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

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

INTERIM REPORT

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EG&G Idaho, Inc.

P. O. Box 1625
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August 22, 1978

Mr. R. E. Tiller, Director
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Idaho Operations Office - DOE
Idaho Falls, ID 83401

TRANSMITTAL OF QUICK LOOK REPORT FOR SEMISCALE MOD-3 BLOWDOWN-REFILL TEST S-07-3 (WR-S-78-017) - DJC-107-78

Ref: (a) R. G. Hanson, Quick Look Report for Semiscale Mod-3 Test S-07-1, Baseline Test Series, WR-S-78-013, EG&G Idaho, Inc. (July 1978)
(b) Semiscale Program, Semiscale EOS Appendix 7, WR-S-78-002, EG&G Idaho, Inc. (March 1978)

Dear Mr. Tiller:

Attached is the quick look report for Semiscale Mod-3 Test S-07-3 which was run August 4, 1978. Test S-07-3 was the fourth experiment conducted in the Semiscale Mod-3 baseline test series (Test Series 7). The test was a blowdown-refill experiment conducted from an initial system pressure of 15.6 MPa, a core inlet temperature of 557 K, and a core flow rate of 9.8 kg/s. The total core power was 2 MW and a flat radial power profile was used. As a result of operational difficulties encountered with the on-line core electrical power control system, the core power decay transient was controlled by the predetermined core power decay from Test S-07-1 (Reference (a)).

As indicated in Reference (b), the objectives of Test S-07-3 were to assess the effects of containment pressure on the end of blowdown conditions and to determine the influence of the on-line power controller on the core thermal response characteristics. However, as a result of the operational difficulties with the on-line power controller, the latter objective was not met by Test S-07-3. Instead, an objective not specified in Reference (b) was added for Test S-07-3. This objective was to assess the effects of locating the system pressurizer in the broken loop hot leg rather than in the intact loop hot leg.

The objectives of Test S-07-3 redefined above were met, and the results of a preliminary analysis of the experimental data are presented in the attached quick look report.

The preliminary analysis of Test S-07-3 (as well as the previous blowdown refill tests conducted in the Mod-3 system) indicates that a delay of about 6 to 8 seconds exists between the time ECC liquid first enters the downcomer and the time the liquid finally reaches the bottom of the lower plenum. The delay appears to be caused by heat transfer from the downcomer walls to the ECC liquid which results in a countercurrent steam flow in the downcomer that is sufficiently large to retard ECC penetration. An evaluation of the downcomer thermal response for Test S-07-3 is currently being performed to determine the magnitude of the heat transfer from the downcomer walls in the Mod-3 system, and to determine if modifications to the current Mod-3 downcomer design can further reduce the hot wall delay time.

Very truly yours,

D. J. Olson, Manager
Semiscale Program

nt

Attachment:
As Stated

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WR-S-78-017

QUICK LOOK REPORT FOR SEMISCALE MOD-3 TEST S-07-3 BASELINE TEST SERIES

SEMISCALE PROGRAM

August 1978

NRC Research and Technical Assistance Report

Prepared for the

U. S. Nuclear Regulatory Commission

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

DEPARTMENT OF ENERGY

IDAHO OPERATIONS OFFICE UNDER CONTRACT EY-76-C-07-1570

PRELIMINARY

QUICK LOOK REPORT FOR

SEMISCALE MOD-3 TEST S-07-3

BASELINE TEST SERIES

by

J. M. Cozzuol

SEMISCALE PROGRAM

Approved: D J Olson
D. J. Olson, Manager
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The information contained in this summary report is preliminary and incomplete. Selected pertinent data are presented in order to draw preliminary conclusions and to expedite the reporting of research results.

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Acknowledgement

Credit is due to those members of the Semiscale Program and its support organizations who contributed to this report. Efforts extended in data reduction and processing, production of data plots, and typing of the document are greatly appreciated. The majority of the analysis in this doeument was performed by J. E. Blakely, H. J. Heydt, G. G. Loomis.

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Summary

This report presents the results of a preliminary analysis of the data from Semiscale Mod-3 Test S-07-3. Test S-07-3 was the third blowdown-refill experiment conducted in the Semiscale Mod-3 baseline test series (Test Series 7).

The test was conducted from an initial pressure of about 15.6 MPa, a core inlet temperature of 557 K, and a core temperature rise of about 37 K. The steady-state core power was 2 MW and the core contained 23 powered rods and 2 unpowered rods (one of the rod locations contained a liquid level probe). A flat radial power profile was used for this test.

The objectives of Test S-07-3 were to evaluate the effects of the location of the system pressurizer (intact loop versus broken loop hot leg) and to assess the effects of containment pressure on the end of blowdown conditions. These objectives were met by comparing results from Tests S-07-3 and S-07-1. The containment pressure for the two tests was 136 kPa and 241 kPa, respectively.

Comparison of results from Tests S-07-3 and S-07-1 during the blowdown period indicate that the location of the pressurizer (intact loop versus broken loop hot leg) did not significantly affect the core thermal response. The pressurizer location had essentially no effect on the departure from nucleate boiling (DNB) response and on the peak cladding temperatures reached during the early part of blowdown. Peak heater rod cladding temperatures of about 1100 K were observed for both tests. The core temperatures for both tests turned over at about 10 s and relatively good cooling was observed until 20 to 30 s. Slightly better cooling observed in Test S-07-3 after about 15 s can be attributed to a somewhat higher steam flow out the top of the core.

An evaluation of the system thermal-hydraulic behavior during the latter stages of blowdown for Tests S-07-3 and S-07-1 indicates that the containment pressure had a considerable effect on the system hydraulic response but did not significantly influence the core thermal behavior. Because of the lower containment pressure for Test S-07-3, refill of the downcomer was not initiated until about 58 s (as compared to about 45 s for Test S-07-1), and lower plenum refill was not completed until about 66 s (as compared to about 55 s for Test S-07-1). The differences in downcomer and lower plenum behavior between the two tests, however, did not significantly influence the core thermal response. The core cladding temperatures were somewhat cooler for Test S-07-3 between about 40 and 60 s, but the cooler temperatures are attributed to the influence of the pressurizer during the early part of blowdown rather than to

the different containment pressures. By the time reflooding of the core was initiated, the cladding temperatures throughout the core were essentially the same for both tests.

The pretest calculations underpredicted the peak cladding temperature during blowdown by about 100 K, and the cladding temperatures during the latter part of blowdown were generally overpredicted. The overprediction of cladding temperatures later in blowdown is due to the fact that the RELAP4 calculation does not predict the draining of the upper head fluid into the upper core region. As a result, the calculated cladding temperatures continued to increase after about 10 s whereas the test results indicated a period of cooling between about 10 and 30 s.

The system pressure was also underpredicted because the saturated break flow rates were overpredicted and also because the steam generation rate in the core region was not calculated accurately. Also, penetration of the downcomer and refill of the lower plenum were calculated to occur earlier than observed in Test S-07-3. The differences may be due to the overprediction of the depressurization rate. Several posttest calculations are currently being analyzed to determine if modifications to the RELAP4 model of the Semiscale system may be required to improve the comparisons between the measured and calculated response.

Introduction

This document presents a preliminary analysis of the data obtained from Semiscale blowdown-refill Test S-07-3, the third blowdown test conducted in the Semiscale Mod-3 baseline test series (Test Series 7). This test series is the first series of experiments to be conducted in the Semiscale Mod-3 facility.

The Semiscale Mod-3 system is the current facility operated by the Semiscale Program. The system design differs significantly from the previous Semiscale systems in several respects including the design of the vessel and broken loop regions. The Mod-3 system has a new vessel which contains a 25-rod, full length (3.66-m) electrically heated core simulator, a full length upper head and upper plenum, and an external downcomer. The broken loop differs from previous systems in that an active pump and steam generator have replaced the resistance simulators used in the Mod-1 system. Unlike previous Semiscale systems, the Mod-3 facility was designed with the capability to investigate the influence of upper head emergency core coolant injection on the core thermal-hydraulics. This capability will not be used during the baseline test series but will be utilized in subsequent test series.

The general objectives of Test Series 7 as outlined in Reference (1) are: (a) to investigate the specific performance characteristics of the Mod-3 system during blowdown, reflood, and integral blowdown-reflood transients and (b) to establish the baseline performance of the Mod-3 system during integral blowdown-reflood experiments with cold leg emergency core coolant injection. The influence of specific Mod-3 features such as the 3.66-m core and the active broken loop components on the system response will be evaluated throughout the course of the test series through a comparative analysis of the Mod-3, Mod-1, and FLECHT-SET data.

Test S-07-3 is the third of three blowdown experiments to be conducted in the Series 7 blowdown test group. As indicated in Reference (1), the principal objective of Test S-07-3 was to assess the effects of containment pressure on the end of blowdown conditions. While the previous two blowdown tests (Tests S-07-1[2] and S-07-2[3]) depressurized to a containment pressure of about 241 kPa absolute (corresponding to conditions in Mod-1 tests), Test S-07-3 depressurized to a containment pressure of about 138 kPa absolute which is more representative of conditions in a pressurized water reactor (PWR) plant with upper head injection (UHI). A comparison of the results from Tests S-07-3 and S-07-1 will be used to determine the effects of the lower containment system pressure on the conditions in the core at the end of blowdown.

As a result of operational difficulties encountered with the on-line core electrical power control system, the objective of evaluating the influence of the core power control system for Test S-07-3 could not be met. However, an objective not specified in Reference (1) was added for Test S-07-3. This objective was to assess the effect of locating the system pressurizer in the broken loop hot leg rather than in the intact loop hot leg. The effect of the pressurizer location will be evaluated by a comparison of results from Tests S-07-3 and S-07-1.

Test S-07-3 was conducted from an initial pressure of about 15.6 MPa, a core inlet temperature of 577 K, and a core temperature rise of about 37 K. The total initial core power was about 2 MW. The core power transient was controlled by a predetermined power decay curve rather than by the on-line power control system. Twenty-three (23) of the 25 rods in the core were powered and 2 were unpowered (one rod location contained a liquid level probe). The system pressurizer discharged into the broken loop hot leg. Ambient temperature ECC fluid was injected into the intact loop cold leg using accumulator, low pressure injection (LPIS), and high pressure injection (HPIS) systems.

The RELAP4/MODF[4]* (Update 55) computer code was used to perform pretest and posttest calculations for Test S-07-3. A detailed description of the model nodalization and input selection, and a summary of the pretest calculation results for Test S-07-3 are contained in Reference (5).

This document contains a summary of the results from Test S-07-3. The actual test conditions, test procedure, and test results are described initially. A discussion of the experimental results and comparisons of the test data to Semiscale Mod-3 Test S-07-1 are then presented. In addition, comparisons of the data to the RELAP4/MODF pretest and posttest calculations are presented.

Test Procedure And Test Conditions

This section describes the test procedure and test conditions for Semiscale Mod-3 Test S-07-3.

Test Procedure. Prior to the initiation of testing, the Mod-3 system (shown in an isometric view in Figure 1) was filled with demineralized water and vented to ensure a liquid-full system. Water in the steam generator feedwater tanks was heated to the desired temperature and the required levels were established in the steam generator secondary sides. The intact loop accumulator water level was established and the accumulator was pressurized with nitrogen gas to the desired pressure. The instruments were then calibrated and zeroed as required and the system was leak checked and hydro checked. After the necessary protective trip controls and peripheral hardware controls (pumps, valves, etc.) had been set, the system was brought up to initial conditions and allowed to equilibrate. When the system had equilibrated and the initial conditions were within the specified tolerances, the test was initiated by rupturing discs in the broken loop to break the system pressure boundary. The transient core power control and broken and intact loop pump speed controls were initiated coincident with the rupturing of the system pressure boundary. The test was terminated when the measurements indicated that the lower plenum of the Mod-3 vessel had filled with water.

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* Idaho National Engineering Laboratory Configuration Control Number H003581B.

Test Conditions and Hardware. A plan view of the Semiscale Mod-3 heated core showing the heater rod cladding thermocouple locations recorded during Test S-07-3 is shown in Figure 2. The azimuthal location (referenced to the intact loop cold leg) and elevations above the bottom of the heated core of the thermocouples on each heater rod are shown in the figure. The thermocouples are located approximately 0.095 cm beneath the cladding surface. The axial power profile for the Semiscale Mod-3 3.66-m rods is shown in Figure 3. As illustrated in the figure, the axial power profile has a step cosine shape and has a peaking factor of about 1.55. The locations of the in-core instrumentation (gamma densitometers and core inlet drag screen) relative to the core axial power profile are shown in Figure 4. The general instrumentation locations for the Mod-3 system are shown in Figure 5. Details of the instrumentation specifications can be found in Reference (1).

The specified and actual test conditions are compared in Table I. In general the conditions were judged as satisfactory to meet the test objectives. The transient normalized power applied to the core is shown in Figure 6, and is the same as the one used during Semiscale Mod-3 Test S-07-1.

Test Results

The principal objectives of Test S-07-3, as defined in the introductory section, were to evaluate the effects of the location of pressurizer (intact versus broken loop hot leg) on the system thermal-hydraulic behavior, and to assess the influence of the containment (pressure suppression system) pressure on the end of blowdown conditions. To accomplish these objectives the test results section is divided into three subsections. The first subsection deals with the overall system thermal-hydraulic response during the period of pressurizer discharge (i.e., between 0 and about 50 s after the initiation of blowdown). Included in this subsection are data comparisons between Test S-07-3 and the first Series 7 blowdown test, Test S-07-1*. The intent of this comparison is to assess the effect of the pressurizer location on the overall system and core thermal-hydraulic behavior with particular emphasis being placed on the core thermal response early in the blowdown period. In the second subsection the effects of the containment pressure on the downcomer ECC penetration characteristics as well as on the overall system and core thermal-hydraulic behavior are examined. The influence of containment pressure is evaluated primarily through a comparison of results from Test S-07-3 with data obtained from Test S-07-1. Since Test S-07-3 will be used to establish initial conditions for a subsequent reflood test at 138 kPa absolute containment pressure (Test S-07-5(1)), the differences in the core response at the end of

* The pressurizer was located in the intact loop hot leg for Test S-07-1 rather than in the broken loop hot leg as was the case for Test S-07-3.

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TABLE I

INITIAL CONDITIONS FOR TEST S-07-3

	Specified	Actual
System Pressure	15.513 MPa	15.6 MPa
Hot Leg Fluid Temperature	594 K	594 K
Cold Leg Fluid Temperature	557 K	557 K
Core Temperature Differential	37 K	37 K
Core Power	2 MW	2.04 MW
Core Inlet Flow Rate	9.77 kg/s	9.55 kg/s
Upper Head Fluid Temperature	557 K	558 K
Pressure Suppression System Pressure	138 kPa	136 kPa
ECC Injection Accumulator		
Actuation Pressure	4137 kPa	4100 kPa
Injection Rate	1.21 kg/s	1.2 kg/s
Fluid Temperature	300 K	300 K
LPIS		
Actuation Pressure	1030 kPa	1400 kPa
Injection Rate	0.16 kg/s	0.23 kg/s
HPIS		
Actuation Pressure	12410 kPa	12400 kPa
Injection Rate	0.062 kg/s	0.062 kg/s

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blowdown have been evaluated to determine the effect of the lower containment pressure on the potential reflood behavior of the Mod-3 core. The final subsection compares the experimental results from Test S-07-1 with RELAP4 pretest calculations.

Effect of Pressurizer Location on the System Thermal-Hydraulic Response During Blowdown

The effects of the location of the pressurizer on the blowdown response of the Mod-3 system were evaluated by comparing data from Tests S-07-3 and S-07-1. An examination of the data for the two tests indicates that the pressurizer location had only a minor effect on the system and core thermal-hydraulic response. Results of the evaluation are presented in the following paragraphs.

The pressurizer location did influence the loop hydraulic response (especially early in blowdown). However, the effects of the pressurizer appear to have been limited primarily to the region near the injection location (i.e., in the intact and broken loop hot legs near the vessel). Figures 7 and 8 compare the volumetric flows in the intact and broken loop hot legs near the vessel for Tests S-07-3 and S-07-1. The comparisons indicate that considerable differences in the hydraulic behavior occurred during the period of liquid and high velocity steam flow from the pressurizer (i.e., between 0 and about 15 s). Once the pressurizer steam flow had decreased sufficiently, the loop hydraulic response for the two tests were not significantly different.

Despite the differences on the loop hydraulic behavior, the pressurizer location was found to have essentially no effect on the core thermal response early in blowdown. Table II presents a listing of the initial heater rod cladding temperatures, the peak temperatures during blowdown, and the times of DNB for each recorded thermocouple in Tests S-07-3 and S-07-1. As indicated in the table, the core thermal response for both tests was characterized by early DNB (less than 0.8 s) in the high power regions of the core. Rod thermocouples in the mid and lower core regions (between 72- and 291-cm) showed a rapid and sustained temperature increase following the early DNB, while heater rod thermocouple locations in the core region below 72-cm and above 291-cm either decreased in temperature at rupture or experienced a gradual dryout. Peak rod cladding temperatures occurred between about 8 and 15 s after rupture throughout the core and were essentially the same for both tests (1100 K).

After about 10 s, the core thermal response showed some sensitivity to the pressurizer location. Figure 9 compares a typical rod cladding temperature on the high power step for Tests S-07-3 and S-07-1. As indicated in the figure, Test S-07-3 exhibited somewhat better cooling

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TABLE II
CORE ROD TEMPERATURE RESPONSE FOR TEST S-07-1 AND S-07-3

Measurement (cm)	T initial (K)		T peak (K)		$(T_p - T_{in.})$(K)		t_{DNB} (s)	
	S-07-1	S-07-3	S-07-1	S-07-3	S-07-1	S-07-3	S-07-1	S-07-3
B1-08	558.5	579.9	643.9	657.6	55.4	77.7	*	*
C2-08	578.4	577.2	652.8	641.7	74.4	64.5	*	*
D2-10	581.4	580.1	653.8	657.6	72.4	77.5	*	*
B2-39	599.0	594.0	709.6	717.8	110.6	123.8	*	*
A4-40	593.4	590.5	691.3	700.2	97.9	109.7	*	*
C3-49	593.1	588.7	716.0	738.5	122.9	149.8	*	*
D3-71	619.3	616.5	845.6	847.8	226.3	231.3	*	*
E1-72	628.8	625.7	832.4	834.1	203.6	208.4	0.9	*
E2-109	652.4	650.3	950.8	944.6	298.4	294.3	0.8	0.8
C3-115	670.0	664.5	1006.3	1010.9	336.3	346.4	1.0	1.0
D1-131	670.3	669.6	1022.2	1016.5	351.9	346.9	0.7	0.2
E1-133	675.8	673.5	1017.0	1005.9	341.2	332.4	0.8	0.6
A5-133	670.1	671.3	1012.9	999.2	342.8	327.9	0.8	0.7
D2-134	671.0	671.3	1043.3	1035.7	372.3	364.4	0.7	0.8
E3-134	676.8	675.5	1032.8	1026.1	356.0	350.6	0.7	0.7
A3-135	663.5	663.3	1022.1	1024.6	358.6	361.3	0.7	0.7
D5-137	672.6	671.3	1003.6	1004.0	331.0	332.7	0.8	0.7
D3-153	667.8	669.5	1055.7	1049.6	387.9	380.1	0.7	0.8
D1-163	668.1	668.4	1013.5	1024.6	345.4	356.2	0.8	0.8
B4-164	678.2	665.4	1051.5	1045.1	373.3	379.7	0.7	0.7
A5-179	698.[illegible]	702.7	1100.2	1100.8	402.0	398.1	0.6	0.5
B3-179	681.3	681.9	1067.7	1062.2	386.4	380.3	0.7	0.6
D2-179	674.7	676.8	1069.7	1062.2	395.0	385.4	0.7	0.7
D5-179	684.6	688.5	1023.1	1026.0	338.5	337.5	0.8	0.7
E1-180	675.8	677.4	1029.8	1035.0	354.0	357.6	0.6	0.5
C2-180	684.8	684.8	1066.1	1062.2	381.3	377.4	0.7	0.5
A4-181	685.0	688.7	1048.2	1038.1	369.2	349.4	0.7	0.8
C3-184	677.2	679.1	1067.5	1061.9	390.3	382.8	0.8	0.8
E2-190	672.7	677.9	1033.9	1025.7	361.2	347.8	0.7	0.4
E4-190	672.0	673.7	1025.2	1017.0	353.2	343.3	0.8	0.7
A4-191	673.4	676.3	1049.2	1056.3	375.8	380.0	0.5	0.6
D3-206	672.5	673.6	1053.4	1043.9	380.9	370.3	0.8	0.6
B2-225	678.3	682.4	1029.3	1018.8	351.0	336.4	0.6	0.9
B3-226	674.6	676.8	1049.3	1039.1	374.7	362.3	0.7	0.7
D3-226	671.3	672.7	1045.9	1026.3	374.6	353.6	0.6	0.7
A3-229	675.1	680.1	1048.8	1050.4	373.7	370.3	0.7	0.7
C3-230	671.9	676.1	1035.8	1026.4	363.9	350.3	0.8	0.8
D1-251	668.5	670.3	948.7	947.8	280.2	277.5	0.8	0.7
D2-254	667.8	671.3	965.4	957.2	297.6	285.9	0.[illegible]	0.8
D5-254	666.4	669.8	950.6	950.3	284.2	280.5	1.1	0.7
C2-277	667.7	671.0	967.0	955.5	299.3	284.5	0.9	0.7
D3-289	648.0	653.1	802.2	826.4	160.3	173.3	0.8	0.7
E3-290	654.9	656.5	830.2	829.5	175.3	173.0	*	*
A3-291	652.5	652.5	774.2	854.5	121.7	202.0	*	*
D1-320	633.3	632.0	639.4	634.0	6.1	2.0	*	*
A5-321	630.0	631.2	671.9	683.7	41.9	52.5	*	*
C2-321	632.4	636.8	636.8	637.6	4.4	0.8	*	*
E1-321	600.0	628.2	600.9	629.4	7.4	1.2	*	*
E4-354	620.5	620.1	621.1	620.1	0.6	0.0	*	*

* indicates thermocouple did not experience DNB

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after about 10 s than did Test S-07-1. The improved cooling was observed throughout most of the core region and can be attributed to a higher volumetric flow out the top of the core for Test S-07-3. A comparison of the volumetric flow near the top of the core region for the two tests (Figure 10) illustrates the somewhat higher flow rate for Test S-07-3.

