

QUICK LOOK REPORT ON  
SEMISCALE MOD-1 TEST S-28-6  
STEAM GENERATOR TUBE RUPTURE TEST SERIES

SEMISCALE PROGRAM

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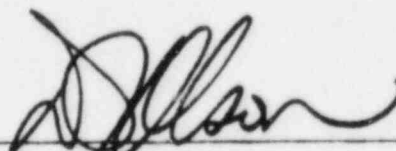
QUICK LOOK REPORT ON  
SEMISCALE MOD-1 TEST S-28-6  
STEAM GENERATOR TUBE RUPTURE TEST SERIES

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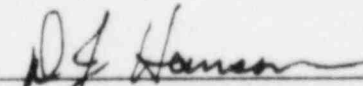
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## SEMISCALE PROGRAM

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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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## Summary

This report presents a preliminary evaluation of the results from Semicale Mod-1 Test S-28-6. 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-28-6 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 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.

The test conditions for Tests S-28-6 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-28-6 simulated the flow from the single-ended rupture of a total of 16 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 the steam generator secondary fluid to atmosphere at a rate equivalent to the tube rupture injection rate.

During the period of steam generator tube rupture injection for Test S-28-6, the secondary-to-primary flow had a significant influence on the hydraulic response of the overall system and core. A 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. 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 315 seconds after rupture.

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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-28-6 were higher than the temperatures in the baseline test (Test S-04-6) or in previous steam generator tube rupture tests. 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 peak temperature of 1258 K attained in Test S-28-6 was about 50 K higher than that obtained in Test S-28-5 (20 tube rupture) and the temperature increase was attributed primarily to the development of nearly stagnated flow conditions within the core following ECC nitrogen flow. An analysis of peak cladding temperature as a function of the secondary-to-primary flowrate indicated a definite temperature peaking for flows equivalent to the single ended rupture of 16 to 20 steam generator tubes.

The overall 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-28-6 was not of sufficient magnitude to cause significant top-down quench of the core, quenching at most locations in the core did not occur until after the initiation of core reflood. 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 and core, but this quenching behavior was not as prevalent as in Test S-28-5 due to the reduced secondary-to-primary flow in Test S-28-6. The entire core was quenched by 600 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 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.

## 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 investigates the thermal and hydraulic phenomena accompanying a hypothesized loss-of-coolant accident (LOCA) in a watercooled 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.

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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) in the Semiscale Mod-1 system. Data from Test Series 28 will be analyzed 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 sixth test in the steam generator tube rupture test series (designated Test S-28-6). Test S-28-6 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 the 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-28-6 fluid was injected at a rate of approximately 0.139 kg/s to simulate flow from the single-ended rupture of 16 tubes<sup>[a]</sup> in a PWR steam generator. The steam generator tube rupture flow was begun at about 40 seconds<sup>[b]</sup> after the initiation of the cold leg break, and continued for the duration of the test (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 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.

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[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.

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The powered heater rod configuration for Tests S-28-6 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 curve for Test S-28-6 is shown in Figure 4 together with the core power decay curve from Test S-04-6 for comparison purposes.

The specified initial conditions and operational variables together with the actual test conditions attained for Test S-28-6 are listed in Table I. Initial prerupture conditions were compared with the specified prerupture conditions. Any differences were judged as not significantly influencing postrupture system behavior.

## Test Results

The primary objective of Test S-28-6 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 16 tubes in a PWR steam generator. The magnitude of the secondary-to-primary flow rate in Test S-28-6 was in the range of tube rupture flow rates for which high rod cladding temperatures could occur<sup>[a]</sup>. A preliminary analysis of the data for Test S-28-6 has been performed to evaluate

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[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 of 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.

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TABLE I  
TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-6

<u>Primary System</u>	<u>Specified Condition</u>	<u>Test Condition</u>
Core Power (AMPCOR-T) (VOLT COR-T) (MW)	1.44	1.41
System pressure (PV+10) (kPa, gage)	15513 + 172	15633
Loop temperature		
Intact loop cold leg (RBU-14) (°K)	557.8 ± 1	557
Intact loop hot leg (RBU-2) (°K)	594.4 ± 1	593
Broken loop hot leg (TFB-30) (°K)	591.7 ± 3	588
Core flow rate (FTV-COREIN) L/min	As required to obtain core $\Delta T$	552
Pressure suppression system		
Tank water temperature (TF-PSS-33) (°K)	Ambient	289
Tank water pressure (P-PSS) (kPa, gage)	155 ± 7	150
Pressurizer water (DPU-PRESLL) (kg)	9.07	8.1
Steam generator feedwater temperature (TFU-SGFW) (°K)	497 ± 6	485
Steam generator secondary liquid level (DPU-SG-SEC) (cm)	295 ± 5	289

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TABLE I (contd)

<u>ECC System</u>	<u>Specified Condition</u>	<u>Test Condition</u>
Accumulator CI-T-1		
Injection location	Intact loop cold leg (Spool piece 14)	
Actuation Pressure (kPa, gage)	4137	4231
Liquid volume (L)	80.1	84
Injection rate (FTU-ACC-1) (L/min)	87	100
N <sub>2</sub> flow duration (sec)	24	28
Accumulator CI-T-2		
Location	Broken loop cold leg (Spool piece 42)	
Actuation Pressure ((kPa, gage)	4137	4230
Liquid volume (L)	16.4	14
Injection rate (FTB-ACC2) (L/min)	28.65	29
Accumulator CI-T-3 (Steam generator secondary simulator)		
Injection location	Intact loop hot leg (Spool piece 6)	
Temperature (TFU-SGS3-B) (°K)	547	530
Initial pressure (PU-SG3-T) (kPa, gage)	7584	8220
Liquid volume (L)	144.4	144
Injection rate (FTU-SGS-H) (L/min)	11.4 ± 1	11.7
Air actuated valve		
Open (seconds after rupture)	40	40
Close (seconds after rupture)	Remains open	Remains open

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TABLE I (contd)

<u>ECC System</u>	<u>Specified Condition</u>	<u>Test Condition</u>
Steam generator secondary fluid discharge		
Initial liquid level (cm)	295	289
Flow rate (L/min)	11.4 $\pm$ 1	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	1140
Injection rate (FTB-LPIS) (L/min)	15.1	17.1
Broken loop LPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	1034	1000
Injection rate (FTB-LPIS) (L/min)	3.6	4.3
Intact loop HPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	12411	12400
Injection rate (FTU-HPIS) (L/min)	1.17	8
Broken loop HPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	12411	12400
Injection rate (FTB-HPIS) (L/min)	0.38	1.2

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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 two 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-28-6 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-28-6 are compared with the previous steam generator tube rupture tests, Test S-28-3 and Test S-28-5 (References 3 and 4).

## System and Core Hydraulic Response During Test S-28-6

The steam generator tube rupture flow for Test S-28-6 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 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-28-6 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-28-6, however, the reverse flow through the intact loop hot leg and downward through the core became increasingly negative at the initiation of the secondary-to-primary flow (40 seconds after rupture). The relatively strong reverse core flow continued until the initiation of nitrogen flow from the intact loop accumulator (64 seconds after rupture). The accumulator nitrogen injection into the intact loop cold leg then terminated the large negative core inlet volumetric flow rate (Figure 6).

Once the intact loop accumulator nitrogen flow ceased (at about 95 seconds after rupture), 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 abatement of the reverse core flow following 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

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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 existed. As a result, the magnitude of the core inlet volumetric flow remained at small, fluctuating positive/negative values for the remainder of the test.

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-28-6 prevented substantial ECC penetration of the vessel downcomer until after the initiation of nitrogen flow from the intact loop accumulator. ECC fluid did penetrate the inlet annulus and the upper portion of the downcomer but was held up in this region by the countercurrent steam flow. Figure 8 shows the presence of subcooled fluid in the downcomer at the -10 cm elevation (10 cm below the cold leg centerline) during the period of ECC injection, prior to nitrogen injection. However, the superheated temperature at the -178 cm elevation following the initiation of the secondary-to-primary flow indicates the presence of high temperature steam from the core. The period of superheat was not evident at the -178 cm elevation in the downcomer in the baseline Test S-04-6 (Figure 9). The effect of the accumulator nitrogen injection was to force the initiation of downcomer refill at about 67 seconds after rupture. The rapid drop in downcomer fluid temperature shown at this time in Figure 8 is evidence of ECC fluid being driven into the downcomer. Figure 10 compares the downcomer collapsed liquid levels obtained from a differential pressure measurement for Tests S-28-6 and S-04-6. As indicated in the figure the downcomer collapsed liquid level for Test S-28-6 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 increase in the downcomer collapsed liquid level at about 67 seconds after rupture for Test S-28-6 was due primarily to the penetration of the downcomer by the ECC fluid stored in the vessel inlet annulus and upper portion of the downcomer during the countercurrent flow phase of the test. The relatively slow refill of the downcomer for Test S-28-6 (as well as for Test S-04-6) after about 100 seconds was accomplished by the intact loop LPIS.

Although the initiation of refill in Test S-28-6 (at 67 seconds) was only slightly later than in Test S-04-6 (at 58 seconds), reflood initiation was delayed significantly in Test S-28-6 as a result of the increased core flow resistance caused by the steam generator tube system flow. Figure 11 compares the collapsed liquid levels obtained from a lower plenum-to-upper plenum differential pressure measurement for Tests S-28-6 and S-04-6. As indicated in the figure, a gradual refill of the vessel lower plenum occurred for Test S-28-6 prior to about 300 seconds. However, because of the increasing liquid level in the downcomer (Figure 10) the downcomer head increased sufficiently to overcome the effect of the

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tube rupture flow. As a result, refill of the lower plenum after about 300 seconds occurred at a considerably increased rate, and reflood of the core was initiated specifically at about 315 seconds after rupture. The initiation of core reflood is indicated in Figure 12 which compares the core inlet fluid density for Tests S-28-6 and S-04-6. About 315 seconds after rupture the density measurement indicates the continuous presence of liquid in Test S-28-6.

Analysis of the system hydraulic response allows the secondary-to-primary flowrate to be classified as being in a "low" or "high" regime. The demarcation between the "low" and "high" secondary to primary flows is illustrated in Figures 13 and 14. The development of negative flows in both the intact loop hot leg and the core for the 0.174 kg/s tube rupture flow condition (Test S-28-5) for an extended period after the cessation of nitrogen injection characterizes this tube rupture flow as being in the "high" range. In contrast, Test S-28-6 exhibits a definite positive flow in the intact loop hot leg after nitrogen injection, as does Test S-28-3. The hydraulic behavior in Tests S-28-6 and S-28-3 is in the "low" secondary-to-primary flow regime and would be expected to produce a bottom up core quench pattern. The core thermal response for Test S-28-6 is discussed in the following section.

## Core Thermal Response During Test S-28-6

The effect of the magnitude of the steam generator tube rupture flow on the core thermal response for Test S-28-6 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-28-3 and S-28-5 for which the secondary-to-primary low rates were 0.104 kg/s and 0.174 kg/s, respectively [a]. Figures 15 through 19 compare typical rod cladding temperatures at the 0.20, 0.36, 0.51, 0.69 and 0.99 m core elevations for the four tests. For Tests S-28-3, S-28-6 and S-28-5, the secondary-to-primary flows resulted in extended periods of rod heat-up and higher peak rod temperatures, than for the baseline Test S-04-6. For Test S-28-6 the extended heat-up occurs because the secondary-to-primary flow results in hydraulic conditions which effectively stagnate the core flow, for a significant period, resulting in relatively poor core cooling. The filling of the downcomer by LPIS injection eventually results in ECC penetration of the core causing the rod temperatures to turn over. By this time the maximum rod cladding temperature in Test S-28-6 had reached 1258 K. For comparison purposes, Table II lists the peak cladding temperatures and the corresponding time of occurrence after

[a] The steam generator tube rupture flow rates for Tests S-28-3 and S-28-5 are equivalent to the flow rates associated with the single-ended rupture of 12 tubes and 20 tubes, respectively, in a PWR steam generator.

