CATEGORY II
CRBRP-ARD-0310

Clinch River Breeder Reactor Plant

VERIFICATION OF THE CRBRP NATURAL CIRCULATION CORE ANALYSES METHODOLOGY WITH DATA FROM FFTF NATURAL CIRCULATION TESTS

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1.0 SUMMARY

To culminate the core portion of the CRBRP Natural Circulation Verification Program (NCVP), pre-test and post-test predictions of the thermal and hydraulic response were made for the FFTF natural circulation tests initiated at three different power levels, utilizing CRBRP methodology for Core Analyses. Prior to these tests, pre-test predictions were made based on assumed test conditions. Following the completion of all three tests, post-test analyses were performed utilizing the actual test boundary conditions and the same methodology used in the pre-test predictions. These predictions provide the basis for the validation of the analytical tools used in making such predictions. The computer codes utilized in these predictions consist of DEMO, COBRA-WC and FORE-2M. Predicted sodium temperatures at the top of the active fuel section for the Fueled Open Test Assemblies (FOTA's) are presented for these tests. Good agreement was obtained between these predictions and measured values. In all cases the predictions were conservative. The acceptance criteria which were established and presented with the pre-test predictions in determining the sodium temperature responses in the core are satisfied. The model and methodology are therefore validated for such design transient analyses.

2.0 INTRODUCTION

The objective of the CRBRP Natural Circulation Verification Program (NCVP) is to analytically and experimentally verify calculational models for the description of the thermal-hydraulic behavior of LMFBR plants when making transition to or operating in the Natural Circulation mode. This verification is an important part of demonstrating the capability of adequate decay heat removal by means of natural circulation in LMFBR's. A major component of the NCVP is the comparison of analytical predictions with experimental data from large systems facilities similar to CRBRP. Natural circulation testing performed at the Fast Flux Test Facility (FFTF) presents an excellent opportunity for such a comparison. A program of pre- and post-test predictions of FFTF tests was therefore implemented. The methodology and approaches are the same as those to be used for the prediction of natural circulation events in CRBRP.

The FFTF natural circulation test conditions encompass the transition into natural circulation cooling from a series of steady state initial conditions for a loop type LMFBR. All the tests were initiated with a reactor scram and main coolant pump trip. The tests included[1]:

A. A transition to natural circulation in the primary loop from 5% reactor power, 75% primary loop flow. The secondary loop pumps coasted down to 10% speed at which time pony motors engaged and provided 10% secondary loop forced flow.

B. A transition to natural circulation in both primary and secondary loops from 35% reactor power and 75% flow (one secondary pony motor was operating during the test).

C. A transition to natural circulation in both primary and secondary loops from 75% reactor power and flow.

D. A transition to natural circulation in both primary and secondary loops from 100% reactor power and flow.

Of these tests, the ones that are of most direct interest for the NCVP are the transients from the three normal power levels, i.e., 35%, 75%, and 100% reactor power levels. This report will present: 1) the CRBRP methodology used for the predictions of reactor inlet flows and temperatures during the transient tests, and the peak channel sodium temperatures at top of the fuel section in both FOTA's; 2) the pre-test predictions on the above-mentioned three transient tests; 3) the post-test assessment of boundary conditions, i.e., decay heat evaluations based on actual power history, and the reactor vessel pressure drop correlations used for the DEMO input; 4) the post-test predictions and comparison with core test data, and 5) discussions and conclusions. The validation of the DEMO code with the plant data will be covered in detail in a separate report.

3.0 ANALYTICAL APPROACH

To analyze LMFBR decay heat removal capability, a system of three computer codes was developed which are used in sequence, i.e.: 1) DEMO code for plant-wide analyses [Reference 2]; 2) COBRA-WC code for core system analyses [Reference 3]; and 3) FORE-2M code for localized core hot rod analyses [Reference 4]. The first code predicts the coupled performance of the reactor, the heat transport systems and the steam generating system. It thus provides the overall system variables such as the flow through the reactor and bulk temperatures entering and exiting the core. The FFTF version of DEMO employed a three region model in the reactor simulation, i.e., fuel, non-fuel, and bypass (Figure 1). This simulation included models for (a) reactor neutron kinetics and decay power, (b) thermal-hydraulic models of the core and surrounding regions, and (c) thermal-hydraulic models of the inlet and outlet plena. The modeling in the DEMO code also substituted a dump heat exchanger for the steam generator. The COBRA-WC code, which accounts for core inter- and intra-assembly flow and heat redistribution, predicts the boundary conditions for a peak rod, or a cluster of rods in fuel and blanket assemblies,[5] given the reactor total flow, pressure drop and core inlet temperature from the DEMO code. With the localized peak channel flows and heat interchange information as a function of time provided by the COBRA-WC code, the hot channel coolant temperature for an individual rod, is predicted by the FORE-2M code which includes uncertainties and also accounts for localized rod effects (e.g., hot spot due to wire wrap spacer, fuel restructuring dynamic fuel/cladding gap conductance, etc.). This hot channel coolant temperature is used as the basis for determining the acceptability of natural circulation during decay heat removal.[6]

Figure 2 shows the interaction between these three codes used in the analysis. Two models were used in the COBRA-WC computation. One modeled a sector of the core and the other, a cluster of assemblies. In addition to the initial reactor inlet flow rate, W_{INLET}, and inlet temperature as a function of time, $T_{in}(\theta)$, the average core ΔP (θ) was provided by the DEMO code for input to the COBRA-WC code using the core model. The reactor inlet flow versus time calculated by the COBRA-WC code was checked with that determined by DEMO, because at high assembly temperatures and low flow, the core flow is

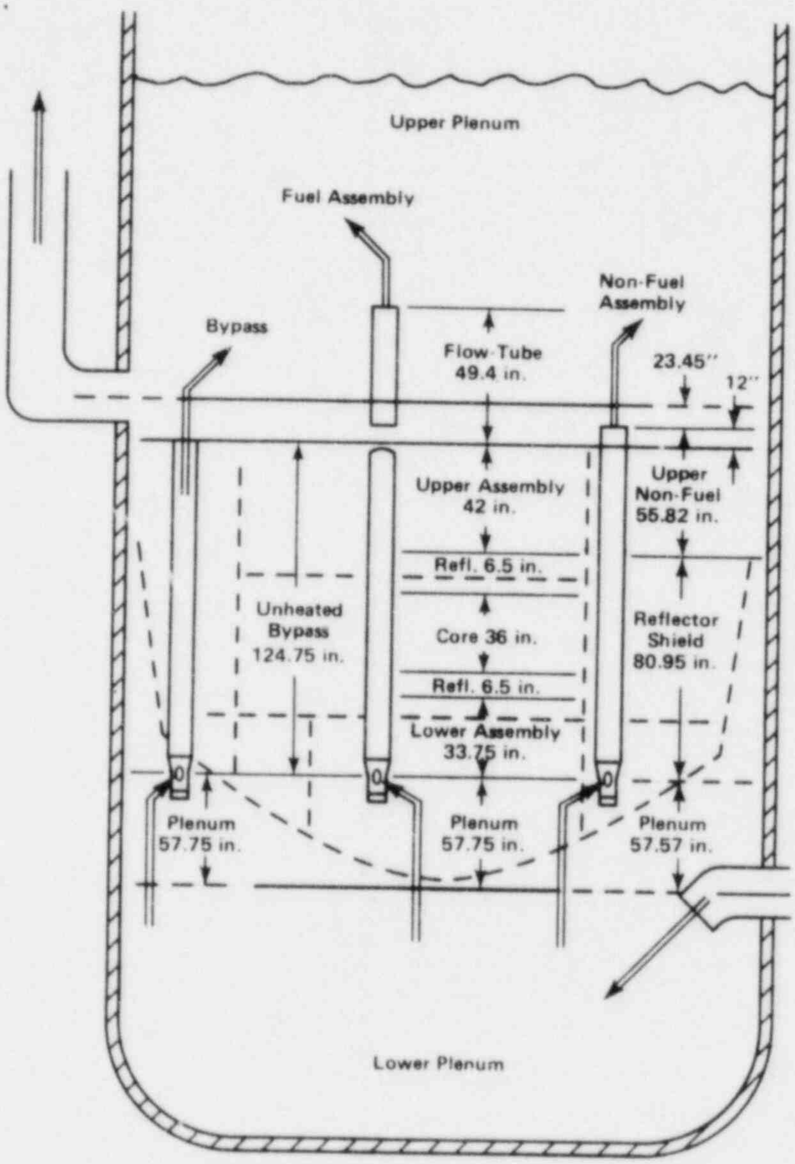


FIGURE 1: ILLUSTRATION OF DEMO MODELING OF FFTF REACTOR

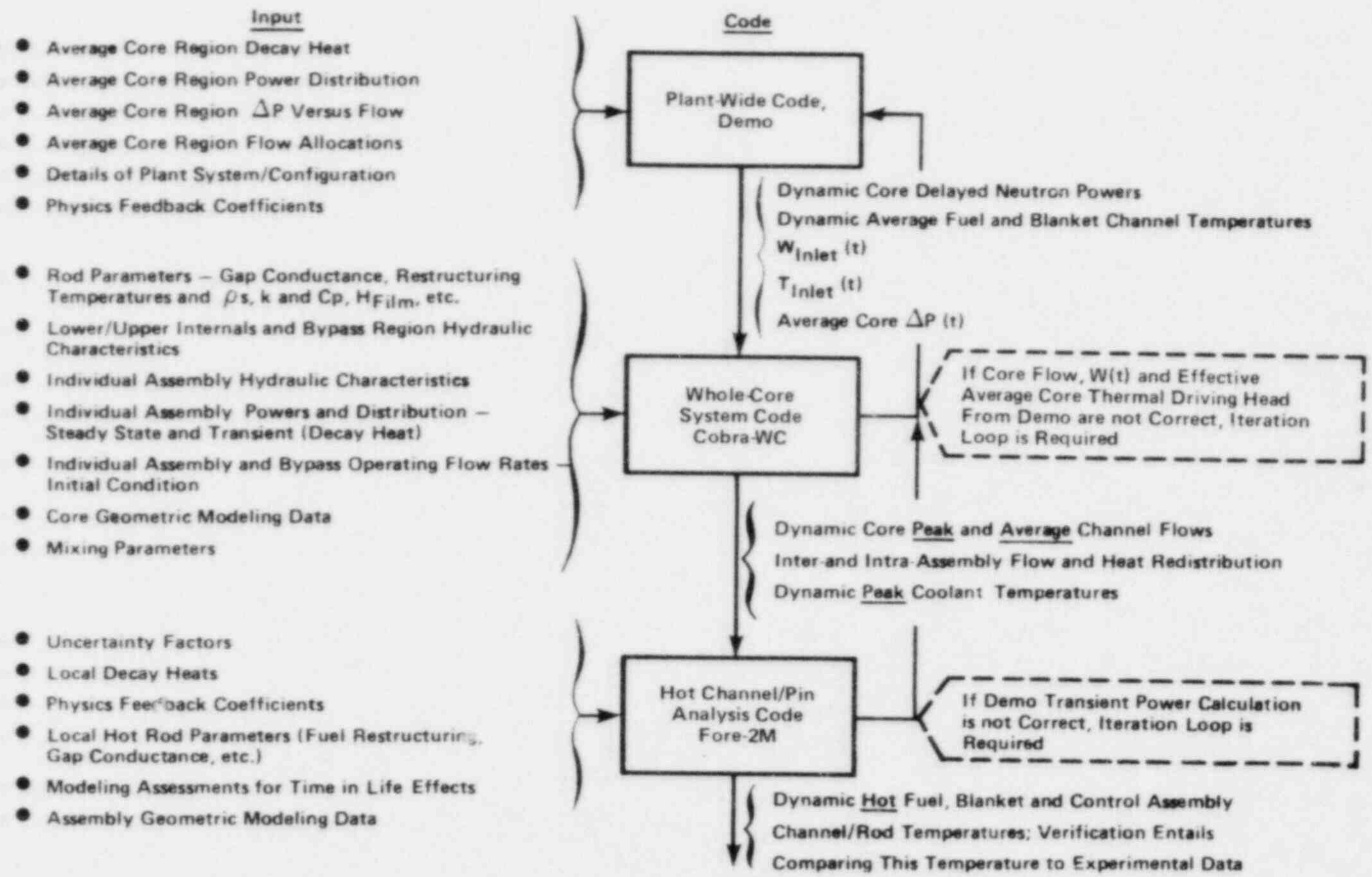


Figure 2. LMFBR NCVP Analysis Procedure

very sensitive to average ΔP and the core thermal driving head. The simplified core model in DEMO requires the ΔP versus flow correlation to be calibrated by COBRA-WC in separate calculations. If the above-mentioned reactor inlet flow do not check, the calculation of DEMO and COBRA-WC will be repeated with a modified ΔP versus flow correlation, which will result in a closer match of the reactor flow between these two codes. This model provided the whole-core flow and heat redistribution analysis of all the parallel core assemblies and bypass regions. By necessity, the calculational matrix or model representation was kept coarse, i.e., a single channel model was utilized for each core assembly in the sector. This yielded assembly flow information and the temporal pressure drop across the assemblies for inputting data to a second COBRA-WC calculation which utilized an assembly-cluster model containing more detailed nodal representation. The latter model provided a better simulation in calculating peak channel coolant temperatures for detailed inter-assembly heat conduction and intra-assembly flow and heat redistribution which have been shown to provide significant benefits.[7] Figures 3 and 4 show the COBRA model of such clusters for the Fuel Open Test Assembly (FOTA's) in Rows 2 and 6.[8] A group of rods rather than individual sub-channels were modeled. The central assembly (FOTA) was simulated by a 37-channel model while the surrounding assemblies were simulated by 19-channel models. The relative position of the rods and COBRA-WC channel representation are shown in Figure 5. The results of this analysis provided data for a further detailed analysis on a single "hot rod" using the FORE-2M code.

A linkage between the COBRA-WC and FORE-2M codes was developed to incorporate the inter- and intra-assembly phenomena into the localized hot rod transient analyses.[6] For each axial node of the hot rod modeled in FORE-2M, a heat balance was performed using the expression for the heat transferred to the coolant at that section, $Q_c(x,\tau)$ as

$$Q_c(x,\tau) = Q_r(x,\tau) + Q_{ex}(x,\tau) \qquad (3\text{-}1)$$

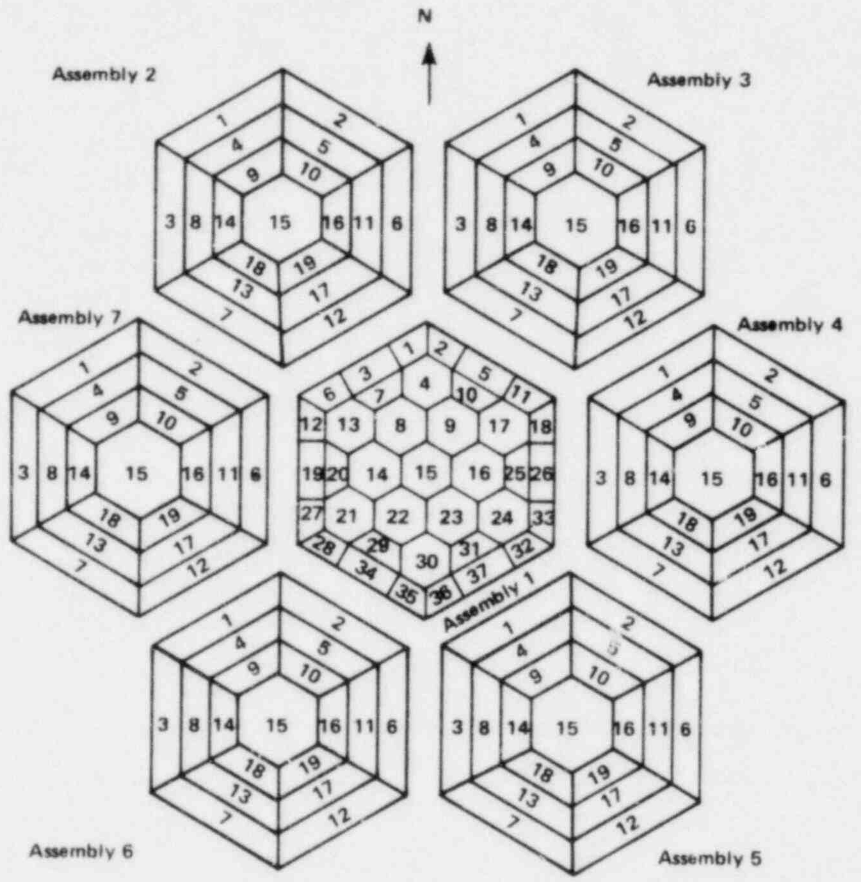


FIGURE 3. COBRA–WC MODEL OF A 7–FUEL ASSEMBLY CLUSTER SURROUNDING ROW 2 FOTA

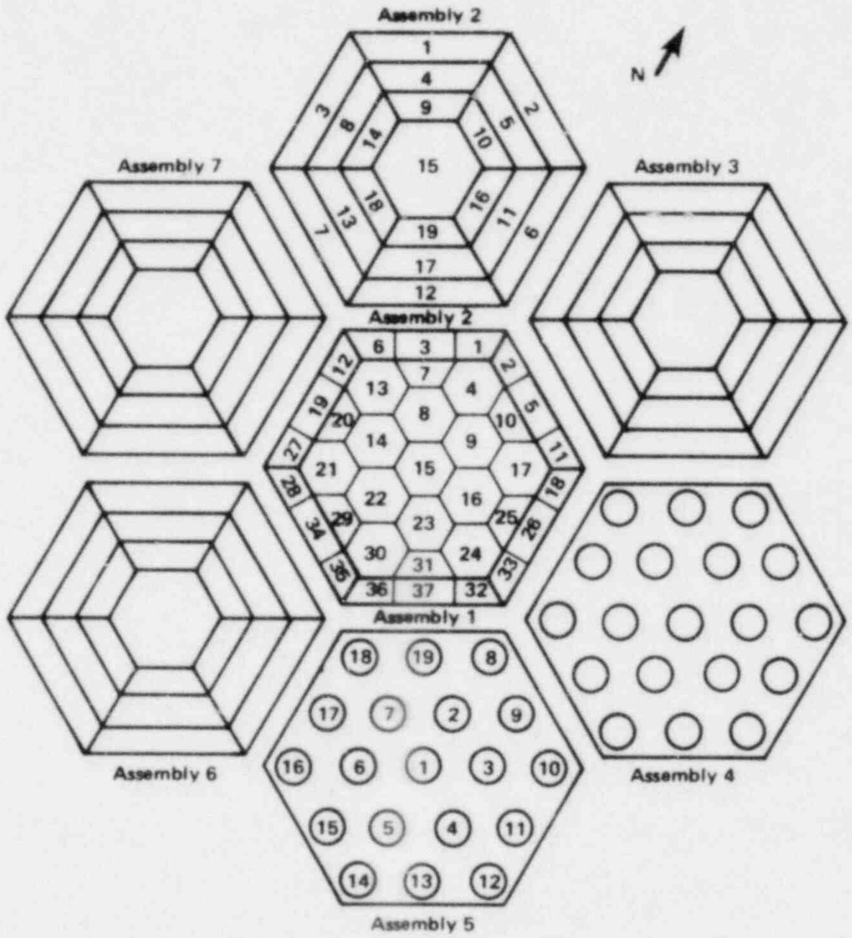
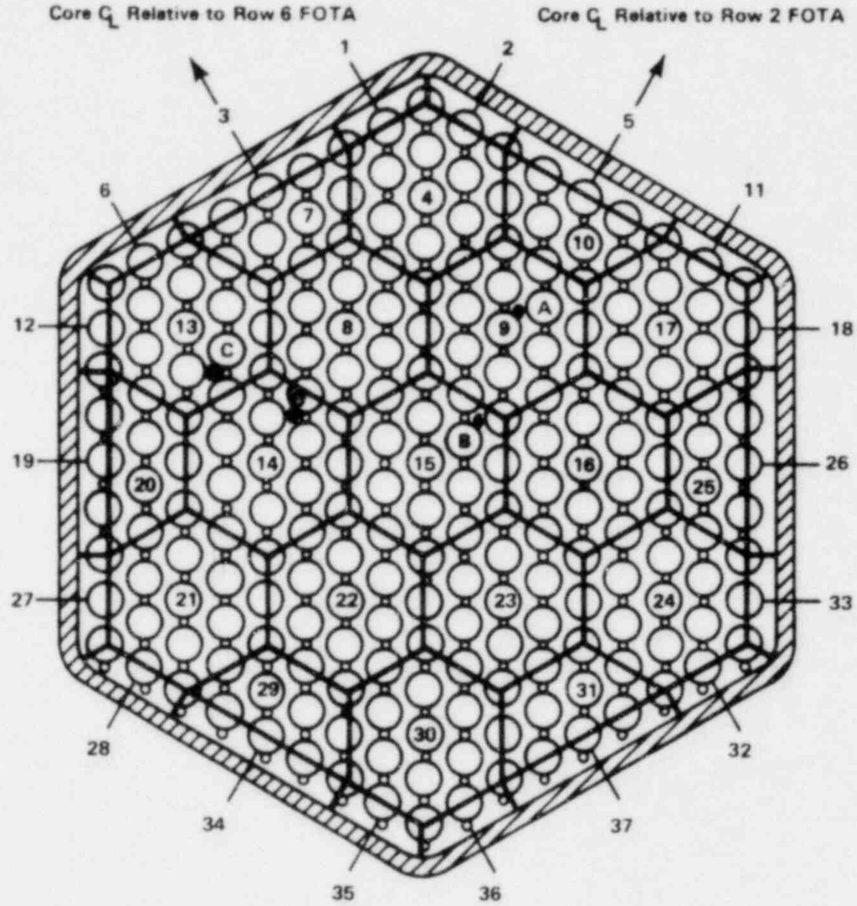


FIGURE 4: COBRA-WC MODEL OF A 7-ASSEMBLY CLUSTER SURROUNDING ROW 6 FOTA CONTAINING FIVE FUEL ASSEMBLIES AND TWO RADIAL REFLECTOR ASSEMBLIES (ASSEMBLIES 4 & 5)



Row 2 FOTA Location A, (Tx1018) Rod (5,7) at Top of Fuel
Row 2 FOTA Location B, (Tx1016) Rod (8,8) at Top of Fuel
Row 6 FOTA Location C, (Tx9018) Rod (4,4) at Top of Fuel
Row 6 FOTA Location D, (Tx9016) Rod (6,6) at Top of Fuel

Figure 5. Typical COBRA-WC 37 Channel Model of a 217 Rod Assembly

where

$Q_r(x,\tau)$ = heat transferred from the rod surface at axial location x and time τ;

$Q_{ex}(x,\tau)$ = coolant heat input or loss due to radial conduction heat transfer and flow redistribution to adjacent coolant channels; directly input from COBRA-WC.

Coupled with this, the axial mass flow rate for each axial node, $G(x,\tau)$ was also input from COBRA-WC analyses. Boundary conditions for the COBRA-WC (e.g., plenum-to-plenum pressure drop and coolant inlet temperature) were furnished by the plant-wide code, DEMO for several cases: i.e., the "best estimate" or nominal case as well as cases with uncertainties. Likewise, corresponding modeling of the core parallel flow network, with regard to pressure drop and decay heat uncertainties, can be used in the COBRA-WC analyses for input to the FORE-2M hot rod temperature predictions.

To evaluate the effects of core uncertainties, hot channel sub-factors were used (Table 1) which were given in Reference 9 with the exception of factors 2 and 3. Similar to those used in steady state calculations, the direct and statistical type factors were conservatively applied by the semi-statistical method. For instance, the factor due to inlet flow uncertainty of -5%[9] to the FOTA was conservatively assumed to occur. The statistical subfactors which were statistically combined include those due to uncertainties in the decay heat calculations ($\pm$25% of decay heat values which gives a $\pm$17% uncertainty in Row 6 FOTA sodium temperature at the time the peak temperature occurs), the pressure drop calculational uncertainties (specified in Reference 10) and the factors for power level measurement, nuclear power distribution and coolant property uncertainties.[6,9] A more pessimistic compilation of uncertainties applicable to conservative design type analyses (not controlled experimental conditions) were described in Reference 6.

The effects of the first two statistical subfactors (i.e., decay heat and pressure drop uncertainties) were determined by comparing different cases with maximum variation of one of the quantities in calculations using all three codes. These different cases are listed in Table 2. The pre-scram

TABLE 1

UNCERTAINTY FACTORS APPLIED TO HOT CHANNEL TEMPERATURES IN FØRE-2M CALCULATIONS FOR N/C TESTS

A. DIRECT SUBFACTORS	
1) INLET FLOW UNCERTAINTY, F_1	1.05
B. STATISTICAL SUBFACTORS (3σ)	
2) PRESSURE DROP CALCULATIONAL UNCERTAINTY EFFECT, F_2	1.15*
3) DECAY HEAT UNCERTAINTY EFFECT, F_3	1.17 (Row 6 FOTA)* 1.13 (Row 2 FOTA)*
4) POWER LEVEL MEASUREMENTS, F_4	1.079
5) COOLANT PROPERTIES, F_5	1.01
6) NUCLEAR POWER DISTRIBUTION, F_6	1.065

*These factors represent values used at the time when peak temperatures occur during the natural circulation transient. They are determined by comparison of different cases in calculations using all three codes. These values are slightly reduced from those used in Reference 11 with 35% power initial conditions (Appendix A).

TABLE 2

CASES IN PRE-TEST ANALYSES EVALUATING
UNCERTAINTY FACTOR ON TEMPERATURE RESPONSES*

Case	Decay Heat	System Pressure Drop	Pre-Scram Irradiation History
I	Nominal	Nominal	a) 1 hr & 56 hr at Initial Power b) 25 hr at Initial Power
II	Nominal	High Side "Design" Value	a) 56 hr at Initial Power b) 25 hr at Initial Power
III	125% of Nominal	Nominal	a) 56 hr at Initial Power b) 25 hr at Initial Power
IV	125% of Nominal	High Side "Design" Value	a) 1 hr & 56 hr at Initial Power b) 25 hr at Initial Power
V	75% of Nominal	High Side "Design Value"	a) 1 hr at Initial Power b) 25 hr at Initial Power

* Slightly different method was used to evaluate uncertainty factor for test initiated from 35% power/75% flow conditions (Appendix A)

irradiation history used in pretest predictions are shown in the last column with option (a) being used for the test initiated from 35% power and option (b), for the other tests. For the natural circulation test predictions, the aforementioned effects are not constant with time due to transient effects. The actual time-dependent factors were used to obtain the 3σ uncertainty on maximum temperature. Table 1 presents values of these factors used at the time when the peak temperature occurs.

4.0 PRE-TEST PREDICTIONS

4.1 DEMO PRE-TEST PREDICTIONS

DEMO pre-test predictions for the FFTF transient natural circulation tests from initial conditions of 100% power, 75% power, and 35% power levels were initially made during the first quarter of 1980. Results of these predictions and the methodology used in the analyses were documented in Reference 10. Subsequent to the issuance of the report, two changes in data input were made to reflect: 1) new information obtained from pressure drop tests of the FFTF fuel assembly inlet nozzle/shield block assembly; and 2) new expected power history for the tests. These pre-test predictions were updated and the results are presented in Reference 12 (Appendix A of this report). Using the new experimental data on pressure drop under low flow conditions through the fuel assembly inlet nozzle/shield blocks[13], the combined pressure drop ($\Delta P_{Combined,COBRA}$) for a series of steady state calculations at various flows were made with the power-to-flow ratio near one. The dynamic pressure drop was then calculated from Equation (4-1),

$$(\Delta P)_{\substack{Dynamic\\DEMO}} = (\Delta P)_{\substack{Combined\\COBRA}} - (\Delta P)_{\substack{Elevation\\DEMO}} \qquad (4\text{-}1)$$

where

$(\Delta P)_{\substack{Combined\\COBRA}}$ = reactor combined gravitational head and dynamic pressure drop calculated by COBRA for given reactor flow;

$(\Delta P)_{\substack{Elevation\\DEMO}}$ = reactor gravitational head as calculated by DEMO for the same flow conditions; and

$\Delta P_{\substack{Dynamic\\DEMO}}$ = dynamic pressure drop to be used for determining the pressure drop correlations.

The resulting dynamic pressure drop correlations for the fuel assembly, the non-fuel assembly and bypass regions are shown in Figure 6, 7 and 8, respectively. Also shown in these figures are reactor pressure drop correlations with high side ("design") pressure drop uncertainties. These uncertainties were applied by increasing the best estimate reactor ΔP values by a factor dependent on the reactor flow. This factor varied from 1.1 at full flow to about 1.38 at 2% flow. These dynamic pressure drop correlations were used in all pre-test predictions.

The post-shutdown decay power was input in tabular form to DEMO as a function of time. Table 3 presents nominal decay power used in the pre-test predictions. A pre-scram power history of 25 hours of operation at 100% and 75% power was assumed for the 100% and 75% cases.[14]* For the 35% power case, 56 hours of operation at 35% power was shown. Other initial conditions are listed in Table 4.

The pre-test predicted primary loop flows as a function of time are shown in Figure 9 for the natural circulation tests initiated at three different power levels. Figure 10 shows the predicted temperature at locations of the hot/cold leg RTD's. The hot leg RTD is located 85 feet upstream of the pump, or 65 feet downstream of the reactor vessel outlet nozzle. The temperature transients for the hot-leg RTD are therefore the reactor vessel exit temperature mitigated by 65 feet of piping. The cold-leg RTD is located just upstream of the cold-leg checkvalve. The temperature transients at the cold-leg RTD are caused by the collapse of the IHX primary outlet temperature onto the secondary inlet temperature at the tube bundle exit.

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*Reference 14 is attached in this report as Appendix B.

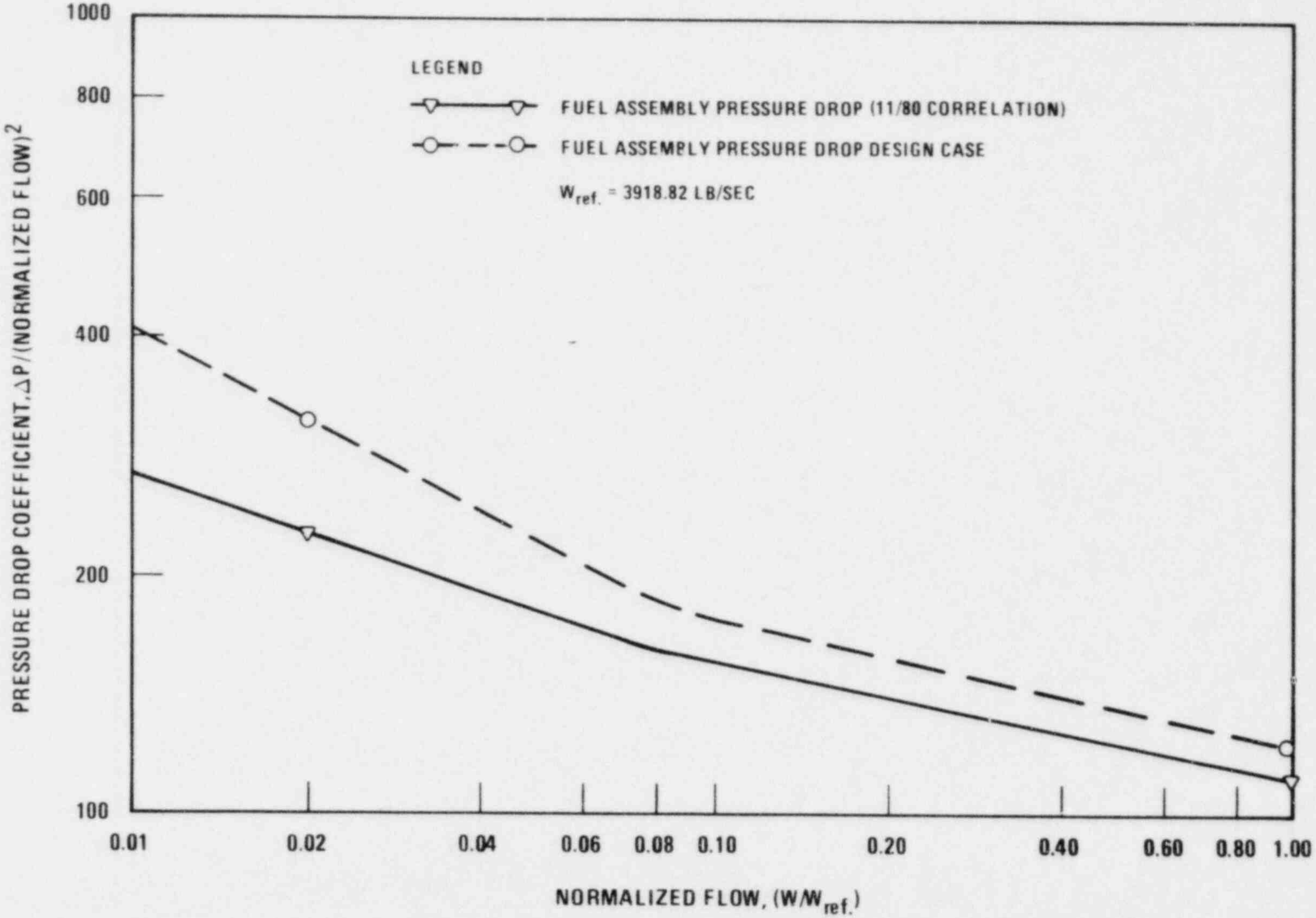

Figure 6. Fuel Assembly Group Dynamic Pressure Drop Correlations used in Pre-Test Predictions

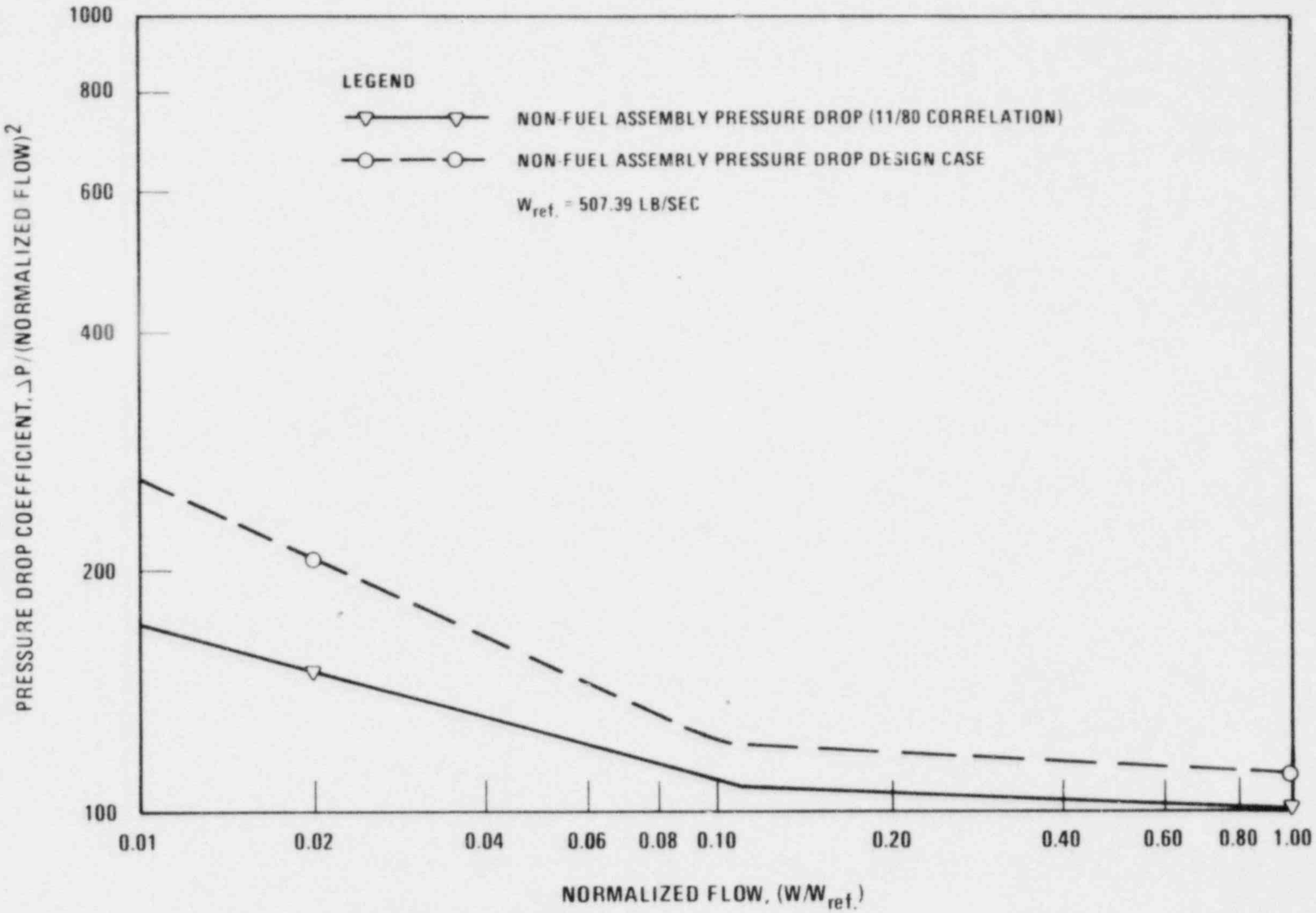

Figure 7. Non-Fuel Assembly Dynamic Pressure Drop Correlations used in Pre-Test Predictions

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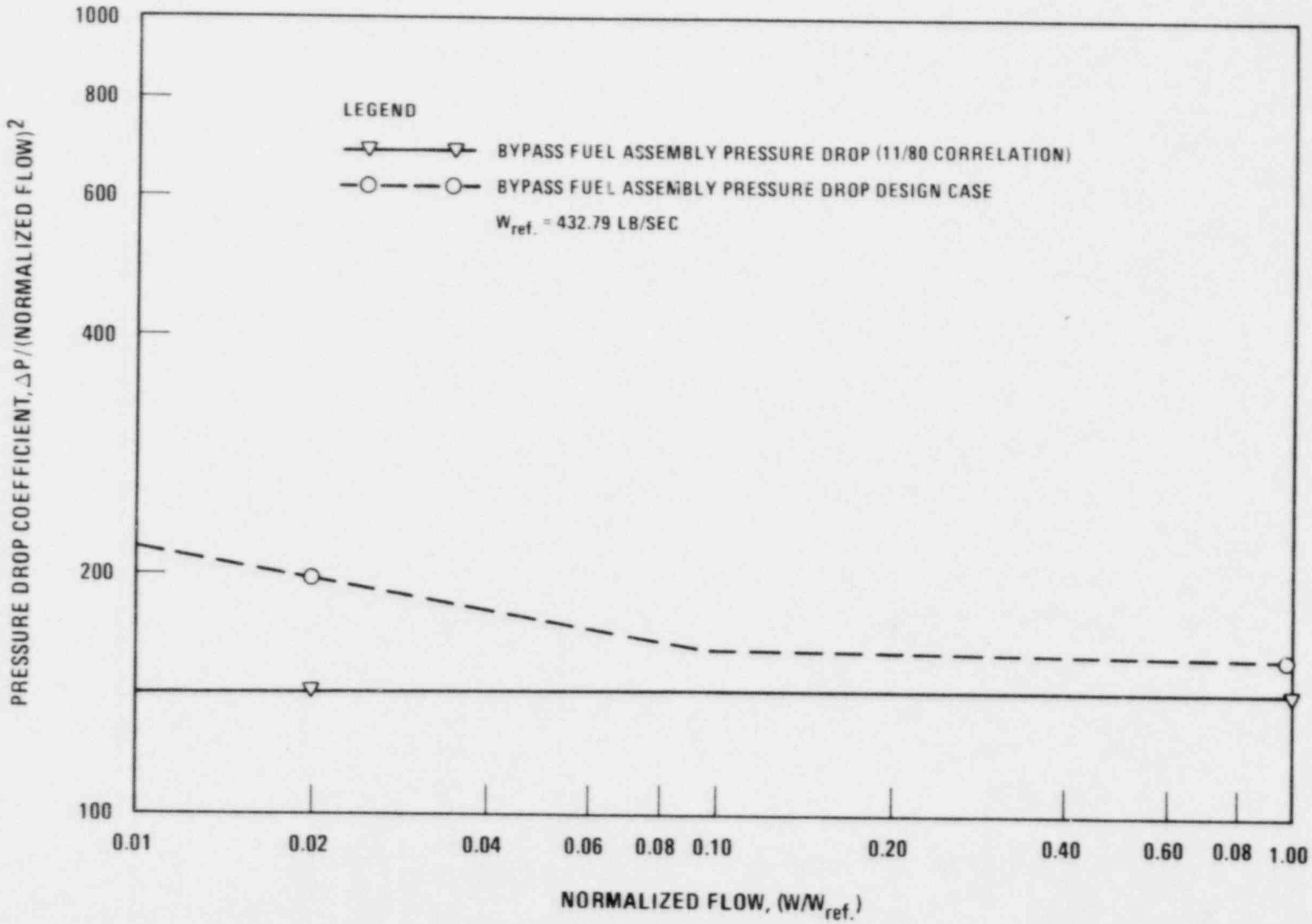

Figure 8. By-Pass Dynamic Pressure Drop Correlations used in Pre-Test Predictions

TABLE 3

DECAY POWER USED IN PRE-TEST PREDICTIONS, MW[12,14]

TIME	FUEL ASSEMBLY			NON-FUEL ASSEMBLY		
	100% Power	75% Power	35% Power	100% Power	75% Power	35% Power
0.0	18.54	13.905	6.769	0.9204	0.6903	0.2829
1.0	17.30	12.975	6.769	0.8708	0.6531	0.2829
2.0	16.52	12.390	6.36	0.8401	0.6301	0.2671
3.0	15.97	11.978	6.111	0.8183	0.6137	0.2580
4.0	15.54	11.655	5.73	0.8010	0.6008	0.2486
5.0	15.18	11.385	5.66	0.7866	0.5900	0.2414
6.0	14.86	11.145	5.459	0.7743	0.5807	0.2360
7.0	14.59	10.943	5.40	0.7634	0.5726	0.2314
8.0	14.34	10.755	5.30	0.7536	0.5652	0.2274
9.0	14.12	10.590	5.22	0.7448	0.5586	0.2237
10.0	13.92	10.440	5.13	0.7368	0.5526	0.2206
20.0	12.51	9.383	4.60	0.6807	0.5105	0.2009
30.0	11.63	8.723	4.30	0.6457	0.4843	0.1891
40.0	10.99	8.243	4.09	0.6203	0.4652	0.1800
50.0	10.49	7.868	3.91	0.6003	0.4502	0.1743
60.0	10.08	7.560	3.77	0.5839	0.4379	0.1686
70.0	9.731	7.298	3.67	0.5699	0.4274	0.1637
80.0	9.430	7.073	3.55	0.5579	0.4184	0.1600
90.0	9.167	6.875	3.47	0.5474	0.4106	0.1566
100.0	8.934	6.701	3.40	0.5381	0.4036	0.1543
200.0	7.512	5.634	2.87	0.4808	0.3606	0.1323
300.0	6.786	5.090		0.4512	0.3384	
400.0	6.312	4.734		0.4316	0.3237	
500.0	5.955	4.466		0.4165	0.3124	
600.0	5.663	4.247		0.4041	0.3031	
700.0	5.411	4.058		0.3932	0.2949	
800.0	5.189	3.892		0.3834	0.2876	
900.0	4.989	3.742		0.3746	0.2810	
1000.0	4.809	3.607		0.3665	0.2749	
2000.0	3.611	2.708		0.3092	0.2319	
3000.0	2.955	2.216		0.2729	0.2047	
4000.0	2.536	1.902		0.2462	0.1847	

TABLE 4

INITIAL CONDITIONS

Parameter	35%/75% Pre-Test Prediction	35%/75% Test**	75%/75% Pre-Test Prediction	75%/75% Test**	100%/100% Pre-Test Prediction	100%/100% Test**
1. Reactor						
A. Power (MWT)	140.0	139.3	300.0	286.4	400.0	399.1
B. Reactor Flow (KGPM)	10.084	9.916	10.085	9.475	13.444	12.034
C. Power History (Hours)	56	8*	25	7*	25	56*
D. Core Inlet Temp (°F)	631.2	629.1	664.0	660.3	679.7	679.0
2. Primary Loop						
A. Hot Leg Temp (°F)	750.4	753.6	922.4	911.3	936.8	938.0
B. Cold Leg Temp (°F)	631.0	629.1	663.3	660.3	679.8	679.0
C. Pump Speed (RPM)	756.0	711.6	762.5	746.4	1009.6	1008.0
3. Secondary Loop						
A. Hot Leg Temp (°F)	722.8	725.5	862.9	855.6	858.8	863.7
B. Cold Leg Temp (°F)	602.0	601.2	602.0	607.5	602.0	604.4
C. Flow (KGPM)	9.9	9.6	9.9	9.8	13.2	13.2
D. Pump Speed (RPM)	661.4	638.4	660.3	639.6	869.3	841.2

* Power history times are times ~ [illegible]% of power at scram.

** Test data represent arithmatic average of all three loops.

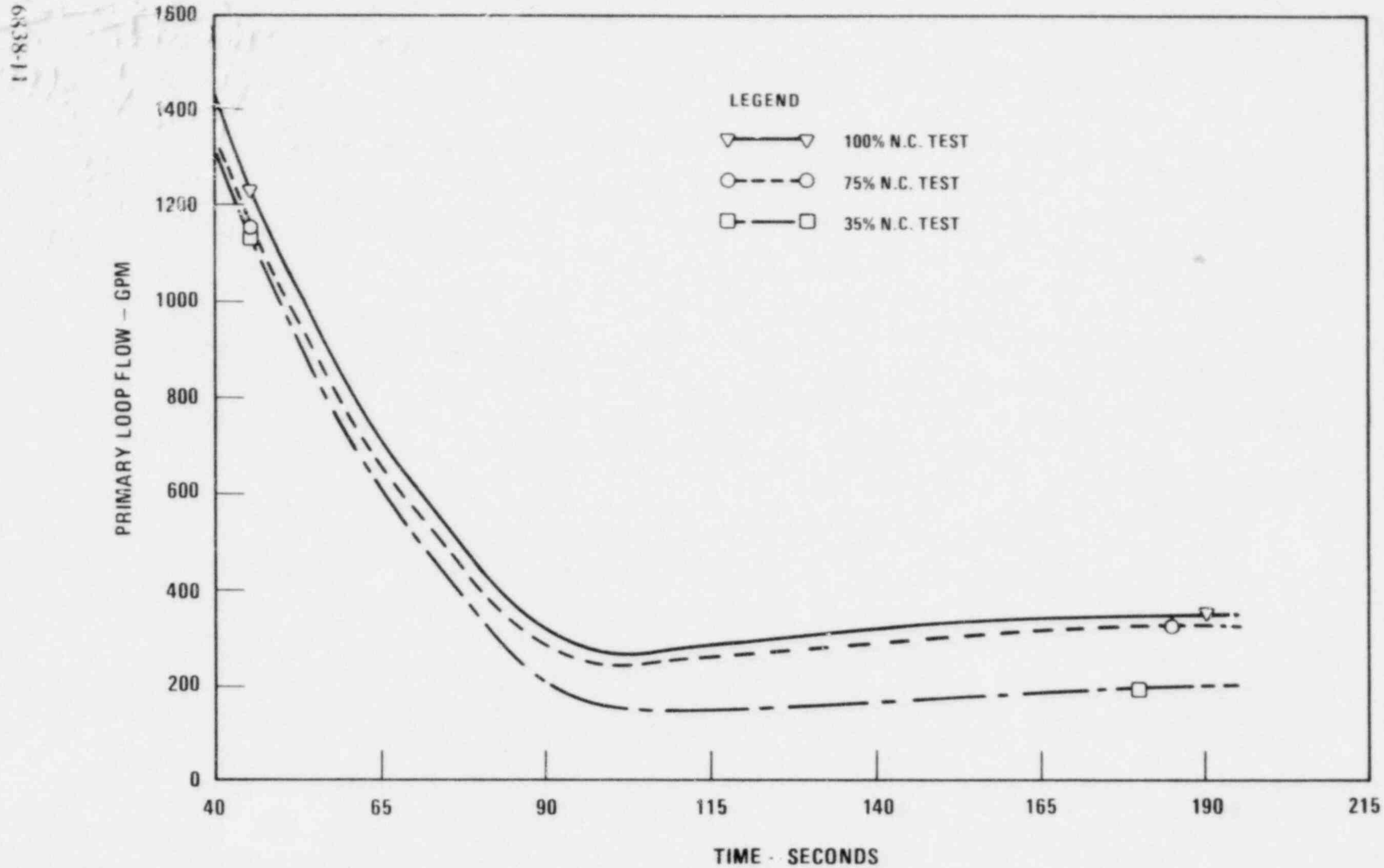

Figure 9. Pre-Test Predictions – Primary Loop Flow as Function of Time for Three Natural Circulation Tests

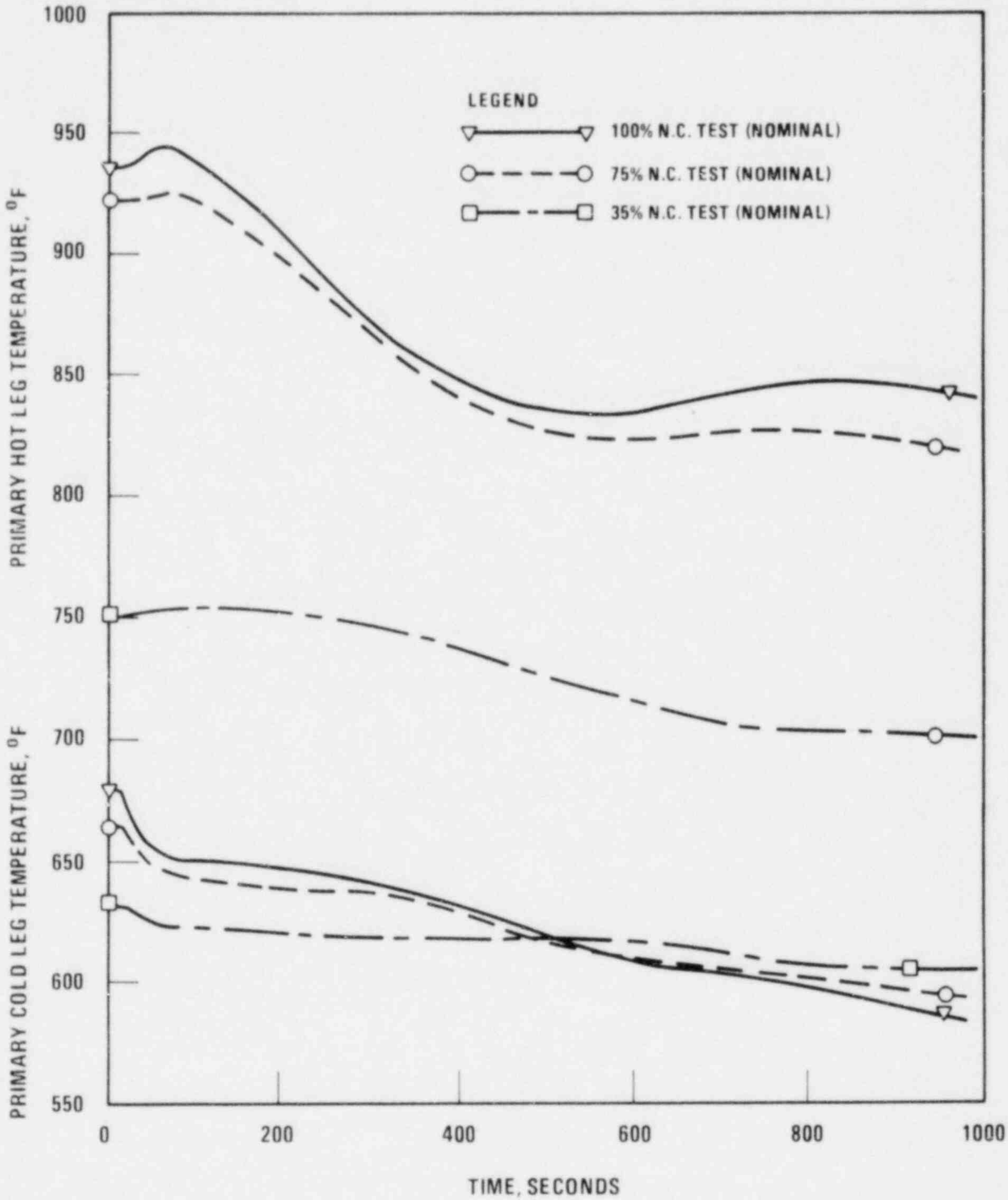

Figure 10. Pre-Test Predictions of Primary Hot/Cold Leg Temperature for Three Natural Circulation Tests

6838-40

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4.2 COBRA-WC PRE-TEST PREDICTION RESULTS

COBRA-WC pre-test predictions of the thermal-hydraulic response of the FFTF core, in particular the instrumented assemblies were made for the first 220 seconds of the natural circulation tests.[8] The nozzle-to-nozzle pressure drop boundary condition obtained from DEMO was used to calculate the 1-D (one channel per assembly) model as described in Section 3. Also provided by DEMO were fission power and vessel inlet and exit coolant temperature as functions of time during the transient.

Major output from the COBRA-WC code consisted of transient flow and coolant temperature information in all assemblies and the peak channel (cluster of subchannels as shown in Figure 5) within specially instrumented assemblies (FOTA's). Such information was used as input data for the selection of peak rod/channel for the single-channel FORE-2M analysis. As stated in Section 3, the inter- and intra-assembly effects of buoyancy and conductive heat transfer are important in reducing the magnitude of temperature gradients. The inter- and intra-assembly flow redistribution can be shown by plotting the ratio of axial flow fraction at node i, and time τ, relative to the initial flow fraction (τ=0), with time. (See Figures 4-14 and 5-11 in Appendix B.)

The effect of inter- and intra-assembly heat transfer is represented by the $Q_{ex}(x,\tau)$ factor, also an output of the COBRA-WC code. It consisted of heat input or loss due to radial temperature gradient (conduction effect) and to cross flow mixing (transport effect) (see Equation 3-1). Figures 4-16 and 5-13 in Appendix B show the variation of this quantity with time for the cases of transient test initiated from 100% and 75% power conditions, respectively.

The COBRA-WC predicted FOTA temperature distributions during the natural circulation tests are reported in Reference 8.

4.3 FORE-2M PRE-TEST PREDICTION RESULTS

Figures 12-17 show results for the Row 2 and Row 6 FOTA's, respectively, as calculated with FORE-2M code using input from the COBRA-WC code. Figure 12 shows sodium temperature predictions at the top of the fuel section for the Row 2 FOTA (T/C location B in Figure 5, or T/C No. TX1016) for the transient initiated from 35% power level. Figure 13 shows similar predictions for the Row 6 FOTA (T/C location C or T/C No. TX9018). Figures 14 and 15 show sodium temperature predictions for Row 2 and Row 6 FOTA, respectively, for the transients initiated from 75% power level. Figures 16 and 17 shown sodium temperature predictions for Row 2 and Row 6 FOTA, respectively, for the transients initiated from 100% power level. Shown in these figures are the nominal (best estimate) case as well as the estimated sodium temperatures with uncertainties, UNC case, as listed in Table 1. The actual temperatures at these subchannels would be expected to correspond with the curve representing the nominal case, and are also shown in the figure. As stated in Reference 14, these actual test data should be less than the T_{UNC} curve representing (3σ) uncertainty case. Thermocouple measui ement uncertainties and time delays were not included in the predictions. The acceptance criterion for sodium temperatures predictions was stated as follows: if the maximum measured temperature at specified channels (corrected for above instrument effects and actual boundary conditions) is less than the curve shown with uncertainties, T_{UNC}, it would follow that DEMO/COBRA-WC/FORE-2M calculations made with design data conservatively envelope the temperature response of the natural circulation test. The model and methodology would therefore be acceptable for such design transient predictions.

