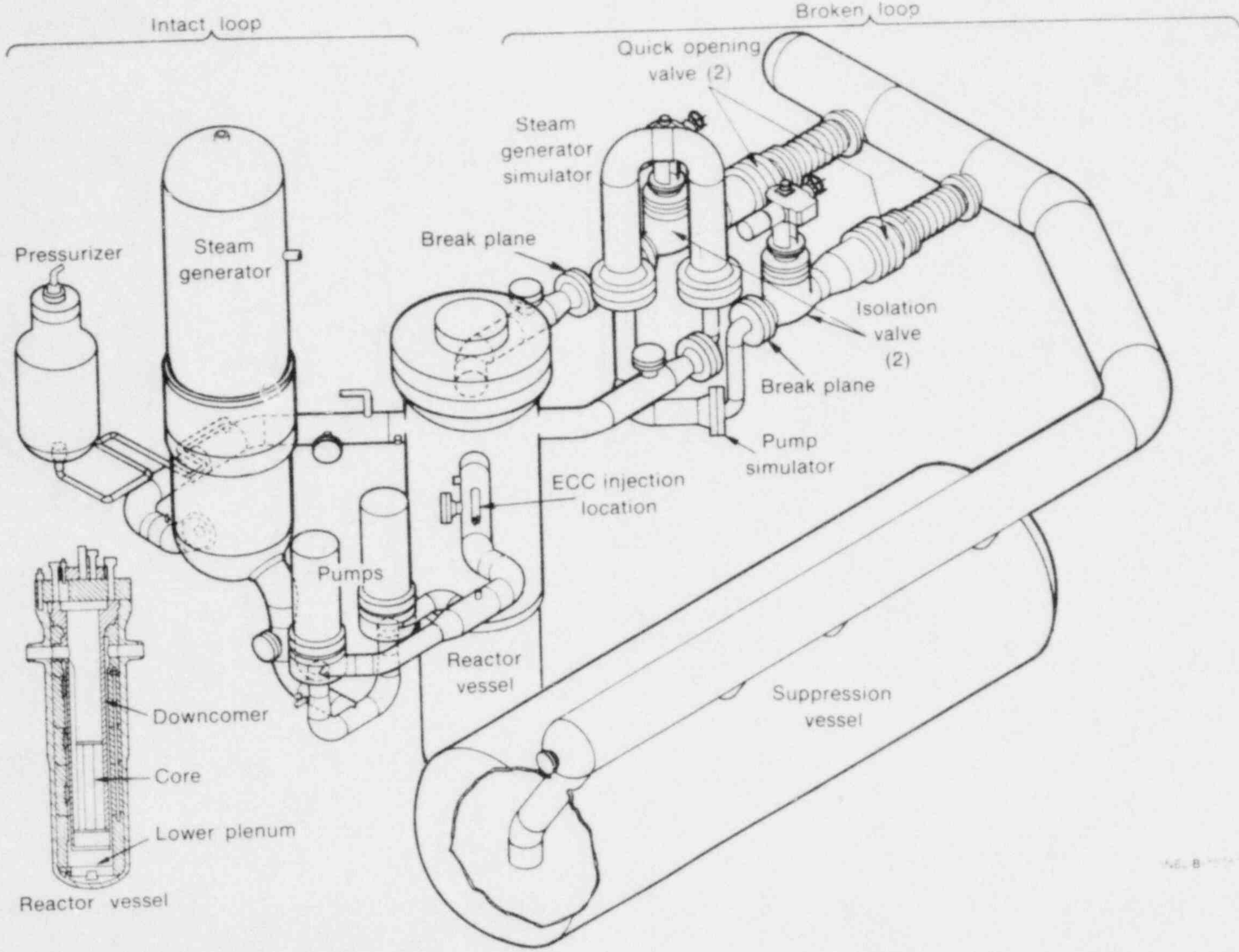

Fig. 22 LOFT Reactor System Configuration

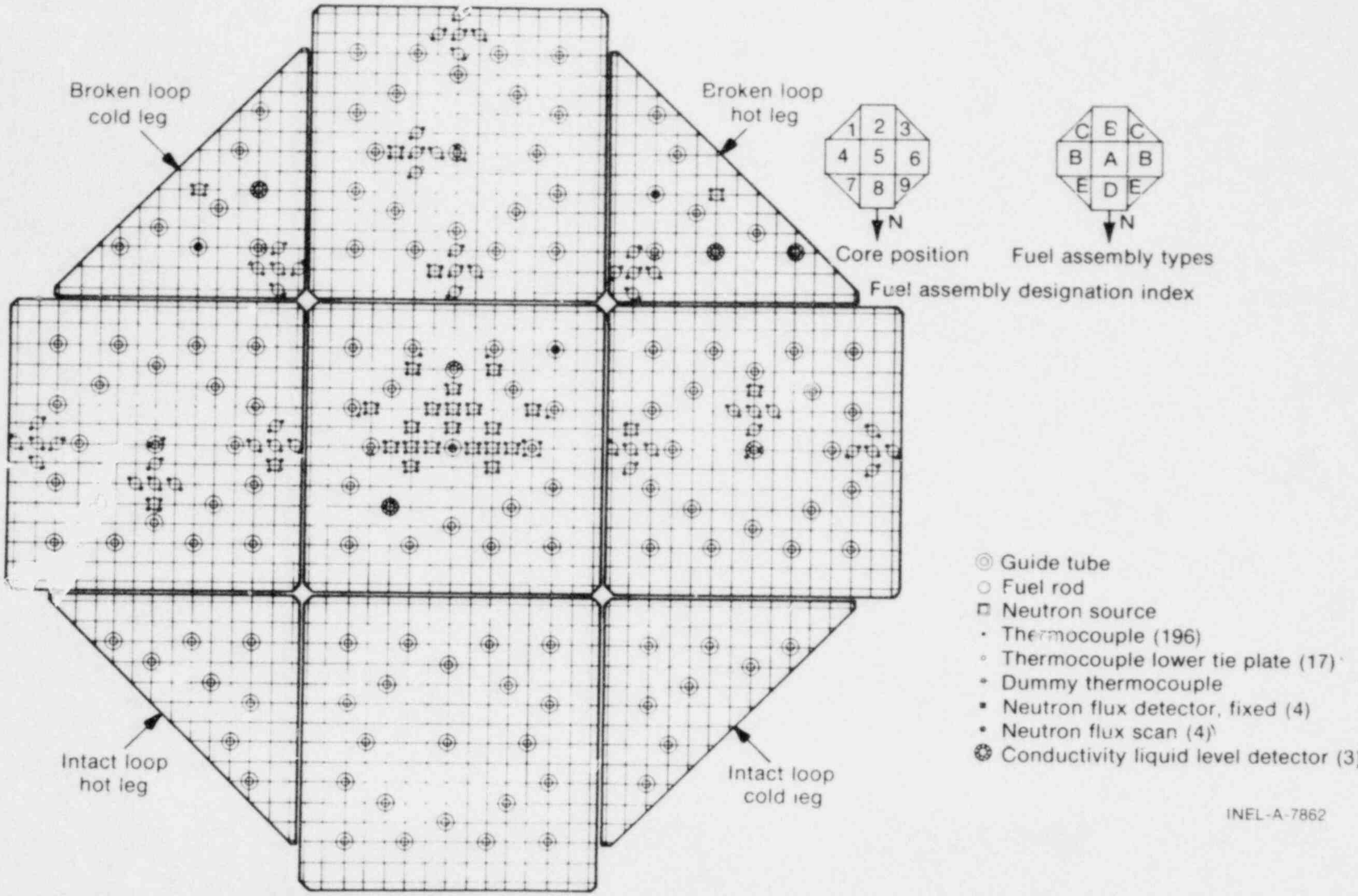

Fig. 23 LOFT Core Configuration and Instrumentation.