Influence of Containment Pressure on System Thermal-Hydraulic Response

As indicated previously, Test S-07-3 was conducted with a containment system pressure of about 136 kPa. To evaluate the effects of the containment system pressure on the resulting system and core thermal-hydraulic response, data from Test S-07-3 have been compared with data from Test S-07-1 for which the containment system pressure was about 241 kPa. Results of the data comparison indicate that the containment system pressure had a considerable influence on the system hydraulic behavior during the latter portion of blowdown. In particular, the downcomer hydraulic behavior and ECC penetration characteristics of the downcomer were considerably different for the two tests. However, the differences in the overall system and downcomer hydraulic behavior had only a minor influence on the core thermal response. The differences in system and downcomer hydraulic behavior as well as in the core thermal behavior for the two tests are discussed in the following paragraphs.

The effect of different containment pressures for Tests S-07-3 and S-07-1 did not become evident until relatively late in the blowdown period. Figure 11 compares the system depressurization rate for the two tests. As expected, the system depressurization rates were nearly identical during the early part of the blowdown transient when flow at the break locations was choked. However, during the latter part of the blowdown transient (i.e., after about 45 s) the system pressures for the two tests diverged considerably as the pressure in Test S-07-3 continued to decrease to the lower containment system pressure. Figures 12 and 13 compare the system pressure (obtained in the intact loop cold leg) with the containment pressure for Test S-07-1 and S-07-3, respectively. As indicated in these figures, the system pressure approaches the containment pressure at about 42 s after rupture for Test S-07-1, but not until about 54 s for Test S-07-3.

In the Mod-3 system, the one-dimensional nature of flow in the downcomer generally delays ECC penetration of the downcomer until the system pressure approaches the containment pressure. Prior to the time the containment pressure is reached, the countercurrent steam flow in the downcomer (driven by the differential pressure between the core region and the containment) is of sufficient magnitude to prevent

ECC from entering the downcomer region. A reduced containment pressure, therefore, tends to increase the duration of the strong downcomer countercurrent steam flow and thus lengthens the time required for ECC to penetrate the downcomer. Figure 14 compares the downcomer volumetric flow rates for Tests S-07-1 and S-07-3 and illustrates the extended period of countercurrent flow in Test S-07-3. Included in the figure are calculations (represented by the dashed lines) of the minimum negative flow required to prevent ECC penetration of the downcomer. The calculations are based on a flooding correlation using the extreme limits of data from Semiscale Mod-1 ECC performance tests (5) to bracket the potential downcomer countercurrent steam flow required to prevent ECC penetration. The comparison of the measured downcomer flows and the calculated minimum downcomer flowrate required to prevent ECC penetration indicates that the magnitude of the measured flow was sufficient to prevent ECC penetration until about 45 s for Test S-07-1 and until about 55 s for Test S-07-3. A comparison of the fluid densities in the upper portion of the downcomer for Tests S-07-1 and S-07-3 (Figure 15) indicates that ECC fluid actually began to enter the downcomer at about 45 and 58 s, respectively, which is in excellent agreement with the times corresponding to the general decrease of the measured velocity below the calculated velocity.

Once ECC penetration of the downcomer begins in the Mod-3 system, heat transfer to the ECC liquid from the hot downcomer walls causes substantial steam generation. The resulting steam flow up the downcomer tends to delay the further penetration of ECC fluid through the downcomer. This phenomenon is termed the "hot wall delay" and is typical of small scale systems such as Semiscale. An evaluation of the downcomer response for Tests S-07-3 and S-07-1 indicates that the different system pressure response for the two tests did not significantly effect the duration of the hot wall delay. A comparison of the fluid density in the lower portion of the downcomer for Tests S-07-2 and S-07-1 is shown in Figure 16. As indicated in the figure, ECC fluid penetrated to the lower portion of the downcomer by about 64 s for Test S-07-3 and by about 53 s for Test S-07-1 which correspond to hot wall delay times of about 6 and 8 s, respectively.

Prior to the time ECC penetration of the downcomer begins in the Mod-3 system, much of the ECC fluid is bypassed to the vessel inlet side of the break. Because of the extended downcomer countercurrent flow period for Test S-07-3, the bypass period was correspondingly longer than for Test S-07-1. The extent of the ECC bypass for each test can be determined by examining the fluid density measurements on the vessel inlet side of the break. The broken loop vessel inlet side density measurement in Figure 17 shows a period of high density liquid until about 62 s after rupture for Test S-07-3 indicating significant bypass until that time. In contrast, Figure 18 shows high density

liquid only until about 53 s after rupture for Test S-07-1. The considerably longer period of ECC bypass for Test S-07-3 indicates that substantially less of the ECC accumulator liquid was available for refill of the downcomer and lower plenum than for Test S-07-1.

The delay in the initiation of downcomer refill for Test S-07-3 resulting from the lower containment pressure (when compared to Test S-07-1) caused a corresponding delay in the refill of the downcomer and lower plenum. The fluid densities near the bottom of the heated section of the core for the two tests (Figure 19) indicates that refill of the lower plenum volume had occurred by about 55 s for Test S-07-1 but not until about 66 s for Test S-07-3. However, although a considerable delay in the initiation of refill was observed for Test S-07-3, the actual refill of the lower plenum required about the same length of time for both tests (i.e., approximately 8 s for Test S-07-3 and 10 s for Test S-07-1) once penetration of the downcomer was initiated. It thus appears that the smaller amount of ECC accumulator liquid that was available for refill in Test S-07-3 did not significantly affect the vessel refill behavior.

Although the differences in containment pressure for Tests S-07-3 and S-07-1 had a considerable influence on the system hydraulic response during the latter part of the test, only minor differences in the core thermal behavior were observed during this period. Figures 20 through 24 compare typical rod cladding temperatures at various elevations in the core for the two tests. For Test S-07-3 the heater rod cladding temperatures in the central portion of the core (i.e., between the 120 and 240 cm elevation) were somewhat cooler between 20 and 60 s than for Test S-07-1. However, the better cooling (at least between 20 and 40 s) is probably an effect of the pressurizer location rather than an effect of the containment pressure. By the time refill of the lower plenum had occurred, however, the cladding temperatures throughout most of the core were essentially the same in both tests. It is thus evident, that during the later portion of blowdown and throughout refill the containment pressure did not significantly affect the core thermal response.

Comparison of Mod-3 Data and RELAP4 Calculated Blowdown Response

Pretest and posttest calculations for Mod-3 Test S-07-3 were made using the RELAP4/MODF Update 55* computer code. The RELAP4 code was used to

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* For the purpose of code configuration control, RELAP4/MODF Update 55 is referenced as program number H003581B.

calculate the thermal-hydraulic response of the Mod-3 system during the blowdown and refill periods. This section presents comparisons between the calculated results and data for Test S-07-3.

Pretest Calculations. The RELAP4 pretest calculation for Test S-07-3 tended to underpredict the peak cladding temperatures during the early blowdown period and overpredict the cladding temperatures later in blowdown. Figures 25, 26 and 27 compare the range of measured cladding temperatures near the 40-cm, 180-cm, and 280-cm elevations with the corresponding predicted temperatures. The results in these figures indicate that the calculations tended to underpredict the peak cladding temperature during the period between 5 and about 10 s. However, except for the 40-cm results, the predicted temperatures did fall within the range of the data. After about 10 s the data indicate a general decrease in rod cladding temperatures which is a result draining of the upper head fluid downward through the core (Reference (2)). Since the RELAP4 calculation did not properly predict the flow of fluid from the upper head region, the calculated cladding temperatures exhibited a gradual increase for the remainder of the blowdown period. As a result, the predicted cladding temperatures in the mid and upper portions of the core were considerably higher than the measured temperatures during the later portion of blowdown.

The system pressure was considerably underpredicted by the pretest calculation during most of the blowdown period as indicated in Figure 28 which compares the measured system pressure with that calculated by RELAP4. The underprediction of the system pressure is caused in part by an overprediction of the break flows. Figure 29 and 30 compare the mass flow rates near the break in the broken loop hot and cold legs with the corresponding predicted flow rates and illustrate the difference between the predicted and measured break flow rates. Another factor which could affect the prediction of the depressurization rate is the calculation of the steam generation rate in the core. In the test, draining of the upper head liquid into the upper portion of the core resulted in considerable steam generation. The RELAP4 calculation, however, does not predict the flow of upper head fluid into the core region. The difference in the upper head liquid penetration of the core between the RELAP4 prediction and the test is illustrated in Figure 31 which compares the measured in-core density with that calculated by the RELAP4 code. As indicated in the figure, a considerable increase in the measured in-core density (that was not predicted by the RELAP4 pretest calculation) occurred between about 10 and 25 s. The underprediction of the system pressure during this period of time is, therefore, consistent with an underprediction of the steam generation rate in the core region.

The calculated and measured lower plenum densities are compared in Figure 32. As indicated in the figure, lower plenum refill was calculated to start about 35 s after rupture, whereas the lower plenum density measurement indicates that refill did not begin until about 63 s. The predicted start time for lower plenum refill was substantially earlier than the measured time because the RELAP4 pretest calculation overpredicts the system depressurization rate. As a result, the calculated system pressure reached the containment pressure faster than during the actual test thus eliminating most of the downcomer countercurrent steam flow earlier in time. ECC penetration of the downcomer therefore, began sooner in the calculation than was observed in the test data.

Posttest Calculations. To evaluate the effect of the differences in break flows between the pretest calculation and the test results, a posttest RELAP4 calculation was performed using the measured vessel inlet side break flow rate to drive the calculation. Use of the measured break flow rate in RELAP4 resulted in an improvement in the calculated system depressurization rate until about 10 s after rupture, as indicated in Figure 33 which compares the measured system pressure with the pressure calculated by RELAP4. However, after 10 s the system pressure was considerably overpredicted. Thus, although overprediction of the break flow rates may have contributed in part to the underprediction of the system depressurization rate in the pretest calculation, other factors also appear to be influencing the resulting calculated system response. Several additional posttest calculations are currently being analyzed to identify these factors and to determine if modifications to the RELAP4 model of the Semiscale system may be necessary.

Conclusions

Results of the data analysis indicate that the location of the pressurizer (intact loop versus broken loop hot leg) did not significantly affect the system and core thermal-hydraulic response. The pressurizer location had essentially no effect on the DNB response and on the peak cladding temperatures reached during the early part of blowdown. Peak heater rod cladding temperatures of about 1100 K were observed for both tests. The core temperatures for both tests turned over at about 10 s and cooling (which coincided with the flashing and draining of upper head fluid) was observed until 20 to 30 s. Slightly better cooling observed in Test S-07-3 after about 15 s can apparently be attributed to a somewhat higher steam flow out the top of the core.

Comparison of the system thermal-hydraulic behavior for Tests S-07-3 and S-07-1 show that the containment pressure had a considerable effect on the system hydraulic response but did not significantly

influence the core thermal behavior. Because of the lower containment pressure for Test S-07-3, refill of the downcomer was not initiated until about 58 s as compared to about 45 s for Test S-07-1. A corresponding delay in refill of the lower plenum to the bottom of the heated section of the core was also observed for Test S-07-3. Refill was not completed until about 66 s for Test S-07-3 as compared to about 55 s for Test S-07-1. The differences in downcomer hydraulic behavior between the two tests did not significantly influence the core thermal response. The core cladding temperatures were somewhat cooler for Test S-07-3 between about 40 and 60 s, but the cooler temperatures can be attributed to the influence of the pressurizer location during the early part of blowdown rather than to the different containment pressures. By the time reflooding of the core was initiated, the cladding temperatures throughout the core were essentially the same for both tests.

A comparison of Test S-07-3 results to the pretest calculation indicate that the peak cladding temperature during blowdown was underpredicted by about 100 K, but that cladding temperatures during the latter part of blowdown were generally overpredicted. The overprediction of cladding temperatures later in blowdown is due to the fact that the RELAP4 calculation does not predict th draining of the upper head fluid into the upper core region. As a result the predicted cladding temperatures continued to increase after about 10 s whereas the test results indicated a period of cooling between about 10 and 30 s.

The pretest calculation overpredicted the system depressurization rate. The system pressure was not predicted accurately possibly because the saturated break flow rates were overpredicted and also because the steam generation rate in the core region was underpredicted. Also, penetration of the downcomer and refill of the lower plenum were calculated to occur earlier than observed in Test S-07-3. The differences may be due to the overprediction of the depressurization rate.

References

1. Semiscale Program, Semiscale EOS Appendix 7, WR-S-78-002, EG&G Idaho, Inc. (March 1978).

2. R. G. Hanson, Quick Look Report for Semiscale Mod-3 Test S-07-1 Baseline Test Series, WR-S-78-013, EG&G Idaho, Inc. (July 1978).

3. E. A. Harvego and R. G. Hanson, Quick Loop Report for Semiscale Mod-3 Test S-07-2 Baseline Test Series, WR-S-78-014, EG&G Idaho, Inc. (August 1978).

4. S. R. Fischer et al., RELAP4/MOD6, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, User's Manual, EG&G Idaho, Inc., CDAP TR003 (January 1978).

5. D. J. Hanson et al., ECC Performance in the Semiscale Geometry, ANCR-1161, Aerojet Nuclear Company, (June 1974).

PRELIMINARY

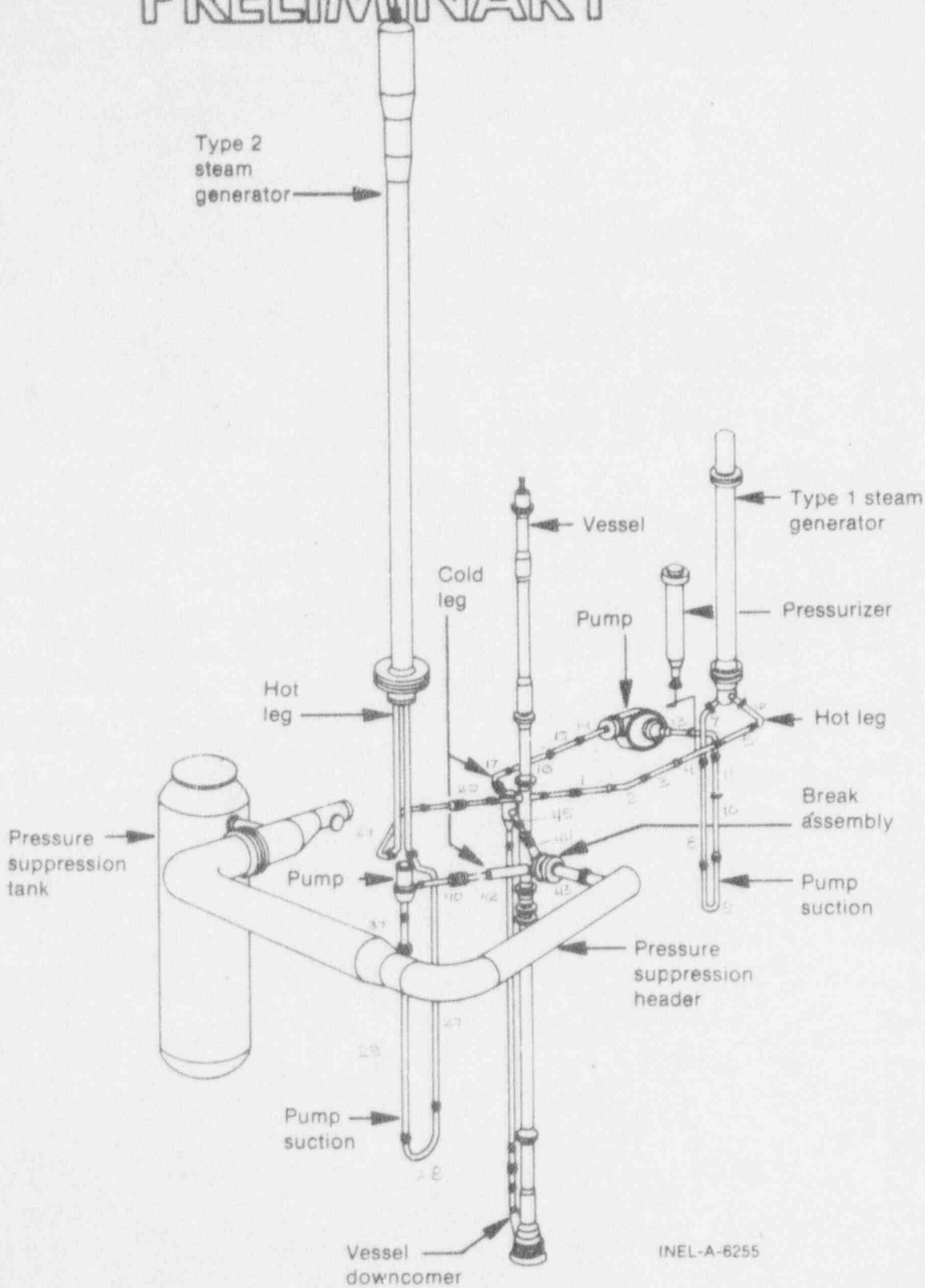

Figure 1. Semiscale Mod-3 System Cold Leg Noncommunicative Break Configuration - Isometric

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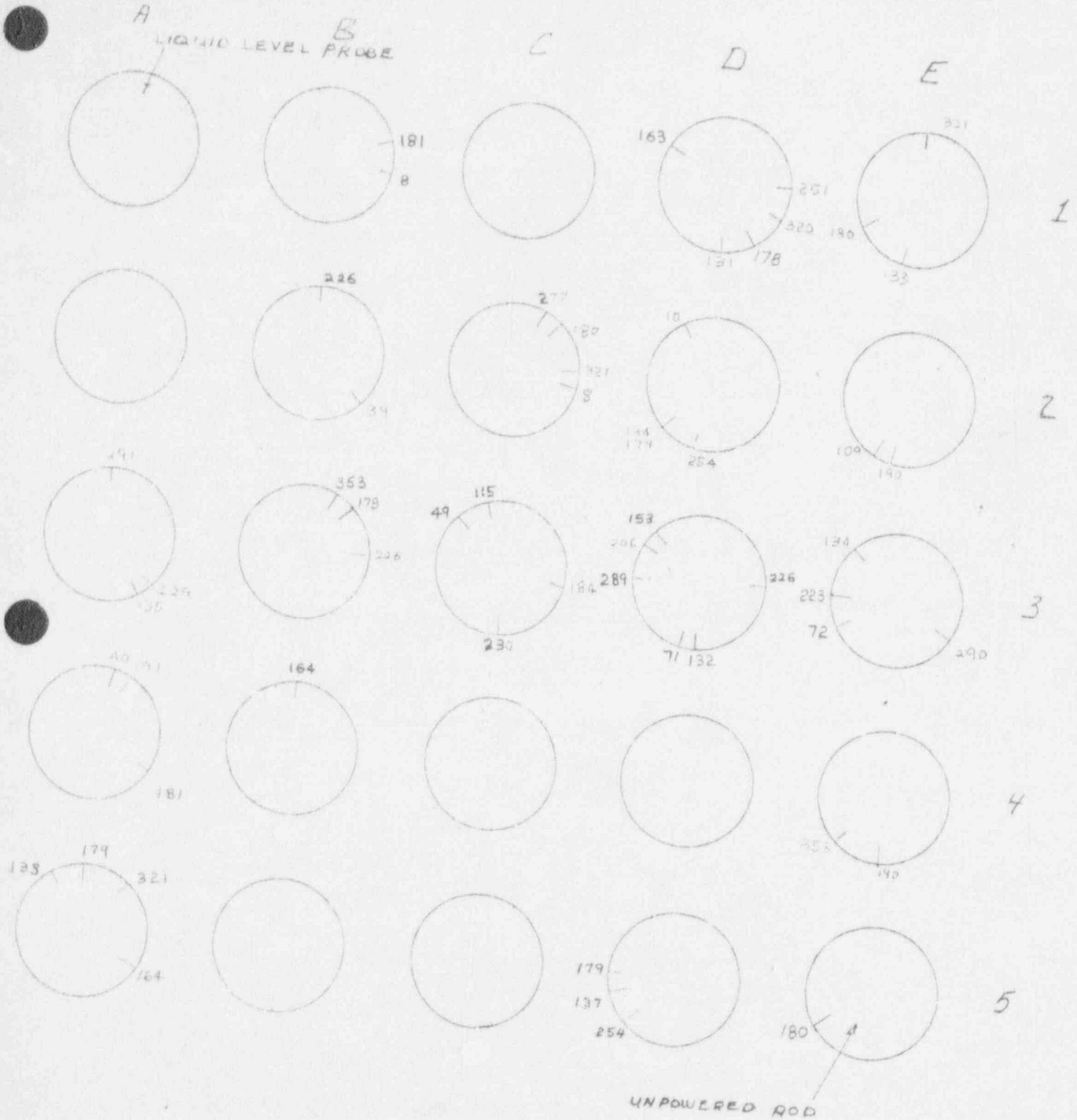

Fig. 2 Location of core rod thermocouples for Test S-07-3.

