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Proceedings of the U.S. Nuclear Regulatory Commission

Eighth Water Reactor Safety Research Information Meeting

Held at
National Bureau of Standards
Gaithersburg, Maryland
October 27-31, 1980

U.S. Nuclear Regulatory Commission

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Date Published: March 1982

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

FINAL AGENDA

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

AT THE

NATIONAL BUREAU OF STANDARDS
ADMINISTRATION BUILDING 101
GAITHERSBURG, MARYLAND

October 27-31, 1980

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MONDAY, OCTOBER 27, 1980

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Chairman: M. L. Picklesimer, NRC

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Chairman: L. H. Sullivan, NRC

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3:30 pm -	BWR Refill-Reflood Program: Overview and Experimental Results	G. W. Burnette, GE
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Seismic Safety Margins Research Program

Chairman: C. W. Burger, NRC

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2:40 pm	Subsystem Response	T. Y. Chuong, LLL
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THURSDAY, OCTOBER 30, 1980
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Chairman: R. F. Abbey, NRC

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11:15 am	Measured Pressure Loads on Model Structures in Simulated Tornado-Like Flow	M.C. Jischke, U of Oklahoma
11:45 am	Automobile Impact Studies	R. Chiapetta, Chiapetta-Welch & Assoc.
12:15 pm	Atmospheric Dispersion Field Experiments to 80 km	I. Van der Hoven, NOAA Air Resources Laboratory

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Chairman: J. Harbour, NRC

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3:00 pm	Reservoir-Induced Seismicity	A. Murphy, NRC
3:45 pm	Northeastern U.S. Seismic Network	P. Pomeroy, NRC Consultant
4:30 pm	Tectonic Features in the Vicinity of the Charleston 1886 Earthquake	J. Behrendt, USGS

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10:00 am	Introduction	C. Z. Serpan, NRC

Irradiation Effects and Neutron Dosimetry

Chairman: C. Z. Serpan, NRC

10:30 am	LWR Dosimetry Improvement Program Overview	W. N. McElroy, HEDL
11:00 am	Reactor Calculation "Benchmark" PCA Blind Tests Results	F.B.K. Kam, ORNL

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Chairman: M. Vagins, NRC

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2:00 pm -	Validation of "Key Curve" Analysis of Elastic-Plastic Fracture Toughness	J. Joyce, USNA
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3:50 pm -	Crack Growth Rate of Irradiated Vessel and of Piping Steels in PWR Environments	H. Watson/ W. Cullen, NRL
4:20 pm -	Crack Arrest Methodology and and Standard Test Methods for RPV Evaluations	G. Irwin/W. Fourney, UMD
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FRIDAY, OCTOBER 31, 1980
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Chairman: M. Vagins, NRC

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FRIDAY, OCTOBER 31, 1980
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Chairman: J. Muscara, NRC

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Chairman: J. Muscara, NRC

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STRUCTURAL ENGINEERING RESEARCH BRANCH

Chairman: G. Bagchi, NRC

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Load Combinations

Chairman: J. O'Brien, NRC

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Event Decoupling (LOCA Plus Earthquake)

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10:20 am	Fracture Mechanics Evaluation	R. D. Streit, LLNL
10:50 am	Results and Conclusion	S. C. Lu, LLNL

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MECHANICAL ENGINEERING RESEARCH PROGRAMS
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11:55 am -	Load Combination Methodology	M. W. Schwartz, LLNL
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Chairman: D. Reiff, NRC

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2:30 pm -	Piping Benchmarks	M. Reich, BNL
3:15 pm -	Snubber Research and Testing	A. Onesto, ETEC
3:45 pm -	Experimental and Preliminary Analytical Results of Coupled Fluid-Structure Interactions During Blowdown of the HDR Vessel	U. Schumann, KfK (FRG)
4:10 pm -	Earthquake Simulation Experiments Performed at the HDR Facility	G. Katzemeier, KfK (FRG)
4:30 pm -	HDR Dynamic Tests of Late 1979	G. Howard, ANCO
5:00 pm -	Predictions of Recirculation Loop Response at HDR to Simulated Blast Excitation	R. Guenzler, INEL

UPPER PLENUM DUMP

by

Yukio SUDO

Japan Atomic Energy Research Institute
(Visiting Engineer Sept. 1979- Sept. 1980, M.I.T.)
Peter GRIFFITH

Professor of Mechanical Engineering
Massachusetts Institute of Technology

Presented at

The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

1. Introduction Combined upper and lower plenum injection of emergency core coolant involves a variety of phenomena which are different from the cold leg injection only. The question to which this experiment is addressed is how does a pool of water on top of the core get through the core during reflood. The understanding of upper plenum dump is the basis for understanding how quench occurs. The objective of this study are to investigate:

(1) Under what conditions an upper plenum dump occurs, that is, the effects of parameters which describe condition of the upper plenum dump.

(2) For what initial temperature conditions is the flow either one or two dimensional through the core.

Singh[1] carried out steady-state experiments using a 2 x 6 array of cartridge heaters simulating fuel rods in a reactor. He used Freon-113 as a coolant and investigated the characteristics of gravity-dominated two phase flow during a PWR reflood. F-113 was supplied into the upper plenum and the same flow rate of F-113 was drained from the lower plenum at the fixed heat flux in the core.

Because of equipment restrictions (heat flux and pump power), he did not investigate the conditions under which the upper plenum dump occured nor could he investigate the effect of the heat flux or the core temperature on upper plenum dump.

2. Experiment Test rig consists of a test section including a core, an electric furnace and a water supply tank. The test section has an upper plenum, a screen core and a skirt attached to the bottom of the screen core. Core was made of stainless steel screens about 0.2 m wide, 0.2 m high and 0.025 m thick. This core has many advantages. The phenomena during reflood is the gravity-dominated phenomena. So, the friction term should be properly simulated in the experiment. The ratio L'/D of core height and hydraulic diameter was designed as closely to that of a reactor as possible. The skirt is provided as one of components and this effect is to be investigated. This test section is set in the lower plenum so that arrangement of components is simulated.

First, a series of transient experiments were carried out to obtain a map of flow pattern by investigating the effect of parameters because the flow pattern is expected to be related intimately to the dump phenomena. The parameters selected are (1) Initial Core Temperature T_{core}= 300 to 600°C, (2) Injected Water Temperature T_{water}= 25 to 100°C, (3) Initial Water Level in the Lower Plenum h_1= 0.02 to 0.14 m and (4) Initial Water Level in the Supply Tank h_2= 0.08 to 0.24 m. The judgement of flow pattern was done by visual observations. The criteria for the determination of flow pattern is whether vapor bubbles are observed to come out from the bottom of the skirt at the bottom of the test section.

Upper plenum water level changes are an inportant part of the problem because flow rate of water through the core should be different depending on the flow patterns. Water flow rate in counter-current flow is restricted by the upward steam flow compared to co-current down flow. In this experimental set-up the supply tank is connected to the upper plenum and the change of water level in supply tank was measured.

3.Analysis The following analysis was carried o t in order to help in understanding of the experimental results.

(1) Effect of the skirt attached to the bottom of the heated core on the characteristics of Water Flow Rate versus Pressure Drop curve. A one-dimensional numerical calculation was carried out by using the drift flux model.

(2) Difference in flow characteristics between a co-current down flow and counter-current flow. In the experiment there was a remarkable difference in the rate of water level in the supply tank depending on the core flow pattern.

(3) The region where water subcooling effects are important was investigated. Water subcooling has a large effect on the conditions under which the upper plenum dump occurs.

4.Results The most significant results are:

(1) The flow pattern in the core is determined by the hydrostatic head in the lower plenum, the hydrostatic head in the upper plenum and the inlet water subcooling.

(2) When the hydrostatic head in the upper plenum is larger than the hydrostatic head in the lower plenum, a co-current down flow occurs with the saturated water injection and a dump occurs through the core.

(3) Conversely, co-current down flow is not realized with the saturated water injection when the hydrostatic head in the upper plenum is less than the hydrostatic head in the lower plenum. In this case, counter-current flow occurs with a much lower water flow rate through the core than in the co-current down flow.

(4) With the subcooled water injection co-current down flow is realized even under the condition that the hydrostatic head in the upper plenum is less than the hydrostatic head in the lower plenum. This is because water subcooling diminishes the amount of steam in the core as well as in the upper plenum. The importance of this effect varies according to the magnitude of water subcooling.

Reference

(1) B. Singh, "Gravity Dominated Two-Phase Flows in Vertical Rod Bundles", ScD. Thesis, Mechanical Engineering Department, M.I.T., 1979.

(2) Y.Y, Hsu, "Proposed Heat Transfer 'Best Estimate' Packages", draft. USNRC, November 1977.

(3) P.Robershotte, "Down-Flow Post Critical Heat Flux Transfer of Low Pressure Water", M.S. Thesis, Mechanical Engineering Department, M.I.T., January 1977.

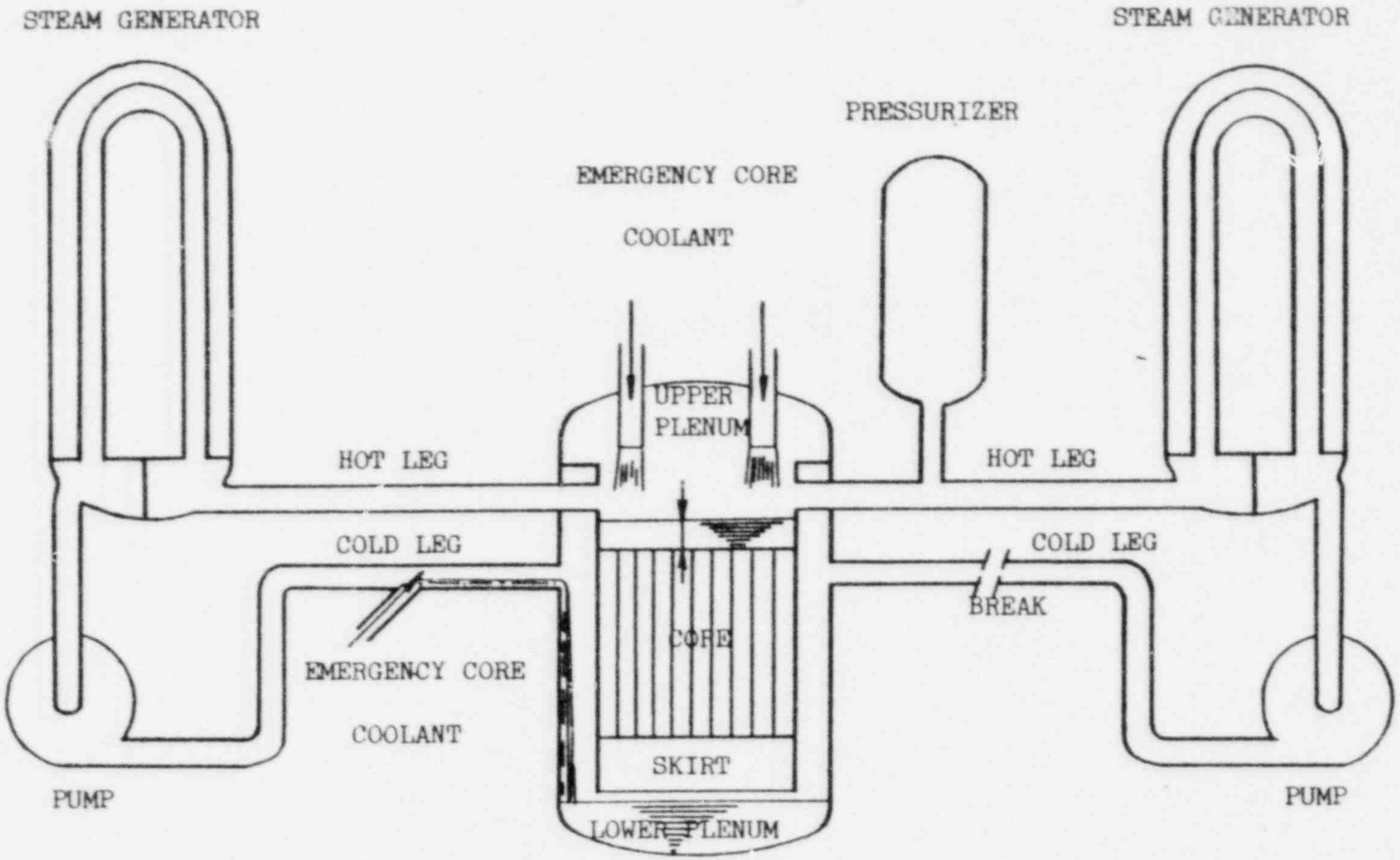


Figure 1. Schematic of a Pressurized Water Reactor During Double-Ended Break With Injection Both into Cold Leg and into Upper Plenum.

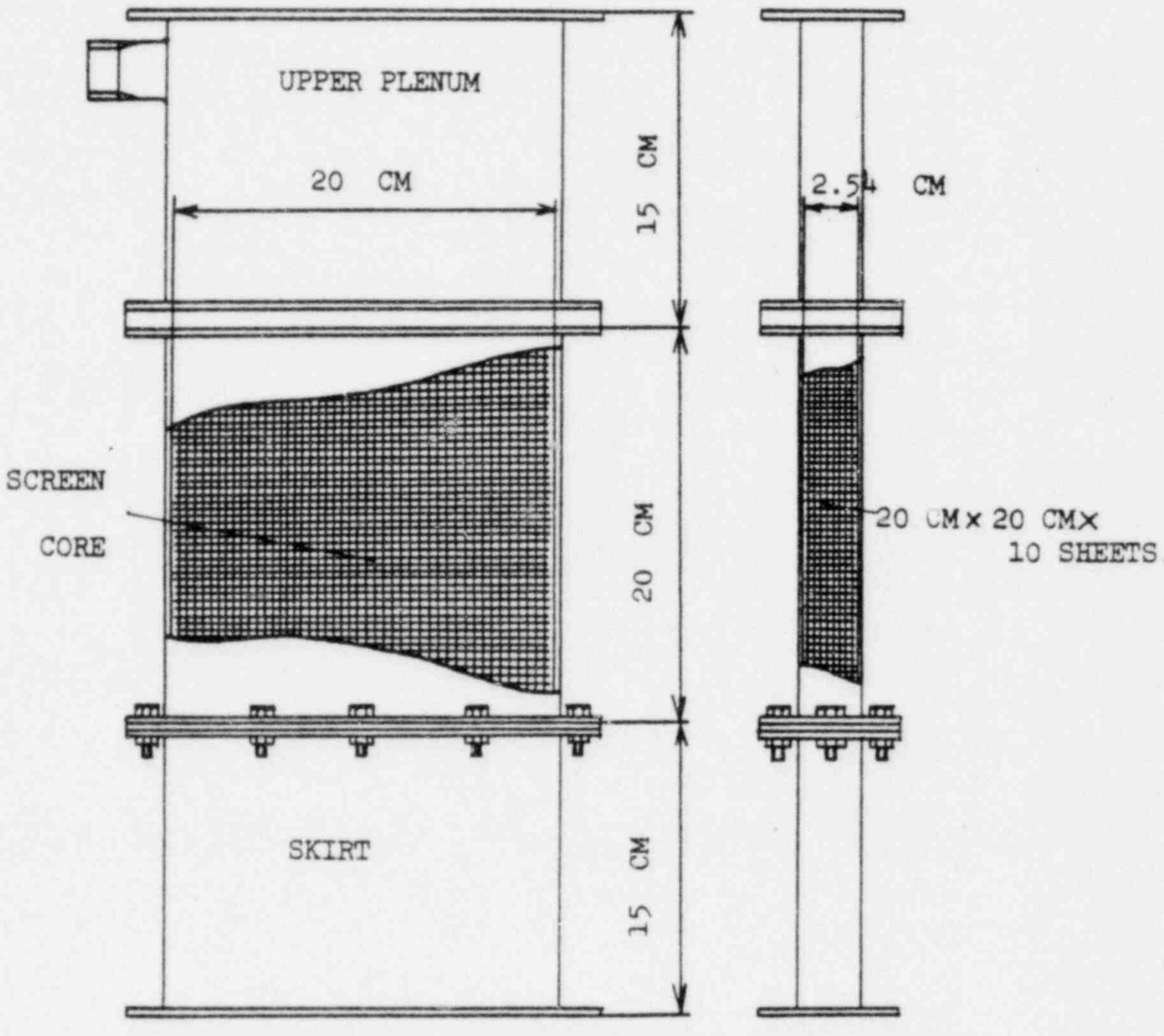


Figure 2. Schematic of Test Section.

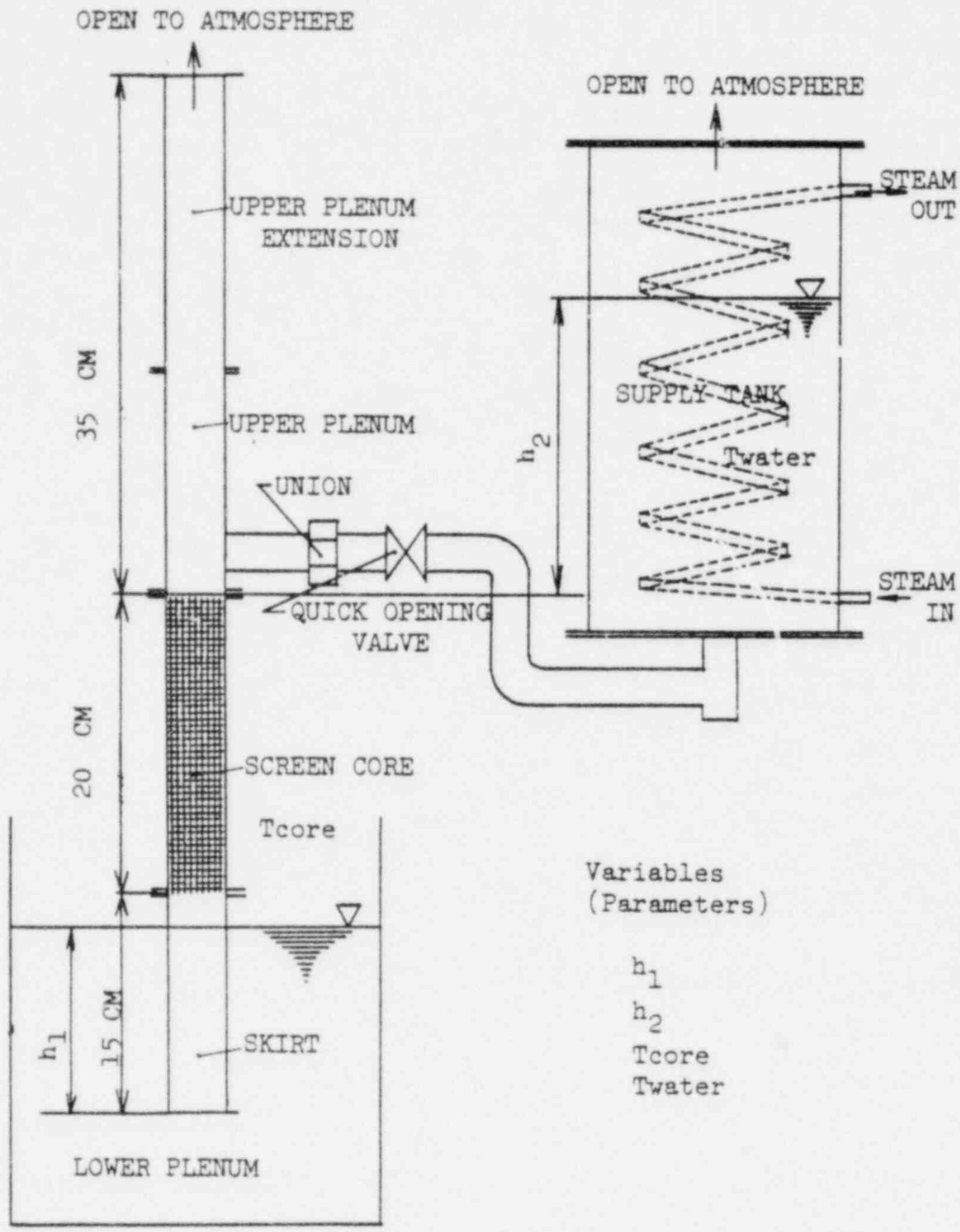


Figure 3. Schematic of Experimental Set-up for Controlled Water Head Experiment.

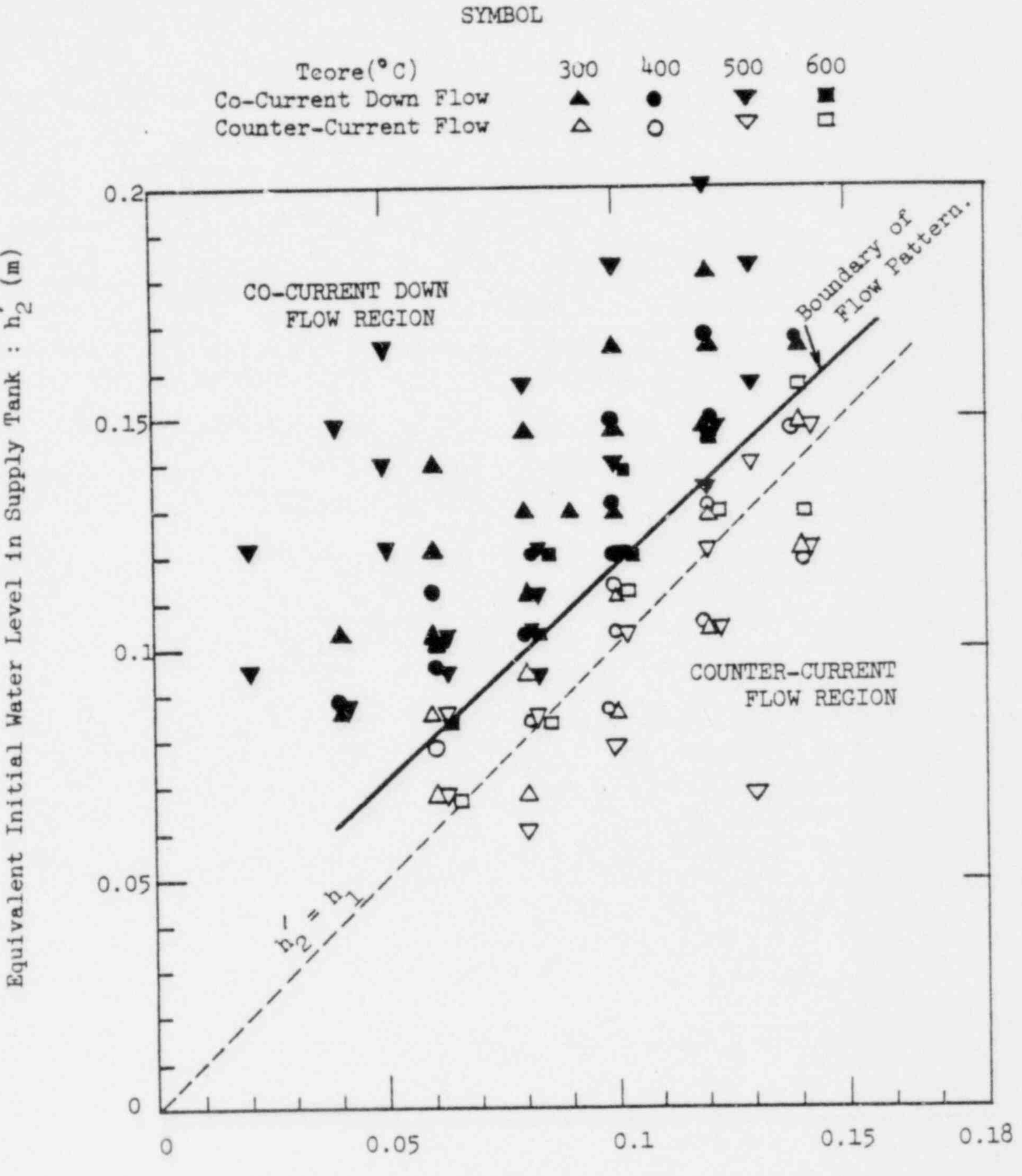


Figure 4. Summary of Observed Results in the Controlled Water Head Experiment with Saturated Water Injection - Relation of h_2 and h_1 on Whether Co-Current Down Flow or Counter-Current Flow Occurs.

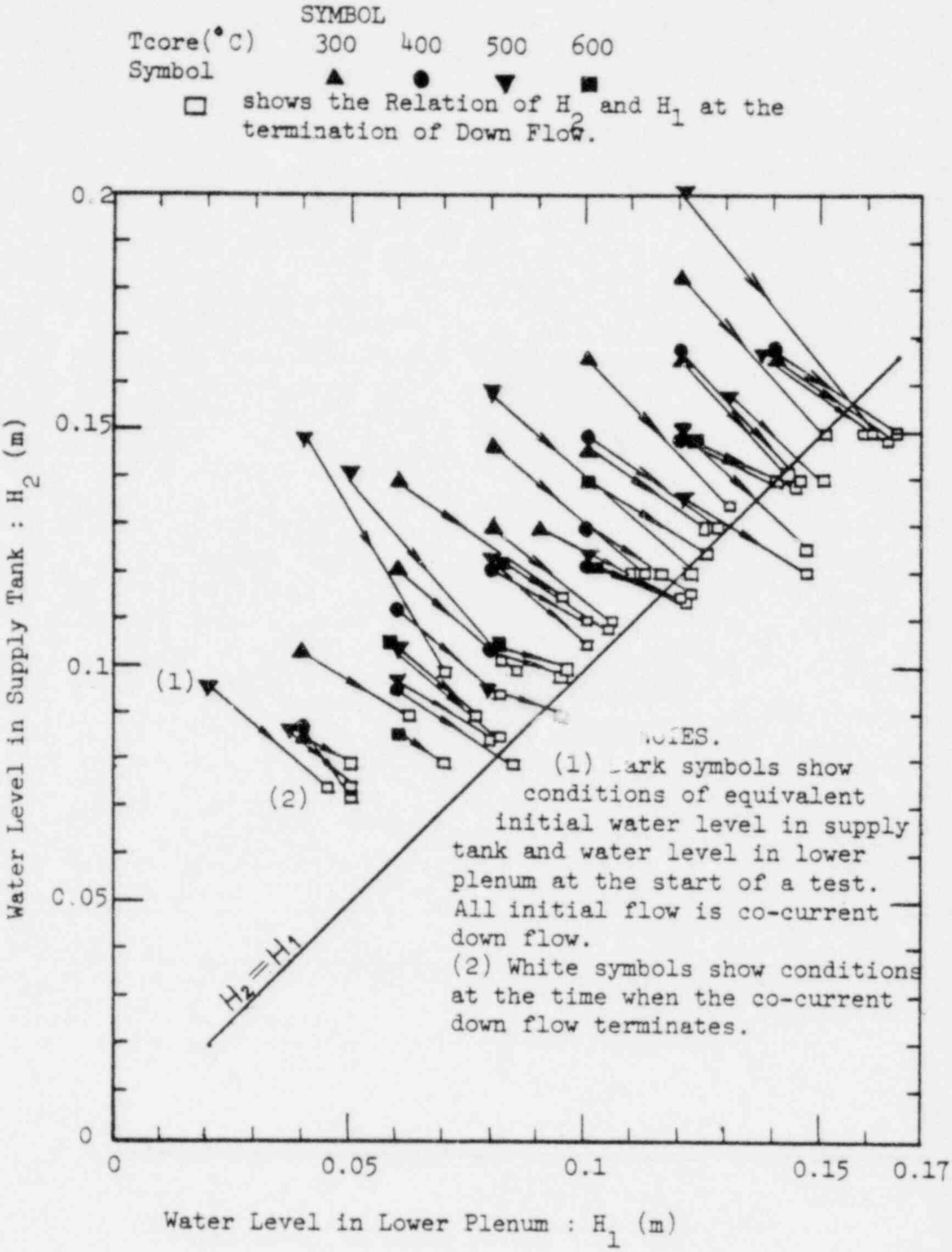


Figure 5. Relations of Water Level in Supply Tank and Water Level in Lower Plenum at the Time When Co-Current Down Flow Terminates during a Test with Saturated Water Injection.

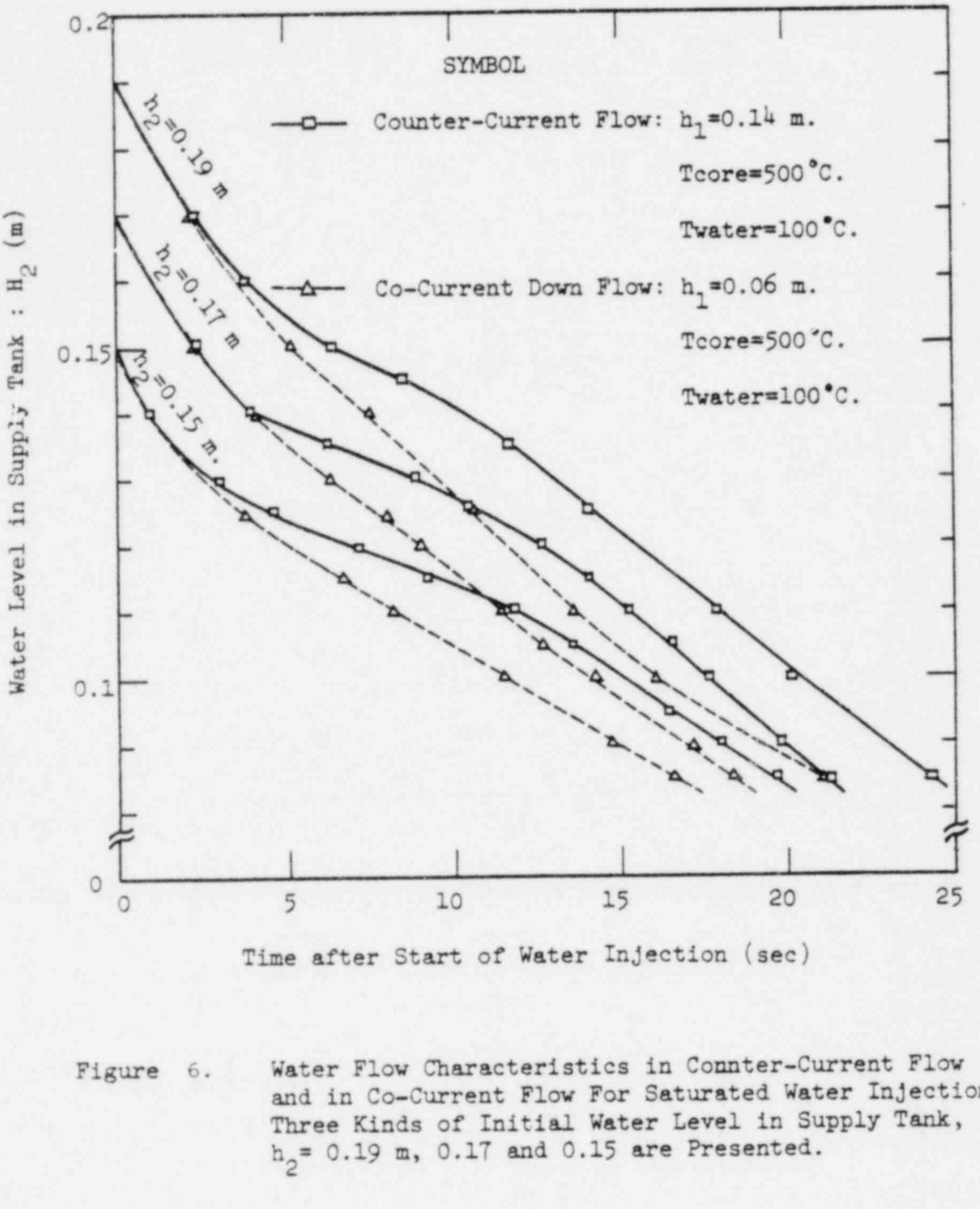


Figure 6. Water Flow Characteristics in Connter-Current Flow and in Co-Current Flow For Saturated Water Injection. Three Kinds of Initial Water Level in Supply Tank, h_2= 0.19 m, 0.17 and 0.15 are Presented.

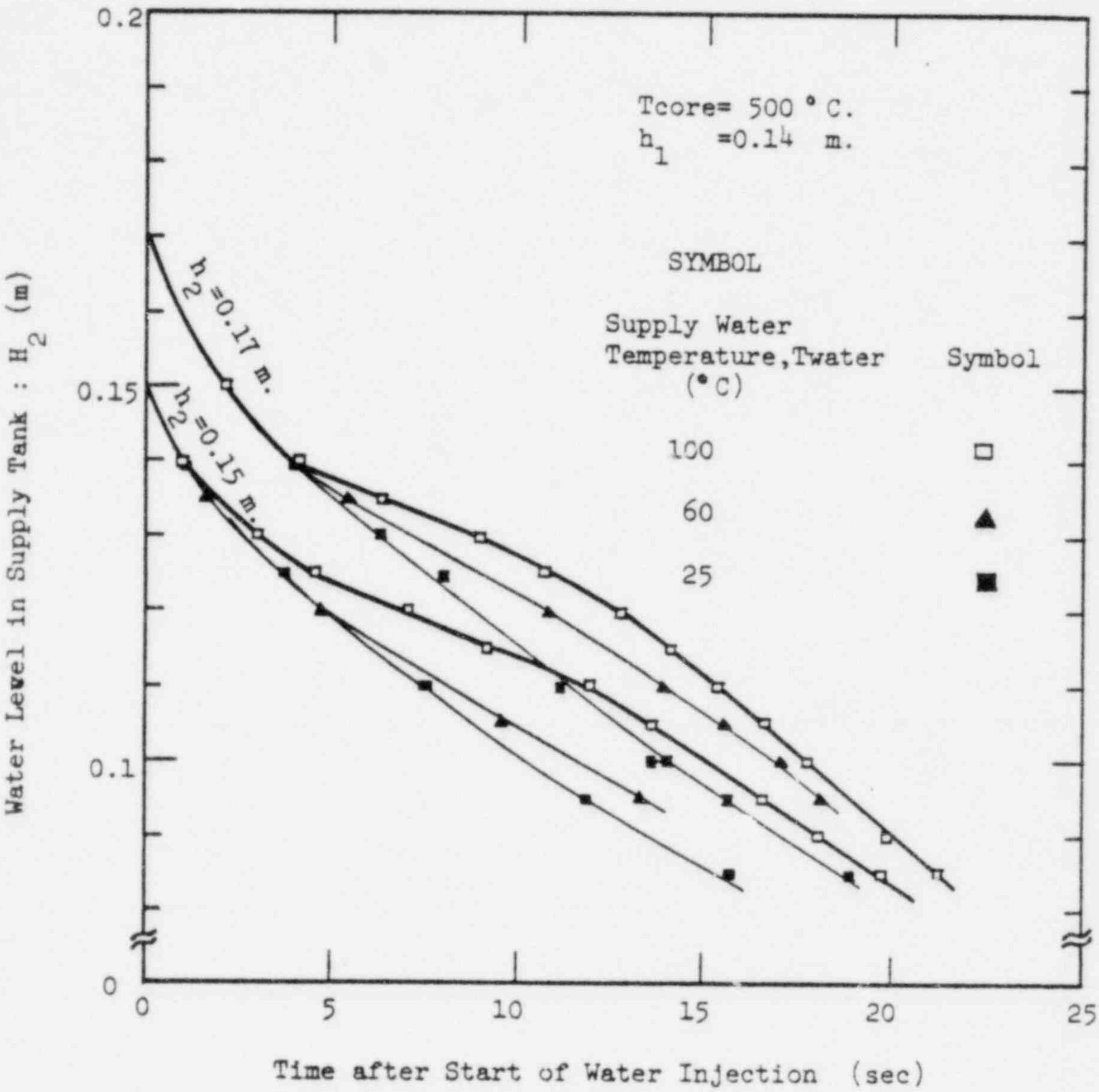


Figure 7. Effect of Water Subcooling on Water Flow Characteristics for the Condition of Initial Core Temperature,Tcore=500°C and Initial Water Level in Lower Plenum, h_1 =0.14 m.
- Water Flow Rates Histories Depend Strongly on Water Subcooling.

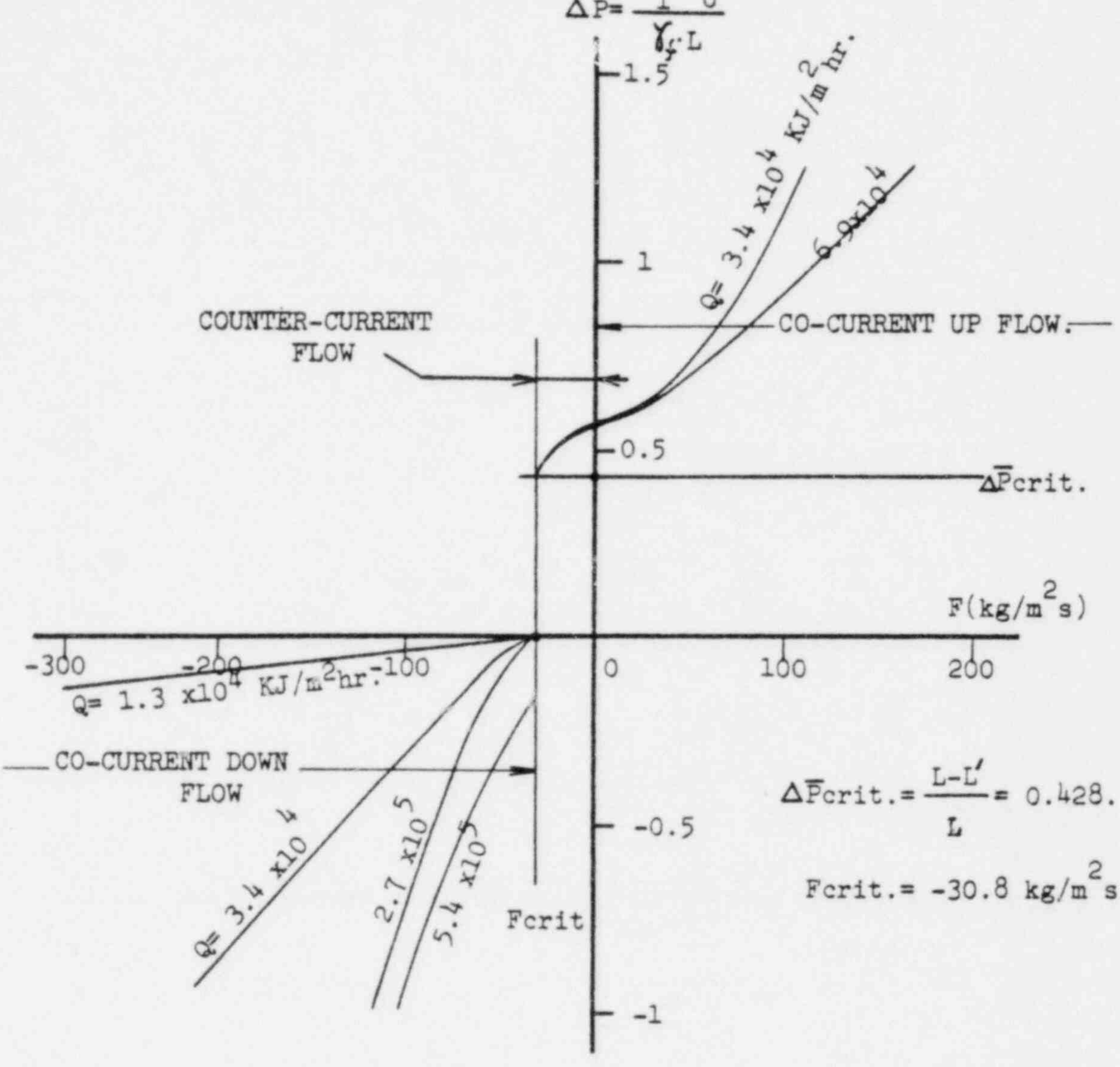


Figure 8. Schematic of Characteristics of Pressure Drop versus Water Flow Rate Curve for the Flow Channel which is Composed of the Core and the Skirt Attached to the Bottom of the Core.

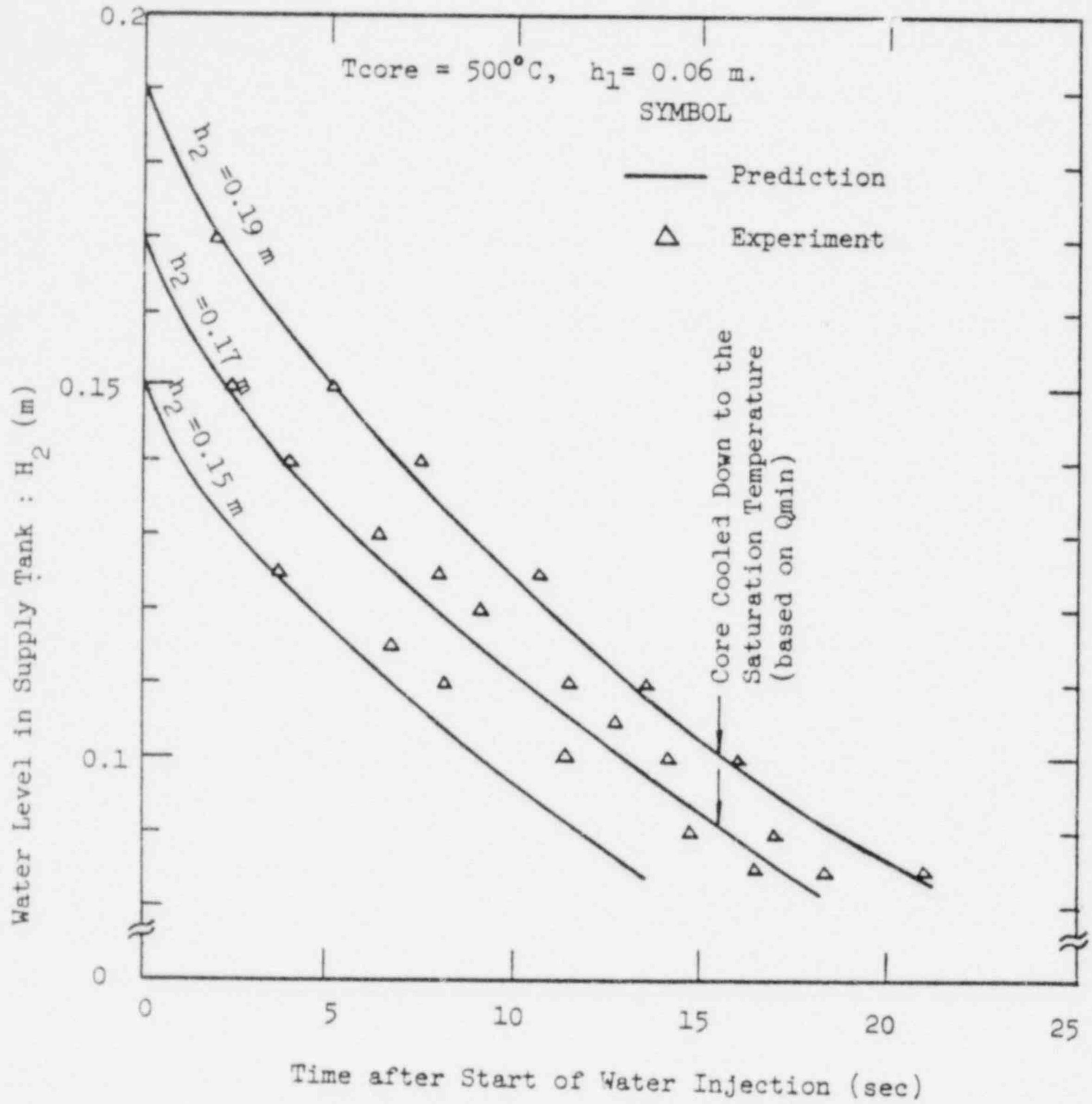


Figure 9. Comparison of Change in Water Level in Supply Tank between the Experimental Results and Predictions for Co-Current Down Flow with Saturated Water Injection.
Tcore = 500°C
h_1 =0.06 m.

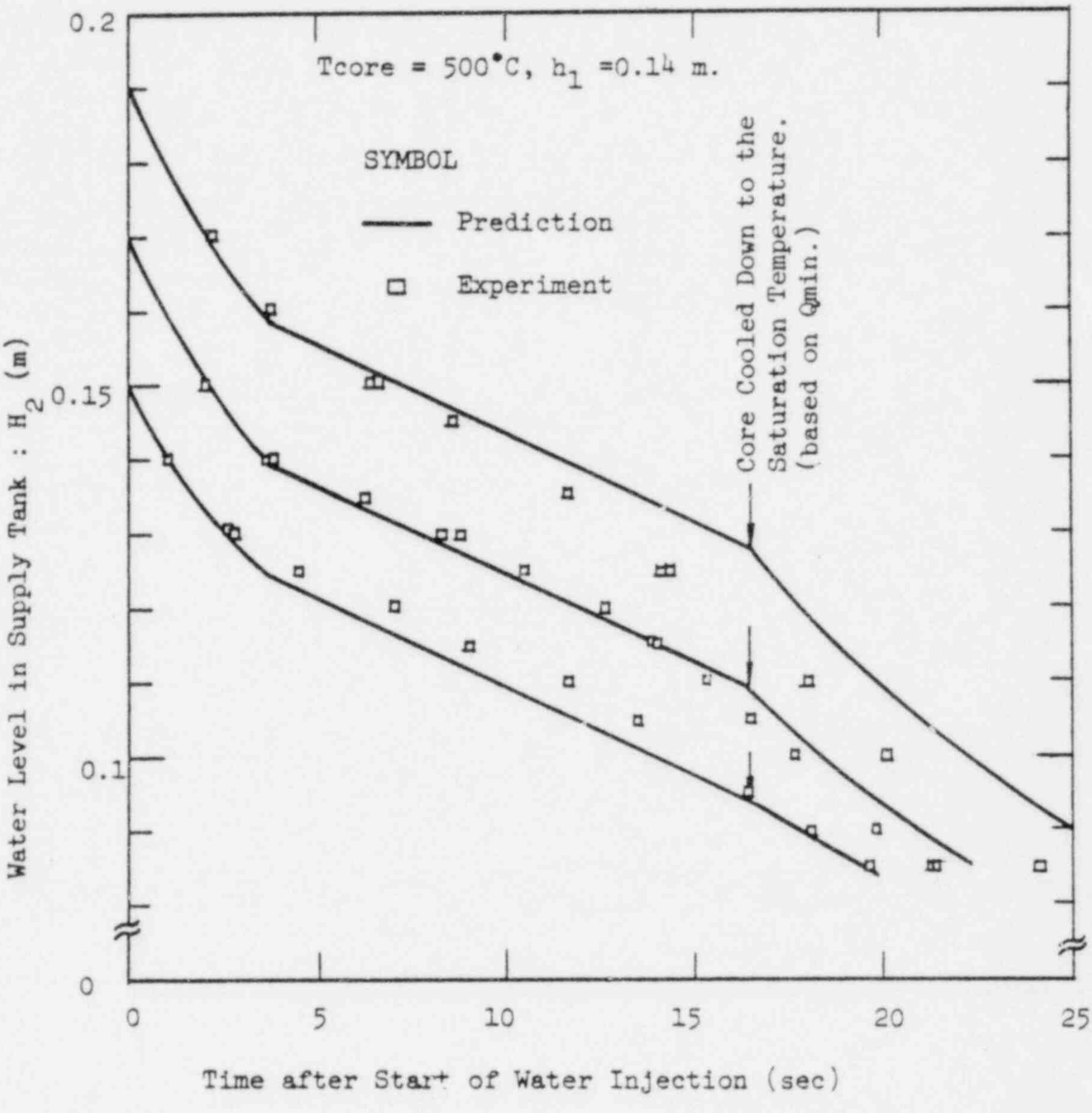


Figure 10. Comparison of Change in Water Level in Supply Tank between the Experimental Results and Predictions for Counter-Current Flow with Saturated Water Injection.
Tcore = 500°C,
h_1 =0.14 m.

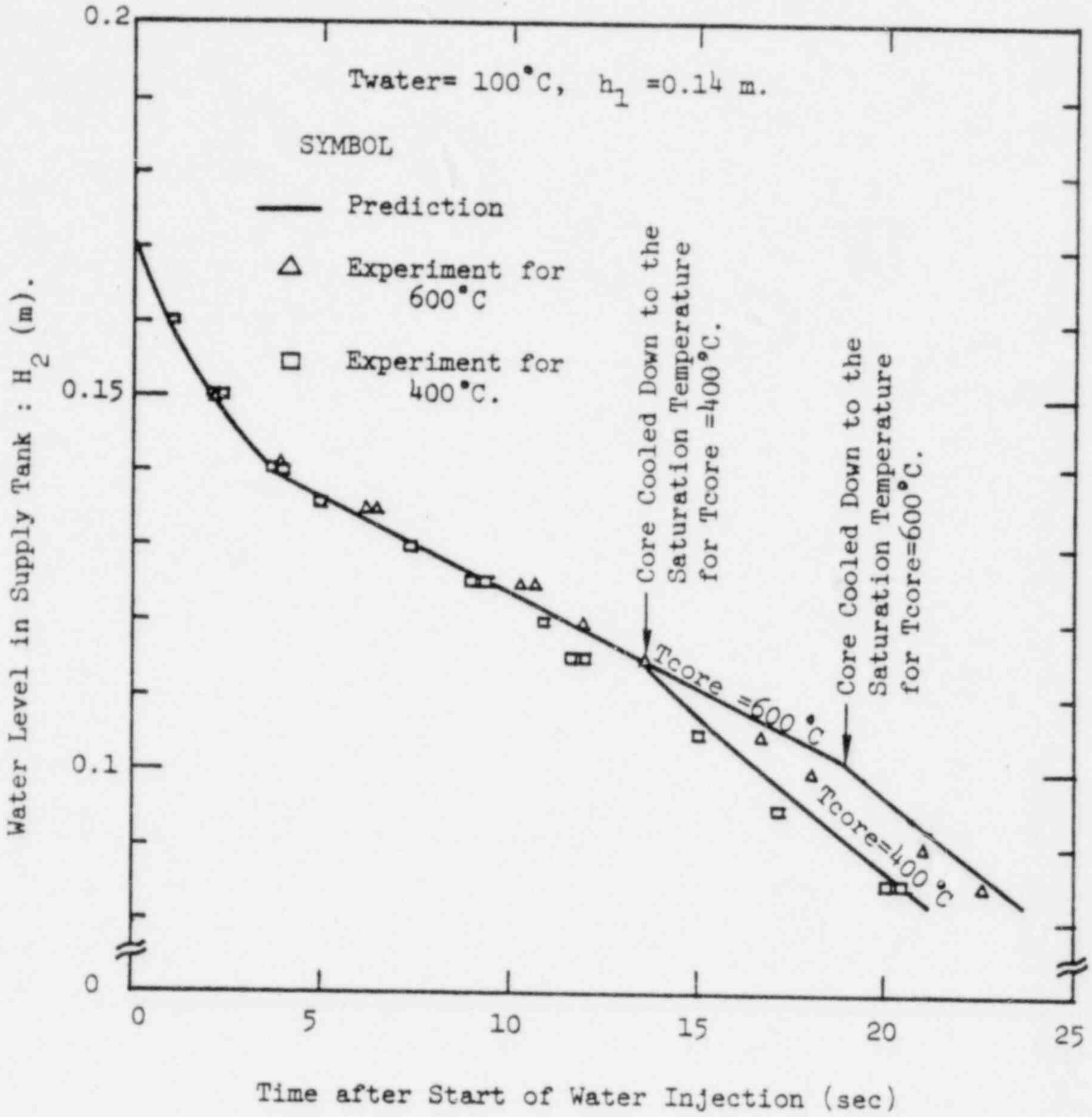


Figure 11. Effect of Initial Core Temperature on the Change in Water Level in Supply Tank for Counter- Current Flow with Saturated Water Injection - Comparisin of Predictions and Experimental Results.

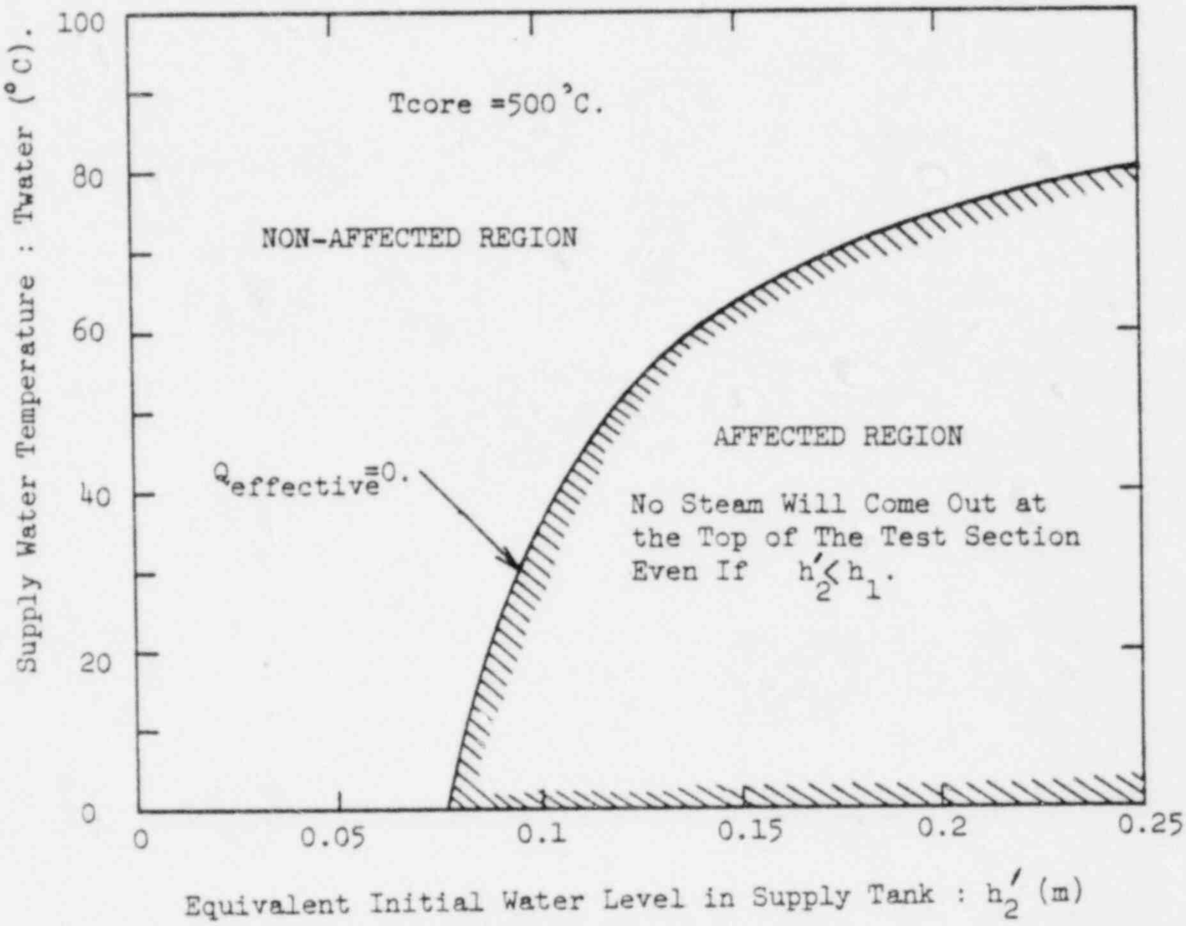


Figure 12. The Boundary of Region Where Water Subcooling Affects the Flow Pattern for Initial Core Temperature, Tcore= 500°C.

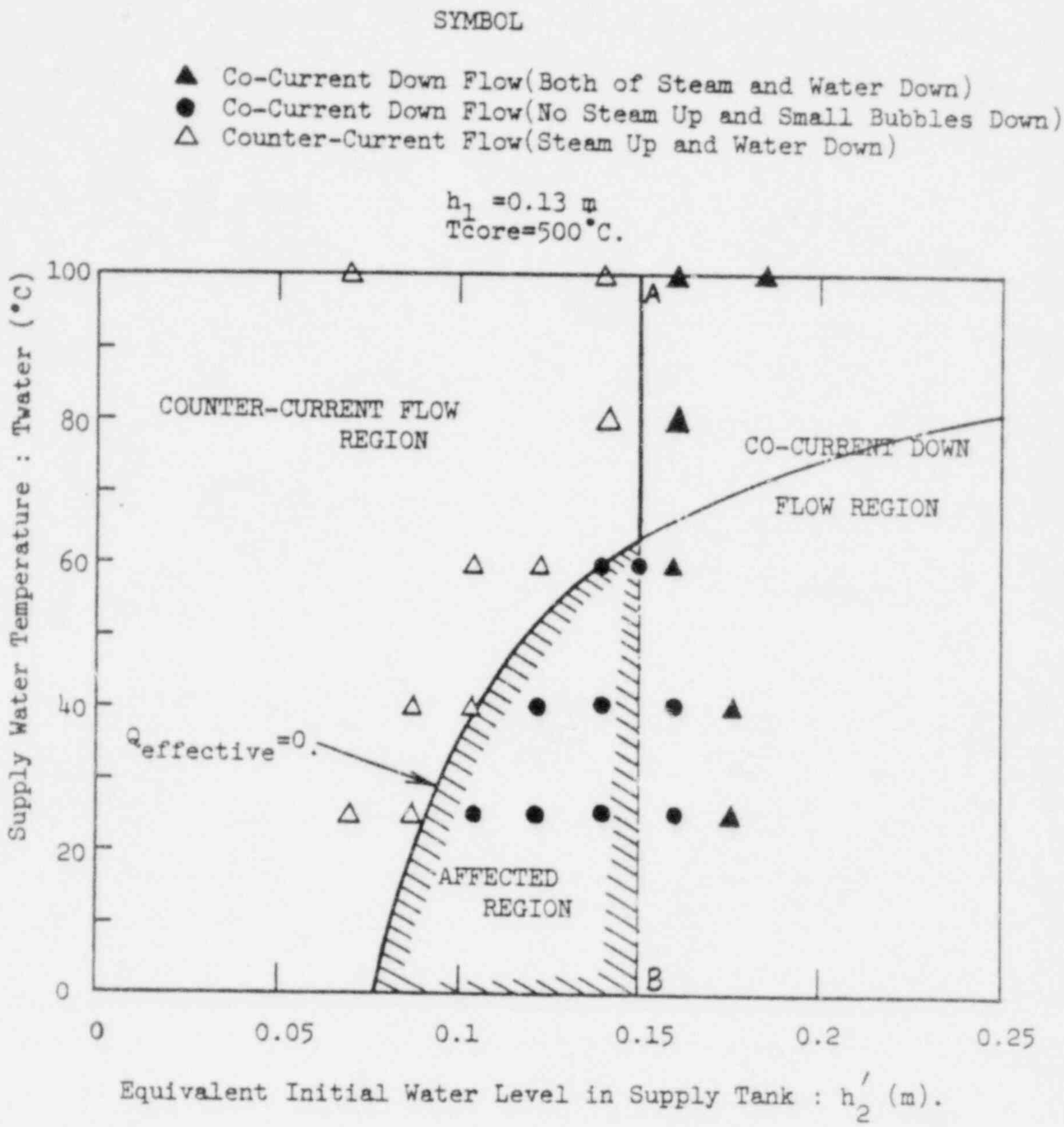


Figure 13. Comparison of the Prediction and Experimental Results for the Transition of Flow Pattern due to Water Subcooling in Gase of Initial Water Level in Lower Plenum, h_1=0.13 m and Initial Core Temperature, Tcore =500°C.

SYMBOL

▲ Co-Current Down Flow(Both of Steam and Water Down)
● Co-Current Down Flow(No Steam Up and Small Bubbles Down)
△ Counter-Current Flow(Steam Up and Water Down)

h_1 =0.06 m,
Tcore=500°C.

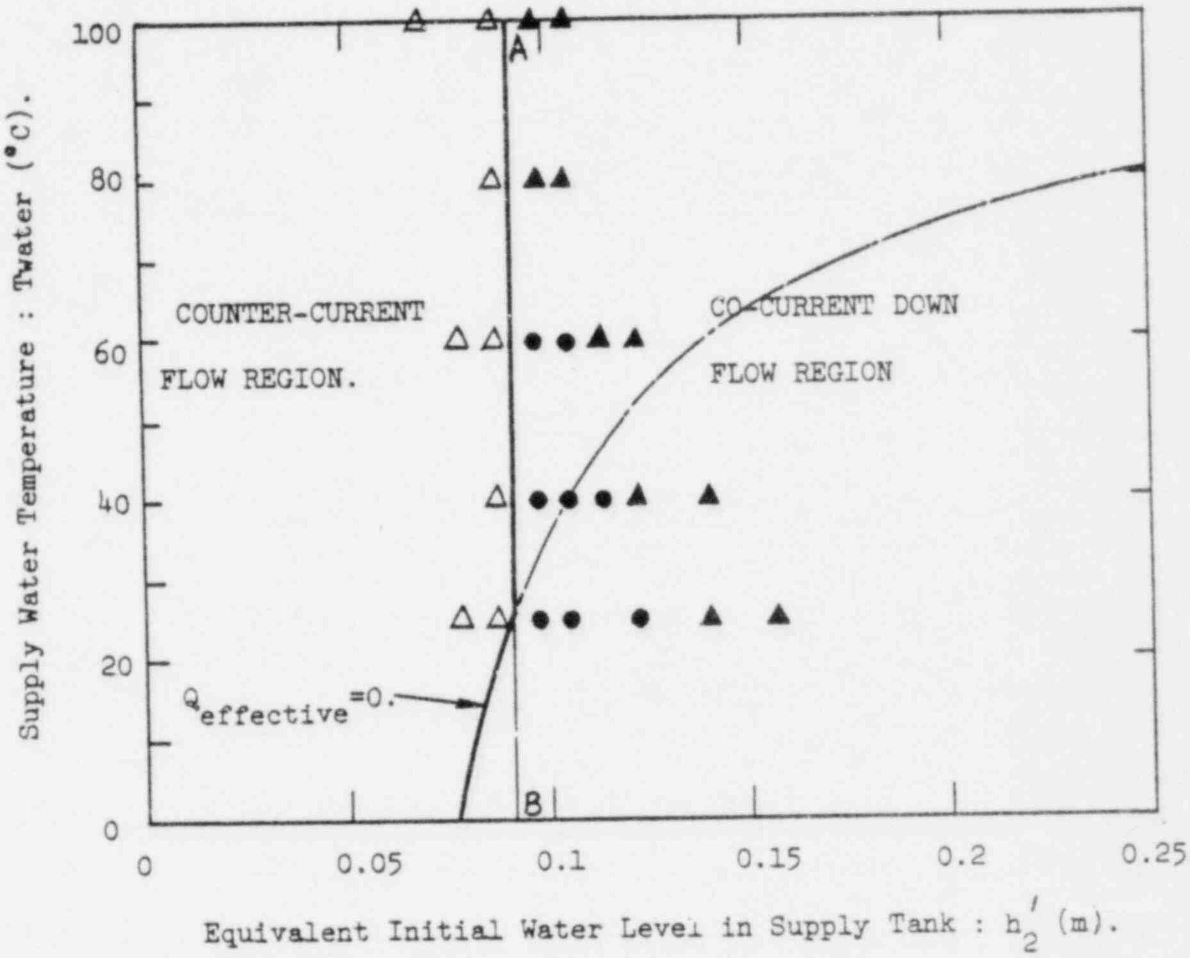


Figure 14. Comparison of the Prediction and Experimental Results for the Transition of Flow Pattern due to Water Subcooling in Case of Initial Water Level in Lower Plenum, h_1 =0.06 m and Initial Core Temperature,Tcore=500°C.

SUMMARY OF CONDENSATION STUDIES

Presented at 8th Water Reactor

Safety Research Information Meeting

October 28, 1980

by

S. G. Bankoff, R. S. Tankin, M. C. Yuen

Northwestern University

SUMMARY OF CONDENSATION RESEARCH AT NORTHWESTERN UNIVERSITY

S. G. Bankoff, R. S. Tankin, M. C. Yuen

The condensation studies being conducted at Northwestern University can be subdivided into four major categories:

1. Stratified Co-Current Horizontal Flow. These studies have been conducted over a wide range of water (.65 kg/sec to 1.45 kg/sec) and steam (.065 kg/sec to .16 kg/sec) flow rates. The inlet water temperature and water height has been varied. A semi-empirical relation for the average Nusselt number in terms of the inlet water and steam Reynolds numbers, degree of subcooled water at the inlet and location from entrance was obtained. An expression for the average Stanton number in terms of the average Reynolds numbers for water and steam and liquid Prandtl number was obtained for data with a wavy interface.

2. Downward Delivery of Water T[illegible] a Simulated Upper Tie Plate. The work that was reported on at the [illegible] Water Reactor Safety Research Information Meeting has been extended to include inertial effects of the jet injection. In this study the nozzle for injecting the water above the upper tie plate is positioned vertically rather than horizontally. At high nozzle positions, the vertical and horizontal configuration yield similar results. For low nozzle positions (near tie plate) the slope of the EOCB curve is much steeper for vertical injection as compared to horizontal injection. The vertical injection results in higher mixing efficiency. Preliminary results indicate that for vertical injection at total dumping, $H^*_{f,d}$ is independent of $H^*_{f,in}$ and depends on the inlet water injection height.

3. Holographic Study of Sprays. A pulsed laser is used to study droplet size distribution of water sprays in an air and steam environment. The resolution obtained from the hologram is better than 100μ. In tests conducted so far,

the spray angle was reduced approximately 25% (from 60° to 47°) when operated in a steam environment as compared to air environment. In addition to change in cone angle, the break-up of droplets occurs sooner in a steam environment. The obtained in these preliminary experiments are compared with the Rosin-Rammler equation and the Nukiyama-Tanasawa equation. At present, the data (droplet location and dimensions) are introduced into a minicomputer and plots are obtained of average droplet size, surface area and volume as a function of radius.

4. Vertical Counter Current Steam-Water Flow in a Flat Plate Geometry. A new test facility has been designed and built to study countercurrent steam-water flow. To eliminate water-steam interaction in the lower plenum, water exiting the test section is sucked from the test section through a porous medium. The instrumentation for this facility is similar to that used in the horizontal co-current channel-thermocouples, pressure transducers, pitot tubes. At present, we are working on a resitivity probe to measure interfacial structure. The data obtained from the earlier facility was compared with the published work of Segev and Collier. Our results are for the channel at 83° and 0° (from the horizontal); whereas Segev and Collier's data are for 17° and 45°. An expression for the average Nusselt number is empirically obtained which is dependent on the inlet water and inlet steam Reynolds numbers, the degree of subcooling of the water, and the distance from the water entrance.

Publications.

1. "Counter Current Flow of Air/Water and Steam/Water Through a Horizontal Perforated Plate" submitted for publication to International Journal of Heat and Mass Transfer.
2. "Stratified Co-Current Flow of Steam/Water in a Horizontal Channel" being prepared for submitting to journal.

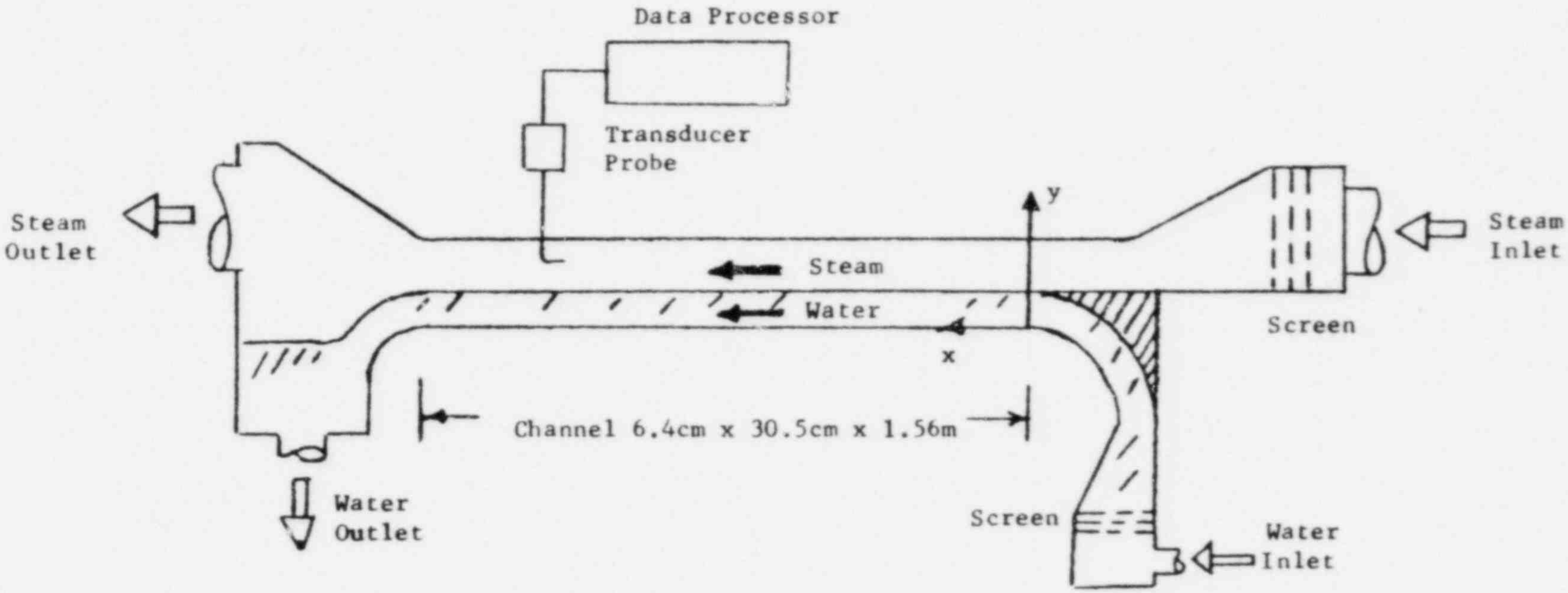


Fig.1-1 Schematic of Horizontal Channel

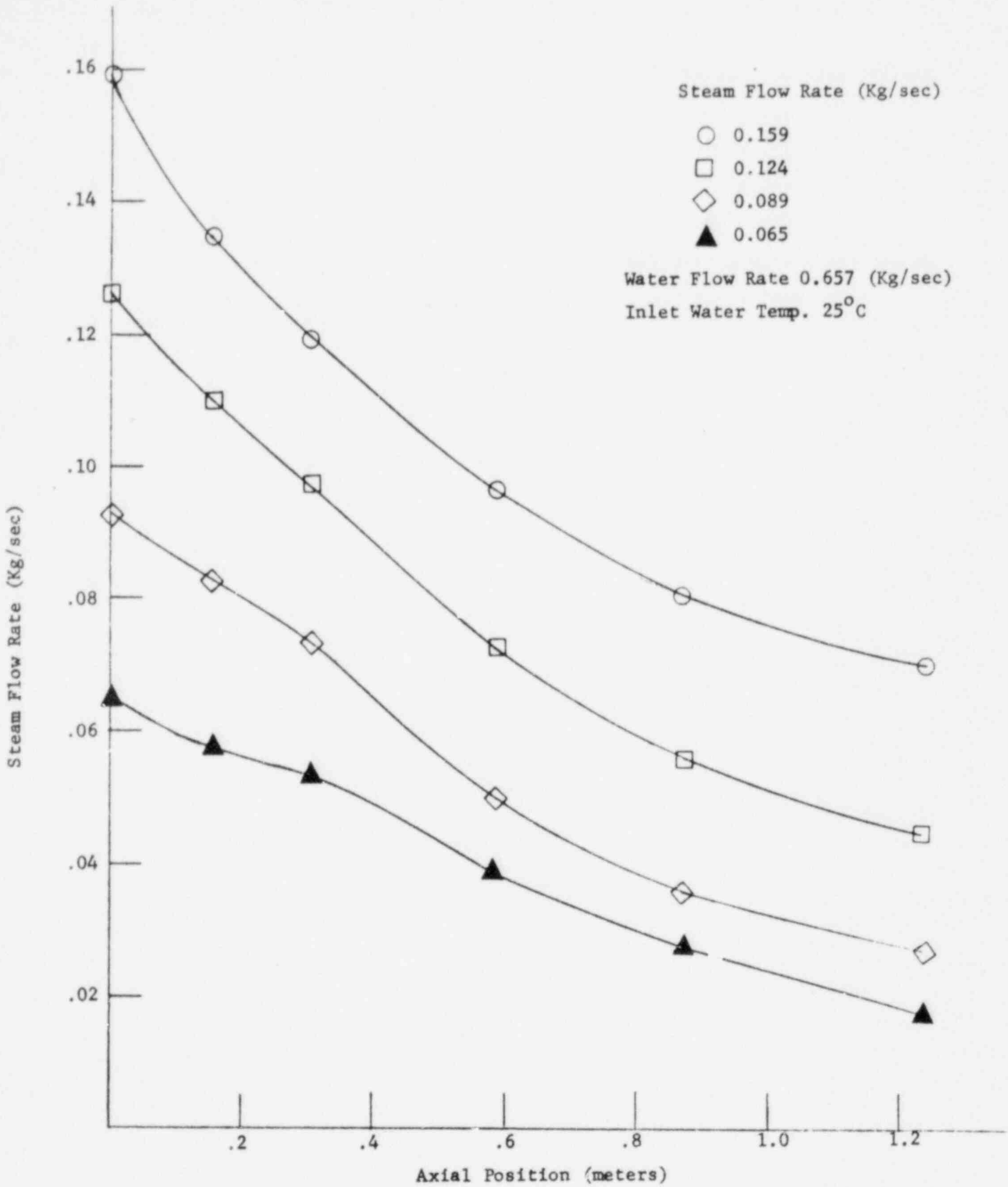


Figure 1-7. Steam flow rate in channel at low water flow rate.

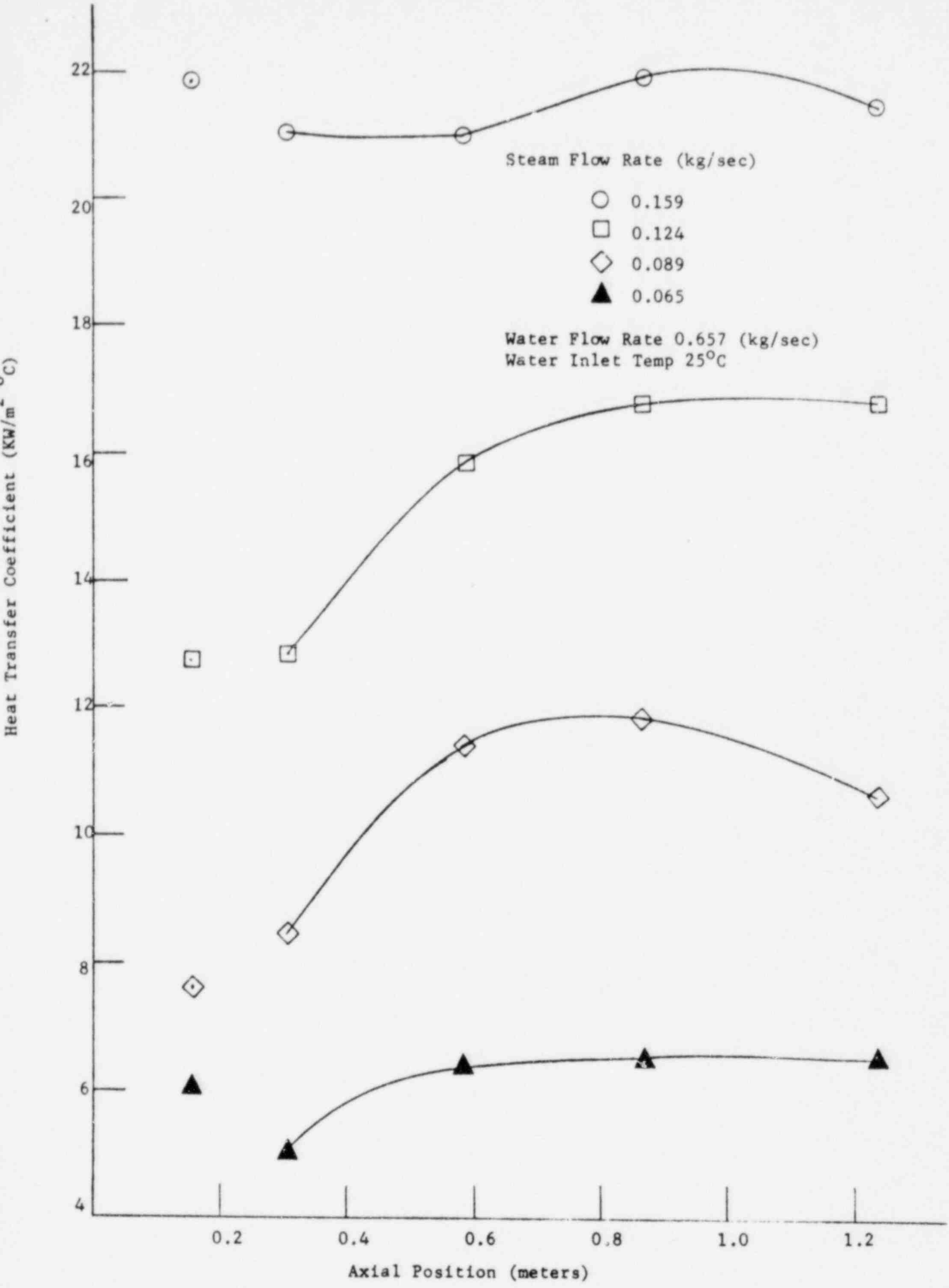


Figure 1-9 Average heat transfer coefficient along channel (smooth interface at entrance for low steam flow rates)

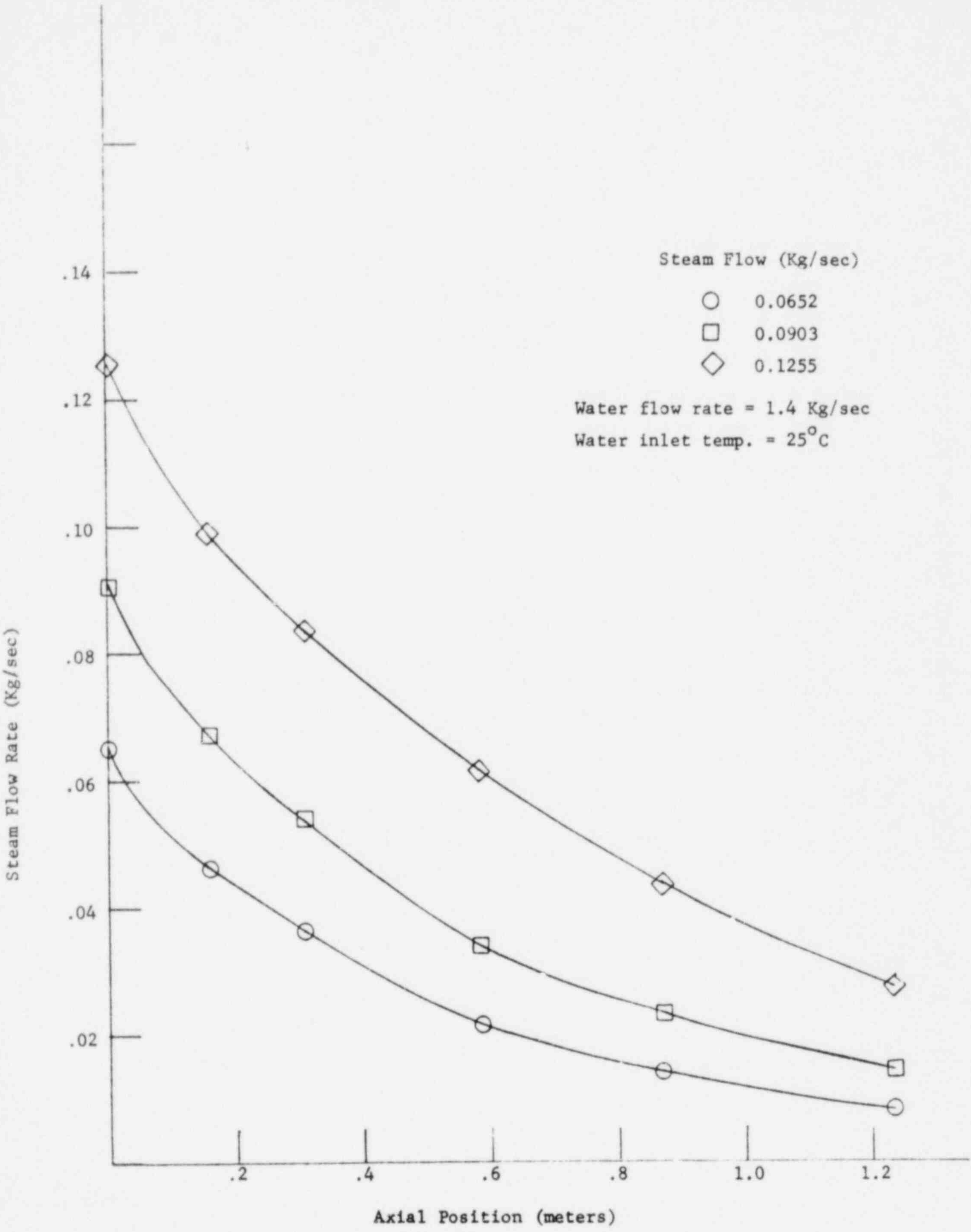


Figure I-10 Steam flow rate in channel

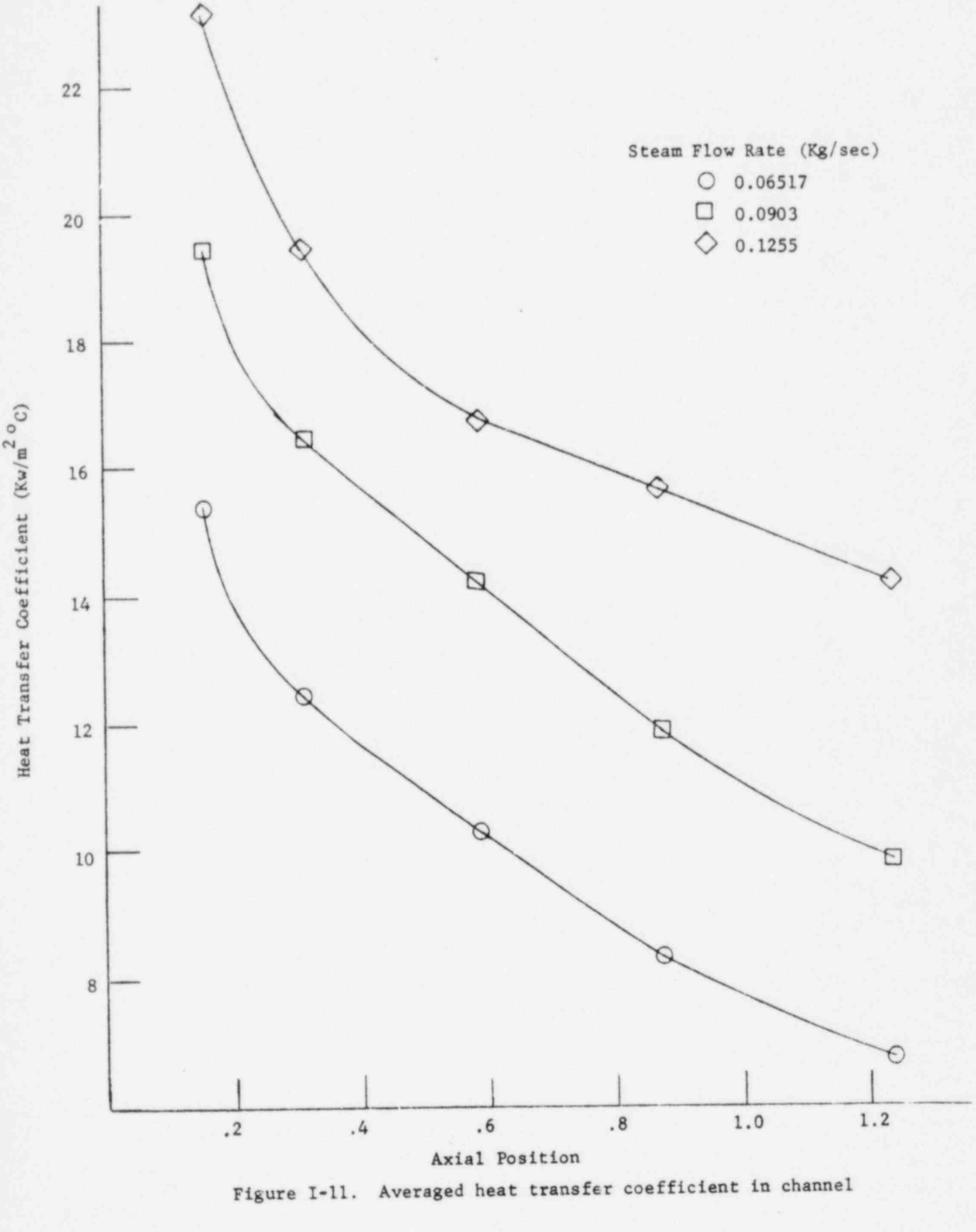


Figure I-11. Averaged heat transfer coefficient in channel

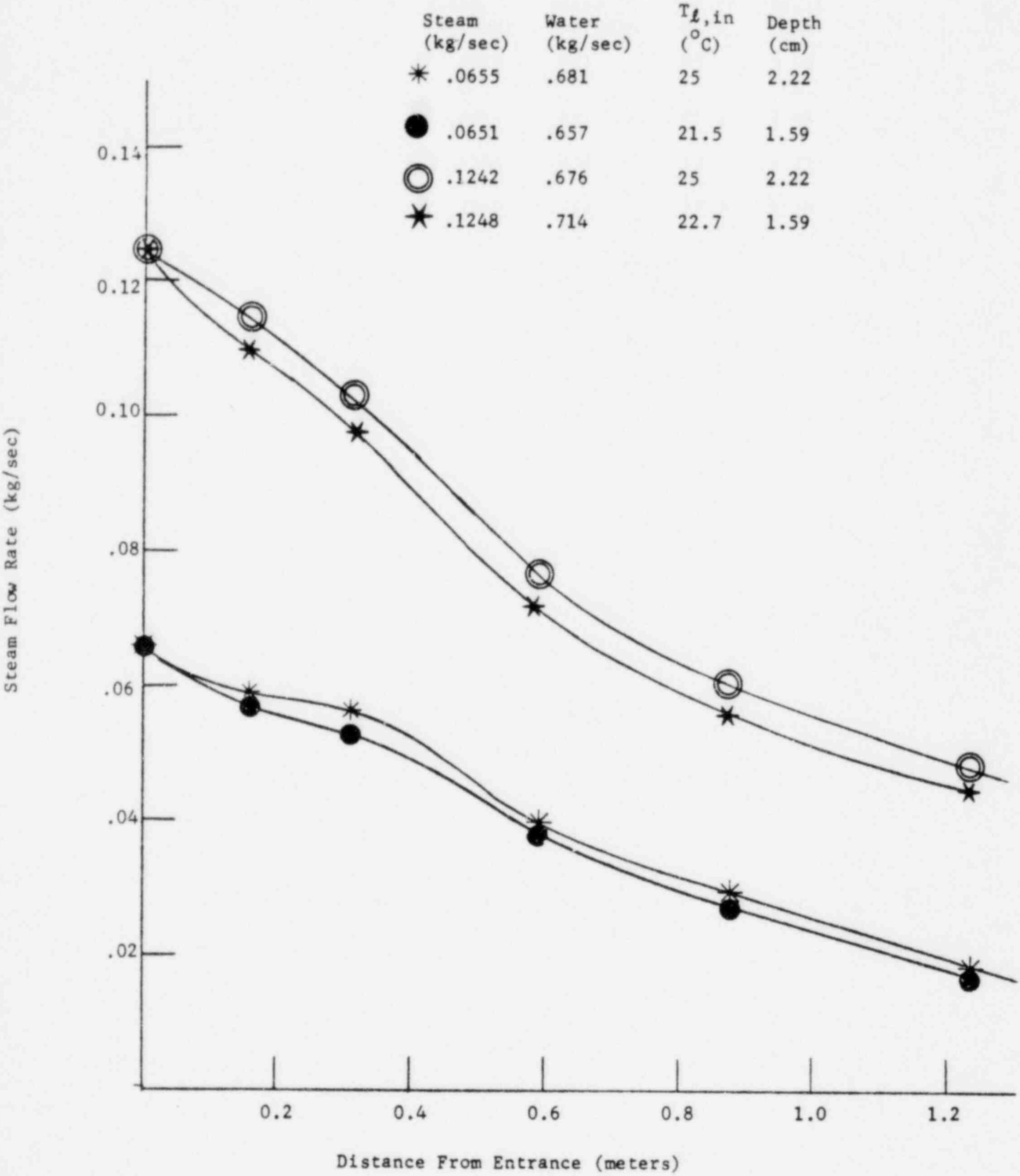


Figure 1-14 Steam flow rate along axial position with high and low entrance depths

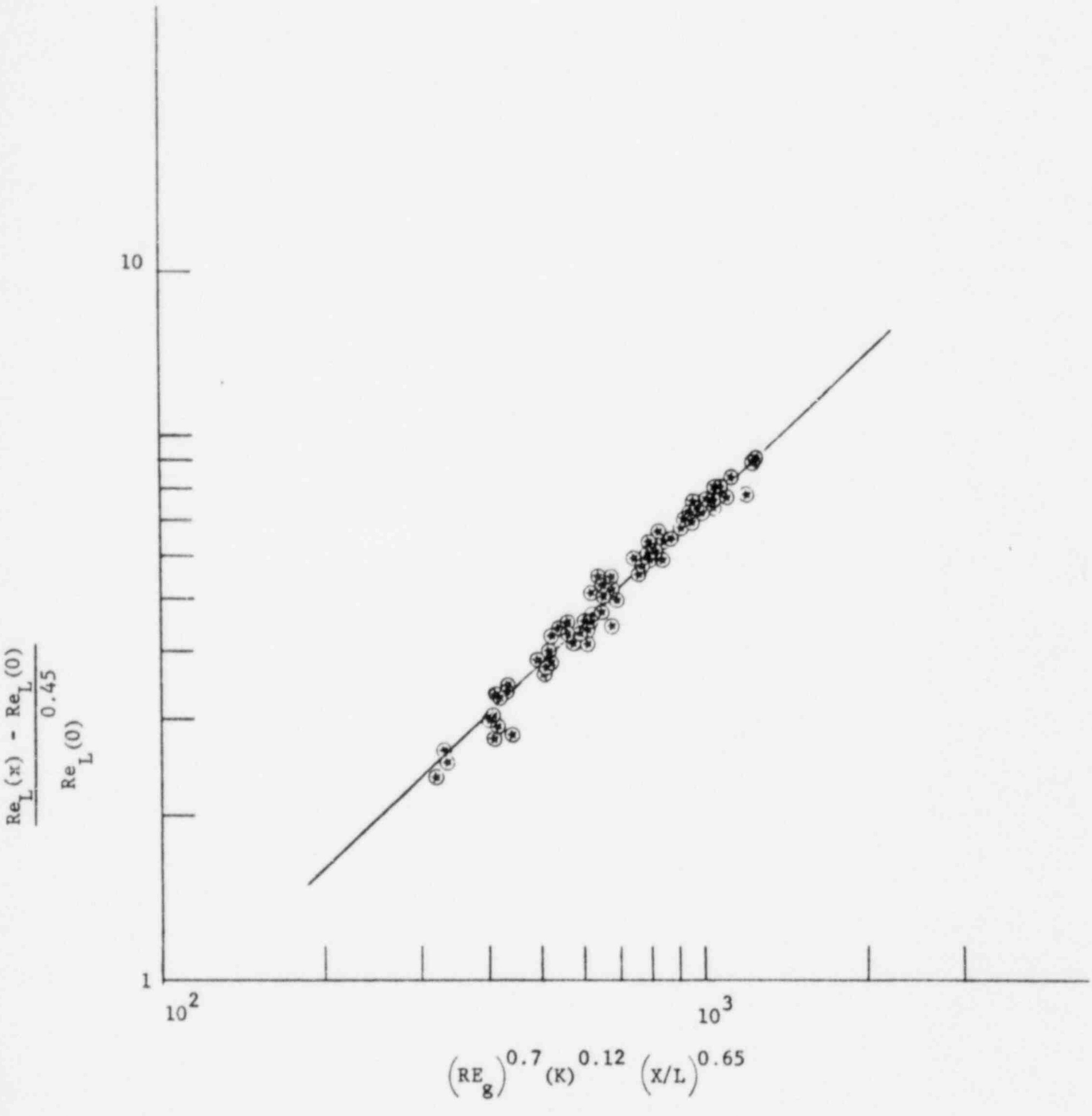


Figure 1-15. Data from 29 tests to determine correlation between Re_L, Re_g, K, and X/L.

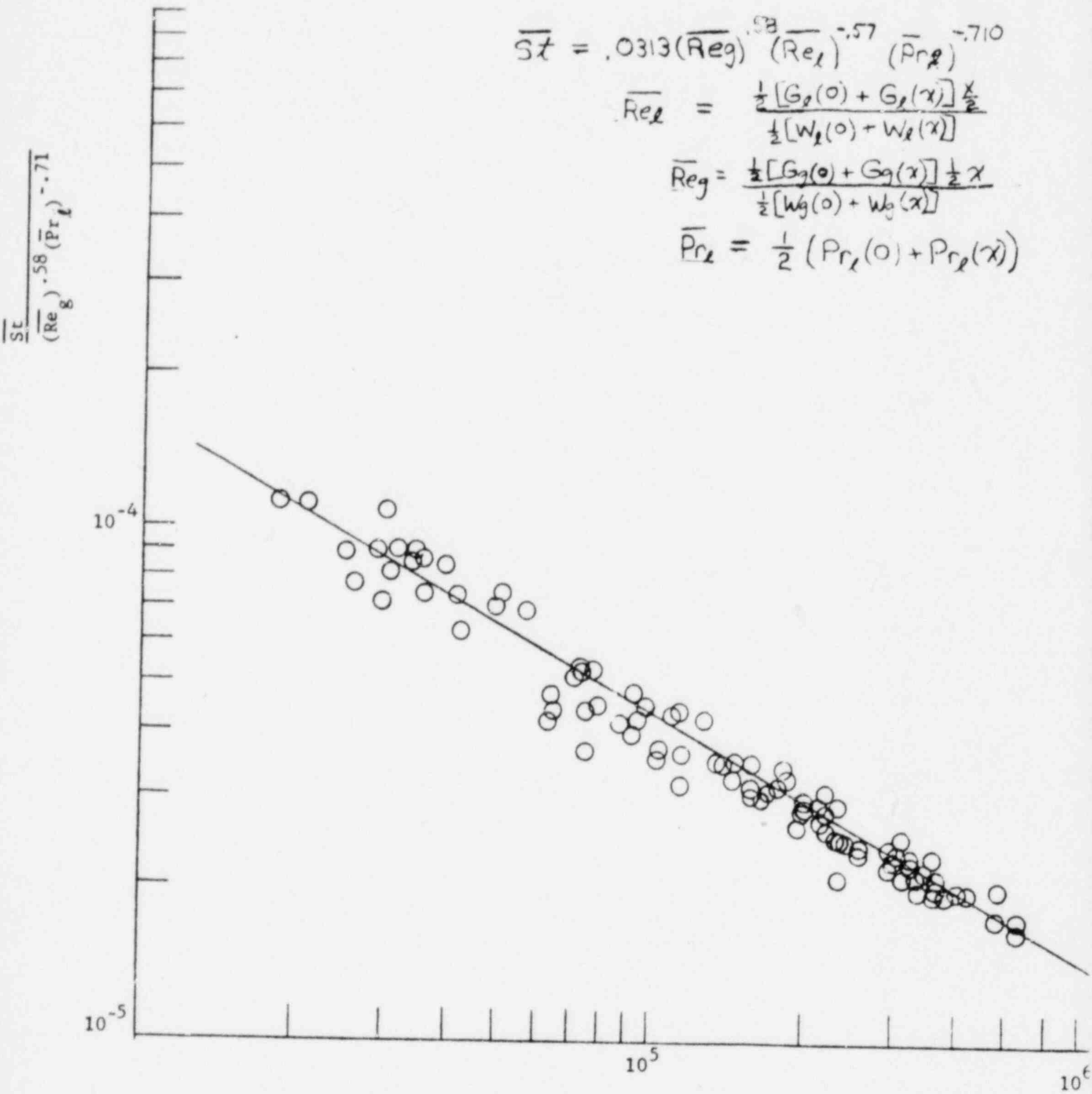


Figure 1-16 Correlation for wavy interface data

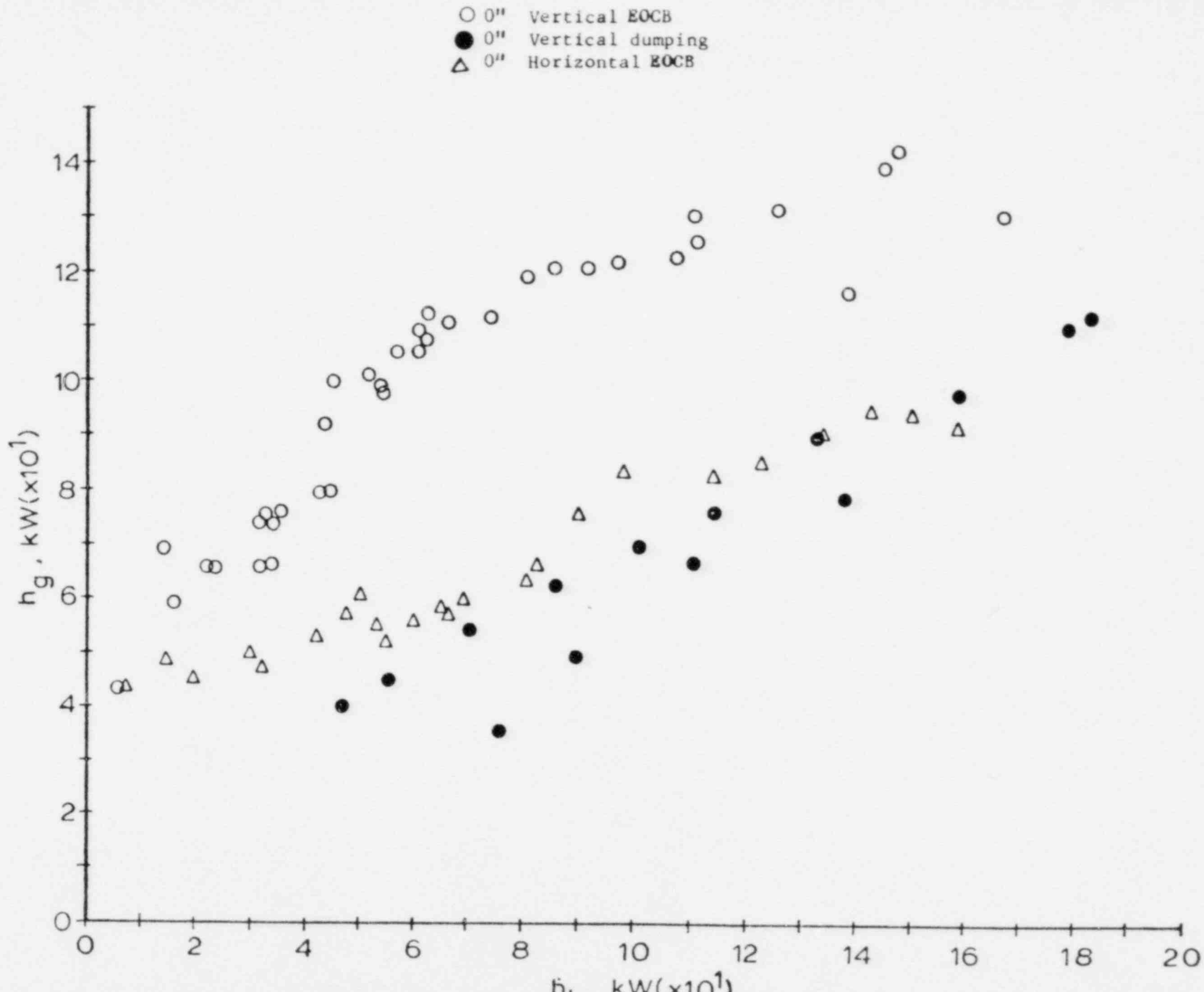


Fig. 6 Total Enthalpy Flux for Vertical Water Injection Directly Through The Hole and Horizontal Water Injection Just Above The Plate.

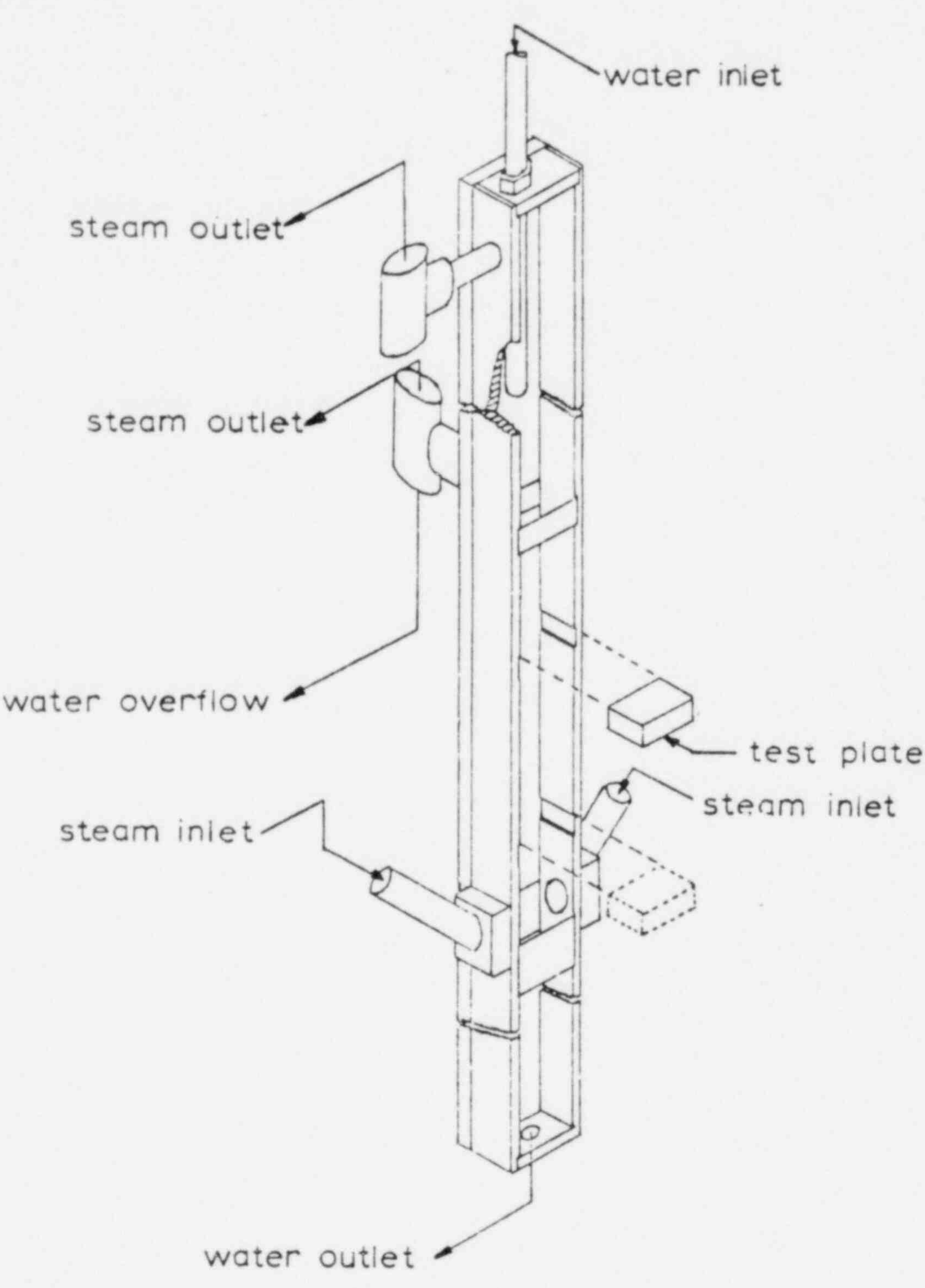


Fig. 1 Schematic Diagram of Apparatus.

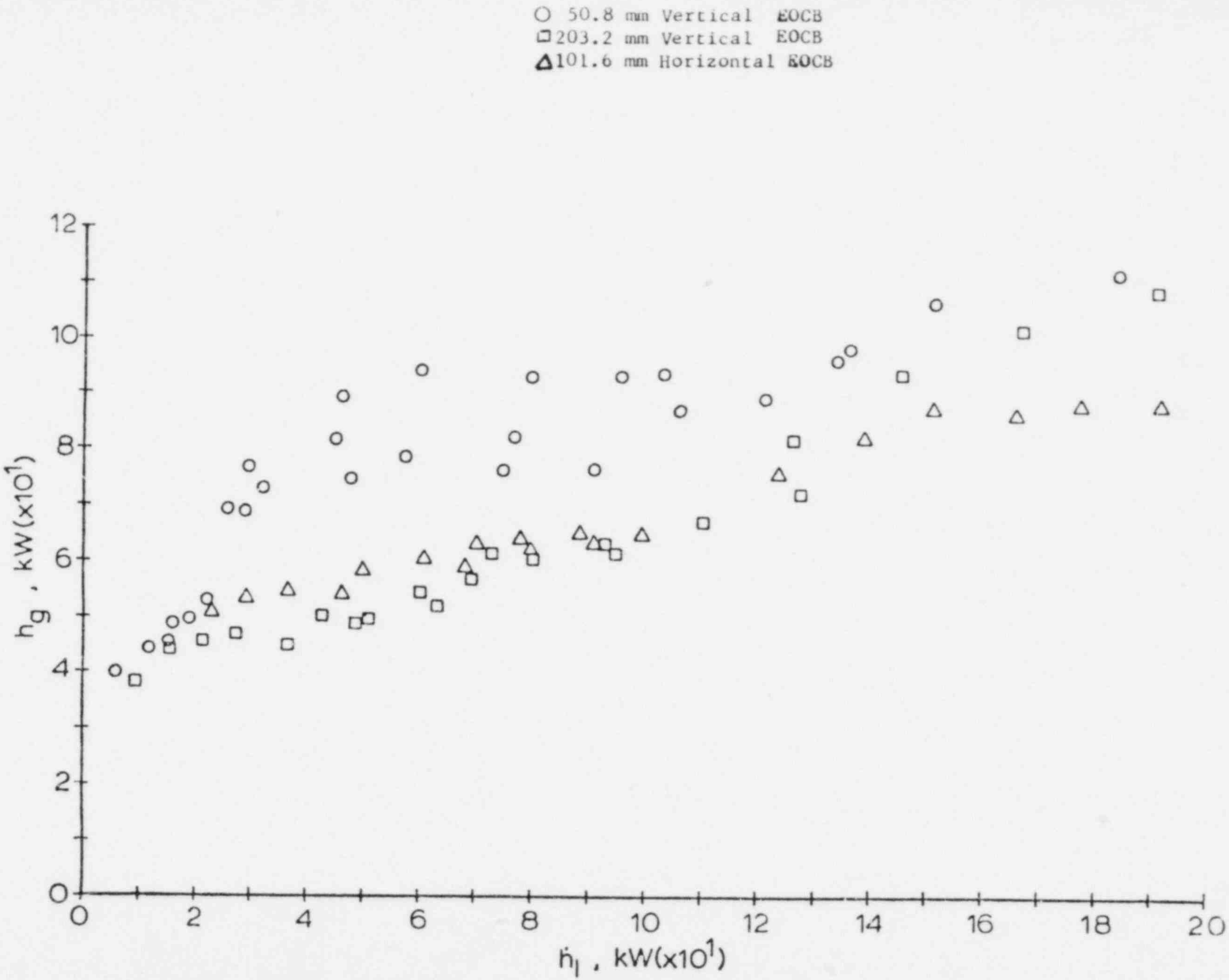


Fig. 5 Comparison of Enthalpy Flux at EOCB Between Horizontal and Vertical Low Water Injection.

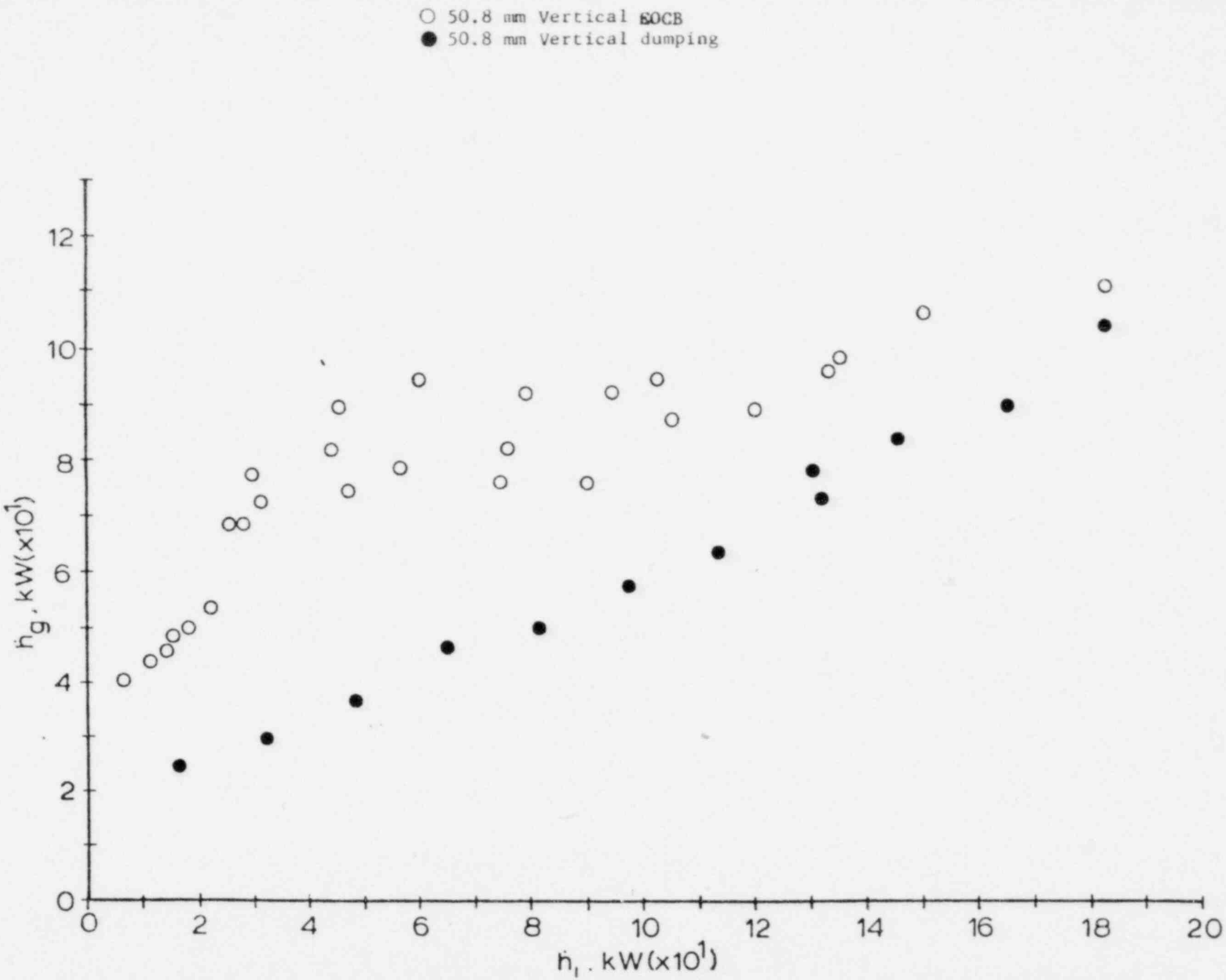


Fig. 4 Total Enthalpy Flux at Weeping and Dumping Thresholds for Low Vertical Water Injection.

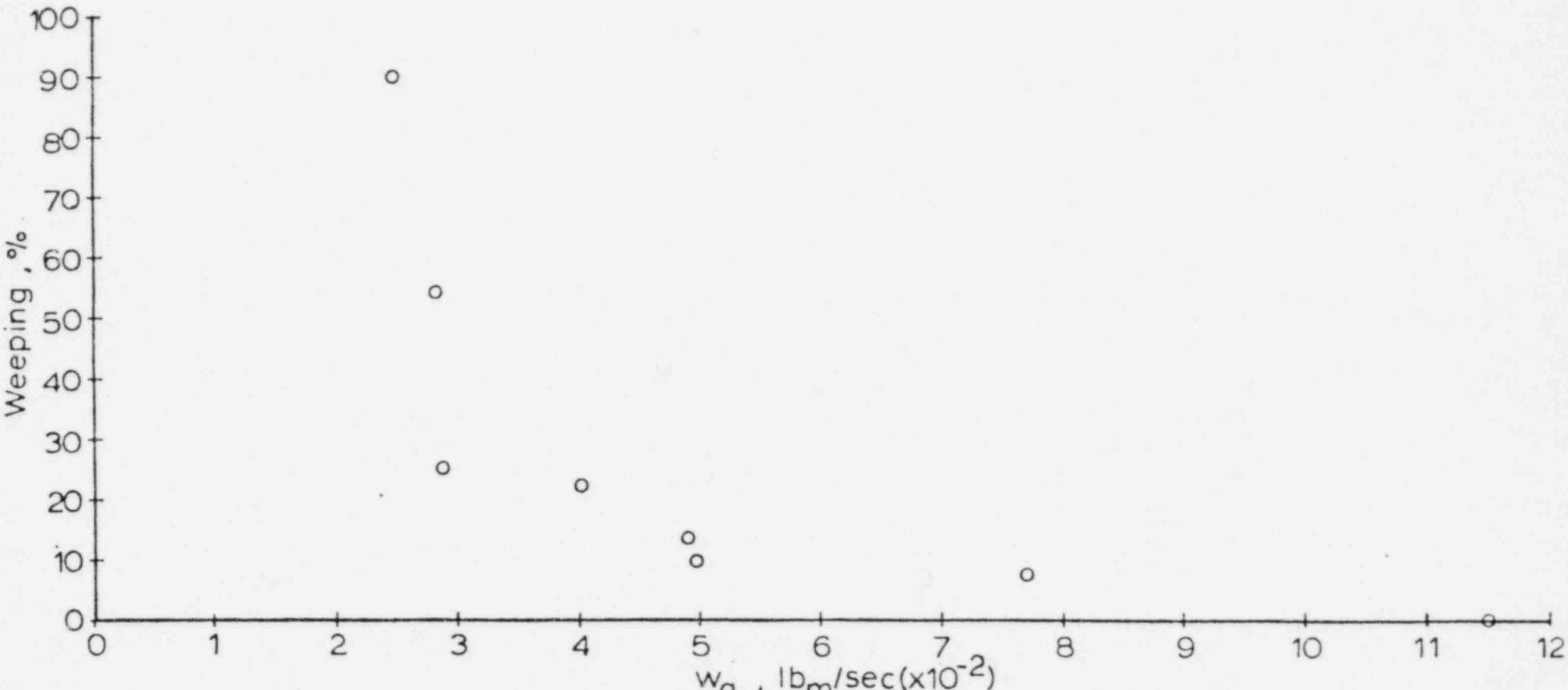


Fig. 7 Weeping Rate at Low Inlet Water Flow Rate (37.8 g/sec) and High Injection Position (203.2 mm).

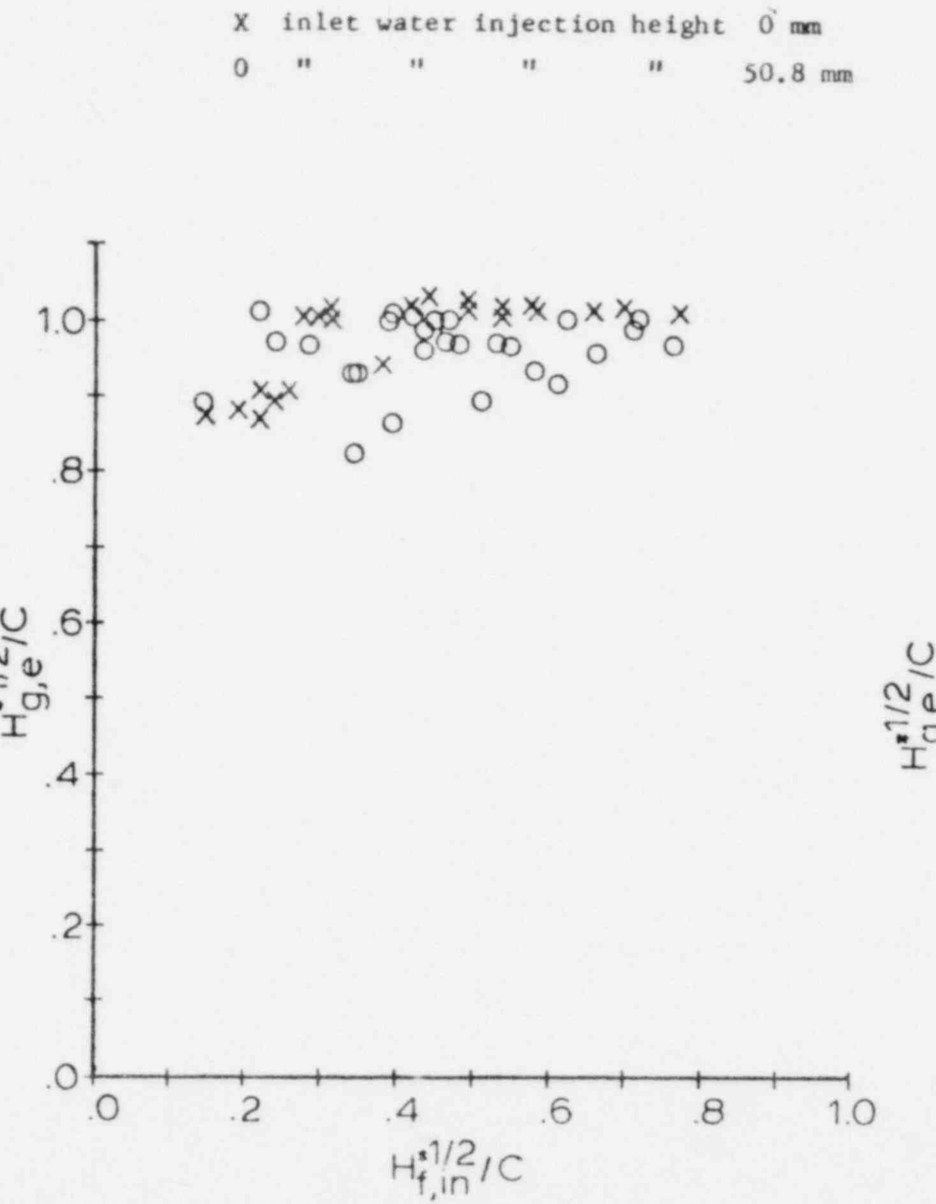


Fig. 8a Correlation of dimensionless effective steam flow rate at weep point, for water injected at 0 mm and 50.8 mm above the test plate.

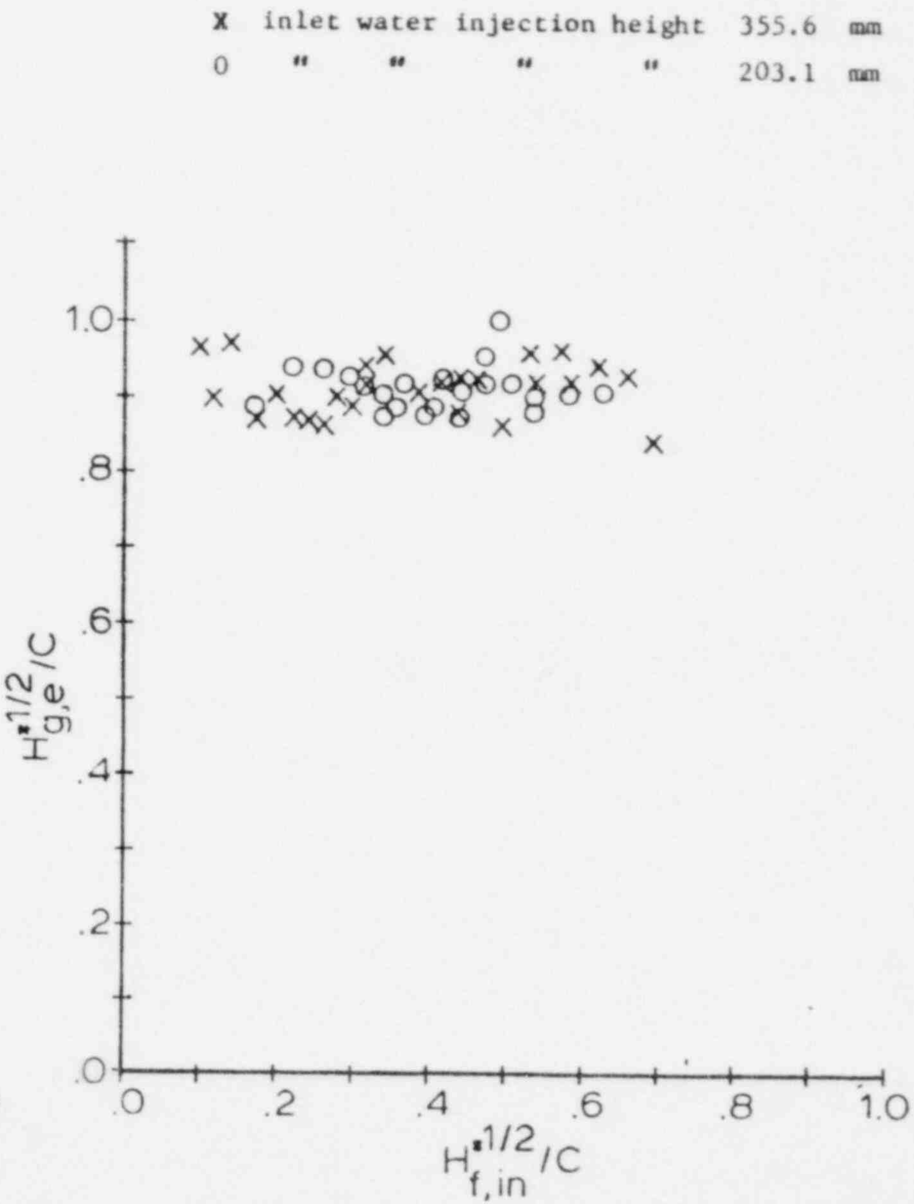


Fig. 8b Correlation of dimensionless effective steam flow rate at weep point, for water injected at 203.2 mm and 355.6 mm above the test plate.

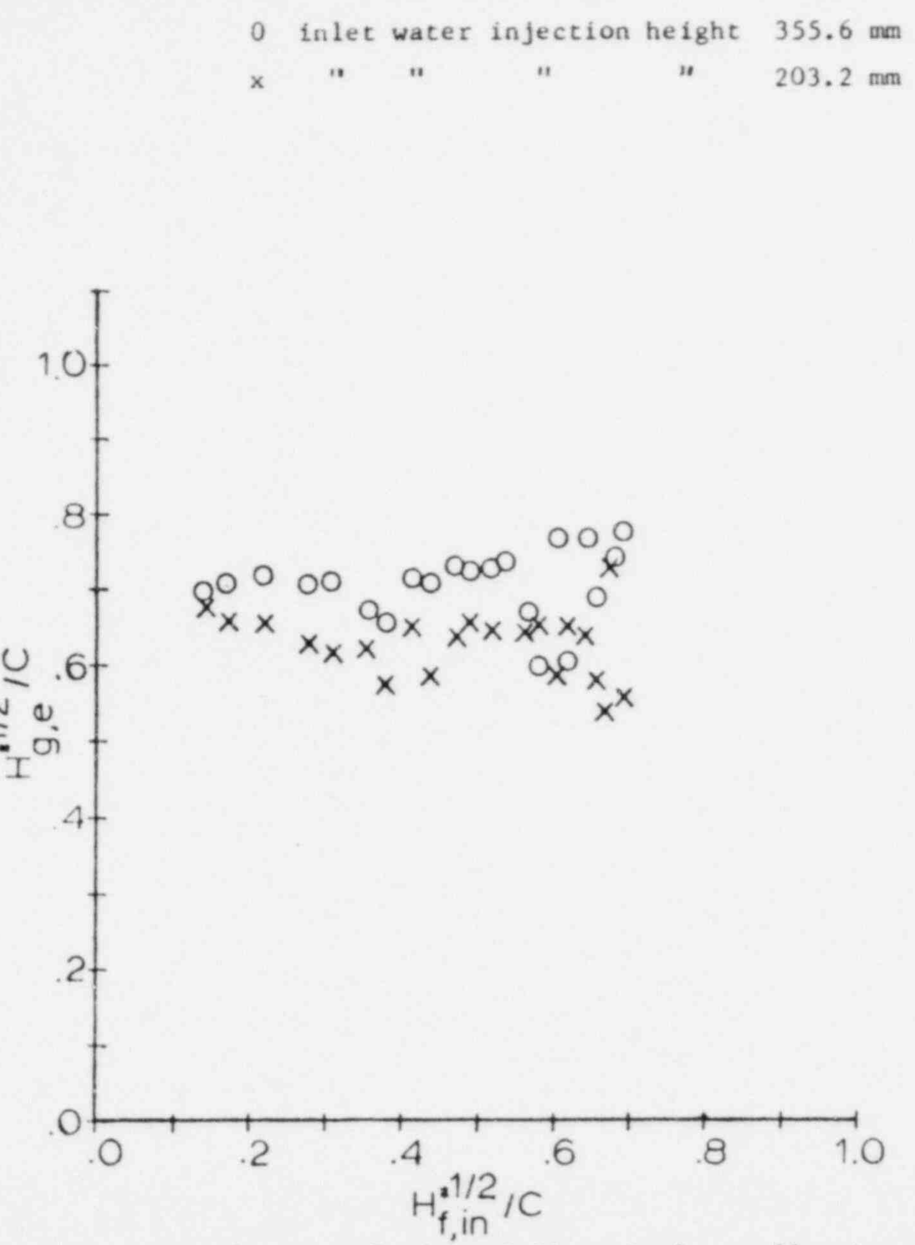


Fig. 9a Correlation of dimensionless effective steam flow rate of total dumping, for water injected at 355.6 mm and 203.2 mm.

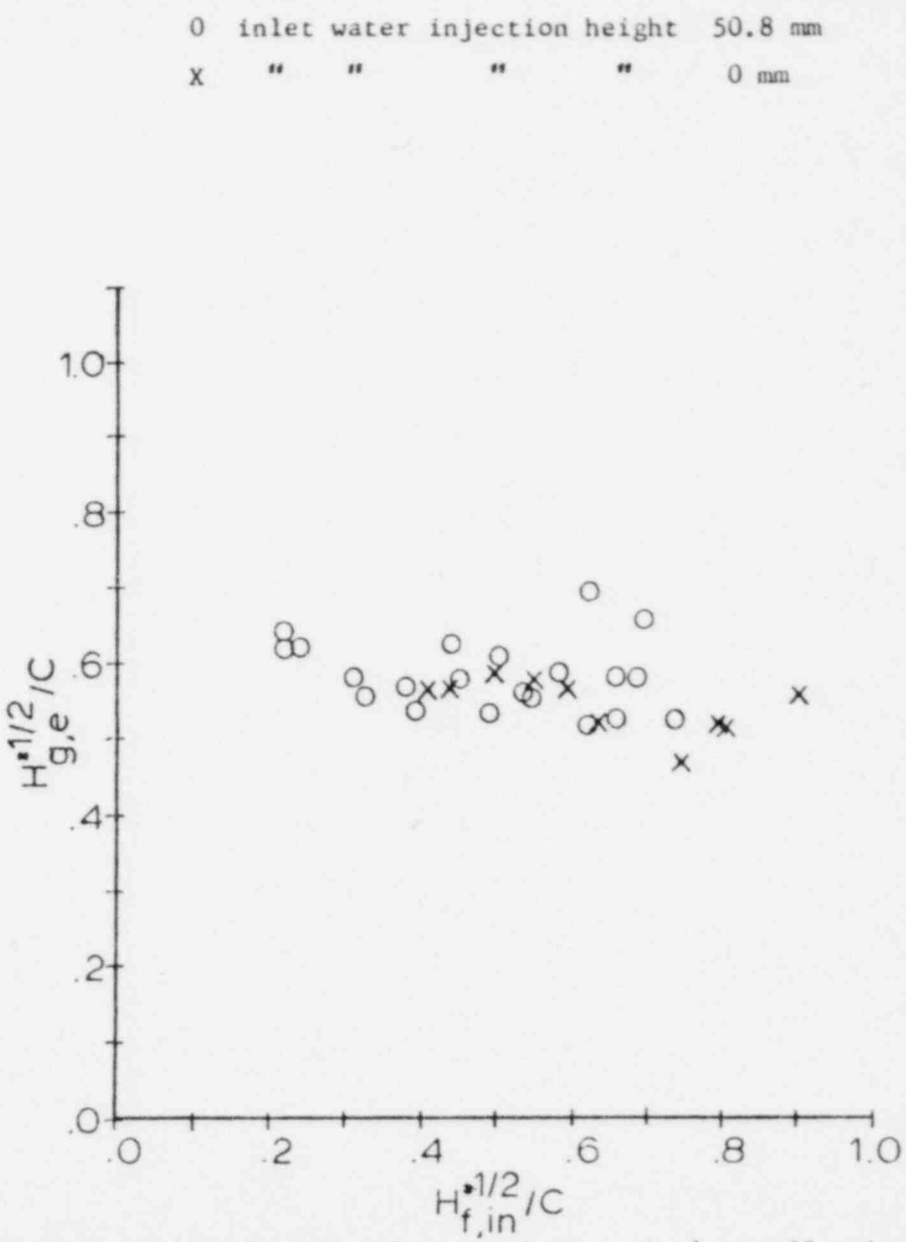


Fig. 9b Correlation of dimensionless effective steam flow rate at total dumping for water injected at 50.8 mm and 0 mm.

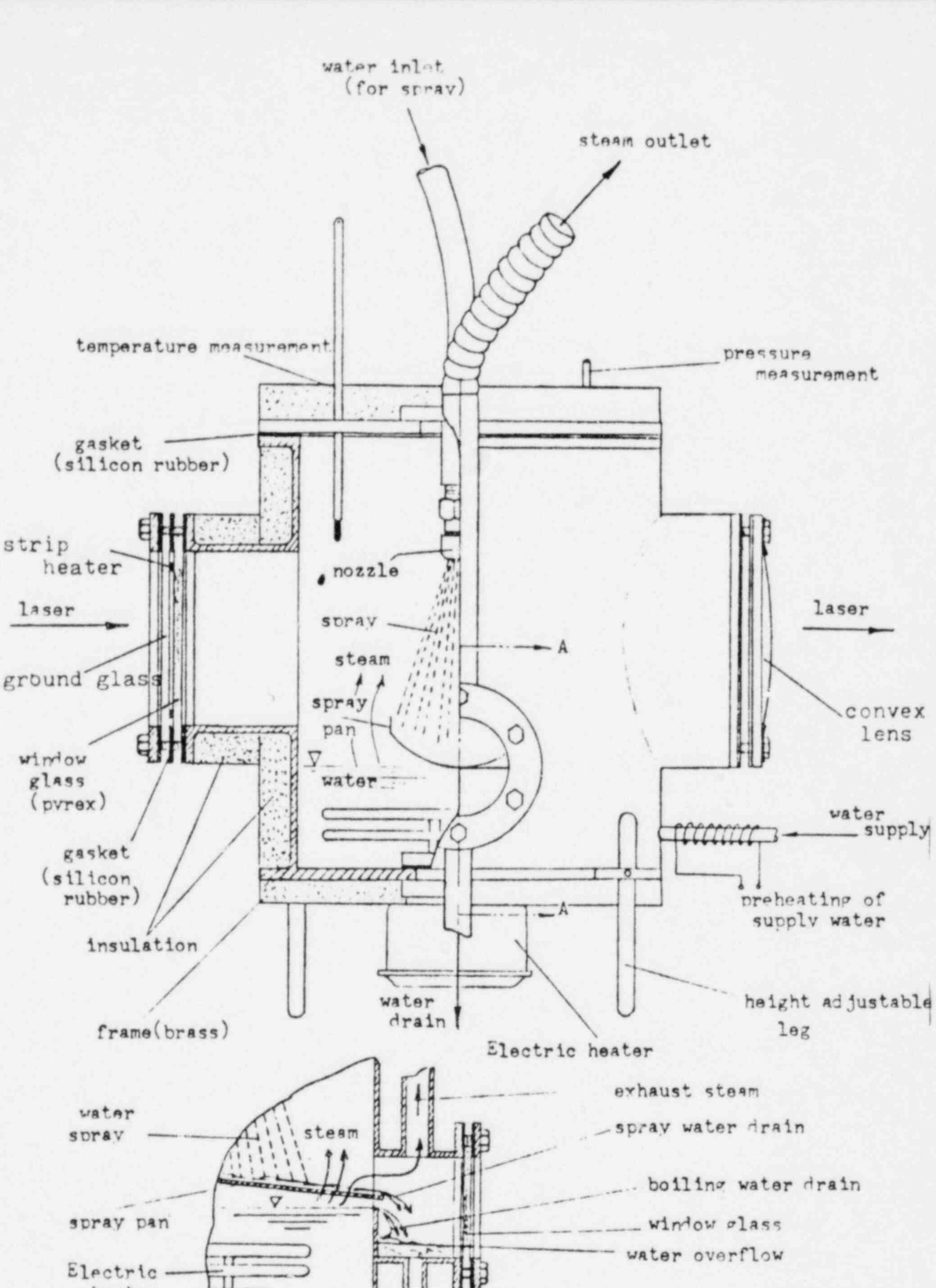


Fig. III-2 Sketch of the Test-section

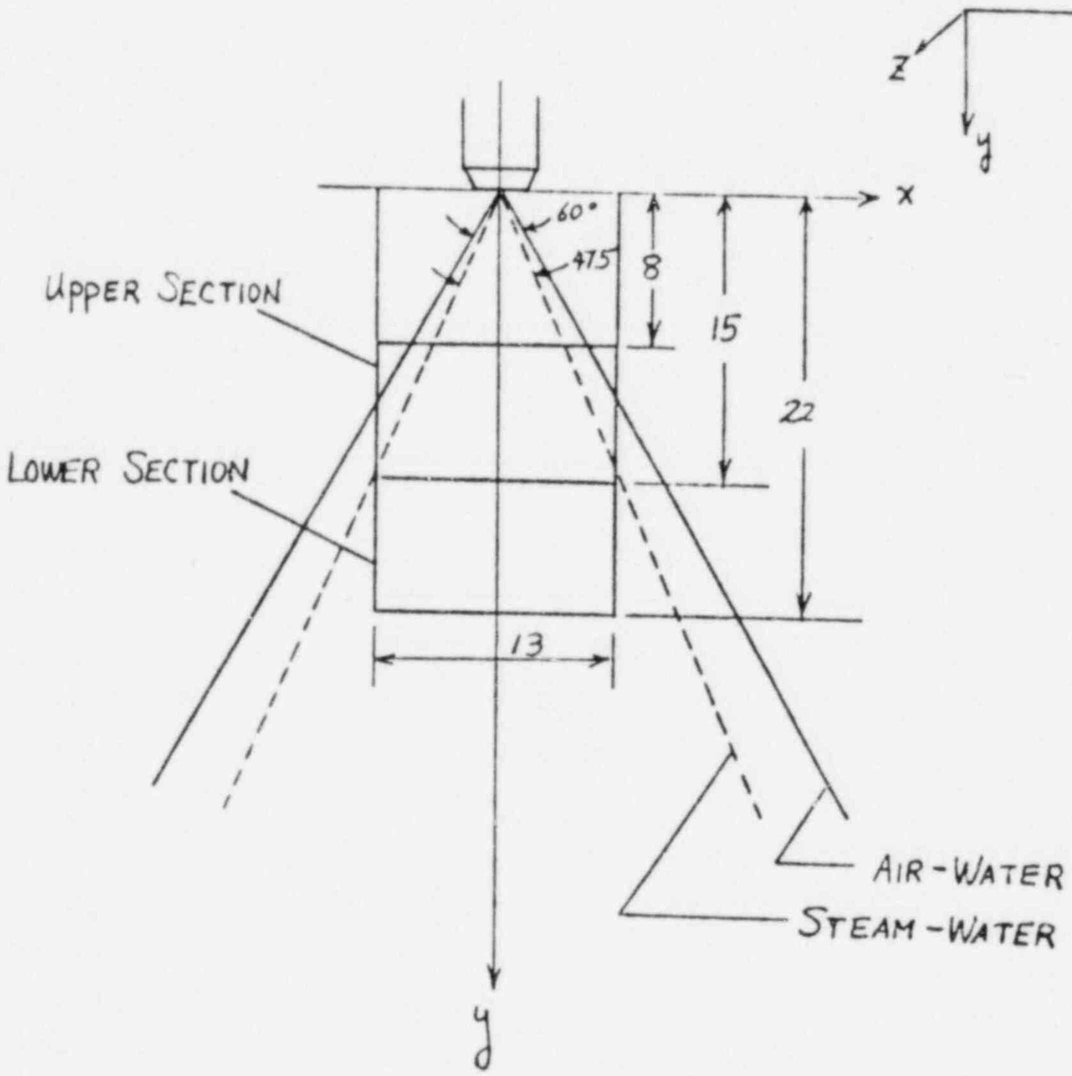
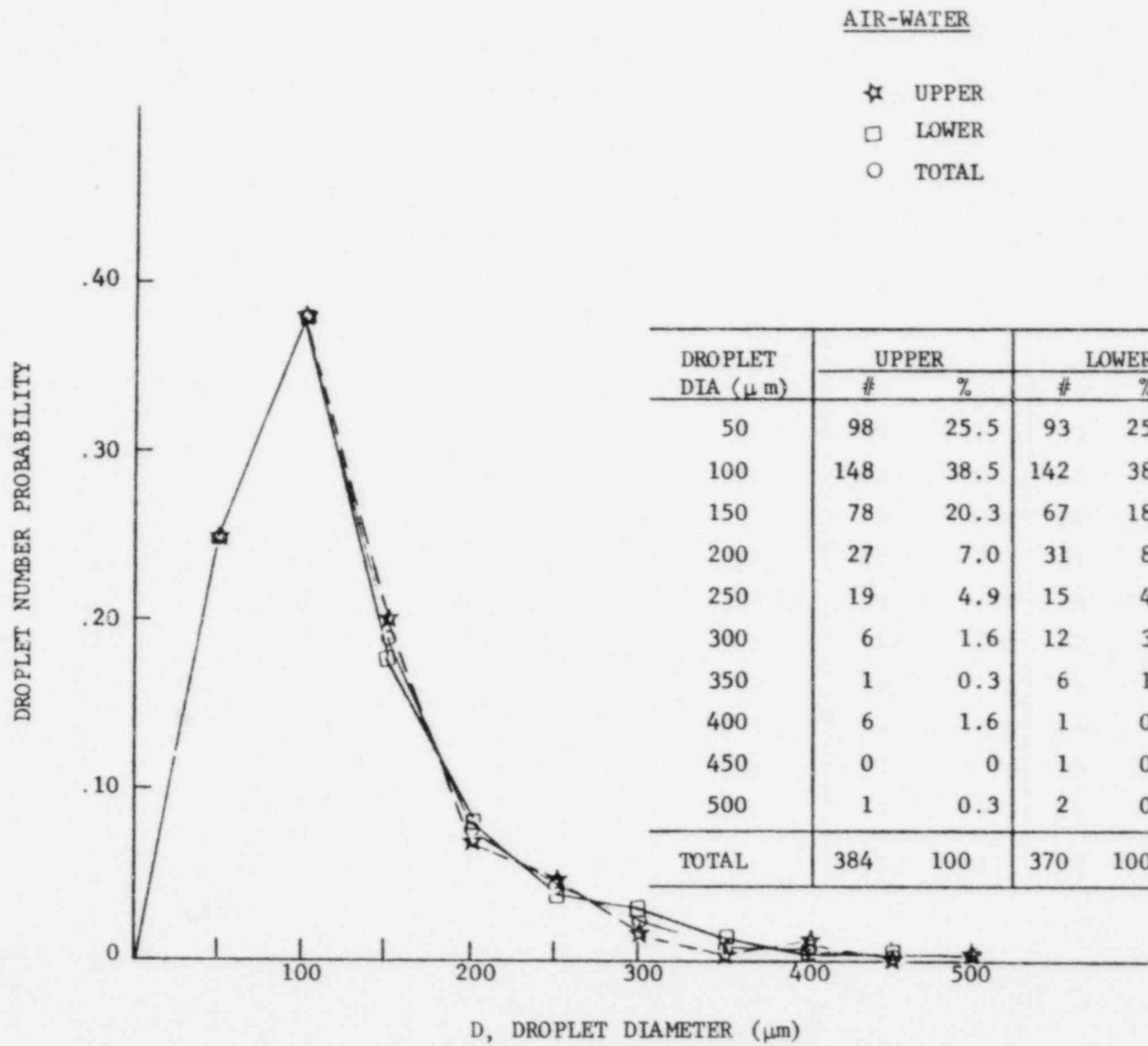
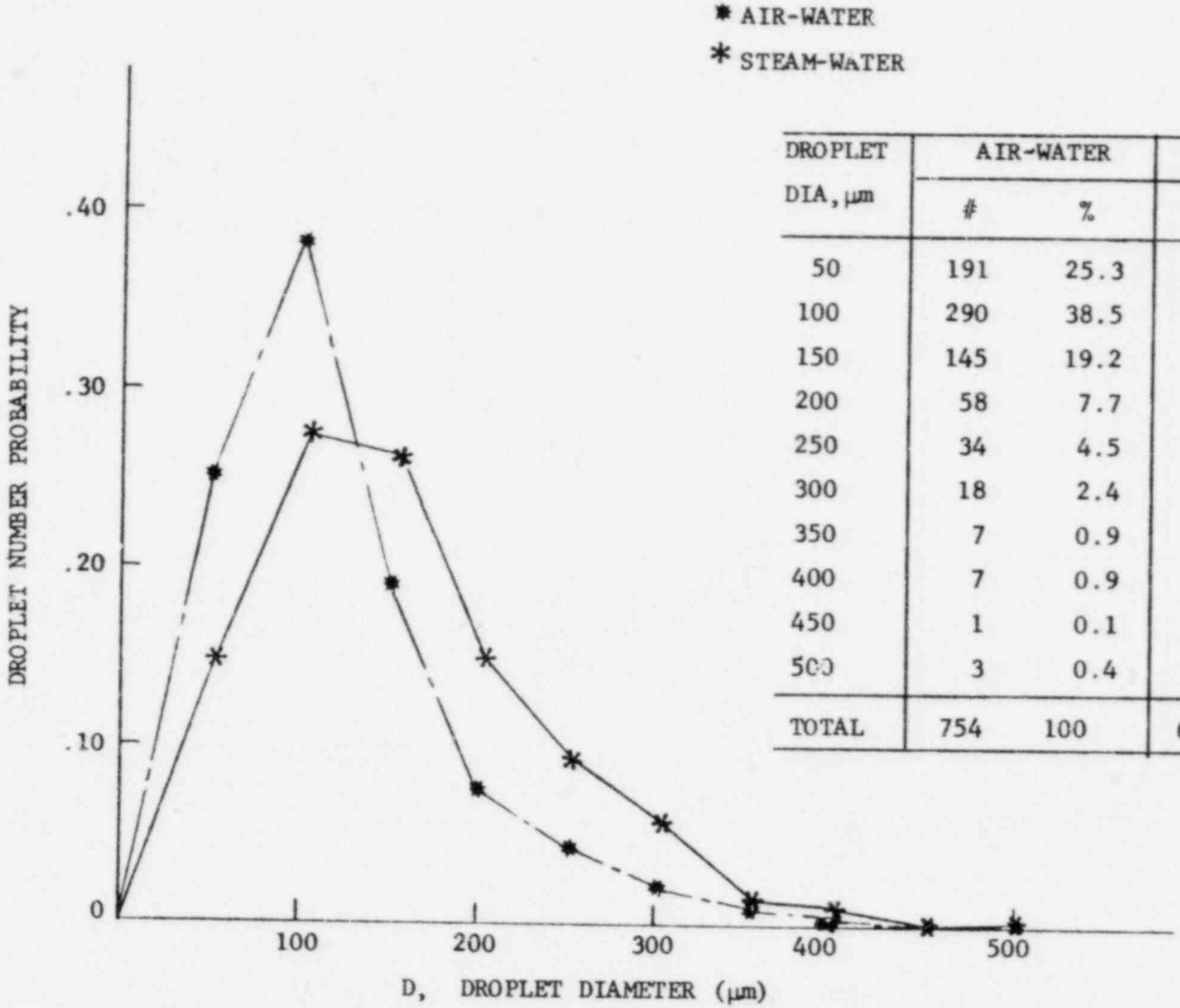


FIG. III-7 Schematic diagram showing regions where holograms were taken.



DROPLET DIA (μm)	UPPER #	UPPER %	LOWER #	LOWER %	TOTAL #	TOTAL %
50	98	25.5	93	25.1	191	25.3
100	148	38.5	142	38.4	290	3[illegible].5
150	78	20.3	67	18.1	145	19.2
200	27	7.0	31	8.4	58	7.7
250	19	4.9	15	4.1	34	4.5
300	6	1.6	12	3.2	18	2.4
350	1	0.3	6	1.6	7	0.9
400	6	1.6	1	0.3	7	0.9
450	0	0	1	0.3	1	0.1
500	1	0.3	2	0.5	3	0.4
TOTAL	384	100	370	100	754	100

FIG. III-8 DROPLET NUMBER PROBABILITY VS DROPLET DIAMETER (WATER SPRAY INJECTED INTO AIR, HOL.#19)



DROPLET DIA, μm	AIR-WATER #	AIR-WATER %	STEAM-WATER #	STEAM-WATER %
50	191	25.3	92	14.6
100	290	38.5	170	27.1
150	145	19.2	163	26.0
200	58	7.7	93	14.8
250	34	4.5	57	9.1
300	18	2.4	36	5.7
350	7	0.9	9	1.4
400	7	0.9	7	1.1
450	1	0.1	0	0
500	3	0.4	1	0.2
TOTAL	754	100	628	100

FIG. III-10 COMPARISON OF DROPLET NUMBER PROBABILITY BETWEEN AIR AND STEAM ENVIRONMENT

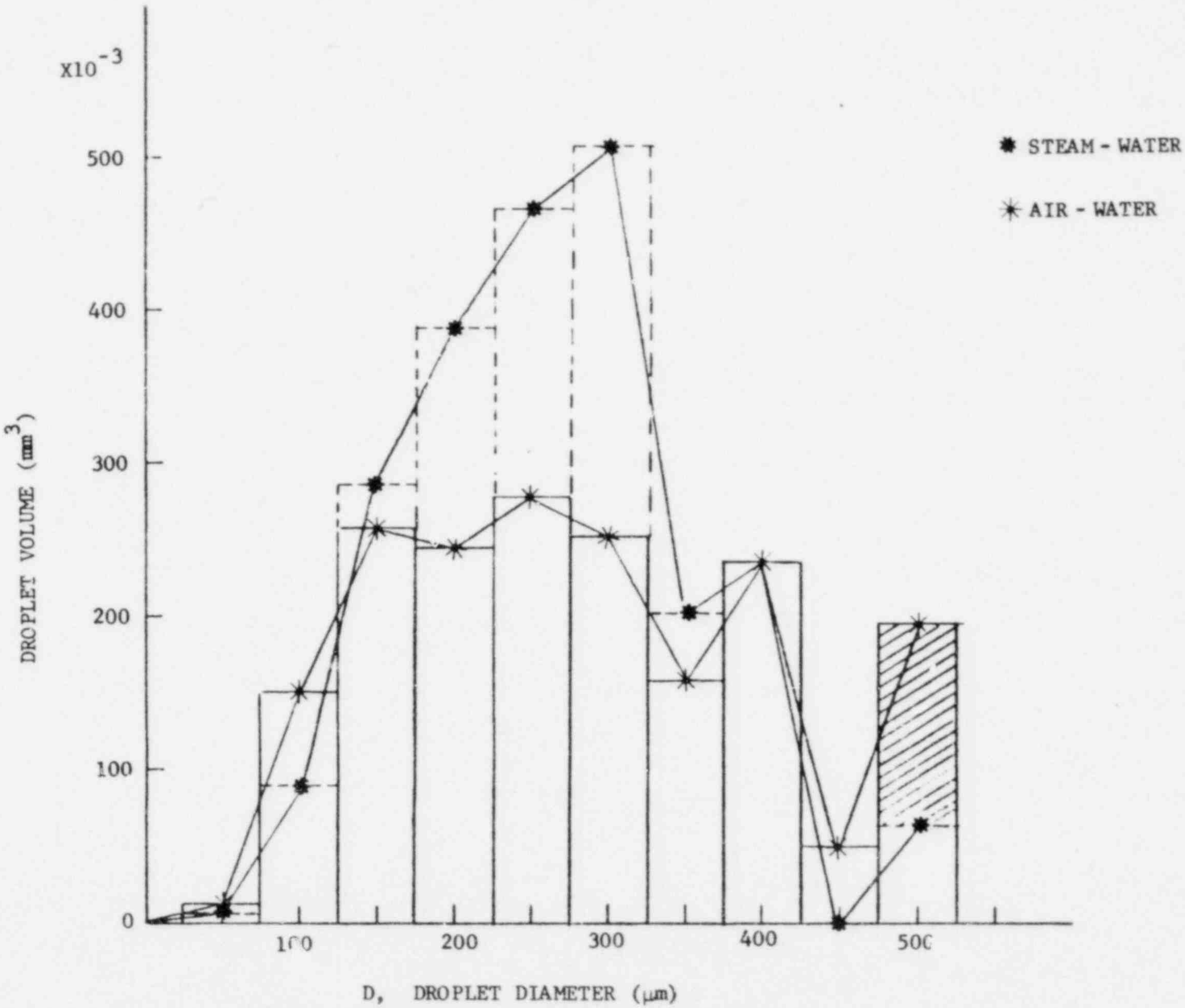


FIG. III-11 DROPLET VOLUME VS DROPLET DIAMETER

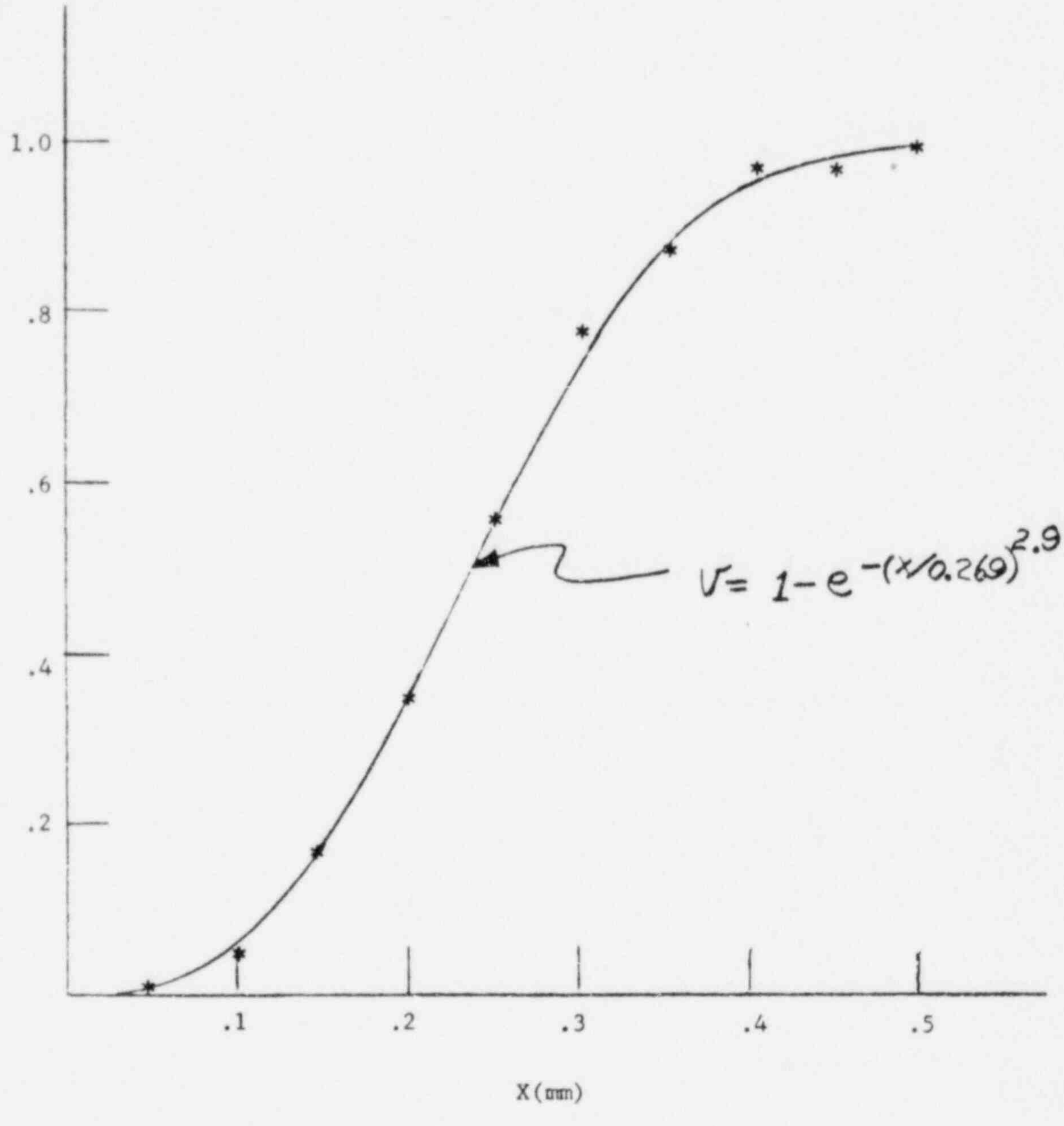


Figure 3-16 Comparison of data with Rosin-Rammler equation for steam-water spray

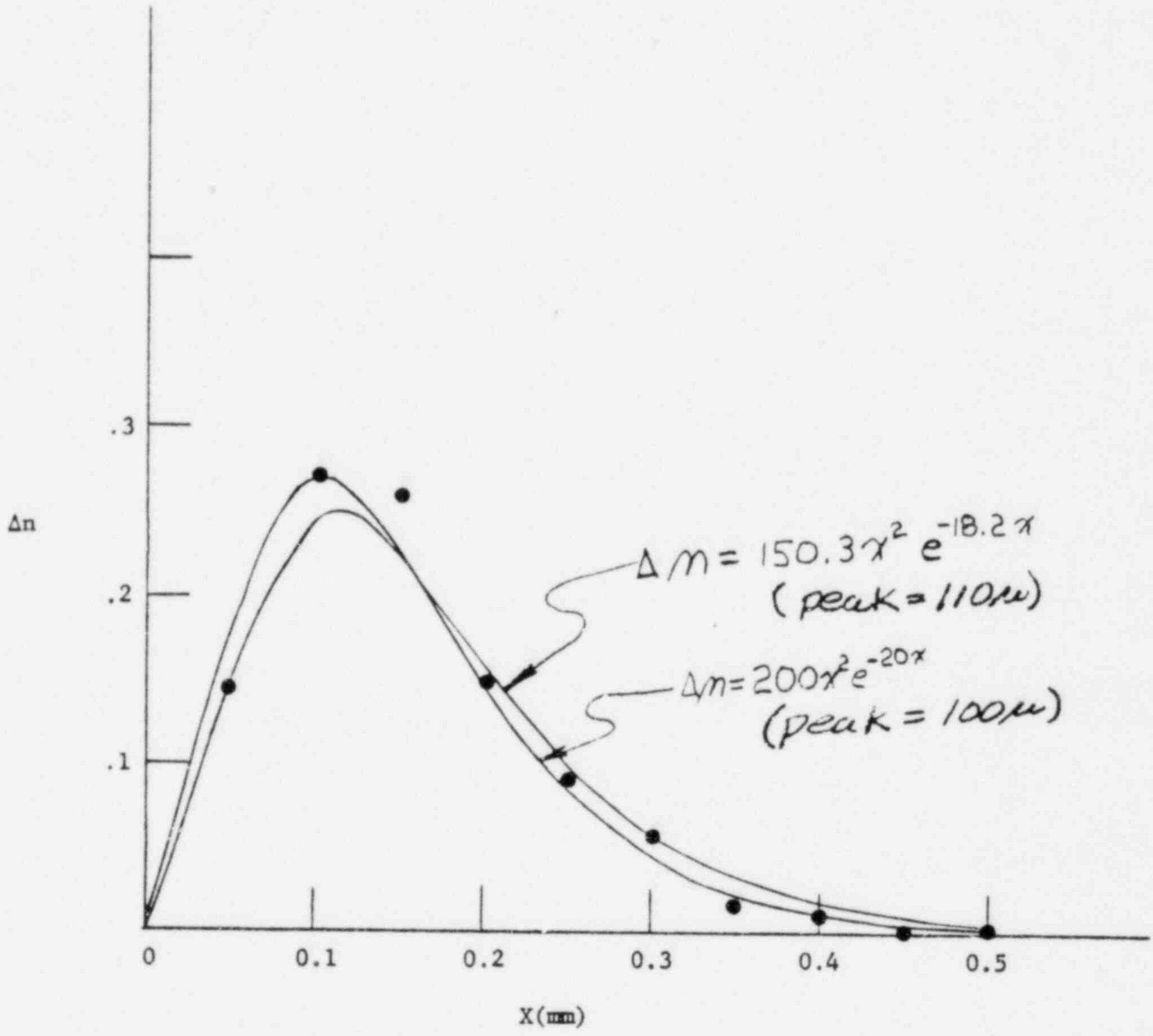
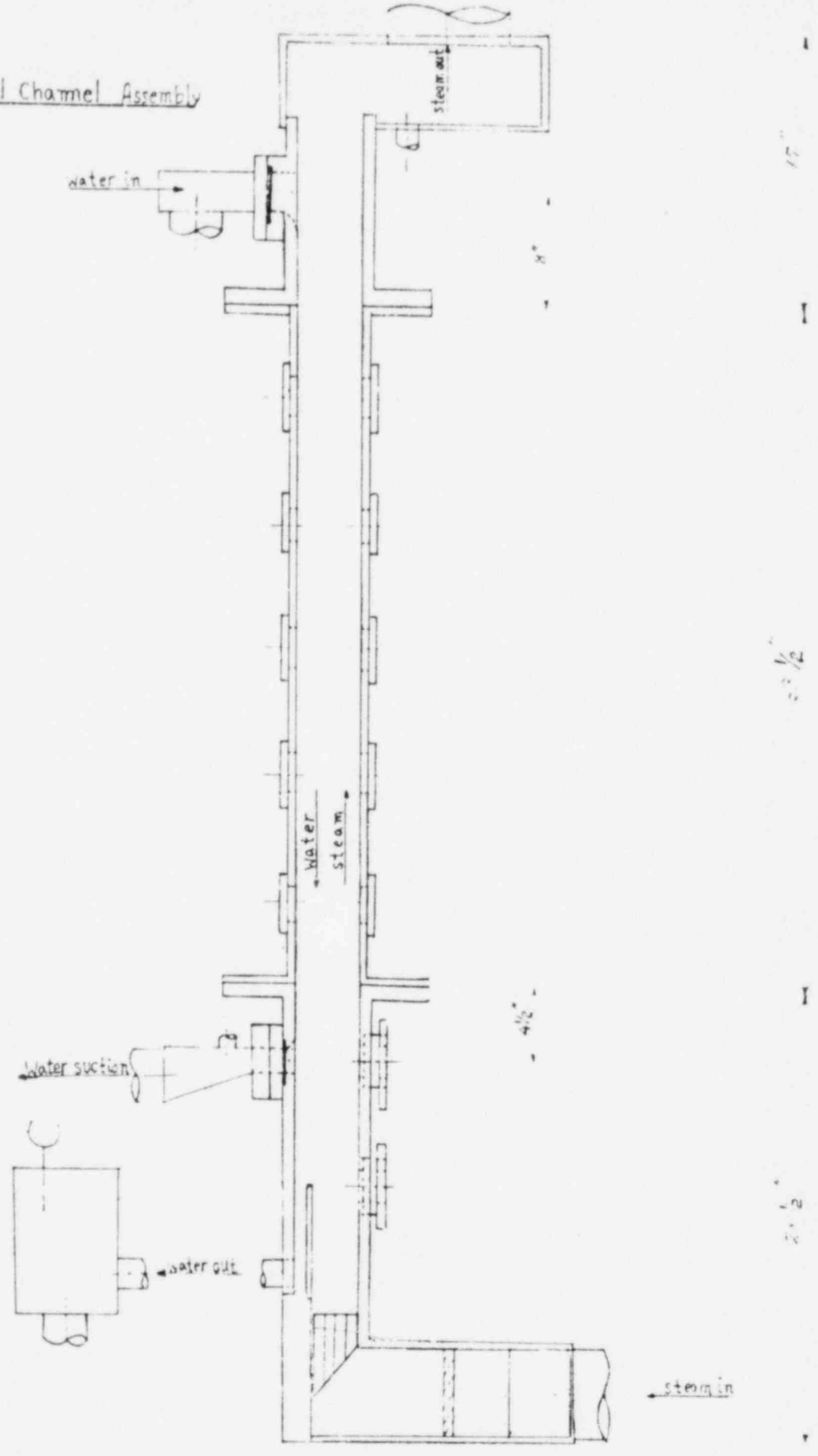


Figure 3-19. Number density distribution compared with Nukiyama-Tanasawa Equation (steam-water).

Vertical Channel Assembly



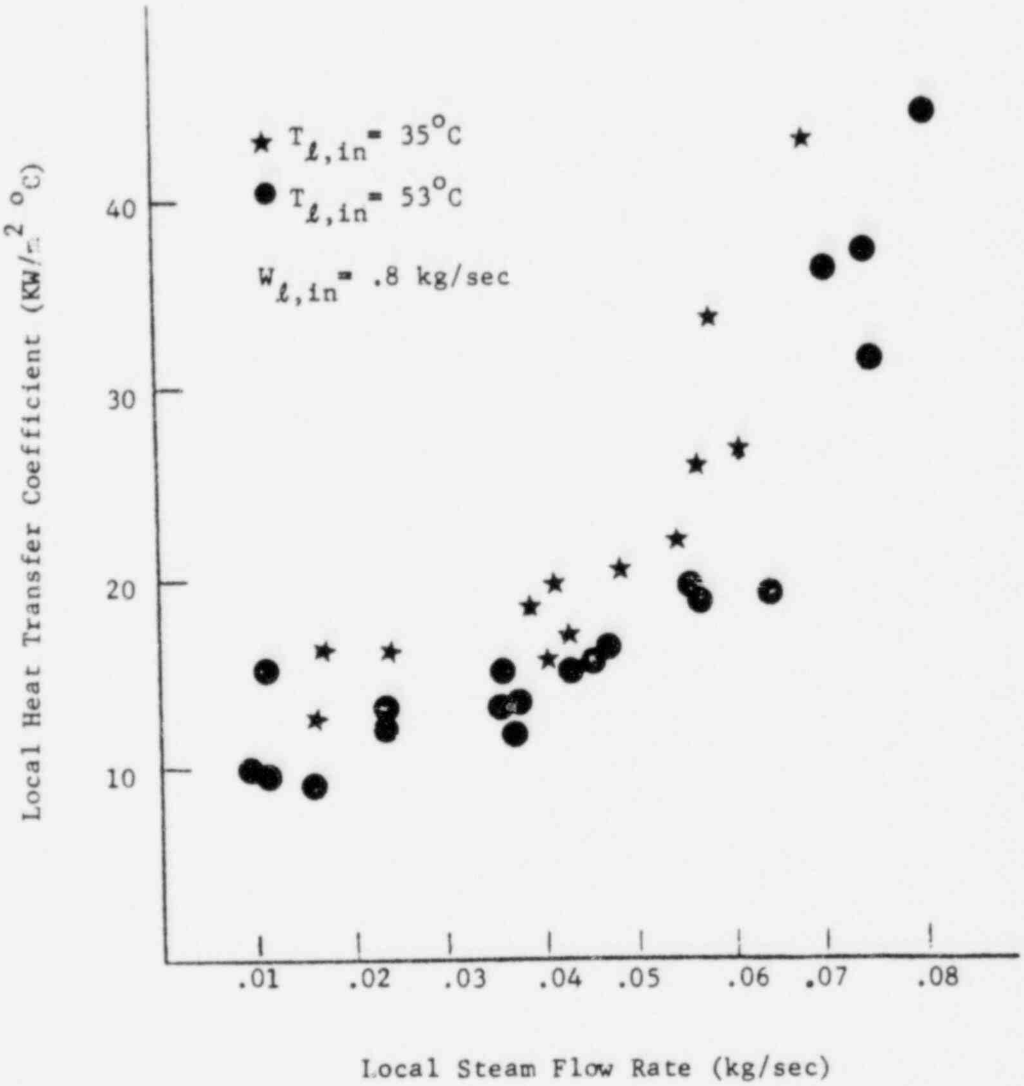


Figure 4-7 Plot of local heat transfer coefficient as a function of local steam flow rate

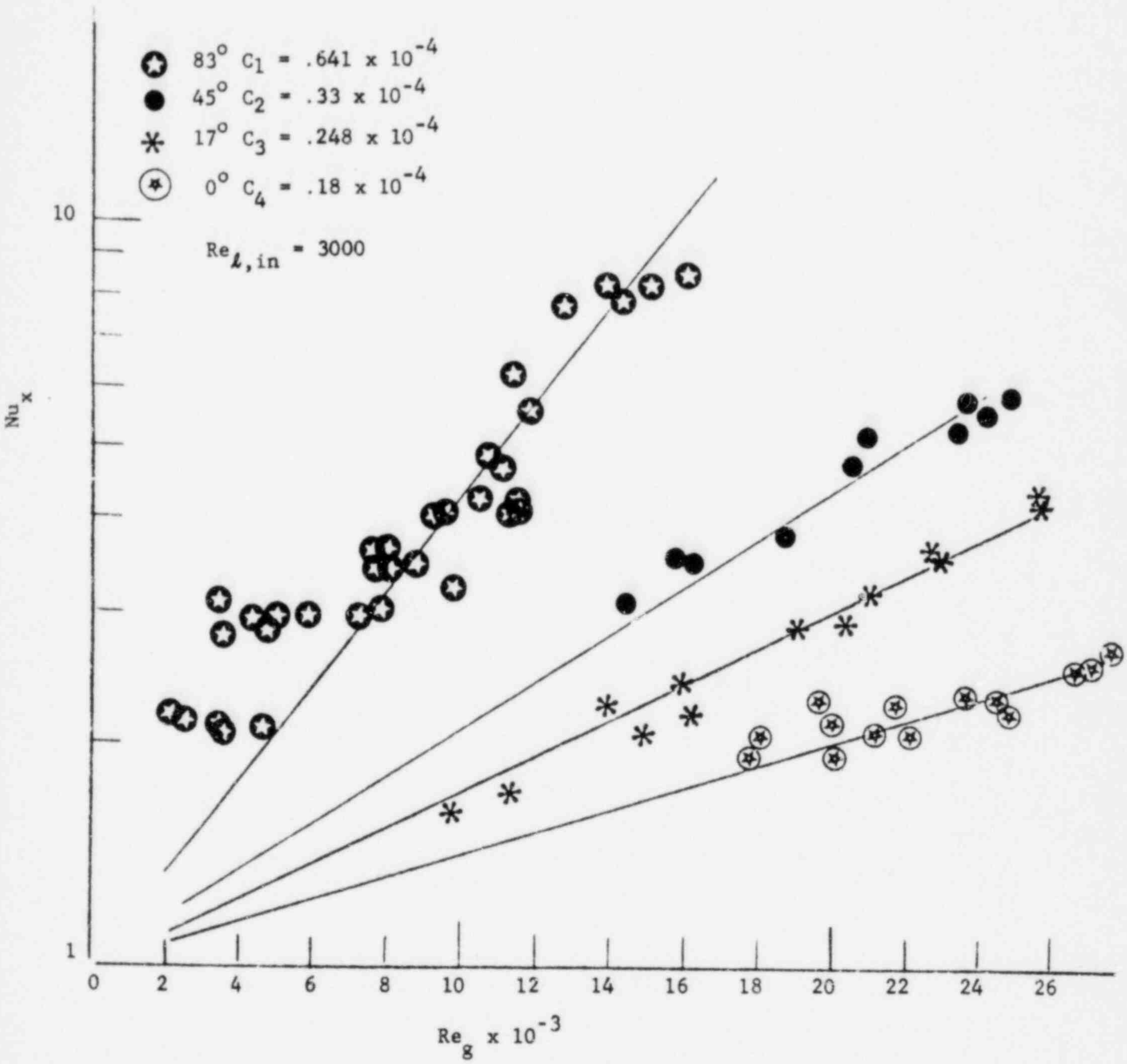


Figure 4.8. Plot of Nusselt number versus gas Reynolds number at various inclination angles.

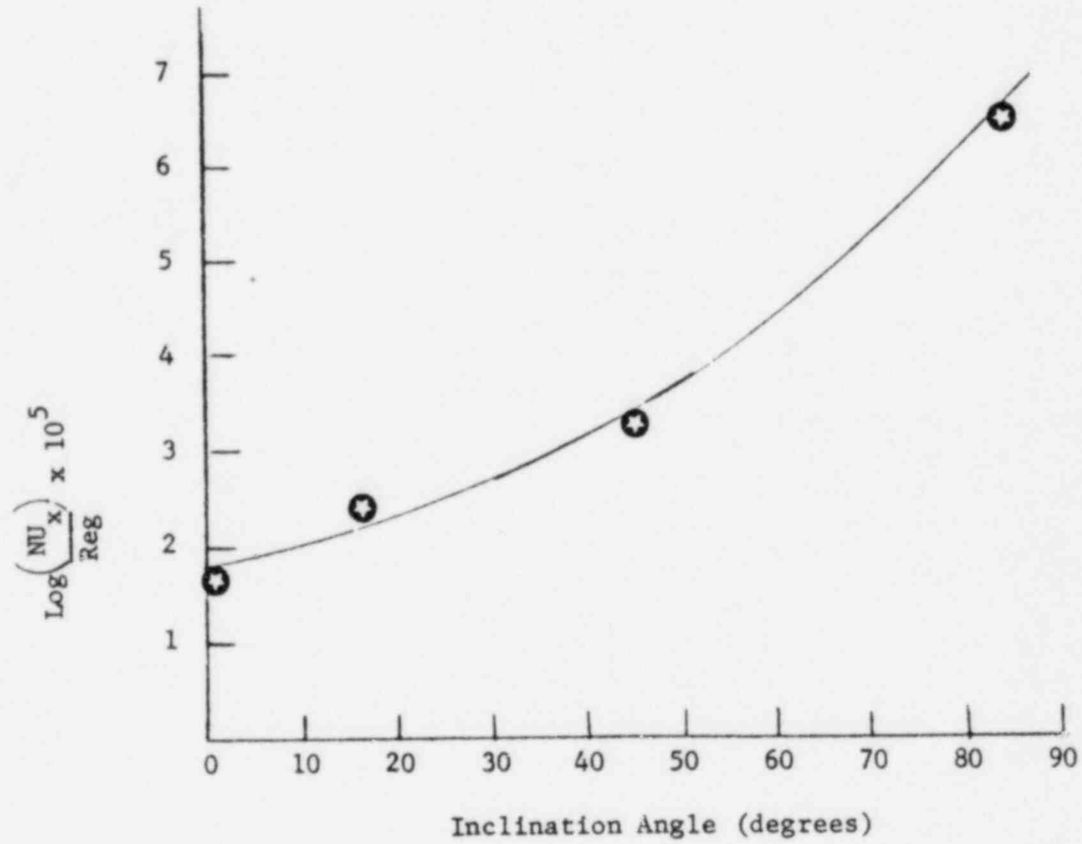


Figure 4-9. Effect of inclination angle on Nusselt number, Re_l = 3000

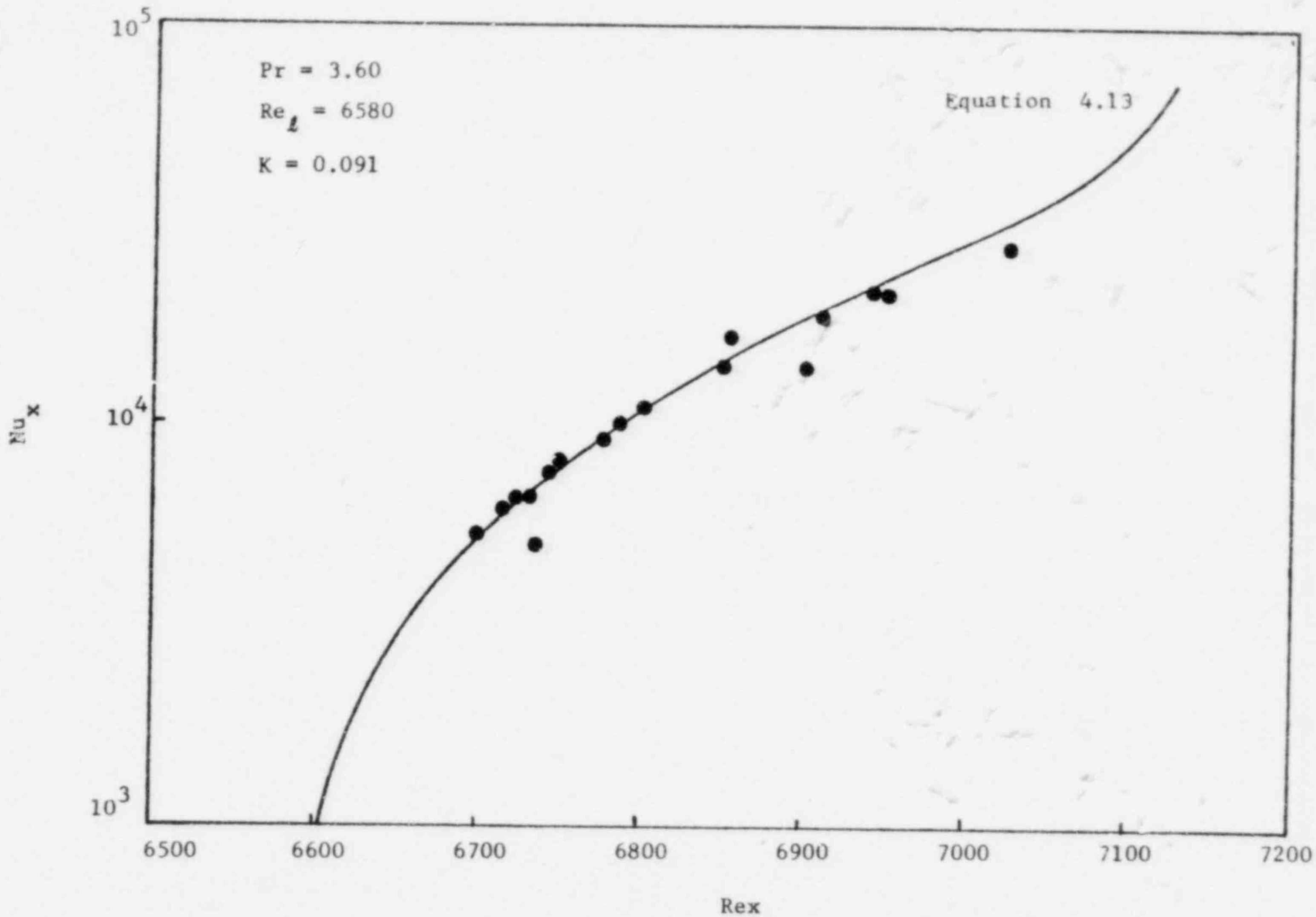


Figure 4-10 Comparison between Equation 4.13 and data

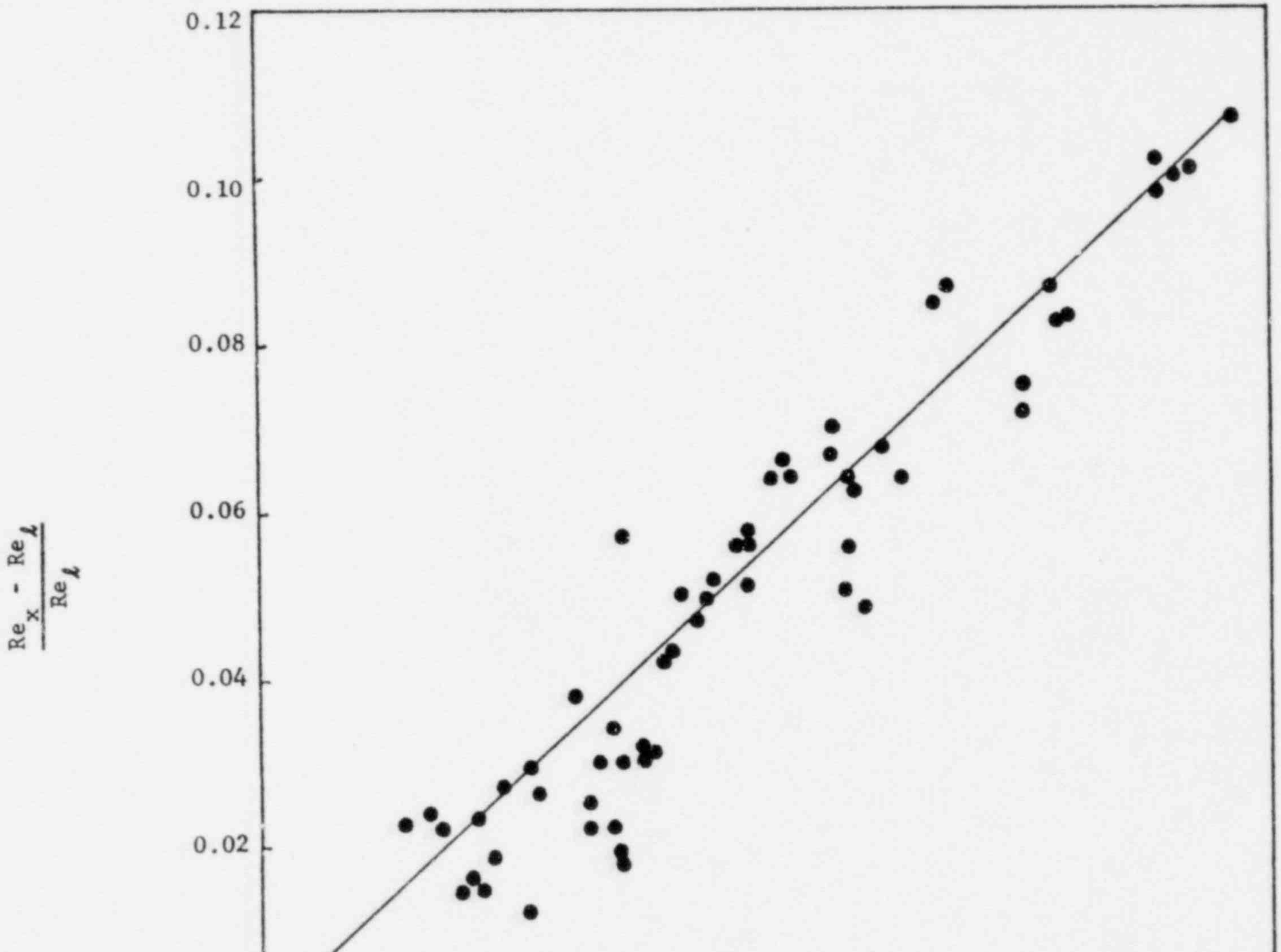


Figure 4-11. Correlation between Re_x, Re_ℓ, Re_g, X/L, K

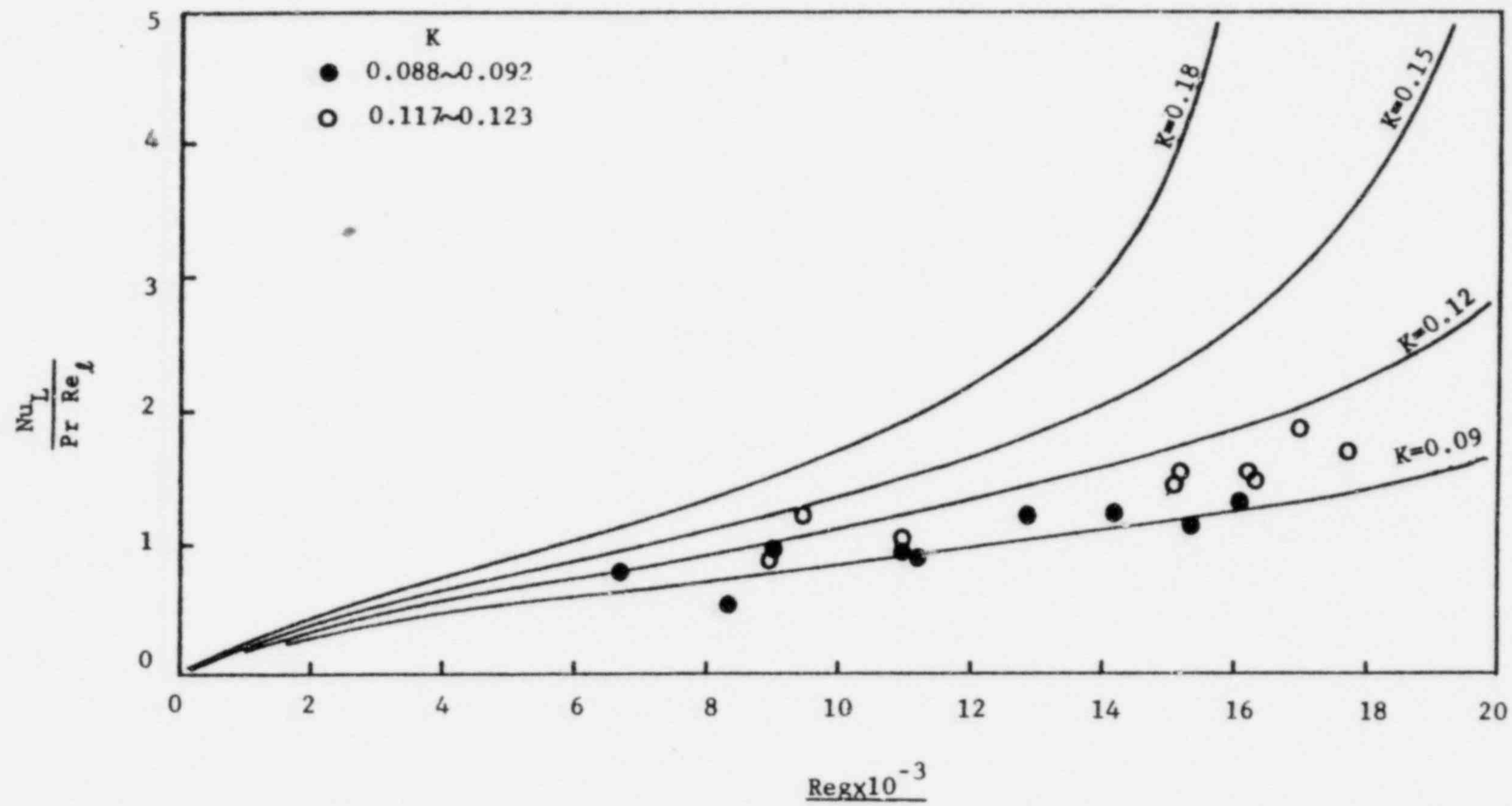


Figure 4-12 Effect of degree of subcooling on condensation rate at the exit (X=L)

RPI RESEARCH IN THE AREA OF PHASE DISTRIBUTION AND SEPARATION PHENOMENA AND LWR INSTABILITY PHENOMENA

R. T. Lahey, Jr.
Rensselaer Polytechnic Institute
Troy, New York 12181

INTRODUCTION

The USNRC is currently sponsoring research at RPI. The work to be reported on herein is being conducted under two separate programs; one dealing with phase distribution and separation phenomena, and the other, LWR instability phenomena.

The basic purpose of the work in the area of phase distribution and separation phenomena is summarized in Table Ia. The significant results obtained during the report period are tabulated in Table Ib. Table IIa summarizes the basic purpose of the research in the area of LWR instability phenomena, and Table IIb tabulates the significant results obtained during the report period.

DISCUSSION - PHASE DISTRIBUTION

An understanding of phase distribution phenomena is closely tied to an understanding of flow regime mechanisms. Moreover, advanced generation LWR safety codes (e.g., TRAC) rely on flow regime maps for specification of the interfacial transfer laws. It is thus quite important that one is able to predict which flow regime exists. Unfortunately, investigators rarely agree on flow regime boundaries. To try to resolve this discrepancy, a careful series of experiments have been conducted. The low pressure air/water experiment conducted at RPI consisted of measuring the chordal average void fraction in a 2.54 cm tube using an x-ray system. The probability density function (PDF) and power spectral density (PSD) function were calculated as well as the first four moments of these distributions. Classical pattern recognition theory techniques were employed to deduce an objective indicator of flow regime.

Figure 1 shows the PDF, PSD and photograph of bubbly flow. Note the PDF is unimodal and PSD, broadband. Figure 2 gives the same information for slug flow; however, in this case, the PDF is bimodal and the PSD is more sharply peaked (at the slug passage frequency). Figure 3 is for annular flow, and is similar to Fig. 1, except the unimodal PDF is concentrated at higher void fractions.

It was found that for all flow rates tested, a constant value of variance (the second moment about the mean) of 0.04 predicted both the bubbly/slug and slug/annular flow regime boundaries. Figure 4 shows the plot of variance, and Fig. 5, the flow regime map (based on the 0.04 criterion). It is significant to note that none of the existing flow regime maps and/or models predicted the measured boundaries (nor did they agree among themselves). This clearly indicates the need for an objective criterion, such as the one developed in this study. More work remains to be done in high pressure steam/water using various geometry test sections to verify the technique; however, it appears to be quite promising.

If detailed calculations are to be made of lateral phase distribution effects (e.g., within rod bundles), then a detailed understanding of geometric and hydrodynamic influences must be developed. To this end, a series of low pressure air/water measurements have been performed in a triangular test section. The local void fraction distribution was obtained using a radio frequency (RF) excited impedance probe, and the local liquid phase velocity profile was determined using a pitot tube. The pitot tube data was reduced with the model of Malnes [1], Shires and Riley [2], and Brandt [3]. This procedure produced a consistent set of data which closely satisfied a global mass balance.

Figures 6 and 7 show the void and liquid velocity profiles for various conditions of flow and quality. It can be clearly seen that there is a pronounced lateral distribution of the vapor phase. Similar phenomena can be expected in more complex geometries (e.g., fuel rod bundles).

Figure 8 is a photograph of the new 2-D test section which has been recently constructed. This test section will be used to obtain data concerning the so-called "chimney effect" during PWR reflood. These data will be used to assess the predictive capability of TRAC for PWR safety analysis and support of the Japanese 2-D experiment.

DISCUSSION - PHASE SEPARATION

A series of low pressure air/water experiments have been conducted in a tee to determine the degree of phase separation. In particular, the static pressure profiles are being measured in the test section shown in Figs. 9 and 10. These data are needed to determine the irreversible loss coefficient. Figures 11 and 12 show the axial pressure profiles in the branch and run for single- and two-phase flows. It can be seen that the flow is fully developed, and thus the determination of loss coefficient should be accurate and consistent. These data will be a valuable check on the predictive capability of advanced generation computer codes such as COBRA-TF and TRAC.

DISCUSSION - STABILITY ANALYSIS

Stability analysis is important in the evaluation of LWR safety. Indeed, one of the operational limits on modern BWR flow control is due to stability limitations. Unfortunately, very little work has been done in this field of technology during the past decade. As a consequence, there are a number of potential safety concerns which are being addressed at RPI.

A systematic evaluation of the effect of friction and gravity on linear stability has shown that, for certain values of these paramters, "islands of instability" can exist in the so-called stable region. Typical islands are shown in Fig. 13. It is also interesting to note in Fig. 14 that one can unify the concept of Ledinegg and density-wave instability. In fact, one can show that the Ledinegg mode is just the zero frequency of the density-wave mode.

The stability boundaries discussed above are based on linearized analyses. These analyses assume very small amplitude oscillations. For finite amplitude oscillations, one must consider the effect of non-linearities. As shown schematically in Fig. 15, if a subcritical bifurcation occurs, one can experience a divergent oscillation in the region of linear stability (if the excitation has a large enough amplitude). In contrast, if, as shown schematically in Fig. 16, a supercritical bifurcation occurs, a limit cycle may occur at reasonably small amplitude in the region of linear instability. This implies that the linear stability boundaries may not be conservative, and can be misleading when appraising stability margin. A modified amplitude-dependent stability boundary is shown in Fig. 17. The amplitude-dependent limit lines were determined using Hopf Bifurcation Theory. It is seen that regions of subcritical bifurcation exist for inlet velocity perturbations ($\varepsilon = \Delta j_i / j_{i_o}$) of 10% and 20%.

Currently most flow decay transients in a BWR are analyzed using techniques which assume the steady-state flow split is maintained during transients. This "initial value" approach does not allow one to predict any superimposed instability which may occur. To investigate the potential for superimposed oscillations during a "boundary value" driven flow decay, a boiling channel was subjected to a transient of the form:

$$\Delta p(t) = \Delta p_{final} + (\Delta p_{initial} - \Delta p_{final})e^{-Kt}$$

The results, shown in Fig. 18, indicate that the heated channel is very underdamped, and thus considerable undershoot and instability occur. Indeed, when one considers the initial value results ($\tilde{j}_i^*(t)$) compared to the more appropriate boundary value results ($j_i^*(t)$), it can be seen that considerable instability is predicted. It is interesting to note that this superimposed instability occurs even though the trajectory of the transient (and the final point) is in the region of linear stability. This implies that superimposed instabilities, and thus premature boiling transition, are possible during rapid flow decays. Obviously, more work is needed to appraise the potential for this phenomena in BWRs.

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[2] Shires, G. L. and Riley, P. J., "The Measurement of Radial Voidage Distribution in Two-Phase Flow by Isokinetic Sampling," AEEW-M650, 1966.

[3] Brandt, F., "Der Reibungdruckverlust von Wasser-Dampf-Gemischen und die Voreilgeschwindigkeit des Daupfes gegenuber dem Wasser in Senkrechten Kesselrohren, Dissertation, Darmstadt, 1958.

TABLE IA

PHASE DISTRIBUTION AND SEPARATION PHENOMENA

SAFETY & LICENSING RELATED ISSUES

- A THOROUGH UNDERSTANDING OF LATERAL PHASE DISTRIBUTION MECHANISMS IS NEEDED IN ORDER TO PROPERLY MODEL THESE EFFECTS IN ADVANCED GENERATION LWR SAFETY CODES

- A THOROUGH UNDERSTANDING OF PHASE SEPARATION MECHANISMS IS NEEDED IN ORDER TO PROPERLY MODEL THESE EFFECTS IN ADVANCED GENERATION LWR SAFETY CODES

- DETAILED PHASE DISTRIBUTION AND SEPARATION DATA IS REQUIRED FOR THE VERIFICATION OF LWR SAFETY CODES AND MODELS

TABLE IB

SIGNIFICANT RESULTS

- AN OBJECTIVE FLOW REGIME INDICATOR HAS BEEN PROPOSED (SHOWS CURRENT MODEL/CORRELATIONS ARE INADEQUATE)

- DETAILED PHASE DISTRIBUTION DATA HAS BEEN TAKEN IN A TRIANGULAR TEST SECTION (SUPPORTS MODEL DEVELOPMENT FOR COBRA-TF)

- A NEW 2-D TEST SECTION HAS BEEN CONSTRUCTED IN WHICH TO INVESTIGATE THE "CHIMNEY" EFFECT DURING PWR REFLOOD (SUPPORTS 2-D EXPERIMENT IN JAPAN THROUGH TRAC VERIFICATION)

- DETAILED ΔP DATA HAS BEEN TAKEN IN A HORIZONTAL TEE TEST SECTION (SUPPORTS PHASE SEPARATION MODEL FOR COBRA-TF, TRAC AND RELAP-5)

TABLE IIA

LWR INSTABILITY PHENOMENA
SAFETY AND LICENSING RELATED ISSUES

- THERMAL-HYDRAULIC INSTABILITY PHENOMENA CURRENTLY LIMITS THE OPERATIONAL (FLOW) CONTROL OF MODERN BWRs

- SUPERIMPOSED INSTABILITIES DURING TRANSIENTS MAY LEAD TO A PREMATURE BOILING TRANSITION. CURRENT BWR LICENSING CODES ARE INCAPABLE OF PREDICTING THIS POTENTIALLY IMPORTANT PHENOMENA

- FINITE AMPLITUDE PERTURBATIONS CAN LEAD TO DIVERGENT OSCILLATIONS IN THE REGION OF LINEAR STABILITY. NONLINEAR EFFECTS OF THIS TYPE ARE CURRENTLY NOT CONSIDERED IN THE LICENSING PROCESS

TABLE IIB

SIGNIFICANT RESULTS

- "ISLANDS OF INSTABILITY" HAVE BEEN DISCOVERED IN REGIONS OF LINEAR STABILITY. THESE ISLANDS EXPLAIN SOME OF THE ANOMALIES SEEN IN PREVIOUS DATA

- THE LEDINEGG INSTABILITY (I.E., FLOW EXCURSIVE) MODE HAS BEEN SHOWN TO BE THE ZERO FREQUENCY LIMIT OF DENSITY-WAVE OSCILLATIONS. THE ANALYTICAL TREATMENT OF THESE IMPORTANT INSTABILITY MODES HAS THUS BEEN UNIFIED

- FINITE AMPLITUDE OSCILLATIONS HAVE STABILITY BOUNDARIES WHICH MAY BE DIFFERENT FROM THE LINEAR STABILITY BOUNDARIES. SINCE A SUBCRITICAL BIFURCATION CAN LEAD TO DIVERGENT OSCILLATIONS IN THE REGION OF LINEAR STABILITY, LINEAR ANALYSIS IS NOT ALWAYS CONSERVATIVE

- SUPERIMPOSED INSTABILITIES HAVE BEEN PREDICTED DURING A FLOW COAST-DOWN EVENT. THIS IS SIGNIFICANT SINCE INSTANTANEOUS FLOW UNDERSHOOT MAY LEAD TO A PREMATURE BOILING TRANSITION

TECHNICAL PAPERS AND REPORTS RESULTING FROM RPI RESEARCH PROGRAM

Topical Reports

(1) Cheng, L. Y., Drew, D. A., Lahey, Jr., R. T., "Virtual Mass Effects in Two-Phase Flow," NUREG/CR-0020 1978.

(2) Honan, T. J. and Lahey, Jr., R. T., "The Measurement of Phase Separation in Wyes and Tees, " NUREG/CR-0557, 1978.

(3) Schell, Susanne L., Gay, R. R., Lahey, Jr., R. T., "The Development of a Side-Scatter Gamma Ray System for the Measurement of Local Void Fraction," NUREG/CR-0677, 1978.

(4) Lahey, Jr., R. T. and Drew, D. A., "An Assessment of the Literature Related to LWR Instability Modes," NUREG/CR-1414, 1980.

(5) Vince, M. A. and Lahey, Jr., R. T., "Flow Regime Identification and Void Fraction Measurement Techniques in Two-Phase Flow," NUREG-in print, 1980.

(6) Achard, J. L., Drew, D. A. and Lahey, Jr., R. T., "The Analysis of Linear and Non-Linear Stability Phenomena in Heated Channels," NUREG-in print, 1980.

RPI Theses (Not Published as Topical Reports)

(1) Saba, N., "An Experimental Technique for the Determination of Steam/Air Fraction," M.S. Thesis, November, 1977.

(2) Sim, S. K., "Analysis of Phase Distribution Mechanisms in Turbulent Two-Phase Pipe Flow," M.S. Thesis, December, 1977.

(3) Shum, F. B., "The Development of a Four-Equation Drift-Flux Computer Code (DRIFT-4)," M.S. Thesis, May, 1978.

(4) Lombardo, N. J., "The Development of a Data Acquisition and Reduction System for the RPI BWR Parallel Channel Effects Experiment," M.E. Thesis, December, 1978.

Technical Papers

(1) Saba, N., Krycuk, G. and Lahey, Jr., R. T., "An Experimental Technique for the Determination of Steam/Air Fraction," ANS Transactions, Vol. 27, 1977.

(2) Block, R. C., Perez-Griffo, M., Singh, U. N. and Lahey, Jr., R. T., "N^{16} Tagging of Water for Transient Flow Measurements," ANS Transactions, Vol. 27, 1977.

Technical Papers (continued)

(3) Perez-Griffo, M., Block, R. C. and Lahey, Jr., R. T., "N^{16} Tagging for Two-Phase Flow Measurements," ANS Transactions, Vol. 30, 1978.

(4) Lahey, Jr., R. T., Krycuk, G and Malaviya, B. K., "A High Intensity X-ray System for Stochastic Measurements of Two-Phase Flows," ANS Transactions, Vol. 30, 1978.

(5) Vince, M. A., Breed, H. E. and Lahey, Jr., R. T., "The Development of a High Temperature Optical Void Probe," ANS Transactions, Vol. 30, 1978.

(6) Gay, R. R., Schell, S. and Lahey, Jr., R. T., "The Side-Scatter Gamma Technique for Local Density Measurements," ANS Transactions, Vol. 30, 1978.

(7) Moreira, S. H., Krycuk, G. and Lahey, Jr., R. T., "Development of a Radio Frequency Local Probe for Void Fraction Measurements," ANS Transactions, Vol. 30, 1978.

(8) 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 Flow," AIChE Ppaer #56b, presented at 71st AIChE Meeting, Miami, 1978.

(9) Drew, D. A., Sim, S. and Lahey, Jr., R. T., "Radial Phase Distribution Mechanisms in Two-Phase Flow," Proceedings of the CSNI Specialists Meeting on Transient Two-Phase Flow, Paris, 1977.

(10) Lahey, Jr., R. T., Vince, M. A., and Krycuk, G., "The Development of an Optical Digital Interferometer," Proceedings of the 2nd Multi-Phase Flow and Heat Transfer Symposium Workshop, April, 1979, Miami, Florida.

(11) Lahey, Jr., R. T. and Drew, D. A., "The Analysis of Phase Distribution in Fully Developed Two-Phase Flows," Proceedings of the 2nd Multi-Phase Flow and Heat Transfer Symposium Workshop, April, 1979, Miami, Florida.

(12) Honan, T. J. and Lahey, Jr., R. T., "The Analysis of Dynamic Bias in Gamma Densitometer Measurements," ANS Transactions, Vol. 32, 1979.

(13) Honan, T. J. and Lahey, Jr., R. T., "The Evaluation of Static Error in Gamma Densitometry," ANS Transactions, Vol. 32, 1979.

(14) Conlon, W. M. and Lahey, Jr., R. T., "Scaling of the RPI Parallel Channel Effects Experiment," ANS Transactions, Vol. 32, 1979.

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(16) Lahey, Jr., R. T., Sim, S. and Drew, D. A., "An Evaluation of Interfacial Drag Models for Bubbly Two-Phase Flows," ASME Symposium Volume on Interfacial Transport Phenomena, 1979 National Heat Transfer Conference, San Diego, California, 1979.

(17) Drew, D. A., Cheng, L. and Lahey, Jr., R. T., "The Analysis of Virtual Mass Effects in Two-Phase Flow," Int. Journal of Multiphase Flow," 5, 1979.

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(18) Drew, D. A. and Lahey, Jr., R. T., "An Analytical Derivation of a Subchannel Void-Drift Model," ANS Transactions, 33, 1979.

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(20) Drew, D. A. and Lahey, Jr., R. T., "Phase Distribution Mechanisms in Turbulent Two-Phase Flow in Channels of Arbitrary Cross Section," submitted for publication in the Journal of Fluids, 1980.

(21) Gay, R. R., Ohkawa, K and Lahey, Jr., R. T., "The Measurement of Local Void Fraction with a Side-Scatter Gamma Technique," Proceedings of 26th International Symposium of Instrument Society of America, May 1980, Seattle, Washington.

(22) Perez-Griffo, M., Block, R. C. and Lahey, Jr., R. T., "Basic Two-Phase Flow Measurements Using ^{16}N Tagging Techniques," Proceedings of ANS/ASME Meeting on Reactor Thermal-Hydraulics, October 1980, Saratoga, New York.

(23) Drew, D. A., and Lahey, Jr., R. T., "Phase Distribution Mechanisms in Turbulent Low Quality Two-Phase Flow in a Circular Pipe," submitted for publication in Journal of Fluid Mechanics, 1980.

(24) Saba, N., Lahey, Jr., R. T. and Corelli, J. C., "An Experimental Technique for Determination of Steam Fraction in Flowing Steam/Air Mixtures," Nuclear Technology, 1980.

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(28) Achard, J. L., Drew, D. A. and Lahey, Jr., R. T., "The Effect of Gravity and Friction on the Stability of Boiling Flow in a Channel," AIChE Symposium Series #199, Vol. 76, Heat Transfer-Orlando, 1980.

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(30) Park, G., Becker, M. and Lahey, Jr., R. T., "The Effect of Radial Non-uniformity on BWR Stability Margins," ANS Transactions, Vol. 35, 1980.

(31) Lahey, Jr., R. T., Cheng, L. Drew, D. A. and Flaherty, J. E., "The Effect of Virtual Mass on the Numerical Stability of Accelerating Two-Phase Flows," Int. Journal of Multiphase Flow, 6, 1980.