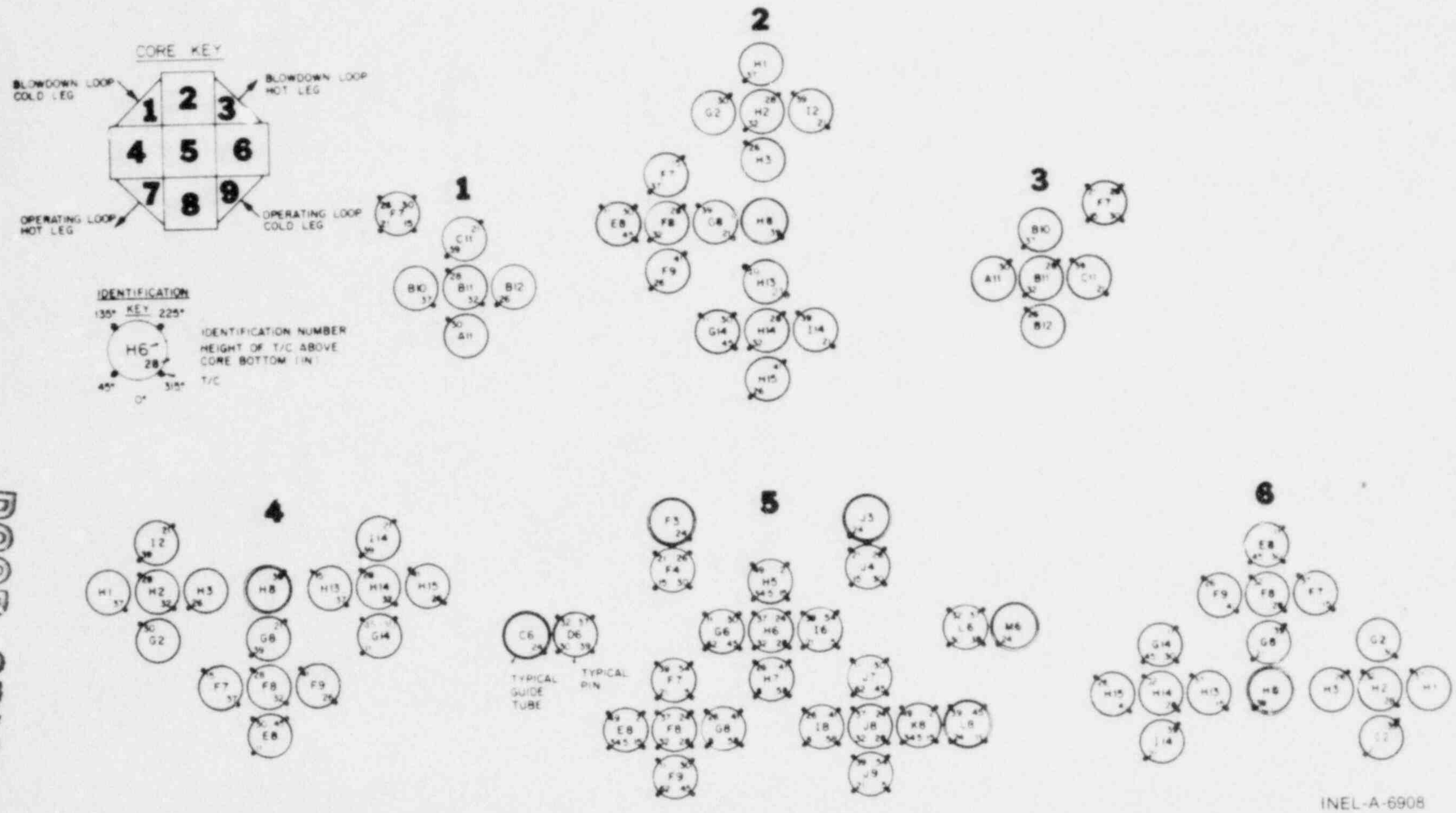

Fig. 24 Axial Positions of LOFT Core Thermocouples.

