INTERIM REPORT

Accession No. ____________

Contract Program or Project Title: Water Reactor Safety Research

Subject of this Document: Heat Transfer Highlights - May 1979

Type of Document: Monthly Report

Author(s): J. C. M. Leung, and R. E. Henry

Date of Document: May 1979

Responsible NRC Individual and NRC Office or Division:

L. B. Thompson and Y. Y. Hsu, 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.

Reactor Analysis and Safety Division
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, Illinois 60439

Prepared for
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

INTERIM REPORT

7910020296

NRC Research and Technical Assistance Report

1072 154

Enclosure 3

Water Reactor Safety Research

Heat Transfer Highlights

May 1979

NOTICE

Reactor Analysis and Safety Division
ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

Transient Critical Heat Flux (J.C.M. Leung and K.A. Gallivan)

(1) Further verification of transient thermal-hydraulic code

The present scheme named CODA which neglects the acoustic phenomena as in COBRA-IV-1,[1] assumes that density (ρ) can be evaluated as a function of enthalpy (h) only. Essentially this assumption decouples the momentum equation from the continuity and energy equations. Currently homogeneous equilibrium flow is assumed, which is a valid approximation particularly for high-pressure systems such as PWRs. An integral momentum equation is used,[2]

$$\frac{d\overline{G}}{dt} = \frac{1}{L} (\Delta P - F) \tag{1}$$

where

$$\overline{G} = \frac{1}{L} \int_o^L G \, dz \ ,$$

$$F = \left(\frac{G^2}{\rho}\right)_{out} - \left(\frac{G^2}{\rho}\right)_{in} + \int_o^L \frac{2f}{D\rho} G|G| dz + \int_o^L \rho g \, dz \ ,$$

and the other variables have the usual meaning. This technique implies that pressure is eliminated as an unknown in the momentum equation and is simply integrated across specified volume boundaries. The continuity and energy equations are written as

$$\frac{\partial \rho}{\partial t} + \frac{\partial G}{\partial z} = 0 \tag{2}$$

$$\frac{\partial h}{\partial t} + \left(\frac{G}{\rho}\right) \frac{\partial h}{\partial z} + \frac{\phi P_h}{\rho A_x} + \left(\frac{1}{\rho}\right) \frac{\partial P}{\partial t} \tag{3}$$

1072 156

The finite-difference form of the equations (1, 2, and 3) was integrated in stepwise manner using a predictor-corrector scheme.

In this monthly report, the present scheme is compared against two existing codes as a further verification run. The standard test problem No. 1 as proposed by Hanco et al.[3] considers an instantaneous heat addition to water flowing in a long pipe. Inlet and exit pressures are fixed at 7.102 MPa and 6.984 MPa, respectively, during the transient. The results of CODA are compared with MECA,[4] a method of characteristics procedure and RAMA,[3] characteristic finite difference scheme. Figure 1 shows close agreement in predicted mass flowrate between CODA and MECA whereas RAMA appears to smear out much of the detailed features of the solution. CODA predicts a slightly lower steady-state mass flowrate beyond 5 sec and this is felt to be caused by the use of a constant friction factor[5] for the two-phase fluid (f = 0.005) in CODA as opposed to the more complicated formula in MECA and RAMA codes.[3] Figure 3 shows good agreement in enthalpy history between CODA and MECA; the slightly higher exit enthalpy beyond 5 sec in CODA is a direct result of the smaller predicted mass flow.

In conclusion, the present scheme has been demonstrated to be comparable to existing codes but presents itself as a simpler and efficient tool in the analysis of small-scale experiments.

(2) Analysis of Transient CHF Data

a. ANL Flow Transients

The first series consists of fifteen 4%/sec flow coastdown tests which were intended to simulate reactor-coolant-system pump trip accidents.[6] Power remained constant to the uniformly heated tube during the transient until being tripped by an overheating protective circuit. The results using the local-condition CHF prediction are reported in the last monthly report[7] but the times-to-CHF plot are reproduced here as Fig. 3 for comparison purposes. The

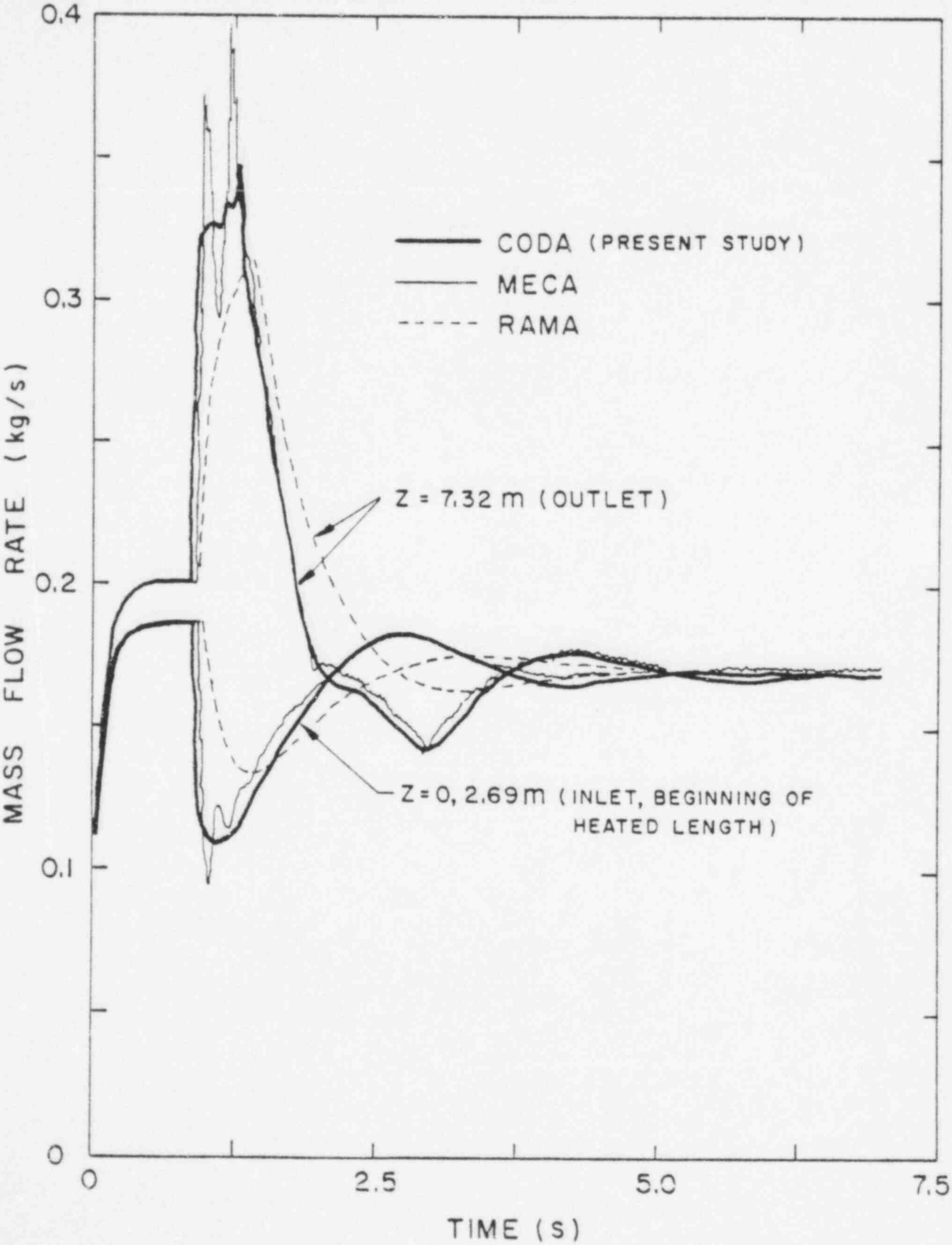

Fig. 1. Comparison of mass flow rate for Hancox problem.

