QUICK LOOK REPORT ON

SEMISCALE MOD-1 TEST S-28-5

STEAM GENERATOR TUBE RUPTURE TESTS

SEMISCALE PROGRAM

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STEAM GENERATOR TUBE RUPTURE TESTS

Author: J. M. Cozzuol

Analysts: G. G. Loomis
O. M. Hanner
M. King

SEMISCALE PROGRAM

Approved: D J Olson

D. J. Olson, Manager
Semiscale Program

Approved: D J Hanson

D. J. Hanson, Manager
Semiscale Experiment Specification & Analysis Branch

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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Summary

This report presents a preliminary evaluation of the results from Semiscale Mod-1 Test S-28-5. This integral blowdown-reflood test was conducted with a break configuration representative of a 200% double-ended offset shear cold leg break, and included the simulation of steam generator tube ruptures at the initiation of vessel refill. Test S-28-5 is part of a series of tests (designated the steam generator tube rupture test series) designed to evaluate the effect of a steam generator secondary-to-primary flow on the system and core thermal-hydraulic response during a large break loss-of-coolant experiment. Previous tests in the steam generator tube rupture test series established upper and lower limits on the range of steam generator tube rupture flow rates for which high rod cladding temperatures could occur. The primary purpose of Test S-28-5 was to provide experimental results of the effect of a steam generator tube rupture flow rate within the range of tube rupture flow rates bounded by the previous tests.

The test conditions for Tests S-28-5 were essentially the same as those of the Series 28 baseline test (Test S-04-6), except for the introduction of the secondary-to-primary mass flow to simulate the steam generator tube ruptures. The tube rupture flow was simulated by a controlled injection from a heated accumulator tank into the intact loop hot leg between the steam generator inlet plenum and the pressurizer. The steam generator secondary-to-primary flow for Test S-28-5 simulated the flow from the single-ended rupture of a total of 20 tubes in 3 of the 4 steam generators in a four-loop PWR. The steam generator tube rupture flow was begun at the initiation of vessel refill (at about 40 seconds after rupture). The water in the heated accumulator tank was maintained at about 547°K (approximately the average temperature of the PWR steam generator secondary fluid at rated load) and 7584 kPa. During the period of tube rupture flow, the heat transfer potential of the intact loop steam generator was simulated by discharging (to atmosphere) the steam generator secondary fluid at a rate equivalent to the tube rupture injection rate.

During the period of steam generator tube rupture injection for Test S-28-5, the secondary-to-primary flow had a significant influence on the hydraulic response of the overall system and core. The relatively strong reverse core flow (and corresponding countercurrent flow in the vessel downcomer) during the early portion of the tube rupture flow period was of sufficient magnitude to prevent ECC penetration of the vessel downcomer until after the initiation of nitrogen flow from the intact loop accumulator. The effect of the accumulator nitrogen injection was to force the initiation of downcomer refill at about 67 seconds after rupture. However, because of the relatively large secondary-to-primary flow, reflood of the core was not initiated until after the downcomer liquid level had increased

sufficiently to allow the downcomer head to overcome the effect of the tube rupture flow. As a result, core reflood was not initiated until about 290 seconds after rupture.

Because of the considerable delay in the initiation of core reflood and the relatively poor cooling in the core during the period of downcomer and lower plenum refill, the peak cladding temperatures obtained during Test S-28-5 were considerably higher than the temperatures in the baseline test (Test S-04-6) or in previous steam generator tube rupture tests with relatively large secondary-to-primary flow rates. The core maximum cladding temperature during reflood for Test S-28-5 was about 1208°K and occurred at the 0.66 meter elevation on Rod F5 at about 315 seconds after rupture. The primary effect of the steam generator tube rupture flow on the core quench behavior was to considerably lengthen the time required for quenching to occur. Since the secondary-to-primary flow rate in Test S-28-5 was not of sufficient magnitude to cause a top-down quench of the core, quenching at most locations in the core did not occur until after the initiation of core reflood. However, several cladding thermocouples in the upper portion of the core on rods near the intact loop hot leg side of the vessel did exhibit early quenching (prior to the initiation of core reflood) as a result of improved cooling from the steam generator secondary fluid entering the vessel upper plenum. The entire core was quenched by about 720 seconds after rupture.

Introduction

As part of the overall Semiscale blowdown and emergency core cooling project conducted by EG&G Idaho, Inc., the Semiscale Mod-1 experimental program is used to investigate the thermal and hydraulic phenomena accompanying a hypothesized loss-of-coolant accident (LOCA) in a water-cooled nuclear reactor system. The general objective of the Semiscale Program is to obtain representative integral and separate effects thermal-hydraulic response data to provide an experimental basis for analytical model development and verification.

The purpose of Test Series 28 (designated the steam generator tube rupture test series) is to investigate the influence of the rupture of steam generator tubes on the core and system response during a hypothetical large break loss-of-coolant accident (LOCA). Data from Test Series 28 will be used to determine the sensitivity of core peak cladding temperatures to the magnitude of the flow rate from the secondary side of the steam generator to the primary system. The data will also be used to evaluate the capability of current models to predict the thermal-hydraulic phenomena that are expected to occur during the refill and reflood phases of a LOCA with steam generator tube ruptures.

This document contains a preliminary analysis of the results obtained from the fifth test (designated Test S-28-5) of the steam generator tube rupture test series conducted in the Semiscale Mod-1 system. Test S-28-5 was conducted with a break configuration representative of a 200% double-ended offset shear cold leg break. The secondary-to-primary flow due to the rupture of steam generator tubes was simulated by injection of fluid at a temperature typical of a PWR steam generator secondary into the intact loop hot leg between the steam generator inlet plenum and the pressurizer. The injection was accomplished using a pressurized water source. For Test S-28-5 fluid was injected at a rate of approximately 0.181 kg/s to simulate flow from the single-ended rupture of 20 tubes[a] in a PWR steam generator. The steam generator tube rupture flow for the test was begun at about 40 seconds[b] after the initiation of the cold

[a] To provide a basis for comparing the relative magnitudes of the tube rupture mass flow rates in the Semiscale Mod-1 system and a PWR, the magnitudes of the steam generator secondary-to-primary mass flow rates in the Mod-1 system are presented in terms of the flow rates associated with a given number of single-ended tube ruptures in a PWR steam generator. The secondary-to-primary mass flow rates used in the Semiscale Mod-1 system for the tube rupture test series are core area scaled to a PWR.

[b] Forty seconds is approximately the time at which vessel refill would have begun in the Semiscale Mod-1 system if the steam generator tube rupture injection had not occurred.

leg break, and continued until about 640 seconds. The change in heat transfer potential of the steam generator was simulated by discharging the steam generator secondary fluid at a rate equivalent to the rate of the tube rupture flow. The system initial conditions and emergency core coolant (ECC) injection parameters for Test S-28-5 were essentially the same as for the baseline test for Series 28, Test S-04-6 (Reference 1).

To assist in understanding the data presented in this report, Figure 1 provides an isometric view of the Mod-1 system together with the general location of the instrumentation. The Semiscale Mod-1 system configuration and the instrumentation for Test Series 28 are described in Reference 2.

The powered heater rod configuration for Test S-28-5 was identical to that for Test S-04-6. Figure 2 shows the core heater rod arrangement and includes the location of unpowered and high-powered heater rods. Thirty-six of the 40 heater rods were powered in each test. Four rods (Rods C3, F3, D5 and F6) were unpowered to make the core bundle more representative of a PWR fuel assembly containing control rod thimbles and instrument tubes. The three center rods (Rods D4, E4, and E5) were operated at a 5% higher peak power density than the remaining 33 powered rods to simulate the radial power profile near a control rod thimble in a PWR fuel assembly. Figure 3 shows the Mod-1 heater rod normalized axial power profile. The low power heater rods had a peak power density of about 37.7 kW/m, whereas the three center rods had a peak power density of about 39.7 kW/m. The initial portion of the core power decay curves for Test S-28-5 is shown in Figure 4. For comparison purposes, the core power decay curve for Test S-04-6 is included in the figure.

The specified initial conditions and operational variables together with the actual test conditions for Test S-28-5 are listed in Table I. Initial prerupture conditions for the test were compared with the specified prerupture conditions and the differences were judged to be minor. These minor differences in conditions prior to rupture for Test S-28-5 did not significantly influence the postrupture system behavior.

A pretest prediction of the thermal-hydraulic response for Test S-28-5 (Reference 3) was obtained using Test S-04-6 data and the FLOOD4 computer code. The system response during the first 40 seconds of Test S-28-5 was expected to be the same as the system response in Test S-04-6. Therefore, Test S-04-6 data was used to provide the initial conditions for the FLOOD4 calculations starting at 40 seconds after rupture. The FLOOD4 code was then used to provide the predictions through the period of the steam generator tube rupture injection until core quenching occurred. The analysis technique used in the test predictions is described in Reference 3. A discussion of results from the test predictions is included in this report.

TABLE I

TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-5

Primary System	Specified Condition	Test Condition
Core Power (AMPCOR-T) (VOLTCOR-T) (MW)	1.44	1.46
System pressure (PV+10) (kPa, gage)	15513 ± 172	15685
Loop temperature		
Intact loop cold leg (RBU-14 (°K)	557.8 ± 1	556
Intact loop hot leg (RBU-2) (°K)	594.4 ± 1	593
Broken loop hot leg (TFB-30) (°K)	591.7 ± 3	594
Core flow rate (FTV-COREIN) L/min	As required to obtain core ΔT	545
Pressure suppression system		
Tank water temperature (TF-PSS-33) (°K)	Ambient	291
Tank water pressure (P-PSS) (kPa, gage)	155 ± 7	148
Pressurizer water (DPU-PRESLL) (kg)	9.07	9.8
Steam generator feedwater temperature (TFU-SGFW) (°K)	497 ± 6	488
Steam generator secondary liquid level (DPU-SG-SEC) (cm)	295 ± 5	284

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TABLE I (contd)

TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-5

ECC System	Specified Condition	Test Condition
Accumulator CI-T-1		
Injection location	Intact loop cold leg (Spool piece 14)	
Actuation Pressure (PU-ACC1) (kPa, gage)	4137	4744
Liquid volume (L)	80.1	87
Injection rate (FTU-ACC-1) (L/min)	87	98
N_2 flow duration (sec)	24	30
Accumulator CI-T-2		
Location	Broken loop cold leg (Spool piece 42)	
Actuation Pressure (PB-ACC2) (kPa, gage)	4137	4285
Liquid volume (L)	16.4	12
Injection rate (FTB-ACC2) (L/min)	28.65	23
Accumulator CI-T-3 (Steam generator secondary simulator)		
Injection location	Intact loop hot leg (Spool piece 6)	
Temperature (TFU-SGS3-B) (°K)	547	544
Initial pressure (PU-SG3-T) (kPa, gage)	7584	7978
Liquid volume (L)	144.4	144
Injection rate (FTU-SGS-H) (L/min)	14.3 ± 0.7	14
Air actuated valve		
Open (seconds after rupture)	40	40
Close (seconds after rupture)	Remains open	Remains ope

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TABLE I (contd)

TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-5

ECC System	Specified Condition	Test Condition
Steam generator secondary fluid discharge		
Initial liquid level (cm)	295	284
Flow rate (L/min)	14.3 ± 0.7	N/A
Air actuated valve		
Opening time (sec)	40	40
Closing time (sec)	Remains open	Remains open
Intact loop LPIS		
Location	Cold leg (Spool piece 14)	
Actuation Pressure (kPa, gage)	1034	958
Injection rate (FTU-LPIS) (L/min)	15.1	17
Broken loop LPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	1034	908
Injection rate (FTB-LPIS) (L/min)	3.6	4.2
Intact loop HPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	12411	15929
Injection rate (FTU-HPIS) (L/min)	1.17	1.3
Broken loop HPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	12411	15688
Injection rate (FTB-HPIS) (L/min)	0.38	1.1

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Test Results

The primary objective of Test S-28-5 was to determine the effect on the system and core thermal-hydraulic response during a large break loss-of-coolant experiment of a simulated steam generator tube rupture flow rate equivalent to the flow associated with the single-ended rupture of 20 tubes in a PWR steam generator. The magnitude of the secondary-to-primary flow rate in Test S-28-5 was in the upper portion of the range of tube rupture flow rates for which high rod cladding temperatures could occur[a]. A preliminary analysis of the data for Test S-28-5 has been performed to evaluate the effect of the steam generator secondary-to-primary flow on the thermal-hydraulic response of the system and core. Results of the analysis are presented in the following three sections. The first section deals with the effect of the secondary-to-primary flow on the overall system and core hydraulic response. Special emphasis is placed on those aspects of the system and core hydraulic response which had a significant influence on the core thermal behavior. The second section is primarily concerned with the core thermal response following the initiation of the steam generator tube rupture flow. Where applicable, results from Test S-28-5 are compared with data from the Series 28 baseline test, Test S-04-6. In addition, to provide an evaluation of the effect on the system and core response of different secondary-to-primary flow rates, results from Test S-28-5 are compared with the previous steam generator tube rupture tests, Test S-28-1 and Test S-28-4 (References 4 and 5). A final section presents comparisons of data for Test S-28-5 with selected results from the pretest calculations.

System and Core Hydraulic Response During Test S-28-5

The steam generator tube rupture flow for Test S-28-5 was initiated at the beginning of vessel refill (about 40 seconds after rupture) and continued for the duration of the test. During the period of injection, the steam generator secondary-to-primary flow had a significant influence

[a] The scaling analysis for Test Series 28 indicated that high rod cladding temperatures could occur in the Semiscale Mod-1 core for secondary-to-primary flow rates of between 0.09 and 0.54 kg/s. This range of tube rupture flow rates in the Semiscale Mod-1 system is equivalent to the flow associated with the single-ended rupture of a total of between 10 and 60 tubes in 3 or 4 steam generators in a 4-loop PWR. (For comparison, note that the 4-loop Trojan PWR has approximately 3300 tubes in each of the 4 steam generators). For secondary-to-primary flow rates in the Semiscale system equivalent to the flow from less than 10 or more than 60 tube ruptures in a PWR steam generator the analysis indicates good core cooling exists during the injection period, thus preventing high cladding temperatures.

on the hydraulic response of the overall system and core. The effects of the tube rupture flow on the system and core hydraulic response are illustrated by comparing volumetric flow rates at various points in the system for Test S-28-5 with the corresponding volumetric flow rates for the baseline test, Test S-04-6. Figures 5 and 6 compare the volumetric flow rates in the intact loop hot leg near the vessel and at the entrance to the core respectively, for the two tests. As indicated in these figures, the intact loop hot leg flow and core inlet flow for Test S-04-6 became positive at about 40 seconds after rupture indicating that vessel refill had begun. For Test S-28-5, however, the reverse flow through the intact loop hot leg and downward through the core became substantially more negative at the initiation of the secondary-to-primary flow (at 40 seconds after rupture). The relatively strong reverse core flow continued until the initiation of nitrogen flow from the intact loop accumulator at about 64 seconds after rupture. The accumulator nitrogen injection into the intact loop cold leg resulted in a reduced core inlet volumetric flow rate (Figure 6) which continued for the duration of the nitrogen flow (until about 90 seconds after rupture).

Once the intact loop accumulator nitrogen flow ceased, the magnitude of the core flow did not increase to the value attained prior to the nitrogen injection even though the secondary-to-primary flow continued at essentially the same rate. The considerably smaller reverse core flow after the period of nitrogen injection can be attributed to a change in the resistance to steam flow in the intact loop cold leg. During the period of accumulator liquid injection into the intact loop cold leg, steam flow from the hot leg side to the cold leg side of the intact loop was limited by the rate of condensation of the steam near the ECC accumulator injection location. The effect of the accumulator nitrogen flow (following the depletion of the accumulator liquid inventory), however, was to clear the remaining liquid from the intact loop cold leg. This phenomenon is illustrated in Figure 7 which shows the fluid density in the intact loop cold leg. Once the liquid had been cleared from the intact loop cold leg and the nitrogen flow had ceased, an additional path for removal of steam from the intact loop occurred. As a result, the magnitude of the core inlet volumetric flow did not increase significantly, but remained at a small negative value until the initiation of core reflood.

The relatively strong reverse core flow (and corresponding countercurrent flow in the vessel downcomer) during the early portion of the tube rupture flow period for Test S-28-5 prevented ECC penetration of the vessel downcomer until after the initiation of nitrogen flow from the intact loop accumulator. The effect of the accumulator nitrogen injection was to force the initiation of downcomer refill at about 67 seconds after rupture. Figure 8 compares the downcomer collapsed liquid levels obtained from a downcomer differential pressure measurement for Tests S-28-5 and S-04-6. As indicated in the figure the downcomer collapsed liquid level for Test S-28-5 did not begin to increase until shortly after the initiation of accumulator nitrogen injection, whereas for Test S-04-6, the

downcomer was essentially full prior to the initiation of nitrogen flow (at about 68 seconds). The considerable increase in the downcomer collapsed liquid level at about 67 seconds for Test S-28-5 was due primarily to penetration of the downcomer by ECC fluid stored in the vessel inlet annulus and upper portion of the downcomer during the downcomer countercurrent phase of the test. The relatively slow refill of the downcomer for Test S-28-5 (as well as for Test S-04-6) after about 100 seconds was accomplished by the intact loop LPIS.

Although the initiation of vessel refill for Test S-28-5 began at about 67 seconds after rupture, the reverse flow through the core (caused by the tube rupture flow) was of sufficient magnitude to prevent the initiation of core reflood until considerably later in the test. Figure 9 compares the collapsed liquid levels obtained from a lower plenum-to-upper plenum differential pressure measurement for Tests S-28-5 and S-04-6. As indicated in the figure, only a partial refill of the vessel lower plenum occurred for Test S-28-5 prior to about 250 seconds. However, because of the increasing liquid level in the downcomer (Figure 8) the downcomer head increased sufficiently by 250 seconds after rupture to overcome the effect of the tube rupture flow. As a result, refill of the lower plenum after about 250 seconds occurred at a considerably increased rate, and reflood of the core was initiated at about 290 seconds after rupture. The initiation of core reflood is indicated in Figure 10 which compares the core inlet fluid density for Tests S-28-5 and S-04-6.

Core Thermal Response During Test S-28-5

The effect of the magnitude of the steam generator tube rupture flow on the core thermal response for Test S-28-5 is illustrated by a comparison of rod cladding temperatures at several elevations in the core with cladding temperatures obtained from the baseline test, Test S-04-6, and from previous steam generator tube rupture tests, Tests S-28-1 and S-28-4 for which the secondary-to-primary low rates were 0.540 kg/s and 0.271 kg/s, respectively[a]. Figures 11 through 14 compare typical rod cladding temperatures at the 0.36, 0.51, 0.69, and 0.84 m elevation in the core for the four tests[b]. As indicated in the figures, the steam generator

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[a] The steam generator tube rupture flow rates for Tests S-28-1 and S-28-4 are equivalent to the flow rates associated with the single-ended rupture of 60 tubes and 30 tubes, respectively, in a PWR steam generator.

[b] Note that the peak cladding temperatures during blowdown for Test S-28-1 are considerably higher than for the other tests. This difference is attributed to a delay in the initiation of the core power decay of about 1 second for Test S-28-1.

secondary-to-primary flow for Test S-28-5 resulted in significantly higher rod cladding temperatures than were observed in the baseline test, or in Tests S-28-1 and S-28-4. In Tests S-28-1 and S-28-4, the tube rupture flow rates were of sufficient magnitude to maintain a strong reverse core flow comprised of a relatively low quality liquid-vapor mixture during the entire tube rupture flow period. As a result, good core cooling existed during the tube rupture injection period, and the maximum core cladding temperatures following the initiation of the tube rupture flow were considerably lower than were observed during the blowdown phase of the tests. In Test S-28-5, however, the secondary-to-primary flow was not of sufficient magnitude to maintain a strong flow downward through the core once refill of the downcomer had been initiated. Thus a prolonged period of rod heatup occurred which led to considerably higher peak cladding temperatures than were observed in Tests S-28-1 and S-28-5. For comparison purposes, Table II lists the peak cladding temperatures and the times after rupture at which the peak cladding temperatures occurred for each thermocouple location monitored during Tests S-28-1, S-28-4, S-28-5, and S-04-6. As indicated in the table, the peak cladding temperature for Test S-28-5 was about 1208°K and occurred at 315 seconds after rupture at the 0.66 meter elevation of Rod F5.

A comparison of the rod quench times for Tests S-28-1, S-28-4, S-28-5 and S-04-6 further illustrates the effects of the rate of the steam generator secondary-to-primary flow on the core thermal response. Table III lists the quench times for all thermocouple locations monitored for the four tests. As indicated in the table, Test S-28-5 exhibited a bottom-up quench behavior for most locations in the core[a] but at a significantly slower rate than was observed in the baseline test (Test S-04-6). The last observed quench for Test S-28-5 occurred at about 720 seconds, approximately 410 seconds after the initiation of core reflood. The bottom-up quench behavior for Test S-28-5 caused by the bottom reflood of the core is considerably different than the top-down quench behavior for Tests S-28-1 and S-28-4 which resulted in the entire core being quenched prior to the initiation of core reflood.

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[a] The small secondary-to-primary flow rate for Test S-28-5 resulted in a preferential flow path through the core on the side adjacent to the intact loop hot leg. As indicated in Table III, the rods adjacent to the intact loop hot leg (eg, rods A4, A5, B5, B6, etc.) exhibit early quench times. This may be attributed to the fact that the rods in the Semiscale Mod-1 system extend through the upper plenum. Apparently, due to the relatively small secondary-to-primary flow, the rods on the intact loop hot leg side of the vessel prevent the liquid portion of the tube rupture flow from penetrating to the opposite side of the vessel. As a result, the cooling of the rods on the side of the vessel adjacent to the intact loop hot leg is better than on the side of the vessel opposite to the intact loop hot leg.