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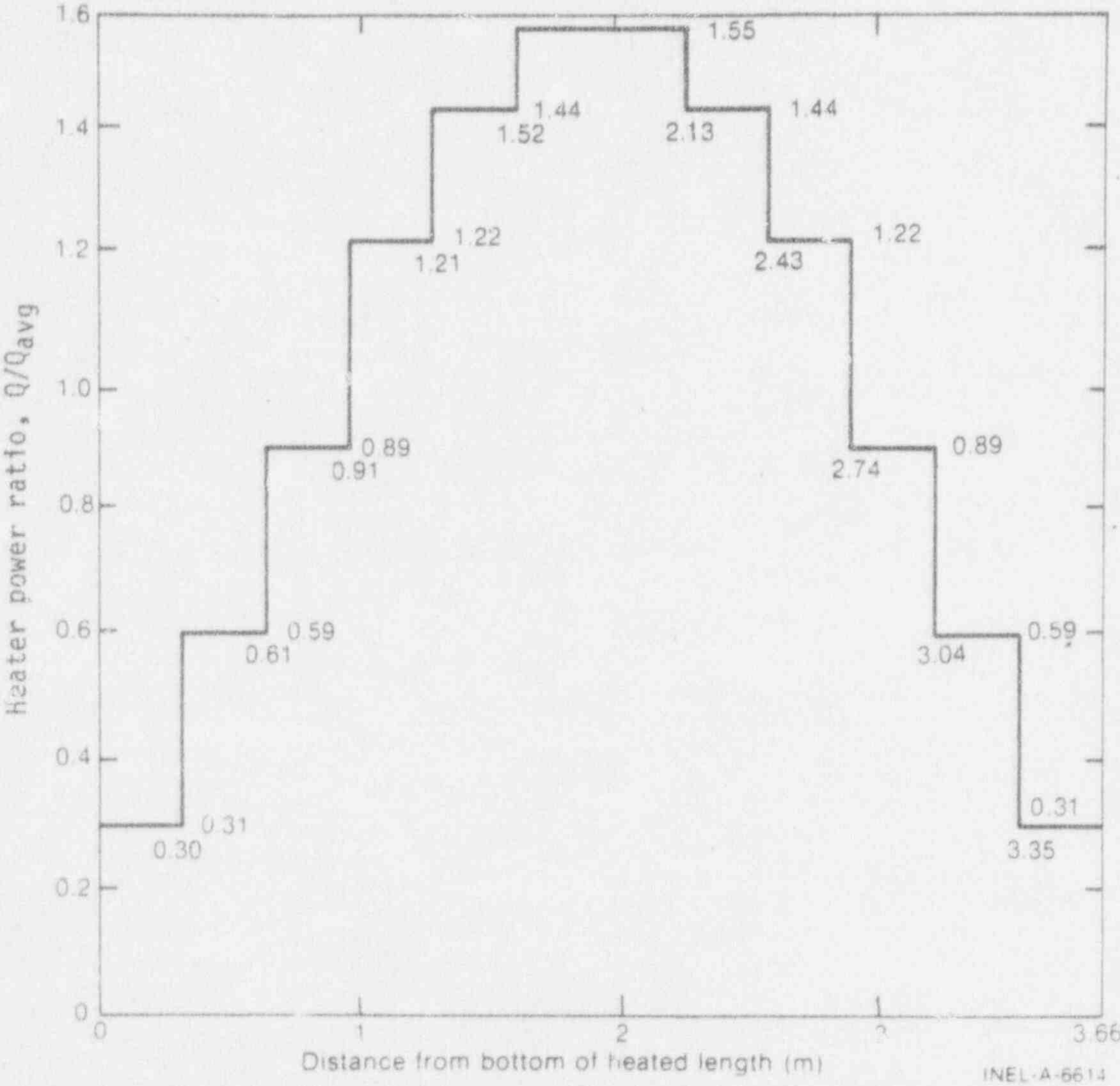

Fig. 3 Semiscale Mod-3 heater rod axial power distribution.

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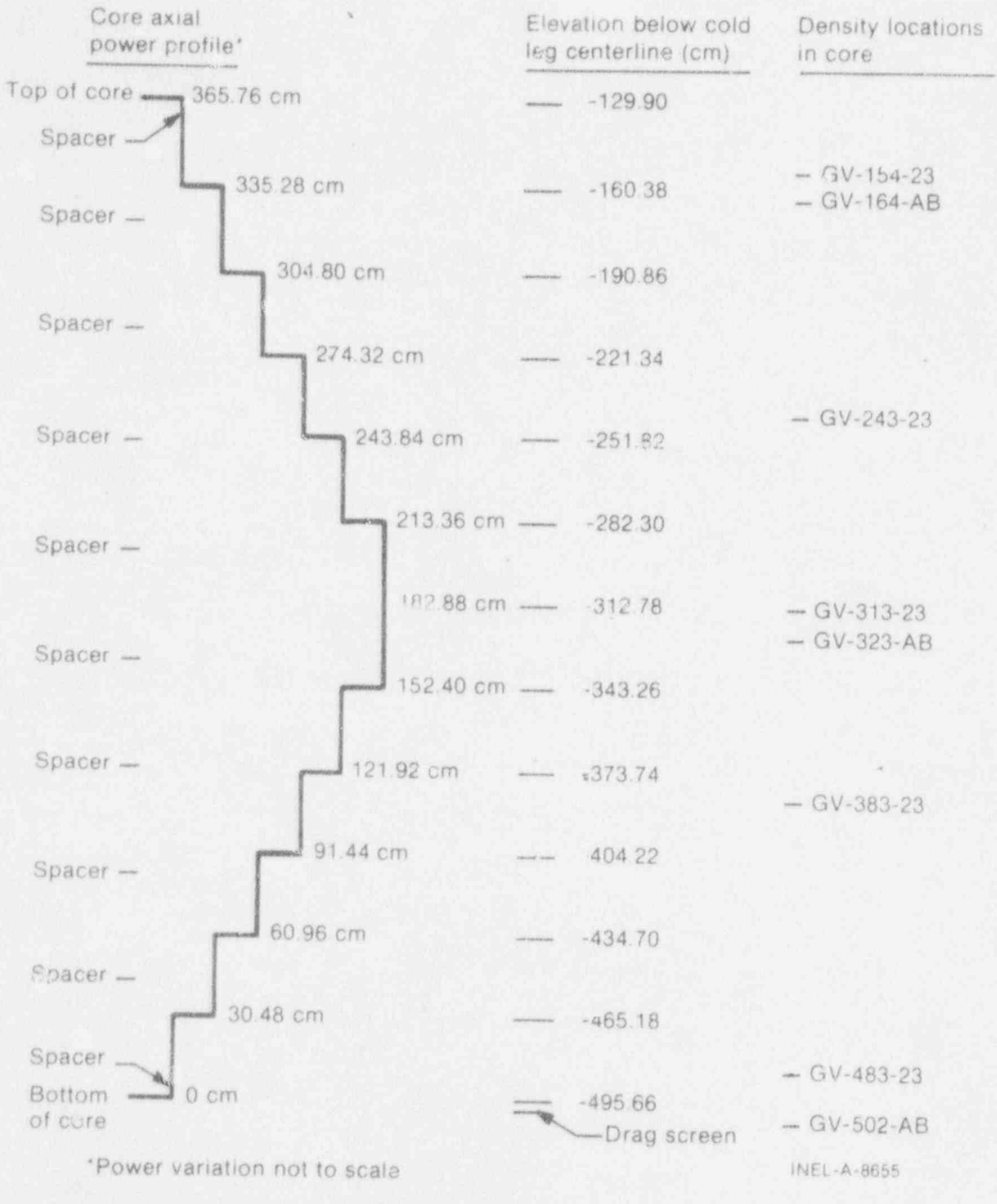

Fig. 4 Axial power profile in relation to vessel instrumentation.

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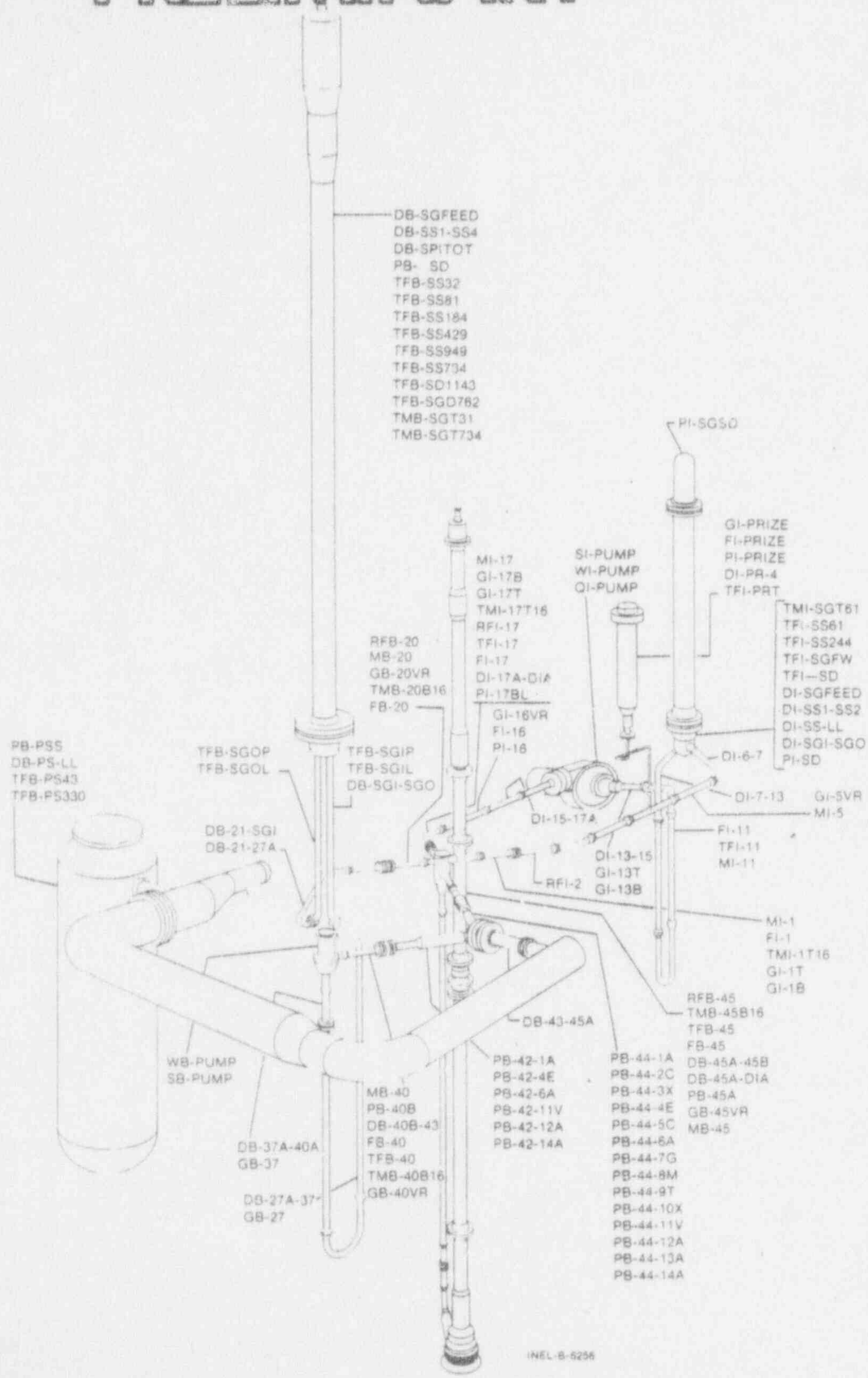

Figure 5. Semiscale Mod-3 System Cold Leg Noncommunicative Break Configuration - Isometric With Available Instrumentation Locations

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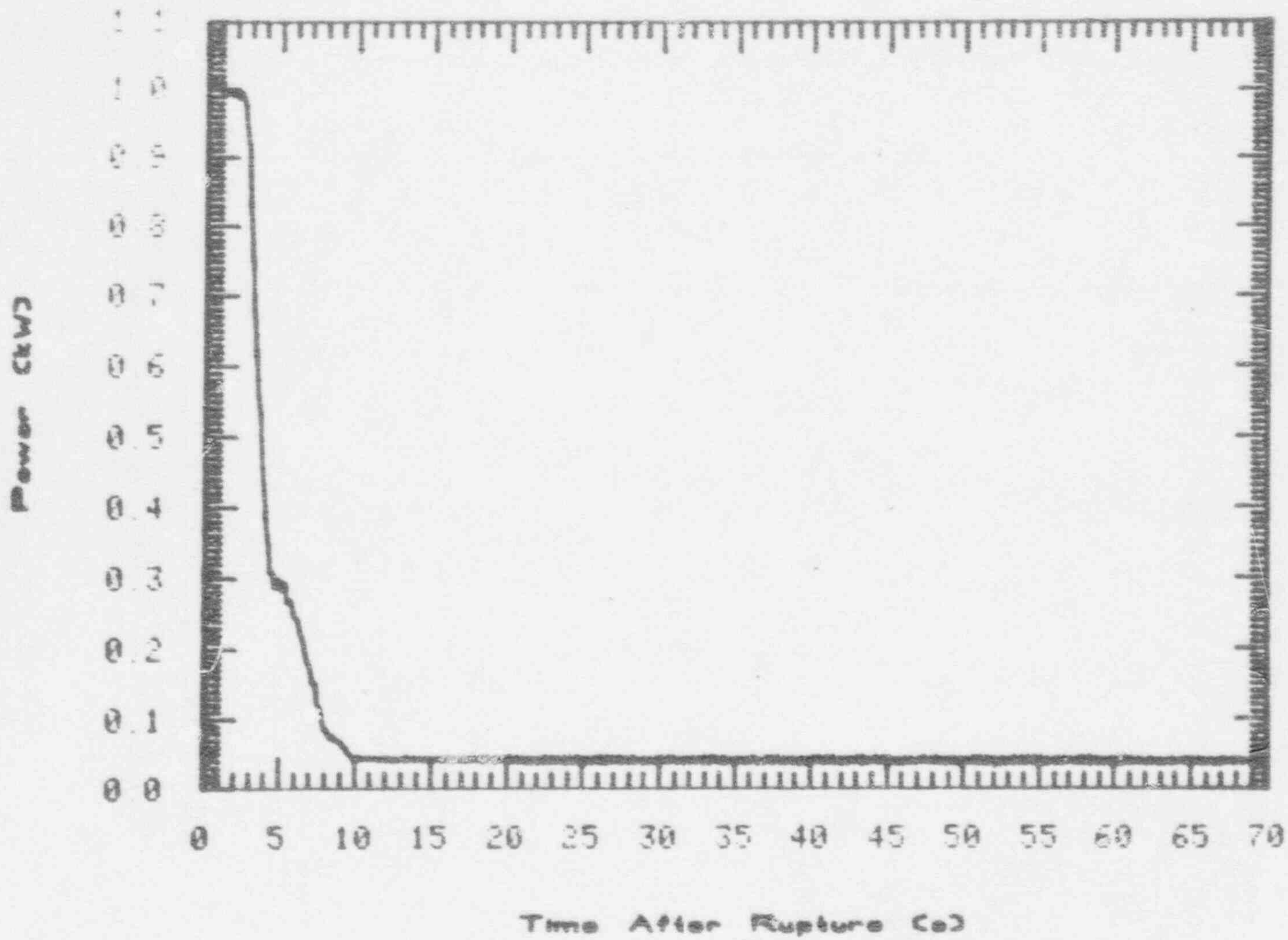

Fig. 6 Normalized core power for Test S-07-3.

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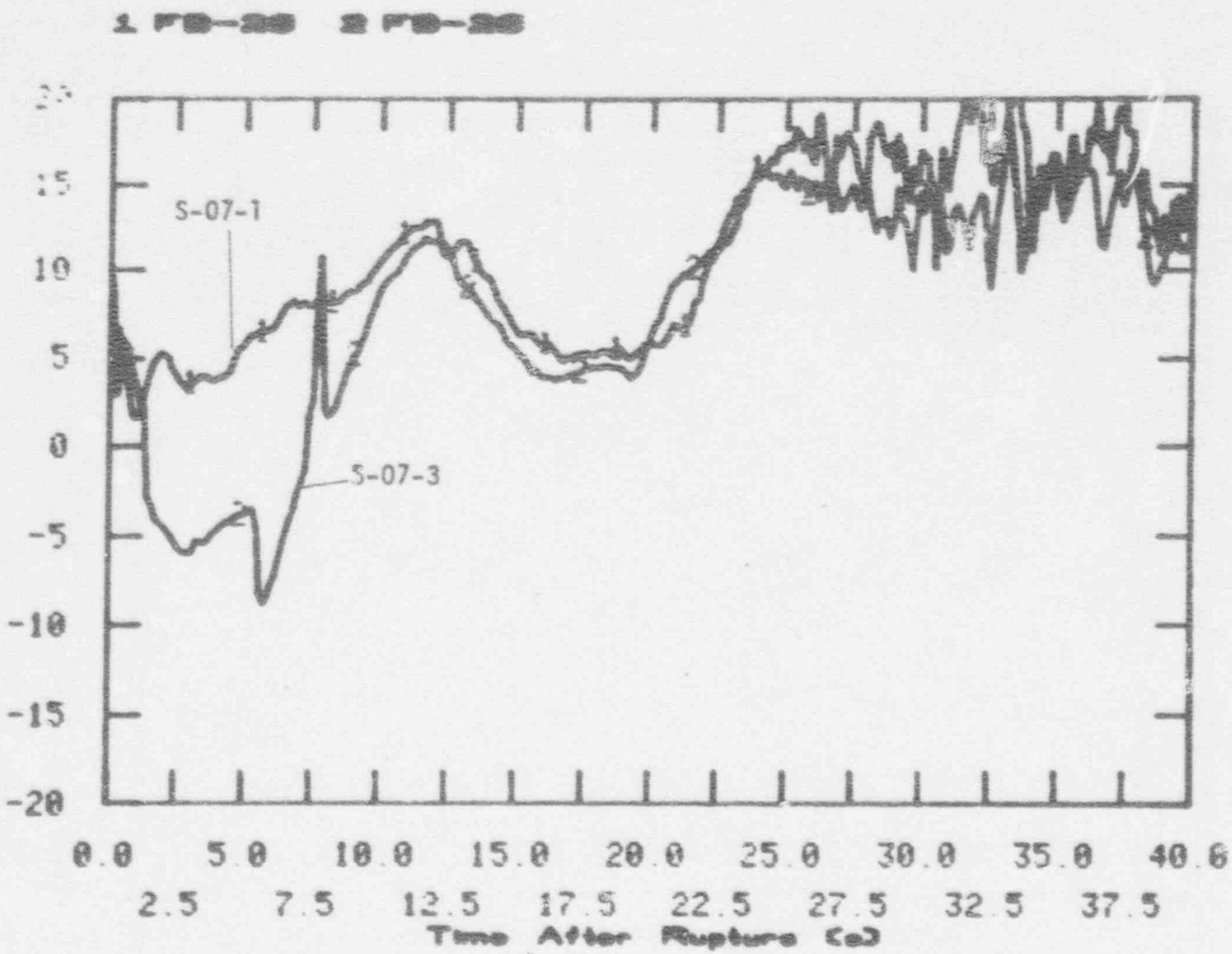

Fig. 7 Comparison of volumetric flow rates in the vessel outlet side of the broken loop for Tests S-07-1 and S-07-3.