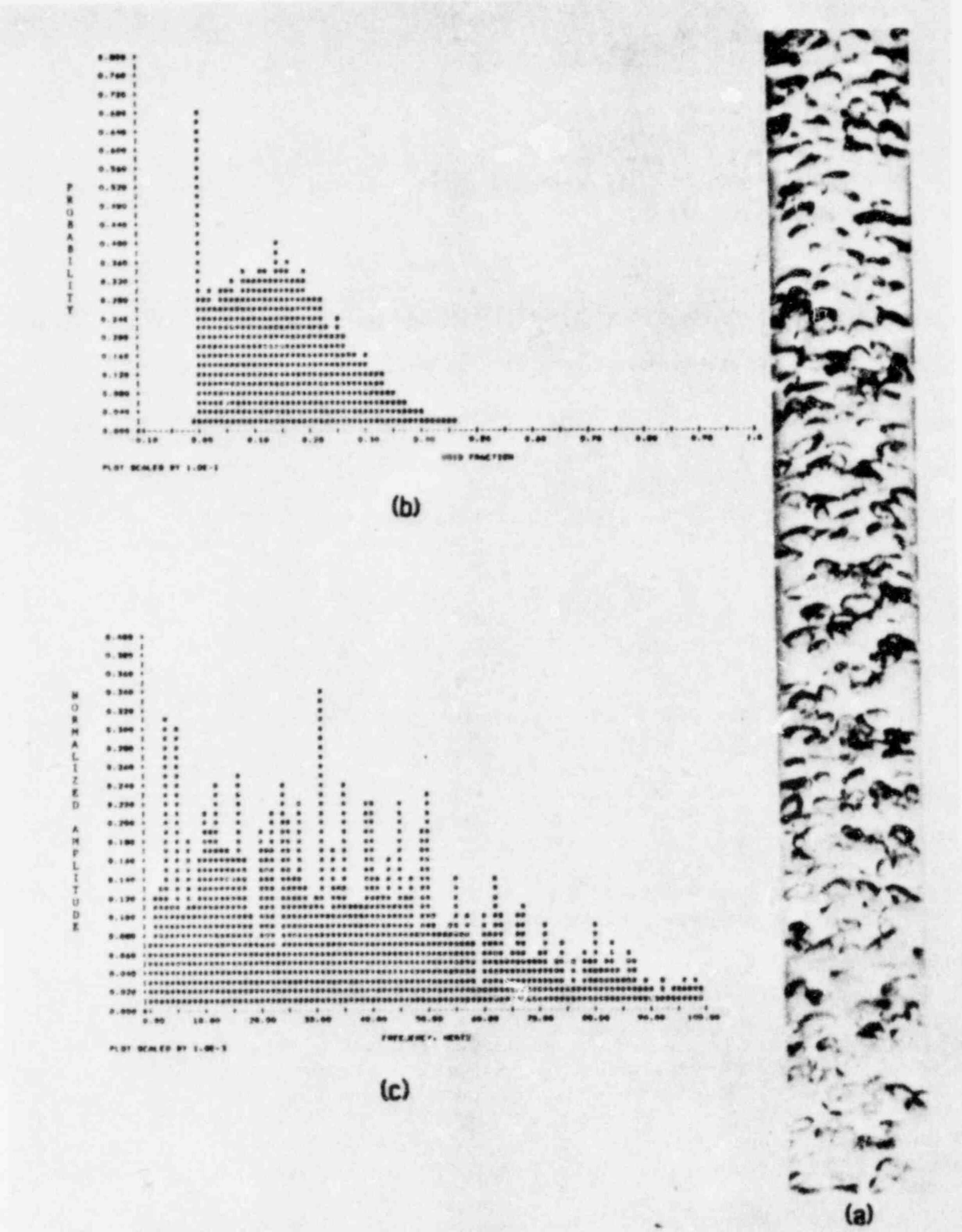
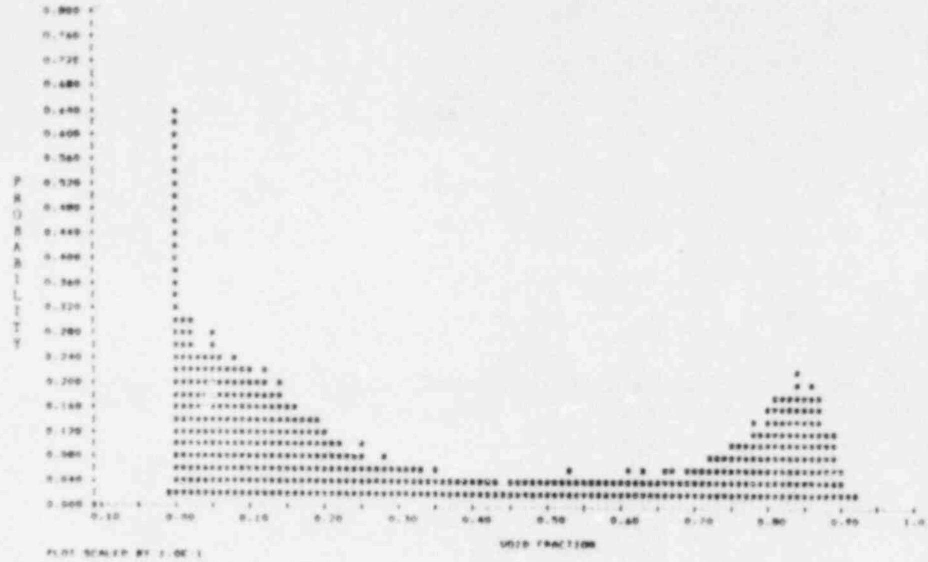
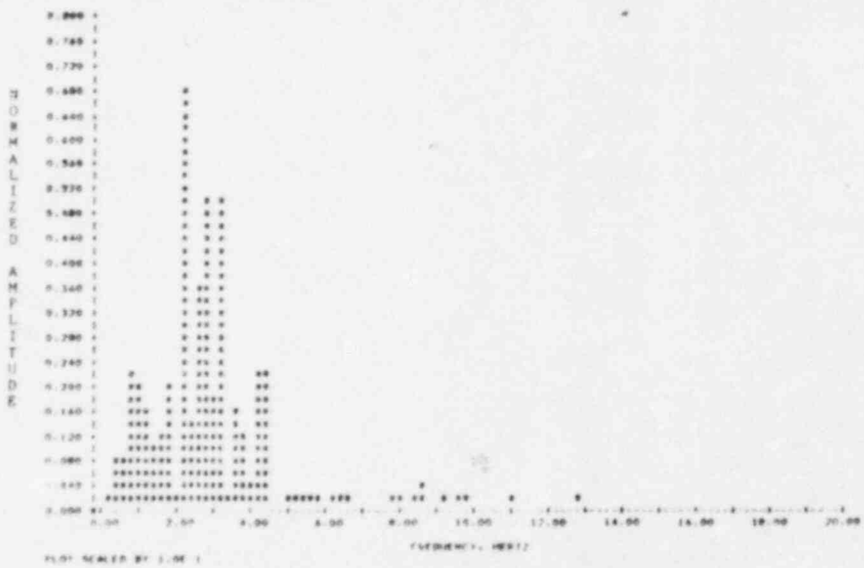


Figure 1
A photograph (a), diamcter PDF (b), and diameter PSD (c), for 13 percent area-averaged void fraction, j_{ℓ} = 0.25 m/sec, j_g=0.073 m/sec



(b)



(c)

(a)

Figure 2

A photograph (a), diameter PDF (b), and diameter PSD (c) for 32 percent area-averaged void fraction, j_ℓ=0.25 m/sec, j_g=0.288 m/sec

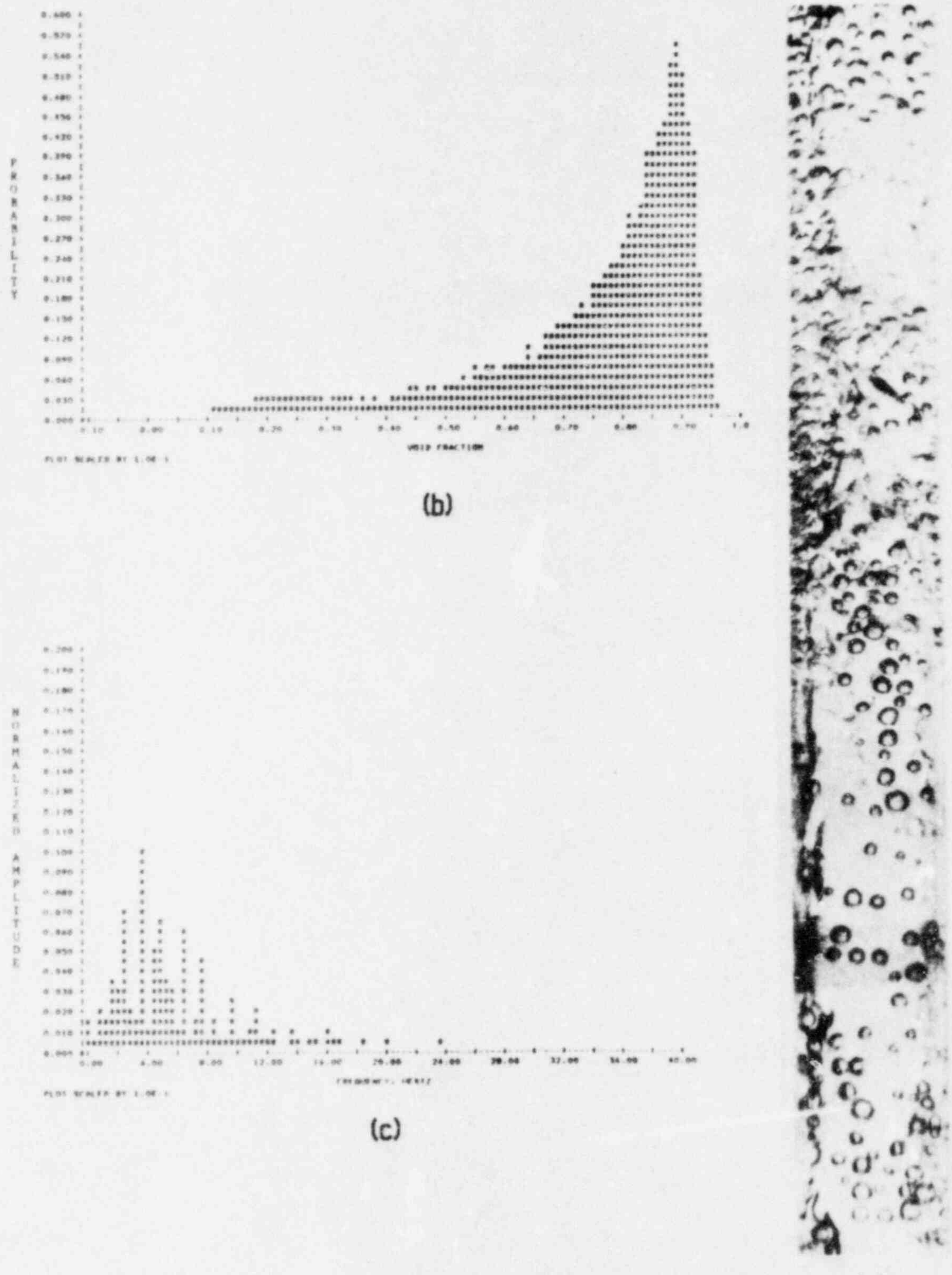


Figure 3

A photograph (a), diameter PDF (b), and diameter PSD (c), for 68 percent area-averaged void fraction, $j_{\ell}=0.25$ m/sec, $j_g=2.67$ m/sec

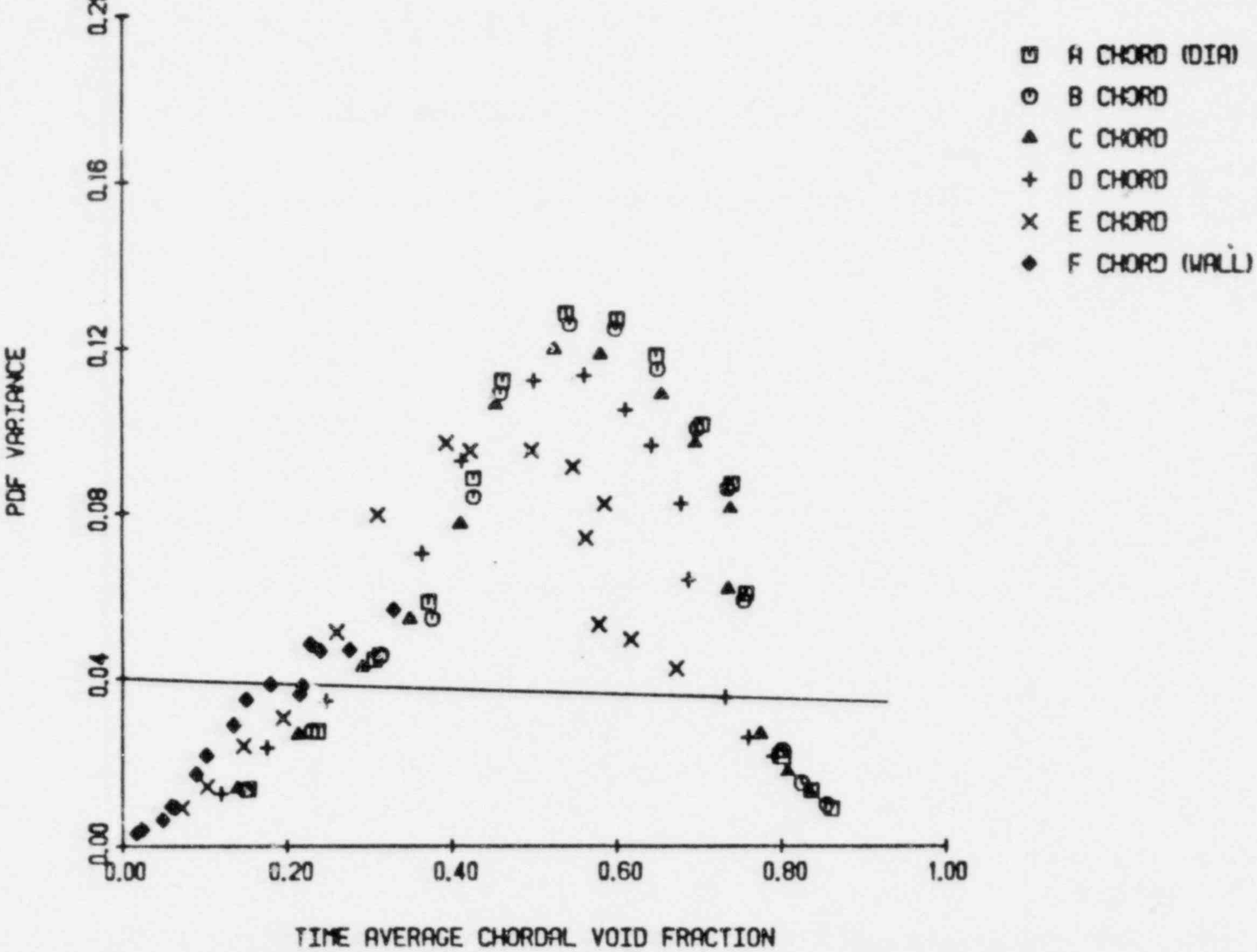


Figure 4 The PDF variance vs. time average chordal void fraction for $j_L = 0.0$ m/sec

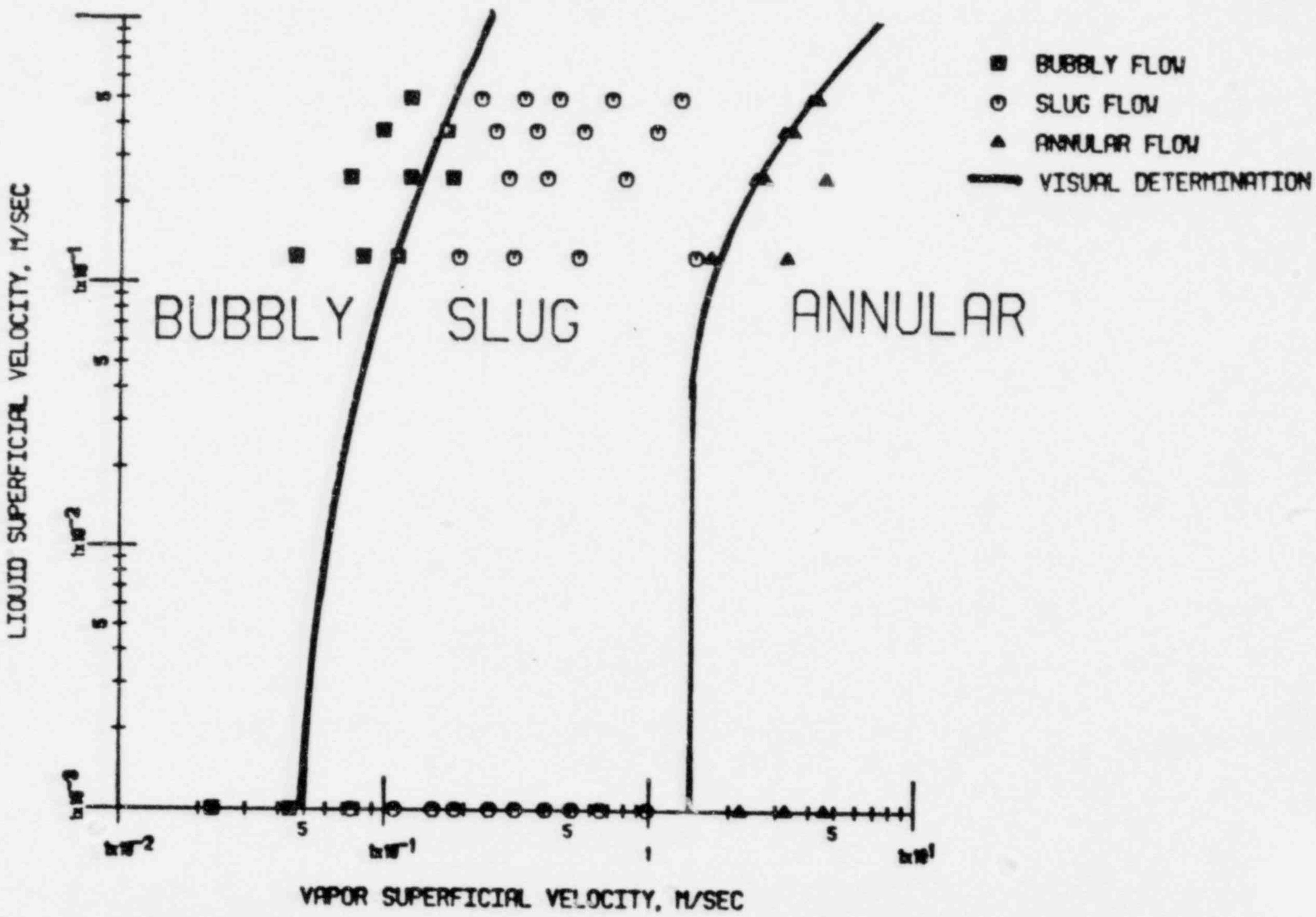


Figure 5 A flow regime map based on a PDF variance of 0.04

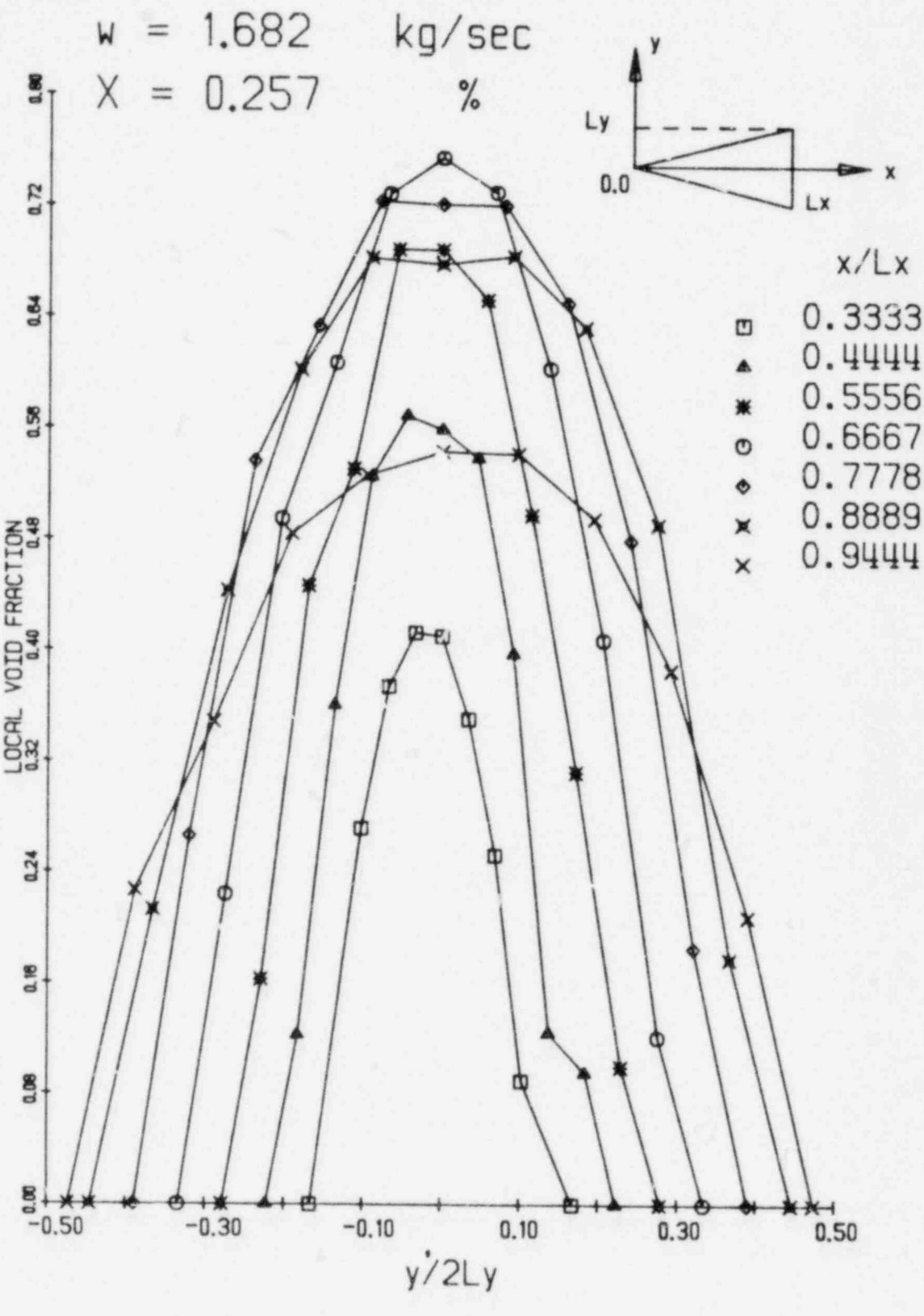
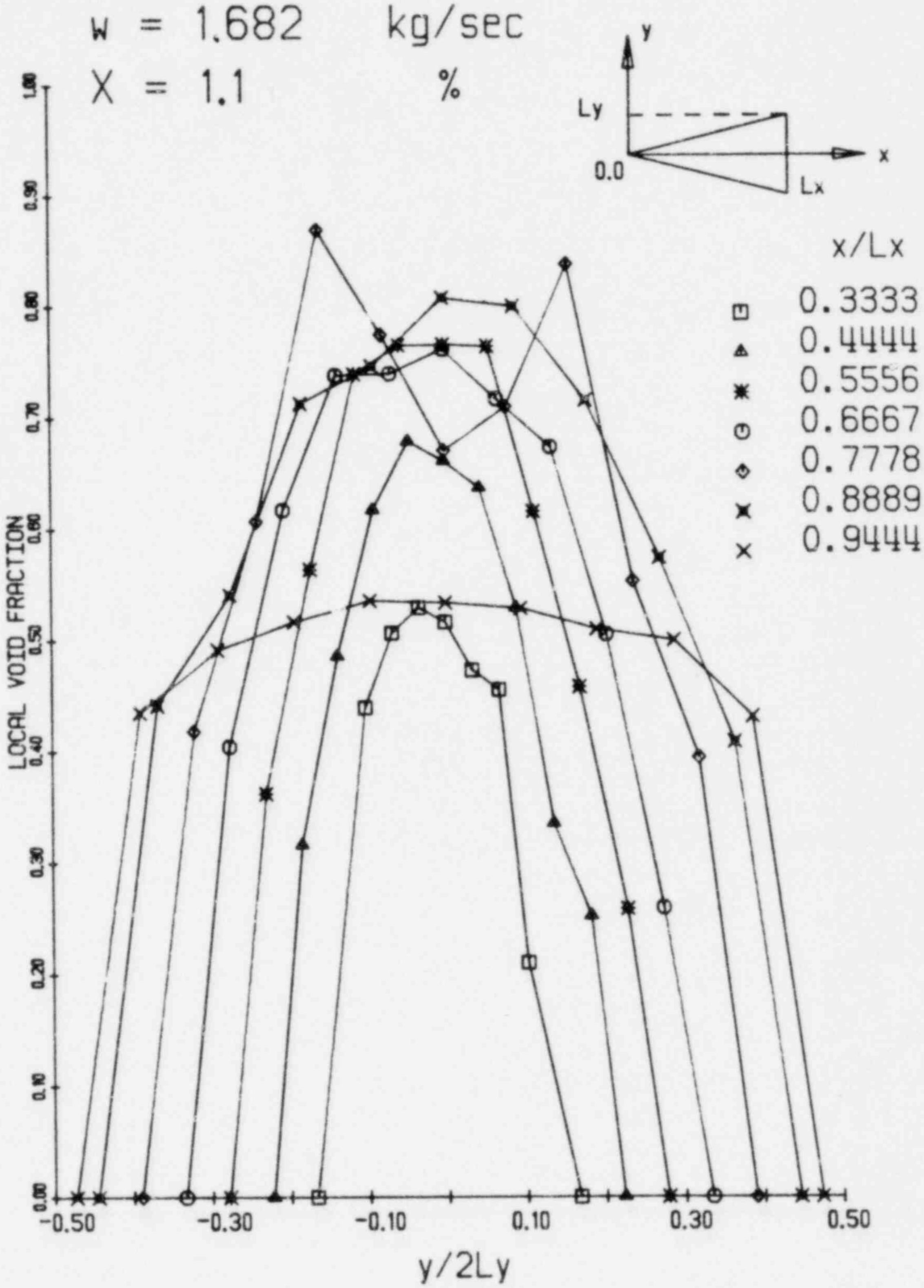
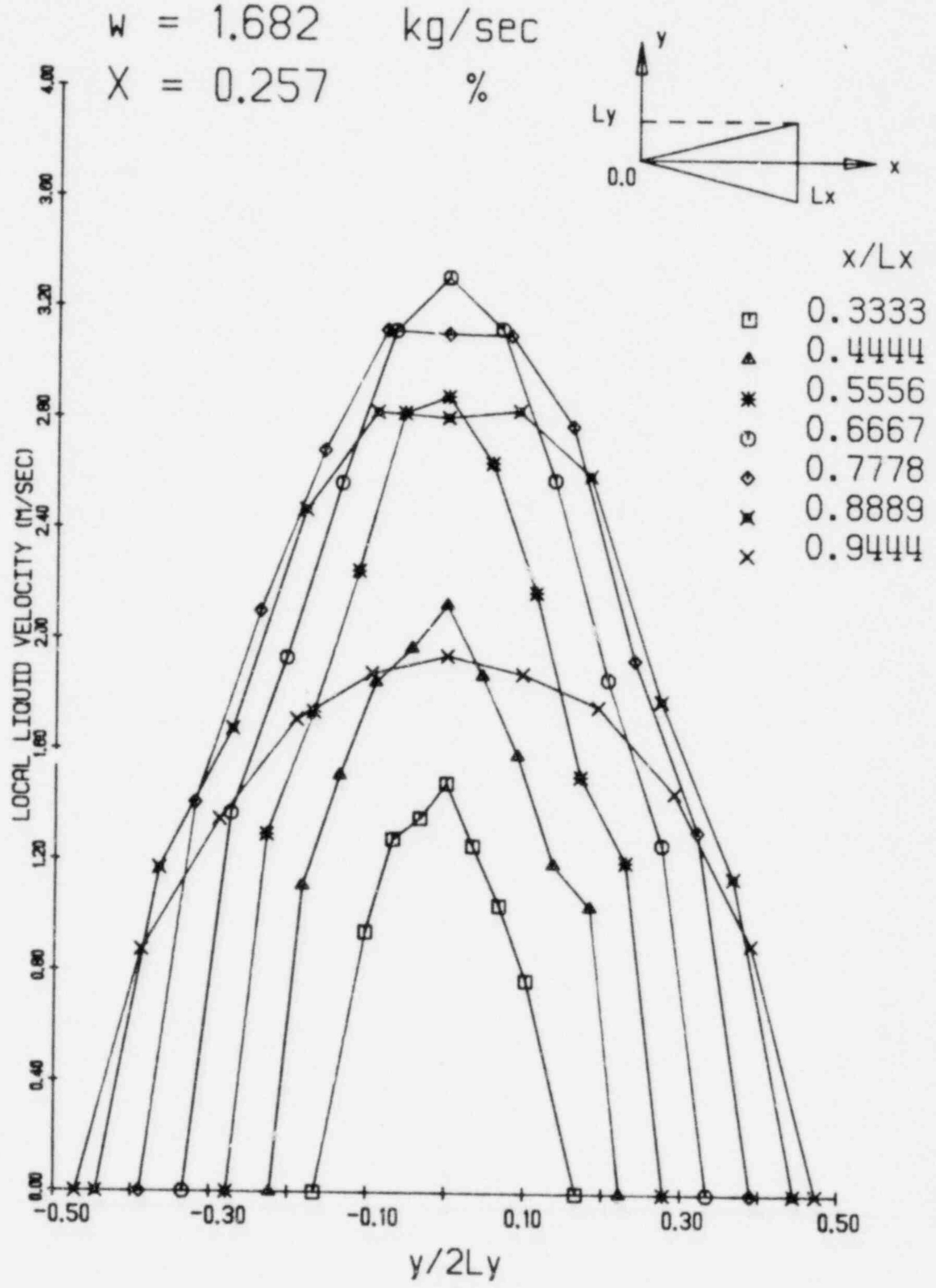

W = 1.682 kg/sec
X = 0.257 %
y
Ly
0.0
x
Lx
x/Lx
0.3333
0.4444
0.5556
0.6667
0.7778
0.8889
0.9444
LOCAL VOID FRACTION
0.00
0.08
0.16
0.24
0.32
0.40
0.48
0.56
0.64
0.72
0.80
-0.50
-0.30
-0.10
0.10
0.30
0.50
y'/2Ly

Figure 6a



W = 1.682 kg/sec
X = 1.1 %
y
Ly
0.0
Lx
x
x/Lx
0.3333
0.4444
0.5556
0.6667
0.7778
0.8889
0.9444
LOCAL VOID FRACTION
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
-0.50
-0.30
-0.10
0.10
0.30
0.50
y/2Ly

Figure 6b



W = 1.682 kg/sec
X = 0.257 %
y
Ly
0.0
x
Lx
x/Lx
0.3333
0.4444
0.5556
0.6667
0.7778
0.8889
0.9444
LOCAL LIQUID VELOCITY (M/SEC)
0.00
0.40
0.80
1.20
1.60
2.00
2.40
2.80
3.20
3.60
4.00
-0.50
-0.30
-0.10
0.10
0.30
0.50
y/2Ly

Figure 6c

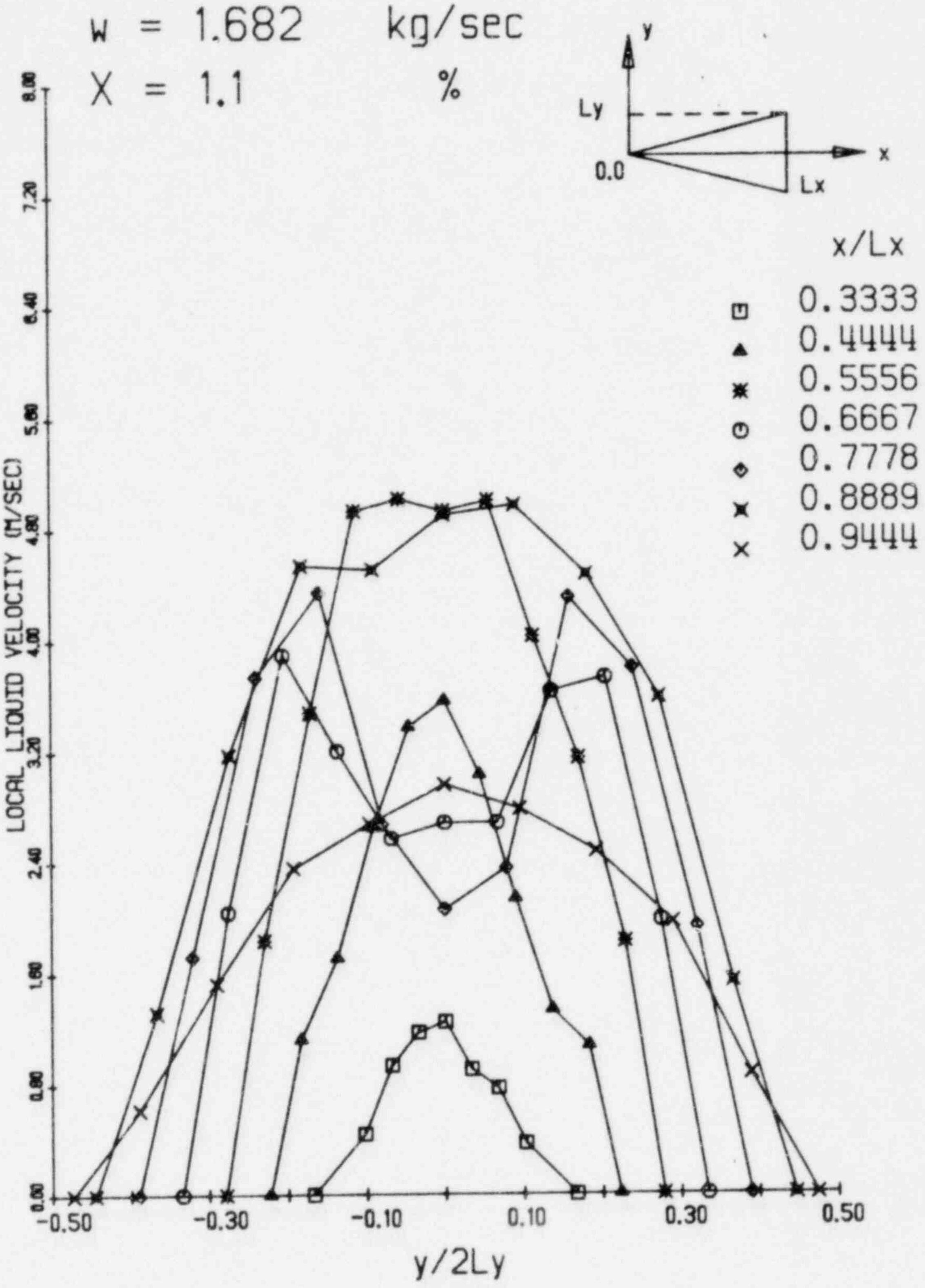
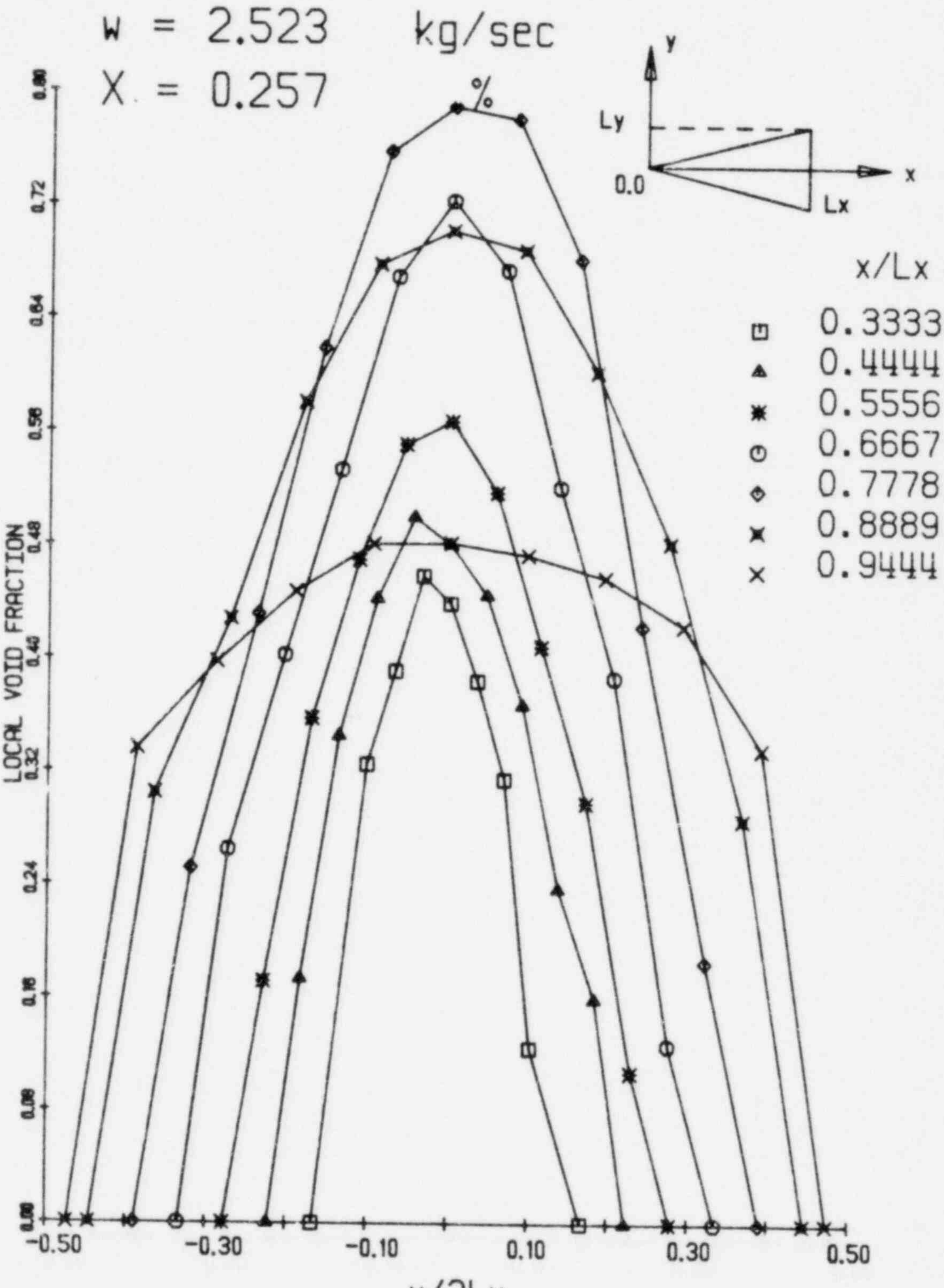

W = 1.682 kg/sec
X = 1.1 %
y
Ly
0.0
x
Lx
x/Lx
0.3333
0.4444
0.5556
0.6667
0.7778
0.8889
0.9444
LOCAL LIQUID VELOCITY (M/SEC)
8.00
7.20
6.40
5.60
4.80
4.00
3.20
2.40
1.60
0.80
0.00
-0.50
-0.30
-0.10
0.10
0.30
0.50
y/2Ly

Figure 6d



W = 2.523 kg/sec
X = 0.257 %
y
Ly
0.0
Lx
x
x/Lx
0.3333
0.4444
0.5556
0.6667
0.7778
0.8889
0.9444
LOCAL VOID FRACTION
0.80
0.72
0.64
0.56
0.48
0.40
0.32
0.24
0.16
0.08
0.00
-0.50
-0.30
-0.10
0.10
0.30
0.50
y/2Ly

Figure 7a

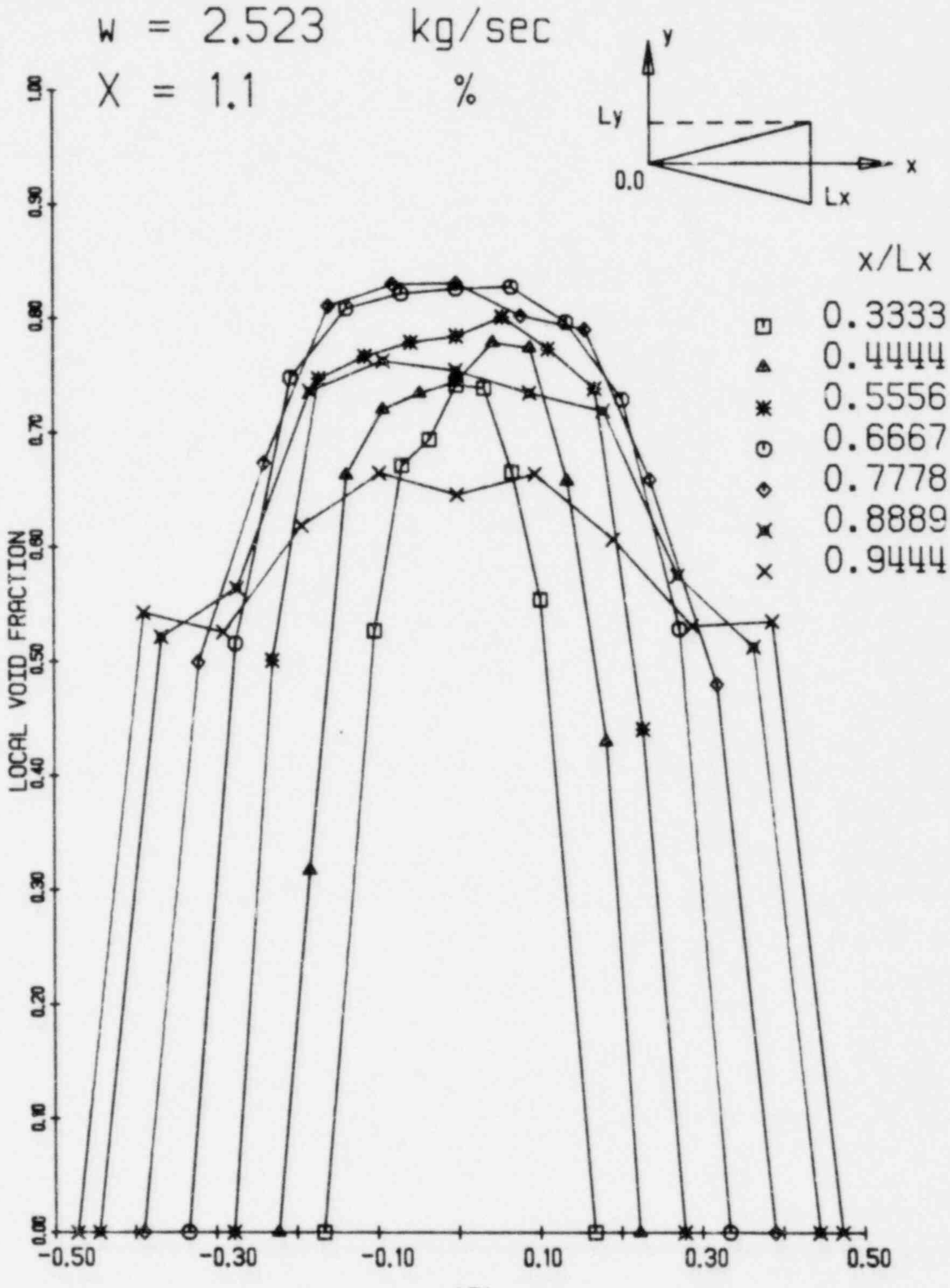
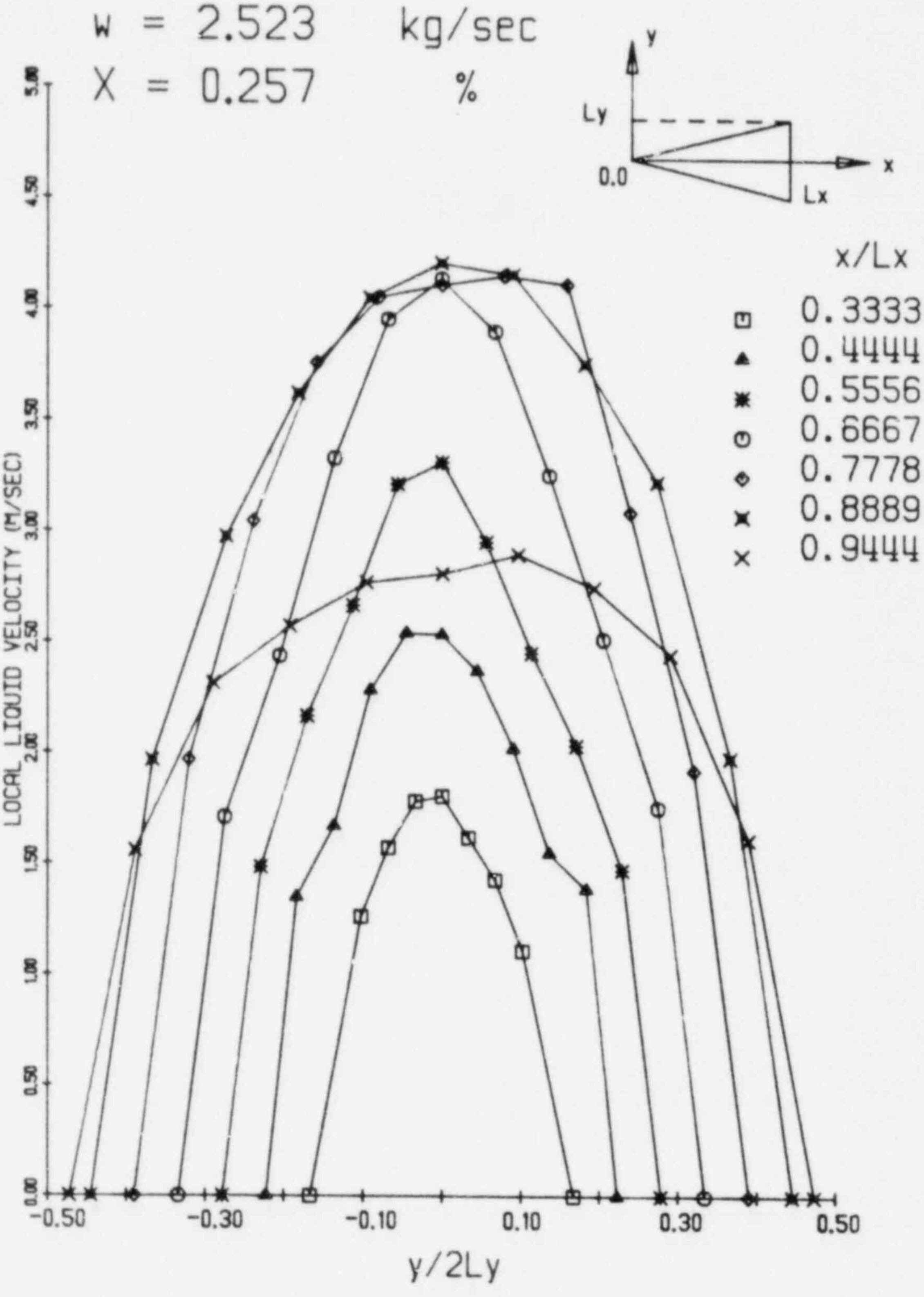
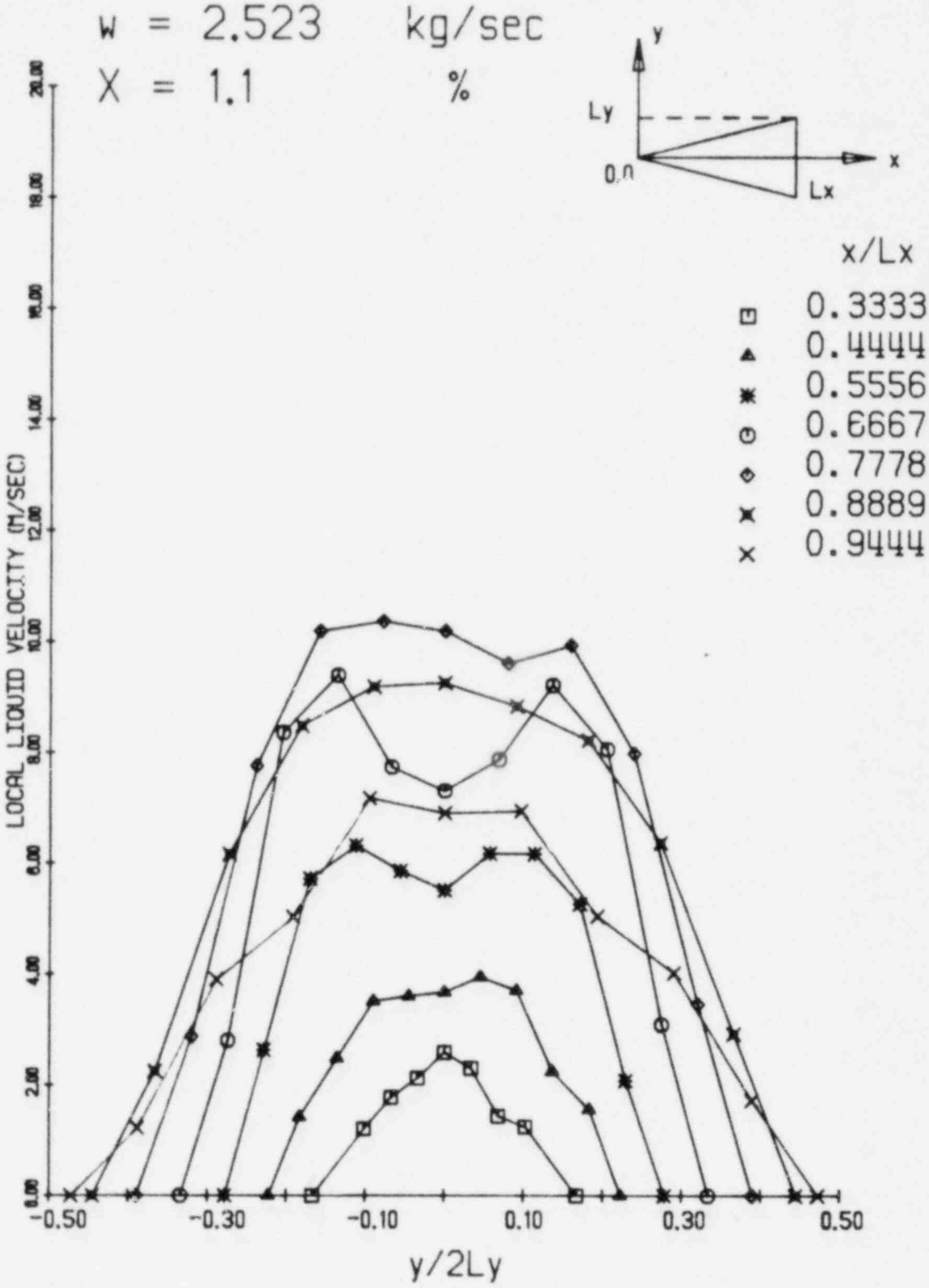

W = 2.523 kg/sec
X = 1.1 %
y
Ly
0.0
Lx
x
x/Lx
0.3333
0.4444
0.5556
0.6667
0.7778
0.8889
0.9444
LOCAL VOID FRACTION
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
-0.50
-0.30
-0.10
0.10
0.30
0.50
y/2Ly

Figure 7b



W = 2.523 kg/sec
X = 0.257 %
y
Ly
0.0
Lx
x
x/Lx
0.3333
0.4444
0.5556
0.6667
0.7778
0.8889
0.9444
LOCAL LIQUID VELOCITY (M/SEC)
5.00
4.50
4.00
3.50
3.00
2.50
2.00
1.50
1.00
0.50
0.00
-0.50
-0.30
-0.10
0.10
0.30
0.50
y/2Ly

Figure 7c



W = 2.523 kg/sec
X = 1.1 %
y
Ly
0.0
Lx
x
x/Lx
0.3333
0.4444
0.5556
0.6667
0.7778
0.8889
0.9444
LOCAL LIQUID VELOCITY (M/SEC)
0.00
2.00
4.00
6.00
8.00
10.00
12.00
14.00
16.00
18.00
20.00
-0.50
-0.30
-0.10
0.10
0.30
0.50
y/2Ly

Figure 7d

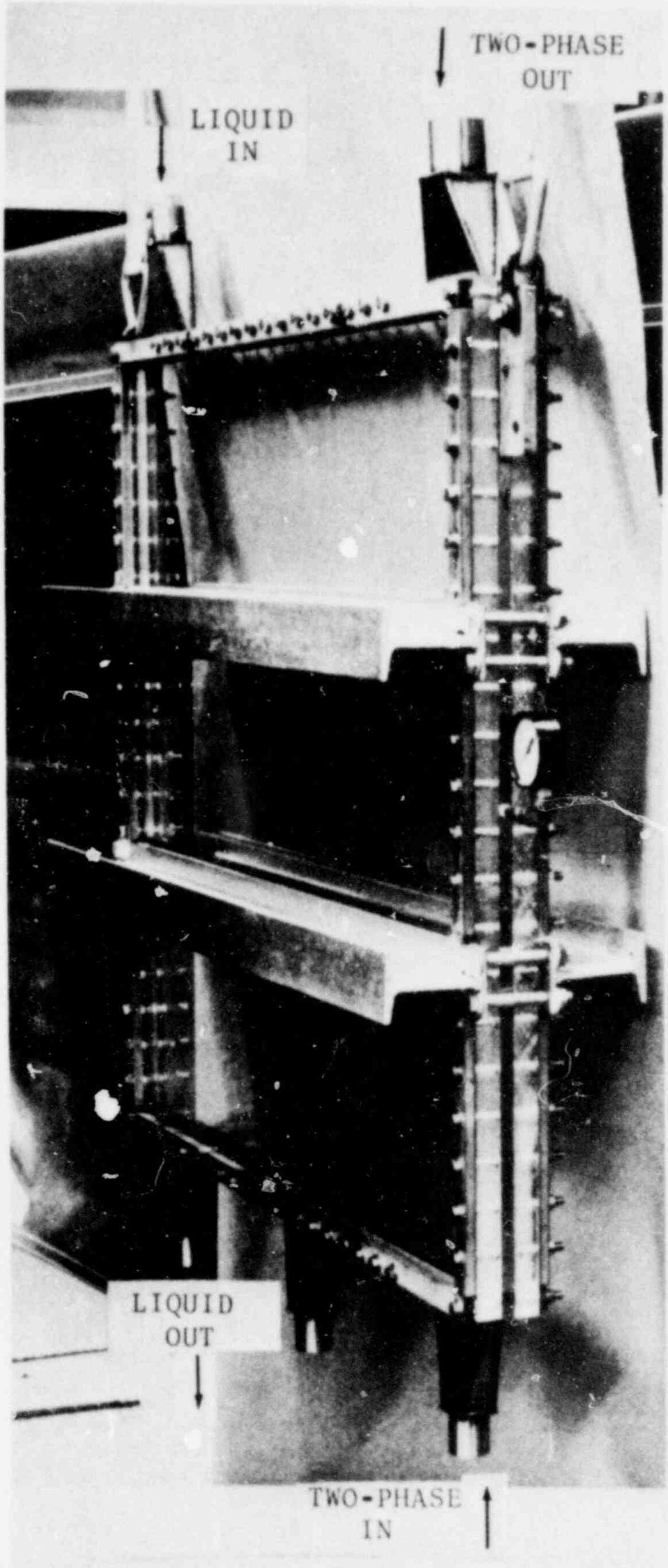


FIGURE 8

2-D Test Section (3' x 3')

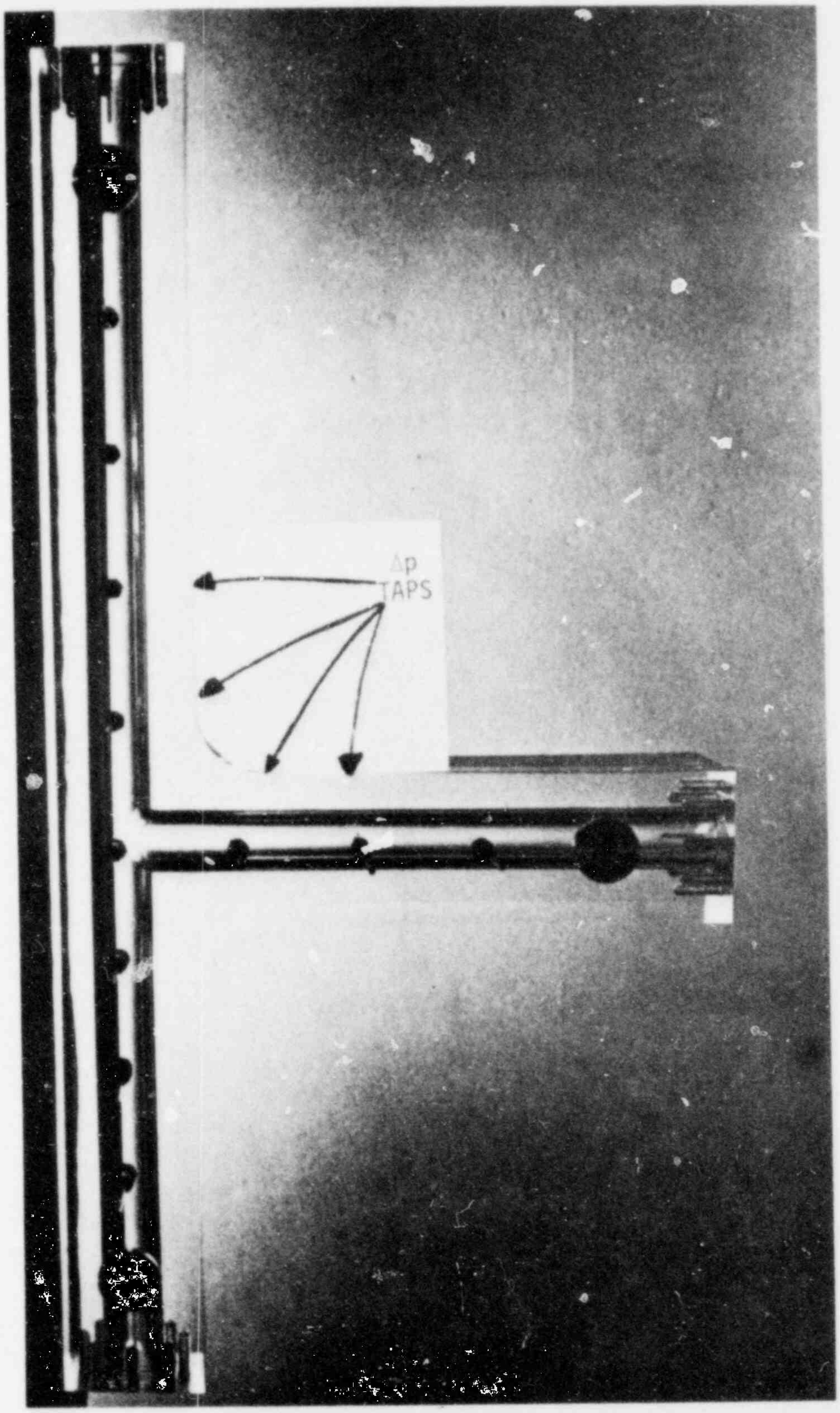


FIGURE 9

Pressure Tap Locations in Tee

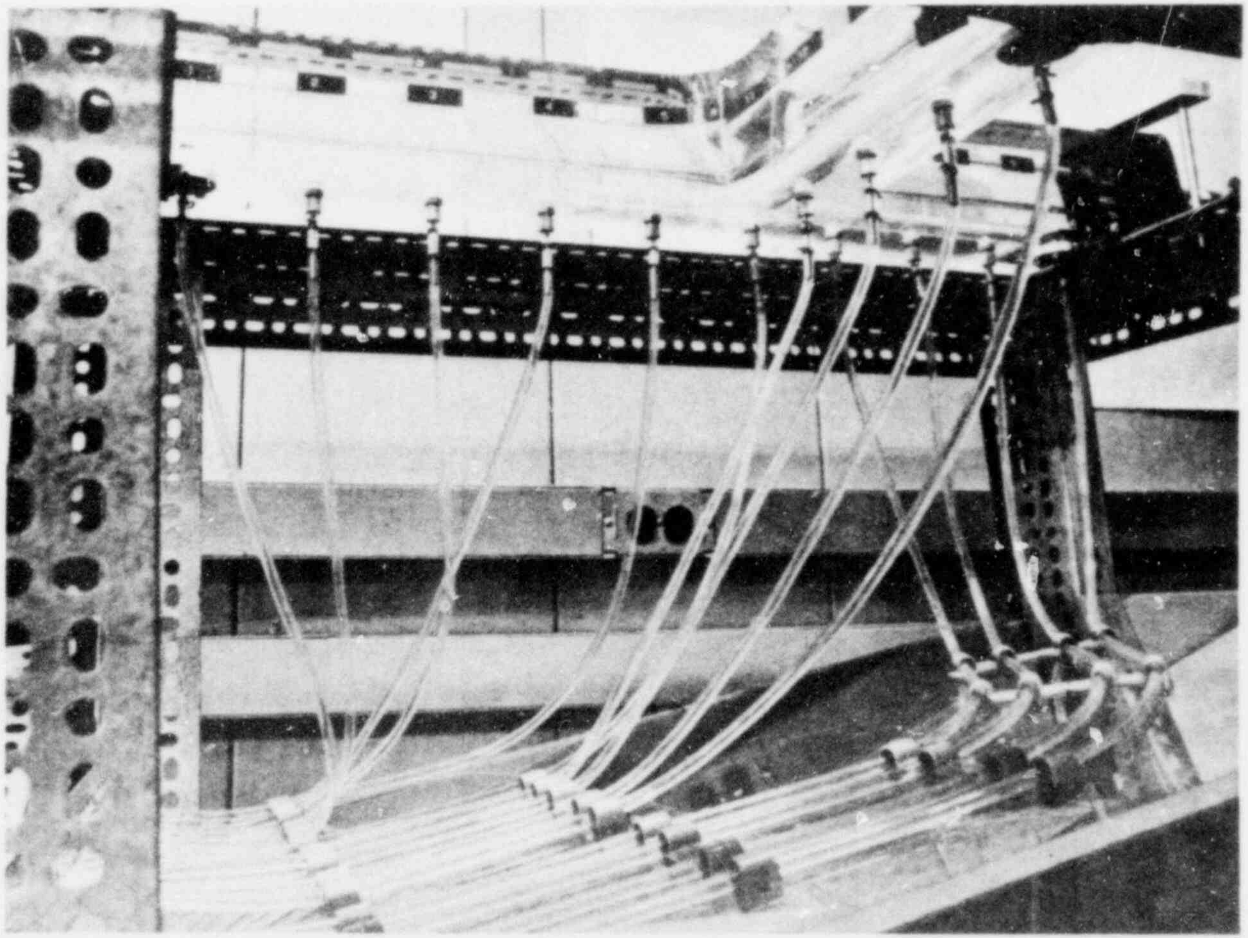
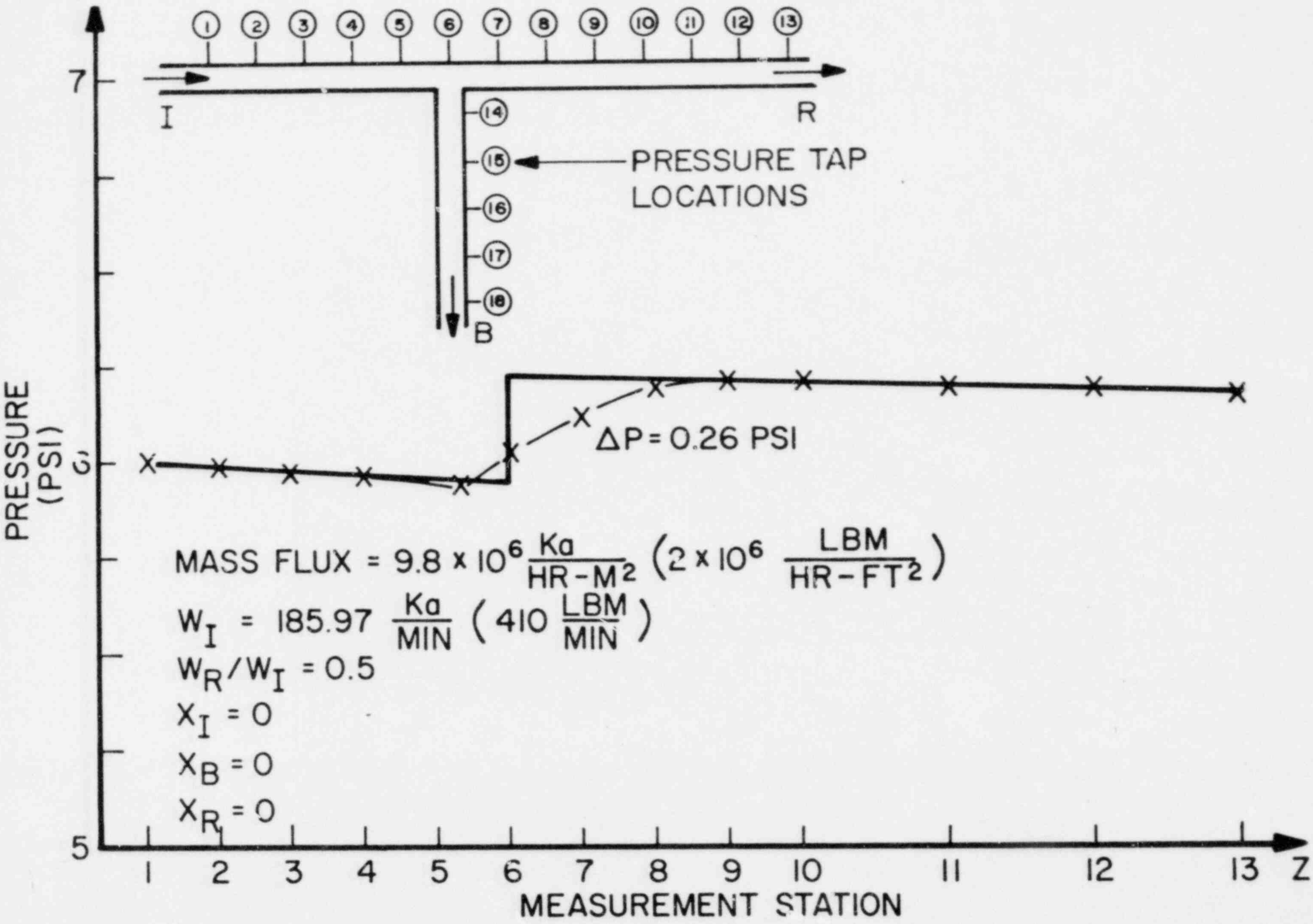


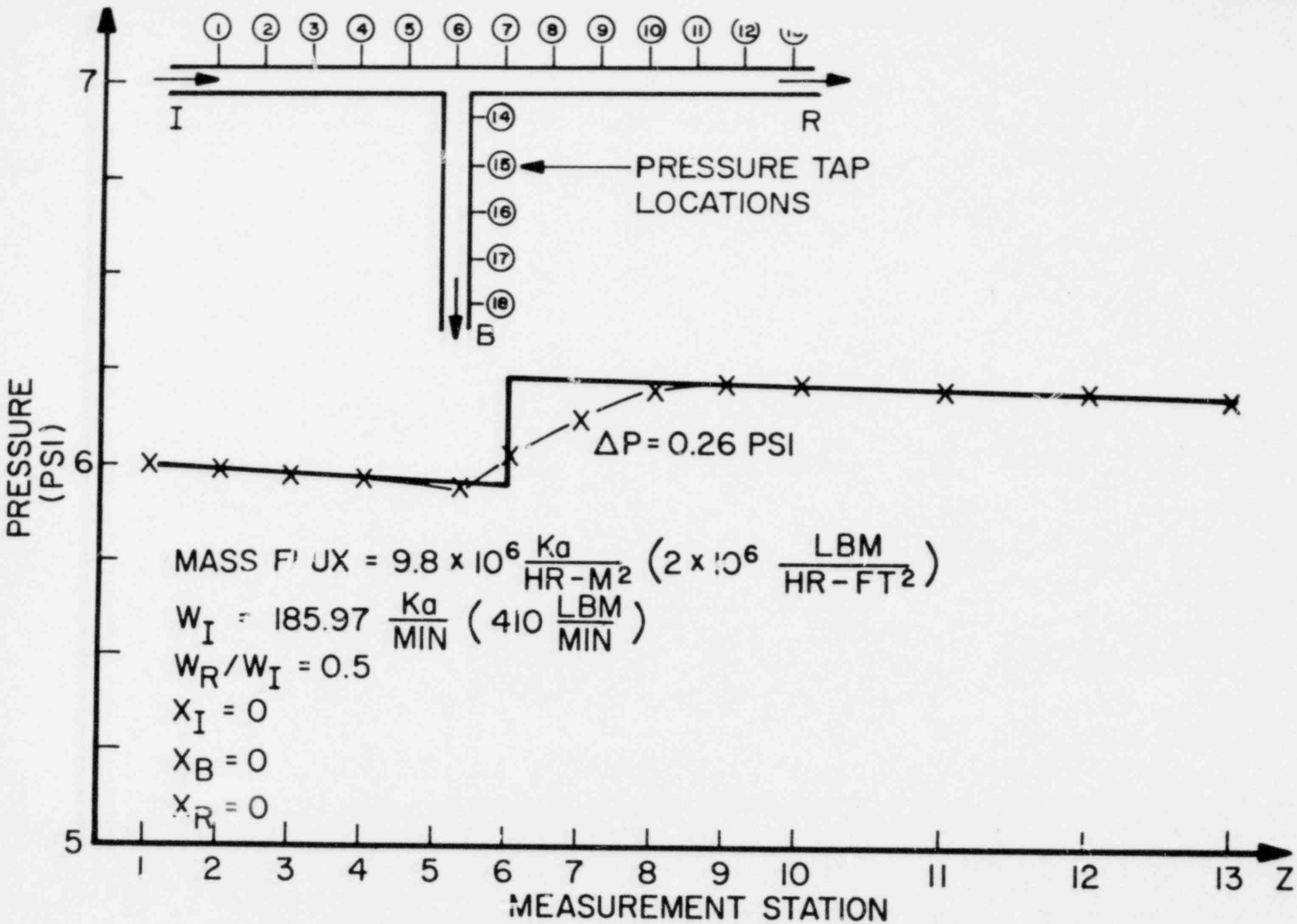
FIGURE 10

Bottom View of test Section



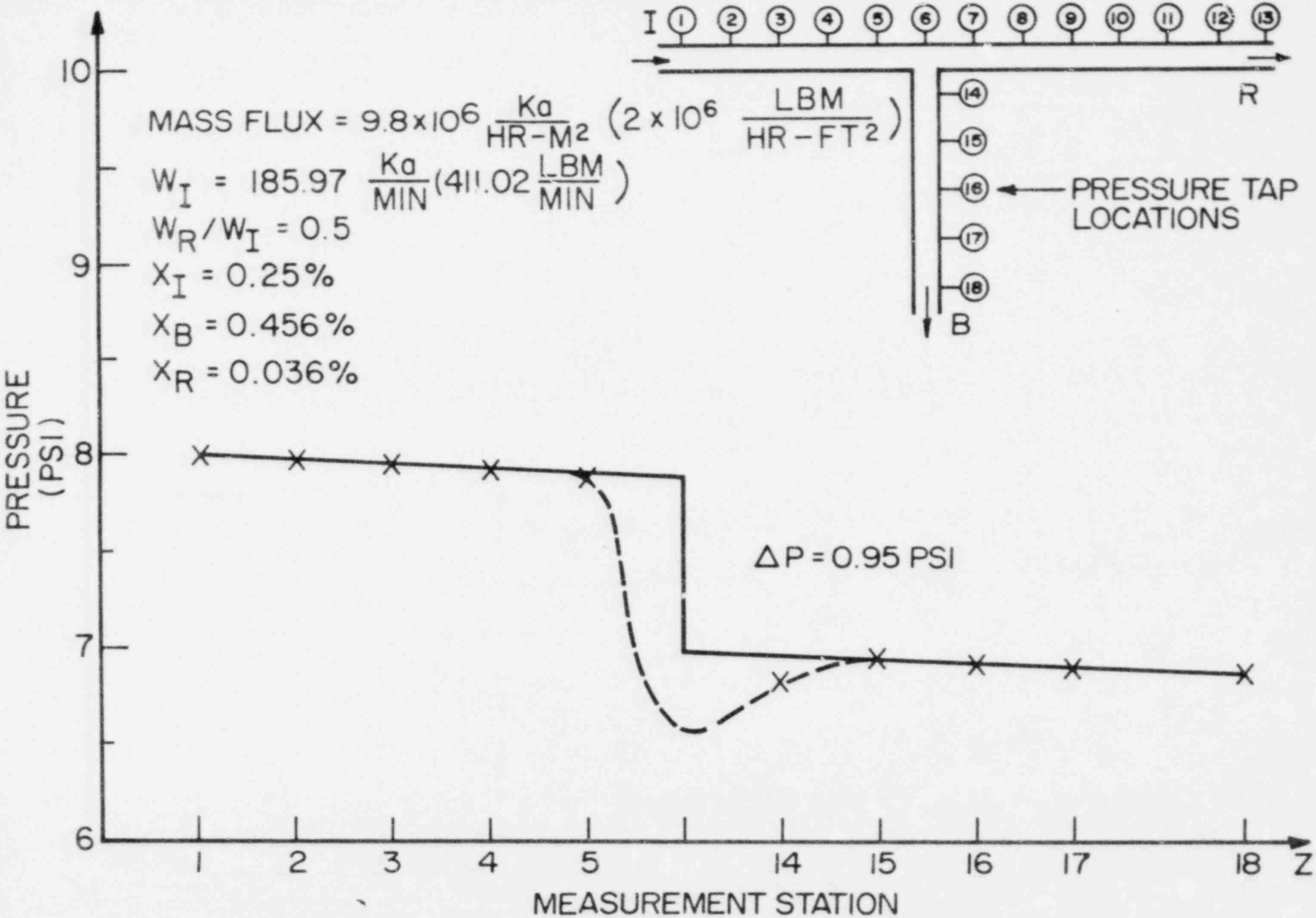
1 2 3 4 5 6 7 8 9 10 11 12 13
14 15 16 17 18
I
R
B
PRESSURE TAP LOCATIONS
$\Delta P = 0.26$ PSI
MASS FLUX = $9.8 \times 10^6 \frac{Kg}{HR-M^2} \left(2 \times 10^6 \frac{LBM}{HR-FT^2}\right)$
W_I = 185.97 $\frac{Kg}{MIN}$ (410 $\frac{LBM}{MIN}$)
$W_R/W_I = 0.5$
$X_I = 0$
$X_B = 0$
$X_R = 0$
PRESSURE (PSI)
7
6
5
1 2 3 4 5 6 7 8 9 10 11 12 13 Z
MEASUREMENT STATION

FIGURE 11A



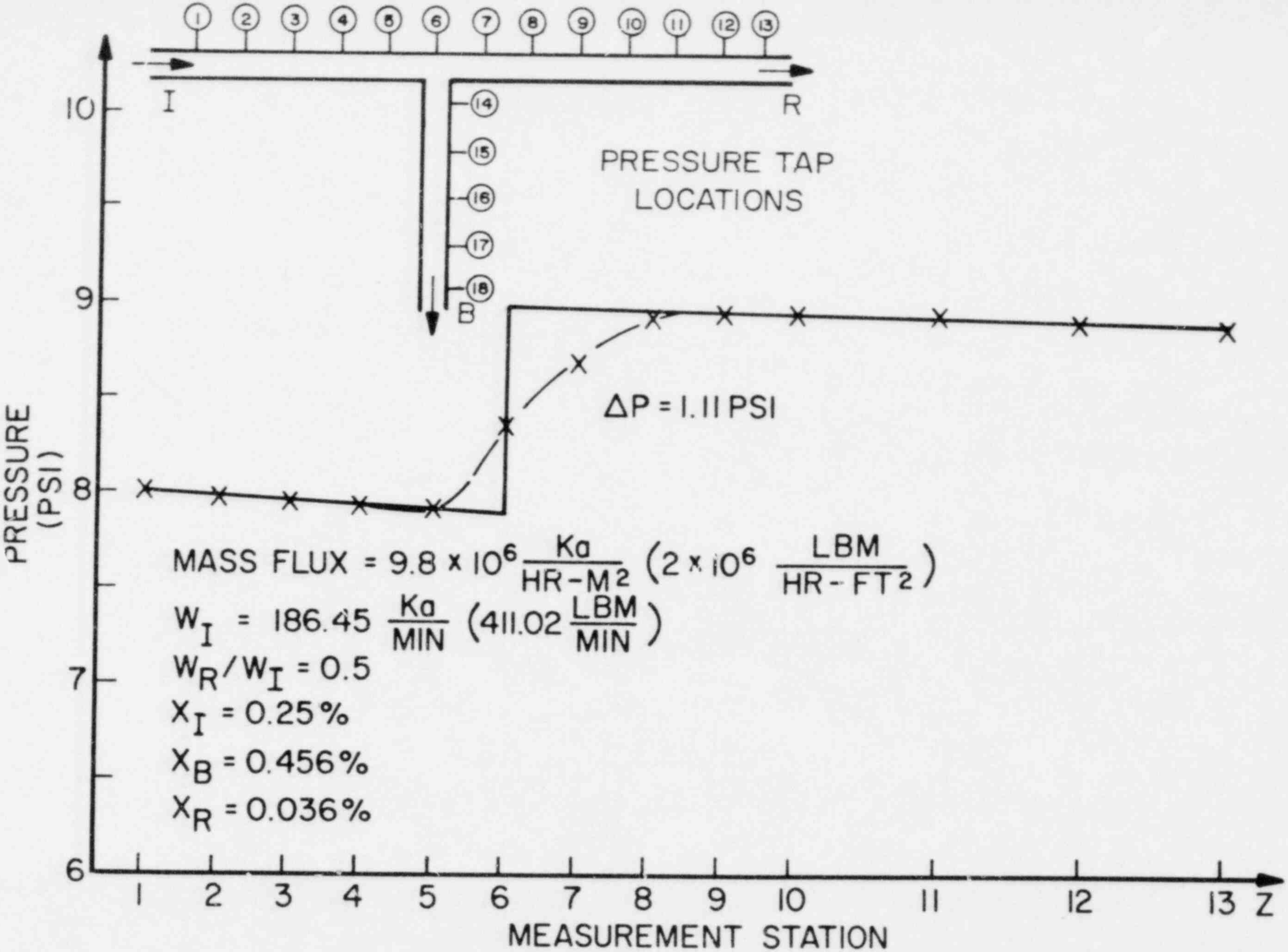
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
I
R
B
PRESSURE TAP LOCATIONS
PRESSURE (PSI)
7
6
5
ΔP = 0.26 PSI
MASS FLUX = 9.8 x 10^6 Kg/HR-M^2 (2 x 10^6 LBM/HR-FT^2)
W_I = 185.97 Kg/MIN (410 LBM/MIN)
W_R/W_I = 0.5
X_I = 0
X_B = 0
X_R = 0
1 2 3 4 5 6 7 8 9 10 11 12 13 Z
MEASUREMENT STATION

FIGURE 11B



I
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
R
B
PRESSURE TAP LOCATIONS
MASS FLUX = 9.8 x 10^6 Kg/HR-M^2 (2 x 10^6 LBM/HR-FT^2)
W_I = 185.97 Kg/MIN (411.02 LBM/MIN)
W_R/W_I = 0.5
X_I = 0.25%
X_B = 0.456%
X_R = 0.036%
ΔP = 0.95 PSI
PRESSURE (PSI)
10
9
8
7
6
MEASUREMENT STATION
Z

FIGURE 12A



1 2 3 4 5 6 7 8 9 10 11 12 13
I
R
14
15
16
17
18
B
PRESSURE TAP LOCATIONS
$\Delta P = 1.11$ PSI
MASS FLUX = $9.8 \times 10^6 \frac{Kg}{HR-M^2} \left(2 \times 10^6 \frac{LBM}{HR-FT^2}\right)$
$W_I = 186.45 \frac{Kg}{MIN} \left(411.02 \frac{LBM}{MIN}\right)$
$W_R / W_I = 0.5$
$X_I = 0.25\%$
$X_B = 0.456\%$
$X_R = 0.036\%$
PRESSURE (PSI)
10
9
8
7
6
1 2 3 4 5 6 7 8 9 10 11 12 13 Z
MEASUREMENT STATION

FIGURE 12B

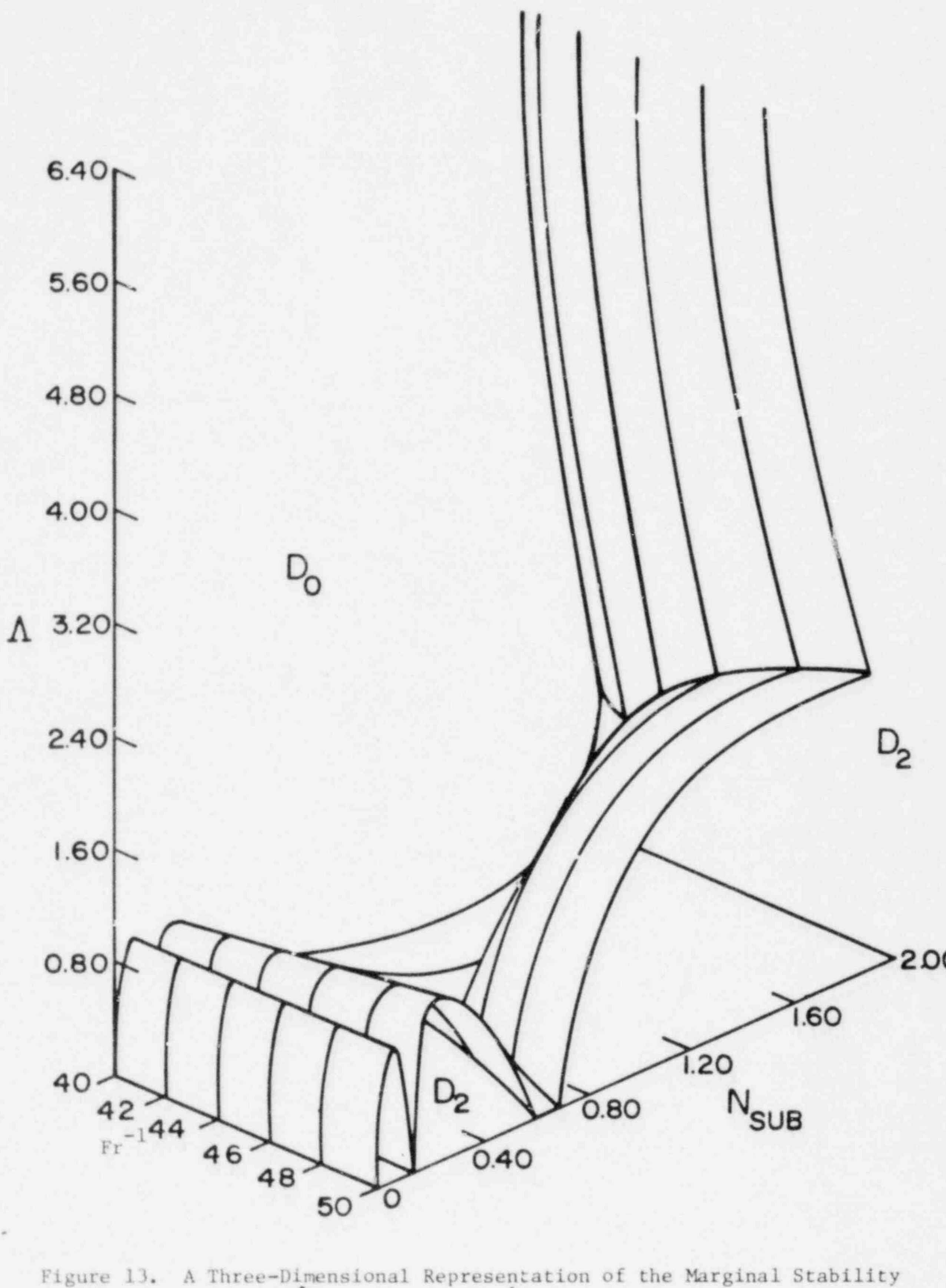


Figure 13. A Three-Dimensional Representation of the Marginal Stability Curves. ($\tilde{j}$=0.3, $40.0 \leq Fr^{-1} \leq 50.0$).

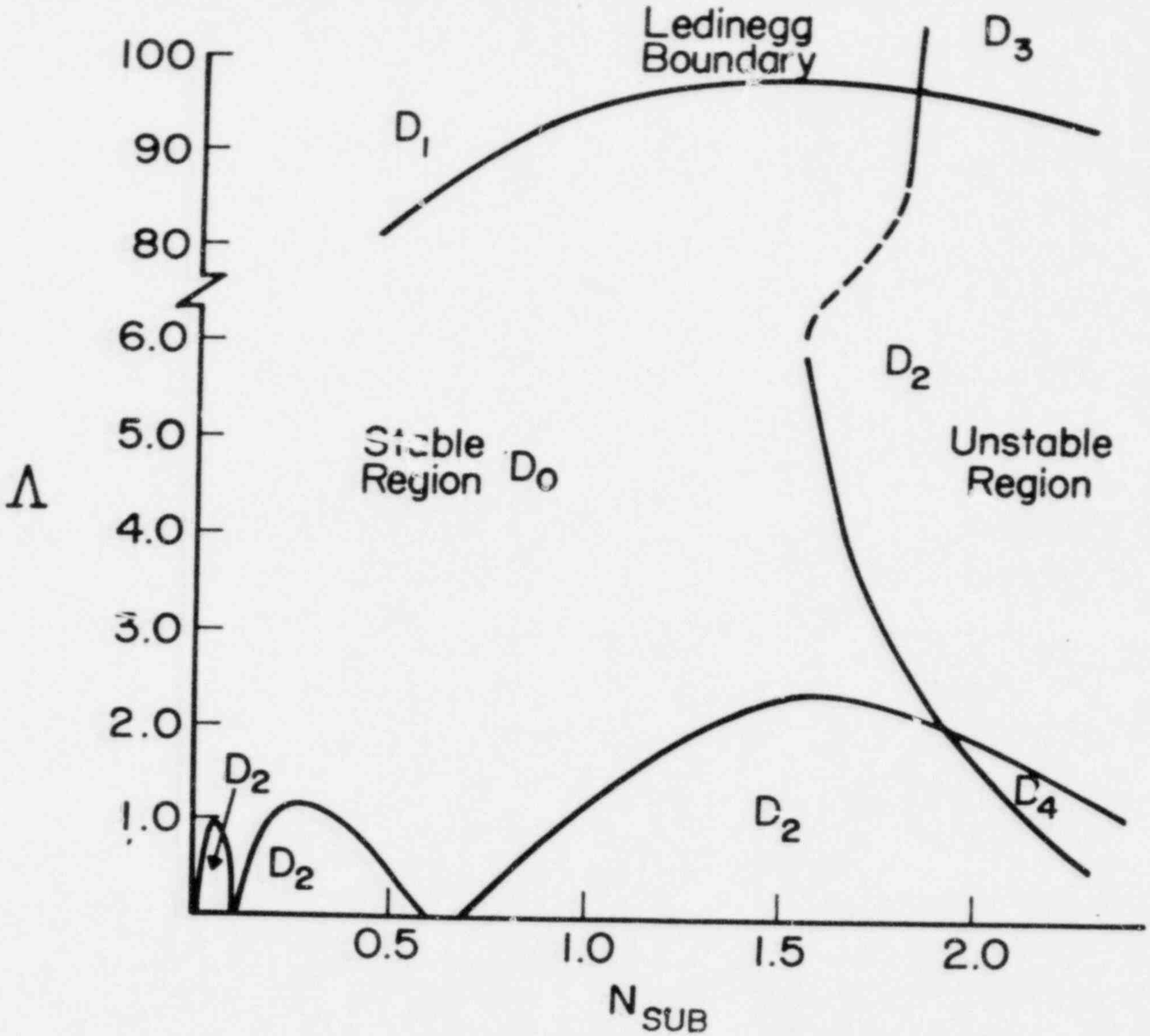


Figure 14. Linear Stability Boundaries, Showing the Ledinegg Boundary. (Fr^{-1}=50, $\tilde{j}$=0.3)

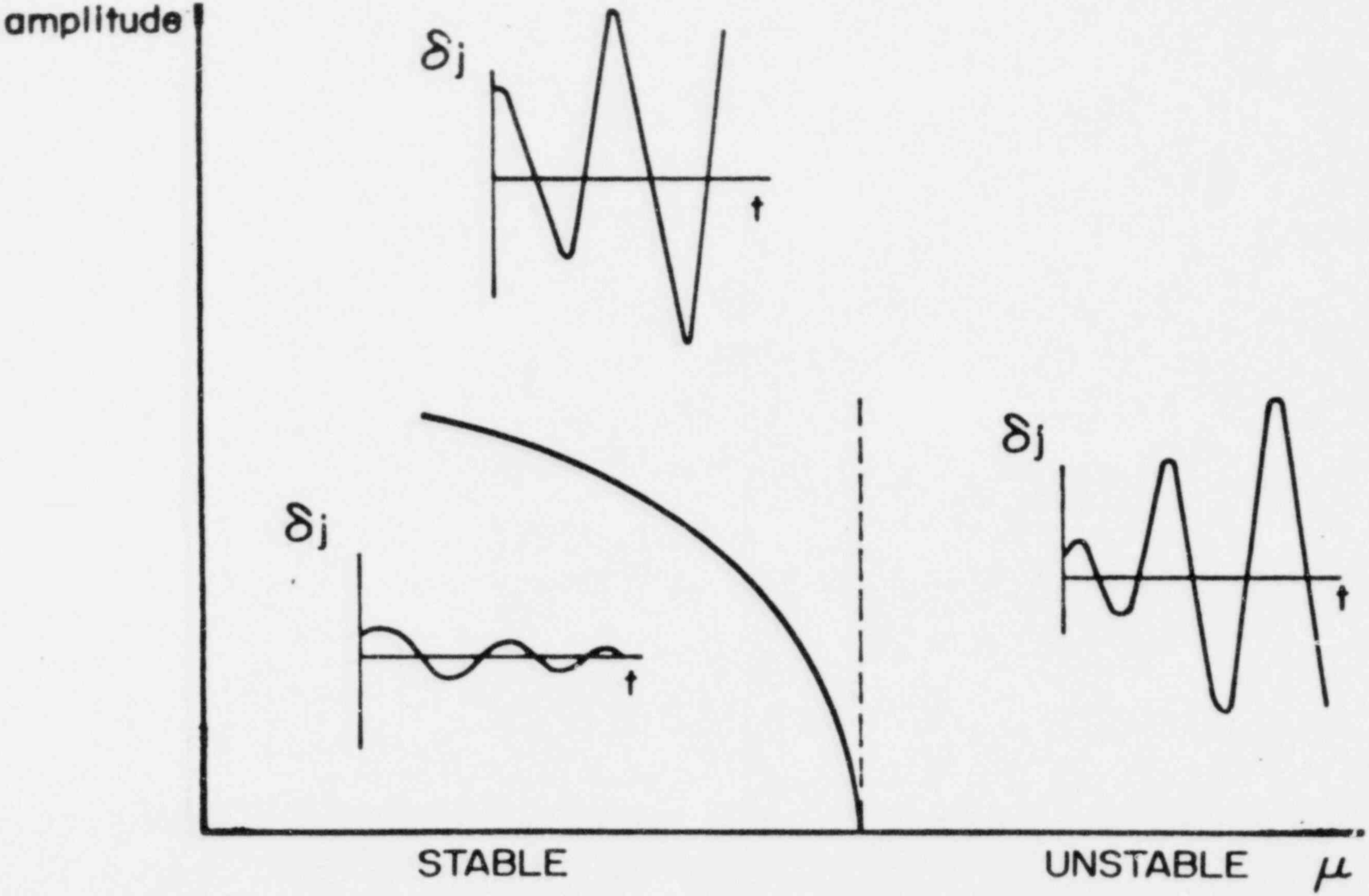


Fig. 15 *Subcritical bifurcation*

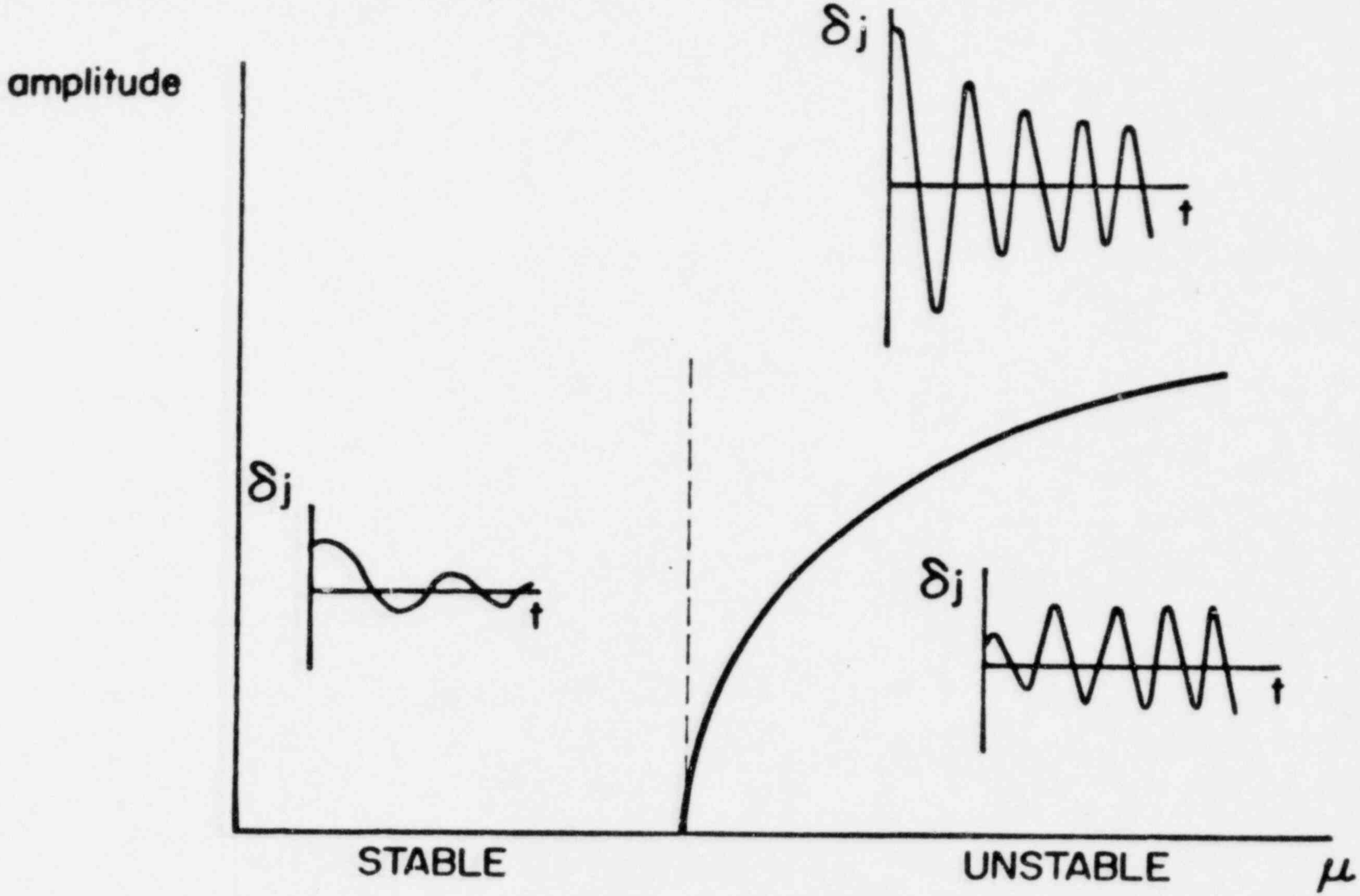


Fig. 16 Supercritical bifurcation

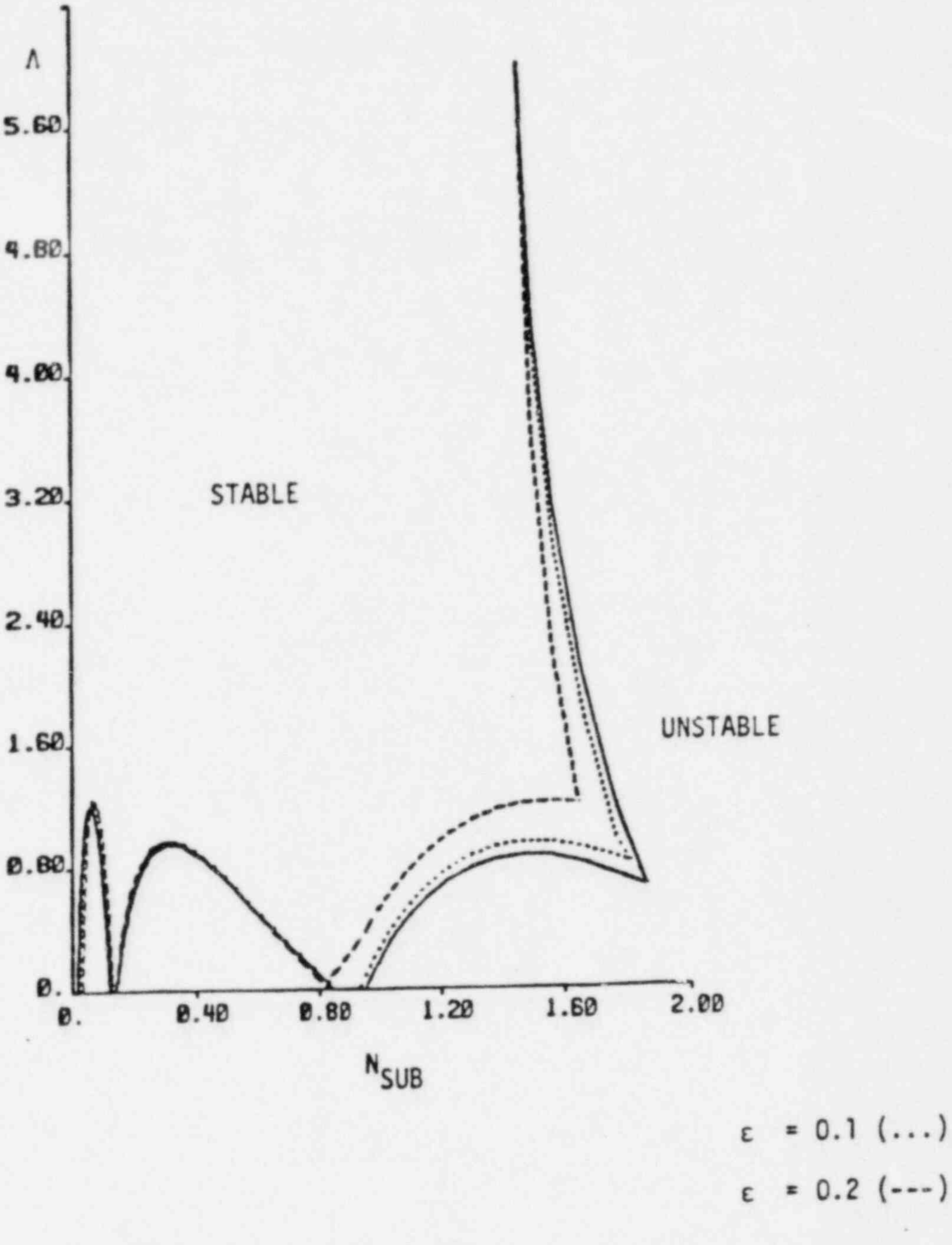


FIGURE 17

FINITE AMPLITUDE INSTABILITY

SURFACES for: $\tilde{j} = 0.325$, $Fr^{-1} = 50$

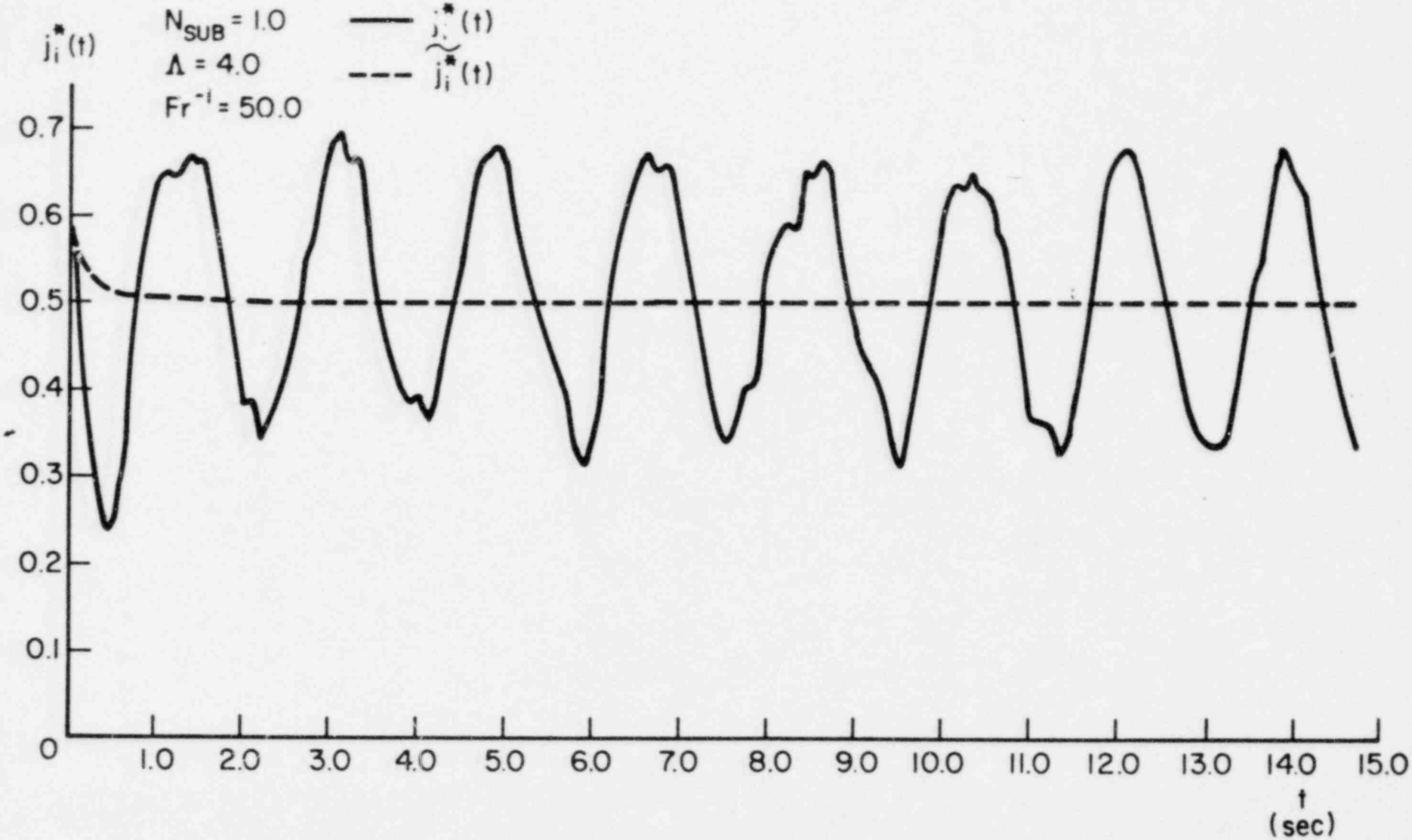

$j_i^*(t)$
$N_{SUB} = 1.0$
$\Lambda = 4.0$
$Fr^{-1} = 50.0$
$j_i^*(t)$
$\tilde{j}_i^*(t)$
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
11.0
12.0
13.0
14.0
15.0
t
(sec)

FIGURE 18

AUXILIARY FEEDWATER RELIABILITY IN PWRS

BY

G. E. EDISON

DIVISION OF SYSTEMS AND RELIABILITY RESEARCH
OFFICE OF NUCLEAR REGULATORY RESEARCH
U. S. NUCLEAR REGULATORY COMMISSION

B A C K G R O U N D

1975 WASH-1400 ESTABLISHES AFW TO BE IMPORTANT SAFETY SYSTEM IN DOMINANT ACCIDENT SEQUENCES IN PWRS

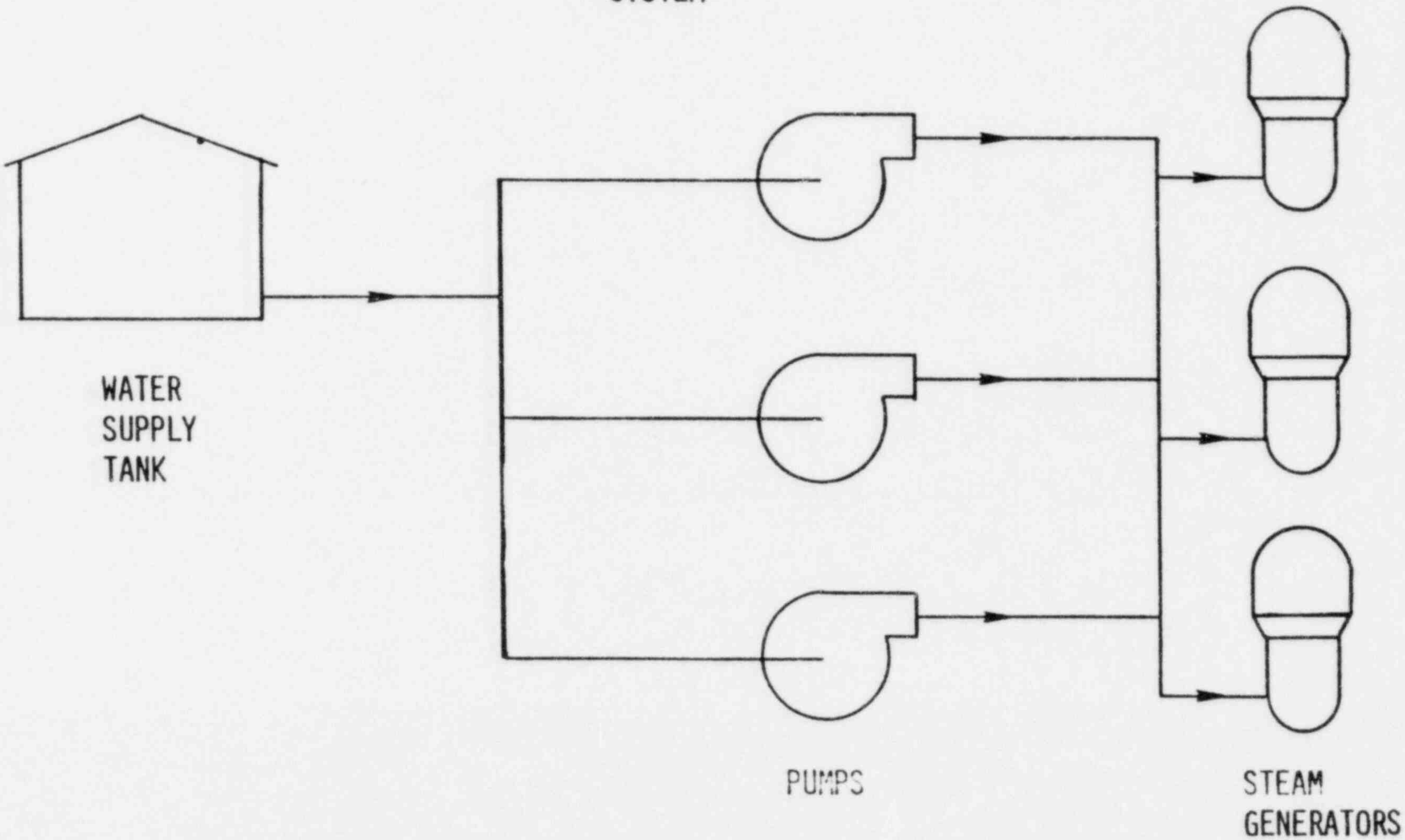
1977-78 STUDIES AT WESTINGHOUSE AND PIECEMEAL INFO AT NRC BEGIN TO DISCOVER RELIABILITY FAULTS IN PWR AUXILIARY FEEDWATER SYSTEMS

MARCH 1979 · TMI-2 ACCIDENT

- FEEDWATER TRANSIENT
- AFW SYSTEM UNAVAILABLE FOR 8 MINUTES

MAY 1979-1980 ALL PWR AUXILIARY FEED SYSTEMS REVIEWED BY NRC WITH UTILITIES, COMPARATIVE RELIABILITY ANALYSES MADE

SCHEMATIC DIAGRAM
AUXILIARY FEEDWATER
SYSTEM



NRC CRITERIA FOR RELIABILITY OF AUXILIARY FEEDWATER SYSTEM

- NO QUANTITATIVE LIMITS FOR SYSTEM RELIABILITY
- AT LEAST 2 FULL-CAPACITY INDEPENDENT FLOW TRAINS THAT INCLUDE DIVERSE POWER SOURCES (BRANCH TECHNICAL POSITION ASB 10-1)
- WITHSTAND SINGLE ACTIVE FAILURE (STANDARD REVIEW PLAN, SECTION 10.4.9)
- OTHER GENERAL DESIGN CRITERIA AND REG. GUIDES SPECIFIED IN SECTION 10.4.9 RELATING TO SURVEILLANCE TESTING, REDUNDANT INSTRUMENTS, LOSS OF OFFSITE POWER, TECHNICAL SPECIFICATIONS

AUXILIARY FEEDWATER RELIABILITY SAFETY ISSUES

- RELIABILITY OF AFW (DECAY HEAT REMOVAL) WHEN MAIN FEEDWATER SYSTEM FAILS
- RELIABILITY OF AFW WHEN OFFSITE POWER IS LOST
- RELIABILITY OF AFW WHEN OFFSITE POWER AND EMERGENCY ONSITE AC POWER FAILS

REFERENCES

- o WASH-1400 (1975)
- o WARD-SR-3045-5 (1978)
- o PROC. ANS MEETING IN L.A., V.3, PG. X.6-1 (MAY 8, 1978)
- o NUREG-0611, APP. III (1980)
- o NUREG-0635, APP. III (1980)
- o NUREG- ? (ANALYSES FOR B&W PLANTS, TO BE PUBLISHED)

RESULTS: ANALYTICAL

- DOMINANT FAILURE MODES IDENTIFIED

  EXAMPLES: (1) AC POWERED LUBE OIL PUMP IN STEAM-DRIVEN PUMP TRAIN

  (2) MANUAL ACTUATION OF AFW

  (3) INTEGRATED CONTROL SYSTEM SIGNAL

- OUTLIER PLANTS WITH RELATIVELY LOW AFW RELIABILITY IDENTIFIED

- AFW RELIABILITY VARIED MORE THAN 2 ORDERS OF MAGNITUDE OVER ALL PWRS

RESULTS: AFW RECOMMENDATIONS BEING IMPLEMENTED AT OPERATING PWRS

- UNDESIRABLE AC POWER DEPENDENCIES ELIMINATED
- OPPORTUNITIES FOR HUMAN ERROR REDUCED
- DIVERSITY ADDED
- SINGLE FAILURE POINTS ELIMINATED

RESULTS: AFW RECOMMENDATIONS BEING IMPLEMENTED AT OPERATING PWRS

- o PROCEDURES CHANGED
- o ALLOWABLE OUTAGE TIME REDUCED
- o AUTOMATIC ACTUATION ADDED
- o ELIMINATE UNDESIRABLE CONTROL CIRCUIT INTERACTIONS WITH AFW
- o OTHERS

ANALYSIS OF PRESSURIZED WATER REACTOR STATION BLACKOUT

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

C. D. Fletcher
B. F. Saffell
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

ANALYSIS OF PRESSURIZED WATER REACTOR STATION BLACKOUT

C. D. Fletcher
B. F. Saffell
EG&G Idaho, Inc.

Station blackout for a pressurized water reactor (PWR) is defined as a loss of offsite and onsite AC power. Offsite power may be lost due to conditions external or internal to the plant. Onsite power may be lost if the diesel generators fail to start and load. Should a station blackout occur, turbine-driven auxiliary feedwater is available as a backup means for removing core decay heat from the primary system. Should the turbine-driven auxiliary feedwater system fail, however, the core decay heat is sufficient to raise the primary system pressure to the code safety valve setpoint beyond which primary system coolant inventory is expelled; a process which eventually leads to core damage. Analysis of this severe accident sequence will serve to enhance understanding of a plant's response to complete station blackout. Information from this study will be used to support resolution of NRC's Unresolved Safety Issue A-44.

This severe accident sequence was studied for a PWR designed by each of the major vendors, Westinghouse (W), Combustion Engineering (CE) and Babcock and Wilcox (B&W). Thermal-hydraulic calculations were performed using the RELAP4/MOD7 computer code. The plants were modeled using best estimate boundary and initial conditions.

The results of the calculations are summarized as follows. Upon loss of offsite power, main feedwater and steam functions are lost. The steam generator secondary pressures rapidly increase and secondary coolant inventory is lost through the safety valves, thereby lowering the secondary level. Secondary system dryout occurs at 2940 s (W), 2900 s (CE), and 200 s (B&W). Before secondary dryout, primary pressure is stabilized by heat removal to the secondaries; following secondary dryout the primary pressure increases rapidly to the code safety valve setpoint. Natural circulation loop flow continues approximately 1200 s past secondary dryout as the secondaries continue to act as heat sinks because the secondary vapor is being superheated. Eventually enough primary inventory is lost through the code safety valve to allow uncovery of the core to begin at 5800 s (W), 6200 s (CE), and 2200 s (B&W). Core damage is estimated to commence approximately 300 s following the beginning of uncovery.

The next phase of study investigated the mitigation of the severe accident by (a) the starting and loading of at least one diesel generator, and (b) the delivery of turbine-driven auxiliary feedwater at intermediate times of the accident. RELAP4/MOD7 calculations were performed to determine the latest times at which these mitigation techniques may be employed in order to avoid core uncovery. The results indicated core uncovery may be avoided if a diesel generator is started by 4800 s (W), and 1840 s (B&W). The CE designed plant does not have motor driven auxiliary feedwater pumps so that at least one turbine-driven auxiliary feedwater pump must be started to avoid core uncovery. The latest effective start times for turbine-driven auxiliary feedwater were determined to be 4200 s (W), 5200 s (CE), and 1600 s (B&W).

Analysis of a complete station blackout with the loss of turbine-driven auxiliary feedwater has resulted in the quantificaton of critical times such as the time for (a) restoration of offsite power, (b) availability of turbine-driven auxiliary feedwater, or (c) initiation of diesel power. The calculated times were found to include a period of natural circulation effects beyond steam generator dryout. Finally, the effectiveness of an alternate decay heat removal system to further mitigate the potential consequences of a complete station blackout is currently being pursued with results to be reported in the near future.

The results of this effort provide assistance to NRC in the resolution of the Unresolved Safety Issue A-44 by identifying the actions and the critical implementation times required to mitigate the potentially severe consequences associated with this event.

PRESSURIZED WATER REACTOR STATION BLACKOUT

by
B.F. SAFFELL

EG&G Idaho, Inc.

STATION BLACKOUT EVALUATION

- DEFINITION AND DESCRIPTION
- ANALYSES
- RESULTS
- SUMMARY

BFS-1

- STATION BLACKOUT DEFINITION

  LOSS OF OFFSITE POWER

  FAILURE TO START DIESEL GENERATORS

- SEVERE ACCIDENT INVESTIGATED

  STATION BLACKOUT WITH A FAILURE

  TO START TURBINE-DRIVEN AUXILIARY FEEDWATER

- UNRESOLVED SAFETY ISSUE A-44

BFS-2

ANALYSES PERFORMED

WESTINGHOUSE

COMBUSTION ENGINEERING

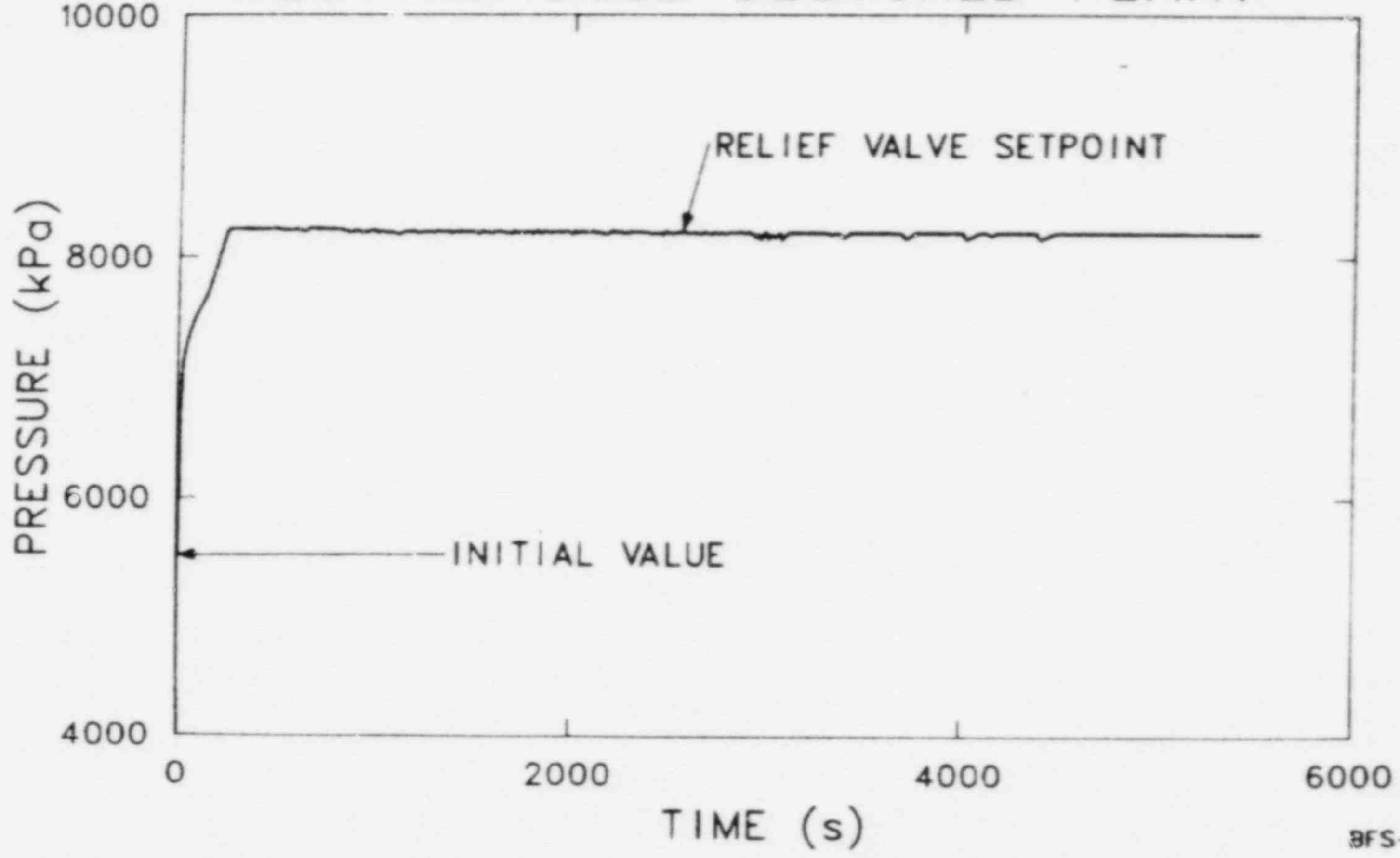
BABCOCK AND WILCOX

RELAP4/MOD7 COMPUTER CODE

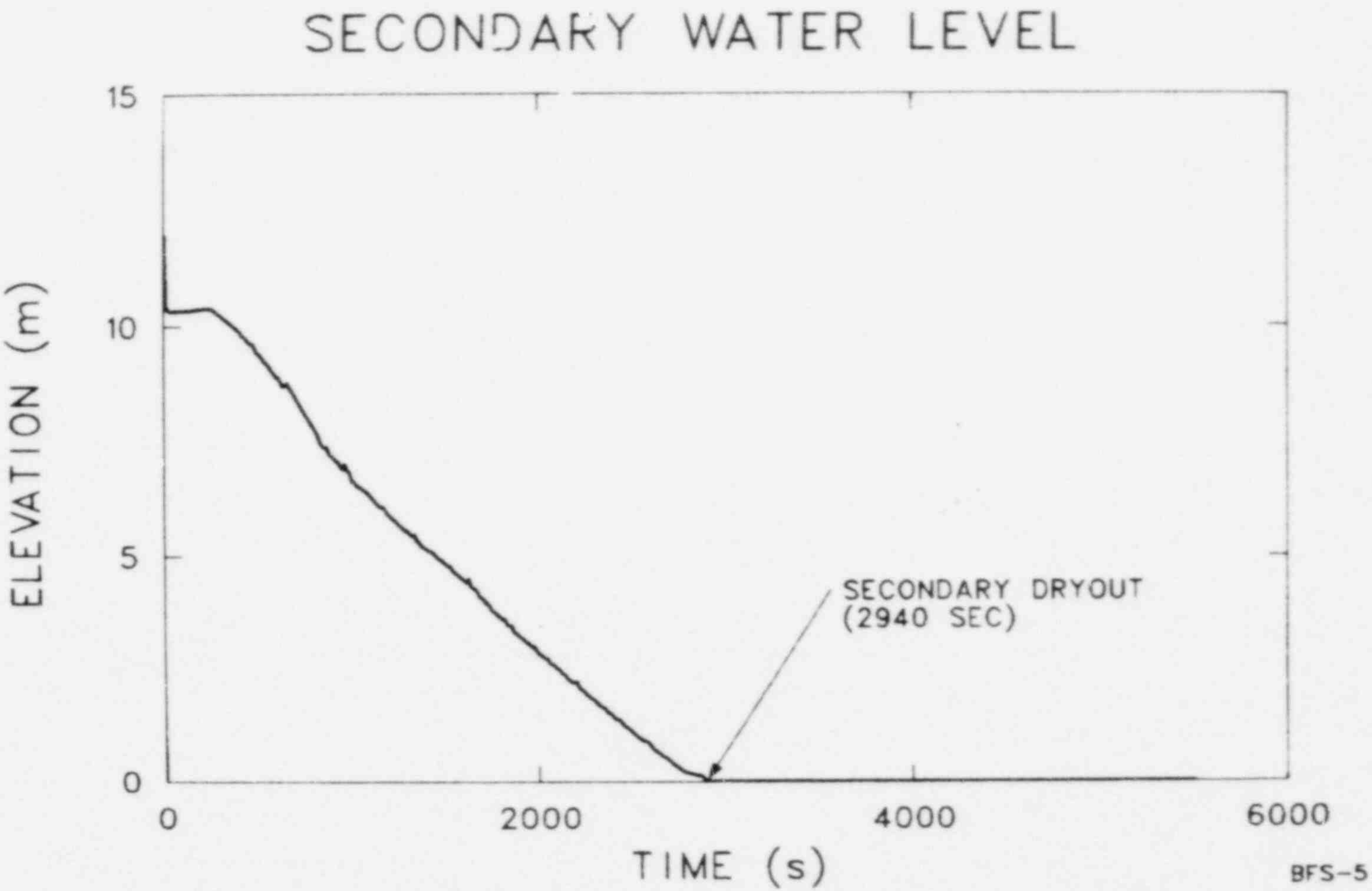
BEST ESTIMATE BOUNDARY AND INITIAL CONDITIONS

BFS-3

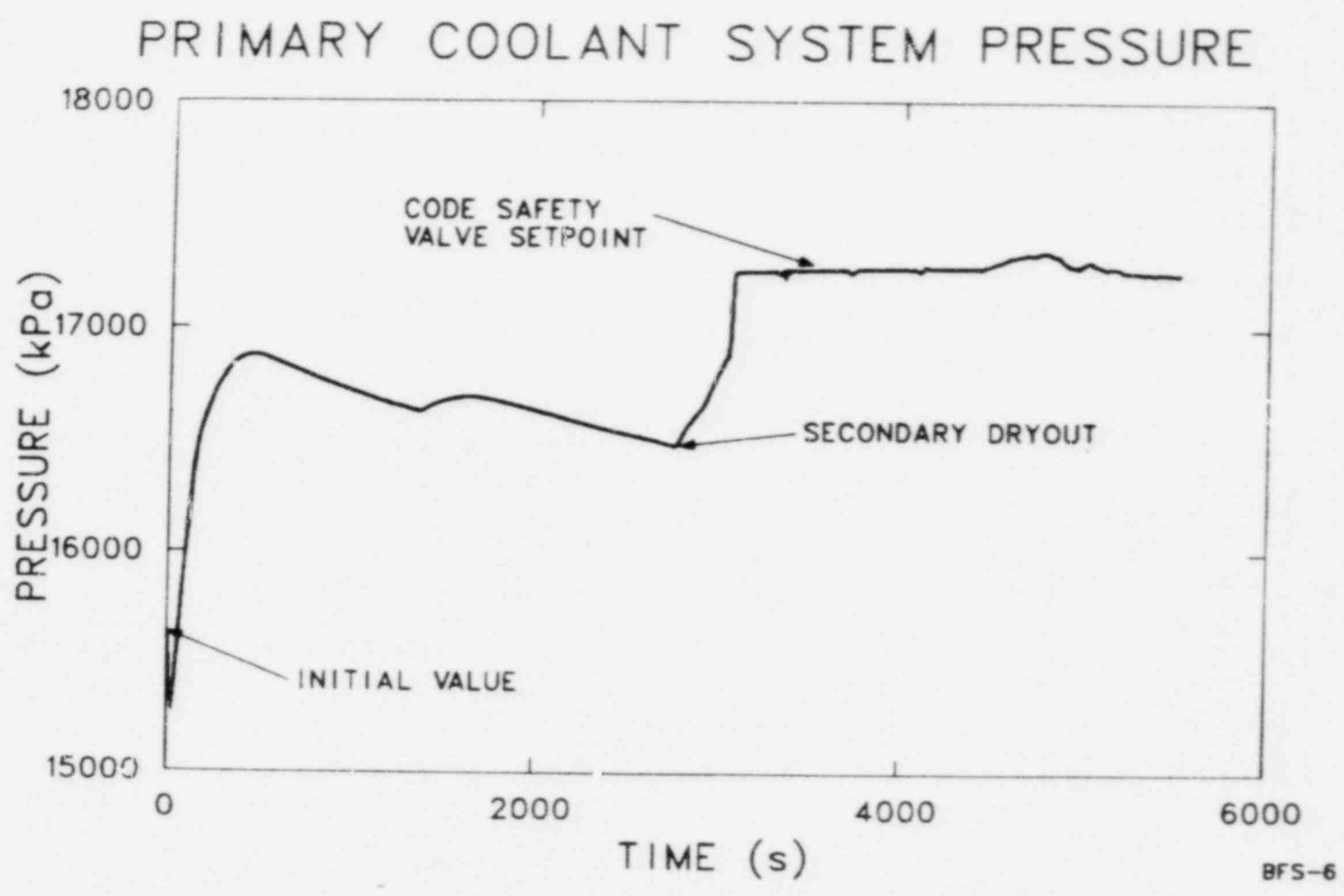
SECONDARY PRESSURE FOR A
WESTINGHOUSE DESIGNED PLANT

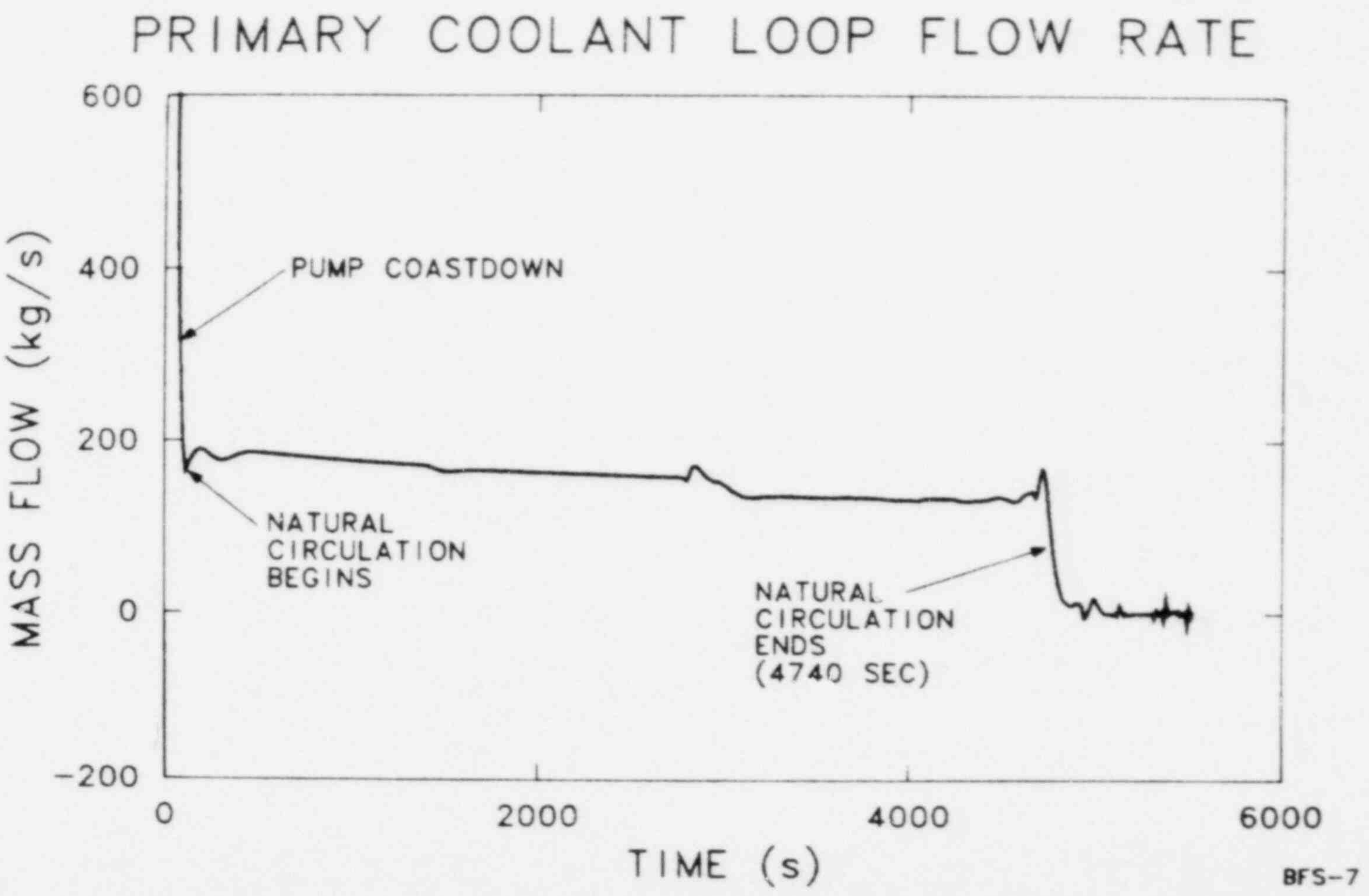


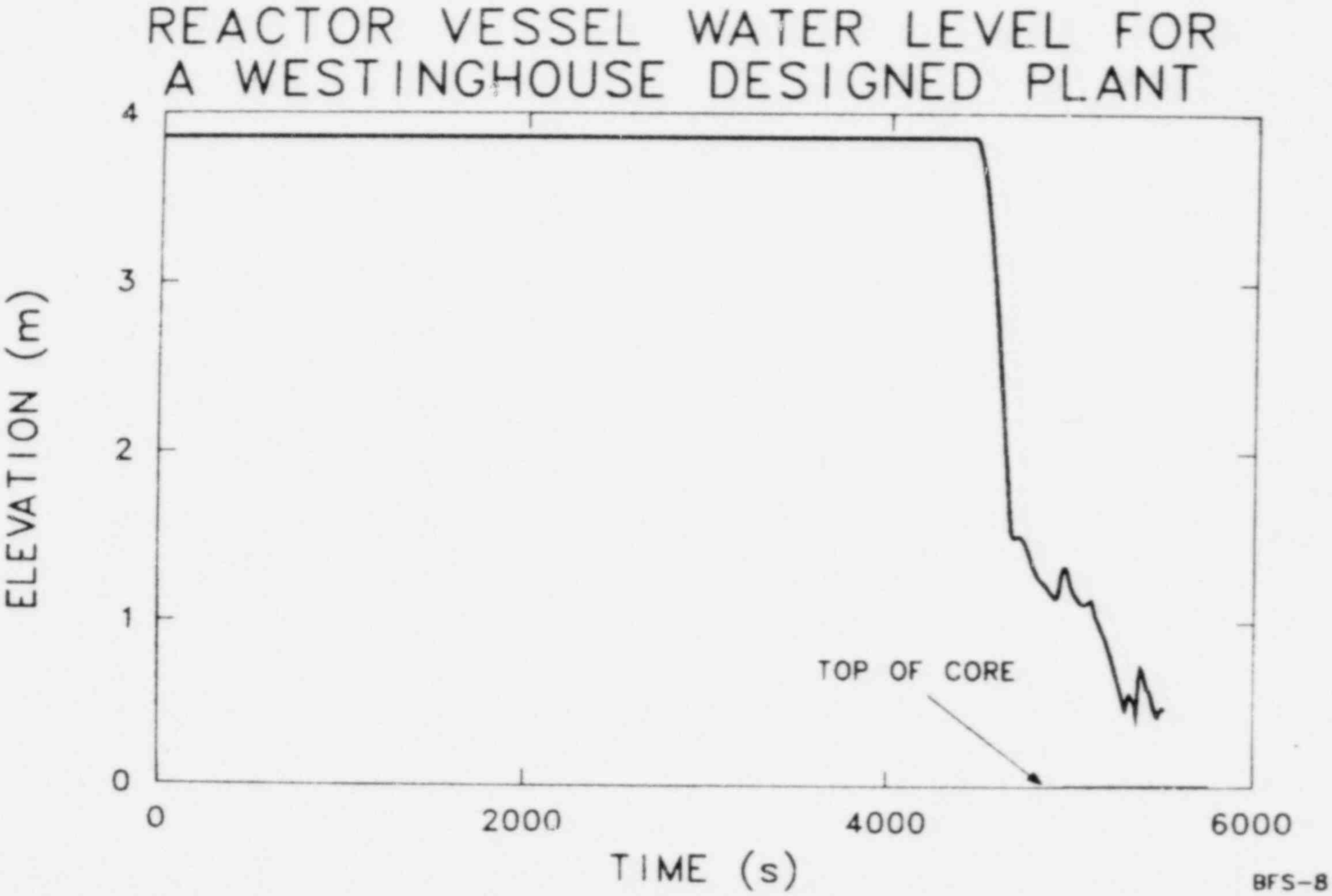
BFS-4



SECONDARY WATER LEVEL
15
10
5
0
ELEVATION (m)
0
2000
4000
6000
TIME (s)
SECONDARY DRYOUT
(2940 SEC)
BFS-5


PRIMARY COOLANT SYSTEM PRESSURE
PRESSURE (kPa)
18000
17000
16000
15000
0
2000
4000
6000
TIME (s)
CODE SAFETY VALVE SETPOINT
SECONDARY DRYOUT
INITIAL VALUE
BFS-6


PRIMARY COOLANT LOOP FLOW RATE
600
400
200
0
-200
MASS FLOW (kg/s)
PUMP COASTDOWN
NATURAL
CIRCULATION
BEGINS
NATURAL
CIRCULATION
ENDS
(4740 SEC)
0
2000
4000
6000
TIME (s)
BFS-7



REACTOR VESSEL WATER LEVEL FOR
A WESTINGHOUSE DESIGNED PLANT
ELEVATION (m)
0
1
2
3
4
0
2000
4000
6000
TIME (s)
TOP OF CORE
BFS-8

RESULTS OF STATION BLACKOUT WITH FAILURE OF TURBINE-DRIVEN AUXILIARY FEEDWATER ANALYSES

PLANT	S.G. DRYOUT (s)	BEGIN CORE UNCOVERY (s)
WESTINGHOUSE	2940	5800
BABCOCK AND WILCOX	200	2200
COMBUSTION ENGINEERING	2900	6200

BFS-9

RESULTS OF STATION BLACKOUT STUDY INCLUDING EFFECTS OF MITIGATING ACTION

PLANT	TURBINE AFW DELAYED START (s)	DIESEL GENERATOR DELAYED START (s)
WESTINGHOUSE	4200	4800
BABCOCK AND WILCOX	1600	1840
COMBUSTION ENGINEERING	5200	--

BFS-11

MITIGATING ACTION

- START AND LOAD AT LEAST ONE DIESEL GENERATOR
- START AND DELIVER TURBINE-DRIVEN AUXILIARY FEEDWATER

SUMMARY

- PWR STATION BLACKOUT WITH FAILURE OF TURBINE DRIVEN AUXILIARY FEEDWATER
- THERMAL-HYDRAULIC CALCULATIONS DEFINE TIMING OF SEQUENCE AND LIMITING TIMES FOR MITIGATING ACTIONS
- RESULTS INPUT TO TAP A-44
- ALTERNATE HEAT REMOVAL SYSTEMS

BFS-12

LA-UR-80-2811

TITLE: LOSS-OF-FEEDWATER TRANSIENTS IN PWRs

AUTHOR(S): Robert D. Burns III

SUBMITTED TO: Eighth Water Reactor Safety Research Information Meeting
Nuclear Regulatory Commission
October 27-31, 1980

los alamos
scientific laboratory
of the University of California
LOS ALAMOS, NEW MEXICO 87545

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St. No. 2629
1/78

DEPARTMENT OF ENERGY
CONTRACT W-7405-ENG. 36

LOSS-OF-FEEDWATER TRANSIENTS IN PWRs

Robert D. Burns III

The TMI-2 accident demonstrated that the nuclear industry must be prepared for events involving multiple equipment failures[1]. The purpose of the LWR severe accident sequence analysis (SASA) research at the Los Alamos Scientific Laboratory (LASL) is to provide the NRC with a technical basis for judging the adequacy of plant operating procedures for dealing with such events. The LASL approach is to (1) identify potential types of multiple equipment failures at specific US commercial nuclear power plants[2], (2) perform computer simulations of postulated accident initiators including anticipated equipment malfunctions and operator actions throughout the event, and (3) determine key primary system responses, timing, and significance of events.

Recent SASA work in LASL's Multifault Accident Analysis Section has focused on loss-of-feedwater (LOFW) transients at a 4-loop Westinghouse nuclear power reactor[3]. This class of accidents was selected because of NRC concern over auxiliary feedwater (AFW) system reliability and because of concern over the capability of plants to "feed and bleed" in total LOFW situations.

In all transients studied, the initiator was loss of main feedwater and reactor coolant pump (RCP) trip, caused by temporary loss of off-site power. Subsequent automatic actions included reactor scram, closure of the main steam isolation valves, and initiation of AFW flow. TRAC-PD2 calculations were designed to study the consequences of AFW delivery rates below the minimum specified in the emergency operating procedures (EOPs) for the reference 4-loop plant.

Six types of LOFW scenarios have been studied, including (1) zero AFW availability (nominal case), (2) initially zero AFW but full recovery after 2 h, (3) zero AFW with steam generator (SG) atmospheric relief valve (ARV) malfunction, (4) zero AFW with high pressure charging flow initiated after 2 h, and (5) zero AFW with delay in reactor scram. Additional cases were considered to study the effects of uncertainties in pressurizer heater/spray operation, operator manual initiation of high pressure charging flow, reactor initial conditions, and RCP and power coastdown characteristics. Nominal case results, rationale for selections of other cases, and lessons learned are summarized below.

(1) Nominal case results indicate three phases of the LOFW transient with zero AFW, in which the behavior of the primary system is significantly different. The first phase (quasi-equilibrium) lasts until the SG inventory boils dry (approximately 70 min.). Natural circulation permits subcooling of the cold leg to the saturation conditions in the SG. Restoration of AFW delivery to at least one SG before the SGs dry out will result in continuation of the first phase and ultimately either normal progression to cold shutdown or return to power.

The second phase (subcooled expansion) begins when heat removal capacity is lost as the SGs boil dry. To accommodate primary water expansion in the vessel, system pressures increase within minutes to the setpoint of the power-operated relief-valves (PORVs), relieving the primary system in subcooled, isobaric expansion at approximately 80 cfm (0.04 m^3/s). The PORVs relieve steam at this rate for approximately 20 min., then water relief continues at the same rate for approximately another 20 min. Natural circulation continues during this phase.

System saturation at the start of the third phase (saturated expansion) drives natural circulation to zero after approximately 10 min. System pressures rise to the pressurizer safety valve setpoint to accommodate approximately 600 cfm (0.3 m^3/s) two-phase flow out the pressurizer valves. Vapor forms in the primary system and clad temperatures begin to rise sharply after approximately 45 min of boiling in the primary system.

(2) A case involving recovery of AFW availability minutes before natural circulation was completely lost in the third phase was selected because operators would likely attempt to restore AFW as soon as possible, and recovery at that time is a bounding case for AFW recovery at any earlier time. Results showed that (a) AFW initiation caused rapid primary depressurization and subsequent reflooding of the primary system by pressurizer water and (b) primary subcooling and natural circulation were re-established after 20-30 min. Operator initiation of increased charging flow in response to low pressurizer level (called for in the EOPs) would result in return to hot shutdown conditions.

(3) A case involving opening of one ARV to its full open position and subsequent failure to close was added as a complication in the LOFW scenarios because reactor operating experience shows that ARVs have malfunctioned in the past and because primary conditions during the first phase of a LOFW transient depend on secondary saturation conditions. Results showed that primary temperatures and pressures fall during the 10 min. required to blowdown the affected SG, but subcooled conditions are maintained even in the absence of pressurizer heaters. The impact on the duration of the first phase is small.

(4) A case involving operator initiation of full charging flow after primary system pressure reached the safety setpoint was selected as a bounding case for scenarios with feed and bleed initiation up to that time. When charging flow was initiated, the vapor-fraction in top quarter of the core was approximately 10 percent. After reaching a maximum of 30 percent about 10 min. later, a slow refill of the core began and completed approximately 2 h later. Results also showed that the primary system eventually returned to subcooled expansion. The circumstances of this are not yet fully understood and remain under study.

(5) A case involving delay in reactor scram was selected to determine the impact of opening the PORVs early in the transient as occurred during the TMI accident. Results indicated that following RCP trip without immediate reactor scram (1) PORVs open after approximately 4 s, safeties after 8 s, and all valves begin to close within seconds of scram initiation and (2) steam relief is accommodated through pressurizer valves up to about 20 s of scram delay, then water relief begins. Scram delay remains under study to include the effects of fuel and water temperature reactivity feedback. (For scram delays requiring pressurizer valve water relief, small breaks in the primary system will probably be forced to accommodate system saturated expansion.)

Work is continuing in the LASL SASA effort to fully document these results and to investigate other LOFW scenarios. Our continuing activities will focus on preparedness for multiple equipment failure events.

REFERENCES

1. R. D. Burns III, "Report of the Technical Assessment Task Foce on WASH 1400 - Reactor Safety Study," Staff Report to the President's Commission on the Accident at Three Mile Island, Washington, DC (October 1979).

2. R. D. Burns III, "A Preliminary Review of Beyond-DBA PWR Accident Sequences," Proc. ANS Thermal Reactor Safety Meeting, Knoxville, TN (April 1980).

3. R. D. Burns III and N. S. DeMuth, "Computer Simulations of Postulated Nuclear Reactor Accidents," Proc. CUBE Conf., Livermore, CA (October 1980).

MULTIFAULT ACCIDENT ANALYSIS SECTION
REACTOR SAFETY ANALYSIS GROUP, Q-7
ENERGY DIVISION
LOS ALAMOS SCIENTIFIC LABORATORY

LOSS OF FEEDWATER TRANSIENTS IN PWRs

EIGHTH WATER REACTOR SAFETY
RESEARCH INFORMATION MEETING
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C.
OCTOBER 28, 1980

GENERIC CATEGORIES OF MFAs DEVELOPED EARLY IN LASL SASA EFFORT

ALL AVAILABLE INFORMATION STUDIED
TO IDENTIFY GENERAL, INCLUSIVE CATEGORIES

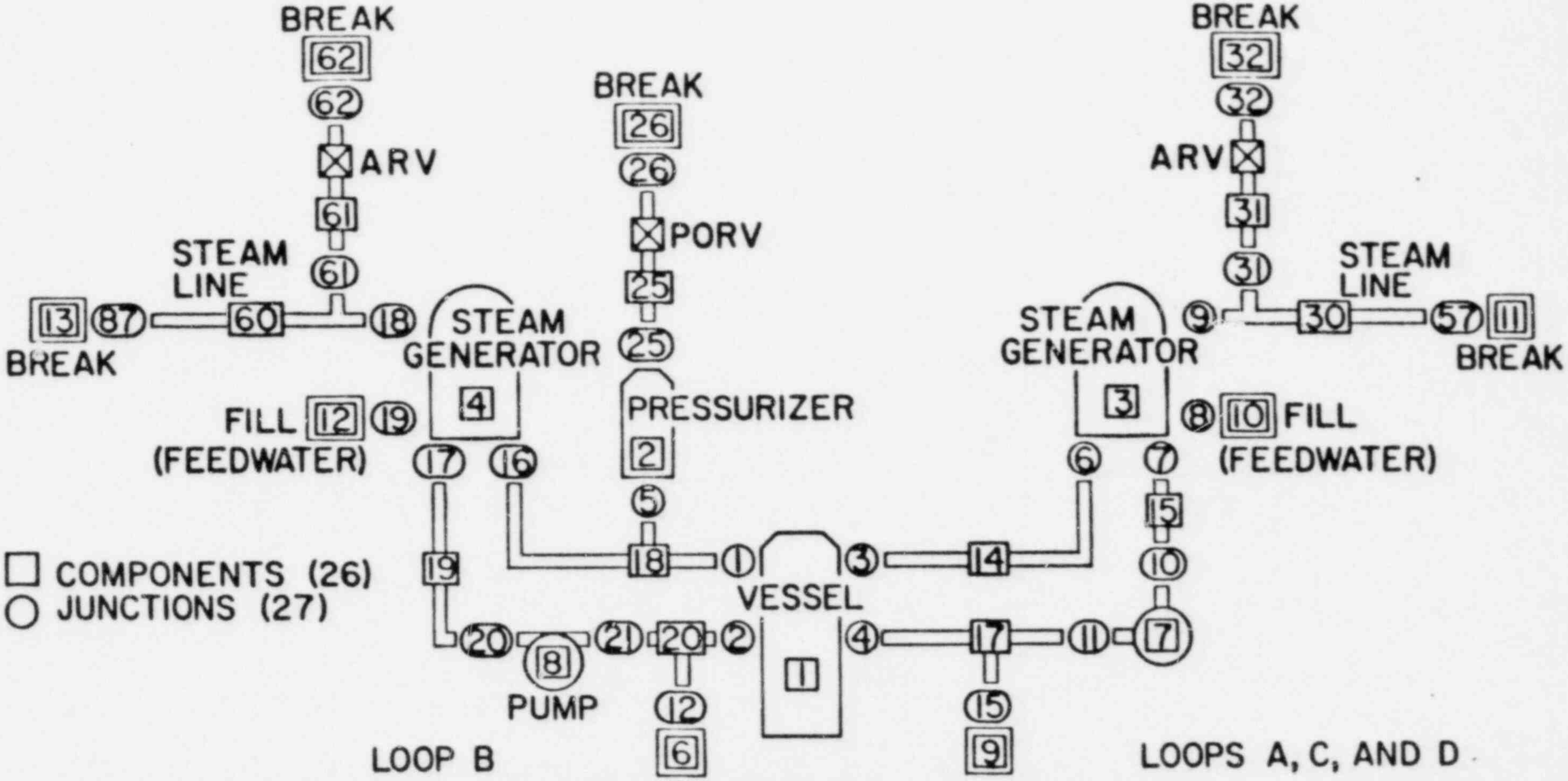
SOURCES:

- WASH-1400
- REACTOR EXPERIENCE(LERs,...)
- FSARs
- ENGINEERING EVALUATIONS OF PLANTS
- VITAL AREA STUDY(FAULT TREES)

LOS ALAMOS SCIENTIFIC LABORATORY

CATEGORIES INCLUDE MULTIPLE EQUIPMENT FAILURE SCENARIOS

1. LOSS OF FEEDWATER (LOFW).
2. STATION BLACKOUT (SB).
3. LOSS OF RESIDUAL HEAT REMOVAL (LORHR).
4. SMALL-BREAK LOCA (S-LOCA).
5. INTERFACING-SYSTEM LOCA (IS-LOCA).
6. PRESSURIZER-VALVE LOCA (PV-LOCA).



ZION-1 PWR
TRAC NODING DIAGRAM

TRAC APPLIED TO SIMULATION OF ZION LOSS OF FEEDWATER SCENARIOS

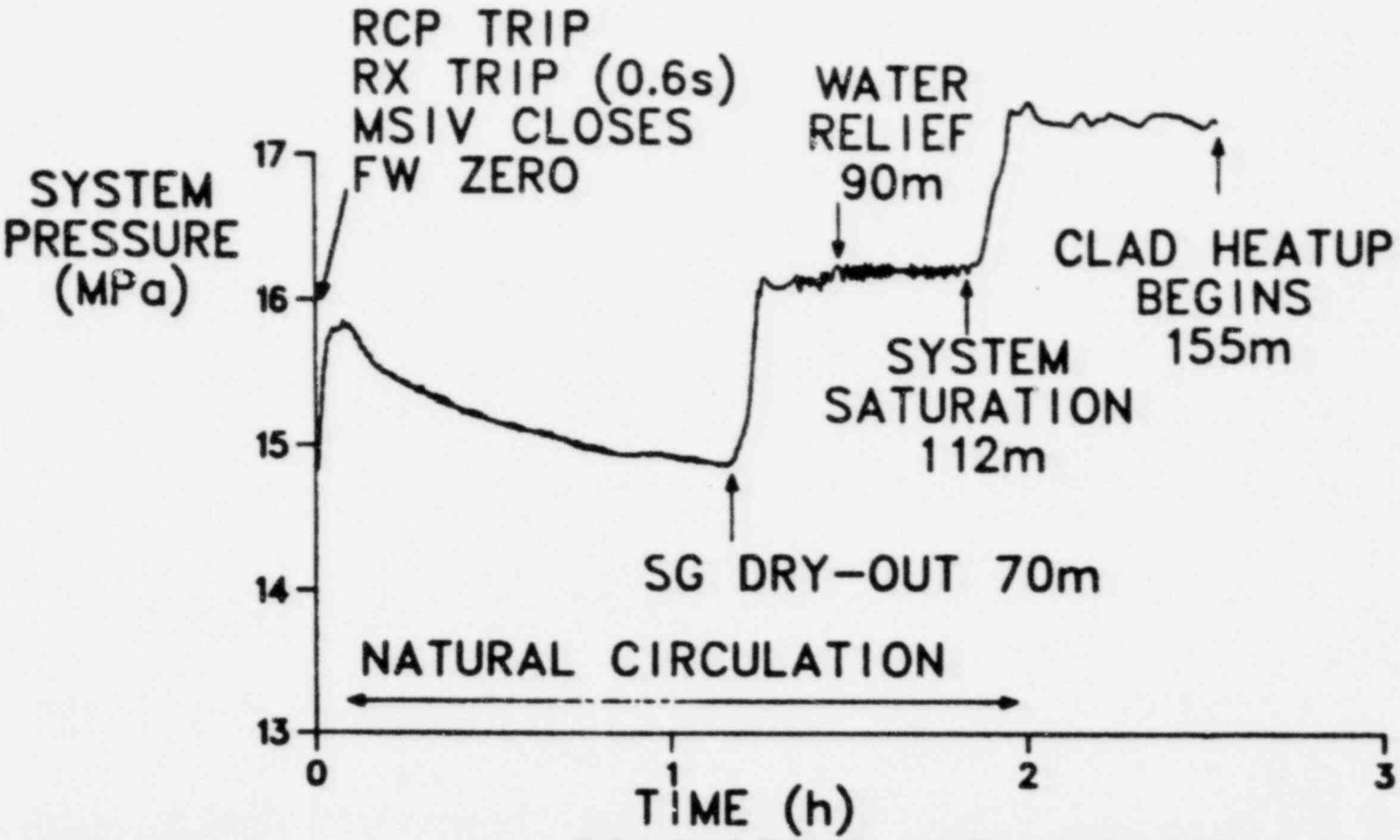
LOFW CASES PRESENTED HERE:

1. NOMINAL SCENARIO (0–3h)
2. DELAYED/DEGRADED AFW (0–10h)
3. STUCK-OPEN ARV (0–1h)
4. FEED/BLEED (0–4h)
5. ATWS (0–1h)

...USE ANALYSES TO FIND AND RESOLVE SPECIFIC SAFETY-RELATED CONCERNS.

LOS ALAMOS SCIENTIFIC LABORATORY

NOMINAL SEQUENCE IS TOTAL LOFW
INITIATED BY LOSS OF OFFSITE POWER



...CONSISTENT WITH SEMISCALE

LOS ALAMOS SCIENTIFIC LABORATORY

ANALYSIS OF STEAM GENERATOR DRY-OUT TIME VERIFIES TRAC RESULT

SG INVENTORY:

160,000 kg * 1.8 MJ/kg = 90 FPS

CORE ENERGY, 0-4200s:

APPROX. 80 FPS (MAINLY DECAY POWER)

FUEL ENERGY, 1350K-550K:

800K * 500 J/kgK * 200,000 kg = 8 FPS

LOS ALAMOS SCIENTIFIC LABORATORY

ANNALYSIS OF PORV RELIEF BASED ON CALCULATED SYSTEM EXPANSION RATES

SUBCOOLED EXPANSION (80–112m)

$$(.003\ m^3/m^3K * 50\ MW)/(.005\ MJ/kgK * 700\ kg/\ m^3)$$

$$= 0.04\ m^3/s \quad (5\ kg/s\ \text{STEAM OR}\ 25\ kg/s\ \text{WATER})$$

SATURATED EXPANSION (112m+)

$h_v - h_l = 0.9\ MJ/kg$ (AT 16.1 MPa)

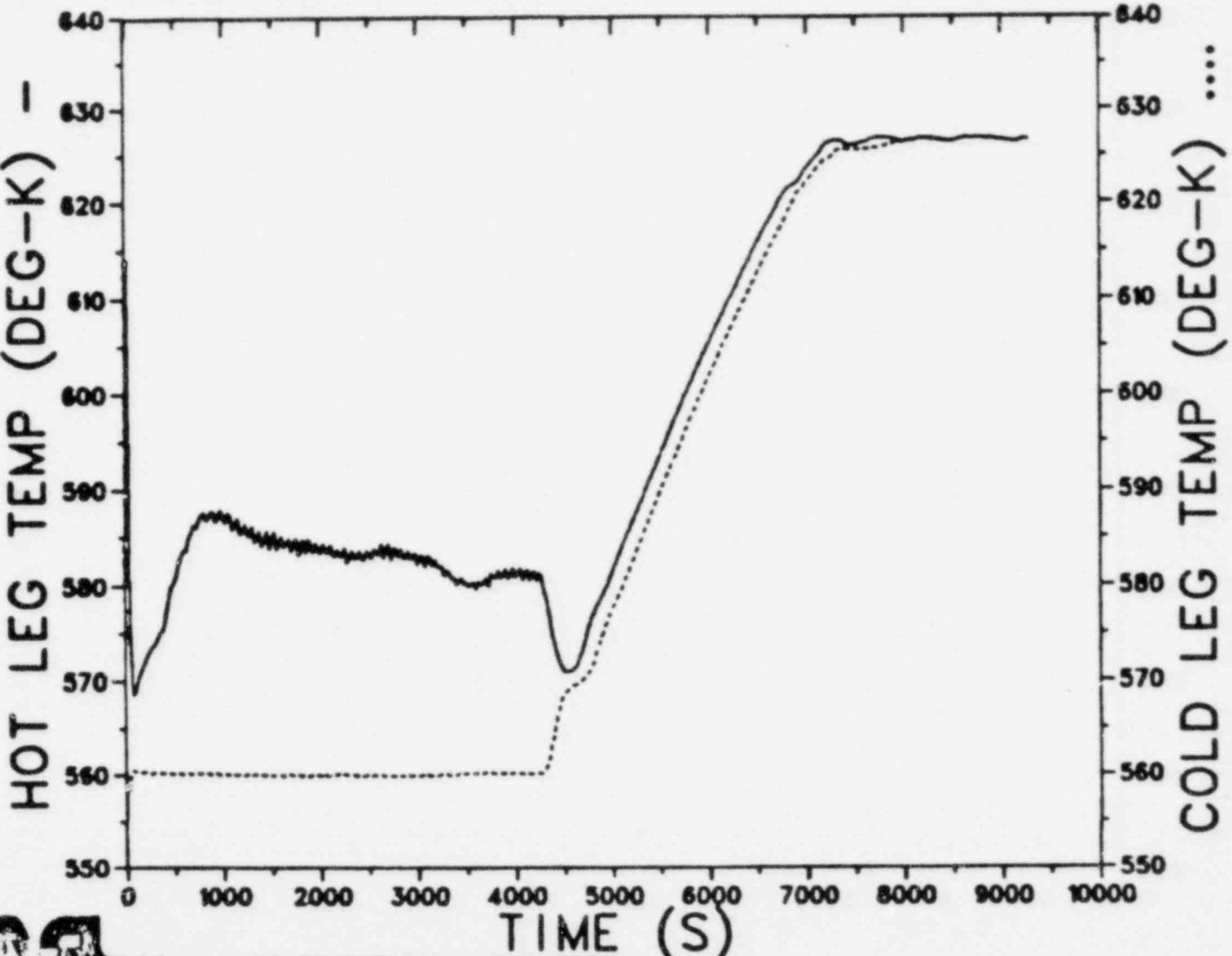
$$v_v - v_l = 0.0075\ m^3/kg$$

$$(0.0075 * 40\ MW)/0.9 = 0.33\ m^3/s$$

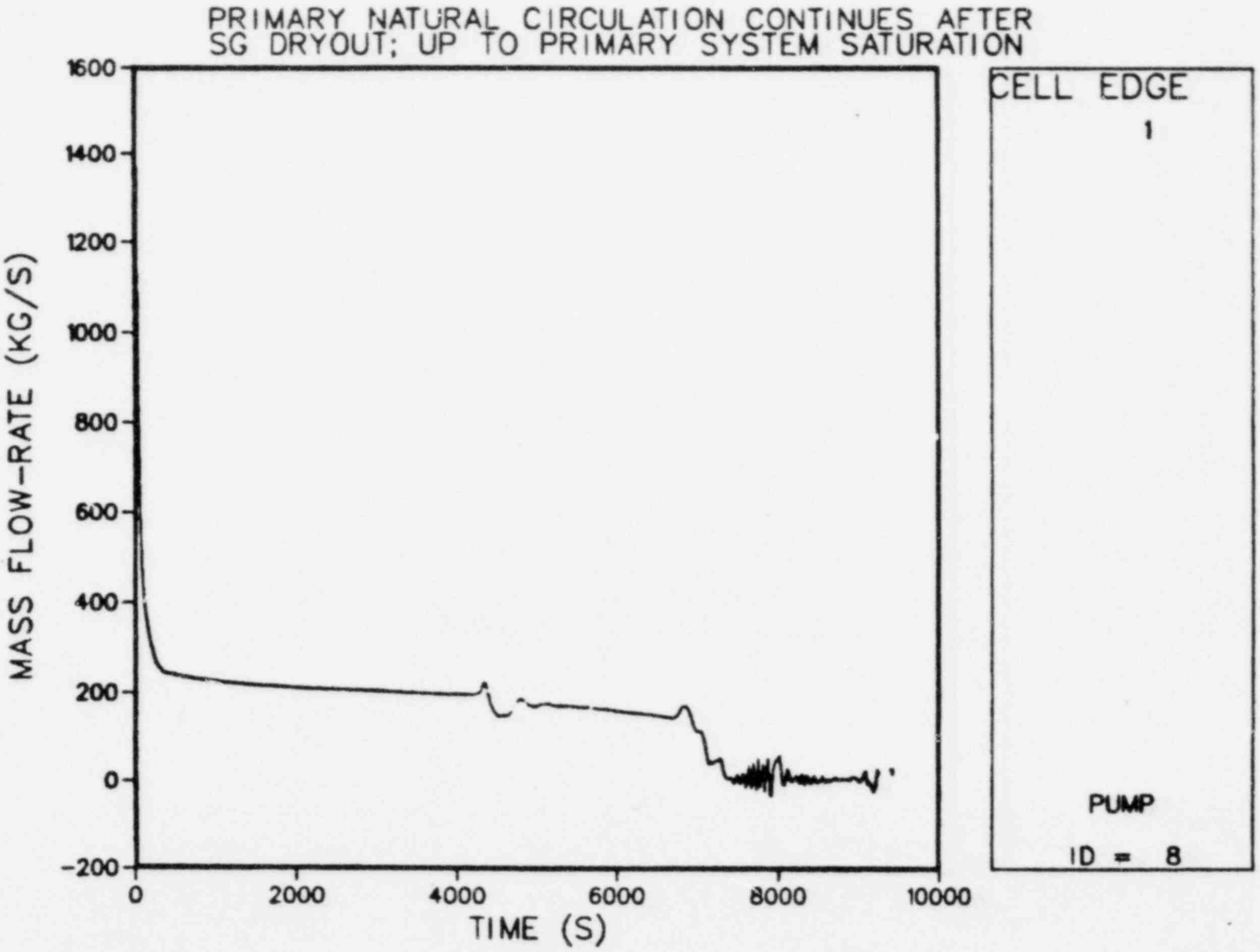
(40 kg/s STEAM OR 200 kg/s WATER)

LOS ALAMOS SCIENTIFIC LABORATORY

COLD LEG TEMPERATURE EQUALS
SECONDARY SATURATION CONDITIONS



LOS ALAMOS SCIENTIFIC LABORATORY


PRIMARY NATURAL CIRCULATION CONTINUES AFTER
SG DRYOUT; UP TO PRIMARY SYSTEM SATURATION
MASS FLOW-RATE (KG/S)
1600
1400
1200
1000
800
600
400
200
0
-200
0
2000
4000
6000
8000
10000
TIME (S)
CELL EDGE
1
PUMP
ID = 8

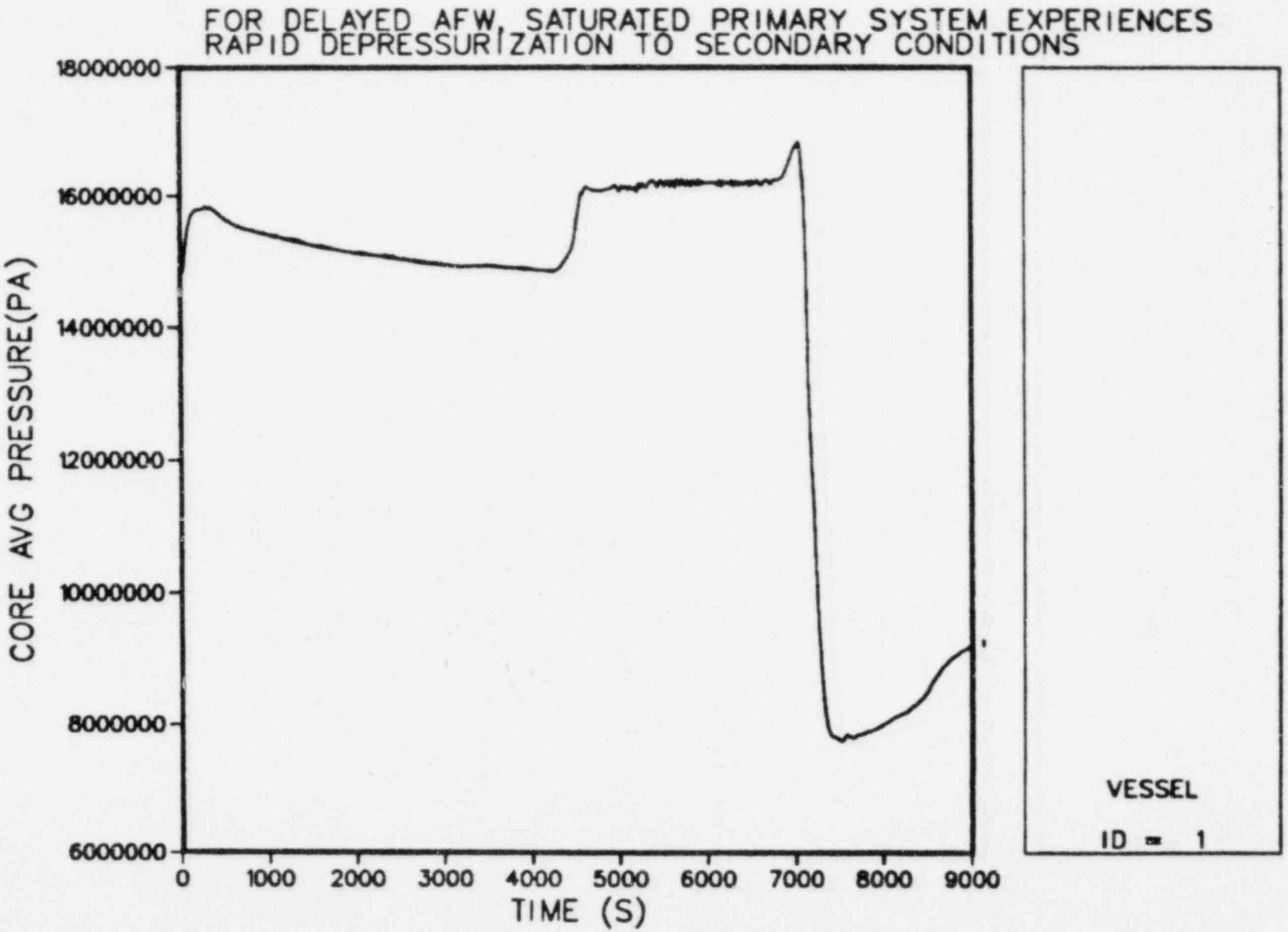
TRAC APPLIED TO SIMULATION OF ZION LOSS OF FEEDWATER SCENARIOS

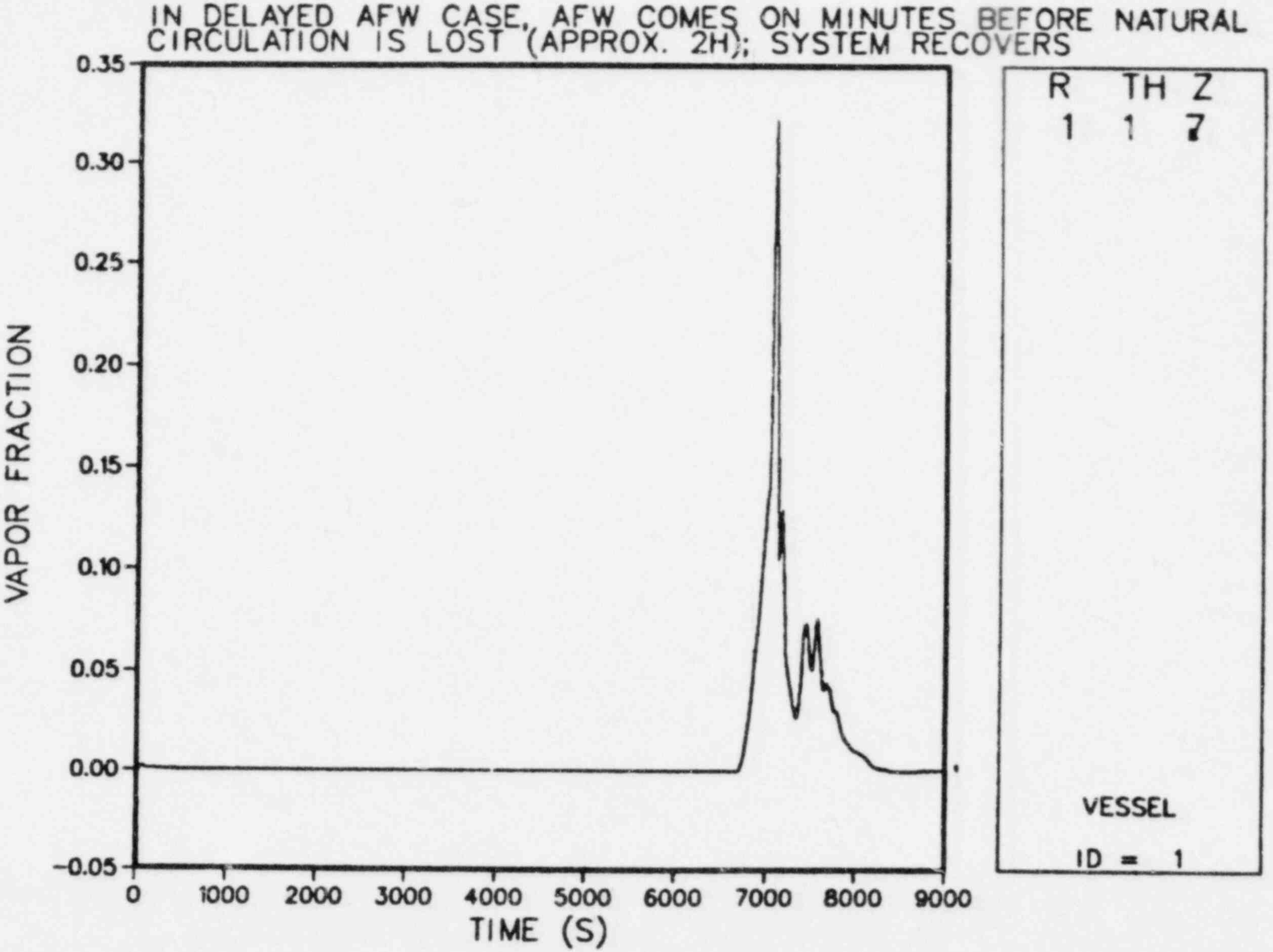
LOFW CASES PRESENTED HERE:

1. NOMINAL SCENARIO (0–3h)
2. DELAYED/DEGRADED AFW (0–10h)
3. STUCK–OPEN ARV (0– 1h)
4. FEED/BLEED (0–4h)
5. ATWS (0– 1h)

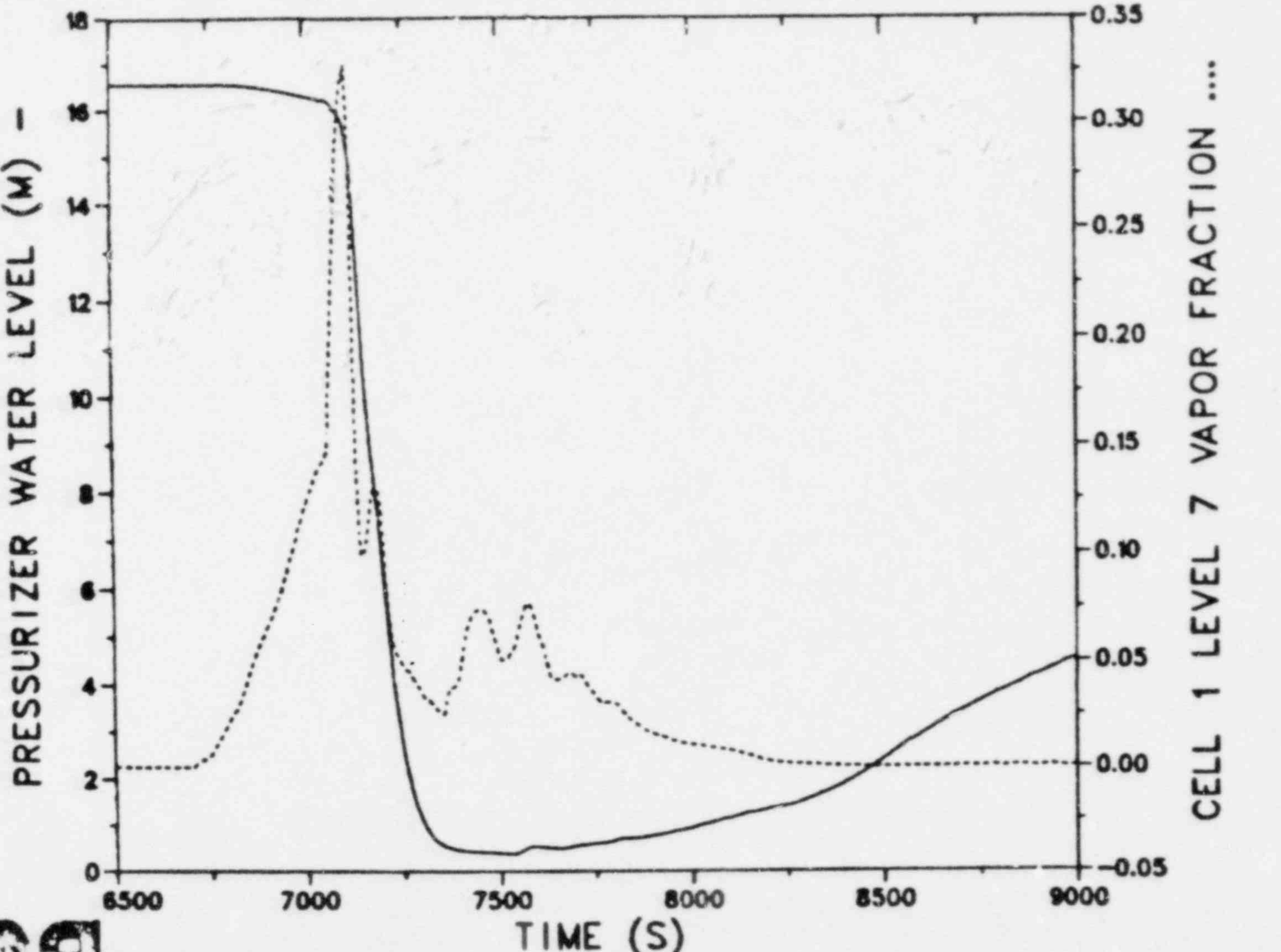
...USE ANALYSES TO FIND AND RESOLVE SPECIFIC SAFETY–RELATED CONCERNS.

LOS ALAMOS SCIENTIFIC LABORATORY

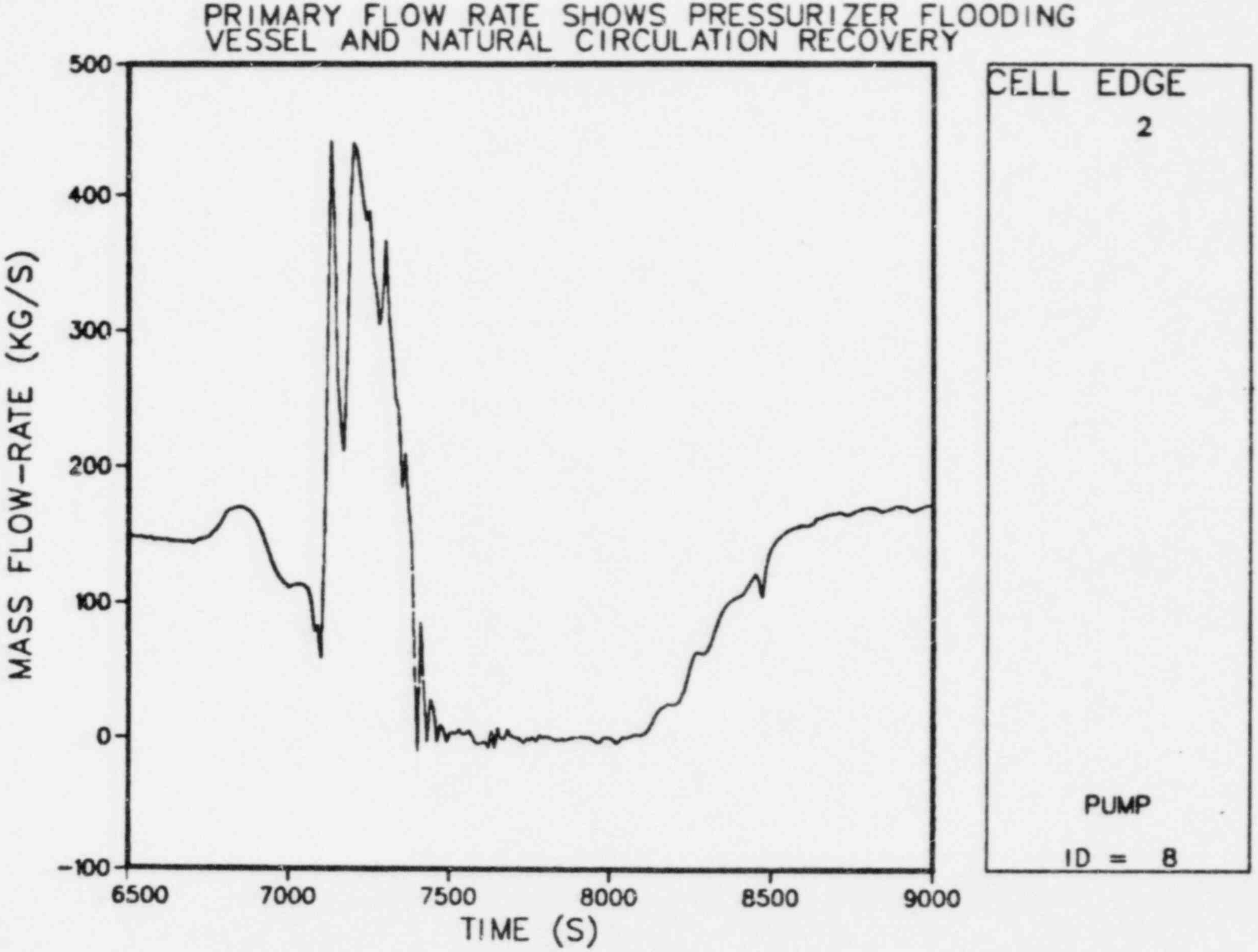

FOR DELAYED AFW, SATURATED PRIMARY SYSTEM EXPERIENCES
RAPID DEPRESSURIZATION TO SECONDARY CONDITIONS
CORE AVG PRESSURE(PA)
18000000
16000000
14000000
12000000
10000000
8000000
6000000
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
TIME (S)
VESSEL
ID = 1


IN DELAYED AFW CASE, AFW COMES ON MINUTES BEFORE NATURAL CIRCULATION IS LOST (APPROX. 2H); SYSTEM RECOVERS
VAPOR FRACTION
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
-0.05
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
TIME (S)
R TH Z
1 1 7
VESSEL
ID = 1

RAPID SYSTEM DEPRESSURIZATION CAUSES PRESSURIZER WATER TO FALL INTO PRIMARY SYSTEM AND REFILL UPPER PLENUM



LOS ALAMOS SCIENTIFIC LABORATORY


PRIMARY FLOW RATE SHOWS PRESSURIZER FLOODING
VESSEL AND NATURAL CIRCULATION RECOVERY
MASS FLOW-RATE (KG/S)
TIME (S)
500
400
300
200
100
0
-100
6500
7000
7500
8000
8500
9000
CELL EDGE
2
PUMP
ID = 8

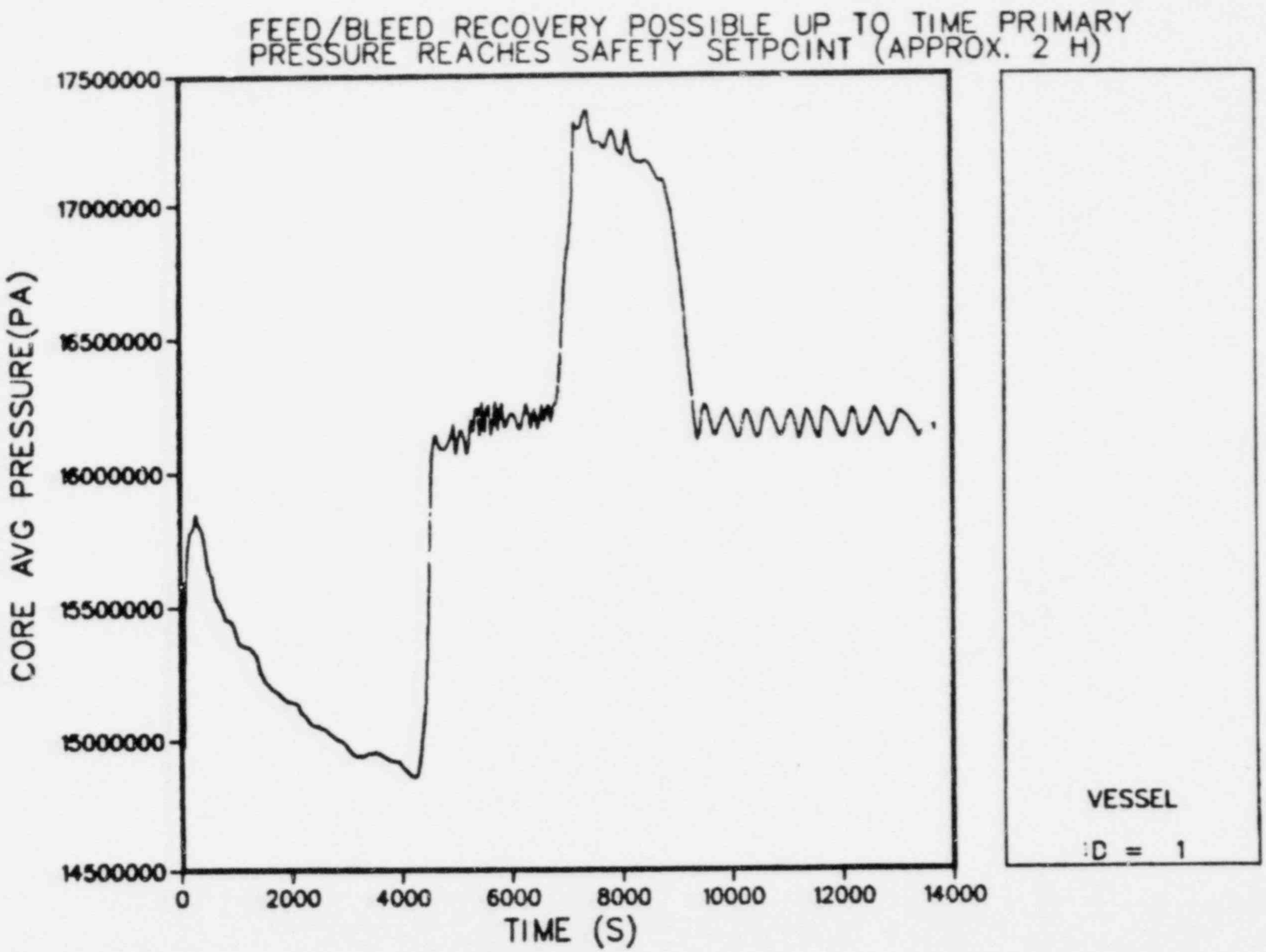
TRAC APPLIED TO SIMULATION OF ZION LOSS OF FEEDWATER SCENARIOS

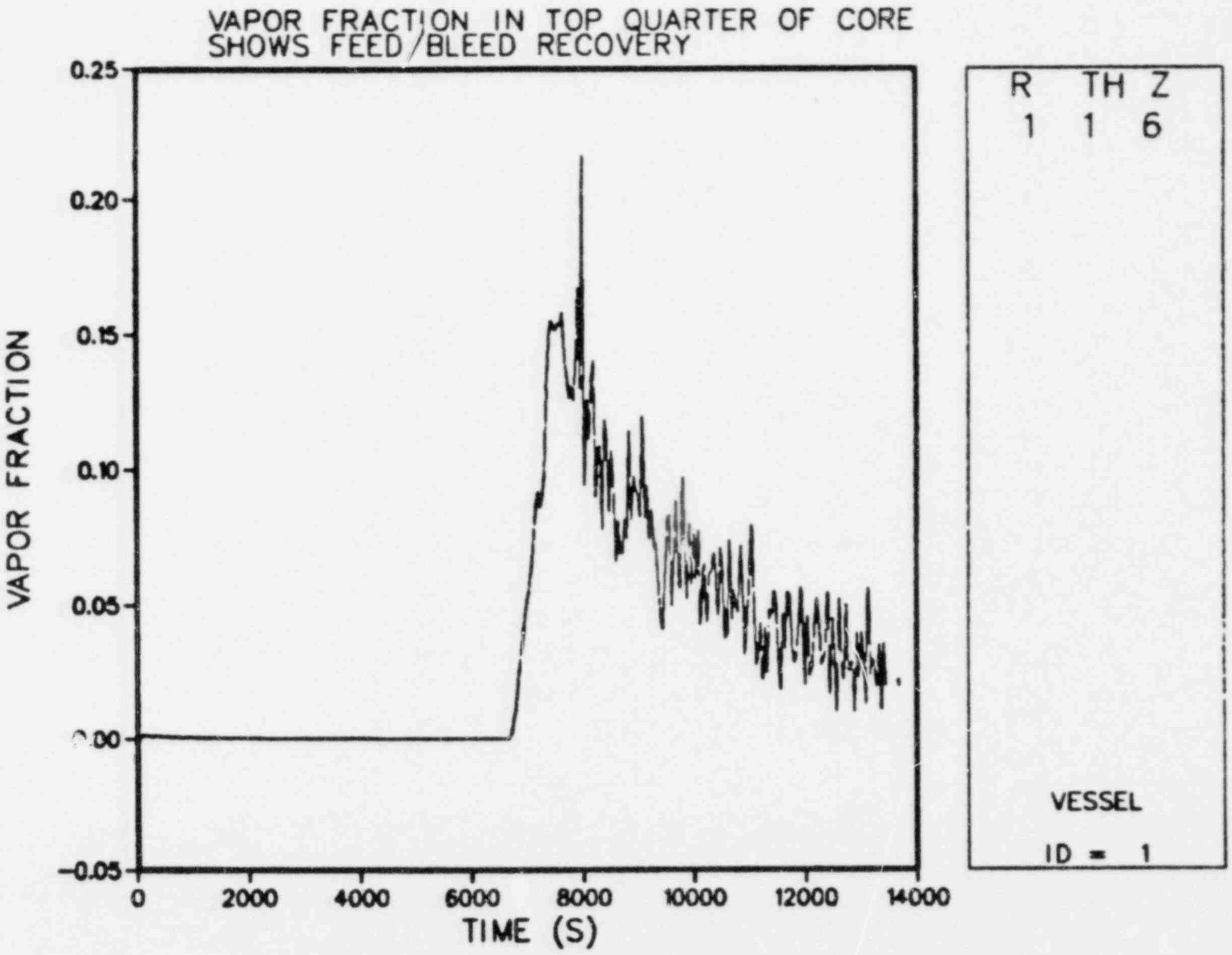
LOFW CASES PRESENTED HERE:

1. NOMINAL SCENARIO (0–3h)
2. DELAYED/DEGRADED AFW (0–10h)
3. STUCK-OPEN ARV (0–1h)
4. FEED/BLEED (0–4h)
5. ATWS (0–1h)

...USE ANALYSES TO FIND AND RESOLVE SPECIFIC SAFETY-RELATED CONCERNS.

LASL LOS ALAMOS SCIENTIFIC LABORATORY


FEED/BLEED RECOVERY POSSIBLE UP TO TIME PRIMARY
PRESSURE REACHES SAFETY SETPOINT (APPROX. 2 H)
CORE AVG PRESSURE(PA)
17500000
17000000
16500000
16000000
15500000
15000000
14500000
0
2000
4000
6000
8000
10000
12000
14000
TIME (S)
VESSEL
:D = 1


VAPOR FRACTION IN TOP QUARTER OF CORE
SHOWS FEED/BLEED RECOVERY
VAPOR FRACTION
TIME (S)
0.25
0.20
0.15
0.10
0.05
0.00
-0.05
0
2000
4000
6000
8000
10000
12000
14000
R TH Z
1 1 6
VESSEL
ID = 1

FURTHER LOFW/ATWS CONSIDERATIONS FOUND IN ANALYSES

1. AFTER 80s AT FULL POWER, SGs DRY-OUT.
2. PORV/SAFETY WATER RELIEF STARTS AFTER ABOUT 20s AT FULL POWER.
3. AT ZION, STEAM RELIEF CAPACITY ADEQUATE IN ATWS – BUT WATER RELIEF MAY NOT BE.
4. REACTIVITY FEEDBACK MUST BE CONSIDERED.

...REQUIRES FURTHER STUDY.

LOS ALAMOS SCIENTIFIC LABORATORY

LESSONS LEARNED IN BASE CASE TRAC SIMULATIONS OF ZION LOFW

1. APPROX. 1 h AVAILABLE FOR INITIATION OF NORMAL AFW RECOVERY.

2. APPROX. 2.5 h NEEDED BEFORE CLAD HEAT-UP BEGINS.

3. INITIATION OF AFW BEFORE NATURAL CIRCULATION LOSS LEADS TO RECOVERY.

4. 30 PERCENT OPERATION OF TURBINE AFW PUMP ADEQUATE FOR RECOVERY.

5. STEAM SIDE DEPRESSURIZATION MAY HAVE SIGNIFICANT REACTIVITY EFFECT.

6. FEED/BLEED RECOVERY POSSIBLE UP TO APPROX. 2 h INTO ACCIDENT.

7. SCRAM DELAY CAN CAUSE EARLY PORV/SAFETY OPENING, EARLIER SG DRY-OUT.

LOS ALAMOS SCIENTIFIC LABORATORY

Severe Core Damage Accident Progression:
Best Estimates and Uncertainties

W. B. Murfin
J. B. Rivard
M. L. Corradini
Sandia National Laboratories
Albuquerque, New Mexico 87185

The progression of severe core damage accidents in two specific PWR plants was recently jointly studied by Sandia National Laboratories and Los Alamos National Scientific Laboratories.[1,2,3] Particular attention was given to threats to containment which might cause release of radioactive material to the atmosphere in core meltdown accidents. Many phenomena are uncertain. Three of particular importance are the coolability of core debris, ultimate containment pressure capability, and steam explosion yield.

If core debris is completely and continuously coolable, the accident progression can be terminated and some threats to containment are completely eliminated. The ultimate strength of the containment is perceived to be more precisely calculable than most other quantities; however, the uncertainty in strength ($\sim \pm 20\%$) is such that at the upper uncertainty limit the containments would survive in almost all accident sequences, and at the lower limit failure would be expected in many accident sequences. It has previously been estimated[4] that steam explosion-generated missiles could penetrate containment with concomitant large releases of radioactive material. If the damage potential of steam explosions is much lower than expected, the maximum consequences of specific meltdown accidents would be lower even though overall risk might not be significantly reduced.

The conclusions drawn in this paper are specific to the two plants studied and should not be extrapolated to other plants.

Core Debris Coolability

Damage severity increases with time in core regions uncovered by boiloff of liquid. Initially, core damage may consist of ballooning, oxidation, and rupture of cladding, collapse of fuel and limited formation of the UO_2-Zr eutectic.[5] This "early" debris tends to have a fairly coarse geometry after quenching (with the possible exception of spalled ZrO_2 chips) which tends to form a rubble bed by downward collapse of part of the core into the undamaged zone. Although the resulting geometry is less permeable than the undamaged core, reintroduction of water may lead to recovery (as at TMI-2). Current models,[6,7,8,9] based on available debris bed experimental data,[8,9,10,11] provide a limited basis for assessing the coolability of early debris.

Later, as the extent and severity of damage increases, molten U-Zr-O liquid[12] flows downward, alternately freezing and melting, tending towards the formation of a coherent pool[13] near the lower edge of the severely damaged region. Introduction of water can result in a steam explosion (see below), nonexplosive quenching and fragmentation,[14] or boiling on the surface of a crust surrounding the melt.[15] Stable cooling of the several configurations which can result is possible in some cases, but assessment of coolability is currently difficult, primarily because of data deficiencies on fragmentation[7] and very deep (~1m) bed behavior.

If a stably-cooled configuration is not attained, the core melt will ultimately penetrate the reactor vessel. Events which may follow include steam explosion with dispersal of debris particulate within and outside the reactor cavity, nonexplosive dispersal of debris particulate within and outside the reactor cavity by rapid steam generation,[16] formation of a static debris bed,[7] and core melt concrete interactions and basemat erosion. Despite the fact that the effects of these events may be crucial in determining containment integrity, estimates of the most probable events are only possible in a few specialized cases.

Ultimate Containment Pressure Capability

Containment construction and materials are so complex that exact structural models are impossible. Approximations are required which may compromise the accuracy of ultimate load calculations.

Ultimate static internal pressure capabilities were computed by several organizations[2,3,17] using different models and different criteria for catastrophic failure. The internal failure pressure for one containment ranged from 110 psig to 126 psig, and from 134 psig to 154 psig for the other. At the highest failure pressure, the containment would have survived almost all accident sequences; at the lowest, failure would be expected for many important sequences. Among the modeling problems and uncertainties are the assumption of axial symmetry, neglect of the effect of penetrations, neglect of internal structures and weight loads, approximations in rebar and tendon modeling, approximations for liner-to-concrete ties, approximations for the ultimate behavior of concrete, and uncertain material properties.

Steam Explosion Yields

Based upon small and large scale experiments,[14] the conversion of thermal energy to work lies in the range of 0 to 3%. When a prototypic fuel material, corium-A[18] (metallic and oxidic) was used, energetic explosions were not observed; the reason for this difference is believed to be due to partial melt solidification.

Analysis of the experiments indicates that the ratio of fuel to water coolant masses in the mixture region of the explosion was less than or equal to one; the possibility exists that the low yield may have been caused by the experimental conditions.

For the reactors studied,[2] a unique value of the explosion energy could not be specified. The actual explosive yield depends on the mass of molten fuel and coolant that can mix and the initial conditions. Fuel melting and fuel/coolant mixing have only been investigated at comparatively small scales (~1-20 kg). These experiments suggest that the molten fuel can mix rapidly with the coolant, although the scaling of the process is unknown.

A current best estimate is that steam explosion yields are lower than previously assumed, although large uncertainties exist. For a typical 3000 MW (th) PWR, steam explosion yields are expected to be in the range of 300-1500 MJ, with the best estimate tending toward the lower value. Steam explosions in this range would be unlikely to cause penetration of containment for the specific plants studied.

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SEVERE CORE DAMAGE ACCIDENT PROGRESSION:

BEST ESTIMATES AND UNCERTAINTIES

W. B. MURFIN

J. B. RIVARD

M. L. CORRADINI

WATER REACTOR SAFETY INFORMATION MEETING

OCTOBER 27-31, 1980

APPLICATION TO LICENSING/SAFETY ISSUES

- LIKELY COURSE OF MELTDOWN ACCIDENTS CAN ONLY BE CRUDELY DELINEATED

- WASH-1400 MAY BE CONSERVATIVE VIS-A-VIS
  - STEAM EXPLOSIONS
  - CONTAINMENT FAILURE LEVELS

- DEBRIS COOLABILITY NEEDS TO BE CAREFULLY ADDRESSED. NEEDED INFORMATION INCLUDES
  - DEBRIS PARTICLE SIZES
  - DISPERSAL
  - BEHAVIOR OF DEEP BEDS
  - BEHAVIOR OF STRATIFIED BEDS

LICENSING/SAFETY ISSUES ADDRESSED
(CLASS 9 ACCIDENT INVESTIGATIONS)

- CAN THE MOST PROBABLE PROGRESSION OF A MELTDOWN ACCIDENT BE DETERMINED?

- IS WASH-1400 CHARACTERIZATION TOO CONSERVATIVE?

- WHAT NEEDS TO BE KNOWN TO ADEQUATELY CHARACTERIZE THE PROGRESSION OF MELTDOWN ACCIDENTS?

RELATED REPORTS

"SUMMARY OF THE ZION/INDIAN POINT STUDY" - NUREG/CR-1409, SAND80-0617

"REPORT OF THE ZION/INDIAN POINT STUDY, VOL. I" - NUREG/CR-1410, SAND80-0617/1

"REPORT OF THE ZION/INDIAN POINT STUDY, VOL. II" - NUREG/CR-1411, LA-8306-MS

"REVIEW OF IN-VESSEL MELTDOWN MODELS" - NUREG/CR-1493, SAND80-0455

"ASSESSMENT OF CORE PENETRATION OF A PWR REACTOR VESSEL AND PARTICULATE DEBRIS COOLABILITY IN TMLB', S2D, AND ABG ACCIDENTS" - NUREG/CR-1518, SAND80-0701

ZION/INDIAN POINT STUDY

- JOINT SNL/LANSL STUDY

- CORE MELTDOWN ACCIDENTS IN WHICH CONTAINMENT INTEGRITY IS BREACHED ABOVEGROUND, WITH PARTICULAR ATTENTION TO POSSIBILITY OF MITIGATION

- UNDERSTANDING OF ACCIDENT PROGRESSION IS DOMINATED BY PHENOMENOLOGICAL UNCERTAINTIES

- THREE EXAMPLES CHOSEN FOR DISCUSSION:

  - CORE DEBRIS COOLABILITY
  - STEAM EXPLOSION YIELD
  - CONTAINMENT FAILURE PRESSURE

IMPORTANCE OF EXAMPLES

- CORE DEBRIS COOLABILITY
  - IF CORE DEBRIS IS CONTINUOUSLY AND PERMANENTLY COOLABLE, ACCIDENT PROGRESSION CAN BE TERMINATED.
  - SOME THREATS TO CONTAINMENT CAN BE AVERTED.
  - COOLABILITY IS AN IMPORTANT QUESTION IN- AND EX-VESSEL.
- CONTAINMENT PRESSURE CAPABILITY
  - CONTAINMENT ULTIMATE PRESSURE CAPABILITY IS PROBABLY CALCULABLE TO CLOSER LIMITS OF ACCURACY THAN MOST OTHER PHENOMENA ($\pm$ 10-20%).
  - HOWEVER, AT UPPER LIMIT OF CALCULATED STRENGTH, CONTAINMENT WILL SURVIVE IN MOST ACCIDENT SEQUENCES, AT LEAST WITH ADDED MITIGATING SYSTEMS. AT LOWER LIMIT, CONTAINMENT FAILS FOR MANY IMPORTANT SEQUENCES, EVEN WITH MITIGATING SYSTEMS.
- STEAM EXPLOSION YIELD
  - PENETRATION OF CONTAINMENT BY STEAM EXPLOSION GENERATED MISSILES WOULD BE A MAJOR CONTRIBUTOR TO THE MAXIMUM CONSEQUENCES (ALTHOUGH NOT TO OVERALL RISK).
  - PRESENT DATA BASE MUCH BETTER THAN AT TIME OF WASH-1400.

DEBRIS COOLABILITY IN WATER

DEPENDS ON

- PARTICLE SIZE
- PACKING
- BED THICKNESS
- DECAY POWER
  - F. P. LOSS
  - GAMMA RAY LOSS
  - DILUTION BY NON-FUEL SPECIES
- FLOW REGIME
  - PUMPED FLOW
  - NAT. CONVECTION W/
    A. THROUGH-FLOW
    B. "U" FLOW
    C. GAS-ADDED FLOW
  - SUBCOOLED OR SATURATED

CORE DEBRIS COOLABILITY

DEBRIS

- FRACTURED/SPALLED Zr/ZrO_2
- FRACTURED PELLETS
- FRAGMENTED (REFROZEN) FUEL/STEEL

PHENOMENA

- HEAT TRANSPORT FROM DEBRIS BY 1- & 2-PHASE CONVECTION OF COOLANT WITHIN DEBRIS
- FORCED OR NATURAL CIRCULATION OF COOLANT THROUGH VESSEL TO HEAT SINK

LIMITS

- DRYOUT--CONTINUOUS, LOCAL STEAM BLANKETING OF A DEBRIS ZONE DUE TO LIQUID STARVATION

CONSEQUENCES

- REMELT OF FUEL BY DECAY HEAT, CONTINUATION OF MELTDOWN

LWR DEBRIS TYPES

- EARLY DEBRIS

  FRACTURED BUT UNMELTED FUEL & CLAD

  SETTLING?

- LATE DEBRIS (IN-VESSEL)

  MELTED AND REFROZEN FUEL, CLAD AND STEEL

  PRESSURE INFLUENCE ON STEAM EXPLOSION

  COARSE (?) PARTICULATE IF NO STEAM EXPLOSION,
  LITTLE DISPERSION

  FINE PARTICULATE w/ STEAM EXPLOSION,
  LARGE DISPERSION

  UNREACTED Zr (?)

- EX-VESSEL DEBRIS

  HIGH Fe FRACTION (CORIUM)

  OXIDATION STATE INFLUENCE ON STEAM EXPLOSION

  DISPERSION POTENTIAL w/ STEAM EXPLOSION

  MELT/CONCRETE INTERACTION
  (GAS-ADDED FLOW)

UNCERTAINTIES FOLLOWING
REINTRODUCTION OF WATER INTO
DAMAGED CORE

LATE (MOLTEN) DEBRIS

- TYPE AND EXTENT OF MELT
  - SIZE, ENTHALPY ABOVE MELT
  - DEGREE OF COHERENCY
- LOCATION OF MELT
  - BLOCKAGES (WATER ACCESS)
  - AXIAL PENETRATION
- EFFECTS OF WATER ENTRY
  - STEAM EXPLOSION, DAMAGE
  - MELT QUENCH, FRAGMENTATION
  - DISPERSAL OF DEBRIS
- DEBRIS BED
  - FORMATION (STRATIFIED?)
  - COOLING
  - REMELT

CONTAINMENT ULTIMATE PRESSURE CAPABILITY

- MOST THREATS (INCLUDING H_2 BURN) CAN BE REPRESENTED BY STATIC OR NEARLY-STATIC PRESSURE.

- COMPLEXITY REQUIRES SIMPLIFYING ASSUMPTIONS AND MODELING.

  - AXIAL SYMMETRY
  - PENETRATIONS
  - INTERNAL STRUCTURES/EQUIPMENT
  - REBAR OR TENDON MODELING
  - LINER-TO-CONCRETE TIES
  - ULTIMATE BEHAVIOR OF CONCRETE
  - MATERIAL PROPERTIES
  - FAILURE CRITERIA

- EFFECT OF INDIVIDUAL ASSUMPTIONS IS PROBABLY SMALL BUT NOT NECESSARILY SO IN AGGREGATE.

METHOD OF CALCULATION

- FINITE DIFFERENCE MODEL
  - NON-LINEAR
  - AXISYMMETRIC
  - LUMPED REBAR
- SLOWLY APPLIED LOAD
  - RAMP
  - SMALL STEPS
- FAILURE CRITERIA
  - GENERAL STATE OF YIELD
  - UNBOUNDED DISPLACEMENT
  - LINER YIELD WITH CONCRETE CRACKING
  - TENDON YIELD
- MATERIAL PROPERTIES
  - GUARANTEED MINIMUM
  - ACTUAL TESTS

RESULTS OF CALCULATIONS

- 4 ORGANIZATIONS
- DIFFERENT MODELS, DIFFERENT FAILURE CRITERIA
- TWO LARGE DRY CONTAINMENTS, ONE REINFORCED CONCRETE, ONE POST-TENSIONED
- ULTIMATE FAILURE PRESSURES:

  R.C. 110-126 PSIG

  P.T. 134-154 PSIG

- FAILURE ASSURED AT LOWER LIMIT FOR MANY SEQUENCES, EVEN WITH VENTING
- SURVIVAL ASSURED AT UPPER LIMIT FOR MOST SEQUENCES, ESPECIALLY WITH VENTING

STEAM EXPLOSION YIELDS

- BASED ON -

  SMALL (1-20 KG) EXPERIMENTS

  CODE CALCULATIONS

- THERMAL-TO-MECHANICAL CONVERSION EFFICIENCY < 3 PERCENT

- POSSIBLY MUCH LOWER FOR CORE MATERIALS

- UNIQUE VALUE CANNOT BE SPECIFIED

- UNCERTAINTIES

  - FUEL/COOLANT MIXED
  - INITIAL CONDITIONS
  - CONVERSION EFFICIENCY
  - SCALING

- EFFECTS EVALUATED BY NON-LINEAR FINITE ELEMENT ANALYSIS OF PRESSURE VESSEL

STEAM EXPLOSION - RESULTS

- BEST ESTIMATE OF YIELD (IN-VESSEL)
  - 300-1500 MJ
  - MORE LIKELY AT LOWER END
- BEST ESTIMATE OF EFFECTS (IN-VESSEL)
  - NO LARGE MISSILES
  - SMALL MISSILES WILL NOT PENETRATE MISSILE SHIELD AND CONTAINMENT
  - CONTAINMENT PENETRATION UNLIKELY
- NO THREAT TO CONTAINMENT EX-VESSEL
- RESULTS ARE PLANT SPECIFIC. CANNOT BE EXTRAPOLATED.

CONCLUSIONS

- CORE DEBRIS COOLABILITY CAN NEITHER BE ASSURED NOR RULED OUT. "BEST ESTIMATES" ARE HIGHLY SUBJECTIVE AND SUBJECT TO EXTENSIVE DEBATE.
  - EXPERIMENTS NOW UNDERWAY MAY IMPROVE ESTIMATES.
- CONTAINMENT FAILURE CALCULATIONS ARE PROBABLY MORE ACCURATE THAN CALCULATIONS OF PRESSURE BUILDUP. HOWEVER, BETTER KNOWLEDGE OF PRESSURE WOULD PROBABLY NOT REMOVE ALL UNCERTAINTY.
- STEAM EXPLOSION YIELDS HAVE A WIDE RANGE OF UNCERTAINTY. HOWEVER, FOR THE SPECIFIC PLANTS INVESTIGATED, EVEN THE UPPER BOUND IS UNLIKELY TO CAUSE CONTAINMENT PENETRATION.

THESE RESULTS ARE PLANT-SPECIFIC AND SHOULD NOT BE EXTRAPOLATED TO OTHER PLANTS WITHOUT CAREFUL ANALYSIS.

