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Summary

An experimental study of inherently safe characteristics of LMFBR systems is being conducted in the Experimental Breeder Reactor II (EBR-II). Experiments have been conducted which show 1) capabilities of heat removal without the aid of active safety systems and 2) capabilities to mitigate a reduction in-core cooling flow or heat rejection to the balance of plant (BOP) without aid of active control or protection systems.

Experiments involving loss of forced reactor cooling flow and loss of all heat rejection to the BOP are planned to be conducted. The purpose of this paper is to summarize the result of the test program by presenting representative test data.

The test series, referred to as the Shutdown Heat Removal Test (SHRT) Program, is being conducted on the Experimental Breeder Reactor II - a 62.5 MWth sodium cooled breeder reactor integrated with plant heat transport steam generating and electrical generating system. A nominal 20 MW of electric power is supplied for local and grid use.

In addition to the direct demonstration of an LMFBR plant's "inherently safe" capabilities, a major objective of the SHRT program is to provide data for computer code validation so that the results of the testing can be extrapolated and applied to other LMFBR designs (as well as to other EBR-II operating conditions). To support this objective an extensive data set is recorded for each test by the plant digital computer based Data Acquisition System. Two special instrumented in-core subassemblies (INSATs) are being used to

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measure in-core temperatures and flows. One INSAT is thermally and hydraulically similar to a fuel driver assembly the other INSAT is similar to a blanket assembly. Data from plant instrumentation is recorded to characterize all the major systems and components of the plant including the primary and secondary sodium system and the steam generating system.

The summary which follows gives a brief description of each type of test that has been conducted to date along with some representative data from each test type.

Type A Tests - Protected Loss of Flow from Power

The purpose of the Type A tests was to demonstrate and provide measurements of the transition to natural circulation cooling following a sudden loss of primary forced circulation (loss of electrical power to the main coolant pumps and auxiliary pump) while at power. Nine tests of this type were conducted. The parameters that were varied from test to test were the initial power and initial flow conditions, and the secondary loop pump mode of operation. The most severe test (SHRT 17) involved a simultaneous loss of electrical power to the main coolant pumps, the auxiliary pump, the secondary loop pump and a reactor scram from 100% power and flow. Figure 1 shows temperatures (predicted and measured) indicative of the response of EBR-II to the SHRT 17 loss of forced circulation. Both the pretest predictions and the measured temperatures correspond to coolant temperature at the top of the active core. The predictions were made with the NATDEMO and HOTCHAN codes prior to conducting the test for which the maximum, nominal and minimum curves denote the HOTCHAN predictions for XX09 with positive uncertainties, without uncertainties and with negative uncertainties, which are labeled "MAX", "NOM", and "MIN", respectively, in the figure. The temperature measurements shown in Fig. 1 are raw data from TTC 31 (the hottest thermocouple) and were taken during testing for monitoring purposes. It should be noted that TTC 31 is one of 13 "top-of-core" (coolant) thermocouples which are all located 0.83 in. below the actual top of fuel. However, because of other compensating effects (i.e., offset of ~ +8°F and center of bundle location), the TTC 31 readings can be directly compared with the true top-of-core predictions without introducing significant error.

The test was initiated by a reactor scram and pump trips at time zero. As shown in the figure the temperature initially decreases rapidly in response to the reactor scram and rapid power decrease. As the pumps coast down to a stop (at about 50 sec), forced flow decreases and the decay power heats up the sodium in the channel. The temperature rise causes an increase of the thermal head which induces sufficient natural circulation flow to terminate the rise in temperature (at about 70 sec). At times beyond the temperature peak, a quasi-equilibrium is approached in which the gradually decreasing rate of heat generation in the core is almost matched by a gradually decreasing rate of heat removal by the coolant. This is indicated by the decreasing flow and temperature in the XX09 subassembly.

As shown in the figure there is excellent agreement between the predicted and measured data. Post-test analysis is necessary to quantify effects of actual test conditions (such as power history) and actual instrumentation offsets on the results. However, subject to further confirmation in the post-test analysis, the data indicates 1) that natural circulation is a safe and reliable method of decay heat removal and 2) that the phenomena governing the transition to natural circulation have been identified and are adequately understood and modeled.

<u>Type B Tests - Delayed Loss of Flow from Shutdown; Isothermal Conditions</u>

The goal of the Type B tests was to provide measurements of the development of thermal heads and natural circulation flows sufficient to cool the core even from an isothermal initial condition. Nine Type B tests were conducted from shutdown conditions from which the decay heat load ranged from very small (SHRT 1 followed a lengthy shutdown) to about 1.6% power for SHRT 2. All of the tests were performed from near isothermal (100% primary flow) conditions, except SHRT 2 which was initiated by tripping the auxiliary pump while the primary pumps were off. The representative data for SHRT 2, which is included in Fig. 2, is consistent with earlier EBR-II test results and indicates 1) adequate natural circulation cooling following a loss of forced circulation during shutdown, and 2) good agreement between measured and pretest predictions.

<u>Type C Tests - Reactivity Feedback Tests</u>

The objective of the Type C tests was to demonstrate and measure the EBR-II reactor power and core temperature response to 1) perturbations in primary coolant flow at a constant inlet temperature and 2) perturbations in primary coolant temperature at the reactor inlet for a constant flow condition. These tests are stepping-stones to the unprotected loss-of-flow and loss-of-heat sink tests to be run in future parts of the SHRT program because they will validate the NATDEMO reactivity feedback modeling.