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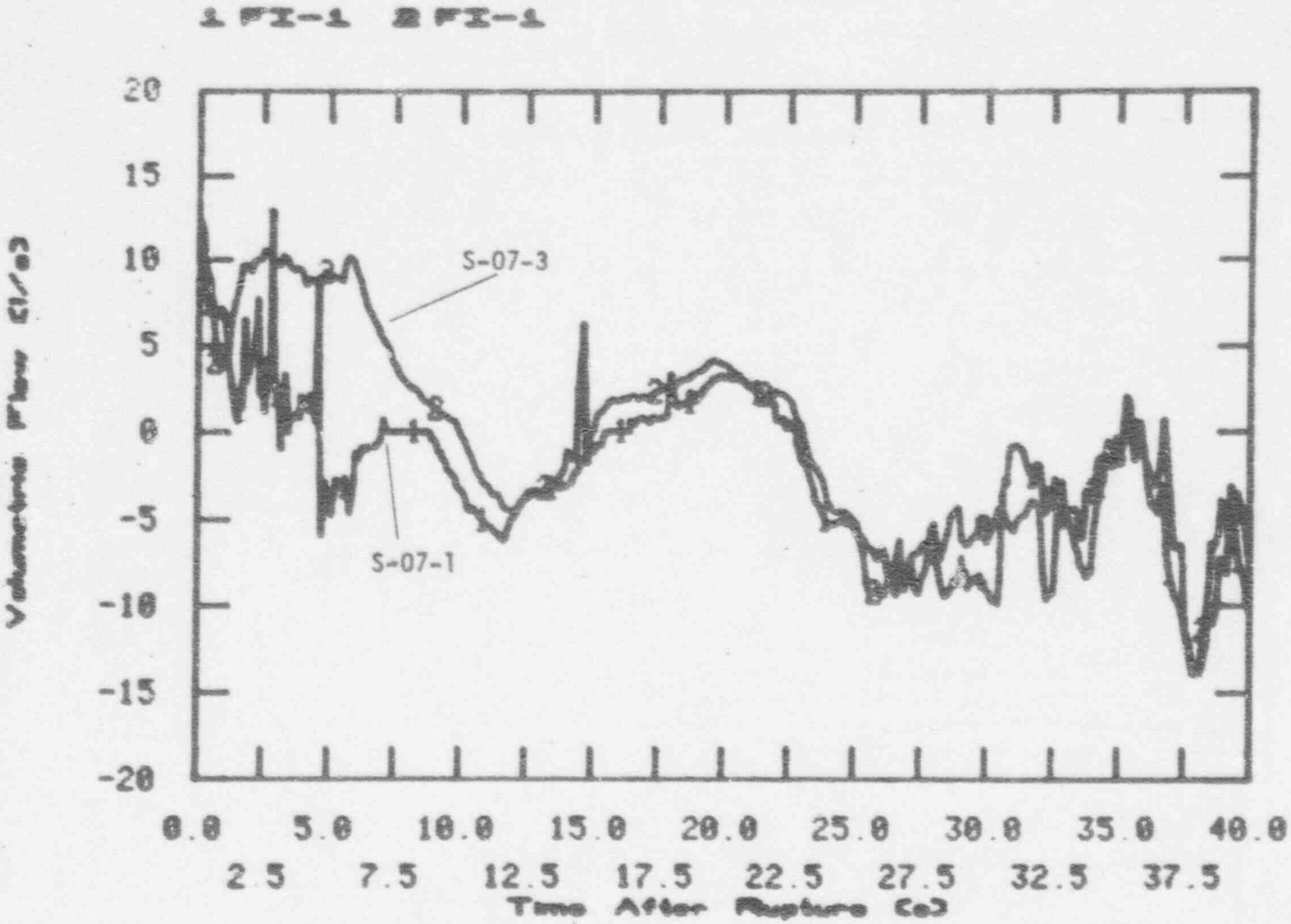

Fig. 8 Comparison of volumetric flow rates in the vessel outlet side of the intact loop for Tests S-07-1 and S-07-3.

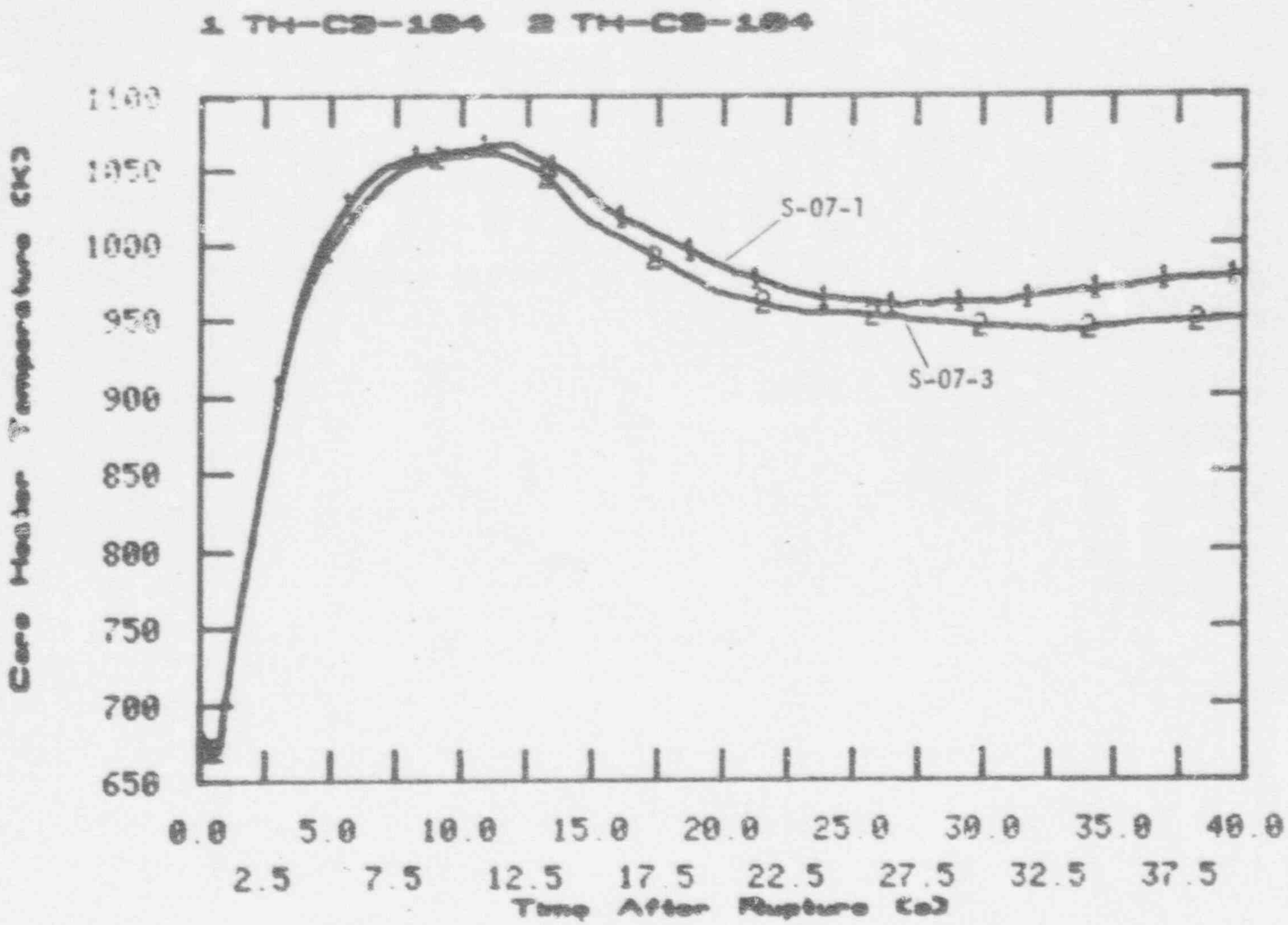

Fig. 9 Comparison of core thermal response between Tests S-07-1 and S-07-2 at the core peak power elevation (184-cm elevation).

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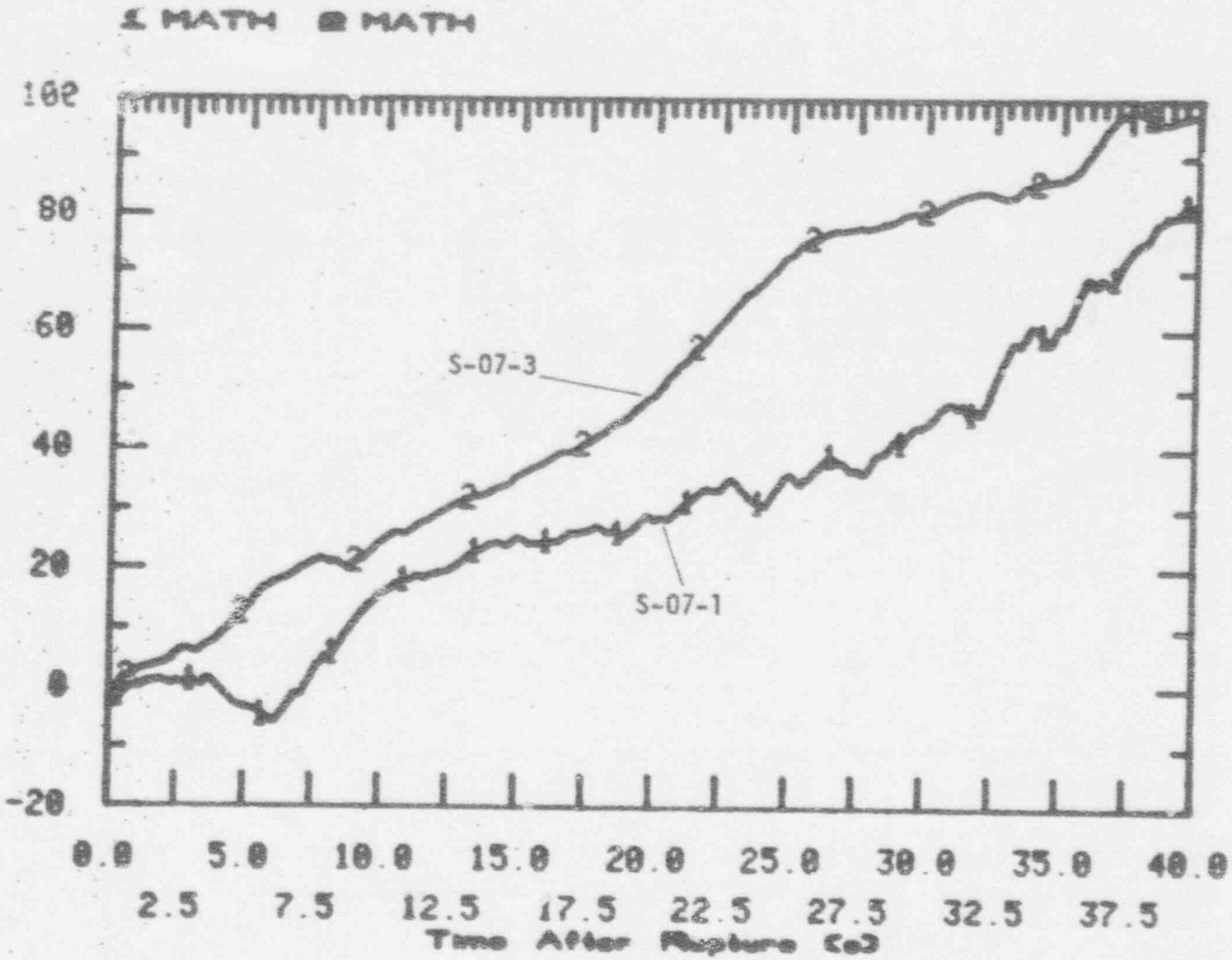

Fig. 10 Comparison of core exit integrated volumetric flow rates for Tests S-07-1 and S-07-3.

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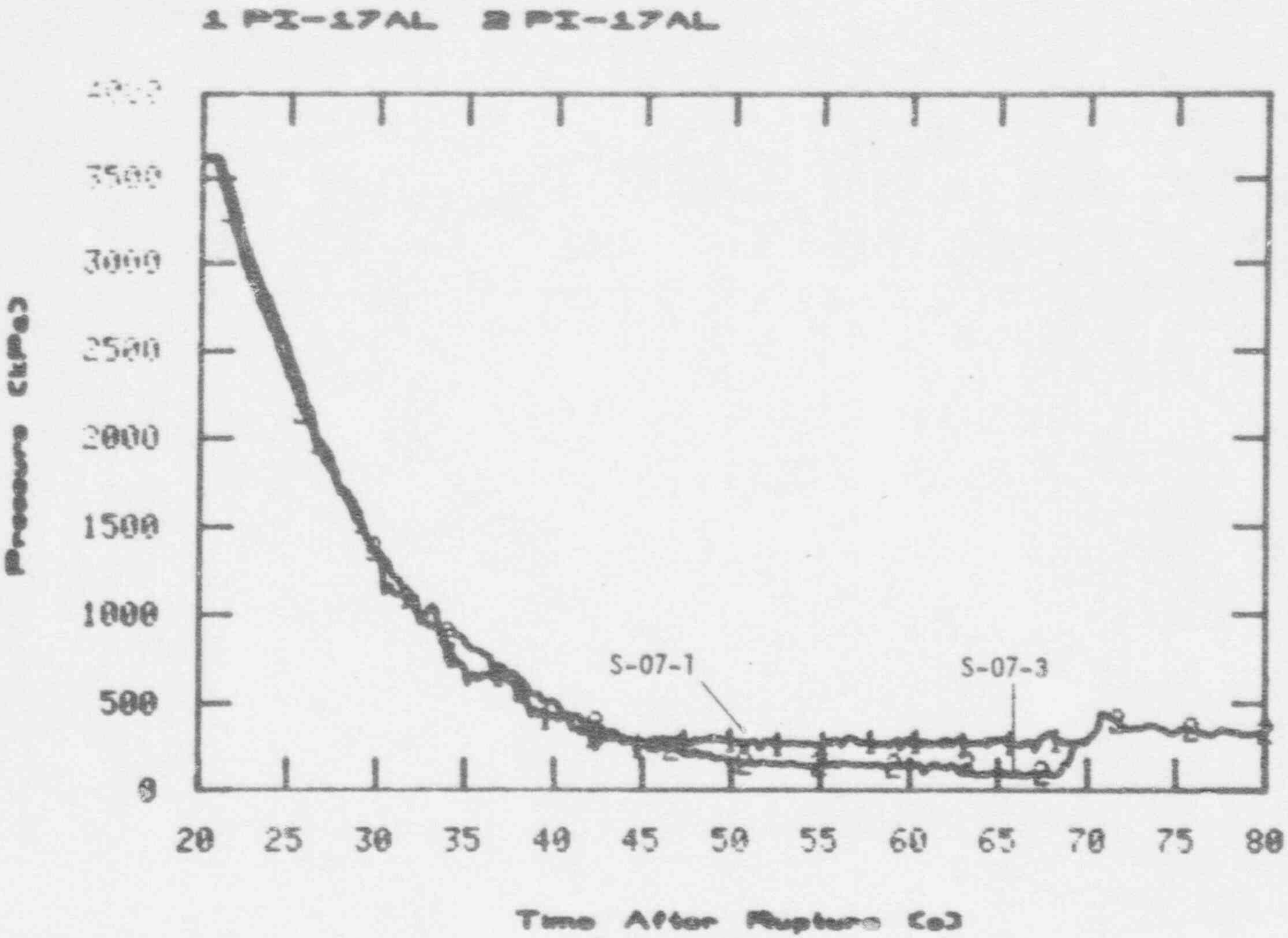

Fig. 11 Comparison of system depressurization for Tests S-07-1 and S-07-3.

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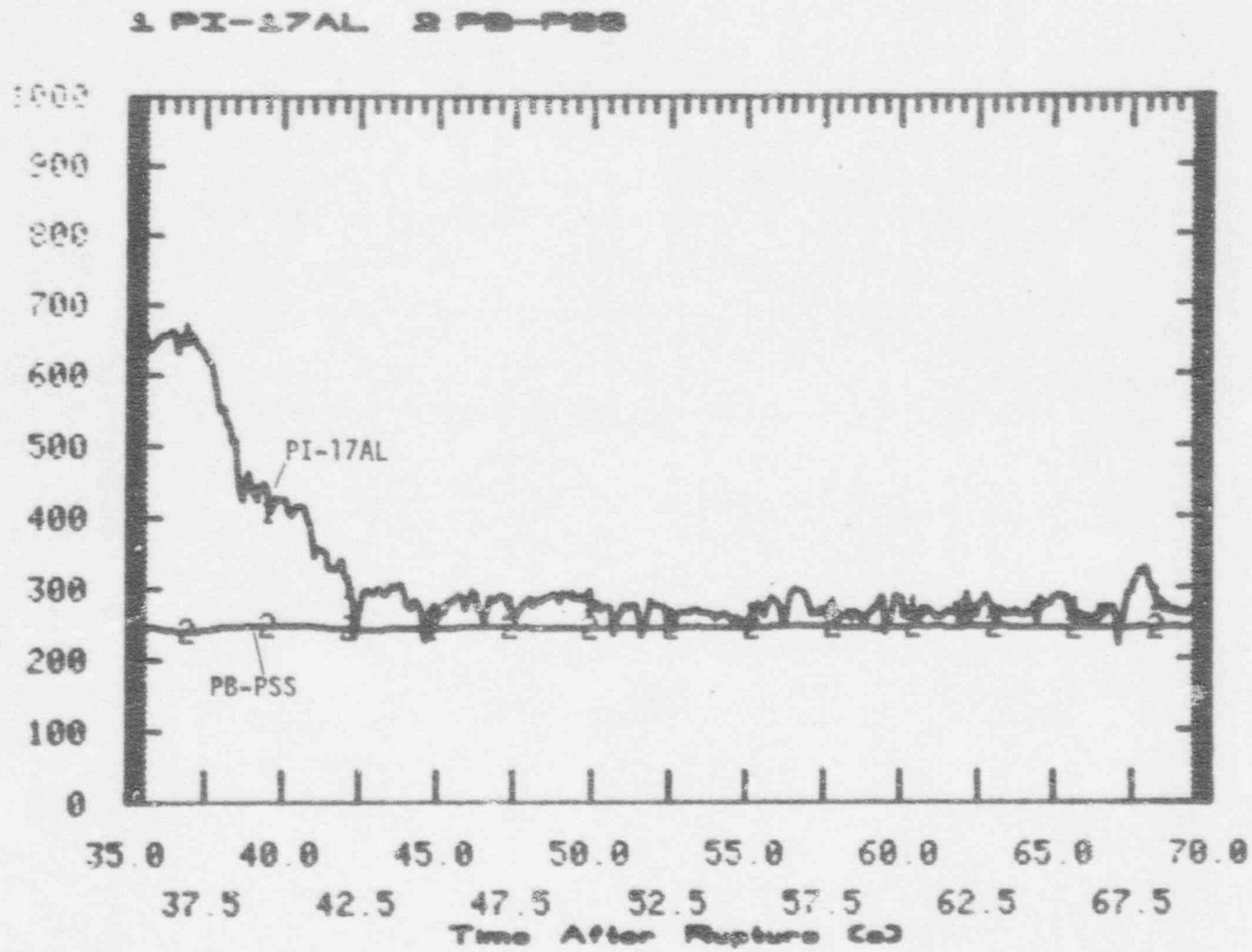

Fig. 12 Comparison of system pressure and containment pressure for Test S-07-1.