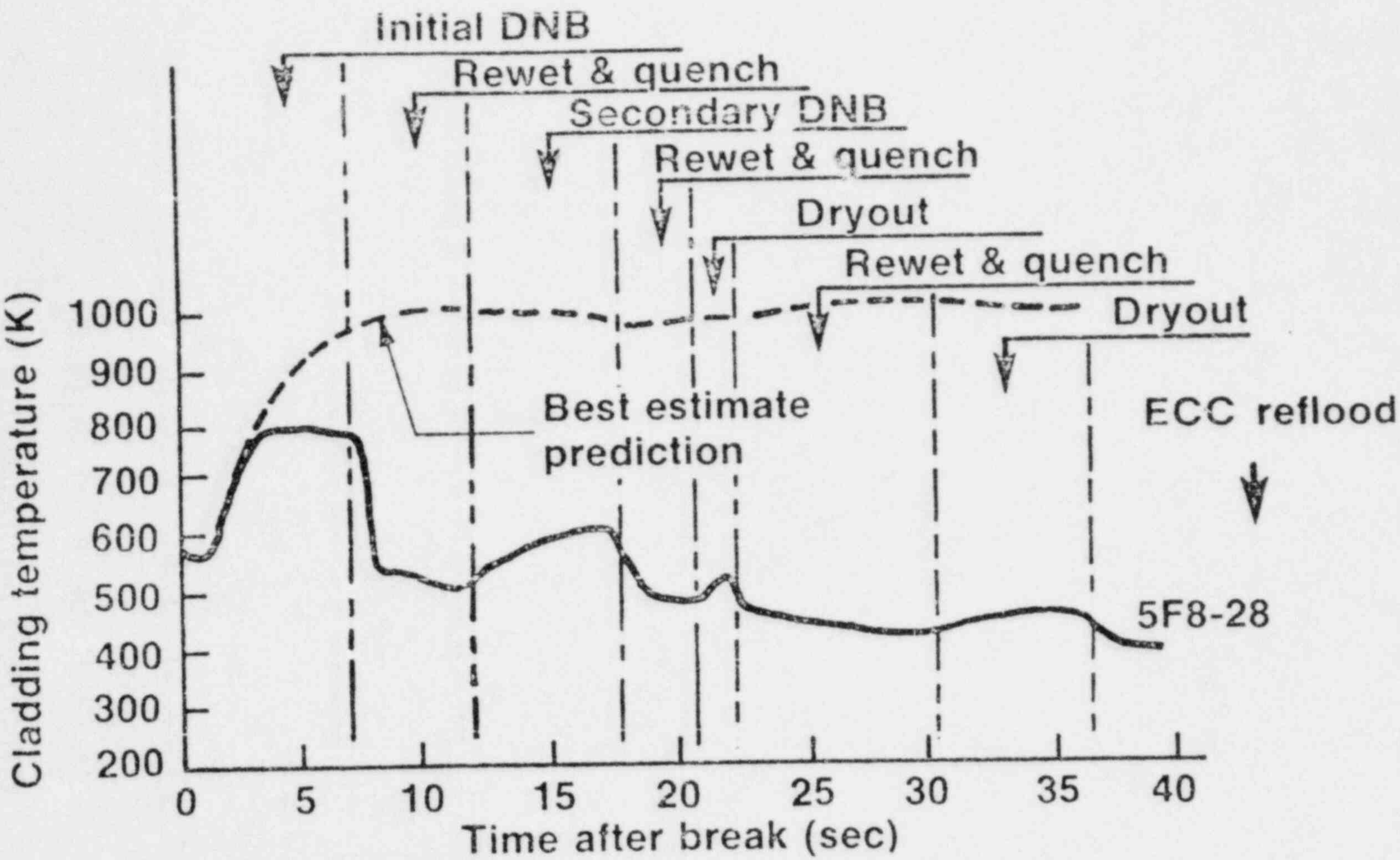

Fig. 25 Measured Cladding Temperature Response for L2-2 (Center Module, Thermocouple 5F8-28).

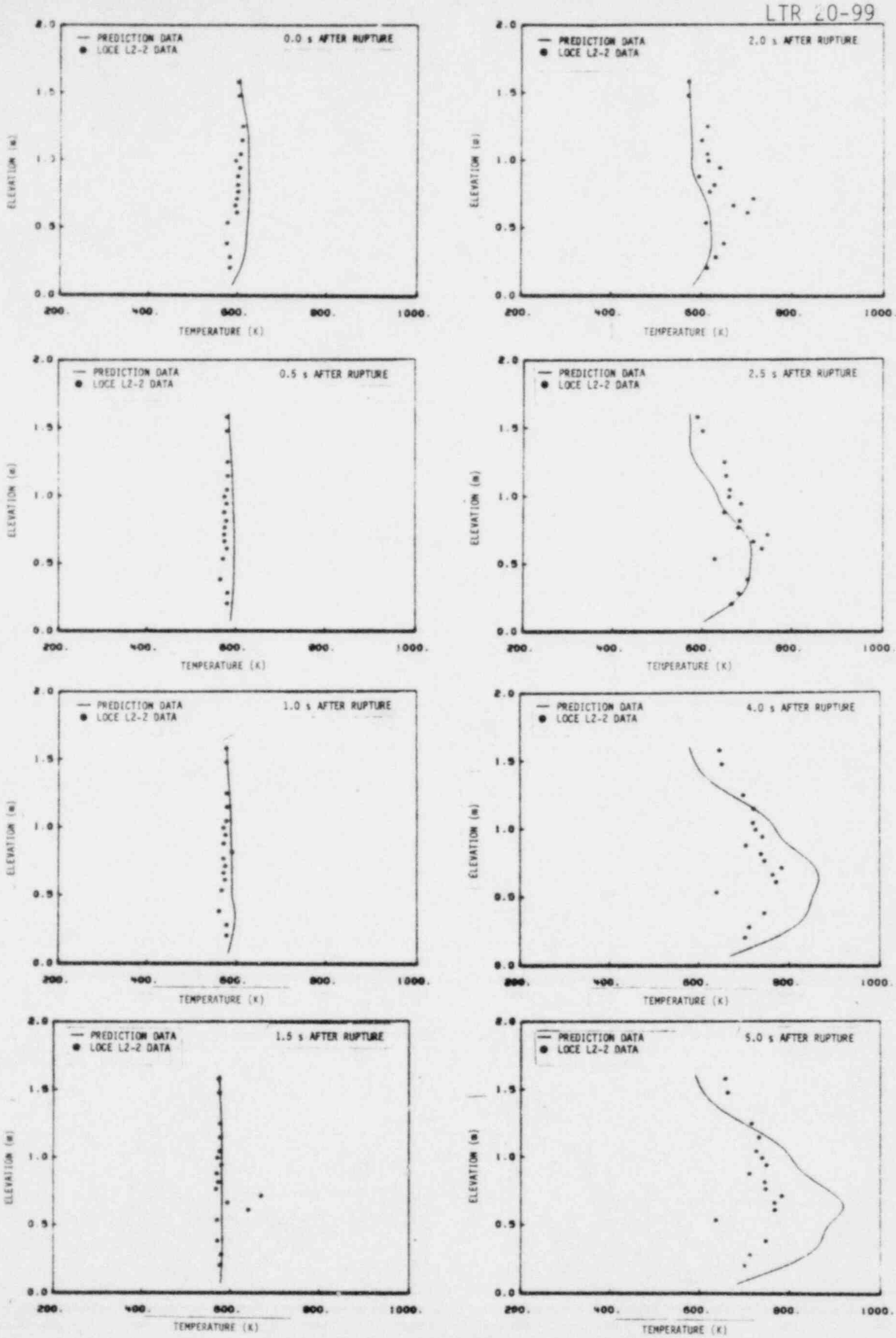

Fig. 26 Initial Boiling Crisis and Rise to Peak Cladding Temperature During 0.0 to 5.0 Seconds after Rupture (Center Module Cluster about Fule Rod 5F8).

illustrate, that almost the entire lengths of the rods experienced a boiling crisis during that time period, and that the boiling crisis began initially at the hot spots on the rod.