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TABLE II

Maximum Core Cladding Temperatures and Times at  
Which Maximum Temperatures Occur for  
Tests S-28-3, S-28-5, S-28-6,  
and S-04-6

Thermocouple I. D.	Test S-04-6 °K (sec)	Test S-28-3 °K (sec)	Test S-28-5 °K (sec)	Test S-28-6 °K (sec)
E3-05	710 (44)	707 ( 60)	886 (284)	805 (290)
C7-07	840 (44)	853 ( 59)	894 (186)	897 (233)
F2-07	765 (52)	743 ( 61)	910 (250)	684 (309)
E6-08	808 (53)	813 ( 62)	1013 (278)	946 (282)
A4-09	885 ( 9)	886 ( 61)	874 ( 10)	930 (208)
E4-09	863 (10)	792 ( 62)	1028 (330)	996 (293)
G3-13	890 (64)	865 (124)	1061 (324)	1047 (286)
D2-14	902 (64)	892 (135)	1022 (329)	1085 (288)
D4-14	860 (64)	879 (135)	1036 (283)	1113 (332)
E8-14	935 (64)	942 (117)	1028 (289)	1082 (288)
F4-14	885 (64)	882 (135)	1137 (341)	1120 (288)
G5-14	921 (58)	893 (124)	1124 (316)	1073 (279)
C7-15	943 (64)	961 (134)	1013 (326)	1121 (282)
C4-20	NA	911 (152)	874 (154)	1110 (310)
J7-20	966 ( 9)	984 (136)	1085 (333)	1157 (250)
E2-20	891 (58)	NA	1103 (341)	NA
E3-20	846 (54)	904 (152)	NA	1174 (307)
E5-20	903 (54)	978 (152)	1155 (356)	1199 (308)
F5-20	888 (54)	921 (138)	1166 (321)	1152 (284)
D1-21	1066 ( 9)	1046 ( 8)	1053 ( 8)	1057 (216)
F2-22	NA	940 (138)	1055 (206)	851 (265)
E3-24	934 (64)	1006 (152)	1117 (339)	1205 (304)
G5-24	982 (62)	1035 (152)	1185 (273)	1138 (216)
D6-25	959 (64)	1072 (152)	1113 (332)	1239 (261)
F2-25	936 (64)	992 (169)	1067 (300)	947 (292)
E5-25	959 (64)	1025 (167)	1162 (354)	1242 (300)
C4-26	NA	1035 (162)	920 (183)	1153 (246)
D8-26	NA	1069 (243)	994 (268)	1118 (238)
F5-26	NA	1048 (167)	1208 (315)	1228 (283)
E4-27	934 (71)	1038 (173)	1160 (356)	1258 (304)
C5-28	948 (71)	1079 (167)	946 (170)	1196 (262)
E6-28	940 (64)	1056 (167)	1177 (330)	1242 (280)
A4-29	NA	1074 (169)	944 ( 10)	1058 (186)
A5-29	1075 ( 9)	1073 (169)	989 ( 9)	1063 (176)
B5-29	NA	1092 (167)	923 (108)	1125 (187)
B6-29	1048 ( 8)	1091 (173)	973 ( 8)	1128 (229)
D3-29	918 (71)	1012 (241)	1045 (331)	1183 (300)
D4-29	914 (71)	1028 (173)	1046 (359)	1236 (325)

PRELIMINARY

# PRELIMINARY

TABLE II (contd)

Thermocouple I. D.	Test S-04-6 °K (sec)	Test S-28-3 °K (sec)	Test S-28-5 °K (sec)	Test S-28-6 °K (sec)
E8-29	997 ( 71)	1097 (163)	1105 (249)	1201 (265)
F4-29	940 ( 71)	1037 (173)	1182 (318)	1218 (286)
G4-29	968 ( 71)	1042 (169)	1152 (259)	1125 (231)
B3-32	819 ( 9)	1007 (250)	809 (162)	980 (289)
H5-32	946 ( 71)	1039 (248)	1053 (240)	974 (211)
B5-33	856 ( 84)	1070 (246)	826 ( 10)	1022 (259)
E1-33	819 (162)	1009 (248)	976 (280)	1023 (253)
E2-33	884 (182)	1029 (251)	1075 (299)	1133 (291)
F5-33	916 ( 71)	1040 (248)	1168 (330)	1181 (321)
G4-33	921 ( 70)	1032 (246)	1104 (310)	1058 (219)
E6-37	810 ( 60)	1026 (251)	1095 (381)	1156 (330)
C2-38	691 ( 72)	927 (281)	834 (332)	916 (230)
G4-38	853 ( 64)	1002 (254)	995 (311)	951 (323)
A4-39	NA	988 (264)	750 ( 10)	751 ( 6)
D3-39	715 ( 75)	994 (264)	993 (356)	1140 (362)
E7-44	768 (189)	1020 (283)	962 (435)	1008 (257)
F4-44	703 ( 84)	988 (292)	1017 (445)	1071 (414)
A5-45	692 ( 8)	972 (307)	669 ( 0)	670 ( 0)
C4-53	619 ( 0)	897 (310)	617 ( 0)	827 (359)
C6-53	638 ( 0)	926 (310)	687 ( 0)	865 (360)
F5-53	609 ( 0)	912 (311)	947 (457)	1009 (420)
E4-55	633 ( 0)	919 (334)	818 (639)	1036 (427)
D2-61	608 ( 0)	783 (338)	707 (639)	765 (486)

PRELIMINARY

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TABLE III

Core Quench Times for Tests S-28-3, S-28-5, S-28-6, and S-04-6

Thermocouple I. D.	Test S-04-6 (sec)	Test S-28-3 (sec)	Test S-28-5 (sec)	Test S-28-6 (sec)
E3-05	77	75	389	361
C7-07	86	124	397	373
F2-07	84	115	397	362
E6-08	88	127	405	379
A4-09	95	144	218	364
E4-09	93	138	410	388
G3-13	108	175	435	424
D2-14	109	177	435	427
D4-14	103	174	436	427
E8-14	107	177	435	424
F4-14	106	176	437	426
G5-14	111	175	438	427
C7-15	135	192	443	438
C4-20	NA	224	236	466
D7-20	191	221	463	461
E2-20	184	NA	NA	NA
3-20	97	222	466	464
C5-20	115	231	470	469
F5-20	112	223	468	461
D1-21	140	251	470	469
F2-22	106	250	473	446
E3-24	106	269	481	479
G5-24	208	274	483	482
D6-25	195	295	494	491
F2-25	196	301	496	488
E5-25	184	283	488	471
C4-26	208	314	193	496
D8-26	217	407	672	538
F5-26	202	315	501	494
E4-27	236	329	513	504
C5-28	220	343	192	507
E6-28	201	330	513	502
A4-29	229	355	114	195
A5-29	230	353	122	236
B5-29	236	345	126	458
B6-29	188	361	186	514
D3-29	242	348	530	508
D4-29	227	348	536	511
E8-29	234	346	529	508

PRELIMINARY



# PRELIMINARY

TABLE III (contd)

Thermocouple I. D.	Test S-04-6 (sec)	Test S-28-3 (sec)	Test S-28-5 (sec)	Test S-28-6 (sec)
F4-29	238	346	530	508
G4-29	240	345	529	506
B3-32	98	397	170	532
H5-32	294	402	380	248
B5-33	183	394	102	333
E1-33	184	402	664	531
E2-33	248	397	664	531
F5-33	265	400	656	531
G4-33	277	397	656	527
E6-37	107	431	680	549
C2-28	80	435	683	528
G4-38	269	436	680	541
A4-39	NA	439	76	107
D3-39	85	439	680	548
E7-44	298	479	704	571
F4-44	99	481	706	573
A5-45	99	482	69	72
C4-53	298	439	71	408
C6-53	63	488	154	556
F5-53	68	498	716	576
E4-55	71	505	No quench	595
D2-61	78	533	No quench	602

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# PRELIMINARY

rupture for each thermocouple recorded during Tests S-28-3, S-28-5 and S-28-6. As indicated in the table, the relative ordering of the peak temperatures are 1258 K at 304 seconds (TH-E4-27) for Test S-28-6, 1208 K at 315 seconds (Th-F5-26) for Test S-28-5, and 1097 K at 163 seconds (TH-E8-29) for Test S-28-3. Figure 20 demonstrates the trend of peak clad temperature as a function of the number of simulated tube ruptures. A definite temperature peaking occurs for the 16 to 20 tube rupture range.

A comparison of the rod quench times for Tests S-28-3, S-28-5, S-28-6, and S-04-6 illustrates the pronounced effect of the steam generator secondary-to-primary flow on core thermal response. Table III lists the quench times for all thermocouple locations recorded for these four tests. As indicated by the table, a definite bottom-up quench behavior is evident for most core locations in each of the tests. The time of quench at the lower elevations varies inversely with the secondary-to-primary flow rate with the baseline Test S-04-6 giving the earliest and most rapid quench time progression (at the lower elevations). The quenching times for Test S-28-6 were similar to those in Test S-28-5 at the lower elevations but were somewhat earlier at the upper core elevations. The latest observed quench time for Test S-28-6 occurred at 602 seconds at 1.63 m above the bottom of the heated core. The same location for Test S-28-5 remained unquenched after 740 seconds, however, the 1.33 m location was quenched by 716 seconds.

## 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-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).
- (4) D. J. Olson Ltr to R. E. Tiller, DJO-179-77, "Transmittal of Quick Look Report for Semiscale Mod-1 Steam Generator Tube Rupture Test S-28-5," (August 19, 1977).

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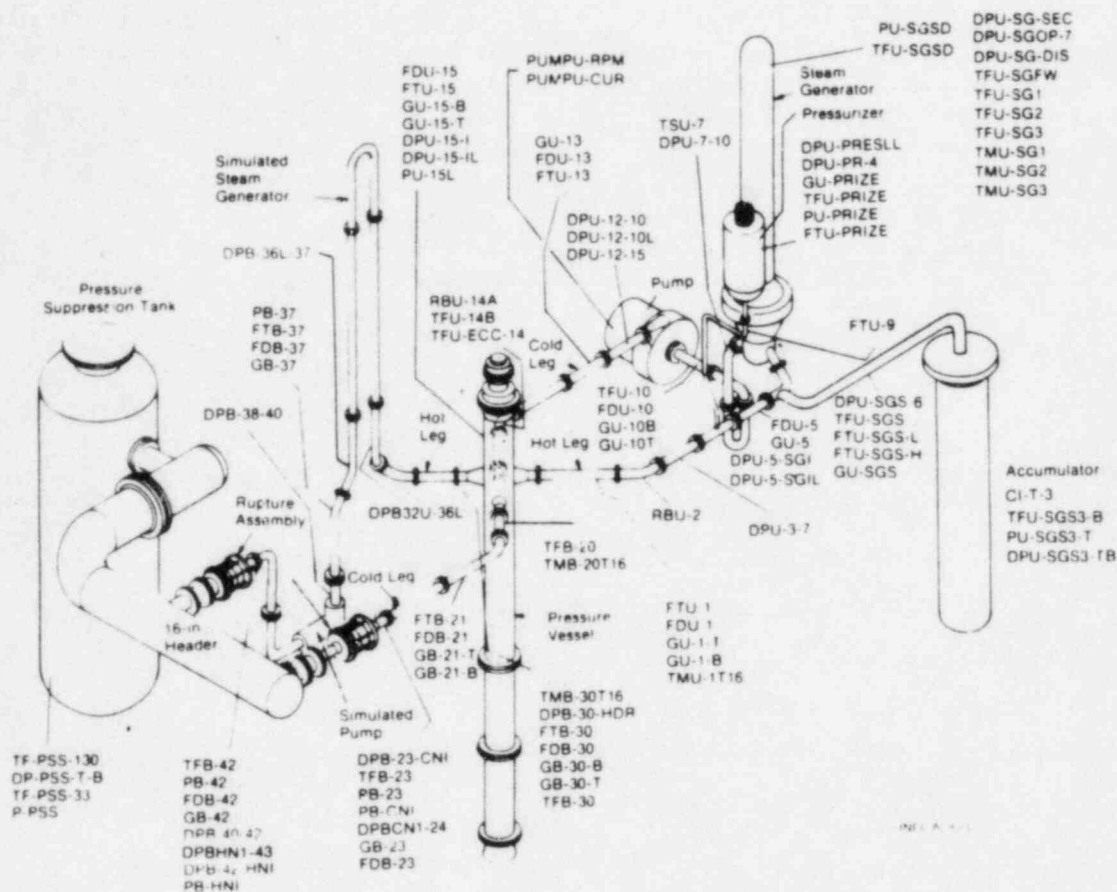


Figure 1. Semiscale Mod-1 System and Instrumentation for Cold  
Break Configuration - Isometric

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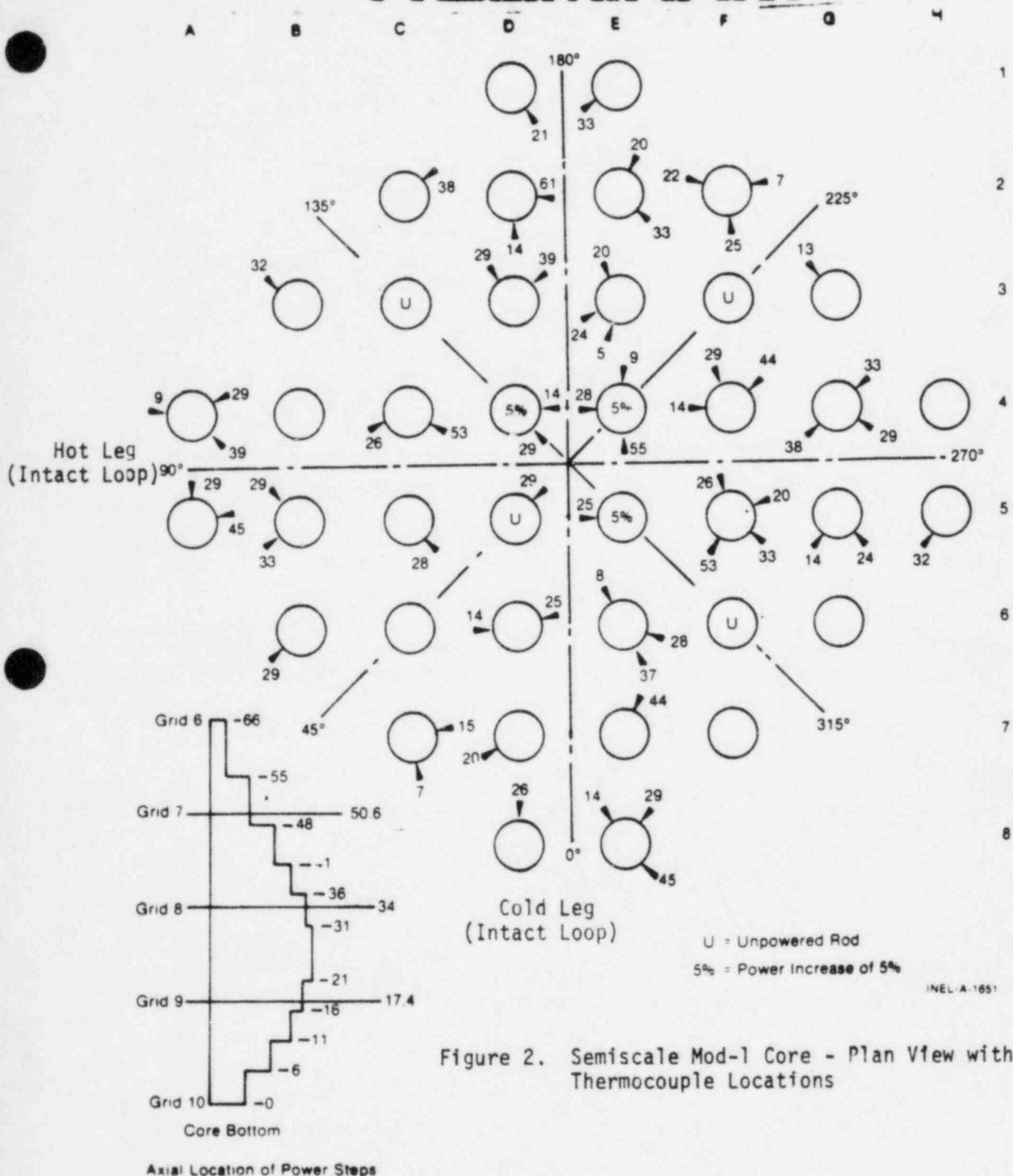


