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LOCA Simulation in NRU Program

Data Report for Thermal-Hydraulic Experiment 3 (TH-3)

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Pacific Northwest Laboratory
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ABSTRACT

A series of in-reactor experiments are being conducted using light-water reactor fuel bundles as part of the Pacific Northwest Laboratory (PNL) Loss-of-Coolant Accident (LOCA) Simulation Program. The fifth experiment in the series of thermal-hydraulic and materials deformation experiments (TH-3) is described in this report. The experiments are being conducted in the National Research Universal (NRU) reactor, Chalk River, Ontario, Canada. The objective of TH-3 was to further refine the feedback control parameters developed in the TH-2 experiment and to re-establish the operability of the loop prior to the subsequent materials deformation and rupture test (MT-3). The TH-3 and MT-3 experiments were planned for the same reactor window and were run within two days of each other. The TH-3 test results insured the success of MT-3 and provided the opportunity to demonstrate the reactor control improvements and to evaluate a new desuperheater concept that would allow the test to run for extended times at high temperatures. The control system improvements and the addition of the new desuperheater resulted in fuel cladding temperatures above 1033K (1400°F) for 340 s. Experimental data and initial results are presented in this report.

SUMMARY

The Loss-of-Coolant Accident (LOCA) Simulation Program is being conducted by Pacific Northwest Laboratory (PNL) to evaluate the thermal-hydraulic and mechanical deformation behavior of full-length light-water reactor (LWR) fuel bundles under LOCA conditions. The test conditions are designed to simulate the heatup, reflood, and quench phases of a large-break LOCA, and the tests are performed in the National Research Universal (NRU) reactor using nuclear fission to simulate low-level decay power typical of these conditions.

Data and initial results from the fifth experiment in the program--thermal-hydraulic experiment 3 (TH-3)--are presented in this document. TH-3 included three tests and had the following major objectives:

- to insure the proper operation of the loop control system (LCS) for the subsequent materials deformation experiment (MT-3)
- to evaluate additional improvements to the data acquisition and control system (DACS) reflood feedback control rate logic developed for TH-2 that were designed to improve the response time and minimize overshoot for better peak fuel cladding temperature control in the range from 1033 to 1089K (1400 to 1500°F) for up to 200 s.
- to evaluate the desuperheater concept for use in MT-3.

The improved variable reflood rate control logic showed that it was possible to maintain temperature control above 1033K (1400°F) for up to 340 s. The desuperheater demonstrated the concept of cooling the exit steam with an upper head spray without measurably affecting the test temperature or outlet pressure.

Thermal-hydraulic data from the TH-3 experiment provided information on steady-state boil-off conditions and made it possible to estimate the heat transfer along the fuel rods during low-reflood steam-cooling conditions. The thermal-hydraulic analysis techniques have been shown to apply to the peak power region for the initial part of the transient but do not apply to steam-cooling conditions caused by prolonged low reflood flow rates.

This report presents graphical data demonstrating fuel cladding temperature control using both preset reflood flow and temperature feedback. Photographs of guard and test fuel used for experiments TH-2 and TH-3 (the same test assembly was used for both experiments) are shown in Appendix A. Preliminary graphical data on test assembly temperatures, cooling flow, and the neutronics environment are presented in the remaining appendices.

CONTENTS

FIGURES

TABLES

INTRODUCTION

The Loss-of-Coolant Accident (LOCA) Simulation Program is being conducted in the National Research Universal (NRU) reactor at Chalk River Nuclear Laboratories (CRNL),[a] Chalk River, Ontario, Canada, by Pacific Northwest Laboratory (PNL).[b] The program is sponsored by the U.S. Nuclear Regulatory Commission (NRC) to evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length, 3% enriched light-water reactor (LWR) fuel rod bundle during the heatup, reflood, and quench phases of a LOCA. Low-level nuclear fission heat was used to simulate the decay heat in the fuel and cladding that is typical of a LOCA.(1)

The program is composed of several thermal-hydraulic and cladding materials deformation experiments. The initial thermal-hydraulic experiment (TH-1), which was performed in October 1980, provided a data base for predicting the quenching characteristics of Zircaloy-clad fuel rods under various reflood conditions.(2) The first materials experiment (MT-1) was performed in April 1981 and used a pressurized cruciform of 11 rods with 1 water tube and 20 unpressurized guard rods. The test rods were pressurized to 3.1 MPa (450 psig). The delay time and the reflood rate were selected to duplicate the TH-1.10 experiment, which reached a peak fuel cladding[c] temperature of 1144K (1600°F). These conditions were achieved, and 6 of the 11 rods ruptured.(3)

In the the second materials experiment (MT-2), which was performed in July 1981, the MT-1 guard rod and shroud assembly was reconstituted underwater and reused with a new cruciform test bundle. One test objective was to perform a low-temperature--1089K (1500°F)--test using variable reflood rates. A malfunction of the test loop resulted in higher temperatures than desired, and 8 of the 11 rods ruptured under adiabatic heatup conditions.(4)

The fourth experiment (TH-2) used a new unpressurized thermal-hydraulic test bundle that was reconstituted in the MT-1/MT-2 guard rod and shroud assembly.(5) This experiment was designed to evaluate the reflood rates necessary to obtain a "flat top" or extended transient from 1033 to 1103K (1400 to 1525°F). The delay time and automatic control system used to control the variable reflood rate in this experiment demonstrated the capability of holding temperatures above 1033K (1400°F) for periods of up to 280 s. The test conditions approached a steady-state boil-off condition in which the quench front is in equilibrium with the reflood rate.

This report presents data and initial results from the third thermal-hydraulic experiment (TH-3), which was performed in November 1981. The experiment was jointly funded by the NRC and the United Kingdom Atomic Energy Authority (UKAEA) to evaluate the thermal-hydraulic test conditions for flat top transient LOCA conditions. The TH-3 experiment used the same test bundle as

(a) Operated by Atomic Energy of Canada Ltd. (AECL).
(b) Operated for the U.S. Department of Energy (DOE) by Battelle Memorial Institute.
(c) Fuel cladding subsequently referred to as cladding.

TH-2 with the addition of several new thermocouples (TCs) and a spray desuperheater. Several modifications were also made to the control logic in the data acquisition and control system (DACS) to improve the reflood rate control and extend the length of the flat top transient.

TH-3 was performed just prior to the third materials experiment (MT-3) to verify the loop control system (LCS) instrumentation and operation. In addition, TH-3 verified the repeatability of the test loop to reproduce the test conditions from the TH-2.14 test. This test was the basic model proposed for the upcoming MT-3 test series and the question of repeatability was important to the success of MT-3. TH-3 also provided the opportunity to adjust the DACS feedback control system first and second derivative weighting functions to improve the temperature response.

The primary objective of TH-3 was to develop more reliable cladding temperature control of a simulated LOCA to insure the success of MT-3. For TH-3, peak cladding temperatures were to range from 1089 to 1172K (1500 to 1650°F) for at least 150 s; reflood water coolant was used to control the temperature.

The results from TH-3 provide thermal-hydraulic response data for full-length nuclear-heated cladding thermal-hydraulic response data in the high alpha temperature range for variable reflood conditions. These conditions extend the existing data base on thermal-hydraulic response to LOCA operating conditions not previously investigated by FLECHT or other out-of-reactor test programs. These tests provided valuable information on the control of quench fronts and two-phase cooling that will be used for subsequent thermal-hydraulic and materials experiments. They also provided information on the quench characteristics of nondeformed rods as compared with deformed rods.

Data from TH-3 will be used in conjunction with previous test results to assess various calculational models for reactor safety analyses and conclusions derived from the large series of electrically heated tests and smaller scale in-reactor tests being conducted elsewhere. The experimental results of the program address 17 specific items outlined in the Code of Federal Regulations 10 CFR 50.46 and 10 CFR 50, Appendix K. These results will be used to provide additional data for model calibration and to help define the primary heat transfer mechanisms for new analytical models. The major contribution of these tests to LWR technology is to reduce the uncertainty on licensing criteria and offer the potential for raising the operating limits on some commercial LWRs.

EXPERIMENT DESCRIPTION

This section describes the components of the test train assembly that were used for TH-3 and details the instrumentation that was provided. The experiment operation, which consisted of one adiabatic and two transient tests, is also described.

TEST TRAIN ASSEMBLY

A schematic of the test train used for the LOCA test program is depicted in Figure 1. The test train (including the head closure, hanger tube, and test assembly) was ~9 m (~30 ft) long. The closure region provided the primary pressure boundary and included penetrations for 183 instrumentation leads. The hanger tube was used to suspend the test bundle and shroud from the head closure plug, and instrument leads were attached to the hanger to protect them during transport and testing. The shroud, which supported the fuel bundle and served as a protective liner during the experiment and transfer operations, also provided proper coolant flow distribution during various stages of the experiment. The stainless steel (SS) shroud consisted of two halves that were clamped together at 17.78-cm (7-in.) intervals and attached at the end fittings. The split shroud design makes it possible to disassemble, examine, and reassemble the test train underwater after its irradiation. The highly instrumented shroud and test assembly was ~4.32 m (14.18 ft) long.

The fuel assembly consisted of a 6 x 6 segment of a 17 x 17 PWR fuel assembly with the four corner rods removed (see Figure 1) and provided a basic fuel array of 32 rods. The 20 guard rods in the outer row reduced the net heat transfer from the inner test rods during the test. There were 11 test fuel rods (see Table 1) and 1 instrument tube, arranged in a cruciform pattern. The 31 unpressurized fuel rods were filled with helium.

The test train instrumentation included: 24 self-powered neutron detectors (SPNDs), 69 fuel rod TCs, 16 steam probe TCs, 41 shroud TCs, 6 carrier TCs, 4 hanger TCs, and 4 closure head TCs. The instrumentation was located at 22 elevations or levels along the test train assembly; each of these levels is defined in Figure 2[(a)] and detailed in Figures 3 through 6. Four TCs were located at an additional level to measure the closure temperature. Additional detail and nomenclature can be found on the blueprints referenced in Figure 3.

Turbine flowmeters and TCs provided the main source of thermal-hydraulic data. Local coolant temperatures were measured with steam probe TCs that protruded into the coolant channel and with TCs attached to the shroud. TCs were also located at the fuel centerline and attached to the inside of the cladding surface to measure azimuthal temperature variations. These cladding TCs were spot welded to the interior cladding surface and monitored the cladding temperature without interference from fuel pellet chips or unintentional TC relocation.

(a) Levels in Figure 2 range from 1 to 21; the twenty-second elevation is Level 18a (located above Level 18).

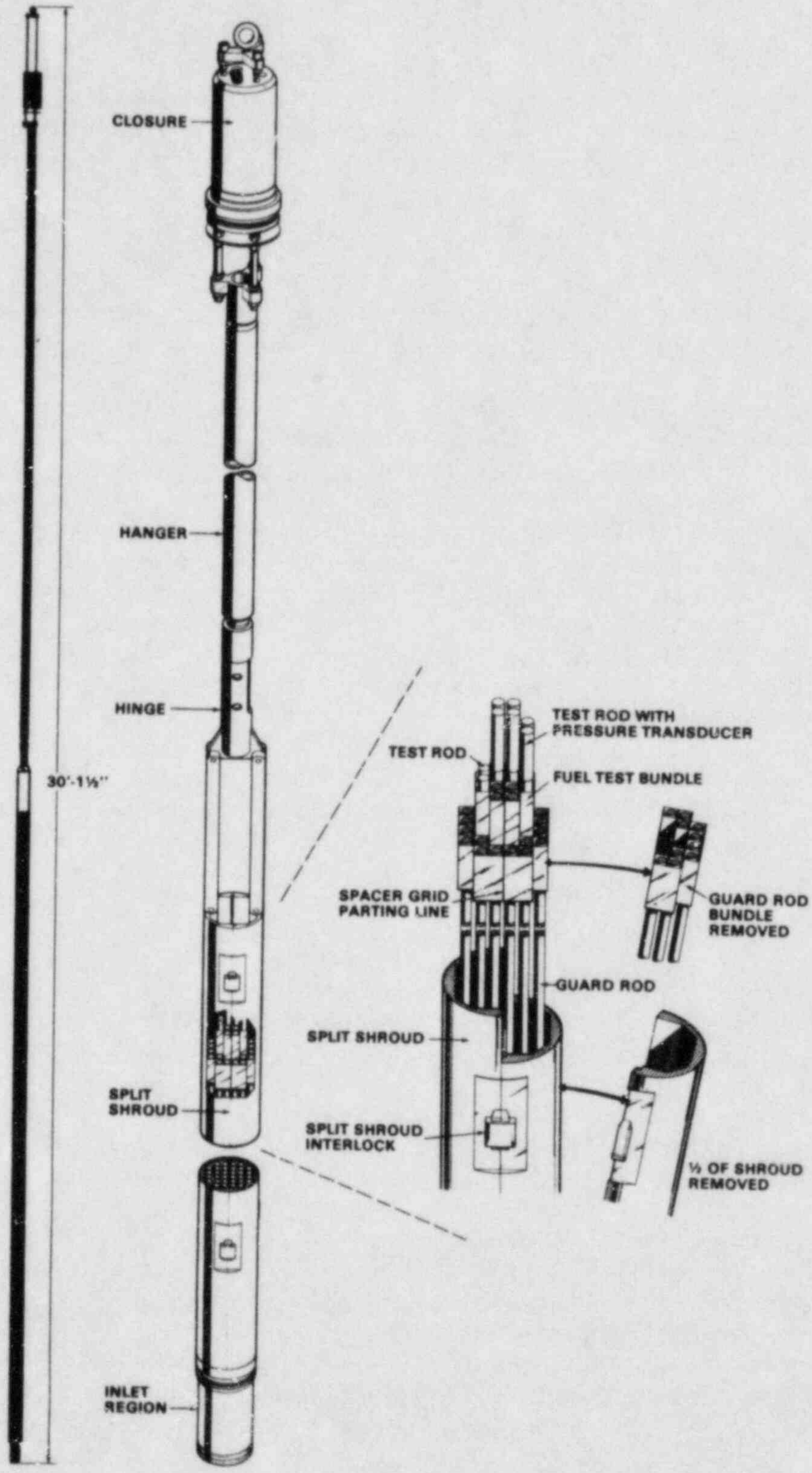


FIGURE 1. Schematic of NRU Loss-of-Coolant Accident Test Train

TABLE 1. Test Fuel Rod Design Variables

Cladding material	Zircaloy-4
Cladding outside diameter (OD)	0.963 cm (0.379 in.)
Cladding inside diameter (ID)	0.841 cm (0.331 in.)
Pitch (rod to rod)	1.275 cm (0.502 in.)
Fuel pellet OD	0.826 cm (0.325 in.)
Fuel pellet length	0.953 cm (0.375 in.)
Active fuel length	3.66 m (12 ft)
Total shroud length	4.32 cm (14.18 ft)
Helium pressurization	unpressurized
Fuel enrichment	2.93% ^{235}U

SPNDs provided neutron flux measurements within the fuel bundle. They could also detect coolant density variations (through flux changes) associated with the quench front that passed each SPND during the reflood phase of the transient. SPND data with regard to coolant density are being evaluated.

The instrument signals were monitored on a real-time basis with the DACS. The recorded data characterized the coolant flow rates, temperature, neutron flux, and operating history.

EXPERIMENT OPERATION

The TH-3 experiment included three successive tests, each having a pretransient and transient phase. All tests used position L-24 in the NRU reactor (see Figure 7). The assembly was oriented in the reactor with side F facing north (fuel rods 2F, 3F, 4F, and 5F faced north).

A preconditioning operation had been conducted prior to the TH-2 experiment and was not repeated for the TH-3 experiment. The average test assembly fuel rod power during preconditioning was ~18.7 kW/m (5.7 kW/ft) with the U-2 loop providing water cooling. Three short runs to full power permitted the fuel to crack and relocate within the cladding. System loop pressure was held at 8.62 MPa (1250 psia).

The pretransient for the TH-3 tests was conducted with steam cooling provided by the U-1 loop at a mass flow rate of ~0.379 kg/s (~3000 lbm/h) and a reactor power of ~7.4 MW. NRU reactor power was increased or decreased as required to maintain the same steam temperature increase across the test assembly. Even though the peak cladding temperature varied from test to test, the total assembly power remained constant because the cooling steam flow rate was maintained as consistently as possible at ~0.379 kg/s (~3000 lbm/h). Test TH-3.01 was an adiabatic test that used no reflood water and was run to determine fuel rod local power. Test assembly power for the adiabatic test was 141.5 kW.

The transient phase for each TH-3 test began when the steam coolant flow was reduced from 0.379 kg/s (3000 lbm/h) to 0 (as indicated by a sharp drop in temperature of the Level 1 TCs); reactor power was maintained at ~7.4 MW.

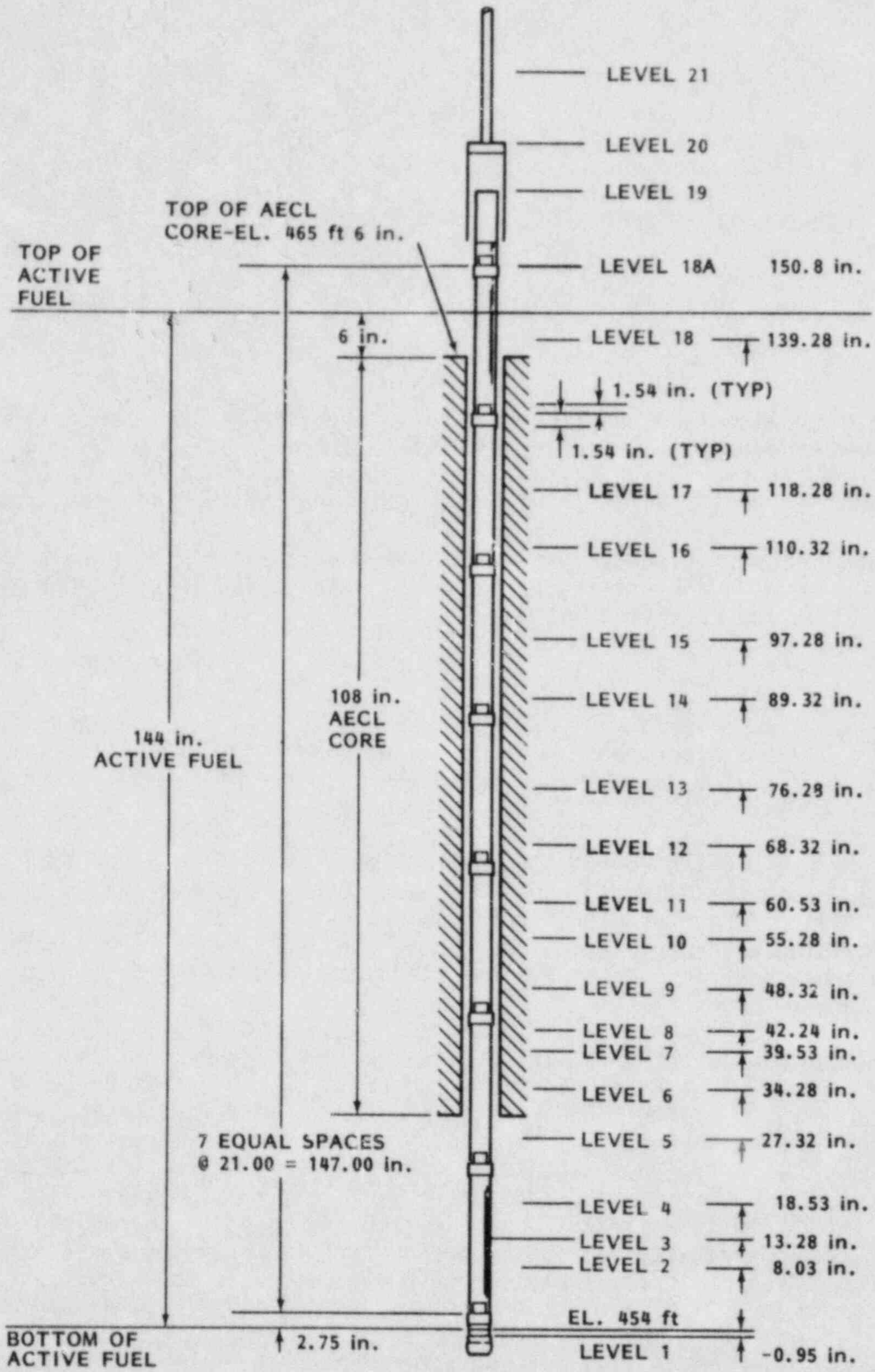


FIGURE 2. Instrumentation Levels in the TH-3 Test Assembly

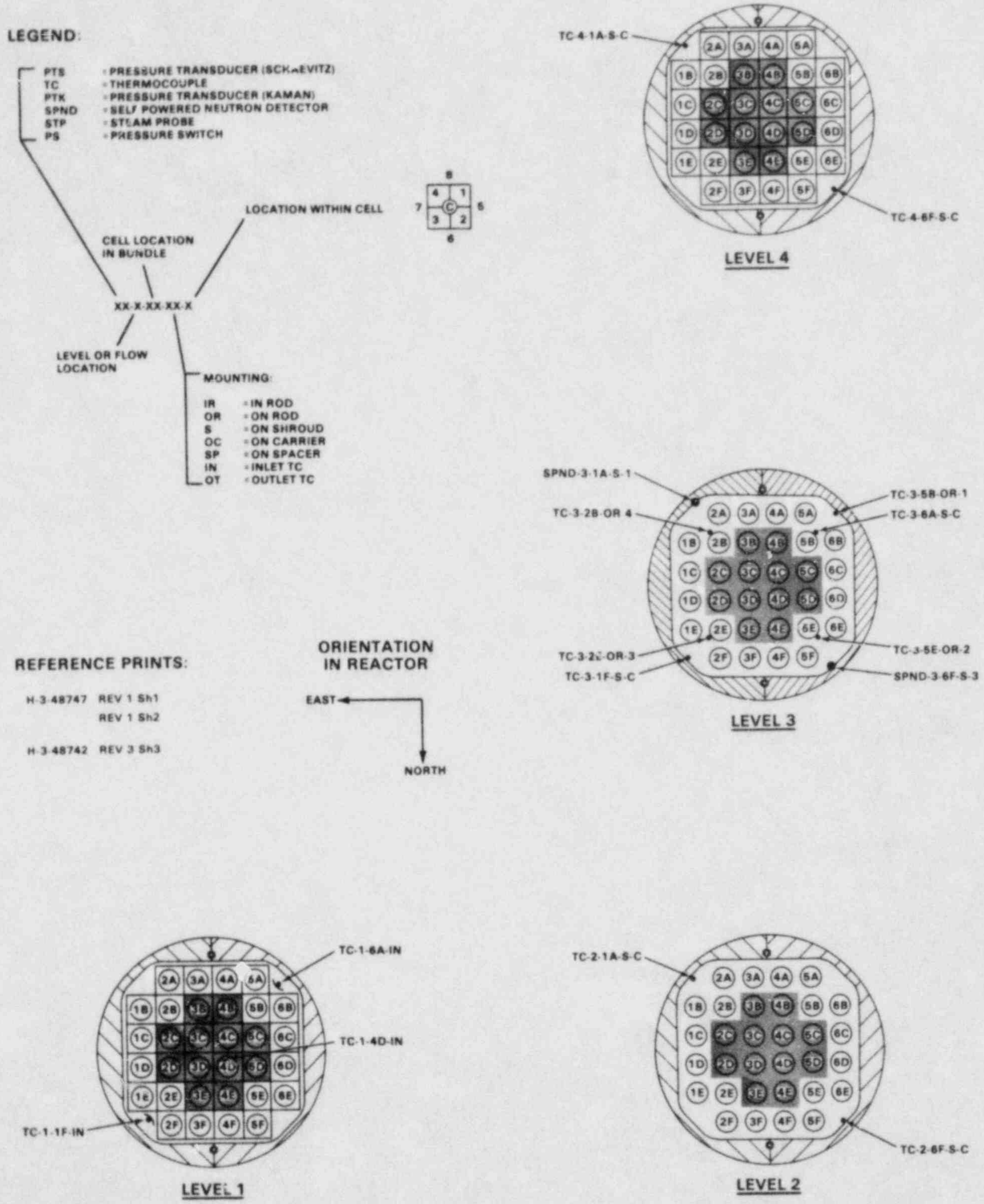


FIGURE 3. Instrumentation at Levels 1 Through 4 in the TH-3 Test Assembly

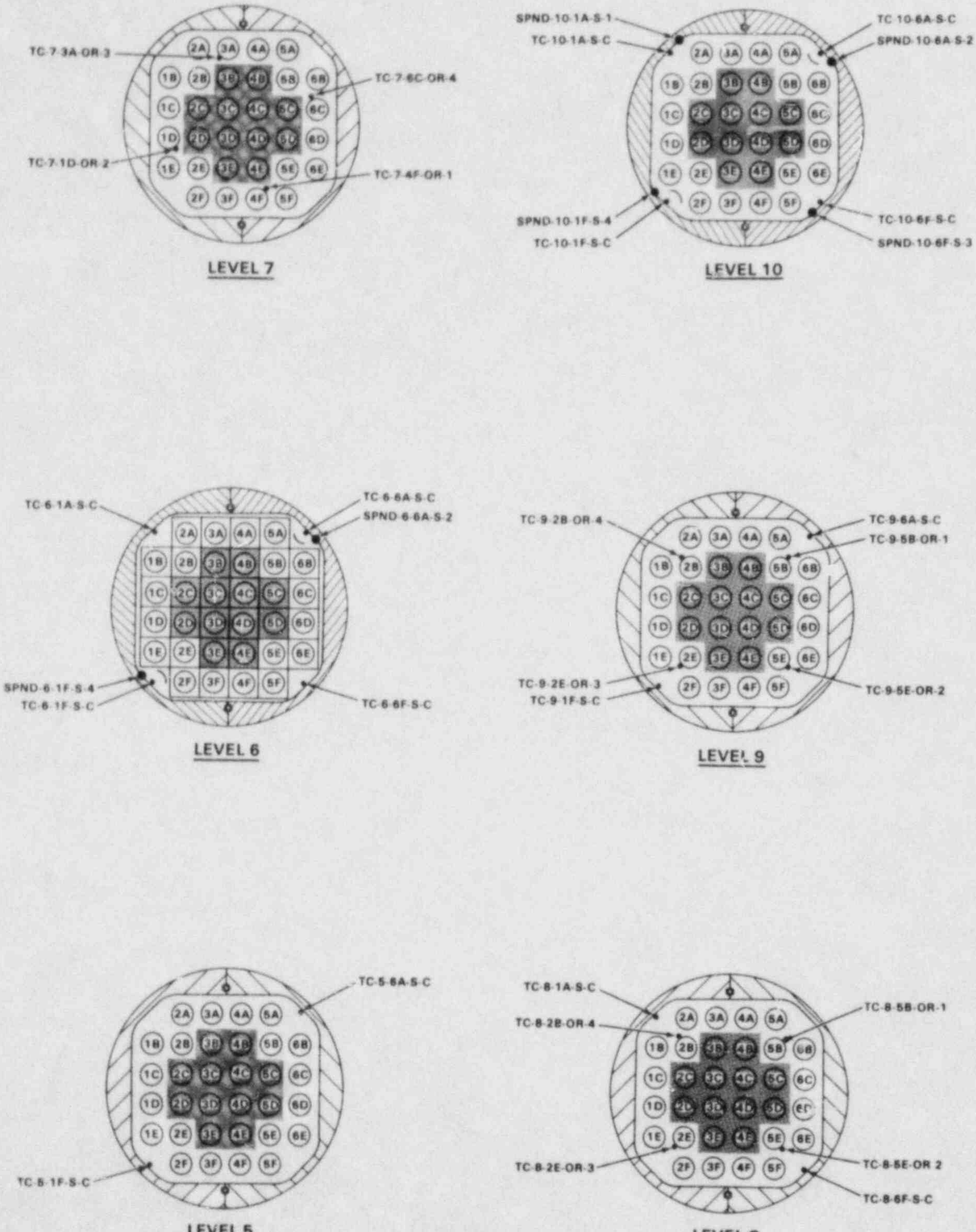


FIGURE 4. Instrumentation at Levels 5 Through 10 in the TH-3 Test Assembly

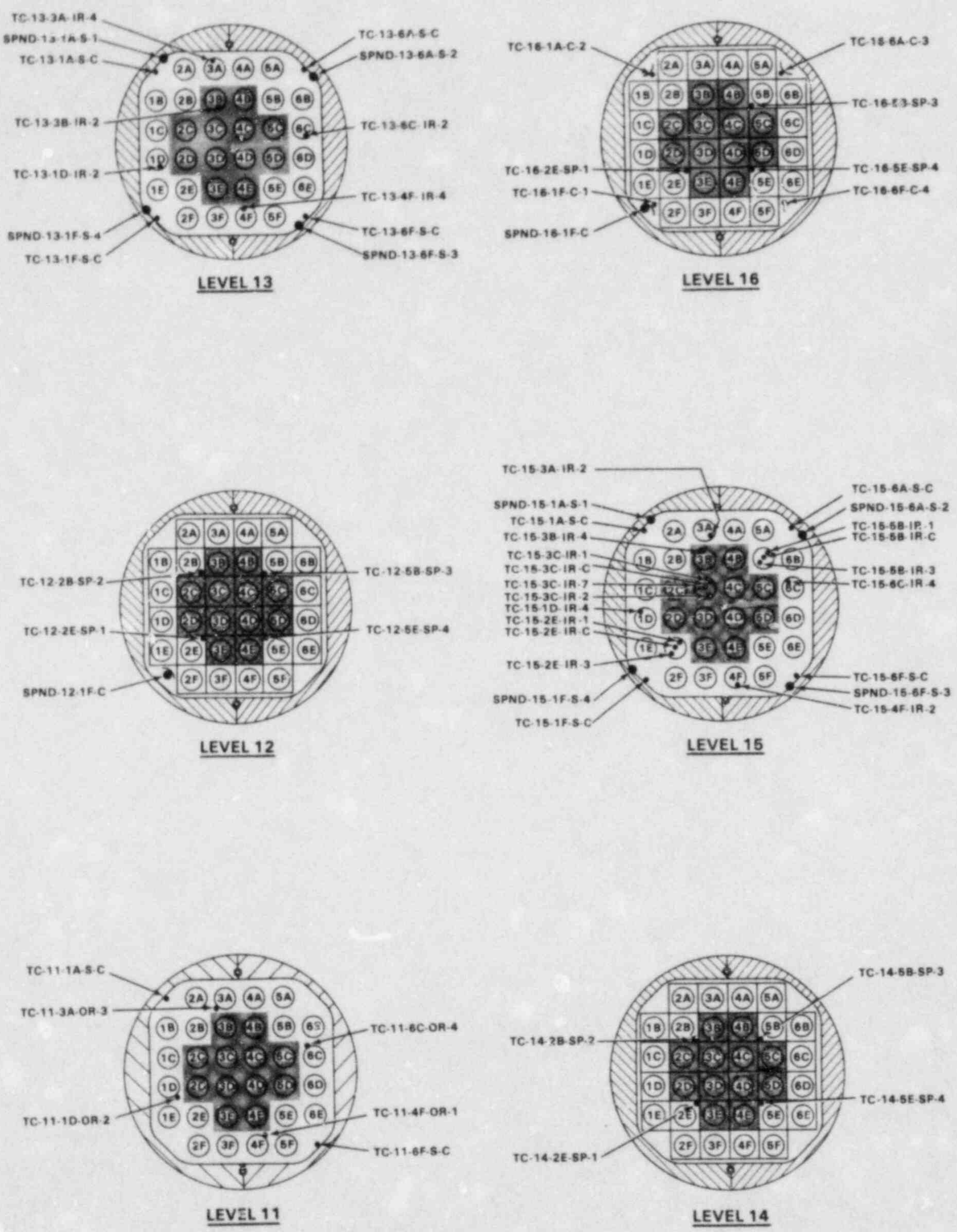


FIGURE 5. Instrumentation at Level 11 Through 16 in the TH-3 Test Assembly

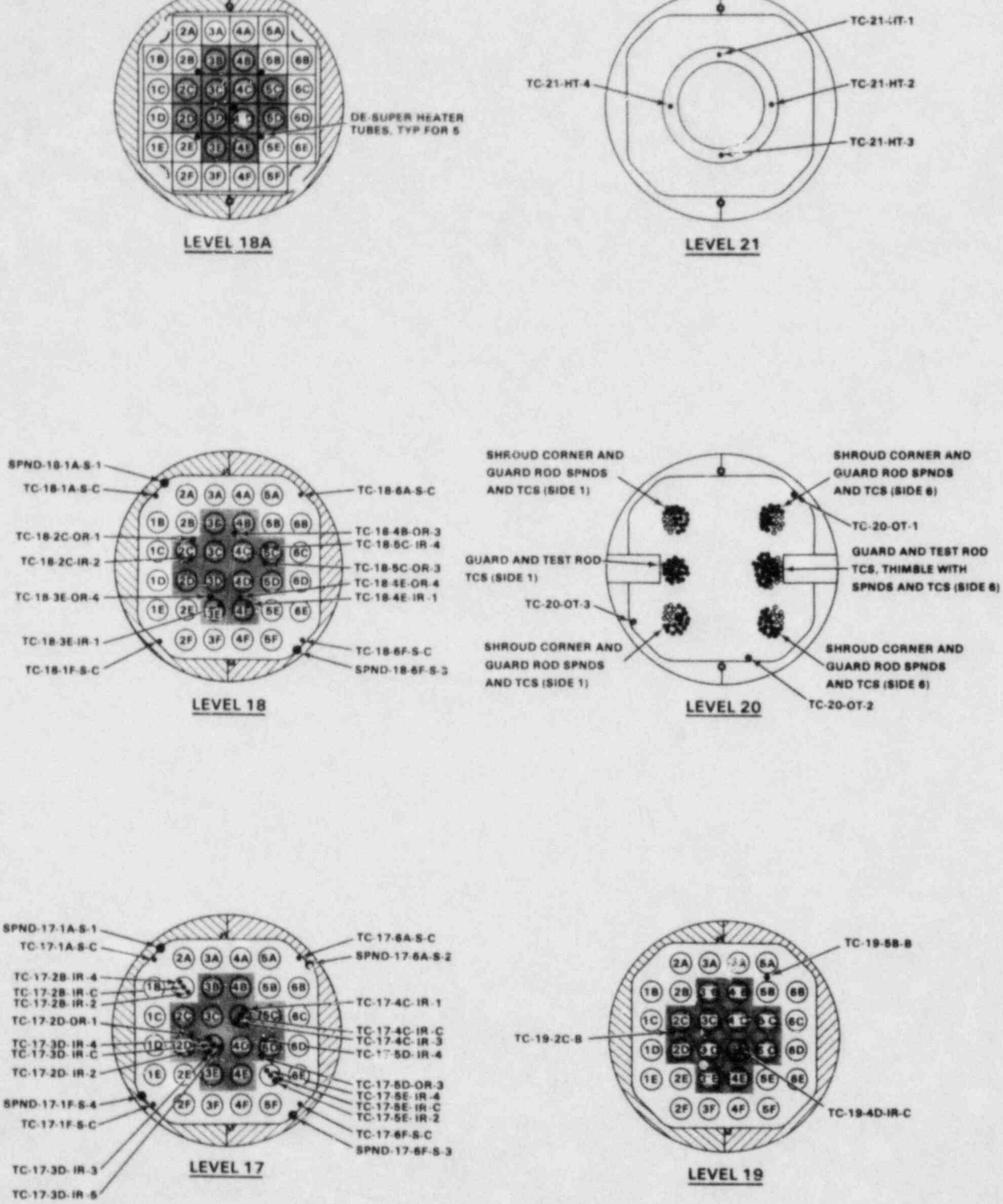
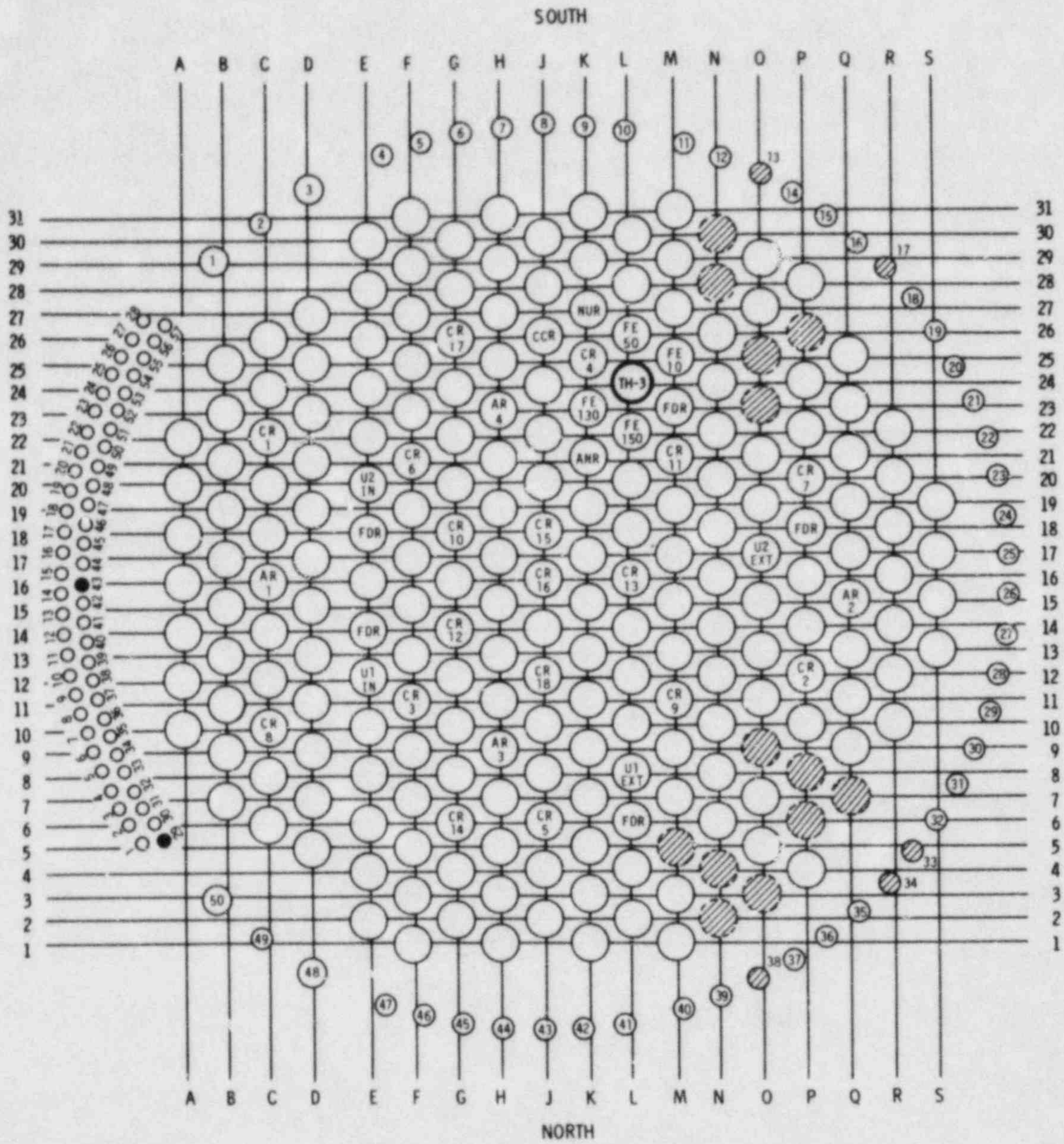


FIGURE 6. Instrumentation at Levels 17 Through 21 in the TH-3 Test Assembly



CR CONTROL ROD

AR ADJUSTER ROD

FDR FLUX DETECTOR ROD

U-1 IN
U-2 IN DOWN FLOW CHANNEL - DUMMY FUEL

U-1 EXT
U-2 EXT UP FLOW CHANNEL - DUMMY FUEL

STANDARD LATTICE POSITION

BLOCKED CHANNEL - NOT FUELED

CCR COBALT CARRIER ROD

NUR NATURAL URANIUM ROD

ANR ALUMINUM NITRIDE ROD

FE 50 FUEL ELEMENT - 50 MWd BURNUP

FE 10 FUEL ELEMENT - 10 MWd BURNUP

FE 130 FUEL ELEMENT - 130 MWd BURNUP

FE 150 FUEL ELEMENT - 150 MWd BURNUP

FIGURE 7. NRU Reactor Core Configuration

Tests TH-3.02 and TH-3.03 used the preprogrammed LCS for 90 s and the DACS with temperature feedback after 90 s. The DACS monitored selected average temperatures (hot spot sensors) at Levels 15 and 17 and used the hottest average to control the reflood rate. The DACS effectively used proportional, integral differential control and translated the error into preselected incremental changes in reflood rate.