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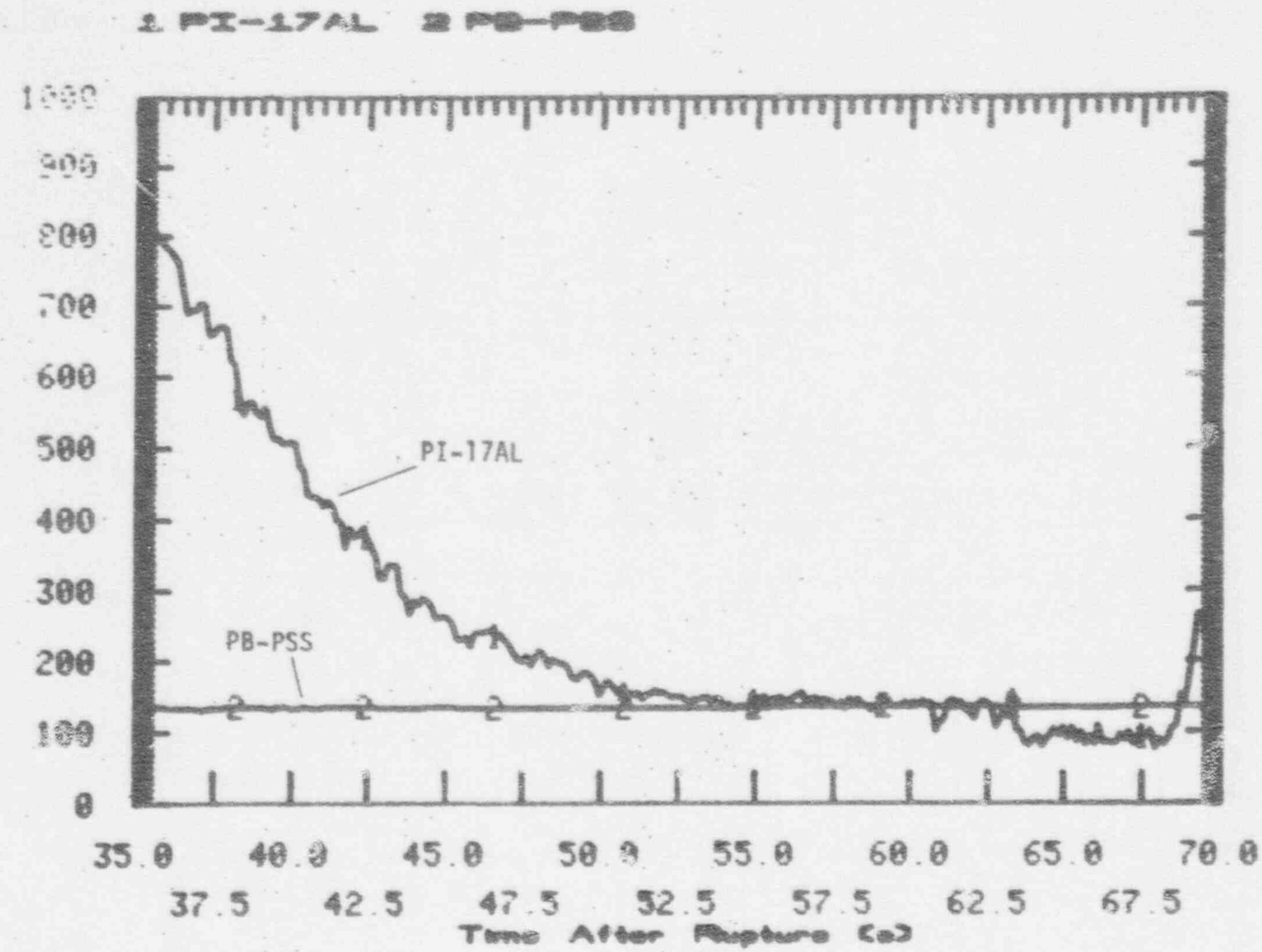

Fig. 13 Comparison of system pressure and containment pressure for Test S-07-3.

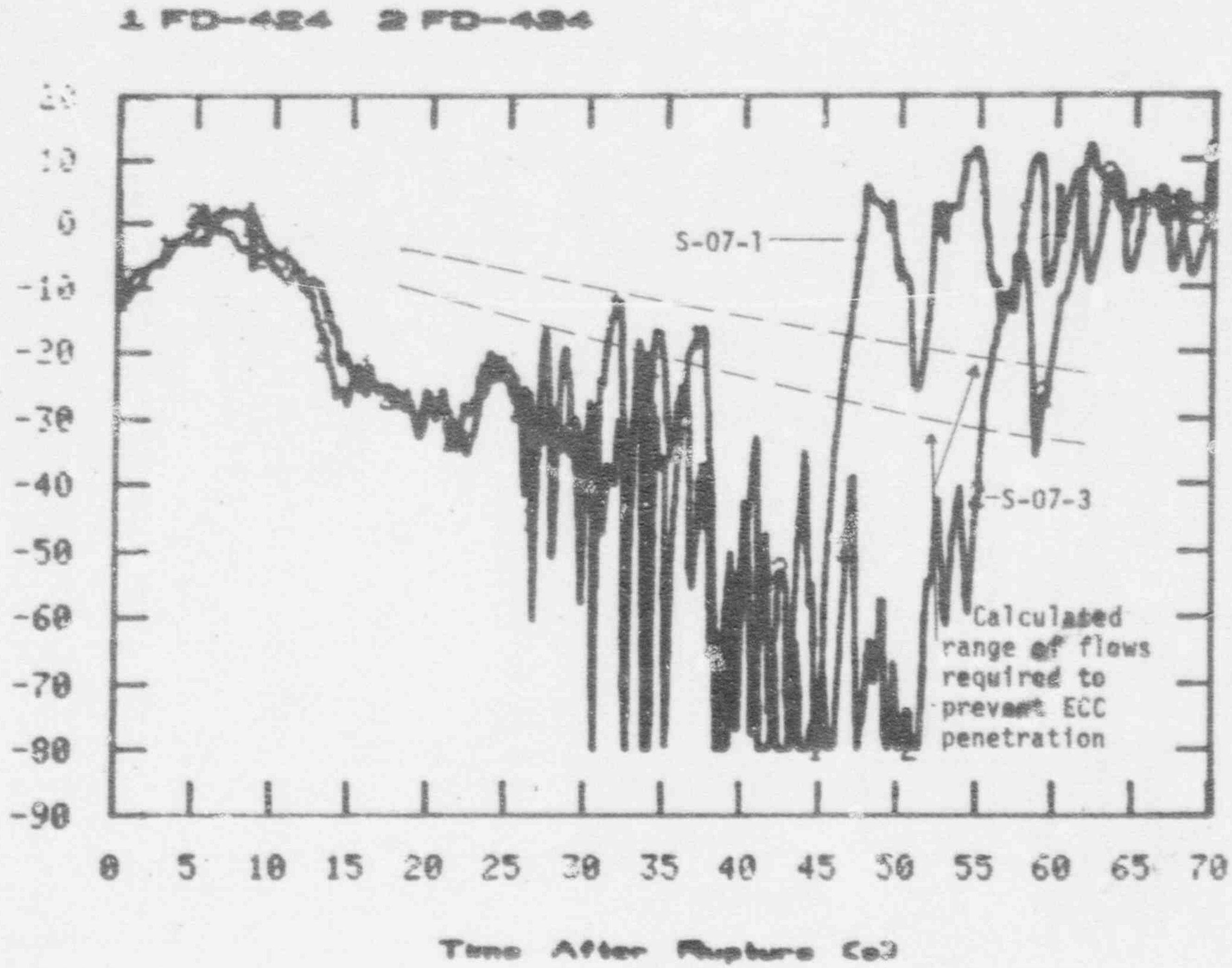

Fig. 14 Comparison of downcomer volumetric flow rates for Tests S-07-1 and S-07-3.

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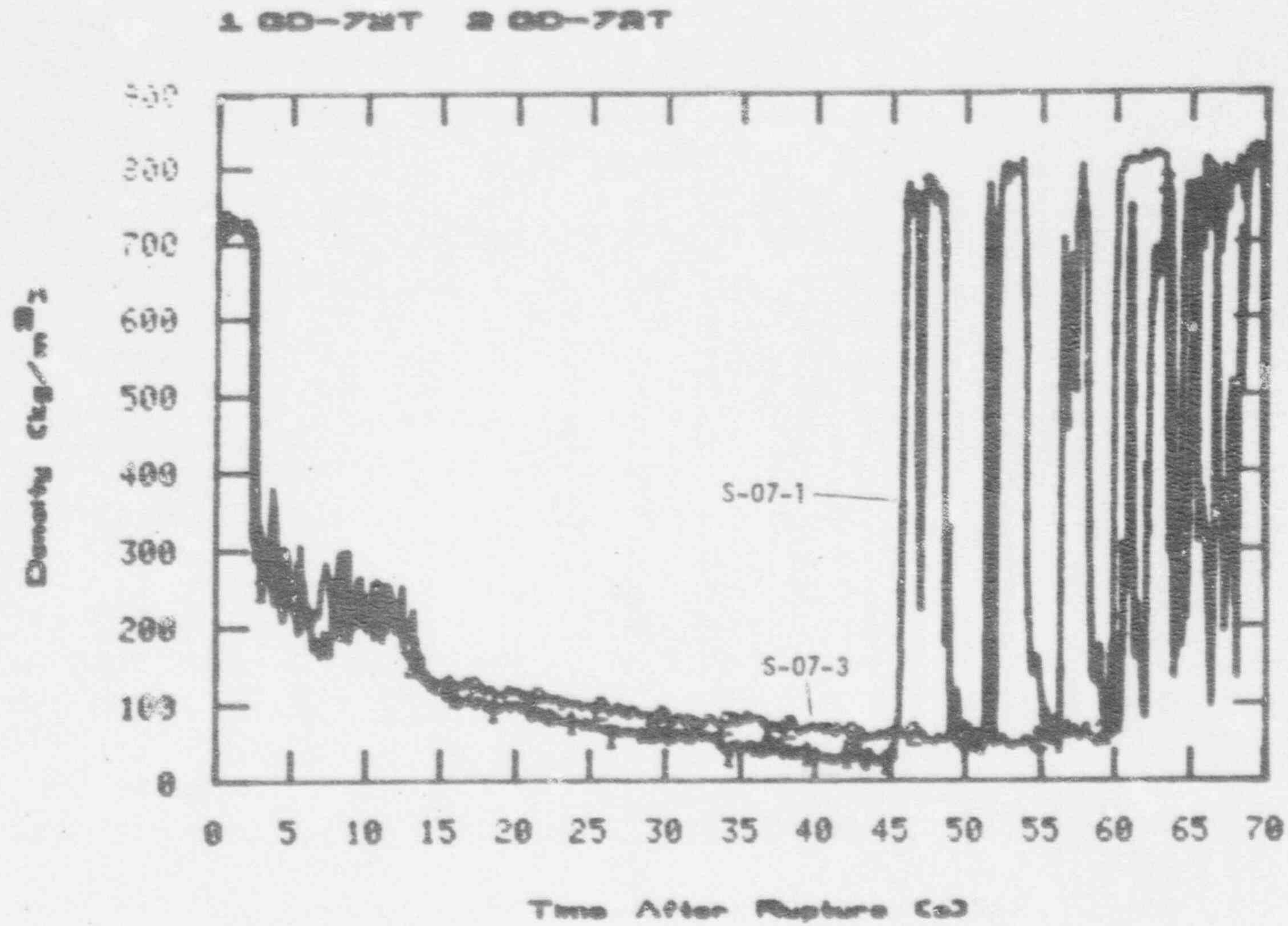

Fig. 15 Comparison of densities near the top of the downcomer for Tests S-07-1 and S-07-3.

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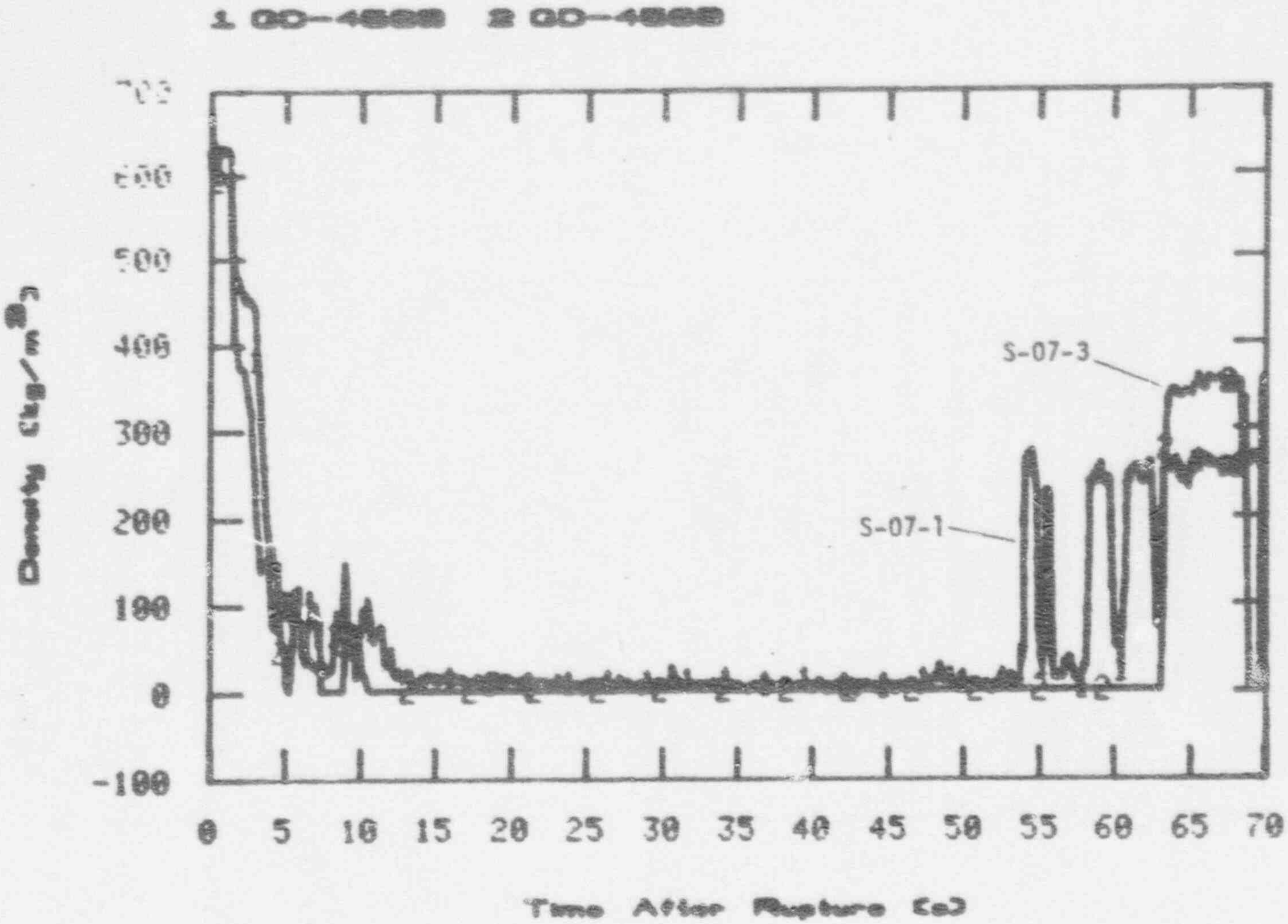

Fig. 16 Comparison of densities near the bottom of the downcomer for Tests S-07-1 and S-07-3.

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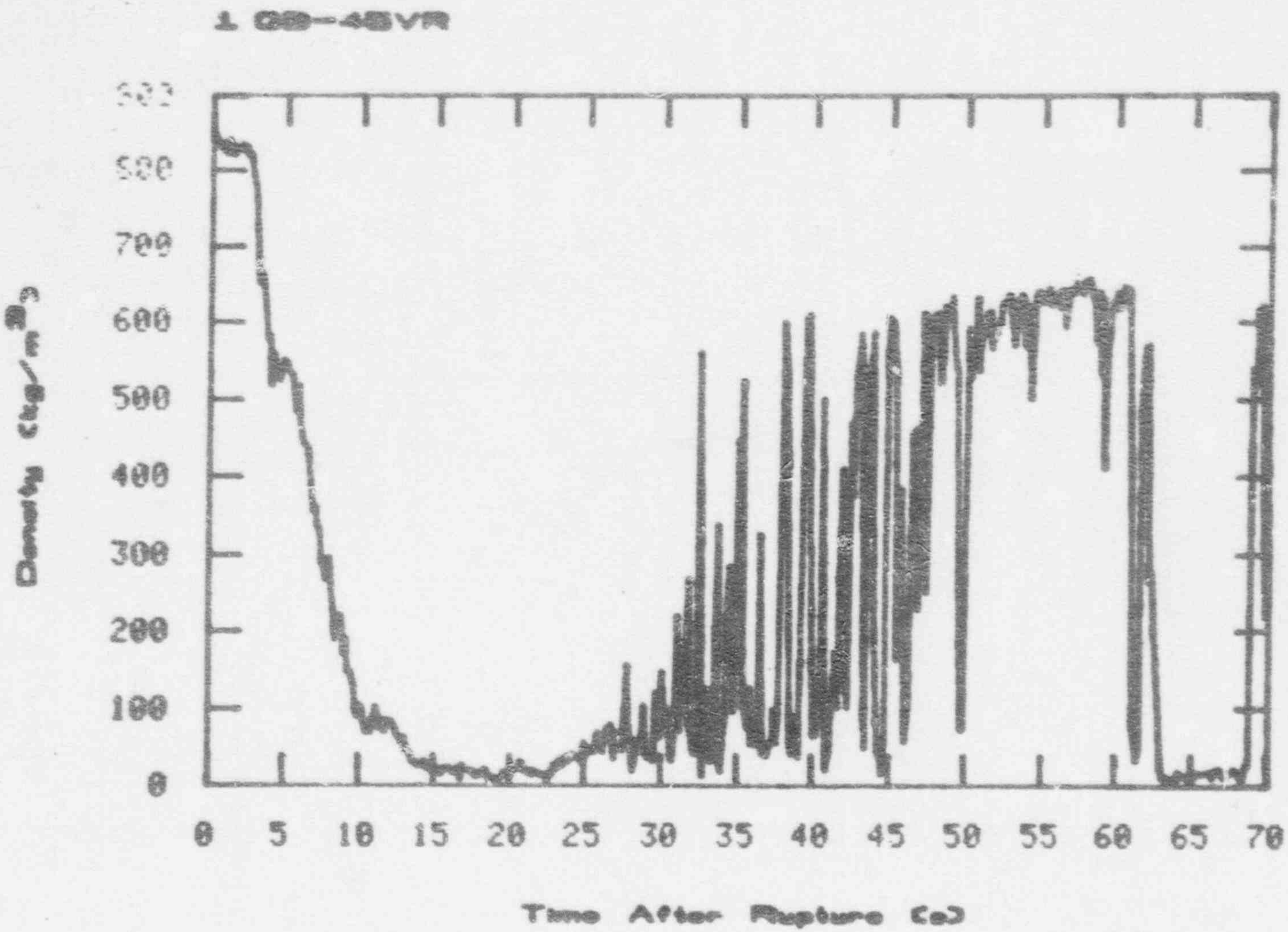

Fig. 17 Fluid density in the vessel inlet side of the broken loop for Test S-07-3.

