QUICK LOOK REPORT ON SEMISCALE MOD-1
TEST S-28-7 THROUGH TEST S-28-12
STEAM GENERATOR TUBE RUPTURE TEST SERIES

SEMISCALE PROGRAM

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QUICK LOOK REPORT ON SEMISCALE MOD-1

TEST S-28-7 THROUGH TEST S-28-12

STEAM GENERATOR TUBE RUPTURE TEST SERIES

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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 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 simulation of steam generator tube ruptures at the initiation of core reflood. Tests S-28-7 through S-28-12 are 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 (Tests S-28-1 through S-28-6) have provided information of the effect on system response of steam generator tube rupture flows initiated at the beginning of vessel (or downcomer) refill. The primary purpose of Tests S-28-7 through S-28-12 was to provide experimental results of the effect on system response of steam generator tube rupture flows initiated at the beginning of core reflood. A comparison results from Tests S-28-7 through S-28-12 with the previous Series 28 tests will allow an evaluation of the effect of the time at which the steam generator tube ruptures occur on the resulting system and core thermal-hydraulic response.

The test conditions for Tests S-28-7 through S-28-12 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 rates for Tests S-28-8, S-28-10, and S-28-11 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 four-loop PWR, whereas the secondary-to-primary flow rates for Tests S-28-7, S-28-9, and S-28-12 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 four-loop PWR. The steam generator tube rupture flow was begun at the initiation of core reflood (at about 62 seconds after rupture) for each test. 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.

An analysis of results from Tests S-28-8, S-28-10, and S-28-11 indicates that relatively small steam generator tube rupture flows (equivalent to flows from more than 12 but less than 20 single-ended tube ruptures in a

PWR steam generator) initiated at the beginning of core reflood can result in a period of flow stagnation in the core. In Tests S-28-8 and S-28-11, the steam generator tube rupture flow rates were of sufficient magnitude to effectively balance the downcomer liquid head following the intact loop accumulator nitrogen injection. As a result, essentially no liquid penetrated beyond the inlet to the core between approximately 80 and 300 seconds after rupture. [After approximately 300 seconds, the downcomer liquid level (and corresponding liquid head) had increased sufficiently under the influence of the LPIS flow to overcome the effects of the steam generator tube rupture flow and a gradual reflooding of the core occurred]. In Test S-28-10, however, the somewhat smaller tube rupture flow rate combined with a considerably larger liquid inventory in the downcomer (at the initiation of accumulator nitrogen injection) resulted in a substantial quantity of liquid penetrating the heated section of the core shortly following the initiation of the intact loop accumulator nitrogen injection. Thus a period of good core cooling existed for Test S-28-10 shortly following the nitrogen injection period.

The stagnation of ECC fluid near the inlet to the core in Tests S-28-8 and S-28-11 during the early portion of the tube rupture flow period resulted in a relatively rapid increase in the rod cladding temperatures which continued until the core power high temperature trip point (1255 K) was reached. The core power tripped off at about 260 seconds after rupture for Tests S-28-8 and at about 262 seconds after rupture for Test S-28-11. However, analyses have been performed to determine how high t[illegible]adding temperatures might have gone for these tests had the core [illegible]ot tripped. The analyses were based on the downcomer hydraulic respo[illegible]nd the rate of cladding temperature increase at the high temperature locations just prior to the core power trip. Results of these analyses indicate that, had the core power not tripped, the peak cladding temperature in Test S-28-8 would probably not have exceeded 1314 K, and the peak cladding temperature for Test S-28-11 would probably not have exceeded 1336 K. Note that these temperatures are considerably below the licensing limit of 1478 K for PWR fuel rods.

During the period of the simulated steam generator tube rupture flow for Tests S-28-7, S-28-9, and S-28-12, the secondary-to-primary flow was the dominant influence on the overall system and core hydraulic response. The introduction of the steam generator tube rupture flow into the intact loop hot leg, at about 62 seconds after rupture, resulted in a strong flow downward through the core which forced liquid from the inlet to the core and emptied most of the downcomer. The strong reverse flow through the core was maintained until about 350 seconds after rupture for each of the tests. By 350 seconds the downcomer liquid level had increased sufficiently under the influence of the LPIS flow to overcome the effect of the steam generator tube rupture flow and the liquid level in the lower plenum began to increase. As a result, bottom flooding of the core was reinitiated at about 368 seconds for Test S-28-7, 360 seconds for Test S-28-9, and 358 seconds for Test S-28-12.

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The core thermal response to relatively large steam generator tube rupture flows was characterized by 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 preferential top-down quenching of rods on the intact loop hot leg side of the vessel is attributed to the fact that the heater rods in the Semiscale Mod-1 core extend through the upper plenum, thus preventing the penetration of the liquid portion of the steam generator tube rupture flow to the opposite side of the vessel. As a result, the liquid portion of the tube rupture flow passed downward through the core on the side of the vessel adjacent to the intact loop hot leg providing excellent cooling and early quenching of the rods. The delayed bottom-up quenching on rods opposite the intact loop hot leg side of the vessel was a result of the bottom flooding of the core initiated after about 350 seconds. The peak cladding temperatures obtained following the initiation of the tube rupture flow occurred shortly after the reiniation 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 Tests S-28-12.

To summarize, results from Tests S-28-7 through S-28-12 indicate that steam generator tube ruptures initiated at the start of core reflood generally do not result in high rod cladding temperatures during a loss-of-coolant experiment in the Semiscale Mod-1 system. The specific range of tube rupture flows equivalent to those associated with the single ended rupture of more than 12 but less than 20 PWR steam generator tubes however, 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 range 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.

Overall, Test Series 28, defines a very narrow band of tube rupture flow rates with a potential for high rod cladding temperatures in the Semiscale Mod-1 system. The band is the same regardless of whether the tube ruptures occur at the start of vessel refill or at the start of reflood. It extends from a flow rate equivalent to the single ended rupture of 16 PWR steam generator tubes as a low limit, to a flow equivalent to the single ended rupture of 20 PWR steam generator tubes as a high limit. 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.

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 phnomena 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, as well as to the time after rupture at which the steam generator tube ruptures occur. 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 from Tests S-28-7 through S-28-12 of the Semiscale Mod-1 steam generator tube rupture test series. Tests S-28-7 through S-28-12 were 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 into the intact loop hot leg, between the steam generator inlet plenum and the pressurizer. The injection was accomplished using a constant pressure water source, with the water being at a temperature typical of a PWR steam generator secondary fluid. The steam generator tube rupture flow for each test was begun at the initiation of core reflood (at about 60 seconds after the initiation of the cold leg break), and continued for the duration of the test (640 seconds) for Tests S-28-8, S-28-10, S-28-11, and S-28-12, and for 464 and 440 seconds, respectively, for Tests S-28-7 and S-28-9. The injected flow rates for the tests covered a range of flows that were equivalent to those associated with the single-ended rupture of between 12 and 35 tubes in a PWR steam generator. 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 Tests S-28-7 through S-28-12 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 Tests S-28-7 through S-28-12 was similar 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 in Tests S-28-7 through S-28-11, and Rods C3, F2, D5, and F6 in Test S-28-12) 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 Tests S-28-7 through S-28-12 is shown in Figure 4.

The specified initial conditions and operational variables together with the actual test conditions for Tests S-28-7 through S-28-12 are listed in Tables I through VI, respectively. Initial prerupture conditions for each test were compared with the specified prerupture conditions and the differences were judged to be minor. These minor differences in conditions prior to rupture did not significantly influence the postrupture system behavior.

Test Results

The following discussion presents results of a preliminary evaluation of the effect of the simulated steam generator tube ruptures on the system and core thermal-hydraulic response during Tests S-28-7 through S-28-12. In considering the system and core hydraulic response following the initiation of the steam generator tube rupture flow, Tests S-28-7 through S-28-12 can be grouped into two distinct regimes of hydraulic behavior. In Tests S-28-8, S-28-10, and S-28-11, the steam generator secondary-to-primary flow rates were sufficiently small (0.139, 0.104, and 0.122 kg/s, respectively) that positive core flow persisted during much of the tube rupture injection period. As a result, bottom reflood of the core was initiated relatively early during the tube rupture injection period for these tests. In Tests S-28-7, S-28-9, and S-28-12, however, the relatively large steam generator secondary-to-primary flow rates (0.261, 0.305, and 0.174 kg/s, respectively) resulted in a strong

TABLE I

TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-7

Primary System	Specified Condition	Test Condition
Core Power (AMPCOR-T) (VOLTCOR-T) (MW)	1.44	1.41
System pressure (PV+10) (kPa, gage)	15513 ± 172	15582
Loop temperature		
Intact loop cold leg (RBU-14 (K)	557.8 ± 1	555
Intact loop hot leg (RBU-2) (K)	594.4 ± 1	571
Broken loop hot leg (TFB-30) (K)	591.7 ± 3	590
Core flow rate (FTV-COREIN) L/min	As required to obtain core ΔT	557
Pressure suppression system		
Tank water temperature (TF-PSS-33) (K)	Ambient	292
Tank water pressure (P-PSS) (kPa, gage)	155 ± 7	173
Pressurizer water (DPU-PRESLL) (kg)	9.07	9.1
Steam generator feedwater temperature (TFU-SGFW) (K)	497 ± 6	486
Steam generator secondary liquid level (DPU-SG-SEC) (m)	2.95 ± .05	2.69

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TABLE I (contd)

ECC System	Specified Condition	Test Condition
Accumulator CI-T-1		
Injection location	Intact loop cold leg (Spool piece 14)	
Actuation Pressure (kPa, gage)	4137	4209
Liquid volume (L)	80.1	91
Injection rate (FTU-ACC 1) (L/min)	87	82
N_2 flow duration (s)	24	32
Accumulator CI-T-2		
Location	Broken loop cold leg (Spool piece 42)	
Actuation Pressure (kPa, gage)	4137	4284
Liquid volume (L)	16.4	12.7
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	538
Initial pressure (PU-SG3-T) (kPa, gage)	7584	7944
Liquid volume (L)	144.4	not available
Injection rate (FTU-SGS-H) (L/min)	21.3	20

PRELIMINARY

TABLE I (contd)

ECC System	Specified Condition	Test Condition
Air actuated valve		
Open (seconds after rupture)	62	62
Close (seconds after rupture)	467	463
Steam generator secondary fluid discharge		
Initial liquid level (cm)	295	289
Flow rate (L/min)	21.3	not available
Air actuated valve		
Opening time (sec)	62	62
Closing time (sec)	467	not available
Intact loop LPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	1034	999
Injection rate (FTB-LPIS) (L/min)	15.1	17.3
Broken loop LPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	1034	1061
Injection rate (FTB-LPIS) (L/min)	3.6	4.2

TABLE I (contd)

ECC System	Specified Condition	Test Condition
Intact loop HPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	12411	15712
Injection rate (FTU-HPIS) (L/min)	1.17	1.4
Broken loop HPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	12411	15543
Injection rate (FTB-HPIS) (L/min)	0.38	1.13

TABLE II

TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-8

Primary System	Specified Condition	Test Condition
Core Power (AMPCOR-T) (VOLTCOR-T) (MW)	1.44	1.42
System pressure (PV+10) (kPa, gage)	15513 ± 172	15538
Loop temperature		
Intact loop cold leg (RBU-14 (K)	557.8 ± 1	556.6
Intact loop hot leg (RBU-2) (K)	594.4 ± 1	593.8
Broken loop hot leg (TFB-30) (K)	591.7 ± 3	691.6
Core flow rate (FTV-COREIN) L/min	As required to obtain core ΔT	539.4
Pressure suppression system		
Tank water temperature (TF-PSS-33) (K)	Ambient	292
Tank water pressure (P-PSS) (kPa, gage)	155 ± 7	152
Pressurizer water (DPU-PRESLL) (kg)	9.07	8
Steam generator feedwater temperature (TFU-SGFW) (K)	497 ± 6	485
Steam generator secondary liquid level (DPU-SG-SEC) (cm)	295 ± 5	287

TABLE II (contd)

ECC System	Specified Condition	Test Condition
Accumulator CI-T-1		
Injection location	Intact loop cold leg (Spool piece 14)	
Actuation Pressure (kPa, gage)	4137	4203
Liquid volume (L)	80.1	82.1
Injection rate (FTU-ACC-1) (L/min)	87	94
N_2 flow duration (sec)	24	26
Accumulator CI-T-2		
Location	Broken loop cold leg (Spool piece 42)	
Actuation Pressure (kPa, gage)	4137	4203
Liquid volume (L)	16.4	12.8
Injection rate (FTB-ACC2) (L/min)	28.65	29.45
Accumulator CI-T-3 (Steam generator secondary simulator)		
Injection location	Intact loop hot leg (Spool piece 6)	
Temperature (TFU-SGS3-B) (K)	547	532
Initial pressure (PU-SG3-T) (kPa, gage)	7584	7930
Liquid volume (L)	144.4	not available
Injection rate (FTU-SGS-H) (L/min)	11.4	10.96

PRELIMINARY

TABLE II (contd)

ECC System	Specified Condition	Test Condition
Air actuated valve		
Open (seconds after rupture)	62	62
Close (seconds after rupture)	open	open
Steam generator secondary fluid discharge		
Initial liquid level (cm)	295	287
Flow rate (L/min)	11.4	not available
Air actuated valve		
Opening time (sec)	62	62
Closing time (sec)	Remains open	not available
Intact loop LPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	1034	1281
Injection rate (FTB-LPIS) (L/min)	15.1	16.2
Broken loop LPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	1034	902
Injection rate (FTB-LPIS) (L/min)	3.6	4.3

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TABLE II (contd)

ECC System	Specified Condition	Test Condition
Intact loop HPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	12411	15881
Injection rate (FTU-HPIS) (L/min)	1.17	1.5
Broken loop HPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	12411	15578
Injection rate (FTB-HPIS) (L/min)	0.38	1.06

PRELIMINARY

TABLE III

TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-9

Primary System	Specified Condition	Test Condition
Core Power (AMPCOR-T) (VOLTCOR-T) (MW)	1.44	1.42
System pressure (PV+10) (kPa, gage)	15513 ± 172	15517
Loop temperature		
Intact loop cold leg (RBU-14 (K)	557.8 ± 1	556.6
Intact loop hot leg (RBU-2) (K)	594.4 ± 1	593.9
Broken loop hot leg (TFB-30) (K)	591.7 ± 3	592
Core flow rate (FTV-COREIN) L/min	As required to obtain core ΔT	534
Pressure suppression system		
Tank water temperature (TF-PSS-33) (K)	Ambient	292
Tank water pressure (P-PSS) (kPa, gage)	155 ± 7	153
Pressurizer water (DPU-PRESLL) (kg)	9.07	8.9
Steam generator feedwater temperature (TFU-SGFW) (K)	497 ± 6	491
Steam generator secondary liquid level (DPU-SG-SEC) (cm)	295 ± 5	275

PRELIMINARY

TABLE III (contd)

ECC System	Specified Condition	Test Condition
Accumulator CI-T-1		
Injection location	Intact loop cold leg (Spool piece 14)	
Actuation Pressure (kPa, gage)	4137	4154
Liquid volume (L)	80.1	79.6
Injection rate (FTU-ACC-1) (L/min)	87	98
N_2 flow duration (sec)	24	26
Accumulator CI-T-2		
Location	Broken loop cold leg (Spool piece 42)	
Actuation Pressure (kPa, gage)	4137	4113
Liquid volume (L)	16.4	12.4
Injection rate (FTB-ACC2) (L/min)	28.65	30
Accumulator CI-T-3 (Steam generator secondary simulator)		
Injection location	Intact loop hot leg (Spool piece 6)	
Temperature (TFU-SGS3-B) (K)	547	514
Initial pressure (PU-SG3-T) (kPa, gage)	7584	7638
Liquid volume (L)	144.4	157
Injection rate (FTU-SGS-H) (L/min)	24.9	25

PRELIMINARY

PRELIMINARY

TABLE III (contd)

ECC System	Specified Condition	Test Condition
Air actuated valve		
Open (seconds after rupture)	62	62
Close (seconds after rupture)	409	open
Steam generator secondary fluid discharge		
Initial liquid level (cm)	295	275
Flow rate (L/min)	24.9	N/A
Air actuated valve		
Opening time (sec)	62	62
Closing time (sec)	409	N/A
Intact loop LPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	1034	1124
Injection rate (FTB-LPIS) (L/min)	15.1	17
Broken loop LPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	1034	970
Injection rate (FTB-LPIS) (L/min)	3.6	4.3