5.0 CALCULATION OF DECAY POWER FROM ACTUAL POWER HISTORY

For the post-test analysis, the actual power hist... ies prior to each test were used in the calculation of decay power. These pre-scram power operating histories are shown in Figures 18, 19, and 20. The decay heat power predictions for fuel assemblies were made independently, using CRBRP methodology for the post-test analysis (Appendix C). This calculation was based on the S4M computer program, using ENDF/B-IV decay energy release rate data.[18] Input reactor rate data was generated from the beginning of

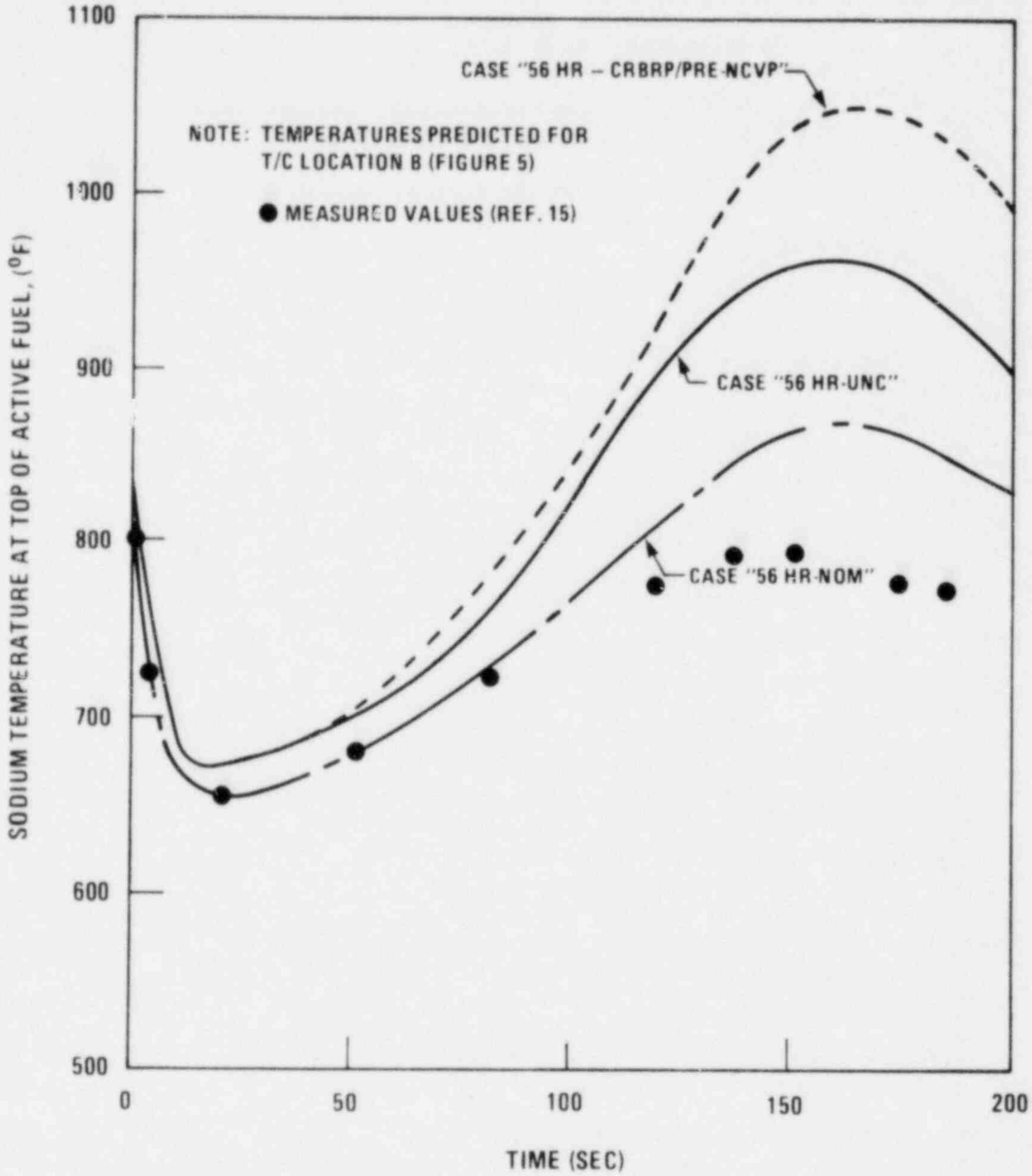

Figure 12. Pretest Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 15) for Row 2 FOTA with 56 Hour Irradiation History (Test Initiated from 35% Power/75% Flow) (TX1016)

6838-2

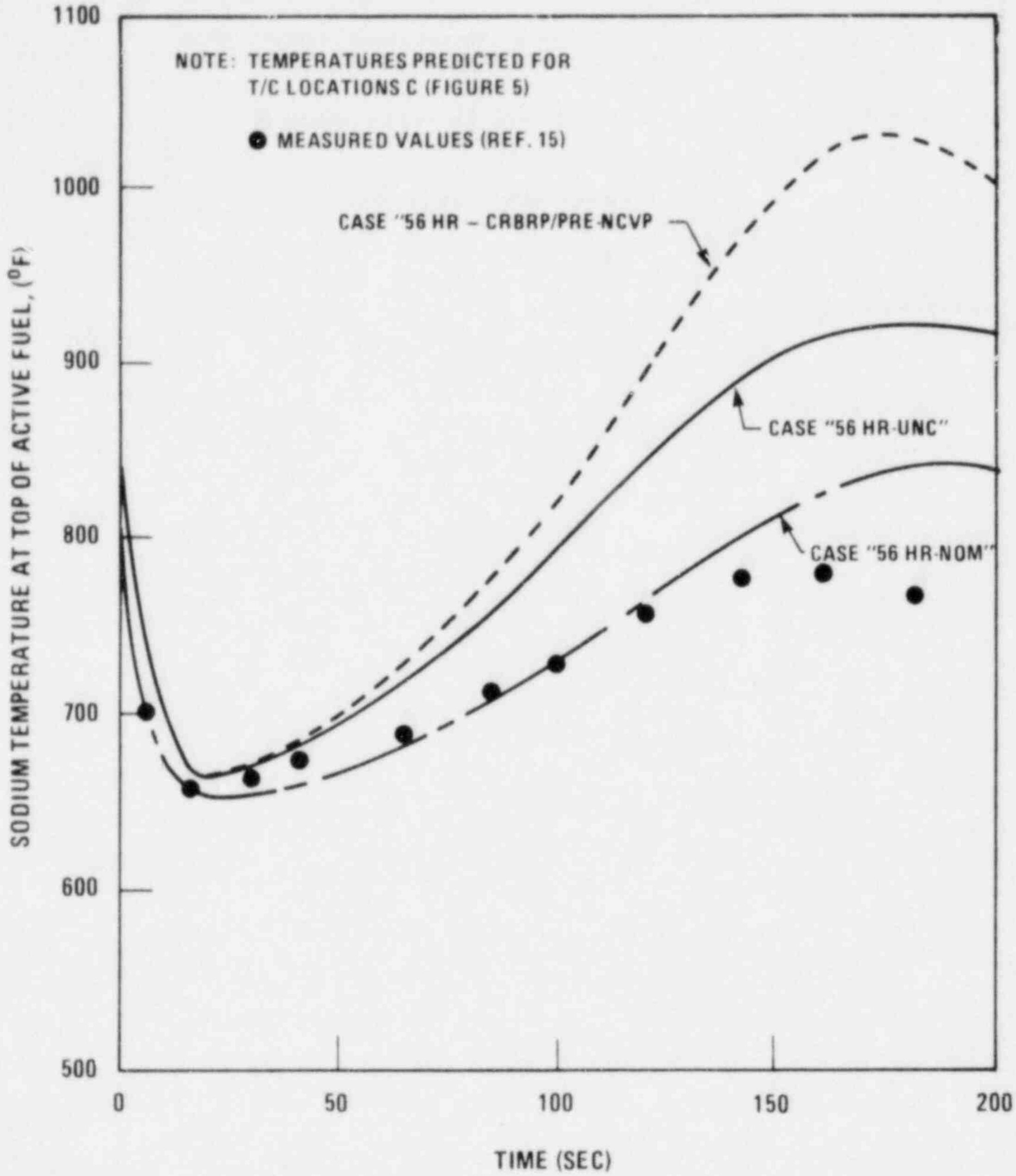

Figure 13. Pretest Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 8) for Row 6 FOTA with 56 Hour Irradiation History (Test Initiated from 35% Power/75% Flow) (TX9018)

6838-1

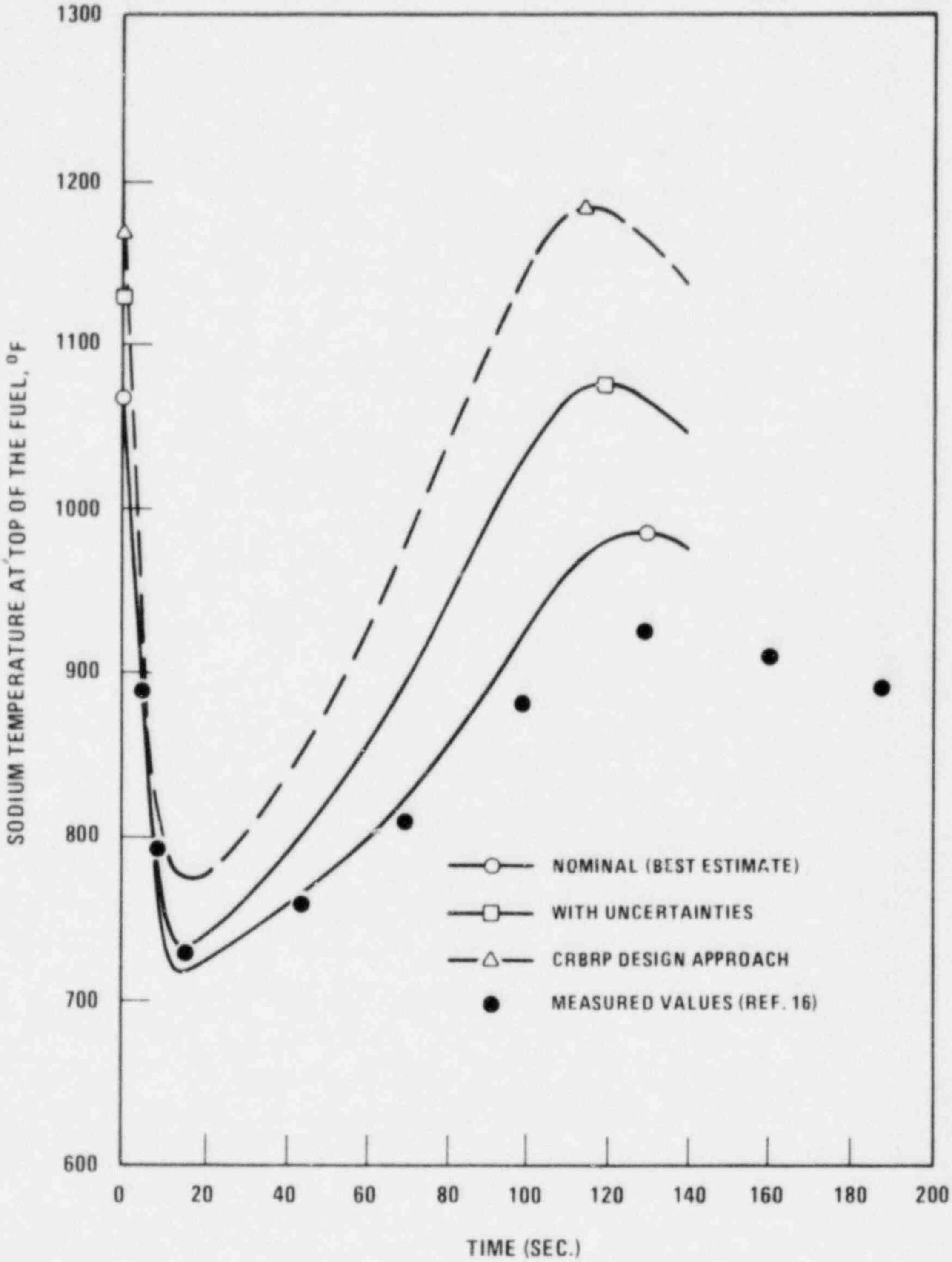

Figure 14. Pretest Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 15) for Row 2 FOTA Initiated at 75% Power/ 75% Flow Conditions (TX1016)

6838-6

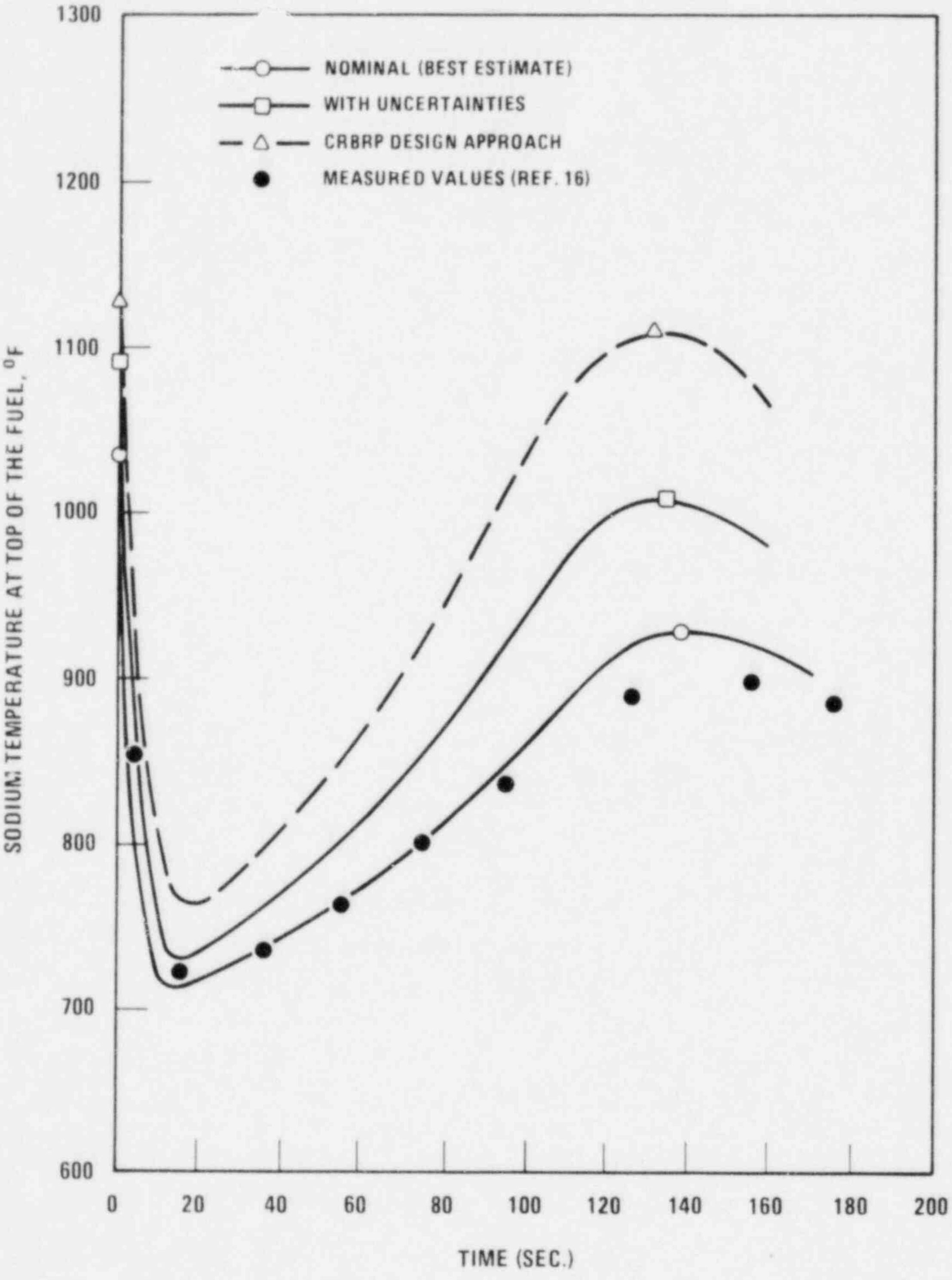

Figure 15. Pretest Sodium Temperature Predictions at Top of the Fuel (Hot Rod Channel 8) for Row 6 FOTA Initiated from 75% Power/75% Flow Conditions (TX9018)

6838-7

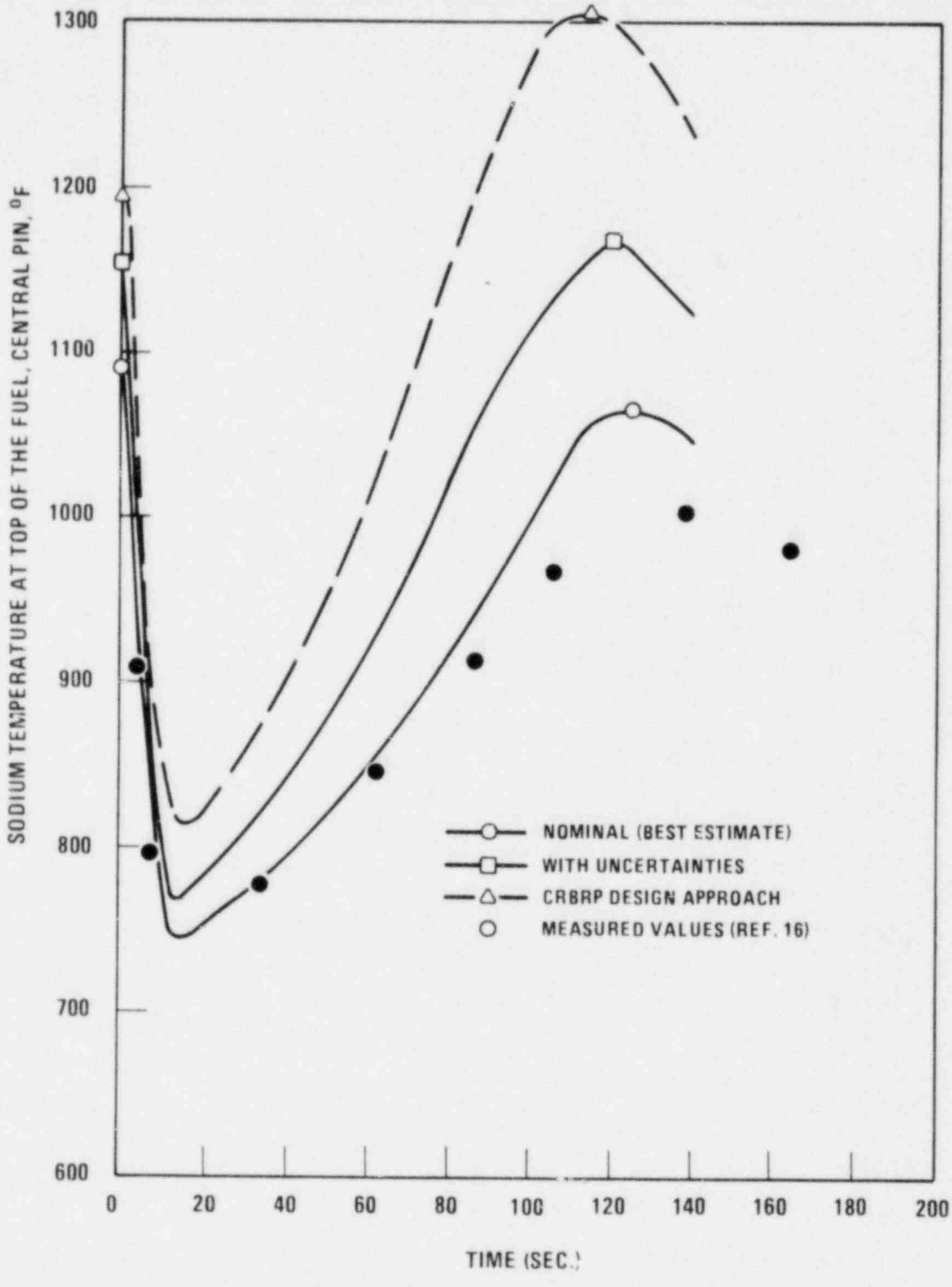

Figure 16. Pretest Sodium Temperature Predictions at Top of the Fuel (Hot Rod in Channel 15) Row 2 FOTA Initiated at 100% Power/100% Flow Conditions (TX1016)

6838-8

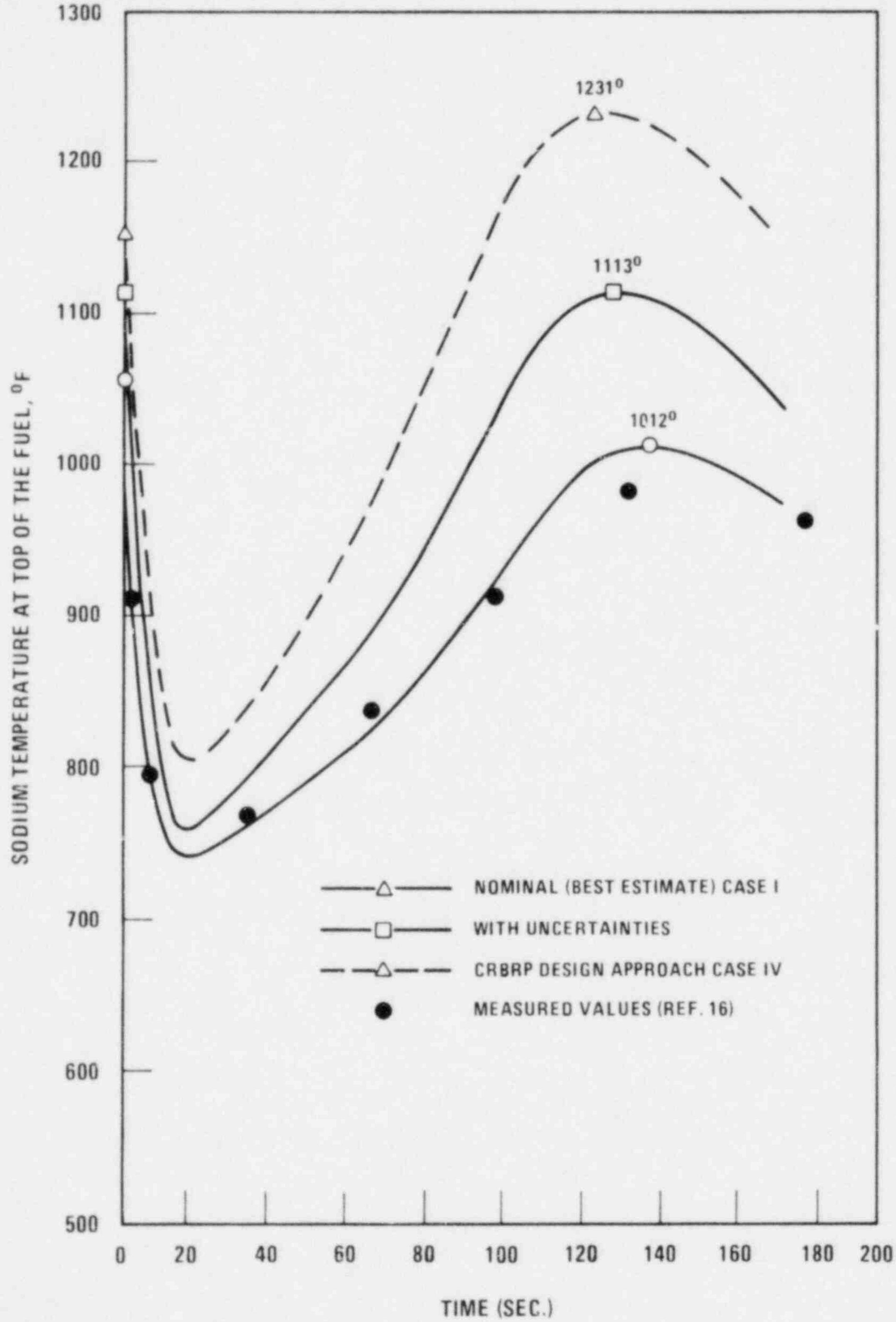

Figure 17. Pretest Sodium Temperature Predictions at Top of the Fuel (Hot Rod in Channel 14) for Row 6 FOTA Initiated at 100% Power/100% Flow Conditions (TX9018)

6838-30

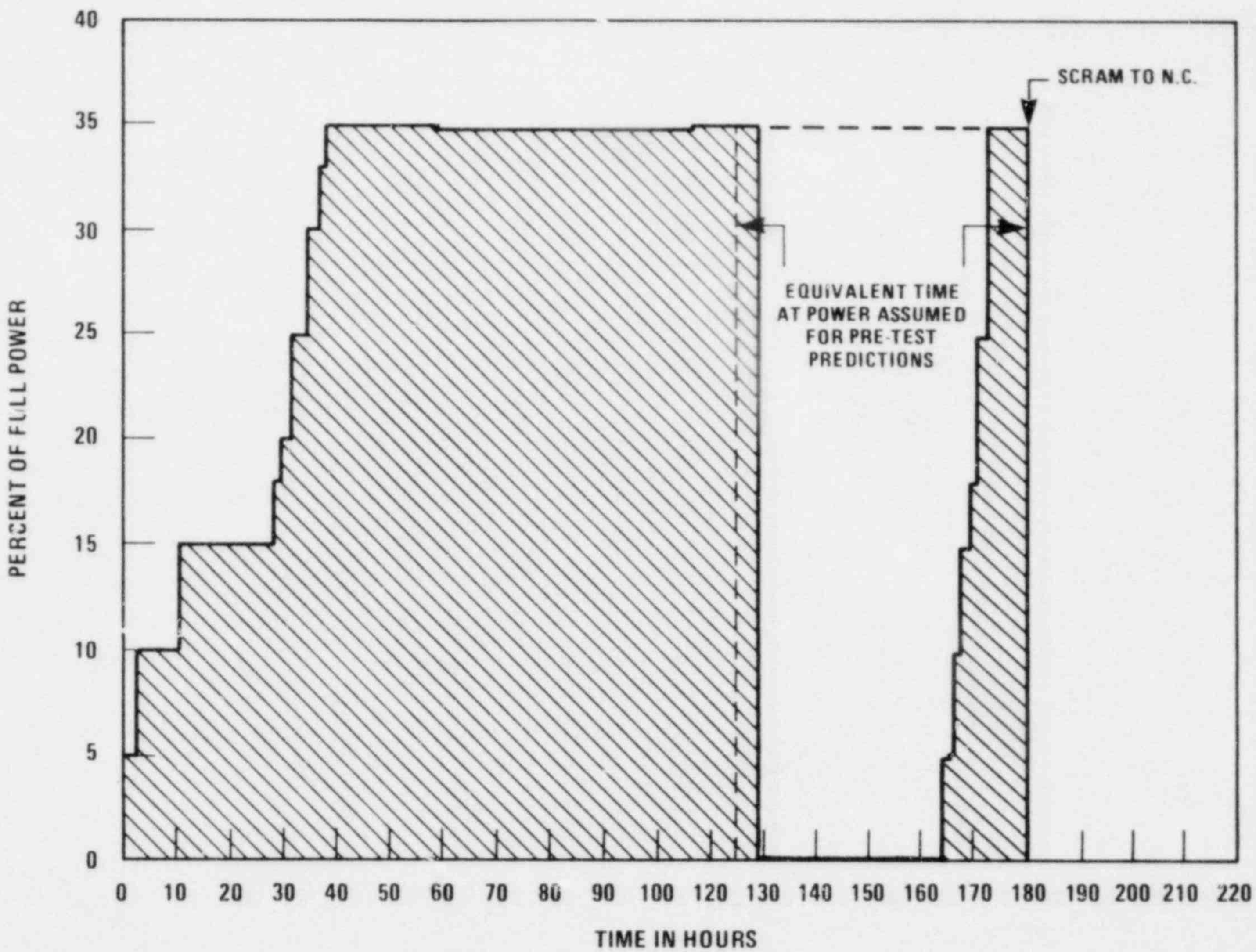

Figure 18. FTR Operating History Prior to N.C. Test from 35% Power Condition

6838-31

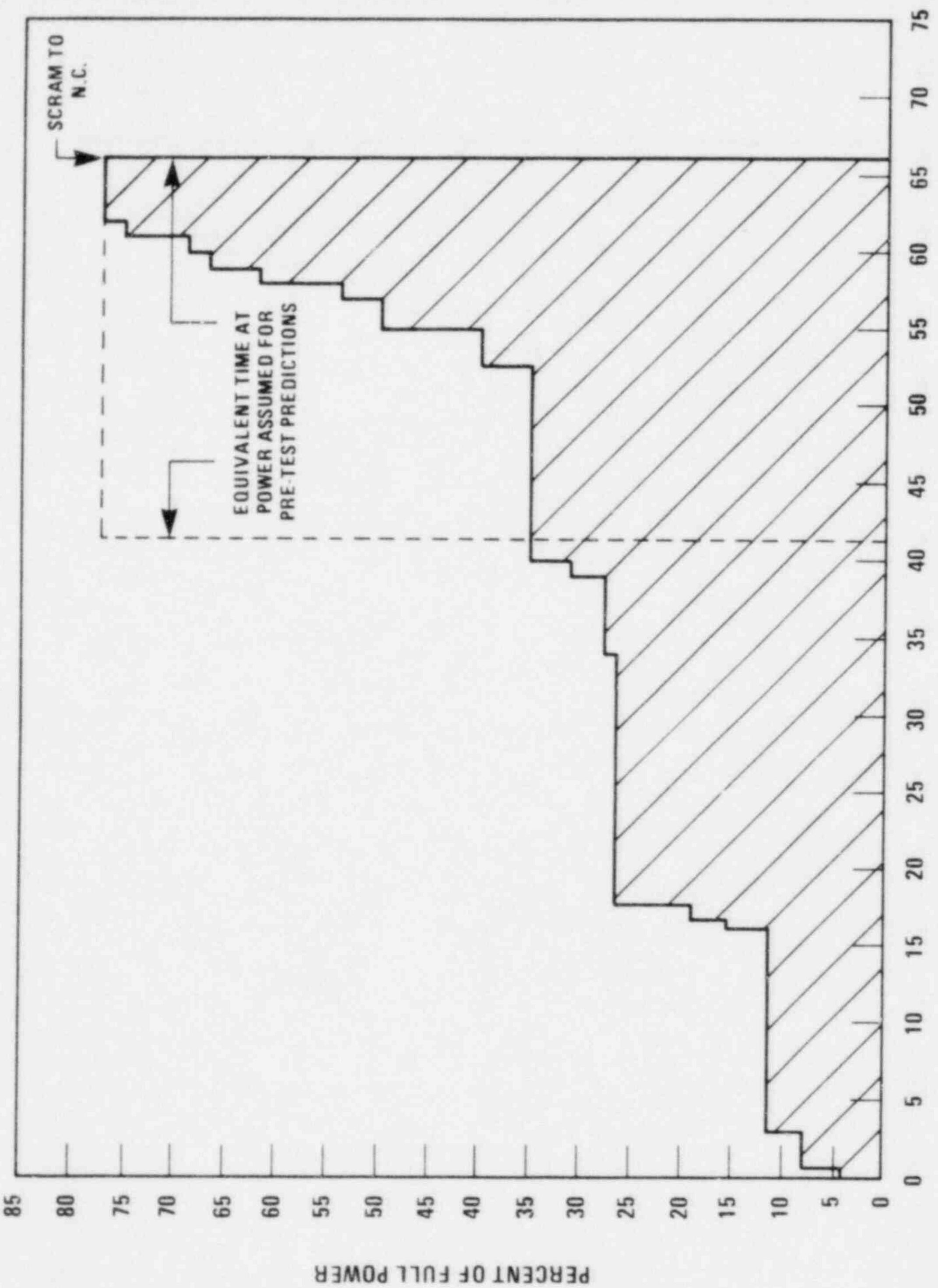

Figure 19. FTR Operating History Prior to N.C. Test from 75% Power Condition

6838-32

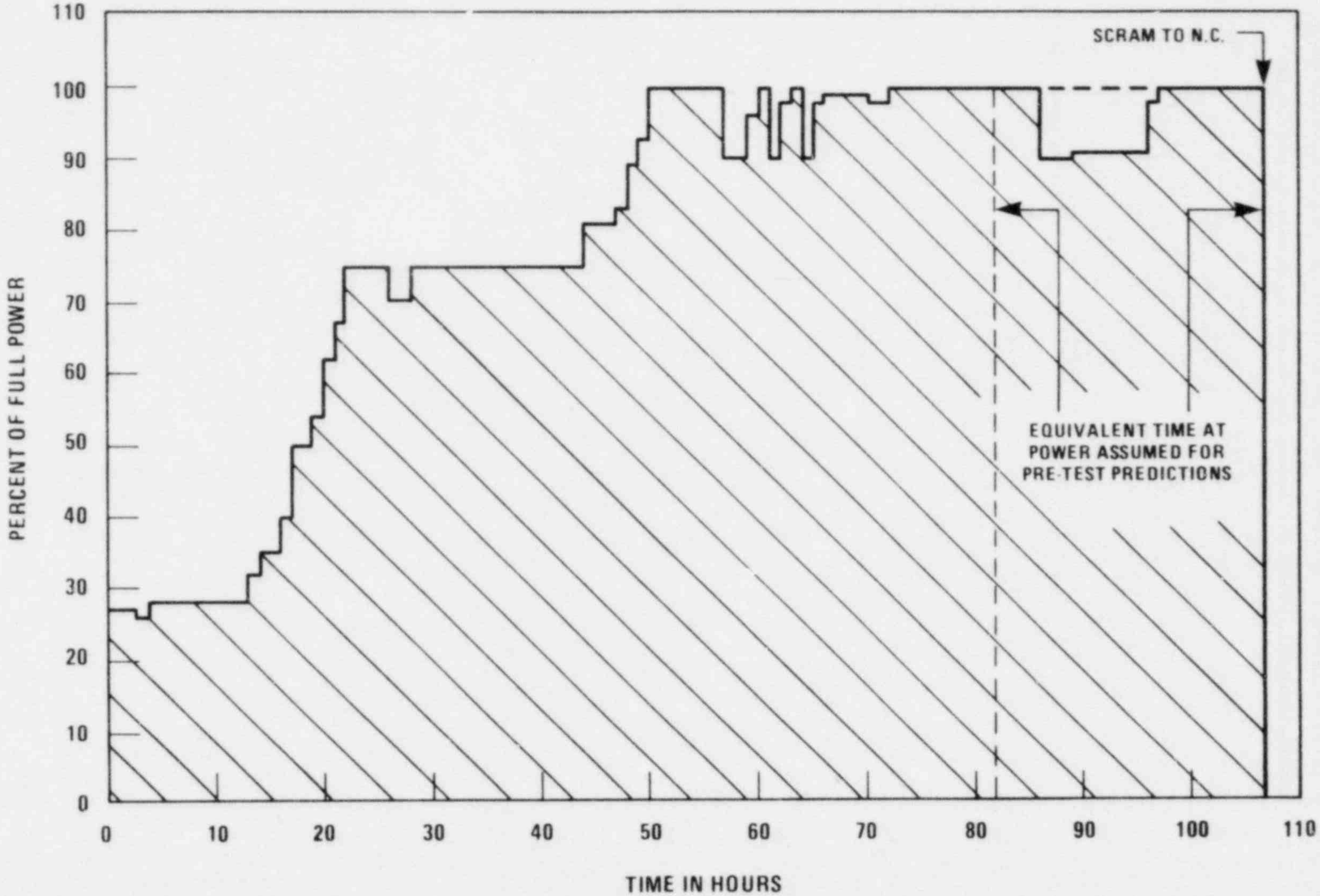

Figure 20. FTR Operating History Prior to N.C. Test from 100% Power Condition

6838-33

equilibrium cycle fission rates provided in Reference 9. Assuming reactor rates vary linearly with the reactor power, the reaction rate versus time model followed the power-time histories provided via Figures 18-20. Decay heat values for fuel assemblies were calculated at small time intervals after shutdown up to 1000 seconds after shutdown (values up to 300 seconds are shown in Table 5). Figure 21 shows the calculated actual decay power in the fuel assemblies for various initial power conditions for the natural circulation tests. Also shown in this figure are the fuel assembly decay power calculations made by the COBRA-WC code which retains the pre-test decay power calculation approach using equivalent time of pre-scram irradiation of power. The two results are seen to be very close to each other. Thus the pre-test calculation approach was confirmed as long as an equivalent power history (pre-scram irradiation) was used which matched the decay power curve closely.

6.0 REACTOR FLOW MATCH

As mentioned in Section 3, the reactor vessel pressure drop correlations used in DEMO were adjusted to have a closer match on the reactor flow. This was done by modifying the reactor pressure drop correlations at low flow range based on calibration calculations done by COBRA-WC as shown in Figures 22, 23 and 24. Using these modified correlations, better agreement in the reactor flow between these two codes was obtained. This comparison is shown in Figures 25, 26 and 27 for pre-scram power levels of 35%, 75% and 100%, respectively. It can be seen that the maximum deviation in the reactor flow during the transient of interest are within 7%, which is less than the scattering of the test data for the primary loop flow (also shown in these figures).[19] The maximum deviation before the iteration was ∿12%. The agreement between the COBRA-WC and DEMO codes is considered very satisfactory.

TABLE 5

FUEL ASSEMBLY DECAY POWER FOR THREE NATURAL CIRUCLATION TESTS (MW)

Time	Pre-Scram Power Level		
	100%	75%	35%
0.0	19.01	13.33	6.536
1.0	17.78	12.42	6.103
2.0	17.05	11.87	5.843
3.0	16.51	11.47	5.654
4.0	16.09	11.16	5.505
5.0	15.74	10.89	5.381
6.0	15.44	10.67	5.273
7.0	15.17	10.47	5.179
8.0	14.93	10.29	5.095
9.0	14.72	10.13	5.108
10.0	14.52	9.980	4.948
12.0	14.17	9.717	4.824
20.0	13.13	8.939	4.455
24.0	12.74	8.650	4.318
30.0	12.26	8.290	4.147
40.0	11.62	7.818	3.923
42.0	11.52	7.737	3.884
50.0	11.13	7.447	3.747
60.0	10.72	7.143	3.602
70.0	10.37	6.886	3.480
78.0	10.13	6.707	3.395
80.0	10.08	6.665	3.375
90.0	9.817	6.472	3.283
96.0	9.677	6.368	3.233
100.0	9.588	6.302	3.202
110.0	9.384	6.150	3.130
116.0	9.272	6.067	3.090
200.0	8.188	5.264	2.707
206.0	8.134	5.223	2.687
300.0	7.469	4.733	2.452

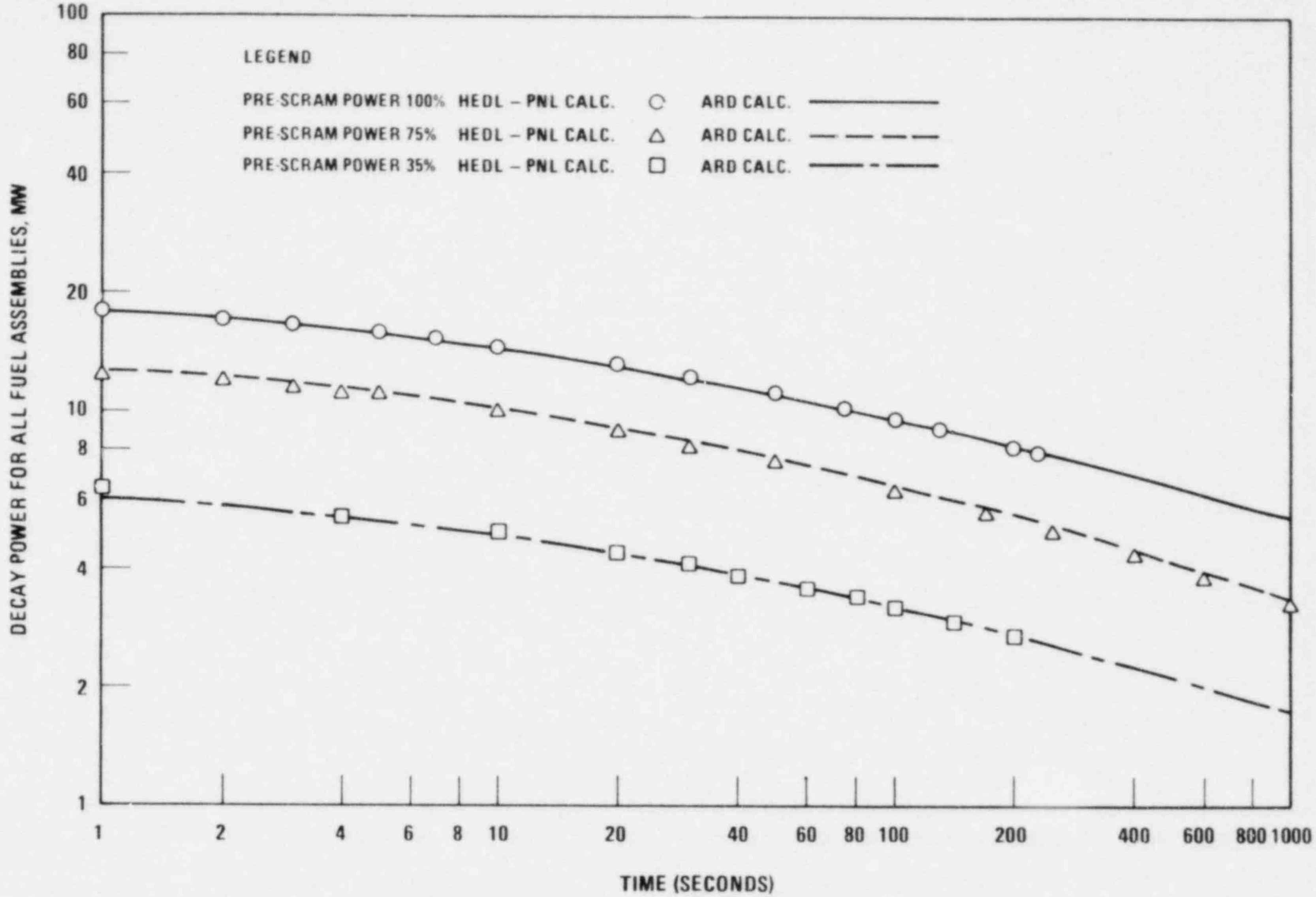

Figure 21. Post Shutdown Decay Heat for Fuel Assemblies (COBRA-WC Post Test Values Vs. ARD Calculations)

6838-34

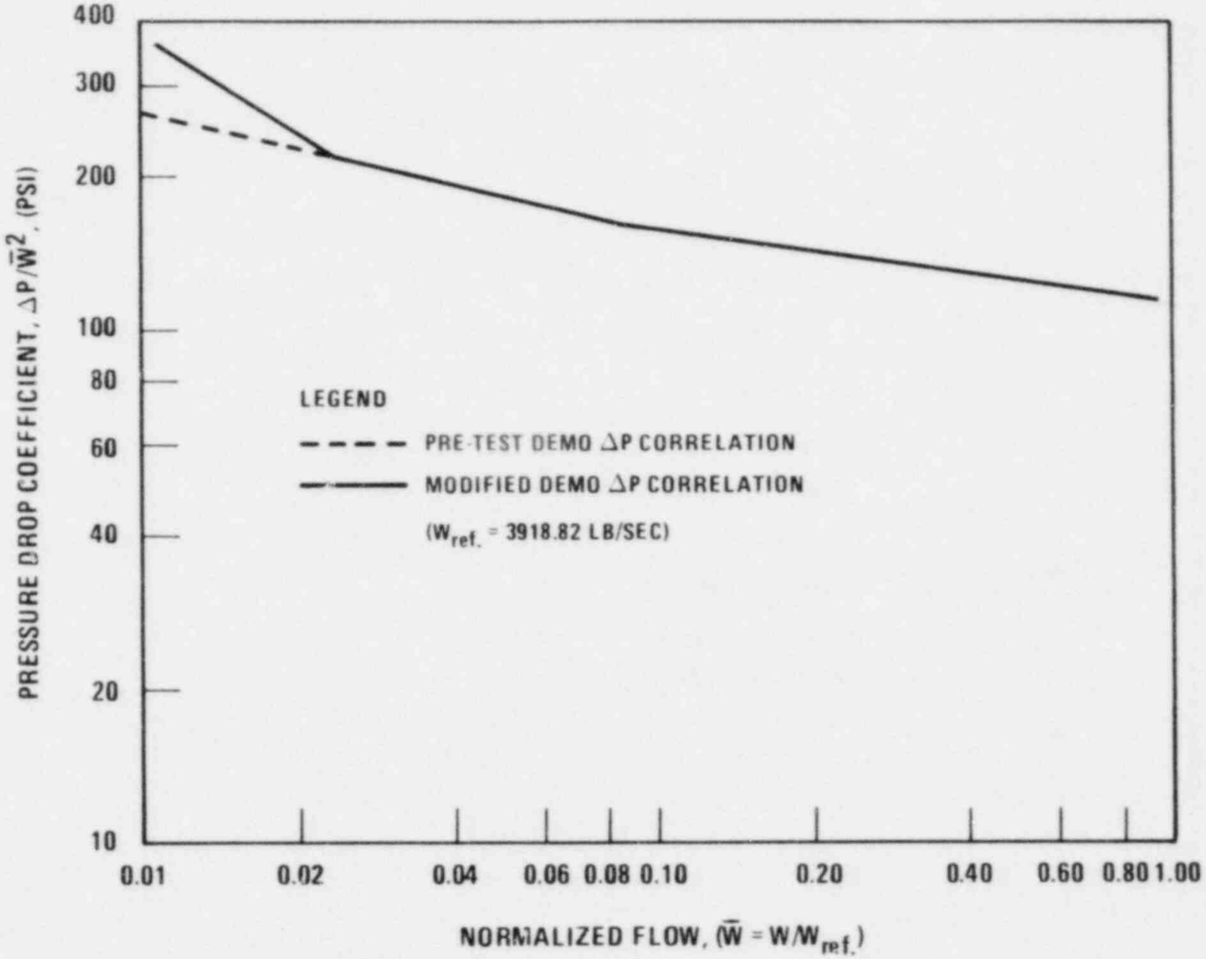

Figure 22. Comparison of the Reactor Dynamic Pressure Drop Correlations for the Fuel Assembly Group

6838-36

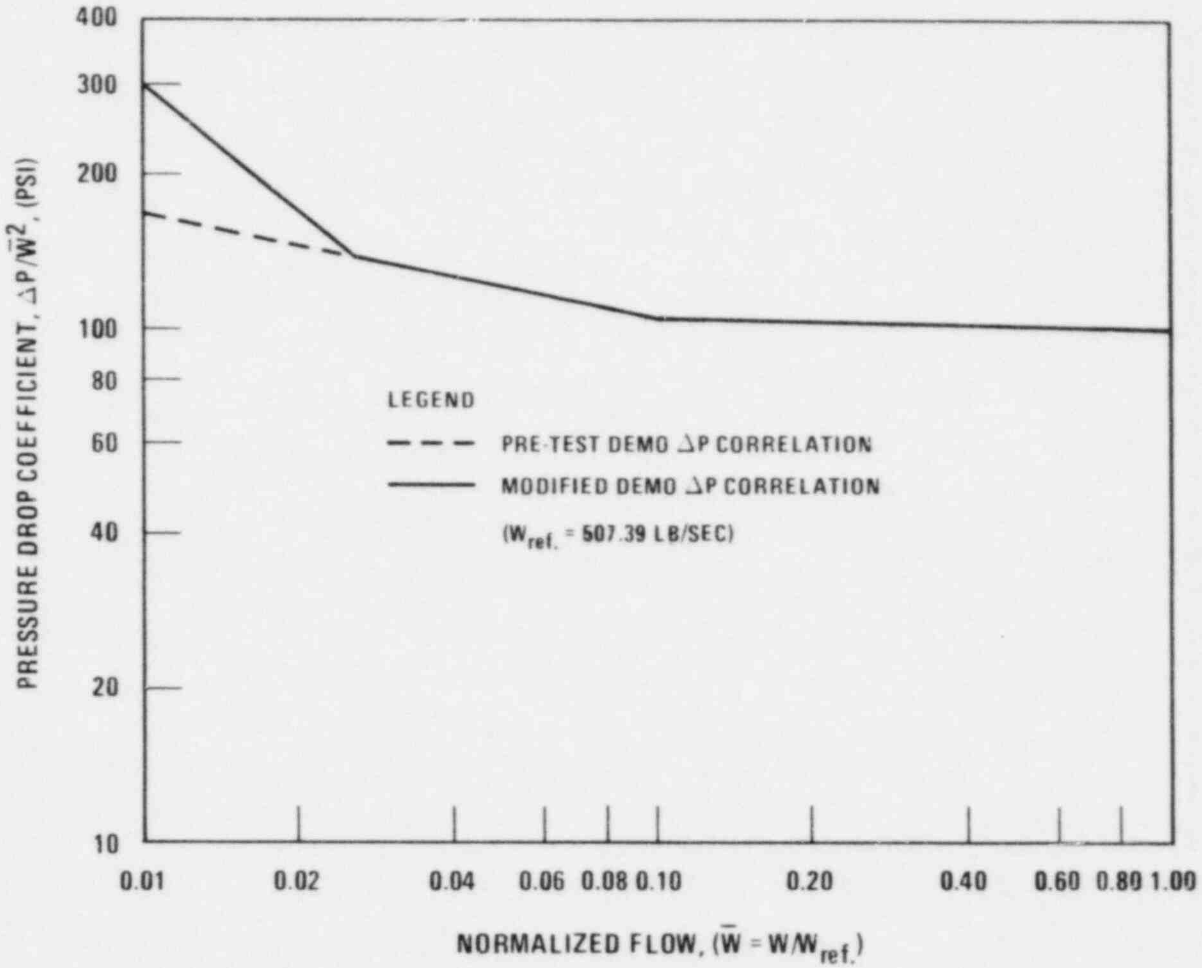

Figure 23. Comparison of the Reactor Pressure Drop Correlations for the Non-Fuel Assembly Group

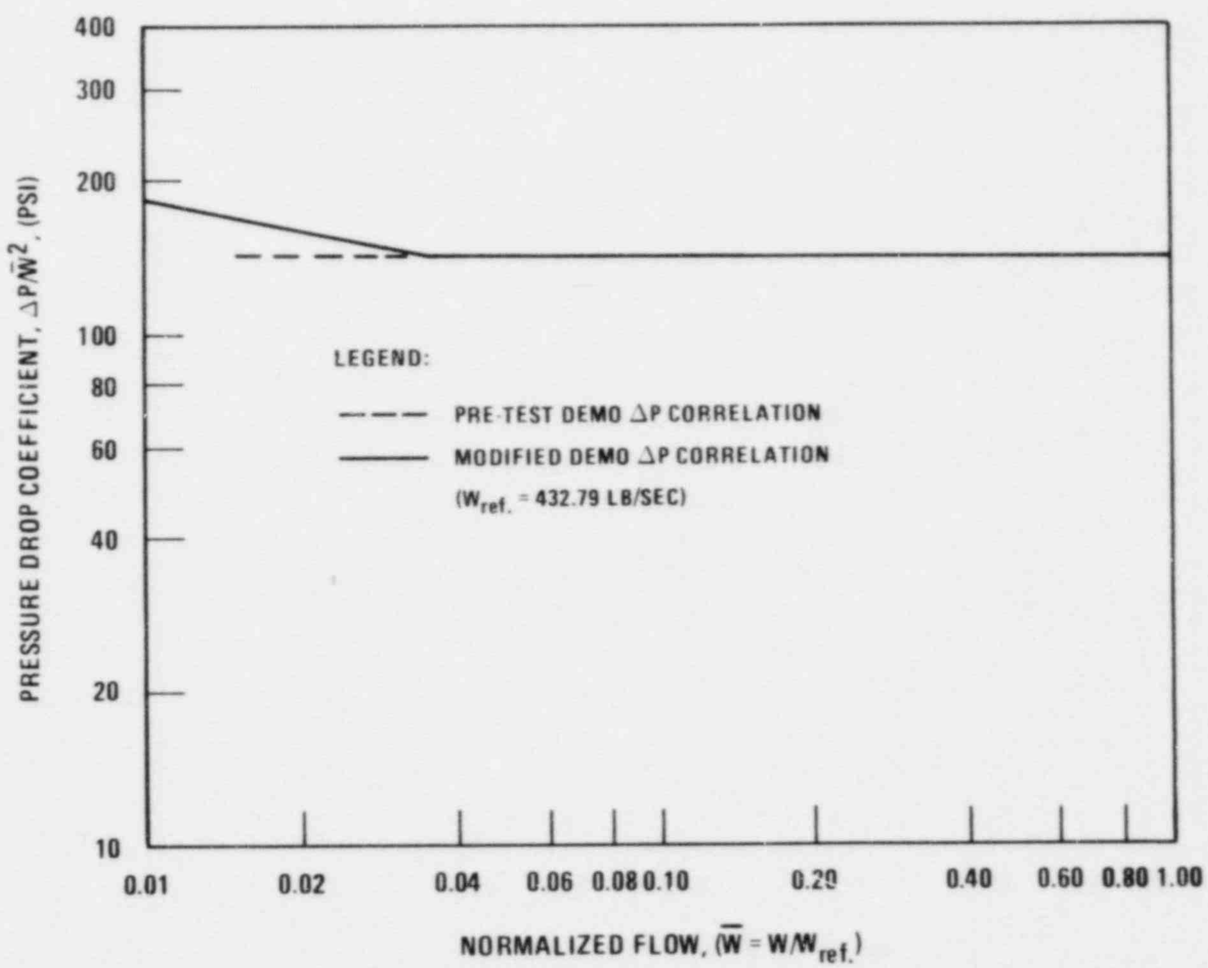

Figure 24. Comparison of the Reactor Pressure Drop Correlations for the Bypass

6838-37

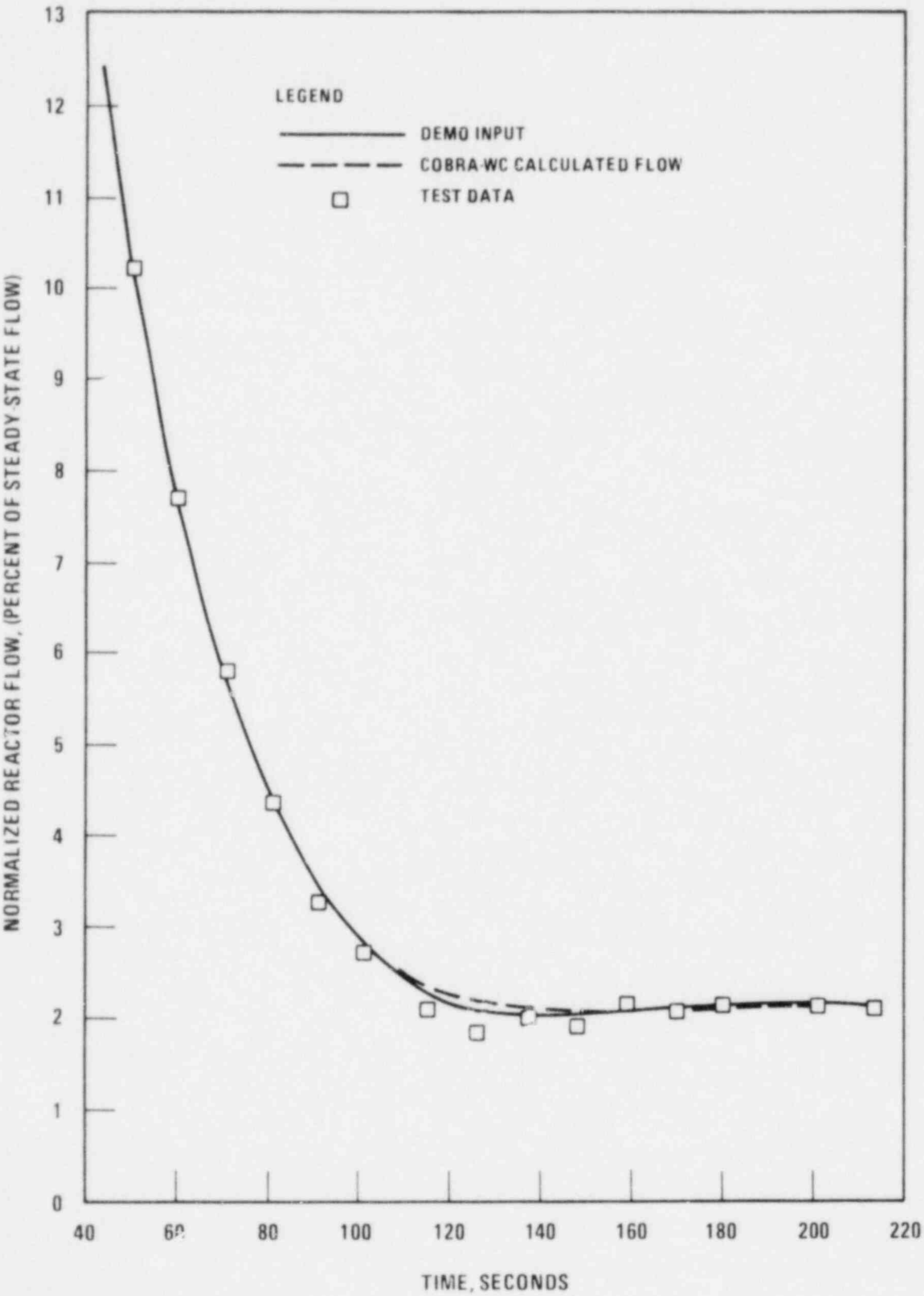

Figure 25 Reactor Flow Comparison (COBRA-WC Vs. DEMO) (Pre-Scram Power at 35% Level)

6838-38

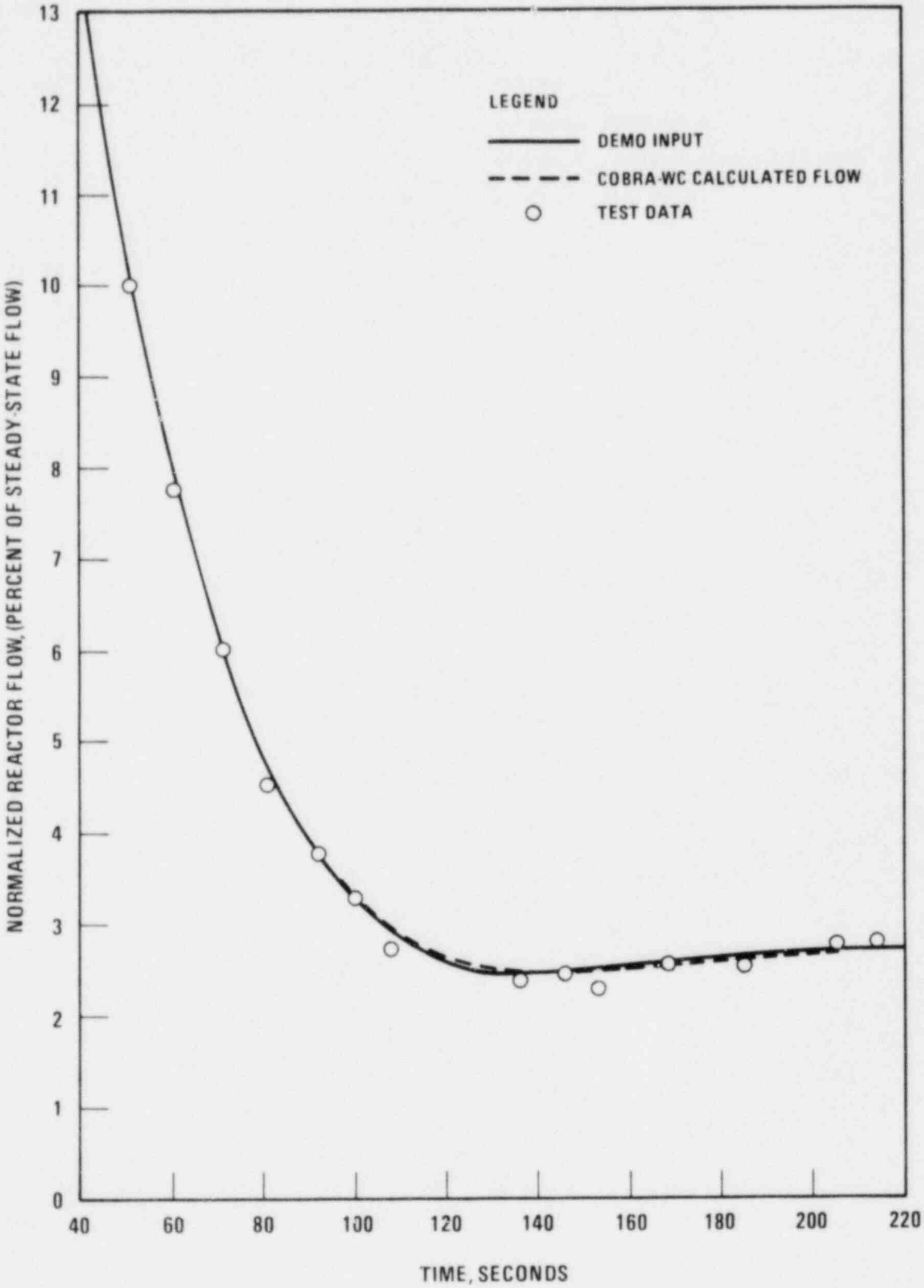

Figure 26. Reactor Flow Comparison (COBRA-WC Vs. DEMO) (Pre-Scram Power at 75% Level)

6838-14

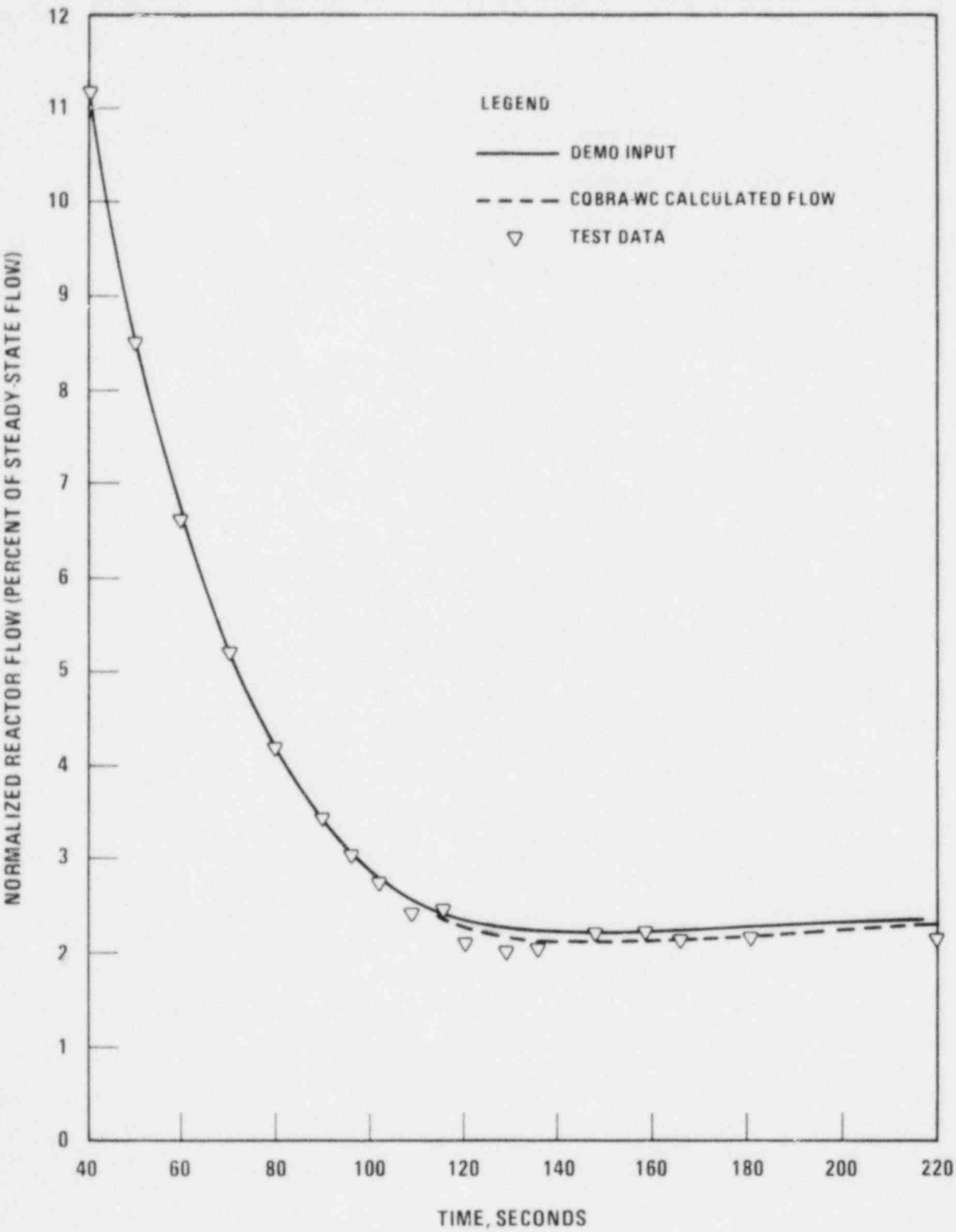

Figure 27. Reactor Flow Comparison (COBRA-WC Vs. DEMO)
(Pre-Scram Power at 100% Level)

7.0 POST-TEST PREDICTIONS AND COMPARISON WITH TEST DATA

Post-test predictions were made using the actual power history, and initial test conditions as shown in Table 4. The description of three natural circulation tests are listed in Table 6.