Eight Type C tests were completed. The first four tests were initiated by perturbing primary flow or inlet temperature from an initial 25% power and flow steady state condition. Two subsequent tests were initiated from 50% power and flow and the final two tests were initiated from 70% power and flow.

The primary flow perturbations were initiated by manually ramping primary pump speed while being careful to keep the flow balanced between the pumps. The sodium temperature at the reactor inlet was maintained as constant as possible by controlling heat rejection via the IHX to the secondary loop with varying secondary loop flow.

Representative raw data for SHRT 25 are shown in Figs. 3 through 6, in which primary flow was perturbed from an initial 70% flow condition. Figure 3 shows the measured core flow through the instrumented subassembly XX09. Figure 4 shows a representative in-core temperature response as measured on TTC 31. The excess reactivity, calculated online by the EBR-II plant DAS is shown in Fig. 5, and the measured power response is shown in Fig. 6. The data indicate that the inherent response of the plant tends to keep power and flow matched, thus keeping core temperatures within the operating range. Detailed analysis of this data will be conducted to quantify reactivity feedback coefficients and confirm the dynamic modeling in the NATDEMO model.

Representative data for SHRT 26 are shown in Figs. 7 through 10. The reactor inlet temperature was perturbed by manually changing the secondary loop flow as shown in Fig. 7 while primary flow was held fixed. The perturbations in secondary loop flow altered the IHX heat rejection resulting in a change in IHX primary outlet temperature and consequent change in reactor inlet temperature. The inlet temperature response is shown in Fig. 8. Reactor inlet temperature variation causes the reactor power to change (Fig. 9) due to reactivity feedback, and consequently affects temperature in the reactor. Figure 10 gives the measured XX09 temperature reading from TTC 31. Note that the increase in inlet temperature causes a significant decrease in power and core outlet temperature. The Type C tests indicate a $\sim$ 2% power reduction occurs for each 1°F increase in reactor inlet temperature. This negative temperature coefficient is larger than has been typically taken credit for in conservative EBR-II pretest predictions.