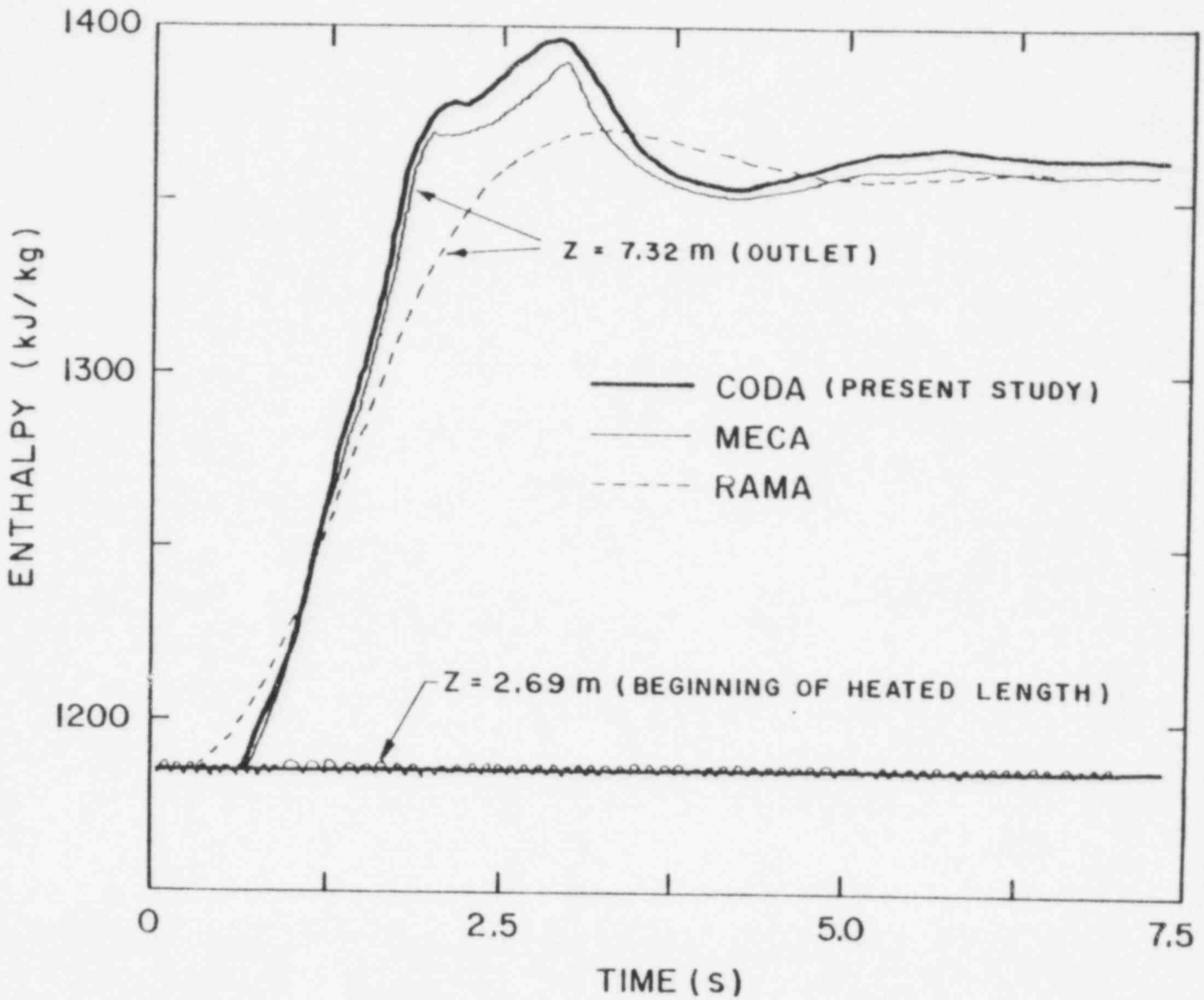

Fig. 2. Comparison of enthalpy for Hancox problem.

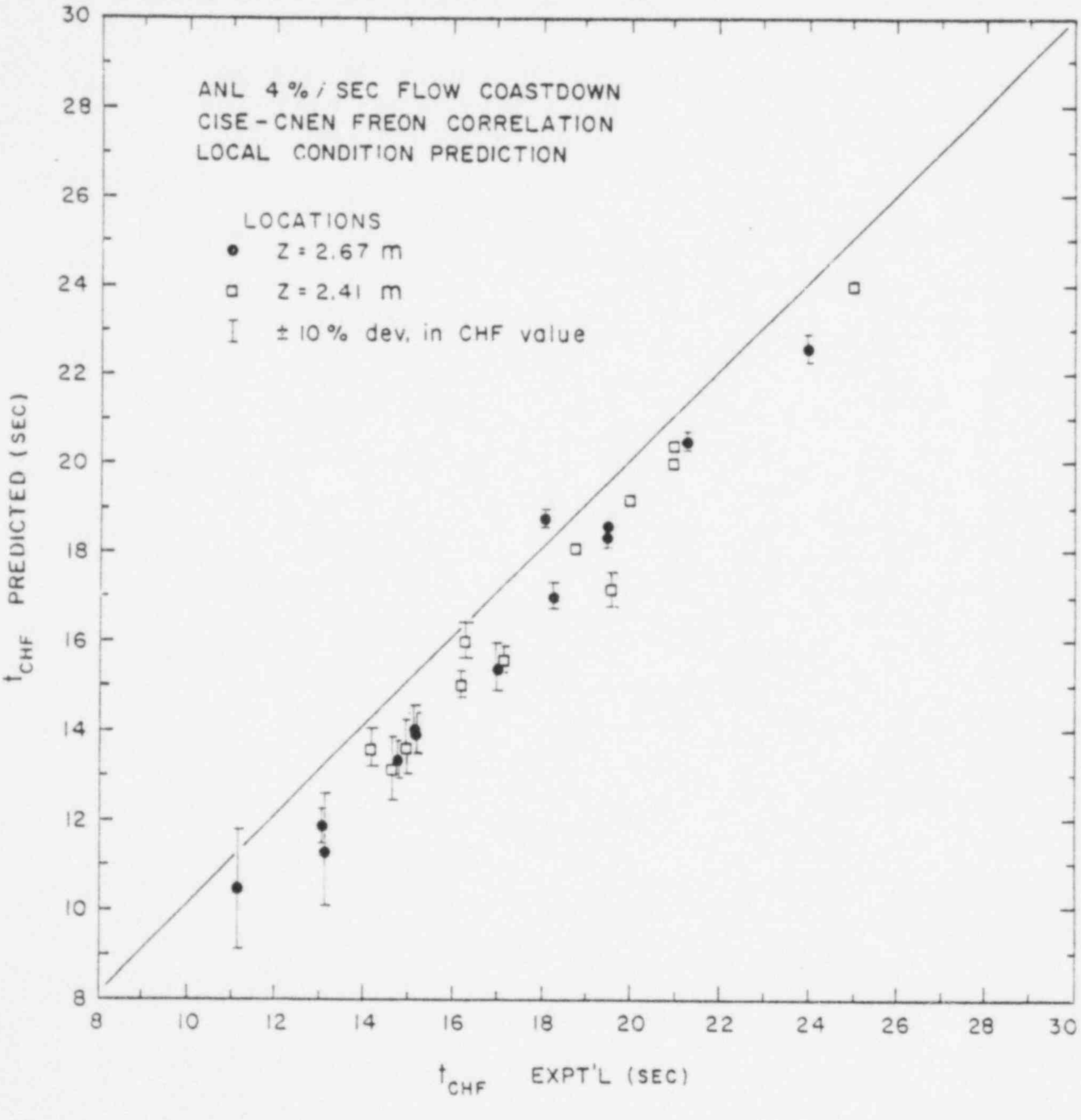

Fig. 3. Prediction of CHF using local-condition hypothesis in 4%/sec flow decay tests.

