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SODIUM BOILING AND MIXED OXIDE FUEL THERMAL BEHAVIOR IN FBR UNDERCOOLING TRANSIENT; W-1 SLSF EXPERIMENT RESULTS

J.M. Henderson
S.A. Wood
D.D. Knight (GE)

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J. M. Henderson and S. A. Wood

Westinghouse Hanford Co.
P.O. Box 1970
Richland, WA 99352

D. D. Knight

General Electric Company
Advanced Reactor Systems Department
P.O. Box 508
Sunnyvale, CA 94086

ABSTRACT

The W-1 Sodium Loop Safety Facility (SLSF) Experiment was conducted to study fuel pin heat release characteristics during a series of LMFBR Loss-of-Piping Integrity (LOPI) transients and to investigate a regime of coolant boiling during a second series of transients at low, medium and high bundle power levels.

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During the last of seven boiling transients, intense coolant boiling produced inlet flow reversal, cladding dryout and moderate cladding melting.

INTRODUCTION

The W-1 Sodium Loop Safety Facility (SLSF) Experiment was the fifth in a series of such experiments sponsored by the Department of Energy as part of the National Fast Breeder Reactor (FBR) Safety Development Program. The Hanford Engineering Development Laboratory (HEDL) has prime responsibility for the W-1 experiment and it was conducted in cooperation with the Advanced Reactor Systems Department of General Electric Company (GE/ARSD). The facility, operated by EG&G Idaho, Inc., is located in the Engineering Test Reactor at the Idaho National Engineering Laboratory near Idaho Falls, Idaho.

W-1 SLSF EXPERIMENT OBJECTIVES

The W-1 experiment had two distinct objectives. The first objective was to evaluate fuel pin heat release characteristics during Loss-of-Piping Integrity (LOPI) accidents. A sequence of four LOPI transients was conducted to collect data at different fuel pin conditions of: a) fresh, unrestructured fuel, b) fresh, restructured fuel, c) irradiated, restructured fuel (with startup cracks healed), and d) irradiated, cracked fuel (after shutdown and startup).

The second objective of the W-1 experiment was to determine the sodium boiling and recovery limits as a function of fuel pin power and coolant flowrate. The eight boiling tests concluded with intentional fuel pin dryout and cladding failure.

These objectives support resolution of safety issues in the third level of defense (LOA-3: maintain containment integrity) in the Department of Energy Fast Reactor Safety Program Plan. Furthermore, the sodium boiling data will expand the data base already obtained in the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility at Oak Ridge National Laboratory.

TEST VEHICLE DESCRIPTION

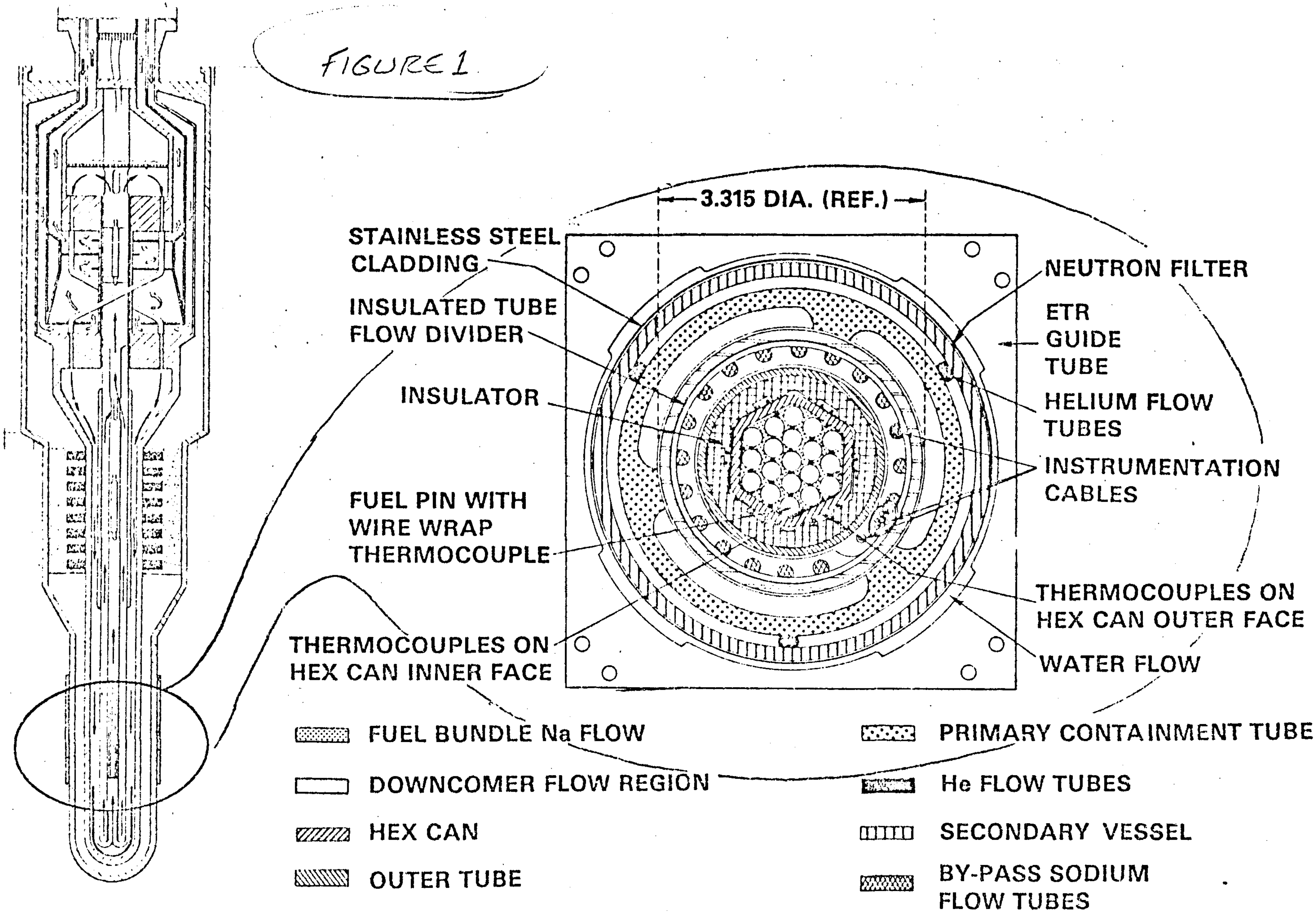
The SLSF in-pile loop (Figure 1), located in the Engineering Test Reactor, is a doubly-contained closed sodium loop test vehicle 8.23 m (27 ft) long and weighing approximately 3400 kg (7500 lbm). The loop consists of a primary and secondary containment vessel, an annular linear induction electromagnetic 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, removable top closure (RTC), and the instrumented test train.

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

SLSF IN-PILE LOOP CROSS SECTION

CROSS-SECTION OF THE W-1 SLSF IN THE LOWER TEST SECTION REGION

FIGURE 1



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SUMMARY

The irradiation phase of the W-1 experiment was conducted between May 27 and July 20, 1979 (Figure 2). The experiment hardware and facility performed as designed, allowing completion of all planned tests (13 transients total).

Test data revealed:

1. Experimental conditions simulating a CRBR LOPI accident at 15% overpower and hot channel conditions did not produce coolant boiling and only minor effects of fuel preconditioning were observed in overall fuel bundle transient thermal performance. These data have been useful in evaluating current transient fuel pin heat release models.

2. The boiling tests verified GE/ARSD predictions of boiling inception and extended the US FBR data base for coolant boiling to 472 W/cm peak linear pin power. These data identify boiling incipience over a range of reactor operating conditions and lend credibility to current thermal-hydraulic codes capable of predicting coolant boiling inception.

3. There was no discernable preboiling coolant superheat in any transient. These data reduced concerns over coolant superheat and subsequent flashing during postulated FBR accidents.