UNION CARBIDE

ORNL

SEVERE ACCIDENT SEQUENCE ASSESSMENT FOR BWRs

M. H. FONTANA

OAK RIDGE NATIONAL LABORATORY

EIGHTH NRC WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

NATIONAL BUREAU OF STANDARDS

OCTOBER 28, 1980

UNION CARBIDE ORNL

SEVERE ACCIDENT SEQUENCE ASSESSMENT (SASA) PROGRAM

OVERALL GOAL/KEY OBJECTIVES

SIGNIFICANTLY INCREASE THE ACTUAL AND PERCEIVED SAFETY OF LWRs BY:

- IDENTIFYING DOMINATING REACTOR ACCIDENTS THAT COULD INVOLVE SEVERE CORE DAMAGE AND/OR THREAT TO FISSION PRODUCT ISOLATION FROM THE ENVIRONMENT;
- DETERMINING, USING BEST ANALYSES, THE BEHAVIOR OF REACTORS DURING THE COURSE OF THESE ACCIDENTS;
- IDENTIFYING, ASSESSING THE EFFECTS OF, AND RECOMMENDING CORRECTIVE ACTION;
- ESTABLISHING FEASIBILITY AND CRITERIA FOR FUNDAMENTAL IMPROVEMENTS IN PLANT DESIGN AND OPERATION; AND
- MAKING INFORMATION AVAILABLE FOR IMPLEMENTATION BY INTERACTING WITH NRC, UTILITIES, AND VENDORS

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SASA PROGRAM RESULTS SHOULD BE USEFUL TO

- NRC-RSR
- NRC-REG
- UTILITIES
- VENDORS

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SASA PROGRAM RESULTS SHOULD BE USEFUL TO

- NRC–RSR:
  - FOR DEVELOPING THE INFORMATION BASE FOR UNDERSTANDING SEVERE ACCIDENTS
  - FOR GUIDING FUTURE R&D
  - FOR GUIDING ADVANCED DESIGNS
- NRC–REG:
  - FOR RULEMAKING
  - FOR GUIDING NRC EMERGENCY RESPONSE CENTERS, UTILITIES, AND GOVERNMENTS IN EVALUATING OPTIONS FOR MANAGING SEVERE ACCIDENTS

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– AND –

SASA PROGRAM RESULTS SHOULD BE USEFUL TO

- UTILITIES:
  - FOR DEVELOPING EMERGENCY PROCEDURES FOR SEVERE ACCIDENTS
  - FOR SPECIFYING MORE ACCIDENT-RESISTANT PLANTS
  - FOR TRAINING OPERATORS
- VENDORS:
  - FOR GUIDING PLANT BACKFITS
  - FOR GUIDING ADVANCED DESIGNS

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SASA PROGRAM SPECIFIC OBJECTIVES – 1

- IDENTIFY AND ASSESS ACCIDENT INITIATORS WITH RESPECT TO THEIR PROBABILITY OF OCCURRENCE AND THEIR POTENTIAL FOR CAUSING SIGNIFICANT DAMAGE, AND IDENTIFY KEY SEQUENCES FOR IN-DEPTH ANALYSIS

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SASA PROGRAM SPECIFIC OBJECTIVES – 2

- ANALYZE SEQUENCES WITH RESPECT TO
  - PHENOMENA (INCLUDING DRIVING FORCES AND FISSION PRODUCT BEHAVIOR),
  - TIMING OF KEY EVENTS,
  - PLANT DYNAMIC RESPONSE,
  - SYSTEMS INTERACTIONS,
  - EQUIPMENT PERFORMANCE, AND
  - OPERATOR PERFORMANCE

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SASA PROGRAM SPECIFIC OBJECTIVES – 3

- IDENTIFY CORRECTIVE ACTION KEYED TO TIME WINDOWS ESTABLISHED BY SEQUENCE ANALYSIS, IDENTIFY REQUIREMENTS FOR IMPLEMENTATION, AND ASSESS SIDE EFFECTS; SUCH ACTION WOULD INCLUDE

  – EQUIPMENT REPAIR,

  – OPERATOR ACTION,

  – OFFSITE SPECIAL PURPOSE EQUIPMENT,

  – EVACUATION

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SASA PROGRAM SPECIFIC OBJECTIVES – 4

- IDENTIFY SAFE STABLE STATES AND HOW THEY MAY BE ATTAINED

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SASA PROGRAM OBJECTIVE – 5

- IDENTIFY INHERENT FISSION PRODUCT RETENTION PHENOMENA

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SASA PROGRAM SPECIFIC OBJECTIVES – 6

- ESTABLISH FEASIBILITY AND CRITERIA FOR IMPROVEMENTS IN
  - PLANT DESIGN
  - INSTRUMENTATION
  - INFORMATION DISPLAYS/OPERATOR PERFORMANCE
  - EMERGENCY PLANNING

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SASA PROGRAM SPECIFIC OBJECTIVES – 7

- IDENTIFY R&D NEEDS

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SASA PROGRAM SPECIFIC OBJECTIVES – 8

- INTERACT WITH NRC, UTILITIES, AND VENDORS

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AS AN INITIAL ATTEMPT AT SEVERE ACCIDENT SEQUENCE ASSESSMENT, ORNL HAS STARTED TO ANALYZE THE BROWNS FERRY-UNIT 1 BOILING WATER REACTOR

THE FIRST SEQUENCE ASSESSED IS LOSS OF OFFSITE AND ONSITE AC POWER (STATION BLACKOUT)

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USES FOR BROWN'S FERRY SASA STUDY

- EVALUATE CONSEQUENCES FOR BLACKOUT
- IMPROVE OPERATING PROCEDURES, OPERATOR TRAINING, LICENSING
- DETERMINE INSTRUMENTATION AND CONTROL REQUIREMENTS
- DETERMINE DC AND AC RELIABILITY REQUIREMENTS
- DEVELOP IMPROVED PLANS FOR MITIGATIVE ACTIONS – ONSITE EMERGENCY PLANS
- DEVELOP IMPROVED OFFSITE EMERGENCY PLANS

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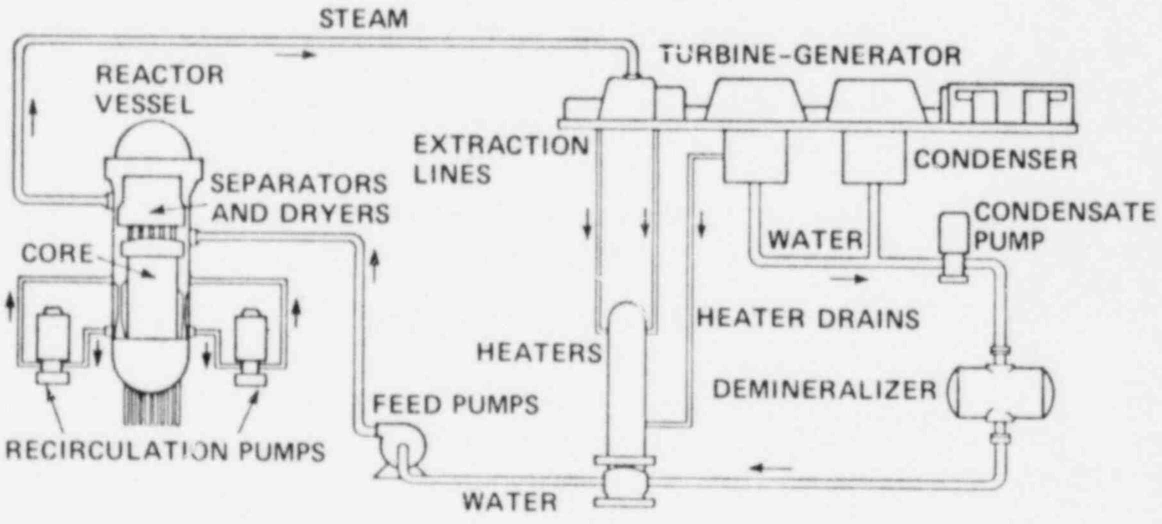
IN ORDER TO ASSIST US IN

- TRACKING ACCIDENT PROGRESSION IN TIME
- IDENTIFYING IMPORTANT EVENTS
- IDENTIFYING IMPORTANT PHENOMENA
- MAINTAINING AWARENESS OF CONCURRENT EVENTS
- IDENTIFYING POTENTIAL CORRECTIVE ACTION

ORNL HAS DEVISED A SEQUENCE PROGRESSION "TIME-LINE" CHART

UNION CARBIDE ORNL

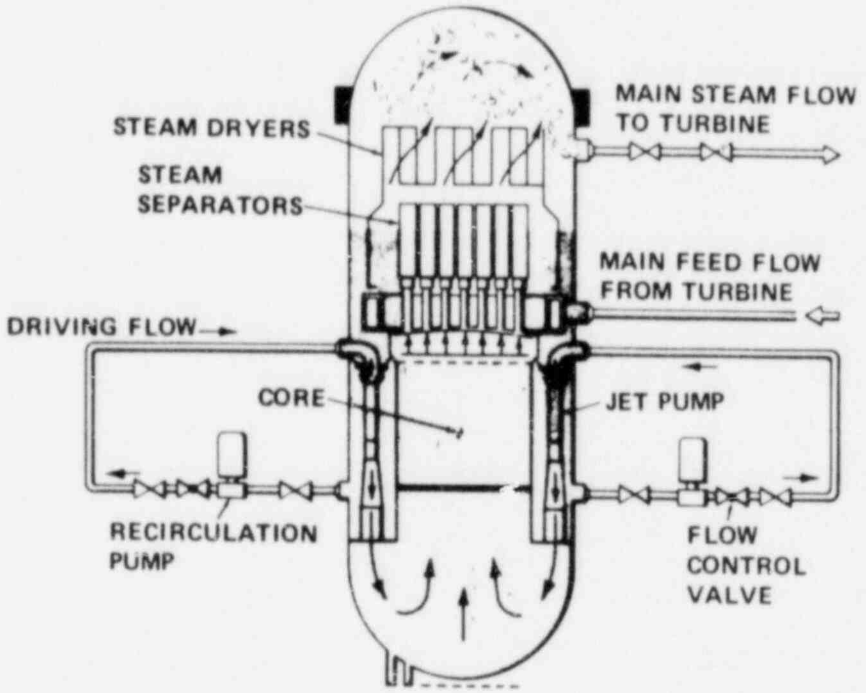
BWR FLOW SCHEMATIC



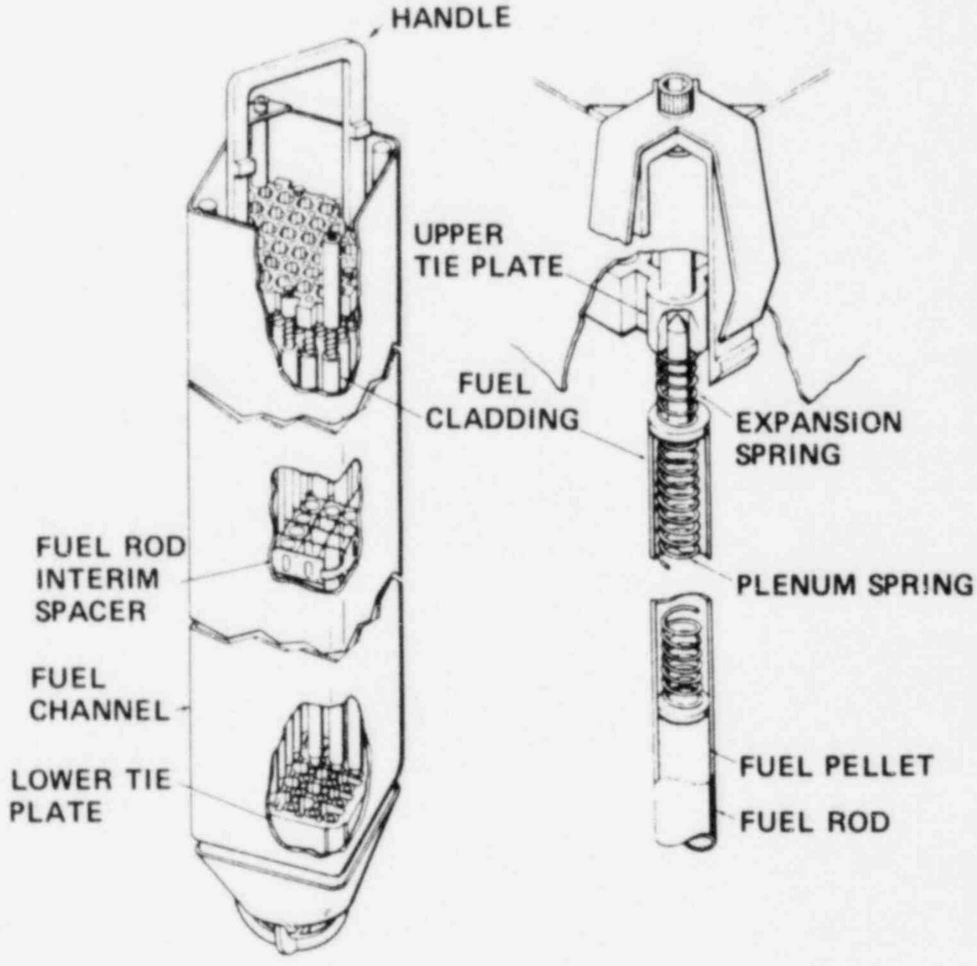
UNION CARBIDE

ORNL

BWR CORE COOLING SYSTEM



BWR FUEL ASSEMBLY



UNION CARBIDE
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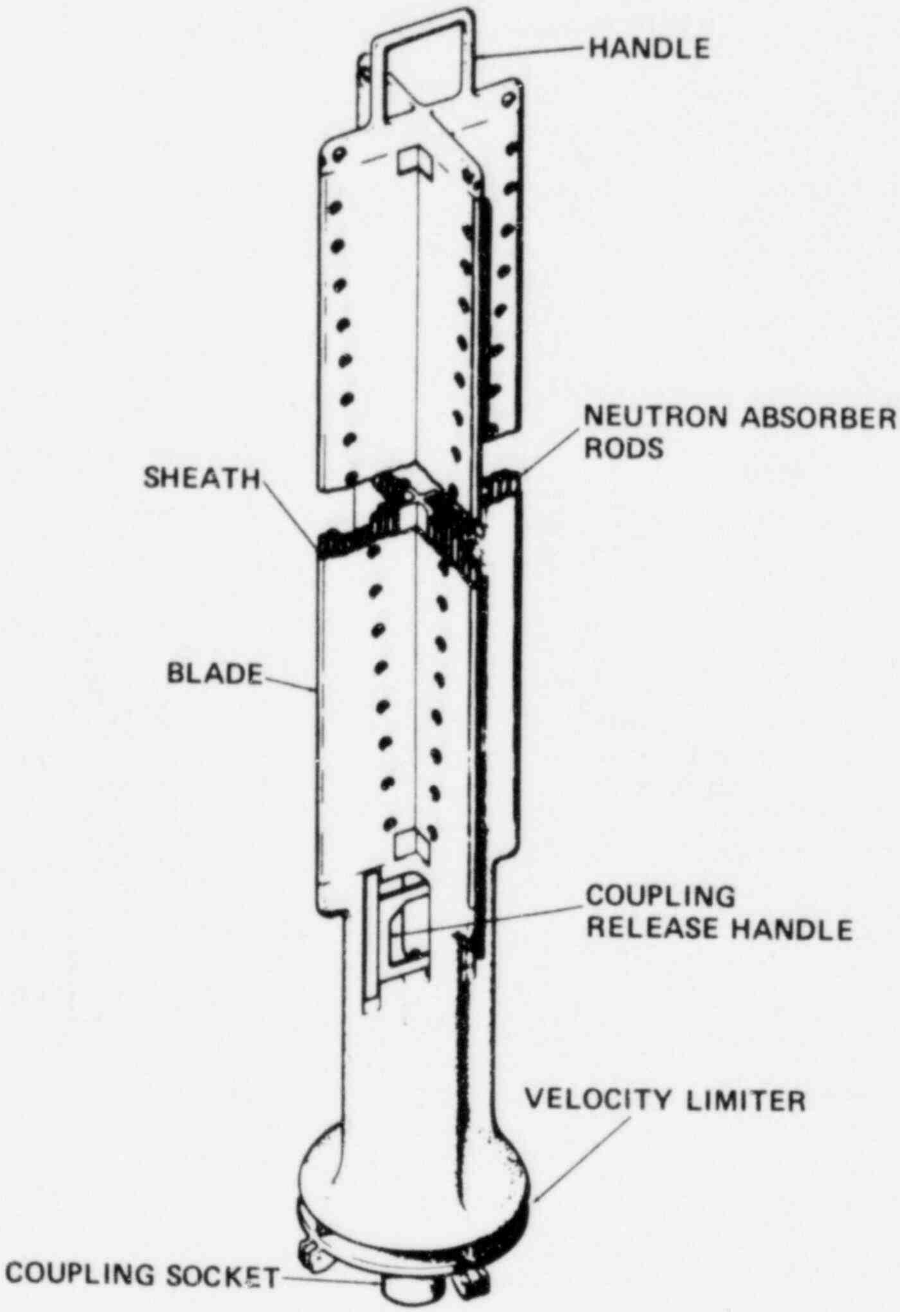
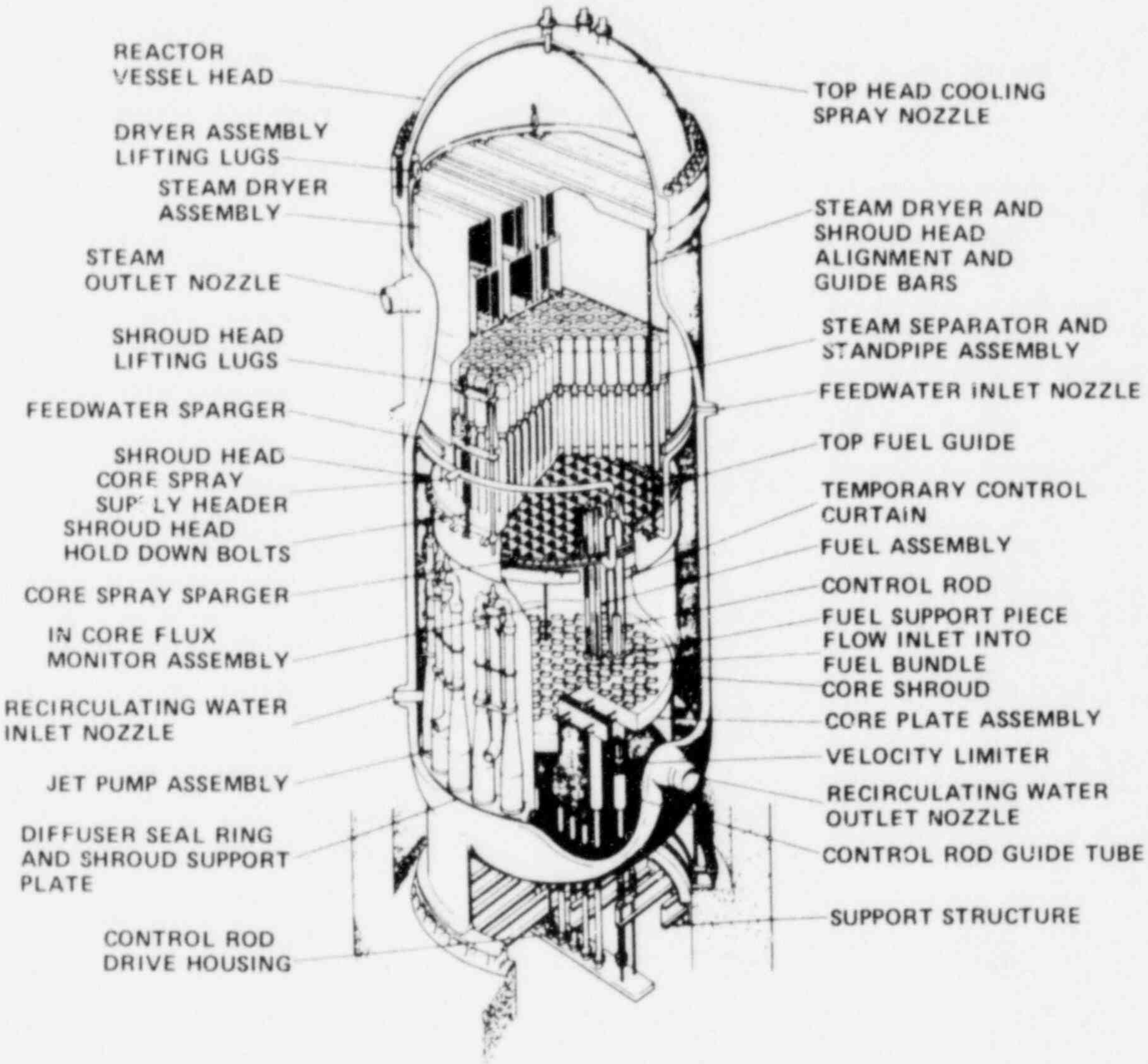
BWR CONTROL ROD



UNION CARBIDE

ORNL

BWR PRIMARY VESSEL INTERNALS



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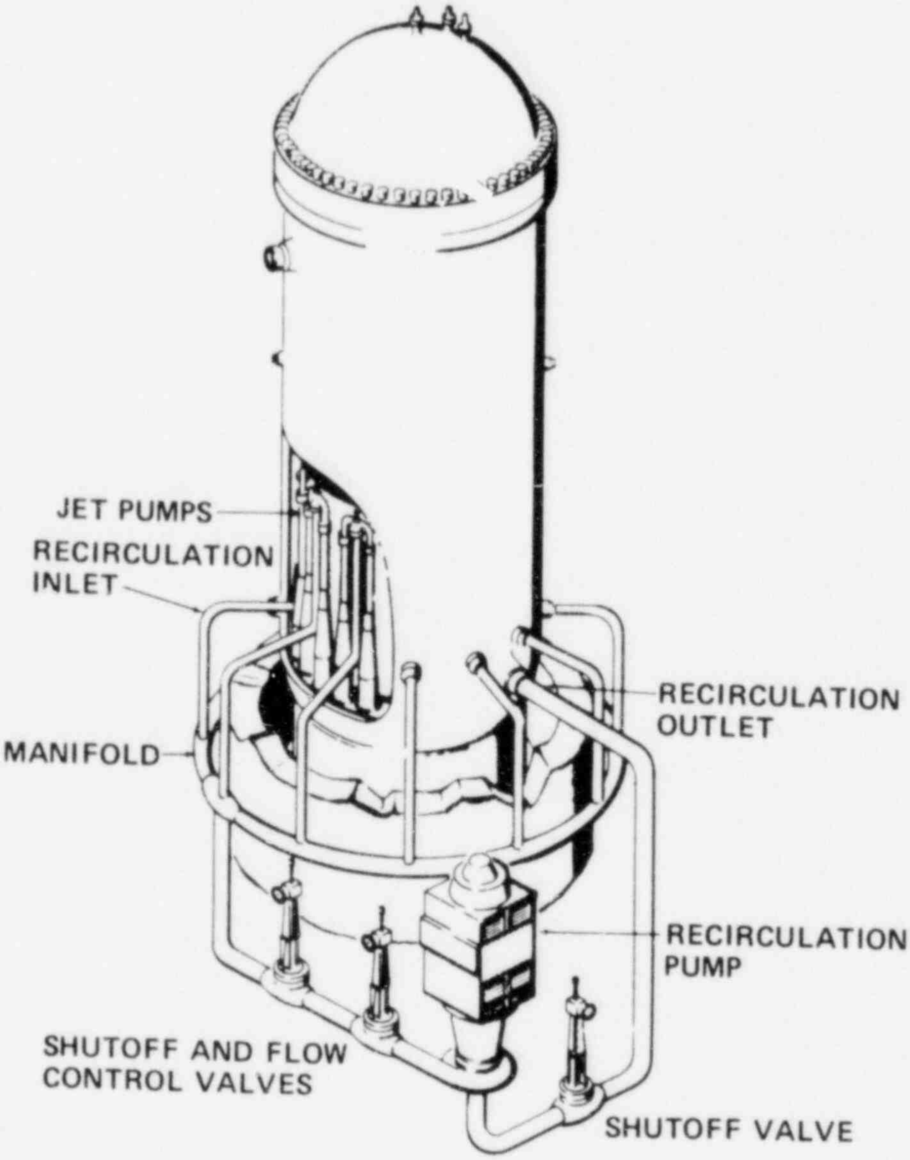
BWR PRIMARY VESSEL AND COOLING SYSTEM



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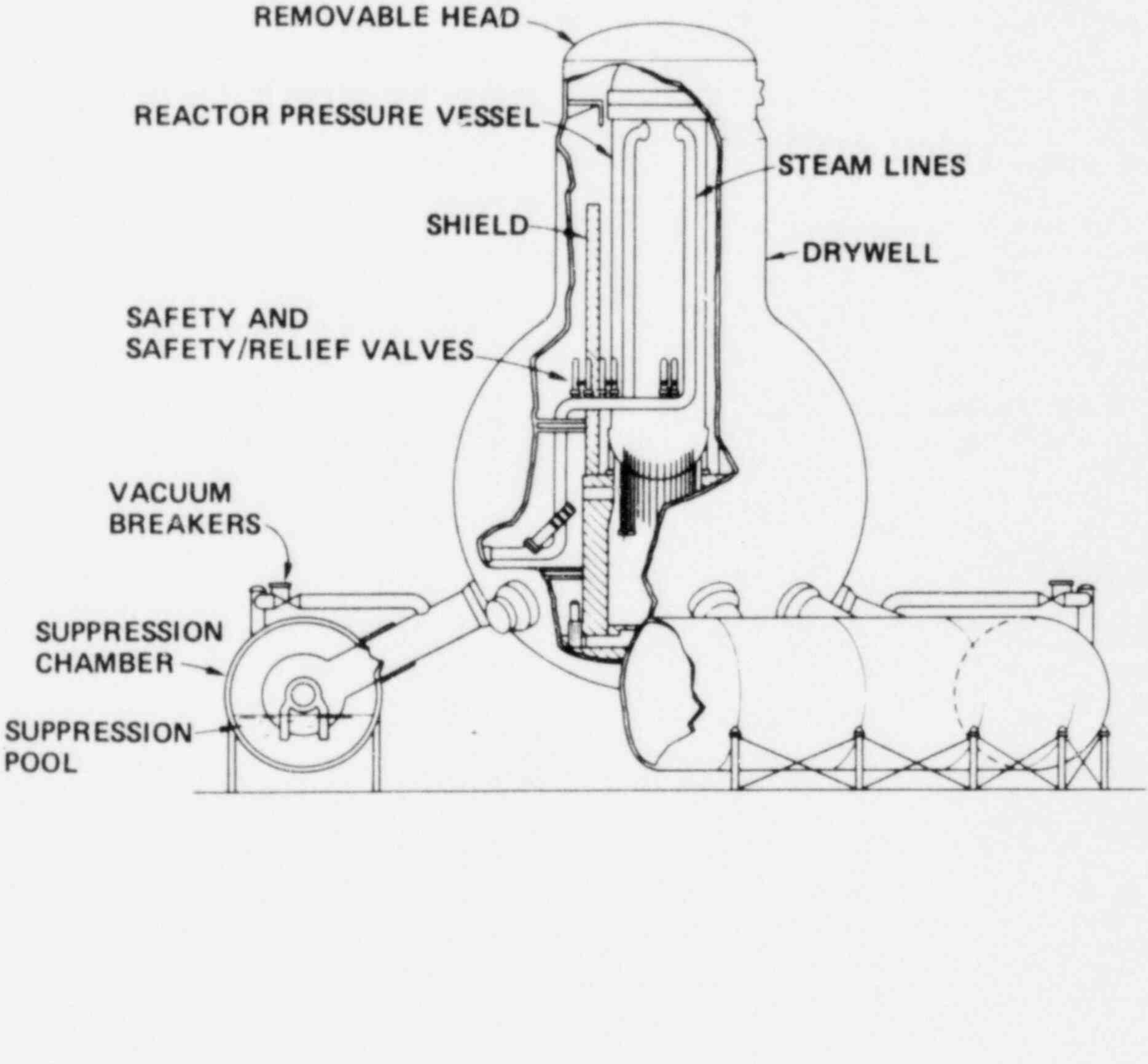
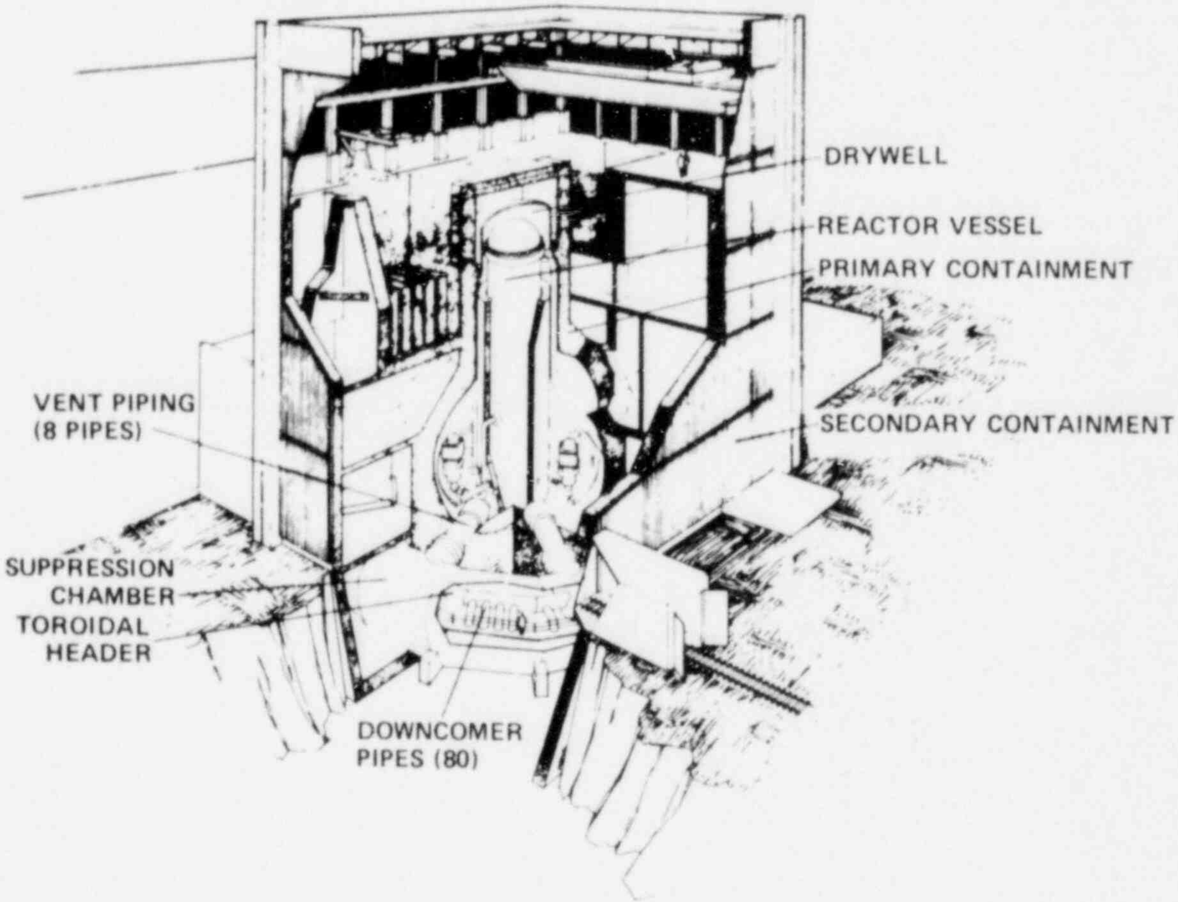
BWR CONTAINMENT SYSTEM (BROWN'S FERRY)



BWR REACTOR BUILDING SHOWING PRIMARY CONTAINMENT SYSTEM ENCLOSED



UNION CARBIDE

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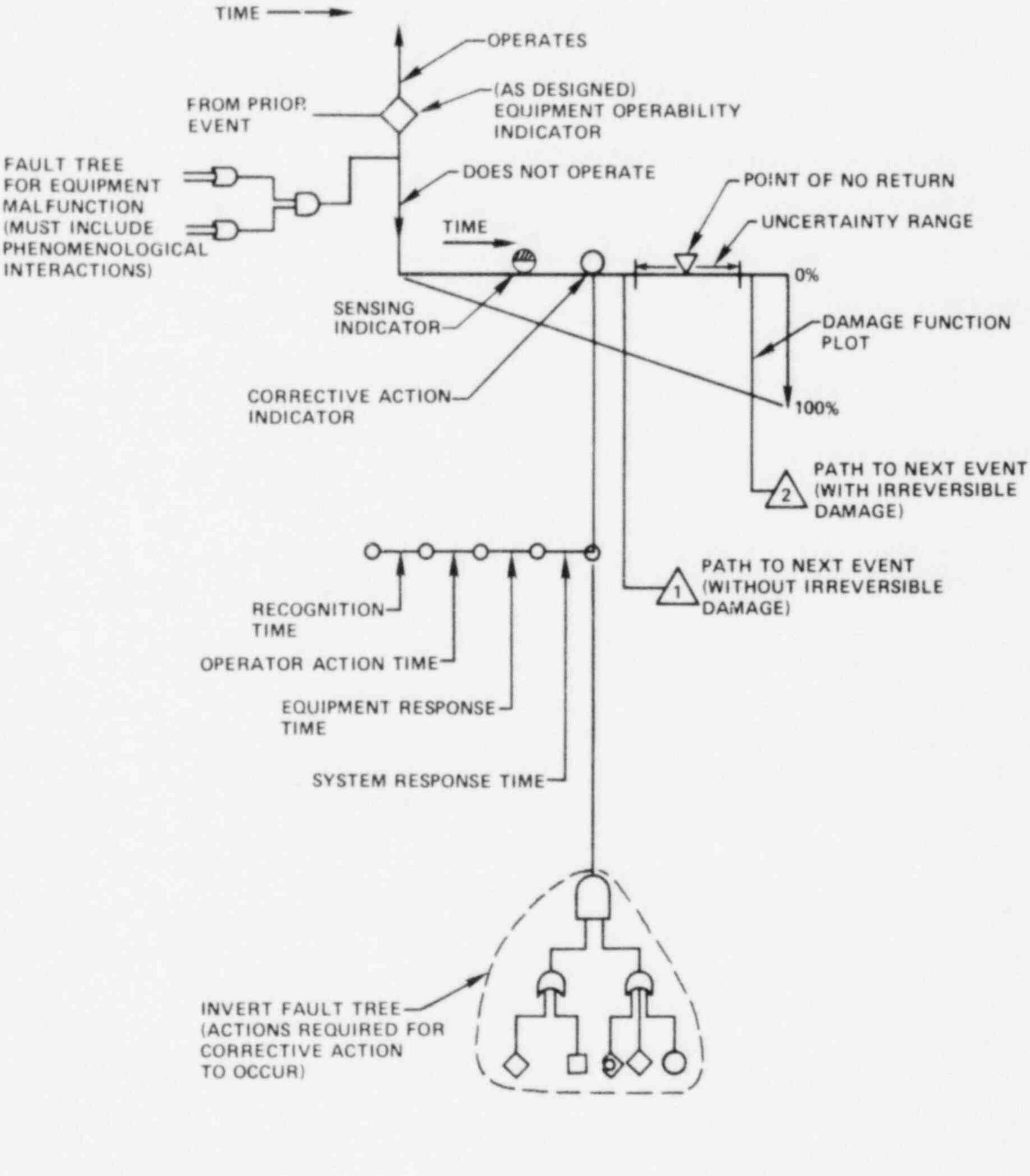
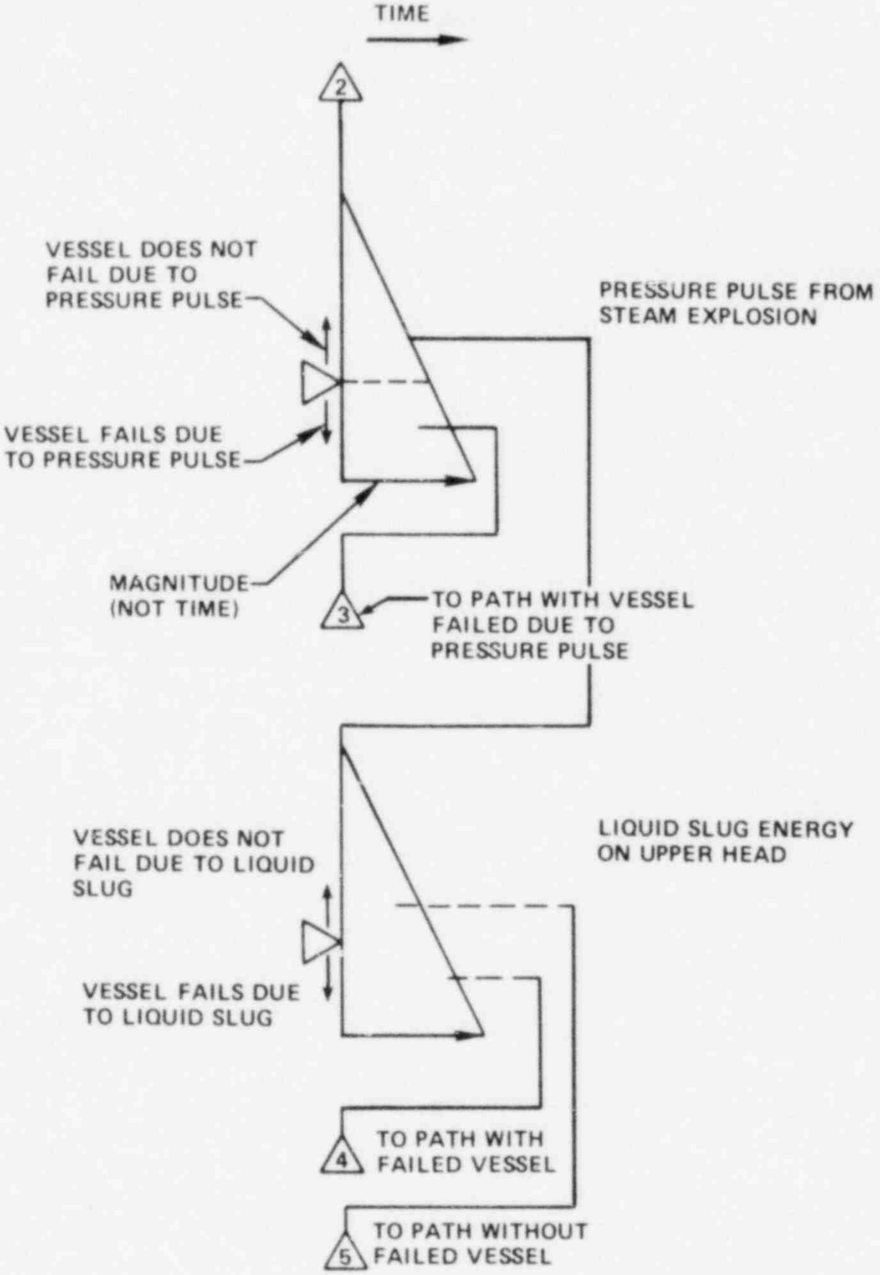
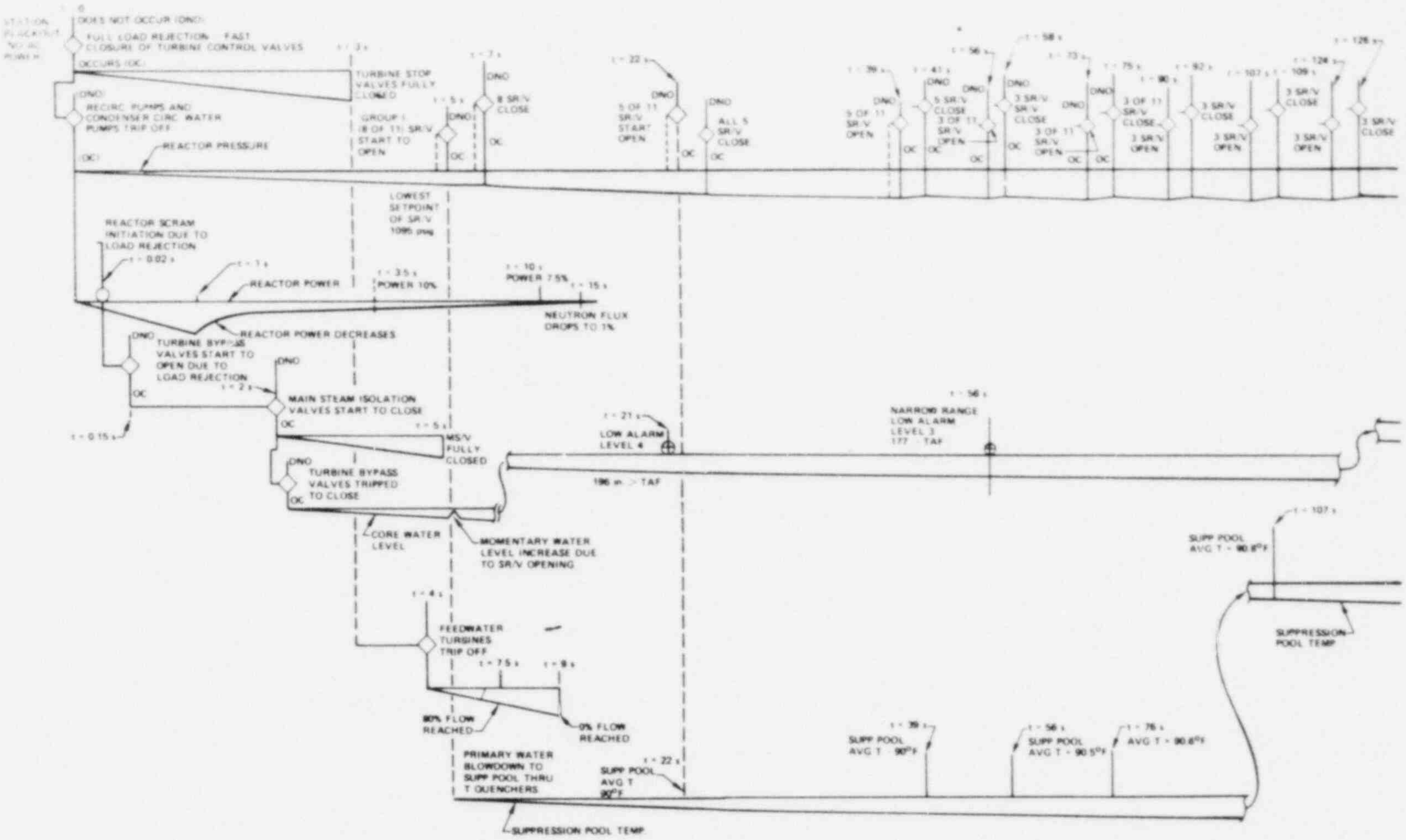
BASIC SEQUENCE PROGRESSION ELEMENTS



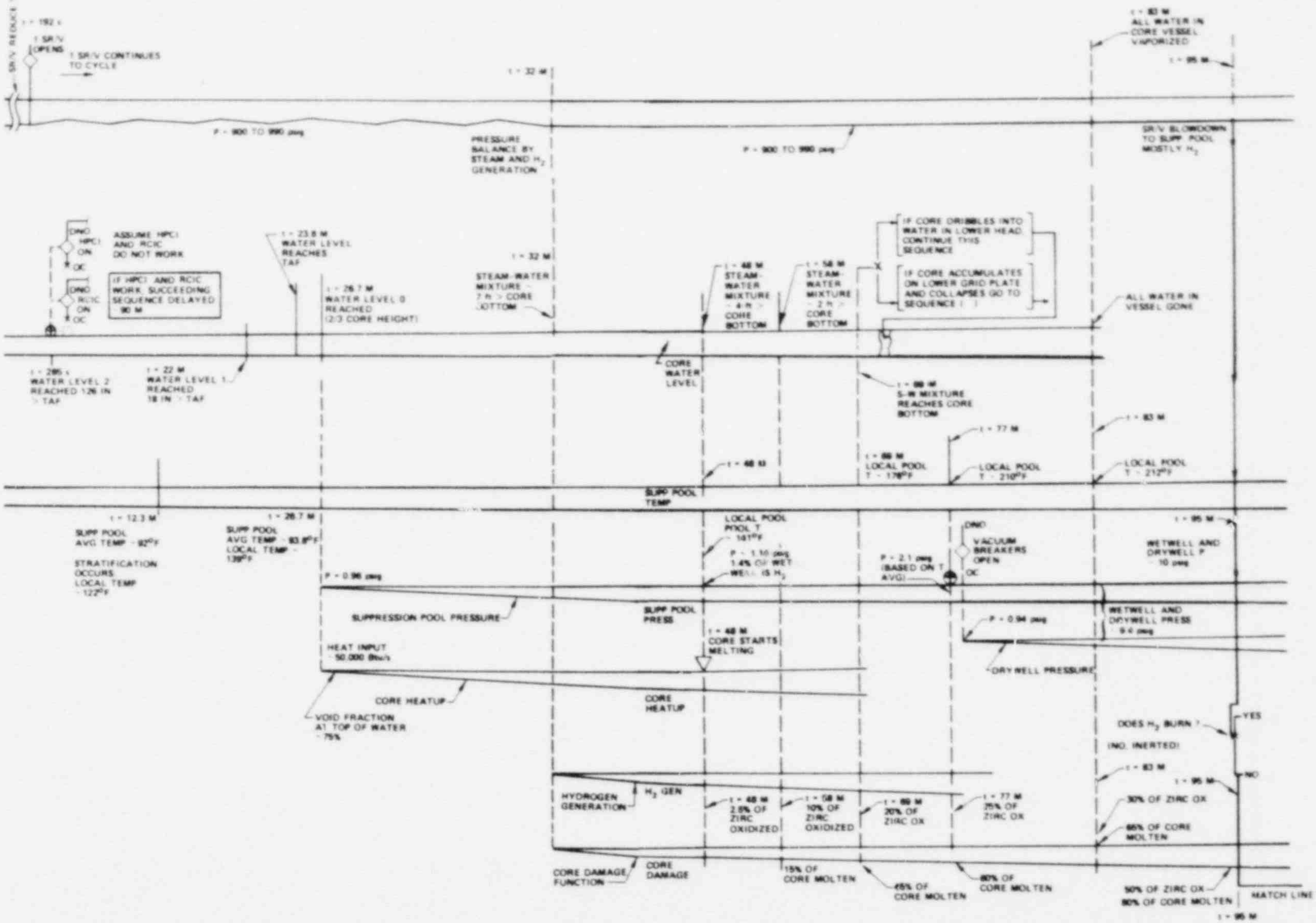
ORNL

PHENOMENOLOGICAL EFFECTS WHICH ARE TIME-INDEPENDENT OR INSTANTANEOUS ARE SHOWN AS FOLLOWS USING STEAM EXPLOSION AS AN EXAMPLE

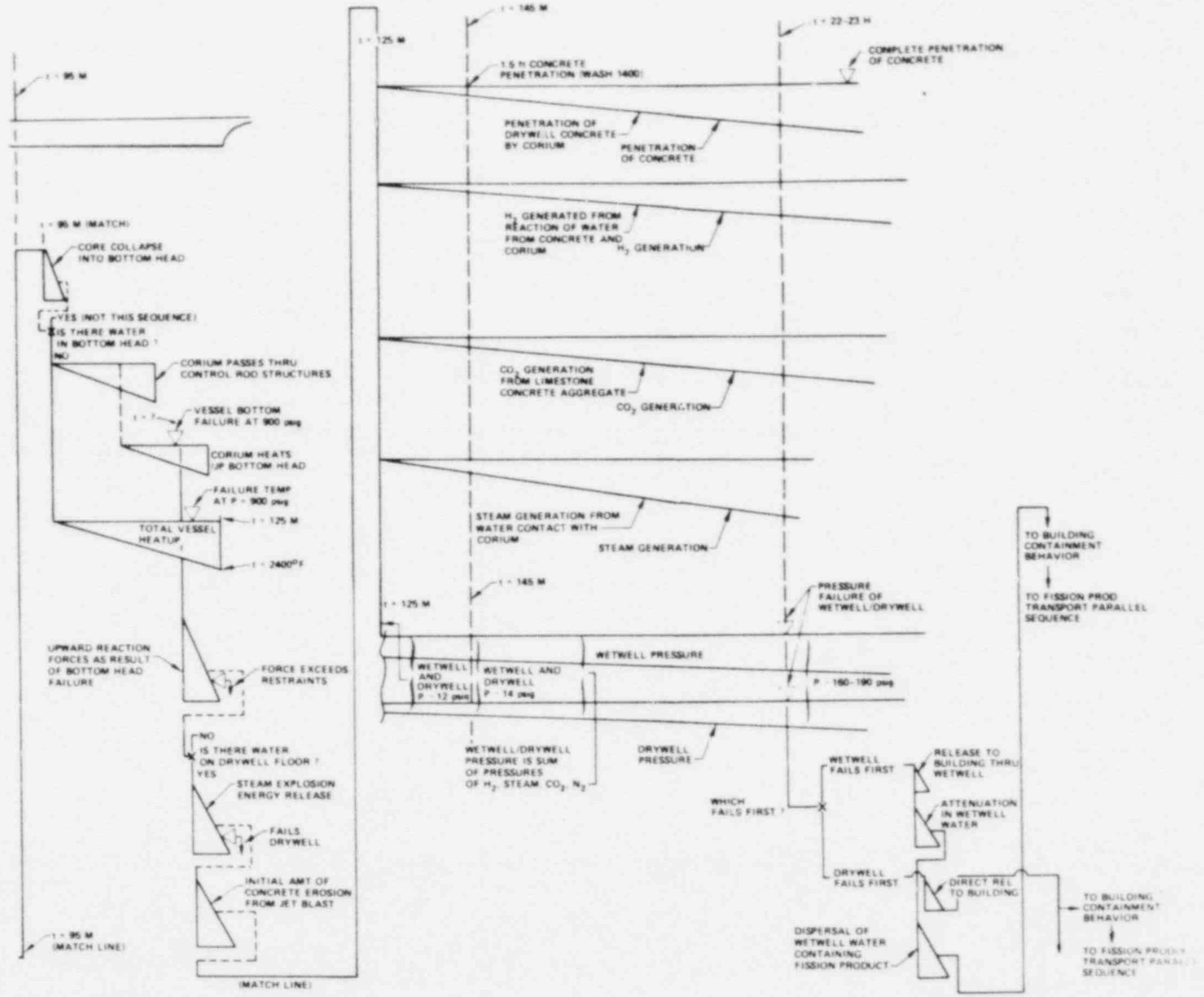




BROWNS FERRY UNIT 1 STATION BLACKOUT SEQUENCE ASSUMING NO NPCI OR RCIC (PRELIMINARY)



SR/V REDUCE TO 1
t = 192 s
1 SR/V OPENS
1 SR/V CONTINUES TO CYCLE
P = 900 TO 990 psig
t = 32 M
PRESSURE BALANCE BY STEAM AND H_2 GENERATION
P = 900 TO 990 psig
t = 83 M
ALL WATER IN CORE VESSEL VAPORIZED
t = 95 M
SR/V BLOWDOWN TO SUPP POOL MOSTLY H_2
DNO
HPCI ON
OC
ASSUME HPCI AND RCIC DO NOT WORK
DNO
RCIC ON
OC
IF HPCI AND RCIC WORK, SUCCEEDING SEQUENCE DELAYED 90 M
t = 23.8 M
WATER LEVEL REACHES TAF
t = 26.7 M
WATER LEVEL 0 REACHED (2/3 CORE HEIGHT)
t = 32 M
STEAM-WATER MIXTURE ~ 7 ft > CORE BOTTOM
t = 48 M
STEAM-WATER MIXTURE ~ 4 ft > CORE BOTTOM
t = 58 M
STEAM-WATER MIXTURE ~ 2 ft > CORE BOTTOM
IF CORE DRIBBLES INTO WATER IN LOWER HEAD, CONTINUE THIS SEQUENCE
IF CORE ACCUMULATES ON LOWER GRID PLATE AND COLLAPSES GO TO SEQUENCE ( )
ALL WATER IN VESSEL GONE
t = 285 s
WATER LEVEL 2 REACHED 126 IN > TAF
t = 22 M
WATER LEVEL 1 REACHED 18 IN > TAF
CORE WATER LEVEL
t = 69 M
S-W MIXTURE REACHES CORE BOTTOM
t = 77 M
t = 83 M
t = 48 M
t = 69 M
LOCAL POOL T ~ 178°F
LOCAL POOL T ~ 210°F
LOCAL POOL T ~ 212°F
SUPP POOL TEMP
t = 12.3 M
SUPP POOL AVG TEMP ~ 92°F
STRATIFICATION OCCURS LOCAL TEMP ~ 122°F
t = 26.7 M
SUPP POOL AVG TEMP ~ 93.8°F
LOCAL TEMP ~ 139°F
LOCAL POOL POOL T ~ 141°F
P ~ 1.10 psig
1.4% OF WETWELL IS H_2
DNO
VACUUM BREAKERS OPEN
OC
P = 2.1 psig (BASED ON T AVG)
t = 95 M
WETWELL AND DRYWELL P ~ 10 psig
P = 0.96 psig
SUPPRESSION POOL PRESSURE
SUPP POOL PRESS
P = 0.94 psig
WETWELL AND DRYWELL PRESS ~ 9.4 psig
DRYWELL PRESSURE
HEAT INPUT ~ 50,000 Btu/s
t = 48 M
CORE STARTS MELTING
CORE HEATUP
CORE HEATUP
VOID FRACTION AT TOP OF WATER ~ 75%
DOES H_2 BURN ?
YES
(NO, INERTED)
NO
HYDROGEN GENERATION
H_2 GEN
t = 48 M
2.5% OF ZIRC OXIDIZED
t = 58 M
10% OF ZIRC OXIDIZED
t = 69 M
20% OF ZIRC OX
t = 77 M
25% OF ZIRC OX
t = 83 M
30% OF ZIRC OX
65% OF CORE MOLTEN
t = 95 M
CORE DAMAGE FUNCTION
CORE DAMAGE
15% OF CORE MOLTEN
45% OF CORE MOLTEN
60% OF CORE MOLTEN
50% OF ZIRC OX
80% OF CORE MOLTEN
MATCH LINE
t = 95 M


t = 95 M
t = 95 M (MATCH)
CORE COLLAPSE INTO BOTTOM HEAD
YES (NOT THIS SEQUENCE)
IS THERE WATER IN BOTTOM HEAD ?
NO
CORIUM PASSES THRU CONTROL ROD STRUCTURES
VESSEL BOTTOM FAILURE AT 900 psig
CORIUM HEATS UP BOTTOM HEAD
FAILURE TEMP AT P = 900 psig
TOTAL VESSEL HEATUP
t = 125 M
t = 2400°F
UPWARD REACTION FORCES AS RESULT OF BOTTOM HEAD FAILURE
FORCE EXCEEDS RESTRAINTS
NO
IS THERE WATER ON DRYWELL FLOOR ?
YES
STEAM EXPLOSION ENERGY RELEASE
FAILS DRYWELL
INITIAL AMT OF CONCRETE EROSION FROM JET BLAST
t = 95 M (MATCH LINE)
(MATCH LINE)
t = 125 M
t = 145 M
t = 22-23 H
1.5 ft CONCRETE PENETRATION (WASH 1400)
COMPLETE PENETRATION OF CONCRETE
PENETRATION OF DRYWELL CONCRETE BY CORIUM
PENETRATION OF CONCRETE
H_2 GENERATED FROM REACTION OF WATER FROM CONCRETE AND CORIUM
H_2 GENERATION
CO_2 GENERATION FROM LIMESTONE CONCRETE AGGREGATE
CO_2 GENERATION
STEAM GENERATION FROM WATER CONTACT WITH CORIUM
STEAM GENERATION
t = 125 M
t = 145 M
PRESSURE FAILURE OF WETWELL/DRYWELL
WETWELL PRESSURE
WETWELL AND DRYWELL P = 12 psig
WETWELL AND DRYWELL P = 14 psig
P = 160-190 psig
WETWELL/DRYWELL PRESSURE IS SUM OF PRESSURES OF H_2, STEAM, CO_2, N_2
DRYWELL PRESSURE
WHICH FAILS FIRST ?
WETWELL FAILS FIRST
RELEASE TO BUILDING THRU WETWELL
ATTENUATION IN WETWELL WATER
DRYWELL FAILS FIRST
DIRECT REL TO BUILDING
DISPERSAL OF WETWELL WATER CONTAINING FISSION PRODUCT
TO BUILDING CONTAINMENT BEHAVIOR
TO FISSION PROD TRANSPORT PARALLEL SEQUENCE
TO BUILDING CONTAINMENT BEHAVIOR
TO FISSION PROD TRANSPORT PARALLEL SEQUENCE

ADVANCED INSTRUMENTATION OVERVIEW

OCTOBER 28, 1980

AFTERNOON SESSION - GREEN AUDITORIUM

PRESENTED BY:

YIH-YUN HSU

U.S. NUCLEAR REGULATORY COMMISSION

U.S. NUCLEAR REGULATORY COMMISSION

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

GAITHERSBURG, MARYLAND

The U.S. Nuclear Regulatory Commission (NRC) sponsored instrumentation research can be divided into two categories: (1) Advanced two-phase flow instrumentation for loss of coolant accident (LOCA) research, and (2) Power plant instrumentation for improved power plant operational safety.

I. Advanced Two-Phase Flow Instrumentation

These advanced two-phase flow instruments are used in test facilities investigating the thermal hydraulic behavior of light water reactors (LWR) during the LOCA. The measured data are then used to assess reactor safety analysis models. These experiments usually call for accurate measurements under severe steam-water conditions. Since most of the commercially available instruments cannot meet the requirements, the NRC and its contractors have been developing many advanced measurement devices and techniques over the past 4-5 years. A large number have been successfully developed and used in the test facilities.

The primary parameters of interest in two-phase flow studies are: density, water/steam phase velocities, mass flow, film flow and visual observations. Some of the instruments and techniques developed for these measurements are: the pulsed neutron activation (PNA) technique, instrumented spool piece, improved gamma densitometers, impedance probe, drag bodies, visual observation probes, optics (such as laser droppler anemometry and holography), and signal analysis and models. As a result of this research, the uncertainty of the measurements has been reduced by a factor of 5-10 in the past five years.

II. Instrumentation for Improved Power Plant Operation Safety

Power plant instrumentation was identified to be crucial for operational safety by the Three Mile Island-2 (TMI) accident investigations. Shortly after TMI-2, the NRC Office of Nuclear Regulatory Research began research efforts on power plant instrumentation. The objectives are: to assess the capability of vendor proposed devices, develop prototypes for the industry to improve reactor safety, and to transfer our LOCA advanced instrumentation technology to the industry. These objectives can be achieved by utilizing our unique LOCA study experience and test facilities.

The on-going program emphases LWR in-vessel liquid level detection to monitor inadequate core cooling. The prototypes under development and evaluation are heated thermocouples and torsional ultrasonic probes. The vendor supplied devices under evaluation are a dp system developed by the Westinghouse Electric Corporation and heated thermocouples developed by Combustion Engineering, Incorporated. All of the devices will be tested and evaluated under normal LWR operation conditions and simulated LOCA conditions in our LOCA experimental facilities.

IN THIS INSTRUMENTATION SESSION, WE HAVE THREE BLOCKS OF PROGRAMS.

1. ADVANCED INSTRUMENTATION PROGRAMS DIRECTLY SPONSORED BY THE SEPARATE EFFECTS RESEARCH BRANCH.

2. INSTRUMENTATION DEVELOPMENT PROGRAM TO SUPPORT THE 2D/3D TEST PROGRAM.

3. ADVANCED INSTRUMENTATION PROGRAM TO SUPPORT MAJOR NRC TEST PROGRAMS AT INEL.

POWER PLANT INSTRUMENTATION

- DEVELOPMENT OF VARIOUS FORMS OF HEATED THERMAL COUPLES FOR LIQUID (OR FROTH) LEVEL MONITORING

- DEVELOPMENT OF ULTRASONIC RIBBON FOR IN-CORE DENSITY PROFILE MEASUREMENT

- DEVELOPMENT OF HEATED THERMAL COUPLES FOR SLOW FLOW MEASUREMENTS

- TESTING OF VARIOUS LIQUID LEVEL MONITORING DEVICES IN NRC THERMAL HYDRAULIC FACILITIES AT ORNL AND INEL

THE OBJECTIVES OF THESE INSTRUMENTATION R/D PROGRAMS ARE:

1. TO REDUCE THE UNCERTAINTIES OF DATA OBTAINED FROM CONFIRMATORY RESEARCH TESTS FOR BETTER CONFIDENCE IN CODE DEVELOPMENT AND CODE ASSESSMENT;

2. A BETTER UNDERSTANDING OF PHYSICAL PHENOMENA RELATING TO SAFETY ISSUES; AND

3. TO IMPROVE THE POWER PLANT CORE-COOLING MONITORING CAPABILITY THROUGH DEVELOPMENT OF SENSORS TO PROVIDE DIRECT INDICATION OF CORE INVENTORY.

MEASURED PARAMETERS & NRC DEVELOPED INSTRUMENTS

	VOID FRACTION	VELOCITY	FLOW RATE	TEMPERATURE	OTHER
PULSED NEUTRON ACTIVATION (ANL)	X	2 PHASE			CALIBRATION
DENSITOMETERS (INEL)	X				
TURBINEMETER (INEL)		X			
DRAG BODIES (INEL & ORNL)			X		
ADVANCED SPOOL PIECE (INEL & ORNL)	X	X	X		
FILM PROBE (ORNL)		FILM VEL.			FILM THICKNESS
STAGNATION PROBE (INEL)		2 PHASE	X		
SIGNAL ANALYSIS (ORNL & INEL)	X	X			

MEASURED PARAMETERS & NRC DEVELOPED INSTRUMENTS (CONTINUED)

	VOID FRACTION	VELOCITY	FLOW RATE	TEMPERATURE	OTHER
DIGITAL INTERFEROMETER (RPI)	MAPPING				
LASER DOPPLER (SUNY-SB)		DROPLET VEL.			DROP SIZE
LASER HOLOGRAM (NWU)	DROPLET				CONDENSATION RATE
SUPERHEAT TEMPERATURE (LEHIGH)				ΔT_{SUP}	
OPTICAL TRANSDUCER (INEL)					FLOW PATTERN
ROD LENS (INEL & LASL)					FLOW PATTERN

RESEARCH INSTRUMENTATION

- SPOOL PIECES WITH MULTI-SAMPLING CAPABILITY AND MORE SENSITIVE COMPONENTS
- LOW ENERGY GAMMA DENSITOMETRY FOR HIGH VOID REGION (GAMMA > 0.9)
- IMPROVEMENT OF ALGORITHMIC FOR VARIOUS MEASUREMENT COMBINATIONS, INCLUDING SPOOL PIECES, BASED UPON TWO-PHASE FLOW MODELING PRINCIPLES
- IN-CORE MEASUREMENTS OF VOID, FLOW AND FILM FLOW THROUGH VARIOUS IMPEDANCE PROBES
- SPATIAL DISTRIBUTION OF DENSITY USING LIQUID LEVEL DETECTORS AND FLUID DISTRIBUTION GRIDS

RESEARCH INSTRUMENTATION (CONTINUED)

- IN-VESSEL VISUAL PROBES
- CORE UPPER PLENUM FLOW MEASUREMENTS
- OPTICAL METHODS SUCH AS LASSER-DOPPLER ANEMOMETRY AND HOLOGRAPHY
- PNA TECHNIQUE FOR FLOW CALIBRATION AND LOW FLOW MEASUREMENTS
- NEW TECHNIQUES USING NON-INTRUSIVE MEASUREMENTS, SUCH AS GAMMA-SCATTERING AND TOMOGRAPHY, ETC.

AS A RESULT OF THE ABOVE IMPROVEMENTS, UNCERTAINTY OF MEASUREMENTS HAVE BEEN REDUCED BY A FACTOR OF 5 - 10 IN THE PAST FIVE YEARS.

PULSED NEUTRON ACTIVATION TECNIQUES
IN WATER REACTOR SAFETY RESEARCH

PAUL KEHLER

COMPONENTS TECHNOLOGY DIVISION

ARGONNE
NATIONAL
LABORATORY

U of C-AUA-USDOE

PRESENTED AT THE

EIGHTH

WATER REACTOR SAFETY RESEARCH

INFORMATION MEETING

NATIONAL BUREAU OF STANDARDS
GAITHERSBURG, MARYLAND

OCTOBER 27-32, 1980

THIS WORK PERFORMED UNDER THE AUSPICES OF THE USNRC.

PULSED NEUTRON ACTIVATION TECHNIQUES

IN WATER REACTOR SAFETY RESEARCH

Paul Kehler
Components Technology Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Introduction

The pulsed Neutron Activation (PNA) technique is a radioactive tracer technique in which the radioactive tracers are produced by irradiating water by a short burst of fast neutrons. The neutrons interact with the oxygen of the water and produce a radioactive isotope of nitrogen through the reaction $O^{16}(m,p)N^{16}$, which decays with a half life of 7.2 sec. under emission of a 6.2 MeV gamma ray.

The high energies of both the neutrons and the gamma rays involved in the PNA process described above makes this technique very suitable for non-intrusive tagging and detection of water in thick-walled pipes and vessels. The short half life of the N^{16} also is desirable from health physics considerations. Thus, the reason why PNA techniques were not more widely used in the past, was the non-availability of small, portable pulsed neutron sources of sufficient output.

Portable pulsed neutron sources available commercially have an output of not more than 10^8 npp (neutrons per pulse). At the Argonne National Laboratory (ANL), we have performed our first reactor safety related PNA tests with classified neutron sources developed by the Sandia Laboratories

(Albuquerque, NM) for programs not related to reactor safety research. These sources have an output of about 2 x 109 npp. Over the past two years, the U.S. Nuclear Regulatory Commission has sponsored the development of a pulsed neutron source suitable for PNA work in the reactor safety research field. Program managers of this project are Y. Y. Hsu and A. L. Hon from the RSR Division of the USNRC. A separate paper is presented at this meeting by Gary Rochau (Sandia Laboratories) on the characteristics of this source.

From the work conducted at the ANL during the past few years, it was concluded that PNA techniques can be usefully applied to three areas of reactor safety research which are listed in Fig. 1. Results of the work performed at the ANL in these three areas during the last year, as well as the effort planned for the next year, are summarized in Fig. 2 and presented in the following:

Calibration of Two-Phase Flow Instruments

In PNA measurements, the activity introduced into the fluid by neutron activation is recorded by a detector located downstream of the neutron source. The detector is hooked up to a multichannel analyser operated as a multiscaler, which stores in separate time channels the number of counts, C, that are registered by the detector during a time interval Δt at the time T after the neutron burst. The C versus T distribution is then evaluated both in terms of transit time and total injected activity (as shown in Fig. 3) for measurement of the mass flow velocity U and density D. The Z in Fig. 3 is the source-detector spacing, A(D) is a relative measure of the injected activity (which is a function of D), and the K-value of the equipment is the measured density for a water-filled pipe ($D=1\ g/cm^3$). It was shown previously[1,2] that, under idealized conditions, both the PNA velocity equation and the PNA density equation are not effected by variations in the flow regime, and validity of these equations for pipes with diameters less than 10 cm was verified experimentally.[3] For larger pipes, both the velocity and the density equations must be corrected by analytically derived factors, as discussed below. Guidelines for the proper planning of PNA test on large pipes will be prepared by ANL in 1981.

PNA Velocity Measurement

Prior to entering two-phase water reactor safety research, ANL has demonstrated that single-phase flow velocities in large pipes (up to 40 cm diameter) can be measured by NPA technique within a few percent (Fig. 4). It is believed that two-phase flow can be measured with a comparable accuracy, especially when the PNA velocity equation is corrected by an analytically derived correction factor prior to the test. Experimentally, the accuracy of PNA measurements could be verified by comparing PNA readings with flow data derived by weight-tank methods. The U.S. Bureau of Standards has, in their local laboratories, large diameter loops terminating in weight tanks, that could be used for such a project. Early in 1981, ANL plans to conduct single-phase PNA flow measurements in 50 cm diameter pipes and to compare these measurements with data derived by weight tank methods and by ultrasonic flowmeters.

The errors introduced into large-pipe PNA measurements are due to the attennation of neutrons and gamma rays in the fluid and to poor collimation of the neutron source and the detector. The effects of neutron attennation in large pipes is shown in Fig. 5, which shows the radial distribution of the induced activity in the downcomer pipe of the PKL, which has an inside diameter of 20 cm. As shown in Fig. 5, the radial activity distribution in pipes of this size can no longer be considered to be uniform; the specific activity at the centerline of the pipe is much lower than the specific activity at the wall. Since the fluid at the wall is moving slower than the fluid at the center, this non-uniform activity distribution in large pipe tends to introduce a negative error into the PNA flow measurement, i.e., the measured mass flow velocity is lower than the actual one. Similarly, due to the attennation of gamma rays, the sensitivity of the detector towards radiation emitted by regions close to the pipe wall is higher than towards radiation that is emitted from the center of large pipes. This will enhance the effects caused by neutron attenuation and the resulting non-uniform radial distribution of activities, and will further reduce the measured velocity.

An increase of the measured velocity, on the other hand, will be caused by poor collimation of the source (and/or the detector),[2] which causes an extended axial distribution of activities along the pipe and a broadening of the measured counts distribution.

The effects of non-uniform radial distributions and of extended axial distributions tend to cancel each other. If one or the other of these two effects predominates, however, a negative or a positive error is introduced into the PNA velocity measurement. At the PKL (at Erlangen, Germany) the proximity of the loop operator from the neutron source necessitated extremely good shielding of the neutron source, which was achieved by mounting the target of the neutron generator at a distance of 35 cm from the wall of the pipe. Since the radial distribution of activities is effected not only by the attennation of neutrons in the fluid but also by a decrease of the neutron fluence from the source by $1/r^2$, the relatively large distance of the neutron source from the pipe caused a relatively "flat" radial distribution of activities in the downcomer of the PKL (Fig. 5). On the other hand, the large source distance, combined with poor source collimation, caused an axial distribution of activities whose dimension was comparable to the source-detector spacing (Fig. 6). It could be expected, therefore, that with the PKL test arrangement, the effect of the axial activity distribution would predominate and that the PNA velocity measurement would be too high if based on the basic, uncorrected velocity equation of Fig. 3.

Some thought was spent on how these effects of the non-uniform radial activity distribution and of the poor source collimation should be compensated by a properly chosen data reduction technique. Correction factors to the basic PNA velocity equation (Fig. 3) could be applied in several different manners. For example, the distribution of the counts, C, could be corrected before entering it into the equation. Or, alternately, an "effective" spacing Z could be derived by correcting the true spacing for effects of non-uniform and/or widely spaced activity distributions. A third method, the one preferred by the author,[2] consists of modifying the time exponent in the basic PNA velocity equation. In this approach, the equation of Fig. is assumed to be a special case (with n=Z) of the general PNA equation shown in Fig. 7.

The one feature of this method of correcting the basic PNA velocity equation that makes it very attractive is, that a physical meaning can be assigned to various values of the exponent n. It was shown[2] than an exponent n=2 must be chosen whenever the velocities of the activated mass elements at the detector are equal to their transit velocities and whenever there ex ıts a known relation between the transit time of a fluid element and its transit velocity. Under adverse experimental conditions, one or both of these two requirements may not be met. For example, a disturbance of the flow between the source and the detector will modify or even destroy the correlation between transit velocity and detector velocity of the activated fluid elements. A bend in the pipe or a DDT rake are examples for such flow disturbances. The most severe disturbance would be caused by a homogenizing screen installed right in front of the detector. Such a screen would completely destroy any relation between the transit velocity and the detector velocity of the activated fluid elements. Nothing would be known about the detector velocities of the individual fluid elements, and one would have to assume a constant value for all of them. Under these conditions, the time exponent n in the general PNA velocity equation, Fig. 7, must be assumed to be $n=1$.

For very extreme cases of poor source collimation (see Fig. 6, which comes close to such a condition), the fundamental relation on which the basic PNA velocity equation rests, i.e., the assumption that the measured transit time of a fluid element is related to its transit velocity, is not satisfied. For the situation depicted in Fig. 6, the fluid elements that are the first ones to arrive at the detector are not necessarily the fastest ones; some of these fluid elements could have slow velocities but could have been activated in close proximity of the detector. For very extreme situations, where there is no correlation between transit times and transit velocities, one must forego an attempt to interpret PNA test data in terms of mass flow velocity, and must satisfy himself with an "average" transit time $\overline{T}$, derived by count-weighted averaging of the measured transit times:

$$\overline{T} = \frac{\sum T\ C}{\sum\ C}$$

This $\overline{T}$ is then used for the calculation of "average velocity $\overline{U}$" by

$$\overline{U} = Z/\overline{T}$$

or

$$\overline{U} = Z\,\frac{\sum\ C}{\sum\ C\ T}$$

The last equation is identical to the equation shown in Fig. 7 with a time exponent of n=0. Thus, physically meaningful interpretations can be given to the time exponents n=2, n=1 and n=0 in the general PNA velocity equations. For large ℓ/D ratios, all these values of n converge towards the same mass flow velocity. For smaller ℓ/D ratios, U does increase with increasing n. This latter fact can be used to adjust the PNA data reduction technique to existing non-ideal experimental conditions, and to choose a data reduction technique that will result in a negligible measurement error. This task of optimizing the data reduction technique can even be more generalized by allowing non-integer values for n, and by allowing values for n that extend beyond 3.

PNASIM, a computer program simulating the PNA velocity measurement technique, was used to derive an optimized time exponent n for the PKL test. PNASIM is an advanced version of ACT-OPT[4] and calculates the number of counts registered by the detector as a function of the transit time. PNASIM calculates the activity of the fluid activated by the pulsed neutron source, the transportation of this activity down the pipe (with a given velocity distribution and a given mixing of the fluid), and the detection of this activity by the detector. A mass flow velocity is then computed for different time exponents n, from the calculated counts distribution. That time exponent n for which the difference between the calculated mass flow velocity and the mass flow velocity given as input to the program is zero, is then specified to be used for the reduction of the actual test data.

Figure 8 is a comparison of actual PKL test data (with intermediate velocity) to a PNASIM simulated distribution. The agreement between test and prediction is good and gives confidence in PNASIM derived data.

Figure 7 shows the accuracy of PNA velocity measurements for different time exponent n. An exponent of n=0.5 was specified by ANL for the reduction of test data taken at the PKL. PNA derived velocity values, using n=0.5, agreed very well with some reference measurements made by PKL.

Another test condition analyzed by PNASIM is the test that we performed at the FAST loop operated by EG&G Idaho, Inc., at the INEL. This test (with still unpublished test data) was briefly described at last year's meeting. At the FAST loop, four neutron sources were positioned much closer to the pipe than the source at the PKL. This caused a relatively "steep" radial activity distribution and a much narrower axial distribution than the one at the PKL. The effects of the radial distribution predominated in the velocity measurement, and the PNA equation had to be corrected for minimum error by introducing values for N that are larger than the one used in the basic equation of Fig. 3. The Fig. 9 shows that with values of $2.5 \leq N \leq 3.5$, error-free velocity readings can be performed over the whole density range. If a value of N=2.9 is specified for all measurements, the error of the velocity reading over the whole density range is less and ±2%.

The discussion given above leads to the conclusion that analysis is necessary for designing a PNA test setup with minimum errors or, alternately, if an optimized arrangement of the source and the detector is not possible, then analysis can be used to correct the velocity data derived by a non-optimized experimental setup.

The errors to be compensated by analysis are of such magnitude that the PNA technique appears feasible for calibration of other two-phase flow velocity measuring systems in large pipes.

PNA Density Measurement

In pipes with diameters less than 10 cm, global density measurements can be performed by using the basic density equation shown in Fig. 3. This equation is not sensitive to flow regimes.[1,3] In larger pipes, however, density measurements must be corrected by analytically derived correction factors, as proposed in last year's meeting. Density correction values were calculated (by the computer program PNASIM) for the FAST experimental setup, as shown in Fig. 10. These correction values are quite severe; in low density regimes the measured values must be multiplied by a factor of more than three. The question of how sensitive the correction factors shown in Fig. 10 are to the flow model used in the computer program PNASIM, has not yet been studied.

Because of the severe correction requirements for PNA density readings taken in large pipes, supplemental global density readings should be very helpful. Accurate density readings probably can be performed concurrently with the PNA measurement by using the high energy neutron source and published neutron scattering techniques.[5] A global density measuring technique based on gamma ray transmission and using the torus detector normally used for PNA tests, is currently being investigated at ANL. Both these neutron and gamma ray techniques would require minimal additions to the standard PNA instrumentation system.

The conclusions of this paragraph are summarized in Fig. 11.

Measurement of Flow Distributions in Large Test Facilities (Fig. 12)

PNA work was started at ANL with flow distribution measurements at the 3-D facility in mind. Hopefully, these tests will be conducted once the 3-D test series begins. The feasibility of flow distribution measurement in large facilities was demonstrated at the LOFT.[6]

Measurements of flow distributions in the Slab Core Test Facility, using PNA techniques, are planned for 1981.

Measurement of Slow Flow Velocities

In the recent past, many large test facilities were operated in a mode simulating small-break loss of coolant accidents of PWR's. These tests last from about half an hour to several hours. The pumps are shut off during these tests, and flow through the system is caused by heat convection through the core. Under these conditions, the flow velocities in the pipes are too low to be measured accurately by conventional equipment such as turbines, drag disks or pressure sensing devices. The PNA technique, on the other hand, works best at slow flow velocities and low void fractions. The usefulness of the PNA technique in measuring slow flow was demonstrated at the PKL[2] and at the LOFT facility[7] (Fig. 13).

The Primaerkreislauf (PKL) facility is operated by the KWU (a subsidiary of Siemens) at Erlangen, Germany. The PKL is a mixed-scale simulation of a 1300 MW_e PWR: it has full vertical dimensions but is scaled down horizontally. The core contains 340 electrically heated "fuel bundles" and has a diameter of 52 cm. The downcomer is simulated by a separate pipe, having an ID of 20 cm. Figure 14 is a schematic diagram of the main components of the facility. The PNA detectors were mounted right over the large flange in the downcomer, and the neutron source was mounted (on a different floor) 1.1 m above the detector.

During the small-break test series conducted at the PKL, in March-May 1980, the flow velocities measured in the downcomer by the PNA technique ranged between .03 m/sec and .4 m/sec. The PNA system was the primary velocity measuring device for this test series. For single-phase flow through the core, a reference flow rate was calculated by thermal balance considerations, based on measurements of the temperature at the bottom and the top of the core, as well as on measurements of the power input to the core. These calculated flow rates agreed very well with the PNA data. Obviously, such reference flow data were not available for two-phase flow through the core.

The first prototype of pulsed neutron sources specifically developed for PNA work by the Sandia Laboratories was used in this test. This source, which worked reliably throughout the whole test at the specified level of 10^{10} neutrons/pulse, will be described in the paper following this one.

A water-cooled, four-segmented NaI Torus detector was mounted right above the large flange on the downcomer. The detector signals were fed through stabilized amplifiers (locked on the Cs^{137} peak) and discriminators (set at 1 MeV) to a Packard Model 9012 multichannel analyzer, which was operated in its multiscaler mode (256 channels). Test data originally stored in the multichannel analyzer were then retrieved and analyzed by a HP 9845B computer.

A single trigger was used to operate the source, to start the multiscaler, and to furnish an interfacing time signal to the DAS of the PKL.

PNA data typical for the PKL test are shown in Fig. 8. The PKL test was the first one in which PNA data were reduced completely automatically. The operator of the computer could not alter the data reduction program. The selection of the start- and the end- channels of the evaluation range, as well as the background correction over this evaluation range, were fully automated. Velocity values derived by this automatic technique were then permanently stored on tape for retrieval of summary results.

EG&G Idaho, Inc. is also using a PNA system for the measurement of slow flow velocities during the L-3 series of LOFT tests.[7] Standard, off-the-shelf NaI detectors are used in their setup. Original difficulties in shielding these detectors from the severe radiation environment, existing even after shutdown of the reactro, were successfully overcome.

References

1. Paul Kehler, "Two-Phase Flow Measurement by Pulsed Neutron Activation Techniques," in "Measurements in Polyphase Flow," edited by David E. Stock, ASME Publications H00121, December 1978.

2. Paul Kehler, "Measurement of Slow Flow Velocities by the Pulsed Neutron Activation Technique," Proc. of the USNRC Review Group Conference on Advanced Instrumentation for Reactor Safety Research, July 29-31, 1980, NUREG/CP-0015.

3. Paul Kehler, "Two-Phase Flow Measurement by Pulsed Neutron Activation Techniques," ANL-NUREG-CT-78-17, January 1978.

4. Paul Kehler, "Feasibility of Calibration of Liquid Sodium Flowmeters by Neutron Activation Techniques," ANL-CT-76-17, July 1976.

5. S. Banerjee and P. Yuen, "A Fast Neutron Scattering Technique for Measuring Void Fractions in Rod Bundles," in "Topics in Two-Phase Heat Transfer and Flow," edited by S. G. Bankoff, ASME Publications H00129, December 1978.

6. Paul Kehler and C. W. Solbrig, "Pulsed Neutron Activation Measurement of Emergency Core Coolant Bypass Flow on the LOFT Reactor," NUREG/ICR-0208, ANL-CT-78-37-2, Revision 1, June 1980.

7. Quick-Look Report on L3-7 Test, EGG-LOFT-5192, June 1980.

FIG. 1:

APPLICATION OF PNA TECHNIQUE FOR WATER REACTOR SAFETY RESEARCH

- CALIBRATION OF TWO-PHASE FLOW INSTRUMENTS
- MEASUREMENT OF FLOW DISTRIBUTIONS IN LARGE TEST FACILITIES
- MEASUREMENT OF SLOW FLOW VELOCITIES

FIG. 2:

PUBLICATIONS ISSUED SINCE LAST YEAR'S MEETING

1. Paul Kehler, "Accuracy of Two-Phase Flow Measurement by Pulsed Neutron Activation Techniques," p. 2483, Vol. 5, Multiphase Transport Fundamentals, Reactor Safety, Applications (Hemisphere Publishing Corp., Washington, D. C., May 1980).

2. Paul Kehler and C. W. Solbrig, "Pulsed Neutron Activation Measurement of Emergency Core Coolant Bypass Flow on the LOFT Reactor," NUREG ICR-0208, ANL-CT-78-37-2, Revision L, June 1980.

3. Paul Kehler, "Measurement of Slow Flow Velocities by the Pulsed Neutron Activation Technique," Proc. of the USNRC Review Group Conference on Advanced Instrumentation for Reactor Safety Research, July 29-31, 1980, NUREG/CP-0015.

FIG. 3:

CALIBRATION OF TWO-PHASE FLOW INSTRUMENTS

- VELOCITY MEASUREMENT

$$U = Z \frac{\sum \frac{1}{T^2} C}{\sum \frac{1}{T} C}$$

- DENSITY MEASUREMENT

$$A(D) = \sum \frac{1}{T} C$$

$$K = A(1)$$

$$D = \frac{A(D)}{K}$$

FIG. 4:

PNA VELOCITY MEASUREMENT

- ACCURACY OF BETTER THAN A FEW % DEMONSTRATED FOR SINGLE-PHASE FLOW
- ACCURACY OF TWO-PHASE FLOW MEASUREMENTS EXPECTED TO BE SIMILAR TO THE ACCURACY OF SINGLE-PHASE MEASUREMENTS
- ACCURACY OF TWO-PHASE FLOW MEASUREMENT IN LARGE PIPES IS INCREASED BY ANALYTICAL MODELING

FIG. 5:

RADIAL ACTIVITY DISTRIBUTION IN THE DOWNCOMER OF THE PKL

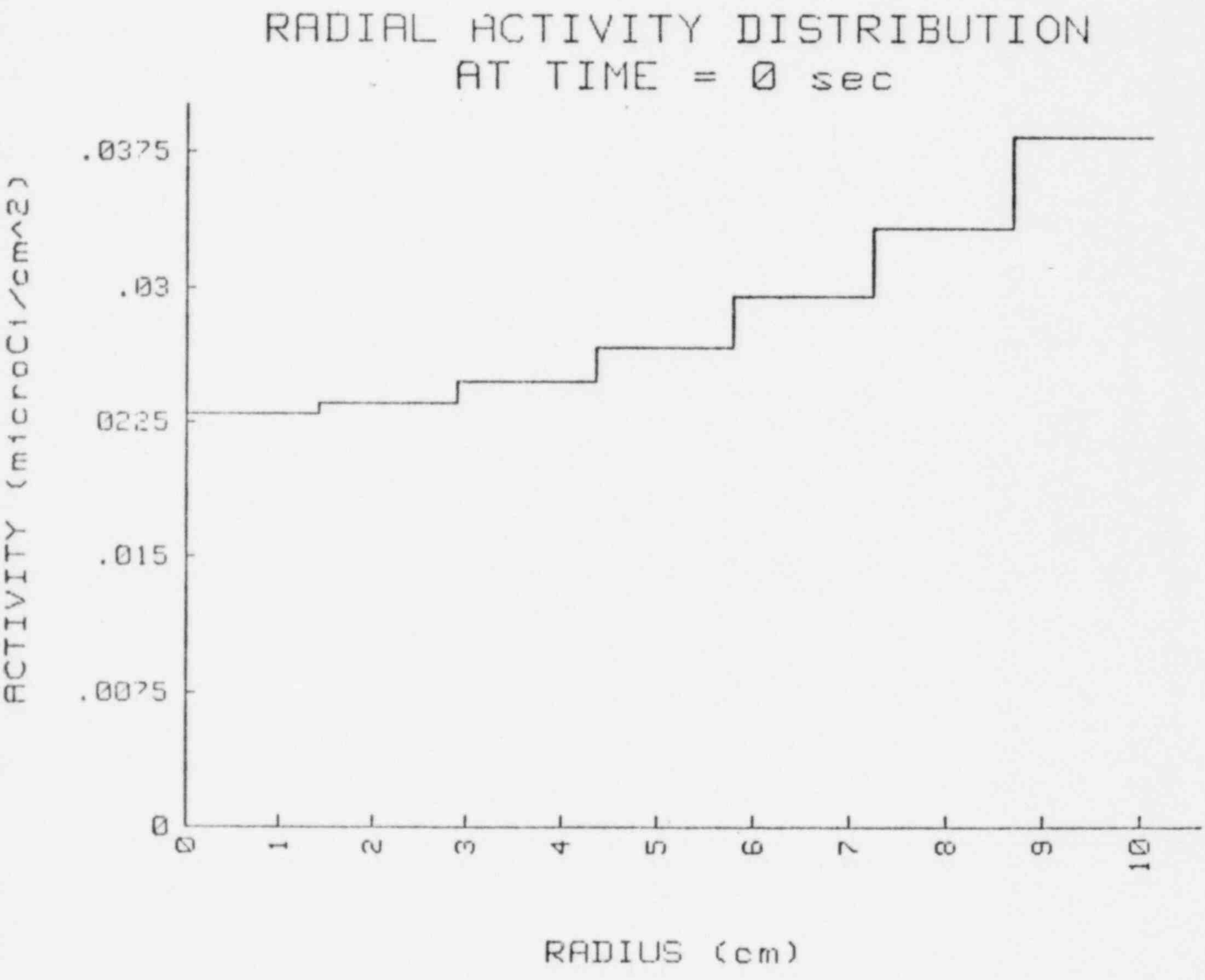
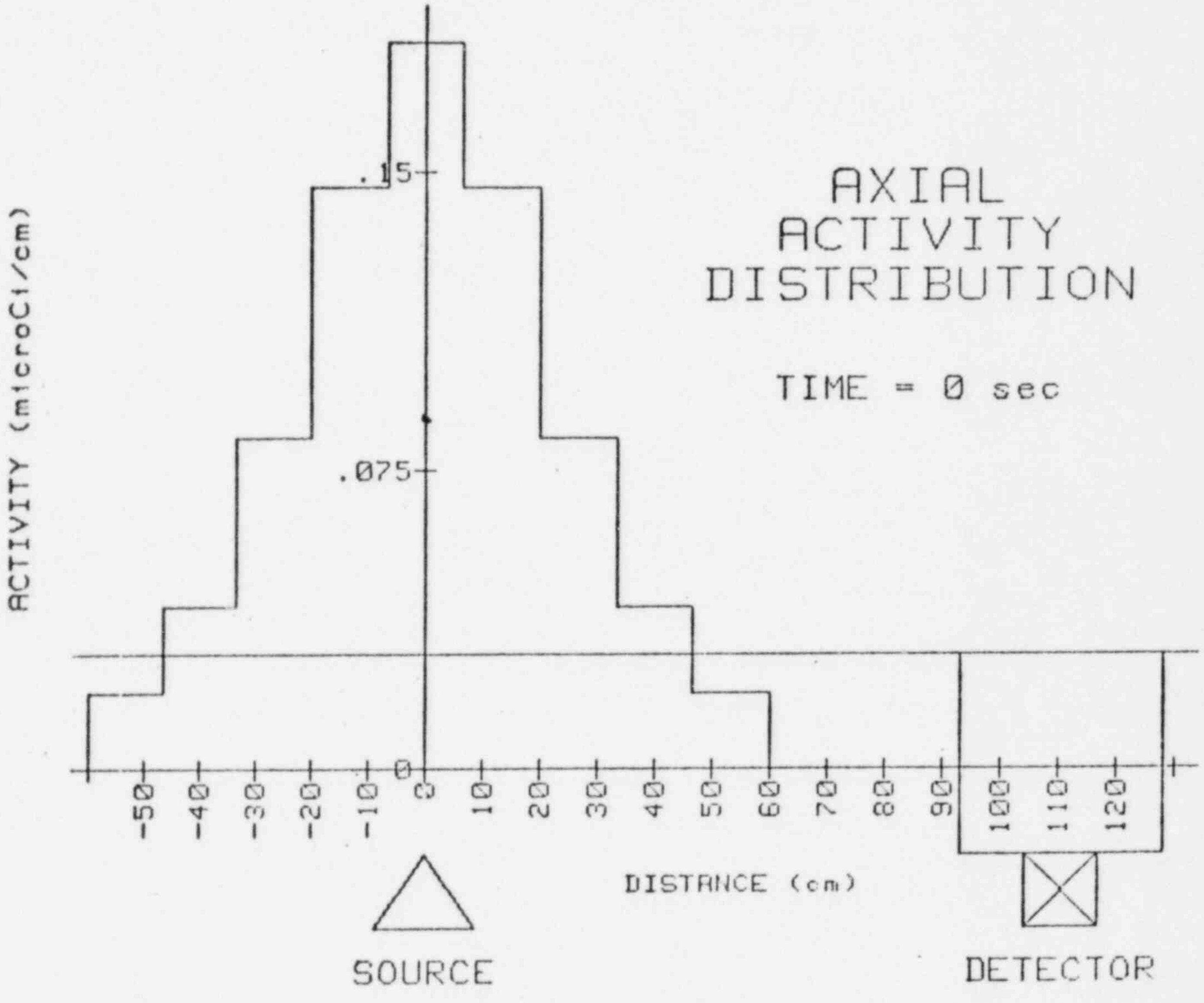


FIG. 6:

AXIAL ACTIVITY DISTRIBUTION IN THE DOWNCOMER OF THE PKL



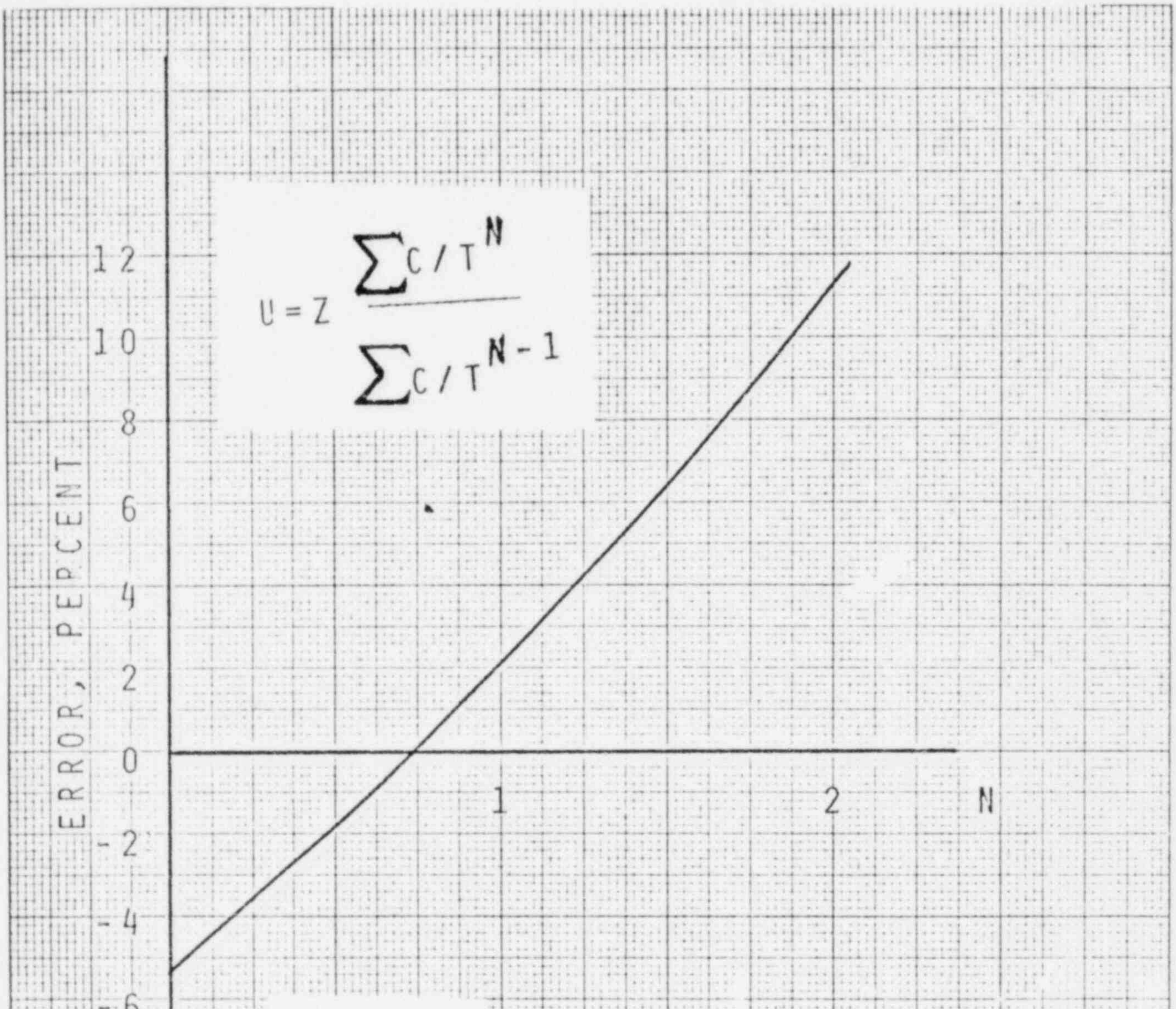


FIG. 7:

ERROR OF
PNA VELOCITY MEASUREMENT
AT THE PKL

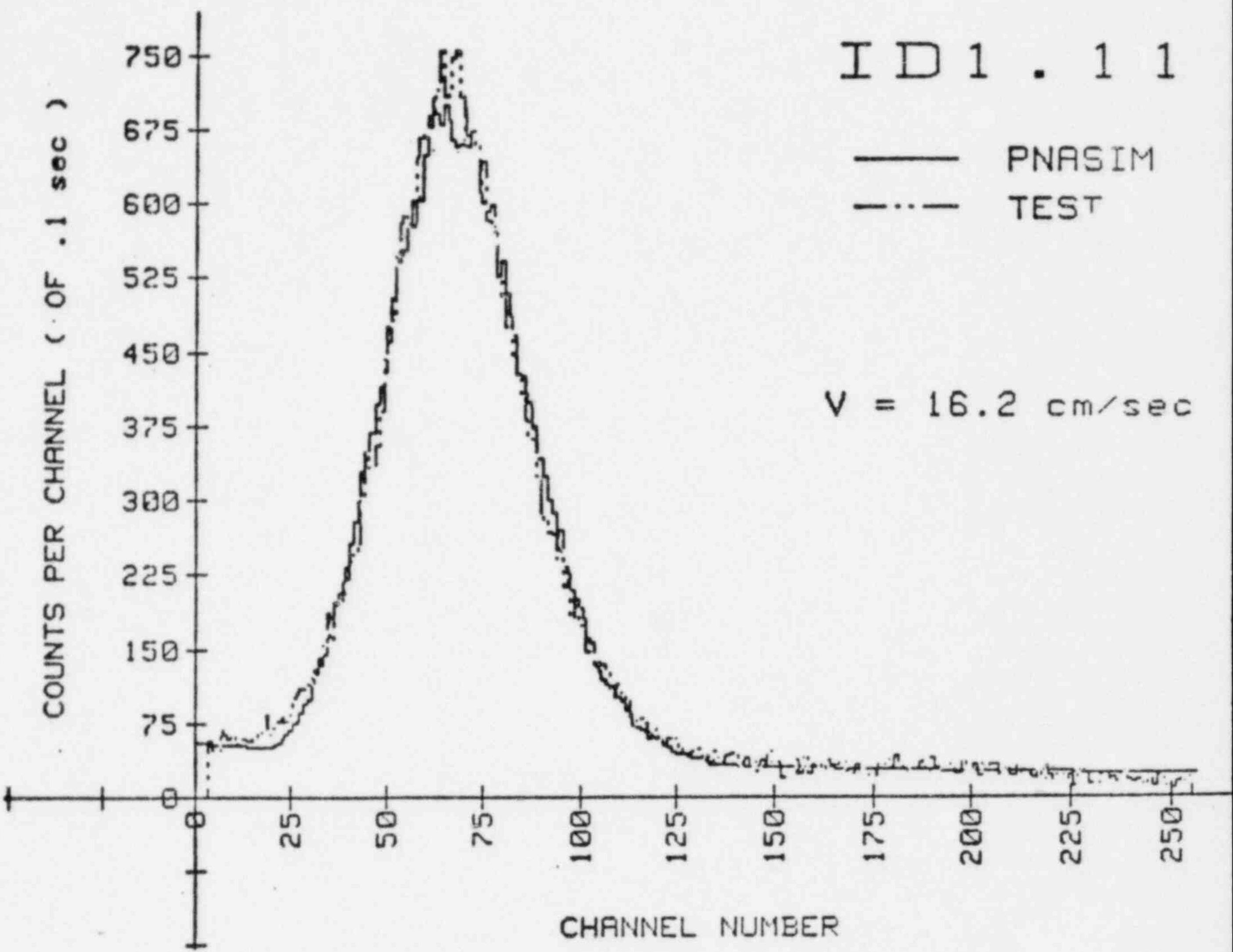


FIG. 8:

COMPARISON OF PREDICTED DISTRIBUTION TO DATA MEASURED AT THE PKL

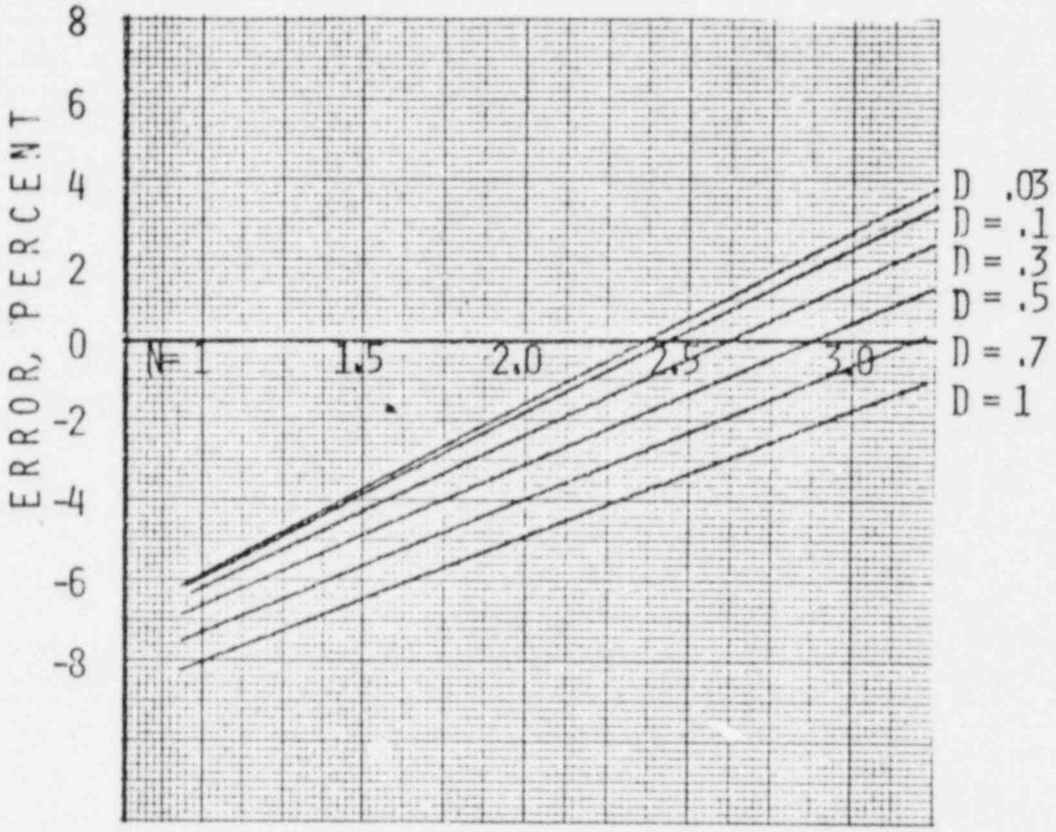


FIG. 9:

ERROR OF PNA VELOCITY MEASUREMENT AT THE FAST LOOP

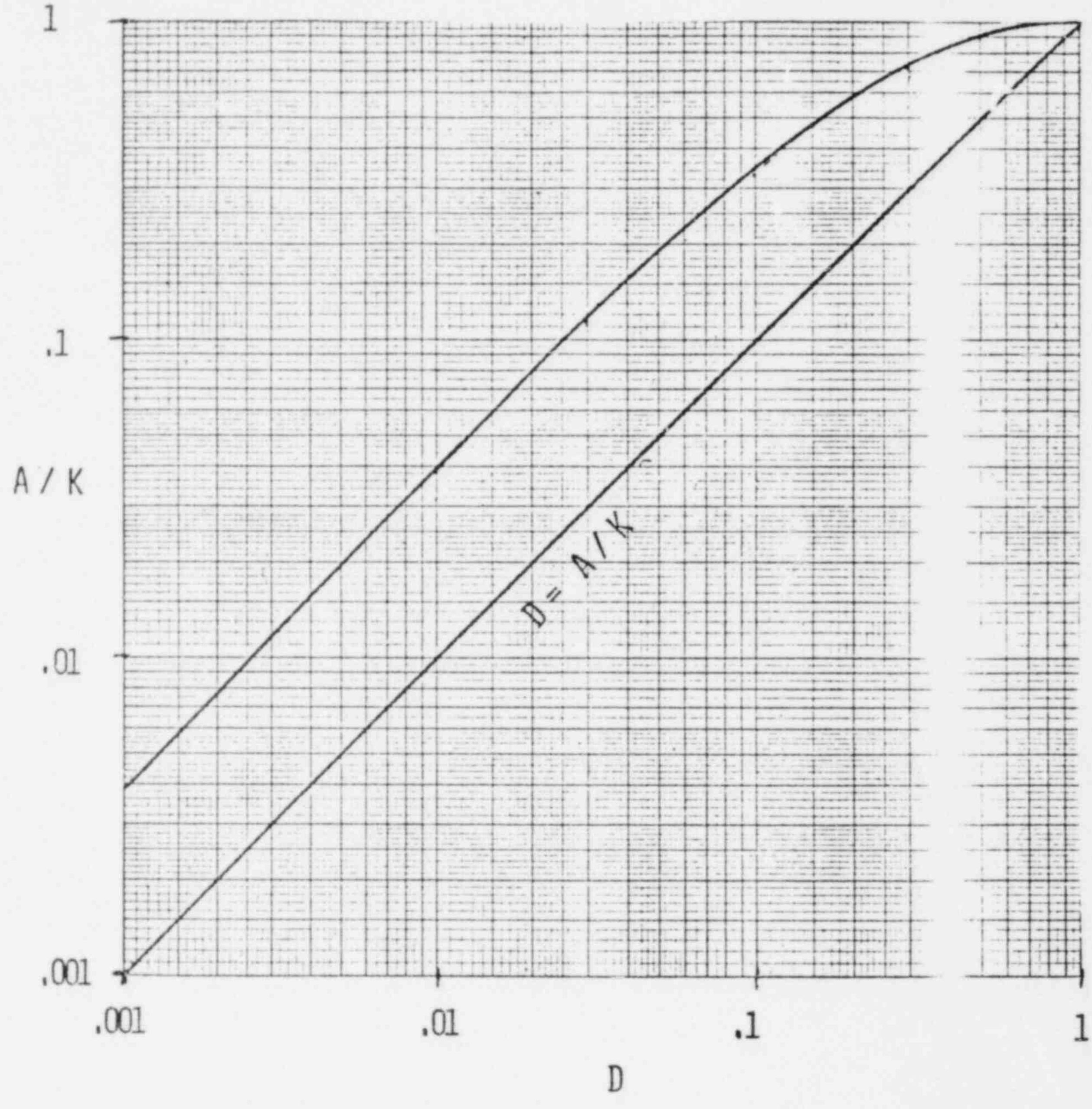


FIG. 10:

CORRECTION OF PNA DENSITY READINGS AT THE FAST LOOP

FIG. 11:

PNA DENSITY MEASUREMENT

- IN PIPES WITH DIAMETERS LESS THAN 10 cm, GLOBAL DENSITY CAN BE MEASURED WITHOUT BEING EFFECTED BY VARIOUS FLOW REGIMES

- FOR PIPES WITH DIAMETERS LARGER THAN 10 cm, GLOBAL DENSITY MEASUREMENTS MUST BE CORRECTED BY ANALYTICALLY DERIVED FUNCTIONS

- GLOBAL DENSITY MEASUREMENTS BY THE PNA TECHNIQUE CAN BE SUPPORTED BY OTHER GLOBAL DENSITY MEASUREMENTS PERFORMED WITH PNA EQUIPMENT

FIG. 12:

MEASUREMENT OF FLOW DISTRIBUTIONS IN LARGE TEST FACILITIES

- DEMONSTRATION AT THE LOFT FACILITY
- TESTS PLANNED AT THE SCTF IN 1981

FIG. 13:

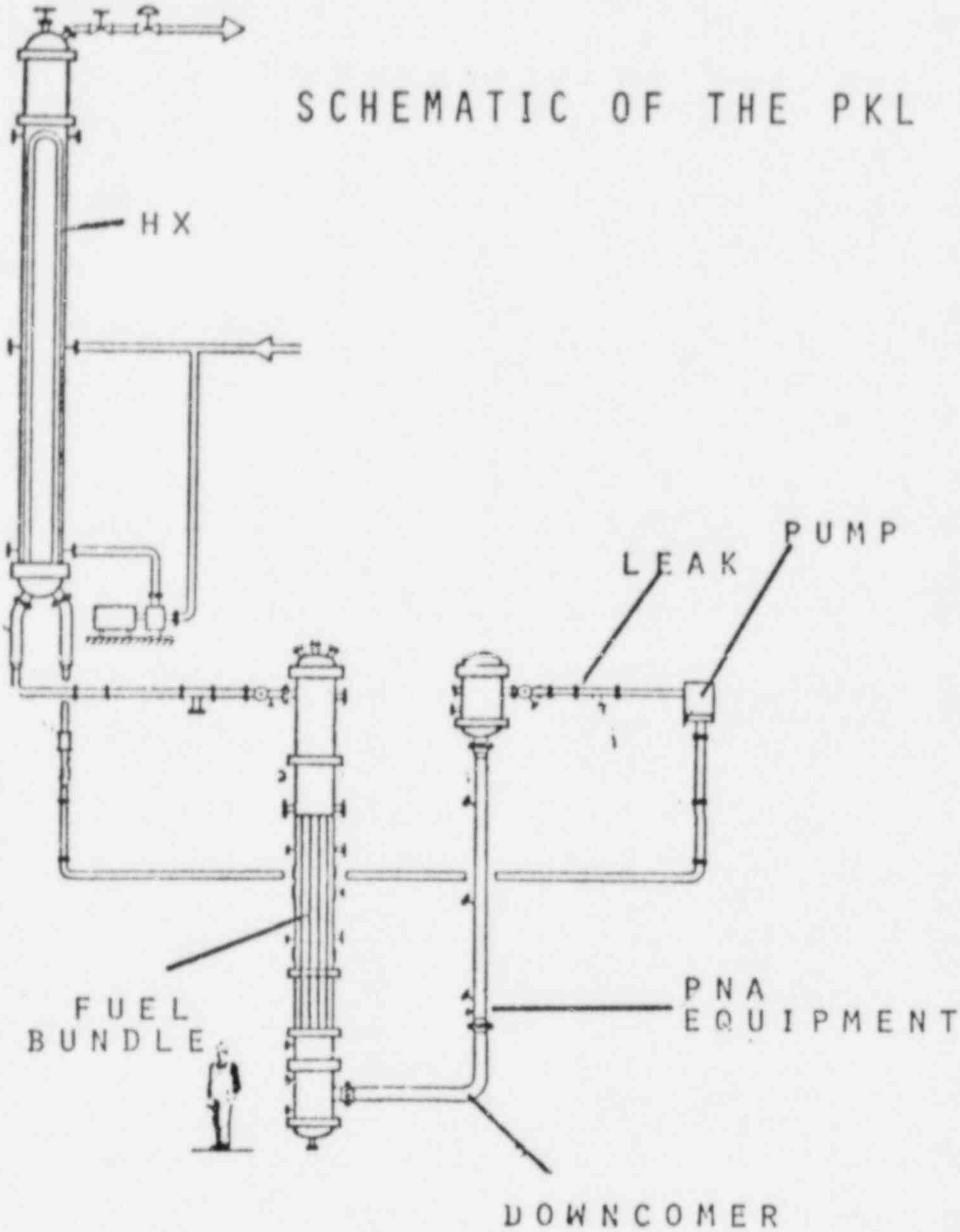
MEASUREMENT OF SLOW FLOW VELOCITIES

- DEMONSTRATED AT THE PKL (ERLANGEN, GERMANY)

- DEMONSTRATED AT THE LOFT FACILITY (EG&G IDAHO, INC.)

FIG. 14:

SCHEMATIC OF THE PKL



A PULSED NEUTRON GENERATOR FOR USE WITH PULSED NEUTRON ACTIVATION TECHNIQUES

SAND80-1902C

Gary E. Rochau
Generator Development Division 2351
Sandia National Laboratories
Albuquerque, New Mexico 87185

A high-output, transportable, pulsed neutron generator has been developed by Sandia National Laboratories for use with Pulsed Neutron Activation (PNA) techniques. The PNA neutron generator generates $>10^{10}$ 14 MeV D-T neutrons in a 1.2 millisecond pulse.

The PNA generator has been designed to meet the requirements listed in Table I. Each operation of the unit will produce a nominal total neutron output of 1.2×10^{10} neutrons. The generator has been designed to be easily repaired and modified. The unit requires no additional equipment for operation or measurement of output. A more complete description of the generator is given in References 1 and 2.

The generator has a minimal operational life of 1000 pulses and can be pulsed up to 12 pulses/minute for 1 minute. After 1 minute, the neutron output falls below 10^{10}. Pulses every 30 seconds can be repeated continuously. High repetition rates (>6 pulses/minute) can cause damage to the neutron tube and shorten the operational life of the generator. When the end of operational life is reached, the neutron tube will require minor servicing to restore the neutron output.

The PNA neutron generator has been segmented into three major component assemblies. Each major assembly contains the individual components which are required to operate the generator. These assemblies, interconnected by up to 60-meter cables, allow the experimenter to conveniently place them in the available space of the experimental area. The major assemblies are shown in Figure 1. The major assemblies, from left to right, are: the Tube-Transformer Assembly (TTA), the Pulse-Forming Network (PFN) box, and the control unit.

The generator utilizes the millisecond pulse (MSP) neutron tube (Figure 2) which was specifically developed for this application. This unclassified tube utilizes a focused deuterium ion beam produced by a specially modified occluded gas ion source. The deuterium beam impinges a 100% tritium-loaded scandium tritide target to produce an isotropic distribution of neutrons with an energy of approximately 14 million electron volts.

The MSP neutron tube is housed in the TTA which is placed at the position where the source of neutrons is desired. The neutron-producing target is located 5.97 cm behind the front surface of the TTA. The TTA is enclosed in a stainless steel cylinder 32.4 cm in diameter and 66 cm long. The cylinder is pressurized to 345 kPa (50 psi) with sulfur hexafluoride gas to provide high-voltage insulation.

A neutron tube was constructed for evaluation of the final neutron generator design. The performance of this evaluation tube is described in Table II. Table II shows that the MSP tube design is capable of surpassing the 1000-operation specification. The design will probably surpass 3000 operations without difficulty. In addition to the evaluation tube, six similar neutron tubes have been constructed for use in neutron generators under construction.

Five neutron generators are being constructed for NRC programs using the PNA technique for flow measurement. An additional PNA neutron generator is under construction for use by Atomic Energy of Canada to investigate a new technique.

A neutron generator was completed in August 1979 to evaluate the final design. This unit was tested to measure the neutron flux distribution (Figure 3) and radiation dose rates (Table III) and to evaluate the neutron monitor (Table IV and Figure 4).

The neutron flux distributions and radiation dose rates were measured to determine an adequate shield design. The flux distribution is essentially uniform over the front surface of the generator and is symmetric. The radiation dose was measured using LiF thermoluminescent dosimeters.

The neutron monitor was evaluated for linearity and gain stability using a lead activation probe, a secondary standard. The gain of the neutron monitor was adjusted so that the digital display would read total neutron output.

The performance of the evaluation neutron generator made it possible to use the unit in the ID test series conducted at the PKL Test Facility of Kraftwerk Union in Erlangen, West Germany, in February 1980.

The neutron generator was installed on the downcomer of the PKL test loop at the location indicated in Figure 5. Biological shielding was required for the generator to reduce radiation exposure to personnel in the area during generator operation. The 4-tonne shield (Figures 6 and 7) was designed for a minimum thickness of 43 cm of polyethylene around the TTA to keep the dose at 1 meter from the source below 4 rem for 2500 pulses. The TTA was enclosed in a cooling jacket inside the shield to keep the TTA temperature below 38°C.

The PFN box was located 6 m from the TTA in an isolated area (Figure 8). The control unit was located 42 m from the TTA location in the control room of the test facility with the other PNA equipment (Figure 9).

This was the first fielding of the PNA neutron generator, and it performed according to specifications. The unit produces a measured average neutron output of 1.2×10^{10} neutrons/pulse and a standard deviation of 3%. The presently installed generator is expected to perform above the specification level throughout the ID test series. After completion of the series, the generator will be returned to Sandia for evaluation testing.

References

1. R. C. Dougherty, G. E. Rochau, R. W. Bickes, Jr., R. J. Walko, and R. S. Berg, "Neutron Generator For Two-Phase Flow Calibration: Annual Progress Report," NUREG/CR-0480, SAND78-2030, November 1978.

2. G. E. Rochau, "Development of a Pulsed Neutron Generator For Two-Phase Flow Measurement," Review Group Conference on Advanced Instrumentation For Reactor Safety Research, NUREG/CP-0007, November 1979.

Table I
PNA NEUTRON GENERATOR

SPECIFICATIONS

Neutron Output:	$>10^{10}$ Neutrons/Pulse
Pulse Duration:	1.2 Milliseconds
Pulse Repetition Rate:	$\leq$12 Pulses/Minute
Lifetime:	$\geq$1000 Pulses (Low Repetition Rate) $\geq$100 Pulses (High Repetition Rate)
Life-Limiting Mechanism:	Neutron Tube
Exclusive Lifetime:	$\geq$10,000 Pulses
Neutron Monitor:	Integral Part of Generator. Sensitive to High Neutron Fluxes, Insensitive to Experimental Geometry.

FEATURES

Repairability:	Completely Demountable
Flexibility:	Easily Modified
Portability:	Two People Required
Safety:	Cannot be Triggered Accidentally
Completeness:	Requires AC Line Power Only
Availability:	Most Parts Commercially Available

Table II
PERFORMANCE OF EVALUATION NEUTRON TUBE

Serial Number:	MSP83C
First Tested:	August 24, 1979
Beginning Output:	$1.3 \times 10^{10}/0.13$
Present Output:	$1.27 \times 10^{10}/0.08$
Source Operations:	3117
Tube Operations:	1968

Performance Parameters:

	Source Current	Accelerator Voltage	Accelerator Current	Target Current	Neutron Rate	Neutron Efficiency
Initial	79.2 A	143 kV	50 mA	213 mA	10^7	49×10^6
Present	84.6 A	142 kV	69 mA	211 mA	10^7	48×10^6

Table III

RADIATION DOSE MEASUREMENTS AT 1 METER

	Dose for 20 Pulses (mR)		Dose/Pulse (mR)	
Angle	Neutron	Gamma	Neutron	Gamma
0	350	6	17.5	0.3
45	320	5	16.0	0.25
90	270	5	13.5	0.25
135	330	4	16.5	0.20
180	160	2	8.0	0.10
225	340	6	17.0	0.30
270	290	5	14.5	0.25
315	390	6	19.5	0.30

Table IV

PNA NEUTRON MONITOR

Detector: Quantrad 600-PIN-RM Silicon Photodiode With
1.14 mm Thick Proton Radiator

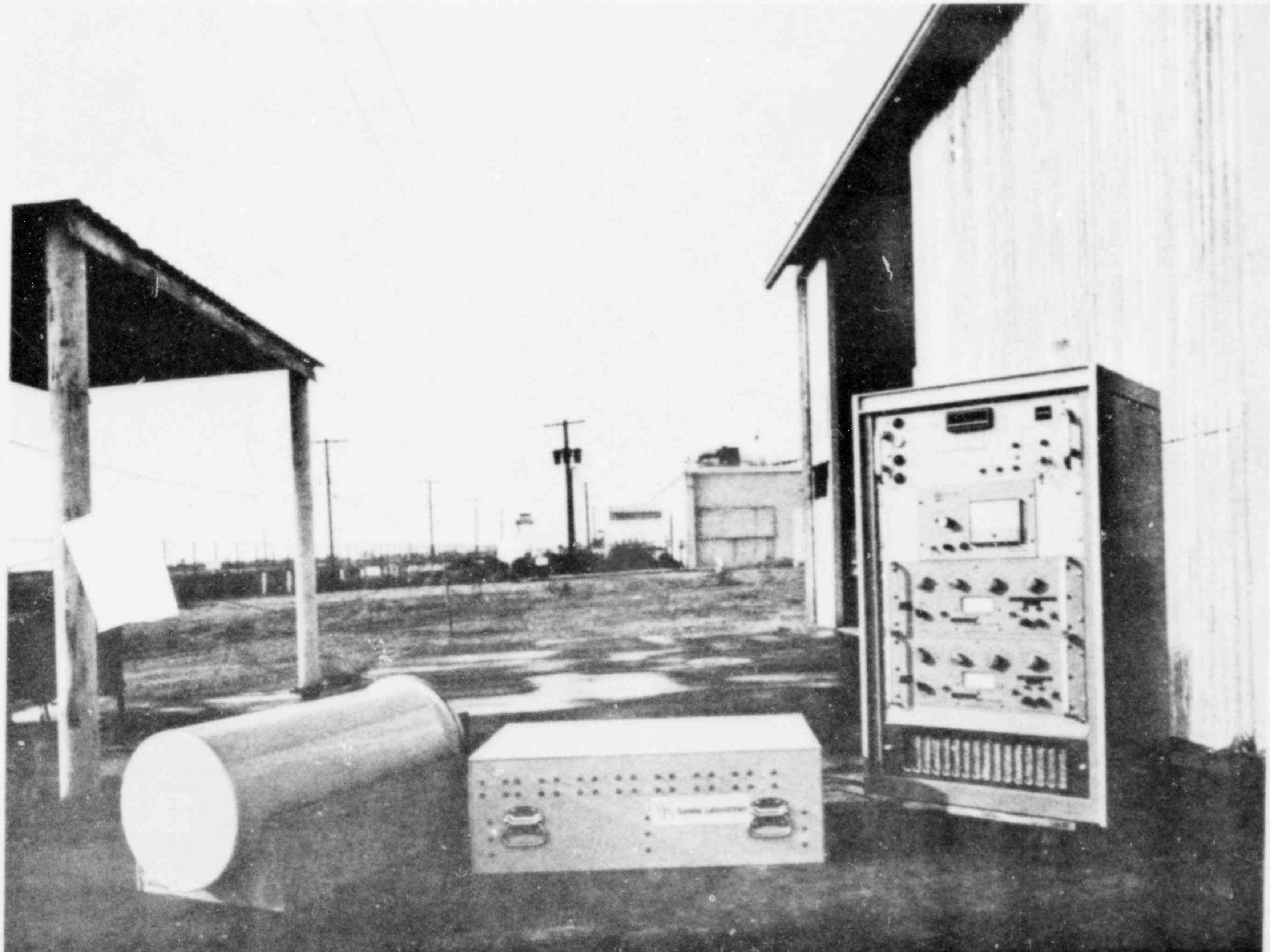
Location: Inside TTA, 12 cm From Neutron-Producing Target

Operation Mode: Integration of Detector Current

Calibration: ± 2% of Secondary Standard (Lead Probe)

Features: LED Self-Check
DVM Readout With BCD Digital Readout
Last Reading Held Until Next Operation

Figure 1



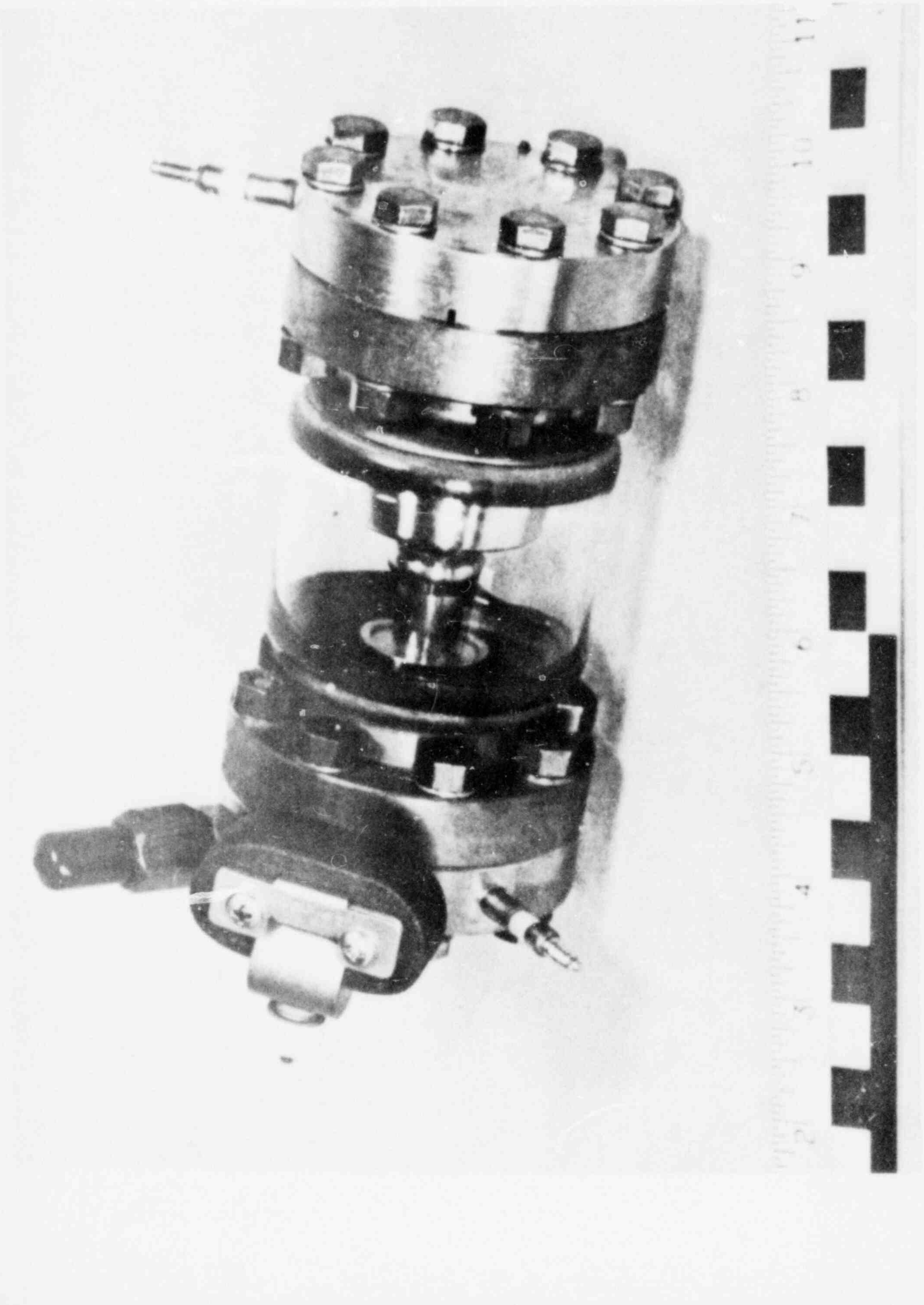
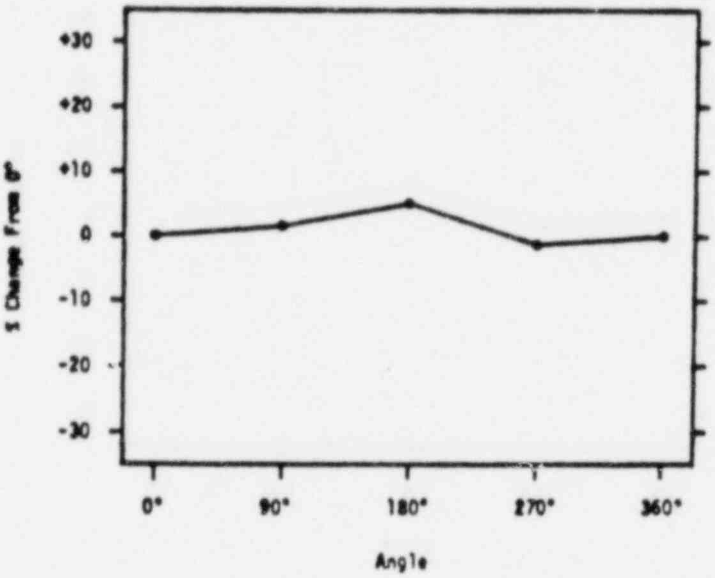
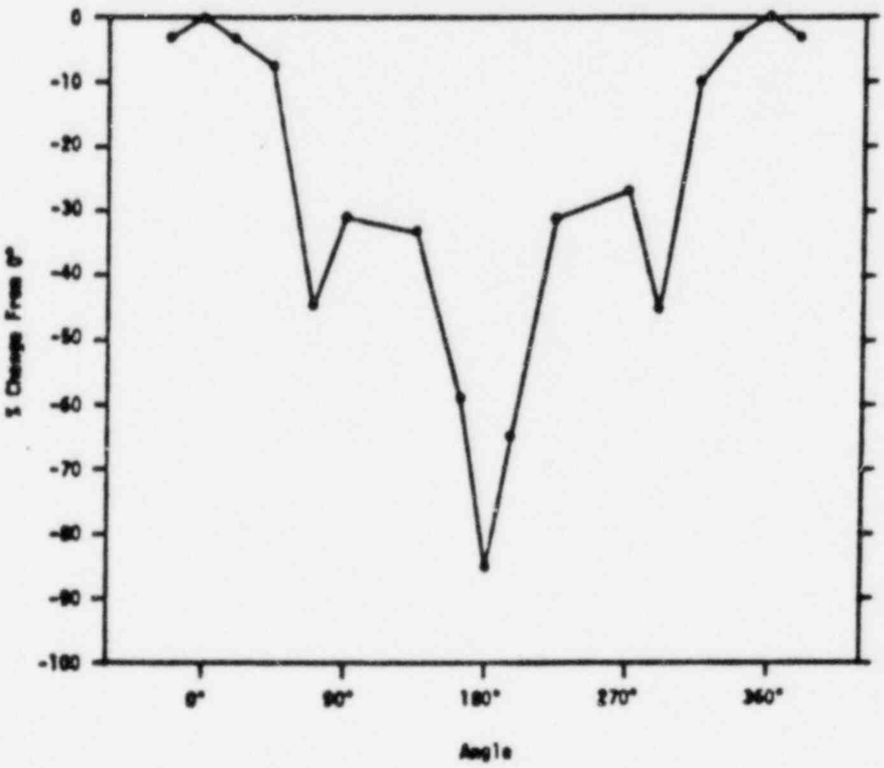


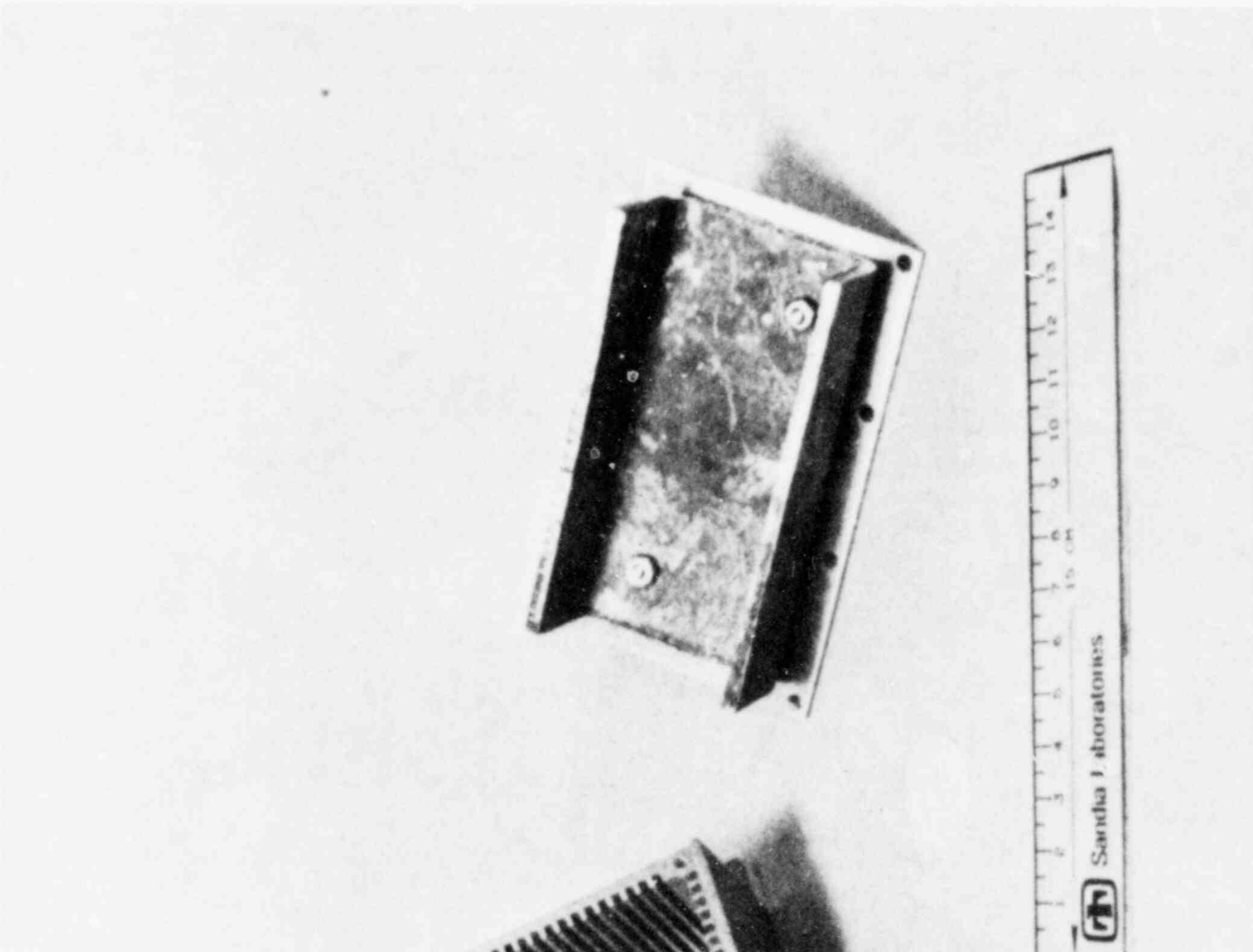
Figure 3

RELATIVE NEUTRON FLUX DISTRIBUTION
TTA ROTATED ABOUT TTA AXIS

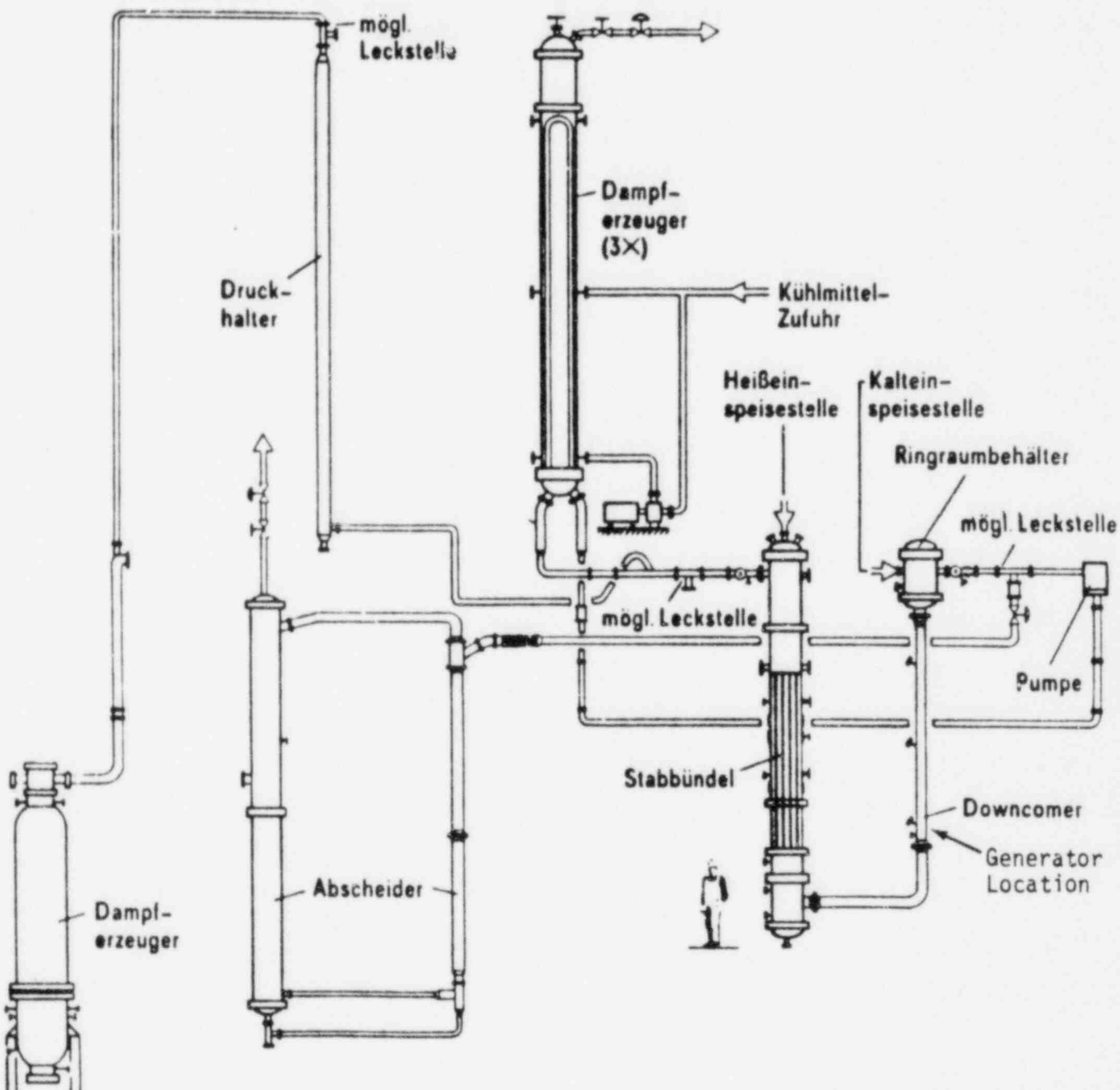


RELATIVE NEUTRON FLUX DISTRIBUTION
TTA ROTATED ABOUT TARGET CENTER




Sandia Laboratories

Kraftwerk Union



Modifikation der Anlage

PKL – Versuche mit kleinen Lecks

E 80 297 C

Figure 5

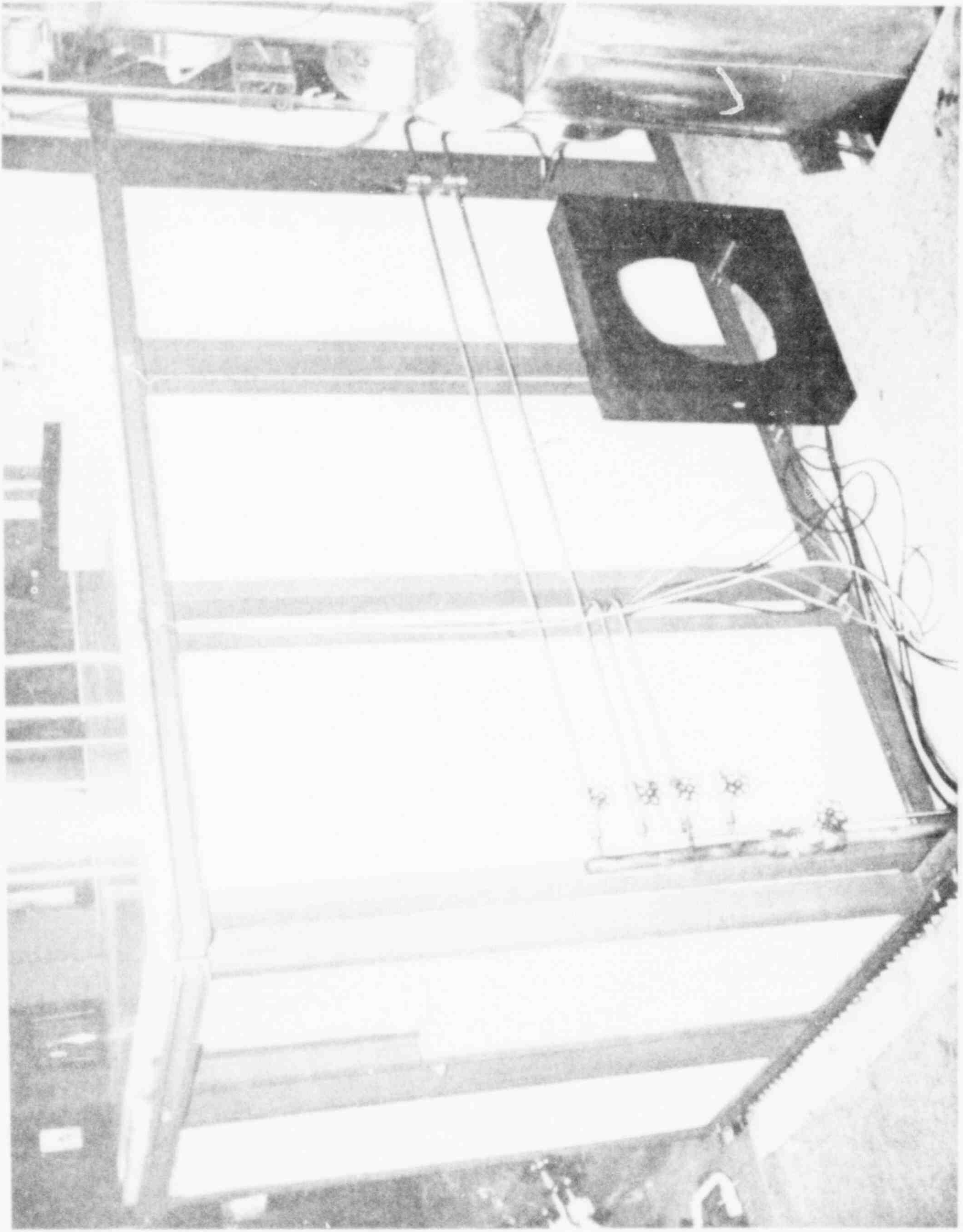


Figure 8

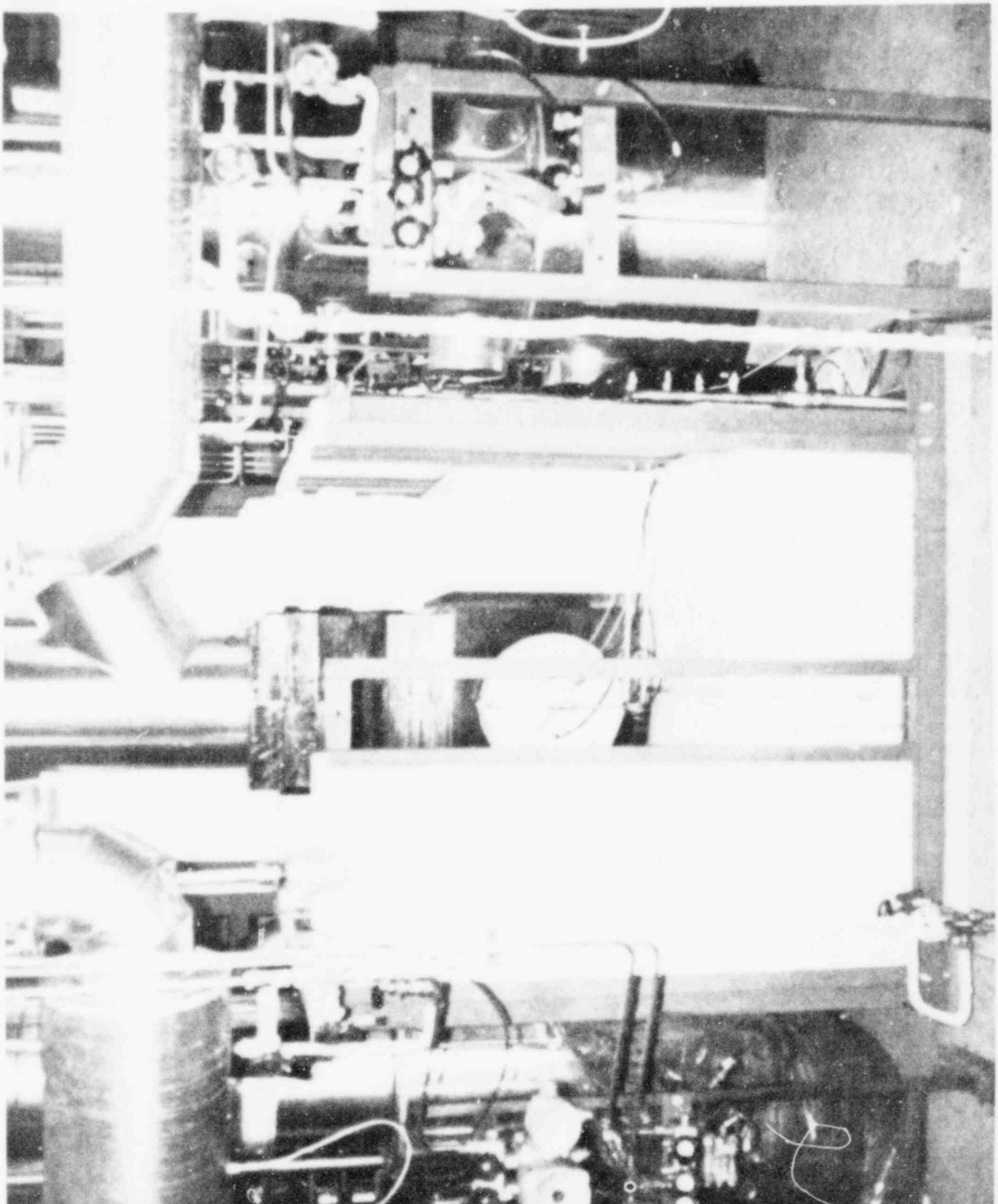


Figure 7

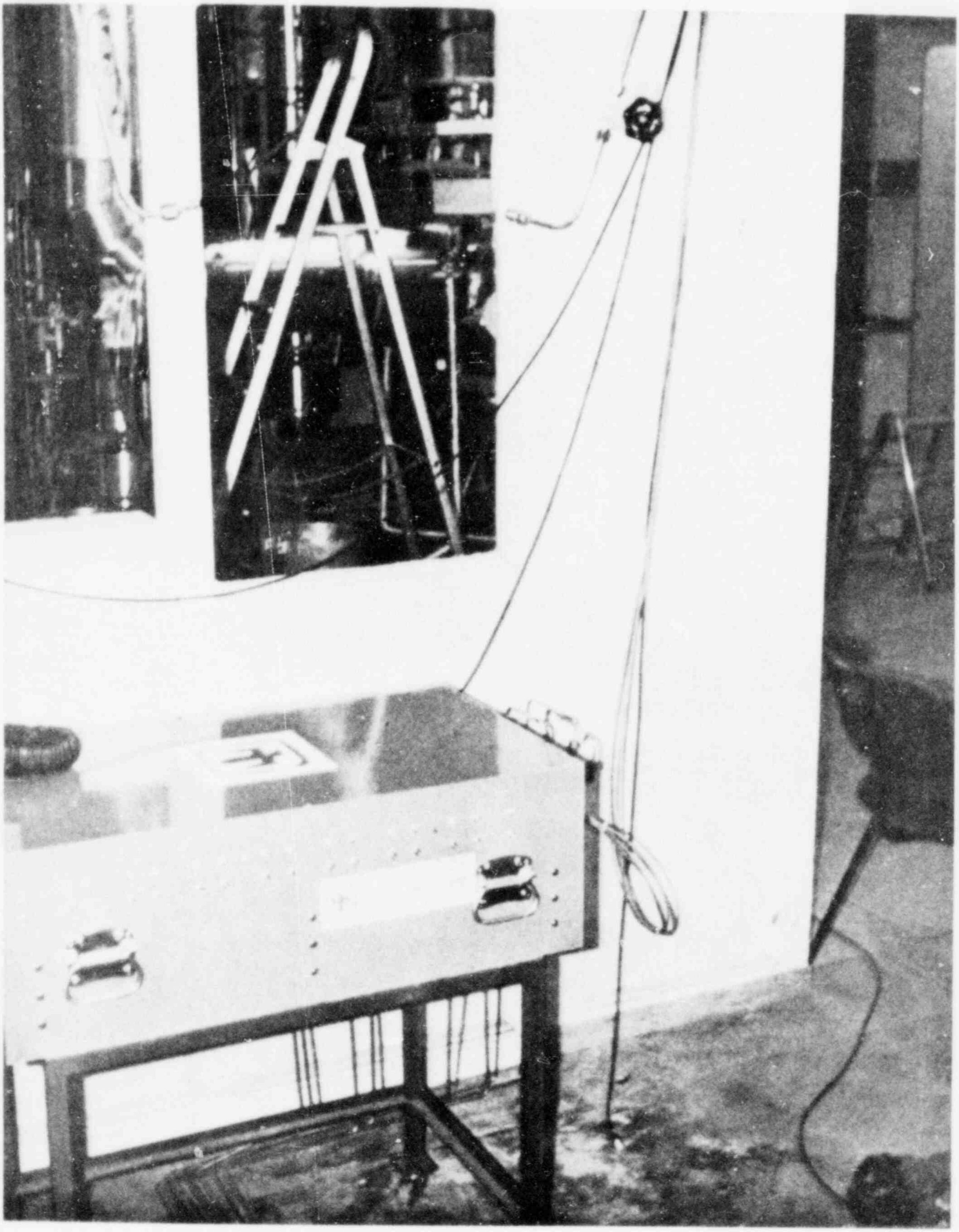


Figure 8

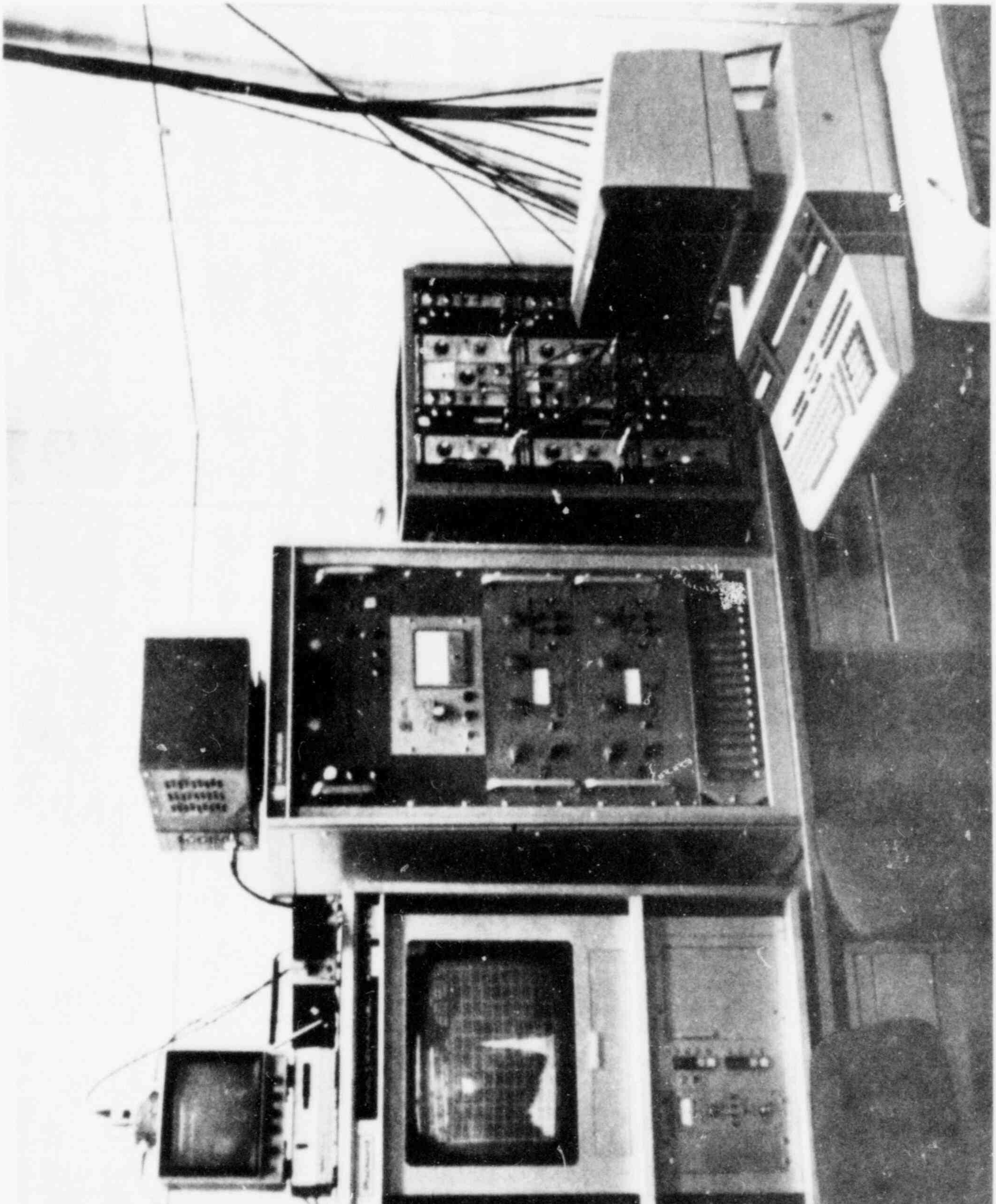


Figure 9

DEVELOPMENT AND EVALUATION OF PWR VESSEL LIQUID LEVEL INSTRUMENTATION AT ORNL*

K. G. Turnage

Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

Presented at the Eighth Water Reactor Safety Research
Information Meeting, October 28, 1980
Gaithersburg, Maryland

The extensive damage that occurred to the reactor at the Three Mile Island accident attests to the fact that more reliable means are needed for detecting inadequate core cooling conditions in PWRs. Thermal and acoustic coolant (level) sensors proposed for use in PWR reactor vessels are being tested under conditions that simulate the thermal and hydraulic conditions of a postulated PWR loss-of-coolant accident. Both natural convection (pumps off) and forced convection (pumps on) two-phase flow tests have been run. The goals are to evaluate the design of the coolant sensors and to determine whether there are conditions under which ambiguous indications of the degree of core cooling might occur.

The thermal devices tested use pairs of K-type thermocouples (TCs) or resistance temperature detectors (RTDs) to sense the cooling capacity of the medium surrounding the device. One of the TCs or RTDs is heated by an electric current passed through a separate wire; the other is primarily influenced by the bulk fluid temperature. The difference between the temperatures of the heated and unheated points (ΔT) is monitored to compensate for variations in the system fluid temperature. For a given heater power, with good cooling conditions (liquid or rapidly flowing two-phase mixtures), the ΔT is relativey low; with poor heat transfer (e.g., stagnant steam), the temperature at the heated junction increases greatly, making the ΔT higher.

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*Research sponsored by Division of Reactor Safety Research, U.S. Nuclear Regulatory Commission under Interagency Agreements DOE 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

A 13-cm-ID steel pressure vessel was used to perform steady-state, natural convection experiments with the devices in saturated water and steam. Pressures in the system were controlled from 0.1 to 10 MPa (14.7 to 1500 psia). In the tests performed to date, the ΔTs for the uncovered state have been significantly greater than those for the covered state. The tests also showed that condensation and deentrainment of liquid on unshielded probes can be a serious problem, particularly at higher pressure. Experiments designed to evaluate several designs of droplet shields with thermal coolant probes are in progress.

A differential heated TC was installed and tested during recent film-boiling experiments in the Thermal Hydraulic Test Facility (THTF) at ORNL. The sensor correctly indicated poor cooling prior to and during rod burdle DNB at pressures from 4.1 to 12.4 MPa (600 to 1800 psi) with outlet flow velocities up to ~3 m/s (10 ft/s). It failed to show inadequate cooling while parts of the rod bundle were in DNB at some higher outlet velocities.

PUBLICATIONS

K. G. Turnage et al., Advanced Two-Phase Flow Instrumentation Program Quarterly Progress Report for October-December 1979, NUREG/CR-1346 (ORNL/NUREG/TM-382), pp. 1-14 (May 1980).

K. G. Turnage et al., Advanced Two-Phase Flow Instrumentation Program Quarterly Progress Report for January-March 1980, NUREG/CR-1647 (ORNL/NUREG/TM-403), (September 1980).

UNION CARBIDE ORNL

PRESENTATION OUTLINE

- OVERVIEW ORNL LIQUID LEVEL DETECTOR PROGRAM
- DEVELOPMENT/EVALUATION THERMAL-TYPE SENSORS
- DEVELOPMENT/EVALUATION ACOUSTIC-TYPE SENSORS
- TEST PLANS

UNION CARBIDE ORNL

PWR LIQUID LEVEL INSTRUMENTATION: A MEANS FOR DETECTING THE APPROACH TO OR THE EXTENT OF INADEQUATE CORE COOLING

DESIRABLE CHARACTERISTICS:

- USEFUL RESPONSE UNDER STAGNANT BOILOFF OR HIGH VOID FRACTION FLOW
- RELIABLE – LONG LIFE, SURVIVE LOCA
- UNAMBIGUOUS – LITTLE OPERATOR INTERPRETATION REQUIRED. NO SPURIOUS INDICATIONS
- TIME RESPONSE – SECONDS

UNION CARBIDE ORNL

ORNL PWR LIQUID LEVEL INSTRUMENT DEVELOPMENT AND EVALUATION

OBJECTIVES:

- IDENTIFY PRACTICAL TECHNIQUES THAT CONFORM TO NRC REQUIREMENTS FOR MAKING IN-VESSEL LIQUID LEVEL MEASUREMENTS
  (LLTF REPORT (NUREG-0578), NRR LETTERS)
- PERFORM AND ANALYZE PROOF-OF-PRINCIPLE TESTING OF SENSORS UNDER "LOCA" THERMAL HYDRAULIC CONDITIONS
- IMPROVE SENSOR DESIGN WHERE APPROPRIATE
- COMMUNICATE WITH INDUSTRY AND NRC/RSR
- IDENTIFY POTENTIAL PROBLEM AREAS

UNION CARBIDE ORNL

SEVERAL IMPORTANT RELATED AREAS HAVE BEEN OUTSIDE THE SCOPE OF THE ORNL LIQUID LEVEL INSTRUMENTATION EFFORT

- DETERMINATION OF WHETHER EXISTING PWR INSTRUMENTATION IS ADEQUATE TO DETECT DEGRADED CORE COOLING CONDITIONS
- DETAILED FUNCTIONAL REQUIREMENTS FOR REACTOR VESSEL LEVEL INSTRUMENTATION
- DESIGN OF PROTOTYPES FOR USE IN PWRs
- FORMAL QUALIFICATION OF NEW INSTRUMENTATION

ORNL

ORNL's ADVANCED TWO-PHASE INSTRUMENTATION PROGRAM IS EVALUATING THREE TYPES OF LIQUID LEVEL INSTRUMENTATION FOR PWR USE

- THERMAL (HEATED TCs OR RTDs)
- ACOUSTIC (WAVEGUIDE/TRANSIT TIME)
- PRESSURE DIFFERENCE

ORNL

TEST VARIABLES

NATURAL CONVECTION TESTS

- MEDIUM
- AMBIENT TEMPERATURE/PRESSURE
- HEATER POWER
- ORIENTATION
- PROBE DESIGN

FORCED CONVECTION TESTS

- VOID FRACTION
- VELOCITY
- PRESSURE
- HEATER POWER
- GEOMETRY

ORNL

ORNL ATPI – LIQUID LEVEL SENSORS

ACCOMPLISHMENTS

- PERFORMED LITERATURE SEARCH – FEASIBILITY STUDY OF VARIOUS TECHNIQUES
- MODIFIED EXISTING FACILITY FOR NATURAL CONVECTION TESTS AT PRESSURES UP TO 1500 psi
- DEVELOPED IN-HOUSE FABRICATION CAPABILITY FOR HEATED JUNCTION THERMOCOUPLES
- BORROWED HTCs FROM NAVAL REACTORS. TESTED UNDER NATURAL AND FORCED CONVECTION
- OBTAINED CHARACTERISTIC CURVES (OUTPUT VS POWER, PRESSURE, MEDIUM) FOR SEVERAL DESIGNS OF HJTC IN NATURAL CONVECTION TEST FACILITY

ORNL

ORNL ATPI – LIQUID LEVEL SENSORS

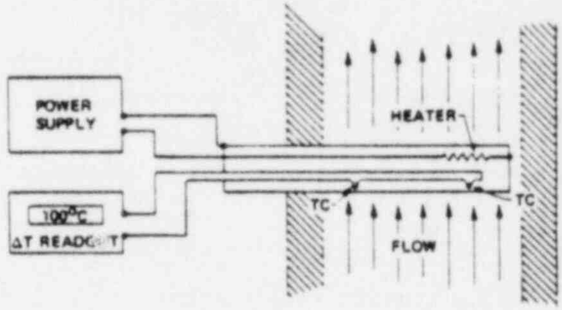
ACCOMPLISHMENTS (CONTINUED)

- PROCURED AND VERIFIED PRINCIPLE OF OPERATION OF RIBBON-TYPE TORSIONAL-EXTENSIONAL PROBE AT HIGH TEMPERATURES AND PRESSURES
- MET WITH B&W, C-E, W TO DISCUSS RLVMS DEVELOPMENT. ARE CONTINUING THIS INTERACTION
- IDENTIFIED NEED FOR SPLASH SHIELDS ON THERMAL PROBES IN BOTH NATURAL AND FORCED CONVECTION
- STUDIED OPERATION OF HJTCs IN THTF DURING RECENT FILM BOILING EXPERIMENTS

THERMAL SENSORS: ADVANTAGES

- CAN MEASURE COOLING CONDITIONS DIRECTLY
- RELATIVELY SMALL
- USE REACTOR-COMPATIBLE MATERIALS
- SIMPLE
- INEXPENSIVE

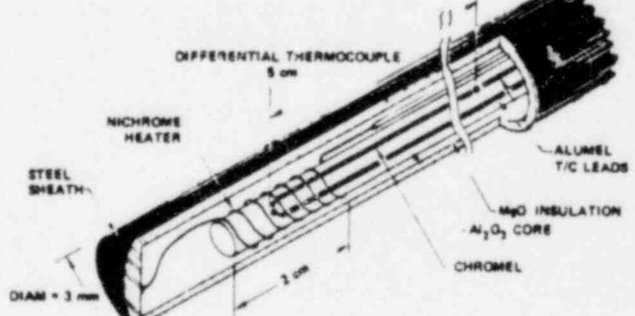
THERMAL SENSORS MEASURE COOLING CONDITIONS DIRECTLY



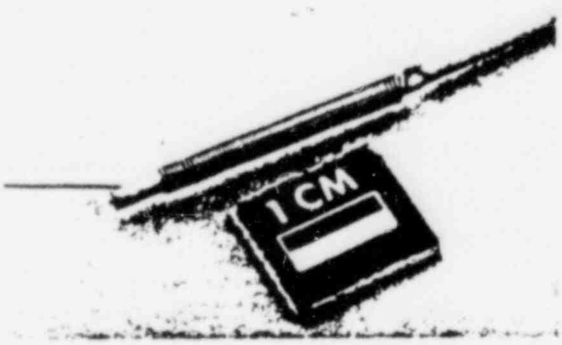
SEVERAL THERMAL-TYPE SENSORS HAVE BEEN TESTED

- NAVY-TYPE HJTC (2 KINDS)
- FCI HEATED RTD (2 DESIGNS)
- ORNL HTC (SEVERAL KINDS)
- OTHERS

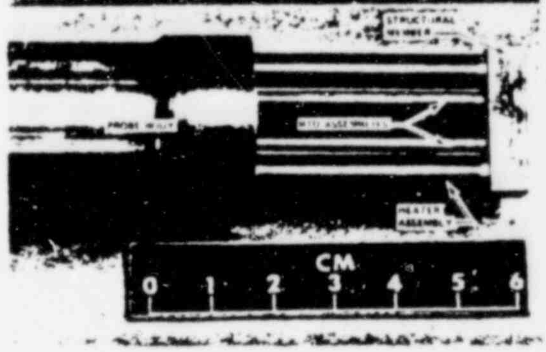
HEATED TC COOLANT SENSORS ARE SMALL AND USE REACTOR-COMPATIBLE COMPONENTS



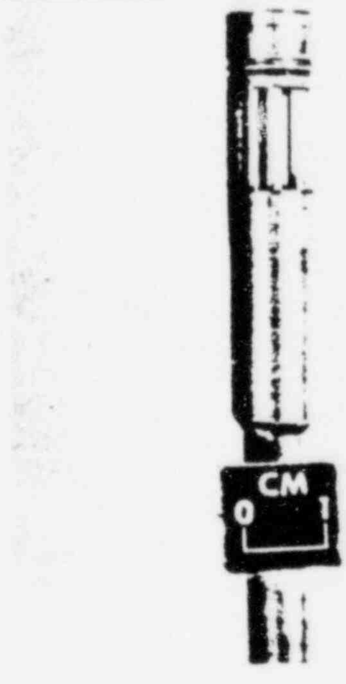
HEATED THERMOCOUPLE SENSOR WITHOUT OUTER SHEATH



A COMMERCIALLY-AVAILABLE HEATED RTD WAS TESTED IN THE NATURAL CONVECTION TEST FACILITY



A HEATED RTD THAT WAS TESTED IS SMALL ENOUGH FOR IN-BUNDLE INSTRUMENT TUBE



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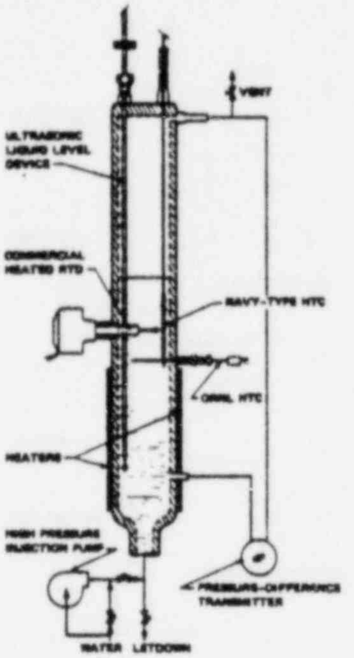
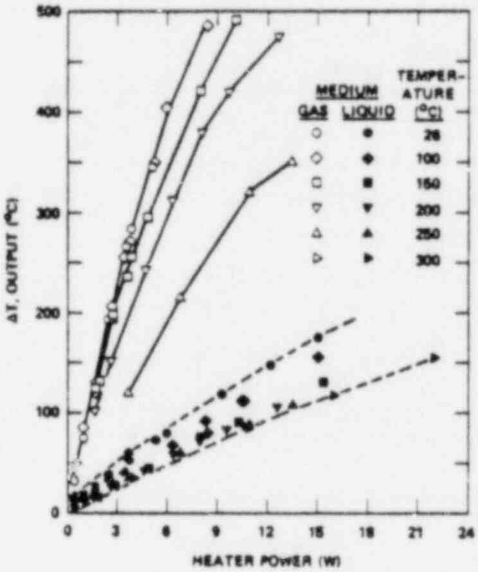
SOME DISADVANTAGES OF THERMAL TYPE SENSORS ARE

- DISCRETE INDICATION – ARRAYS NEEDED
- CAN BE AFFECTED BY ENTRAINED LIQUID ABOVE LIQUID-VAPOR INTERFACE
- MAY NOT BE USEFUL DURING NORMAL CONDITIONS (DIFFERENTIAL TC)
- HEATER CAN FAIL

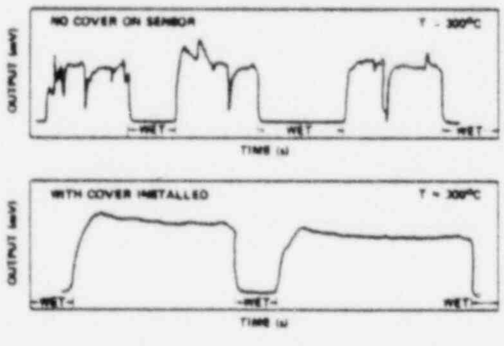
A PRESSURIZER WAS USED TO PERFORM HIGH PRESSURE NATURAL CONVECTION TESTS WITH SEVERAL DEVICES



OUTPUT FROM NAVY-TYPE HTC PROBE IN NATURAL CONVECTION TO STEAM AND WATER CLEARLY INDICATES MEDIUM PRESENT



THERMAL-TYPE LEVEL SENSORS SHOULD BE SHIELDED FROM SPLASHING BY ENTRAINED LIQUID



A STANDPIPE ARRANGEMENT MAY PROTECT SENSOR FROM SPLASHING AND ALLOW ADEQUATE TIME RESPONSE

CHARACTERISTICS REQUIRE INVESTIGATION

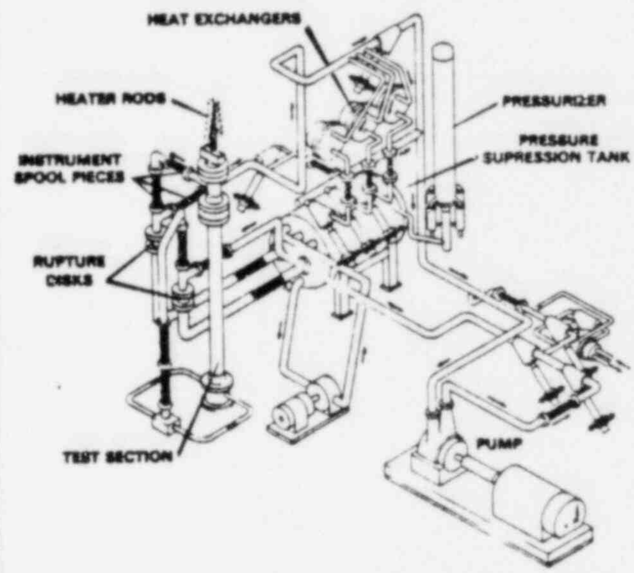
- DOES IT INDICATE COLLAPSED LIQUID LEVEL?
- WHAT ARE EFFECTS OF VELOCITY?
- HOW LARGE SHOULD DRAIN AND VENT HOLES BE?

PERFORMANCE OF HTCS UNDER VARIOUS LOCA CONDITIONS ARE BEING STUDIED WITH THE THERMAL HYDRAULIC TEST FACILITY (THTF) AT ORNL

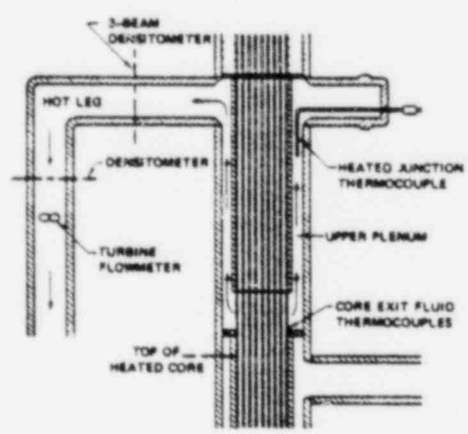
THTF

- A HIGH PRESSURE, SINGLE-LOOP SEPARATE EFFECTS LOCA EXPERIMENT
- 8 X 8, 12-ft ACTIVE HEIGHT, ELECTRICALLY-HEATED CORE SIMULATOR WITH ~500 frs T/Cs
- CURRENTLY PERFORMING TESTS TO PROVIDE NRC-REQUESTED LOCA HEAT TRANSFER DATA
- TEST SECTION OUTLET WELL INSTRUMENTED FOR VOID FRACTION AND VELOCITY

BDHT EXPERIMENTS ARE RUN IN THE THTF



HEATED THERMOCOUPLE RESPONSE IN THTF UPPER PLENUM WAS RELATED TO RESPONSE OF TEST SECTION AND OUTLET PIPING INSTRUMENTATION



AN EXPERIMENTAL HTC SENSOR WAS EVALUATED DURING THTF STEADY STATE FILM BOILING EXPERIMENTS

- FLOW RATES AND CORE POWERS SIMULATED LOCKED ROTOR OR ROD EJECTION ACCIDENT
- SINGLE HTCs LOCATED IN UPPER PLENUM, NEAR TEST SECTION OUTLET
- OUTPUT ΔT MONITORED BEFORE, DURING, AND AFTER PERIODS OF FILM BOILING IN ROD BUNDLE
- RESPONSE OF TEST SENSOR RELATED TO FRS TEMPERATURES AND FLUID CONDITIONS AT TEST SECTION OUTLET

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TEST RESULTS FROM THTF FILM BOILING EXPERIMENTS WITH HJTC PROBE

- SENSOR OPERABLE AFTER >10 h AT LOCA CONDITIONS
- INDICATED POOR COOLING PRIOR TO AND DURING ROD BUNDLE DNB AT 600, 900, 1200 AND 1800 psi WITH OUTLET VELOCITIES UP TO ~10 fps
- FAILED TO SHOW INADEQUATE COOLING WHILE PARTS OF ROD BUNDLE WERE IN DNB AT SOME HIGHER OUTLET VELOCITIES

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CONCLUSIONS FROM TESTING OF THERMAL SENSORS TO DATE

- RELIABLE IN WET/DRY CONDITIONS IF PROPERLY SHIELDED
- AS TESTED CAN INDICATE POOR COOLING UNDER LOCA CONDITIONS IF VELOCITIES/ CORE POWER ARE RELATIVELY LOW
- ADDITIONAL DEVELOPMENT NEEDED TO DETECT POOR COOLING IF CORE POWERS AND FLOW VELOCITIES ARE HIGH

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FUTURE TEST OBJECTIVE:

A COMPREHENSIVE EVALUATION OF MOST PROMISING METHODS IN HIGH PRESSURE TWO-PHASE FLOW

TESTS WILL INVOLVE:

- REALISTIC GEOMETRIES
- FLUID CONDITIONS LIKE MOST IMPORTANT ACCIDENTS
- RELIABLE INDEPENDENT MEASUREMENTS OF LOCAL VOID FRACTION, VELOCITY
- BOTH RAW AND PROCESSED ("CONTROL ROOM") SENSOR OUTPUTS
- IN FLOW STREAM WITH HEATED ROD BUNDLE TO ALLOW EVALUATION OF SENSOR PERFORMANCE AS DETECTOR OF INADEQUATE COOLING

UNION CARBIDE ORNL

ATPI–LIQUID LEVEL SENSORS

NEAR TERM PROGRAM PLANS

- COMPLETE EVALUATION OF HJTC SENSORS IN THTF DURING REMAINING BUNDLE BOILOFF EXPERIMENTS
- STUDY STANDPIPE-TYPE SPLASH SHIELDS (ESPECIALLY LEVEL INSIDE SHIELD VS VOID FRACTION) IN AIR-WATER AND STEAM-WATER TEST FACILITIES AT ORNL
- CONSTRUCT SHIELDED HJTC ARRAYS. INSTALL AND TEST IN SEMISCALE AT INEL
- FABRICATE AND TEST HTC ELECTRONICS DEVELOPED BY J. V. ANDERSON (INEL)
- DEVELOP IMPROVED PACKAGING FOR ULTRASONIC SENSOR
- PARTICIPATE IN EVALUATION OF W ΔP SYSTEM AT INEL

EVALUATION OF HJTCs IN THTF DURING BUNDLE BOILOFF AND REFLOOD

- OCTOBER 1980
- DECAY HEAT CORE POWER
- SMALL BREAK THERMAL HYDRAULICS
- TWO DISTINCT SPLASH SHIELD DESIGNS CONCURRENTLY TESTED

THE INSTRUMENT DEVELOPMENT LOOPS AT ORNL ARE USED TO SIMULATE LOCA FLOW BEHAVIOR IN A REACTOR UPPER PLENUM

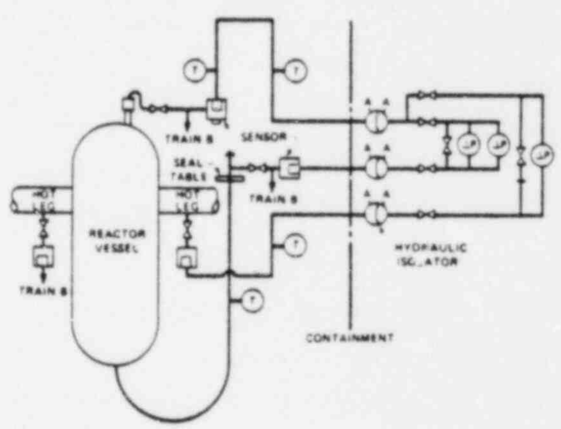
- INSTRUMENTS FOR LOCAL VOID FRACTION AND VELOCITY MEASUREMENTS INSTALLED AND WORKING
- REALISTIC GEOMETRIES
- AIR-WATER AND STEAM-WATER
- FLOW VISUALIZATION POSSIBLE

A HTC ARRAY AND A PRESSURE DIFFERENCE LIQUID LEVEL SYSTEM ARE TO BE INSTALLED IN SEMISCALE DURING FUTURE SYSTEM EFFECTS TESTS

OBJECTIVES:

- EVALUATION AT BEST ESTIMATE LOCA T-H CONDITIONS
- BEHAVIOR IN CONJUNCTION WITH FULL-LENGTH CORE
- TIME RESPONSE, RESOLUTION LIMITATIONS
- CONSIDERATION OF VARIETY OF ACCIDENT TYPES

A PRESSURE DIFFERENCE-BASED LEVEL SYSTEM DEVELOPED BY WESTINGHOUSE IS TO BE STUDIED USING THE SEMISCALE FACILITY



ULTRASONIC COOLANT LEVEL SENSOR

G. N. MILLER

Instrumentation and Controls Division

Oak Ridge National Laboratory

Oak Ridge, Tennessee 37830

Presented at

8th Reactor Reactor Research Information Meeting

National Burea of Standards

Gaithersburg, Maryland

October 28, 1980

ULTRASONIC COOLANT LEVEL SENSOR

The ultrasonic torsional wave level sensor has been proposed as a means of measuring liquid level in PWR's. The sensor element itself is a flattened metal rod. A torsional wave can be excited in the sensor element from outside the reactor vessel by a simple coil surrounding a magnetostrictive segment of the sensor element. The exciting coils can be located outside the pressure boundry and away from the severe environment inside the reactor vessel.

A disadvantage of the torsional wave sensor is the dependence of the velocity of propagation on the temperature of the sensor element. Tests were conducted at ORNL in a heated, pressurized water vessel to test the concept of using the temperature dependence of the velocity of the extensional wave to measure the temperature of the sensor and then using this information to correct velocity of the torsional wave for temperature. A probe was designed and fabricated to use both torsional and extensional waves on the same element, greatly simplifying the construction of the sensor.

This report documents the laboratory tests at ORNL of an ultrasonic sensor designed to locate a water/steam interface over the range of temperatures and pressures encountered in a PWR. The "active length" of the sensor consists of a 0.76 m long flattened wire, stainless steel 304, of 1 x 2 mm cross section.

Referring to Fig. 1, the entire waveguide was spring-tensioned and tied with fine wire to lie approximately along the axis of a 3 m long sheath of SS304 tubing, 10 mm OD x 1 mm wall. This tubing was perforated to allow temperature and water to equilibrate quickly.

The two coils are separated by approximately 10 cm so that each transducer can be biased and alternately driven to maximize the energy in each of the torsional or extensional modes, while minimizing spurious echoes from the other mode. When the Joule tranducer is energized to produce the extensional wave, electronic blanking is used to disregard spurious echoes. When the Wiedemann transducer is energized to produce the torsional mode, the blanking times are increased because of slower propagation of the torsional wave.

Figure 2 is a drawing of the test vessel used for evaluation of the probe. Electrical heaters on the lower half of the outside wall of the Pressurizer are used to heat the vessel and fluid to the desired temperature. The pressure containment volume included the portion of the probe wall outside the pressure vessel as can be seen from Fig. 2. A high pressure pump was used to raise the liquid level. The high temperature test was always run with the vessel full at start. Measurements were then made with decreasing fluid level. A calibration run was made at atmospheric pressure with level first increasing and then decreasing measure hysteresis. As can be seen from Figs. 3 and 4 there were less than 0.5% total error from nonlinearity and hysteresis at 66°C.

Figure 3 is a plot of actual data for the extensional and torsional round trip transit times as a function of temperature. It should be noted that the time data does not include the blanking interval.

Figure 4 is the plot of change in time vs level. The data for the reference curve in this figure was taken on this probe at 66°C. The ordinate is graduated in 100 nanosecond increments; this is the round trip transit time from the blanking interval. The total round trip transit time is obtained by adding the blanking interval. The 10% and 90% points on the differential pressure level measurement transmitter corresponded to the 0 and 72 cm points on the ultrasonic level sensor.

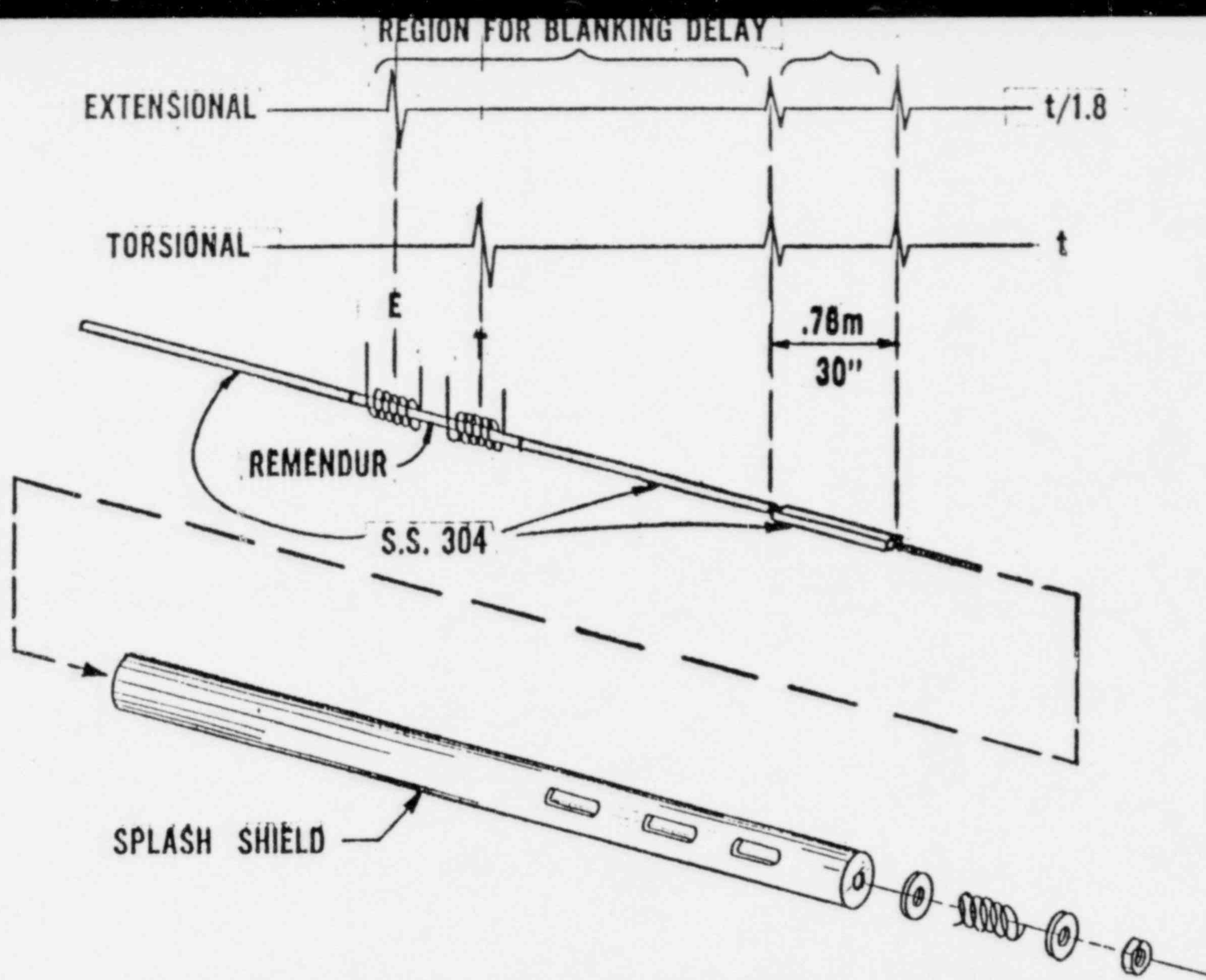
Figure 5 is a plot of error (% of full span) versus percent level in the data taken at the various temperatures. The worst case error for these tests was less than 3%. This accuracy can theoretically be improved by several techniques. The measurement resolution of the present Panatherm intervalometer is 100 nanoseconds. The combination of using the extensional wave to determine the temperature of the probe; using this temperature to correct the torsional data and calculating level results in a theoretical resolution of about 2 mm (about 2.7% of full span). Averaging can improve the results. Improved accuracy would result from smaller time resolution (a 20 MHz clock rather than 10 MHz), multiple trips through the liquid, and using sensor materials with a larger change in velocity for level change.

The results of the tests on the combination ultrasonic extensional-torsional wave level sensor showed that both waves can be excited in the same probe and that the temperatures derived from the extensional wave data can be used to correct the temperature dependence of the torsional wave signal. These tests were conducted over a temperature range of 25 to 300°C.

The objective of this program is to develop a liquid level sensor for use inside the reactor vessel of pressurized water reactors to meet the proposed NRC requirements for an unambiguous indication of inadequate core cooling. Reactor vendors and designers have further indicated a desire for a level detection device which is not event dependent.

Since under normal operating conditions, a pressurized water reactor is completely filled with coolant, an instrument which always indicates "full" runs a strong chance of being disregarded in the event a loss of coolant accident should occur. It is, therefore, imperative that there be a means for checking the operation of the level indicator during normal reactor operation. The combination extensional-torsional wave ultrasonic probe meets these requirements in that, with a zoned probe, the output of the device can be used not only to indicate level, but also temperature and density profiles. Correlation of these outputs with other plant sensor indications would provide a self-checking capability for the level probe. Furthermore, if the probe were located so that one zone could be confidently assumed to be completely surrounded with a medium of uniform characteristics, then the probe itself can be self-calibrating.

Continued development of this device will proceed now that the crucial experiment described in this paper has shown that the level indication can be temperature compensated over a wide range of temperatures. Attention can now be turned to the engineering and design problems necessary for a working level detector. Some of these considerations are: redesign to remove the excitation coils from inside the pressure and temperature boundry; improved resolution; a more rugged sensor element; possible use of mode conversion to generate the torsional wave; the optimum number of zones; the location of the zones and mounting and support of the sensor element. Further testing needs to be done in flowing systems, and to check the sensitivity of the probe to plant noise pickup.



DRAWING OF PROBE

FIGURE 1.

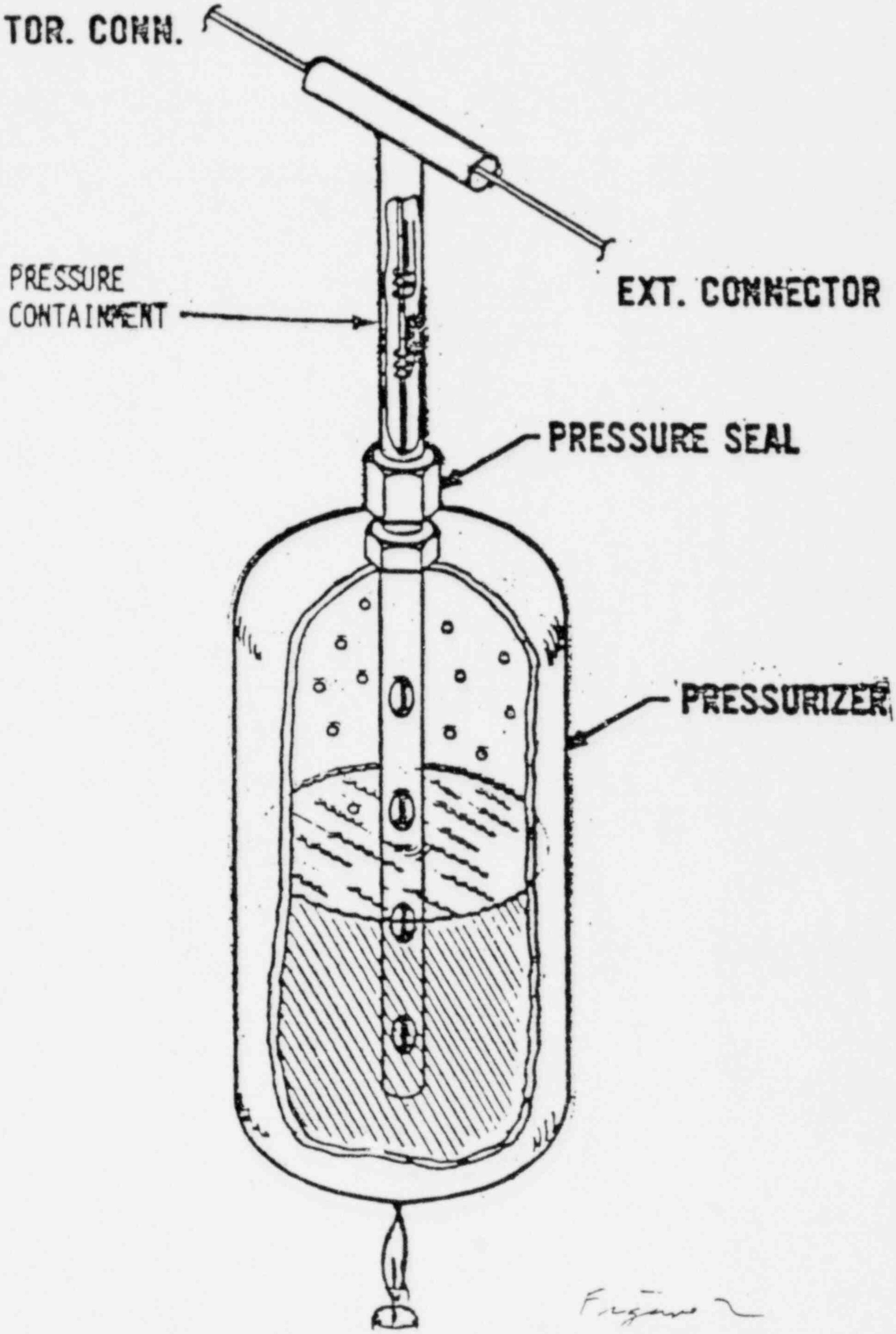
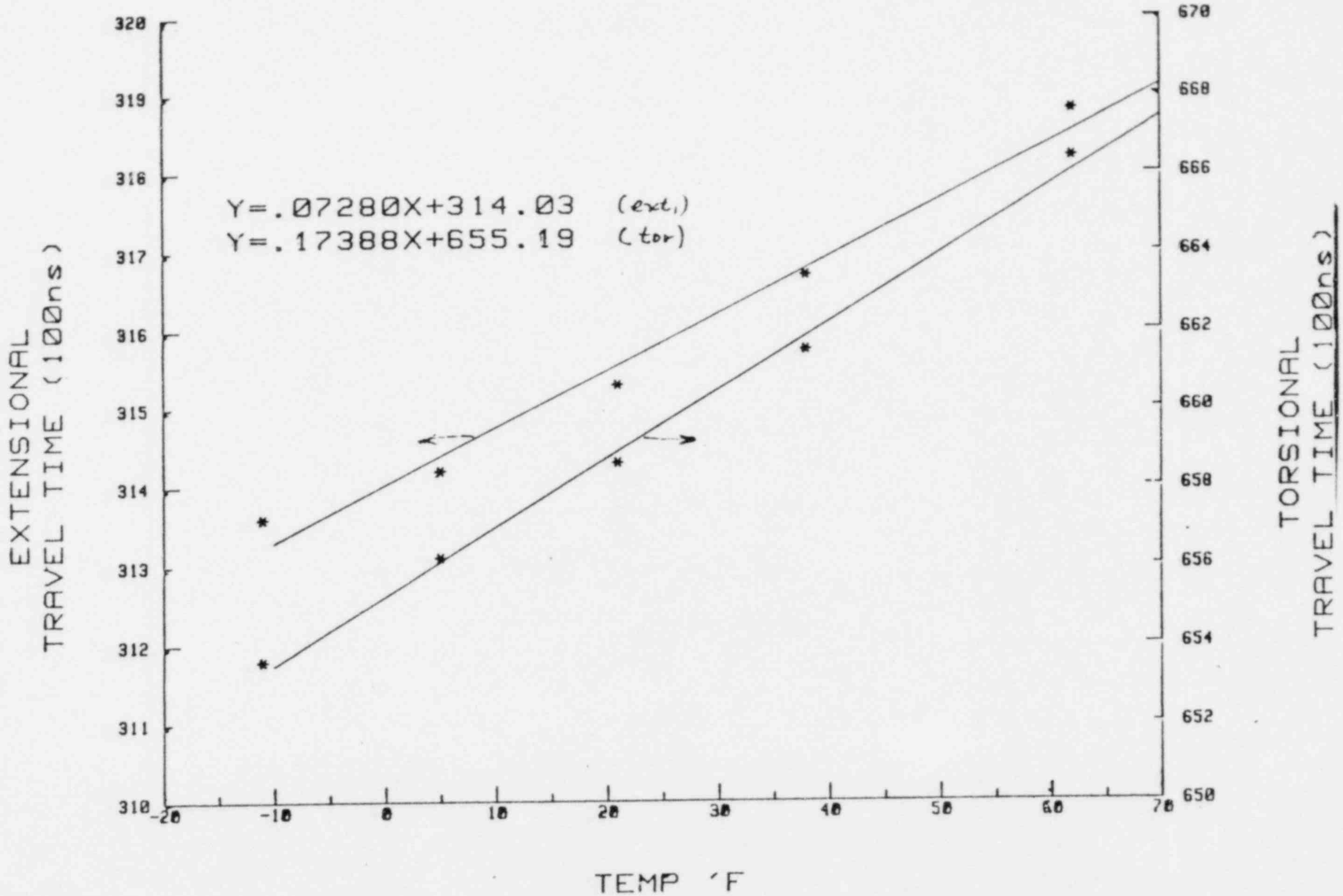
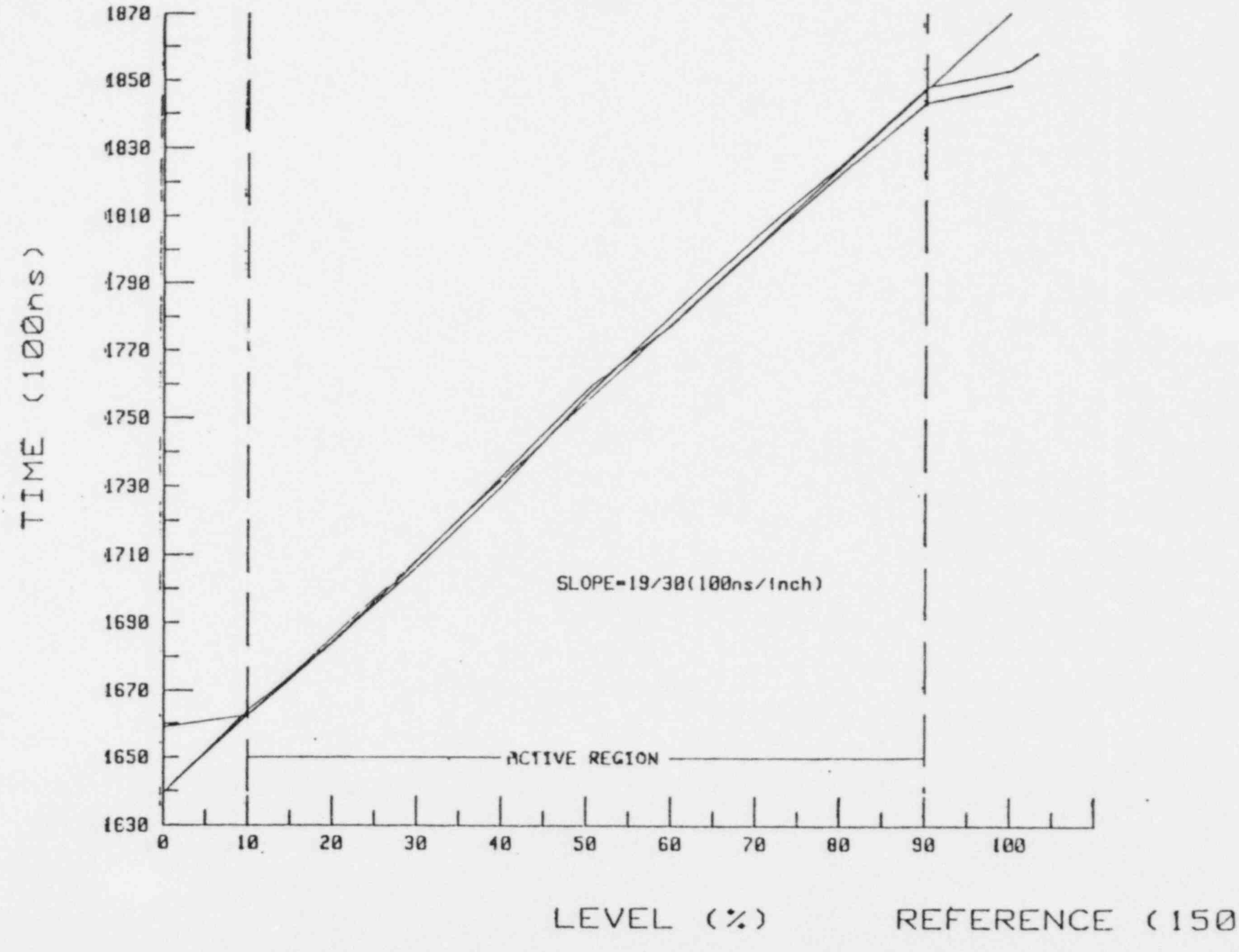

TOR. CONN.
EXT. CONNECTOR
PRESSURE
CONTAINMENT
PRESSURE SEAL
PRESSURIZER

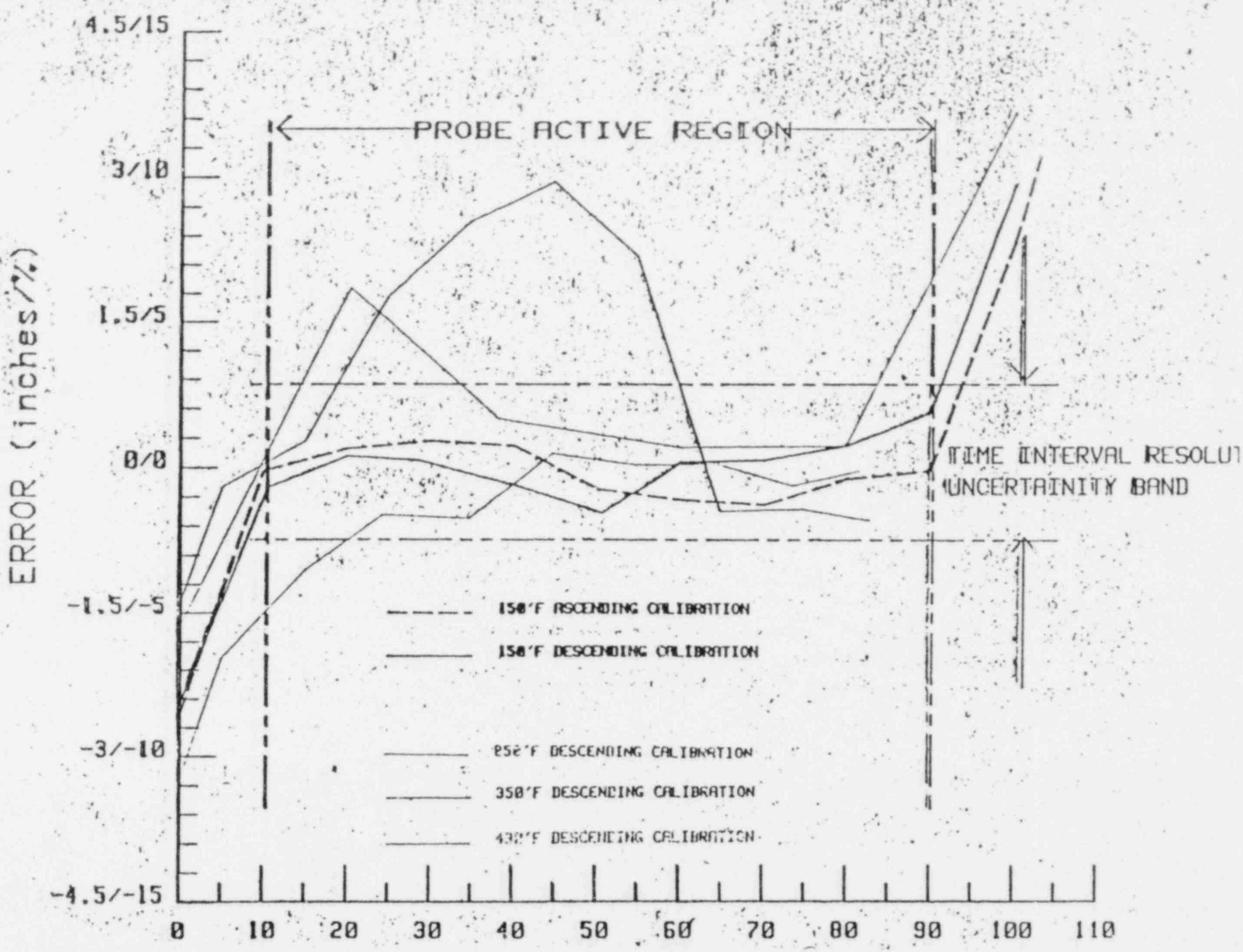
Figure 2

TEMPERATURE CHARACTERISTICS OF PROBE
CLEARLY EXHIBITS A LINEAR RELATIONSHIP





TIME (100ns)
1870
1850
1830
1810
1790
1770
1750
1730
1710
1690
1670
1650
1630
SLOPE=19/30(100ns/inch)
ACTIVE REGION
0
10
20
30
40
50
60
70
80
90
100
LEVEL (%)
REFERENCE (150


PROBE ACTIVE REGION
ERROR (inches/%)
4.5/15
3/10
1.5/5
0/0
-1.5/-5
-3/-10
-4.5/-15
0 10 20 30 40 50 60 70 80 90 100 110
TIME INTERVAL RESOLUT
UNCERTAINITY BAND
150'F ASCENDING CALIBRATION
150'F DESCENDING CALIBRATION
252'F DESCENDING CALIBRATION
350'F DESCENDING CALIBRATION
DESCENDING CALIBRATION

PUBLICATIONS

1. Advanced Two Phase Flow Instrumentation Program Quarterly Program Report for October to December, 1979.

   NUREG/CR - 1346 May 1980 - K. G. Turnage, C. E. Davis, R. L. Anderson and G. N. Miller.

2. Advanced Two Phase Flow Instrumentation Program Quarterly Program Report for January to March 1980.

   NUREG/CR 1647 September 1980 - K. G. Turnage, C. E. Davis, R. L. Anderson and G. N. Miller.

Avantages of the torsion-wave probe are:

1. only a steel ribbon in the reactor

2. transducers easily made radiation resistant

3. output can also yield density and temperature profiles

Some disadvantages are:

1. may be noise sensitive

2. instrumentation is more expensive

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The following goals were set in january, 1980:

1. Develop a proof-of-principle probe which combines torsional and extensional excitation

2. Characterize the probe

3. Test the probe

   - at room temperature
   - at elevated temperature
   - at elevated pressure

4. make recommendation to NRC concerning further development

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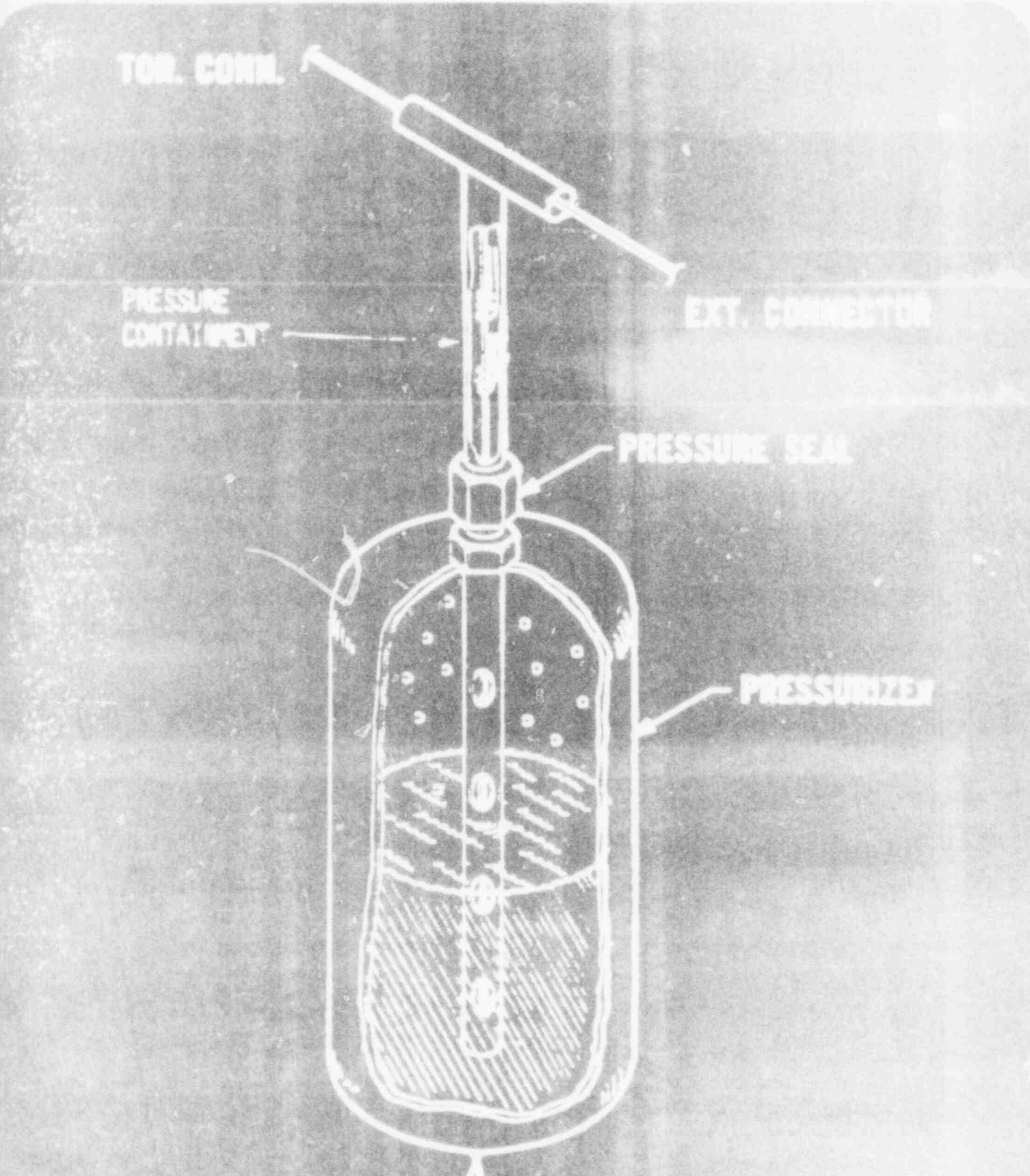
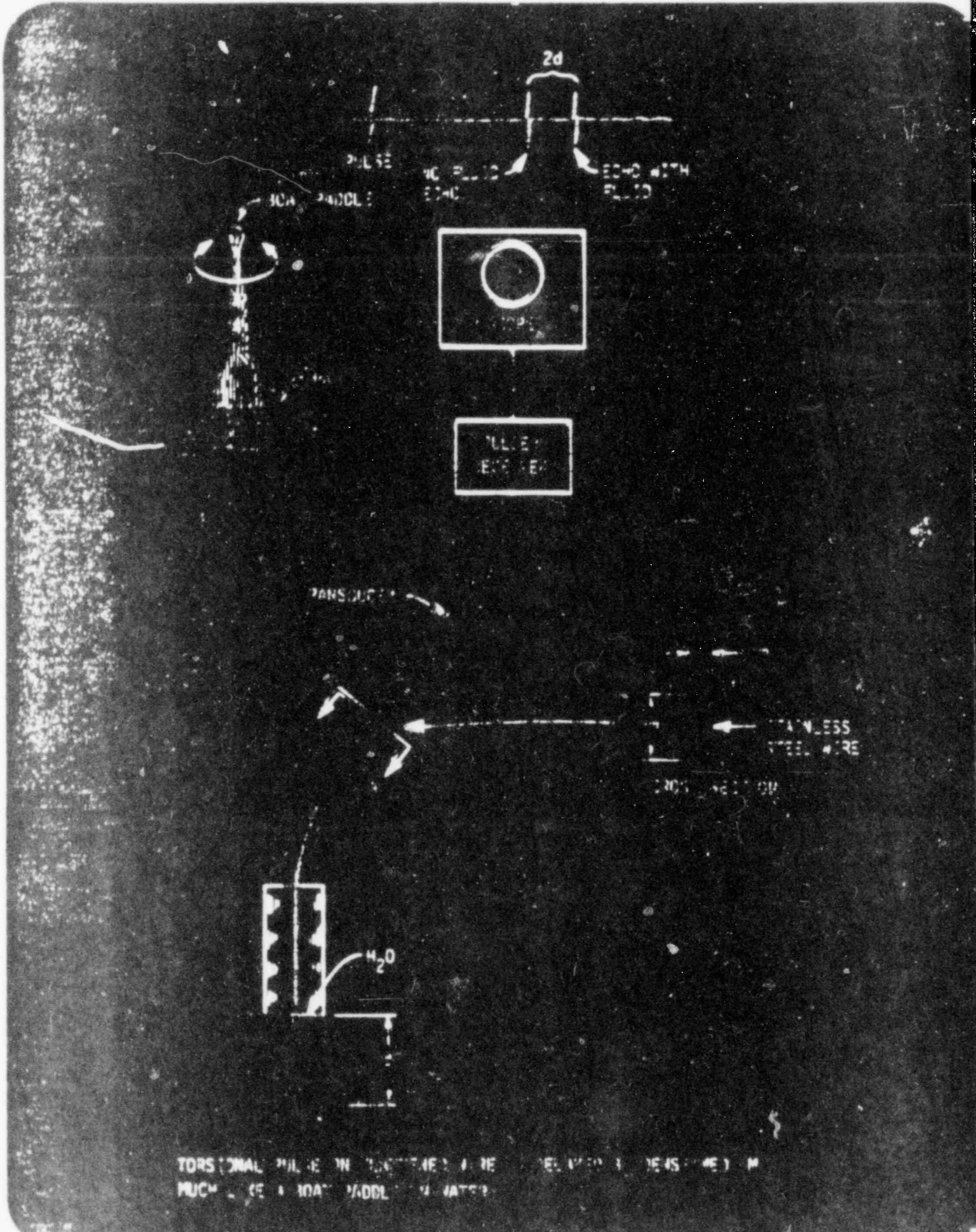
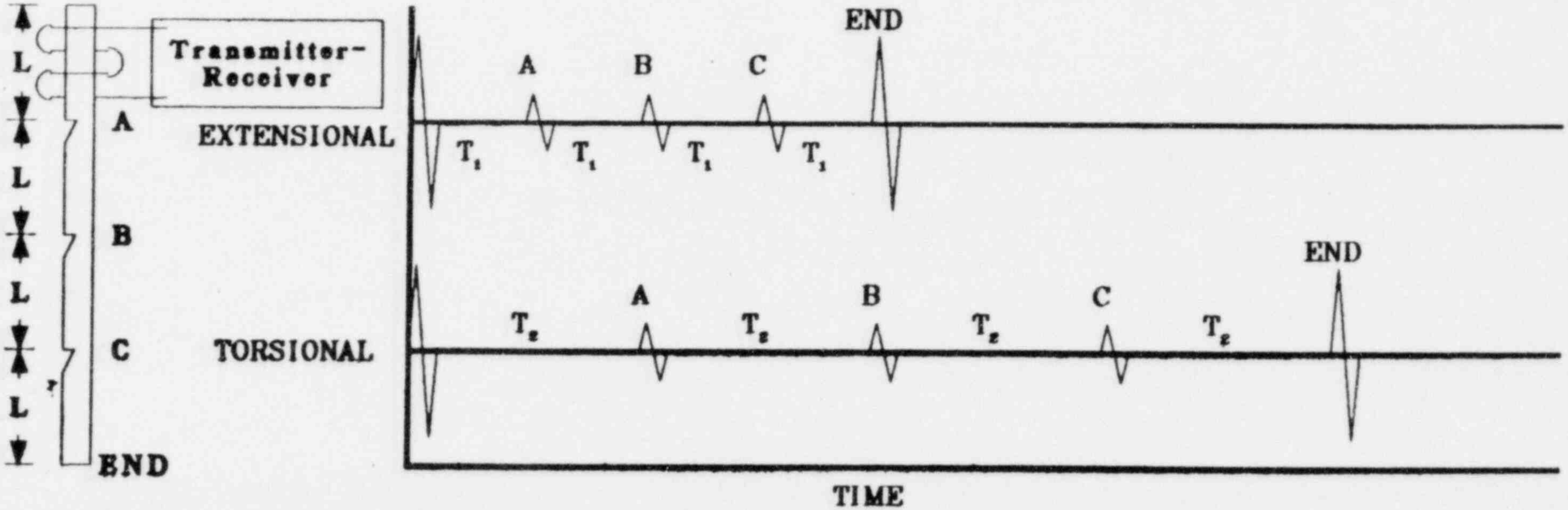

TOR. CONN.
PRESSURE
CONTAINMENT
EXT. CONNECTOR
PRESSURE SEAL
PRESSURIZER


2d
PULSE
H_2O

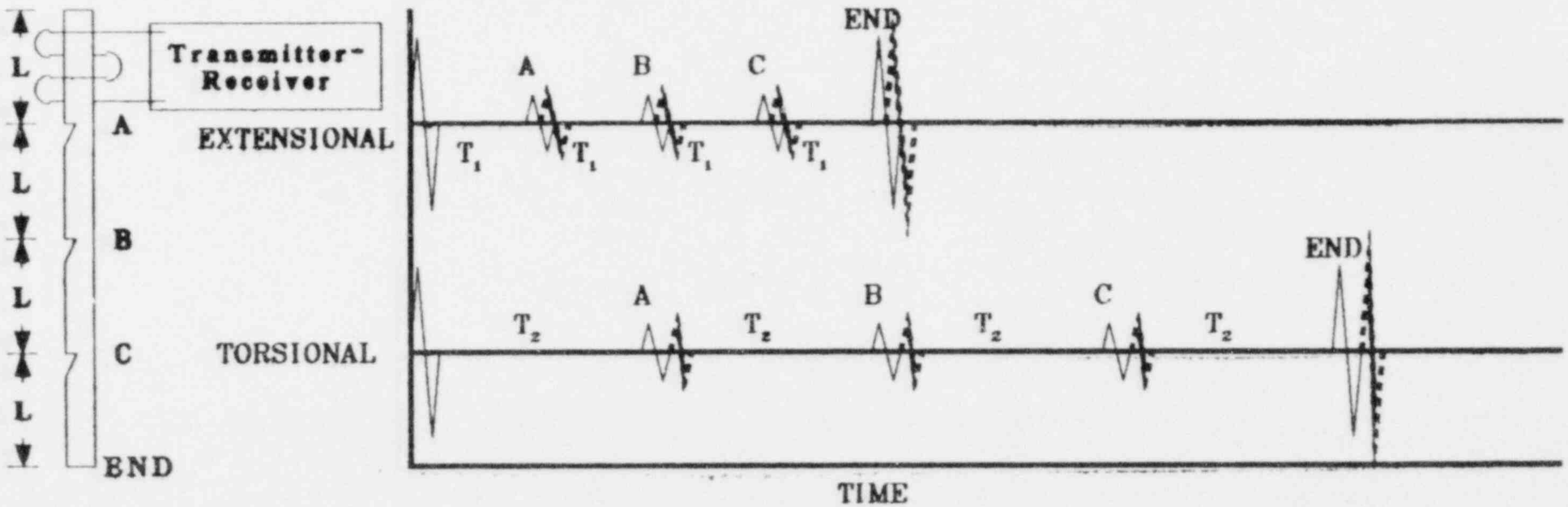
The velocity of an extensional wave is about twice the torsional velocity in stainless steel.



ornl

The velocity of an extensional wave is about twice the torsional velocity in stainless steel.

The velocity of both extensional and torsional waves decreases with increasing temperature.

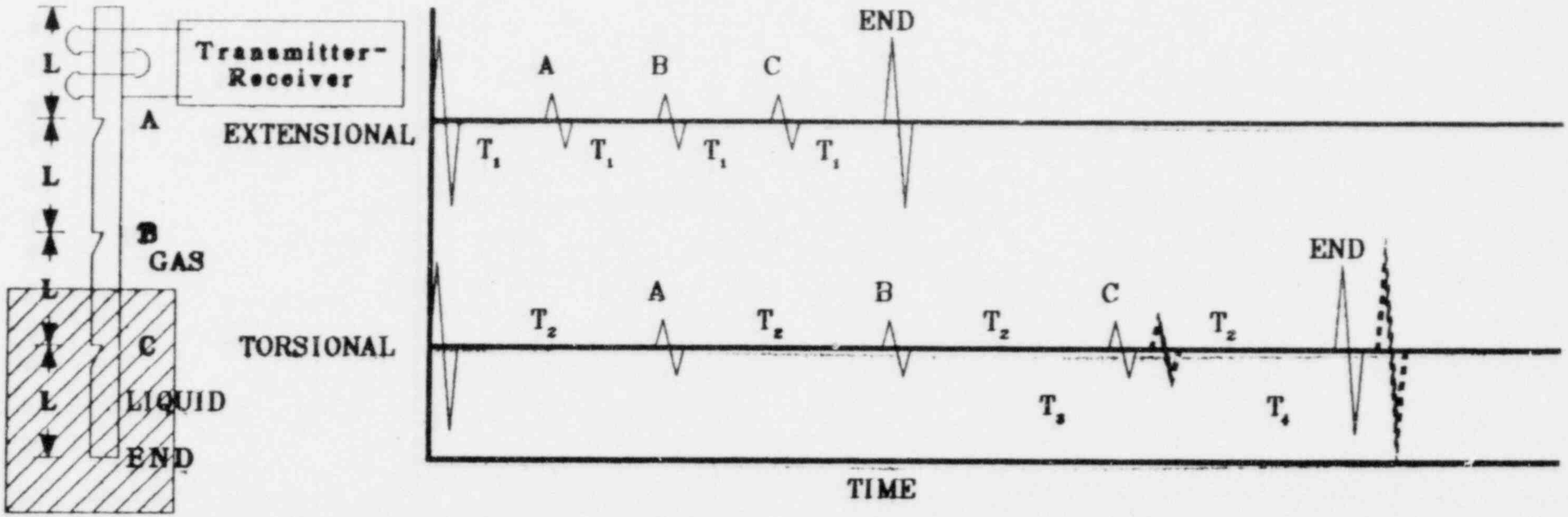


ornl

The velocity of an extensional wave is about twice the torsional velocity in stainless steel.