Fig. 1

PREDICTED AND MEASURED TEMPERATURES OF SHRT 17

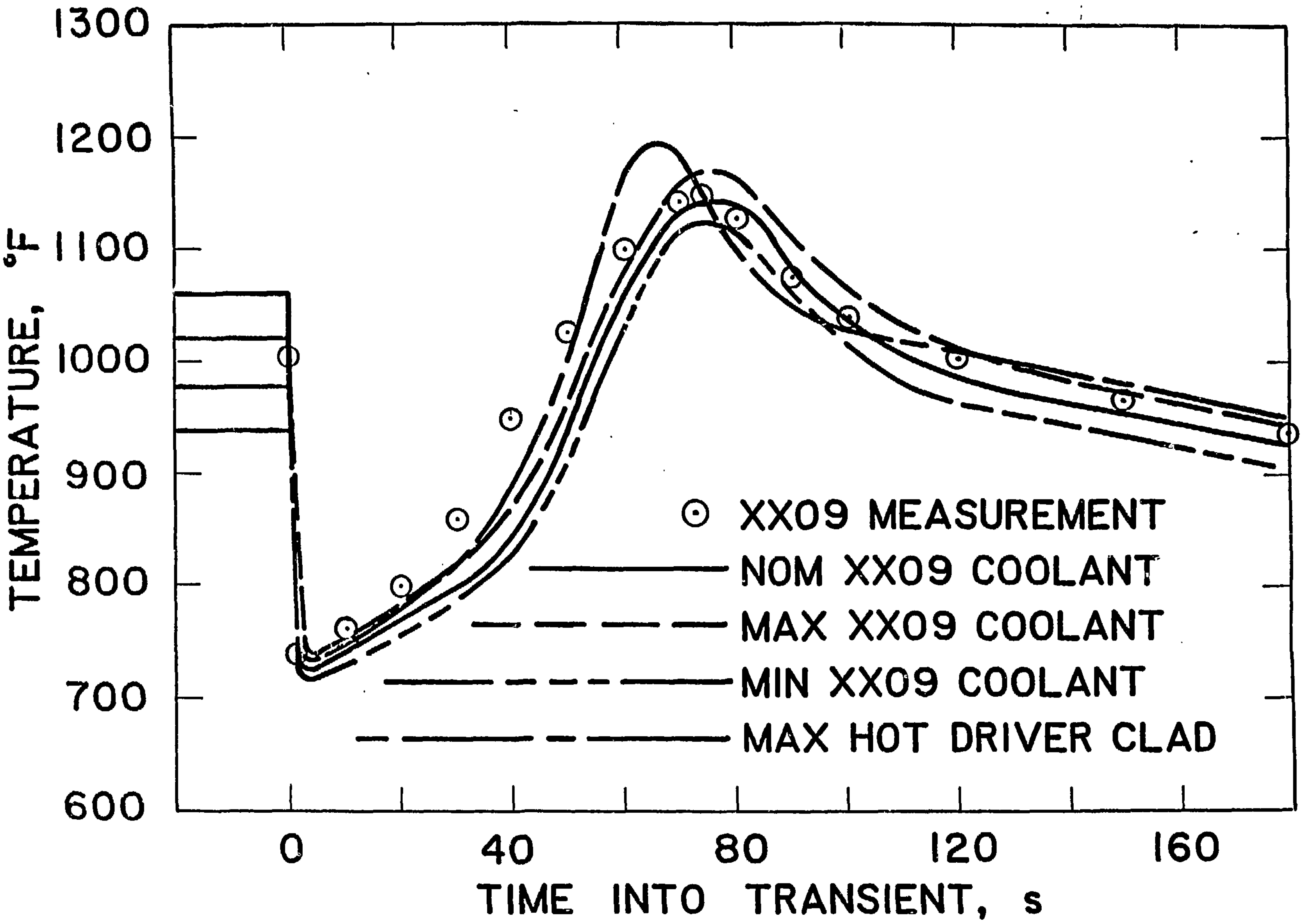

Fig. 2

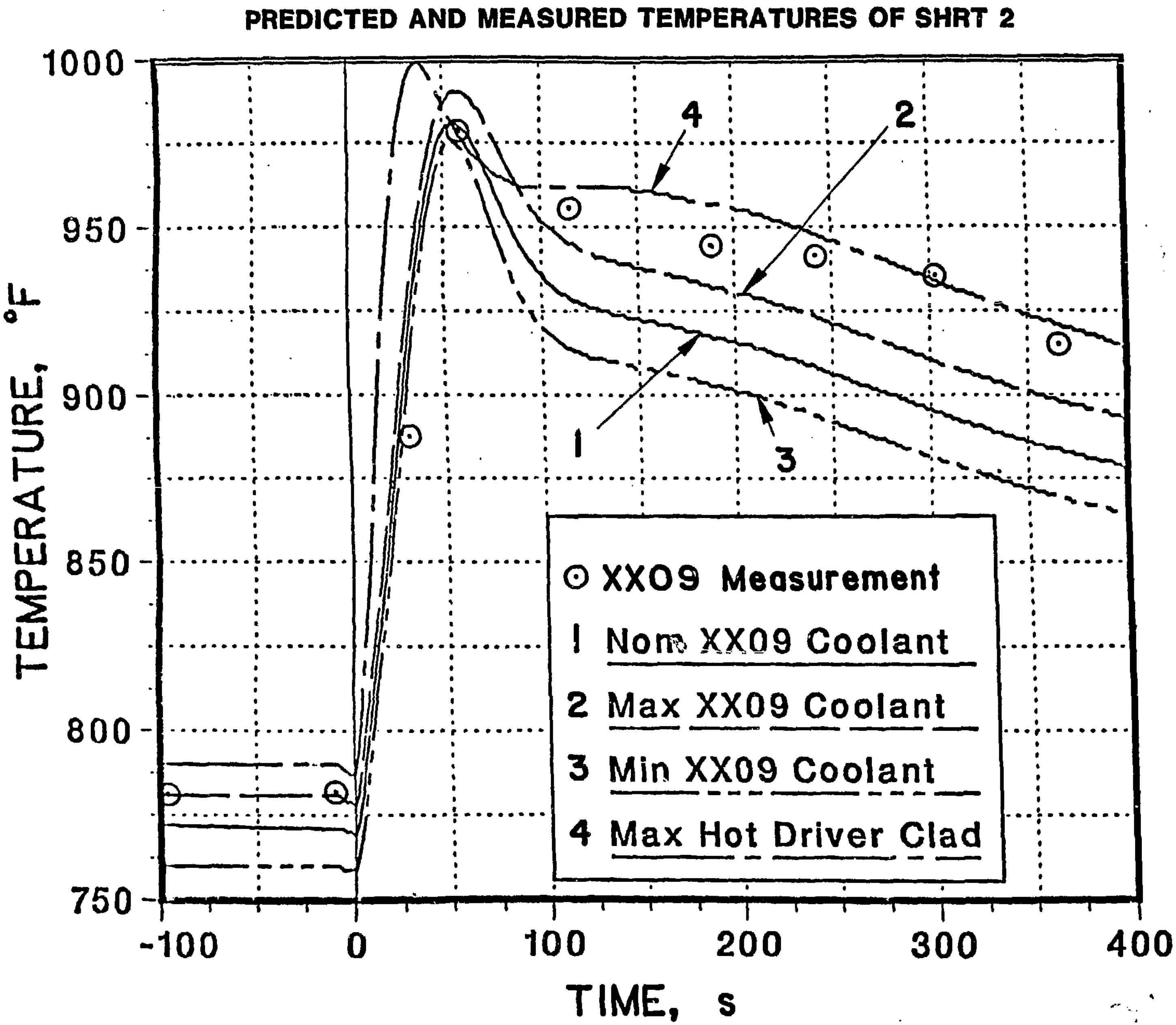
PREDICTED AND MEASURED TEMPERATURES OF SHRT 2
1000
950
900
850
800
750
TEMPERATURE, °F
-100
0
100
200
300
400
TIME, s
4
2
1
3
XX09 Measurement
1 Nom XX09 Coolant
2 Max XX09 Coolant
3 Min XX09 Coolant
4 Max Hot Driver Clad

