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HEDL W-1 SLSF EXPERIMENT LOPI TRANSIENT AND BOILING TEST RESULTS

J.M. Henderson
S.A. Wood
R.B. Rothrock

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SUMMARY

The W-1 Sodium Loop Safety Facility (SLSF) experiment was designed to study the heat release characteristics of fast reactor fuel pins under Loss-of-Piping-Integrity (LOPI) accident conditions and determine stable sodium boiling initiation and recovery limits in a prototypic fuel pin bundle array.

The results of the experiment address major second level of assurance (LOA-2) safety issues and provide increased insight and understanding of phenomena that would inherently terminate hypothesized accidents with only limited core damage.

The irradiation phase of the experiment, consisting of thirteen individual transients, was performed between May 27 and July 20, 1979. The final transient produced approximately two seconds of coolant boiling, cladding dryout, and incipient fuel pin failure. The facility and test hardware performed as designed, allowing completion of all planned tests and achievement of all test objectives.

HEDL W-1 SLSF EXPERIMENT LOPI TRANSIENT AND BOILING TEST RESULTS

I. INTRODUCTION

The W-1 Sodium Loop Safety Facility (SLSF) experiment if the fifth in a series of experiments sponsored by the Department of Energy as part of the National Fast Breeder Reactor (FBR) Safety Assurance Program. The experiments are being conducted under the direction of both Argonne National Laboratory and the Hanford Engineering Development Laboratory (HEDL). W-1 was the first in the series of HEDL SLSF experiments and was conducted in cooperation with the Advanced Reactor Systems Division of General Electric Company (GE/ARSD). The facility, located in the Engineering Test Reactor (ETR) at the Idaho National Laboratory (INEL), is operated by EG&G Idaho, Inc.

The irradiation phase of the W-1 SLSF experiment was conducted between May 27 and July 20, 1979, terminating with incipient fuel pin cladding failure during the final boiling transient.

The experiment hardware and facility performed as designed, allowing completion of all planned tests and test objectives.

The combined results from the W-1 SLSF experiment are expected to help resolve FBR safety issues in the areas of:

- Heat release characteristics of FBR fuel pins during loss-of-piping integrity (LOPI) accident conditions,
- Sodium boiling initiation and void progression characteristics,
- Coolant boiling conditions required to produce incipient fuel pin failure.

The improved knowledge in these areas will permit refinement of the models employed in whole-core loss-of-flow safety analyses.

II. OBJECTIVES

The W-1 experiment has two distinct and separate objectives. The primary objective is to evaluate fuel pin heat release characteristics during LOPI accident flow and power conditions. The Clinch River Breeder Reactor (CRBR) conditions during this accident are being used as representative of FBR's. A sequence of four LOPI transients were to be conducted to collect data at the different fuel pin conditions of: a) fresh, unrestructured fuel, b) fresh, restructured fuel, c) irradiated, uncracked fuel (after steady-state irradiation), and d) irradiated, cracked fuel (after shutdown and startup).

The second objective of the W-1 experiment is to determine stable sodium boiling and recovery limits (the boiling window) as a function of fuel pin power and bundle flow rates. These boiling tests were to culminate with incipient fuel pin failure to determine the range of power/flow ratios over which stable sodium boiling exists.

These objectives are consistent with resolution of major second level FBR safety assurance (LOA-2) issues identified in the Fuel Pin Failure Mechanisms Program Plan.(1) They will provide increased insight and understanding of phenomena that inherently terminate hypothesized accidents with only limited core damage. Furthermore the boiling window data will expand sodium boiling data already obtained in the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility at Oak Ridge National Laboratory.

III. EXPERIMENT DESCRIPTION

The SLSF in-pile loop, located in the Engineering Test Reactor (Figure 1), is a doubly-contained closed sodium loop test vehicle 8.23 m (27 ft) long and weighing approximately 3400 kg (7500 lbm). The loop consists of a primary and secondary containment vessel, an annular linear induction electromagnetic pump (ALIP), a tube-and-shell, sodium-to-helium heat exchanger (HX), a 0.1 cm (40 mil) thick cadmium thermal neutron filter, loop sensors, removeable top closure (RTC), and the instrumented test train (Figures 2 and 3).

Figure 1. SODIUM LOOP SAFETY FACILITY (SLSF)

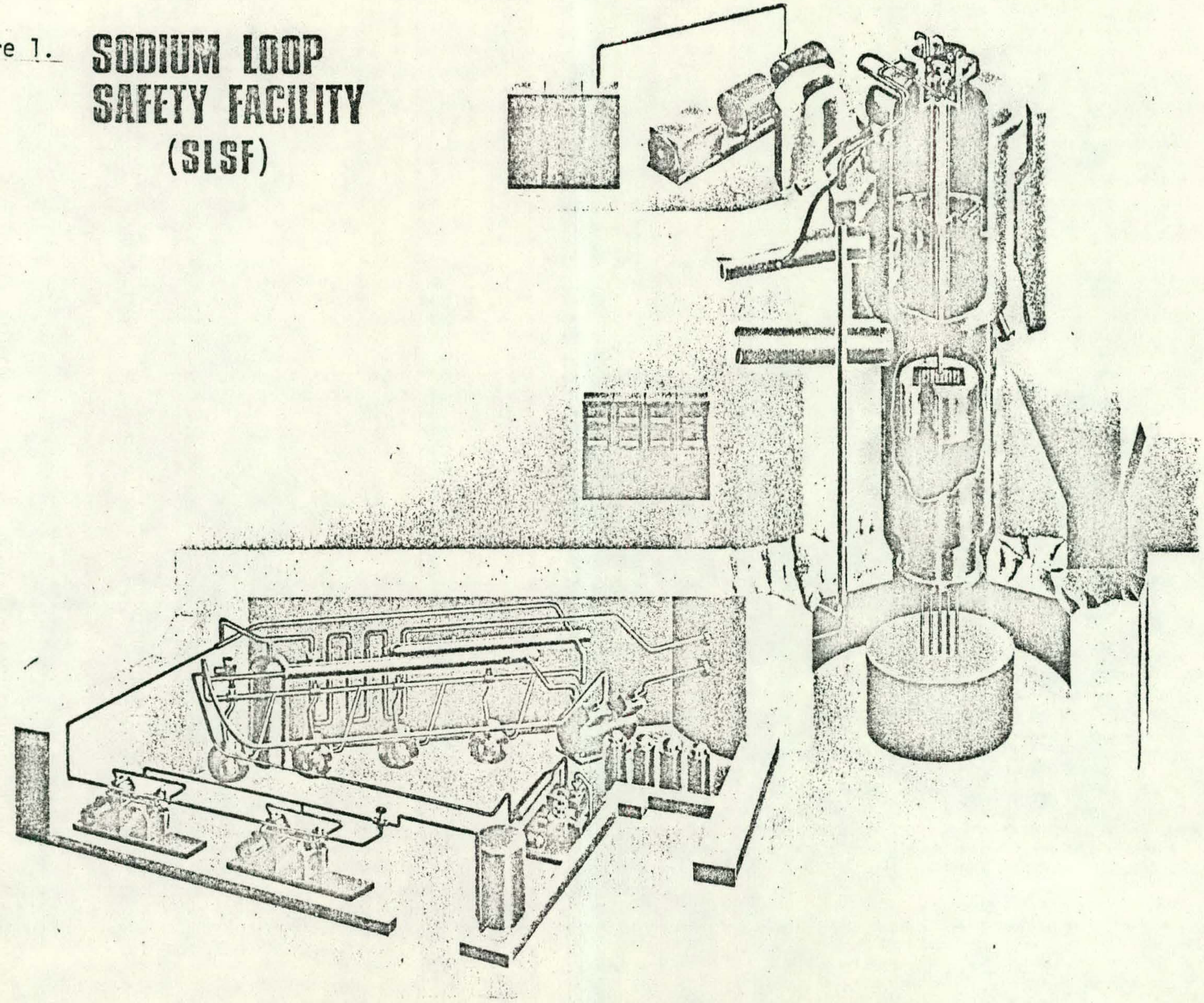


Figure 2-

HEDL SLSF EXPERIMENT ASSEMBLY
UPPER SECTION

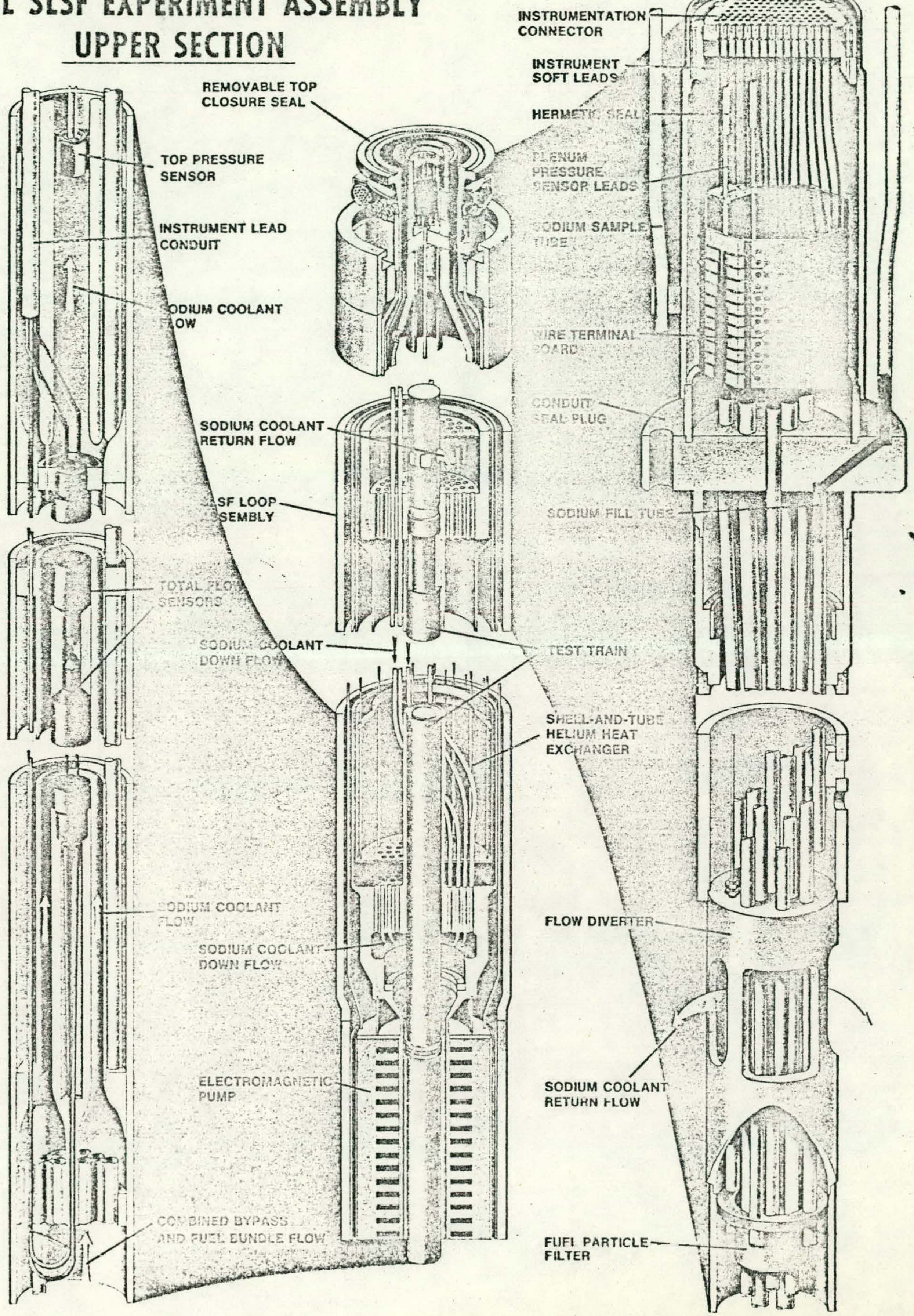
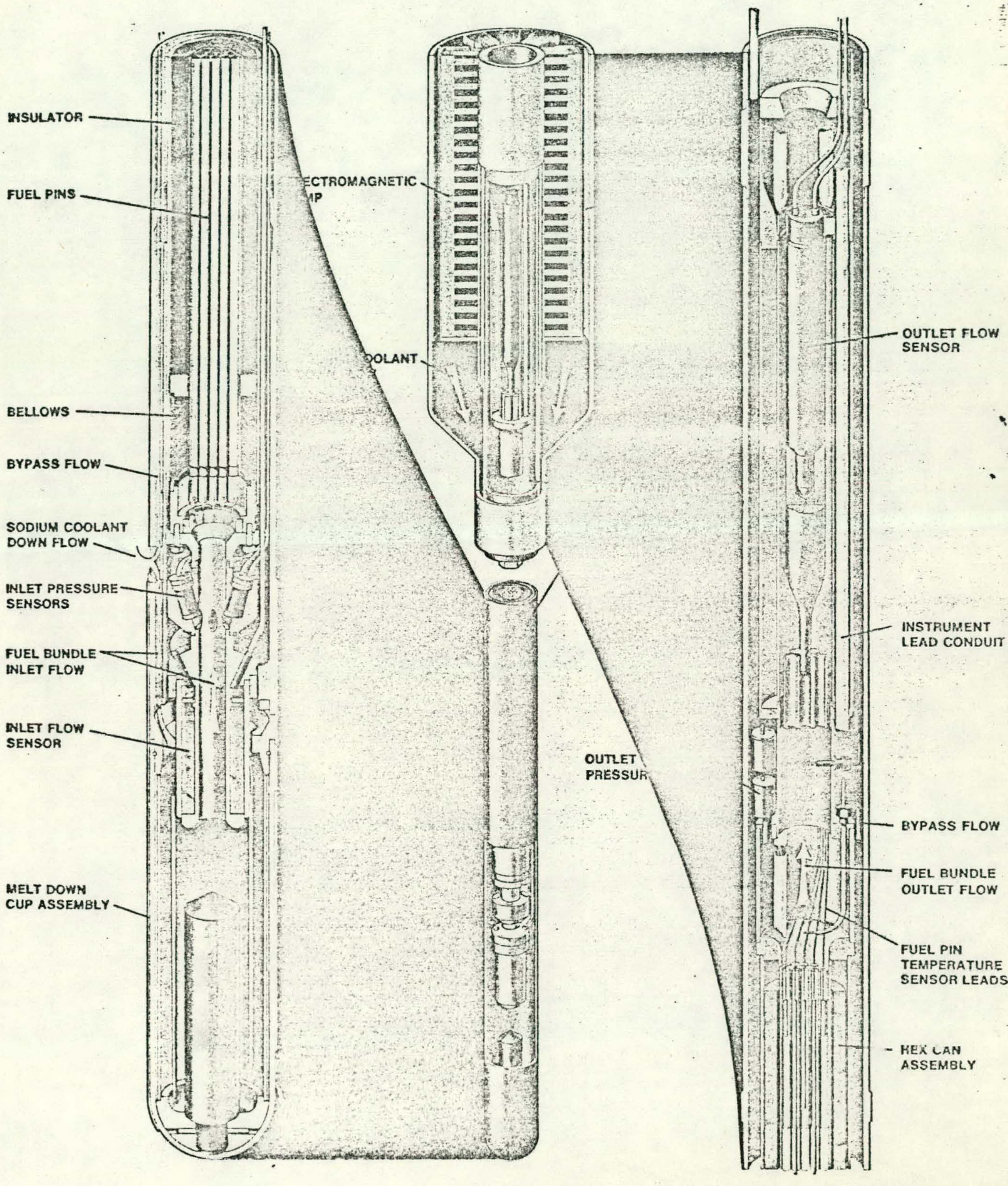