EXPERIMENT CONDITIONS AND RESULTS

The test conditions measured during the experiment are described in Table 2. The table also includes reactor trip criteria, the type of test (adiabatic or transient), and the type of reflood control (preprogrammed LCS reflood flow or DACS with temperature feedback). LCS and DACS control parameters are detailed in Tables 3 and 4, respectively.

The adiabatic heatup test (TH-3.01) was performed to determine local fuel power. Test rod and shroud heatup rates for the adiabatic test are shown in Figures 8 through 10.

Tests TH-3.02 and TH-3.03 used LCS preprogrammed reflood flow rates for the first 90 s and DACS-controlled reflood rates with cladding temperature feedback for the remainder of the test. These two tests were used to refine the DACS parameters for the desired control and repeatability. Figures 11 and 12 demonstrate the results where selected hot spot sensors were used. Test repeatability using the DACS is shown in Figures 13 through 15, where test TH-3.03 is compared with test TH-2.14.[5] Average test rod inside cladding temperatures at Levels 13, 15, and 17 are shown.

The effects of spray cooling on the fuel and shroud at Level 18 are shown in Figure 16, and the effects of spray cooling on coolant temperature at Level 19 and the outlet pressure are shown in Figure 17. Spray cooling is shown turned on when its line is low.

Appendices A through F summarize the preliminary data resulting from the TH-3 experiment. The data are arranged as follows:

Appendix A - Guard and Test Fuel Photographs
Appendix B - Pretransient Test Assembly Temperatures
Appendix C - Transient Fuel and Cladding Temperatures
Appendix D - Test Coolant and Shroud Temperatures
Appendix E - Neutron Flux
Appendix F - Reflood Flow Measurements.

TABLE 2. Measured Experimental Operating Conditions

Parameter	Preconditioning[a]	Reflood Calibration	Test TH-3.01	Test TH-3.02	Test TH-3.03
Reactor power, MW	127	0	7.4	7.4	7.4
Test assembly power, kW		0	141.5	134.6	133.5
Coolant	U-2 water	U-1 steam/ reflooding	U-1 steam/ reflooding	U-1 steam/reflooding	
Coolant flow, kg/s (lbm/h)	0 to 16.30 (0 to 129,400)	0.0254, 0.508 (1.0, 2.0)	0.379 (3010)	0.379 (3004)	0.379 (3009)
Reflood delay, s	NA[b]	NA	NA	9	3
Reflood rates, m/s (in./s)	NA	NA	NA	0.0823 (3.24) for 8 s; 0.0549 (2.16) for 40 s; 0.0366 (1.44) for 16 s; 0.0244 (0.96) for 20 s; 0.0188 (0.74) for 28 s; 0.0127 (0.50) for 162 s	0.0828 (3.26) for 8 s; 0.0574 (2.26) for 40 s; 0.0371 (1.46) for 16 s; 0.0224 (0.88) for 28 s; 0.0124 (0.49) for 18 s; 0.0191 (0.75) for 40 s; 0.0097 (0.38) for 96 s; 0.0147 (0.58) for 28 s; 0.0102 (0.40) for 130 s
Transient initiation time[c]	NA	NA	11/10/81 22:13:42	11/10/81 23:33:09	11/11/81 0:48:26
Pretransient cladding temperatures, K (°F)	NA	NA	723 (842)	723 (842)	717 (830)
Peak cladding temperature (PCT), K (°F)	700 (800)		1008 (1354.4)	1318 (1912)	1283 (1850)
Reactor conditional trip criteria (PCT), K (°F)	NA	NA	978 (1300)	1172 (1650)	1200 (1700)
PCT turnaround time,[d] s	NA	NA	35	193	257
Bundle quench time,[d] s	NA	NA	NA	277	407
Type of test	NA	Reflood	Adiabatic	Transient	Transient
Type of reflood control	NA	LCS[e]	NA	DACS[f] after 90 s	DACS after 90 s

(a) Preconditioning was performed prior to the TH-2 experiment.
(b) Not applicable.
(c) Transient initiated by termination of steam flow.
(d) Time after initiation of transient.
(e) LCS = loop control system.
(f) DACS = data acquisition and control system.

TABLE 3. LCS Preset Control Reflood Rates for TH-3.03

LCS Control Rate, in./s (m/s)	Total Transient Test Time, s
Delay	0 to 5
2.2 (0.0559)	5 to 50
1.5 (0.0381)	50 to 65
1.0 (0.0254)	65 to 85
0.8 (0.0203)	85 to 90
(a)	

(a) Control was transferred to DACS.

TABLE 4. DACS Discrete Reflood Feedback Control Parameters for TH-3.03

DACS Discrete Reflood Control Rate, in./s (m/s)	DACS Parameters	
	Value[a]	Weight
1.3 (0.0330)	Ax"	A = 6
1.1 (0.0279)	Bx'	B = 10
0.9 (0.0229)	Cx	C = 1
0.8 (0.0203)	DΔT	D = 4
0.7 (0.0178)		
0.6 (0.0152)		
0.5 (0.0127)		
0.4 (0.0102)		

(a) where $x' = \frac{\Delta\ \text{(temperature)}}{\Delta\ \text{(time)}}$

ΔT = averaging time in seconds.

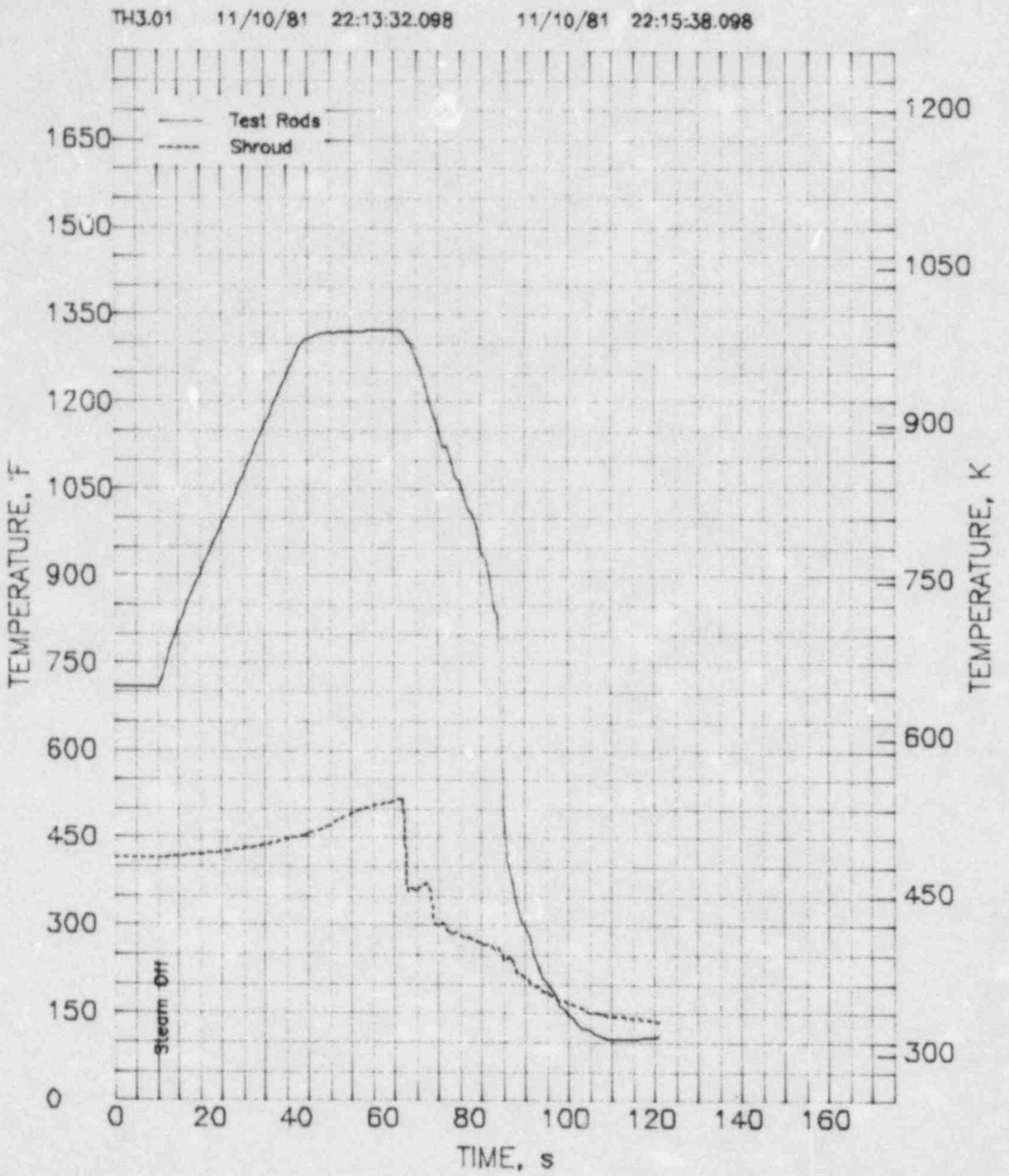


FIGURE 8. Average Test Rod and Shroud Temperatures at Level 13 for Adiabatic Test TH-3.01

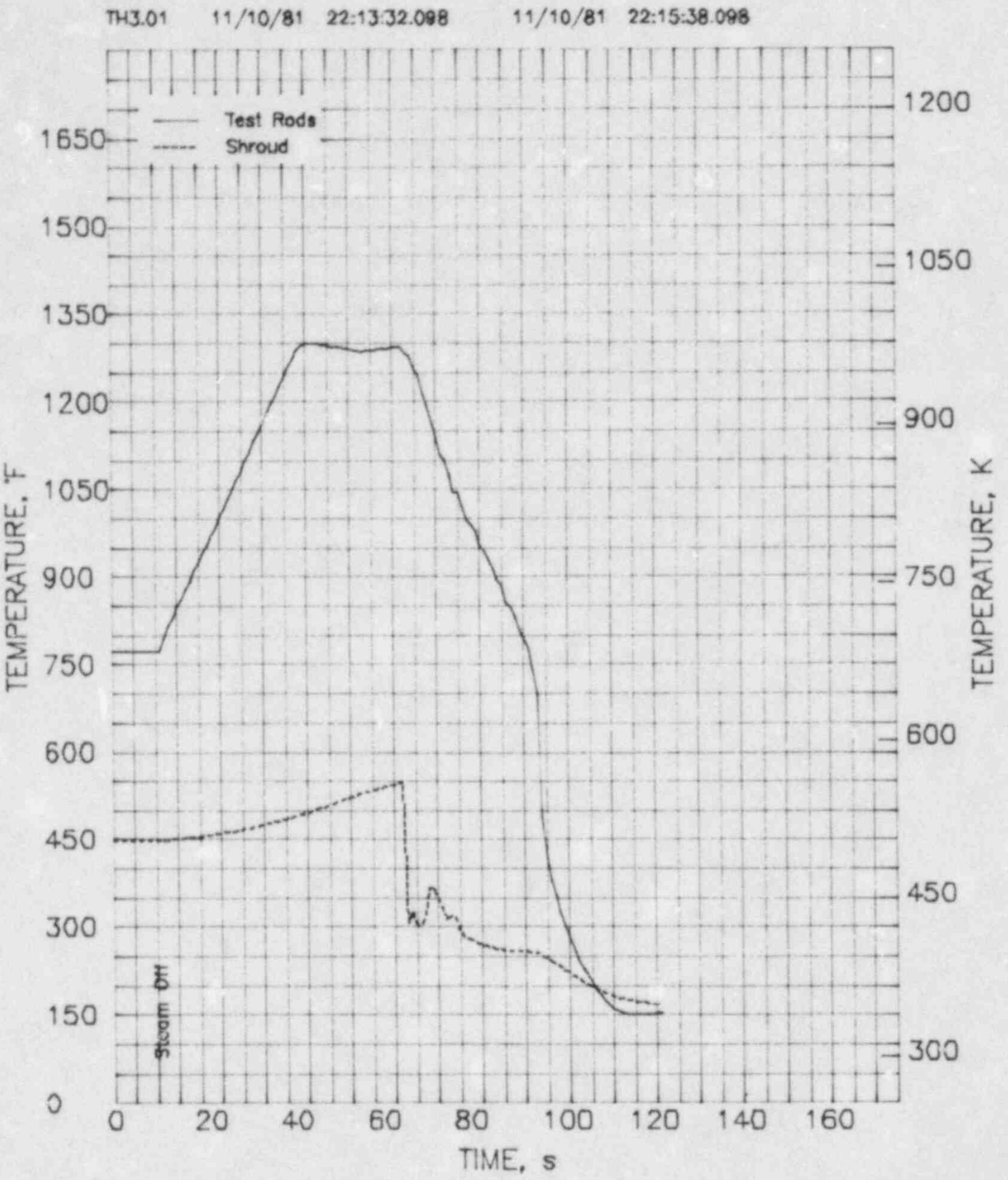


FIGURE 9. Average Test Rod and Shroud Temperatures at Level 15 for Adiabatic Test TH-3.01

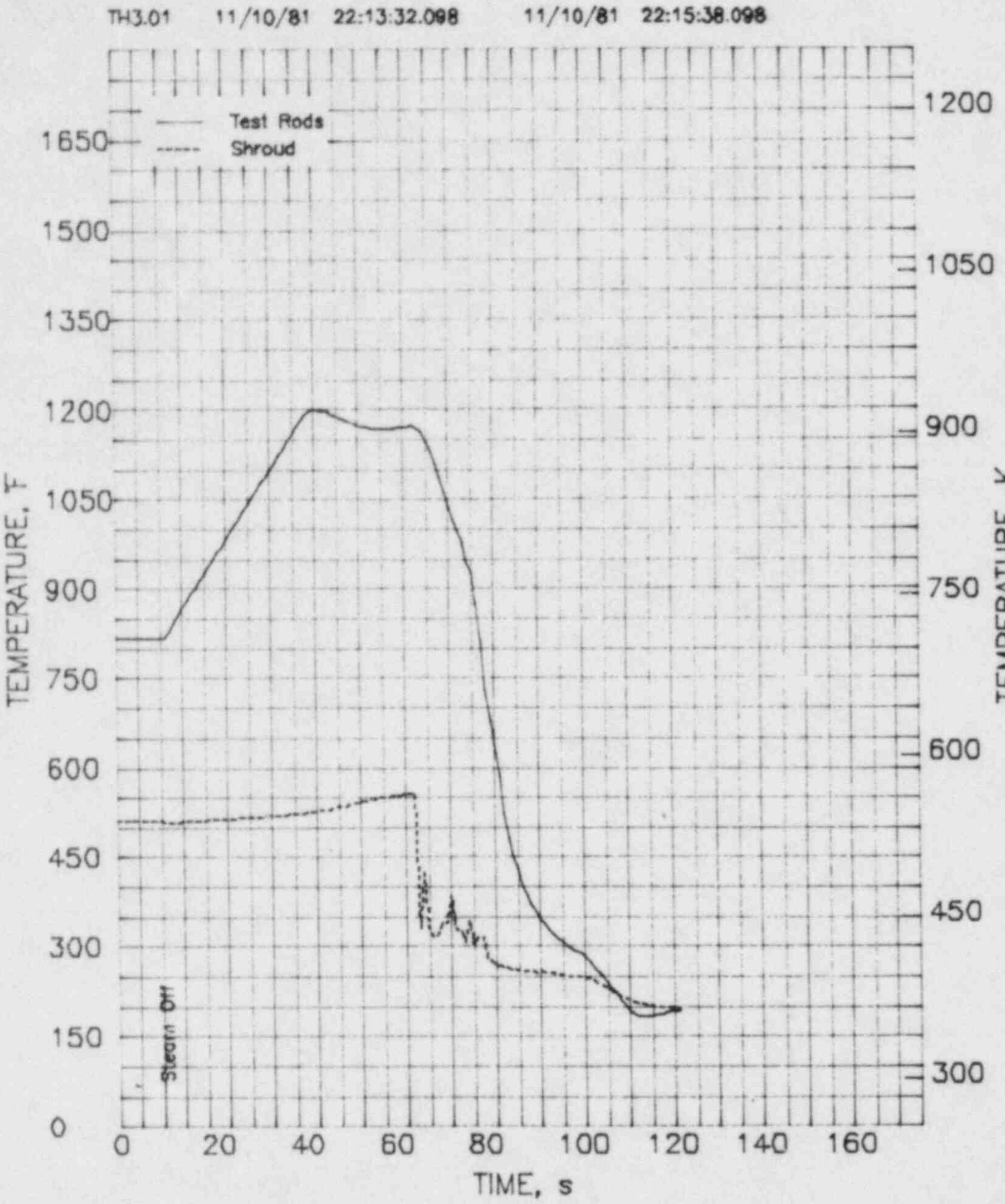


FIGURE 10. Average Test Rod and Shroud Temperatures at Level 17 for Adiabatic Test TH-3.01

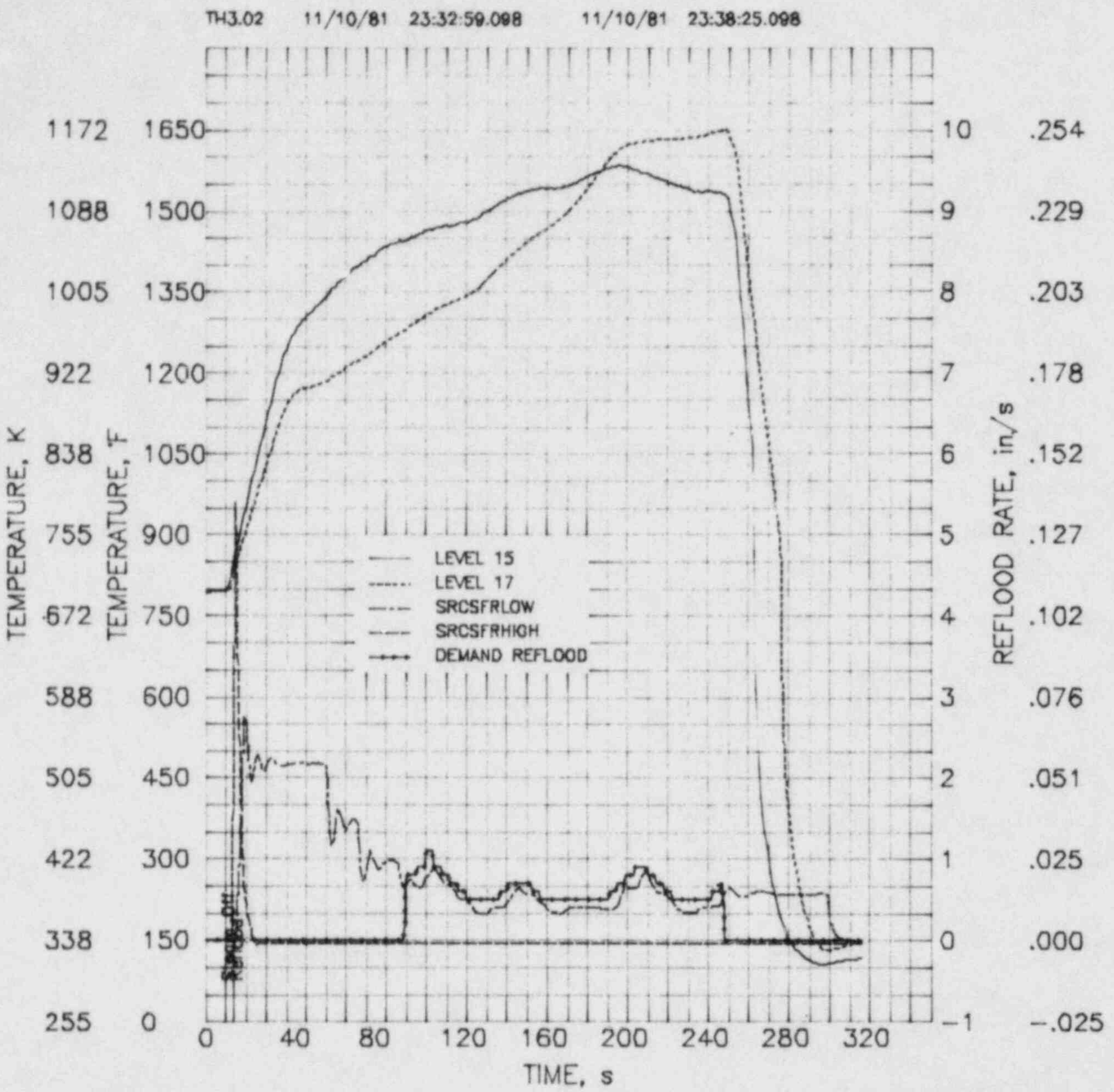


FIGURE 11. Hot Spot Sensor Averages and Demand and Actual Reflood Flow Rates at Levels 15 and 17 for TH-3.02 (DACS--90 s after steam shut off)

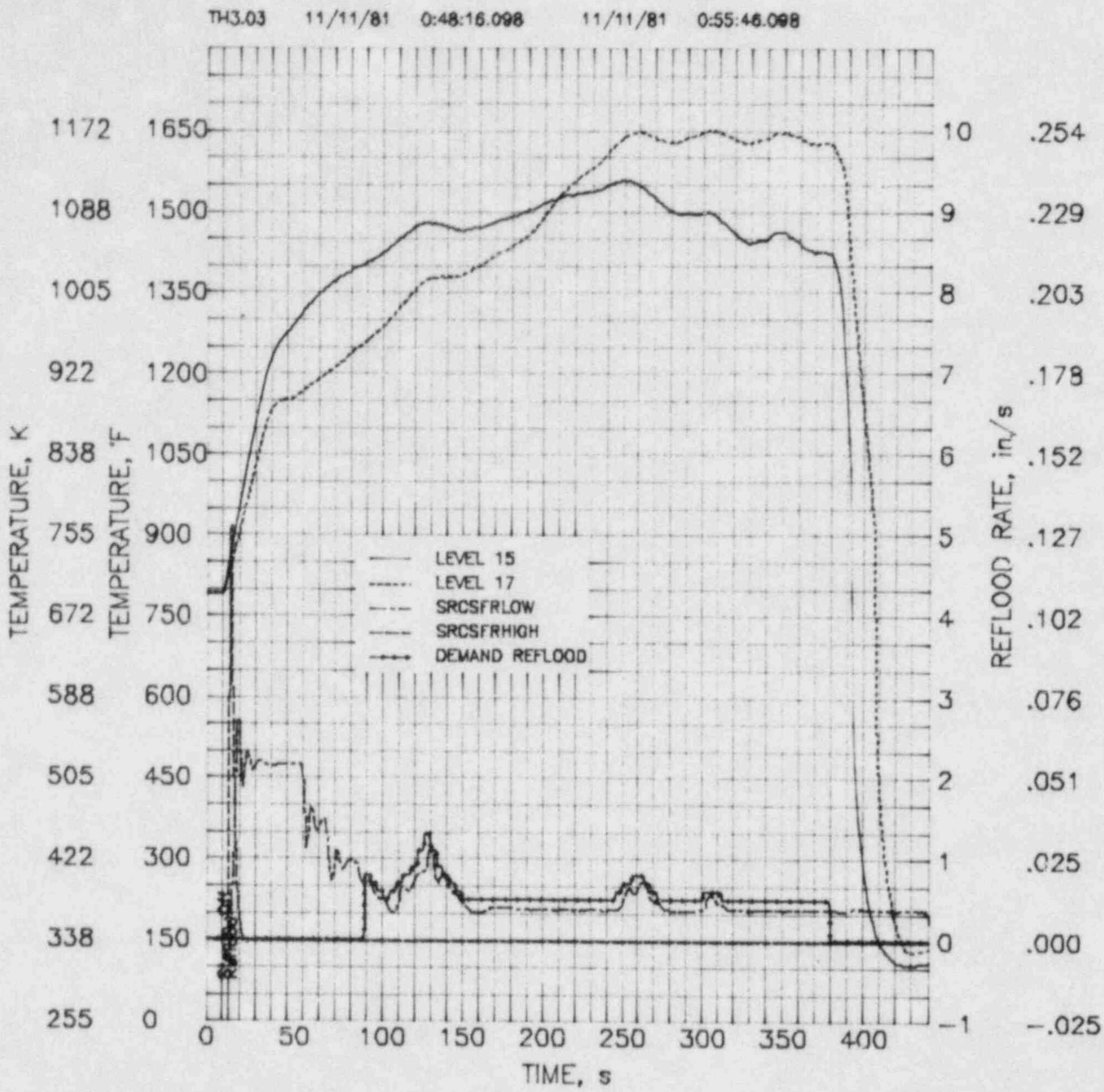


FIGURE 12. Hot Spot Sensor Averages and Demand and Actual Reflood Flow Rates at Levels 15 and 17 for TH-3.03 (DACS--90s after steam shut off)

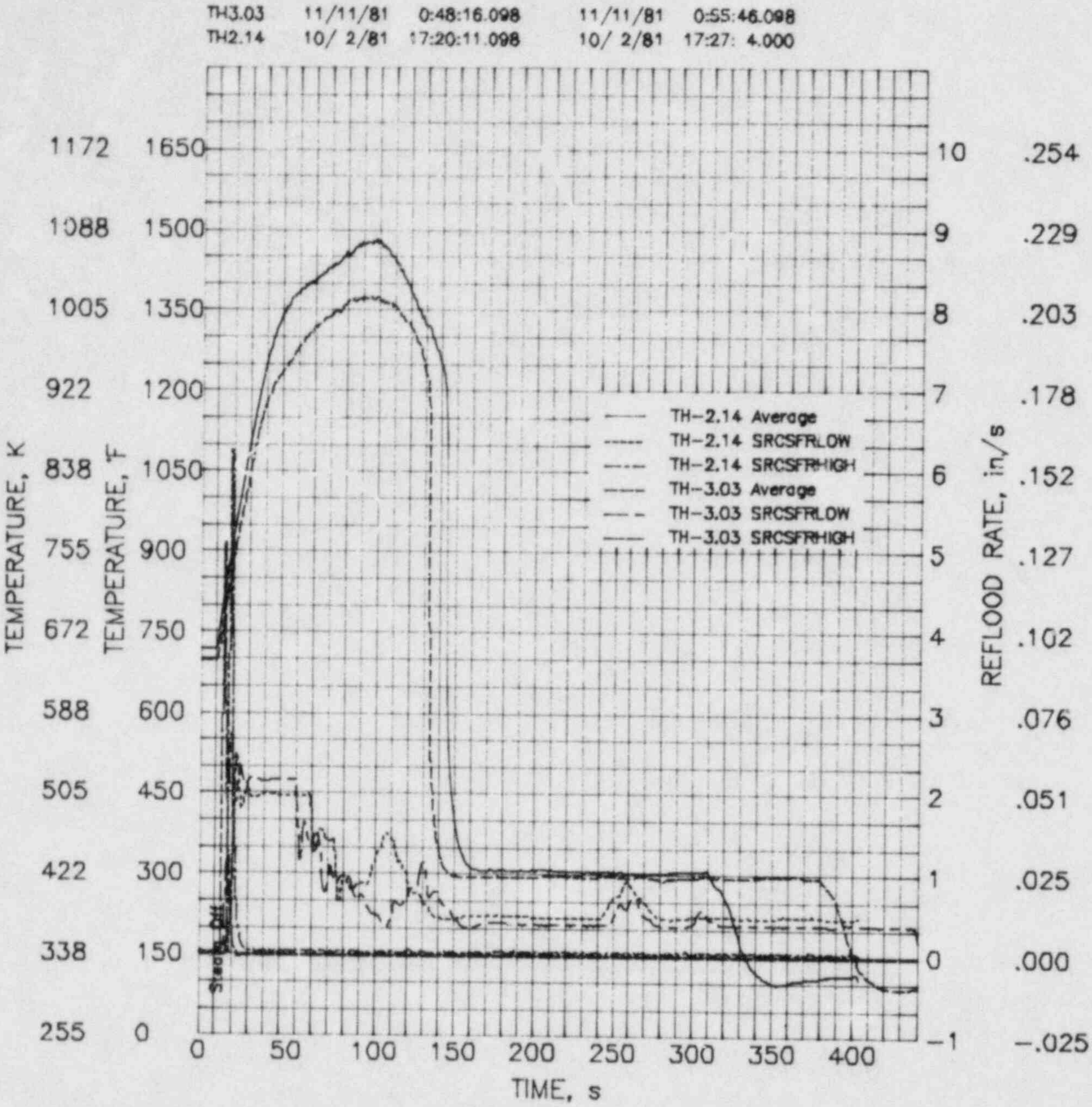


FIGURE 13. Comparison of Tests TH-3.03 and TH-2.14 Showing Level 13 Average Test Rod Temperatures and Reflood Flow Rates

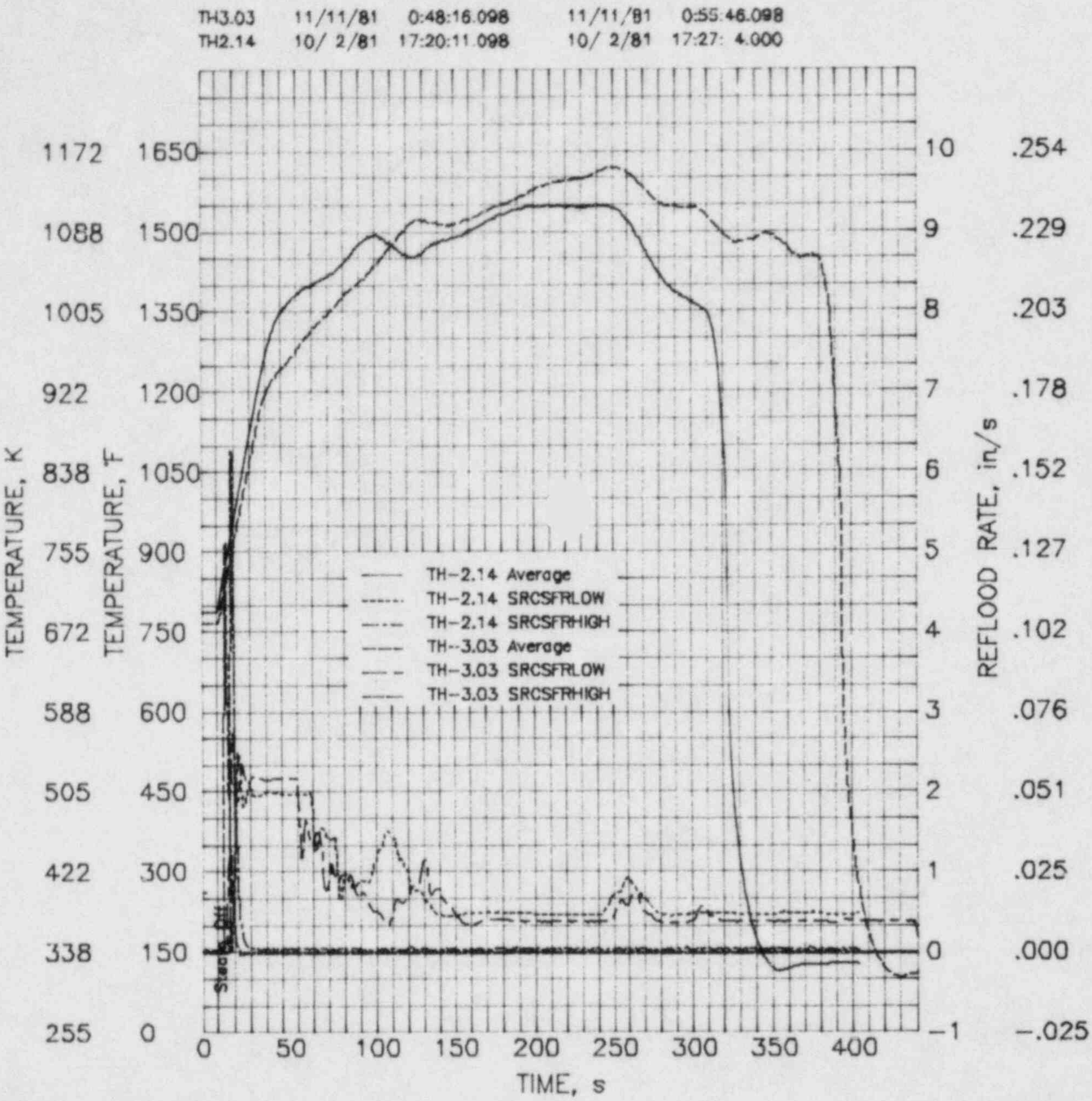


FIGURE 14. Comparison of Tests TH-3.03 and TH-2.14 Showing Level 15 Average Test Rod Temperatures and Reflood Flow Rates

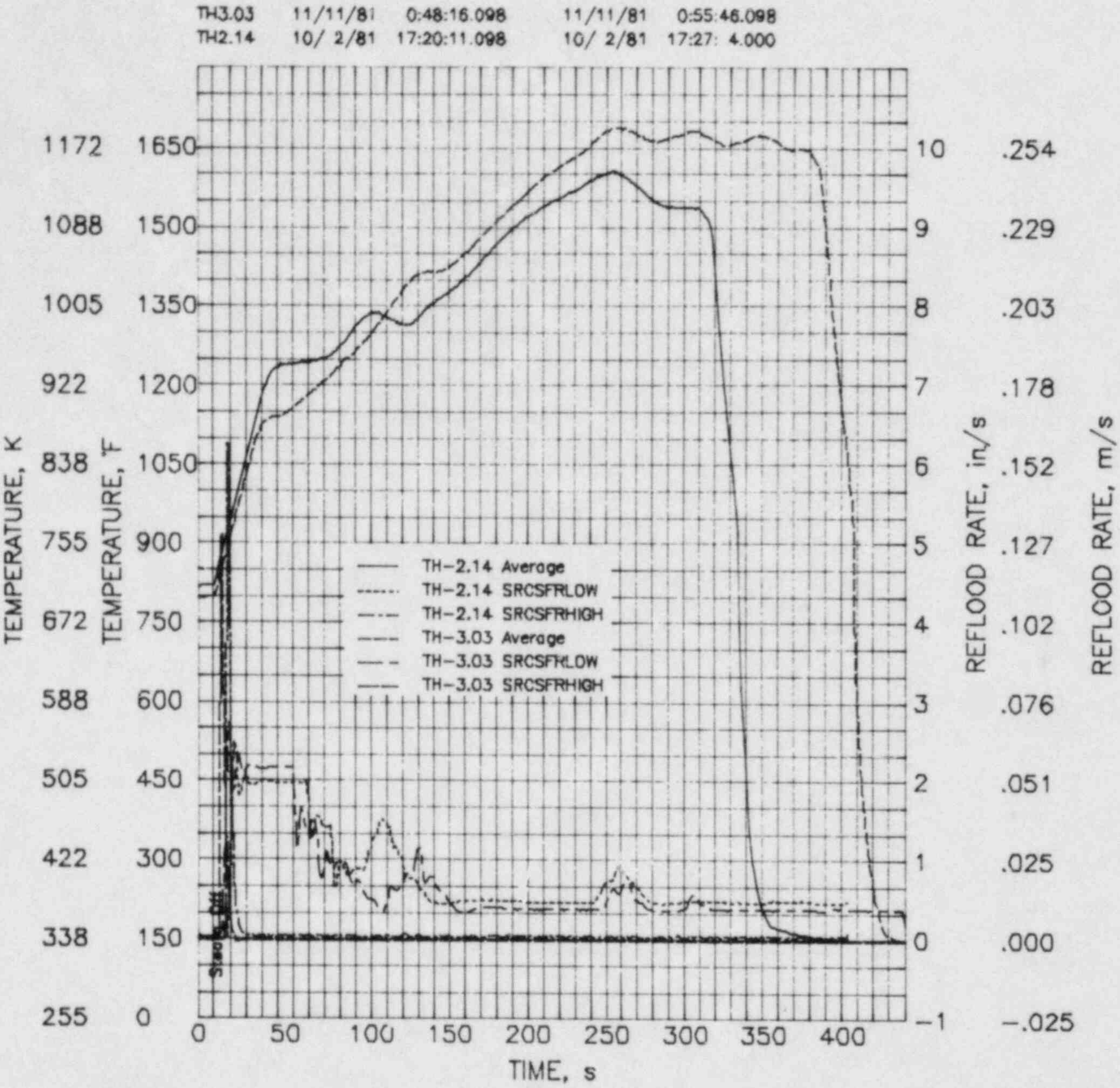


FIGURE 15. Comparison of Tests TH-3.03 and TH-2.14 Showing Level 17 Average Test Rod Temperatures and Reflood Flow Rates

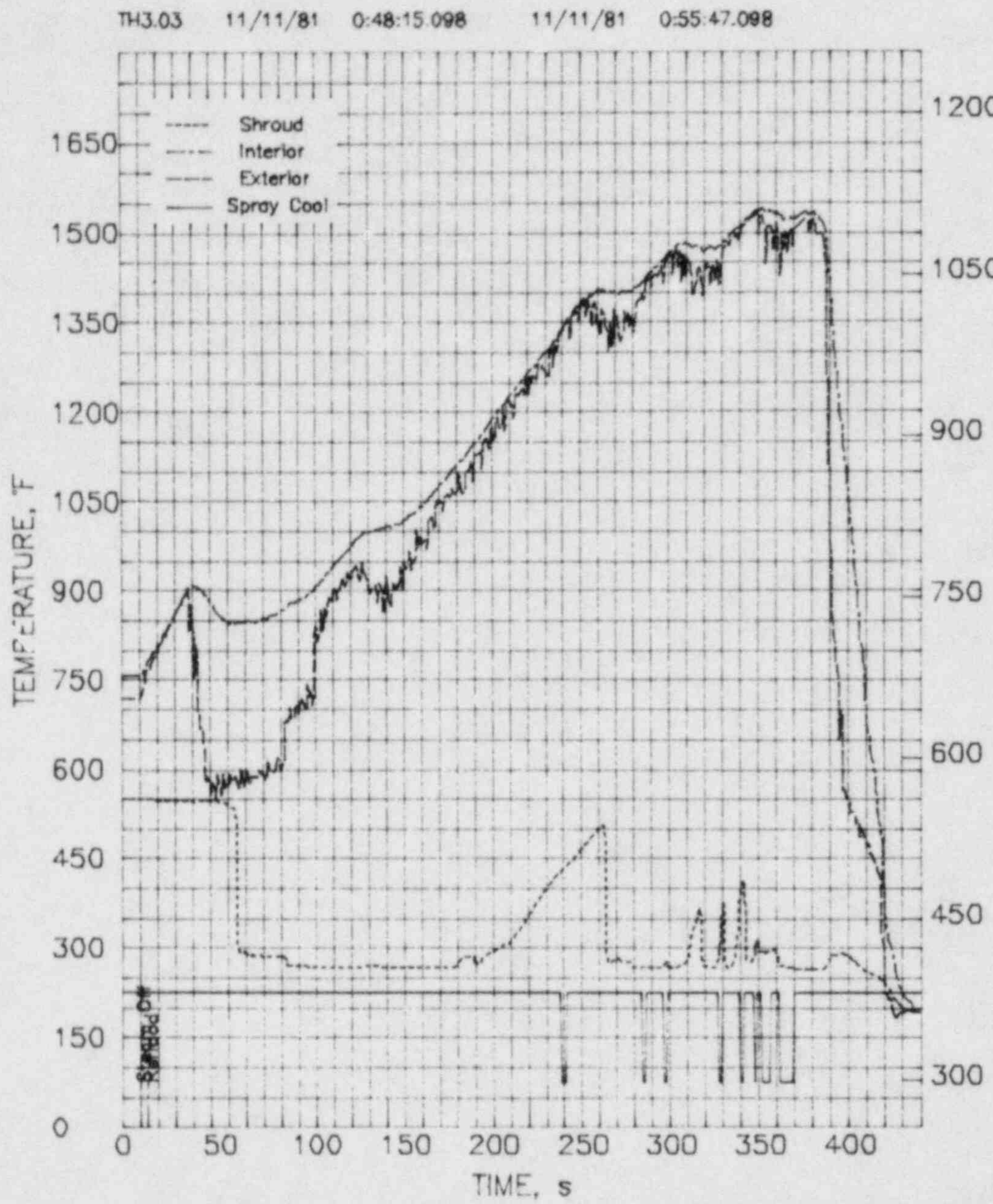


FIGURE 16. Spray Cooling Effects on the Fuel and Shroud at Level 18 for TH-3.03

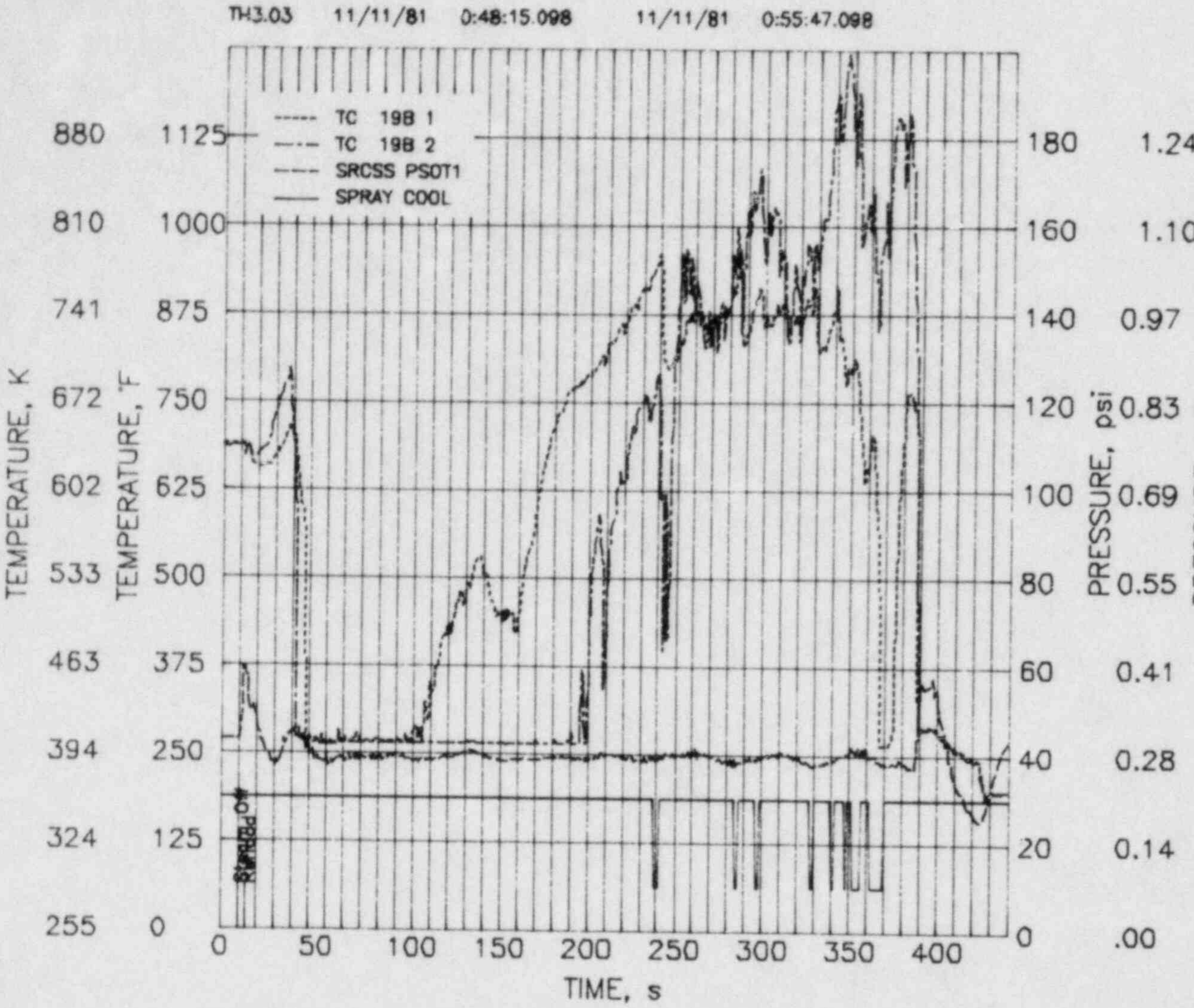


FIGURE 17. Spray Cooling Effects on Coolant Temperature at Level 19 and Outlet Pressure

INSTRUMENTATION FAILURE

The test rods used in this experiment were previously used in the TH-2 experiment, and the shroud and guard rods were previously used in TH-2, MT-2, and MT-1. Due to this extended use, some of the instrumentation was not operational for the TH-3 experiment. Table 5 lists the test assembly instrumentation that failed before or during the TH-3 experiment.