Figure 2. Semiscale Mod-1 Core - Plan View with Thermocouple Locations

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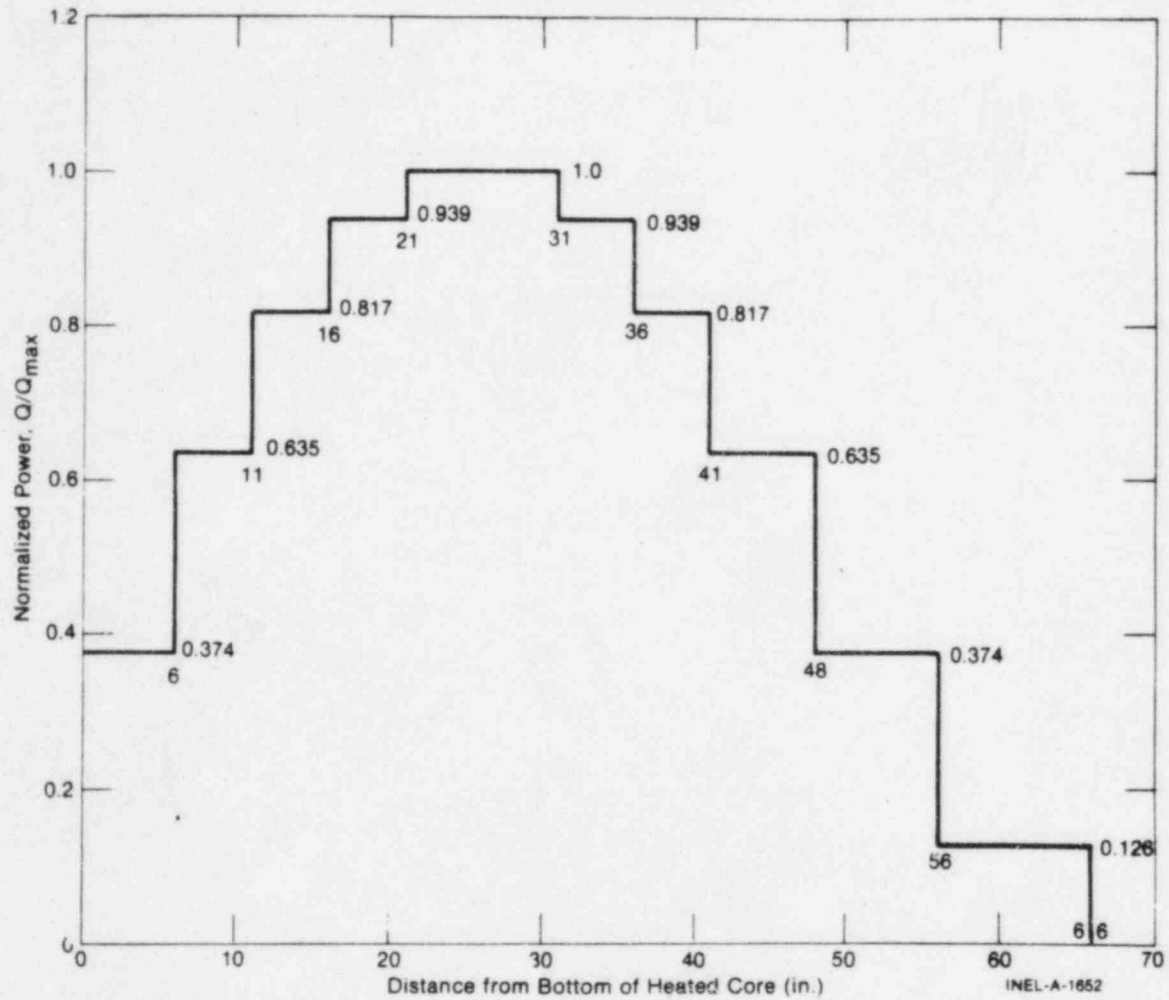


Figure 3. Semiscale Mod-1 Axial Power Profile

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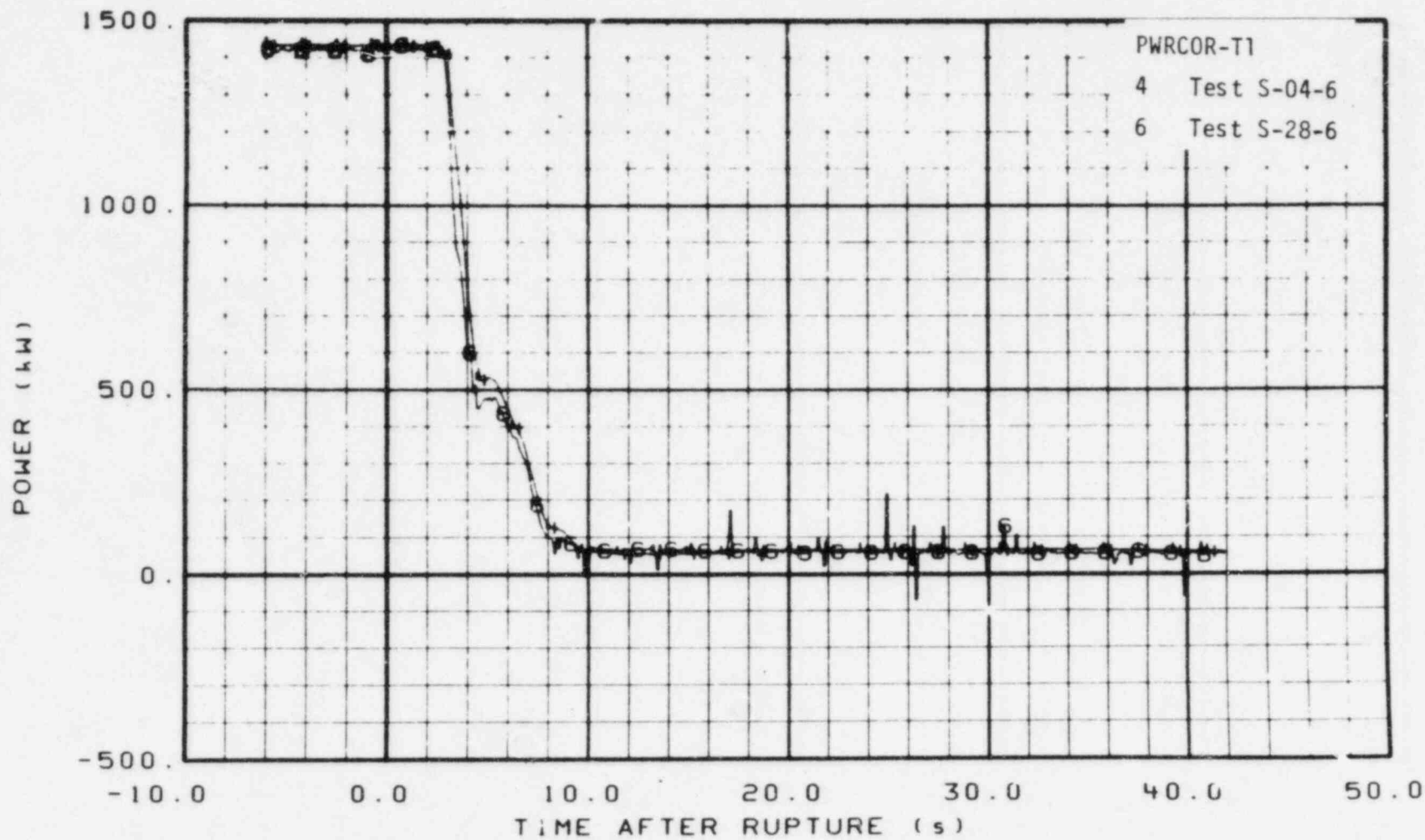


Figure 4. Initial Core Power Decay - Tests S-28-6 and S-04-6

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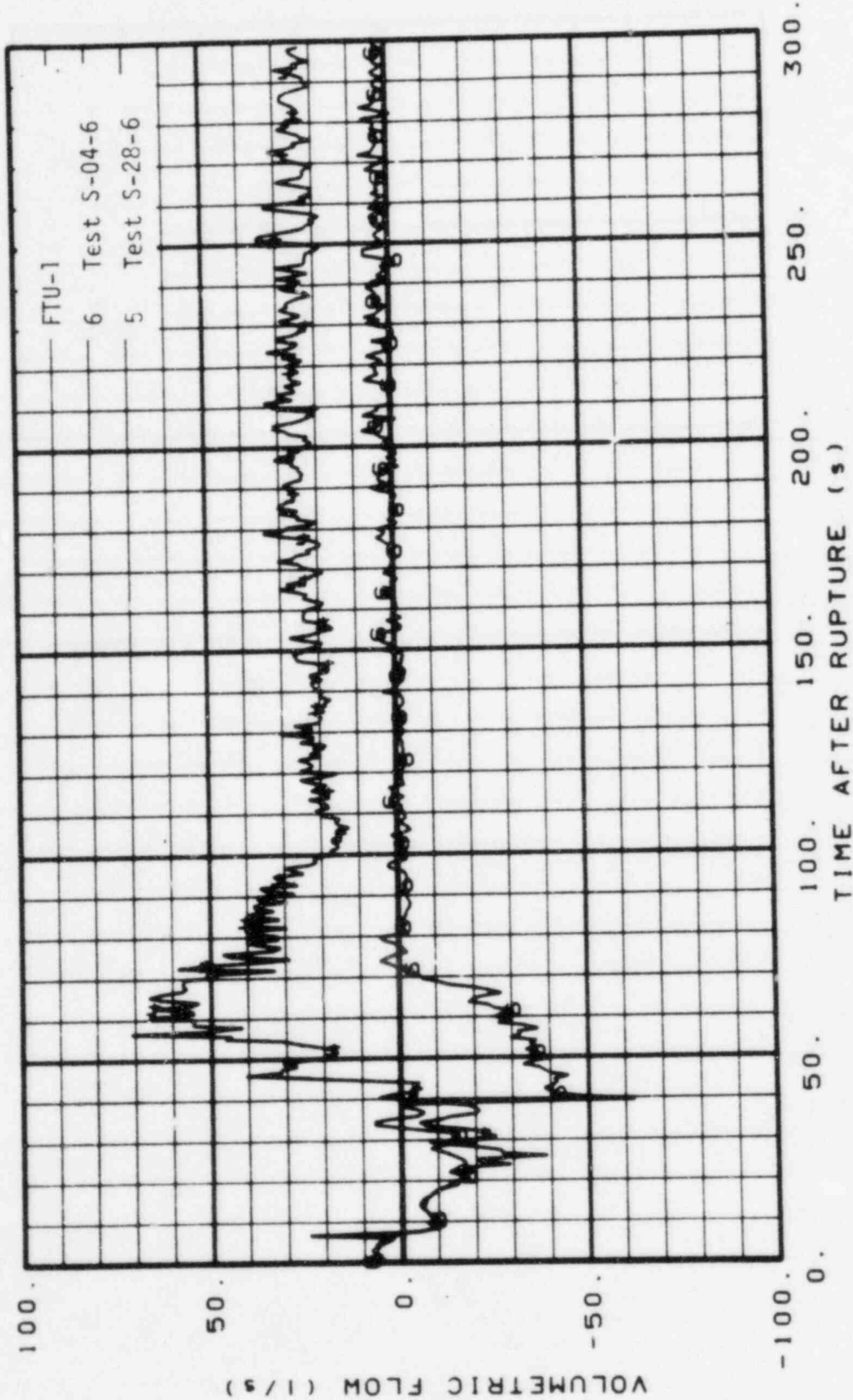