TABLE III

INITIAL CONDITIONS FOR LOFT NUCLEAR LOCE L2-2

Parameter	EOS Specified Value(2)	Measured Value
Primary Coolant System		
Mass flow rate (kg/s)(a)	--	194.2
Pressure (MPa)	15.6 ± 0.1	15.64
Temperature (T_h) (K)	587.59 ± 7.2	580.4
Boron concentration (ppm)	As required	838
Cold leg temperature (K)	--	557.7
Reactor Vessel		
Power level (MW)	--	24.88
Maximum linear heat generation rate (kW/m)	26.2	26.37
Control rod position (centimeters above full-in position)	137.2 ± 1.3	137
Pressurizer		
Steam volume (m^3)	--	0.353
Water volume (m^3)	--	0.607
Water temperature (K)	As required to establish pressure	619
Pressure (MPa)	15.6 ± 0.1	15.62
Level (cm)	113 ± 17.8	108.9
Broken Loop		
Hot leg temperature (K)	587.6 $^{+0}_{-444}$	
Near vessel	--	561.2
Near break	--	542.9

TABLE III (Continued)

Parameter	EOS Specified Value(2)	Measured Value
Broken Loop (Continued)		
Cold leg temperature (K)	563.8 $^{+0}_{-14}$	
Near vessel	--	555
Near break	--	538.3
Steam Generator Secondary Side[b]		
Water level (cm)	320	314
Water temperature (K)	--	553
Pressure (MPa)	--	6.35
Mass flow rate (kg/s)	--	12.67
ECC Accumulator A		
Gas volume (m^3)	--	1.05
Water volume injected (m^3)	--	1.68
Pressure (MPa)	4.22 ± 0.17	4.11
Temperature (K)	305.4 ± 8.3	300.8
Boron concentration (ppm)	3100	3301
Liquid level (m)	2.045 ± 0.03	2.01
Suppression Tank		
Liquid level (cm)	127 ± 2.54	135.07(d)
Gas volume (m^3)	--	53.3
Liquid volume (m^3)	--	31.9
Downcomer submergence (cm)[a]	--	48.73
Water temperature (K)	356 ± 3.6	352.0[d]

TABLE III (Continued)

Parameter	EOS Specified Value(2)	Measured Value
Suppression Tank (Continued)		
Pressure (gas space) (MPa)	0.086 $\pm$ 0.007	0.123[d]

(a) Calculated.

(b) Not controlled.

(c) Based on average submergence of four downcomers.

(d) Out of specification but did not affect results.

It has been hypothesized that the boiling crisis and initial rewetting can be attributed to the fluid flow dynamics in the intact loop cold leg and broken loop hot and cold legs, as illustrated in Figures 27 and 28. Figure 27 is the momentum flux measured above Fuel Assembly 1. This figure shows flow stagnation above the core within 0.5 seconds after rupture, which continues until about 2.3 seconds after rupture. Core flow reversal also occurs from essentially time zero, and choking in the broken loop cold leg at about 2 to 4 seconds when conditions at the break point reach saturation. Broken loop cold leg flow choking can be seen in Figure 28, which presents the intact and broken cold leg mass flows. Points A and B for the broken cold leg indicate break flow reduction corresponding to the choking.

Prior to the cold leg choking, guide tube temperature measurements indicate that flow stagnation at the bottom of the core occurred at about 0.8 to 1.0 second, and lasted until about 2.0 seconds. The core stagnation thereby resulting because of inlet and outlet stagnation during the period of time from about 0.5 to 2.0 seconds corresponds to the time when the core experienced the initial boiling crisis. (e.g., axial temperature profiles in Figure 26). This early boiling crisis is probably a heat flux dominated mechanism, such as a departure from nucleate boiling (DNB), since coolant is available, but the capability of the coolant to remove heat has been reduced because of flow stagnation.

Figure 27 indicates a positive core flow beginning at around 2 to 3 seconds. This is also evident in Figure 25, which shows that about this time the temperature history curve slope at the 76-cm elevation for rod 5F8, observably changes. Figure 28 shows that at about 3.8 seconds the intact loop cold leg flow overtakes the broken loop cold leg flow. Another very significant change in cladding temperature slope can be seen corresponding to this time in Figure 25, indicating a further increase in positive core flow.

The increase in core flow resulted in additional rod cooling and eventual rewetting of all the rods in the core. Figure 29 illustrates

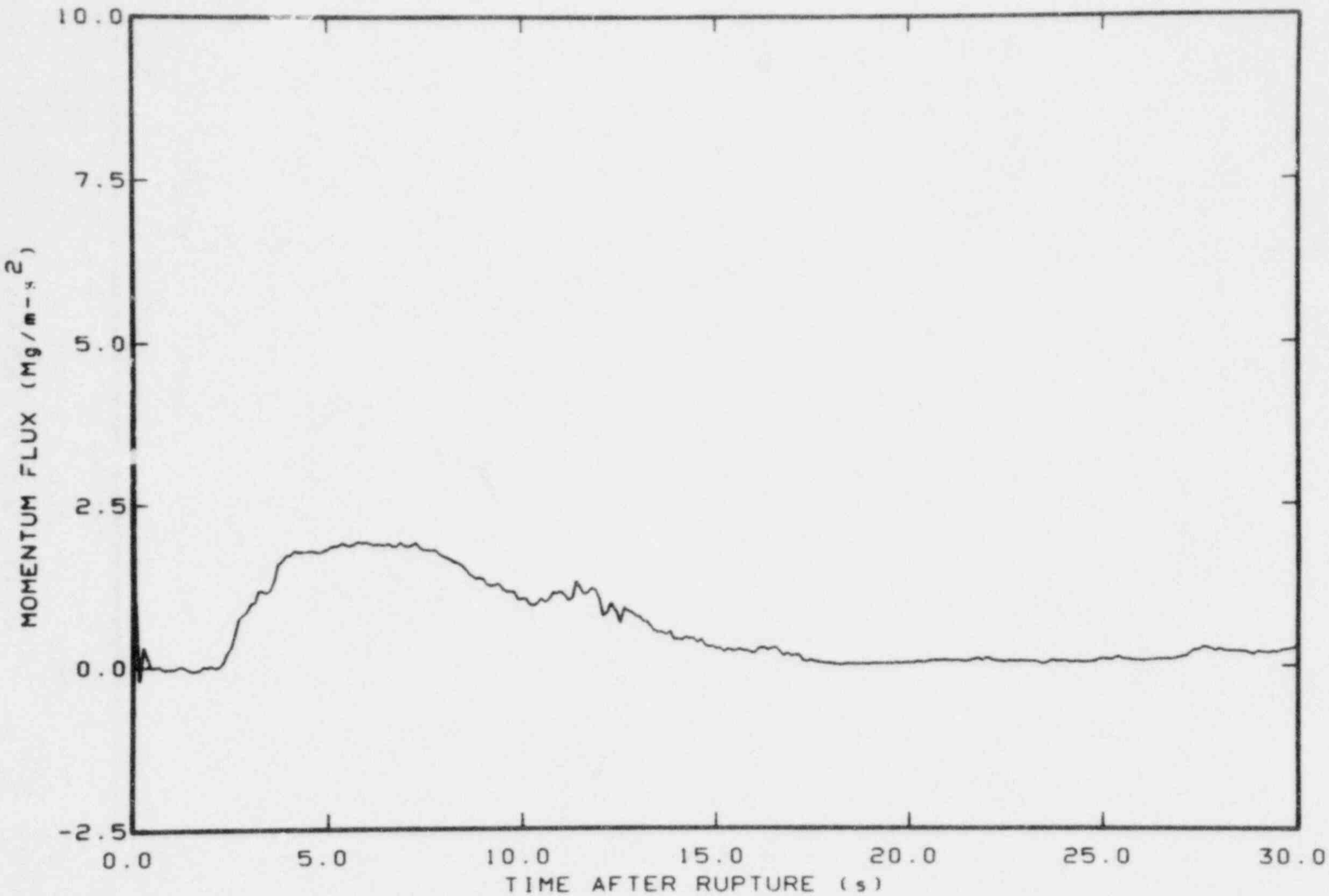

Fig. 27 Momentum Flux above Fuel Assembly 1 during LOFT Test L2-2.

