PRETEST PREDICTIONS FOR FFTF REACTOR AND PRIMARY

LOOP TRANSIENT NATURAL CIRCULATION TEST FROM 5% POWER

(TS-51-5A008, PART 1)

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<u>CONTENTS</u>

LIST OF TABLES

LIST OF FIGURES

1. BACKGROUND

Detailed planning for the first FFTF test of the primary loop natural circulation performance has been completed. The first test is a transient test involving reactor scram from 5% power operation, to be conducted generally as described in Reference 1, except that secondary pony motors will be kept operating for this test. The predictions provided in this letter will be used in conjunction with post-test evaluation and similar information from further tests at higher reactor power levels to demonstrate the plant decay heat removal capability and provide required model verification as discussed in References 2 and 3.

The decision to operate the secondary pony motors in this test was made for two reasons. One was an operational desire to permit the focus of the attention of attending personnel on the primary loop response. The second was a desire to reduce the scope of the possible primary loop uncertainties, recognizing that the low power level of this particular test places many important plant parameters below levels for which the decay heat removal evalution model was developed. Secondary pony motor operation was judged to provide some incremental certainty regarding IHX temperature response so that the test would principally be addressing reactor performance. Additional efforts subsequent to Reference 1 have focused on defining the test uncertainties and establishing acceptance limits for this test. Appendix B to this letter, provided for information, is typical of the evaluation performed, though the acceptance criteria discussed in that paper have since been revised based on rescheduling of the plant test sequence. Current schedules call for the 5% test to be performed on about November 16, with a second test from 35% power to follow on about December 1. Both tests will precede the full power demonstration. Because the 35% power test is expected to be a more accurate test, the decision to proceed with the power demonstration will be based on the results of that test, rather than on the 5% test as had previously been planned.

The remaining primary loop natural circulation tests described in Reference 1, including the steady-state test series and the transient tests from 75 and 100% reactor power, are presently scheduled to be performed in late February and March of 1981.

2. TEST DESCRIPTION, SCRAM FROM LOW POWER (5%)

The first test planned to address natural circulation in the FFTF primary loop and reactor vessel is a plant scram from 5% power (20 Mw), 75% flow. The primary pump pony motors will be de-energized just prior to reactor scram so that the reactor will undergo a transition to natural circulation following automatic trip of the pump main motors upon scram. The test will be conducted prior to power operation above 5% power and after steady operation for at least 1 hour at 5% power. For this test the secondary loops will be operated with pump pony motors and with cold leg temperature controlled (via air flow modulation) during the transient. This test will permit attention to be focused on core and reactor responses to natural circulation by minimizing potential for perturbations from IHX's or natural circulation phenomena in the secondary system. The Test Specification TS-51-5A008 gives a detailed description of this test (Appendix A).

3. SAFETY MODEL PREDICTIONS

The FFTF safety model is the version of the IANUS computer program used to generate the analyses of natural circulation behavior documented in Reference 4 and FSAR Chapter 15.1.3. The IANUS model is discussed in more detail in Reference 5. The safety model predictions are to be compared with the test result in order to demonstrate that the combined model parameter uncertainties are within the design allowances.[3] The safety model was developed, however, to address the transition to natural circulation from full power equilibrium operating conditions. Some of the model assumptions appropriate for the design evaluation are clearly inappropriate for this beginning-of-life, low power transient test. Accordingly, the assumptions have been adjusted in four instances, as discussed below. The adjustments have been made in a manner consistent with the safety model development to maintain a relationship between the safety model prediction and the 5% test directly comparable to that between the design safety model and the design event. With these adjustments, we expect the comparison of the 5% test result with its corresponding safety model prediction to provide meaningful feedback on the model conservatism.

The changes made to the safety model for the 5% test predictions involve four important differences between the test conditions and the equilibrium operation, on which the original model was based. These differences include:

1. replace design high power assembly nominal parameters with those of the instrumented (Fuel Open Test Assembly or FOTA) high power assembly in the test core load;
2. replace design power uncertainty with the larger value (20%) applicable to the test conditions, recognizing that the reactor will not have been operated at a sufficiently high power level to obtain a good thermal power calibration;
3. replace end-of-life decay power curve with a beginning of life curve applicable to the test condition (but then apply comparable uncertainty allowances);
4. replace flow dependent inter-assembly hot channel factor based on design temperature rise with a similarly calculated curve at a temperature rise appropriate to the test condition.

Each of these four changes is discussed in more detail below.

This adjusted safety model was used to predict the peak transient temperature as measured by a fast responding thermocouple in the Row 2 FOTA. This thermocouple, designated TX1016 in the instrument list, is also referred to as HF011 T8/8. This thermocouple is located one inch above the active fuel zone near the assembly center as shown in Figure 3.1.

3.1 Adjustments for Experimental Conditions

In order to make the safety model prediction for the 5% test have the same relationship to the test, as the FSAR predictions have to the design basis events, four significant adjustments have been made to the original model as listed above. The normal reactor inlet temperature for operation at 5% power is 596°F. This value, which will be the test inlet temperature, was also used in the predictions. The inlet temperature has no appreciable affect on core temperature rise, thermal head, or natural circulation performance, however, so this is not a significant adjustment.

The design end-of-life high-power-assembly power was 1.439 times the average assembly in that core. The Row 2 FOTA, (a high-power-assembly in the test core loading) experiences a power which is 1.2774 times that of the average assembly in the test core loading. The overall hot channel factor table, Table 3.1, illustrates how this factor is applied (line I.1).

At the time the 5% power scram to natural circulation test is performed, the reactor will have achieved only 5% power. This power level is too low for accurate thermal power calibration to be performed. Therefore the power measurement uncertainty is estimated to be 20% rather than the 8.7% which is characteristic of higher power operation. Table 3.1 illustrates the application of this factor (line 2 under STATISTICAL).