Fig. 3

MEASURED FLOW IN INSTRUMENTED SUBASSEMBLY XX09 OF SHRT 25

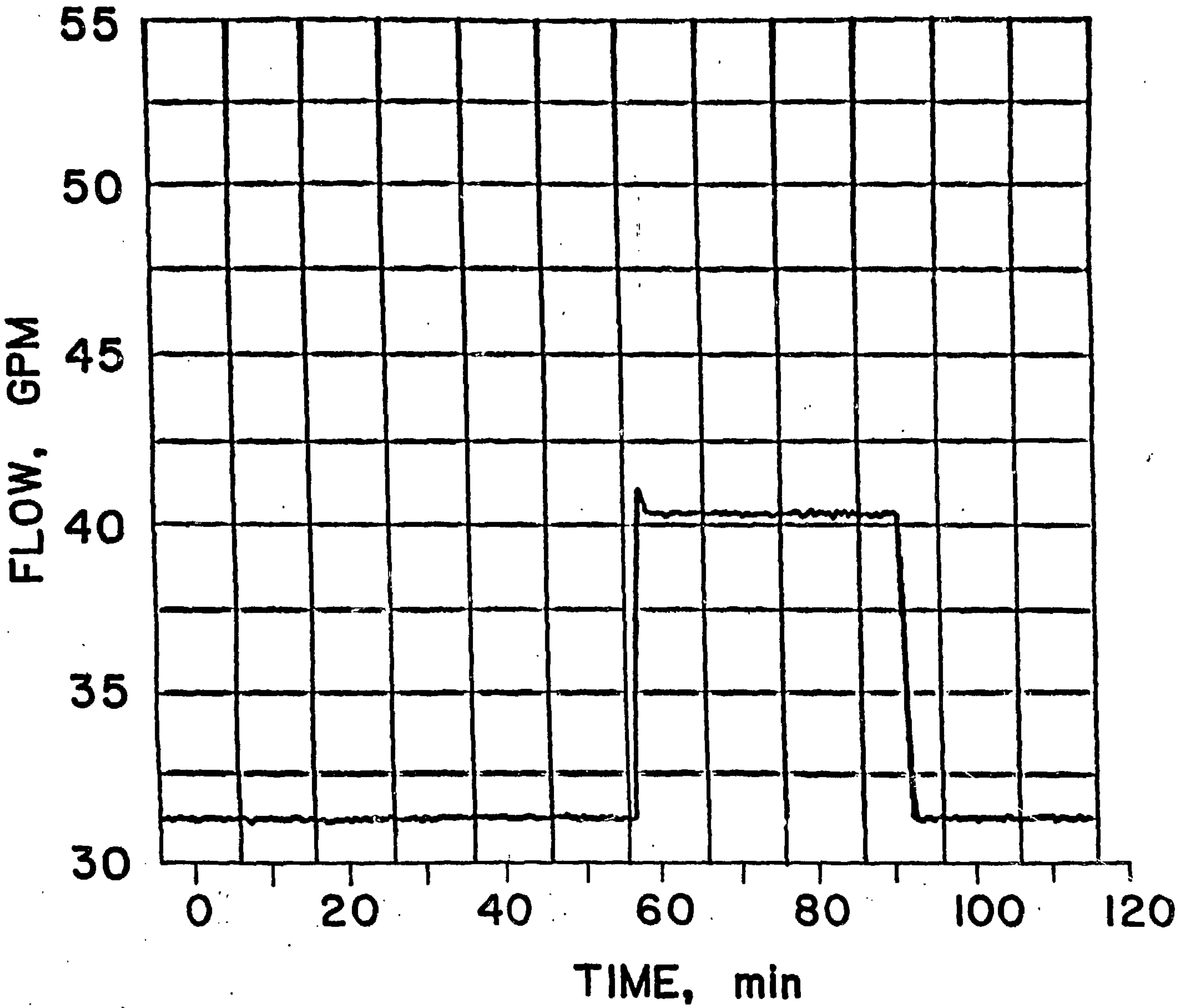

Fig. 4

TOP OF CORE TEMPERATURE MEASURED IN XX09 OF SHRT 25