1072 160

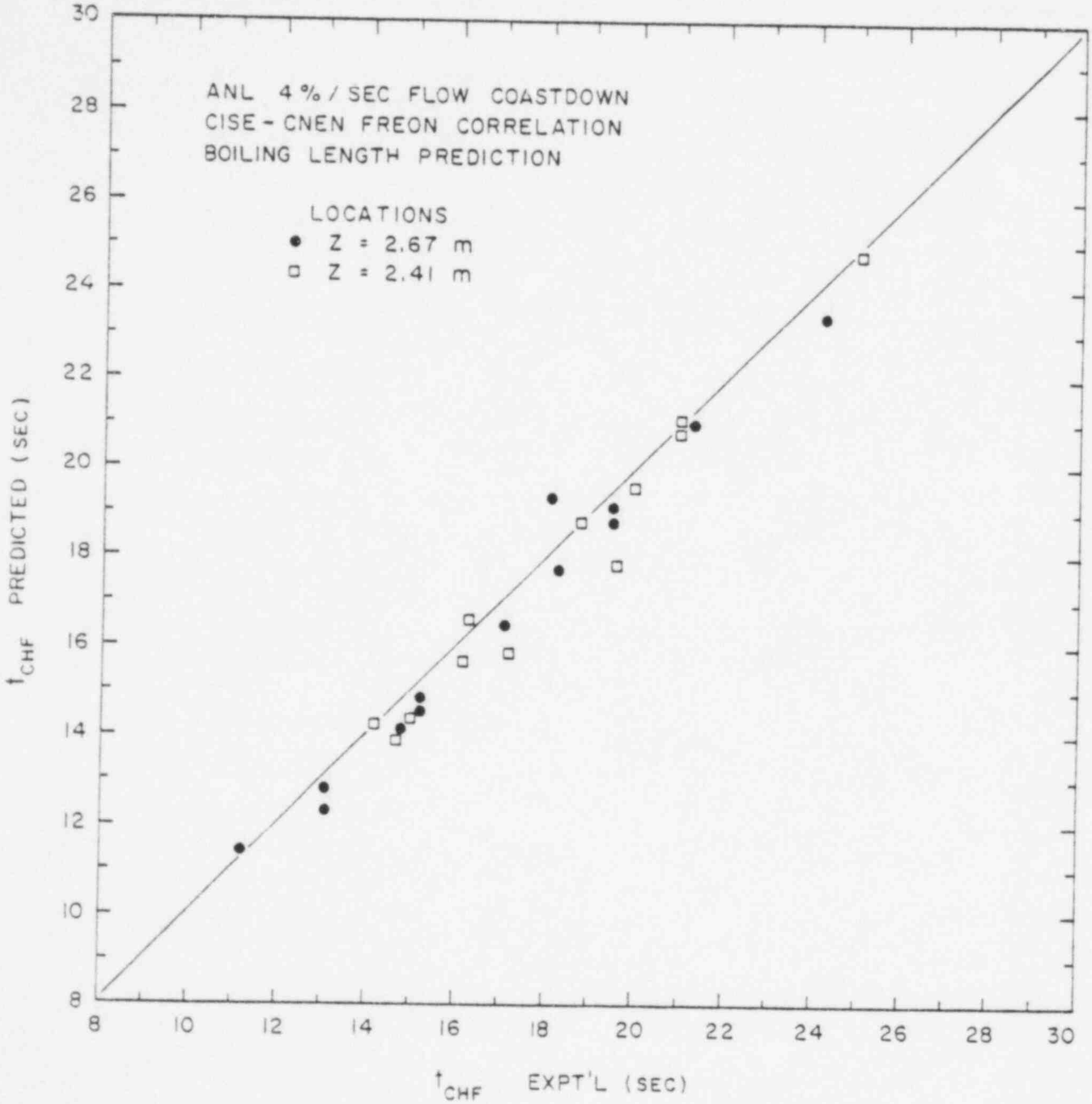

Fig. 4. Prediction of CHF using boiling-length hypothesis in 4%/sec flow decay tests.

predicted times appear to be slightly conservative, in the order of 1 sec. Another well-known CHF hypothesis is the boiling-length or hydrodynamic-condition hypothesis,

$$x_{CHF} = x(L_b, G, P) \tag{4}$$

where L_b is the boiling length. This hypothesis has been shown to yield better prediction of steady-state CHF data than the local-condition hypothesis[8,9] for nonuniform heat flux data. In the present prediction of transient CHF, the boiling length is taken to be the instantaneous saturated two-phase region in the tube. The results of the boiling-length analysis is shown in Fig. 4 where better agreement is observed.

The second series consists of five inlet-flow blockage experiments conducted in an outlet-peaking test section. Table 1 lists the measured times to CHF at the last three heat-flux zones as well as the predicted times using both hypotheses. It is of interest to note that dryout occurred first in the second last zone (zone D) and this predicted by the local-condition hypothesis in this case leads to a better prediction in terms of times to CHF and its propagation. The results are summarized graphically in Fig. 5.

b. ANL Combined Flow and Pressure Transients

These are exit-break blowdown experiments conducted in both chopped-cosine and inlet-peaking heat-flux test sections. The rapid depressurization of the heated section is accompanied by an inlet-flow stoppage under constant power input. The prediction of CHF using the local condition hypothesis for Tests DB-170 and DB-315 is shown in Figs. 6 and 7, respectively. In general good agreement is obtained.

c. Combustion Engineering (CE) Double-Ended Break Experiment

CE conducted a number of blowdown tests in a uniformly heated tube from initial pressures typical of PWRs.[10] One such test ST022 which has been

TABLE 1. COMPARISON OF TIME-TO-CHF FOR FLOW TRANSIENTS IN AN OUTLET PEAKING TEST SECTION

	Zone C $z = 2.36$ m $\phi/\bar{\phi} = 1.35$ t_{CHF} (sec)			Zone D $z = 2.57$ m $\phi/\bar{\phi} = 1.16$ t_{CHF} (sec)			Zone E $z = 2.67$ m $\phi/\bar{\phi} = 0.81$ t_{CHF} (sec)		
RUN ID	EXP	LC	BL	EXP	LC	BL	EXP	LC	BL
4231	4.3	4.0	4.6	3.7	3.6	4.0	4.6	3.9	3.9
4232	3.5	3.3	3.7	2.8	3.0	3.4	3.9	3.2	3.2
4241	4.9	4.9	>5.0	4.2	4.3	4.8	5.1	4.4	4.6
4242	3.4	3.8	>4.0	2.3	2.6	3.4	2.6	2.8	3.2
4243	1.9	1.8	2.6	1.4	1.2	1.6	1.8	1.5	1.5