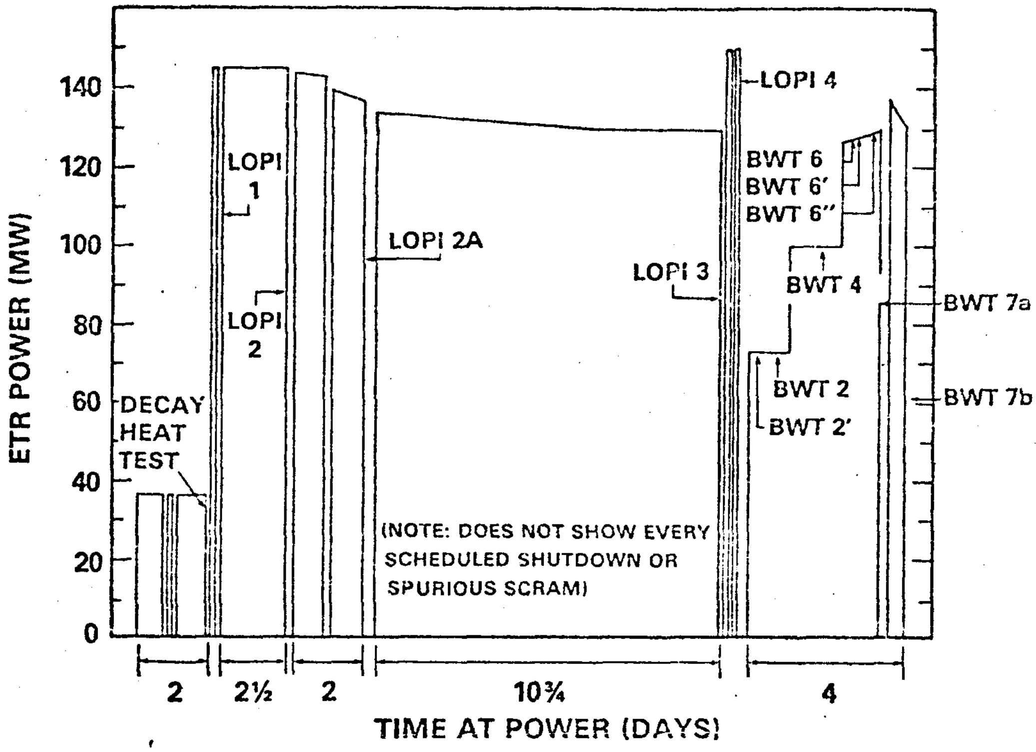
4. The boiling test data clearly show the three-dimensional aspects of coolant void progression. These data are directly applicable to the evaluation of sodium coolant boiling models, once the size and heat sink effect of the 19-pin bundle are accounted for.

5. In the final boiling transient, two seconds of coolant boiling was produced. The upper half of the fuel bundle voided, resulting in test section inlet flow reversal, cladding dryout and the expected fuel pin failure.

Post-test examination revealed that eight of the nineteen fuel pins incurred cladding breaches. Four of the breached pins showed definite cladding melting and relocation in both axial directions. Two inner ring fuel pins were self-welded to the center fuel pin. Five of the breached pins incurred circumferential, brittle fracture type cracks, suggesting cladding breaches caused by rapid quenching during the post-transient flow return.

In addition to the coolant boiling data obtained during the W-1 experiment, unique steady-state fuel pin performance data were obtained from the in-fuel and wire wrap thermocouples. Data showing the effects of fuel restructuring and gap closure and the effects of startup cracking and crack healing on fuel temperatures were obtained from the four in-fuel thermocouples which survived the entire experiment. These data, along with detailed coolant temperature profile data, were used to analyze fuel pin heat transfer models in the SIEX[1] fuel pin performance computer code.

FIGURE 2. ETR POWER HISTORY FOR THE W-1 SLSF EXPERIMENT



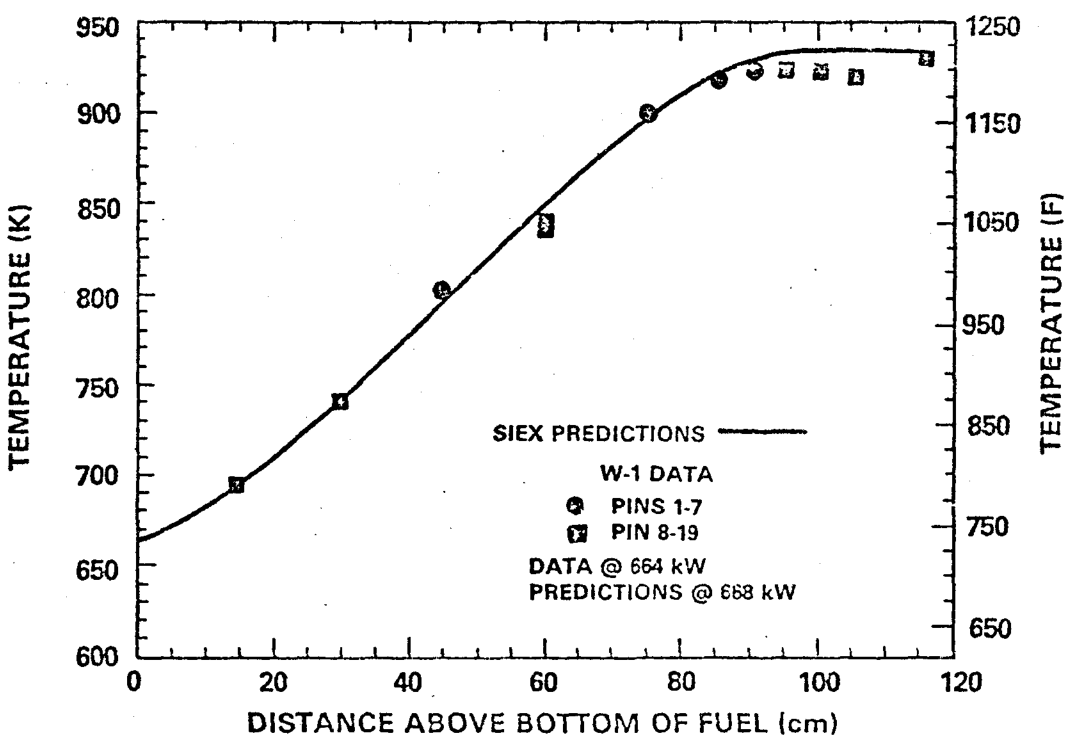


FIGURE 3. W-1 SLSF EXPERIMENT AXIAL COOLANT TEMPERATURE PROFILE

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STEADY STATE DATA

Coolant temperature data recorded during steady-state operations were very close to pretest predictions (Figure 3). 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 fuel restructuring models in fuel pin thermal performance codes.

TRANSIENT DATA

Twelve major transients were conducted during the W-1 experiment. The results of these transients are summarized in Table 1. Four LOPI transients were originally scheduled; five were conducted. LOPI 2A, a repeat of LOPI 2, was required because a premature reactor scram occurred during LOPI 2.

The LOPI transients were conducted from full fuel bundle power conditions of 658 kW (nominal), yielding 384.5 W/cm (11.7 kW/ft) average fuel pin linear power. This corresponds to a 15% overpower condition of the highest power FFTF fuel bundle.

During the power maneuvering just prior to LOPI 4, one of the thermocouples used in the heat balance calculations failed, causing an error in the initial LOPI 4 power level (i.e., an additional 7% overpower condition). Fuel bundle wire-wrap thermocouples indicated that saturated coolant conditions existed for approximately 0.5 seconds.

With the exception of the overpower condition during LOPI 4, the maximum coolant temperatures dropped slightly as the fuel bundle irradiation time increased. The fuel centerline temperatures also dropped as fuel restructuring progressed.

The Boiling Window Test series was designed to investigate a hypothesized regime of power-to-coolant flowrate ratios which could cause "stable" coolant boiling conditions. The sodium boiling tests were conducted at three different average fuel pin power levels; 1) 200 W/cm (6.1 kW/ft), 2) 296 W/cm (9.0 kW/ft), and 3) 384.5 W/cm (11.7 kW/ft).

Radial and axial expansion of the boiling front was evident from the wire-wrap thermocouple coolant temperature data. The data clearly show that at higher heat flux levels, the boiling tends to be more intense. The fuel bundle inlet flow reduction is caused by an increased pressure drop generated by sodium void progression through the fuel bundle coolant channels.

During Boiling Window Test 7b', the sodium void progression and corresponding pressure drop increase caused inlet flow reversal, cladding dryout and moderate cladding melting. The transient was stopped by an ETR scram at 3.5 seconds and a return to full coolant pump power. Details of Boiling Test 7b' are given in Table 2.

TABLE 1

W-1 SLSF EXPERIMENT
TRANSIENT SUMMARY

Transient	Average Pin Power	Minimum Flow*	Peak Na Temperature	Boiling Duration
LOPI 1	377 W/cm	0.47 Kg/s	1211 K	N/A
LOPI 2	410 W/cm	0.47 Kg/s	1216 K	N/A
LOPI 2A	385 W/cm	0.47 Kg/s	1216 K	N/A
LOPI 3	377 W/cm	0.47 Kg/s	1200 K	N/A
LOPI 4	411 W/cm	0.47 Kg/s	1230 K	0.5 s
BWT 2'	207 W/cm	0.47 Kg/s	1224 K	0.8 s
BWT 4	313 W/cm	0.68 Kg/s	1250 K	1.0 s
BWT 6	386 W/cm	0.88 Kg/s	1247 K	0.2 s
BWT 6'	391 W/cm	0.88 Kg/s	1249 K	0.1 s
BWT 6"	396 W/cm	0.88 Kg/s	1280 K	0.9 s
BWT 7a'	393 W/cm	0.78 Kg/s	1288 K	1.6 s
BWT 7b'	391 W/cm	0.75 Kg/s	1374 K	2.0 s

* Programmed fuel bundle inlet low flow point

TABLE 2

BOILING TEST 7b'

Time	Event
0 s	Start of transient.
1.5 s	Boiling initiation - detected by inlet flow reduction
1.5 - 2.1 s	Boiling progression from center-to-edge-of-bundle at the top-of-the-fuel (TOF) region of the bundle.
2.8 s	Inlet flow reversal.
3.1 s	Boiling at axial midplane.
3.5 s	ETR scram. Cladding dyrout at the fuel bundle axial midplane.