TABLE 5. TH-3 Test Assembly Instrumentation Failure Status

Instrument	Time of Failure
TC-3-2B-OR-4	between TH-2.14 and TH-3.01
TC-3-6A-S-C	before TH-2.01 transient
TC-4-1A-S-C	between TH-2.14 and TH-3.01
TC-4-6F-S-C	during TH-2.04 transient
TC-6-1A-S-C	between TH-2.14 and TH-3.01
TC-6-6A-S-C	before TH-2.01 preconditioning
TC-7-6C-OR-4	before TH-2.01 preconditioning
TC-7-4F-OR-1	before TH-2.01 preconditioning
TC-7-1D-OR-2	before TH-2.01 preconditioning
TC-7-3A-OR-3	during TH-2.01 transient
TC-8-2E-OR-3	between TH-2.14 and TH-3.01
TC-8-1A-S-C	between TH-2.14 and TH-3.01
TC-8-5B-OR-1	before TH-2.01 preconditioning
TC-8-6F-S-C	before TH-2.01 preconditioning
TC-8-5E-OR-2	during TH-2.04 transient
TC-9-5E-OR-2	between TH-2.14 and TH-3.01
TC-9-2E-OR-3	between TH-2.14 and TH-3.01
TC-10-1A-S-C	before TH-2.01 preconditioning
TC-10-1F-S-C	between TH-2.14 and TH-3.01
TC-11-6C-OR-4	before TH-2.01 preconditioning
TC-11-4F-OR-1	during TH-2.05 transient
TC-12-5B-SP-3	between TH-2.14 and TH-3.01
TC-12-5E-SP-4	between TH-2.14 and TH-3.01
TC-12-2B-SP-2	before TH-2.01 preconditioning
TC-13-4F-IR-4	between TH-2.14 and TH-3.01
TC-13-1A-S-C	between TH-2.14 and TH-3.01
TC-14-5B-SP-3	before TH-2.01 preconditioning
TC-14-2B-SP-2	before TH-2.01 preconditioning
TC-14-5E-SP-4	between TH-2.14 and TH-3.01
TC-15-3C-IR-2	before TH-2.01 preconditioning
TC-15-1F-S-C	before TH-2.01 preconditioning
TC-15-4F-IR-2	between TH-2.14 and TH-3.01
TC-15-3C-IR-C	between TH-2.14 and TH-3.01
TC-15-5B-IR-3	between TH-2.14 and TH-3.01

TABLE 5. (contd)

Instrument	Time of Failure
TC-15-5B-IR-C	between TH-2.14 and TH-3.01
TC-15-5B-IR-1	between TH-2.14 and TH-3.01
TC-15-1A-S-C	between TH-2.14 and TH-3.01
TC-16-6A-C-3	between TH-2.14 and TH-3.01
TC-16-6F-C-4	between TH-2.14 and TH-3.01
TC-16-2E-SP-1	before TH-2.01 preconditioning
TC-16-2B-SP-2	removed at reconstitution of TH-2
TC-16-5E-SP-4	before TH-2.01 preconditioning
TC-17-5E-IR-2	between TH-2.14 and TH-3.01
TC-17-6F-S-C	during TH-3.01
TC-17-1A-S-C	between TH-2.14 and TH-3.01
TC-17-2D-OR-1	between TH-2.14 and TH-3.01
TC-17-3D-IR-4	between TH-2.14 and TH-3.01
TC-17-3D-IR-3	before TH-2.01 preconditioning
TC-17-5E-IR-4	between TH-2.07 and TH-2.08
TC-17-5D-OR-3	during TH-2.11 transient
TC-18-6A-S-C	between TH-2.14 and TH-3.01
TC-18-1F-S-C	between TH-2.14 and TH-3.01
TC-18-1A-S-C	between TH-2.14 and TH-3.01
TC-19-4D-IR-P	between TH-2.14 and TH-3.01
TC-20-OT-1	between TH-2.14 and TH-3.01
TC-21-HT-1	before TH-2.01 preconditioning
TC-21-HT-2	not connected to DACS
TC-21-HT-3	not connected to DACS
TC-21-HT-4	not conntected to DACS
SPND-3-6F-S-3	before TH-2.01 preconditioning
SPND-6-1F-S-4	before TH-2.01 preconditioning
SPND-6-6A-S-2	between TH-2.14 and TH-3.01
SPND-10-6A-S-2	before TH-2.01 preconditioning
SPND-13-1F-S-4	before TH-2.01 preconditioning
SPND-13-1A-S-1	before TH-2.01 preconditioning
SPND-13-6A-S-2	before TH-2.01 preconditioning
SPND-15-1A-S-1	before TH-2.01 preconditioning
SPND-16-1F-C	before TH-2.01 preconditioning
SPND-17-6A-S-2	before TH-2.01 preconditioning
SPND-17-1F-S-4	before TH-2.01 preconditioning
SPND-18-6F-S-3	between TH-2.13 and TH-2.14
SPND-18-1A-S-1	before TH-2.01 preconditioning

THERMAL-HYDRAULIC ANALYSIS AND COMPARISON OF TEST DATA

A preliminary analysis of the test data was conducted to evaluate the repeatability of test conditions required to obtain a flat top transient in the 1033 to 1102K (1400 to 1525°F) temperature range. The results of this analysis are presented in this section, and the test conditions determined for the TH-3.03 test are presented in Table 2.

COMPARISON OF TH-2.12, TH-2.14, AND TH-3.03 TEST DATA

The TH-3 experiment was performed to evaluate the repeatability of operating conditions necessary to obtain an extended or flat top transient in the high alpha temperature range. The automatic control system used to control the variable reflood rates in this series demonstrated the capability of holding temperatures above 1033K (1400°F) for periods of up to 280 s. The ability of the automatic control system to repeat the experiment is demonstrated by the time-temperature graphs for TH-2.12, TH-2.14, and TH-3.03 at Levels 13, 15, and 17 (see Figures 18 through 20).

The heat transfer coefficients for TH-2.12, TH-2.14, and TH-3.03 are calculated based on local assembly powers and the difference between the average fuel rod temperature and an artificial coolant temperature, which was assumed to be at saturated conditions for 40 psia (267°F). Local assembly powers were calculated by means of a heat conduction technique for both the fuel rods and the shroud. Local heat transfer coefficients for TH-2.12, TH-2.14, and TH-3.03 at Levels 13, 15, and 17 will be used as input to predictive codes to provide information on test rod ballooning and deformation (see Figures 21 through 23).

ASSEMBLY POWER CALCULATIONS

The test assembly powers during the pretransients for TH-3.01, TH-3.02, and TH-3.03 were calculated by a calorimetric method. The test assembly powers were determined from the flow rates provided by the U-2 loop instrumentation and inlet and outlet steam temperatures obtained by averaging the test assembly inlet (Level 1) TC readings and the outlet (Level 20) TC readings. Using the test assembly temperatures rather than the loop temperatures eliminated the effect of heat losses in the loop piping. Average rod powers were calculated to be 1.207, 1.152, and 1.142 kW/m (0.368, 0.351, and 0.348 kW/ft), respectively, for TH-3.01, TH-3.02, and TH-3.03.

A heat conduction calculation based on the cladding temperature ramp rates during the nominally adiabatic heatup period between the times when the steam cooling was turned off and the reflood cooling was turned on was used to determine local fuel rod powers for all tests. Total assembly powers are calculated by adding local fuel rod powers to those of the shroud. An additional test (TH-3.01) was performed to obtain heatup rate data. Integration of the axial power distribution so obtained gave an average fuel rod power of 1.332 kW/m (0.406 kW/ft). The average rod power for the ramp rate calculation was 10.4%

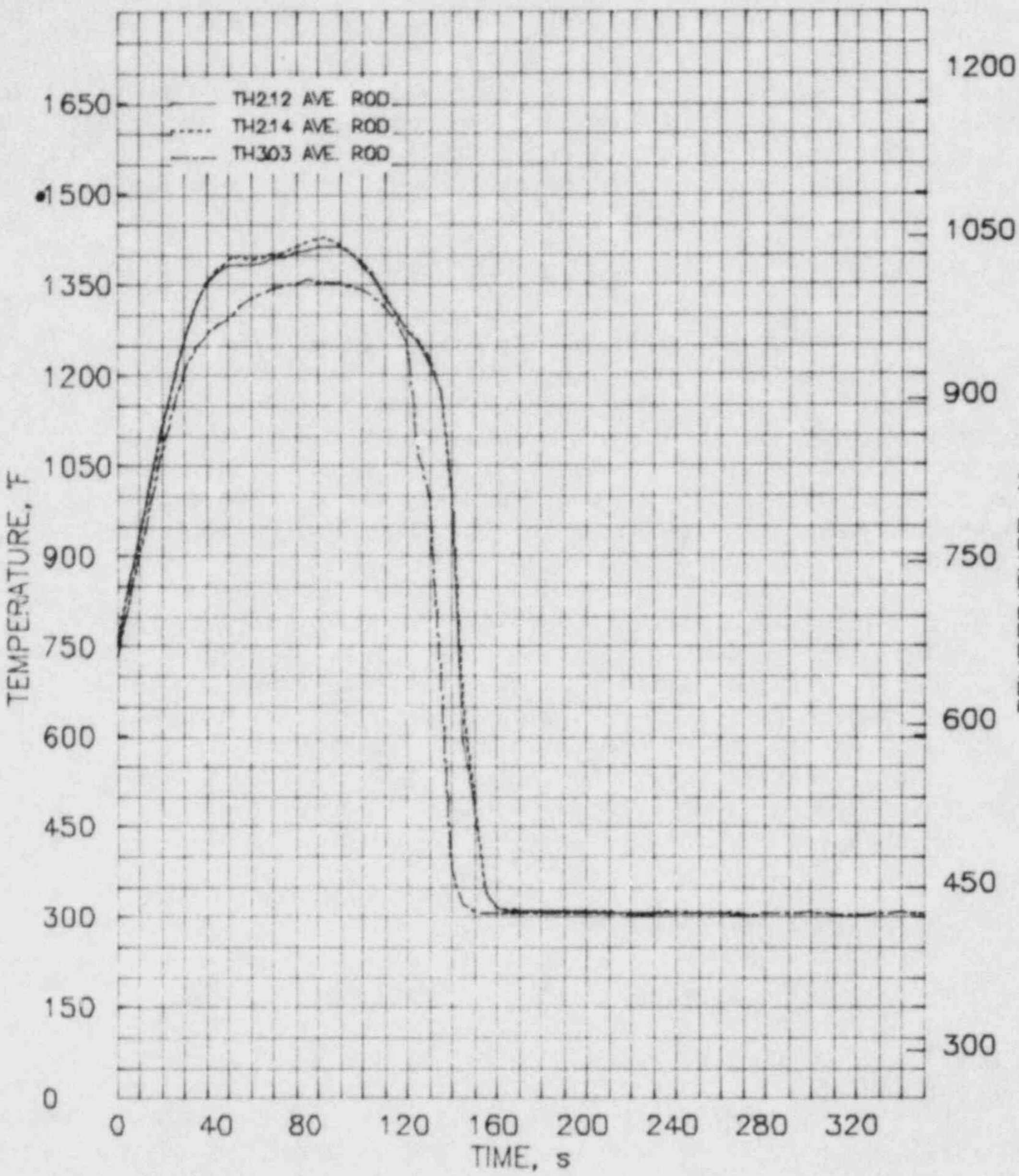


FIGURE 18. Average Cladding Temperatures at Level 13 for TH-2.12, TH-2.14, and TH-3.03

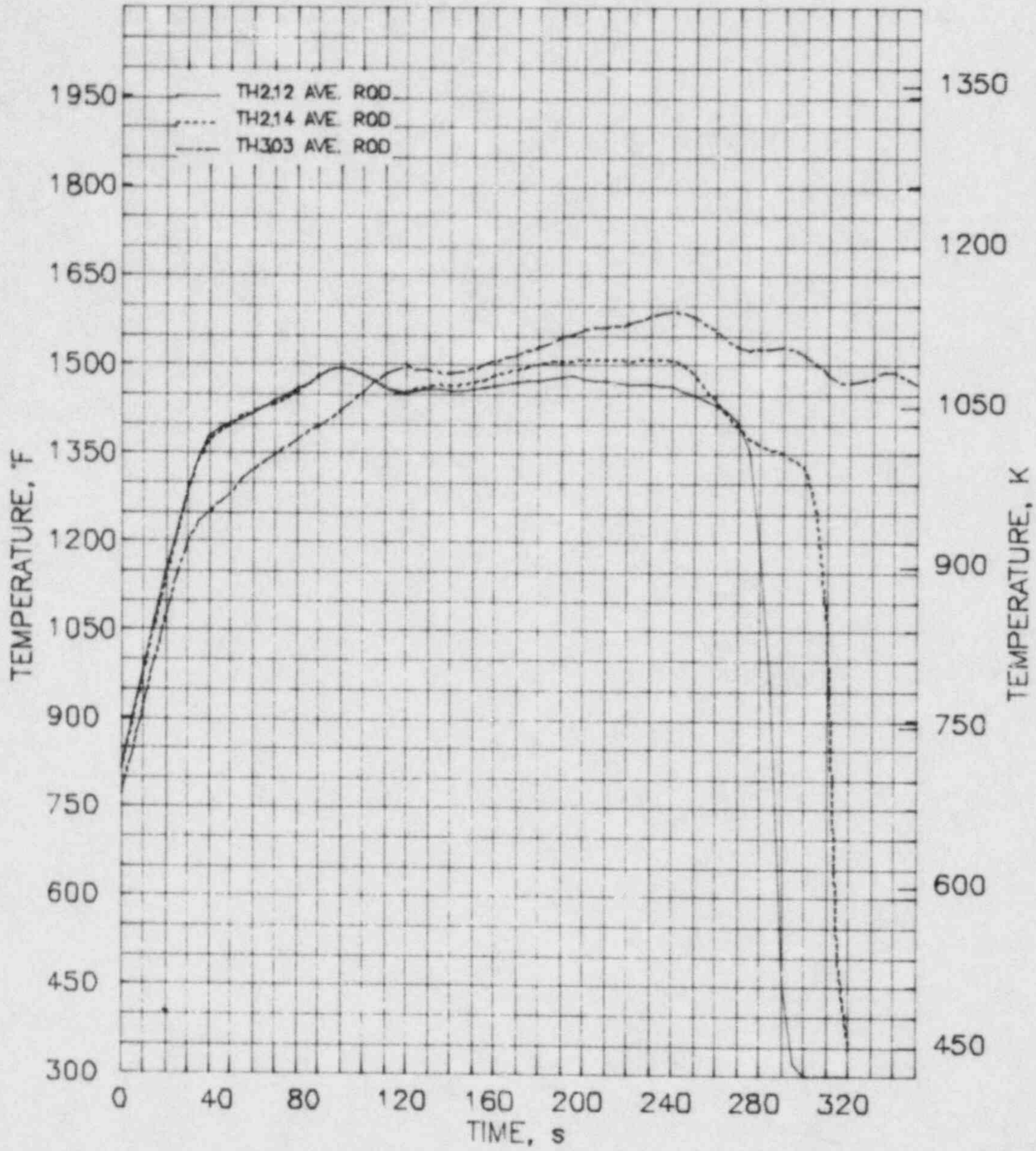


FIGURE 19. Average Cladding Temperatures at Level 15 for TH-2.12, TH-2.14, and TH-3.03

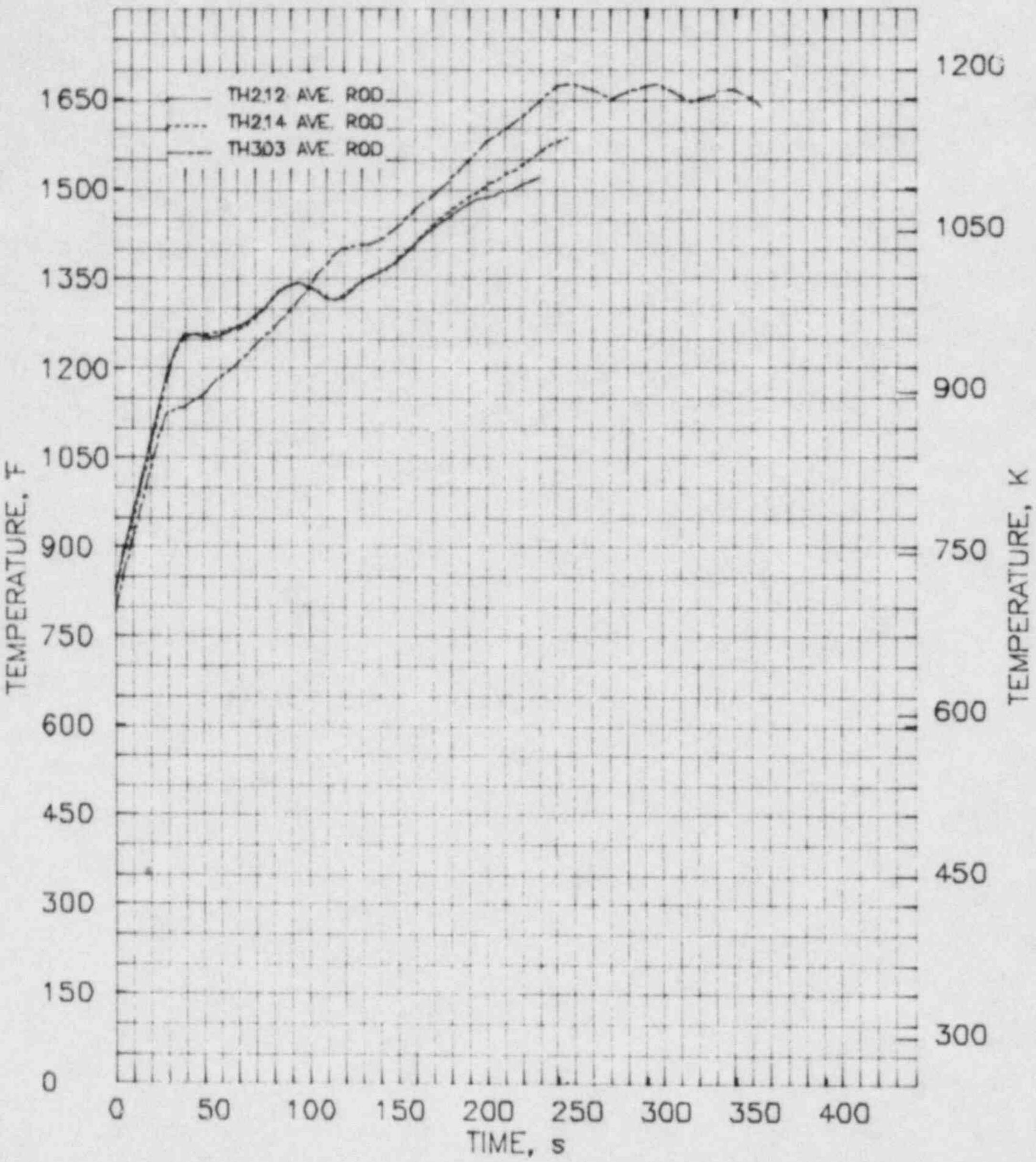


FIGURE 20. Average Cladding Temperatures at Level 17 for TH-2.12, TH-2.14, and TH-3.03

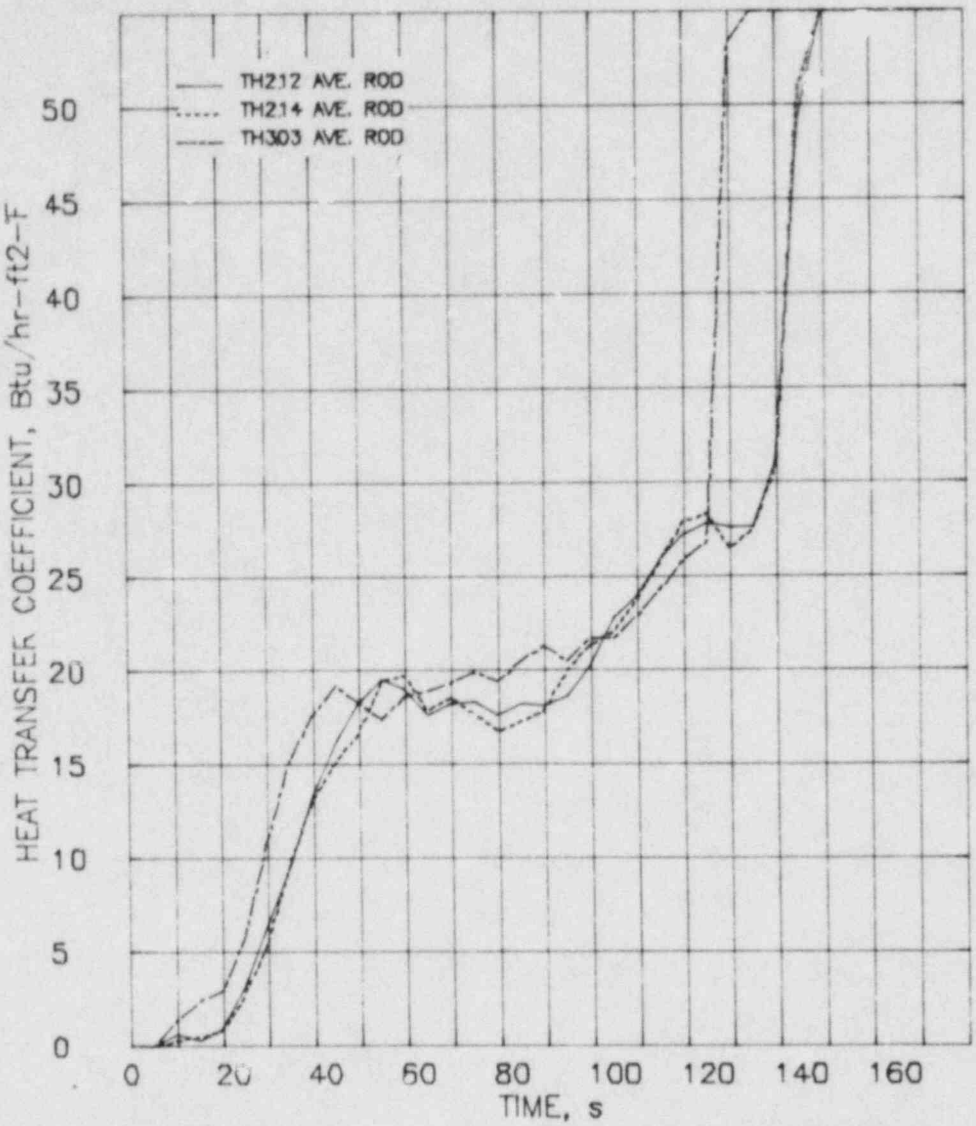


FIGURE 21. Heat Transfer Coefficients at Level 13 for TH-2.12, TH-2.14, and TH-3.03

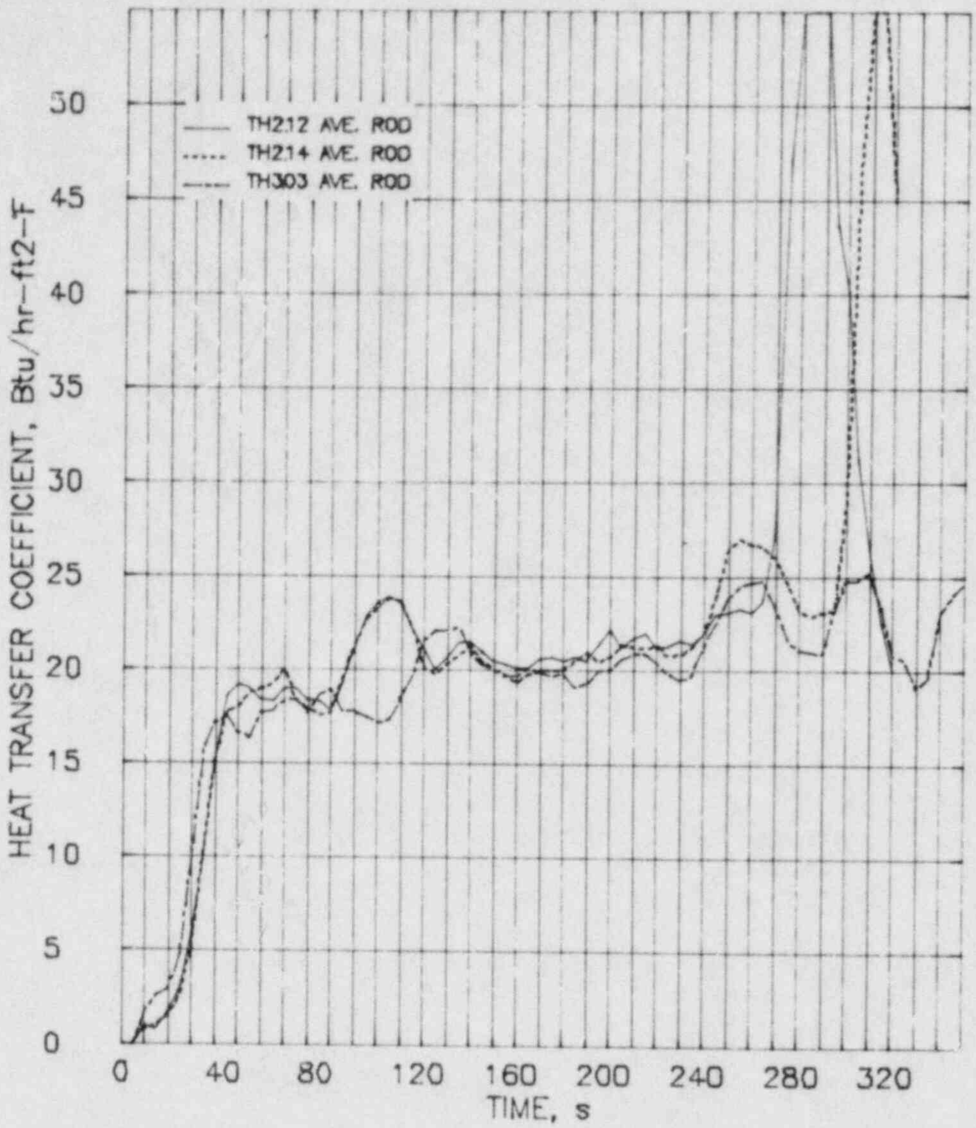


FIGURE 22. Heat Transfer Coefficients at Level 15 for TH-2.12, TH-2.14, and TH-3.03

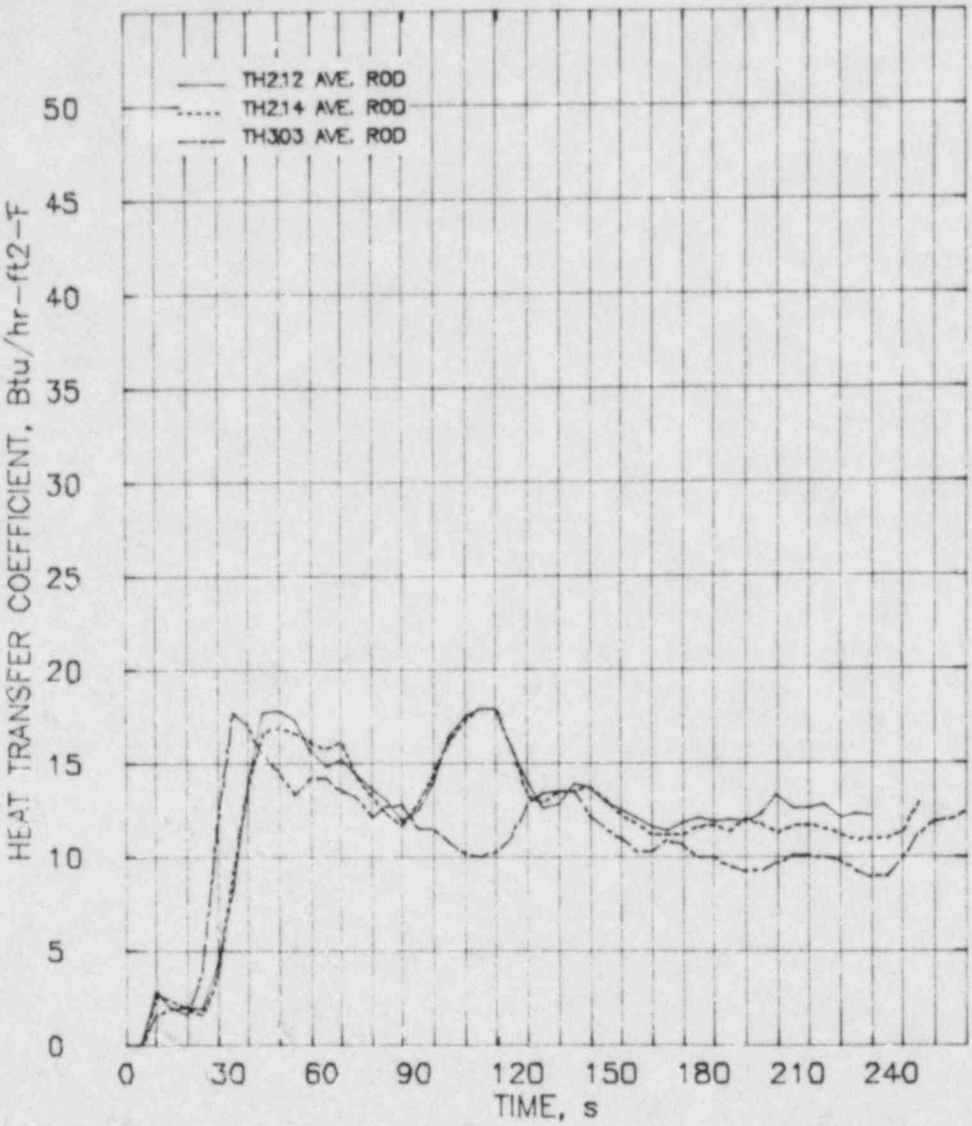


FIGURE 23. Heat Transfer Coefficients at Level 17 for TH-2.12, TH-2.14, and TH-3.03

greater than for the calorimetric calculation for TH-3.01. The heatup rate method of determining local fuel power was used for post-test TRUMP[6] code calculations as described in the following section.

TRUMP CODE POST-TEST ANALYSIS

Post-test calculations for the TH-3 experiment were made using heat transfer coefficients determined from the FLECHT correlation[7] as input to the TRUMP heat conduction code.

TH-3.03 calculations involved evaluating the test data to determine specific delay times, reflood rates, and their durations. The FLECHT correlation input was modified to accommodate the local axial rod power as determined from an adiabatic heatup test and the changes in reflood rates used during the course of the transients for TH-3.03. The calculated heat transfer coefficients were then used as input to the TRUMP code. Measured cladding temperature data are compared with the post-test analysis from the TRUMP-FLECHT calculation for the temperature ramp values of the cladding in Figures 24 through 26. These post-test calculations involved evaluating the test data to continue improving the ability of TRUMP to calculate cladding temperatures for a given set of test parameters for up to 90 s to provide improved pretest calculations for subsequent tests. After the initial 90-s period at which time peak cladding temperatures was expected to be between 1033 and 1103K (1400 and 1525°F), transient control was shifted to the DACS.