Note: $\phi/\bar{\phi}$ = local heat flux to average heat flux

EXP = experimental measurement

LC = local-condition prediction

BL = boiling-length prediction

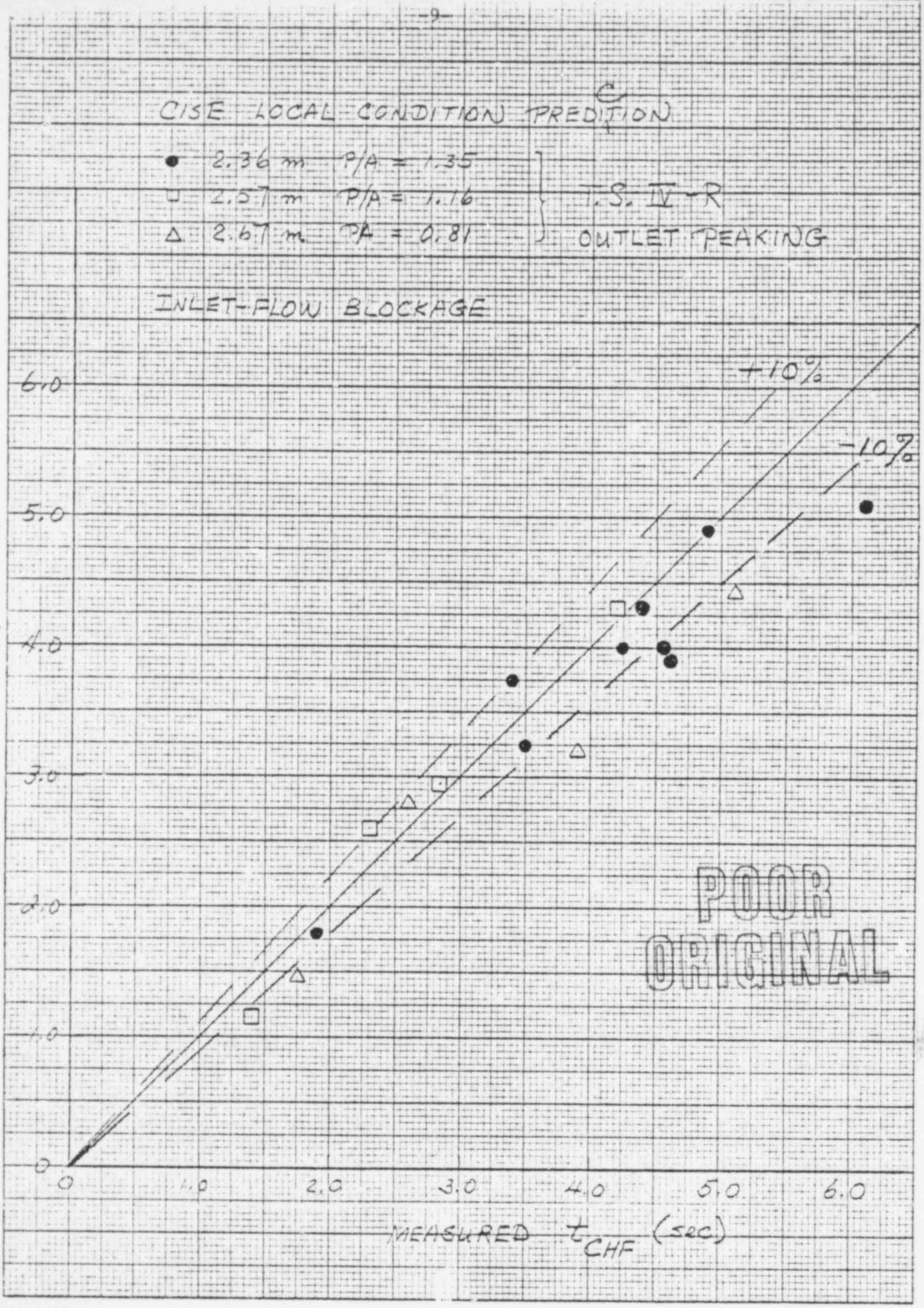

Fig. 5. Prediction of CHF using local-condition hypothesis in inlet flow stoppage tests.

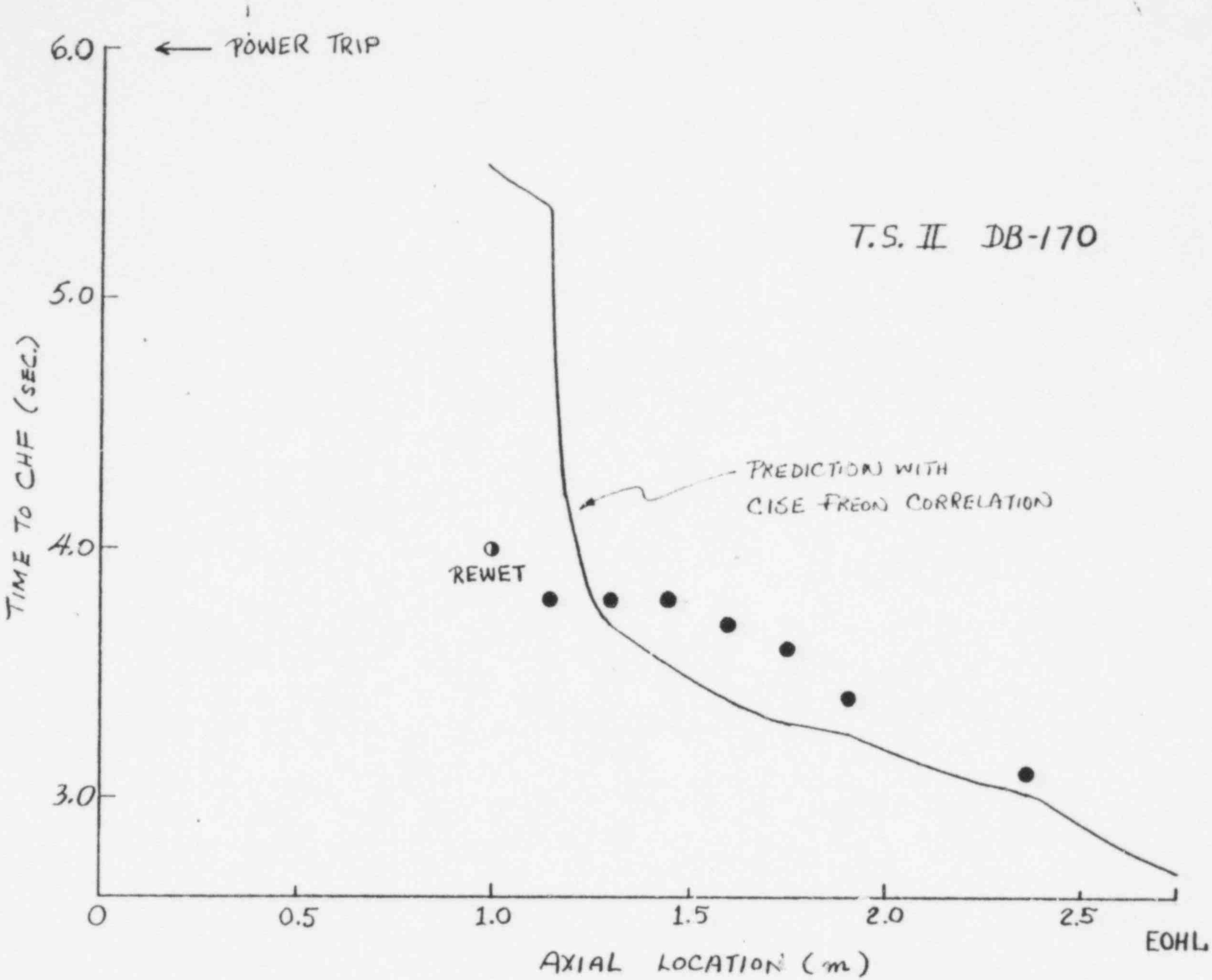

Fig. 6. Prediction of CHF during the outlet break test DB-170.