To verify the core modelling in COBRA-WC/FORE-2M codes for the natural circulation tests, the actual FFTF reactor flow was used in DEMO to calculate the average core ΔP which, along with the reactor inlet temperature and dynamic core delayed neutron powers, were used in the post-test analyses. These data were used by COBRA-WC in the same manner as in the pre-test predictions (Section 4.2) to yield the information concerning the inter- and intra-assembly phenomena which was in turn inputted to FORE-2M for the localized hot rod analyses. The sodium temperature at peak channels of the instrumented, open test fuel assemblies were thus calculated. Other initial test conditions as listed in Table 4, as well as the information related to the decay power (Section 5.0) were used in the COBRA-WC and FORE-2M codes. Thus, the post-test predictions were made based on nominal values. The results are shown in Figures 28-39.

Figure 28 shows the post-test sodium temperature nominal predictions at the top of the fuel section for the hot rod in Channel 15 (Location B Figure 5) of the Row 2 FOTA, initiated at 35% power/75% flow conditions. Also shown are the measured data from the thermocouple No. TX1016[15]. The agreement between the two curves is excellent, especially the peak temperatures (both at 800°F). Figures 29 and 30 show the temperatures of the same channel for initial conditions at 75% power and 100% power, respectively. The agreement is good with the predicted peak temperature being 10°F (or 3% of the temperature rise) higher (conservative) than measured in Figure 29 and 35°F (or 10% of the temperature rise) higher than measured in Figure 30. Note, in all these cases, the predicted temperatures at initial time (steady-state) are comparable with measured values to within 20°F also in the conservative direction. The predicted time at which peak temperatures occur is 15-20 seconds later than measured which is related to the slight mismatch of reactor flow transient between DEMO and COBRA-WC (Figures 25, 26 and 27). As a second check, the thermocouple readings of TX1018 are compared with corresponding

TABLE 6

TEST DESCRIPTION

Pre-Scram Power/Flow Conditions	35%/75%	75%/75%	100%/100%
1. Date and Time of Test	12/10/80 14:03:45	3/12/81 5:59:00	3/18/80 10:04:00
2. Time at Test Power* Prior to Scram	8 hours	7 hours	56 hours
3. Duration of Decay Heat Removal Under Natural Circulation	$\sim$ 1 hour	$\sim$ 2.2 hours	$\sim$ 36 hours
4. Pony Motor Operation	Seconday Loop 1 Pony Motor Remained Operational	None	None
5. Time of Peak Core Temperature (Row 2 FOTA)	$\sim$ 140 seconds	$\sim$ 135 seconds	$\sim$ 130 seconds
6. Time of Minimum Primary Flow	$\sim$ 120 seconds	$\sim$ 125 seconds	$\sim$ 130 seconds
7. Average Time for Primary Pumps to Stop	$\sim$ 120 seconds	$\sim$ 120 seconds	$\sim$ 135 seconds

* Power History Times are Times > 90% Test Power at Scram

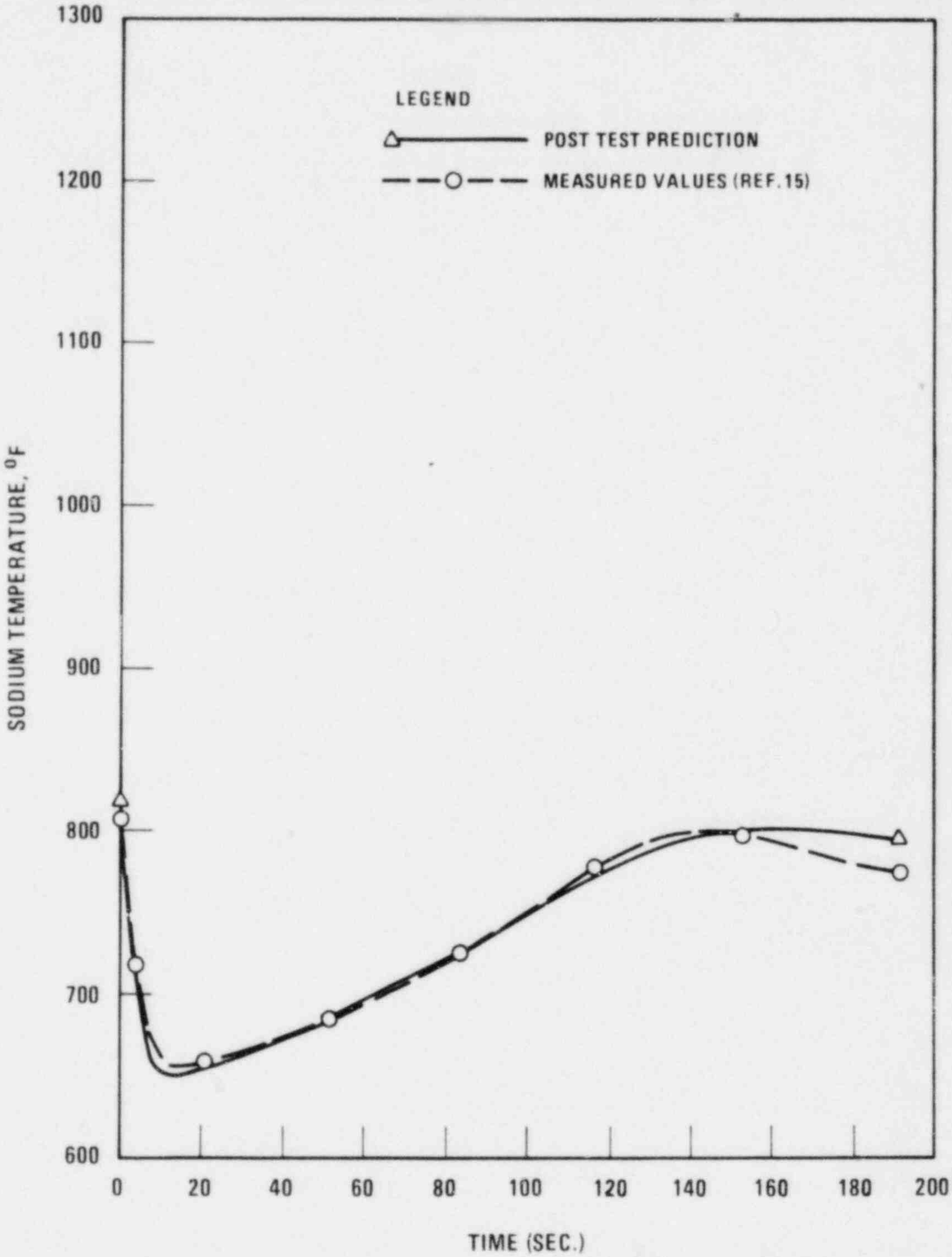

Figure 28. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 15) Row 2 FOTA Initiated at 35% Power/75% Flow Conditions (TX1016)

6838-16

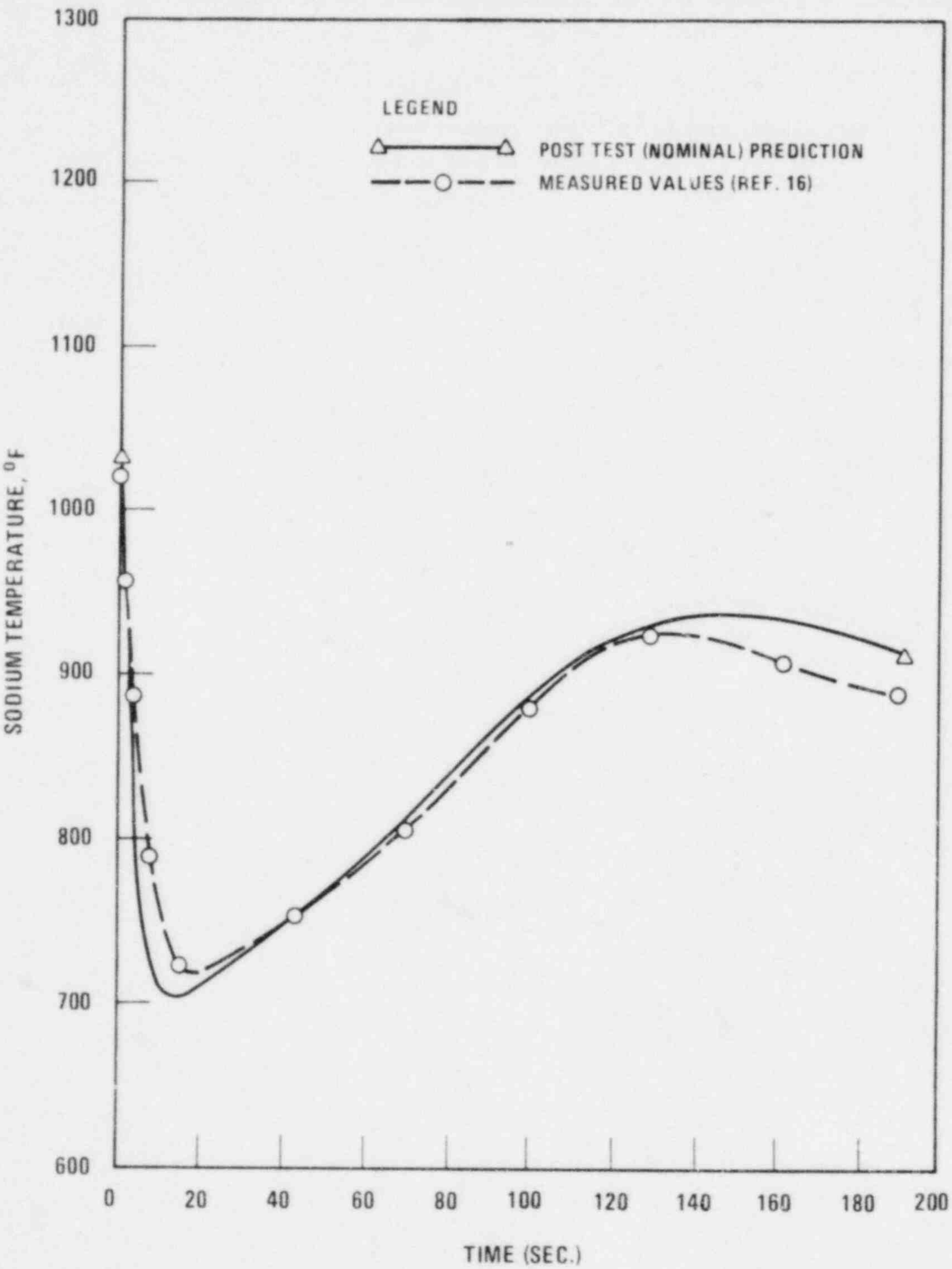

Figure 29. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 15) Row 2 FOTA Initiated at 75% Power/75% Flow Conditions (TX1016)

6838-17

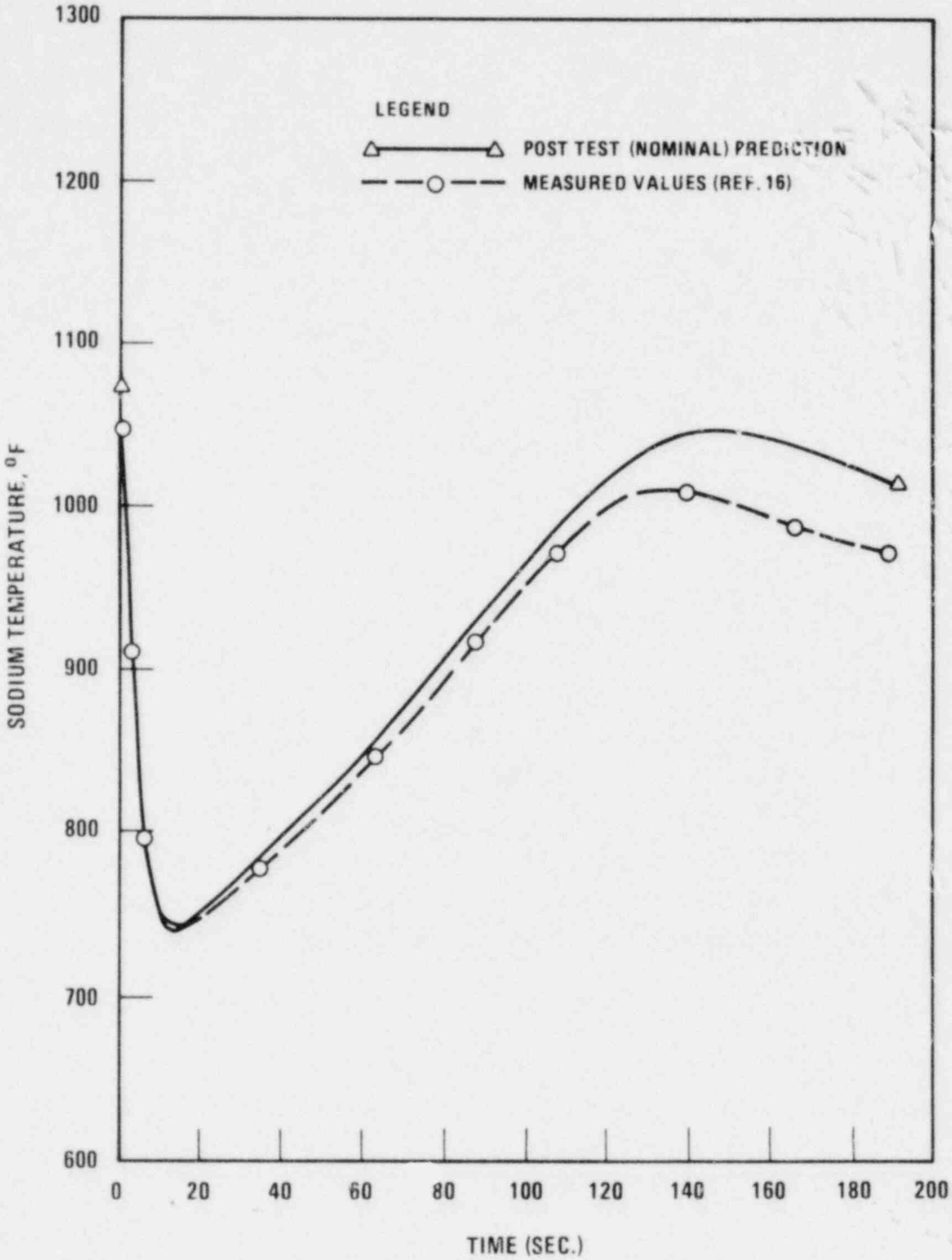

Figure 30. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 15) Row 2 FOTA Initiated at 100% Power/100% Flow Conditions (TX1016)

6838-18

predictions for the hot rod in channel 9 (Location A, Figure 5) for the three cases in Figures 31, 32 and 33. Very similar agreement is observed with the best agreement occurring for the test initiated at 35% power condition. For the test initiated at 100% power condition, the predicted peak temperature is only 30°F (or 9% of the temperature rise) higher than the measured temperature.

Similar curves for the post-test sodium temperature predictions at the top of the fuel section for the hot rod in channel 14 (Location D, Figure 5) and for the hot rod in channel 13 (Location C, Figure 5) of Row 6 FOTA are shown in Figures 34-39. The agreement between the prediction and measured values is similar to and somewhat better than that for the Row 2 FOTA (i.e., the difference in peak temperature is less than 6% of the peak temperature rise). Thus it may be concluded that these peak temperature predictions are validated by the experiments.

8.0 DISCUSSIONS AND CONCLUSIONS

By using different cases during the pre-test predictions, the effects of uncertainties in decay heat and pressure drop calculations upon the "hot channel" sodium temperature were determined. This was the basis for the determination of the uncertainty subfactors F_2 and F_3 listed in Table 1. The decay heat uncertainty effect, F_3 was found to be different for the two FOTA's but less dependent on the initial conditions. The use of Table 1 is consistent with the methodology to be applied to the CRBRP.

The calculation of decay power in accordance with the actual pre-scram power history and the reactor flow match performed in the post-test predictions removed some of the uncertainties in the pre-test predictions. The post-test predictions of the sodium temperatures (Figures 28-39), therefore, agree better with the measured data. The deviations in peak sodium temperature are small, which are due primarily to the mismatch between measured and calculated reactor flow during this part of the transient.

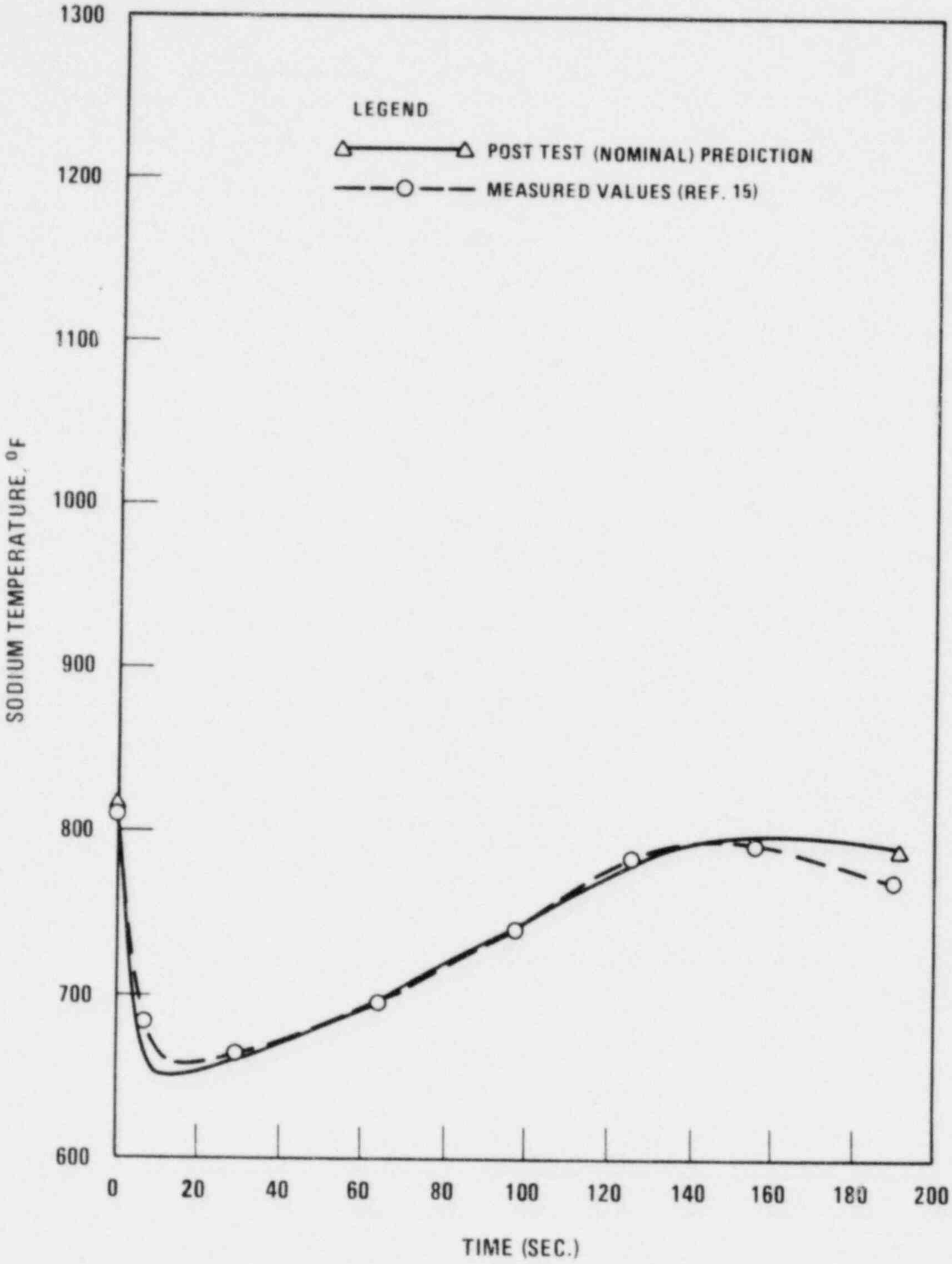

Figure 31. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 9) Row 2 FOTA Initiated at 35% Power/75% Flow Conditions (TX1018)

6838-19

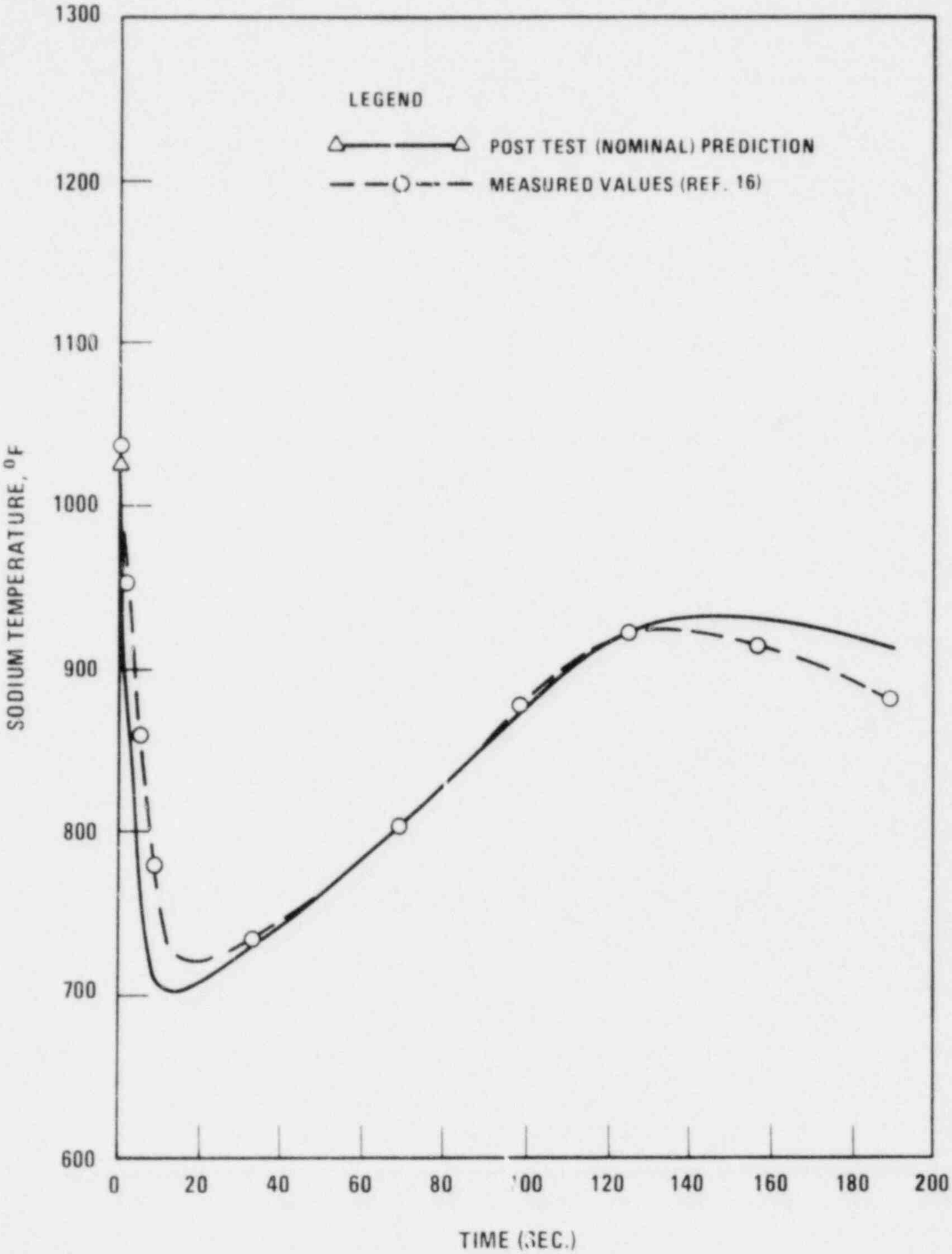

Figure 32. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 9) Row 2 FOTA Initiated at 75% Power/75% Flow Conditions (TX1018)

6838-20

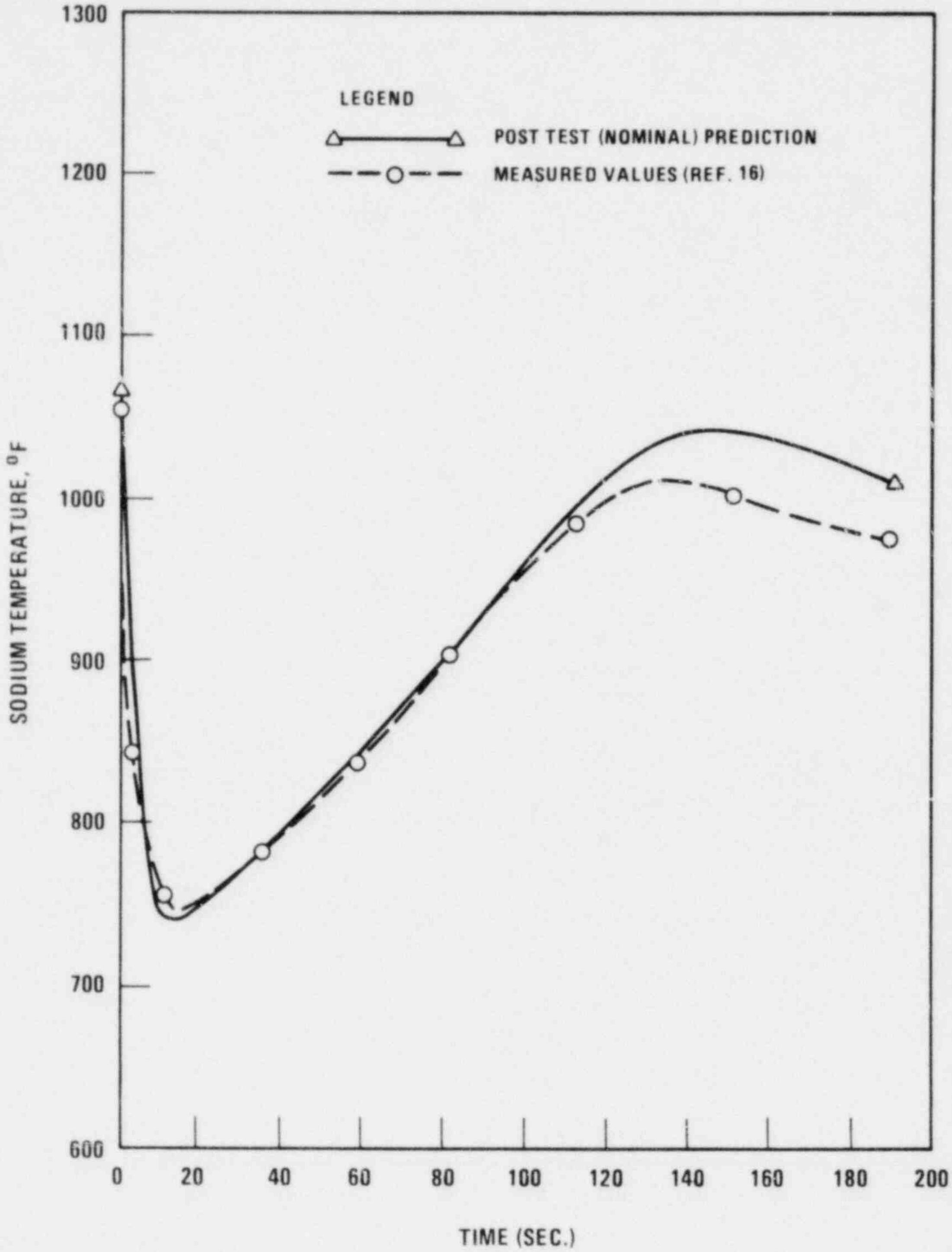

Figure 33. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 9) Row 2 FOTA Initiated at 100% Power/100% Flow Conditions (TX1018)

6838-21

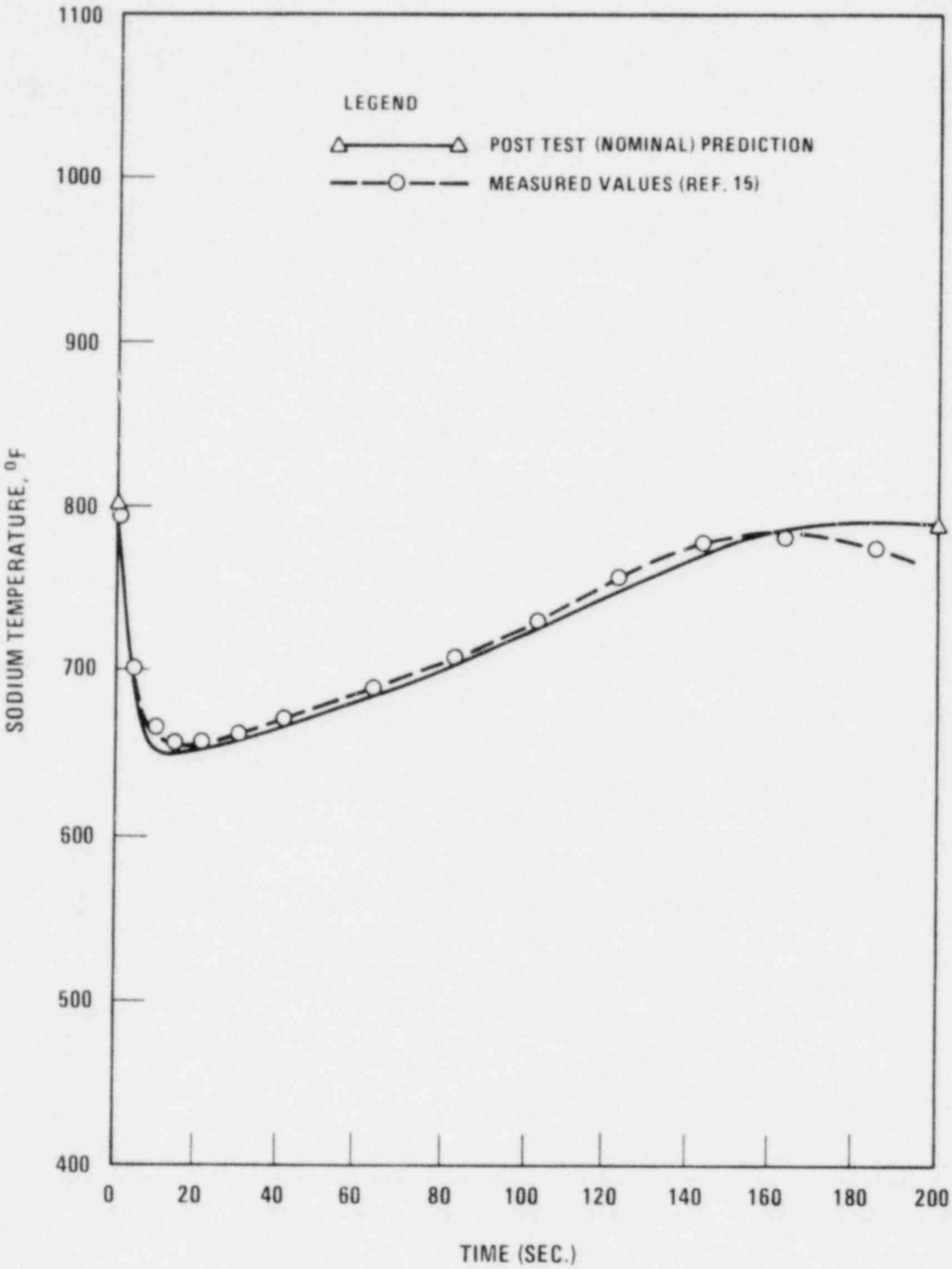

Figure 34. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 13) Row 6 FOTA Initiated at 35% Power/75% Flow Conditions (TX9018)

6838-22

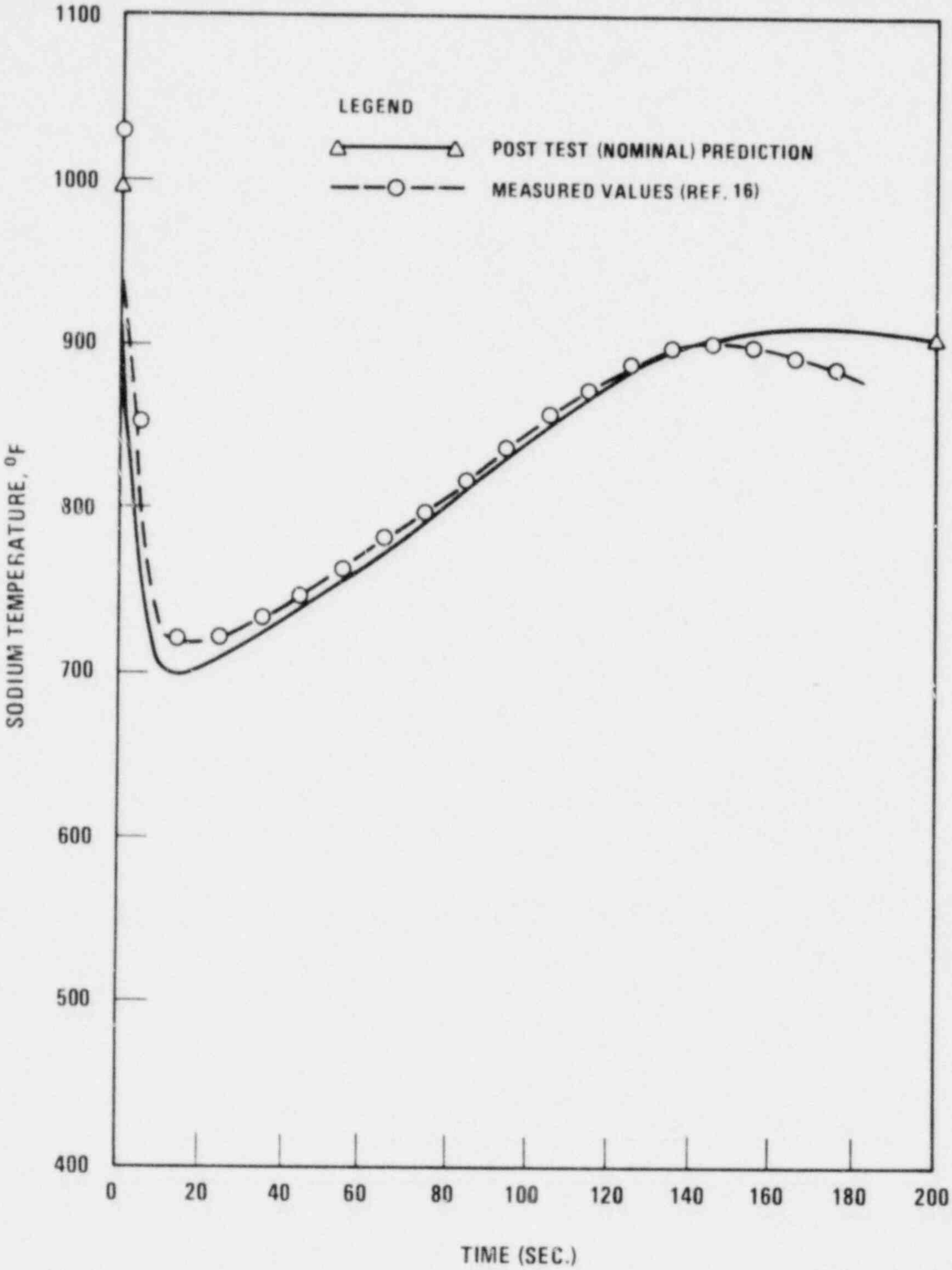

Figure 35. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 13) Row 6 FOTA Initiated at 75% Power/75% Flow Conditions (TX9018)

6838-23

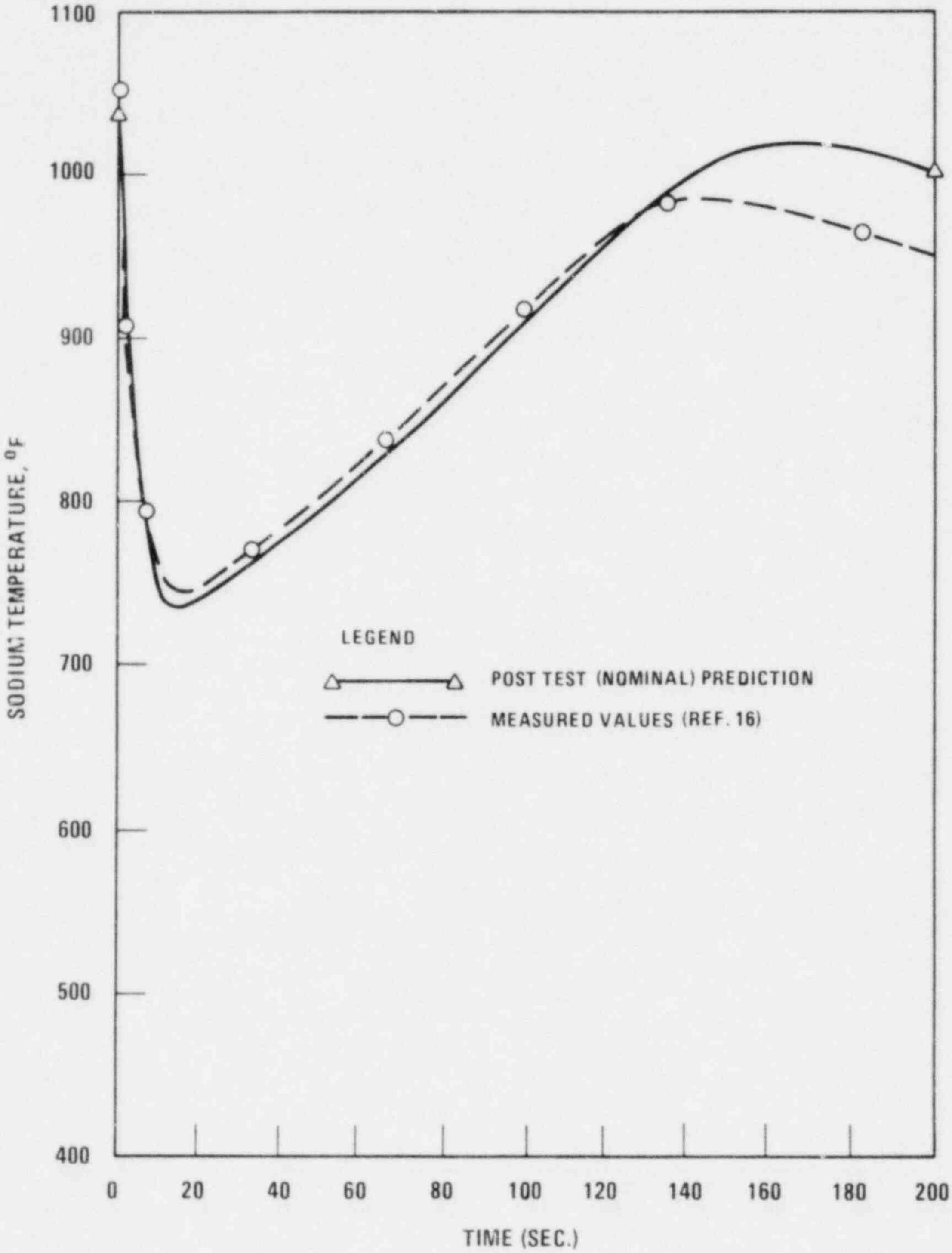

Figure 36. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 13) Row 6 FOTA Initiated at 100% Power/100% Flow Conditions (TX9018)

6838-24

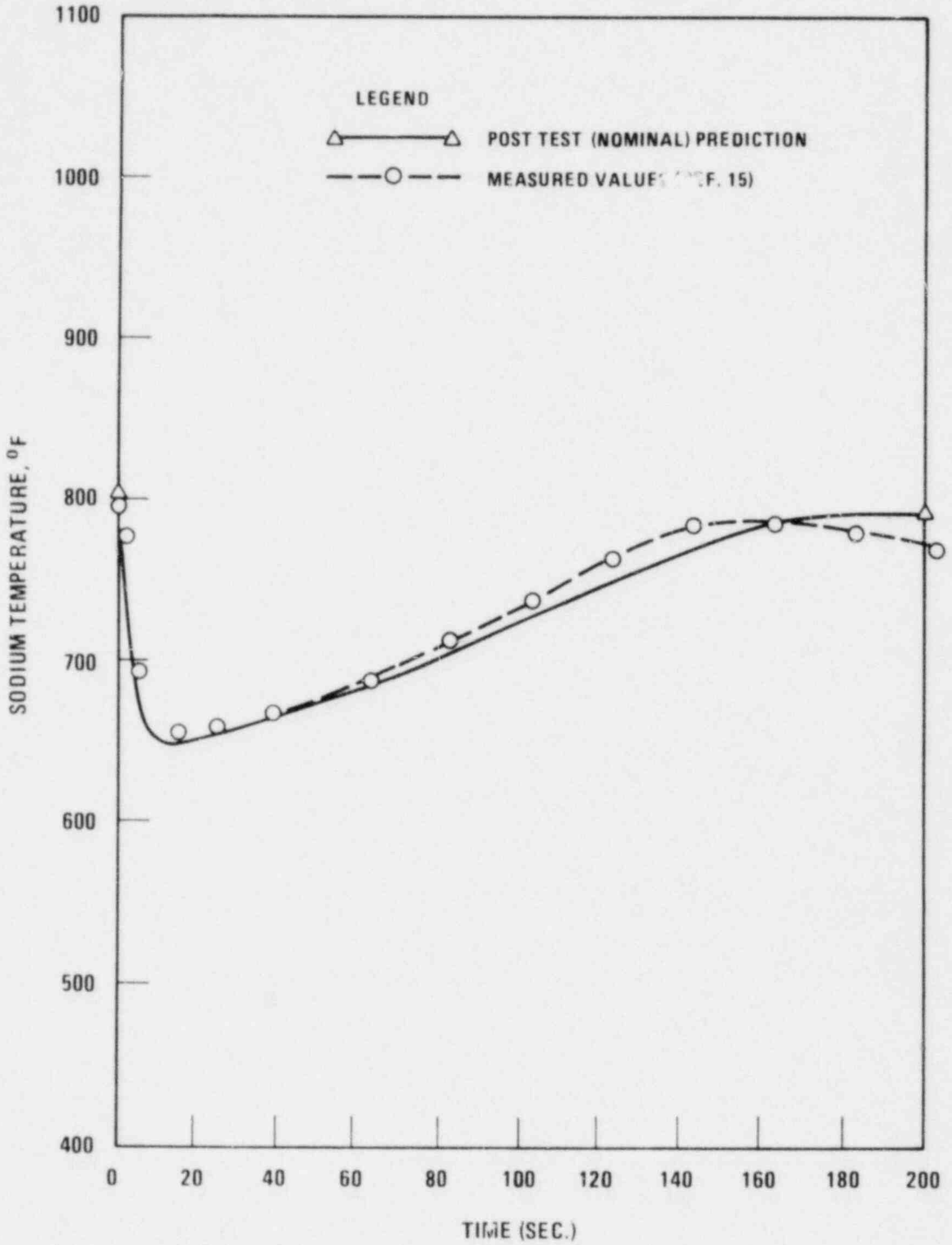

Figure 37. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 14) Row 6 FOTA Initiated at 35% Power/75% Flow Conditions (TX9016)

6838-25

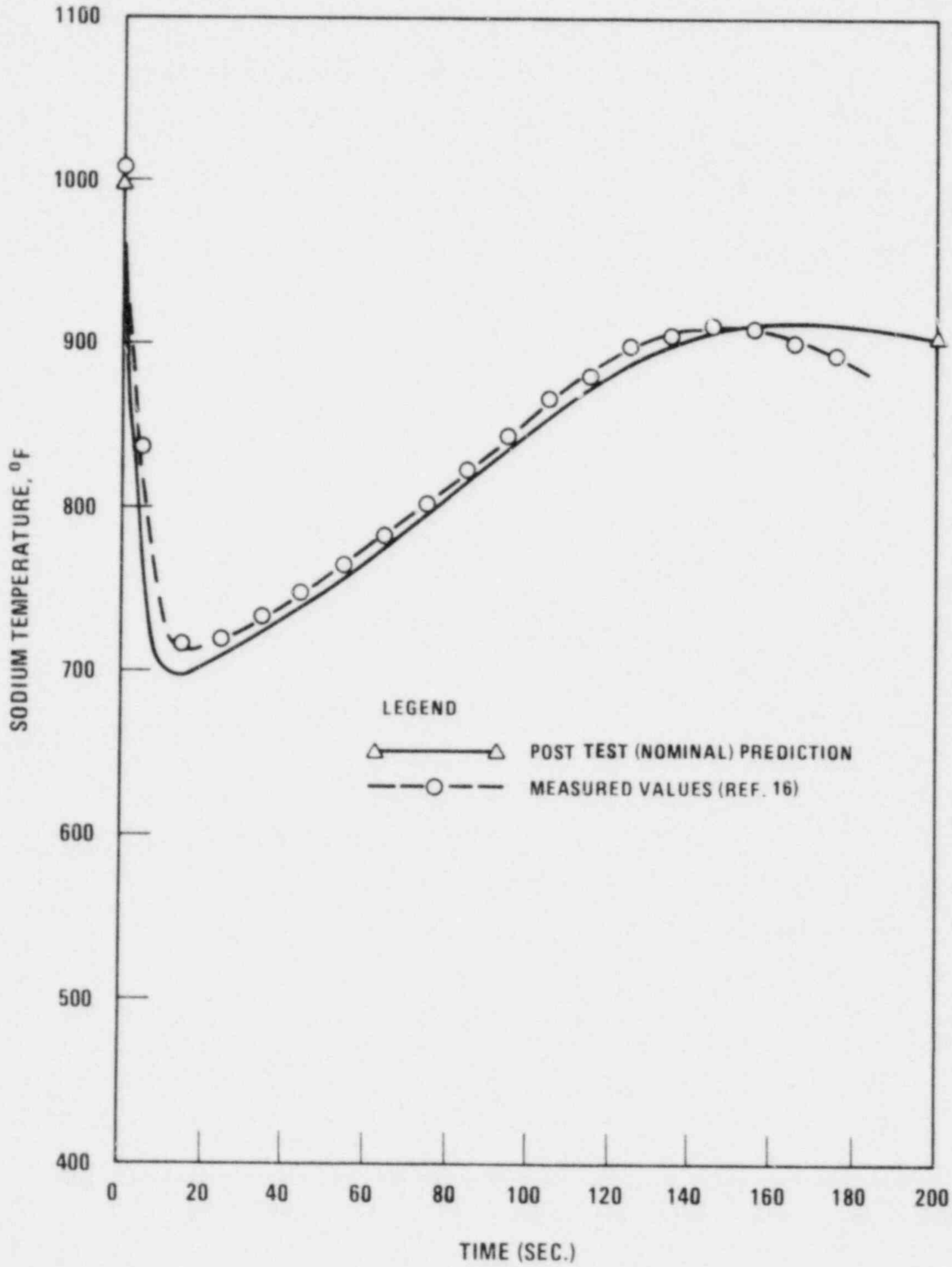

Figure 38. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 14) Row 6 FOTA Initiated at 75% Power/75% Flow Conditions (TX9016)

6838-26

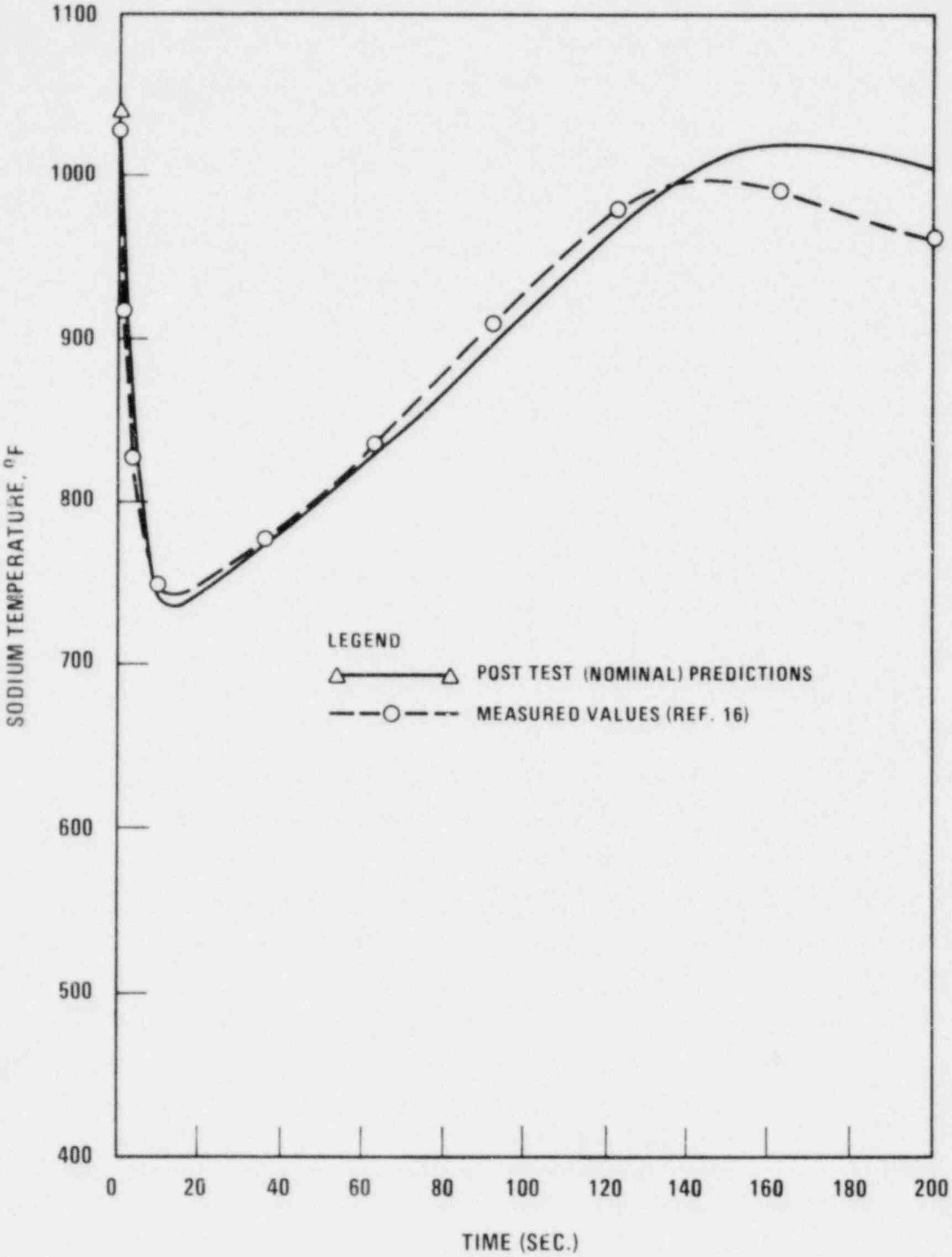

Figure 39. Post Test Sodium Temperature Predictions at Top of the Fuel Section (Hot Rod in Channel 14) Row 6 FOTA Initiated at 100% Power/100% Flow Conditions (TX9016)

6838-27

Some temperature anomaly was found in the measured temperatures near the center of bundle. A non-distorted rod bundle with a nearly flat radial power distribution (Row 2 FOTA) and with fairly uniform temperature assemblies surrounding it should have its maximum temperature occurring at the central rods. However, the thermocouple data indicate the maximum temperature at steady state is occurring at location A (Figure 5, midway between the center and edge rods).[20] This phenomenon existed also during the transient (Figure 40). For this reason, a comparison between post-test prediction and measured data on thermocouple location TX1018 for Row 2 FOTA was also made. The agreement appears to be consistent. In the case with Row 6 FOTA, TX9016 thermocouple reading at location D (Figure 5) was also compared with the calculated values because it became the peak temperature during the transient (Figure 41).

In conclusion, the use of the CRBRP methodology developed for the prediction of natural circulation test performance has provided a sound basis for similar evaluations. The uncertainty factor used in the evaluation of peak sodium temperature was established by means of conservative models. The post-test analyses for tests with three different pre-scram power levels resulted in good agreement with the measured data, with the predicted values always in the conservative direction. The maximum measured sodium temperatures at the peak channels are less than the curve shown with uncertainties (Figures 12-17) thereby meeting the acceptance criteria established previously.[12] It therefore follows that the calculations made with design data conservatively envelop the temperature response of the natural circulation tests. The model and methodology are therefore validated for such design transient analyses. Also shown in these figures are maximum sodium temperatures determined by the fixed flow fraction approach as reported in Reference 21 which are consistently even higher than the measured values. This further demonstrates the conservatism in this approach.

9.0 ACKNOWLEDGEMENT

The work involving the COBRA-WC computer code was performed under US/DOE Contract DE-AC06-76RL01830 by Battelle Pacific Northwest Laboratory (PNL).

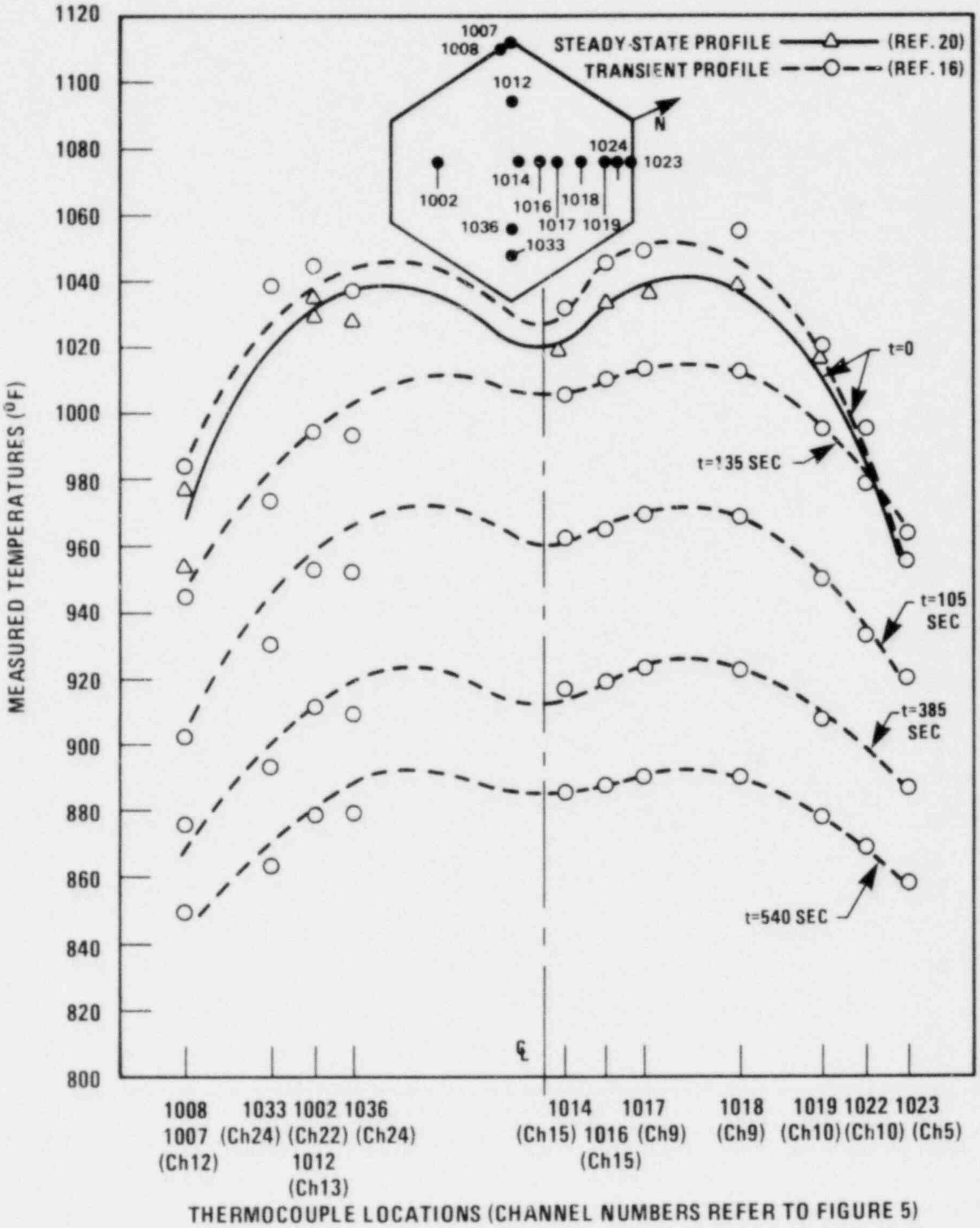

Figure 40. Steady-State and Transient Temperature Profiles (Row 2 FOTA) Initiated at 100% Power

6838-28

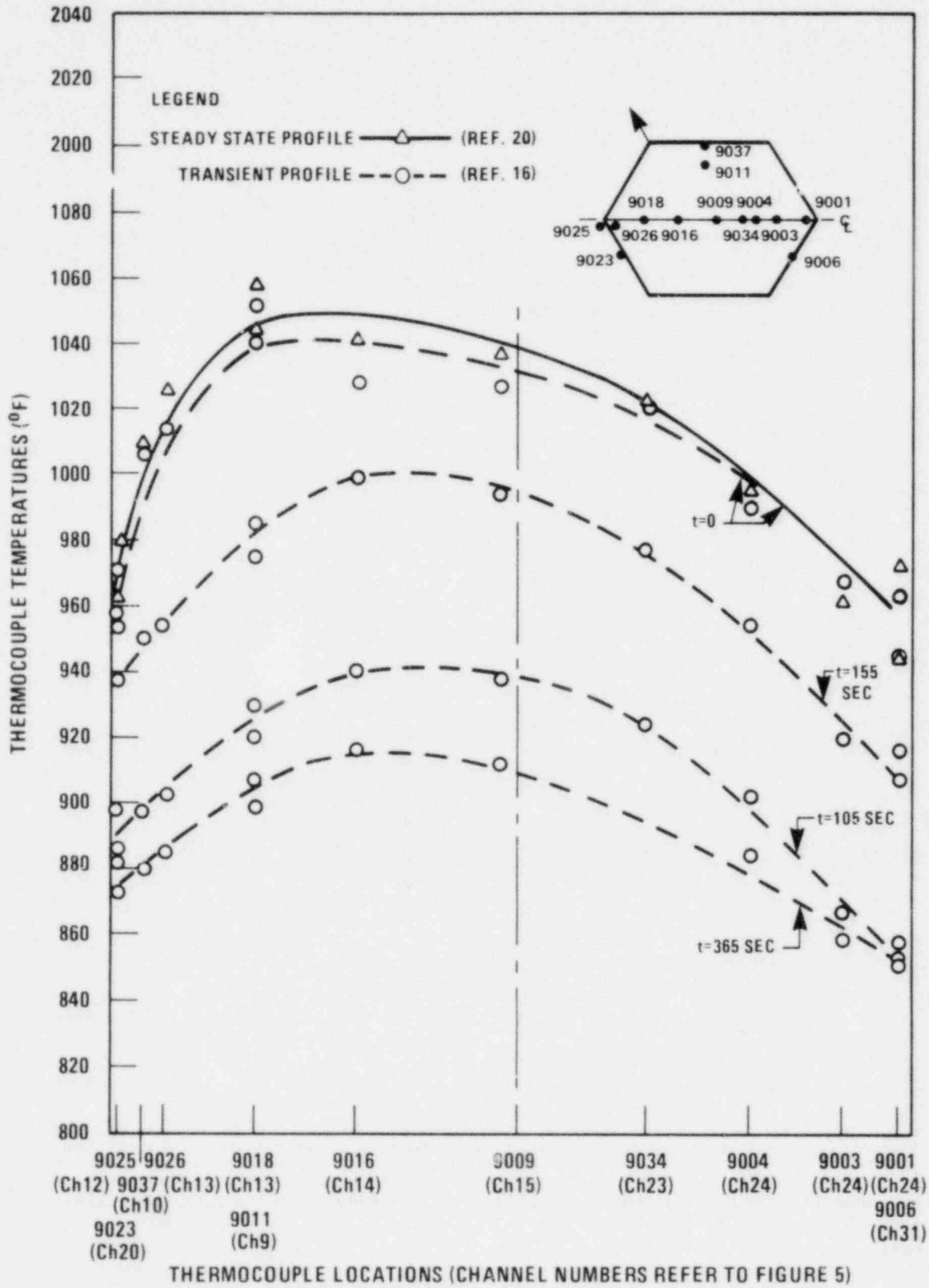

Figure 41. Steady-State and Transient Temperature Profiles (Row 6 FOTA) Initiated at 100% Power

6838-29

The authors are indebted to Thomas L. George, W. A. Prather, D. R. Rector and Dr. E. V. Khan of PNL for providing these calculations. Acknowledgement is also due to Harry Johnson and Dr. W. T. Nutt for their assistance in obtaining FFTF information.