TABLE 2 (CON'T)

Time	Event
3.7 s	Dryout extended up to the TOF region of the fuel bundle.
3.8 s	Inlet flow recovery.
3.9 s	Peak fuel centerline temperature: 2800 K.
4.0 s	Peak wire-wrap thermocouple temperature measurement: 1374 K.
5.6 s	Full recovery of fuel bundle flowrate to pretransient levels.

POST-TEST ANALYSIS

The post-test analysis of the W-1 experiment was conducted to evaluate both the W-1 data and the capability of safety and design computer codes to predict the W-1 fuel bundle transient response.

The computer codes used to evaluate the W-1 data were SIEX,[1] COBRA-3[2] and SAS-3D.[3] The SIEX code was used to predict steady-state fuel pin thermal performance, restructuring, fission gas release, and the fuel-cladding gap conductance. The COBRA-3 code was used to analyze the fuel pin and coolant thermal-hydraulic response during the LOPI transients and Boiling Window Tests. COBRA-3 calculations were reliable only to boiling inception. The SAS-3D code was used to analyze boiling progression, fuel pin cladding dryout, and subsequent cladding melting.

The results of the SIEX calculations of fuel restructuring and fuel pin thermal performance are shown on Figure 4, where fuel centerline temperatures are plotted versus time. The SIEX code underpredicted fuel centerline temperatures by as much as 220 K (400°F). Possible sources of error are in the fuel thermal conductivity model, the fuel-cladding gap conductance model, or the radial power profile in the fuel.

In order to reduce uncertainty in the COBRA-3 transient calculations, an attempt was made to reduce the fuel temperature error by conducting a parametric study with the SIEX code. Parameters that affect the calculated fuel temperature were varied to determine which produced the best match to the measured fuel temperature data.

It was determined that lowering the fuel thermal conductivity provided a close match between fuel centerline temperature predictions and data at both elevations of the in-fuel thermocouples (TE 22-3 @ 56 cm and TE 22-5 @86 cm above the bottom of the fuel). Using a higher fuel-cladding gap xenon concentration, effectively decreasing the gap conductance, gave fuel temperatures that agreed with TE 22-3 data, but the predictions at the 86 cm elevation were much higher than the TE 22-5 data.

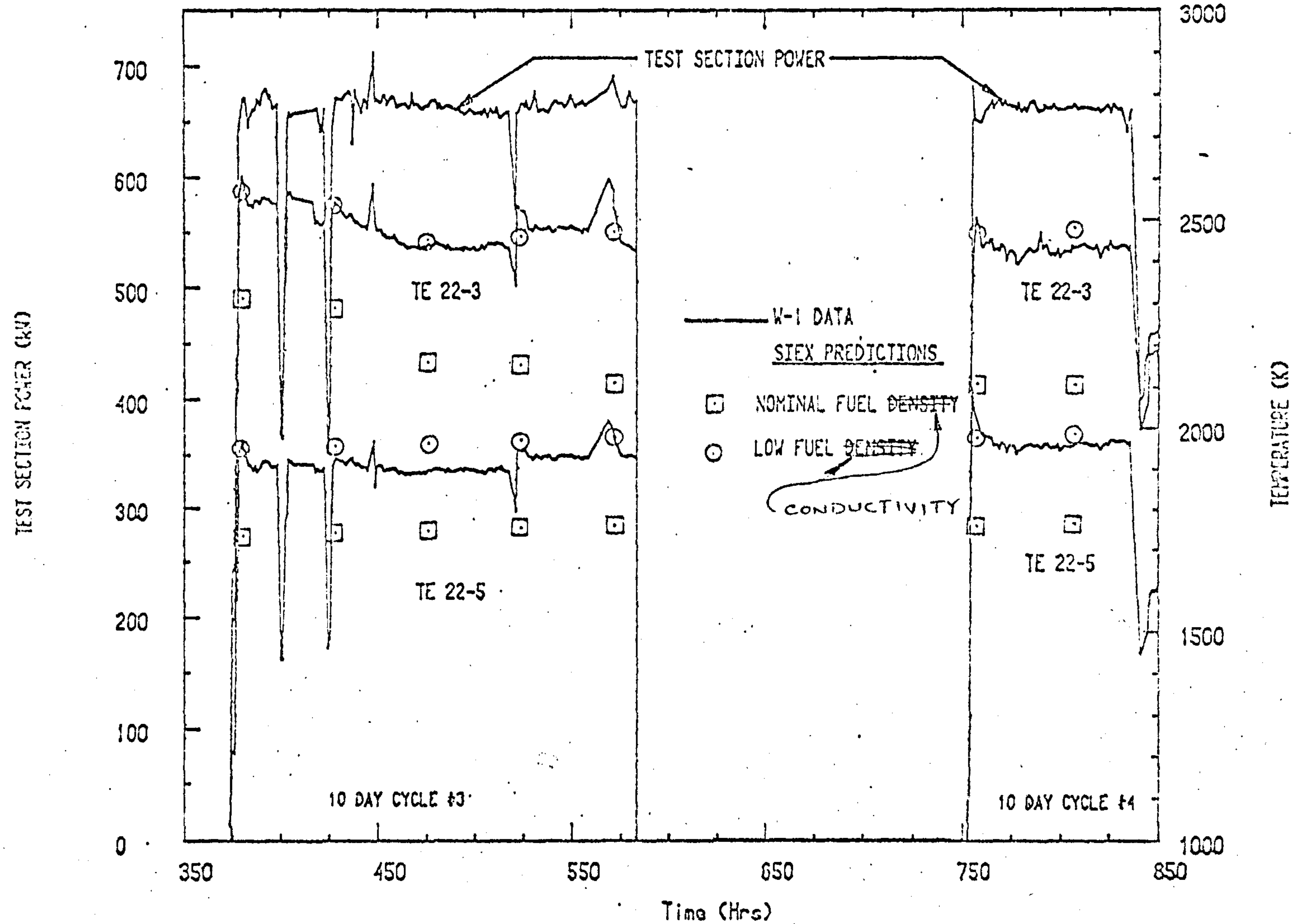

FIGURE 4.

W-1 SLSF EXPERIMENT IRRADIATION HISTORY
TEST SECTION POWER & FUEL CENTERLINE TEMPERATURES

The lower fuel conductivity case SIEX predictions of fuel restructuring and fission gas release matched the W-1 fuel pin post-irratiation examination data very closely. These predictions were therefore used as input to the COBRA-3 code for the thermal-hydraulic analysis of the W-1 transients.

The COBRA-3 predictions of coolant temperatures are plotted with test data for LOPI 3 and Boiling Test 7b' in Figures 5 and 6. The agreement between predicted wire-wrap thermocouple response and the actual W-1 data varies from good to excellent.

The SAS-3D calculation results of Boiling Test 7b' are shown in Figure 7. The predictions of coolant and cladding temperature are very close to the W-1 wire-wrap thermocouple data, demonstrating the effectiveness of the SAS-3D code in analyzing this transient.

CONCLUSIONS

LOPI ACCIDENT EVALUATION

The W-1 experiment LOPI accident simulation demonstrated that the protected CRBR LOPI accident (i.e., pipe rupture and corresponding flow reduction followed by an automatic plant scram) is a benign event. The W-1 LOPI transients were conducted at 384.5 W/cm (11.7 kW/ft) average (nominal) pin power levels. This corresponds to a 15% overpower condition for the highest power FFTF assembly. Only LOPI 4, conducted at 23% overpower conditions for FFTF, generated coolant boiling, the boiling was very mild, lasting only 0.5 seconds.

There were only small changes in coolant temperature during the progression of LOPI transients. Maximum coolant temperatures dropped with progressing fuel pin irradiation and the corresponding fuel microstructure changes. It appears that a LOPI type transient becomes less severe with increasing core burnup, although as previously mentioned, the change is small.

BOILING PROGRESSION

The W-1 coolant boiling tests were designed to extend the THORS facility sodium boiling data base into the normal range of operating powers for a full-size breeder reactor. One major observation in the shift to higher powers is the acceleration of boiling progression from the center coolant channels to the outer coolant channels. The corresponding increase in coolant pressure drop rapidly leads to flow reversal.