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TABLE II

Maximum Core Cladding Temperatures and Times at Which Maximum Temperatures Occur for Tests S-28-1, S-28-4, S-28-5, and S-04-6

Thermocouple I. D.	Test S-28-5 °K (sec)	Test S-28-1 °K (sec)	Test S-28-4 °K (sec)	Test S-04-6 °K (sec)
E3-05	886 (284)	730 (39)	743 (124)	710 (44)
C7-07	894 (186)	864 (39)	858 (94)	840 (44)
F2-07	910 (250)	767 (39)	793 (125)	765 (52)
E6-08	1013 (278)	822 (10)	837 (109)	808 (53)
A4-09	874 (10)	928 (10)	882 (9)	885 (9)
E4-09	1028 (330)	807 (39)	801 (107)	863 (10)
G3-13	1061 (324)	904 (10)	850 (109)	890 (64)
D2-14	1022 (329)	965 (10)	838 (105)	902 (64)
D4-14	1036 (283)	899 (10)	817 (10)	860 (64)
E8-14	1028 (289)	975 (9)	916 (109)	935 (64)
F4-14	1137 (341)	881 (10)	844 (109)	885 (64)
G5-14	1124 (316)	908 (10)	880 (109)	921 (58)
C7-15	1013 (326)	955 (10)	902 (39)	943 (64)
C4-20	874 (154)	913 (10)	831 (9)	NA
D7-20	1085 (333)	1010 (10)	896 (39)	966 (9)
E2-20	1103 (341)	924 (10)	NA	891 (58)
E3-20	NA	910 (10)	834 (9)	846 (54)
E5-20	1155 (356)	951 (10)	868 (39)	903 (54)
F5-20	1166 (321)	906 (10)	834 (109)	888 (54)
D1-21	1053 (8)	1125 (10)	1050 (8)	1066 (9)
F2-22	1005 (206)	1003 (10)	908 (8)	NA
E3-24	1117 (339)	1033 (45)	875 (9)	934 (64)
G5-24	1185 (273)	970 (10)	913 (109)	982 (62)
D6-25	1113 (332)	971 (10)	899 (39)	959 (64)
F2-25	1067 (300)	969 (10)	875 (9)	936 (64)
E5-25	1162 (354)	968 (10)	880 (10)	959 (64)
C4-26	920 (183)	980 (10)	879 (10)	NA
D8-26	994 (268)	872 (9)	843 (40)	NA
F5-26	1208 (315)	963 (10)	884 (10)	NA
E4-27	1160 (356)	938 (10)	841 (9)	934 (71)
C5-28	946 (170)	943 (10)	951 (10)	948 (71)
E6-28	1177 (330)	945 (10)	887 (109)	940 (64)
A4-29	944 (10)	1058 (10)	933 (10)	NA
A5-29	989 (9)	1123 (9)	914 (8)	1075 (9)
B5-29	923 (108)	971 (10)	890 (40)	NA
B6-29	973 (8)	1064 (9)	980 (8)	1048 (8)
D3-29	1045 (331)	964 (10)	863 (9)	918 (71)
D4-29	1046 (359)	939 (10)	840 (10)	914 (71)

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TABLE II (contd)

Thermocouple I. D.	Test S-28-5 °K (sec)	Test S-28-1 °K (sec)	Test S-28-4 °K (sec)	Test S-04- °K (sec)
E8-29	1105 (249)	999 (10)	930 (39)	997 (71)
F4-29	1182 (318)	940 (10)	868 (9)	940 (71)
G4-29	1152 (259)	953 (10)	869 (9)	968 (71)
B3-32	809 (162)	875 (10)	700 (10)	819 (9)
H5-32	1053 (240)	906 (9)	818 (110)	946 (71)
B5-33	826 (10)	871 (10)	828 (10)	856 (84)
E1-33	976 (280)	907 (10)	821 (8)	819 (162)
E2-33	1075 (299)	926 (10)	838 (8)	884 (182)
F5-33	1168 (330)	863 (9)	819 (9)	916 (71)
G4-33	1104 (310)	894 (10)	821 (9)	921 (70)
E6-37	1095 (381)	780 (10)	769 (109)	810 (60)
C2-38	834 (332)	779 (11)	686 (100)	691 (72)
G4-38	995 (311)	748 (10)	735 (130)	853 (64)
A4-39	750 (10)	808 (10)	711 (10)	NA
D3-39	993 (356)	791 (10)	688 (99)	715 (75)
E7-44	962 (435)	656 (0)	722 (129)	768 (189)
F4-44	1017 (445)	675 (10)	670 (109)	703 (84)
A5-45	669 (0)	677 (13)	669 (0)	692 (8)
C4-53	617 (0)	618 (0)	618 (0)	619 (0)
C6-53	687 (0)	638 (0)	638 (0)	638 (0)
F5-53	947 (457)	613 (0)	612 (0)	609 (0)
E4-55	881 (639)	631 (0)	631 (0)	633 (0)
D2-61	707 (639)	608 (0)	608 (0)	608 (0)

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TABLE III

Core Quench Times for Tests S-28-1, S-28-4, S-28-5, and S-04-6

Thermocouple I. D.	Test S-28-5 (sec)	Test S-28-1 (sec)	Test S-28-4 (sec)	Test S-04-6 (sec)
E3-05	389	94	275	77
C7-07	397	124	388	86
F2-07	397	147	390	84
E6-08	405	111	389	88
A4-09	218	99	153	95
E4-09	410	91	197	93
G3-13	435	115	407	108
D2-14	435	106	240	109
D4-14	436	78	148	103
E8-14	435	184	411	107
F4-14	437	75	220	106
G5-14	438	100	267	111
C7-15	443	110	281	135
C4-20	236	78	124	NA
D7-20	463	122	348	191
E2-20	NA	121	NA	184
E3-20	466	77	194	97
E5-20	470	77	180	115
F5-20	468	78	221	112
D1-21	470	112	227	140
F2-22	473	145	434	106
E3-24	481	43	181	106
G5-24	483	81	208	208
D6-25	494	83	181	195
F2-25	496	122	444	196
E5-25	488	71	170	184
C4-26	193	67	107	208
D8-26	672	73	182	217
F5-26	501	66	181	202
E4-27	513	67	147	236
C5-28	192	59	110	220
E6-28	513	78	197	201
A4-29	114	69	97	229
A5-29	122	71	110	230
B5-29	126	59	102	236
B6-29	186	74	118	188
D3-29	530	63	128	242
D4-29	536	64	114	227
E8-29	529	131	455	234

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TABLE III (contd)

Thermocouple I. D.	Test S-28-5 (sec)	Test S-28-1 (sec)	Test S-28-4 (sec)	Test S-04-6 (sec)
F4-29	530	58	162	238
G4-29	529	66	175	240
B3-32	170	63	96	98
H5-32	380	72	164	294
B5-33	102	51	90	183
E1-33	664	94	179	184
E2-33	664	82	464	248
F5-33	656	56	160	265
G4-33	656	60	160	277
E6-37	680	63	159	107
C2-28	683	56	106	80
G4-38	680	57	145	269
A4-39	76	50	81	NA
D3-39	680	53	106	85
E7-44	704	73	320	298
F4-44	706	50	118	99
A5-45	69	47	74	99
C4-53	71	45	71	298
C6-53	154	50	85	63
F5-53	716	50	115	68
E4-55	No quench	46	87	71
D2-61	No quench	63	115	78

Comparison of Selected Data to Calculations

A test prediction of the thermal-hydraulic response characteristics for Test S-28-5 was performed using the FLOOD4 computer code. Detailed description of the analysis technique used in the calculations and additional predicted results for Test S-28-5 are given in Reference 3.

Comparisons of the predicted and measured system and core thermal-hydraulic parameters for Tests S-28-5 indicate that considerable differences exist between the predicted and measured results. These differences are apparently due primarily to the assumptions made in the pretest analysis. The assumptions that refill of the downcomer and lower plenum would not begin until after the steam generator tube rupture flow ended (at about 646 seconds after rupture), and that reflood of the core would not begin until about 720 seconds after rupture (based on the time required to fill the lower plenum once refill had been initiated) were not consistent with the refill and reflood response observed in Test S-28-5. As a result, the pretest analysis did not adequately model the thermal-hydraulic phenomena occurring in the vessel and core during Test S-28-5.

Based on the considerable differences in the calculated and measured thermal-hydraulic response for Test S-28-5 it is evident that further development of the pretest analysis technique would be required to properly model the phenomena that occurred in the Semiscale Mod-1 system during steam generator tube rupture phase of the test.

PRELIMINARY

References

(1) H. S. Crapo, B. L. Collins, and K. E. Sackett, "Experiment Data Report for Semiscale Mod-1 Tests S-04-5 and S-04-6 (Baseline ECC Tests)", TREE-NUREG-1045, January 1977.

(2) D. J. Olson Ltr to P. E. Litteneker, DJO-125-77, "Transmittal of Semiscale EOS Appendix 28", June 1977.

(3) D. J. Olson Ltr to R. E. Tiller, DJO-156-77, "Test Prediction of Semiscale Mod-1 Integral Test S-28-5", July 18, 1977.

(4) D. J. Olson Ltr to R. E. Tiller, DJO-151-77, "Transmittal of Quick Look Report for Semiscale Mod-1 Steam Generator Tube Rupture Tests S-28-1 and S-28-2", July 6, 1977.

(5) D. J. Olson Ltr to R. E. Tiller, DJO-162-77, "Transmittal of Quick Look Report for Semiscale Mod-1 Steam Generator Tube Rupture Tests S-28-3 and S-28-4", July 27, 1977.