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[12] Ltr. R. L. Copeland, CRBRP P/O to D. G. Eisenhut, NRC, PS:80:359, "Pre-Test Prediction of FFTF Natural Circulation," dated November 26, 1980. (Attached to this report as Appendix A.)

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APPENDIX A

Pre-Test Prediction of FFTF Natural Circulation

(Letter PS:80:359, R. L. Copeland, CRBRP P/O
to D. G. Eisenhut, NRC, dated
November 28, 1980)

Docket No. 50-537

(W/Attachments I, III, IV)

Department of Energy
Clinch River Breeder Reactor
Plant Project Office
P.O. Box U
Oak Ridge, Tennessee 37830
Docket No. 50-537

November 28, 1980

Mr. Darrell G. Eisenhut, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Eisenhut:

PRE-TEST PREDICTION OF FFTF NATURAL CIRCULATION

Reference: Letter S:L:1195, A. R. Buhl to R. S. Boyd, "Natural Circulation Decay Heat Removal Verification Plan," dated June 21, 1976.

This letter transmits information concerning the pre-test prediction of the results of the natural circulation test scheduled for early December 1980 at the Fast Flux Test Facility (FFTF). The pre-test prediction is based on CRBRP methodology and computer codes as discussed in the reference. This pre-test prediction is provided to clearly document that the data and methodology which will be used in the post-test analyses were selected prior to the test.

The pre-test prediction assumes reactor scram and trip of the sodium circulating pumps from a pre-existing condition of 35% of full power and 75% of full flow. A test with similar conditions is scheduled for December 1980. Additional tests will be performed for conditions of (1) 75% power and 75% flow and (2) 100% power and 100% flow. Pre-test predictions utilizing CRBRP methodology and computer codes will be performed for those tests. The results will be provided to you prior to the tests.

Following completion of all three natural circulation tests, post-test analyses will be performed utilizing the actual test boundary conditions. The pre-test predictions are, of necessity, based on boundary conditions (e.g., power history and heat sink temperature) selected by the analysts. It is unlikely that these boundary conditions will be duplicated during the actual tests. Accordingly, the post-test analyses will be used for comparison with experimental data. A report will be compiled to consolidate all of the pre-test and post-test predictions and discuss any differences between those analytical results and the test data.

The pre-test and post-test predictions will be considered "successful" if they demonstrate (1) that the Project methodology provides a reasonable characterization of the phenomena controlling natural circulation flow and heat transfer in both the reactor and the heat transport loops and (2) that use of the methodology results in a conservative prediction of the critical plant parameters (e.g., core hot channel sodium temperature and heat transport system flow rates). Section 8.2 of Attachment II (WARD-94000-00321) discusses the methodology utilized to generate the acceptance boundary curves for the pre-test predictions. The same methodology will be utilized to generate acceptance boundary curves for the post-test predictions. The prediction methodology will be considered acceptable if the measured flows for the tests are equal to or greater than the acceptance boundary curves.

The contents of each of the attachments are as follows:

Attachment I	Overall prediction methodology description
Attachment II	WARD-94000-00321, "DEMØ Pre-Test Predictions for the FFTF Transient Natural Circulation Tests," base case prediction in terms of heat transport system flow rates (WARD-NC-94000-5 issued in January 1982 TIC)
Attachment III	Parametric variation and sensitivity case predictions in terms of heat transport system flow rates
Attachment IV	Base and sensitivity case predictions in terms of reactor core hot channel temperatures
Attachment V	Archive listing of input data for FØRE-2M and CØBRA-WC cases
Attachment VI	Microfiche archive copy of DEMØ base case input, output, and program modifications

We would be pleased to answer any questions you may have on the attachments or to discuss the analyses at your convenience.

Sincerely,

Raymond L. Copeland

Raymond L. Copeland
Acting Assistant Director
for Public Safety

PS:80:359

Attachments

cc: Standard Distribution
Service List
Licensing Distribution

ATTACHMENT I

PREDICTION OF FAST FLUX TEST FACILITY NATURAL CIRCULATION TEST RESULTS USING CLINCH RIVER BREEDER REACTOR PLANT PROJECT METHODOLOGY

PREDICTION OF FFTF NATURAL CIRCULATION TEST RESULTS USING CRBRP PROJECT METHODOLOGY

The objective of the CRBRP Natural Circulation Verification Program (NCVP) is to analytically and experimentally verify numerical models for the description of the thermal-hydraulic behavior of LMFBR plants when making transition to or operating in the natural circulation mode. This verification is an important part of demonstrating the capability of adequate decay heat removal by means of natural circulation in LMFBR's. A description of the NCVP program was provided to NRC in a letter dated June 21, 1976. In addition, the program approach to meeting this objective has been extensively published through various open literature publications and through conferences (Refs. 1, 2, 3, 4). A major component of the NCVP is comparison of analytical predictions with experimental data for experiments in facilities other than CRBRP. Planned natural circulation testing at the Fast Flux Test Facility (FFTF) presents an excellent opportunity to compare Project predictions with experimental data. A program of pre-test and post-test prediction of the FFTF tests is in progress. This attachment provides an overview of the analytical tools, approaches and data utilized for the FFTF test predictions. To the extent practical, the tools, approaches and data are the same as those to be used for prediction of natural circulation scenarios in CRBRP.

The computer codes utilized in the predictions are DEMO, FORE-2M and COBRA-WC. These 3 codes, not a single code, were developed for performing the desired predictions because a single code would be too large to be of practical use for design applications. A brief description of each code and its role in the prediction follows:

A. DEMØ (Ref. 5)

DEMØ, a system-wide code, predicts the coupled performance of the reactor, heat transfer systems, and the steam generating system (including piping and plena heat capacity effects). To provide accurate results for the plant as a whole, yet maintain the code as an amenable tool with regard to computer storage requirements and running time, localized phenomena (which do not affect the system as a whole) are not given detailed modeling. An example of this is an item such as local flow/heat redistribution between and within core assemblies at low flow. DEMØ employs a three region model (fuel, non-fuel and bypass) for the core. To consider these localized phenomena with the required resolution takes a separate computer code itself (i.e., CØBRA-WC was selected for this purpose). In general then, given a natural circulation event, DEMØ provides a verified prediction of the overall system state variables such as net flow through the reactor and bulk temperatures entering and exiting the core.

B. CØBRA-WC (Ref. 6)

The CØBRA-WC code, which accounts for core inter- and intra-assembly flow and heat redistribution, predicts the boundary conditions for a peak rod in the fuel and blanket assemblies given the reactor boundary conditions such as total reactor inlet flow and bulk temperature from DEMØ, individual core assembly powers, and individual core assembly thermal-hydraulic information.

C. FORE-2M (Ref. 7)

Given the localized peak channel flows and heat interchange as a function of time, FØRE-2M predicts the hot channel coolant temperature for natural circulation which includes considering uncertainties in the predictions. This hot channel coolant temperature is used as the basis for determining the acceptability of natural circulation for decay heat removal of LMFBR's. To have a verified hot rod prediction, one must properly account for such effects as: a) fuel restructuring; b) fuel-cladding gap conductance; c) stored heat in the rod; d) statistical significance of physics and engineering uncertainties in power, flow, dimensions, properties, etc.; e) localized hot spots on the rod due to the wire wrap or pellet eccentricity; f) localized decay heat variations; and g) localized rod power variations in the assembly.

Since FØRE-2M is used for core hot rod analysis with input from DEMØ and CØBRA-WC, extreme detail in nodalization can be used to assure accurate temperature predictions. This level of detail cannot be part of the CØBRA and DEMØ predictions since the storage requirements and computer running times would make the codes impractical. Thus, FØRE-2M is used to predict verified hot channel coolant temperatures. [Note: In addition, the code provides a detailed nuclear physics prediction for the prompt and delayed neutron fission power as a function of time from shutdown with the capability of considering Doppler, sodium density, and radial/axial core expansion feedback plus the effects of control system worth and PPS functions. This information is used to verify the DEMØ prediction of core power versus time].

The above analysis procedure is outlined by Figure 1.

The planned FFTF tests encompass the transition into natural circulation cooling from a series of steady state initial conditions. All the tests will be initiated with a reactor scram and main coolant pump trip. The tests will include:

A) A transition to natural circulation in the primary loop from 5% reactor power, 75% primary loop flow. The secondary loop pumps will coast down to 10% speed at which time pony motors engage and provide 10% secondary loop forced flow.

B) A transition to natural circulation in both primary and secondary loops from 35% reactor power and 75% flow.

C) A transition to natural circulation in both primary and secondary loops from 75% reactor power and flow following test subset B.

D) A transition to natural circulation in both primary and secondary loops from 100% reactor power and flow.

The parts of this testing that are of most direct interest for NCVP qualification are the transients run from the three normal power points, i.e., 100%, 75% and 35% reactor power - test subsets B, C and D above.

The pre-test predictions of the tests consists of performing nominal best-estimate calculations plus uncertainty evaluations of the nominal calculation. The reactor flows and inlet temperature versus time for each of these tests is calculated with an FFTF version of the DEMØ code which substitutes a dump heat exchanger for the steam generator. The post-test analysis will assess the effects of actual test conditions as opposed to those used in the analyses, effects of measured uncertainties and consistency of the models in DEMØ, CØBRA-WC and FØRE-2M with test results.

Two basic boundary conditions employed in the pre-test predictions are power history prior to scram (i.e., the decay powers) and the DHX sodium outlet temperature. When the actual values for these parameters are input to the programs used for the pre-test predictions, the results should agree closely with the actual measurements.

In addition, the post-test analysis will include an assessment of the accuracy of the data used in the pre-test predictions. The objective will be to determine, to the extent possible with FFTF instrumentation, how much of the difference between the best-estimate calculations and the measurements is due to uncertainties in the data used in the predictions.

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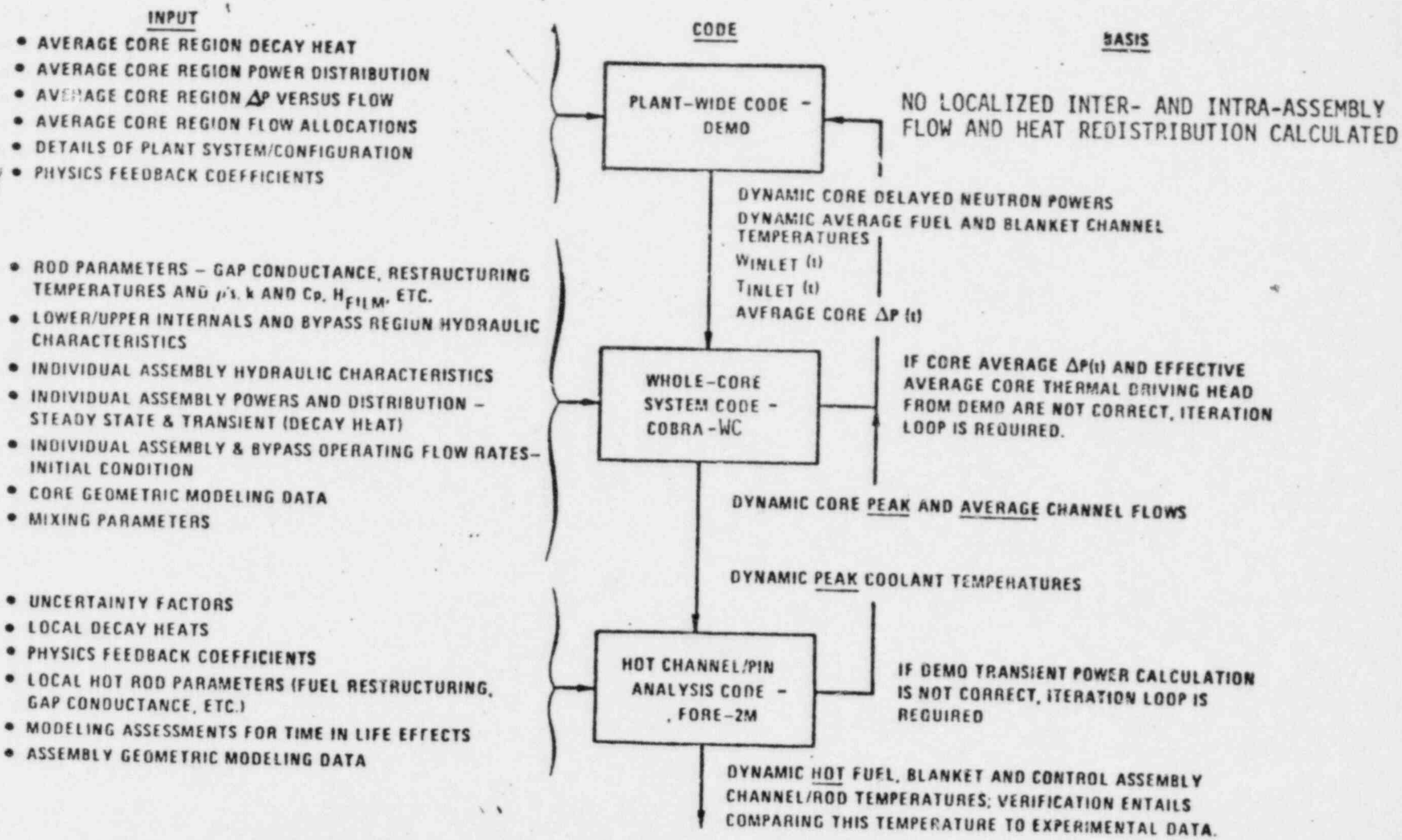


FIGURE 1: LMFBR NCVP ANALYSIS PROCEDURE

ATTACHMENT III

DEMØ PRE-TEST PREDICTIONS OF REACTOR INLET FLOWS AND TEMPERATURES DURING THE FFTF TRANSIENT NATURAL CIRCULATION TEST

ATTACHMENT III

PRE-TEST PREDICTIONS OF REACTOR INLET FLOWS
AND TEMPERATURES DURING THE FFTF
TRANSIENT NATURAL CIRCULATION TEST

Pre-test predictions of reactor inlet flows and temperatures for the FFTF transient natural circulation test from initial conditions of 35% power and 75% flow were made during the first quarter of CY80 and documented as a part of the Attachment II report WARD-94000-00321 published in March, 1980. Subsequent to the issuance of this report, there have been two changes in data input for these predictions which have an impact on the results. These are:

1) The initial predictions assumed a power history for the test which is equivalent to 1 hour of operation at 35% of 400 MW_t power. The actual power history will be in excess of 5 hours of operation at 35% power and may be as long as 56 hours.

2) Pressure drop tests of the fuel assembly inlet nozzle/shield block assembly conducted at low flows typical of natural circulation have shown that the Δ P's at low flows are less than those assumed in the analysis reported in the March predictions.

The significance of these two aspects is the following:

1) Higher decay heats associated with the longer times at power will cause measured temperatures within the core (in the FOTA's) to be higher than those which have been predicted for the 1 hour power history. It was thus necessary to evaluate and quantify this effect. It also means, because of higher core temperature, that the flows will be higher because of the increase in the thermal driving heads.

2) The reactor flows will be somewhat higher for a given thermal head due to the reduced reactor Δ P's associated with the inlet nozzle/shield block assembly.

For these reasons, it was decided to supplement the analyses described in the WARD-94000-00321 with additional analyses of several cases to provide data for FORE-2M analyses so that the upper bound on fuel assembly temperatures could be developed.

Four additional cases for the 35% power/75% flow test were analyzed. Each of these cases was calculated with the same program used to compute the results given in the March report with the exception of decay powers and reactor Δ P's. The new decay powers used for the 56 hour decay heats are shown in Table 1. For those cases using a "maximum" decay power, these values were multiplied by 1.25. The difference in the correlations for the reactor Δ P versus flow for the fuel assembly, non-fuel assembly and by-pass regions are shown in Figures 1 through 3. When the case is identified as a maximum Δ P case, the uncertainties given in the March report were used.

The four cases (in addition to the "best estimate" 35% power/75% flow case given in the March report) are as follows:

CASE 1: Same as the "best estimate" case given in WARD-94000-00321 except a nominal decay heat based on an assumed power history equivalent to 56 hours of operation at 35% power.

CASE 2: Same as CASE 1 except maximum decay heats (+ 25% uncertainty) for a 56 hour power history.

CASE 3: Same as CASE 1 except the reactor pressure drops based on the revised correlations (Figures 1 through 3).

CASE 4: Same as CASE 1 except that high side ("design") pressure drop uncertainties (as described in WARD-94000-00321) were applied to the pressure drop correlations.

The new reactor vessel pressure drop correlations used for the average fuel, non-fuel, and bypass assembly are shown in Figure 1 through 3, respectively. These correlations were calculated from data taken from COBRA-WC analyses which, in turn, were based on the new experimental data on pressure drop through the fuel assembly inlet nozzle/shield blocks from HEDL. The methodology used in the development of these correlations was the same as that described in the March report.

Figure 4 shows a comparison of the predicted primary loop flow in GPM between the base case (results reported in WARD-94000-00321) and the four additional runs. As expected, comparison between the base case and case number 1 shows that higher decay heat (56 hours of operation) resulted in higher primary loop natural circulation flow. The maximum difference in predicted flow due to the new decay heat effect alone is less than 15% for the first 300 second transient. Comparison between the base case and case number 3 shows further increase in the predicted primary loop natural circulation flow due to the lower reactor vessel pressure loss coefficient correlations used in case 3. The maximum difference in predicted flow between these two cases is approximately 20%. Finally, the comparison also shows that case 4 utilizing the "design" pressure drops and the nominal decay heat for the 56 hrs. power history resulted in the most severe transient in terms of natural circulation flow. Figure 5 shows the same comparison on an expanded scale for clarity.

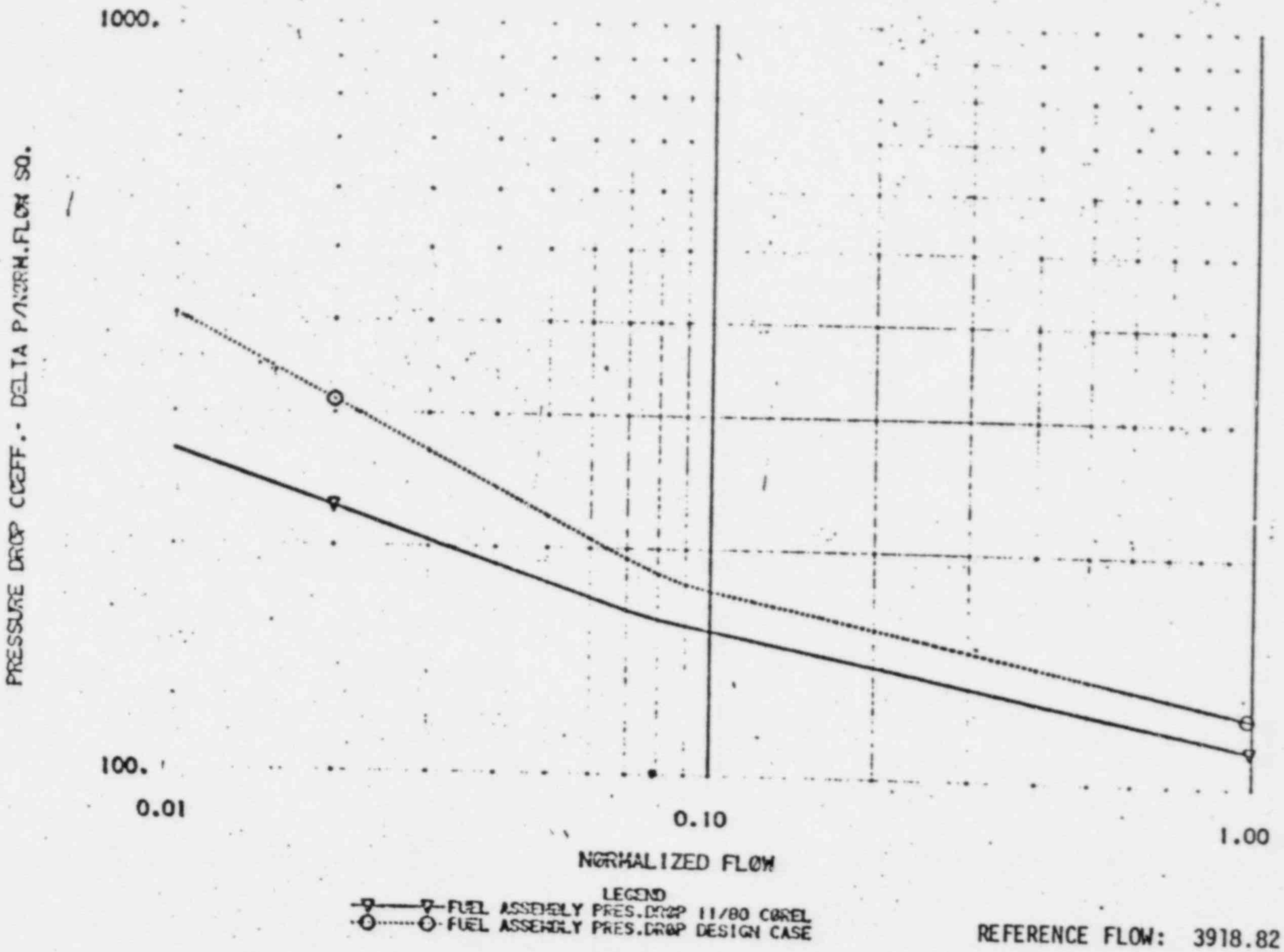
TABLE 1

DECAY HEAT* FOR 56 HR. INITIAL OPERATION AT 35% FULL POWER

TIME (SEC)	FUEL DECAY HEAT (MW)	N-F/A DECAY HEAT (MW)
0.0	6.769	0.283
1.0	6.769	0.283
3.0	6.111	0.258
6.1	5.459	0.233
12.0	4.998	0.215
24.0	4.488	0.196
42.0	4.053	0.179
60.6	3.769	0.168
78.8	3.561	0.160
97.0	3.398	0.154
115.2	3.266	0.149
133.3	3.157	0.145
151.5	3.064	0.141
169.7	2.984	0.138
187.9	2.915	0.135
206.0	2.843	0.132

*Neutronic heating needs to be added to the decay heat for total heat to fuel assembly and nonfuel assembly.

DYN PRES.DRØP R.V.FUEL ASSY   1



REFERENCE FLOW: 3918.82 LB/SEC

Figure 1

DYN PRES.DROP NON-FUEL ASSY

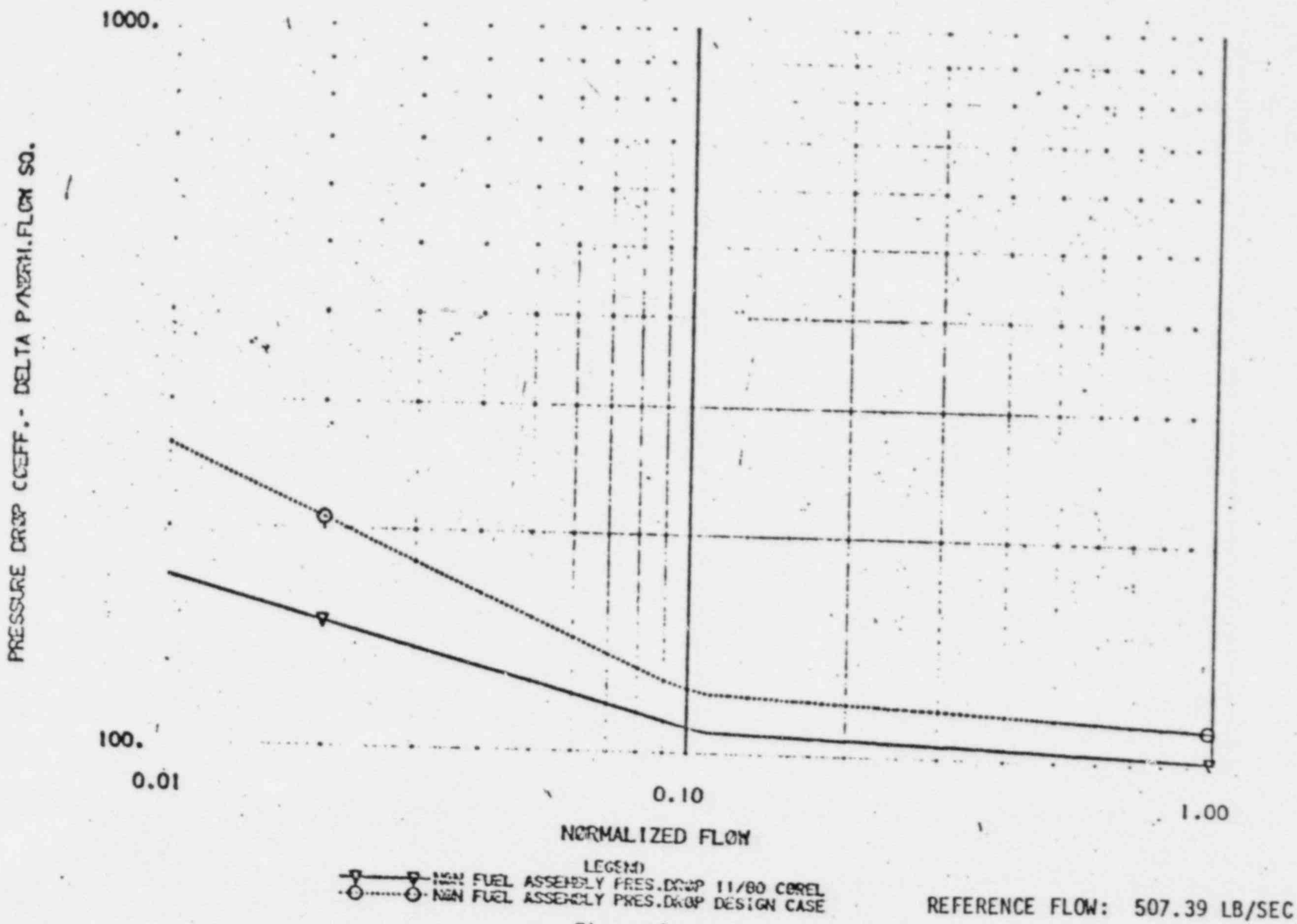
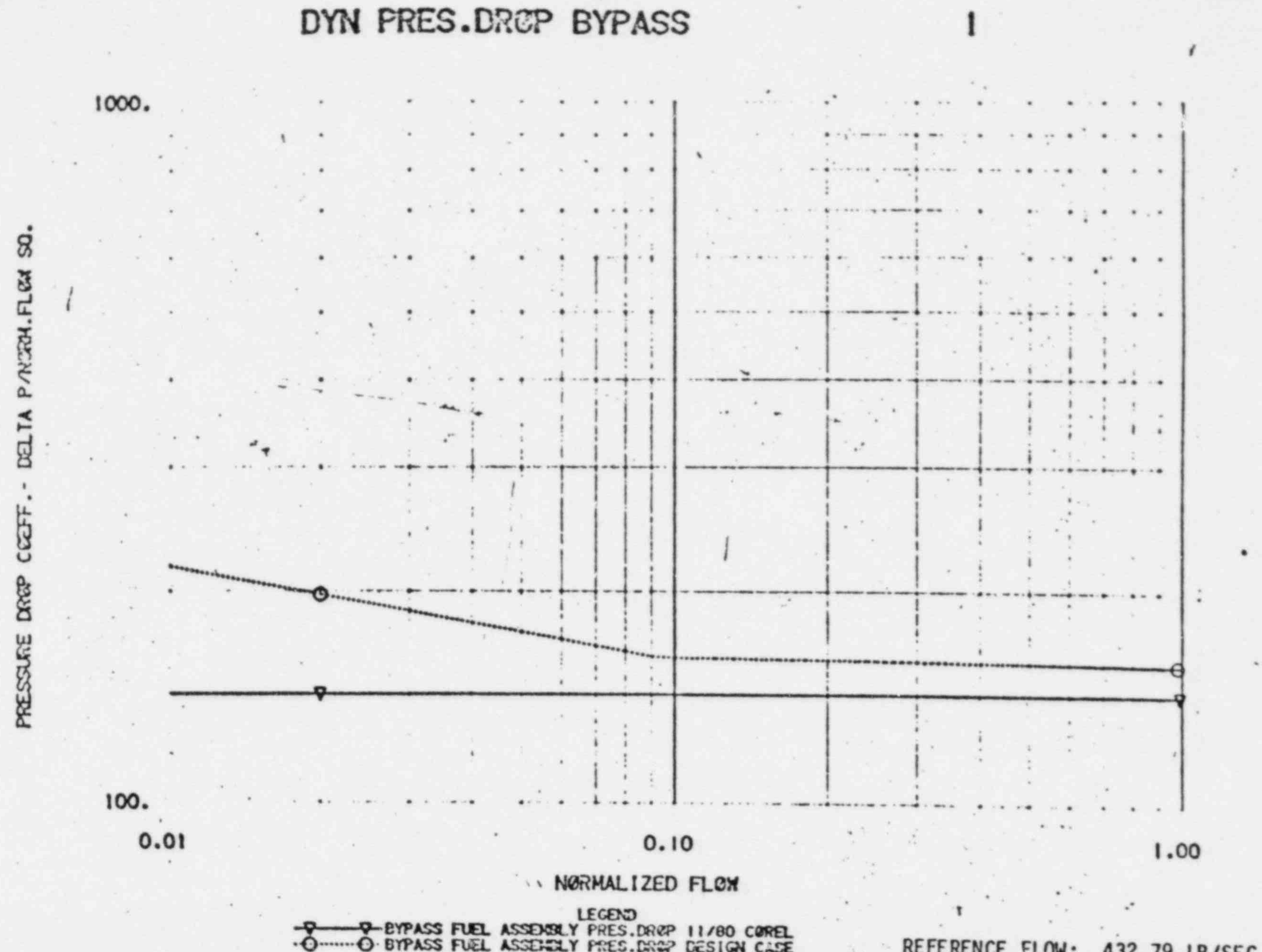


Figure 2


DYN PRES.DRØP BYPASS
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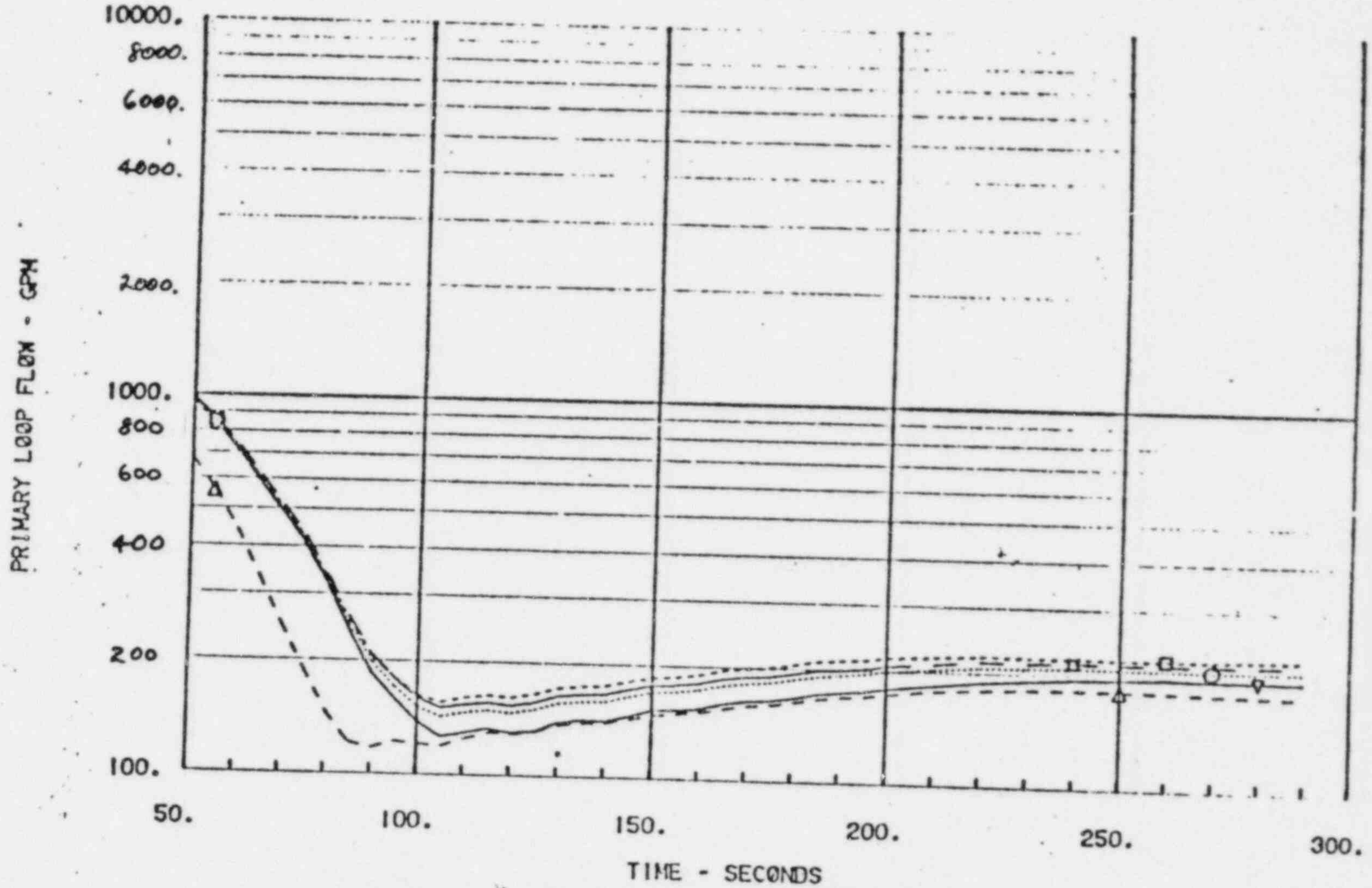
1000.
100.
PRESSURE DRØP CØEFF.- DELTA P/NØRM.FLØW SQ.
0.01
0.10
1.00
NØRMALIZED FLØW
LEGEND
BYPASS FUEL ASSEMBLY PRES.DRØP 11/80 CØREL
BYPASS FUEL ASSEMBLY PRES.DRØP DESIGN CASE
REFERENCE FLOW: 432.79 LB/SEC

Figure 3

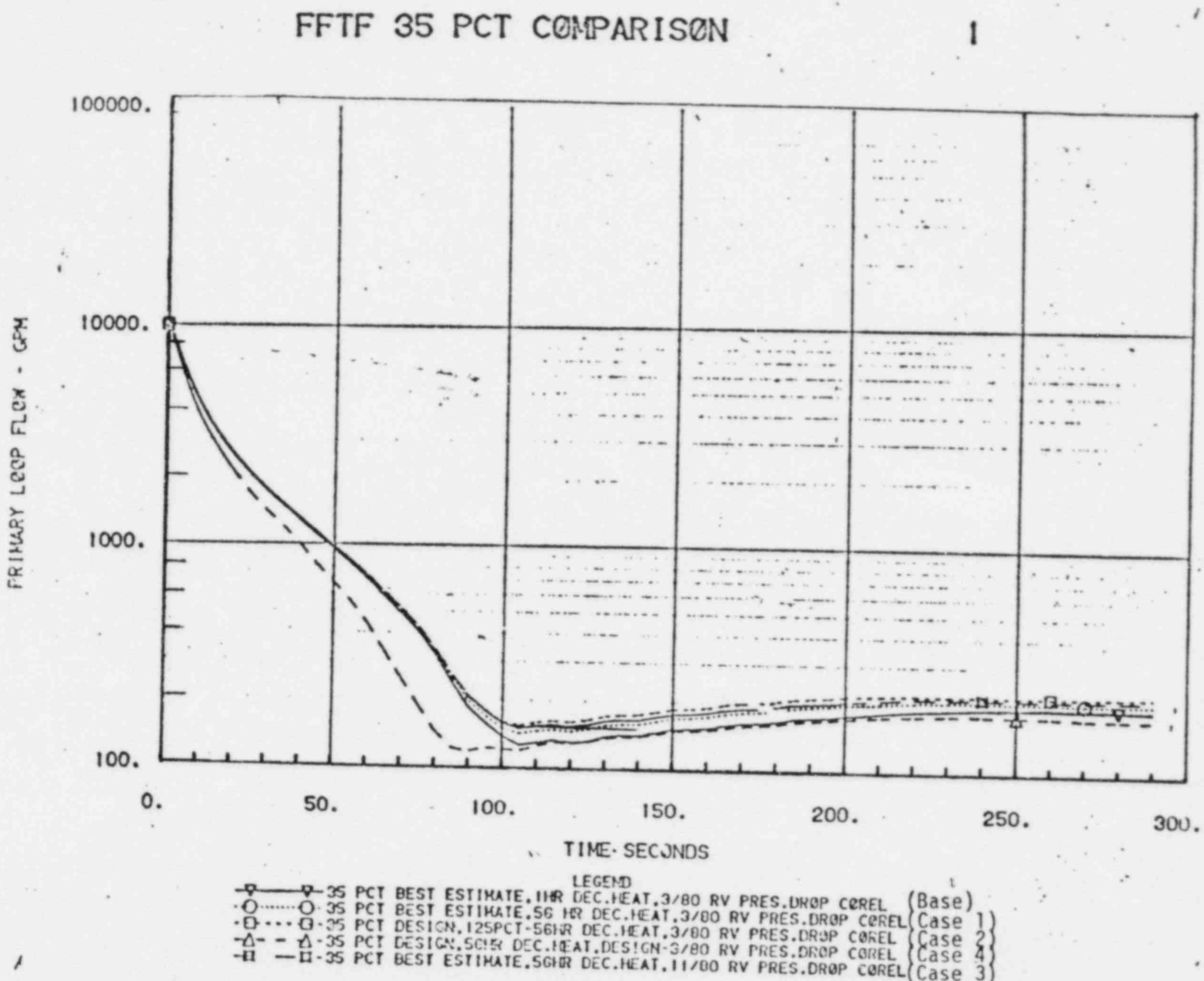
FFTF 35 PCT COMPARISON



LEGEND

-▽——▽-35 PCT BEST ESTIMATE, IHR DEC.HEAT, 3/80 RV PRES DROP COREL (Base)
-O······O-35 PCT BEST ESTIMATE, SGHR DEC.HEAT, 3/80 RV PRES DROP COREL (Case 1)
-□----□-35 PCT DESIGN, 125PCT - SGHR DEC.HEAT, 3/80 RV PRES DROP COREL (Case 2)
-△- - △-35 PCT DESIGN, SGHR DEC.HEAT, DESIGN - 3/80 RV PRES DROP COREL (Case 4)
-⌗ —⌗-35 PCT BEST ESTIMATE, SGHR DEC.HEAT, 11/80 RV PRES DROP COREL (Case 3)

Figure 5


FFTF 35 PCT CØMPARISØN
PRIMARY LØØP FLØW - GPM
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TIME SECONDS
LEGEND
35 PCT BEST ESTIMATE.1HR DEC.HEAT.3/80 RV PRES.DRØP CØREL (Base)
35 PCT BEST ESTIMATE.56 HR DEC.HEAT.3/80 RV PRES.DRØP CØREL (Case 1)
35 PCT DESIGN.125PCT-56HR DEC.HEAT.3/80 RV PRES.DRØP CØREL (Case 2)
35 PCT DESIGN.56HR DEC.HEAT.DESIGN-3/80 RV PRES.DRØP CØREL (Case 4)
35 PCT BEST ESTIMATE.56HR DEC.HEAT.11/80 RV PRES.DRØP CØREL (Case 3)

Figure 4

ATTACHMENT IV

CRBRP NCVP PRE-TEST PREDICTION OF
FOTA TEMPERATURES FOR FFTF NATURAL CIRCULATION TEST
INITIATED AT 35% POWER AND 75% FLOW CONDITIONS

WARD-D-0274
Category II

CRBRP NCVP PRE-TEST PREDICTION OF FOTA TEMPERATURES FOR FFTF NATURAL CIRCULATION TEST INITIATED AT 35% POWER AND 75% FLOW

by

R. D. Coffield
K. D. Daschke
Y. S. Tang

November 1980

REVIEWED: Y. S. Tang
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APPROVED: R. A. Markley
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Core T&H Analysis

R. M. Vijuk
R. M. Vijuk, Manager
Reactor Analysis and Core Design

1. Introduction

The purpose of this report is to present pre-test predictions of the thermal and hydraulic responses of the fueled open test assemblies (FOTA's) in the FTR for the FFTF natural curculation plant startup test initiated at 35% power and 75% flow. The 35% power test is one of the series of natural circulation tests to be performed in the FFTF during acceptance testing. Pre-test predictions for the FFTF 75 and 100% power tests will also be perfomed and documented in a forthcoming report. A summary description of the FTR core configuration is given in the appendix.

The natural circulation tests demonstrate the capability of adequate decay heat removal by means of natural circulation and provide the basis for the verification of the analytical tools used to predict the flow and temperature responses in LMFBRs. The COBRA-WC and FORE-2M codes as well as DEMO have been used for the core analyses. These predictions are the final step in an extensive verification program (Refs. 1, 2 and 12) on these codes which includes comparisons with test data. existing analytical solutions, and recognized codes as well as comparisons with numerical solutions and sensitivity analyses. The detailed description and results of these verifications of the COBRA-WC and FORE-2M codes will be published in future reports.

During the transition to and operation in the natural convection cooling mode, the effect of a power-to-flow ratio greater than that at steady state is experienced. Consequently, core temperatures increase and natural convection phenomena such as inter- and intra-assembly flow redistribution become significant once low flow conditions are reached. In the CRBR or FTR the core thermal head becomes increasingly significant relative to the form and friction losses across the core for flows below 5% of full flow. Coupled with the flow redistribution, significant heat redistribution on an inter- and intra-assembly basis occurs throughout the core due to radial temperature differentials and an increased flow transport time. Both of these effects (i.e., natural convection flow and heat redistribution) have been found to significantly reduce maximum core temperatures as demonstrated by the EBR-II natural circulation experiments (Ref. 3) and Brookhaven investigations (Ref. 4).

2. ANALYTICAL APPROACH

To analyze LMFBR decay heat removal capability, a system of three computer codes has been developed and is used in sequence, i.e., 1) DEMØ, for plant-wide analyses (Ref. 5); 2) CØBRA-WC, for core system analyses (Ref. 6); and 3) FØRE-2M for localized core hot rod analyses (Ref. 7). The last two codes provide the two-stage calculational approach for the core analyses under natural circulation conditions. Utilizing DEMØ boundary conditions, a detailed whole-core flow and heat redistribution analyses of all the parallel core assemblies and bypass regions is first performed by the CØBRA-WC code. This is done for a sector of symmetry of the reactor core. By necessity, the calculational matrix or nodal representation, especially distant from the area of interest, is less detailed than that in the area of interest, i.e., FOTA. Clusters of rods rather than individual subchannels are modeled. The results of this analysis provide the effect of reactor inter- and intra-assembly flow and heat redistribution on the temperature response. The data are then used for a detailed analysis on a single, "hot rod" using the FØRE-2M code. These latter analyses include effects of localized phenomena and uncertainties in nuclear/thermal-hydraulic/mechanical data.

A linkage between the CØBRA-WC and FØRE-2M codes has been developed and verified to incorporate the inter- and intra-assembly phenomena into the localized hot rod natural circulation analyses (Ref. 8). For each axial node of the hot rod modeled in FØRE-2M, a heat balance is performed using the expression for the heat transferred to the coolant at that section, $Q_c(\chi,\tau)$ as

$$Q_c\ (\chi,\tau) = Q_r\ (\chi,\tau) + Q_{ex}\ (\chi,\tau) \tag{1}$$

where

$Q_r\ (\chi,\tau)$ = heat transferred from the rod surface at axial location χ and time τ;

$Q_{ex}(\chi,\tau)$ = coolant heat input or loss due to radial conduction and mixing heat transfer and flow redistribution to adjacent coolant channels; directly input from CØBRA-WC.

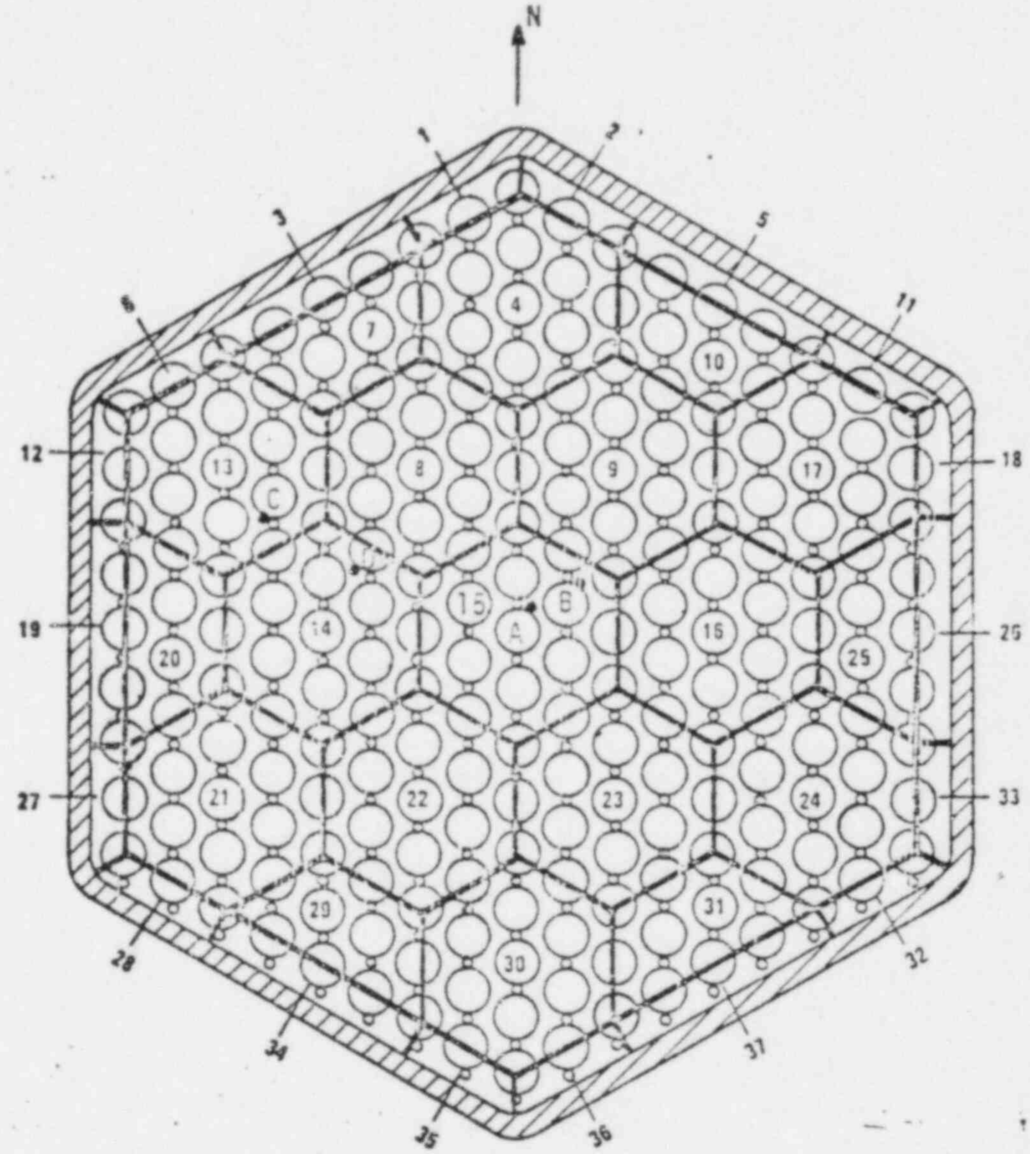
Coupled with this, the axial mass flow rate for each axial node $G(\chi,\tau)$ is also input from CØBRA-WC analyses. Both $Q_{ex}(\chi,\tau)$ and $G(\chi,\tau)$ are based on nominal conditions in the CØBRA-WC code. It is conservative to use $Q_{ex}(\chi,\tau)$ and $G(\chi,\tau)$ data in this fashion because these values are lower than those calculated for the hot channel temperature conditions, and thus, result in a conservatively higher predicted hot channel temperature.

Boundary conditions for the CØBRA-WC and/or FØRE-2M analyses (e.g., plenum-to-plenum pressure drop and coolant inlet temperature) are furnished by the plant-wide code, DEMØ (Ref. 9) for several cases: the "best estimate" or nominal case as well as cases with uncertainties. Likewise, a corresponding modeling of the core parallel flow network, with regard to pressure drop and decay heat uncertainties, can be used in the CØBRA-WC analyses for input to the FØRE-2M hot rod temperature predictions. For the 35% test, only the case with nominal conditions and minimum assumed irradiation history was explicitly calculated for both the FOTA 2 and 6 via the DEMØ/CØBRA-WC/FØRE-2M sequential method as an initial base run. For other cases, the method was used implicitly as will be described later in this section.

Figure 1 shows the model used by the CØBRA-WC code for the FFTF FOTA's to maintain both a reasonable computer running time and adequate level of detail (Ref. 10). In this case, a 217-rod FFTF fuel assembly has been divided into 37 channels. Each channel shown is represented by one rod with power and flow conditions corresponding to the average for all rods contained in it. For the interior, hexagonal shaped regions, this averaged rod would represent a total of 12 rods (other assemblies were modeled in less detail).

To evaluate the effects of core uncertainties, hot channel factors are used (Table 1) in FØRE-2M. Similar to those used in steady state calculations, the direct and statistical type factors are conservatively applied by the semi-statistical method. For instance, the inlet flow uncertainty of -5% (Ref. 11) to the FOTA is conservatively assumed to occur. The statistical subfactors which are statistically combined include those due to uncertainties in the decay heat calculations (±25%), the pressure drop calculational uncertainties (specified in Ref. 9) and the factors for power level measurement, nuclear power distribution and coolant property uncertainties as used in Ref. 8.

The effect of the recent FFTF shield/orifice assembly pressure loss coefficient data (from HEDL) was accounted for in these predictions. The procedure for evaluating such a global effect as well as those due to decay heat and pressure drop calculational uncertainties was to utilize DEMØ code results (Appendix A) to determine flow variations. The FØRE-2M code was then used to determine resultant temperature changes. Similarly, the effect of the core



NOTES: Location A for Row 2 FOTA on Rod (9,9) FFTF Designation
Location B for Row 2 FOTA on Rod (8,8) FFTF Designation
Location C for Row 6 FOTA on Rod (4,4) FFTF Designation
Location D for Row 6 FOTA on Rod (6,6) FFTF Designation

FIGURE 1

TYPICAL CØBRA-WC 37 CHANNEL MODEL OF A 217-ROD ASSEMBLY

TABLE 1

HOT CHANNEL FACTORS APPLIED TO HOT ROD TEMPERATURES IN FØRE-2M CALCULATIONS FOR N/C TESTS

("1 Hr-UNC" AND "56 Hr-UNC" CASES)

A.	DIRECT SUBFACTORS	
	1) INLET FLOW UNCERTAINTY	1.05
B.	STATISTICAL SUBFACTORS (3σ)	
	2) DECAY HEAT UNCERTAINTY EFFECT	1.20*
	3) PRESSURE DROP CALCULATIONAL UNCERTAINTY EFFECT	1.20*
	4) POWER LEVEL MEASUREMENTS	1.08
	5) COOLANT PROPERTIES	1.01
	6) NUCLEAR POWER DISTRIBUTION	1.065

*These factors represent values used at the time when peak temperatures occur during the natural circulation transient.

inter- and intra-assembly flow and heat redistribution (Ref. 8) was accounted for via the CØBRA-WC/FØRE-2M linkage. This was done by comparing the sodium temperatures utilizing DEMØ/CØBRA-WC/FØRE-2M sequential computations with the same analysis which neglects the CØBRA-WC computations for $Q_{ex}(\chi,\tau)$ and $G(\chi,\tau)$. These results were then applied to the FØRE-2M temperature predictions as bias factors for the "56 Hr-NOM" and "UNC" cases (defined in Section 3). This is a conventional method of accounting for perturbations on a multi-parameter problem. For natural circulation test predictions, the aforementioned effects including the hot channel factors are not constant with time, due to transient effects. The actual time-dependent factors were used to obtain the 3σ uncertainty maximum temperatures. Table 1 presents values of these factors used at the time when the peak temperature occurs.

3. CASE DEFINITION AND BOUNDARY CONDITIONS

The boundary conditions of the reactor, such as initial power, inlet and outlet plenum temperatures (as a function of time), reactor inlet flow and the nozzle-to-nozzle pressure drop (as a function of time) are provided by the system code, DEMØ. Detailed description of CØBRA-WC modeling and output provided to the FØRE-2M code will be described in a separate report.

Since the 35% power/75% flow initial condition test is the first of the significant dynamic tests which are part of the FFTF Acceptance Test Procedure, the allowable decay heat after shutdown will most likely be minimized. This is accomplished by only allowing low irradiation histories (burnup) on the core prior to the initiation of the test. The minimum irradiation time is estimated to be 1 hour at 35% power, and the maximum irradiation time is estimated to be 56 hours at 35% power. Thus, these irradiation histories should bracket the actual irradiation time before shutdown. The effect of such variation in decay heat on the temperature response is significant in this case, and will be shown separately. As mentioned in Reference 9, this and some other boundary conditions used in the calculation are assumed conditions. It is likely that post-test analysis made with decay power based on actual pre-test irradiation history will be necessary. Other boundary conditions such as the pump coast-down characteristic and intermediate loop pony motor operation should also be accounted for in the post-test analyses.

The following cases are therefore analyzed:

- Case "1 Hr-NOM" - Nominal case with no uncertainties, but includes flow and heat redistribution with only 1 hour irradiation history at 35% core power. Initial base runs were performed with DEMØ, CØBRA-WC, and FORE-2M; corrections were made for the recent shield/orifice assembly pressure loss coefficient data. System parameters from Reference 9 were utilized.

- Case "56 Hr-NOM" - Nominal case as in "1 Hr-NOM" except with 56-hour irradiation history at 35% core power. Runs were performed with DEMØ and FØRE-2M; correction factors were applied for 1) inter and intra assembly flow and heat distribution and 2) shield/orifice assembly pressure loss coefficient. System parameters listed in Appendix A were utilized.

- Case "56 Hr-UNC" - Case with 3σ uncertainties including decay heat, ΔP characteristics, flow and heat redistribution for 56-hour irradiation history at 35% core power. Runs were made with DEMØ and FØRE-2M; correction factors were applied for 1) inter and intra assembly flow and heat distribution and 2) shield/orifice assembly pressure loss coefficient. System parameters listed in Appendix A and uncertainties listed in Table 1 were utilized.

* Case "1 Hr-UNC" - Same case as in "56 Hr-UNC" except with 1-hour irradiation history at 35% core power. Correction factors used in the "56 Hr-UNC" case were adjusted to account for 1-hour decay heat levels and applied to the "1 Hr-NOM" case.

* Case "CRBRP/PRE-NCVP" - Current CRBRP conservative design approach (which neglects the effects of inter- and intra-assembly heat and flow redistribution and combines the hot channel factors conservatively) for 1-hour and 56-hour irradiation history at 35% core power. DEMØ and FØRE-2M runs were made utilizing Reference 9 system parameters for the 1 hour case and Appendix A system parameters for the 56 hour case.

Shutdown power data for the 1-hour irradiation cases are provided in Reference 9. The correspondence data for the 56-hour cases are given in Appendix A.

As shown in Figure 1, the FOTA is simulated by the CØBRA-WC code by a 37-channel model. For the Row 2 FOTA, channels 9, 10 and 15 are of the most interest, with channel 15 representing the hottest channel in this assembly with a very nearly flat power profile. Because of the nuclear flux distribution and heat transfer to the surrounding assembly of the Row 6 FOTA, channels 8, 13 and 15 are of most interest, with channel 13 representing the hottest channel.

Other initial conditions are (Ref. 9):

* Inlet temperature: 631°F
* Nozzle-to-nozzle pressure drop: 67.2 psid
* Initial core flow at 10,082 gpm
* Primary hot leg temperature at 750°F
* Secondary loop flow at 75 $\pm$1% of 13,200 gpm
* Secondary cold leg temperature at 602°F
* All six heat transport system (HTS) pony motors de-energized (subsequent to analyses reported here, current FFTF plans call for one intermediate pony motor to be operating).

4. RESULTS OF CALCULATIONS

Figures 2 through 5 present the calculation results in terms of coolant temperatures at the axial level where maximum temperatures occur which corresponds to the top of the active fuel section of the Row 2 and Row 6 FOTA's (applicable to two T/C locations in the FTR, as shown in Figure 1). For each FOTA, there are six cases analyzed. The "1 Hr-NOM", "56 Hr-NOM" cases (defined in Section 3) represent the nominal case sodium temperatures. The actual temperature in these channels would be expected to correspond with these two curves, depending upon the irradiation history. Also shown in Figure 2 are the results of the initial base run described in Section 2. Due to uncertainties, as discussed in Section 2, two additional curves are presented in Figures 2 through 5 representing the "1 Hr-UNC" and "56 Hr-UNC" cases defined in Section 3. The actual temperatures at the specified locations should not exceed these two curves representing (3σ) uncertainty cases. Thermocouple measurement uncertainties and time delays are not included in the predictions. Thus, the acceptance criterion is that if the maximum measured temperature at specified channels (corrected for above instrument effects and actual boundary conditions) is less than the curve, it would follow that the DEMØ/CØBRA-WC/FØRE-2M calculations made with design data conservatively envelope the temperature response of the natural circulation test. Also shown in Figures 2 through 5 is the maximum temperature based on the current CRBRP design method (CRBRP/PRE-NCVP). This method uses DEMØ calculated reactor flows based on maximum system ΔP's and maximum decay heat, does not consider the benefits of core inter- and intra-assembly flow and heat redistribution, and combines the hot channel factors conservatively.

