TEST PREDICTION OF THE SEVENTH SEMISCALE MOD-1 TEST SERIES STEAM GENERATOR TUBE RUPTURE TESTS TEST S-28-1 (REVISION 1)

SEMISCALE PROGRAM

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TEST PREDICTION OF THE
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by

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SEMISCALE PROGRAM

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SUMMARY

This document contains a pretest prediction of the Semiscale Mod-1 system thermal-hydraulic response for Test S-28-1 (Revision 1). Test S-28-1 (Revision 1) will be the third integral blowdown reflood test to be performed in the steam generator tube rupture test series. The primary objectives of this test are to set an upper limit on the range of steam generator tube ruptures over which high peak cladding temperatures can result and to provide a data base for comparison between the test data and the analysis results used to specify Test Series 28.

The initial conditions for Test S-28-1 (Revision 1) will be as specified for Test S-28-1 in Appendix 28 of the Semiscale Experimental Operating Specification (EOS)[1]. Injection from an accumulator into the intact loop hot leg just upstream of the steam generator inlet plenum will begin at 40 seconds after rupture to simulate the steam generator tube ruptures. The test will be run with an injection rate of 0.54 kg/s to simulate the rupture of 60 steam generator tubes.

The break configuration will represent a full size (200%) double-ended offset shear cold leg break. The test will be initiated at an initial core power of 1.44 MW (with 36 powered rods) and the ANS power decay curve will be used during the reflood portion of the test. Emergency core coolant (ECC) from the intact loop high pressure injection system (HPIS), the accumulator, and the low pressure injection system (LPIS) will be injected into the intact loop cold leg. Accumulator, HPIS, and LPIS injection will also be used in the broken loop pump simulator discharge. The pressure suppression system pressure will be maintained at about 241 kPa during the blowdown and the reflood portions of the test.

The predictions for Test S-28-1 (Revision 1) were developed from Test S-04-6 test data and the FLOOD4 computer code. The system response during the first 40 seconds of Test S-28-1 (Revision 1) is expected to be the same as the system response in Test S-04-6. Therefore, Test S-04-6 data is provided to give an indication of the expected system thermal-hydraulic response for the first 40 seconds of Test S-28-1 (Revision 1). The FLOOD4 computer code was used to provide predictions for the remainder of the transient. Since the heat transfer and entrainment correlations used in the FLOOD4 code have not been extensively tested against data, the prediction is expected to follow the trends of the data, but may not exactly calculate the oscillating flows and the rod temperatures. Also, the calculation of quench times is strongly dependent on the rod temperature distribution and system pressure at the initiation of reflood. Small differences in these parameters can significantly affect the reflood calculations. In addition, the FLOOD4 code does not account for downcomer wall heat transfer during the refill and reflood transient. Previous test data indicates that liquid depletion in the downcomer, which is due to downcomer wall heat transfer, can also significantly affect the core response during refill and reflood.

Since the initial conditions for Test S-28-1 (Revision 1) and Test S-04-6 (the baseline test) are the same, the system response should be the same until 40 seconds after rupture. The peak temperature in the core should be approximately 1075 K at 8 seconds after rupture. This temperature is expected to occur on a rod on the perimeter of the core. The peak temperature should be about 994 K when the injection simulating

the tube ruptures begins. Test S-04-6 data indicates the system pressure should reach 241 kPa (containment pressure) by 40 seconds. Fluid saturation conditions at 6 MPa and 549 K should be present in the steam generator secondary at 40 seconds.

FLOOD4 calculations indicate the peak temperature in the core should decline from 994 K at 40 seconds to 828 K at 242 seconds after rupture. This temperature decrease is due to the reverse steam flow through the core caused by the injection simulating the flow from the ruptured tubes. After the injection into the intact loop hot leg near the steam generator ended at 242 seconds after rupture (the time at which the steam generator secondary would empty if 60 tubes ruptured), the FLOOD4 model indicated that during lower plenum refill the peak temperature in the core would increase to 1021 K. Lower plenum refill was accomplished by the LPIS and HPIS only because the intact loop accumulator would be depleted of water at approximately 70 seconds after rupture. Reflood of the core by the LPIS and HPIS is expected to start at about 316 seconds after rupture. The core hot spot is expected to quench about 417 seconds after rupture (101 seconds after reflood) and the whole core is expected to quench by 434 seconds after rupture.

I. INTRODUCTION

This report contains the predictions of the Semiscale Mod-1 system thermal-hydraulic response for Test S-28-1 (Revision 1) which will be the third integral blowdown-reflood test in the steam generator tube rupture test series. The report identifies the prerupture system conditions and presents the expected behavior of key variables with particular emphasis placed on the predicted response of the electrically-heated core. Test S-04-6 data[2] (the baseline test for Test S-28-1 (Revision 1)) was used to indicate the expected system blowdown response, since the response in Test S-28-1 (Revision 1) should be the same during this period. The FLOOD4 [3] models used to predict the system response over the remainder of the transient are described.

The primary objectives of Test S-28-1 (Revision 1) are to set an upper limit on the range of steam generator tube ruptures over which high peak cladding temperatures are expected to occur and to provide a data base for evaluation of the analysis techniques used to specify Test Series 28. The test conditions for Test S-28-1 (Revision 1) are identical to those of the baseline Test S-04-6 except for the introduction of accumulator injection into the intact loop hot leg just upstream of the steam generator inlet plenum to simulate the steam generator tube ruptures. The test will be run with an injection rate of 0.54 kg/s to simulate the rupture of 60 steam generator tubes. The change in heat transfer potential of the steam generator will be simulated by discharging the steam generator secondary fluid over the simulated tube rupture period. The water in the accumulator will be near saturation conditions at 547 K (approximately the average temperature of the pressurized water reactor (PWR) steam

generator secondary fluid at rated load) and 5.9 MPa. The total volume of water injected to simulate the tube rupture flow is 0.144 m^3, which is core area scaled from three PWR steam generators at rated load. The injection will begin at 40 seconds after the cold leg break to simulate the steam generator tube ruptures. During steam generator liquid injection, the accumulator pressure will be maintained by a nitrogen supply. The injection will be terminated before the accumulator water is completely exhausted to prevent nitrogen injection into the primary system. The initiation of the tube ruptures at 40 seconds was selected because preliminary analysis showed that when the tube ruptures occurred at this time the highest peak cladding temperatures occurred (see Figure 1). The highest temperatures occurred because tube rupture at this time was assumed to prohibit refill of the downcomer and lower plenum, and an adiabatic heat-up of the core was assumed to take place while the downcomer and lower plenum were refilled.

Emergency core coolant (ECC) for Test S-28-1 (Revision 1) will be injected into the intact loop cold leg and broken loop pump simulator discharge. The Mod-1 ECC systems in operation in both loops will include the accumulator injection system (AIS), the high pressure injection system (HPIS), and the low pressure injection system (LPIS).

The operating conditions for Test S-28-1 (Revision 1) are summarized in Table I. The test will be conducted at an initial core power of 1.44 MW and an initial flow rate of 9.5 X 10^{-3} m^3/s. The radial core power profile will be peaked for this test. The three high power rods will have a peak power density of 39.7 kW/m and the other 33 low power rods will have a peak power density of 37.7 kW/m. Four rods will be

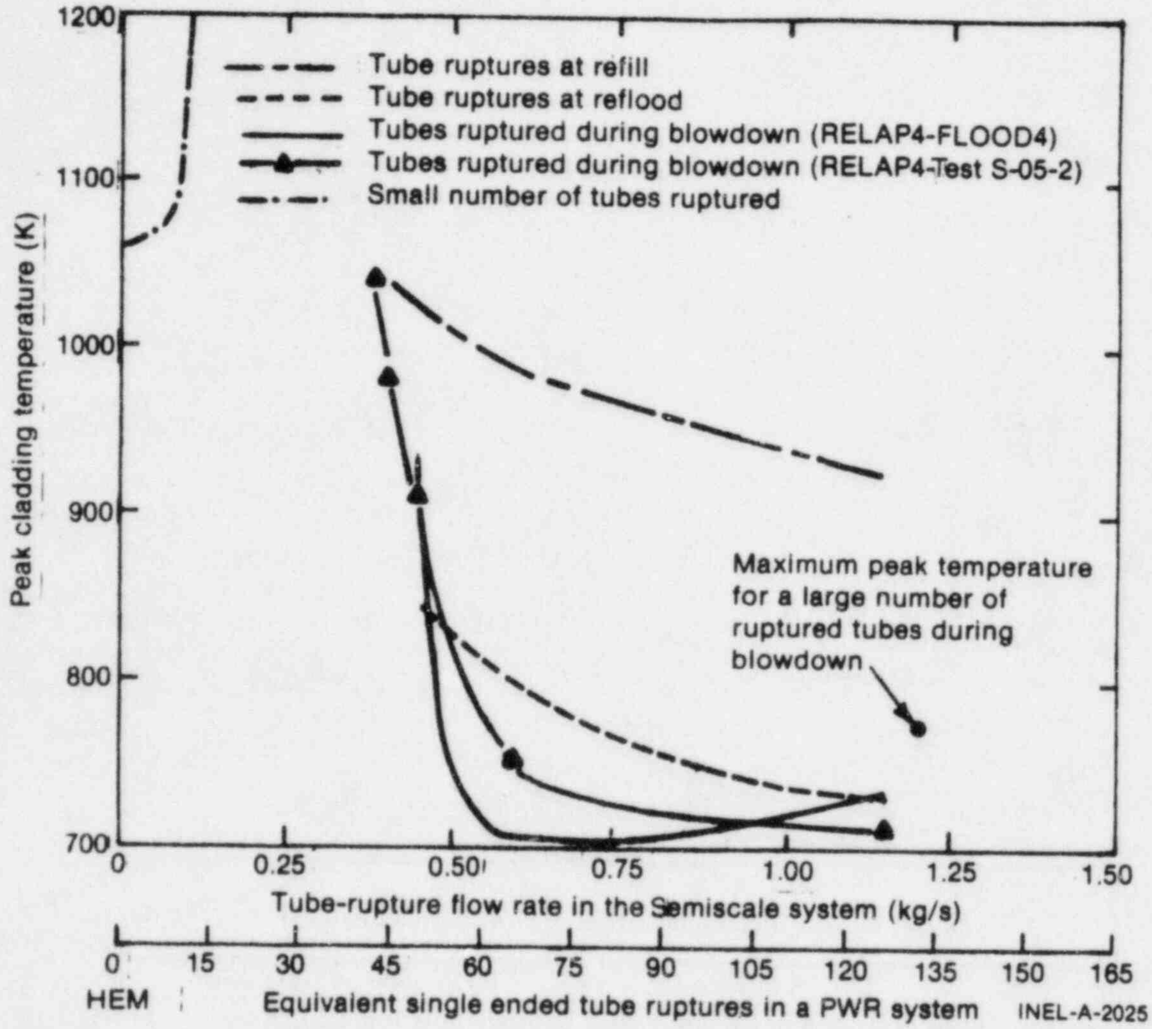


Figure 1. The Influence of the Tube Rupture Flow Rate or the Number of Ruptured Tubes on the Peak Cladding Temperature

Table I

Test S-28-1 (Revision 1) Description and Initial Conditions

Parameter	Initial Value
Break Size	200%[a]
Break Type	Cold Leg
Intact Loop Resistance	Low[b]
Nominal Initial System Pressure	15.5 MPa
Hot Leg Fluid Temperature	594 K
Cold Leg Fluid Temperature	558 K
Core Temperature Difference	36 K
Core Power	1.44 MW
Core Initial Inlet Flow Rate	7.1 kg/s
Power Decay	Figure 3
Pump Speed Control	Allowed to coast down to approximately 61% of initial rpm, then maintain at 61% of initial rpm.
ECC Injection	
Accumulator	
Location	Intact Loop Cold Leg
Actuation Pressure	4.1 MPa
Liquid Volume	0.08 m^3
Gas Volume	0.053 m^3
Line Resistance	659 $\frac{MPa\ sec^2}{kg\ m^3}$
Injection Rate	$1.45 \times 10^{-3}\ m^3/s$
Nitrogen Valve	Open for 24 seconds after accumulator empty of water

Table I (contd)

Test S-28-1 (Revision 1) Description and Initial Conditions

Parameter	Initial Value
HPIS	
Location	Intact Cold Leg
Actuation Pressure	12.4 MPa
Injection Rate	1.96×10^{-5} m^3/s
LPIS	
Location	Intact Cold Leg
Actuation Pressure	1.03 MPa
Injection Rate	2.52×10^{-4} m^3/s
Tube Rupture Simulator	
Steam Generator Accumulator	
Location	Just Upstream From The Intact Loop Steam Generator Inlet Plenum
Actuation Time	40.0 Seconds
Closure Time	242 Seconds
Liquid Volume	0.144 m^3
Gas Volume	4.8×10^{-2} m^3
Temperature	547 K
Injection Rate	7.15×10^{-4} m^3/s

(a) 200% break refers to a simulated double-ended offset shear break in the broken loop with each break nozzle having an area of 0.000243 m^2. The 200% break has a break area-to-system volume ratio equivalent to that ratio for a double-ended offset shear break in the cold leg of one loop of a four-loop pressurized water reactor.

(b) Low system resistance refers to the size of orifices located at the inlet and outlet of the intact loop steam generator. The low system resistance orifices have an approximate 4.06 cm diameter hole. The total system resistance with the low resistance orifices is properly scaled to the LOFT system.

unpowered with their locations chosen to give the same core configuration as in Test S-04-6. The fluid temperature at the core inlet will be 558 K and the core outlet fluid temperature will be 594 K. The axial power profile will be skewed toward the bottom of the heated core as shown in Figure 2. The power decay will follow the electrical power decay curve shown in Figure 3. The pressure suppression system for Test S-28-1 (Revision 1) will be controlled to maintain a containment pressure of 241 kPa throughout the blowdown and the reflood portions of the test.

Section II of this report presents a brief description of the analysis methods and FLOOD4 models used in these predictions, and the results of the calculations. Section III presents the more significant conclusions arising from the predictions. A more detailed discussion of the FLOOD4 model is included in Appendix A.