Figure 5. Comparison of Intact Loop Hot Leg Volumetric Flow Near the Vessel - Tests S-28-6 and S-04-6

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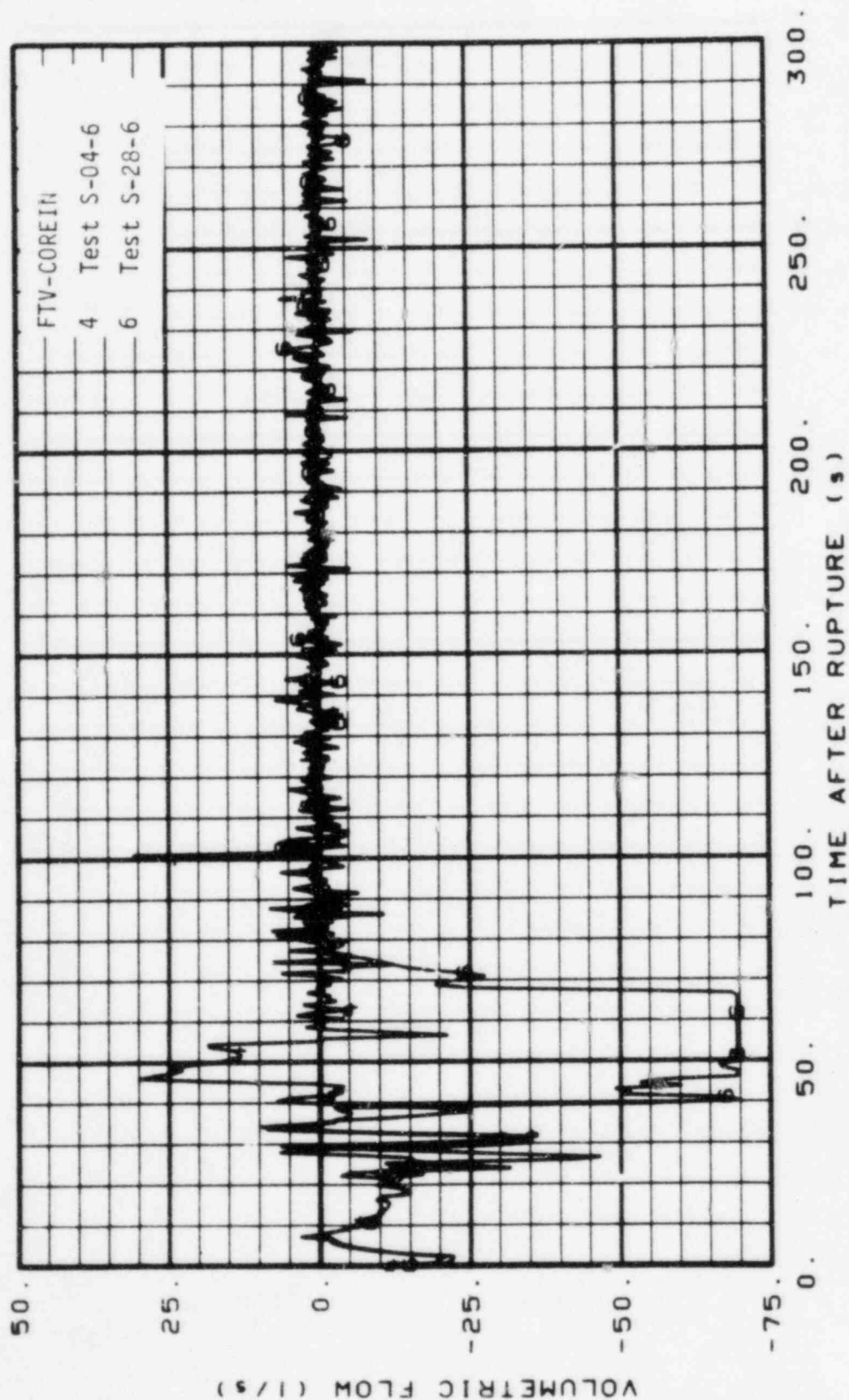


Figure 6. Comparison of Core Inlet Volumetric Flow - Tests S-28-6 and S-04-6

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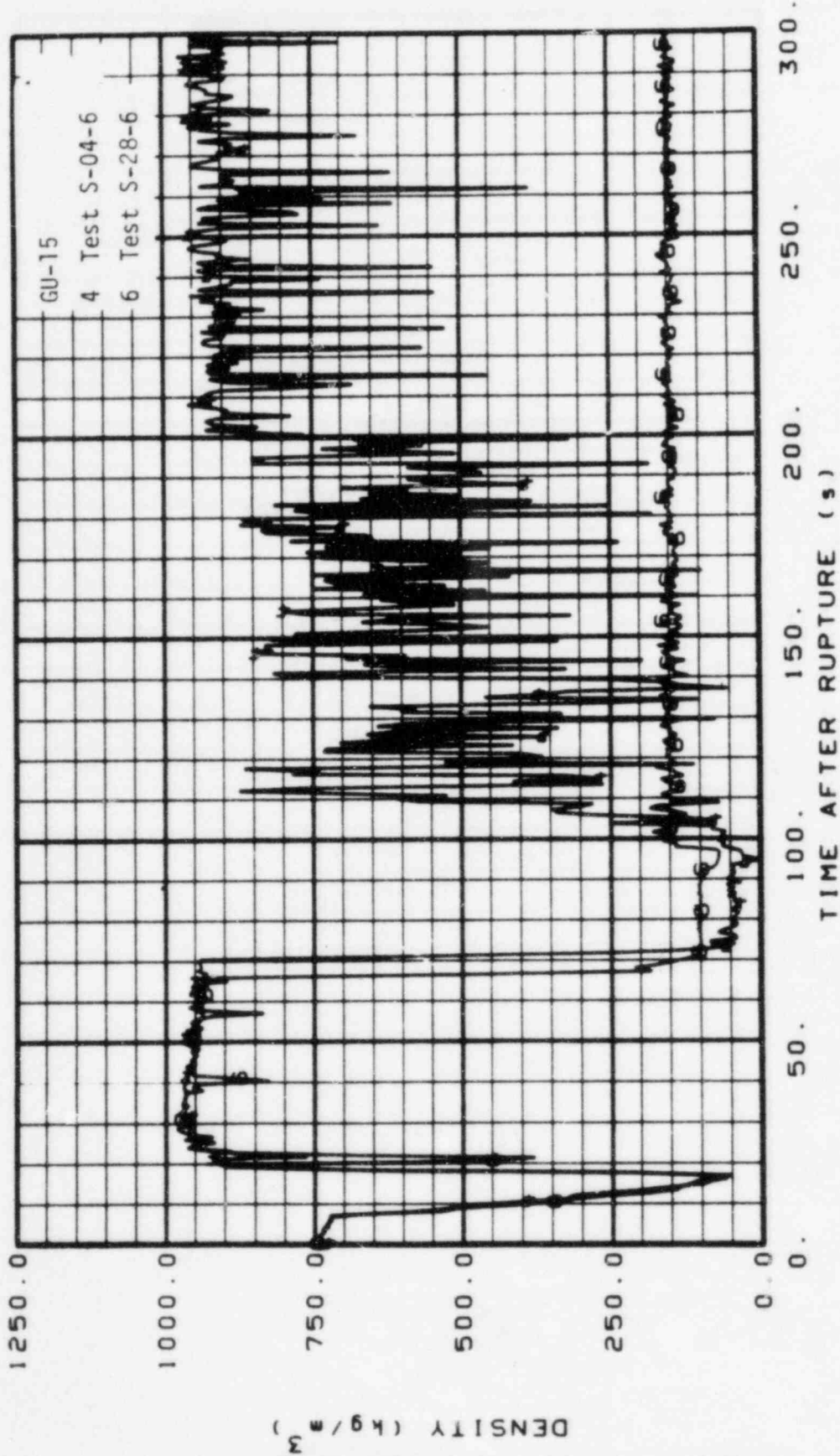
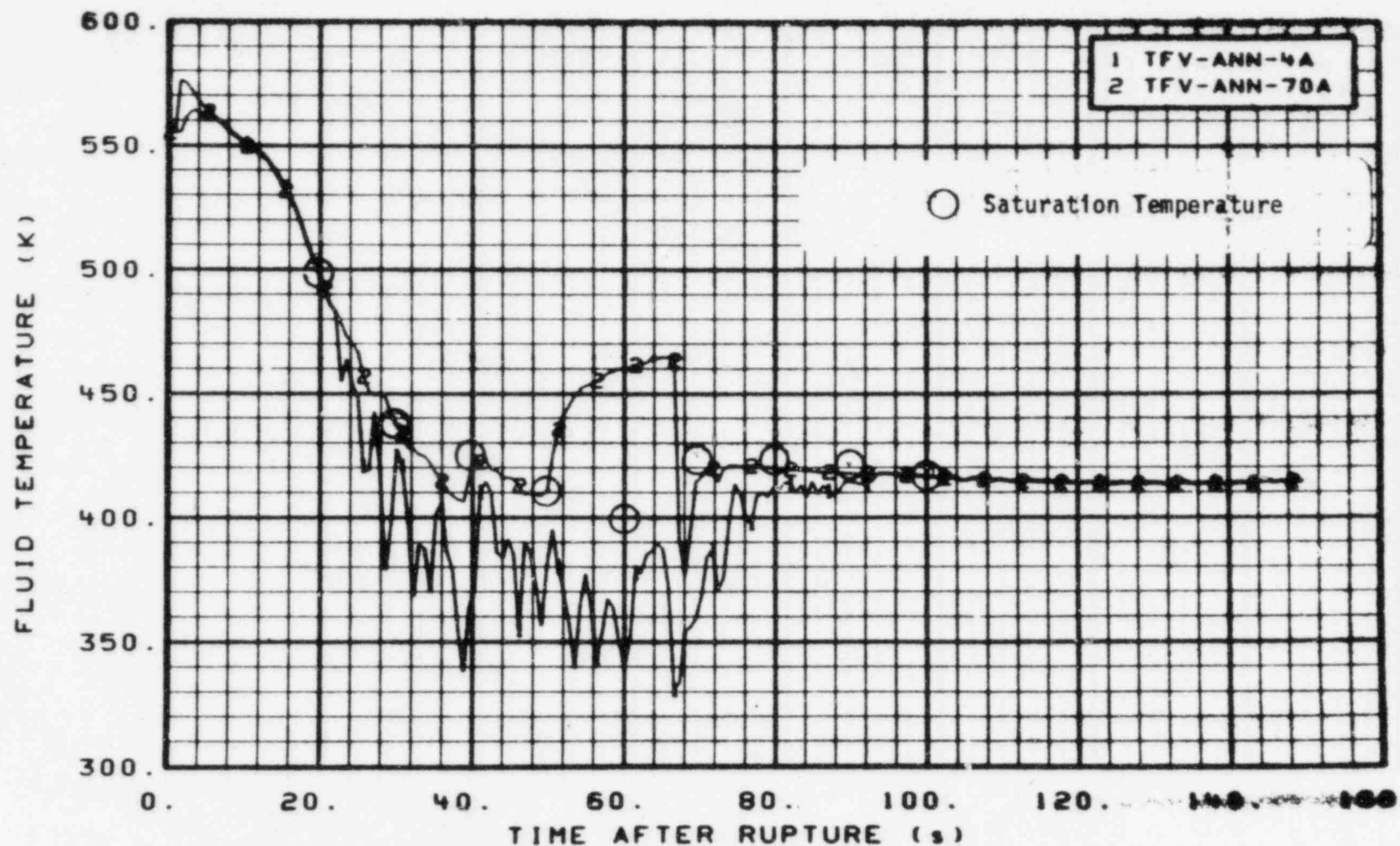


Figure 7. Comparison of Fluid Density in the Intact Loop Cold Leg Near the Vessel - Tests S-28-6 and S-04-6

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TFV-ANN-4A = (1)Vessel Elevation -10 cm, TFV-ANN-70A = (2) Vessel Elevation -178 cm

Figure 8. Test S-28-6 Downcomer Fluid Temperatures at the -10 and -178 cm Elevations