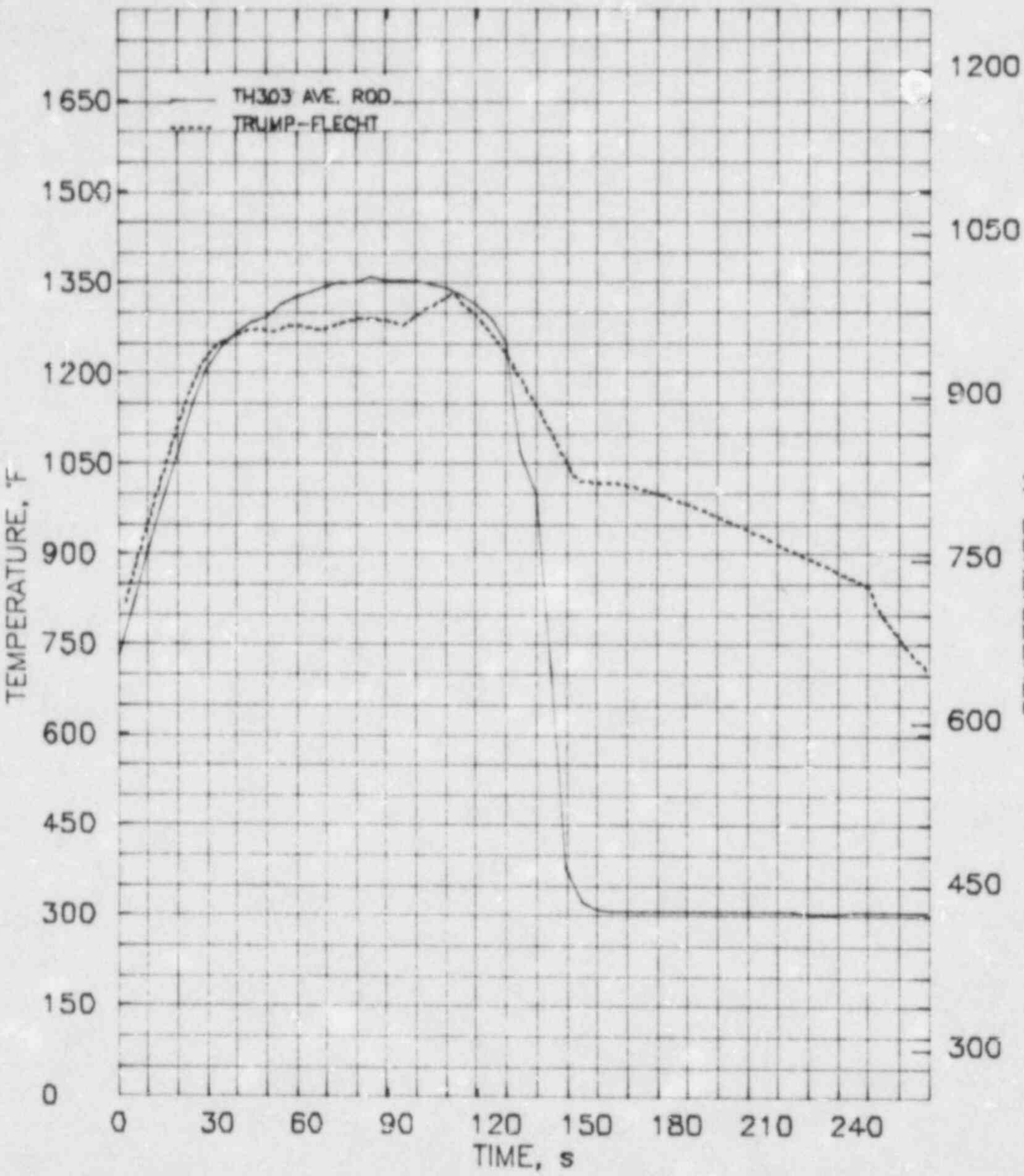


FIGURE 24. Comparison of TH-3.03 Average Cladding Temperature with TRUMP-FLECHT Calculation for Level 13

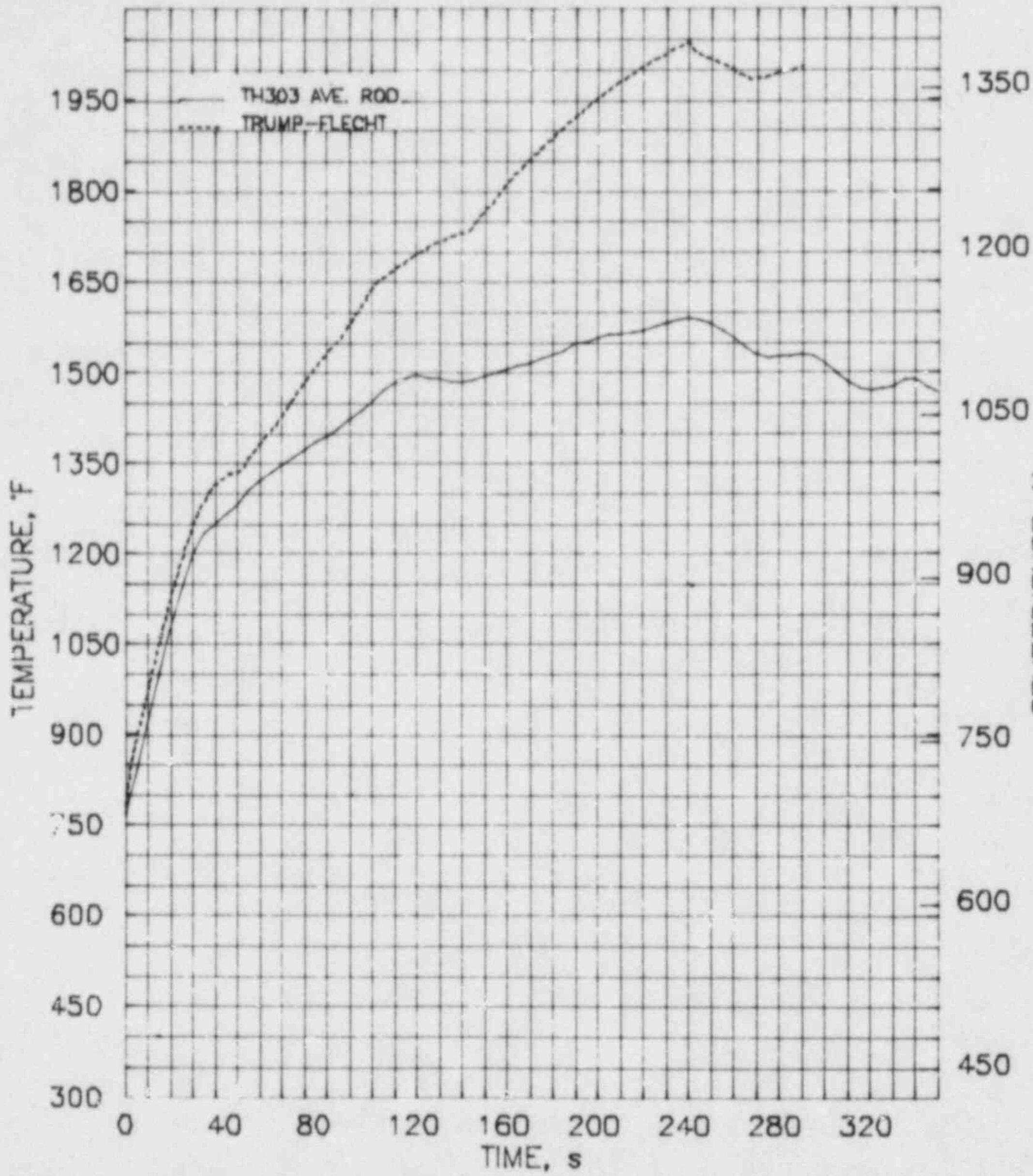


FIGURE 25. Comparison of TH-3.03 Average Cladding Temperature with TRUMP-FLECHT Calculation for Level 15

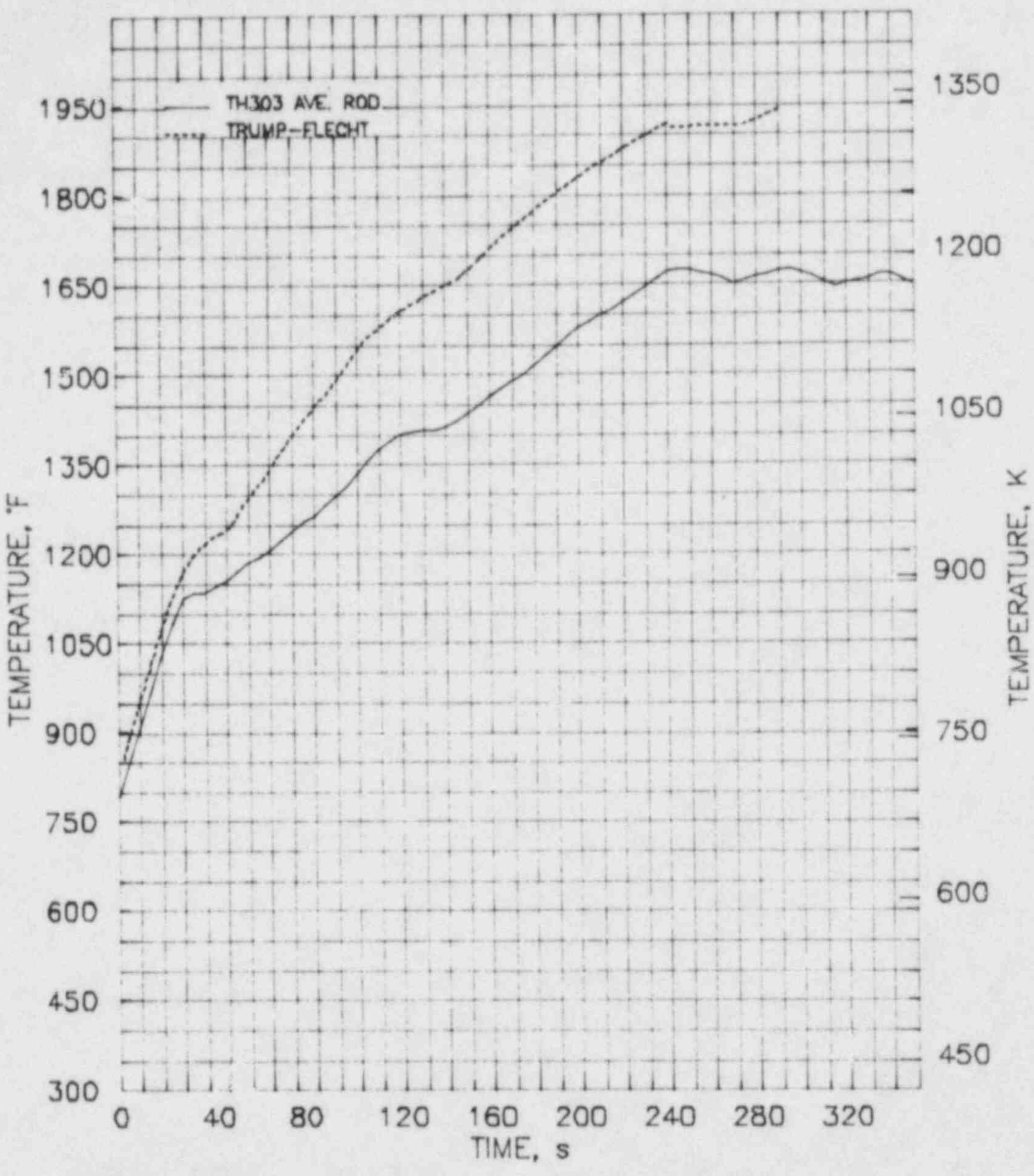


FIGURE 26. Comparison of TH-3.03 Average Cladding Temperature with TRUMP-FLECHT Calculation for Level 17

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1. Hann, C. R. 1979. Program Plan for LOCA Simulation in the National Research Universal (NRU) Reactor. PNL-3056, Pacific Northwest Laboratory, Richland, Washington.

2. Mohr, C. L., et al. 1980. Prototypic Thermal-Hydraulic Experiments in NRU to Simulate Loss-of-Coolant Accidents. NUREG/CR-1882, PNL-3681, Pacific Northwest Laboratory, Richland, Washington.

3. Russcher, G. E., et al. 1981. Materials Test-1 LOCA Simulation in the NRU Reactor. NUREG/CR-2152, Vol. 1, PNL-3835, Pacific Northwest Laboratory, Richland, Washington.

4. Mohr, C. L., et al. Materials Test-2 LOCA Simulation in the NRU Reactor. NUREG/CR-2509, PNL-4155, Pacific Northwest Laboratory, Richland, Washington.

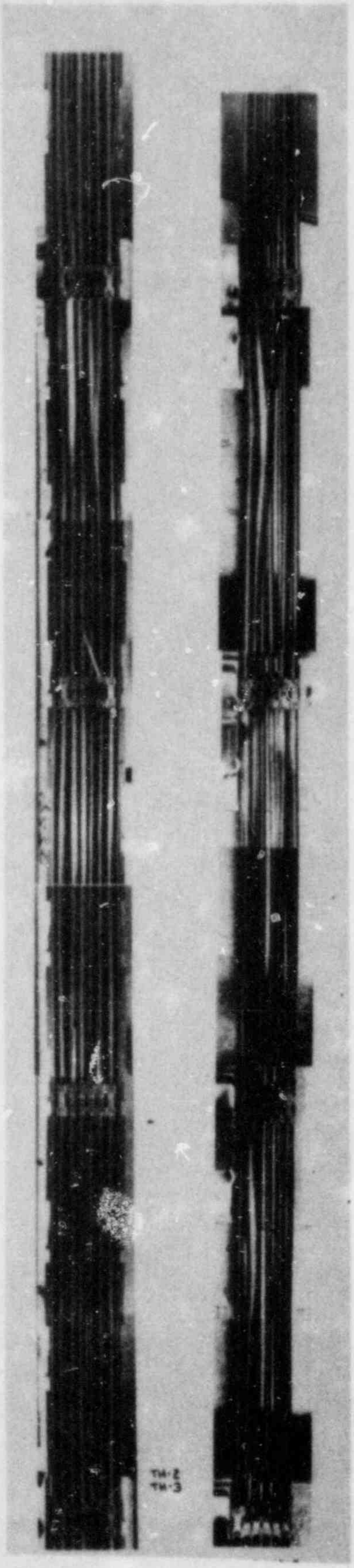
5. Mohr, C. L., et al. 1982. LOCA Simulation in NRU Program - Thermal-Hydraulic Experiment 2 Data Report. NUREG/CR-2526, PNL-4164, Pacific Northwest Laboratory, Richland, Washington.

6. Siefken, L. J., et al. June 1979. FRAP-T5: A Computer Code for the Transient Analysis of Oxide Fuel Rods. NUREG/CR-0840, TREE-1282, EG&G Idaho, Inc., Idaho Falls, Idaho.

7. Lilly, G. P., et al. 1977. PWR FLECHT Cosine Low Flooding Rate Test Series Evaluation Report. WCAP-8838, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

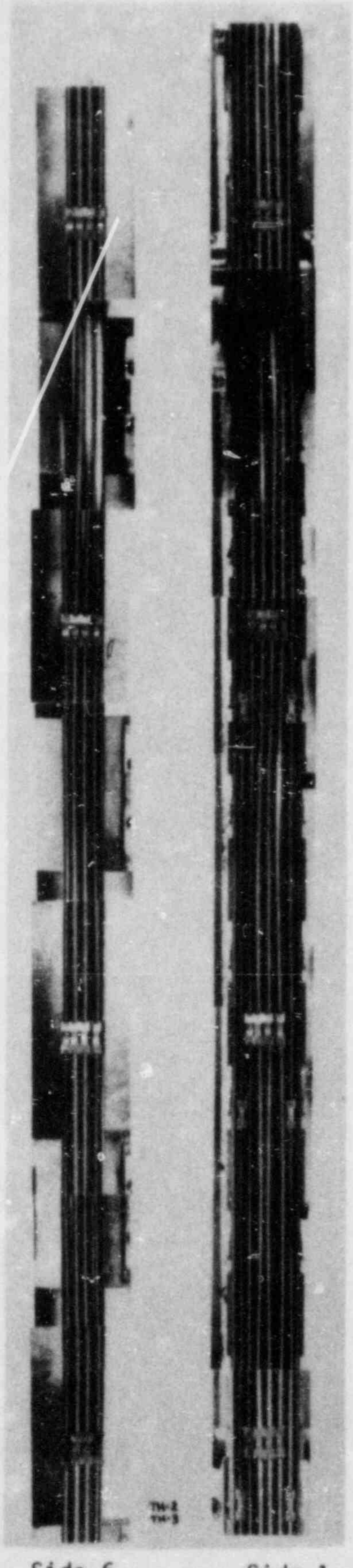
APPENDIX A

GUARD AND TEST FUEL PHOTOGRAPHS



Guard Bundle Side 6 Guard Bundle Side 1

FIGURE A.1 TH-2 and TH-3 Guard Fuel Bundle; Sides 1 and 6



Side 6 Side 1

FIGURE A.2. TH-2 and TH-3 Test Fuel Bundle; Sides 1 and 6

APPENDIX B

PRETRANSIENT TEST ASSEMBLY TEMPERATURES

APPENDIX B

PRETRANSIENT TEST ASSEMBLY TEMPERATURES

Pretransient temperature data summaries are presented in this appendix. Steam at ~0.379 kg/s (~3000 lbm/h) was used to cool the test assembly, and test assembly backpressure was maintained at 0.28 MPa (40 psia).

Diagonal temperature profiles from one shroud corner to the opposite shroud corner at Levels 13, 15, and 17 are shown in Figures B.1 through B.3. Axial temperature profiles obtained from steam probes, carrier thermocouples (TCs), and shroud TCs (all located in the shroud corners) are shown in Figures B.4 through B.6. Average steam probe axial temperatures are shown in Figures B.7 through B.9.

The remainder of this appendix consists of the following graphical data:

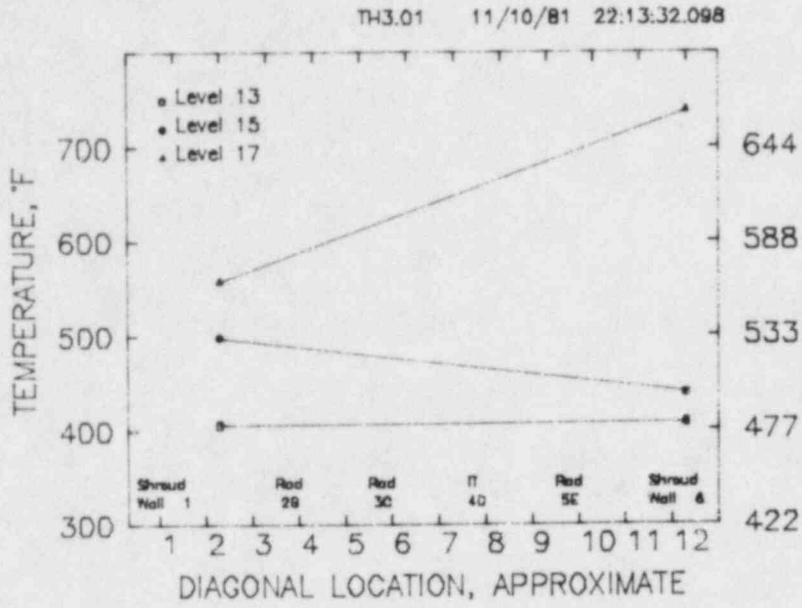


FIGURE B.1. Diagonal Temperature Profile Across Test Assembly Coolant During Pretransient for TH-3.01

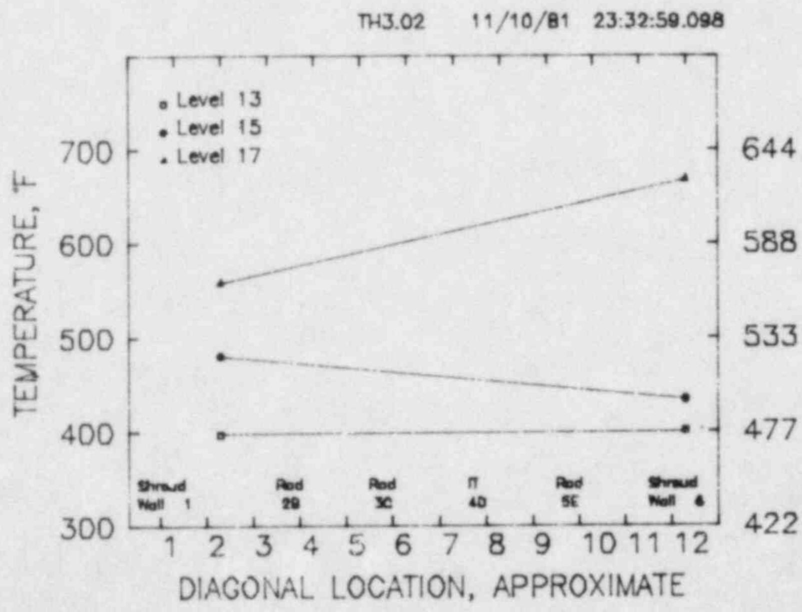


FIGURE B.2. Diagonal Temperature Profile Across Test Assembly Coolant During Pretransient for TH-3.02

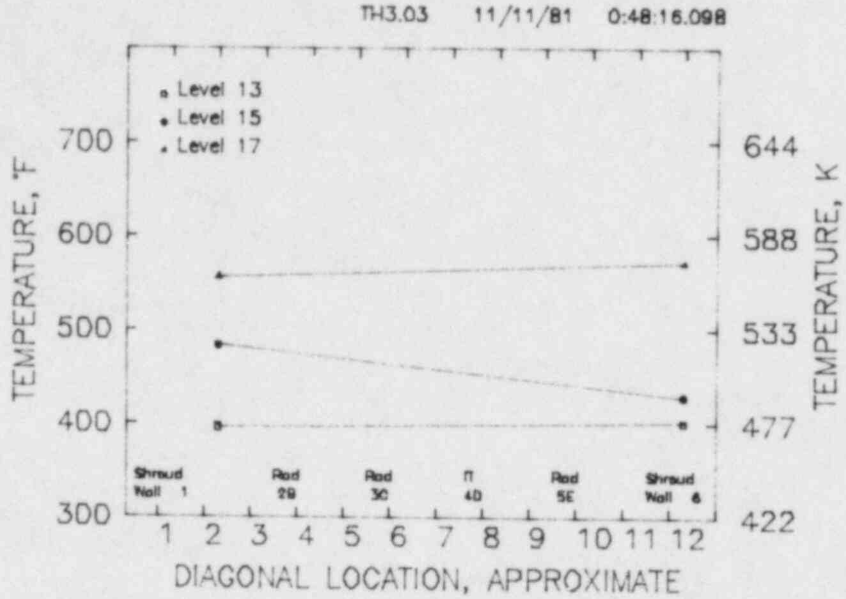


FIGURE B.3. Diagonal Temperature Profile Across Test Assembly Coolant During Pretransient for TH-3.03

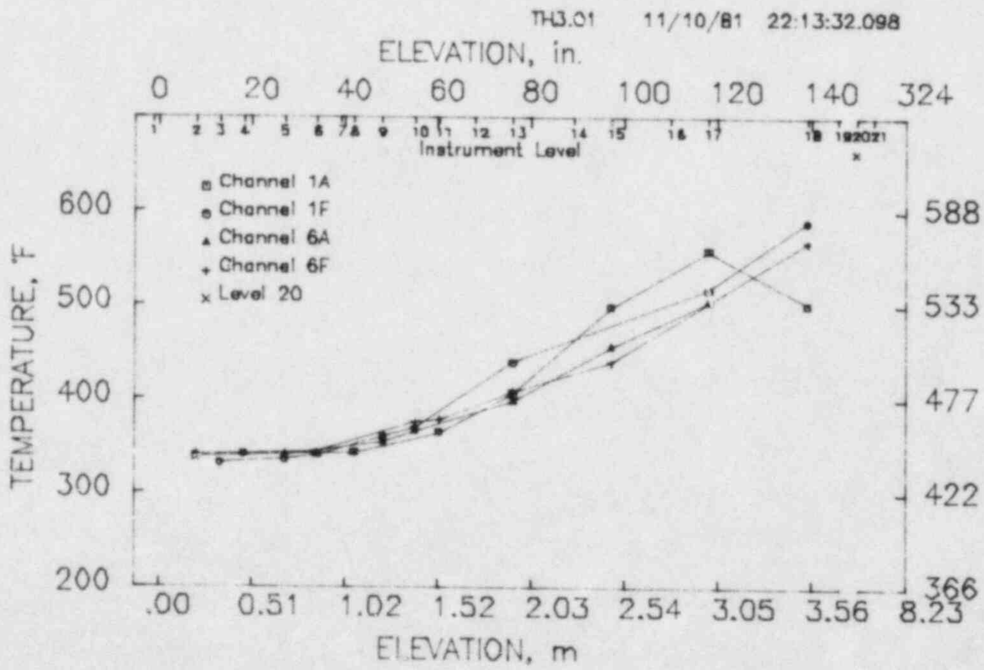


FIGURE B.4. Shroud Axial Temperature Profiles During Pretransient for TH-3.01

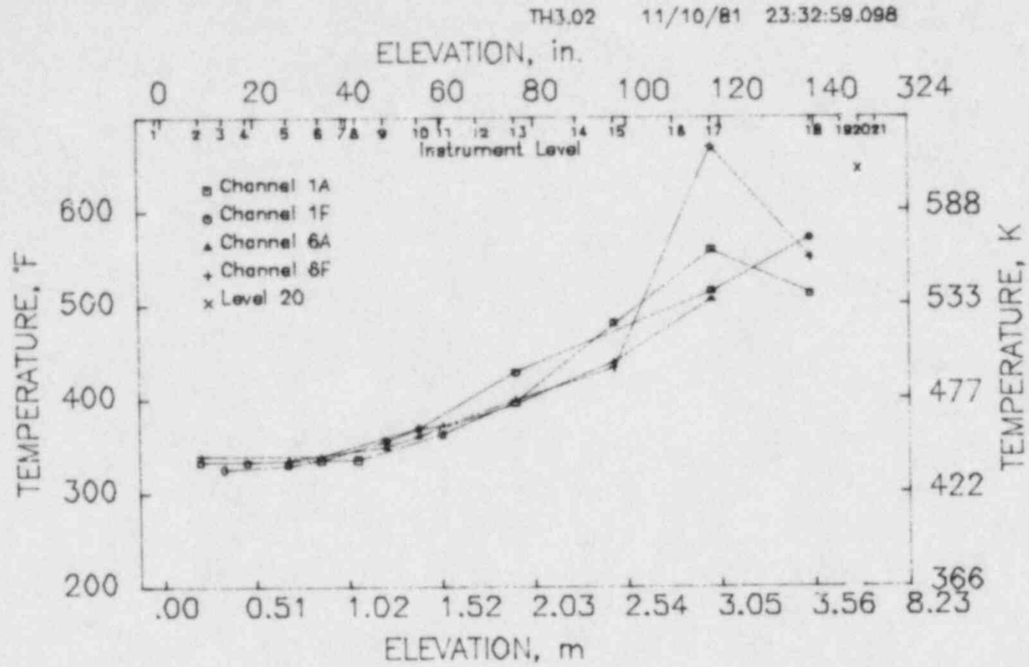


FIGURE B.5. Shroud Axial Temperature Profiles During Pretransient for TH-3.02

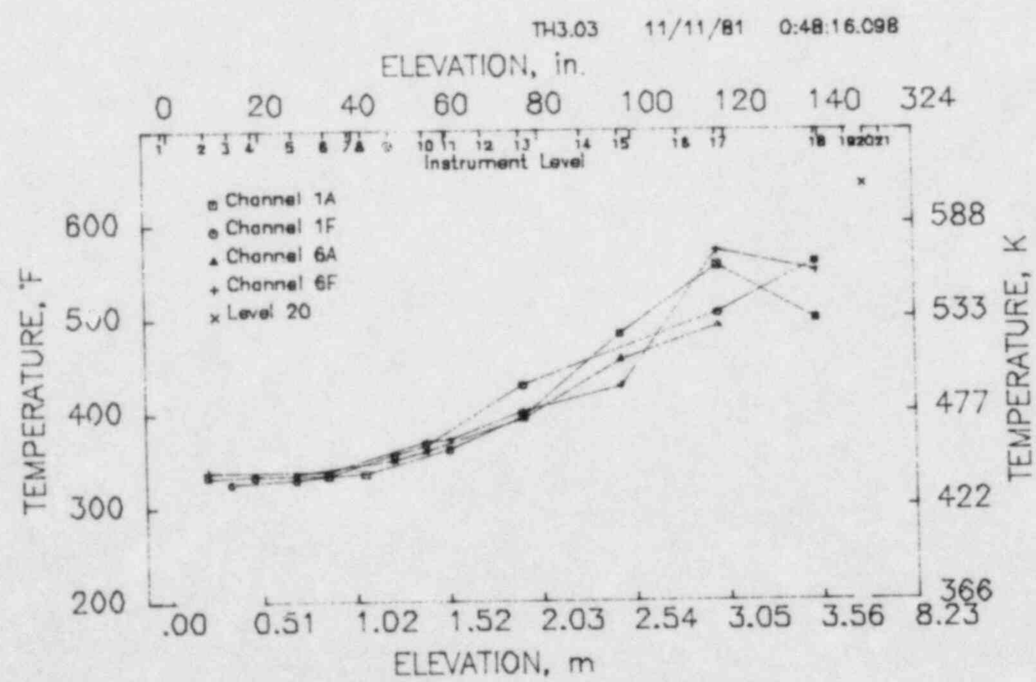


FIGURE B.6. Shroud Axial Temperature Profiles During Pretransient for TH-3.03

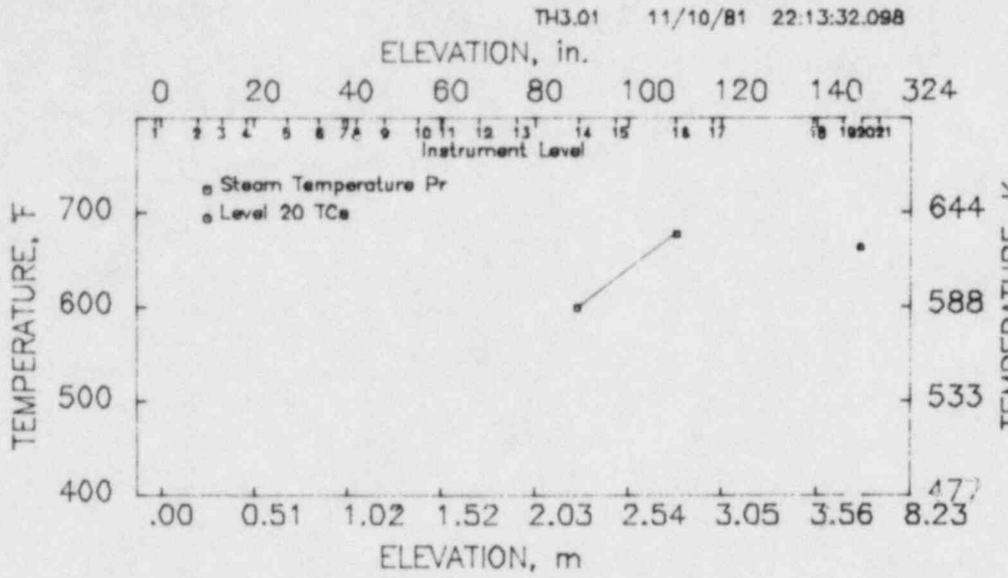


FIGURE B.7. Average Steam Probe Temperature Profile (in steam) During Pretransient for TH-3.01

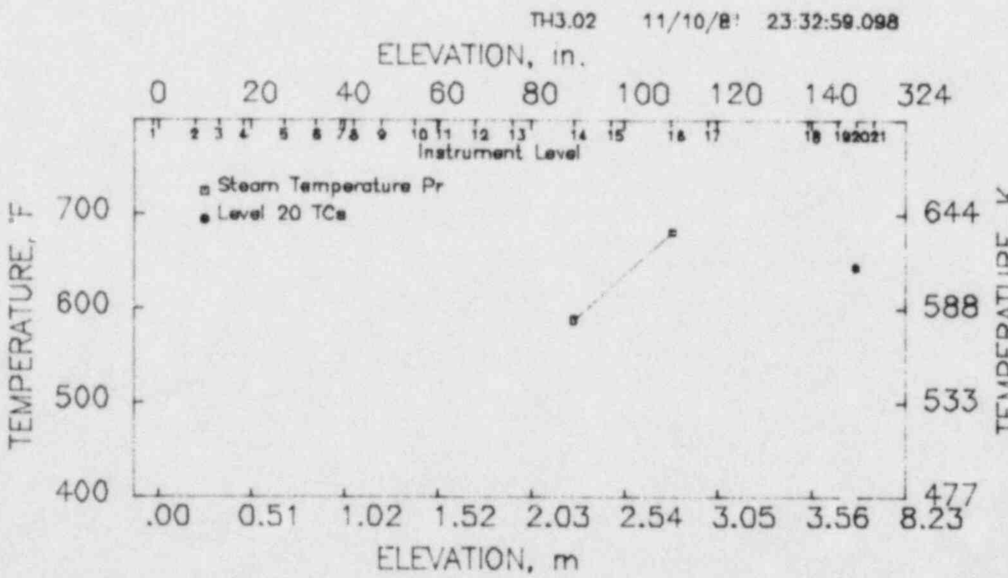


FIGURE B.8. Average Steam Probe Temperature Profile (in steam) During Pretransient for TH-3.02

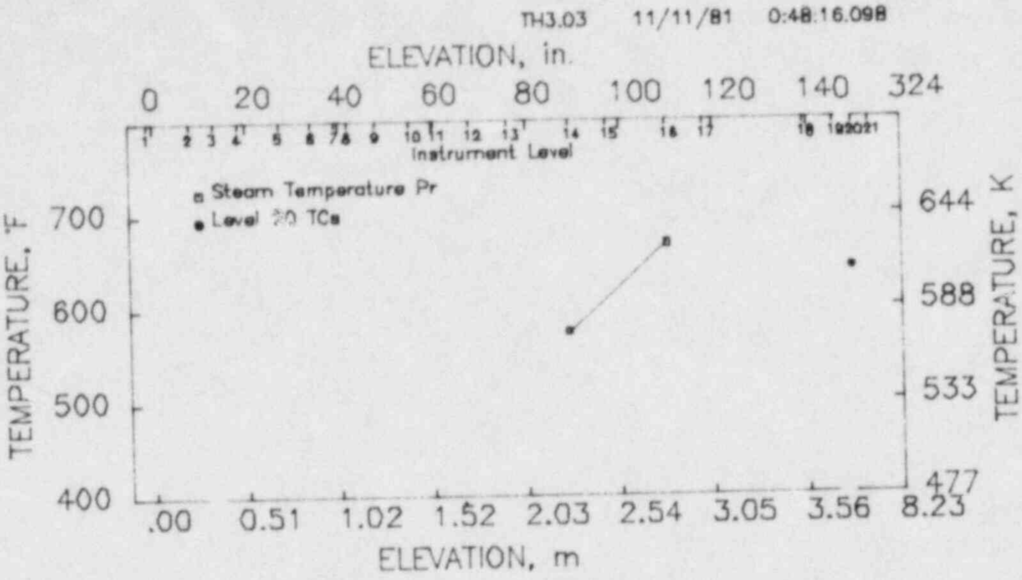


FIGURE B.9. Average Steam Probe Temperature Profile (in steam) During Pretransient for TH-3.03

APPENDIX C

TRANSIENT FUEL AND CLADDING TEMPERATURES

APPENDIX C

TRANSIENT FUEL AND CLADDING TEMPERATURES

Transient fuel and cladding temperatures are presented in this appendix. The test assembly environment during the transient can be steam, water, or both with a changing neutron flux depending partially on control rod position and the changing reactivity of water and steam. The test assembly backpressure was maintained at ~0.28 MPa (40 psia).

Selected test rod center and cladding inner temperatures for Levels 13, 15, and 17 are shown in Figures C.1 through C.9. Shroud and test train axial temperature profiles (see Figures C.10 and C.11) show the effects of reflooding and the axial power profile on temperature distributions throughout the transient.

Average fuel temperatures for test and guard fuel rods at Levels 13, 15, and 17 in test TH-3.03 are shown in Figures C.12 through C.14. These figures demonstrate the resultant average guard and test fuel temperatures when using hot spot sensor feedback control.

The effect of spray cooling on the fuel and shroud at Level 18 is shown in Figure C.15; the effect of spray cooling on coolant temperature at Level 19 and the outlet pressure are shown in Figure C.16. Spray cooling is shown turned on when its line is low.

The remainder of this appendix consists of the following graphical data:

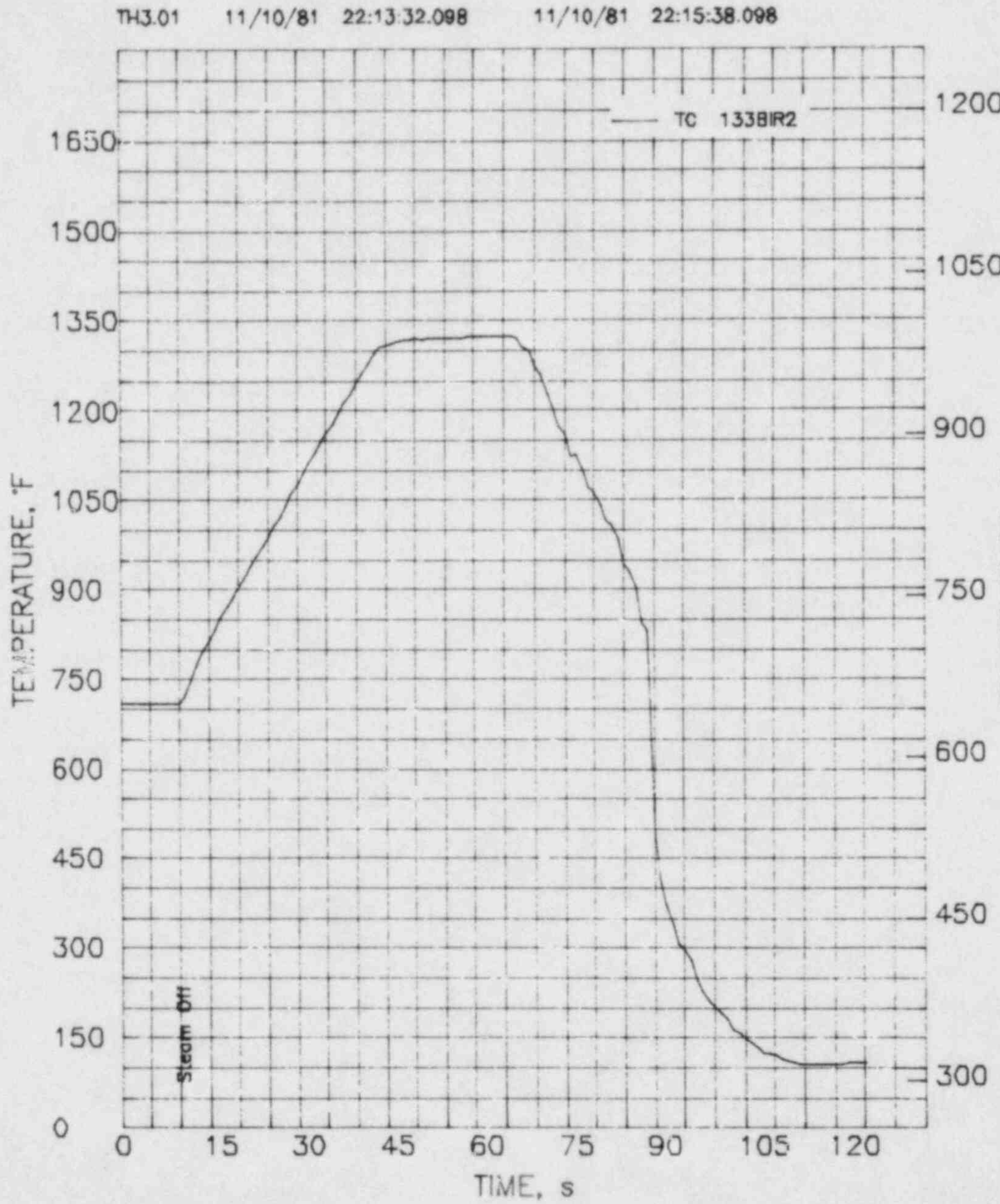


FIGURE C.1. Test Rod Interior Cladding Temperature History at Level 13 During TH-3.01 Transient

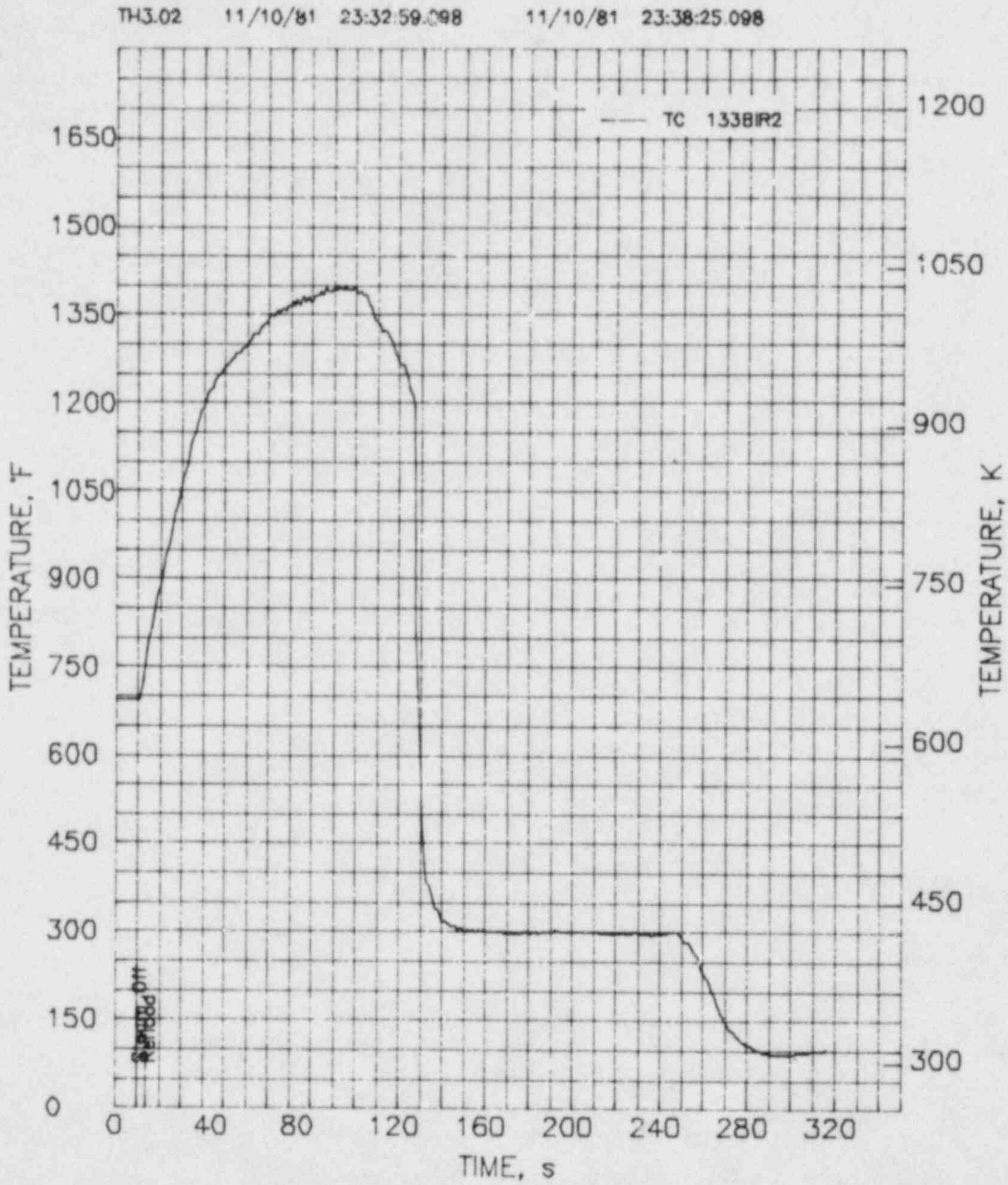


FIGURE C.2. Test Rod Interior Cladding Temperature History at Level 13 During TH-3.02 Transient