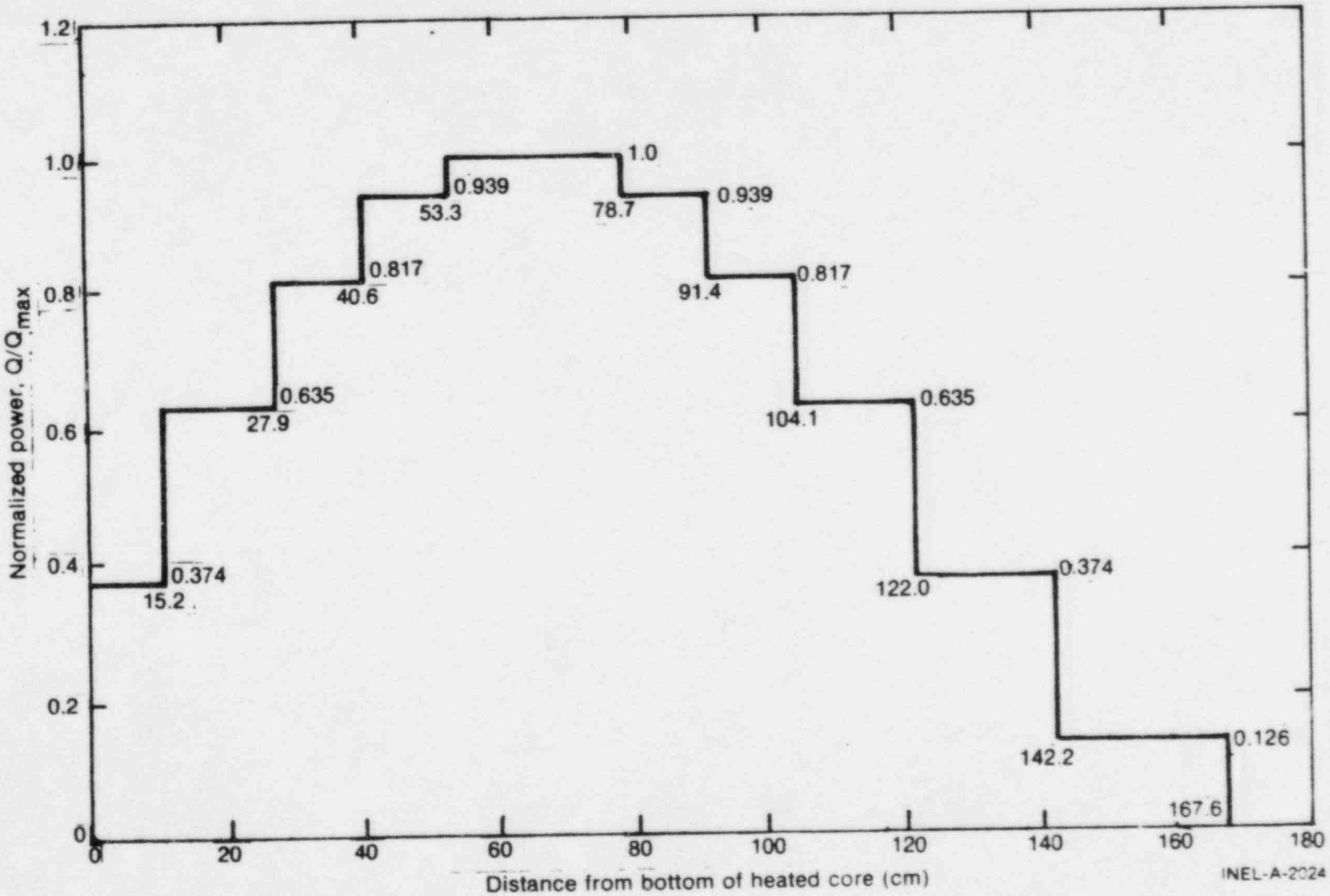


Figure 2. Normalized Axial Power Profile for Test S-28-1 (Revision 1)

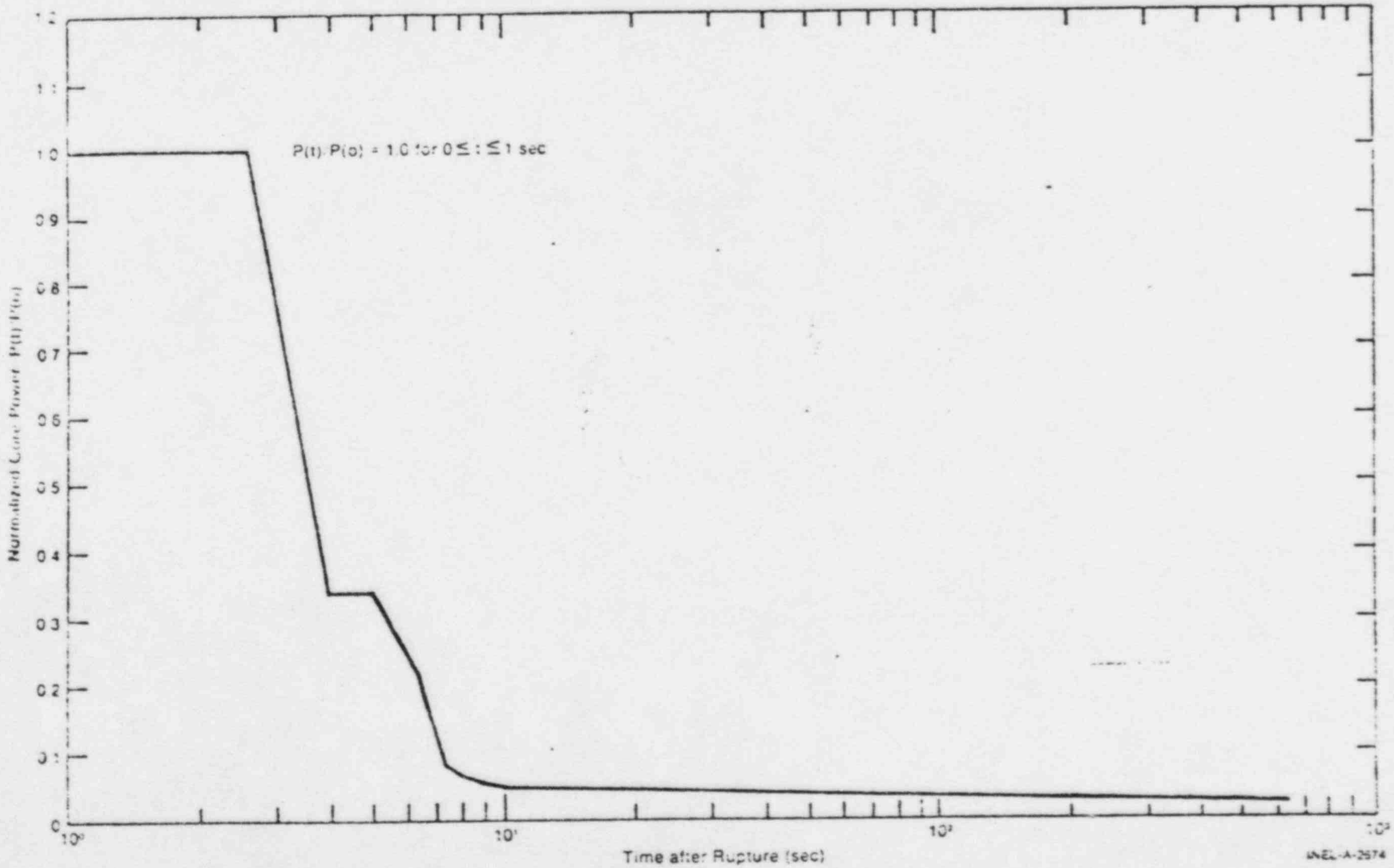


Figure 3. Decay Heat Curve for Test S-28-1 (Revision 1)

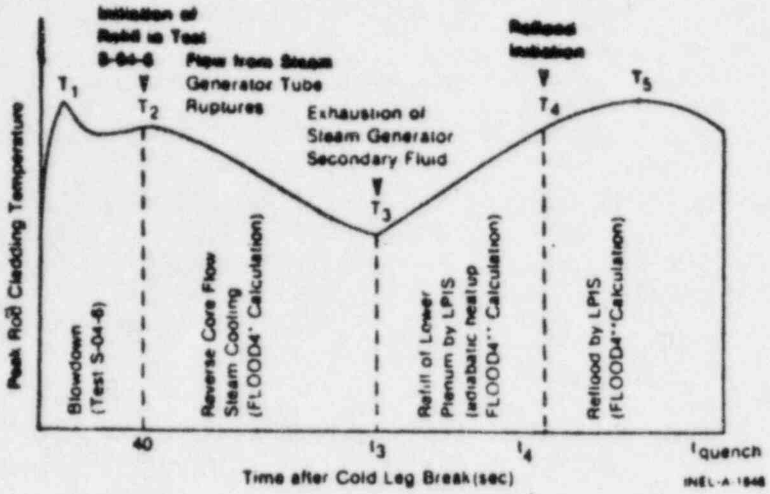
II. PREDICTIONS OF STEAM GENERATOR TUBE RUPTURE TEST S-28-1 (REVISION 1)

1. METHOD OF ANALYSIS

The analysis of a loss-of-coolant accident (LOCA) involving steam generator tube ruptures required that a different method of analysis be used to predict the response of the Semiscale Mod-1 system because of the long transient (over 316 seconds to initiation of reflood) expected for this type of LOCA. The long transient was a factor in deciding against using the analysis methods used for previous pretest predictions which consisted of using RELAP4 to predict the blowdown and refill response and FLOOD4 to predict the reflood reponse of the system. An initial study of the type of phenomena expected in Test S-28-1 (Revision 1) and the type of calculations required indicated that the best way to proceed would be to use the methods developed for the scoping analysis described in EOS Appendix 28, Addendum 28-A (see Reference 1). This method is described below and summarized in Figure 4.

The transient for Test S-28-1 (Revision 1) can be divided into four main time periods. These periods consist of: (1) the blowdown period prior to the steam generator tube rupture, (2) a period of reverse core flow after the tubes rupture and lasting until the steam generator secondary empties, (3) heat-up of the core as the lower plenum is refilled by the LPIS, and HPIS, and (4) core reflood by the LPIS and HPIS.

The baseline test for Test S-28-1 (Revision 1) is Test S-04-6. Test S-28-1 (Revision 1) will differ from Test S-04-6 only in that it will include steam generator accumulator injection into the intact loop hot leg just upstream of the steam generator inlet plenum beginning at



Total Rupture Mass Flow (kg/s)	Temperature (K)					Time (sec)			Heat Transfer# Coefficient (kW/m^2-.K)
	T_1	T_2	T_3	T_4	T_5	t_3	t_4	t_5	
	1075	994	828	1020	1031	242	316	324	.14

* For historical configuration control, load module FLOOD4 102 (configuration control number HB00120IB) was used for this section of the study.

** Load module FLOOD4 103 (configuration control number HB00121IB) was used for this section of the study.

\# Steam cooling heat transfer coefficient.

Figure 4. Analysis Technique Used in Pretest Prediction for Test S-28-1 (Revision 1)

40 seconds after rupture to represent the flow from the ruptured tubes. Since the initial conditions for Tests S-28-1 (Revision 1) and S-04-6 will be the same, the Mod-1 system response during the first 40 seconds of the transient of Test S-28-1 (Revision 1) should be the same as the system response in Test S-04-6. Because of the similarity expected between the two tests, Test S-04-6 data is used to indicate the expected system thermal-hydraulic response prior to 40 seconds.

The period of reverse steam flow through the core, which is caused by the injection simulating the tube ruptures, lasts until 242 seconds after rupture. This end time is based on how long it would take to empty the steam generator secondary at a tube rupture flow rate of 0.54 kg/s (60 tubes ruptured). It was assumed during this period that the core would be cooled by single phase forced convection heat transfer to steam. To estimate the amount of core cooling which would occur, the FLOOD4 program (load module FLOOD4/102)[a] was used in the following manner. From Test S-04-6 data, the peak temperature in the core at 40 seconds was 994 K. This temperature was used as input into the FLOOD4 code, and a cosine curve fit was used to calculate an initial axial temperature profile at the start of the reverse core flow period. The heat transfer from the rods to the steam during the period of reverse core flow was calculated by the FLOOD4 program using a heat transfer coefficient calculated from the Dittus-Boelter correlation. The heat transfer coefficient that was calculated for a tube rupture flow rate of 0.54 kg/s was 0.14 kW/m^2-K. Since the current version of FLOOD4 is not able to calculate sustained periods of negative core flow, a special

[a] For the purpose of historical configuration control, FLOOD4/102 is referenced as program number H001201B.

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The period of reverse steam flow through the core, which is caused by the injection simulating the tube ruptures, lasts until 242 seconds after rupture. This end time is based on how long it would take to empty the steam generator secondary at a tube rupture flow rate of 0.54 kg/s (60 tubes ruptured). It was assumed during this period that the core would be cooled by single phase forced convection heat transfer to steam. To estimate the amount of core cooling which would occur, the FLOOD4 program (load module FLOOD4/102)[a] was used in the following manner. From Test S-04-6 data, the peak temperature in the core at 40 seconds was 994 K. This temperature was used as input into the FLOOD4 code, and a cosine curve fit was used to calculate an initial axial temperature profile at the start of the reverse core flow period. The heat transfer from the rods to the steam during the period of reverse core flow was calculated by the FLOOD4 program using a heat transfer coefficient calculated from the Dittus-Boelter correlation. The heat transfer coefficient that was calculated for a tube rupture flowrate of 0.54 kg/s was 0.14 kW/m^2-K. Since the current version of FLOOD4 is not able to calculate sustained periods of negative core flow, a special

[a] For the purpose of historical configuration control, FLOOD4/102 is referenced as program number H00120IB.

version of FLOOD4 (FLOOD4/102), in which the initial axial temperature profile was reversed and positive core flow was modeled, was constructed to perform this calculation. The magnitude of the steam flow through the core was estimated by assuming that 30% of the water injected to simulate the flow from the ruptured tubes flashed to steam on entering the intact loop hot leg, and that a flow split between the intact loop, core, and broken loop occurred where 65.2% of the steam flowed through the core. The flow split was estimated on the basis of the intact and broken loop and core hydraulic resistances. This core flow was then used in the calculation of the heat transfer coefficient from the Dittus-Boelter correlation and in the FLOOD4 calculation. The assumption of single phase heat transfer to steam from 40 to 242 seconds is probably a conservative assumption as the core flow during this period is expected to be a two-phase mixture of water and steam and heat transfer to the liquid portion of the flow in the core was neglected in the analysis of the heat transfer during this period.

The temperature distribution at the end of the reverse core flow calculation was assumed to be the rod temperature distribution at the initiation of the refill period. All intact loop ECC injected during the period of tube rupture flow is assumed to bypass out the break. The time period for refill was estimated by assuming that the lower plenum must be refilled by the LPIS and HPIS flow alone. (The refill period was calculated to be about 74 seconds). The rod temperature distribution at the initiation of refill was input into the FLOOD4 code (load module FLOOD4/103)[a] and an adiabatic heat-up option (h = 0.0) was used for

[a] For the purpose of historical configuration control, FLOOD4/103 is referenced as program number H00121IB.

74 seconds to calculate the temperature at the initiation of reflood. The reflood of the core using LPIS and HPIS flow was then calculated with the FLOOD4 code (load module FLOOD4/103).

It is anticipated that several of the assumptions used in the analysis of the refill and reflood periods may result in higher calculated rod cladding temperatures than would actually occur. The use of an adiabatic heat-up (h = 0.0) may be somewhat conservative and, therefore, result in higher predicted peak rod cladding temperatures than would occur in the experiment. The use of a heat transfer coefficient of 30 W/m^2-K during the heat-up would decrease the peak rod cladding temperature by about 83 K. The potential accumulation of water in the lower plenum during the emptying of the steam generator was not included in estimating the volume of water that must be supplied by the LPIS and HPIS to completely fill the lower plenum. If a smaller volume of LPIS and HPIS liquid were needed to fill the lower plenum, the adiabatic heat-up would occur for a shorter period of time, which would also result in lower calculated rod cladding temperatures.

The FLOOD4 code does not account for liquid depletion in the downcomer because it does not calculate downcomer wall heat transfer. This downcomer wall heat transfer, noted in previous Semiscale tests, could impede the refilling of the lower plenum by the LPIS and HPIS. In this case, the experimental refill and reflood response could be somewhat different than the predicted response.

2. FLOOD4 MODEL DESCRIPTION

The FLOOD4/102 computer code was used to predict the core thermal response during the reverse steam flow portion of Test S-28-1 (Revision 1) (40 to 242 seconds after rupture) and FLOOD4/103 was used to predict the

refill and reflood portions of the test (242 to 450 seconds after rupture). The FLOOD4 code is a recently developed reflood analysis tool and, therefore, is undergoing evaluation and improvement as more test data becomes available. Figure 5 shows the FLOOD4 model of the Semiscale system. A more detailed discussion of the code and a listing of the input to the models is contained in Appendix A.