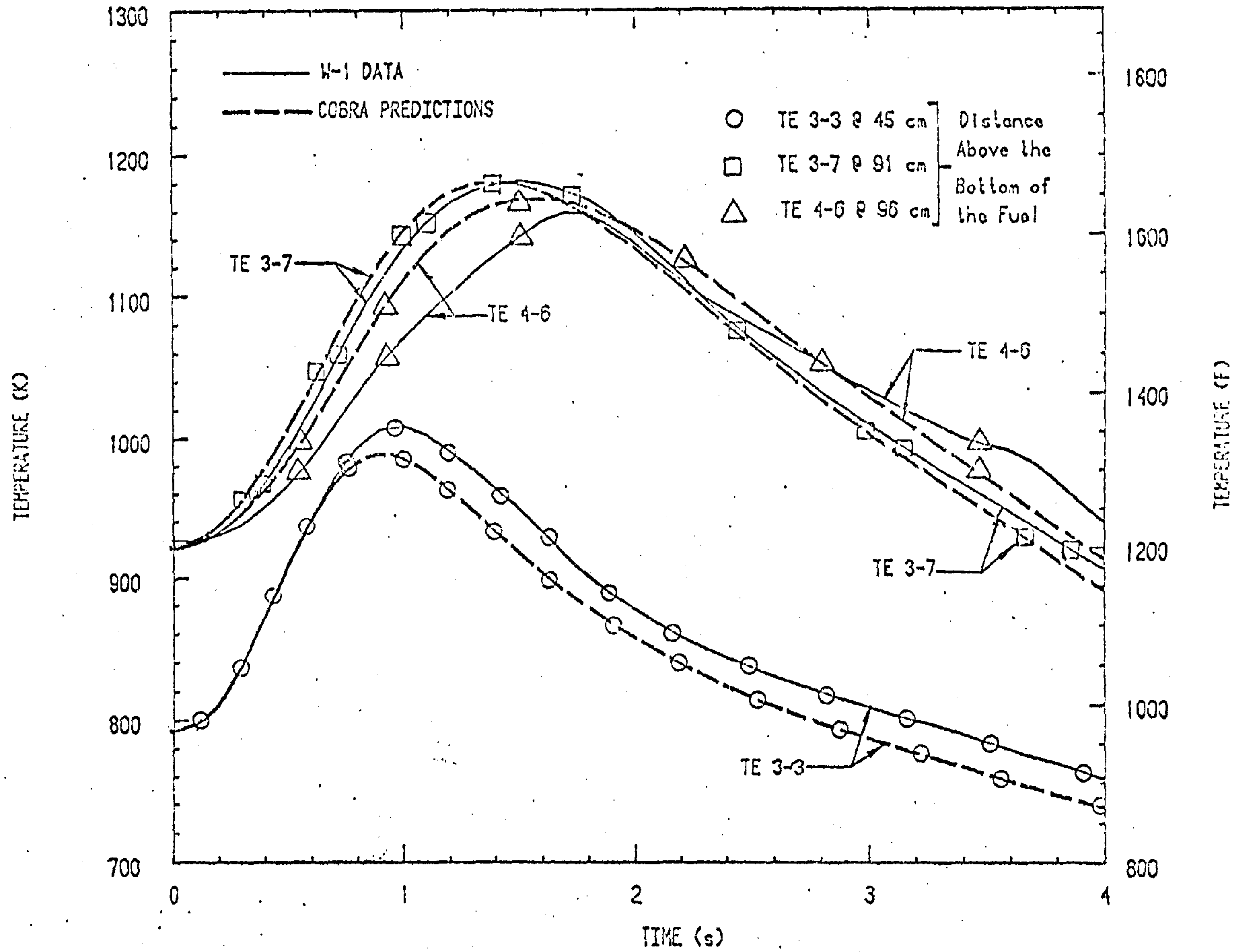

FIGURE 5.

W-1 SLSF EXPERIMENT LOSS-OF-PIPING INTEGRITY TRANSIENT #3
WIRE-WRAP THERMOCOUPLE DATA vs PREDICTIONS

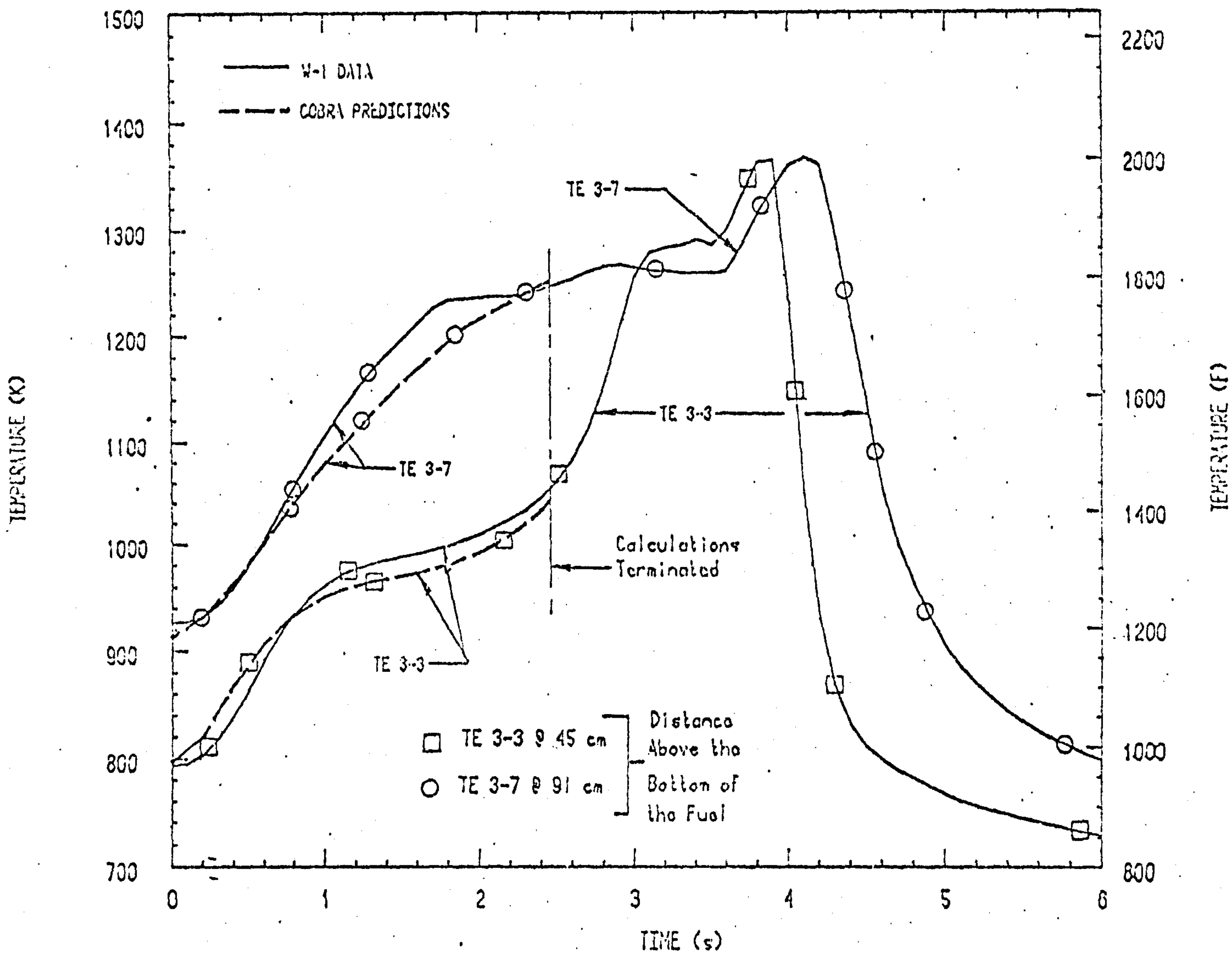

FIGURE 6.