1072 165

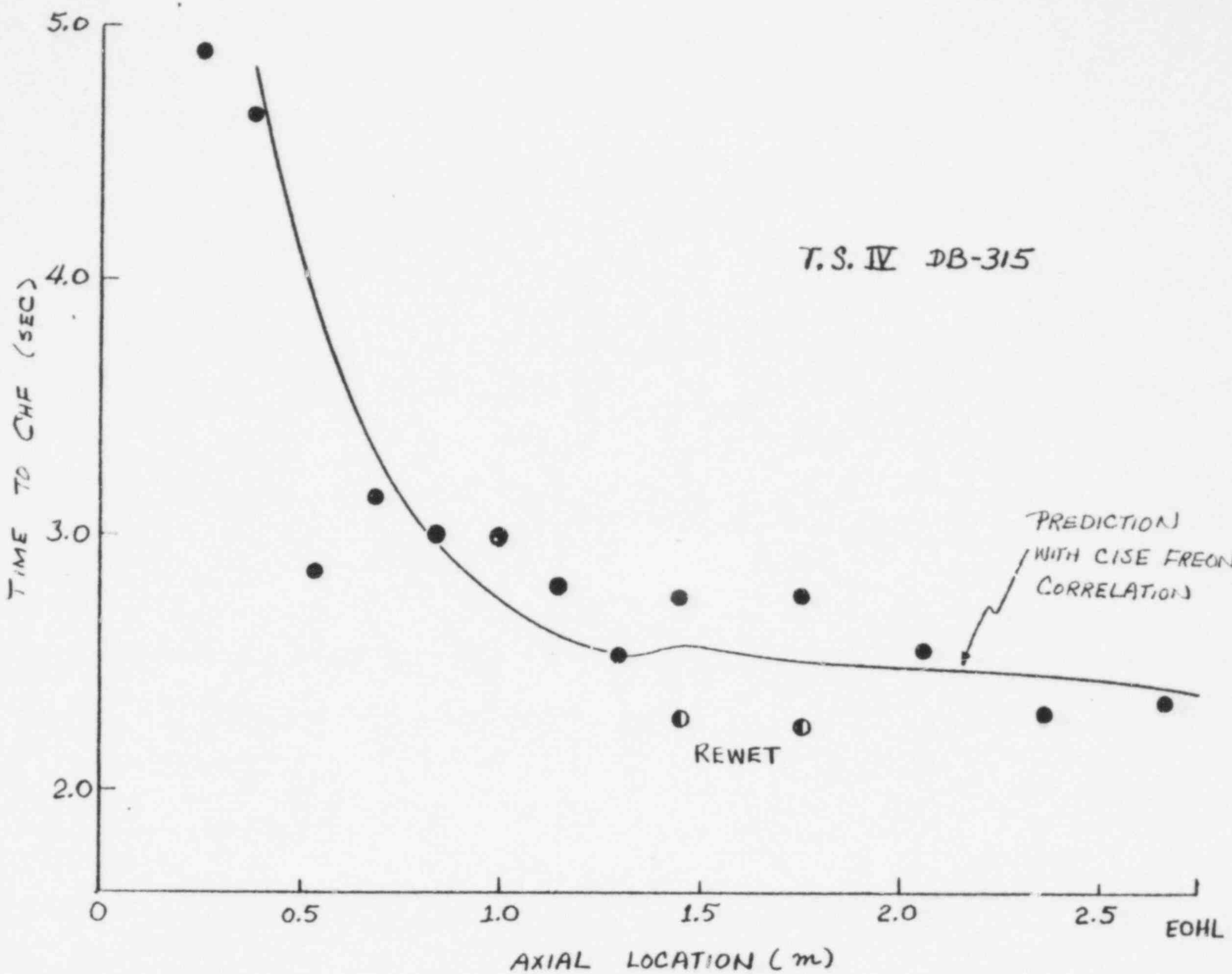

Fig. 7. Prediction of CHF during the outlet break test DB-315

1072 166

-11-

reported in Ref. 11 is taken for analysis. The initial conditions are 16.0 MPa, inlet temperature of 296°C, mass velocity of 3530 kg/m^2s, and test-section heat flux of 461 kW/m^2. The inlet and exit breaks are governed by orifices, their respective diameters being 0.326 cm and 0.250 cm. The reported test-section pressure drop was first used as input boundary condition, yielding the inlet mass velocity as shown in Fig. 8. The measured mass velocity (combination of densitometer and turbine meter) is also shown for comparison. Although there is good agreement in the maximum reversed flowrate, the predicted value lags behind generally. Similar results have been reported for the code.[12] However, it is felt in the present case that flow reversal in the inlet portion of the test section should not take about 300 ms as predicted by the T.S. ΔP. Therefore, the measured inlet mass velocity was subsequently employed in the input boundary condition. The predictions for CHF onset are shown in Fig. 9, and the Biasi correlation is seen to predict CHF onset well but underestimates the region in CHF, whereas CISE correlation is able to predict the region in CHF although it underestimates the time of crisis slightly. In generality, the CHF is a direct result of near depletion of liquid in test section.

d. ORNL THTF Bundle Test 181

THTF Test 181[13] was a mild transient with a 20% inlet break and 45 powered rods at 37.5 kW/rod (relatively low power). The initial pressure of 14.2 MPa dropped rapidly to about 6.9 MPa and remained at such a level for a few seconds. Wall thermocouples did not exhibit and rapid cooling early in the transient and so constant heat-flux values were used in the analysis. Also a complete mixing approach is taken in the calculation of thermal-hydraulic quantities in the core in spite of the presence of four unheated rods. The prediction was performed using the measured pressure history in the core and the measured flowrate in the vertical inlet spool