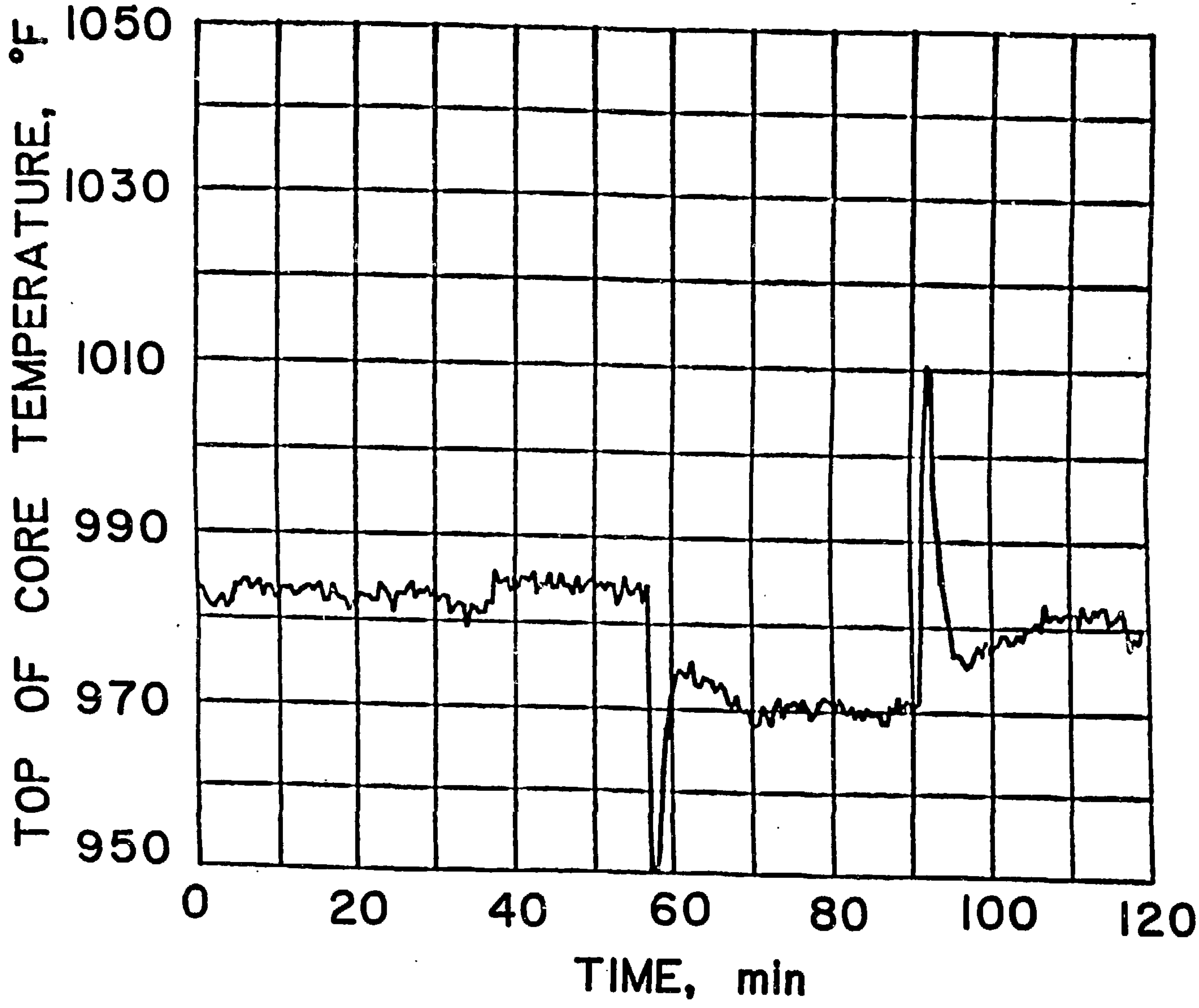

Fig. 5

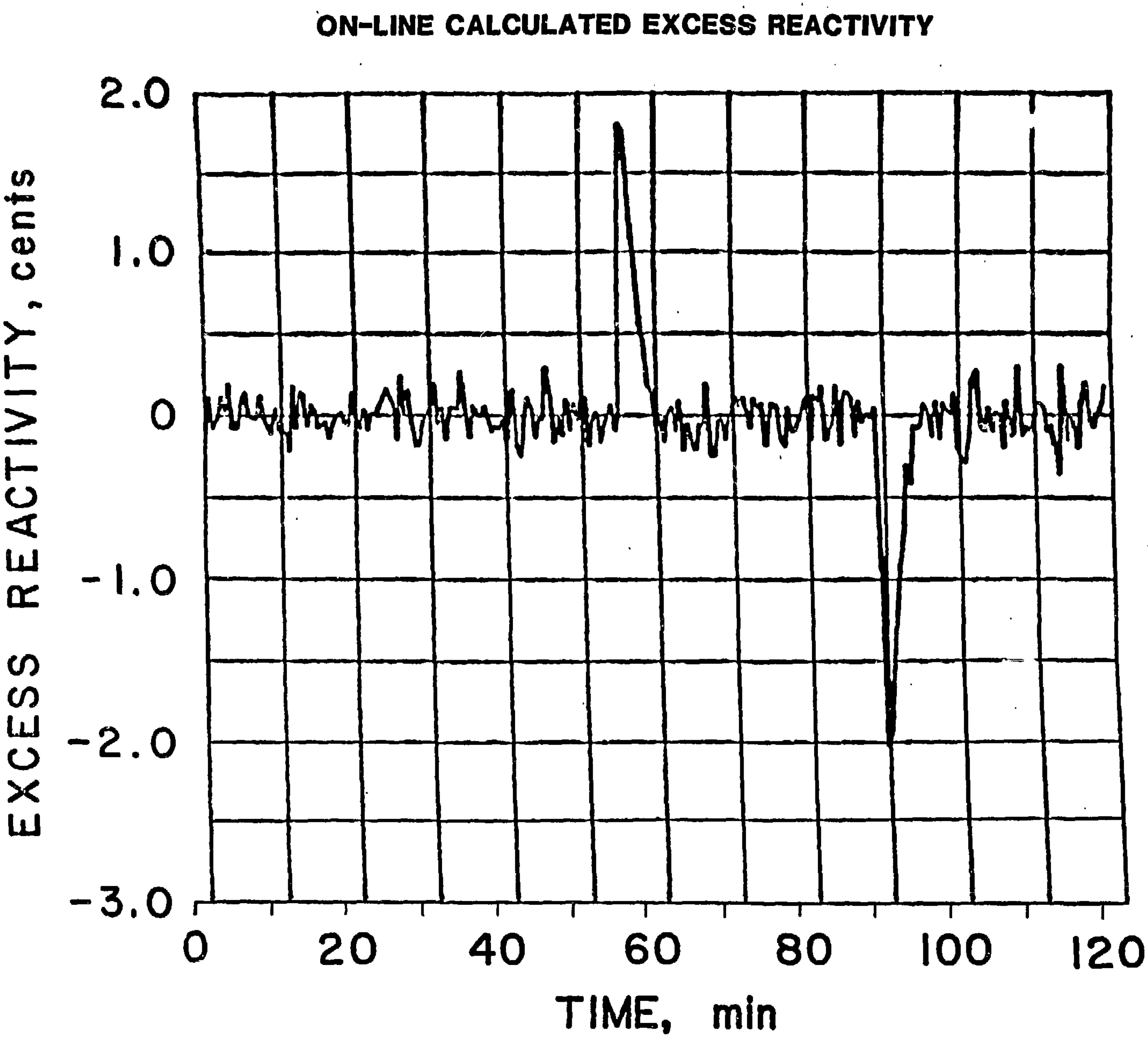
ON-LINE CALCULATED EXCESS REACTIVITY
EXCESS REACTIVITY, cents
2.0
1.0
0
-1.0
-2.0
-3.0
0
20
40
60
80
100
120
TIME, min