The predictions presented in the FSAR utilized an early conservative method of determining decay heat. This early method yielded values ~10% higher over the first five minutes after scram than the current more recently developed model. In addition, the FSAR safety model allowed a 25% error band on the decay heat, i.e., the calculated value was increased by 1.25. Thus the FSAR safety model used a value (1.10 x 1.25 =) 1.38 times the present decay heat model. The factor of 1.38 has been retained for this prediction. The decay heat was calculated for the expected power history prior to the test, using the minimum one hour at 5% power required by the test specification. The decay power at ~200 seconds controls the predicted peak temperature rise for this test. Therefore the calculated decay power at 200 seconds was multiplied by 1.38, and the time-at-power for the safety model was set to cause the resulting decay heat curve to pass through this point. The decay heat has thus been adjusted to the experimental conditions retaining the conservatism of the FSAR safety model. The resultant curve of decay heat versus time after scram is given in Figure 3-2. Variations in experimental conditions may require recomputation of the predictions to afford a valid comparison.

Flow distribution between assemblies at very low flows improves (flow increases to hot assemblies) with increasing power to flow ratio. The safety model for the design basis event uses a flow dependent flow distribution factor based on steady-state FLODISC[6] calculations performed using a power to flow ratio of unity. The design basis effective power to flow ratio at the time of the peak temperature is in excess of unity so the model is conservative.

The 5% power scram to natural circulation test will be performed with an initial power of 5% of the design basis event, and less than 4% (less than 1 MW versus ~25 MW) of the decay power of the design basis event. As a result, the effective power to flow ratio at the time of the peak temperature will also be low for the test case relative to the design basis event and the safety model function is therefore not applicable.

A new curve was generated for the 5% test using the FLODISC code with a power to flow ratio of 1/15 based on the steady state power condition (5% power to 75% flow). This curve is provided as Figure 3.3. The flow dependent flow distribution curve from the safety evaluation[4] is shown in Figure 3.4. Comparison of Figures 3.3 and 3.4 make evident the fact that, with the low power to flow ratio of 1/15 for the 5% test, less flow redistribution occurs at very low flow rates. To ensure that FLODISC was used in a manner consistent with the original design basis studies, a repeat calculation was performed with power to flow equal to unity, and Figure 3.4 was reproduced.

3.2 Results of the Safety Model Analysis

The results of the safety model analysis are provided in Figures 3-5 through 3-7. The first of these, Figure 3-5, provides the time dependent values of subassembly inlet temperature and the HF011 T8/8 temperature sensor. The inlet temperature remains essentially constant during the test. The important curve for assessing model conservatism is the fast thermocouple at the top of the core, HF011 T8/8. This analysis predicts a peak temperature of 664°F during the transient.

Figure 3-6 provides the predicted total primary flow following the initial decay from 75%. The minimum in flow at ~2 minutes can be clearly seen on this figure. The power during the transient is shown in Figure 3-7. Neutron power and decay power are shown separately along with total power.

4. NOMINAL PREDICTIONS

In order to demonstrate the degree of conservatism in the nominal design and the safety margins provided by this design, a best estimate prediction was performed. For this purpose a nominal version of IANUS was prepared. Table 4.1 compares this model to the safety model. The results of testing which has occurred since the safety model was established, both at component testing facilities and in the FFTF itself, have been included. Further the predictions for the thermocouple response are not based on the steady-state FLODISC model which was develoepd for IANUS, but on a multi-assembly thermal-hydraulic code developed at HEDL called CORA. This code uses the system conditions predicted by IANUS as boundary conditions and provides a detailed analysis of the FOTA assembly,including the effect of radial heat transfer from assembly to assembly.

Using the nominal data, IANUS runs were performed simulating the total FFTF behavior. The core inlet pressure, inlet temperature, decay heat, reactor vessel level, and upper plenum temperature resulting from this simulation were provided as inputs to the CORA code. Because CORA models clusters of assemblies, consistent physical modeling was checked using a CORA simulation of an average fuel assembly and comparing this with the average fuel assembly modeled by IANUS.

The core thermal/hydrualics code, CORA, was used to simulate the Row 2 FOTA as a central assembly in a cluster of 19 assemblies. The results of this simulation are reported in HEDL-TC-1778 (Appendix C) as is a similar simulation for the Row 6 FOTA. The prediction for TX1016 in Figure 4.1 is taken from Appendix C as is the predicted upper assembly thermocouple response in Figure 4.2.

Total prediction uncertainty based on IANUS and CORA modeling uncertainties (both structural and numerical) as well as plant condition uncertainties (which may persist even in a post-test analysis) are also included in Appendix B . The uncertainty in the predicted peak ΔT is $\sim 15°F$. A test temperature to prediction temperature deviation which falls within this bound, would indicate probable validity of the "best estimate" models with anticipated uncertainty bounds. Larger deviations, if any, will provide an initial focus for more detailed evaluation.

5. PLANT COMPONENT TRANSIENTS

The thermal transients in this test are expected to be unimportant in magnitude and rate of change. Since temperature differences throughout the test will be below 50°F (except across core region), no significant thermal stresses are expected.

6. SUMMARY

A safety model for a specific thermocouple in the Row 2 FOTA during a FFTF natural circulation test from 5% reactor power has been defined. This model includes appropriate adjustments for differences between test conditions and the equilibrium design case for which the original safety evaluation model was developed. Consistency in methodology with the development of the original was maintained. The safety model prediction for the Row 2 FOTA thermocouple has been prepared based on an assumed power history at the time of the scram. Similarly, the best estimate model has been described with the role of IANUS and CORA specified, and the nominal prediction prepared. In either case experimental conditions may require a post-test update of the predictions using identical methodology, but with the actual test conditions.