5. CONCLUSIONS

The following conclusions can be made from the results obtained in this report:

a) The transients of the cases studied are very mild since the expected (nominal) peak transient ΔT's are less than 55°F higher than the initial steady state temperature rises as shown in Figures 3 and 5 for two different FOTA's and the maximum irradiation history.

b) The accuracy of the predictions is expressed in terms of hot channel factors as conventionally used in reactor design. By considering 3σ uncertainties statistically, the maximum temperatures of these two FOTA's are also shown in Figures 2 through 5. The measured temperatures in the respective FOTA's should therefore not exceed "1 Hr-UNC" or "56 Hr-UNC" curves with a 99.9% confidence level depending on the irradiation history of the core before shutdown. The "56 Hr-UNC" case yields the highest temperature.

c) The CRBRP/PRE-NCVP approach predictions are very conservative with respect to the nominal temperatures of the FFTF FOTA under natural circulation conditions.

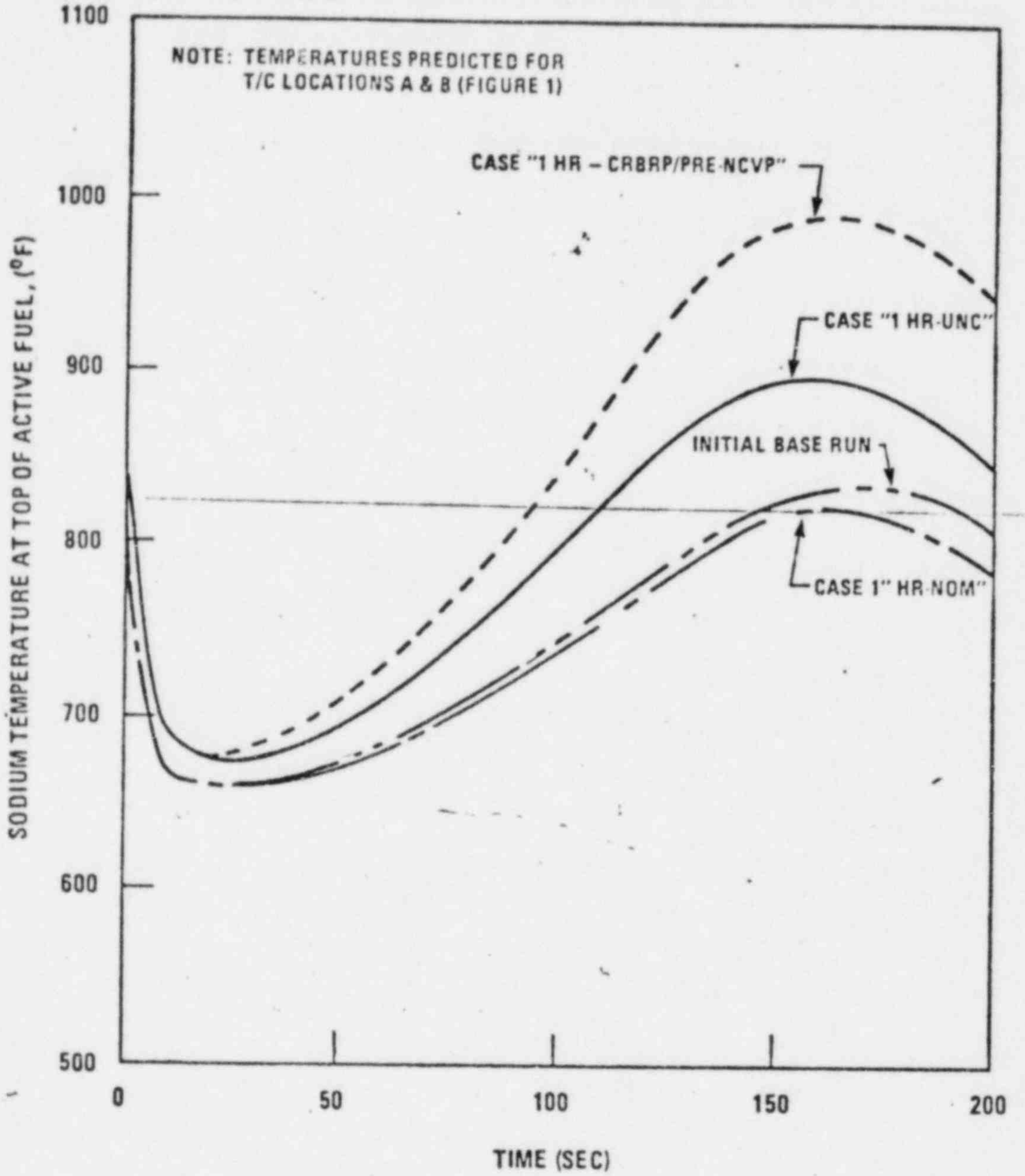


Figure 2. Sodium Temperatures at Top of Active Core Axial Position for Row 2 FOTA with 1 Hour Irradiation History (Test Initiated from 35% Power/75% Flow)

5005-1

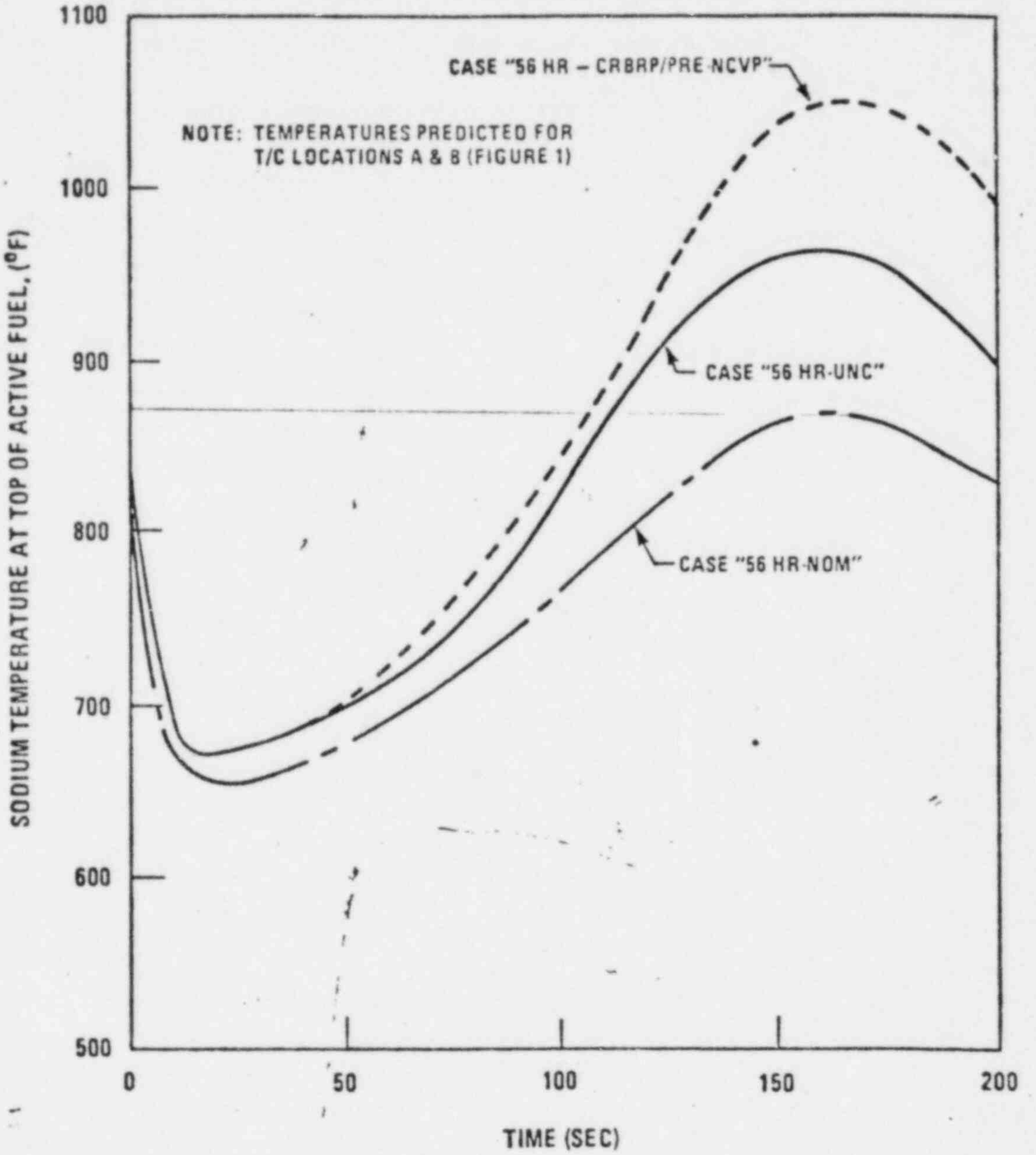


Figure 3. Sodium Temperatures at Top of Active Core Axial Position for Row 2 FOTA with 56 Hour Irradiation History (Test Initiated from 35% Power/75% Flow)

5005-2

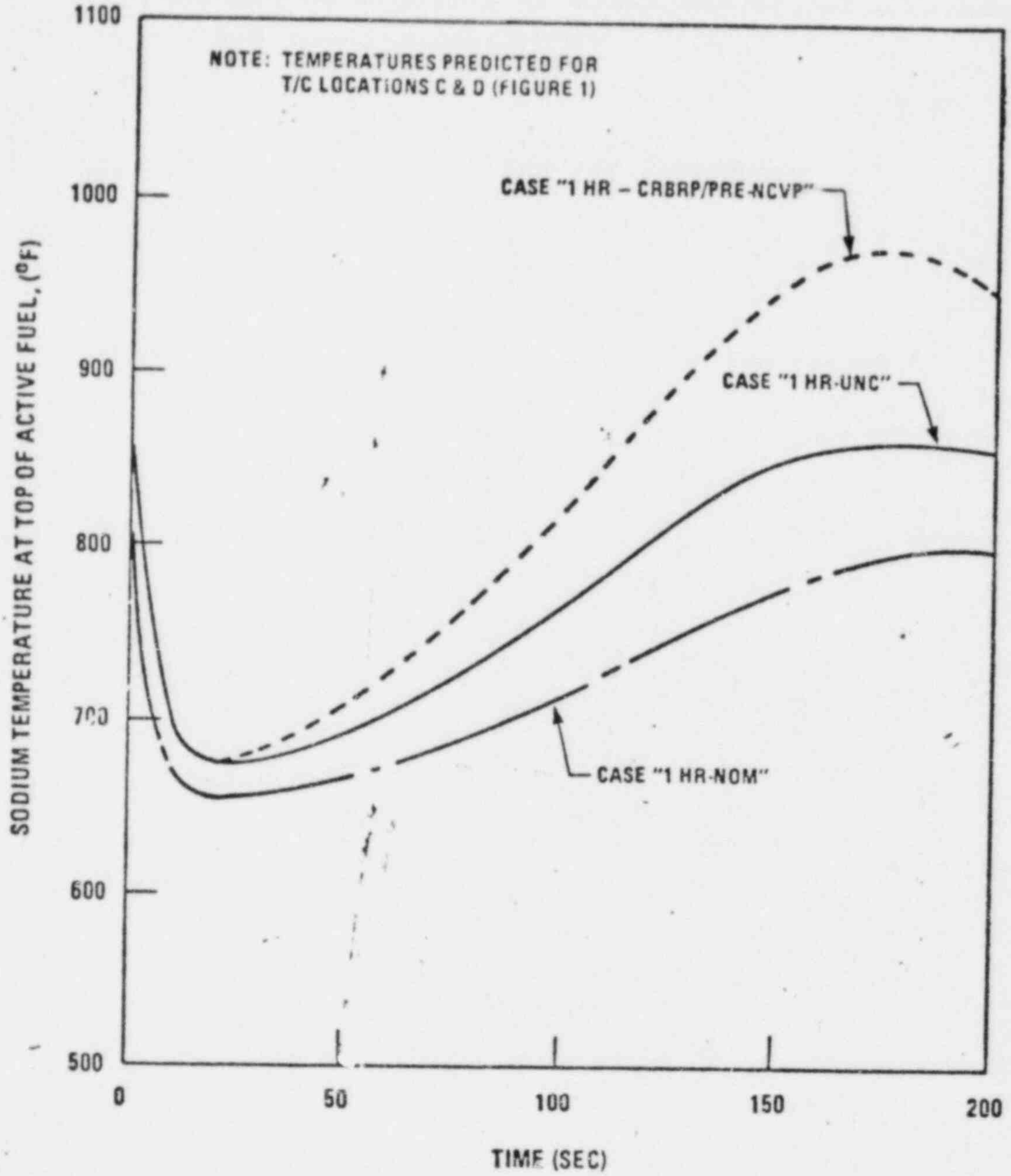


Figure 4. Sodium Temperatures at Top of Active Core Axial Position for Row 6 FOTA with 1 Hour Irradiation History (Test Initiated from 35% Power/75% Flow)

5005-3

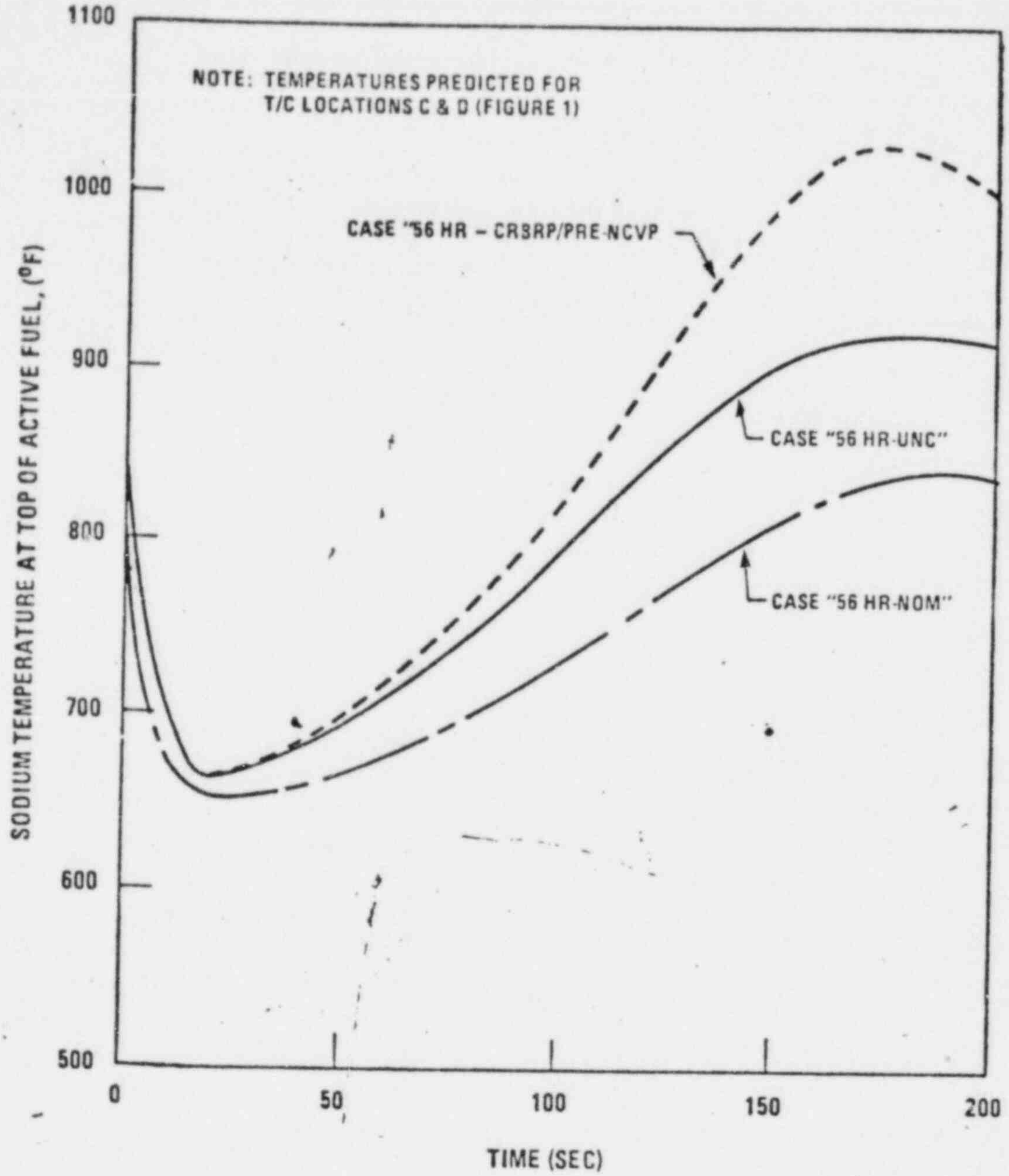


Figure 5. Sodium Temperatures at Top of Active Core Axial Position for Row 6 FOTA with 56 Hour Irradiation History (Test Initiated from 35% Power/75% Flow)

5005-4

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12. "Verification of Natural Circulation in Clinch River Breeder Reactor," transmitted to NRC via letter dated June 21, 1976.

APPENDIX A-A

DESCRIPTION OF FTR CORE CONFIGURATION AND SELECTED INPUT DATA

The schematic diagram of FTR is shown in Figure A-1. The core map is shown in Figure A-2. Approximately 1/6 of the core is modeled in the CØBRA-WC code, while FØRE-2M uses an annular model of a single rod in the assembly (FOTA's).

Each FOTA is a 40-ft. long core assembly, consisting of a lower 12-ft. in-core fueled section and an upper 28-ft. instrument stalk section. The lower section is essentially the same as a regular core driver except for the addition of pin bundle thermocouples in the FOTA. The locations of these two FOTA's within the core are shown in Figure A-2. The Row 2 FOTA is surrounded by 6 driver assemblies at two of its faces. Of these two assemblies, inter-assembly heat transfer should be greater for the Row 6 FOTA.

Table A-1 presents the 56-hour decay heat data obtained from HEDL.

Table A-2 indicates typical results from DEMØ for the transient reactor inlet nozzle flow for the various conditions evaluated. This information was used in performing the analyses for Figures 2, 3, 4 and 5 as described in Section 2 of this report.

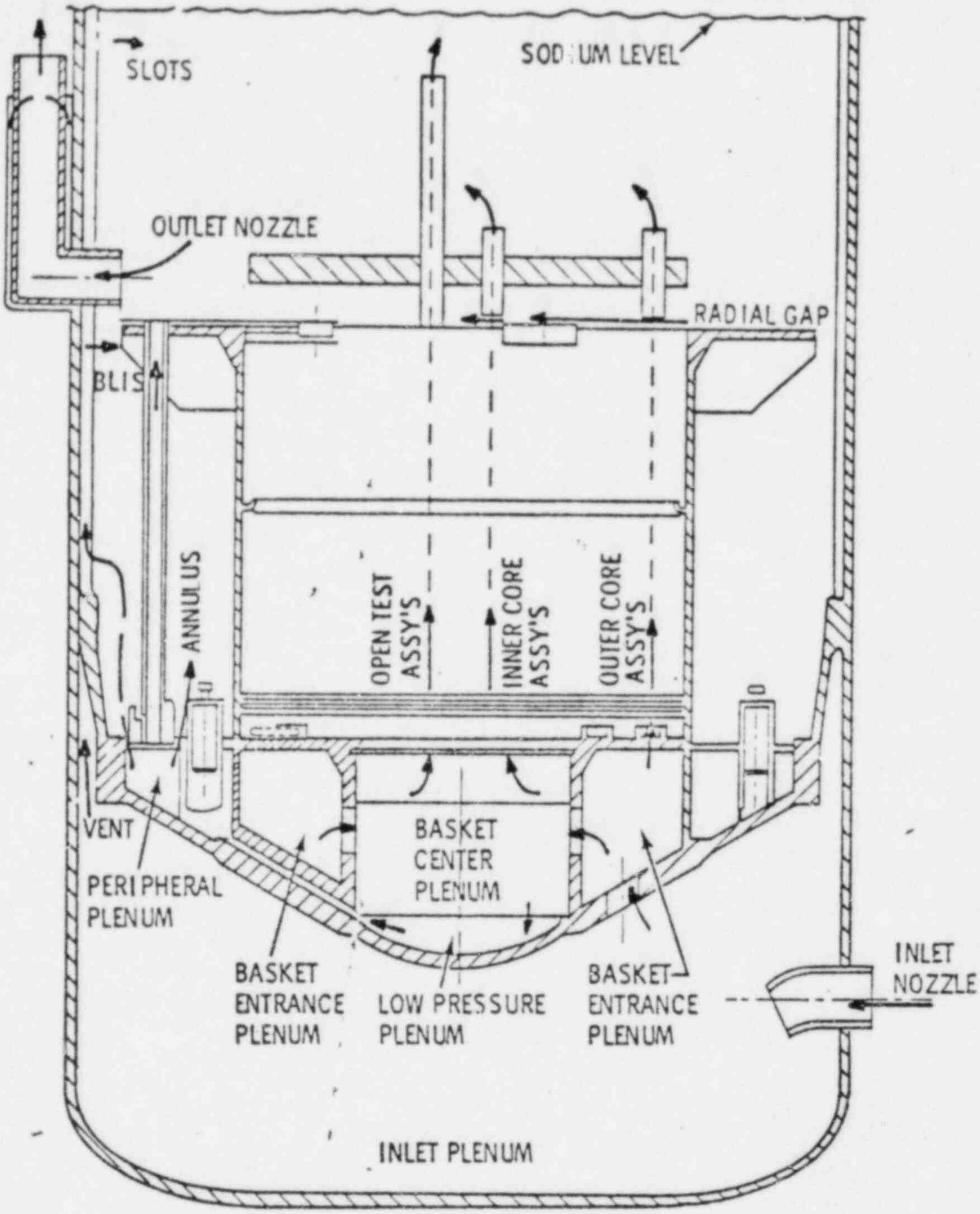
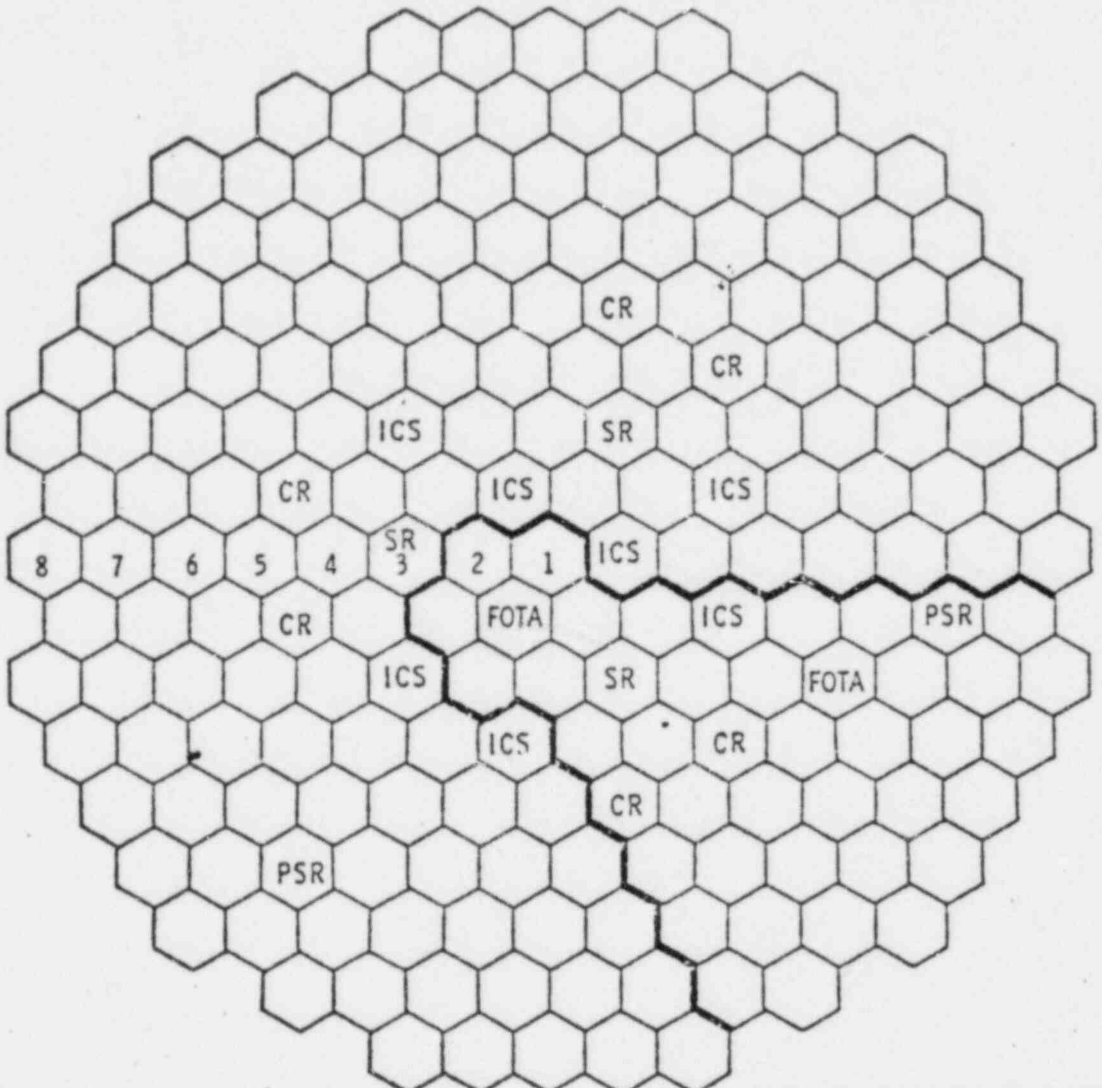


FIGURE A-1

FFTF REACTOR VESSEL ILLUSTRATING CØBRA-WC MODELED FLOW PATHS



FOTA - FUELED OPEN TEST ASSEMBLY
CR - CONTROL ROD
SR - SAFETY ROD
PSR - PERIPHERAL SAFETY ROD
ICS - IN-CORE SHIM
UNMARKED, ROWS 1 - 6 (DRIVERS)
UNMARKED, ROWS 7 - 9 (REFLECTORS)

FIGURE A-2

FFTF CORE MAP OUTLINING THE SECTOR MODELED WITH CØBRA-WC

TABLE A-1

56-HOUR DECAY HEAT DATA*

TIME (SEC.)	FUEL DECAY**	NON-FUEL DECAY**
0.0	0.05000	0.02147
1.0	0.05000	0.02147
3.0	0.04514	0.01958
6.1	0.04032	0.01769
12.0	0.03692	0.01634
24.0	0.03315	0.01486
42.0	0.02994	0.01359
60.6	0.02784	0.01277
78.8	0.0263	0.01215
97.0	0.0251	0.01168
115.2	0.02412	0.01129
133.3	0.02332	0.01097
151.5	0.02263	0.01069
169.7	0.02204	0.01046
187.9	0.02153	0.01025
206.1	0.0210	0.01004

* Delayed neutron power same as for 1-hour case from Reference 9.

** Fraction of steady state reference power; for fuel assemblies the reference power is 135.4 MW and for non-fuel assemblies, 13.2 MW.

TABLE A-2

TYPICAL DEMØ OUTPUT DATA FOR FIGURES 2,3,4 AND 5 ANALYSES

TIME (SEC.)	DEMØ REACTOR INLET NOZZLE FLOW, (lb/sec)				
	RUN 1	RUN 2	RUN 3	RUN 4	RUN 5
0.0	3668.0	3668.0	3668.0	3668.0	3668.0
10.0	1584.7	1584.6	1379.0	1584.6	1581.5
60.0	256.96	259.6	161.3	260.4	266.2
100.0	50.6	55.1	44.0	59.2	58.2
155.0	55.7	62.1	54.3	68.2	64.8
200.0	65.2	72.7	63.0	79.2	75.3

- Run 1: 1-hour nominal conditions;
- Run 2: 56-hour nominal conditions;
- Run 3: Same as Run 2, but with ΔP uncertainty;
- Run 4: Same as Run 2, but with decay heat uncertainty;
- Run 5: Same as Run 2, but with recent FFTF shield/orifice assembly pressure loss coefficient data from HEDL.

NOTE: Core inlet temperature can be maintained at 631°F for the first 200 seconds based on above DEMØ studies.

APPENDIX B

Pre-Test Prediction of FFTF Natural Circulation Tests

(Letter PS:81:080, R. L. Copeland, CRBRP P/O
to D. G. Eisenhut, NRC, dated February 26, 1981)

Docket No. 50-537

(Attachment WARD-D-0281, "CRBRP Natural Circulation
Verification Program Pre-Test Predictions for FFTF
Natural Circulation Tests Initiated at 100% and
75% Power/Flow Conditions")

APPENDIX B

Department of Energy
Clinch River Breeder Reactor
Plant Project Office
P.O. Box U
Oak Ridge, Tennessee 37830

Docket No. 50-537

February 26, 1981 + with attachments

MAR 02 1981

Mr. Darrell G. Eisenhut, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Eisenhut:

PRE-TEST PREDICTION OF FFTF NATURAL CIRCULATION TESTS

References: (1) Letter PS:80:359, R. L. Copeland to D. G. Eisenhut, "Pre-Test Prediction of FFTF Natural Circulation," dated November 28, 1980.

(2) Letter, S:L:1195, A. R. Buhl to R. S. Boyd, "Natural Circulation Decay Heat Removal Verification Plan," dated June 21, 1976.

This letter transmits information concerning the pre-test prediction of the results of the natural circulation tests scheduled for early March 1981 at the Fast Flux Test Facility (FFTF). These pre-test predictions and those previously transmitted in Reference (1) are being supplied to establish a basis for demonstrating that the methodology and the computer codes, as discussed in Reference (2), are capable to reasonably characterize the phenomena controlling natural circulation. These pre-test predictions are provided to clearly document that the data and methodology which will be used in the post-test analyses were selected prior to the test.

These pre-test predictions assume reactor scram and trip of the sodium circulating pumps from pre-existing conditions of 1) 100% of full power and 100% of full flow, and 2) 75% of full power and 75% of full flow. Pre-test predictions utilizing CRBRP methodology and computer codes for tests from 35% power and 75% of flow were transmitted in Reference (1).

Following completion of all three natural circulation tests, post-test analyses will be performed utilizing the actual test boundary conditions. The pre-test predictions are, of necessity, based on boundary conditions (e.g., power history and heat sink temperature) selected by the analysts. It is unlikely that these boundary conditions will be duplicated during the actual tests. Accordingly, the post-test analyses will be used for comparison with experimental data. A report will be compiled to consolidate all of the pre-test predictions and post-test analyses and discuss any differences between those analytical results and the test data.

The pre-test predictions and post-test analyses will be considered successful if they demonstrate 1) that the Project methodology provides a reasonable characterization of the phenomena controlling natural circulation flow and heat transfer in both the reactor and the heat transport loops and 2) that use of the methodology results in a conservative prediction of the critical plant parameters (e.g., core hot channel sodium temperature and heat transport system flow rates). Included with the predictions are acceptance boundary curves for the pre-test predictions. The methodology used to generate these acceptance curves are similar to those used in Reference (1) and will be the same methodology to generate acceptance boundary curves for the post-test analyses. The prediction methodology will be considered acceptable if the measured flows for the tests are equal to or greater than the acceptance boundary curves.

Included as an attachment to this letter is "CRBRP Natural Circulation Verification Program Pre-Test Predictions for FFTF Natural Circulation Tests Initiated at 100% and 75% Power/Flow Conditions," WARD-D-0281, dated February 1981. Appendices A, B, and C, of this report, which contain an archival listing of data used by the computer codes, are being provided to the addressee only.

We would be pleased to answer any questions you may have on the attachments or to discuss the analyses at your convenience.

Sincerely,

Raymond L. Copeland

Raymond L. Copeland
Acting Assistant Director
for Public Safety

PS:81:080

Attachments:
As stated

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WARD-D-0281

CRBRP NATURAL CIRCULATION VERIFICATION PROGRAM PRE-TEST PREDICTIONS FOR FFTF NATURAL CIRCULATION TESTS INITIATED AT 100% AND 75% POWER/FLOW CONDITIONS

by

A. Cheung
R. D. Coffield
K. D. Daschke
Y. S. Tang

FEBRUARY 1981

APPROVED: R. A. Markley
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Performance Analysis

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1. SUMMARY

Pre-test predictions of the thermal and hydraulic response were made for the FFTF natural circulation tests initiated at 100% power/100% flow and 75% power/75% flow conditions. These predictions were performed under the Natural Circulation Verification Program (NCVP) to provide the basis for the verification of the analytical tools used in making such predictions. Computer codes utilized in these predictions are DEMØ, CØBRA-WC and FØRE-2M. Pre-test predictions of sodium temperatures are presented at the top of the active fuel section for the Fueled Open Test Assemblies (FOTA's) for these tests. Also presented are predictions of primary and secondary loop flow rates, and hot and cold leg temperature at the locations where they will be measured. The acceptance criterion for the DEMØ code for the system natural circulation analysis is shown in terms of the primary loop flow using a combination of pressure drop and decay power uncertainties which give the minimum flow for the natural circulation transient. If the actual measured flow is greater than the acceptance limit, it would follow that DEMØ calculations for the minimum flow design case (Case V in this report) conservatively envelope the actual natural circulation flow, and are therefore acceptable for the DEMØ flow predictions. The acceptance criterion for all three codes for the analyses is shown by the sodium temperature responses in the FOTA's. If the maximum measured temperature at specified channels (corrected for instrument effects and actual boundary conditions) is less than the predicted hot channel temperature (with known uncertainties accounted for), it would follow that the DEMØ/CØBRA-WC/FØRE-2M calculations made with the design data conservatively envelope the temperature response of the natural circulation test. The model and methodology would therefore be acceptable for such design transient predictions.

2. INTRODUCTION

The purpose of this report is to present pre-test predictions of the thermal and hydraulic responses of the FFTF plant including the fueled open test assemblies (FOTA's) for the FFTF natural circulation plant startup tests initiated at 100% power/flow and 75% power/flow conditions. These are the last two and most severe tests in the series of natural circulation tests to be performed in the FFTF during acceptance testing.

The FFTF tests encompass the transition into natural circulation cooling from a series of steady state initial conditions. All the tests will be initiated with a reactor scram and main coolant pump trip. The test series will include (Ref. 1):

a) A transition to natural circulation in the primary loop from 5% reactor power, 75% primary loop flow. The secondary loop pumps will coast down to 10% speed at which time pony motors engage and provide 10% secondary loop forced flow.

b) A transition to natural circulation in both primary and secondary loops from 35% reactor power and 75% flow.(*)

c) A transition to natural circulation in both primary and secondary loops from 75% reactor power and flow.

d) A transition to natural circulation in both primary and secondary loops from 100% reactor power and flow.

(*)One of the secondary loop pony motors was engaged during the test.

Pre-test predictions for tests initiated at 35% power/75% flow were previously reported in Reference 2. All predictions were performed under the Natural Circulation Verification Program (NCVP) using the CRBRP methodology. The objective of this verification program is to analytically and experimentally verify numerical models for the description of the thermal-hydraulic behavior of LMFBR plants when making transition to or operating in the natural circulation mode.

A system of three computer codes has been developed and is used in sequence: i.e., DEMØ, for plant-wide analyses (Ref. 3); CØBRA-WC, for core system analyses (Ref. 4); and FØRE-2M, for localized core hot rod analyses (Ref. 5). A brief summary of each of these codes follows:

A. DEMØ (Ref. 3)

DEMØ, a system-wide code, predicts the coupled performance of the reactor, heat transfer systems and the steam generating system (including piping and plena heat capacity effects). To provide accurate results for the plant as a whole, yet maintain the code as an amenable tool with regard to computer storage requirements and running time, localized phenomena (which do not affect the system as a whole) are not given detailed modeling. An example of this is an item such as local flow/heat redistribution between and within core assemblies at low flow. The FFTF version of DEMØ employs a three region model (fuel, non-fuel and bypass) for the core. The code substitutes a dump heat exchanger for the steam generator. To consider these localized phemomena with the required resolution takes a separate computer code itself (i.e., CØBRA-WC was selected for this purpose). In general, DEMØ provides a prediction of the overall system state variables such as net flow through the reactor and bulk temperatures entering and exiting the core.

B. CØBRA-WC (Ref. 4)

The CØBRA-WC code, which accounts for core inter- and intra-assembly flow and heat redistribution, predicts the boundary conditions for a peak rod in the fuel and blanket assemblies given the reactor boundary conditions such as total reactor flow pressure drop and core inlet temperature from DEMØ, individual core assembly powers, and individual core assembly thermal-hydraulic characteristics. Analyses of this type phenomena under natural convection cooling requires detailed core/reactor modeling because of the strong interaction between the fuel assemblies, blanket assemblies, control assemblies, plus other core regions and bypass flows which all act as highly coupled parallel flow paths with heat transfer between them. Typical modeling details of the core assemblies is given in Section 3.

C. FØRE-2M (Ref. 5)

Given the localized peak channel flows and heat interchange as a function of time, FØRE-2M predicts the hot channel coolant temperature for natural circulation which includes considering uncertainties in the predictions. This hot channel coolant temperature is used as the basis for determining the acceptability of natural circulation for decay heat removal of LMFBR's. To have a verified hot rod prediction, one must properly account for such effects as:

a) fuel restructuring; b) fuel-cladding gap conductance; c) stored heat in the rod; d) statistical significance of physics and engineering uncertainties in power, flow, dimensions, properties, etc.; e) localized hot spots on the rod due to the wire wrap or pellet eccentricity; f) localized decay heat variations; and g) localized rod power variations in the assembly.

Since FØRE-2M is used for core hot rod analysis with input from DEMØ and CØBRA-WC, extreme detail in nodalization can be used to assure accurate temperature predictions. This level of detail cannot be part of the CØBRA and DEMØ predictions since the storage requirements and computer running times would make the codes impractical. [Note: In addition, the code provides a detailed nuclear physics prediction for the prompt and delayed neutron fission power as a function of time from shutdown with the capability of considering Doppler, sodium density, and radial/axial core expansion feedback plus the effects of control system worth and PPS functions. This information is used to verify the DEMØ prediction of core power versus time].

During the transition to and operation in the natural convection cooling mode, the effect of an increasing power-to-flow ratio approaching or exceeding that of steady state can be experienced. Consequently, core temperatures increase and natural convection phenomena such as inter- and intra-assembly flow redistribution become significant once low flow conditions are reached. In the CRBR or FTR, the core thermal head becomes increasingly significant relative to the form and friction losses across the core for flows below 5% of full flow. Coupled with the flow redistribution, significant heat redistribution on an inter- and intra-assembly basis occurs throughout the core due to radial temperature differentials and an increased flow transport time. Both of these effects (i.e., natural convection flow and heat redistribution) have been found to significantly reduce maximum core temperatures as demonstrated by the EBR-II natural circulation experiments (Ref. 6) and Brookhaven investigations (Ref. 7).

The pre-test predictions of the tests consists of performing nominal (best-estimate) calculations plus uncertainty evaluations of the nominal calculation. The post-test analyses will assess the effects of actual test conditions as opposed to those used in the analyses, effects of measured uncertainties and consistency of the models in DEMØ, CØBRA-WC and FØRE-2M with test results. For instance, two important boundary conditions employed in the pre-test predictions are power history prior to scram (which affects the decay powers) and the DHX sodium outlet temperature.

In addition, the post-test analyses will include an assessment of the accuracy of the data used in the pre-test predictions. The objective will be to determine, to the extent possible with FFTF instrumentation, how much of the difference between the best-estimate calculations and the measurements is due to uncertainties in the data used in the predictions.

3. ANALYTICAL APPROACH

Figure 3-1 shows the interaction between three codes used in the CRBRP natural circulation analysis procedure. The first code (DEMØ) provides core delayed neutron powers, dynamic average region temperatures and inlet flow, and average core pressure drops. The last two codes (CØBRA-WC and FØRE-2M) provide the two-stage calculational approach for the core analyses under natural circulation conditions. Utilizing DEMØ boundary conditions, a detailed whole-core flow

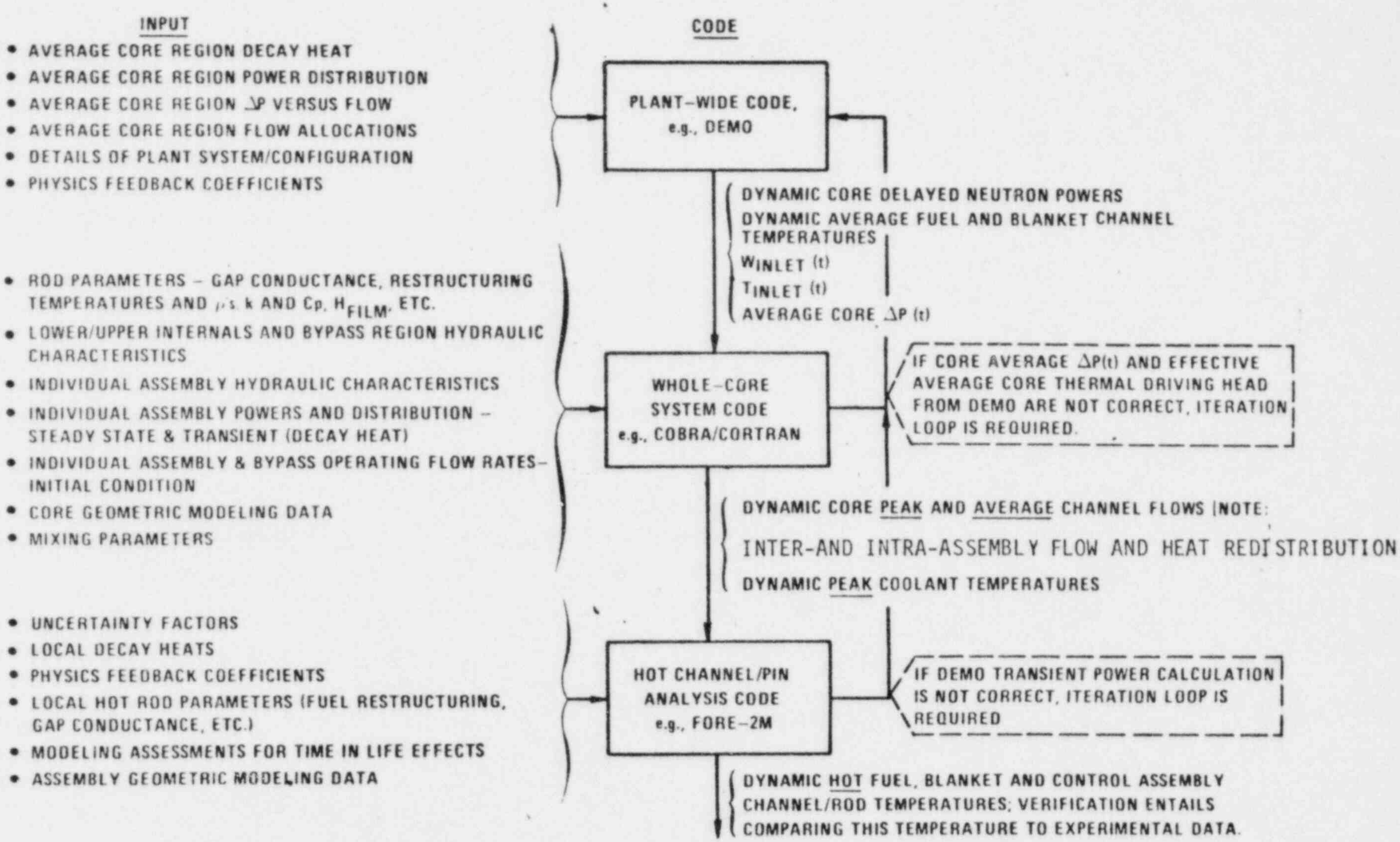

FIGURE 3-1: LMFBR NCVP ANALYSIS PROCEDURE

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and heat redistribution analyses of all the parallel core assemblies and bypass regions is first performed by the CØBRA-WC code. This is done for a sector of symmetry of the reactor core including a bypass flow. By necessity, the calculational matrix or nodal representation, especially distant from the area of interest, is less detailed than that in the area of interest, i.e., FOTA. Two step calculations are thus used with the CØBRA-WC code. The first calculation utilizes a single-channel model for each core assembly in the sector of symmetry of the reactor core. This yields assembly flow information and the temporal pressure drop across the fuel assembly for inputting data to a second calculation which contains more detailed FOTA analysis. The latter calculation uses a seven-assembly cluster where detailed inter-assembly heat conduction and intra-assembly flow and heat redistribution may be simulated. Figures 3-2 and 3-3 show the model of such a cluster for the FOTA in Rows 2 and 6. A group of rods rather than individual subchannels are modeled. The central assembly (FOTA) is simulated by a 37-channel model while the surrounding assemblies are simulated by 19-channel models. The relative position of rods and CØBRA-WC channel representation is shown in Figure 3-4. Each channel shown is represented by one rod with power and flow conditions corresponding to the average for all rods contained in it. For the interior, hexagonal shaped regions, this average rod would represent a total of 12 rods. The results of this analysis provide the effect of reactor inter- and intra-assembly flow and heat redistribution on the temperature response. The data are then used for a detailed analysis on a single, "hot rod" using the FØRE-2M code. These latter analyses include effects of localized phenomena and uncertainties in nuclear/thermal-hydraulic/mechanical data.

A linkage between the CØBRA-WC and FØRE-2M codes has been developed and verified to incorporate the inter- and intra-assembly phenomena into the localized hot rod natural circulation analyses (Ref. 8). For each axial node of the hot rod modeled in FØRE-2M, a heat balance is performed using the expression for the heat transferred to the coolant at that section, $Q_c(\chi,\tau)$ as

$$Q_c(\chi,\tau) = Q_r(\chi,\tau) + Q_{ex}(\chi,\tau) \tag{3-1}$$

where

$Q_r(\chi,\tau)$ = heat transferred from the rod surface at axial location χ and time τ;

$Q_{ex}(\chi,\tau)$ = coolant heat input or loss due to radial conduction and mixing heat transfer and flow redistribution to adjacent coolant channels; directly input from CØBRA-WC.

Coupled with this, the axial mass flow rate for each axial node $G(\chi,\tau)$ is also input from CØBRA-WC analyses. Boundary conditions for the CØBRA-WC (e.g., plenum-to-plenum pressure drop and coolant inlet temperature) are furnished by the plant-wide code, DEMØ for several cases: the "best estimate" or nominal case as well as cases with uncertainties. Likewise, a corresponding modeling of the core parallel flow network, with regard to pressure drop and decay heat uncertainties. can be used in the CØBRA-WC analyses for input to the FØRE-2M hot rod temperature predictions.

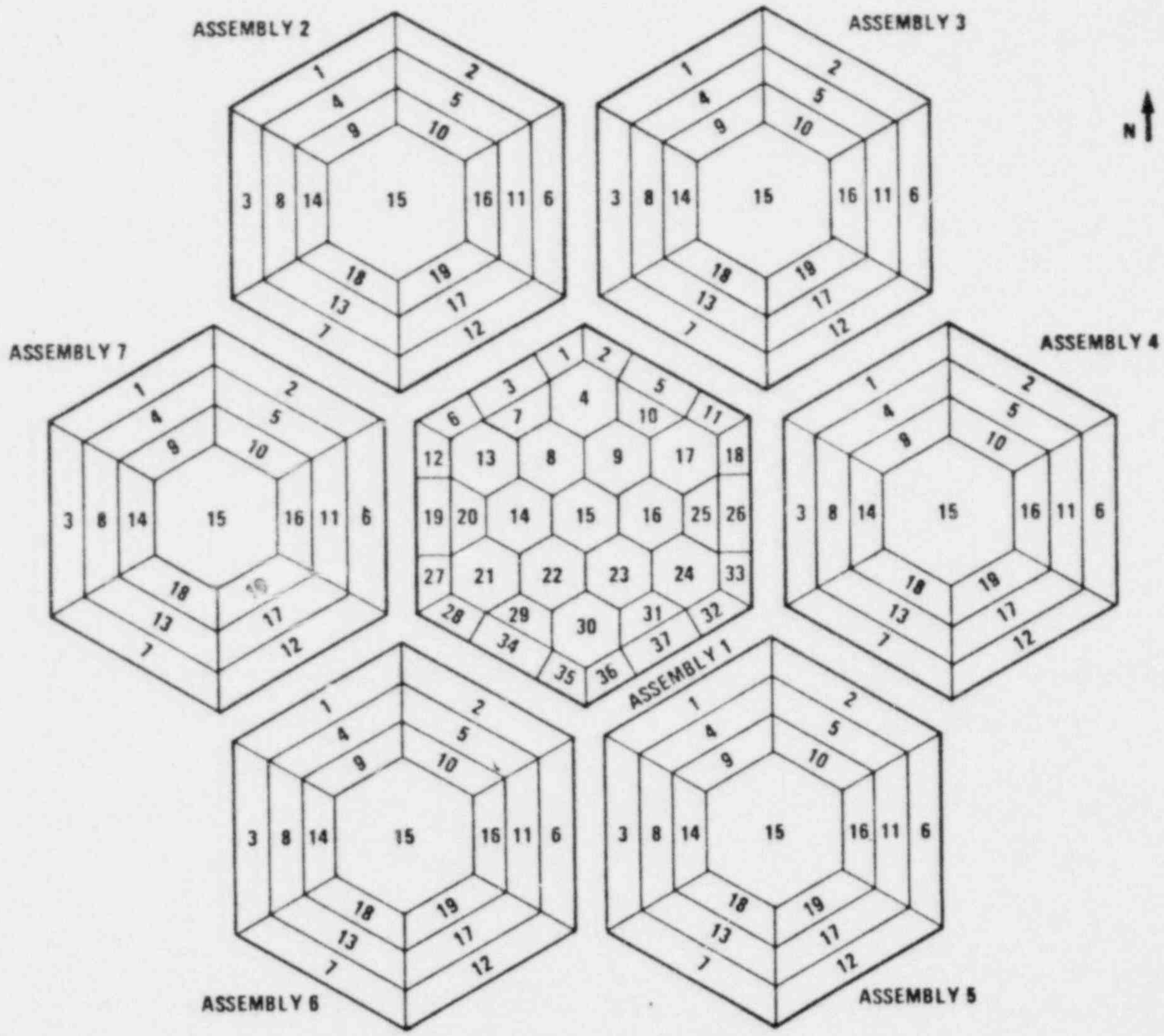


Figure 3-2. COBRA-WC Model of a 7-Assembly Cluster Surrounding Row 2 FOTA

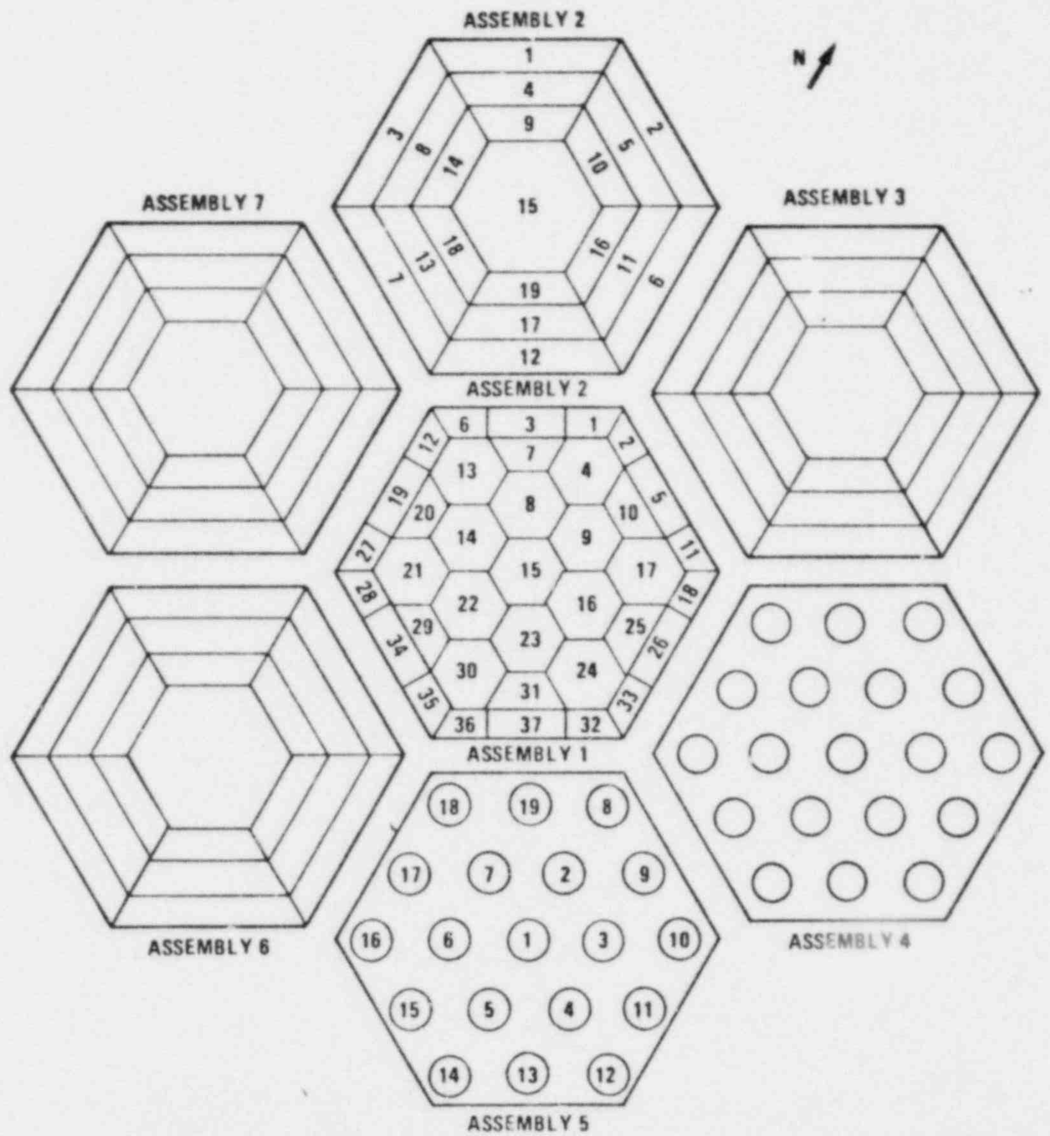
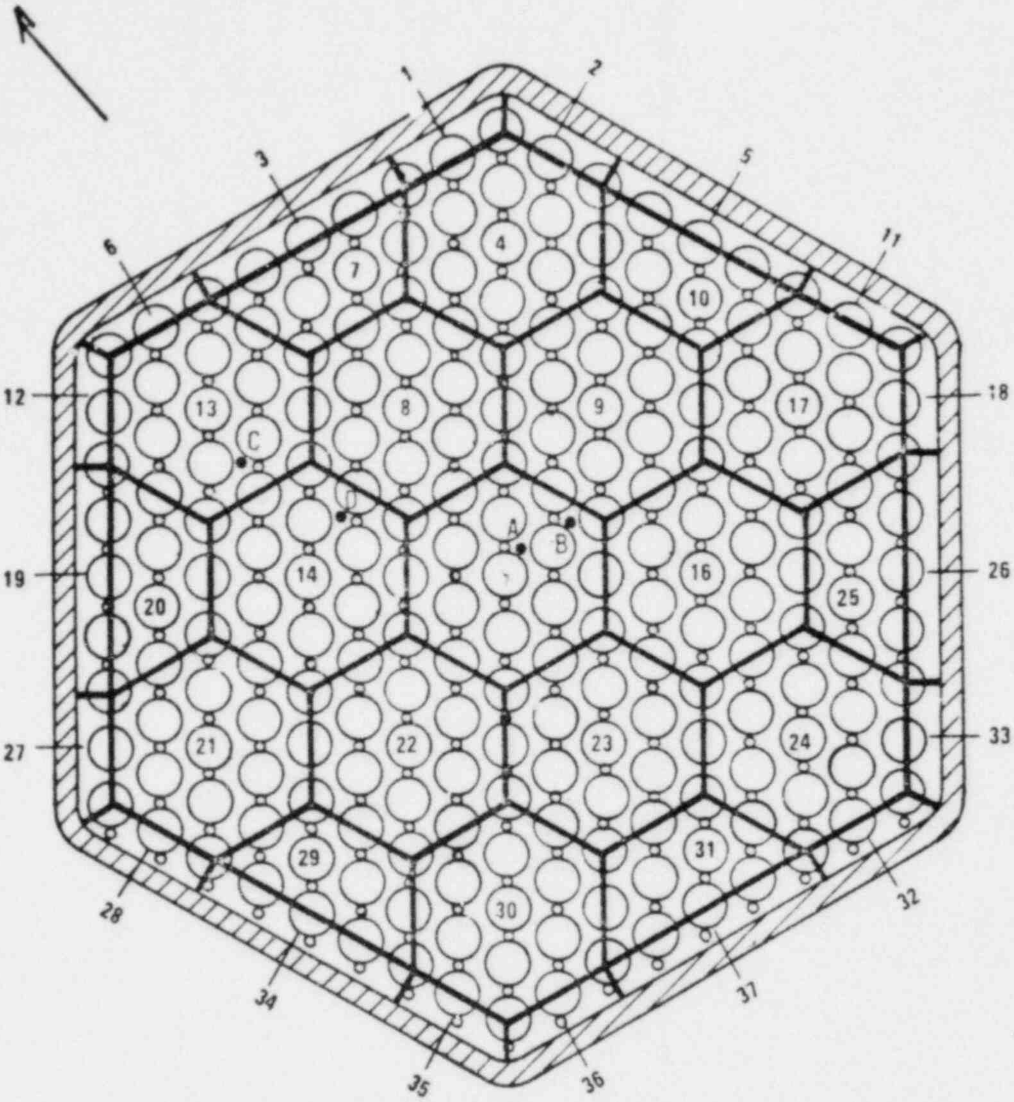


Figure 3-3. COBRA-WC Model of a 7-Assembly Cluster Surrounding Row 6 FOTA



Row 2 FOTA Location A, (Tx1014) Rod (9,9) at Top of Fuel
Row 2 FOTA Location B, (Tx1016) Rod (8,8) at Top of Fuel
Row 6 FOTA Location C, (Tx9018) Rod (4,4) at Top of Fuel
Row 6 FOTA Location D, (Tx9016) Rod (6,6) at Top of Fuel

FIGURE 3-4

TYPICAL CØBRA-WC 37 CHANNEL MODEL OF A 217 ROD ASSEMBLY

To evaluate the effects of core uncertainties, hot channel factors are used (Table 3-1). Similar to those used in steady state calculations, the direct and statistical type factors are conservatively applied by the semi-statistical method. For instance, the factor due to inlet flow uncertainty of -5% (Ref. 9) to the FOTA is conservatively assumed to occur. The statistical subfactors which are statistically combined include those due to uncertainties in the decay heat calculations ($\pm$25% of decay heat values which gives a $\pm$17% uncertainty in Row 6 FOTA sodium temperature at the time the peak temperature occurs), the pressure drop calculational uncertainties (specified in Reference 10) and the factors for power level measurement, nuclear power distribution and coolant property uncertainties. A more pessimistic compilation of uncertainties applicable to conservative design type analyses (not controlled experimental conditions) are described in Reference 8.