Fig. 6

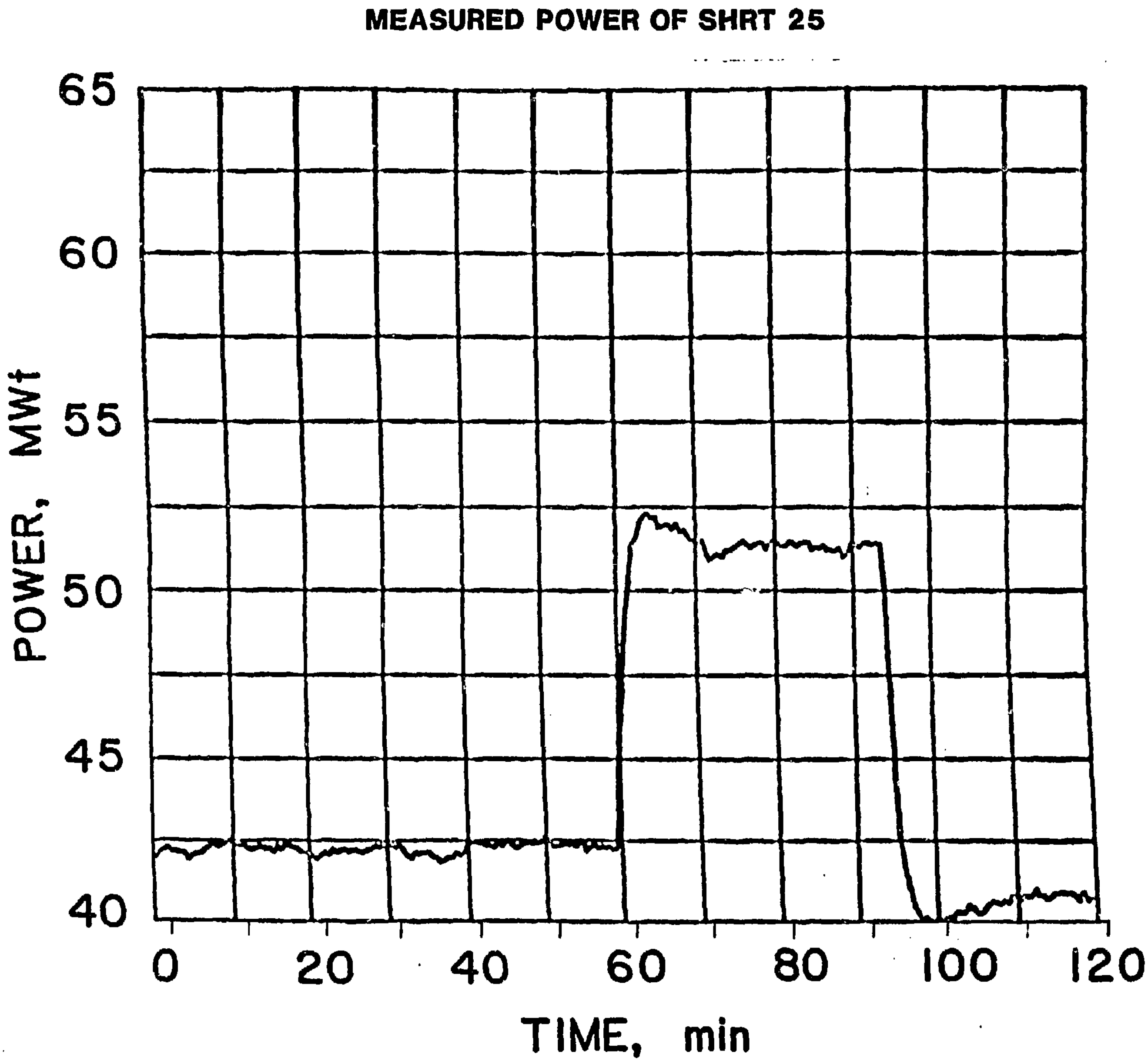
MEASURED POWER OF SHRT 25
65
60
55
50
45
40
POWER, MWt
0
20
40
60
80
100
120
TIME, min