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TABLE III (contd)

ECC System	Specified Condition	Test Condition
Intact loop HPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	12411	12400
Injection rate (FTU-HPIS) (L/min)	1.17	1.9
Broken loop HPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	12411	12400
Injection rate (FTB-HPIS) (L/min)	0.38	0.35

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TABLE IV

TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-10

Primary System	Specified Condition	Test Condition
Core Power (AMPCOR-T) (VOLTCOR-T) (MW)	1.44	1.41
System pressure (PV+10) (kPa, gage)	15513 ± 172	15605
Loop temperature		
Intact loop cold leg (RBU-14 (K)	557.8 ± 1	557
Intact loop hot leg (RBU-2) (K)	594.4 ± 1	594
Broken loop hot leg (TFB-30) (K)	591.7 ± 3	592
Core flow rate (FTV-COREIN) L/min	As required to obtain core ΔT	538
Pressure suppression system		
Tank water temperature (TF-PSS-33) (K)	Ambient	285
Tank water pressure (P-PSS) (kPa, gage)	155 ± 7	168
Pressurizer water (DPU-PRESLL) (kg)	9.07	8.15
Steam generator feedwater temperature (TFU-SGFW) (K)	497 ± 6	482
Steam generator secondary liquid level (DPU-SG-SEC) (cm)	295 ± 5	209

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TABLE IV (contd)

ECC System	Specified Condition	Test Condition
Accumulator CI-T-1		
Injection location	Intact loop cold leg (Spool piece 14)	
Actuation Pressure (kPa, gage)	4137	4292
Liquid volume (L)	80.1	82
Injection rate (FTU-ACC-1) (L/min)	87	94
N_2 flow duration (sec)	24	24.8
Accumulator CI-T-2		
Location	Broken loop cold leg (Spool piece 42)	
Actuation Pressure (kPa, gage)	4137	4196
Liquid volume (L)	16.4	12.2
Injection rate (FTB-ACC2) (L/min)	28.65	28
Accumulator CI-T-3 (Steam generator secondary simulator)		
Injection location	Intact loop hot leg (Spool piece 6)	
Temperature (TFU-SGS3-B) (K)	547	535
Initial pressure (PU-SG3-T) (kPa, gage)	7584	7853
Liquid volume (L)	144.4	not available
Injection rate (FTU-SGS-H) (L/min)	8.55	8.51

TABLE IV (contd)

ECC System	Specified Condition	Test Condition
Air actuated valve		
Open (seconds after rupture)	62	62
Close (seconds after rupture)	open	open
Steam generator secondary fluid discharge		
Initial liquid level (cm)	295	209
Flow rate (L/min)	855	not available
Air actuated valve		
Opening time (sec)	62	62
Closing time (sec)	Remains open	open
Intact loop LPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	1034	1292
Injection rate (FTB-LPIS) (L/min)	15.1	17
Broken loop LPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	1034	1064
Injection rate (FTB-LPIS) (L/min)	3.6	4.3

TABLE IV (contd)

ECC System	Specified Condition	Test Condition
Intact loop HPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	12411	15874
Injection rate (FTU-HPIS) (L/min)	1.17	1.32
Broken loop HPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	12411	15564
Injection rate (FTB-HPIS) (L/min)	0.38	1.13

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TABLE V

TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-11

Primary System	Specified Condition	Test Condition
Core Power (AMPCOR-T) (VOLTCOR-T) (MW)	1.44	1.41
System pressure (PV+10) (kPa, gage)	15513 ± 172	15639
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	591
Core flow rate (FTV-COREIN) L/min	As required to obtain core ΔT	541
Pressure suppression system		
Tank water temperature (TF-PSS-33) (K)	Ambient	299
Tank water pressure (P-PSS) (kPa, gage)	155 ± 7	148
Pressurizer water (DPU-PRESLL) (kg)	9.07	8.55
Steam generator feedwater temperature (TFU-SGFW) (K)	497 ± 6	474
Steam generator secondary liquid level (DPU-SG-SEC) (cm)	295 ± 5	283

PRELIMINARY

TABLE V (contd)

ECC System	Specified Condition	Test Condition
Accumulator CI-T-1		
Injection location	Intact loop cold leg (Spool piece 14)	
Actuation Pressure (kPa, gage)	4137	4223
Liquid volume (L)	80.1	31.7
Injection rate (FTU-ACC-1) (L/min)	87	82
N_2 flow duration (sec)	24	29.9
Accumulator CI-T-2		
Location	Broken loop cold leg (Spool piece 42)	
Actuation Pressure (kPa, gage)	4137	4252
Liquid volume (L)	16.4	13.85
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	533
Initial pressure (PU-SG3-T) (kPa, gage)	7584	8232
Liquid volume (L)	144.4	not available
Injection rate (FTU-SGS-H) (L/min)	9.96	9.83

PRELIMINARY

TABLE V (contd)

ECC System	Specified Condition	Test Condition
Air actuated valve		
Open (seconds after rupture)	62	61
Close (seconds after rupture)	open	open
Steam generator secondary fluid discharge		
Initial liquid level (cm)	295	283
Flow rate (L/min)	9.96	not available
Air actuated valve		
Opening time (sec)	62	61
Closing time (sec)	Remains open	open
Intact loop LPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	1034	1585
Injection rate (FTB-LPIS) (L/min)	15.1	17.4
Broken loop LPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	1034	990
Injection rate (FTB-LPIS) (L/min)	3.6	4.35

PRELIMINARY

PRELIMINARY

TABLE V (contd)

ECC System	Specified Condition	Test Condition
Intact loop HPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	12411	15976
Injection rate (FTU-HPIS) (L/min)	1.17	1.06
Broken loop HPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	12411	15609
Injection rate (FTB-HPIS) (L/min)	0.38	1.06

PRELIMINARY

PRELIMINARY

TABLE VI

TEST AND PRERUPTURE CONDITIONS FOR TEST S-28-12

Primary System	Specified Condition	Test Condition
Core Power (AMPCOR-T) (VOLTCOR-T) (MW)	1.44	1.42
System pressure (PV+10) (kPa, gage)	15513 ± 172	15585
Loop temperature		
Intact loop cold leg (RBU-14 (K)	557.8 ± 1	559
Intact loop hot leg (RBU-2) (K)	594.4 ± 1	593
Broken loop hot leg (TFB-30) (K)	591.7 ± 3	592
Core flow rate (FTV-COREIN) L/min	As required to obtain core ΔT	542
Pressure suppression system		
Tank water temperature (TF-PSS-33) (K)	Ambient	294
Tank water pressure (P-PSS) (kPa, gage)	155 ± 7	150
Pressurizer water (DPU-PRESLL) (kg)	9.07	8.3
Steam generator feedwater temperature (TFU-SGFW) (K)	497 ± 6	480
Steam generator secondary liquid level (DPU-SG-SEC) (cm)	295 ± 5	228

PRELIMINARY

PRELIMINARY

TABLE VI (contd)

ECC System	Specified Condition	Test Condition
Accumulator CI-T-1		
Injection location	Intact loop cold leg (Spool piece 14)	
Actuation Pressure (kPa, gage)	4137	4192
Liquid volume (L)	80.1	81.6
Injection rate (FTU-ACC-1) (L/min)	87	85
N_2 flow duration (sec)	24	27
Accumulator CI-T-2		
Location	Broken loop cold leg (Spool piece 42)	
Actuation Pressure (kPa, gage)	4137	4193
Liquid volume (L)	16.4	13
Injection rate (FTB-ACC2) (L/min)	28.65	22
Accumulator CI-T-3 (Steam generator secondary simulator)		
Injection location	Intact loop hot leg (Spool piece 6)	
Temperature (TFU-SGS3-B) (K)	547	537
Initial pressure (PU-SG3-T) (kPa, gage)	7584	8071
Liquid volume (L)	144.4	not available
Injection rate (FTU-SGS-H) (L/min)	14.26	14.36

PRELIMINARY

PRELIMINARY

TABLE VI (contd)

ECC System	Specified Condition	Test Condition
Air actuated valve		
Open (seconds after rupture)	62	62
Close (seconds after rupture)	open	open
Steam generator secondary fluid discharge		
Initial liquid level (cm)	295	228
Flow rate (L/min)	14.26	not available
Air actuated valve		
Opening time (sec)	62	62
Closing time (sec)	Remains open	open
Intact loop LPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	1034	1072
Injection rate (FTB-LPIS) (L/min)	15.1	17.4
Broken loop LPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	1034	1378
Injection rate (FTB-LPIS) (L/min)	3.6	4.3

PRELIMINARY

TABLE VI (contd)

ECC System	Specified Condition	Test Condition
Intact loop HPIS		
Location	Cold leg (Spool piece 14)	
Actuation pressure (kPa, gage)	12411	1378
Injection rate (FTU-HPIS) (L/min)	1.17	11.34
Broken loop HPIS		
Location	Cold leg (Spool piece 42)	
Actuation pressure (kPa, gage)	12411	10397
Injection rate (FTB-HPIS) (L/min)	0.38	1.28

PRELIMINARY

reverse core flow, thus preventing the initiation of core reflood until late in the tube rupture flow period. Therefore, to facilitate a complete discussion of the effects of the steam generator secondary-to-primary flow on the system and core thermal-hydraulic response, results from Tests S-28-8, S-28-10, and S-28-11, and results from Tests S-28-7, S-28-9, and S-28-12 are presented in separate sections. The first section discusses the overall system hydraulic response and core thermal response for Tests S-28-8, S-28-10, and S-28-11, while the second section provides a similar discussion for Tests S-28-7, S-28-9, and S-28-12.

System and Core Response During Tests S-28-8, S-28-10, and S-28-11

The primary objective of Tests S-28-8, S-28-10, and S-28-11 was to determine the effect on the system and core thermal-hydraulic response, during a large break loss-of-coolant experiment, of relatively small simulated steam generator tube rupture flows initiated at the beginning of core reflood (at about 62 seconds after rupture). The secondary-to-primary flow rates for Tests S-28-8, S-28-10, and S-28-11 were equivalent to the flow rates associated with the single-ended rupture of a total of 16, 12, and 14 tubes[a], respectively, in a 3 of 4 steam generators in a 4-loop PWR[b]. An analysis of the data for Tests S-28-8, S-28-10, and S-28-11 has been performed to evaluate the effects of the different steam generator secondary-to-primary flow rates 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 effects of the secondary-to-primary flows 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.

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

[b] For comparison, note that the 4-loop Trojan PWR has approximately 3300 tubes in each of the 4 steam generators.

System and Core Hydraulic Response to Relatively Small Steam Generator Tube Rupture Flows Initiated at the Beginning of Core Reflood

The simulated steam generator tube rupture flows for Tests S-28-8, S-28-10, and S-28-11 were initiated at about 62 seconds after rupture and continued for the duration of the tests. During these tests, the steam generator tube rupture flow had a strong influence on the overall system and core hydraulic response. The effects of the tube rupture flow on the system and core hydraulics are illustrated by comparing the core inlet volumetric flow rates, as well as the downcomer and core collapsed liquid levels, for the three tests. Figure 5 compares the core inlet volumetric flow rates for Tests S-28-8, S-28-10, and S-28-11. As indicated in the figure, the core flow became strongly negative at the initiation of the tube rupture flow. The strong flow downward through the core, resulting from the tube rupture flow, was of sufficient magnitude to force liquid from the core inlet and to cause a rapid depletion of liquid in the downcomer[a]. Figure 6 compares the downcomer collapsed liquid level (obtained from a downcomer differential pressure measurement) for Tests S-28-8, S-28-10, and S-28-11, while Figure 7 compares the core collapsed liquid level (obtained from a lower plenum to upper plenum differential pressure measurement) for the same tests. The depletion of liquid from the downcomer and core is indicated by the rapid drop in the downcomer and core collapsed liquid levels at about 62 seconds after rupture. For Tests S-28-8 and S-28-11, the reverse core flow rates were sufficiently large to essentially empty the downcomer prior to the initiation of nitrogen flow from the intact loop accumulator (at about 65 seconds after rupture). The somewhat lower tube rupture flow rate and corresponding smaller reverse flow in the core for Test S-28-10 resulted in a slower rate of removal of liquid from the downcomer, such that only about half of the downcomer liquid inventory had been depleted by the time accumulator nitrogen injection began. Thus, the downcomer liquid inventory at the start of the accumulator nitrogen injection was strongly dependent on the steam generator tube rupture flow rate.

The timing of the initiation of nitrogen injection from the intact loop accumulator had a disproportionately large effect on the system response, because the liquid inventory in the downcomer at the time of the initiation of nitrogen injection, to a large degree, determined the ensuing core hydraulic response. The injection of accumulator nitrogen into the intact loop cold leg stopped the depletion of liquid from the downcomer,

[a] Note that refill of the downcomer and the initiation of core reflood had begun shortly before the initiation of the steam generator tube rupture flow. As a result, a considerable liquid level in the downcomer had been established prior to the initiation of the tube rupture flow.

and forced additional liquid back into the downcomer and lower plenum for each of the tests as indicated in Figures 6 and 7 by the rapid increases in the downcomer and core collapsed liquid levels after about 65 seconds. Because the downcomer had been essentially emptied prior to the nitrogen injection in both Tests S-28-8 and S-28-11, the additional liquid forced into the downcomer by the nitrogen injection did not increase the liquid level in the lower plenum much above the inlet to the core barrel. For Test S-28-10, however, as a result of the somewhat lower tube rupture flow, the liquid inventory available in the downcomer at the start of nitrogen injection was substantially larger than in either Tests S-28-8 or S-28-10. Thus, the additional liquid forced into the downcomer and lower plenum by the nitrogen injection resulted in considerably higher liquid levels in both the downcomer and core.

The injection of accumulator nitrogen into the intact loop cold leg resulted in the reinitiation of core reflood in each of the tests. However, because of the substantial differences in the downcomer liquid inventory at the initiation of accumulator nitrogen injection, the core hydraulic response was considerably different for Test S-28-10 than for Tests S-28-8 and S-28-11. In Test S-28-10, flooding of the core was reinitiated at about 68 seconds and resulted in a substantial liquid level within the heated section of the core that was maintained until about 95 seconds after rupture[a]. In contrast, for Tests S-28-8 and S-28-11 the initial increase in core liquid level did not result in significant penetration of liquid above the bottom of the heated section of the core. In addition, the steam generator tube rupture flows resulted in a very slow rise of liquid level in the core (about 0.17 cm/s for Test S-28-8 and about 0.13 cm/s for Test S-28-11) between approximately 80 and 300 seconds after rupture. As a result, very little liquid penetrated into the core until considerably after 300 seconds for either Tests S-28-8 or S-28-11. The eventual increase in core reflood rate is a result of the gradual increase in the downcomer liquid level under the influence of the intact loop LPIS. After approximately 300 seconds, the downcomer liquid level and corresponding liquid head had increased sufficiently to overcome the effect of the steam generator tube rupture flows.

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[a] The liquid level in the core began to decrease after 95 seconds because of a corresponding decrease in the downcomer liquid level resulting from downcomer mass depletion. The downcomer mass depletion phenomenon in the Semiscale Mod-1 system is due to a nontypically high heat transfer rate from the core barrel and vessel filler walls to the downcomer liquid which results in boiling of the liquid, and is peculiar to the Semiscale Mod-1 system.

Core Thermal Response to Relatively Small Steam Generator Tube Rupture Flows Initiated at the Beginning of Core Reflood

The rupture of relatively small numbers of steam generator tubes at the initiation of core reflood can lead to high cladding temperatures in the Mod-1 system if a relatively prolonged period of core flow stagnation is maintained. The effects of the core hydraulic response on the core thermal response are illustrated by a comparison of rod cladding temperatures at various elevations in the core for Tests S-28-8, S-28-10 and S-28-11. Figures 8 through 11 compare typical rod cladding temperatures at the 0.36, 0.64, 0.74 and 0.99 m elevations in the core for the three tests. As indicated in these figures, the core thermal response for Tests S-28-8 and S-28-11 was very similar following the initiation of the steam generator tube rupture injection. The similarity in core thermal response for the two tests in consistent with the observed downcomer and core hydraulic response. The stagnation of the ECC fluid near the inlet to the core during the early portion of the tube rupture flow period resulted in a relatively rapid increase in the rod cladding temperatures which continued until the core power high temperature trip point (about 1255 K) was reached. The core power was tripped at about 260 seconds after rupture for Test S-28-8 and about 262 seconds after rupture for Test S-28-11.