FLOOD4/102 is a modified version of the FLOOD4 code. It was modified to allow the FLOOD4 heat-up option to be used to predict the rod cladding and core fluid temperature response during the period of reverse steam flow. Since the current version of the FLOOD4 heat-up option is not able to model sustained periods of negative core flow, the code was modified internally to invert the initial axial temperature profile and then positive core flow was used in the FLOOD4 model to enable it to calculate the heat transfer from the rods to the steam flow.
In this way, the cooling of the rod temperatures due to the reverse steam flow from 40 to 242 seconds after rupture was calculated. The temperature profile at 242 seconds was then input into the FLOOD4 model to calculate the refill and reflood response. FLOOD4/103 was the version of the FLOOD4 code used to make this calculation. The FLOOD4 heat-up option was used to model the assumed adiabatic heat-up of the rods while the lower plenum was refilled by the LPIS and HPIS. The temperature profile at the end of the adiabatic heat-up was used as the initial temperature profile at the start of reflood. The reflood of the core by the LPIS and HPIS was then calculated with the FLOOD4 code. The rod axial temperature distribution at 40 seconds after rupture is shown in Figure 6, and at the start of refill in Figure 7. The rod axial temperature

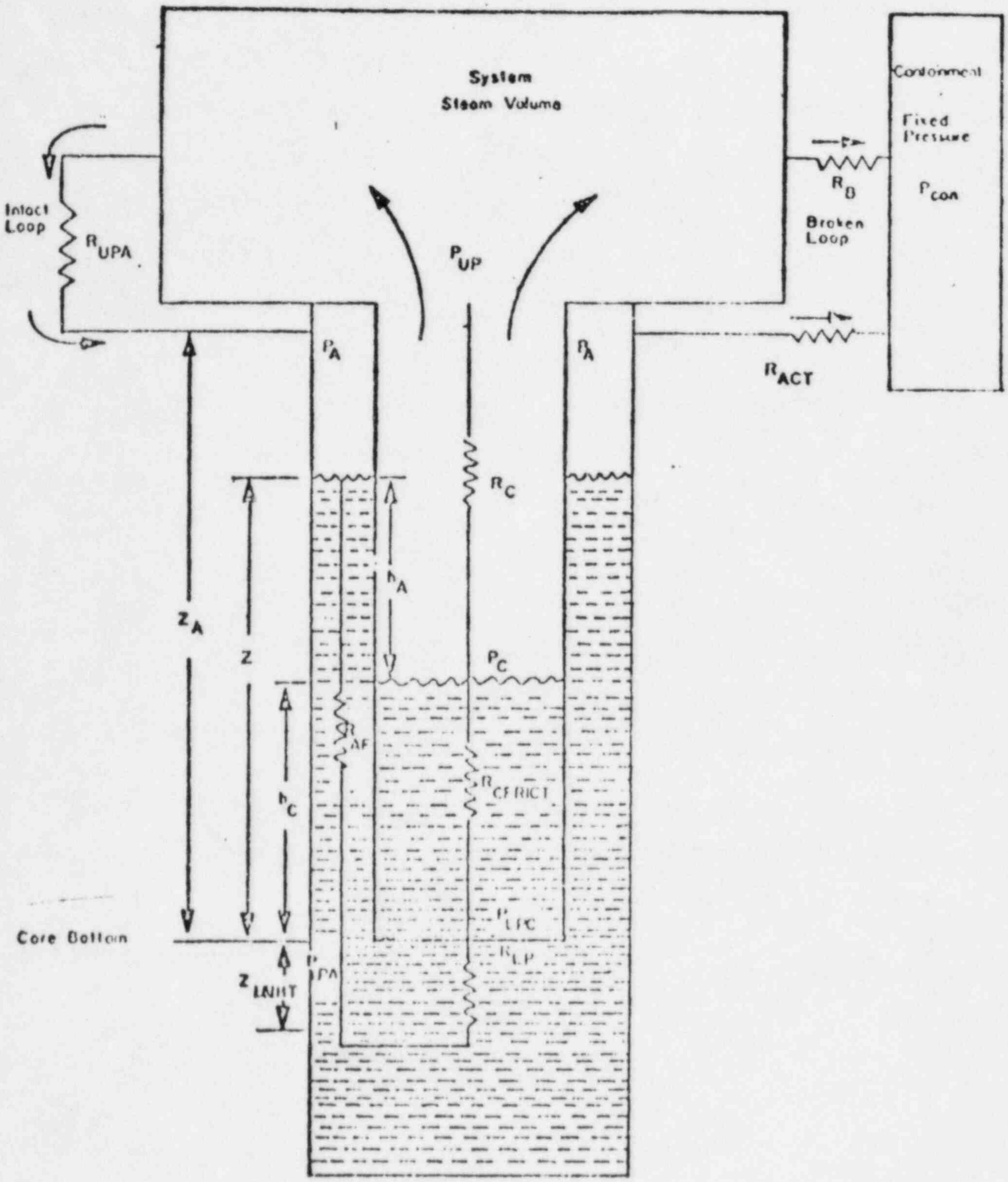

Figure 5. FLOOD4 Model of Semiscale System

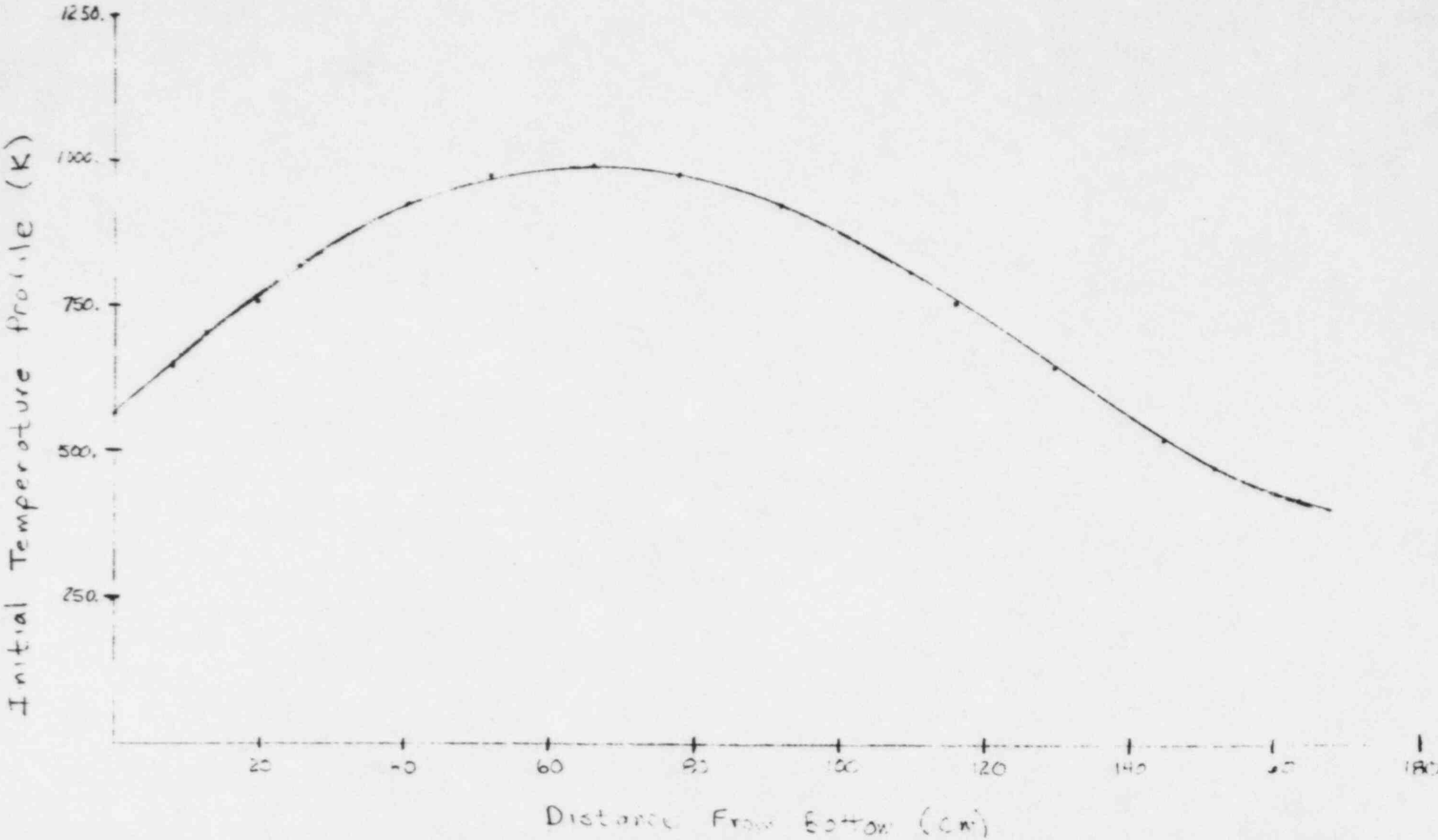

Figure 6. Rod Temperature Profile at Initiation of Simulated Tube Rupture Flow

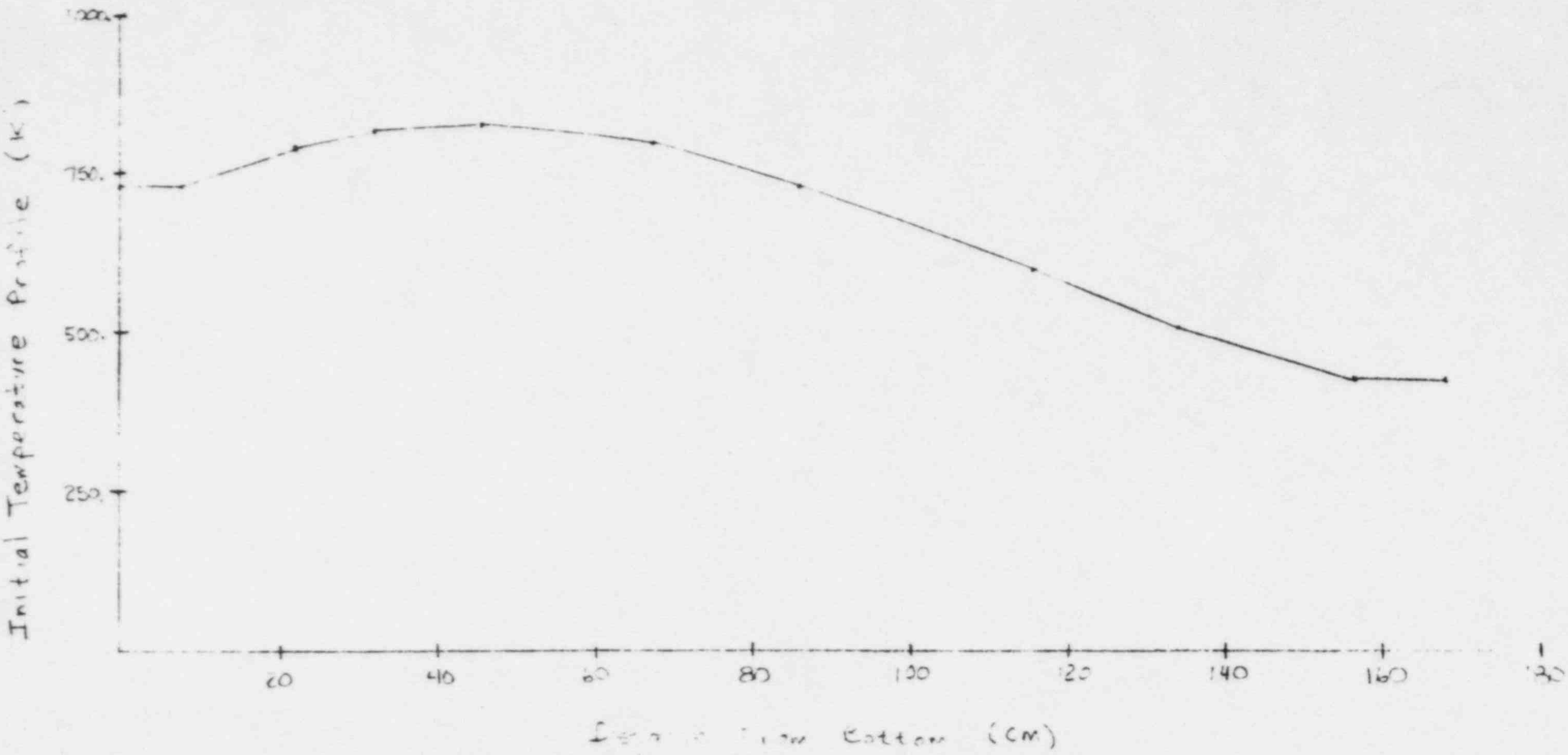

Figure 7. Rod Temperature Profile at Start of Refill

distribution used at the start of reflood is shown in Figure 8. Table II lists the initial conditions used in the FLOOD4 model at the start of reflood for Test S-28-1 (Revision 1). The FLOOD4 calculations for Test S-28-1 (Revision 1) provide a prediction of the thermal-hydraulic response for the reverse core flow, refill, and reflood processes over the time period from 40 seconds to 450 seconds following rupture.

3. PREDICTIONS OF THE SEMISCALE MOD-1 SYSTEM RESPONSE

Predicted behavior of key system parameters for Test S-28-1 (Revision 1) are presented and discussed in this section.

3.1 Blowdown Response Prior to Steam Generator Tube Rupture

Since the initial conditions for Tests S-04-6 and S-28-1 (Revision 1) are the same, the system response in Test S-28-1 (Revision 1) should be essentially the same as in Test S-04-6 until steam generator injection into the intact loop between the pressurizer and the steam generator inlet plenum begins at 40 seconds after rupture. A detailed discussion of the system thermal-hydraulic response in Test S-04-6 is contained in Reference 2 and therefore only a brief discussion is included here. Several results from the blowdown period of Test S-04-6 which are of interest in Test S-28-1 (Revision 1) are described below.

The peak temperature in the core during the blowdown period of Test S-04-6 occurred on a rod located on the perimeter of the core and reached approximately 1075 K at 8 seconds after rupture. Test data shows this temperature declined to 994 K at 40 seconds as shown in Figure 9. Test S-04-6 data indicated the system pressure had reached containment pressure (241 kPa) at 40 seconds after rupture (Figure 10). The mass flow at the core inlet (see Figure 11) at this point in time is negative (flow out of the core) and approximately .25 kg/s in magnitude.