The effects of the first two statistical subfactors (i.e., decay heat and pressure drop uncertainties) are determined by comparing different cases with maximum variation of one of the quantities in calculations using all three codes. These different cases are listed in the next two sections. For natural circulation test predictions, the aforementioned effects are not constant with time due to transient effects. The actual time-dependent factors were used to obtain the 3σ uncertainty on maximum temperature. Table 3-1 presents values of these factors used at the time when the peak temperature occurs.

4. PRE-TEST PREDICTIONS FOR NATURAL CIRCULATION TEST INITIATED AT 100% POWER/FLOW CONDITIONS

4.1 Cases Analyzed for Predictions

Several cases are analyzed with the initial reactor condition of 100% power and 100% flow before shutdown. The maximum irradiation history of 25 hours full power is also assumed. The parameters that are varying in these different cases are the decay heat and system pressure drop (including core pressure drop) calculations. Thus, the effect on temperature response of these uncertainties is determined by comparing results of different cases analyzed:

- Case I - Nominal conditions, including nominal decay heat and system pressure drop, without any uncertainties. This case provides the best estimate of the expected plant and core performance during the transient.
- Case II - Same as Case I, except that high side "design" pressure drop uncertainties were applied to the entire plant system. The uncertainties assigned to each component or system were based on typical design data which is biased to include all high side pressure drop data.
- Case III - Same as Case I, except maximum decay power (125% of the nominal decay heat shown in Table 4.1) was assumed to account for calculational uncertainty on decay power.
- Case IV - Theproposed CRBRP design approach which is the case with maximum uncertainty. It assumes both high side "design" system pressure drop and maximum decay power (125% nominal decay heat). Other known hot channel subfactors such as factors for power level, nuclear calculational uncertainties, coolant properties and inlet flow maldistribution are also included in the FØRE-2M analysis. In addition, because this is used for design purposes, several more hot channel subfactors, such as power control band uncertainty, fissile fuel maldistribution, subchannel flow area and flow distribution calculational uncertainty are also included.

TABLE 3-1

HOT CHANNEL FACTORS APPLIED TO HOT CHANNEL TEMPERATURES IN FØRE-2M CALCULATIONS FOR N/C TESTS

A.	DIRECT SUBFACTORS	
	1) INLET FLOW UNCERTAINTY, F_1	1.05
B.	STATISTICAL SUBFACTORS (3σ)	
	2) PRESSURE DROP CALCULATIONAL UNCERTAINTY EFFECT, F_2	1.15*
	3) DECAY HEAT UNCERTAINTY EFFECT, F_3	1.17 (Row 6 FOTA)* 1.13 (Row 2 FOTA)*
	4) POWER LEVEL MEASUREMENTS, F_4	1.079
	5) COOLANT PROPERTIES, F_5	1.01
	6) NUCLEAR POWER DISTRIBUTION, F_6	1.065

*These factors represent values used at the time when peak temperatures occur during the natural circulation transient. They are determined by comparison of different cases in calculations using all three codes.

o Case V - Same as Case IV, except the minimum decay power (75% of the nominal decay power) was assumed. This case provides minimum natural circulation flow, and is therefore used for the acceptance criterion for the DEMØ code to predict the natural circulation flow. No temperature prediction of the FOTA's was made because it does not represent a worst condition.

4.2 DEMØ Analysis

Pre-test predictions for the FFTF transient natural circulat on tests from initial conditions of 100% power/100% flow; and 75% power/75% flow were initially made during the first quarter of 1980. Results of these predictions and the methodology used in the analyses were documented in Reference 10. Subsequent to the issuance of this report, two changes in data input were made to reflect: 1) new information obtained from pressure drop tests of the fuel assembly inlet no??le/shield block assembly; and 2) new expected power history for the tests. Bec use of these changes the pre-test predictions for the 100% power and the 75% power cases described in Reference 10 have been updated and the results are presented in Sections 4.3 and 5.3 of this report for the 100% power and the 75% power tests, respectively. With the exception of decay powers and reactor pressure drop correlations, the program and boundary conditions used to compute the results were the same as discussed in Reference 10 and will not be repeated here. Changes made to the program are discussed in the following sections.

The reactor vessel dynamic pressure drop correlations were determined by the same method outlined in Reference 10, i.e.:

$$\Delta P_{\substack{\text{Dynamic}\\ \text{DEMØ}}} = \Delta P_{\substack{\text{Total}\\ \text{CØBRA}}} - \Delta P_{\substack{\text{Static}\\ \text{DEMØ}}} \qquad (4\text{-}1)$$

where

$\Delta P_{\substack{\text{Total}\\ \text{CØBRA}}}$ = total reactor pressure drop calculated by CØBRA for given reactor flow;

$\Delta P_{\substack{\text{Static}\\ \text{DEMØ}}}$ = static reactor pressure drop as calculated by DEMØ for the same flow conditions; and

$\Delta P_{\substack{\text{Dynamic}\\ \text{DEMØ}}}$ = dynamic pressure drop to be used for determining the pressure drop correlations.

Using the new experimental data on pressure drop under low flow conditions through the fuel assembly inlet nozzle/shield blocks (Ref. 11), the total pressure drop ($\Delta P_{\text{Total CØBRA}}$) for a series of steady state calculations at various flows with the power-to-flow ratio near one were made. The dynamic pressure drop was then calculated from Equation 4-1. The resulting dynamic pressure drop correlations for the fuel assembly, the non-fuel assembly and bypass regions are shown in Figure 4-1, 4-2 and 4-3, respectively. Also shown in these figures are reactor pressure drop correlations with high side ("design") pressure drop uncertainties. These uncertainties were applied by increasing the best estimate reactor ΔP values by a factor dependent on the reactor flow. This factor varied from 1.1 at full flow to about 1.38 at 2% flow. These dynamic pressure drop correlations were used in both the 100% and 75% power pre-test predictions.

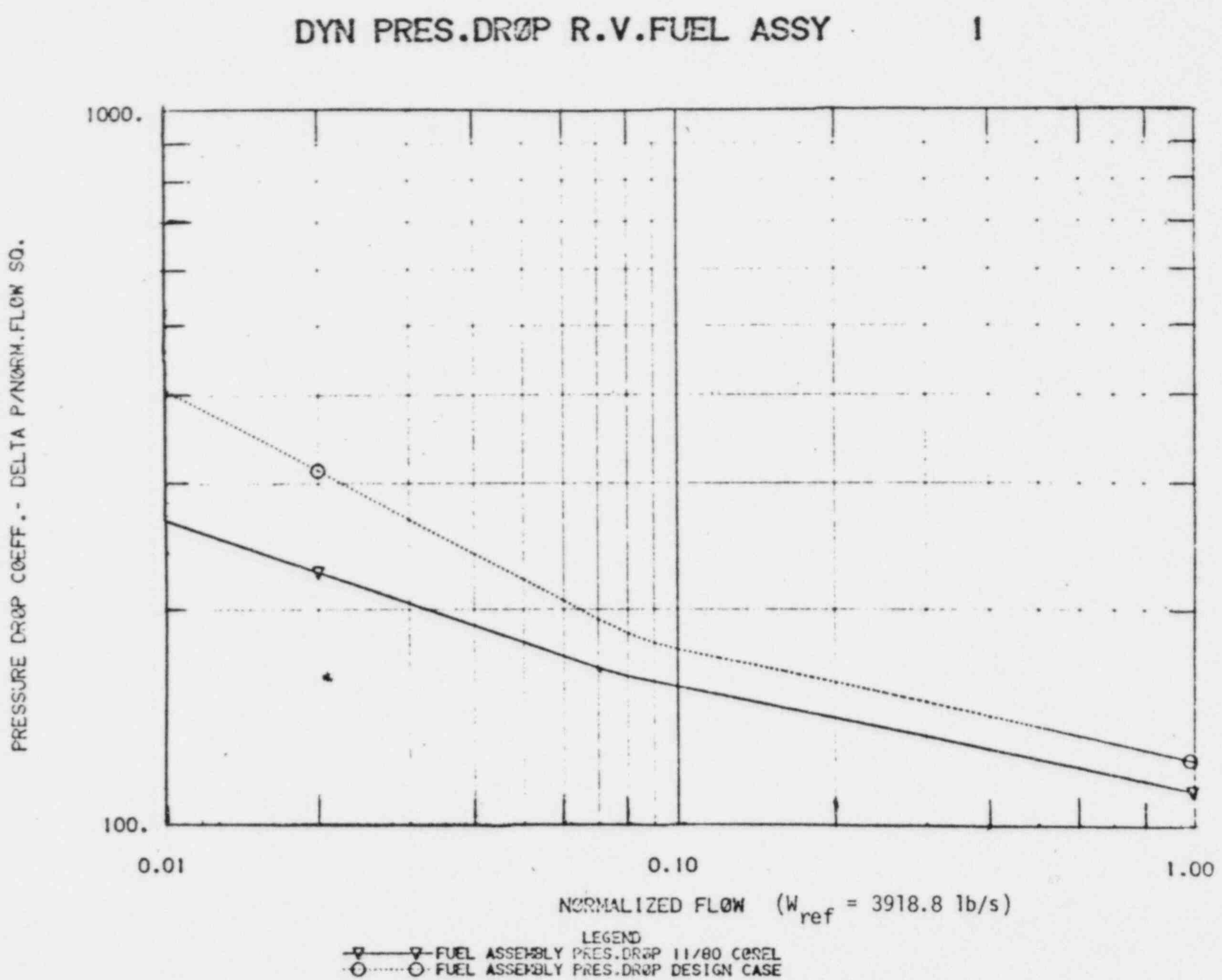


Figure 4-1 Fuel Assembly Group Dynamic Pressure Drop Correlations

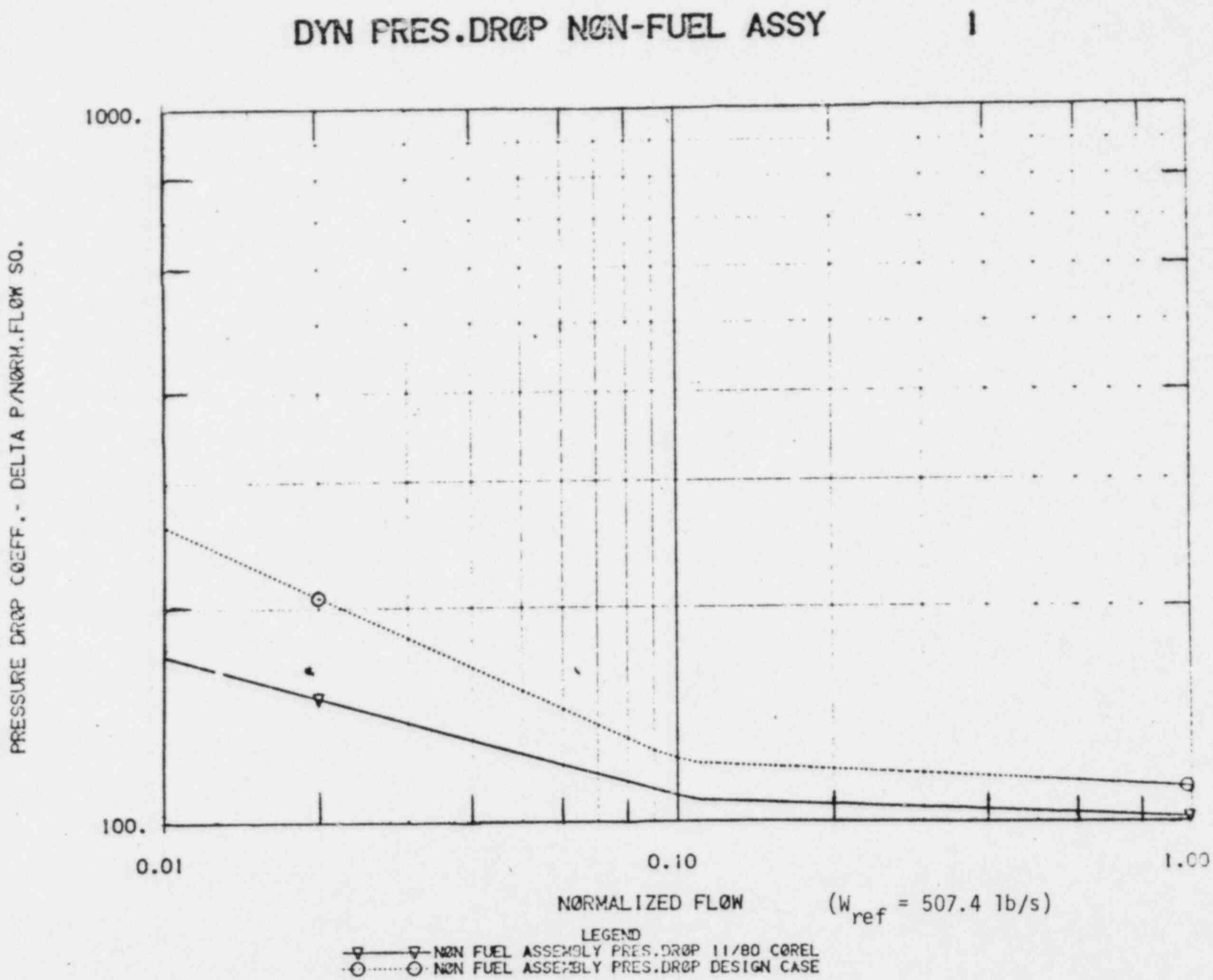


Figure 4-2 Non-Fuel Assembly Dynamic Pressure Drop Correlations

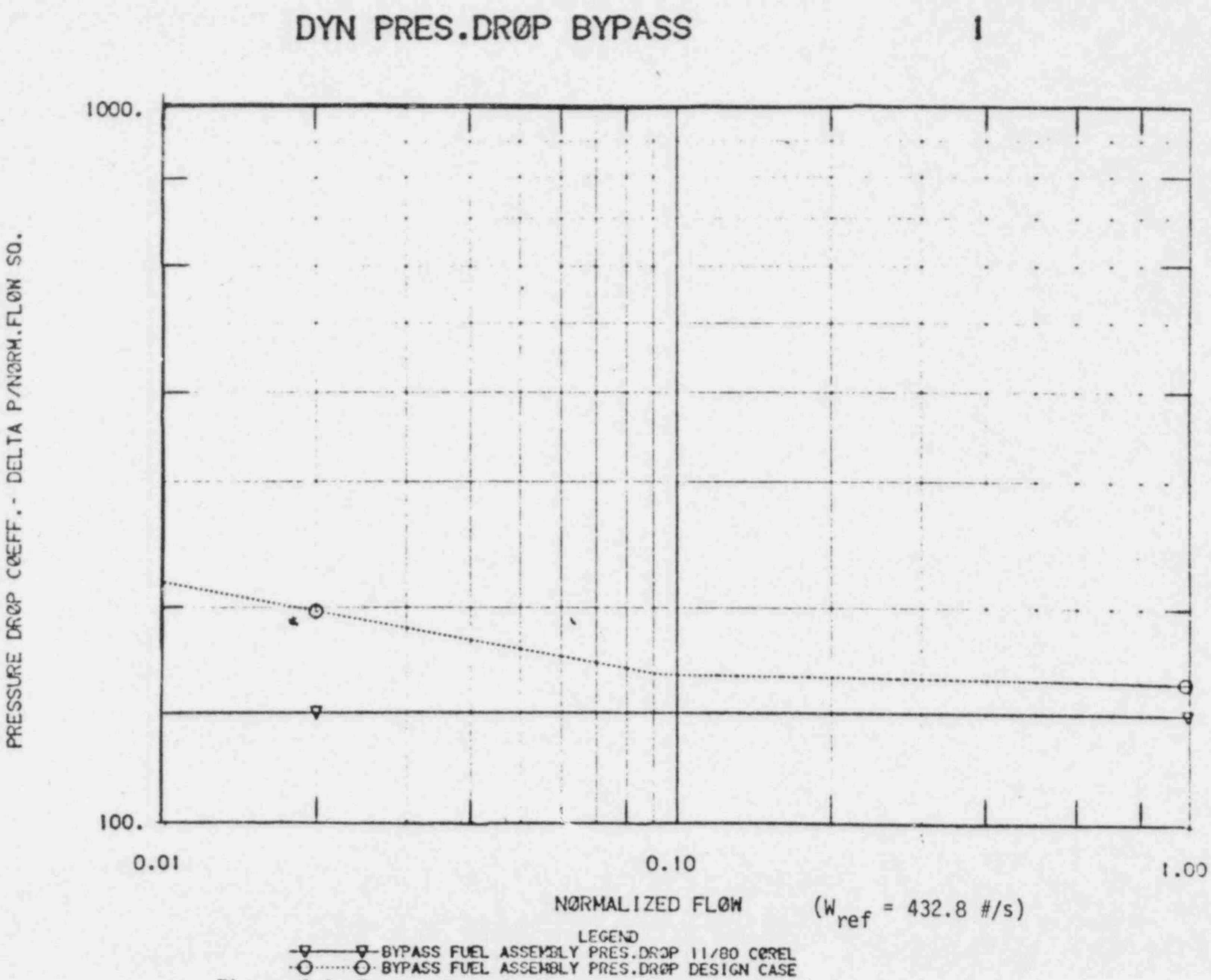


Figure 4-3 Bypass Dynamic Pressure Drop Correlations

The post-shutdown decay power is input in tabular form as a function of time. Reference 10 assumed 25 hours of operations history for the 100% power case and only 1 hour for the 75% case. In the updated predictions, the power history for both the 100% and 75% case assumes 25 hours of operations at 100% or 75% power prior to shutdown. The values of decay power used for the best estimate predictions are shown in Table 4-1 for the fuel assemblies and the non-fuel assemblies. For those cases using a "maximum" decay power (Cases III and IV), these values were multiplied by 1.25; and for the case (Case V) using a "minimum" decay power, these values were multiplied by 0.75. No startup period was considered in the decay power determinations.

DEMØ Input for 100% Power/100% Flow Initial Conditions

For the 100% power case the plant conditions at the beginning of the test are specified as:

1) Reactor power at 100% ± 1% of 400 MW;
2) Primary loop flow at 100% ± 1% of 13,443 gpm;
3) Primary cold leg temperature at 680 ± 5°F;
4) Primary hot leg temperature at 938 ± 5°F;
5) Secondary loop flow at 100% ± 1% of 13,200 gpm;
6) Secondary cold leg temperature at 602°F; and
7) All six heat transport system (HTS) pony motors de-energized.

The transient will be initiated by scramming the reactor, causing a trip of all six heat transport system (HTS) sodium pumps. The pumps will coast down to zero speed and the plant will undergo a transition to natural circulation flow.

In addition to the best estimate case, four additional cases were analyzed in order to determine effects of maximum uncertainties in the pressure drop data and the decay power data. The five cases analyzed are described in Section 4-1.

4.3 DEMØ Results for 100% Power/100% Flow Initial Conditions

The necessary dimensional input and initial conditions for these analyses are given in the update listing in Appendix A. Major results from these analyses are presented in Figures 4-4 through 4-13. Results for Case I through Case IV are plotted together because they were used in subsequent analyses by CØBRA-WC/FØRE-2M. Results for Case V were presented separately because they were only used in design analyses where flows were used as the limiting criterion.

Figure 4-4 shows a comparison of the predicted primary loop flow in GPM for Cases I through IV. for the best estimate case (Case I) the primary flow drops to a minimum value of about 2% (270 gpm) of rated conditions 100 seconds after the start of transient. It then recovers to about 3% (410 gpm) after about 600 seconds. The drop in flow at 100 seconds is due to the decrease in thermal head resulting from the decrease in power-to-flow ratio immediately following the plant trip. The reactor upper plenum and the reactor outlet piping are filled with cold sodium, causing a reduction in the primary natural head, which drops to a minimum around 40 seconds. When the pumps stop and the flow is supported only by the thermal head, the flow drops to a minimum, while the reactor temperature increases to a maximum value. When the reactor temperature

TABLE 4.1 DECAY HEAT - MW

TIME	FUEL ASSEMBLY		NON-FUEL ASSEMBLY	
	100% Power	75% Power	100% Power	75% Power
0.0	18.54	13.905	0.9204	0.6903
1.0	17.30	12.975	0.8708	0.6531
2.0	16.52	12.390	0.8401	0.6301
3.0	15.97	11.978	0.8183	0.6137
4.0	15.54	11.655	0.8010	0.6008
5.0	15.18	11.385	0.7866	0.5900
6.0	14.86	11.145	0.7743	0.5807
7.0	14.59	10.943	0.7634	0.5726
8.0	14.34	10.755	0.7536	0.5652
9.0	14.12	10.590	0.7448	0.5586
10.0	13.92	10.440	0.7368	0.5526
20.0	12.51	9.383	0.6807	0.5105
30.0	11.63	8.723	0.6457	0.4843
40.0	10.99	8.243	0.6203	0.4652
50.0	10.49	7.868	0.6003	0.4502
60.0	10.08	7.560	0.5839	0.4379
70.0	9.731	7.298	0.5699	0.4274
80.0	9.430	7.073	0.5579	0.4184
90.0	9.167	6.875	0.5474	0.4106
100.0	8.934	6.701	0.5381	0.4036
200.0	7.512	5.634	0.4808	0.3606
300.0	6.786	5.090	0.4512	0.3384
400.0	6.312	4.734	0.4316	0.3237
500.0	5.955	4.466	0.4165	0.3124
600.0	5.663	4.247	0.4041	0.3031
700.0	5.411	4.058	0.3932	0.2949
800.0	5.189	3.892	0.3834	0.2876
900.0	4.989	3.742	0.3746	0.2810
1000.0	4.809	3.607	0.3665	0.2749
2000.0	3.611	2.708	0.3092	0.2319
3000.0	2.955	2.216	0.2729	0.2047
4000.0	2.536	1.902	0.2462	0.1847

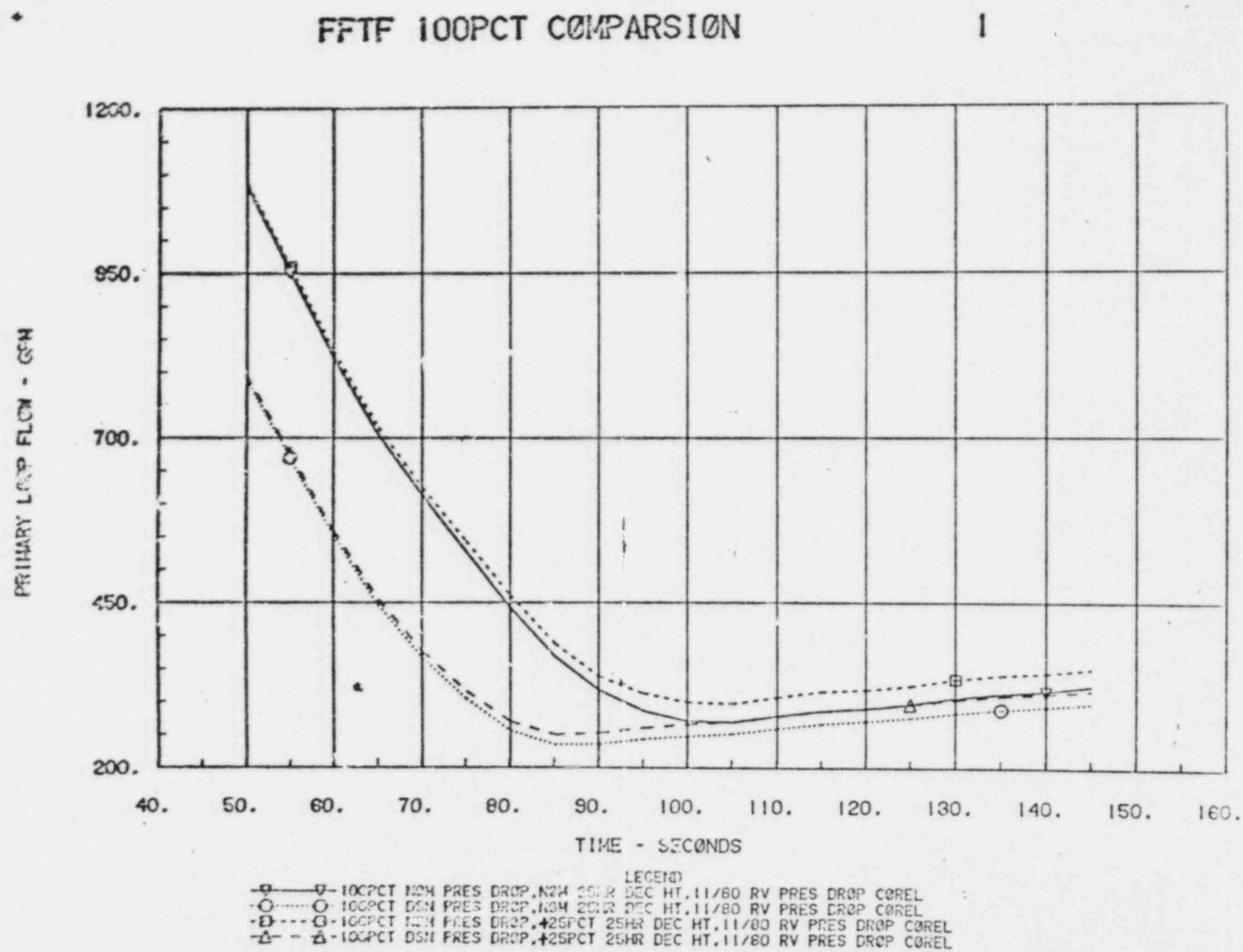


Figure 4-4 DEMO Predicted Primary Loop Flow at 100% Power/100% Flow Initial Conditions (No correction for instrument inaccuracy)

increases, the thermal heads recover and the flow recovers. Thus, it can be seen that the minumum flow reached by the reactor and the time required for it to recover has a significant effect on the maximum reactor temperatures. As expected, high side pressure drop uncertainties lower the predicted primary loop flow due to the added flow resistance while maximum decay heat uncertainty increases the predicted loop flow because of the increased thermal head. Figure 4-5 shows the same predicted flow for Case V with high side pressure drop and minimum decay heat uncertainties. As can be seen when the uncertainties were combined in this way, the predicted primary loop flow drops to a minimum of 220 gpm or 1.6% of initial flow versus 270 gpm or 2.0% of initial flow for the nominal case and reaches this minimum flow about 15 seconds earlier than the nominal case.

Figure 4-6 shows the predicted temperature at the location of the hot leg RTD for Cases I through IV. This instrument is located 85 feet upstream of the pump. The transients depicted in this figure are caused by the reactor vessel exit temperature mitigated by about 65 feet of piping. Both the high side pressure drop uncertainties and the maximum decay heat uncertainty apparently have the effect of smoothing out the initial transient (Cases II through IV). Figure 4-7 shows the hot leg RTD temperature for Case V. Again, the combined effect of the high side pressure drop and the minimum decay heat uncertainties gives rise to a less severe transient in the predicted hot leg temperature.

Figure 4-8 shows the predicted temperature at the primary cold leg RTD for Cases I through IV. This instrument is located just upstream of the cold leg check valve. The transient occurring early on as seen at the cold leg RTD location, is caused by the collapse of the IHX primary outlet temperature onto the secondary inlet temperature at the tube bundle exit and the mitigation of this transient due to heat transfer between the sodium and the IHX lower plenum and cold leg piping. This collapse occurs because as the flows decrease, heat is transferred fore efficiently from the primary to the secondary side. High pressure drop uncertainties increase the predicted cold leg temperature as a result of changes in the primary and secondary loop flow ratio combined with changes in the above mentioned mitigating effects which are more pronounced with lower flows. Maximum decay heat uncertainty, on the other hand, lowers the cold leg temperature slightly because of the slight increase to the primary loop flow. Figure 4-9 shows the same primary cold leg RTD temperature for Case V. Here the effect of the high pressure drop and minimum decay heat uncertainties works together to increase the predicted cold leg temperature.

Figure 4-10 shows the acceptance limit curve for DEMØ for the 100% transient natural circulation analyses. The parameter used in this acceptance criterion is in terms of the primary loop flow. This selection is based primarily on the consideration that the available primary flow is the key parameter in the determination of natural circulation decay heat removal capability. Also shown in this figure are results from the best estimate predictions (Case I) and the minimum flow design predictions (Case V). The acceptance limit curve was developed from the flows calculated in Case V by:

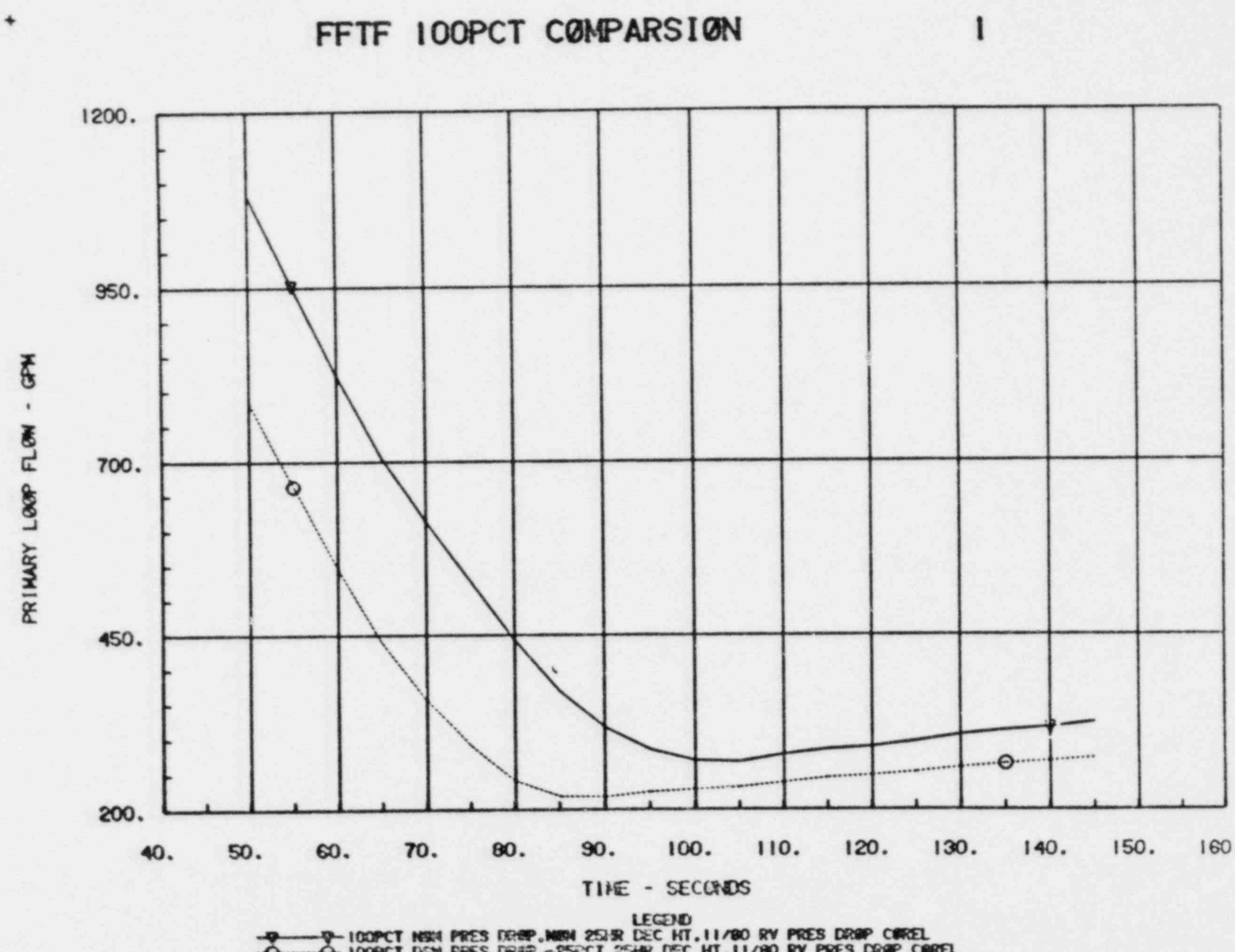


Figure 4-5 DEMO Predicted Primary Loop Flow at 100% Power/100% Flow Initial Conditions (Case V) (No correction for instrument inaccuracy)

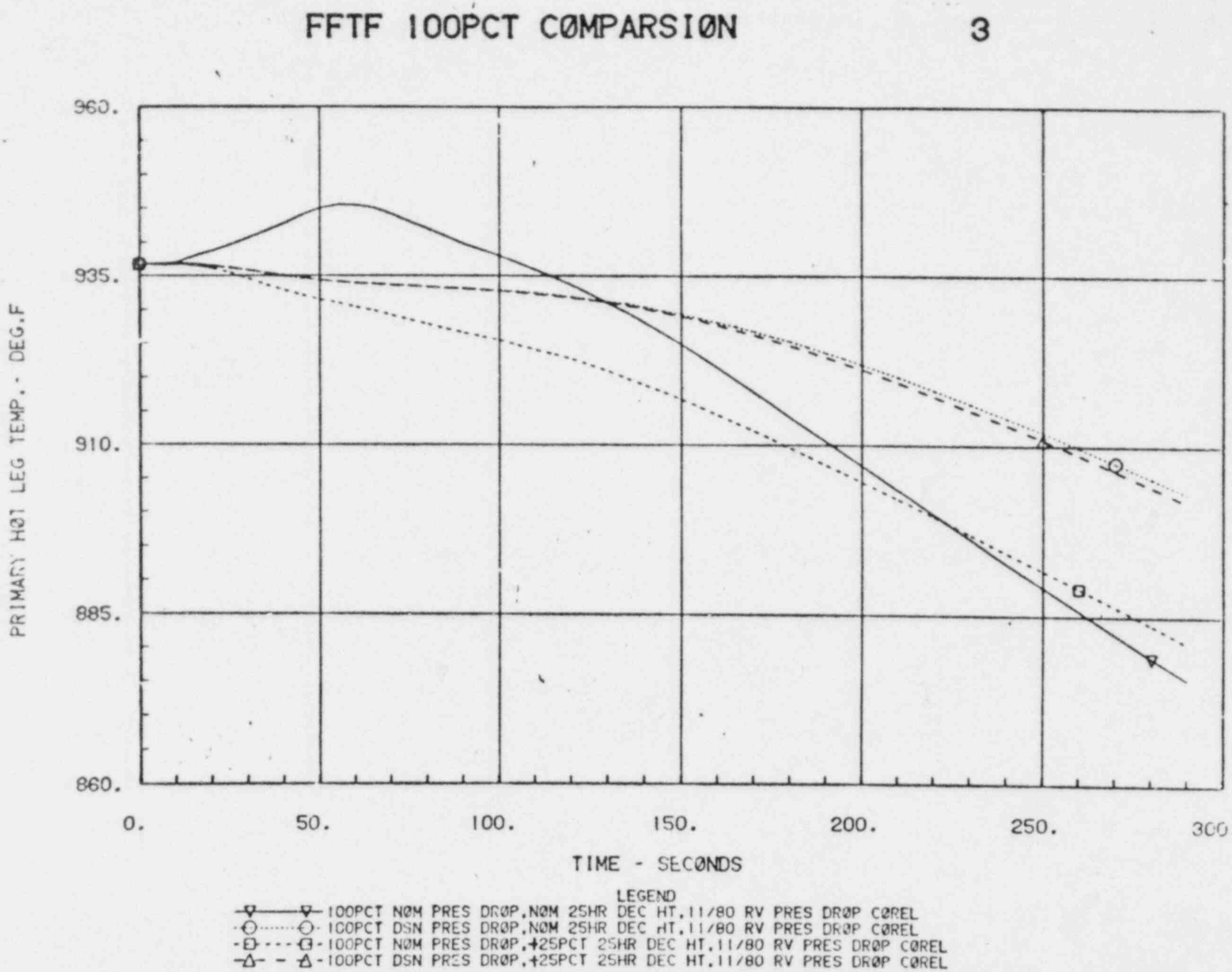


Figure 4-6 DEMO Predicted Primary Hot Leg RTD Temperature (100% Power/100% Flow Initial Conditions) (No correction for instrument inaccuracy)

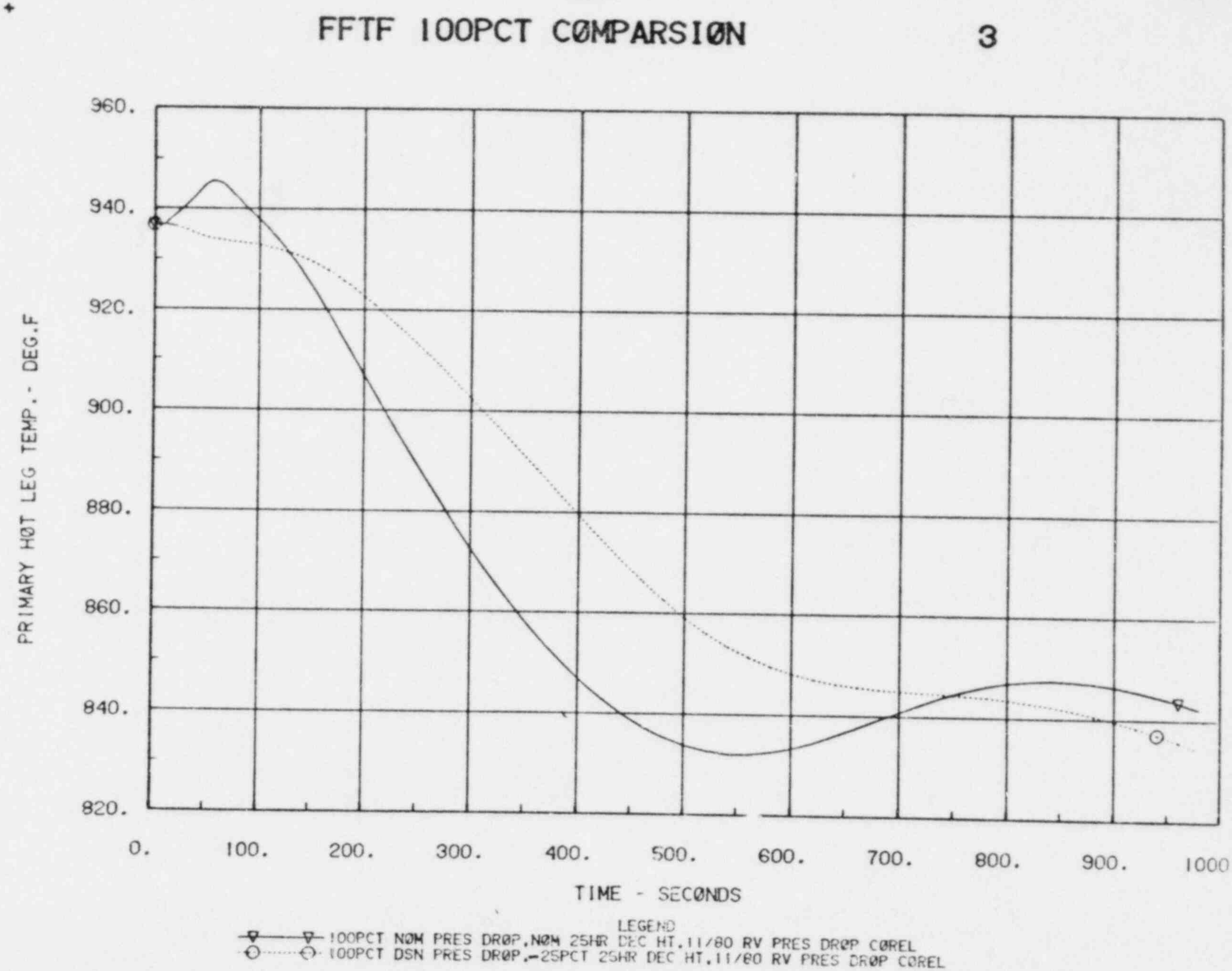


Figure 4-7 DEMO Predicted Primary Hot Leg Temperature (100% Power/100% Flow Initial Conditions, Case V) (No correction for instrument inaccuracy)

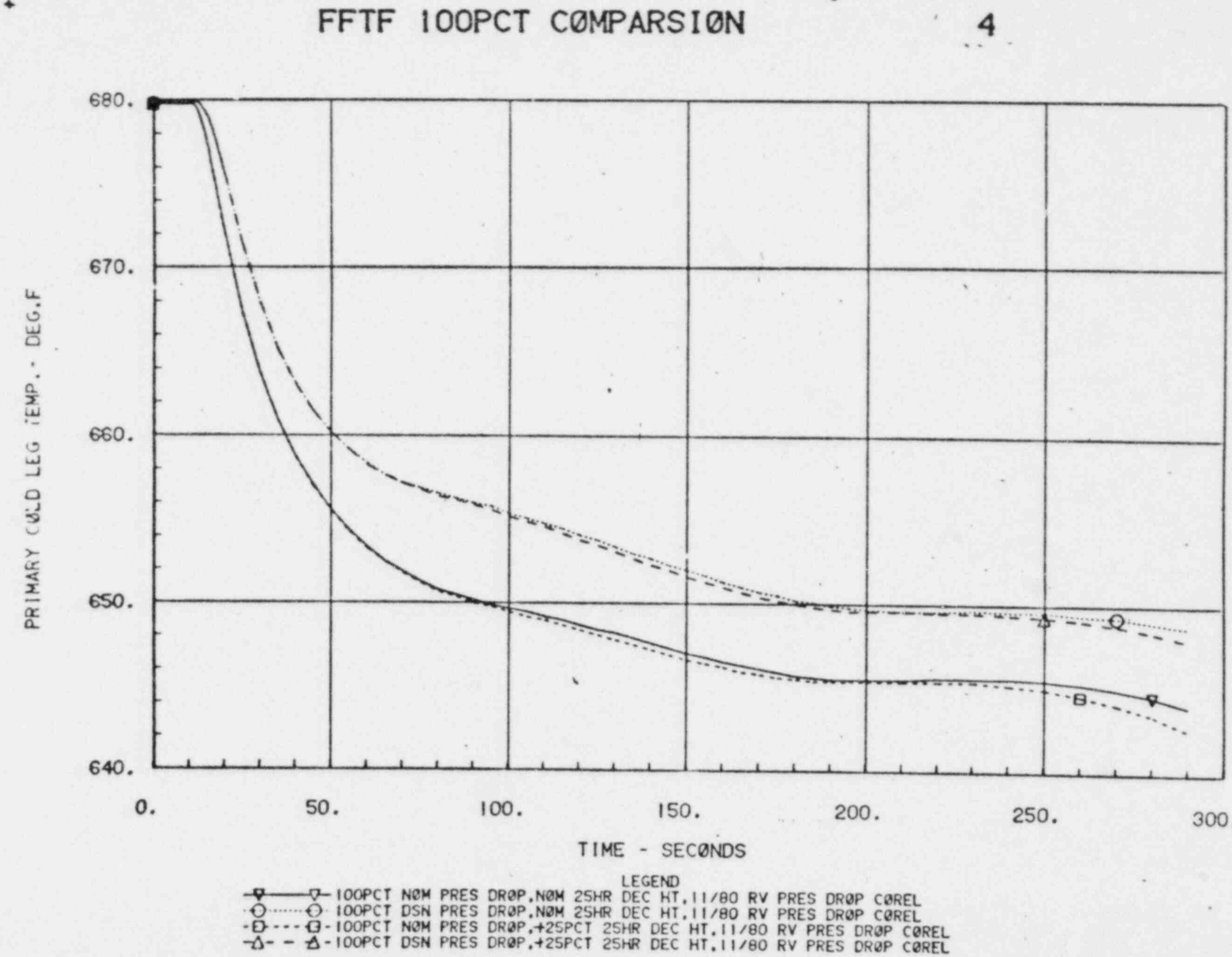


Figure 4-8 DEMO Predicted Primary Cold Leg RTD Temperature (100% Power/100% Flow Initial Conditions) (No correction for instrument inaccuracy)

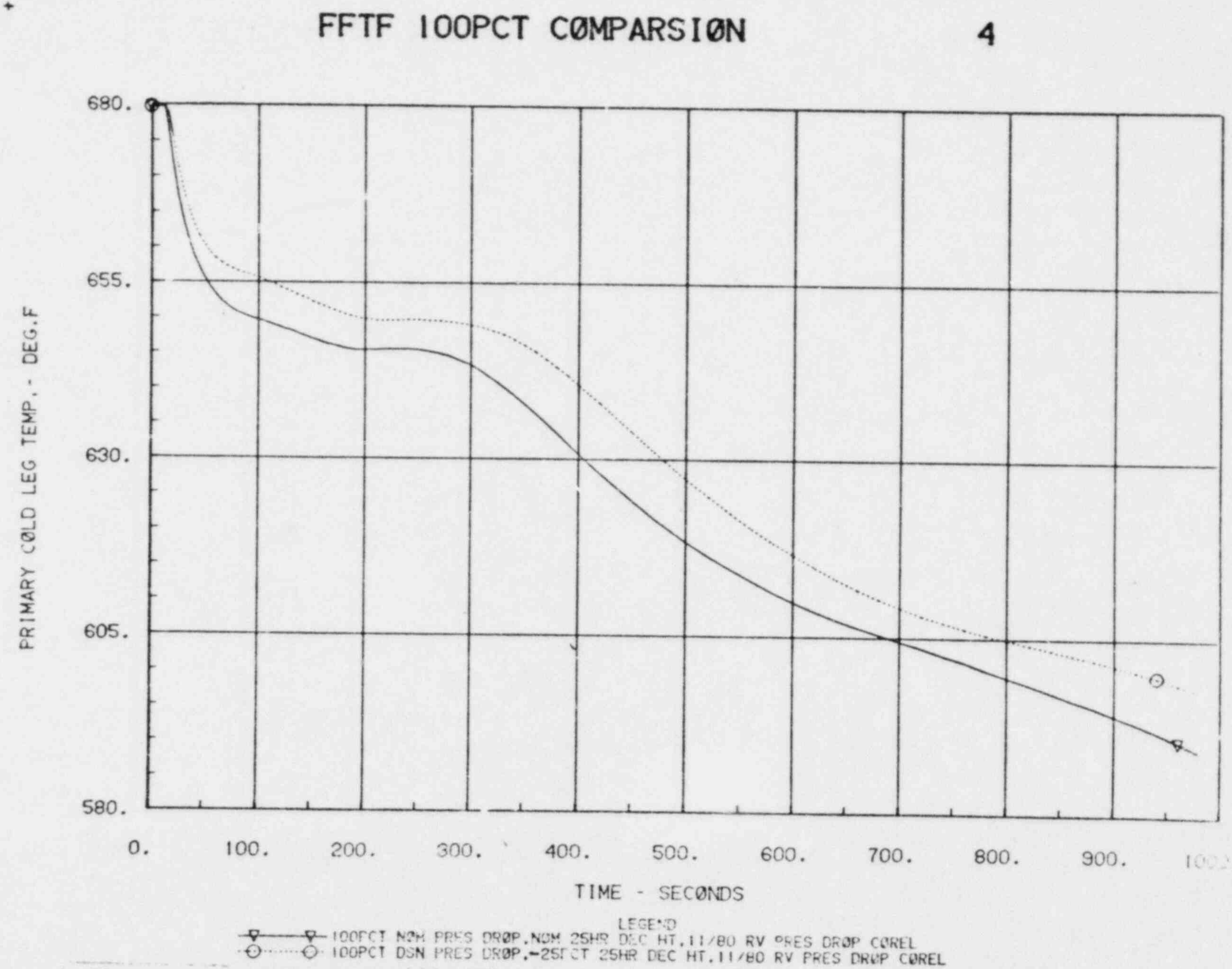
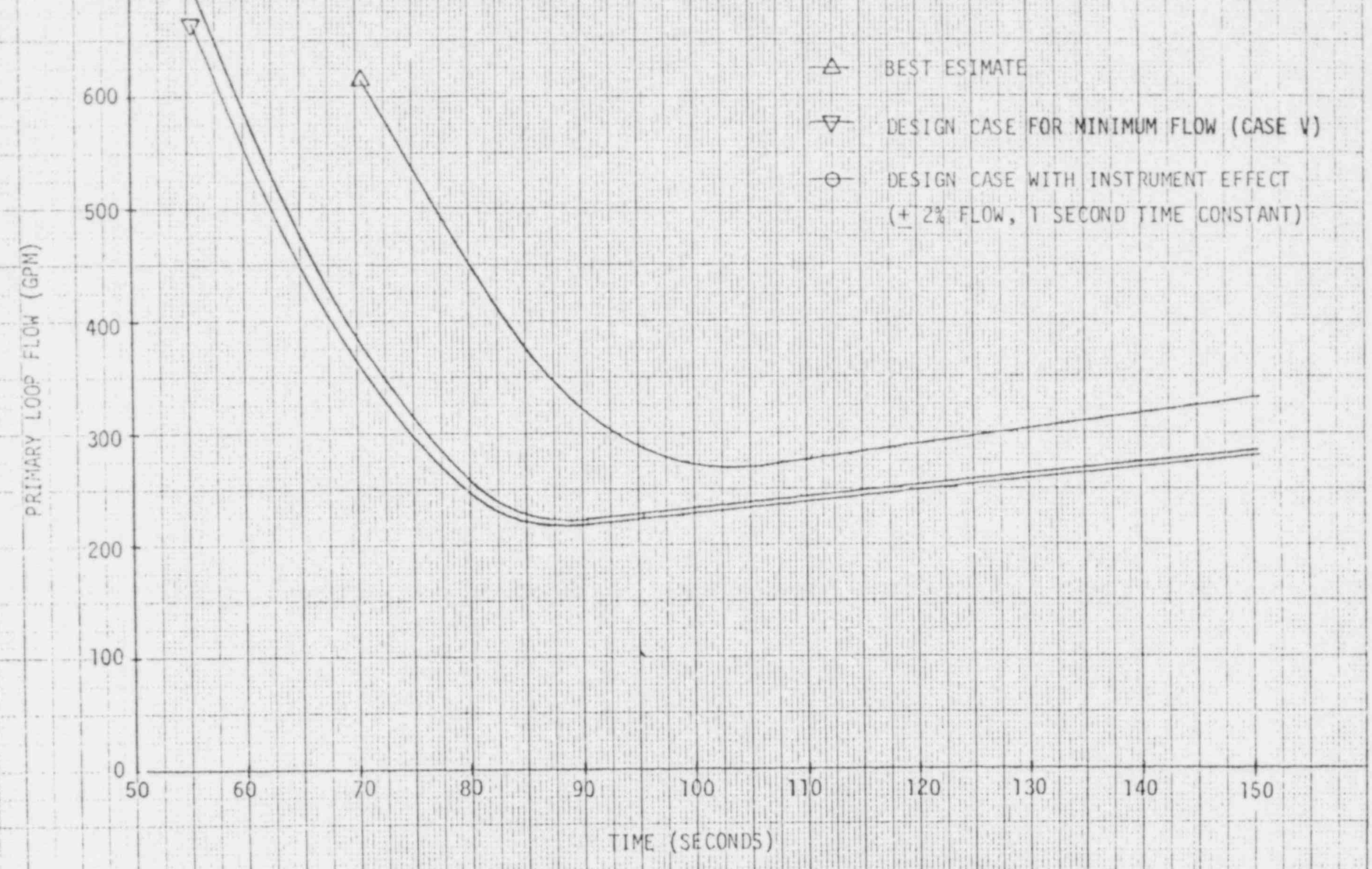


Figure 4-9 DEMO Predicted Primary Cold Leg Temperature (100% Power/100% Flow Initial Conditions, Case V) (No correction for instrument inaccuracy)

Figure 4-10 ACCEPTANCE LIMIT ON FLOW FOR THE 100% POWER CASE



1) Increasing the flow by 2% of the calculated value to account for the magnetic flow meter inaccuracies; and

2) Adjusting the calculated flow with a one second first order lag to account for the magnetic flow meter time constant.

This acceptance limit, therefore, further biases the predicted flows in the conservative direction and should represent the minimum acceptable measured flows. If the actual measured flows are greater than this acceptance limit, then it would follow that a DEMØ calculation made with the minimum flow design case (Case V) conservatively envelope the actual natural circulation flows. This particular acceptance limit is related only to flows as computed by DEMØ. In-core temperature acceptance criteria would be based on a combination of high side decay heat as well as pressure drop uncertainties and would, therefore, be based on a different flow curve (see Figure 4-4).

Figure 4-11 shows the predicted secondary loop flow for Cases I and V. This flow remains higher than the primary side, for both the best estimate case (Case I) and the minimum flow design case (Case V). The corresponding predictions for the secondary hot leg RTD and cold leg RTD temperatures are shown in Figures 4-12 and 4-13. The secondary hot leg RTD is located upstream of the tee, before the DHX inlet. This secondary cold leg RTD is located 60 feet downstream of the secondary pump outlet.

4.4 CØBRA-WC Results for 100% Power/100% Flow Initial Conditions

Major output from the CØBRA-WC code consists of transient flow and coolant temperature information in all assemblies including the peak channel (cluster of subchannels as shown in Figure 3-3) within specific assemblies (FOTA's). The former information is used as input data for FØRE-2M analysis, and the latter information provides a basis for the selection of peak rod/channel for the single-channel analysis. The inter- and intra-assembly flow redistribution are reflected in the parameter

$$\frac{G(\chi,\tau)}{G(o,\tau)}\ ,\ \text{and}\ \frac{G(o,\tau)}{G(o,o)}\ , \qquad (4\text{-}1)$$

where $G(\chi,\tau)/G(o,\tau)$ is the ratio of the axial mass velocity at axial (node) location χ to that at the inlet of assembly at time τ; $G(o,\tau)/G(o,o)$ is the normalized inlet mass velocity based on initial mass velocity (τ=o). Since inlet velocity is assumed uniform in each assembly, the normalized inlet mass velocity for a subchannel is the same as the initial mass velocity for the entire assembly. Thus, the normalized axial mass velocity at node location χ becomes

$$\frac{G(\chi,\tau)}{G(\chi,o)} = \frac{G(\chi,\tau)}{G(o,\tau)} \cdot \frac{G(o,\tau)}{G(o,o)} \cdot \frac{G(o,o)}{G(\chi,o)} \qquad (4\text{-}2)$$

Combining Equation 4-2 with the expression for normalized reactor flow, $G_R(\tau)/G_R(o)$, will yield,

$$\frac{G(\chi,\tau)}{G(\chi,o)} \cdot \frac{G_R(o)}{G_R(\tau)} = \frac{G(\chi,\tau)/G_R(\tau)}{G(\chi,o)/G_R(o)} = \frac{G_i(\tau)/G_R(\tau)}{G_i(o)/G_R(o)} \qquad (4\text{-}3)$$

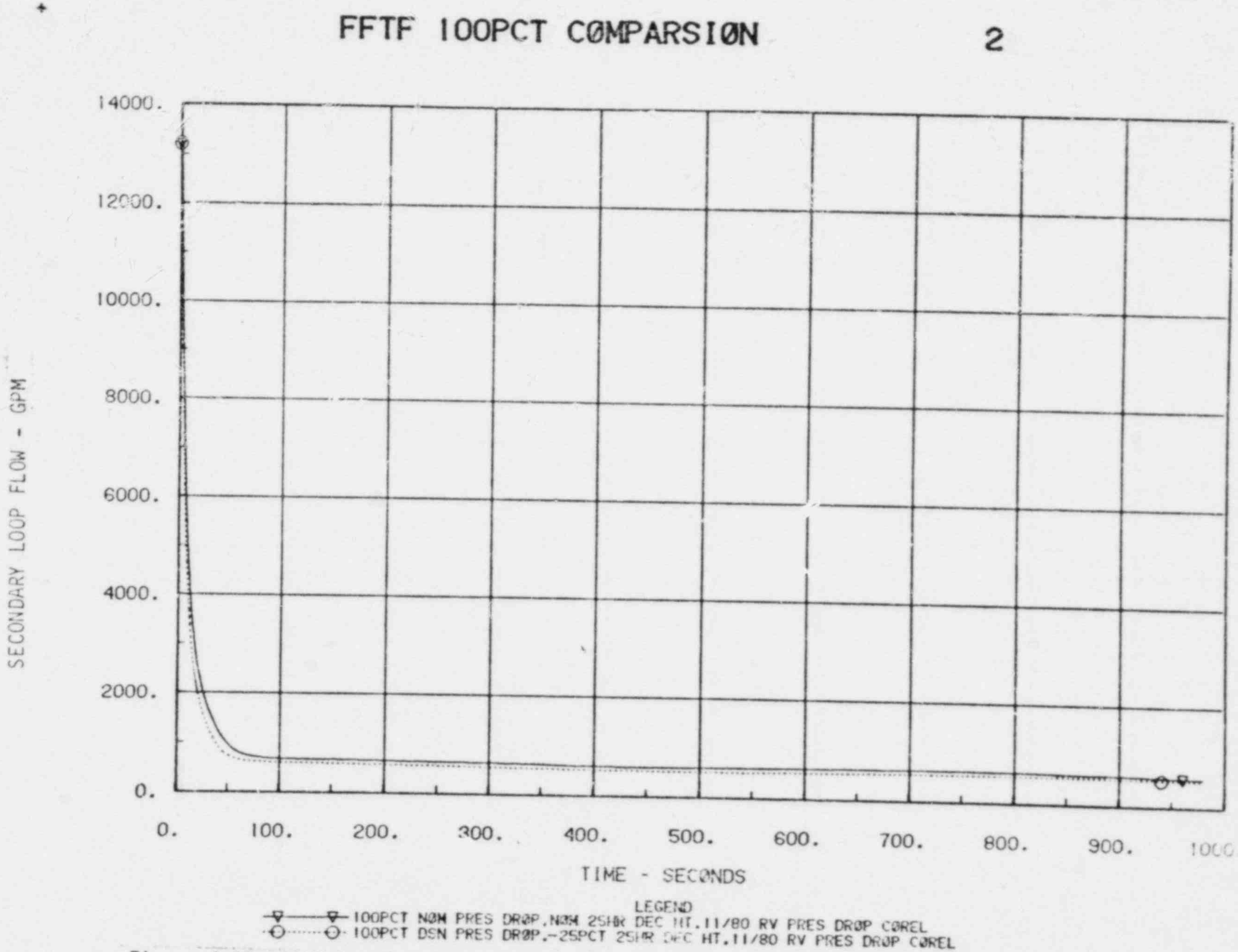


Figure 4-11 DEMO Predicted Secondary Loop Flow at 100% Power/100% Flow Initial Conditions)
(No correction for the instrument inaccuracy)

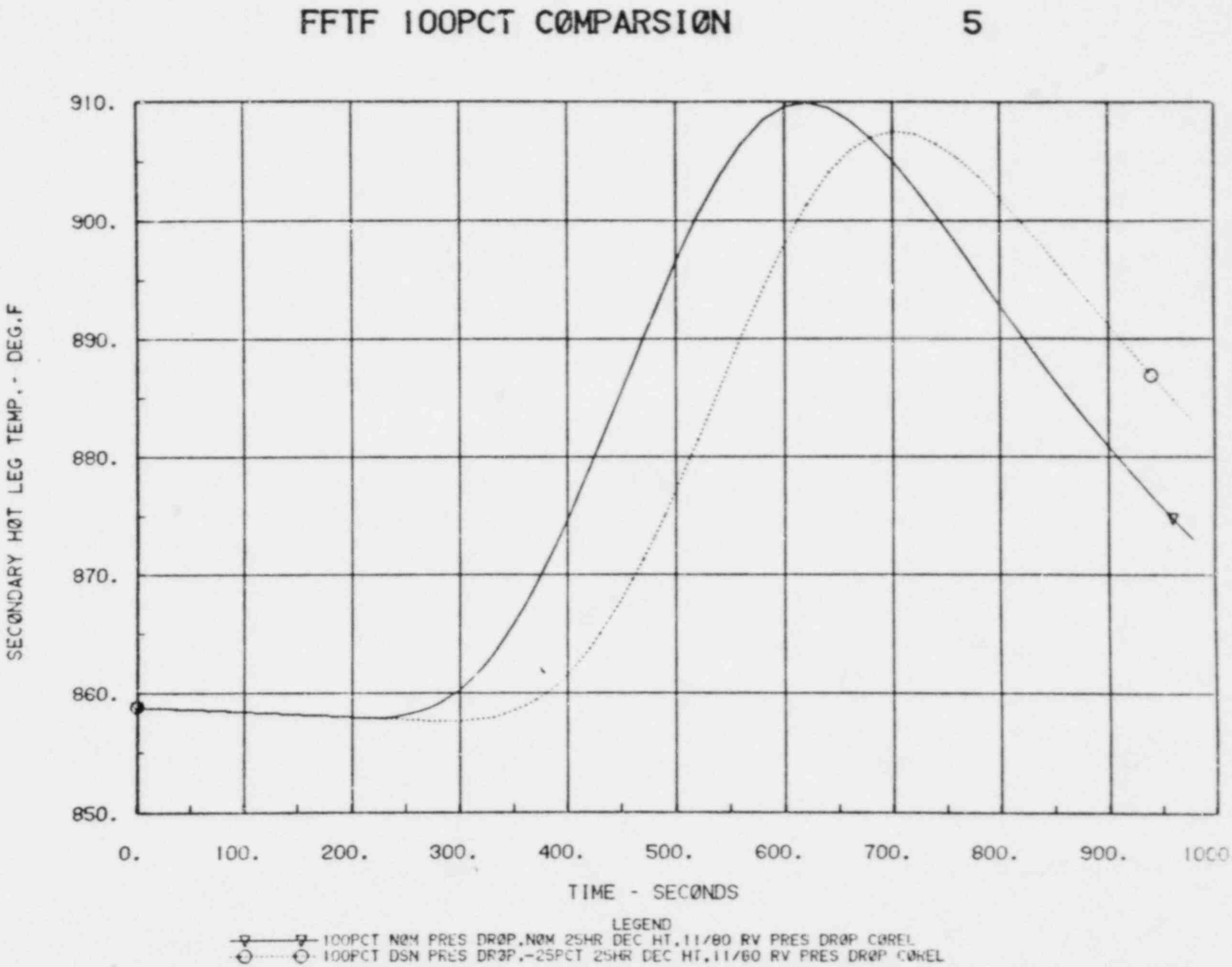


Figure 4-12 DEMO Predicted Secondary Hot Leg RTD Temperature (100% Power/100% Flow Initial Conditions) (No correction for instrument inaccuracy)

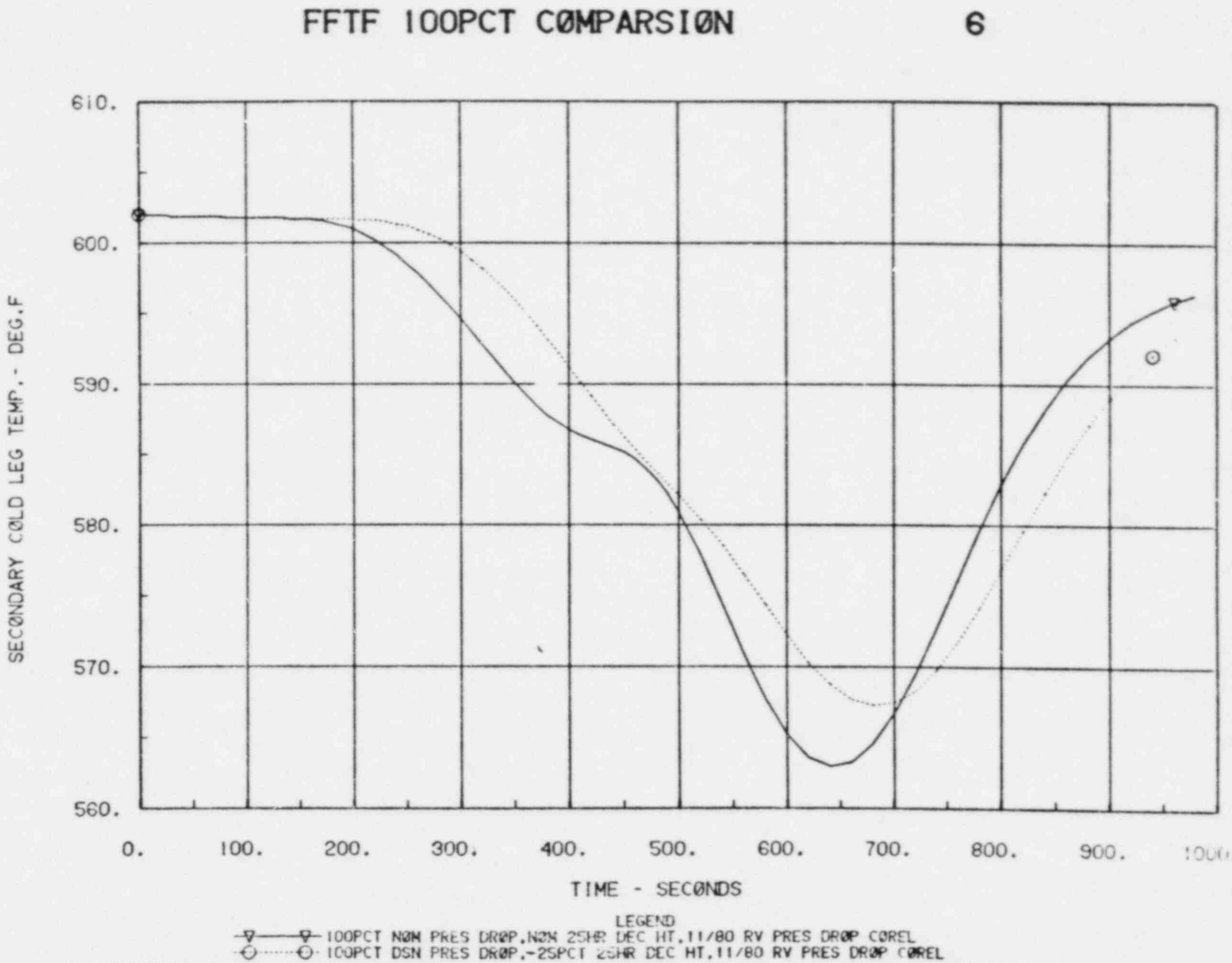


Figure 4-13 DEMO Predicted Secondary Cold Leg RTD Temperature (100% Power/100% Flow Initial Conditions) (No correction for instrument inaccuracy)

Note that Equation 4-3 is the expression of axial flow fraction at node i (location χ) and time τ, relative to the initial flow fraction (τ=0). When this ratio is greater than 1, the channel axial flow fraction is gaining flow relative to initial (steady state) conditions. Figure 4-14 shows this ratio for the Row 2 FOTA initiated from 100% power/100% flow conditions at two axial nodes (i.e., i=1 and 7, bottom and top nodes within the active fuel section, respectively). For i=1, this ratio represents the effect of inter-assembly flow redistribution with time. At τ=125 sec. the ratio peaks which reflects the maximum buoyancy effect on flow. The difference between i=1 and i=7, represents the gaining in axial flow fraction at the top node relative to the bottom node, a measure of intra-assembly flow redistribution (or cross flow mixing). Similar variations of the ratio for the Row 6 FOTA are shown in Figure 4-15.