To determine how high the peak cladding temperatures may have gone for Tests S-28-8 and S-28-11 had core power not tripped, an analysis of the cladding temperature response was performed, based on the downcomer hydraulic response and the rate of cladding temperature rise at the peak temperature locations just prior to the core power trip. Results from a previous test which simulated the rupture of 16 steam generator tubes at the beginning of vessel refill (see Reference 3) indicate that rod cladding temperatures would turn over when the downcomer collapsed liquid level approached the 2.0 m elevation above the bottom of the lower plenum. If the assumption is made that a similar turnover in cladding temperatures would have occurred in Test S-28-8 when the downcomer liquid level reached the 2.0 m elevation (see Figure 6), then the peak cladding temperatures would have been reached at approximately 310 seconds after rupture. The heatup rate at the peak cladding temperature location at the time the core power tripped (260 seconds) was approximately 0.83 K/s. If this rate had continued until 310 seconds, then cladding temperatures would have increased a maximum of 42 K above the cladding temperatures at 260 seconds. Thus, the peak cladding temperature in Test S-28-8 would probably not have exceeded 1314 K had the core power not tripped. A similar analysis for Test S-28-11 indicates that peak cladding temperatures would have increased approximately 58 K higher (giving a peak cladding temperature of 1336 K) had the core power not tripped. Both of these maximum temperatures are considerably below the 1478 K licensing limit. For comparison purposes Table VII lists the peak cladding temperatures following the initiation of tube rupture flow at each thermocouple location in the core for Tests S-28-8, S-28-10, and S-28-11.

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TABLE VII

MAXIMUM CLADDING TEMPERATURES FOLLOWING INITIATION OF STEAM GENERATOR TUBE RUPTURE FLOW FOR TESTS S-28-8, S-28-10, AND S-28-11

THERMOCOUPLE I.D.	TEST S-28-8 (K)	TEST S-28-10 (K)	TEST S-28-11 (K)
E3-05	706	673	679
C7-07	831	819	825
F2-07	781	715	749
E6-08	854	782	823
A4-09	915	843	884
E4-09	891	764	836
G3-13	995	813	968
D2-14	1056	839	1036
D4-14	1057	825	1032
E8-14	1046	885	1036
F4-14	1048	822	1017
G5-14	1042	843	1014
C7-15	1094	896	1084
C4-20	1107	825	1107
D7-20	1144	896	1144
E3-20	1112	777	1097
E5-20	1164	868	1145
F5-20	1111	827	1089
D1-21	1096	948	1085
F2-22	1081	854	1052
E3-24	1207	887	1191
G5-24	1173	906	1132
D6-25	1257	936	1256
E5-25	1232	890	1219
F2-25	1132	871	1110
C4-26	1238	922	1235
D8-26	1164	999	1177
F5-26	1233	920	1208
E4-27	1247	932	1234
C5-28	1272	977	1278
E6-28	1245	941	1238
A4-29	1164	980	1178
A5-29	1166	987	1178
B5-29	1240	1004	1249
B6-29	1211	1001	1221
D3-29	1229	938	1232

PRELIMINARY

PRELIMINARY

TABLE VII (contd)

THERMOCOUPLE I.D.	TEST S-28-8 (K)	TEST S-28-10 (K)	TEST S-28-11 (K)
D4-29	1245	932	1247
D5-29	1198	702	1201
E8-29	1220	1001	1220
F4-29	1224	935	1198
G4-29	1179	947	1126
B3-32	1103	926	1127
H5-32	1079	960	1032
B5-33	1187	991	1210
E1-33	1105	945	1119
E2-33	1172	954	1177
F5-33	1190	957	1189
G4-33	1132	953	1096
E6-37	1131	919	1152
C2-38	989	812	1025
G4-38	1010	896	988
A4-39	951	906	1003
D3-39	1102	870	1128
E7-44	1027	939	1057
F4-44	985	871	994
A5-45	867	873	920
C4-53	863	769	904
C6-53	879	822	930
F5-53	889	782	901
E4-55	911	781	928
D2-61	688	663	697

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Although the tube rupture flow rate for Test S-28-10 was not significantly different than for Tests S-28-8 and S-28-11, the differences in the downcomer and core hydraulic response just prior to and following the accumulator nitrogen injection (as discussed in the previous section) resulted in a considerably less severe core thermal response for Test S-28-10. The penetration of liquid into the core following the initiation of the intact loop accumulator nitrogen flow for Test S-28-10 resulted in a period of good core cooling. As indicated in Figures 8 through 11, core cladding temperatures decreased considerably between about 70 and 120 seconds after rupture, and quenching of the lower and upper portions of the core occurred. The gradual heatup of the core after 120 seconds is a result of the drop in downcomer liquid level and corresponding drop in the core liquid level between about 80 and 100 seconds after rupture as discussed in the previous section. The peak cladding temperature observed in Test S-28-10 during the tube rupture injection period was about 1004 K and occurred on Rod B5 at the 0.74 m elevation.

<u>System and Core Response During Tests S-28-7, S-28-9, and S-28-12</u>

The primary objective of Tests S-28-7, S-28-9, and S-28-12 was to determine the effects on the system and core thermal-hydraulic response during a large break loss-of-coolant experiment of relatively large steam generator tube rupture flows initiated at the beginning of core reflood. The secondary-to-primary flow rates for Tests S-28-7, S-28-9, and S-28-10 were equivalent to the flow rates associated with 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. An analysis of the data for Tests S-28-7, S-28-9, and S-28-12 has provided insight into the phenomena which occur in the Semiscale Mod-1 system as a result of the simulated rupture of comparatively large numbers of steam generator tubes at the initiation of core reflood. The first part of the following discussion is concerned with an evaluation of the overall system and core hydraulic response to the steam generator tube rupture flows. The second section deals primarily with the core thermal response.

<u>System and Core Hydraulic Response to Relatively Large Steam Generator Tube Rupture Flows Initiated at the Beginning of Core Reflood</u>

The simulated steam generator tube rupture flows for Tests S-28-7, S-28-9, and S-28-12 were initiated at about 62 seconds after rupture, and continued for the duration of the test for Test S-28-12, and until 464 and 440 seconds, respectively for Tests S-28-7 and S-28-9. During the period of steam generator tube rupture injection for the three tests, the secondary-to-primary flow was the dominant influence on the overall system and core hydraulic response. A comparison of the downcomer and core liquid levels for the three tests illustrates the effects of the relatively strong secondary-to-primary flow. Figure 12 compares

the downcomer collapsed liquid levels for Tests S-28-7, S-28-9, and S-28-12, while Figure 13 compares the core collapsed liquid levels for the same tests. The introduction of the steam generator tube rupture flow into the intact loop hot leg at about 62 seconds after rupture, forced liquid from the core inlet and resulted in emptying most of the downcomer prior to the initiation of the intact loop accumulator nitrogen flow at about 65 seconds after rupture[a]. The accumulator nitrogen injection into the intact loop cold leg was not of sufficient magnitude to overcome the effect of the relatively large tube rupture flow rates in each of the three tests. As a result, only small amounts of liquid were forced into the vessel lower plenum under the influence of the accumulator nitrogen flow, and essentially no liquid penetrated into the bottom of the heated section of the core. As indicated in Figure 13, the liquid level increased slightly following the accumulator nitrogen injection for each test, but then remained at a relatively constant level considerably below the heated section of the core until about 350 seconds after rupture. By 350 seconds, however, the downcomer liquid level had increased sufficiently (Figure 12) under the influence of the low pressure injection system to overcome the effect of the tube rupture flow. As a result, a gradual increase in the liquid level in the lower plenum occurred, and flooding of the core was reinitiated by about 368 seconds for Test S-28-7, 360 seconds for Test S-28-9, and 358 seconds for Test S-28-12.

Although the strong reverse core flows (resulting from the tube rupture flows) were able to empty the downcomer prior to nitrogen injection in Tests S-28-7, S-28-9 and S-28-12, a gradual refill of the downcomer occurred after about 150 seconds even though the tube rupture flows continued at the same rate. A comparison of the core inlet volumetric flow rates for the three tests provides an indication of the phenomena which result in the refill of the downcomer. The core inlet volumetric flow rates for Tests S-28-7, S-28-9, and S-28-12 are shown in Figure 14. As shown in the figure, the reverse flow through the core became substantially more negative at the initiation of the secondary-to-primary flow and continued strongly negative until the initiation of nitrogen flow from the intact loop accumulator. It was during this period of strong reverse core flow, that the downcomer was essentially emptied. The accumulator nitrogen injection resulted in a considerable reduction in the core flow rates, the reduction being maintained even after the nitrogen flow ceased, about 95 seconds after rupture. The considerably smaller reverse core flow rates after the period of nitrogen injection can be attributed to a change in the loop resistance due to steam flow in the intact loop cold leg. Prior to accumulator nitrogen injection,

[a] As indicated in Figures 12 and 13, partial refill of the vessel downcomer and the initiation of core reflood had occurred prior to the initiation of the secondary-to-primary flow at about 62 seconds after rupture.

the cold leg piping near the accumulator injection point was filled with subcooled liquid. Thus, steam flow from the hot leg side to the cold leg side of the intact loop was limited by the rate of condensation at the liquid-vapor interface near the accumulator injection location. As a result of this flow limitation the steam generator tube rupture flow was able to maintain a strong flow through the alternate path, downward through the core. The effect of the accumulator nitrogen flow, however, was to clear liquid from the intact loop cold leg providing an additional path for removal of steam from the intact loop, therefore reducing the tendency for the rupture flow to penetrate through the core. Thus, the magnitude of the core inlet volumetric flow rates, following the nitrogen injection period, did not increase to the pre-nitrogen injection values, but remained at a considerably reduced negative value until the initiation of core reflood. As a result, the downcomer was able to refill under the influence of the LPIS.

Core Thermal Response to Relatively Large Steam Generator Tube Rupture Flows Initiated at the Beginning of Core Reflood

The core thermal response to relatively large steam generator tube rupture flows initiated at the beginning of core reflood is characterized by relatively early quenching of rods on the side of the core adjacent to the intact loop hot leg and by considerably delayed quenching of rods on the side of the core opposite the intact loop hot leg. Figures 15 through 18 compare typical rod cladding temperatures at the 0.23 and 0.74 m elevations in the core for rods both adjacent to and opposite the intact loop hot leg side of the core. As indicated in the figures, early quenching occurred at both elevations on the rods directly underneath the intact loop hot leg (Figures 15 and 17), while delayed quenching occurred on the rods opposite the intact loop hot leg (Figures 16 and 18). The preferential early quenching of rods adjacent to the intact loop hot leg side of the vessel is attributed to the fact that the rods in the opposite side of the vessel. As a result, the liquid portion of the tube rupture flow passes downward through the core on the side adjacent to the intact loop hot leg providing excellent cooling and early rod quenching.

A comparison of rod quench times for the three tests indicates a top-down quench pattern occurred on those rods adjacent to the intact loop hot leg side of the vessel whereas a bottom-up quench pattern occurred on rods opposite the intact loop hot leg side of the vessel. Table VIII lists the quench times for each thermocouple location in the core for Tests S-28-7, S-28-9, and S-28-12. As indicated in the table, a top-down quench behavior existed for each of the tests on many of the rods near the intact loop hot leg side of the vessel prior to about 300 seconds after rupture. The top-down quench behavior during this period of time resulted from the strong flow downward through the core of a liquid-vapor mixture from the steam generator secondary. The largest number of early quenches occurred in Test S-28-9 which had the largest tube rupture

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TABLE VIII

CORE QUENCH TIMES FOR
TESTS S-28-7, S-28-9, AND S-28-12

THERMOCOUPLE I.D.	TEST S-28-7 (sec)	TEST S-28-9 (sec)	TEST S-28-12 (sec)
E3-05	419	345	420
C7-07	424	266	427
F2-07	429	416	423
E6-08	434	342	437
A4-09	219	155	215
E4-09	436	243	444
G3-13	467	445	472
D2-14	467	313	475
D4-14	261	152	475
E8-14	469	272	476
F4-14	467	445	472
G5-14	470	451	475
C7-15	470	211	480
C4-20	211	126	253
D7-20	481	269	498
E3-20	484	244	500
E5-20	316	189	504
F5-20	482	250	501
D1-21	492	323	503
F2-32	492	468	483
E3-24	397	207	510
G5-24	503	248	510
D6-25	247	167	516
E5-25	288	172	515
F2-25	50[illegible]	477	503
C4-26	188	108	219
D8-26	481	154	546
F5-26	494	195	520
E4-27	247	167	527
C5-28	181	115	193
E6-28	323	193	526
A4-29	151	108	125
A5-29	160	117	138
B5-29	160	108	133
B6-29	177	122	166
D3-29	210	141	529

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TABLE VIII (contd)

THERMOCOUPLE I.D.	TEST S-28-7 (sec)	TEST S-28-9 (sec)	TEST S-28-12 (sec)
D4-29	199	123	529
D5-29	62	84	496
E8-29	496	528	532
F4-29	317	171	532
G4-29	521	210	528
B3-32	157	100	158
H5-32	276	168	247
B5-33	142	93	116
E1-33	541	528	543
E2-33	535	481	544
F5-33	289	163	544
G4-33	518	190	542
E6-37	227	150	551
C2-38	175	111	532
G4-38	326	165	551
A4-39	113	84	95
D3-39	170	120	544
E7-44	319	199	573
F4-44	185	128	575
A5-45	96	78	85
C4-53	53	60	109
C6-53	56	83	120
F5-53	56	120	592
E4-55	60	111	586
D2-61	169	No Quench	605

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flow rate, while the smallest number of early quenches occurred in Test S-28-12 which had the lowest tube rupture flow rate. The considerably higher tube rupture flow rate for Test S-28-9 resulted in better penetration of the heated section of the core (both axially and radially) by the liquid portion of the tube rupture flow which thus caused a significantly higher number of early rod quenches than occurred in either Tests S-28-7 or S-28-12. Once flooding of the core was reinitiated (at about 360 seconds after rupture for each of the tests) a bottom up quench pattern occurred. Quenching of the entire core was accomplished by about 550 seconds for Test S-28-7, 530 seconds for Test S-28-9, and 605 seconds for Test S-28-12.

The effect of an increase in the steam generator tube rupture flow on the peak core thermal response for Tests S-28-7, S-28-9, and S-28-12 was to cause a considerable decrease in the peak cladding temperatures in the core. Table IX lists the peak cladding temperatures for each thermocouple in the core following initiation of the tube rupture flow for the three tests. As indicated in the table, the peak cladding temperatures at most thermocouple locations were the highest for Test S-28-12 in which the tube rupture flow rate was the lowest, and were the lowest for Test S-28-9 which had the highest tube rupture flow rate and correspondingly better cooling. The maximum cladding temperatures observed during the tube rupture flow period were 1035 K for Test S-28-7, 969 K for Test S-28-9, and 1154 K for Test S-28-12.

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-192-77, "Transmittal of Quick Look Report for Semiscale Mod-1 Steam Generator Tube Rupture Test S-28-6," September 9, 1977.