Fig. 7

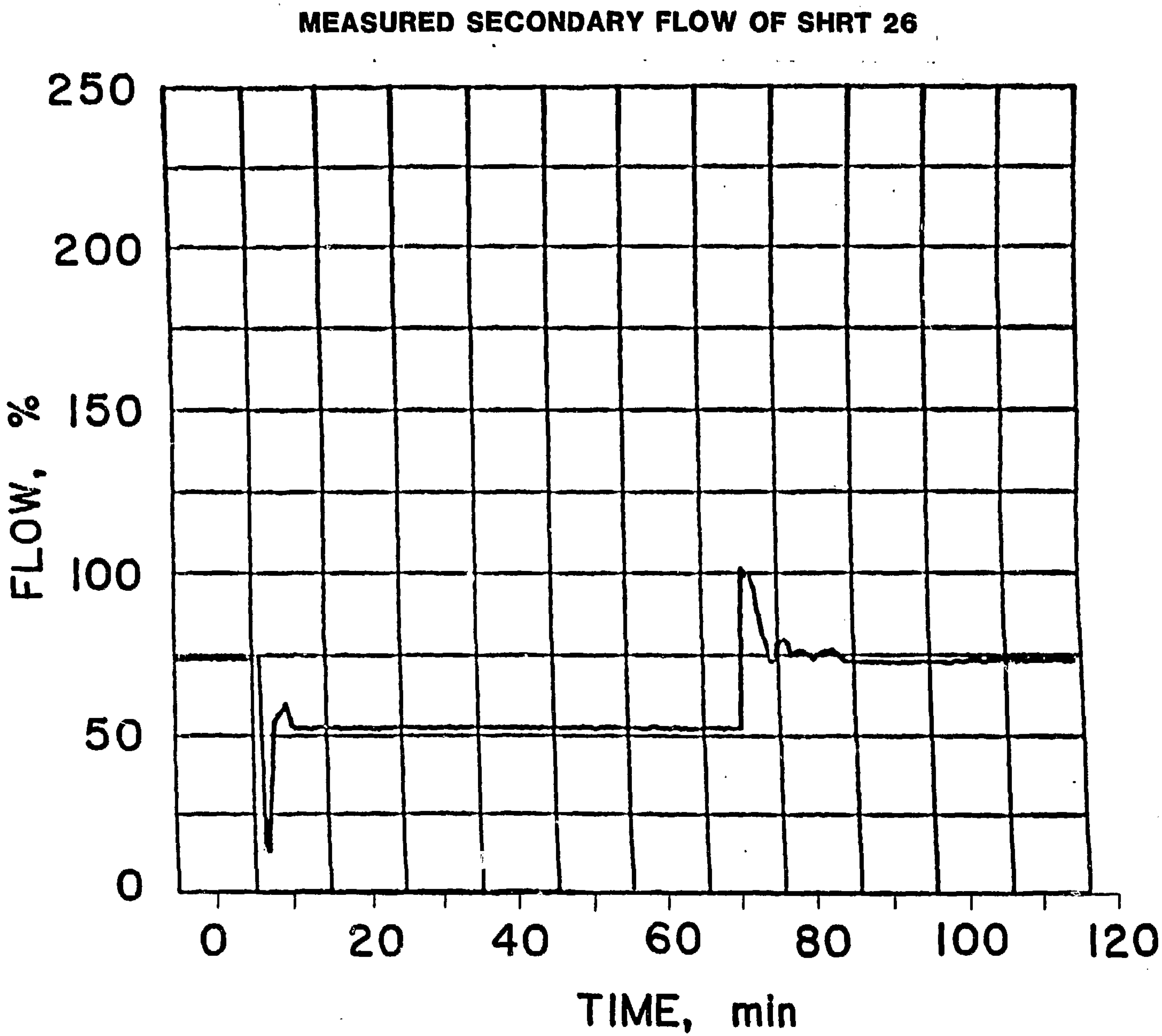
MEASURED SECONDARY FLOW OF SHRT 26
FLOW, %
250
200
150
100
50
0
0
20
40
60
80
100
120
TIME, min