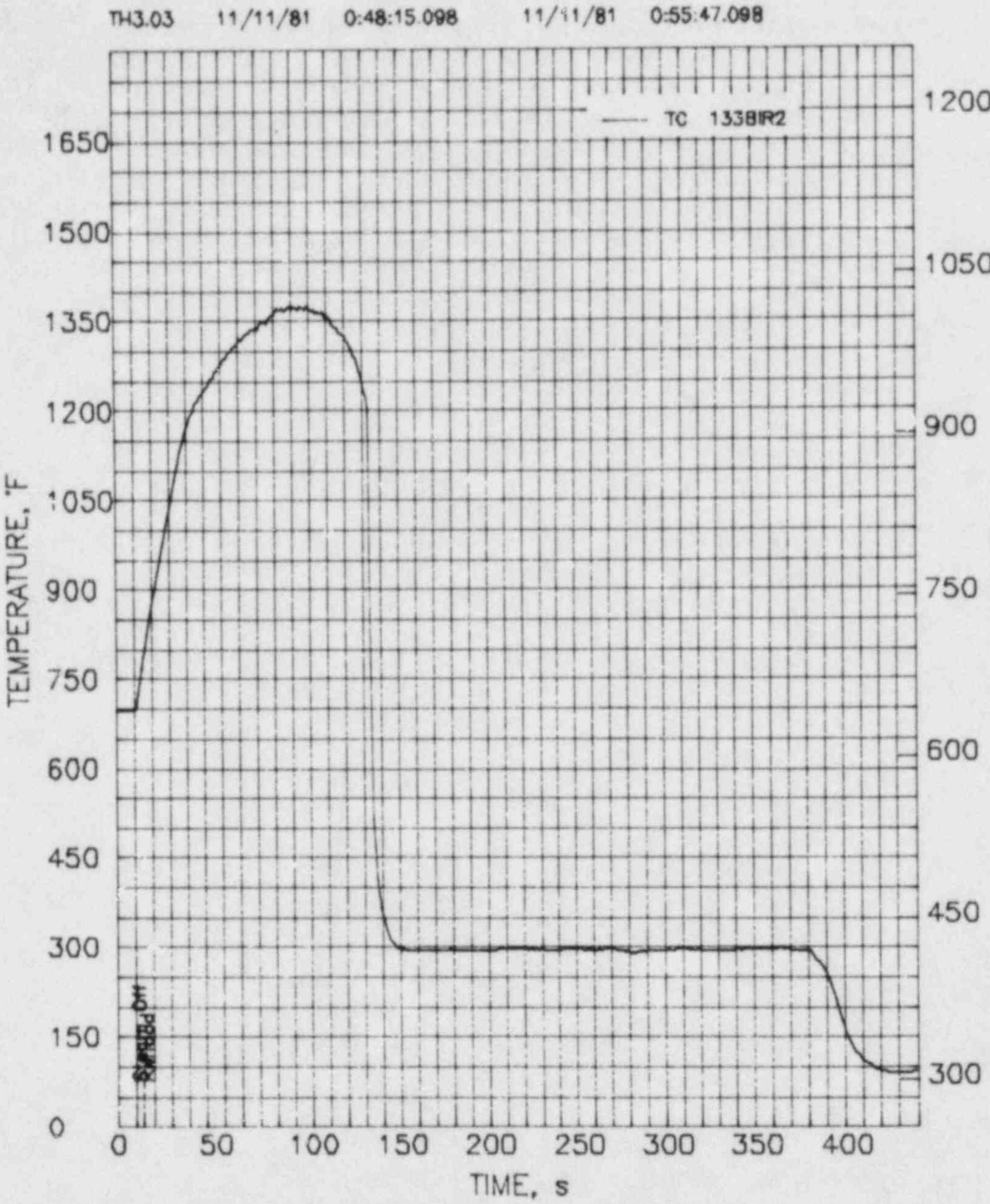


FIGURE C.3. Test Rod Interior Cladding Temperature History at Level 13 During TH-3.03 Transient

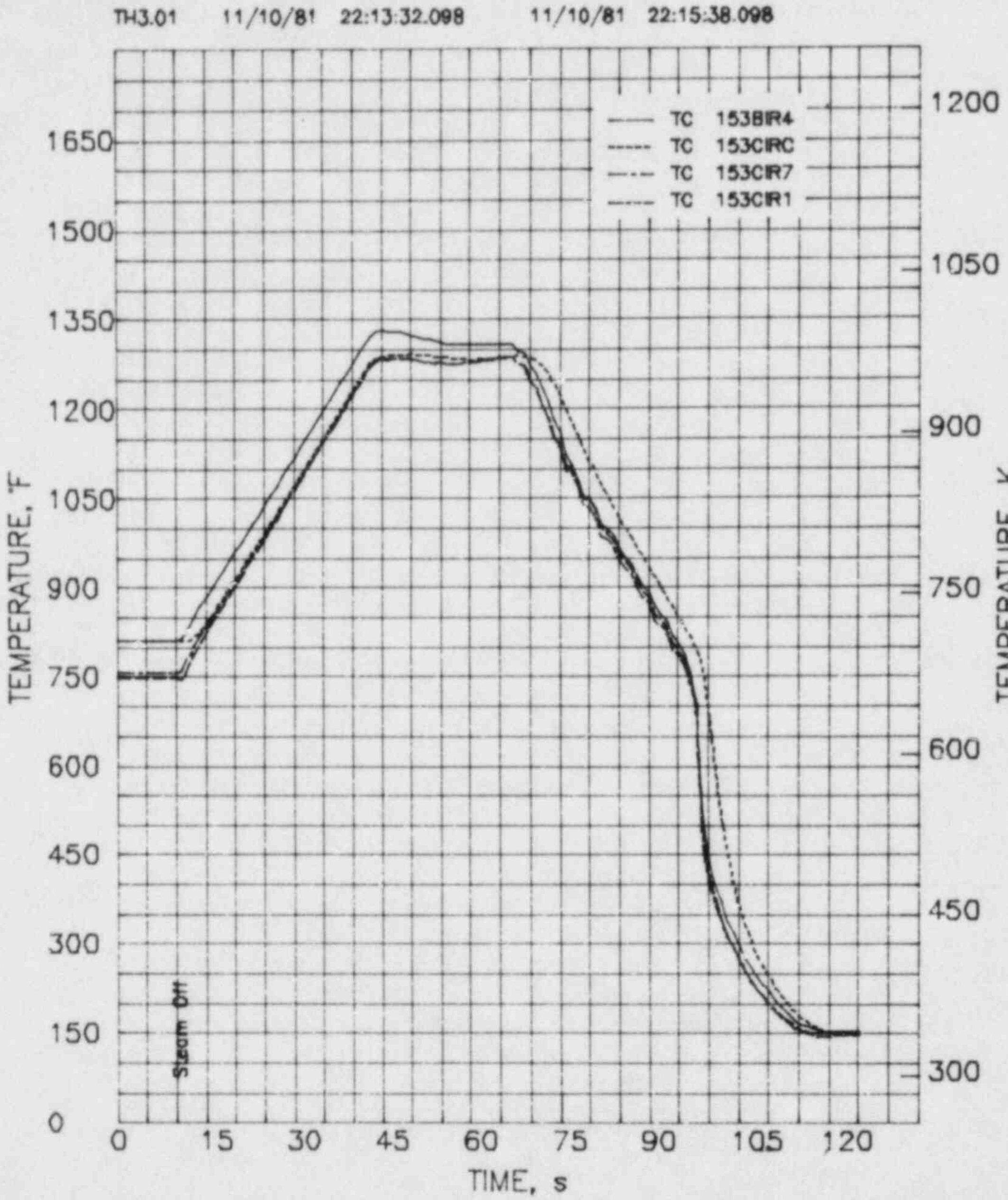


FIGURE C.4. Test Rod Center and Interior Cladding Temperature Histories at Level 15 During TH-3.01 Transient

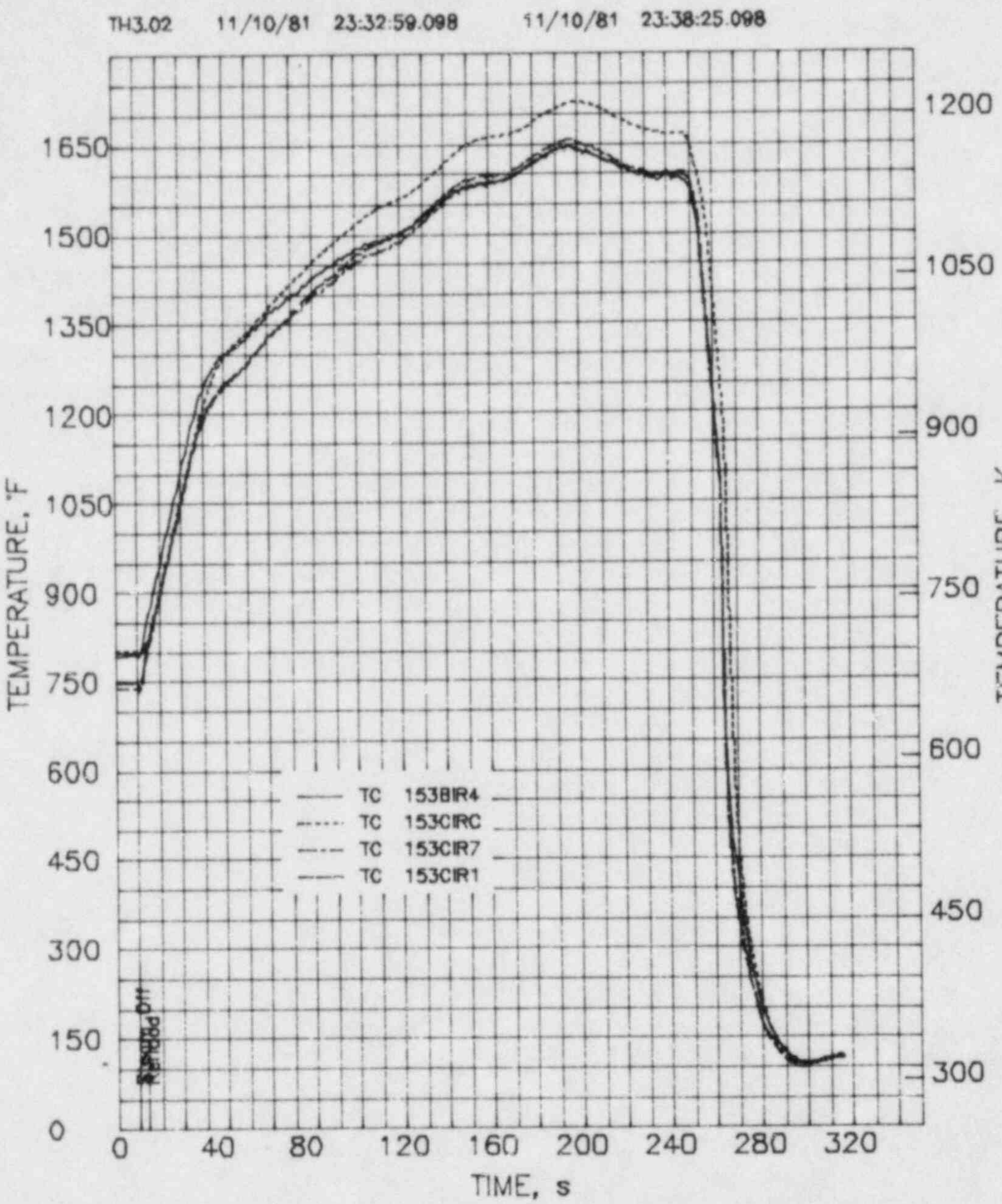


FIGURE C.5. Test Rod Center and Interior Cladding Temperature Histories at Level 15 During TH-3.02 Transient

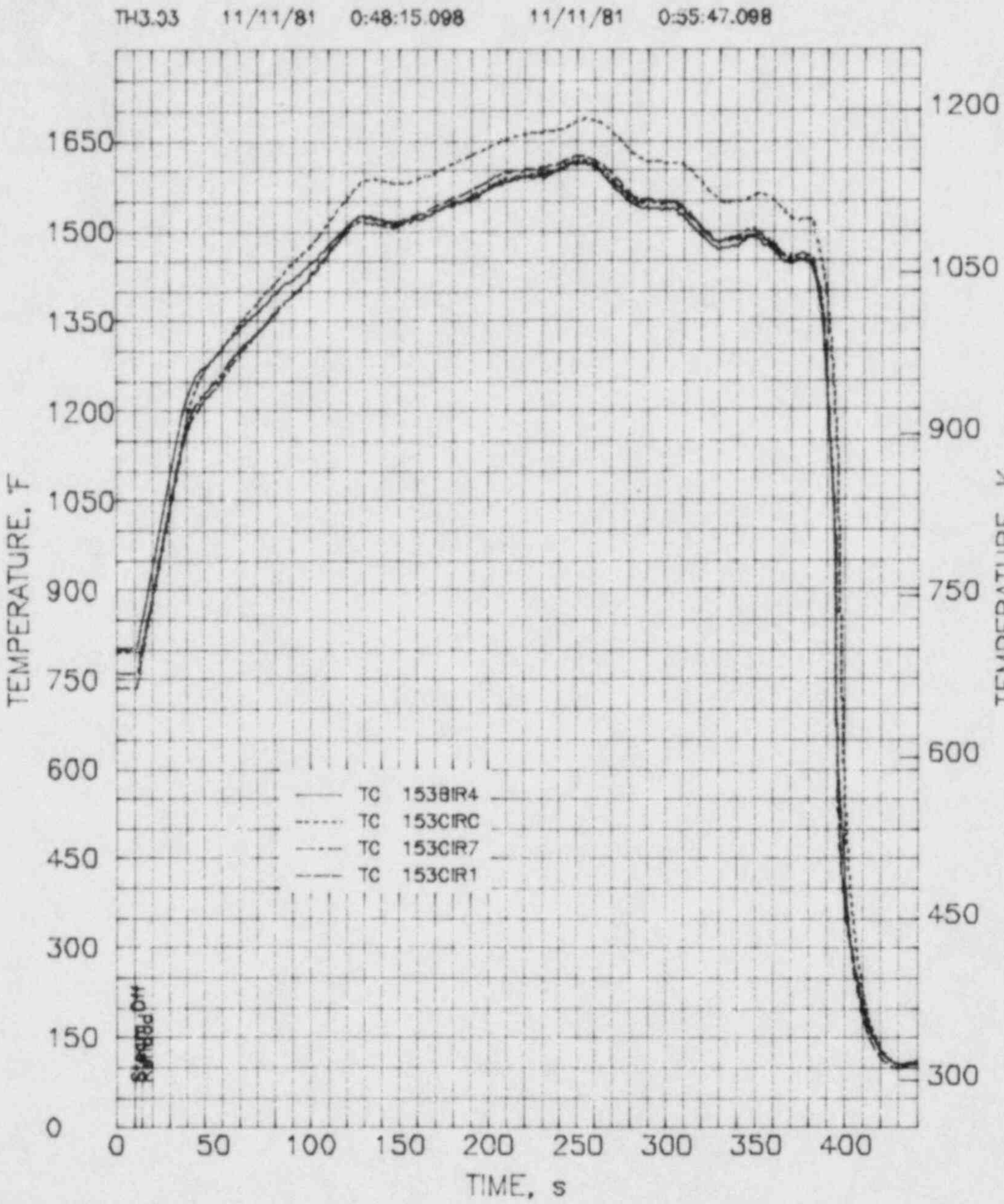


FIGURE C.6. Test Rod Center and Interior Cladding Temperature Histories at Level 15 During TH-3.03 Transient

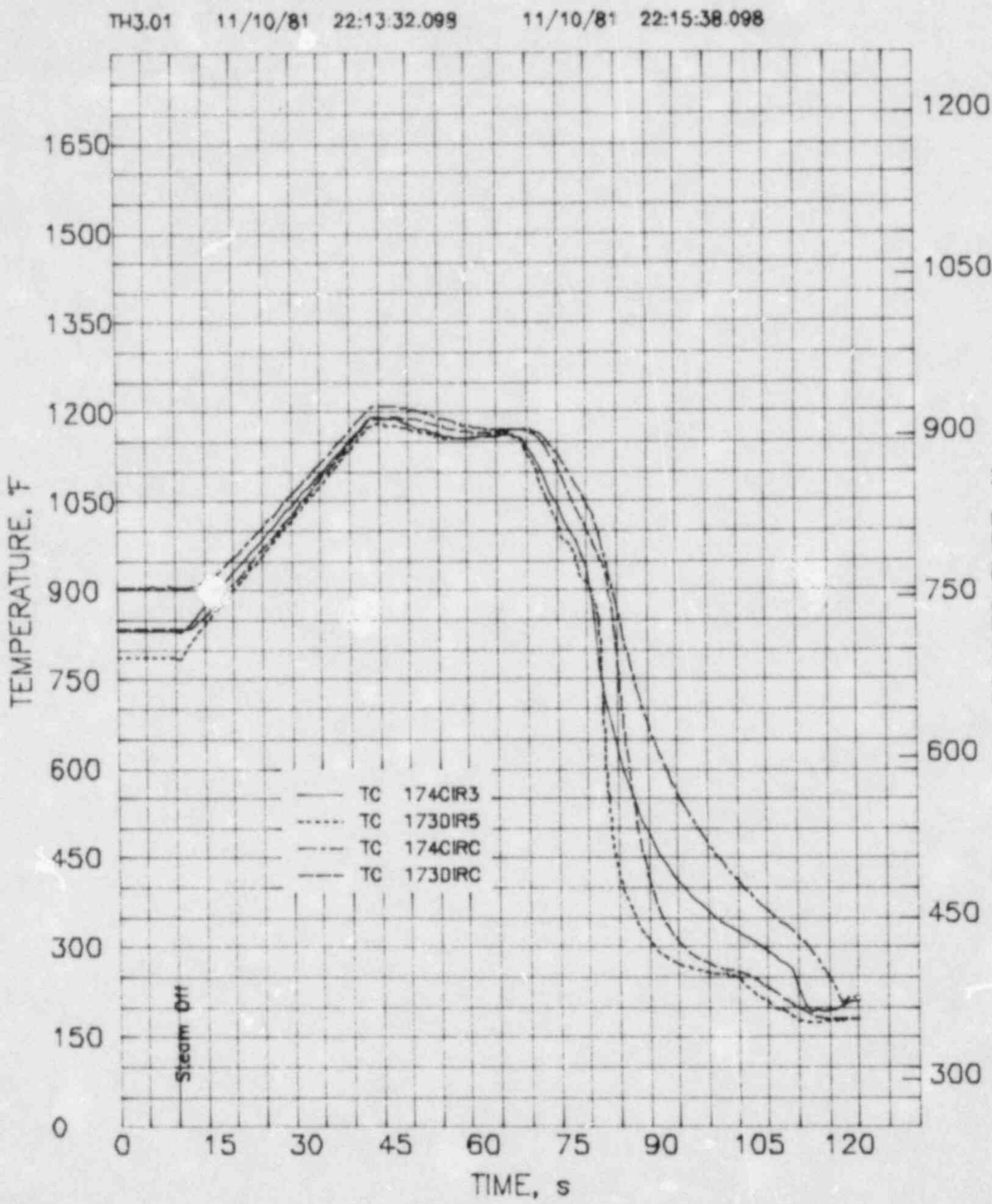


FIGURE C.7. Test Rod Center and Interior Cladding Temperature Histories at Level 17 During TH-3.01 Transient

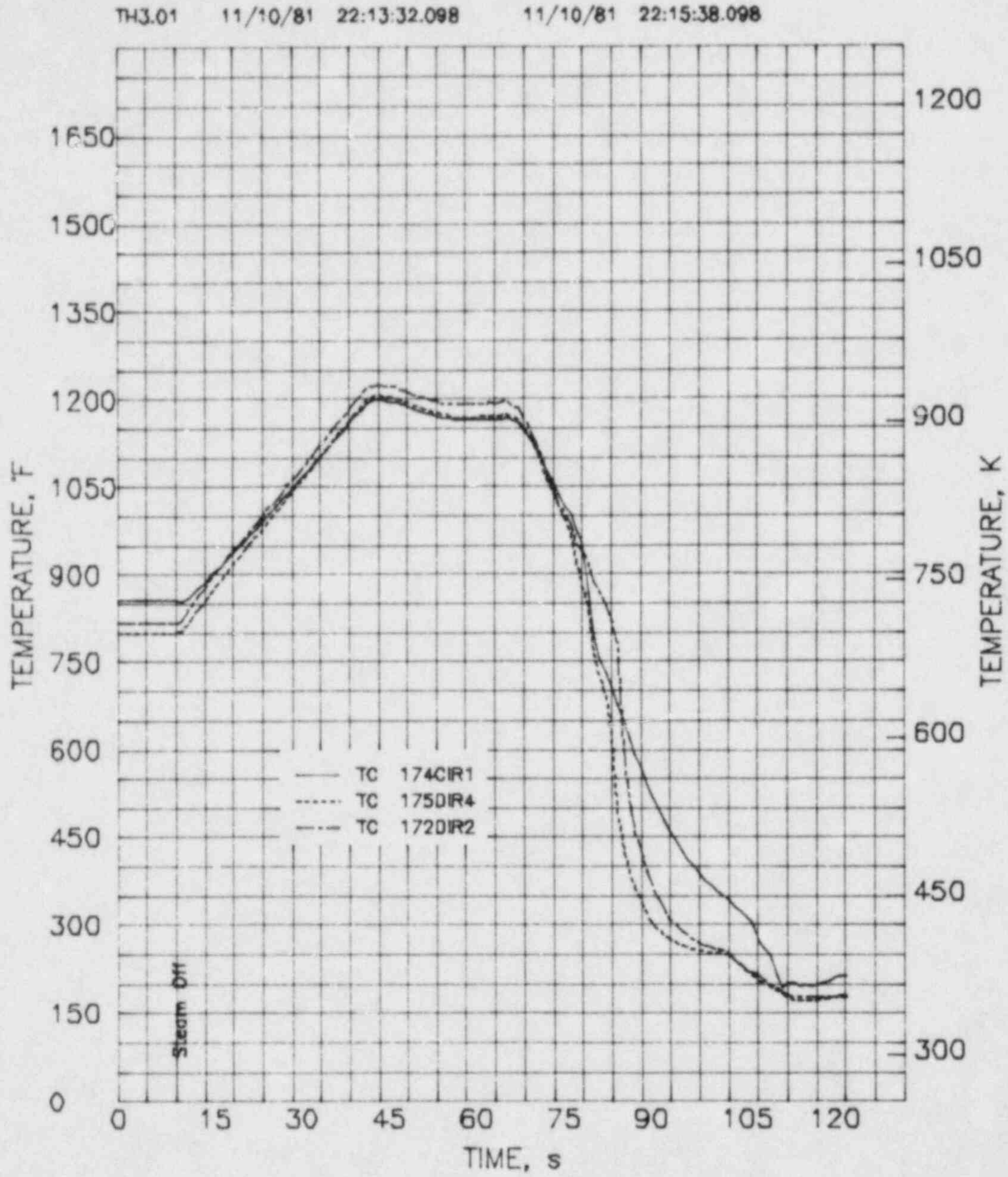


FIGURE C.7. (contd)

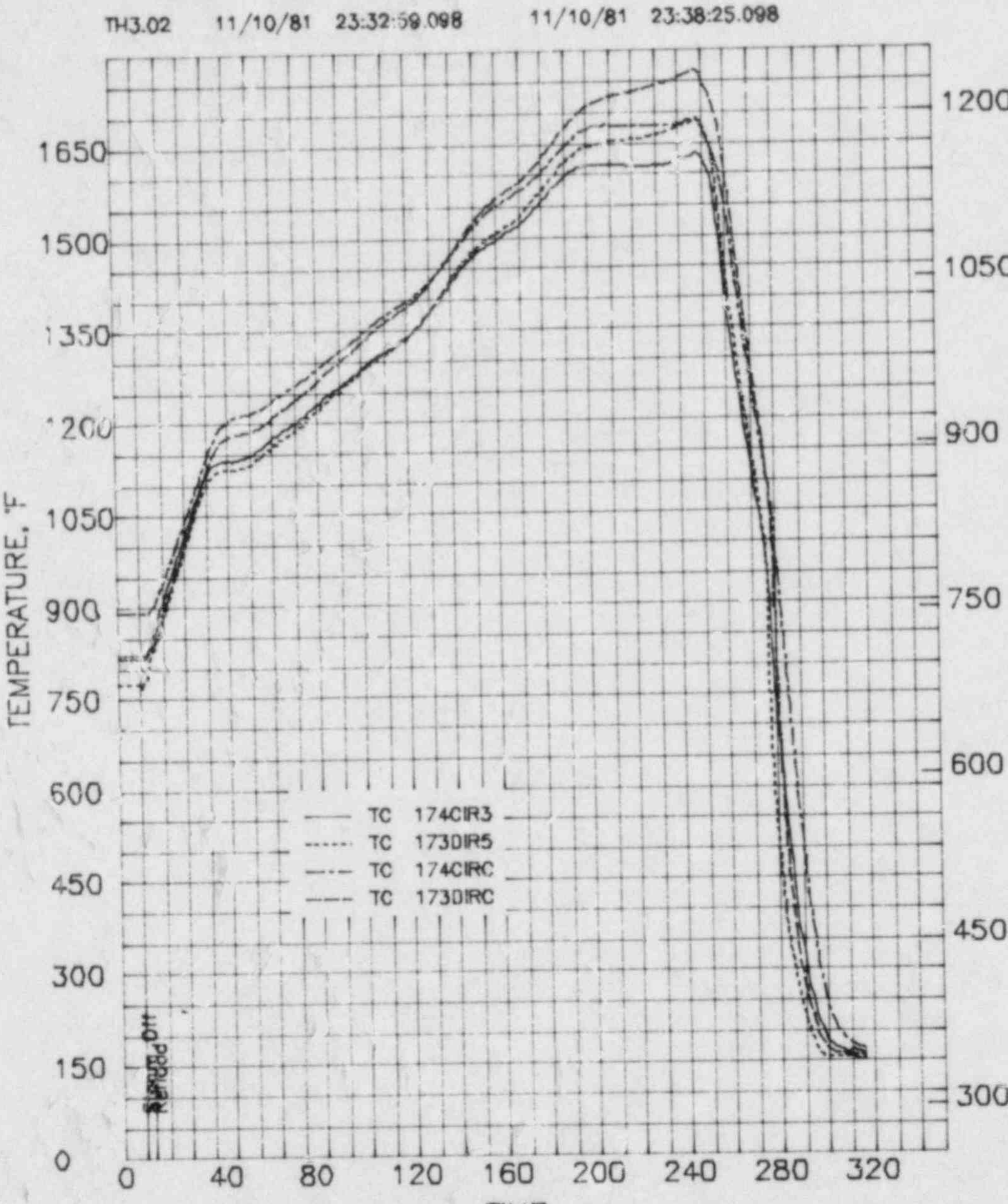


FIGURE C.8. Test Rod Center and Interior Cladding Temperature Histories at Level 17 During TH-3.02 Transient

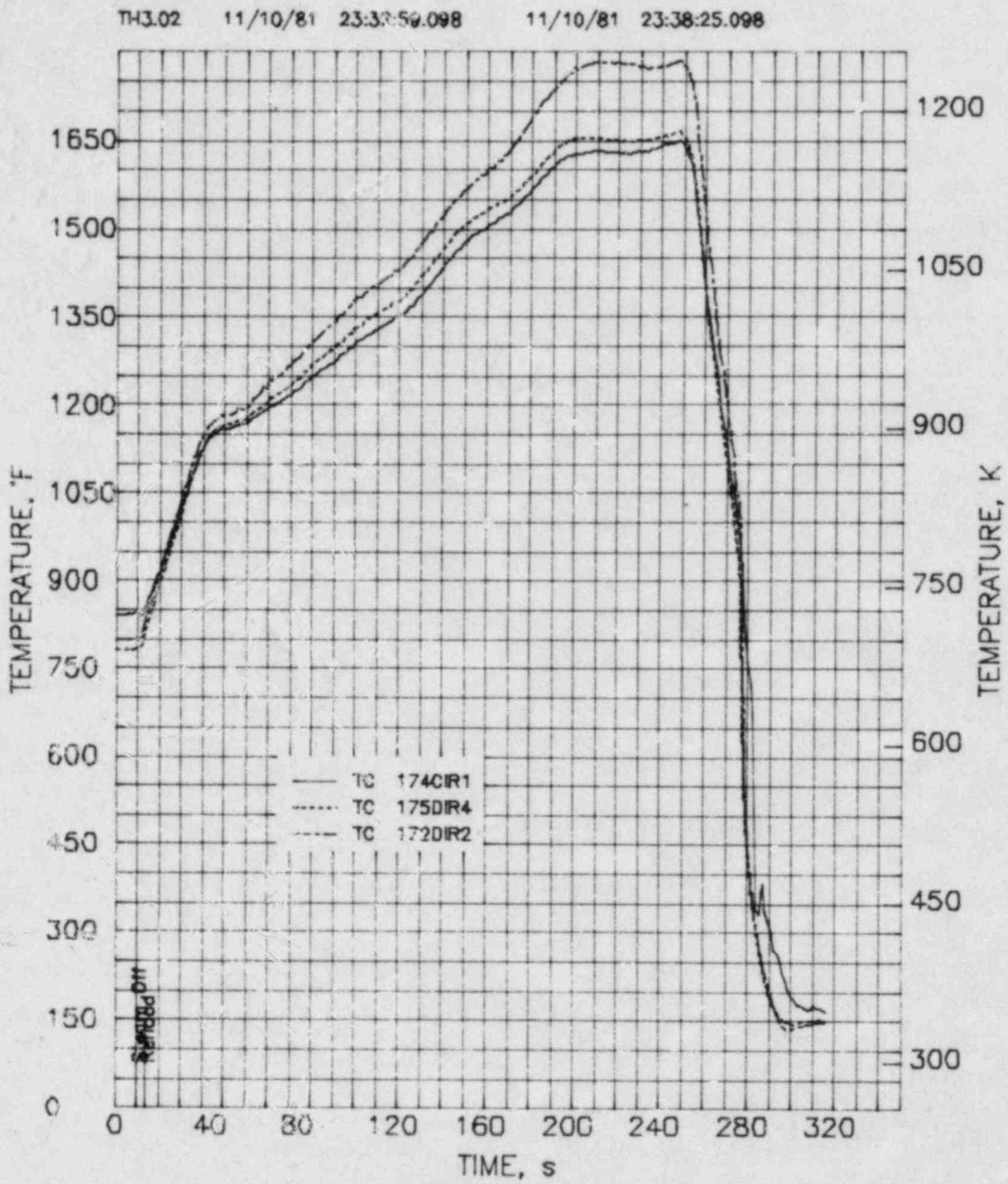


FIGURE C.8. (contd)

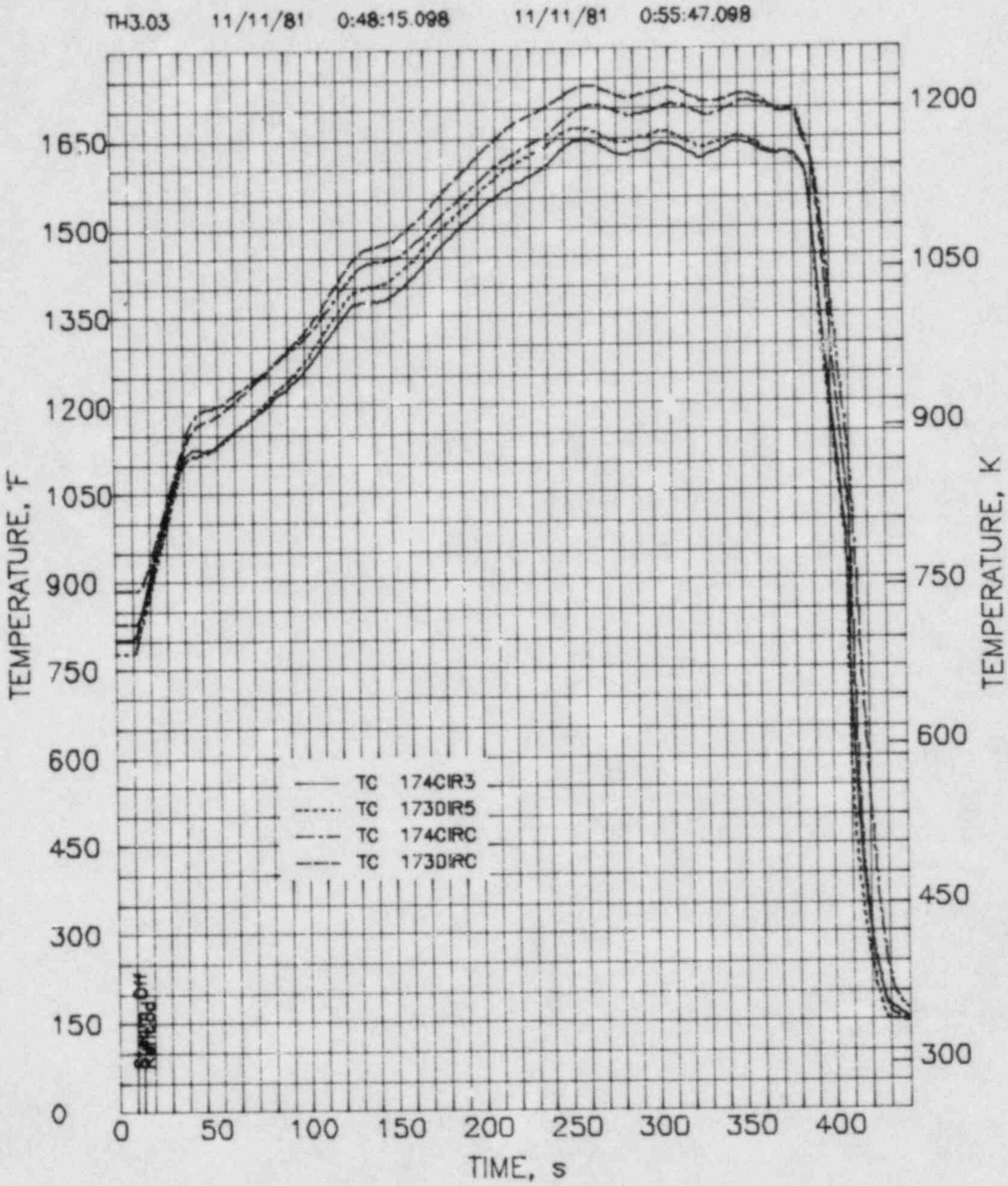


FIGURE C.9. Test Rod Center and Interior Cladding Temperature Histories at Level 17 During TH-3.03 Transient

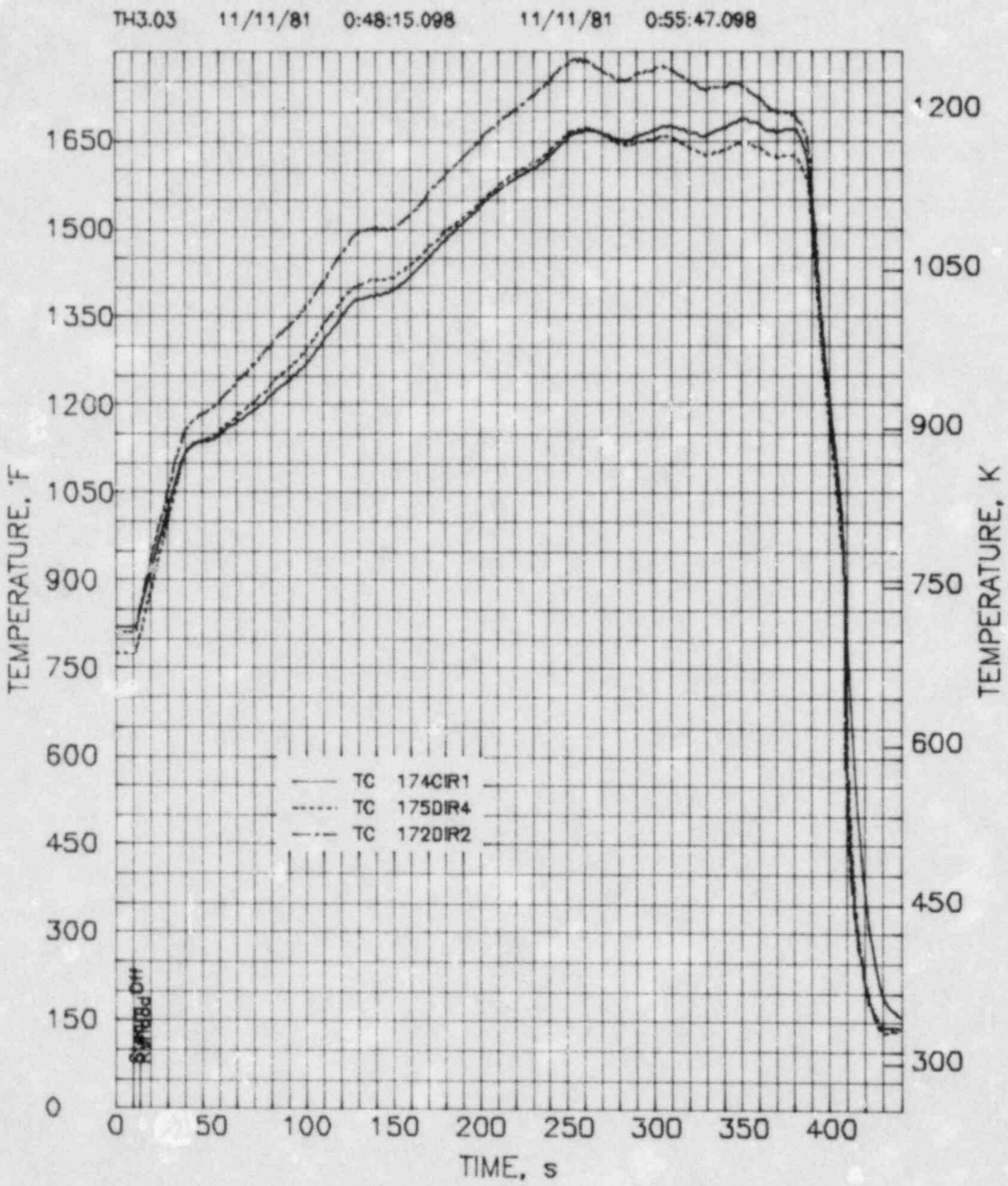


FIGURE C.9. (contd)