PRELIMINARY

TABLE IX

MAXIMUM CLADDING TEMPERATURES FOLLOWING INITIATION OF STEAM GENERATOR TUBE RUPTURE FLOW FOR TESTS S-28-7, S-28-9, AND S-28-12

THERMOCOUPLE I.D.	TEST S-28-7 (K)	TEST S-28-9 (K)	TEST S-28-12 (K)
E3-05	809	721	956
C7-07	854	828	882
F2-07	844	773	818
E6-08	901	829	1039
A4-09	874	860	854
E4-09	863	792	1077
G3-13	905	821	1033
D2-14	922	842	1011
D4-14	916	816	1022
E8-14	938	888	946
F4-14	968	840	1135
G5-14	975	866	1058
C7-15	968	905	965
C4-20	928	821	882
D7-20	1002	906	1049
E3-20	903	803	1115
E5-20	947	870	1135
F5-20	916	851	1106
D1-21	962	942	974
F2-22	937	863	863
E3-24	964	863	1144
G5-24	1009	908	1115
D6-25	1035	929	1125
E5-25	955	871	1154
F2-25	936	878	889
C4-26	988	847	900
D8-26	939	879	984
F5-27	956	871	1154
E4-27	916	822	1154
C5-28	1005	887	919
E6-28	992	910	1140
A4-29	989	939	928
A5-29	1009	969	946
B5-29	1021	932	927
B6-29	1017	952	945
D3-29	889	769	1055
D4-29	898	769	1058

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TABLE IX (contd)

THERMOCOUPLE I.D.	TEST S-28-7 (K)	TEST S-28-9 (K)	TEST S-28-12 (K)
D5-29	654	598	964
E8-29	1031	948	1040
F4-29	918	811	1135
G4-29	943	841	1057
B3-32	876	728	772
H5-32	882	792	919
B5-33	938	765	799
E1-33	841	793	928
E2-33	858	827	1049
F5-33	854	744	1115
G4-33	873	780	993
E6-37	816	769	1068
C2-38	759	698	846
G4-38	833	753	946
A4-39	836	692	715
D3-39	786	712	991
E7-44	775	729	965
F4-44	744	686	1021
A5-45	778	650	660
C4-53	563	578	576
C6-53	578	594	595
F5-53	570	591	891
E4-55	577	598	845
D2-61	545	547	661

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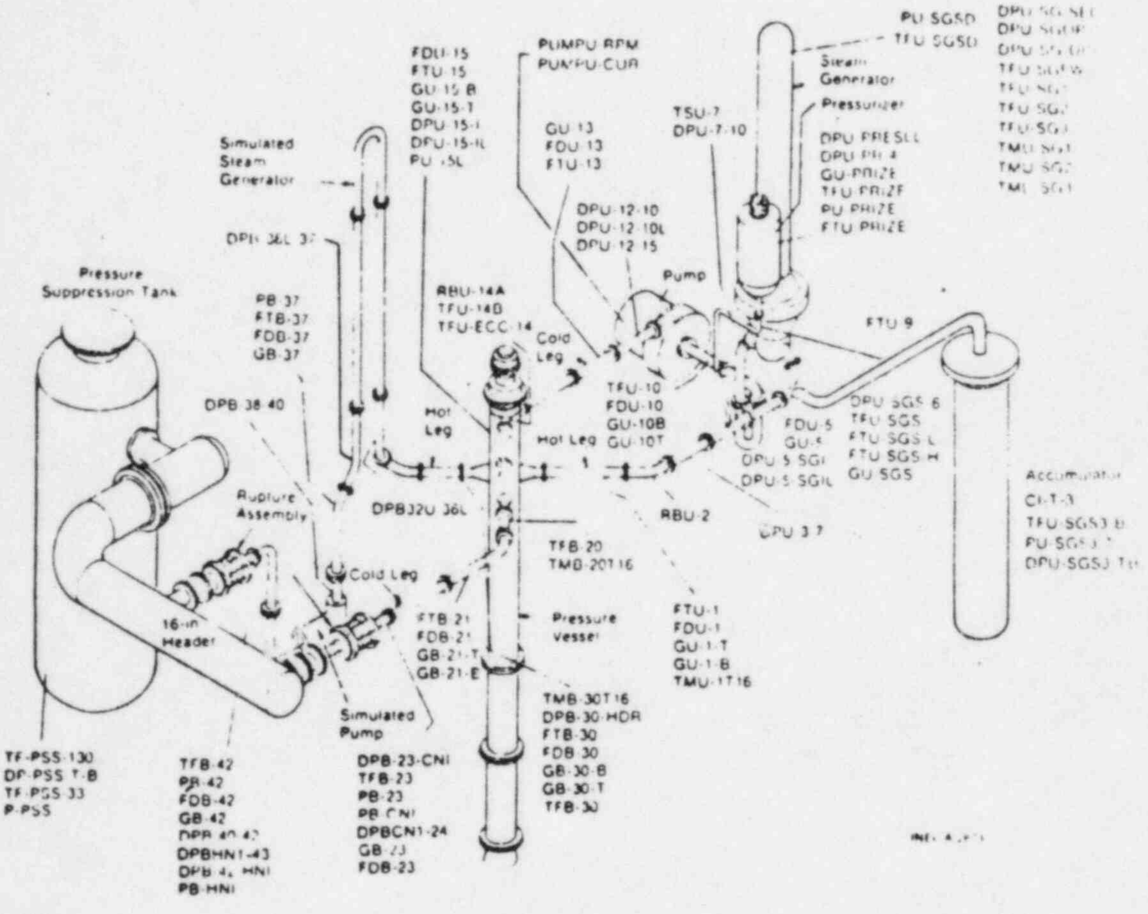

Figure 1. Semiscale Mod-1 System and Instrumentation for Cold Leg Break Configuration - Isometric

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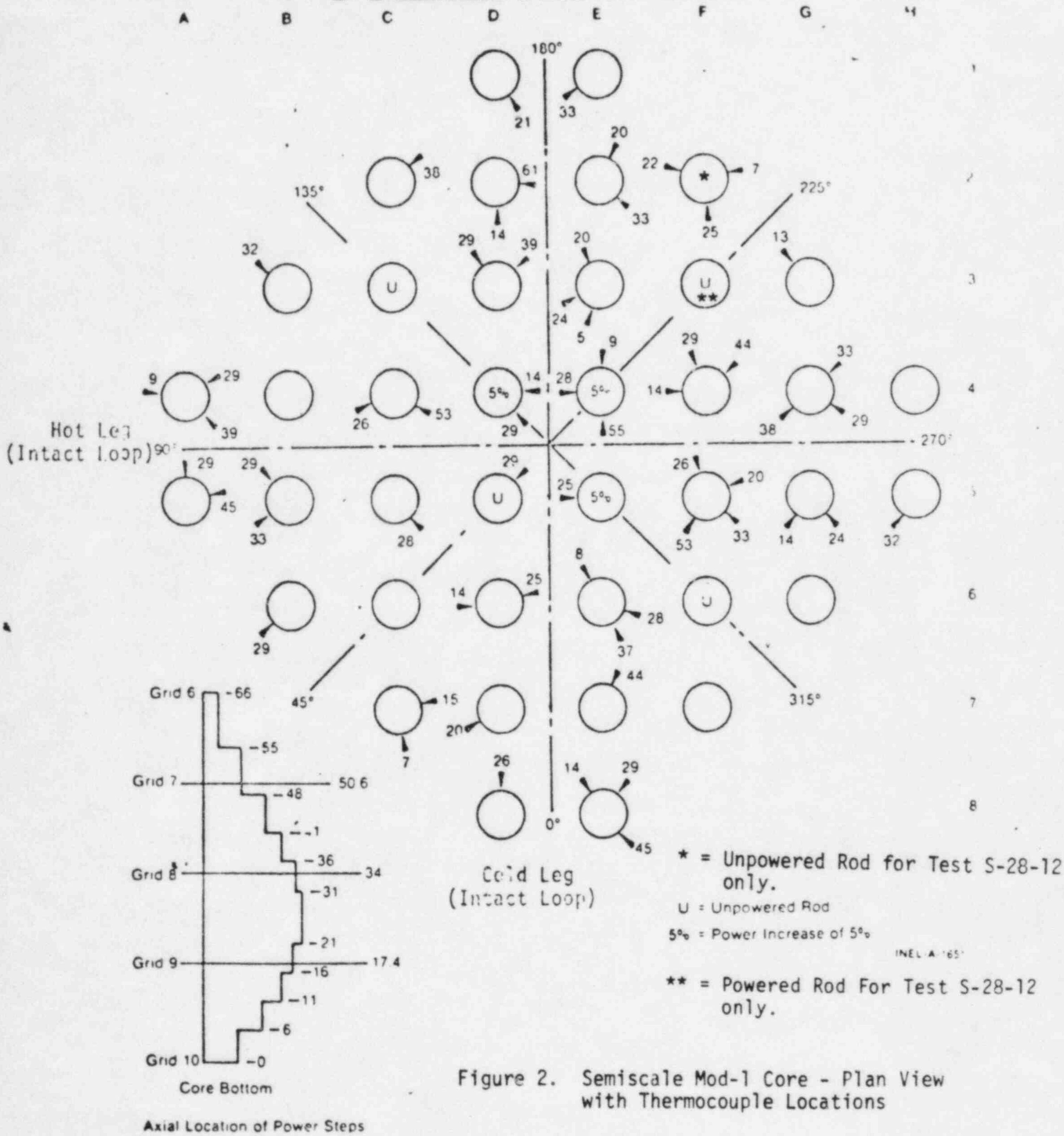

Figure 2. Semiscale Mod-1 Core - Plan View with Thermocouple Locations

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PRELIMINARY

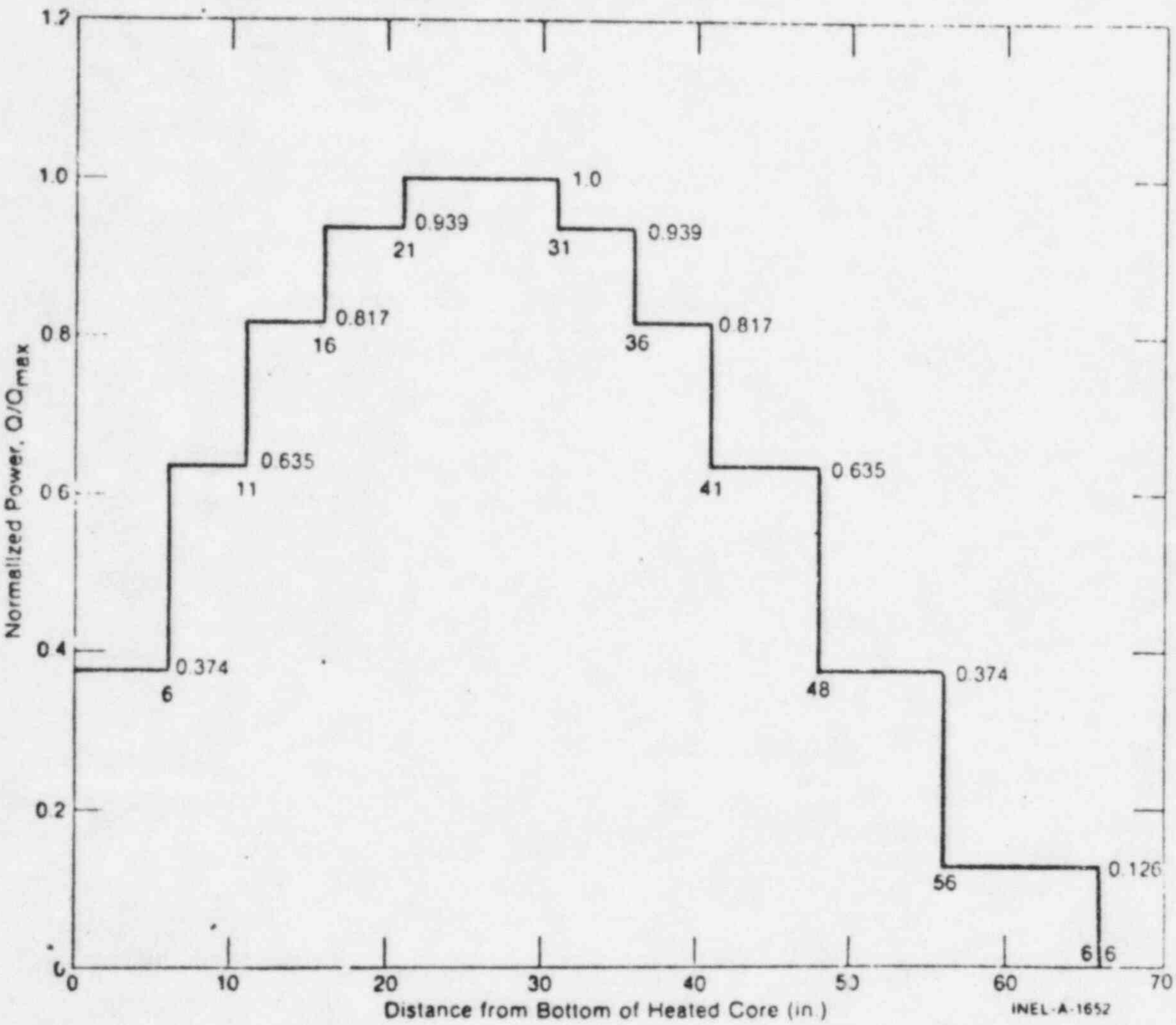

Figure 3. Semiscale Mod-1 Axial Power Profile

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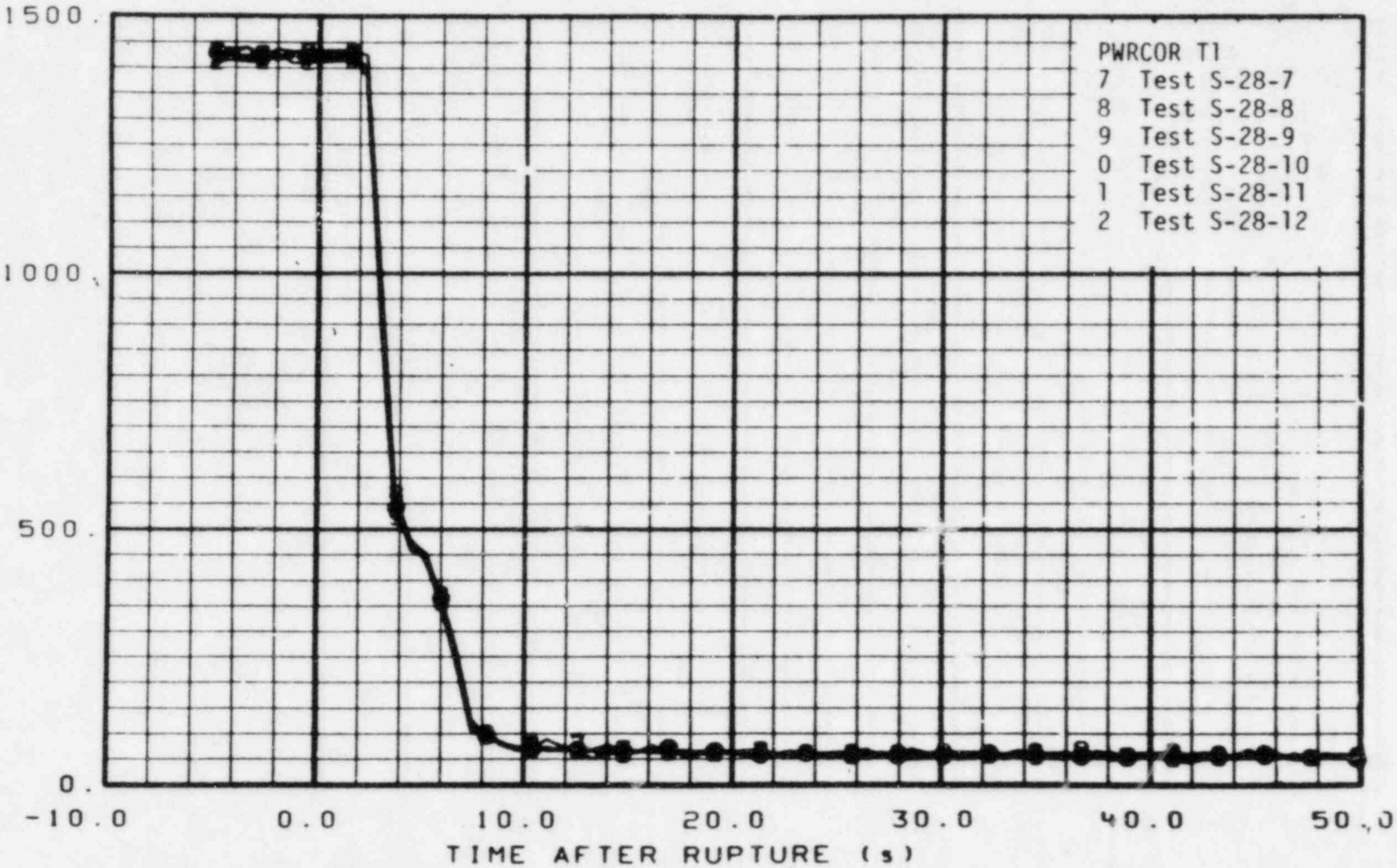