PRELIMINARY

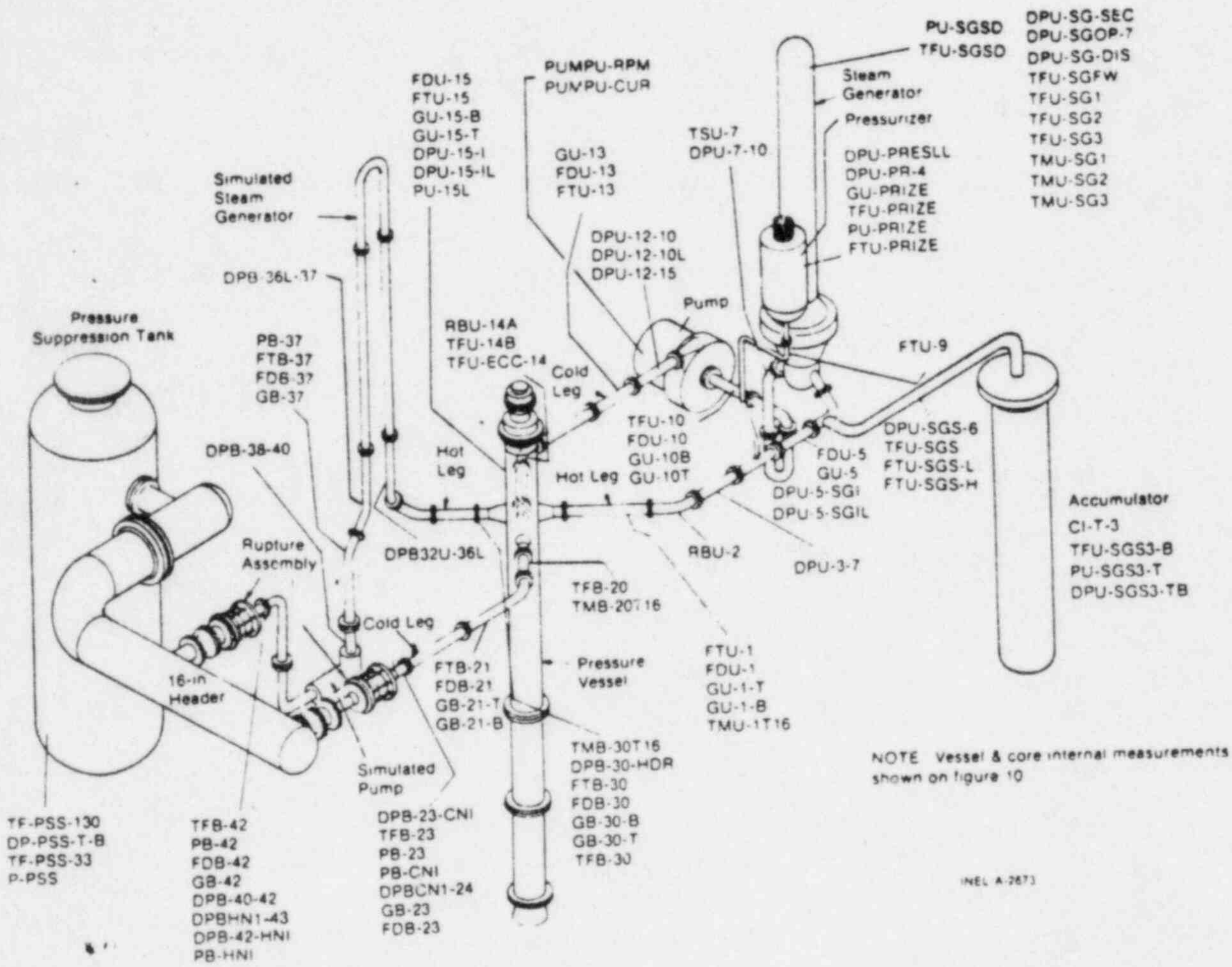

Figure 1. Semiscale Mod-1 System and Instrumentation for Cold Leg Break Configuration - Isometric

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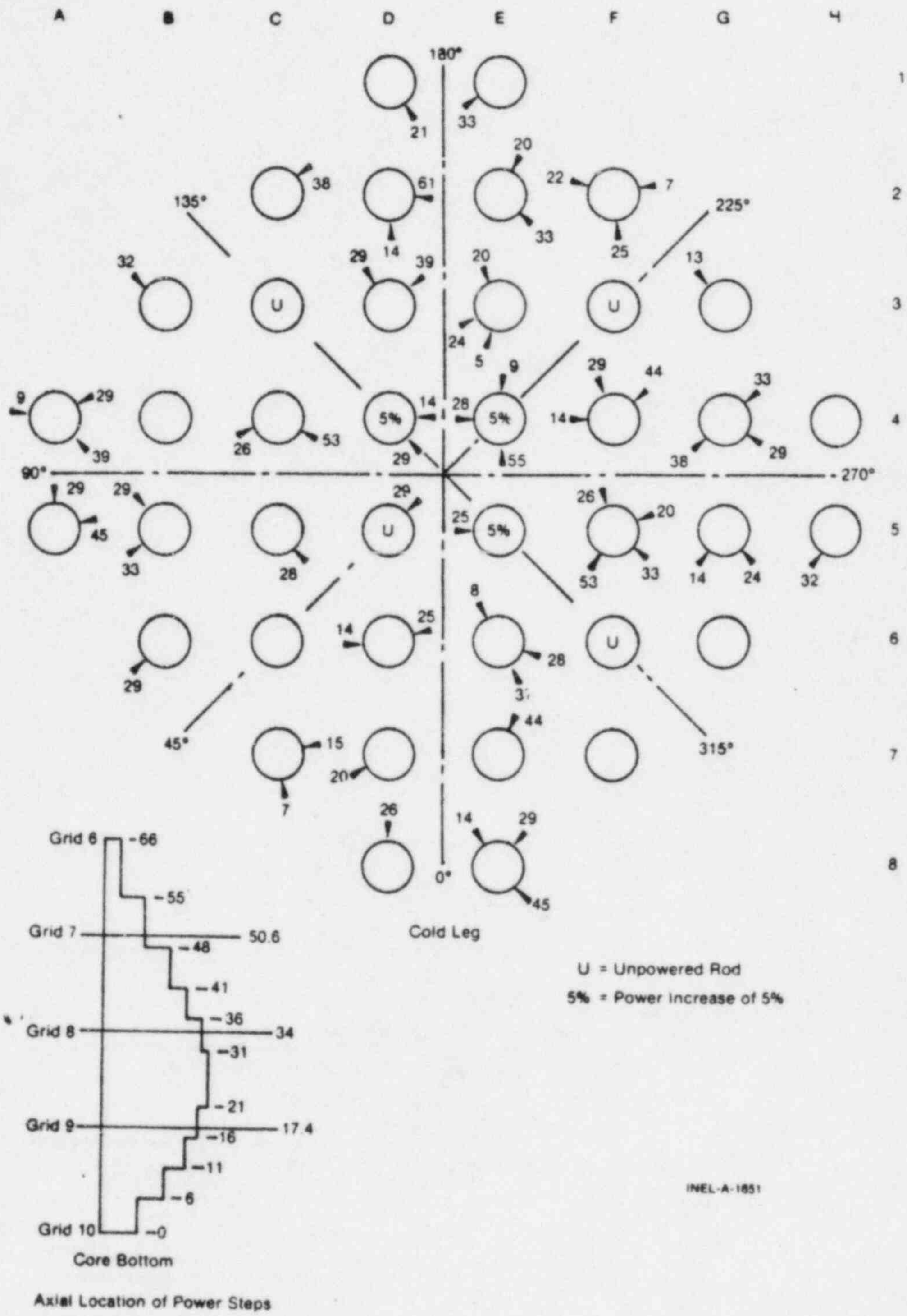