7. REFERENCES

1. "Interim Summary of FFTF Natural Circulation Test Plans," dated September 1977.
2. "Safety Evaluation Report related to operation of Fast Flux Test Facility," NUREG-0358, August 1978, and Supplement 1, May 1979.
3. Letter, R. L. Ferguson, FFTFPO to William Gammill, USNRC, responding to "Transmittal of Meeting Summary, April 28, 1978, Natural Circulation," FTF:GDB:E457, dated July 5, 1978.
4. "FFTF Natural Circulation Evaluation: Transition to Natural Circulation," HEDL-TC-557, April 1976.
5. "Simulation of the Overall FFTF Plant Performance," HEDL-TC-556, S. L. Additon, et al, April 1976.
6. Novendstern, F. H., "FLODISC-A Computer Code for Calculations of Flow Distributions in Paralleled Channels," FRT-695, March 7, 1972.

TABLE 3.1

OVERALL HOT CHANNEL FACTORS

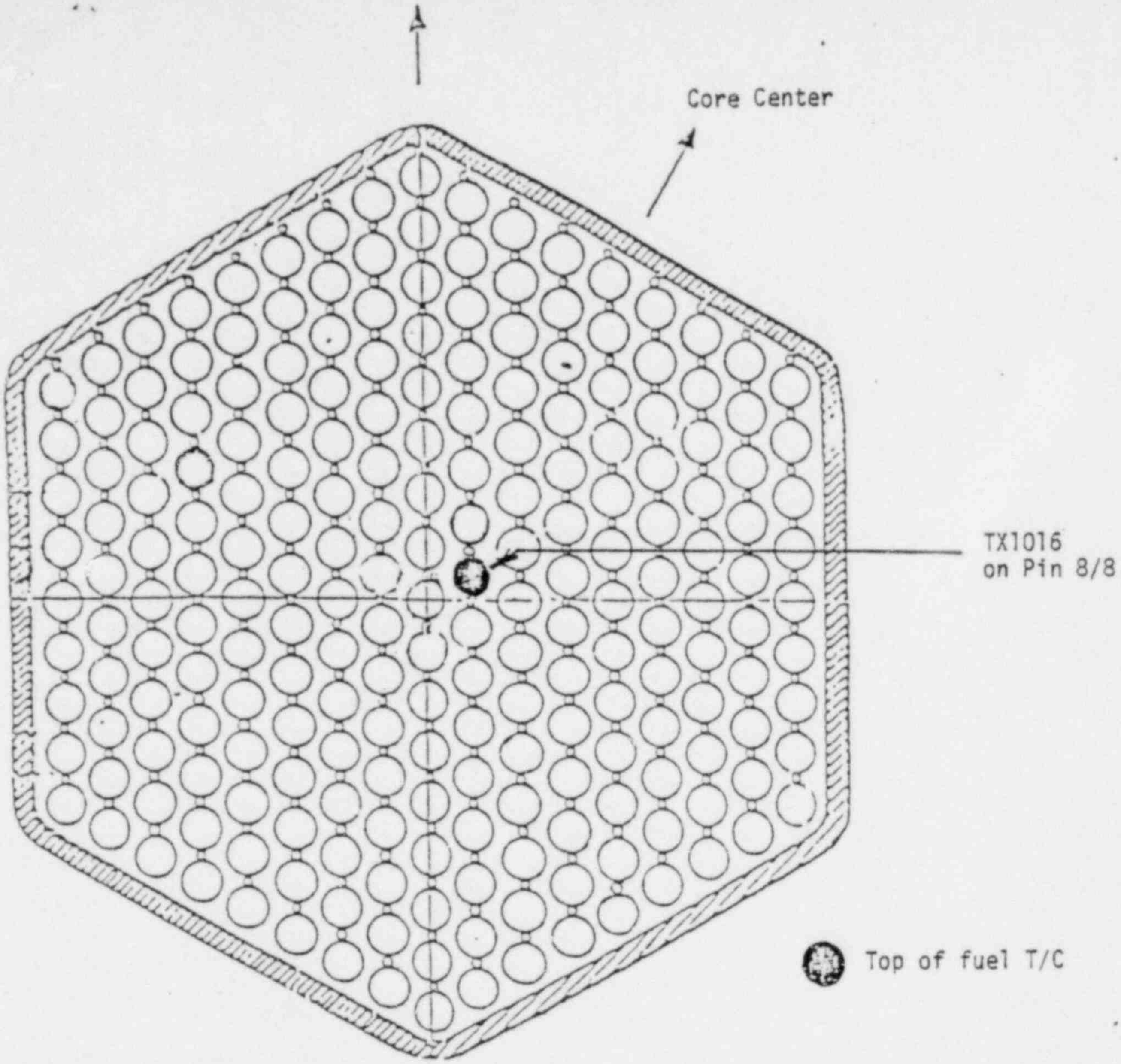
FOR 5% TEST

	Safety Model Hot Channel	FOTA Safety Model Hot Channel
I. HIGH POWER ASSEMBLY FACTORS		
DIRECT		
1. NUCLEAR POWER DISTRIBUTION	1.439	1.2774
2. FLOW ORIFICING	0.9174	0.9174
TOTAL	1.320	1.1719
II. HOT CHANNEL FACTORS	COOLANT	COOLANT
DIRECT		
1. INLET FLOW MALDISTRIBUTION	1.05	1.05
2. INTRASUBASSEMBLY FLOW MALDISTRIBUTION	1.14	1.14
3. INTERCHANNEL COOLANT MIXING	1.00	1.00
4. POWER CONTROL BAND	1.02	1.02
5. WIRE WRAP PEAKING	-	-
DIRECT SUBTOTAL	1.22	1.22
STATISTICAL (3σ)		
1. FISSILE FUEL MALDISTRIBUTION	1.035	1.035
2. POWER LEVEL MEASUREMENT	1.087	1.200
3. NUCLEAR POWER DISTRIBUTION	1.060	1.060
4. ROD DIAMETER, PITCH & BOW	1.011	1.011
5. FILM COEFFICIENT	-	-
6. CLAD CONDUCTIVITY & THICKNESS	-	-
7. NON-EQUILIBRIUM RATED CORE	1.02	1.02
STATISTICAL SUBTOTAL 3σ	1.114	1.213
HOT CHANNEL FACTOR TOTALS (DIRECT COMBINED WITH 3σ)	1.359	1.480
OVERALL HOT CHANNEL FACTORS	1.794	1.734