W-1 SLSF EXPERIMENT HIGH POWER BOILING TEST 7b'
WIRE-WRAP THERMOCOUPLE DATA vs PREDICTIONS

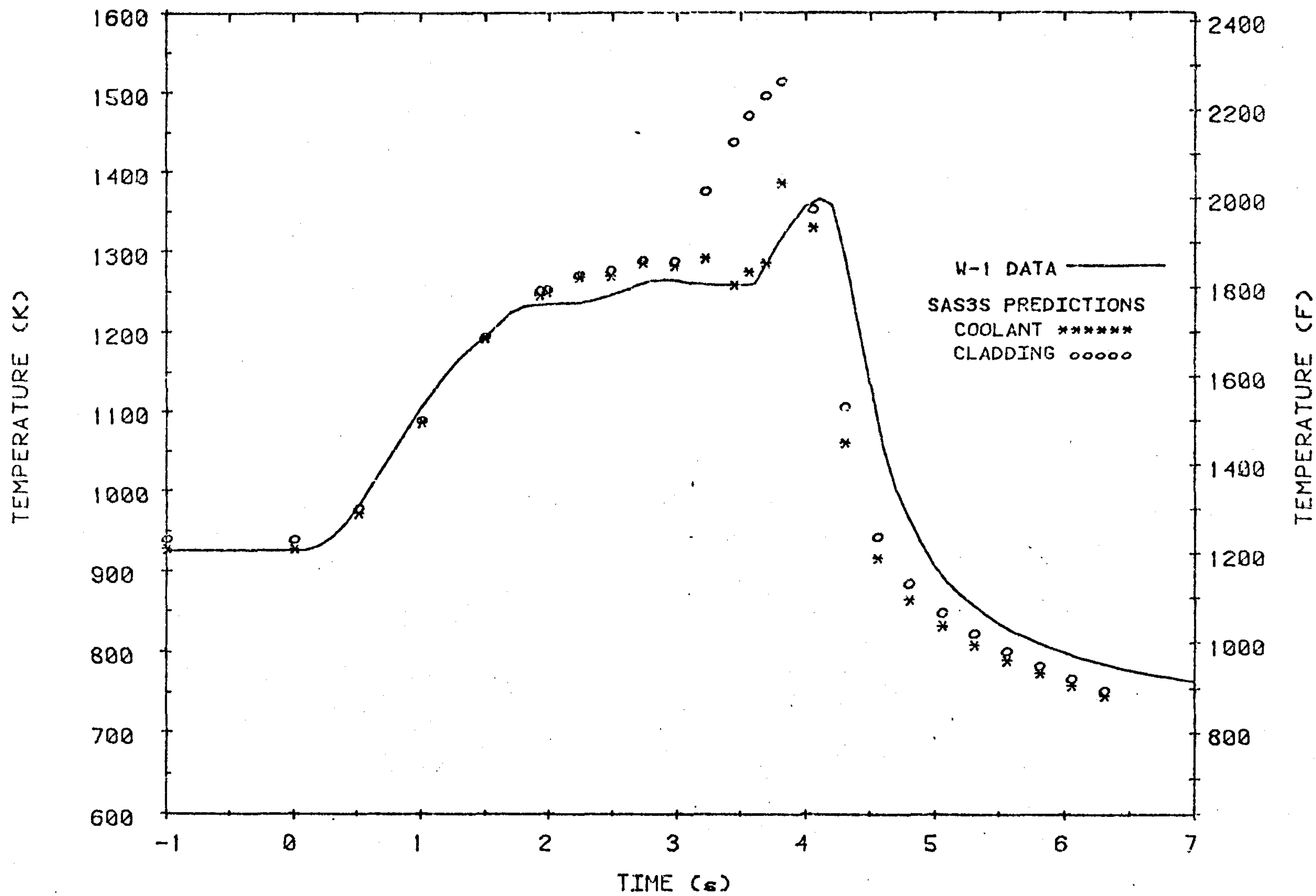

Figure ~~23~~ 7. W-1 SLSF EXPERIMENT BOILING WINDOW TEST 7b'
WIRE-WRAP THERMOCOUPLE DATA VS SAS3D PREDICTIONS

It is therefore concluced that the Boiling Window, defined as the range between incipient boiling and boiling which inevitably leads to flow reversal and cladding dryout/melting, narrows rapidly with increasing pin power, possibly vanishing completely at the medium-to-high power levels used in the W-1 experiment.

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2. W. W. Marr, "COBRA-3M: A Digital Computer Code for Analyzing Thermal-Hydraulic Behavior in Pin Bundles," ANL 8131, March 1975.

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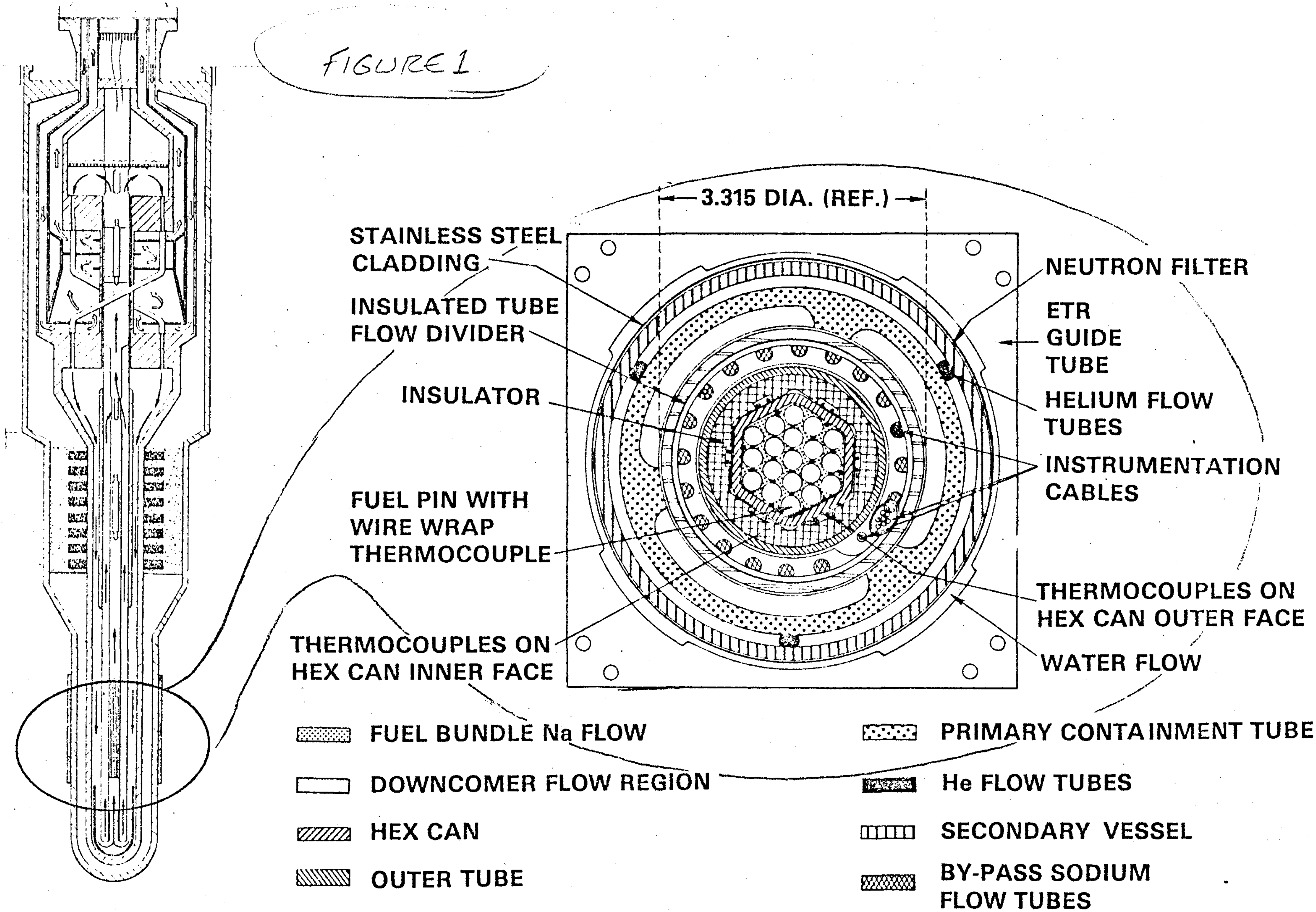
SLSF IN-PILE LOOP CROSS SECTION

CROSS-SECTION OF THE W-1 SLSF IN THE LOWER TEST SECTION REGION



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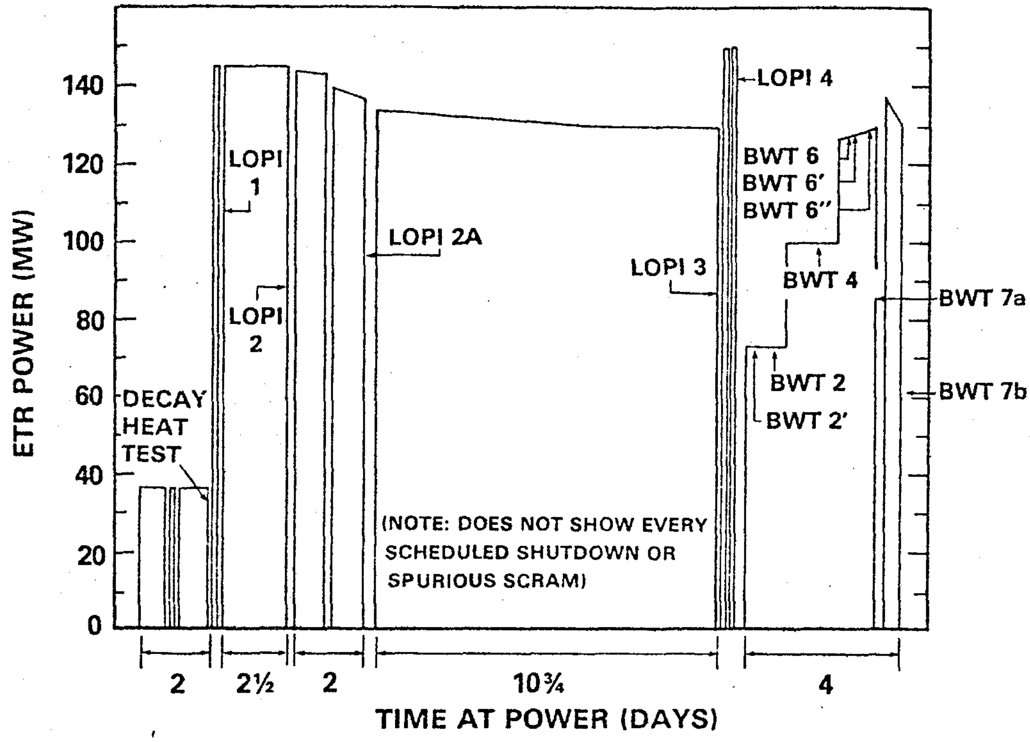