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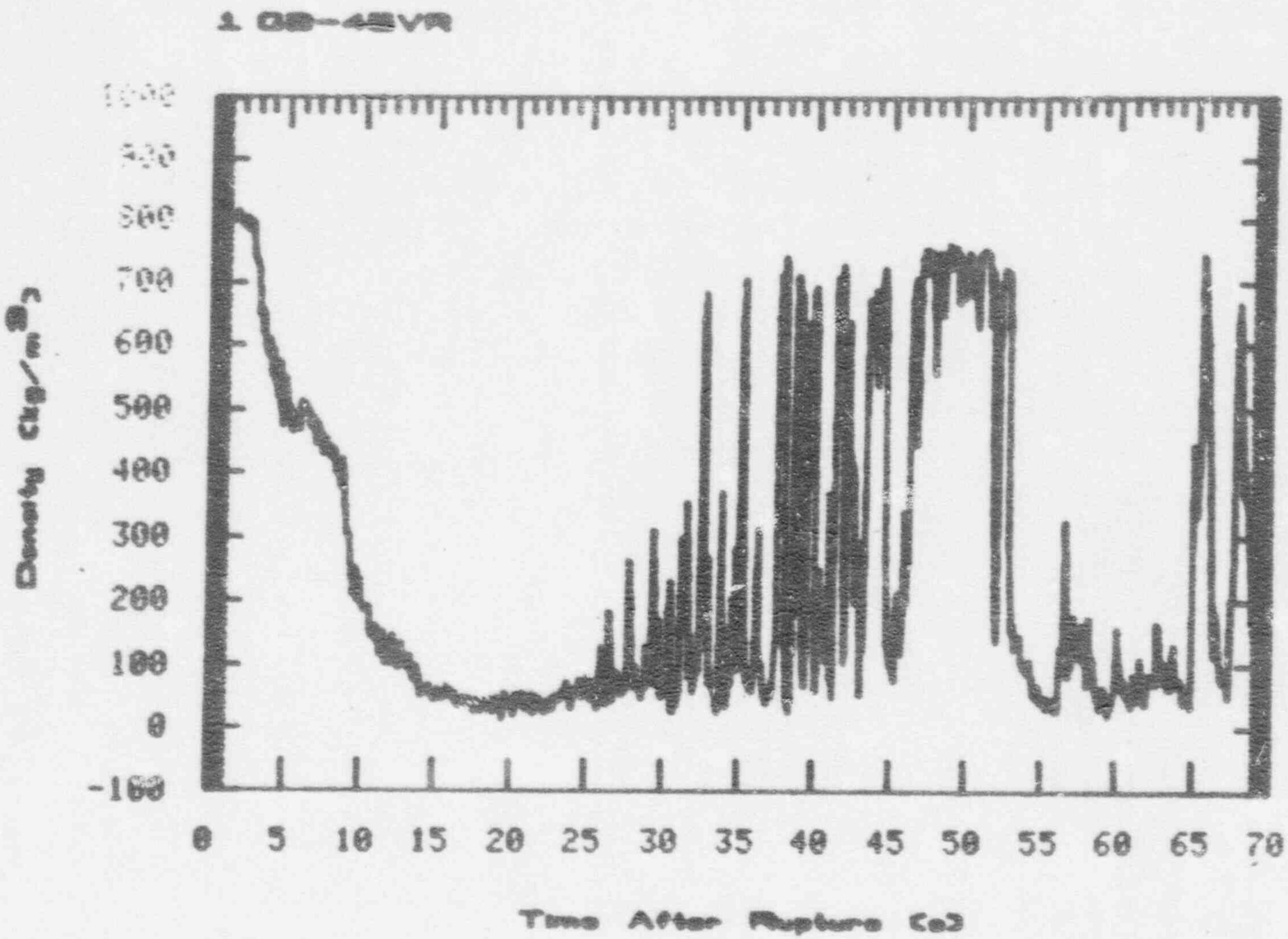

Fig. 18 Fluid density in the vessel inlet side of the broken loop for Test S-07-1.

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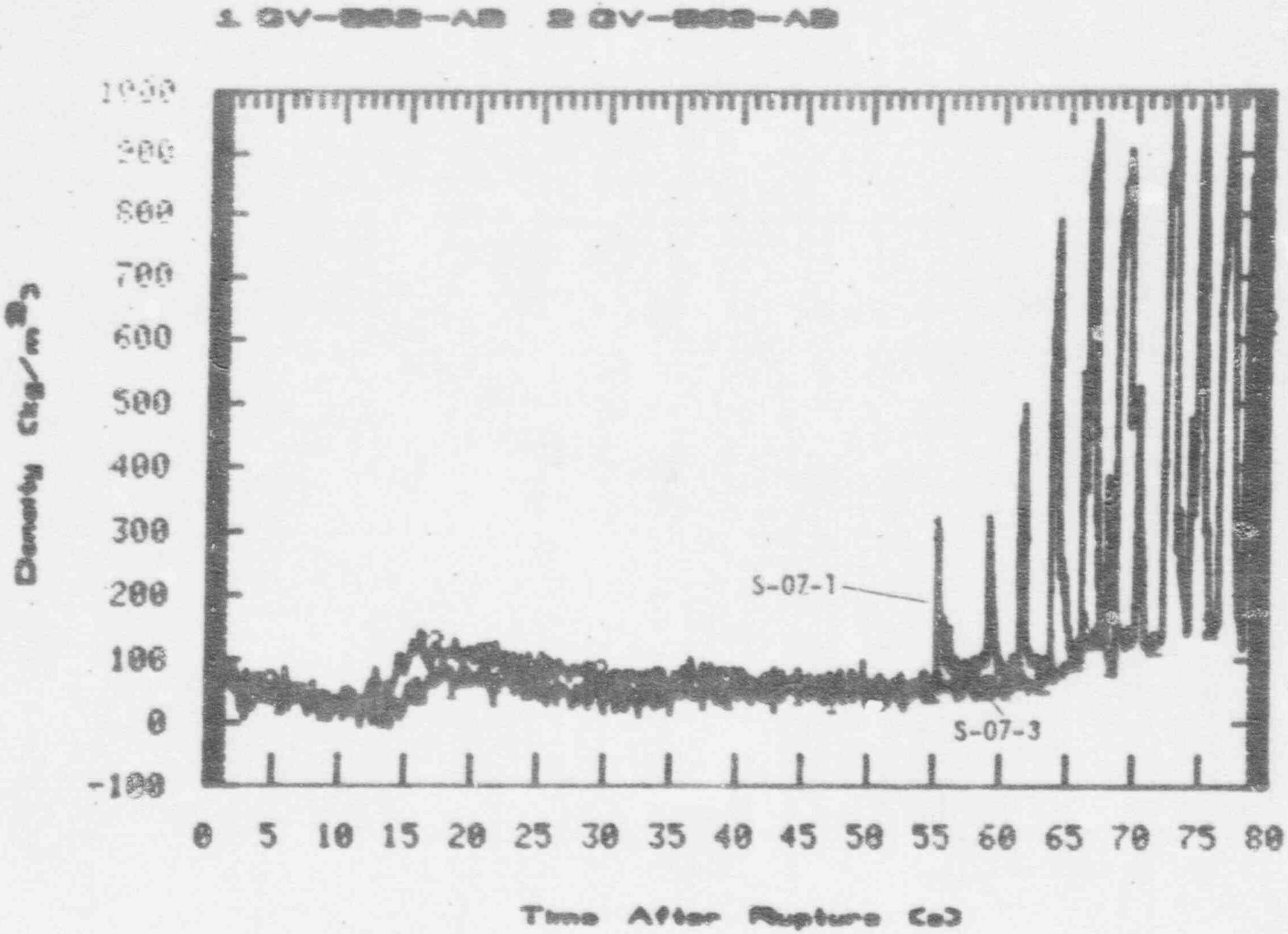

Fig. 19 Comparison of the core inlet densities for Tests S-07-1 and S-07-3.

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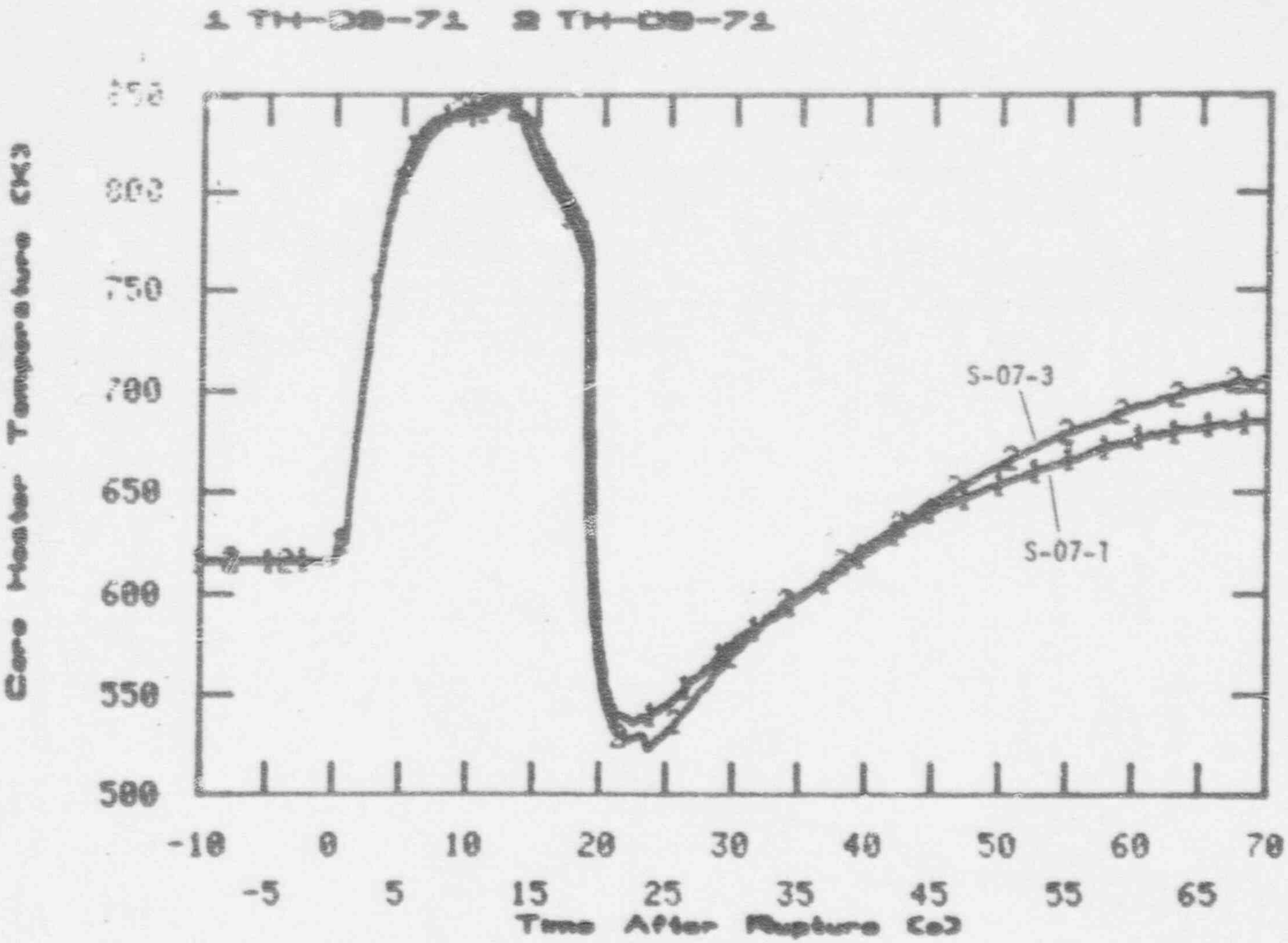

Fig. 20 Comparison of core thermal response at the 71-cm elevation for Tests S-07-1 and S-07-3.

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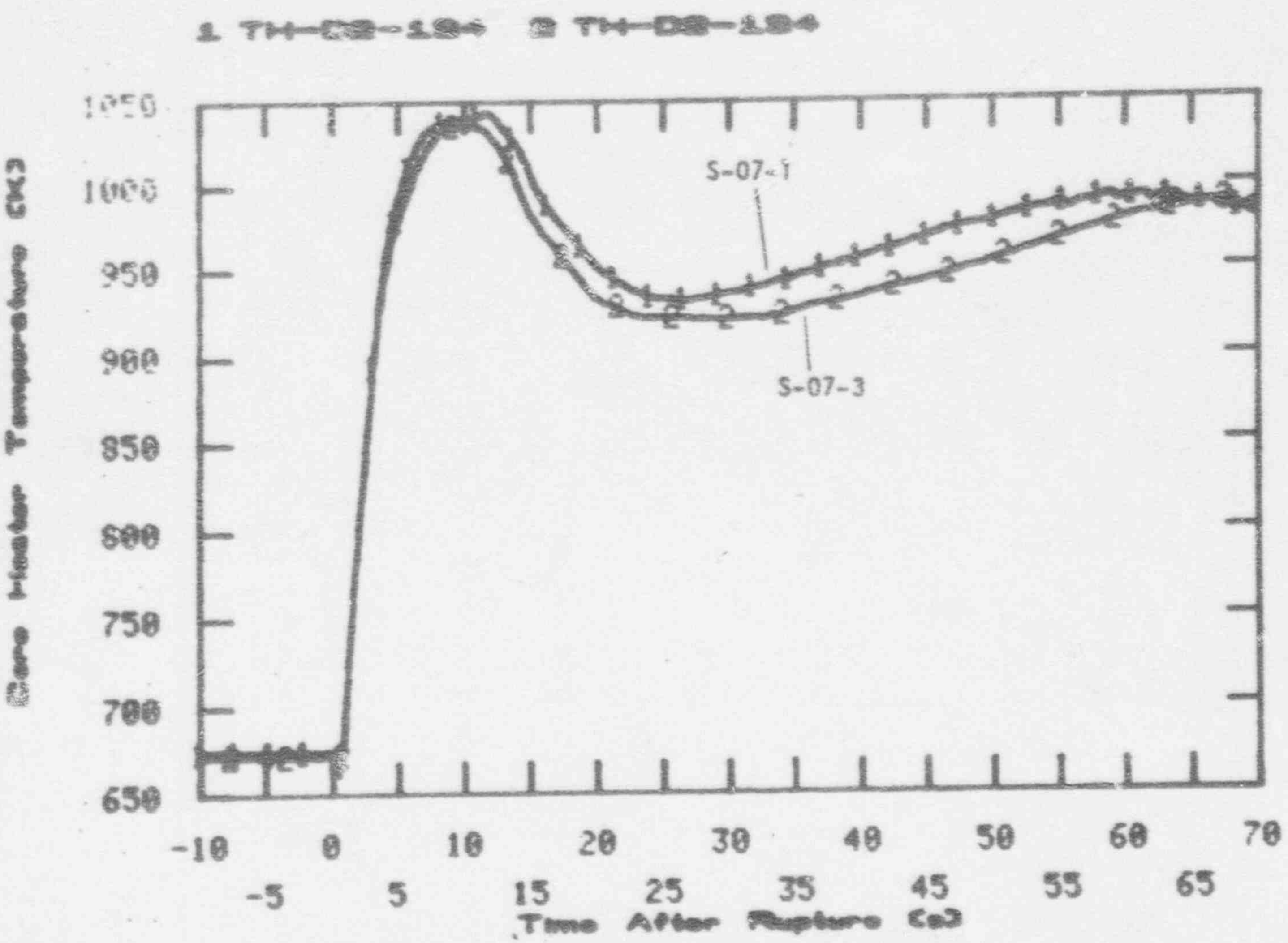

Fig. 21 Comparison of core thermal response at the 134-cm elevation for Tests S-07-1 and S-07-3.

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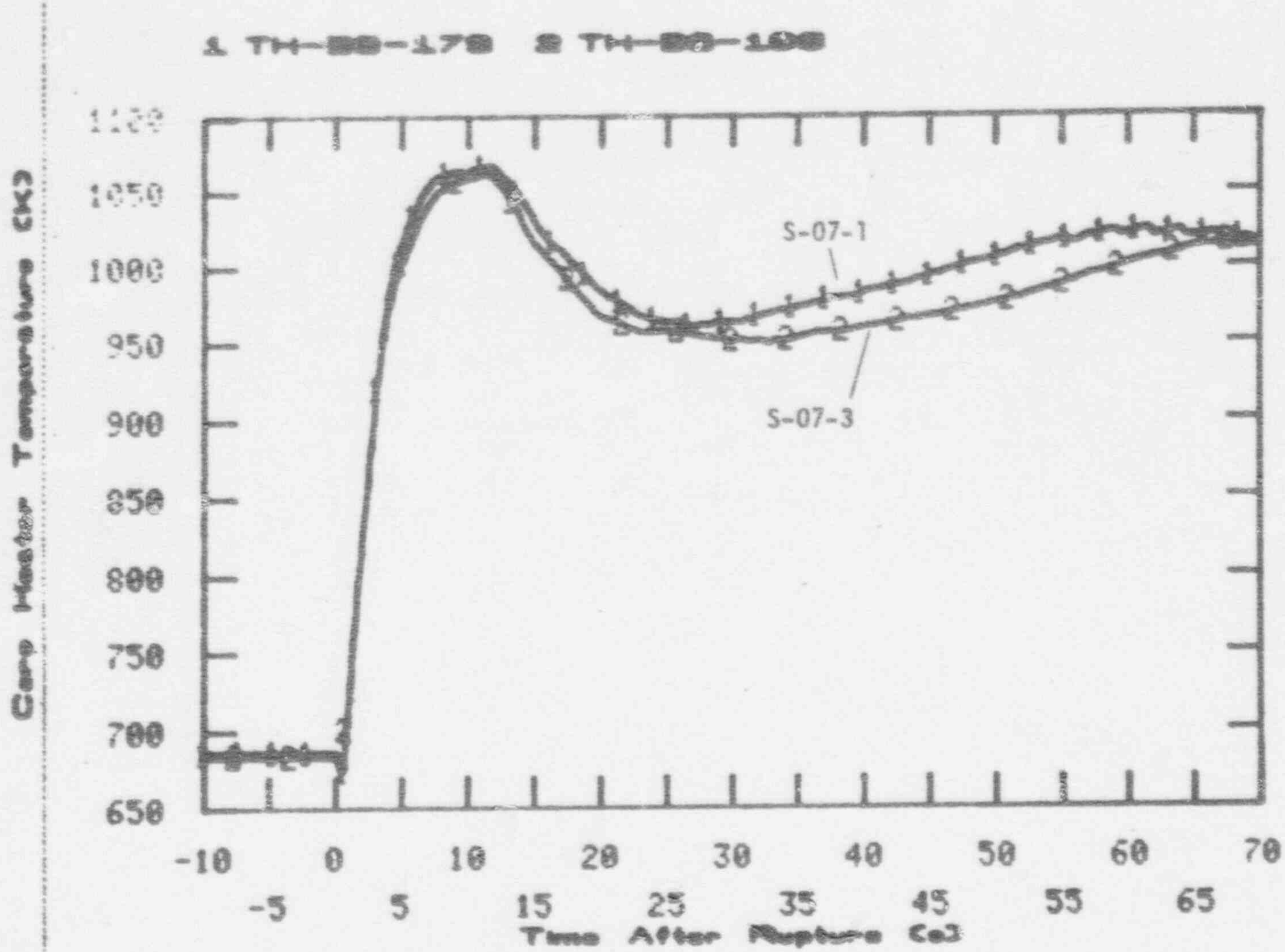

Fig. 22 Comparison of core thermal response at the 180-cm elevation for Tests S-70-1 and S-07-3.