Figure 3. **HEDL SLSF EXPERIMENT ASSEMBLY**

LOWER SECTION



During loop operation, 661 K (730 F) sodium flows through the downcomer region (between the test train and primary tube) to the bottom of the loop where the flow reverses direction and is split into the bundle flow at 1.95 kg/sec (4.29 lbm/sec) and by-pass flow at 3.98 kg/sec (8.77 lbm/sec) (Figures 4 and 5). The flow rises up through the fuel bundle and by-pass regions of the test train and recombines above the outlet flow sensor. The sodium, now 755 K (900 F), passes up through the remainder of the test train to the top plenum region, reverses direction, and flows down through the HX and ALIP returning to the downcomer annulus.

The heart of the W-1 loop is the instrumented test train. It is approximately 7.9 m (26 ft) long and contains 19 FTR size fuel pins with CRBR type axial blankets in a hexagonal bundle array. The test train contains seventy-six (76) thermocouples, sixteen (16) pressure transducers, and four (4) sodium flowmeters. In addition, the center seven fuel pins have annular fuel pellets over the length of the active section and in-fuel thermocouples placed in the pins to measure fuel temperatures at three different elevations. The outer twelve pins have solid pellets (Figure 6). Fuel enrichments were selected to produce a flat power profile.

IV. PRELIMINARY DATA EVALUATION

Initial data evaluation of the more important test parameters was completed during the irradiation phase of the experiment. Data on failure locations and magnitude, and fuel pin characterization will be obtained during post-test examination.

Initial evaluation of the data reveals the following:

- Experiment conditions simulating a 15 percent overpower, hot channel, CRBR LOPI transient did not result in coolant boiling (which was predicted). These data show the conservativeness of the CRBR design and survivability of the plant under these severe accident conditions.

Figure 4. SLSF IN-PILE LOOP CROSS SECTION

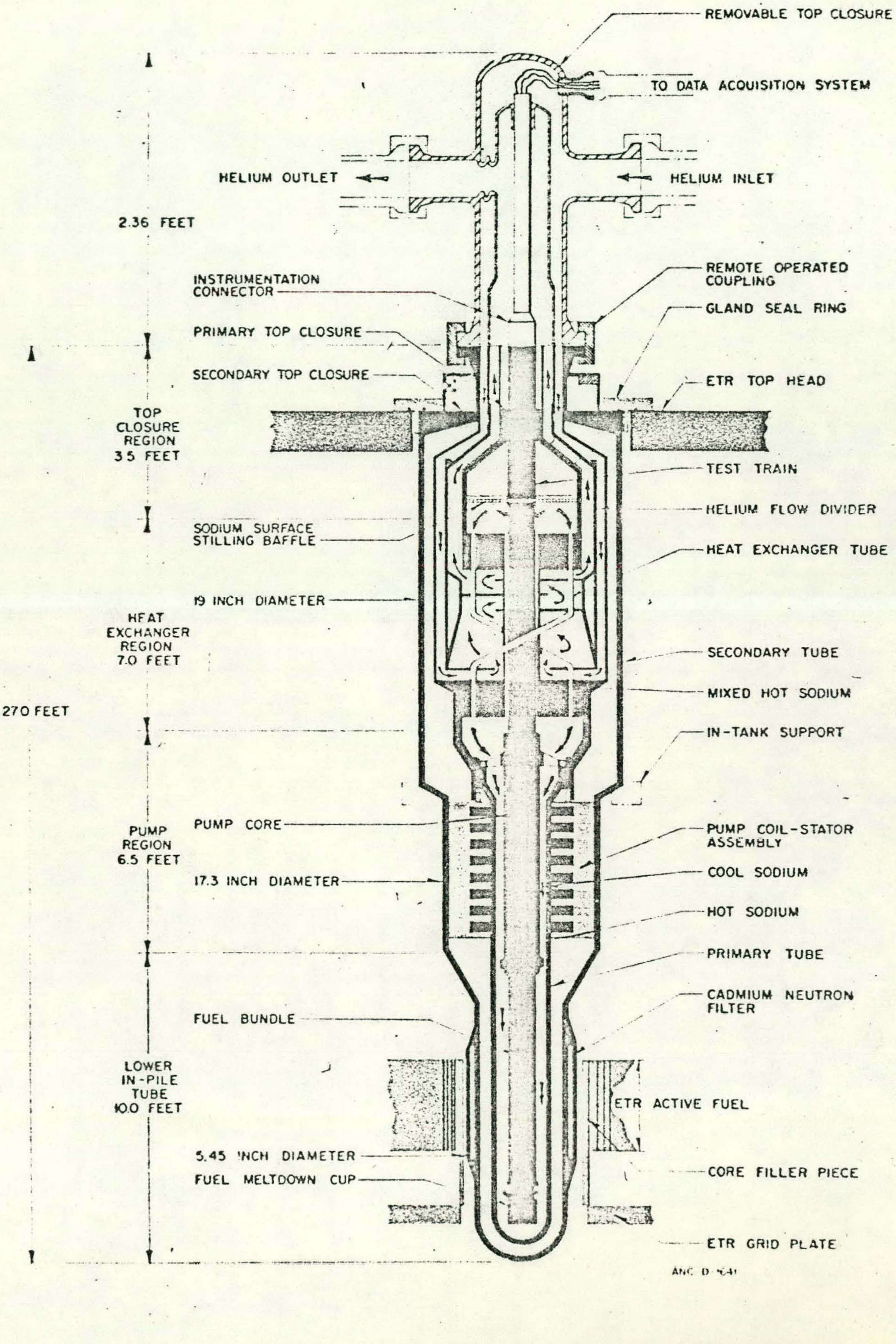


Figure 5. **CROSS-SECTION OF THE W-1 SLSF IN THE LOWER TEST SECTION REGION**

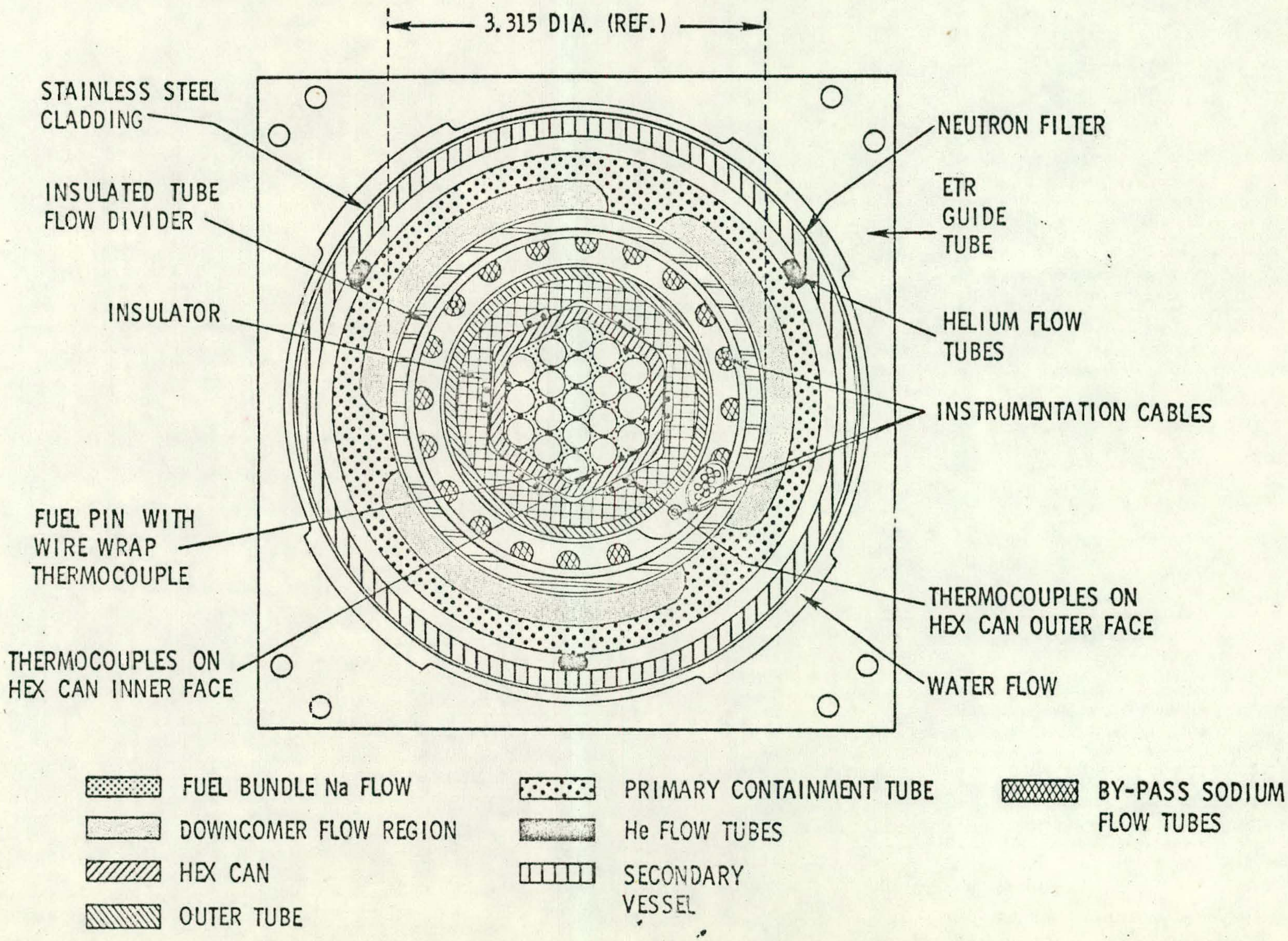


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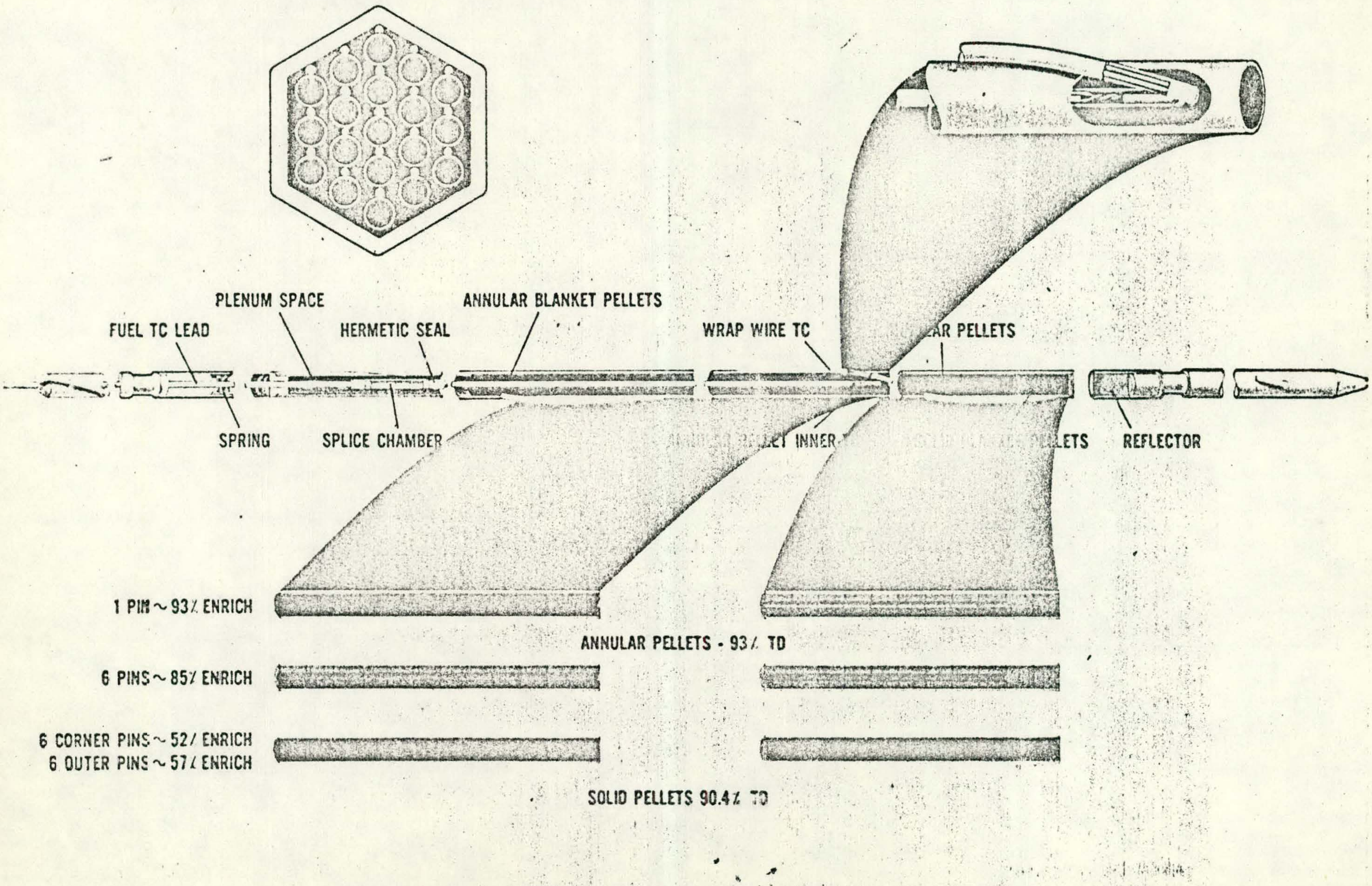
Figure 6. **SLSF INSTRUMENTED PINS**