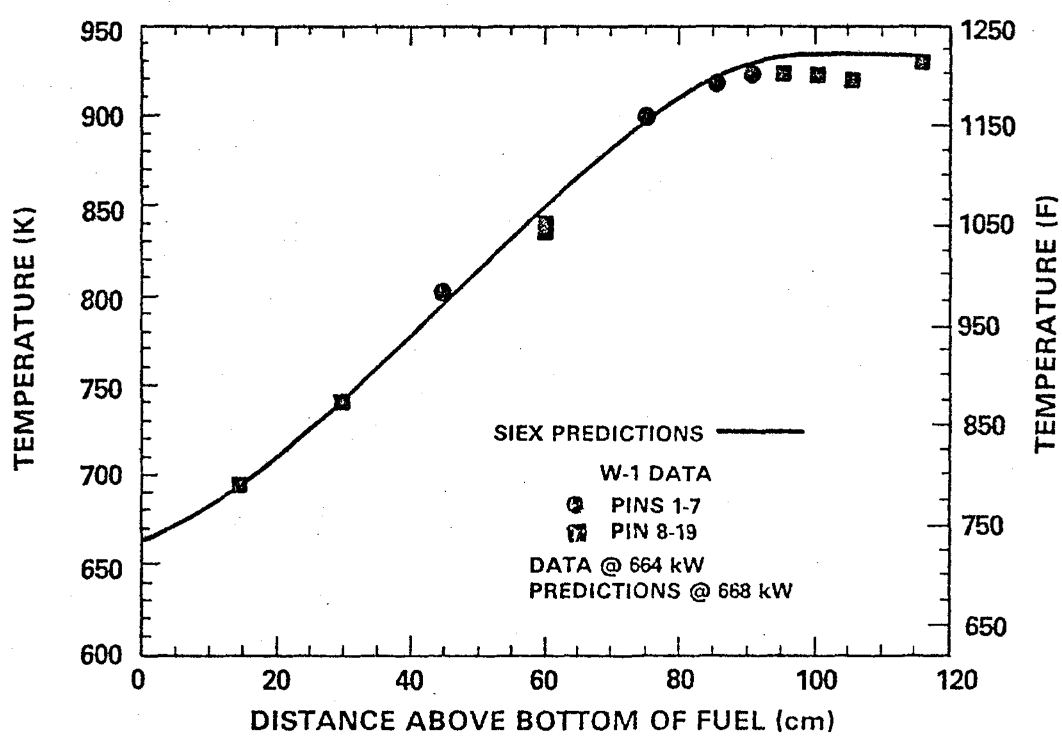

FIGURE 3. **W-1 SLSF EXPERIMENT AXIAL COOLANT TEMPERATURE PROFILE**

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The Boiling Window Test series was designed to investigate a hypothesized regime of power-to-coolant flowrate ratios which could cause "stable" coolant boiling conditions. The sodium boiling tests were conducted at three different average fuel pin power levels; 1) 200 W/cm (6.1 kW/ft), 2) 296 W/cm (9.0 kW/ft), and 3) 384.5 W/cm (11.7 kW/ft).

Radial and axial expansion of the boiling front was evident from the wire-wrap thermocouple coolant temperature data. The data clearly show that at higher heat flux levels, the boiling tends to be more intense. The fuel bundle inlet flow reduction is caused by an increased pressure drop generated by sodium void progression through the fuel bundle coolant channels.

During Boiling Window Test 7b', the sodium void progression and corresponding pressure drop increase caused inlet flow reversal, cladding dryout and moderate cladding melting. The transient was stopped by an ETR scram at 3.5 seconds and a return to full coolant pump power. Details of Boiling Test 7b' are given in Table 2.

TABLE 1

W-1 SLSF EXPERIMENT
TRANSIENT SUMMARY

Transient	Average Pin Power	Minimum Flow*	Peak Na Temperature	Boiling Duration
LOPI 1	377 W/cm	0.47 Kg/s	1211 K	N/A
LOPI 2	410 W/cm	0.47 Kg/s	1216 K	N/A
LOPI 2A	385 W/cm	0.47 Kg/s	1216 K	N/A
LOPI 3	377 W/cm	0.47 Kg/s	1200 K	N/A
LOPI 4	411 W/cm	0.47 Kg/s	1230 K	0.5 s
BWT 2'	207 W/cm	0.47 Kg/s	1224 K	0.8 s
BWT 4	313 W/cm	0.68 Kg/s	1250 K	1.0 s
BWT 6	386 W/cm	0.88 Kg/s	1247 K	0.2 s
BWT 6'	391 W/cm	0.88 Kg/s	1249 K	0.1 s
BWT 6"	396 W/cm	0.88 Kg/s	1280 K	0.9 s
BWT 7a'	393 W/cm	0.78 Kg/s	1288 K	1.6 s
BWT 7b'	391 W/cm	0.75 Kg/s	1374 K	2.0 s

* Programmed fuel bundle inlet low flow point

TABLE 2

BOILING TEST 7b'

Time	Event
0 s	Start of transient.
1.5 s	Boiling initiation - detected by inlet flow reduction
1.5 - 2.1 s	Boiling progression from center-to-edge-of-bundle at the top-of-the-fuel (TOF) region of the bundle.
2.8 s	Inlet flow reversal.
3.1 s	Boiling at axial midplane.
3.5 s	ETR scram. Cladding dyrout at the fuel bundle axial midplane.

TABLE 2 (CON'T)

Time	Event
3.7 s	Dryout extended up to the TOF region of the fuel bundle.
3.8 s	Inlet flow recovery.
3.9 s	Peak fuel centerline temperature: 2800 K.
4.0 s	Peak wire-wrap thermocouple temperature measurement: 1374 K.
5.6 s	Full recovery of fuel bundle flowrate to pretransient levels.

POST-TEST ANALYSIS

The post-test analysis of the W-1 experiment was conducted to evaluate both the W-1 data and the capability of safety and design computer codes to predict the W-1 fuel bundle transient response.

The computer codes used to evaluate the W-1 data were SIEX,[1] COBRA-3[2] and SAS-3D.[3] The SIEX code was used to predict steady-state fuel pin thermal performance, restructuring, fission gas release, and the fuel-cladding gap conductance. The COBRA-3 code was used to analyze the fuel pin and coolant thermal-hydraulic response during the LOPI transients and Boiling Window Tests. COBRA-3 calculations were reliable only to boiling inception. The SAS-3D code was used to analyze boiling progression, fuel pin cladding dryout, and subsequent cladding melting.

The results of the SIEX calculations of fuel restructuring and fuel pin thermal performance are shown on Figure 4, where fuel centerline temperatures are plotted versus time. The SIEX code underpredicted fuel centerline temperatures by as much as 220 K (400°F). Possible sources of error are in the fuel thermal conductivity model, the fuel-cladding gap conductance model, or the radial power profile in the fuel.

In order to reduce uncertainty in the COBRA-3 transient calculations, an attempt was made to reduce the fuel temperature error by conducting a parametric study with the SIEX code. Parameters that affect the calculated fuel temperature were varied to determine which produced the best match to the measured fuel temperature data.

It was determined that lowering the fuel thermal conductivity provided a close match between fuel centerline temperature predictions and data at both elevations of the in-fuel thermocouples (TE 22-3 @ 56 cm and TE 22-5 @86 cm above the bottom of the fuel). Using a higher fuel-cladding gap xenon concentration, effectively decreasing the gap conductance, gave fuel temperatures that agreed with TE 22-3 data, but the predictions at the 86 cm elevation were much higher than the TE 22-5 data.