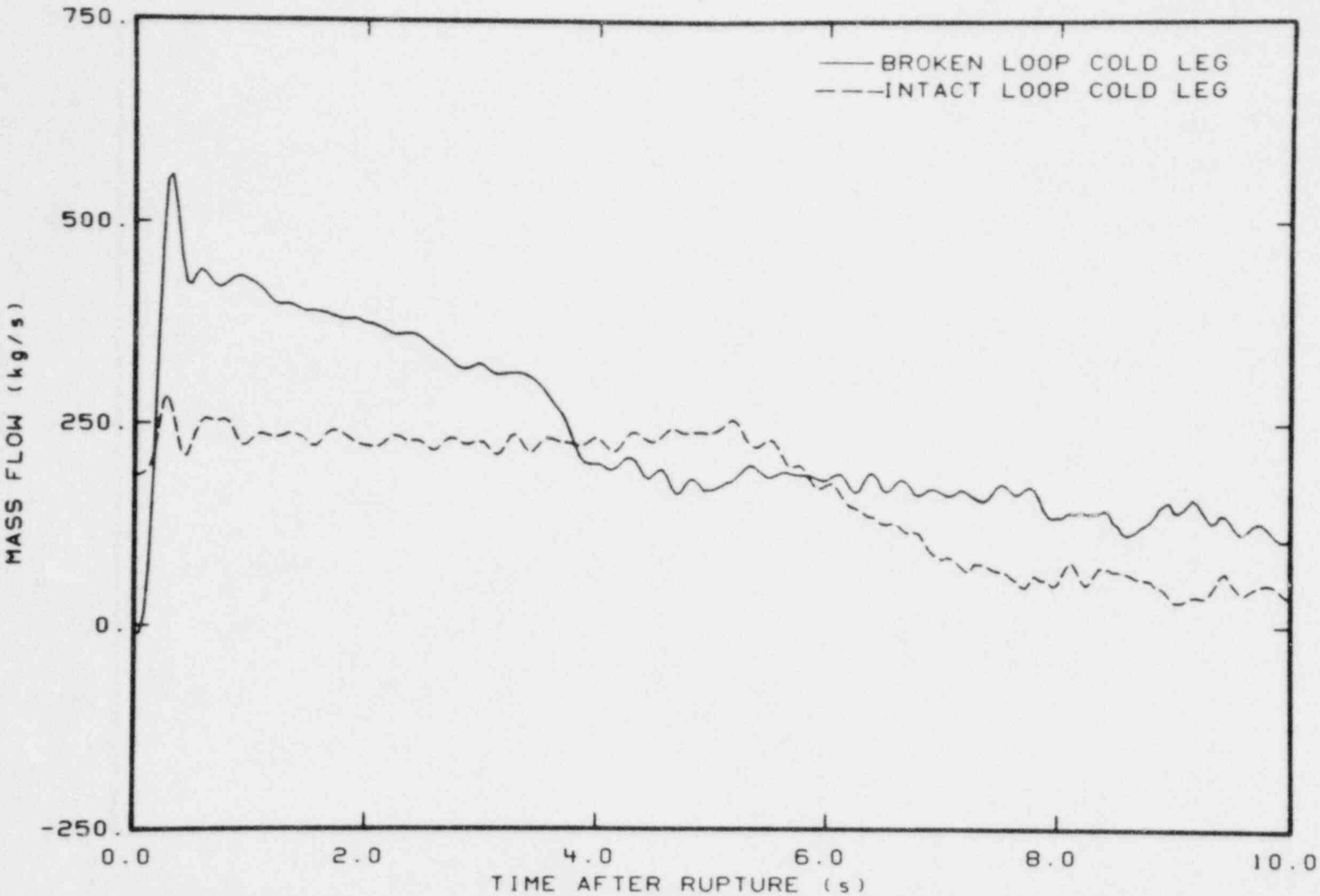

Fig. 28 LOFT L2-2 Mass Flow Rate Histories for the Intact and Broken Loop Cold Legs.