Figure 2. Semiscale Mod-1 Core - Plan View with Thermocouple Locations

PRELIMINARY

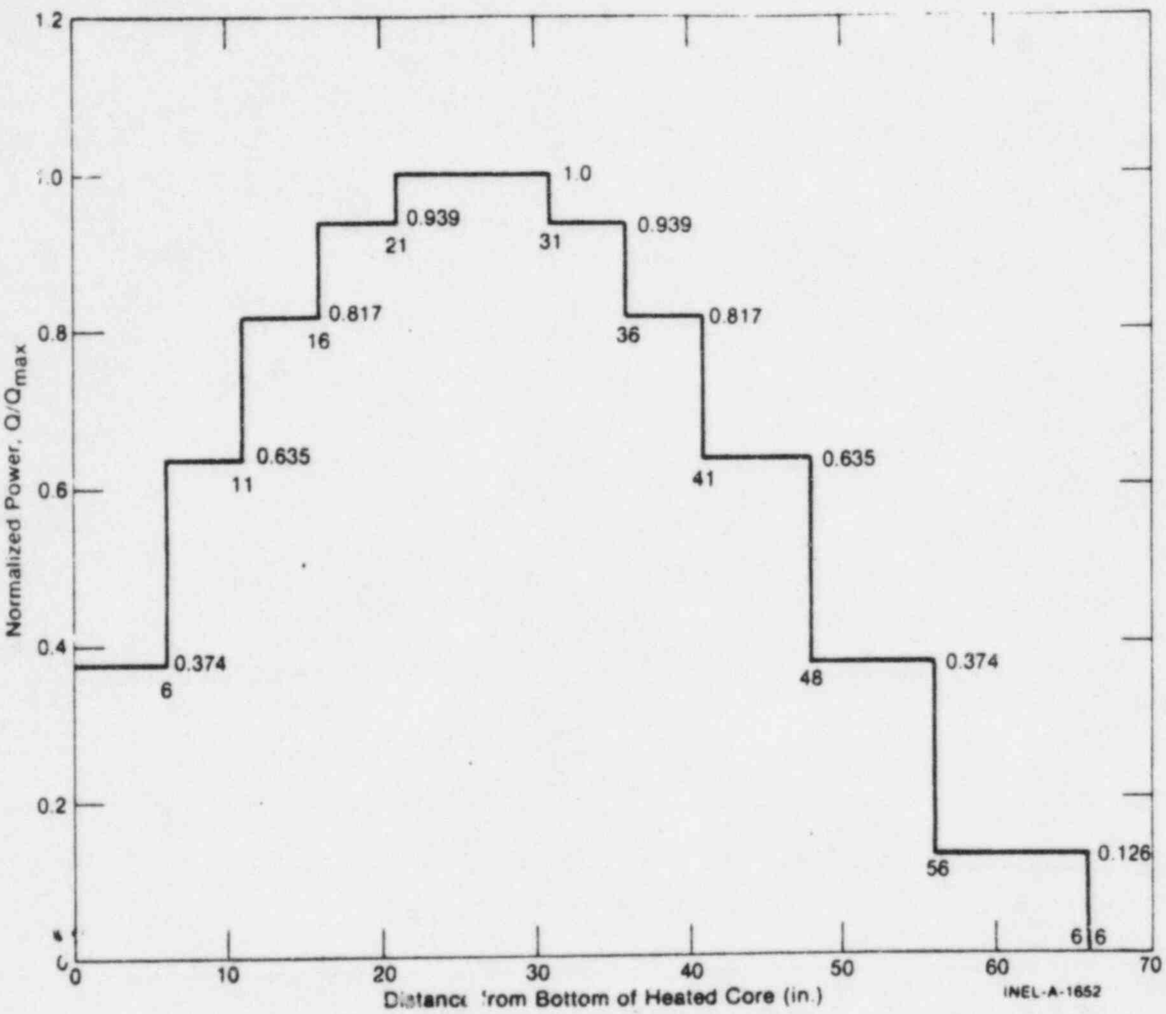

Figure 3. Semiscale Mod-1 Axial Power Profile

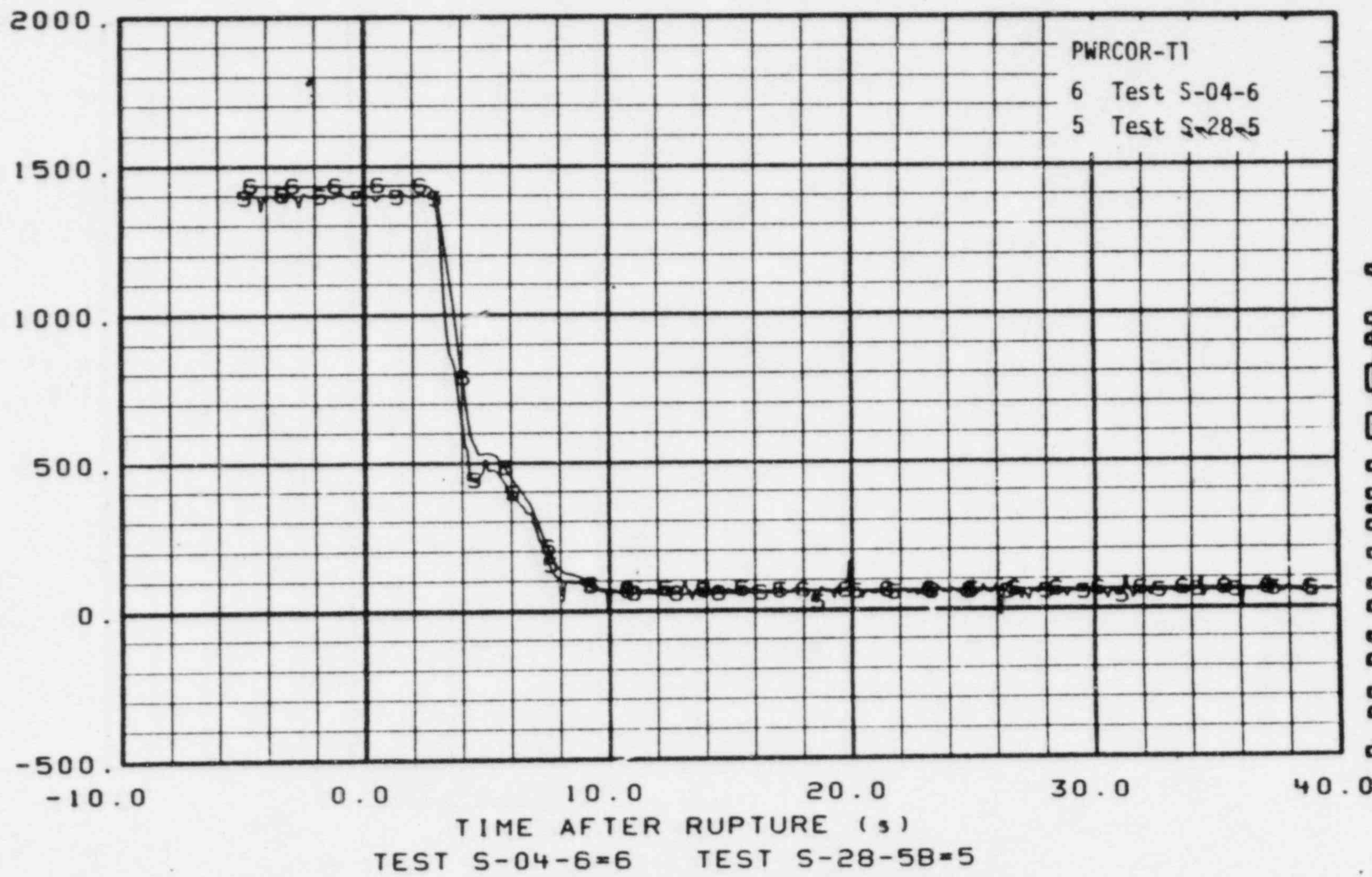

Figure 4. Initial Core Power Decay - Tests S-28-5 and S-04-6

PRELIMINARY

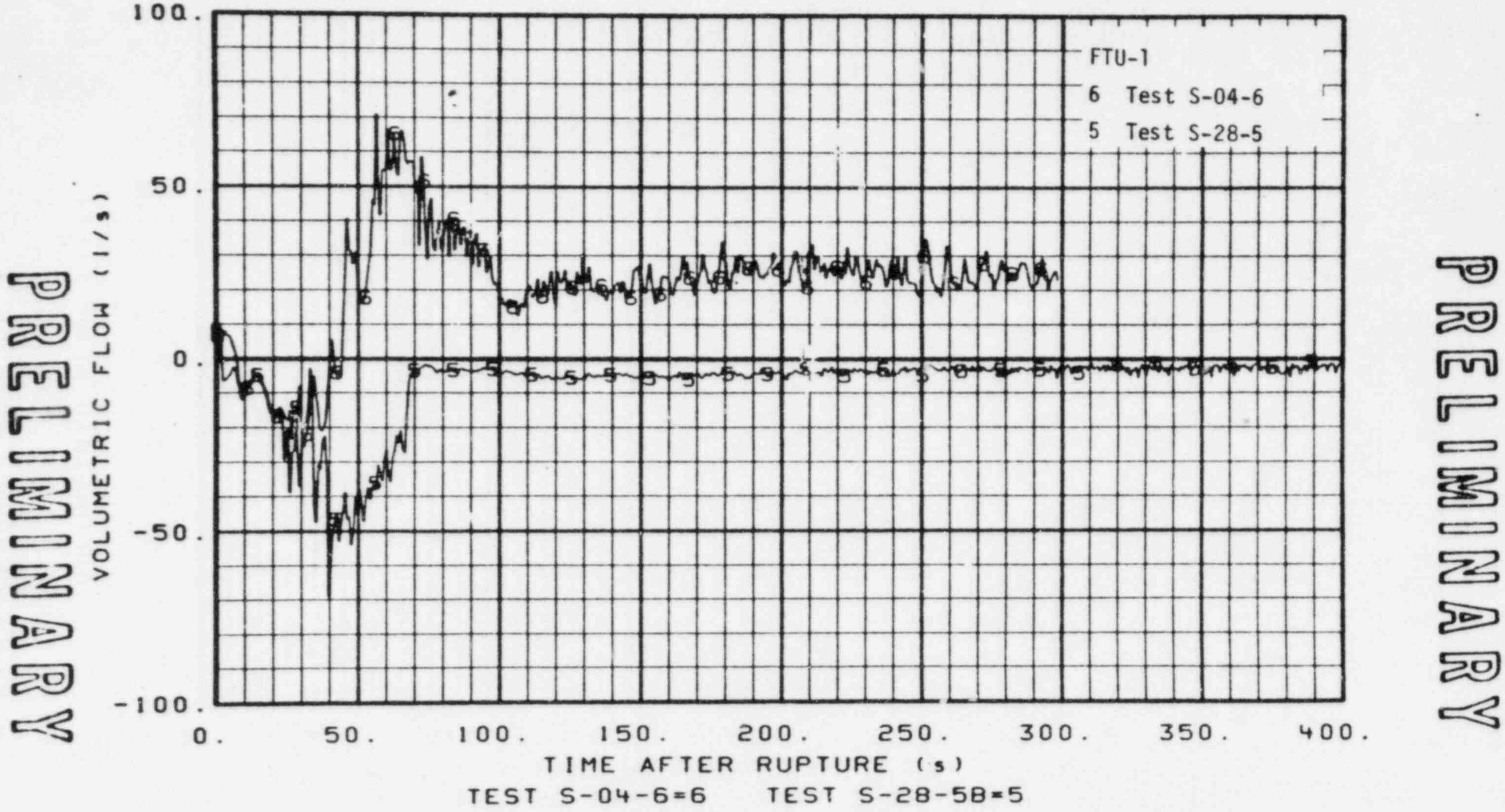

Figure 5. Comparison of Intact Loop Hot Leg Volumetric Flow Near the Vessel - Tests S-28-5 and S-04-6