TABLE 4.1 NOMINAL MODEL VERSUS SAFETY MODEL SUMMARY

Model Feature or Parameter	Nominal Model	Safety Model
Computer Program Used	IANUS, CORA	IANUS, Steady-State FLODISC
Reactor Core Flow Distribution	Dynamic Model of Parallel Channels	Hot Channel Based on Quasi-Steady State Analysis
Reactor Bypass Flow	Dynamic Model	Fixed Based on Steady State
Decay Power	Based on HEDL-TME 77-13	Based on HEDL-TME-71-27 with 25% Uncertainty
Reactor Pressure Drop	Based on Plant Data	+20% Uncertainty
Pump Stopped Rotor Pressure Drop	Fit to LMEC Test Data	IANUS Design Equation (+15% Uncertainty)
Loop Pressure Drops	Based on Plant Experimental Data	Design Values in IANUS
DHX Post-Scram Response	Based on Plant Experimental Data	IANUS Design Equations
Pump Coastdowns	Fit to LMEC and Plant Data	IANUS Design Equations

1) The Safety Model is that Model used to predict the design case transient in FSAR Section 15 and HEDL-TC-557.



Core Location 1202
View from Top

FIGURE 3.1 Row 2 FOTA Thermocouple Location

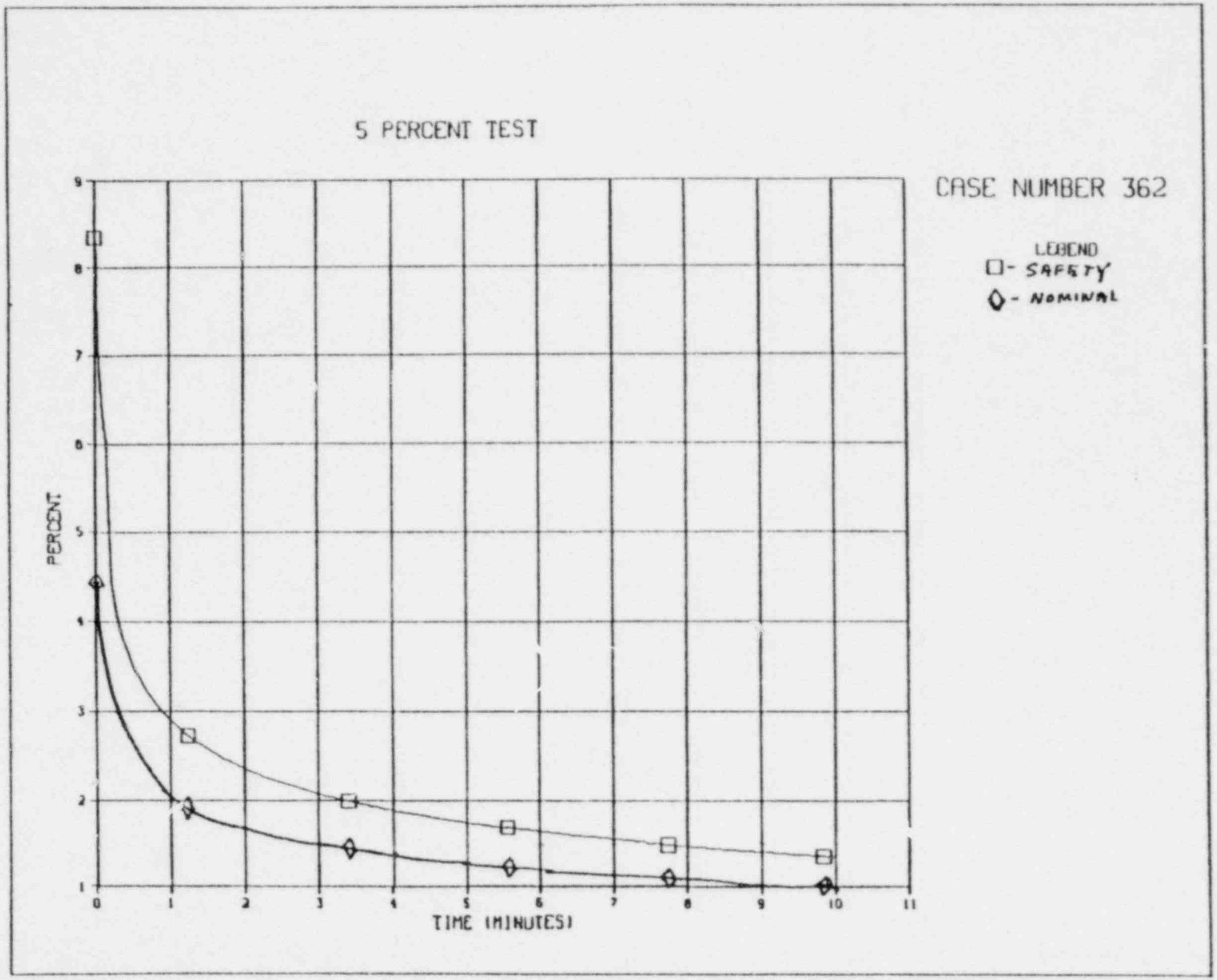