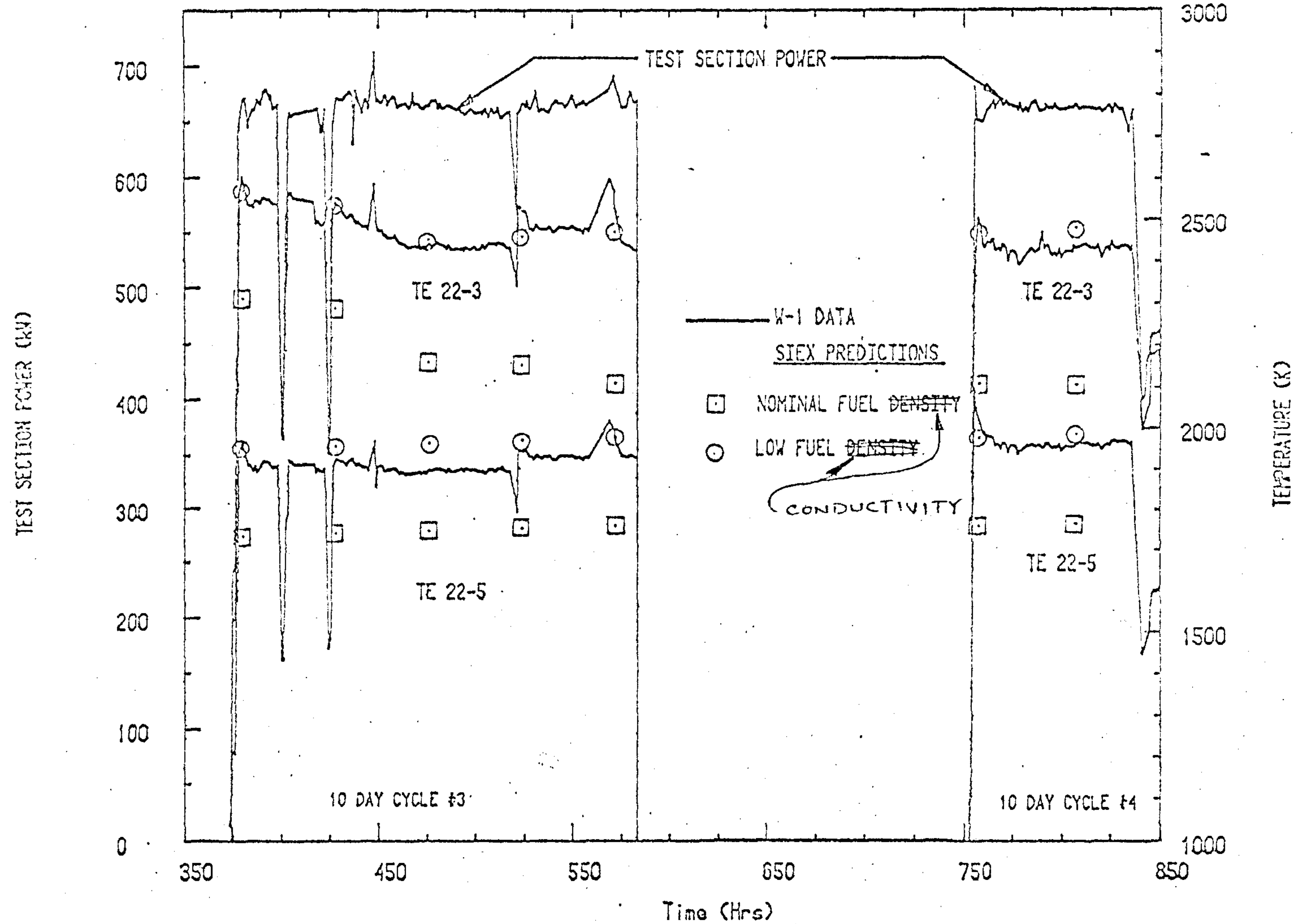

FIGURE 4.

W-1 SLSF EXPERIMENT IRRADIATION HISTORY
TEST SECTION POWER & FUEL CENTERLINE TEMPERATURES

The lower fuel conductivity case SIEX predictions of fuel retructuring and fission gas release matched the W-1 fuel pin postirratiation examination data very closely. These predictions were therefore used as input to the COBRA-3 code for the thermal-hydraulic analysis of the W-1 transients.

The COBRA-3 predictions of coolant temperatures are plotted with test data for LOPI 3 and Boiling Test 7b' in Figures 5 and 6. The agreement between predicted wire-wrap thermocouple response and the actual W-1 data varies from good to excellent.

The SAS-3D calculation results of Boiling Test 7b' are shown in Figure 7. The predictions of coolant and cladding temperature are very close to the W-1 wire-wrap thermocouple data, demonstrating the effectiveness of the SAS-3D code in analyzing this transient.

CONCLUSIONS

LOPI ACCIDENT EVALUATION

The W-1 experiment LOPI accident simulation demonstrated that the protected CRBR LOPI accident (i.e., pipe rupture and corresponding flow reduction followed by an automatic plant scram) is a benign event. The W-1 LOPI transients were conducted at 384.5 W/cm (11.7 kW/ft) average (nominal) pin power levels. This corresponds to a 15% overpower condition for the highest power FFTF assembly. Only LOPI 4, conducted at 23% overpower conditions for FFTF, generated coolant boiling, the boiling was very mild, lasting only 0.5 seconds.

There were only small changes in coolant temperature during the progression of LOPI transients. Maximum coolant temperatures dropped with progressing fuel pin irradiation and the corresponding fuel microstructure changes. It appears that a LOPI type transient becomes less severe with increasing core burnup, although as previously mentioned, the change is small.

BOILING PROGRESSION

The W-1 coolant boiling tests were designed to extend the THORS facility sodium boiling data base into the normal range of operating powers for a full-size breeder reactor. One major observation in the shift to higher powers is the acceleration of boiling progression from the center coolant channels to the outer coolant channels. The corresponding increase in coolant pressure drop rapidly leads to flow reversal.

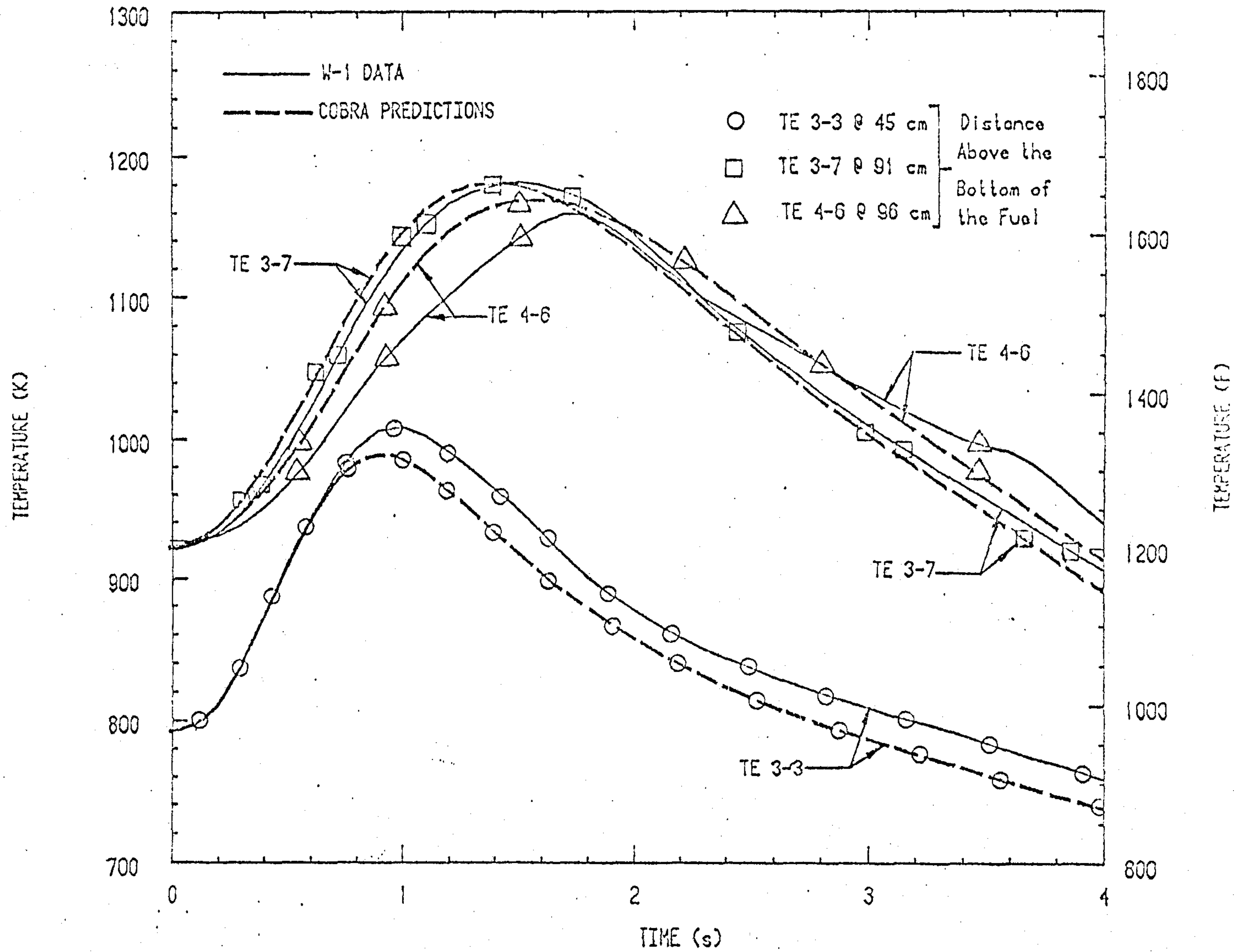

FIGURE 5.