Figure 4. Initial Core Power Decay

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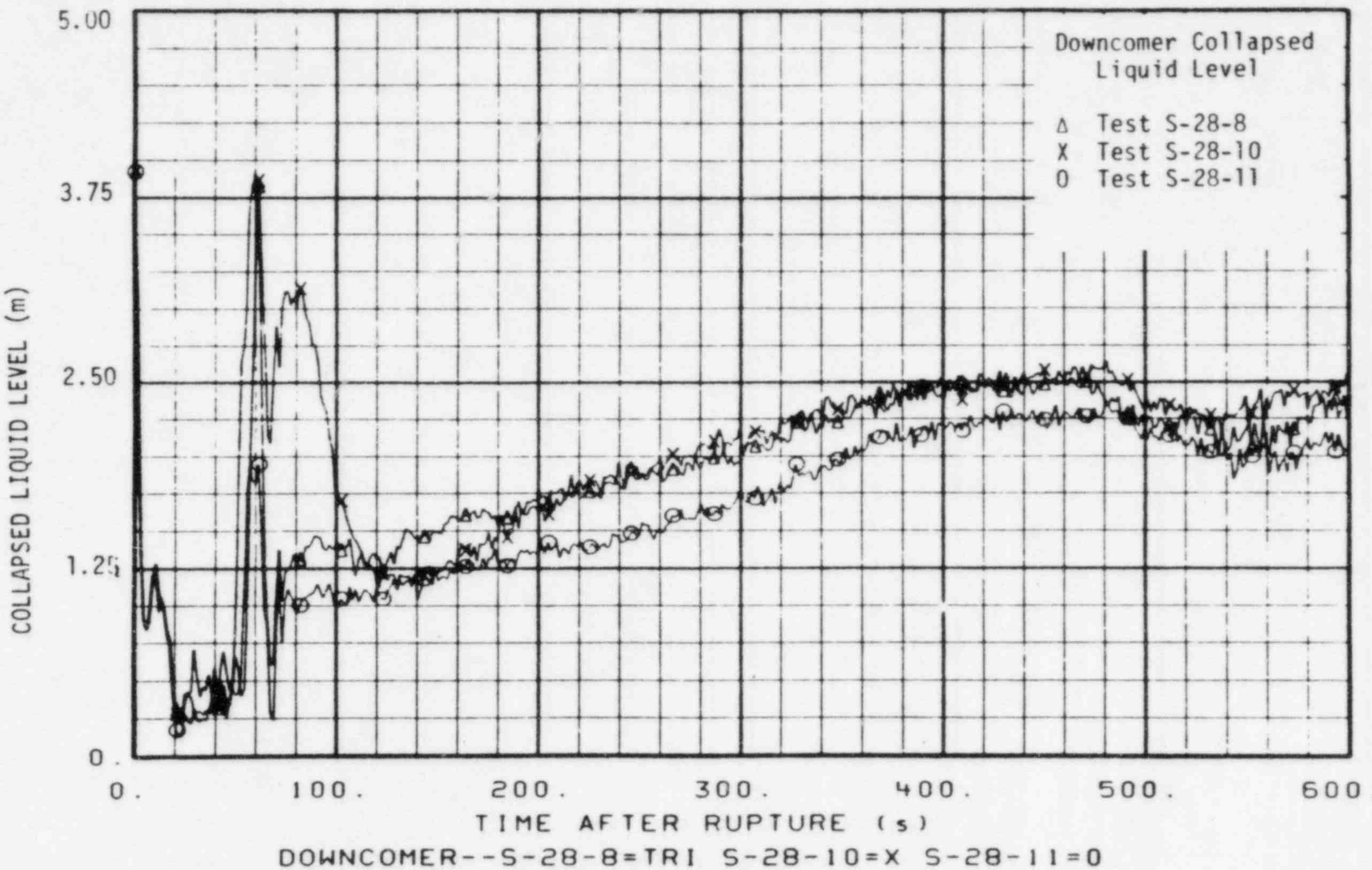

Figure 5. Comparison of Downcomer Collapsed Liquid Levels - Tests S-28-8, S-28-10, and S-28-11

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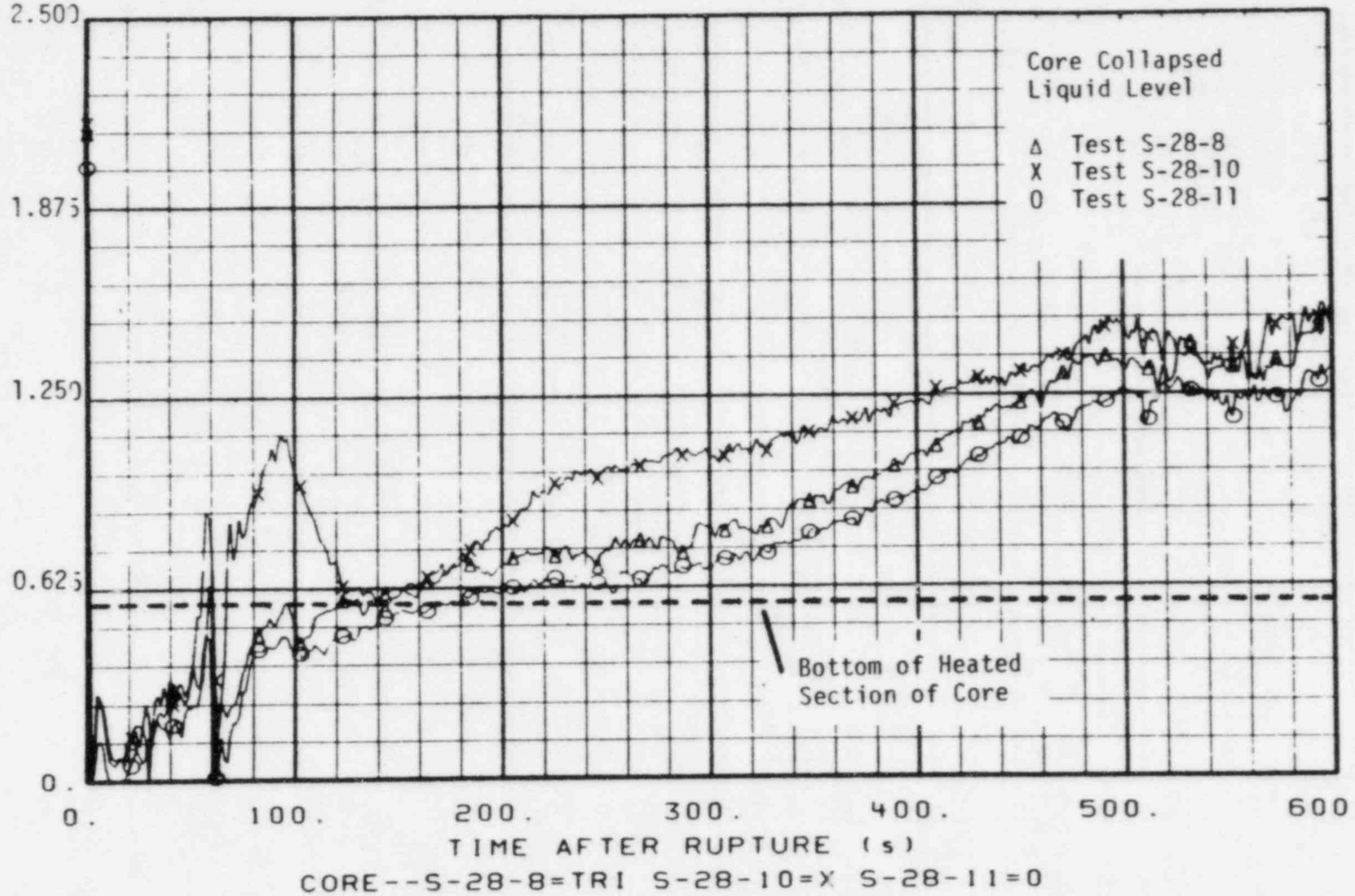

Figure 6. Comparison of Core Collapsed Liquid Levels - Tests S-28-8, S-28-10, and S-28-11

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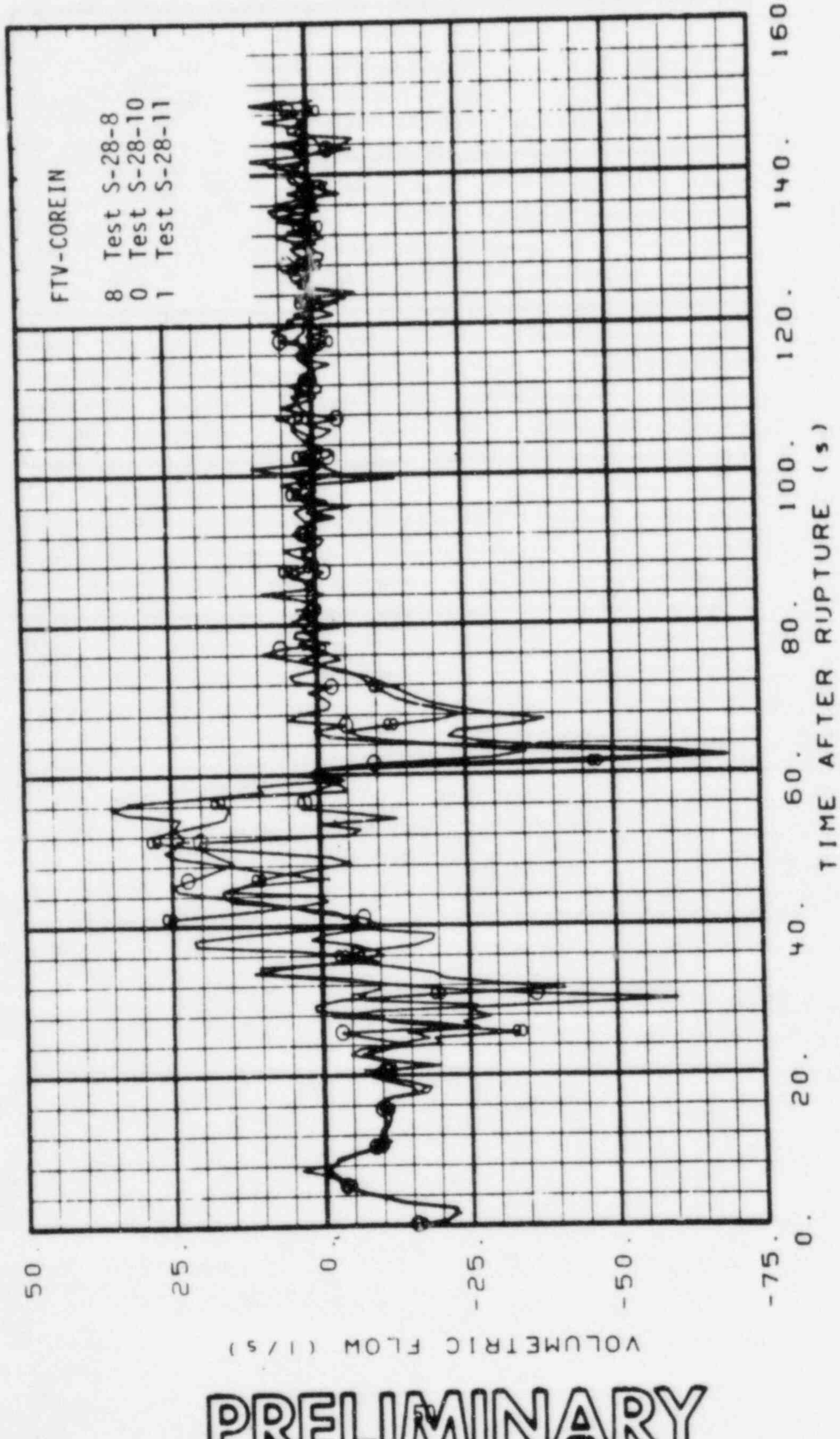

Figure 7. Comparison of Core Inlet Volumetric Flow Rates - Tests S-28-8, S-28-10, and S-28-11

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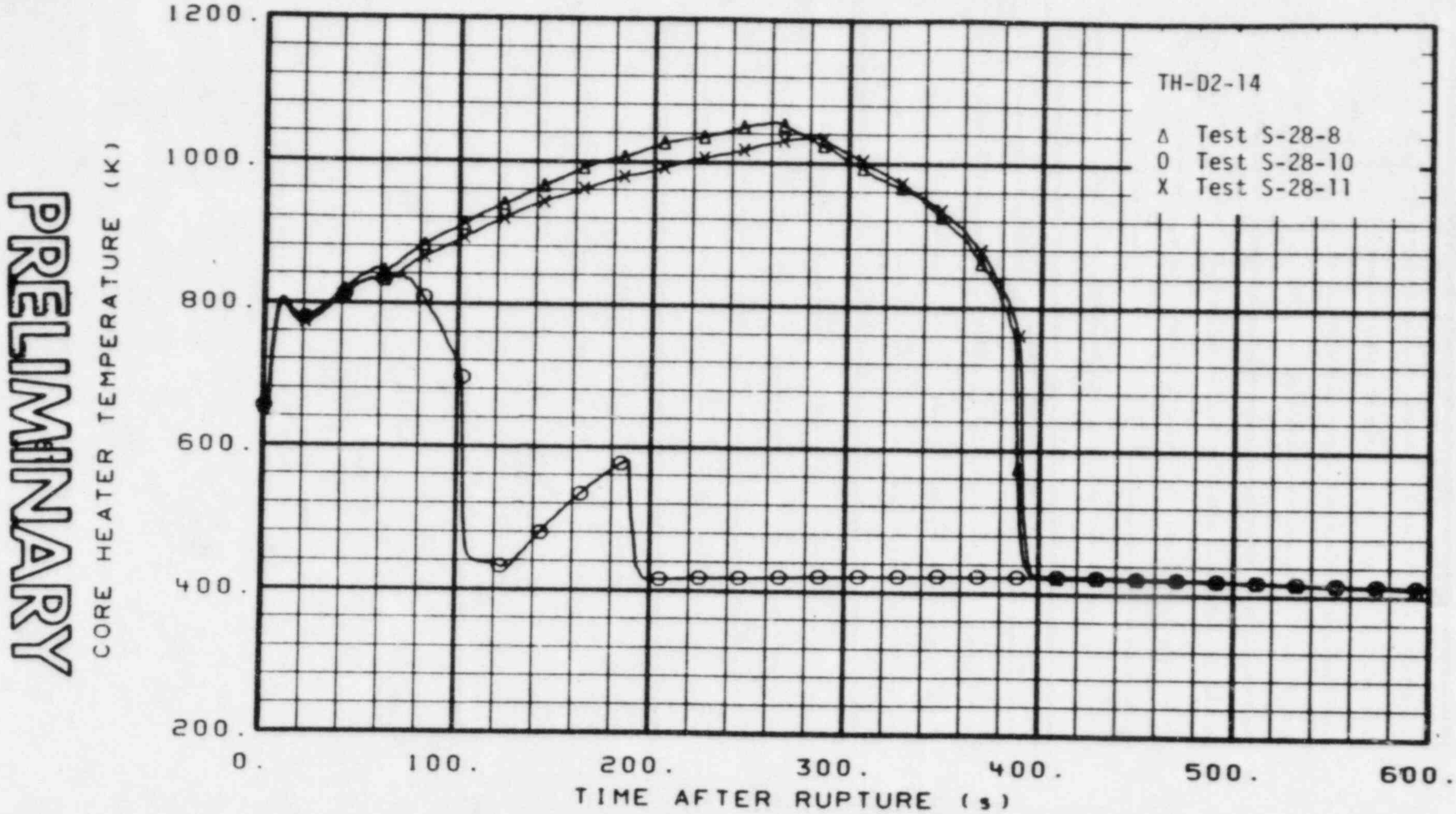

Figure 8. Comparison of Cladding Temperatures on Rod D2 at the 0.36 Meter Elevation - Tests S-28-8, S-28-10, and S-28-11