FIGURE 3.2 5% Power Test Decay Power

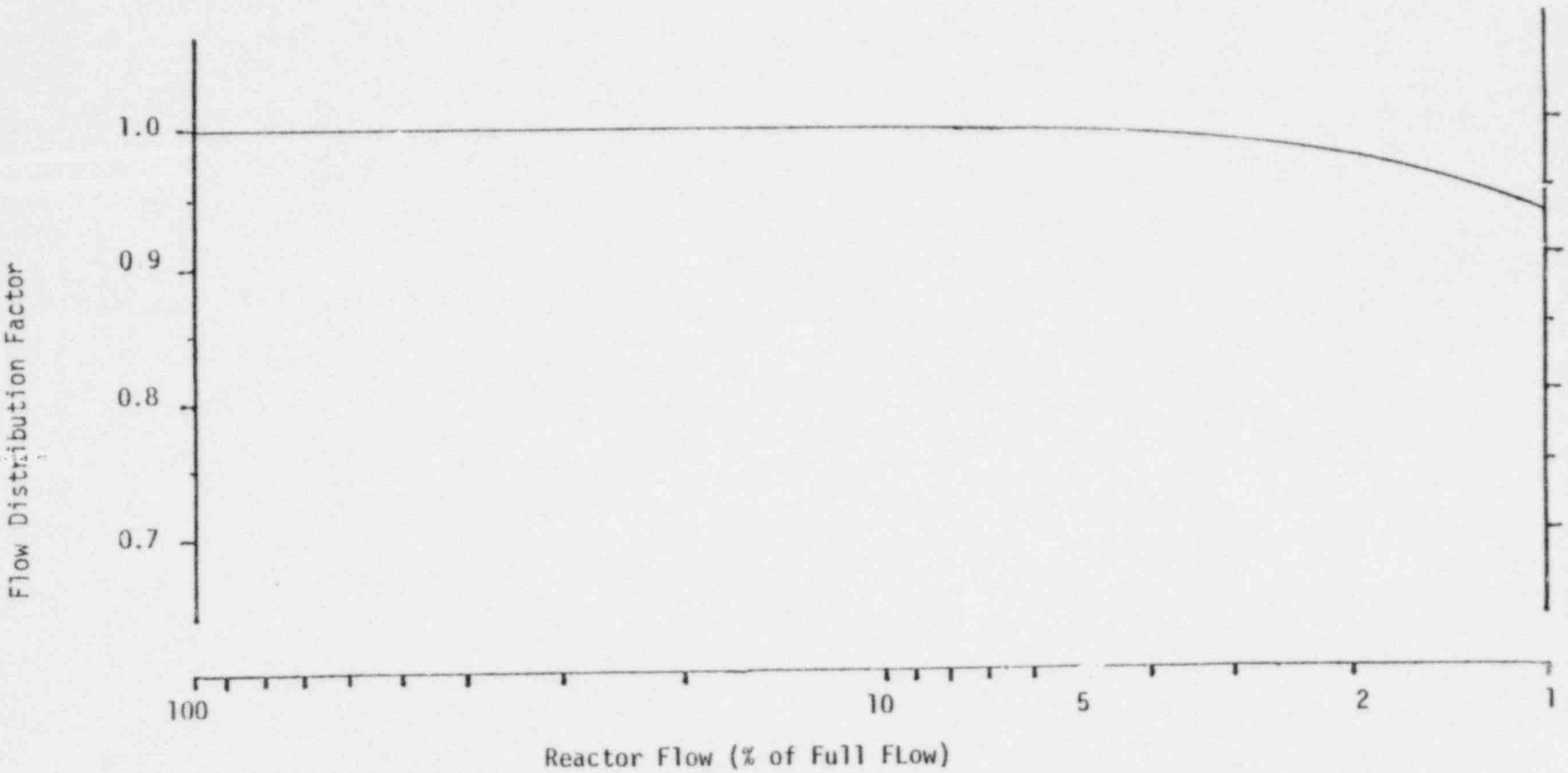


FIGURE 3.3 FOTA Inter-Assembly Flow Distribution Factor for 5% Power

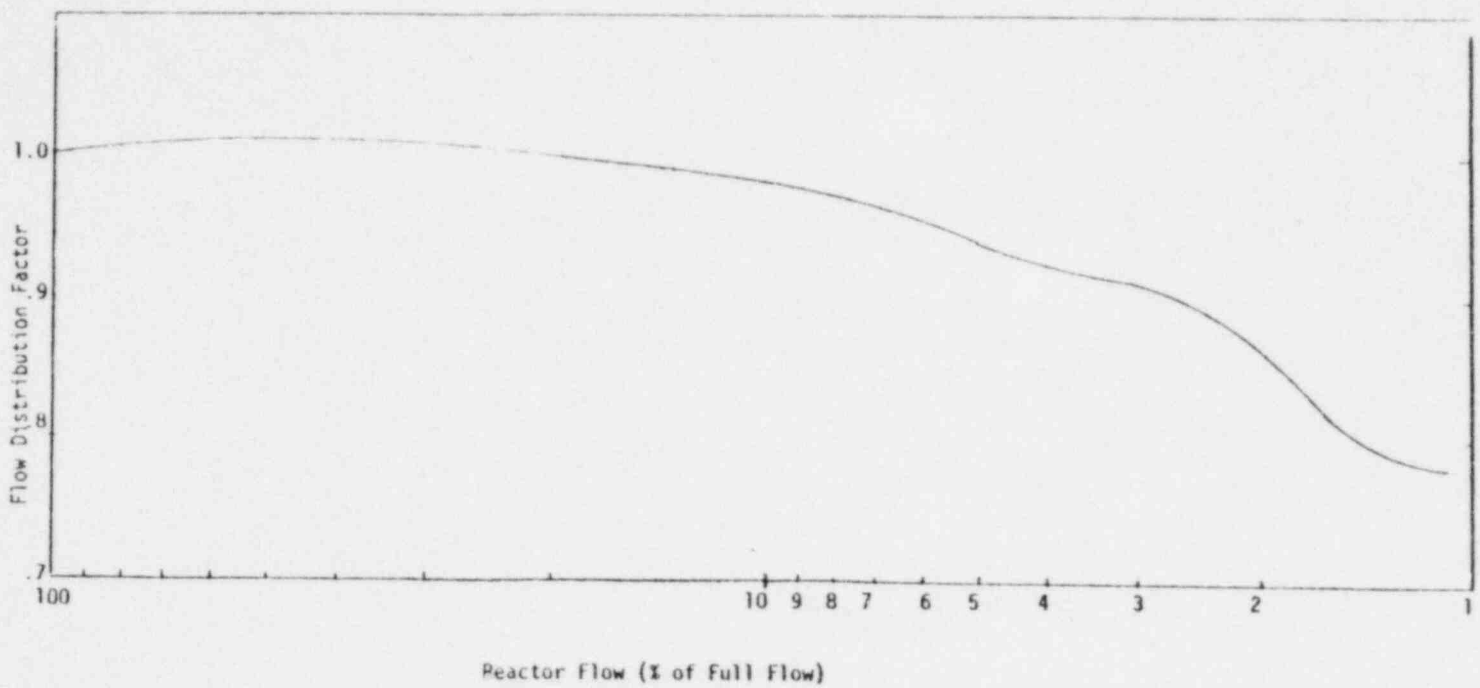


FIGURE 3.4 Hot Channel Inter-Assembly Flow Distribution Factor

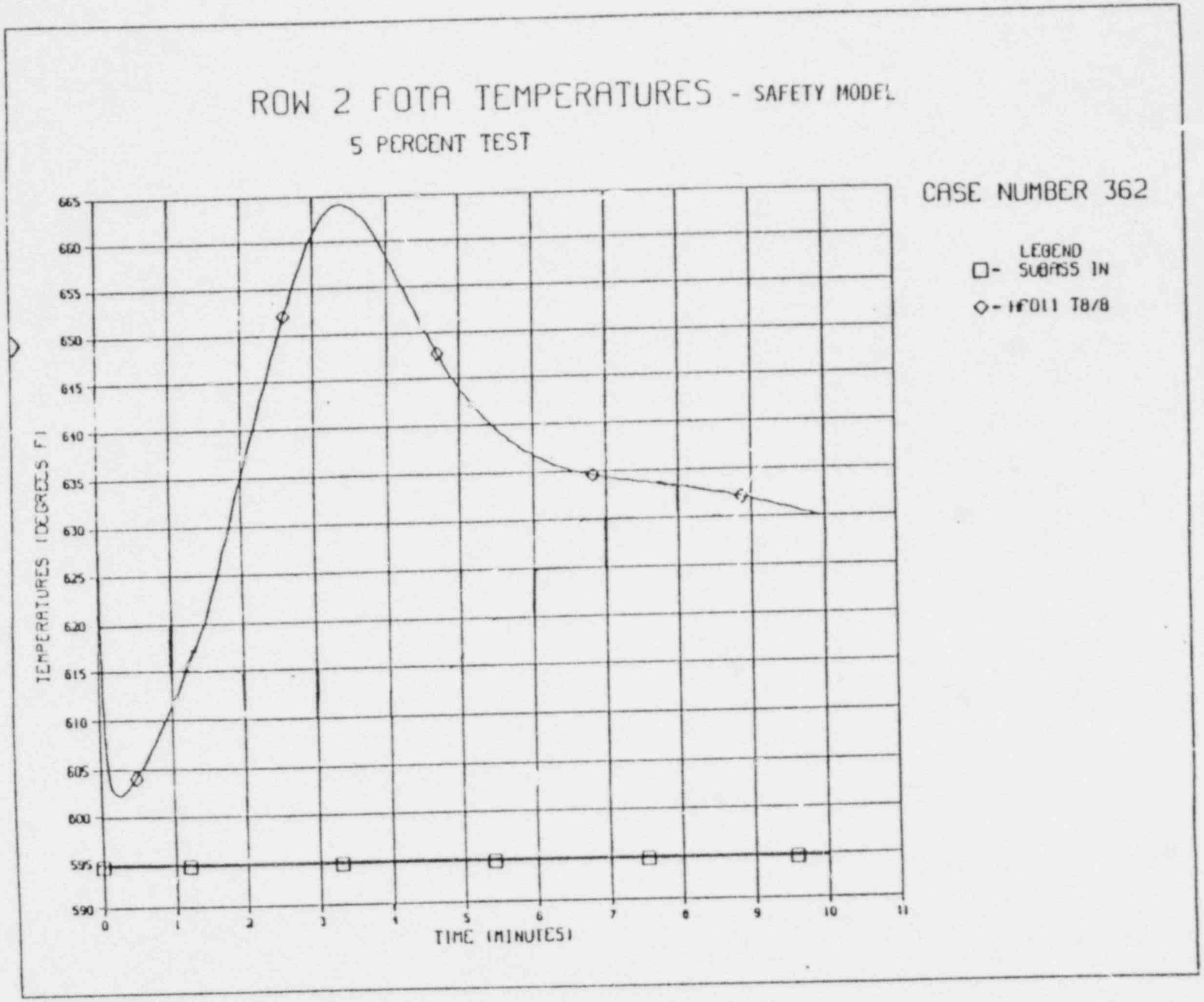

ROW 2 FOTA TEMPERATURES - SAFETY MODEL
5 PERCENT TEST
CASE NUMBER 362
LEGEND
□- SUBASS IN
◇- HFO11 T8/8
TEMPERATURES (DEGREES F)
665
660
655
650
645
640
635
630
625
620
615
610
605
600
595
590
0
1
2
3
4
5
6
7
8
9
10
11
TIME (MINUTES)