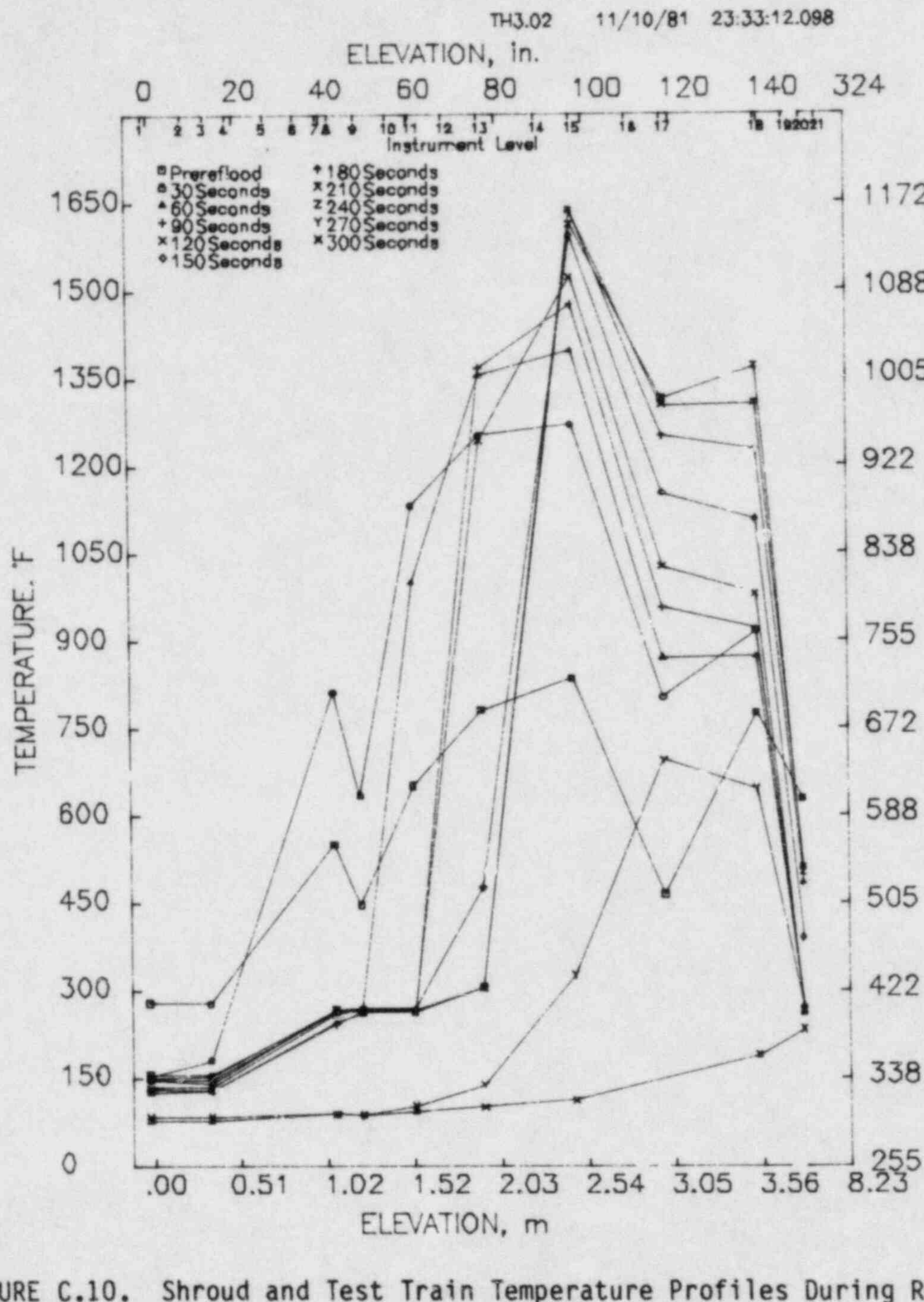


FIGURE C.10. Shroud and Test Train Temperature Profiles During Reflood at 30-s Intervals During TH-3.02 Transient

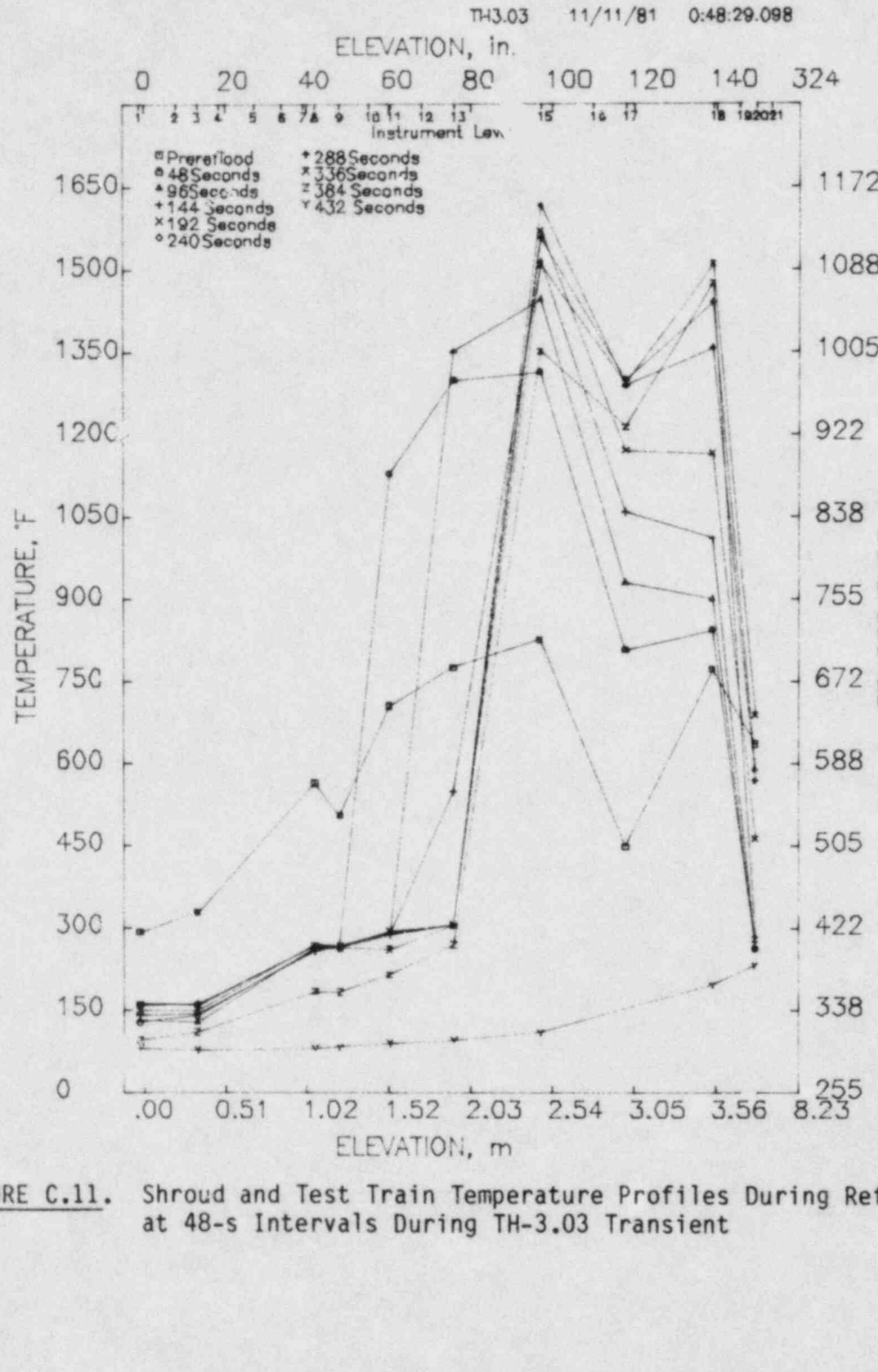


FIGURE C.11. Shroud and Test Train Temperature Profiles During Reflood at 48-s Intervals During TH-3.03 Transient

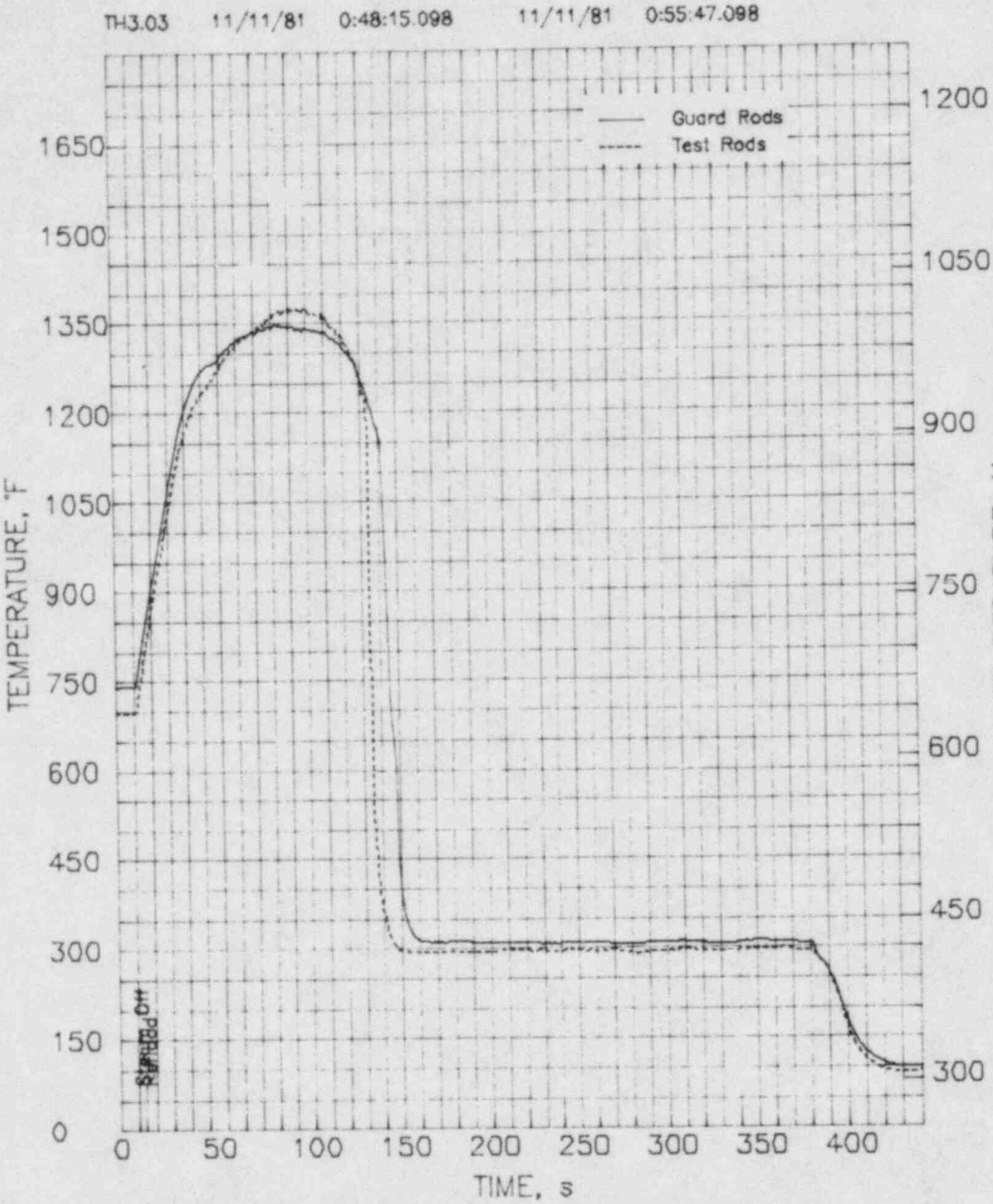


FIGURE C.12. Average Guard and Test Rod Interior Cladding Temperatures at Level 13 for TH-3.03

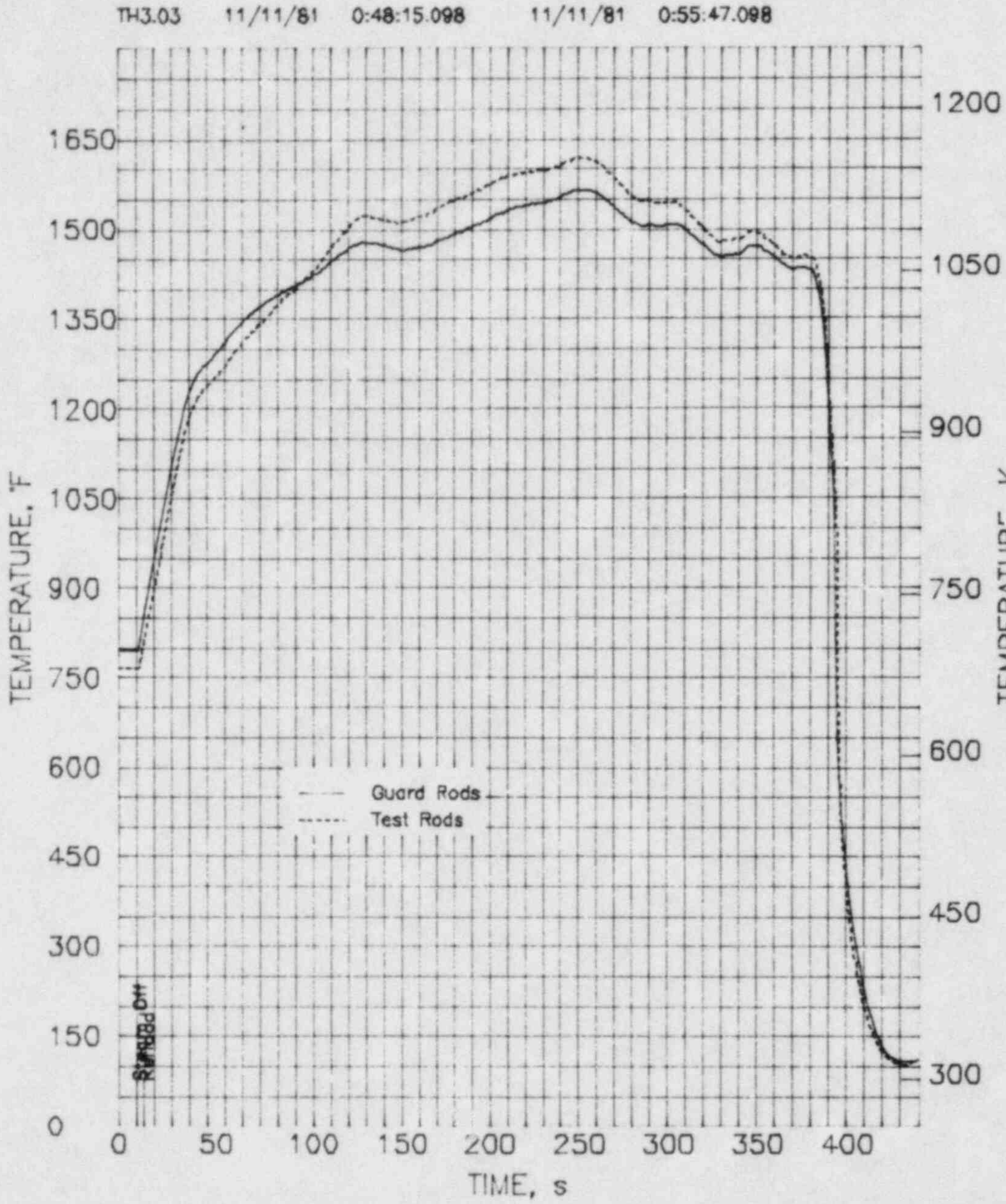


FIGURE C.13. Average Guard and Test Rod Interior Cladding Temperatures at Level 15 for TH-3.03

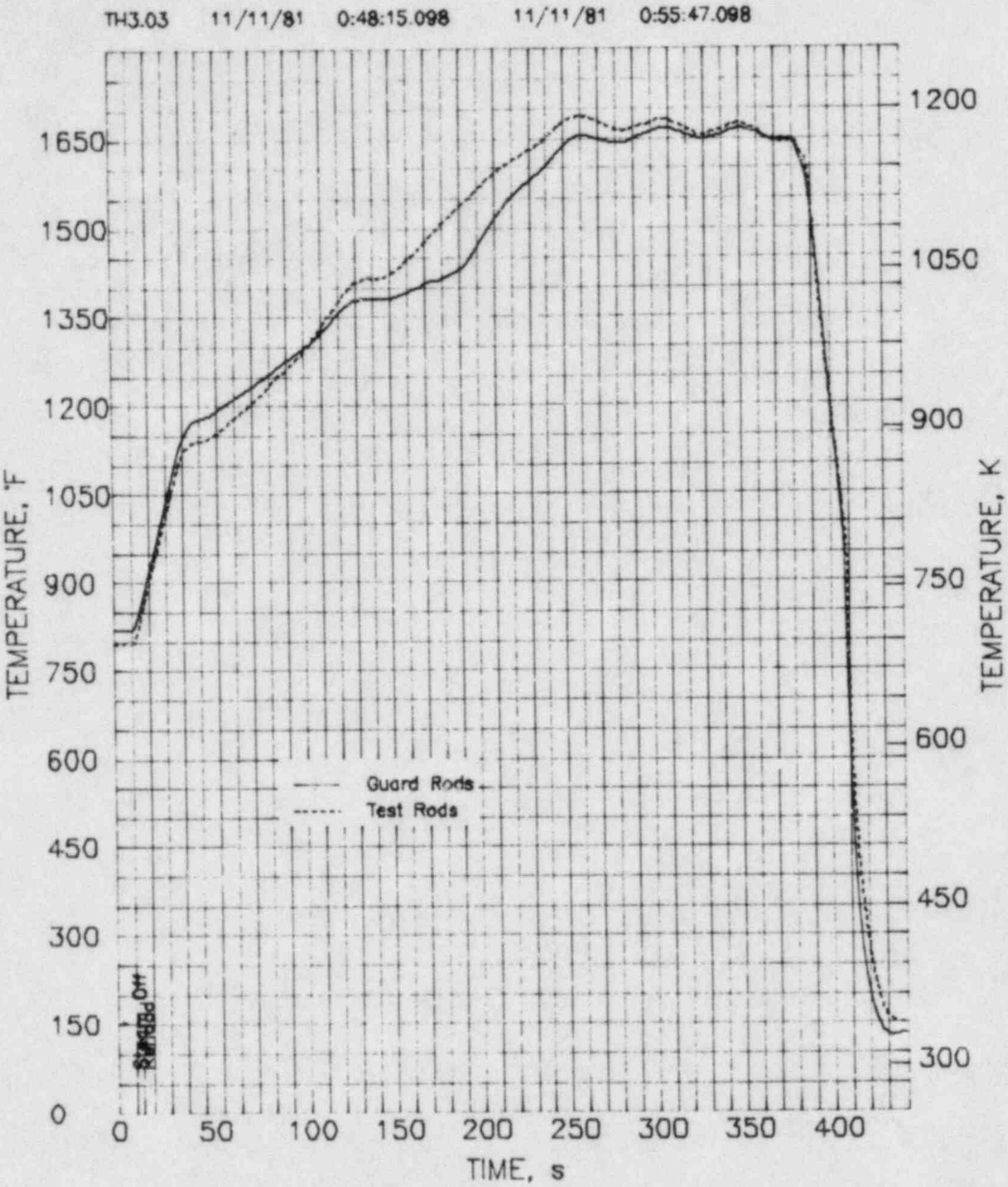


FIGURE C.14. Average Guard and Test Rod Interior Cladding Temperatures at Level 17 for TH-3.03

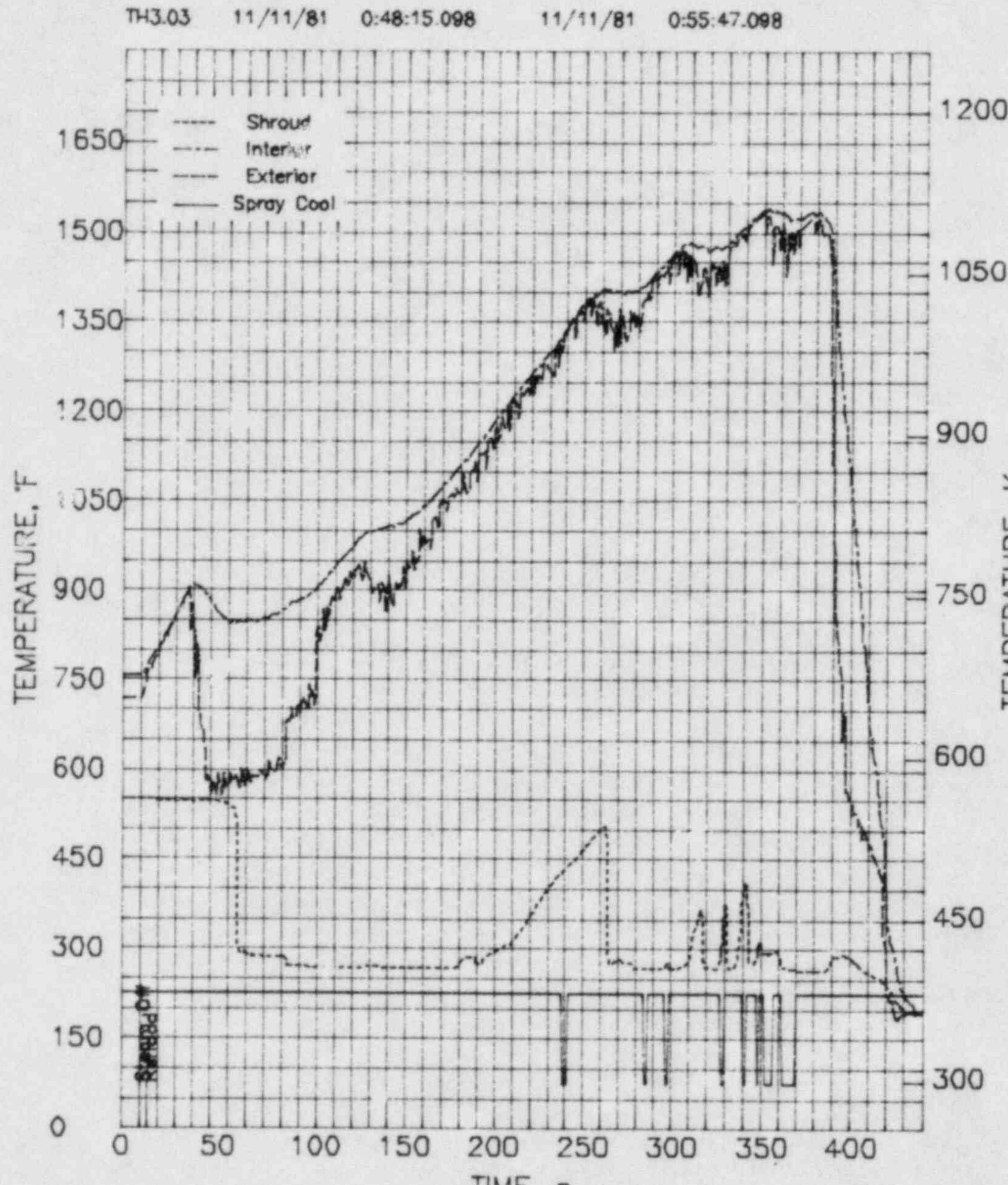


FIGURE C.15. Spray Cooling Effects on the Fuel and Shroud at Level 18 for TH-3.03

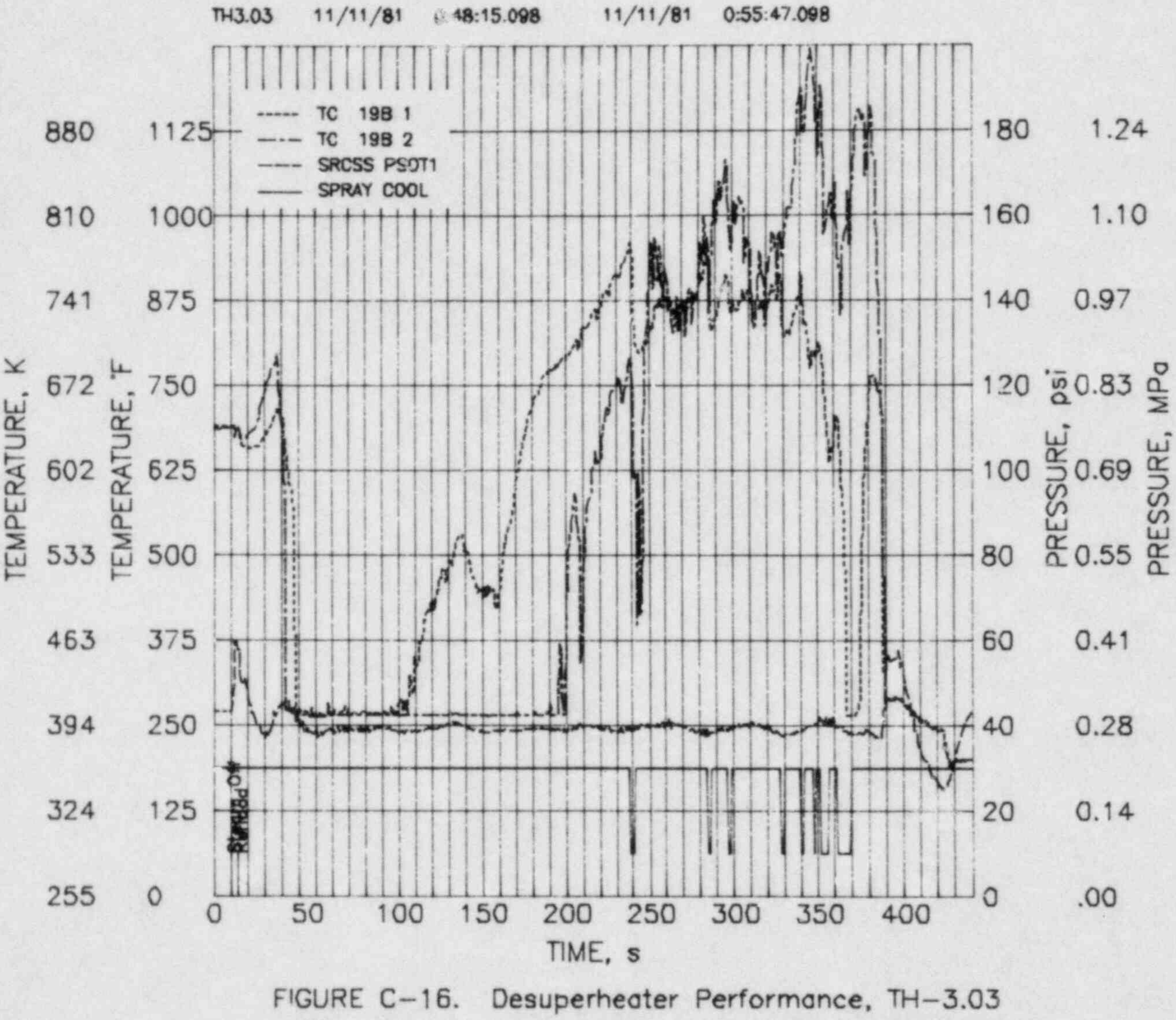


FIGURE C.16. Spray Cooling Effects on Coolant Temperature at Level 19 and Outlet Pressure

APPENDIX D

TEST COOLANT AND SHROUD TEMPERATURES

APPENDIX D

TEST COOLANT AND SHROUD TEMPERATURES

It is difficult to measure the temperature of the coolant once reflooding occurs because much of the time the coolant is in two phases: liquid and steam. Thermocouple (TC) readings of coolant flow are thus a rough indication of the steam and water temperature with the additional effects of the response time and the in situ environment. Test assembly inlet, outlet, and steam probe TCs at Levels 12, 14, and 16 are shown in Figures D.1 through D.3

Shroud temperatures at Levels 10, 13, 15, and 17 are shown in Figures D.4 through D.15. It is important to know the shroud temperature to insure that the shroud does not overheat; if it did, it could warp and thereby delay removal of the test assembly from the test site.

Shroud temperatures tend to be cooler than coolant temperatures, which may be an indication that heat is transferred radially outward, allowing the shroud TCs to quench before the coolant TCs. There may be a tendency for upper level shroud and coolant TCs to be hotter, which might be caused by less quenching by entrained liquid at the higher levels.

The remainder of this appendix consists of the following graphical data:

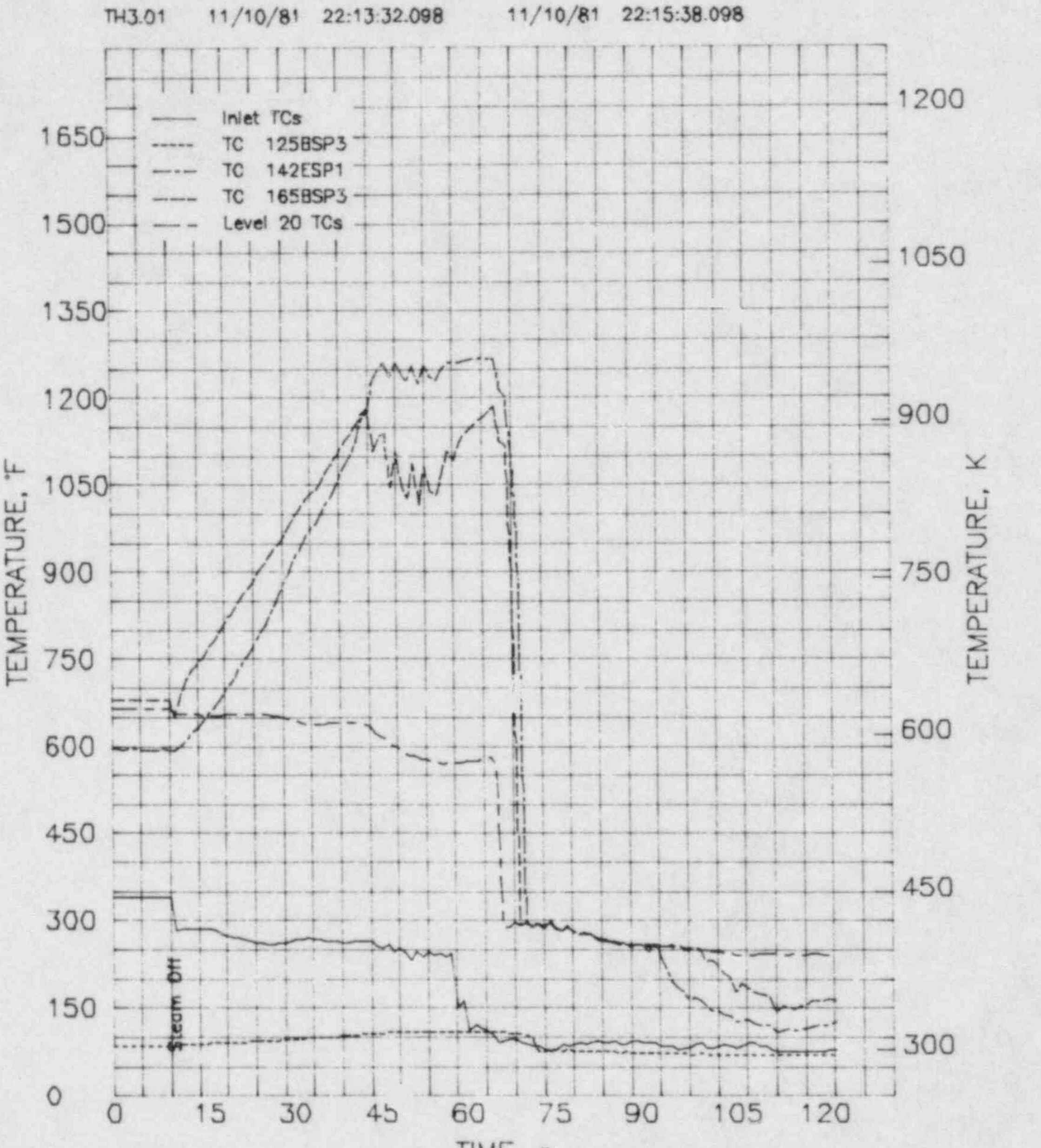


FIGURE D.1. Steam Probe, Inlet, and Outlet TC Temperature Histories for Levels 1 to 20 During TH-3.01 Transient

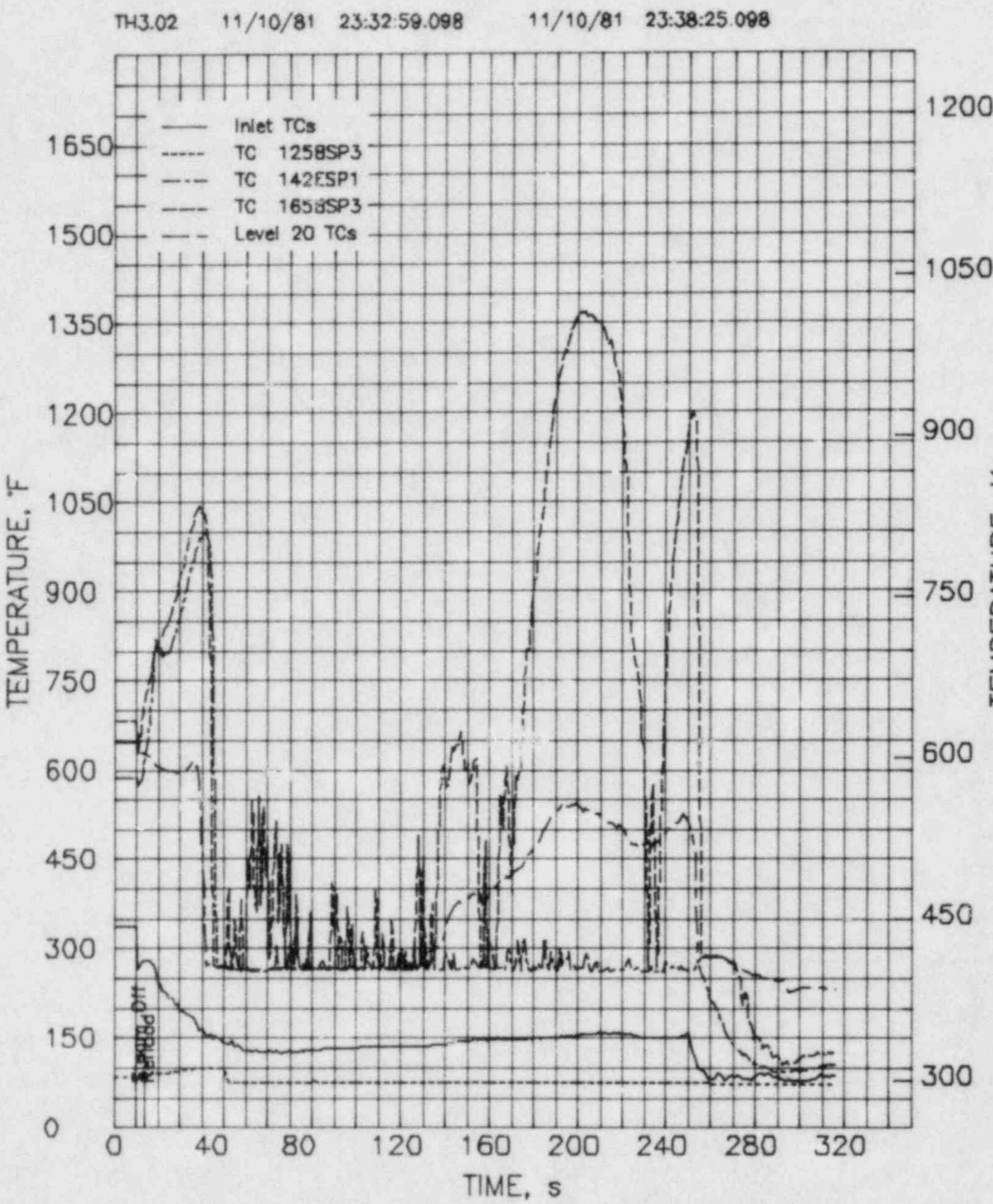


FIGURE D.2. Steam Probe, Inlet, and Outlet TC Temperature Histories for Levels 1 to 20 During TH-3.02 Transient

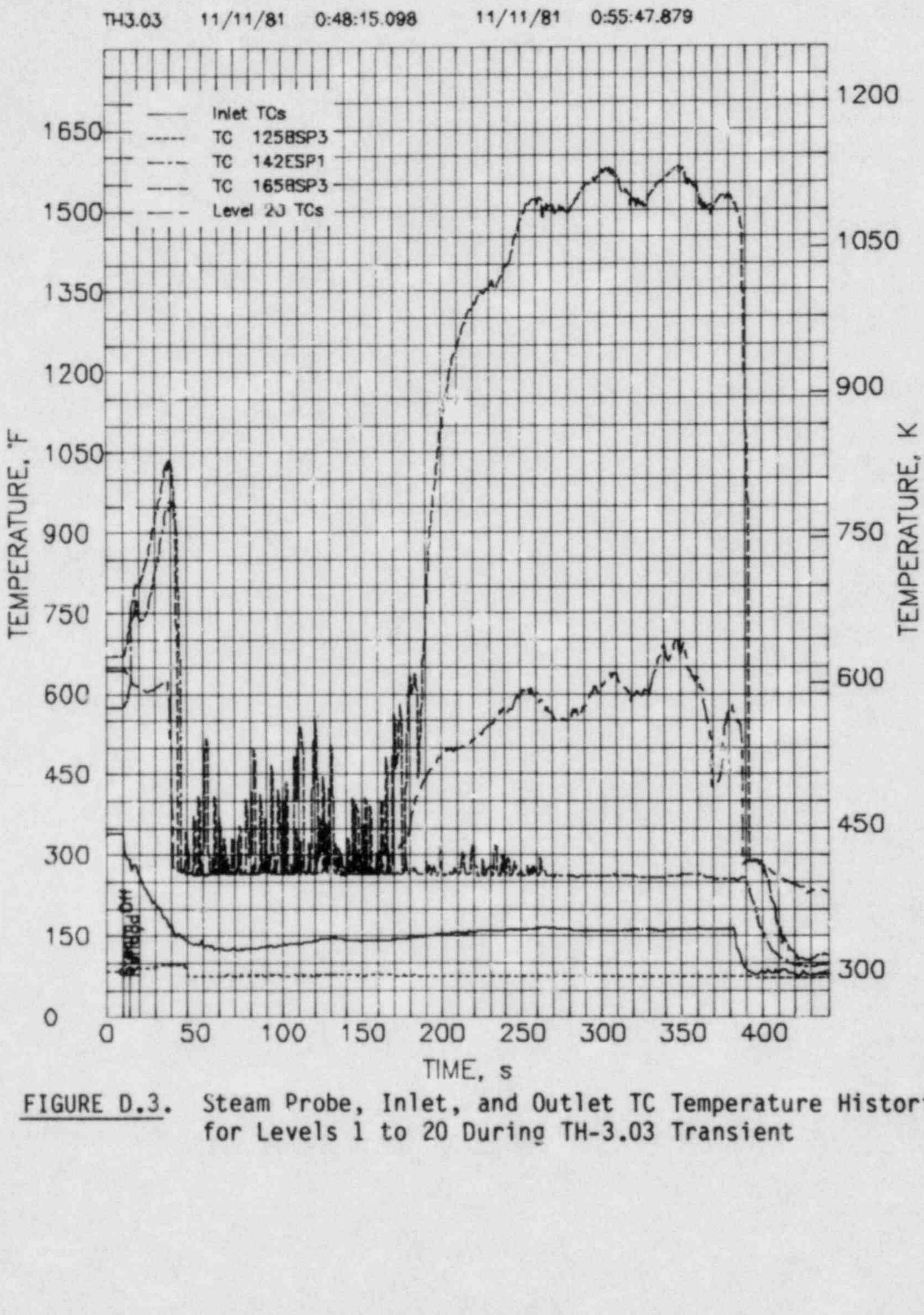


FIGURE D.3. Steam Probe, Inlet, and Outlet TC Temperature Histories for Levels 1 to 20 During TH-3.03 Transient