Changes in liquid level affect only the velocity of the torsional wave.

$$\text{Level} = L + L\cdot\left[\frac{T_3 - T_2}{T_4 - T_2}\right]$$



The velocity of torsional ultrasonic waves in a rectangular waveguide is dependent on the *density* of the surrounding medium.

$$v = K \sqrt{\frac{\mu}{\rho}} \left[1 + \frac{\rho}{2\rho_s} \left(1 - \frac{1}{K} \right) \right]$$

The velocity of an extensional wave is:

$$v = \sqrt{\frac{Y}{\rho_s}}$$

where ρ = density of surrounding medium
ρ_s = density of sensor material
μ = shear modulus
Y = Young's modulus
K = shape factor (less than one)

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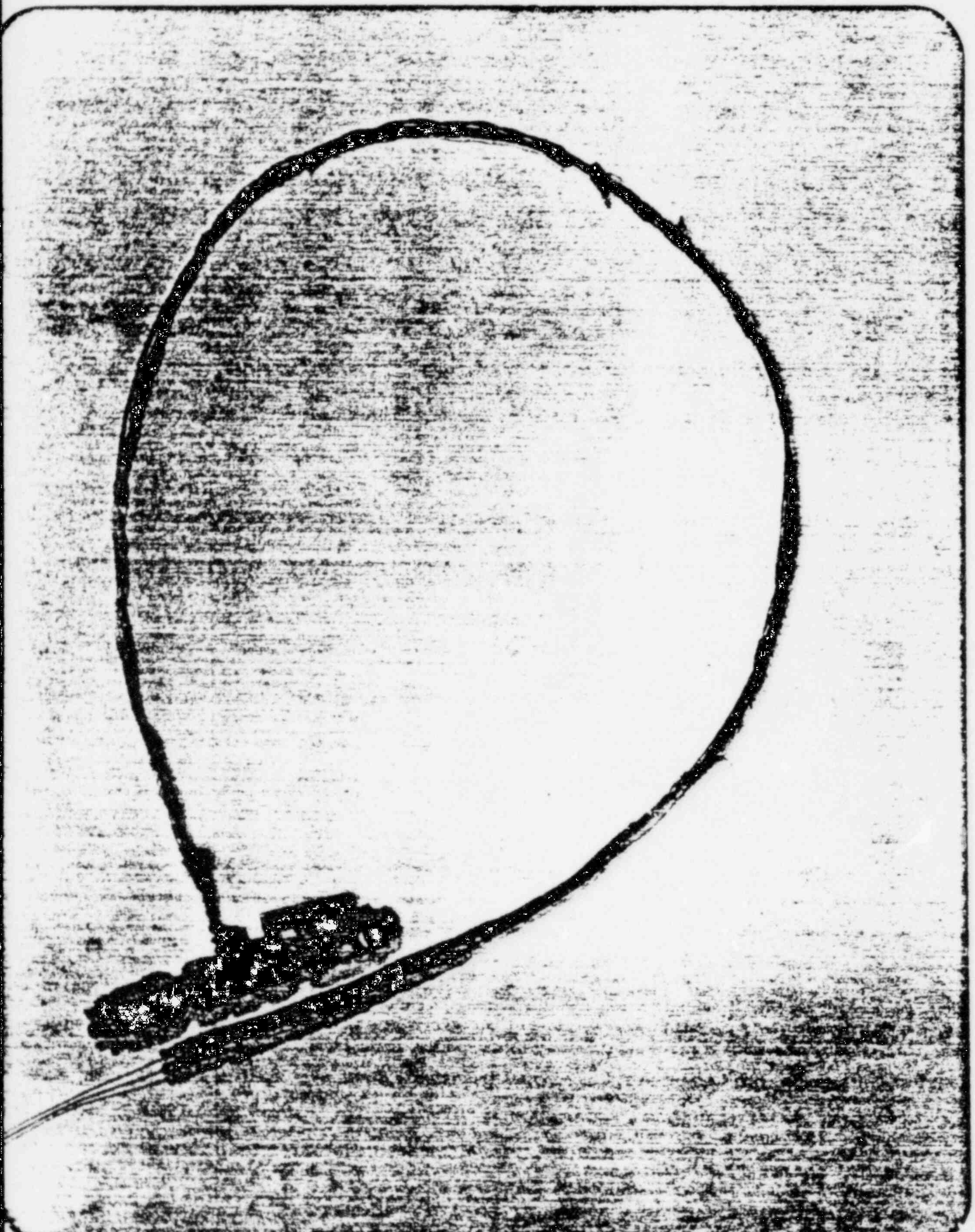
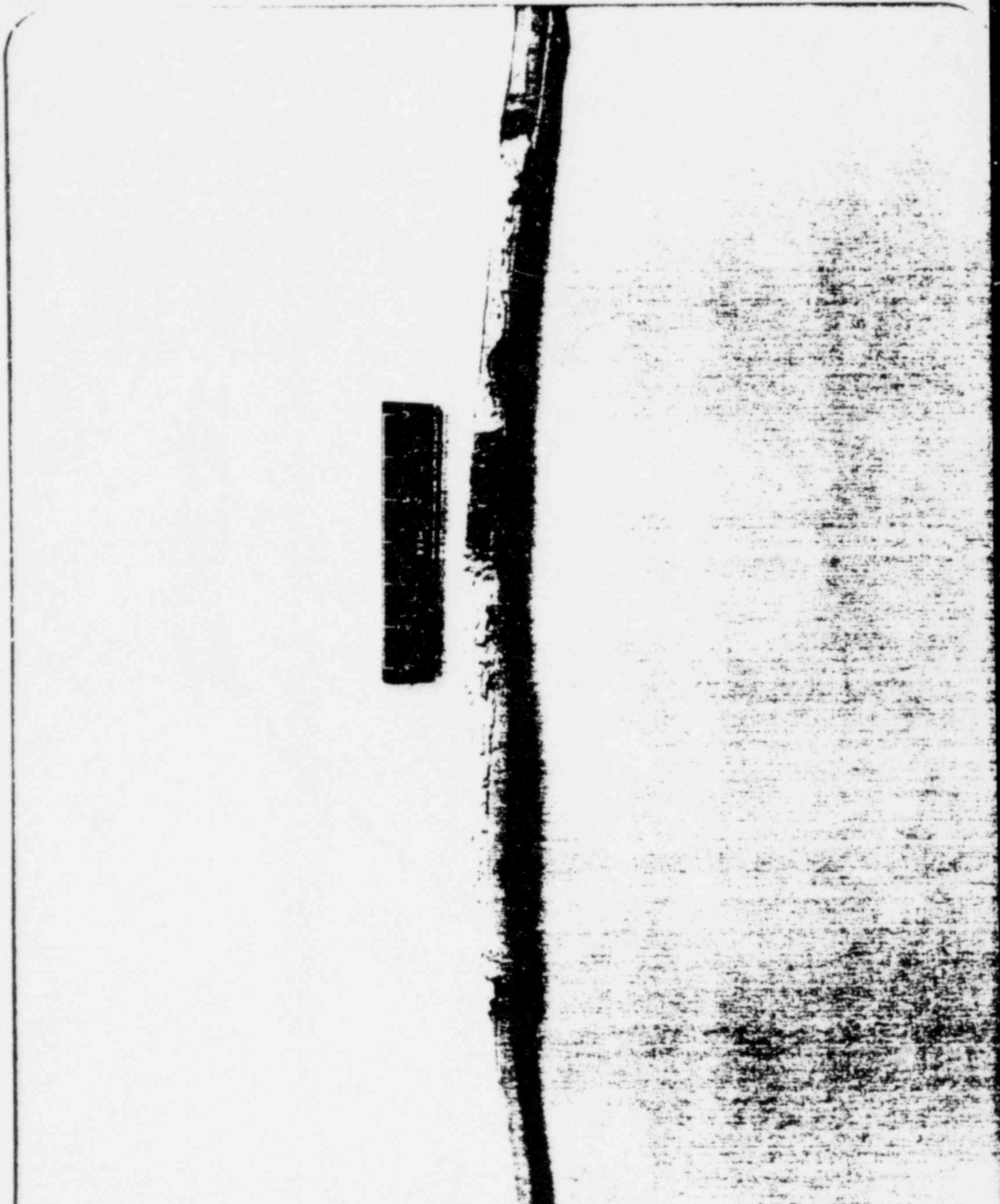
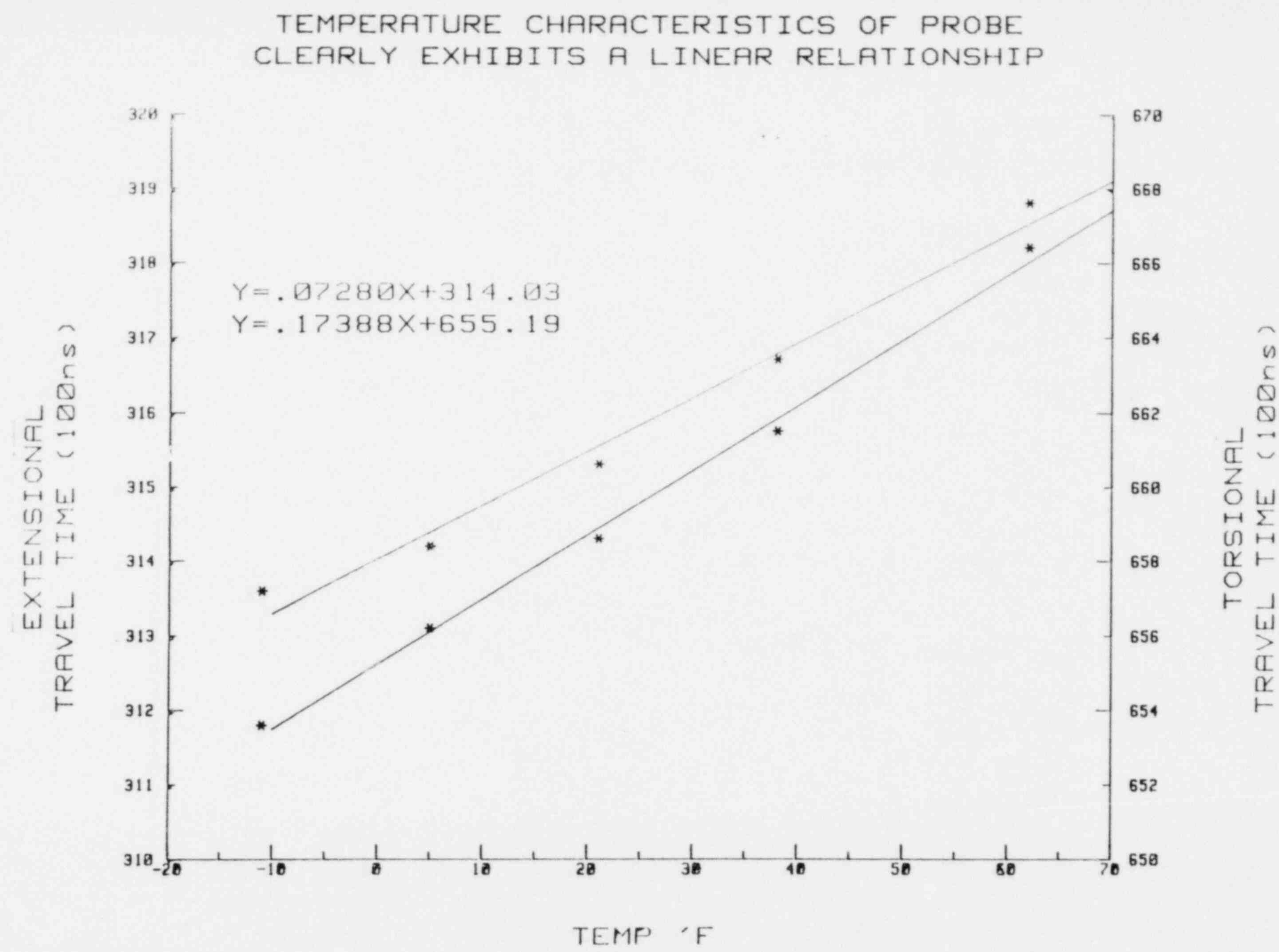
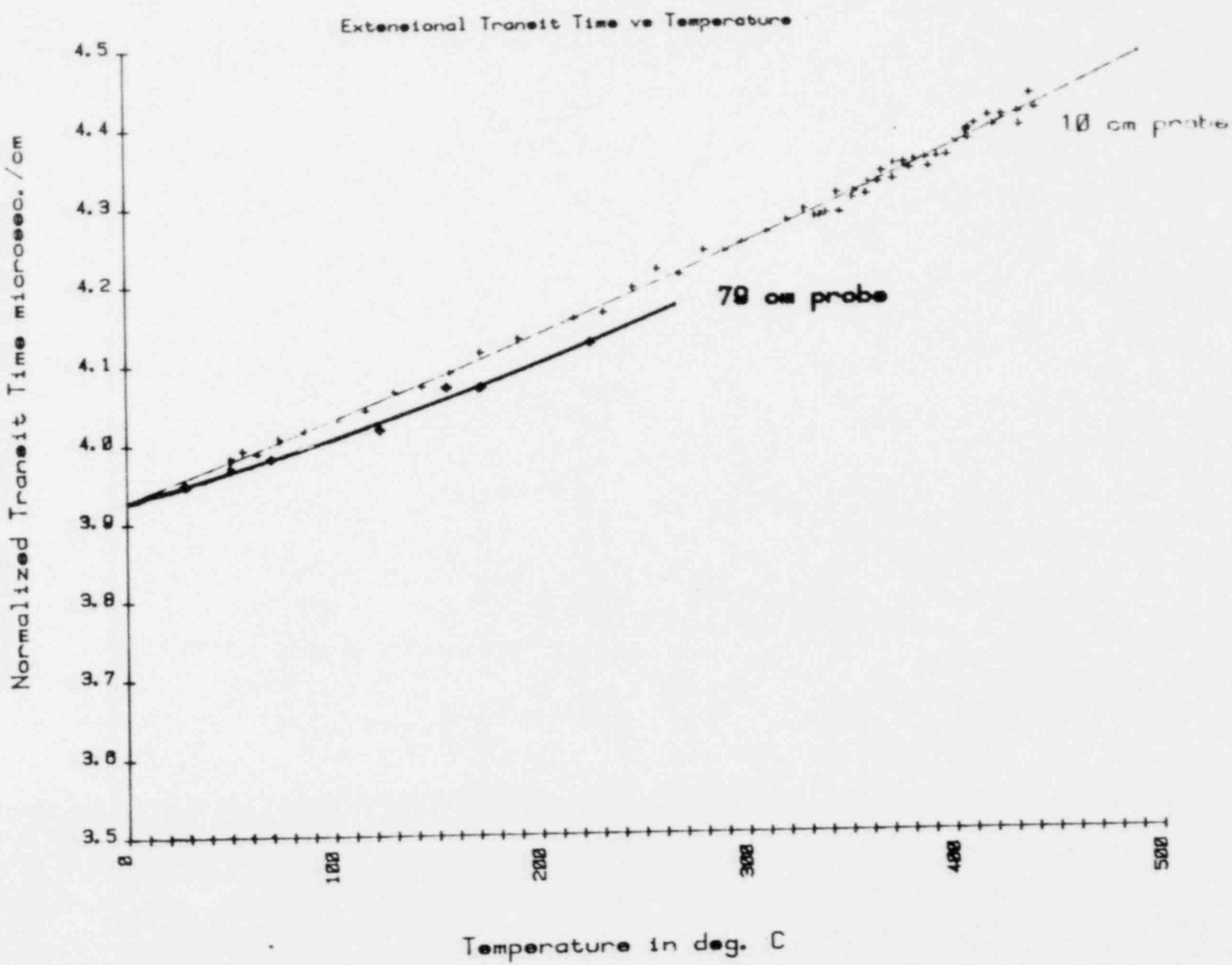
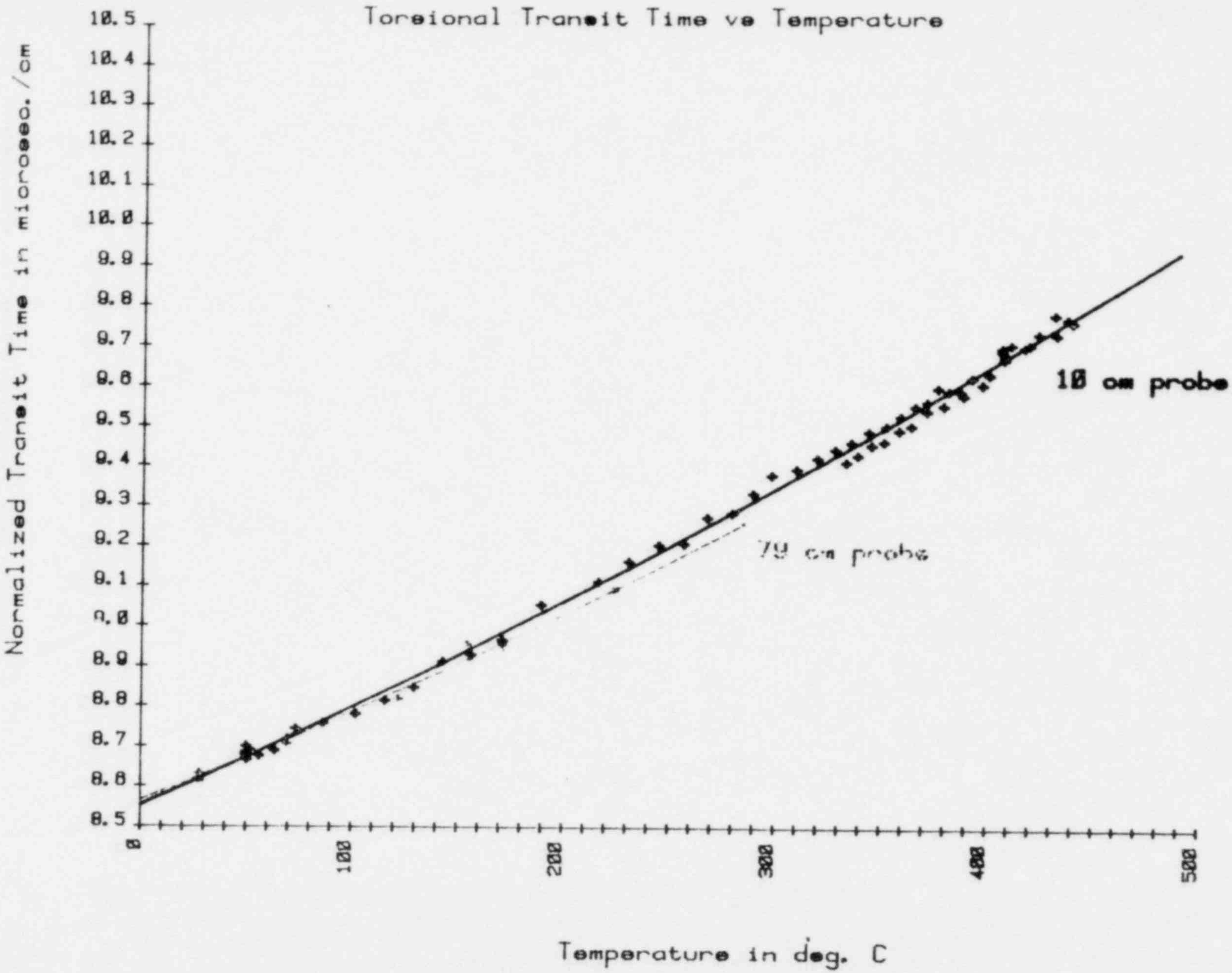


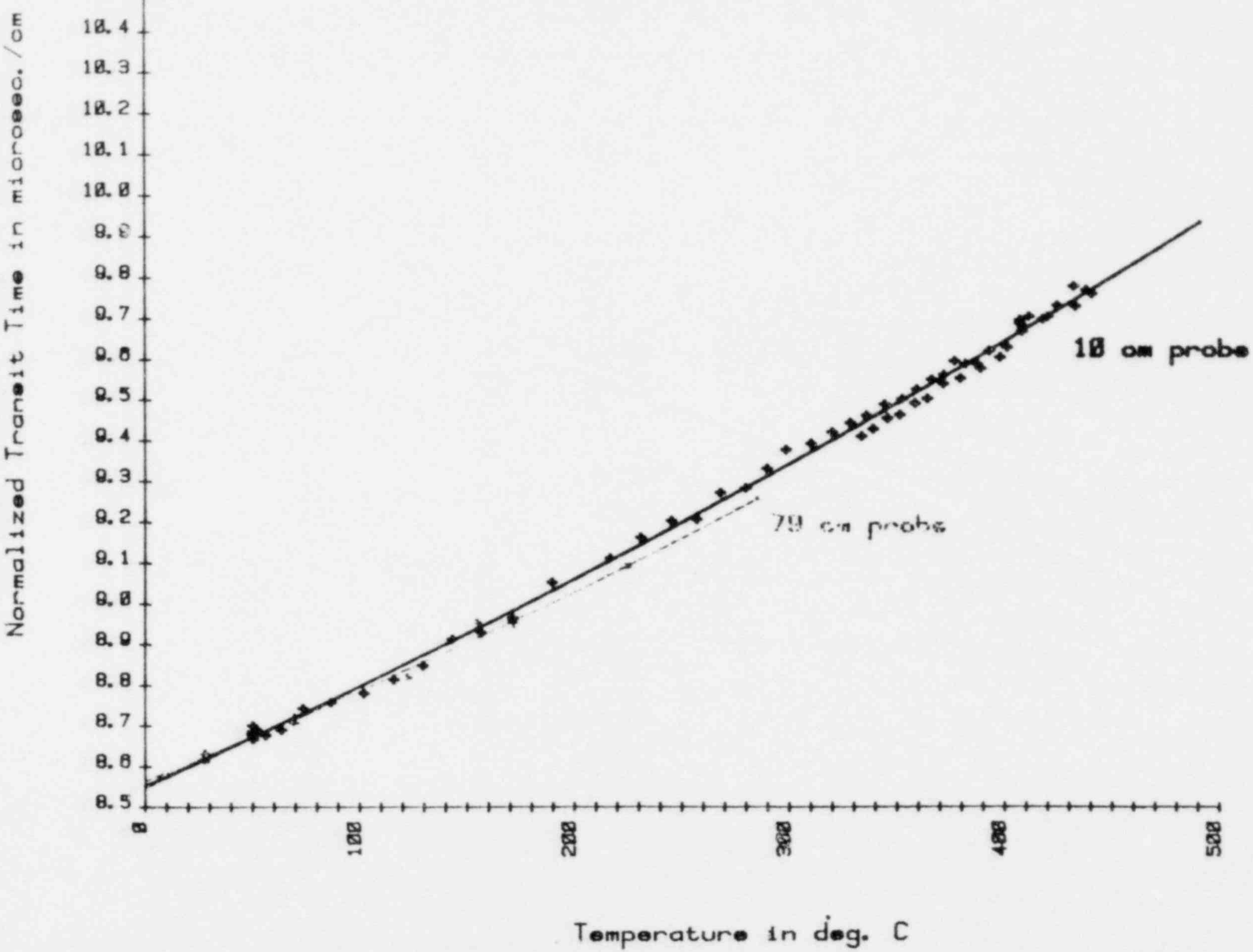


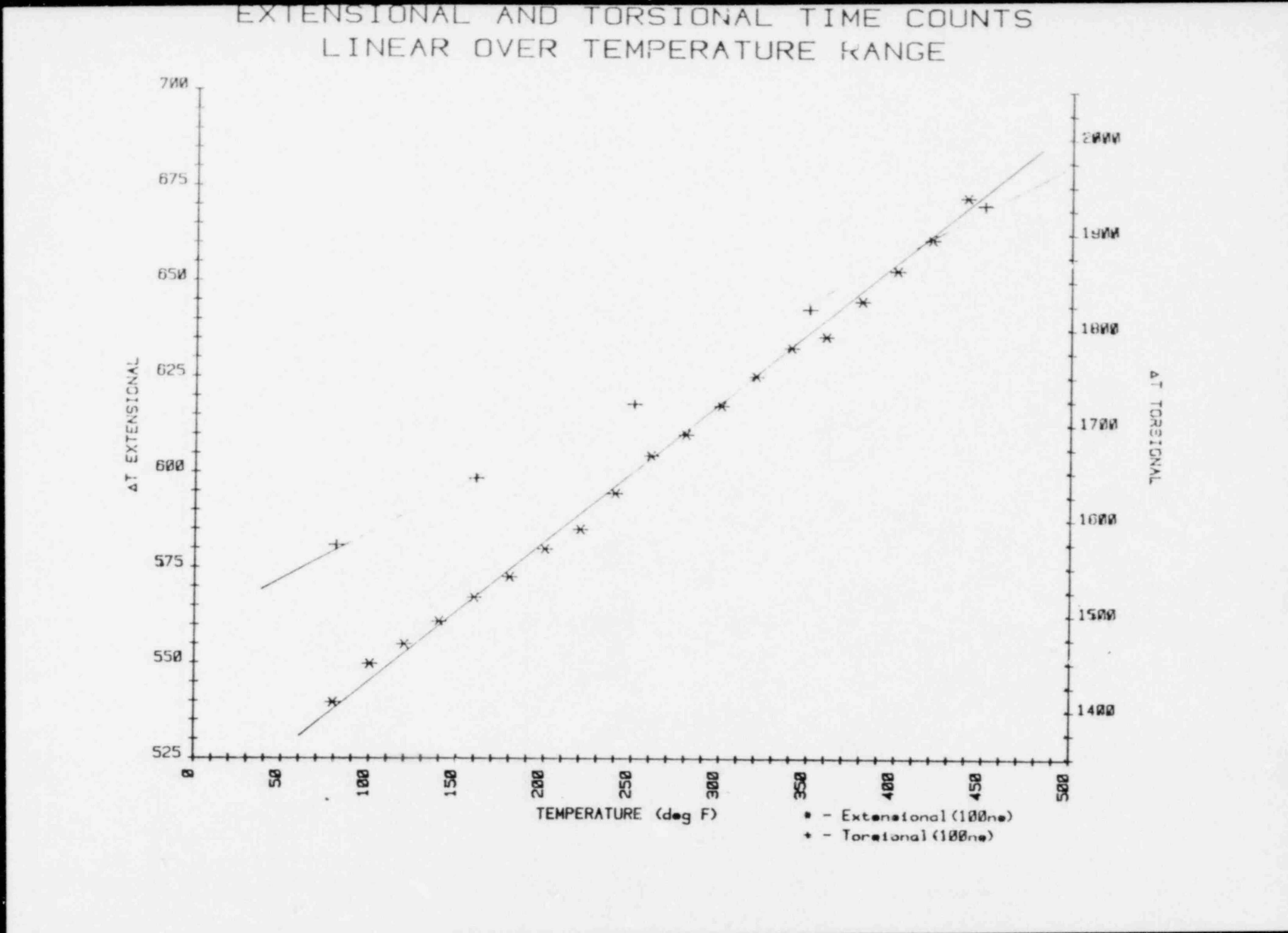

TEMPERATURE CHARACTERISTICS OF PROBE
CLEARLY EXHIBITS A LINEAR RELATIONSHIP
Y=.07280X+314.03
Y=.17388X+655.19
EXTENSIONAL
TRAVEL TIME (100ns)
TORSIONAL
TRAVEL TIME (100ns)
TEMP 'F
320
319
318
317
316
315
314
313
312
311
310
670
668
666
664
662
660
658
656
654
652
650
-20
-10
0
10
20
30
40
50
60
70

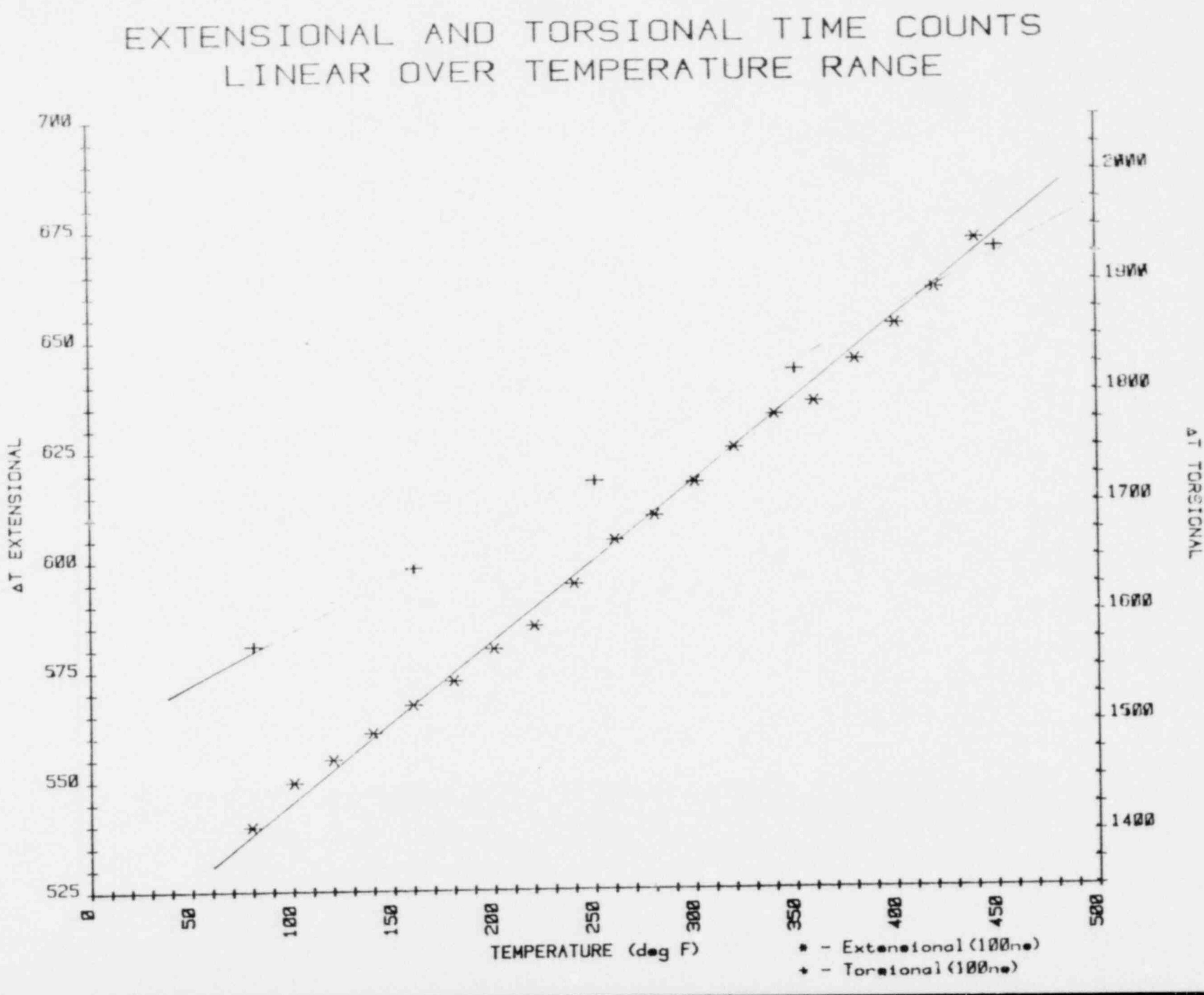

Extensional Transit Time vs Temperature
Normalized Transit Time microsec./cm
4.5
4.4
4.3
4.2
4.1
4.0
3.9
3.8
3.7
3.6
3.5
0
100
200
300
400
500
Temperature in deg. C
10 cm probe
70 cm probe

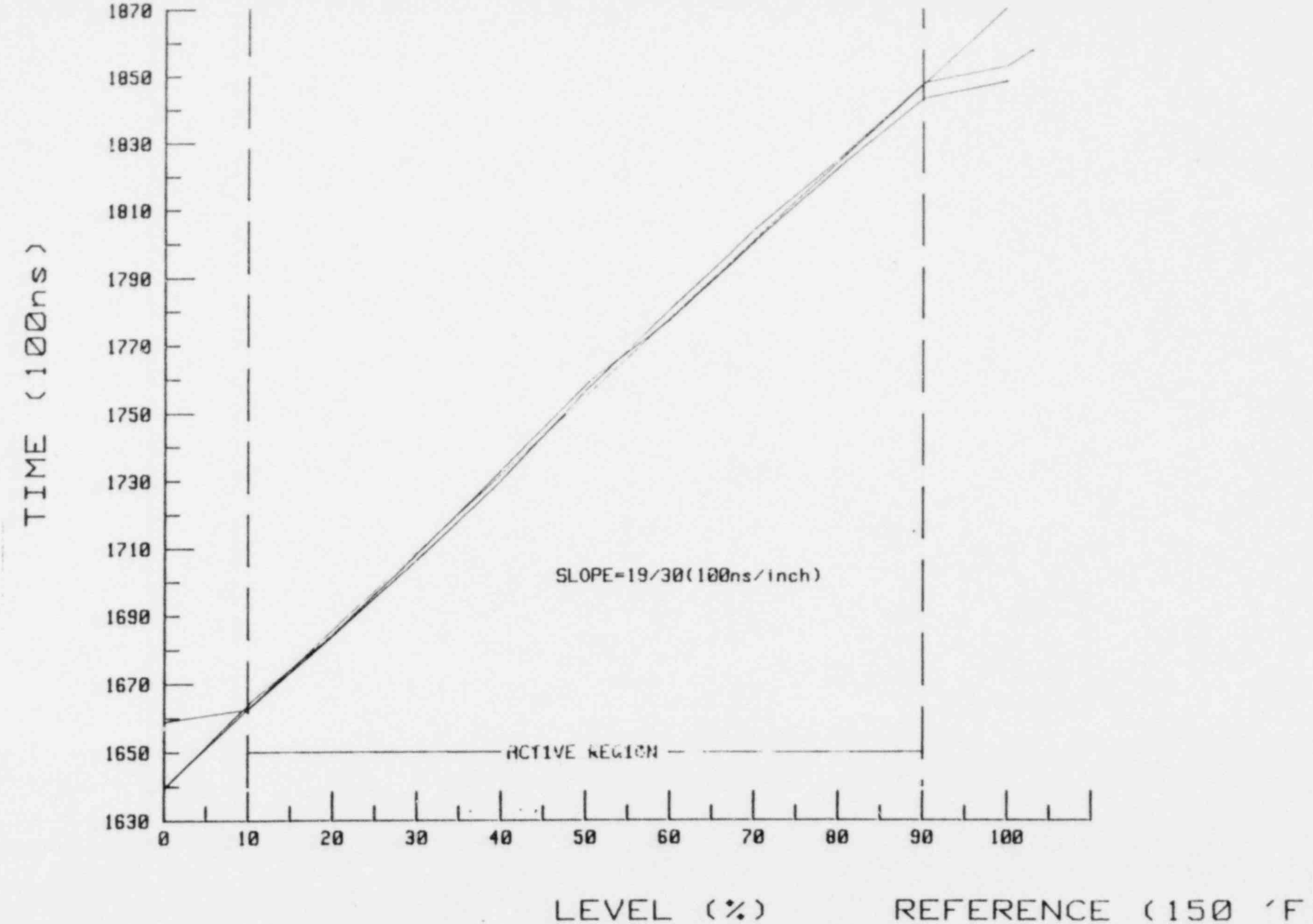

Torsional Transit Time vs Temperature
Normalized Transit Time in microsec./cm
Temperature in deg. C
10 cm probe
79 cm probe
10.5
10.4
10.3
10.2
10.1
10.0
9.9
9.8
9.7
9.6
9.5
9.4
9.3
9.2
9.1
9.0
8.9
8.8
8.7
8.6
8.5
0
100
200
300
400
500

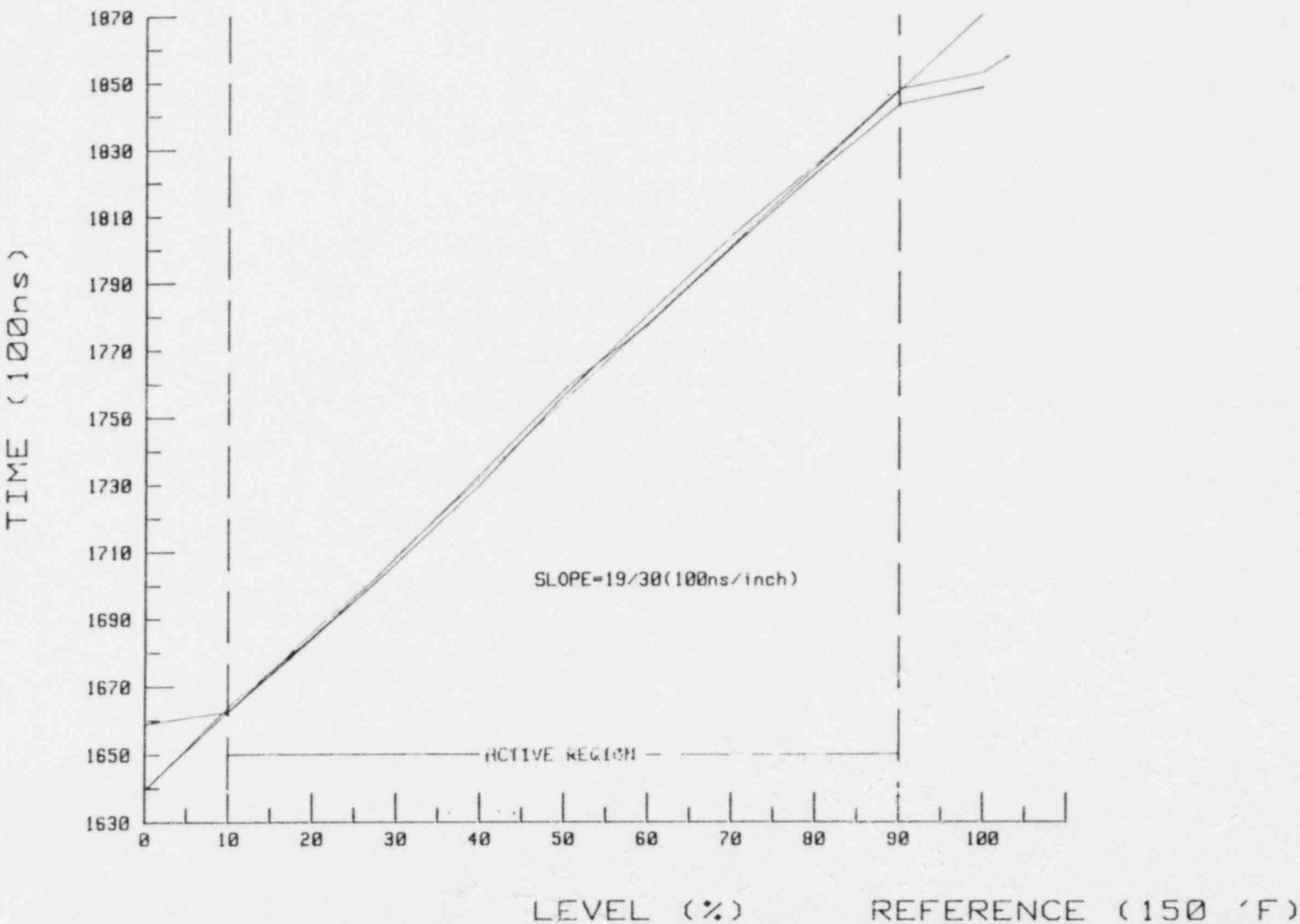
Torsional Transit Time vs Temperature

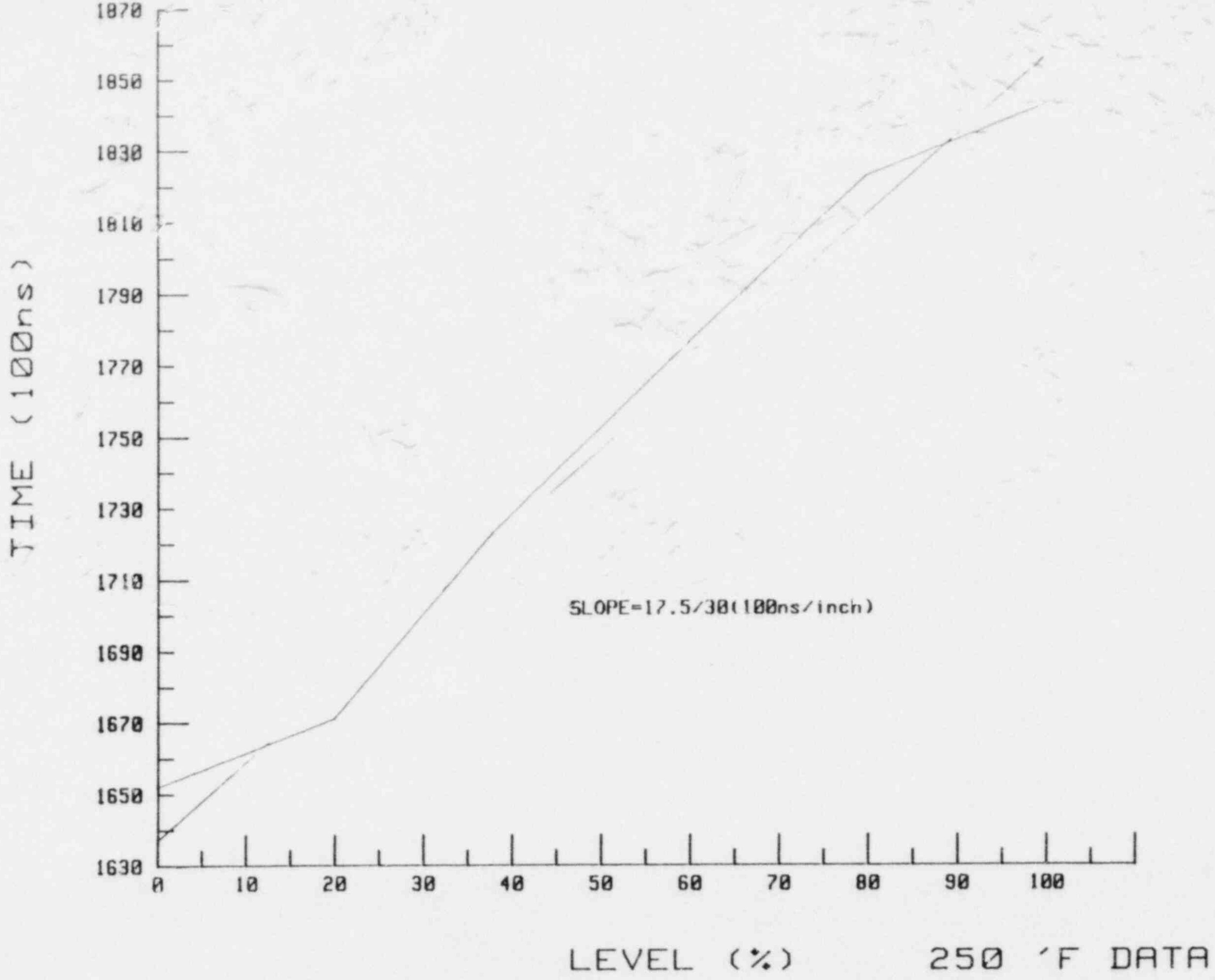



EXTENSIONAL AND TORSIONAL TIME COUNTS
LINEAR OVER TEMPERATURE RANGE
ΔT EXTENSIONAL
ΔT TORSIONAL
700
675
650
625
600
575
550
525
2000
1900
1800
1700
1600
1500
1400
0
50
100
150
200
250
300
350
400
450
500
TEMPERATURE (deg F)
* - Extensional(100ns)
+ - Torsional(100ns)

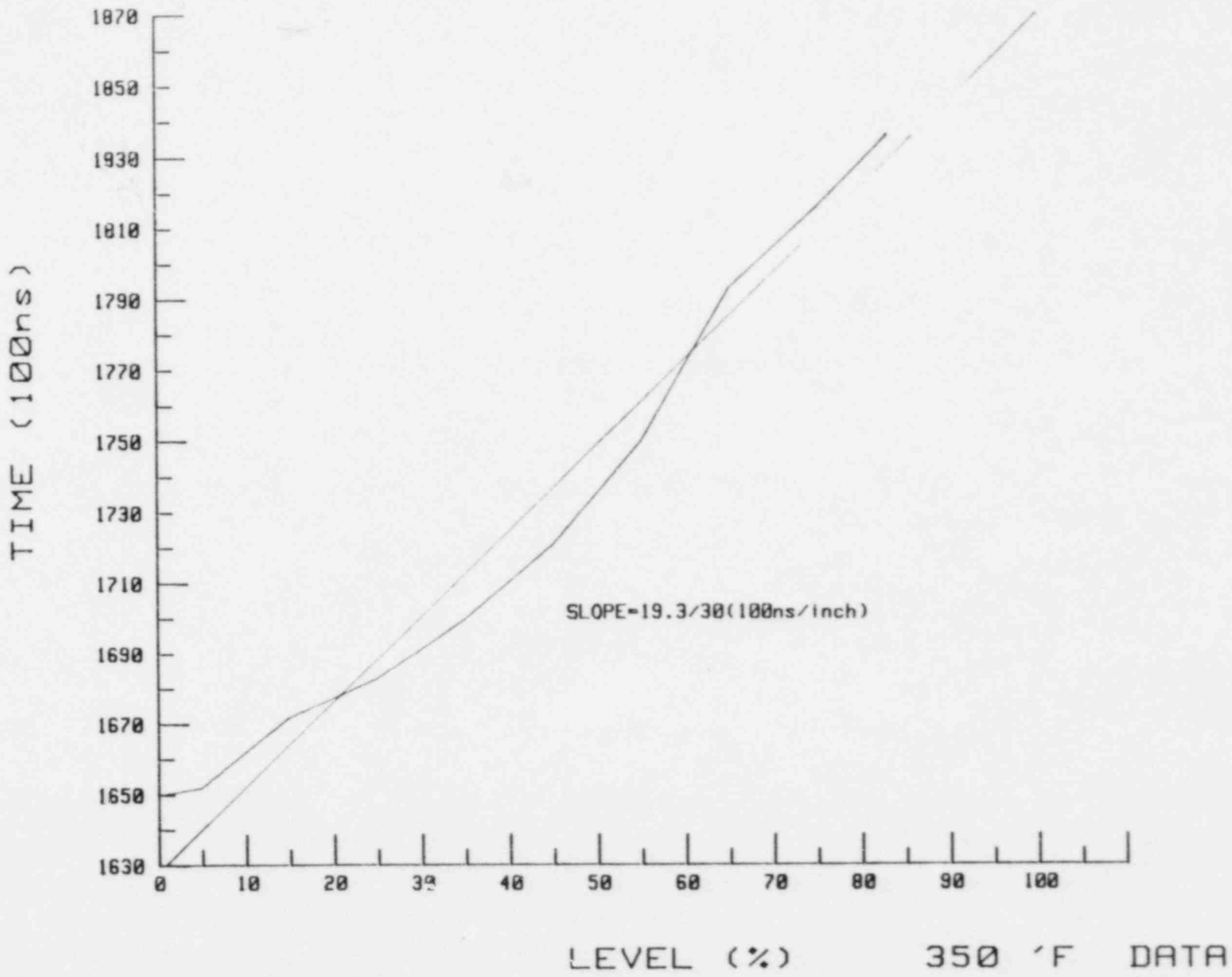

EXTENSIONAL AND TORSIONAL TIME COUNTS
LINEAR OVER TEMPERATURE RANGE
ΔT EXTENSIONAL
ΔT TORSIONAL
TEMPERATURE (deg F)
700
675
650
625
600
575
550
525
2000
1900
1800
1700
1600
1500
1400
0
50
100
150
200
250
300
350
400
450
500
* - Extensional (100ns)
+ - Torsional (100ns)


TIME (100ns)
1870
1850
1830
1810
1790
1770
1750
1730
1710
1690
1670
1650
1630
0
10
20
30
40
50
60
70
80
90
100
SLOPE=19/30(100ns/inch)
ACTIVE REGION
LEVEL (%)
REFERENCE (150 'F)

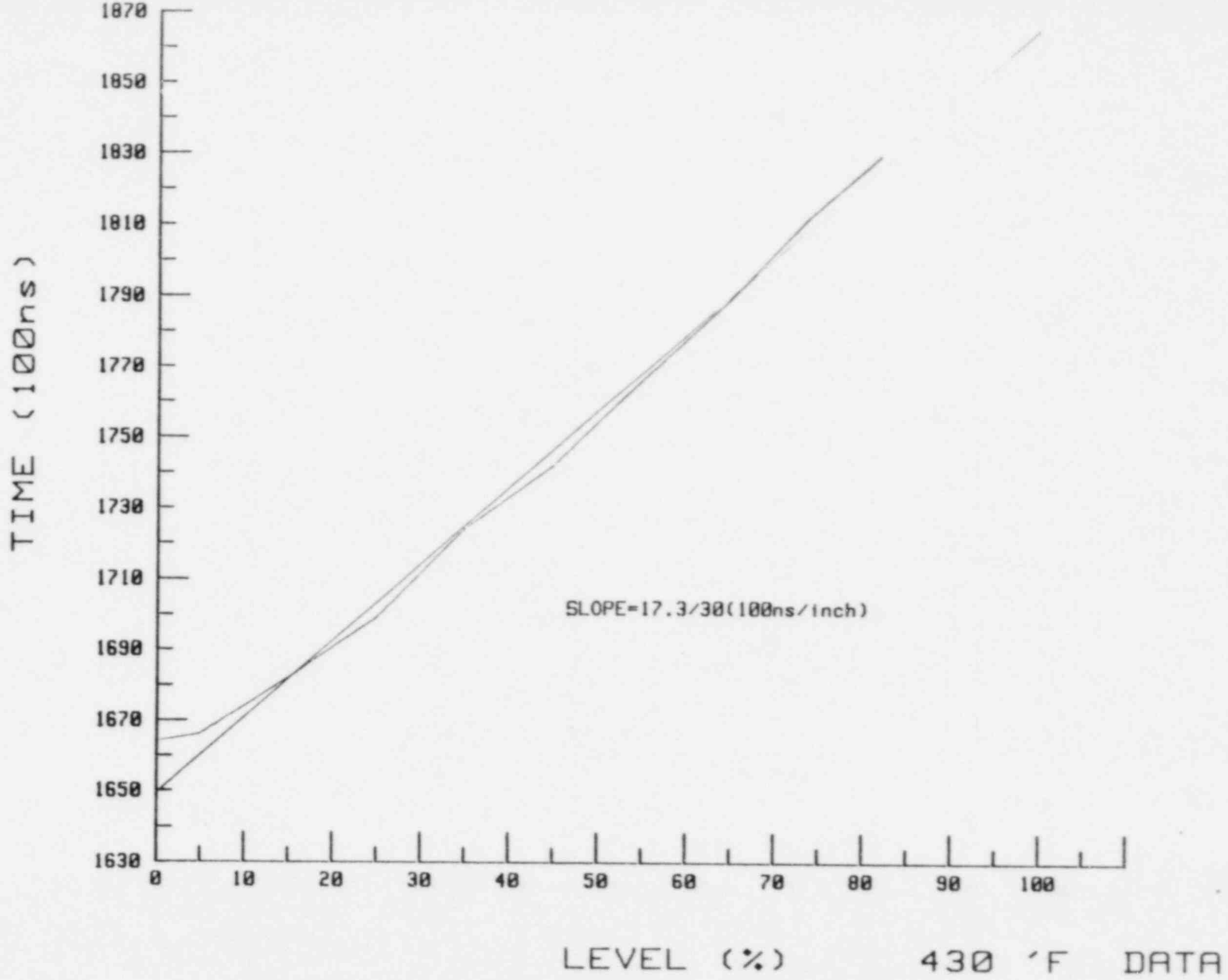

TIME (100ns)
1870
1850
1830
1810
1790
1770
1750
1730
1710
1690
1670
1650
1630
0
10
20
30
40
50
60
70
80
90
100
SLOPE=19/30(100ns/inch)
ACTIVE REGION
LEVEL (%)
REFERENCE (150 'F)



TIME (100ns)
1870
1850
1830
1810
1790
1770
1750
1730
1710
1690
1670
1650
1630
0
10
20
30
40
50
60
70
80
90
100
SLOPE=17.5/30(100ns/inch)
LEVEL (%)
250 'F DATA



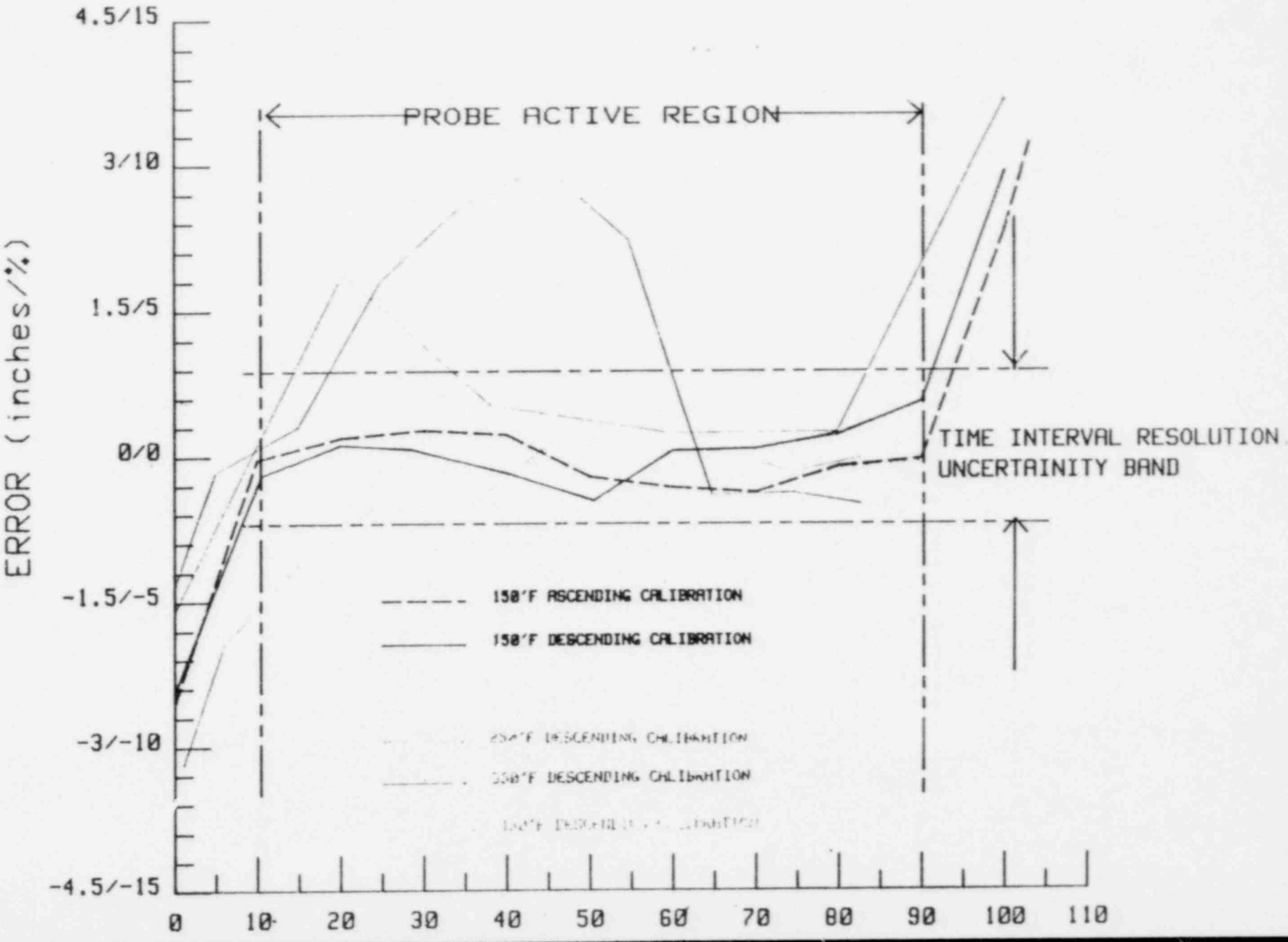
TIME (100ns)
1870
1850
1930
1810
1790
1770
1750
1730
1710
1690
1670
1650
1630
SLOPE=19.3/30(100ns/inch)
0
10
20
30
40
50
60
70
80
90
100
LEVEL (%)
350 ′F DATA

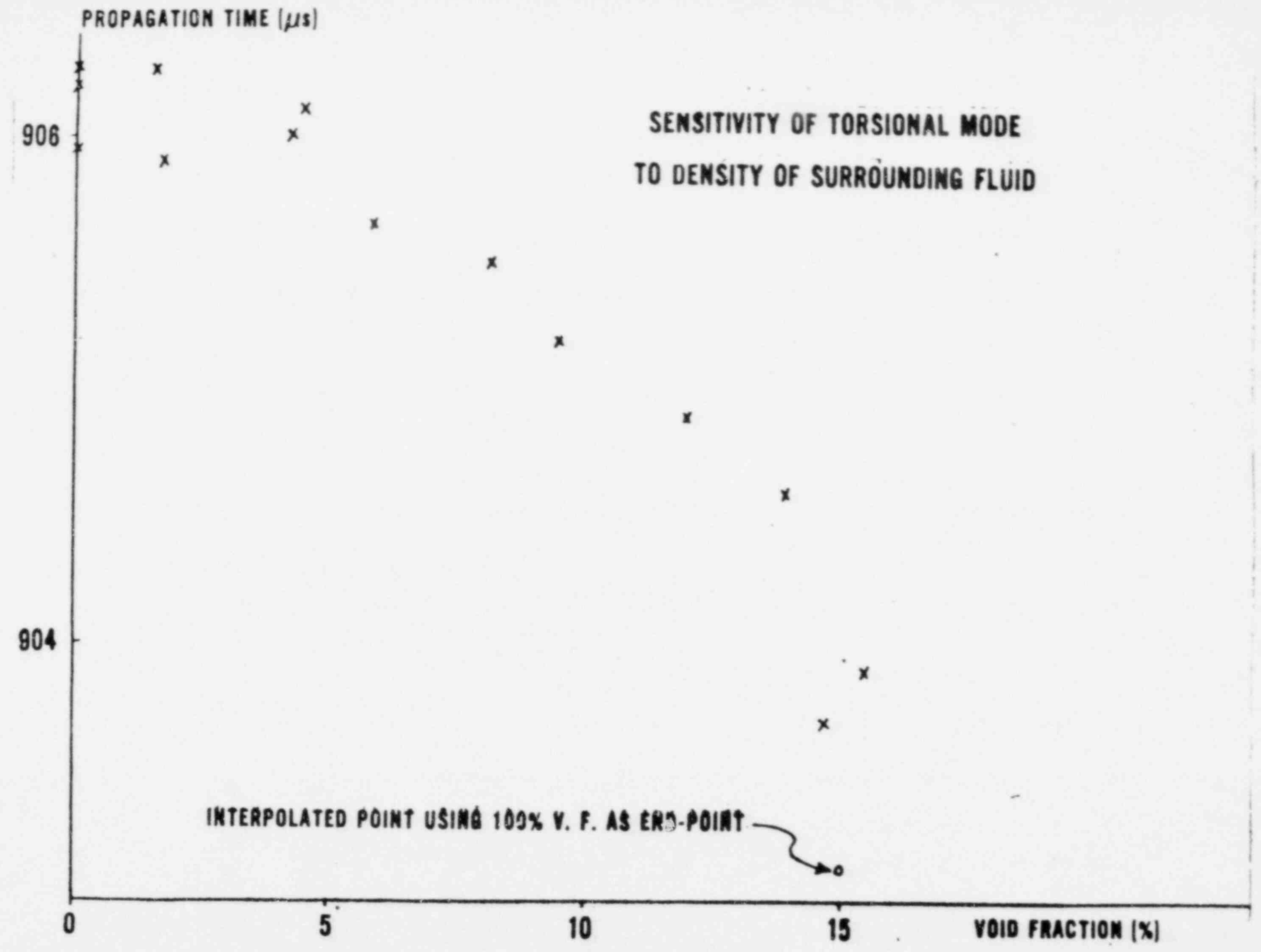


1870
1850
1830
1810
1790
1770
1750
1730
1710
1690
1670
1650
1630
TIME (100ns)
SLOPE=17.3/30(100ns/inch)
0
10
20
30
40
50
60
70
80
90
100
LEVEL (%)
430 ′F DATA

PROBE WELL BEHAVED IN ACTIVE REGION

ORNL 1980





PROPAGATION TIME (μs)
906
904
SENSITIVITY OF TORSIONAL MODE
TO DENSITY OF SURROUNDING FLUID
INTERPOLATED POINT USING 100% V. F. AS END-POINT
0
5
10
15
VOID FRACTION (%)

Development plans for the torsional-ultrasonic level probe include:

1. Develop new probe with the intent of solving various problems:

   - locate coils outside pressure boundry
   - Remendur temperature problem
   - pressure seal
   - strength
   - good communication with coolant fluid
   - measurement of coolant density

2. Test at elevated temperature and pressure

3. Test under two-phase flow conditions

4. Test with typical plant acoustic noise

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ORNL WS-13556

IN-VESSEL INSTRUMENTATION FOR HIGH-TEMPERATURE TRANSIENT TWO-PHASE FLOWS

B. G. EADS
ADVANCED INSTRUMENTATION FOR REFLOOD STUDIES PROGRAM

presented at

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING
GAITHERSBURG, MARYLAND
OCTOBER 27–31, 1980

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IN-VESSEL INSTRUMENTATION FOR HIGH-TEMPERATURE TRANSIENT TWO-PHASE FLOWS*

Introduction

A program under the sponsorship of the United States Nuclear Regulatory Commission (USNRC) was initiated in 1977 at the Oak Ridge National Laboratory (ORNL) to develop instrumentation for application in PWR safety experimental facilities. The program, Advanced Instrumentation for Reflood Studies (AIRS), is specifically to develop instrumentation for measurement of in-vessel local fluid phenomena in safety experiments designed to investigate the refill and reflood phases of the PWR loss-of-coolant accident. However, the technology being developed has general applicability to the measurement of two phase fluid flow. The objective of the ORNL program is to develop techniques and systems for measuring fluid flow in-core, deentrainment in the upper plenum, liquid fallback from the upper plenum into the core, and flow across the core-upper plenum interface. To attain this objective, liquid film thickness and velocity, two-phase flow velocity, void fraction and momentum flux must be measured.

Liquid film thickness and film velocity measurement systems are being implemented utilizing concepts developed at Lehigh University.[1,2] Film sensor development at ORNL is limited to

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*Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, under Interagency Agreements DOE 40-551-75 and 40-552-75 with the U.S. Department of Energy under Contract W-7405-eng-26 with the Union Carbide Corporation.

adaptation of the present techniques to the environment of a refill/reflood facility. Electrical impedance sensors have been developed for measurement of two-phase flow local velocities and void fractions for a variety of in-vessel geometries. Techniques using commercially available hardware have been developed which make it possible to measure transient differential pressures (DP) with long sensing lines. These techniques are being applied to low-range DP measurements within the upper end box and across the end box tie-plate. A unique and innovative drag body is being developed for upper end box momentum flux measurements. The drag body is an integral part of the end box and introduces no disturbance to flow.

REFERENCES

1. Chen, John C., et al, "Investigation of Post-CHF Heat Transfer for Water-Cooled Reactor Application and Development of Two-Phase Flow Instrumentation, Progress Report April 1, 1977 to June 30, 1977," LU-NUREG-PR771.
2. Chen, John C., et al, "Investigation of Post-CHF Heat Transfer for Water-Cooled Reactor Application and Development of Two-Phase Flow Instrumentation, Progress Report January 1, 1978 to March 31, 1978," LU-NUREG-PR781.
3. Del Tin, G. and Negrini, A., "Development of the Electrical Impedance Probes for Void Fraction Measurements in an Air-Water Mixture," 2nd Multi-Phase Flow and Heat Transfer Symposium-Workshop, Miami Beach, Florida, April 16-18, 1979.
4. Carrard, G. and Ledwidge, T. J., "Measurements of Slip Distribution and Average Void Fraction in an Air-Water Mixture," Progress in Heat and Mass Transfer, Vol. 6, (Proceedings of the International Symposium on Two-Phase Systems 1971), Hestroni, et al, Editor, pp. [illegible]-418, Pergamon Press, Oxford 1972.

ORNL WS-13668

UNION CARBIDE ORNL

THE OVERALL OBJECTIVES OF THE INTERNATIONAL 2D/3D REFILL AND REFLOOD PROGRAM

- TO STUDY THE STEAM BINDING EFFECT DURING REFLOOD FOR VARIOUS ECCS COMBINATIONS
- TO STUDY THE REFLOOD FLOW DISTRIBUTION (CHIMNEY EFFECT) IN A HEATED CORE
- TO STUDY THE FLOW HYDRODYNAMICS IN THE CORE, DOWNCOMER AND UPPER PLENUM DURING REFILL AND REFLOOD

ORNL WS-13557

A VARIETY OF MEASUREMENT METHODS HAVE BEEN UTILIZED IN THE ORNL DEVELOPMENT OF TRANSIENT TWO-PHASE FLOW INSTRUMENTATION FOR HIGH-TEMPERATURE IN-VESSEL MEASUREMENTS

- LONG-LINE DIFFERENTIAL PRESSURE FOR UPPER END BOX TIE-PLATE
- DRAG BODY MOMENTUM FLUX FOR UPPER END BOX TIE-PLATE
- THIN LIQUID FILMS
  - THICKNESS UTILIZING AN ELECTRICAL CONDUCTANCE SENSOR
  - VELOCITY UTILIZING AN ELECTROLYSIS POTENTIAL SENSOR
- VELOCITY AND VOID FRACTION UTILIZING ELECTRICAL IMPEDANCE SENSORS
  - IN-CORE
  - FREE FIELD

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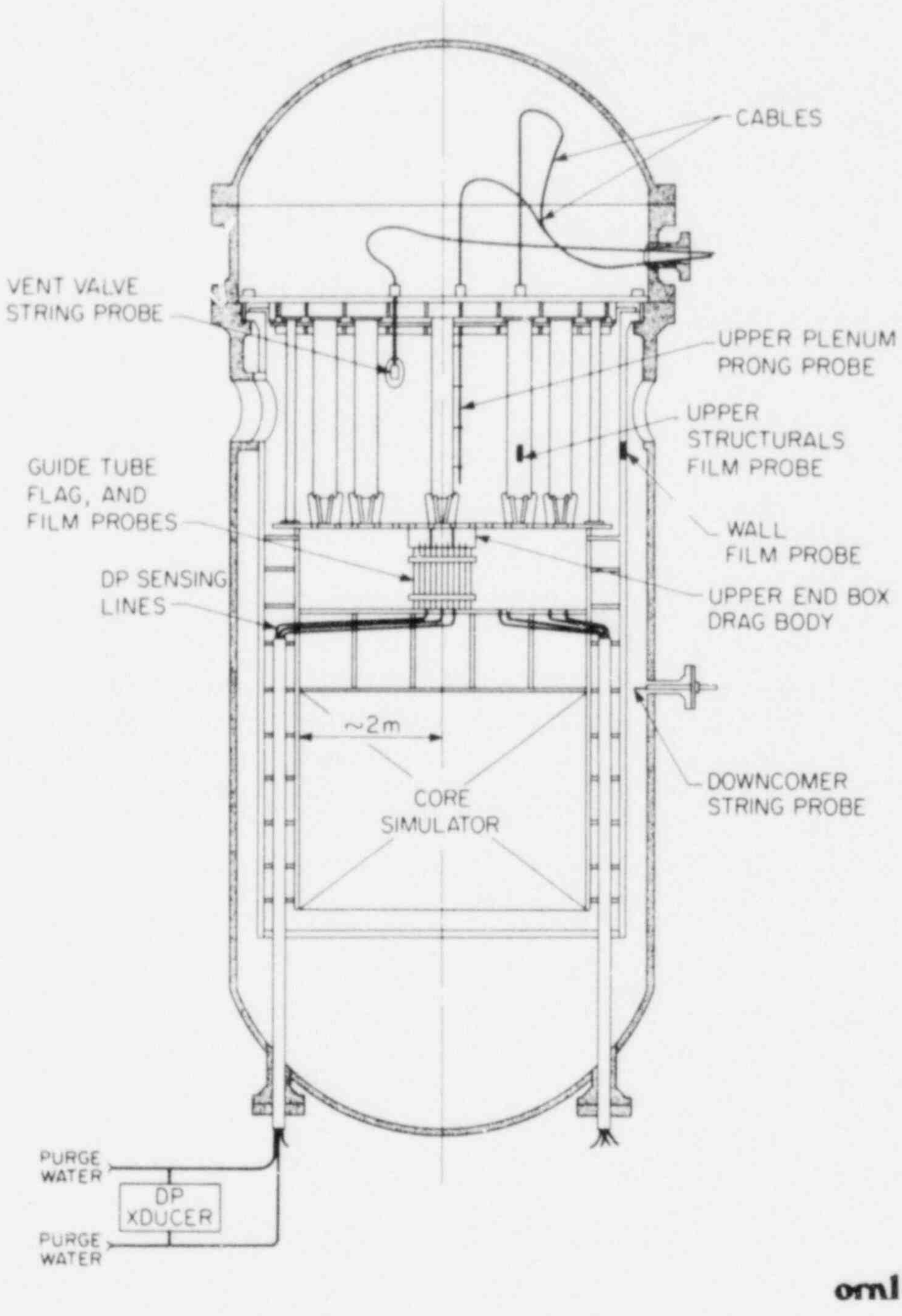
ORNL-DWG 80-18433

IN-VESSEL SENSORS OF MANY DIFFERENT GEOMETRIES HAVE BEEN DEVELOPED



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ORNL WS-13593

TECHNIQUES WERE DEVELOPED FOR LONG LINE, LOW RANGE, TRANSIENT DIFFERENTIAL PRESSURE MEASUREMENTS

SPECIFIED	ACHIEVED
LINE LENGTH 15 METERS	15 METERS CONTINUOUS LIQUID PURGED
827 KPa, 170°C	827 KPa, 170°C
0.2 TO 20 IN-H_2O (0.05 TO 5 KPa)	1 TO 20 IN-H_2O (0.25 TO 5 KPa)
3 Hz BANDWIDTH	10 Hz
NEGLIGIBLE VIBRATION AND WATER HAMMER EFFECT	MINIMIZED BY HYDRAULIC RESISTANCE
LOWEST POSSIBLE ERROR	$<$ 0.3 IN H_2O (0.075 KPa)

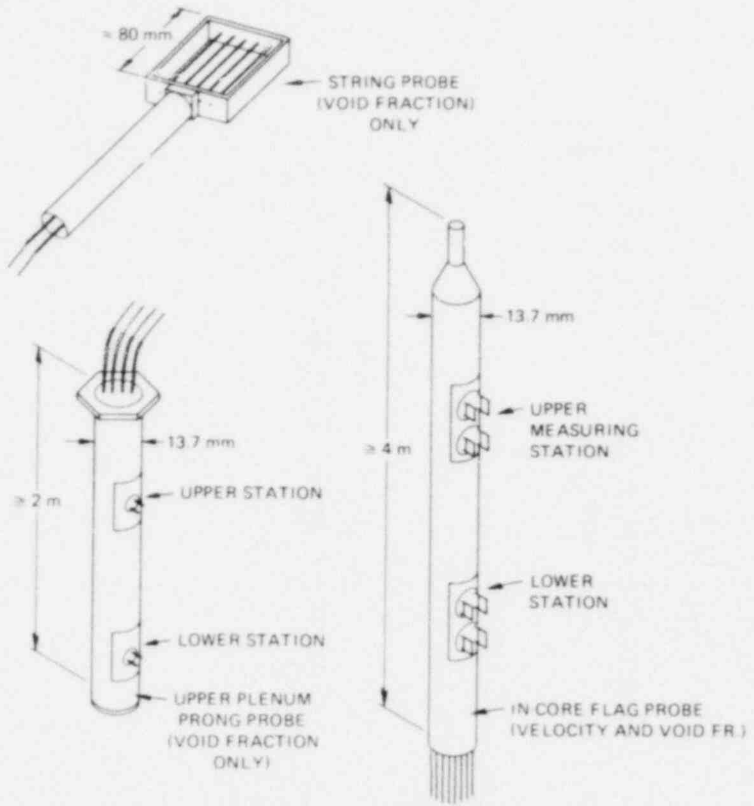
ORNL WS-13594

A UNIQUE AND INNOVATIVE DRAG BODY HAS BEEN DEVELOPED TO MEASURE TRANSIENT MOMENTUM FLUX AT THE CORE/UPPER PLENUM INTERFACE WITHOUT INTRODUCING OBSTRUCTION TO THE FLOW

- USES EXISTING END BOX STRUCTURE
- STRAIN GAGE BASED, BIDIRECTIONAL
- HIGH SENSITIVITY, 1/1000
- HIGH SIGNAL/NOISE RATIO
- HIGH RESONANT FREQUENCY, ~100 Hz
- MAINTAINS SENSITIVITY AND STABILITY
  - HIGH TEMPERATURES, TO 315°C
  - TRANSIENT FLOW CONDITIONS (CYCLIC LOADING)
  - THERMAL SHOCK DUE TO QUENCH

ORNL-DWG 80-14283R

THREE TYPES OF IMPEDANCE PROBES HAVE BEEN DEVELOPED FOR MEASUREMENT OF HIGH-TEMPERATURE STEAM/WATER TWO-PHASE FLOWS IN REACTOR SAFETY EXPERIMENTS



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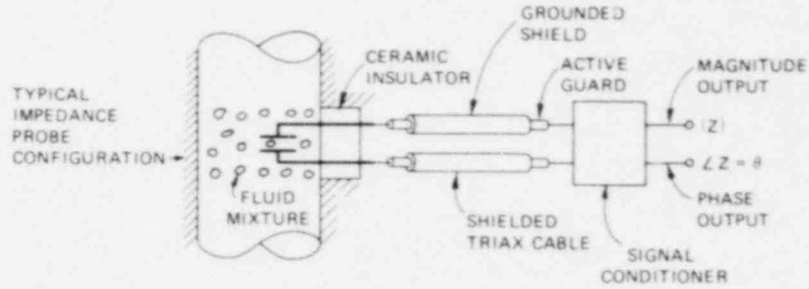
ORNL-DWG 80-14284R

A UNIQUE COMBINATION OF HIGH-TEMPERATURE SENSOR TECHNOLOGY, SOPHISTICATED SIGNAL CONDITIONING ELECTRONICS AND SIGNAL ANALYSIS METHODS MAKES POSSIBLE AN IN-VESSEL VOID FRACTION MEASUREMENT



- IMPEDANCE $Z = R + jX_c = |Z| e^{j\theta}$
- ADMITTANCE $Y = \frac{1}{Z} = G + jWC = |Y| e^{-j\theta}$
- MIXTURE CONDUCTANCE $G_{mix} = \frac{\sigma_m}{\beta}$
- MIXTURE CAPACITANCE $C_{mix} = \frac{\epsilon_m}{\beta}$
- FLUID PROPERTIES
  - σ_ℓ = CONDUCTIVITY OF LIQUID (X/cm)
  - σ_g = CONDUCTIVITY OF GAS
  - σ_m = AVG MIXTURE CONDUCTIVITY
  - ϵ_ℓ = PERMITTIVITY OF LIQUID (F/cm)
  - ϵ_g = PERMITTIVITY OF GAS
  - ϵ_m = AVG MIXTURE PERMITTIVITY
- VOID FRACTION (α) RELATIONSHIPS

$$\left.\begin{array}{l}\sigma_m = f(\sigma_\ell, \sigma_g, \alpha)\\ \epsilon_m = f(\epsilon_\ell, \epsilon_g, \alpha)\end{array}\right\}\ \text{SAME FUNCTION}$$

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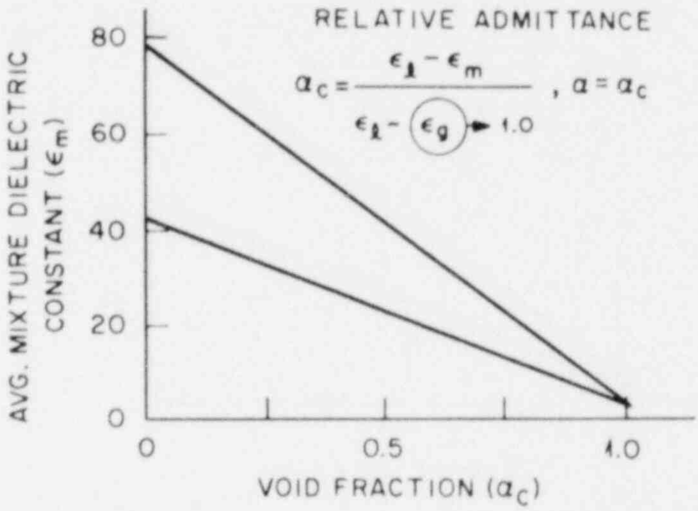
ORNL-DWG 80-14285A

FOR FLOW REGIMES WITH NO DISPERSED DROPLETS THE RELATIONSHIP BETWEEN VOID FRACTION AND THE AVERAGE MIXTURE DIELECTRIC CONSTANT CAN BE DESCRIBED BY A LINEAR MODEL



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ORNL WS-13595

A METHOD HAS BEEN DEVELOPED TO IDENTIFY AND COMPENSATE FOR FLOW REGIME EFFECTS ON IMPEDANCE MEASUREMENT OF VOID FRACTION BY USING WEINER'S EQUATION

$$\alpha = 1 - [1 - \alpha_c] \left[\frac{\epsilon_\ell + n}{\epsilon_\ell (1 - \alpha_c) + (\alpha_c + n)} \right]$$

α = ACTUAL OR TOTAL VOID FRACTION

α_c = RELATIVE ADMITTANCE VOID FRACTION (ASSUMING LINEAR RELATIONSHIP)

n = EMPIRICAL DISTRIBUTION FACTOR

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ORNL WS-13596

ASSUMPTIONS REQUIRED IN THE PROPOSED FLOW REGIME COMPENSATION METHOD FOR IMPEDANCE PROBE

- THAT WEINER'S MODEL APPLIES TO THE REGION OF THE SENSOR VOLUME WHERE THE FLOW IS DISPERSED
- THAT THE DISPERSED REGION IS ELECTRICALLY IN PARALLEL WITH A LIQUID BRIDGE
- GAS IS THE CONTINUOUS PHASE AND LIQUID IS DISPERSED PHASE

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ORNL WS-13597

MEASURED IMPEDANCE PHASE ANGLE DATA IS USED TO DETECT THE PRESENCE OF DISPERSED DROPLETS AND TO DETERMINE THE DISTRIBUTION FACTOR n

- α_c AND THE LOSS ANGLE γ ARE MEASURED VALUES
- CALCULATE γ_{MAX} (LOSS ANGLE WITH NO DISPERSED DROPLETS)
- CALCULATE $S = 1 - [\tan\gamma / \tan\gamma_{MAX}]$
- CALCULATE:

$$n = \left[\frac{A(1-\alpha_c)^{0.3}}{S}\right] + 1$$

  A = PROBE CALIBRATION CONSTANT
- CALCULATE α USING WEINER'S EQUATION

$$\alpha = 1 - [1 - \alpha_c]\left[\frac{\epsilon_\ell + n}{\epsilon_\ell(1-\alpha_c) + (\alpha_c + n)}\right]$$

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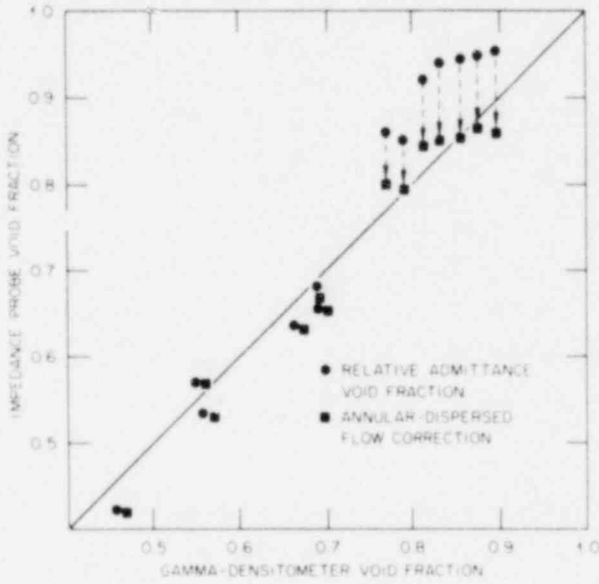
ORNL-DWG 80-14290R

SCTF FLAG PROBE VOID FRACTION RESULTS ILLUSTRATE THE EFFECT OF THE DISPERSED FLOW CORRECTION. AIR/WATER TESTS APRIL 1980



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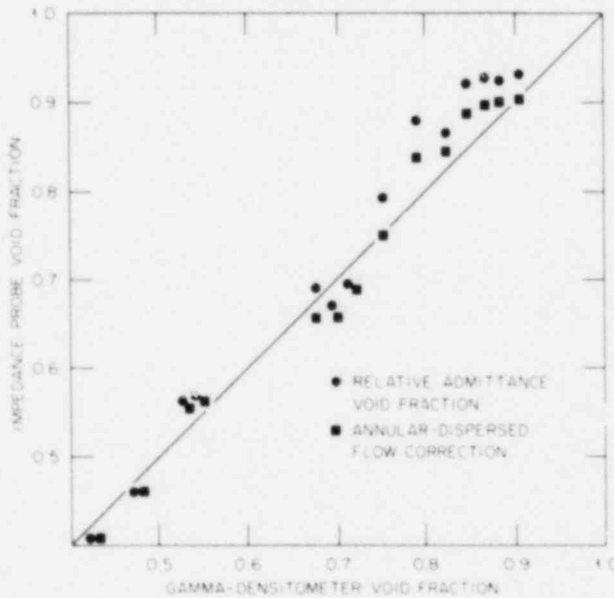
ORNL-DWG 80-14291R

SCTF PRONG PROBE vs GAMMA DENSITOMETER VOID FRACTIONS FROM STEADY STATE AIR/WATER TESTS



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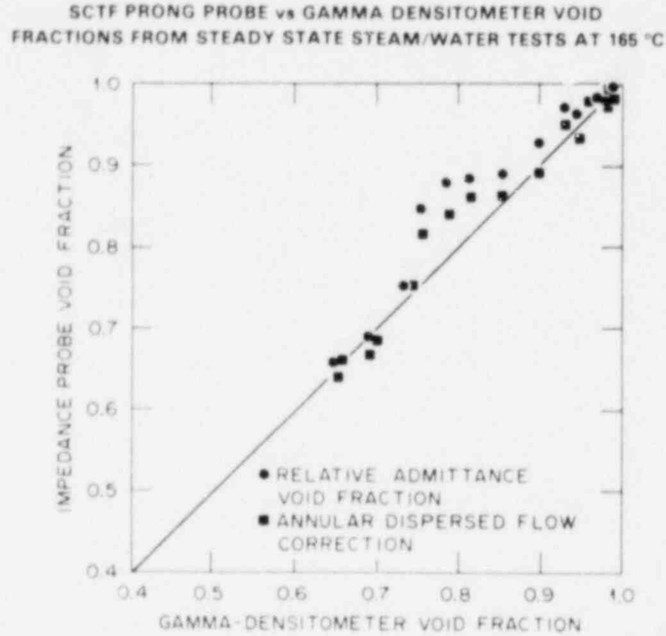
ORNL-DWG 80-14292R



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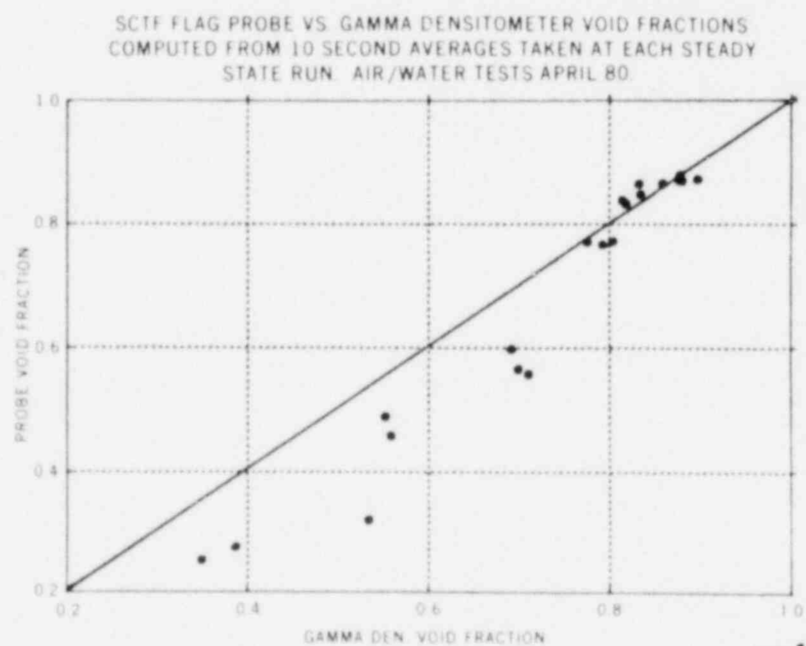
ORNL-DWG 80-18502



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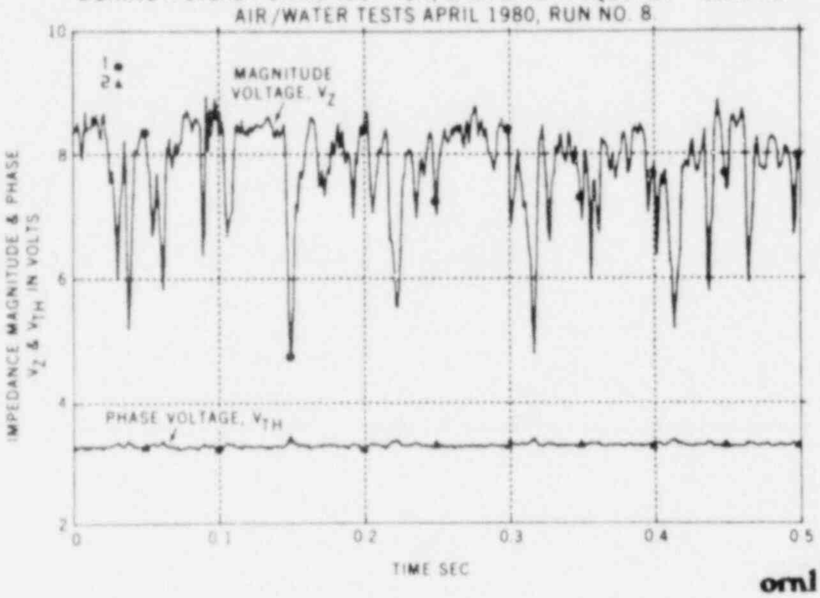
ORNL - DWG 80-18501

SCTF FLAG PROBE SIGNALS VS. TIME
DURING A STEADY STATE TEST RUN, SAMPLING FREQUENCY = 1000 HZ.
AIR/WATER TESTS APRIL 1980, RUN NO. 8.



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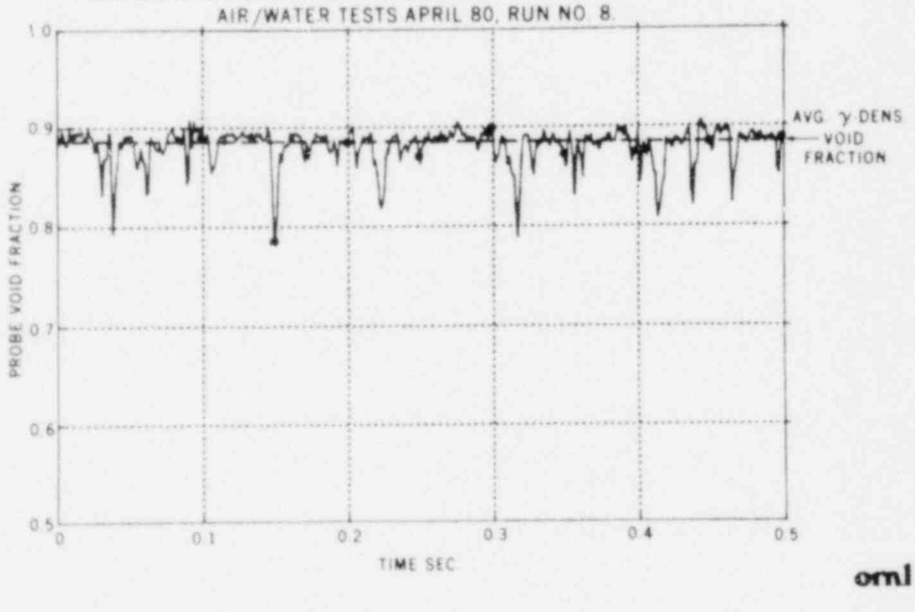
ORNL - DWG 80-18500

SCTF FLAG PROBE VOID FRACTION VS. TIME
DURING STEADY STATE TEST RUN, SAMPLING FREQUENCY = 1000 HZ.
AIR/WATER TESTS APRIL 80, RUN NO. 8.



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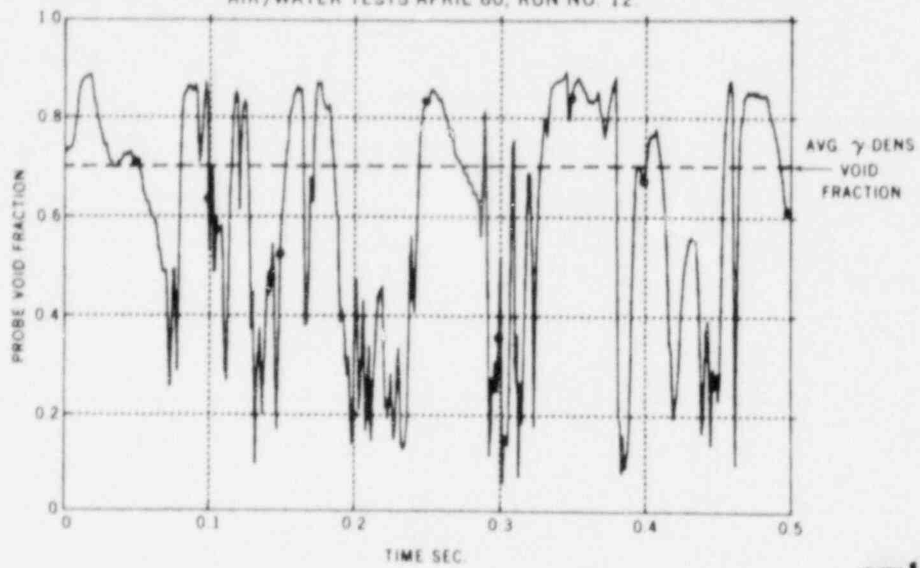
ORNL-DWG 80 18502

SCTF FLAG PROBE VOID FRACTION VS. TIME
DURING STEADY STATE TEST RUN, SAMPLING FREQUENCY = 1000 HZ.
AIR/WATER TESTS APRIL 80, RUN NO. 12



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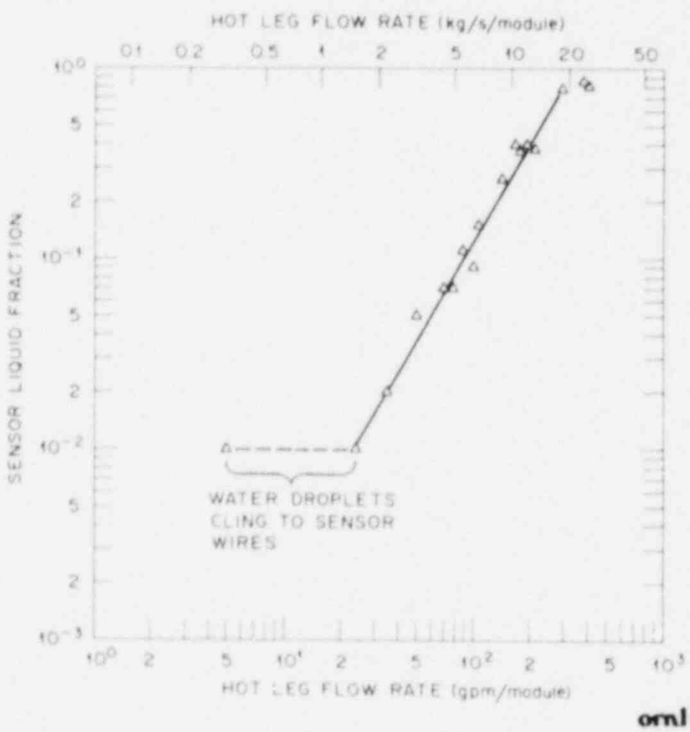
ORNL-DWG 80-18401

LIQUID FRACTION JUST ABOVE TIE-PLATE MEASURED WITH STRING PROBE IN 3-MODULE AIR-WATER IDL, HOT LEG INJECTION-NO AIR FLOW.



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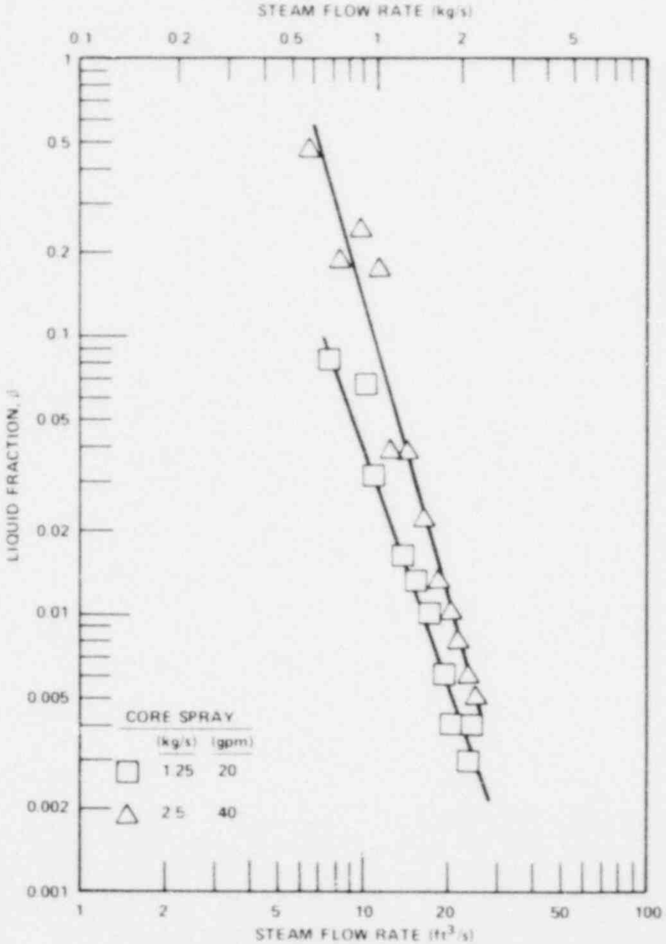
ORNL-DWG 80-18556

STRING PROBE MEASUREMENT RESULTS ABOVE TIE-PLATE IN STEAM/WATER IDL INDICATE HIGH SENSITIVITY IN HIGH VOID FRACTION FLOWS



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ORNL WS-13598

SUMMARY AND CONCLUSION

ORNL HIGH-TEMPERATURE IN-VESSEL INSTRUMENTATION DEVELOPMENTS INCLUDE:

- A CERMET INSULATOR-SEAL THAT IS THERMAL SHOCK RESISTANT
- A VARIETY OF HIGH-TEMPERATURE SENSOR FABRICATION TECHNIQUES
- LONG-LINE TRANSIENT DP MEASUREMENT
- UNIQUE HIGH-SENSITIVITY END BOX DRAG BODY WITH NO FLOW OBSTRUCTION
- FILM THICKNESS SENSOR AND ELECTRONICS
- FILM VELOCITY SENSOR AND ELECTRONICS
- IMPEDANCE PROBES IN THREE DIFFERENT GEOMETRIES WITH ASSOCIATED ELECTRONICS

VOID FRACTION MEASUREMENT BY IMPEDANCE PROBES HAVE BEEN DEMONSTRATED TO HAVE

- EXCELLENT AGREEMENT WITH GAMMA DENSITOMETER
- HIGH SENSITIVITY IN HIGH VOID FRACTION S/W AND A/W FLOWS
- HIGH FREQUENCY RESPONSE

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OVERVIEW OF 2D/3D INSTRUMENTATION
DEVELOPED AT EG&G IDAHO, INC.

Presented at
The Eight Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

R. E. Rice
EG&G Idaho, Inc.

Idaho National Enginnering Laboratory
Idaho Falls, Idaho 83415

OVERVIEW OF 2D/3D INSTRUMENTATION DEVELOPED AT EG&G IDAHO

R. E. Rice
EG&G Idaho, Inc.

Instruments provided by EG&G Idaho Support the overall objectives of the 2D/3D Program. Test results further the understanding of full scale-accident phenomena, such as steam binding, end-of-bypass, upper plenum deentrainment, and multidimensional core reflood behavior. Results obtained in the Japanese Cylindrical Core and Slab Core Test Facilities (CCTF, SCTF) and the German Primary Coolant Loop (PKL) and Upper Plenum Test Facility (UPTF) are also used to assess the TRAC thermal-hydraulic Code. Four catagories of measurements are made: density, liquid inventory, local velocity, and mass flow rate.

Two generations of soft X-ray densitometers have evolved in our support of 2D/3D. The first system utilized combinations of low energy sources to provide a three-path density measurement in 75 to 150 mm spool pieces. These systems featured an LN_2-cooled germanium or silicon detector and pulse height discrimination to separate the combined signal into path densities. A second generation of low energy densitometers has also been developed to improve performance, reduce system costs, and to minimize maintenance requirements. Details of this measurement system will be presented in a companion paper.[1]

Liquid level measurements to date in 2D/3D have relied on conductivity probe devices,[2] which have provided a large volume of liquid level data to date on 2D/3D. A new system has been developed to improve liquid level results and provide a global indication of liquid inventory using a grid of optical wet/dry sensors. Details of this measurement will be presented in a companion paper.[3]

Several instrument systems are used on 2D/3D to collect local velocity data in single-phase flow. Applications include use of variable reluctance drag transducers and miniature turbine meter probes, both calibrated to yield single-phase local velocities. A new device has been developed to measure extremely low liquid velocities, as occurring at the core inlet during reflood. A water-cooled reference thermocouple is paired with a surface thermocouple penetrating into the free stream. The differential output is calibrated to yield accurate liquid velocities down to 0.025 m/s. Signals from "upstream" and "downstream" resistance thermometers are also compared to yield direction of flow.

The most difficult measurements to make accurately are of two-phase mass flow. Several sets of combined transducer outputs are utilized on the 2D/3D facilities to make these measurements on pipes of various configurations. The PKL , CCTF,[4] and SCTF have been provided 75 to 150 mm diameter spool pieces, each containing a full flow drag screen, turbine meter and a three-beam densitometer. The SCTF is being provided a 116 by 737 mm oval shaped spool piece with a four-target drag rake and a four-beam densitometer. The UPTF will be equipped with flow measurement stations in five 750 mm diameter pipes, each containing a six-target drag rake and a six-beam densitometer system. The development approach on these measurements is to verify the mass flow algorithms through two-phase testing. Two-phase flow accuracies of $\pm$15% full scale have been demonstrated for spool pieces up to 150 mm diameter.

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1. J. B. Colson, "Low Energy-Sodium Iodide Gamma Densitometer for 2D/3D Program," Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, MD, October 27-31, 1980.

2. R. C. Greninger, H. K. Meyer-Christians, CCTF Operation and Maintenance Manual Conductivity Liquid level Measurement Systems (CLLMS), EGG-3D-5046 (December 1979).

3. R. P Evans, B. L. Watson, "An Optical Liquid Level Detector for High Temperature/Pressure Water Environment," Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, MD, October 27-31, 1980.

4. G. H. McCarty, CCTF Operation and Maintenance Manual Instrumented Spool Piece and Downcomer Drag Disk Flow Measurement Systems, EGG-3D-5105 (March 1980).

OVERVIEW OF 2D/3D INSTRUMENTATION DEVELOPED AT EG&G IDAHO, INC.

by

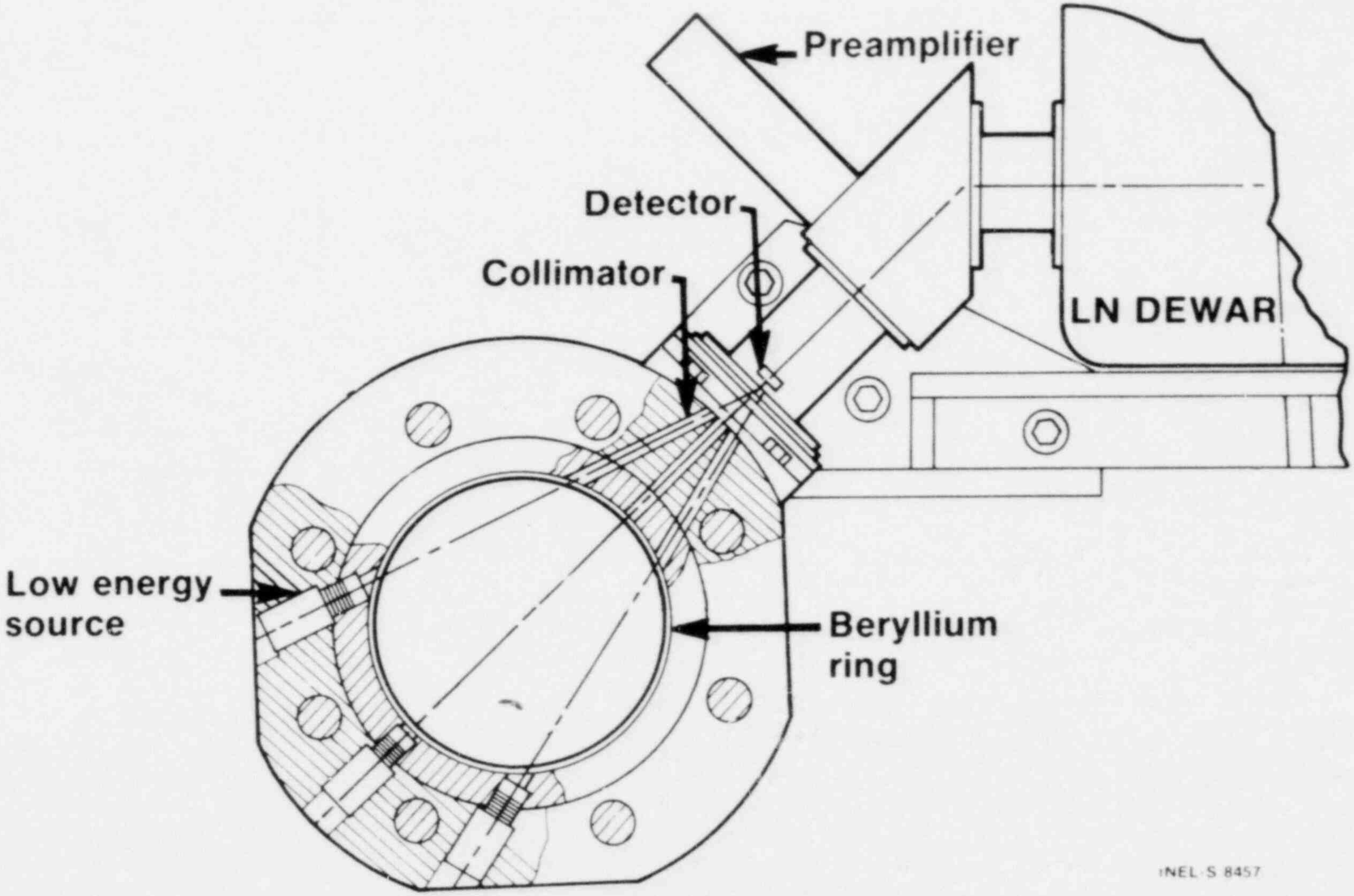
R.E. RICE

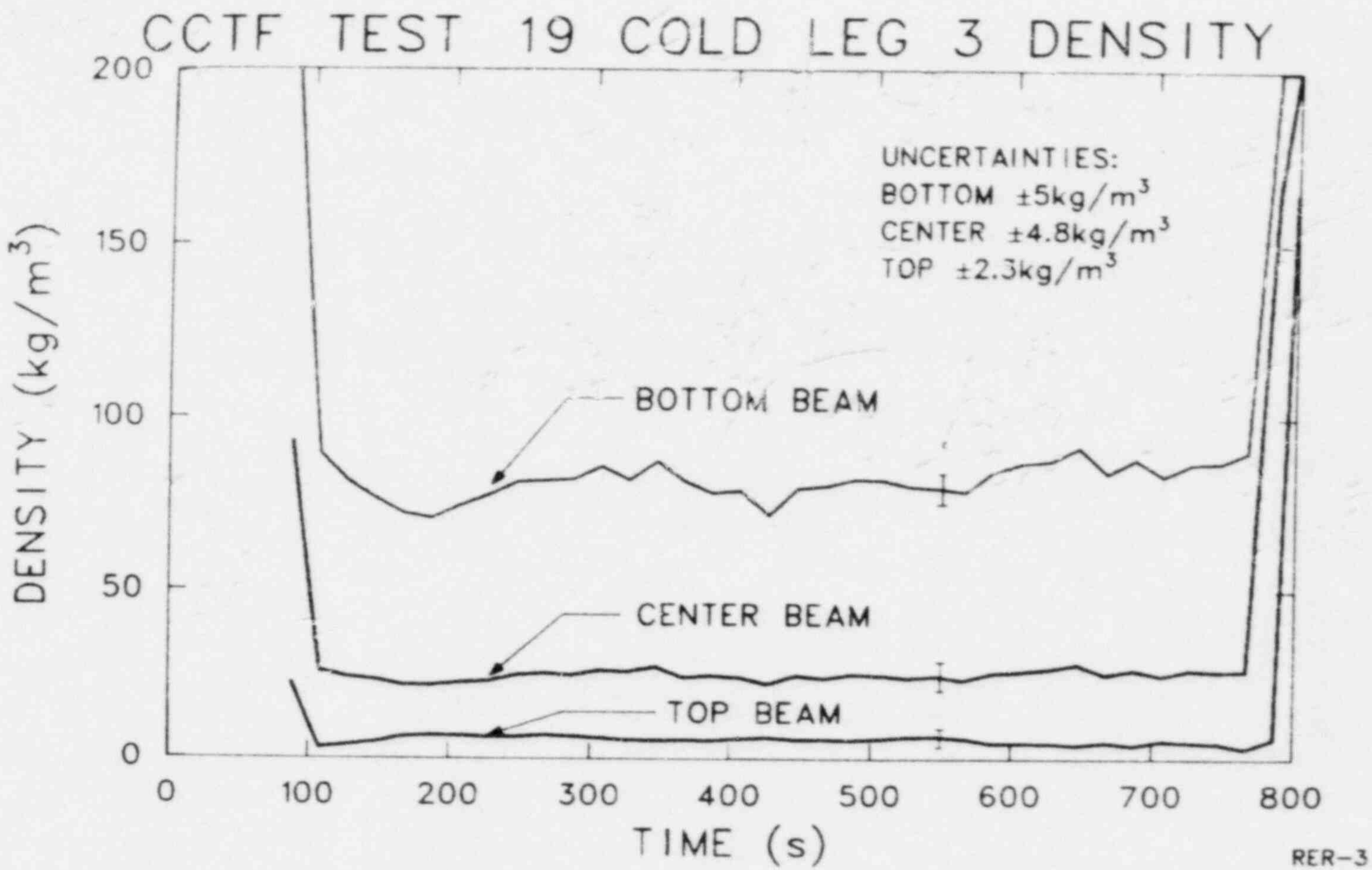
EG&G Idaho, Inc.

EG&G IDAHO SCOPE OF SUPPLY FOR 2D/3D PROGRAM

MEA. / FACILITY	DENSITY	LIQUID INVENTORY	LOCAL VELOCITY	MASS FLOW RATE
PKL (GERMANY)		92 CONDUCTIVITY PROBES	4 TURBO PROBES	4 PIPES 75 TO 100 mm
UPTF (GERMANY)		1200 OPTICAL SENSORS	67 TURBO PROBES	5 PIPES 750 mm
CCTF (JAPAN)		219 CONDUCTIVITY PROBES 272 OPTICAL SENSORS	4 DRAG DISKS 14 TURBO PROBES 4 COOLED TC	8 PIPES 150 mm
SCTF (JAPAN)	19 PATH DENSITIES	186 CONDUCTIVITY PROBES	3 DRAG DISKS 16 TURBO PROBES	2 PIPES 150 mm 1 ELLIPTICAL PIPE

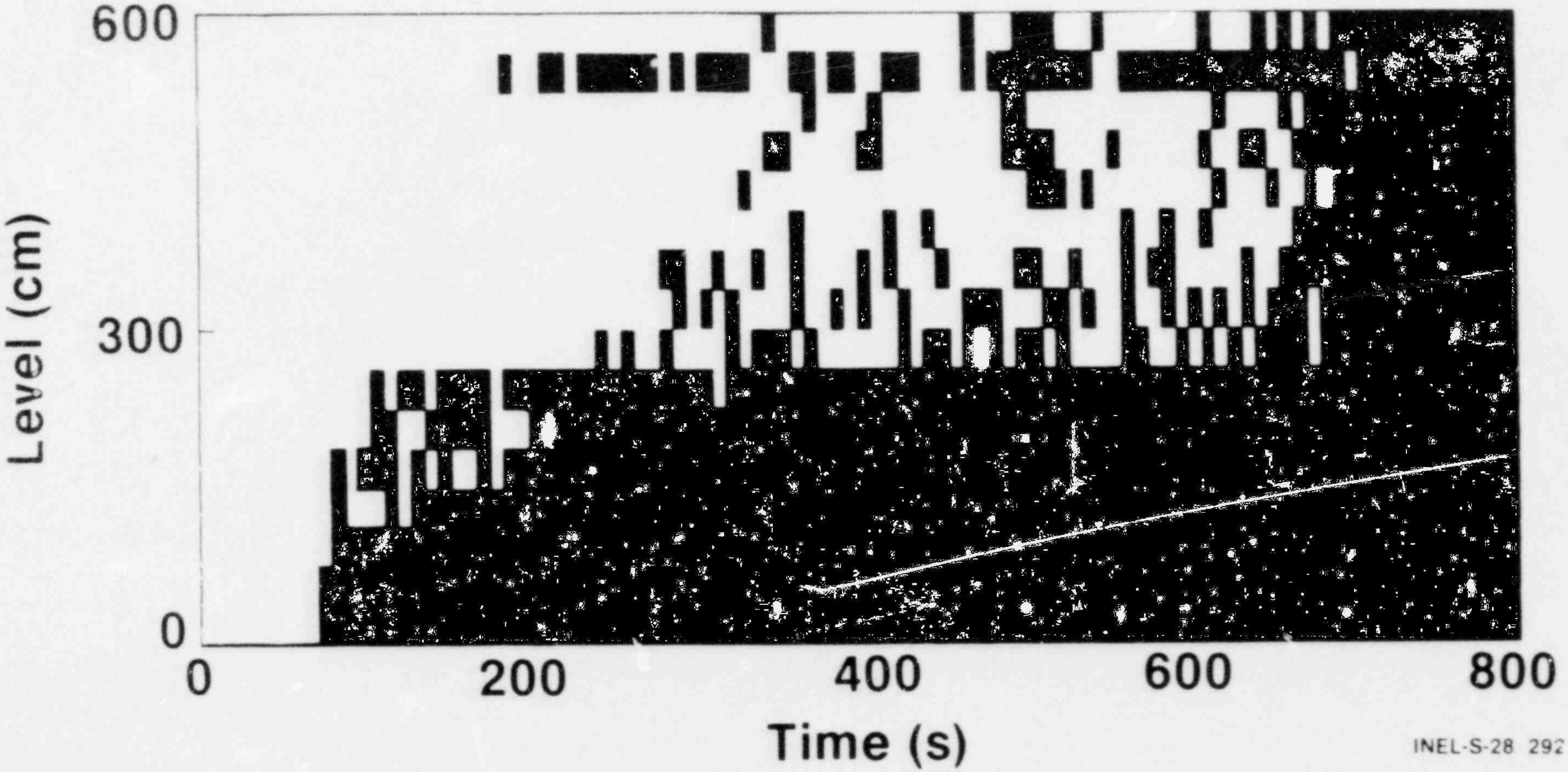
Three Beam Densitometer

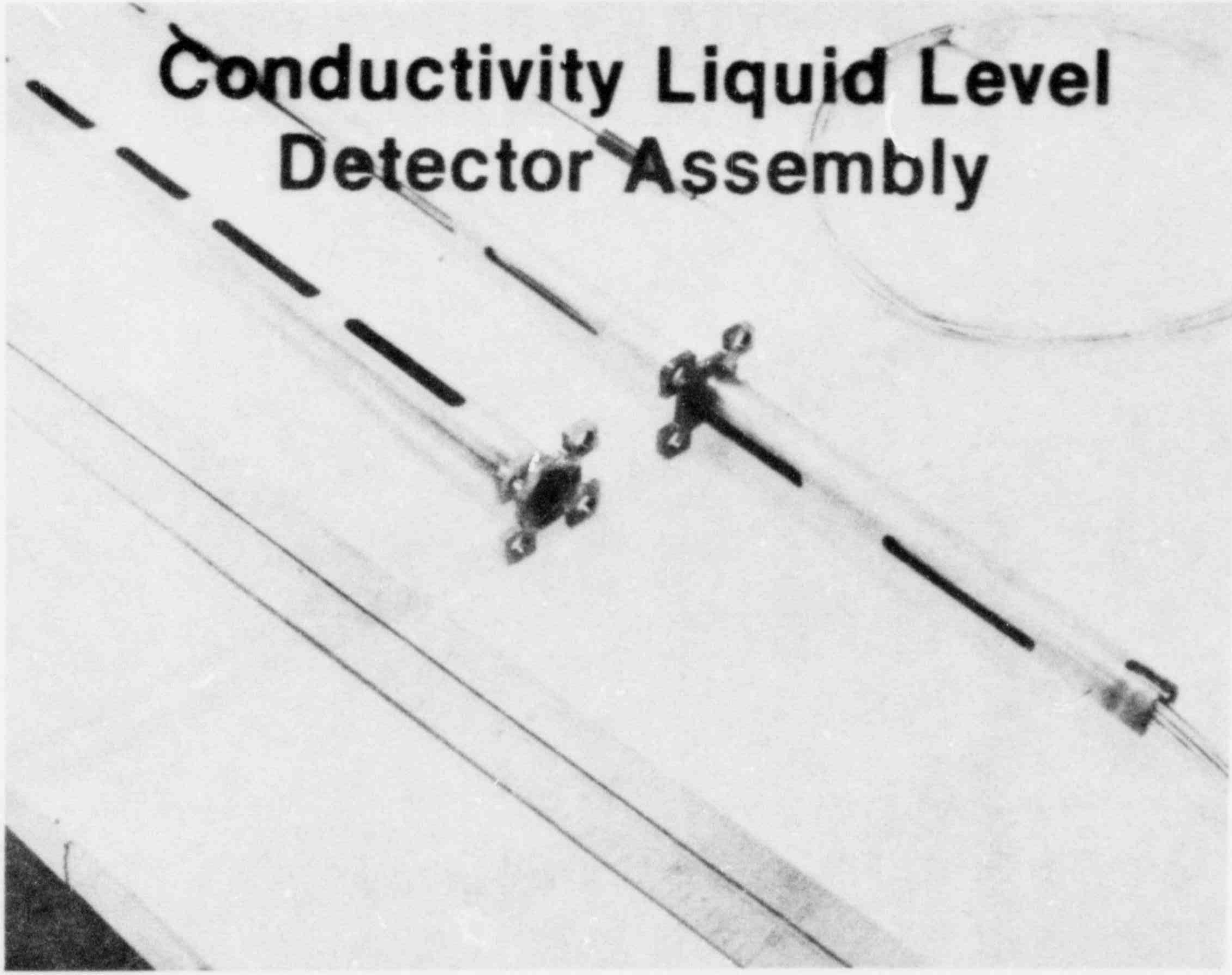




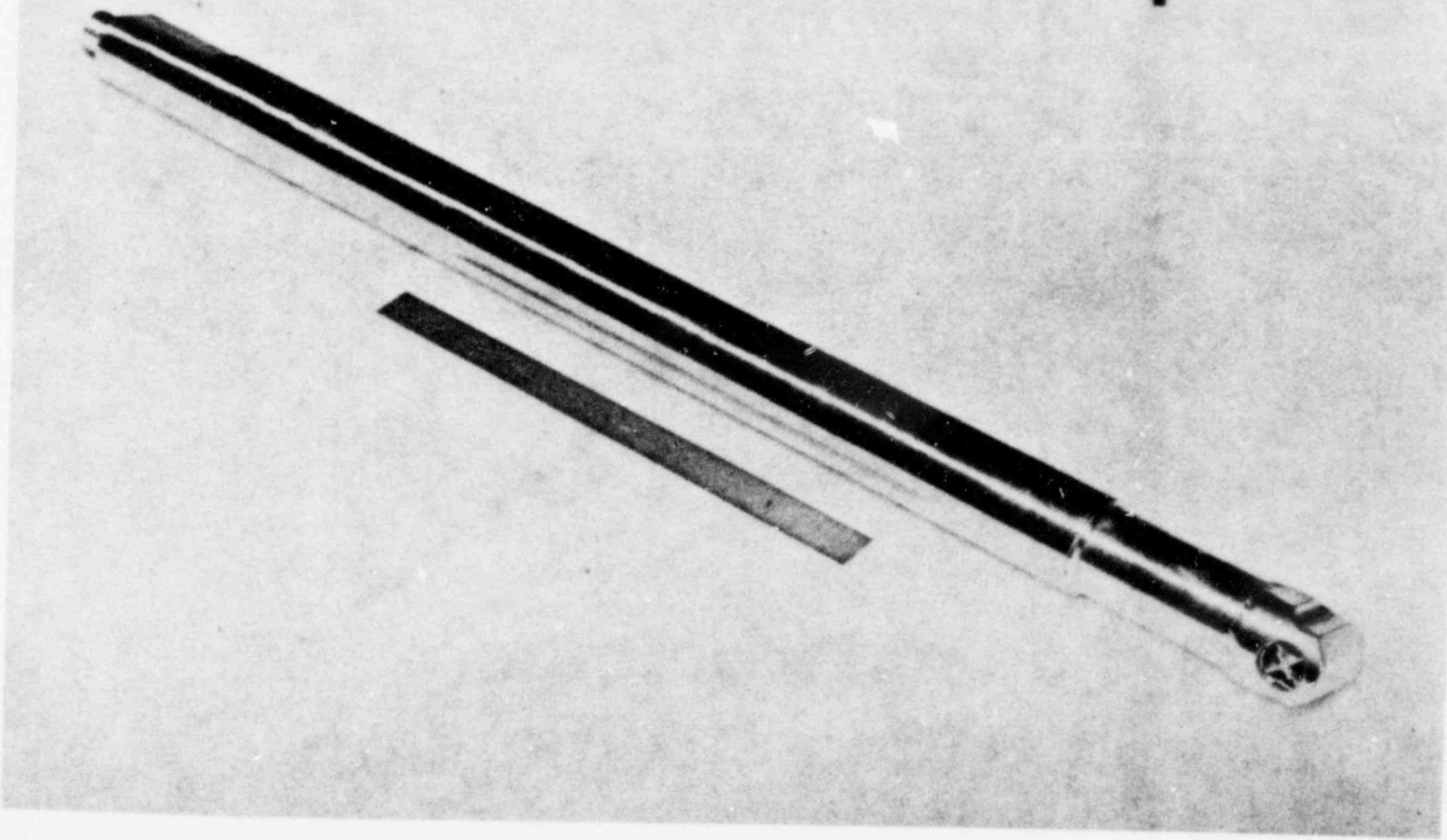
CCTF TEST 19 COLD LEG 3 DENSITY
UNCERTAINTIES:
BOTTOM ±5kg/m³
CENTER ±4.8kg/m³
TOP ±2.3kg/m³
BOTTOM BEAM
CENTER BEAM
TOP BEAM
DENSITY (kg/m³)
TIME (s)
0
50
100
150
200
0
100
200
300
400
500
600
700
800
RER-3

CCTF Test 12 Liquid Level
Core Wide Range

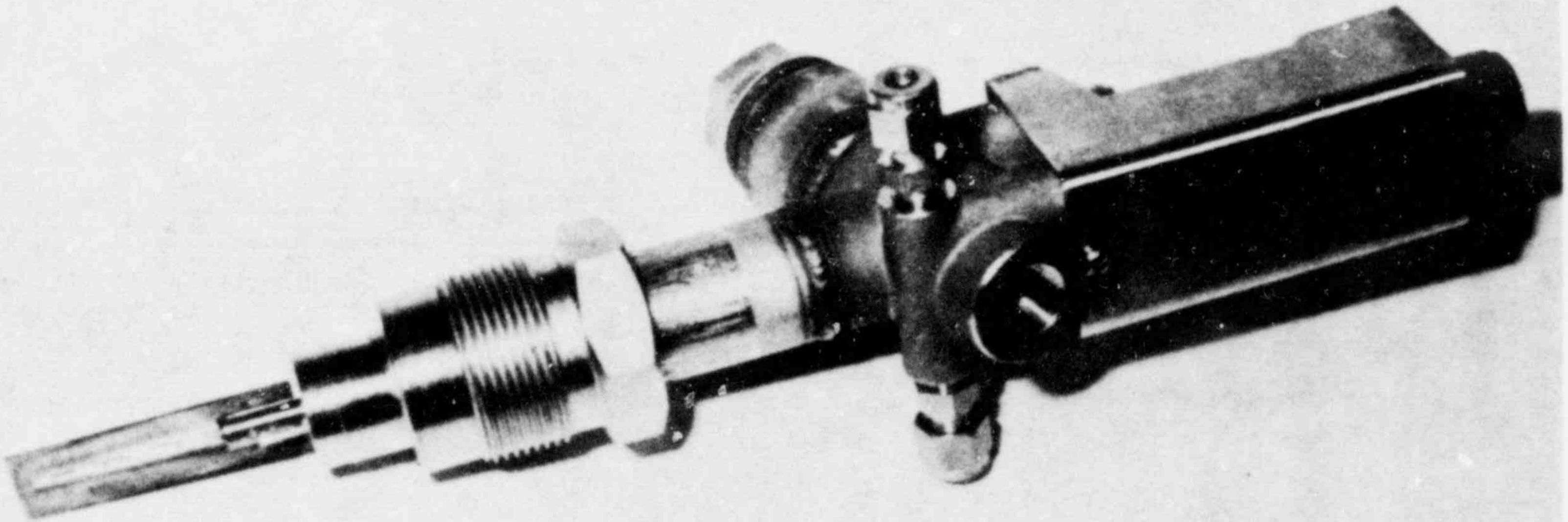



Conductivity Liquid Level
Detector Assembly

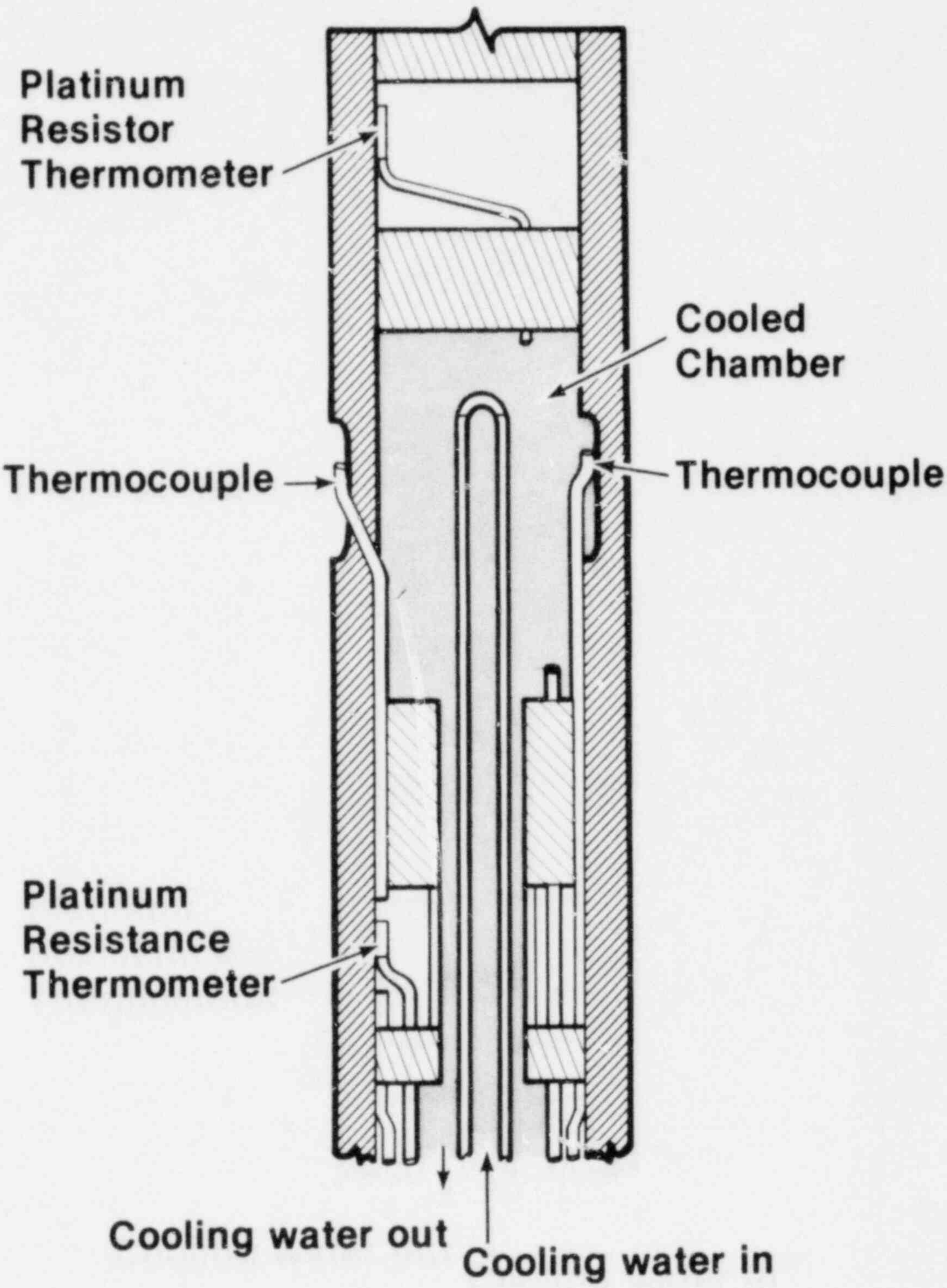
PKL Upper Plenum Turboprobe



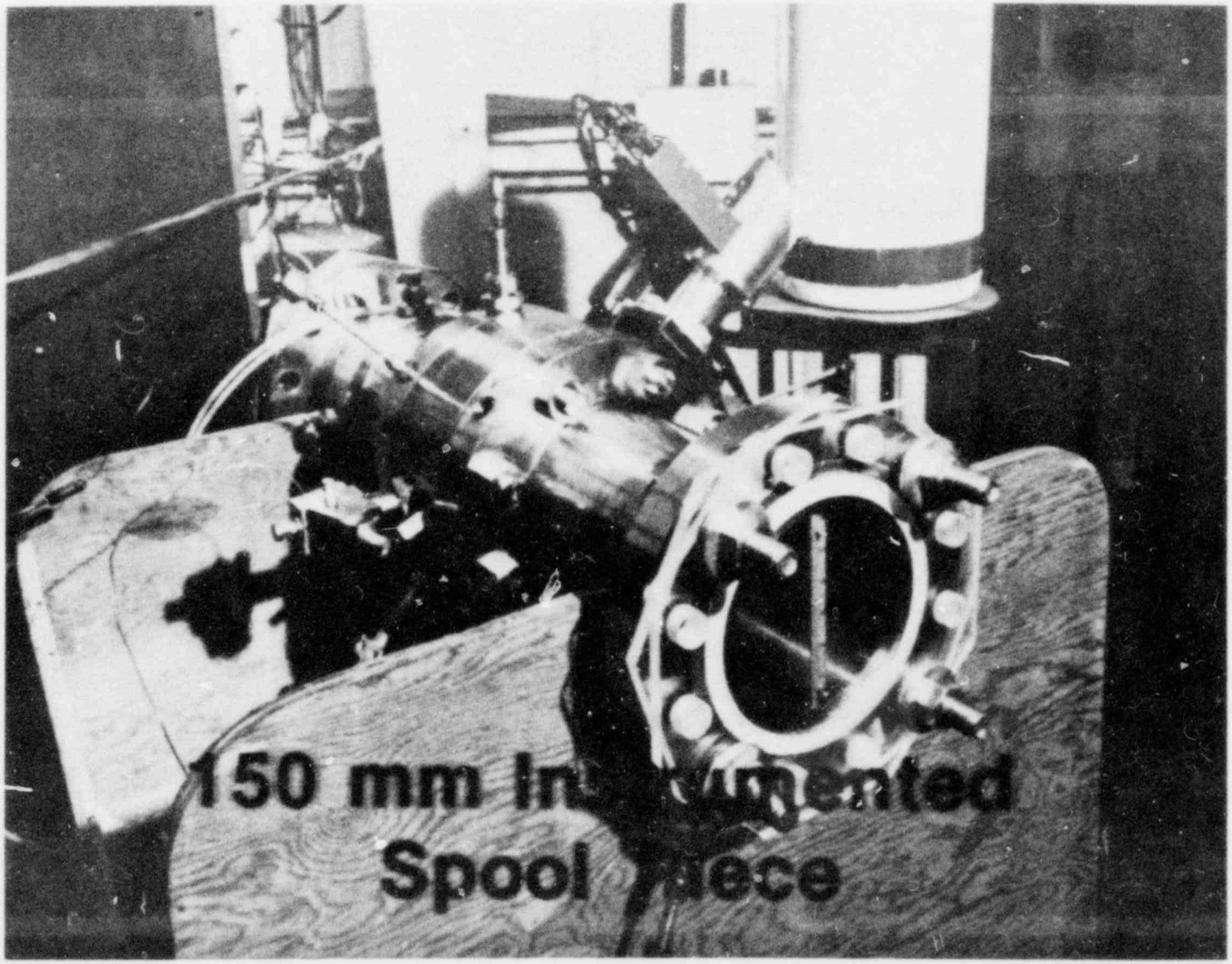
Variable Reluctance
Drag Transducer



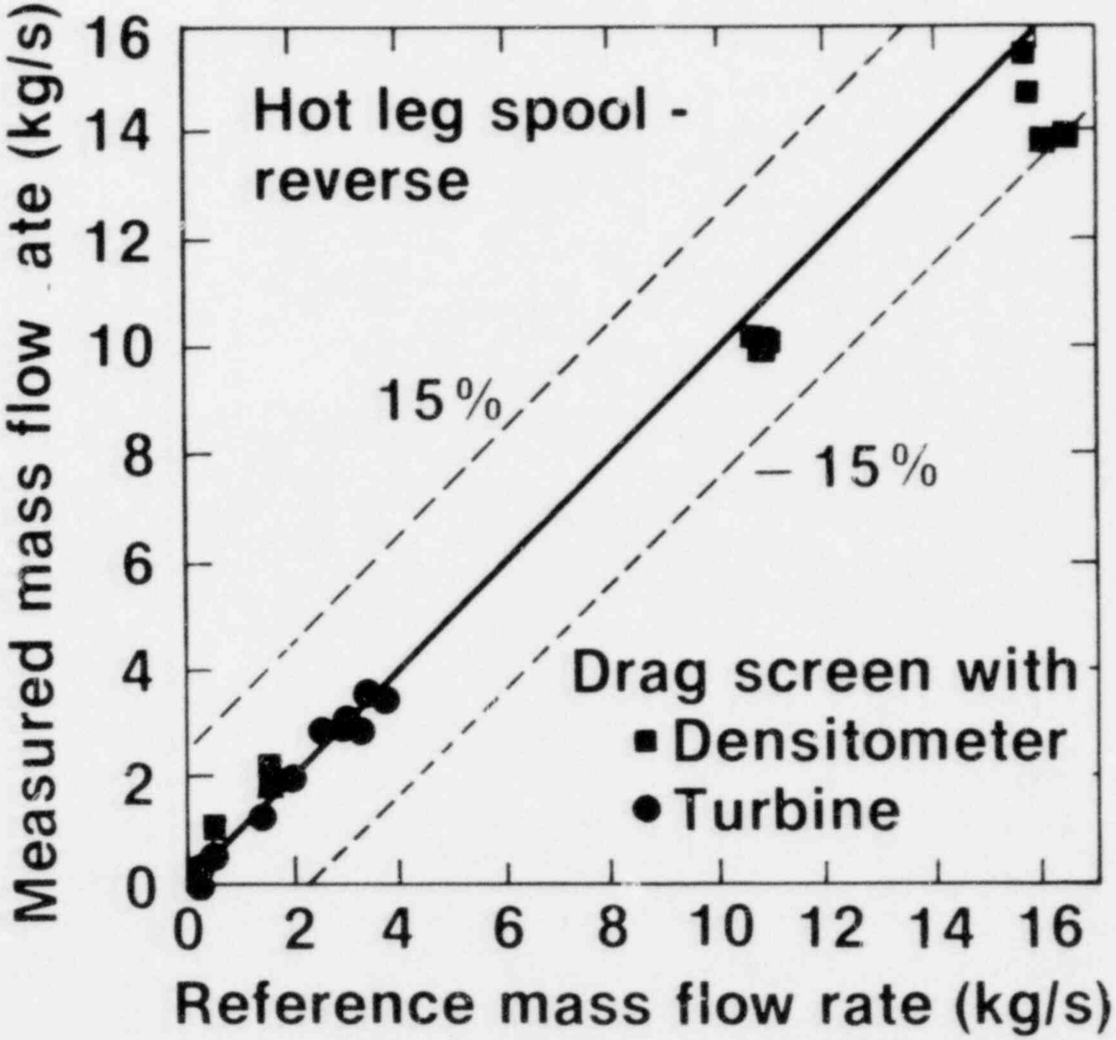
"Cooled Thermocouple Velocimeter"



INEL-S-28 238


150 mm
Spool

150-mm Spool Piece Two Phase Flow Test Results



INEL-S-19 784 Rev. 1

LOW ENERGY SODIUM IODIDE GAMMA DENSITOMETER FOR 2D/3D PROGRAM

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

J. B. Colson
R. R. Rohrdanz
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

LOW ENERGY SODIUM IODIDE GAMMA DENSITOMETER FOR 2D/3D PROGRAM

J. B. Colson
R. R. Rohrdanz
EG&G Idaho, Inc.

A gamma densitometer has been designed, fabricated, and tested for use in the Slab Core Test Facility (SCTF) in Tokai, Japan, to measure the coolant fluid density during the refill and reflood phases of a simulated loss-of-coolant accident. The SCTF is part of the 2D/3D Program in support of NRC objectives as described in a companion paper.[1] The design is based on the attenuation of a gamma beam as a function of density.

The densitometer was designed to maximize accuracy at high void fraction, low density two-phase flow in the range of 0.7 to 70 kg/m^3 and measure up to 1000 kg/m^3 at reduced performance. Environmental criteria included a severe environment of superheated steam up to 1073 K, 1.2 MPa at the pressure boundary, a 10 gauss 50 Hz magnetic field, and a 333 K thermal shock. Other design criteria included 350 ms response time, SI (metric) hardware design, less than 2 mr/hr surface radiation from the source, 300 series stainless steel construction, and restricted envelope limitations.

Two models were designed for application at four different regions of the test facility for a total of 23 density measurements. One model was designed to measure the fluid density of the coolant in the core. As such, the gamma beam was collimated to be 2 mm wide by 39 mm high and focused between heater rods in adjacent simulated fuel bundles. The beam path through the fluid is about 250 mm long. The other model is designed to measure the density in the core end box, the upper plenum, and the hot leg piping where the area is more open. In these locations a circular gamma beam collimated to 25 mm in diameter is used.

To meet the design criteria, a low energy source was selected to have the best sensitivity and accuracy over the desired operating range of densities. The Am-241 gamma ray at 60 keV was selected. This source has the added advantages of long half life, hence, no significant decay and is easily shielded to a low radiation level. Since Am-241 is essentially a surface source due to its self shielding, the maximum usable source strength is about one curie. This provides a maximum count rate to the detector of about 500,000 counts/s. This high count rate is necessary to minimize statistical error in the measurement.

To detect and process this count rate, a high speed system was developed using a feedback stabilized photomultiplier tube in conjunction with a sodium iodide detector. The feedback circuit controls the photomultiplier high voltage to stabilize the pulse height output. High speed electronics are used to minimize the effects of dead time in the counting amplifier. The amplifier output is passed through a single channel analyzer into a scaler providing a digital output summing the counts in a 300 ms interval. The outputs of all 23 channels are combined into a single serial RS-232C format to be processed by the facility data acquisition system.

This design has eliminated the need for liquid nitrogen cooling of previous low energy designs using Ge/Si detectors. Also no dead time correction is needed and the use of direct digital processing has eliminated any error due to analog-to-digital and digital-to-analog conversion.

The statistical error due to the random decay of the source and the calibration errors contribute the major portion of the measurement error. The statistical error is calculated to be less than 1 kg/m^3 + 0.3% of reading for a 95% confidence level over the range of 0.7 to 70 kg/m^3. The calibration errors were shown to be less than 1.7 kg/m^3 + 0.7% of reading.

REFERENCE

1. R. E. Rice, "Overview of 2D/3D Instrumentation Developed at EG&G Idaho, Inc.," Eight Water Reactor Safety Research Information Meeting, Gaithersburg, MD, October 27-31, 1980 (preoccdings to be published).

LOW ENERGY SODIUM IODINE GAMMA DENSITOMETER FOR 2D/3D PROGRAM

by

J.B. COLSON

EG&G Idaho, Inc.

OUTLINE

THEORY

GEOMETRY

ELECTRONICS

HARDWARE

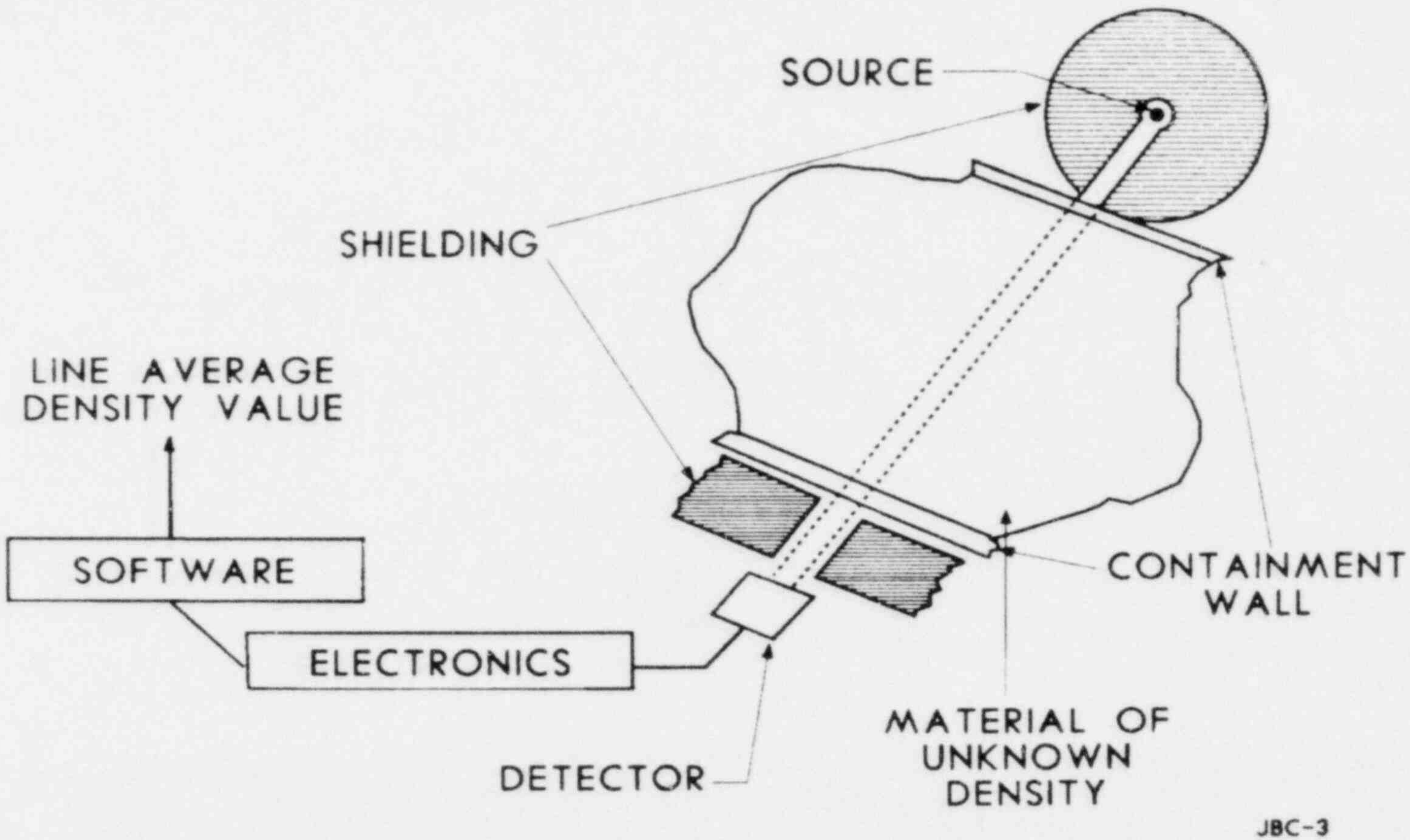
STATISTICAL ERROR

CALIBRATION DATA

CONCLUSIONS

JBC-2

GAMMA DENSITOMETER CONCEPT



DENSITOMETER THEORY

$$I = \frac{EAS}{R^2} A_T$$

I = DETECTOR COUNT RATE, PULSES/S

E = DETECTOR EFFICIENCY

A = COLLIMATION AREA, m^2

S = SOURCE STRENGTH, PHOTONS/S/STERADIAN

R = SOURCE TO DETECTOR DISTANCE, m

A_T = BEAM ATTENUATION

JBC-4

BEAM ATTENUATION

$A_T = A_w A_a A_n A_f$

A_w = VESSEL WALL ATTENUATION

A_a = AIR PATH ATTENUATION

A_n = DETECTOR HOUSING ATTENUATION

A_f = FLUID ATTENUATION

$A_i = e^{-\gamma_i X_i \rho_i}$, $i = w,a,n,f$

γ_i = MASS ATTENUATION COEFFICIENT, m^2/kg

X_i = PATH LENGTH, m

ρ_i = AVERAGE DENSITY kg/m^3

JBC-5

MEASUREMENT EQUATION

$$I = I_o e^{-\rho_f/B}$$

$$B = \frac{1}{X_f \gamma_f}$$

$$I_o = \frac{EAS}{R^2} A_w A_a A_n$$

SOLVING FOR DENSITY

$$\rho_f = B \ln \frac{I_o}{I}$$

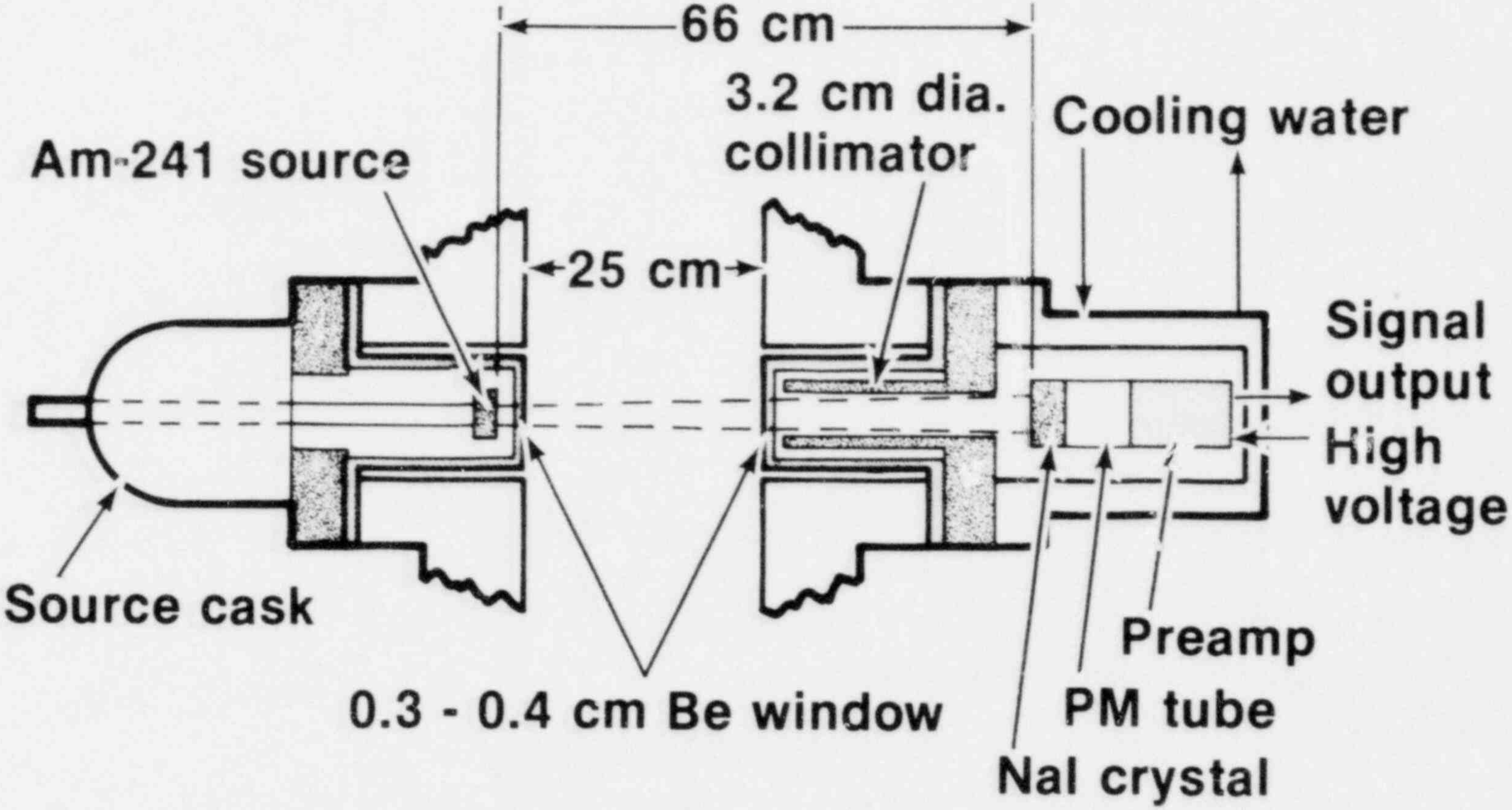
JBC-6

DESIGN CRITERIA

FLUID	STEAM/WATER
DUAL RANGE	0.7 kg/m^3 TO 70 kg/m^3 TO 1000 kg/m^3
ACCURACY	3% RANGE
PRESSURE	1.2 MPa
TEMPERATURE	1073 K AT BOUNDARY
THERMAL SHOCK	333 K
MAGNETIC FIELD	10 GAUSS AT 50 HZ

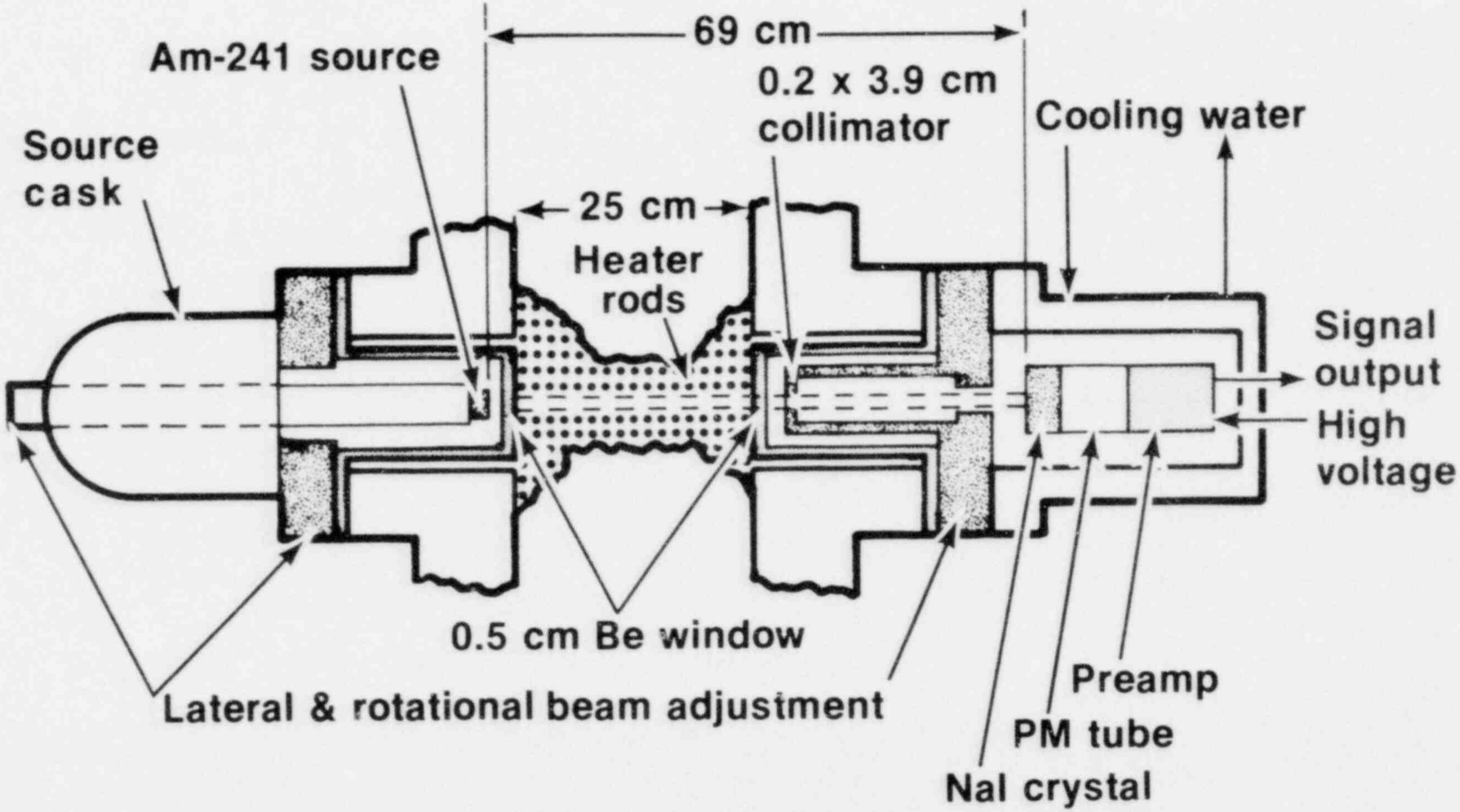
JBC-13

Circular Beam Densitometer



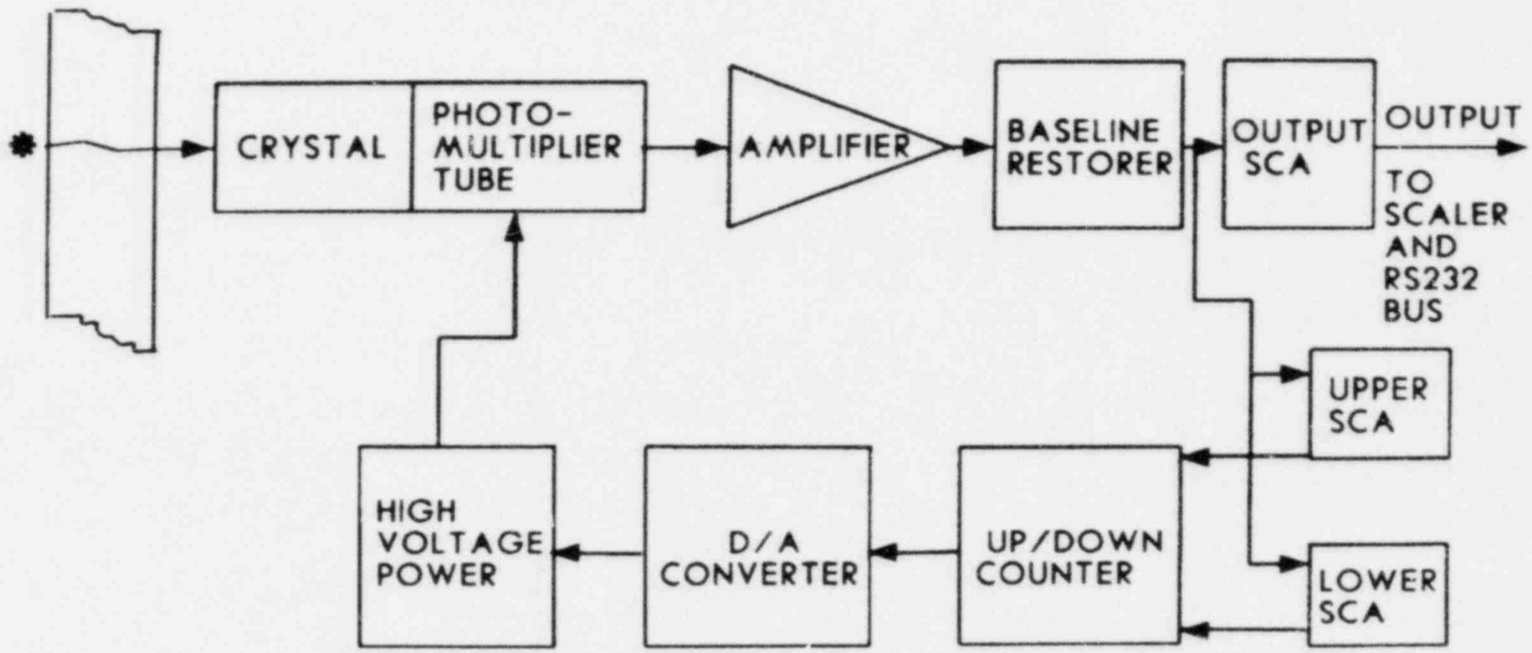
INEL-S-28 234

Rectangular Beam Densitometer

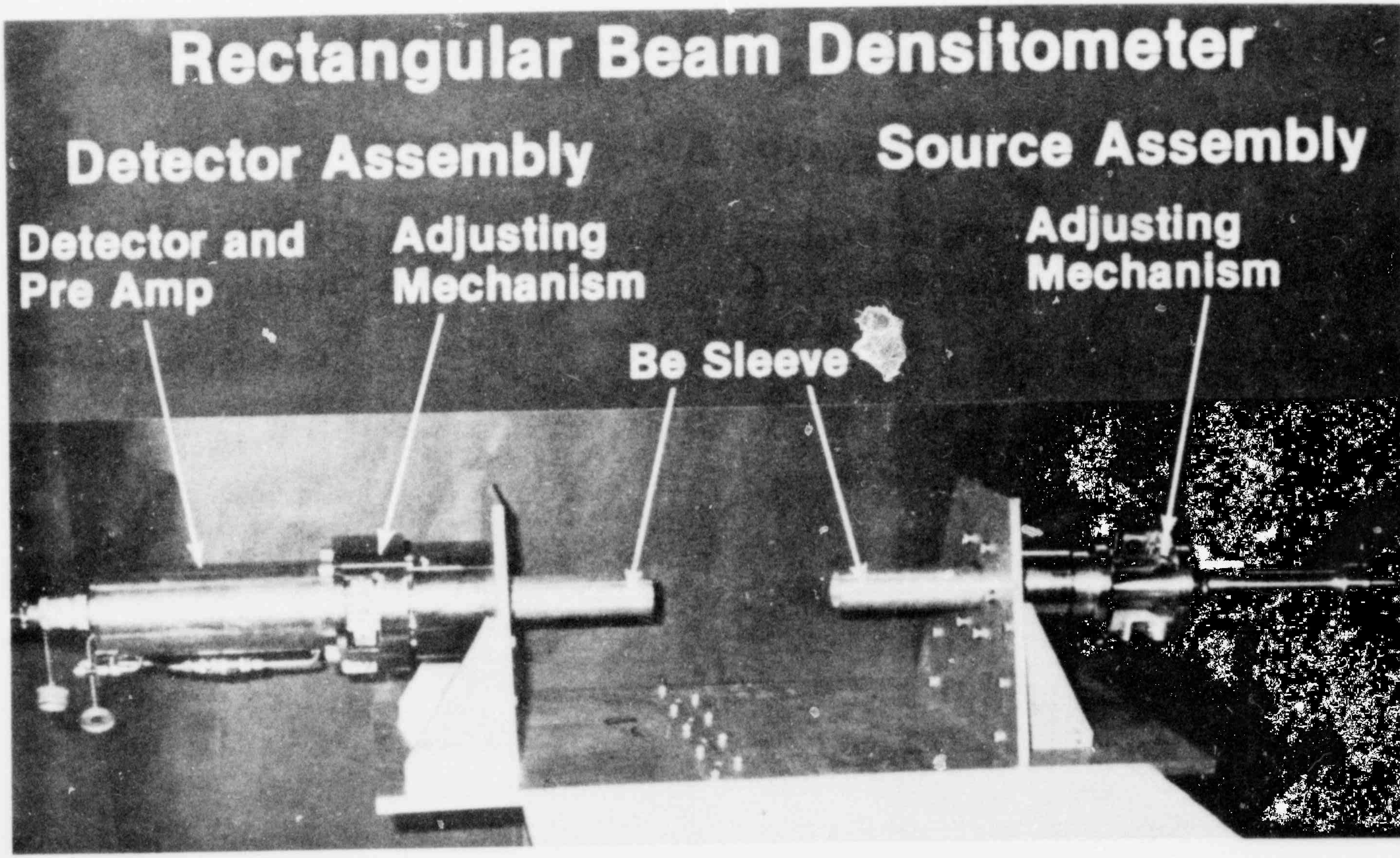


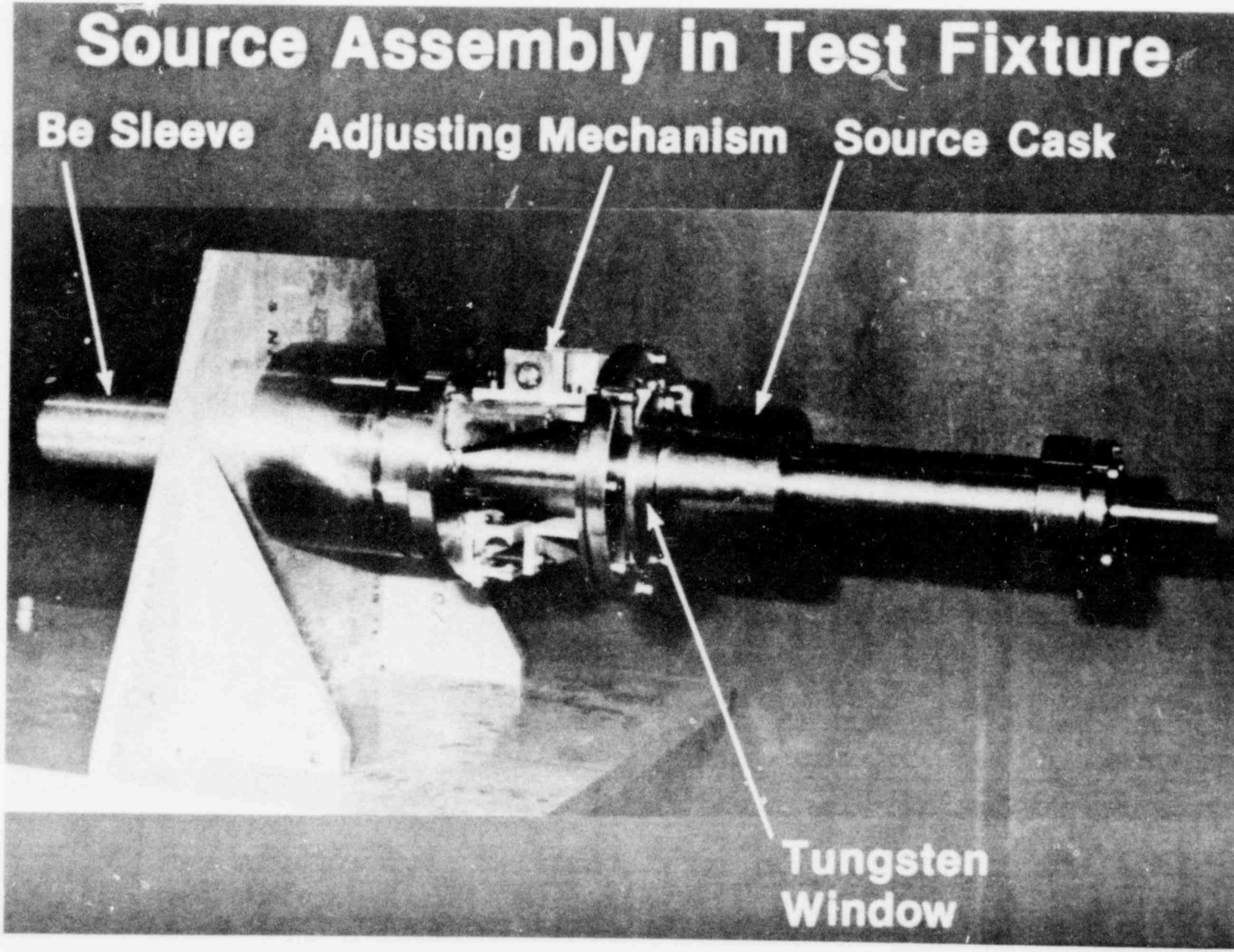
INEL-S-28 235

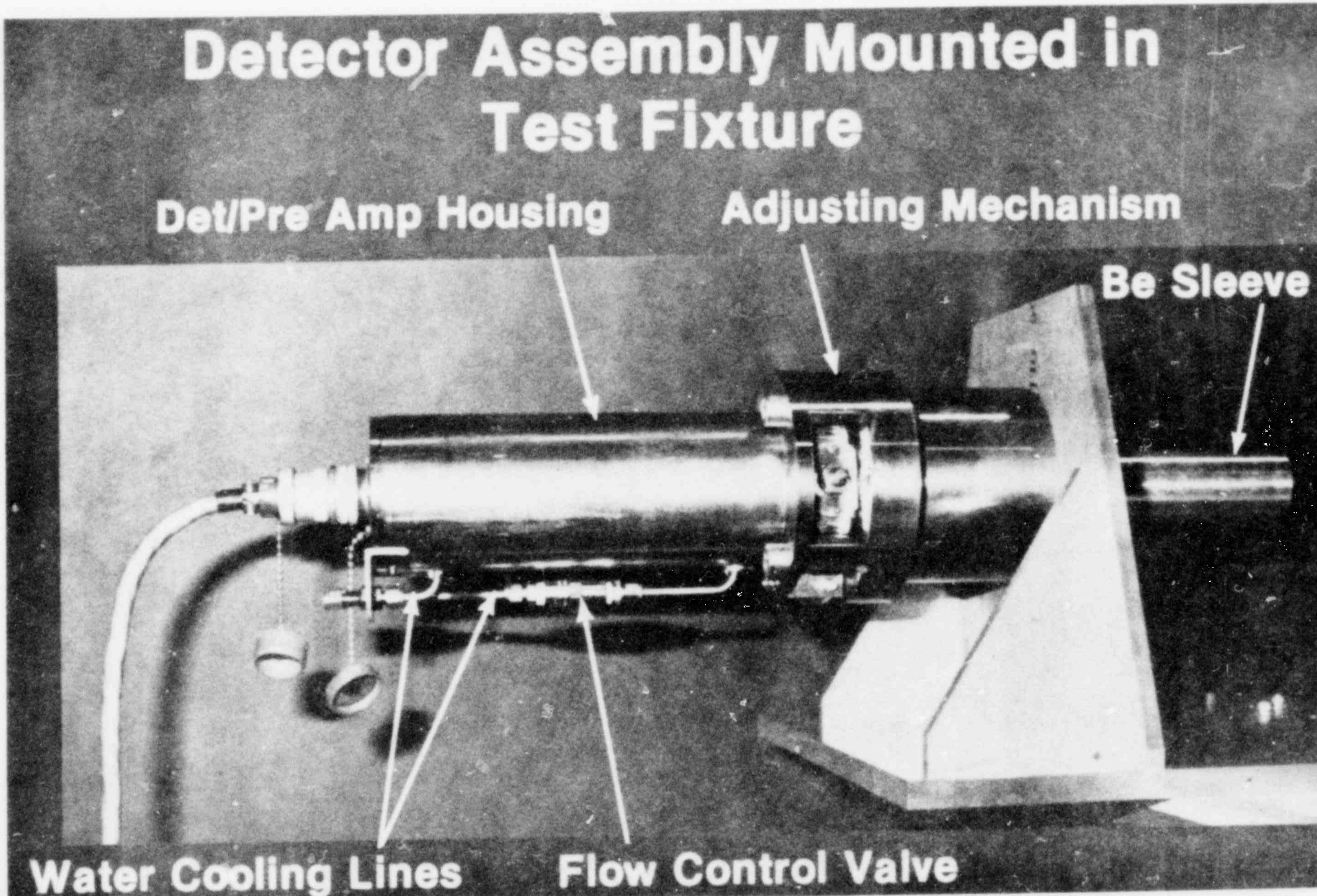
HIGH SPEED ELECTRONIC BLOCK DIAGRAM



JBC-7


Rectangular Beam Densitometer
Detector Assembly
Source Assembly
Detector and Pre Amp
Adjusting Mechanism
Be Sleeve
Adjusting Mechanism


Source Assembly in Test Fixture
Be Sleeve
Adjusting Mechanism
Source Cask
Tungsten
Window


Detector Assembly Mounted in
Test Fixture
Det/Pre Amp Housing
Adjusting Mechanism
Be Sleeve
Water Cooling Lines
Flow Control Valve

STATISTICAL ERROR

$$\rho = B \ln \frac{I_o}{I} \qquad . \; I = I_o e^{-\rho/B}$$

$$\frac{\Delta \rho}{\rho} = - B \frac{\Delta I}{I}$$

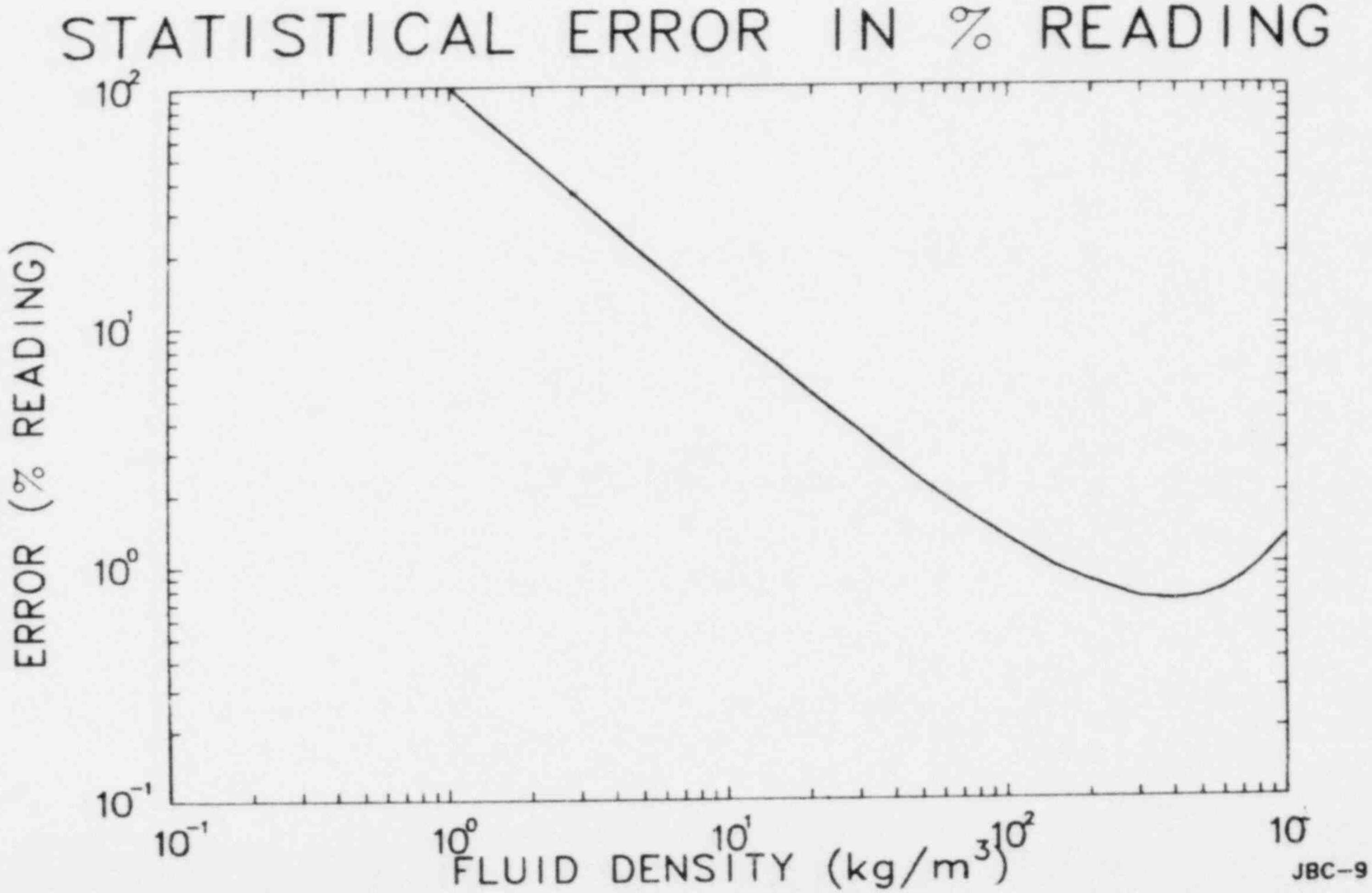
$\sigma = (IT)^{-1/2}$ = STANDARD DEVIATION

T = COUNTING TIME

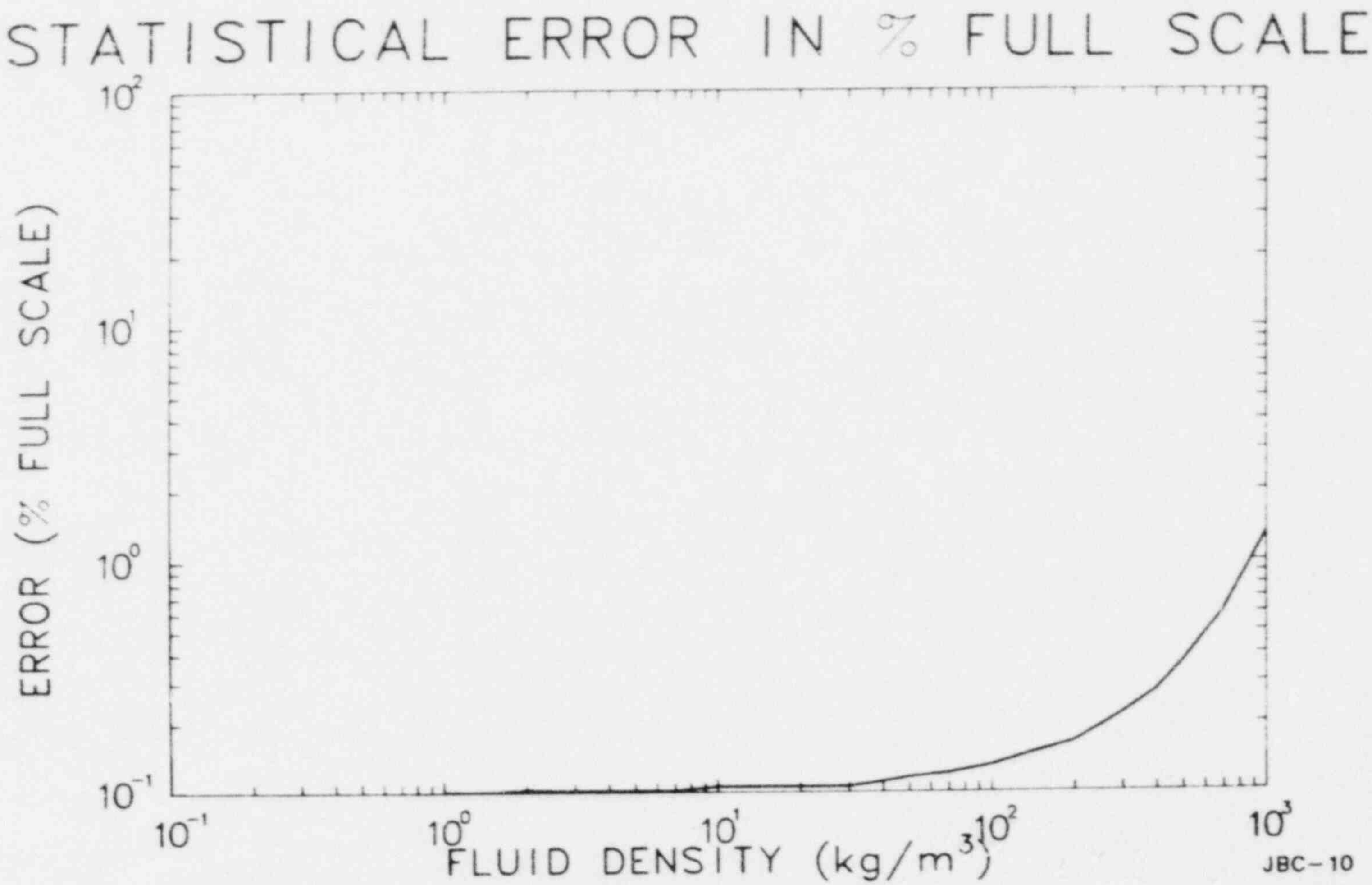
$\frac{\Delta I}{I} = 2\sigma$ FOR 95% CONFIDENCE LEVEL

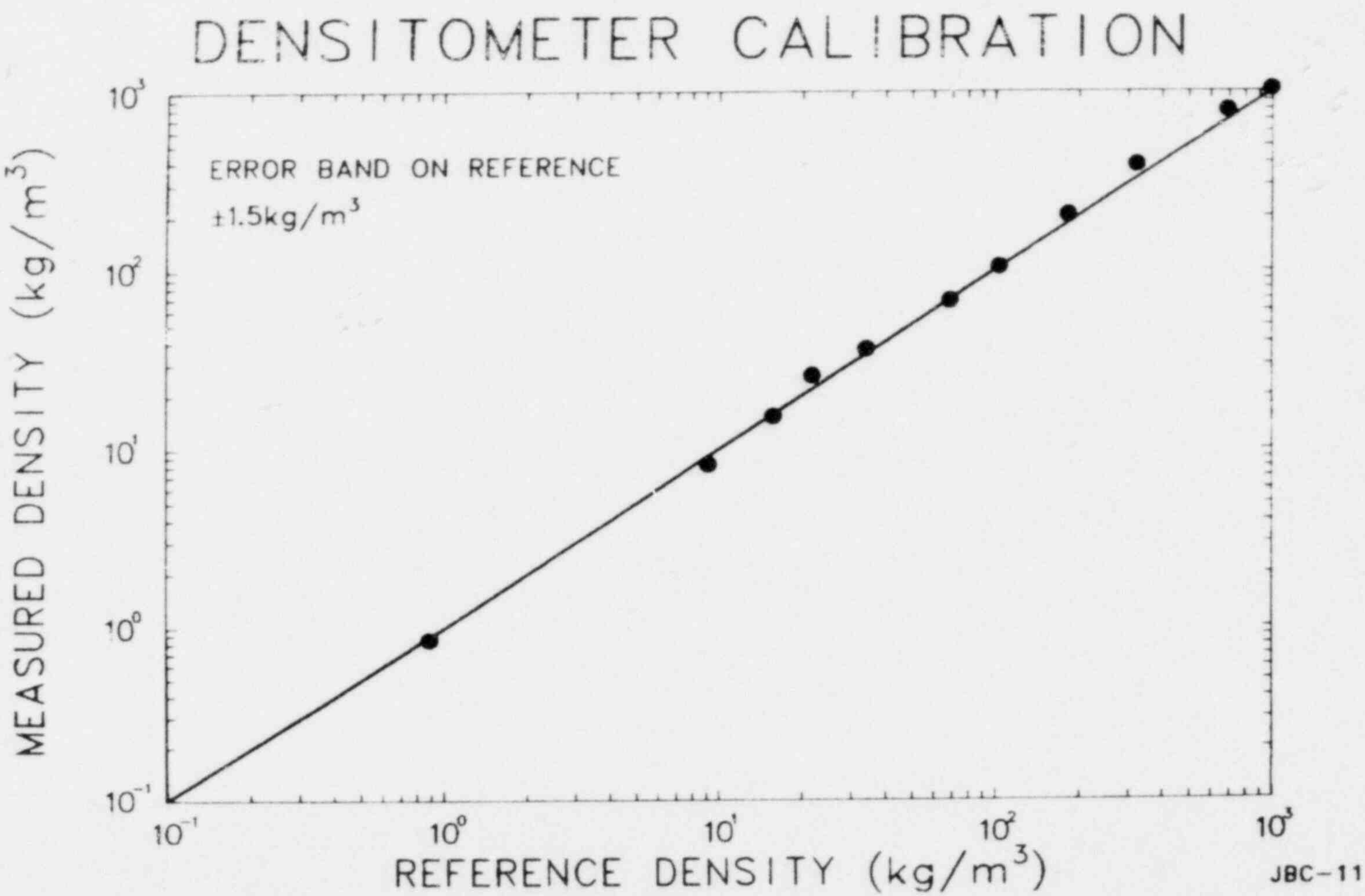
$$\frac{\Delta \rho}{\rho} = - \frac{2B}{\rho (IT)^{1/2}} = - \frac{2Be^{-\rho/2B}}{\rho (I_o T)^{1/2}}$$

JBC-8



STATISTICAL ERROR IN % READING
ERROR (% READING)
10^2
10^1
10^0
10^{-1}
10^{-1}
10^0
10^1
10^2
FLUID DENSITY (kg/m^3)
JBC-9


STATISTICAL ERROR IN % FULL SCALE
ERROR (% FULL SCALE)
10^2
10^1
10^0
10^-1
10^-1
10^0
10^1
10^2
10^3
FLUID DENSITY (kg/m^3)
JBC-10


DENSITOMETER CALIBRATION
ERROR BAND ON REFERENCE
±1.5kg/m³
MEASURED DENSITY (kg/m³)
REFERENCE DENSITY (kg/m³)
10⁻¹
10⁰
10¹
10²
10³
JBC-11

SUMMARY RESULTS

ERROR	RANGE 70 kg/m^3	RANGE 1000 kg/m^3
STATISTICAL	1.0 kg/m^3 + 0.3% Rd	0.3 kg/m^3 + 1.3% Rd
CALIBRATION	1.7 kg/m^3 + 0.7% Rd	1.3 kg/m^3 + 1.2% Rd
FULL RANGE	2.7%	1.9%

JBC-14

AN OPTICAL LIQUID LEVEL DETECTOR FOR HIGH TEMPERATURE/PRESSURE WATER ENVIRONMENT

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

B. L. Watson
R. P. Evans
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

AN OPTICAL LIQUID LEVEL DETECTOR FOR HIGH TEMPERATURE/PRESSURE WATER ENVIRONMENTS

B. L. Watson
R. P. Evans
EG&G Idaho, Inc.

An optical liquid level detector has been developed for the 2D/3D Program to meet the need for a fast-responding, liquid level detector (LLD) requiring a minimum of data interpretation. NRC use of these instruments is further described in a companion paper "Overview of 2D/3D Instrumentation Developed at EG&G Idaho, Inc." Design goals were established as 20 ms rise time, binary single parameter transducer output (wet/dry), 283 to 623 K operating temperature range, 1 to 20 bars operating pressure, electromagnetic immunity to 10 gauss fields at 50 hertz, and 523 K thermal shock.

The design chosen to meet these requirements utilizes an optical transducer consisting of a 45 degree sapphire cone metallized into a 200-series nickel housing. Dimensions of the transducer are approximately 0.3175 cm (1/8 in.) diameter by 1.27 cm (1/2-in.) long. Two optical fibers couple the transducer to the local signal conditioners and are protected by 0.157 cm (0.062 in.) diameter stainless steel tubing. Within the signal conditioner module continuous light generated by a high intensity tungsten lamp is coupled into an input fiber and transmitted to the optical transducer. The amount of light coupled into the return fiber by internal reflection within the transducer is a function only of the optical refractive index at the surface; dry producing a maximum return signal, wet returning a minimum. Detection is performed by a discrete photodiode, discriminated into a computer compatible binary signal, multiplexed, and stored on magnetic tape. Unique system components are the aluminum-coated optical fibers which are required to meet the 623 K temperature environment and the nickel-sapphire optical transducer

Performance parameters have been measured on prototype units with the following results: Signal/noise ratio 400/1, Dry/wet ratio 60/1, Optical throughput 25 dB (15 m fiber lengths), and electrooptic response time 1 ms.

Characterization tests have been performed to ensure accurate steam/water discrimination over the range of operating environments and include both falling drop and traveling void response time tests, autoclave testing at elevated temperature and pressure, and numerous laboratory bench scale experiments. These tests have demonstrated that under single phase conditions this design produces a distinct binary output with transition rise times limited only by the recording instrument or the interface velocity. Interface rise times are extremely fast (10 ms) due to the small active area on the transducer (approximately 2 mm^2). The small active area also tends to produce binary signals during two-phase operation although signal integration prior to recording often produces a variable amplitude signal. Variable amplitude signals can also be generated by high density steam although the hydrophobic property of sapphire and vertical positioning of the transducers minimize the effect. Of primary importance to the steam/water detector function is the large ratio that exists between steam and liquid water (30 to 1 minimum). This allows a threshold to be set which clearly distinguishes liquid water from any steam state.

In summary, an optical liquid level detector has been developed which meets or exceeds established design goals. Production design has been initiated with application to the Japanese Cylindrical Core Test Facility and the German Upper Plenum Test Facility fluid distribution grid projects.

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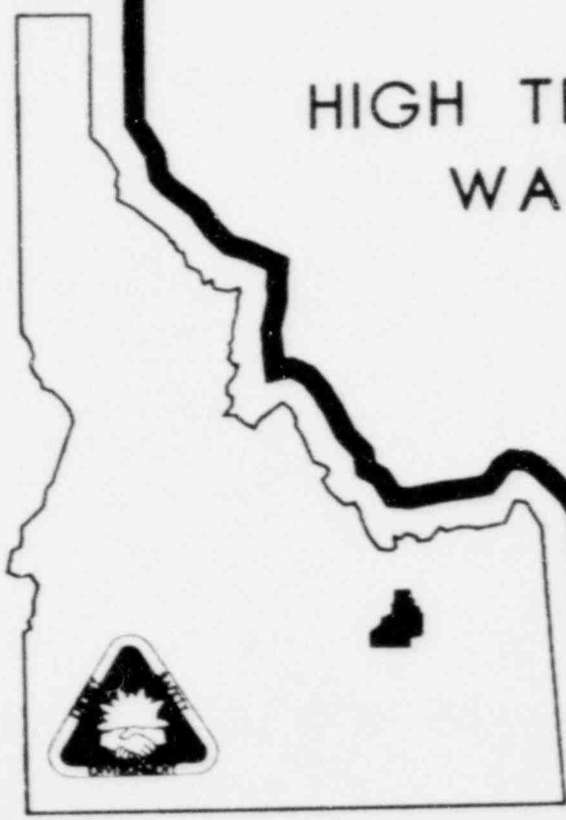
1. N. Abuaf, O. C. Jones, Jr., G. A. Zimmer, Optical Probe for Local Void Fraction and Interface Velocity Measurements, BNL NUREG 50791-NRC-2, (March 1978).

2. M. D. Rourke, V. L. Jones, and H. R. Friedrich, "Strain Induced Excess Loss in Aluminum-Coated Optical Wave Guides," (submitted to) Journal of Applied Physics Letters, (May 1980).

AN OPTICAL LIQUID LEVEL DETECTOR FOR HIGH TEMPERATURE/PRESSURE WATER ENVIRONMENT

by

B.L. WATSON



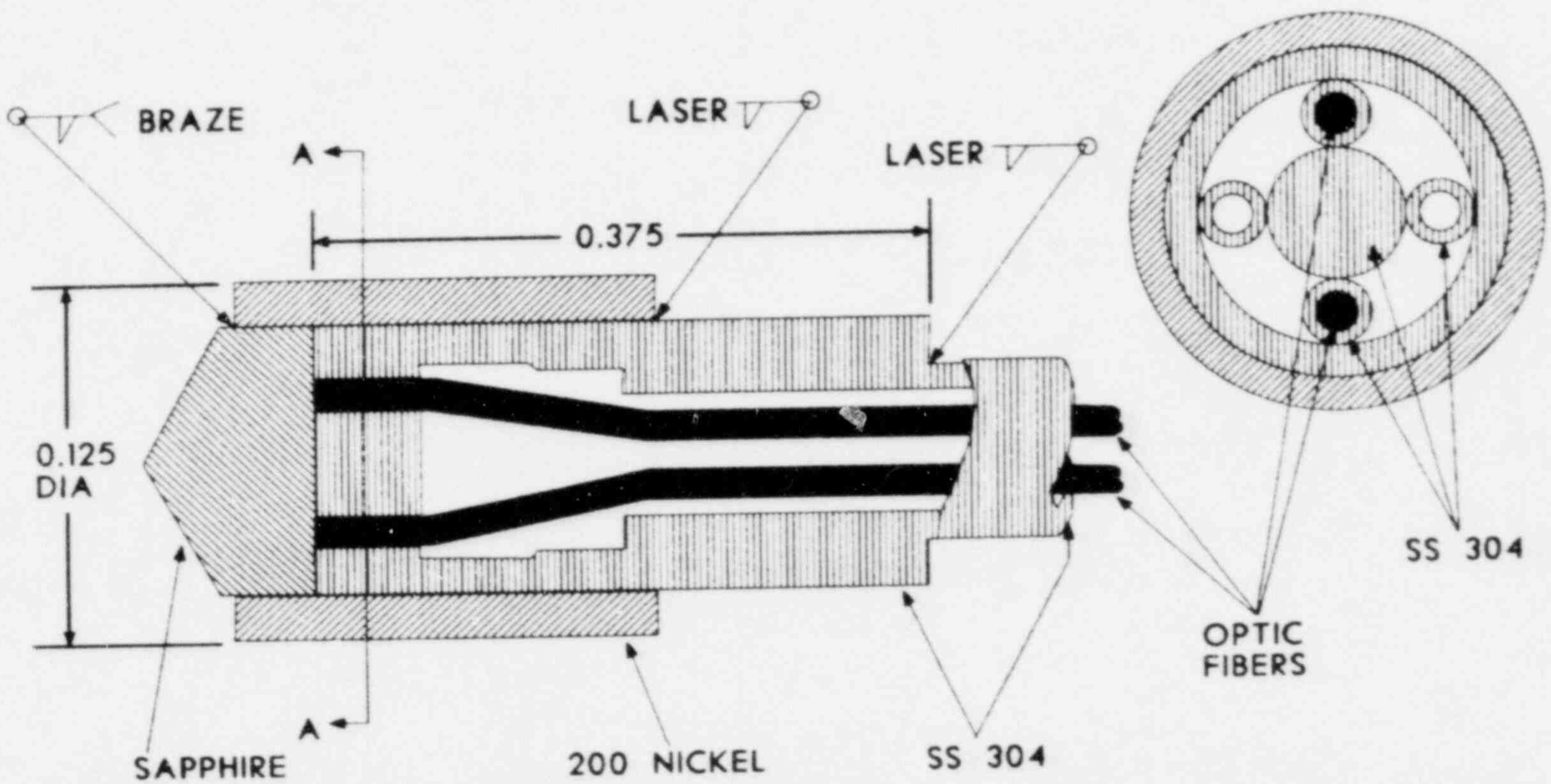
EG&G Idaho, Inc.

DESIGN GOALS

- RISE TIME <20 ms
- BINARY SINGLE PARAMETER TRANSDUCER REPONSE
- 10° TO $350^{\circ}C$ OPERATING TEMPERATURE
- 1 TO 20 BARS OPERATING PRESSURE
- 10 GAUSS 50 hz E.M. IMMUNITY
- $250^{\circ}C$ THERMAL SHOCK

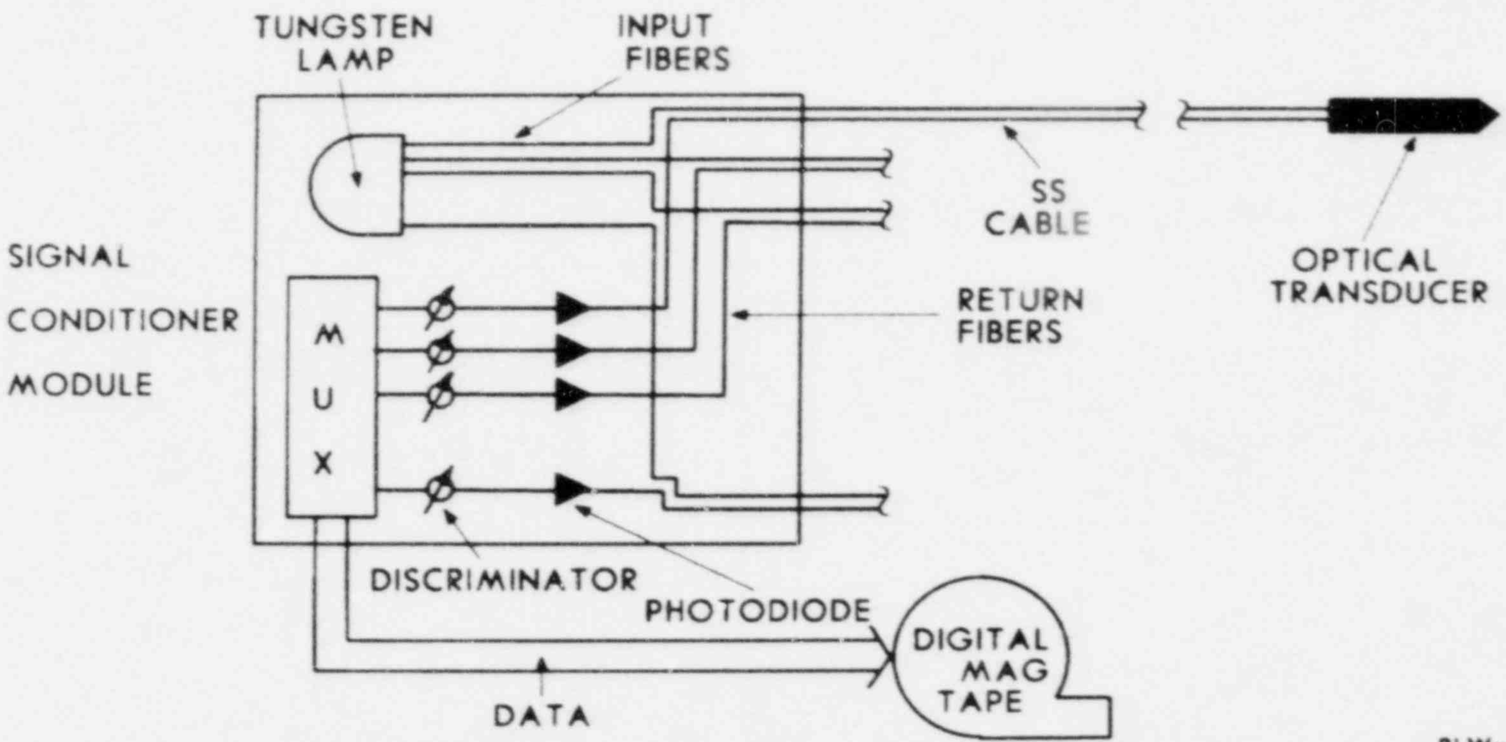
BLW-2

OPTICAL TRANSDUCER



BLW-3

SYSTEM DESIGN



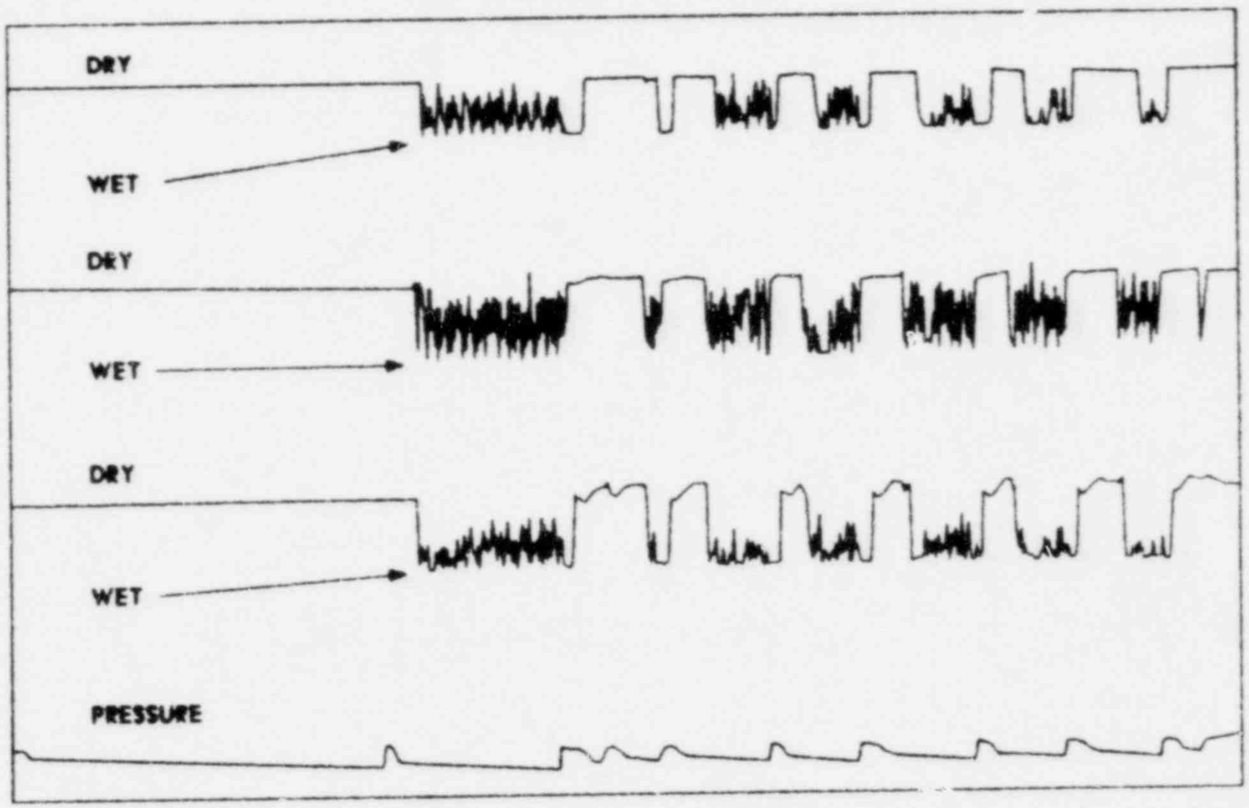
BLW-4

SYSTEM PERFORMANCE

SIGNAL/NOISE RATIO	400/1
DRY/WET RATIO	60/1
OPTICAL THROUGHPUT (15 M FIBER LENGTHS)	25 db
ELECTROOPTIC RESPONSE TIME	<1 ms

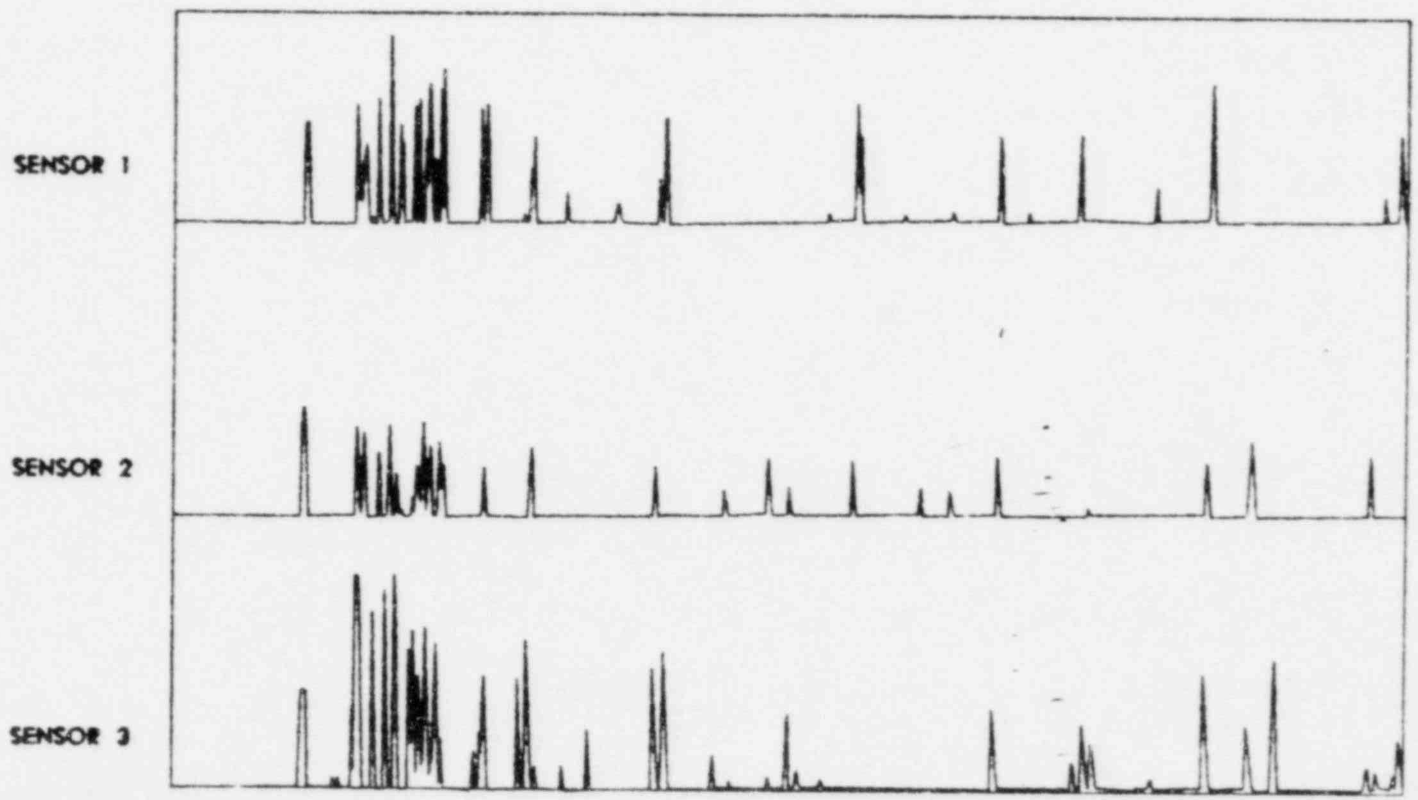
BLW-5

AUTOCLAVE DATA



BLW-6

BUBBLY FLOW DATA



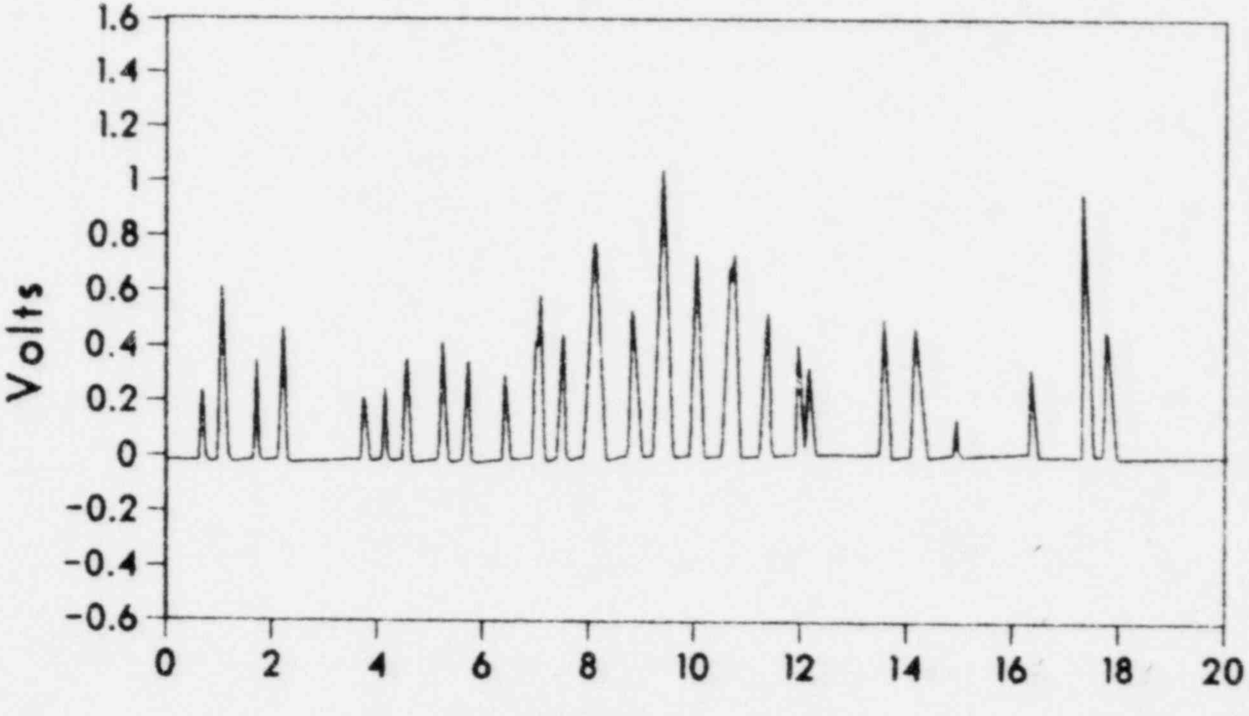
BLW-7

LOFT TEST SUPPORT FACILITY TESTING

- REALISTIC ENVIRONMENT
- CORRELATING INSTRUMENTATION
  - VIDEO CAMERAS
  - TURBINE FLOWMETERS
  - GAMMA DENSITOMETERS
- TEST GOALS
  - EVALUATE TWO-PHASE PERFORMANCE
  - VERIFY ACCURATE STEAM/WATER DISCRIMINATION
  - ASSESS ADDITIONAL MEASUREMENT CAPABILITIES

BLW-8

RESPONSE TIME DATA



8 ms/mm

Recorder Bandwidth 110 hz

BLW-9

ADVANCED INSTRUMENTATION PROJECT

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

W. H. Roach
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

ADVANCED INSTRUMENTATION PROJECT

W. H. Roach
EG&G Idaho, Inc.

SUMMARY

The advanced Instrumentation Branch of EG&G Idaho, in close cooperation with Water Reactor Research programs and the USNRC, prepares and develops instrumentation which may be utilized in the NRC assessment of the safety aspects of nuclear reactors.

During the past fiscal year, the Advanced Instrumentation Branch has completed several projects and studies, begun investigation into several new projects, and continued development of on-going tasks.

COMPLETED PROJECTS

A gamma ray tomographic densitometer system, including software for data reduction, has been developed and tested.[1] A system for 35.56 cm (14-in.) diameter piping has been fabricated and is ready for installation at the Water Reactor Test Facility.

Studies have been completed on a gamma ray scattering technique.[2] The method has the potential for providing density information in two-phase flow.

In the field of liquid level detectors, a simple, economical system using a heated thermocouple transducer and electronic controller has been developed and proof-tested.[3]

CURRENT PROJECTS

Small break accidents can lead to conditions where the steam generator becomes an important link in overall system response.[4] Instrumentation is being developed for the detection of U-tube voiding and for local heat flux measurements. Voiding is detected by a thermocouple placed at the top of the U-tube which detects change in heat conduction due to flow change. Heat flux measurements utilize a sandwich thermocouple principle.

A holographic camera system has been developed and is now being used to investigate bubble growth phenomena and mass and heat transport across a steam-water interface under laboratory conditions.

To assist in assessing the validity of phase velocity models in two-phase flow codes, a laser doppler velocimeter system has been developed and is under test.[5]

Work continues in the development of miniature zircaloy-sheathed thermocouples for use in embedded or internal fuel rod temperature measurements. Units to 0.5 mm in diameter have been produced and embedding techniques perfected.

PROPOSED TASKS

- Development of improved ultrasonic thermometry for temperatures to 3273 K range.

- Characterization of cladding thermocouple time response and perturbation effects under various coolant and temperature regimes.

- Evaluatation of infrared techniques for applicability to Water Reactor Safety Measurements.

- Feasibility studies of methods of sampling and identifying hydrogen gas in a PWR environment.

- Development of image analysis techniques to assist in the study of two-phase flow phenomena.

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2. A. G. Baker, "Gamma Scattering," NRC Review Group Conference on Advanced Instrumentation for Reactor Safety Research, Oak Ridge, Tenn., July 29-31, 1980, as published in the proceedings (NUREG/CP-0008).

3. J. V. Anderson, C. L Jeffery, "Heated Thermocouple Liquid Level System," NRC Review Group Conference on Advanced Instrumentation for Reactor Safety Research, Oak Ridge, Tenn., July 29-31, 1980, as published in the proceedings (NUREG/CP-0008).

4. J. Wolf, "Steam Generator Instrumentation," Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Md., October 27-31, 1980, proceedings to be published.

5. M. Wilson, "Optical Instrumentation in Two-Phase Flow: Laser Doppler Anemometer," Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Md., October 27-31, 1980, proceedings to be published.

OVERVIEW OF EG&G IDAHO ADVANCED INSTRUMENTATION PROGRAM

by

W.H. ROACH

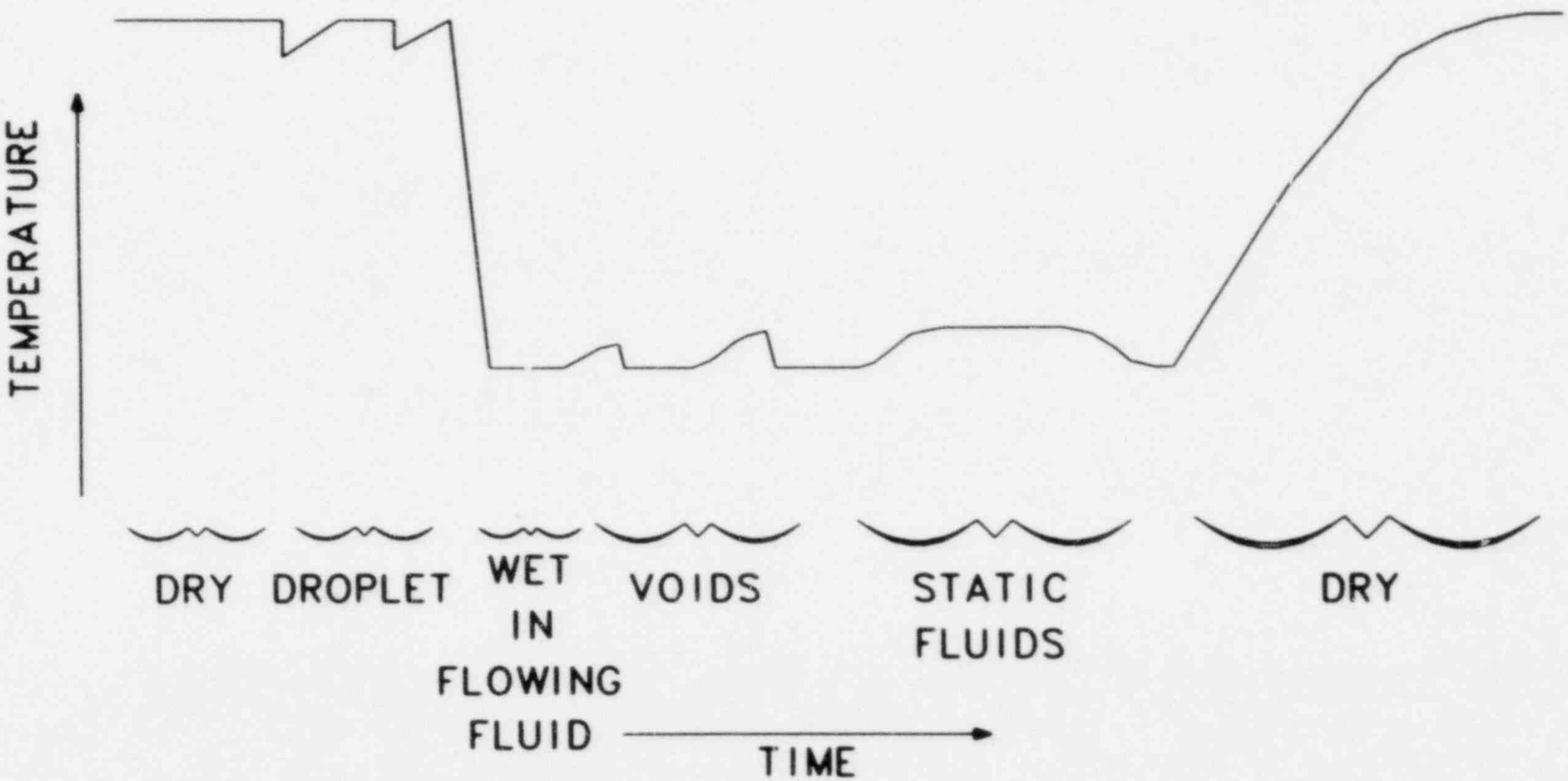
EG&G Idaho, Inc.

FY-80 COMPLETED PROJECTS

- GAMMA RAY TOMOGRAPHIC DENSITOMETER
- GAMMA RAY SCATTERING DENSITOMETER
- HEATED THERMOCOUPLE LIQUID LEVEL SYSTEM

WHR-1

POSTULATED TRANSDUCER RESPONSE



FY-80 CURRENT PROJECTS

- STEAM GENERATOR INSTRUMENTATION
- HOLOGRAPHIC STUDIES
- LASER DOPPLER VELOCIMETER
- MINIATURE ZIRCALOY-SHEATHED THERMOCOUPLE

WHR-2

HOLOGRAPHIC STUDIES

- BUBBLE GROWTH
- MASS AND HEAT TRANSPORT
- MULTIPLE PULSE HOLOGRAMS

WHR-3

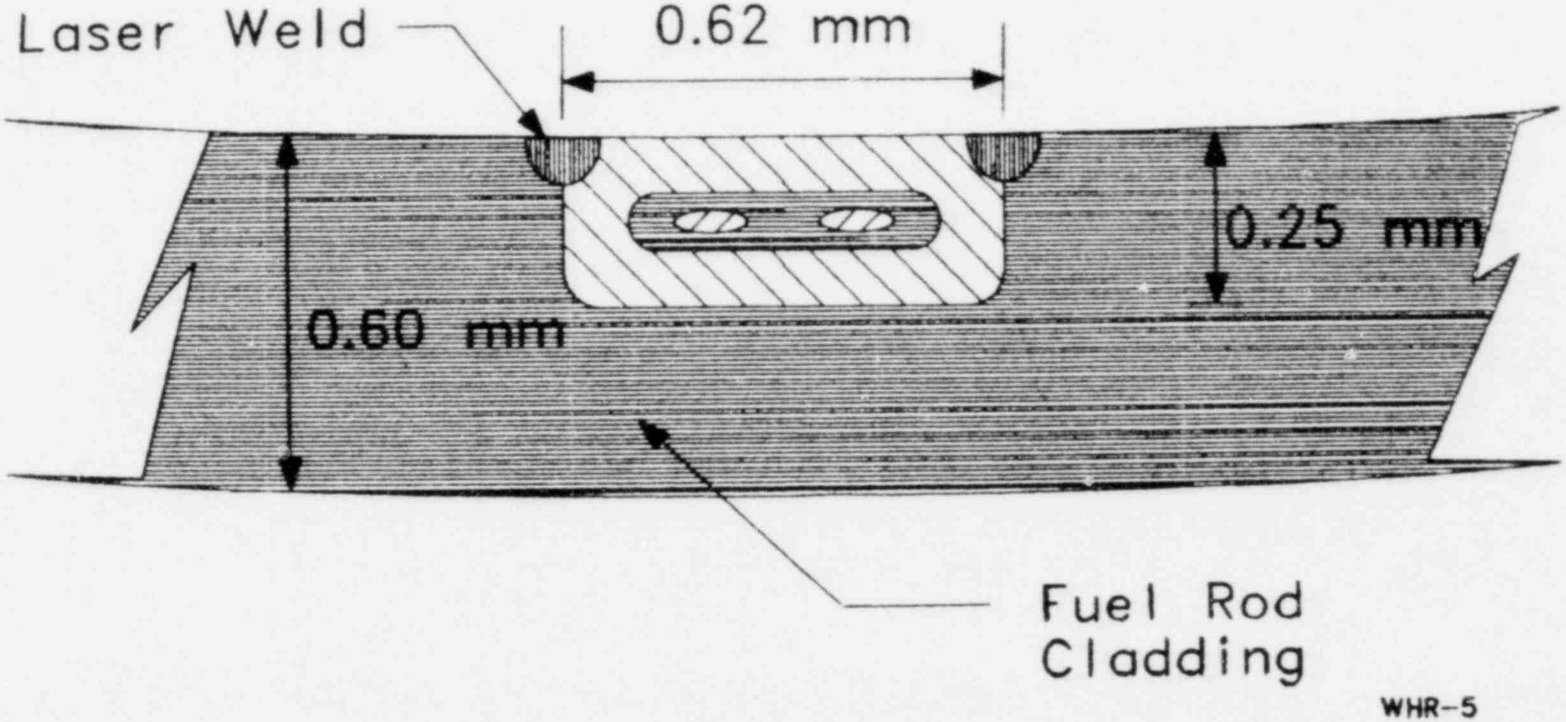
MINIATURE ZIRCALOY-SHEATHED THERMOCOUPLES

- 0.5 mm DIAMETER
- EMBEDDING TECHNIQUES

    50 mm SECTIONS, FLATTENED

WHR-4

EMBEDDED ZIRCALOY-SHEATHED THERMOCOUPLE ON INNER FUEL ROD CLADDING SURFACE



FY-81 TASKS

- ULTRASONIC TEMPERATURE MEASUREMENT
- CLADDING THERMOCOUPLE RESPONSE
- INFRARED TECHNOLOGY
- INCONDENSABLE GAS ANALYSIS
- IMAGE ANALYSIS

WHR-6

LASER DOPPLER ANEMOMETRY INSTRUMENTATION
OF TWO-PHASE FLOWS

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

M. L. Wilson
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

LASER DOPPLER ANEMOMETRY INSTRUMENTATION
OF TWO-PHASE FLOWS

M. L. Wilson
EG&G Idaho, Inc.

The assessment of safety related computer codes is an important part of the NRC Water Reactor Safety Research Program. Advanced computer codes have been developed using two-fluid models which allow separate phase velocities. Determination of these phase velocities require instrumentation capable of making measurements in two-phase flows. Conventional single-phase instrumentation such as turbines and drag devices do not have the response and range necessary to obtain phase velocity. Also the presence of these devices change the flow by introducing obstructions and pressure drops which can cause flow disturbances.

To accurately make measurements of the separate phase velocities in two-phase flows, the instrumentation should have a wide dynamic range, quick response time, and make a nonintrusive measurement. Laser Doppler Anenometer (LDA) systems have been constructed which have a 10^5 dynamic range (0.001 to 100 m/s) velocity measurement capability, response times of the order of 50 μs, and make a nonintrusive measurement with only light beams present in the flow. Also LDA systems measure velocity directly and do not require calibration. It is for those reasons that LDA techniques are currently under development at the INEL for use in some two-phase flows. Reference 1 gives an overview of LDA theory and describes their operation.

Several authors have used LDA techniques to instrument two-phase flows (References 2-5). In most of this work air-water bubbly two-phase flows were investigated. It was decided that this type of flow should be investigated initially in the Advanced Instrumentation Branch at INEL. A 38 mm square pexiglass test section was constructed which could be installed in several flow facilities. Work was done in both horizontal and vertical two-phase flows, with void fraction ranging from 0 to 0.25.

The LDA system consisted of a 15 milliwatt helium-neon laser, beam splitter, frequency shifter, beam expander, and both forward and backscatter receiving optics. The signal processing was done by a digital frequency counter specifically designed for LDA applications. Velocity information from the signal processor was recorded using a microcomputer and stored on a disk for latter processing. With the present system, data from the LDA can be stored on disk at a rate of 1000 data points per second yielding a 1 ms response time. Software has been developed to calculate mean velocity, probability distribution function (PDF), turbulance intensity, and higher order moments including variance, skewness, and kurtosis.

Good data was obtained from the LDA system up to 0.25 void fraction at which point the flow was sufficiently opaque that the laser beams were substantially attenuated. In horizontal two-phase flows, increasing the air flow while holding the water flow constant caused the calculated mean velocity and turbulance intensity to increase as was expected, while the PDF stayed smooth and symmetrical about the mean. However, in vertical flows the PDF was asymmetric with a tail which extended out well beyond the mean velocity. It is felt that this tail on the high side of the PDF is caused by bubbles inducing water movement as they pass through the measurement volume under buoyant forces. In the flows tested, many bubble sizes were present each with a different terminal rise velocity. A LDA measuring these velocities would have a distribution from 0 to 40 cm/s superimposed upon the water flow distribution. It appears that the PDF properly predicts this distribution. In addition, with increasing void fraction, the probability and extent of the PDF tail increases as expected.

Further work needs to be done in several areas. Increasing the data rate of the data aquisition system, improved analytical techniques, and investigation of other two-phase flow types is currently planned. Construction of a LDA system using laser diodes to decrease size and increase reliability is also underway. It appears that LDA techniques can be of great utility in measuring phase velocities in some two-phase flows.

REFERENCES

1. F. Durst, Principles and Practice of Laser Doppler Anemometry, Academic Press, 1976

2. W. E. R. Davies, "Velocity Measurements in Bubbly Two-Phase Flows Using Laser Doppler Anemometry," UTIAS Technical Note 184, 185.

3. K. Ohba, "Simultaneous Measurement of Local Liquid Velocity and Void Fraction in Bubbly Flows Using a Gas Laser," Technological Reports of Osaka University, Vol.-26, No.-1328-1336, October 1976, pp. 547-556.

4. F. Durst, Laser Doppler Measurements in Two-Phase Flows, Proceedings of the LDA Symposium, Copenhagen, 1975, DK-2740, P. O. Box 70, Skovlunde, Denmark, June 1976.

5. S. L. Lee, "Laser Doppler Anemometry Technique Applied to Two-Phase Dispersed Flows,"Two-Phase Flow Instrumentaion Review Group Meeting, January 13-14, 1977, NUREG-0375.

LASER DOPPLER ANEMOMETRY INSTRUMENTATION OF TWO-PHASE FLOWS

by

M. L. WILSON

EG&G Idaho, Inc.