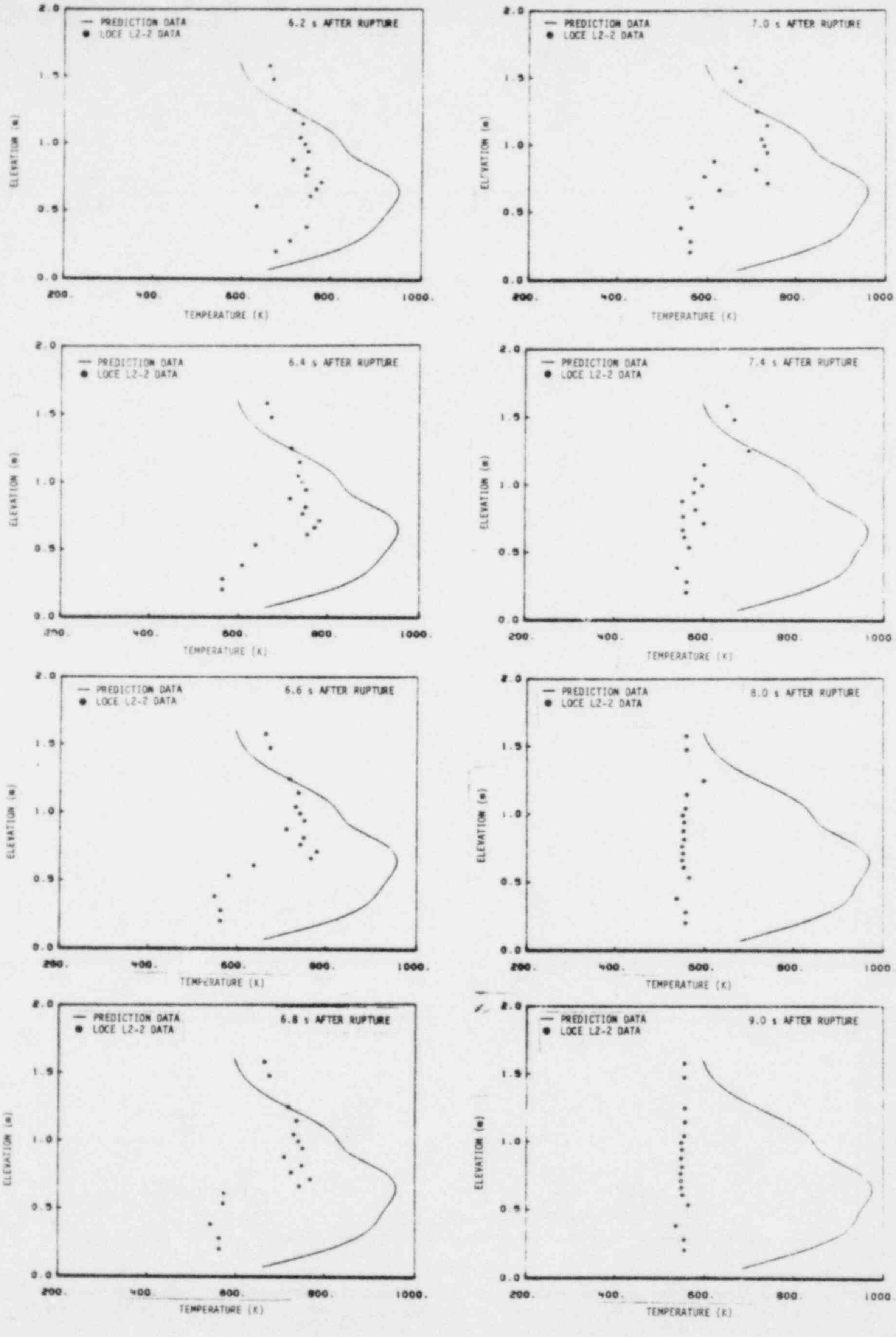

Fig. 29 Initial Rewet (from Bottom) during 6.0 to 9.0 Seconds after Rupture for LOFT Test L2-2 (Center Module Cluster about Fuel Rod 5F8).

the axial quenching history between 6.2 and 9.0 seconds, and demonstrates that the quench proceeded from rod bottom to top. Figure 30 illustrates rod temperature histories from 0.0 to 10.0 seconds for rods in fuel assemblies 4, 5, and 6. As might be expected, the rod in assembly 6, which is closest to the intact loop cold leg (see Figure 23) rewet first, with rewets in assemblies further removed from the intact cold loop occurring later in time.

The rods were quenched for a period of about 4 seconds, after which a boiling crisis occurred for a second time. Figure 28 illustrates that by this time the broken cold leg flow has again overtaken the intact cold leg flow, and Figure 27 illustrates that although core flow is still positive, it is being reduced. By this point in time, the significant coolant depletion has probably precipitated a dryout of the fuel rods. Subsequent rewets (see Figure 25) have been attributed to entrained liquid in the upper plenum regions running down onto the rods. The rods are at a low temperature that would result in rewetting if coolant is available. The final rewet and quench of the rods at about 37 seconds resulted from the reflooding of the core by ECC injection.

3. PBF LOCE DATA

PBF is a test reactor with a right circular cylinder core geometry and a rated operating capacity of 30-MW. An in-pile test tube with coolant flow separate from the core coolant flow passes through the center of the core. The PBF LOC-11 experiment[38,39] was comprised of three tests (Tests A, B, C). For the LOC-11 experiment four fuel rods isolated in separate flow shrouds were placed in the in-pile tube, which had been modified so a cold leg break LOCA could be simulated.

The cladding surface temperature history for one of the rods (all rods behaved essentially the same) in the LOC-11A test is shown in Figure 31. This history shows that a rewet (RNB) occurred during this

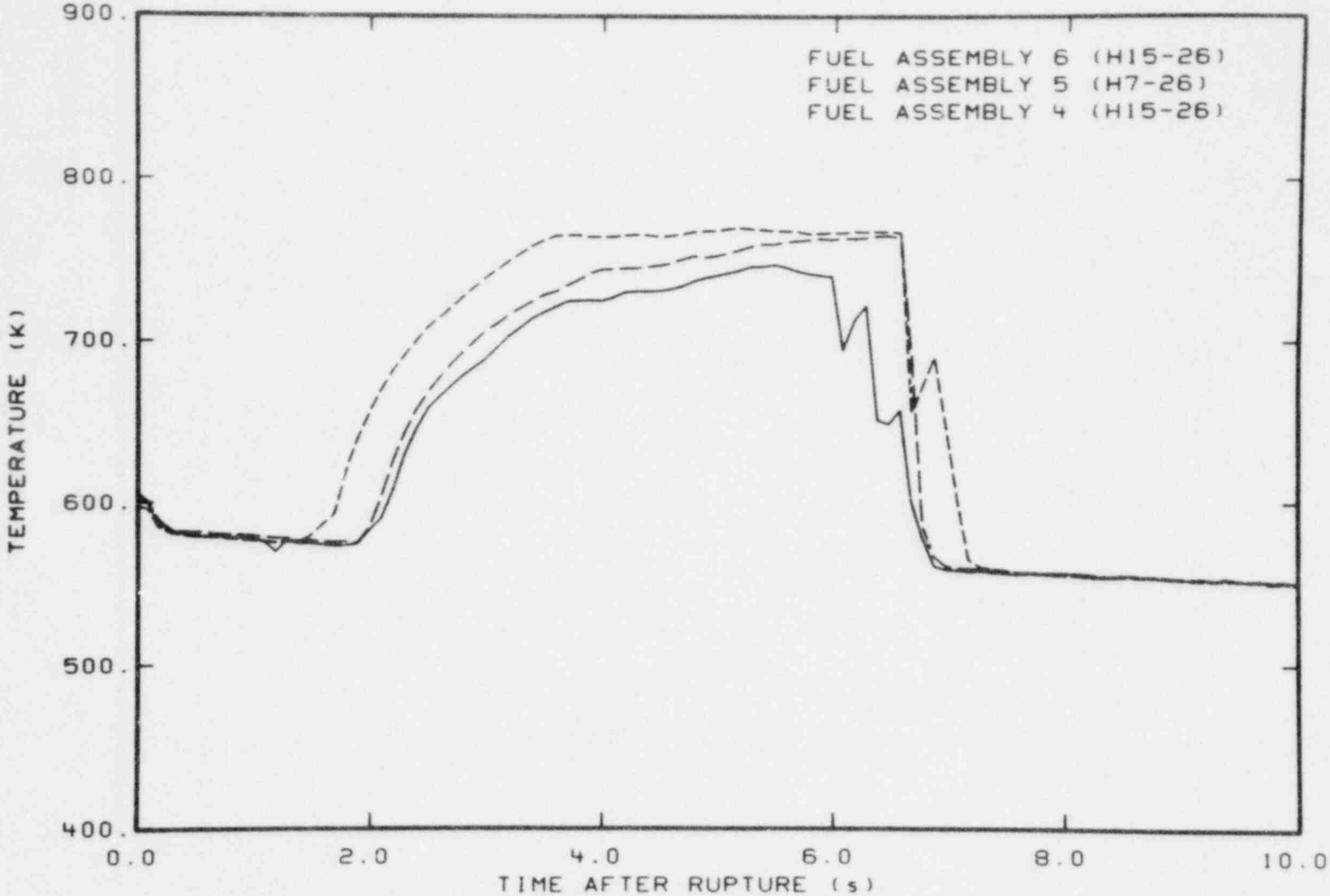

Fig. 30 Peak Cladding Temperature Histories for LOFT Test L2-2 in Fuel Assemblies 4, 5, and 6.