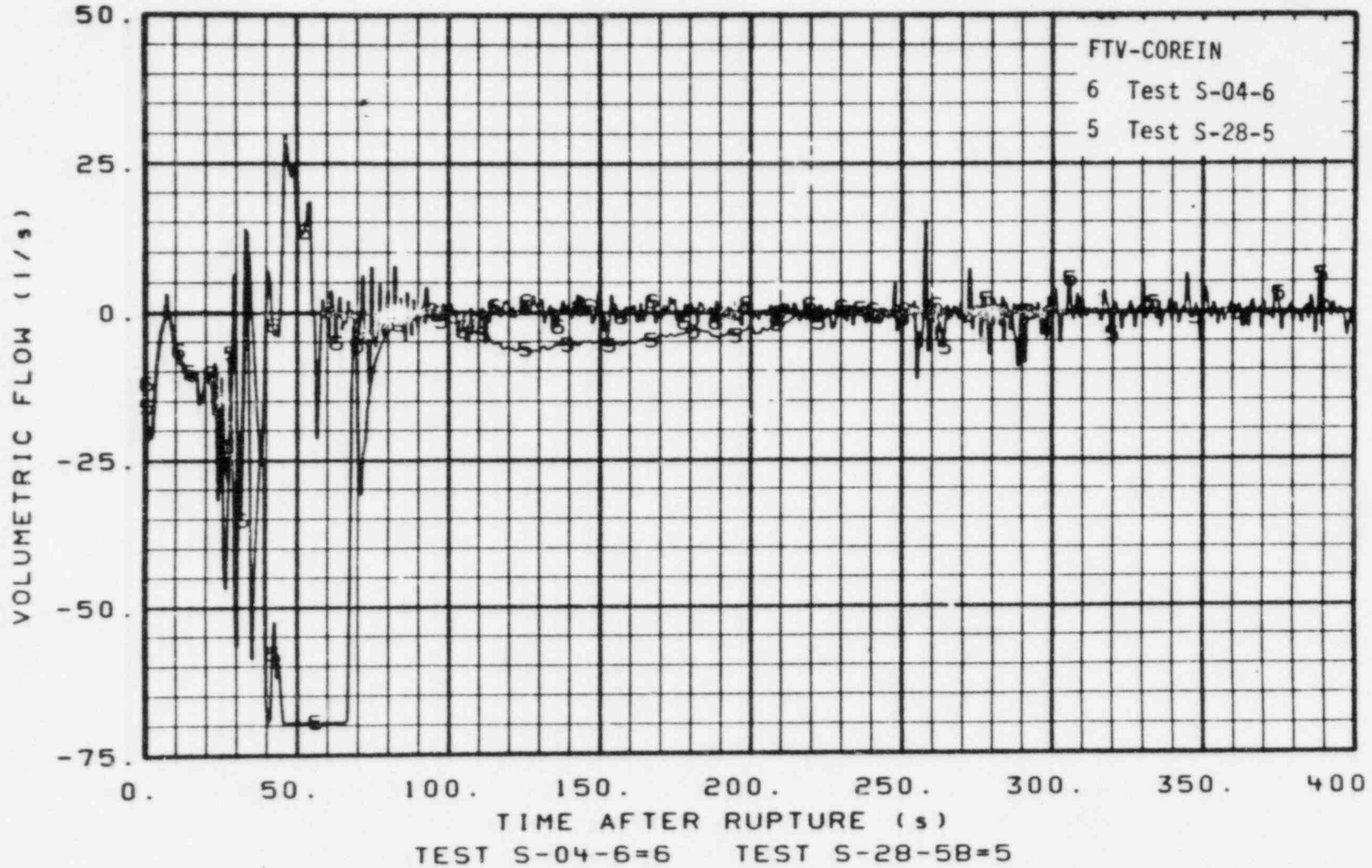

Figure 6. Comparison of Core Inlet Volumetric Flow - Tests S-28-5 and S-04-6

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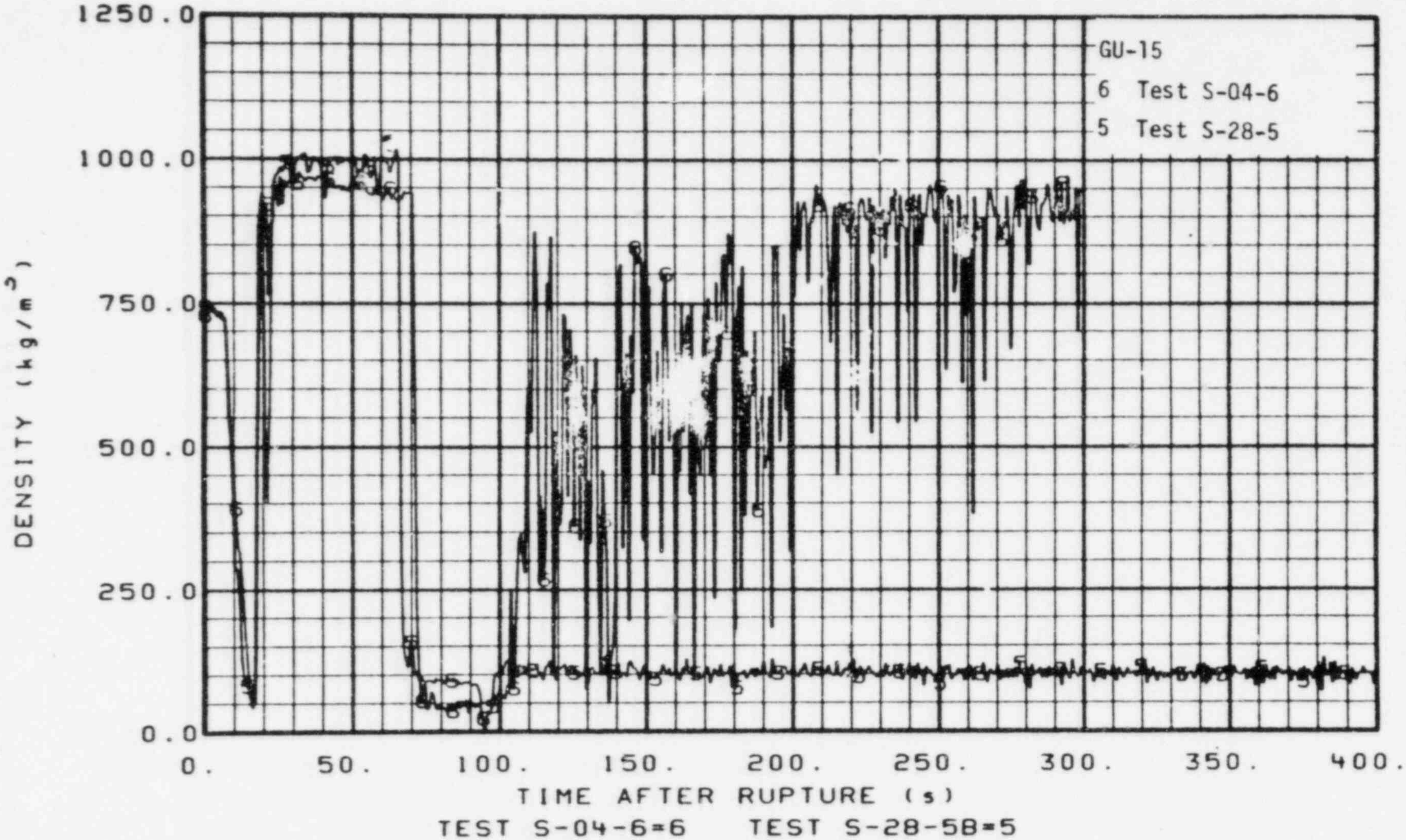

Figure 7. Comparison of Fluid Density in the Intact Loop Cold Leg Near the Vessel - Tests S-28-5 and S-04-6

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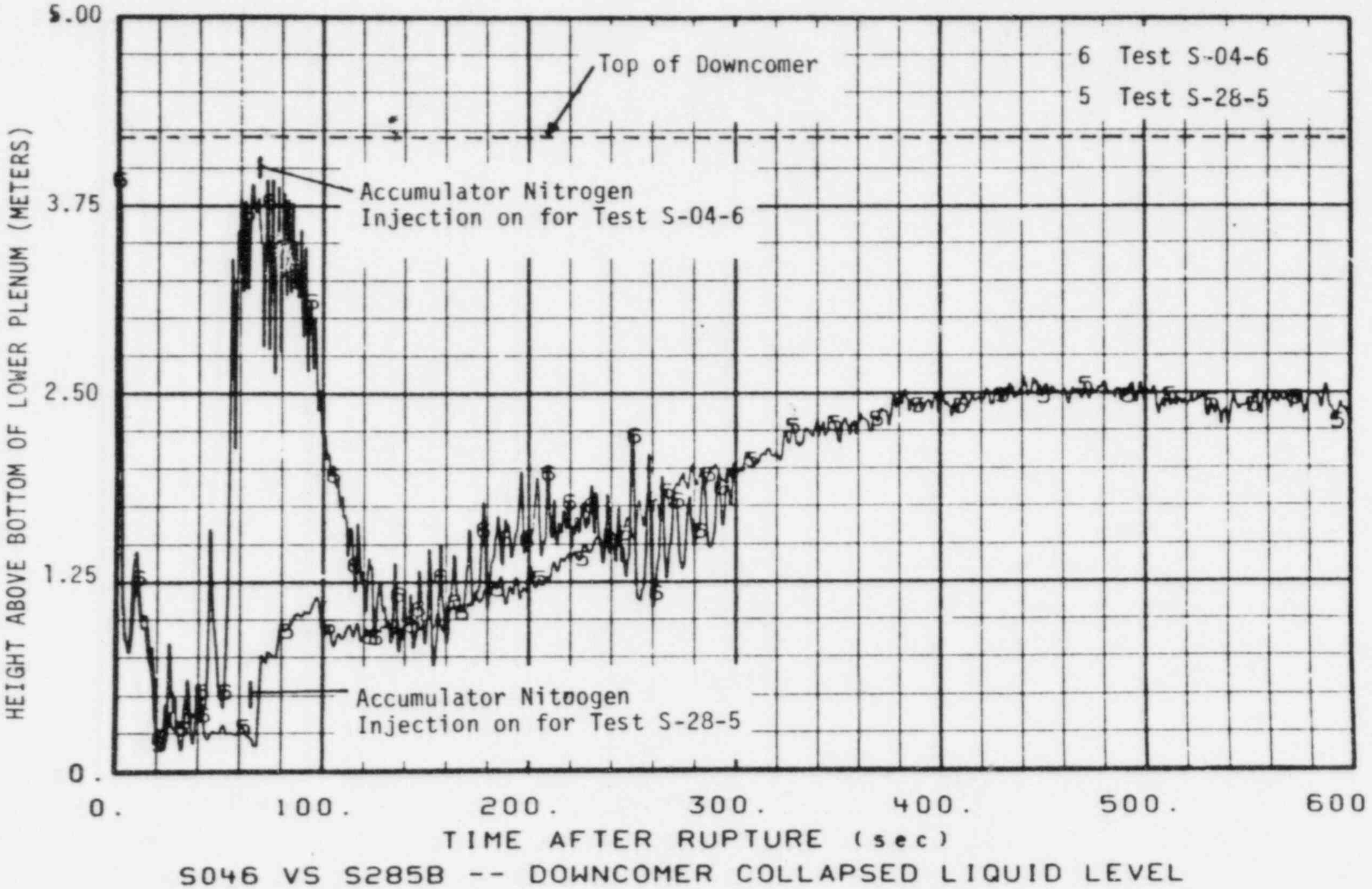

Figure 8. Comparison of Downcomer Collapsed Liquid Level -
Tests S-28-5 and S-04-6