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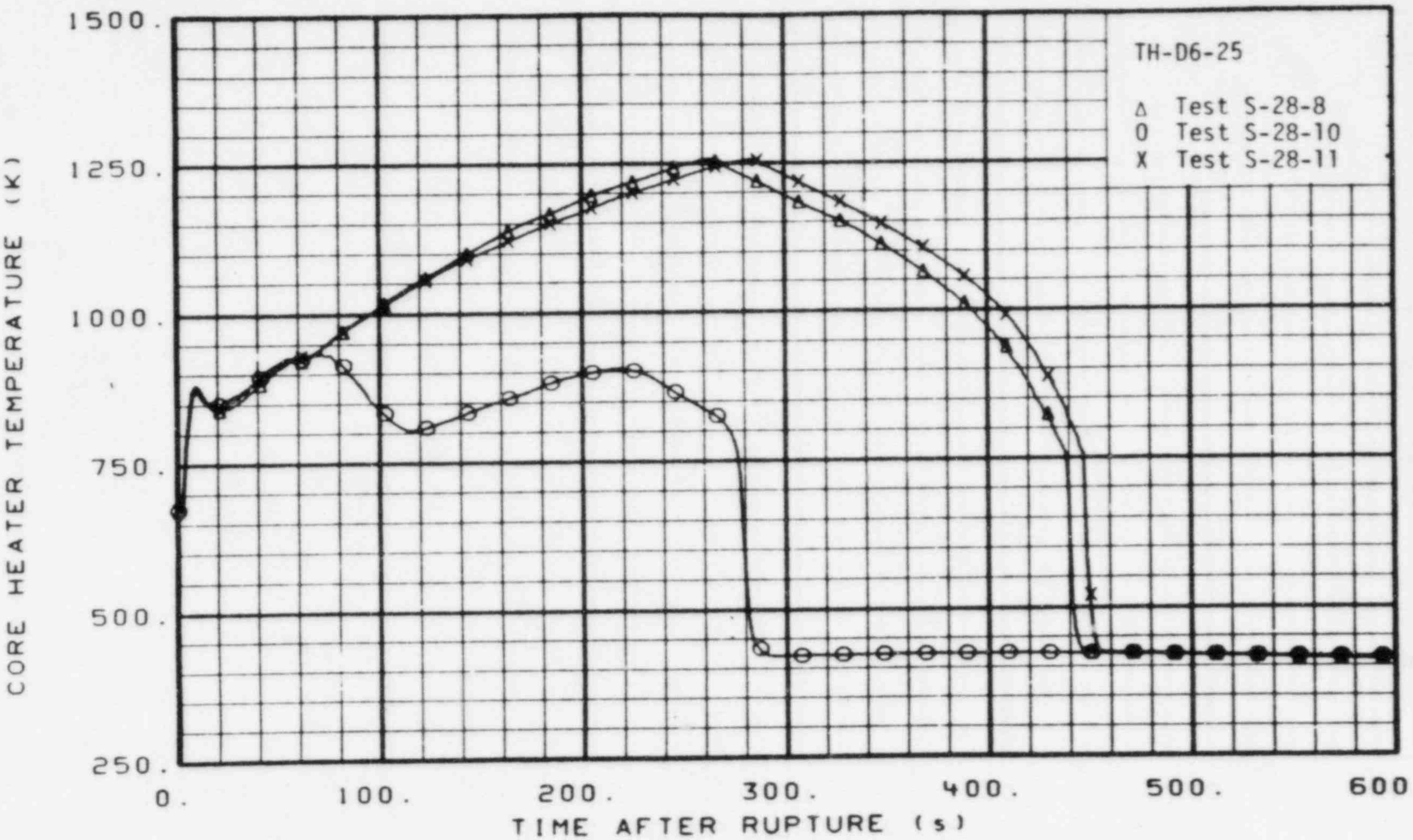

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Figure 9. Comparison of Cladding Temperatures of Rod D6 at the 0.64 Meter Elevation - Tests S-28-8, S-28-10, and S-28-11

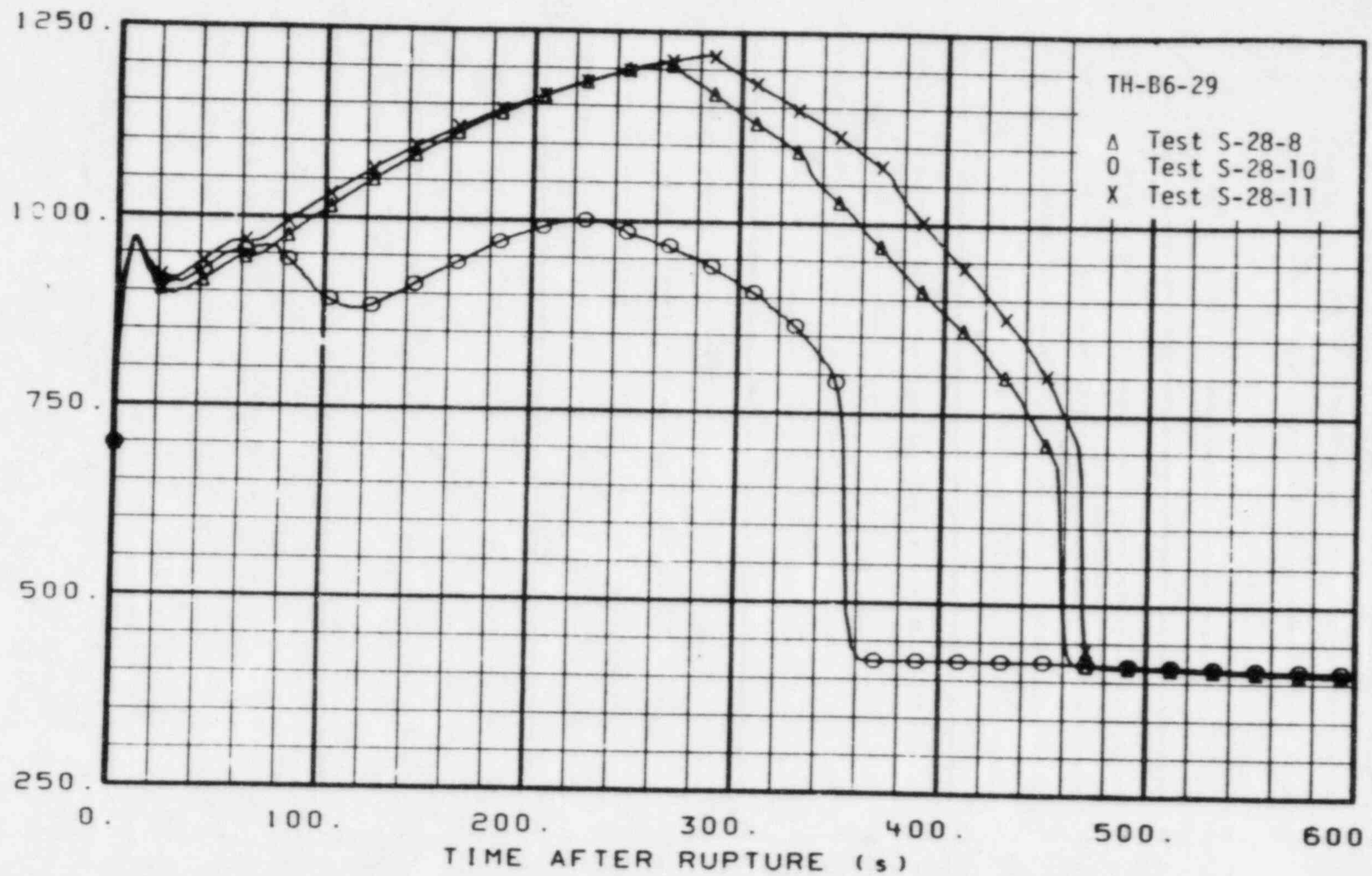

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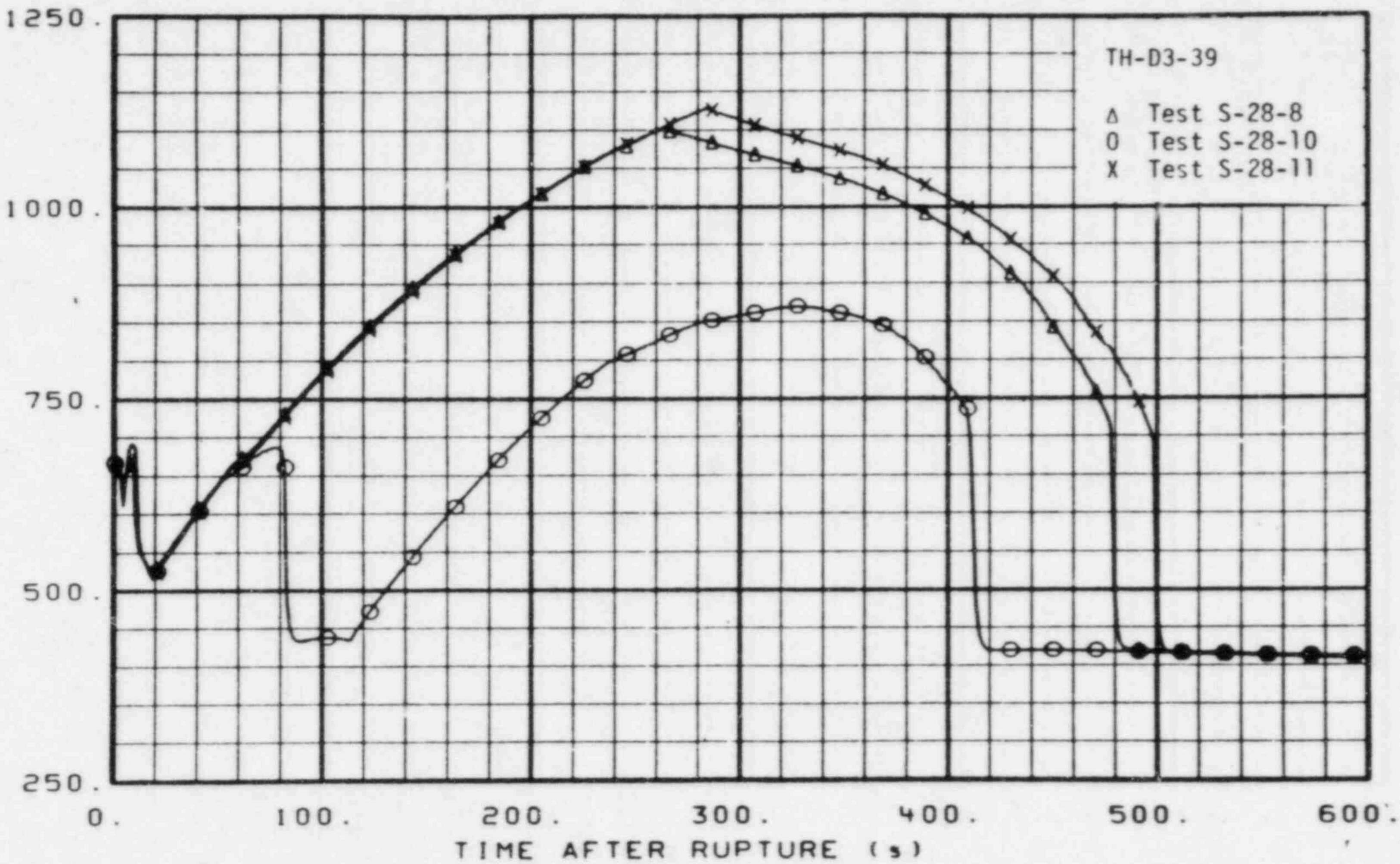
Figure 10. Comparison of Cladding Temperatures on Rod D3 at the 0.99 Meter Elevation - Tests S-28-8, S-28-10, and S-28-11



PRELIMINARY

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Figure 11. Comparison of Cladding Temperatures on Rod D3 at the 0.99 Meter Elevation - Tests S-28-8, S-28-10, and S-28-11

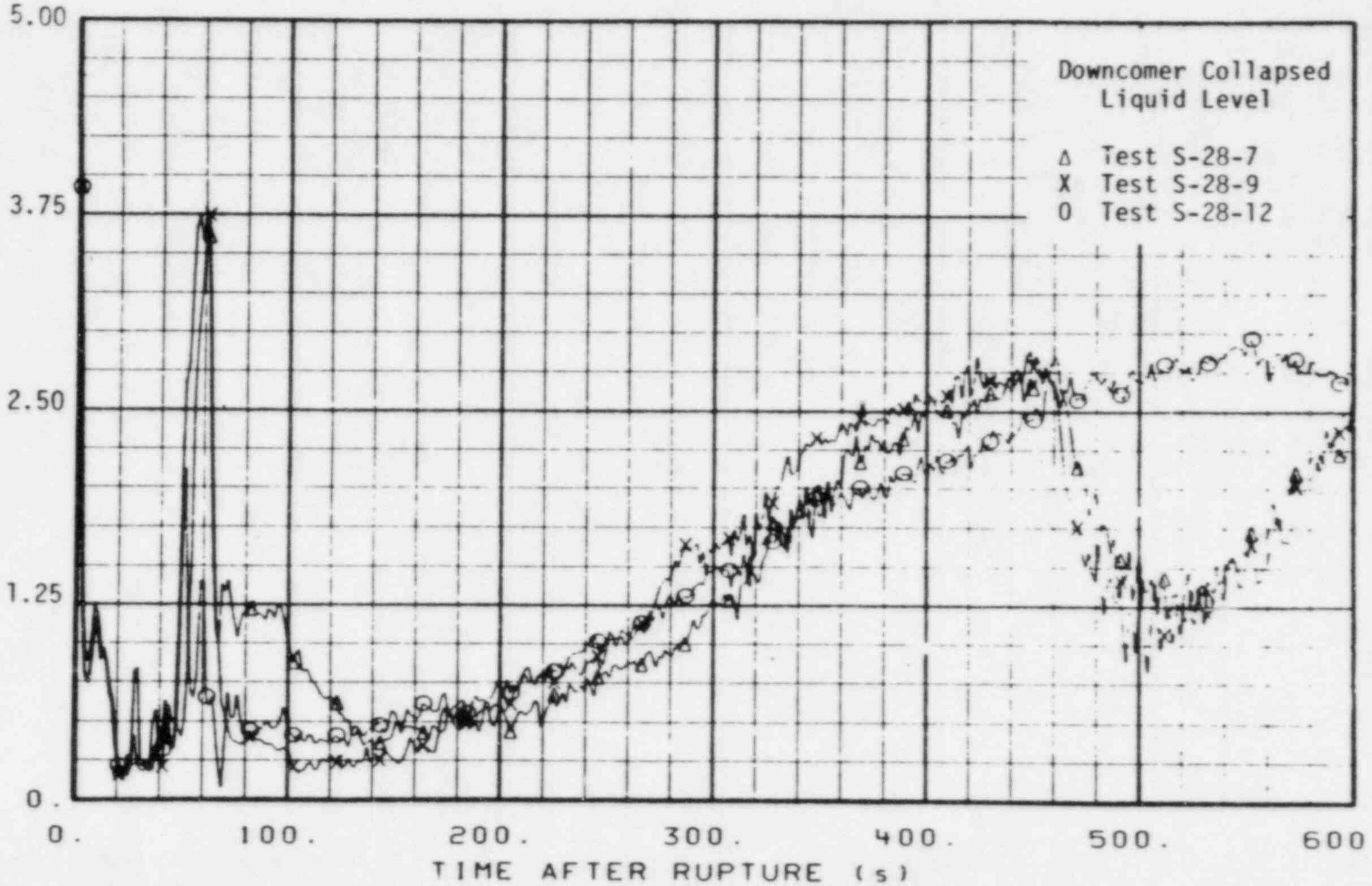

Figure 12. Comparison of Downcomer Collapsed Liquid Levels - Tests S-28-7, S-28-9, and S-28-12.