INSTRUMENTATION REQUIREMENTS

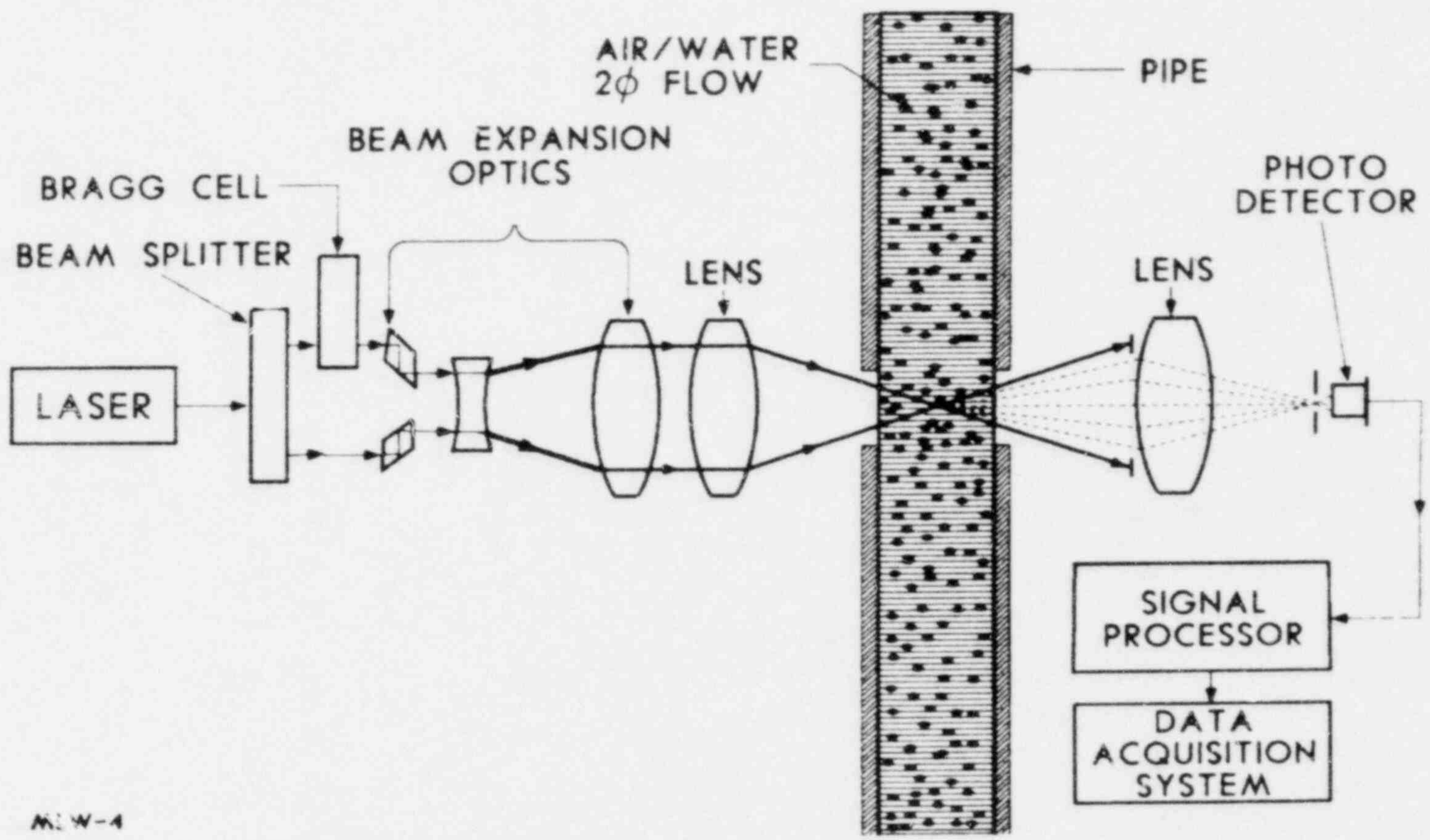
- QUICK RESPONSE
- WIDE DYNAMIC RANGE
- NONINTRUSIVE
- DIRECT MEASUREMENT
- REQUIRE NO TWO-PHASE CALIBRATION

MLW-2

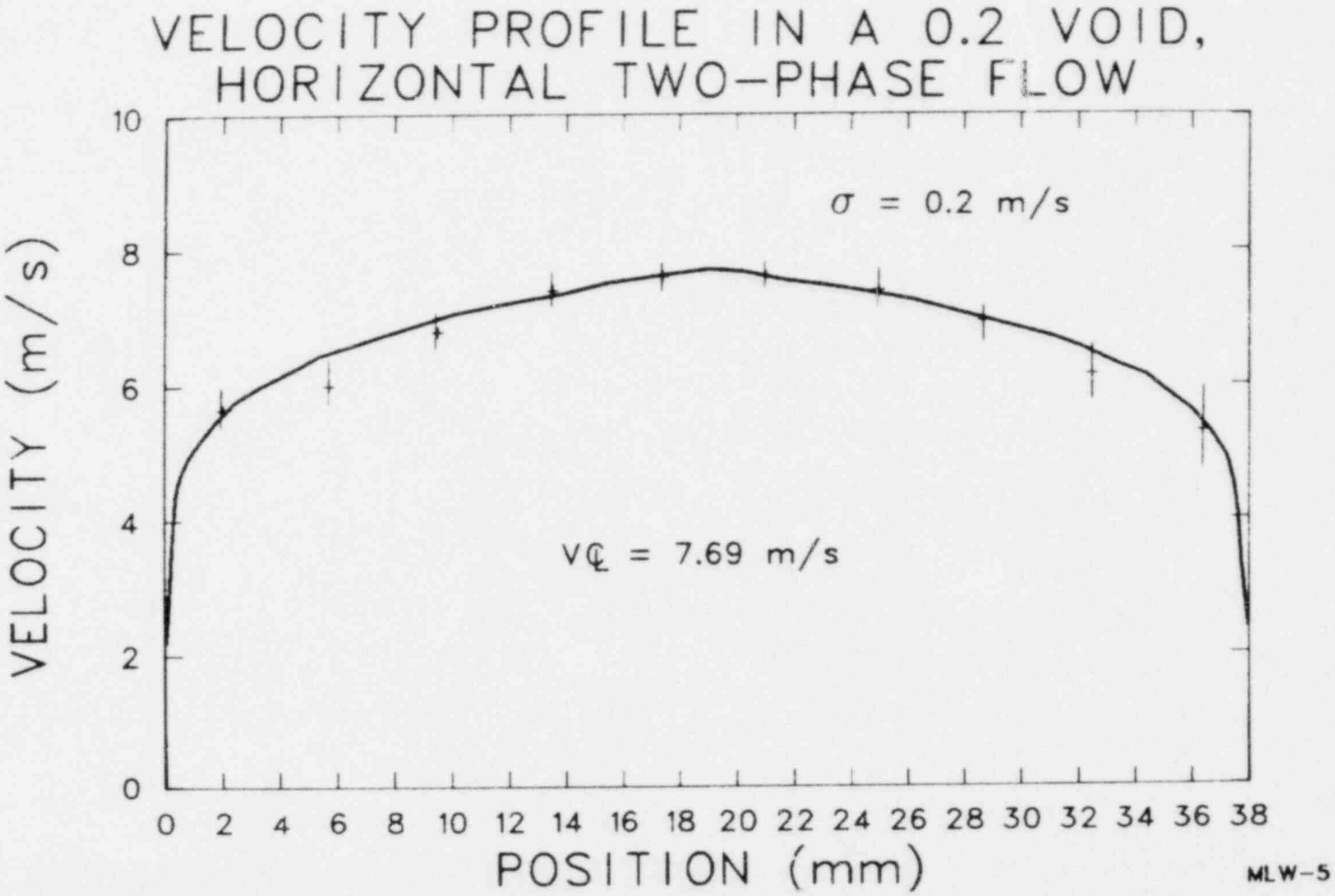
LDA MEASUREMENT CAPABILITIES

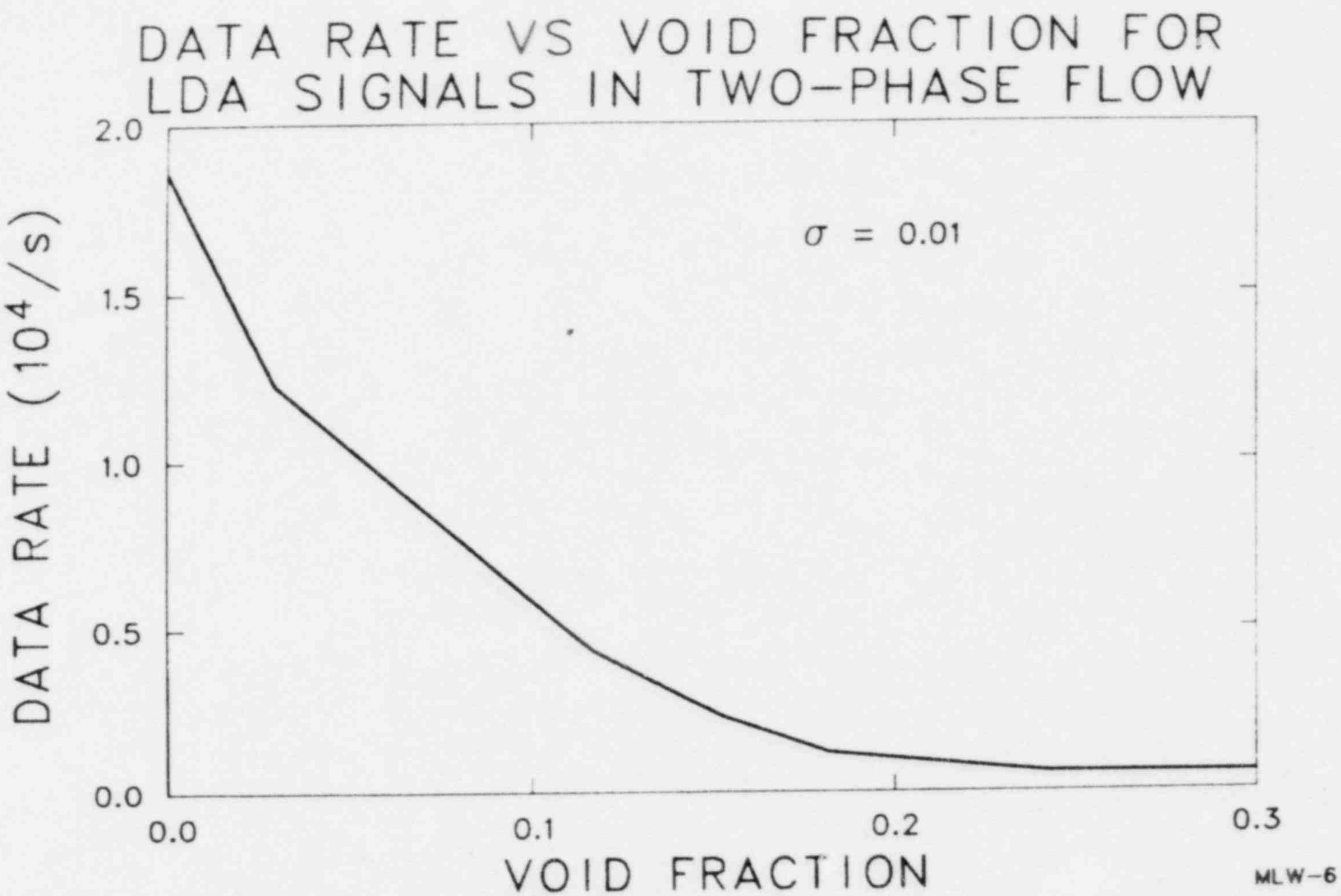
- RESPONSE TIME (50 s)9
- DYNAMIC RANGE (0.001-100 m/s)
- DIRECT MEASUREMENT OF VELOCITY
- ABSOLUTE MEASUREMENT REQUIRING NO CALIBRATION
- NONINTRUSIVE MEASUREMENT

MLW-3

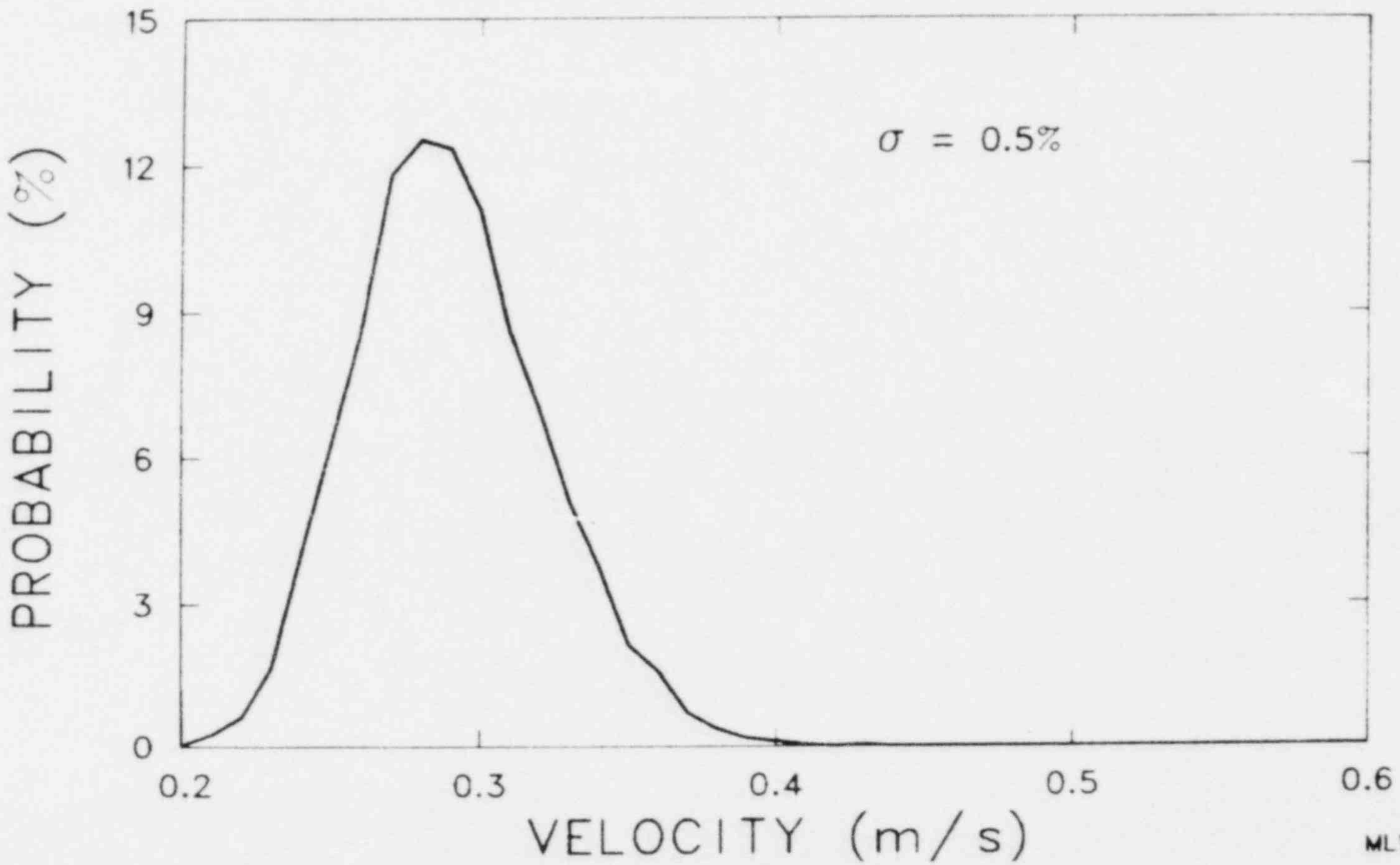

AIR/WATER
2ϕ FLOW
PIPE
BEAM EXPANSION
OPTICS
BRAGG CELL
BEAM SPLITTER
PHOTO
DETECTOR
LENS
LENS
LASER
SIGNAL
PROCESSOR
DATA
ACQUISITION
SYSTEM

M[illegible]W-4

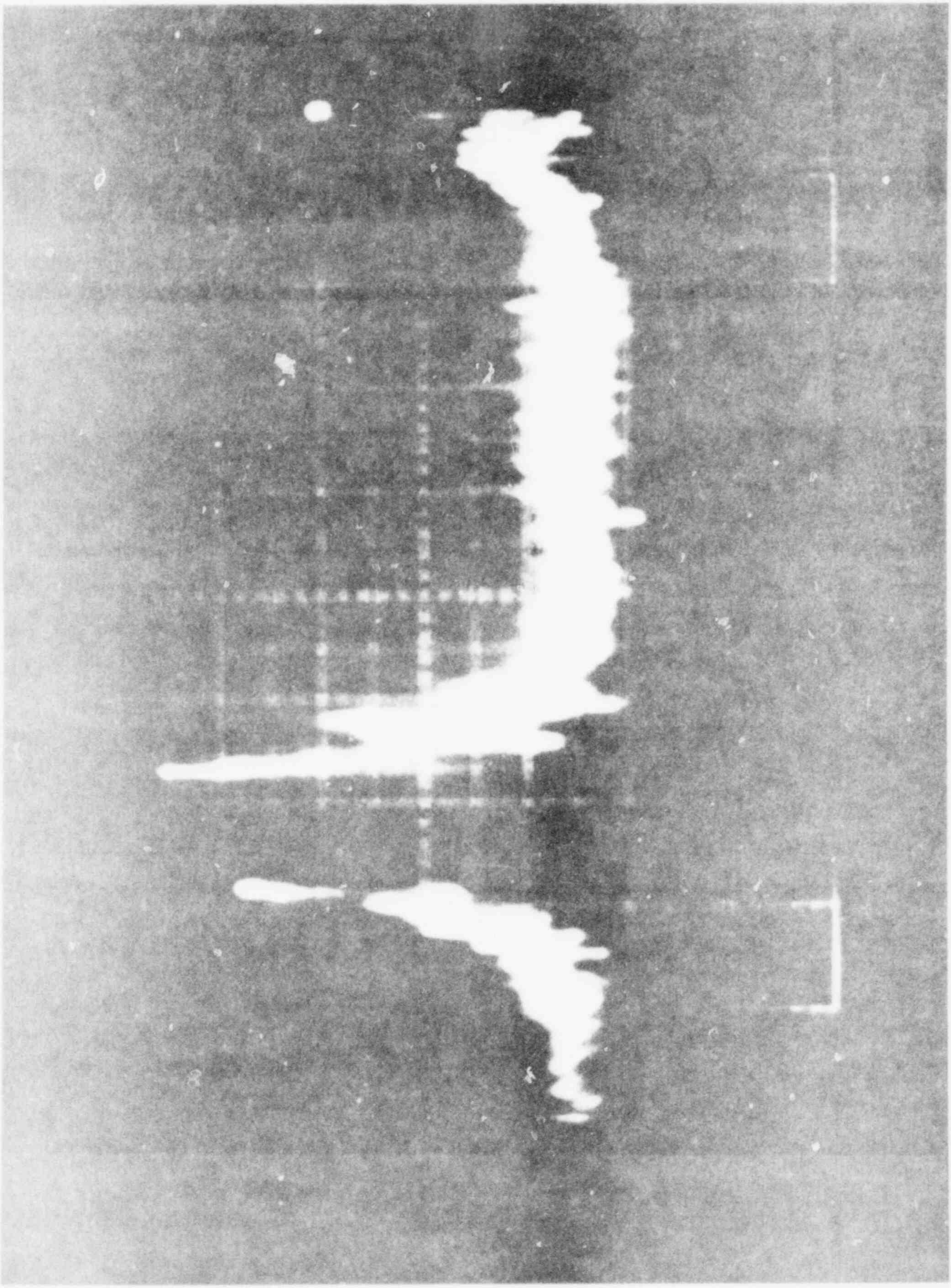

VELOCITY PROFILE IN A 0.2 VOID,
HORIZONTAL TWO-PHASE FLOW
σ = 0.2 m/s
V℄ = 7.69 m/s
VELOCITY (m/s)
POSITION (mm)
0
2
4
6
8
10
0 2 4 6 8 10 12 14 16 18 20 22 24 26 28 30 32 34 36 38
MLW-5


DATA RATE VS VOID FRACTION FOR
LDA SIGNALS IN TWO-PHASE FLOW
σ = 0.01
DATA RATE (10⁴/s)
2.0
1.5
1.0
0.5
0.0
0.0
0.1
0.2
0.3
VOID FRACTION
MLW-6

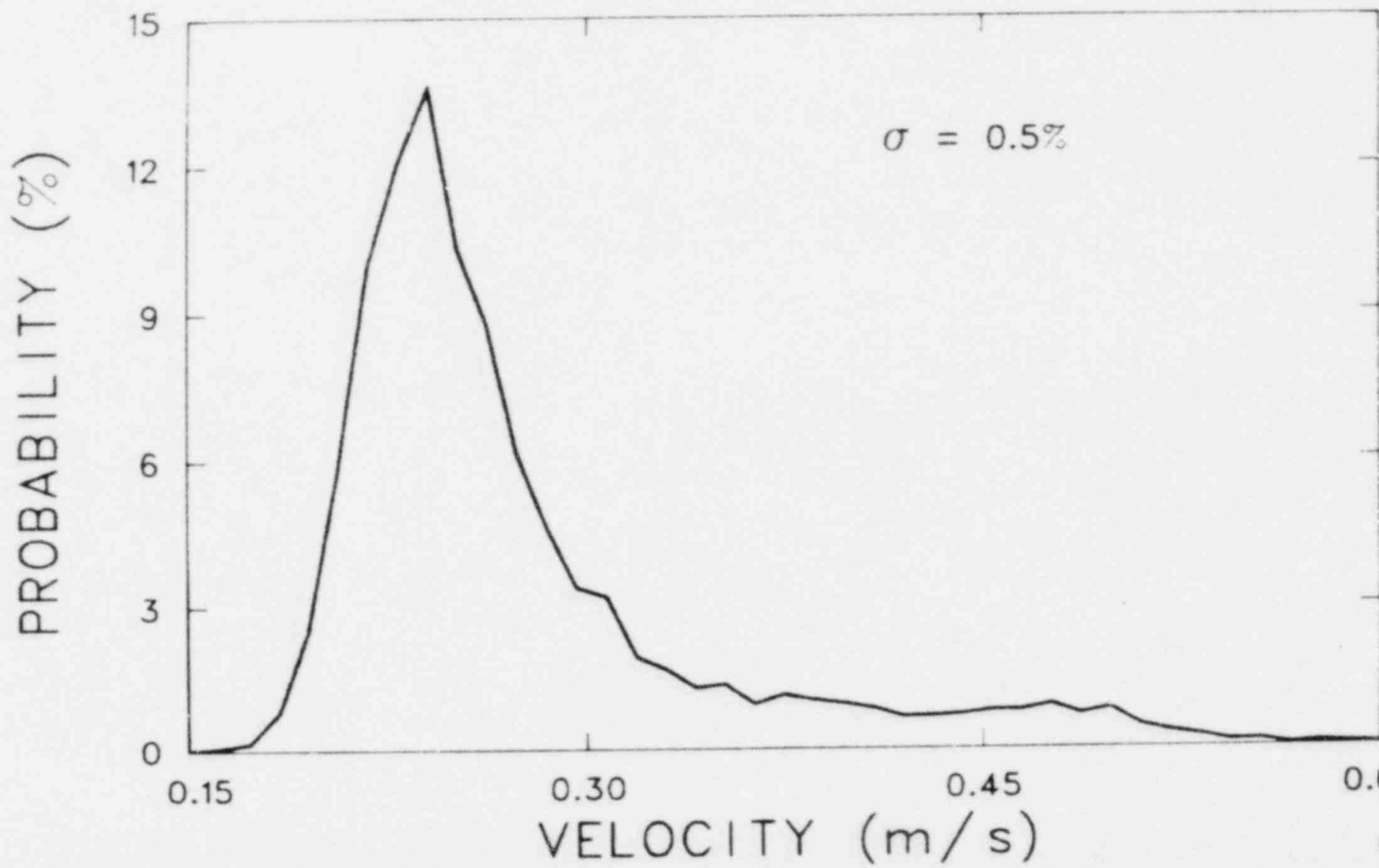
PDF OF ALL WATER FLOW



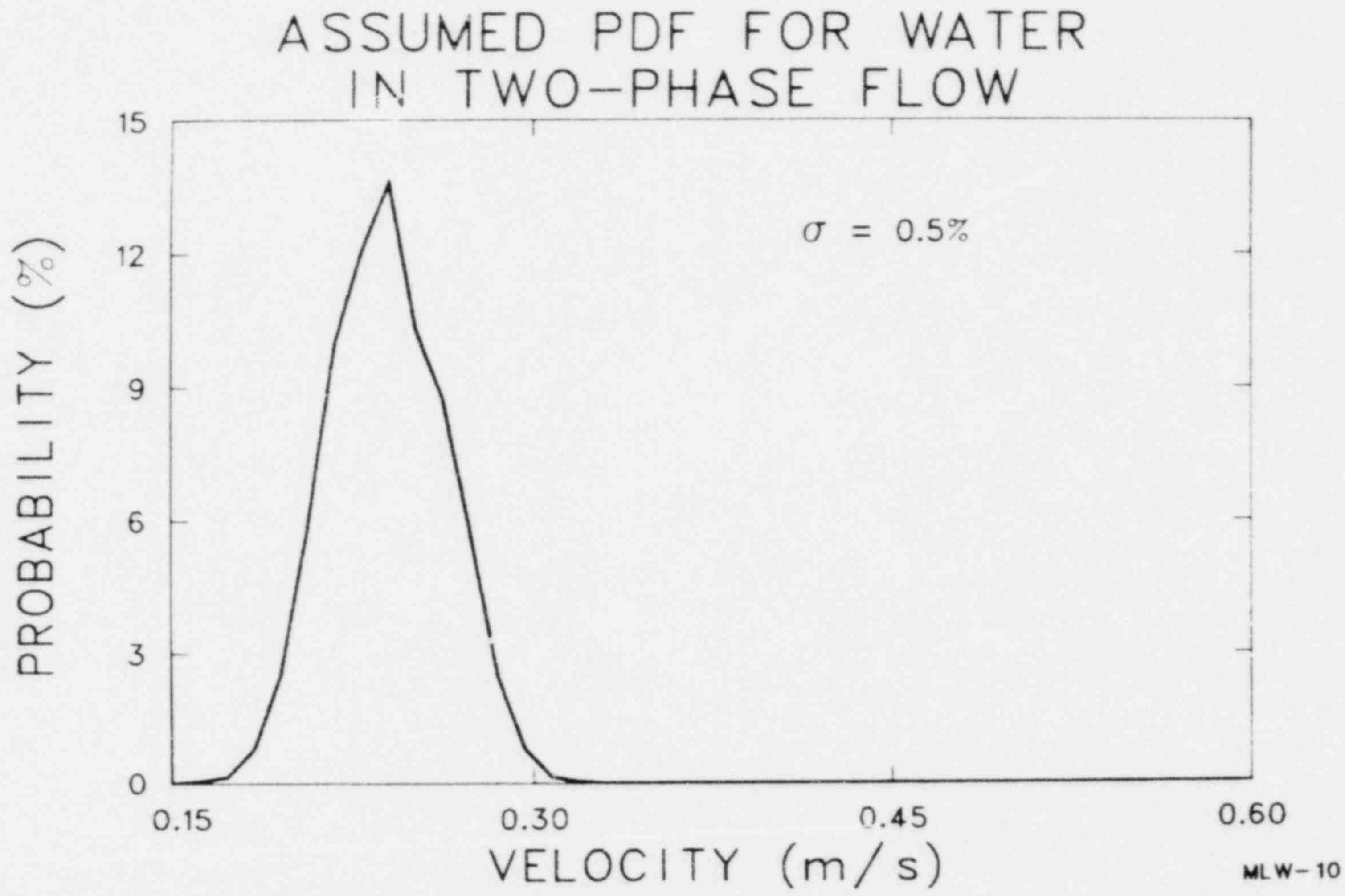
MLW-7



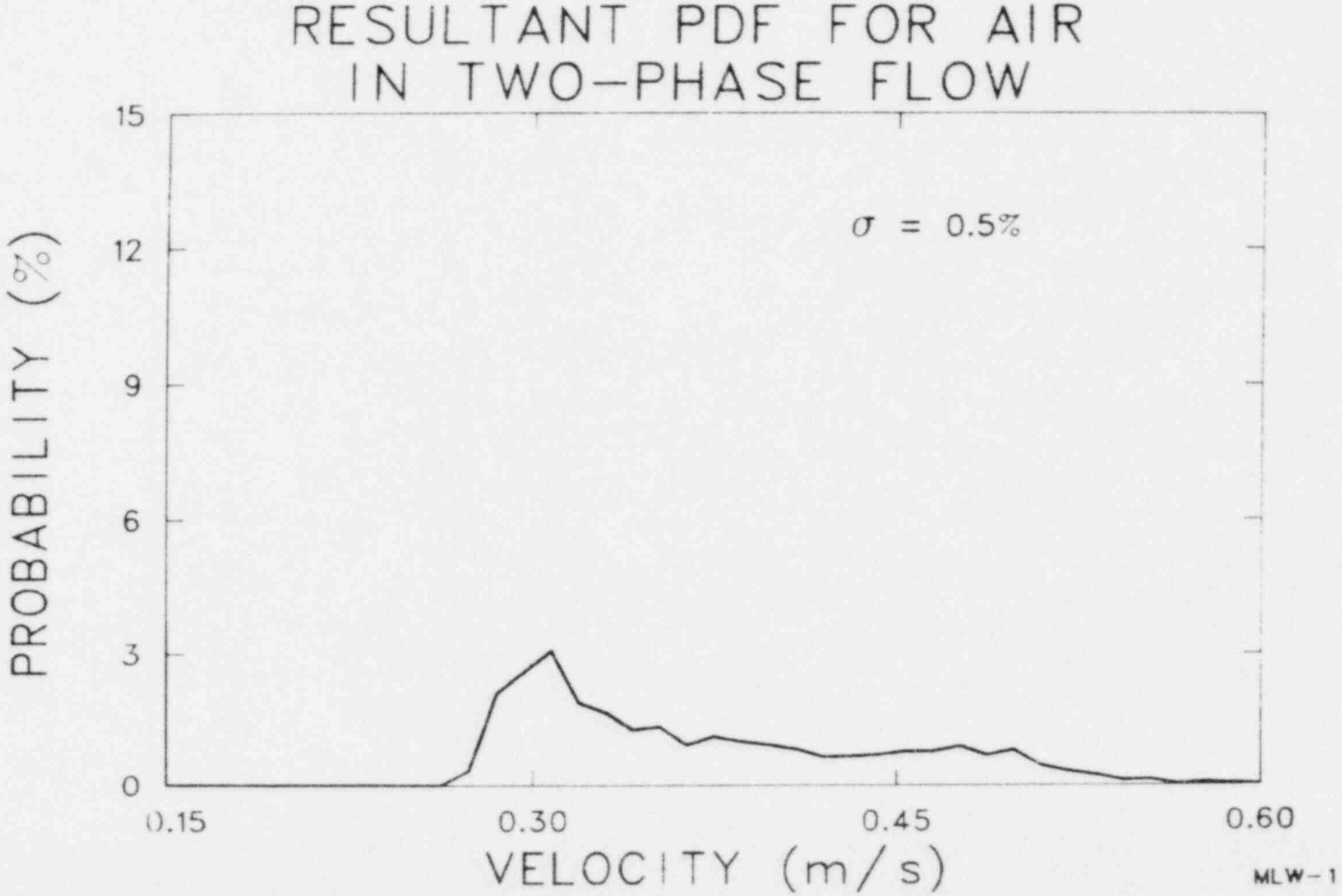
PDF OF AIR/WATER TWO-PHASE FLOW



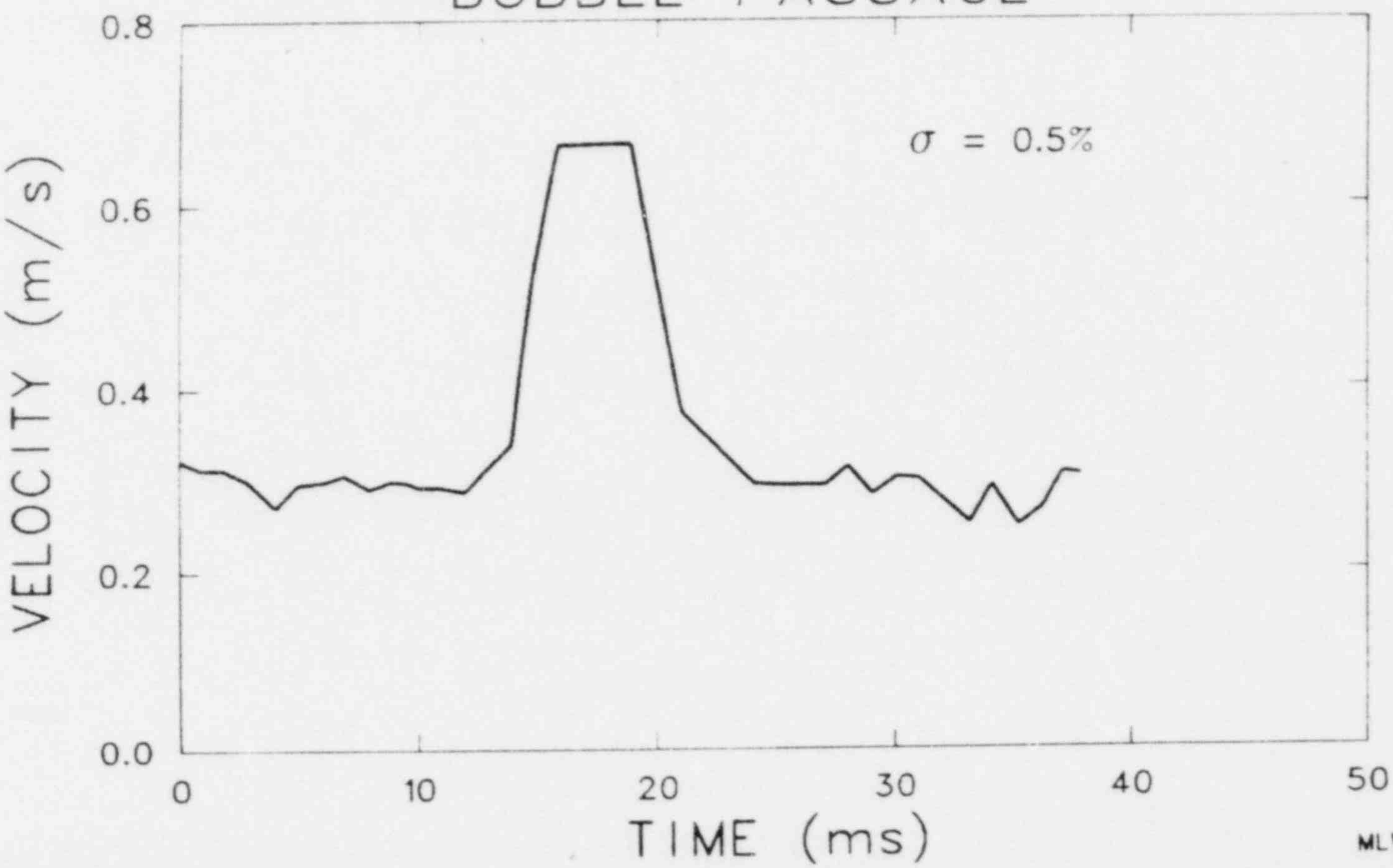
MLW-9



ASSUMED PDF FOR WATER
IN TWO-PHASE FLOW
PROBABILITY (%)
15
12
9
6
3
0
σ = 0.5%
0.15
0.30
0.45
0.60
VELOCITY (m/s)
MLW-10


RESULTANT PDF FOR AIR
IN TWO-PHASE FLOW
σ = 0.5%
PROBABILITY (%)
15
12
9
6
3
0
0.15
0.30
0.45
0.60
VELOCITY (m/s)
MLW-11

VELOCITY VS TIME DURING BUBBLE PASSAGE



MLW-12

SUMMARY

- LDA TECHNIQUES WORK IN BUBBLY TWO-PHASE FLOWS UP TO 0.25 VOID FRACTION

- VELOCITY PROFILES, MEAN VELOCITY, PDF, AND HIGHER ORDER MOMENTS OF THE PDF WERE OBTAINED IN A TWO-PHASE BUBBLY FLOW

- THE INFLUENCE OF BUBBLES IN A VERTICAL TWO-PHASE FLOW WERE DETECTABLE

MLW-13

FUTURE WORK

- INCREASE DATA RATE OF THE DATA ACQUISITION SYSTEM
- DEVELOP ANALYTICAL TECHNIQUES TO OBTAIN SLIP FROM LDA DATA
- INVESTIGATE OTHER TWO-PHASE FLOWS
- CONSTRUCT A LASER DIODE LDA SYSTEM TO IMPROVE MEASUREMENT CAPABILITIES

MLW-14

STEAM GENERATOR INSTRUMENTATION

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

Dr. J. R. Wolf
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

STEAM GENERATOR INSTRUMENTATION

Dr. J. R. Wolf
EG&G Idaho, Inc.

As a result of the Three Mile Island incident, considerable effort has gone into investigating safety aspects of the small break accident. Because of the slow depressurization rate in this type of accident, the steam generator plays an important part in determining plant thermal response.

In the early stages of a large break where it is not necessary to cool the plant over a long period of time, the steam generator has a less important role. As a result of plant safety emphasis on large break accidents, very little instrumentation has been developed for steam generator applications.

During a small break loss-of-coolant accident, current NRC regulations require that the main coolant pumps be shut off and natural circulation used to cool the reactor. Under certain conditions, steam and any noncondensible gasses which have been produced in the core will pass into the steam generator and collect at the highest point of the system. This point is the top of the U-bend in the tube bundle or the inlet (hot leg) pipe in a once through steam generator design.

When the primary coolant contains a two-component mixture of noncondensible gasses and water, heat transfer and removal through the tube bundle will be inhibited. Because little or no heat is removed in the steam generator, the normal inlet to outlet primary coolant density gradient disappears. The natural circulation mode is driven by this density gradient in the steam generator tube bundle and when it is no longer present, natural circulation is lost. When this occurs, the noncondensible gases collect on the top of the U-tube.

In order to properly monitor the steam generator and to determine if natural circulation has been inhibited, instrumentation is being developed to measure local heat flux in the tube bundle and to detect voiding caused by noncondensible gas collection in the U-tube.

A calorimetric thermopile device is being developed to measure local U-tube heat flux. It consists of a stainless steel half cylinder which is attached directly to a steam generator U-tube. Two 0.025 cm Type E thermocouples are embedded in the inner and outer surfaces of the probe to measure its differential temperature. The local U-tube heat flux is determined by using this differential temperature and calculating the heat conduction through the probe. Development work is currently underway to characterize the device and to determine if attaching it to the outside of a tube will cause a perturbation to normal local heat flux.

The boundary layer voiding detector is being developed in order to detect steam voiding in the tube bundle. The detector is a thermocouple placed on a tube wall and covered with an electric heater. The heat loss from the heater as determined by the temperature of the thermocouple is caused by the coolant flow in the tube. When the tube voids with steam, the temperature of the heater increases due to the steam's poor thermal conductivity. A rapid increase in heater temperature is an indication of voiding. Laboratory tests with voiding caused by water injection indicate that the device is capable of detecting U-tube voiding.

REFERENCE

1. S. B. Englert, J. R. Fincke, J. R. Wolf, "Low Flow Velocimeters and Voiding Detectors," NRC Review Group Conference on Advanced Instrumentation for Reactor Safety Research, Oak Ridge, Tennessee, July 29-31, 1980, as published in the proceedings (NUREG/CP-0008).

STEAM GENERATOR INSTRUMENTATION

By

DR. J.R. WOLF

EG&G Idaho, Inc.

OUTLINE

- NEED FOR STEAM GENERATOR INSTRUMENTATION
- BOUNDARY LAYER VOIDING DETECTOR
- LOCAL U-TUBE HEAT FLUX MEASUREMENTS
- SUMMARY

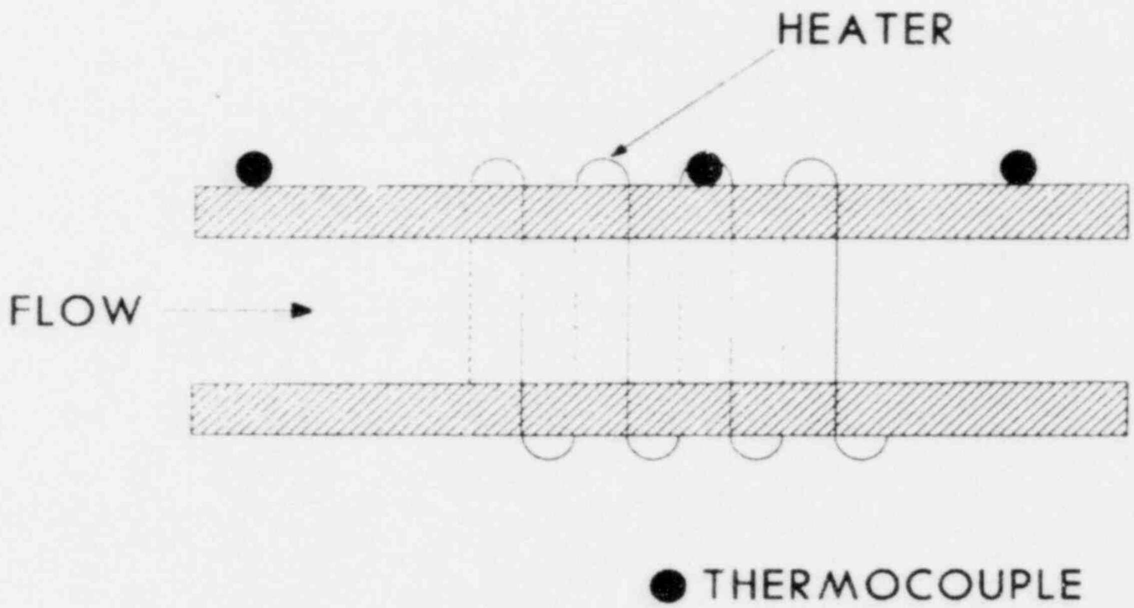
NEW INSTRUMENTATION NEEDS

NEW INSTRUMENTATION NEEDS TO MAKE SMALL BREAK STEAM GENERATOR MEASUREMENTS

- U-TUBE VOIDING
- LOCAL U-TUBE HEAT FLUX

JRW-3

BOUNDARY LAYER VOID DETECTOR CONFIGURATION

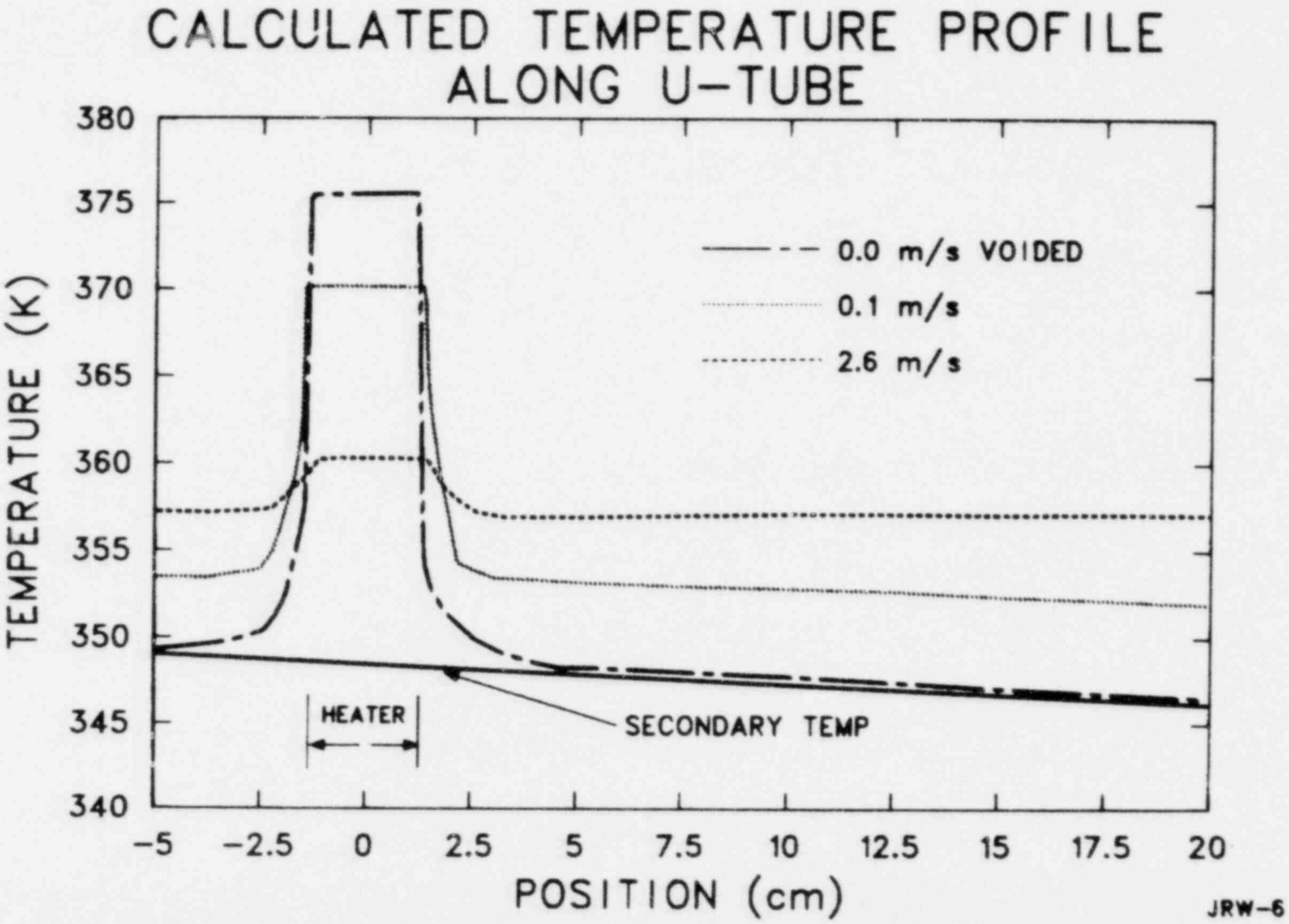


JWR-4

STEADY STATE MODEL

- SERIES OF ONE-DIMENSIONAL RELAXATION EQUATIONS
- CONVECTIVE HEAT TRANSFER TO PRIMARY AND SECONDARY WITH FORCED AND NATURAL CONVECTION
- AXIAL CONDUCTION CONSIDERED

JRW-5

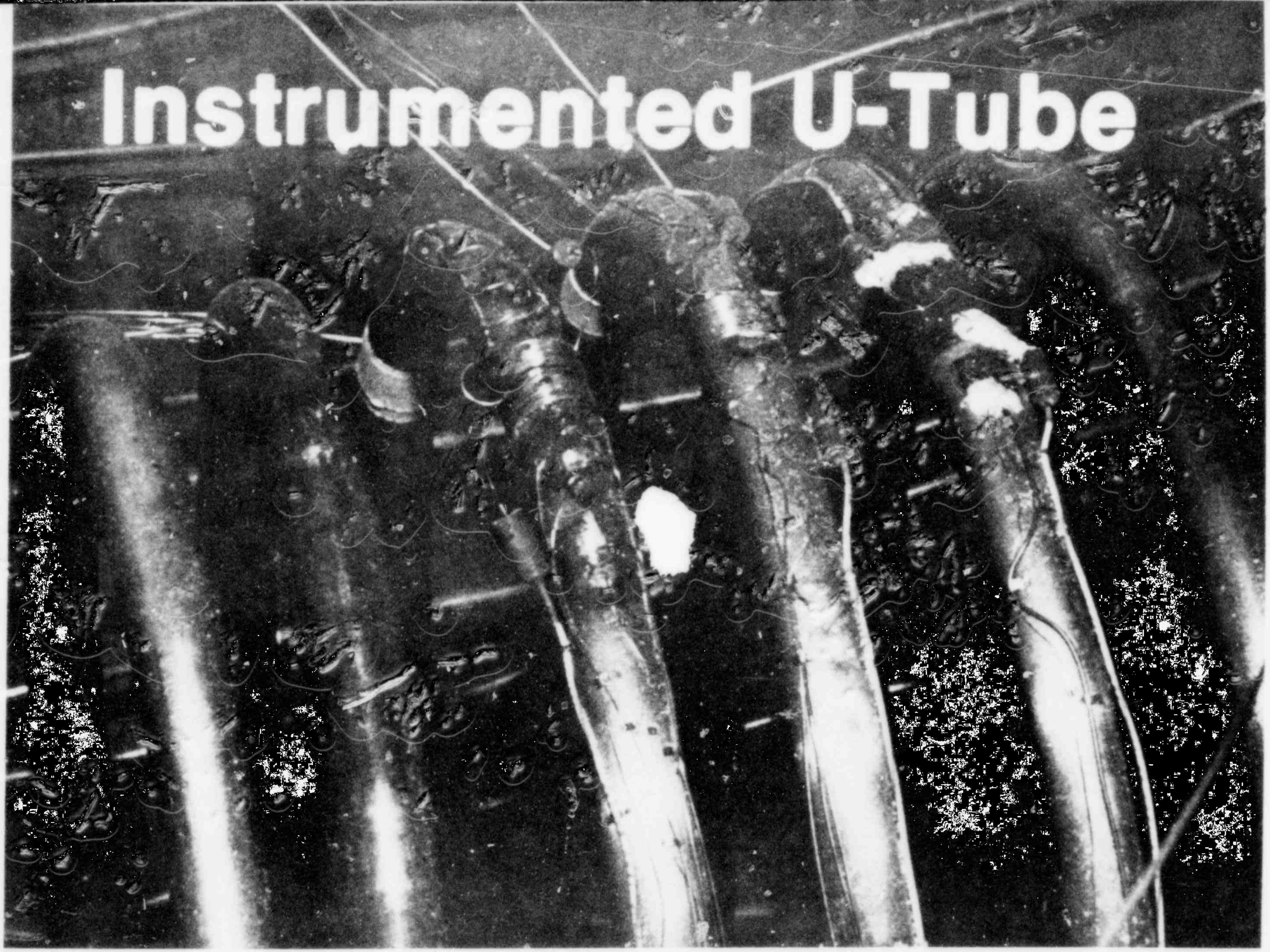


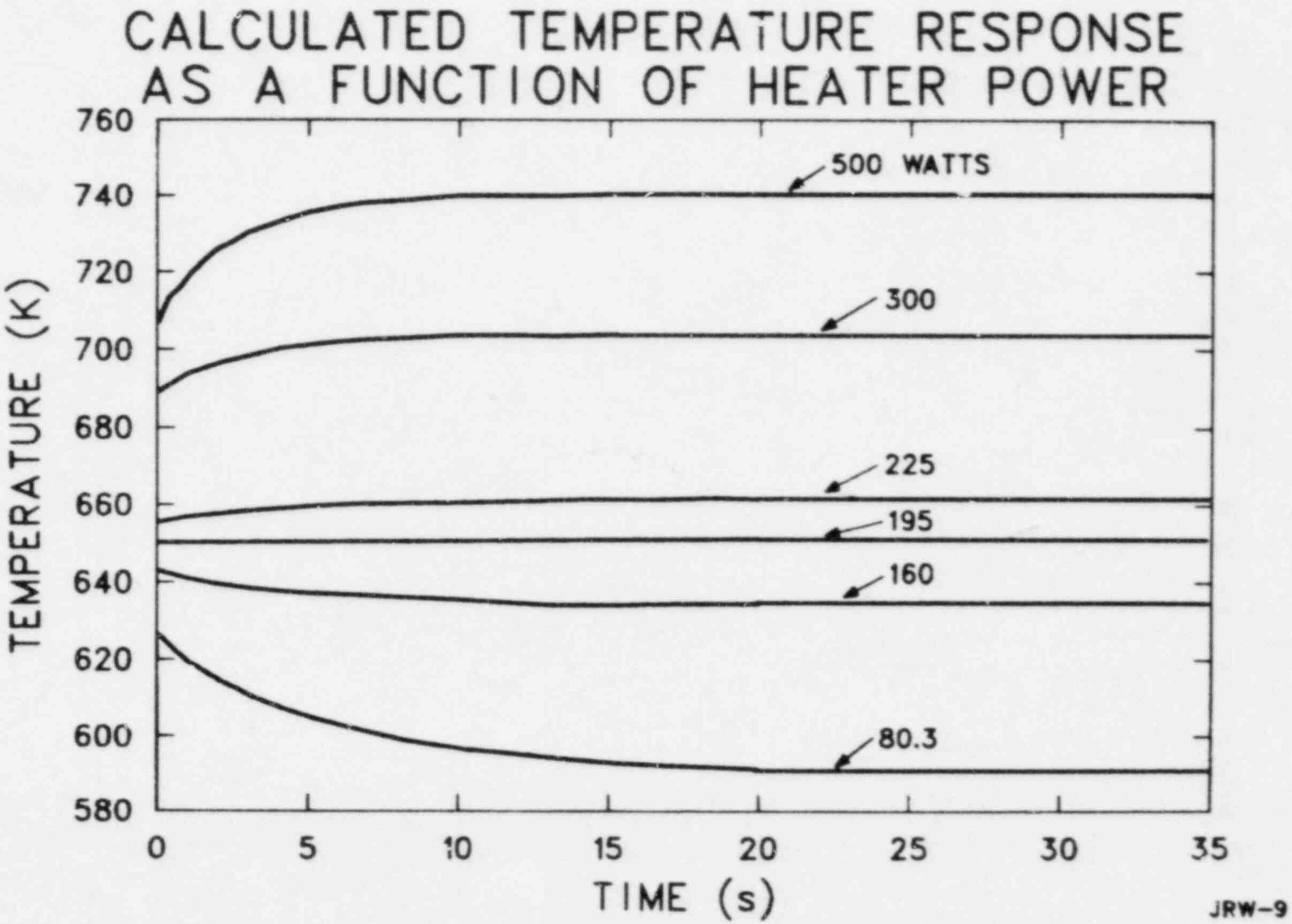
CALCULATED TEMPERATURE PROFILE
ALONG U-TUBE
TEMPERATURE (K)
380
375
370
365
360
355
350
345
340
0.0 m/s VOIDED
0.1 m/s
2.6 m/s
HEATER
SECONDARY TEMP
-5
-2.5
0
2.5
5
7.5
10
12.5
15
17.5
20
POSITION (cm)
JRW-6

TRANSIENT MODEL

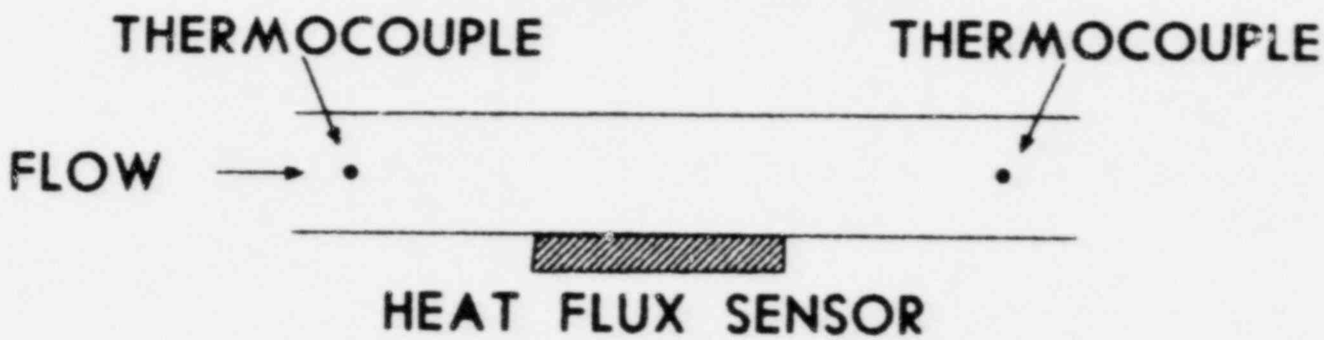
- LUMPED PARAMETER SYSTEM
- CONVECTIVE HEAT TRANSFER TO PRIMARY AND SECONDARY WITH FORCED AND NATURAL CONVECTION
- NO AXIAL CONDUCTION
- NO RADIAL TEMPERATURE DISTRIBUTION

JRW-7


Instrumented U-Tube


CALCULATED TEMPERATURE RESPONSE
AS A FUNCTION OF HEATER POWER
TEMPERATURE (K)
760
740
720
700
680
660
640
620
600
580
TIME (s)
0
5
10
15
20
25
30
35
500 WATTS
300
225
195
160
80.3
JRW-9

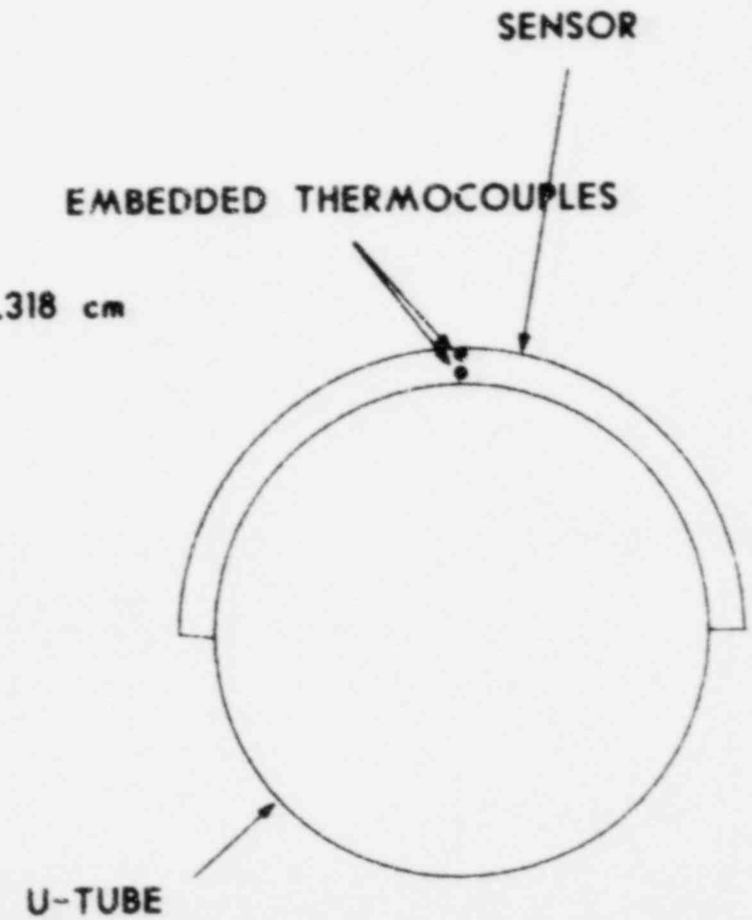
HEAT BALANCE



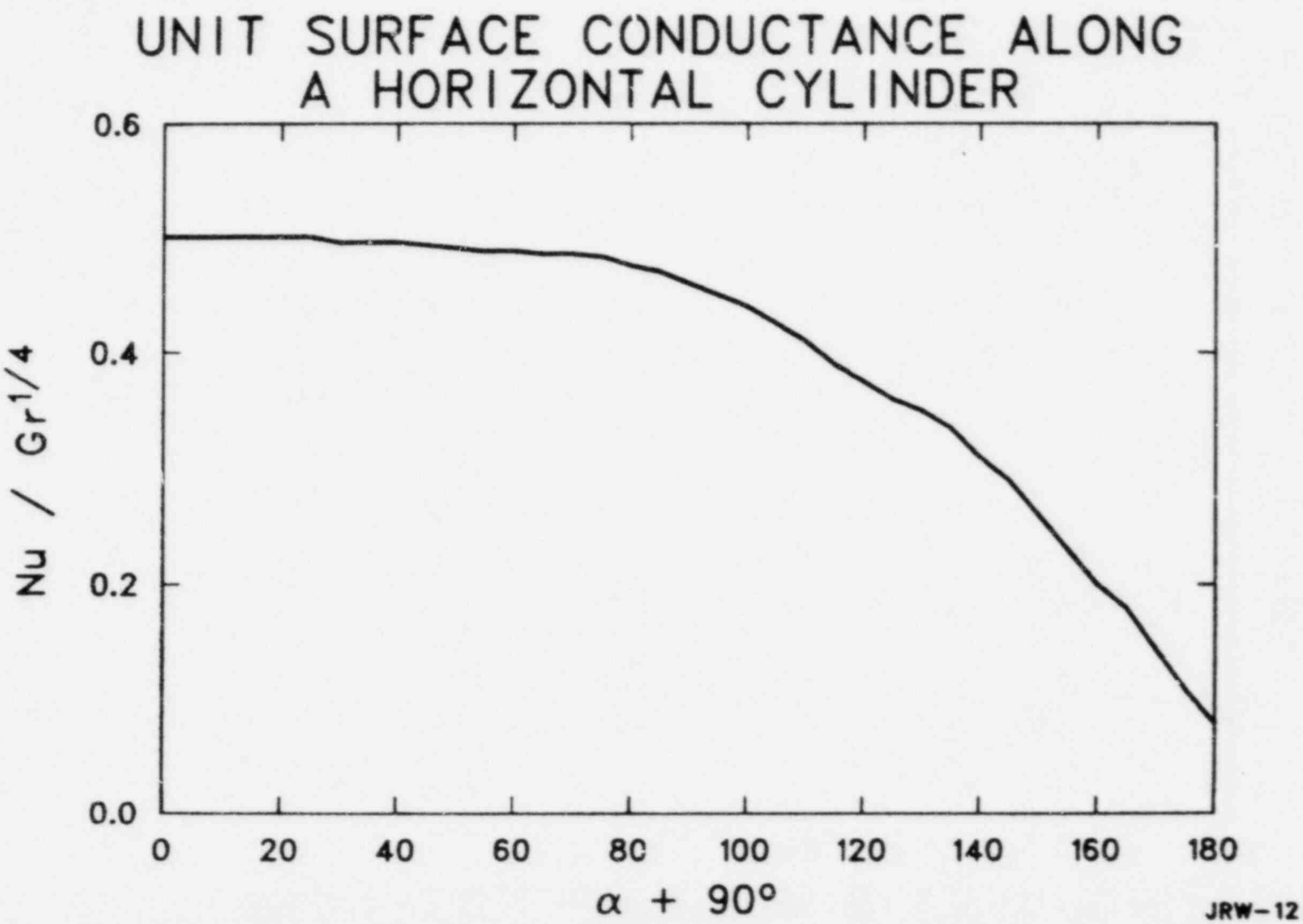
$$\frac{\dot{q}}{A} = \frac{C_p \rho V A_x \Delta T}{A_T} = C\left[\frac{2\pi k L\left[T_i - T_o\right]}{A_F \ln\left[\frac{r_o}{r_i}\right]}\right]$$

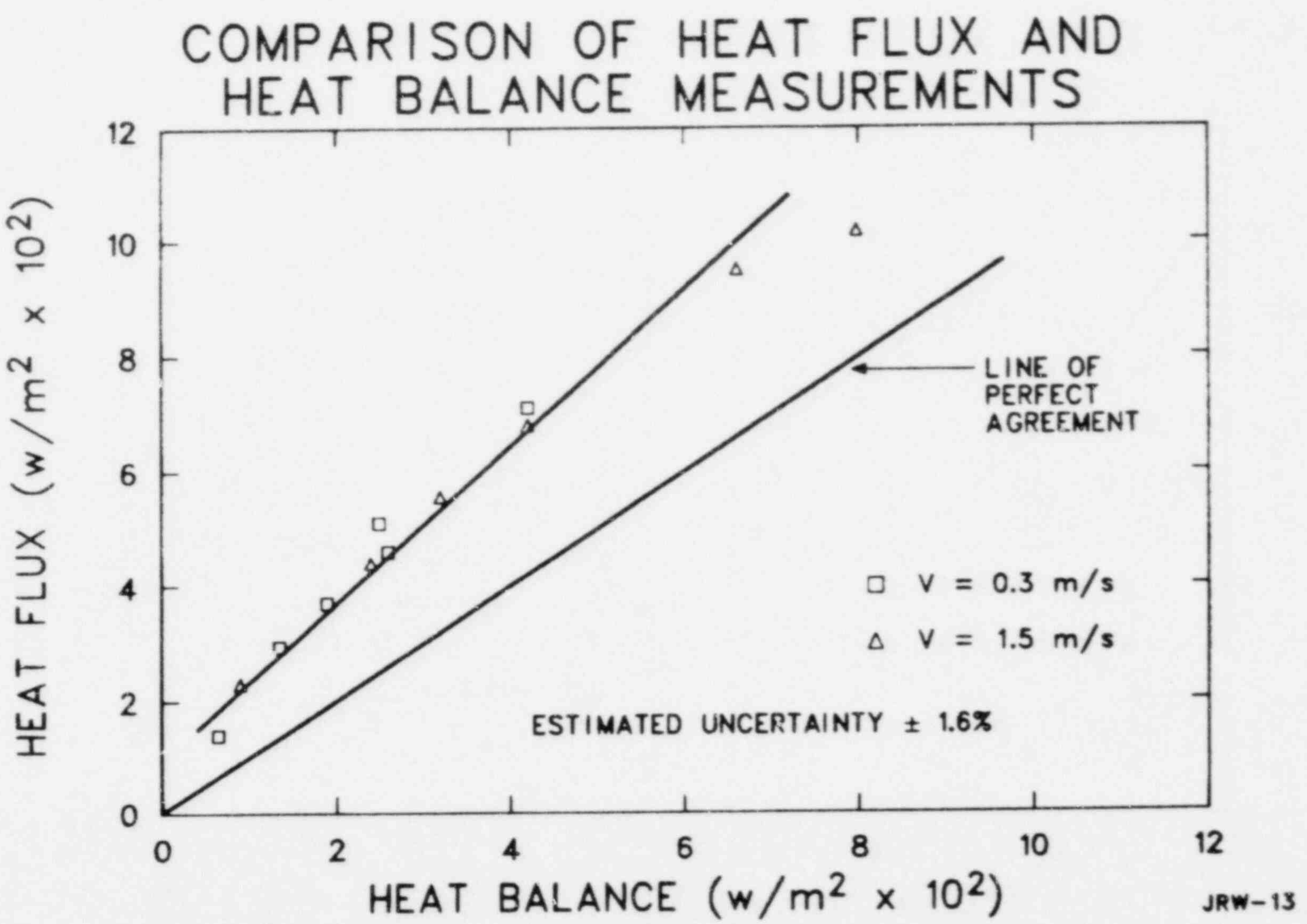
JRW-10

HEAT FLUX SENSOR



JRW-11


UNIT SURFACE CONDUCTANCE ALONG
A HORIZONTAL CYLINDER
0.6
0.4
0.2
0.0
Nu / Gr$^{1/4}$
0
20
40
60
80
100
120
140
160
180
α + 90°
JRW-12


COMPARISON OF HEAT FLUX AND
HEAT BALANCE MEASUREMENTS
HEAT FLUX (w/m² x 10²)
HEAT BALANCE (w/m² x 10²)
0
2
4
6
8
10
12
LINE OF
PERFECT
AGREEMENT
□ V = 0.3 m/s
△ V = 1.5 m/s
ESTIMATED UNCERTAINTY ± 1.6%
JRW-13

SUMMARY

- MODELED BLVD TC LOCATION AND TRANSIENT RESPONSE
- COMPLETED LABORATORY PROTOTYPE TESTS TO SHOW BLVD FEASIBILITY
- COMPLETED DESIGN AND FABRICATION OF LOCAL U-TUBE HEAT FLUX PROBE

JRW-14

NON-INTRUSIVE DENSITY PROFILE DETERMINATION:

GAMMA BEAM DENSITOMETER, TOMOGRAPHY AND SCATTERING

NED KONDIĆ
US NRC/WRSR-LOFT

PRESENTED AT THE

EIGHT WATER REACTOR SAFETY RESEARCH

INFORMATION MEETING

HELD AT THE NATIONAL BUREAU OF STANDARDS
GAITHERSBURG, MARYLAND

OCTOBER 27-31, 1980

NRC CONTRACTOR FOR THE WORK REPORTED: INEL/EG & G IDAHO FALLS, ID

NON-INTRUSIVE DENSITY PROFILE DETERMINATION: GAMMA BEAM DENSITOMETER, TOMOGRAPHY AND SCATTERING

GOAL: FIND OUT, OR MAKE THE BEST POSSIBLE ESTIMATE OF THE TWO-PHASE FLOW MASS DISTRIBUTION IN A PIPE CROSS-SECTION (Y, Z - PLANE)

MAIN PHYSICAL PRINCIPLES INVOLVED IN THE METHODS DISCUSSED:

1. RADIATION ATTENUATION ALONG THE BEAMS

    1.1. PRIMARY PHOTONS

    1.2. SECONDARY (SCATTERED) PHOTONS

2. RADIATION GENERATION WITHIN THE MASS

    2.1. PRIMARY GENERATION, FROM TAGGED OR EXCITED FLUID ATOMS

    2.2. SECONDARY GENERATION, FROM ATOMS INVOLVED IN SCATTERING

ALL THE ALTERNATIVES/METHODS DISCUSSED -- EXCEPT FOR 2.1. -- REQUIRE AN EXTERNAL GAMMA SOURCE, OR AN ARRAY OF SUCH SOURCES.

THE GEOMETRY OF PRIMARY INTEREST IS: CYLINDRICAL PIPE, BUT, IN PRINCIPLE, OTHER, EVEN IRREGULAR SHAPES OF FLUID CONTAINERS OR DUCTS, CAN BE USED.

GAMMA-BEAM DENSITOMETER

USES PRIMARY ATTENUATION OF 2-6 (OR MORE) BEAMS.
THROUGH COMPARATIVE COMPUTER ALGORITHMS, DIRECT READING (DETECTOR COUNT RATES) CAN BE RELIABLY INTERPRETED, SO THAT THEY RENDER:

- AVERAGE DENSITY AND
- FLOW PATTERN PARAMETERS (CHARACTERISTICS)

DATA ANALYSIS COVERS FOLLOWING FLOW PATTERNS: HOMOGENOUS, TILTED STRATIFIED, AND ECCENTRIC ANNULAR.

TOMOGRAPHY

IT USES EITHER PRIMARY ATTENUATION OR BOTH, PRIMARY GENERATION AND PRIMARY ATTENUATION (IN PRESENT EXPERIMENTS, PRIMARY GENERATION IS NOT USED).

CONTRARY TO (FIXED) MULTIBEAM GAMMA-BEAM DENSITOMETER, TOMOGRAPHY C A N GIVE CROSS-SECTIONAL DENSITY DISTRIBUTION WITHOUT ASSUMPTIONS OR AUXILIARY DATA.

SCATTERING

IT USES ATTENUATION OF THE PR MARY AND SECONDARY BEAMS, AS WELL AS THE SECONDARY GENERATION OF PHOTONS (GAMMAS). THUS, INHERRENTLY, IT CONTAINS MORE (HIDDEN) INFORMATION FOR EACH COUNT (RATE) REGISTERED, I.E. DETECTED. MATHEMATICAL ANALYSIS, NECESSARY TO EXTRACT FINAL RESULTS (LOCAL DENSITY VALUES) FROM THE COUNT RATES, DEPENDS ON THE GEOMETRICAL ARRANGEMENT CHOSEN FOR THE EXPERIMENTAL ASSEMBLY: SEVERAL OF THEM ARE FEASIBLE. ELIMINATION OF SOURCES AND DETECTORS MOVEMENT IS ACHIEVED BY USING 2 STATIONARY SOURCES (EXTERNAL) AND EITHER MULTIPLE DETECTORS OR 1 - 2 DETECTORS (STATIONARY IN BOTH CASES). IN THE LATTER OPTION, ENERGY DISCRIMINATION OF SCATTERED PHOTONS (USING COMPTON ENERGY/ANGLE EQUATION) DEFINES THE LOCATION OF THE SCATTERING POINT (MASS ELEMENT). NO ASSUMP'S OR AUX.DATA NEEDED.

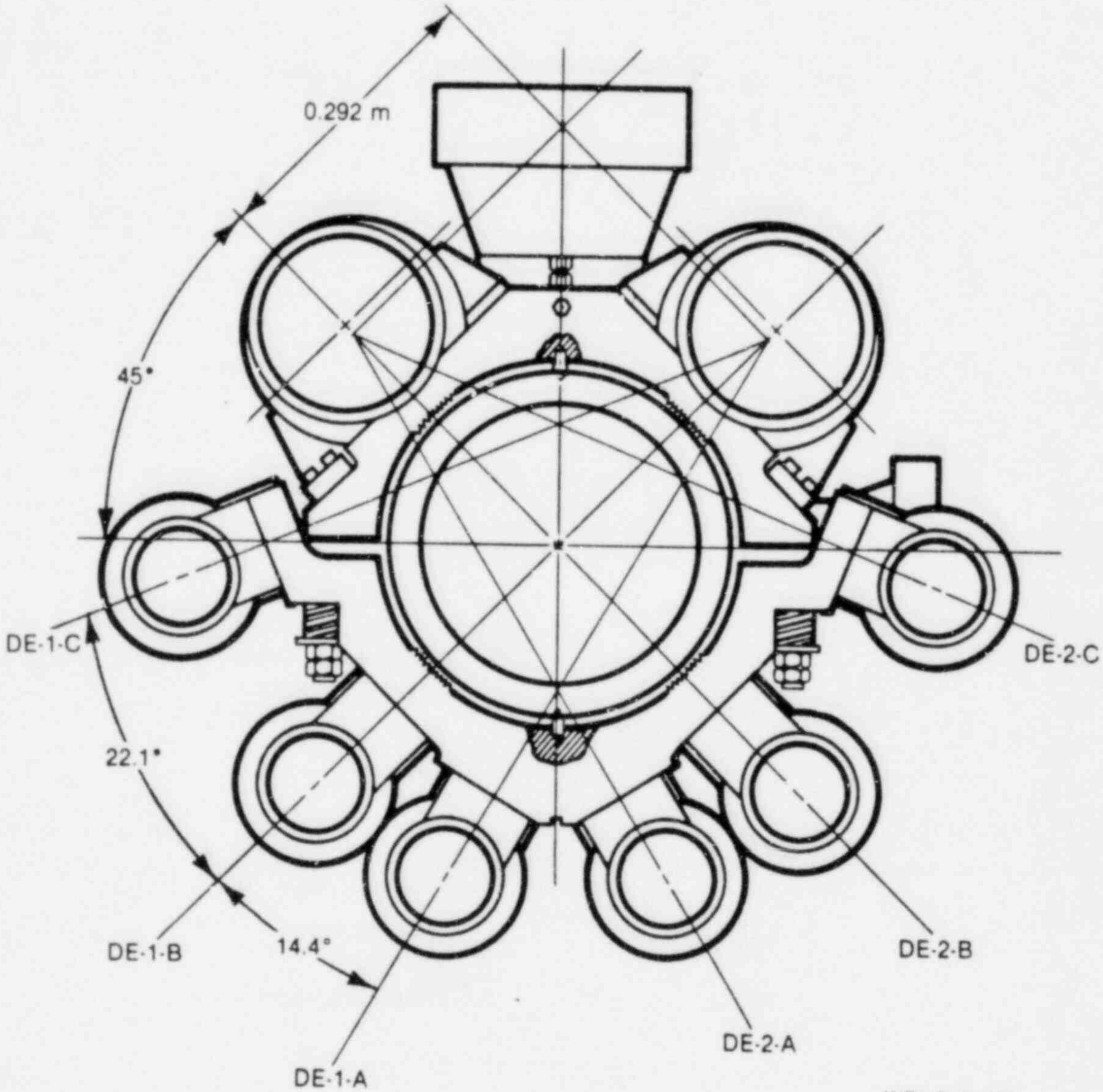


Figure 2. Cross-section of densitometer installation.
EXPERIMENTAL ASSMBLY USED AT THE TRANSIENT FLOW CALIBRATION FACILITY
(LOCATED IN W Y L E LABORATORIES,- NORCO, CALIFORNIA)

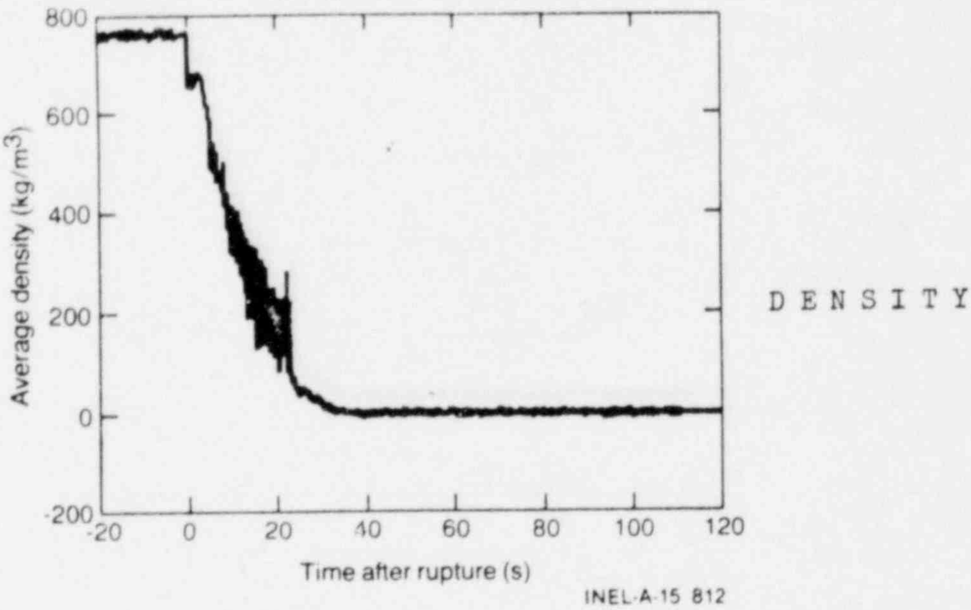


Figure 7. Average density from six-beam densitometer (Test IA1).

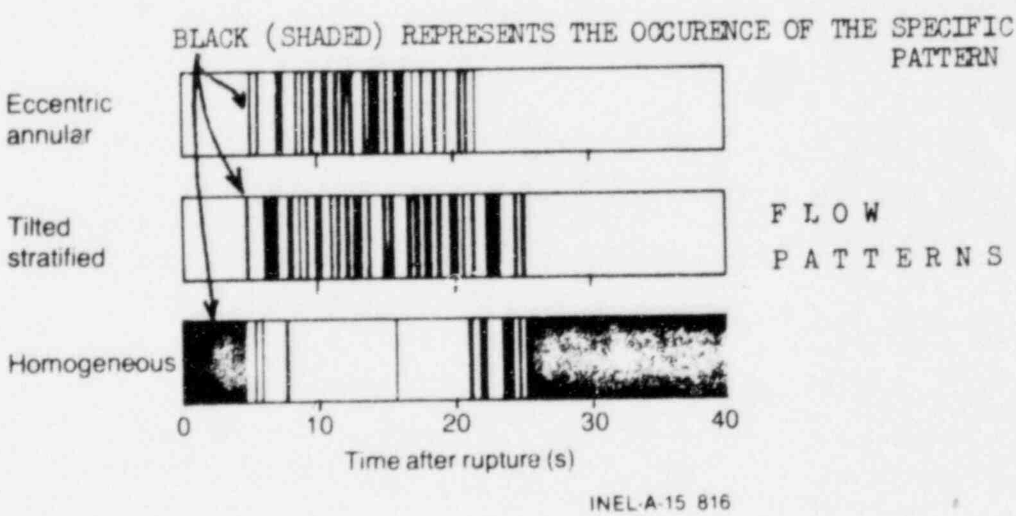


Figure 8. Density profile from six-beam densitometer (Test IIIA1).

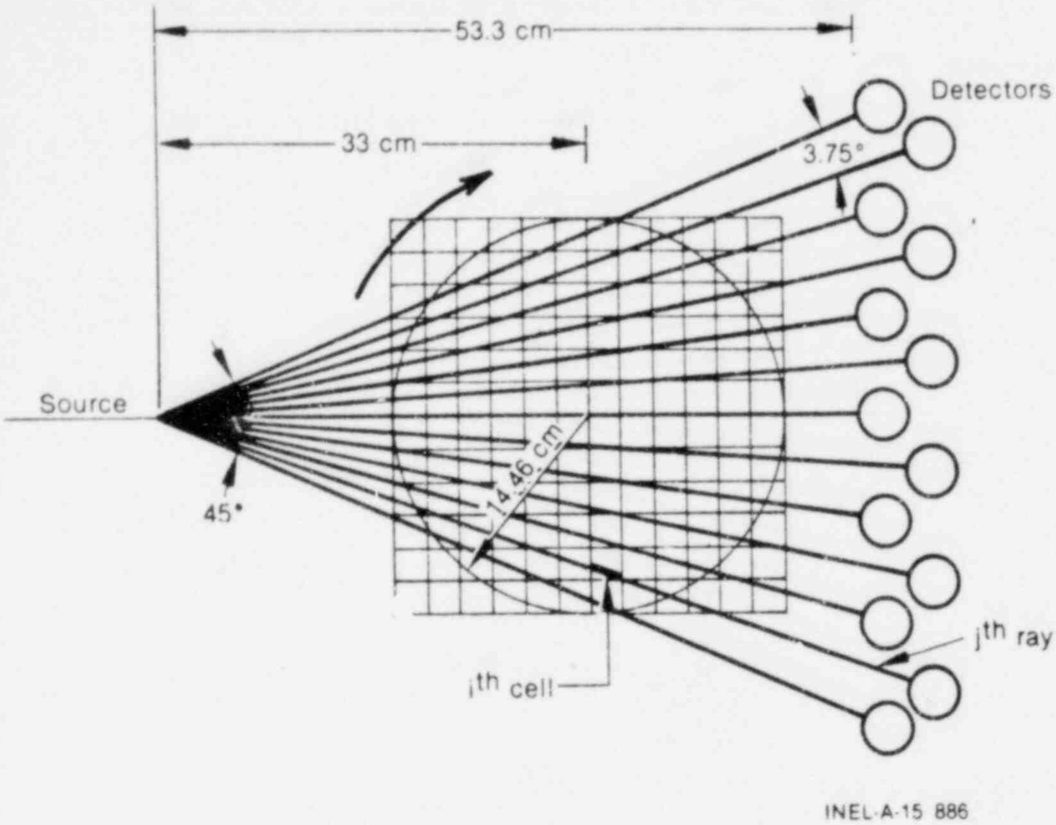


Figure 1. Tomographic densitometer ray geometry and pixel array used for iterative reconstruction. (The density field is bounded by the circle, which contains N cells along a diameter. The contribution of the i^{th} cell to the j^{th} ray (heavy line) is the weighting factor W_{ij}.

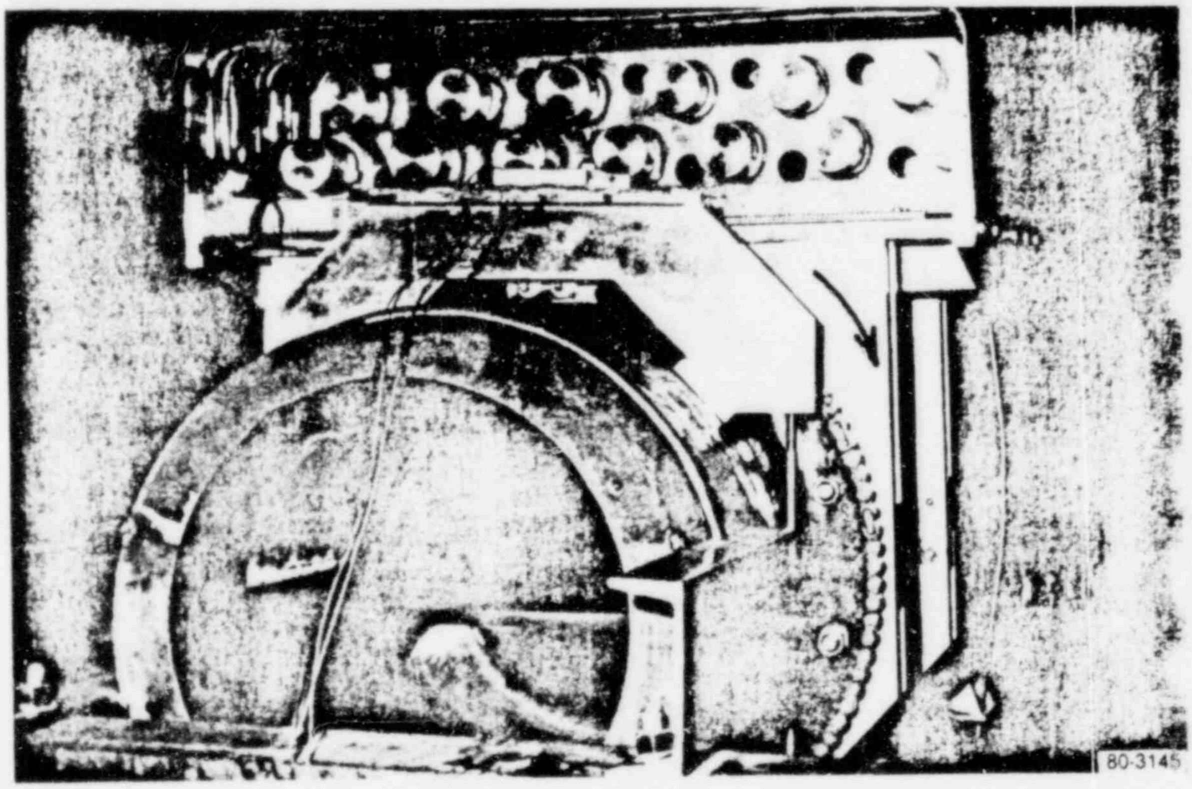
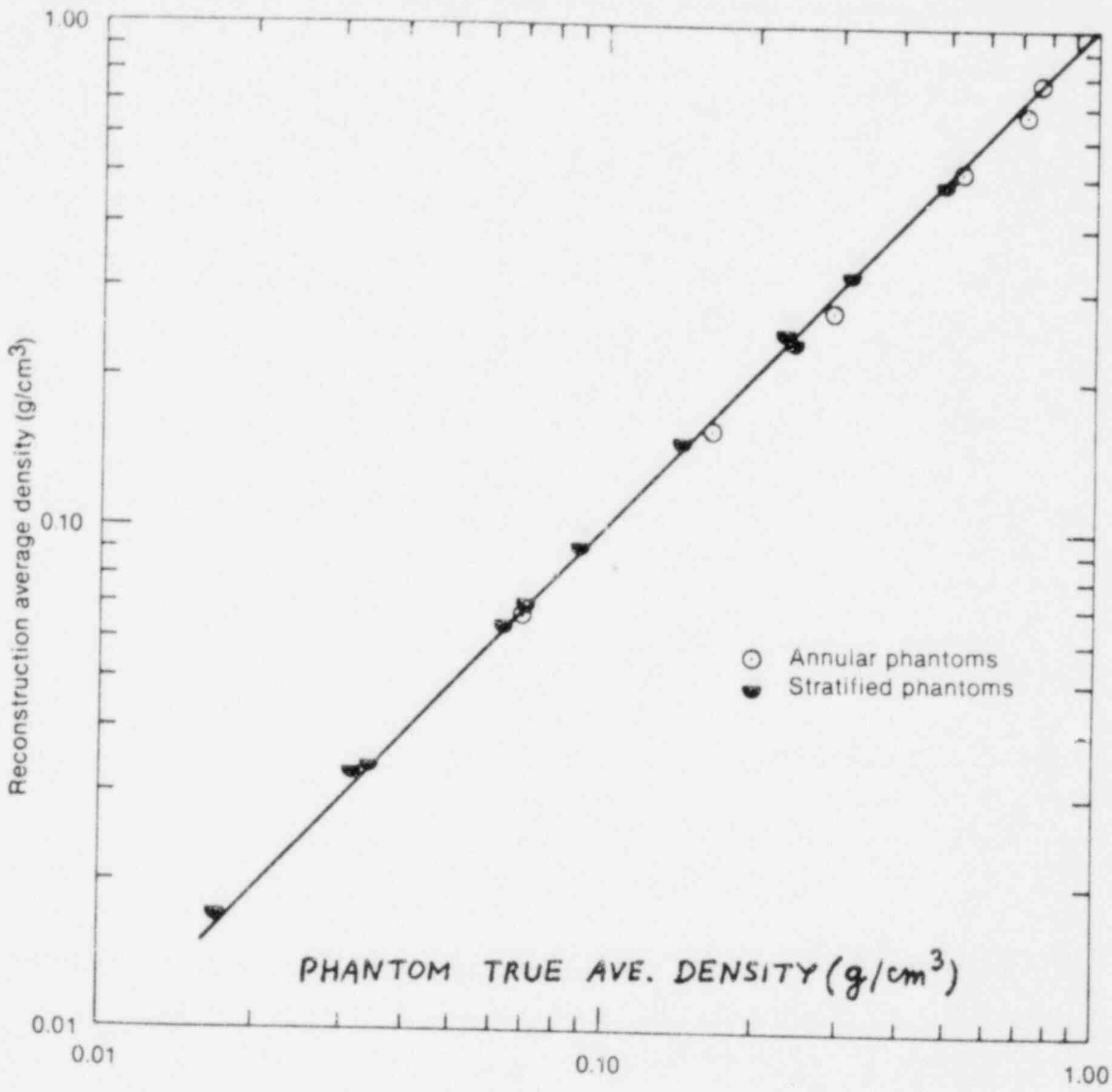
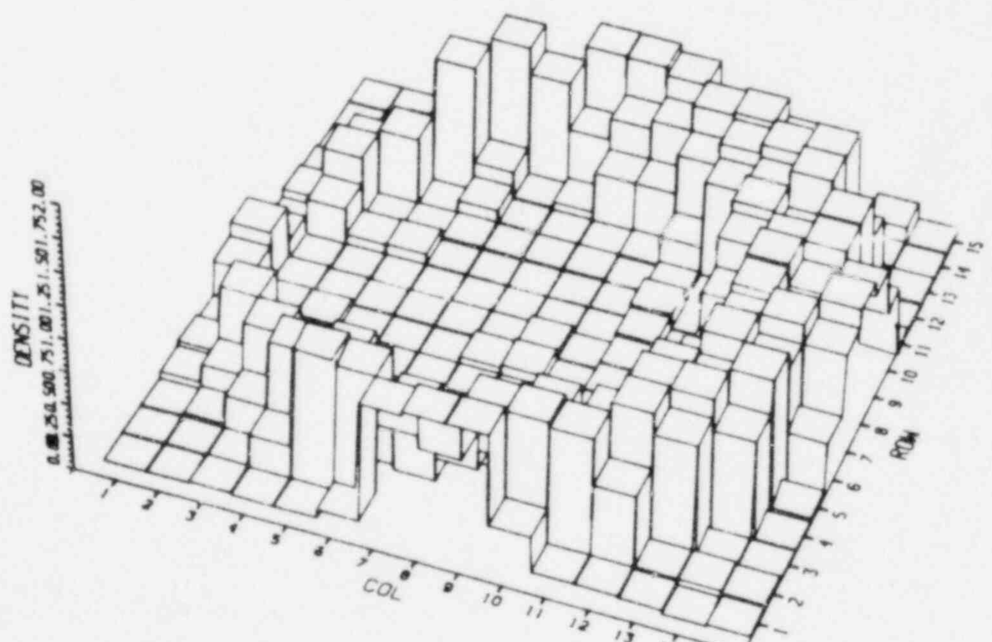


Figure 3. Tomographic densitometer detector array.


1.00
0.10
0.01
0.01
0.10
1.00
Reconstruction average density (g/cm3)
Annular phantoms
Stratified phantoms
PHANTOM TRUE AVE. DENSITY (g/cm3)


DENSITY
COL
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1 2 3 4 5 6 7 8 9 10 11 12 13 14 15
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15

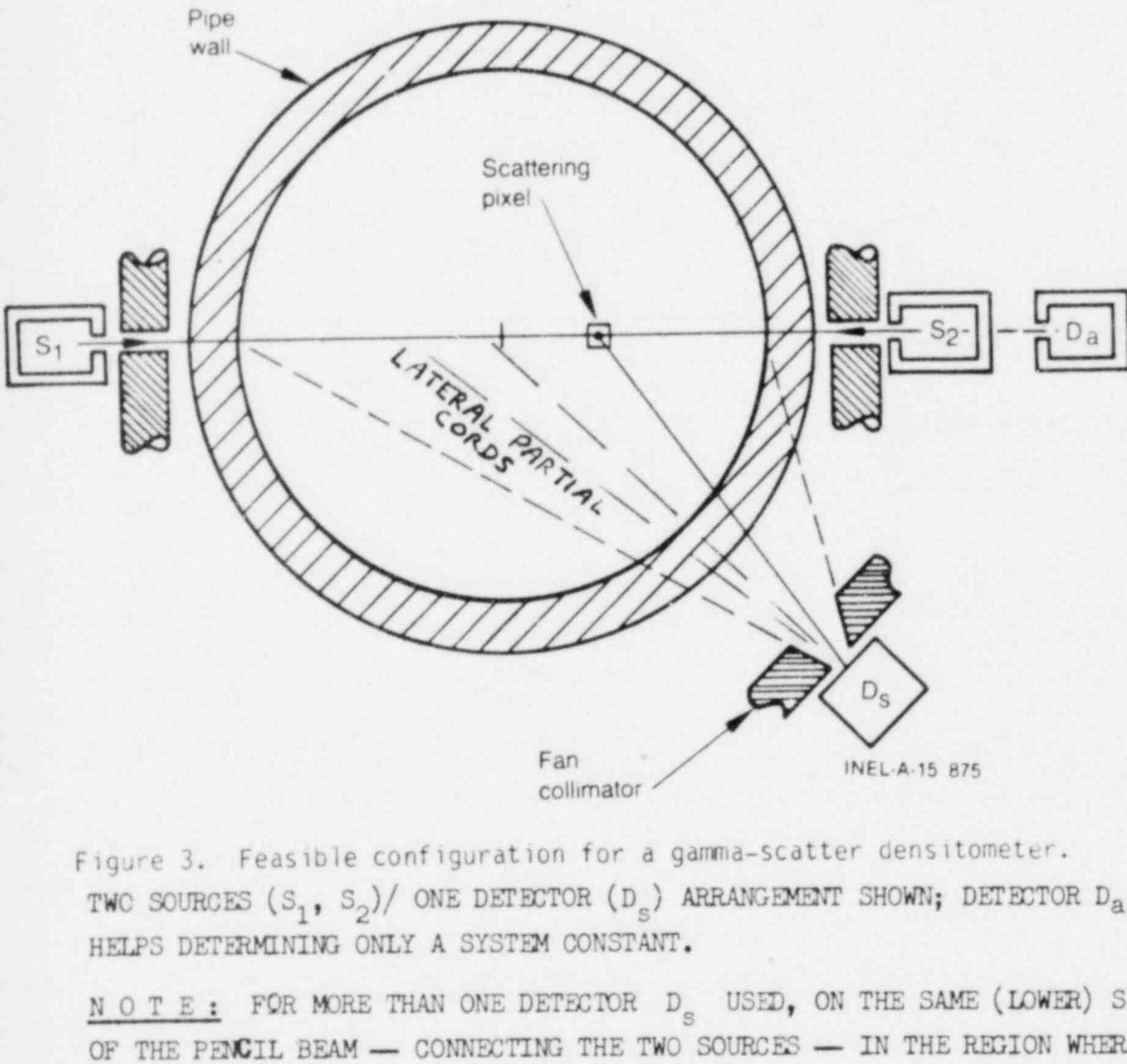


Figure 3. Feasible configuration for a gamma-scatter densitometer.
TWO SOURCES (S_1, S_2)/ ONE DETECTOR (D_s) ARRANGEMENT SHOWN; DETECTOR D_a HELPS DETERMINING ONLY A SYSTEM CONSTANT.

N O T E : FOR MORE THAN ONE DETECTOR D_s USED, ON THE SAME (LOWER) SIDE OF THE PENCIL BEAM — CONNECTING THE TWO SOURCES — IN THE REGION WHERE LATERAL PARTIAL CORDS OVERLAP, CONDITIONS START TO BE MET FOR A TOMOGRAPHIC RECONSTRUCTION OF THE DENSITY FIELD IN THAT REGION. THIS WOULD BE DONE ANALYTICALLY, WITHOUT ANY ADDITION OF SOURCES, DETECTORS OR MOVEMENT OF THE EXPERIMENTAL ASSEMBLY.

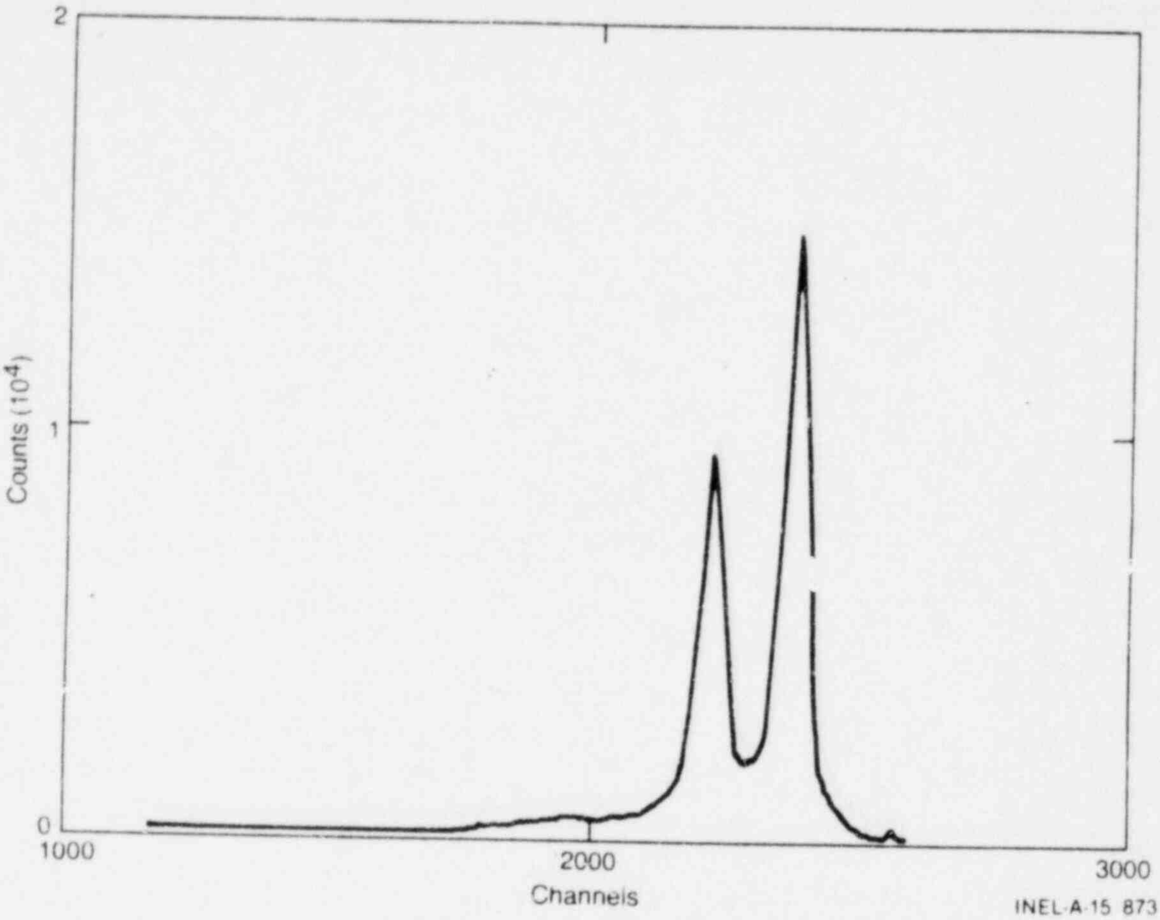


Figure 8. Scattering spectrum for 2nd-and-9th-pixels phantom.

CONCLUSION

TOMOGRAPHY AND SCATTERING -- CONTRARY TO SOME VIEWS -- CAN BE USED TO DETERMINE CONCRETE NUMERICAL VALUES OF THE CROSS-SECTIONAL DENSITY FIELD ALSO IN TRANSIENT TURBULENT TWO-PHASE FLOW, WITHOUT CHANGING THE DATA ANALYSIS PROCEDURE COMPLETED BY NGW. A MERELY QUANTITATIVE MODIFICATION OF THE EXPERIMENTAL SETUP (AFFECTING THE COST) WOULD REQUIRE FOLLOWING:

1. AN ARRAY OF STATIONARY SOURCES AND DETECTORS AROUND THE PIPE (FOR TOMOGRAPHY)
2. AN INCREASED STRENGTH OF GAMMA SOURCES (KCURIE OR MORE) OR FLASH X-RAY GENERATORS; DETECTOR REQUIREMENT REMAINS SAME AS FOR STEADY STATE(FOR SCATTER.)
3. COMPUTER SOFTWARE TO DECIPHER (TOMOGR.) OR ENHANCE THE ACCURACY (SCATT.).

BASED ON ALL OF THE ABOVE, <u>TOMOGRAPHIC DENSITOMETER</u> WAS CHOSEN AS THE REFERENCE INSTRUMENT FOR DENSITY DISTRIBUTION DETERMINATION IN THE NEW STEADY-STATE CALIBRATION FACILITY (AT INEL/EG&G, IDAHO FALLS). GEOMETRY AND FLUID CONDITIONS ARE SAME AS AT LOFT, I.E. P/T OF THE PRIMARY LOOP OF COMMERCIAL PWR'S WITH 14 IN. O.D. PIPE.

Presentation at the Eighth NRC Information Meeting

on October 29, 1980

INTRODUCTION

TECHNICAL PERSPECTIVE AND OBJECTIVES OF 2D/3D RESEARCH PROGRAM

by L. S. Tong

During the reflood stage of a loss-of-coolant accident (LOCA) in a PWR, the liquid droplets carried out by the upward steam flow from the core may pass through the upper plenum and be carried over into the steam generator. The evaporation of this carryover water in the steam generators may increase the upper plenum pressure and hence reduce the core liquid level, leaving the fuel rod partially uncovered and uncooled. This phenomena is called steam binding. In the present licensing criteria based on a conservative analysis of cold-leg ECCS injection, liquid carried out of the core is assumed to be all vaporized in the steam generators. This assumption leads to an overly high pressure in the upper plenum because, in fact, the crowded internals in the upper plenum serve as strong moisture separators, and the amount of liquid carried through the upper plenum into the steam generators should be greatly reduced by de-entrainment. Experimental data is needed in order to determine the amount of de-entrainment of moisture in the upper plenum so that a realistic estimate of the steam binding effect in a PWR can be made. The calculated steam binding effect contribution to the maximum peak clad temperature in a LOCA with a cold-leg ECC injection system could be as high as 270^{o}F.

For a system in which the ECC is injected into the upper plenum, the effect of steam binding competes with the condensation effect of the subcooled injection water. Other concerns with upper plenum ECC injection are the distribution of injection water in the upper plenum and the fallback of the accumulated water into the core. These phenomena are believed to be strongly influenced by the size or scale of the testing facility. Therefore, large-scale tests of the above phenomena are needed if realistic answers to these questions are to be obtained.

During the blowdown and refill stages of a large-break LOCA in a PWR, the ECC water injected at the cold legs may not immediately penetrate through the downcomer into the lower plenum and provide for core cooling, because of the momentum effect of the upward steam flow in the downcomer. The flow transient induced by the condensation of steam and ECC water in the upper annulus of the downcomer also could delay the lower plenum filling. These delays would also result in an increase of peak clad temperature. The prediction of the exact amount of time delay is difficult because the flow pattern in the downcomer is oscillatory and gap-size dependent. At present, Appendix K conservatively assumes that all water injected into the downcomer during blow-down is bypassed. A recent empirical correlation of ECC penetration has been developed based on small-scale test data and is waiting for confirmation, utilizing data from large-scale tests.

During a small-break LOCA, the effectiveness of natural circulation and reflux boiling in providing core cooling for various anomalous transients requires confirmation. Specifically, the CCFL in the hot leg and steam generator in a reflux boiler mode of core cooling should be tested with a large-scale model.

The effectiveness of upper plenum ECC injection for condensing the steam bubble in the upper plenum and for rapid cooling of a partially uncovered core during a small-break LOCA also needs testing.

The facilities designed for conducting the above tests are shown in Table 1.

The objectives of the 2D/3D program are summarized in the following:

1. To study the effectiveness of various ECCS during reflood for a large-break LOCA (including cold-leg injection, combined hot-leg and cold-leg injection, lower plenum injection and vent valve) by measuring:

    - the liquid carryover and fallback at upper core support plate,
    - the de-entrainment of liquid in upper plenum,
    - the pressure difference between the upper plenum and the top of downcomer,
    - the pressure drop across steam generators.

2. To study the effectiveness of various ECCS during refill for a large-break LOCA by measuring:

    - ECC penetration in downcomer and lower plenum filling during refill,
    - downcomer flow transient induced by the condensation of steam by ECC water during refill,
    - U-tube flow oscillation during refill,
    - pool height and temperature of water accumulated in upper plenum under combined injection during refill and reflood.

3. To study the events leading to core uncovery during a small-break LOCA by measuring:

   - net flow rate out of the vessel in a natural circulation of the system,
   - heat transfer in the steam generators,
   - condensing steam bubble in upper plenum and core cooling of a partially uncovered core through hot-leg injection,
   - phase separation and counter-current flow limit in the hot leg during reflux boiling,
   - local liquid level and fluid temperatures in the core, upper plenum, exit nozzle, downcomer and lower plenum.

4. To study convective flow and fuel clad temperature distribution inside a heated core under the following conditions:

   (a) during reflood for a large-break LOCA (chimney effect),

   (b) during core uncovery of a small-break LOCA,

   (c) under conditions of core flow blockage during reflood, by measuring:

   - density distribution in the core,
   - velocity distribution in the core,
   - location and clad temperatures at hot spots.

TABLE 1

2D/3D TEST FACILITIES

SCOPE OF TESTING	GERMANY	JAPAN	USNRC
INTEGRAL TESTS • LARGE-BREAK LOCA (ALTERNATE ECCS) • SMALL BREAK (NAT. CIRC. & CORE UNCOV.)	PKL 340-ROD (FULL-HEIGHT CORE, 3-LOOP)	CCTF 2000-ROD (FULL-HEIGHT CORE 4-LOOP)	ADVANCED INSTR. DESIGN SUPPORT TRAC ANALYSIS
LARGE-SCALE SEPARATE EFFECTS TESTS • ECC PENETRATION & BYPASS • STEAM BINDING COUPLING • FLOW BLOCKAGE • CCFL & PHASE SEP. (SMALL BREAK)	UPTF FULL-SCALE PWR VESSEL FULL-SCALE DOWNCOMER	SCTF 2000-ROD (FULL-HEIGHT 6-FT. RADIAL SLAB)	ADVANCED INSTR. DESIGN SUPPORT TRAC ANALYSIS

NOTE: PKL - PRIMARKREISLAUF
CCTF - CYLINDRICAL CORE TEST FACILITY
SCTF - SLAB CORE TEST FACILITY
UPTF - UPPER PLENUM TEST FACILITY

A SYNOPSIS OF PKL SMALL BREAK TESTS

D. Hein, F. Winkler
Kraftwerk Union AG, Erlangen

Paper presented at the 8th Water Reactor Saftey Research Information Meeting
October 27 to 31, 1980 - Gaithersburg, U.S.A

A Synopsis of Small Break Tests

1. Introduction

The KWU PKL test facility /1/ was designed to examine the system behaviour of a PWR during the refill and reflood phase of a loss of coolant accident with the option to include the end of blowdown phase. The design pressure of the loop is 35 bars. This fact allowed to modify PKL in a relatively short time to run tests to answer questions arising after the TMI accident in the field of small break LOCAs.

As opposed to large breaks, small breaks are characterized not by voiding of the primary system and subsequent fast reflooding, but rather by maintaining sufficient coolant inventory over an extended high pressure period. In addition to the energy transport out of the primary system due to the break flow, an additional heat sink is needed to remove heat from the primary system. This additional heat sink is provided by the steam generators whose secondary side is cooled down automatically by 100 K/h. To avoid overheating of the core during a small break LOCA the interest has to be concentrated on two items:

- sufficient water inventory to keep the core covered,
- energy transport from the core to the steam generators at different water inventories.

The main features for small break experiments, discussed in international cooperation, are shown in fig. 1. In this list those problems are marked, in which PKL tests can contribute.

Integral Behaviour	investigated in PKL
● Free Convection 2 Phase Integral Behaviour (Steady State Tests)	×
● Transient Energy and Mass Transfer	×
Separate Effects	
● Heat Transfer in Steam Generator	×
● Mixture Level in Core, Downcomer etc.	×
● Reflux Boiler/Reflux Condenser	×
● Countercurrent Flow in Primary Horizontal Loop Pipes	
● Core Uncovered Heat Transfer	(×)
● Influence of Non – Condensible Gas	×
● Flow Blockage in Core	
● Nuclear Feedback	

Fig. 1: Main Features for Small Break Experiments

2. Test Facility

The PKL test facility - the program is supported by the German Minister of Research and Technology - represents a typical KWU 1300 MWe 4 Loop PWR on a model scale of 1:134 /1/. It was designed to simulate the behaviour of the entire primary system during the refill and reflood phase of a LOCA /2, 3/ In view of the importance of the driving gravity forces during reflood as well as for natural circulation, all elevations correspond to actual reactor dimensions. The test facility is designed for maximum pressure of 35 bars and can cover the important phase of small break LOCA from 35 bars down to 10 bars, the pressure at which the low pressure injection is initiated. In figure 2 the PKL test facility is shown as modified for small break tests. The test bundle simulating the core consists of 340 electrically heated rods. The three loops - one with double capacity simulating two loops - contain active steam generators whose secondary side can be cooled down.

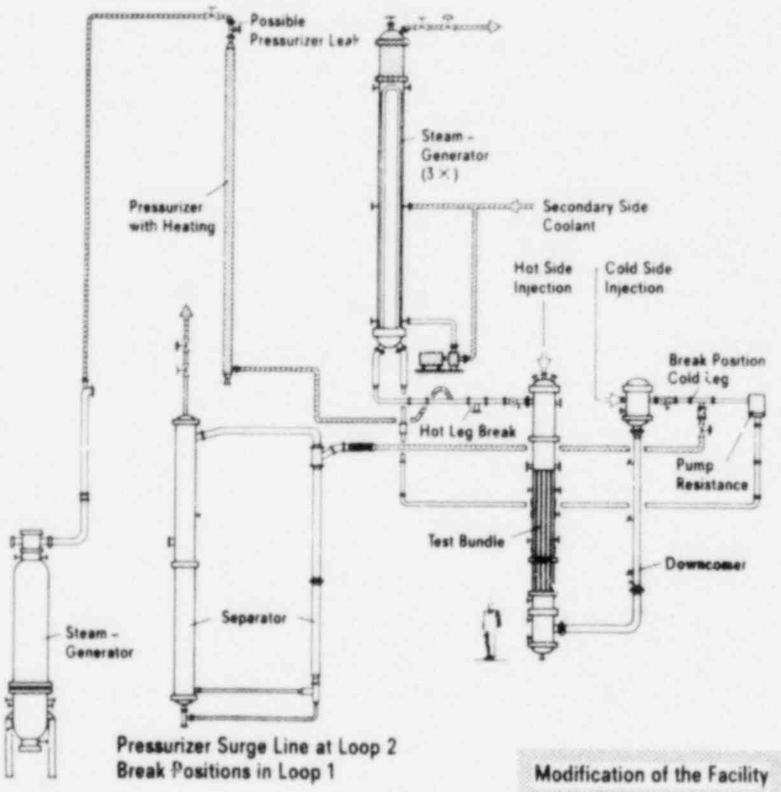


Fig. 2: PKL - Arrangement for Small Break Tests

3. Test Matrix

The objectives of the PKL small break tests are

- to enhance the understandings of the phenomena, energy transport and transport mechanisms pertaining to small break LOCAs.

- to aid the development of models describing these phenomena
- To provide a data base for code development and assessment

To attain these objectives steady state tests without break and ECC injection were carried out to study the energy transport mechanism with full and reduced water inventories in the primary loops.

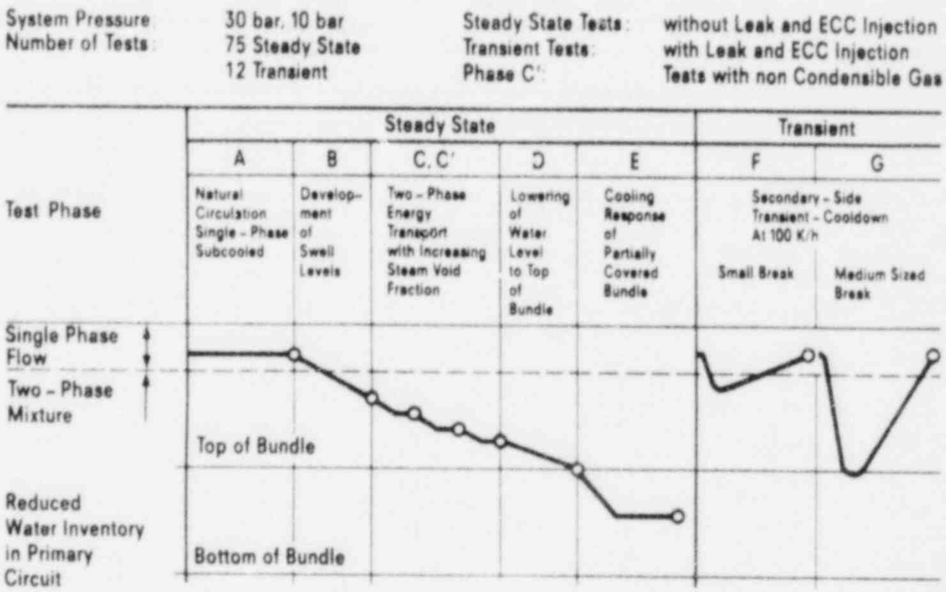


Fig. 3: Matrix of the PKL - Small Break Tests

4. Results

4.1 Steady State Tests

The thermohydraulics and heat transfer mechanisms during a small break LOCA cover a wide range form natural circulation with single or two-phase flow to phase separation with counter current flow of steam and water. Starting with single phase natural circulation, the water inventory in the test facility was reduced step by step until reflux condenser behaviour in the steam generator was established. The test conditions and the main test results are listed in figure 4.

ID1	Bundle Power	Water ** Inventory	Mode of Energy Transport	Flow Rate $\dot{m}$	$\Delta\vartheta_{p-s}$	p_p	p_s	Δp_{p-s}
–	kW	%		kg/s	K	bar	bar	bar
4	402	100	Subcooled Natural Circulation	4.5	17	28.8	18.8	10.0
5	625	100	Subcooled Natural Circulation	5.4	25	29.7	17.8	11.9
6*	404	99	Single Phase Natural Circulation	5.4	16	30.1	23.3	6.8
7*	405	96	Single Phase Natural Circulation	4.9	16	30.0	23.3	6.7
8	409	95	Two Phase Circulation	9.1	3 – 5	30.0	28.8	1.2
9	410	93	Two Phase Circulation	7.5	3 – 5	29.8	28.7	1.1
10	413	87	Two Phase Circulation	4.2	3 – 5	30.5	29.6	0.9
11	411	80	Two Phase Circulation	3.2	3 – 5	30.0	29.1	0.9
12	412	84	Two Phase Circulation	1.7	3 – 5	30.2	29.7	0.5
13	412	80	Reflux Condenser	0	2 – 4	30.2	30.0	0.2
14	411	51	Reflux Condenser	0	2 – 4	29.5	29.5	0
15	641	53	Reflux Condenser	0	2 – 4	29.2	29.1	0.1

* Some Steam in Upper Plenum
** Water Inventory Primary Side Pressurizer Not included

Fig. 4: Summary of Results from Test ID 1

For this sequence of steady state tests, results are shown in figure 5 for the mass flow rate at different values of water inventory within the primary system. With single phase natural circulation the mass flow rate is nearly constant for different degrees of subcooling. An increase in fluid temperature results in steam production in the core. But, while this vapour is being collected in the upper dome of the vessel, natural circulation remains unchanged.

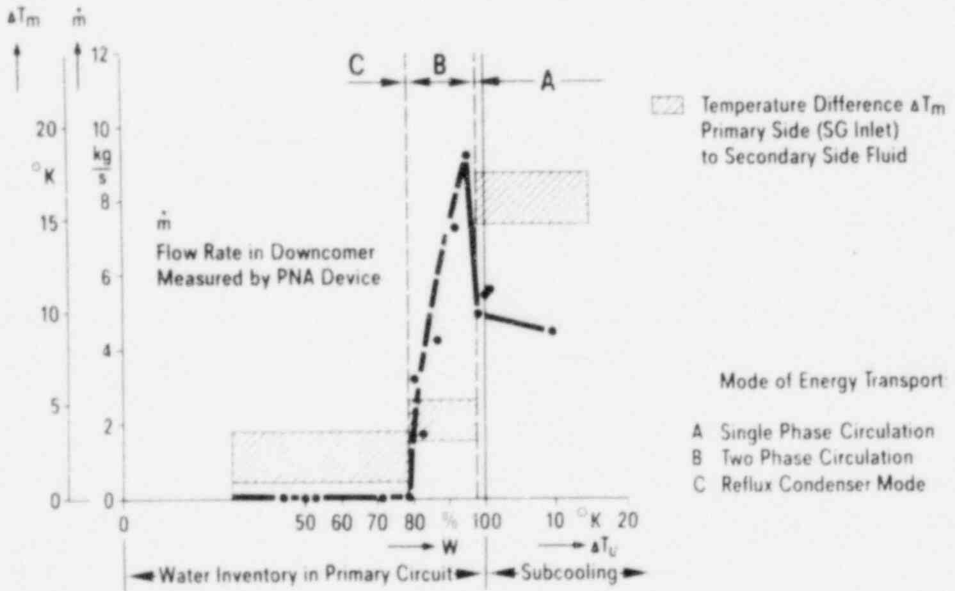


Fig. 5: Results from Tests ID 1

When the upper dome of the vessel is completely filled with vapour a further reduction of the water inventory changes the single phase conditions in the hot leg and the steam generator into a two-phase mixture. This results in an additional driving force for natural circulation and the flow rate increases. With a further reduction of the water inventory, phase separation occurs at the top of the U-tubes and the flow circulation slows down significantly and appears to be non-existent at about 80 % water inventory.

Considering heat transfer, represented by the temperature difference between primary and secondary side, a 16 K temperature difference is needed to transfer 400 kW from the primary to the secondary side fluid by single phase natural circulation. With two-phase energy transport the heat transfer is enhanced by condensation resulting in a reduction of the temperature difference down to about 5 K. After the breakdown of the natural circulation due to phase separation the energy is transferred by boiling and condensation only. Vapour is produced in the core and flows upwards via the hot legs to the steam generators where it condenses. The condensate flows back to the vessel. This counter current flow situation and heat transfer mode - well known from heat pipes - is calles "reflux condenser mode" and is a very effective energy transport mechanism. Thus the temperature difference between the primary and secondary side is only about 2 K.

These tests proved that the decay heat from the core can be transferred to the heat sink by natural circulation or even better by heat transfer in the reflux condenser mode: natural circulation is not a requirement for the energy transport from the core to the steam generators.

While running the test facility with a water inventory at which the energy is transported in the reflux condenser mode the influences of

- core power
- secondary side water level
- non-condensible gas

were investigated as shown in fig. 6. These tests were single loop tests: all loops except loop 1 were blocked off.

Test-Nr.	Bundle Power kW	System Pressure bar	SG Water Level m	Non-Condensible Gas (N_2) kg
ID 14	200/300/400	30	9*	-
ID 15	200/300/400	10	9*	-
ID 16	200/300/400	10	9*	-
ID 17	160	10	9*/7/5	-
ID 18	160	30	9*/7/5	-
ID 19	160	10	9*	0,12/0,24/0,36

*U-Tubes Completely Covered

Fig. 7: Reflux condenser mode, influence of power

The influence of a reduced secondary side water level on the energy transport during der reflux condenser mode is shown in fig. 8. It can be seen that the temperature difference increases with reduced heat exchange area. The temperature difference doubles from 2 K with fully covered U-tubes to about 4 K with only half of the U-tubes area available for heat transfer.

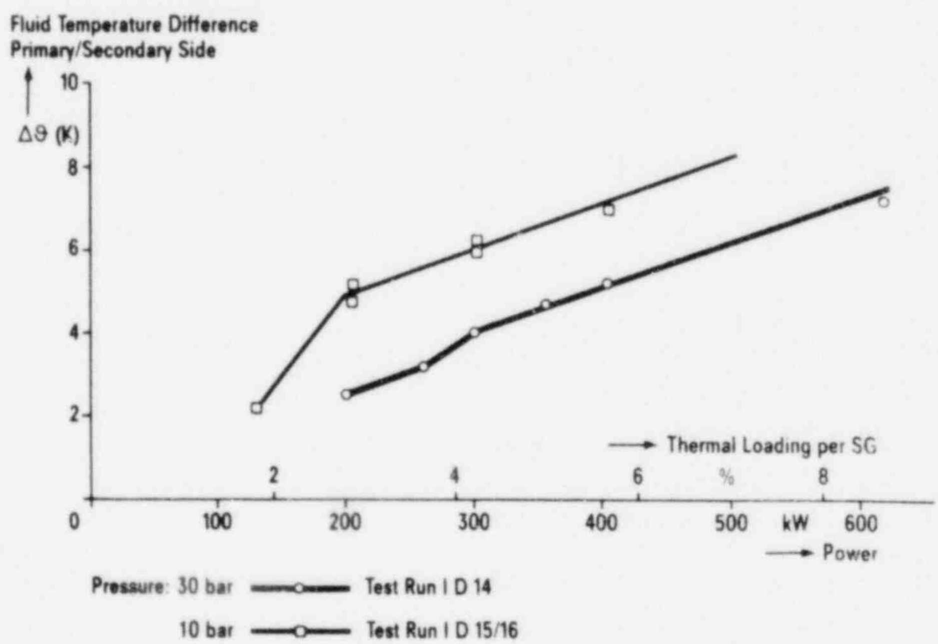


Fig. 8: Reflux condenser mode, influence of secondary side water level

The influence of a reduced secondary side water level on the energy transport during the reflux condenser mode is shown in figure 8. It can be seen that the temperature difference increases with reduced heat exchange area. The temperature difference doubles from 2 K with fully covered U-tubes to about 4 K with only half of the U-tubes area available for heat transfer.

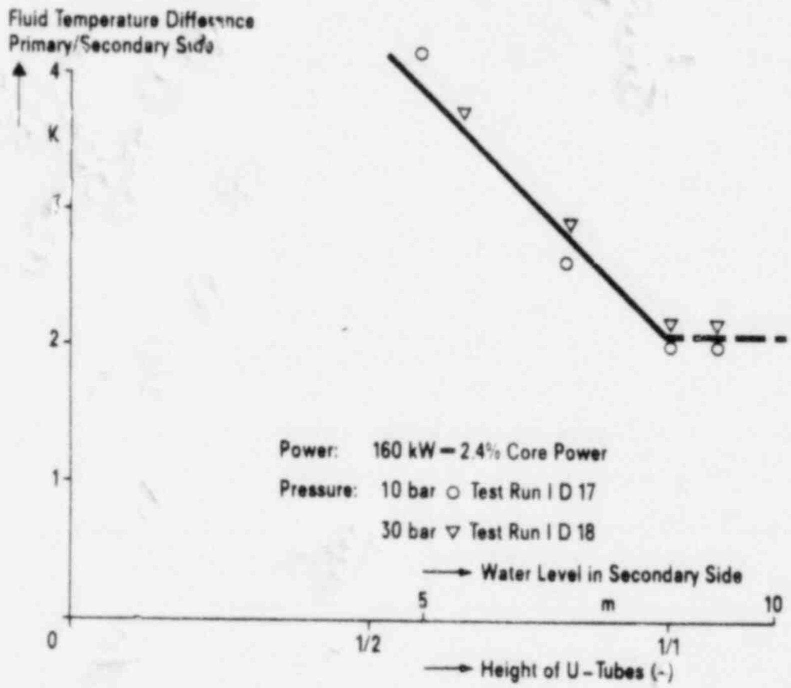


Fig. 8: Reflux condenser mode, influence of secondary side water level

The influence of the presence of non-condensible gases in the steam generators during the reflux condenser mode was also examined. During this test (fig. 9) nitrogen was injected in steps into the steam generator to determine its effect on the heat transfer.

The maximum injected mass of 0,35 kg corresponds to that amount of gas that would come from all of the nitrogen dissolved in the accumulator water or in the water of the flooding tank and 100 % of the fission gas in the plena of all fuel rods. With this high amount of non-condensible gas at a system pressure of only 10 bars, the temperature difference has increased to about 8 K. This value is significantly lower than the temperature difference measured for single phase natural circulation (16 K).

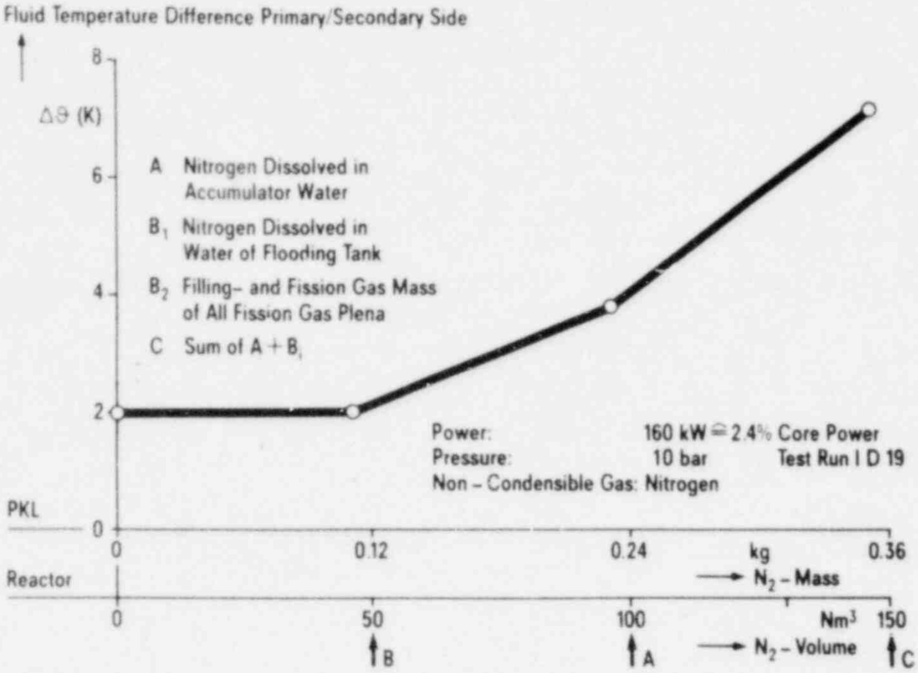


Fig. 9: Reflux condenser mode, influence of non-condensible gas (N_2)

4.2 Transients

To provide information about the time history of small break LOCAs, transient tests were carried out to

- investigate the different phases of the accident including the transition from one energy transport mode to another.
- produce a data base for code assessment.

The test matrix for these transients is shown in fig. 10.

All these tests were started with a water-filled system at a pressure of 30 bars. The transients were initiated by

- opening the break
- starting the ECC injection
- cooling down the secondary side by 100 K/h.

Number of Runs	Break Size	Break Position	Injection Location	Remarks	Run No
6	Small Break	k	K		ID 2
		k	H		ID 3
		h	K		ID 12
		h	H		ID 13
		p	K		ID 8
		p	H		ID 9
5	Medium Break	k	K		ID 5
		k	H	No Feed	ID 6
		k	K	Water to 1 SG	ID 7
		h	K		ID 10
		h	H		ID 11

Initial Conditions: 30 bar Subcooled Single Phase Flow

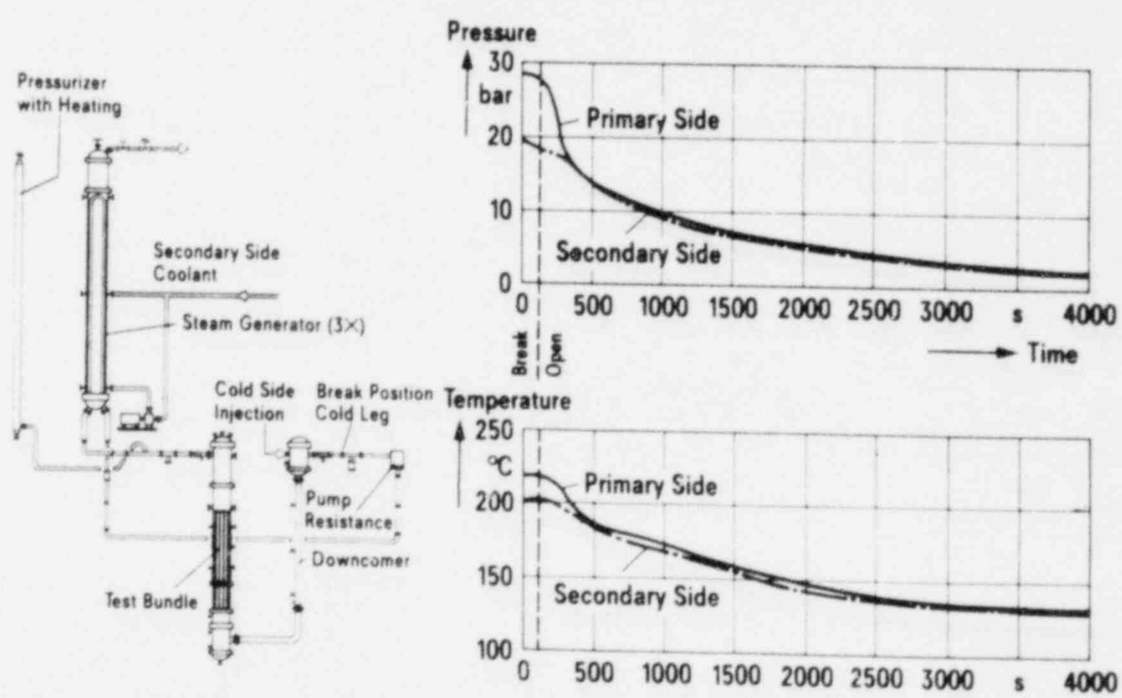
Boundary Conditions: 400 kW Bundle Power
100 K/h Cool Down of Secondary Side
1(2) kg/s Injection Rate

Break Position: k – Cold Leg
h – Hot Leg
p – Pressurizer

Injection Location: K – Cold Side
H – Hot Side

Fig. 10: Transient Tests with Small and Medium Size Breaks

An example of a transient test is shown in figures 11 and 12. Although the water inventory is reduced considerably (fig. 12), it can be seen from fig. 11, that the primary side closely follows the cooldown of the secondary side. This demonstrates the effectivness of the steam generator as a heat sink and confirms the results of the steady state tests.



Pressure and Temperature Transients at Primary and Secondary Side

Fig. 11: Test ID 5 with Cold Leg Break and Cold Side Injection

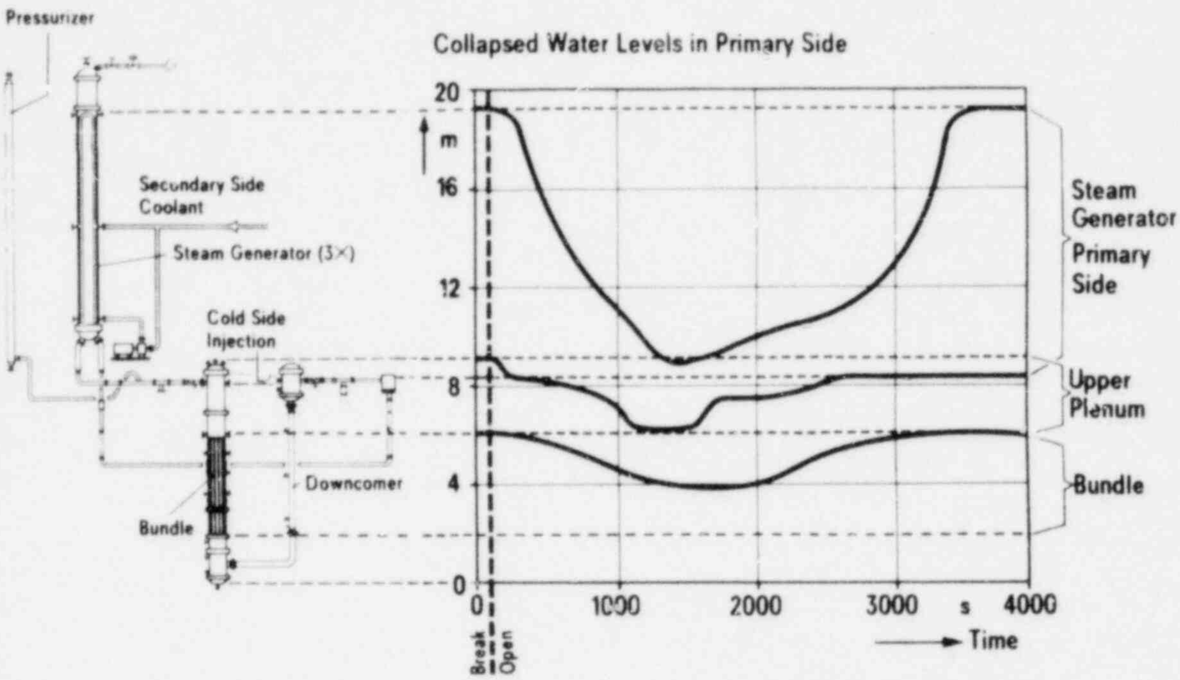


Fig. 12: Test ID 5 with Cold Leg Break and Cold Side Injection

5. Conclusions

The control of small break LOCAs in KWU-PWRs is assured by automatic measures in the short term transient and by manual measures in the long term transient. In addition to the design of the safety injection pumps, a significant feature is the immediate cooldown of the secondary side to obtain an additional heat sink. Energy transport from the core in single and two phase flow under typical transient conditions was verified by tests at KWU in the PKL-test facility.

These tests proved that the decay heat from the core can be transferred to the heat sink by natural circulation or even better by heat transfer in the reflux condenser mode: natural circulation is not a requirement for the energy transport from the core to the steam generators.

The conclusions drawn from steady state and transient tests confirm that the removal of energy from the core is assured as long as water mixture covers the core and heat can be transferred via the secondary side of the steam generators.

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/3/ D. Hein. H. Watzinger
Status of Experimental Verification of ECCS Efficiency
ENS/ANS Topical Meeting on Nuclear Power Reactor Safety, Oct. 1978, Brussels

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Errata for Paper
" A SYNOPSIS OF PKL SMALL BREAK TESTS"
D. Hein, F. Watzinger

Test-Nr.	Bundle Power kW	System Pressure bar	SG Water Level m	Non-Condensible Gas (N_2) kg
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ID 17	160	10	9*/7/5	-
ID 18	160	30	9*/7/5	-
ID 19	160	10	9*	0,12/0,24/0,36

*U-Tubes Completely Covered

Fig. 6 : Single loop tests, reflux condenser mode

To assure that at higher power (decay heat more than 2 %) no limitation occurs in the energy transport with the reflux condenser mode, tests were run up to a maximum power equivalent to 8 % decay heat. The results are plotted in figure 7 . Higher power results in an almost linear increase of the temperature difference. It is noted that the curve for a 10 bar system pressure is parallel to the 30 bar pressure curve. The higher steam density at the higher pressure results in a better heat transfer.

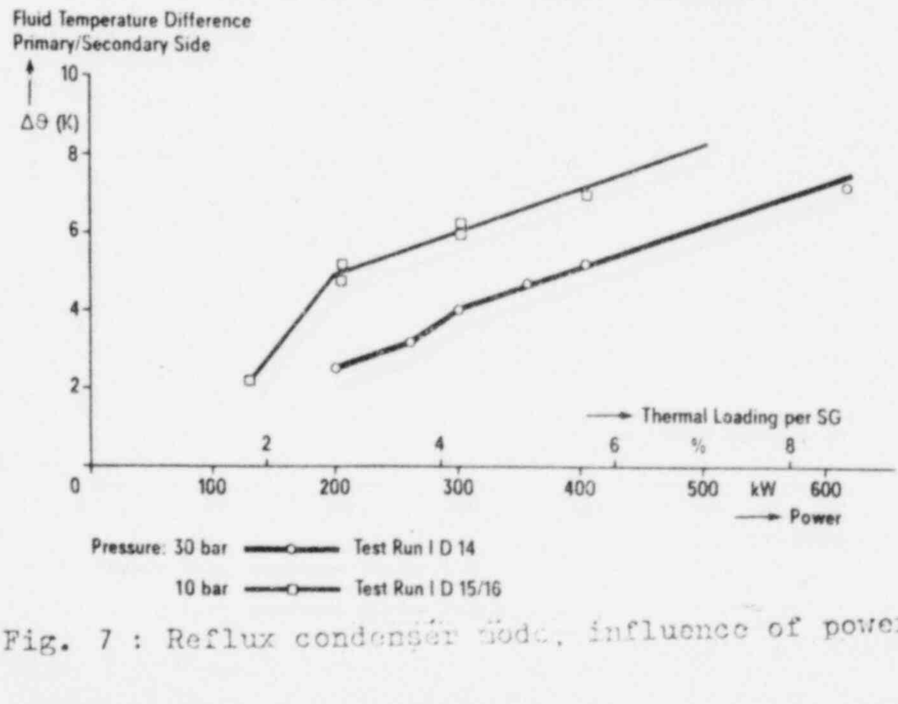


Fig. 7 : Reflux condenser mode, influence of power

The German 2D/3D UPTF Program

by

E.F. Hicken and K.R. Hofmann

presented at

VIII. WRSRIM, October 27 to 31, 198o

The German 2D/3D UPTF Program

1. Objectives

The overall objective of the trilateral 2D/3D program between Japan, United States and Germany is the coordinated analytical and experimental investigation of the ECC behavior in a PWR during the refill and reflood phase under large break LOCA conditions. The 2D/3D program will conclude the large break LOCA research by investigating the two and three dimensional thermohydraulic behavior in the reactor vessel and its impact on core cooling in large scale experiments. Appropriate computer codes including TRAC, K-FIX, COBRA and T-FIX are developed and assessed.

JAERI is conducting tests in the integral Cylindrical Core Test Facility (CCTF)and will investigate the core behavior during refill and reflood in the full scale Slab Core Test Facility (SCTF) while the USNRC is providing advanced instrumentation, code development and assessment, design calculations and test analyses.

The German contribution to the program includes design, construction and operation of the Upper Plenum Test Facility (UPTF) to determine the three-dimensional thermohydraulic behavior in the upper plenum and downcomer during the last part of blowdown, refill and reflood phases using an external steam supply for simulation of the core flow into the upper plenum.

.I.?C recently proposed also to perform separate effects tests in UPTF for investigation of phenomena occuring during small break LOCA.

2. UPTF Test Facility

The test facility represents the pressure vessel of a German 13oo MW PWR including the upper plenum, upper plenum internals and the downcomer in real dimensions. The core is replaced by a steam water mixing device to simulate the two phase core flow into the upper plenum during the refill and reflood phases of the transient. Three intact loops and one broken loop with opening valves to simulate the break flow are attached to the pressure vessel. The steam generator behavior in the PWR is simulated by steam separators. The primary coolant pumps are represented as flow resistances. The broken loop contains two steam separators to allow both hot and cold leg break simulation. A pressure suppression system with additional steam injection capability is available to control the break flow back-pressure. ECCS flow for cold leg and combined hot and cold leg injection is provided by 4 accumulators simulating also the low pressure injection systems. The steam for the UPTF operation is provided by a conventional power plant.

3. Operating Procedure

After heat up of the system with steam saturated water will be stored in the steam generator simulators and the lower plenum.
At about 10 bars the valves in the broken loop legs will be opened and ECCS injection from the accumulators starts. Flashing of the stored water in the steam generator simulators and lower plenum provides the necessary conditions for the end of blowdown phase. The refill phase is marked by steam back flow from the containment simulator which is made possible by supplementary steam injection.
During the reflood phase large quantities of steam and water are injected through the core simulator. Oscillations of the downcomer water column and the core can be achieved by a controlled injection mode up to 0.4 Hz. The test will be completed after the core region is flooded from the bottom or top.

4. Instrumentation

Advanced two phase flow instrumentation provided by USNRC will be used to investigate the thermohydraulic phenomena specifically at the core/upper plenum interface, in the upper plenum, the downcomer and in the 4 primary loops.

5. Test plans

According to a preliminary time schedule for the UPTF construction, testing will start in mid 1985. At least 40 tests are planned including separate effects tests and integral tests with cold leg, combined hot and cold leg and upper plenum injection mode. Other US-type reactor-systems, which will be tested additionally are B&W and CE ECCS injection, for which provisions are made in the UPTF.

Coupling of some UPTF and SCTF tests will be done by iteratively matching the flow conditions at the core/upper plenum interface.

THE GERMAN 2D/3D UPTF PROGRAM

BY

E. F. HICKEN AND K. R. HOFMANN

(8. WRSRIM , OCT. 27 - 31 , 1980)

OBJECTIVE OF THE 2D/3D PROGRAM

COORDINATED ANALYTICAL AND EXPERIMENTAL STUDY OF THE THERMOHYDRAULIC BEHAVIOR OF EMERGENCY CORE COOLANT
DURING
THE REFILL AND REFLOOD PHASE OF A LOSS-OF-COOLANT ACCIDENT IN A PRESSURIZED WATER REACTOR

BMFT : UPTF
JAERI : CCTF AND SCTF
USNRC: ANALYSIS (TRAC) AND INSTRUMENTATION

Fig.1

OBJECTIVES OF THE 2D/3D PROGRAM (CONT.)

- EFFECT OF VARIOUS ECC INJECTIONS ON STEAM BINDING FOR LARGE BREAK LOCA
- FLOW HYDRODYNAMICS IN PWR VESSEL DURING REFILL AND REFLOOD FOR LARGE BREAK LOCA
- FLOW AND TEMPERATURE DISTRIBUTION IN A HEATED CORE

PROPOSED :

- EVENTS LEADING TO CORE UNCOVERY AND/OR RECOVERY FOR SMALL BREAK LOCA

Fig.2

OBJECTIVE OF UPTF

DETERMINE THE THREE - DIMENSIONAL THERMAL - HYDRAULIC BEHAVIOR OF THE FLUID IN THE UPPER PLENUM AND DOWNCOMER DURING THE REFILL AND REFLOOD PHASES

USING AN EXTERNAL STEAM SUPPLY FOR SIMULATION OF THE REACTOR CORE

Fig.3

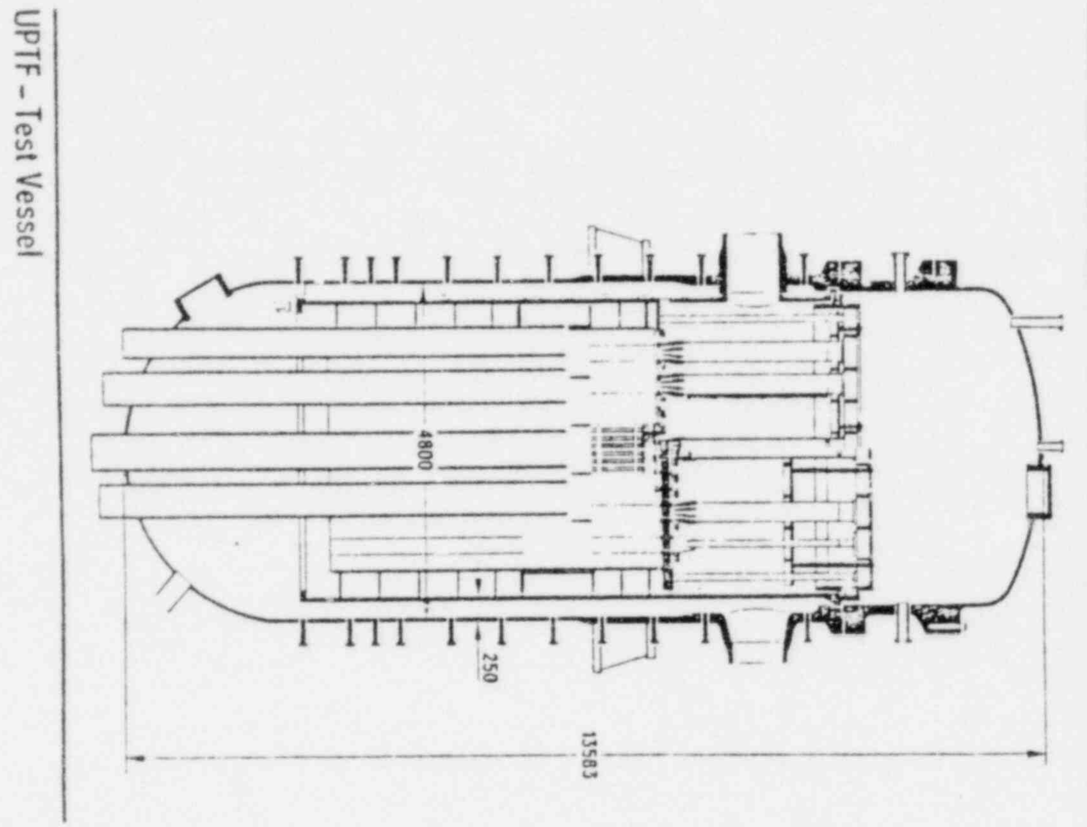
- **Simulation of the Reactor**
  - Upper Plenum
  - Upper Core Support Plate
  - Upper End Box
  - Shortened Fuel Element Dummies
  - Downcomer
  - Intact Loop
  - Broken Loop with S. G Simulation

- **The Effect of the Core on Processes**
  - in the Remainer of the Primary System
  - is Simulated by a Steam Water Injection System

- **Upper Plenum Geometries to be Invastigated**
  - KWU, $\overline{W}$, Japan
  - B & W Vent Valves
  - CE - Alignment Plate

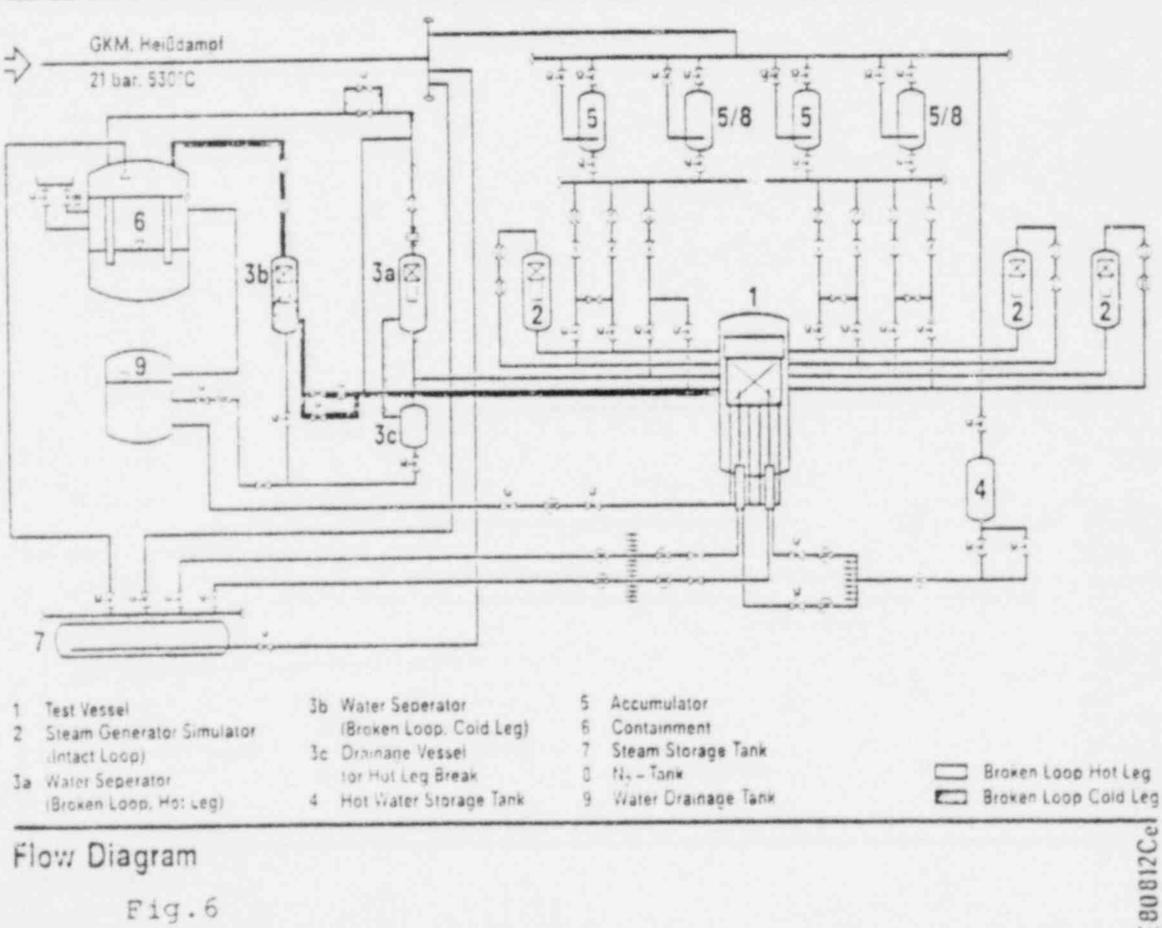
- **ECCS:**
  - Cold Leg Injection
  - Combined Injection

Fig.4



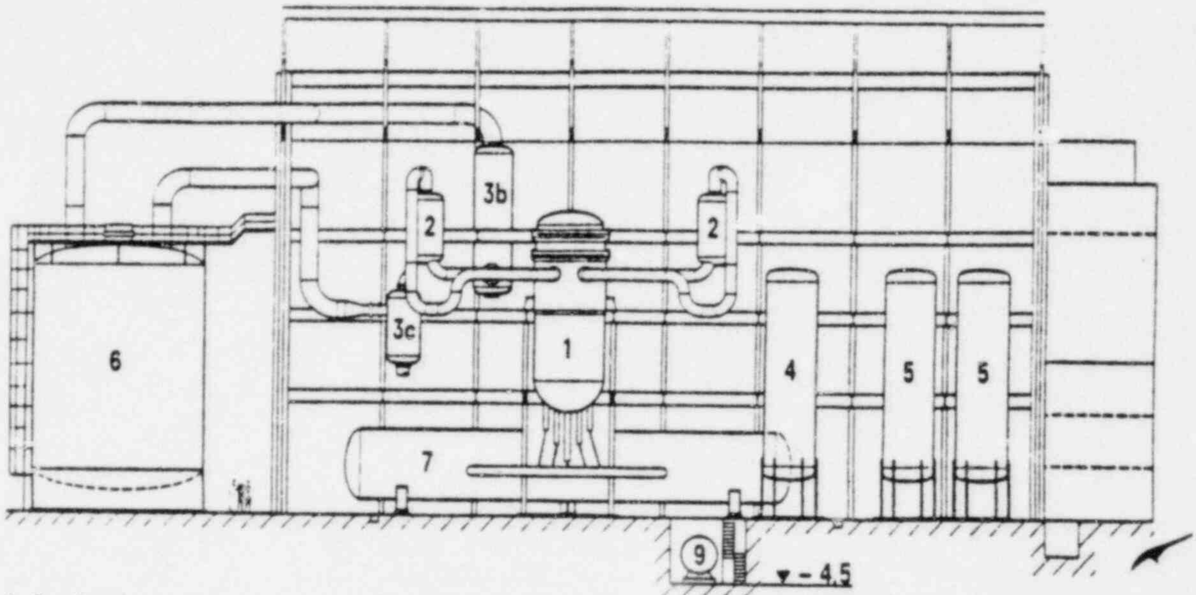
UPTF - Test Vessel
4800
250
13583

Fig.5



GKM. Heißdampf
21 bar. 530°C
1
2
3a
3b
3c
4
5
5/8
6
7
9
1 Test Vessel
2 Steam Generator Simulator (Intact Loop)
3a Water Seperator (Broken Loop, Hot Leg)
3b Water Seperator (Broken Loop, Cold Leg)
3c Drainage Vessel for Hot Leg Break
4 Hot Water Storage Tank
5 Accumulator
6 Containment
7 Steam Storage Tank
8 N_2 - Tank
9 Water Drainage Tank
Broken Loop Hot Leg
Broken Loop Cold Leg
Flow Diagram
E80812Ce

Fig.6

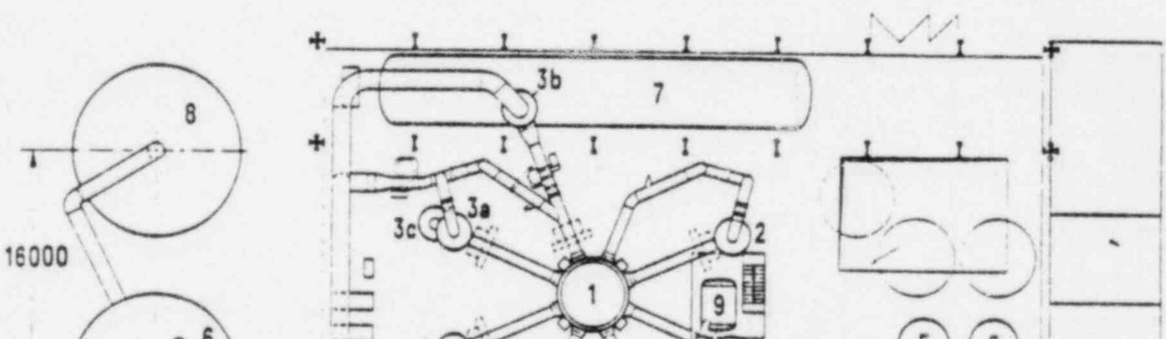


1 Test Vessel
2 Steam Generator Simulator. (intakt Loop)
3b Water Seperator (Broken Loop, Cold Leg)
3c Drainage Vessel for Hot Leg Break
4 Hot Water Storage Tank
5 Accumulators
6 Containment
7 Steam Storage Tank
9 Drain Storage Tank

UPTF Layout
Front View

Fig.7

E 79 1215 e a



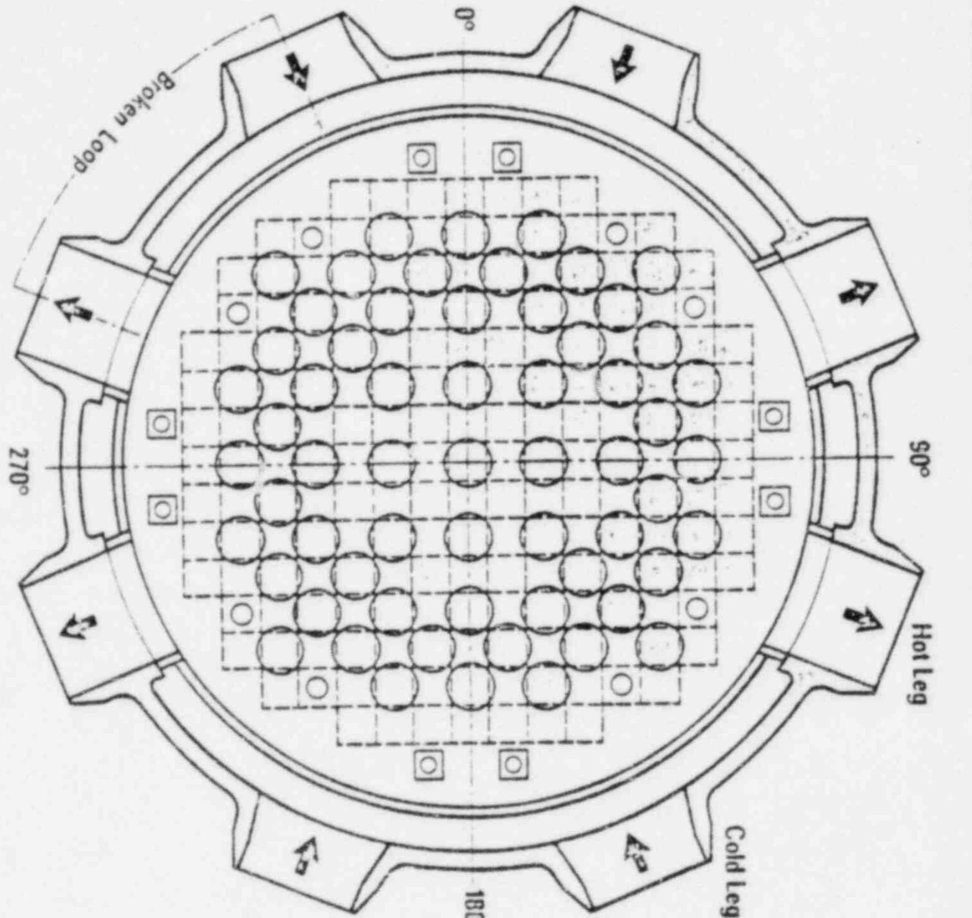
1 Test Vessel
2 Steam Generator Simulator. (Intakt Loop)
3a Water Seperator (Broken Loop, Hot Leg)
3b Water Seperator (Broken Loop, Cold Leg)
3c Drainage Vessel for Hot Leg Break
4 Hot Water Storage Tank
5 Accumulators
6 Containment
7 Steam Storage Tank
8 Demineralized Water Tank
9 Drain Storage Tank

UPTF Layout
Top View

Fig.8

E 79 1216 e a

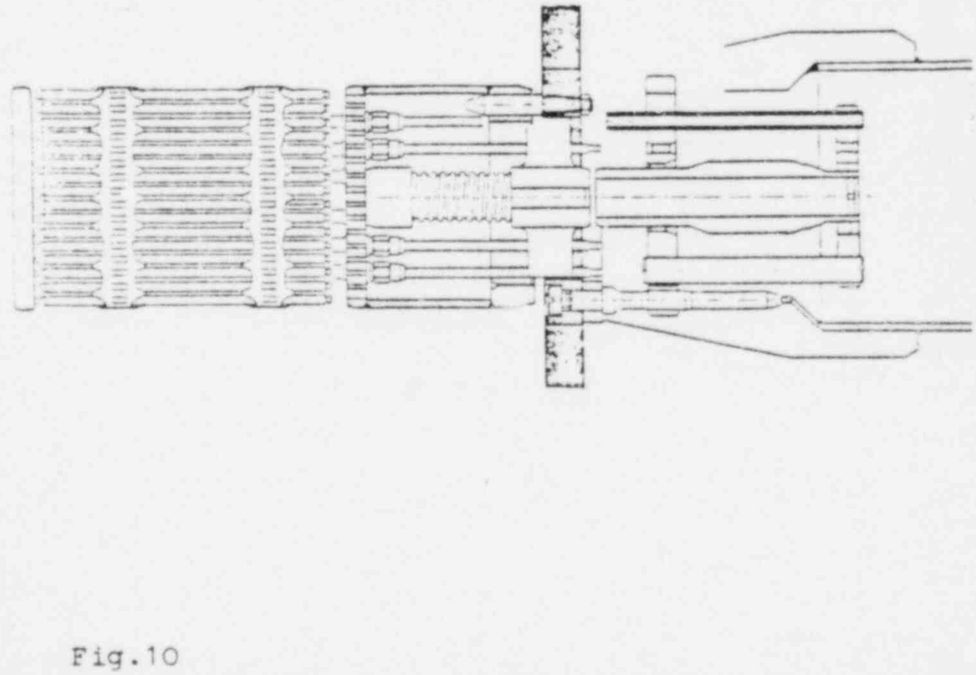
Upper Plenum (UPTF)
Cross Section



E 79 1213 e a

Fig.9

End Box (UPTF)
Fuel Element with CRA Spider



E 79 1214 Ce a

Fig.10

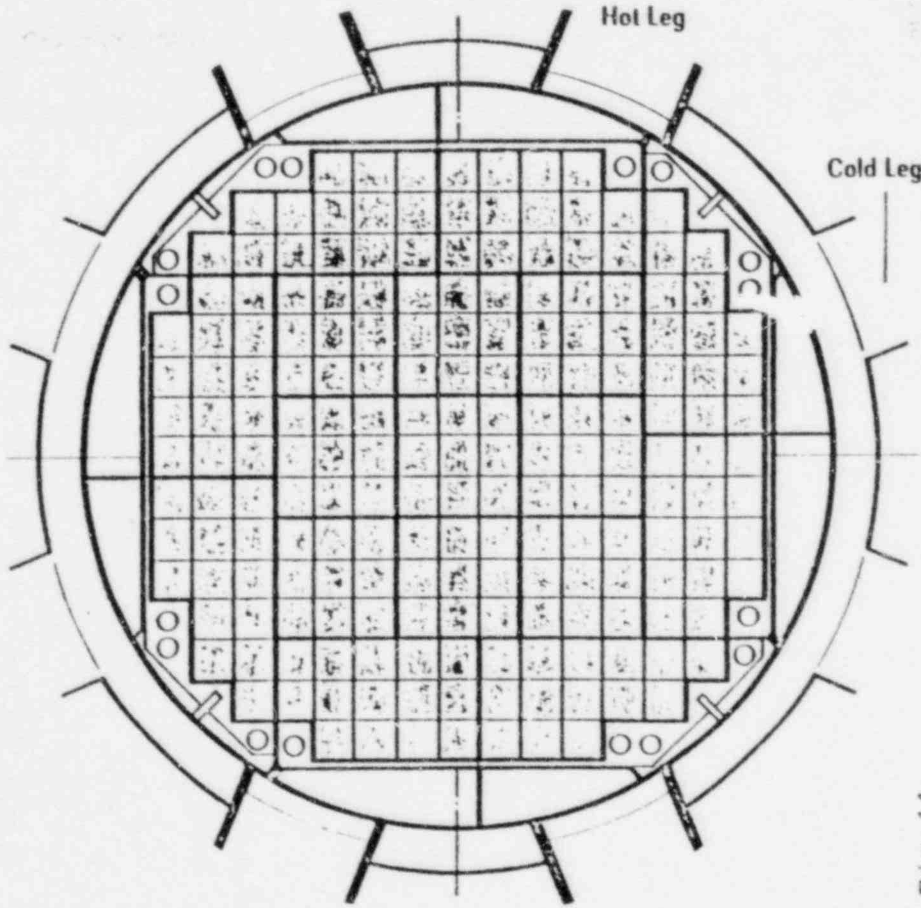


Fig. 11

Reactor Core
Cross Section

E 79 1213 e

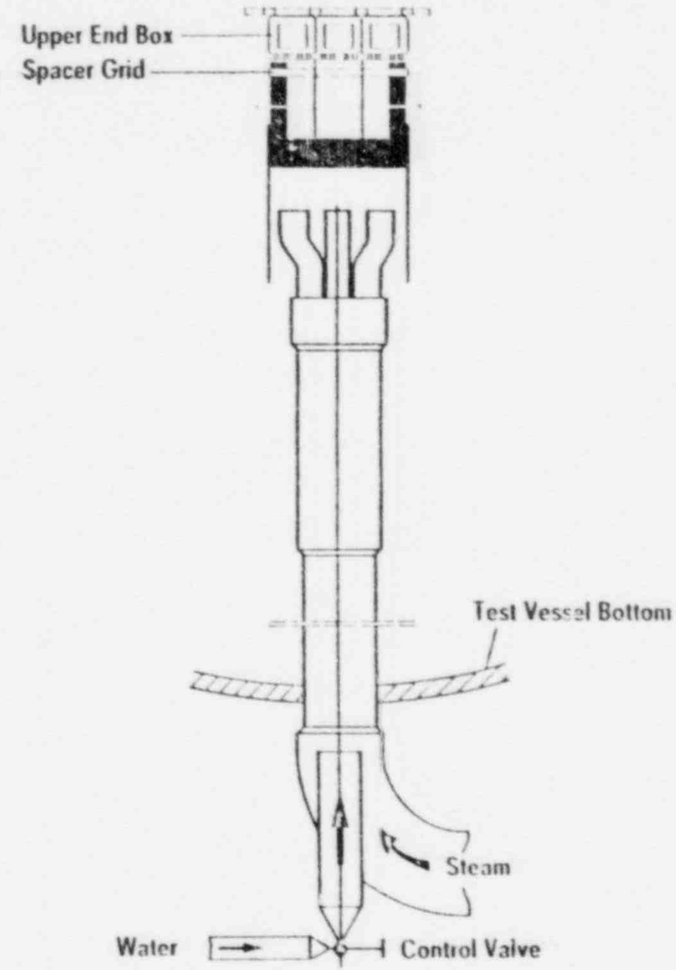
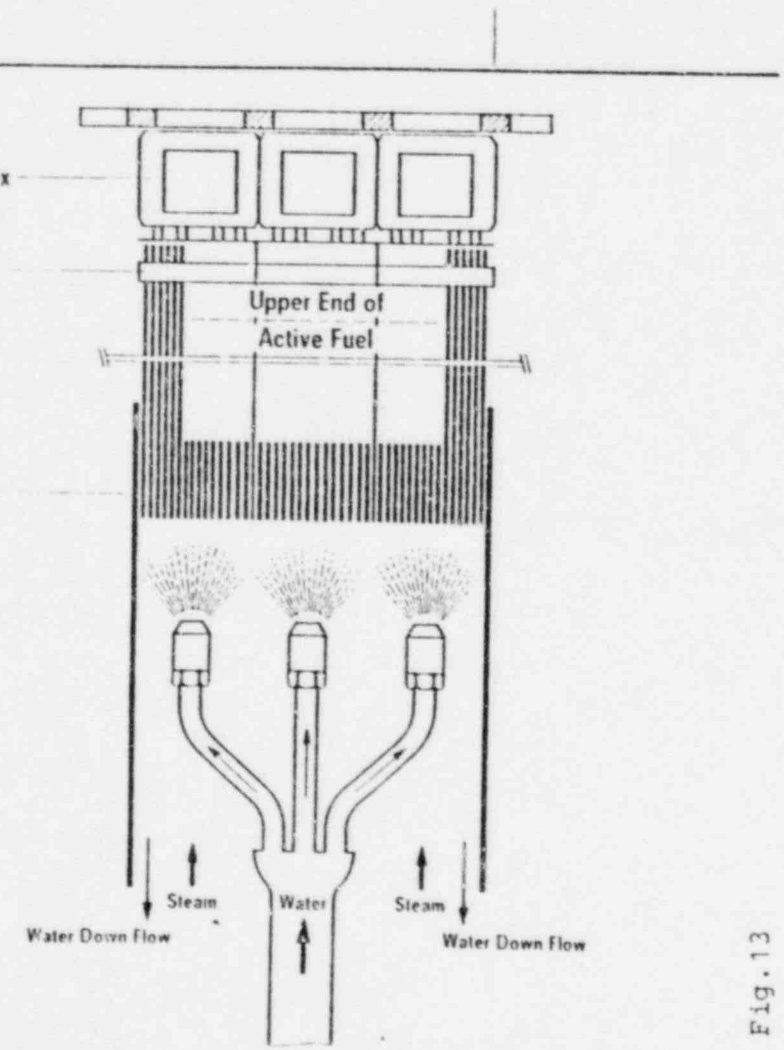


Fig. 12

Core Injection Zone (UPTF)
External Mixing Concept

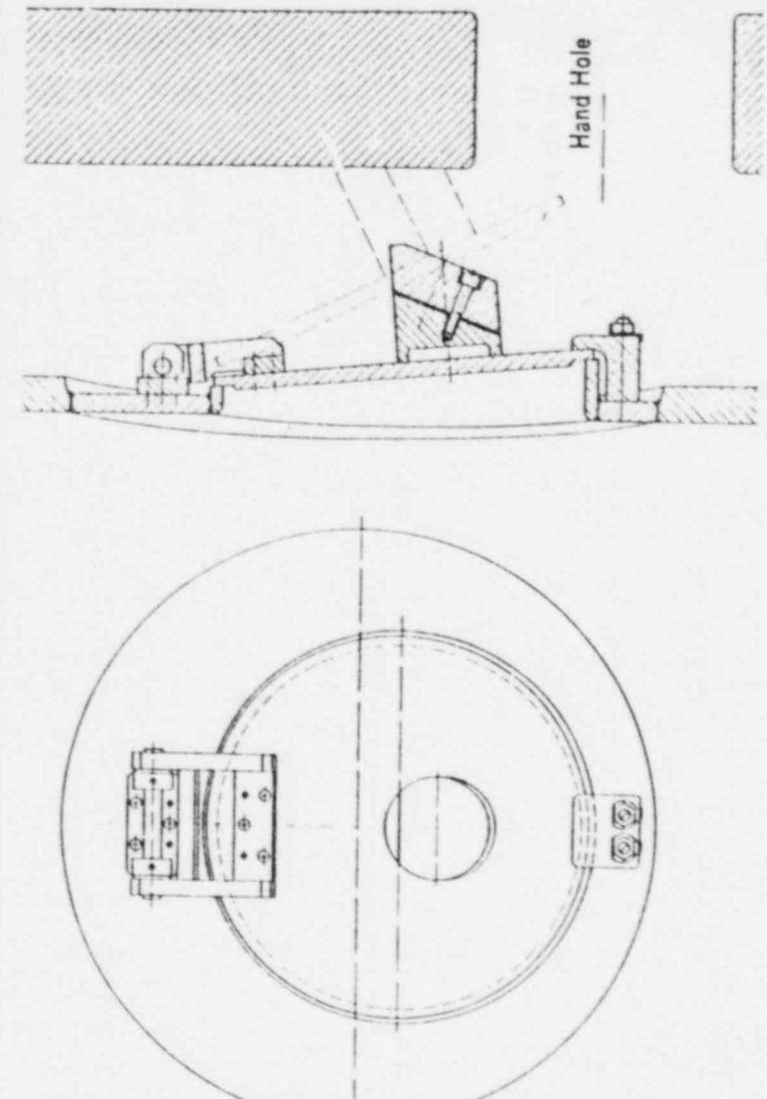
E 80 217 e



Core Injection Zone (UPTF)
Nozzle Concept

Fig.13

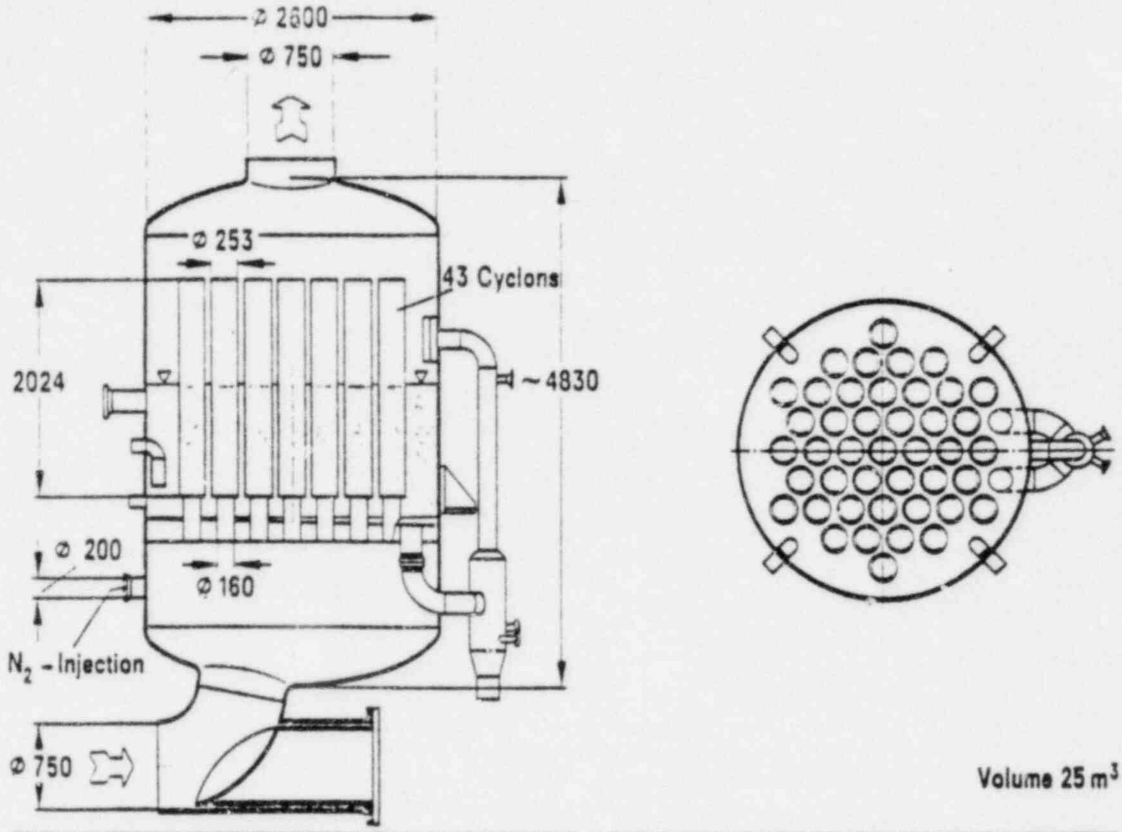
E 80 218 e



Vent Valve (UPTF)

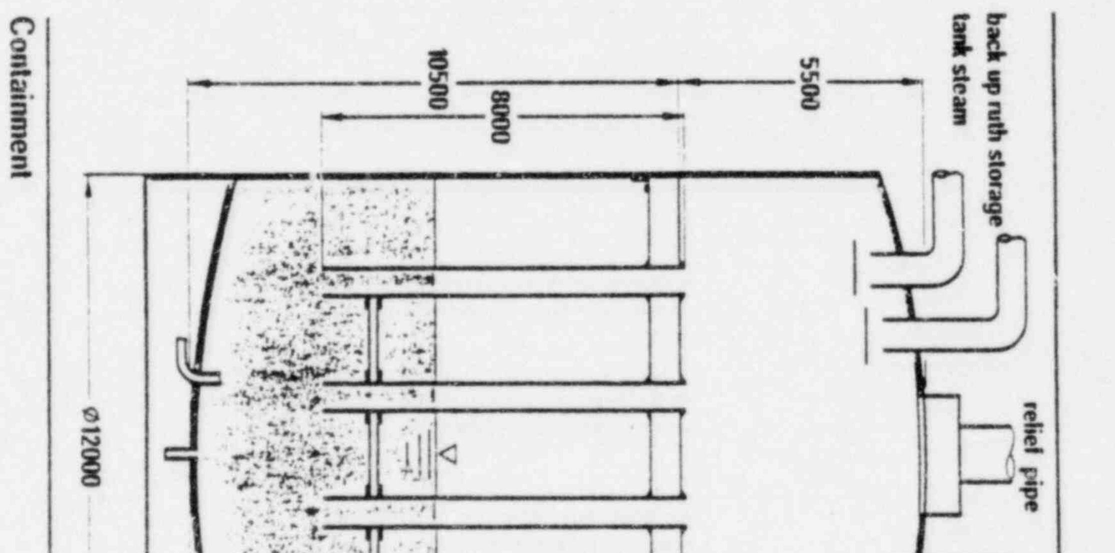
Fig.14

E791232e



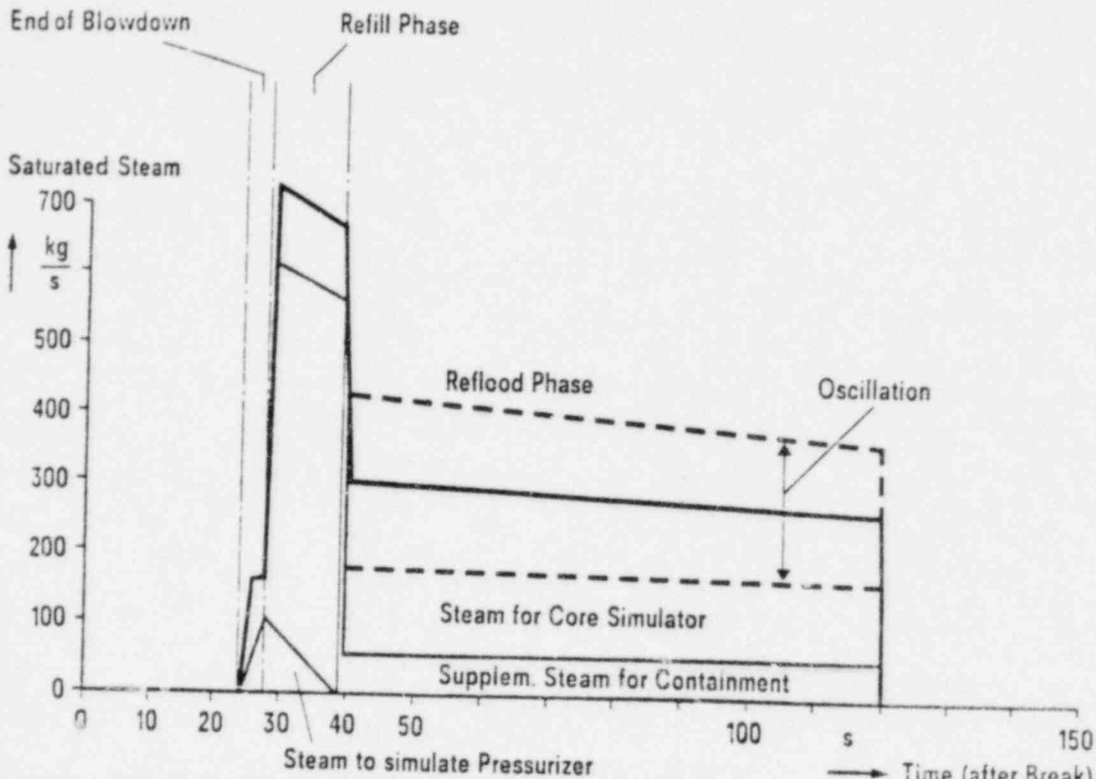
Steam Generator Simulator (Water Separator) – UPTF
Broken Loop, Hot Leg
Fig.15

E 791230e



E 791222 e

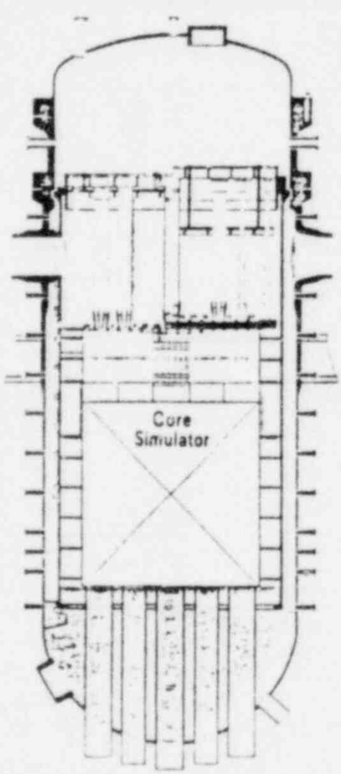
Fig.16



UPTF – Steam Demand for Double Ended Breaks
(Cold Leg, Hot Leg)

Fig.17

E 79 1219 el



Instrumentation in Test Vessel	Upper Plenum	Core–Upper Plenum Interface	Core	Lower Plenum	Down-comer
Fluid Temperature	x	x		x	x
Fluid Temperature below End Box		x			
Superheat Probe below End Box		x			
Wall Temperature					x
Pressure	x				
DP to Containment	x				x
DP Upper Plenum – Downcomer	x				x
DP Vent Valve	x				x
DP axial		x			x
DP horizontal					x
DP collapsed Level	x			x	
LLD	x		x	x	
FDG	x	x			x
Film Probes	x				
Impedence Probes (Band Probes)	x				
Video Probes	x				
Turbines vertical	x				x
Turbines horizontal	x	x			
Turbines Vent Valves					x
Flow Modules		x			
Conductivity Probe				x	

Instrumentation in Test Vessel

Fig.18

E 80 230 Ce

PRELIMINARY UPTF TEST MATRIX

	MINIMUM NUMBER OF TESTS
SEPARATE EFFECTS TESTS	
- DOWNCOMER	6
- UPPER PLENUM	4
- SG TUBE BREAK	2
INTEGRAL TESTS	
- COMBINED INJECTION	12
- COLD LEG INJECTION (INCL. B&W, CE)	12
- ALTERNATE ECCS	6
	40

Fig.19

UPTF SCHEDULE (PRELIMINARY)

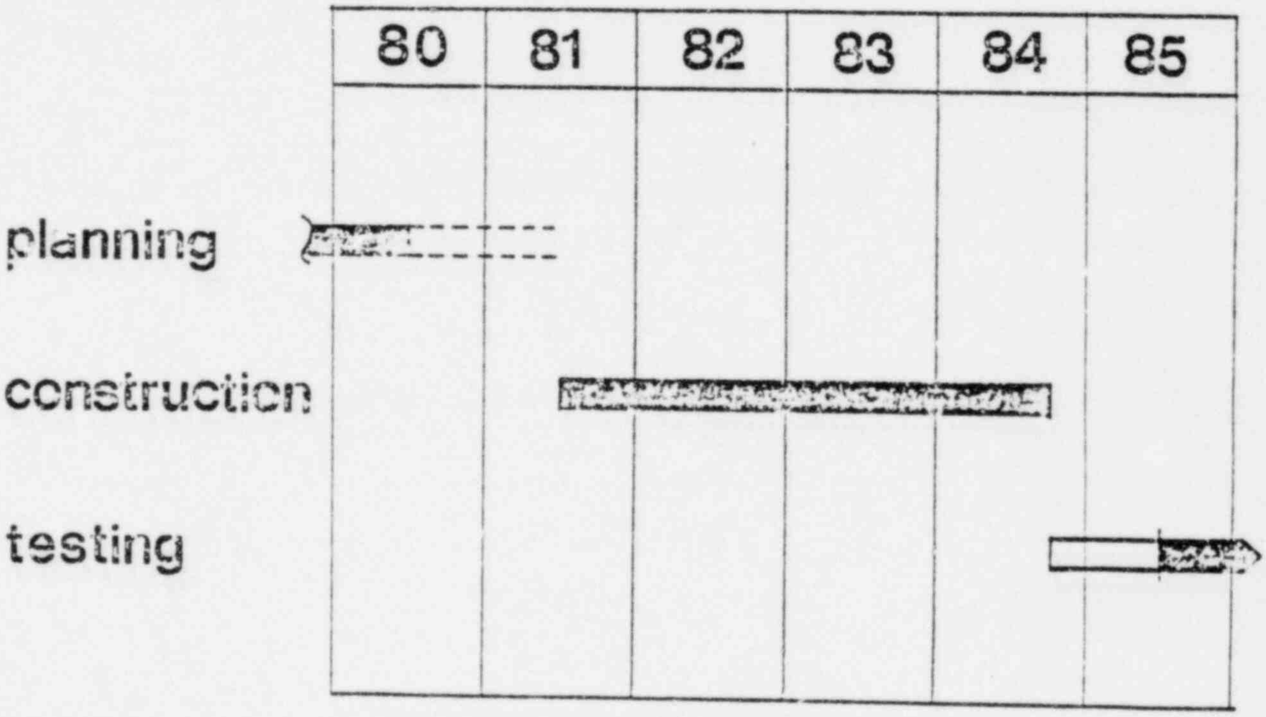


Fig.20

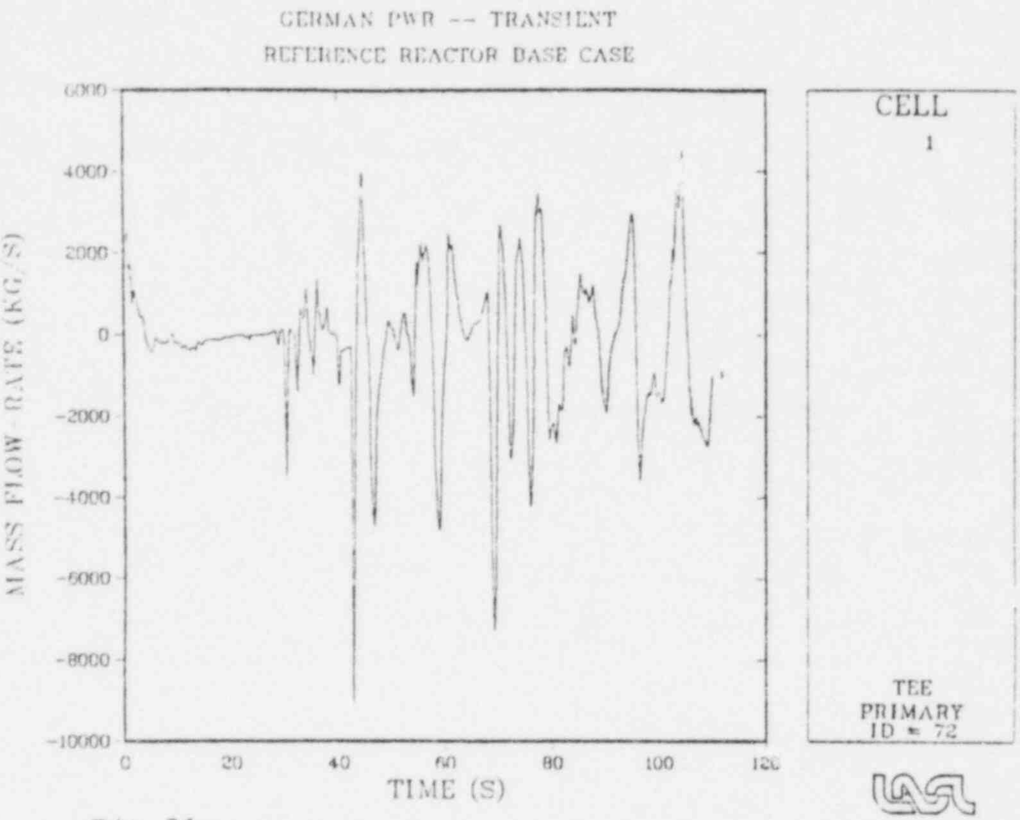

GERMAN PWR -- TRANSIENT
REFERENCE REACTOR BASE CASE
6000
4000
2000
0
-2000
-4000
-6000
-8000
-10000
MASS FLOW RATE (KG/S)
0
20
40
60
80
100
120
TIME (S)
CELL
1
TEE
PRIMARY
ID = 72

Fig.21

RESULTS OF CCTF CORE I TESTS [1)]

Yoshio Murao
Kenmei Hirano
Masao Nozawa

Japan Atomic Energy Research Institute
At the Eighth Water Reactor Safety Research
Information Meeting

October 29, 1980

More than twenty refill and reflood tests were conducted at the JAERI's Cylindrical Core Test Facility (CCTF) since March, 1979. The thermo-hydraulic and system behaviors observed in these tests are discussed in this report.

The objectives of CCTF are:

(1) Demonstration of effectiveness of ECCS in PWR during refill and reflood phases of LOCA.

(2) Provision of information for analytical modeling of thermo-hydrodynamic phenomena of refill and reflood phases in PWR LOCA.

(3) Verification of reflood analysis code "REFLA"[(1),(2)] and US-developed three-dimensional code "TRAC".

REFLA code system is under development at JAERI and a one-dimensional reflood analysis code, REFLA-1D, is currently operational[(2)]. In order to evaluate reflood phenomena in reactors, it is important to establish a sufficient data base for realistically modeling the refill and reflood phenomena that can be extrapolated to real reactors.

The CCTF with its integral simulation of a PWR system and extensive measurement systems has already yielded a large body of information on the overall thermo-hydraulic behavior and system effects during refill and reflood phases of PWR LOCA. The features of the test facility and instrumentation are described in the following slides. The pressure vessels of CCTF and PWR are compared in Fig.1. The flow diagram of CCTF is illustrated in Fig.2. Injection condition of ECC water for base case test is shown in Fig.3. Test conditions of the CCTF Core I are listed in Table 1.

General trends

In previous report[(2),(3)], the following results were described:

(1) The core and the downcomer thermo-hydrodynamic behaviors were nearly one-dimensional.

(2) Each intact loop had nearly same flow characteristics and the parallel channel oscillation did not occur in loops.

---

1) This work was performed under the contract between the Sciece and Technology Agency of Japan and JAERI to demonstrate the effectiveness of ECCS during reflood period of a hypothetical LOCA of a PWR.

(3) Some amount of entrained water from the core was accumulated on the upper support plate and almost all the water entering hot legs evaporated in the steam generators.

(4) The steam from steam generators entered cold legs and was condensed by the subcooled emergency core cooling water without significant thermo-hydrodynamic coupling.

(5) Quenching in the core was nearly one-dimensional and one-directional except at the top of the center bundles.

Some of these phenomena will be shown in a movie later on. Most of these phenomena were observed in other tests such as the parameter effect tests. The phenomena observed in CCTF were similar to the model assumed in the safety evaluation model.

Parameter effects on clad surface temperature

The effects of ECC flow rate, system pressure and initial clad temperature on the temperature transients of peak-powered rod are plotted on Figs.4 to 6. These tendencies of parameter effects are easy to estimate from small scale reflood experiments. For example, the temperature rise, which is defined as the difference between the peak clad temperature and the initial clad temperature, and the quench time of midplane in CCTF tests and PWR-FLECHT low flooding tests are compared in Fig.7. The flooding velocity in these CCTF tests were about 2 cm/sec during LPCI injection.

System effect

In order to examine the system effects, the downcomer head and the flow resistance across a broken loop and intact loops are indicated in Figs.8 and 9, respectively. For demonstration of the cold downcomer and the lower plenum injection effects, the downcomer head of FLECHT-SET coupling test (2714B equivalent) is plotted on Fig.8. The slow water accumulation in downcomer is caused by the "downcomer bypass" and the low saturated head is caused by the "hot wall effect". The difference in the flow resistance across a broken loop and intact loops is manifested as the pressure drop at the broken cold leg nozzle. The acceleration of water by the steam induces this pressure drop. The pressure drop increases with the steam velocity. The mass balance relation of quasi-steady state can be written as shown in Fig.10. It is found that the flooding rate increases with the pressure drop at the broken cold leg nozzle and does not change significantly with the steam generation rate in the core. The flooding rates of tests with initial clad temperature variation are plotted against time on Fig.11. Though the steam generation rate in the core is thought to increase with initial clad temperature, the flooding rate of each test is found to be nearly same at about 2 cm/sec during LPCI period. The measured mass balance is indicated in Fig.12.

The system effects can be summerized in Table 2.

Requirement for system model

The information necessary for evaluation of the flooding rate is as follows:

(1) Downcomer bypass phenomenon.

(2) Reduction of downcomer effective head due to hot wall effect.

(3) Pressure drop at broken cold leg nozzle.

(4) De-entrainment rate in upper plenum.

Acknowledgement

The authors are much indebted to Mr. H. Akimoto, Mr. T. Ohkubo, Mr. J. Sugimoto, and Mr. T. Sudo for planning and carrying out the CCTF tests.

References

(1) Murao, Y. : Analytical study of thermo-hydrodynamic behavior of reflood-phase during LOCA, J. Nucl. Sci. Technol., 16[11] pp.802 ~ 817, (1979).

(2) Murao, Y. et al. : Experimental and analytical modeling of the reflood-phase during PWR-LOCA, ASME HTD-Vol.8, Experimental and analytical modeling of LWR safety experiments, p.23 ~ 29, (1980).

(3) Nozawa, M. : Japanese safety research programs - ROSA, CCTF, and SCTF, Seventh water reactor safety research information meeting, (1979).

FEATURE OF TEST FACILITY

- 1/21 SIZE OF 1100 MWe CLASS PWR IN VOLUMETRIC SCALING

- FULL LENGTH IN VERTICAL DIMENSION
  (AVERAGED LENGTH OF HEAT TRANSFER TUBE OF S.G. IS 5.3 M SHORTER THAN ACTUAL LENGTH)

- SYSTEM WITH 4 LOOP PIPINGS AND COMPONENTS
  (ONE BROKEN AND THREE INTACT LOOPS)

- ANNULAR DOWNCOMER WITH CONTROL DEVICE OF WALL TEMPERATURE
  (VOLUME OF CORE BYPASS REGION IS INCLUDED IN VOLUME OF DOWNCOMER) , GAP OF ANNULUS : 61.5 MM

- 8 x 8 RODS BUNDLE WITH 7 NON-HEATED RODS AND 57 RODS OF THREE DIFFERENT POWER LEVEL

- 32 BUNDLES ARRANGED THREE POWER ZONE

- 8/15 SCALED UPPER PLENUM STRUCTURE

FEATURE OF INSTRUMENTATION

- 900 T/CS ON CORE CLADDING SURFACE

- 101 T/CS IN CORE (WALL,FLUID,STEAM TEMPERATURE MEASUREMENT)

- TOTAL MEASURING CHANNELS 1600 CH.