PRELIMINARY

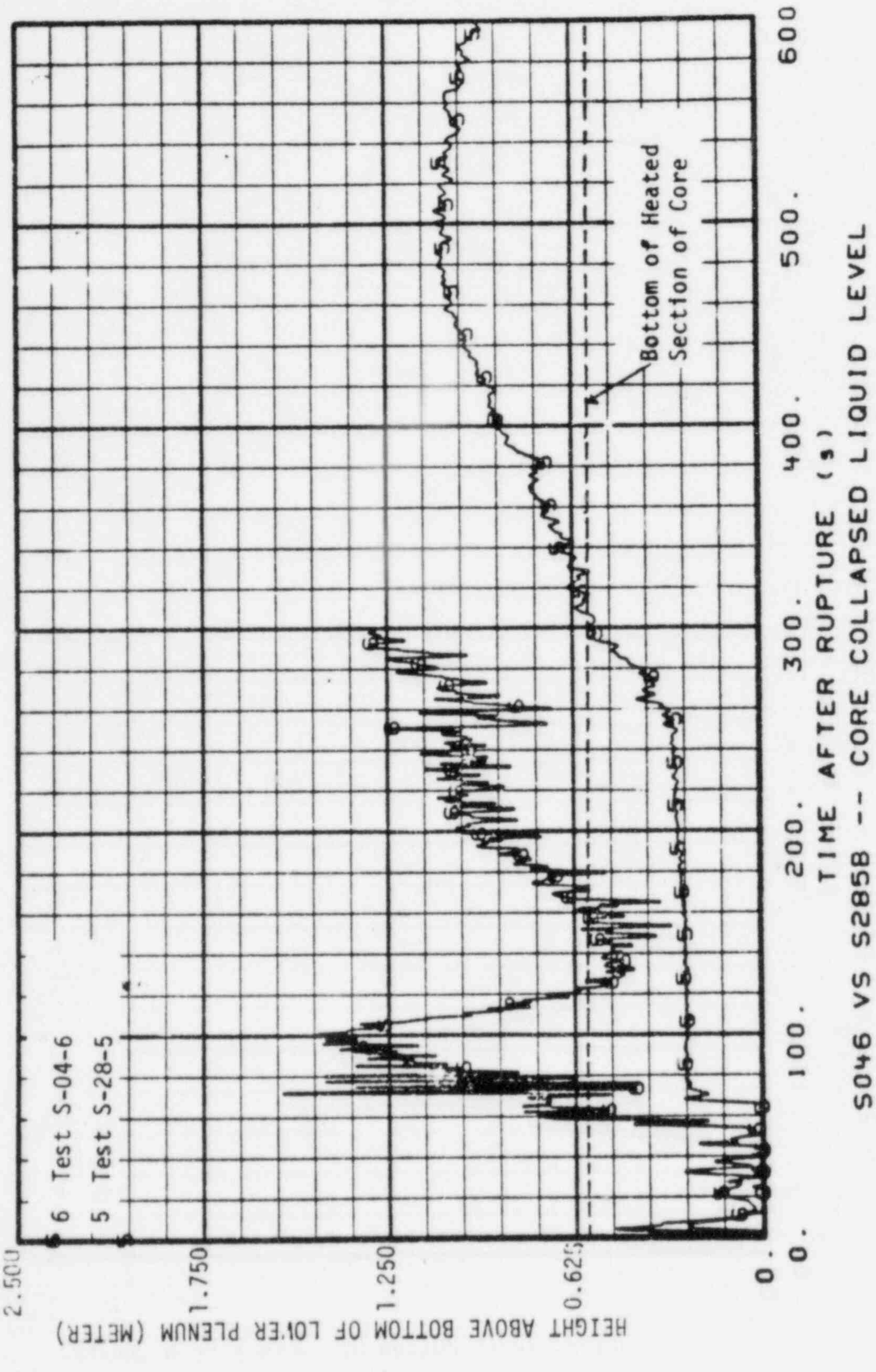

Figure 9. Comparison of Core Collapsed Liquid Level - Tests S-28-5 and S-04-6

PRELIMINARY

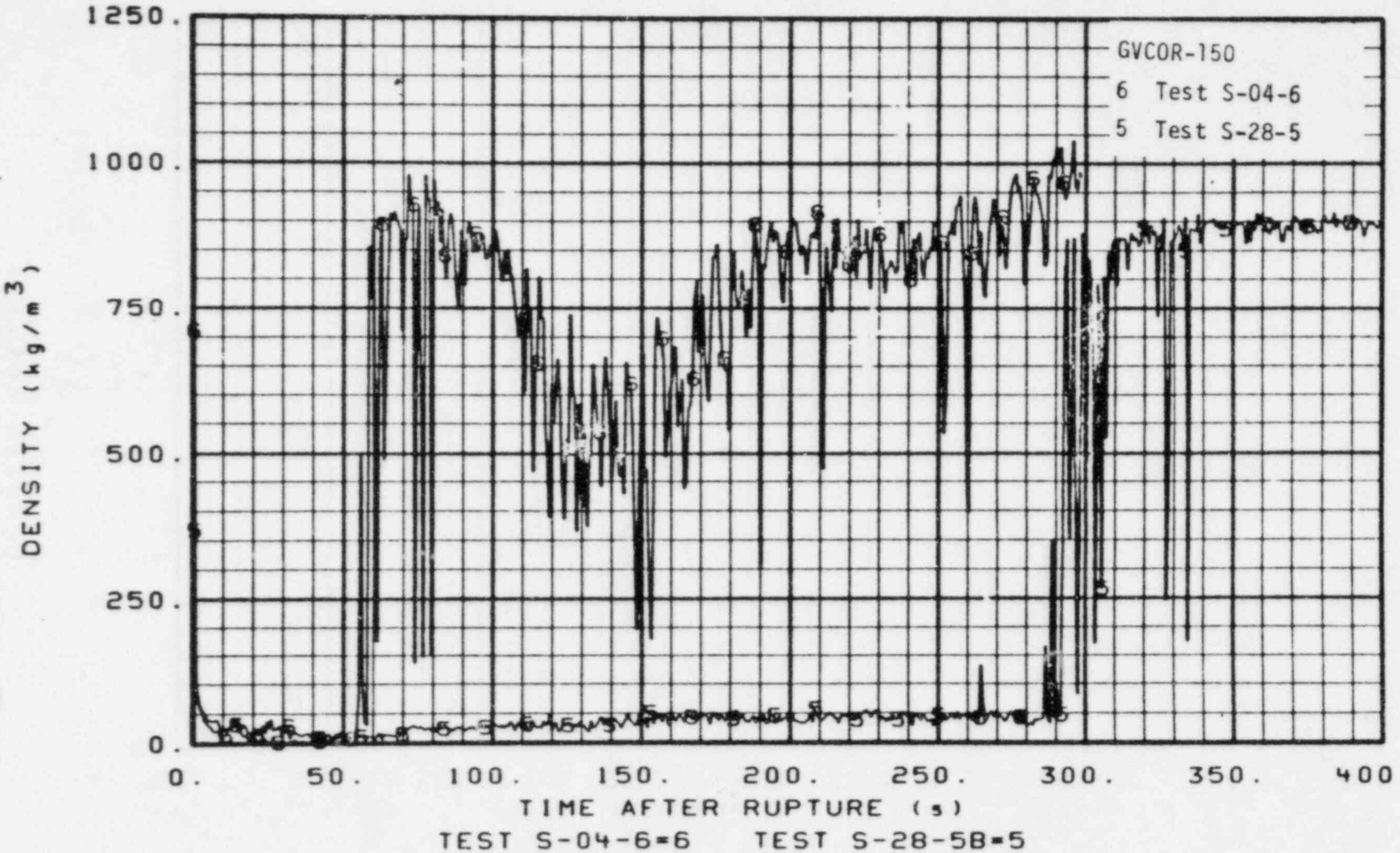

Figure 10. Comparison of Core Inlet Fluid Density - Tests S-28-5 and S-04-6

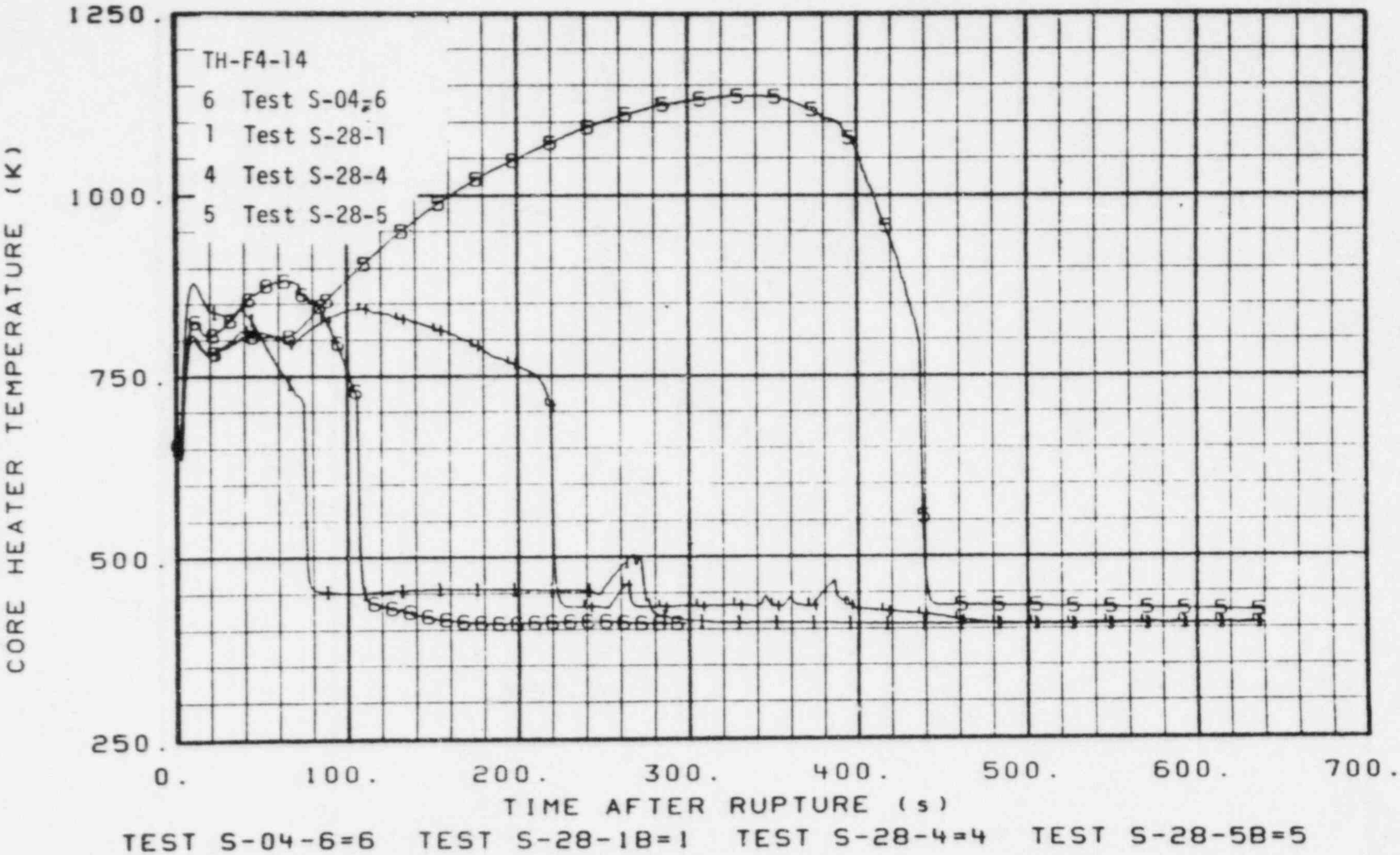

Figure 11. Comparison of Rod Cladding Temperatures on Rod F4 at the 0.36 Meter Elevation - Tests S-28-1, S-28-4, S-28-5 and S-04-6

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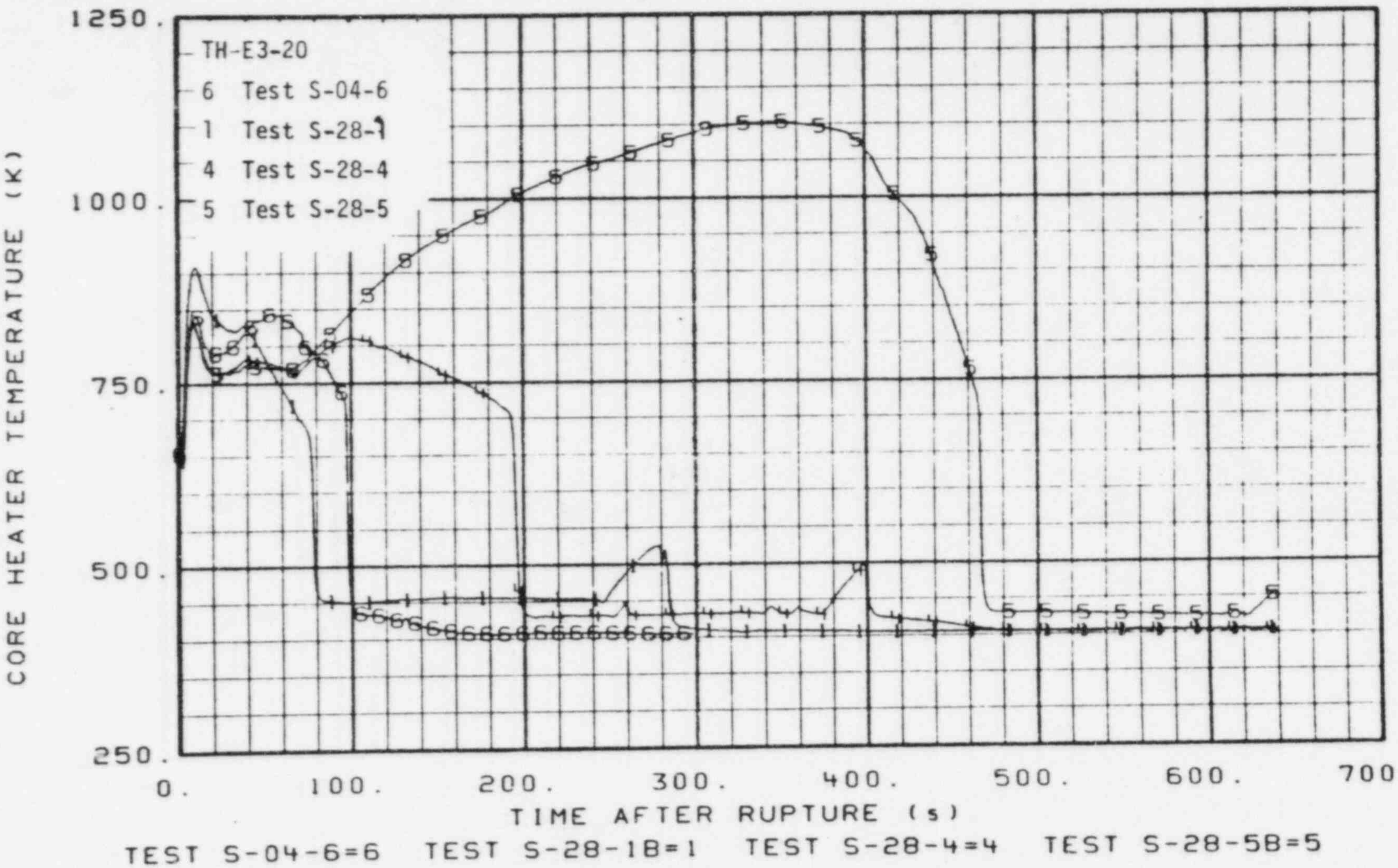