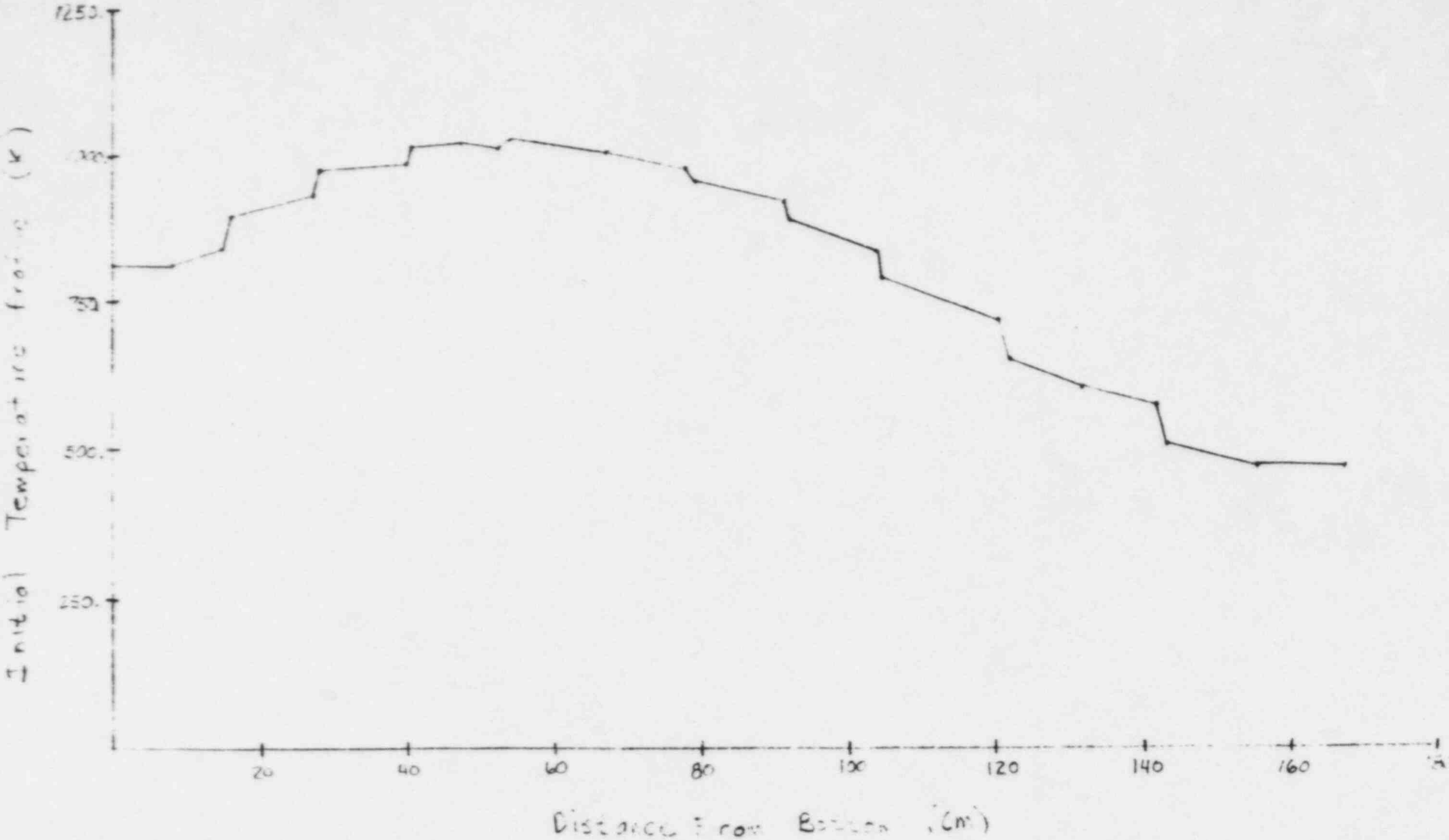

Figure 8. Rod Temperature Profile at Start of Reflood

Table II

Semiscale Mod-1 Initial Conditions At The Start of Reflood for Test S-28-1 (Revision 1)

Parameter	Initial Value
Containment Pressure	241 KPa
Temperature of ECC	
At Bottom of Heated Length	411 K (saturated)
ECC Injection Rate Cold Leg Intact Loop	
316 Seconds to Completion	$2.72 \times 10^{-4}\ m^3/s$
Peak Rod Power Density	0.99 kW/m
Power Profile	Stepped (Figure 2)
Power Decay	Refer to Figure 3
Peak Initial Rod Temperature	1021 K
Temperature Profile	Refer to Figure 8

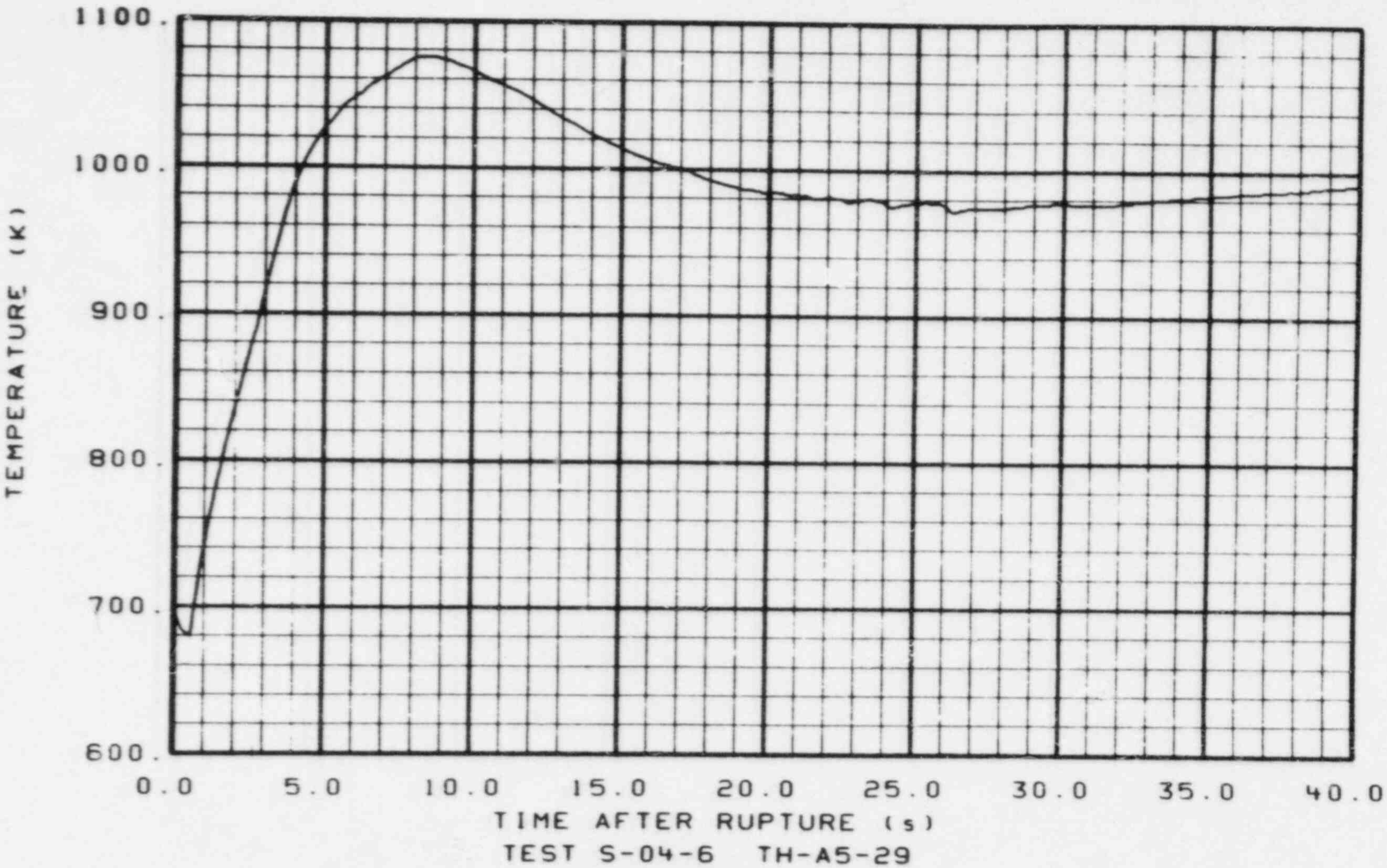

Figure 9. Measured Cladding Temperature Rod A5 (74 cm Elevation) During Blowdown in Test S-04-6

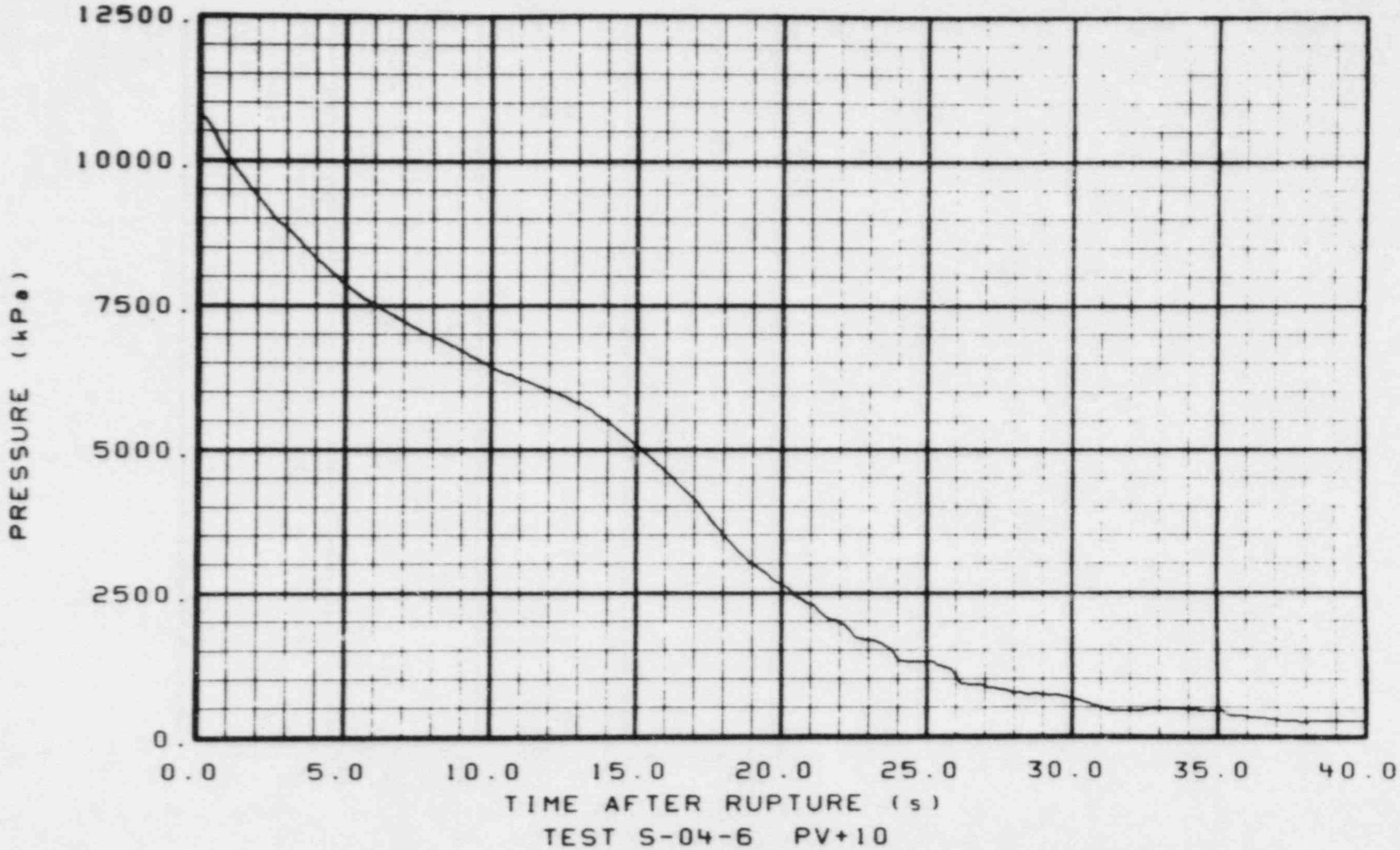

Figure 10. Measured Pressure in the Upper Plenum During Blowdown in Test S-04-6

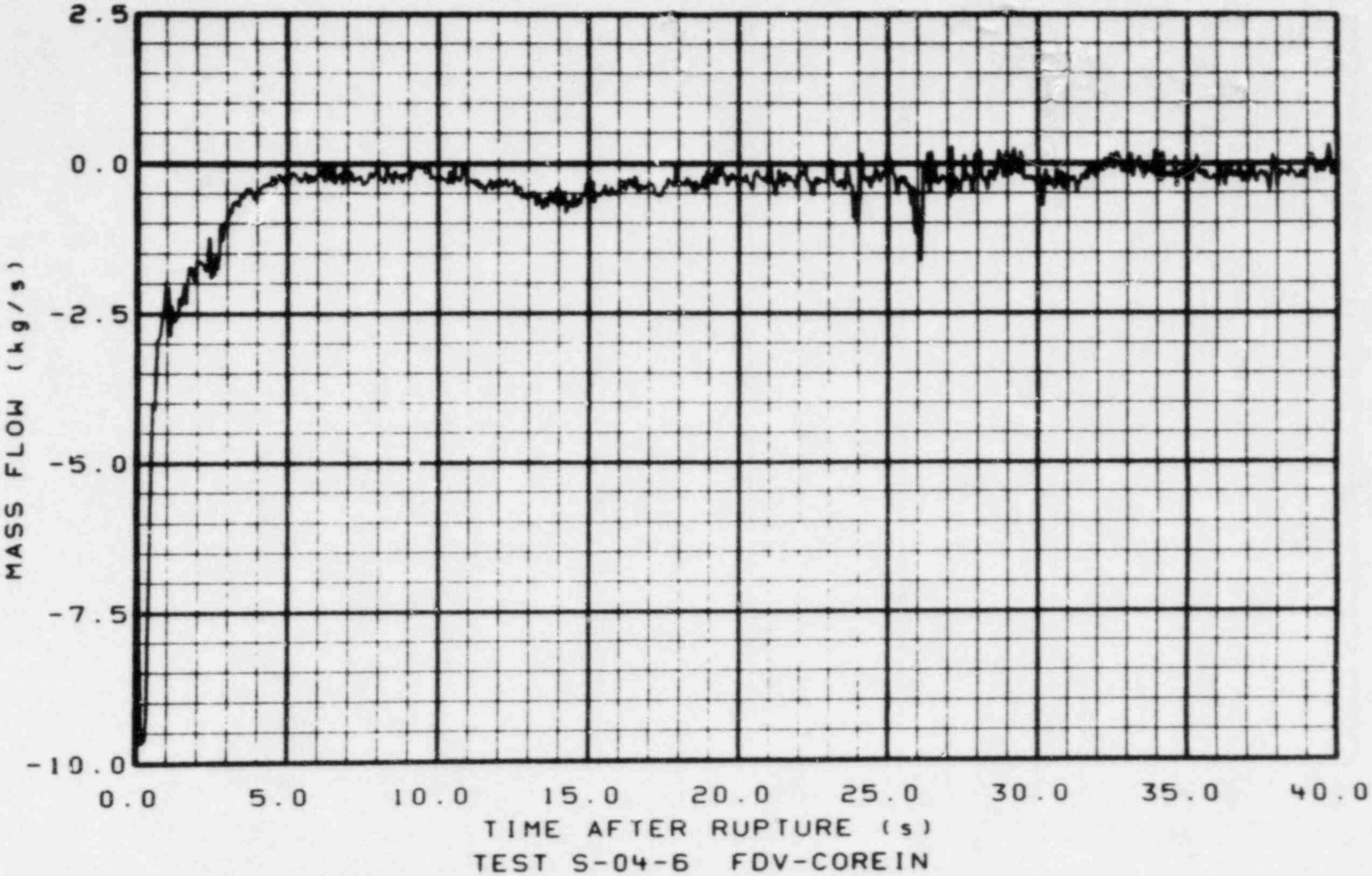

Figure 11. Measured Mass Flow at the Core Inlet During Blowdown in Test S-04-6

The steam generator secondary pressure (Figure 12) at 40 seconds was approximately 6.0 MPa and the secondary fluid temperature (Figure 13) was 549 K (saturation conditions).

3.2 System Response During Reverse Core Flow and Refill

The FLOOD4 calculation of the period of reverse steam flow through the core predicted that the peak temperature in the core would decline from 994 K at 40 seconds after rupture to 828 K at 242 seconds after rupture (compare Figures 6 and 7). Single phase heat transfer to steam was assumed during this period of reverse core flow. Somewhat better heat transfer may occur during the test because liquid is present in the flow. At 242 seconds the injection representing the tube rupture flow ended. During the heat up of the core while the lower plenum was refilled by the LPIS and HPIS, the FLOOD4 model showed the peak temperature rising from 828 K to 1021 K at the beginning of reflood (compare Figures 7 and 8). Reflood from the bottom is estimated to start at 316 seconds after rupture.