FIGURE 3.5

PRIMARY LOOP FLOW INFORMATION

5 PERCENT TEST

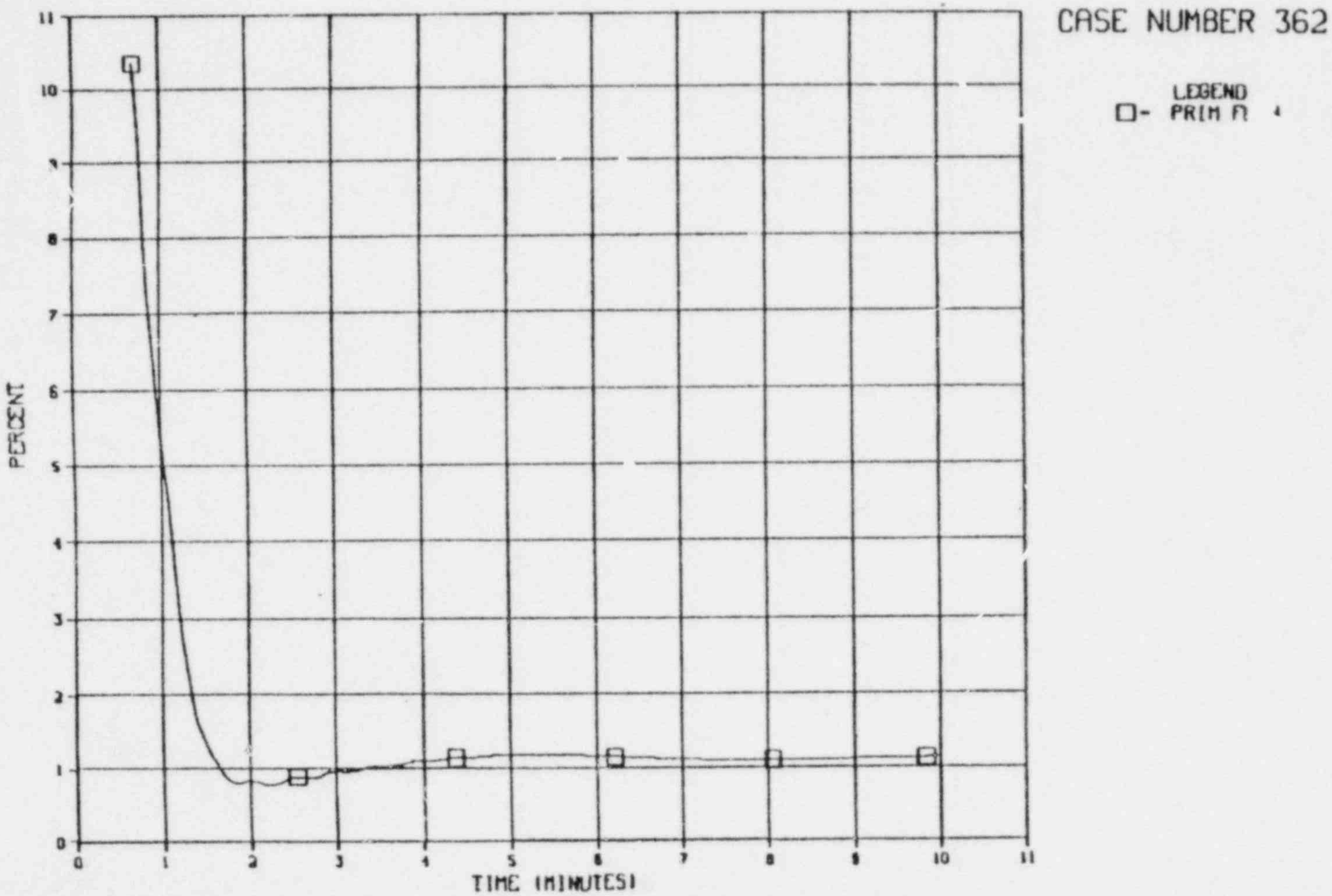


FIGURE 3.6

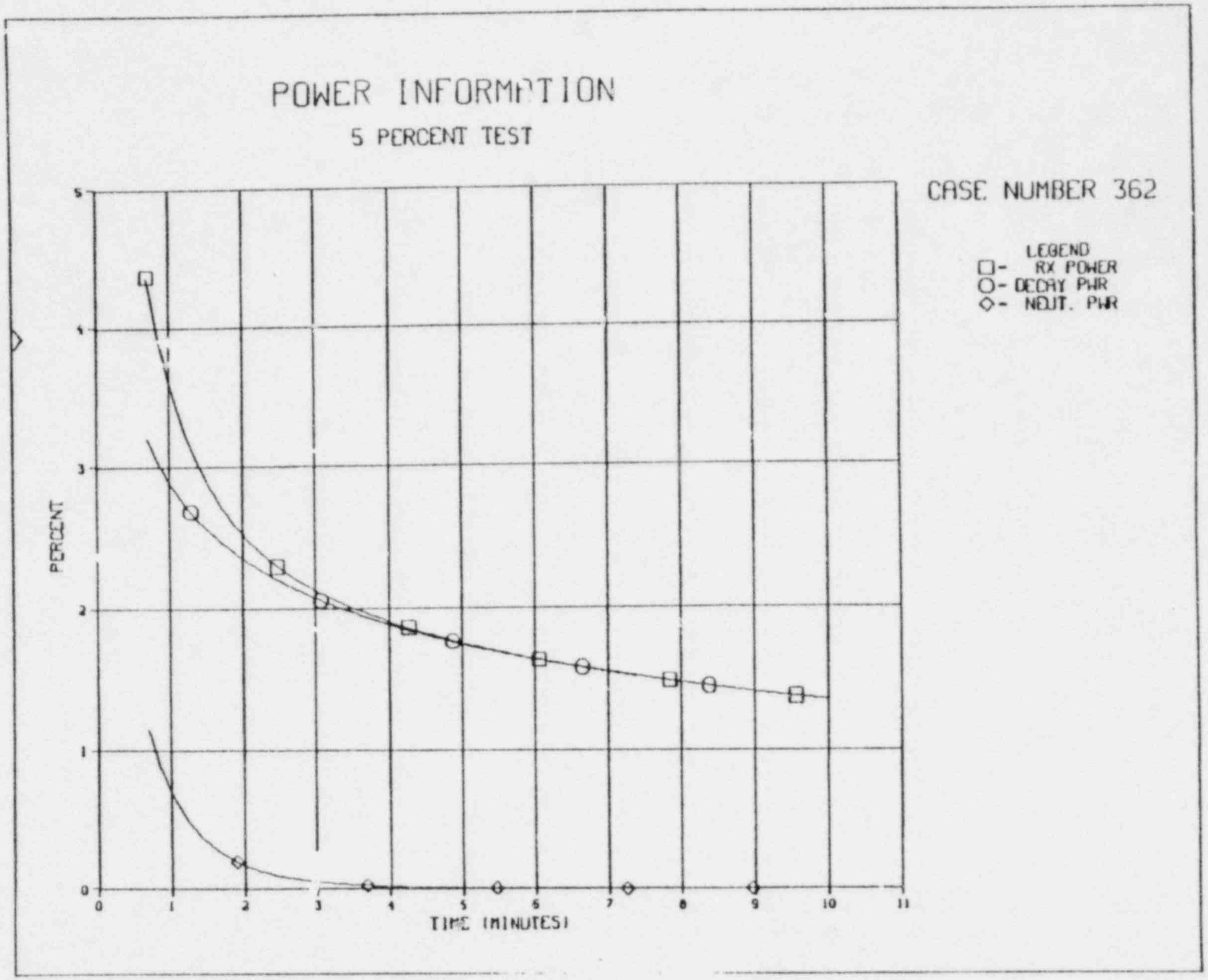

POWER INFORMATION
5 PERCENT TEST
CASE NUMBER 362
LEGEND
□- RX POWER
○- DECAY PWR
◇- NEUT. PWR
PERCENT
0
1
2
3
4
5
0