The intra-assembly flow redistribution also contributes to sodium enthalpy change which is a factor included in the $Q_{ex}(\chi,\tau)$ As shown in Equation 3-1, this term accounts for inter- and intra-assembly heat transfer effects, and cannot be shown explicitly. It consists of that due to the radial temperature gradient (conduction effect) and to cross flow mixing (transport effect) while the heat flux from the rod surface also includes the stored energy in the fuel rod. Figure 4-16 shows the typical variation in power data for the peak rod at the top node of the active fuel section for the Row 2 FOTA under nominal conditions. There appears to be three periods during the natural circulation transient with respect to the mode of energy transfer from the rod to the "peak channel" at the top fuel node:

1) Period (0 to 15 sec.) - The heat transfer from the rod is larger than the power generated due to stored heat effects from the fuel internals. The absolute value of $Q_{ex}(\chi,\tau)$ decreases due to the overall assembly cooling during most of this period, i.e., radial temperature gradient in sodium collapses.

2) Period (15 to 95 sec.) - The coolant temperature is monotonically increasing. The heat deposition to the coolant is less than the power generated since part of this power has to first increase the fuel temperature to a sufficient level to drive the heat generated into the coolant. The absolute value of $Q_{ex}(\chi,\tau)$ increases gradually due to the radial temperature gradient buildup in the bundle and flow redistribution effects.

3) Period (95 to 140 sec.) - After 95 seconds, the $Q_{ex}(\chi,\tau)$ decreases due to a decrease in radial temperature gradient in the bundle with high inter- and intra-assembly flow redistribution. Eventually, the power generated is lower than the heat transferred from the fuel to the coolant due to the relative increase in coolant flow (which increases the fuel-to-coolant temperature difference) in this channel coupled with the decrease in power generation.

It is shown that during the transient, $Q_{ex}(\chi,\tau)$ reaches a maximum value of 33% of the heat from the rod surface in the Row 2 FOTA. By utilizing this parameter along with axial mass velocity ratios, $G(\chi,\tau)/G(o,\tau)$, as input to the single-channel FØRE-2M code, the inter- and intra-assembly heat redistribution effect is thereby included in the peak channel coolant temperature predictions.

Typical CØBRA-WC code input data for the nominal case natural circulation test initiated at 100% power/100% flow conditions are shown in Appendix B.

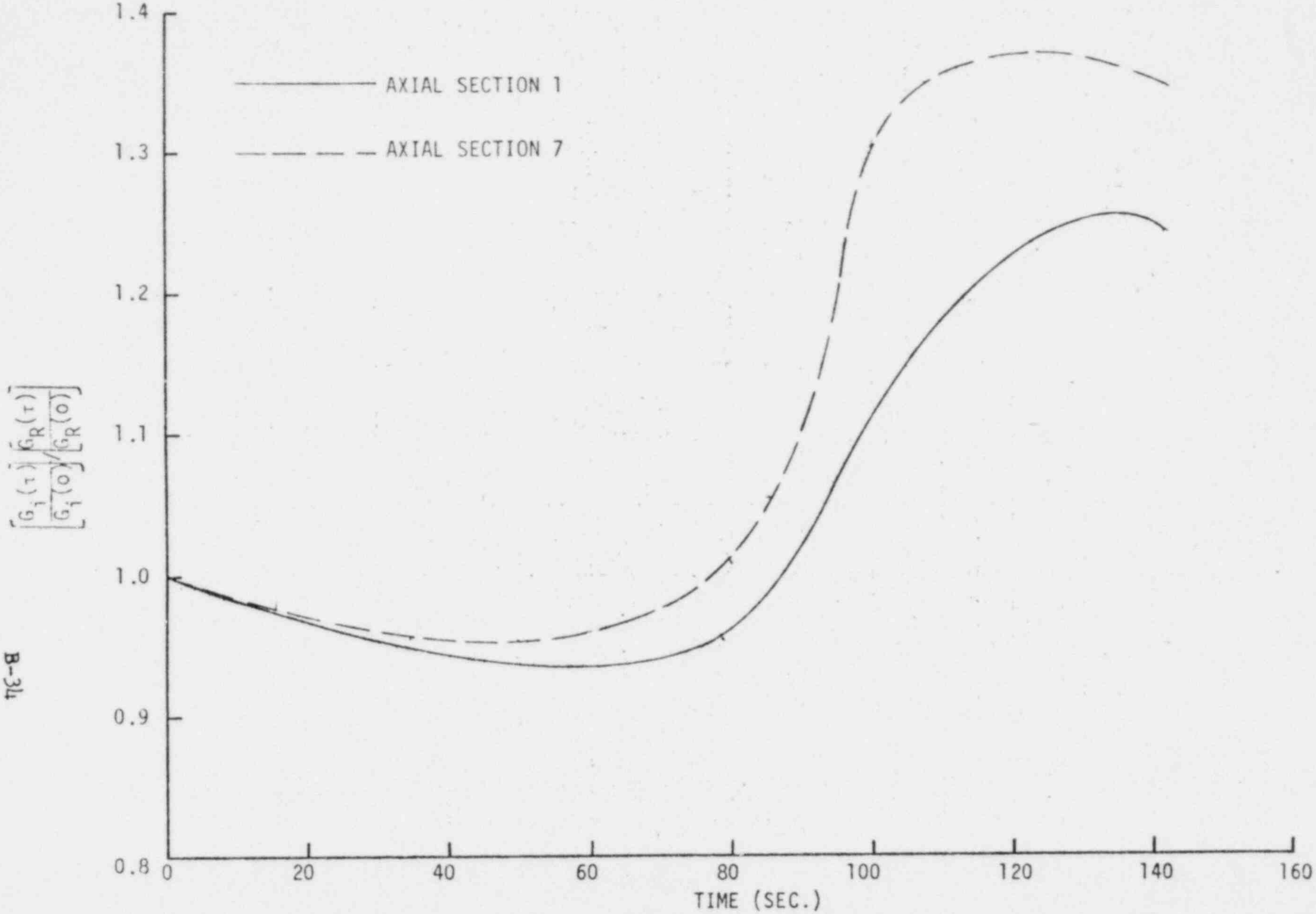


FIGURE 4-14: PEAK CHANNEL FLOW RELATIVE TO REACTOR FLOW COASTDOWN (ROW 2 FOTA NOMINAL CASE INITIATED AT 100% POWER/100% FLOW)

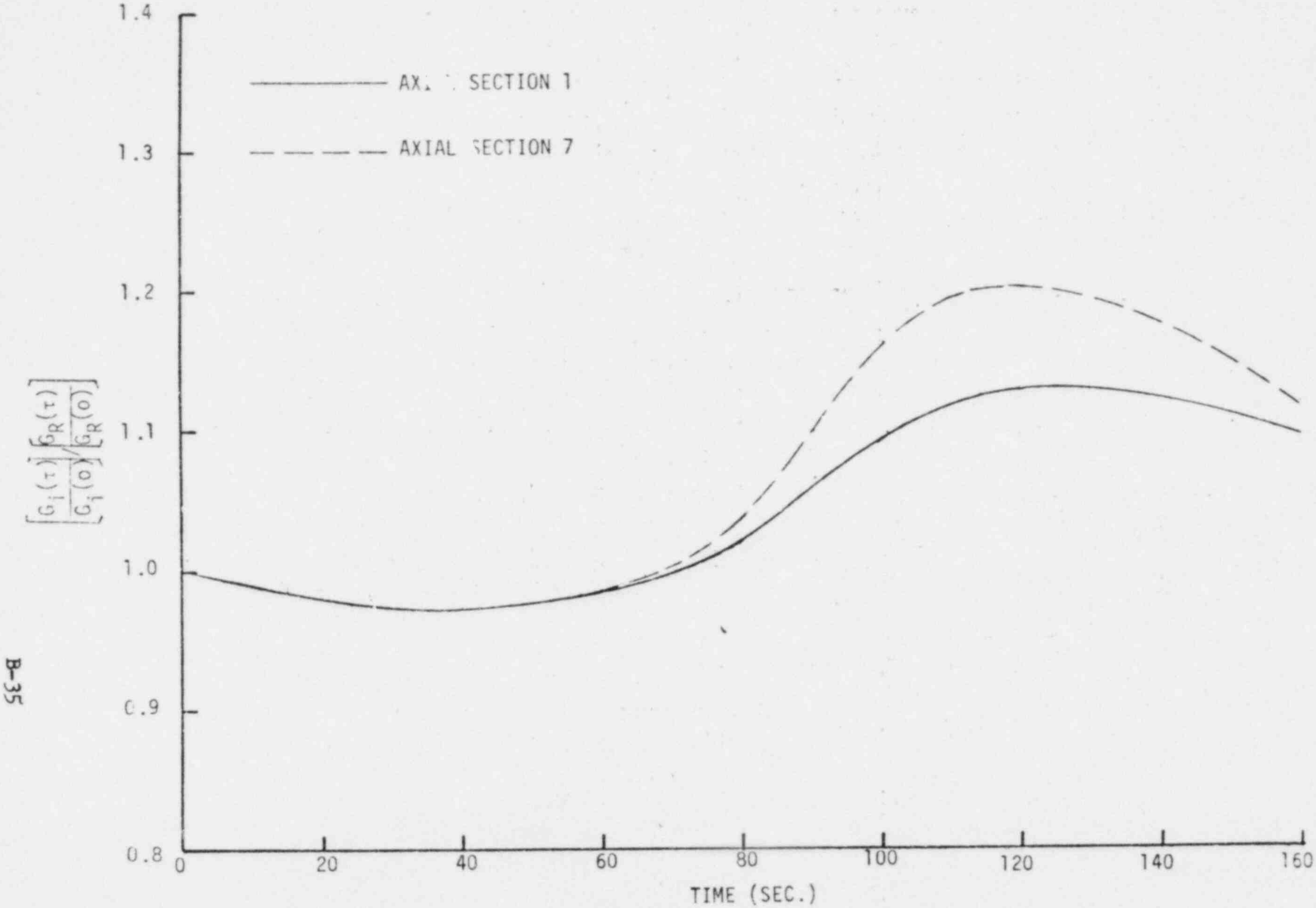


FIGURE 4-15: PEAK CHANNEL FLOW RELATIVE TO REACTOR FLOW COASTDOWN (ROW 6 FOTA NOMINAL CASE INITIATED AT 100% POWER/100% FLOW)

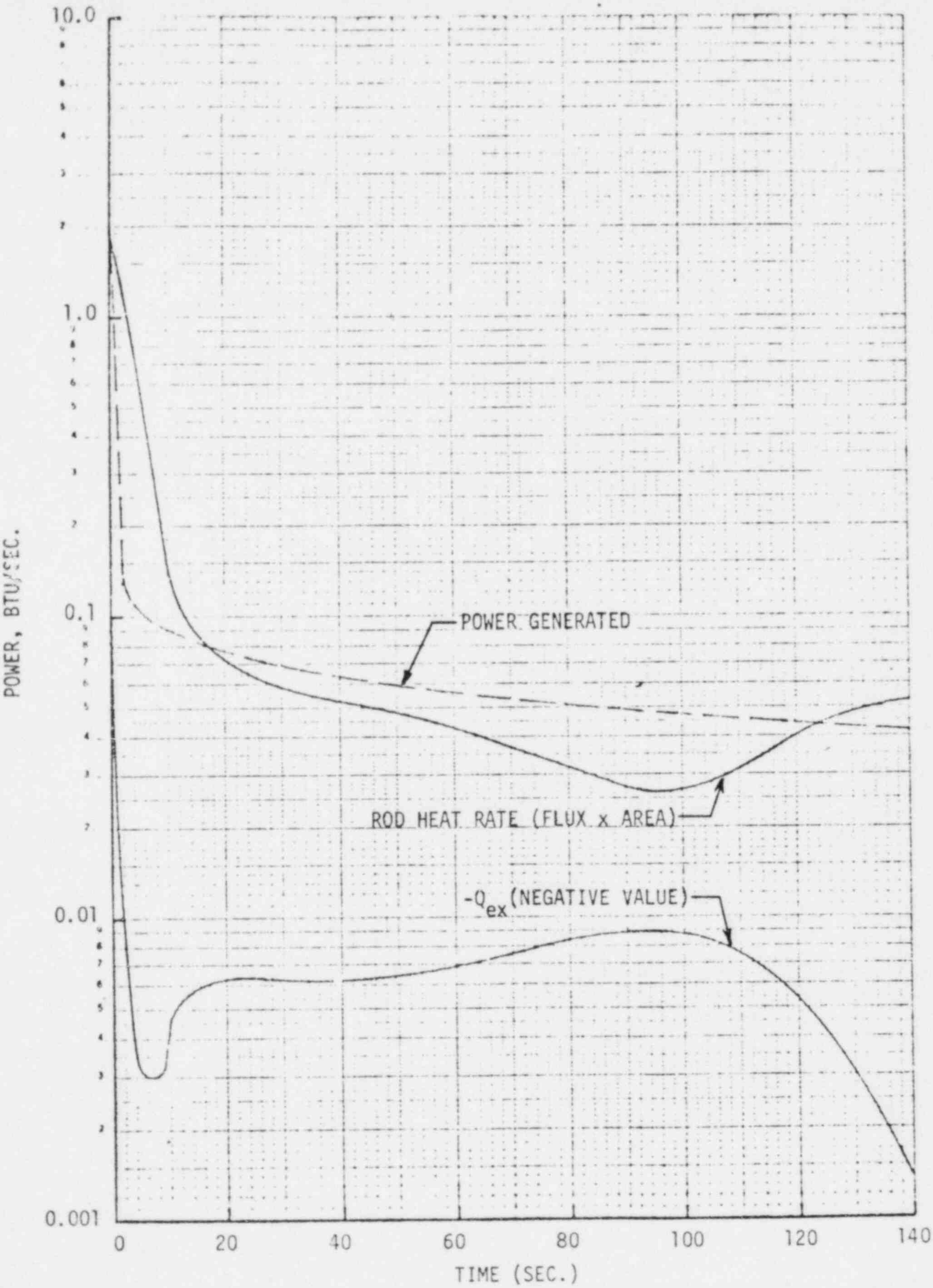


FIGURE 4-16: VARIATION IN POWER DATA FOR HOT ROD AT TOP NODE OF ACTIVE FUEL LENGTH (ROW 2 FOTA) FOR THE NOMINAL CASE INITIATED AT 100% POWER/100% FLOW CONDITIONS

4.5 FØRE-2M Results for 100% Power/100% Flow Initial Conditions

Figures 4-17 and 4-18 show results for the Row 2 and Row 6 FOTA's, respectively, as calculated with FØRE-2M using input from the CØBRA-WC computer code. These transient sodium coolant temperatures are for the top of the active core axial position in Channel 15 for the Row 2 FOTA and Channel 8 for the Row 6 FOTA. These locations represent hot spots where maximum temperatures are occurring in these rod bundles. The effect of the following conditions are thus defined relative to each other:

- Nominal conditions (Case I);
- Nominal conditions except pressure drops increased to design values (Case II); and
- Nominal conditions except a +25% decay heat increase (Case III).

The transient differences in core ΔT between these cases become part of the uncertainty analysis as will be described in Section 4.6.

The sodium temperature estimation for Case IV with maximum uncertainties and using the conservative CRBRP design approach is shown in Figures 4-19 and 4-20, for the respective FOTA's. These Case IV predictions represent the most conservative temperature predictions from the design point-of-view.

Typical FØRE-2M code edited input data for the nominal case are shown in Appendix C.

4.6 Effects of Known Uncertainties on the Sodium Temperature Response

From the results of FØRE-2M analyses on Cases I, II and III, the predicted sodium temperature rise at the top node of the active fuel section can be found. The effects of uncertainties in decay heat and pressure drop calculations on "hot channel" temperatures (see Table 3-1) are thereby determined:

$$\text{Hot Channel Subfactor due to Pressure Drop Calculational Uncertainty} = F_2 = \frac{\Delta T_{II}}{\Delta T_I} \quad (4\text{-}4)$$

$$\text{Hot Channel Subfactor due to Decay Heat Uncertainty} = F_3 = \frac{\Delta T_{III}}{\Delta T_I} \quad (4\text{-}5)$$

where ΔT_I, ΔT_{II} and ΔT_{III} are temperature rise for Cases I, II and III, respectively (the definition of these cases are given in Section 4.1). These hot channel subfactors are combined with known hot channel subfactors, F_4, F_5 and F_6 listed in Table 3-1, using the semi-statistical method (Ref. 12). Thus, the sodium temperature at the top of the fuel section with (3σ) uncertainties, is calculated by the expression

$$\Delta T_{UNC} = T_I + \Delta T_{DIR} + \Delta T_{STAT} \quad (4\text{-}6)$$

and

$$\Delta T_{DIR} = (F_1 - 1)\,(\Delta T_I) \quad (4\text{-}7)$$

$$\Delta T_{STAT} = [(F_2-1)^2+(F_3-1)^2+(F_4-1)^2+(F_5-1)^2+(F_6-1)^2]^{1/2} \cdot (\Delta T_I) \quad (4\text{-}8)$$

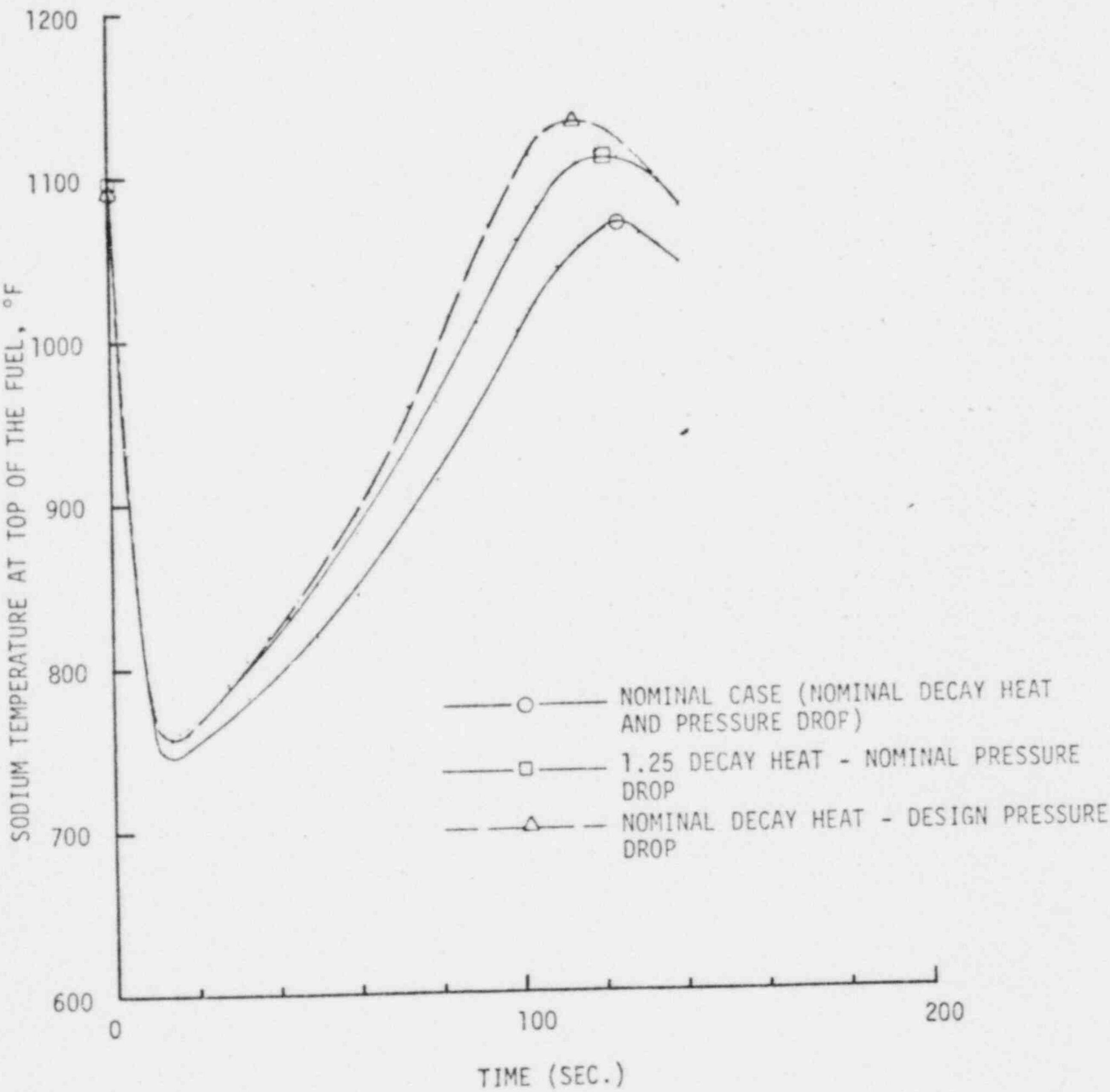


FIGURE 4-17: CALCULATED SODIUM TEMPERATURE AT TOP OF THE FUEL (CHANNEL 15) FOR ROW 2 FOTA WITH DIFFERENT CASES INITIATED FROM 100% POWER/100% FLOW CONDITIONS (Tx1016)

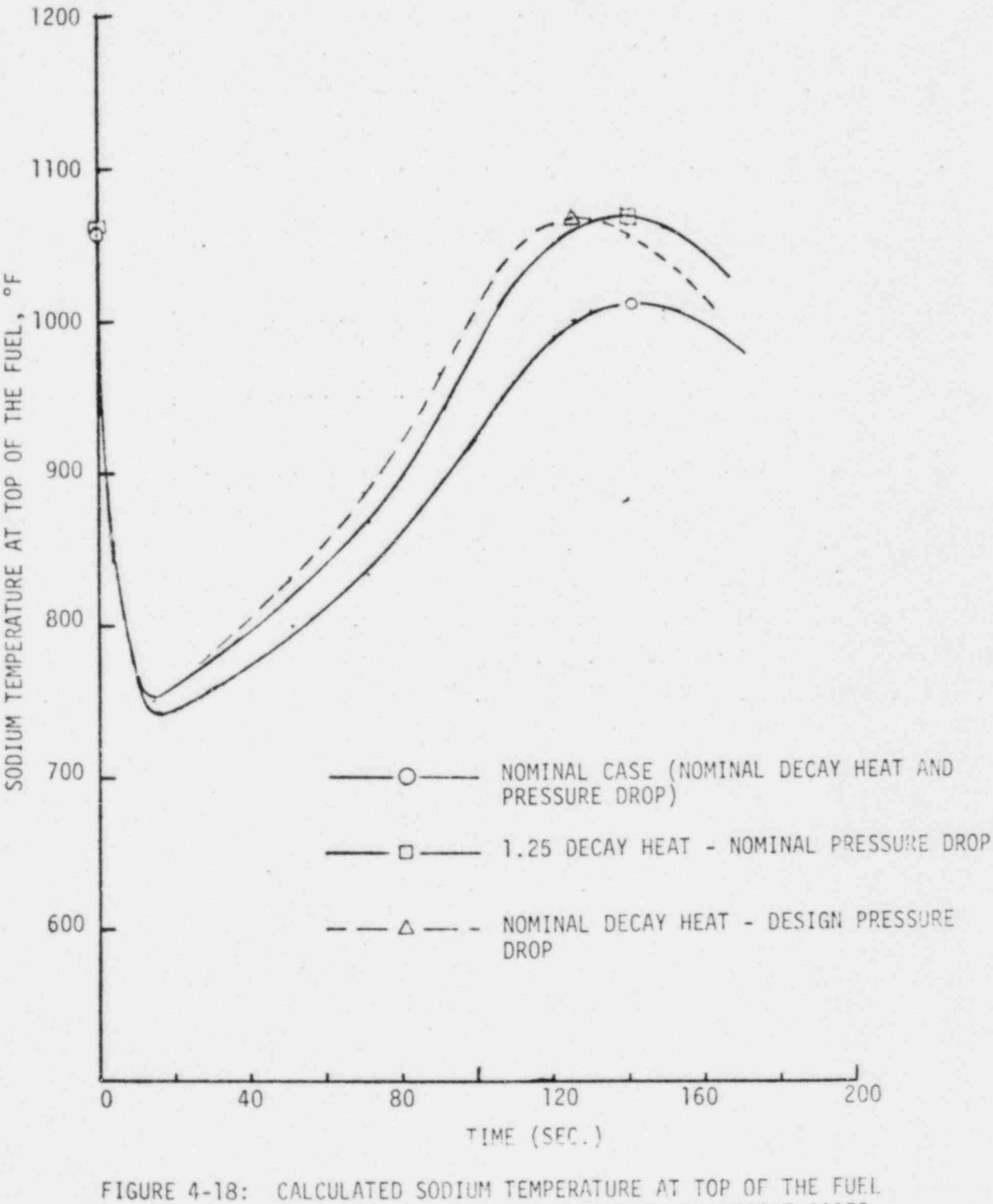


FIGURE 4-18: CALCULATED SODIUM TEMPERATURE AT TOP OF THE FUEL (CHANNEL 8) FOR ROW 6 FOTA WITH DIFFERENT CASES INITIATED FROM 100% POWER/100% FLOW CONDITIONS (Tx9018)

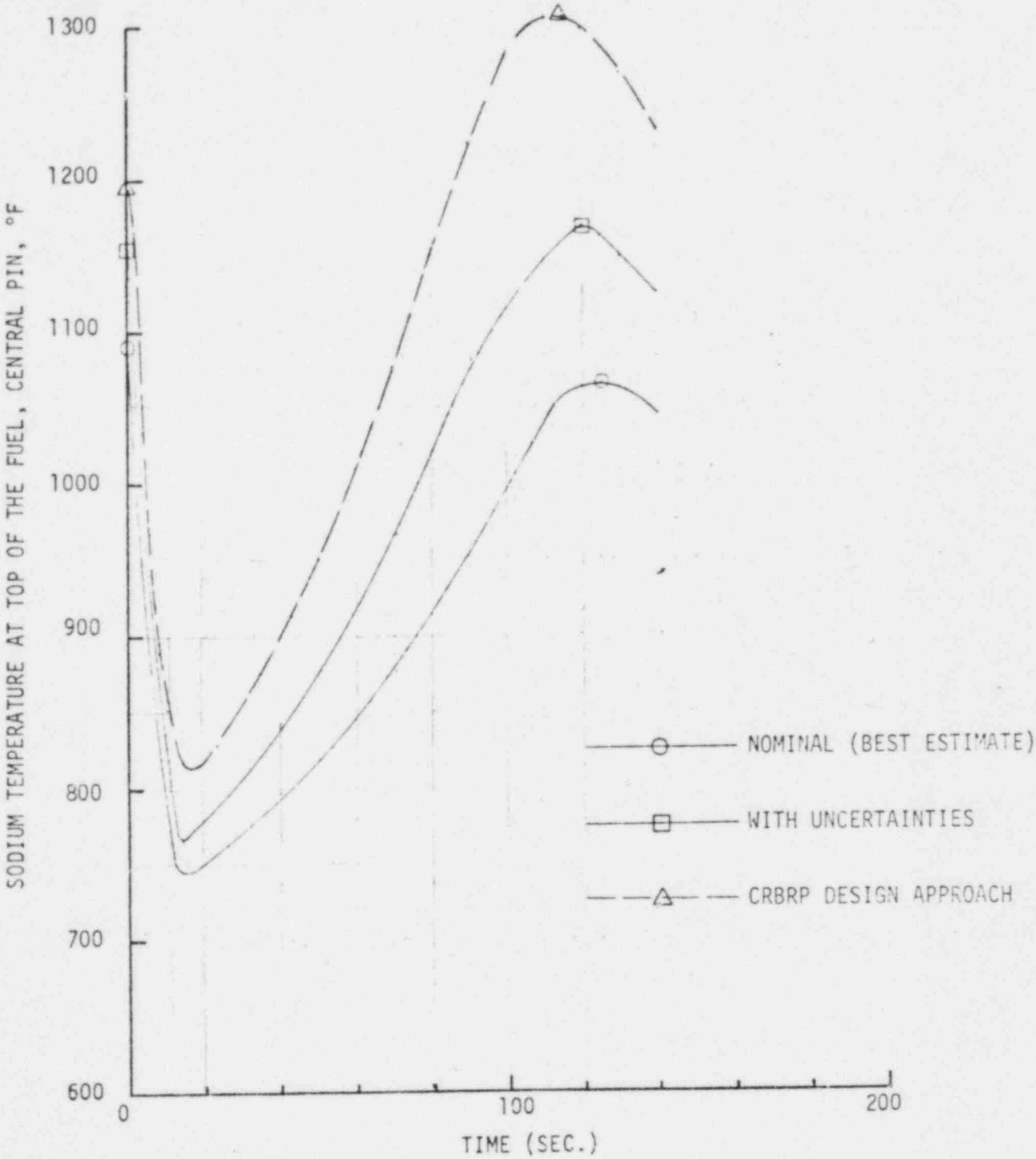


FIGURE 4-19: SODIUM TEMPERATURE PREDICTIONS AT TOP OF THE FUEL (HOT ROD IN CHANNEL 15) ROW 2 FOTA INITIATED AT 100% POWER/100% FLOW CONDITIONS (Tx1016)

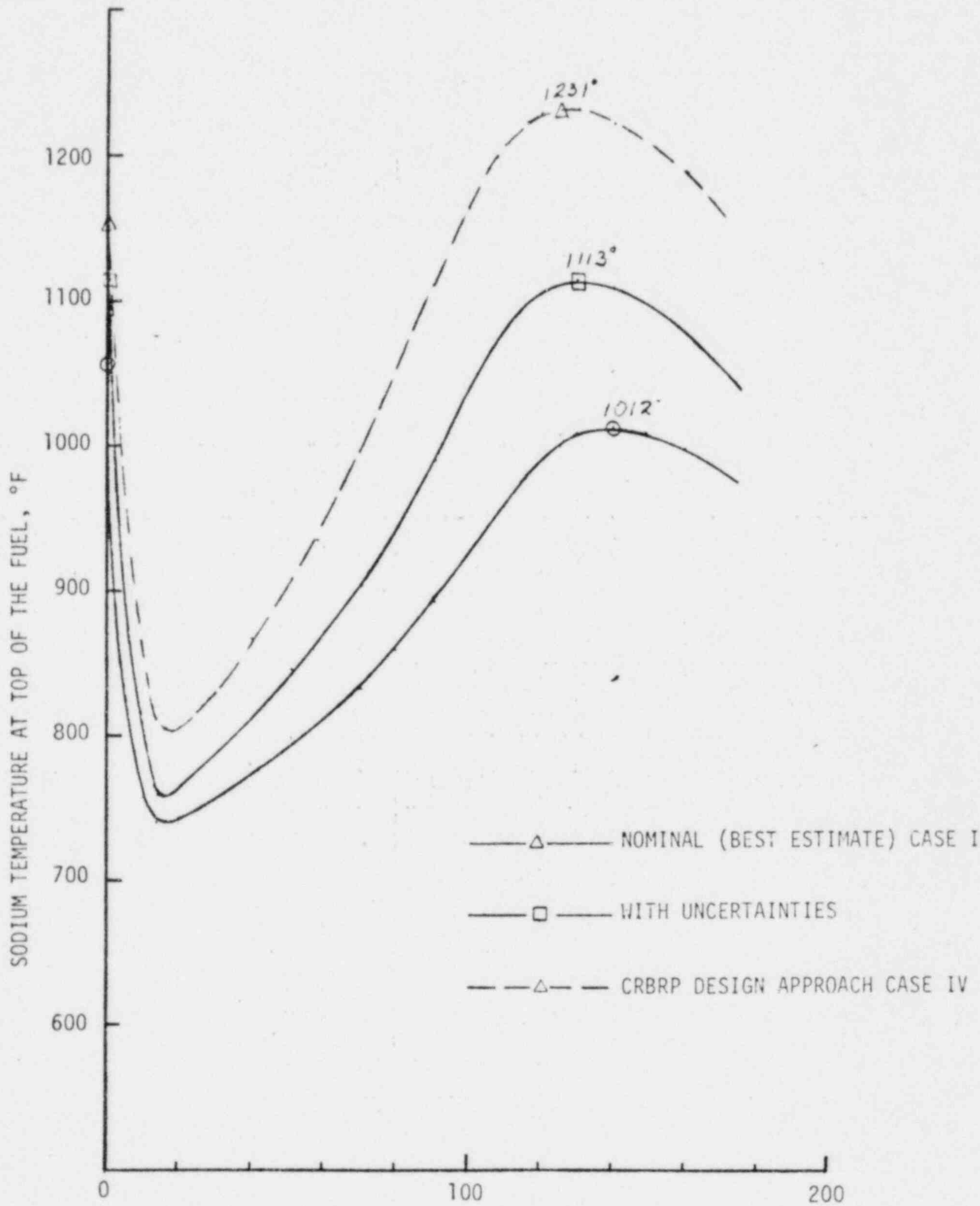


FIGURE 4-20: SODIUM TEMPERATURE PREDICTIONS AT TOP OF THE FUEL (HOT ROD IN CHANNEL 8) FOR ROW 6 FOTA INITIATED AT 100% POWER/100% FLOW CONDITIONS (Tx9018)

4.7 Sodium Temperature Predictions at the Top of the Active Fuel Section

Figure 4-19 shows the calculated sodium temperature with uncertainties, T_{UNC}, for the Row 2 FOTA, Channel 15 or thermocouple (Tx1016). Also shown in the figure are the nominal case (Case I) sodium temperature predictions initiated from 100% power and 100% flow conditions and the CRBRP design approach (Case IV) predictions. Similar temperature predictions for Row 6 FOTA, Channel 8 are shown in Figure 4-20. The actual temperatures at these channels would be expected to correspond with the curve representing the nominal case. As stated in the previous section, the CRBRP design approach, on the other hand, gives the most conservative temperature estimation because of assumed boundary conditions and extra margins with hot channel factors. The actual test temperatures should not exceed the T_{UNC} curve representing (3σ) uncertainty case. Thermocouple measurement uncertainties and time delays are not included in the predictions. The acceptance criterion for sodium temperatures can therefore be stated as follows: if the maximum measured temperature at specified channels (corrected for above instrument effects and actual boundary conditions) is less than the curve shown with uncertainties, T_{UNC}, it would follow that the DEMØ/CØBRA-WC/FØRE-2M calculations made with design data conservatively envelope the temperature response of the natural circulation test. The model and methodology would therefore be acceptable for such design transient predictions.

4.8 Qualitative Discussion of Other Uncertainties

This section addresses other uncertainties (from those described in Section 4.6) of the analyses performed for the 100% power/100% flow natural circulation test. The areas described have only recently become identified and are due to the benefits of the test data from the 35% power/75% flow teat and/or steady state operation in the 75% to 100% power range. The effects on the temperature predictions of these conditions are not incorporated in the pre-test predictions. However, they will be evaluated in the post-test analysis phase of the NCVP. It is anticipated that otheritems, which affect the pre-test predictions, will also be identified after the natural circulation testing is completed and as more operating data are accumulated. A general conclusion drawn is that the net effect of all the phenomena to be described will result in the pre-test predictions presented in this report (even on a nominal Case I basis) being conservative. The degree of conservatism cannot be quantitatively provided at this time.

Decay Heat

The latest FFTF test-specification revision for the 100% power/100% flow test indicates that 24 to 30 hours of full power operation are allowable prior to initiating the natural circulation test. Also, current estimates indicate that it will take in the order of 13 hours to reach full power. The decay heats used for the NCVP pre-test predictions assume a full power condition of 25 hours prior to the test (with no startup history accounted for). This has a small effect on the predictions, however, even if the worst case allowable times should be experienced prior to the test. For example, even if 36 hours would elapse at full power prior to the test, the decay heat at the time of the maximum Row 2 FOTA temperature would only be approximately 3% larger than the decay heat used. This would increase the maximum nominal temperature by less than 7°F. Such an effect is enveloped by the power uncertainties considered in Case III.

Reactor Outlet Plenum Stratification Model

Data from the 35% power/75% flow test indicate that the reactor outlet plenum has a fairly uniform axial temperature distribution from steady state through the first 120 seconds of the transient; outlet plenum stratification then starts. These temperature measurements extend from the vortex suppressor plate down to ~3 feet above the horizontal baffle. The data indicate that a homogeneous mixing model would be valid for the outlet plenum during the first 120 seconds before stratification occurs. The stratification model is needed at t≥120 seconds. The DEMØ code conservatively used a stratified type outlet plenum model throughout the transient. The effect of this approach is to have lower outlet plenum temperatures leaving the reactor outlet nozzle and entering the elevated outlet piping. As a consequence, the hot leg coolant is too cool relative to anticipated actual temperature and consequently, the buoyancy affected system flow rates would be under-predicted. On the other hand, if a homogeneous mixing model would have been used the system flow would be more accurately predicted up through the time of peak core temperatures (~120 seconds); after this time, the system flow would be over-predicted, which would cause an under-prediction of the FOTA temperatures after the peak temperature. In the post-test analyses, an improved outlet plenum model may be necessary. At that time, experimental data will be available for constructing a more phenomenological model for FFTF reactor outlet plenum mixing. The net effect on the pre-test predictions is that the current model should result in conservative FOTA temperature predictions due to the under-prediction of primary system flow rates.

In conclusion, several fact rs have been discussed which would make even the nominal case (Case I) predi.tions become conservative. Since these uncertainties cannot be quantified at this time, the lower bound of the estimated sodium temperature therefore cannot be specified.

5. PRE-TEST PREDICTIONS FOR NATURAL CIRCULATION TEST INITIATED AT 75% POWER AND 75% FLOW CONDITIONS

5.1 Cases Analyzed for Predictions

The five cases analyzed with the initial reactor condition of 75% power and 75% flow before shutdown are the same as presented in Section 4.1. The maximum irradiation history of 25 hours operation at 75% power is also assumed although in the actual test, a somewhat shorter irradiation time is anticipated. The parameters that are varying in different cases are, again, the decay heat and system pressure drop calculations. The effects on temperature response of uncertainties of these parameters is therefore determined by comparing results of different cases analyzed as in the previous cases.

5.2 DEMØ Input for 75% Power/75% Flow Initial Conditions

For the 75% power case the plant conditions at the beginning of the test are specified as:

1) Reactor power at 75% $\pm$ 1% of 400 MW;
2) Primary loop flow at 75% $\pm$ 1% of 13,443 gpm;
3) Primary cold leg temperature at 662 $\pm$ 5°F;
4) Primary hot leg temperature at 920 $\pm$ 5°F;
5) Secondary loop flow at 75% $\pm$ 1% of 13,200 gpm;
6) Secondary cold leg temperature at 602°F; and
7) All six heat transport systme (HTS) pony motors de-energized.

Here again, the transient will be initiated by scramming the reactor, causing a trip of all six HTS sodium pumps. While the pumps coast down to zero speed, the plant will undergo a transition to natural circulation flow.

Pressure drop uncertainties and decay heat uncertainty have the same meaning and were applied in the same manner as described in Section 4.2 for the 100% cases. The necessary dimensional input and initial conditions for these analyses can be found in the update listing in Appendix A.

5.3 DEMØ Results for 75% Power/75% Flow Initial Conditions

Results for the 75% power analyses are presented in Figures 5-1 through 5-10. Again, results for Cases I through IV are presented together while those for Case V are shown separately for the same reasons indicated in Section 4.3.

Figure 5-1 shows the predicted primary loop flow for Cases I through IV. The results are similar to those for the 100% case. Here the best estimate case (Case I) drops to a minimum flow of 1.9% (250 gpm) of rated conditions at 100 seconds before it recovers to around 2.6% (350 gpm). The drop in flow here is again caused by the reduction in thermal head resulting from the rather abrupt reduction in the power-to-flow ratio immediately after scram. The high side pressure drop uncertainties lower the predicted flow and the maximum decay heat uncertainty raises the predicted flow as in the 100% case. Figure 5-2 shows the same flow predictions for Case V. The predicted flow for Case V drops to a minimum of 1.5% (200 gpm) and reaches this minimum flow about 15 seconds earlier than the best estimate case.

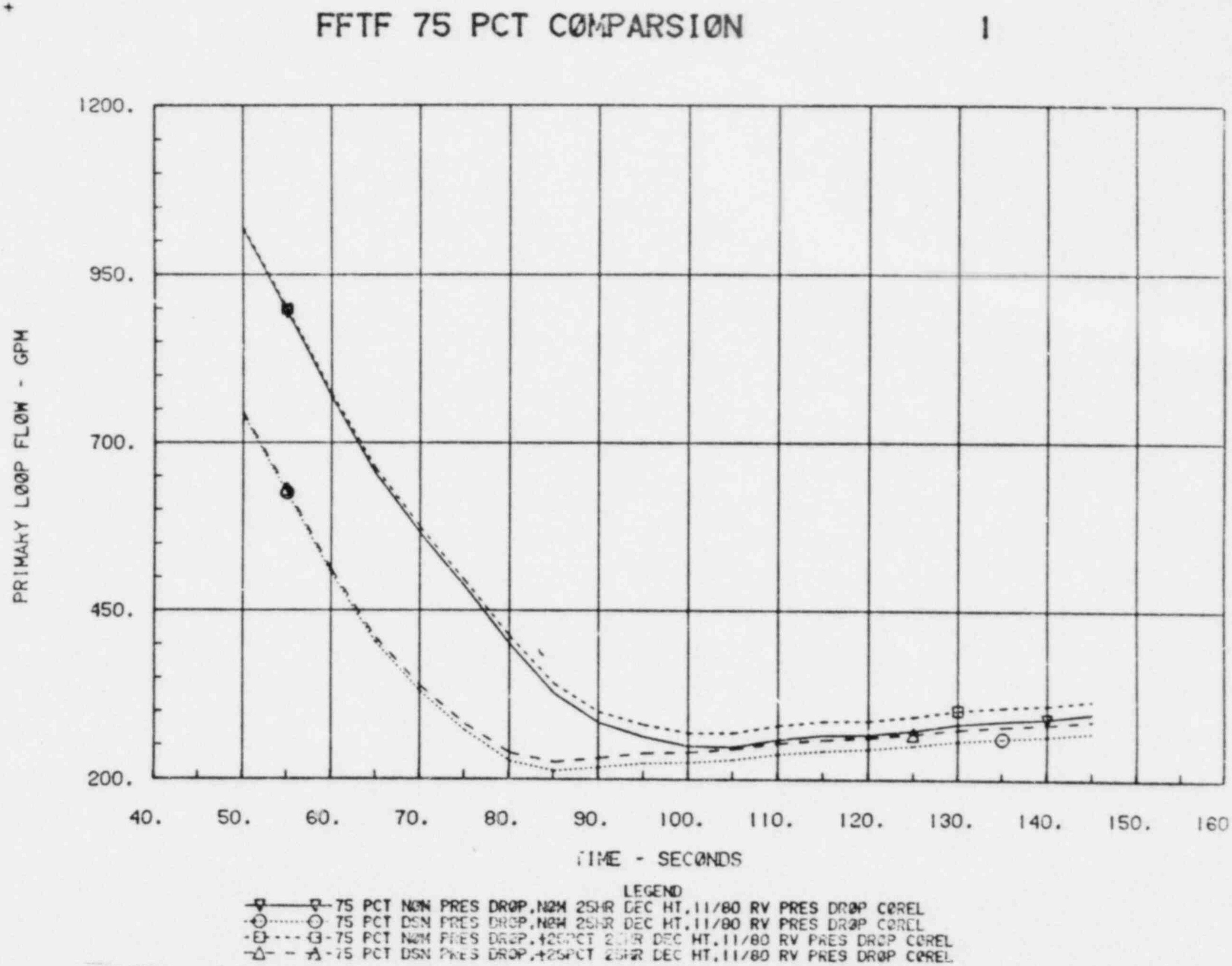


Figure 5-1 DEMO Predicted Primary Loop Flow at 75% Power/75% Flow Initial Conditions (No correction for instrument inaccuracy)

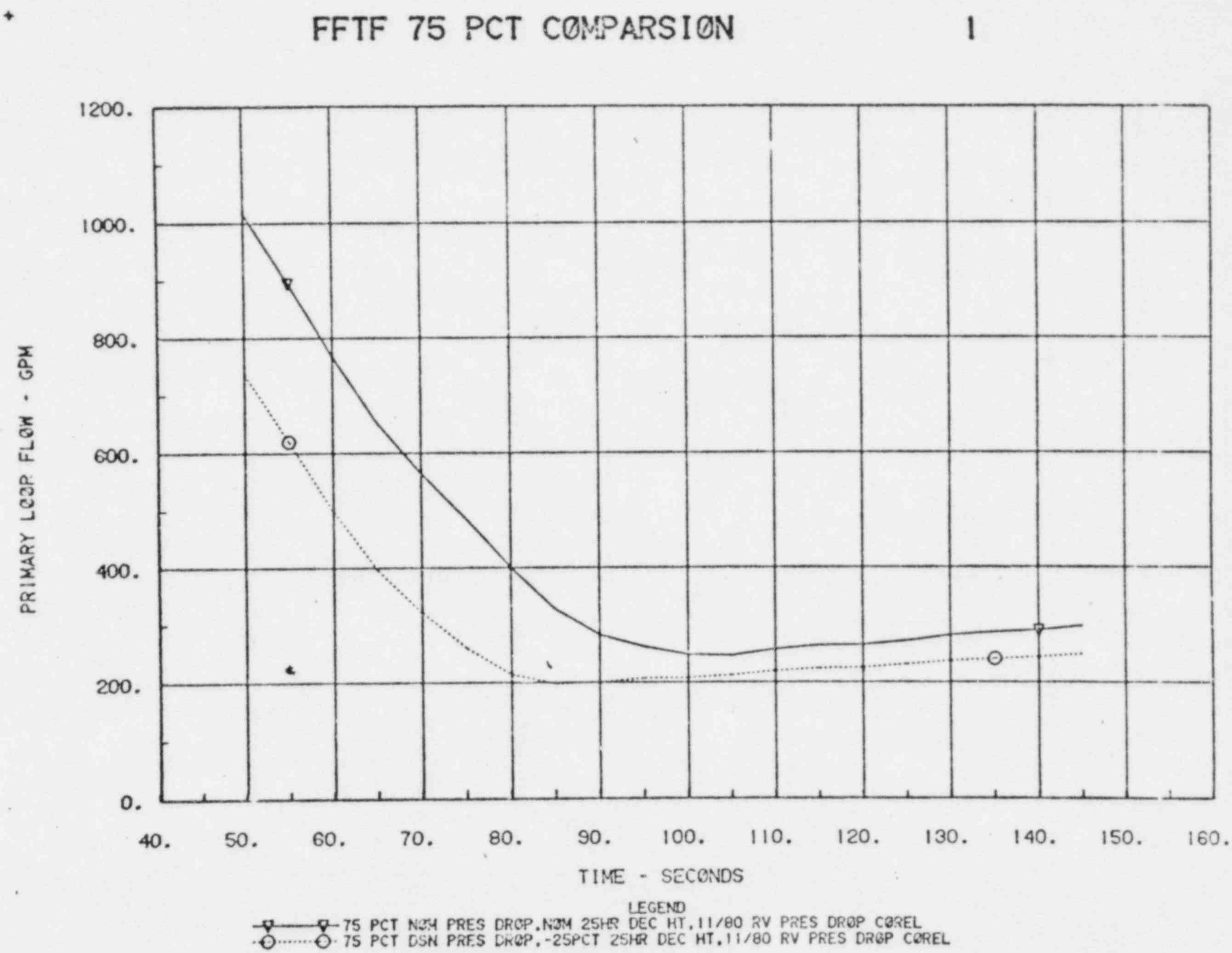


Figure 5-2 DEMO Predicted Primary Loop Flow at 75% Power/75% Flow Initial Conditions (No correction for instrument inaccuracy)

Figure 5-3 shows the predicted hot leg RTD temperatures for Cases I through IV, and Figure 5-4 shows the same temperature predicted by Case V. The shape of the transient curves produced by these runs are quite similar to those for the 100% case. These transients are caused by the rapid decrease in reactor outlet temperature mitigated through 65 feet of hot leg piping.

The primary cold leg RTD temperatures are shown in Figure 5-3 for the first four cases analyzed, and in Figure 5-6 for the fifth case. Again, the results are quite similar to those for the 100% case.

The acceptance limit curve for the 75% power/75% flow test is shown in Figure 5.7. This acceptance limit curve is constructed in the same manner as described for the 100% case by biasing the Case V calculated flow to account for the magnetic flow meter inaccuracies and time constant. This curve again represents the lowest acceptable limit for the measured flow under the specified test conditions. If the measured flow falls above this acceptance limit, DEMØ calculations would then be verified.

Figure 5-8 shows the predicted secondary loop flow for Cases I and IV. As with the 100% power case, this secondary flow remains higher than the primary flow. Figure 5-9 shows the secondary hot leg RTD temperatures and Figure 5-10 shows the secondary cold leg RTD temperatures. These curves are similar in shape to the 100% curves.

5.4 CØBRA-WC Results for 75% Power/75% Flow Initial Conditions

The results of CØBRA-WC analyses for the nominal case (Case I) are shown in Figures 5-11 and 5-12 in terms of axial flow fraction at the top and bottom nodes relative to the initial flow fraction (=0). Figure 5-11 shows this ratio for Row 2 FOTA initiated from 75% power/75% flow conditions. Figure 5-12 shows the ratio for Row 6 FOTA. The trends are similar to that of Section 4.4 (Figures 4-11 and 4-12), although the maximum ratio is about 6% lower in comparison with the case for 100% power/100% flow initial conditions. The lower buoyancy effect is expected because of the lower transient temperatures in the case for 75% power/75% flow initial conditions.

Figure 5-13 shows the typical variation in power data for the peak rod at the top node of the active fuel section for Row 2 FOTA under nominal conditions. As in Section 4.4, three distinctive periods can be similarily observed with respect to the mode of energy transfer from the rod to the peak channel.

5.5 FØRE-2M Results for 75% Power/75% Flow Initial Conditions

The results of FØRE-2M analyses for the three different cases (I, II and III) are shown in Figures 5-14 and 5-15. Figure 5-14 presents the sodium temperatures at the top of fuel section of Row 2 FOTA, Channel 15 (T/C Tx1016). Cases II and III with maximum uncertainty in system pressure drop or decay heat give rise to higher temperatures than nominal cases. As shown in Figure 5-1, the reactor flow reaches a minimum at a lower value and about 15 seconds earlier than the nominal case. Correspondingly, the peak channel sodium temperature reaches a maximum at a higher value and about 15 seconds earlier than the nominal case. For the case of maximum uncertainty decay heat, the sodium flow reaches a minimum at the same time (Figure 5-1). Similarly, the peak channel sodium temperatures in these two cases reach a maximum at the same time. Similar results are shown in Figure 5-15 for Row 6 FOTA, Channel 8 (T/C Tx9018).