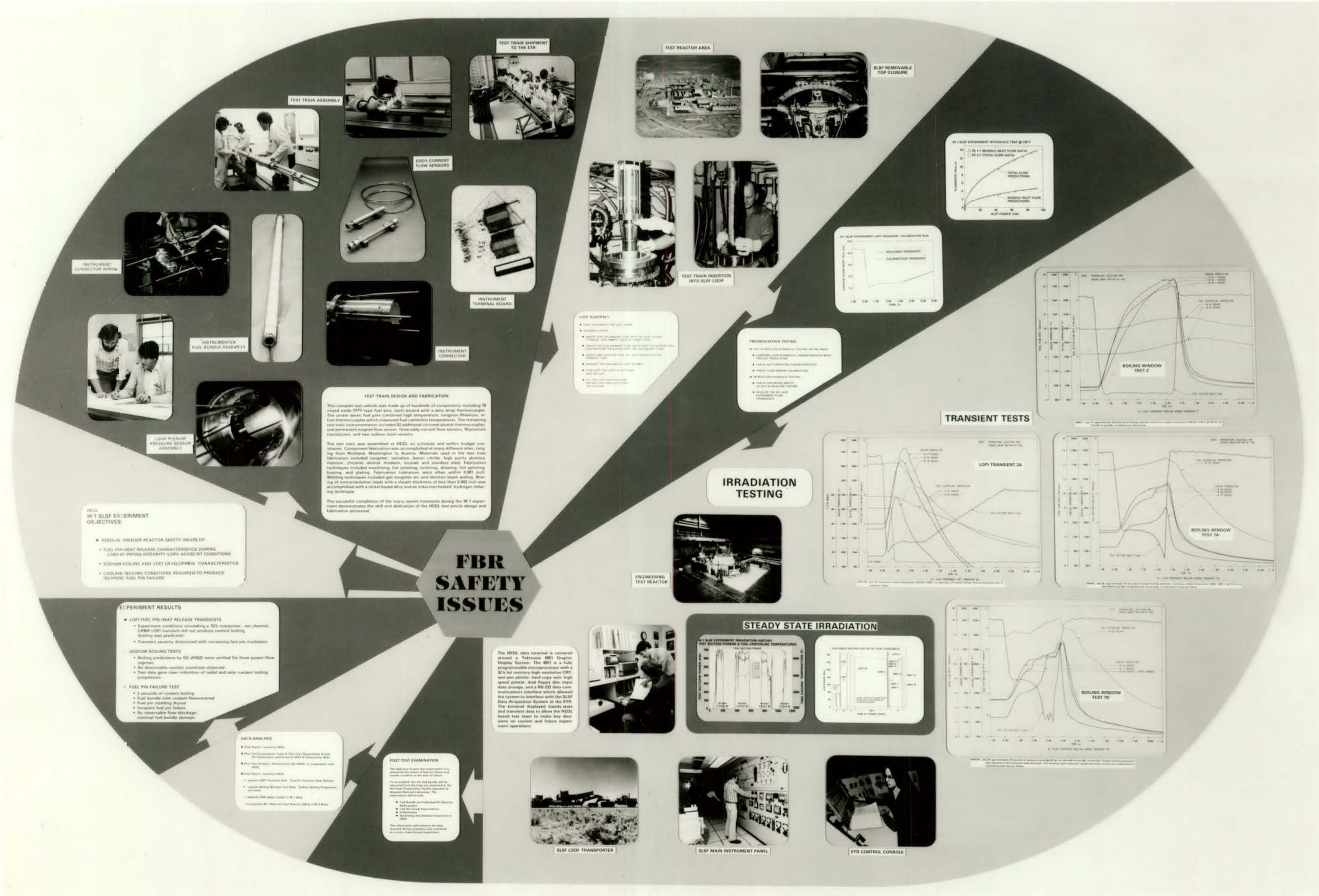

FBR SAFETY ISSUES
TEST TRAIN SHIPMENT TO THE ETR
TEST TRAIN ASSEMBLY
EDDY CURRENT FLOW SENSORS
INSTRUMENT CONNECTOR WIRING
INSTRUMENT TERMINAL BOARD
INSTRUMENTED FUEL BUNDLE ASSEMBLY
INSTRUMENT CONNECTOR
LOOP PLENUM PRESSURE SENSOR ASSEMBLY
TEST TRAIN DESIGN AND FABRICATION
This complex test vehicle was made up of hundreds of components including 19 mixed oxide FFTF type fuel pins, each wound with a wire wrap thermocouple. The center seven fuel pins contained high temperature, tungsten Rhenium, in fuel thermocouples which measured fuel centerline temperatures. The remaining test train instrumentation included 53 additional chromel alumel thermocouples, one permanent magnet flow sensor, three eddy current flow sensors, 16 pressure transducers, and two sodium level sensors.
The test train was assembled at HEDL on schedule and within budget constraints. Component fabrication was accomplished at many different sites, ranging from Richland, Washington to Austria. Materials used in the test train fabrication included tungsten, tantalum, boron nitride, high purity alumina, rhenium, chromel, alumel, rhodium, inconel, and stainless steel. Fabrication techniques included machining, hot pressing, sintering, drawing, hot spinning, brazing, and plating. Fabrication tolerances were often within 0.001 inch. Welding techniques included gas tungsten arc and electron beam welding. Brazing of instrumentation leads with a sheath thickness of less than 0.003 inch was accomplished with a nickel based alloy and an induction heated, hydrogen reducing technique.
The successful completion of the many severe transients during the W-1 experiment demonstrates the skill and dedication of the HEDL test article design and fabrication personnel.
HEDL
W-1 SLSF EXPERIMENT OBJECTIVES:
RESOLVE BREEDER REACTOR SAFETY ISSUES OF:
FUEL PIN HEAT RELEASE CHARACTERISTICS DURING LOSS-OF-PIPING INTEGRITY (LOPI) ACCIDENT CONDITIONS
SODIUM BOILING AND VOID DEVELOPMENT CHARACTERISTICS
COOLANT BOILING CONDITIONS REQUIRED TO PRODUCE INCIPIENT FUEL PIN FAILURE
EXPERIMENT RESULTS
LOPI FUEL PIN HEAT RELEASE TRANSIENTS
Transient severity diminished with increasing fuel pin irradiation
SODIUM BOILING TESTS
No discernable coolant inlet-outlet observed
FUEL PIN FAILURE TEST
2 seconds of coolant boiling
Fuel bundle inlet coolant flow reversal
Fuel pin cladding dryout
Incipient fuel pin failure
No observable flow blockage; minimal fuel bundle damage
DATA ANALYSIS
POST TEST EXAMINATION
TEST REACTOR AREA
SLSF REMOVABLE TOP CLOSURE
TEST TRAIN INSERTION INTO SLSF LOOP
LOOP ASSEMBLY
PREIRRADIATION TESTING
TRANSIENT TESTS
BOILING WINDOW TEST 2'
LOPI TRANSIENT 2A
BOILING WINDOW TEST 7A'
BOILING WINDOW TEST 7B'
IRRADIATION TESTING
ENGINEERING TEST REACTOR
STEADY STATE IRRADIATION
The HEDL data terminal is centered around a Tektronix 4051 Graphic Display System. The 4051 is a fully programmable microprocessor with a 32 k bit memory high resolution CRT, wet pen plotter, hard copy unit, high speed printer, dual floppy disc mass data storage, and a RS-232 data communications interface which allowed the system to interface with the SLSF Data Acquisition System at the ETR. The terminal displayed steady state and transient data to allow the HEDL based test team to make key decisions on current and future experiment operations.
SLSF LOOP TRANSPORTER
SLSF MAIN INSTRUMENT PANEL
ETR CONTROL CONSOLE

PHOTO NO. 8001830 -1cn

- The boiling window tests verified GE/ARSD pretest predictions for incipient boiling in the center coolant channels and extended the range of data from the THORS tests to 472 W/cm (14.4 kW/ft) peak linear pin power. These data will better identify boiling incipience over a range of reactor conditions, ensuring the conservative nature of current and future reactor designs.

- There was no discernable coolant superheat in any transient prior to coolant boiling. These data reduce the concern over coolant superheat during large breeder reactor accident conditions.

- Steady-state coolant radial temperature gradients were less than 5.5 K, giving good approximation of coolant temperatures in the interior of large FBR fuel bundles. However, radial temperature gradients increased to approximately 80K during the W-1 transients, affecting the timing of radial void progression. This was probably due to a heat sink effect caused by the hex can.

  There is evidence that flow diversion from center to outer coolant channels caused by boiling in the center channels, accelerated boiling initiation in the outer channels. This evidence is comprised of recorded coolant temperatures that show increases in the rate of outer channel temperature rise corresponding in time with center channel coolant void expansion. These data will be useful in the development of FBR subassembly coolant boiling models, once the effect of the radial temperature gradient has been compensated for.

- In the final boiling window test approximately two seconds of coolant boiling were observed. The upper half of the fuel bundle voided, resulting in test section inlet flow reversal, cladding dryout and fuel pin failure. The delay between the onset of coolant boiling and fuel pin failure show the conservative nature of current safety code modeling in this area.

- Fuel bundle damage appears to be minimal with no coolant flow blockage detectable.

V. W-1 SLSF EXPERIMENT OPERATION

The irradiation phase of the W-1 experiment was conducted between May 27 and July 20, 1979. Initial full test section power, 668 kW, was produced at an ETR power of 143 MW (145 MW was predicted). Figure 7 shows the predicted power coupling for all test sections powers. During the later stages of the experiment, the ETR power at full test section power lowered to between 120 and 130 MW. This change in power coupling was caused primarily by the gradual build up of fission product decay heat in the test fuel bundle. As the fraction of fission product induced power increased, the ETR power required to maintain full test section power decreased. Another factor which is thought to have changed the power coupling was neutron flux peaking in the SLSF corner of the ETR. This was the result of ETR control rod manipulations required to offset fission product accummulation in the ETR fuel.

The irradiation history is shown in Figures 8 through 12 . There were very few spurious scrams during the experiment due to recent improvements to reactor and loop plant protective systems.

B. Steady State Loop Operation

Coolant temperature data recorded during steady-state operations were very close to pretest predictions (Figure 13). These data show the validity of the fuel pin-to-coolant heat transfer modeling in steady-state fuel pin performance and thermal-hydraulic codes.

Fuel temperature data were much higher than expected throughout the experiment while fuel restructuring occurred more slowly than expected. These data will be valuable in the calibration of burnup models in fuel pin thermal performance codes. Plots of some fuel temperature data and test section power with time are shown in Figures 14 through 18 .