W-1 SLSF EXPERIMENT LOSS-OF-PIPING INTEGRITY TRANSIENT #3
WIRE-WRAP THERMOCOUPLE DATA vs PREDICTIONS

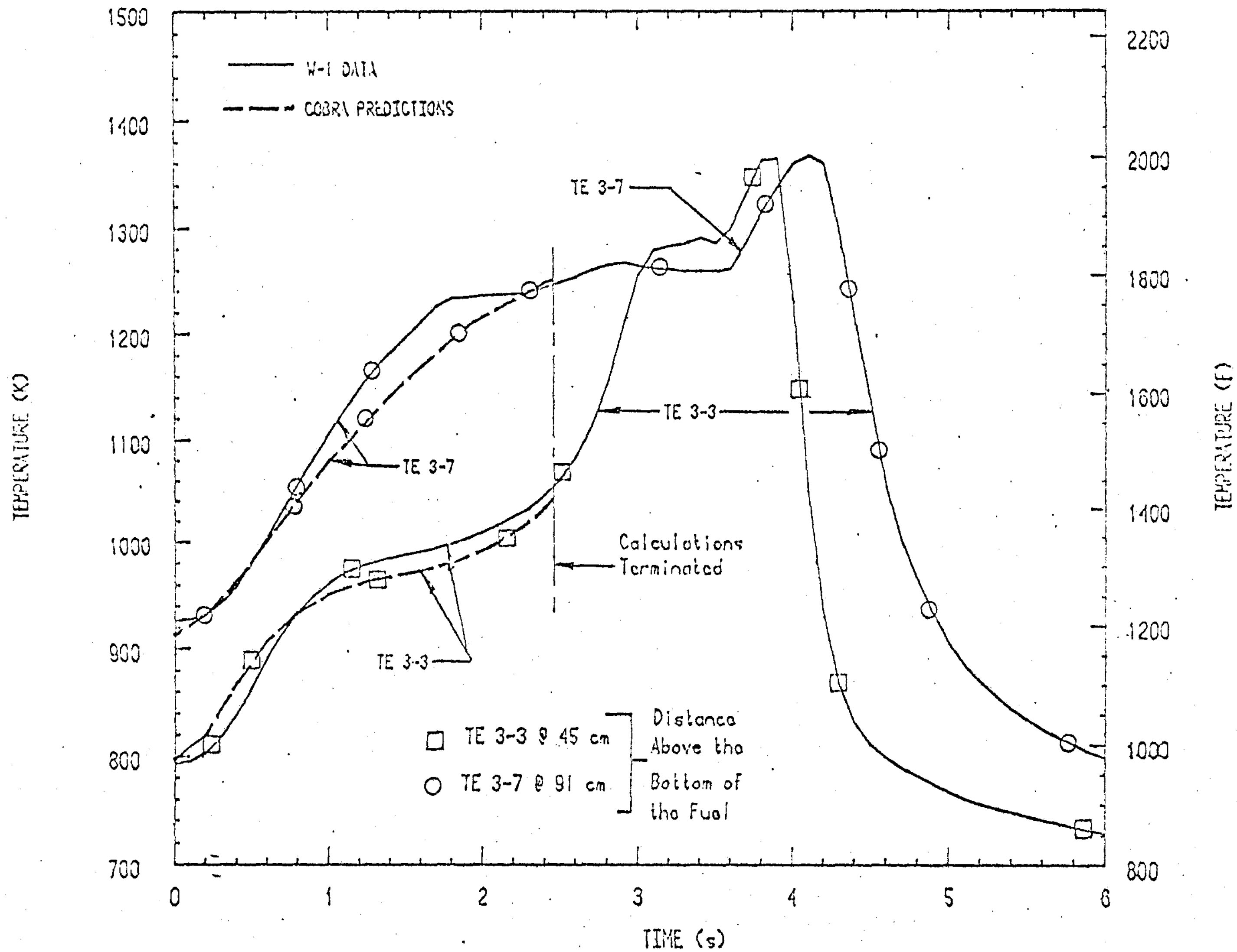

FIGURE 6.

W-1 SLSF EXPERIMENT HIGH POWER BOILING TEST 7b'
WIRE-WRAP THERMOCOUPLE DATA vs PREDICTIONS

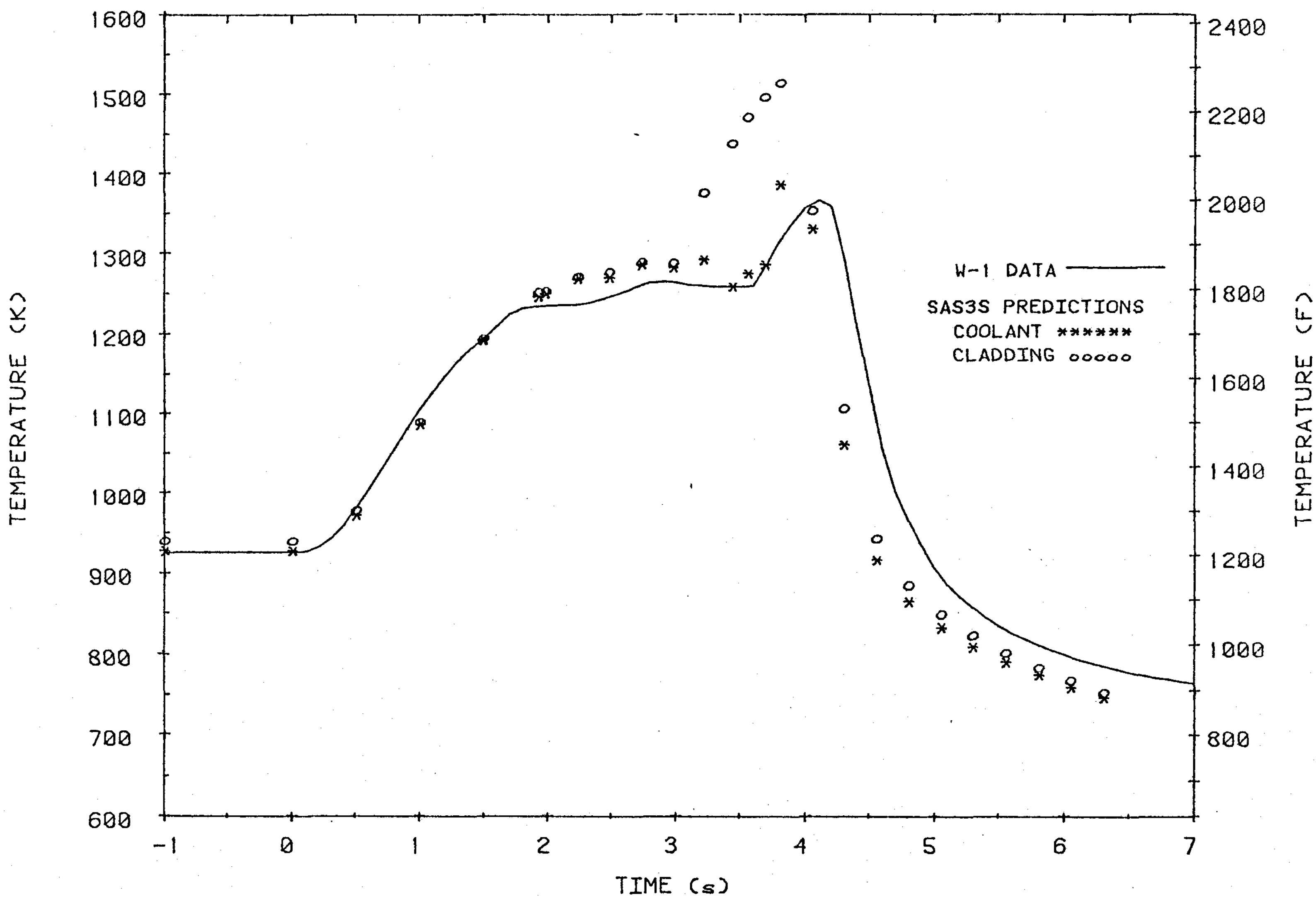

Figure ~~23~~ 7. W-1 SLSF EXPERIMENT BOILING WINDOW TEST 7b′
WIRE-WRAP THERMOCOUPLE DATA VS SAS3D PREDICTIONS

It is therefore concluced that the Boiling Window, defined as the range between incipient boiling and boiling which inevitably leads to flow reversal and cladding dryout/melting, narrows rapidly with increasing pin power, possibly vanishing completely at the medium-to-high power levels used in the W-1 experiment.

REFERENCES

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3. M. G. Stevenson, et al., "Current Status and Experimental Basis of the SAS LMFBR Accident Analysis Code System," Proc. Fast Reactor Safety Meeting, Beverly Hills, CA, CONF-740401, pg. 1303, (April 2-4, 1974).