Fig. 8

MEASURED REACTOR INLET TEMPERATURE OF SHRT 26

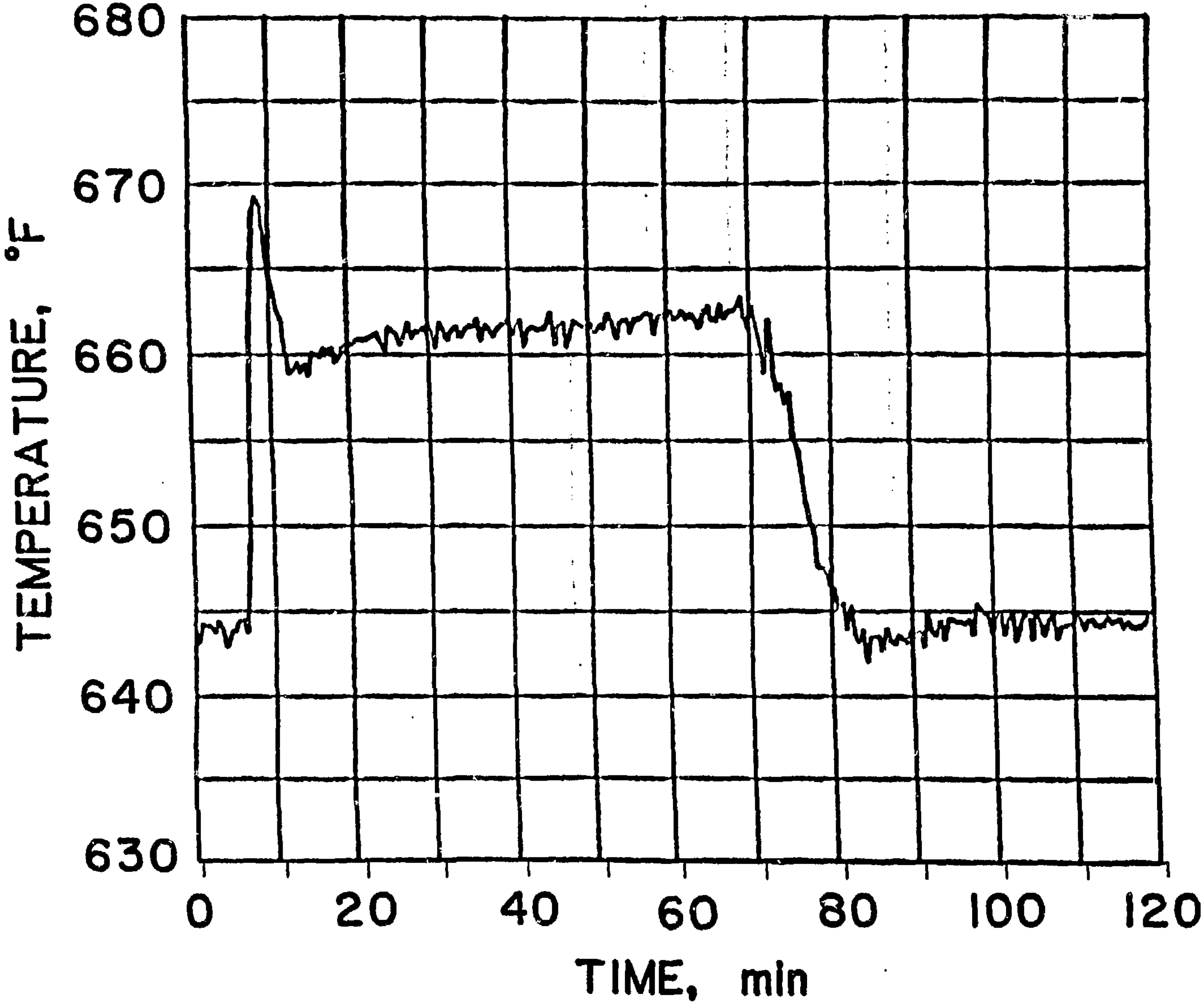

Fig. 9

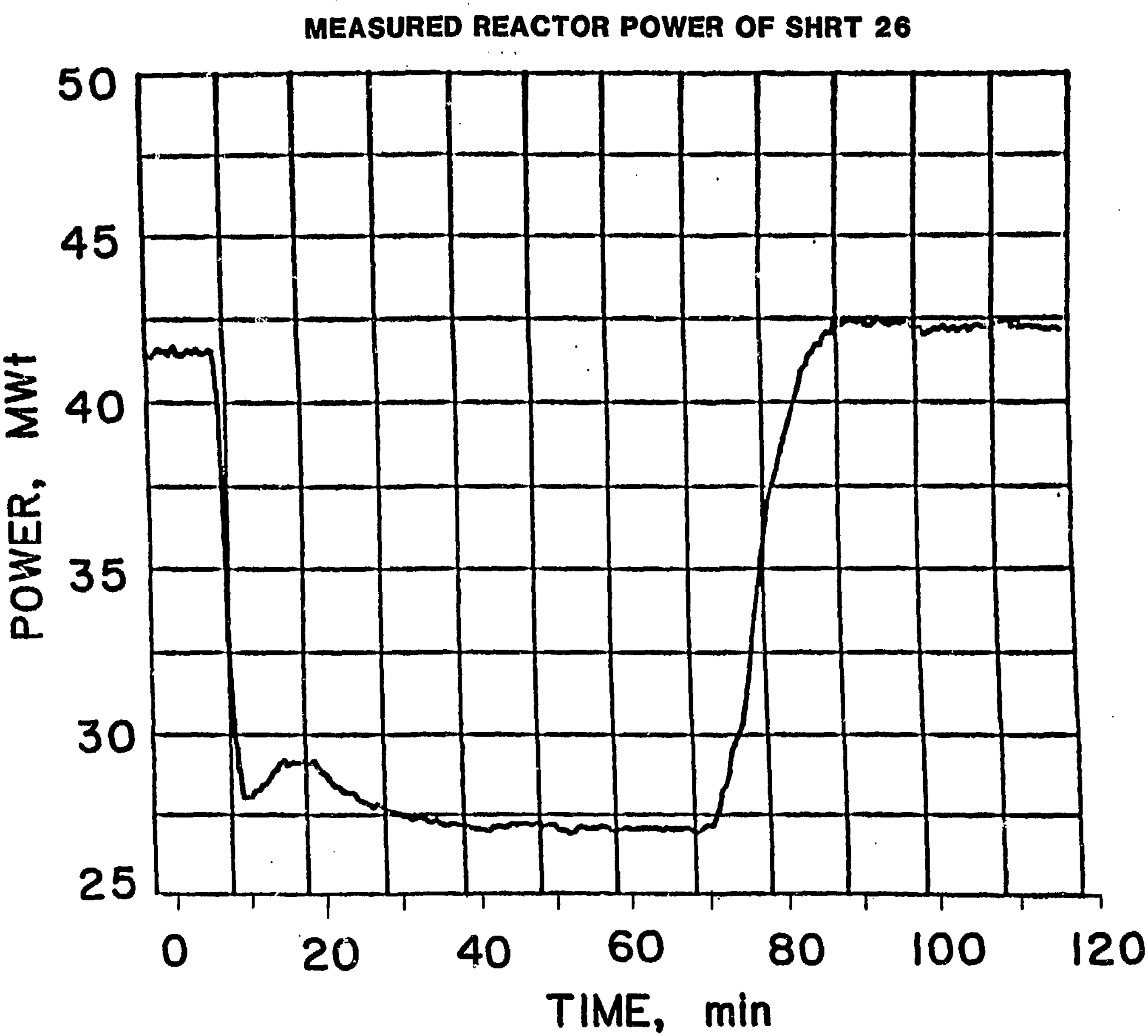
MEASURED REACTOR POWER OF SHRT 26
50
45
40
35
30
25
POWER, MWt
0
20
40
60
80
100
120
TIME, min

Fig. 10

MEASURED COOLANT TEMPERATURE