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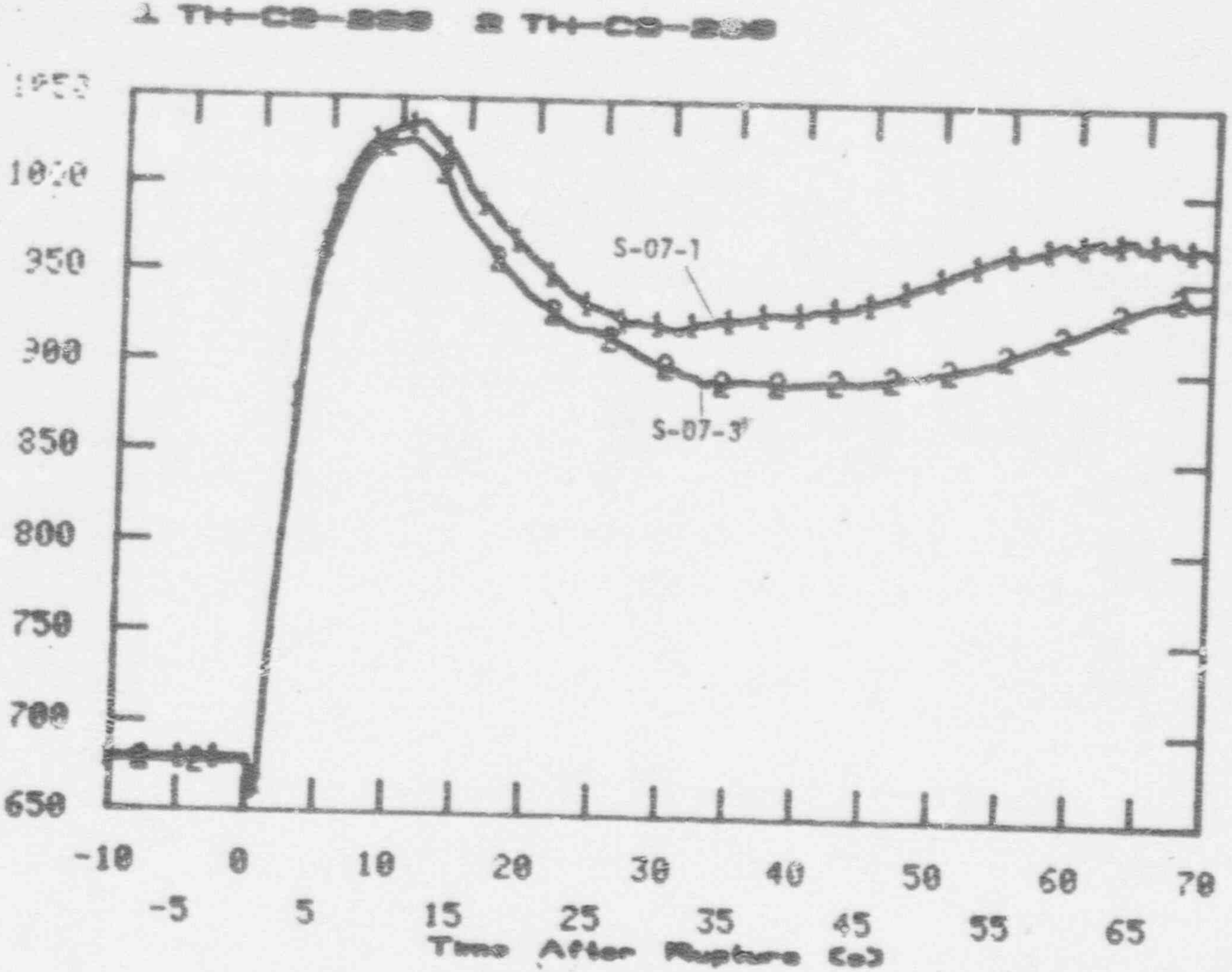

Fig. 23 Comparison of core thermal response at the 290-cm elevation for Tests S-07-1 and S-07-3.

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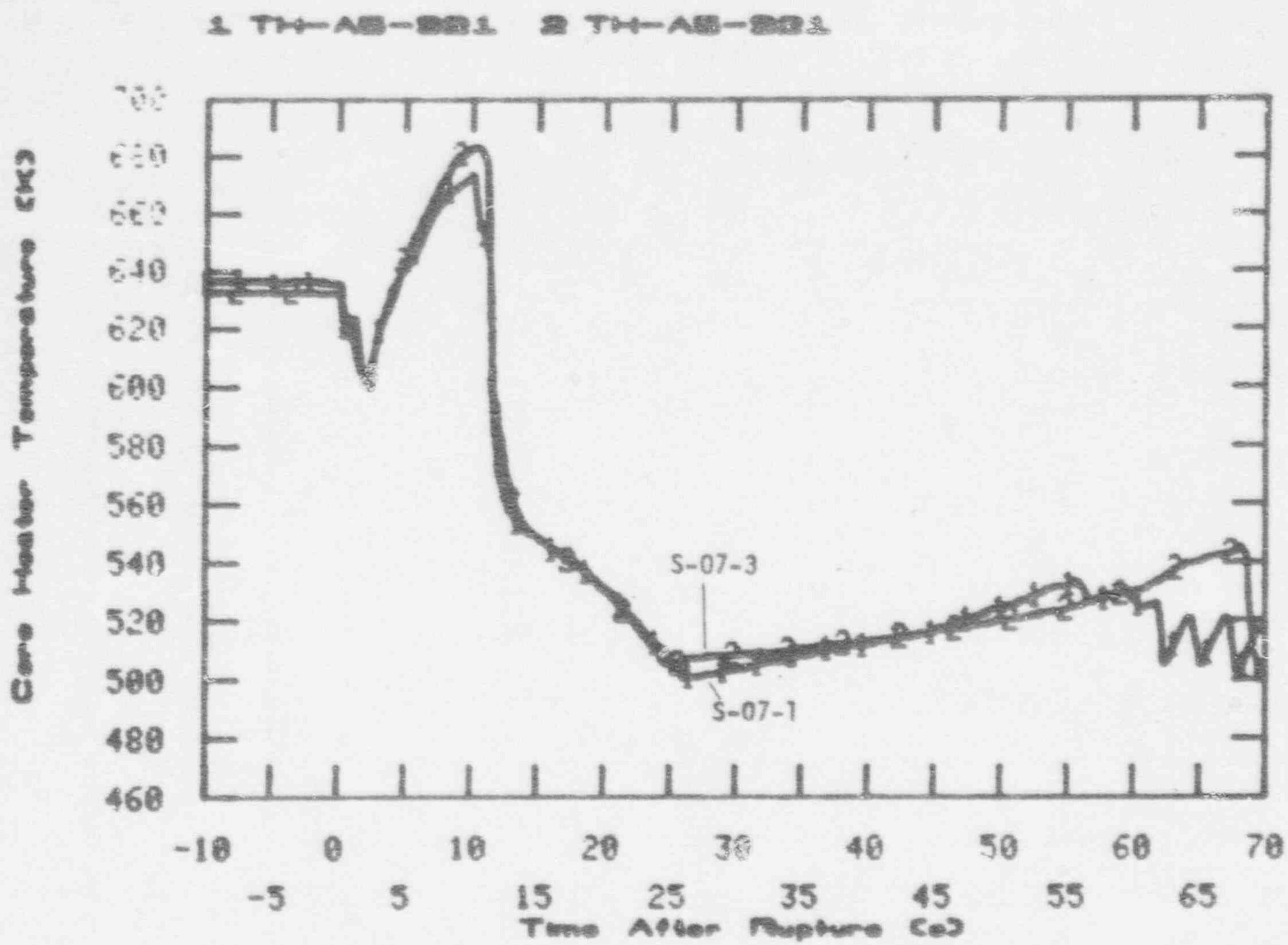

Fig. 24 Comparison of core thermal response at the 321-cm elevation for Tests S-07-1 and S-07-3.

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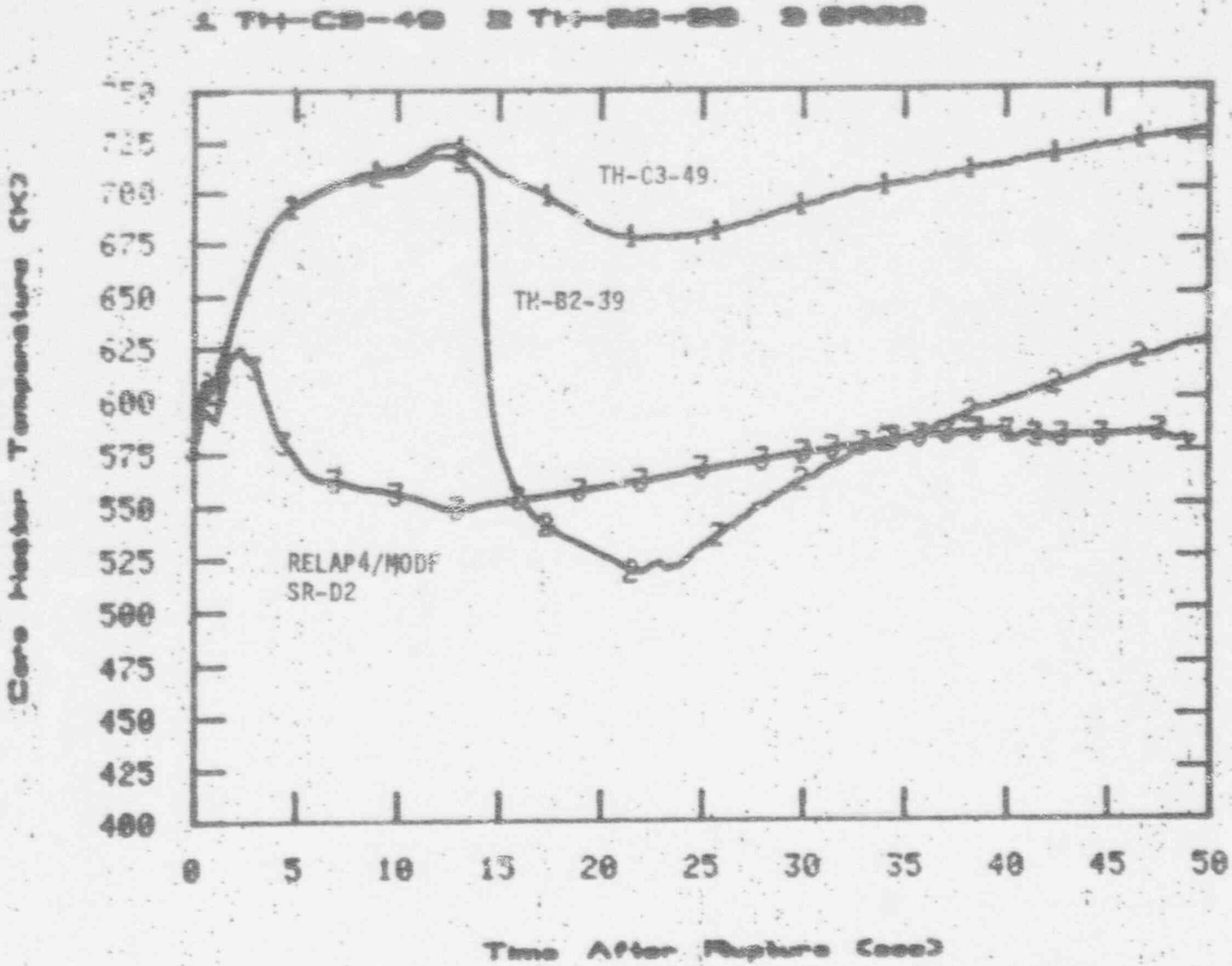

Fig. 25 Comparison of the core thermal response near the bottom of the core for Test S-07-3 data and the RELAP4/MODF pretest calculation.

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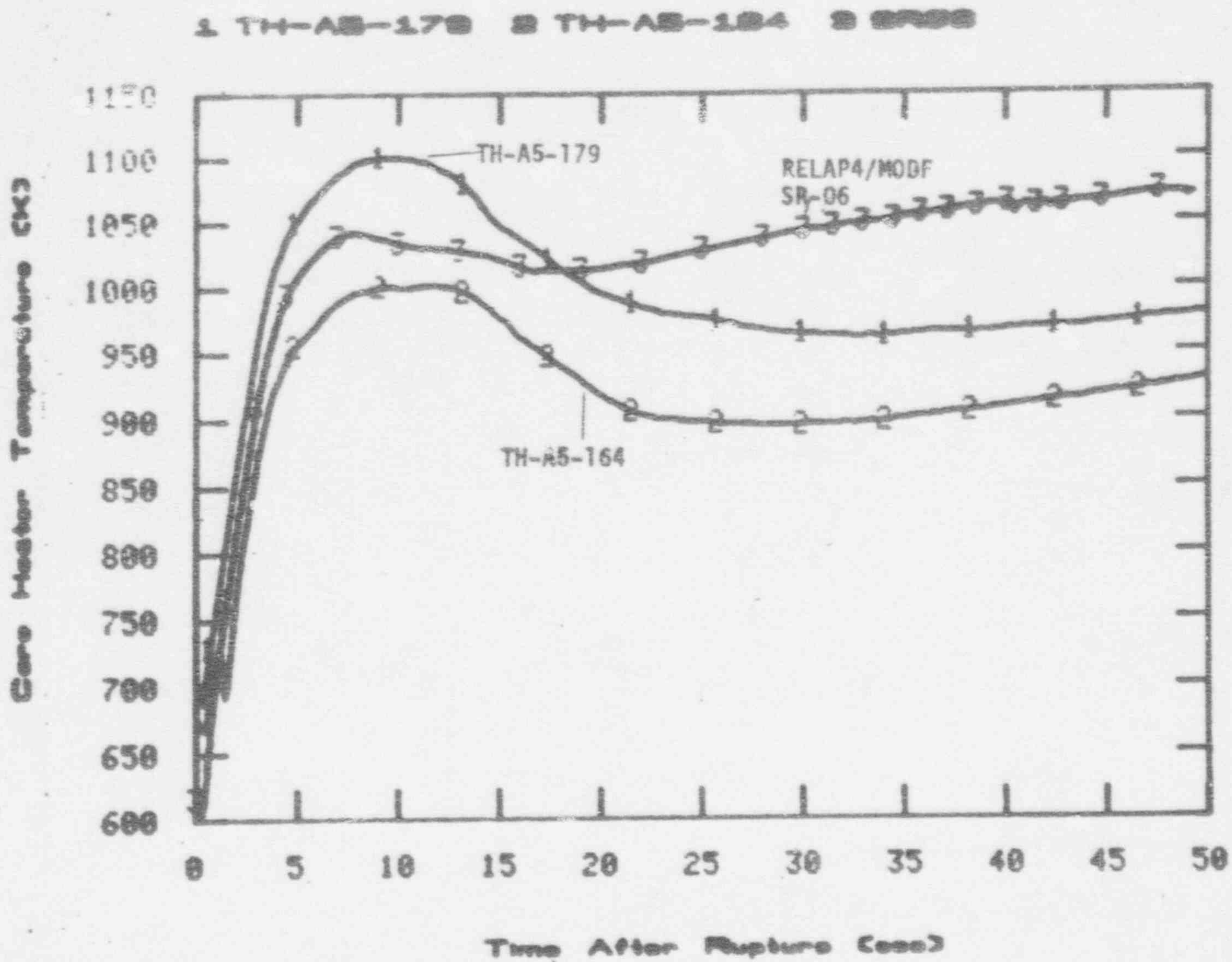

Fig. 26 Comparison of the core thermal response near the middle of the core for Test S-07-3 data and the RELAP4/MODF pretest calculation.

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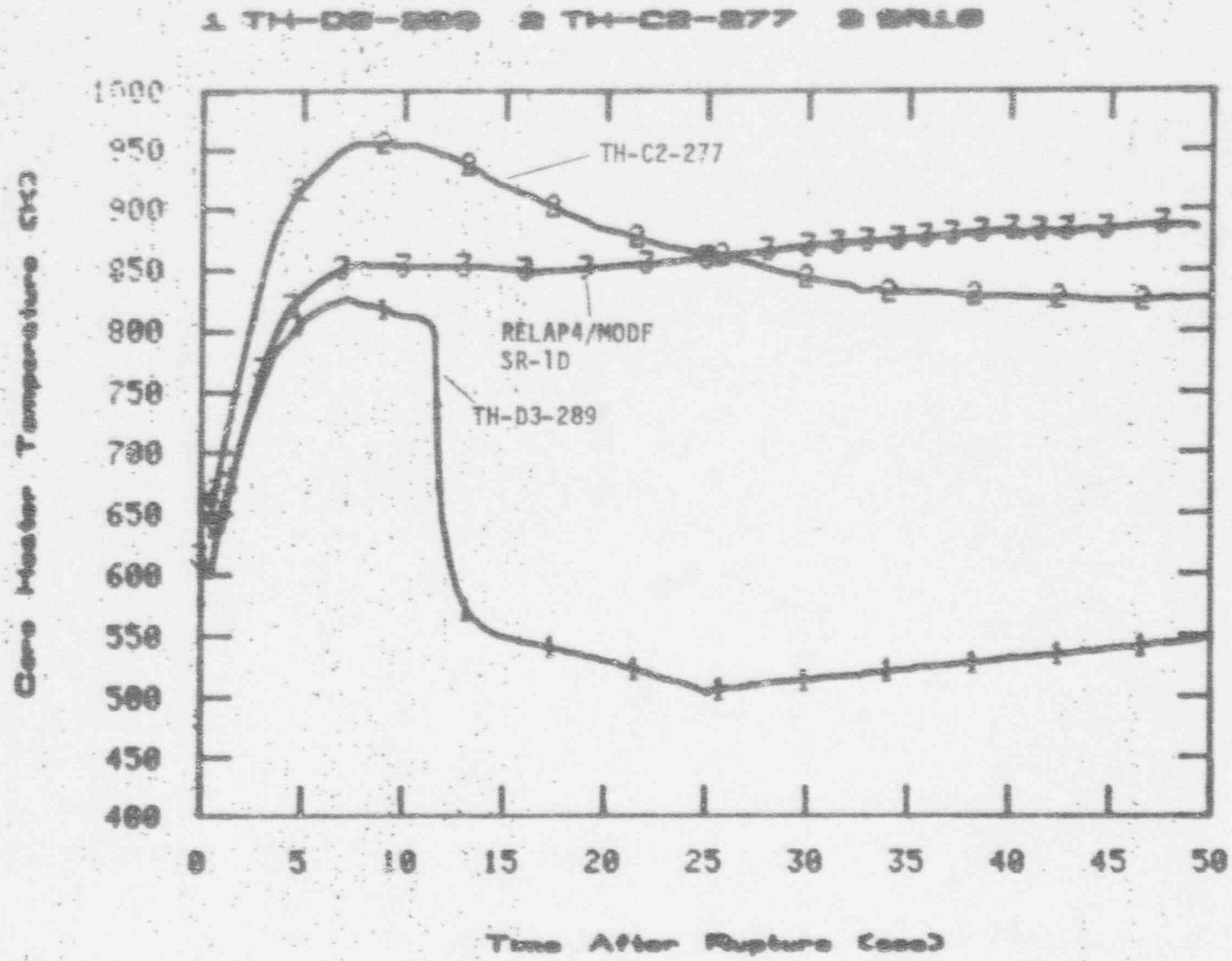

Fig. 27 Comparison of the core thermal response near the top of the core for Test S-07-3 data and the RELAP4/MODF pretest calculation.

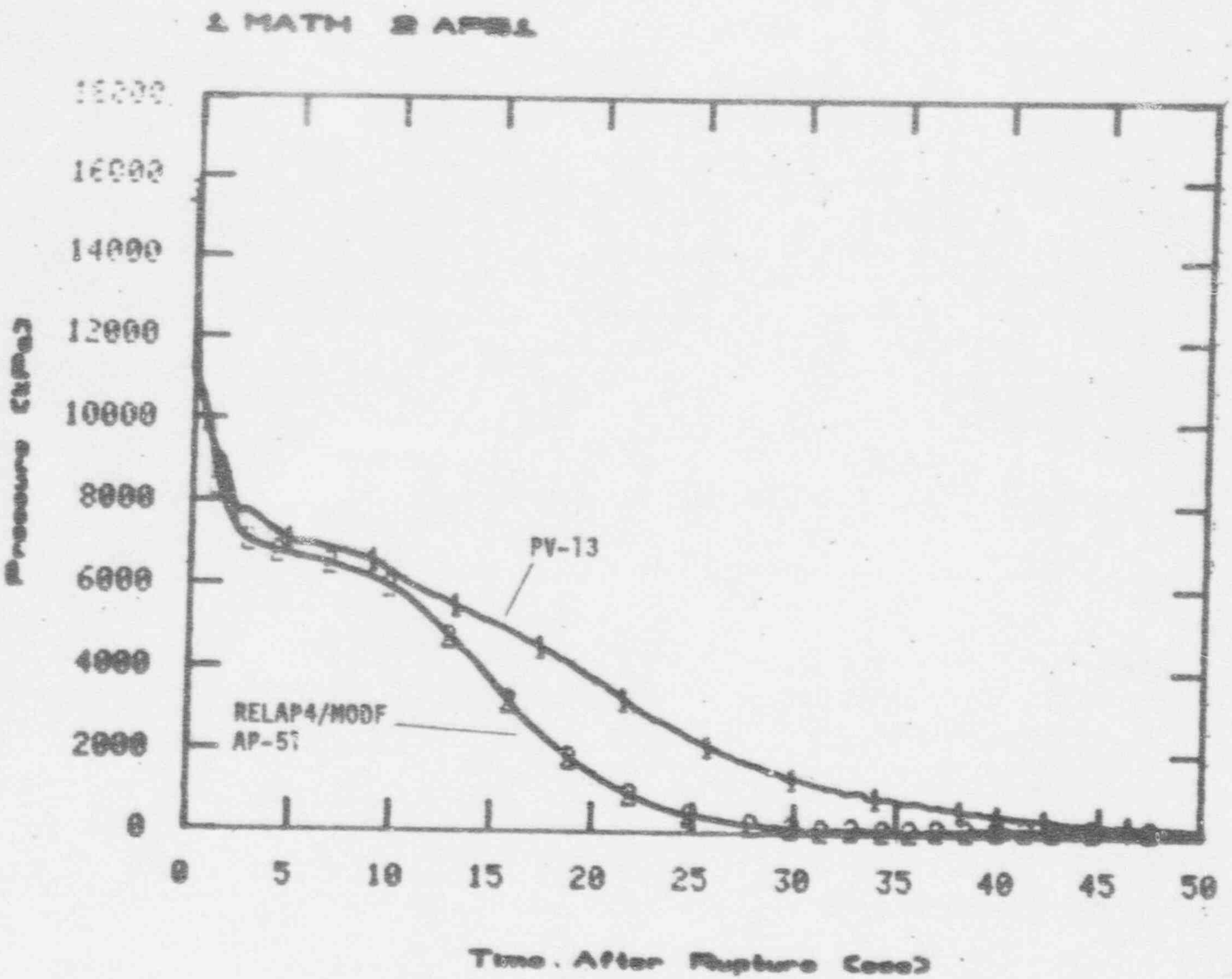

Fig. 28 Comparison of system depressurization rate for Test S-07-3 data and the RELAP4/MODF pretest calculation.