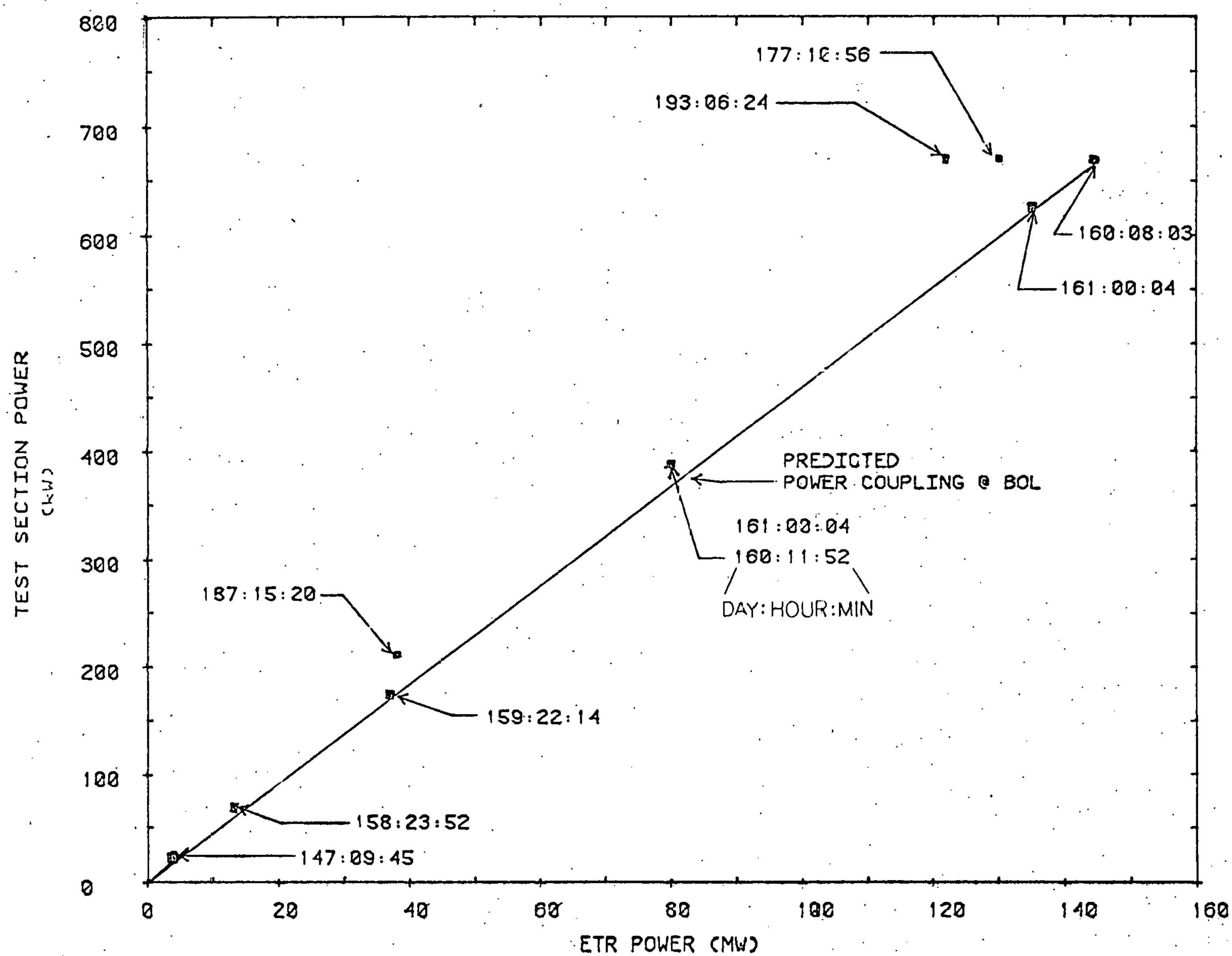


FIGURE 7. W-1 SLSF EXPERIMENT PREDICTED vs MEASURED POWER COUPLING

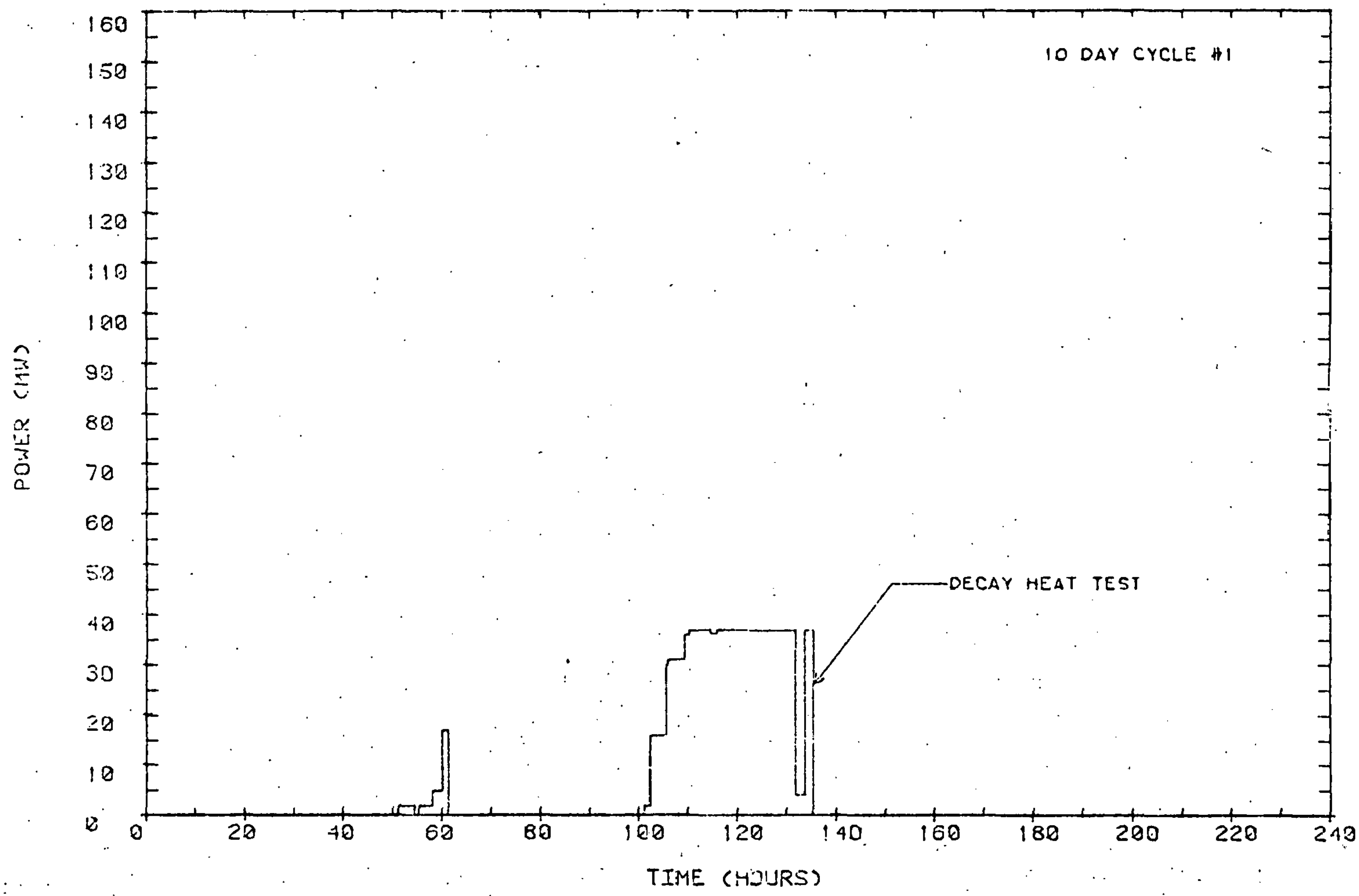


Figure 8. ETR POWER HISTORY FOR THE W-1 SLSF EXPERIMENT

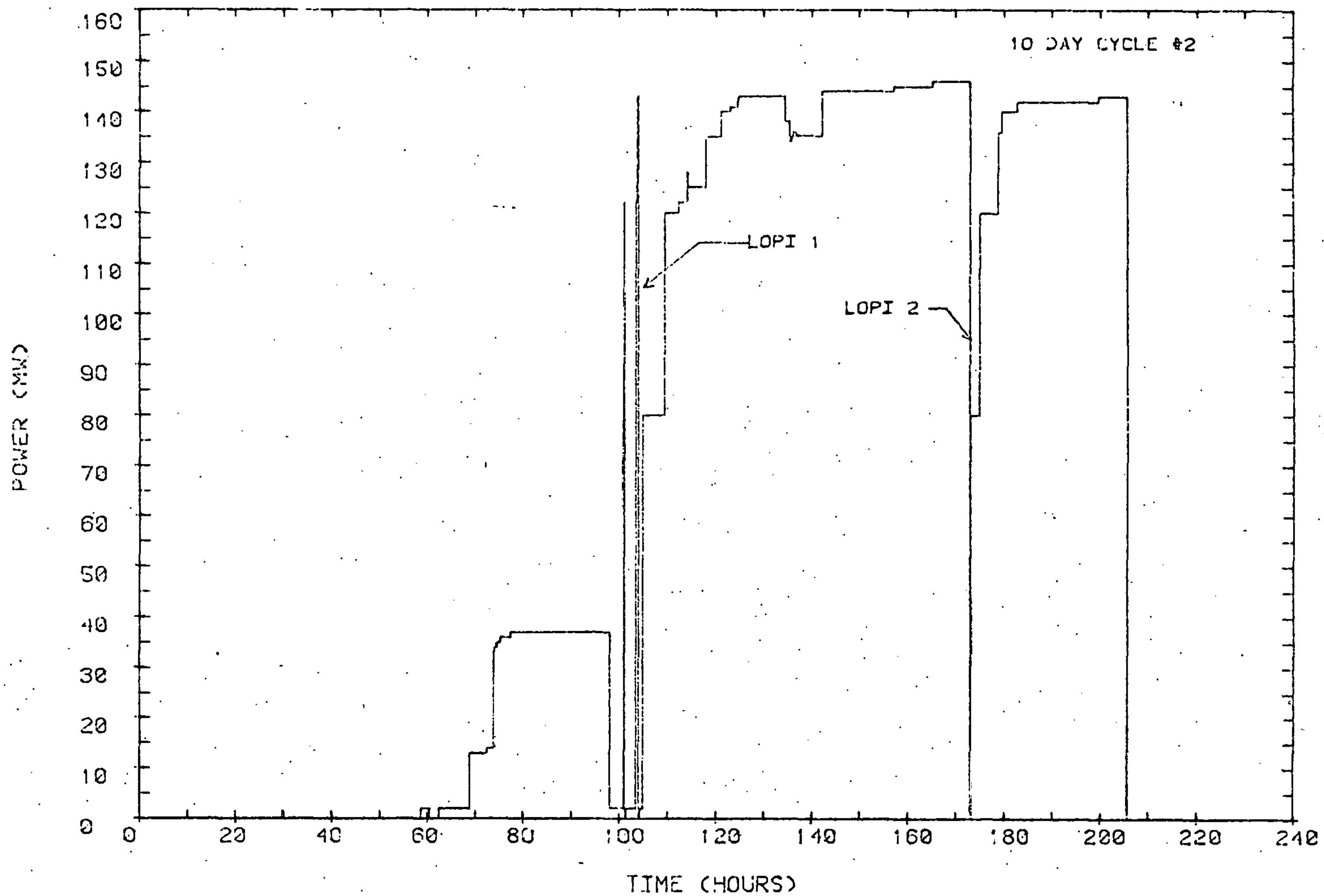


Figure 9. ETR POWER HISTORY FOR THE W-1 SLSF EXPERIMENT

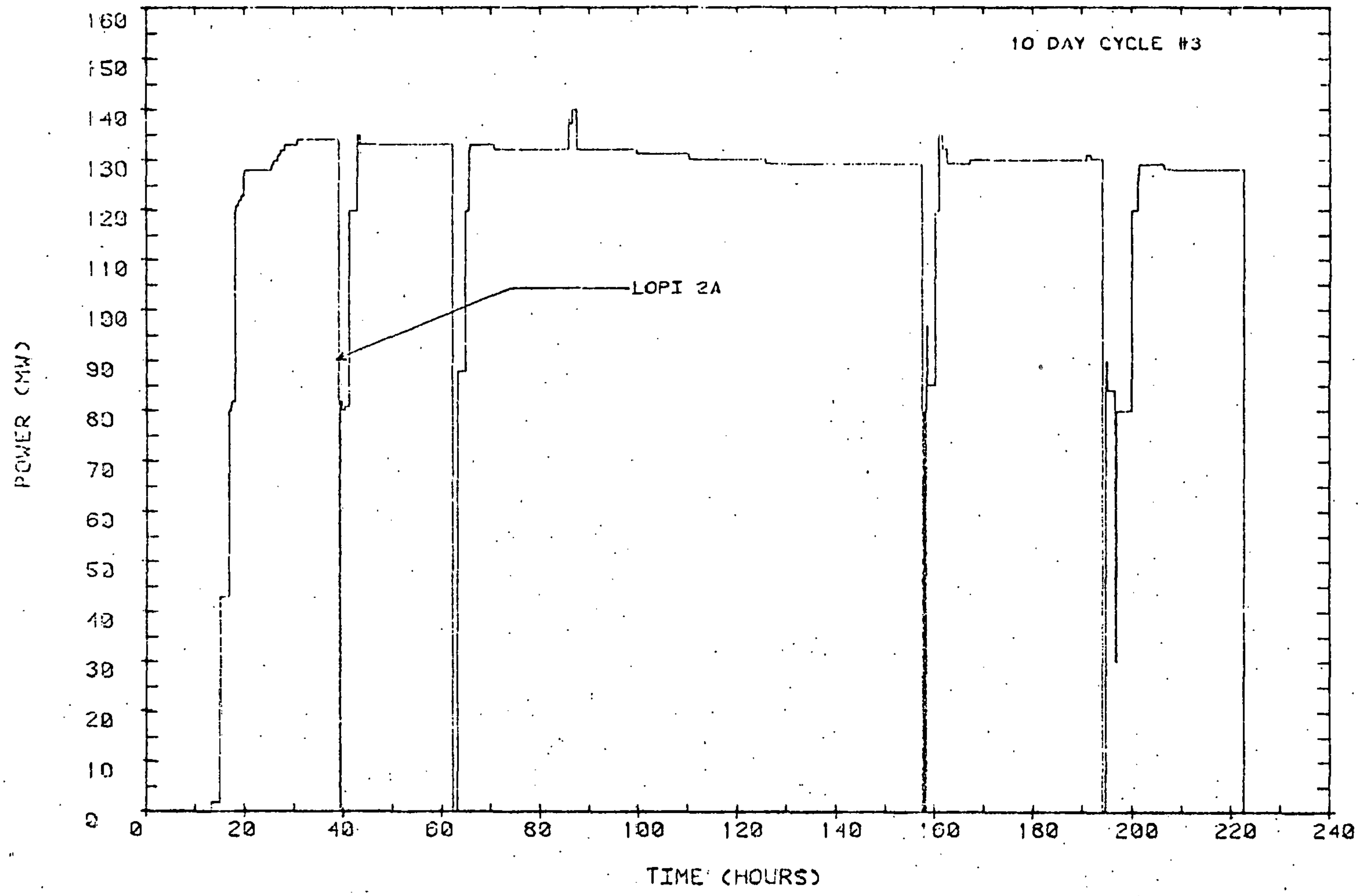


Figure 10. ETR POWER HISTORY FOR THE W-1 SLSF EXPERIMENT

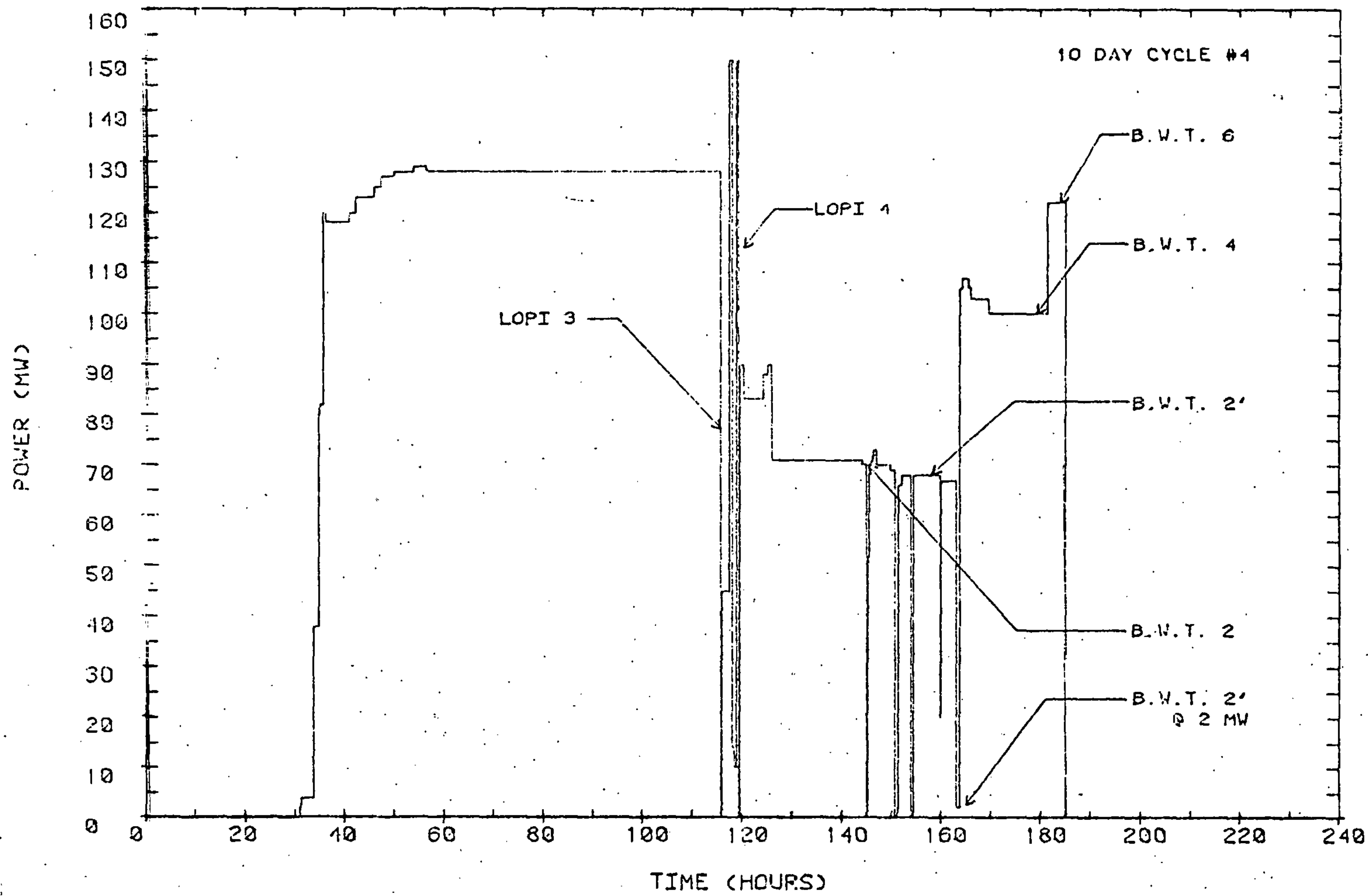


Figure 11. ETR POWER HISTORY FOR THE W-1 SLSF EXPERIMENT

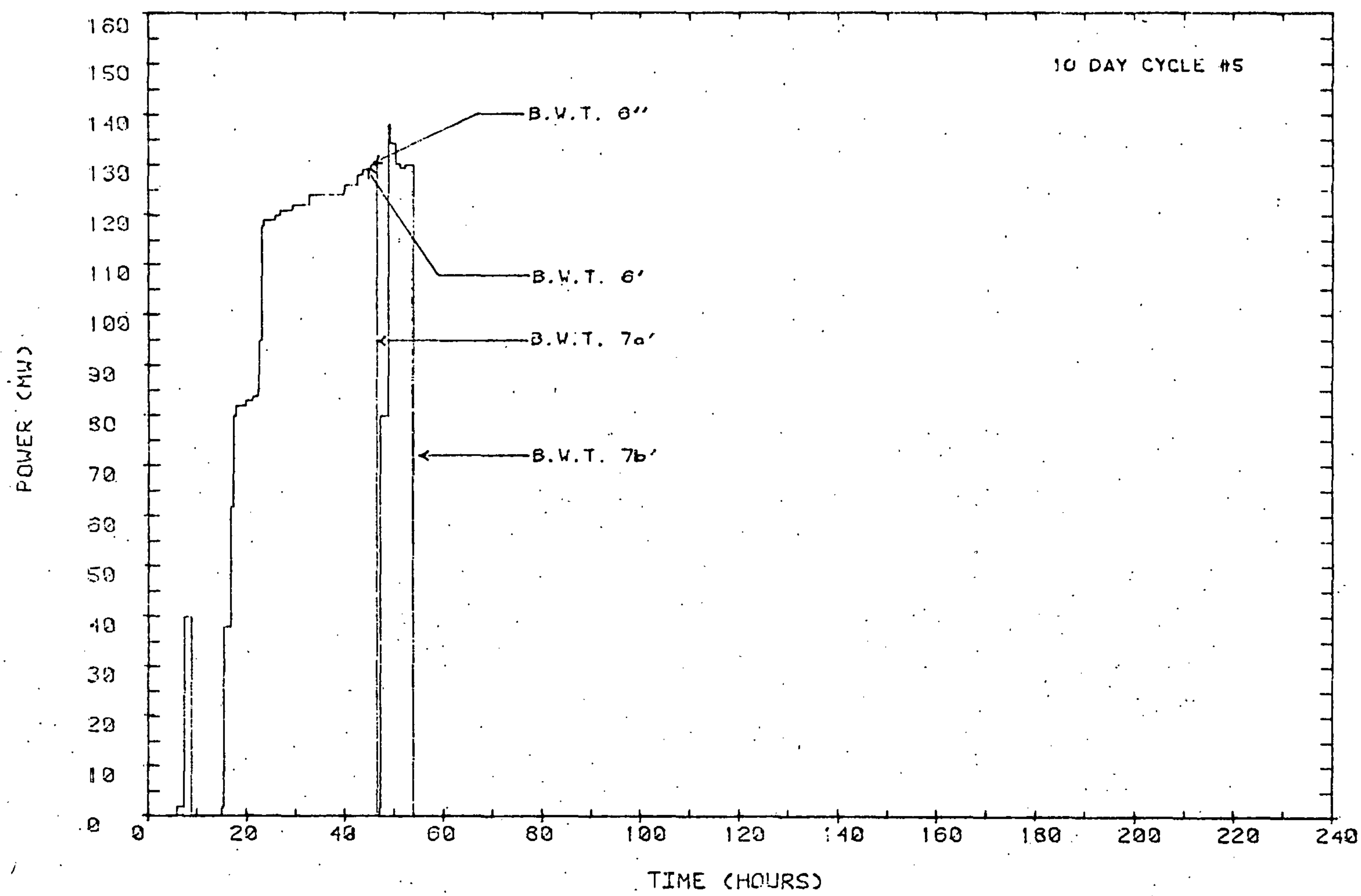


Figure 12. ETR POWER HISTORY FOR THE W-1 SLSF EXPERIMENT

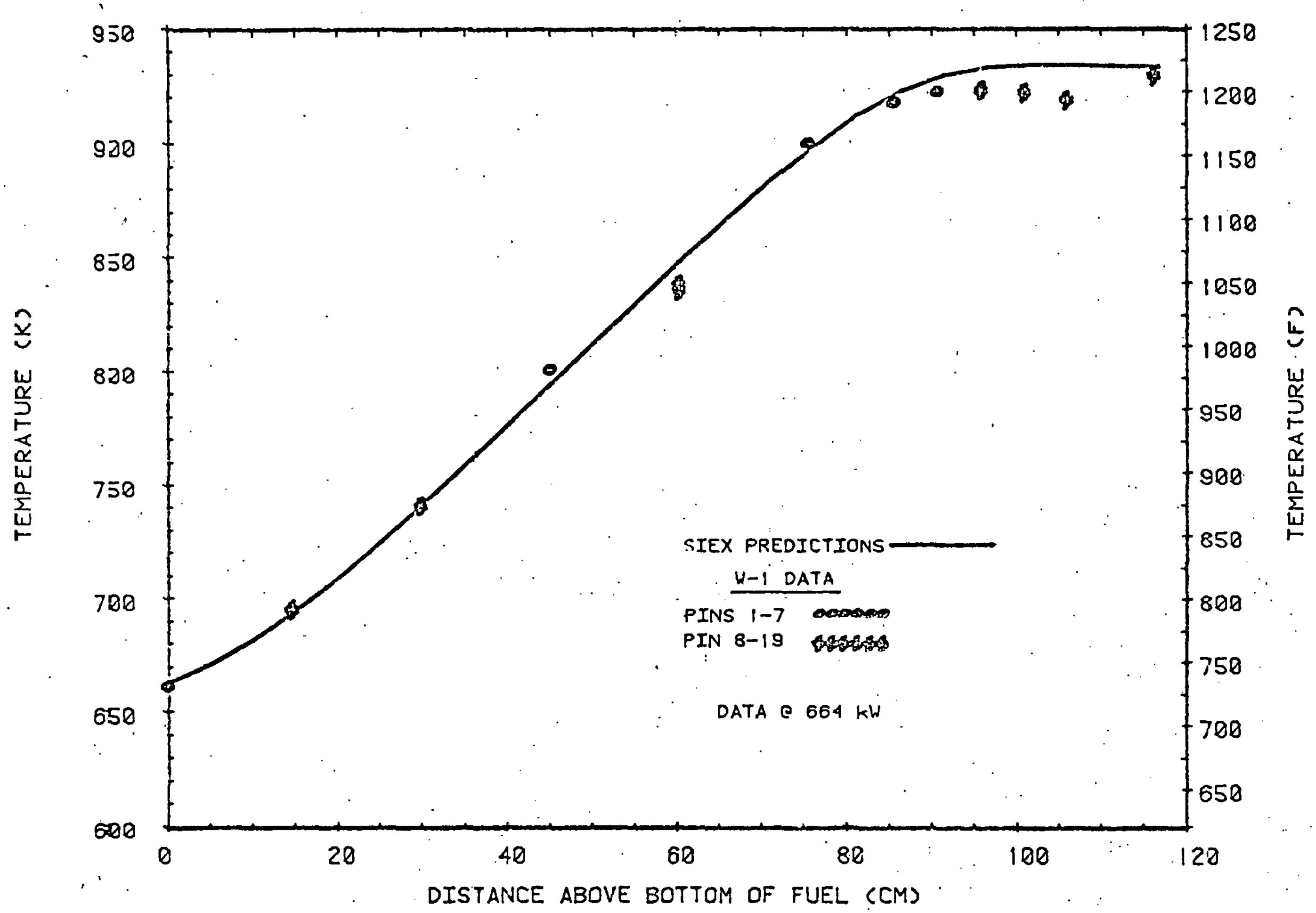


Figure 13. W-1 SLSF EXPERIMENT AXIAL COOLANT TEMPERATURE PROFILE

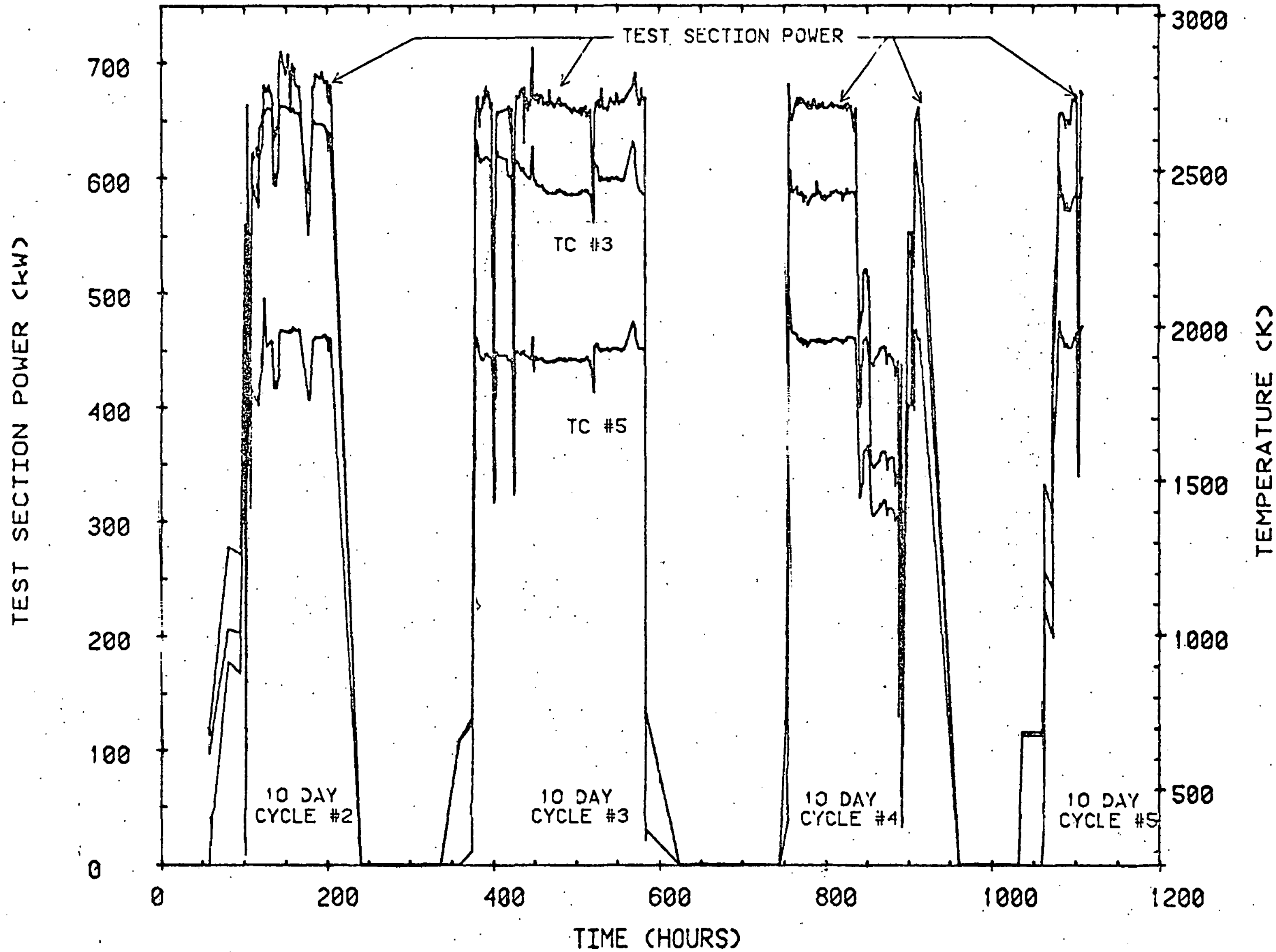


Figure 14. W-1 SLSF EXPERIMENT IRRADIATION HISTORY
TEST SECTION POWERS & FUEL CENTERLINE TEMPERATURES

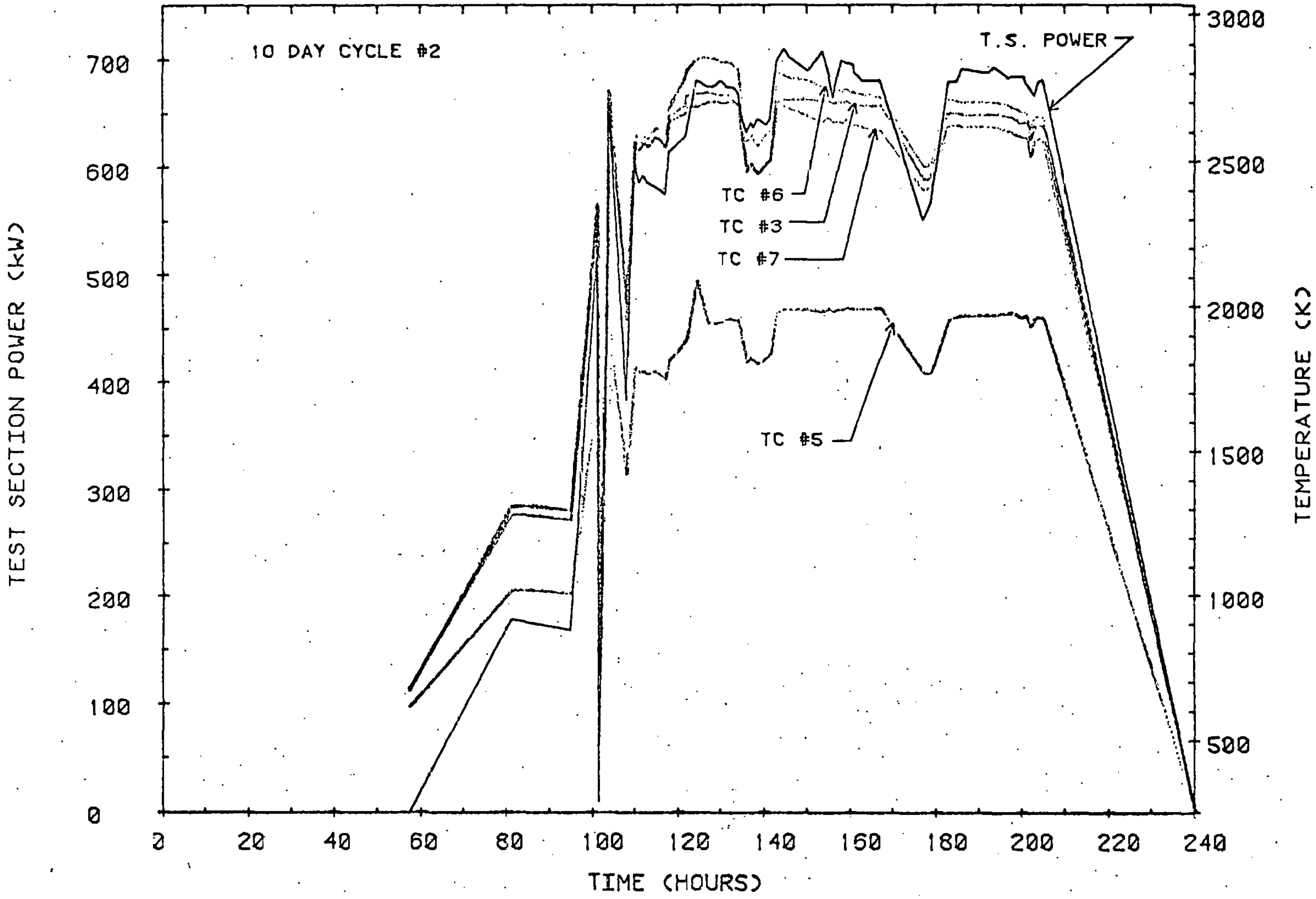


Figure 15. W-1 SLSF EXPERIMENT IRRADIATION HISTORY
TEST SECTION POWERS & FUEL CENTERLINE TEMPERATURES

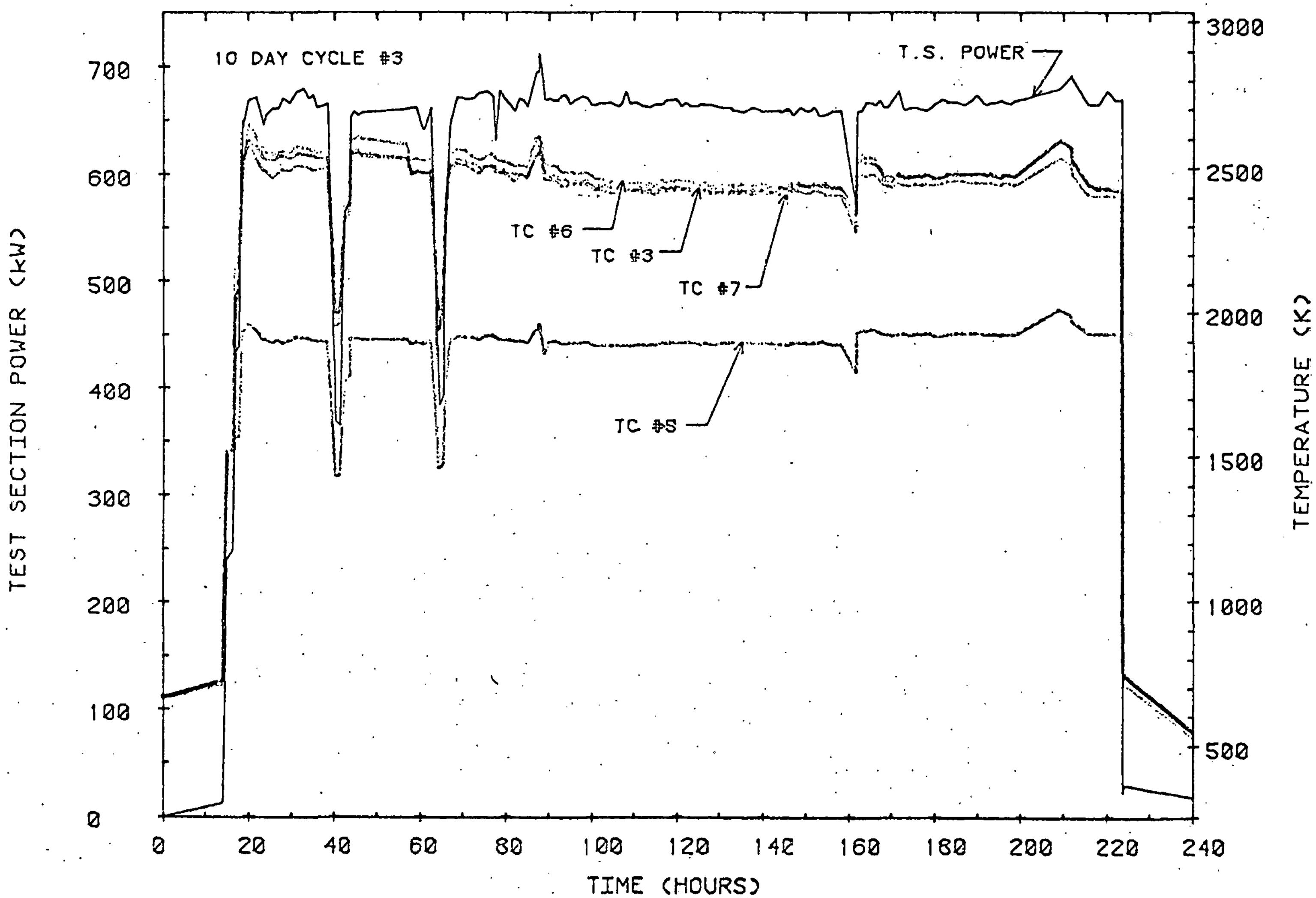


Figure 16. W-1 SLSF EXPERIMENT IRRADIATION HISTORY
TEST SECTION POWERS & FUEL CENTERLINE TEMPERATURES

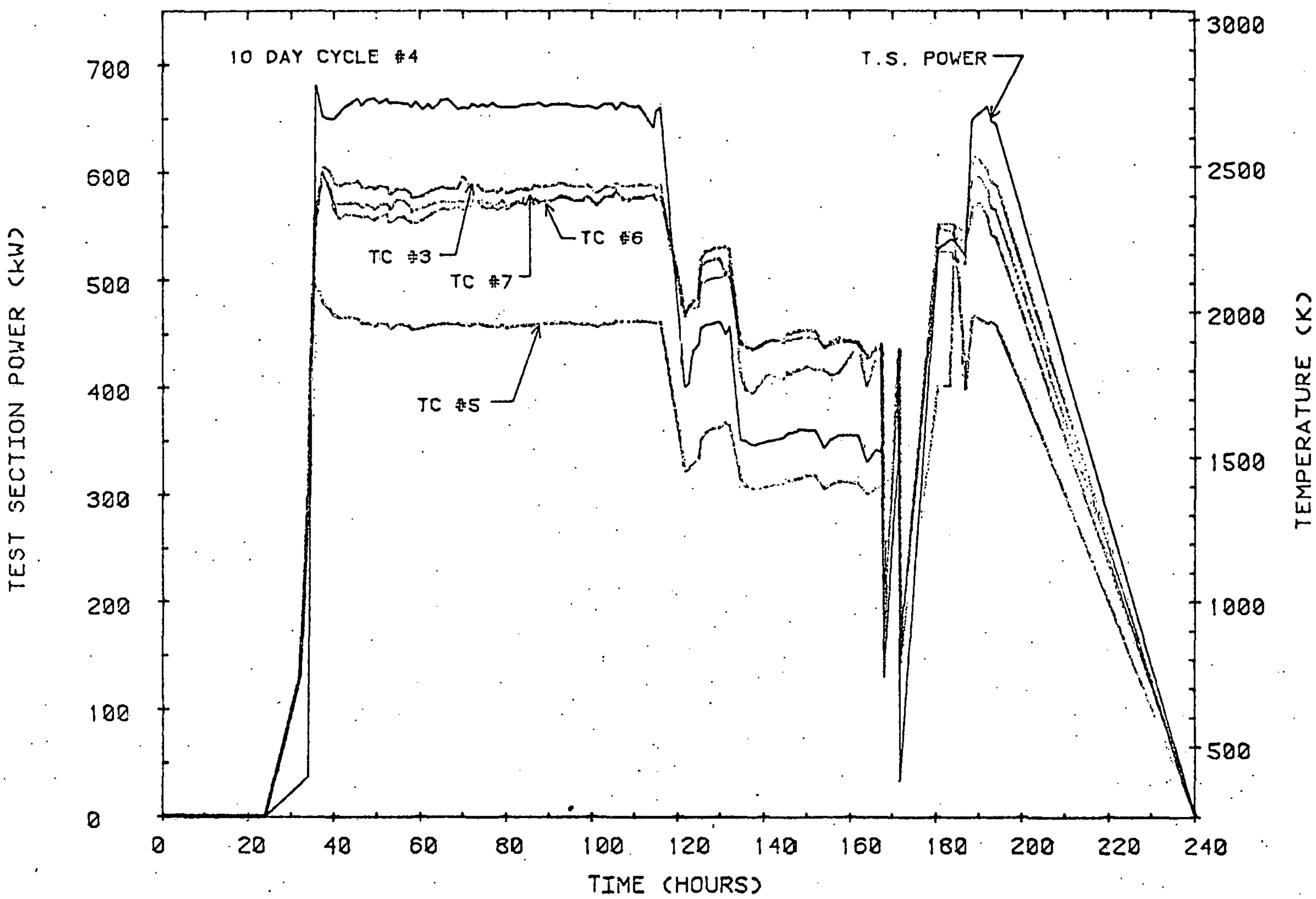


Figure 17. W-1 SLSF EXPERIMENT IRRADIATION HISTORY
TEST SECTION POWERS & FUEL CENTERLINE TEMPERATURES

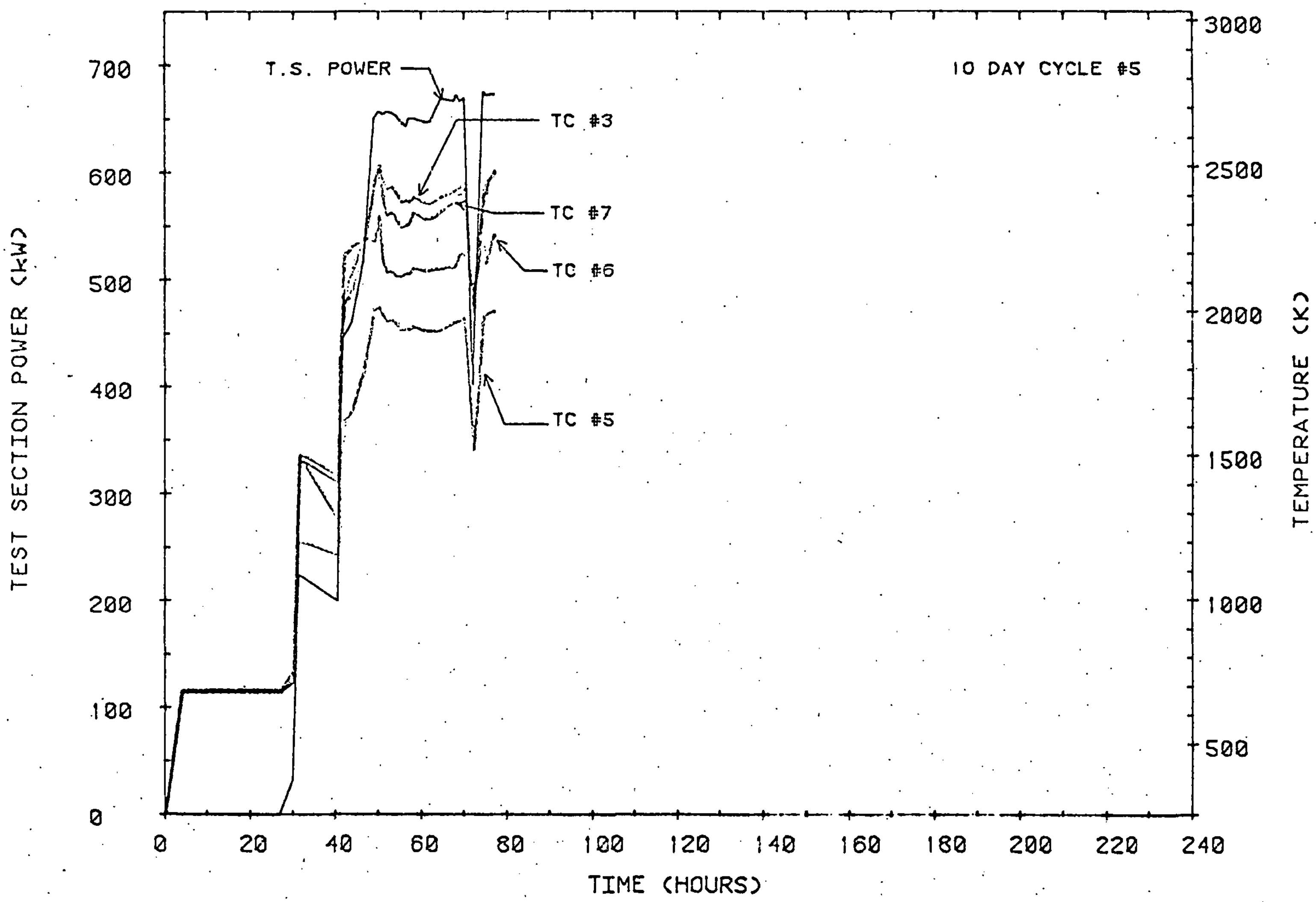


Figure 18. W-1 SLSF EXPERIMENT IRRADIATION HISTORY
TEST SECTION POWERS & FUEL CENTERLINE TEMPERATURES

C. Transient Operation

The following is a summary of the transients conducted during the W-1 experiment. Included are the date of transient and key operating information.