The FLOOD4 code cannot account for the downcomer mass depletion phenomena noted in previous Semiscale tests. This phenomena is a result of an excessively large amount of energy transfer from the downcomer walls to the downcomer gap after the liquid is depleted from the accumulator. Since the accumulator empties at approximately 70 seconds after rupture, and refill of the lower plenum must be accomplished by LPIS and HPIS flow alone, the downcomer mass depletion could cause the measured refill phenomena to be different than the predicted response.

3.3 System Response During Reflood

The FLOOD4 code was used to predict the system response during reflood for Test S-28-1 (Revision 1). Hand calculations were done to determine the time at which reflood would commence. These calculations

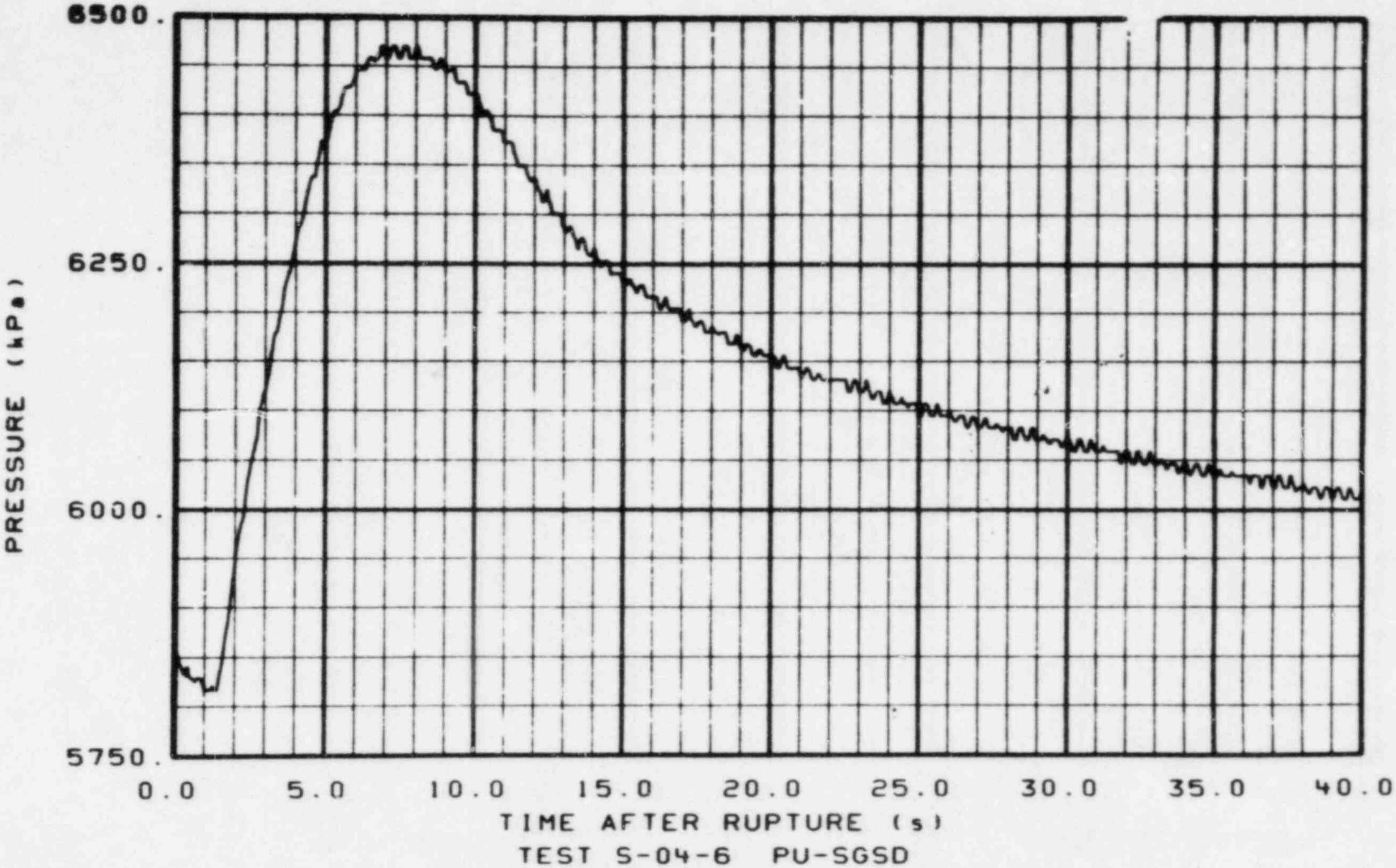

Figure 12. Measured Pressure in the Steam Generator Secondary During Blowdown in Test S-04-6

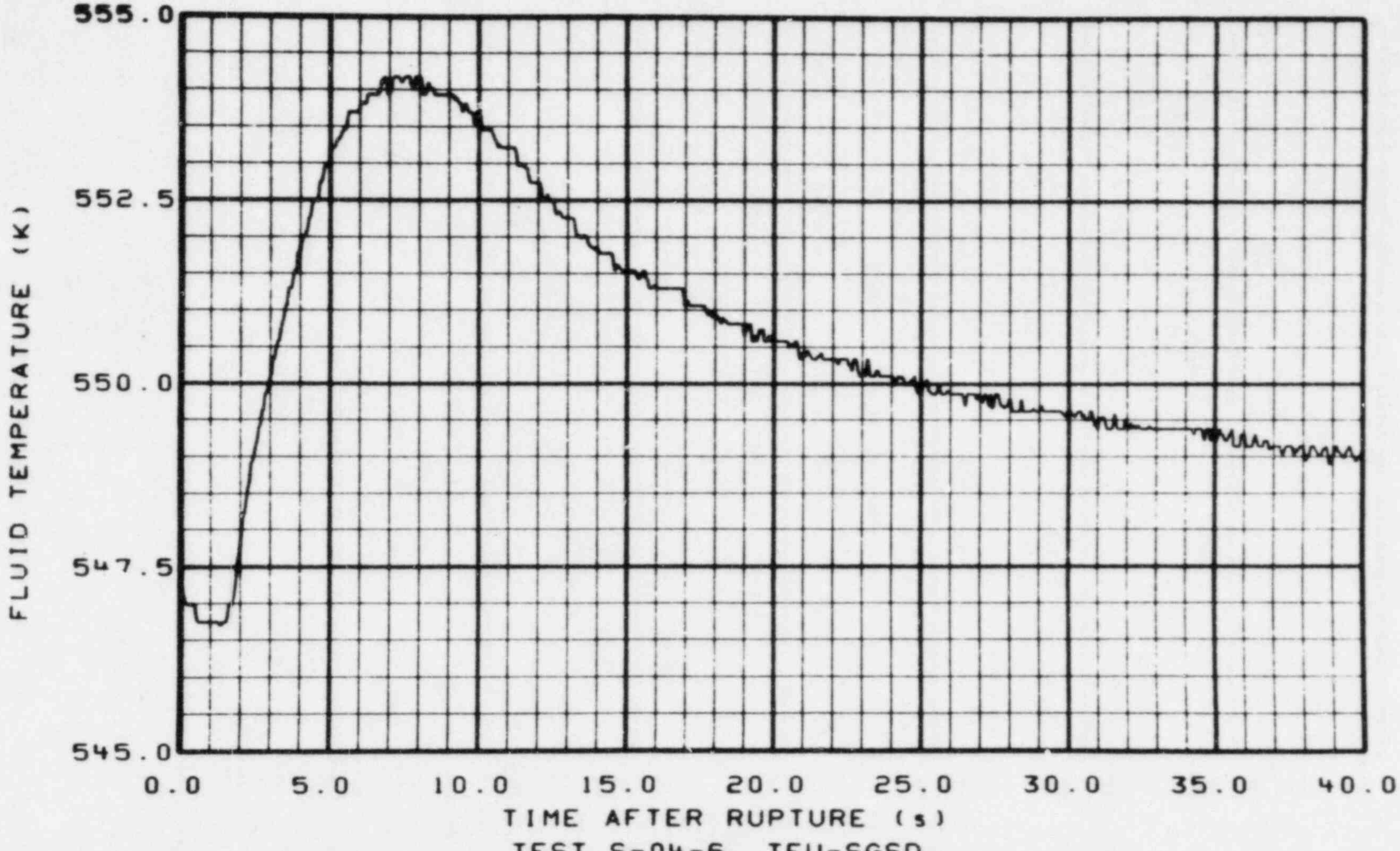

Figure 13. Measured Fluid Temperature in the Steam Generator Secondary During Blowdown in Test S-04-6

were based on the time for the steam generator secondary to empty, and the lower plenum to refill by the LPIS and HPIS. The results indicated core reflood in Test S-28-1 (Revision 1) would start at approximately 316 seconds. The calculated reflood results presented in this section were done assuming the rod power densities at the time of reflood were reflective of the high power rod power density. This is a conservative assumption since the core power profile will be radially peaked during Test S-28-1 (Revision 1).

Figure 14 shows that the FLOOD4 calculations predict that the core inlet flow oscillates both positively and negatively as for previous predictions. A comparison of the data in Figure 15, a plot of differential pressure between the upper plenum and the inlet annulus, to the data in Figure 16, a plot of the steam flow from the upper plenum, shows that the differential pressure between the upper plenum and the inlet annulus follows the steam generation in the core. The oscillations in the steam flow from the upper plenum are related to the oscillations in the core flow rate shown in Figure 14. A comparison of the data in these figures shows that the amplitude of the steam flow and differential pressures are directly related to the amplitude of the core flow oscillations as expected. The downcomer annulus liquid level shown on Figure 17 also shows oscillations. These oscillations are primarily responsible for the oscillations in the core flow and steam generation in the core. The manometer type oscillations result when the liquid level builds up in the downcomer and forces liquid into the core. Some of the liquid forced into the core is vaporized to steam on contact with the heater rods. This steam generation causes a small pressure increase which tends to force some fluid out of the core and also entrain some liquid up the core. The level in the downcomer then builds up again and the