- 48 VIEW WINDOWS WITH TV CAMERA, 16 MM MOVIE AND 35 MM STILL CAMERA

- US-PROVIDED SPOOL PIECE SYSTEM AND LIQUID LEVEL DETECTOR SYSTEM

- COLOR IMAGE DISPLAY SYSTEM

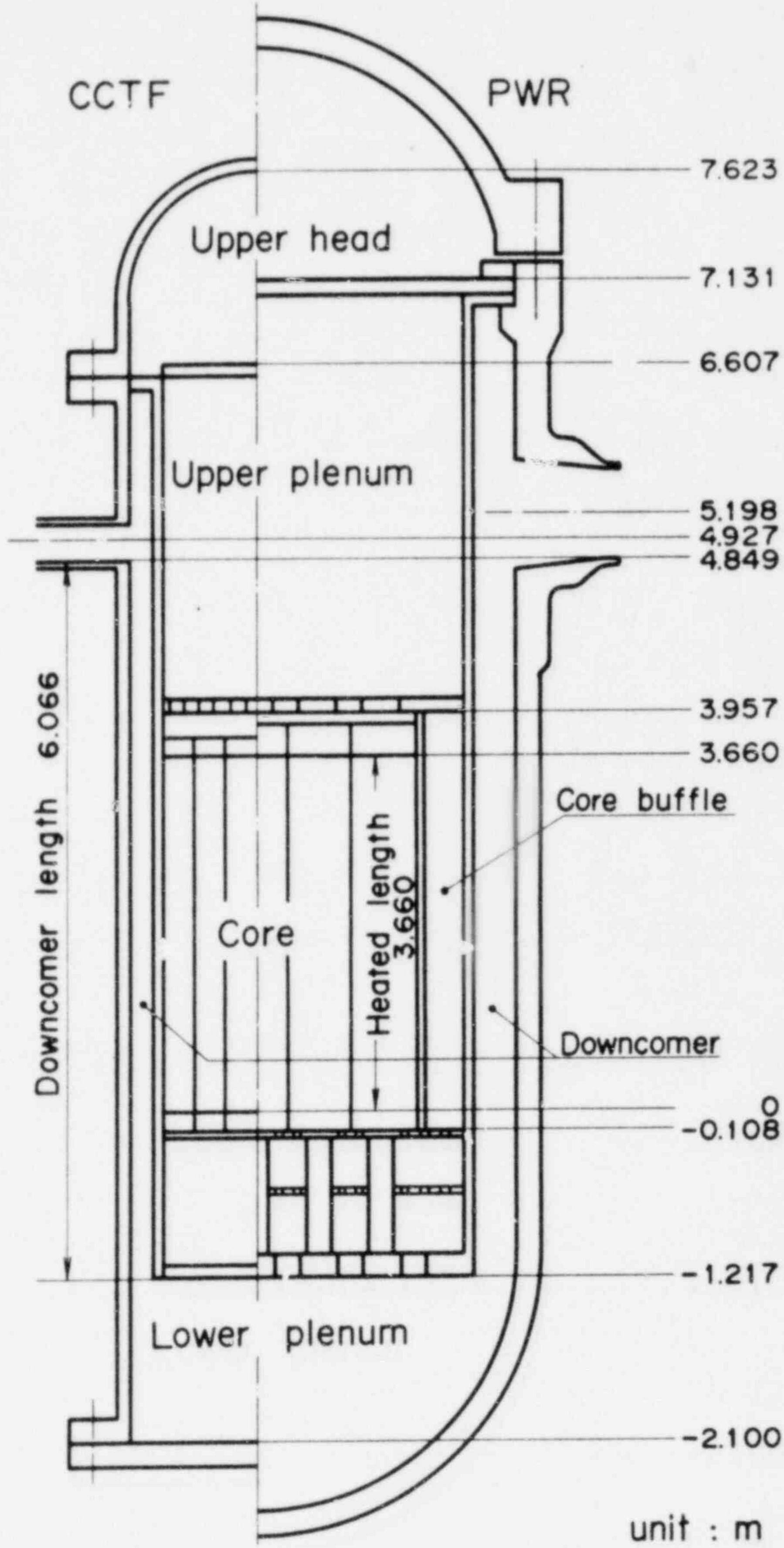


Fig. 1 Comparison of pressure vessel of CCTF and PWR

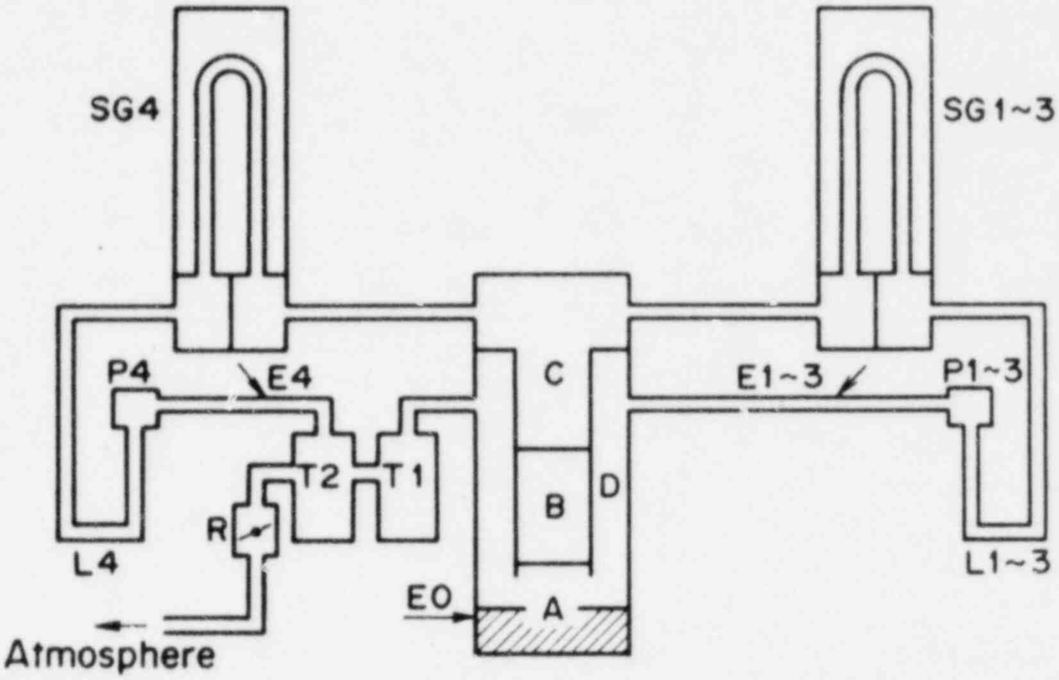


Fig. 2 Flow diagram of Cylindrical Core Test Facility (CCTF)

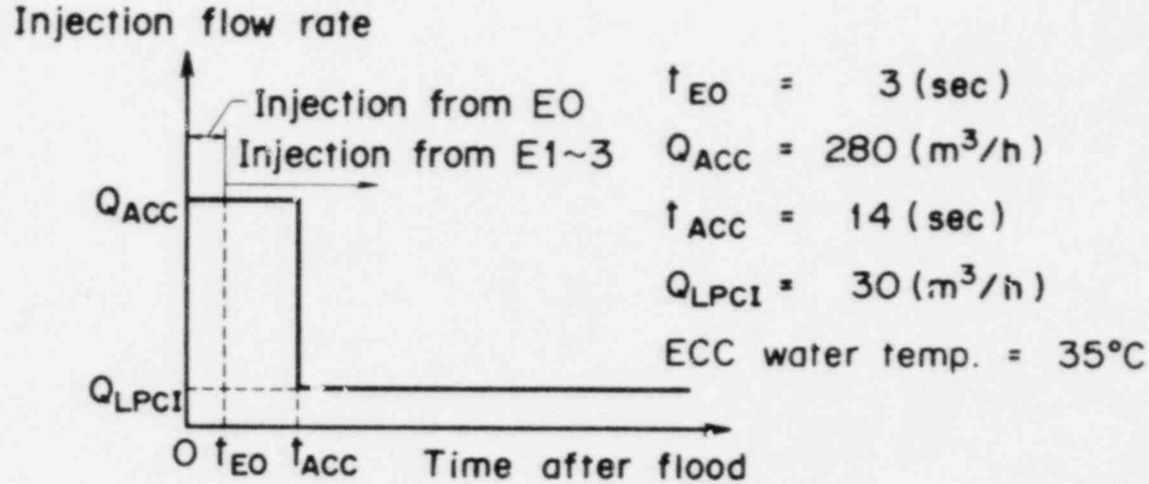


Fig. 3 Injection condition of ECC water
(Base case)

TABLE 1 CCTF CORE I TEST CONDITIONS
(PARAMETER EFFECT TEST)

LINEAR POWER (CORE AVERAGE)	1.4 KW/M
RADIAL POWER PROFILE	1.15, 1.10, 0.89
AXIAL PEAKING FACTOR	1.49
LOCAL PEAKING FACTOR	1.1
STRUCTURE TEMPERATURE	T_{SAT}
DOWNCOMER WALL TEMPERATURE	$T_{SAT} + \sim 80^{o}C$
S.G. SECONDARY SIDE WATER TEMP.	$265^{o}C$
INITIAL PEAK CLAD TEMP.	<u>600,</u> 700, $800^{o}C$
SYSTEM PRESSURE	1.5, <u>2.0,</u> 3.0, 4.2 KG/CM^2

ECCS INJECTION CONDITION	$Q_{ACC}(M^3/H)$	$T_{ACC}(SEC)$	$Q_{LPCI}(M^3/H)$
BASE CASE	280	14	30
LOW ACC	240	14	30
SHORT ACC	280	11	30
HIGH LPCI	280	14	60
LOW LPCI	280	14	15 ~ 20

ECC WATER TEMPERATURE $35^{o}C$

______ BASE CASE CONDITION

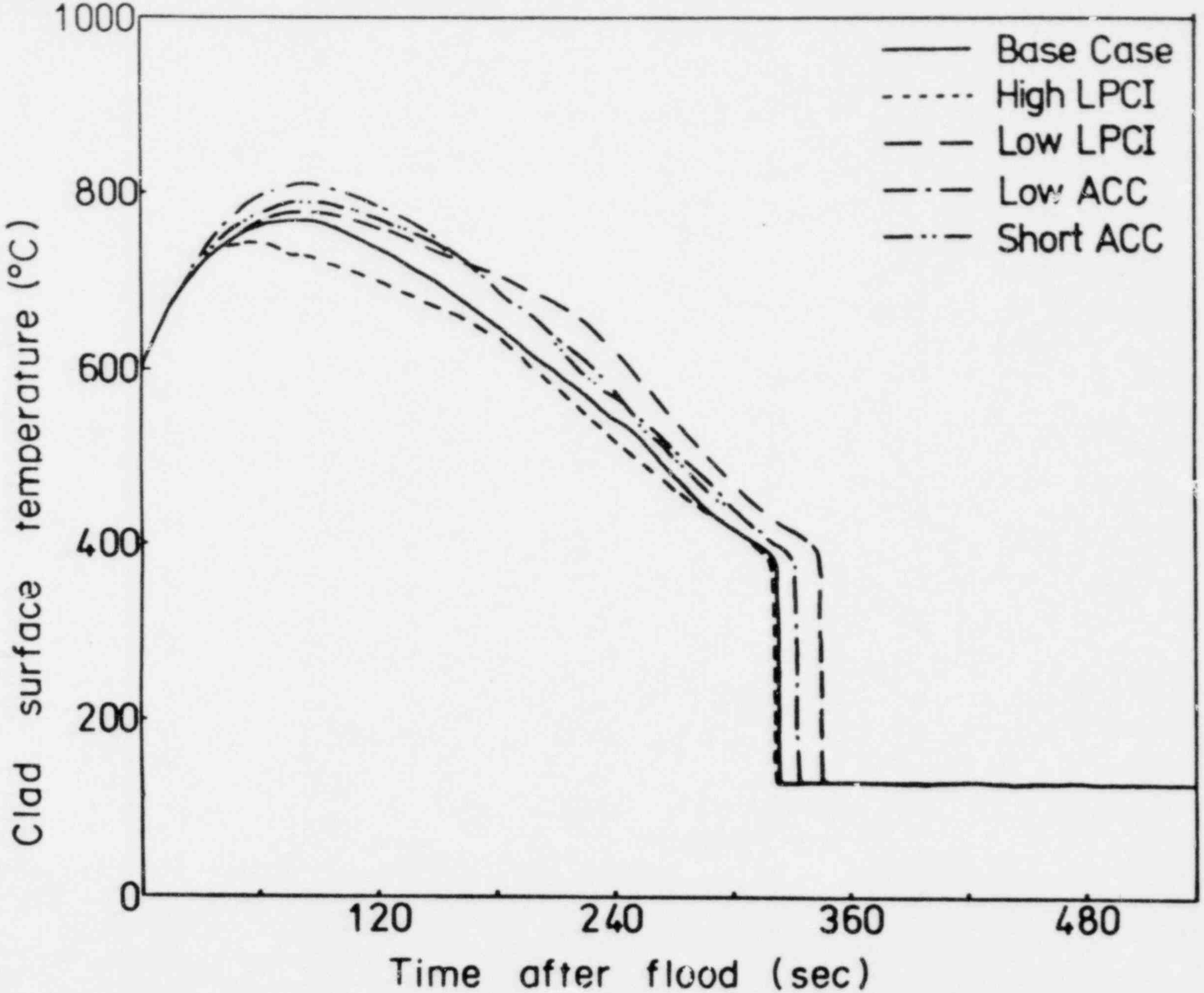


Fig. 4 Midplane temperature of peak-powered rod

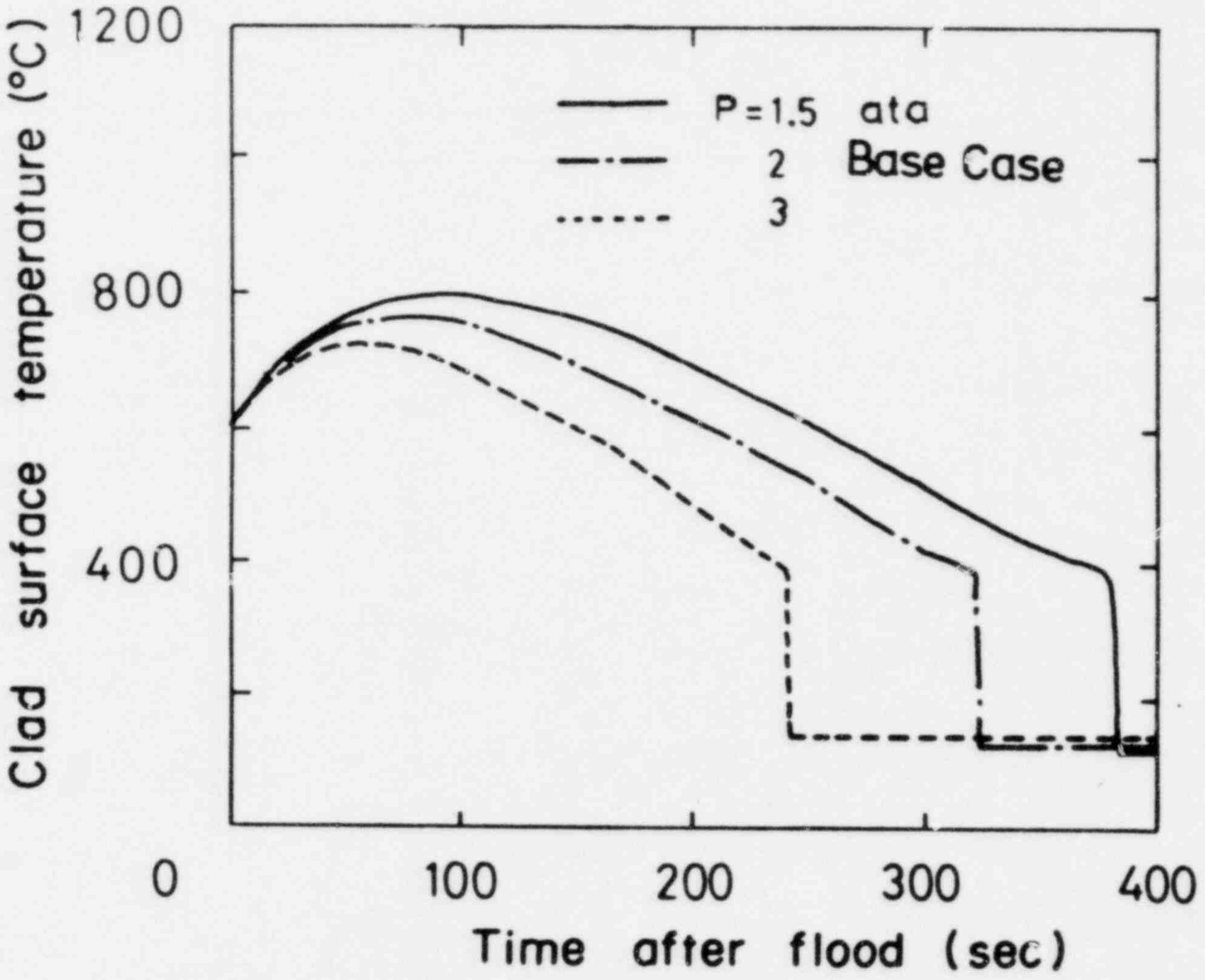


Fig. 5 Midplane temperature of peak-powered rod

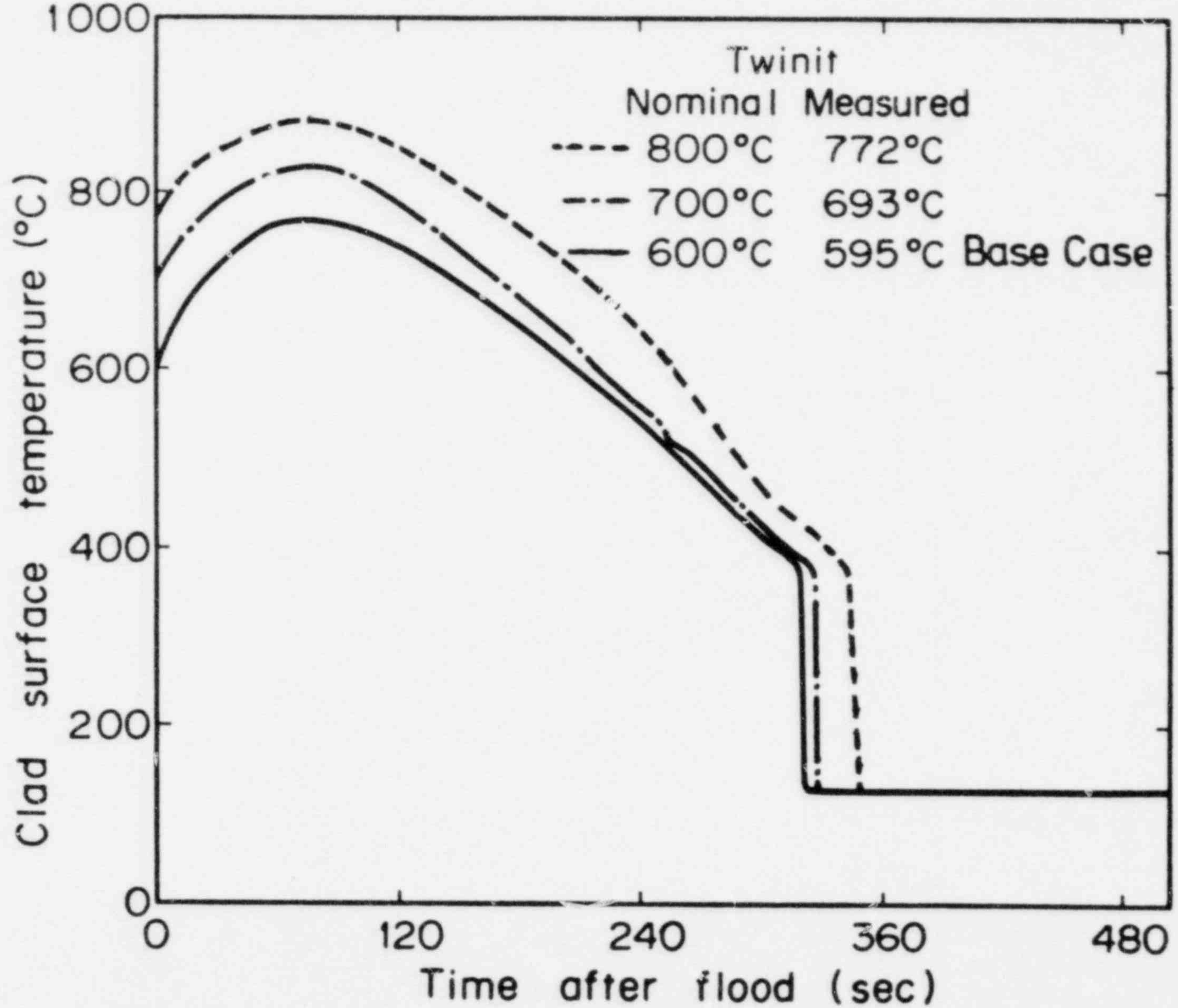


Fig. 6 Midplane temperature of peak-powered rod
(Effect of initial clad temperature)

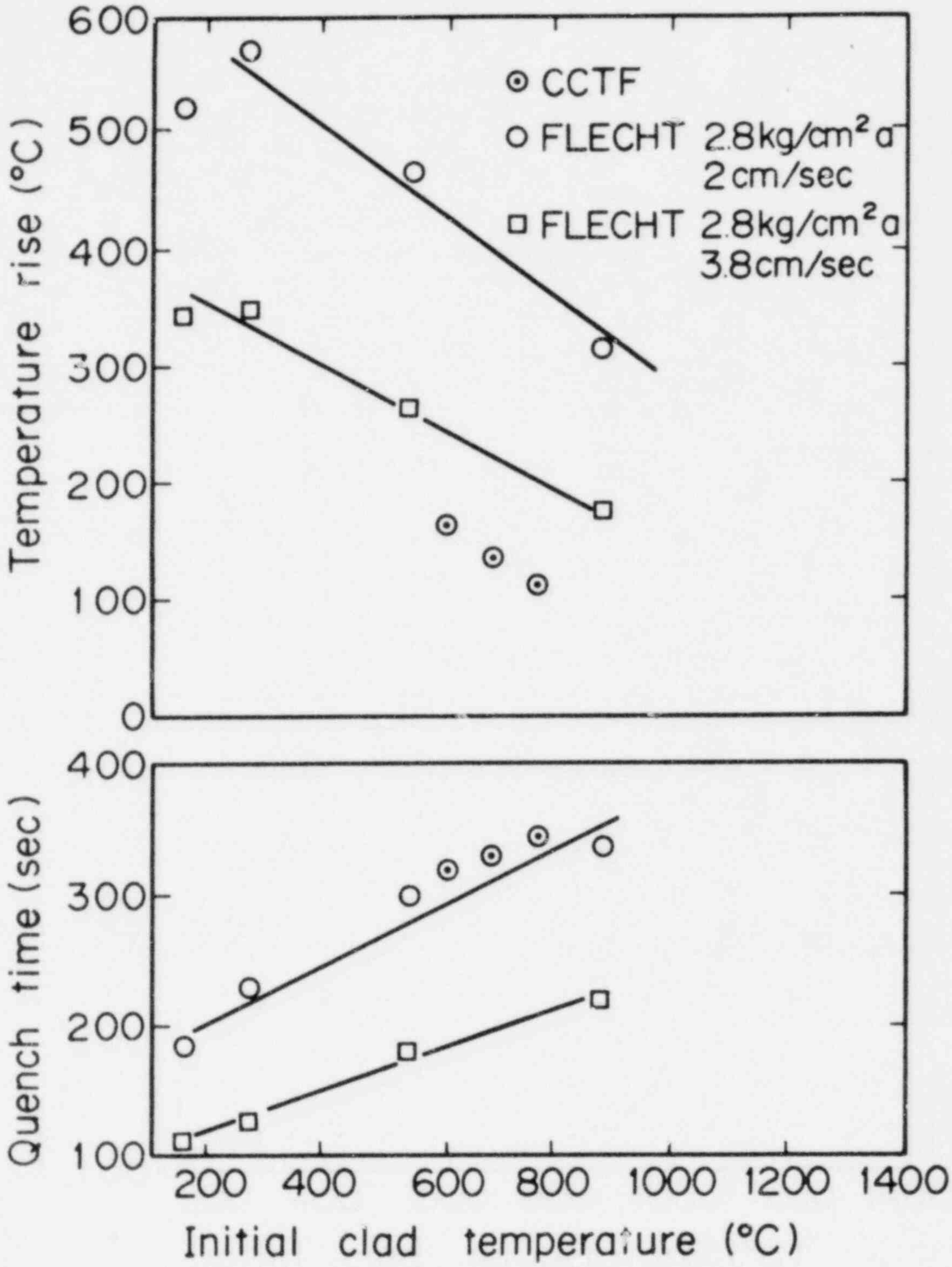


Fig. 7 Comparison of temperature rises and quench times versus initial clad temperature in CCTF tests and FLECHT

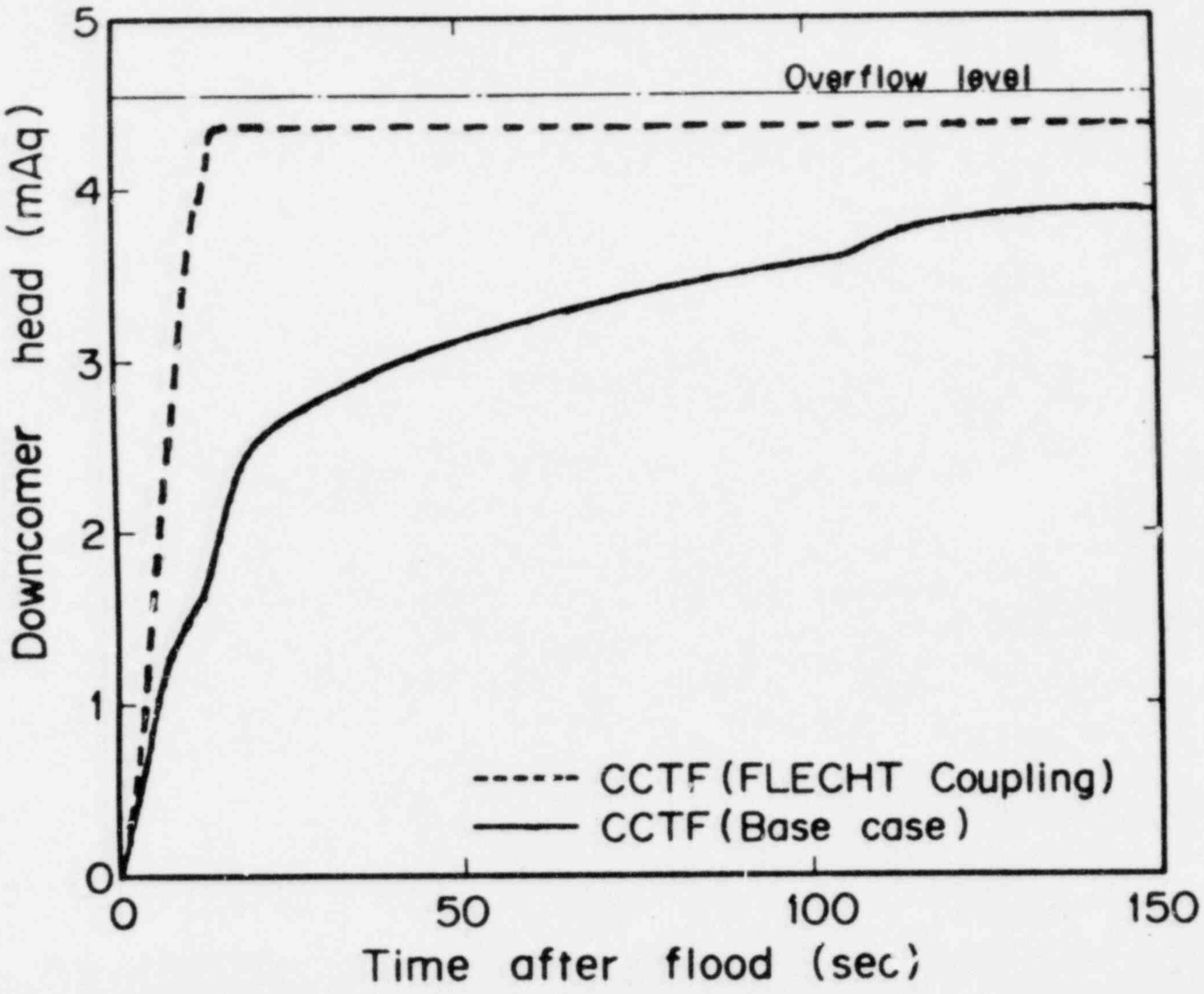


FIG. 8 DOWNCOMER HEAD IN CCTF TESTS OF BASE CASE TEST AND FLECHT-SET COUPLING TEST

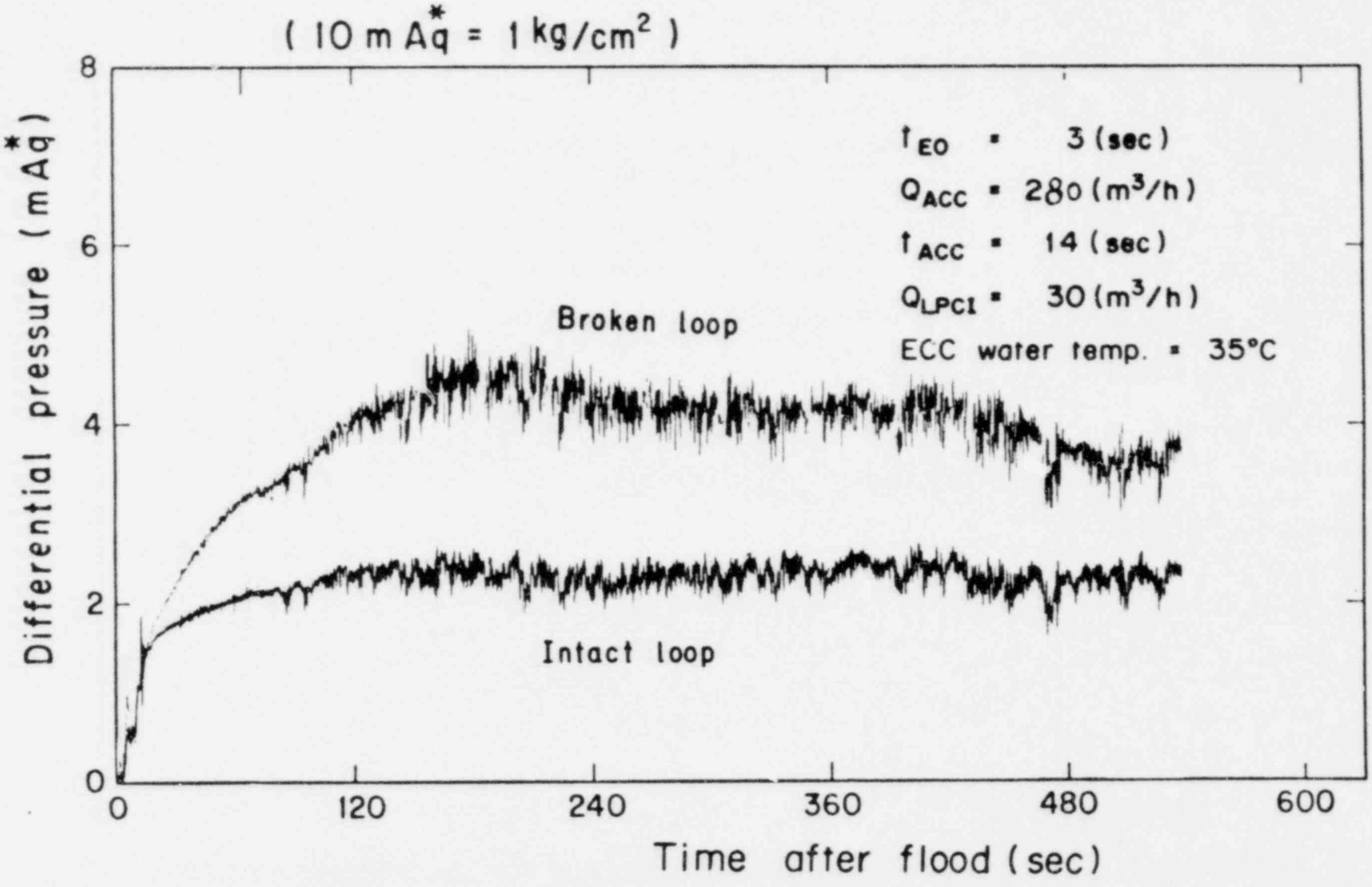
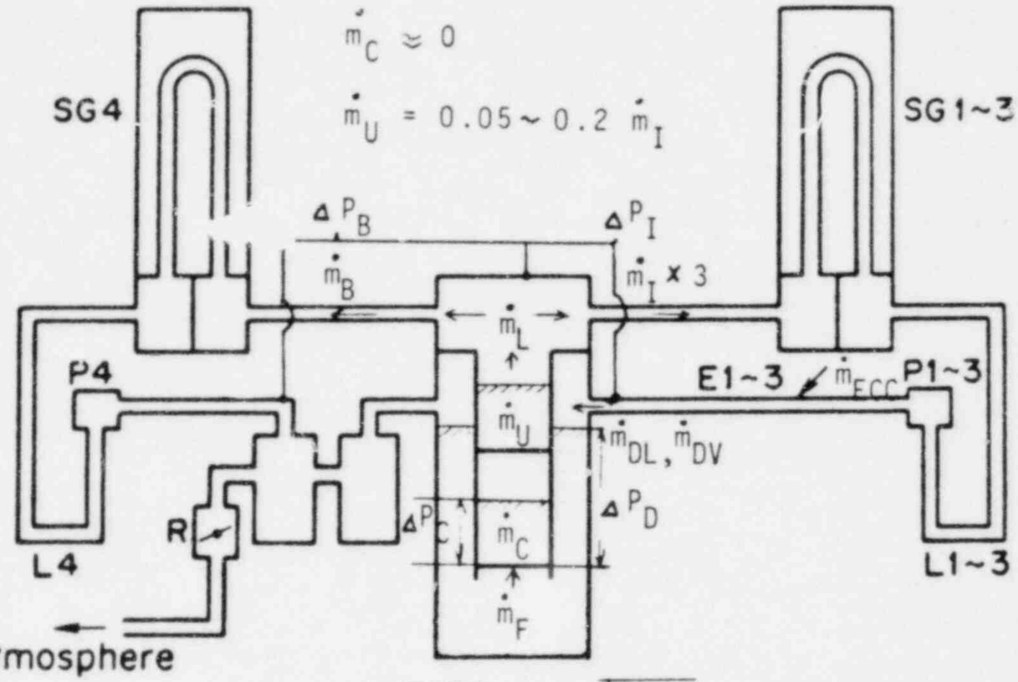


Fig. 9 Loop flow resistance (Base case)

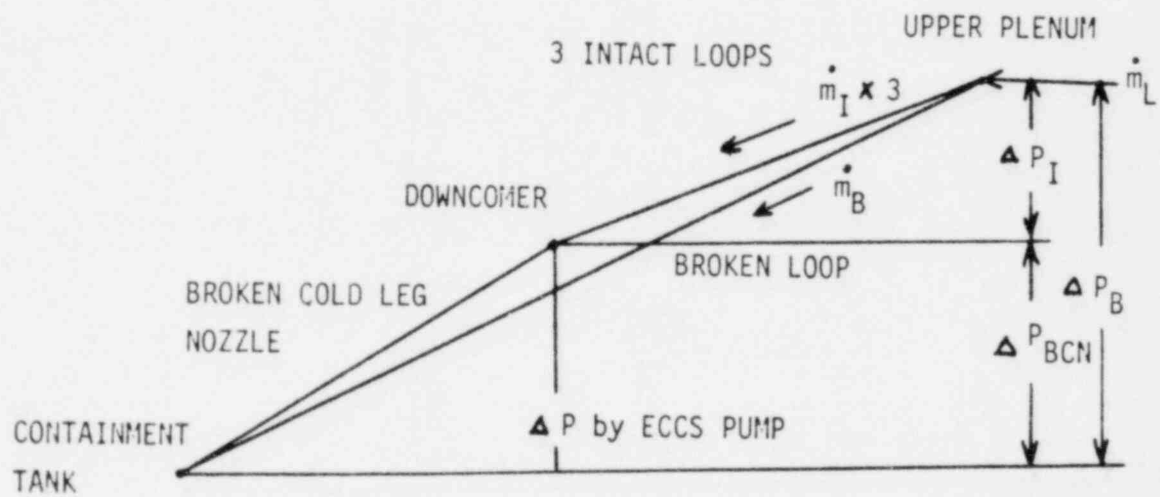
$$\dot{m}_F = \dot{m}_C + \dot{m}_U + \dot{m}_L$$

$$\dot{m}_C \approx 0$$

$$\dot{m}_U = 0.05 \sim 0.2\ \dot{m}_I$$



$$\dot{m}_L = \left(\sqrt{\frac{P_B}{P_I}} + 3\right)\sqrt{\frac{2\rho}{K}}\ S_L\sqrt{\Delta P_D - \Delta P_C}$$



$$\dot{m}_L = \dot{m}_B + 3 \times \dot{m}_I = \left(\sqrt{\frac{P_B}{P_I}} + 3\right) m_I$$

$$\frac{k}{2\rho S_L^2}\dot{m}_I^2 = \Delta P_I = \Delta P_D - \Delta P_C$$

FIG. 10 RELATION OF MASS BALANCE

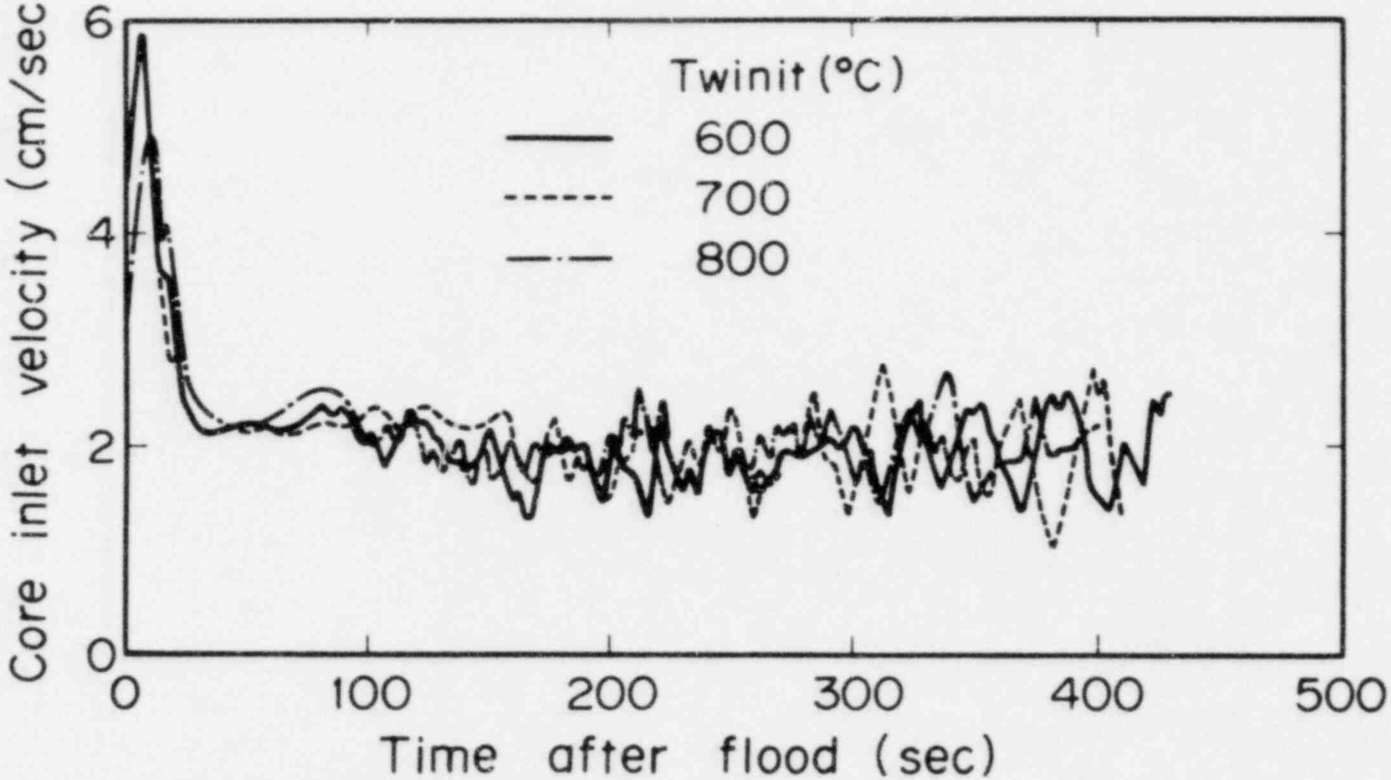


Fig. 11 Core inlet velocity (Effect of initial clad temperature)

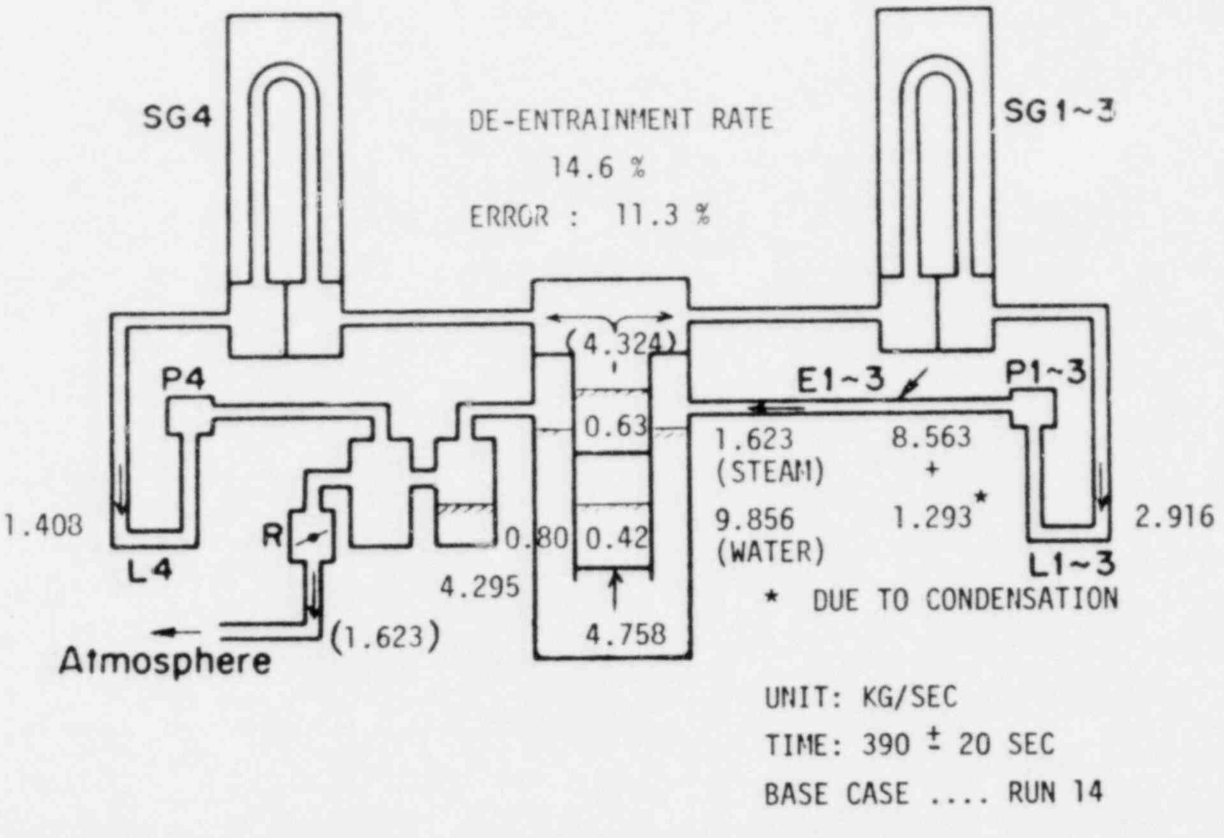


FIG. 12 MEASURED MASS BALANCE IN SYSTEM

TABLE 2 SUMMARY OF SYSTEM EFFECT

ACC FLOW RATE +	INITIAL WATER ACCUMURATION +	ΔP_D +	$\dot{M}_F$ +
LPCI FLOW RATE +	STEAM FLOW - WATER FLOW +	DOWNCOMER BYPASS - ΔP_{BCN} - $\Delta P_B/\Delta P_I$ -	$\dot{M}_F$ +
SYSTEM PRESSURE +	STEAM VELOCITY -	DOWNCOMER BYPASS - ΔP_{BCN} -, $\dot{M}_U$ +	$\dot{M}_F$ +
INITIAL CLAD TEMP. +			$\dot{M}_F$ CONST.
DOWNCOMER WALL TEMP. +	EFFECTIVE HEAD -	ΔP_D +	$\dot{M}_F$ -

LA-UR -80- 3086

TITLE: TRAC ANALYSIS SUPPORT FOR THE 2[illegible]D PROGRAM

AUTHOR(S): Ken A. Williams, Q-8

SUBMITTED TO: Eighth Water Reactor Safety Information Meeting
October 27-31, 1980
Washington, D.C.

LOS ALAMOS SCIENTIFIC LABORATORY
Post Office Box 1663 Los Alamos, New Mexico 87545

TRAC ANALYSIS SUPPORT FOR THE 2D/3D PROGRAM

by

Ken A. Williams
Energy Division
Los Alamos Scientific Laboratory

The 2D/3D program may be broadly described as analysis support to a multinational research program on refill and reflood in light water reactors (LWRs) under accident conditions. LASL's role in this program includes the application of the TRAC computer code to prediction of full-scale LWR transients and to simulation of large-scale, multidimensional, German and Japanese experimental test facilities. The use of TRAC in this program is t provide design assistance, pretest predictions, and post-test analyses for the above mentioned facilities; and most importantly, allow extrapolation of results from facility to facility (coupling) and to actual reactor behavior. The final goal of this program is an assessed, best-estimate, computer code for prediction of the course of postulated transients in large light water reactor systems.

We have recently completed two calculations of a 200% double-ended cold-leg break in a reference German pressurized water reactor (GPWR) having combined hot- and cold-leg ECC (Emergency Core Coolant) injection. In the calculation reported herein the intact loop coolant pumps were allowed to coastdown to a low speed, while in the other case, the pump speeds were maintained constant after 30 s. The primary conclusion is that by 110 s the transient is terminated with the core completely filled with liquid and with all fuel rods quenched. This TRAC-PD2 calculation differs significantly from previous calculations in that subcooled hot-leg ECC water pools in the upper plenum and penetrates into the core, and results in early downward rod quenching. The steam generated from this cooling provides a large reverse core steam flow rate that delays lower plenum filling and bottom quenching. However, when the cold leg ECC water begins filling the core, bottom quenching progresses rapidly due to the previous core cooling. Moreover, there does not appear to be core steam binding as a result of liquid pools in both the lower plenum and upper plenum. This calculation shows considerable multidimensional behavior in the vessel, especially with regard to liquid penetration into the core and rod temperatures. The highest rod temperatures and latest quenches occur in the azimuthal zones connected to the broken loop. The peak rod temperature of 875 K occurs at 6 s during blowdown, with this rod finally being quenched at 106 s.

During the past year over 40 design calculations were performed for the Slab Core Test Facility (SCTF). The majority of these calculations addressed the necessity of an additional steam supply for prototypical simulations. While our calculations concluded that an extra steam source was unnecessary to get positive steam flow through the core, such a source would add operational flexibility to the system.

In support of the Cylindrical Core Test Facility (CCTF) we have performed post-test analyses of four tests and provided a pretest prediction for an additional test. Comparisons are presented for CCTF Run 20 clad temperature data and the TRAC-PD2 post-test calculation. The comparisons include statistical information to illustrate the range and standard deviation of the experimental data. The calculated heater rod surface temperatures are in agreement with the data for elevations from the core entrance to about 0.2 m above the core midplane. This is true for both peak clad temperatures and quench times. For elevations above this level the PD2 results are good until slightly after peak clad temperatures are reached (i.e., 300-400 s). After this period the surface heat transfer is under predicted, resulting in over prediction of temperatures and quench times. However, internal modifications to PD2 have produced much better agreement with data at these upper core locations. The total liquid carryover rate is in good agreement with data, as the liquid pool on the upper core support plate is calculated to be approximately 0.25-0.30 m deep.

The 2D/3D TRAC analysis near term activities are summarized in the final figure, showing activity on all 2D/3D facilities and reference LWR's.

2D/3D ANALYSIS STAFF

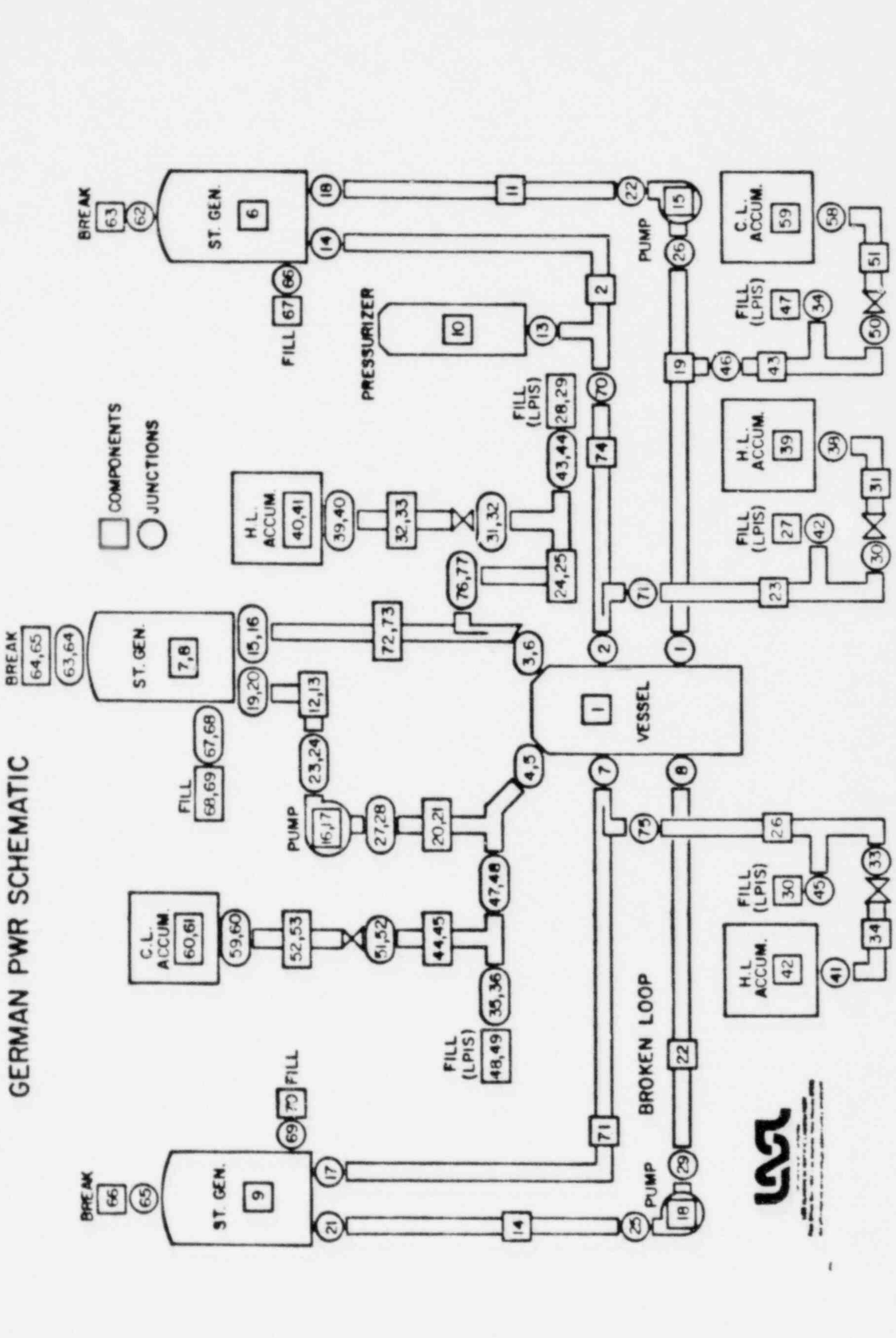
K. WILLIAMS	ANALYSIS SECTION LEADER
S. SMITH	SCTF / PKL
R. FUJITA	JAERI RESIDENT ENGINEER
F. MOTLEY	GPWR
T. BROWN	CCTF
M. CAPPIELLO	UPTF

GERMAN REFERENCE REACTOR CALCULATION

- FULL POWER: 3765 MW_T
- DOUBLE-ENDED COLD LEG BREAK
- COMBINED HOT AND COLD LEG ECC INJECTION
- ACCIDENT SIMULATION THROUGH COMPLETE CORE REFLOODING

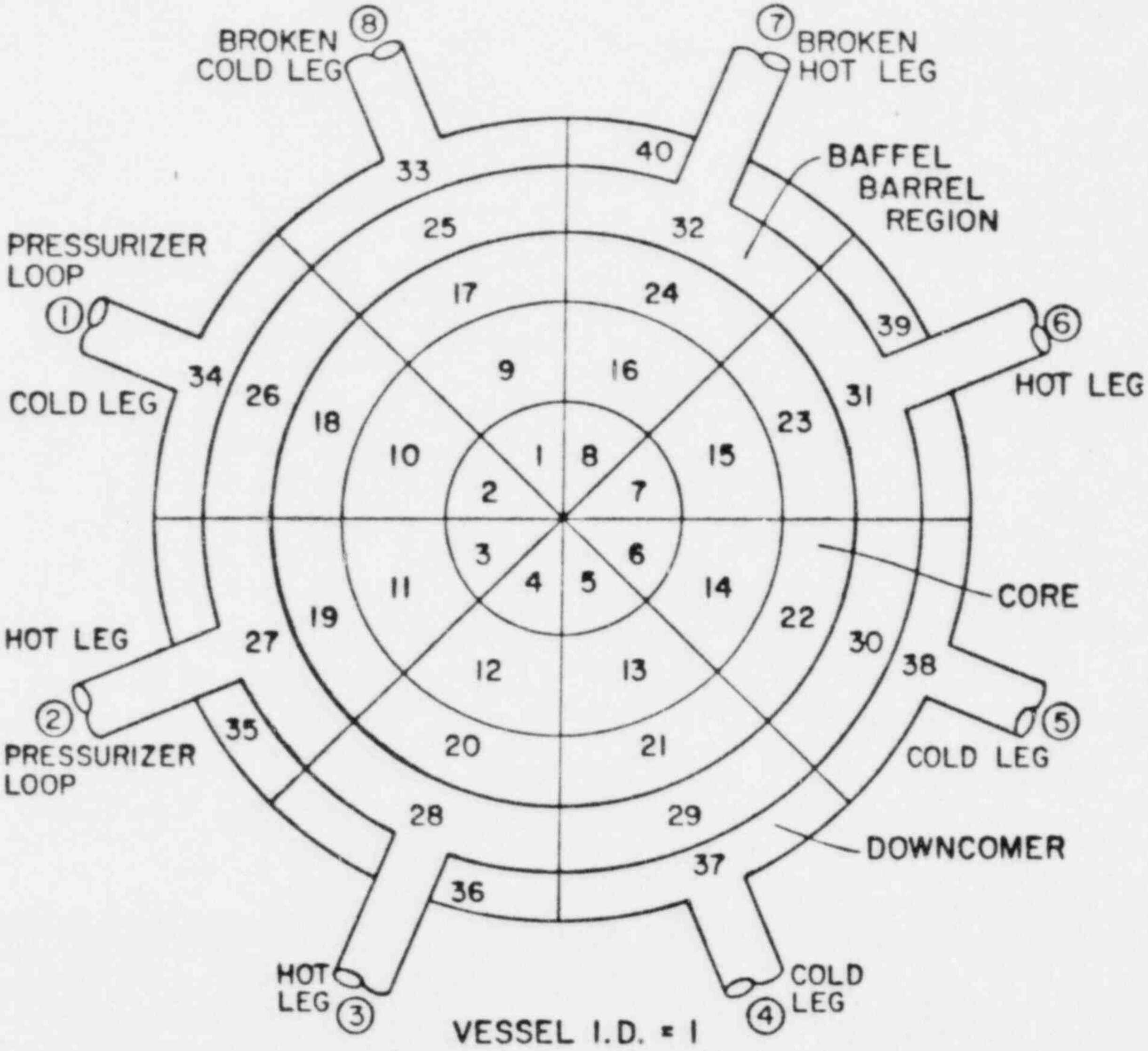
LASL

STS:10/80 INFMTQ

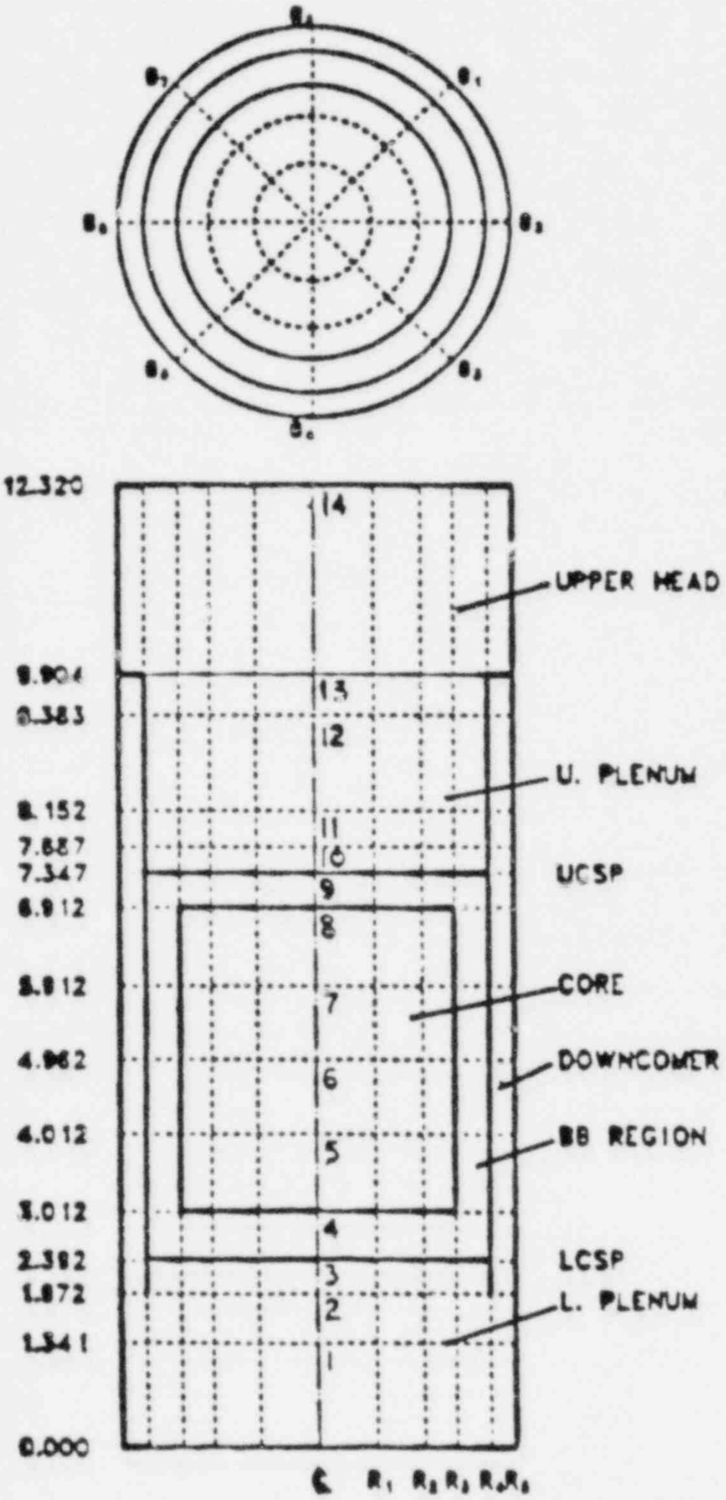


TRAC model schematic for German PWR

Horizontal vessel noding



GERMAN PWR - TRANSIENT REFERENCE REACTOR BASE CASE

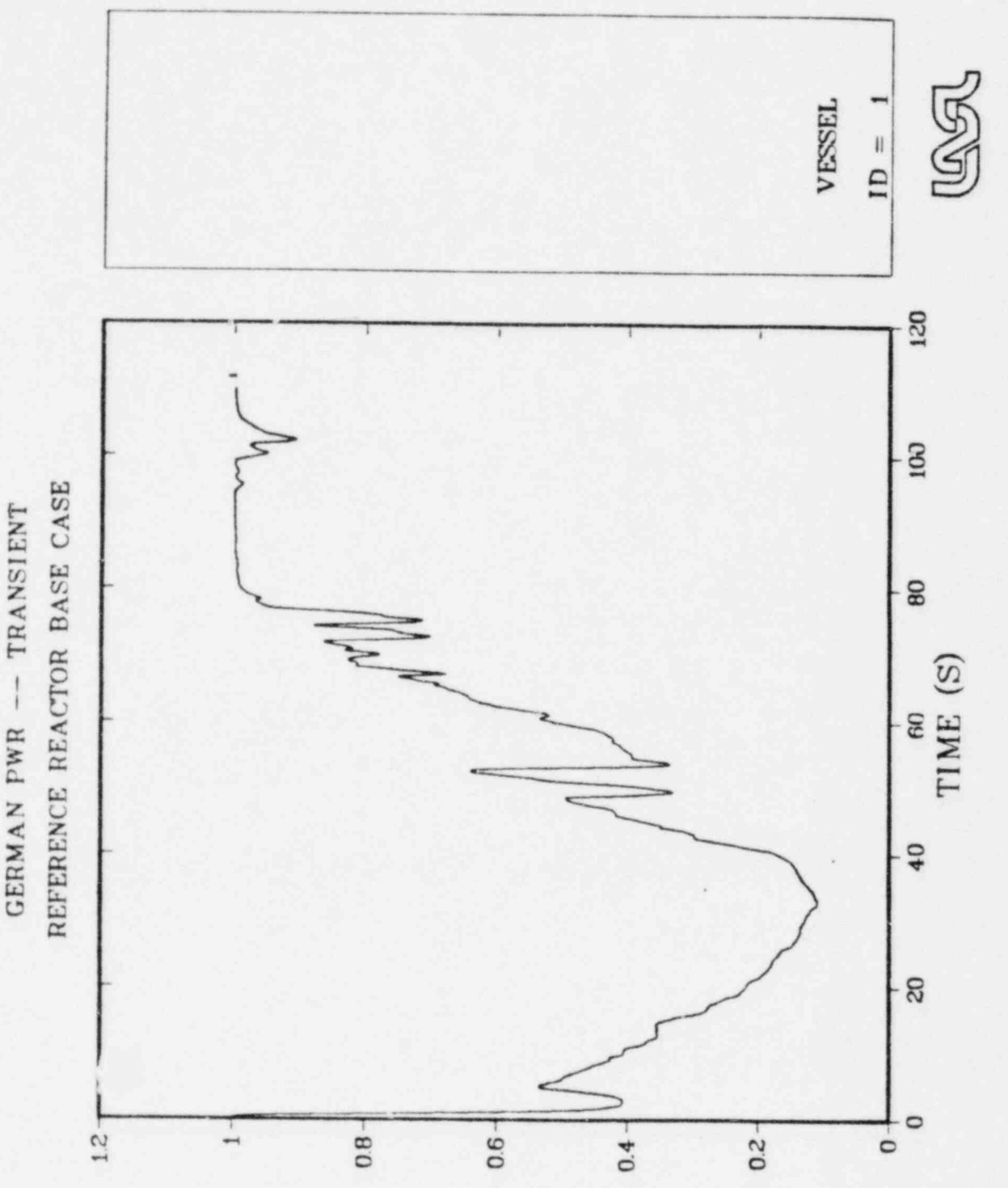


R_1 = .750
R_2 = 1.350
R_3 = 1.750
R_4 = 2.185
R_5 = 2.500

θ_1 = 45.0
θ_2 = 90.0
θ_3 = 135.0
θ_4 = 180.0
θ_5 = 225.0
θ_6 = 270.0
θ_7 = 315.0
θ_8 = 360.0

GERMAN PWR VESSEL NODING DIAGRAM

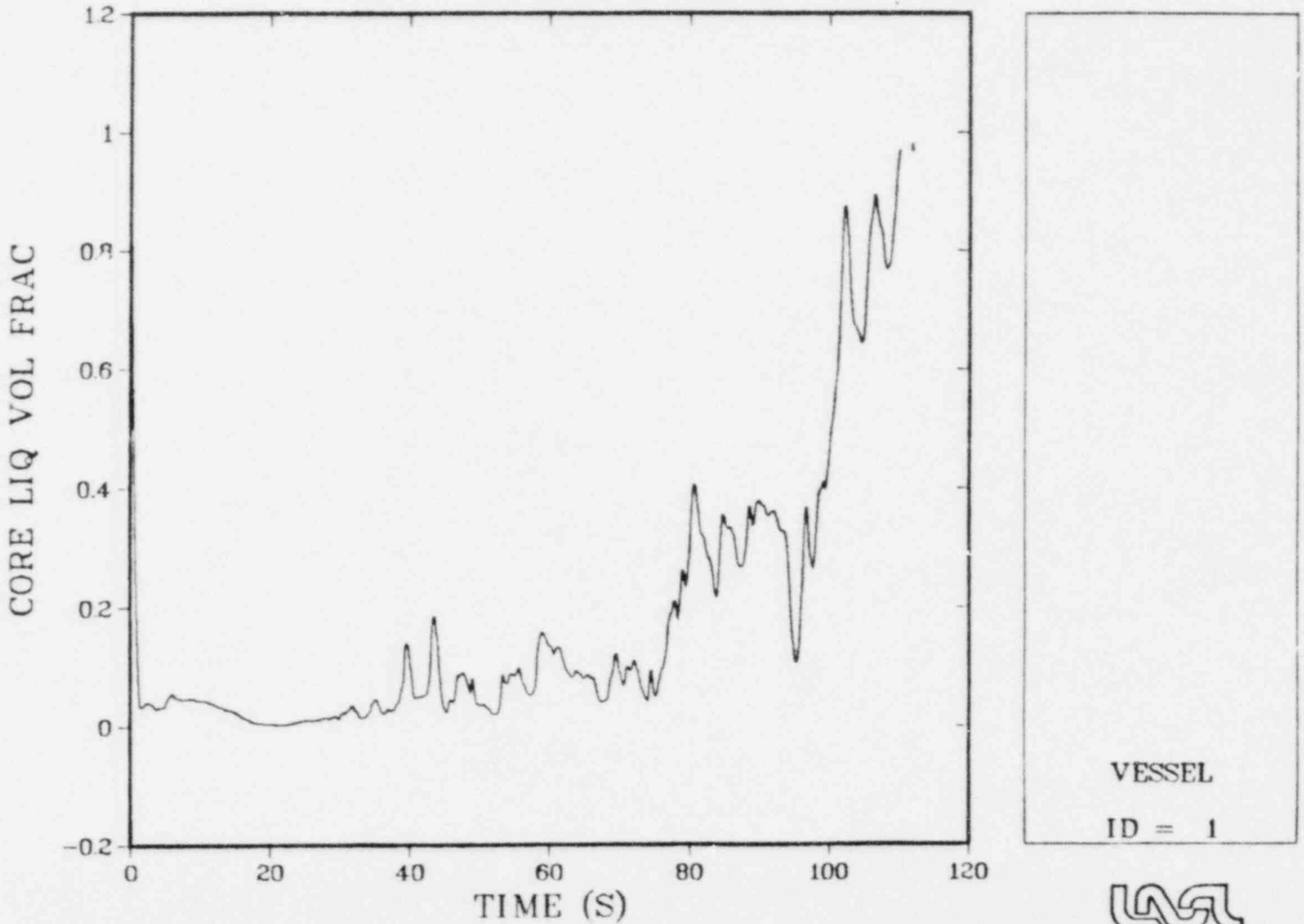
Axial vessel noding



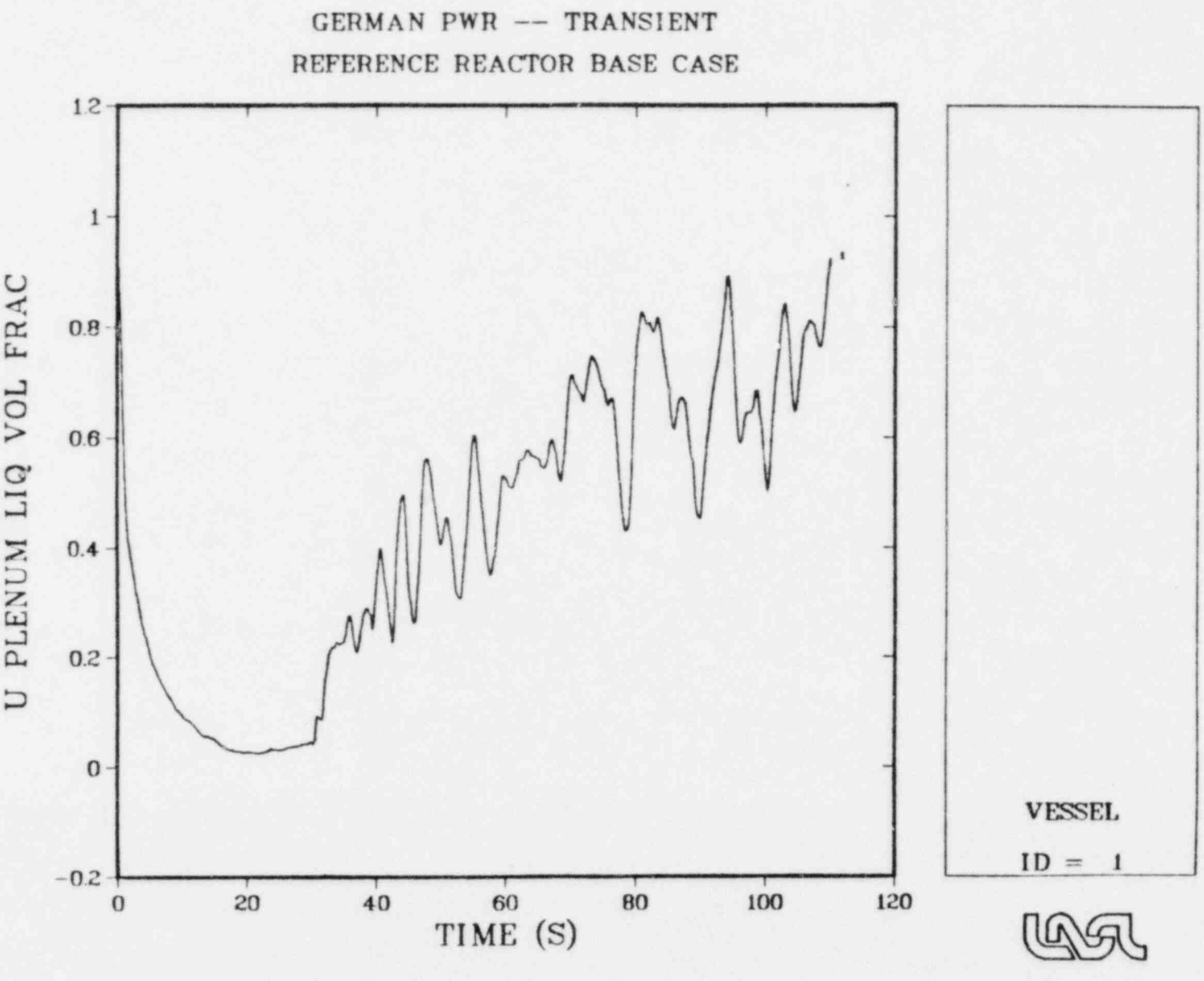
Lower Plenum Liquid Volume Fraction

GERMAN PWR -- TRANSIENT

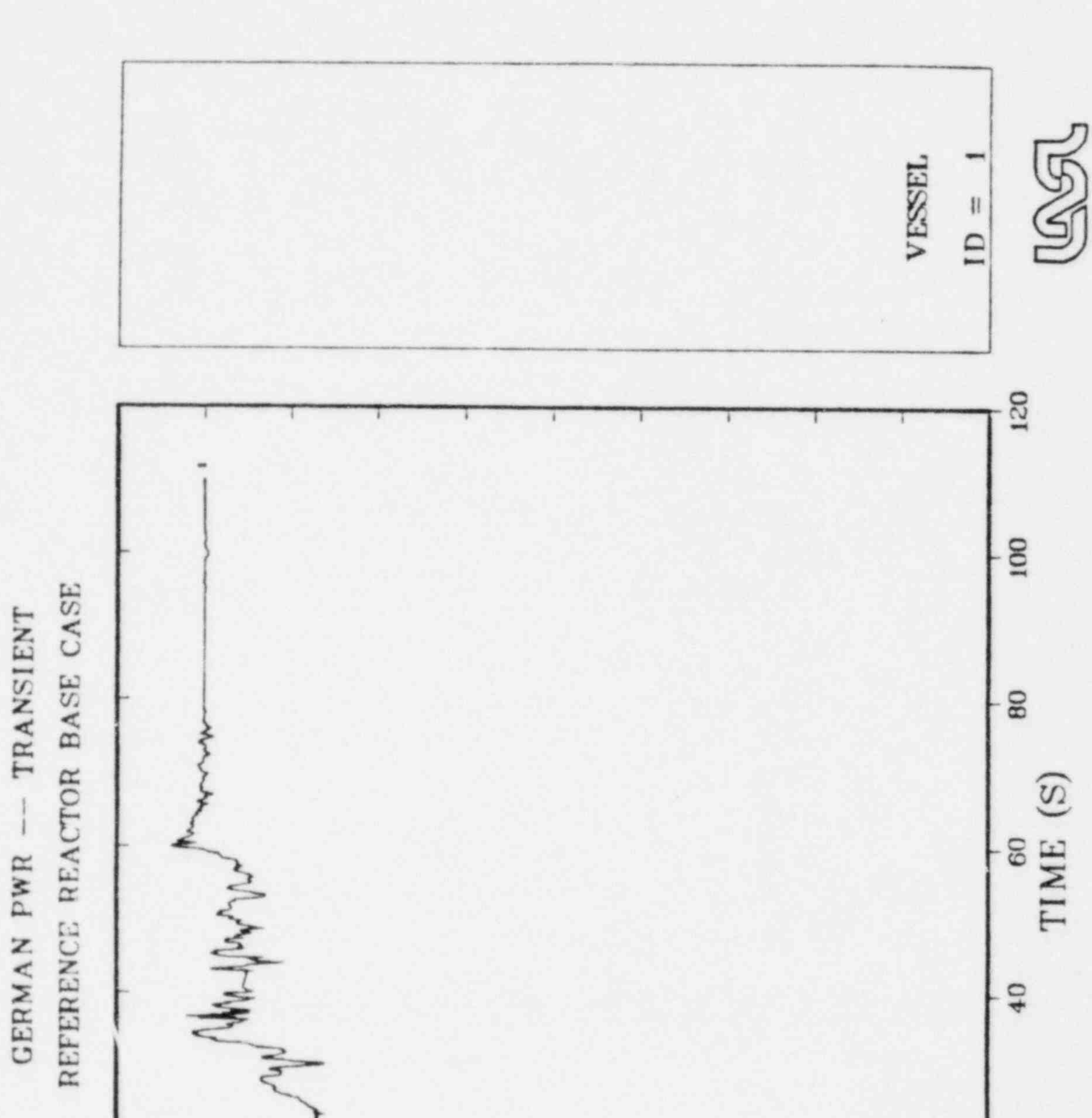
REFERENCE REACTOR BASE CASE



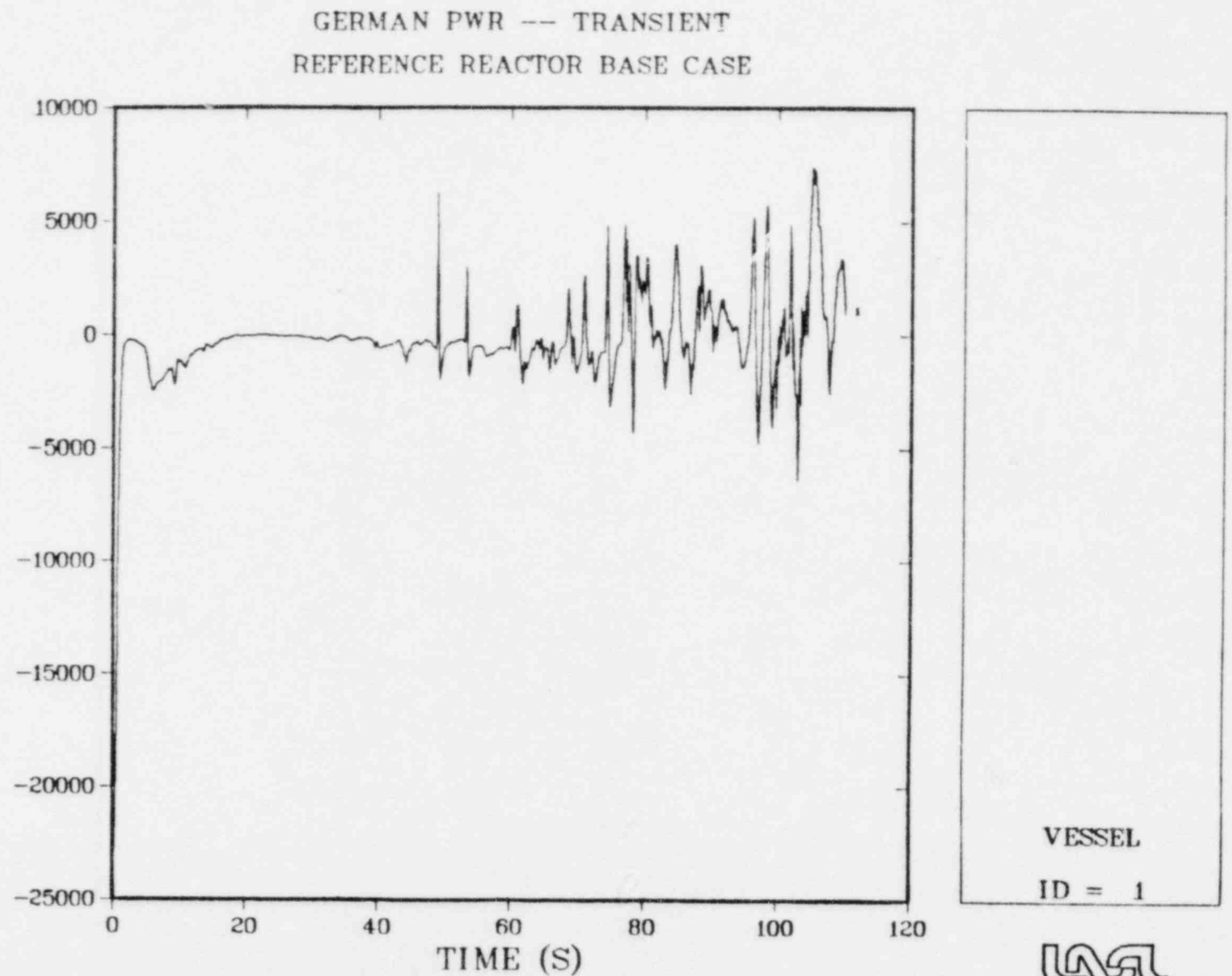
Core liquid volume fraction



Upper Plenum liquid volume fraction



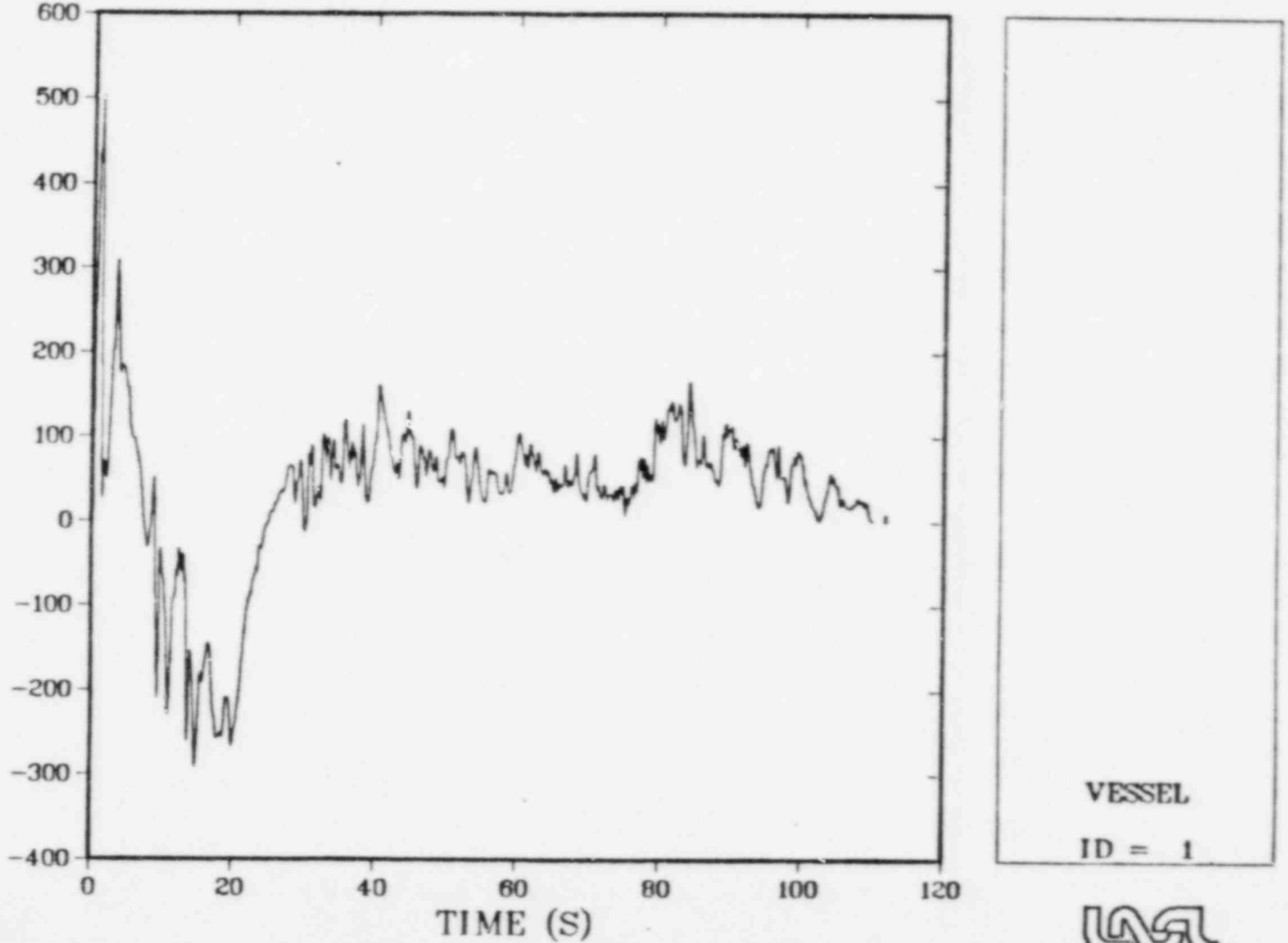
Core inlet vapor mass flow rate



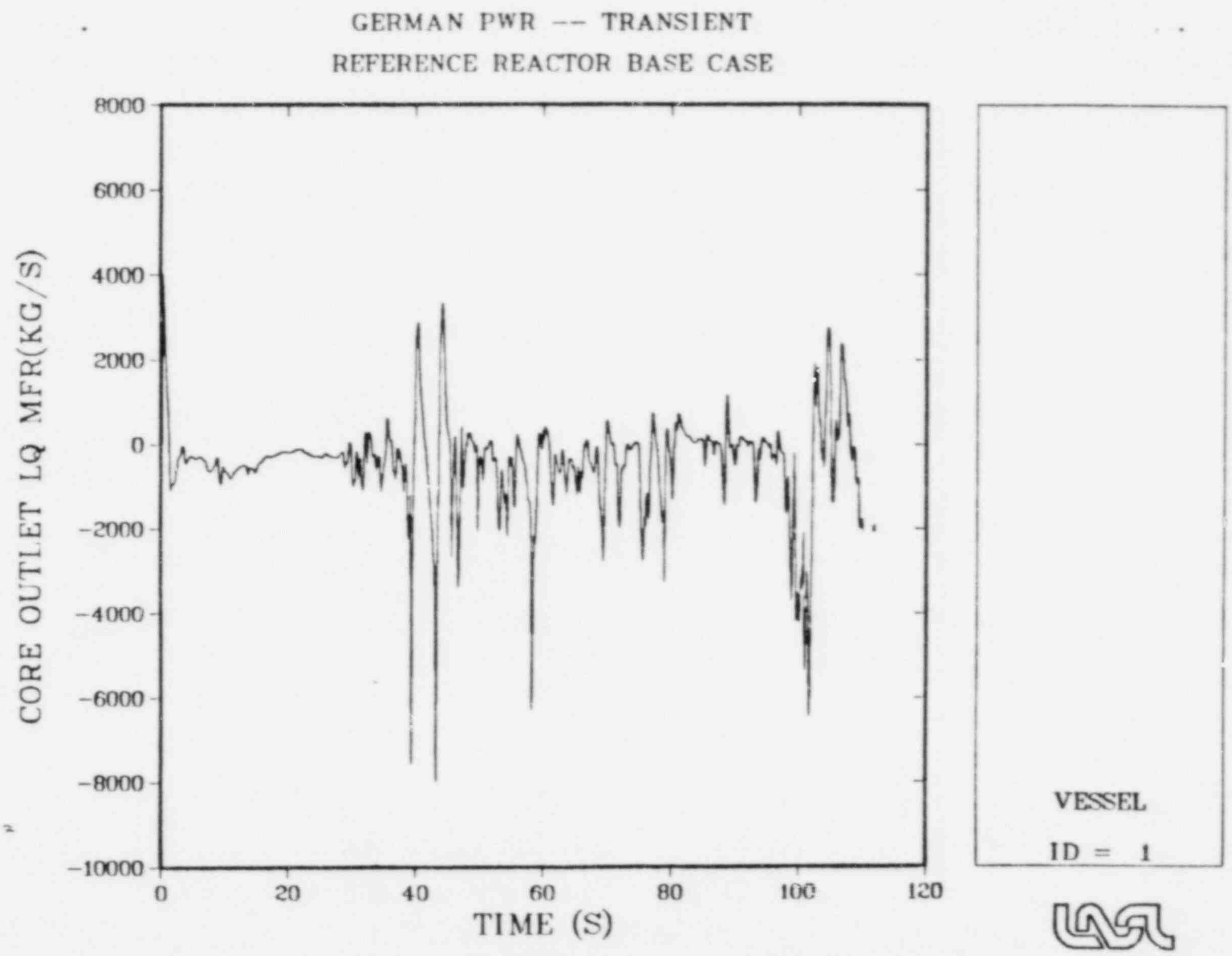
Core inlet liquid mass flow rate

GERMAN PWR -- TRANSIENT

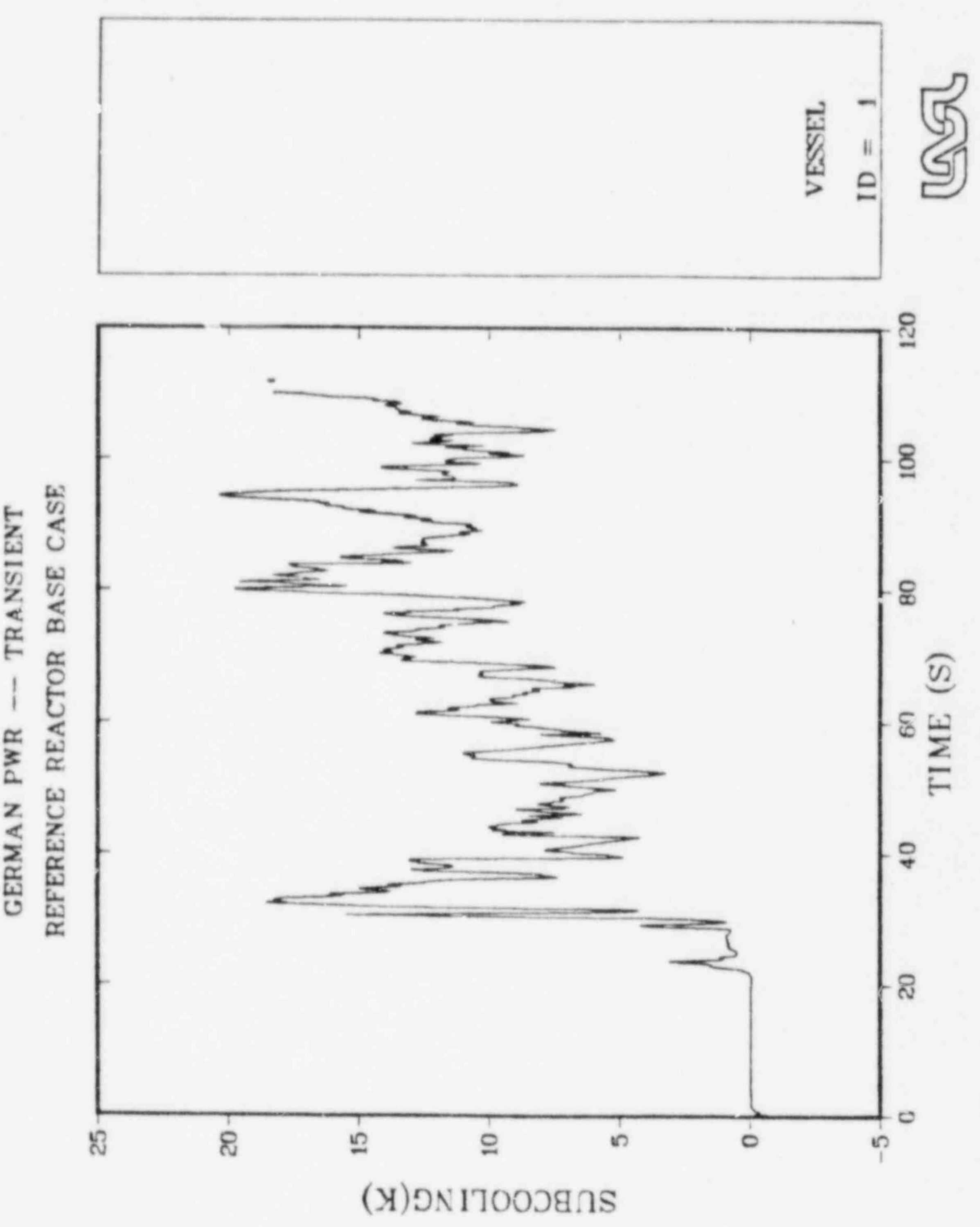
REFERENCE REACTOR BASE CASE



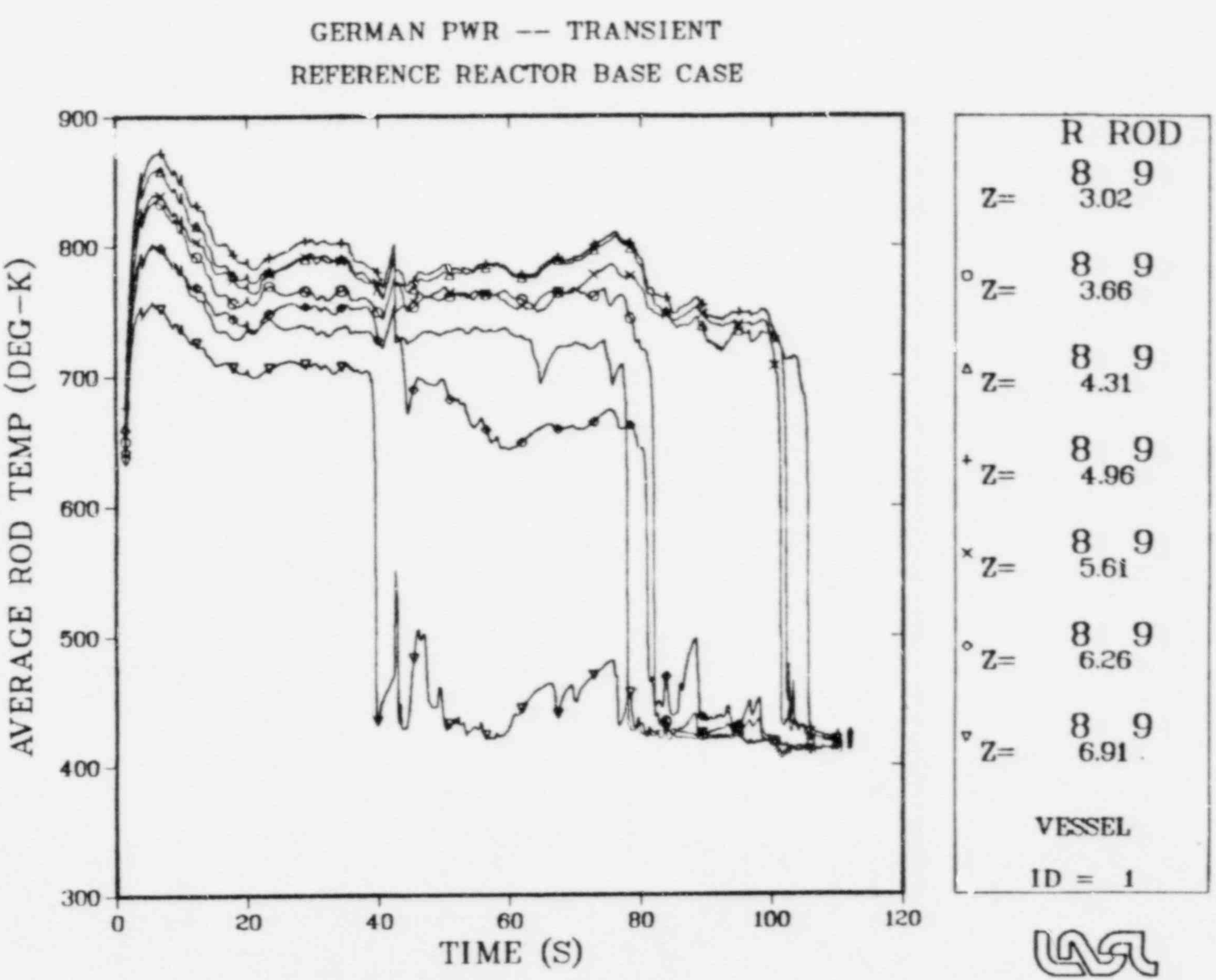
Core outlet vapor mass flow rate



Core outlet liquid mass flow rate

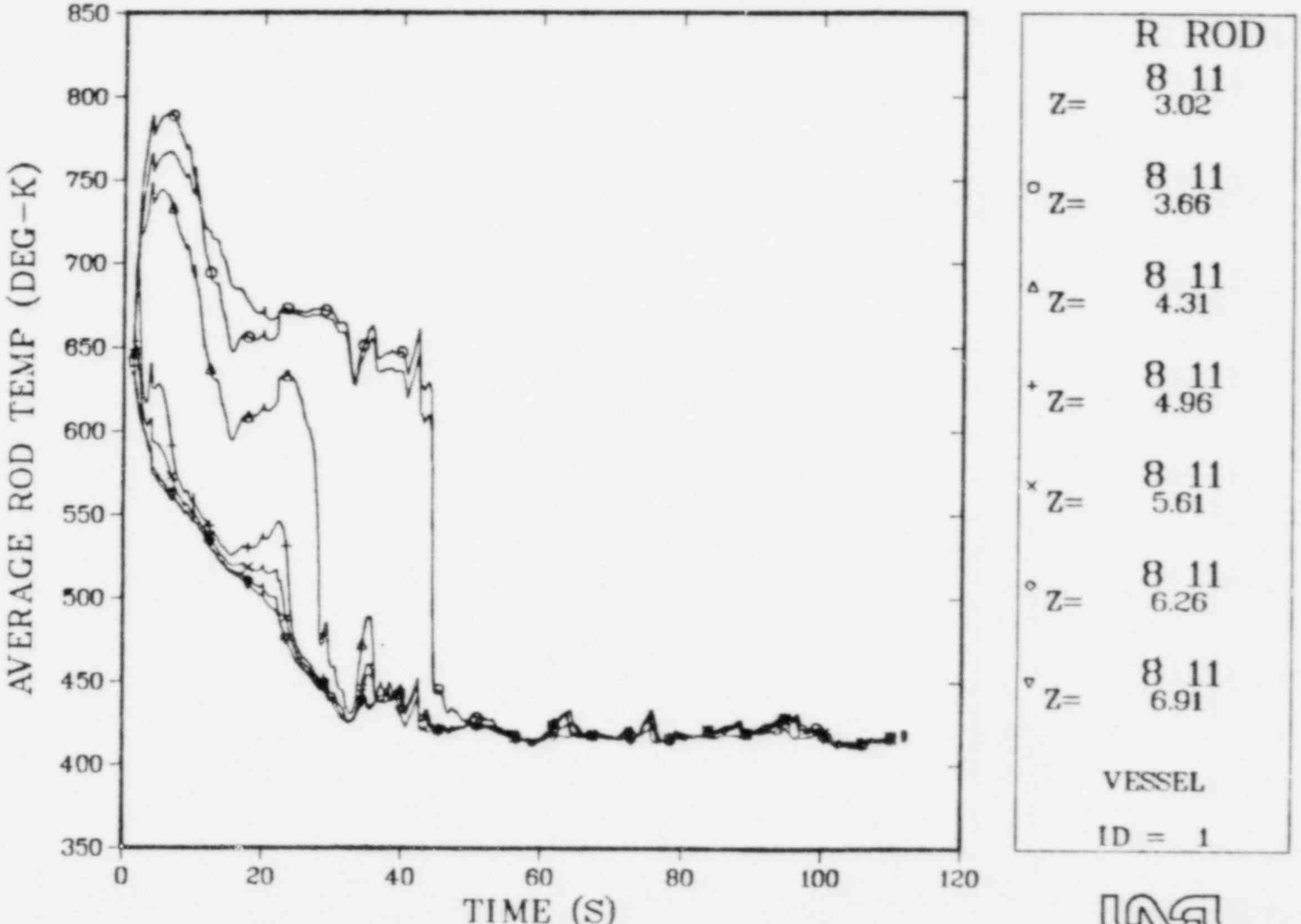


Average subcooling in upper plenum

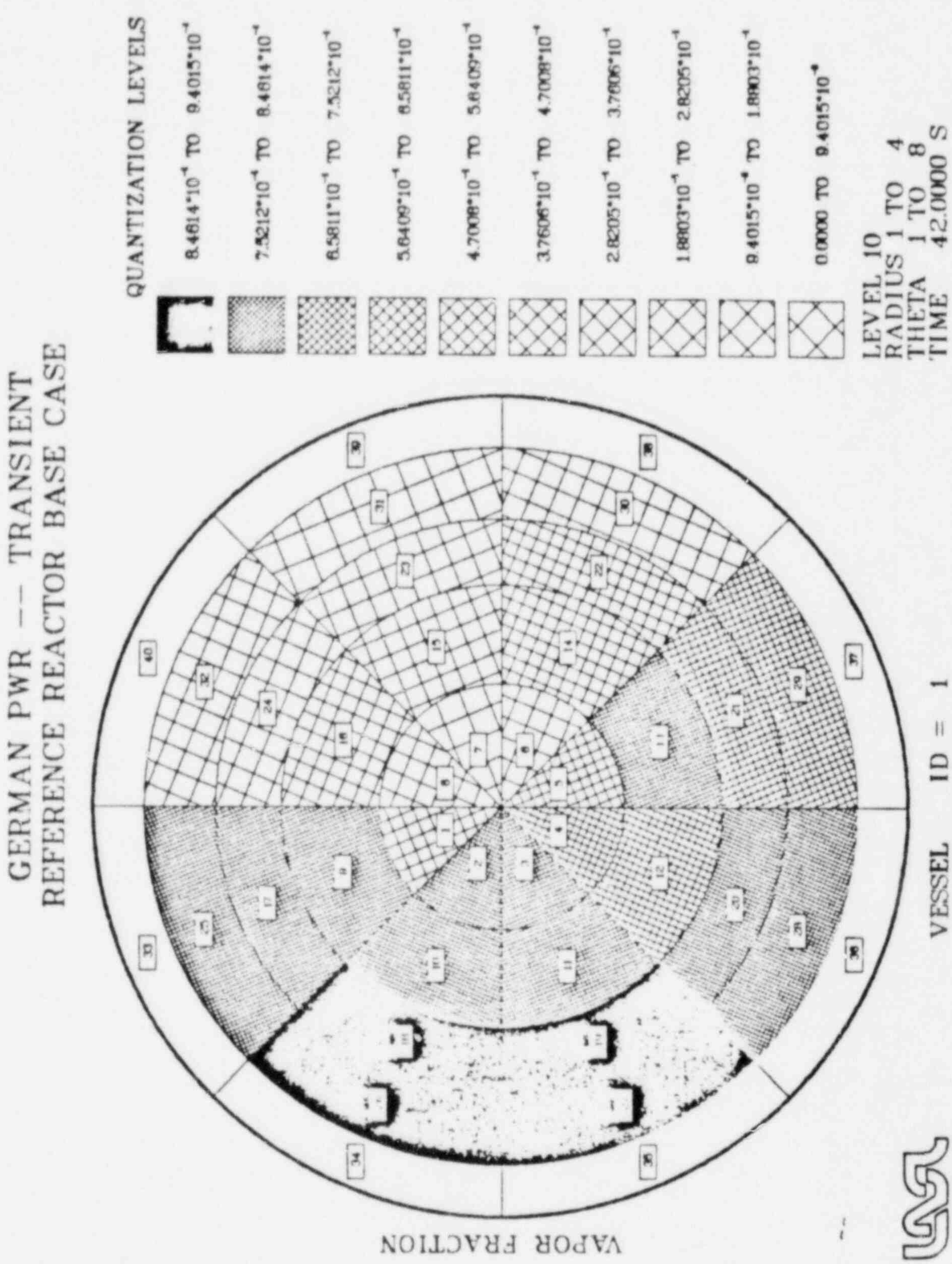


Rod 9 average surface temperature

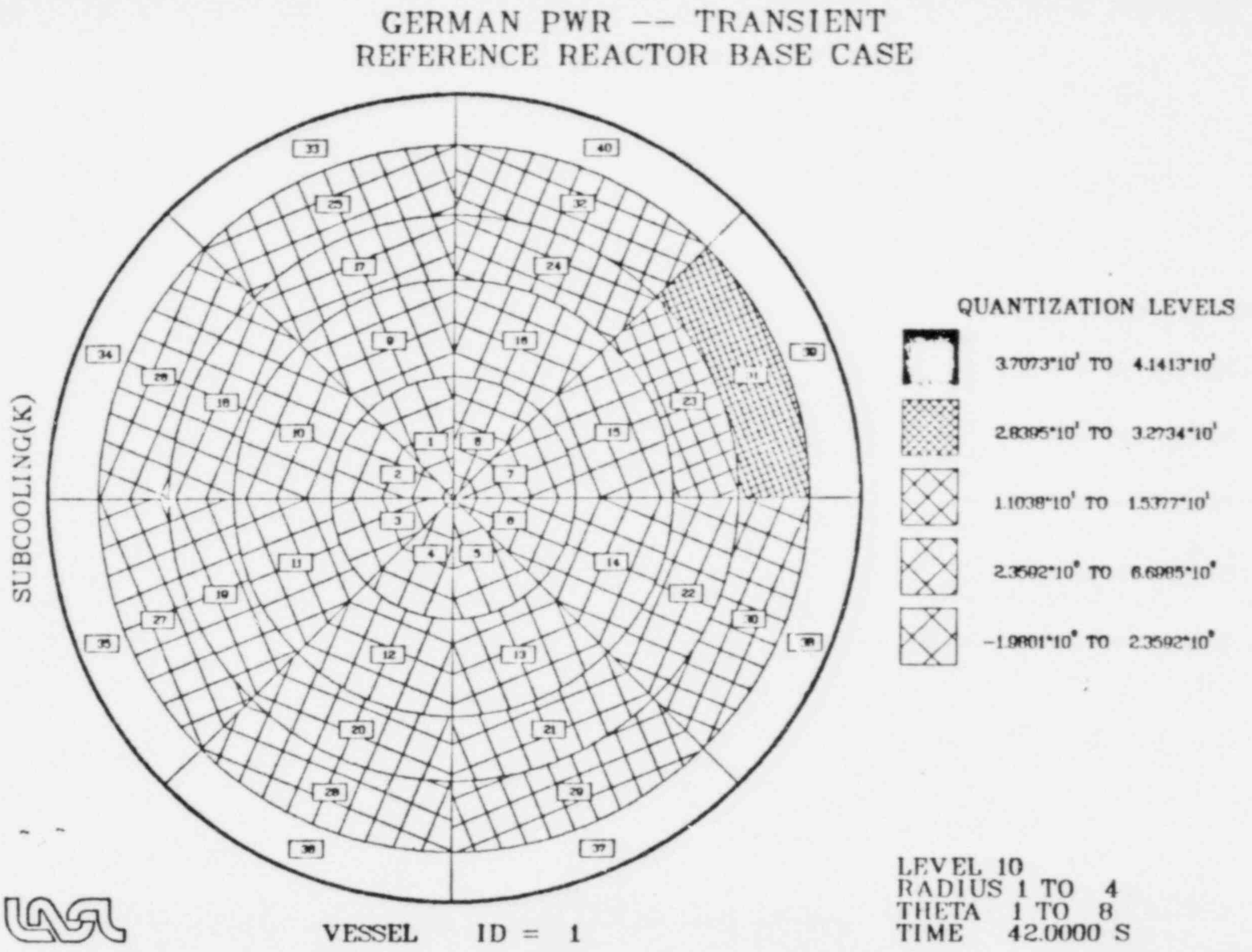
GERMAN PWR -- TRANSIENT
REFERENCE REACTOR BASE CASE



Rod 11 average surface temperature

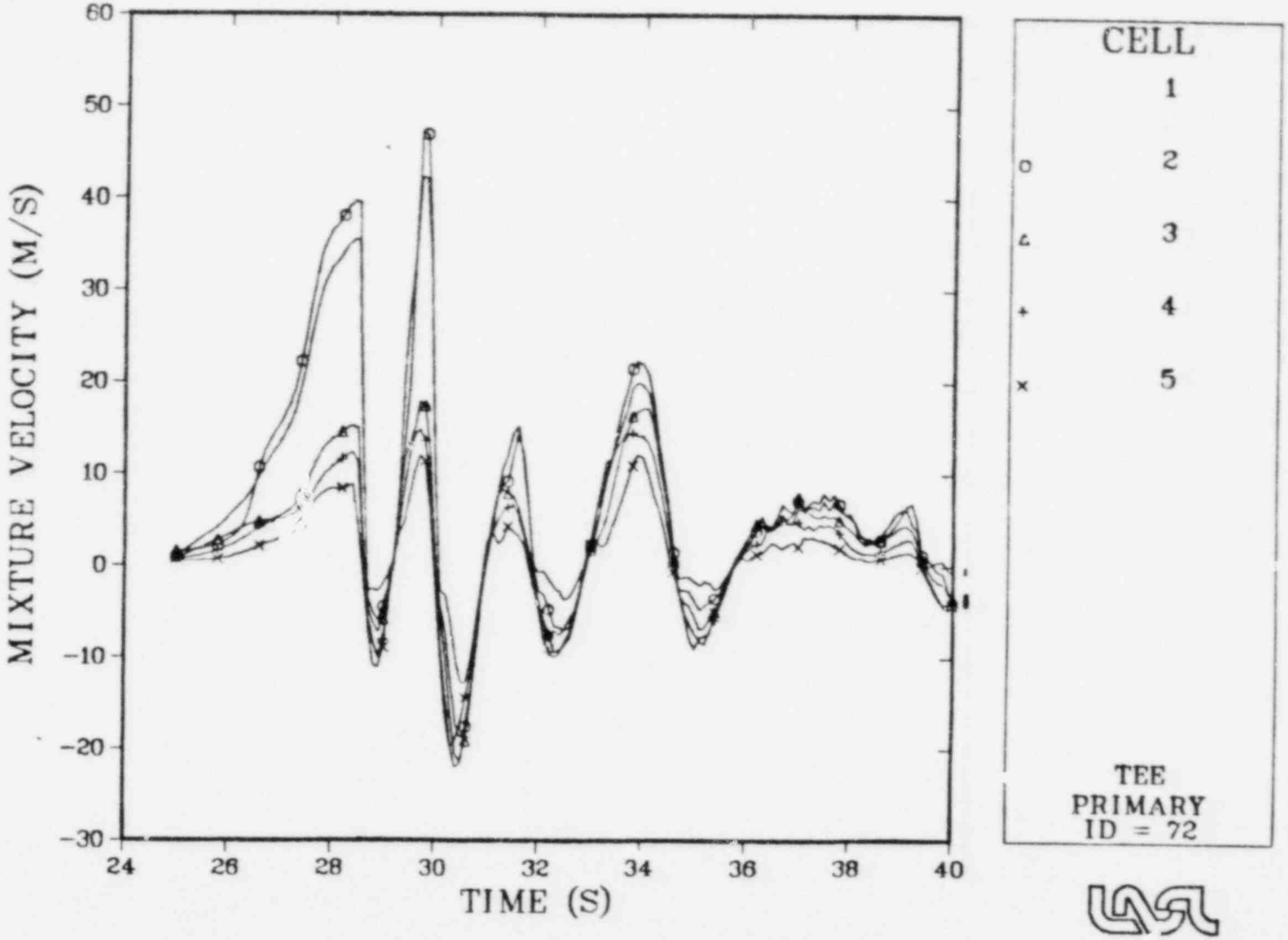


Upper core support plate void fraction at 42 sec.

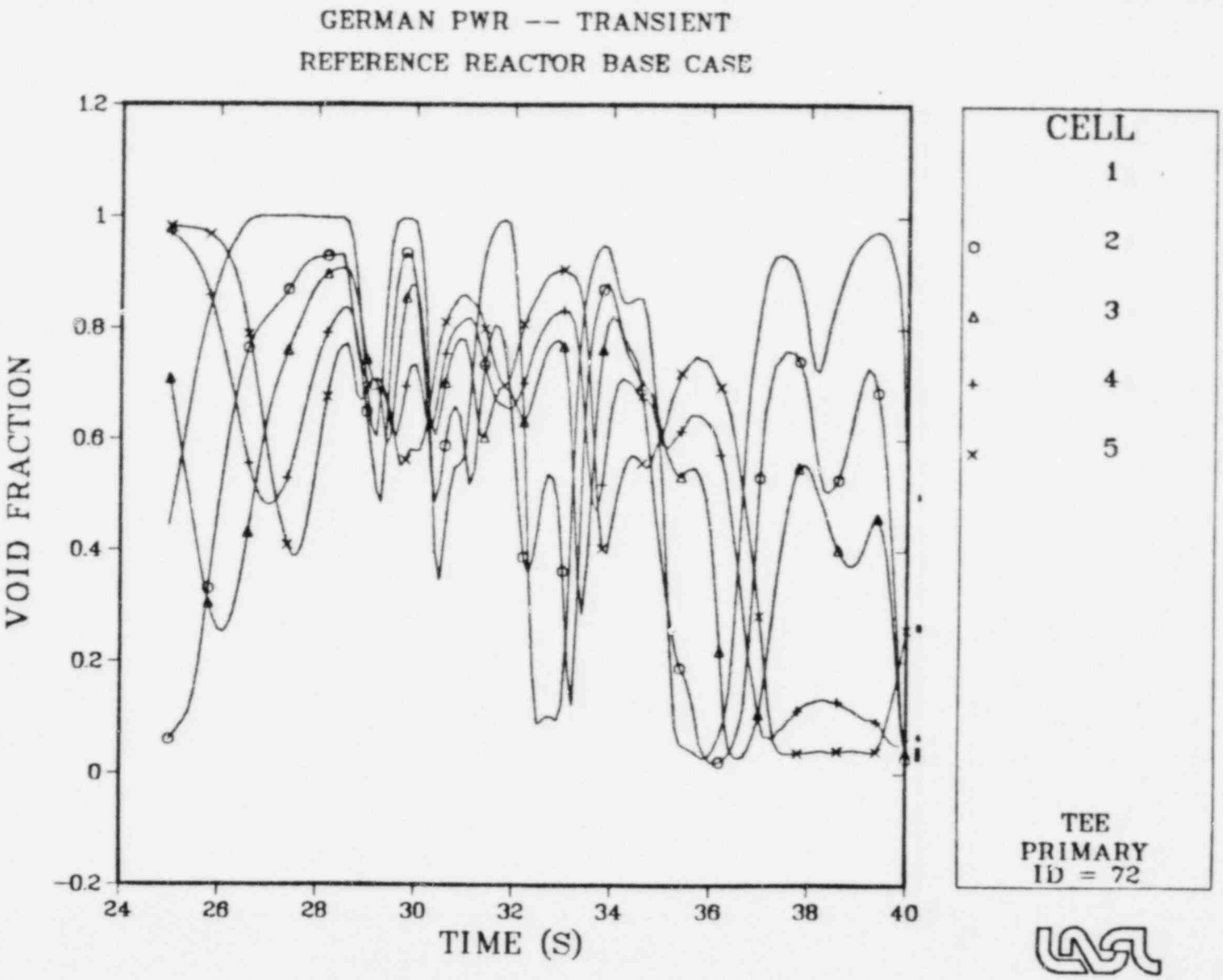


Upper core support plate liquid subcooling at 42 sec.

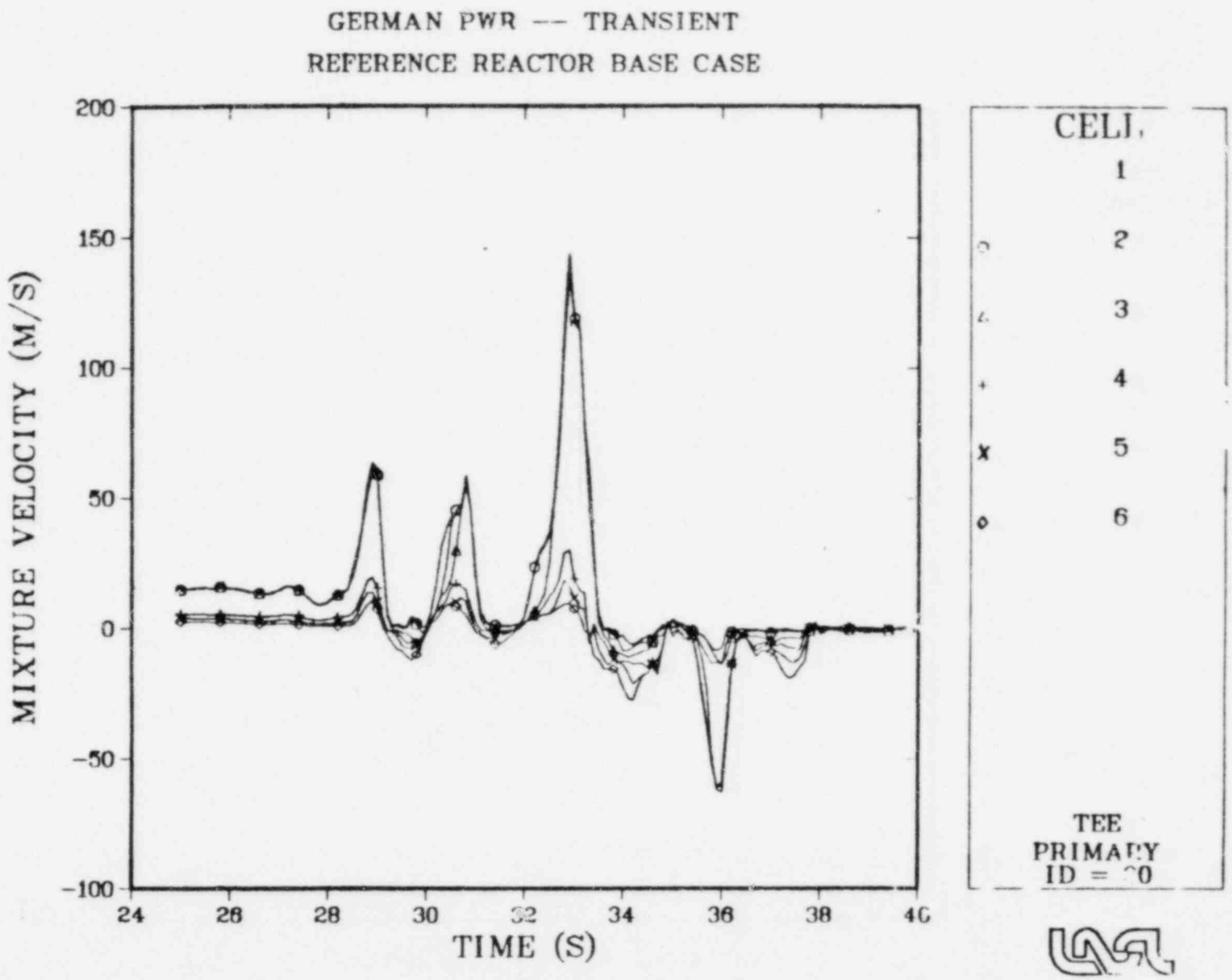
GERMAN PWR -- TRANSIENT
REFERENCE REACTOR BASE CASE



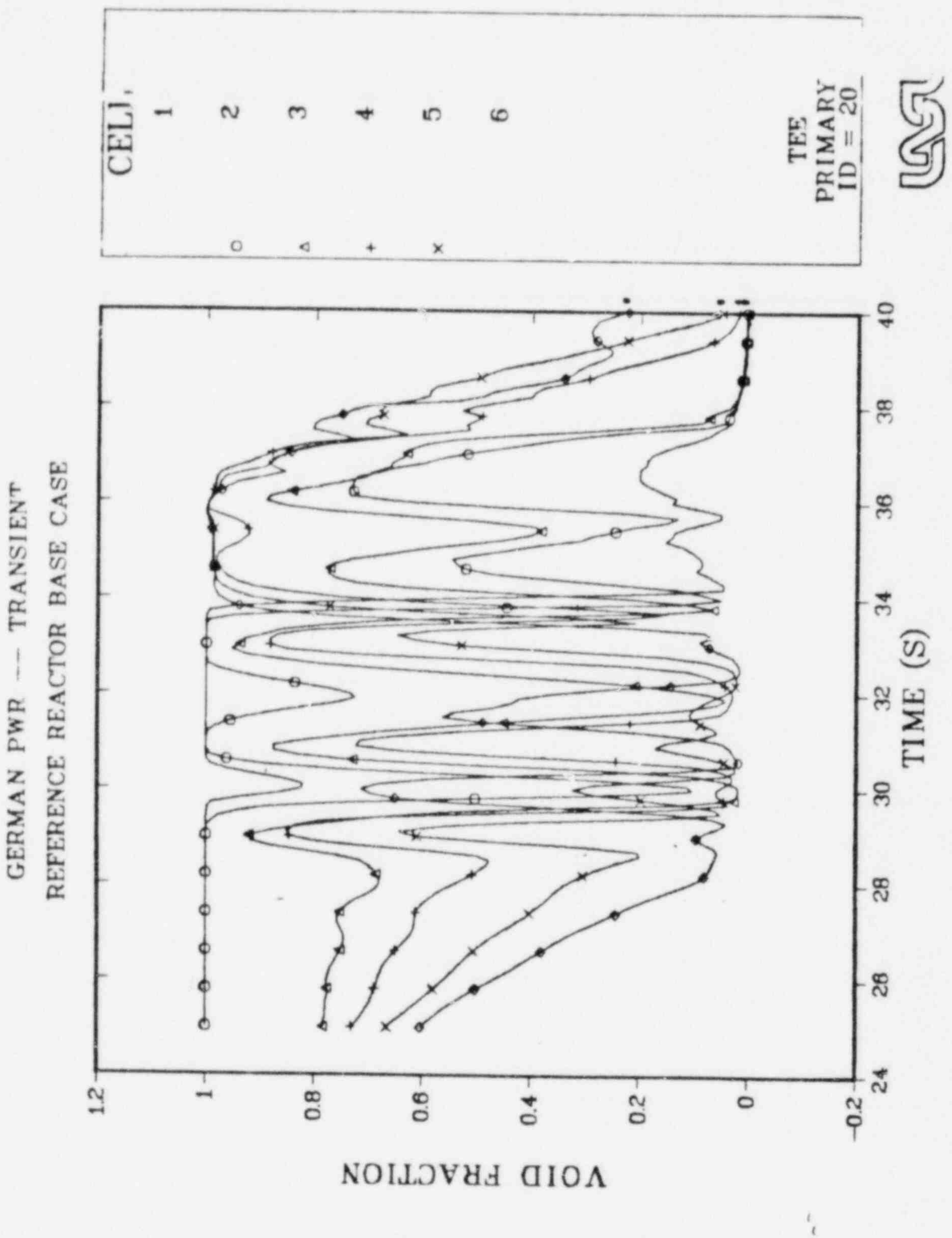
Mixture velocity in each cell of intact loop hot leg from 25 to 40 sec.



Void fraction in each cell of intact loop hot leg from 25 to 40 sec.



Mixture velocity in each cell of intact loop cold leg from 25 to 40 sec.



Void fraction in each cell of intact loop cold leg from 25 to 40 sec.

SLAB CORE TEST FACILITY CALCULATIONS

- POWER: 6.65 MW_T
- DOUBLE-ENDED COLD LEG BREAK
- COMBINED HOT AND COLD LEG ECC INJECTION
- UNBLOCKED CORE

SCTF COMBINED INJECTION

53 CASES WERE PERFORMED WITH TRAC DURING FISCAL 1980

- STEAM SUPPLY STUDY - 41 CASES
- SENSITIVITY STUDY - 11 CASES
- REFLOOD CALCULATION - 1 CASE

STS:10/80 INFMTG

SCTF STEAM SUPPLY STUDY CONCLUDED EXTRA STEAM SUPPLY UNNECESSARY BUT WOULD ADD FLEXIBILITY

. 5 BASIC CASES VARYING ECC TEMPERATURES

. 11 PARAMETRIC VARIATIONS OF THESE CASES

. RESULTS OF 50 S TRANSIENT:

- SUBCOOLED LIQUID IN COLD-LEG ECC RESULTED IN LOWER PLENUM REFILL

- SUBCOOLED LIQUID IN HOT-LEG ECC FORMED UPPER PLENUM POOL AND GAVE MORE LIQUID FALLBACK

- BEGAN QUENCHING FROM BOTTOM AND TOP

- EXTRA STEAM SOURCE UNNECESSARY TO GET POSITIVE STEAM FLOW THROUGH CORE

CALCULATION OF SCTF TRANSIENT THROUGH REFLOOD SHOWED SIMILARITY TO GPWR CALCULATION

. TRAC-PD2; GPWR INITIAL CONDITIONS

. CALCULATED UNTIL CORE COMPLETELY QUENCHED AT ABOUT 120 S OF GPWR TIME (90 S SCTF TIME

. TRANSIENT BEHAVIOR:
  - LOWER PLENUM FILLS AT 40 S
  - QUENCHING BOTH BY FALLING FILM AND BY BOTTOM REFLOOD
  - RODS FULLY QUENCHED AT 90 S
  - VESSEL LIQUID-FILLED AT 100 S

LASL

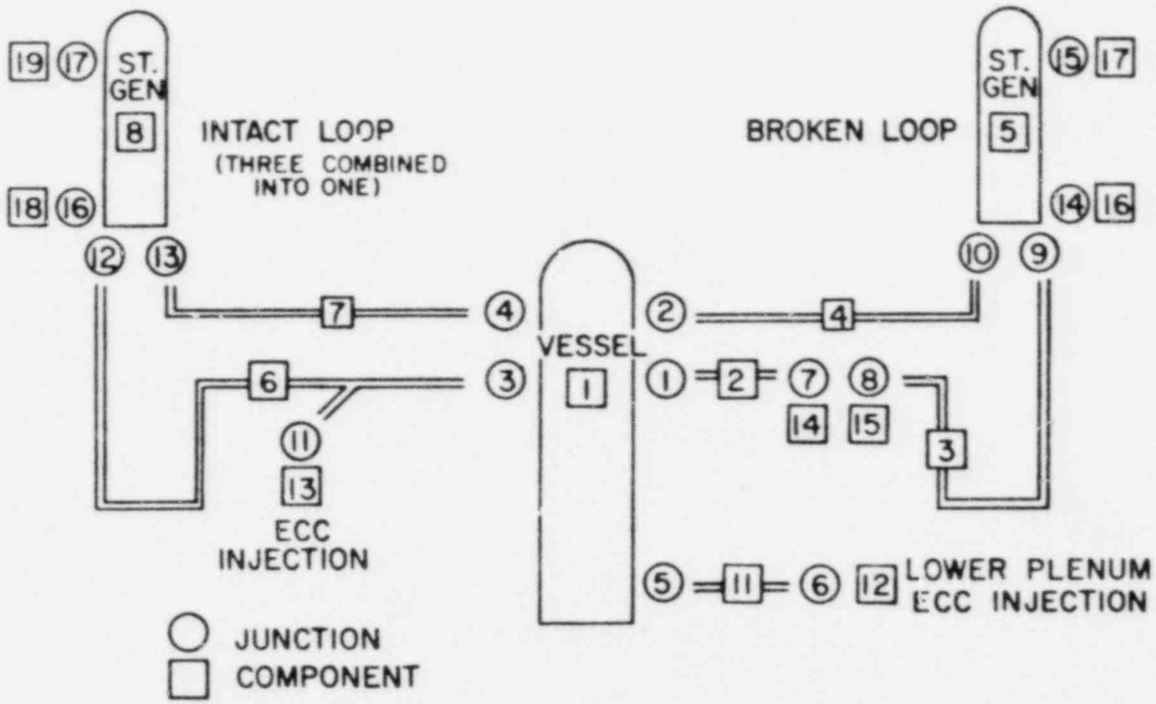
STS 10/80 INFMTQ

CYLINDRICAL CORE TEST FACILITY CALCULATIONS

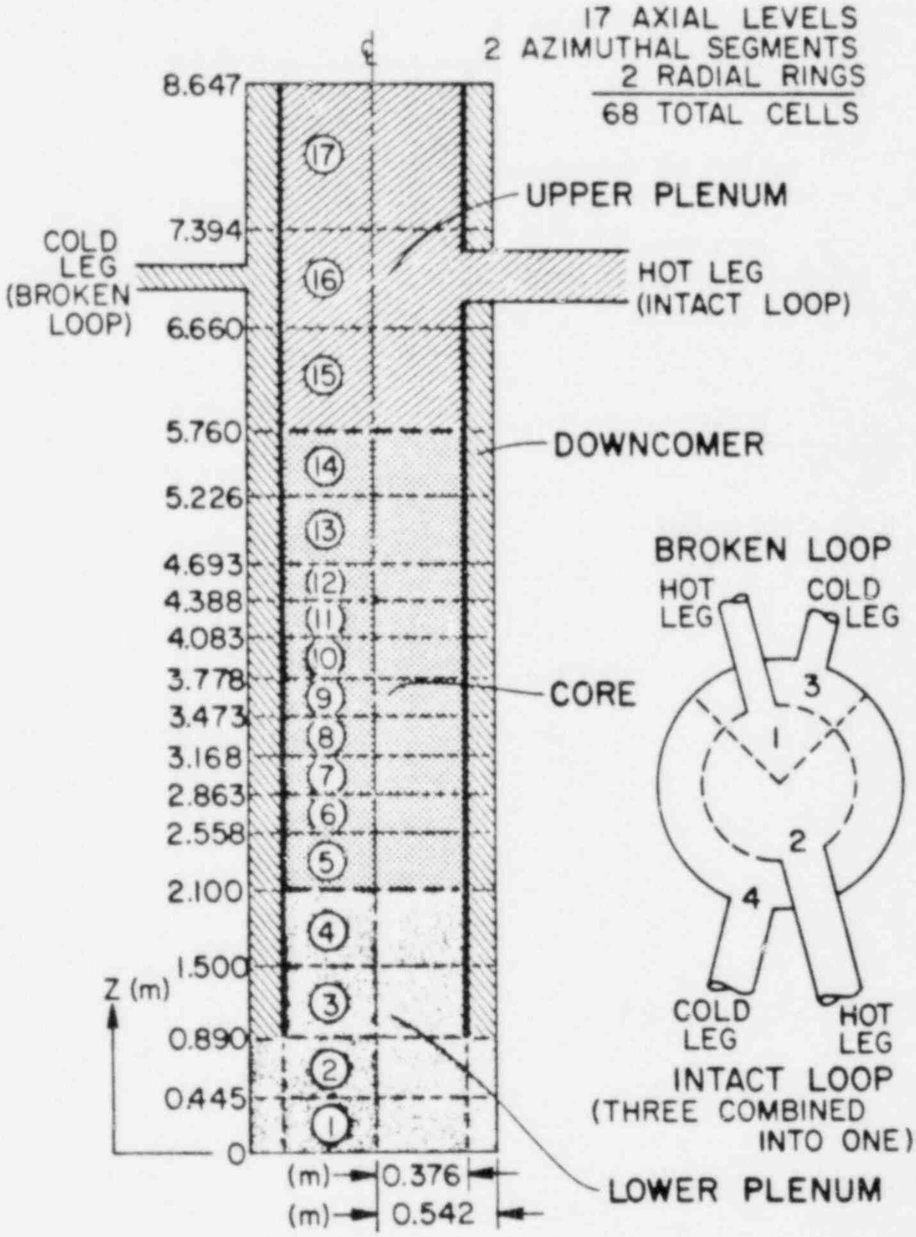
CCTF TESTS ANALYZED

- C1-1
- C1-16
- C1-11

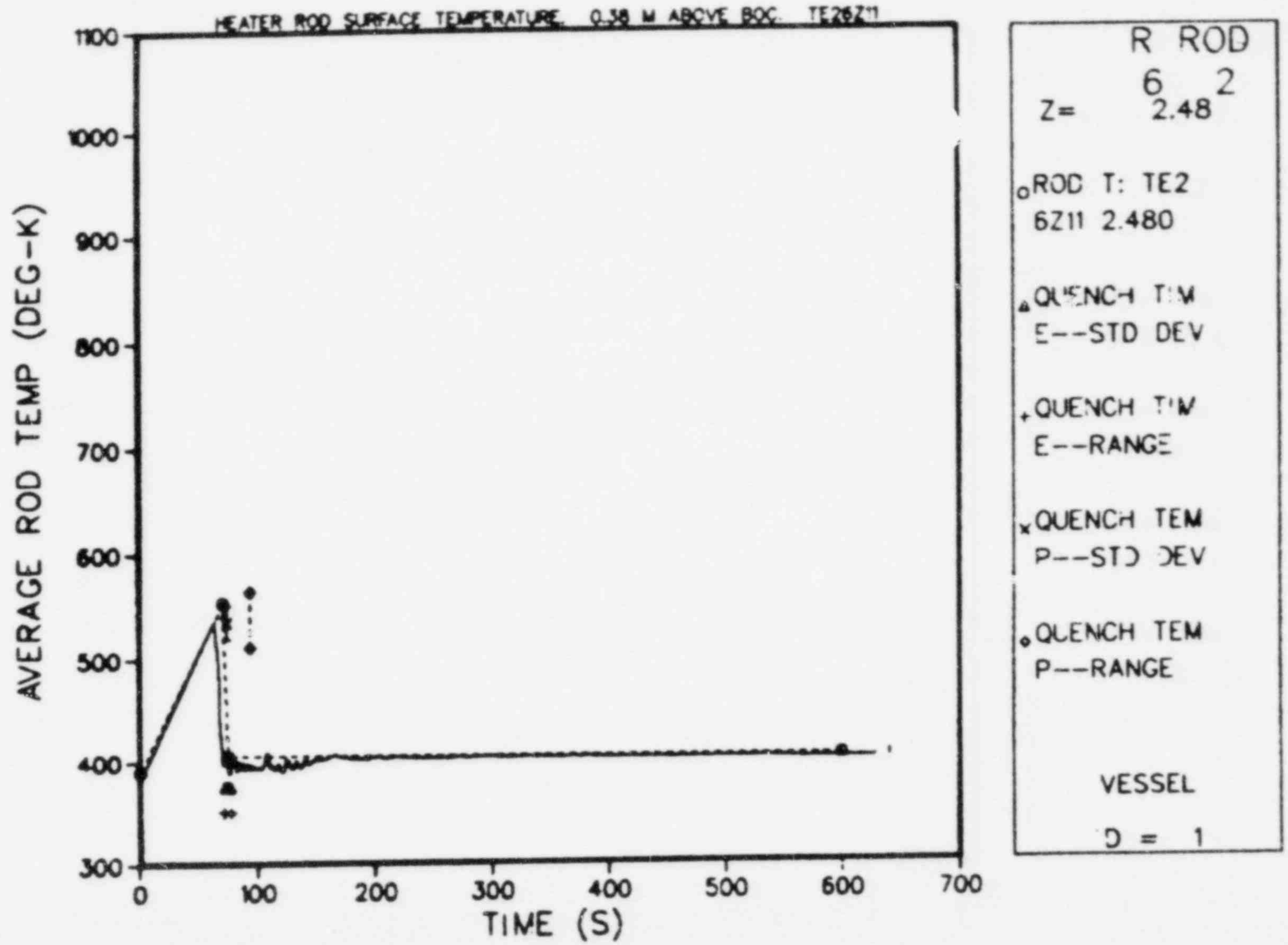
STS:10/80 INFMTQ



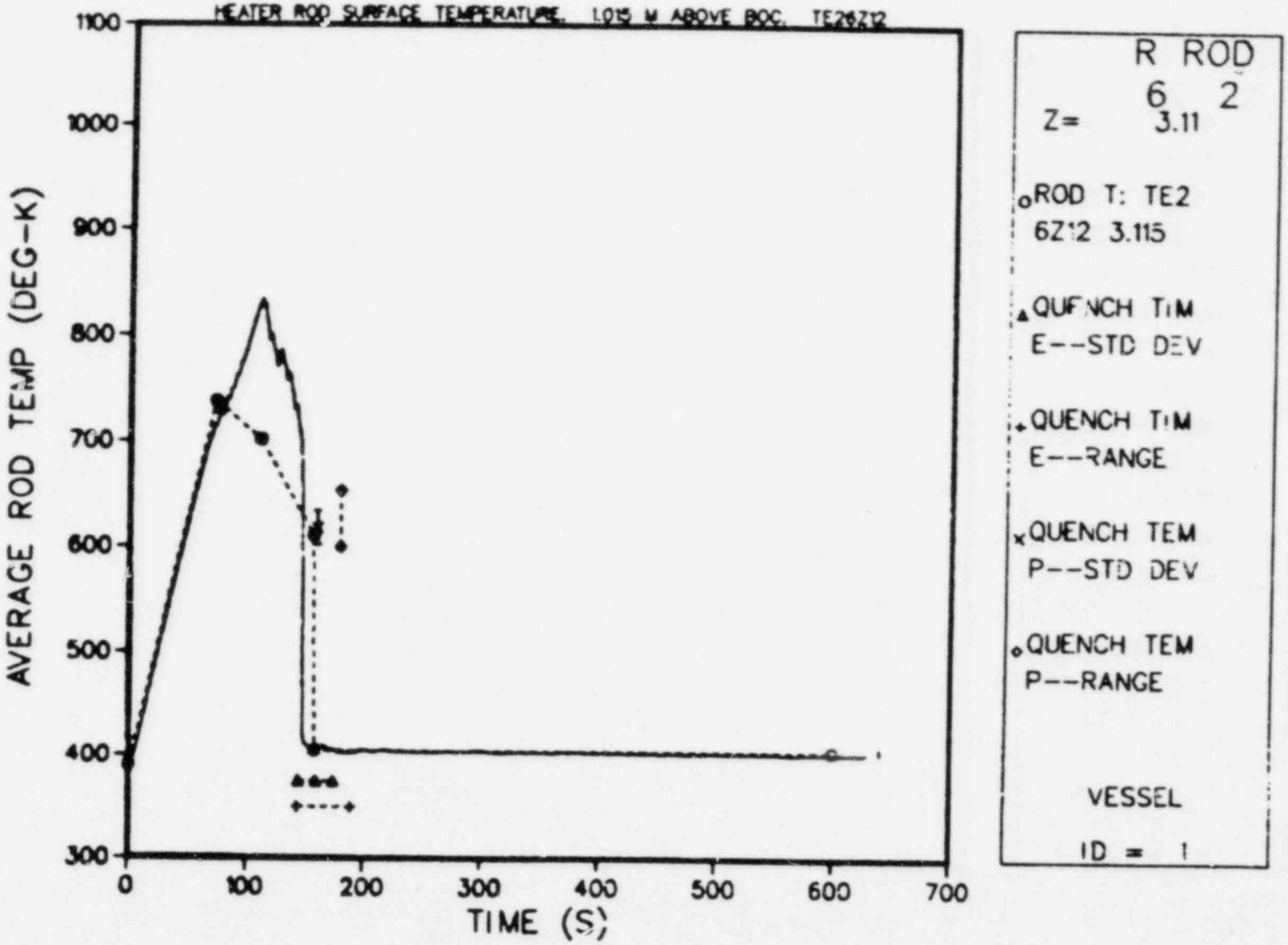
TRAC COMPONENT SCHEMATIC FOR CCTF



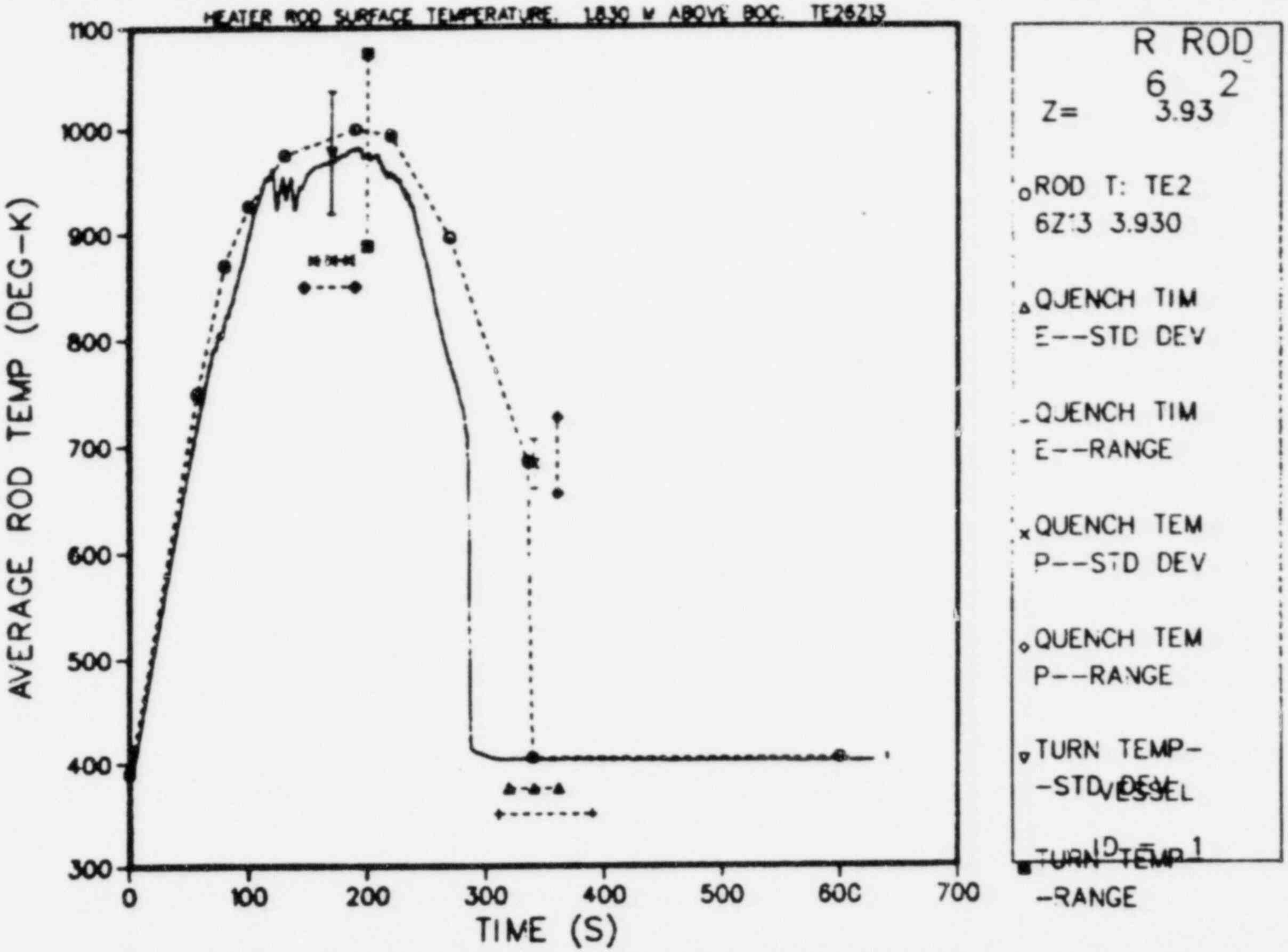
CCTF VESSEL NODING


HEATER ROD SURFACE TEMPERATURE, 0.38 M ABOVE BOC, TE26Z11
AVERAGE ROD TEMP (DEG-K)
1100
1000
900
800
700
600
500
400
300
0
100
200
300
400
500
600
700
TIME (S)
R ROD
6 2
Z= 2.48
ROD T: TE2
6Z11 2.480
QUENCH TIM
E--STD DEV
QUENCH TIM
E--RANGE
QUENCH TEM
P--STD DEV
QUENCH TEM
P--RANGE
VESSEL
D = 1

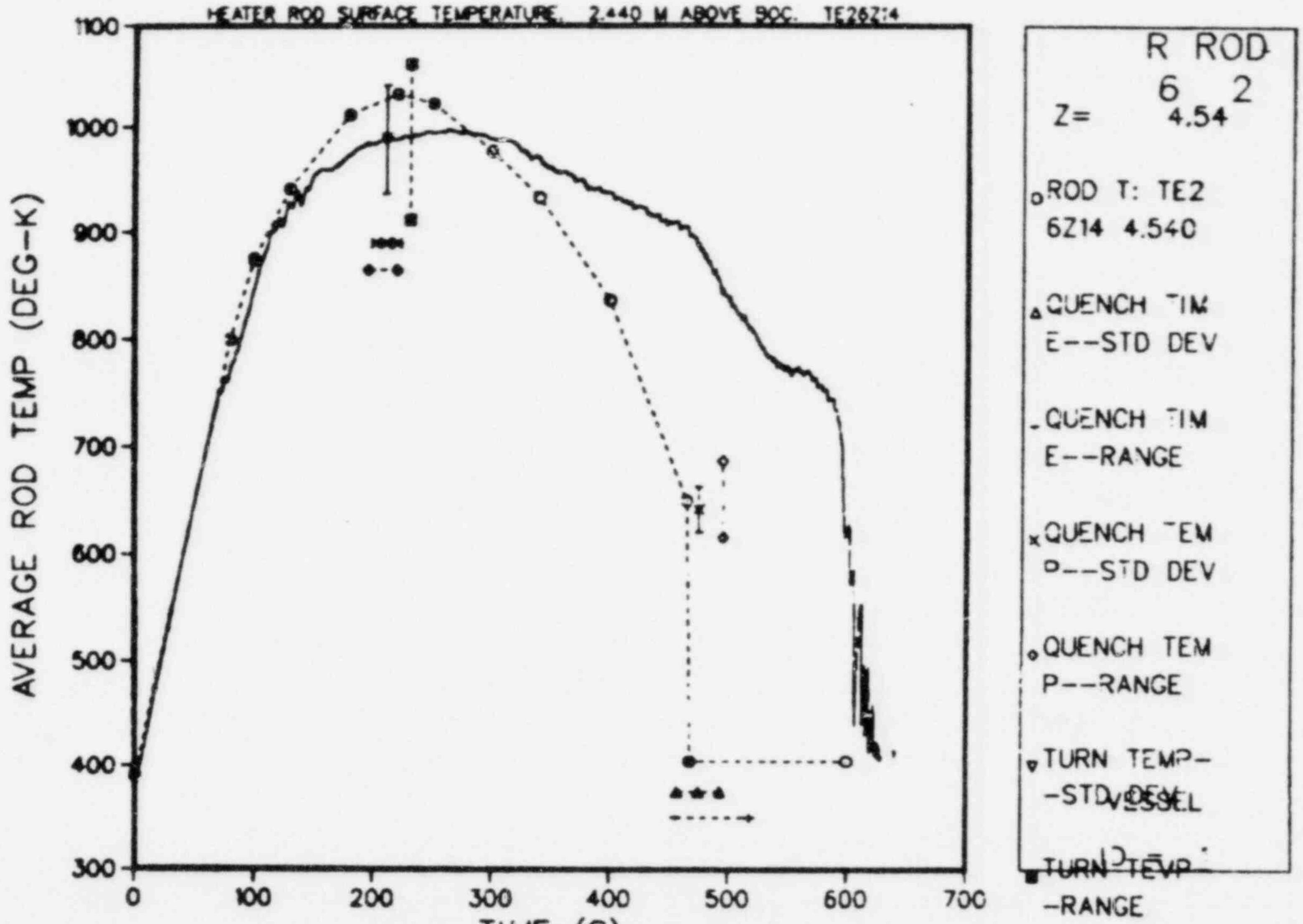
CCTF RUN 20


HEATER ROD SURFACE TEMPERATURE, 1.015 M ABOVE BOC, TE26Z12
AVERAGE ROD TEMP (DEG-K)
TIME (S)
1100
1000
900
800
700
600
500
400
300
0
100
200
300
400
500
600
700
R ROD
6 2
Z= 3.11
ROD T: TE2
6Z12 3.115
QUENCH TIM
E--STD DEV
QUENCH TIM
E--RANGE
QUENCH TEM
P--STD DEV
QUENCH TEM
P--RANGE
VESSEL
ID = 1

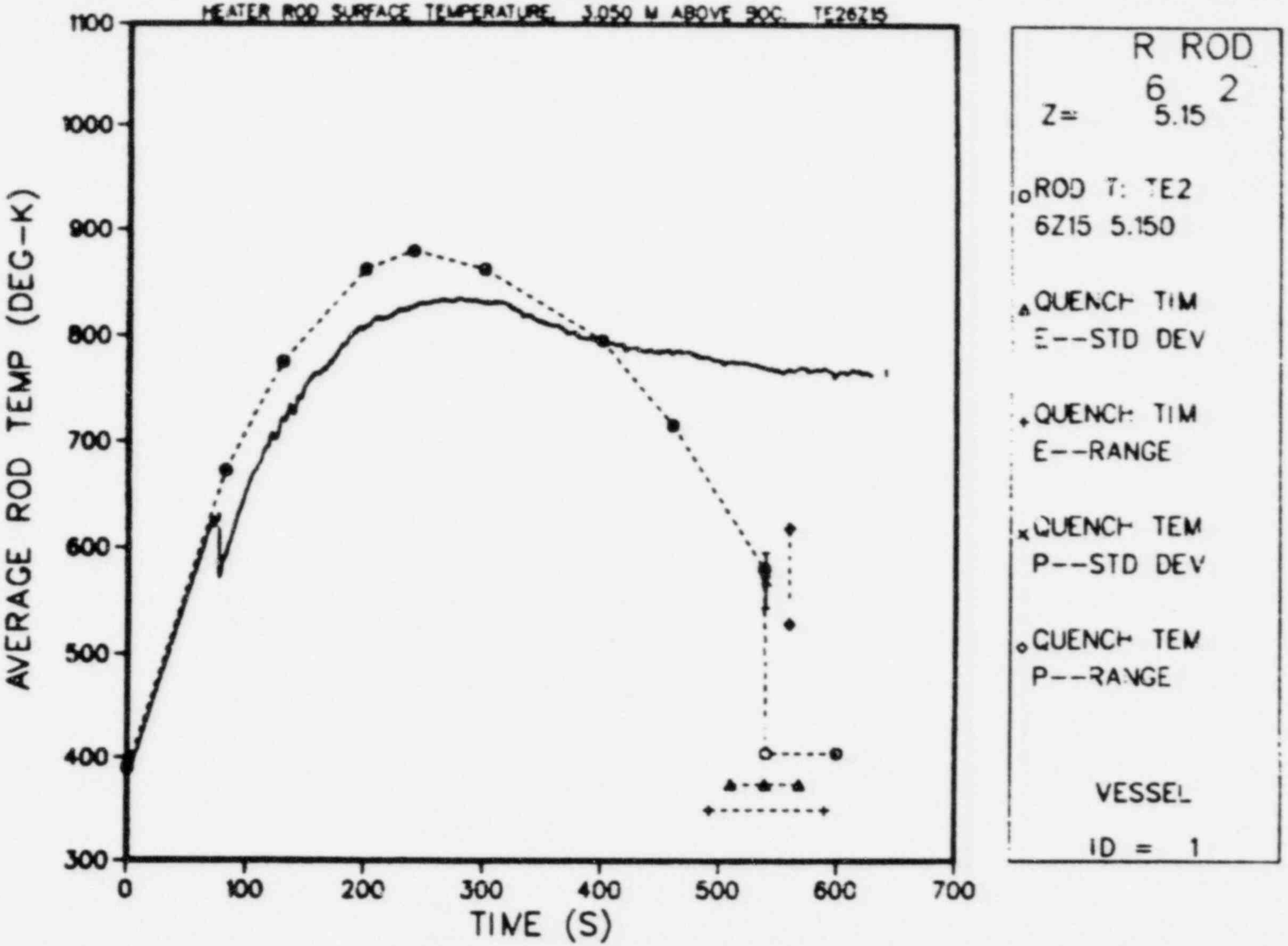
CCTF RUN 20


HEATER ROD SURFACE TEMPERATURE, 1.830 M ABOVE BOC, TE26Z13
AVERAGE ROD TEMP (DEG-K)
TIME (S)
R ROD
6 2
Z= 3.93
ROD T: TE2 6Z13 3.930
QUENCH TIME--STD DEV
QUENCH TIME--RANGE
QUENCH TEMP--STD DEV
QUENCH TEMP--RANGE
TURN TEMP--STD DEV
VESSEL
TURN TEMP--RANGE

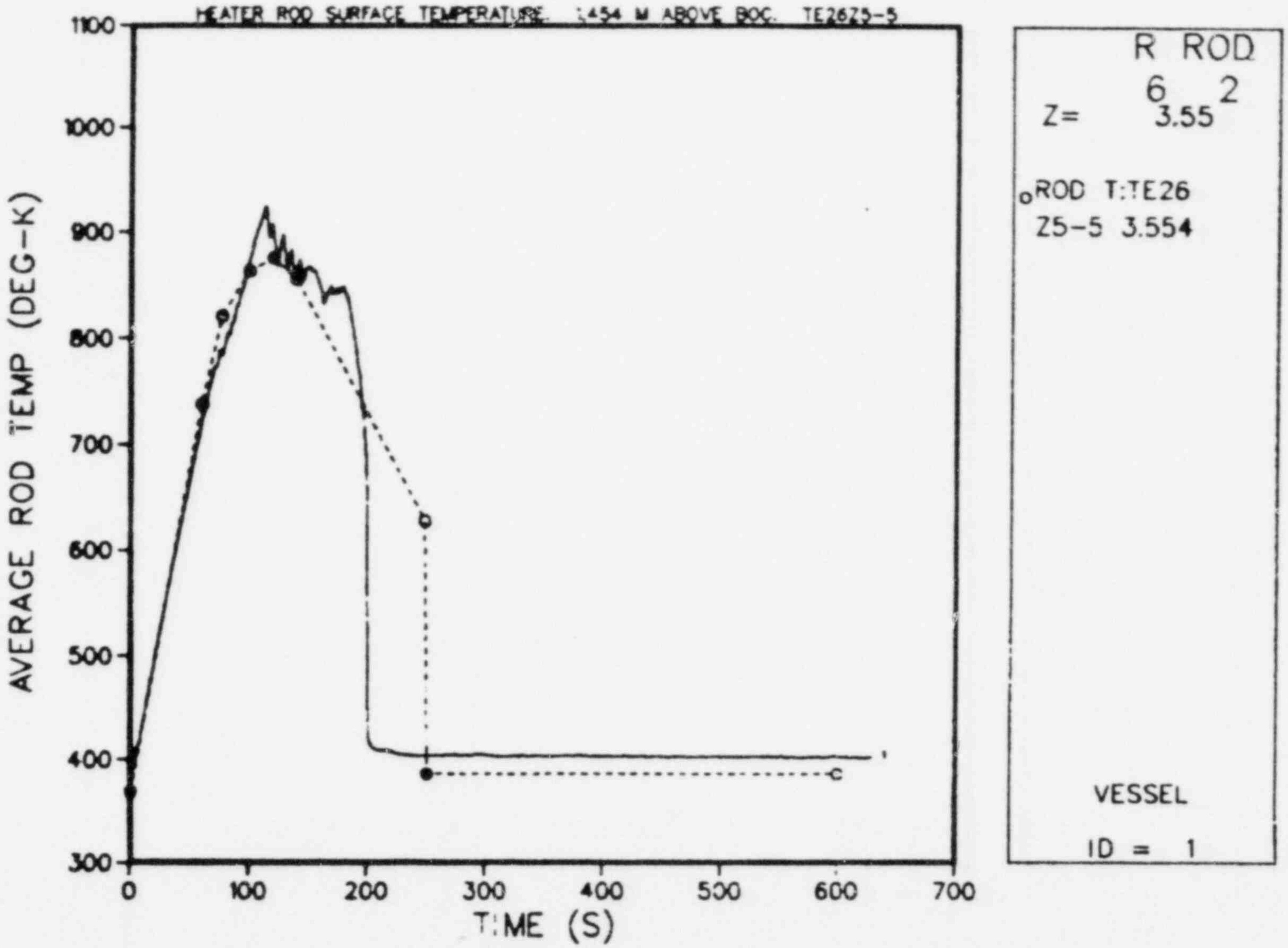
CCTF RUN 20


HEATER ROD SURFACE TEMPERATURE, 2.440 M ABOVE BOC. TE26Z14
AVERAGE ROD TEMP (DEG-K)
TIME (S)
R ROD
6 2
Z= 4.54
ROD T: TE2 6Z14 4.540
QUENCH TIME--STD DEV
QUENCH TIME--RANGE
QUENCH TEMP--STD DEV
QUENCH TEMP--RANGE
TURN TEMP--STD DEV
VESSEL
TURN TEMP--RANGE

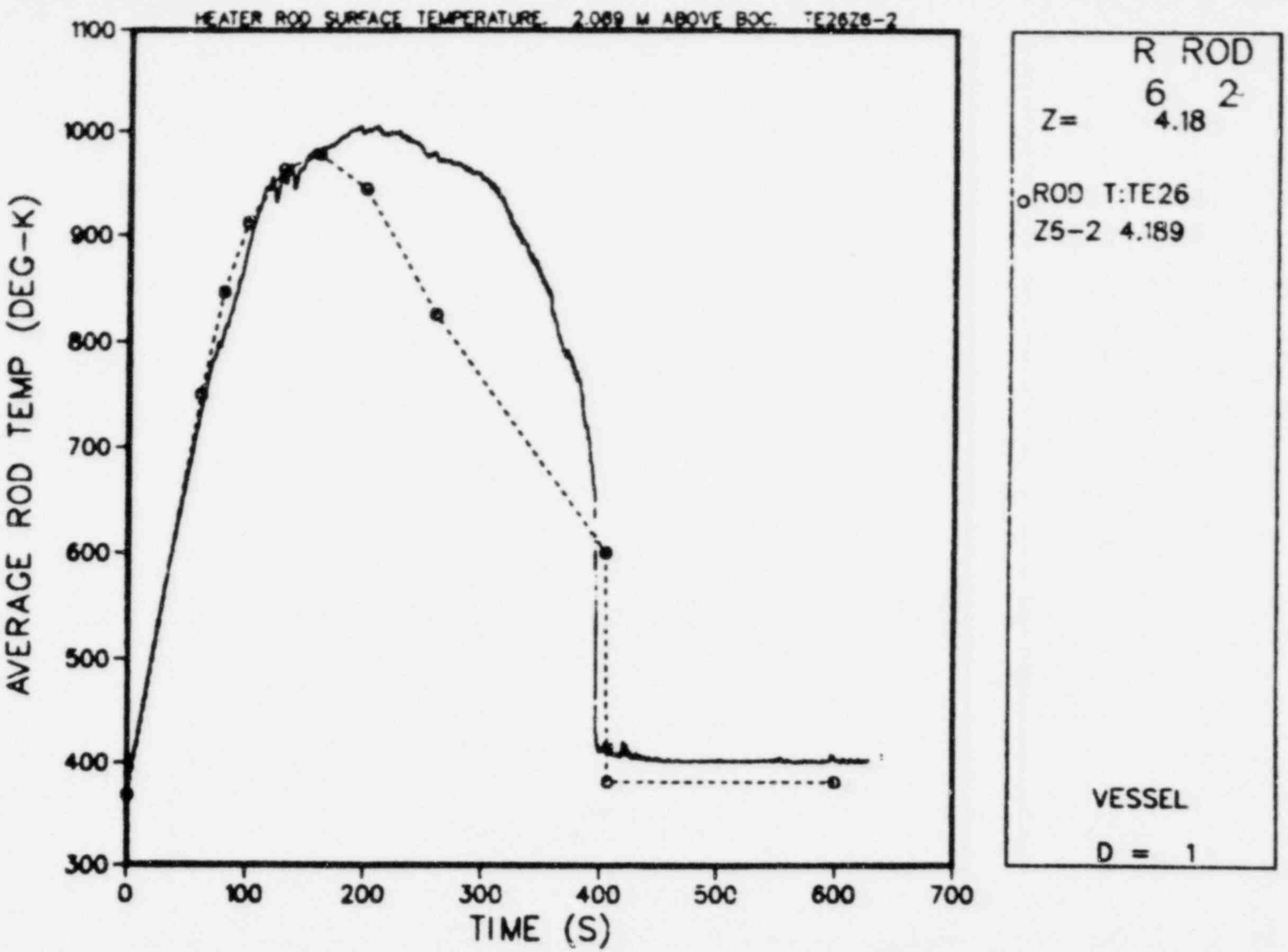
CCTF RUN 20


HEATER ROD SURFACE TEMPERATURE, 3.050 M ABOVE BOC, TE26Z15
AVERAGE ROD TEMP (DEG-K)
TIME (S)
1100
1000
900
800
700
600
500
400
300
0
100
200
300
400
500
600
700
R ROD
6 2
Z= 5.15
ROD T: TE2
6Z15 5.150
QUENCH TIM
E--STD DEV
QUENCH TIM
E--RANGE
QUENCH TEM
P--STD DEV
QUENCH TEM
P--RANGE
VESSEL
ID = 1

CCTF RUN 20

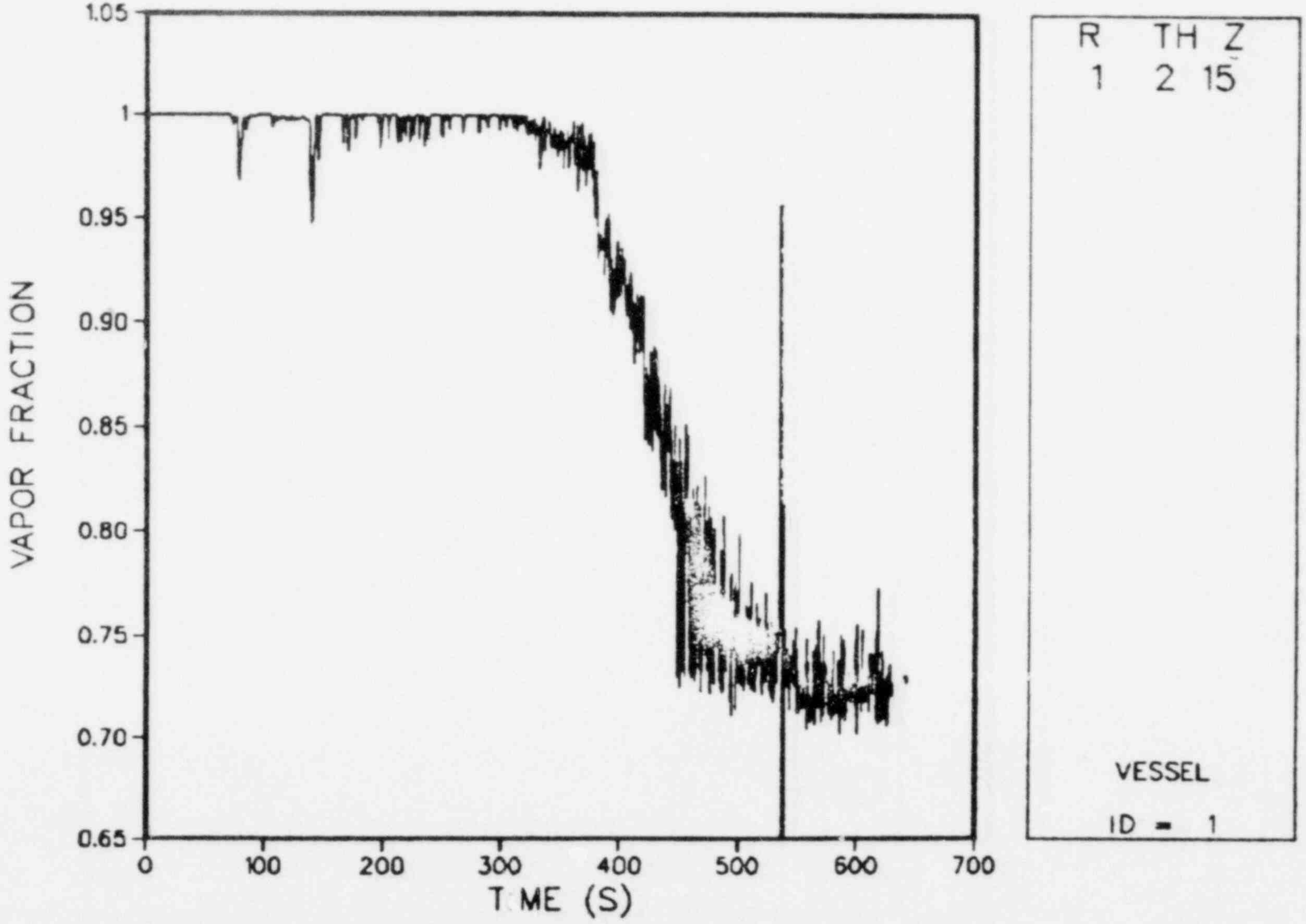

HEATER ROD SURFACE TEMPERATURE. 3.454 M ABOVE BOC. TE26Z5-5
AVERAGE ROD TEMP (DEG-K)
TIME (S)
R ROD
6 2
Z= 3.55
ROD T:TE26
Z5-5 3.554
VESSEL
ID = 1

CCTF RUN 20


HEATER ROD SURFACE TEMPERATURE. 2.089 M ABOVE BOC. TE26Z6-2
AVERAGE ROD TEMP (DEG-K)
TIME (S)
1100
1000
900
800
700
600
500
400
300
0
100
200
300
400
500
600
700
R ROD
6 2
Z= 4.18
ROD T:TE26
Z6-2 4.189
VESSEL
D = 1

CCTF RUN 20

Void Fraction at Axial Level (0.9 m high)
Above Upper Core Support Plate



CCTF RUN 20

2D/3D TRAC ANALYSIS NEAR TERM ACTIVITIES

. CCTF DATA COMPARISONS WITH ALL MEASUREMENTS; ANALYSIS OF ADDITIONAL TESTS

. SCTF REFILL/REFLOOD CALCULATIONS INCLUDING CORE BLOCKAGES

. UPTF DESIGN CALCULATIONS WITH EMPHASIS ON CORE SIMULATOR AND LOOP BEHAVIOR

. CODE "ASSESSMENT/MODEL CHANGES" FOR ANALYSIS OF 2D/3D TYPE FACILITIES

LASL

STS:10/80 INFMTA

UNION CARBIDE
ORNL

DAVID G. THOMAS
S. K. COMBS
M. E. BAGWELL
INSTRUMENT DEVELOPMENT
LOOP PROGRAM

MEASUREMENT OF TWO-PHASE FLOW AT THE CORE UPPER PLENUM INTERFACE UNDER SIMULATED REFLOOD CONDITIONS

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

GAITHERSBURG, MARYLAND
OCTOBER 27-31, 1980

MEASUREMENT OF TWO-PHASE FLOW AT THE CORE UPPER PLENUM INTERFACE UNDER SIMULATED REFLOOD CONDITIONS*

David G. Thomas, S. K. Combs and M. E. Bagwell
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

The Instrument Development Loop (IDL) Program is part of the International 2D/3D Refill and Reflood Experimental and Research Program. The principal experimental facilities in the International Program are the Slab Core Experiment in Japan and the Upper Plenum Test Facility (UPTF) in Germany. Among the objectives of the international program are: the study of the steam binding effect during reflood for various emergency core cooling combinations; the study of the reflood flow distribution (chimney effect) in a heated core; and the study of the flow hydrodynamics in the core, downcomer and upper plenum during refill and reflood.

A major problem is coupling the results to be obtained at the two major experiments. One approach is to measure the flows at the interface boundary of the two experiments and attempt to match them as closely as possible. Therefore the two major objectives of the IDL Program were to simulate expected flows at the core/upper plenum interface during the reflood phase of a postulated LOCA and to develop instrumentation systems for mass flow measurement at the core/upper plenum interface.

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*Research sponsored by Division of Reactor Safety Research, U.S. Nuclear Regulatory Commission under Interagency Agreements DOE 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

Two experimental facilities were used in these studies: a three-bundle air/water loop and a one-bundle steam/water loop. Both loops represent full-scale vertical sections of the UPTF, extending from spray nozzles to the top of the upper plenum and including a short length of dummy fuel rods, upper end boxes, core support plate and control rod guide tubes.

Since testing was completed on this program just within the last month, all results must be considered as preliminary and are subject to change in the final report.

Three flow regimes were identified and studied: (1) all liquid down, (2) counter-current flow in which gas (or vapor) goes up and liquid goes both up and down, and (3) cocurrent flow in which both gas (or vapor) and liquid go up. Instruments necessary to measure mass flow under these conditions are (1) Tie-plate drag body or equivalently ΔP across tie plate, (2) free field turbine meter located above the tie plate, (3) temperature, (4) pressure, and (5) collapsed liquid level ΔP measurement. The tie-plate drag body was unique because it utilized part of the end box as a drag body and all transducers were contained within structural members of the end box. This meant that this instrument sampled a large amount of the flow with minimum disturbance to the flow.

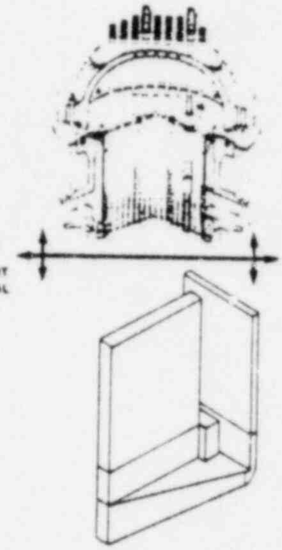
Some of the significant achievements of the IDL program include:

The tie-plate drag body was developed and tested successfully; measurement with tie-plate drag body was shown to be equivalent to the ΔP measurement; the tie-plate drag body gave a useful measurement in pure downflow situations and the combination of drag/turbine correlates with mass flow for high upflow.

THE OVERALL OBJECTIVES OF THE INTERNATIONAL 2D/3D REFILL AND REFLOOD PROGRAM

- TO STUDY THE STEAM BINDING EFFECT DURING REFLOOD FOR VARIOUS ECCS COMBINATIONS
- TO STUDY THE REFLOOD FLOW DISTRIBUTION (CHIMNEY EFFECT) IN A HEATED CORE
- TO STUDY THE FLOW HYDRODYNAMICS IN THE CORE, DOWNCOMER AND UPPER PLENUM DURING REFILL AND REFLOOD

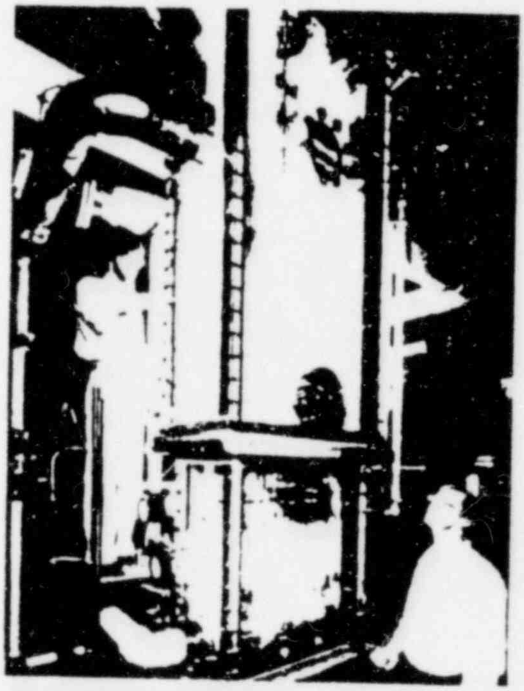
THE PRINCIPAL EXPERIMENTAL FACILITIES IN THE 2D/3D REFILL AND REFLOOD PROGRAM ARE SLAB CORE AND UPTF



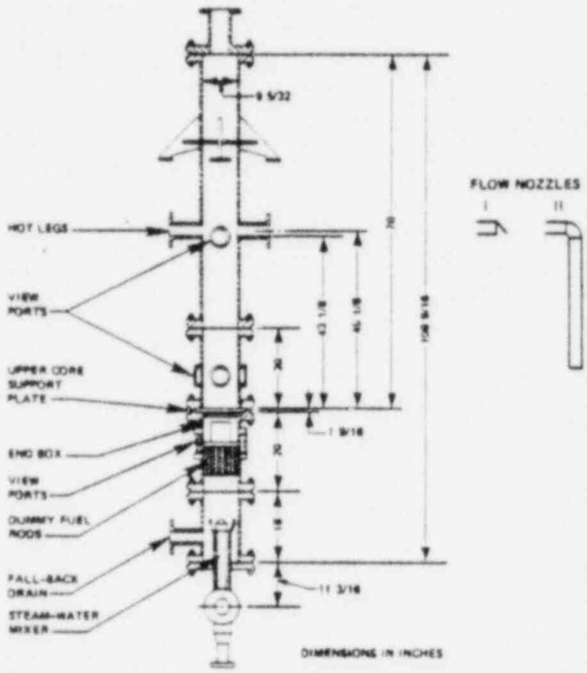
PRINCIPAL OBJECTIVES OF INSTRUMENT DEVELOPMENT LOOP (IDL) PROGRAM

- SIMULATE EXPECTED FLOWS AT THE CORE/UPPER PLENUM INTERFACE DURING THE REFLOOD PHASE OF A POSTULATED LOCA
- SCOPE POSSIBLE INSTRUMENTATION SCHEMES FOR MASS FLOW MEASUREMENT AT CORE-UCSP INTERFACE
- EVALUATE INSTRUMENT ACCURACY
- DEVELOPMENT OF MASS FLOW MEASUREMENT SYSTEM
- PHENOMENOLOGICAL STUDIES

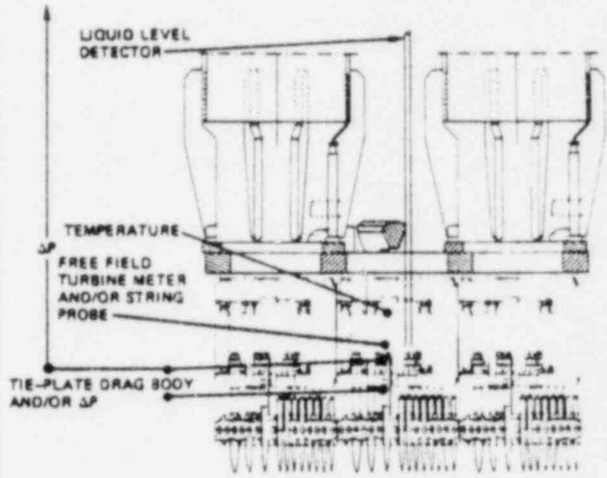
FLOWS ARE SIMULATED IN A THREE MODULE TRANSPARENT REPRESENTATION OF THE UPTF USING AIR AND WATER



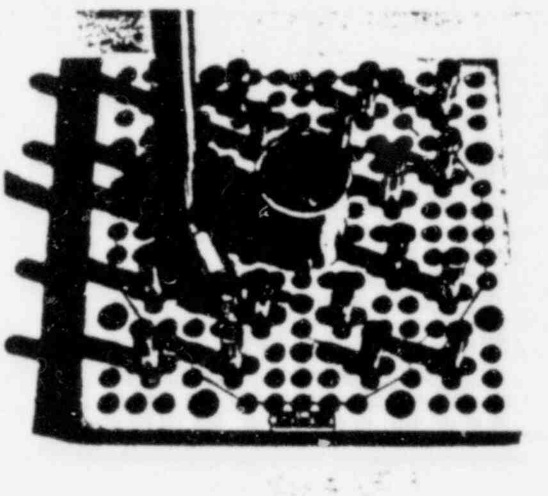
IDL STEAM WATER LOOP HAS THE CAPABILITY OF INJECTING HOT-LEG WATER IN TWO DIFFERENT CONFIGURATIONS



INSTRUMENTATION SCHEME PROPOSED BY THE UNITED STATES



THREE KEY INSTRUMENTS AT CORE UPPER PLENUM INTERFACE ARE TIE-PLATE DRAG BODY, TIE PLATE TURBINE AND TIE PLATE ΔP



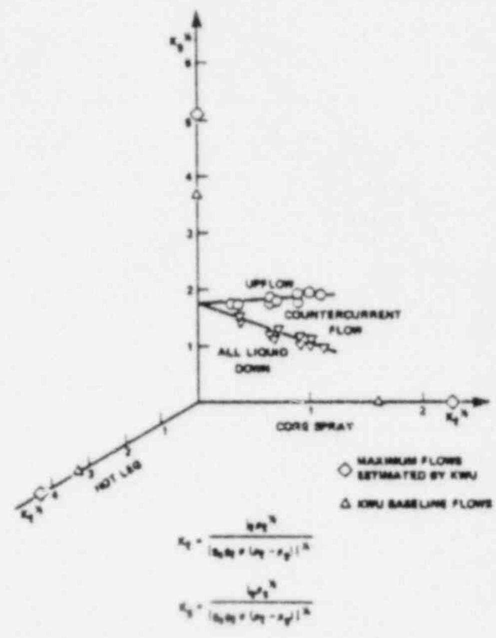
TEST MATRIX FOR IDL STEAM/WATER TESTS

TEST SERIES	PRESSURE (psia)	CORE SPRAY SUBCOOLING (°F)	CORE SPRAY GPM	HOT-LEG TEMPERATURE (°F)	HOT-LEG GPM	NUMBER OF TESTS	CUMULATIVE NUMBER OF TESTS
10	66	-	-	-	-	10	10
11	100	-	-	-	-	14	24
12	65	10	35	-	-	7	31
13	65	10	5	-	-	6	37
14	65	10	20	-	-	14	51
15	66	10	55	-	-	12	63
16	100	10	50	-	-	18	81
17	100	10	5	-	-	13	94
18A	100	10	20	-	-	18	112
19	100	10	35	-	-	16	128
20	100	10	44	-	-	14	142
21	66	10	44	-	-	10	152
22	66	10	35	-	-	12	164
23	30	10	5	-	-	8	172

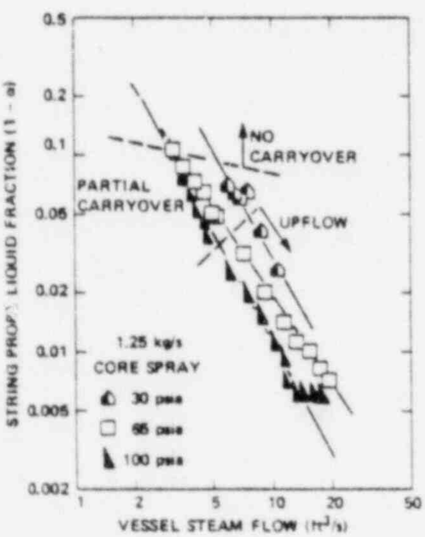
TEST MATRIX FOR IDL STEAM/WATER TESTS

TEST SERIES	PRESSURE (psia)	CORE SPRAY SUBCOOLING (°F)	CORE SPRAY GPM	HOT-LEG TEMPERATURE (°F)	HOT-LEG GPM	NUMBER OF TESTS	CUMULATIVE NUMBER OF TESTS
[illegible]	30	10	20	–	–	7	179
[illegible]	30	10	20	–	–	5	184
[illegible]	60	–	–	–	–	10	194
77	30	–	–	–	–	3	197
[illegible]	100	–	–	–	–	13	210
[illegible]	100	10	20	[illegible]	108	5	215
[illegible]	[illegible]	10	20	[illegible]	[illegible]	4	219
[illegible]	100	–	–	100	[illegible]	6	225
27	100	–	–	100	[illegible]	6	231
[illegible]	[illegible]	–	–	[illegible]	[illegible]	6	[illegible]
[illegible]	[illegible]	10	20	[illegible]	[illegible]	6	[illegible]
[illegible]	[illegible]	–	–	[illegible]	[illegible]	7	[illegible]
[illegible]	[illegible]	–	–	100	100	7	[illegible]
[illegible]	[illegible]	–	–	100	[illegible]	7	[illegible]

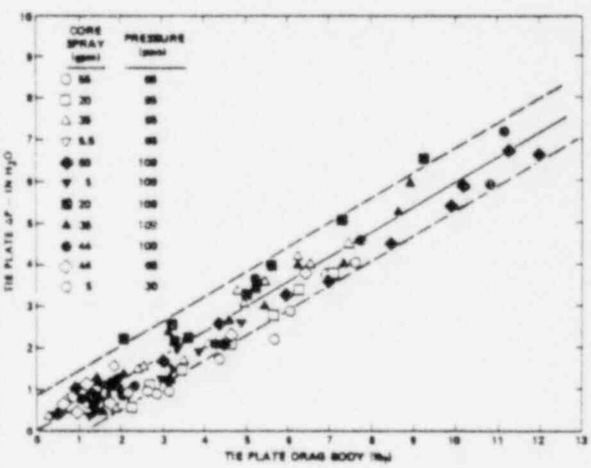
FLOW REGIMES OBSERVED IN STEAM WATER LOOP



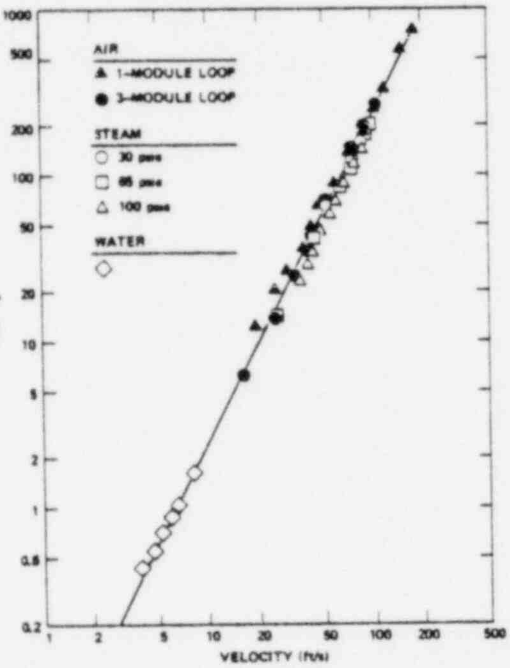
LIQUID FRACTION JUST ABOVE TIE-PLATE MEASURED WITH STRING PROBE IN STEAM/WATER LOOP



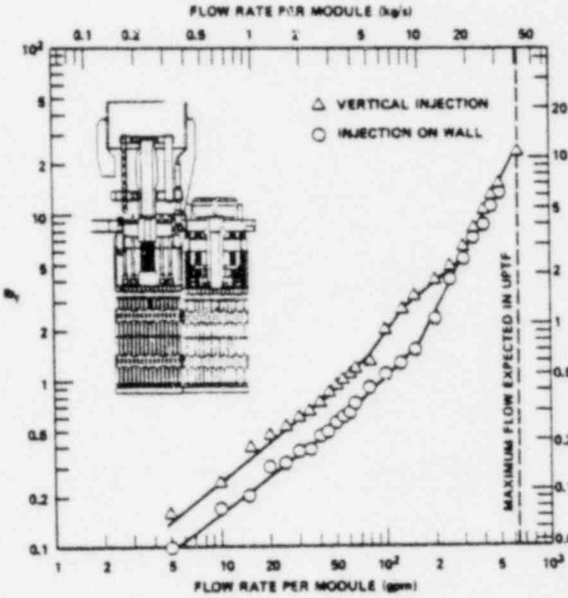
TIE-PLATE DRAG BODY AND TIE PLATE ΔP MEASUREMENT SHOW A 1:1 CORRESPONDENCE



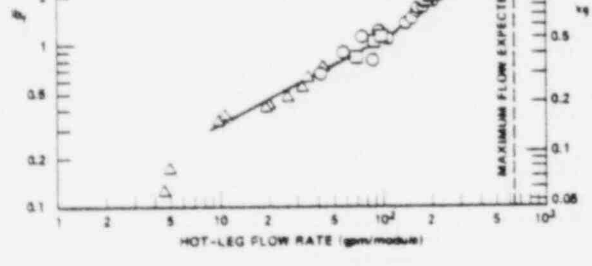
IN SINGLE PHASE FLOW, TIE-PLATE DRAG BODIES YIELDED A GOOD CALIBRATION CURVE FOR A WIDE RANGE OF FLUID DENSITIES IN BOTH 1 AND 3 MODULE LOOPS



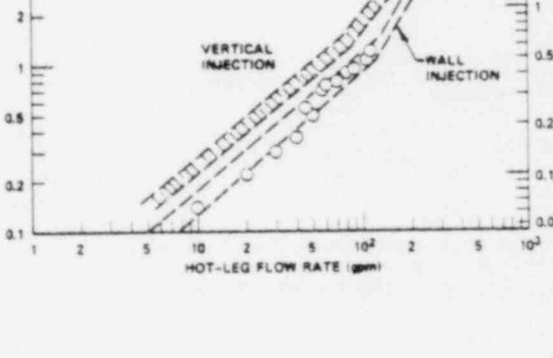
DOWNFLOW CALIBRATION OF TIE-PLATE DRAG BODY IN SINGLE MODULE LOOP



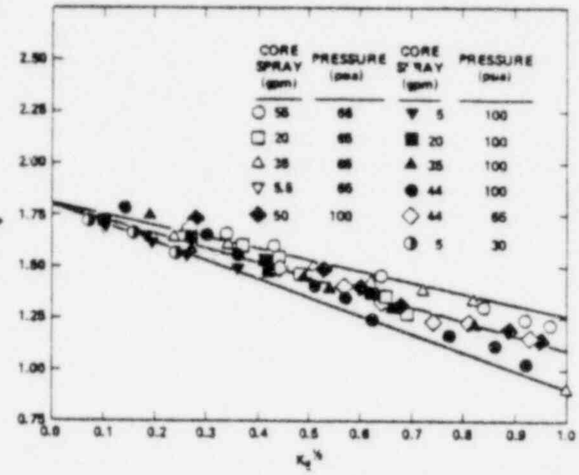
DOWNFLOW CALIBRATION OF SIMULATED TIE-PLATE DRAG BODY IN THREE MODULE AIR/WATER LOOP



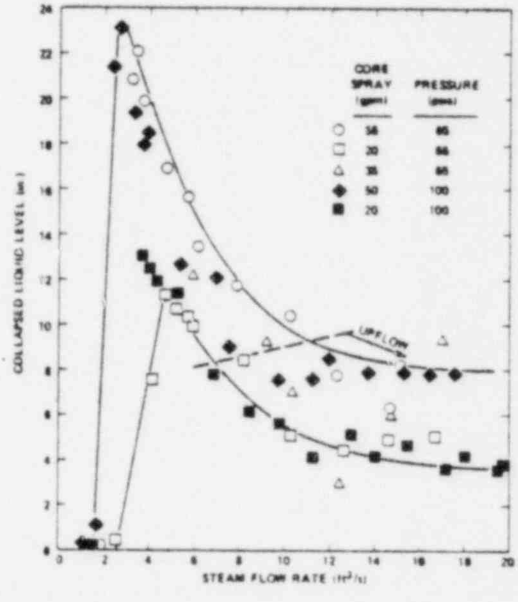
AFTER TIE-PLATE DRAG BODY WAS INSTALLED IN STEAM/WATER LOOP, DOWNFLOW RESULTS WERE IN GOOD AGREEMENT WITH PREVIOUS WALL INJECTION STUDIES



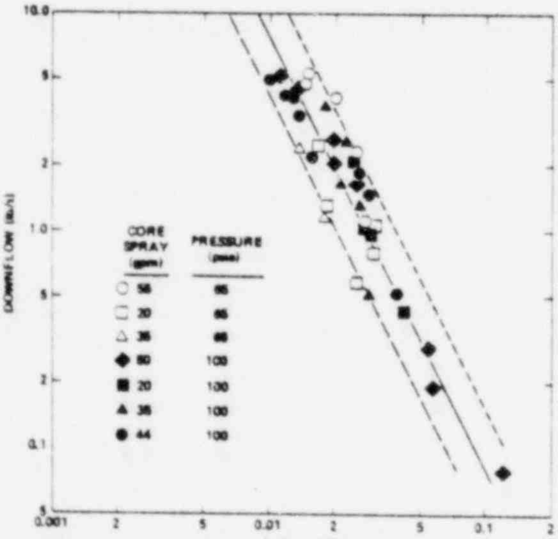
STEAM/WATER CCFL RESULTS FOR SATURATED CORE SPRAY



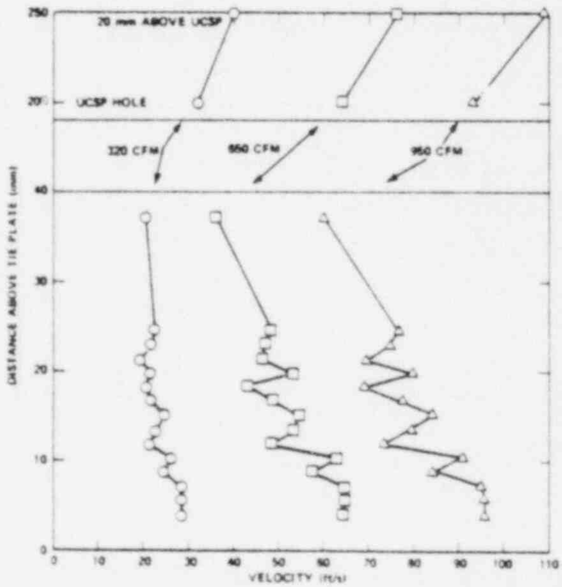
COLLAPSED LIQUID LEVEL AT TIE-PLATE IN STEAM/WATER TESTS



PRELIMINARY CORRELATION FOR DOWNFLOW IN COUNTER CURRENT FLOW REGIME



TIE-PLATE TURBINE METER READING DEPENDED ON DISTANCE METER WAS LOCATED ABOVE TIE PLATE



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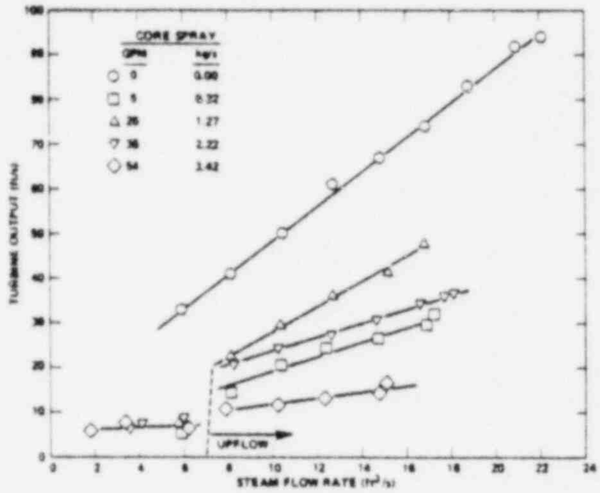
TIE-PLATE TURBINE METER OUTPUT DECREASED AS WATER CONTENT OF CORE SPRAY INCREASED



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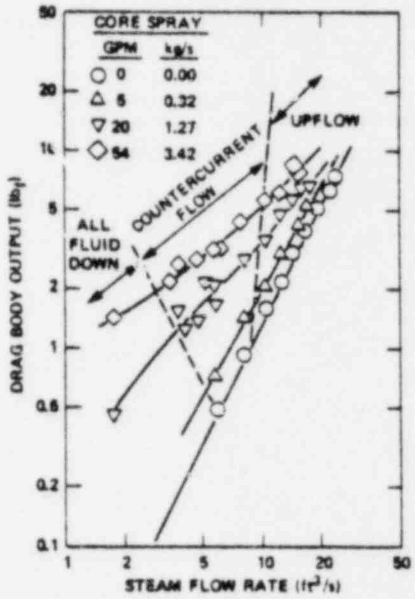
TIE-PLATE DRAG BODY OUTPUT INCREASED AS WATER CONTENT OF CORE SPRAY INCREASED



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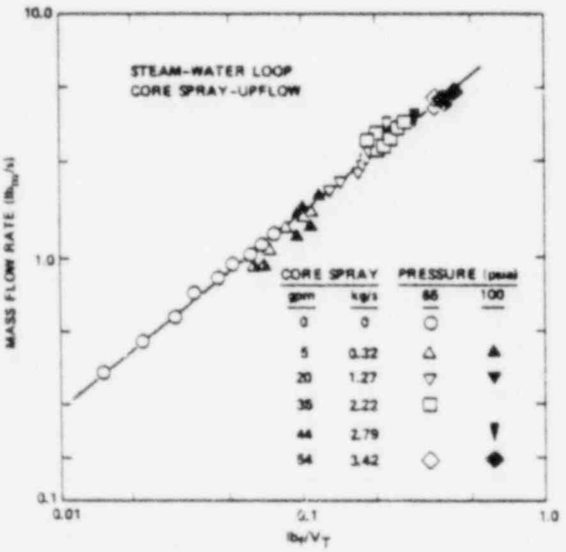
IN HIGH FLOW UPFLOW, MASS FLOW RATE IS WELL CORRELATED BY THE TIE-PLATE DRAG BODY AND THE PLATE TURBINE



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MODEL EQUATIONS FOR CALCULATING MASS FLOW RATE THROUGH TIE-PLATE $\left[G = \frac{lb_m}{(s)(module)}\right]$

P R E L I M I N A R Y

- HIGH FLOW UPFLOW

$$G = \frac{2g_c}{K_{2\phi}}\left(\frac{F_z}{A_{SDB}V_T}\right)$$

- DOWNFLOW
  - NO SEAL ON TIE PLATE

$$G = \frac{g_c F_z}{\sqrt{2g_c h}}\ \frac{A_{STP}}{A_{SDB}}$$

  - SEAL ON TIE PLATE

$$G = \frac{C_1}{\rho A_0}\sqrt{\frac{2g_c F_z}{\rho_\ell A_{SDB}}}$$

- COUNTERCURRENT FLOW

$$G = \text{LIQUID DOWN} = C_2\left(\frac{\rho_{mst}}{\rho_g F_z}\right)^2$$

P R E L I M I N A R Y

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RESULTS OF IDL PROGRAM

- TIE-PLATE DRAG BODY WAS DEVELOPED AND TESTED SUCCESSFULLY
- MEASUREMENT WITH TIE-PLATE DRAG BODY WAS SHOWN TO BE EQUIVALENT TO ΔP MEASUREMENT
- TURBINE METERS WERE SHOWN TO HAVE SERIOUS PROBLEMS IN LOW UPFLOW
- TURBINE METERS WERE USEFUL IN HIGH UPFLOW
- ΔP IS NOT A USEFUL MEASUREMENT IN SOME DOWNFLOW SITUATIONS
- TIE-PLATE DRAG BODY GIVES A USEFUL MASS FLOW MEASUREMENT IN PURE DOWNFLOW SITUATIONS
- TIE-PLATE DRAG BODY AND COLLAPSED LIQUID LEVEL MAY GIVE A USEFUL MASS FLOW MEASUREMENT IN COUNTERCURRENT FLOW
- DEMONSTRATED THAT DRAG/TURBINE CORRELATES WITH MASS FLOW FOR HIGH UPFLOW

SEPARATE EFFECTS PROGRAM

SESSION SUMMARY

OCTOBER 29, 1980

AFTERNOON SESSION - RED AUDITORIUM

PRESENTED BY:

WILLIAM D. BECKNER

U.S. NUCLEAR REGULATORY COMMISSION

U.S. NUCLEAR REGULATORY COMMISSION

EIGHT WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

GAITHERSBURG, MARYLAND

This session includes a number of papers summarizing separate effects research applicable to both pressurized water reactors (PWR) and boiling water reactors (BWR). These programs are called separate effects because the research focuses on specific phenomena or an isolated period during a postulated accident. All of these programs were originally designed to study the large-break loss of coolant accident (LOCA). However, most of the research under these programs has been reoriented toward more general applicability to other accidents such as the small break LOCA.

The Two Loop Test Apparatus (TLTA), jointly sponsored by the U.S. Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI) and the General Electric Company (GE), is a model of a BWR utilizing a single full sized electrically simulated fuel channel. While originally designed to study heat transfer and the hydraulics of the blowdown phase of the LOCA, the TLTA has evolved to test nearly the entire LOCA transient and thus the TLTA is now nearly an intergral test facility. In the last year, the TLTA was also used to simulate two small-break LOCA transients. These tests were conducted at the request of the NRC Three Mile Island Lessons Learned Task Force to evaluate the calculational methods used to specify operator actions during small-breaks and other transients. In addition, separate effects tests of heat transfer during core uncovery were conducted. Thus the TLTA program has been oriented away from the area of large-break LOCA. All planned testing in the TLTA has now been completed and the program sponsors are evaluating the possibility of upgrading the facility to better simulate small-break and other non-LOCA transients.

The BWR Refill Reflood program is also a joint NRC, EPRI, GE sponsored program studying a number of separate effects phenomena. This program is closely linked to the BWR TRAC effort discussed in the previous analysis development session. A number of new experiments are being conducted and previously obtained GE data are being released under this program for use in model development and code assessment. Direct input to the BWR TRAC program is also provided by a significant model development effort. A summary of this model development work will be presented. In addition, a presentation will be made of experiments attempting to simulate the heat from hot rods using steam injection (adiabatic injection technique). While these experiments are not directly applicable to reactors, they are of great importance to planned future experimental programs. Two large multidimensional test programs - the 30° Sector Steam Test Facility (SSTF) which is a BWR test facility under this program and the Upper Plenum Test Facility (UPTF) which is a PWR test under the International 2D/3D program - will both use adiabatic injection of steam to simulate the core heat. The results of the adiabatic injection tests will provide an indication of our ability to conduct large multidimensional experiments without the prohibitively high cost of a large heated core.

The Thermal Hydraulic Test Facility (THTF) is a separate effects PWR heat transfer facility using a new highly instrumented 8 X 8 array of electrically simulated fuel pins. Recent testing has concentrated on filling gaps in the data base for conditions during the blowdown phase of the LOCA and extending the data base to conditions applicable during other accidents such as small-breaks. Topics to be presented include transient film boiling, high pressure steam cooling and high pressure reflood.

The Full Length Emergency Core Heat Transfer - System Effects and Separate Effects Test (FLECHT-SEASET) is a joint, NRC, EPRI and Westinghouse Electric Corporation sponsored program. This program is involved in the study of reflood and related phenomena. The presentation will include findings concerning void distribution, heat transfer - including the relative components of the various heat transfer mechanisms, and the effect of flow blockages.

ROD BUNDLE HEAT TRANSFER RESEARCH FOR WATER REACTOR SAFETY*

J. D. White

Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

The Nuclear Regulatory Commission (NRC) is funding a program at Oak Ridge National Laboratory (ORNL) to study heat transfer from rod bundles in nuclear reactor accident situations. This program is called the Blowdown Heat Transfer Program. The program scope has changed significantly from emphasis on large break Loss-of-Coolant Accidents (LOCAs) to emphasis on small-break accidents and on expected reactor transients. The primary purpose of the program at present is to produce rod bundle data which can be used to provide insight into the thermal hydraulic behavior of light water reactor cores during hypothetical accidents.

Toward this end, the Thermal-Hydraulic Test Facility (THTF) at ORNL is used to supply appropriate pressure, temperature, and flow conditions to a 64-rod, 12-ft long bundle in a predetermined manner. Resultant heat fluxes, surface temperatures, and fluid conditions (quality, enthalpy, void fraction) are representative of reactor accident situations. More than 1200 instruments are monitored 20 times/s during a test.

This report discusses the results of two types of tests conducted in calendar year 1980. The first test series was conducted to investigate heat transfer rates in an uncovered bundle (like a small-break LOCA). This test series also included high pressure reflood tests (also representative of a small-break LOCA). The uncovered bundle tests had low steaming rates and effectively no entrainment. An equilibrium model seemed adequate for determination of steam enthalpy. Steam temperatures up to 1100°F were measured (after a radiation correction) by in-bundle thermocouples. At Reynolds numbers from 3000 to 12000, and at pressures from 400 to 1200 psi, steam cooling was effective – the heat transfer coefficients varied from 17 to 33 B/h ft^2°F. ORNL found that a Dittus-Boelter type correlation with fluid properties evaluated at the heated wall temperature used in conjunction with a simple radiation model agreed very well with the data. The radiation heat flux was approximately 20-30% of the total heat flux. Mixture level swell was determined to be proportional to volumetric vapor generation rate; the swell varied from 15 to 111%. Reflood data at pressures from 350 to 1100 psi were obtained. The ratio of quench front velocity to inlet flooding velocity was in the range 40-50% in the THTF experiments. (Further details can be obtained by calling the Test Engineer, T. M. Anklam, at 615-574-0772.)

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*Research sponsored by Division of Reactor Safety Research, U.S. Nuclear Regulatory Commission under Interagency Agreements DOE 40-551-75 and 40-552-75 with the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

Sixteen film boiling tests (upflow) were run in the THTF. Three of these tests were conducted under transient conditions; these tests were designed and executed so that the pressure, flow and heat flux variations were moderated sufficiently to allow the generation of good quality transient film boiling data. The remaining 13 tests were conducted under steady-state conditions. The resultant data encompasses a pressure range of 600-1800 psi, mass flux $1\text{-}8 \times 10^5$ $lb_m/h\ ft^2$, and heat flux $5 \times 10^4 - 3.2 \times 10^5$ $B/h\ ft^2$. Analysis of the data is still preliminary. Early results from the transient tests indicate that the Groeneveld 5.7 and 5.9 correlations agree with the ORNL transient data better than the Dougall-Rohsenow correlation (using RELAP4 calculated fluid conditions). Analysis of the steady-state data has begun. (For further details, contact the Test Engineer, C. B. Mullins, at 615-574-0767.)

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JIM WHITE
MANAGER
PWR–BDHT PROGRAM

ROD BUNDLE HEAT TRANSFER RESEARCH FOR WATER REACTOR SAFETY

PRESENTED TO

EIGHTH WATER REACTOR SAFETY RESEARCH

INFORMATION MEETING

OCTOBER 29, 1980

GAITHERSBURG, MARYLAND

UNION CARBIDE ORNL

THIS DISCUSSION WILL DESCRIBE:

- NEED FOR HEAT TRANSFER INFORMATION
- ORNL PROGRAM TO PROVIDE THIS INFORMATION
- BUNDLE UNCOVERY TEST RESULTS
- FILM BOILING TEST RESULTS

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IMPORTANT CURRENT DATA NEEDS HAVE BEEN IDENTIFIED AND DOCUMENTED

- TYPES OF ACCIDENTS
  - SMALL BREAK LOCA, LOCKED ROTOR, ROD EJECTION, LOSS OF AC/DC POWER, LOCA
- HEAT TRANSFER MODE/FLOW REGIME
  - FILM BOILING, TRANSITION BOILING, LEVEL SWELL, REWET, CONVECTION AND RADIATION TO STEAM

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THE PROGRAM HAS THREE OBJECTIVES

- PRODUCE DATA NEEDED TO ASSESS THE APPLICABILITY OF CURRENTLY USED CORRELATIONS
- SUGGEST NEW OR MODIFIED CORRELATIONS IF REQUIRED
- BENCHMARK THERMAL-HYDRAULIC CODES

BLOWDOWN HEAT TRANSFER PROGRAM

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EMPHASIS IS ON CONFIRMATION OF ANALYTICAL TOOLS USED TO PREDICT THE BEHAVIOR OF LWR CORES

- NUCLEAR APPLICATIONS?
- FULL RANGE OF CONDITIONS?
- STEADY-STATE CORRELATIONS?

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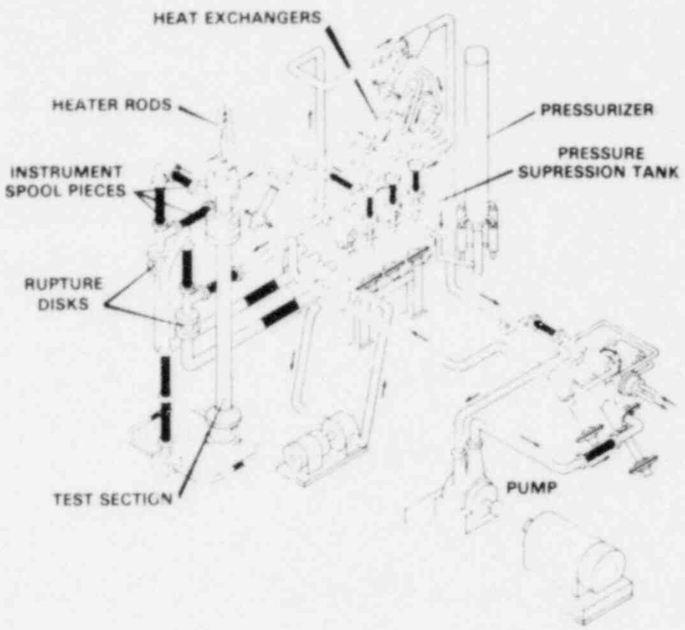
EXPERIMENTS ARE RUN IN THE THTF



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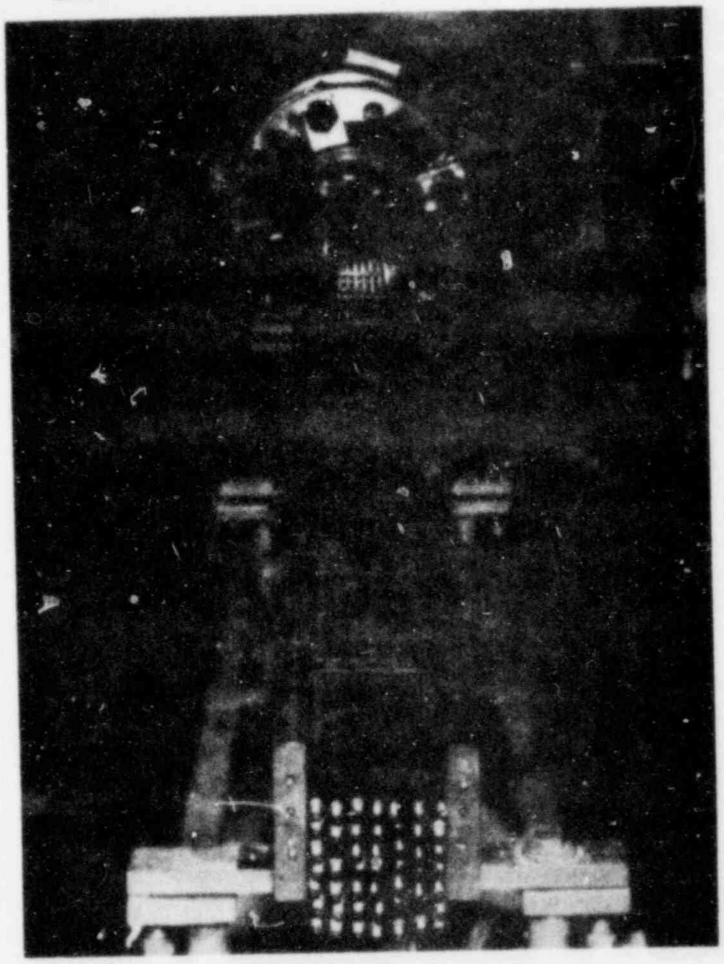
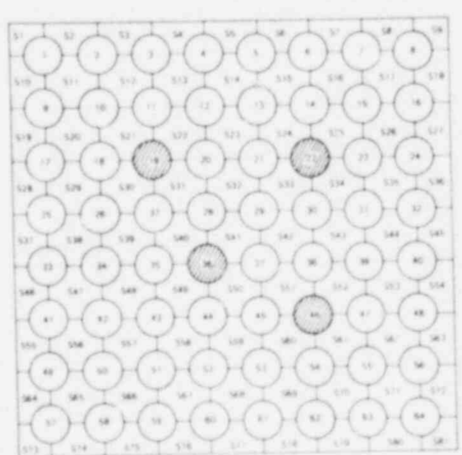
LATEST TEST BUNDLE HAS IMPROVED PERFORMANCE FEATURES

- STATE OF THE ART MEASUREMENTS OF T_s, φ_s
- ROD BUNDLE PROTOTYPIC OF CURRENT LWR DESIGNS
- POWER CAPABILITY
  - INDIVIDUAL ROD 0–150 kW
  - BUNDLE 0–9.0 MW
- CAPABILITY TO REPLACE INDIVIDUAL RODS IN-SITU
- INSTRUMENTATION FOR LOCAL FLUID CONDITIONS MEASUREMENTS



BUNDLE 3 IS AN 8 X 8 ARRAY REPRESENTATION
OF A 17 X 17 PWR SUBASSEMBLY



INACTIVE RODS

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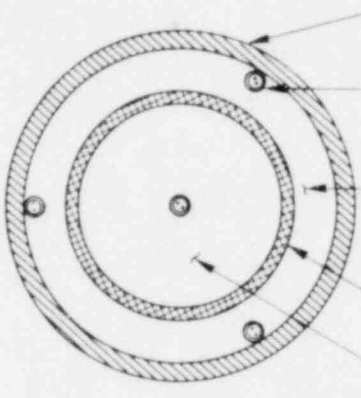
THE FUEL PIN SIMULATORS WERE DESIGNED TO PROVIDE ACCURATE DETERMINATIONS OF SURFACE HEAT FLUX AND TEMPERATURE



HEAT ROD CROSS SECTION – 0.374-in. DIAM

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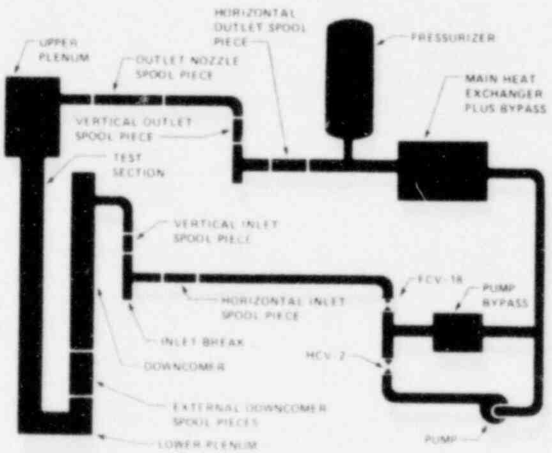
BUNDLE FLUID CONDITIONS MUST BE KNOWN TO ASSESS CORRELATIONS AND MODELS

- TRANSIENT THERMAL HYDRAULICS CODE WITH HYDRAULIC AND ENERGY BOUNDARY CONDITIONS
- BUNDLE FLUID CONDITIONS ARE CALCULATED USING ORNL MODIFIED VERSION OF RELAP4 (RLPSFLUX)



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THREE TYPES OF TESTS WERE CONDUCTED THIS YEAR

• BUNDLE BOILOFF/REFLOOD	25 TESTS
• FILM BOILING	
TRANSIENT	3 TESTS
STEADY-STATE	13 TESTS
• LARGE BREAK LOCA	1 TEST

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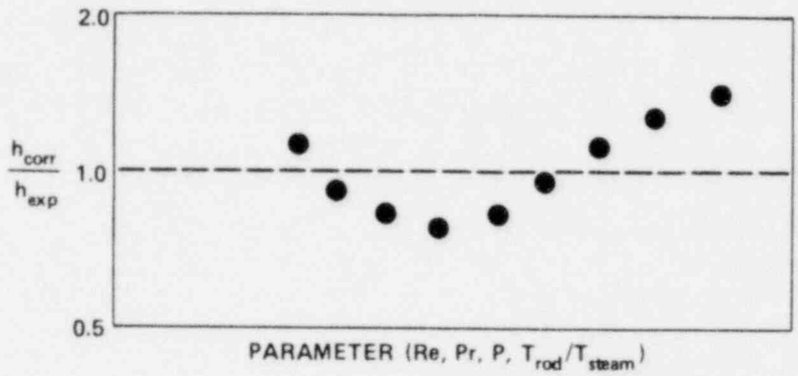
THE PROGRAM HAS 3 TYPES OF OUTPUTS

- EXPERIMENTAL DATA
- CALCULATED RESULTS
- ANALYSIS OF RESULTS

ORNL PROVIDES COMPARISONS OF HEAT TRANSFER CORRELATIONS WITH DATA FOR EACH TEST



- PRELIMINARY WITHIN 60 DAYS
- FINAL DATA WITHIN FY-198[illegible]

SMALL BREAK HEAT TRANSFER TESTS

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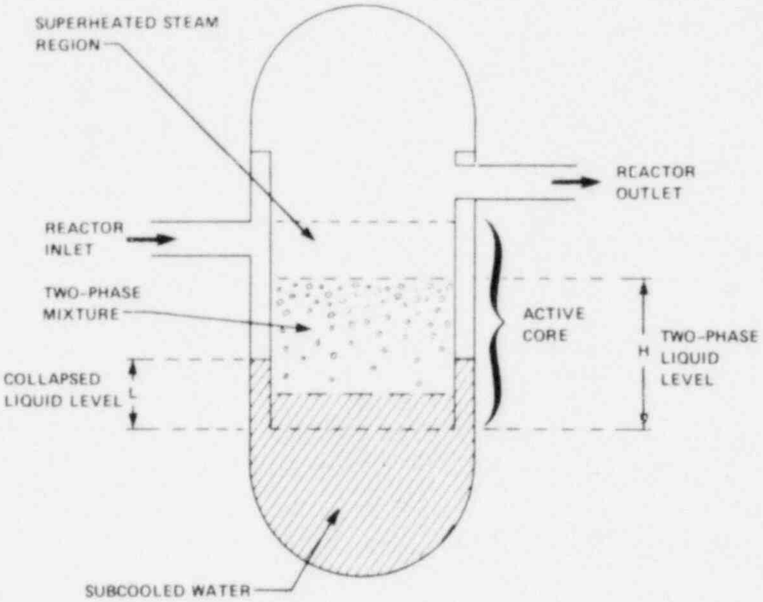
DURING CERTAIN POSTULATED PWR SMALL BREAK LOCA SCENARIOS, THE REACTOR CORE UNCOVERS AND THEN RECOVERS



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QUESTIONS PERTINENT TO THE UNDERSTANDING OF CORE THERMAL HYDRAULIC BEHAVIOR DURING SMALL BREAK LOCAs

- STEAM ENTHALPY RISE?
- CLADDING TO STEAM HEAT TRANSFER?
- TWO-PHASE MIXTURE LEVEL SWELL?
- QUENCH BEHAVIOR DURING SMALL BREAK REFLOOD?

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WE HAVE RUN 25 TESTS INVESTIGATING HEAT TRANSFER IN BOILOFF/REFLOOD SITUATIONS

- PRESSURES: 350–1700 psia
- POWER LEVEL: 0.1–0.6 kW/ft
- STEAM TEMPERATURES: 800–1500°F
- DEPTH OF UNCOVERY: 8.5 ft
- FLOODING RATE: 1.1–9.0 in./s

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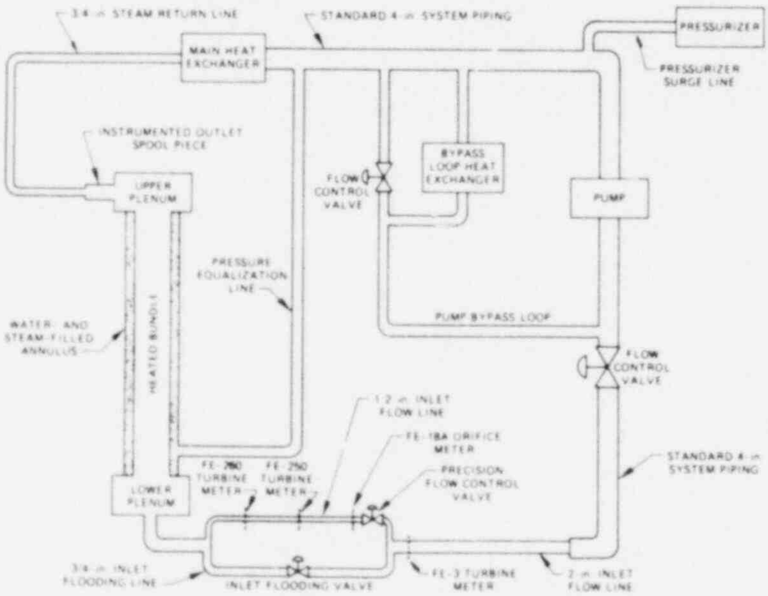
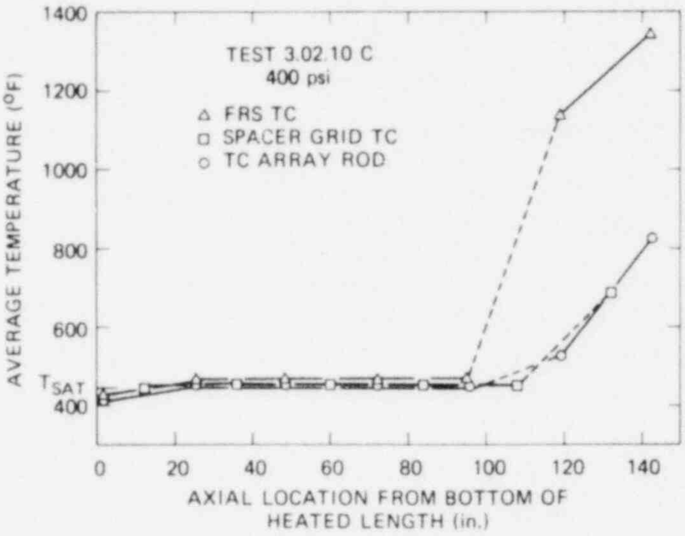
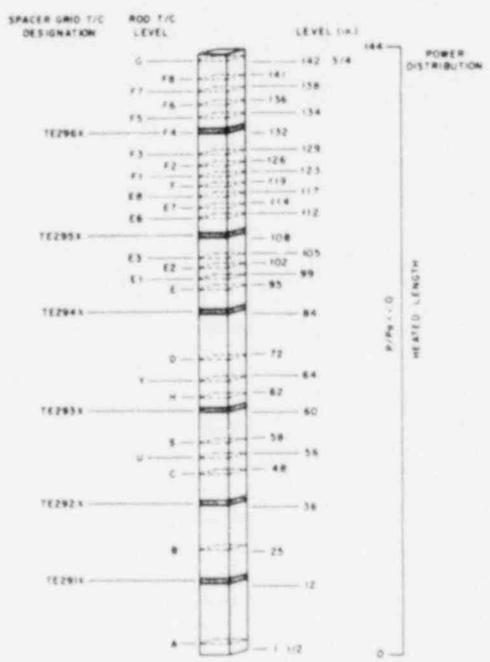
THE THTF IN BUNDLE BOILOFF/REFLOOD CONFIGURATION



IN THE FIRST SET OF TESTS, THE BUNDLE WAS UNCOVERED TO A DEPTH BELOW THE TOP TWO THERMOCOUPLE LEVELS



NEW FUEL PIN SIMULATORS, NEW DP CELLS, AND A NEW IN-BUNDLE DENSITOMETER WERE ADDED TO THE THTF



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BUNDLE UNCOVERY TEST RESULTS WERE CALCULATED DIRECTLY FROM EXPERIMENTAL MEASUREMENTS

- NO LARGE COMPUTER CODE WAS REQUIRED
- STEAM TEMPERATURES WERE DEDUCED FROM LOCAL FLUID TEMPERATURE MEASUREMENTS
- ROD HEAT FLUXES AND SURFACE TEMPERATURES DID NOT REQUIRE AN INVERSE CODE
- INLET AND OUTLET MASS FLUXES WERE EQUAL WITHIN MEASUREMENT UNCERTAINTY (5%)
- THERMODYNAMIC STATE OF STEAM DEDUCED FROM MEASURED PRESSURE AND TEMPERATURE (NO ENTRAINMENT)
- RADIATION HEAT TRANSFER CALCULATIONS BASED ON LITERATURE VALUES FOR STEAM (Q_{RAD} TO STEAM ≈ 20–30% TOTAL)

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HEAT LOSSES DURING THE FIRST SET OF TESTS VARIED CONSIDERABLY

TEST	HEAT LOSS (%)
C	5.1%
D	4.5%
E	1.2%
F	22%
G	8.1%
H	6.1%

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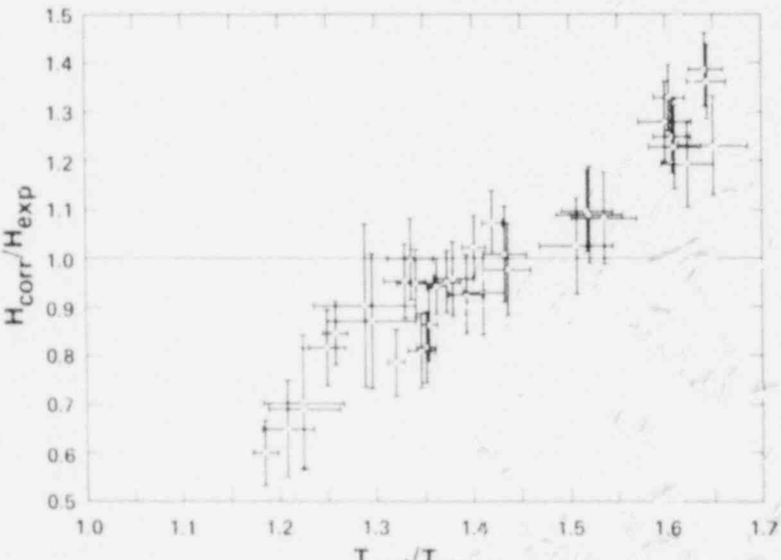
CLADDING TO STEAM HEAT TRANSFER

- TOTAL HEAT TRANSFER COEFFICIENT VARIED FROM 17-33 B/hr ft^2 ^{o}F
- EXISTING MODELS PREDICTED h's WITHIN 40% OF EXPERIMENTAL h's
- EXISTING MODELS DID NOT PREDICT SOME OF THE TRENDS IN THE DATA
- ORNL HAS SUGGESTED SOME CHANGES IN THE WAY THE MODELS ARE USED

UNION CARBIDE ORNL

EXISTING HEAT TRANSFER MODELS DO NOT PREDICT ALL TRENDS OBSERVED IN DATA



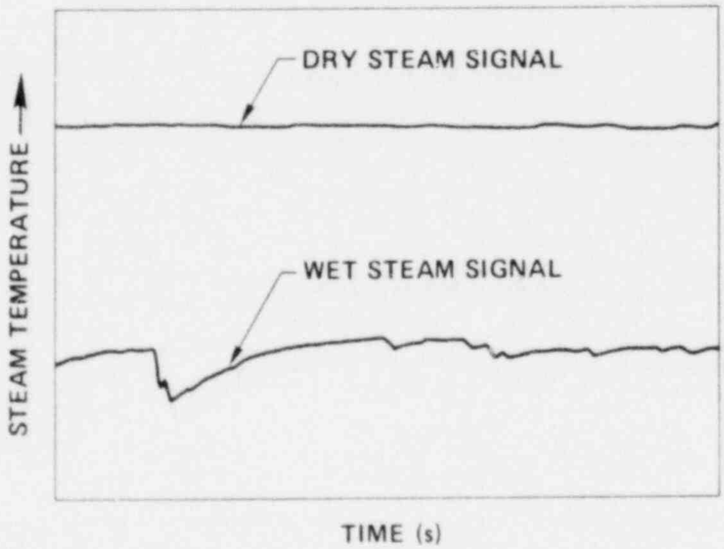
UNION CARBIDE ORNL

AN EQUILIBRIUM MODEL WAS ADEQUATE FOR THE ORNL UNCOVERED BUNDLE DATA

- COMPARISONS OF EXPERIMENTAL STEAM TEMPERATURE WITH CALCULATED STEAM TEMPERATURES
- CALCULATED STEAM VELOCITIES WERE LOW
- LACK OF ENTRAINMENT WAS DEDUCED FROM FLUID THERMOCOUPLE BEHAVIOR

UNION CARBIDE ORNL

FLUID THERMOCOUPLE OUTPUTS WERE USED TO DEDUCE LACK OF ENTRAINMENT



UNION CARBIDE
ORNL

ORNL HAS SUGGESTED A HEAT TRANSFER MODEL THAT PROVIDES GOOD AGREEMENT WITH DATA

- $h_{TOTAL} = h_{CONV} + h_{RAD}$
- $h_{CONV} = 0.021 \dfrac{K_W}{D_W} Re_W^{0.8} Pr_W^{0.4}$
- ORNL RADIATION MODEL

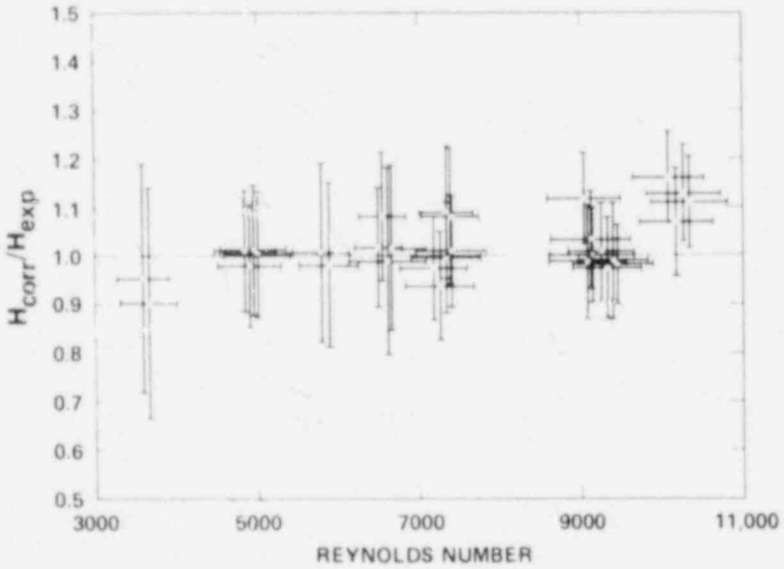
  GREY ISOTHERMAL VAPOR

  RADIATION PROPERTIES OF STEAM DEDUCED FROM LITERATURE

  RADIATION TO COLD RODS

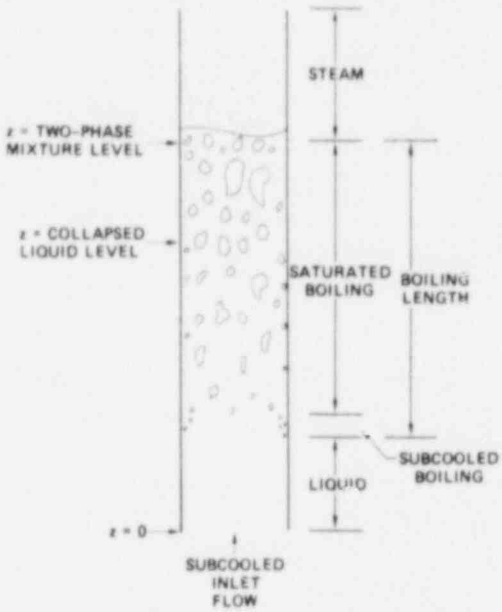
UNION CARBIDE
ORNL

ORNL HEAT TRANSFER MODEL PROVIDES GOOD AGREEMENT WITH DATA



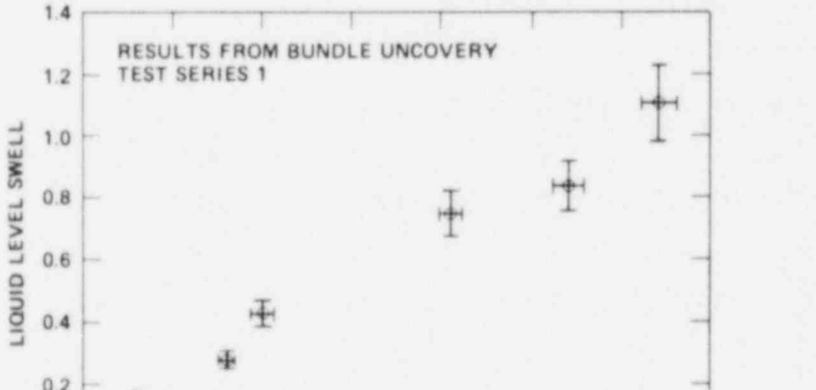
UNION CARBIDE ORNL

BUNDLE UNCOVERY TESTS ARE TYPICAL OF LOW FLOW BOILING IN A VERTICAL CHANNEL



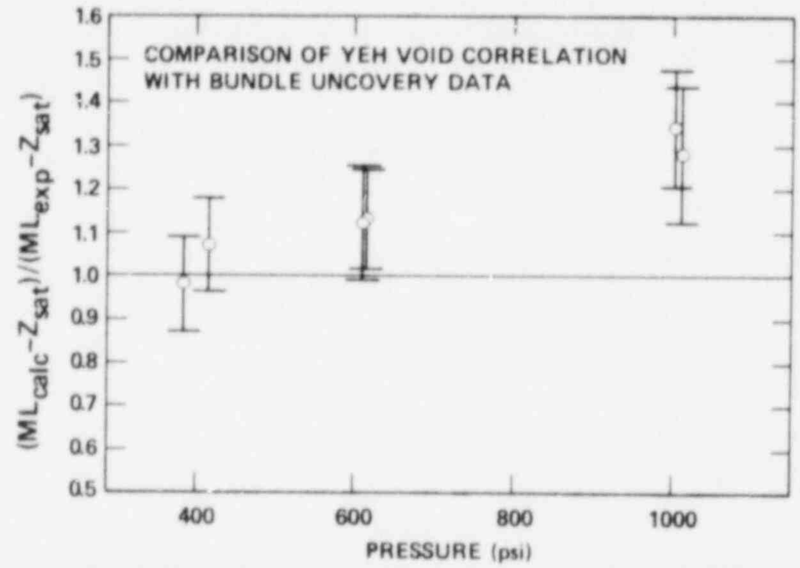
UNION CARBIDE ORNL

LIQUID LEVEL SWELL WAS ROUGHLY PROPORTIONAL TO VOLUMETRIC VAPOR GENERATION RATE



UNION CARBIDE ORNL

THE YEH CORRELATION SEEMS TO OVERPREDICT LEVEL SWELL AT HIGHER PRESSURES



UNION CARBIDE ORNL

REFLOOD TESTS

- 8 X 8 BUNDLE 12 ft LONG
- FLAT POWER PROFILE
- PRESSURES 350–1100 psi
- FLOODING RATE 1.1–9.0 in./s
- LINEAR POWER 0.25 kW/ft–0.42 kW/ft
- INITIAL FRS TEMPERATURE 1460°F–800°F

UNION CARBIDE ORNL

REFLOOD TEST CONCLUSIONS

- AT HIGH FLOODING RATE AND HIGH WALL TEMPERATURES, COLLAPSED LIQUID LEVEL $\geq$ QUENCH LEVEL. SUGGESTS INVERTED ANNULAR FILM BOILING
- QUENCH FRONT VELOCITY $\cong$40–50% OF FLOODING VELOCITY (NOT DIRECTLY APPLICABLE TO REACTOR CASE)
- NO GROSS LIQUID CARRYOVER WAS NOTED IN ANY TEST. UPPER PLENUM PROBABLY ACTING AS A STEAM SEPARATOR
- ROD QUENCH TEMPERATURES RANGED FROM $833^{o}F \rightarrow 960^{o}F$

UNION CARBIDE ORNL

SMALL BREAK HEAT TRANSFER CONCLUSIONS

- HEAT TRANSFER DATA
  - $3{,}500 \leq Re \leq 10{,}200$
  - $800^{o}F \leq T_W \leq 1466^{o}F$
  - $1.2 \leq T_W/T_S \leq 1.65$
  - $350 \leq PRESS \leq 1000$ psia
  - $0.25 \leq POWER \leq 0.42$ kW/ft
- HEAT TRANSFER COEFFICIENTS 17–33 B/hr ft^2 ^{o}F
  $22\% \leq Q_{RAD}/Q_{TOTAL} \leq 37\%$
- EXISTING MODELS PREDICT EXPERIMENTAL HEAT TRANSFER REASONABLY WELL
  - MOSTLY WITHIN $\pm$30%
  - DO NOT PREDICT ALL OF DATA TRENDS
- VARIETY OF RADIATION MODELS COUPLED WITH A DITTUS-BOELTER LIKE CONVECTIVE CORRELATION WITH PHYSICAL PROPERTIES EVALUATED AT HEATED SURFACE TEMPERATURE PROVIDE GOOD AGREEMENT WITH DATA
- LEVEL SWELL VARIED FROM 15–111% AND WAS ROUGHLY PROPORTIONAL TO VOLUMETRIC VAPOR GENERATION RATE

UNION CARBIDE ORNL

ORNL FILM BOILING TESTS

UNION CARBIDE ORNL

DATA FROM THE UPFLOW FILM BOILING TESTS ARE BEING COMPARED WITH PREDICTIONS USING SEVERAL CORRELATIONS

GROENEVELD 5.7 AND 5.9 (RELAP)

DOUGALL-ROHSENOW (RELAP AND TRAC)

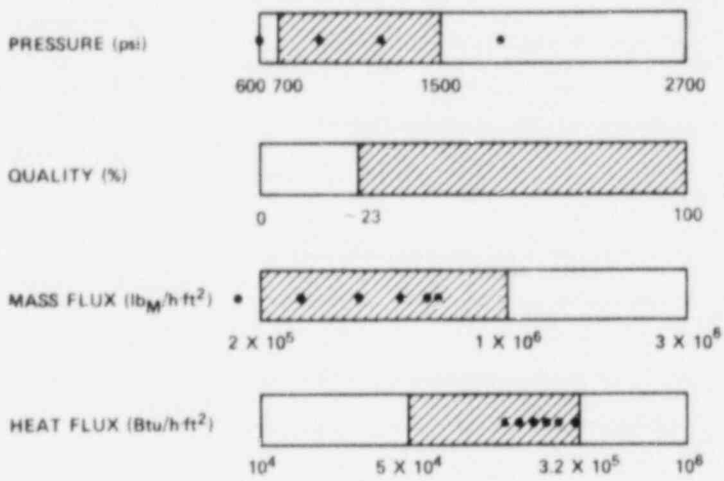
BROMLEY (TRAC)

GROENEVELD-HADALLER (COBRA/TRAC)

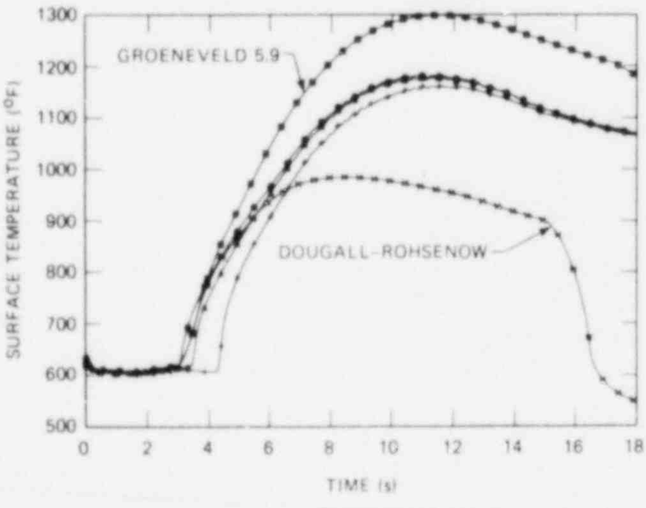
FORSLUND-ROHSENOW (COBRA/TRAC)

MODIFIED BROMLEY (COBRA/TRAC)

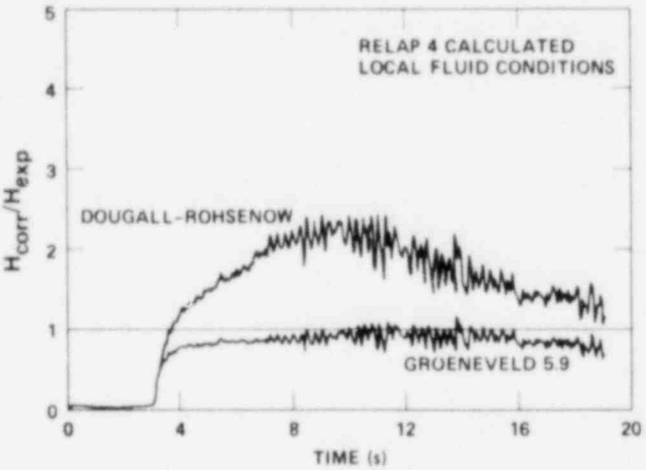
FILM BOILING DATA HAS BEEN OBTAINED OVER A WIDE RANGE OF CONDITIONS



RELAP4M5 UNDERPREDICTS THE ROD SURFACE TEMPERATURES IN ORNL TESTS USING THE DOUGALL-ROHSENOW CORRELATION



THE GROENEVELD 5.9 CORRELATION AGREES WITH THE ORNL TRANSIENT FILM BOILING DATA BETTER THAN THE DOUGALL-ROHSENOW CORRELATION



BWR BLOWDOWN / EMERGENCY CORE COOLING INTEGRAL PROGRAM (TLTA LARGE AND SMALL BREAK)

G. L. Sozzi

Contributors:
W. S. Hwang
L. S. Lee
R. Muralidharan
D. S. Seely

General Electric Company
Nuclear Engineering Department
San Jose, California, USA

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National Bureau Of Standards
Gaithersburg, Maryland

BACKGROUND

Emergency core cooling (ECC) systems are designed to maintain fuel cladding temperatures below specified limits, even under a wide range of loss of inventory or hypothetical small and large pipe break loss of coolant accidents (LOCA). The BWR Blowdown/Emergency Core Coolant (BD/ECC) Program is an experimentally based program to investigate the integral effects of ECC injection during a hypothetical LOCA. This program is sponsored by the Nuclear Regulatory Commission (NRC), the Electric Power Research Institute (EPRI), and the General Electric Company (GE).