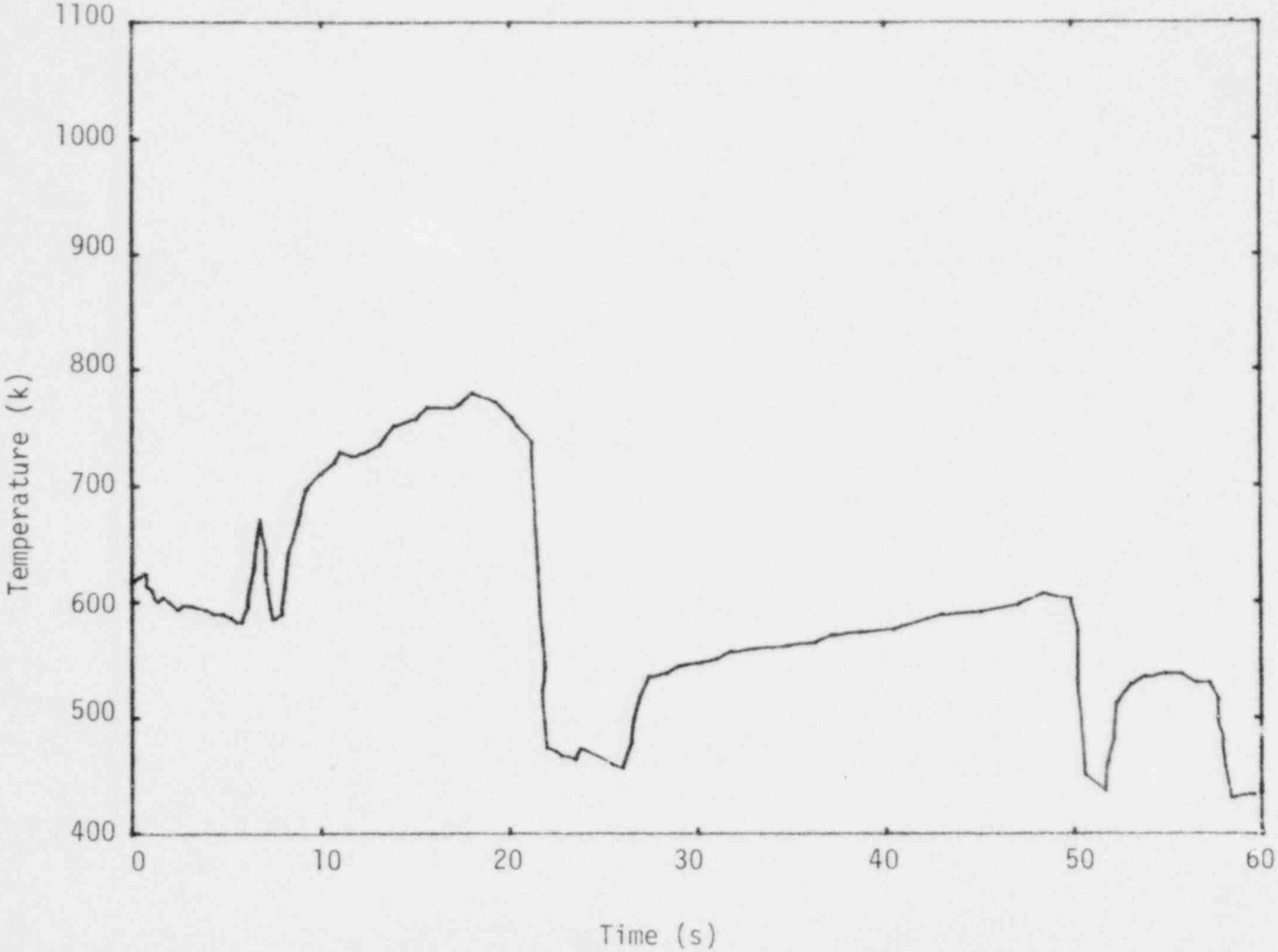

Fig. 31 Cladding Surface Temperature History at the 0.533 m, 0° Elevation for Rod 611-1, PBF Test LOC-11A.

test. This rewet was caused by the cycling of the cold and hot blowdown leg valves and is not necessarily typical of a LOCA however. All four rods experienced the rewet at the same time.

The cladding surface temperature history for one of the rods in the LOC-11C test is shown in Figure 32. The blowdown valves had been repaired prior to this test, and no rewets were observed in this test.

The limited amount of data from LOC-11 does not make it possible to draw any conclusions with regard to RNB during blowdowns with nuclear fuel rods. Obtaining additional nuclear LOCE data is imperative, since it is not conclusive that nuclear fuel rods will have the same rewetting characteristics as electrical heated rods during a blowdown because of material property differences and differences in the mode of powering the rods.

Another question that remains to be resolved because of the lack of nuclear data is whether anomalies in nuclear fuel rods, such as asymmetric pellet stacking or gap width, would produce the differences in RNB behavior from rod to rod for similar coolant conditions as was seen with the Semiscale heater rods. This question is a small part of the larger question of how well electrical heated rods simulate a nuclear fuel rod in any kind of test.