1. LOPI 1: June 9, maximum coolant temperature 1211 K (1720 F), no indication of boiling ($T_{SAT} \approx$ 1230.4 K (1755 F)); time at full power (472 W/cm [14.4 kW/ft] peak linear power) prior to transient: ~15 minutes.

2. LOPI 2: June 12, maximum coolant temperature 1216.5 K (1730 F), no indication of boiling. Examination of data showed that the reactor power was high; time at full power prior to transient: ~2 days.

3. LOPI 2A: June 21, maximum coolant temperature 1214.3 K (1726 F), no indication of boiling. (Figure 19); time at full power prior to transient: 4 days.

4. LOPI 3: July 9, maximum coolant temperature 1197 K (1695 F), no indication of boiling: total time at full power prior to transient: ~15 days.

5. LOPI 4: July 9, maximum coolant temperature 1229.3 K (1753 F), data showed about 0.5 sec of boiling. Examination of data revealed that LOPI 4 was run at approximately 5% overpower (496 W/cm (15.12 kW/ft)) due to failure of one of the thermocouples used for test section power calculations. Time at full power prior to transient: 10 minutes.

6. Boiling Window Test (BWT) 2: July 11, reactor scram occurred 3.5 seconds into the transient necessitating a repeat of the transient.

7. BWT2': July 11, approximately 0.8 second of boiling observed; maximum coolant temperature 1224.3 K (1744 F) @ 246 W/cm (7.5 kW/ft) and 24% of full flow for 4.0 seconds (Figure 20).

8. BWT4: July 12, approximately 1.0 second of boiling observed; maximum coolant temperature 1244.3 K (1780 F) @ 364 W/cm (11.1 kW/ft) and 35% of full flow for 4.0 seconds (Figure 21).

9. BWT6: July 13, approximately 0.2 seconds of boiling observed at the end of the transient; maximum coolant temperature 1244.3 K (1780 F) @ 472 W/cm (14.4 kW/ft) and 45% of full flow for 3 seconds.