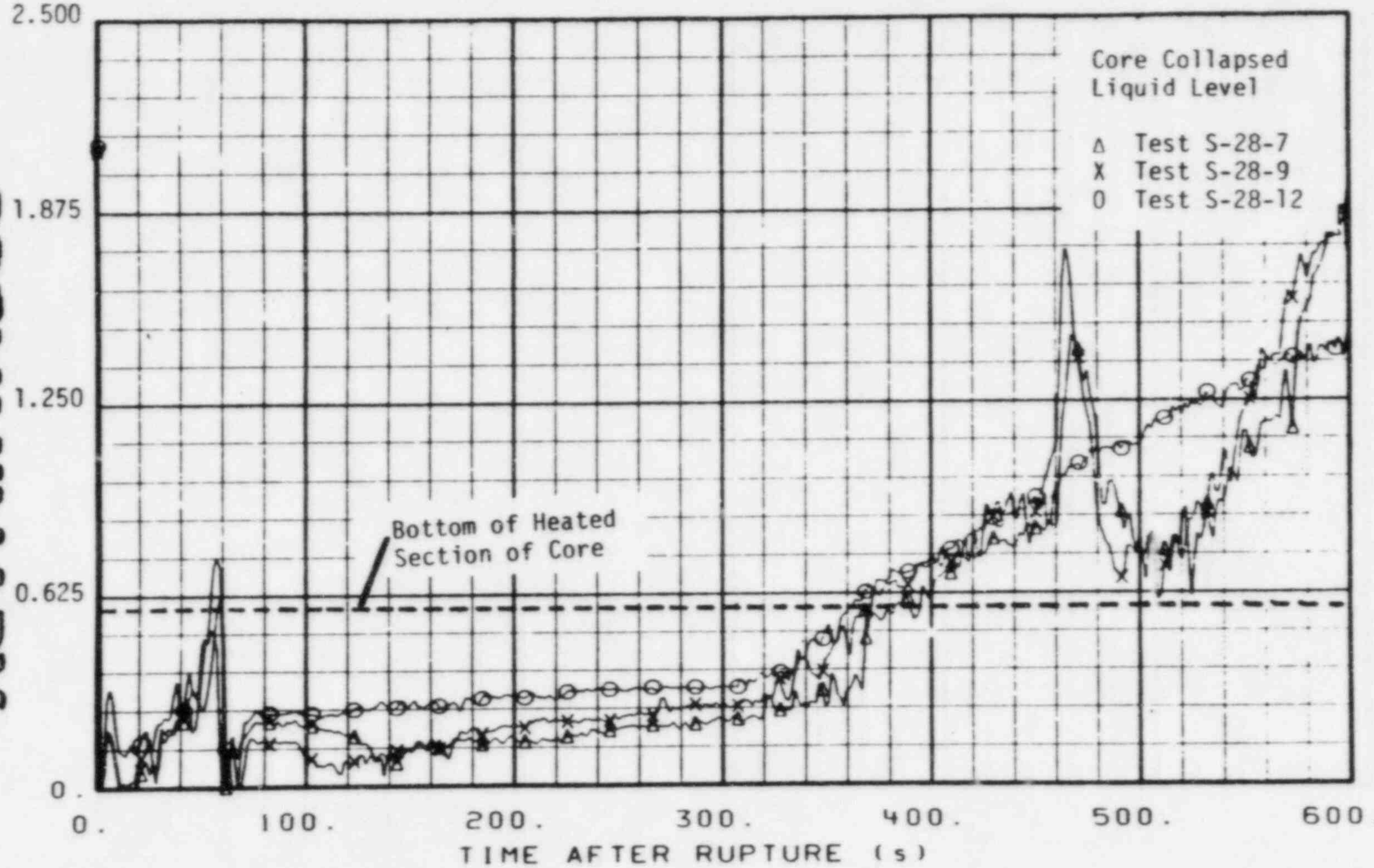

Figure 13. Comparison of Core Collapsed Liquid Levels -
Tests S-28-7, S-28-9, and S-28-12

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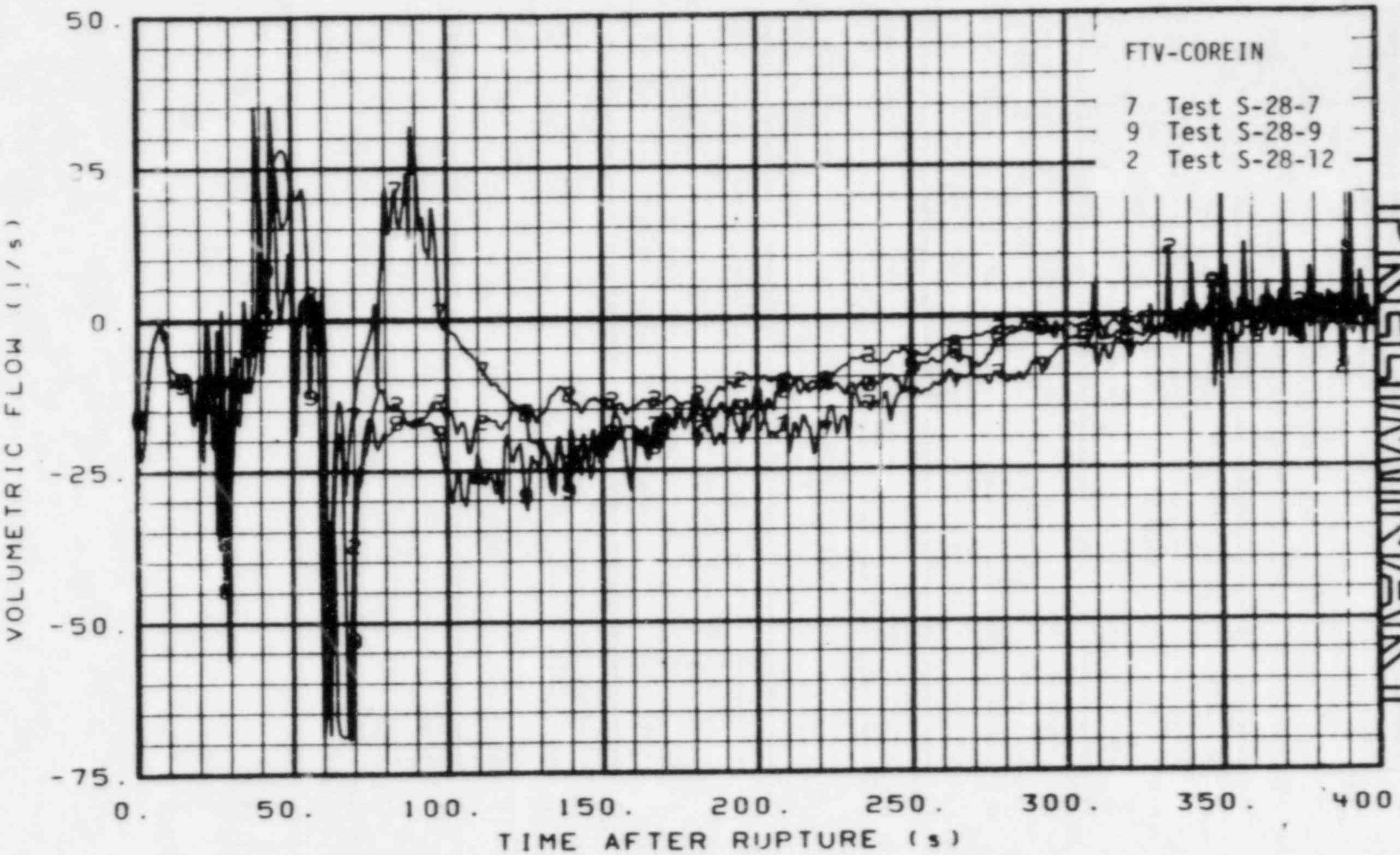

Figure 14. Comparison of Core Inlet Volumetric Flow Rates - Tests S-28-7, S-28-9, and S-28-12

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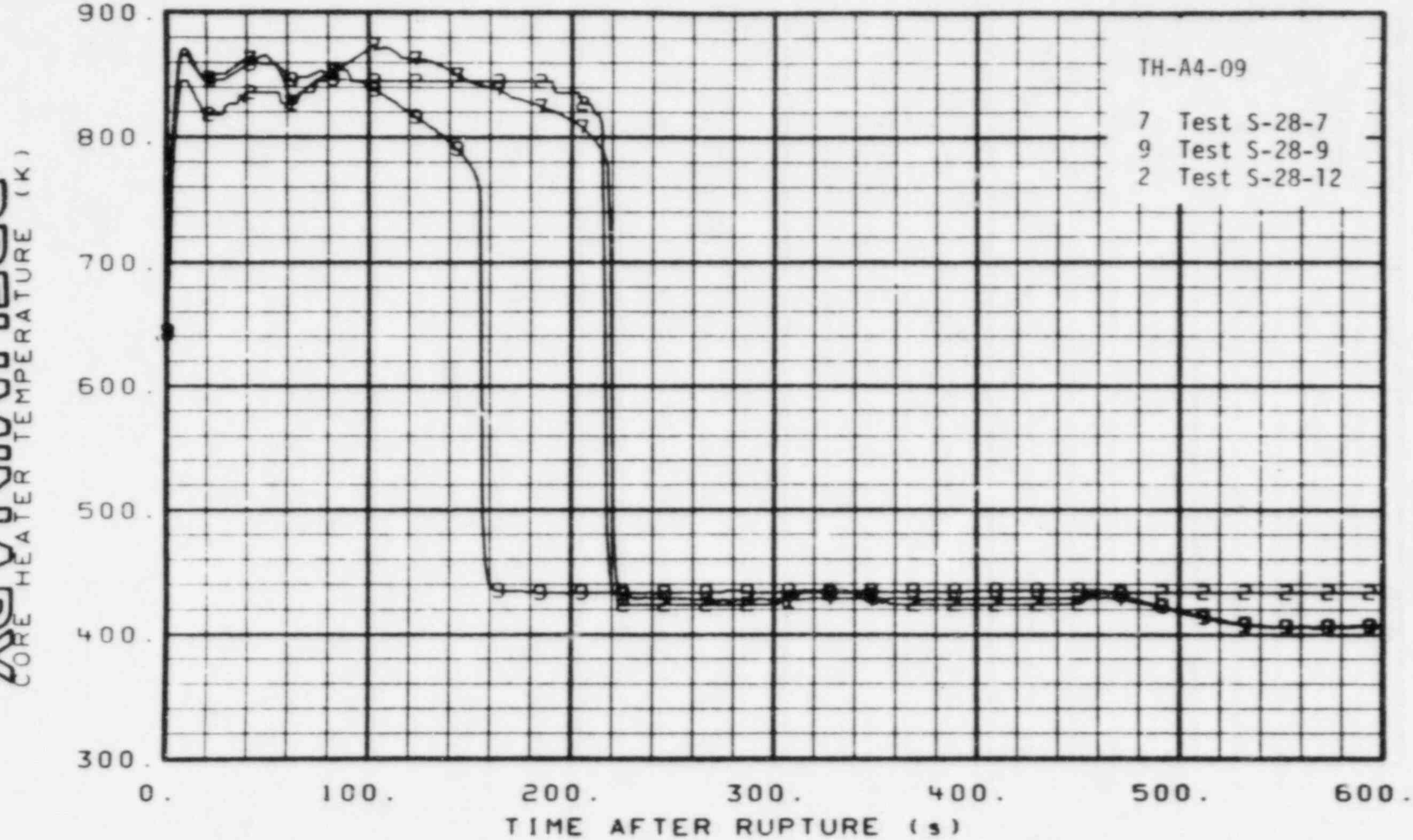
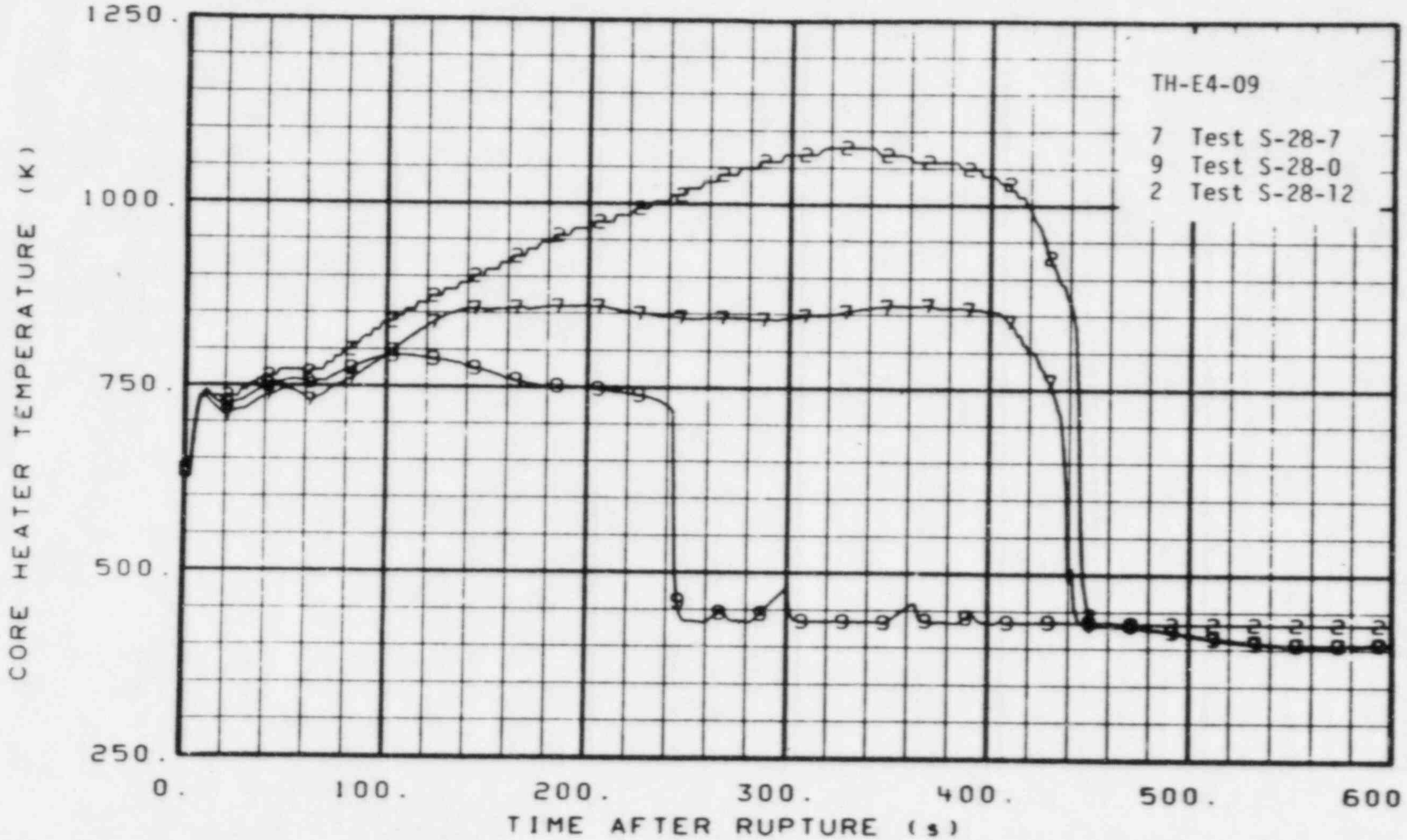


Figure 15. Comparison of Cladding Temperatures on Rod A4 at the 0.23 Meter Elevation - Tests S-28-7, S-28-9, and S-28-12



PRELIMINARY

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Figure 16. Comparison of Cladding Temperatures on Rod E4 at the 0.23 Meter Elevation - Tests S-28-7, S-28-9, and S-28-12