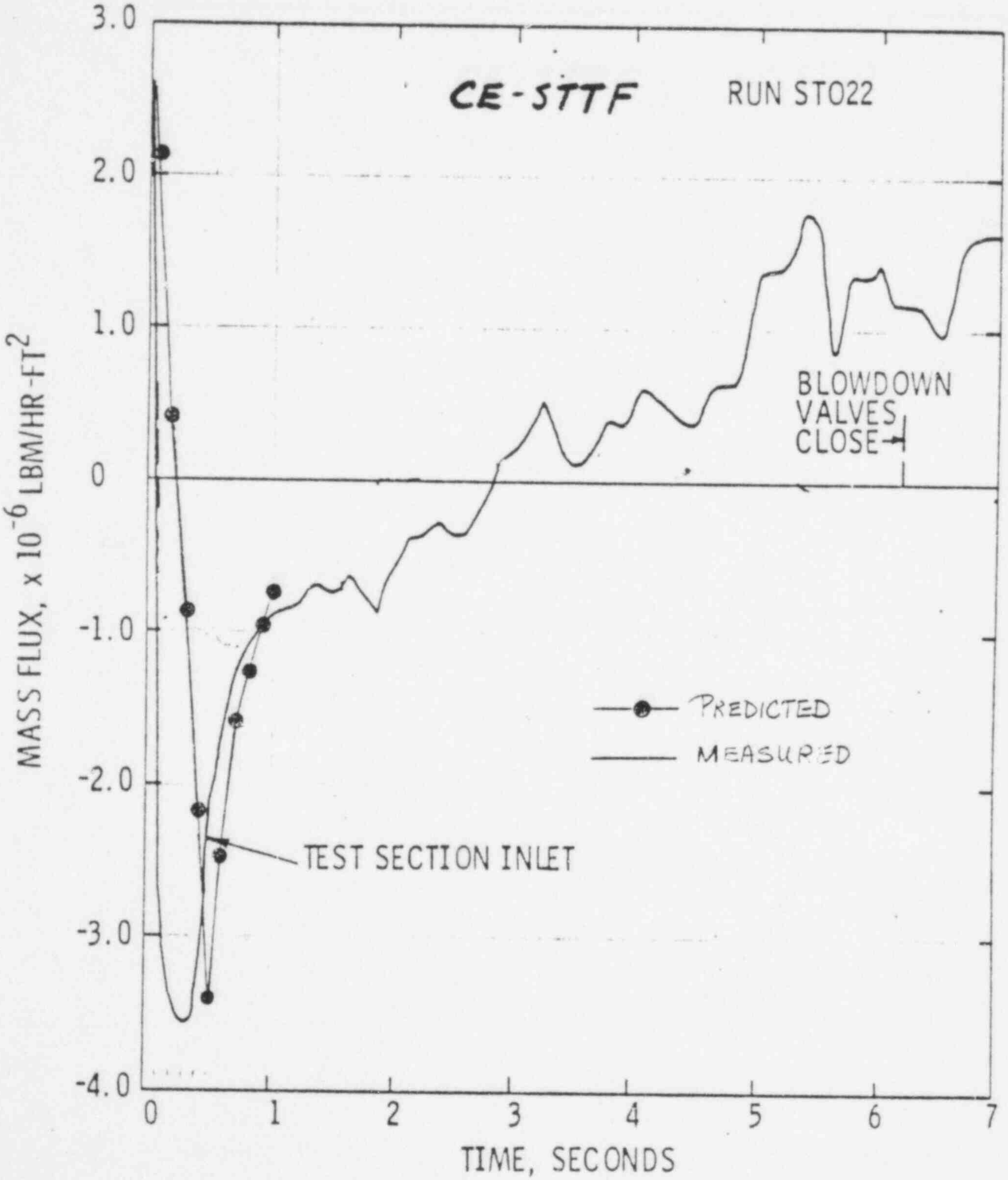

Fig. 8. Comparison of predicted and measured inlet mass velocity.

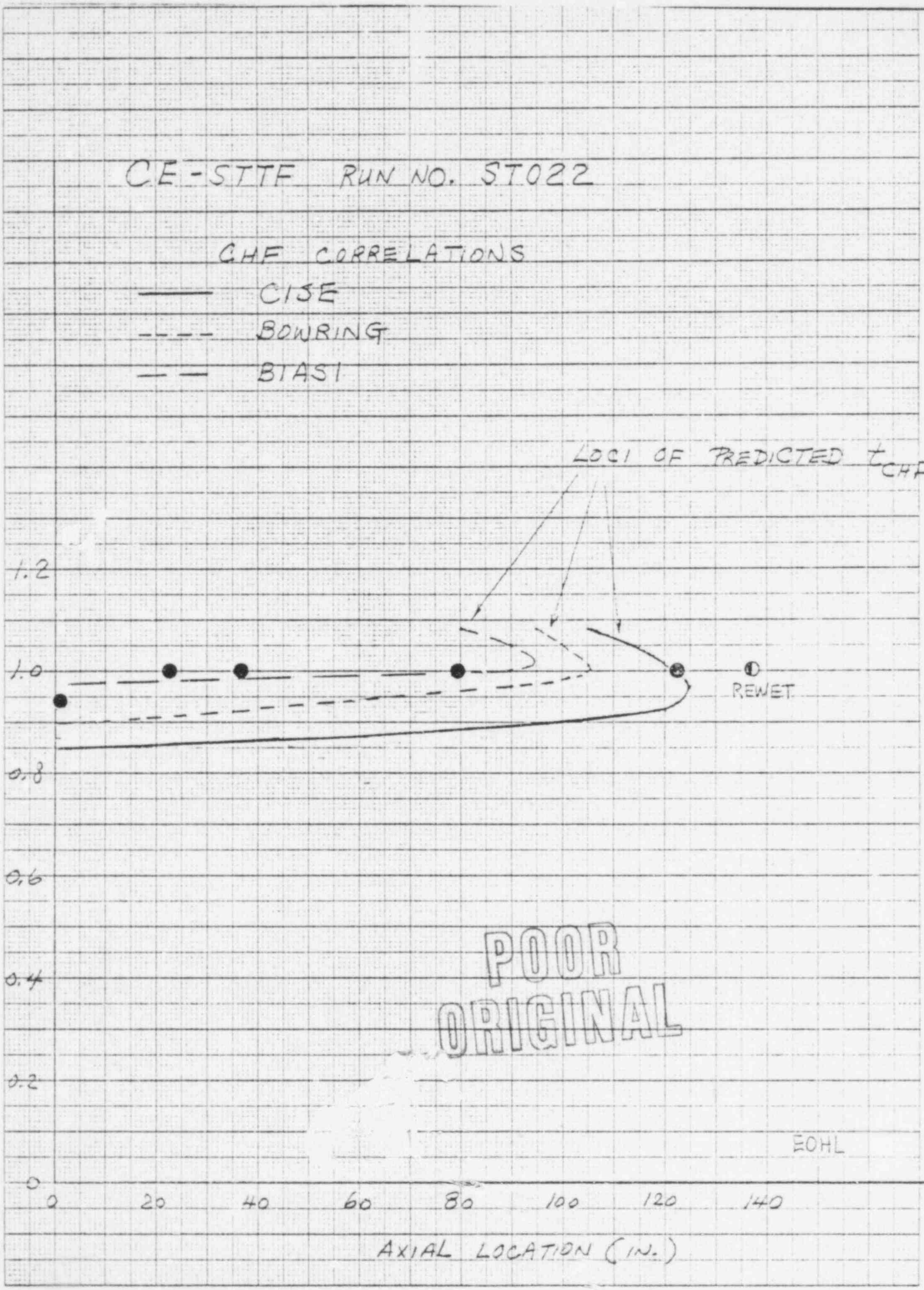

Fig. 9. Predicted times to CHF in CE single-tube blowdown test.

piece. The results are indicated in Fig. 10 where CISE and Biasi correlations were able to predict the onset of dryout in the high-power region remarkably well. The result using the GE bundle correlation is also shown for comparison. The calculation was terminated at about 2.3 sec with the presence of highly superheated steam calculated in the middle of the core. Again in this test, CHF was a result of high quality two-phase mixture. The 20% small inlet break leads to a fluid stagnation or levitation condition somewhere in the heated section.

The results obtained this month continue to demonstrate the adequacy of using steady-state CHF correlations (in the form employed by the local-condition hypothesis) in predicting CHF during transients. More bundle data will be included in future analysis.