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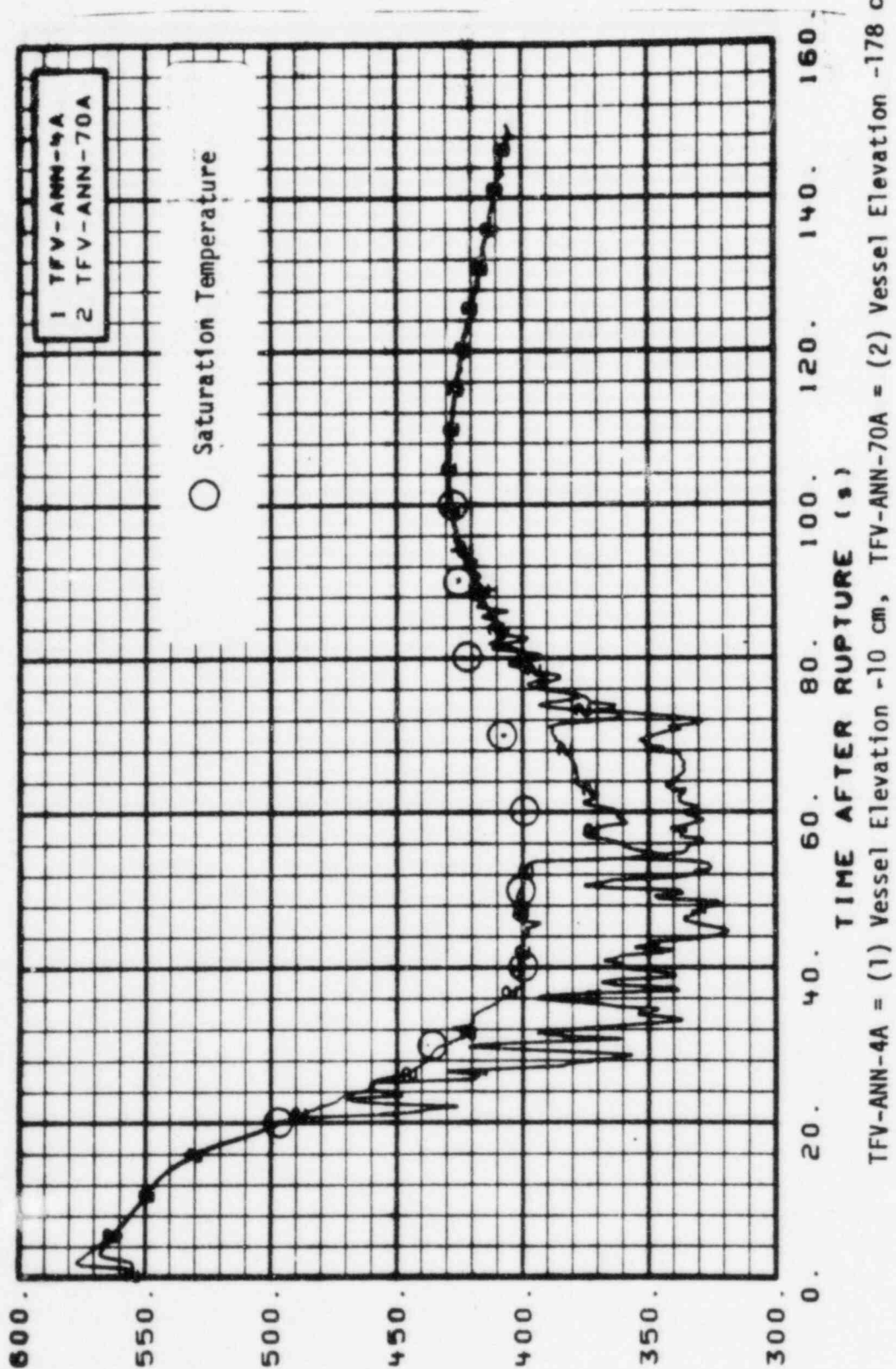


Figure 9. Test S-04-6 Downcomer Fluid Temperatures at the -10 and -178 cm Elevations

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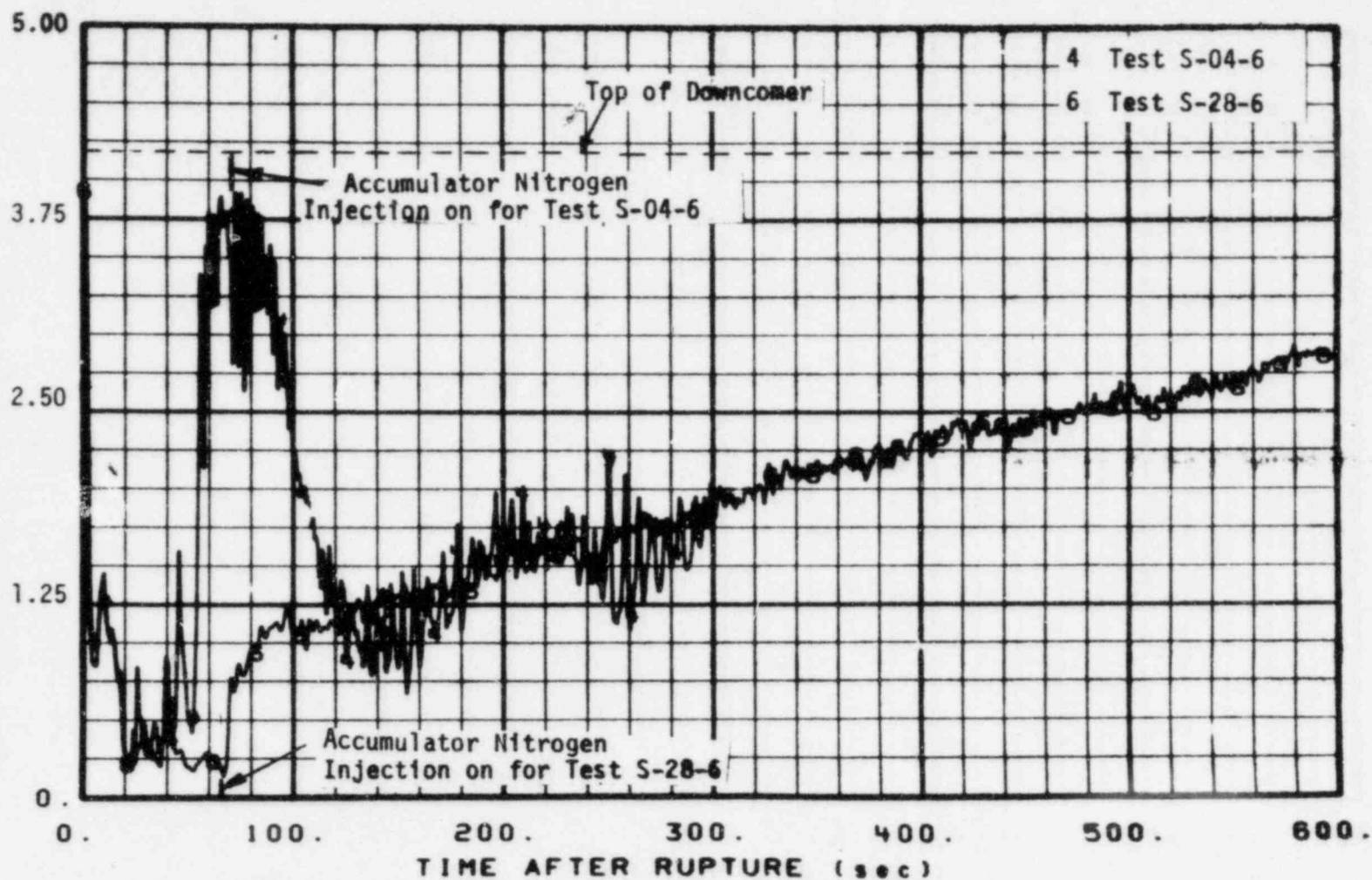


Figure 10. Comparison of Downcomer Collapsed Liquid Level - Tests S-28-6 and S-04-6

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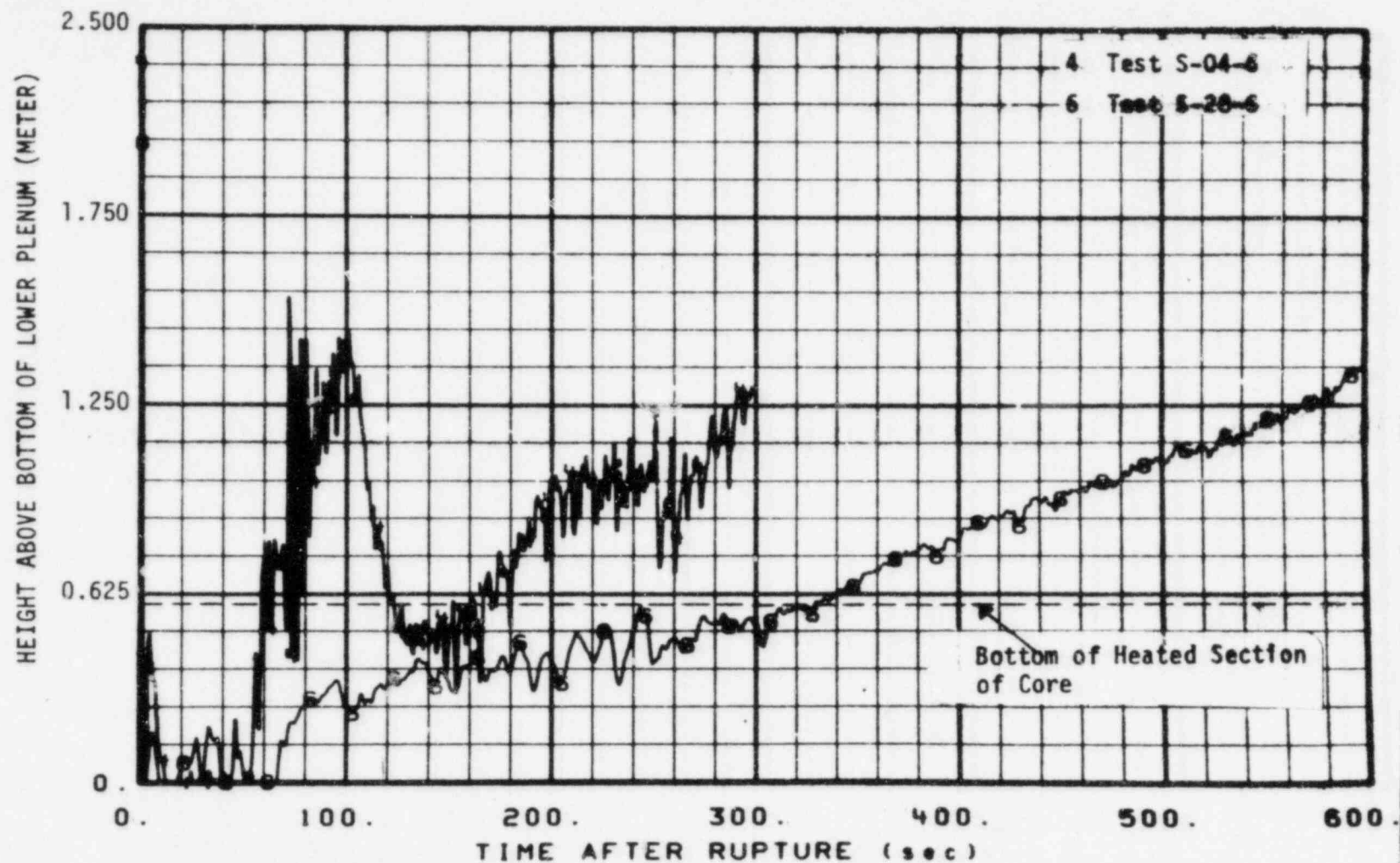


Figure 11. Comparison of Core Collapsed Liquid Level - Tests S-28-6 and S-04-6

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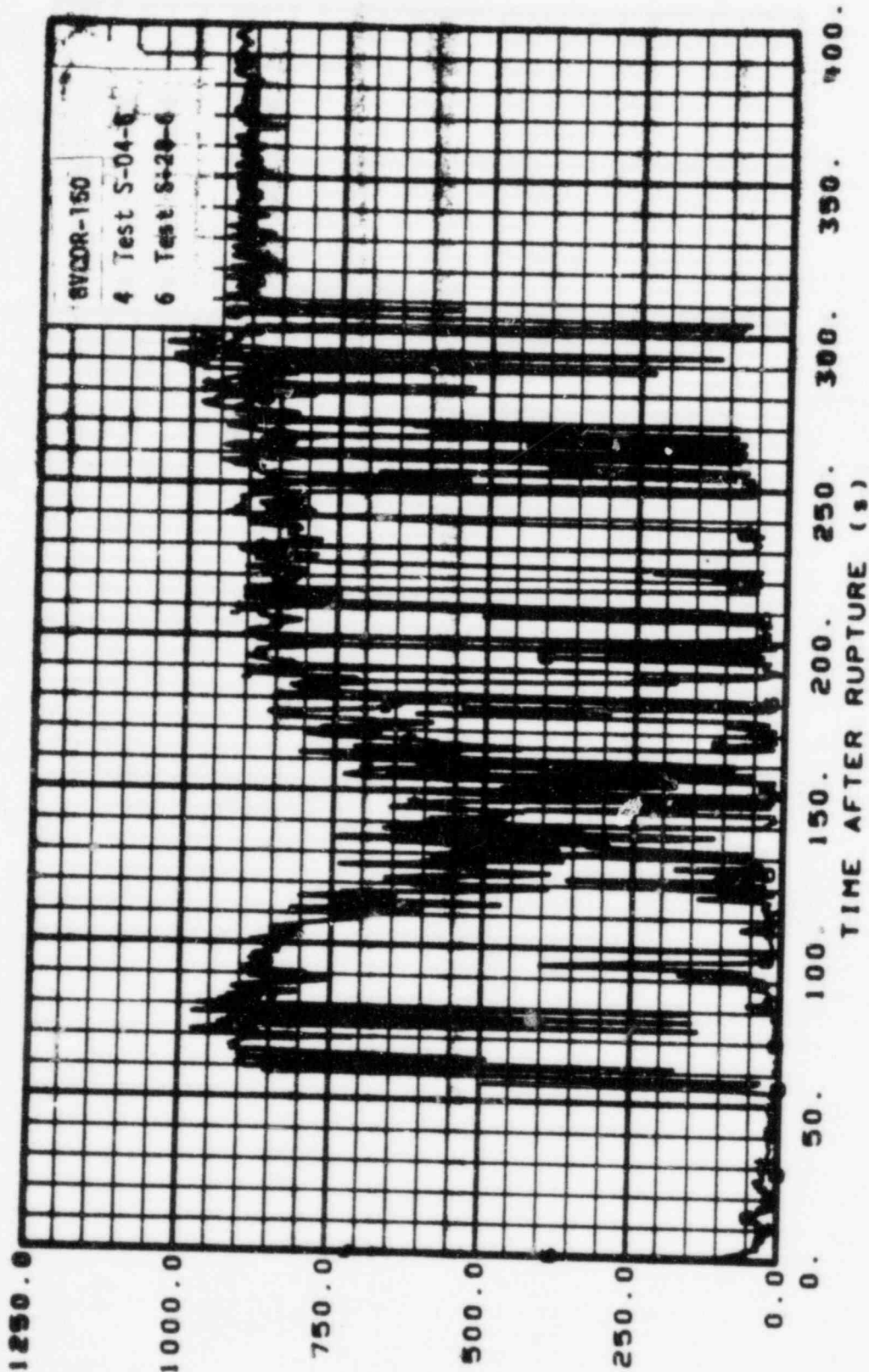