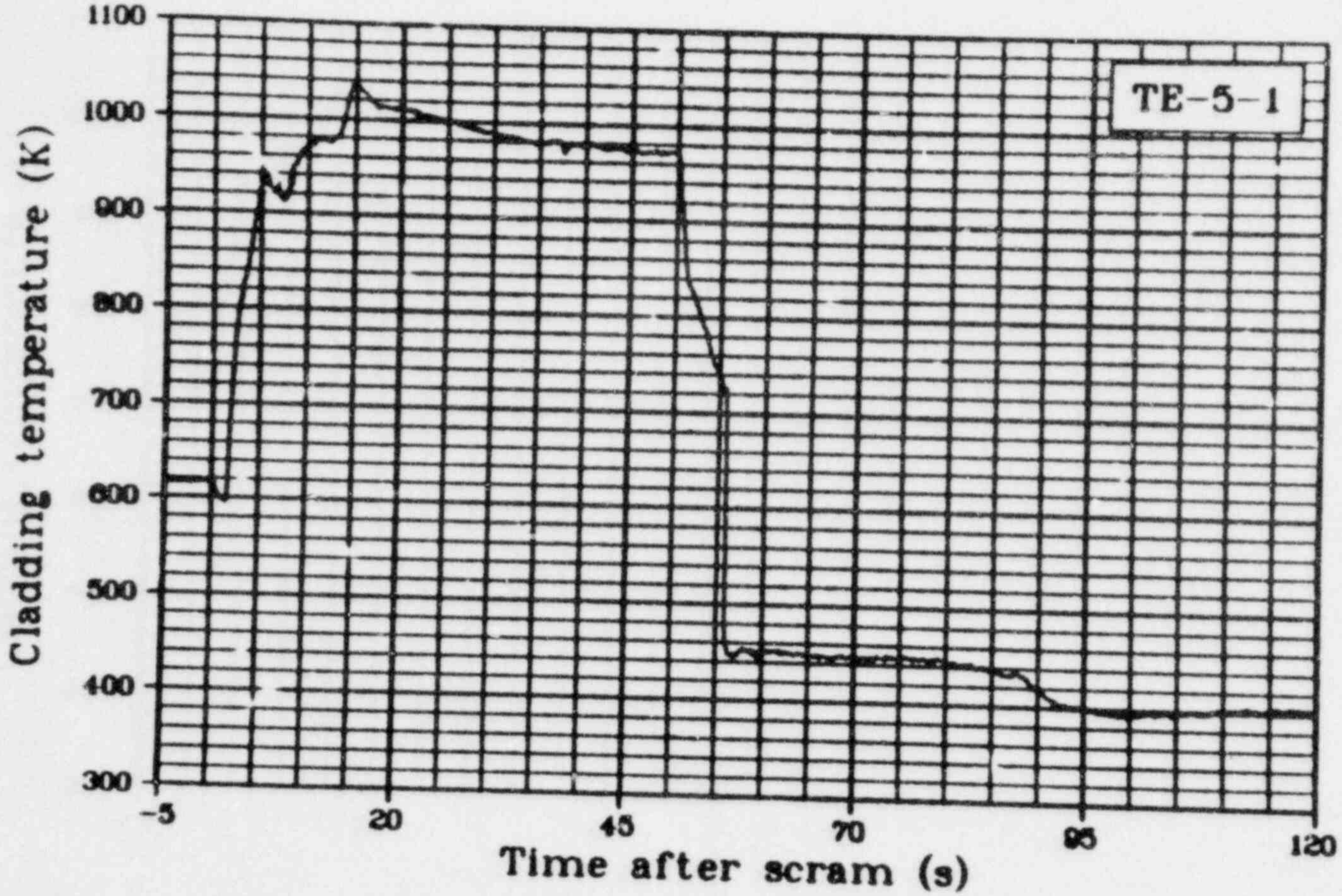

Fig. 32 Cladding Surface Temperature History at 0.533 m, 0° Elevation for Rod 611-1, PBF Test LOC-11C.

IV. CONCLUSIONS

The occurrence of RNB and rod rewetting during the blowdown phase of a LOCE after the critical heat flux has been exceeded has been clearly demonstrated in the Semiscale tests and LOFT L2-2 test performed at INEL. Rewetting has significant importance because it serves as a mechanism which can reduce the peak cladding temperature experienced by a rod during a LOCE. At the present time the conservative assumptions present in the Evaluation Model Requirements of Appendix K, 10 CFR 50 do not permit the consideration of RNB after the critical heat flux has been exceeded, and the use of transition boiling (which encompasses rewetting) cannot be employed once the surface temperature exceeds the saturation temperature by 167 K.

The characterization of rewetting is essentially a problem of adequately describing (in a quantitative manner) the transition boiling regime which represents the bridge between nucleate boiling (essentially 100% surface wetting by liquid)and film boiling (essentially 0% surface wetting by liquid). The transition boiling regime is bounded by the critical heat flux and the minimum film boiling heat flux. An upper bound minimum film boiling temperature (surface temperature at the minimum film boiling heat flux) can be estimated based on thermodynamic considerations, but the lack of reliable data and inconsistency between existing transition boiling correlations for forced convection two-phase flow severely limits any attempts to reliably predict rewetting or RNB for a hot surface in two-phase forced convection flow.

The data trends evident in the Semiscale data are consistent with information obtained in other rewetting tests, and suggests many factors that influence RNB. The rewetting phenomena were found to be affected by interaction between local rod power densities and local coolant vapor flow. The dependence of rewetting on these two factors was clearly manifested in the axial distribution of RNB. Factors influencing these two variables, such as unpowered adjacent rods, cold walls, differences in local power, or nozzle geometry affecting the coolant break flow, could significantly influence the occurrence of

RNB and were evident in changes in the axial distribution of rewets when these parameters were changed. While no radial pattern of RNB was evident, the interplay of possible cold wall effects and inherent rod differences could have masked a radial dependence that might have been present if all the heater rods were identical. It appears that thermal hydraulic parameters such as local qualities and power densities are dominating factors, with individual rod differences coming into play when the quality and power density borderline between RNB or non-RNB conditions. The data were found to be very repeatable with respect to individual rods for similar conditions.

The dramatic rewetting and RNB observed in LOFT L2-2 graphically demonstrates the viability of rewetting as a cooling mechanism in a LOCE blowdown, and the significant impact this phenomenon can have on peak cladding temperatures. The rewetting of the rods can be correlated with causal thermal-hydraulic events in the intact and broken loop cold legs, and the uniformity of the core response to the rewetting suggests that the RNB phenomenon is not an atypical or isolated occurrence, but a phenomenon resulting from the interaction of fuel rods with major thermal hydraulic conditions. Further tests are needed to determine if individual rod characteristics may influence rewetting in the same manner as seen in the Semiscale tests.

The Semiscale and LOFT L2-2 data have demonstrated that RNB is a realistic blowdown phenomenon and the elimination of any consideration of rewetting of the cladding surface or RNB during blowdown is distinctly conservative in Appendix K Evaluation Models. However, to be of practical significance in the safety analysis of nuclear reactors, reliable calculational models must be developed which can adequately predict rewetting and describe transition boiling based on calculated blowdown thermal-hydraulic conditions. The development of such models will require significant analytical and experimental research to correlate significant parameters and identify two-phase flow transition boiling mechanisms.

The development of rewetting models will also need to address the question of whether individual rod characteristics play an important role in rewetting, as seen in the Semiscale tests. The acquisition of further nuclear LOCE data is especially crucial, since it is important to discover if anomalies in nuclear fuel rod construction can influence a rod's rewetting characteristics when thermal-hydraulic states border between rewetting and non-rewetting conditions (unlike LOFT L2-2 where a hydraulic event clearly dominated as the rewetting mechanism).

In summary, the Semiscale, LOFT and PBF LOCE data have demonstrated that assumptions in Evaluation Model Requirements with regard to RNB and rewetting are conservative in nature. The demonstrated occurrence of rewetting during blowdown and subsequent reductions in peak cladding temperatures reached by the fuel rods during the LOCE transient and the lack of adequate experimental data and analytical correlations to describe the transition boiling regime encompassing this phenomenon emphasizes the need for further research in this area. To be of practical significance in the safety analysis of nuclear reactors, reliable calculational models must be developed that can adequately predict rewetting and characterize transition boiling based on conditions calculated to occur during a blowdown.

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