10. BWT6': July 19, suspect low test section power; just touched saturation temperature @ just less than 472 W/cm (14.4 kW/ft) and 45% of full flow for 3.5 seconds.

11. BWT6": July 19, approximately 0.9 second of coolant boiling; maximum coolant temperature 1277.6 K (1840 F) @ 472 W/cm (14.4 kW/ft) and 45% of full flow for 3.5 seconds.

12. BW7A': July 20, approximately 1.6 seconds of coolant boiling observed, maximum coolant temperature 1284.3 K (1852 F); no indication of fuel pin failure @ 472 W/cm (14.4 kW/ft) and 40% of full flow for 3.0 seconds.

13. BW7B': July 20, approximately 2.0 seconds of boiling occurred before cladding dryout @ 472 W/cm (14.4 kW/ft) and 38% of full flow for 3.0 seconds. Details are given in Table 1 and Figure 22 .

After the conclusion of the W-1 experiment, the SAS3D code was utilized to simulate the final W-1 transient involving cladding dryout and pin failure. The timing of significant events during the transient is shown in Table .

Figure 23 shows the coolant temperature measured near the top of the fueled region with a W-1 wire-wrap thermocouple, along with coolant and cladding temperatures calculated by the SAS code for the dryout transient.

Predicted coolant temperatures matched the W-1 thermocouple data very closely (± 25 K) at this elevation. Thermocouple response time adjustments to the SAS coolant temperature predictions would make the agreement much closer through the last 2 seconds of the transient. It is apparent that the SAS prediction of cladding melting and fuel pin failure are conservative.

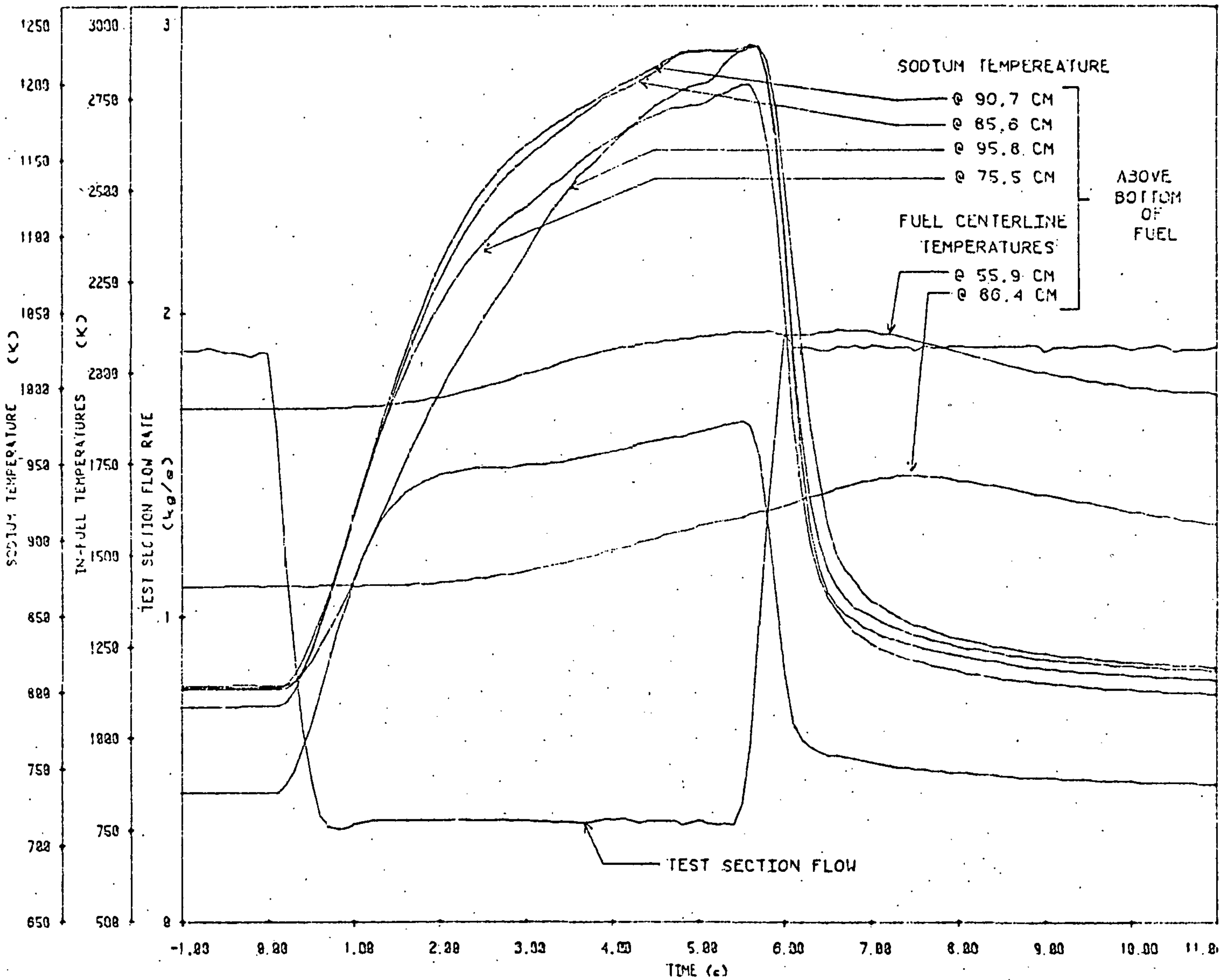


Figure 20. W-1 SLSF EXPERIMENT BOILING WINDOW TEST 2'-

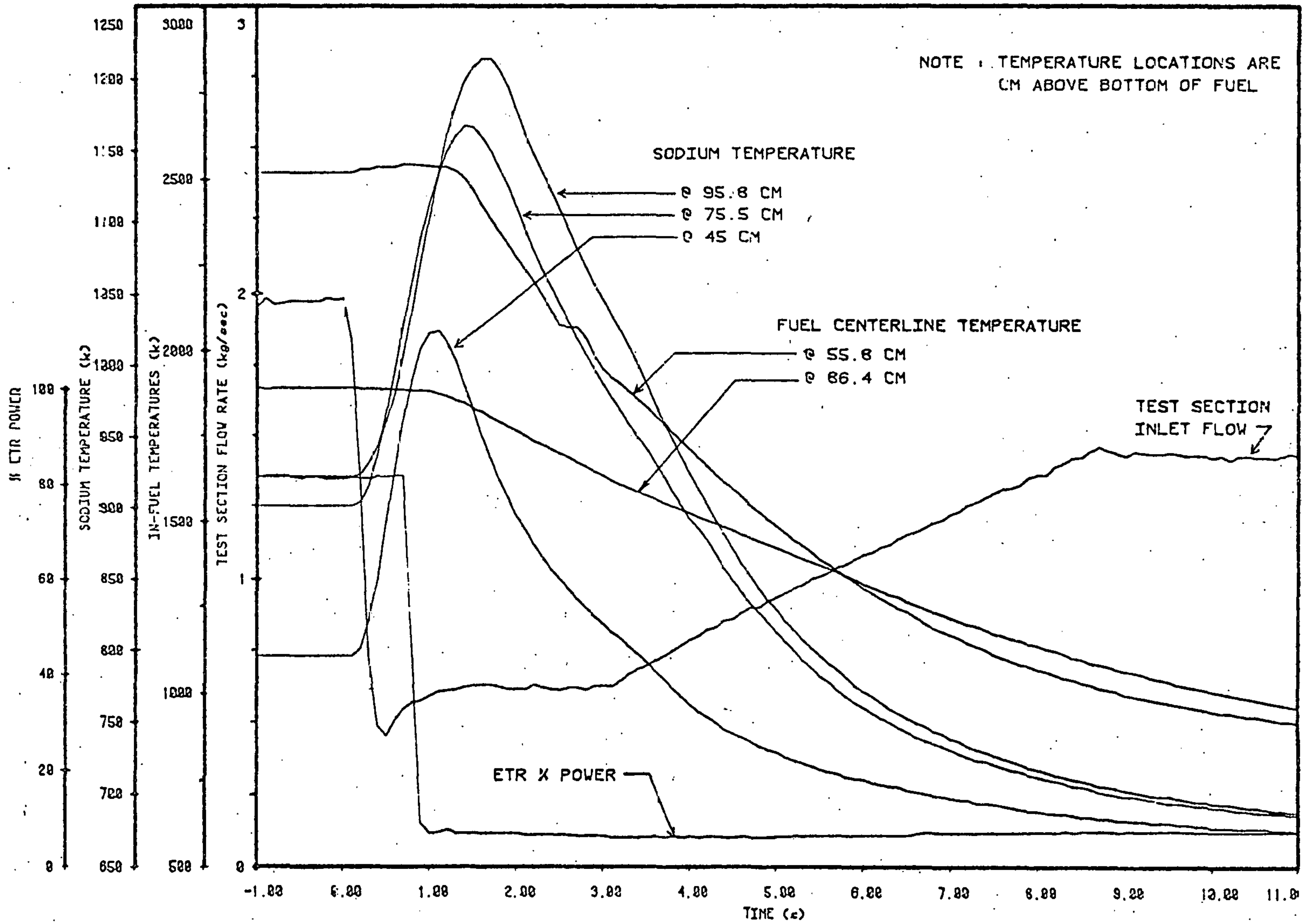