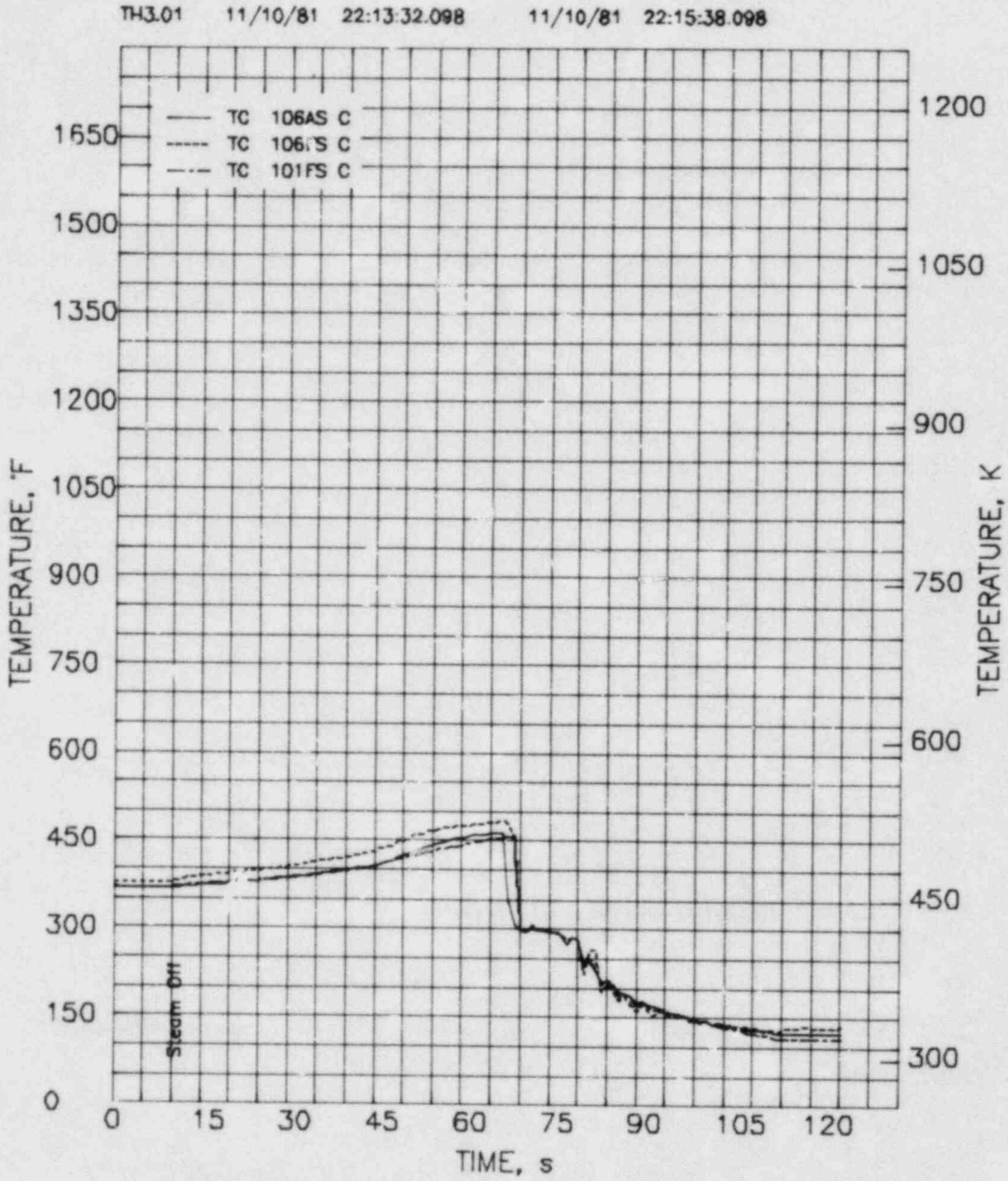


FIGURE D.4. Shroud Temperature Histories at Level 10 During TH-3.01 Transient

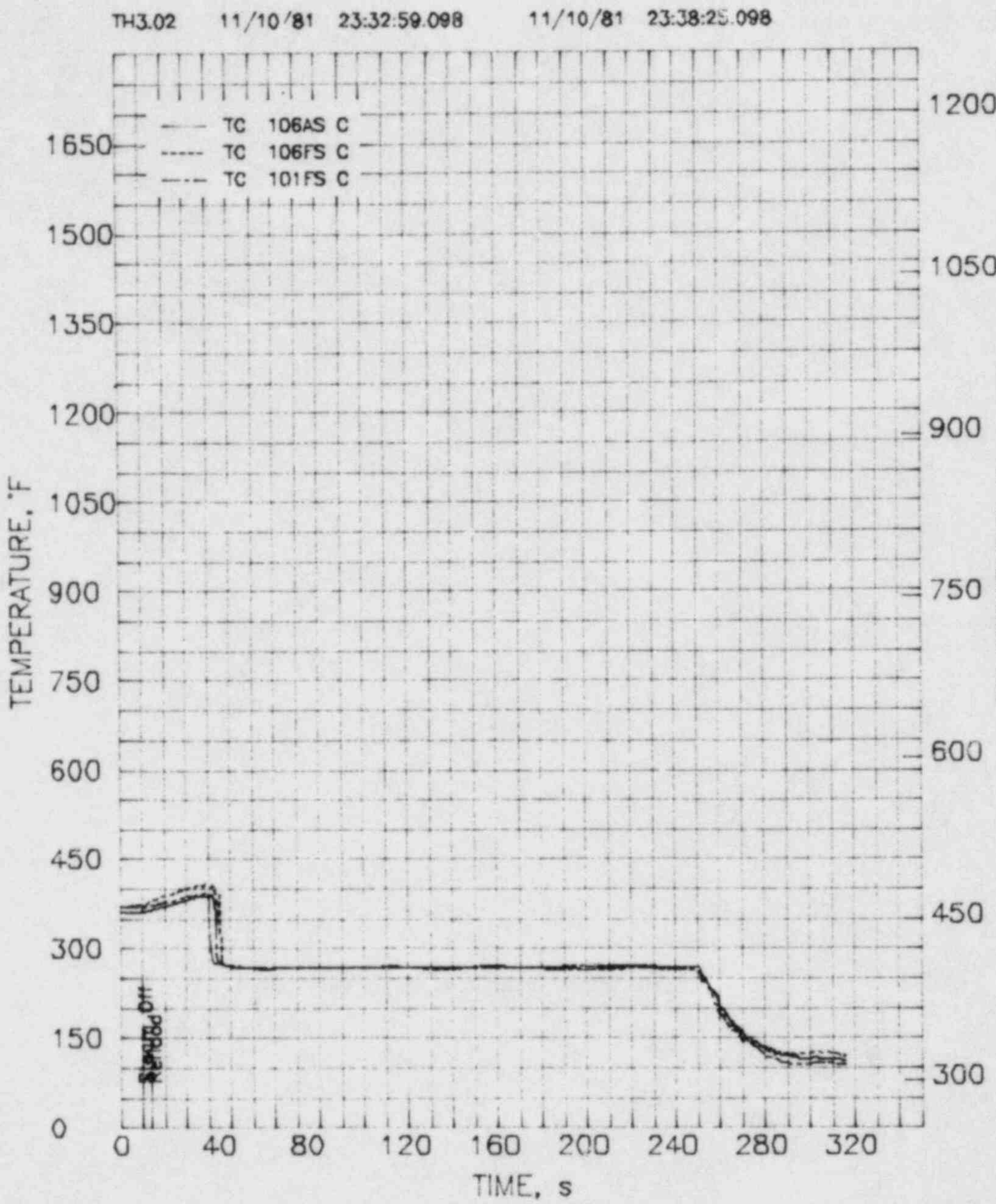


FIGURE D.5. Shroud Temperature Histories at Level 10 During TH-3.02 Transient

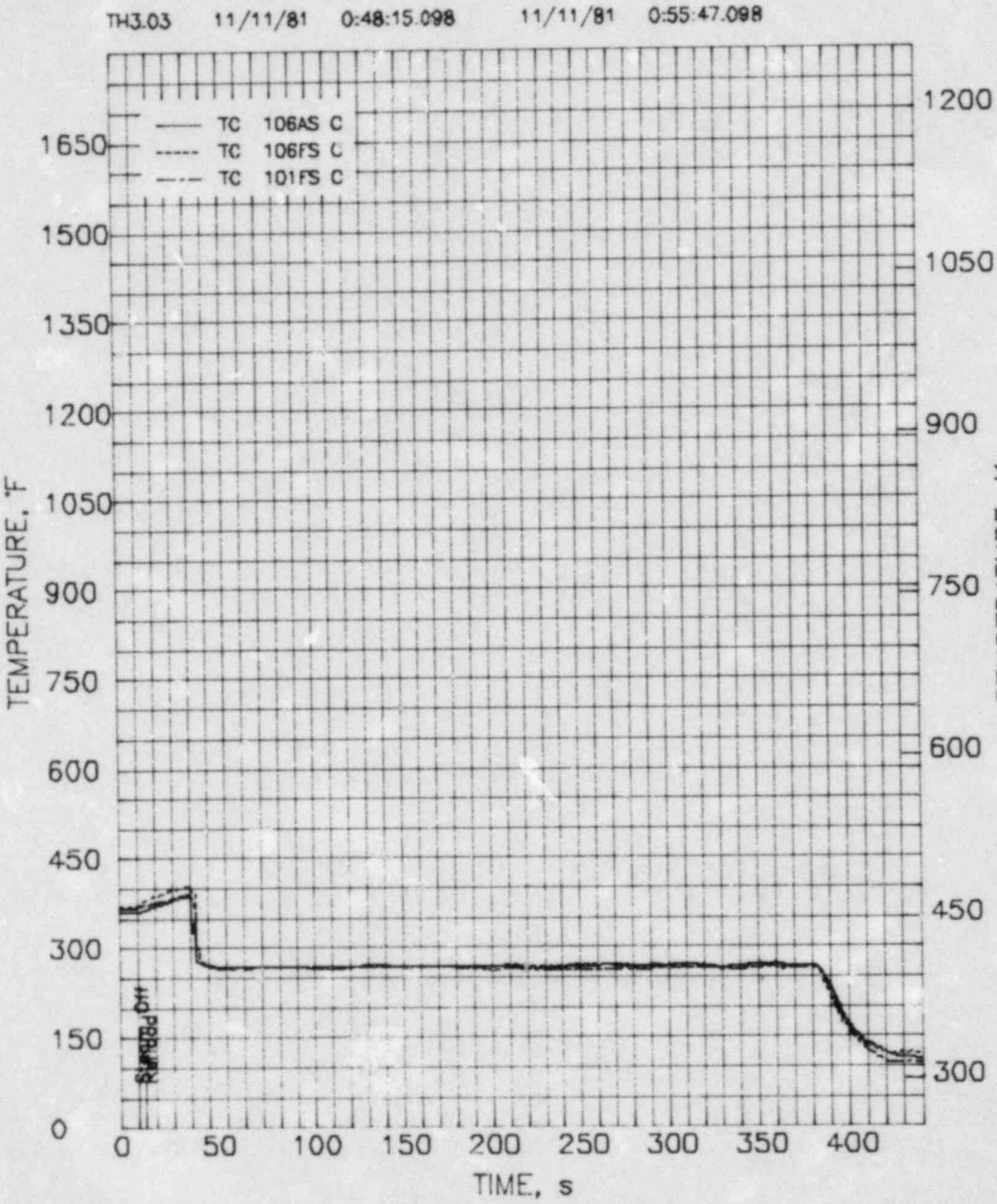


FIGURE D.6. Shroud Temperature Histories at Level 10 During TH-3.03 Transient

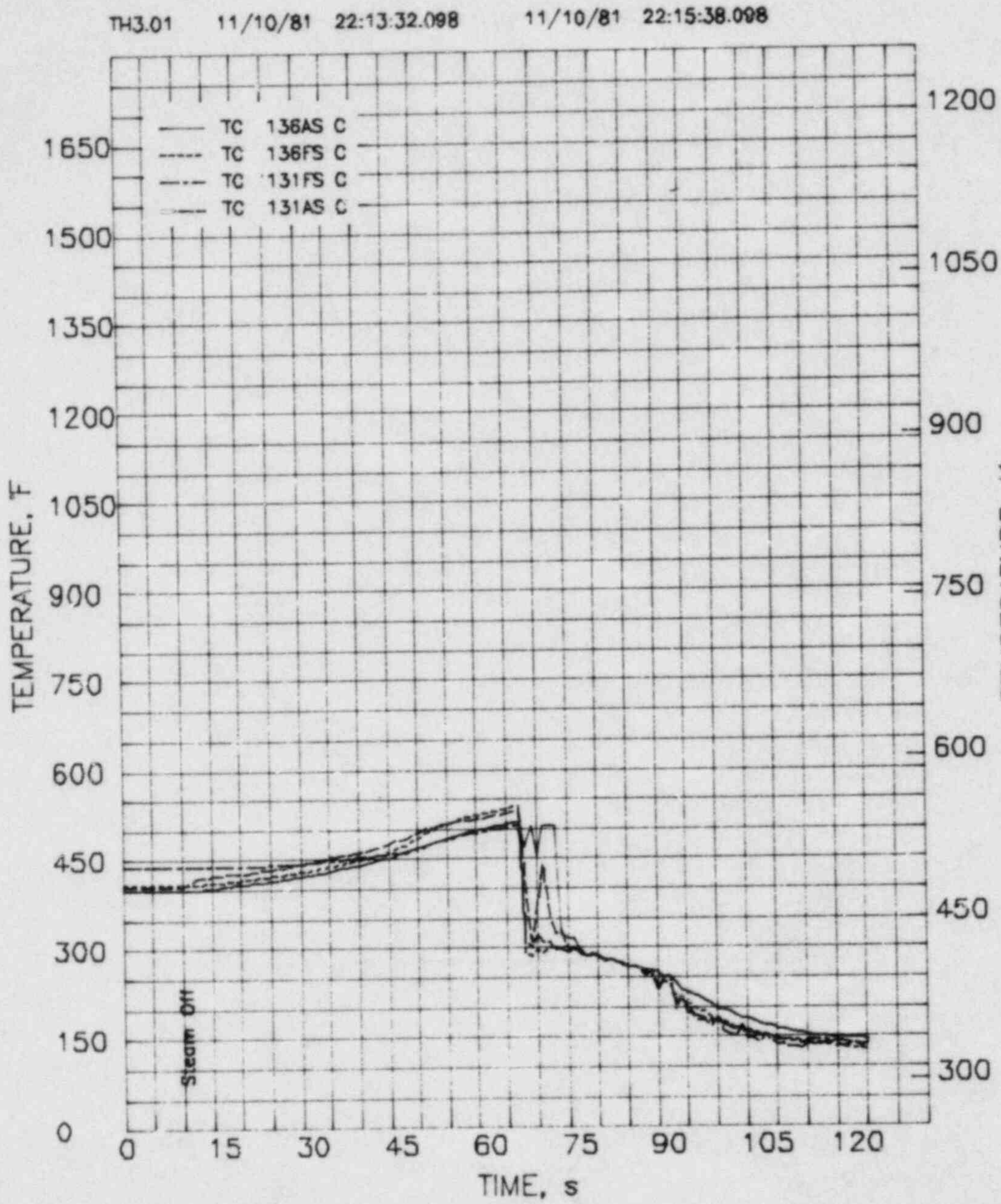


FIGURE D.7. Shroud Temperature Histories at Level 13 During TH-3.01 Transient

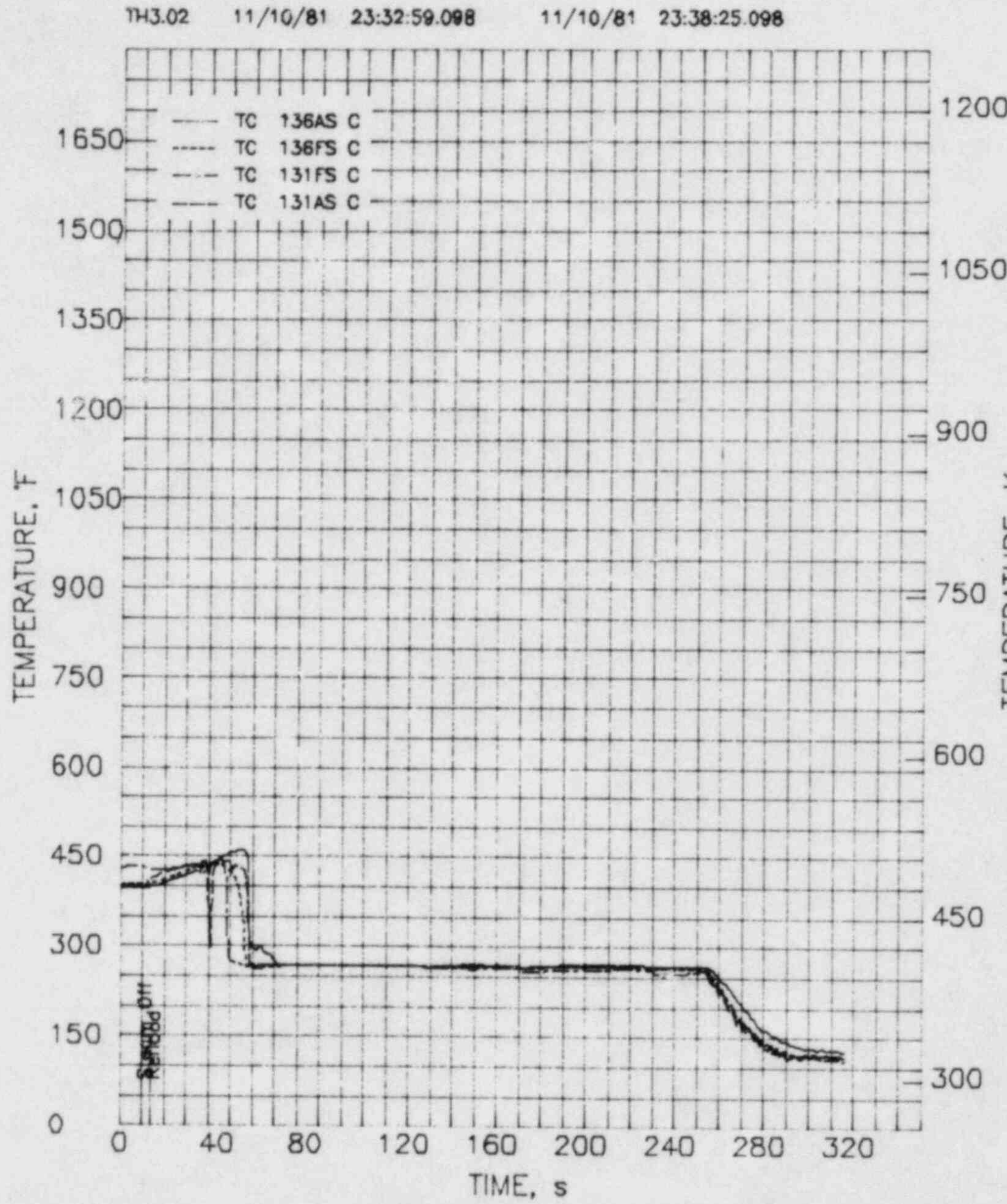


FIGURE D.8. Shroud Temperature Histories at Level 13 During TH-3.02 Transient

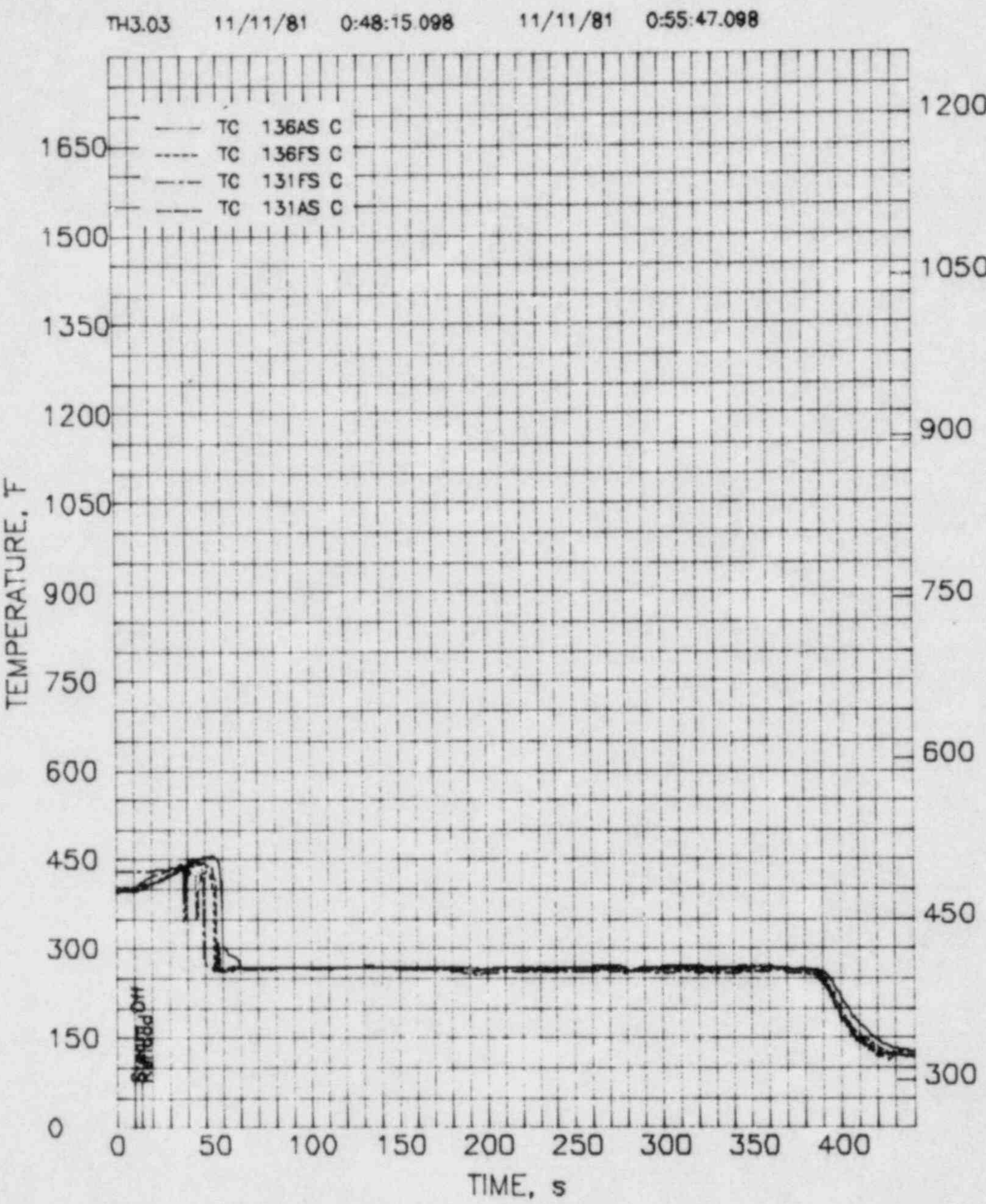


FIGURE D.9. Shroud Temperature Histories at Level 13 During TH-3.03 Transient

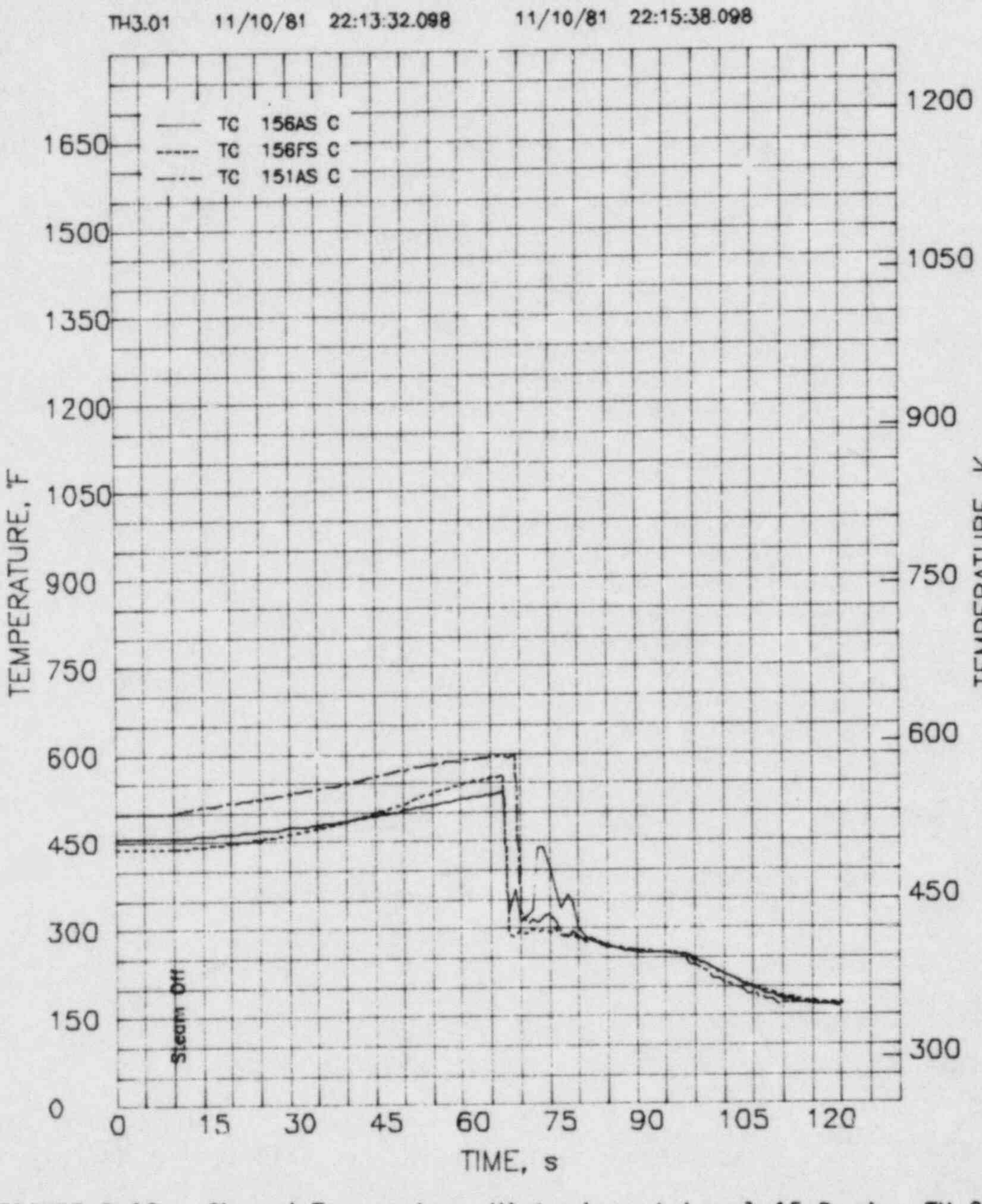


FIGURE D.10. Shroud Temperature Histories at Level 15 During TH-3.01 Transient

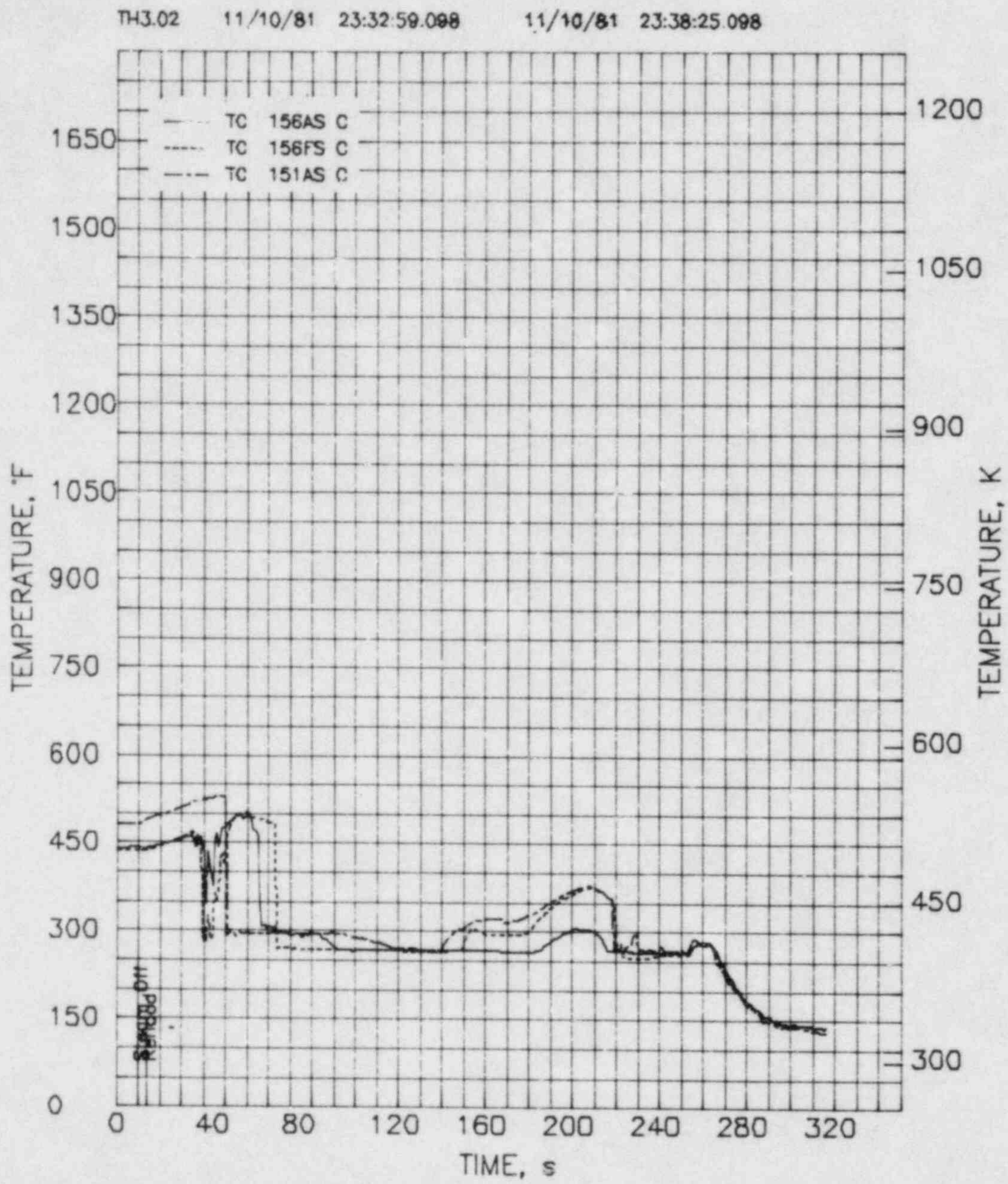


FIGURE D.11. Shroud Temperature Histories at Level 15 During TH-3.02 Transient

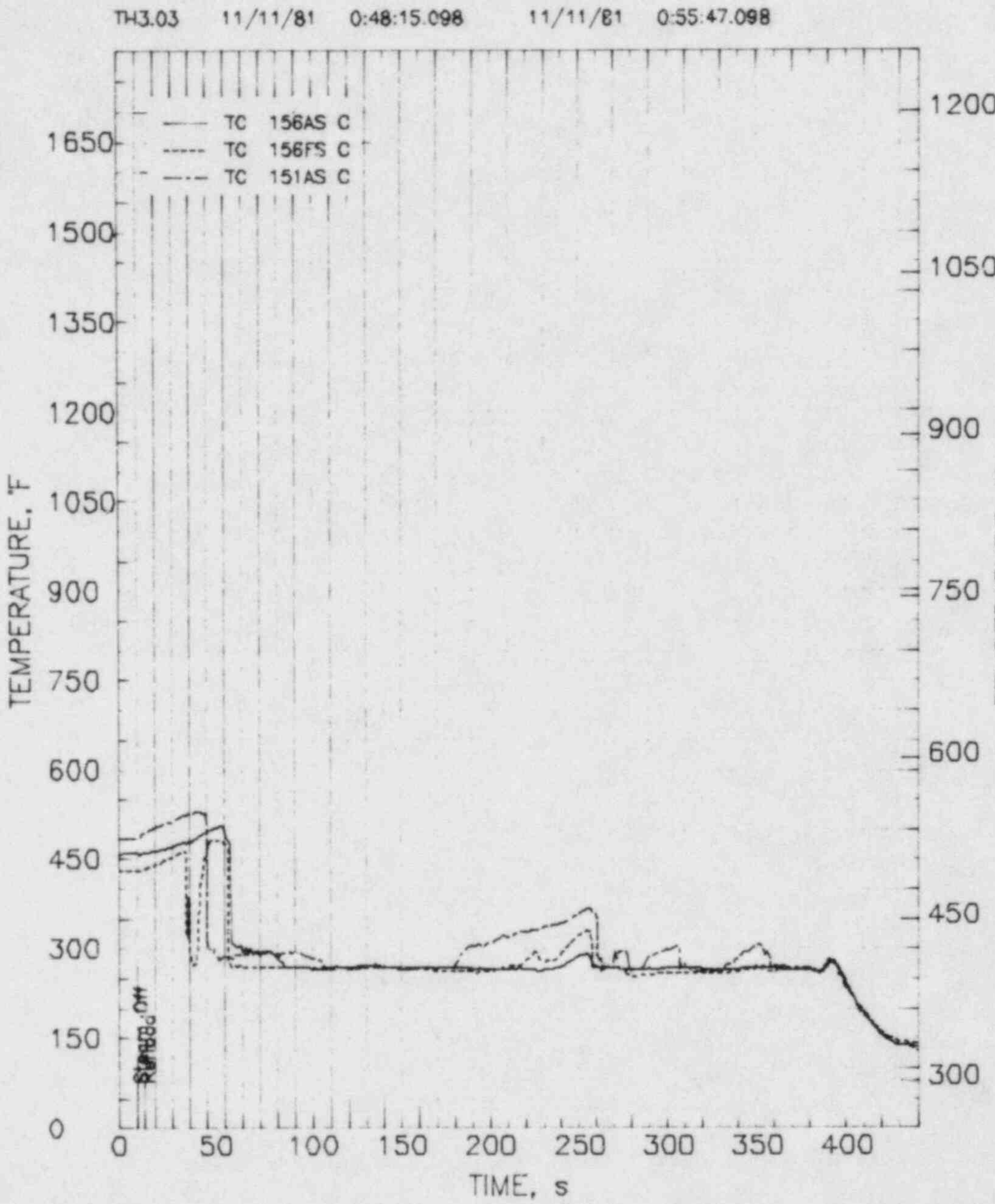


FIGURE D.12. Shroud Temperature Histories at Level 15 During TH-3.03 Transient

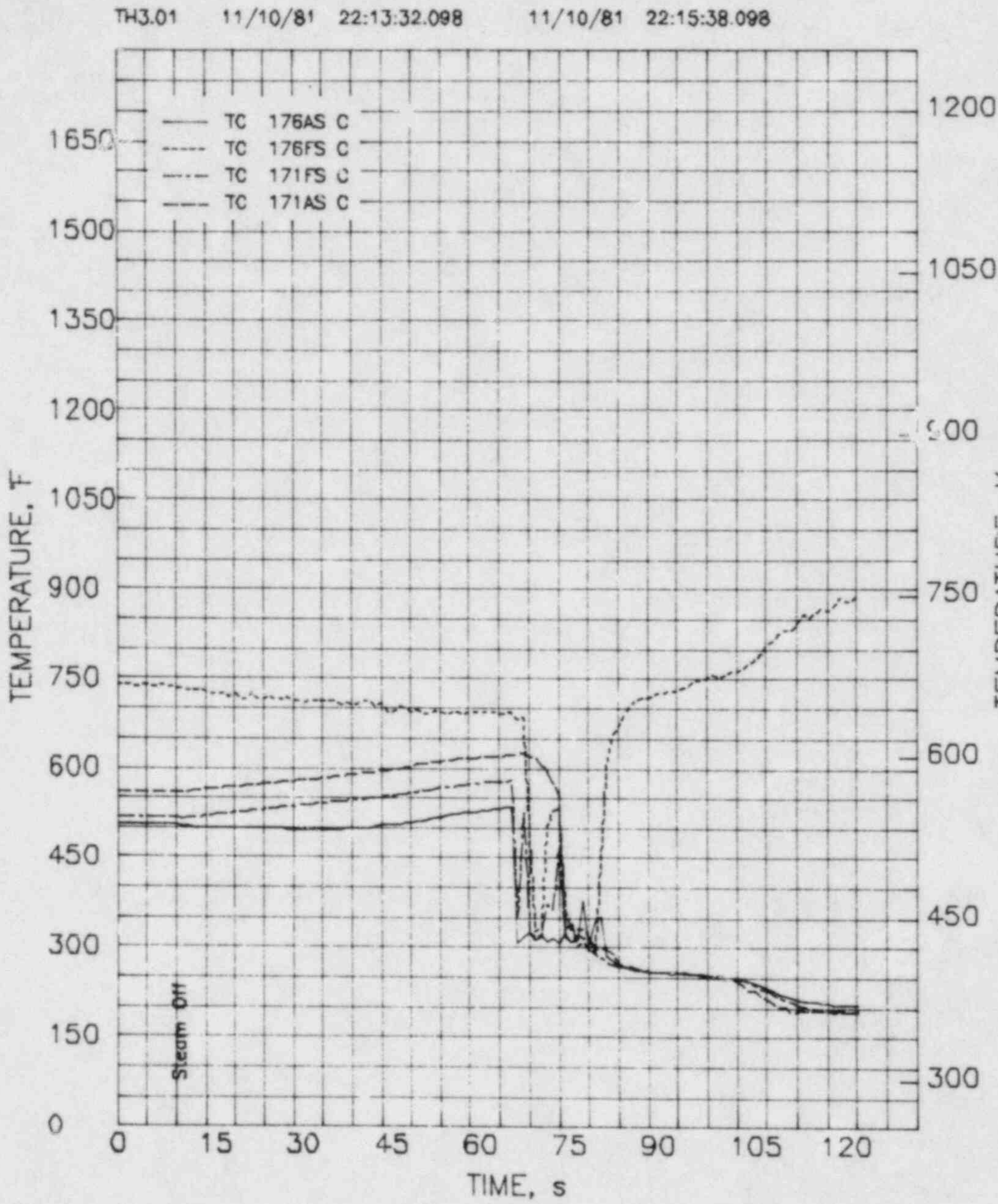


FIGURE D.13. Shroud Temperature Histories at Level 17 During TH-3.01 Transient

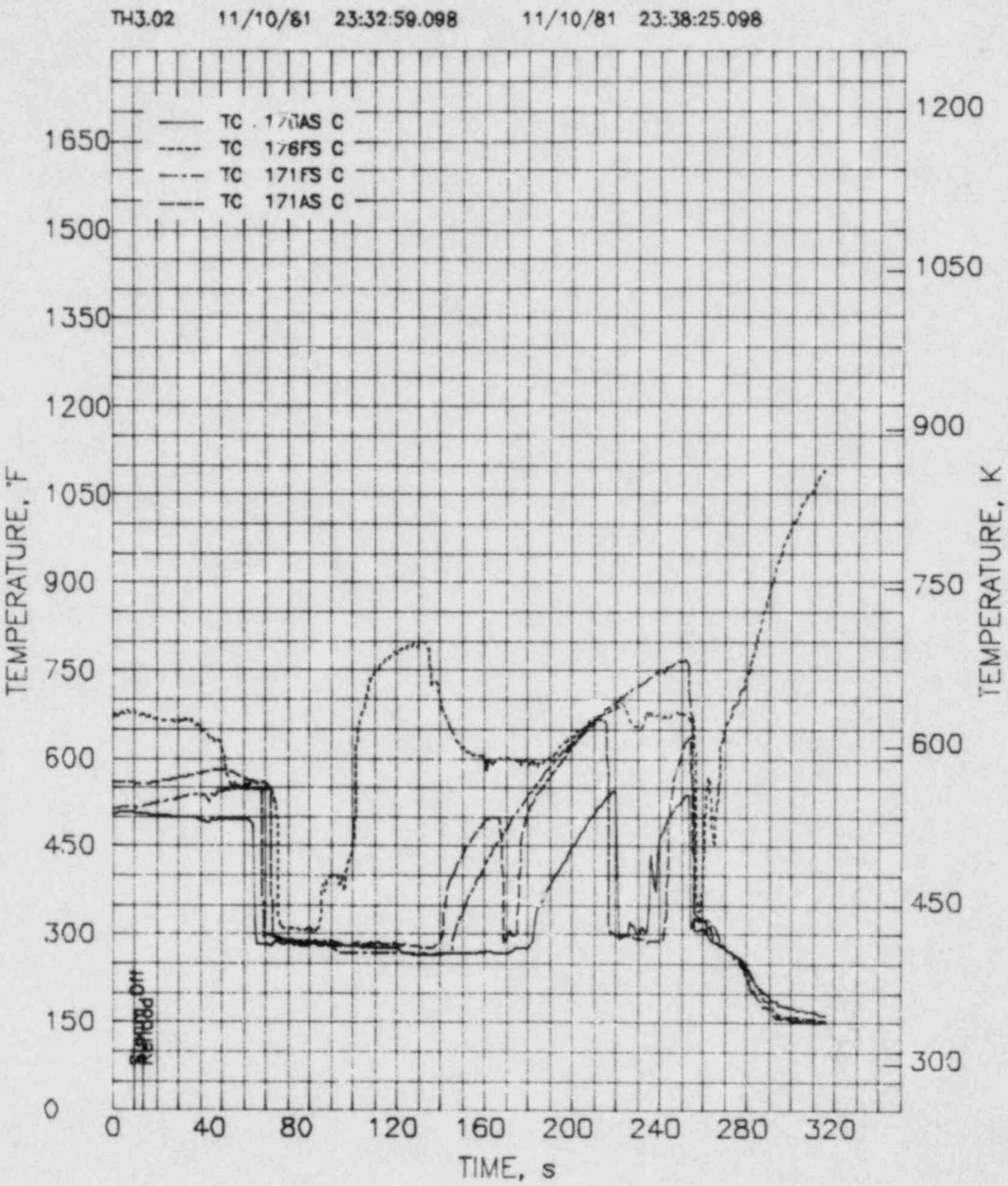


FIGURE D.14. Shroud Temperature Histories at Level 17 During TH-3.02 Transient

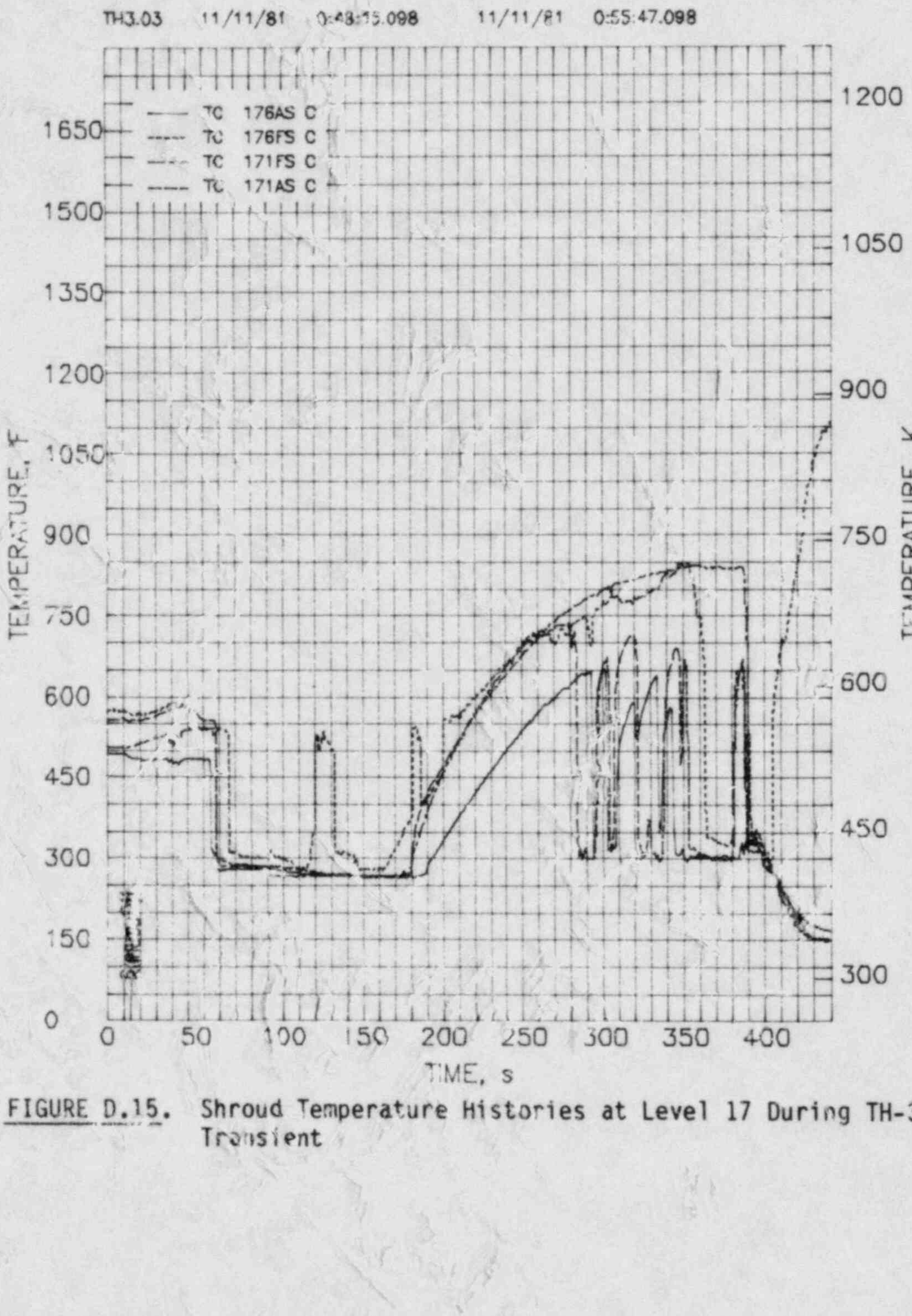


FIGURE D.15. Shroud Temperature Histories at Level 17 During TH-3.03 Transient

APPENDIX E

NEUTRON FLUX

APPENDIX E

NEUTRON FLUX

Neutron flux was measured during preconditioning and during pretransient and transient tests.