The principal objective of the BD/ECC Program is to obtain and evaluate basic BD/ECC data from test system configurations which have performance characteristics similar to a BWR during a hypothetical LOCA. Other objectives include the determination of the degree to which current LOCA models describe the observed phenomena, and where necessary, development of improved physical interpretation of the governing phenomena.

The BD/ECC program can be considered as an extention of the BWR Blowdown Heat Transfer (BDHT) Program [1] which was completed in late 1975. The BD/ECC program is divided into several test phases (see Table 1) which are designed to investigate different portions or variations of the BWR LOCA responses.

During the first phase of the BD/ECC Program, basic system response information was investigated. A building block approach to evaluate the effects of various ECC systems, operating independently and in combination, was adopted. While the original experimental plan emphasized the large break hypothetical design basis accident, two tests were included to investigate the more probable small pipe break transisents. The planned small break test series was accelerated in response to the accident at TMI-2. In addition, this test phase was expanded to include uncovery (boil-off) tests under slow loss-of-inventory transients.

The Two Loop Test Apparatus (TLTA) shown in Figure 1 was the BWR system simulator used to provide the thermal hydraulic response of a BWR. Main features of this system simulator include: a full size, electrically heated bundle that is capable of duplicating the power output of a fuel bundle from full initial power to the decay heat power. Also included is a coolant injection system to supply the ECC flow rates.

The TLTA has been modified to meet the primary objective of each testing phase with the overall objective of maintaining a real-time, thermal-hydraulic system response.

STATUS

The 7 x 7 BDHT [1] and 8 x 8 BDHT [2] test phases, and all planned testing as part of the BD/ECC-1A [3] phase have been completed. The 7 x 7 and 8 x 8 BDHT tests investigated mainly the blowdown portion of the LOCA transient and as such, the ECC system was not activated. Results from these tests serve as a baseline from which the effectiveness of the ECC system can be evaluated. Large and small break tests with various combinations of ECCS injection parameters were performed in the BD/ECC-1A phase.

The salient results from the earlier test phases are included in Table 1. The BD/ECC-1A test phase recently completed is summarized below; The key results are discussed.

RESULTS

(a) Reference Large Break LOCA Test

The overall system response for this reference test is depicted by the system pressure response (Figure 2) and the mixture level response (Figures 3 and 4). Bundle heat-up generally occurs following bundle uncovery well after lower plenum flashing. Local dryouts are seen in a few locations as lower plenum flashing tapers off. These early (∿ 20 sec) dryouts, however, are all rewetted by fall-back cooling at about the same time as the HPCS begins to inject. Bulk heat-up of the bundle begins when the liquid continuum in the bundle collapses following the uncovery of the jet pumps exit in the lower plenum. Since the heat up occurs after HPCS injection, its severity is mitigated by the ECC fluid draining into the bundle. As a result, the maximum PCT was limited to below 700°F (375°C) as shown in Figure 5.

The bundle is completely quenched as it refloods. This reflood occurs prior to completely refilling of the lower plenum. The accumulation of ECCS inventory within the bundle is prevented from draining into the lower plenum due to counter current flow limiting (CCFL) at the lower end of bundle, ie, at the inlet orifice.

Similar results were observed for the overall system response for the peak power test except that the bundle temperature was higher as expected. The early, local dryouts were more pronounced in this test - maximum temperature of 1050°F was reached before rewetting. However, the maximum temperature during the ensuing bulk heat-up was below 800°F (427°C).

(b) Small Break Test

Two small break tests under conditions of non-degraded and degraded ECC systems were recently completed.The non-degraded test response was governed merely by the net imbalance of inventory leakage out the break and inventory make up due to activation of the high pressure core spray injection system. Prior to HPCS injection, the mixture level in the system dropped slowly. Following HPCS injection, the level eventually recovered as expected. The mixture level remained above the top of the core region throughout the test and therefore the bundle did not heat up.

The degraded small break test was conducted with the assumption of a failed high pressure system. For this test the automatic depressurization system (ADS) was activated when the mixture level dropped to near the top of the core region.The ADS allows steam to discharge through the steam lines and results in a rapid system drpressurization (See Fig.6). As the system pressure decreased to below the shut-off head of the low pressure ECC Systems (LPCS & LPCI), ECCS fluid was injected into and eventually refilled the system.

In conjunction with the testing and as part of the BWR-Owners Group activities associated with TMI, pretest predictions were made for these small break tests using the BWR small break licensing analysis methods. Figures 6 to 8 show the predicted and measured response for the degraded ECCS small break test. The overall system pressure response was closely predicted, the inventory and correspondingly the mixture level within the bundle core region was underpredicted. This lead to a predicted bundle dryout and heat up which was not observed in the test (Fig. 8).

From the test results, especially in comparison with the small break pretest predictions, it becomes evident that CCFL at regional boundaries is an important phenomena. Whereas CCFL at the upper tie plate might adversely restrict the upper plenum inventory from cooling the bundle, CCFL at the side entry orifice can beneficially prevent the bundle inventory from draining into the lower plenum. The net effect, as observed from the BWR system simulator is that the bundle refloods and the inventory provides significant cooling to the bundle. This latter mechanism is conservatively neglected in BWR analysis methods, which leads to the underprediction of coolant inventory and mixture level shown in Figure 7.

As with the large break tests, the system and bundle heat-up response are governed primarily by the movement of the mixture level within the bundle core region.

(c) Separate Effects - Boil Off Tests

A series of boiloff tests was completed in the TLTA. Saturated liquid was allowed to boil off from within the bundle under conditions of constant pressure (between 200 to 800 psia) at decay heat power. The slow boil off lead to uncovery within the bundle with nucleate pool boiling below the mixture level and steam cooling above. Comparisons were made with single channel analysis methods and the heat transfer rates were well predicted using standard single phase heat transfer correlations (e.g. Dittus-Boelter).

The void fraction distribution below the mixture level was also found to be in good agreement with the drift-flux model used in the single channel method (See Figure 9).

FUTURE PROGRAM DIRECTION

Discussions between the Program sponsors (NRC,EPRI and GE) have been under way for well over a year related to the future direction of this Program and the need for a new experimental facility. At present the favored approach is an upgraded single bundle, complete integral facility for purposes of testing small break LOCA's, loss-of-inventory threatening transients, and other multiple failure events.

REFERENCES

1. R. Muralidharan, "BWR Blowdown Heat Transfer Final Report" GEAP 21214. General Electric Company, San Jose, CA., Feb. 1976.

2. W. S. Hwang "BWR Blowdown/Emergency Core Cooling Program 64-ROD Bundle Blowdown Heat Transfer (8 x 8 BDHT) Final Report" GEAP-NUREG-23977, General Electric Co., San Jose, CA., Sept. 1978.

3. NUREG/CR-1154, GEAP-21304-15 "BWR Blowdown Emergency Core Cooling Fifteenth Quarterly Progress Report, July 1 - September 30, 1979", General Electric Co., San Jose, Ca., Feb. 1980

4. NUREG/CR-1154, GEAP-21304-14 "BWR BD/ECC Fourteenth Quarterly Progress Report, April 1 - June 30, 1979"

5. NUREG/CR-1154,GEAP-21304-18 "BWR BD/ECC Eighteenth Quarterly Progress Report, April 1 - June 30, 1980"

6. NUREG/CR-1154, GEAP 21304-17 "BWR BD/ECC Seventeenth Quarterly Progress Report, January 1 - March 31, 1980".

TABLE 1, TWO LOOP TEST APPARATUS (TLTA) - CHRONOLOGICAL SUMMARY OF RESULTS

PROGRAM PHASE	TLTA CONFIGURATION	OBJECTIVE	STATUS	SIMULATION BASES	RESULTS
7 x 7 BDHT	TLTA-1	BASELINE BWR DATA	COMPLETED 1975	-BWR/4 -BDHT ONLY -7 x 7 FULL SIZE BUNDLE -FULL BUNDLE POWER (4.55 MW)	• BUNDLE HEAT UP GOVERNED BY UNCOVERY • PCT MARGIN IDENTIFIED (~ 1000°F) • IMPROVED PHENOMENA UNDERSTANDING
8 x 8 BDHT	TLTA-2	VARIATION BUNDLE	COMPLETED 1976	-BWR/4 -BDHT ONLY -8 x 8 FULL SIZE BUNDLE -FULL BUNDLE POWER (6.5 MW	• PCT FOR 8 x 8 BUNDLE < PCT for 7 x 7 • NO NEW PHENOMENA
	TLTA-3	BWR/4 AND 6 TIE BACK	COMPLETED 1977	-BWR/6 -BDHT ONLY -8 x 8 FULL SIZE BUNDLE -FULL BUNDLE POWER (5.05 & 6.5 MW)	• BWR/6 DEPRESSURIZATION SLOWER COMPARED WITH BWR/4 AS EXPECTED
	TLTA-4	BASELINE DATA WITH NO ECC	COMPLETED 1978	-BWR/6 -BDHT ONLY - x 8 FULL SIZE BUNDLE -FU. BUNDLE POWER (5.05 & 6.5 MW) -UPPER TIE PLATE MOCKUP	• SYSTEM DEPRESSURIZATION SLOWER WITH IMPROVED JET PUMP SIMULATION (EXTENDED TAIL PIPE) • CCFL AT BUNDLE INLET HOLDS UP INVENTORY IN BUNDLE, DELAYS UNCOVERY AND ENHANCES HEAT TRANSFER
BD/ECC-1A	TLTA-5	EARLY (< 100 sec.) ECC INTERACTION	COMPLETED 1979	-BWR/6 -ECCS INJECTION, MULTIPLE FAILURE -8 x 8 FULL SIZE BUNDLE -FULL BUNDLE POWER (2.6 TO 6.5 MW) -ECCS PARAMETER VARIATIONS	• DEPRESSURIZATION WITH ECC INJECTION • ECCS EFFECTIVE IN REDUCING PCT • CCFL AT BUNDLE INLET DELAYS HEAT UP
	TLTA-5A	BD/ECC INTERACTION WITH IMPROVED SIMULATION (REFLOOD & POWER)	COMPLETED 1980	-BWR/6 -ECCS INJECTION, MULTIPLE FAILURE -8 x 8 FULL SIZE BUNDLE -FULL BUNDLE POWER (5.05 & 6.5 MW) -ECCS PARAMETERS VARIATION	• ECCS EFFECTIVE IN COOLING BUNDLE • MAX. PCT < 1000°F • BUNDLE REFLOODS BEFORE L.P. REFILLS COMPLETELY DUE TO CCFL AT SEO
	TLTA-5B	SMALL BREAK SCOPING TEST	COMPLETED 1980	-BWR/6 -HIGH PRESSURE ECC INJECTION ON HIGH DRYWALL PRESSURE -8 x 8 FULL SIZE BUNDLE -FULL BUNDLE POWER AFTER 7 SEC.	• NO CORE UNCOVERY • NO BUNDLE HEAT-UP • NO NEW PHENOMENA
	TLTA-5C	SMALL BREAK TEST BASELINE DATA	COMPLETED 1980	-BWR/6 -HPCS DEACTIVED FOR DEGRADED TEST -ADS ACTIVATED AND DELAYED 120 SEC. -LPCS PLUS 2/3 LPCI -8 x 8 FULL SIZE BUNDLE	• BUNDLE REFLOODED BY ECC FLUID AND CCFL AT SEO. • NO BUNDLE HEAT-UP • SYSTEM REFILLED
	TLTA-5A	BUNDLE UNCOVERY, BOIL-OFF SEPARATE EFFECTS	COMPLETED 1980	-BWR/6 -8 x 8 FULL SIZE BUNDLE -DECAY HEAT BUNDLE POWER -STEADY SYSTEM PRESSURE VARIATION -STEADY DECAY HEAT VARIATION	• HEAT TRANSFER RATES WELL PREDICTED BY STANDARD CORRELATIONS (E.G. DITTUS-BOELTER) • VOID DISTRIBUTION AGREES WELL WITH DRIFT-FLUX MODEL
BD/ECC-1B		INTEGRAL BLOWDOWN TO REFLOOD TEST. NON-LOCA TRANSIENTS TEST	FACILITY DESIGN BEING STUDIED		
BD/ECC-2		SEPARATE EFFECTS ECC TEST AT HIGH TEMPERATURE, ALTERNATE POWER PROFILE	ELIMINATED		
NON JET PUMP		BD/ECC PHENOMENA IN NON-JET PUMP CONFIGURATION	ELIMINATED		

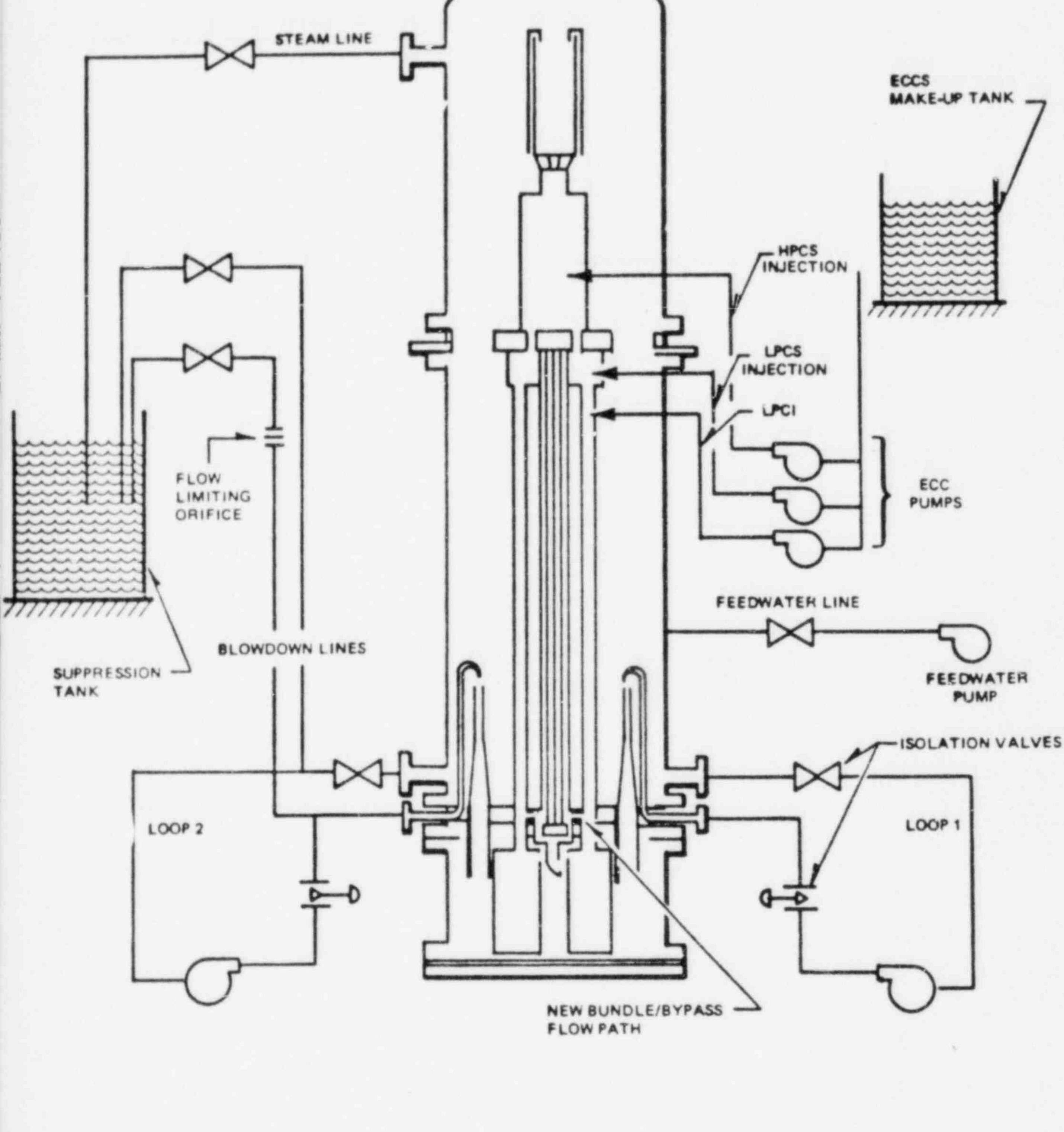


Figure -1. Two-Loop Test Apparatus Configuration 5A (TLTA-5A) with Emergency Core Cooling Systems

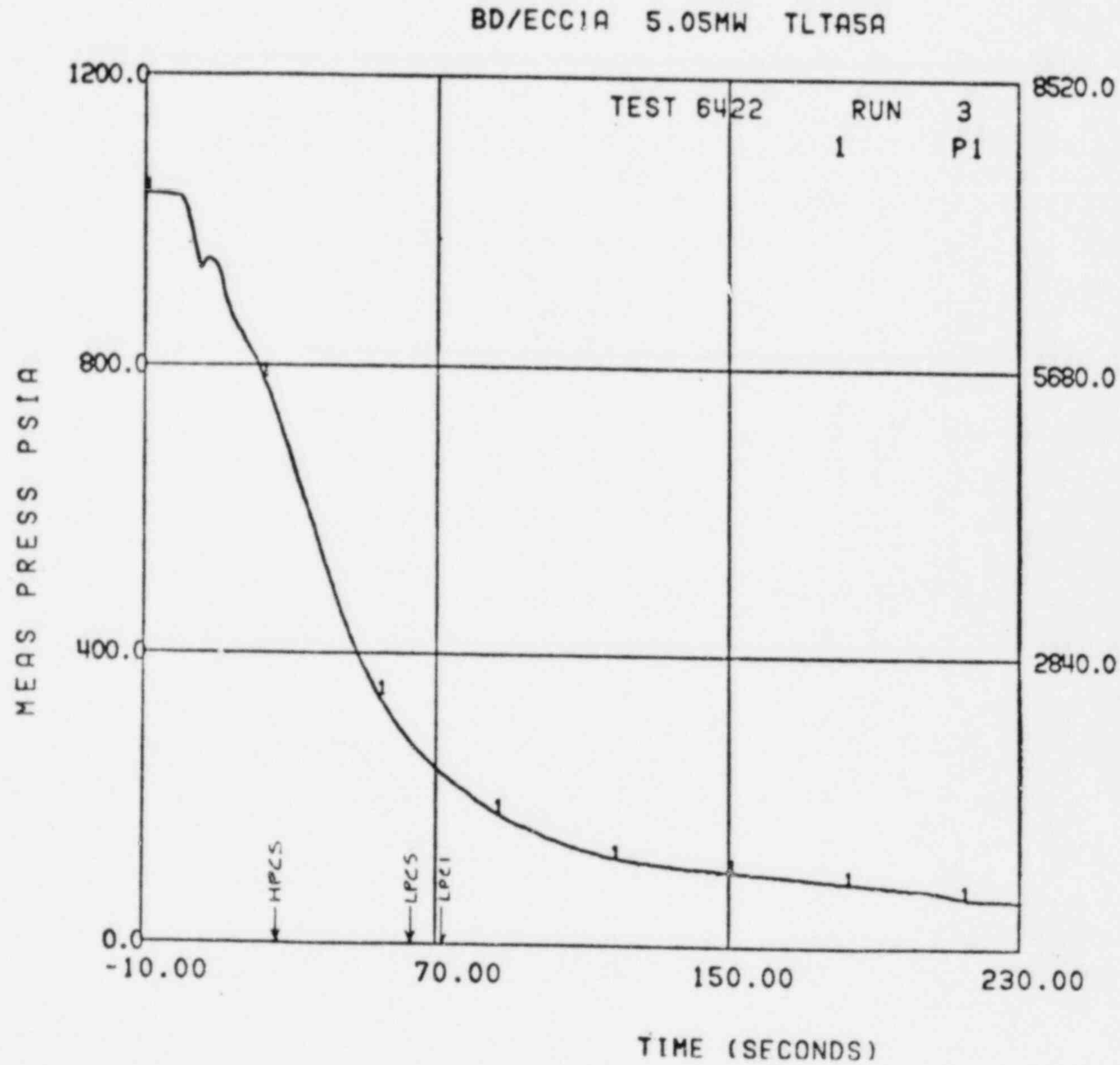


Figure 2, Steam Donue pressure

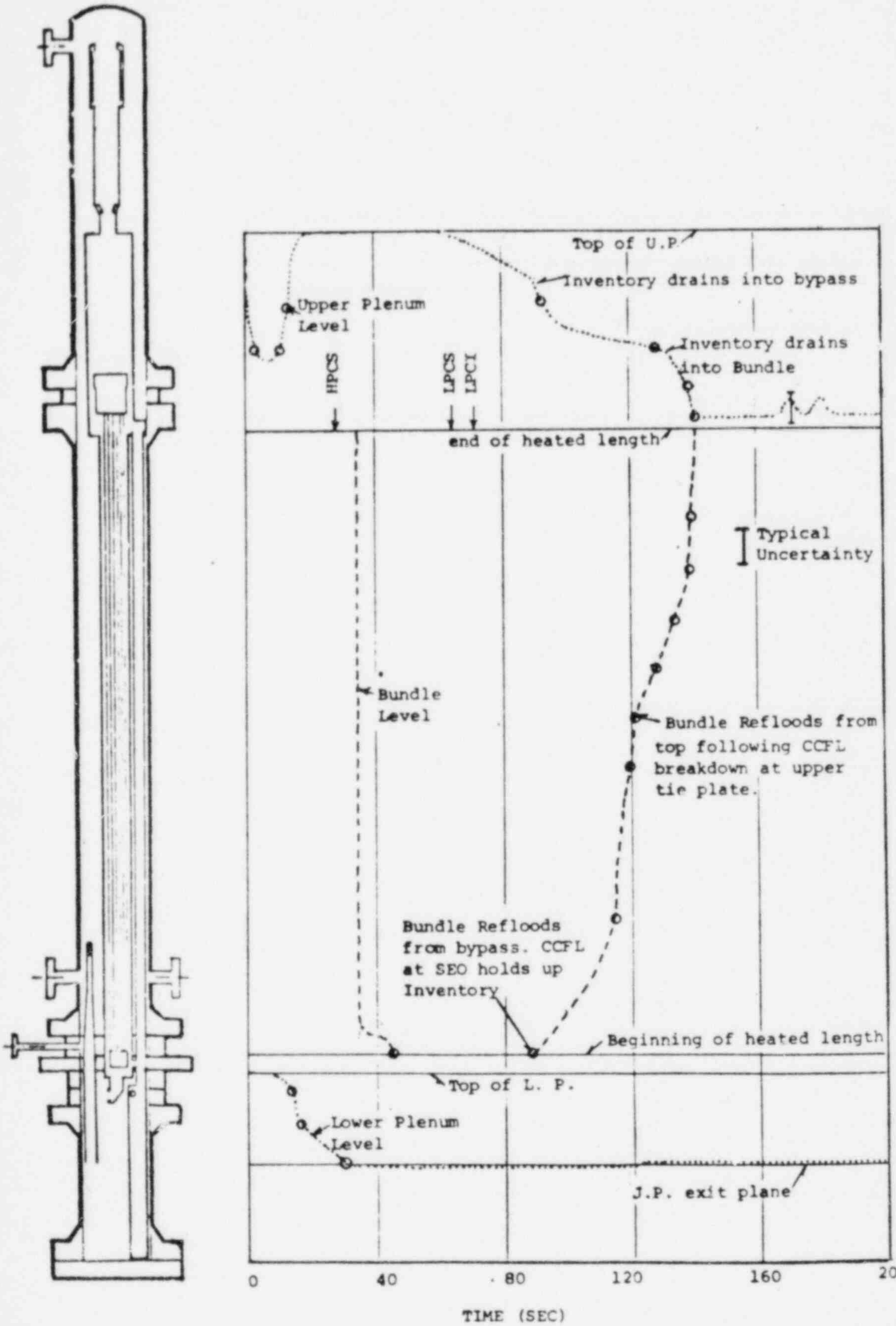


Figure 3, Mixture level along the bundle path for TLTA 5A Test 6422 Run 3 (Ave. power, Ave. ECC rates)

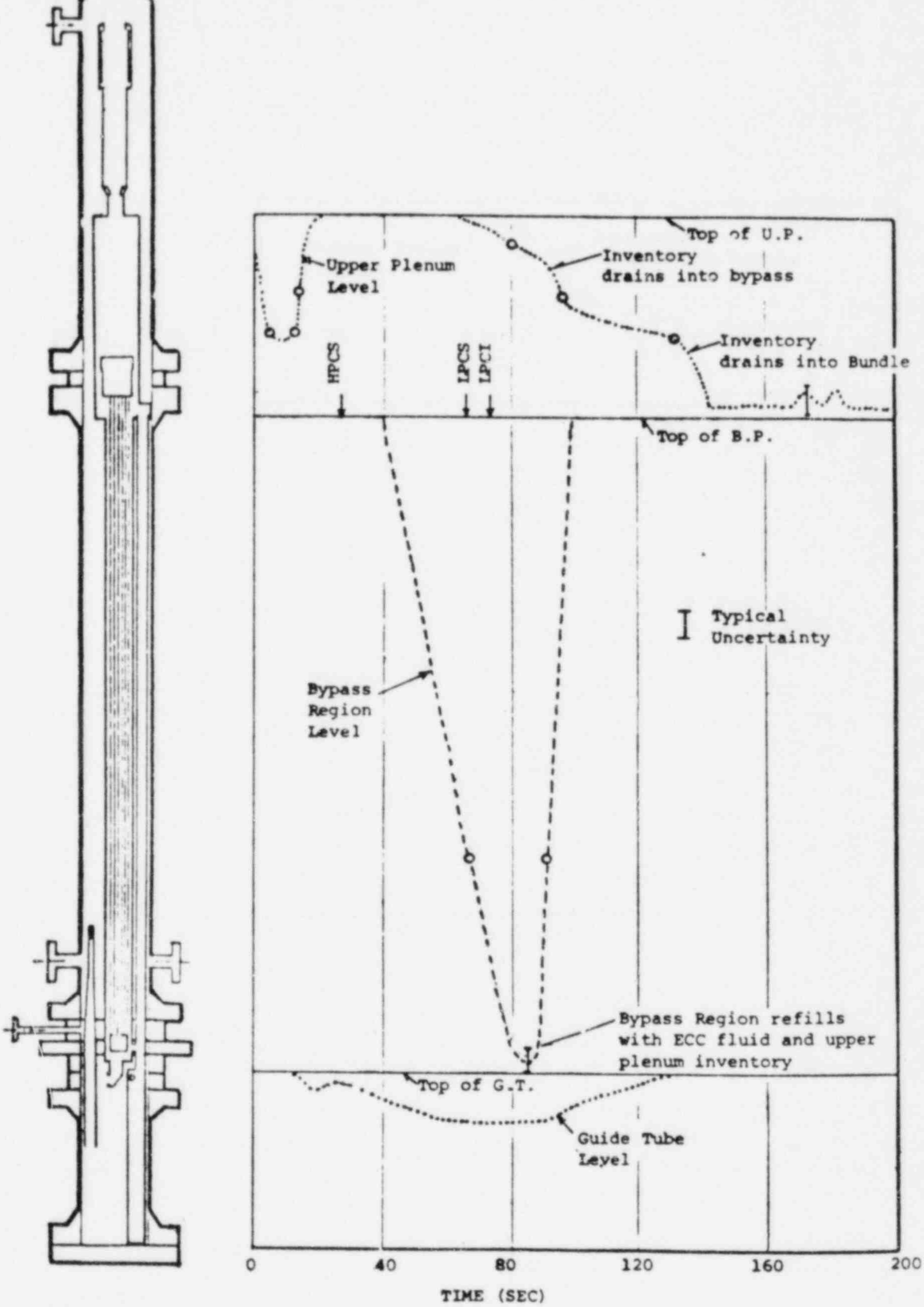


Figure 4, Mixture level along the bypass path for TLTA 5A Test 6422 Run 3 (Ave. power, Ave. ECC rates)

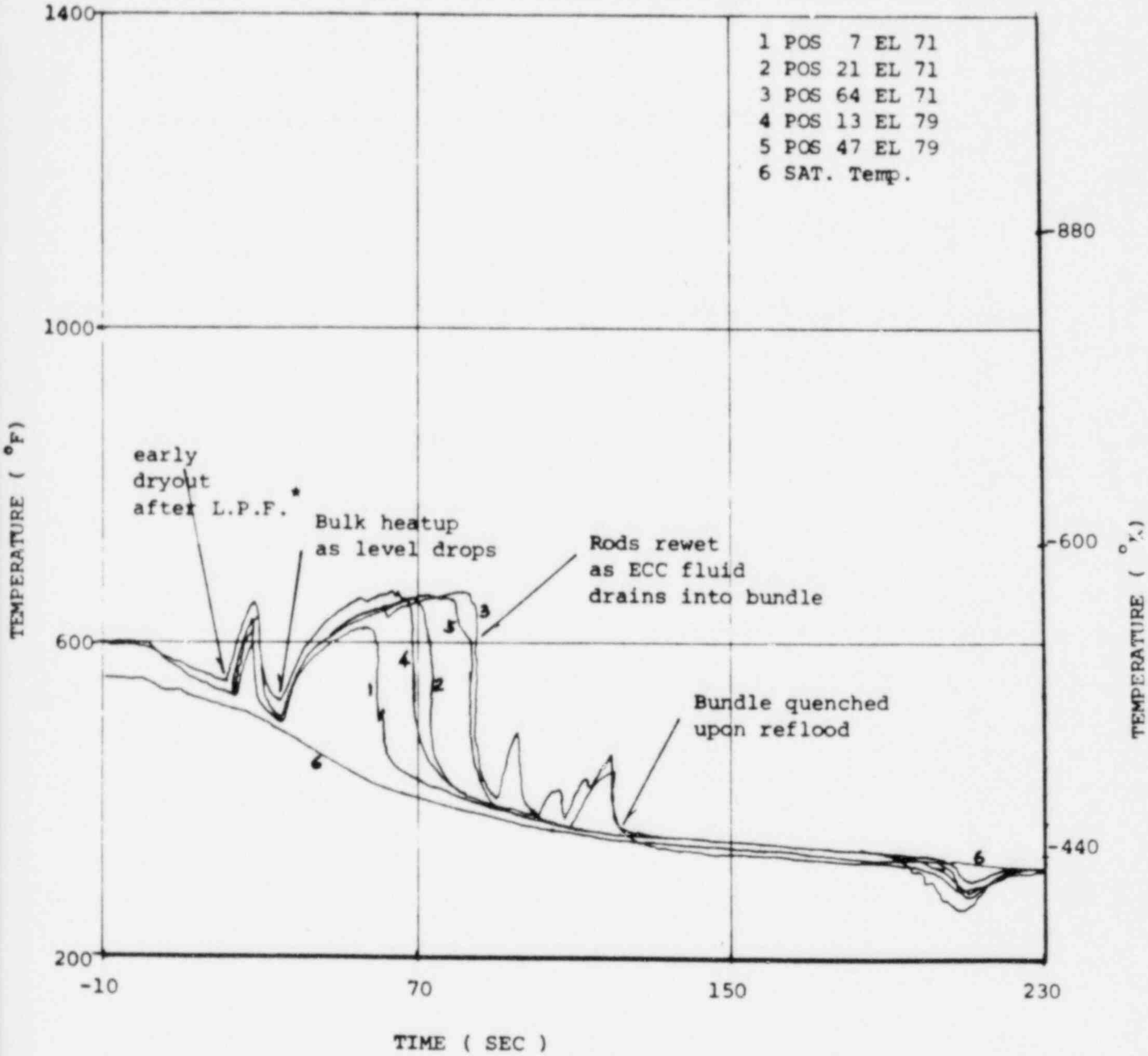


Figure 5, Peak Power Region cladding temperatures for TLTA-5A Test 6422 Run 3 (Average Power, Average ECC Tests)

*Note: Lower plenum flashing

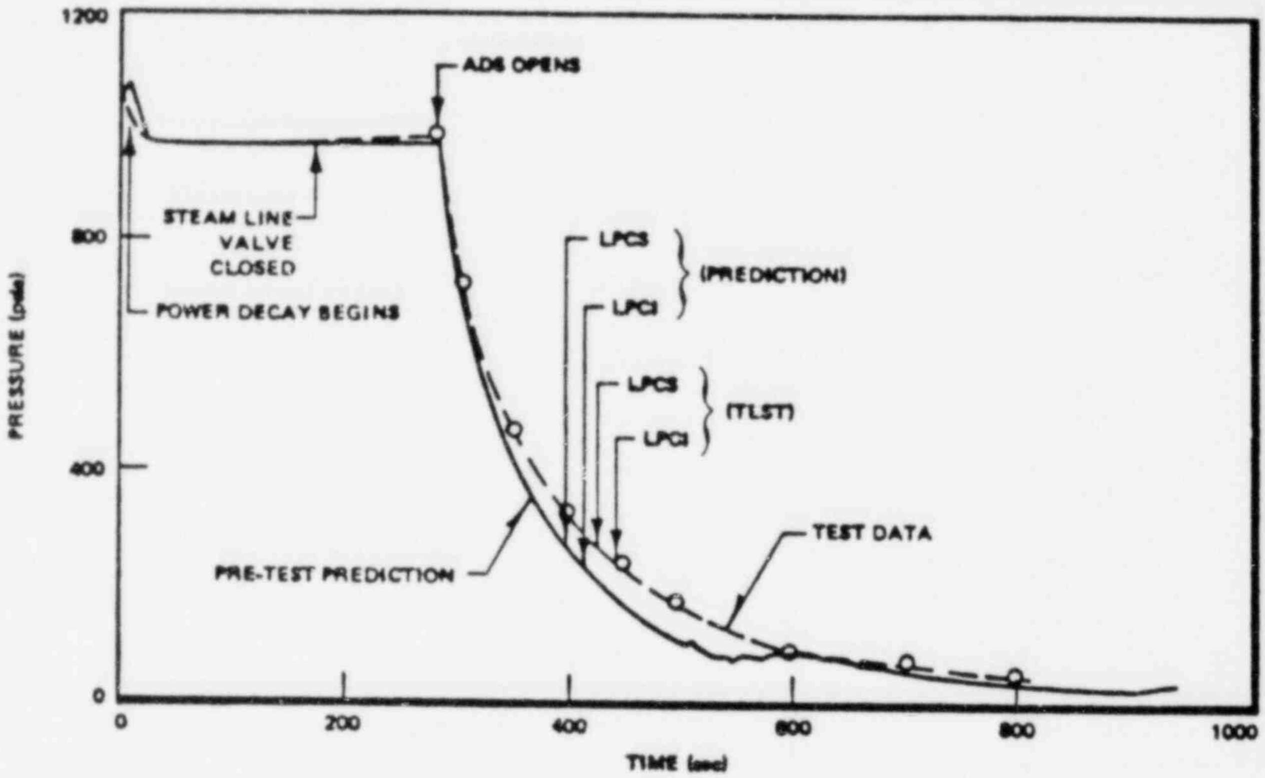


Figure 6, Comparison of System Pressures, TLTA Small Break Test II.

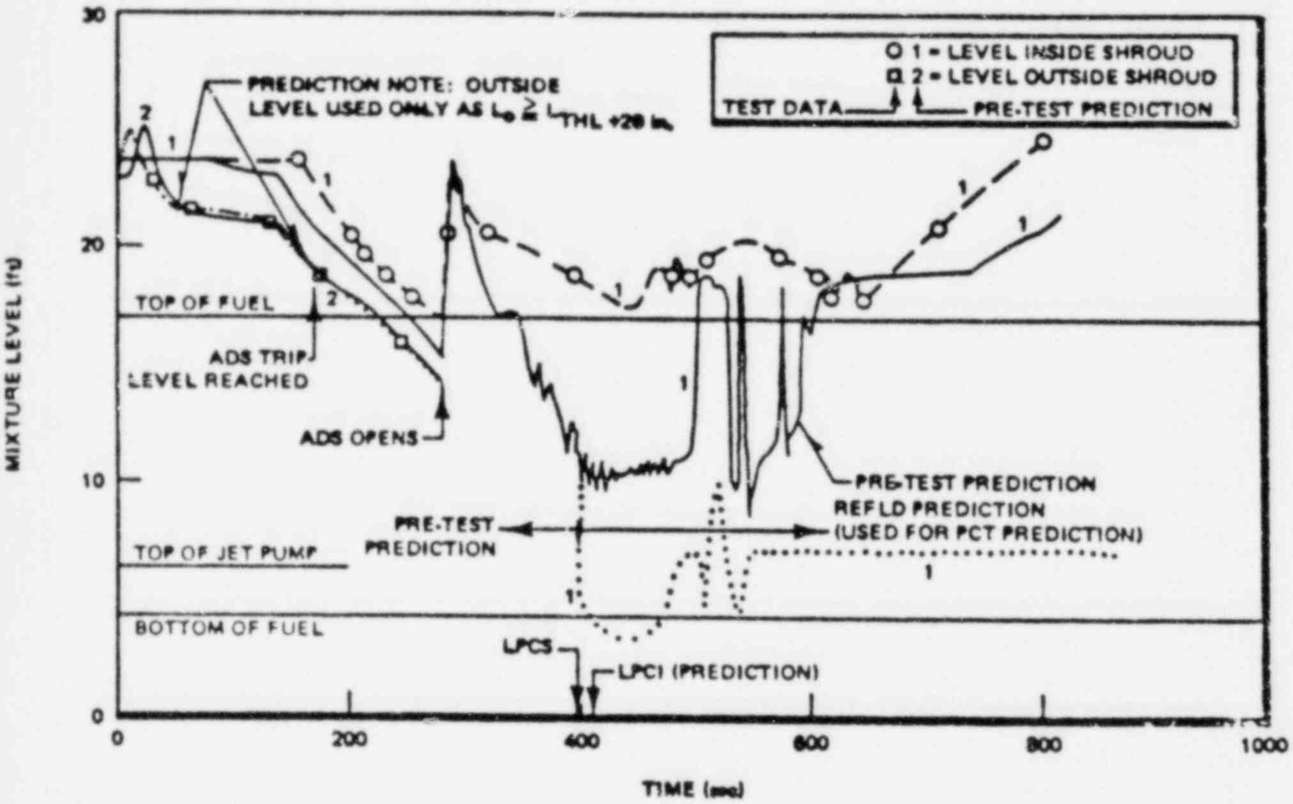


Figure 7, Comparison of Levels, TLTA Small Break Test II.

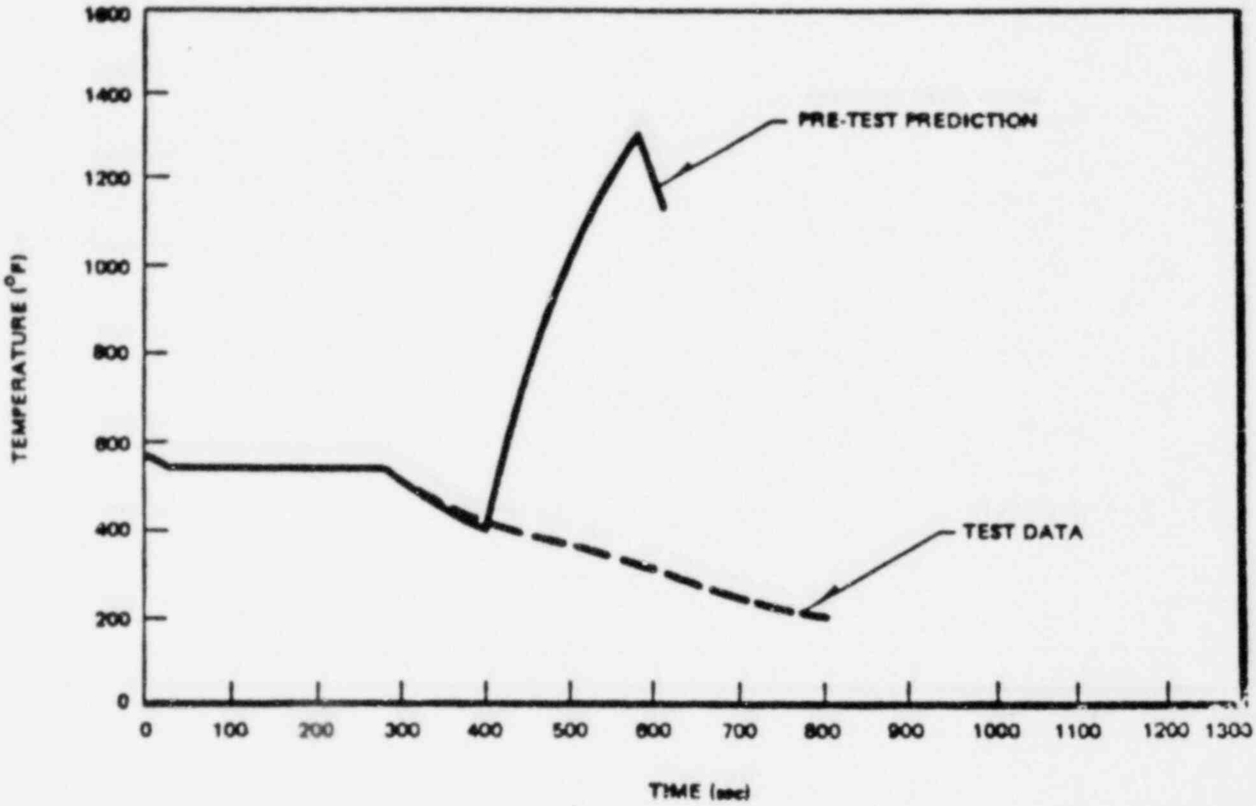


Figure 8, Comparison of Peak Cladding Temperatures, TLTA Small Break Test II.

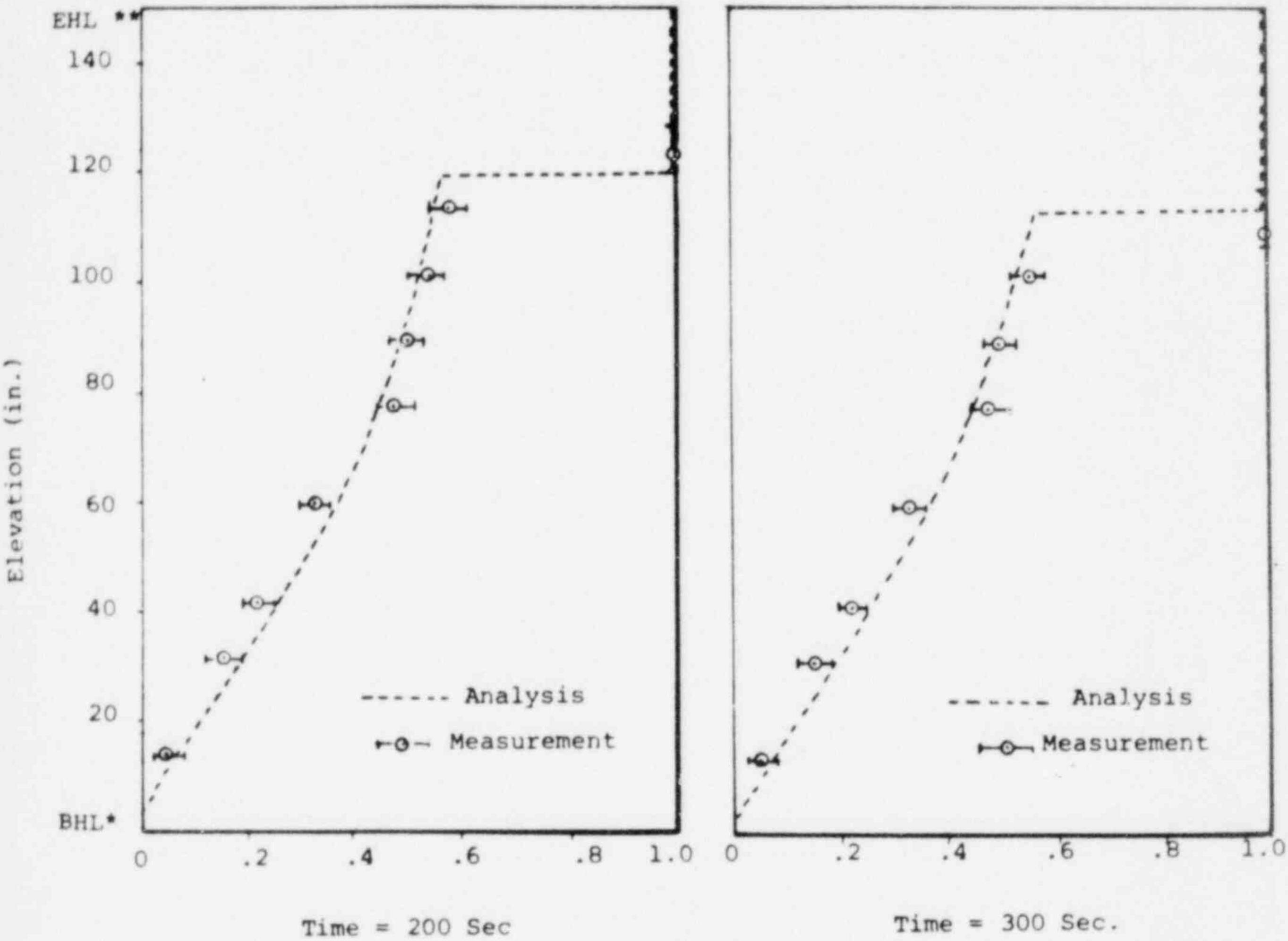


Figure 9, Comparison of bundle void distributions at selected times.

Note: * Beginning of heated length
** End of heated length

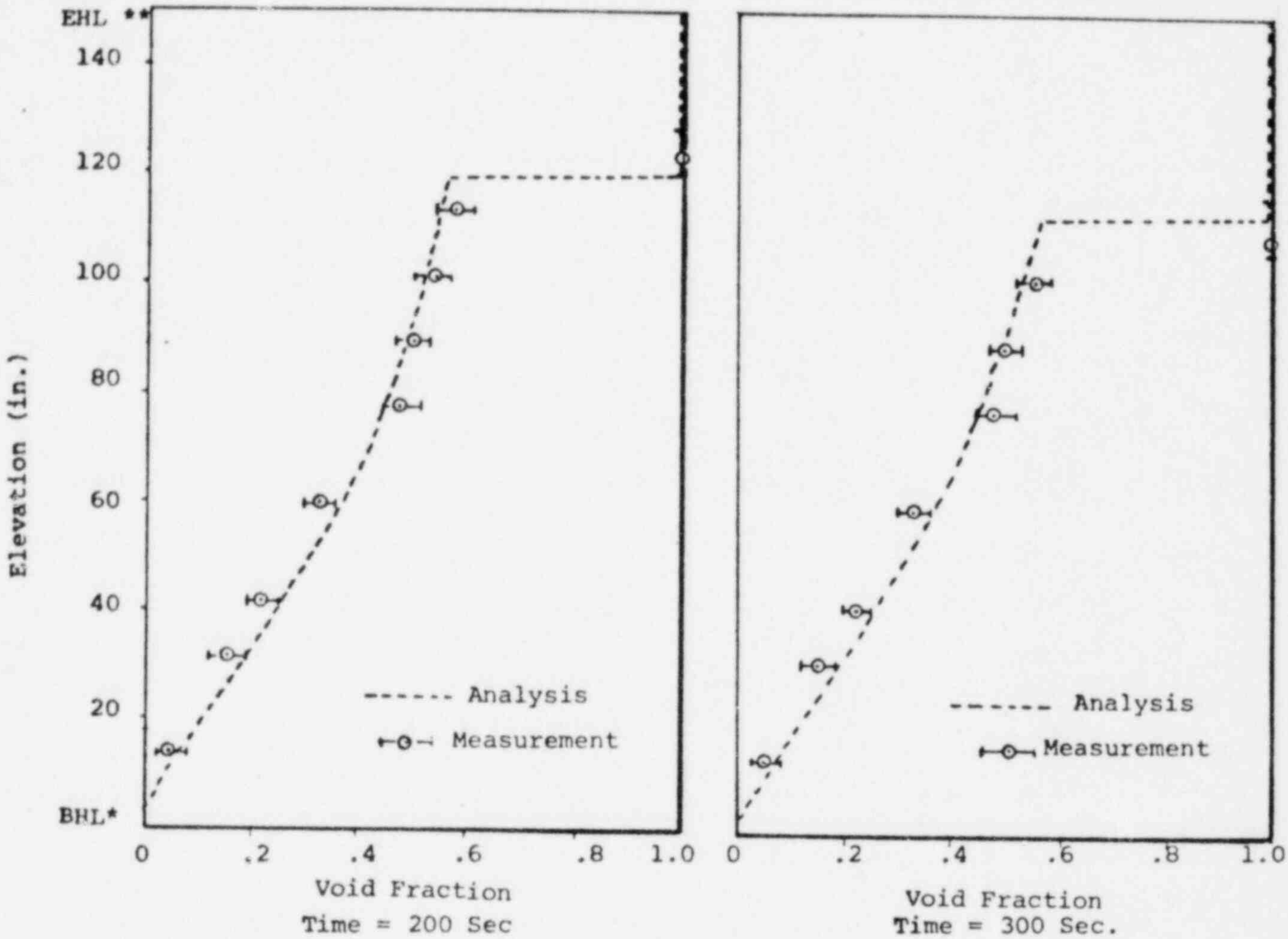


Figure 9, Comparison of bundle void distributions at selected times.

Note: * Beginning of heated length
** End of heated length

BWR REFILL-REFLOOD PROGRAM

OVERVIEW AND EXPERIMENTAL RESULTS

GW Burnette

GENERAL ELECTRIC COMPANY
NUCLEAR FUEL AND SERVICES ENGINEERING DEPARTMENT
SAN JOSE, CALIFORNIA

FOR PRESENTATION AT:

The 8th Water Reactor Safety Research Information Meeting
October 29, 1980
National Bureau of Standards
Gaithersburg, Maryland

PROGRAM SPONSORS:

US NUCLEAR REGULATORY COMMISSION
ELECTRIC POWER RESEARCH INSTITUTE
GENERAL ELECTRIC COMPANY

BWR REFILL-REFLOOD PROGRAM

OVERVIEW AND EXPERIMENTAL RESULTS

GW Burnette

The BWR Refill-Reflood Program is jointly sponsored by the U.S. Nuclear Regulatory Commission, Electric Power Research Institute and General Electric Company. The program will address the thermal-hydraulic behavior of BWR's during the refill and reflood phases of postulated LOCA's on a generic basis. A primary output of this program will be a set of best estimate models which can be used for realistic LOCA predictions. These models are being developed in conjunction with INEL and are suitable for incorporating into the TRAC-BWR system code. The experimental data from this program will be a major factor in the assessment of these models.

The program features a balanced combination of realistic model development and appropriate supporting experiments for model development and qualification. Separate effects tests are included for use in realistic model development while large scale system experiments are planned for use in independent qualification. A full radius, 30° Sector Facility at Lynn, Massachusetts is the primary large scale facility to be used within the program, but data from many other experiments will also be utilized for model assessment.

Early results from the program have confirmed that the GE core spray distribution prediction methodology [1] is sufficiently general to apply to an alternate geometry (BWR/4) and set of spray conditions in the 30° Sector (previously confirmed for a BWR/6 system design). The 30° Sector is presently being modified in preparation for conducting transient refill-reflood tests.

Development of a steam injection technique for simulating fuel bundles in the 30° Sector has been achieved using a single bundle system facility. Separate effects tests for model development are now underway in this single bundle system. A number of preliminary models for simulating BWR component performance and thermal-hydraulic phenomena have been developed. These results and the status of model development are detailed in the Model Development presentation.

Reference

1. SA Sandoz, et. al., "Core Spray Design Methodology Confirmation Tests", NEDO-24712, August, 1979.

BWR REFILL-REFLOOD PROGRAM

PROGRAM OVERVIEW	GW BURNETTE
EXPERIMENTAL RESULTS AND STATUS	GW BURNETTE
MODEL DEVELOPMENT AND STATUS	JGM ANDERSEN

SPONSORED BY:

USNRC,	PMG MEMBER - WD BECKNER
EPRI,	PMG MEMBER - M MERILO
GE,	PMG MEMBER - GW BURNETTE

OCTOBER 29, 1980

REFILL-REFLOOD PROGRAM OVERVIEW

OBJECTIVES

PAYOFF:

QUALIFIED BEST ESTIMATE LOCA METHODS

SPECIFIC OBJECTIVES:

- IMPROVED UNDERSTANDING OF PHENOMENA CONTROLLING REFILLING AND REFLOODING OF THE BWR
- BASIS FOR AND SUPPORT TO DEVELOPMENT AND QUALIFICATION OF BWR THERMAL-HYDRAULIC LOCA CODES
- BASIS FOR ASSESSING ASSUMPTIONS USED IN ESTABLISHING BWR LOCA SAFETY MARGINS

GWB
10/29/80

REFILL-REFLOOD PROGRAM OVERVIEW

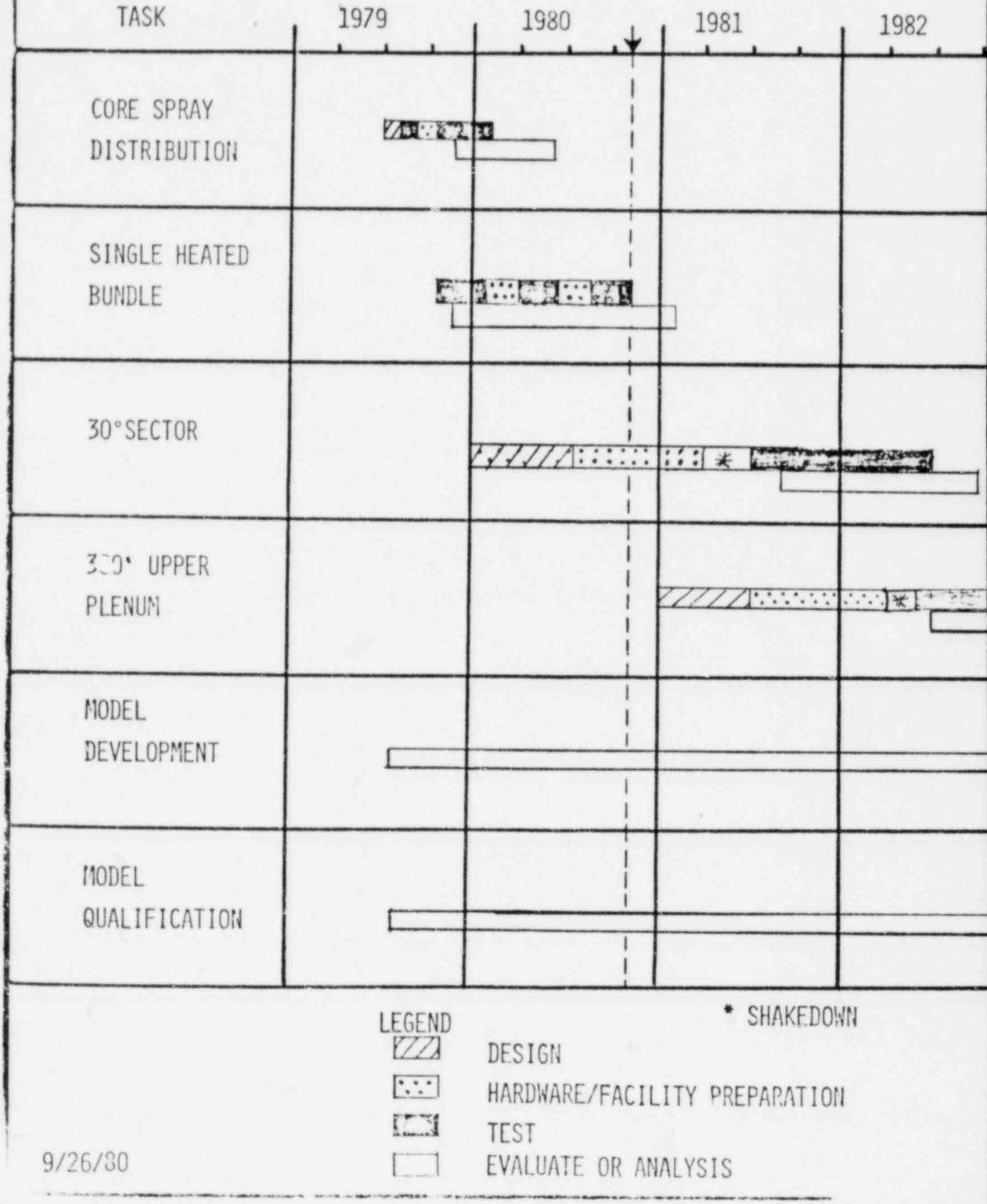
MAIN ELEMENTS AND STATUS

TASK	STATUS
CORE SPRAY DISTRIBUTION	COMPLETE
SINGLE HEATED BUNDLE	SYSTEM EFFECTS TESTS COMPLETE ADIABATIC STEAM INJECTION TESTS COMPLETE SEPARATE EFFECTS TESTS IN PROGRESS
CCFL/REFILL SYSTEM EFFECTS TESTS	MODIFICATIONS UNDERWAY MEASUREMENT PLAN FINALIZED
360° UPPER PLENUM TESTS	NOT STARTED
MODEL DEVELOPMENT	MANY BASIC MODELS DEVELOPED/IMPROVED (CONSTITUTIVE AND HEAT TRANSFER) SINGLE CHANNEL MODEL STARTED
MODEL QUALIFICATION	TASK PLANNING COMPLETE

GWB
10/29/80

REFILL-REFLOOD PROGRAM OVERVIEW

SCHEDULE



9/26/80

REFILL-REFLOOD PROGRAM EXPERIMENTAL RESULTS

SINGLE BUNDLE SYSTEM TESTS

OBJECTIVES

- IDENTIFY AND EVALUATE PHENOMENA CONTROLLING THE REFILL AND REFLOOD PHASE OF A BWR LOCA
- DEVELOP AN ADIABATIC INJECTION TECHNIQUE FOR THE 30° SECTOR FACILITY
- OBTAIN SEPARATE EFFECTS THERMAL-HYDRAULIC PERFORMANCE DATA FOR MODEL DEVELOPMENT

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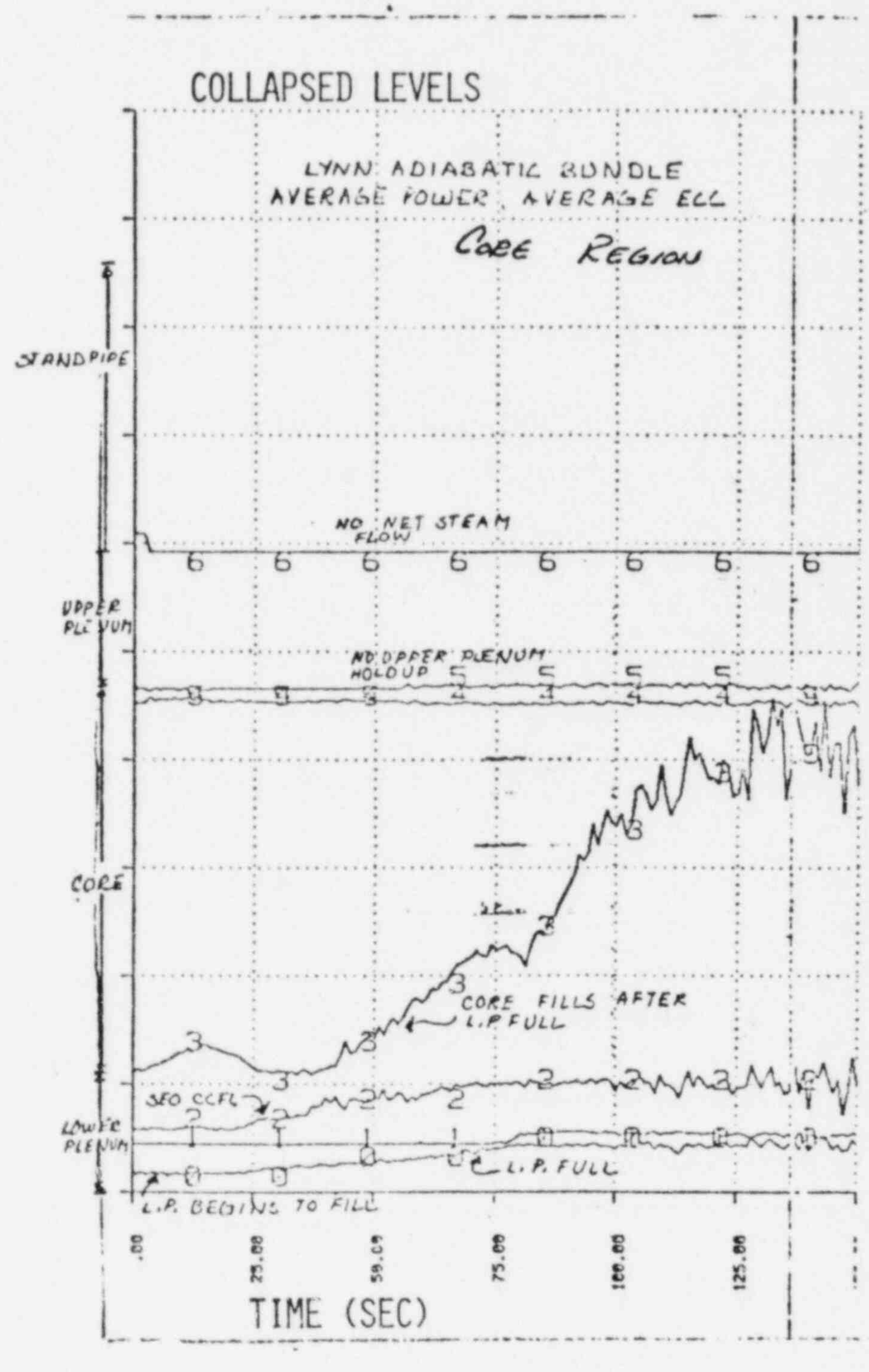
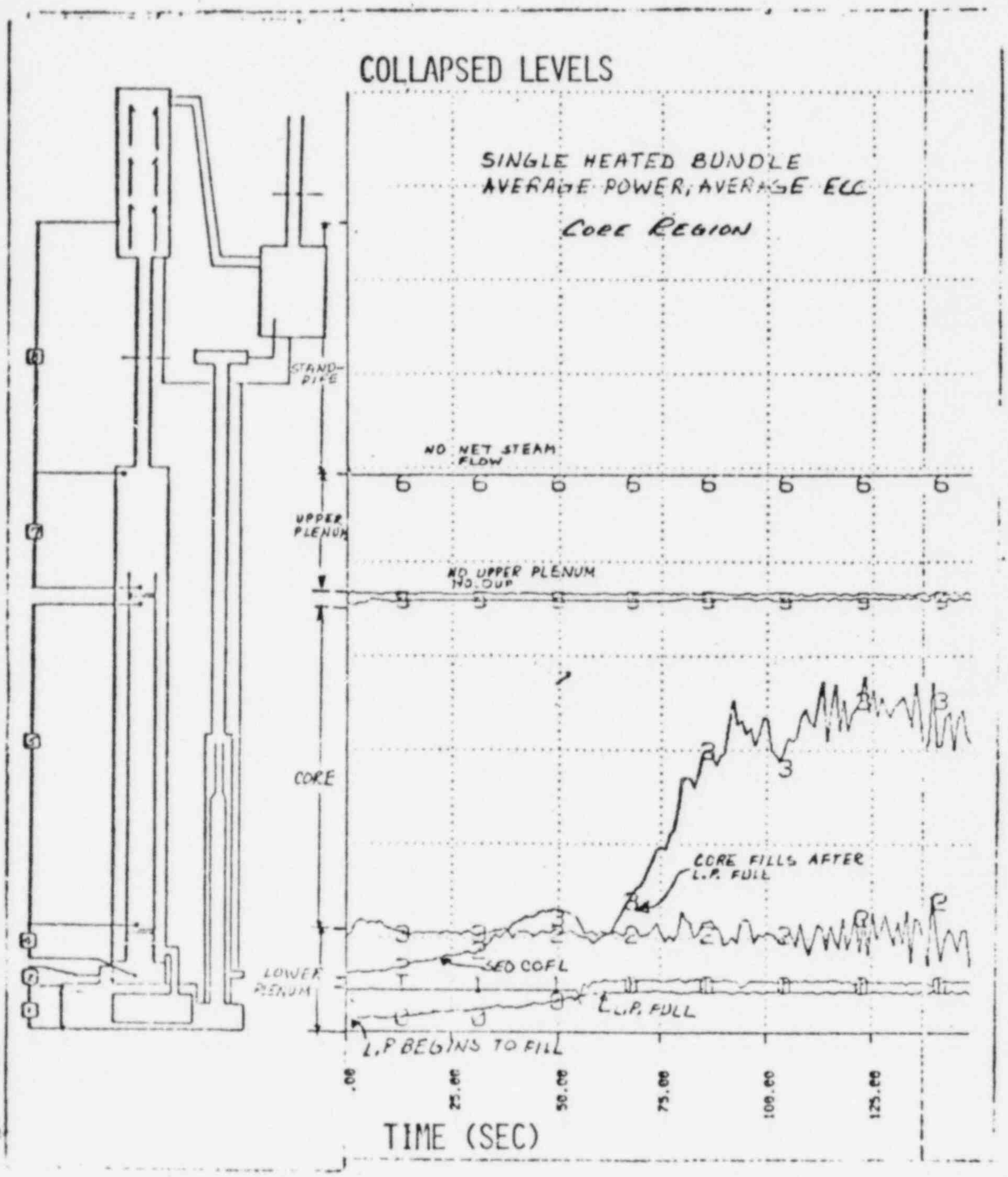
SINGLE BUNDLE SYSTEM TESTS

ADIABATIC STEAM INJECTION



SINGLE BUNDLE SYSTEM TESTS

HEATED BUNDLE



SINGLE BUNDLE SYSTEM TESTS

PRELIMINARY CONCLUSIONS

- SIMILAR LOWER PLENUM AND CORE REGION REFILL CHARACTERISTICS
- REFILLING INSENSETIVE TO BUNDLE-UPPER PLENUM FEEDBACK OVER EXPECTED CONDITION RANGE
- STEAM INJECTION CAN BE USED TO SIMULATE HEATED BUNDLES IN 30° SECTOR

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10/29/80

REFILL - REFLOOD PROGRAM EXPERIMENTAL RESULTS

30° SECTOR FACILITY

BACKGROUND

- BWR RESPONSE INFLUENCED BY CCFL BREAKDOWN IN UPPER PLENUM
- BREAKDOWN TIME DEPENDENT UPON MIXING
- GOOD MIXING MODELS NEEDED FOR REALISTIC RESPONSE PREDICTIONS

OBJECTIVE

- OBTAIN REALISTIC, LARGE SCALE REFILL-REFLOOD PERFORMANCE DATA FOR MODEL QUALIFICATION/DEVELOPMENT

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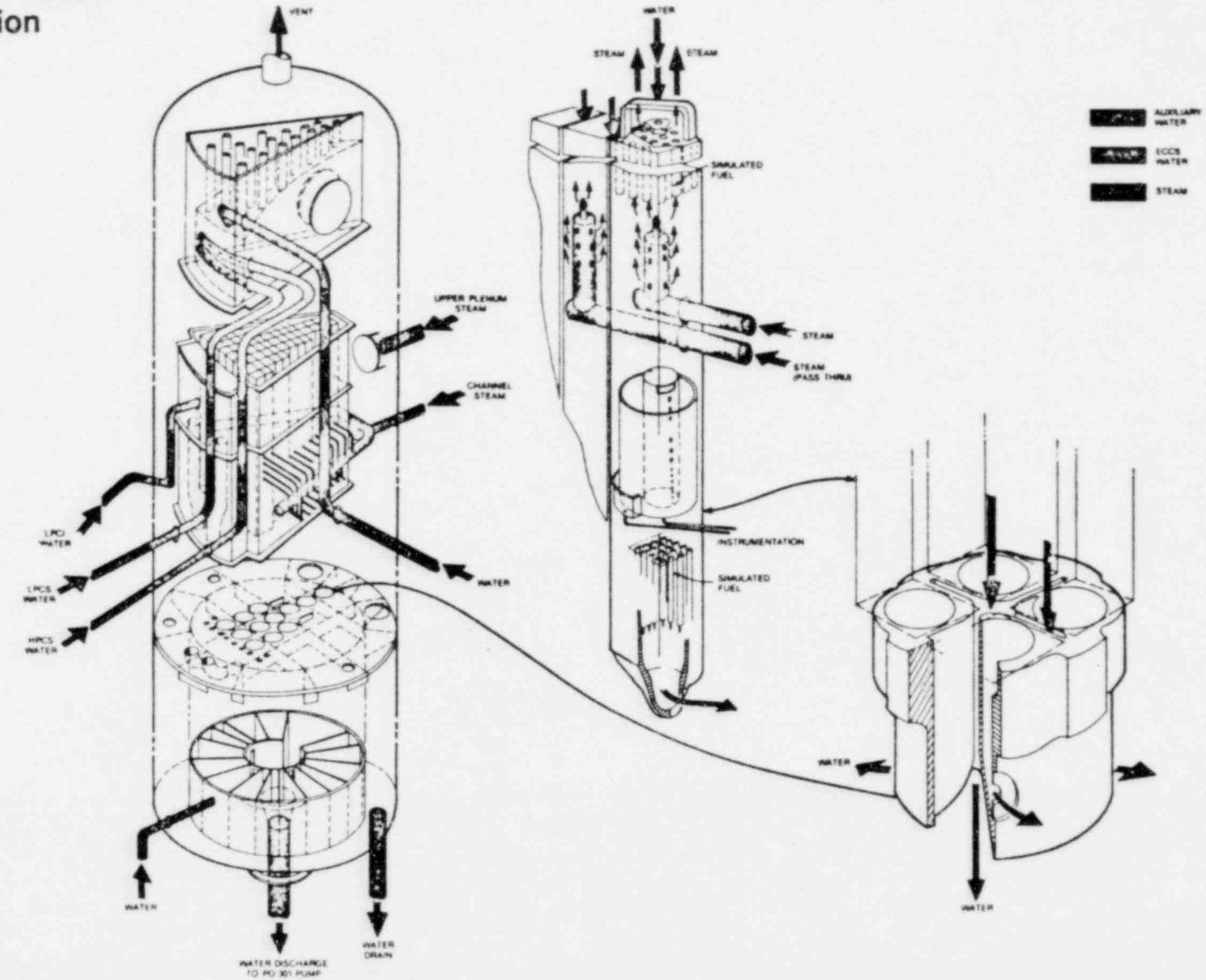
30° Steam Sector Test Section



30° SECTOR FACILITY

MODIFICATION NEEDS

FACILITY HARDWARE

- BLOWDOWN SYSTEM
- INITIALIZATION EQUIPMENT
- EXCESS VOLUME VENT SYSTEM
- VESSEL INTERNALS

TEST INSTRUMENTATION (AUGMENTATION)

- REGIONAL FLUID INVENTORIES AND LEVELS
- TEMPERATURE FIELDS IN ECC INJECTION REGIONS
- LOCAL SUBCOOLING MEASUREMENTS
- SYSTEM BOUNDARY CONDITIONS

DATA ACQUISITION/DATA REDUCTION

- INCREASED MEASUREMENT CAPABILITY
- INCREASED INSTRUMENT/DATA QA CHECKS
- DERIVED QUANTITIES FOR MODEL ASSESSMENT

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10/19/80

30° SECTOR FACILITY

MILESTONES

MAJOR HARDWARE INSTALLED	4Q80
COMPLETE SYSTEM INSTALLATION	1Q81
SHAKEDOWN COMPLETE	2Q81
BEGIN TESTING	3Q81

BWR REFILL-REFLOOD PROGRAM

MODEL DEVELOPMENT FOR TRAC-BD

GENERAL ELECTRIC

PROGRAM SPONSORS:

U. S. NUCLEAR REGULATORY COMMISSION

ELECTRIC POWER RESEARCH INSTITUTE

GENERAL ELECTRIC COMPANY

JGM ANDERSEN

OCTOBER 1980

OBJECTIVE

Best Estimate Simulation of a BWR LOCA Transient.

TASK

Develop Models for BWR Components and Phenomena of Importance for BWRs.

JGM Andersen

October 1980

BWR COMPONENT MODELS

- JET PUMP
  - MIXING
  - MIXING AND IRREVERSIBLE LOSSES
  - REVERSIBLE LOSSES

- STEAM SEPARATOR
  - PRESSURE DROP
  - CARRY OVER AND CARRY UNDER

- DRYER
  - PRESSURE DROP

- UPPER PLENUM
  - MIXING
  - VOID AND ENTHALPY DISTRIBUTION
  - SUBCOOLED CCFL BREAKDOWN

- FUEL BUNDLE (INEL)

JGM ANDERSEN
OCTOBER 1980

MODELS FOR IMPORTANT BWR PHENOMENA

- Counter Current Flow Limitation
- Void Fraction and Pressure Drop
  - Interface Shear
  - Wall Friction
- Heat Transfer
  - Interface Heat Transfer
  - Wall Heat Transfer

JGM Andersen
October 1980

SCHEDULE

- FIRST VERSION — END 1980
  - JET PUMP
  - SIMPLE SEPARATOR AND DRYER MODELS
  - SIMPLE UPPER PLENUM MODEL
  - CCFL
  - VOID FRACTION AND PRESSURE DROP
  - HEAT TRANSFER
- FINAL VERSION — MID 1982
  - SEPARATOR AND DRYER MODELS
  - UPPER PLENUM MODEL

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JET PUMP MODEL

- BASED ON TRAC TEE COMPONENT
- MODEL DEVELOPMENT
  - MOMENTUM EQUATION MODIFIED TO IMPROVE PREDICTION OF REVERSIBLE LOSSES AT AREA CHANGES.
  - MIXING PROCESS IN MIXING REGION.
  - MIXING LOSSES.
  - IRREVERSIBLE LOSSES IN DRIVE LINE, SUCTION, AND DISCHARGE.
  - FORWARD AND REVERSE FLOW.
- ASSESSED ON INEL JET PUMP DATA

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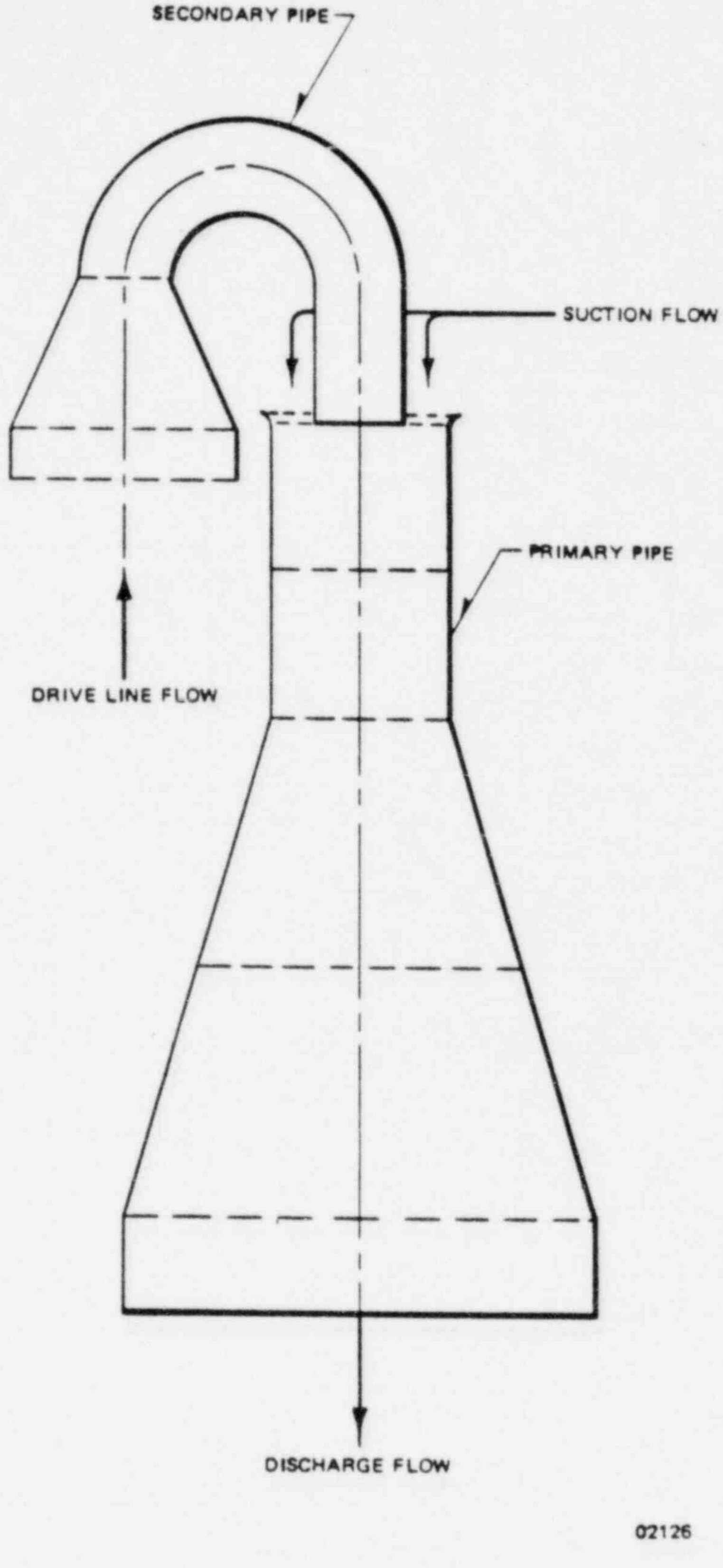

Figure 12. Two-Fluid Jet Pump Model II

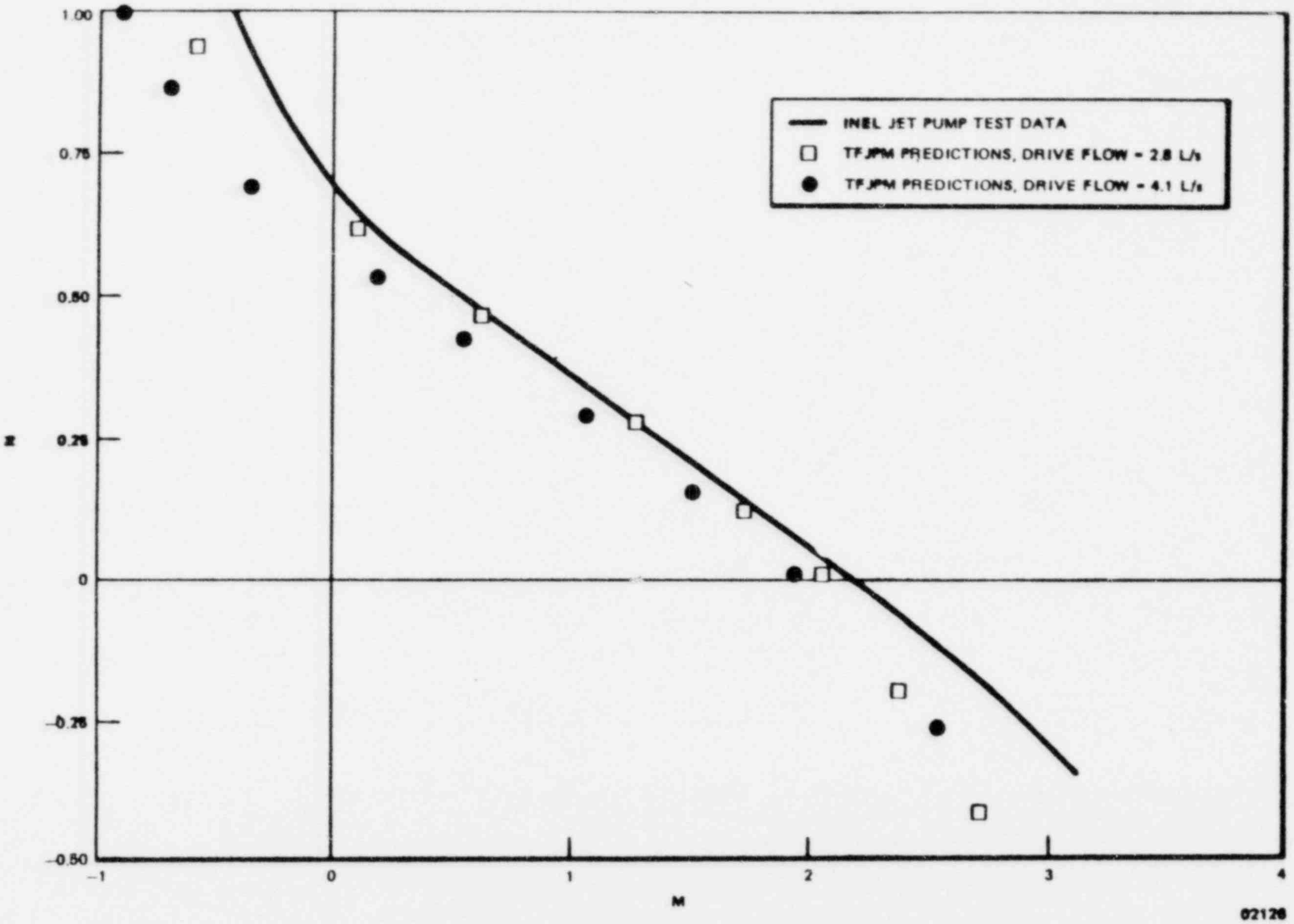

Figure 16. Comparison of TFJPM Predictions with INEL Test Data for +ve Drive Flow

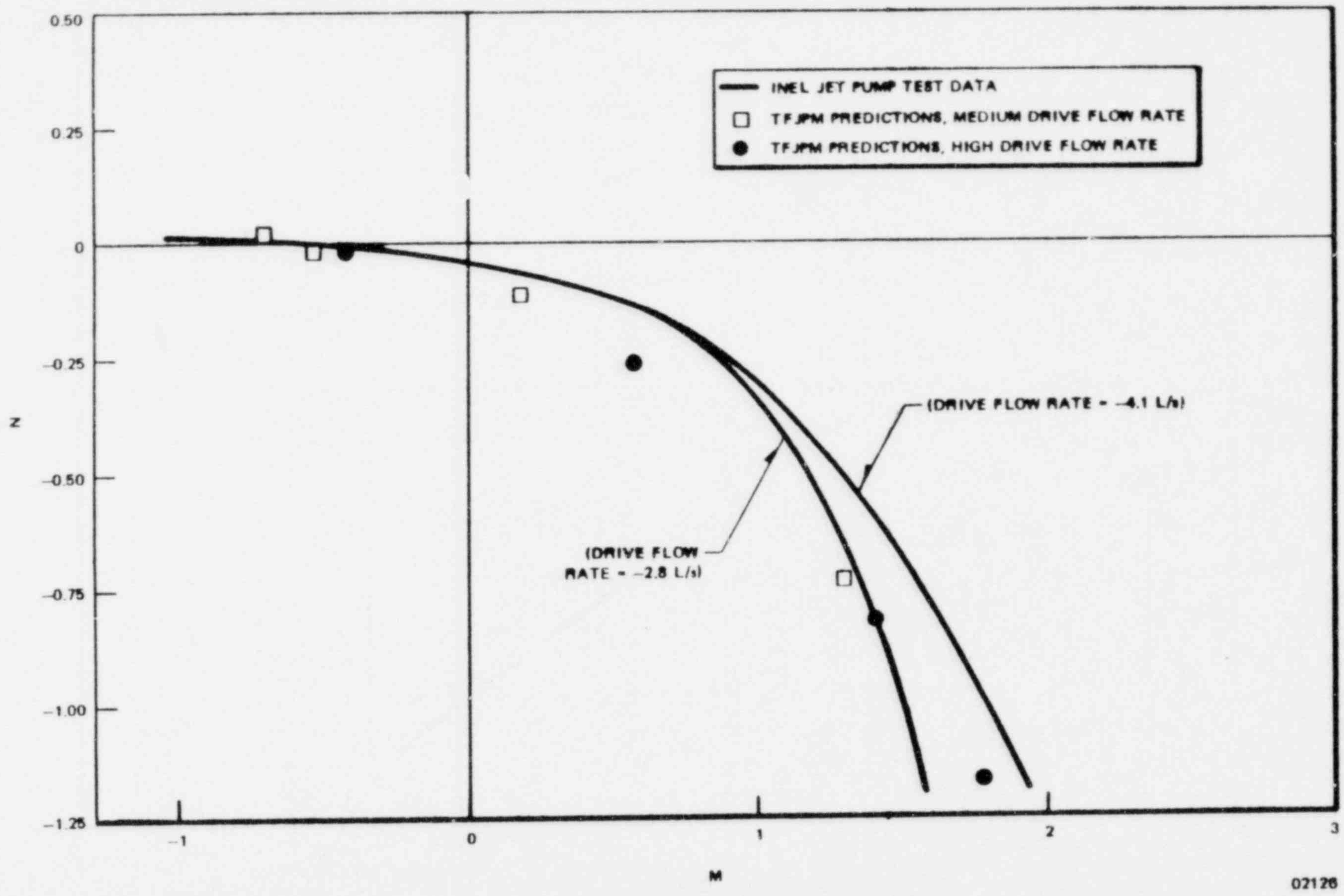

Figure 17. Comparison of TFJPM Predictions with INEL Test Data for -ve Drive Flow

VOID FRACTION MODEL

INTERFACE SHEAR

- Based on Void Fraction Data
- Model Features
  - Interfacial Force Depends on:
    - Drag due to relative motion of phases
    - Shear due to phase distribution
    - Phase and velocity distribution
    - Wall friction governed shear field
  - Match to Counter Current Flow Limiting Data
- Flow Regimes
  - Bubbly/Churn Flow
  - Annular Flow
  - Dispersed Droplet Flow
  - Single Phase Flow
- Assessed on Separate Effects Tests

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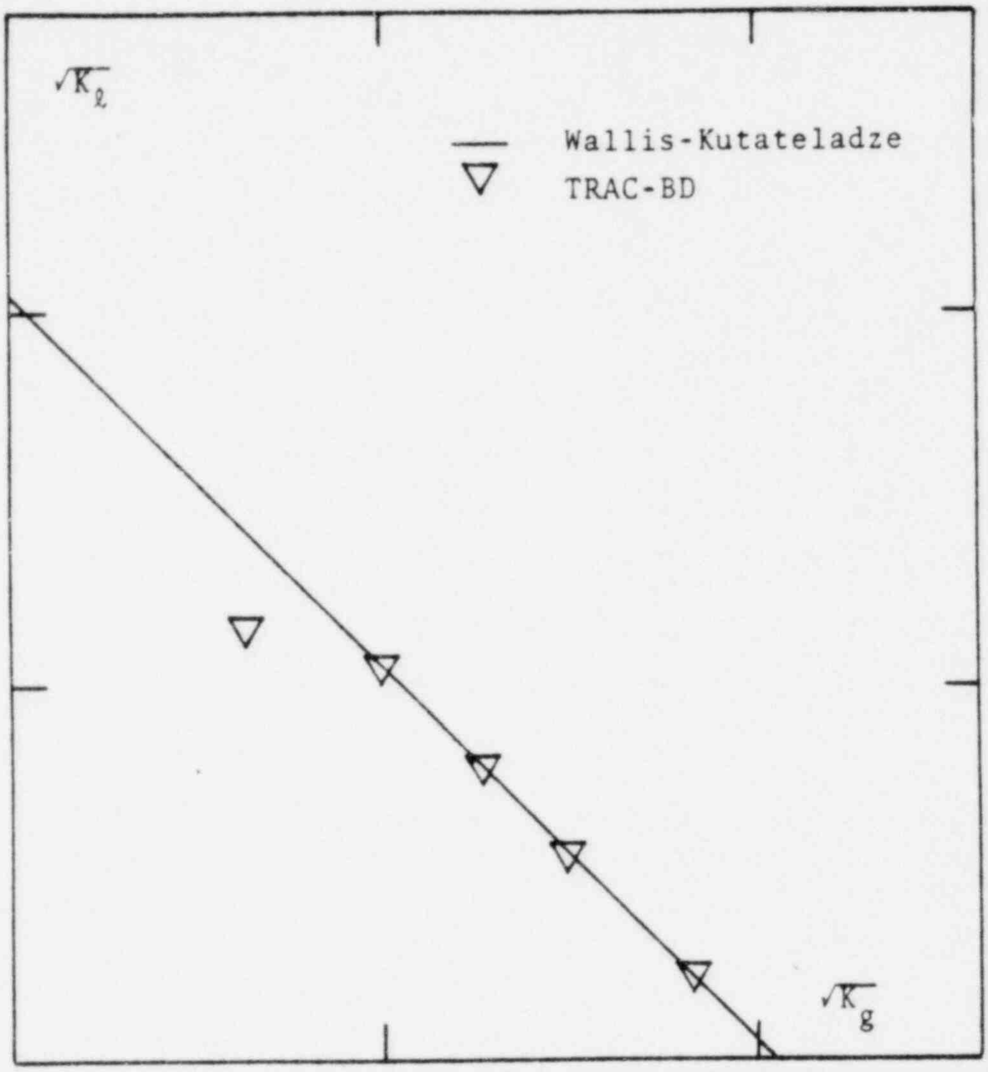
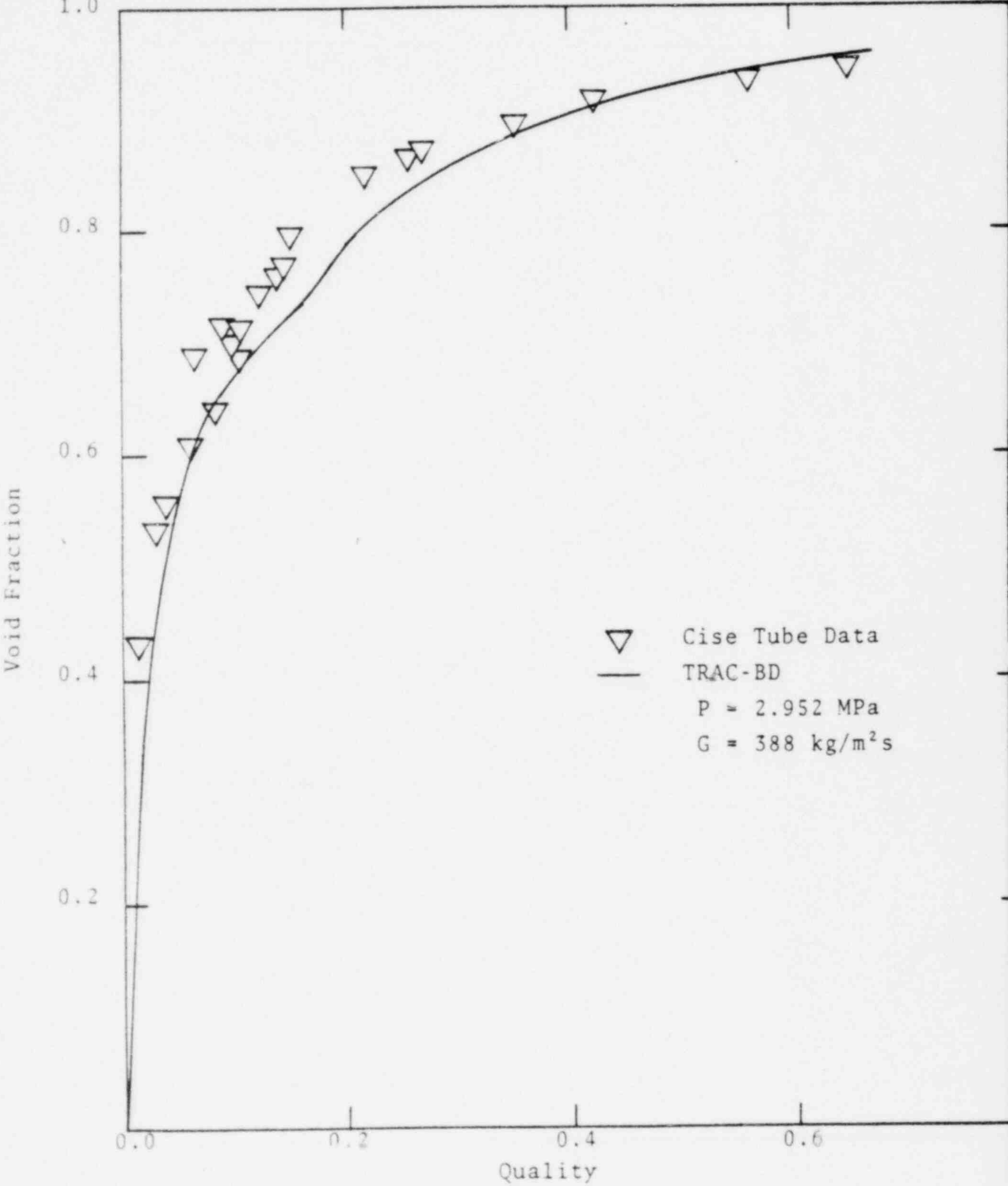
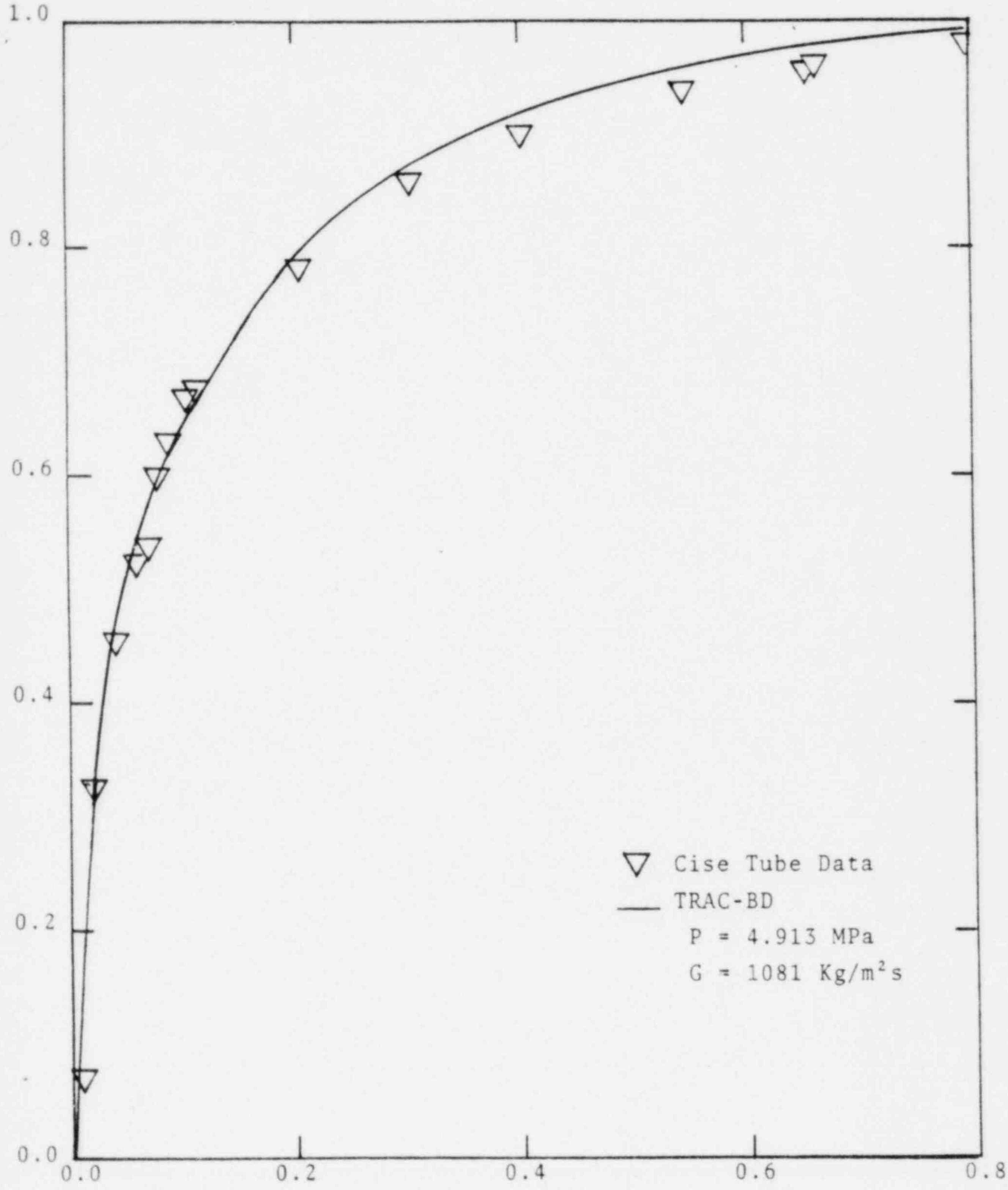


CCFL

Preliminary Assessment of TRAC-BD Models



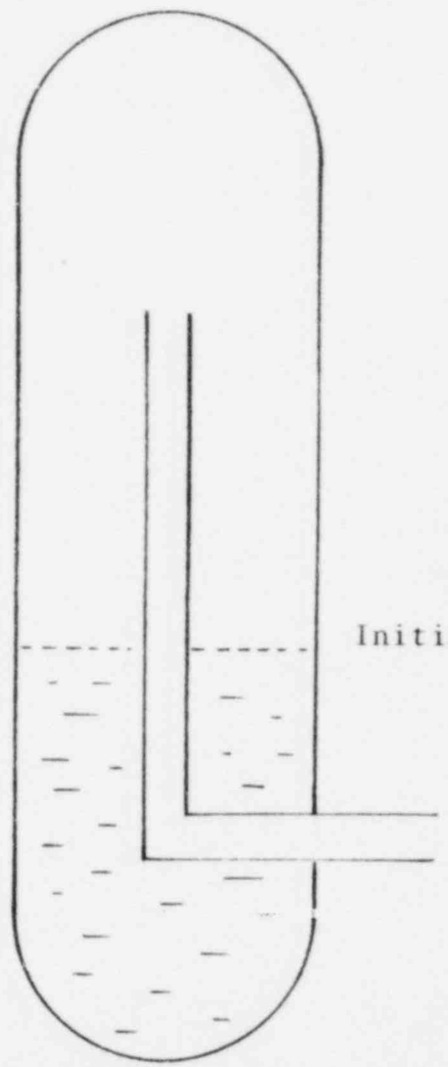
Interface Shear
Preliminary Assessment of TRAC-BD Models



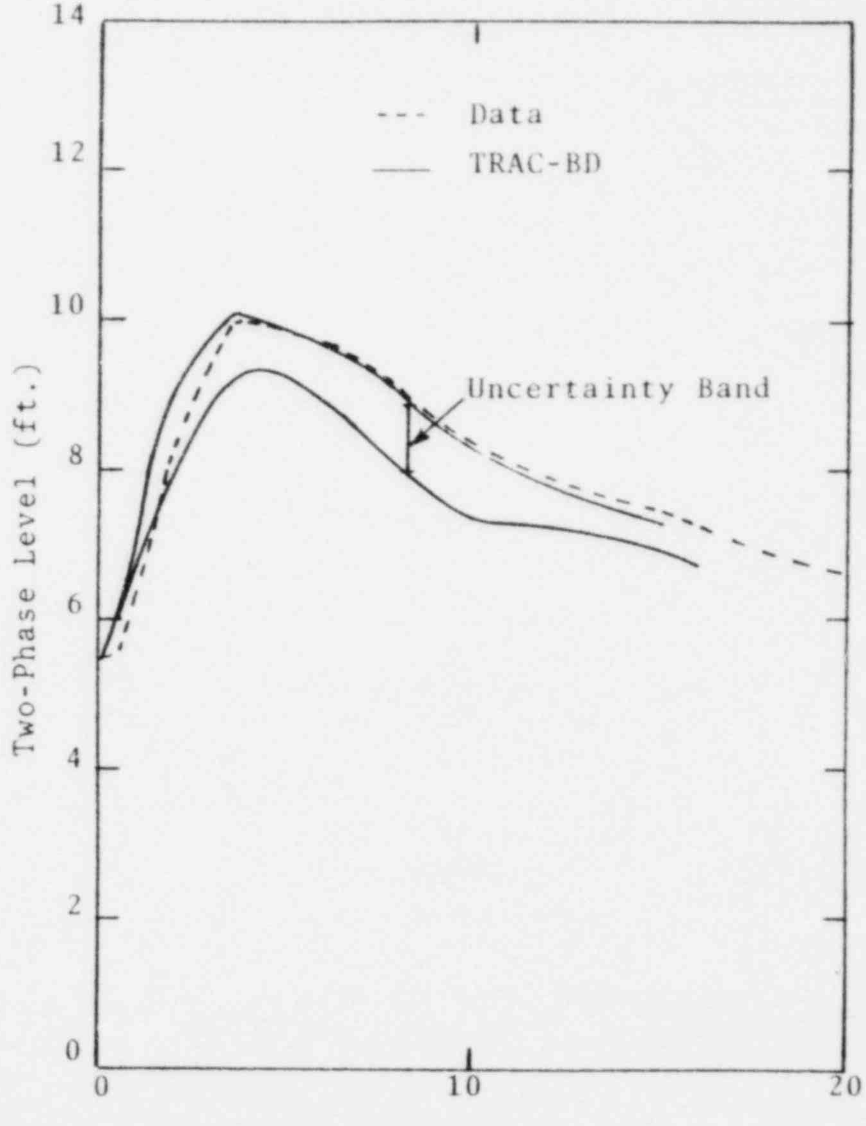
Quality

Interface Shear

Preliminary Assessment of TRAC-BD Models



Level Swell Test

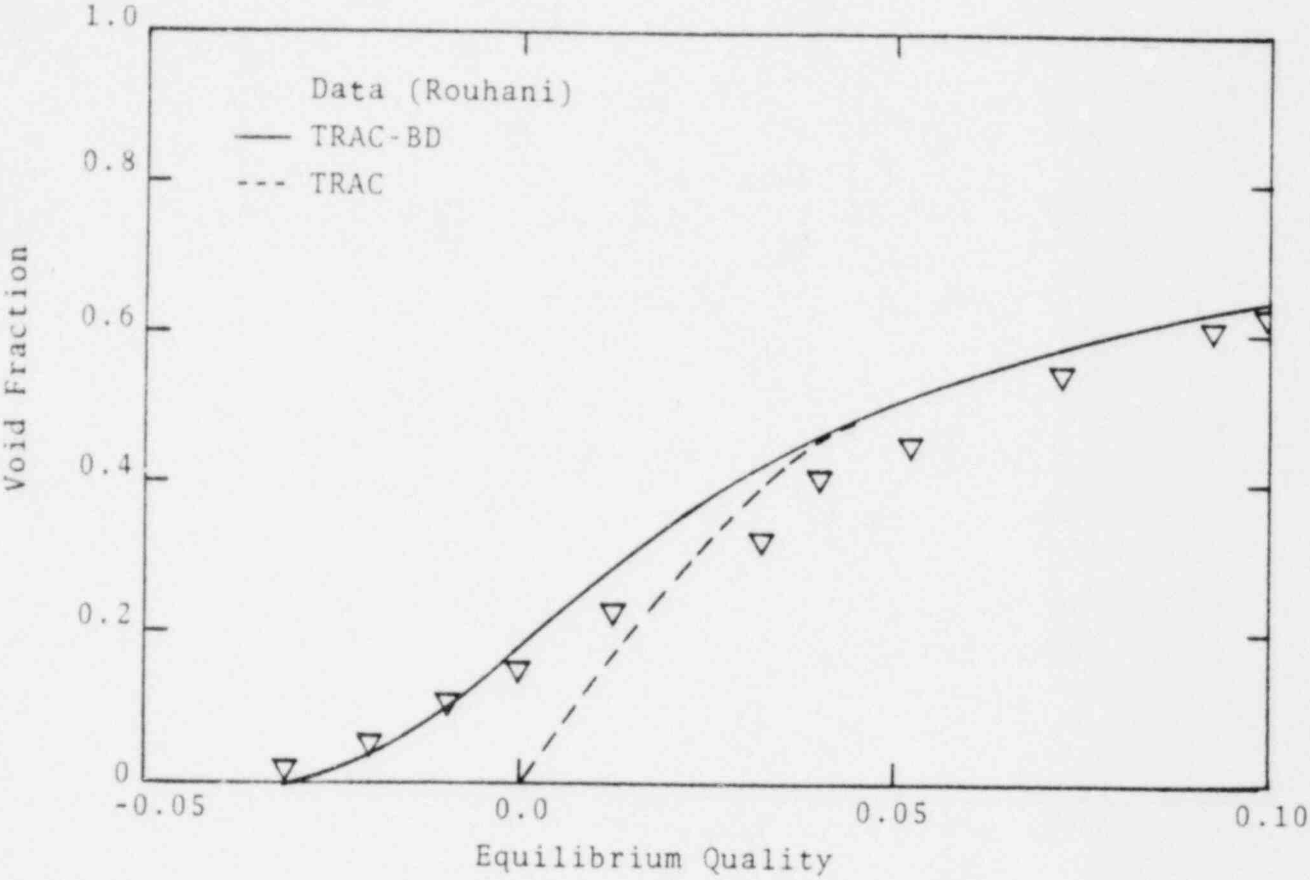


Preliminary Assessment of TRAC-BD Models

HEAT TRANSFER IMPROVEMENT

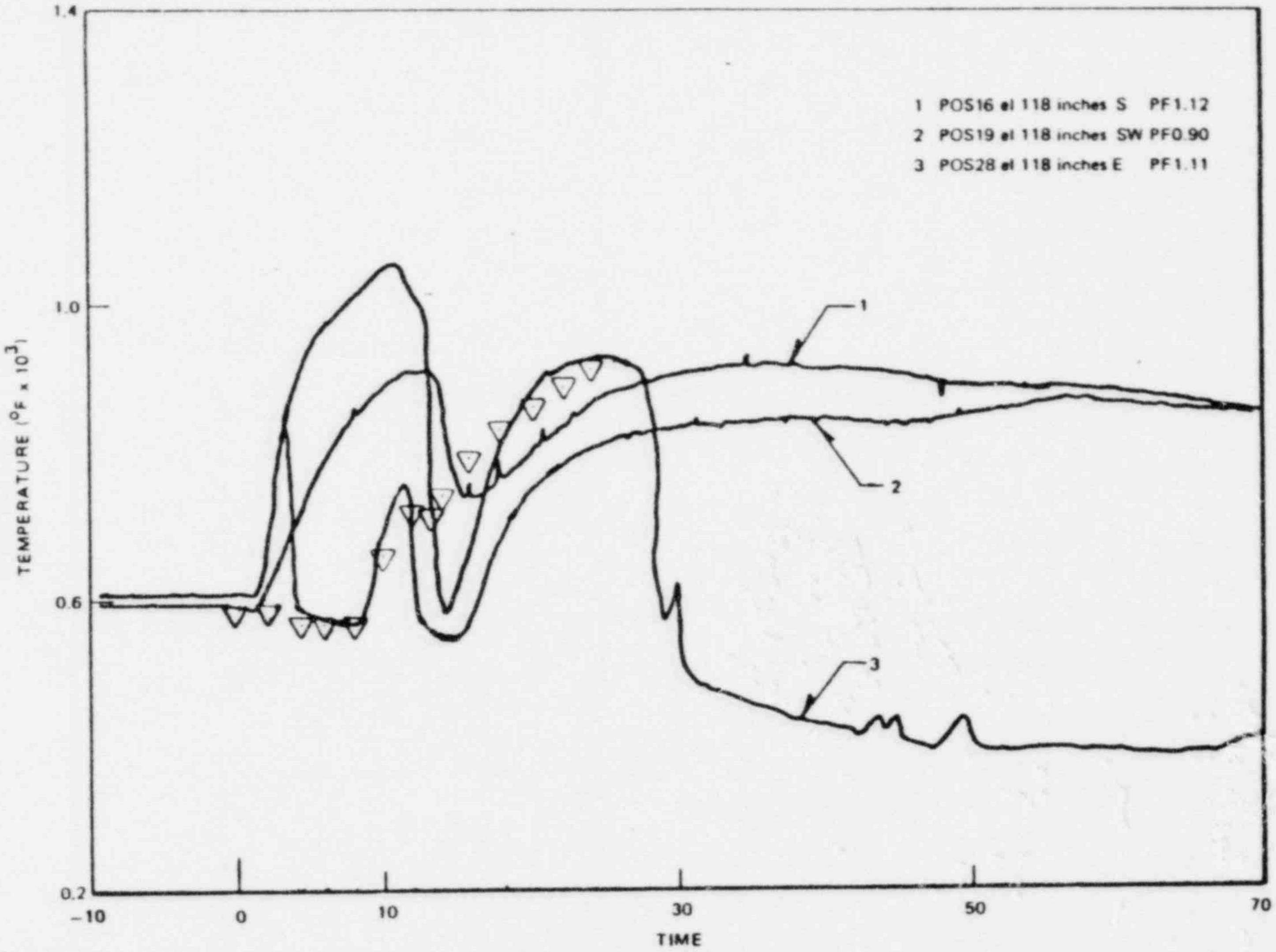
- Subcooled Boiling
  - Mechanistic Heat Transfer Model (Bowring, Rouhani)
  - Net Vapor Generation (Saha-Zuber)
- Critical Heat Flux
  - Boiling Length Correlation (CISE-GE)
- Thermal Radiation
  - All Surfaces are Grey
  - All Surfaces have Uniform Temperature
  - All Surfaces Emit Radiation Uniformly
  - Two-Phase Flow Absorbs and Emit Radiation
  - Semi-Gray Model
  - First Order Anisotropic Transport Correction

JGM Andersen
October 1980

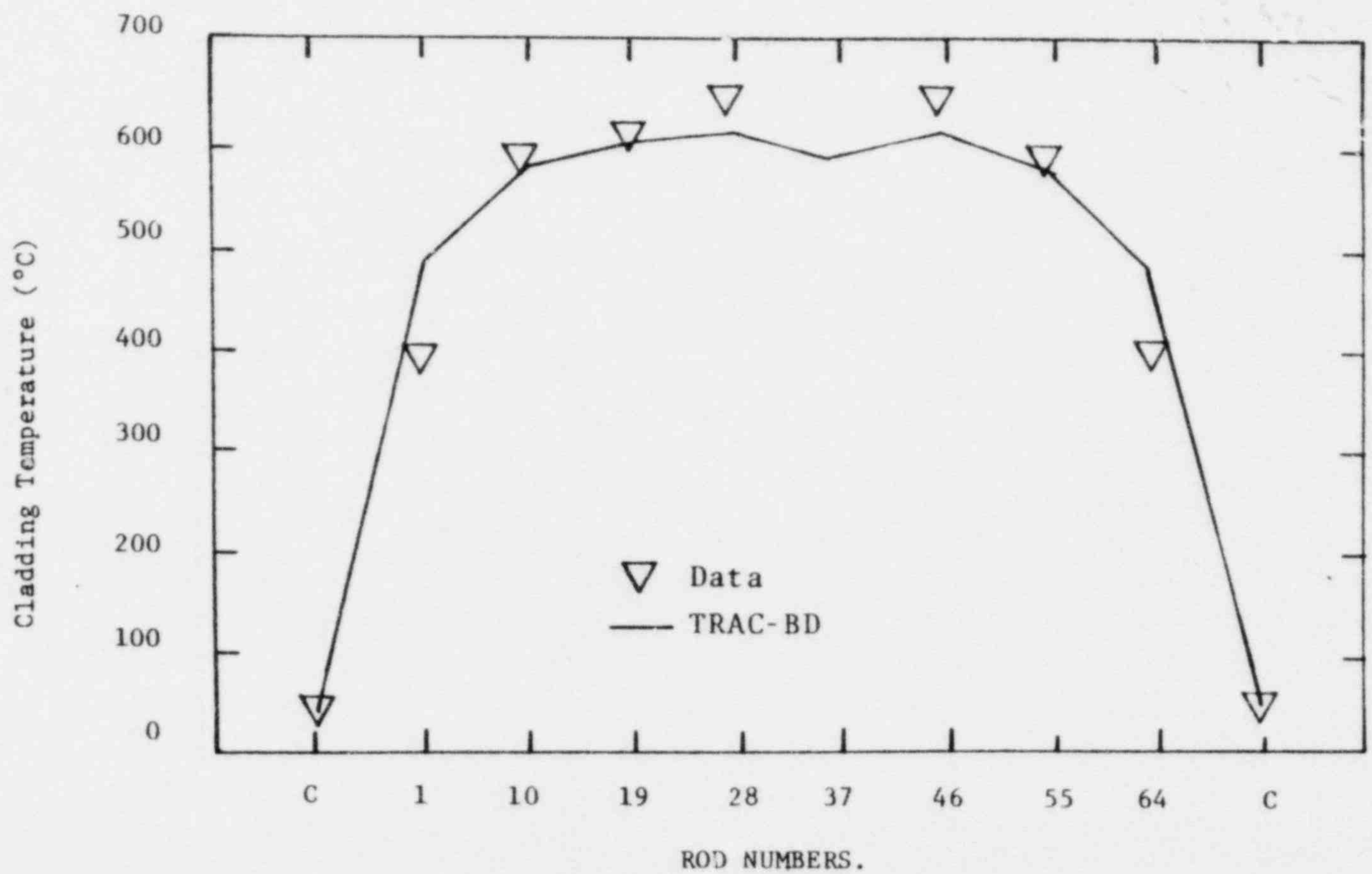


SUBCOOLED BOILING

Preliminary Assessment of TRAC-BD Models



Test 4904 Run 45



Radiation Calculation
Assessment of TRAC-BD Models

STATUS

- Development of first version of component and phenomena models of TRAC BD complete.

- Testing and developmental assessment through 1980.

- Independent assessment of models in 1981.

- Development of final models in 1981.

JGM Andersen
October 1980

BWR REFILL-REFLOOD PROGRAM

MODEL DEVELOPMENT FOR TRAC-BD

JGM Andersen

GENERAL ELECTRIC COMPANY
NUCLEAR FUEL AND SERVICES ENGINEERING DEPARTMENT
SAN JOSE, CALIFORNIA

FOR PRESENTATION AT:

The 8th Water Reactor Safety Research Information Meeting
October 29, 1980
National Bureau of Standards
Gaithersburg, Maryland

PROGRAM SPONSORS:

US NUCLEAR REGULATORY COMMISSION
ELECTRIC POWER RESEARCH INSTITUTE
GENERAL ELECTRIC COMPANY

BWR REFILL-REFLOOD PROGRAM

MODEL DEVELOPMENT FOR TRAC-BD

JGM Andersen

A part of the Refill-Reflood Program is concerned with the development of models for the BWR version of TRAC. The main goal is to develop models that allow a best estimate simulation of all components and phenomena in a BWR system. The BWR components to be modelled are:

- Jet Pump
- Steam Separator
- BWR Fuel Bundle (done by EG&G)
- Steam Dryer
- Upper Plenum Phenomena

The basic phenomena of importance for the BWR LOCA transient are

- Interface shear and wall friction
- Interface and wall heat transfer
- Entrainment and deposition

The development of models for BWR components and phenomena is being done in two steps:

- Development of an intermediate set of component and phenomena models that allow a reasonably good simulation of a BWR LOCA transient. This task is to be completed in 1980.
- Assessment of the intermediate models and development of final models for a best estimate BWR-LOCA simulation. This task is to be completed in 1982. Furthermore, in this period the models are to undergo independent qualification.

A jet pump model, including appropriate losses (form, mixing and other irreversible), has been developed. This model has been used to predict small scale as well as full scale jet pump performance with good results. Simple models for the steam separators and dryers, giving full separation of the phases, have also been developed.

A new methodology that allows the correlation of interface shear and wall friction based on void fraction and pressure drop data has been developed. The main new feature is that the model for the interface drag and shear accounts for the effect of the phase and velocity distribution in the calculation of the average relative velocity. Furthermore, the interfacial force accounts for the effect of drag, phase distribution and wall friction. Based on this model a new set of constitutive correlations for the interface shear and drag has been developed. The model has been tested against void fraction data, with good results. The heat transfer models have been upgraded to include several new phenomena. The major improvements are the inclusion of subcooled boiling and thermal radiation. The latter is particularily important for BWR's during spray cooling. Furthermore, the critical heat flux correlation has been replaced with a boiling length correlation, which improves the prediction of early boiling transition in high power fuel bundles during a DBA-LOCA.

The above accomplishments represent the scope of the intermediate step in the model development. The modesl will undergo an extensive developmental assessment during the remainder of 1980. The development of the final models will start in 1981.

NRC/EPRI/WESTINGHOUSE
FULL LENGTH EMERGENCY CORE HEAT TRANSFER
SEPARATE EFFECTS AND SYSTEMS EFFECTS TESTS
(FLECHT-SEASET):

161-UNBLOCKED BUNDLE
21-ROD BUNDLE FLOW BLOCKAGE
RESULTS

Presented by
L. E. Hochreiter
Westinghouse Nuclear Safety Department
Nuclear Technology Division

At
Eighth Water Reactor Safety Information Meeting
October 29, 1980

The FLECHT-SEASET program is a NRC/ Electrical Power Research Insititute (EPRI)/Westinghouse cooperative research and development effort who's goal is to improve our understanding of large break LOCA reflood phenomena as well as the different natural circulation cooling modes typical of small break LOCA. The detailed objectives of the program are given in Figure 1. The FLECHT-SEASET program can be subdivided into two major subtasks each of which addresses current PWR licensing and research needs. Those subtasks are, rod bundle flow blockage; and system response during reflood and natural circulation.

The rod bundle flow blockage program utilizes three separate experimental programs as shown in Figure 2. The goal of flow blockage portions of the FLECHT-SEASET program is to provide experimental data and analysis which can be used to address the current Appendix K steam cooling-flow blockage rule at low flooding rates. This particular portion of the program has received more attention due the concerns raised in NUREG-0630 on LOCA burst strain and blockage models used by the vendors and NRC staff.

The systems effects test portion of the FLECHT-SEASET program also utilizes three separate experiments as shown in Figure 3. The steam generator separate effects test results were discussed at last year's information meeting. The upper plenum flooding tests will be conducted at INEL to obtain the flooding behavior of the FLECHT-SEASET upper plenum. The systems effects test facility is presently under construction and testing will be initiated during the second quarter of 1981. Natural circulation cooling modes will be investigated in this scaled facility which utilizes two, full height, well instrumented multitube steam generators. Single phase, two-phase, and reflux condensation cooling modes will be investigated. The effects of different secondary side heat sinks, non-condensible gas injection and ECC injection will be investigated on the stable cooling modes for the primary system. Large break LOCA reflood systems effects tests will also be conducted in the same facility.

In this presentation, the results from the 161-unblocked bundle tests as well as the 21-rod bundle flow blockage tests will be discussed in detail. For those interested, a listing of all the published FLECHT-SEASET reports is given in Table 1.

A.) 161-Unblocked Bundle Program Results

The objectives of the 161-Unblocked bundle test program are given in Figure 4.

The program included heat transfer experiments on, forced reflooding, gravity feed reflooding, steam cooling, and bundle boil-off experiments. The cross-section of the 161-rod bundle is shown in Figure 5 and the flow schematic of the facility is shown in Figure 6. Examples of the forced flooding reflood heat transfer, clad temperature response, and measured vapor superheat temperature are shown in Figures 7 and 8 for a low flooding rate experiment (1"/sec) and a high flooding rate test (6"/sec) to indicate the differences in heat transfer regimes. The low flooding rate data (<1.5in/sec) are primarily in the dispersed flow heat transfer regime in which the heat transfer mechanisms include radiation to surfaces, drops, and vapor; as well as forced convection to steam flow. In the high flooding rate heat transfer regime, the heat transfer mechanisms are inverted annular film boiling and radiation to the liquid core. The lower flooding rate regimes are of more interest since the heat transfer is lower and it is within this regime that the calculated LOCA peak clad temperature occurs. Two basic approaches have been used to analyze the data from this test series. The first results in the development of an empirical correlations for the rod bundle quench and the heat transfer above the quench front. The second approach is to perform a mass and energy balance above the quench front to obtain the split in the local rod heat flux in to the different heat transfer mechanisms such as radiation to drops, vapor, surfaces, and forced convection to superheated steam.

In the first approach, an empirical quench front and heat transfer correlation has been developed. This correlation, while it is empirical, has been formulated using non-dimensionless physical parameters and does fit the FLECHT 15x15 cosine power shape, FLECHT 15x15 skewed power shape, and present FLECHT-SEASET 17x17 cosine power shape data. Examples of the correlations and data are given in Figure 9 - 11 for tests which preserve the same integral of power to flow ratio. An example of how the data overlaps is given in Figure12 which indicates that the integral of power method correlates the data to a reasonable degree.

The more detailed analysis of the data is currently still in progress at this writing. Mass and energy balance calculations have been performed for the key experiments such that the non-equilibrium quality and equilibrium quality can be calculated from the data. The assumptions used in these calculations have been presented at previous information meeting and are given in the FLECHT and FLECHT-SEASET reports. However, the key measurement needed to obtain the energy split between vaporization and superheating is the non-equilibrium vapor temperature. Examples of the non-equilibrium vapor temperature distribution for the 1"/sec test is shown in Figure 13. The calculated non-equilibrium and equilibrium quality is given in Figures 14 and 15. The difference between the qualities indicates the large amount of energy stored in the vapor as superheat.

Using the measured flows and calculated qualities, a model for the droplet motion, and a radiation heat transfer network, the individual wall heat flux components can be evaluated as shown in Figure 16. The key parameters are the droplet size and velocity assumed at the transition/dispersed flow interface. Previous FLECHT studies have used drop sizes estimated from movies. Droplet photography received more attention in the FLECHT-SEASET program and high quality, high speed movies were taken using techniques developed at NASA Lewis. Droplet velocities and sizes were obtained from the movies and a droplet spectrum was used to calculate the droplet radiation heat flux component. An example of the droplet velocity and size data is shown in Figure 17. The split of the heat flux components are shown in Figure 18 for the droplet spectrum and a sauter mass mean droplet size.

Additional data will be analyzed in a similar fashion to examine flooding rate and pressure effects. The resulting heat flux splits are directly related to the relative importance of the different heat transfer mechanisms and can be used for best estimate code assessment.

Pure steam cooling tests were also conducted in the 161-rod bundle tests at low vapor Reynolds numbers, typical of small break LOCA conditions. The data was analyzed using both a one-dimensional and subchannel energy balances to obtain the vapor temperature. Only steady-state data was utilized to simplify the analysis. The resulting data was correlated and compared to the Dittux-Boelter correlation and is shown in Figure 19. The data lies significantly above the Dittus-Boelter correlation at low Reynolds numbers and merges with the Dittus-Boelter correlation at higher Reynolds numbers. This data trend is consistant with other single phase data in rod bundles with a pitch-to-diameter ratio of 1.33.

B.) 21-Rod Bundle Flow Blockage Program.

The objectives of the 21-rod bundle flow blockage program are given in Figure 20. A loop schematic for the 21-rod bundle test facility is shown in Figure 21 and a cross section of the test section is shown in Figure 22. Currently a total of seven blockage configurations will be examined in the 21-rod bundle program. A listing of the different configuration is given in Figure 23. The FLECHT-SEASET flow blockage program has been designed to compliment both the Karlsruhe FEBA program and the international 2D/3D slab core flow blockage program in Japan. Two basic sleeve shapes will be used to simulate the ballooned and burst fuel rods. The sleeves which will be used are shown in Figure 24. These sleeves will slide over heater rods which have detailed thermocouple measurements down stream of the blockage.

Vapor superheat measurements are made in the bundle by specially built subchannel steam probes and bare fluid thermocouples which are hung from the grids in both the upflow and downflow directions.

At this writing, test have been completed on the first three bundle configurations; (unblocked reference, 9 center rods blocked, all 21-rods blocked). Comparisons of the blocked configureations to the reference unblocked configurations for a 0.91 in/sec test are shown in Figures 25 and 28 for both the heat transfer and the clad temperature. The vapor temperature measurements at selected elevations for the three tests are also shown in Figure 29 and 30. While data analysis efforts on these tests are just beginning some preliminary conclusions can be drawn from the data.

It appears that the blockage acts to break up the drops either by atomization or by the additional shear caused by the steam flow acceleration through the blockage region. The result of the droplet breakup is to locally desuperheat the vapor as shown in Figure 29 immediately downstream of the blockage. The increased liquid fraction surface area increases both the steam-droplet heat transfer and the radiation to the droplets. The local turbulence downstream of the blockage is also enhanced by the flow acceleration through the blockage. The net result of these increased heat transfer mechanisms is to locally increase the rod total heat transfer coefficient as a function of the distrance from the blockage plane. This results in blocked bundle arrays with flow bypass.

In conclusion, it is felt that the FLECHT-SEASET program is generating both data and resulting analysis effects which can help validate best estimate models, provide a basis for reassessing licensing criteria, and examine small break PWR cooling modes.

TABLE 1
FLECHT-SEASET
REPORTS ISSUED TO DATE
(CONTRACT NO. NRC-04-77-127)

REPORT NO.	TITLE	AUTHOR
1	PWR FLECHT Separate Effects and System Effects Tests (SEASET) Program Plan	C. E. Conway L. E. Hochreiter M. C. Krepinevich H. W. Massie, Jr. E. R. Rosal R. C. Howard
2	PWR FLECHT-SEASET Steam Generator Separate Effects Task Task Plan Report	L. E. Hochreiter R. C. Howard M. J. Loftus W. Kavalkovich H. W. Massie, Jr. M. C. Krepinevich M. C. McGuire A. E. Tome
3	PWR FLECHT-SEASET Unblocked Bundle, Forced and Gravity Reflood Task Task Plan Report	L. E. Hochreiter C. E. Wonway C. E. Dodge M. C. Krepinevich H. W. Massie, Jr. E. R. Rosal T. E. Sobek M. M. Valkovic
4 NUREG/CR-1366	PWR FLECHT-SEASET Steam Generator Separate Effects Task Data Report	R. C. Howard M. F. McGuire L. E. Hochreiter

5 NUREG/CR-1370	PWR FLECHT-SEASET 21-Rod Bundle Flow Blockage Task Task Plan Report	L. E. Hochreiter R. A. Basel R. J. Dennis N. Lee H. W. Massie, Jr. M. J. Loftus E. R. Rosal M. M. Valkovic
6 NUREG/CR-1531	PWR FLECHT-SEASET 161-Rod Bundle Flow Blockage Task Task Plan Report	L. E. Hochreiter N. Lee M. F. McGuire H. W. Massie, Jr. M. J. Loftus M. M. Valkovic

W/NRC/EPRI FLECHT-SEASET PROGRAM

THE GOALS OF THE PROGRAM ARE TO:

- ENHANCE THE UNDERSTANDING OF THE PHYSICS OF REFLOOD PHENOMENA IN PWRS.

- AID IN THE IMPROVEMENT OR FURTHER DEVELOPMENT OF THERMAL-HYDRAULIC MODELS AND/OR COMPUTER CODES FOR THE REFLOOD PHASE IN PWRS.

- AID IN THE VALIDATION OF BEST ESTIMATED THERMAL-HYDRAULIC MODELS AND/OR COMPUTER CODES FOR THE REFLOOD PHASE IN PWRS AND AID IN IMPROVING THE UNDERSTANDING OF SAFETY MARGINS ASSOCIATED WITH CURRENT LICENSING EVALUATION MODELS AND CRITERIA.

- BROADEN THE DATA BASE FOR PWR LOCA-ECCS SAFETY EVALUATIONS TO PERMIT A COORDINATED REAPPRAISAL OF EXISTING LICENSING CRITERIA.

- PROVIDE POST TMI DATA AND ANALYSIS ON REFLUX COOLING AND NATURAL CIRCULATION.

FIGURE 1

FLECHT-SEASET TASKS

17 X 17 UNBLOCKED FLECHT TESTS

- GEOMETRY EFFECTS, DATA BASE FOR BLOCKAGE
- PROVIDE DATA FOR REFLOOD CODE DEVELOPMENT/VERIFICATION (TRAC, RELAP MOD-6)

21 - ROD BUNDLE TESTS

- ASSESS BLOCKAGE GEOMETRY AND CONFIGURATION EFFECTS
- PROVIDE DATA FOR BLOCKAGE ANALYSIS METHOD

17 X 17 BLOCKED BUNDLE FLECHT TESTS

- BLOCKAGE AND BYPASS EFFECTS
- ADDRESS CURRENT LICENSING CRITERIA
- ASSESS BLOCKAGE ANALYSIS METHOD

TO ASSESS THE APP K STEAM COOLING/FLOW BLOCKAGE RULE AND TO PROVIDE A DATA BACE FOR A RULE CHANGE

FIGURE 2

FLECHT-SEASET TASKS

STEAM GENERATOR TESTS

- OBTAIN SG HEAT RELEASE CHARACTERISTICS
- PROVIDE DATA FOR STEAM GENERATOR MODEL DEVELOPMENT/VERIFICATION (TRAC, RELAP-MOD 6)

FLECHT-SEASET (F/S) UPPER PLENUM TESTS (AT EG&G)

- OBTAIN F/S UPPER PLENUM COUNTER FLOW CHARACTERISTICS

FLECHT-SEASET SYSTEMS EFFECTS TESTS

- SYSTEM RESPONSE DURING REFLOOD
- REFLOOD SYSTEM CODE ASSESSMENT
- TMI RELATED NATURAL CIRCULATION, 2ϕ NATURAL CIRCULATION, REFLUX COOLING CONFIGURATIONS

ASSESS SAFETY MARGIN, PROVIDE DATA/ANALYSIS FOR IMPROVED REFLOOD SYSTEMS CODE, CODE VERIFICATION

FIGURE 3

UNBLOCKED BUNDLE OBJECTIVES

- AID IN THE DEVELOPMENT OR VERIFICATION OF RELFOOD CALCULATIONAL MODEL
- ESTABLISH A BASELINE DATA SET TO EVALUATE FLOW BLOCKAGE EFFECTS
- EVALUATE EFFECTS OF ROD DIAMETER AND PITCH ON HEAT TRANSFER AND PROVIDE A SINGLE FLECHT CORRELATION FOR HEAT TRANSFER
- PROVIDE DATA, EVALUATE, AND DEVELOPE A SINGLE PHASE STEAM COOLING HEAT TRANSFER CORRELATION

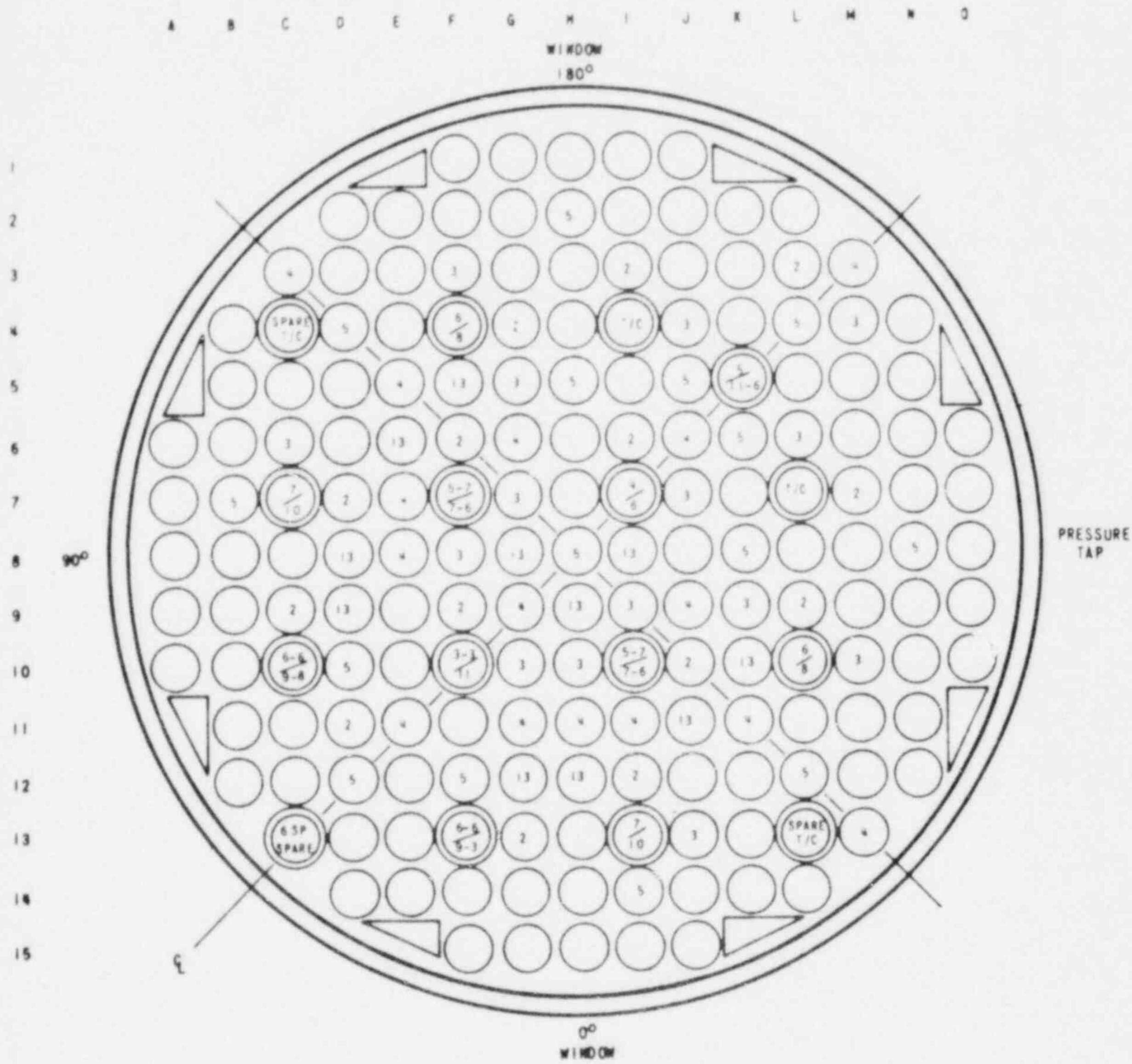
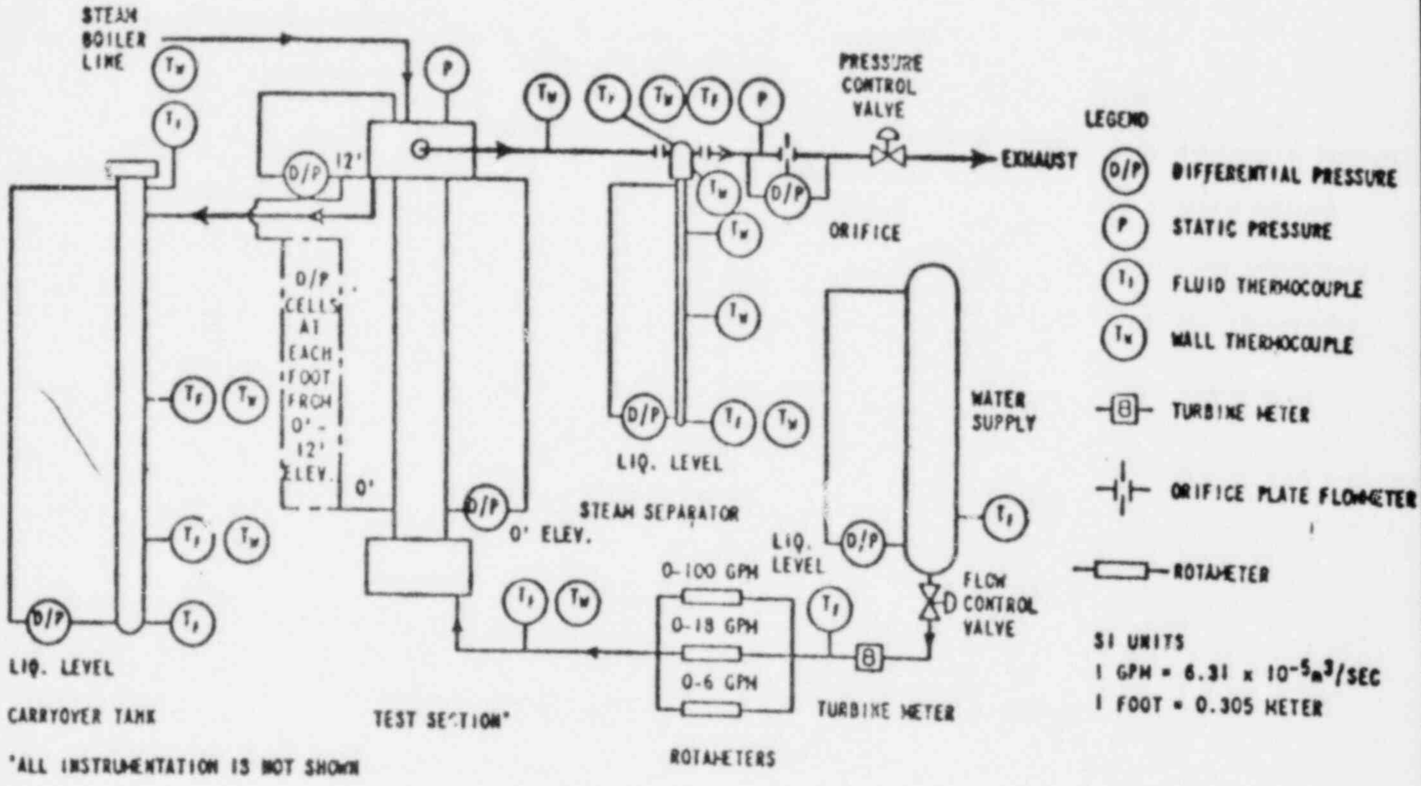


Figure 5 Cross Section 17 x 17 Unblocked Bundle Instrumentation



FLECHT-SEASET UNBLOCKED BUNDLE LOOP SCHEMATIC

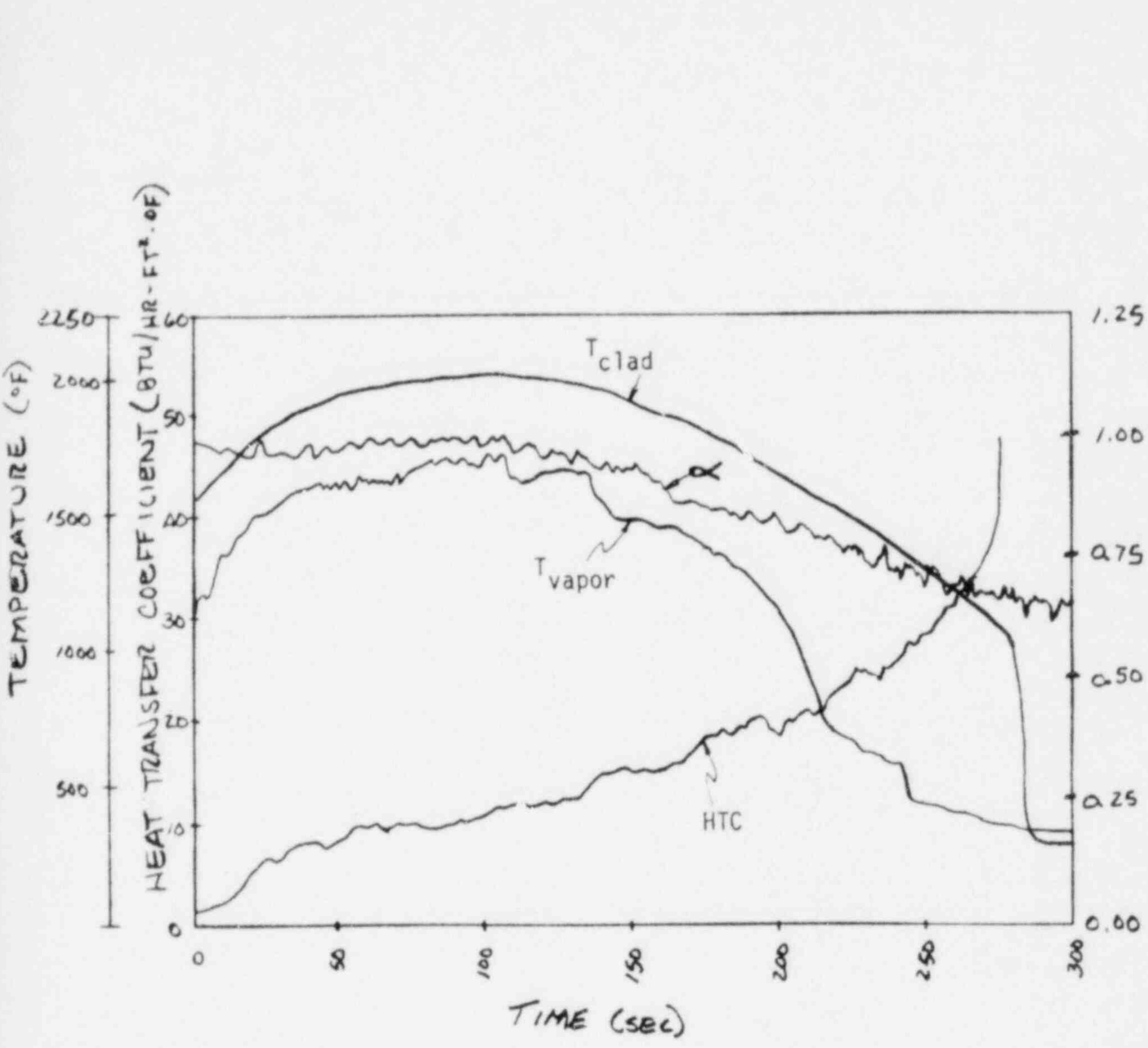


FIGURE 7. Test 31504, 1"/Sec, 40 psia

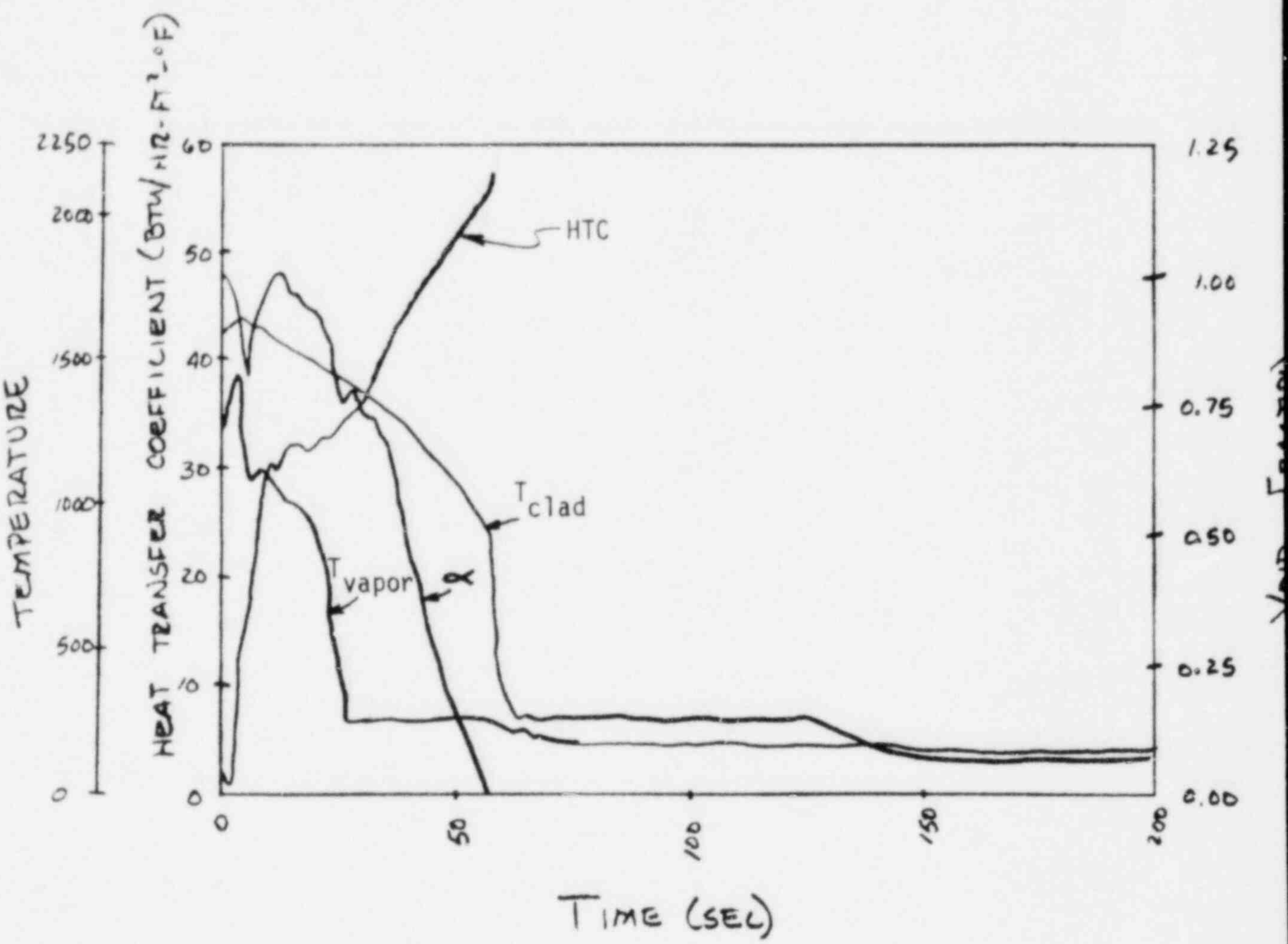
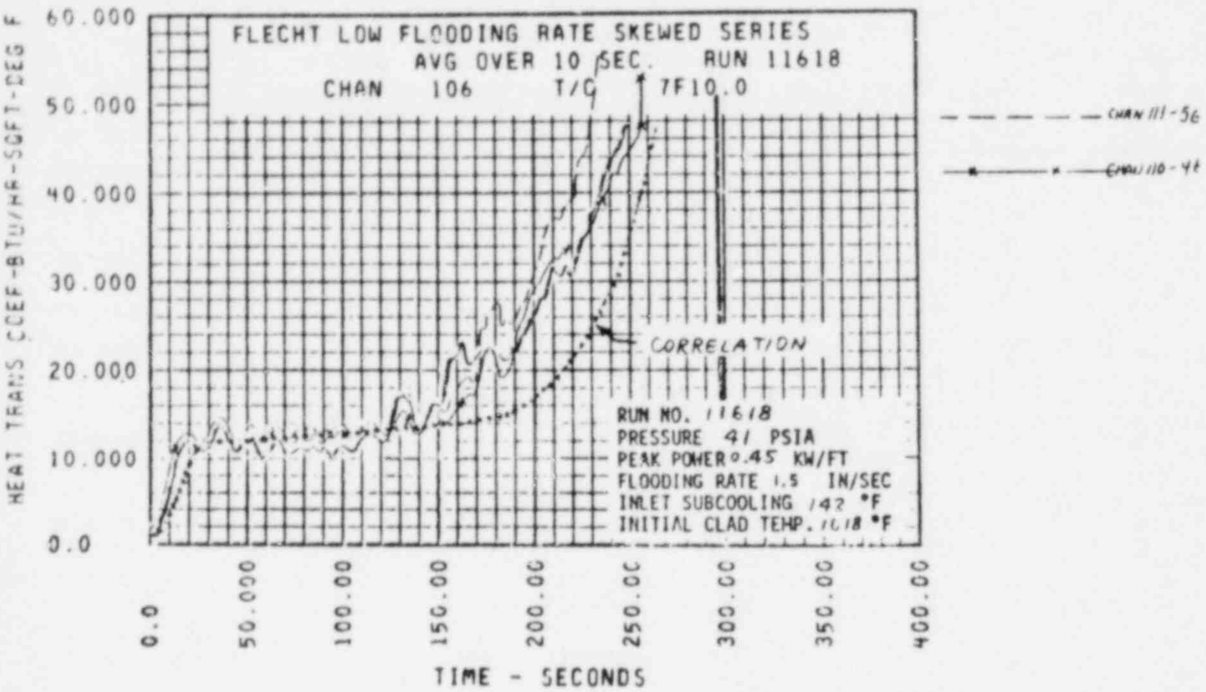
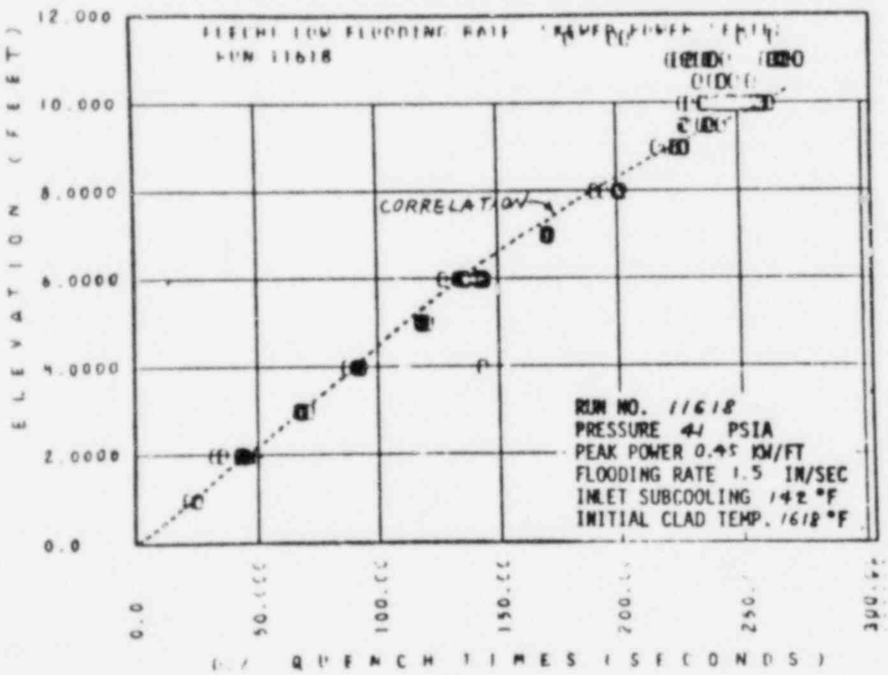


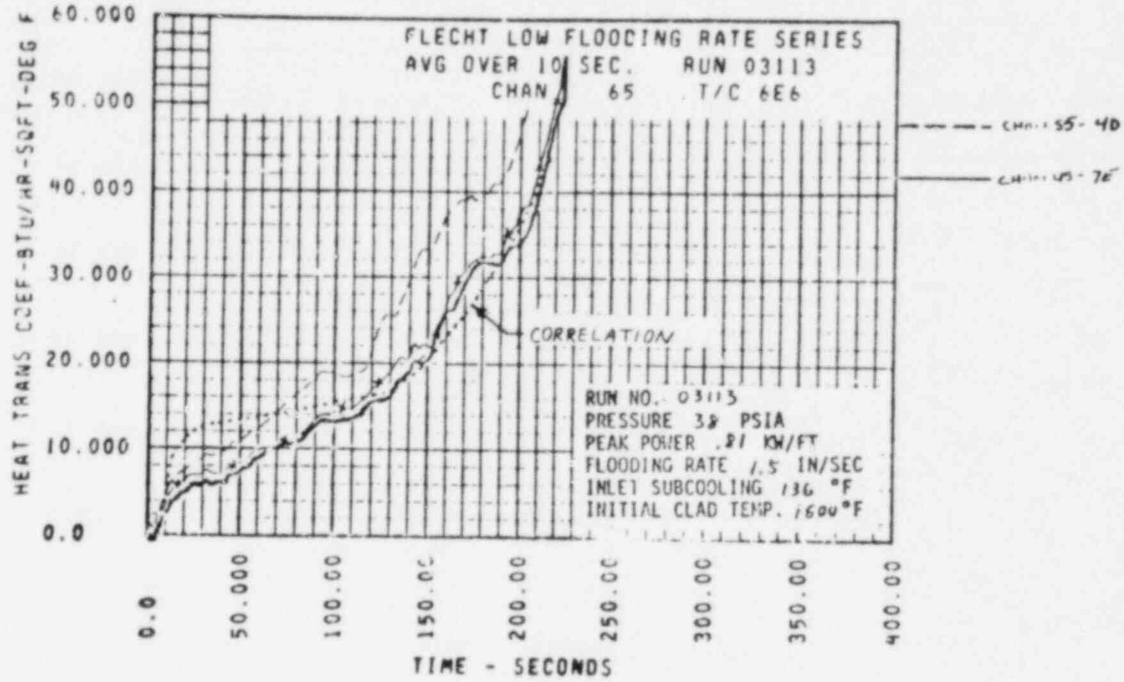
FIGURE 8
Test 3170?, 6"/Sec, 40 psia

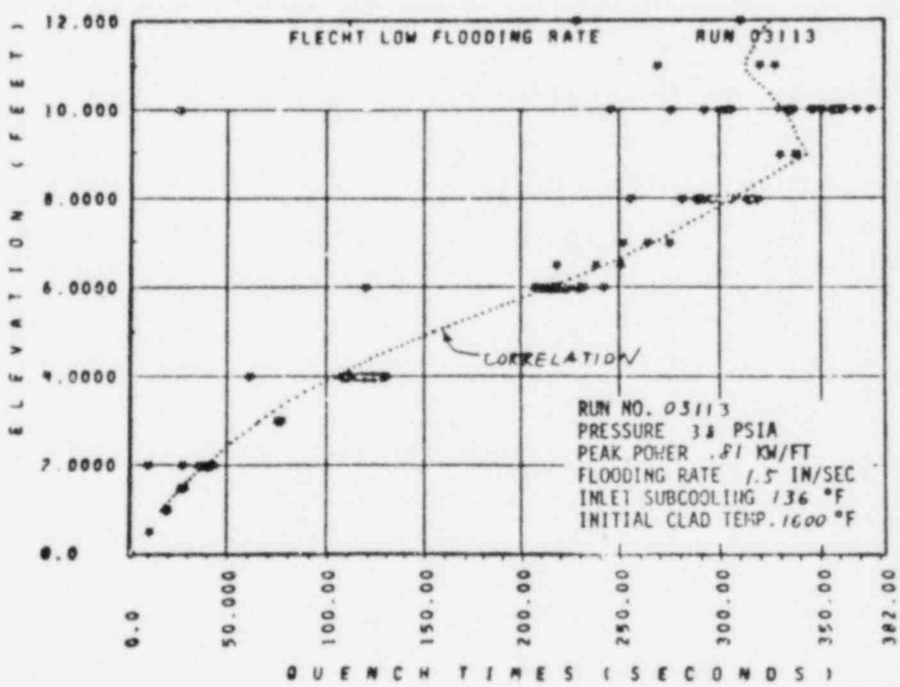




15x15 SKEWED POWER

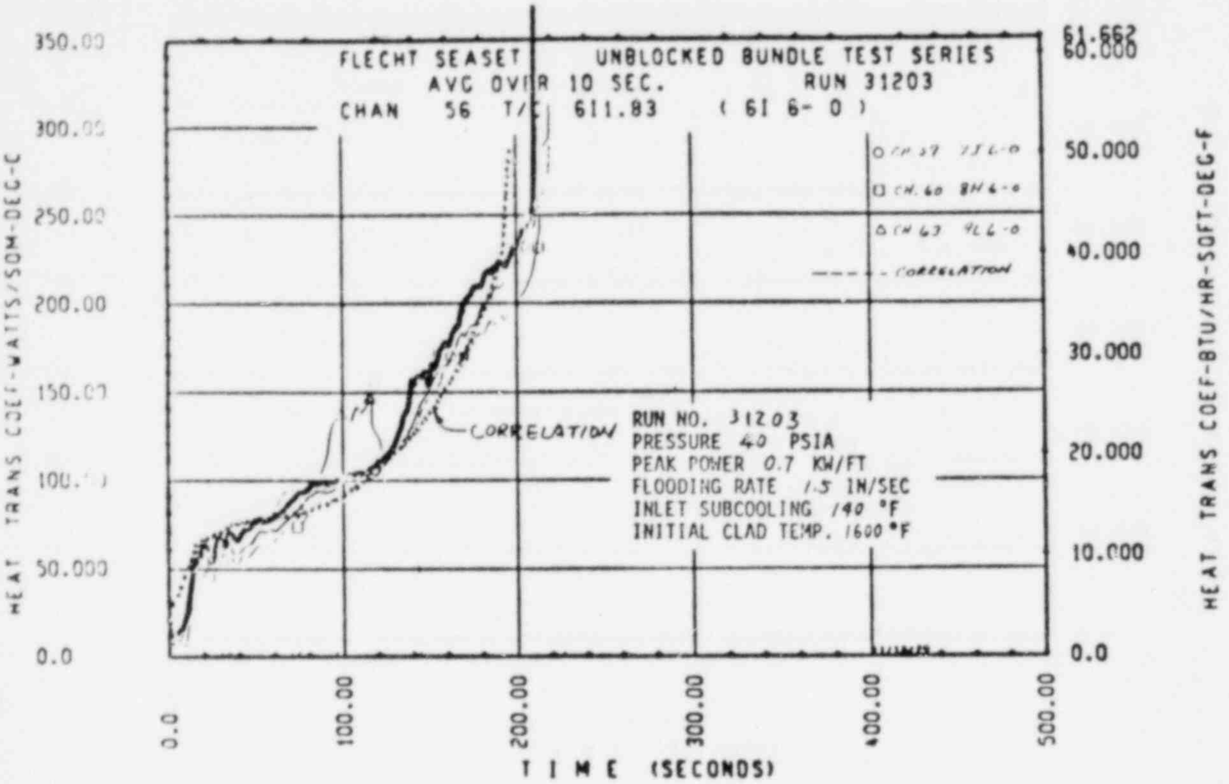
FIGURE 10

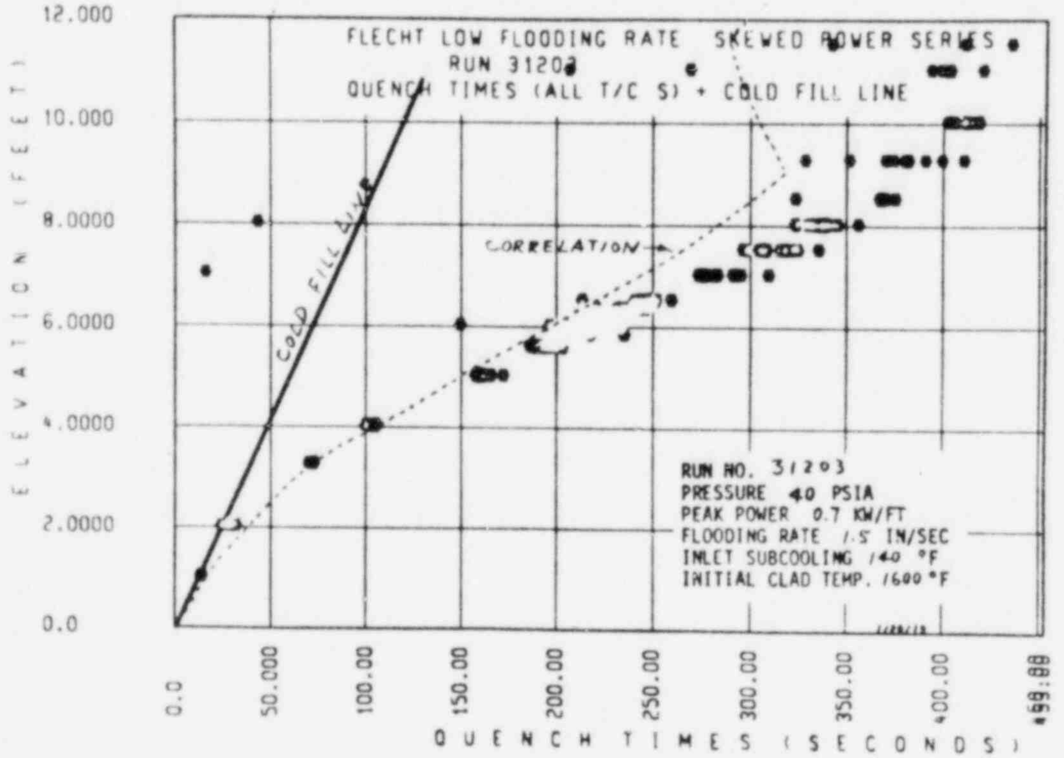




15x15 COSINE POWER

FIGURE 9





17x17 Cosine Power

FIGURE 11

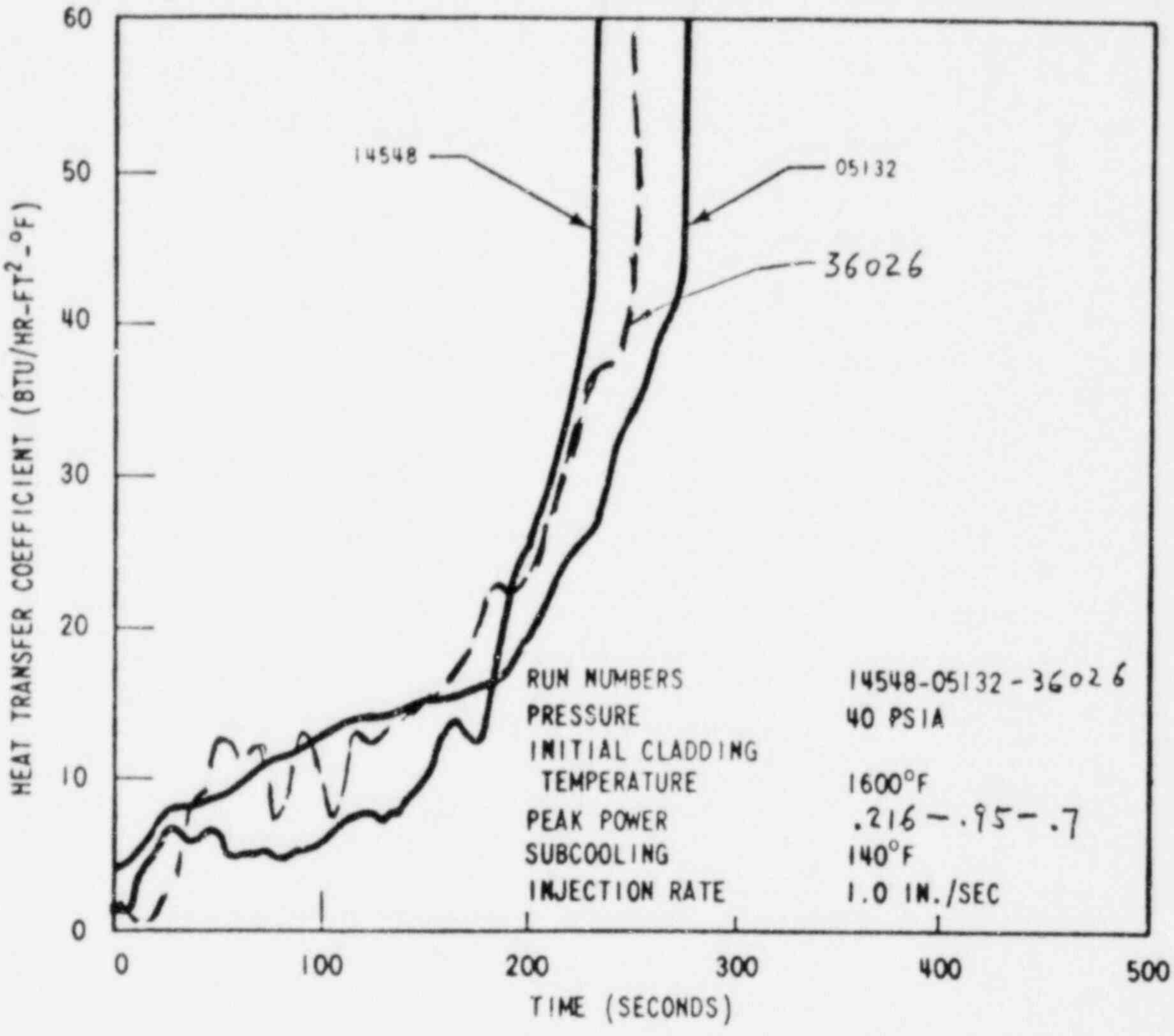


Figure 12 Heat Transfer Coefficient Versus Time for High Temperature Comparison Tests

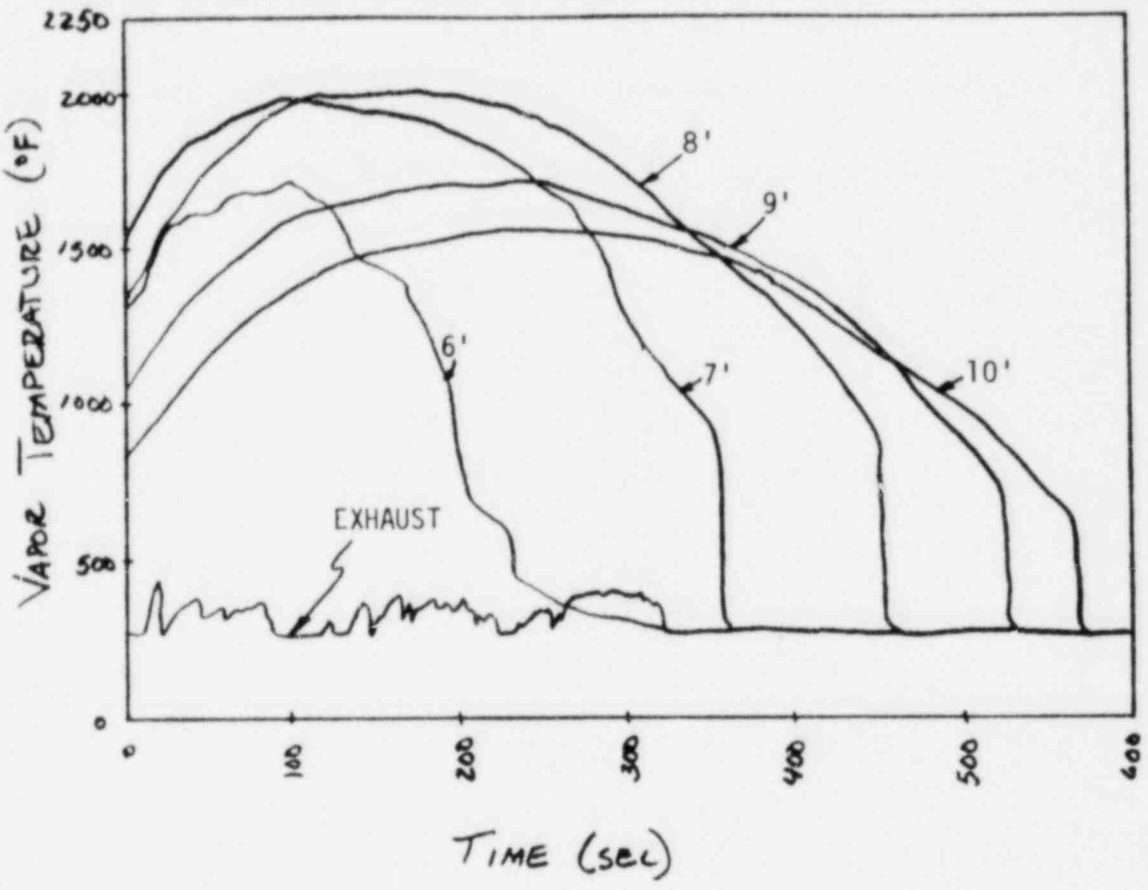


FIGURE 13
Vapor Superheat Data for Test 31504, 1"/Sec, 40 psia

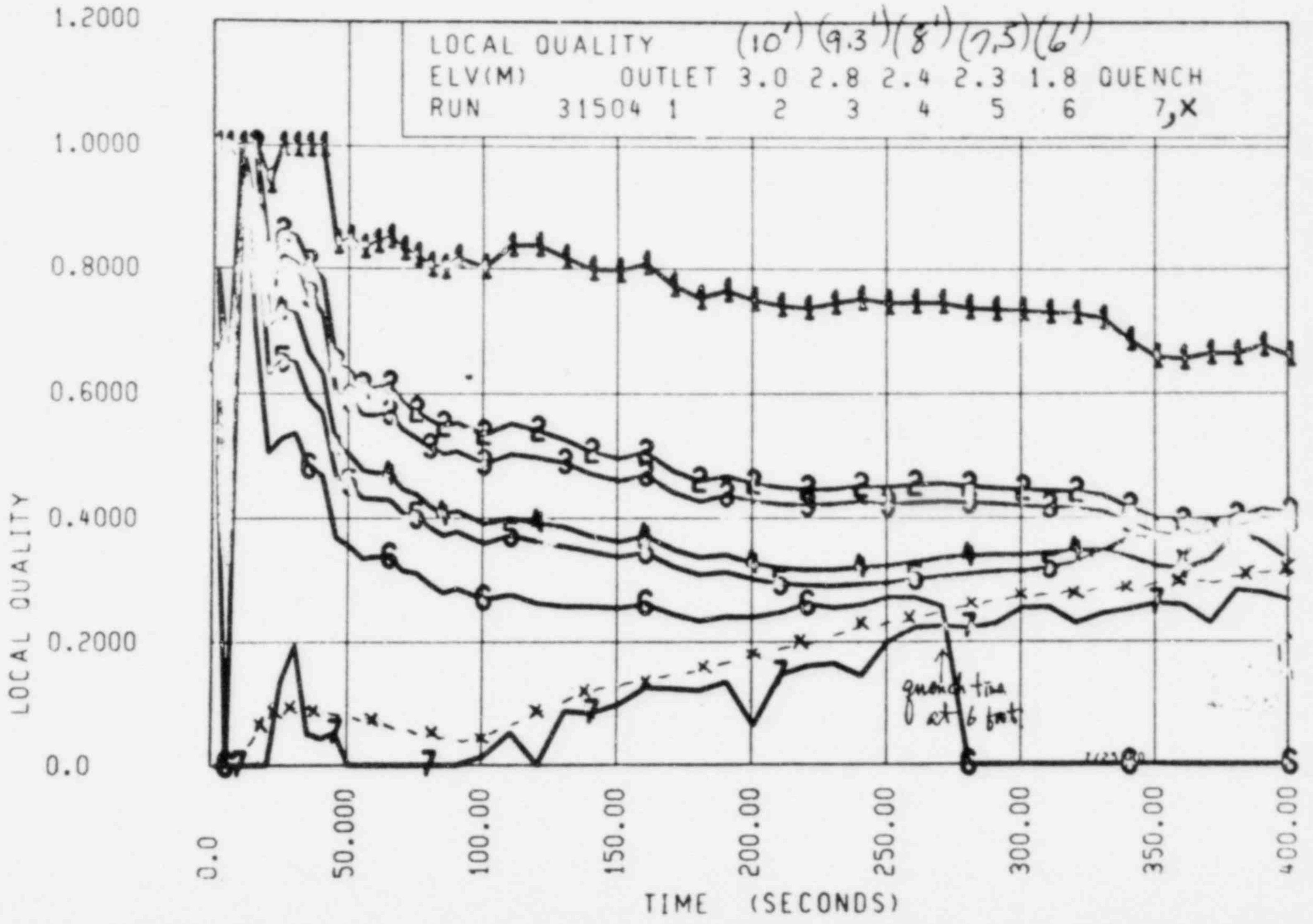


FIGURE 14

NON-EQUILIBRIUM QUALITY FOR TEST 31504 (1"/SEC)

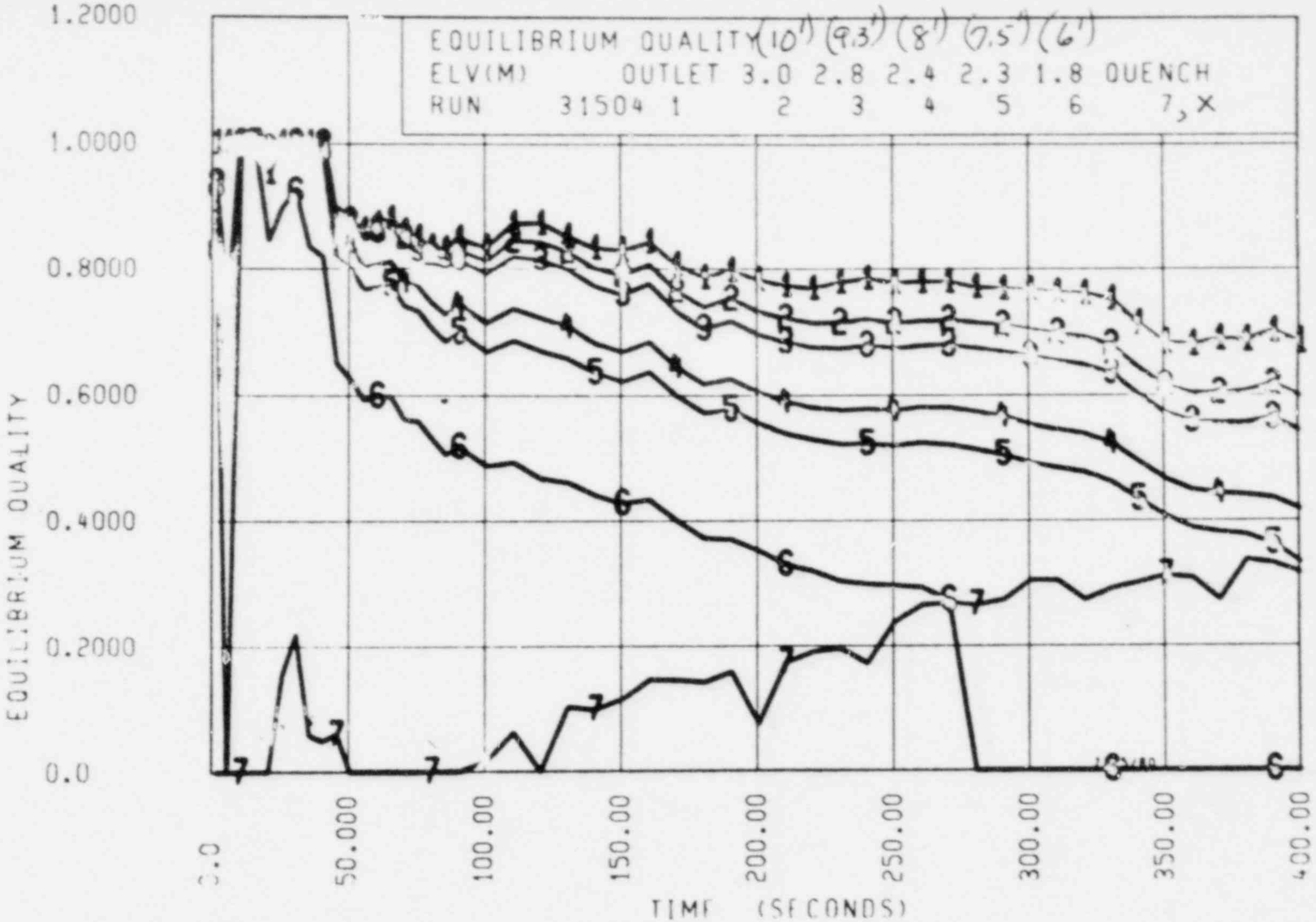


FIGURE 15
EQUILIBRIUM QUALITY FOR TEST 31504 (1"/SEC)

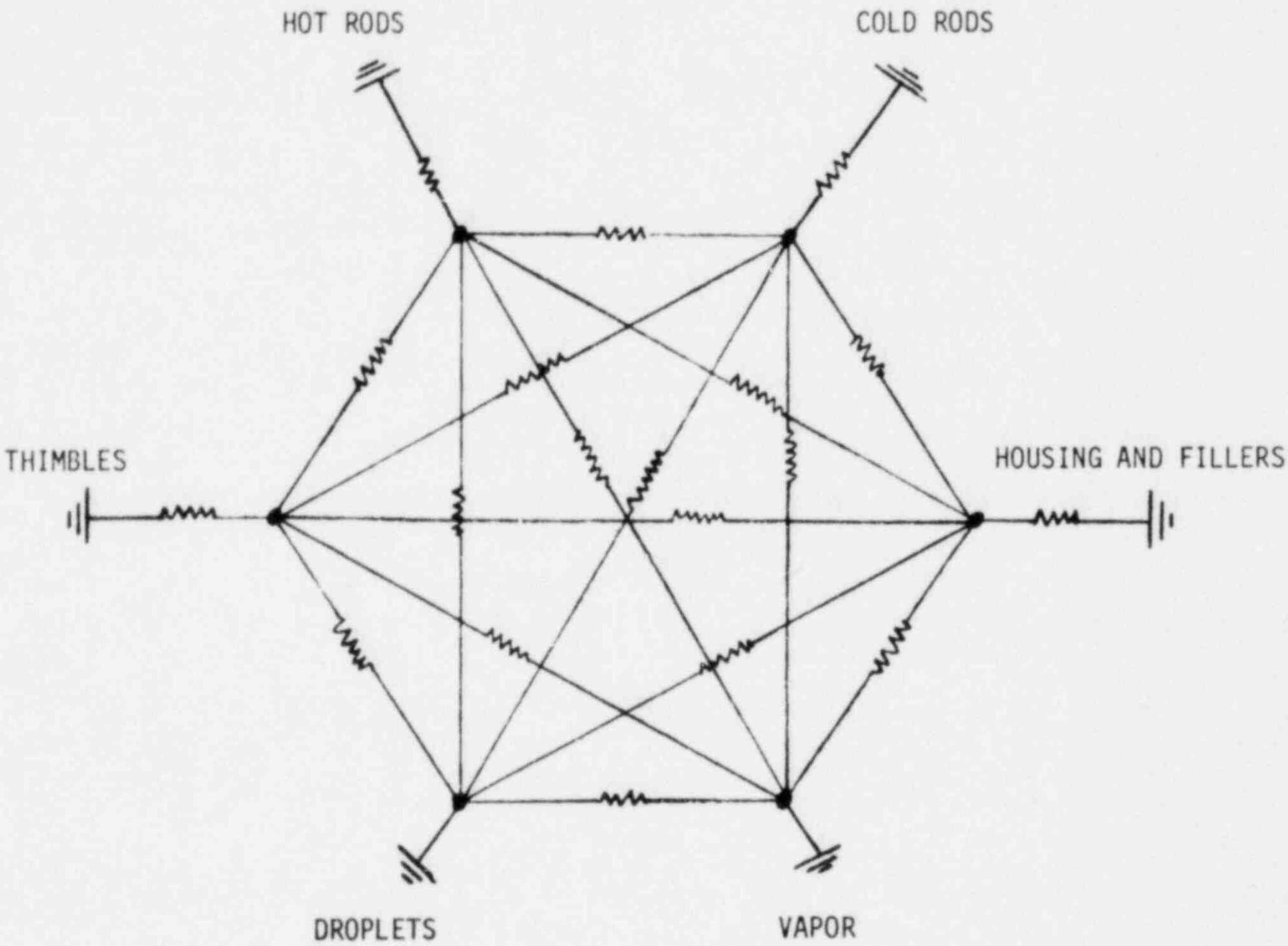
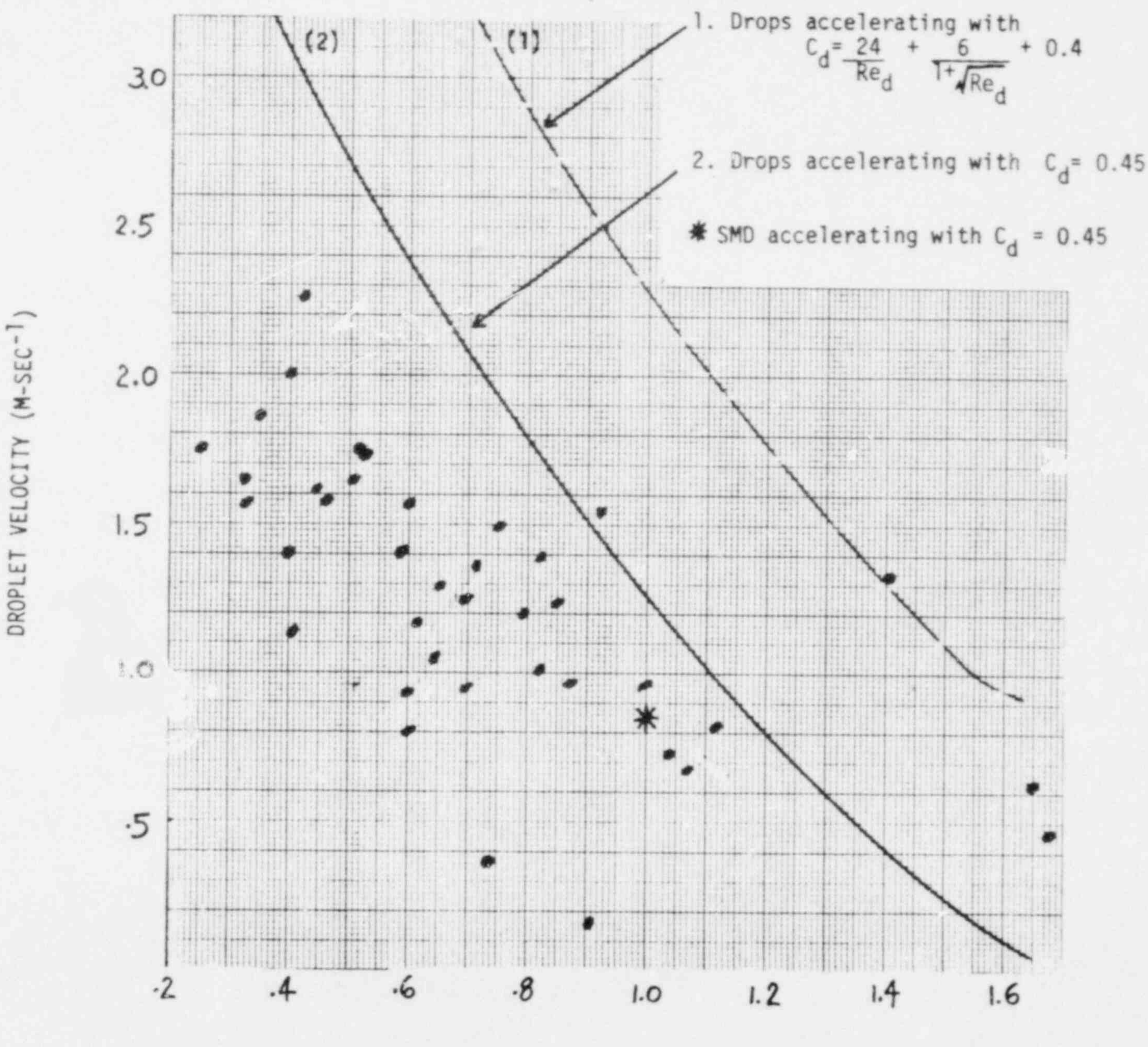


FIGURE 15 RADIATION NETWORK FOR CALCULATING RADIATION HEAT TRANSFER IN FLECHT-SEASET 161-ROD BUNDLE



1. Drops accelerating with $C_d = \frac{24}{Re_d} + \frac{6}{1+\sqrt{Re_d}} + 0.4$
2. Drops accelerating with $C_d = 0.45$
* SMD accelerating with $C_d = 0.45$
(2)
(1)
DROPLET VELOCITY (M-SEC^{-1})
3.0
2.5
2.0
1.5
1.0
.5
.2
.4
.6
.8
1.0
1.2
1.4
1.6
DROPLET DIAMETER (MM)

FIGURE 18 HEAT FLUX SPLIT

RUN 31504, 200 SEC (QUENCH FRONT AT 4.97FT)

DROPLET DATA TAKEN FROM MOVIE AT 6 FT, CALCULATED SMD AT ZA = .00327 FT.

ASSUME DROPLET ACCELERATING WITH C_D = 0.45

HEAT RELEASE	DROPLET SPECTRUM		SAUTER MEAN DROP	
(BTU/SEC-FT)	6'	10'	6'	10'
Q'_t (HOT RODS)	70.03	14.95	70.03	14.95
Q'_{cv} (HOT RODS)	52.26 (75%)	7.43 (50%)	53.80 (77%)	7.42 (50%)
Q'_r (HOT RODS)	17.77 (25%)	7.52 (50%)	16.23 (33%)	7.53 (50%)
Q'_t (COLD RODS)	46.58	12.16	46.58	12.16
Q'_{cv} (COLD RODS)	31.37 (67%)	3.63 (30%)	32.23 (69%)	3.62 (30%)
Q'_r (COLD RODS	15.21 (33%)	8.53 (70%)	14.25 (31%)	8.54 (70%)
Q'_r (THIMBLE)	-1.39	-0.66	-1.77	-0.66
Q'_r (HOUSING)	-4.78	-10.60	-5.28	-10.61
Q'_r (DROP)	-24.37 (74%)	-3.28 (20%)	-20.03 (65%)	-3.31 (21%)
Q'_r (VAPOR)	-2.46 (25%)	-1.43 (9%)	-3.46 (11%)	-1.48 (9%)

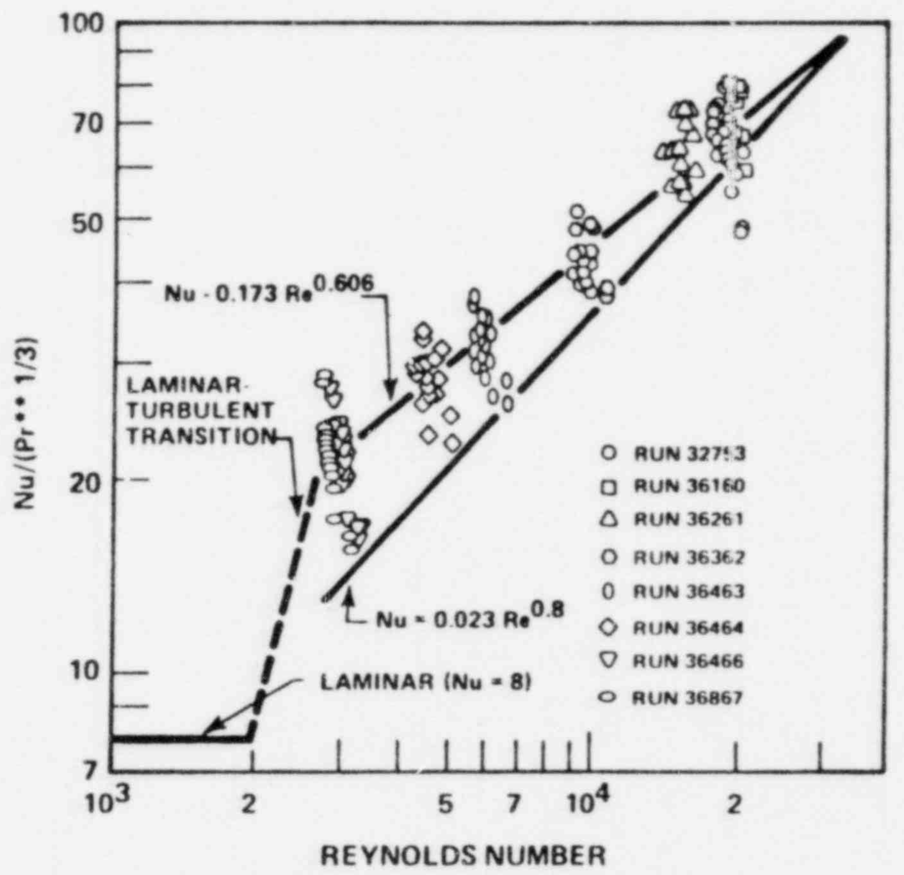


Figure 19 Data-Based Nusselt Number Versus Reynolds Number for Eight Steam Cooling Tests

21-ROD BUNDLE TASK OBJECTIVES

- TO OBTAIN, EVALUATE, AND ANALYZE THERMAL HYDRAULIC DATA USING 21-ROD BUNDLES TO DETERMINE THE EFFECTS OF FLOW BLOCKAGE GEOMETRY VARIATION ON THE REFLOOD HEAT TRANSFER

- TO GUIDE THE SELECTION OF A BLOCKAGE SHAPE FOR USE IN THE LARGE BLOCKED BUNDLE TASK

- TO DEVELOP AN ANALYTICAL OR EMPIRICAL METHOD FOR USE IN ANALYZING THE BLOCKED BUNDLE HEAT TRANSFER DATA

FIGURE 20

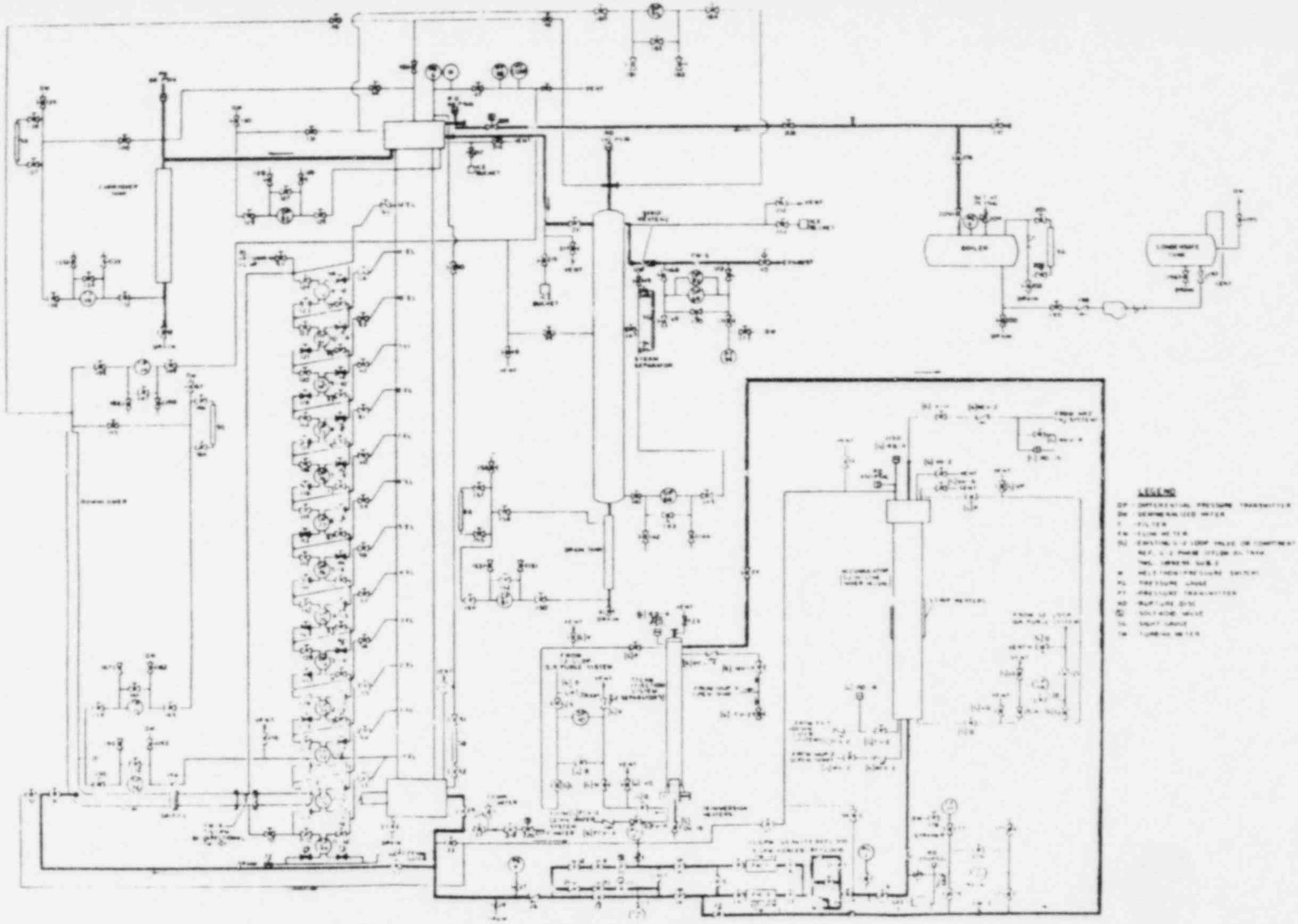


Figure 21 Schematic Diagram -
FLECHT SEASET 21-Rod Bundle
Flow Diagram
(Drawing No. 1541E68)

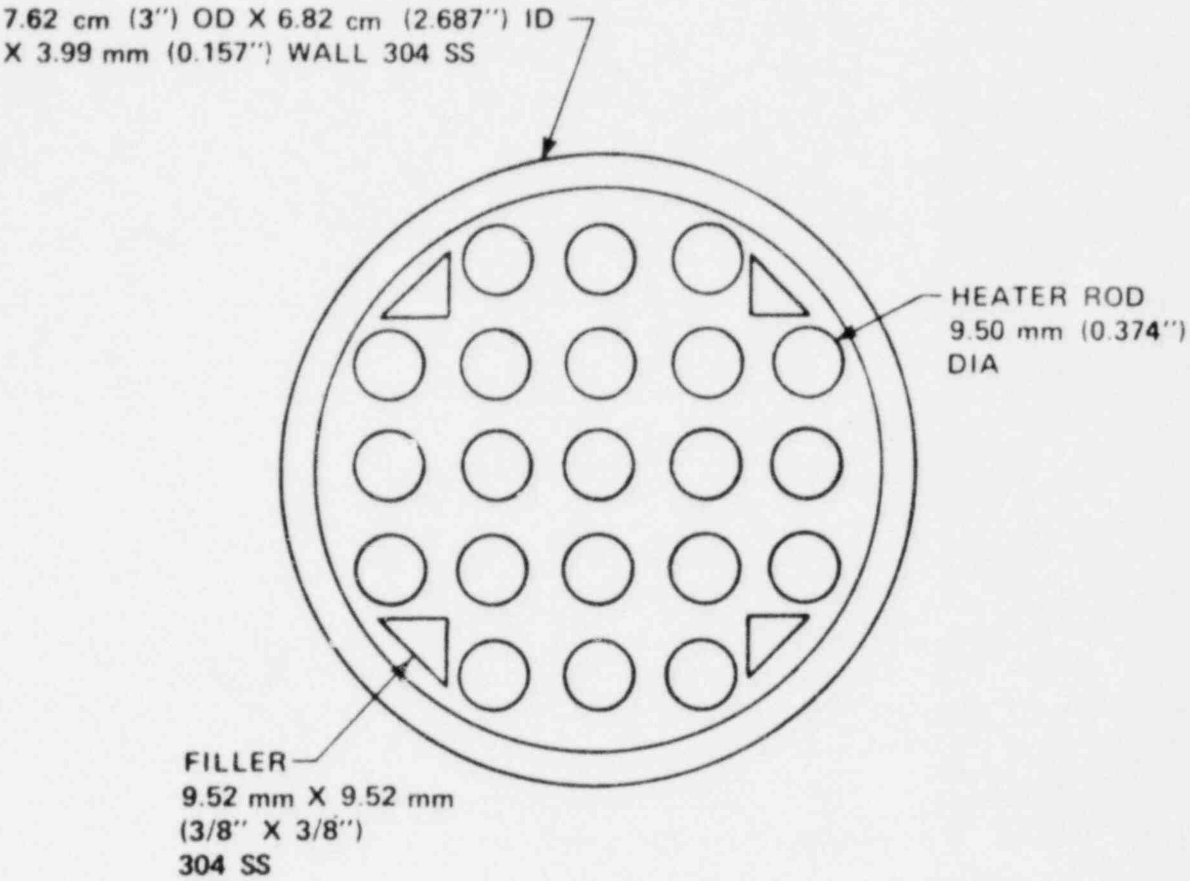


Figure 22 21-Rod Bundle Test Section Cross Section

- 21-ROD BUNDLE PROGRAM WILL TEST:

A- UNBLOCKED REFERENCE

B- 9 RODS BLOCKED COPLANAR, 62%, SHORT CONCENTRIC SLEEVE

C- 21 RODS BLOCKED COPLANAR, 62%, SHORT CONCENTRIC SLEEVE

D- 21 RODS BLOCKED NON COPLANAR, SHORT CONCENTRIC SLEEVE

E- 21 RODS BLOCKED NON-COPLANAR, LONG NON-CONCENTRIC SLEEVE

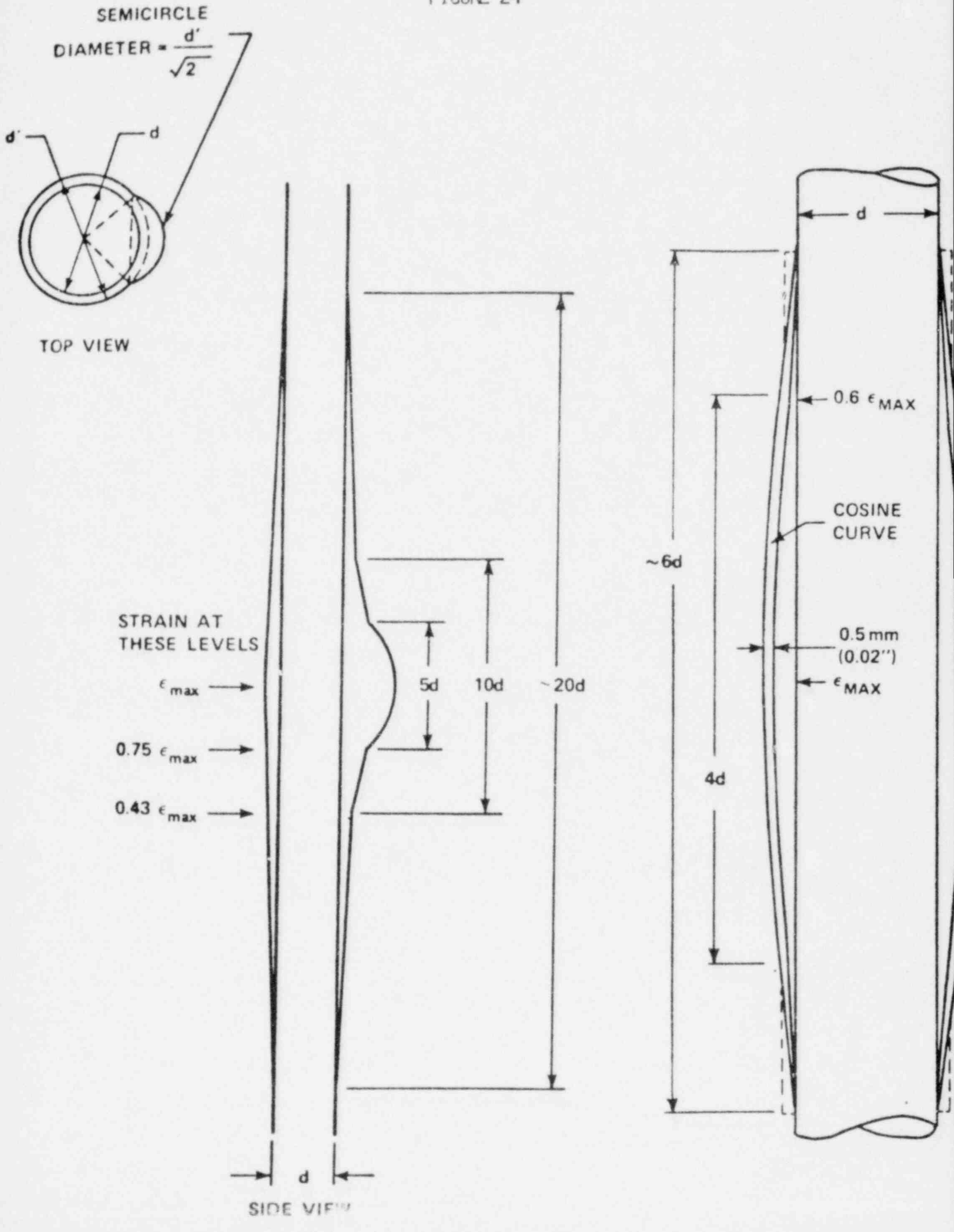
F- THE WORST SHAPE, NON-COPLANAR, MORE STRAIN

G- TO BE DETERMINED

- 161-ROD BUNDLE WILL BLOCK TWO 21 ROD BUNDLE ISLAND WITH WORST SHAPE DETERMINED FROM 21-ROD BUNDLE. TWO TEST SERIES WILL BE PERFORMED.