Figure 19. W-1 SLSF EXPERIMENT LOPI TRANSIENT 2A

TABLE 1

PRELIMINARY SCENARIO OF BOILING WINDOW

TEST 7B'

	Time	Event
1.	0 Sec	Start of Transient
2.	1.5 Sec	Test section inlet flow decay due to boiling in bundle.
3.	1.75 Sec	Boiling initiation measured by center channel wire wrap thermocouples at the top of the fuel (TOF) region of the bundle.
4.	1.9 Sec	Boiling indicated by wire wrap thermocouples in first ring coolant channels at the TOF region of the bundle.
5.	2.1 Sec	Boiling indicated by hex can inner face thermocouples at the TOF region of the bundle. This shows a radial progression of boiling in the bundle.
6.	2.8 Sec	Inlet flow reversal and start of test section flow oscillations.
7.	3.1 Sec	Boiling indicated by wire wrap thermocouples at fuel bundle axial midplane. Clear indication of axial progression of boiling.
8.	3.5 Sec	Reactor Scram. Dryout conditions measured by center channel wire wrap thermocouples and hex can inner face thermocouples at the fuel bundle axial midplane.
9.	3.7 Sec	Dryout measured by wire wrap thermocouples and hex can inner face thermocouples at the TOF region of the bundle.
10.	3.8 Sec	Inlet flow recovers to greater than 0 lbm/s and starts recovery to full flow conditions.
11.	3.9 Sec	In-fuel thermocouples measured peak fuel temperatures about 2811 K (4600°F). No apparent fuel melting.
12.	4.0 Sec	Peak wire wrap thermocouple temperatures of 1374°K (2013°F). Dryout condition over 18" of fuel column length.
13.	5.6 Sec	Inlet flow back to full flow conditions (Programmed full flow @ 4.0 sec).
14.	1-5 Min.	SLSF radiation area monitors in ETR TOP Dome show rapid rise in count rate.
15.	5-6 Min.	On-Line Cover Gas Sampling System (OLCS) detection of fuel pin cladding breach.