Figure 12. Comparison of Rod Cladding Temperatures on Rod E3 at the 0.51 Meter Elevation - Tests S-28-1, S-28-4, S-28-5 and S-04-6

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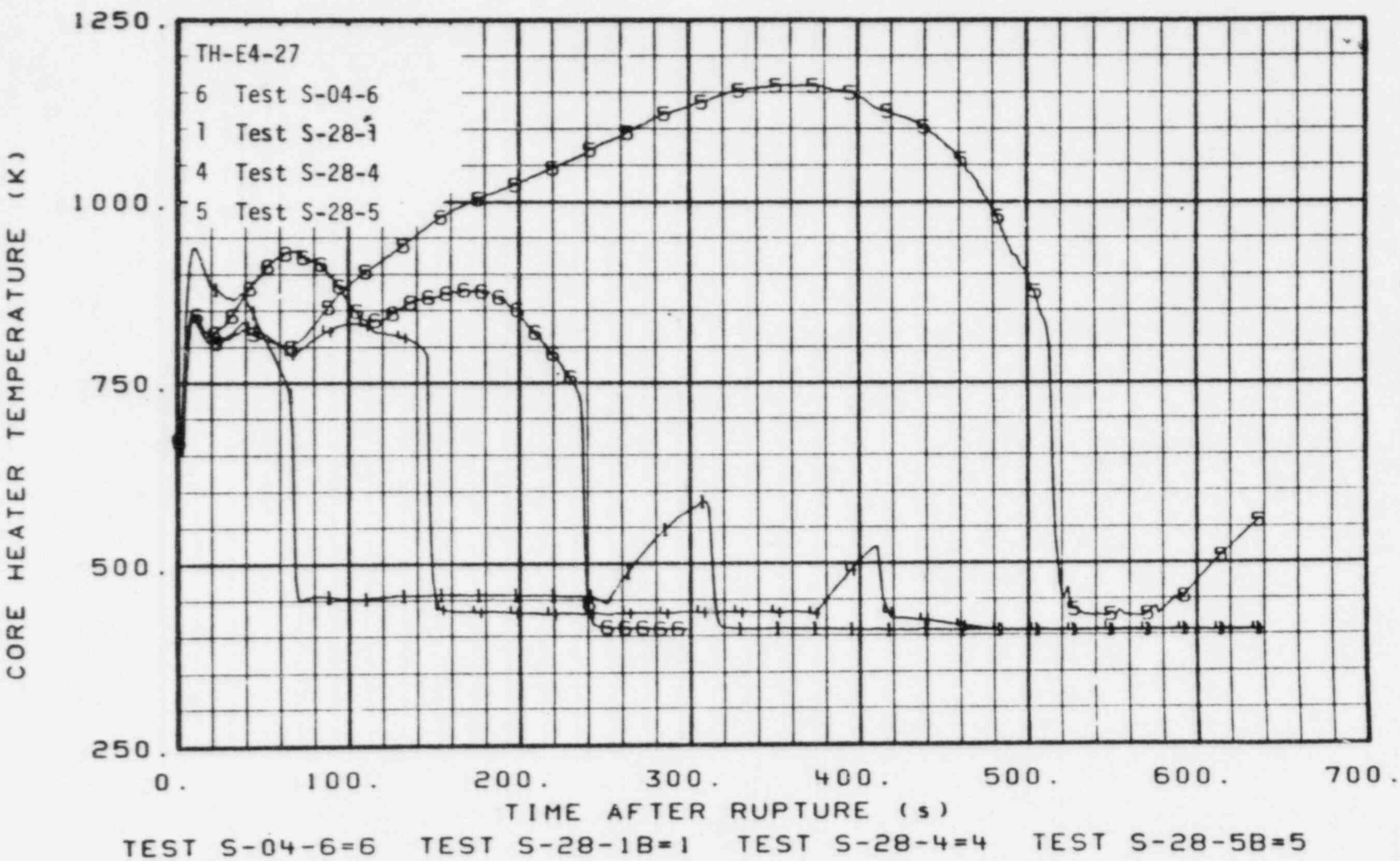

Figure 13. Comparison of Rod Cladding Temperatures on Rod E4 at the 0.69 Meter Elevation - Tests S-28-1, S-28-4, S-28-5 and S-04-6

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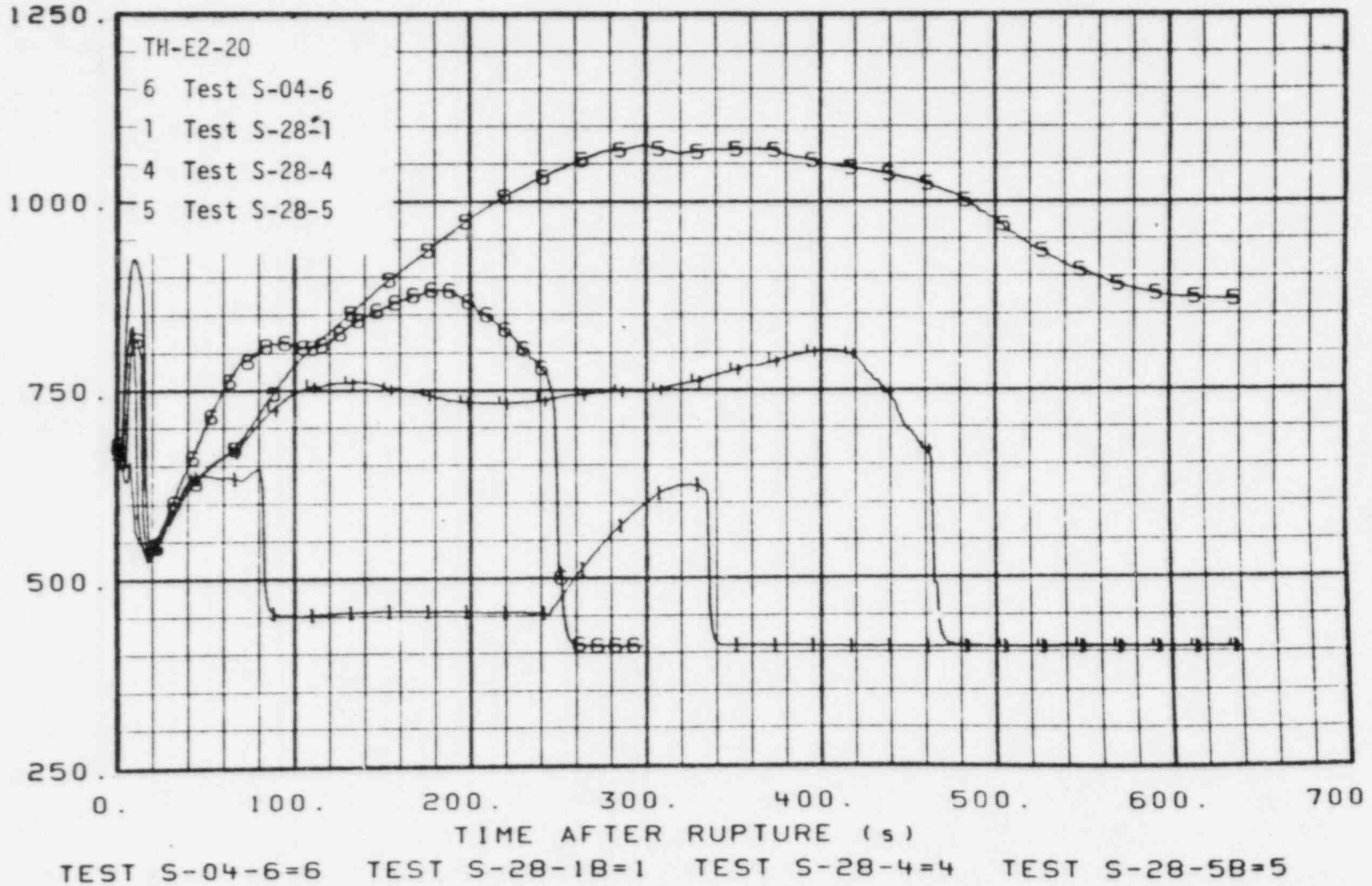

Figure 14. Comparison of Rod Cladding Temperatures on Rod E2 at the 0.84 Meter Elevation - Tests S-28-1, S-28-4, S-28-5, and S-04-6

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EG&G Idaho, Inc.

P. O. Box 1625
Idaho Falls, Idaho 83401

September 9, 1977

Mr. R. E. Tiller, Director
Reactor Operation and Program Division
Idaho Operations Office - ERDA
Idaho Falls, Idaho 83401

TRANSMITTAL OF QUICK LOOK REPORT FOR SEMISCALE MOD-1 STEAM GENERATOR TUBE RUPTURE TEST S-28-6 DJO-192-77

Dear Mr. Tiller:

Attached is the quick look report for Semiscale Mod-1 Test S-28-6 which was performed August 19, 1977. This integral blowdown-reflood test was conducted with a break configuration representative of a 200% double-ended offset shear cold leg break, and included the injection of a heated liquid into the intact loop hot leg close to the steam generator. The injection simulated the single ended rupture, at the time of refill, of a total of 16 tubes in 3 of the 4 steam generators in a four-loop PWR. Previous tests in the steam generator tube rupture test series established preliminary upper and lower limits on the range of steam generator tube rupture flow rates for which high rod cladding temperatures could occur. The primary purpose of Test S-28-6 was to refine and narrow this range of tube rupture flowrates by providing experimental data at a specific tube rupture flowrate which could potentially result in very high core heater rod cladding temperatures.

During the period of steam generator tube rupture injection in Test S-28-6, the secondary-to-primary flow had a significant influence on the thermal-hydraulic response of the system and core. The secondary-to-primary flow rate (0.139 kg/s) was large enough to delay the initiation of significant reflood until about 315 seconds after the cold leg rupture. As a result of the delay in the initiation of core reflood and the relatively poor cooling in the core during the period of downcomer and lower plenum refill, the peak cladding temperatures in Test S-28-6 were higher than the temperatures obtained in the baseline test (Test S-04-6) or in the other steam generator tube rupture tests with the tube rupture occurring at the start of vessel refill. The core maximum cladding temperature during reflood for Test S-28-6 was about 1258 K and occurred at the 0.69 m elevation on rod E4 at about 304 seconds after rupture.

The results of Test S-28-6, together with results from previous tests in the series, allowed the range of steam generator tube rupture flowrates, for which high rod cladding temperatures could occur, to be narrowed to

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those equivalent to the single ended rupture of from 16 tubes (low limit) to 20 tubes (high limit) in 3 of the 4 steam generators in a four-loop PWR. Three PWR steam generators contain about 10,000 tubes, so this range covers the rupture of only about 0.04% of the total number of tubes present.

Very truly yours,

D. J. Olson, Manager
Semiscale Program

OMH:emw

Attachment

cc: R. W. Barber, ERDA
R. S. Brodsky, ERDA
W. W. Bixby, NRC - 2
R. S. Boyd, NRC
S. Fabic, NRC
R. B. Foulds, NRC
R. F. Fraley, ACRS - 21
S. H. Hanauer, NRC
G. Kelly, NRC
S. Levine, NRC
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