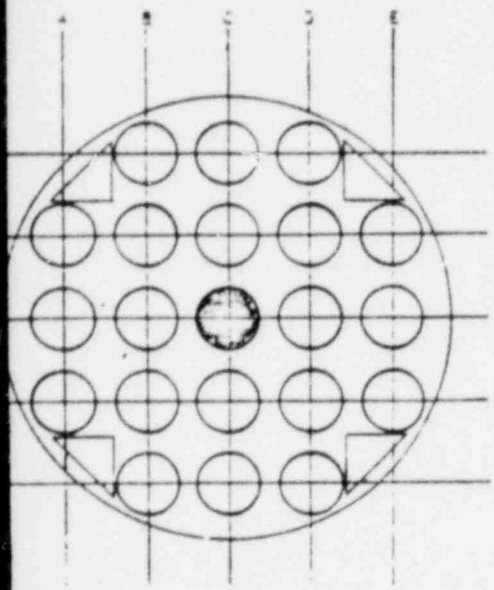
FIGURE 23

FIGURE 24

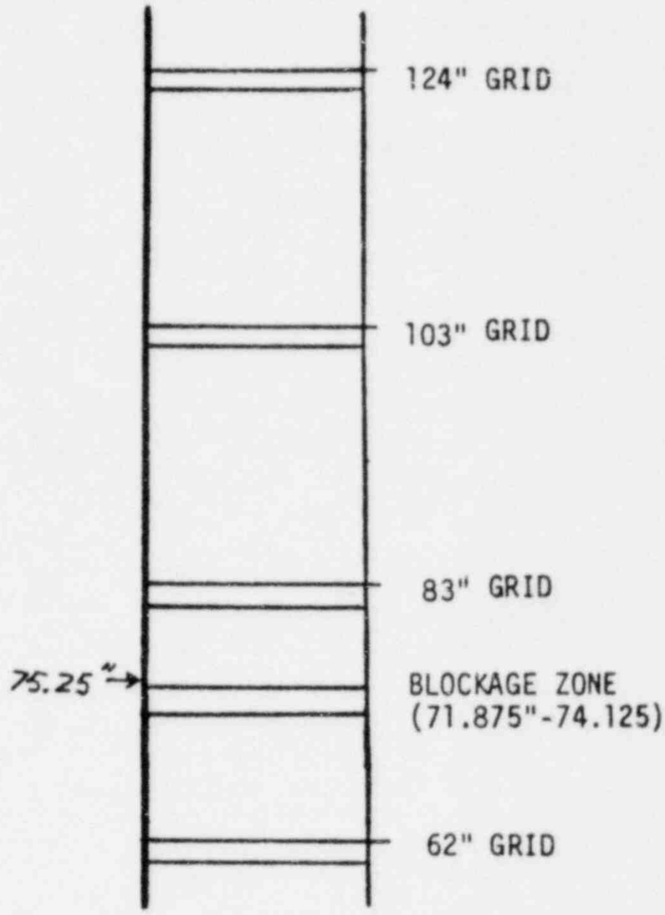


SEMICIRCLE
DIAMETER = d'/√2
d'
d
TOP VIEW
STRAIN AT
THESE LEVELS
ε max
0.75 ε max
0.43 ε max
5d
10d
~20d
d
SIDE VIEW
d
~6d
0.6 ε MAX
COSINE
CURVE
4d
0.5 mm
(0.02")
ε MAX

RADIAL LOCATION



AXIAL LOCATION



RUN NO.
42606A - UNBLOCKED
42306B - 9 RODS BLOCKED, 62%
42506C - 21 RODS BLOCKED, 62%

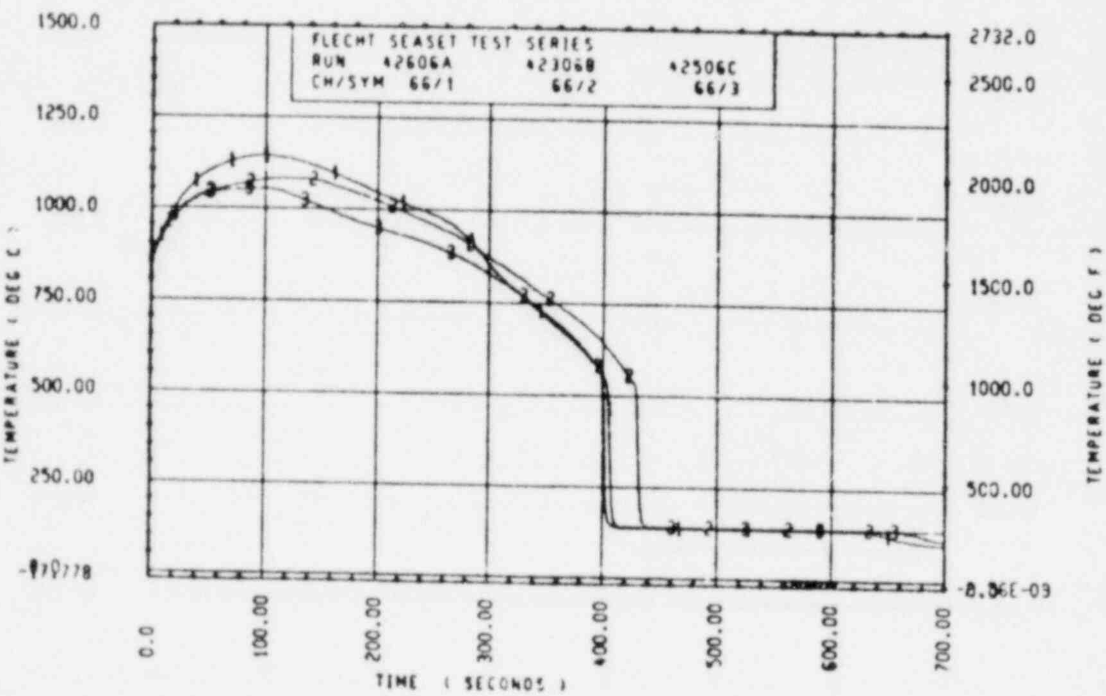
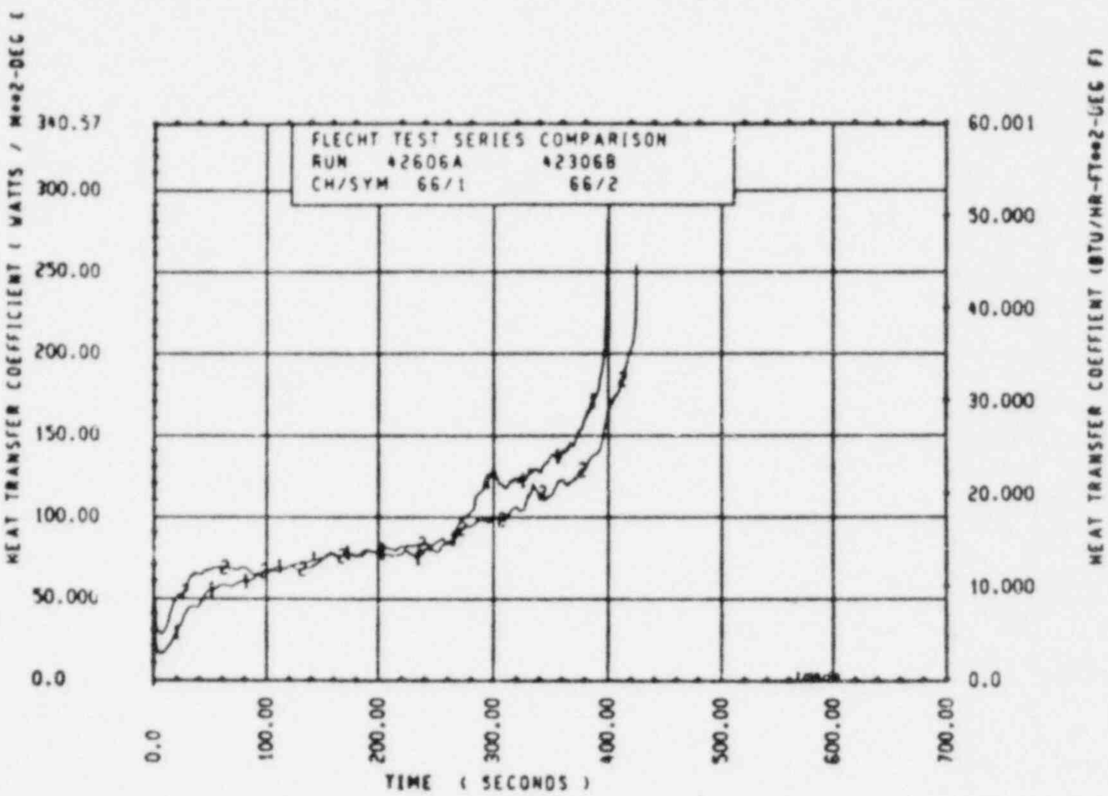


FIGURE 25A


FLECHT TEST SERIES COMPARISON
RUN #2606A #2306B
CH/SYM 66/1 66/2
HEAT TRANSFER COEFFICIENT (WATTS / M**2-DEG C
HEAT TRANSFER COEFFIENT (BTU/HR-FT**2-DEG F)
TIME (SECONDS)

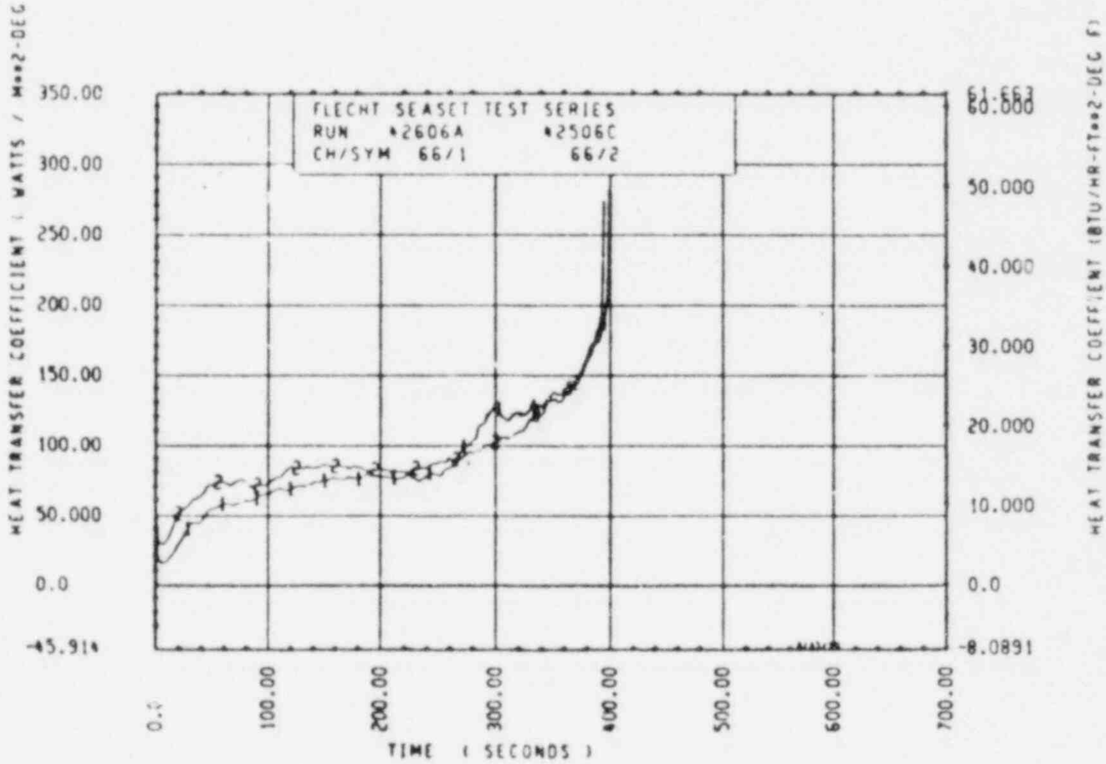

FLECHT SEASET TEST SERIES
RUN #2606A #2506C
CH/SYM 66/1 66/2
HEAT TRANSFER COEFFICIENT (WATTS / M**2-DEG C
HEAT TRANSFER COEFFIENT (BTU/HR-FT**2-DEG F)
TIME (SECONDS)

FIGURE 25B

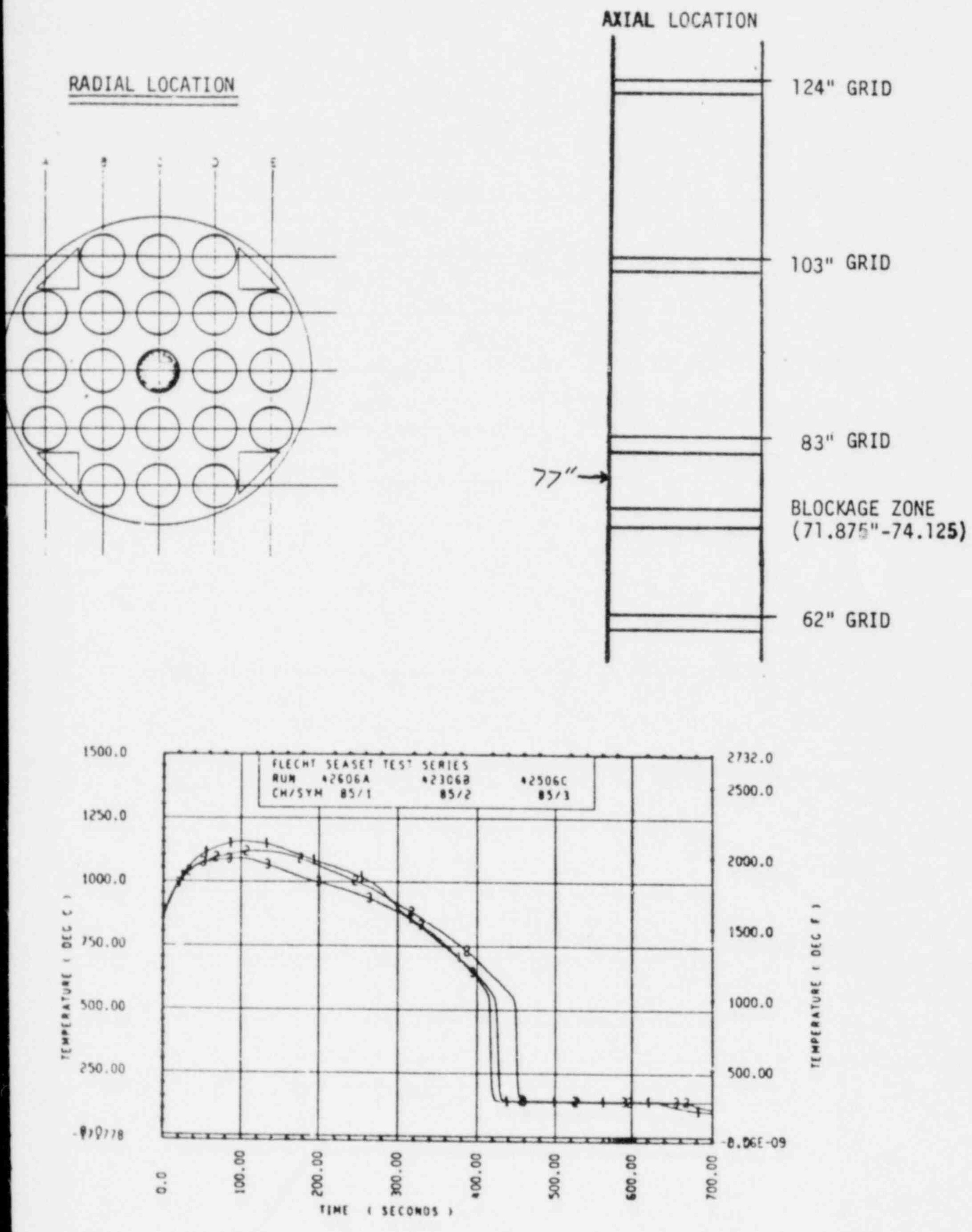
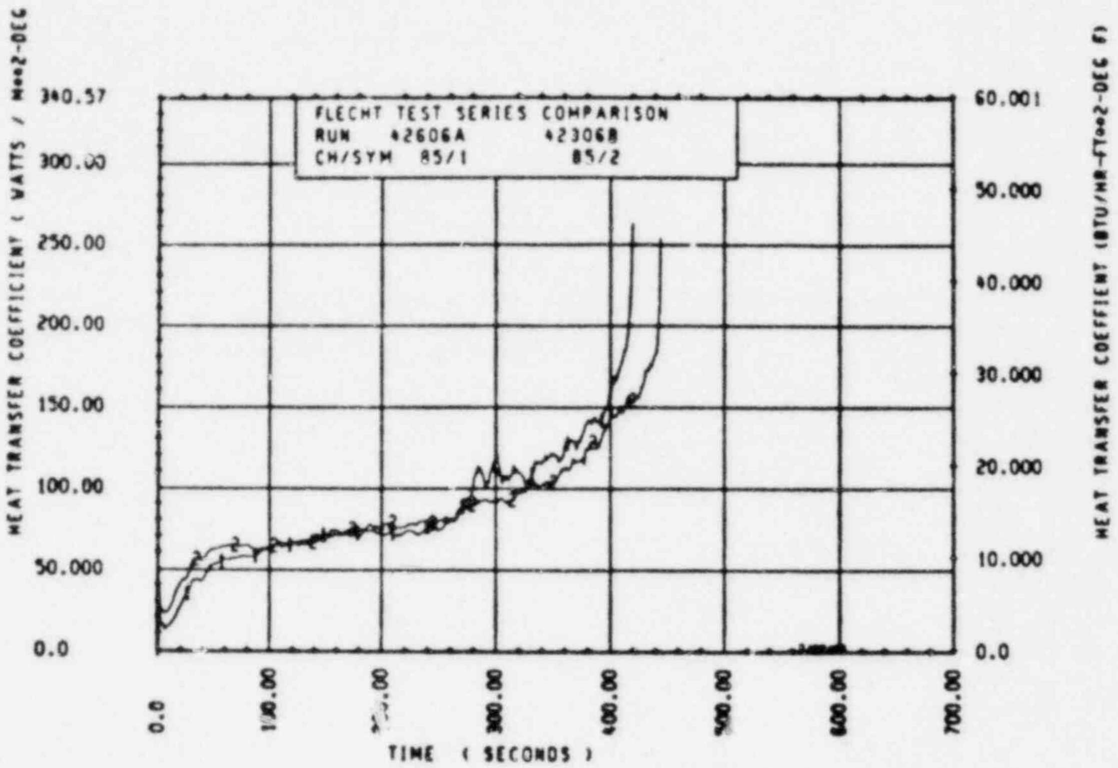

RADIAL LOCATION
A
B
C
D
E
AXIAL LOCATION
124" GRID
103" GRID
83" GRID
77"
BLOCKAGE ZONE
(71.875"-74.125)
62" GRID
FLECHT SEASET TEST SERIES
RUN #2606A #2306B #2506C
CH/SYM 85/1 85/2 85/3
1500.0
1250.0
1000.0
750.00
500.00
250.00
TEMPERATURE (DEG C)
2732.0
2500.0
2000.0
1500.0
1000.0
500.00
-0.86E-09
TEMPERATURE (DEG F)
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
TIME (SECONDS)

FIGURE 26A


FLECHT TEST SERIES COMPARISON
RUN 42606A 42306B
CH/SYM 85/1 85/2
HEAT TRANSFER COEFFICIENT (WATTS / M**2-DEG C
HEAT TRANSFER COEFFIENT (BTU/HR-FT**2-DEG F)
TIME (SECONDS)

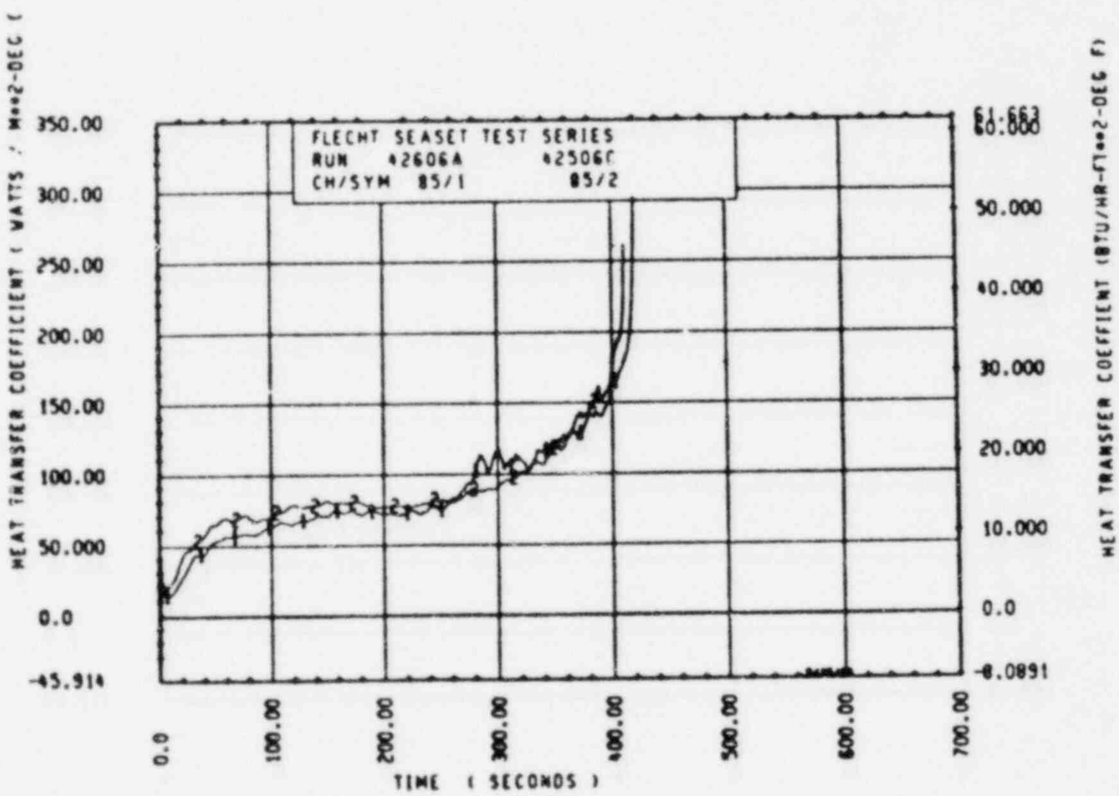

FLECHT SEASET TEST SERIES
RUN 42606A 42506C
CH/SYM 85/1 85/2
HEAT TRANSFER COEFFICIENT (WATTS / M**2-DEG C
HEAT TRANSFER COEFFIENT (BTU/HR-FT**2-DEG F)
TIME (SECONDS)

FIGURE 26B

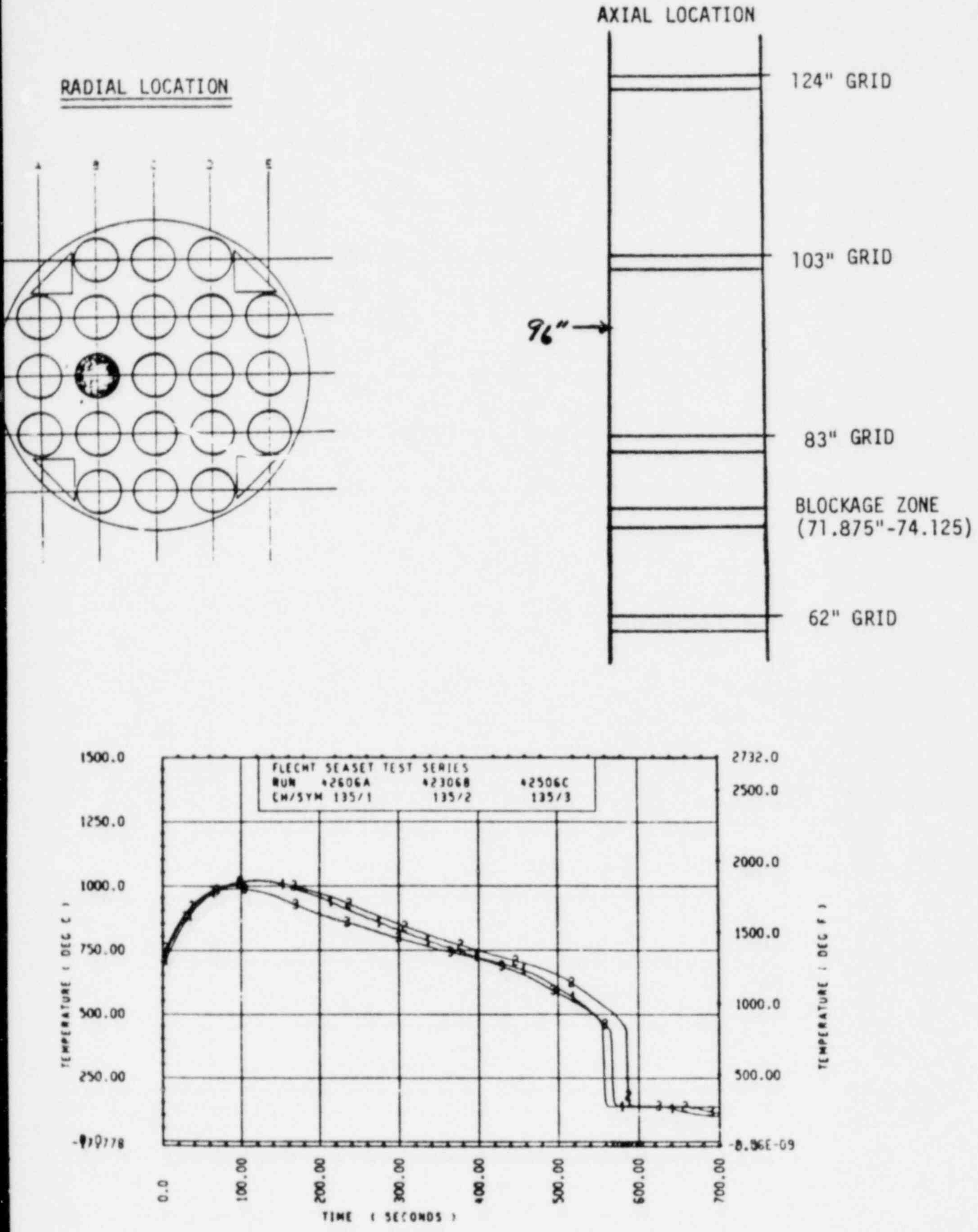
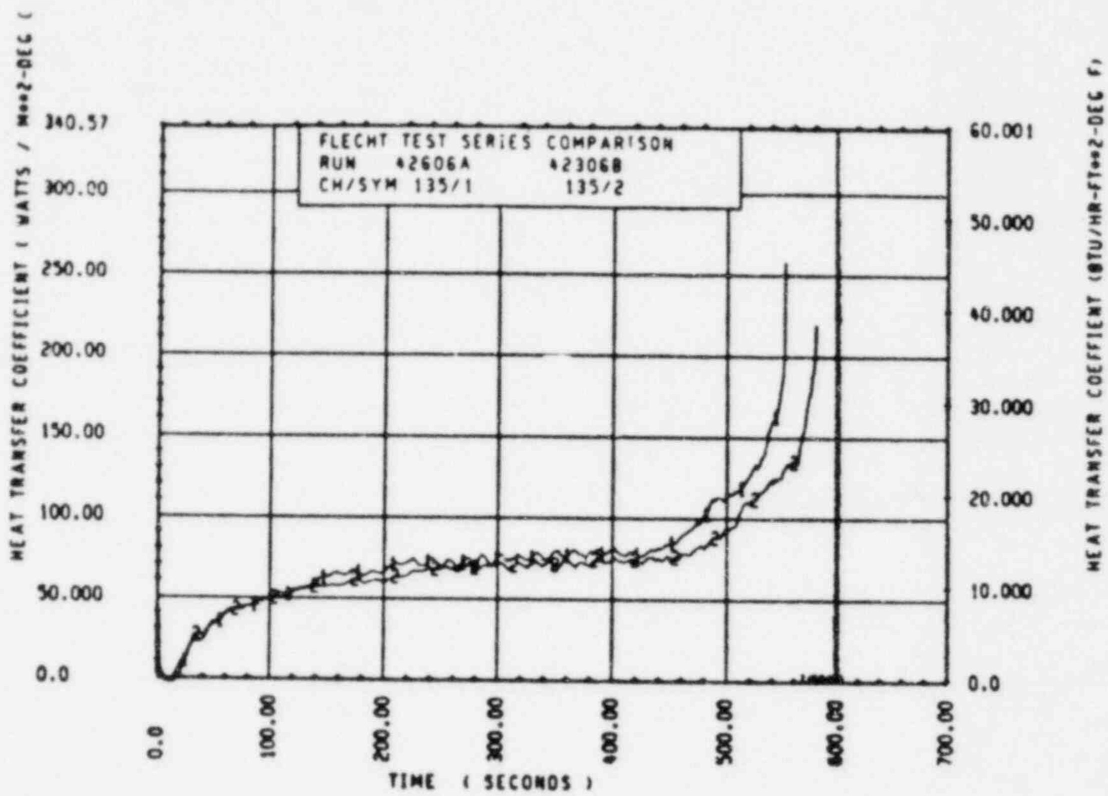

AXIAL LOCATION
RADIAL LOCATION
124" GRID
103" GRID
96"
83" GRID
BLOCKAGE ZONE
(71.875"-74.125)
62" GRID
FLECHT SEASET TEST SERIES
RUN 42606A 42306B 42506C
CH/SYM 135/1 135/2 135/3
1500.0
1250.0
1000.0
750.00
500.00
250.00
2732.0
2500.0
2000.0
1500.0
1000.0
500.00
TEMPERATURE (DEG C)
TEMPERATURE (DEG F)
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
TIME (SECONDS)

FIGURE 27 A


FLECHT TEST SERIES COMPARISON
RUN 42606A 42306B
CH/SYM 135/1 135/2
HEAT TRANSFER COEFFICIENT (WATTS / M**2-DEG C
HEAT TRANSFER COEFFIENT (BTU/HR-FT**2-DEG F)
TIME (SECONDS)

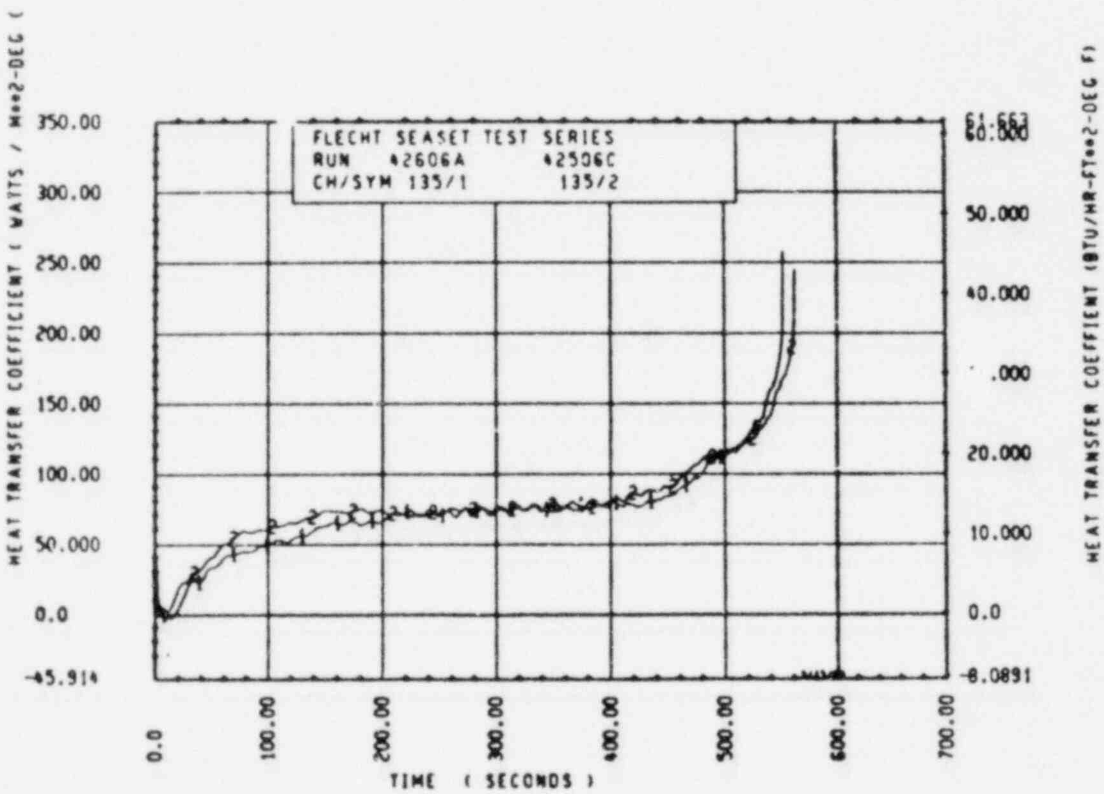
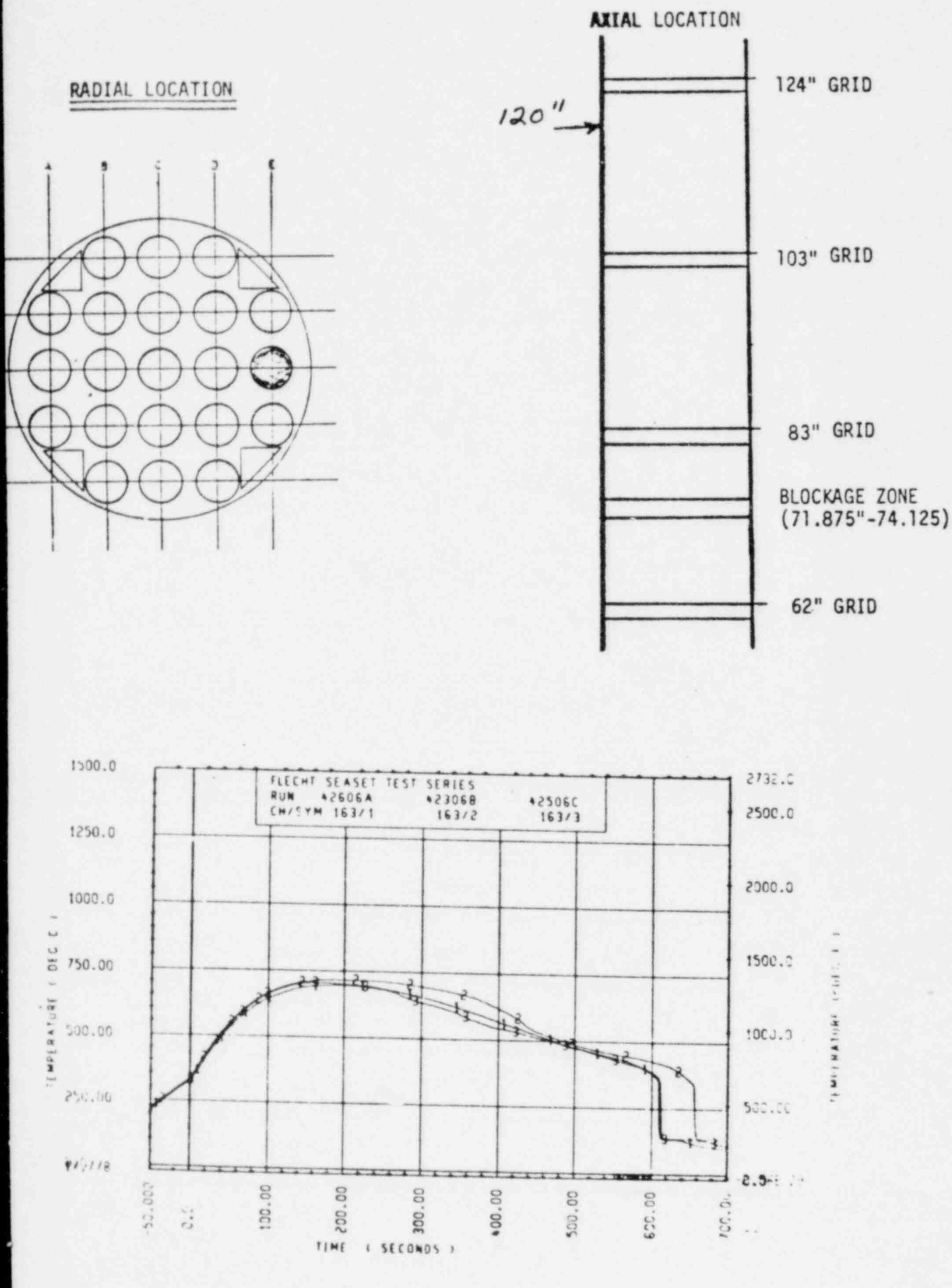

FLECHT SEASET TEST SERIES
RUN 42606A 42506C
CH/SYM 135/1 135/2
HEAT TRANSFER COEFFICIENT (WATTS / M**2-DEG C
HEAT TRANSFER COEFFIENT (BTU/HR-FT**2-DEG F)
TIME (SECONDS)

FIGURE 27B



RADIAL LOCATION
AXIAL LOCATION
124" GRID
120"
103" GRID
83" GRID
BLOCKAGE ZONE
(71.875"-74.125)
62" GRID
FLECHT SEASET TEST SERIES
RUN 42606A 42306B 42506C
CH/SYM 163/1 163/2 163/3
1500.0
1250.0
1000.0
750.00
500.00
250.00
2500.0
2000.0
1500.0
1000.0
100.00
200.00
300.00
400.00
500.00
600.00
TIME (SECONDS)

FIGURE 28

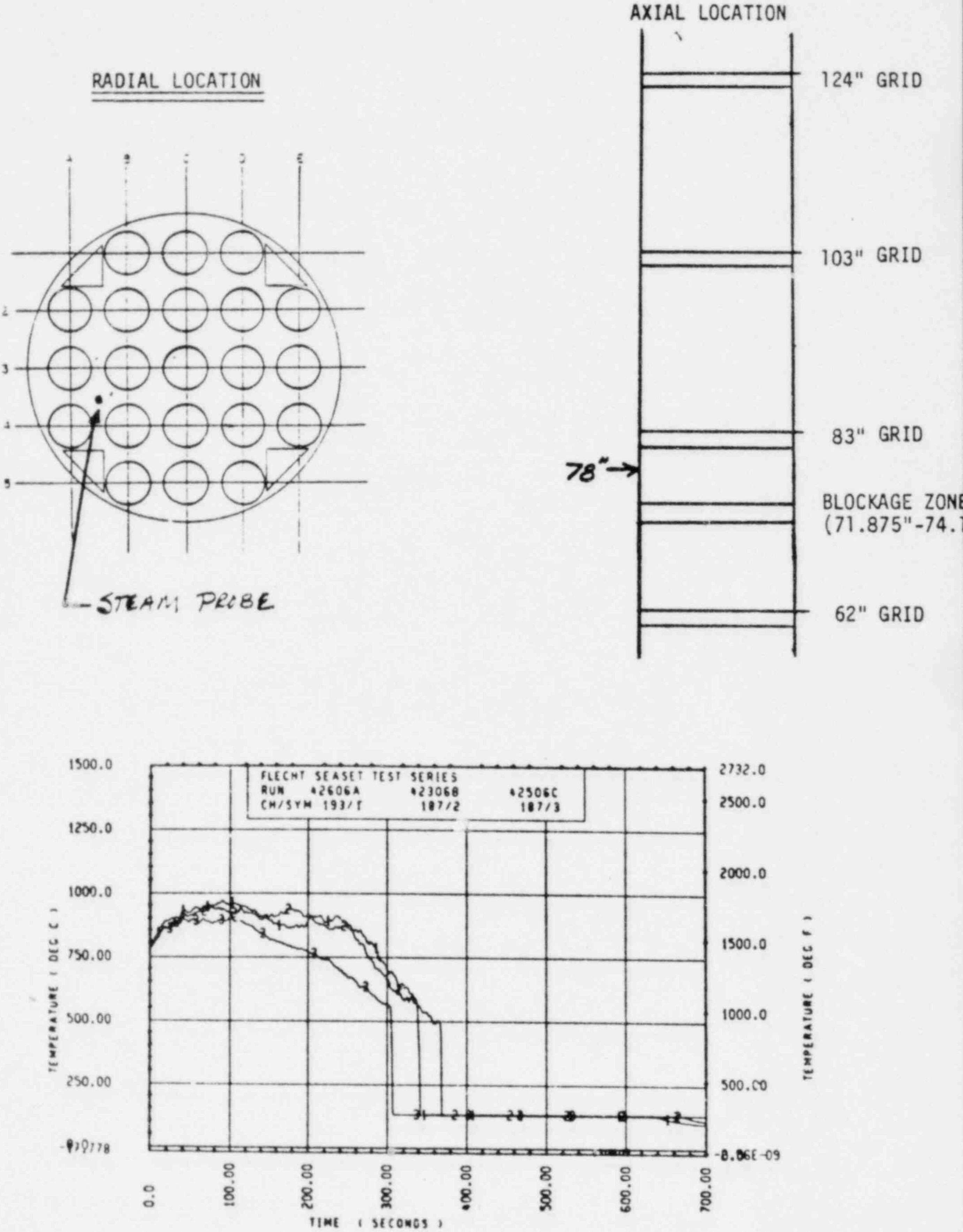

RADIAL LOCATION
STEAM PROBE
AXIAL LOCATION
124" GRID
103" GRID
83" GRID
78"
BLOCKAGE ZONE
(71.875"-74.
62" GRID
FLECHT SEASET TEST SERIES
RUN 42606A 42306B 42506C
CH/SYM 193/1 187/2 187/3
TEMPERATURE (DEG C)
TEMPERATURE (DEG F)
TIME (SECONDS)
1500.0
1250.0
1000.0
750.00
500.00
250.00
2732.0
2500.0
2000.0
1500.0
1000.0
500.00
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00

FIGURE 29

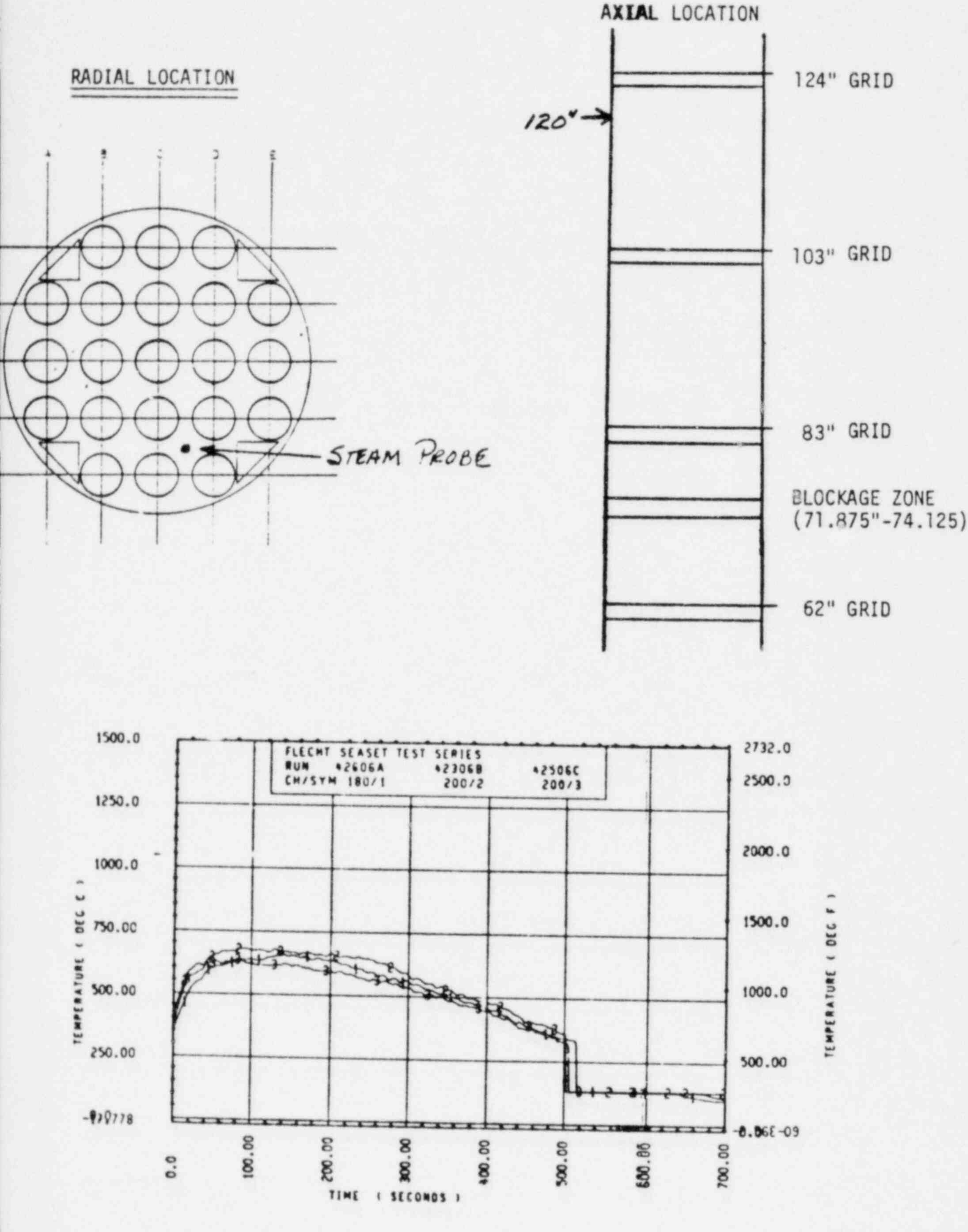

RADIAL LOCATION
STEAM PROBE
AXIAL LOCATION
120"
124" GRID
103" GRID
83" GRID
BLOCKAGE ZONE
(71.875"-74.125)
62" GRID
FLECHT SEASET TEST SERIES
RUN 42606A 42306B 42506C
CH/SYM 180/1 200/2 200/3
TEMPERATURE (DEG C)
TEMPERATURE (DEG F)
TIME (SECONDS)

FIGURE 30

Methodology Study for Qualification Testing of Wire and Cable at LOCA Condition

K.Yoshida, Y.Nakase, S.Okada, M.Ito, Y.Kusama
S.Tanaka, Y.Kasahara, S.Machi

Japan Atomic Energy Research Institute
Takasaki Radiation Chemistry Research Establishment
Takasaki, 370-12, Japan

An extensive study on a simulated testing method of a loss-of-coolant-accident(LOCA) and an accelerated aging method for electric cable materials is going forward at Takasaki Radiation Chemistry Research Establishment, JAERI. There are some problems to be solved in the qualification tests performed on the basis of suggestions by IEEE standards. A long period of the tests, which is more than several months when occasion demands, and a construction of an apparatus to simulate combined thermal and radiation environments of LOCA require agreat deal of cost. It is, therefore, an urgent issue to investigate an accelerating condition in order to shorten the test period of LOCA simulation and to study the possibility of a sequential test in which thermal and radiation environments are imposed on the cables separately and sequentially.

We built an apparatus to test the cables under a simulated LOCA condition, where environments of radiation, high temperature steam and chemical spray are combined. The simulator is designed to heat the inside of a pressure vessel up to 150°C from 20°C within 5sec and to control the temperature constant with accuracy of ±1°C in the range from 20°C to 200°C by saturated steam. The temperature is controlled to trace various sorts of LOCA profiles by using a program generator. Simultaneously, the cables in the vessel are exposed by radiation from Co-60 source(200 or 30 kci) placed at the center of the vessel. The designed dose rate is 1 or 0.25 Mrad/hr, and the uniformity ratio of the exposure dose is within 1.1, at the position of the cables winded on a mandrel set up in the vessel. The cables are connected with the electrical loading circuits through specially designed penetrations.

We studied the change of mechanical properties and insulation resistance of the sheets of insulating and jacketing materials used in the cables such as ethylene-propylene rubber and chloro-sulfonated polyethylene, at various stages LOCA conditons, for instance, after the pre-conditioning, after the first transient part with high temperature and at the end of the simulation. A typical LOCA profile for PWR and BWR, which cable makers are required to pass by electric power corporations in Japan, was studied. As the conditions of the latter cooling period, which occupies a large portion of the profile, we adopted four different temperature(85, 102, 120, 130°C), intending to determine an accelerating condition from a time- temperature relationship.

No correlation between the values of insulation resistance and the temperature was found. The correlation between the mechanical properties and the temperature was not so explicit as to determine an accelerating condition with sufficient accuracy. On the other hand, the values of absorbed radiation dose including the pre-conditioning showed clear correlation with those of elongation and toughness measured at various stages of the simulation. It suggests that radiation is more effective on the degradation than other environments such as a rapid heating at the first stage of LOCA. The result may provide a useful clue to determine the accelerating condition.

Comparison of simultaneous, sequential and reverse sequential methods in LOCA test were conducted. In the sequential method, the sheets were exposed to steam and chemical spray environments with a typical LOCA profile for PWR after 150 Mrad irradiation, while in reverse sequential method the environmental exposure was befor the irradiation. Both stages were combined in the simultaneous method. As to elongation at break, the deterioration in the three methods do not show a remarkable difference. On the contrary, a tensile strength at break was quite different. The sequential method is the most destructive and the reverse sequential method has minor effect on the strength of materials. The degree of deterioration by simultaneous method is intermediate of two method.

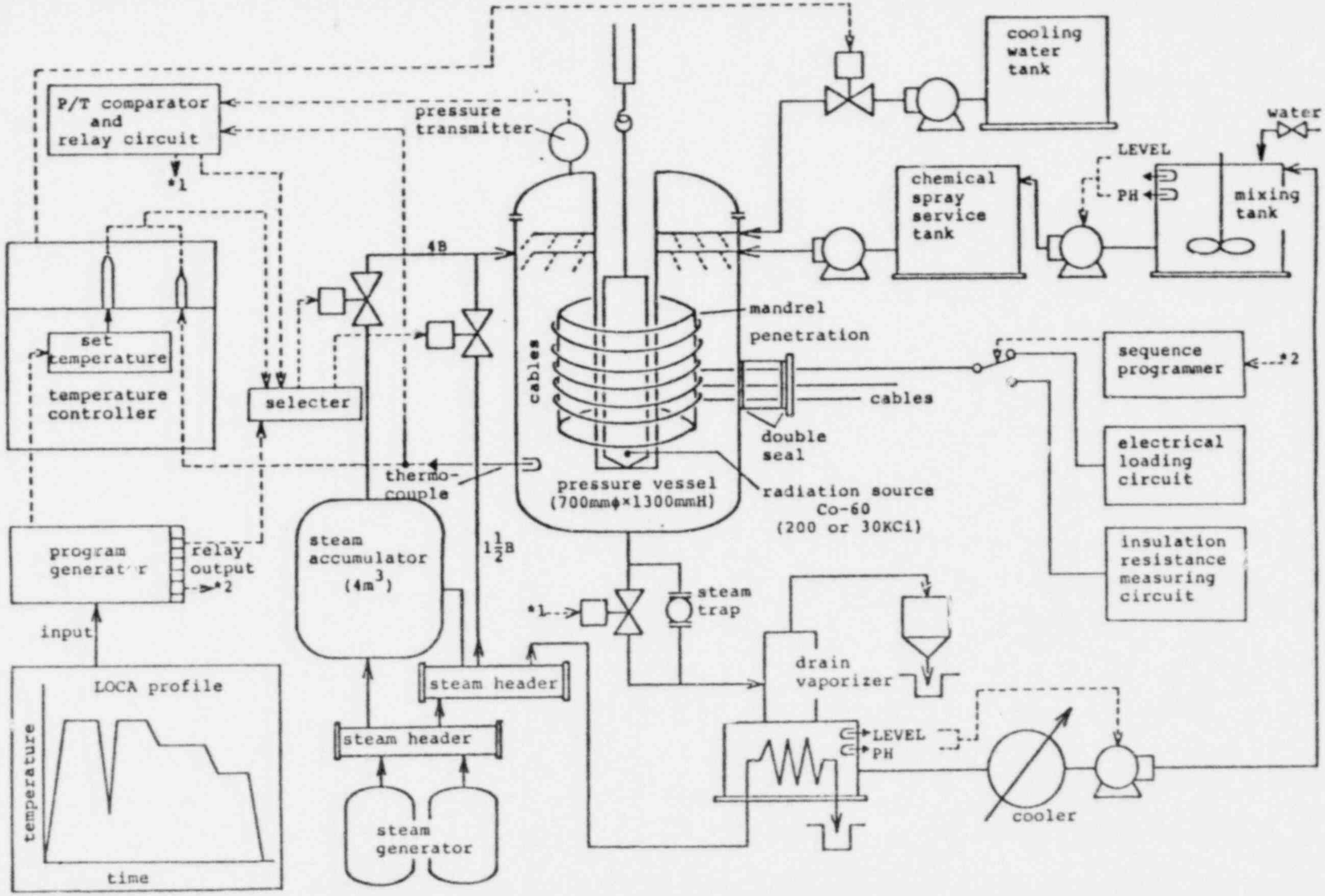


Fig.1 Flow Diagram of SEAMATE-II

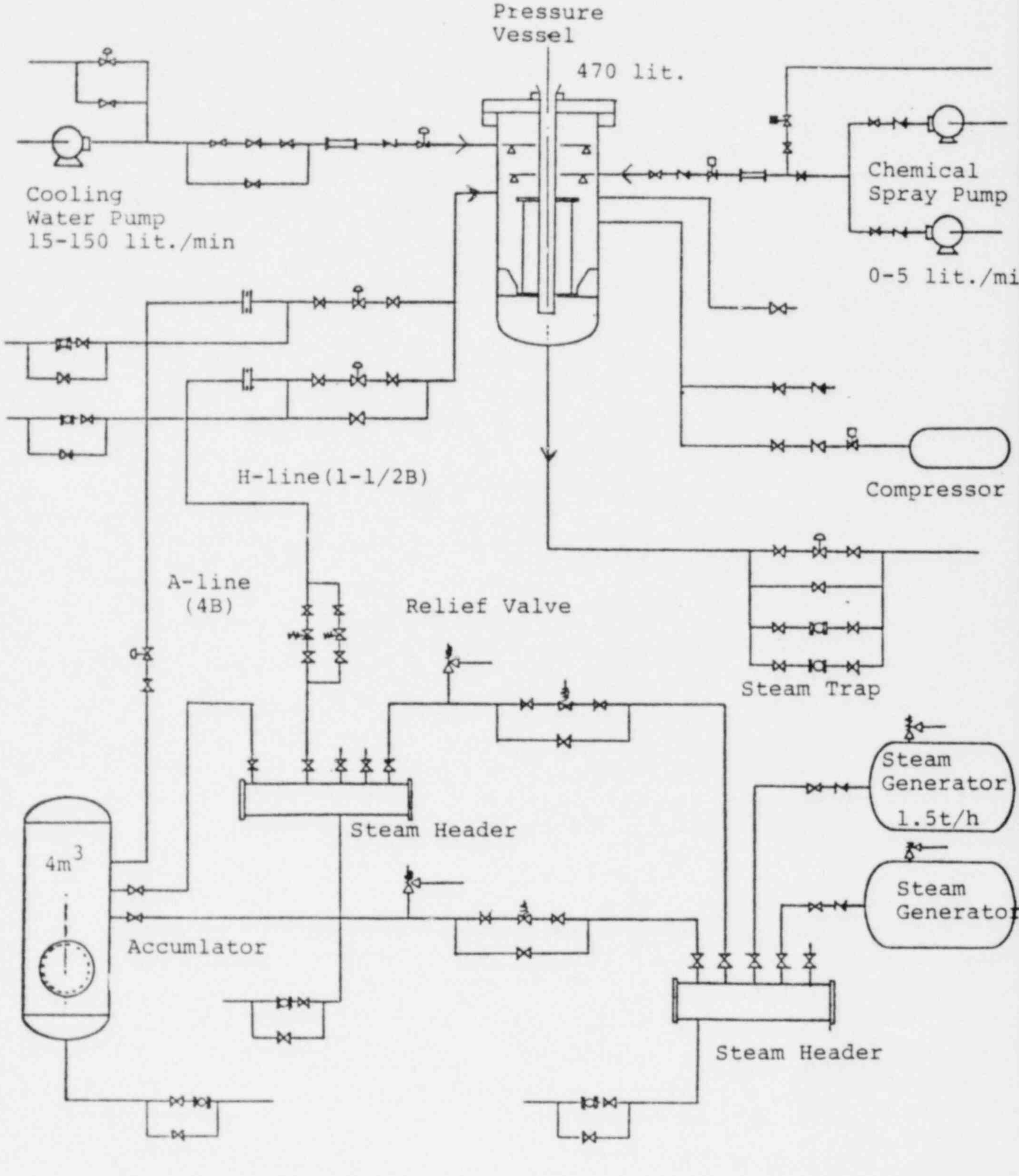


Fig. 2 Flow Sheet of Steam Supply, Water Supply and Chemical Spray Line of SEAMATE

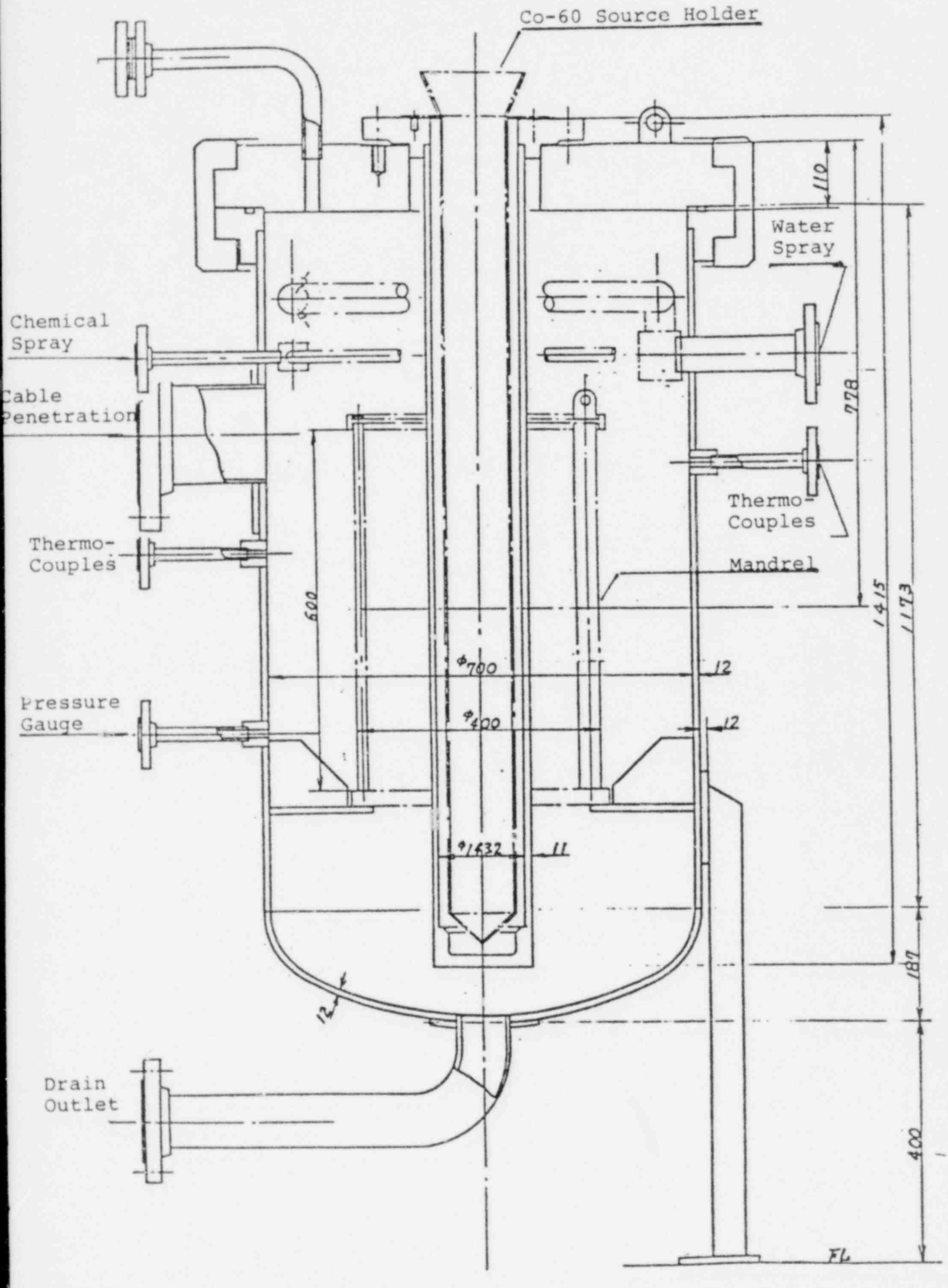


Fig.3 Cross Section of Pressure Vessel

Specification of Apparatus

Size of Pressure Vessel	700I.D x 1300 H
Maximum Pressure	25 kg/cm^2G
Maximum Temperature	250°C
Size of Mandrel	400 I.D x 600
Quick Heating Time	20 to 150 within 10sec(design.)
	" " 4 " (measured)
Maximum Steam Supply	10,000 kg/h
Rate of Chemical Spray	0-5 lit./min
Amount of Co-60	30 kci and 200 kci
Capacity of Electrical Loading	600V and 90A
Number of Cable Loaded	9 Cables(3 conductors)
Dose Rate	0.25 and 1.0 Mrad/h(designed)
Dose Rate Uniformity	1.1 (designed)

Uniformity Ratio of Dose Rate

Vertical	1.04 - 1.14	(for 200 kci)
"	1.04 - 1.15	(for 30 kci)
Radial	1.03 - 1.06	(for 200 kci)
"	1.07 - 1.15	(for 30 kci)

Air Equivalent Dose Rate at Cable Position

200 kci	1.09 Mrad/h	(without cooling water)
"	0.95 Mrad/h	(with cooling water)
30 kci	0.21 Mrad/h	(without cooling water)

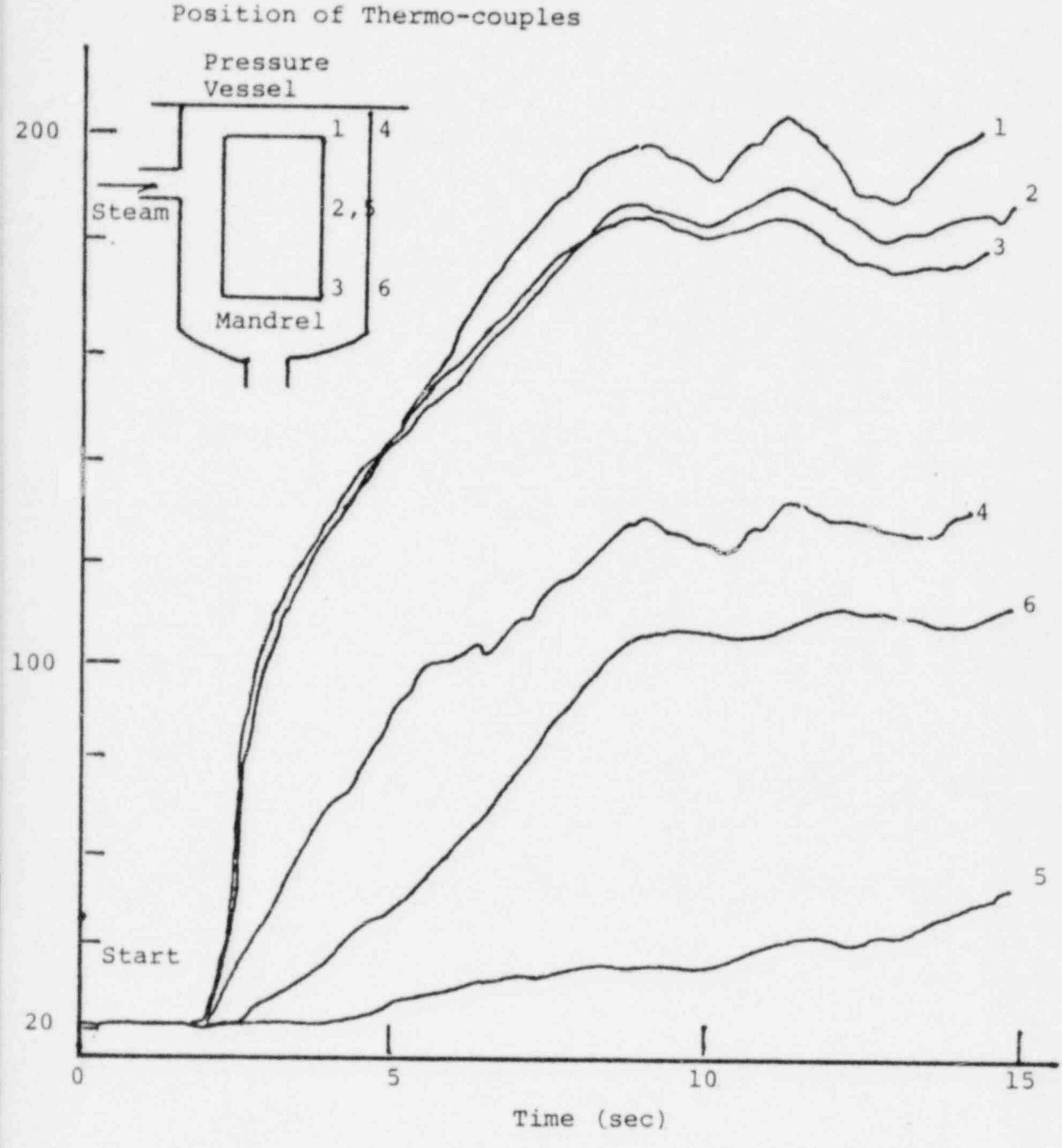


Fig. 4 Quick Heating by Steam and Temperature Distribution in Pressure Vessel (heating program: 20-200°C within 15 sec)
Position of Thermo-couples: 1,2,3 on Mandrel
4,6 on PV Wall
5 in Cable

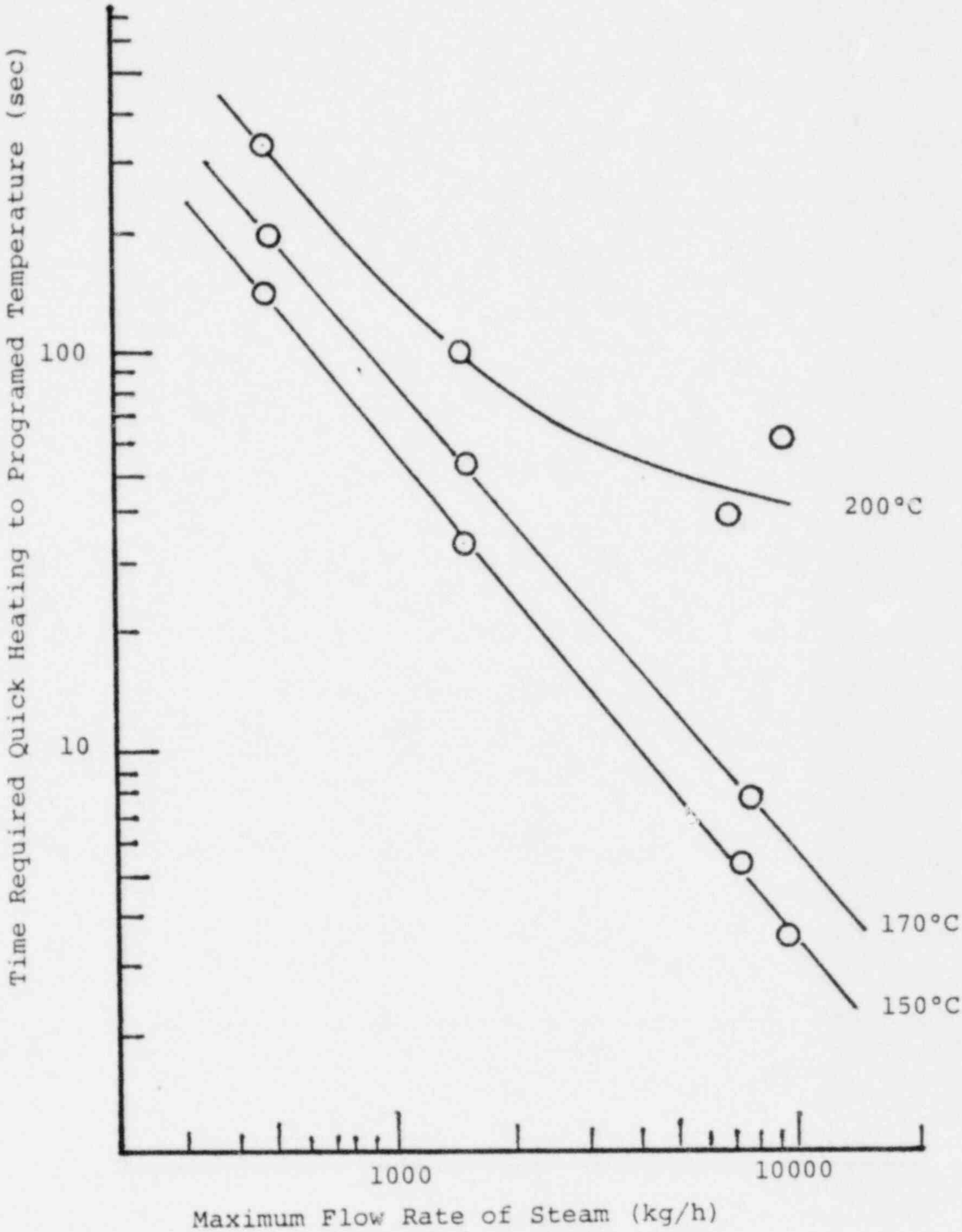


Fig. 7 Relation between Quick Heating Time and Steam Flow Rate

Methodological Study of LOCA Testing

Present Program

1 Shortening a Latter Cooling Period of LOCA for PWR and BWR by Arrhenius Approach

2 Comparison between Simultaneous and Sequential LOCA Testing

Next Program

1 Equivalence of the Shortened one-week Test to the one-month LOCA Test

2 Possibility to estimate a Deterioration of Cable Using a Degradation Data Obtained from a Sheet of Insulating and Jacketing Materials

3 Influence of Changes in LOCA Profiles on Degradation of Elastomeric Materials

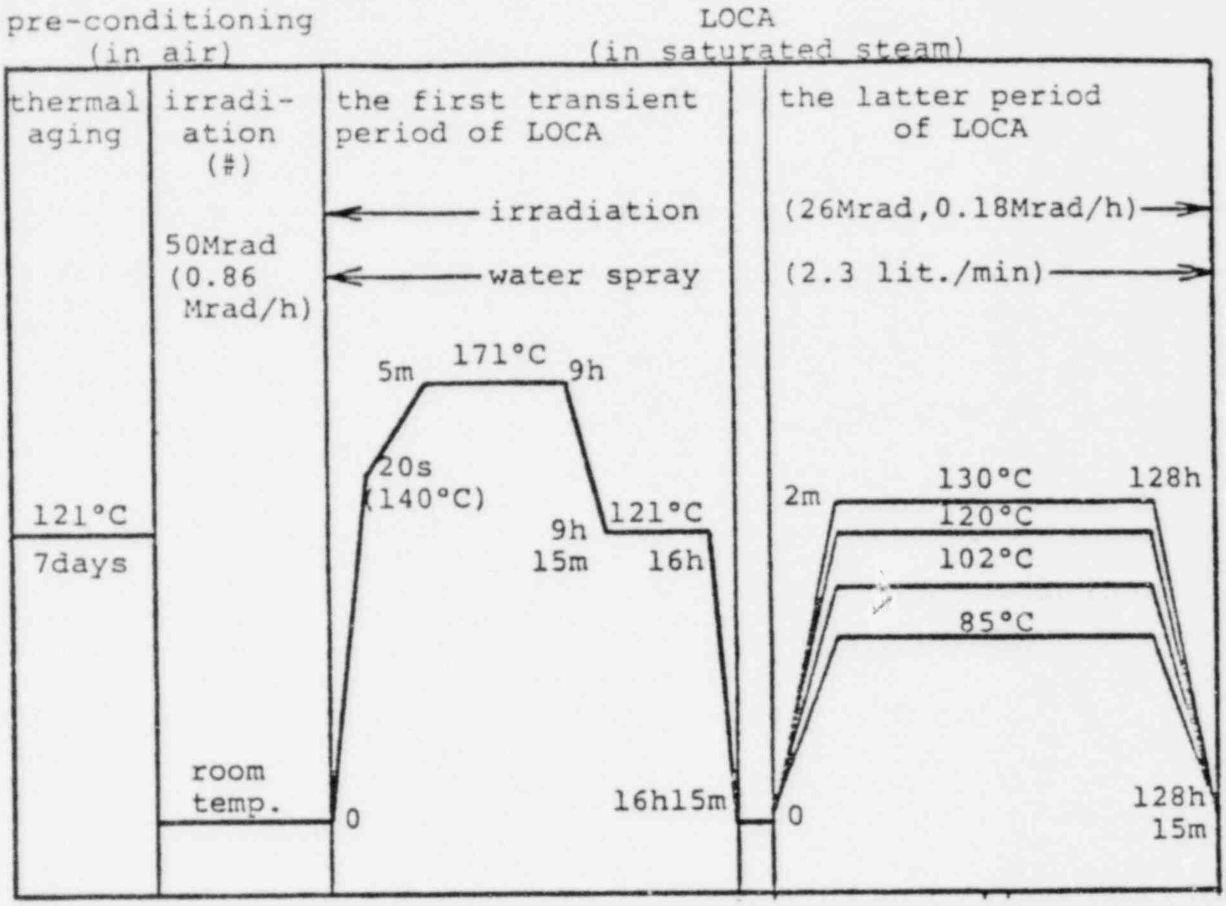


Fig. 8 Tested Profile for BWR

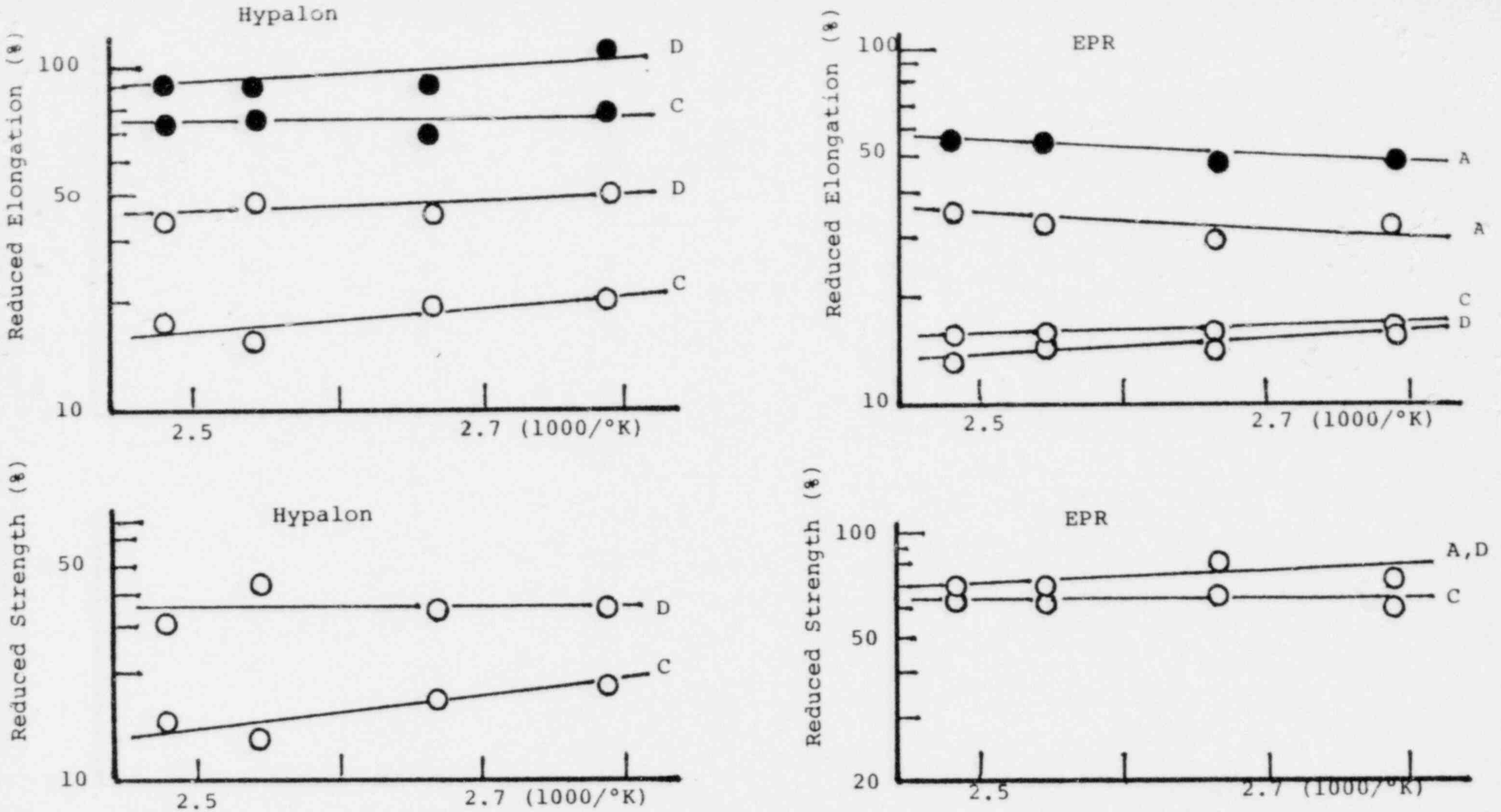


Fig 9 Effect of Temperature in Latter Part of LOCA Profile for BWR on Elongation and Strength of Hypalon and EPR

(○ with Aging , ● without Aging)

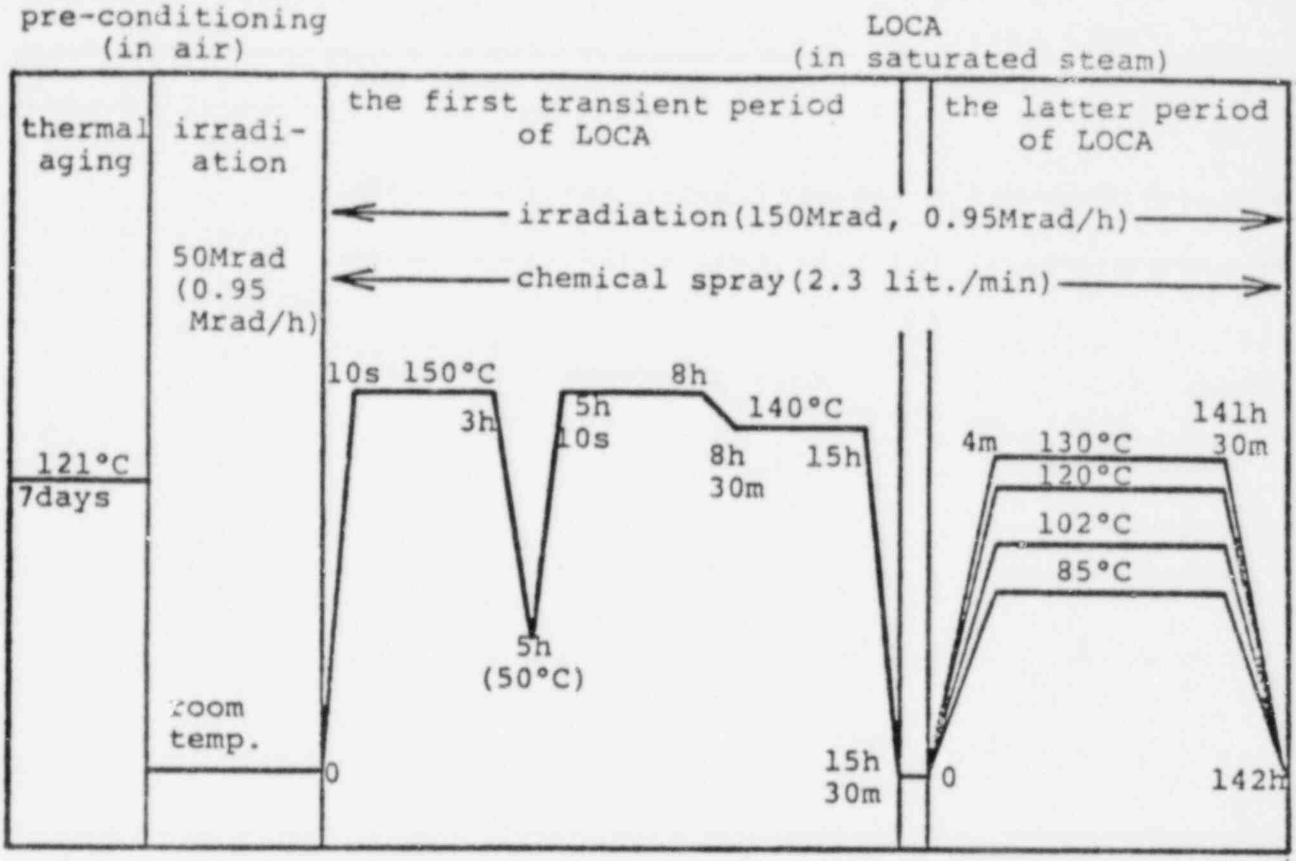


Fig.10 Tested Profile for PWR

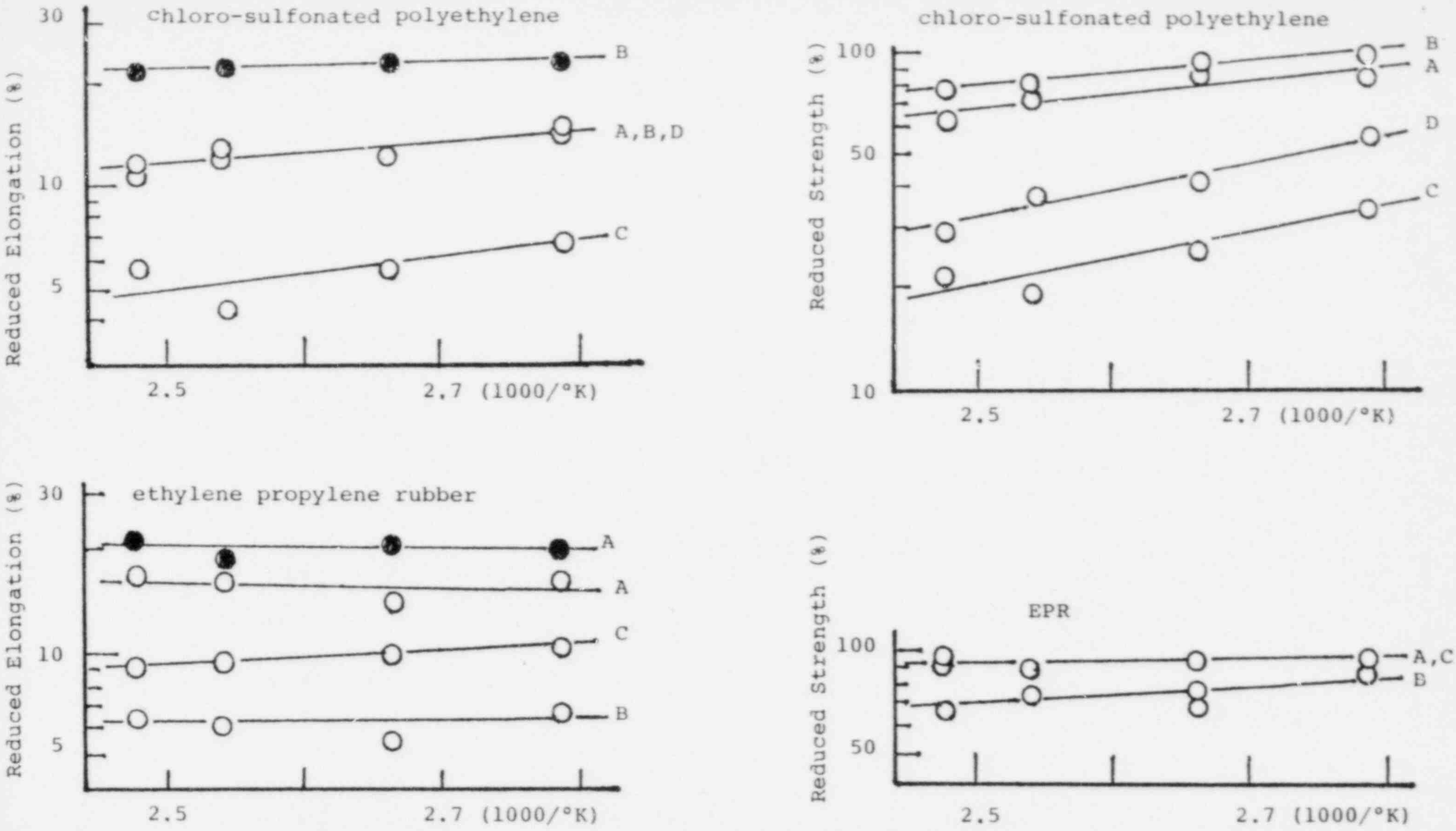


Fig. 11 Effect of Temperature in Latter Part of LOCA for PWR on Strength and Elongation of Chloro-sulfonated polyethylene and Ethylene propylene rubber (○ with Aging, ● without Aging)

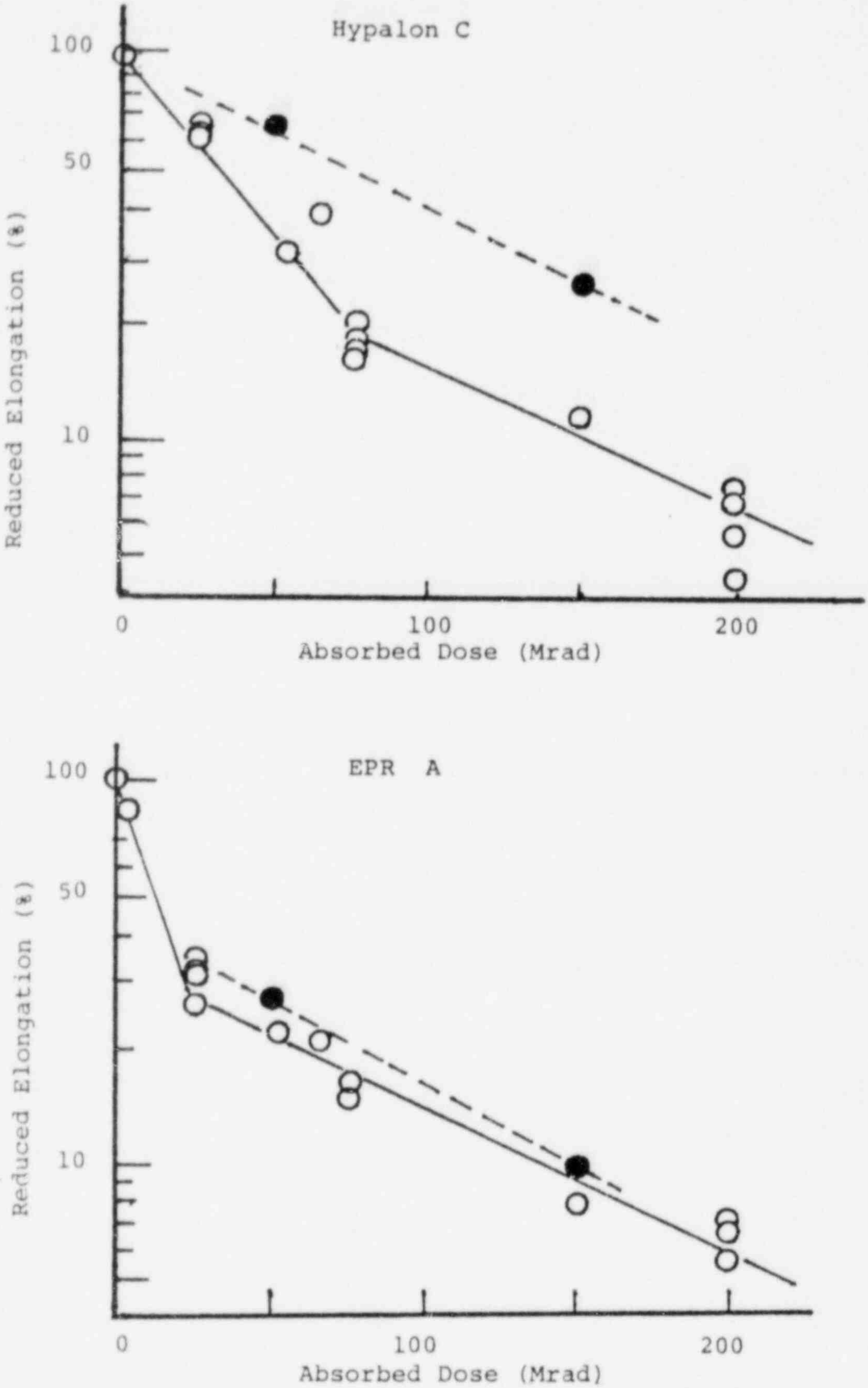


Fig. 12 Relationship between Absorbed Dose and Elongation at Various LOCA Stages
(○ with LOCA, ● Irradiation only)

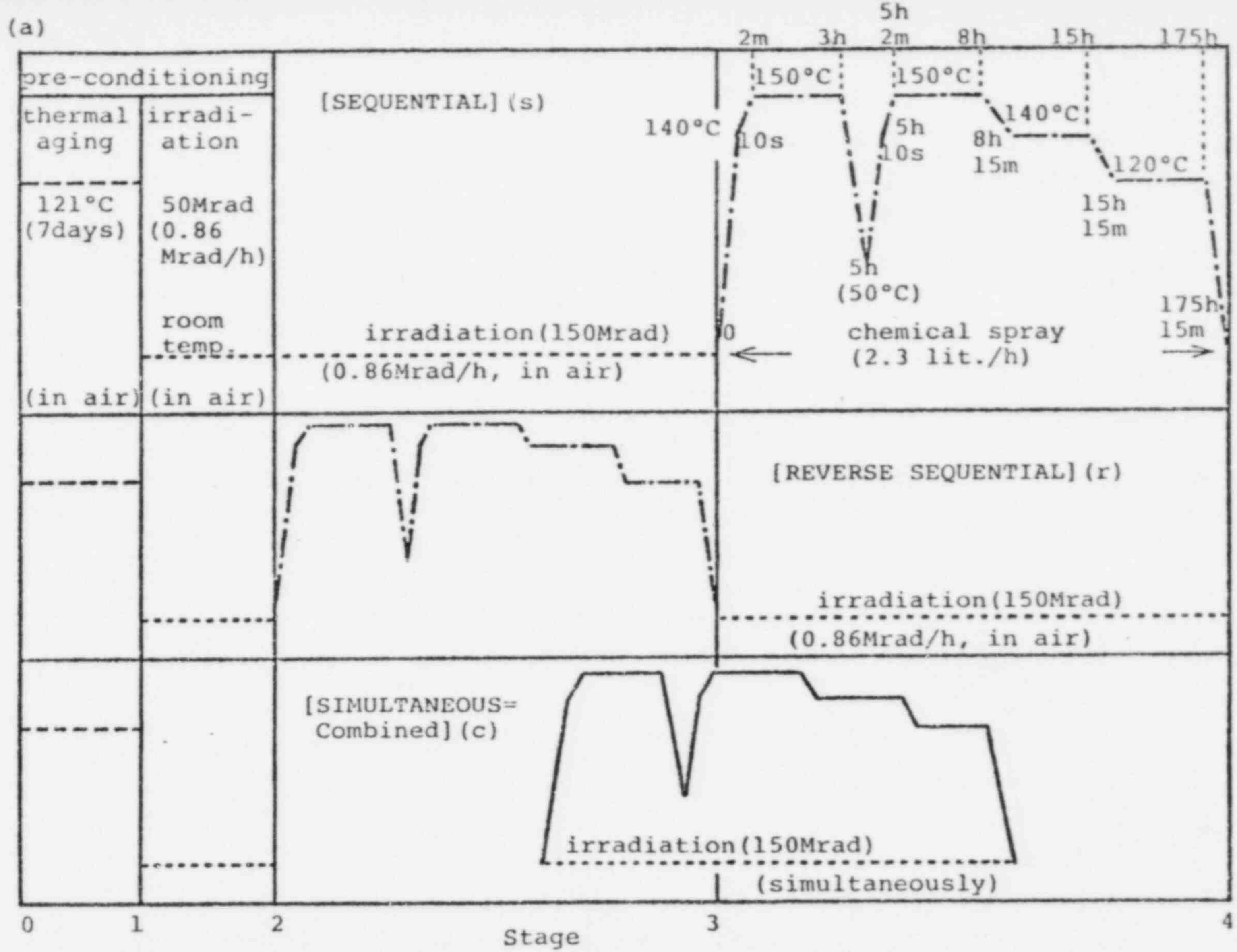
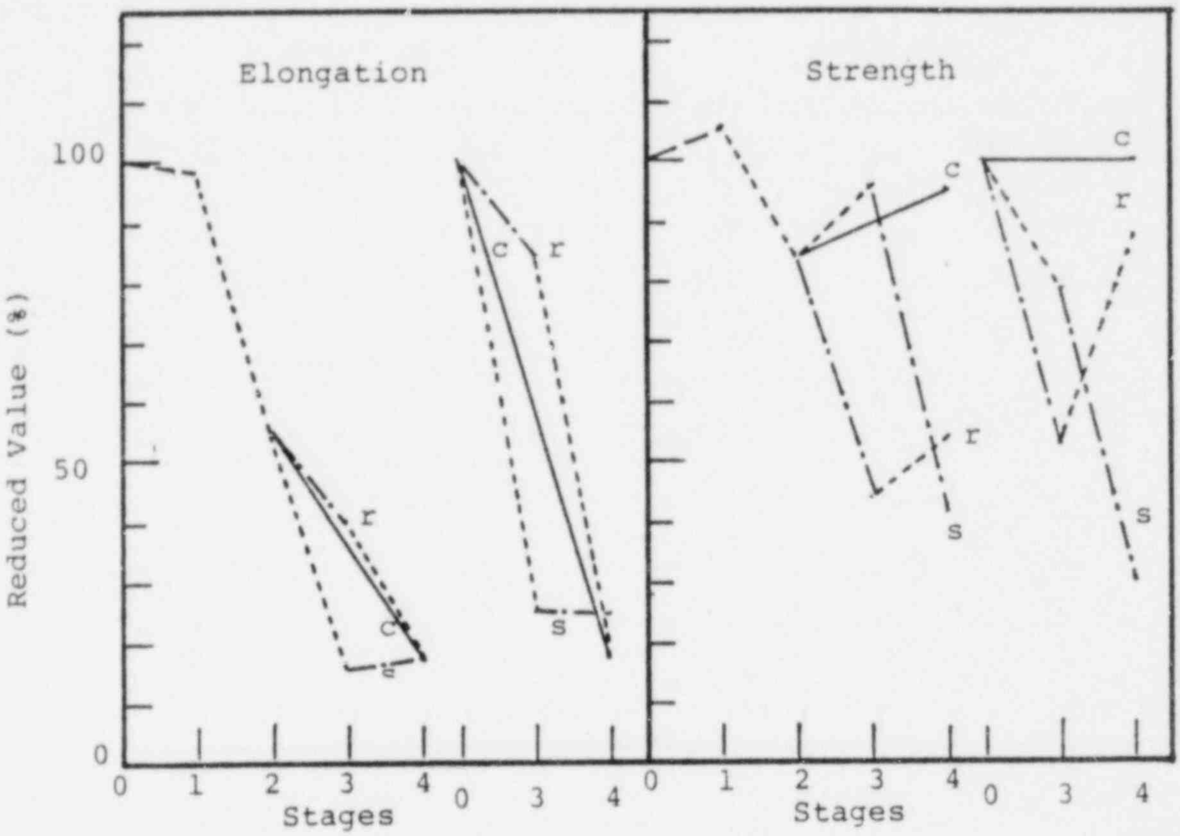


Fig.13 Tested Profiles for Simultaneous, Sequential and Reverse Sequential Method

EPR A



EPR C

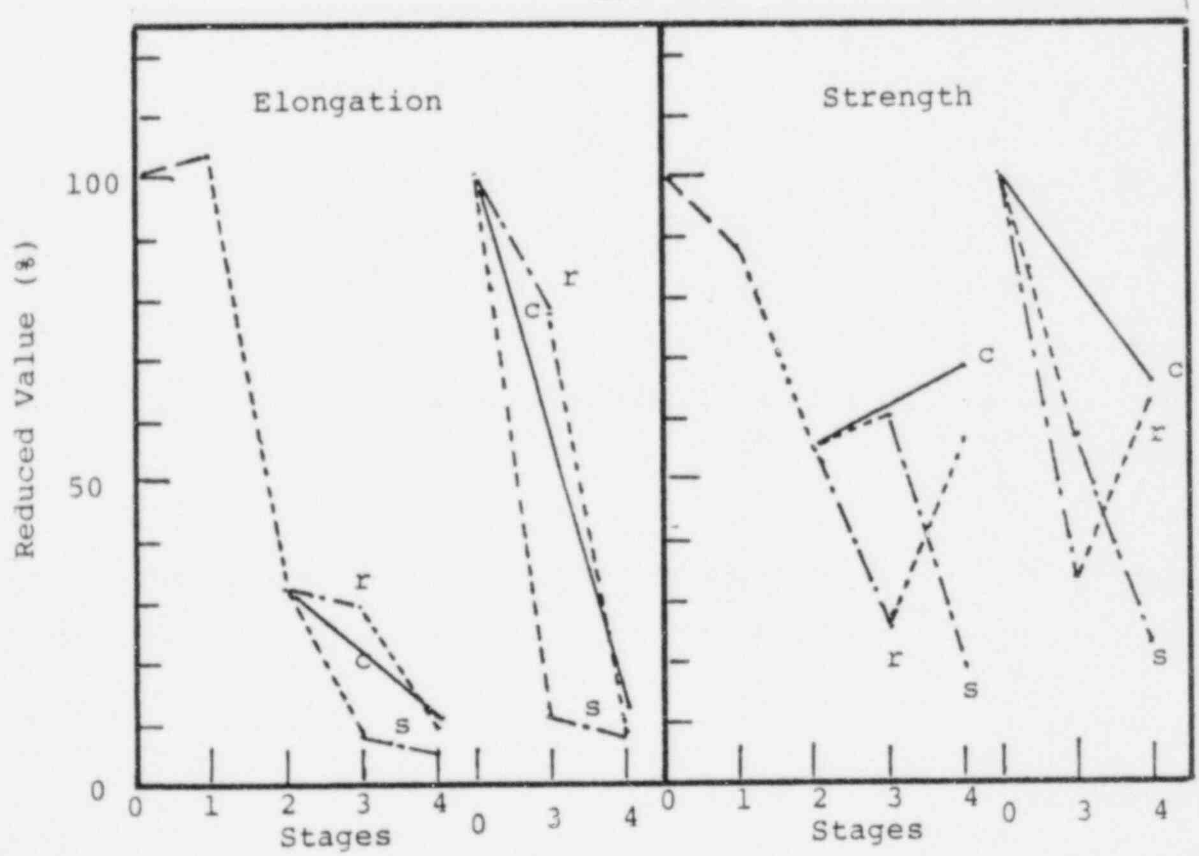


Fig.14 Comparison of Simultaneous, Sequential andReverse sequential LOCA Tests (EPR)

SAND80-2422C

USNRC Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980

Fire Protection Research Program at Sandia National Laboratories*

Leo J. Klamerus
Systems Safety Technology Division 4442
Sandia National Laboratories
Albuquerque, New Mexico 87185

ABSTRACT

Sandia National Laboratories is executing a program for the Nuclear Regulatory Commission to provide data needed for confirmation of the suitability of current design standards and regulatory guides for fire protection and control in water reactor power plants. This paper summarizes the activities of this ongoing program through October 1980. Characterization of electrically initiated fires revealed a margin of safety in the separation criteria of Regulatory Guide 1.75 for such fires in IEEE-383 qualified cable. However, tests confirmed that these guidelines and standards are not sufficient, in themselves, to protect against exposure fires. This paper describes both small and full scale tests to assess the adequacy of fire retardant coatings and full scale tests on fire shields to determine their effectiveness. It also describes full scale tests to determine the effects of walls and ceilings on fire propagation between cable trays. Some small-scale scoping tests have been conducted to investigate the effects of varying the furnace pressure on cable penetration performance in the ASTM-E-119 Fire Test. The Sandia Fire Research Facility has been completed and a series of tests have been run to assess the effectiveness of Halon-1301 as a suppression system in extinguishing deep-seated cable-tray fires. It was found that given sufficient soak times Halon systems are effective in extinguishing such fires.

INTRODUCTION

The Office of Nuclear Regulatory Research of the United States Nuclear Regulatory Commission is conducting confirmatory research in areas considered important to protecting the health and safety of the public. Fire protection, established by NUREG-0050, "Recommendations Related to Browns Ferry Fire," is one area of such research.

The objectives of the Fire Protection Research Project at Sandia National Laboratories are to:

(1) provide data either to confirm the suitability of current design standards and regulatory guides for fire protection and control in light water reactor power plants, or to indicate areas where they should be updated;

*This report documents part of the Fire Protection Research (FPR) Program being conducted by Sandia National Laboratories for the United States Nuclear Regulatory Commission under Interagency Agreement DOE 40-550-75.

(2) obtain data to facilitate either modification or generation of standards and guides (changes are to be made where appropriate to decrease the vulnerability of the plant to fire, provide for better control of fires, mitigate the effects of fires on plant safety systems, and remove unnecessary design restrictions);

(3) Obtain fire effects data and assess improved equipment, design concepts, and fire prevention methods that can be used to reduce vulnerability to fire.

PROGRAM RESULTS

Cable-Tray Separation

In support of some of the provisions of NRC Regulatory Guide 1.75 "Physical Independence of Electric Systems," tests were conducted at Sandia with varying separation distances to determine the minimum separation necessary for cables most susceptible to fire. Vertical separation distances from 152 cm (5 ft) down to 26.7 cm (10.5 in) and horizontal separation distances from 91 cm (3 ft) down to 23 cm (9 in) were tested. For electrically initiated fires in a horizontal open-space configuration, it was determined that a fire will not propagate from the ignited tray to adjacent trays. These tests were conducted with fire retardant IEEE-383 qualified cable, 12-gage single-conductor and 12-gage triplex wire, utilizing both uniform and random-pattern cable packing.

Tests were also conducted with an experimental exposure (fuel) fire. The objective was to determine whether cable-tray separation alone is sufficient to prevent fire propagation between trays and between redundant safety divisions if an exposure fire resulted in a fully developed cable-tray fire.

The type and size of the worst-case exposure fire that must be considered for licensing are based on a fire-hazard analysis and will vary from plant to plant. Accordingly, no attempt was made to define a design-basis fire for the exposure-fire tests. Single-tray tests were conducted to find a reasonable set of conditions that would result in a fully developed cable-tray fire. The experimental exposure fire was then used in full-scale cable-tray exposure-fire tests. Propane burners were used to start an exposure fire in one tray, with a barrier placed between it and the tray above. When a fully developed fire was obtained in the first tray, the burners were turned off and the barrier was removed. This method allows experimental study of fire propagation from tray to tray under specific conditions and without the exposure fire affecting the other cable trays.

As noted above, a series of tests were conducted on arrays of cable trays, with both electrical and exposure-fire initiation. An array of 14 closely spaced cable trays was used to simulate a single safety division. Simulated redundant safety divisions were separated by the required 152 cm (5 ft) vertical and 91 cm (3 ft) horizontal distance. The principal conclusion was that a fully developed fire in the bottom cable tray of a stacked array may propagate to a redundant safety division without fire suppression systems (as expected). On the other hand, electrically initiated fires (IEEE-383 qualified cable) do not propagate because they do not result in a fully developed cable tray fire.

In order to determine the characteristics of a cable-tray fire in cable tunnels or in areas where structural walls are close enough to the tray to influence the fire, some of the tests were repeated to include the effect of walls and ceilings. The preliminary indication is that there is a greater chance of fire propagation under these conditions than with a similar configuration in an open area. It was shown that both the weight loss and heat flux at the top tray follow an inverse square law relationship with the distance to the corner.

In typical plant installations, cable trays are oriented vertically at some locations and in others are oriented both vertically and horizontally. Vertical cable trays have been and will be tested in both the open-space configuration and with walls and ceilings close enough to affect the fire.

The first full-scale vertical fire test was to demonstrate the effectiveness of a ceramic fiber blanket and automated fire suppression system to protect cables in a vertical cable tray configuration that is currently permitted by separation criteria guidelines. An open pool fire fueled by liquid hydrocarbon [$7.6L^3$ (2 gal) of heptane] was used.

Three open-head sprinklers were located above the trays and connected to a separate manually operated water supply. Three dummy sprinkler heads without connection to the water supply were suspended near each open head. During the test the three dummy heads were monitored electrically to determine the times at which the fusible links were activated. In order to bracket the allowable response times, it was intended that only after activation of all three dummy heads in one location would the water system be manually operated. Two smoke detectors were also located in the test area.

The fire burned for about 40 minutes with the ionization detector activating at 11 s and the photoelectric detector activating at 14 s. Two of the fusible links at the closest sprinkler location activated (one at 52 s, the other at 54 s) but the third did not activate at all, consequently, no water was supplied. At 3 min 13 s, a short circuit between conductors was indicated. At 3 min 55 s, erratic measurements were recorded for the conductors in another tray indicating the existence of intermittent short circuits. In all cable trays except one, thermal damage of cables was observed near the base [8 to 15 cm (3 to 6 in) above the fire pan].

<u>Effectiveness of Fire Shields</u>

Sandia National Laboratories has completed a series of tests using different fire shields:

- ceramic wool blanket over ladder tray

- solid bottom tray with no cover
- solid cover on ladder tray with no vents
- vented cover on solid bottom tray
- 2.54 cm (1 in) fire barrier (thermal board) between trays.

The results of the tests showed that all fire shield designs offered some protection. None of the cable which passed the flame retardency test in IEEE Std 383-1974 ignited. It is possible to ignite the cable which did not pass this flame retardancy test; however, no propagation was observed past the fire shields.

Experiments are planned to study the methodology for testing the fire retardancy of seals and penetrations. Some small-scale scoping tests have been conducted to investigate the effects of varying the furnace pressure on cable penetration performance in the ASTM-E-119 Fire Test.

Effectiveness of Fire-Retardant Coating Materials

The objective of this portion of the program is to provide information on the effectiveness of fire-retardant coating materials when used in typical cable-tray installations.

A survey of coating materials available for use in cable trays was initiated in August 1976. Generic types were chosen for testing and evaluation in small- and large-scale cable systems tests. Small-scale tests on basic coating properties have been conducted by using six coatings and two cable types. Full-scale tests were conducted using both single and double trays.

While the results showed that all coatings offer a measure of additional protection, there was a wide range in the relative effectiveness of the different coatings tested. No propagation to the second tray was observed in any of two-tray tests in which cable that passed the IEEE Std 383-1974 test was used. (Propagation was observed in three tests involving cable which did not pass the IEEE Std 383-1974 test.) Overall, a good correlation was obtained between small-scale and large-scale tests.

Halon-1301 Suppression Tests

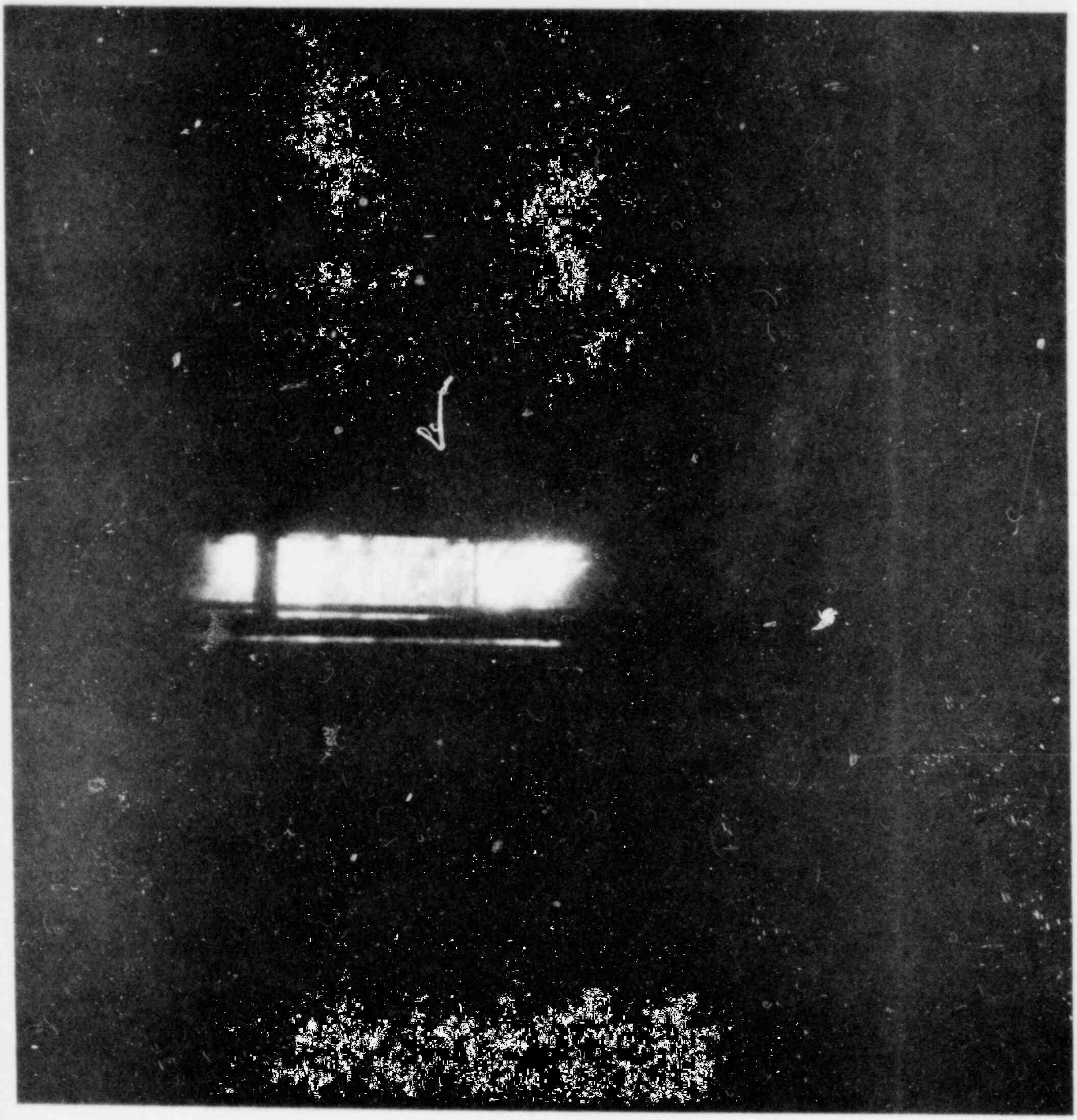
Seven full scale cable tray fire tests have been conducted at Sandia's Fire Research Facility to provide confirmatory data on Halon 1301 as a suppression measure for such fires. Five of these tests used a 6% Halon-1301 concentration as a fire suppression agent while two of the tests used a lack of ventilation (oxygen deprivation) as a suppression technique. Results of three tests which used IEEE-383 qualified cable in a horizontal configuration were as follows: 1) A 45-minute Halon soak did not allow re-ignition; 2) a 10-minute Halon soak did not allow re-ignition; 3) a 4-minute Halon soak did allow re-ignition.

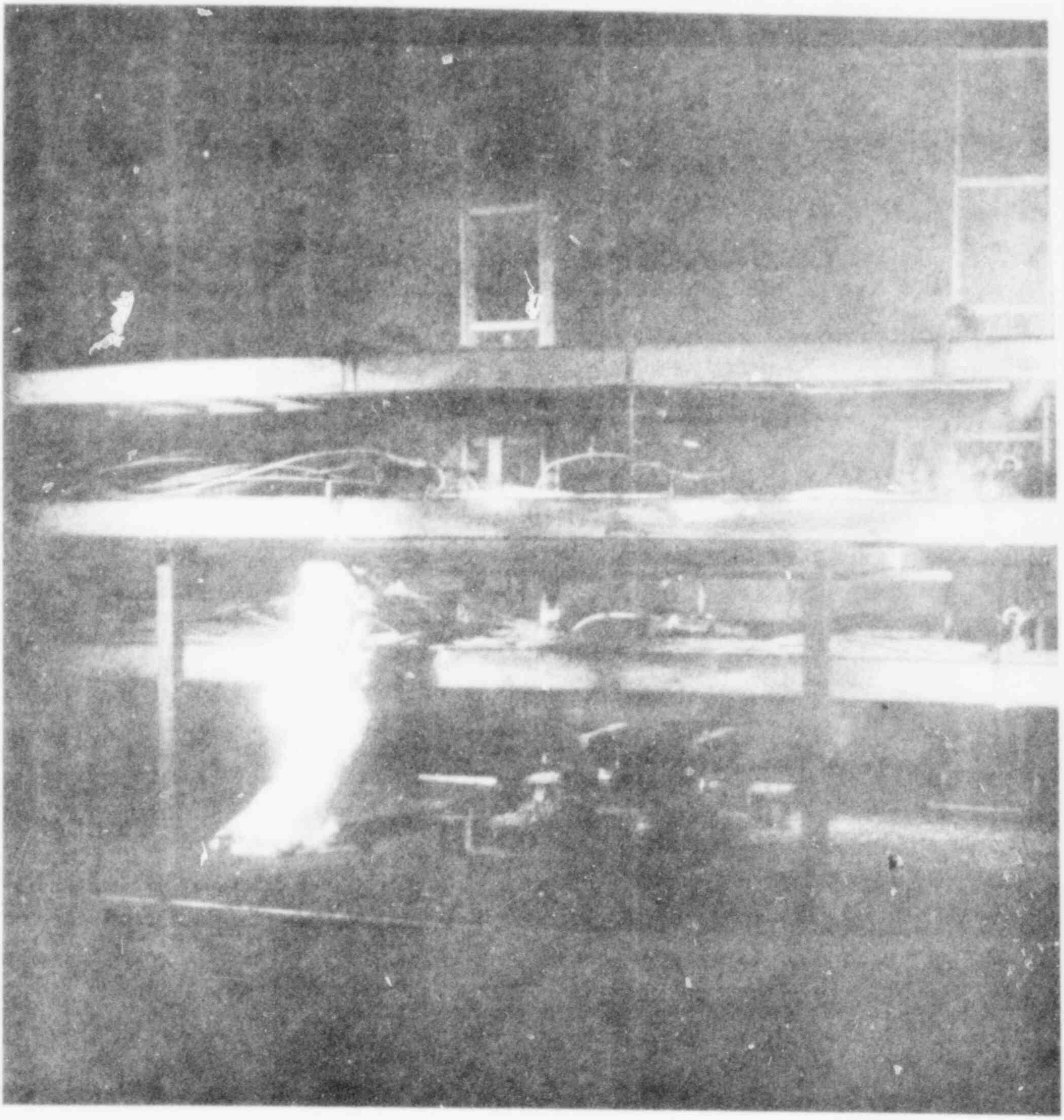
Results of two tests which used PE/PVC non-qualified cable were as follows: 1) a horizontal configuration with a 16-minute Halon soak did not allow re-ignition; 2) a vertical array test with a 5-minute Halon soak did not allow re-ignition. Results of two tests which used a horizontal configuration of IEEE-383 qualified cable and no Halon were as follows: 1) a 45-minute "buttoned up" period did not allow re-ignition when the ventilation system was turned on; 2) a 10-minute "buttoned up" period did allow re-ignition when the ventilation system was turned on.

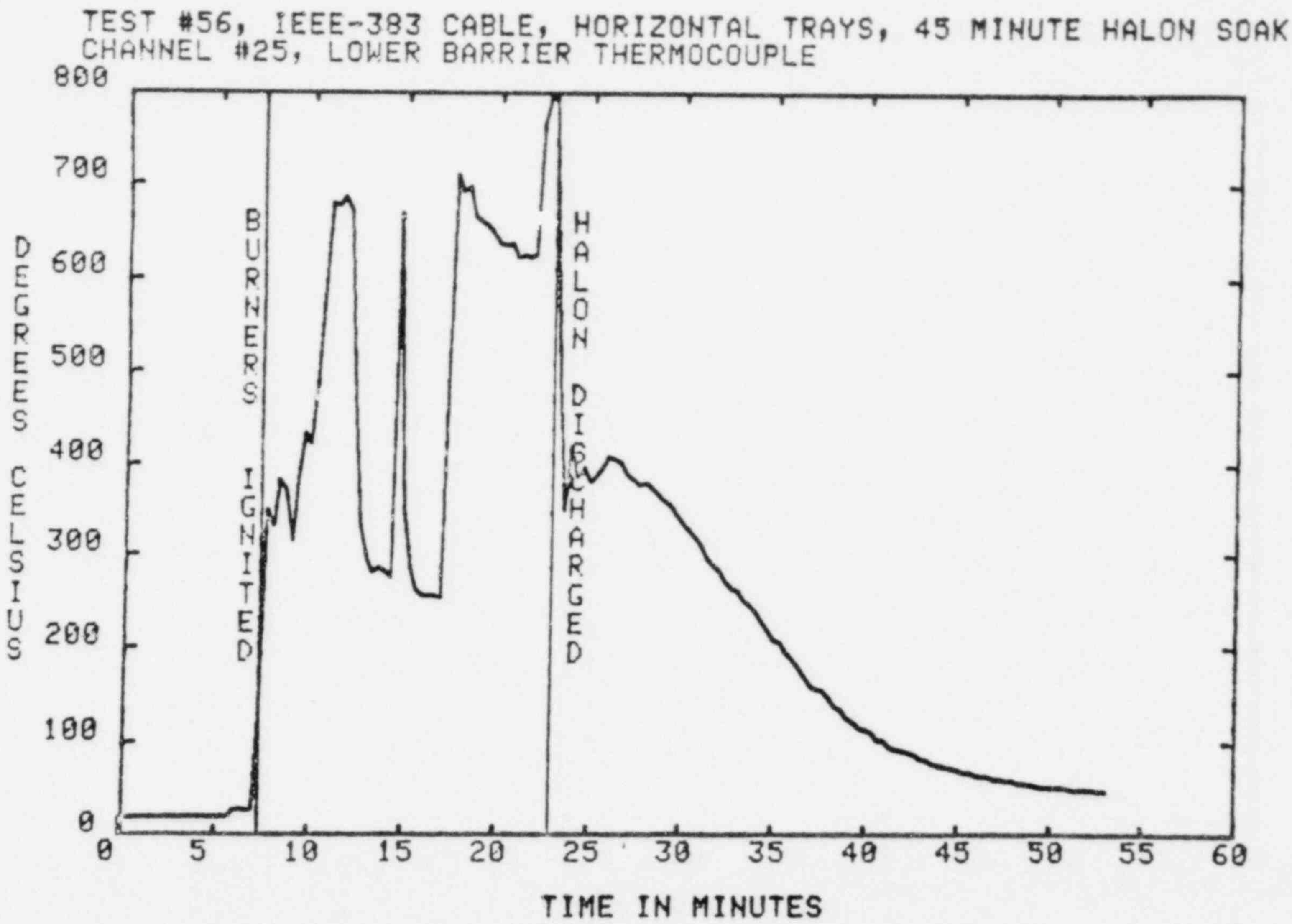
These results indicate that at least a 10-minute soak period should be used for Halon suppression systems before the room is entered. The closing of fire dampers in a room is a valuable aid in suppressing the fire, and might be adequate by itself if given sufficient time before the fire brigade enters the room. The critical question is: "How long does it take for the exposed hot surfaces of the cable insulation to cool below its ignition temperature?" The tests described here attempted to answer this question by providing temperatures taken at the surface of several exposed cables. Cooling time will be influenced somewhat by the ambient temperature and to a large extent by internal cable bundle temperatures. These temperatures have also been recorded for the fire tests described. Deep seated fires were obtainable in cable trays using IEEE-383 qualified cable but were not seen in the tests using unqualified cable where flaming was more easily acquired.

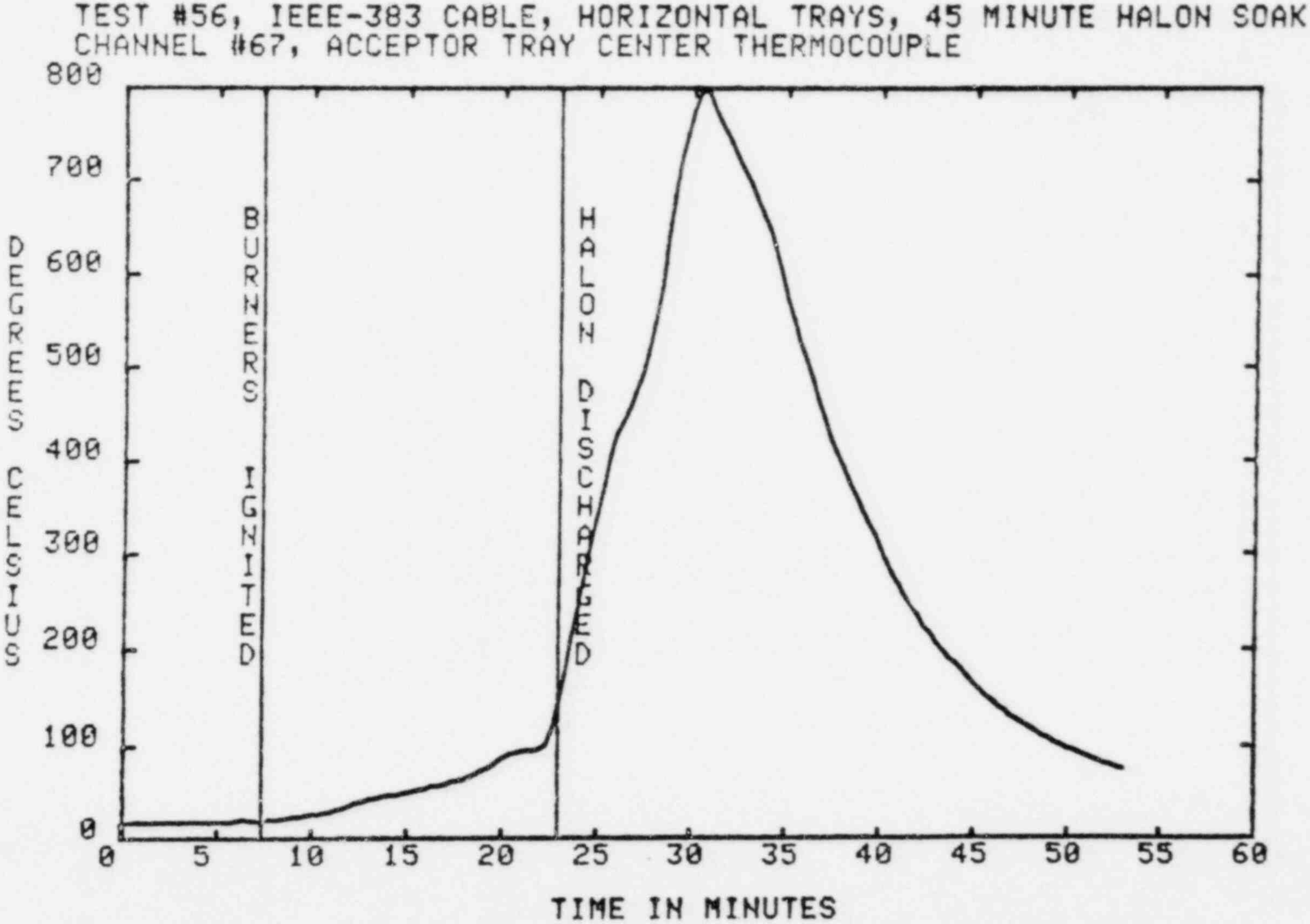
Future Work

More full scale testing will be completed to assess the effectiveness of CO_2 and water as suppression systems in extinguishing deep-seated cable tray fires. These results will be compared with each other as well as the results obtained from the Halon-1301 suppression tests. Full-scale replication testing of actual plant configurations and fire protection systems will be implemented. Confirmatory data on both line and point fire detection systems will be obtained and an in-situ test method developed. The evaluation of penetration fire stop methodology will be continued.

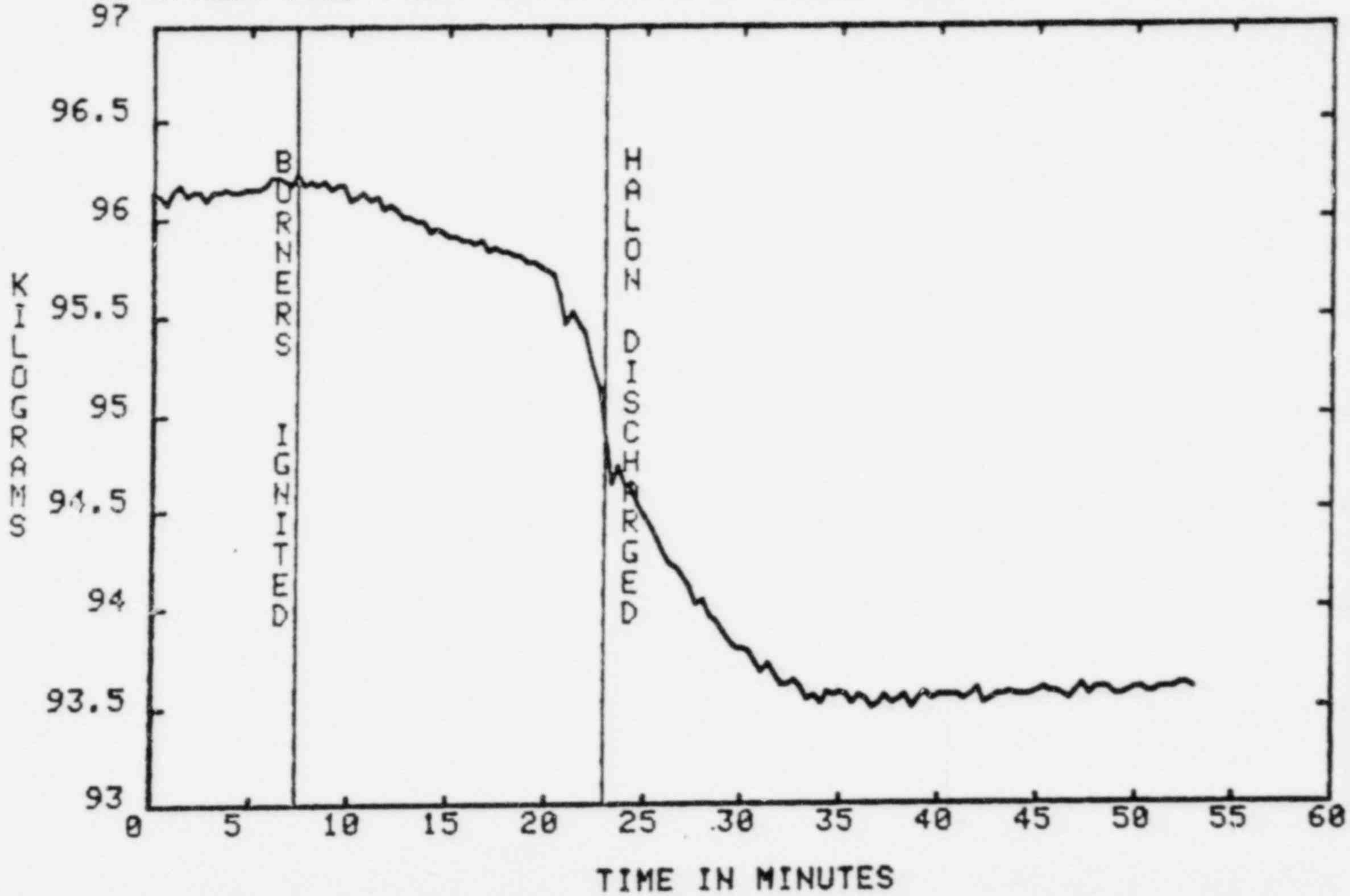


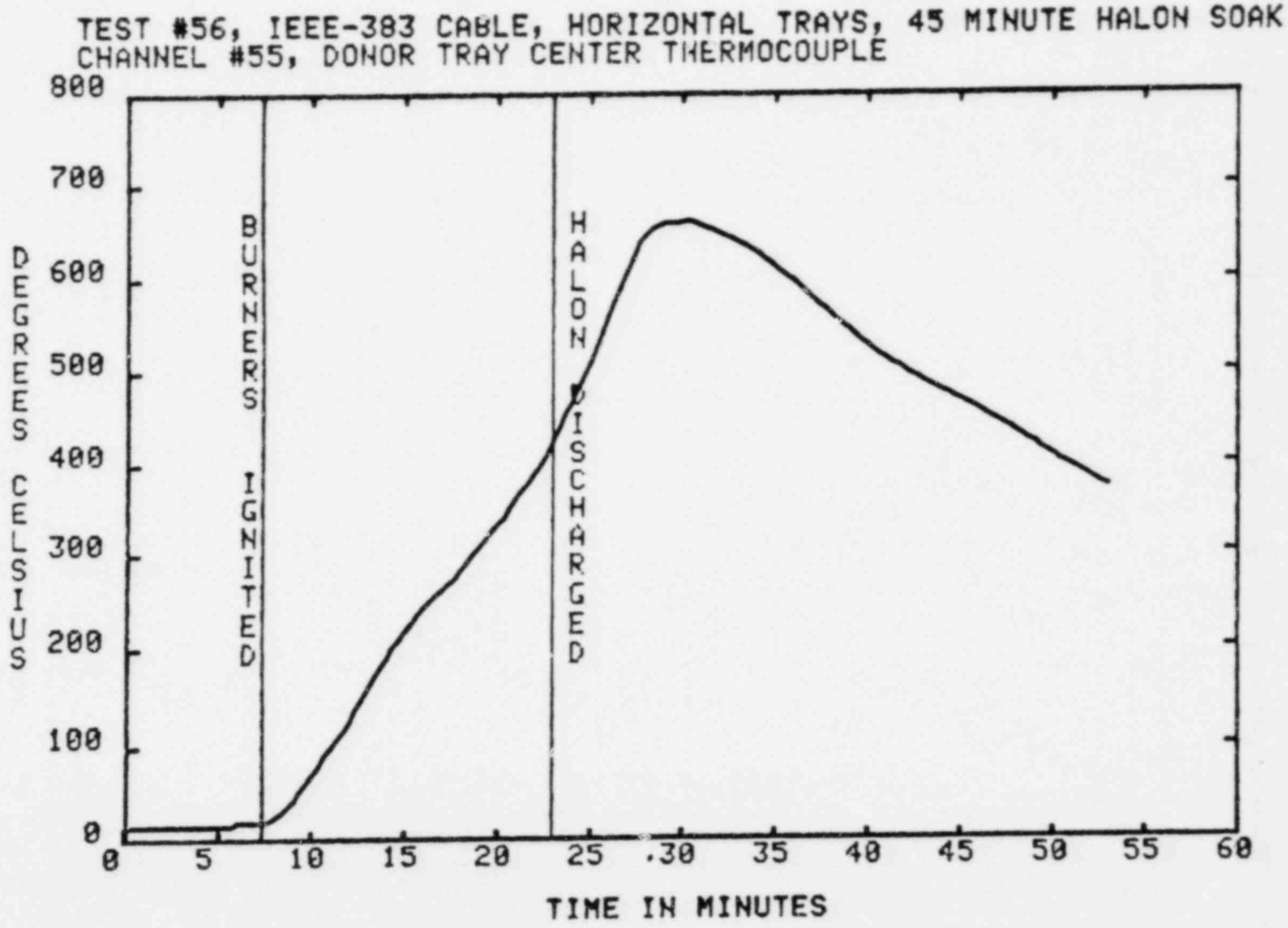


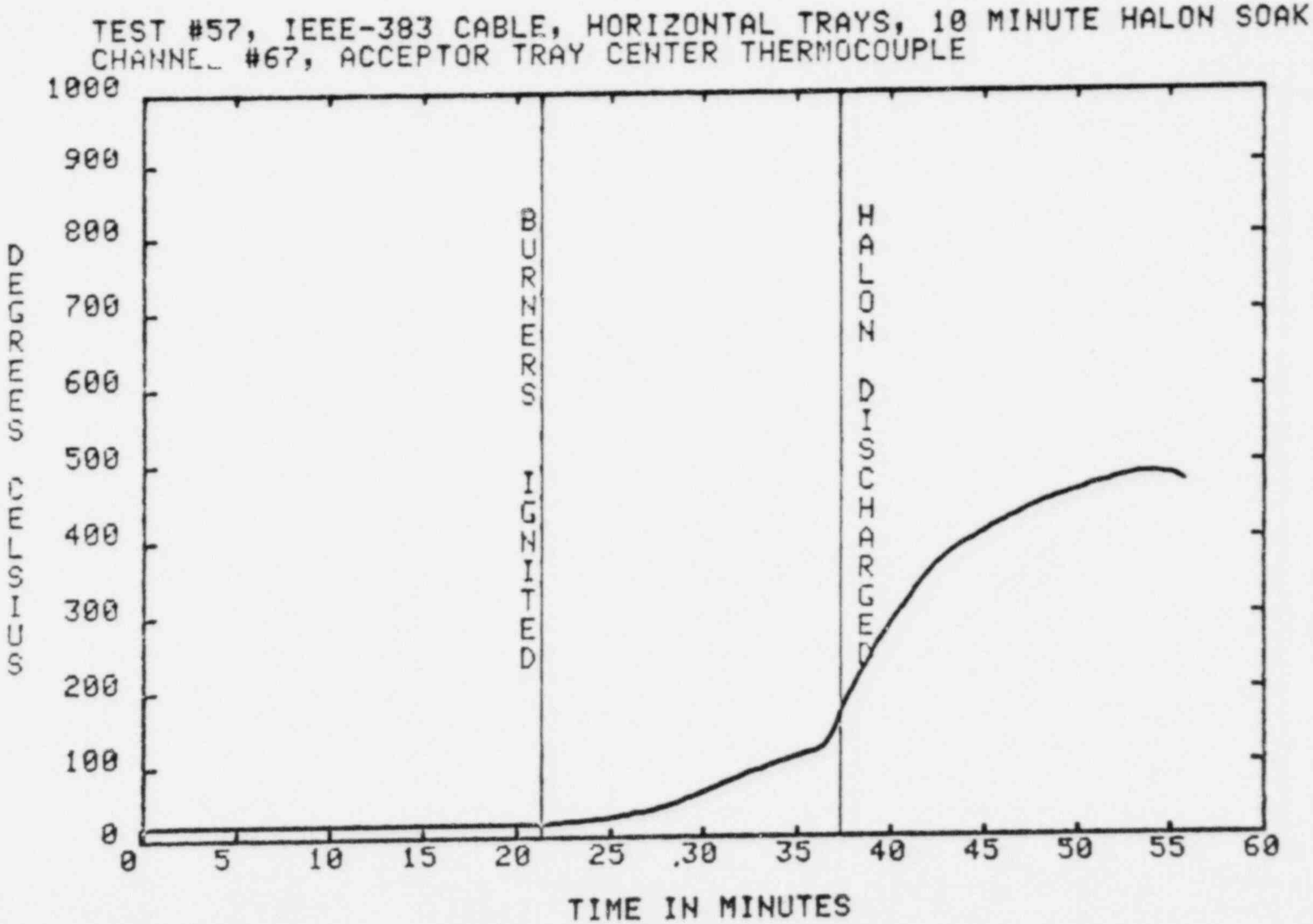

TEST #56, IEEE-383 CABLE, HORIZONTAL TRAYS, 45 MINUTE HALON SOAK
CHANNEL #25, LOWER BARRIER THERMOCOUPLE
DEGREES CELSIUS
0
100
200
300
400
500
600
700
800
BURNERS IGNITED
HALON DISCHARGED
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES


TEST #56, IEEE-383 CABLE, HORIZONTAL TRAYS, 45 MINUTE HALON SOAK
CHANNEL #67, ACCEPTOR TRAY CENTER THERMOCOUPLE
DEGREES CELSIUS
800
700
600
500
400
300
200
100
0
BURNERS IGNITED
HALON DISCHARGED
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES

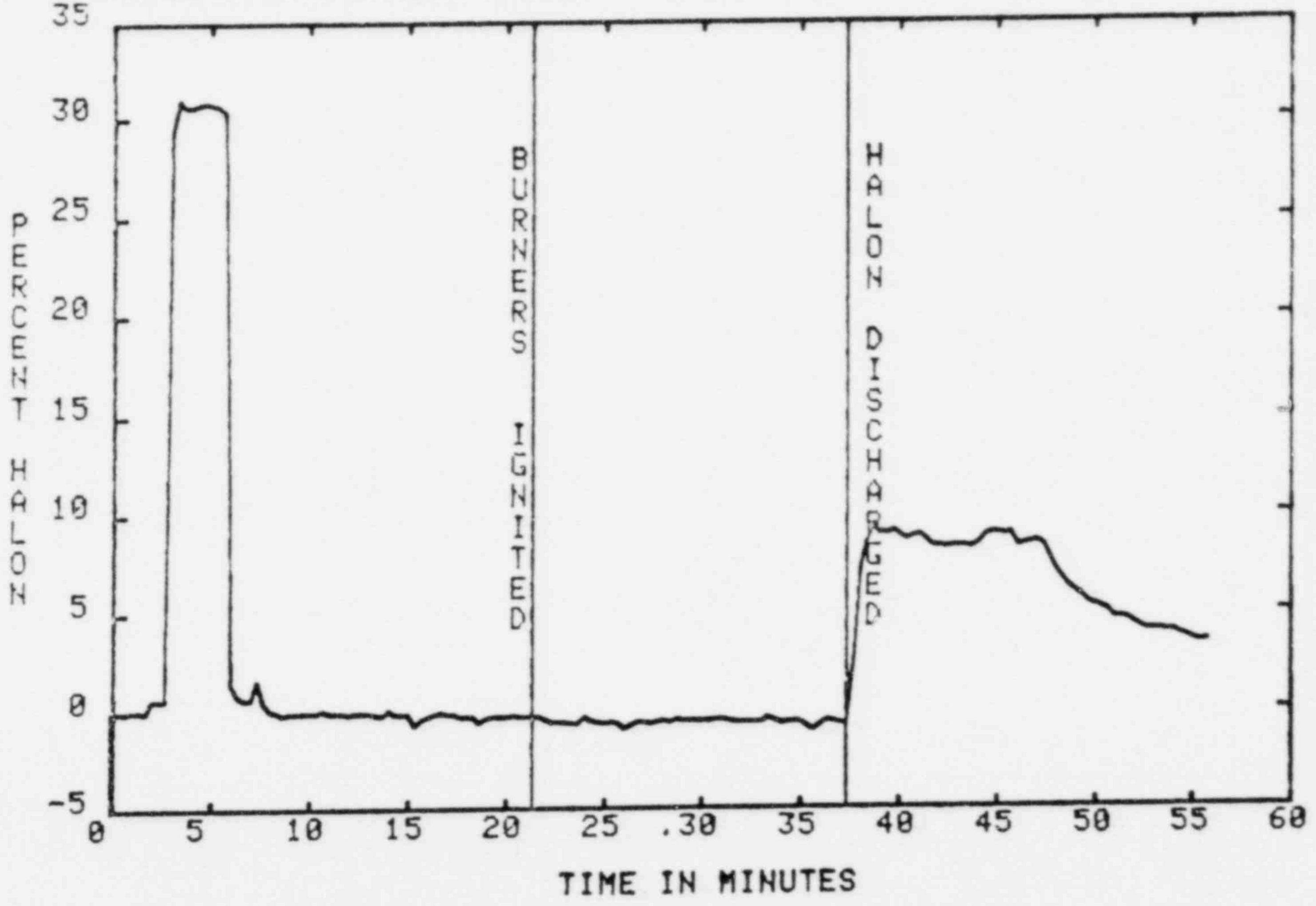
TEST #56, IEE_-383 CABLE, HORIZONTAL TRAYS, 45 MINUTE HALON SOAK
CHANNELS #80, #81: TOTAL MASS OF DONOR TRAY

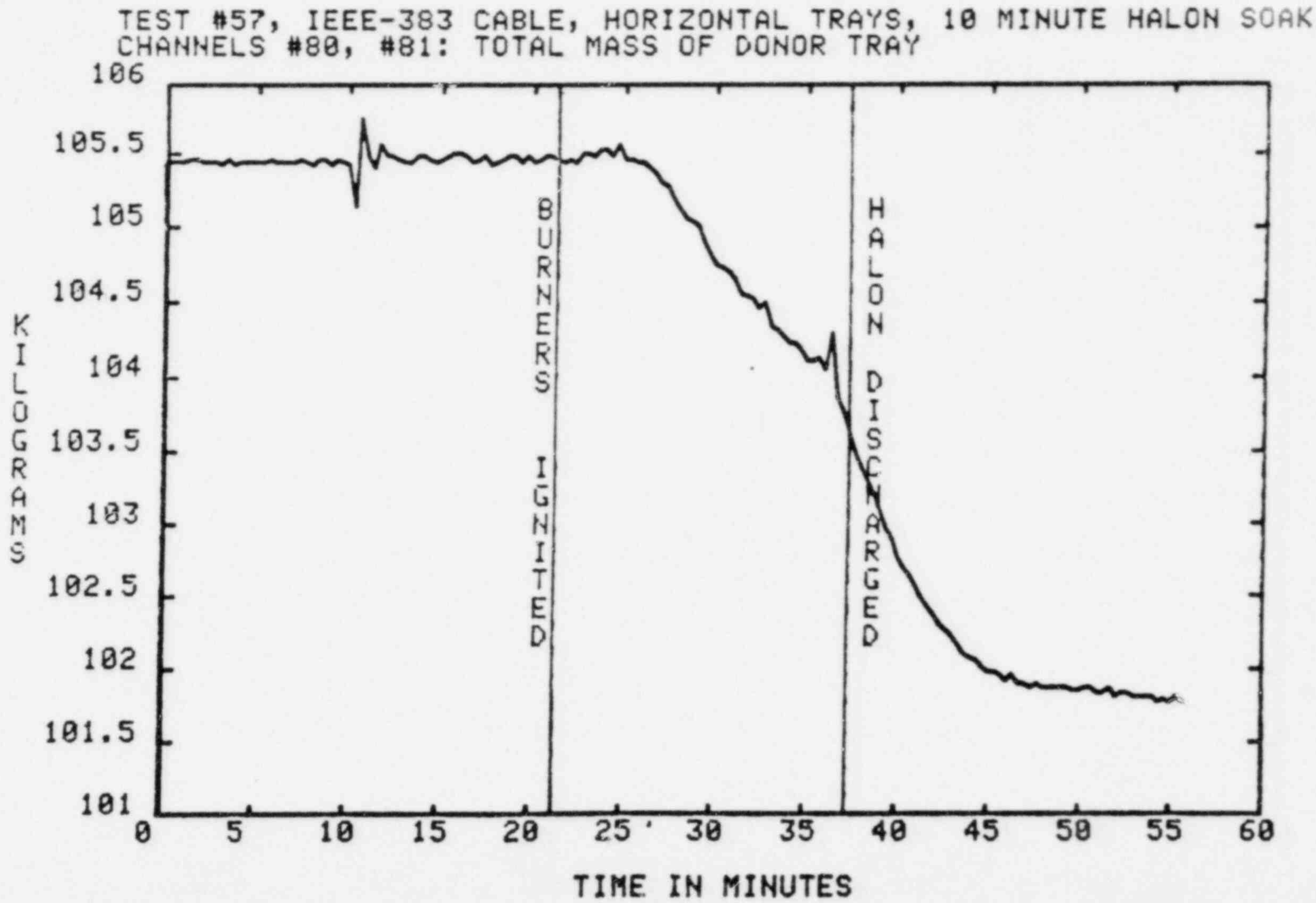



TEST #56, IEEE-383 CABLE, HORIZONTAL TRAYS, 45 MINUTE HALON SOAK
CHANNEL #55, DONOR TRAY CENTER THERMOCOUPLE
DEGREES CELSIUS
0
100
200
300
400
500
600
700
800
BURNERS IGNITED
HALON DISCHARGED
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES

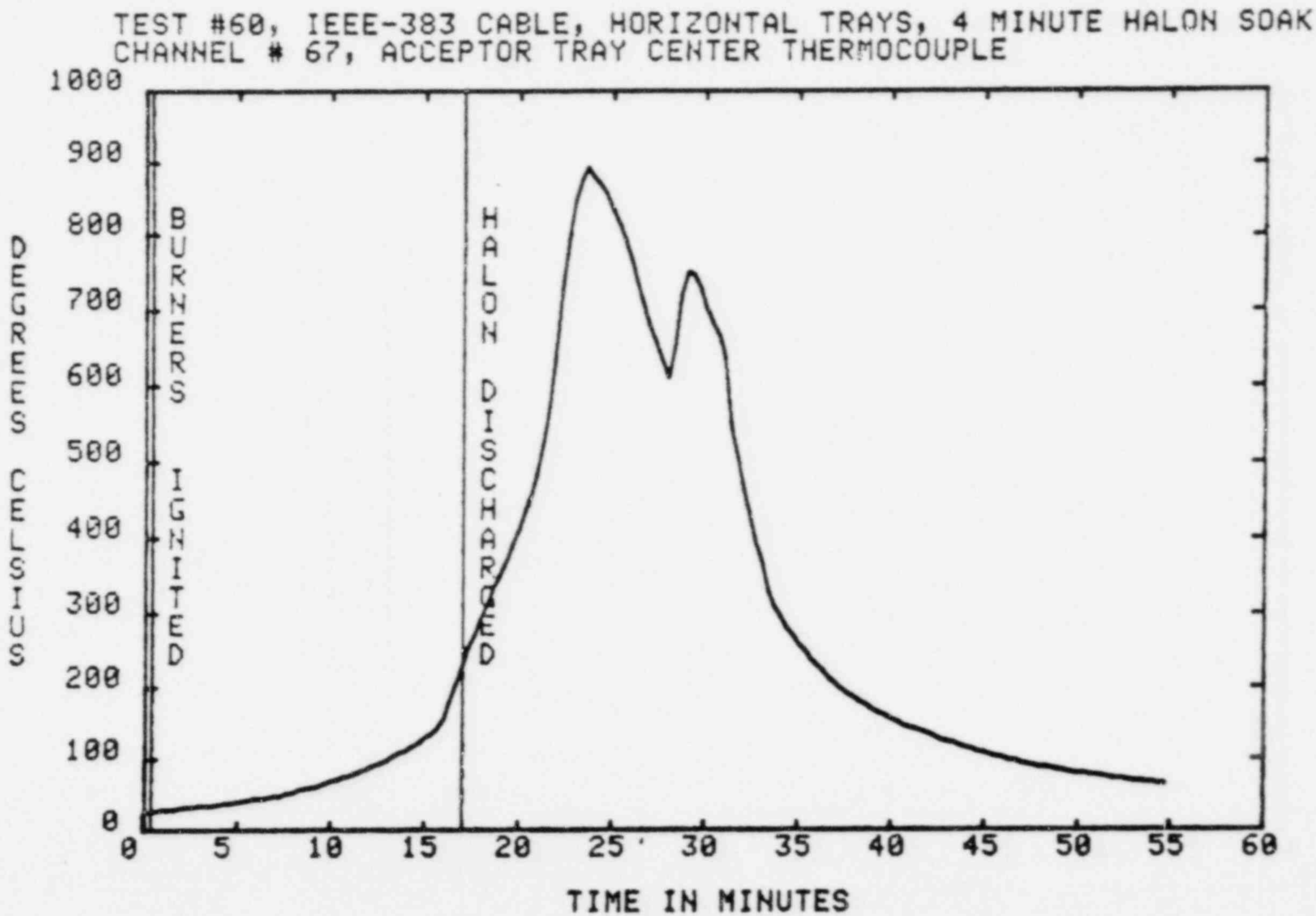

TEST #57, IEEE-383 CABLE, HORIZONTAL TRAYS, 10 MINUTE HALON SOAK
CHANNEL #67, ACCEPTOR TRAY CENTER THERMOCOUPLE
DEGREES CELSIUS
1000
900
800
700
600
500
400
300
200
100
0
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES
BURNERS IGNITED
HALON DISCHARGED

TEST #57, IEEE-383 CABLE, HORIZONTAL TRAYS, 10 MINUTE HALON SOAK
CHANNEL #73, PERCENT HALON IN BURN ROOM

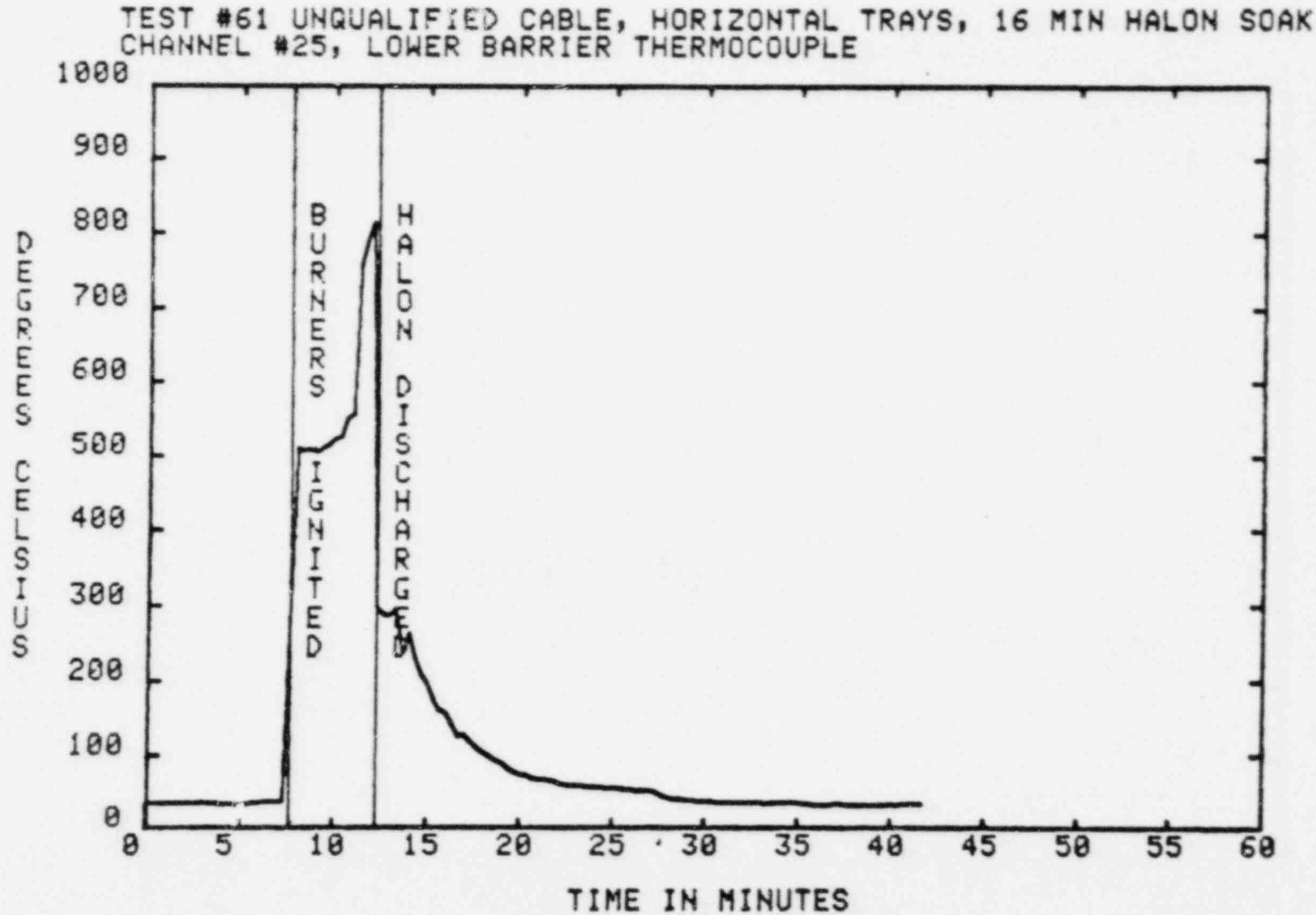


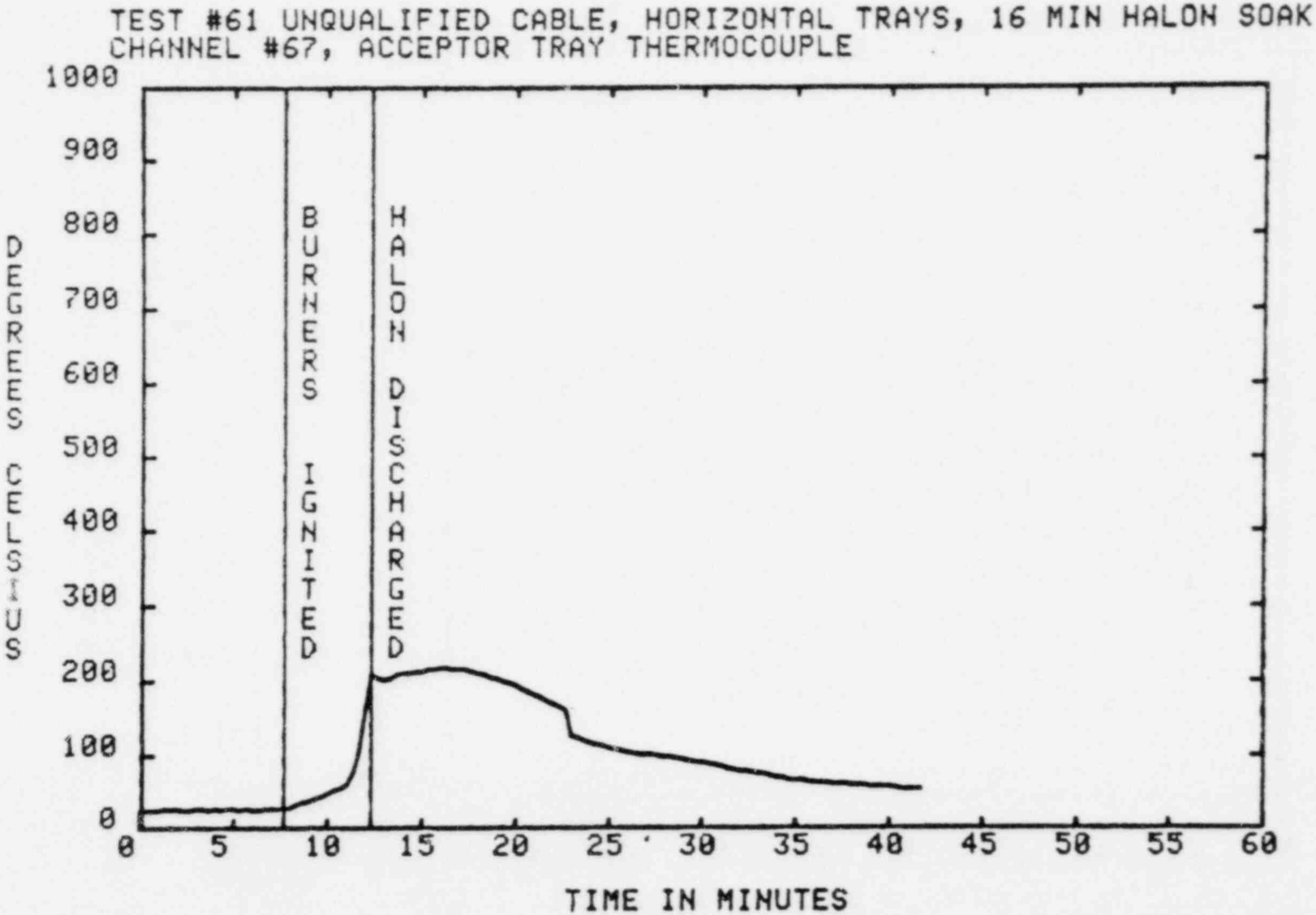


TEST #57, IEEE-383 CABLE, HORIZONTAL TRAYS, 10 MINUTE HALON SOAK
CHANNELS #80, #81: TOTAL MASS OF DONOR TRAY
KILOGRAMS
106
105.5
105
104.5
104
103.5
103
102.5
102
101.5
101
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES
BURNERS IGNITED
HALON DISCHARGED

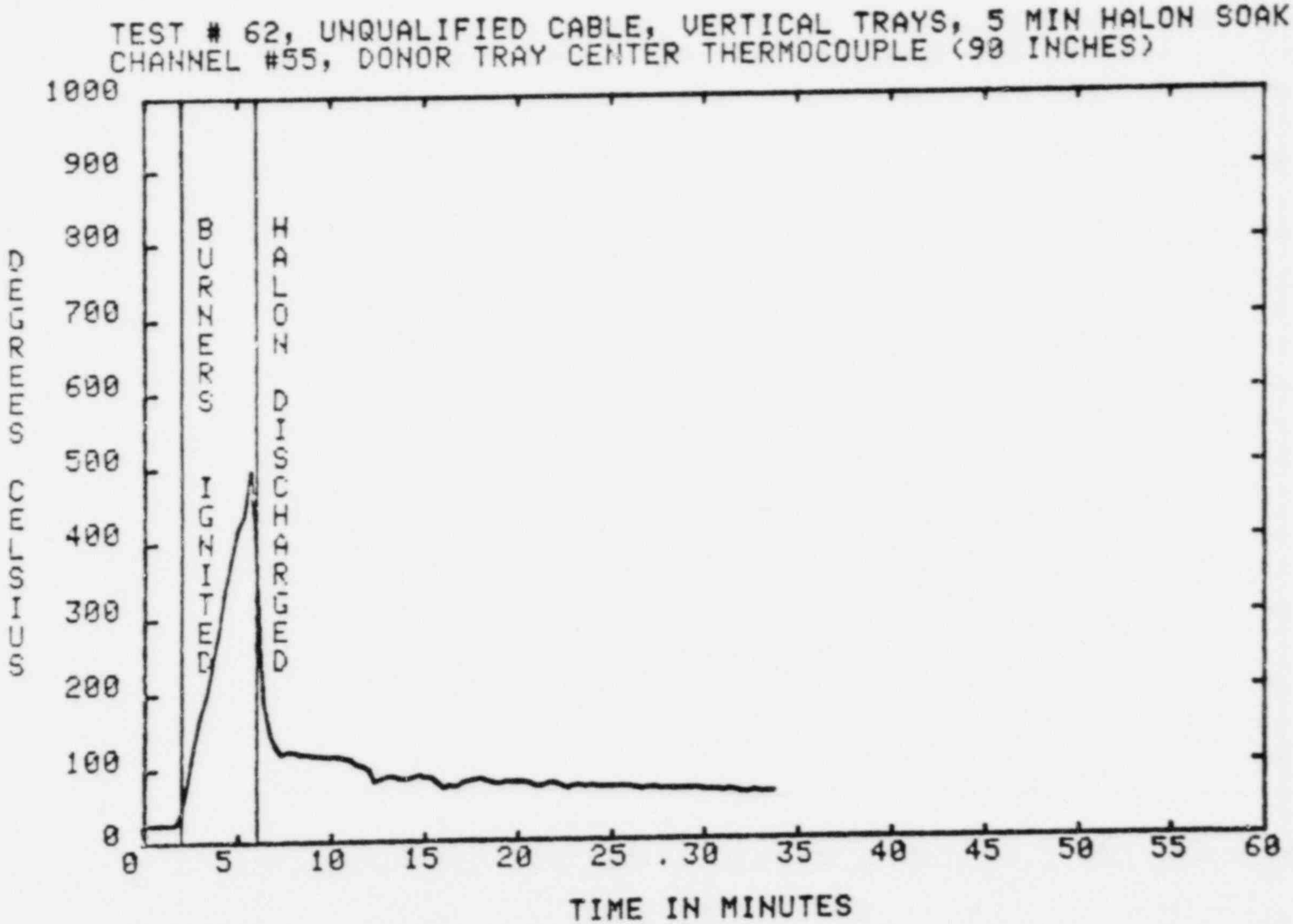


TEST #60, IEEE-383 CABLE, HORIZONTAL TRAYS, 4 MINUTE HALON SOAK
CHANNEL # 67, ACCEPTOR TRAY CENTER THERMOCOUPLE
DEGREES CELSIUS
BURNERS IGNITED
HALON DISCHARGED
1000
900
800
700
600
500
400
300
200
100
0
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES

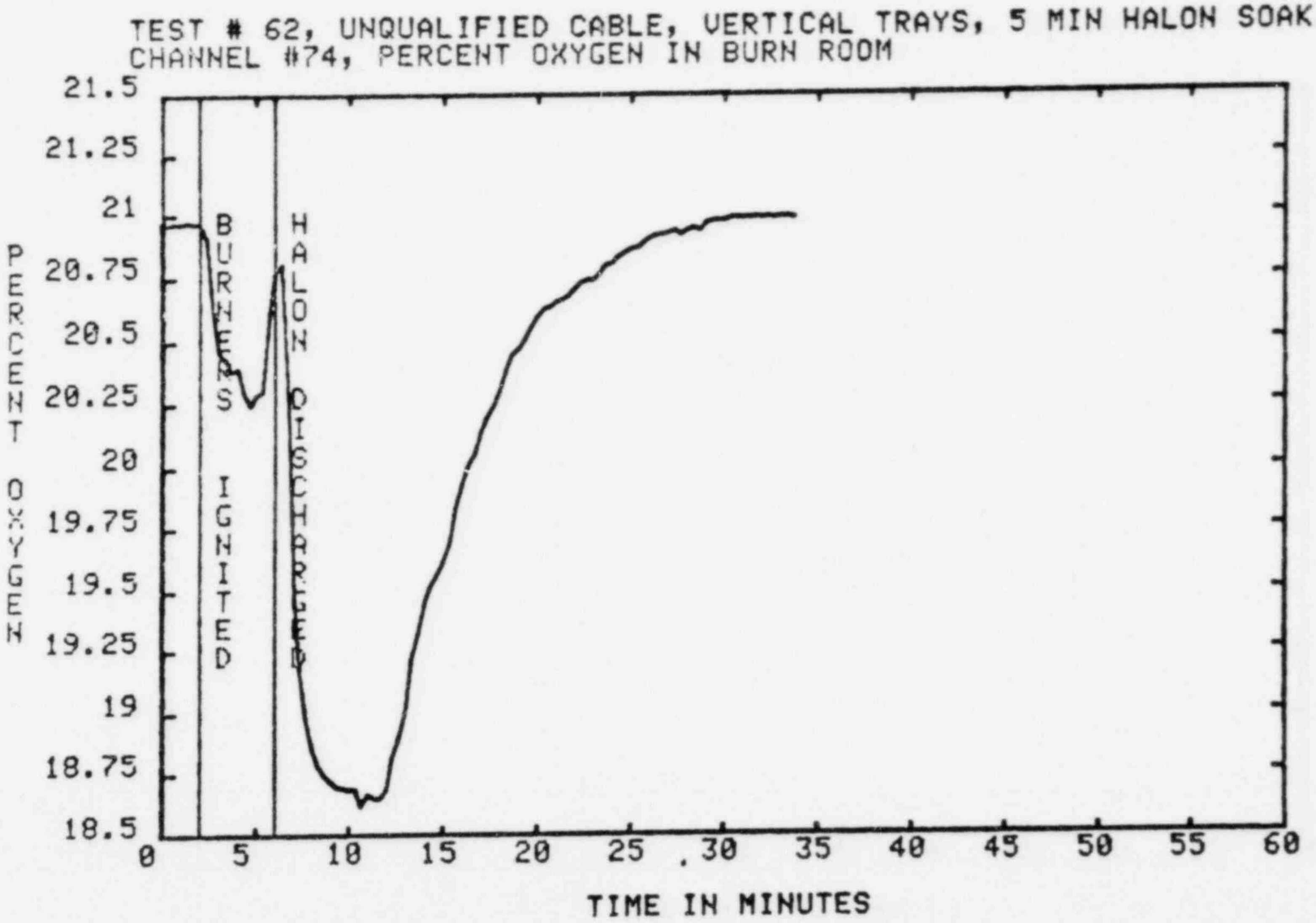

TEST #61 UNQUALIFIED CABLE, HORIZONTAL TRAYS, 16 MIN HALON SOAK
CHANNEL #25, LOWER BARRIER THERMOCOUPLE
DEGREES CELSIUS
1000
900
800
700
600
500
400
300
200
100
0
BURNERS IGNITED
HALON DISCHARGED
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES



TEST #61 UNQUALIFIED CABLE, HORIZONTAL TRAYS, 16 MIN HALON SOAK
CHANNEL #67, ACCEPTOR TRAY THERMOCOUPLE
DEGREES CELSIUS
0
100
200
300
400
500
600
700
800
900
1000
BURNERS IGNITED
HALON DISCHARGED
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES



TEST # 62, UNQUALIFIED CABLE, VERTICAL TRAYS, 5 MIN HALON SOAK
CHANNEL #55, DONOR TRAY CENTER THERMOCOUPLE (90 INCHES)
DEGREES CELSIUS
1000
900
800
700
600
500
400
300
200
100
0
BURNERS IGNITED
HALON DISCHARGED
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES


TEST # 62, UNQUALIFIED CABLE, VERTICAL TRAYS, 5 MIN HALON SOAK
CHANNEL #74, PERCENT OXYGEN IN BURN ROOM
PERCENT OXYGEN
21.5
21.25
21
20.75
20.5
20.25
20
19.75
19.5
19.25
19
18.75
18.5
BURNERS IGNITED
HALON DISCHARGED
0
5
10
15
20
25
30
35
40
45
50
55
60
TIME IN MINUTES

Fire Protection System Modeling: The Fire Resistance of Walls Penetrated by Electric Cables

L. W. Hunter and S. Favin

The Johns Hopkins University
Applied Physics Laboratory
Johns Hopkins Road
Laurel, MD 20810

Presentation to the Reactor Operational Safety Program Session (R. Feit, Chairman) at the Eighth Water Reactor Safety Research Information Meeting, on October 29, 1980.

INTRODUCTION

In nuclear power plants, electric cables frequently penetrate walls. An important safety issue is the fire resistance of the penetration seal, Fig. 1. A fire burning on one side of the wall heats the metal conductors in the exposed cables. The metal has a high thermal conductivity and tends to channel heat through the wall to the unexposed side.

This paper calculates the cable temperature on the unexposed side as a function of the intensity of the fire, the cable size and placement, the wall thickness, and the thermal properties of the materials involved. These calculations are offered to help in the selection of safe designs of penetration seals and in the interpretation of tests [1-3].* The calculations suggest that a simple formula may predict whether the backface cable temperature remains less than a preset safety limit.

The kind of installation considered here is one in which cables are poked through an opening in the wall and then the remaining space in the opening is sealed with a fire resistant filler material. The sealant is directly exposed to the fire; the neglect of shields which are sometimes present makes a pessimistic case of interest from a safety viewpoint. No conduits pass through the seal; a conduit would strongly affect the heat flow and a different analysis would be required. The same is true for a seal in a metal bulkhead. Electric current in the cables is assumed to be shut off when the fire is detected.

In practice, many different kinds of cables can pass through one penetration seal and the distribution of cables may be irregular. It is of interest for the present, however, to assume that the cables are equivalent and uniformly spaced in a symmetrical two-dimensional array.

Many sealants do not ablate significantly. Any reactions or phase changes that take place leave behind inert residue. When exposed to a steady

* Numbers in brackets designate References at end of paper.

(developed) fire, these seals eventually reach homogeneous, steady conditions in which temperatures are as high as possible. We calculate the steady temperatures over a range of fire conditions. These results are useful for predicting when the seal is safe. To an approximation, the results may be applied at each instant when the fire intensity varies with time.

Thermal properties of the cables and the wall are taken to be constant, since the uncertainty in their values in practical applications can be comparable to their variations with temperature. In addition, the present results suggest that the thermal conductivity of the wall has only a small effect in steady state.

BASIC MATHEMATICAL DESCRIPTION

The heat balance equations are developed in this section for a seal which has attained homogeneous, steady conditions.

The cables are represented by homogeneous cylinders, while the sealant and wall together are represented by a homogeneous wall, Fig. 2. The fire on one side imparts heat to these cables and the wall. From $z = 0$ to $z = \ell$, the cables make perfect thermal contact with the wall and heat is conducted into the wall in the r and z directions. Radial temperature gradients are expected to be small in the cables. Heat is lost to the unexposed room at $z \geq \ell$. The convective heat fluxes, radiative fluxes, temperatures, and thermal conductivities are shown in Fig. 2. The labeling is as follows:

1 - fire
2 - unexposed room
3 - cable
No number label - wall

Since the cables are assumed to be equivalent and uniformly distributed, a good approximation is that the radial temperature distribution has an extremum ($\partial T/\partial r = 0$) at some distance $r = b$ from the center of any cable. The maximum allowed cable (or wall) temperature on the unexposed side is denoted T_m.

It is possible to linearize the heat fluxes in such a way that predictions that the seal is safe remain valid. The linearizations shown in Figs. 3 and 4 achieve the desired result in terms of the equilibrium fire-side temperature, T_1, given by

$$H_1'(T_1' - T_1) + F_1 - \sigma T_1^4 = 0, \tag{1}$$

and the effective heat transfer coefficients

$$H_1 = H_1' + 4\sigma T_1^3 \tag{2}$$

and

$$H_2 = H_2' + \sigma(T_m + T_2)(T_m^2 + T_2^2). \tag{3}$$

When the backface cable temperature is too hot, then the fluxes leaving the wall and cables, Fig. 4, are underestimated, while the influxes, Fig. 3, are overestimated; therefore, the calculated backface temperature will be higher than the true value, which already exceeds T_m. The calculation will properly predict that the seal fails.

The linearization is accurate when the backface temperatures are near the decision point, T_m, and when the fire is intense, for then the fire-side cable and wall temperatures are close to equilibrium.

The heat balance equations which determine the temperature of a cable may now be written as

$$\frac{d^2T_3}{dz^2} + \frac{2H_1}{ak_3}(T_1 - T_3) = 0, \text{ for } z < 0, \tag{4}$$

$$\frac{d^2T_3}{dz^2} + \frac{2k}{ak_3}\frac{\partial T}{\partial r} = 0, \text{ for } 0 < z < \ell,\ r = a, \tag{5}$$

$$\frac{d^2T_3}{dz^2} - \frac{2H_2}{ak_3}(T_3 - T_2) = 0, \text{ for } \ell < z, \tag{6}$$

subject to the boundary conditions that $T_3 \to T_1$ as $z \to -\infty$, $T_3 \to T_2$ as $z \to \infty$ and that T_3 and dT_3/dz are continuous at $z = 0$ and $z = \ell$. The wall temperature satisfies

$$\frac{\partial^2 T}{\partial z^2} + \frac{1}{r}\frac{\partial}{\partial r}\left(r\frac{\partial T}{\partial r}\right) = 0, \text{ for } 0 < z < \ell,\ a < r < b, \tag{7}$$

subject to

$$-k\frac{\partial T}{\partial z} = H_1(T_1 - T) \text{ at } z = 0 \tag{8}$$

$$-k\frac{\partial T}{\partial z} = H_2(T - T_2) \text{ at } z = \ell \tag{9}$$

$$\frac{\partial T}{\partial r} = 0 \text{ at } r = b \tag{10}$$

and

$$T = T_3 \text{ at } r = a. \tag{11}$$

MATHEMATICAL FORMULAS

The solution of the cable temperature equations inside the wall may be written

$$\Theta_3 \equiv \frac{T_1 - T_3}{T_1 - T_2} = \frac{z^* + \eta_1^{-1}}{1 + \eta_1^{-1} + \eta_2^{-1}} +$$

$$+ \beta[\eta_1 + \eta_1\eta_2 + \eta_2]^{-1} \int_0^1 dz^{*\prime} G(z^*, z^{*\prime}) \left.\frac{\partial\Theta}{\partial r^*}\right|_{r^* = a^*} , \tag{12}$$

where

$$z^* \equiv z/\ell,\ z^{*\prime} \equiv z'/\ell,\ r^* \equiv r/\ell,\ a^* \equiv a/\ell, \tag{13}$$

$$\eta_1 \equiv \left(\frac{2aH_1}{k_3}\right)^{\frac{1}{2}} \frac{\ell}{a}\ ,\ \eta_2 \equiv \left(\frac{2aH_2}{k_3}\right)^{\frac{1}{2}} \frac{\ell}{a}\ , \tag{14}$$

$$\beta \equiv 2\,\frac{k}{k_3}\,\frac{\ell}{a}\ , \tag{15}$$

and

$$\Theta \equiv \frac{T_1 - T}{T_1 - T_2} \tag{16}$$

The kernel is

$$G(z^*, z^{*\prime}) = \begin{cases} [1 + \eta_1 z^{*\prime}][1 + \eta_2(1 - z^*)] & \text{if } z^{*\prime} < z^*, \\ [1 + \eta_1 z^*][1 + \eta_2(1 - z^{*\prime})] & \text{if } z^* < z^{*\prime}. \end{cases} \tag{17}$$

Thus, the cable temperature depends on the wall temperature T nearby.

It follows from Eqs. (7)-(10) that the wall temperature T near the cable has an expansion of the form

$$\Theta = \frac{z^* + (H_1^*)^{-1}}{1 + (H_1^*)^{-1} + (H_2^*)^{-1}}$$

$$+ \sum_n \frac{C_n}{\alpha_n^*}\left[\frac{I_0(\alpha_n^* r^*)K_1(\alpha_n^* b^*) + I_1(\alpha_n^* b^*)K_0(\alpha_n^* r^*)}{I_0(\alpha_n^* a^*)K_1(\alpha_n^* b^*) + I_1(\alpha_n^* b^*)K_0(\alpha_n^* a^*)}\right] Z_n^*(z^*), \tag{18}$$

where

$$H_1^* \equiv \frac{\ell H_1}{k}\ ,\ H_2^* \equiv \frac{\ell H_2}{k}\ , \tag{19}$$

and

$$b^* \equiv \frac{b}{\ell}\ . \tag{20}$$

The I and K are modified Bessel functions. The $\alpha_n^\star$ are the positive roots of

$$(\alpha^\star)^2 - (H_1^\star + H_2^\star)\alpha^\star \cot \alpha^\star - H_1^\star H_2^\star = 0 \;, \tag{21}$$

while the functions $Z_n^\star$ are given by

$$Z_n^\star(z^\star) = \tag{22}$$

$$\frac{(\alpha_n^\star)^{\frac{1}{2}}\left[\sin(\alpha_n^\star z^\star) + \frac{\alpha_n^\star}{H_1^\star}\cos(\alpha_n^\star z^\star)\right]}{\left\{\left[-\frac{1}{2}\sin\alpha_n^\star \cos\alpha_n^\star + \frac{\alpha_n^\star}{2}\right] + \frac{\alpha_n^\star}{H_1^\star}\sin^2\alpha_n^\star + \left(\frac{\alpha_n^\star}{H_1^\star}\right)^2\left[\frac{1}{2}\cos\alpha_n^\star \sin\alpha_n^\star + \frac{\alpha_n^\star}{2}\right]\right\}^{\frac{1}{2}}}$$

The expansion coefficients C_n in Eq. (18) are unknown at this point. The matching condition at r = a, Eq. (11), remains to be applied. This condition is equivalent to the following set of equations which determine the C_n:

$$\sum_n M_{mn}C_n = -(\alpha_n^\star)^{-1}(1 + \eta_1^{-1} + \eta_2^{-1})^{-1} \times$$

$$[Z_m^\star(0)(1 - H_1^\star\eta_1^{-1}) - Z_m^\star(1)(1 - H_2^\star\eta_2^{-1})], \tag{23}$$

where

$$M_{mn} = \delta_{mn} - \frac{\alpha_m^\star \beta G_{mn}}{(\eta_1 + \eta_1\eta_2 + \eta_2)}\left[\frac{I_1(\alpha_n^\star a^\star)K_1(\alpha_n^\star b^\star) - I_1(\alpha_n^\star b^\star)K_1(\alpha_n^\star a^\star)}{I_0(\alpha_n^\star a^\star)K_1(\alpha_n^\star b^\star) + I_1(\alpha_n^\star b^\star)K_0(\alpha_n^\star a^\star)}\right] \tag{24}$$

in which

$$G_{mn} \equiv \int_0^1 dz^\star \int_0^1 dz^{\star\prime} Z_m^\star(z^\star)G(z^\star, z^{\star\prime})Z_n^\star(z^{\star\prime}) \tag{25}$$

$$= (\alpha_m^\star)^{-2}(\eta_1 + \eta_1\eta_2 + \eta_2)\delta_{mn}$$

$$+ (\alpha_m^\star)^{-2}(\alpha_n^\star)^{-2}Z_m^\star(0)Z_n^\star(0)[H_1^\star - \eta_1][H_1^\star + H_1^\star\eta_2 + \eta_2]$$

$$+ (\alpha_m^\star)^{-2}(\alpha_n^\star)^{-2}Z_m^\star(1)Z_n^\star(1)[H_2^\star - \eta_2][H_2^\star + H_2^\star\eta_1 + \eta_1]$$

$$+ (\alpha_m^\star)^{-2}(\alpha_n^\star)^{-2}[Z_m^\star(0)Z_n^\star(1) + Z_n^\star(0)Z_m^\star(1)][H_1^\star - \eta_1][H_2^\star - \eta_2]. \tag{26}$$

It is possible to evaluate the cable temperature (at $z = \ell$) by Eq. (18), considering that the cable and wall temperatures are equal at r = a:

$$\frac{T_3 - T_2}{T_1 - T_2} = \frac{(H_2^*)^{-1}}{1 + (H_1^*)^{-1} + (H_2^*)^{-1}} - \sum_n (\alpha_n^*)^{-1} C_n Z_n^*(1). \tag{27}$$

This series converges well except when k is very small. Then a more convenient series is obtained by substituting Eq. (18) into Eq. (12):

$$\frac{T_3 - T_2}{T_1 - T_2} = \frac{\eta_2^{-1}}{1 + \eta_1^{-1} + \eta_2^{-1}} - \frac{\beta}{\eta_1 + \eta_1\eta_2 + \eta_2} \times$$

$$\sum_n (\alpha_n^*)^{-2} \left[\frac{I_1(\alpha_n^* a^*)K_1(\alpha_n^* b^*) - I_1(\alpha_n^* b^*)K_1(\alpha_n^* a^*)}{I_0(\alpha_n^* a^*)K_1(\alpha_n^* b^*) + I_1(\alpha_n^* b^*)K_0(\alpha_n^* a^*)}\right] \times$$

$$[(H_1^* - \eta_1)Z_n^*(0) + (\eta_1 + \eta_1 H_2^* + H_2^*)Z_n^*(1)]C_n \;. \tag{28}$$

This result converges well except when a or k_3 is very small. Together, Eqs. (27) and (28) cover all cases.

NO STUBS

In practice, the cables emerging from both sides of a seal do not extend on forever, as assumed in Fig. 2. They might bend or intertwine, for example, and, for convenience in fire tests, the cables might be chopped off to leave only short stubs on either side of the seal. This section addresses the limiting case in which the cables are sheared off at the wall faces and no stubs are left at all. It is expected that practical cases are bounded between no stubs and the infinite cables.

The heat balance equations for the length of cable left inside the seal are

$$\frac{d^2T_3}{dz^2} + \frac{2k}{ak_3}\frac{\partial T}{\partial r} = 0,\ 0 < z < \ell,\ r = a, \tag{29}$$

$$-k_3\frac{dT_3}{dz} = H_1(T_1 - T_3) \text{ at } z = 0, \tag{30}$$

$$-k_3\frac{dT_3}{dz} = H_2(T_3 - T_2) \text{ at } z = \ell. \tag{31}$$

The wall equations, Eqs. (7)-(10), and the matching condition, Eq. (11), still apply.

The solution of these equations may be developed as in the previous section. It turns out that the no stub formulas for Θ_3 and Θ, where $0 < z < \ell$, are very similar in appearance to the infinite cable results. The only difference is that η_1 and η_2, defined by Eq. (14), are replaced by new groups:

$$\eta_1 \rightarrow \omega_1 \equiv \frac{H_1\ell}{k_3}\ ,\ \eta_2 \rightarrow \omega_2 \equiv \frac{H_2\ell}{k_3}\ . \tag{32}$$

NUMERICAL RESULTS

This section presents calculated steady cable temperatures on the unexposed side of the wall. The calculations explore the influence of the fire intensity, the cable size and placement, the wall thickness and the thermal properties of the materials involved.

The calculations are based on the two series, Eqs. (27) and (28). By evaluating both series wherever possible, cross checks on the results may be obtained and truncation errors kept within bounds.

The independent parameters which characterize the seal are varied one at a time around nominal values listed in Table 1. These data apply to #12 AWG (American Wire Gage) single-conductor cables spaced one radius apart. The thermal conductivity k_3 assigned to the cables is that of copper. The conductivity k of the wall is somewhat arbitrary because, in the model, the wall includes the sealant; it will turn out, however, that k is not important in steady state. The nominal values of T_1', F_1 and H_1' roughly represent a fully developed cable tray fire [4, 5]. The value H_2' is affected by air circulation, but plays only a minor role since radiant losses tend to be larger than the convective losses in the unexposed room. The maximum safe cable temperature T_m is put at 800K ($\simeq 1000^{\circ}F$) at which point the cables could present a fire hazard in the unexposed room.

Figures 5-9 show qualitatively how the backface temperature T_3 of infinite cables is affected by the characteristics of the seal and the fire intensity. Near nominal conditions, T_3 is made cooler by increases in the cable spacing (b) and the wall thickness (ℓ). The backface cable temperature is made hotter by increases in the fire temperature (T_1'), the fire radiation (F_1), the thermal conductivity of the wall (k) or cable (k_3), and the cable radius (a). Further discussion of the results is given in the figure captions.

The heat fluxes to and from the seal were linearized, Figs. 3 and 4, in such a way that the calculated (approximate) backface cable temperature T_3 exceeds the safety limit T_m when the exact T_3 does so. Figures 5-9 were calculated with T_m = 800K. It is seen that $T_3/T_m < 1$ throughout Figs. 5-9. Hence the exact backface cable temperature would remain less than 800K in all the situations considered.

DISCUSSION

For cables packed to within a separation distance of 6 radii ($b = 4a$), Fig. 7 shows that the limiting temperature as $b \to a$ is an accurate approximation near nominal conditions. The limit is given by

$$\frac{T_3 - T_2}{T_1 - T_2} \to \frac{\eta_2^{-1}}{1 + \eta_1^{-1} + \eta_2^{-1}} \quad \text{as } b \to a. \tag{33}$$

This formula promises to be convenient in practical applications.

Equation (33) describes steady states in which no heat flows from the cables into the wall. The wall conductivity (k) drops out completely. This result justifies the way the model treats the wall and sealant together as one homogeneous wall. Heat does not flow into the wall under nominal conditions because the wall is as hot as the cables; this behavior is indeed observed in practice [6].

We have seen that one of the variables in fire tests is the length of exposed cable stubs. If the cables are sheared off at both the front and back faces of the wall, Eqs. (32) and (33) combine to show that the backface cable temperature is given approximately by

$$\frac{T_3 - T_2}{T_1 - T_2} \to \frac{\omega_2^{-1}}{1 + \omega_1^{-1} + \omega_2^{-1}} \tag{34}$$

This formula gives $T_3 = 779K$ under nominal conditions, compared to $T_3 = 479K$ with infinite cables. Thus, shortening the stubs would make the fire test significantly more severe, perhaps even making the test unrealistic. This possibility is well known to fire safety engineers [6].

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NOMENCLATURE

a	=	cable radius
a^*	=	Eq. (13)
b	=	one half the distance between axes of nearest neighbor cables
b^*	=	Eq. (20)
C_n	=	expansion coefficient in Eq. (18)
F_1	=	radiant heat flux from the fire
$G(z^*,z^{*'})$	=	Eq. (17)
G_{mn}	=	Eq. (25)
H_1'	=	heat transfer coefficient from the fire, Fig. 3
H_1	=	Eq. (2)
H_1^*	=	Eq. (19)
H_2'	=	heat transfer coefficient into the unexposed room, Fig. 4
H_2	=	Eq. (3)
H_2^*	=	Eq. (19)
I, J, K	=	modified Bessel functions
k	=	thermal conductivity of wall
k_3	=	thermal conductivity of cable
ℓ	=	thickness of wall
M_{mn}	=	Eq. (24)
r	=	radial coordinate
r^*	=	Eq. (13)
T	=	wall temperature
T_1'	=	fire temperature
T_1	=	Eq. (1)

NOMENCLATURE (page 2)

T_2 = unexposed room temperature

T_3 = cable temperature

T_m = maximum safe back face temperature

z = axial coordinate

z^* = Eq. (13)

Z_n^* = Eq. (22)

α_n^* = a root of Eq. (21)

β = Eq. (15)

δ = Kronecker delta function

η_1 = Eq. (14)

η_2 = Eq. (15)

θ = Eq. (16)

θ_3 = Eq. (12)

σ = Stefan-Boltzmann constant

ω_1 = Eq. (32)

ω_2 = Eq. (32)

TABLE 1

Nominal Values of the Controlling Parameters

a = 1.8 mm (0.070 inch)

b = 2.7 mm (0.11 inch)

k_3 = 3.5 $Wcm^{-1}K^{-1}$ (2.0×10^2 Btu ft^{-1} hr^{-1} $^oF^{-1}$)

ℓ = 15 cm (6 inch)

k = 0.40 $Jm^{-1}s^{-1}K^{-1}$ (0.23 Btu ft^{-1} hr^{-1} $^oF^{-1}$)

T_1' = 1300K (1880°F)

F_1 = 2.2 Wcm^{-2} (7.0×10^3 Btu ft^{-2} hr^{-1})

H_1' = 40 $Jm^{-2}s^{-1}K^{-1}$ (7.0 Btu ft^{-2} hr^{-1} $^oF^{-1}$)

T_2' = 300K (80°F)

H_2' = 4.0 $Jm^{-2}s^{-1}K^{-1}$ (0.70 Btu ft^{-2} hr^{-1} $^oF^{-1}$)

T_m = 800K (980°F)

FIGURE CAPTIONS

1. Cables penetrating a wall between a fully developed fire and an unexposed room.

2. Parameters controlling the heat flow from a fire through a cable penetration seal to an unexposed room.

3. Heat fluxes from the fire. The linearized flux and its slope are exact at equilibrium.

4. Heat fluxes into the unexposed room. The linearized flux is exact at thermal equilibrium and at the maximum allowed backface temperature, T_m.

5. The cable temperature at the unexposed side of the seal, evaluated for various cable radii (a) and spacings (b) near nominal conditions, $(a/\ell) \times 10^3 = 12$ and $a/b = 0.67$. The dashed line shows the backface temperature the wall would reach in the absence of cables. The comparison with the no cable case shows that the cables heat the wall at the unexposed side for practical values of a and b.

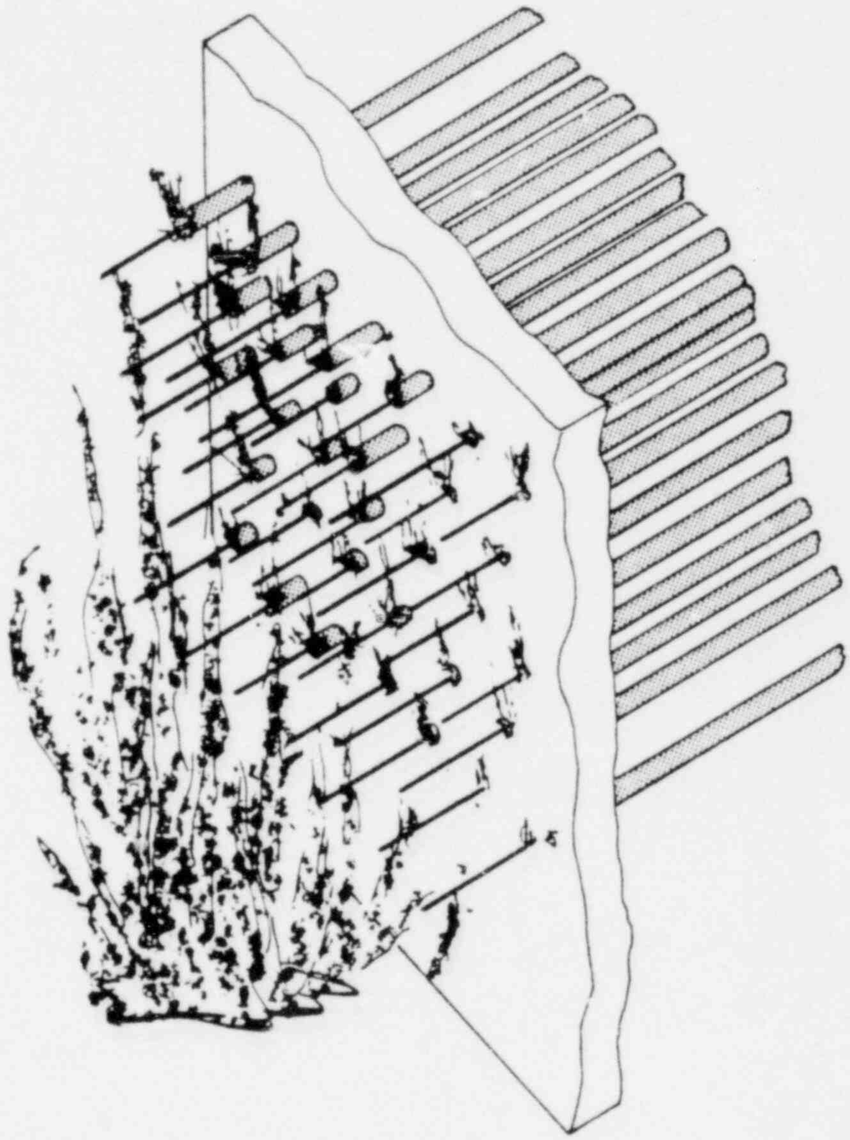
6. The effect of the thermal conductivity (k) of the wall and the cable spacing (b) on the cable temperature on the unexposed side. Nominal conditions are at $(k/k_3) \times 10^4 = 11$ and $b/a = 1.5$. As $k \rightarrow 0$, the cables become thermally isolated and their temperature approaches the $b/a = 1$ value for all b/a. The wall cools the cables below this limit.

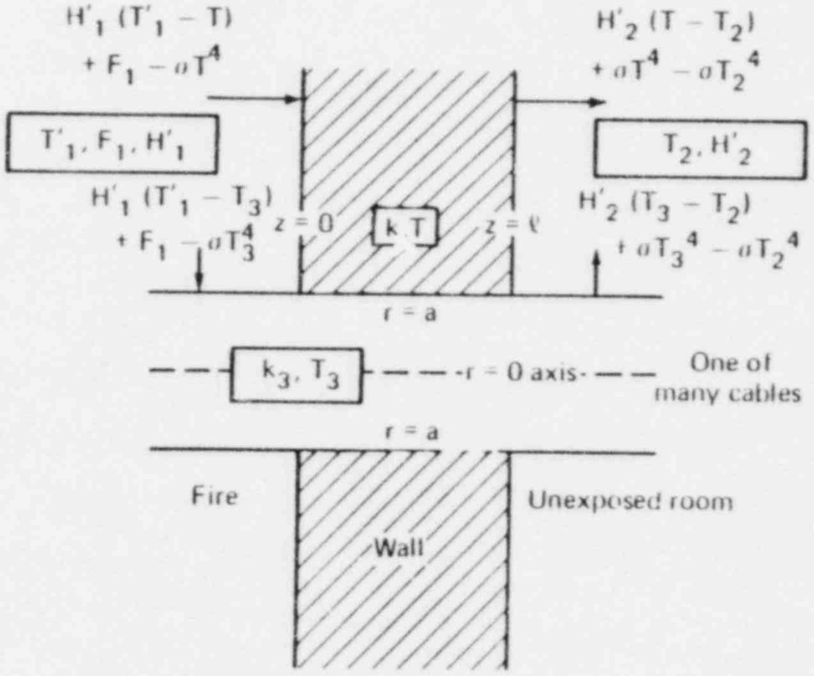
7. The cable temperature, at the unexposed side of the seal, evaluated for various cable spacings (b) and wall conductivities (k) near nominal conditions, $b/a = 1.5$ and $(k/k_3) \times 10^4 = 11$. When b is large enough, the cables are isolated from each other and their temperature becomes independent of b. When b is small enough ($b/a < 4$), the wall rises to cable temperature and ceases to absorb heat from the cables. Then the temperature is again independent of b. The transition occurs around $b/a = 10$.

8. The cable temperature at the unexposed side of the seal, evaluated for various wall thicknesses (ℓ) and cable spacings (b) near nominal conditions, $\ell/a = 83$ and $b/a = 1.5$. The wall thickness is the main defense against the fire.

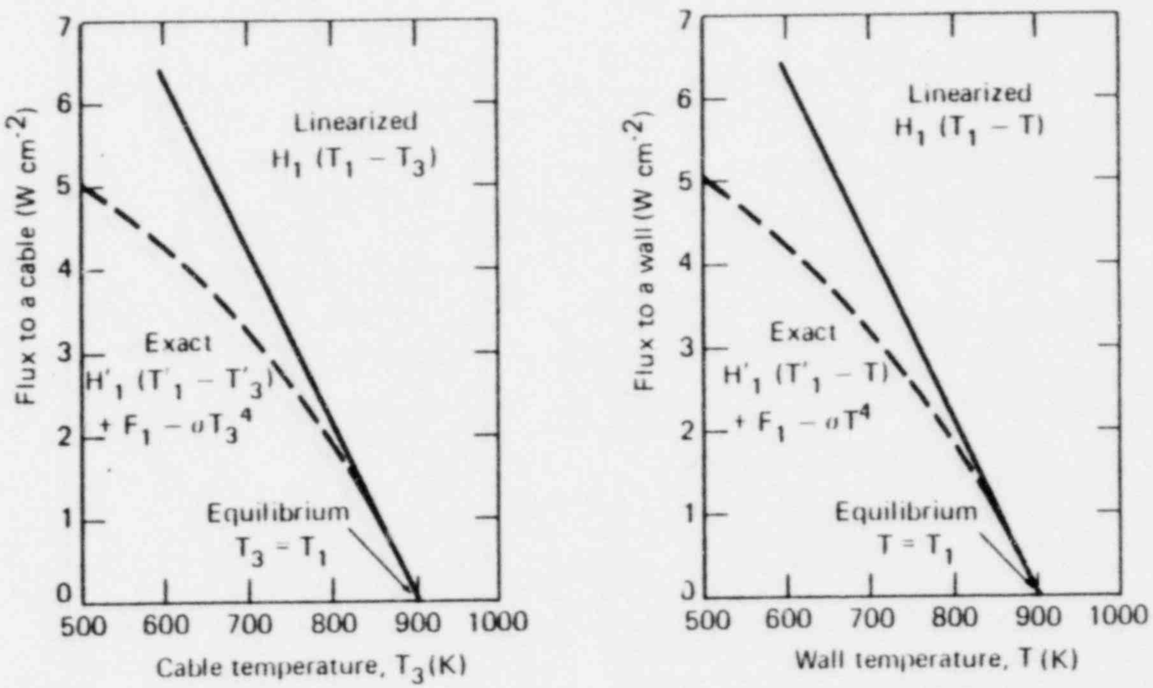
9. The cable temperature at the unexposed side of the seal, evaluated for various radiant fluxes from the fire (F_1) and fire temperatures (T_1'), near nominal conditions, $F_1/\sigma T_m^4 = 0.947$ and $T_1'/T_m = 1.625$.

THE JOHNS HOPKINS UNIVERSITY
APPLIED PHYSICS LABORATORY
LAUREL, MARYLAND

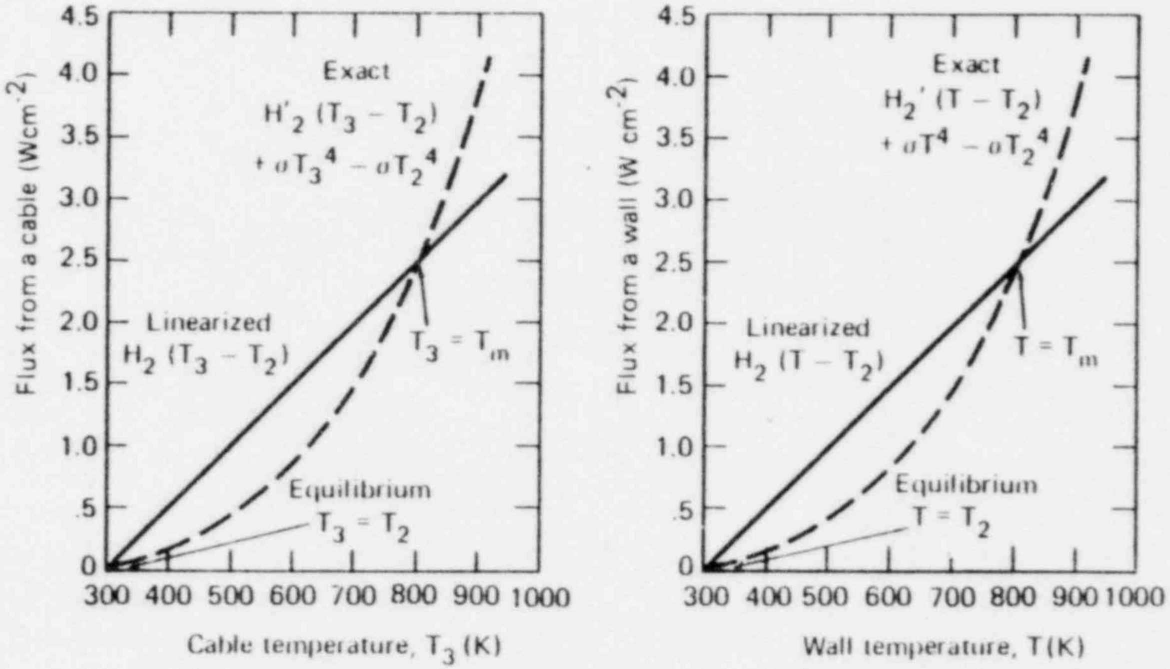




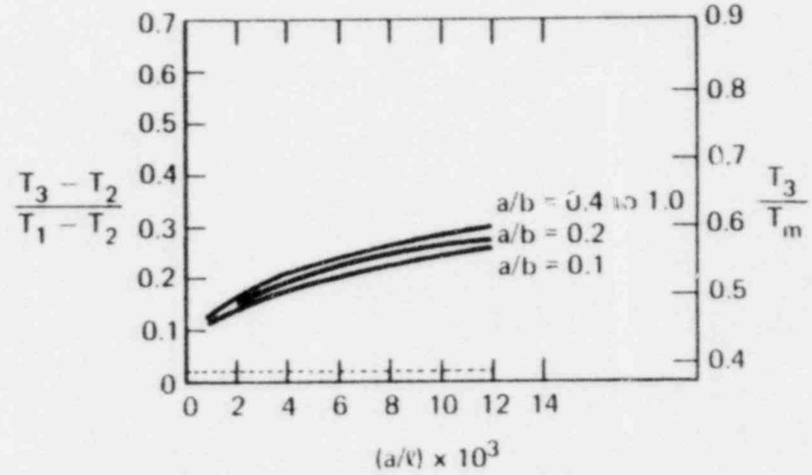
$H'_1 (T'_1 - T) + F_1 - \sigma T^4$
T'_1, F_1, H'_1
$H'_1 (T'_1 - T_3) + F_1 - \sigma T_3^4$
$z = 0$
k, T
$z = \ell$
$H'_2 (T - T_2) + \sigma T^4 - \sigma T_2^4$
T_2, H'_2
$H'_2 (T_3 - T_2) + \sigma T_3^4 - \sigma T_2^4$
$r = a$
k_3, T_3
$r = 0$ axis
One of many cables
$r = a$
Fire
Wall
Unexposed room



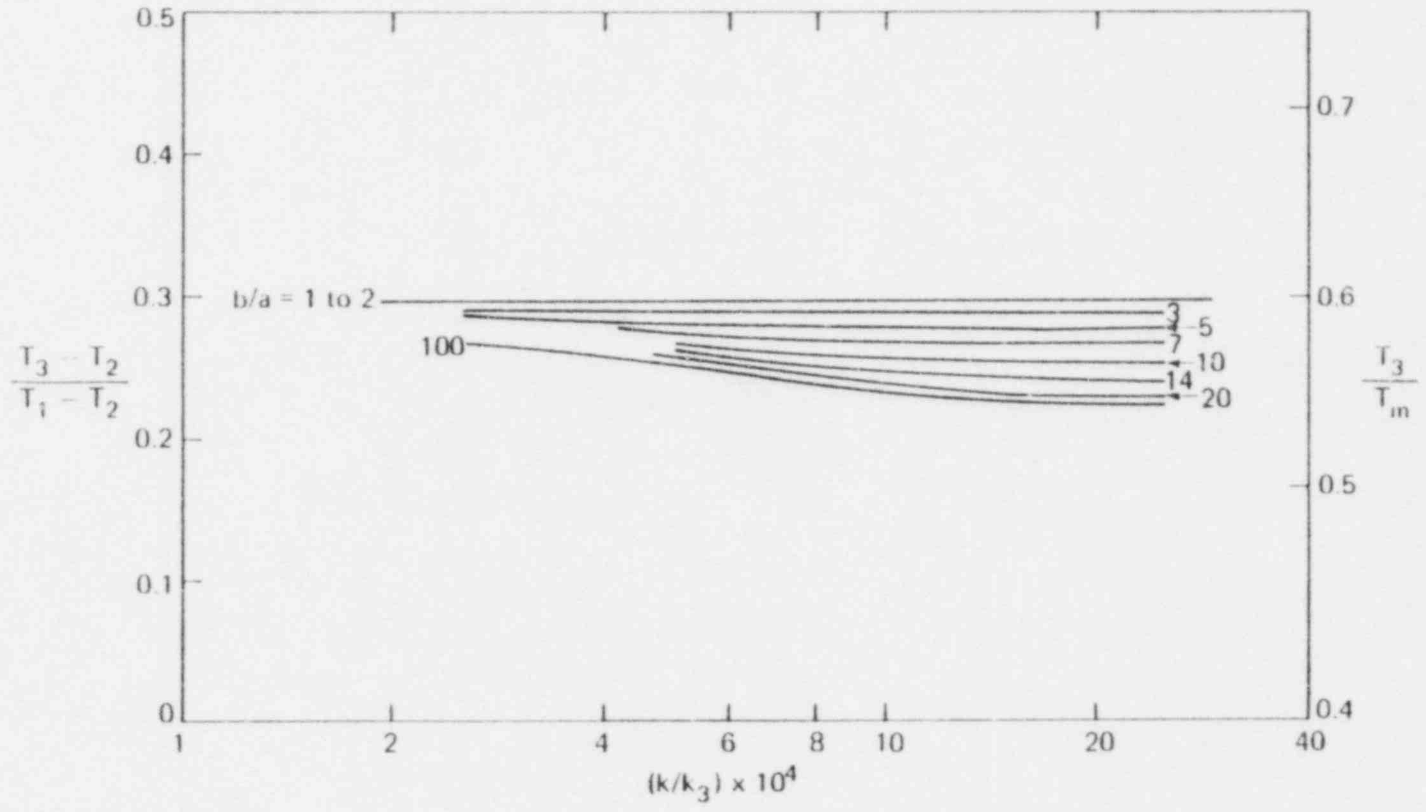
Flux to a cable (W cm⁻²)
Linearized
$H_1 (T_1 - T_3)$
Exact
$H'_1 (T'_1 - T'_3)$
$+ F_1 - \sigma T_3^4$
Equilibrium
$T_3 = T_1$
Cable temperature, T_3 (K)
Flux to a wall (W cm⁻²)
Linearized
$H_1 (T_1 - T)$
Exact
$H'_1 (T'_1 - T)$
$+ F_1 - \sigma T^4$
Equilibrium
$T = T_1$
Wall temperature, T (K)



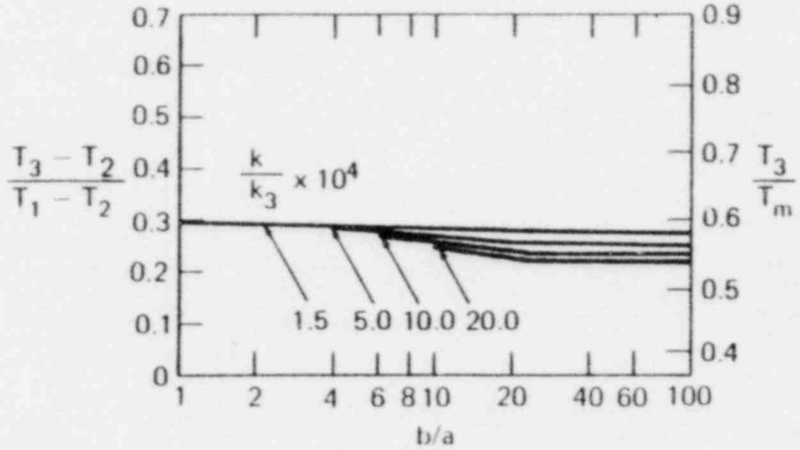
4.5
4.0
3.5
3.0
2.5
2.0
1.5
1.0
.5
0
300 400 500 600 700 800 900 1000
Flux from a cable (Wcm^{-2})
Cable temperature, T_3 (K)
Exact
$H'_2 (T_3 - T_2)$
$+ \sigma T_3^4 - \sigma T_2^4$
Linearized
$H_2 (T_3 - T_2)$
$T_3 = T_m$
Equilibrium
$T_3 = T_2$
Flux from a wall (W cm^{-2})
Wall temperature, T (K)
Exact
$H_2' (T - T_2)$
$+ \sigma T^4 - \sigma T_2^4$
Linearized
$H_2 (T - T_2)$
$T = T_m$
Equilibrium
$T = T_2$

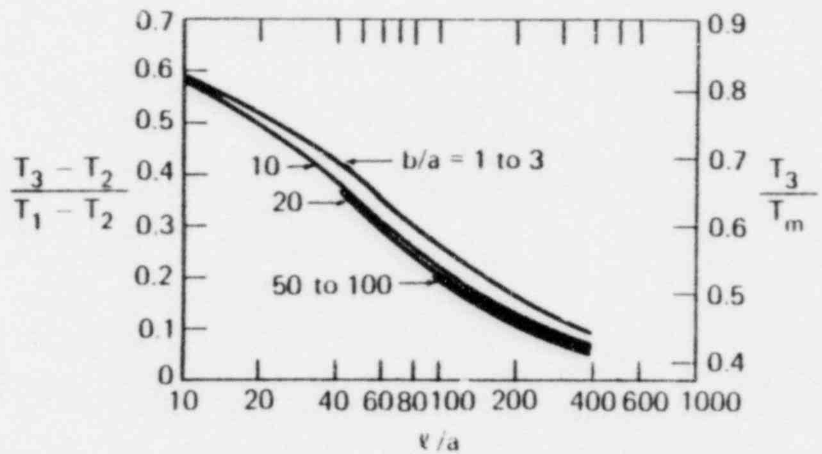


$\frac{T_3 - T_2}{T_1 - T_2}$
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
0 2 4 6 8 10 12 14
$(a/\ell) \times 10^3$
a/b = 0.4 to 1.0
a/b = 0.2
a/b = 0.1
$\frac{T_3}{T_m}$
0.9
0.8
0.7
0.6
0.5
0.4

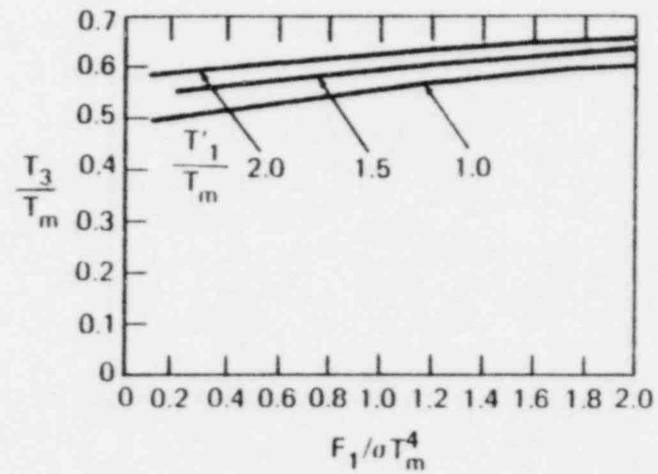


0.5
0.4
0.3
0.2
0.1
0
$\frac{T_3 - T_2}{T_1 - T_2}$
0.7
0.6
0.5
0.4
$\frac{T_3}{T_m}$
b/a = 1 to 2
3
5
7
10
14
20
100
1
2
4
6
8
10
20
40
$(k/k_3) \times 10^4$


$\frac{T_3 - T_2}{T_1 - T_2}$
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
$\frac{k}{k_3} \times 10^4$
1.5
5.0
10.0
20.0
0.9
0.8
0.7
0.6
0.5
0.4
$\frac{T_3}{T_m}$
1
2
4
6
8
10
20
40
60
100
b/a



$\frac{T_3 - T_2}{T_1 - T_2}$
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
10 20 40 60 80 100 200 400 600 1000
ℓ/a
b/a = 1 to 3
10
20
50 to 100
0.9
0.8
0.7
0.6
0.5
0.4
$\frac{T_3}{T_m}$



$\frac{T_3}{T_m}$
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
$\frac{T'_1}{T_m}$
2.0
1.5
1.0
0 0.2 0.4 0.6 0.8 1.0 1.2 1.4 1.6 1.8 2.0
$F_1/\sigma T_m^4$

Summary

Evaluation Of IEEE 383
Cable Flame Test Method

by

Leon J. Przybyla
Underwriters Laboratories Inc.

For Presentation At The

Eight Water Reactor Safety Research

Information Meeting

Gaithersburg, Marlyand

October 27-31, 1980

Evaluation Of IEEE 383
Cable Flame Test

INTRODUCTION:

The evaluation of the IEEE 383 cable flame test method has been completed. This evaluation was conducted as part of the fire research program conducted at Underwriters Laboratories and sponsored by the Nuclear Regulatory Commission.

OBJECTIVE:

The objective was to assess the IEEE 383 cable flame test method and recommend modifications which would better define the method with respect to repeatability and reproducibility.

DISCUSSION:

Five test parameters were considered as being significant to define a cable flame test method. These were the environment, test equipment, sample, flame source, and performance measurement. The IEEE 383 test method was evaluated with respect to these conditions. Separate effects experiments were conducted to investigate the sensitivity of results to changes in these test parameters.

FINDINGS:

Test Equipment - It was found that the sample support should be completely specified as to its size and construction, including the shape, dimensions and spacing of ladder rungs. The use of pressure gauges in combination with flame temperature measurement is not adequate for controlling the fuel and air to the flame source. The use of rotameters with compensation for the gas densities is recommended.

Environment - The environment surrounding the sample needs to be controlled. An enclosure, as shown in Fig. 1, is suitable for that purpose. Ventilation through the enclosure of 1500 ± 300 CFM (708 ± 14.2 l/s); and an initial air temperature of 75 ± 5 F (24 ± 3 C) are recommended.

Flame Source - Propane should be supplied to the burner at a rate corresponding to an energy release rate of 70,000 ± 1600 BTU/Hr (20,517 ± 469 W) with 163 ± 10 SCFH of air. The position of the burner head should be more definitely specified. A position 24 ± 1/8 in. (610 ± 3 mm) above and 3 ± 1/8 in. (76 ± 3 mm) behind the cable tray is recommended.

Sample - The cable sample should be preconditioned to a temperature of 75 ± 5 F (24 ± 3 C). The cable sample should be fastened to the tray with metallic ties every 18 in. (457 mm) along the cable tray.

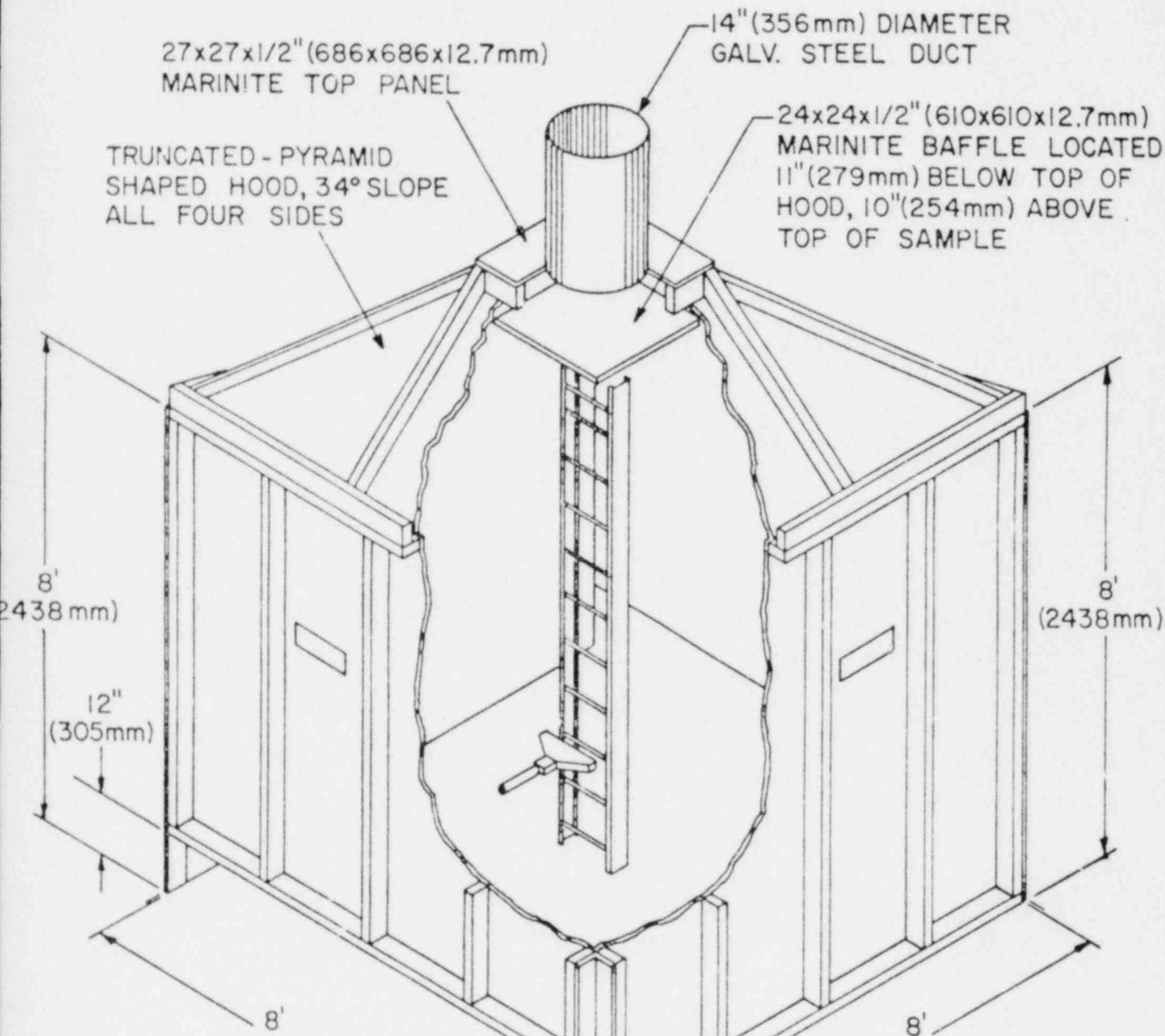
Performance Measurement - A definition of cable damage should be included in the Standard. Although more or less sophisticated determinations of jacket and insulation properties might be conceived for assessing damage, these do not appear to be necessary. A definition of damage as melting, blistering, or charring appears to be sufficient for this test.

FURTHER TESTING:

The analysis and modifications are based upon a limited number of experiments conducted at one facility. It is recommended that round robbin testing be undertaken to establish the reporducibility of this test.

CABLE TEST ENCLOSURE

NOTE: EXPERIMENT NOS. 1-9 CONDUCTED IN ENCLOSURE WITHOUT DUCTED HOOD.



ENCLOSURE AND HOOD CONSTRUCTED OF 1/2"(12.7mm) THICK GYPSUM WALLBOARD ON NOM. 2x4"(51x102mm) LUMBER FRAMEWORK. UPPER 24"(610mm) OF HOOD PROTECTED WITH 1/4"(6.4mm) THICK CERAMIC BOARD. INTERIOR PAINTED FLAT BLACK.

FIG. 1

EVALUATION OF IEEE 383
CABLE FLAME TEST METHOD

OBJECTIVE

- ASSESS THE IEEE 383 CABLE FLAME TEST
- RECOMMEND MODIFICATIONS WHICH WOULD BETTER DEFINE THE TEST METHOD

TEST PARAMETERS

- ENVIRONMENT
- EQUIPMENT
- FLAME SOURCE
- SAMPLE
- PERFORMANCE MEASUREMENT

FINDINGS

EQUIPMENT

- DEFINE CABLE TRAY
- USE ROTAMETERS FOR FLOW CONTROL

ENVIRONMENT

- ENCLOSURE
- VENTILATION ESTABLISHED (1500 $\pm$ 300 CFM)
- INITIAL AIR TEMPERATURE (75 $\pm$ 5 F)

FLAME SOURCE

- FLAME COMPOSITION

  70,000 $\pm$ 1600 BTU/HR

  163 $\pm$ SCFH AIR
- LOCATION

  24 $\pm$ 1/8 IN. HEIGHT

  3 $\pm$ 1/8 IN. BEHIND TRAY

SAMPLE

- PRECONDITIONED 75 $\pm$ 5 F
- SUPPORT WITH METALLIC TIES EVERY 18 IN.

PERFORMANCE MEASUREMENT

- DAMAGE

FURTHER TESTING

ROUND ROBIN TESTING BE UNDERTAKEN TO CONFIRM THE REPRODUCIBILITY OF THIS TEST

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