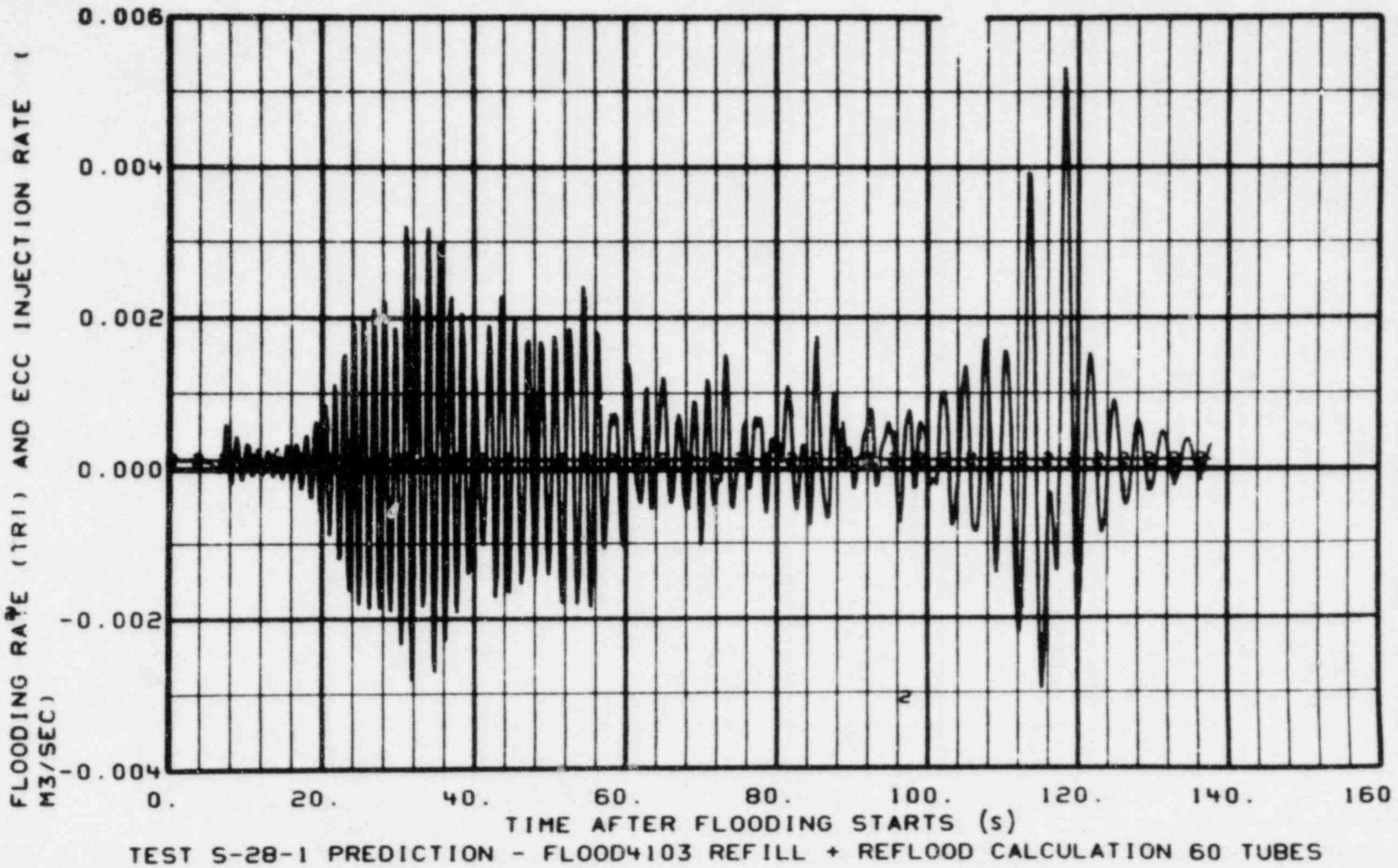

Figure 14. FLOOD4 Prediction of Core Inlet Volumetric Flow Rate

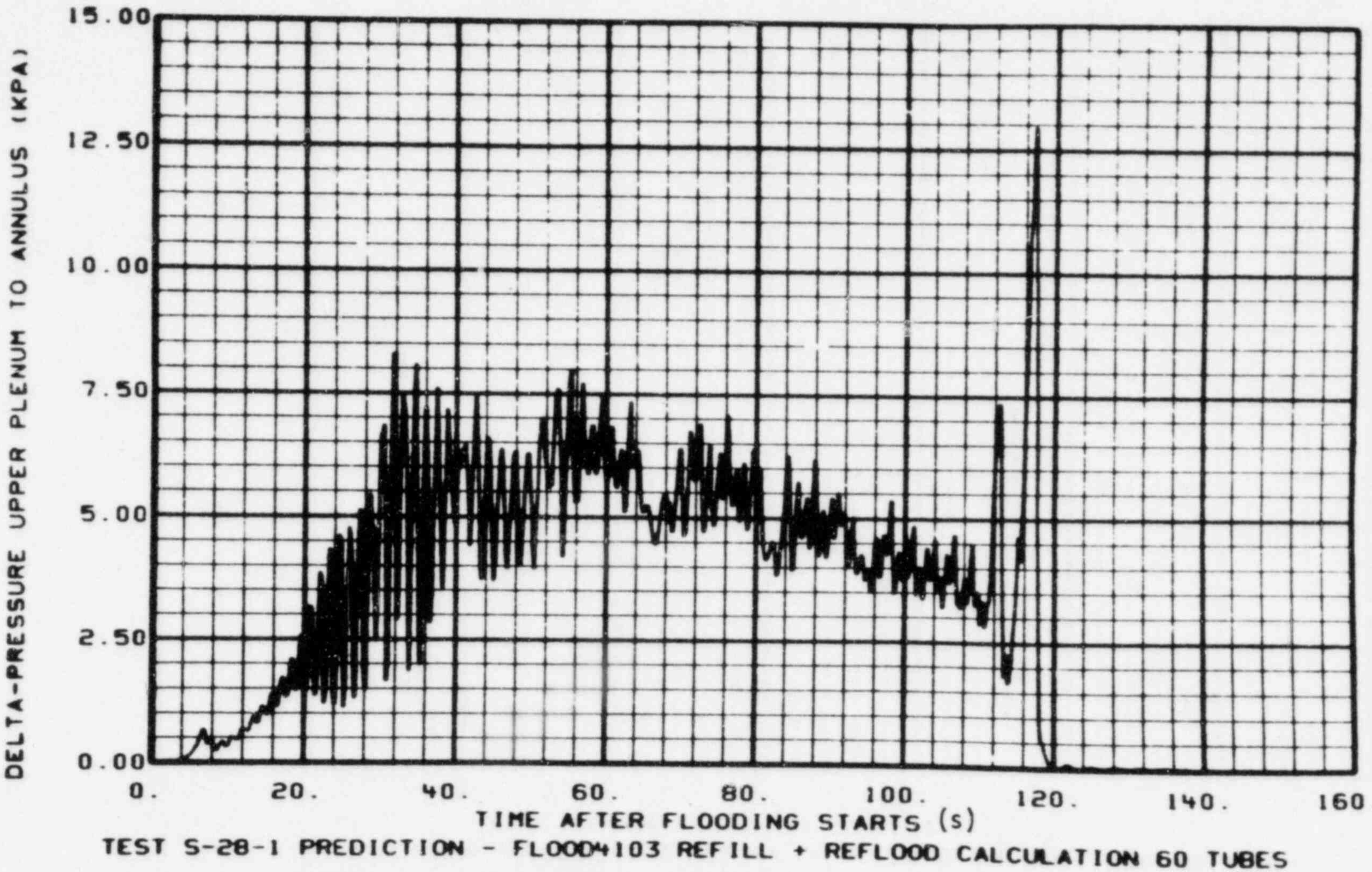

Figure 15. FLOOD4 Prediction of Differential Pressure Between the Upper Plenum and Inlet Annulus

process is repeated causing the oscillatory flows and levels shown in Figures 14 through 17. The core quench levels shown in Figure 17 indicate that the core quenched from both the top and the bottom and that all rod surfaces will be quenched by about 118 seconds after the initiation of reflood, or 434 seconds after rupture. The rate of change of liquid level for the downcomer annulus relative to the core is slightly different because the annulus water that enters the core and is evaporated to steam is not included in the core collapsed level calculation. The collapsed core liquid level is used only as a calculational parameter for heat transfer specification. That the core collapsed liquid level is shown to be higher than the heated length does not indicate that the core is completely full of liquid. The collapsed liquid level includes liquid in the regions above the heated length which would fill the core if collapsed. This collapse will not occur because steam flow in the core causes entrainment.

The FLOOD4 code cannot account for the downcomer mass depletion phenomena noted in previous Semiscale tests. This phenomena is a result of an excessively large amount of energy transfer from the downcomer walls to the fluid in the downcomer gap after the liquid is depleted from the accumulator. The mass depletion from the downcomer causes a reduction in the downcomer liquid head which in turn causes a reduction in the core reflood driving potential (the annulus water level shown in Figure 17 would be lower if mass depletion were taken into account). The depletion could cause the measured reflood phenomena to be somewhat different than the predicted reflood response.

The FLOOD4 predicted thermal response of the Semiscale Mod-1 core for the reflood portion of Test S-28-1 (Revision 1) is presented in

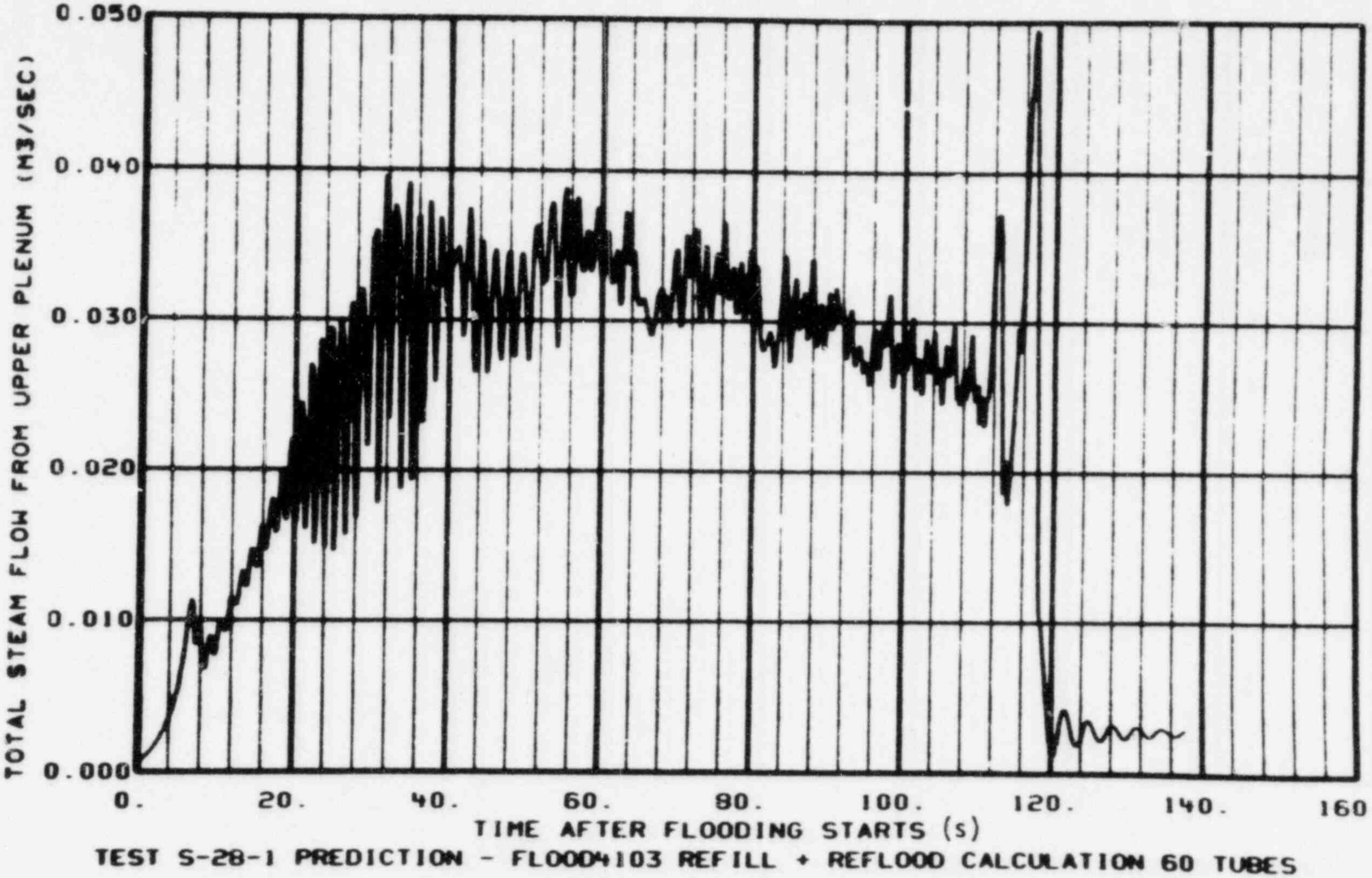

Figure 16. FLOOD4 Prediction of Total Steam Flow from the Vessel Upper Plenum During Reflood

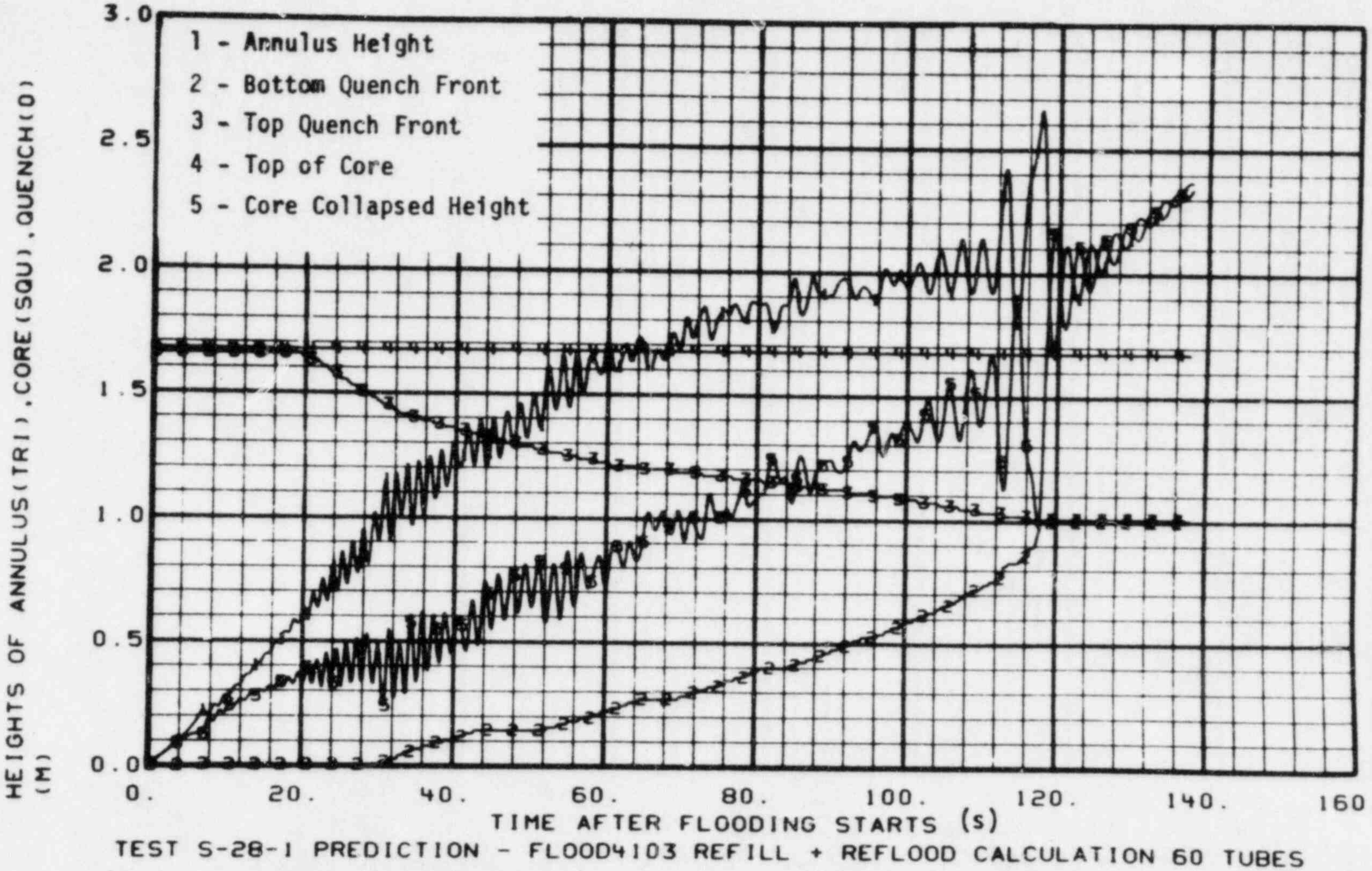

Figure 17. FLOOD4 Prediction of Water and Quench Front Heights During Reflood

Figures 18 and 19. Core elevations of 20, 36, 53, 63.5, and 99 cm above the heated length were chosen for presentation of predicted parameters because they correspond to existing core heater rod thermocouple measurements. The predicted heat transfer coefficients rapidly increase to a fairly constant value (Figure 18) similar to the dispersed flow heat transfer coefficients observed in previous tests. In Figure 18, the lower elevations are shown to have the higher heat transfer coefficients in this dispersed flow film boiling regime because the fluid quality in the lower core region is lower relative to the fluid quality in the upper core region. When the quench front approaches an elevation, the heat transfer coefficient increases very rapidly as the heat transfer regime switches to transition boiling. After a rod position is quenched the heat transfer regime is nucleate boiling and forced convection to liquid with fairly constant heat transfer coefficients between 2.0 and 3.0 kW/m^2-K. The predicted rod surface temperature response for various elevations is shown in Figure 19. The temperature at each elevation shows a continual decrease while in the dispersed flow film boiling regime until the quench front approaches that elevation; then the rod surface temperature decreases very rapidly. This rapid decrease in temperature is a result of the prediction of transition boiling. All the predicted rod surface temperature responses approach a constant value corresponding to nucleate boiling and forced convection to liquid. The predicted surface heat flux at the hot spot (63.5 cm elevation) is shown in Figure 20. The quench time is about 417 seconds after rupture (101 seconds after reflooding starts) for this location and the critical heat flux during quench is about 1250.0 kW/m^2.

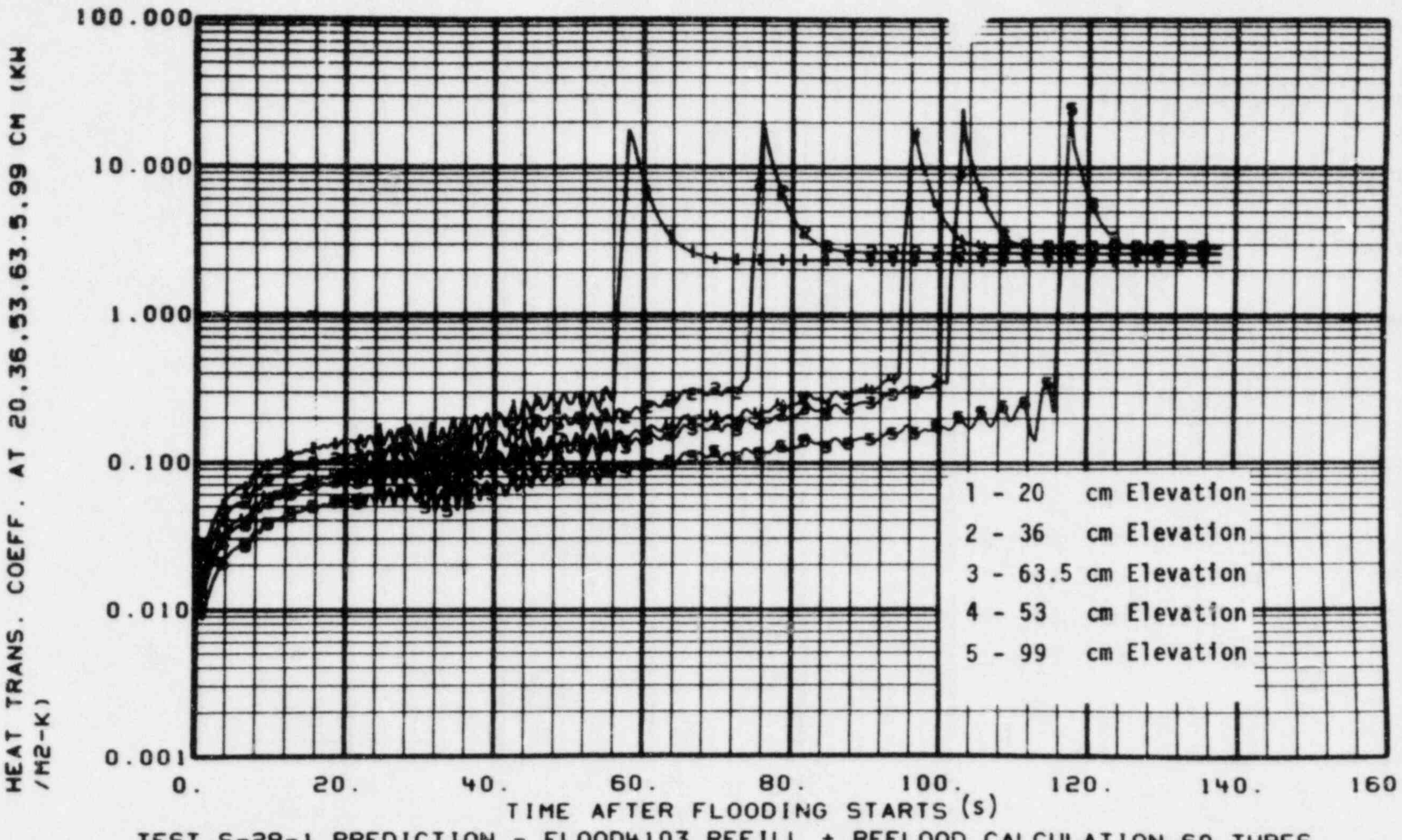

Figure 18. FLOOD4 Prediction of the Heat Transfer Coefficient at the 20, 36, 53, 63.5, and 99 cm Elevations