1
2
3
4
5
6
7
8
9
10
11
TIME (MINUTES)

FIGURE 3.7

FIGURE 4.1 Calculated Row 2 FOTA Top of Fuel Coolant Temperatures for T/C TX1016 on Pin 9

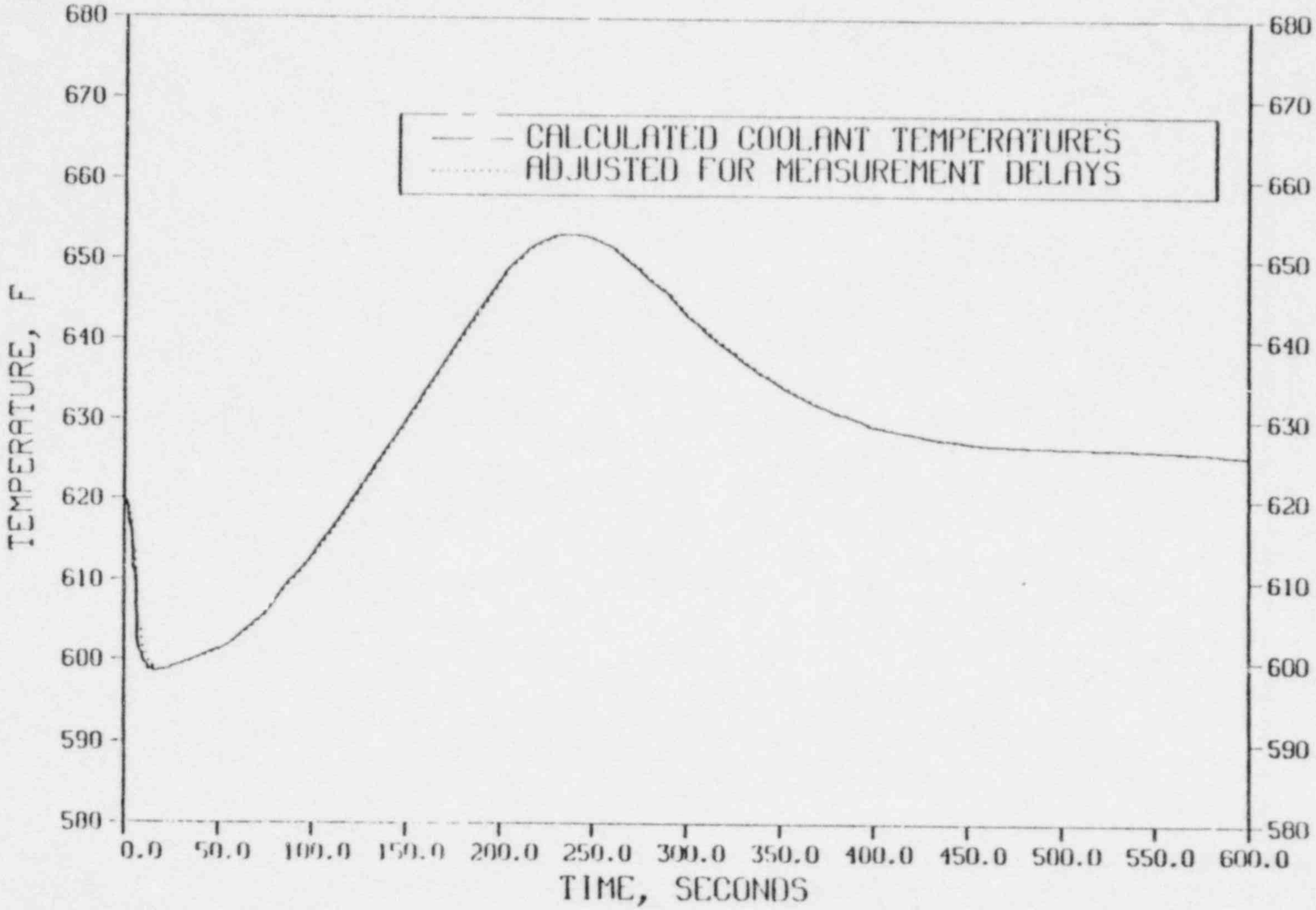


FIGURE 4.2 Calculated Row 2 FOTA Instrument Stalk Coolant Temperatures at T/C Location