PRECONDITIONING AND PRETRANSIENT NEUTRON FLUX

Neutron flux was measured inside the test assembly on the wall of the shroud and elsewhere in the National Research Universal (NRU) reactor. The internal flux was measured by self-powered neutron detectors (SPNDs) in opposite corners of the stainless steel shroud and at several elevations, ranging from 0.337 to 3.538 m (13.3 to 139.3 in.) above the bottom of the fuel column. These SPNDs provided both a measure of the radial neutron flux gradient and the neutron flux distribution over the vertical axis of the test assembly. Because neutron flux was measured in opposite corners of the test assembly, radial neutron flux gradients are evident. Figure E.1 presents the axial distribution of neutron flux measured in each corner.

Flux detector rods in the NRU reactor provided an independent measure of the neutron flux axial distribution at two other locations. In Figure E.2, neutron flux measured inside and outside the test assembly is compared. The magnitude of the difference is predominantly due to the addition of structural material (stainless steel) and absorber material (light water) in a heavy-water-moderated reactor. However, the normalized neutron flux distributions are quite comparable, especially when platinum flux detector rods are corrected for burnup.

The only obvious difference among these flux distributions is the location of the peak flux measured inside the test assembly--fuel rods 3.66 m (12 ft) long--compared with the peak measured in the NRU reactor--fuel rods 2.74 m (9 ft) long. The core midplanes are offset by 0.305 m (1 ft).

TRANSIENT NEUTRON FLUX

To investigate the possibility of changing neutron flux during a simulated loss-of-coolant accident (LOCA) (while the NRU reactor power is held constant), neutron flux histories are presented with time histories of control rod movement. During the course of the transient, light water was injected to reflood the test train. In a heavy-water-moderated reactor, this injection acts as a mild absorber; consequently, the dynamic control rod is withdrawn. Control rod positioning is displayed in Figure E.3 as is the average neutron flux time history measured at Level 15.

It has been proposed that the measured neutron flux is affected by the changing temperature of test assembly SPNDs and their instrument leads. The effect is primarily due to dielectrical resistivity changes in the leads due to a high-temperature, ionizing environment. High-temperature calibration of these sensors can be further compensated by SPND characteristics measured in high-temperature transients of the initial thermal-hydraulic experiment. However, because the temperature of the SPNDs and their instrument leads (attached to the shroud) remained cool throughout the experiment, temperature compensation of these data is not appropriate. The temperatures remained well within the SPND calibration temperature range.

The remainder of this appendix consists of the following graphical data:

<u>Figure E.1</u>. Axial Distribution of Pretransient Neutron Flux for TH-3.03

<u>Figure E.2</u>. Normalized Axial Distribution of Pretransient Neutron Flux for TH-3.03

<u>Figure E.3</u>. Changing Control Rod 15 Position and Neutron Flux During TH-3.03

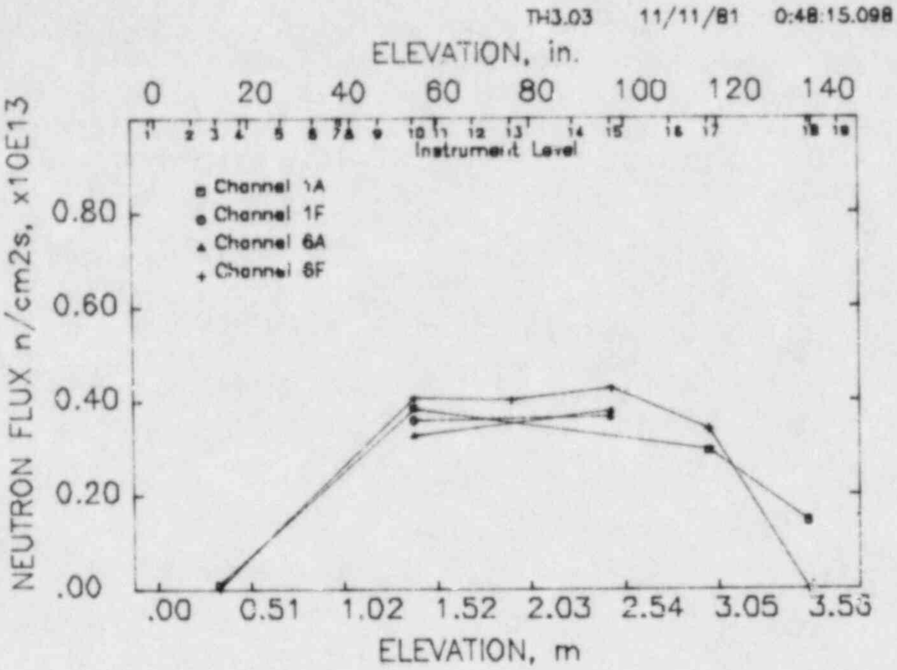


FIGURE E.1. Axial Distribution of Pretransient Neutron Flux for TH-3.03

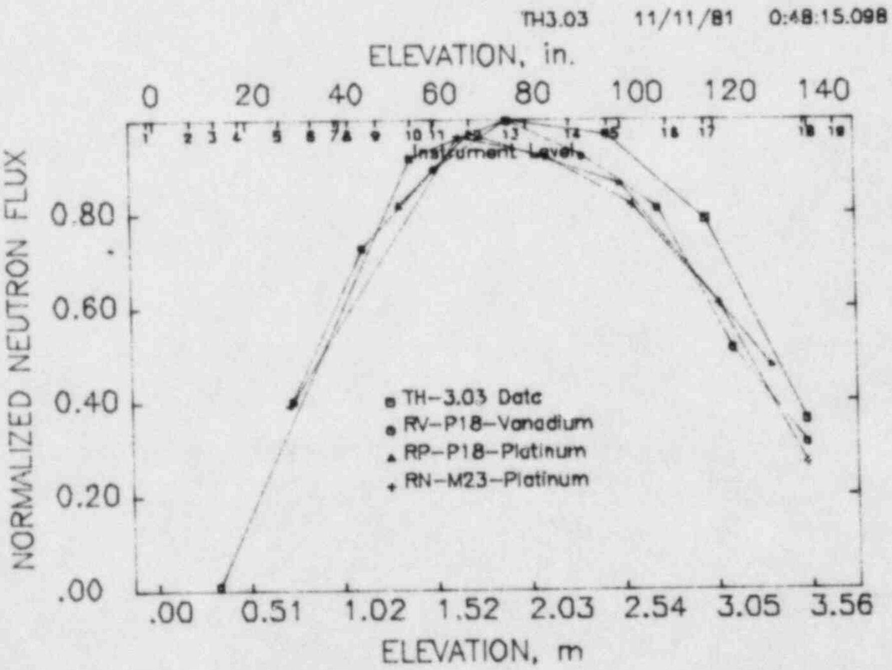


FIGURE E.2. Normalized Axial Distribution of Pretransient Neutron Flux for TH-3.03

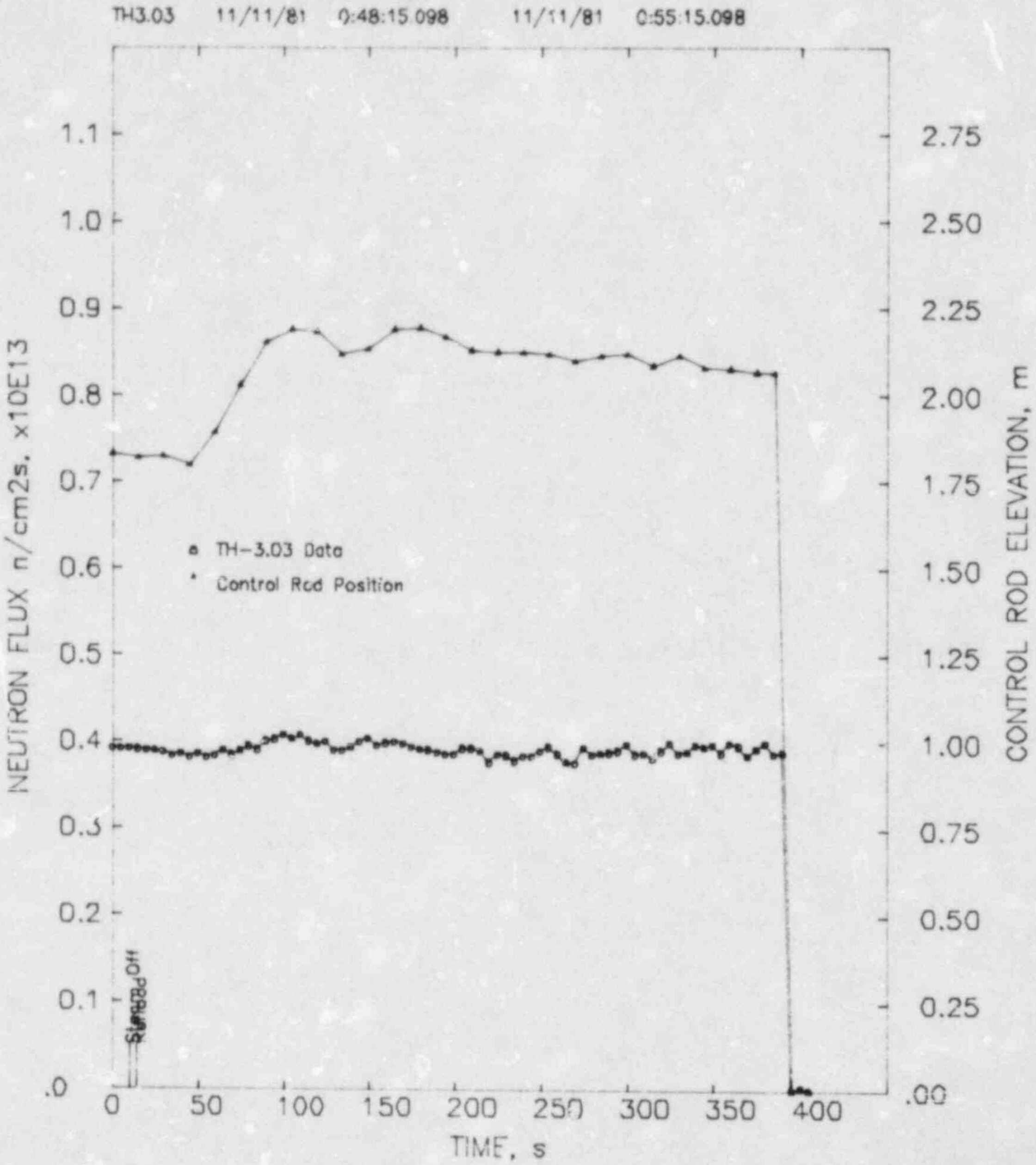


FIGURE E.3. Changing Control Rod 15 Position and Neutron Flux During TH-3.03

APPENDIX F

REFLOOD FLOW MEASUREMENTS

APPENDIX F

REFLOOD FLOW MEASUREMENTS

The reflood flow system included a Fisher-Porter turbine flowmeter in the high flow rate line and series-connected Barton and Fisher-Porter turbine flowmeters in a parallel low flow rate line. A parallel standby reflood line that provided emergency reflood coolant was not used during the TH-3 experiment.

The reflood control system was calibrated before the first transient using steam probe data to monitor the water/steam interface during reflood operation. Prior to the pretransient test phase, two reflood flow tests were performed at 0.0254 and 0.0508 m/s (1.0 and 2.0 in./s) to calibrate the reflood loop (see Figures F.1 and F.2 for the flow rate recordings).

Steam probe temperature histories provided independent measurements of the reflood coolant level (and reflood flow rate) in the test assembly. Figures F.3 and F.4 provided the data for the pretest reflood rate calibrations by showing the time required between subsequent level quenches.

Transient test starting times and reflood delay times depend on the flow conditions at the bottom of the active fuel. These flow conditions are related to the temperature response of thermocouples (TCs) located at Level 1, which is 0.013 m (0.5 in.) below the active fuel. The transients start when the steam coolant is shut off, as determined from a quick drop in temperature at Level 1. The reflood initiation times occur when the reflood water quenches the TCs at Level 1, as indicated by a second quick drop in temperature.

At the start of transient tests TH-3.02 and TH-3.03, the loop control system (LCS) used a fast reflood rate to bring the reflood coolant level up to the bottom of the fuel rods (Level 1). After that, the LCS used additional preset reflood rates. Tests TH-3.02 and TH-3.03 used the preprogrammed LCS for 90 s and the data acquisition and control system (DACS) with temperature feedback after 90 s. The DACS monitored selected temperatures (hot spot sensors) at Levels 15 and 17 and used the hottest average for control. The reflood rates for TH-3.02 and TH-3.03 as measured by turbine flowmeters are shown in Figures F.5 and F.6.

The remainder of this appendix consists of the following graphical data:

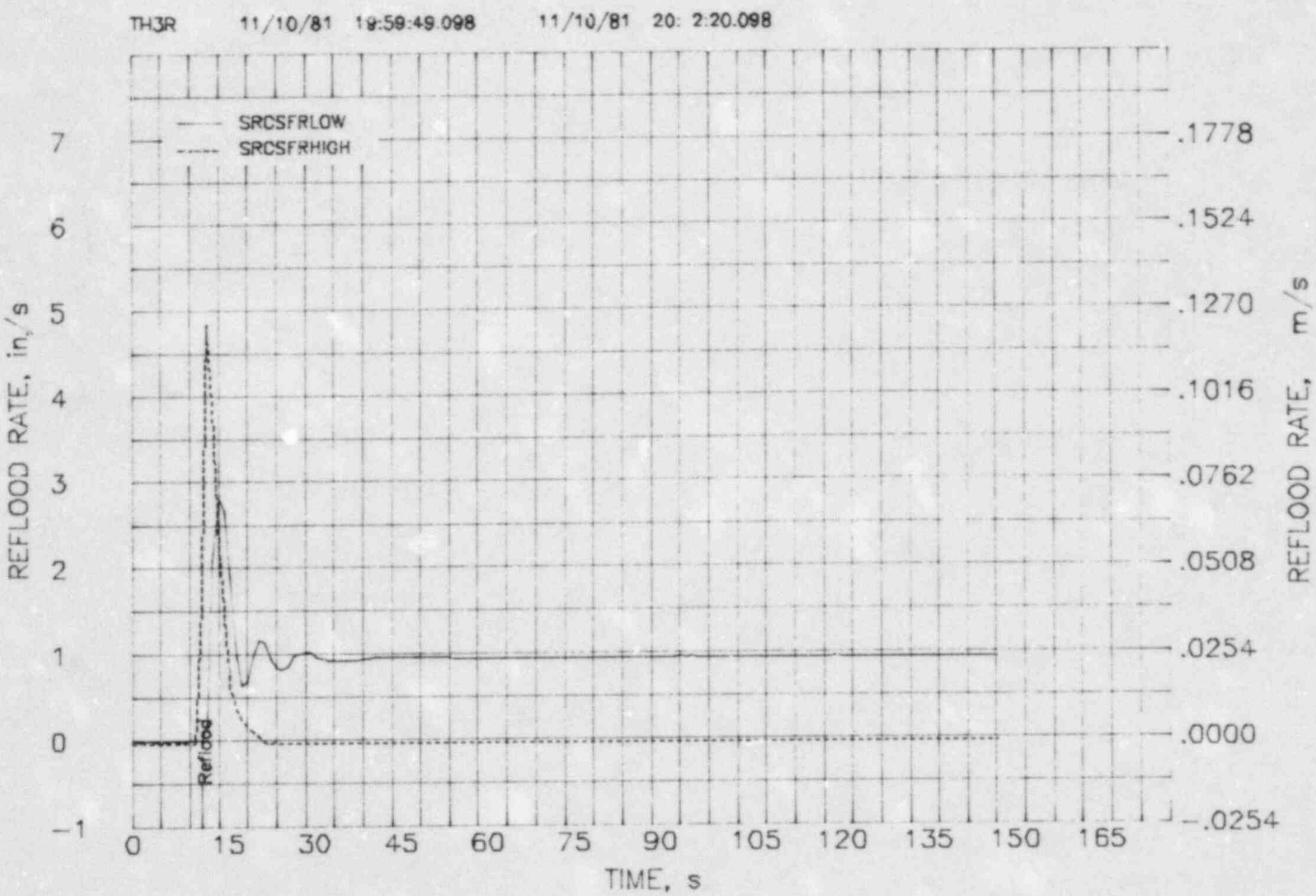


FIGURE F.1. Pretest Reflood Flow Rate for Calibration Test 1

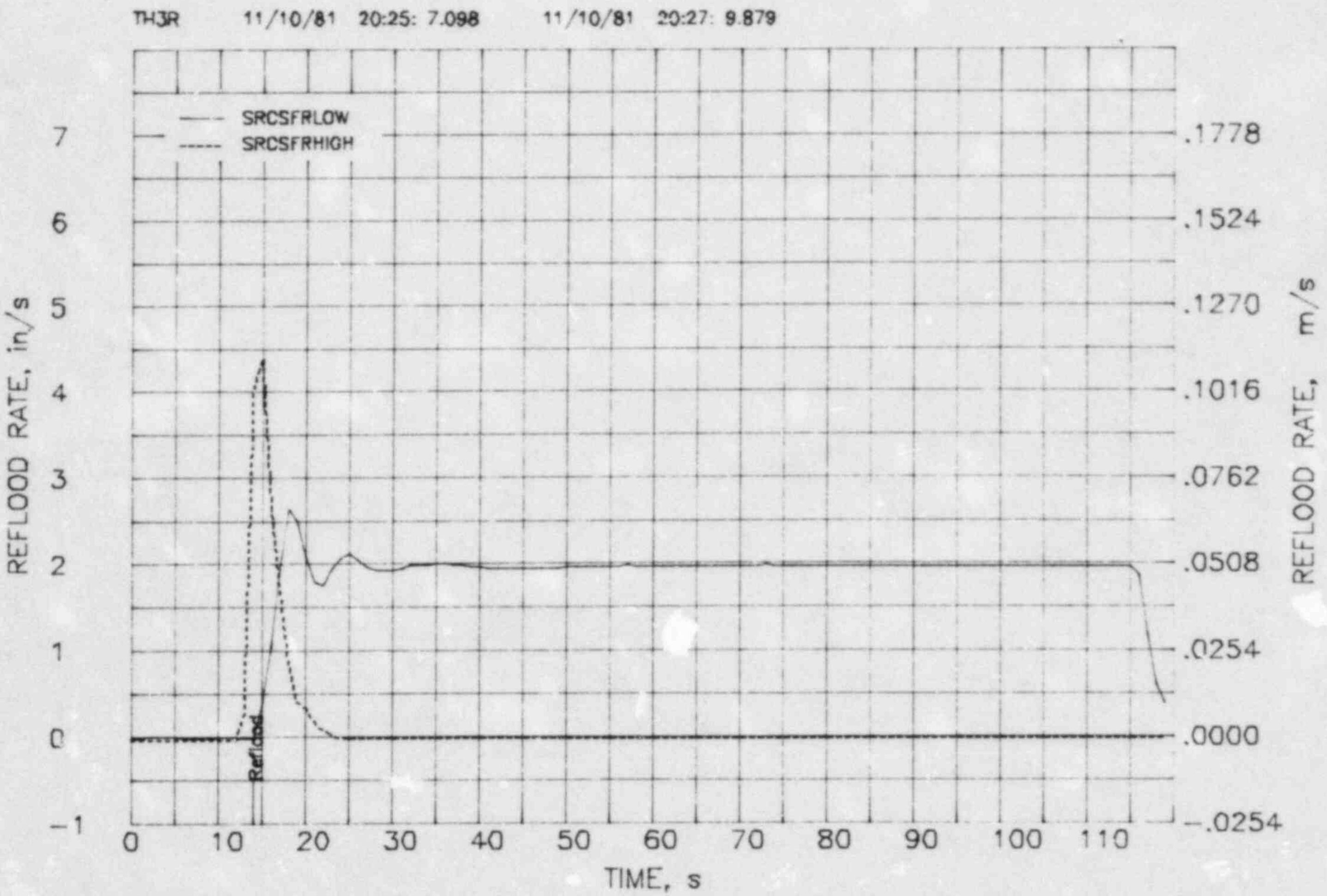


FIGURE F.2. Pretest Reflood Flow Rate for Calibration Test 2

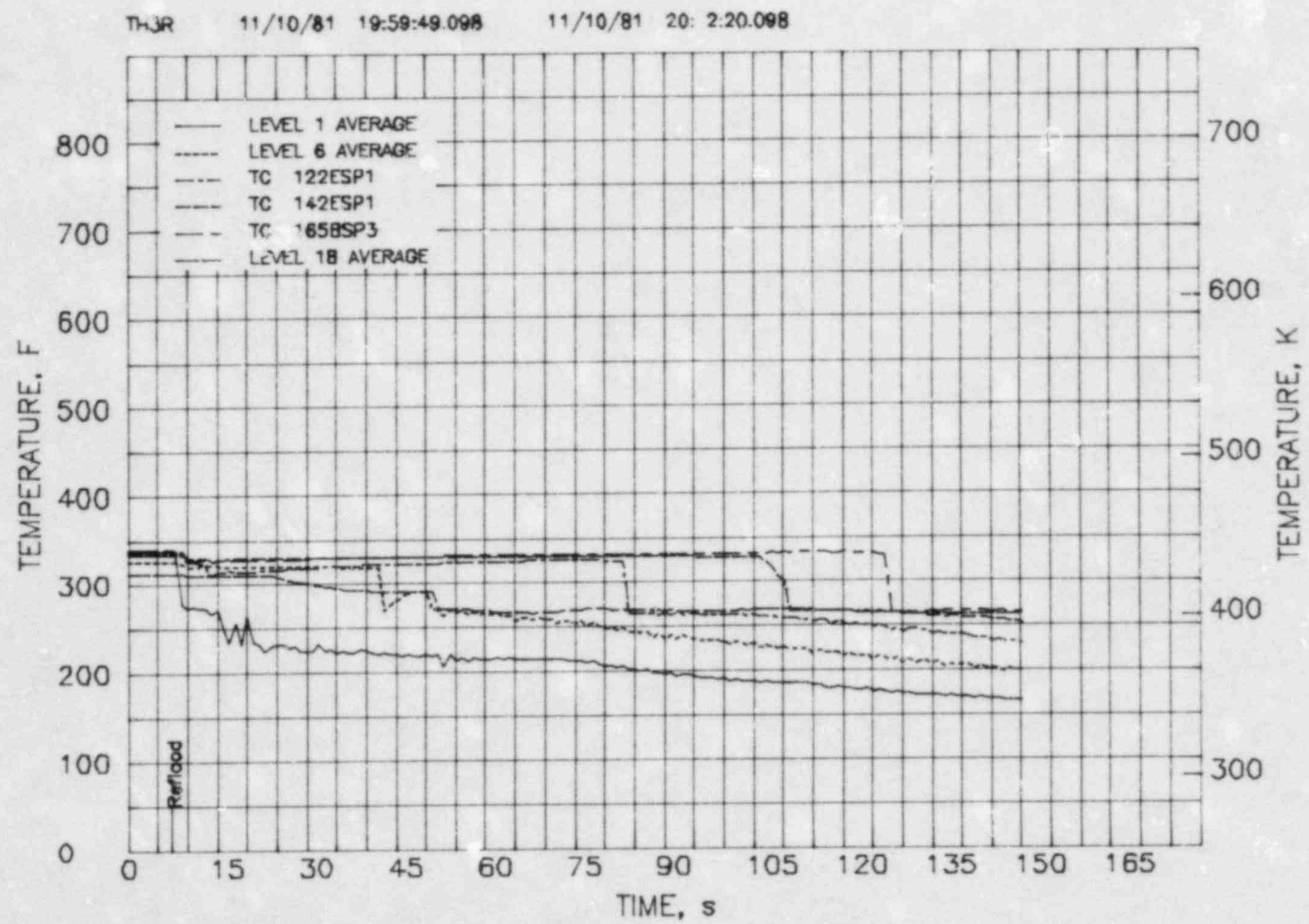


FIGURE F.3. Steam Probe and Shroud Temperatures for Reflood Calibration Test 1

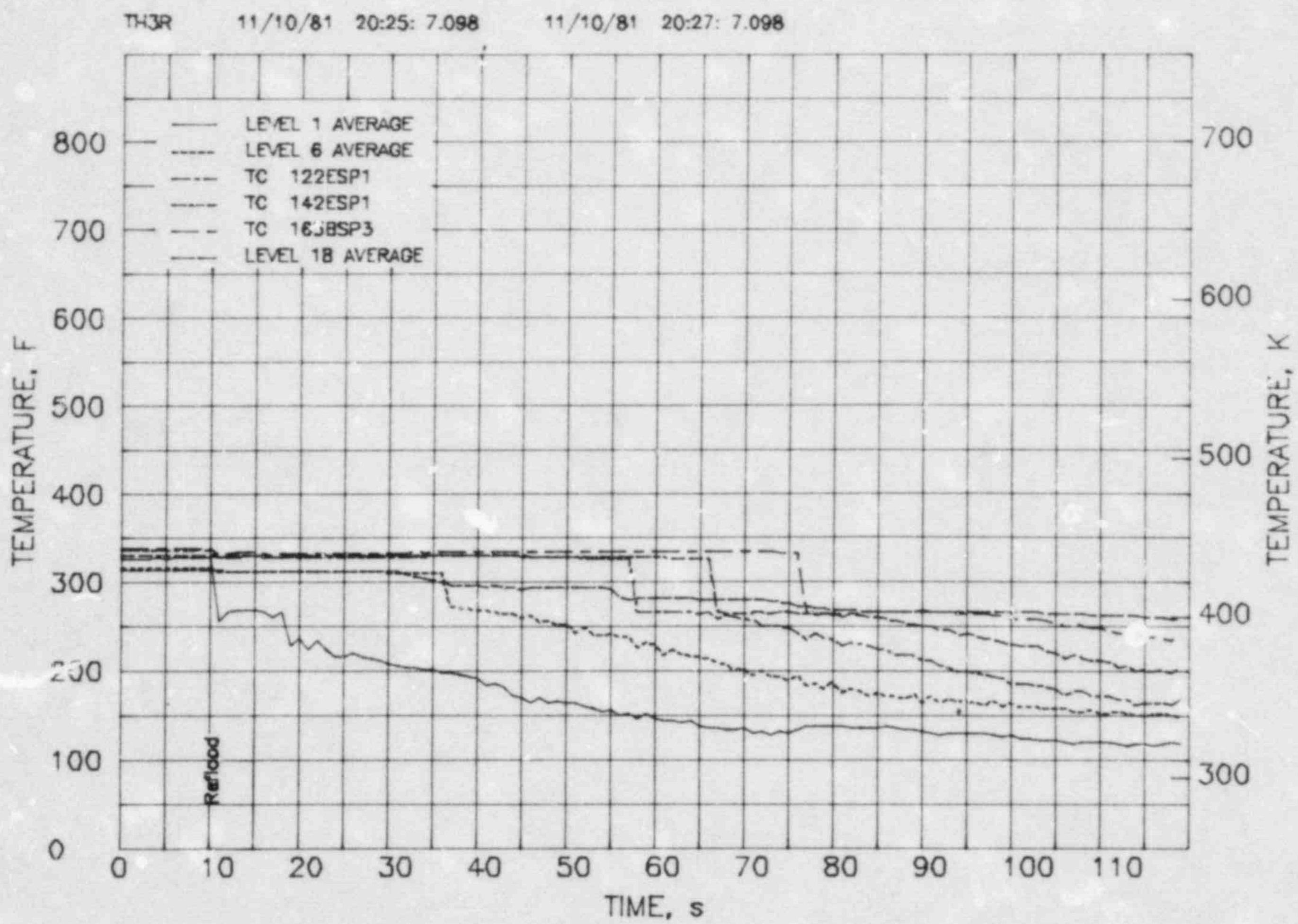


FIGURE F.4. Steam Probe and Shroud Temperatures for Reflood Calibration Test 2

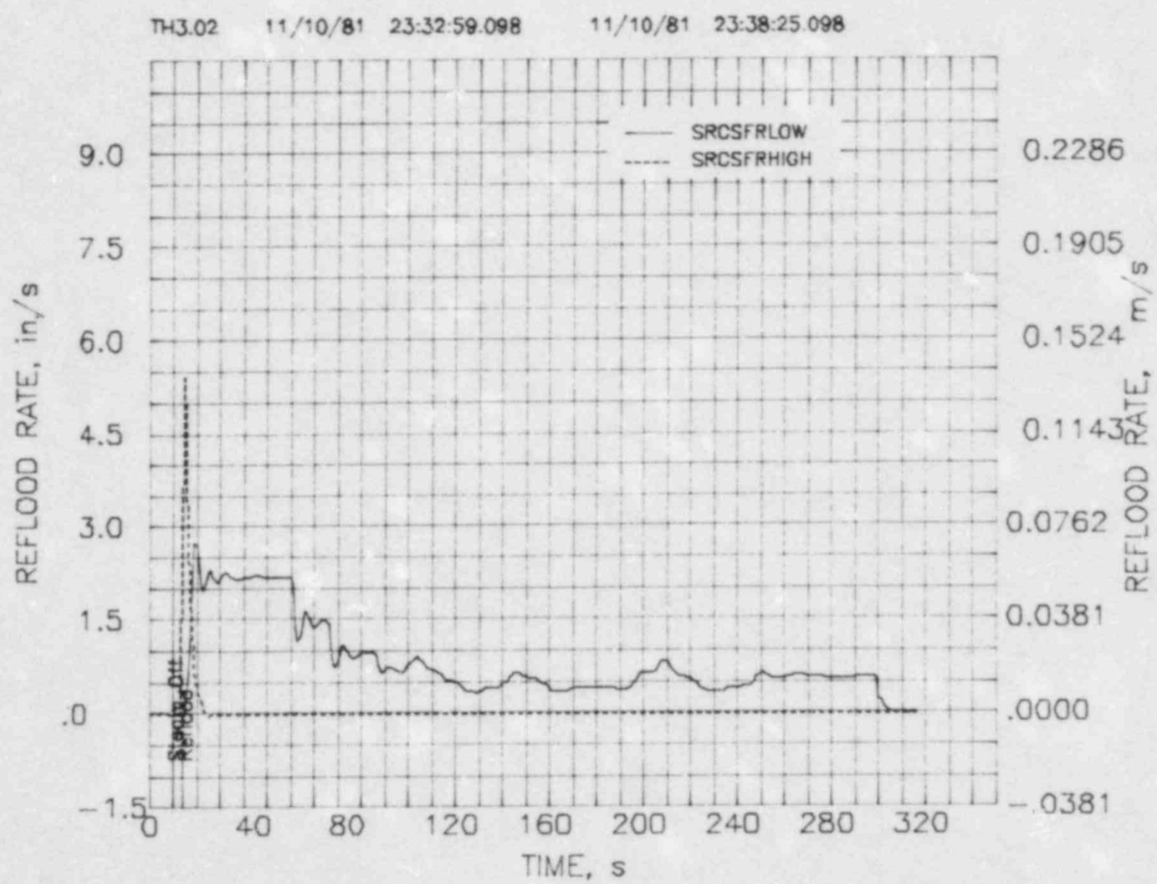


FIGURE F.5. Turbine Flowmeter Data for TH-3.02

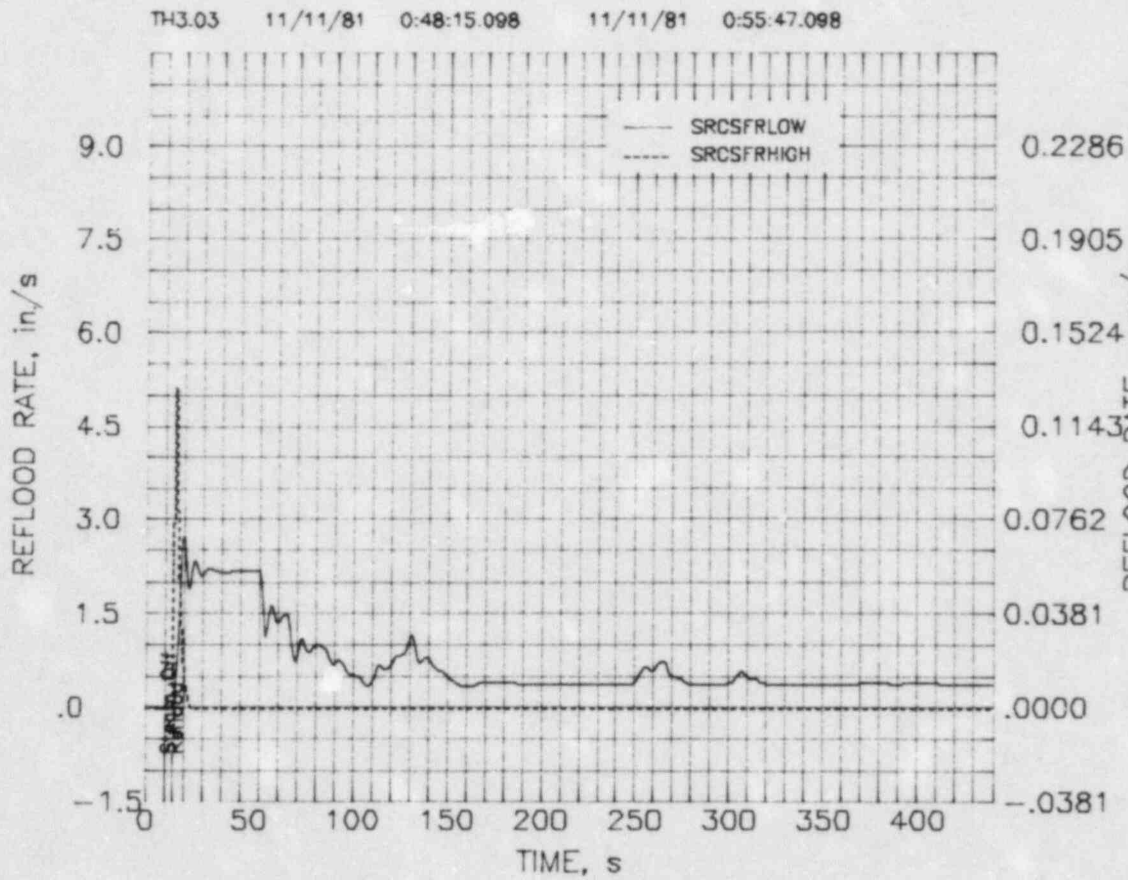


FIGURE F.6. Turbine Flowmeter Data for TH-3.03

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A series of in-reactor experiments are being conducted using light-water reactor fuel bundles as part of the Pacific Northwest Laboratory (PNL) Loss-of-Coolant Accident (LOCA) Simulation Program. The fifth experiment in the series of thermal-hydraulic and materials deformation experiments (TH-3) is described in this report. The experiments are being conducted in the National Research Universal (NRU) reactor, Chalk River, Ontario, Canada. The objective of TH-3 was to further refine the feedback control parameters developed in the TH-2 experiment and to re-establish the operability of the loop prior to the susequent materials deformation and repture test (MT-3). The TH-3 and MT-3 experiments were planned for the same reactor window and were run within two days of each other. The TH-3 test results insured the success of MT-3 and provided the opportunity to demonstrate the reactor control improvements and to evaluate a new desuperheater concept that would allow the test to run for extended times at high temperatures. The control system improvements and the addition of the new desuperheater resulted in fuel cladding temperatures above 1033K (1400°F) for 340 s. Experimental data and initial results are presented in this report.

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