Figure 12. Comparison of Core Inlet Fluid Density -  
Tests S-28-6 and S-04-6

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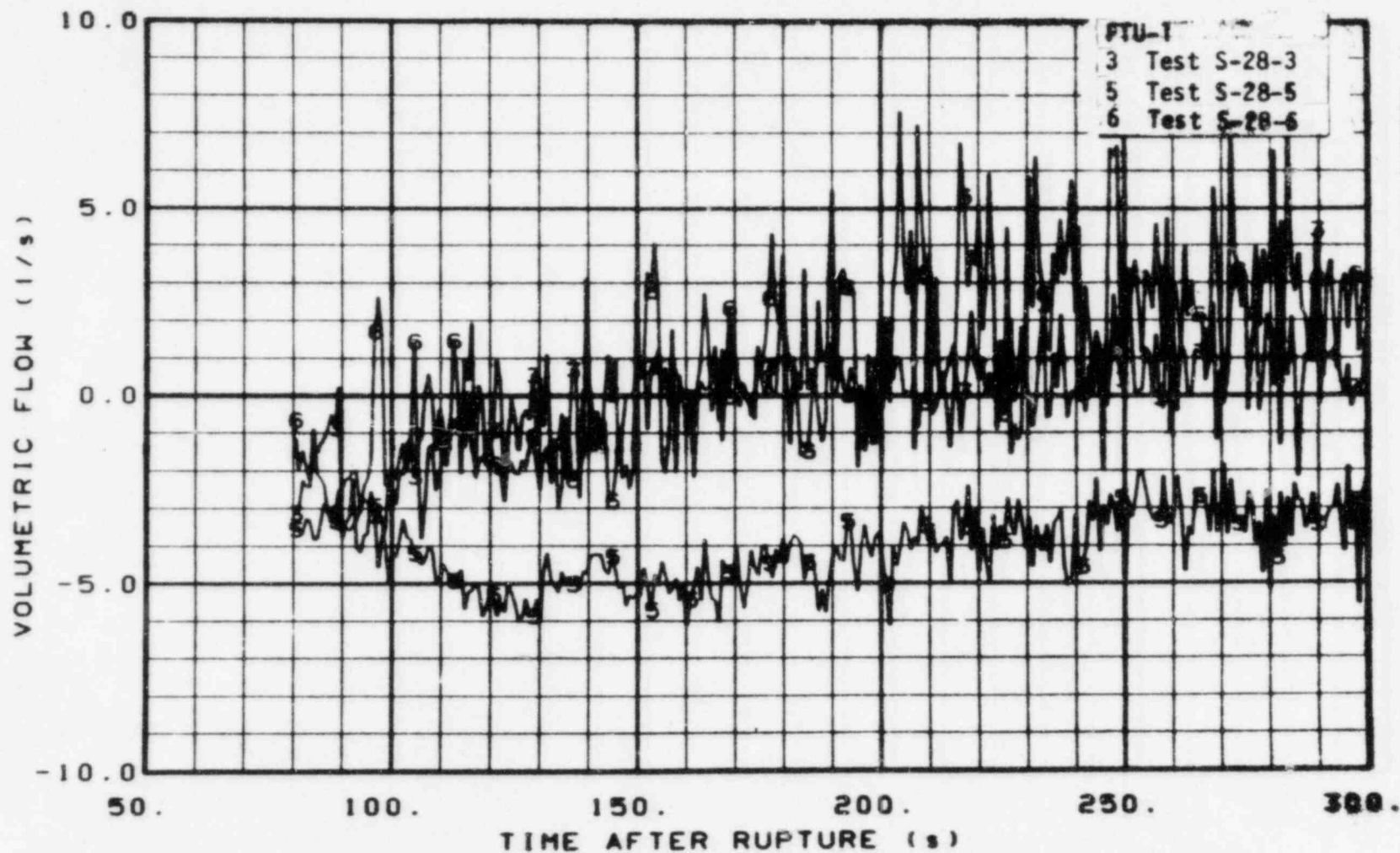


Figure 13. Comparison of Intact Loop Hot Leg Volumetric Flow During Refill - Tests S-28-3, S-28-5 and S-28-6

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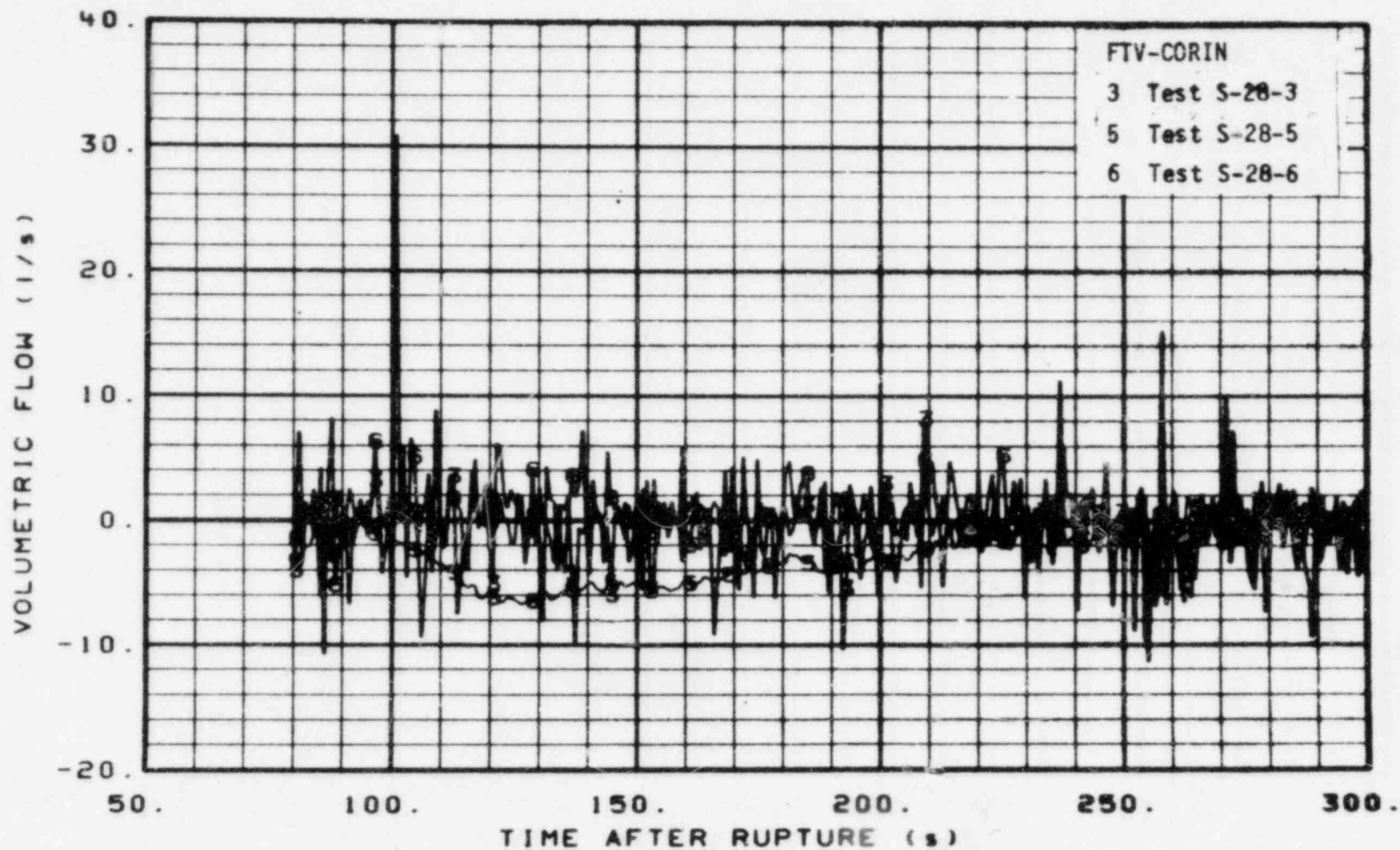


Figure 14. Core Inlet Volumetric Flow During Refill - Tests S-28-3, S-28-5 and S-28-6

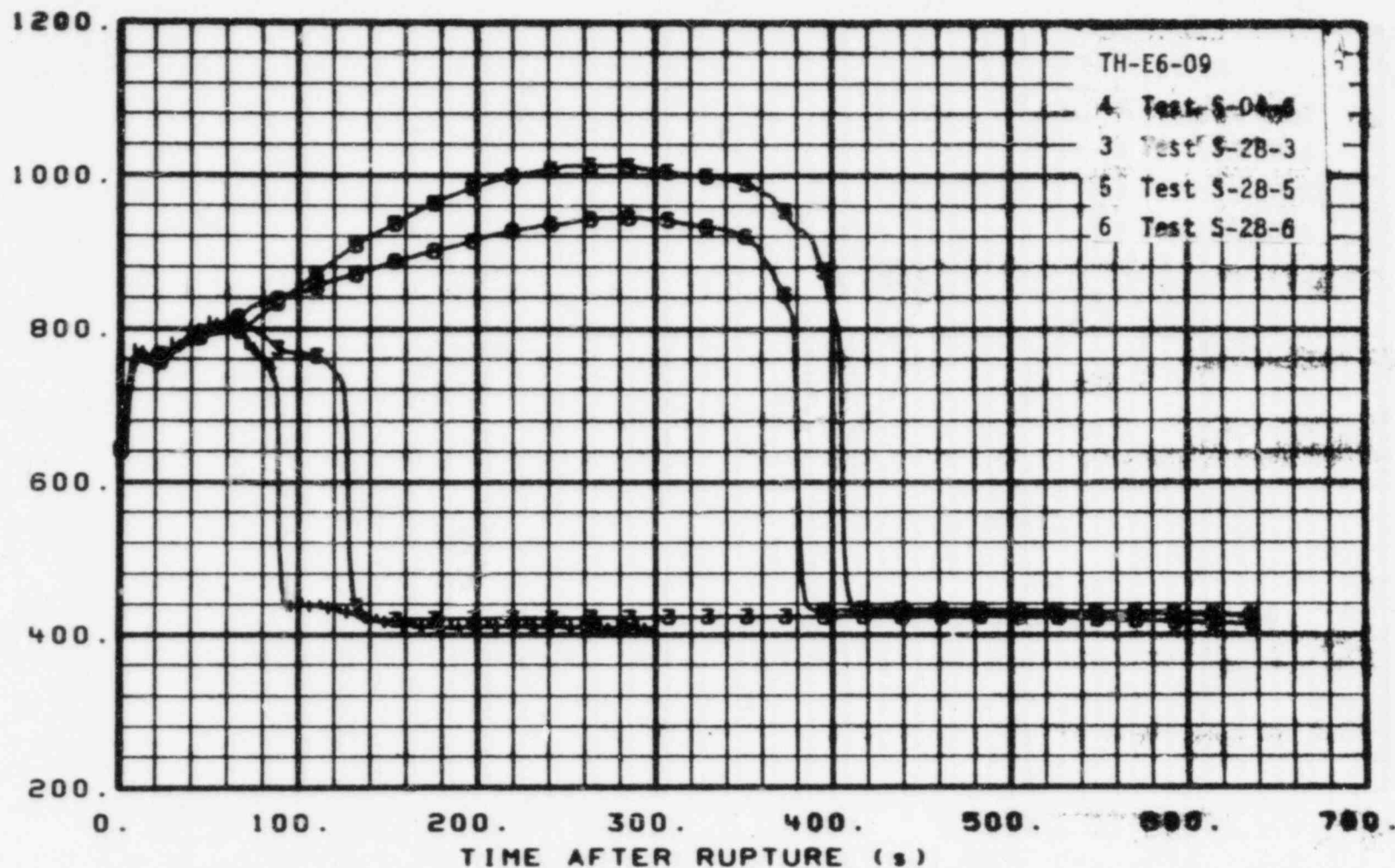
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CORE HEATER TEMPERATURE (K)



PRELIMINARY

Figure 15. Comparison of Rod Cladding Temperatures for Rod E6 at 0.20 m Above Bottom of Heated Core - Tests S-28-3, S-28-5, S-28-6 and S-04-6