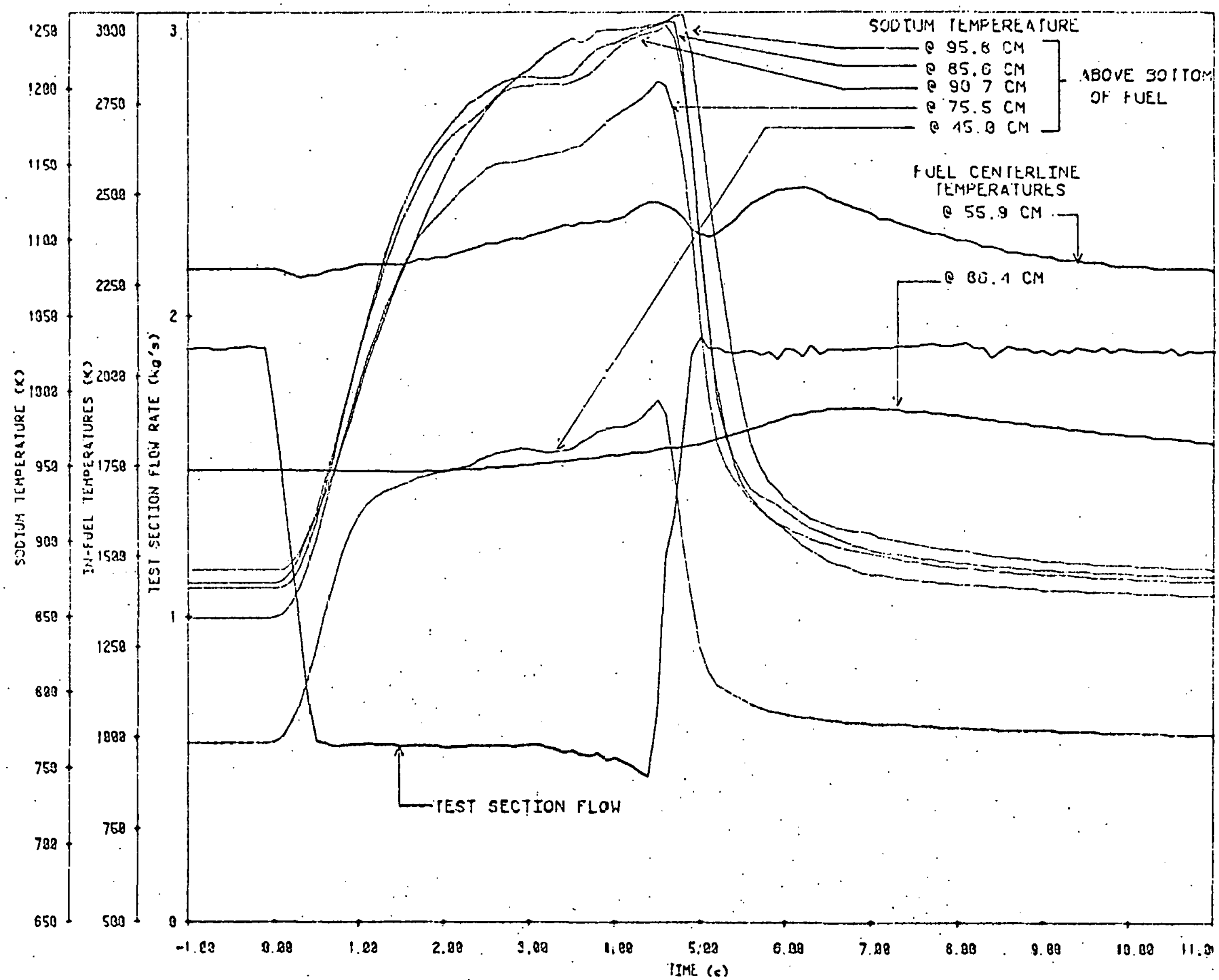


Figure 21. W-1 SLSF EXPERIMENT BOILING WINDOW TEST 4

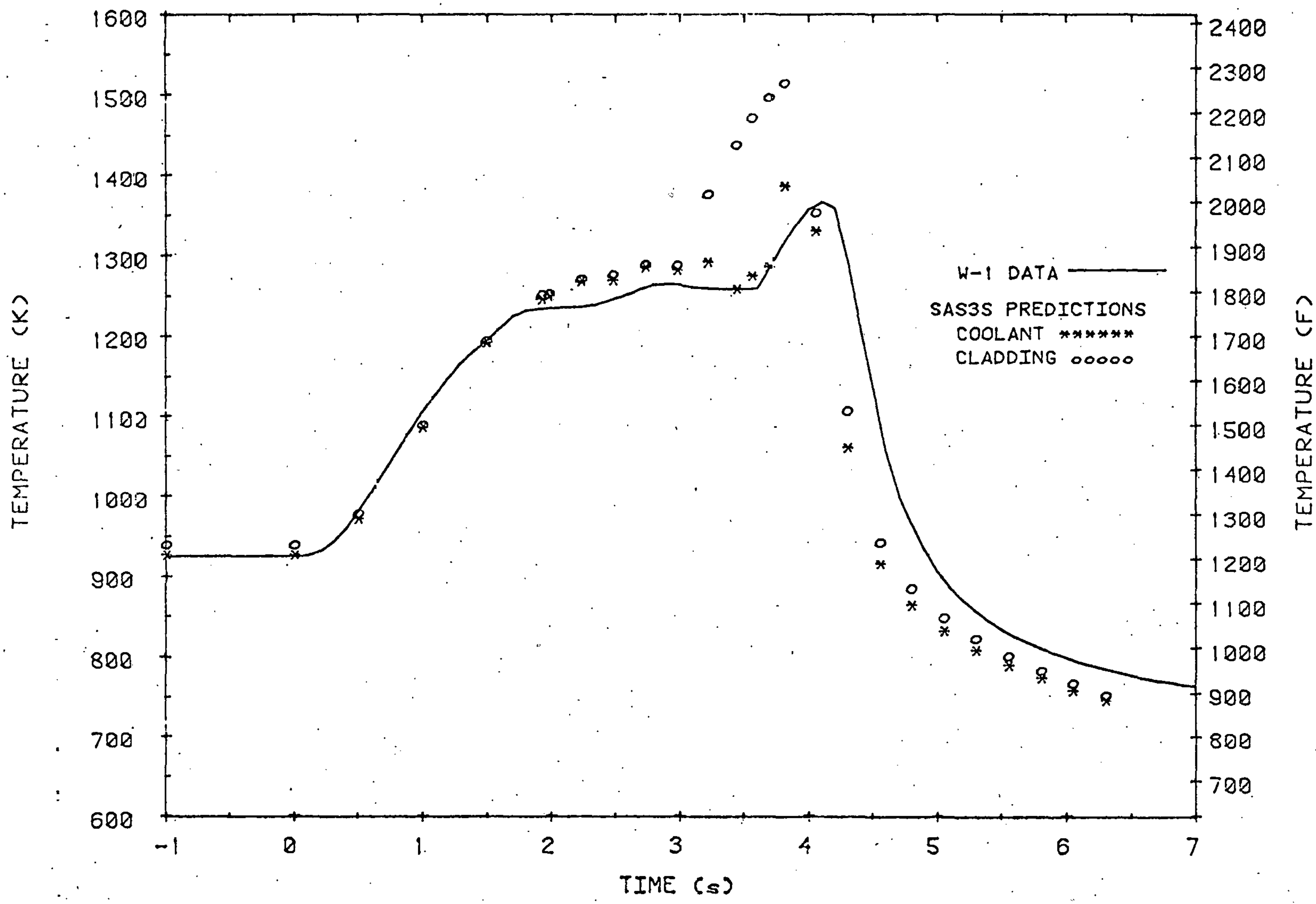


Figure 23. W-1 SLSF EXPERIMENT BOILING WINDOW TEST 7b'
WIRE-WRAP THERMOCOUPLE DATA VS SAS3D PREDICTIONS

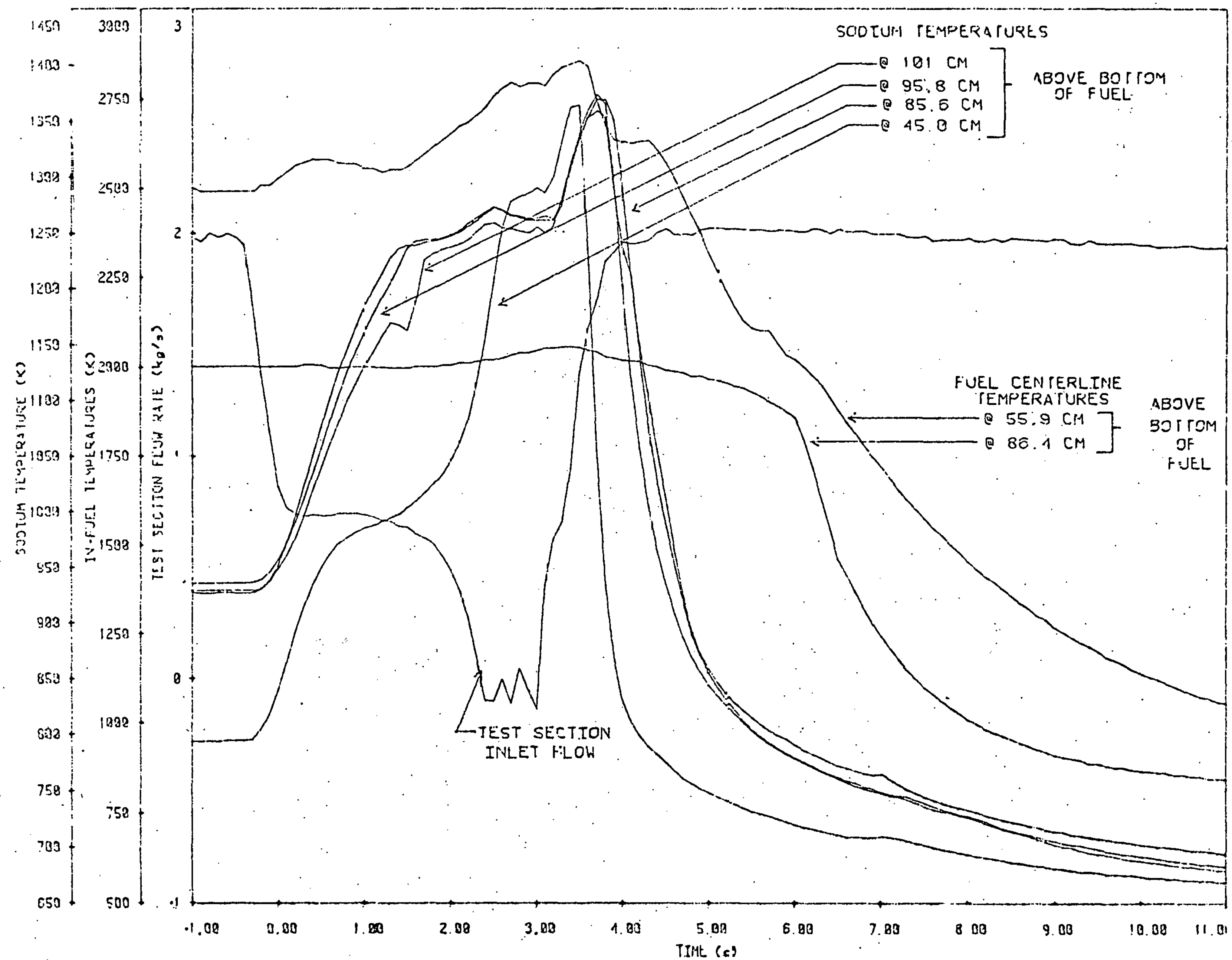


Figure 22. W-1 SLSF EXPERIMENT BOILING WINDOW TEST 7b'

TABLE

Event	Initial Occurrance Seconds into Transient W-1	SAS
Boiling	1.9	1.91
Dryout	3.5	2.75
Cladding Melting	~4.0	3.39