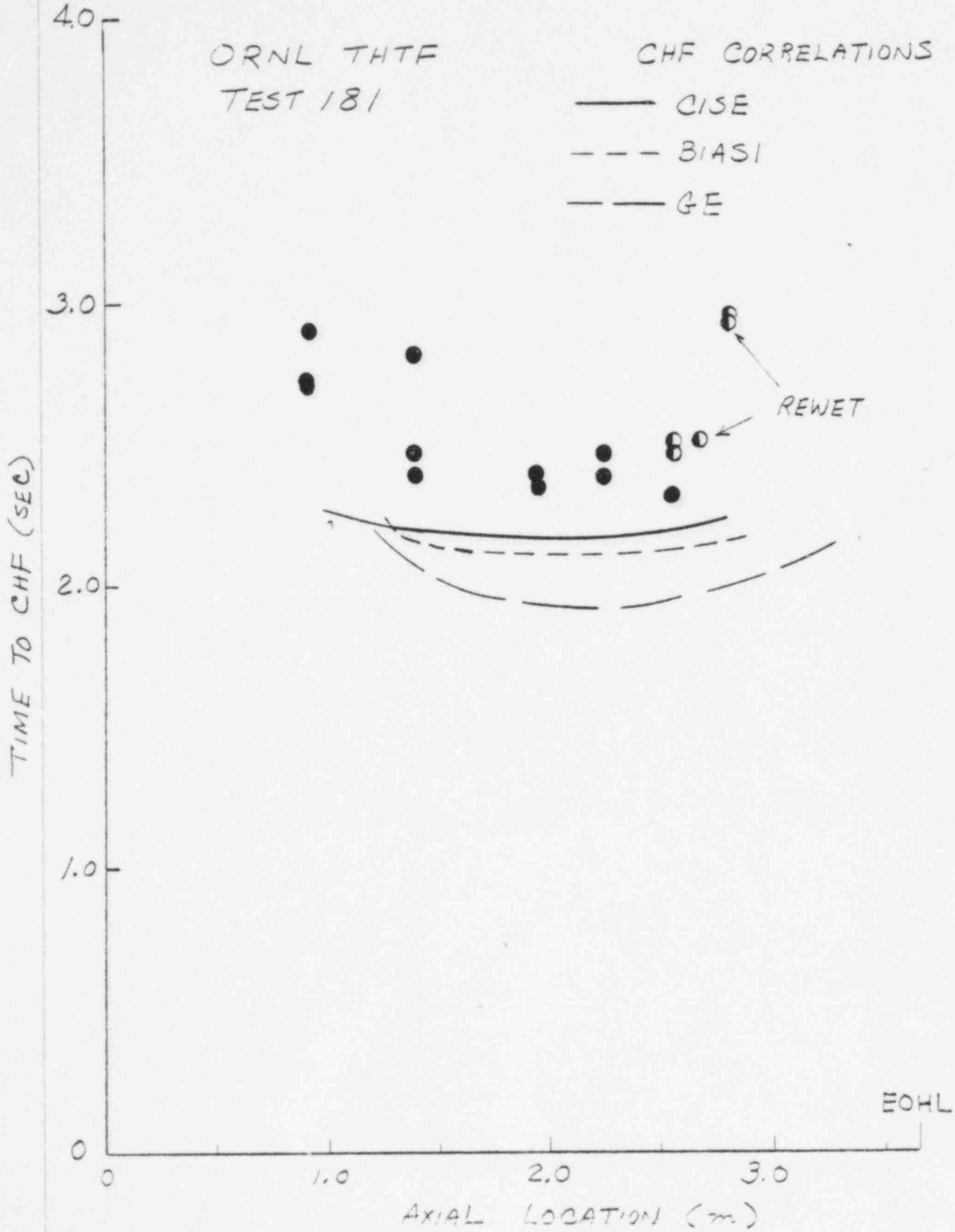

Fig. 10. Predicted times to CHF in ORNL THTF test 181.

1072 171

References

1. C. L. Wheeler et al., "COBRA-IV-I: An Interim Version of COBRA for Thermal Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores," BNWL-1962 (1976).
2. J. E. Meyers, "Hydrodynamic Models for the Treatment of Reactor Thermal Transients," Nucl. Sci. Eng., 10, p. 269 (1961).
3. W. T. Hancox and S. Banerjee, "Numerical Standards for Flow-Boiling Analysis," Nucl. Sci. and Eng., 64, p. 106 (1977).
4. W. T. Hancox, W. G. Mathers, and D. Kawa, "Analysis of Transient Flow-Boiling; Application of the Method of Characteristics," Paper 42, AIChE 15th Natl. Heat Transfer Conf., San Francisco, CA (1975).
5. G. B. Wallis, One-Dimensional Two-Phase Flow, McGraw-Hill (1969).
6. R. C. Kern et al., "Qualification of PWR Transient Analysis Methods with Plant Startup Measurements," Trans. Am. Nucl. Soc., 32, 447 (1979).
7. Water Reactor Safety Research Monthly Progress Report for April 1979, Reactor Analysis & Safety Division of ANL.
8. R. T. Lahey et al., "The Effect of Non-uniform Axial Heat Flux on Critical Power," Proc. Heat and Fluid Flow in Water Reactor Safety, 193 (1977).
9. J.C.M. Leung and R.E. Henry, "Transient Critical Heat Flux," Light Water Reactor Safety Research Quarterly Progress Report, Oct-Dec 1978, ANL-79-18 (1979).
10. H. N. Guerrero etal., "Single Tube and Rod Bundle Blowdown Heat Transfer Experiments Simulating Pressurized Water Reactor LOCA Conditions," ASME Paper 76-HT-11, St. Louis (1976).
11. CE/EPRI PWR Blowdown Heat Transfer Program, Quarterly Technical Progress Report No. 3, April 1 to June 30 (1975).

12. P. Saha, "THOR, Development of a Computer Code for the Prediction of Thermal Hydraulics of Reactors," presentation at Sixth Water Reactor Safety Information Mtg., Gaithersburg (1978).

13. W. G. Craddick *et al.*, Quick Look Report on Thermal Hydraulic Test Facility Test 181, April 26 (1979).

Two-Fluid Model of Two-Phase Flow in a Pin Bundle of a Nuclear Reactor
(T. C. Chawla and M. Ishii)

An accurate prediction of both single and two-phase thermal hydraulics of a pin bundle for thermal is of extreme importance both to the design and the safety of these reactors. The fluid flow and heat transfer in a pin bundle is extremely complex, consequently with the exception of the formulation by Chawla and Ishii [1], all previous attempts (see for example [2-5]) at formulating the governing equations both for single and two-phase flows have utilized heuristic macroscopic balances using finite control volumes (e.g., subchannels) for mass, momentum, and energy. It has long been recognized that the cross-sectional area averaging is a very useful tool in formulating governing equations for fluid flow and heat transfer in very complex geometries. As a result of area averaging, the three-dimensional field equations are reduced to quasi-one-dimensional forms. As a consequence of area averaging, the information on changes of variables in direction normal to the main flow direction is basically lost within a subchannel. The transfer of the momentum and energy between the fluid and wall is expressed by empirical correlations or by simplified models. However, the application of area averaging alone in a pin bundle geometry does not yield a complete description of the momentum exchange between the channels at the interchannel boundaries. This consideration in turn has led to the use of segment averaging along the interchannel boundary of momentum equation in transverse (to the gap between the pins) direction to provide an equation for crossflow. These formal procedures were utilized for the first time by Chawla and Ishii [1] in the formulation of the governing equation for drift flux model of two-phase flow in a pin bundle geometry.