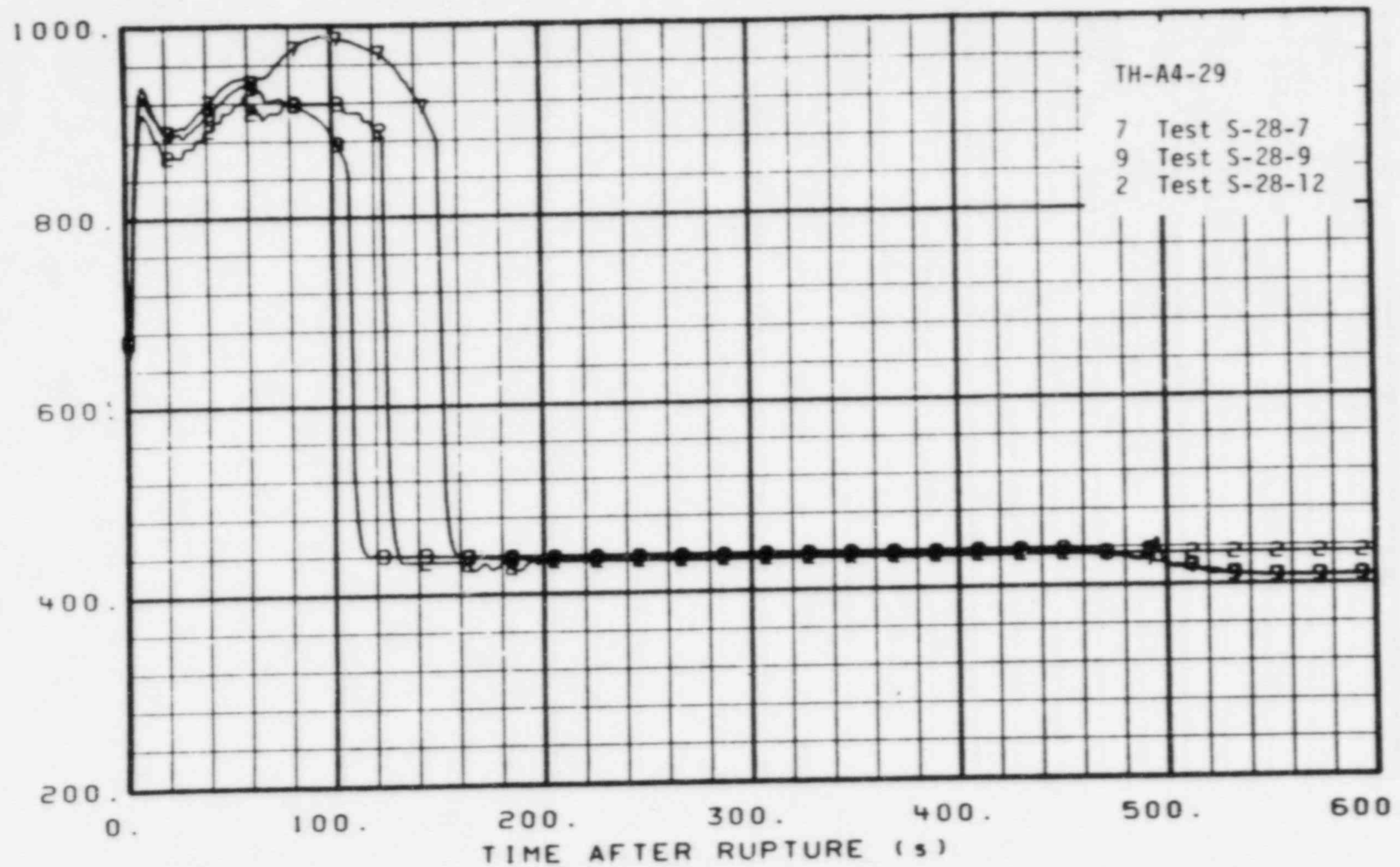

Figure 17. Comparison of Cladding Temperatures on Rod A4 at the 0.82 Meter Elevation - Tests S-28-7, S-28-9, and S-28-12

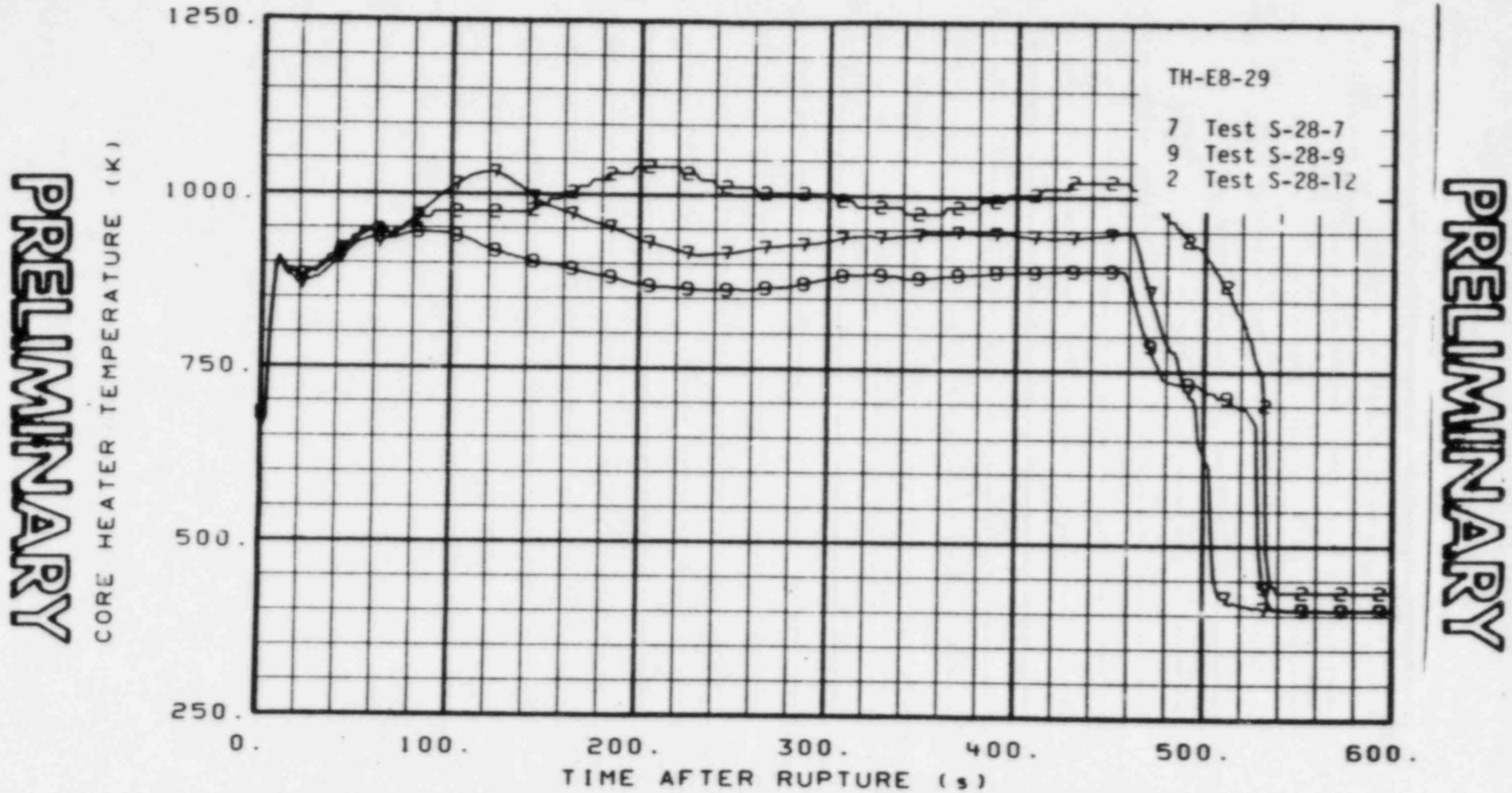

Figure 18. Comparison of Cladding Temperatures on Rod E8 at the 0.82 Meter Elevation - Tests S-28-7, S-28-9, and S-28-12

EG&G Idaho, Inc.

P. O. Box 1625
Idaho Falls, Idaho 83401

June 3, 1977

Mr. P. E. Litteneker, Chief
Reactor Safety Behavior Branch
Reactors Division
Idaho Operations Office - ERDA
Idaho Falls, Idaho 83401

TRANSMITTAL OF SEMISCALE EOS APPENDIX 28 - DJO-125-77

Reference: (a) P. E. Litteneker Ltr to D. J. Olson, Test Series EOS 28, May 26, 1977
(b) L. S. Tong Ltr to P. E. Litteneker, Test Series EOS 28, May 20, 1977

Dear Mr. Litteneker:

The enclosed Semiscale Experimental Operating Specification (EOS) Appendix 28, has been revised in accordance with the instructions in the referenced letters. Since several of the comments are best resolved by further clarification rather than modification to the document, a summary of the resolutions to the comments has been prepared and included as an attachment to this letter.

Resolution of the comments concerning the actual testing sequence and the details of the specified secondary to primary flowrates did not occur in time for the printing of the EOS. The tests run as part of Test Series 28 will take place in a sequence and with flowrates agreed with NRC and will be properly documented.

Very truly yours,

D J Olson

D. J. Olson, Manager
Semiscale Program

JMC:elp

Attachment

FOIA-84-884

C1

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
H. J. C. Kouts, NRC
S. Levine, NRC
W. C. Lyon, NRC
T. G. McCreless, ACRS
T. E. Murley, NRC
J. A. Norberg, NRC
T. M. Novak, NRC
D. F. Ross, NRC
Z. R. 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
G. Fader, CE
G. Farber, IfR
R. Gay, EPRI
P. Griffith, MIT
R. W. Kiehn, EG&G Idaho
W. Kirchner, LASL
M. Levenson, EPRI
W. Loewenstein, EPRI - 2
P. A. Lottes, ANL
H. P. Pearson, EG&G Idaho - 6
W. Riebold, JRCE
H. Seipel, DBF&T
D. G. Thomas, HNL
D. Trent, PNL
T. A. Zordan, W
R. J. Beers, ERDA-ID

Attachment
DJO-125-77
June 3, 1977

RESOLUTION OF NRC COMMENTS ON APPENDIX 28 TO THE 1-1/2 LOOP SEMISCALE EXPERIMENTAL OPERATING SPECIFICATION

COMMENT 1, Page 28-2. Item (1) should clarify that since the area associated with the rupture of a tube is related to how it ruptures, flow rate is directly related to flow area and loosely related to the number of tube ruptures unless offset shear is assumed.

RESOLUTION. In item (1), the "number of tube ruptures" refers specifically to single ended tube ruptures in a PWR steam generator. The magnitudes of the steam generator secondary to primary mass flow rates are presented in terms of the flow rates associated with a given number of single ended tube ruptures to provide a basis for comparing the relative magnitudes of the mass flow rates in the Semiscale Mod-1 system and in a PWR system.

COMMENT 2, Page 28-3 General. It would be helpful to the reader to provide headings segregating small tube ruptures from large tube ruptures and subheadings in each section for the time of occurrence of the tube ruptures.

28-2.0a - Small Tube Ruptures

i - Blowdown

ii - Refill

iii - Reflood

2.0b - Large Number of Ruptures

i - Blowdown

ii - Refill

iii - Reflood

RESOLUTION. Comment incorporated.

COMMENT 3, Page 28-3, 2nd paragraph, line 2. "...the number of steam generator tubes (ie, steam generator secondary to primary flow rate..." should be added.

RESOLUTION. Comment incorporated.

COMMENT 4, Page 28-3, last line. "very small number (less than 12)" tube ruptures used here must refer to the LPWR. This point should be clarified and possibly somewhere in the report note:

(a) ratio of ruptured tubes to total tubes assumed,

(b) what assumptions are made concerning distribution of breaks within the PCS loops.

RESOLUTION. Comments incorporated. (a) The number of tubes ruptured is given in Figure 28-A-1 as a percentage of the total number of tubes in three of the four steam generators in a 4-loop PWR. (b) The assumption was made that the tube ruptures occurred at (or near) the inlet plenum of the intact loop steam generator.

COMMENT 5, General. Does the Homogeneous Equilibrium assumption in RELAP4 tend to reduce the effect of tube rupture for any part of the LOCA? Which part? The effect during refill and reflood as well as late in blowdown appear to be potentially significant. This may be observed in analysis/experimental data comparisons and if so should lead to improved analysis methods because this could be increasingly significant with increase in system size.

RESOLUTION. RELAP4 was not used in the analysis of any of the tests actually to be run in Test Series 28. In each test in the series it is assumed that the tube ruptures occur at the initiation of refill.

Test data from the baseline test (Test S-04-6) provided the conditions at the time of the steam generator tube ruptures for the analysis and FLOOD4 calculations were used from that point. In the FLOOD4 calculations only the steam flow from the secondary to primary mass flow was considered. It is expected that considering only the steam flow provides conservatism in the calculations.

COMMENT 6, Page 28-8, 1st paragraph. What happens to the secondary side of the steam generator during this series if flow is coming in from the accumulator. The secondary side should blowdown at a rate consistent with the accumualtor discharge rate.

RESOLUTION. The steam generator secondary system fluid will be discharged starting at the same time as the initiation of injection from the accumulator which simulates the tube rupture flow. The rate of discharge of the secondary system fluid will be controlled so that the emptying time approximates the accumulator injection period. Requirements for control of the steam generator secondary discharge valves (an air actuated valve and a needle valve) for this test series are discussed in Addendum 28-B.

COMMENT 7, Page 28-8, 2nd paragraph, 4th line. What were the assumptions in the analysis which would bias the calculations in the direction of high peak cladding temperature? These should be delineated.

RESOLUTION. A discussion of the assumptions in the analysis which biased the calculations in the direction of high peak cladding temperatures is included in Addendum 28-A.

COMMENT 8, Page 28-8, 2nd paragraph. One test should be conducted with a high number of simulated steam generator tube ruptures.

RESOLUTION. A test simulating the rupture of 100 steam generator tubes will be conducted.

COMMENT 9, Page 28-7, 1st paragraph. Adiabatic heatup appears to be an EM rather than a realistic assumption. Is this assumption used in periods of zero core flow?

RESOLUTION. The assumption of adiabatic heatup is used during the refill phase of the calcualtion when zero core flow may occur. Although the assumption of adiabatic heatup is conservative, it does not appear to be unreasonable since results from previous Semiscale tests (in particular Test S-05-2) indicate that core flow stagnation (or very low core flow rates) can occur during the refill period.

COMMENT 10, Page 28-11. The test program should start with a high number of simulated tube ruptures and work back towards low numbers of simulated ruptures. The reverse should hold true for small numbers of tube ruptures (12). Tests should cover 4, 8, and 11 simulated tube ruptures and the test sequence should start with the lowest number of tubes (4) and increase to 11.

RESOLUTION. Changes to the testing sequence for Test Series 28 will be decided in further communications between EG&G and NRC.

COMMENT 11, Figure 28-A-1. When relating tube rupture flow rates to the number of tubes ruptured an equivalent single ended tube rupture is assumed. Why? Explain in report.

RESOLUTION. Comment incorporated. The assumption of single ended tube ruptures was made to provide a basis for comparing the relative magnitude

of the steam generator secondary to primary mass flow rates in the Semiscale Mod-1 system to that in a PWR system.

COMMENT 12, Page 28-A-9. Reference is made to Test S-05-2 in the highlight section, yet results from Test S-04-6 are used. What is the baseline?

RESOLUTION. Test S-04-6 is the baseline test, and data from Test S-04-6 were used (where applicable) as input for several of the calculations performed. Results from Test S-05-2 were used only to provide an additional estimate of the peak rod cladding temperature during reflood when tube ruptures occurred during blowdown.

COMMENT 13, Pages 28-A-17/18. How come the initial temperature conditions for refill and reflood are the same for 18 seconds (40 versus 58)?

RESOLUTION. The initial temperature conditions used in the calculations performed to evaluate the effect of the rupture of a relatively large number of steam generator tubes prior to the initiation of refill were based directly on data from Test S-04-6. It was determined from Test S-04-6 that refill began at about 40 seconds and that a maximum core cladding temperature of about 1450°F occurred at approximately the same time. To provide conservatism, the same set of initial temperature conditions was used for the calculations performed to evaluate the effect of the rupture of a relatively large number of steam generator tubes at the initiation of reflood.

<u>COMMENT 14, Page 28-B-3</u>. The accumulator valve should remain open for the duration time it would take the secondary to completely discharge. This should vary from test to test.

<u>RESOLUTION</u>. Table 28-B-I lists the times for opening and closing the accumulator valves to achieve the injection of the required volume of secondary fluid into the primary system based on a given secondary to primary mass flow rate. The steam generator secondary system fluid will be controlled so that the emptying time approximates the accumulator injection period.

<u>COMMENT 15, Page 28-C-2</u>. Why was core area scaling used instead of volume scaling?

<u>RESOLUTION</u>. Core area scaling was used for both the total volume and the mass flow rates used to simulate the rupture of steam generator tubes in a PWR because it is a more appropriate scaling criteria for the Semiscale Mod-1 system during reflood than is volume scaling.

<u>COMMENT 16, Page 28-A-5</u>. Why does FLOOD4 predict lower rod cladding temperatures for low reflood rates?

<u>RESOLUTION</u>. The FLOOD4 code predicts low temperatures compared with some FLECHT data. It is a developmental code and the reasons for the low temperatures are not known at this time.

<u>COMMENT 17, Page 28-A-7</u>. Did the refill calculation account for hot walls?

<u>RESOLUTION</u>. The refill calculation did not account for the hot wall delay in the Semiscale Mod-1 system.