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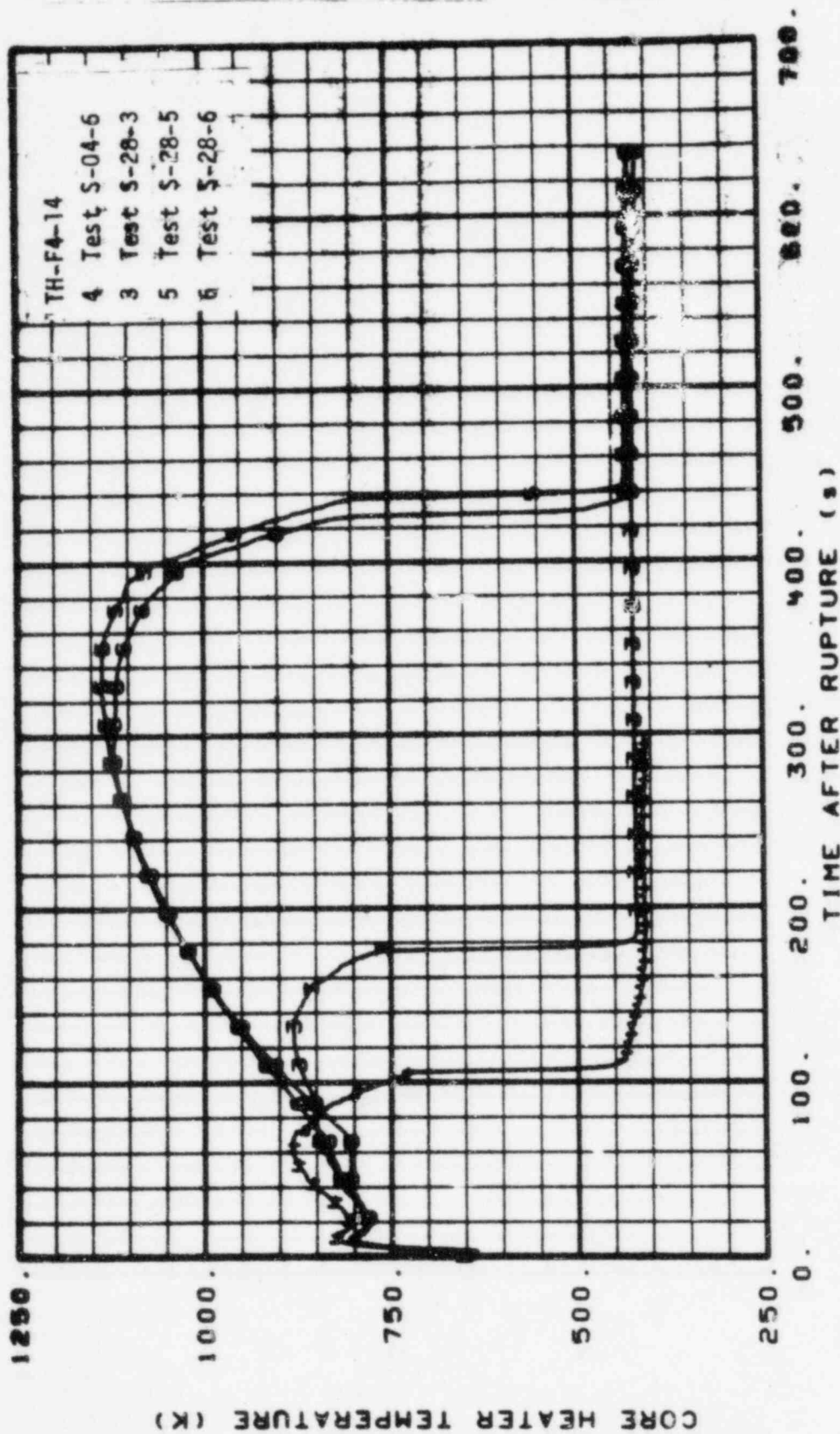
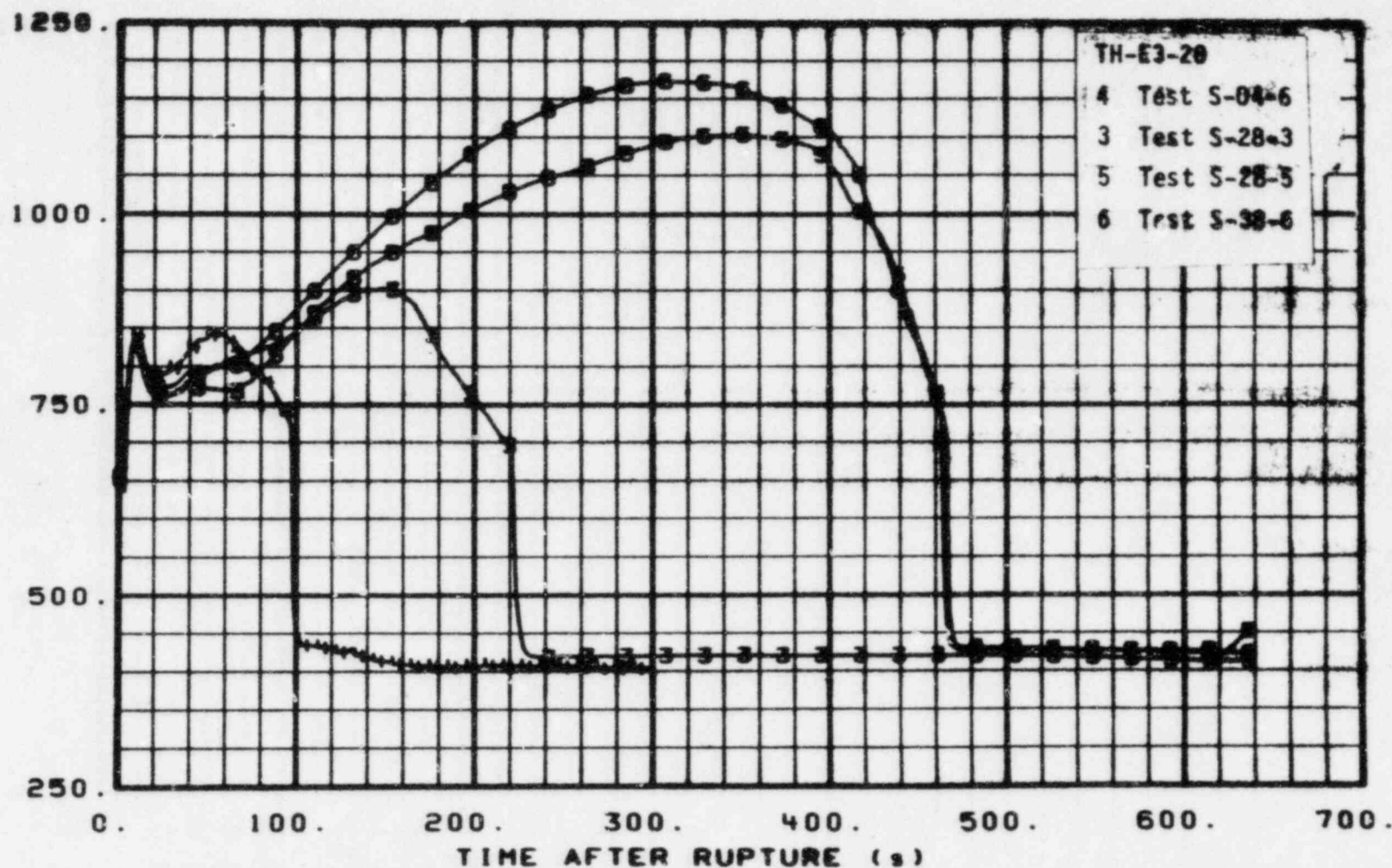


Figure 16. Comparison of Rod Cladding Temperatures for Rod F4 at 0.36 m Above Bottom of Heated Core - Tests S-28-3, S-28-5, S-28-6 and S-04-6

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PRELIMINARY

CORE HEATER TEMPERATURE (K)



PRELIMINARY

Figure 17. Comparison of Rod Cladding Temperatures for Rod E3 at 0.51 m Above Bottom of Heated Core - Tests S-28-3, S-28-5, S-28-6 and S-04-6



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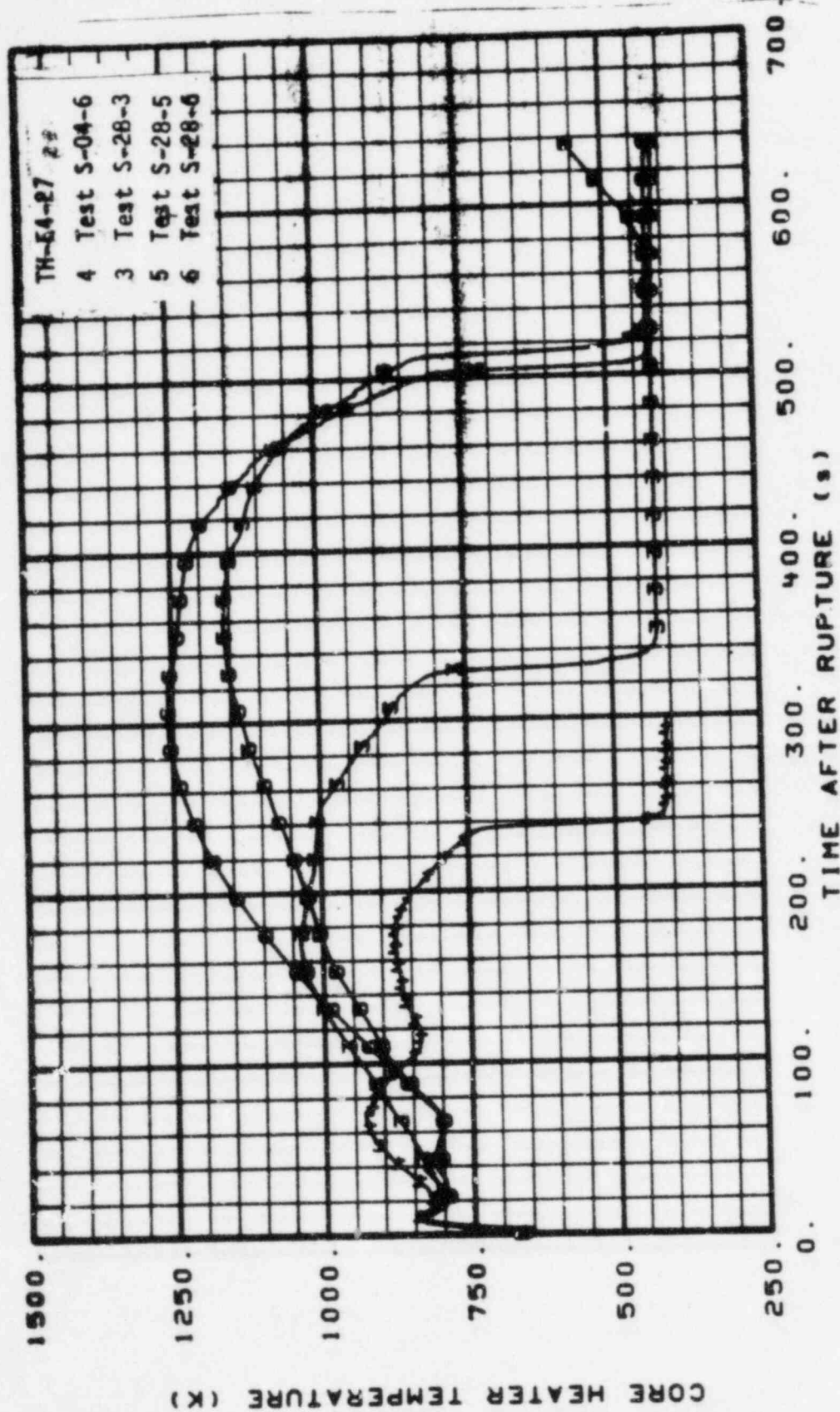


Figure 18. Comparison of Rod Cladding Temperatures on Rod E4 at 0.69 m Above Bottom of Heated Core - Tests S-28-3, S-28-5, S-28-6 and S-04-6