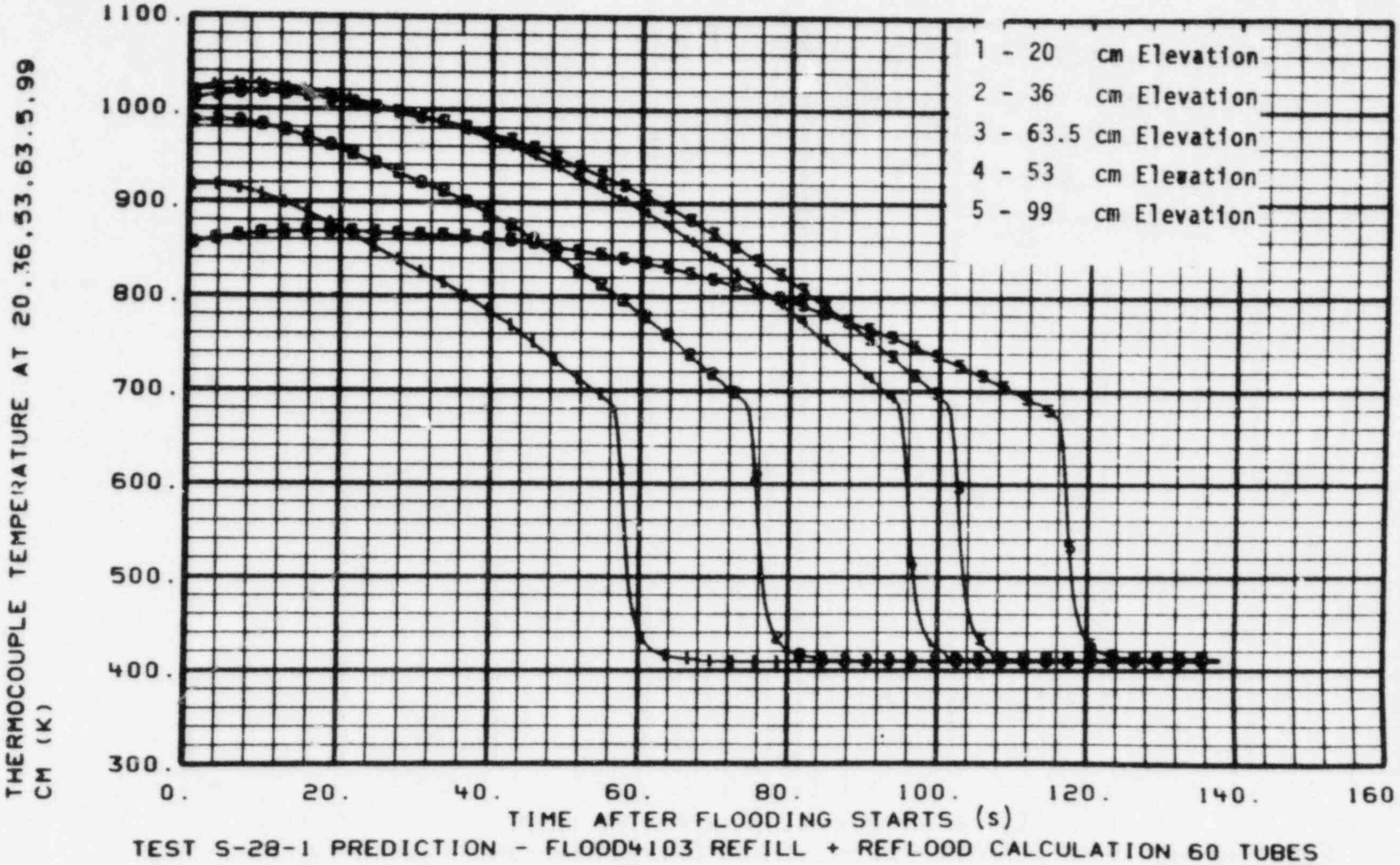

Figure 19. FLOOD4 Prediction of the Rod Temperatures at the 20, 36, 53, 63.5, and 99 cm Elevations During Reflood

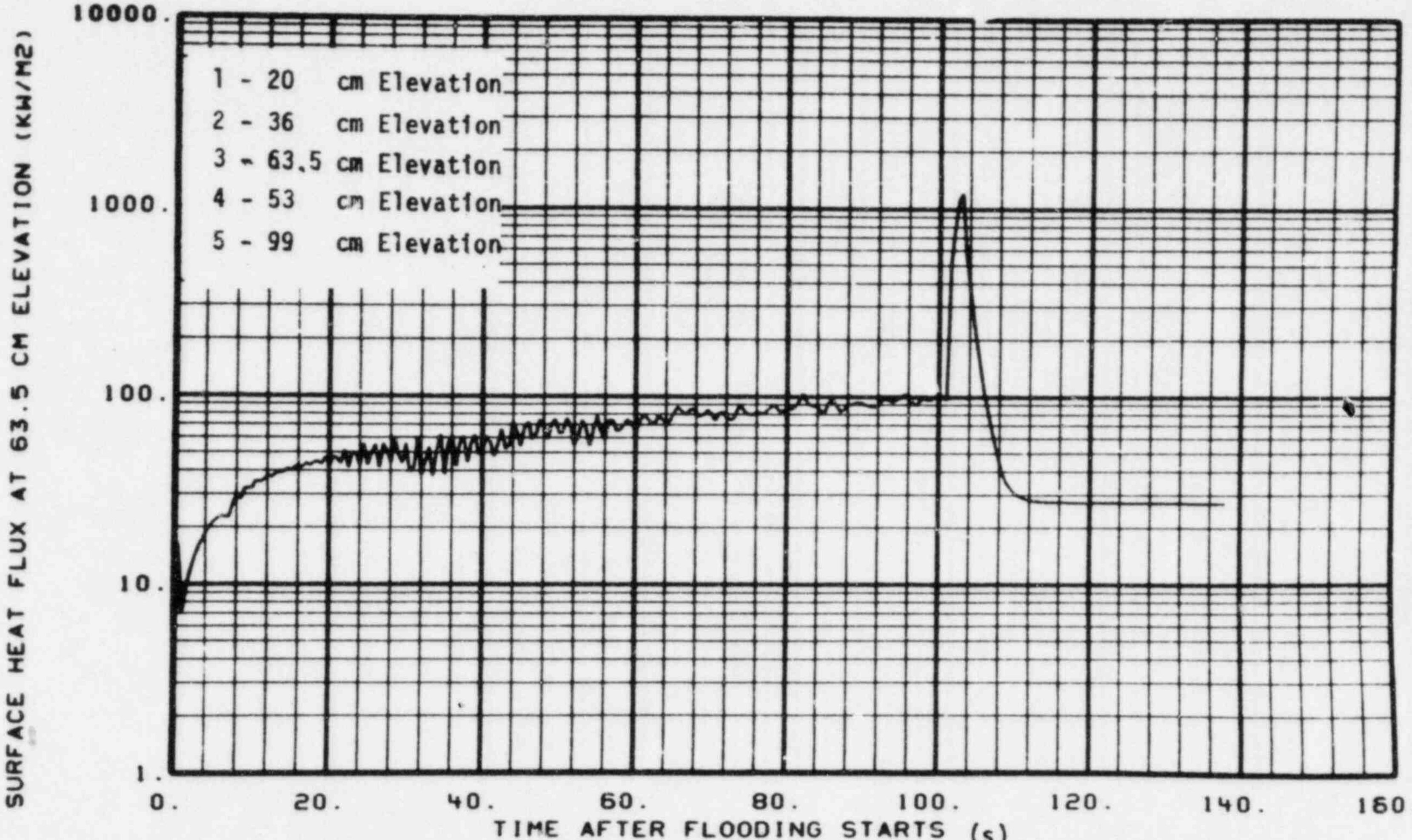

Figure 20. FLOOD4 Prediction of the Surface Heat Flux at the Rod Hot Spot (63.5 cm Elevation) During Reflood

In summary, the peak rod temperature in the core during blowdown should be approximately 1075 K at 8 seconds after rupture. This peak temperature should decline to 994 K at 40 seconds after rupture. During the period of reverse steam flow through the core the peak rod temperature should decline to 828 K. During the refill period the calculations indicated the peak core temperature would increase to 1021 K at the beginning of reflood. The peak temperature increases slightly during reflood to 1030 K before the rod hot spot quenches at 101 seconds after reflood or 417 seconds after rupture. This rod surface temperature response is summarized in Figure 21.

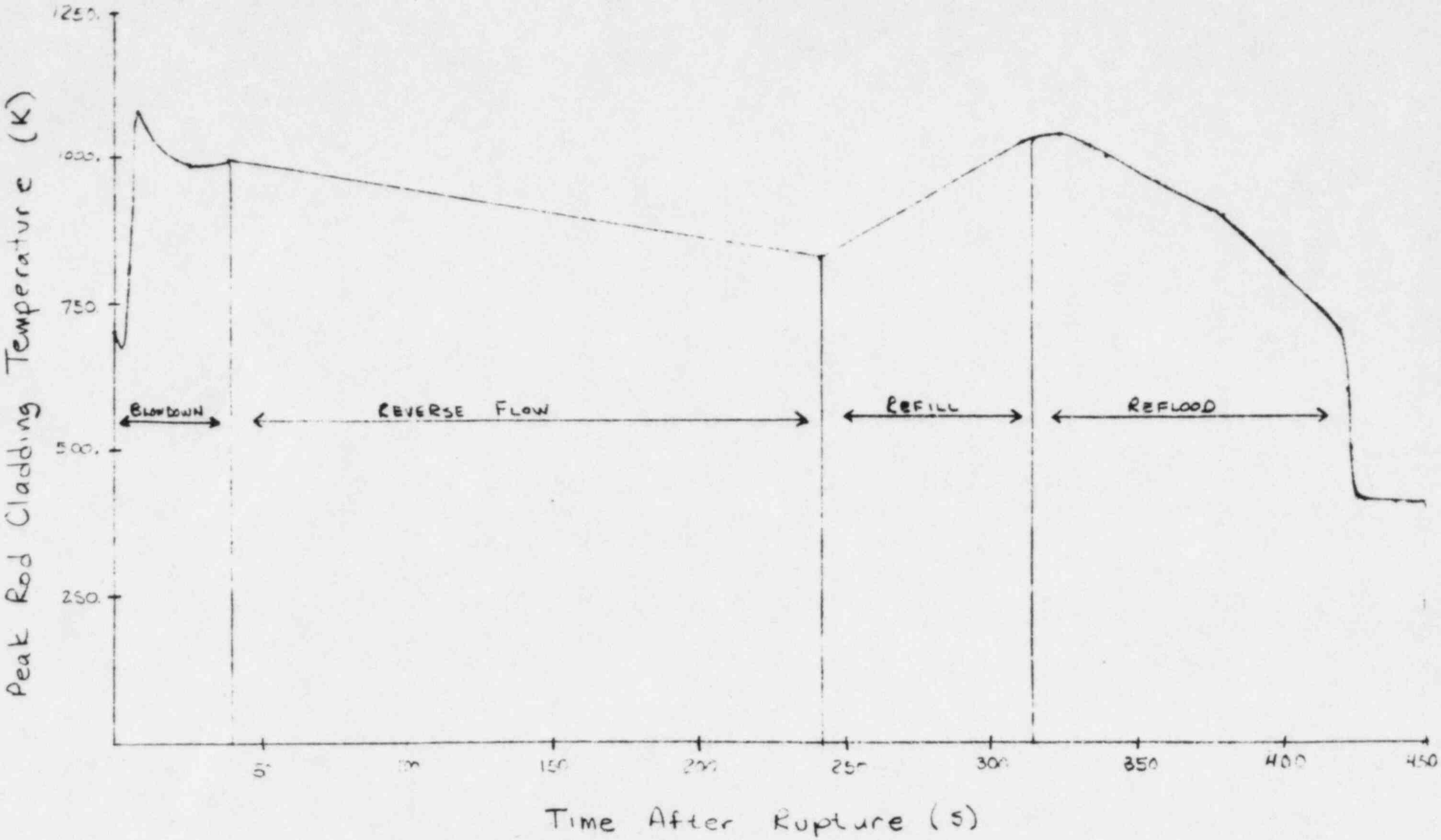

Figure 21. Peak Rod Cladding Temperature During Test S-28-1 (Revision 1)

III. CONCLUSIONS

The conclusions relative to the use of Test S-04-6 data to indicate the blowdown response and the FLOOD4 predictions over the rest of the transient for the Semiscale Mod-1 Test S-28-1 (Revision 1) are as follows:

(1) The system and core thermal response should be essentially identical for Tests S-28-1 (Revision 1) and S-04-6 until 40 seconds after rupture because the initial conditions for the two tests are the same.

(2) The peak temperature during blowdown should be approximately 1075 K at 8 seconds after rupture and should occur on a rod located on the core perimeter. This temperature should decline to 994 K at 40 seconds after rupture.

(3) The reverse steam flow through the core, which is caused by the tube ruptures, should cool the core over the period from 40 to 242 seconds after rupture. The peak temperature at 242 seconds should be approximately 828 K. This predicted response was based on single phase heat transfer to steam and did not account for any heat transfer to the liquid present in the core flow. Better cooling in the core may result from any liquid present in the flow.

(4) Heat-up of the core while the LPIS and HPIS refill the lower plenum should result in a peak temperature rise from 828 K to 1021 K over the period of 242 to 316 seconds after rupture.

(5) The start of reflood from the bottom is estimated to begin at 316 seconds after rupture.

(6) The hot spot elevation (63.5 cm from the bottom of the heated length) is expected to quench at about 417 seconds after rupture or about 101 seconds after the start of reflood. The whole core is predicted to quench by 118 seconds after the start of reflood (434 seconds after rupture).

IV. REFERENCES

(1) D. J. Olson Ltr to P. E. Litteneker, DJO-125-77, Transmittal of EOS Appendix 28, June 3, 1977.

(2) J. O. Zane Ltr to P. E. Litteneker, Zan-250-76, Transmittal of Quick Look Report for Semiscale Mod-1 Integral Blowdown Reflood Tests S-04-5 and S-04-6, October 15, 1976.