The drift flux model (or mixture model) is formulated by considering the mixture as a whole, rather than two-phases separately. The drift flux model thus requires only four field equations namely, continuity, momentum, and energy equations for the mixture, and the continuity equation for one of the phases, say vapor [1,6]. On the other hand, two-fluid model of a two-phase flow is formulated in terms of two sets of conservation equations governing the balance of mass, momentum, and energy of each phase [7]. Since the macroscopic fields of one phase are not independent of the other phase, the interaction terms which couple the transport of mass, momentum, and energy of each phase across the interphases appear in the field equation whereas in the drift flux model which considers the mixture as whole, these interaction terms cancel each other. In the two-fluid model formulation, the transport processes of each phase are expressed by their own balance equations, therefore it is expected that the model can predict more detailed changes and phase interactions than the drift flux model. Although the drift flux model is simpler than the two-fluid model, it requires some drastic constitutive assumptions since it has only four field equations in contrast to six field equations in the two-fluid model. Therefore, it is natural that some of the characteristics of two-phase flow in the drift model will be lost. The drift flux model is generally useful in analyzing two-phase flows where there exists a strong coupling between the motion of two phases, and the information desired is the response of total mixture and not that of each constituent phase separately; for example, in the dynamic analysis of two-phase flow systems where the response of total system is desired such as in the analysis of thermohydraulic flow instability problem in the boiling channels [8,9]. Two-phase flow problems involving a sudden acceleration of one phase may not be appropriately described by the drift flux model. In these cases, inertia terms of each phase should be considered, that is, by use of two-fluid model.

Previous studies have indicated that unless phasic interaction terms are accurately modelled [10] in the two-fluid model, the numerical instabilities are frequently encountered in the numerical solution of two-fluid models. A recent study by Lahey <u>et al</u>. [11] has demonstrated that virtual mass originating from momentum interaction between the two phases had a considerable effect on improving numerical stability and efficiency. Another approach to achieving numerical stability is the inclus of "artificial viscosity" in the numerical algorithm to damp out high frequency oscillations occurring possibly due to imprecise modelling. This approach is currently being followed by Amsden and Harlow [12] in their two-fluid digital computer codes. In spite of these shortcomings of two-fluid model, there is however no substitute available for modelling accurately two-phase phenomena where two phases are weakly coupled.

The objective of the present study is to obta the governing equations for two-fluid model for two-phase flows in a pin bundle geometry. For this purpose, we view two-phase flow as a field which is subdivided into two turbulent single-phase regions with moving boundaries separating the two constituent phases such that the differential balances for turbulent flow hold for each subregion and for the interface, wherein the latter differential balances which account for singular characteristics of the interface, we further assume that all interfaces are identical and have the same interface velocity. With the exception of the interface velocity, we assume all other singular transferrable properties of the interface are turbulent in nature. Since the turbulent fields in each subregion are unsteady because of the moving and deforming interfaces, one must view conceptually these turbulent balances as ensemble averages of local instantaneous differential field equations for single phase and of the local instantaneous differential jump conditions. These ensemble averages are constructed with an assumption that all the samples of two-phase flows in an ensemble are statically identical such that if all observed at a given instant of time, a given point in each sample is surrounded by the same phase or is located at the interface implying the structure of two-phase flows and the geometry of interface are identical between the samples. By assuming further that the two-phase flow is temporarily stationary, then by ergodic hypothesis the ensemble averages at a given instant of time become equal to temporal averages. With the three-dimensional field equations thus obtained, we perform the Eulenian area averaging over the cross-sectional area of each phase in a given channel and segment averaging of transverse momentum equation along the phase intercepts at the interchannel boundaries. To simplify the governing equations obtained as a result of these operations, we invoke the assumption that the motion of the fluid in each phase is dominantly in axial direction, that is the transverse components of velocity are small compared to axial components. We further assume that within a channel the variation of axial component of velocity is much stronger than the variation along the axial direction. We also assume that similar arguments can also be applied to the variation of enthalpy in a channel. As a result o[illegible] considerations, we obtain two sets of continuity, momentum, and energy equations describing motion of each phase in the axial direction. The phasic interaction terms which appear in these equations are governed by interfacial transfer conditions obtained from interface differential balances. The segment averaged transverse momentum equation for each phase provides the governing equation for cross flow.

It is demonstrated that the governing equations obtained in this study are based on formal and sound physic l basis and are indispensable if physically correct methods are desired for analyzing two-phase flows which require modelling by two-fluid model in a pin bundle.

1072 177

REFERENCES

1. Chawla, T. C. and Ishii, M. "Equations of Motion for Two-Phase Flow in a Pin Bundle of a Nuclear Reactor," Int. J. Heat Mass Transfer, Vol. 21, 1978, pp. 1057-1068.

2. Stewart, C. W., Wheeler, G. L., Cena, R. J., McMonagle, C. A., Cuta, J. M., and Trent, D. S., "COBRA-IV: The Model and the Method," Battelle Pacific Northwest Laboratory Report BNWL-2214, NRC-4, 1977.

3. Rowe, D. S., "A Mathematical Model for Transient Subchannel Analysis of Rod-Bundle Nuclear Fuel Elements," J. Heat Transfer, Vol. 95, 1973, pp. 217-221.

4. Bowring, R. W., "HAMBO, A Computer Programme for the Subchannel Analysis of the Hydraulic and Burnout Characteristics of Rod Clusters, Part 1, General Description," United Kingdom Atomic Energy Authority Report AEEW-R524, 1967.

5. St. Pierre, C. C., "SASS Code 1, Subchannel Analysis for the Steady State," Atomic Energy of Canada Limited Report AECL-APPE-41, 1966.

6. Zuber, N., "Flow Excursions and Oscillations in Boiling, Two-Phase Flow Systems with Heat Addition," in Proceedings of EURATOM Symposium on Two-Phase Flow Dynamics, Commission of European Communities, Brussels, Vol. 1, 1967, pp. 1070-1089.

7. Ishii, M., Chawla, T. C., and Zuber, N., "Constitutive Equation for Vapor Drift Velocity in Two-Phase Annular Flow," AIChE Journal, Vol. 22, 1976, pp. 283-289.

8. Ishii, M. and Zuber, N., "Thermally Induced Flow Instabilities in Two-Phase Mixtures," Paper No. B5-11, 4th International Heat Transfer Comference, Paris, France, 1970.

9. Saha, P., "Thermally Induced Two-Phase Flow Instabilities, Including the Effect of Thermal Nonequilibrium Between the Phases," Ph.D. Thesis, Georgia Institute of Technology, Atlanta, 1974.

10. Boure, J., "On a Unified Presentation of the Non-Equilibrium Two-Phase Flow Models," ASME Symposium Volume, Non-Equilibrium Two-Phase Flows, 1975.

11. Lahey, Jr., R. T., Cheng, L. Y., Drew, D. A., and Flaherty, J. E., "The Effect of Virtual Mass on the Numerical Stability of Accelerating Two-Phase Flows," Presented at AIChE 7th Annual Meeting at Miami Beach, Florida, 1978.

12. Amsden, A. A., and Harlow, F. H., "K-TIF: A Two-Fluid Computer Program for Downcomer Flow Dynamics," Los Alamos Scientific Laboratory, LA-6994, NRC-4, 1978.