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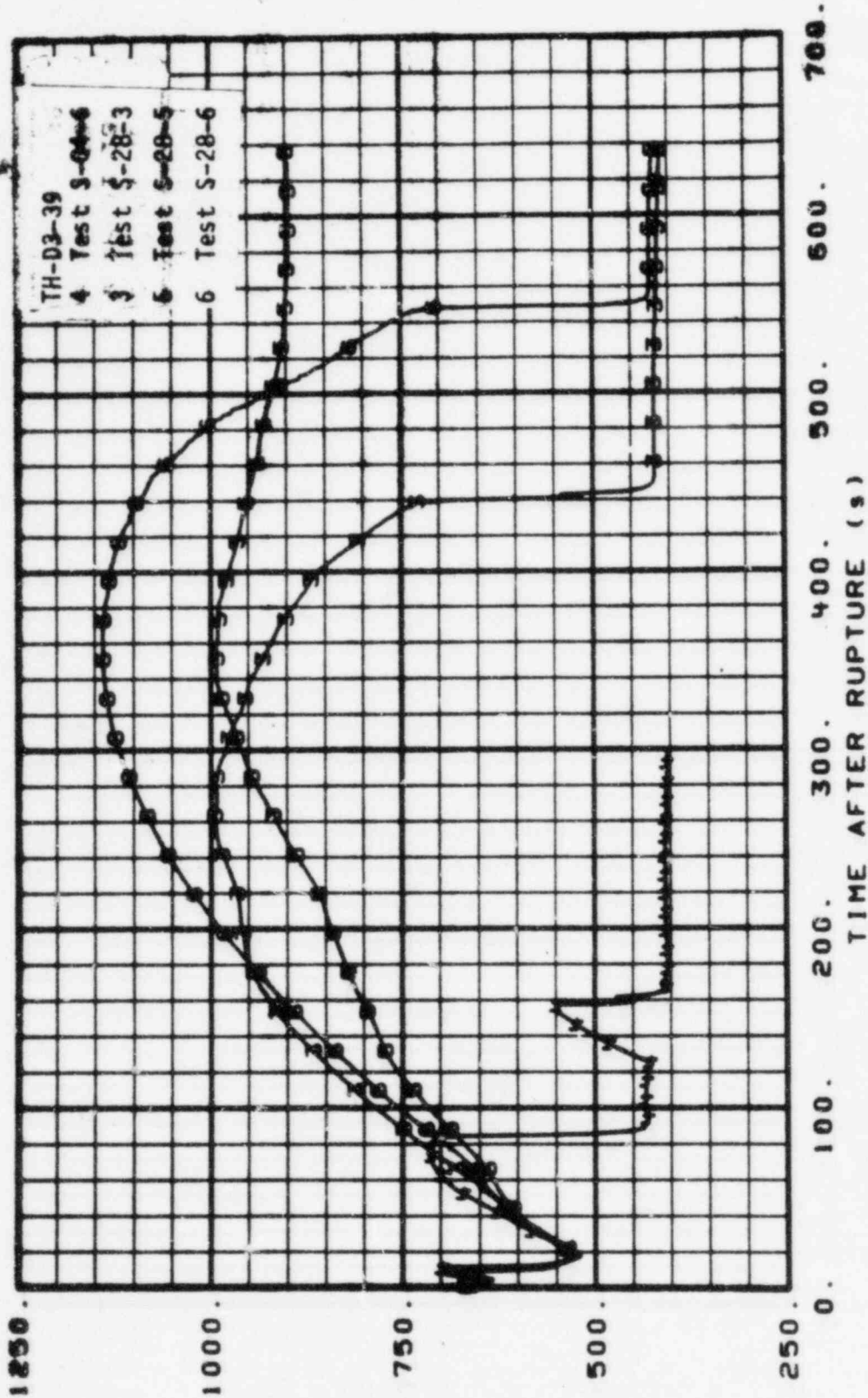


Figure 19. Comparison of Rod Cladding Temperatures on Rod D3  
at 0.99 m Above Bottom of Heated Core - Tests S-28-3,  
S-28-5, S-28-6 and S-04-6

CORE HEATER TEMPERATURE (K)

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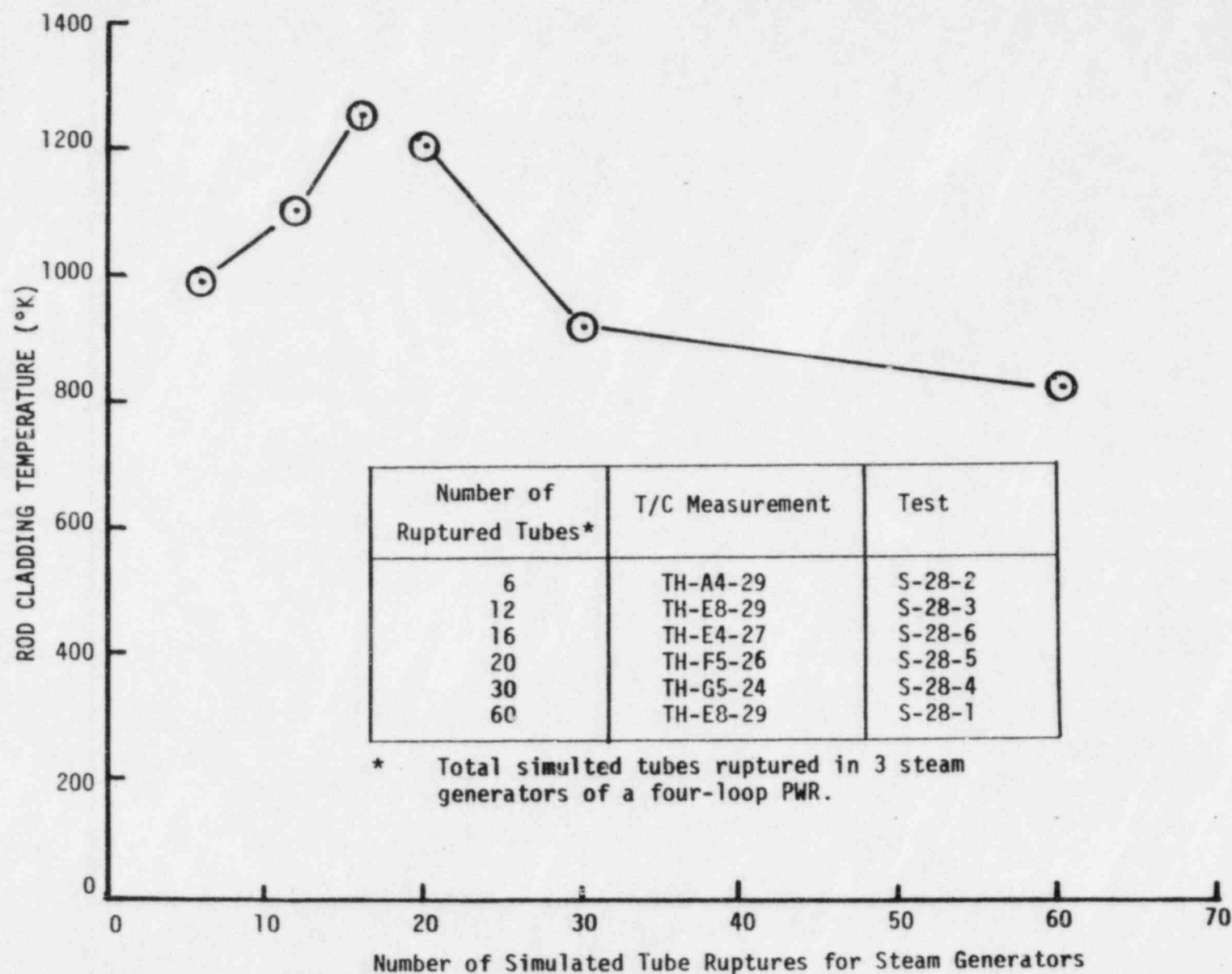


Figure 20. Maximum Temperatures of Core Heater Rod Cladding Resulting from Simulated, Steam Generator Tube Ruptures

September 30, 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 TESTS S-28-7 THROUGH S-28-12 DJO-199-77

- Reference: (1) 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-28-3," (July 6, 1977).
- (2) 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).
- (3) D. J. Olson Ltr to R. E. Tiller, DJO-179-77, "Transmittal of Quick Look Report for Semiscale Mod-1 Steam Generator Tube Rupture Test S-28-5," (August 19, 1977).
- (4) D. J. Olson Ltr to R. E. Tiller, DJO-192-77, "Transmittal of Quick Look Report for Semiscale Mod-1 Steam Generator Tube Rupture Test S-28-6," (September 9, 1977).

Dear Mr. Tiller:

Attached is the Quick Look Report for Semiscale Mod-1 Tests S-28-7 through S-28-12. These integral blowdown-reflood tests were conducted with a break configuration representative of a 200% double-ended offset shear cold leg break, and included the injection of a heated liquid from a pressurized accumulator tank to simulate a steam generator tube rupture flow initiated at the beginning of core reflood. The primary objective of Tests S-28-7 through S-28-12 was to evaluate the effects on the system and core thermal-hydraulic response of simulated steam generator tube rupture flow rates over a range of flow rates which could lead to high rod cladding temperatures. The steam generator secondary-to-primary flow rates for Tests S-28-8, S-28-10 and S-28-11 were relatively small and simulated the flow from the single-ended rupture of a total of 16, 12, and 14 tubes, respectively, in 3 of 4 steam generators in a 4-loop PWR. The secondary-to-primary flowrates in these tests were not of sufficient magnitude to maintain a reverse flow through the core during most of the tube rupture flow period. The secondary-to-primary flow rates for Tests S-28-7, S-28-9, and S-28-12 were relatively large (i.e., of sufficient magnitude to maintain a strong reverse flow through the core for much of the tube rupture injection period), and simulated the flow from the single-ended rupture of a total of 30, 35 and 20 tubes, respectively, in 3 of 4 steam generators in a 4-loop PWR.

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C12

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September 30, 1977  
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An analysis of results from Tests S-28-8, S-28-10, and S-28-11 indicates that with some relatively small steam generator tube rupture flows, initiated at the beginning of core reflood, a period of core flow stagnation can result, which in turn may lead to relatively high cladding temperatures. In Tests S-28-8 and S-28-11, the tube rupture flow rates were of sufficient magnitude to effectively balance the downcomer liquid head during the period between 100 and 300 seconds after rupture. As a result, very little liquid penetrated into the core from the lower plenum and a period of relatively rapid increase in rod cladding temperatures occurred. The rod cladding temperature heatup rates were sufficiently large to cause the cladding temperatures to exceed the high temperature trip point (1255 K) thus causing core power to be tripped at about 260 seconds for Test S-28-8 and at about 262 seconds for Test S-28-11. However, an analysis of the rod cladding temperature response based on the downcomer hydraulic response and the rate of cladding temperature increase at the high temperature locations just prior to the core power trip, indicates that the cladding temperatures would probably not have exceeded 1314 K for Test S-28-8 and 1336 K for Test S-28-11, had the core power not tripped. Both of these temperatures are considerably below the licensing limit of 1478 K. Turnover of rod cladding temperatures would probably have occurred shortly after 300 seconds for these tests had the core power not tripped.

During the period of steam generator tube rupture injection for Tests S-28-7, S-28-9, and S-28-12, the secondary-to-primary flow rates were of sufficient magnitude to maintain a relatively strong flow downward through the core, thus providing relatively good cooling in the core. The core thermal response for these tests was characterized by an early top-down quenching of rods on the side of the core adjacent to the intact loop hot leg, and by considerably delayed bottom-up quenching of rods on the side of the core opposite the intact loop hot leg. The peak cladding temperatures in these tests occurred shortly after the reinitiation of core reflood. The core maximum cladding temperature observed was 1035 K for Test S-28-7, 969 K for Test S-28-9, and 1154 K for Test S-28-12.

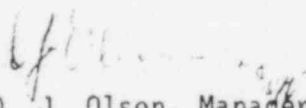
To summarize, results from Tests S-28-7 through S-28-12, as well as from the previous steam generator tube rupture tests, Tests S-28-1 through S-28-6 (References 1, 2, 3, and 4), indicate that steam generator tube ruptures generally do not result in high rod cladding temperatures during loss-of-coolant experiments in the Semiscale Mod-1 system, regardless of whether the ruptures occur at the start of reflood or at the time of refill. However, the specific range of tube rupture flows equivalent to those associated with the single ended rupture of more than 12 but less



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than 20 PWR steam generator tubes does provide a potential for elevated rod cladding temperatures. Tests with tube rupture flows equivalent to the single ended rupture of up to 16 steam generator tubes indicated that, although high cladding temperatures could result, the peak cladding temperatures would be considerably below the 1478 K licensing limit. The narrow band of tube rupture flows equivalent to the single ended rupture of more than 16 but less than 20 PWR steam generator tubes was not explored experimentally and currently remains as a range providing a potential for relatively high rod cladding temperatures. This narrow band covers only about 0.04% of the total number of tubes present in three of the four steam generators in a four-loop PWR.

Very truly yours,

  
D. J. Olson, Manager  
Semiscale Program

JMC:emw

Attachment

cc: R. W. Barber, ERDA  
R. S. Brodsky, ERDA  
W. W. Bixby, NRC - 2  
R. S. Boyd, NRC  
S. Fabric, NRC  
R. B. Foulds, NRC  
R. F. Fraley, ACRS - 21  
S. H. Hanauer, NRC  
G. Kelly, NRC  
S. Levine, NRC  
W. C. Lyon, NRC  
T. G. McCreless, ACRS  
T. E. Murley, NRC  
T. M. Novak, NRC  
D. F. Ross, NRC  
Z. P. Rosztoczy, NRC  
R. M. Scroggins, NRC  
B. Sheron, NRC  
D. E. Solberg, NRC  
V. Stello, NRC

R. L. Tedesco, NRC  
L. S. Tong, NRC - 2  
J. Block, CREARE  
G. F. Brockett, ITI  
D. M. Chapin, MPR  
J. Cudlin, B&W  
R. Denning, BCL  
R. B. Duffey, EPRI  
G. Fader, CE  
G. Farber, IFR  
P. Griffith, MIT  
R. W. Kiehn, EG&G Idaho  
W. Kirchner, LASL  
M. Levenson, EPRI  
W. Loewenstein, EPRI - 2  
P. A. Lottes, ANL  
J. V. Miller, W  
H. P. Pearson, EG&G Idaho - 6  
W. Riebold, JRCE  
H. Seipel, DBF&T  
D. G. Thomas, HNL  
D. Trent, PNL  
R. J. Deers, ERDA-ID