(3) J. O. Zane Ltr to P. E. Swanson, Zan-235-75, Test Prediction of the Third Mod-1 Semiscale Test Series, Reflood Heat Transfer Tests, Tests S-03-1, S-03-2, and S-03-3, November 19, 1975.

APPENDIX A

FLOOD4 COMPUTER CODE

APPENDIX A

FLOOD4 COMPUTER CODE

The FLOOD4 computer code is a recently developed analysis tool used to predict core reflood behavior in water reactors. The methods and models used in FLOOD4 are currently undergoing evaluation and improvement.

The FLOOD4 code couples the system hydraulics using the momentum equation for the core, lower plenum, and downcomer with the heat transfer and steam generation in the core region. Liquid which rises in the downcomer to a height greater than the cold leg is assumed to be lost from the system. The steam within the system is lumped into one gas volume and the perfect gas law is used to calculate the relationship between the steam pressure, mass, volume, and temperature. Figure A-1 illustrates the hydraulic coupling of the Semiscale system used in the FLOOD4 model. The core is represented in FLOOD4 by a series of axially stacked conduction nodes which have a specified initial temperature and energy generation rate. The heat transfer coefficient applied to a node depends on the mode of heat transfer which is determined from the elevation of the node, the elevation of the water, and the temperature of the node. Four different heat transfer modes are used to define the boiling curve and one mode is defined for forced convection to single-phase liquid below the quench level. The reference temperature used for the heat transfer calculation is T_{sat} for nodes above the water level in the core and T_{bulk} for nodes below the quench level core.

The fluid entering the upper plenum is a mixture of the steam generated in the core and entrained water. The amount of steam is determined from the heat flux at the nodes above the quench front and

the amount of entrained water is a function of the steam flow rate, the collapsed water level above the quench front, the pressure, and the core hydraulic diameter. Since all of the rods are assumed to be identical, the calculation is performed for one subchannel and rod and then multiplied by the number of rods in the core to obtain the total steam flow. The axial temperature distribution at the start of the injection simulating the tube ruptures (40 seconds) was determined by using the peak temperature in the core at that time and using a cosine curve fit to determine the temperature distribution. The initial axial temperature distribution of the rods at the start of reflood was determined by using temperature distribution at the time the FLOOD4 reverse core flow steam cooling calculation terminated as input into FLOOD4 and performing a core heat-up calculation.

FLOOD4 has several new features which are still experimental and include: (1) capability for upper plenum injection which include a condensation mode, (2) capability to have liquid fall back from the upper plenum if the core steam velocity goes below a certain value, and (3) vaporization of entrained liquid in the intact loop steam generator. Future predictions will attempt to include these features where applicable in an effort to better simulate expected behavior.

Table A-1 is a copy of the input to the FLOOD4 calculation for reverse core flow steam cooling calculation and Table A-II is a copy of the input to the refill and reflood calculation for Test S-28-1 (Revision 1).

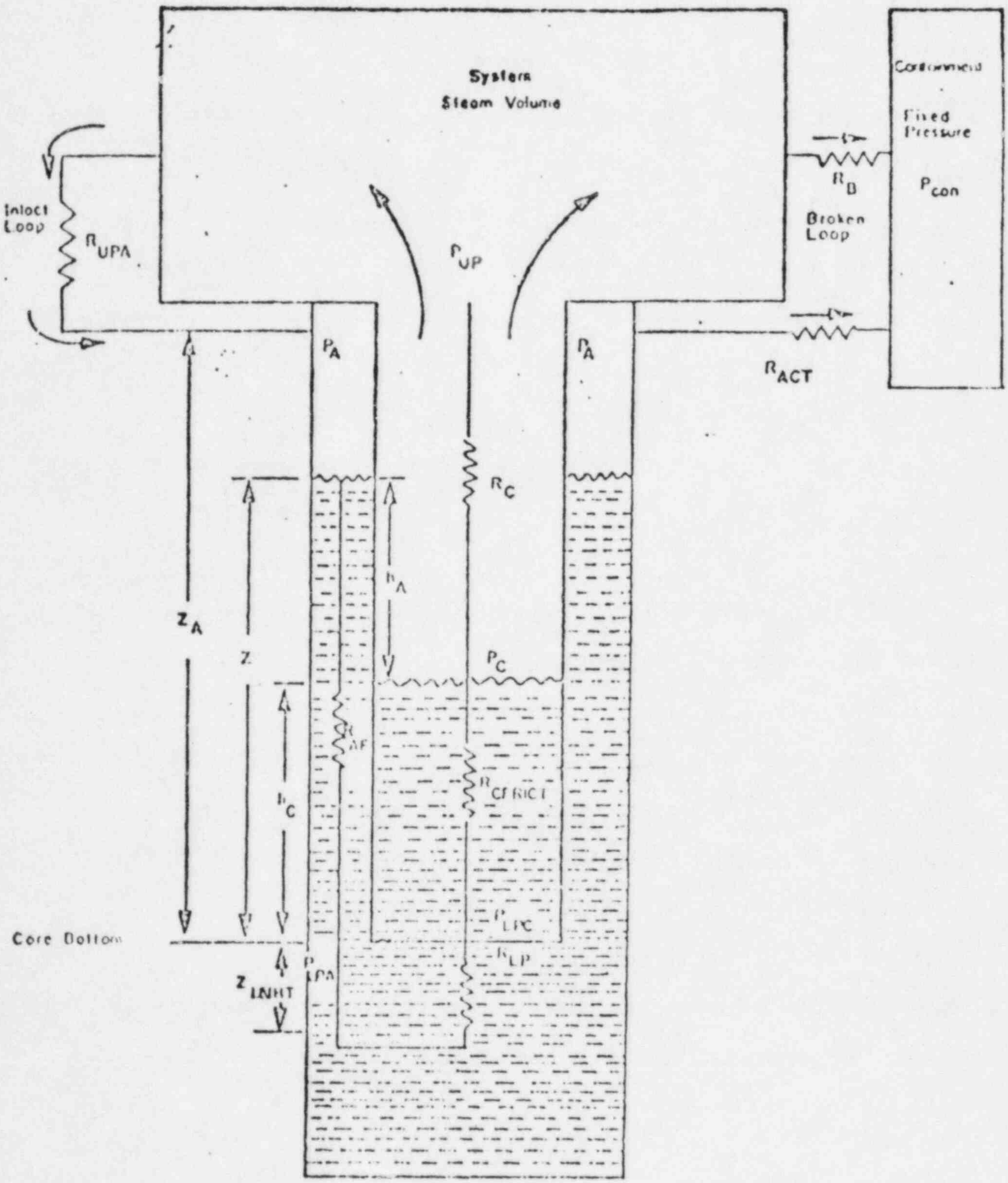

Figure A-1. FLOOD4 Model of Semiscale Mod-1 System

Table A-I. FLOOD4 Input L   ngs for the Reverse Core Flow Steam Co   , Calculation

```
0000000001111111111222222222233333333334444444444555555555566666666667777777778
1234567890123456789012345678901234567890123456789012345678901234567890123456789 0

  T  [illegible] S-28-1 PREDICTION - FLOOD4 .2 REVERSE STEAM FLOW CALCULATION
1 T     AFTER FLOODING STARTS (SEC)
2 FL   DING RATE(TRI) AND ECC INJECTION RATE (IN/SEC)           -100.        100.
3 HEIGHTS OF ANNULUS(TRI),CORE(SOL),QUENCH( ) (FT)              -2.          12.0
4 T TAL STEAM FLOW FROM UPPER PLENUM (GPM)
5   TA-PRESSURE UPPER PLENUM-ANNULUS (PSI)                       0.0          8.0
6 T  [illegible]OUPLE TEMPERATURE AT 6,14,21,25,39 IN. (F)       0.          220.
7 S   ACE HEAT FLUX AT 25 INCH ELEVATION (BTU/HR-FT2)           1000.        1000000.
8 HEAT TRANS.COEFF.AT 6,14,21,25,39 IN. (BTU/HR-FT2-F)          1.           [illegible]
9 DISTANCE FROM BOTTOM (IN)                                     0.           70.
- INITIAL TEMPERATURE PROFILE (F)                               0.           1600.
11 FLOODING RATE(TRI) AND ECC INJECTION RATE (GPM)
12 LIQUID LEVEL AND CORE EXIT QUALITY                            0.0          1.0
13 LIQUID LEVEL AND CORE EXIT STEAM FLOW (LBS/SEC)
14 CORE EXIT LIQUID VAPORIZED IN STEAM GENERATOR (LB/SEC)
15 CORE EXIT AND COLLAPSED LEVEL STEAM VELOCITY (FT/SEC)
-1
   0.55        0.52        50.        .55      300.0
  .[illegible] [illegible] .0569      30.[illegible] 0.      58.4   .0001408     264.0
   1.8         2.63        1.67       .1445     .085
 0            .0.376       .05        1.        .760      .6      0.0          0.3
   5.5         0.0         0.0        .27       .12       3.5     3.0
  .001         500.   200         .001      500.   200    .001    500.   200   100
 26[illegible].6  11.0
-1
   9.6         11.0
-1
   5.63        11.9
-1
   0.52        11.0
-1
 2            .00465      0.00       175.      940.      1.0     0.00
   0.6         .25        75380.      0.0      264.      36.0    264.         1330.
 1     1      [illegible]   40.0       0.5      242.      .05     2.00          0.2
   36.        940.        1.0        1.0      1.0       .995    .118
 1     1      2         .055        0.
 2     2      4         .[illegible]  1.
 3     4      2         .[illegible]36  [illegible].
 [illegible] [illegible] [illegible] .[illegible]  [illegible].
 6     [illegible]  1   .[illegible]29  0.
-1
 17   28   50   43   78   4    2    2   10   5   5
 1    [illegible]  0.5   0.374    825.
 13   [illegible]  0.5   0.635    1015.
 23   [illegible]  0.5   0.817    1304.
 33   [illegible]  0.5   0.939    1504.
 43   [illegible]  0.5   1.000    1563.
 63   [illegible]  0.5   0.939    1505.
 73   [illegible]  0.5   0.817    1246.
 83   [illegible]  0.5   0.635    1091.
 97   [illegible]  0.5   0.374    8[illegible]7.
 113  1[illegible]2  0.5   0.126    553.
-1

0000000001111111111222222222233333333334444444444555555555566666666667777777778
1234567890123456789012345678901234567890123456789012345678901234567890123456789 0
```

Table A-II. FLOOD4 Input Listing for the Refill and Reflood Calculation

```
TEST S-2[illegible]-1 PREDICTION - FLOOD4[illegible] REFILL + REFLOOD CALCULATION [illegible]
TIME AFTER FLOODING STARTS (SEC)
FLOODING RATE(TRI) AND ECC INJECTION RATE (IN/SEC)          -100.
HEIGHTS OF ANNULUS(TRI),CORE(SEC),[illegible] (FT)           -2.
TOTAL STEAM FLOW FROM UPPER P[illegible]
[illegible] UPPER PLENUM-ANNULUS (PSI)                          0.0
[illegible] TEMPERATURE AT 8,14,21,25,34 IN. (F)              0.
[illegible] HEAT FLUX AT 25 INCH ELEVATION (BTU/HR-FT2)        1000.
HEAT TRA S.COEFF.AT 8,14,21,25,34 IN. (BTU/HR-FT2-F)          1.
[illegible] FROM BOTTOM (IN)                                   0.
[illegible] TEMPERATURE PROFILE (F)                            0.
FLOODING RATE(TRI) AND ECC INJECTION RATE (GPM)
[illegible] AND CORE EXIT QUALITY                              0.0
[illegible] AND CORE EXIT STEAM FLOW (LBS/SEC)
[illegible] VAPORIZED [illegible] (LB/SEC)
CORE EXIT AND COLLAPSED LEVEL STEAM VELOCITY (FT/SEC)
-1
   0.5[illegible]      0.55        50.       0.55     300.00
   .[illegible]        .[illegible] .0589    [illegible]   0.       53.4   .[illegible]
   11.[illegible]      2.[illegible] 1.57    .0445     .[illegible]5
 0                     .0037[illegible] .05   1.       .[illegible]
   [illegible]         [illegible] 0.0      .07    1.[illegible]
   [illegible]         [illegible] 200     .001     500.  [illegible]   .[illegible]
  25[illegible]        11.[illegible]
-1
   9.6         11.0
-1
   5.[illegible]        11.9
-1
   0.5[illegible]       11.0
-1
 2             .00468       0.00        175.       940.        1.0
   0.6          0.25     75[illegible]0.     0.[illegible]     25[illegible]    3[illegible]
 1   1        0           242.0        0.5     [illegible]   .[illegible]5
  35.          940.        1.0        1.0       1.0      .5[illegible]
   1     2          .055          0.
                    .0[illegible]         1.
                    .0[illegible]         0.
   [illegible] [illegible] 1    .010          0.
   8    8    1      .029          0.
-1
 17    [illegible]    50    43    78     4      7    2    10    5    5
                  0.5        0.374       859.
                  0.5        0.635       [illegible]
                  0.5        0.[illegible]       10[illegible].
                  0.5        0.939       1030.
                  0.5        1.000       9[illegible]4.
                  0.5        0.939       876.
                  0.5        0.817       769.
                  0.5        0.635       642.
 97   112         0.5        0.374       4[illegible].
 1[illegible]3   13[illegible]   0.5    0.126       326.
-1
```

EG&G Idaho, Inc.

P. O. Box 1625
Idaho Falls, Idaho 83401

June 28, 1977

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

TEST PREDICTION OF SEMISCALE MOD-1 INTEGRAL TEST S-28-3 - DJO-146-77

Reference: D. J. Olson Ltr to P. E. Litteneker, DJO-125-77
Transmittal of Semiscale EOS Appendix 28,
June 3, 1977

Dear Mr. Tiller:

Enclosed is the test prediction document for Test S-28-3 of the steam generator tube rupture test series. Details of the system description and initial test conditions were transmitted in the referenced letter.

The primary objectives of Test S-28-3 are to aid in defining the core temperature response for a **small number of steam generator tube ruptures** and to probe into the range of steam generator tube ruptures indicated by the analysis used in the specification of Test Series 28 to result in high peak cladding temperatures. Test S-28-3 will be a 200% double-ended cold leg break simulation. The rupture of approximately 12 steam generator tubes will be simulated by a flow rate of 0.104 kg/s from accumulator injection into the intact loop hot leg between the pressurizer and the steam generator inlet plenum. The injection will begin at 40 seconds after the initiation of the cold leg break to simulate the steam generator tube ruptures. The change in heat transfer potential of the steam generator will be simulated by discharging the steam generator secondary fluid during the simulated tube rupture period. The system initial conditions and emergency core coolant injection parameters are the same as Test S-04-6, the baseline test for Test Series 28.

The blowdown and refill response should be essentially the same as Test S-04-6 (0 to 58 seconds after rupture). The predictions for the reflood portion of the transient were performed with the FLOOD4 code. Experimental results from Test S-04-6 indicate the peak rod temperature during blowdown should occur at 8 seconds after rupture when a maximum of 1075 K was achieved. Initiation of the simulated tube ruptures at 40 seconds after rupture is not expected to delay refill in Test S-28-3

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C5

because of the small tube rupture flow rate specified for this test. The peak temperature reached during reflood was approximately 1158 K. When the FLOOD4 calculation ended at 600 seconds after reflood (658 seconds after rupture), the core hot spot had not yet quenched. This delay in the quenching of the core is thought to be due to steam binding in the intact loop and upper plenum which retarded the core flooding rate. The rod temperature at the core hot spot had declined to 1055 K when the calculation ended.

Very truly yours,

D J Olson

D. J. Olson, Manager
Semiscale Program

CPF:emw
Enclosure as stated

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