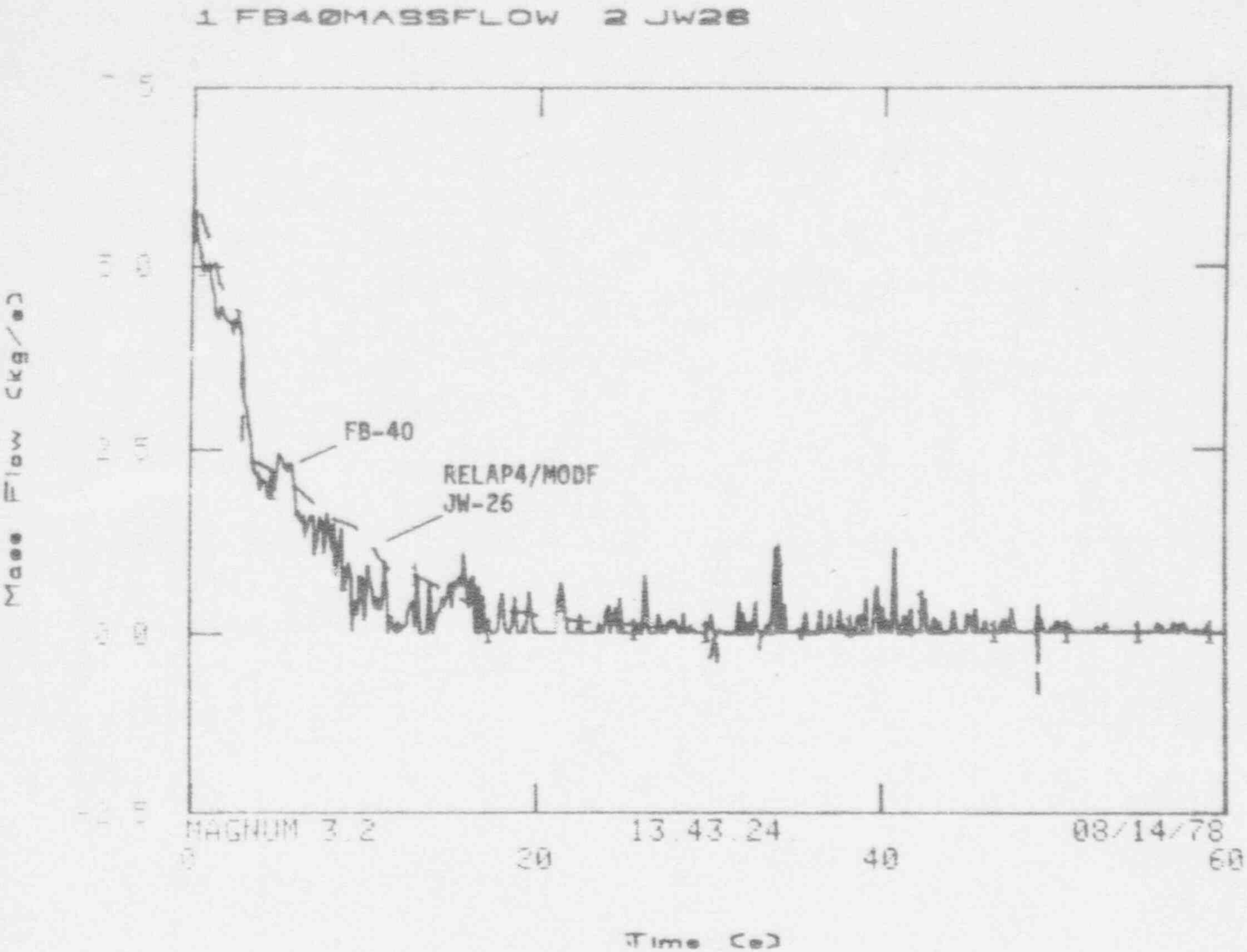

Fig. 29 Comparison of broken loop hot leg flow for Test S-07-3 data and the RELAP4/MODF pretest calculation.

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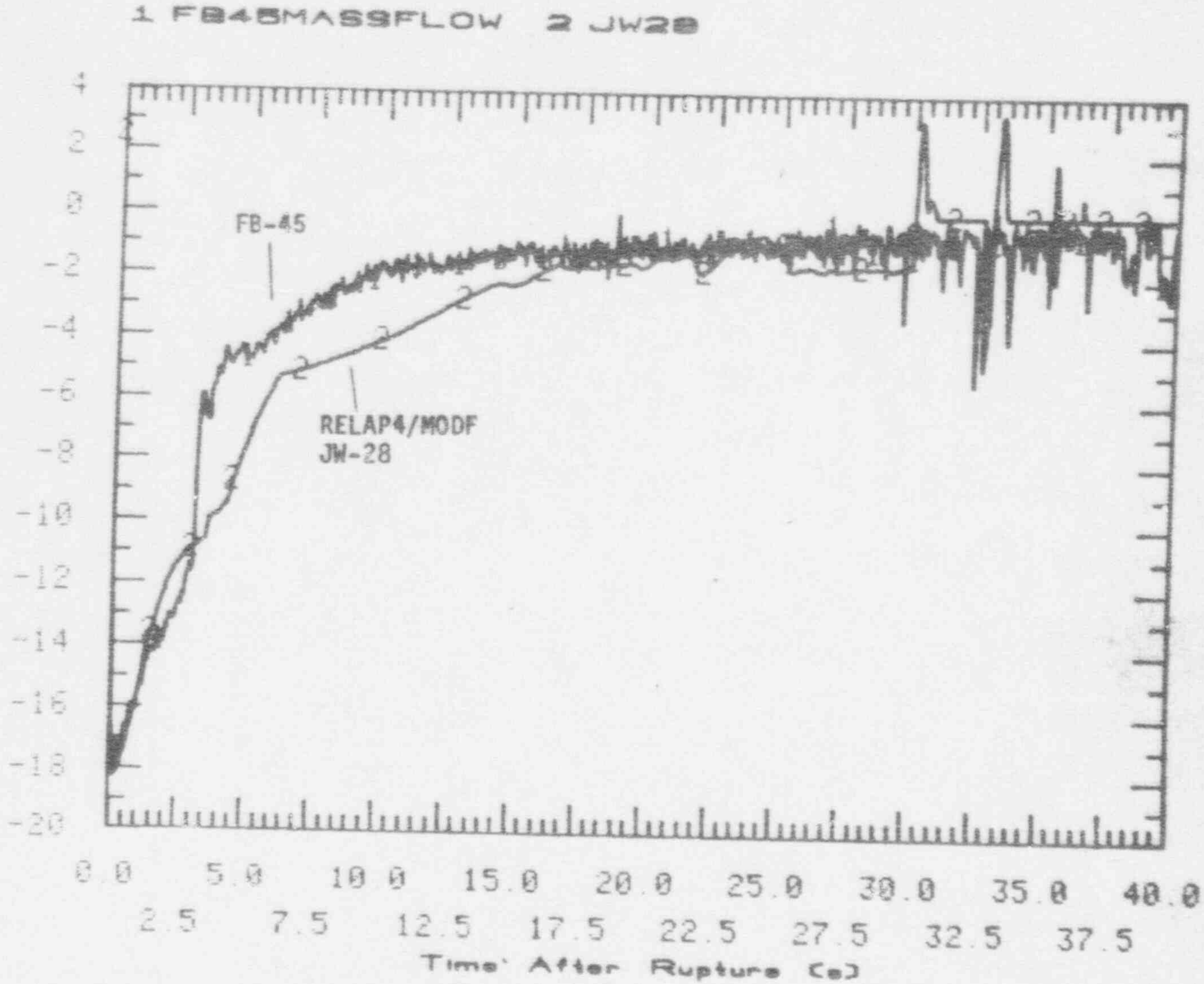

Fig. 30 Comparison of the broken loop cold leg flow for Test S-07-3 data and the RELAP4/MODF pretest calculation.

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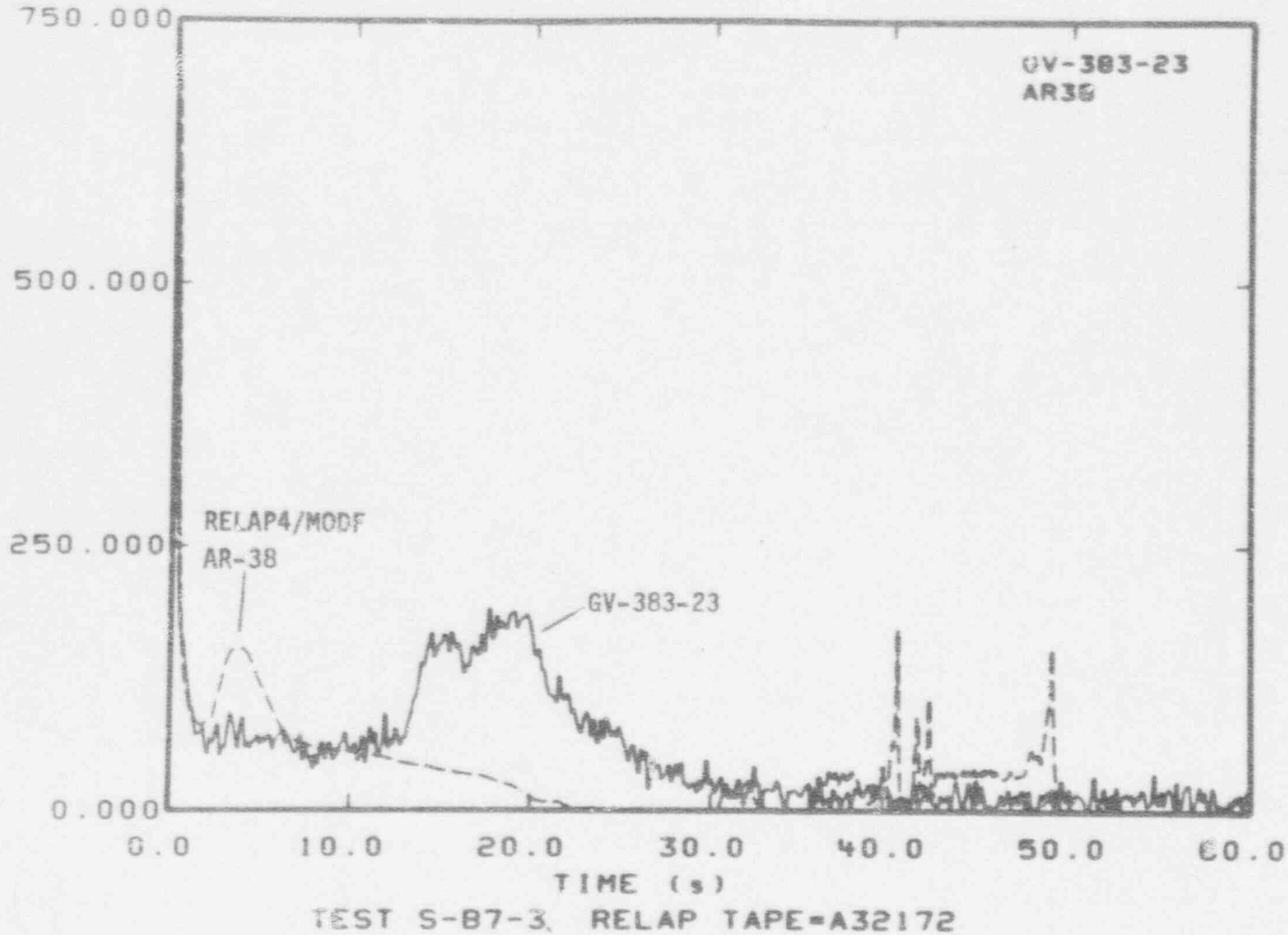

Fig. 31 Comparison of density in the midsection of the core for Test S-07-3 data and the RELAP4/MODF pretest calculation.

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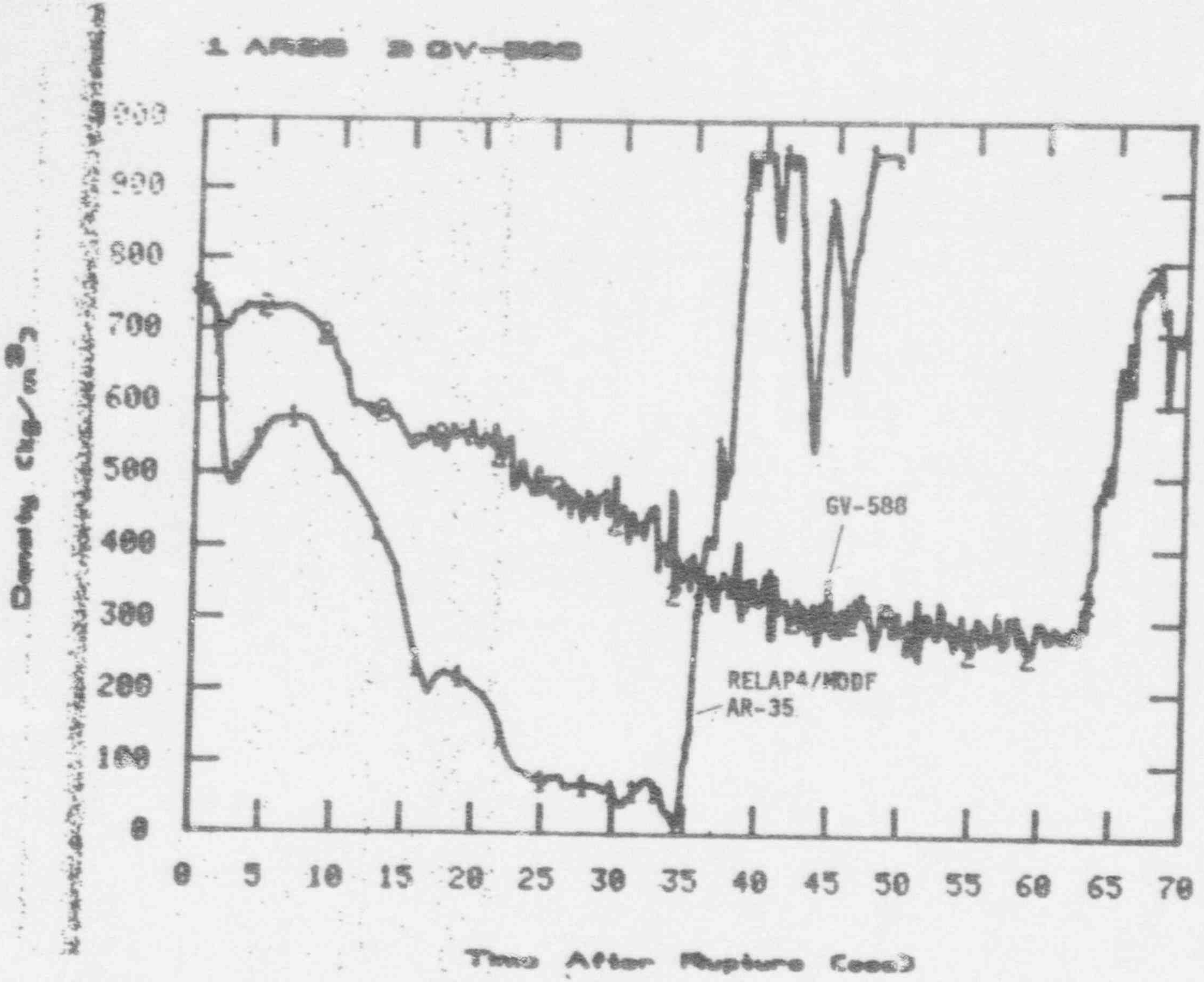

Fig. 32 Comparison of density in the lower plenum for Test S-07-3 data and the RELAP4/MODF pretest calculation.

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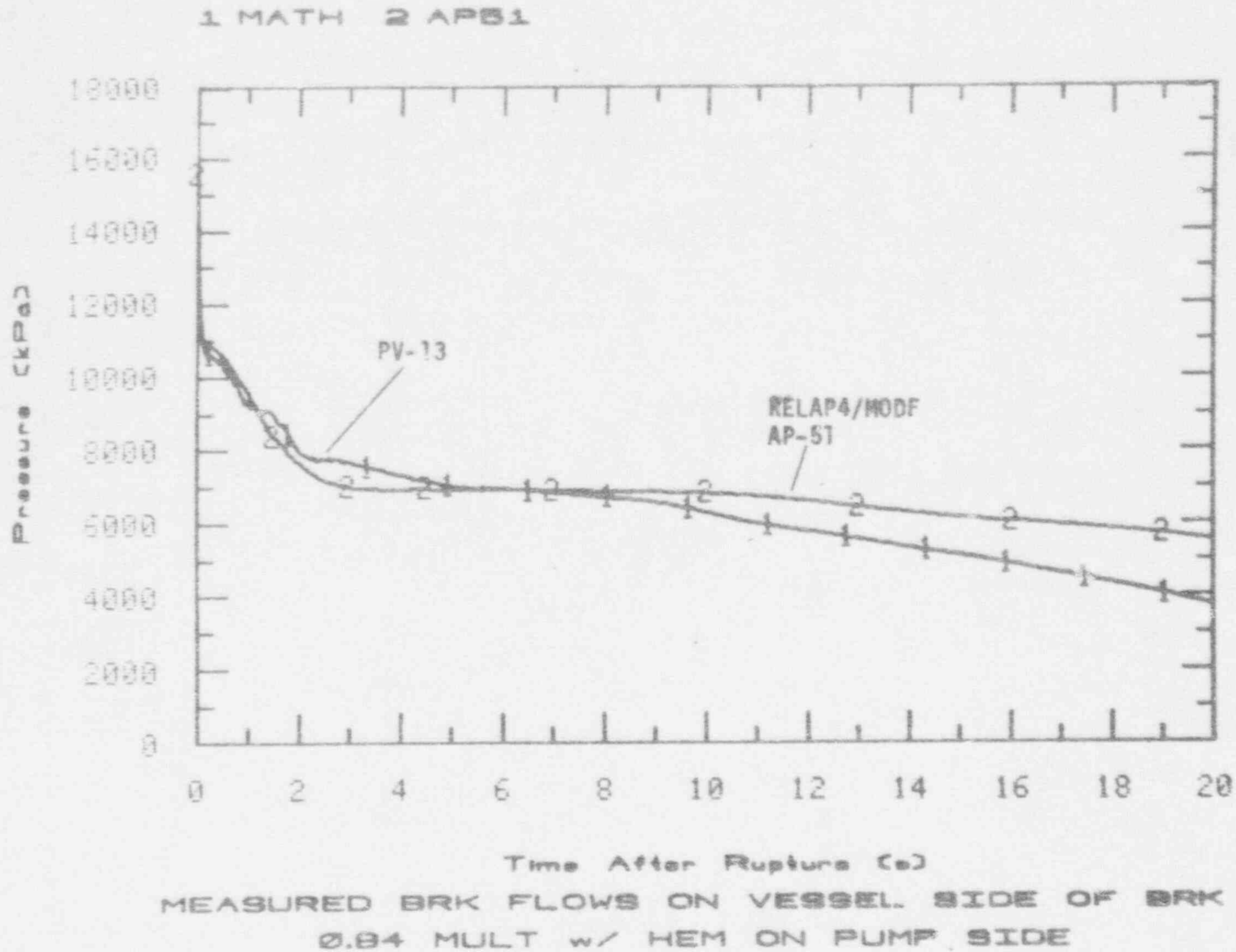

Fig. 33 Comparison of system depressuration rate for Test S-07-3 data and the RELAP4/MODF posttest calculation.