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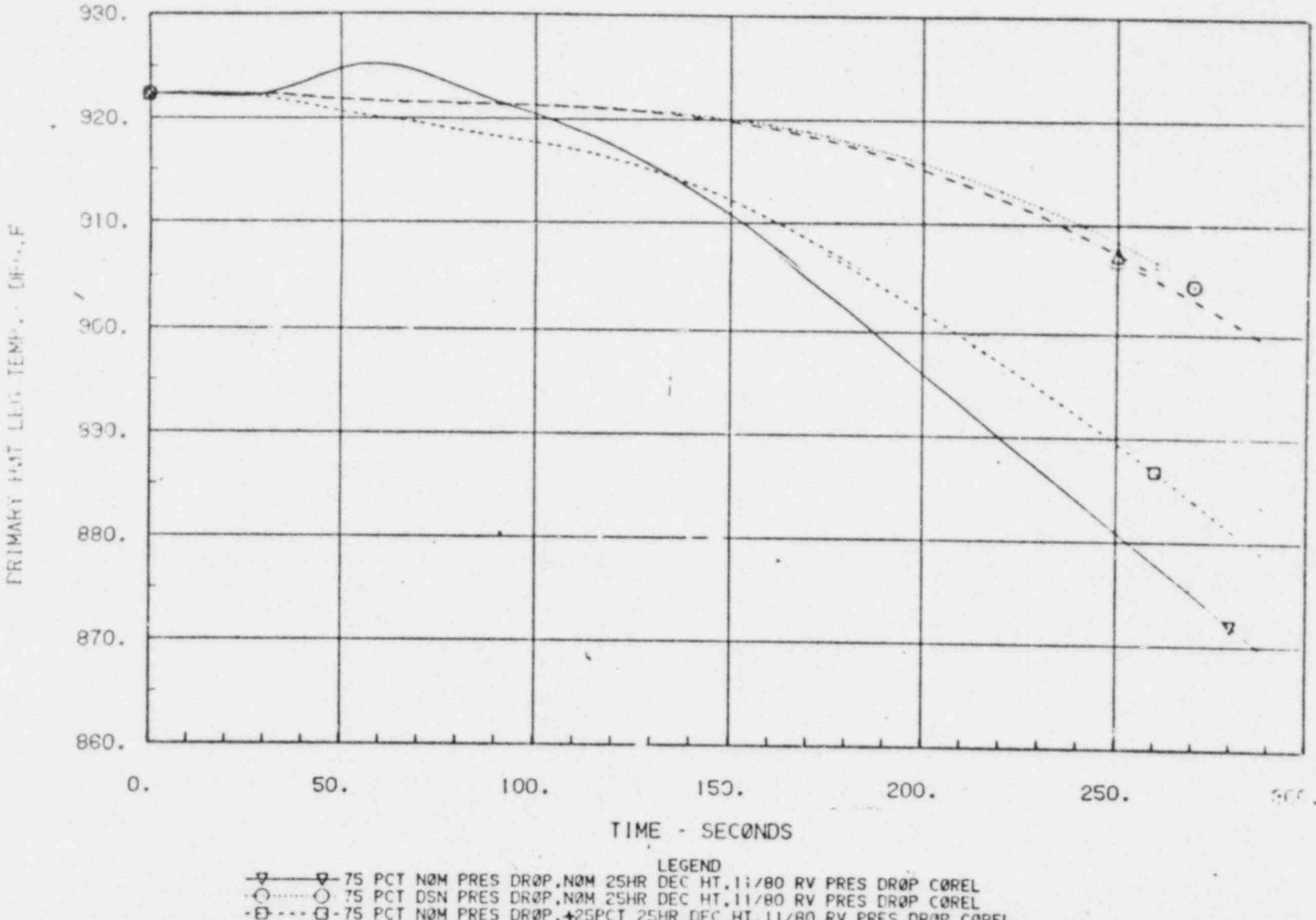

Figure 5-3 DEMO Predicted Primary Hot Leg RTD Temperature (75% Power/75% Flow Initial Conditions) (No correction for instrument inaccuracy)

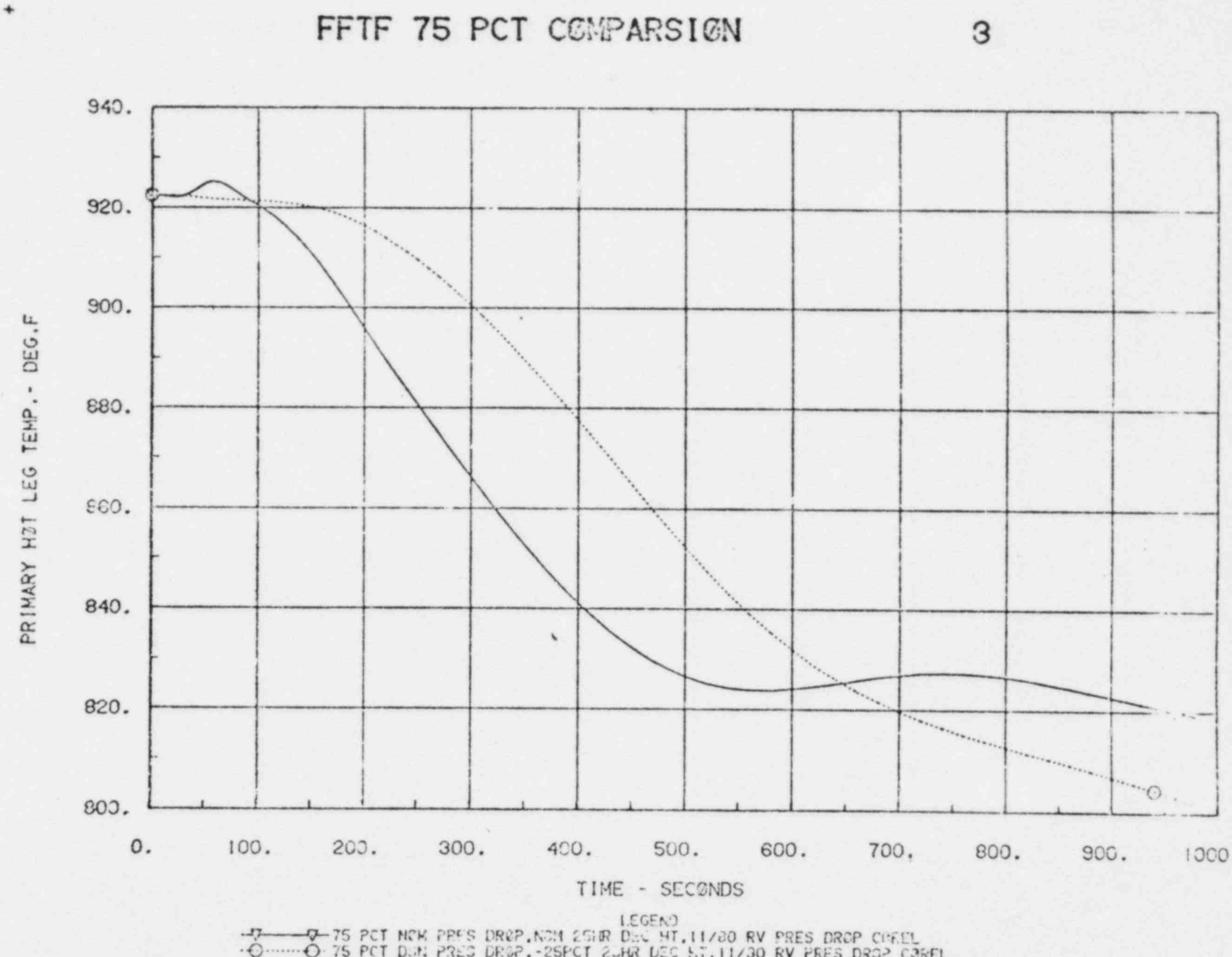

Figure 5-4 DEMO Predicted Primary Hot Leg RTD Temperature (75% Power/75% Flow Initial Conditions, Case V) (No correction for instrument inaccuracy)

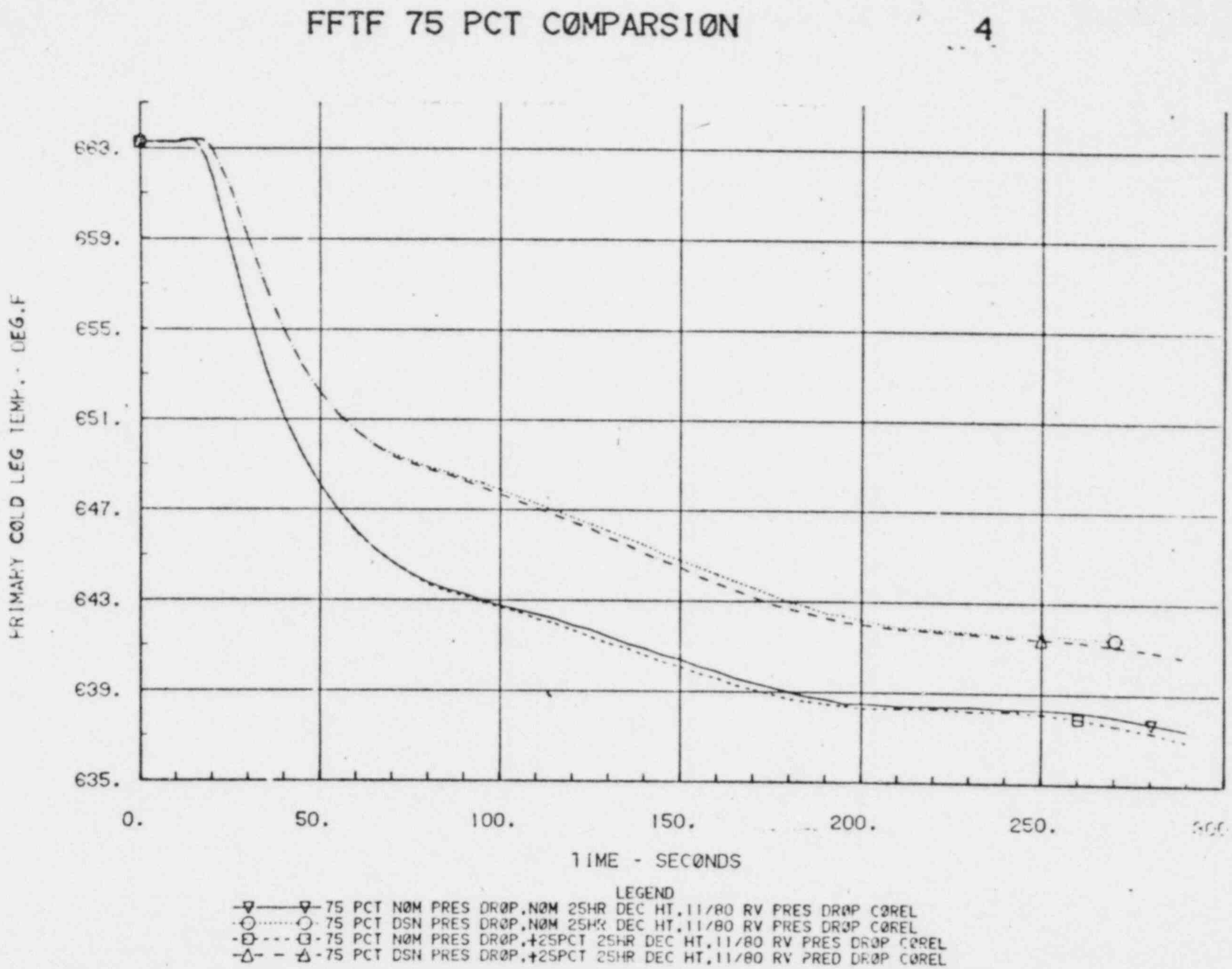

Figure 5-5 DEMO Predicted Primary Cold Leg RTD Temperature (75% Power/75% Flow Initial Conditions) (No correction for instrument inaccuracy)

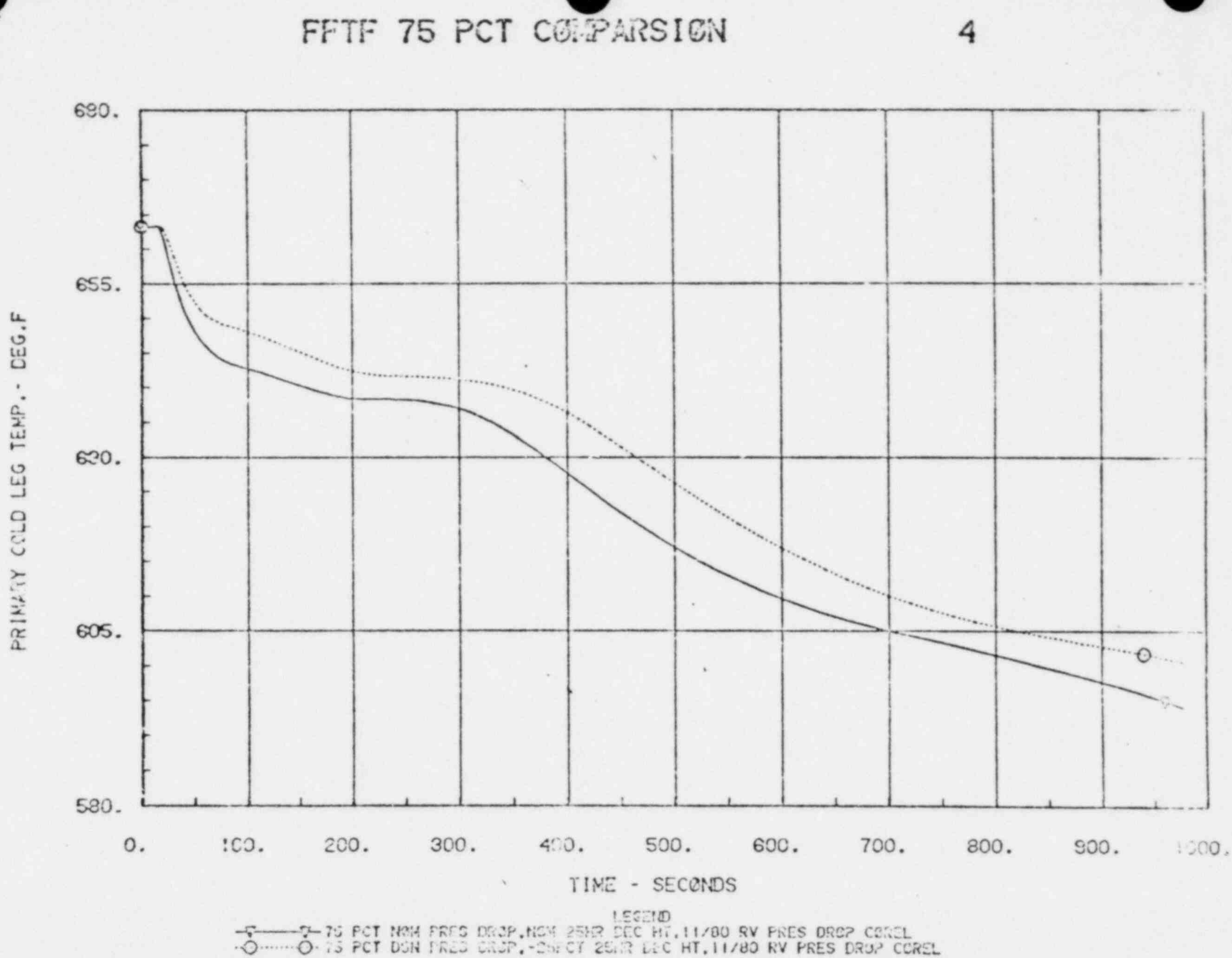

Figure 5-6 DEMO Predicted Primary Cold Leg RTD Temperature (75% Power/75% Flow Initial Conditions, Case V) (No correction for instrument inaccuracy)

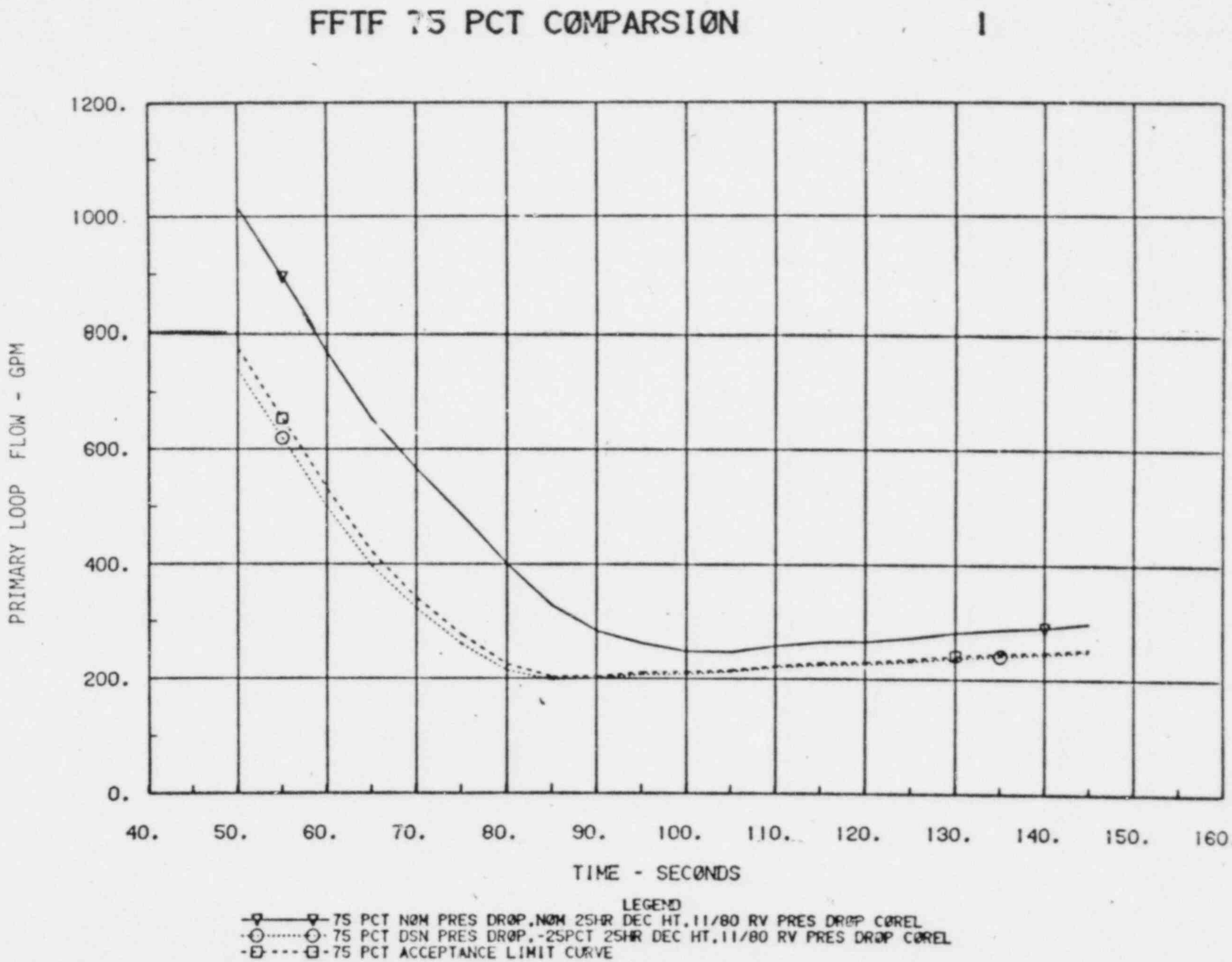

Figure 5-7 Acceptance Limit Curve for the 75% Power Case.

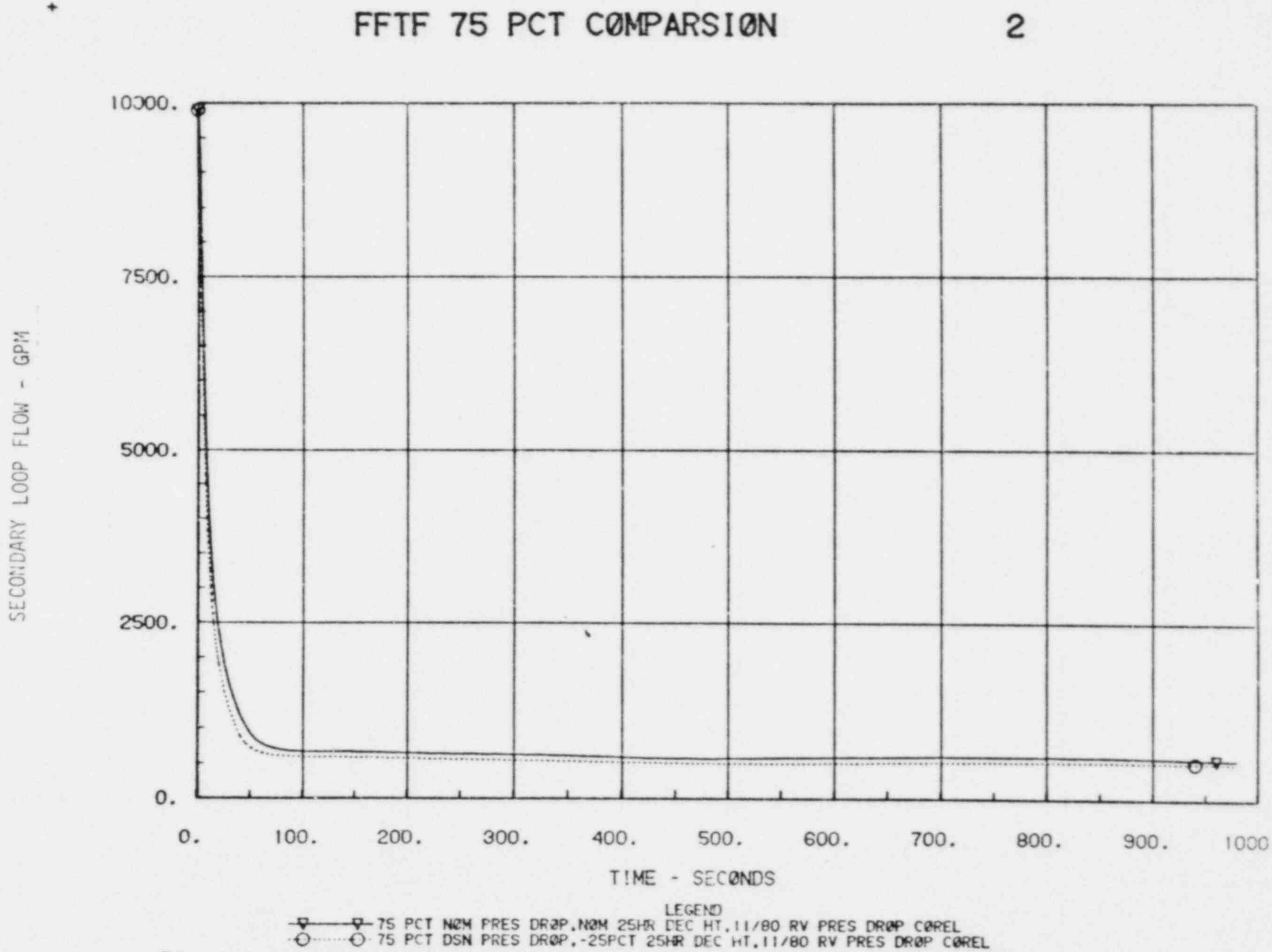

Figure 5-8 DEMO Predicted Secondary Loop Flow for 75% Power/75% Flow Initial Conditions
(No correction for instrument inaccuracy)

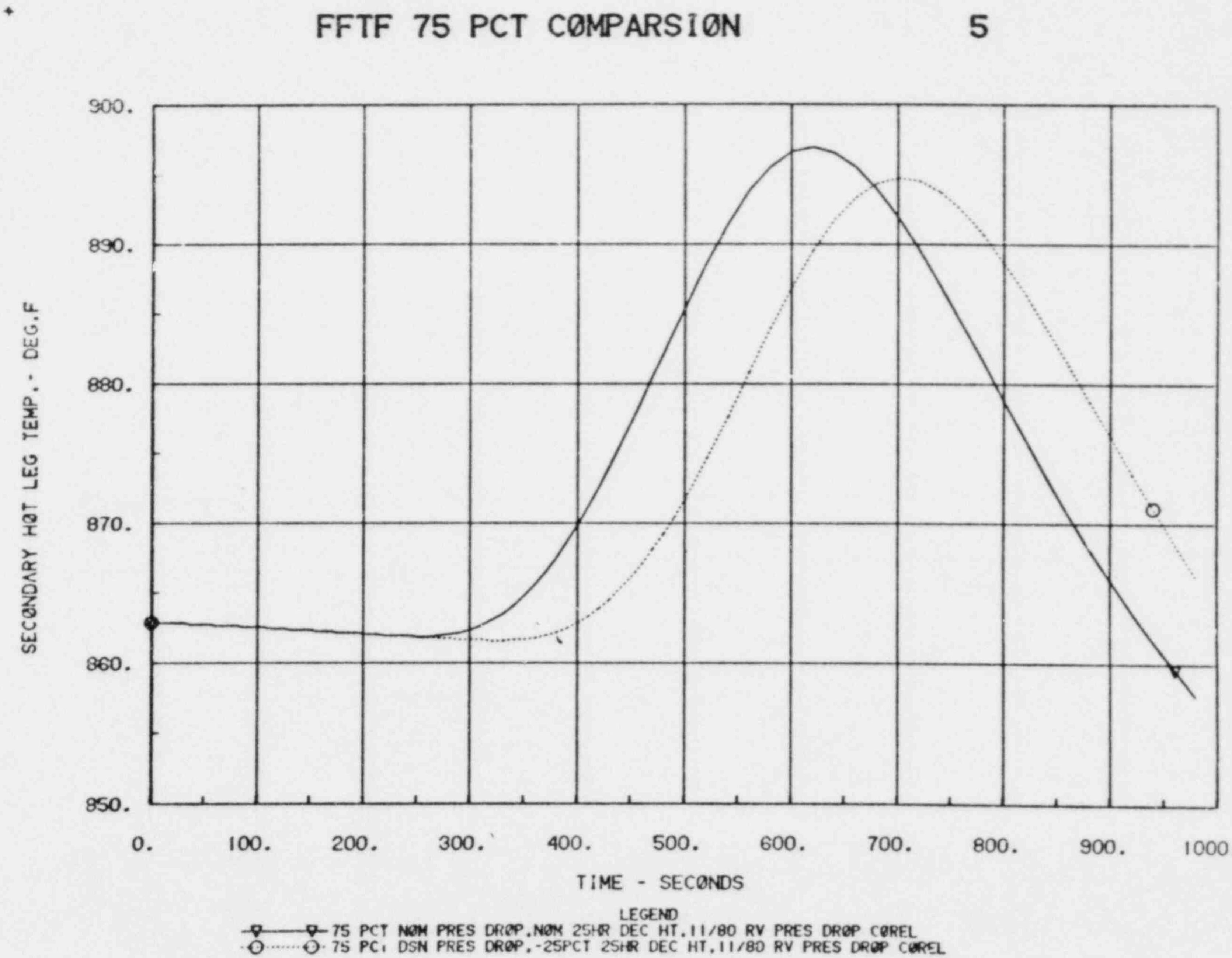

Figure 5-9 DEMO Predicted Secondary Hot Leg RTD Temperature (75% Power/75% Flow Initial Conditions) (No correction for instrument inaccuracy)

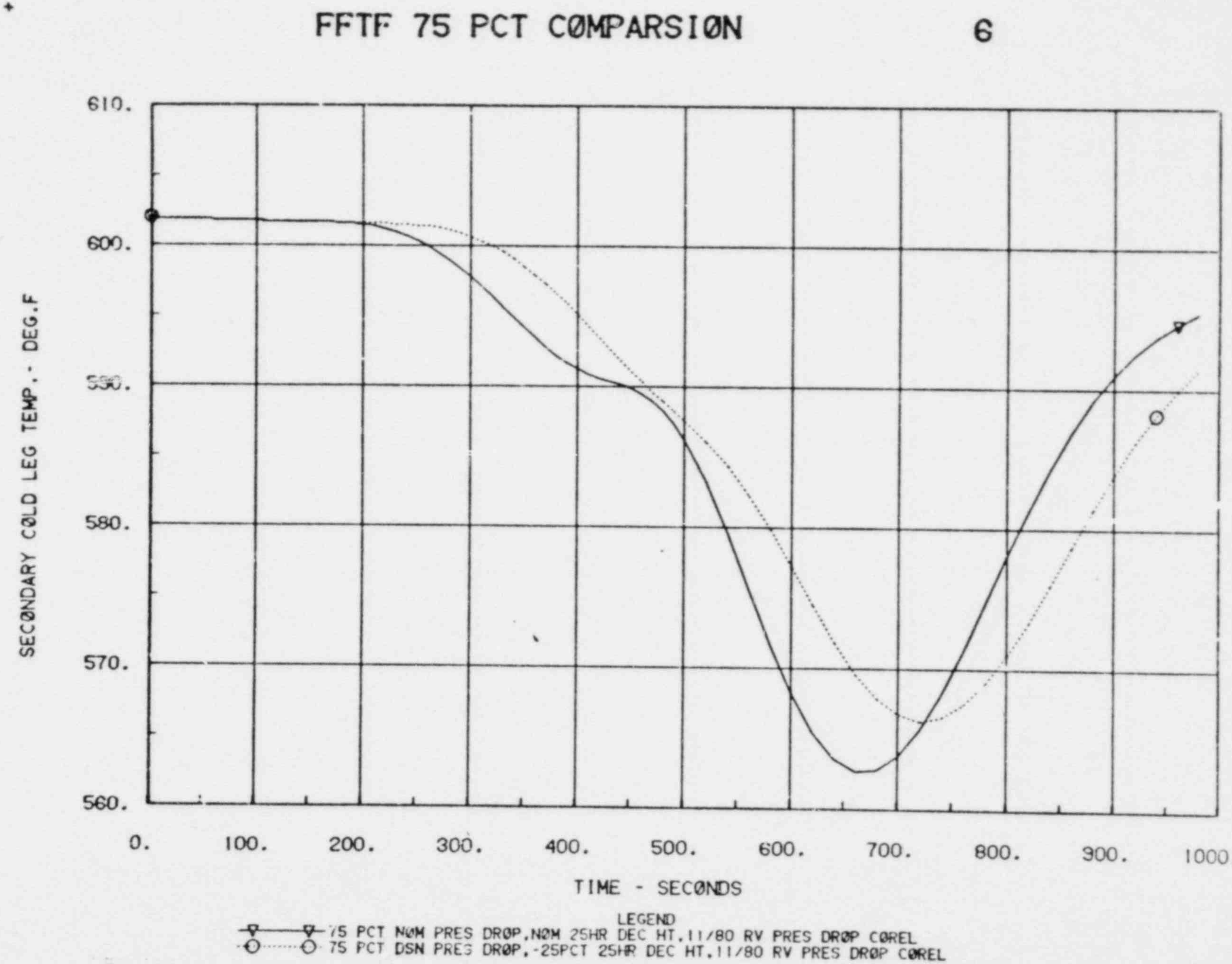

Figure 5-10 DEMO Predicted Secondary Cold Leg RTD Temperature (75% Power/75% Flow Initial Conditions) (No correction for instrument inaccuracy)

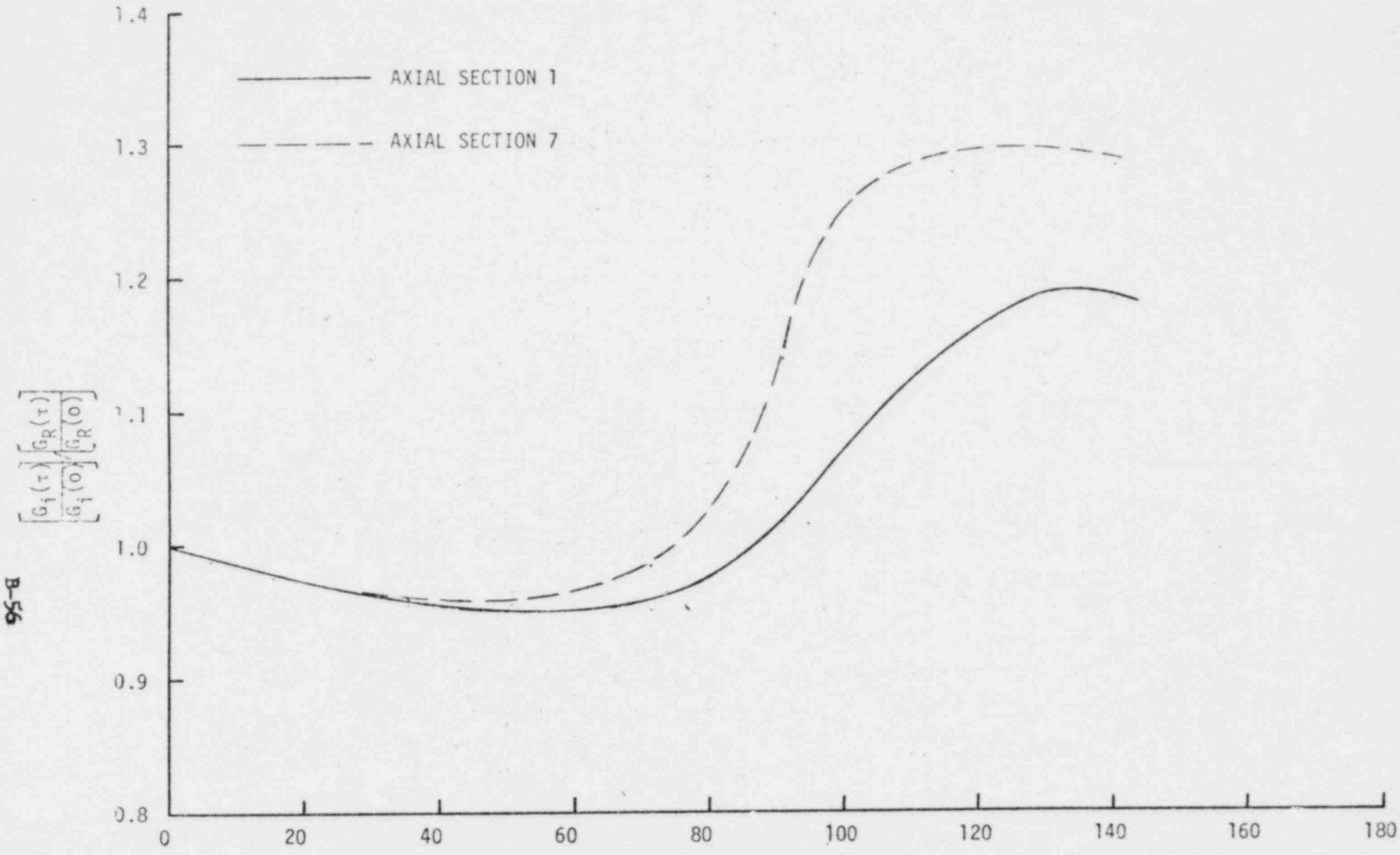

FIGURE 5-11: PEAK CHANNEL FLOW RELATIVE TO REACTOR FLOW COASTDOWN (ROW 2 FOTA NOMINAL CASE INITIATED AT 75% POWER/75% FLOW CONDITIONS)

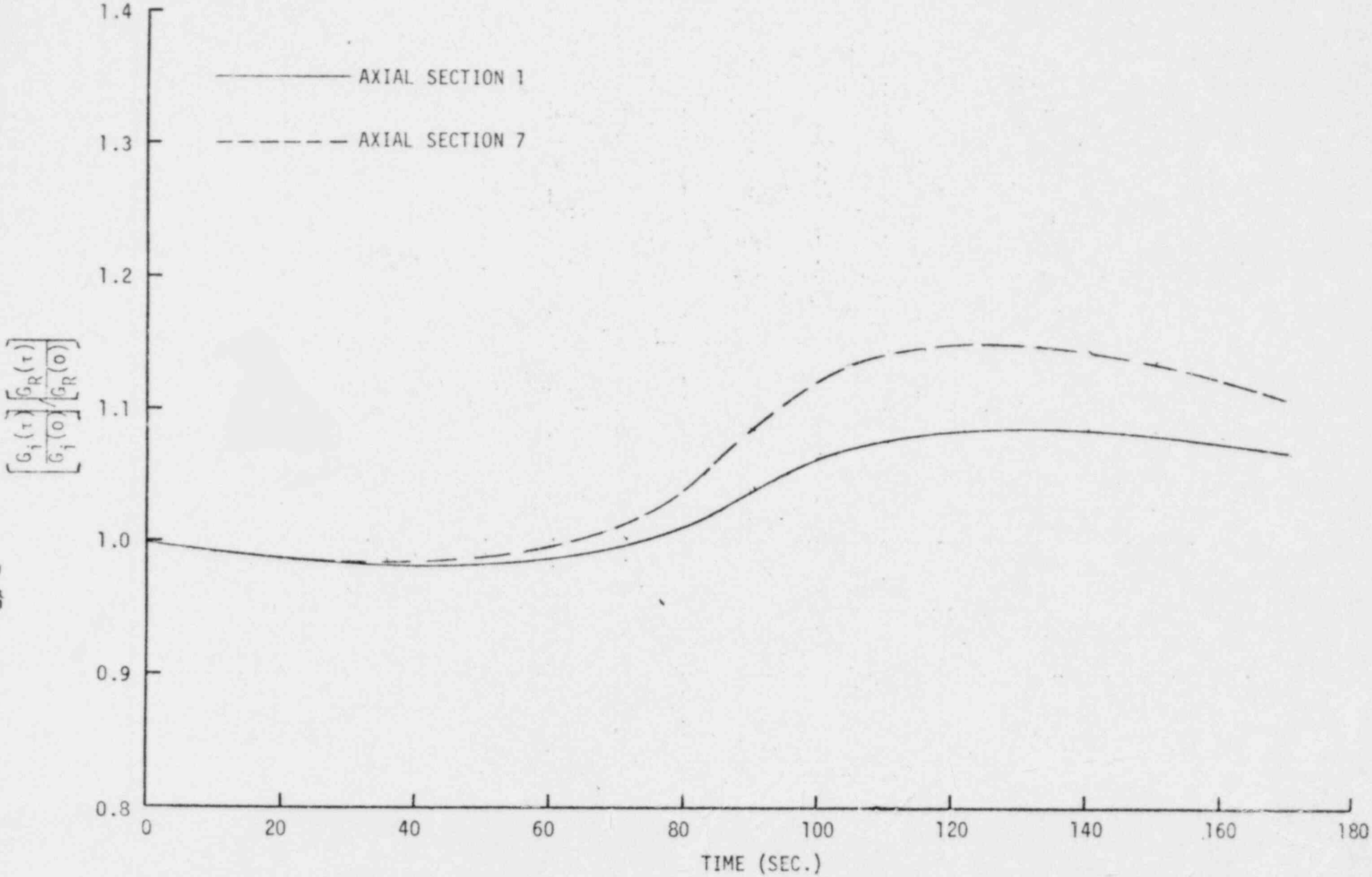

FIGURE 5-12: PEAK CHANNEL FLOW RELATIVE TO REACTOR FLOW COASTDOWN (ROW 6 FOTA NOMINAL CASE INITIATED AT 75% POWER/75% FLOW CONDITIONS)

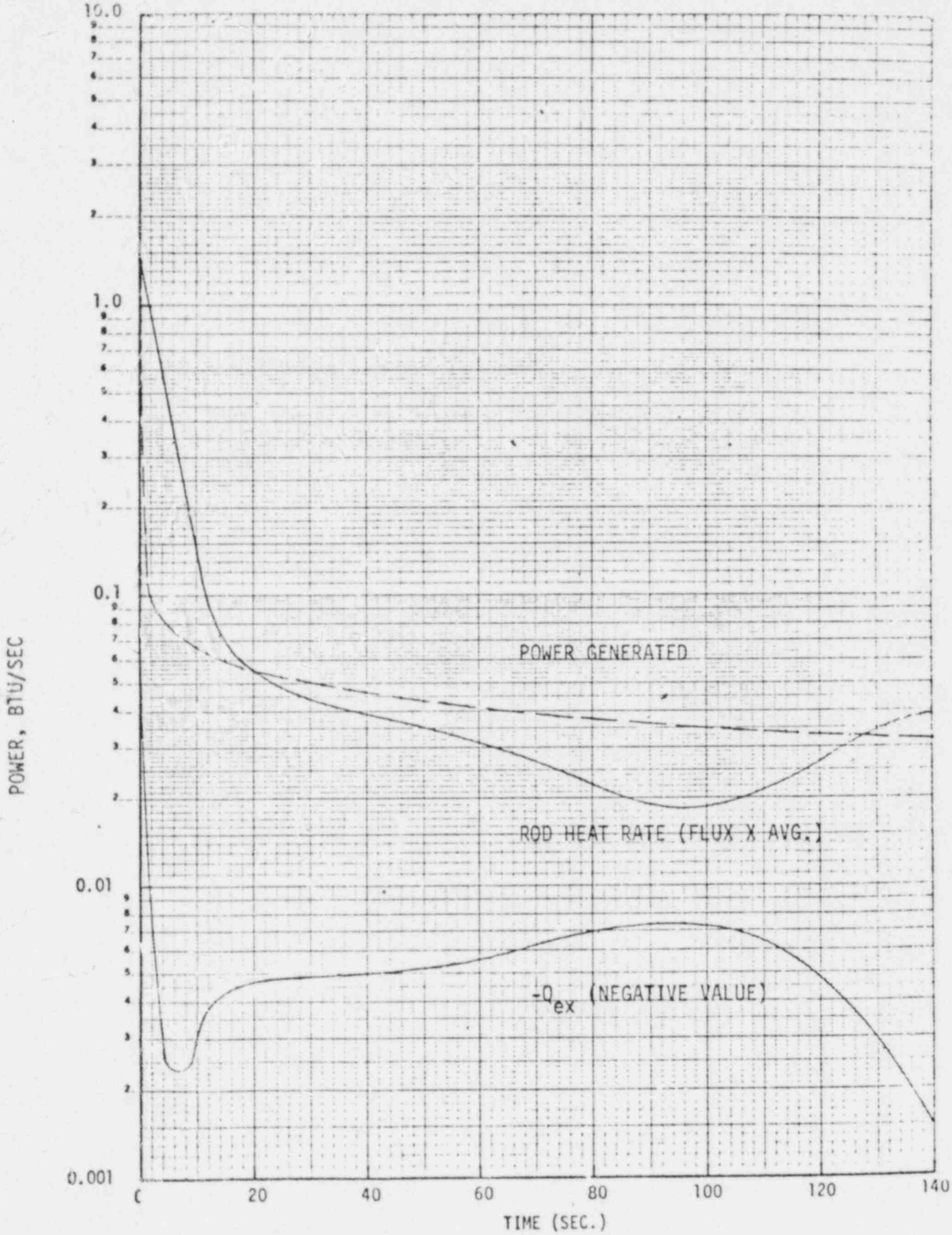

FIGURE 5-13: VARIATION IN POWER DATA FOR HOT ROD AT TOP NODE OF THE ACTIVE FUEL (ROW 6 FOTA) FOR THE NOMINAL CASE INITIATED AT 75% POWER/75% FLOW CONDITIONS

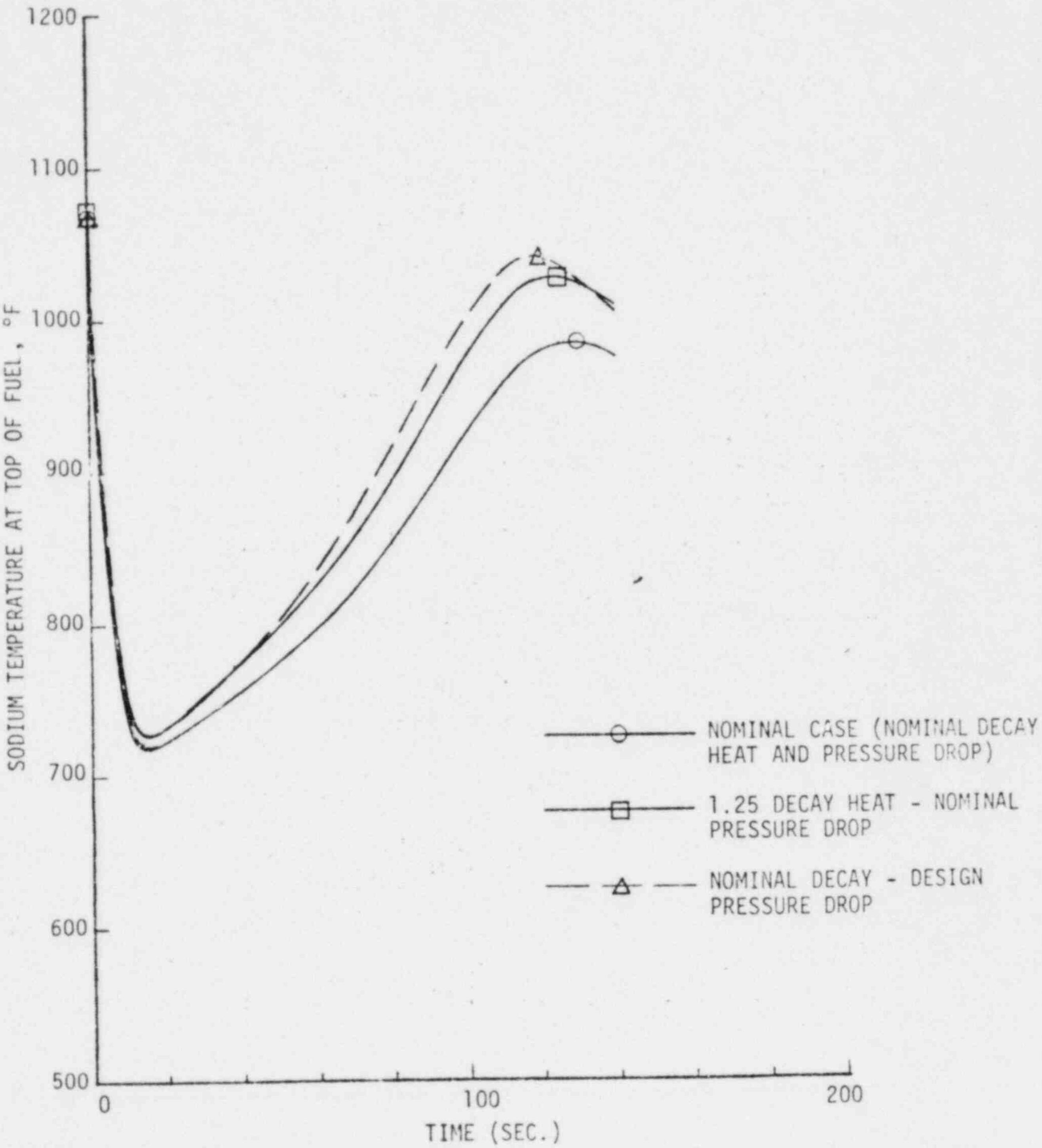

FIGURE 5-14: CALCULATED SODIUM TEMPERATURE AT TOP OF THE FUEL (CHANNEL 15) FOR ROW 2 FOTA WITH DIFFERENT CASES INITIATED AT 75% POWER/75% FLOW CONDITIONS (Tx101C)

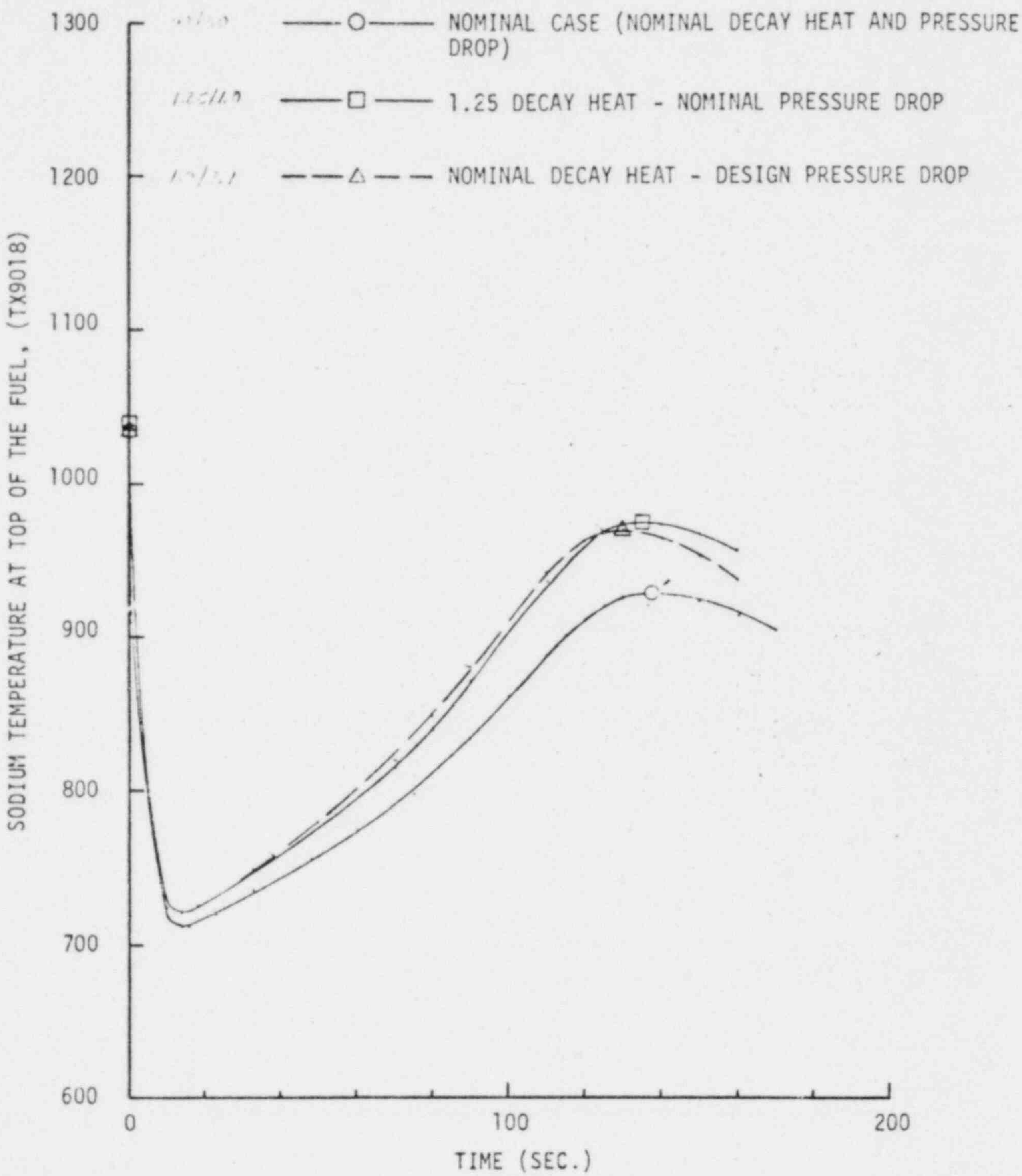

FIGURE 5-15: CALCULATED SODIUM TEMPERATURE AT TOP OF THE FUEL SECTION (CHANNEL 8) FOR ROW 6 FOTA WITH DIFFERENT CASES INITIATED AT 75% POWER/75% FLOW CONDITIONS (Tx9018)

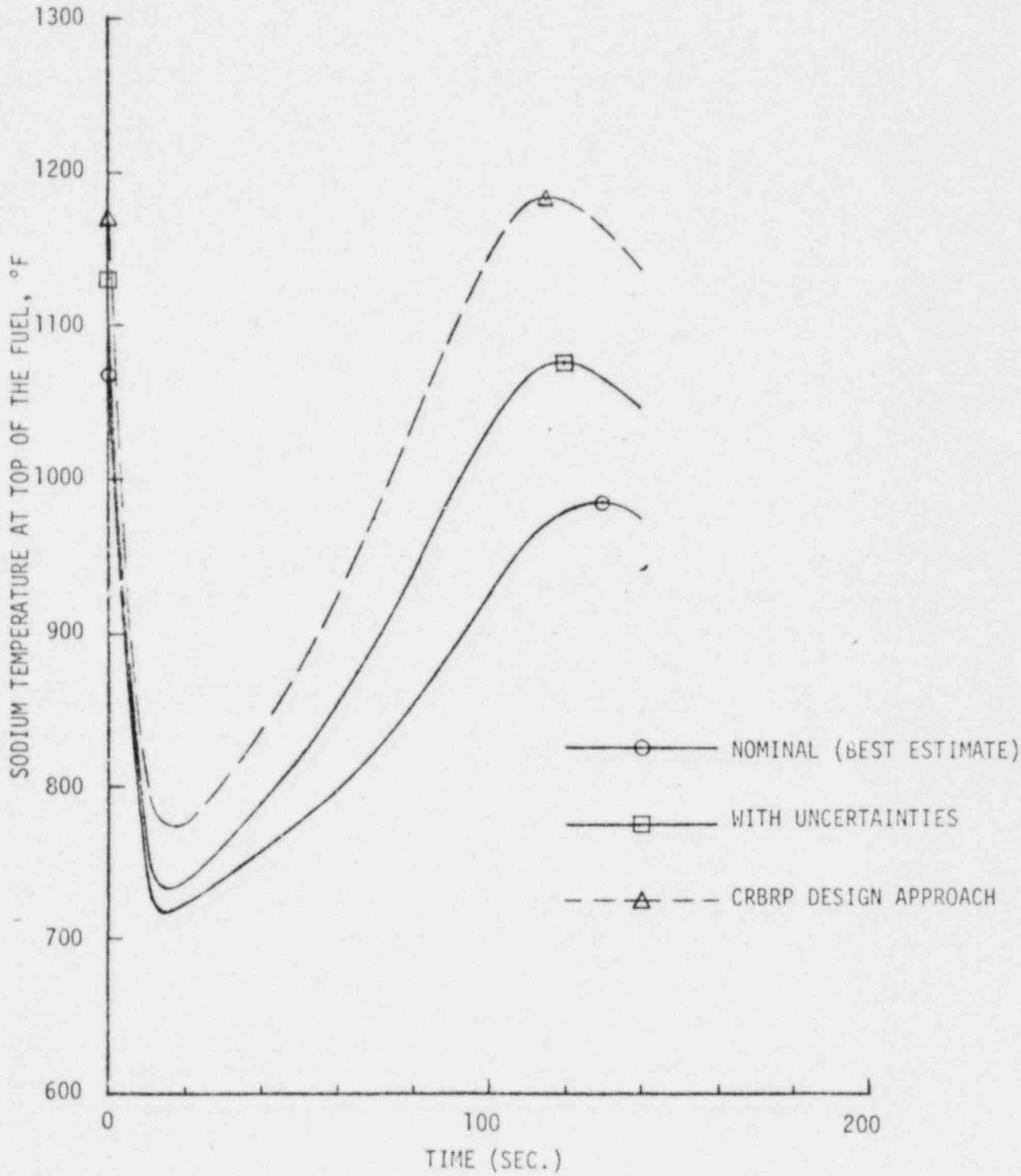

FIGURE 5-16: CALCULATED SODIUM TEMPERATURE PREDICTIONS AT TOP OF THE FUEL SECTION (HOT ROD IN CHANNEL 15) FOR ROW 2 FOTA INITIATED AT 75% POWER/75% FLOW CONDITIONS (Tx1016)

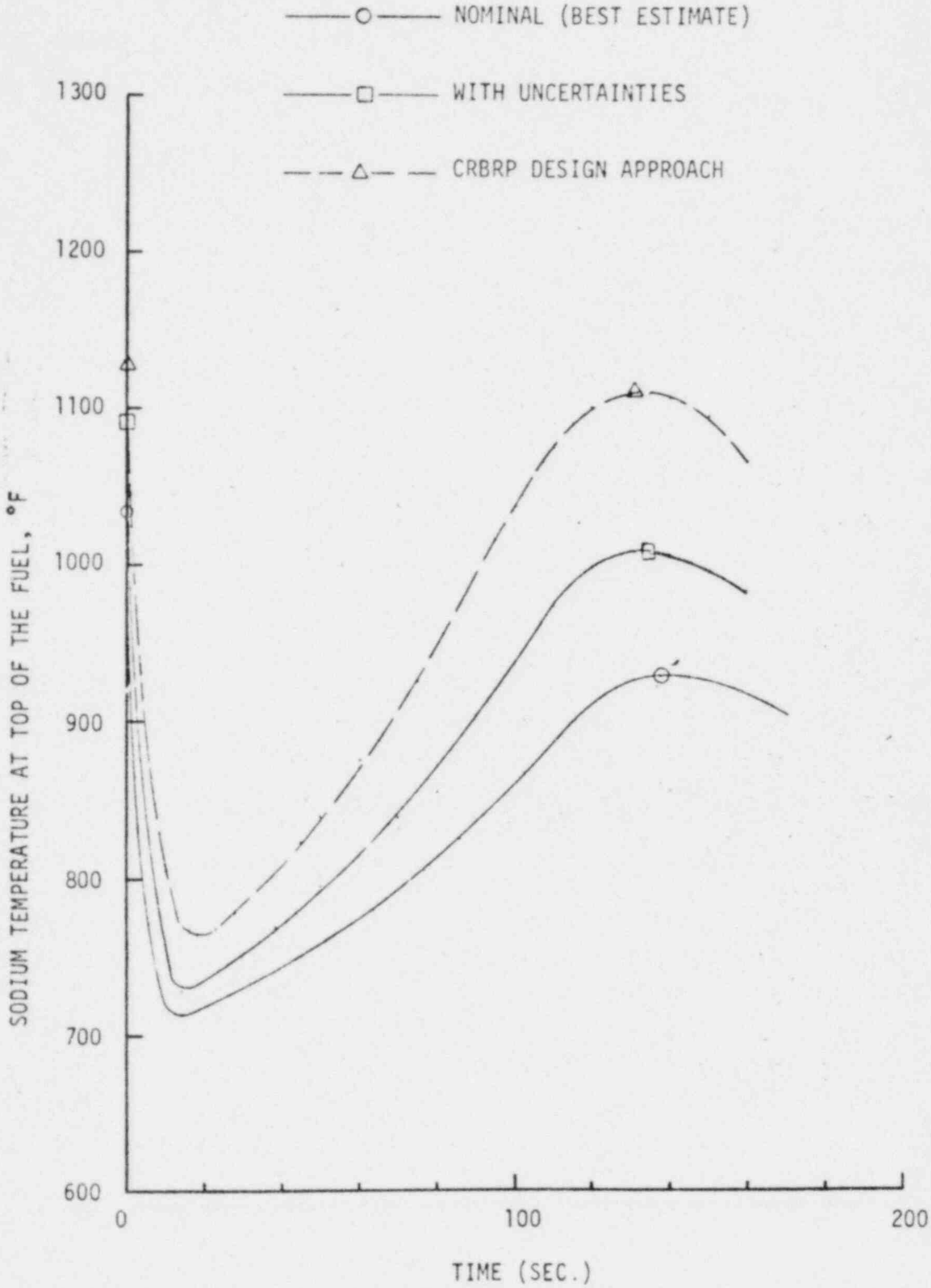

FIGURE 5-17: SODIUM TEMPERATURE PREDICTIONS AT TOP OF THE FUEL (HOT ROD CHANNEL 3) FOR ROW 6 FOTA INITIATED FROM 75% POWER/75% FLOW CONDITIONS (Tx9018)

The sodium temperature estimation for Case IV with maximum uncertainties in CRBRP design approach is shown in Figure 5-16 and 5-17. As in Section 4.5, Case IV represents the most conservative temperature predictions from the design point-of-view.

As can be noted by comparing Figures 5-16 and 4-19, hot rod temperatures for 75% power/75% flow initial conditions are significantly less than that for 100% power/100% flow initial conditions. Although both were initiated from power-to-flow equal unity conditions and proportional decay heats were used, the difference is more than that due to core inlet temperature between the two cases. This is because the primary system flow at natural circulation is not proportional to the decay powers (see Figures 4-4 and 5-1). The factor affecting primary system flow (for the time at which peak temperature is attained) is the steady state primary system hot-to-cold leg ΔT. Since this ΔT is approximately the same for both the 100% and 75% initial conditions, the system flow is rearly the same for both, but the decay power is only 75% for the latter case. Thus, significantly lower temperatures result.

5.6 Effects of Known Uncertainties on the Sodium Temperature Response

From the above FØRE-2M results for Cases I, II and III, the effects of uncertainties in decay heat and pressure drop calculations on the "hot channel" sodium temperature are determined in the same manner as in Section 4.5. Table 5-1 shows the comparison of these effects with different initial conditions for Rows 2 and 6 FOTA's. The hot channel effect of these uncertainties is different for the two FOTA's but less dependent on the initial conditions; 75% and 100% initial power/flow.

5.7 Sodium Temperature Predictions at the Top of the Active Fuel Section

Figure 5-16 shows the calculated sodium temperature T_{UNC} for the Row 2 FOTA, Channel 15 or thermocouple (Tx1016). Also shown in the figure are the nominal case (Case I) sodium temperature predictions initiated from 75% power/75% flow conditions and the CRBRP design approach (Case IV) predictions. Similar temperature predictions for thw Row 6 FOTA Channel 8 are shown in Figure 5-17. As in the previous case initiated at 100% power and 100% flow conditions, the actual temperatures at these channels would be expected to correspond with the curve representing the nominal case.

The CRBRP design approach gives the most conservative temperature estimation because of assumed boundary conditions and extra margins with hot channel factors. The actual temperatures should not exceed the T_{UNC} curve representing (3σ) uncertainty case for the test. The same acceptance criterion for the sodium temperature prediction as stated in Section 4.7 applies here.

TABLE 5-1

COMPARISON EFFECTS OF UNCERTAINTIES IN DECAY HEAT AND PRESSURE DROP CALCULATIONS ON SODIUM TEMPERATURE

		SODIUM TEMPERATURE RISE, °F				
FOTA	INITIAL CONDITIONS	NOMINAL CASE (CASE I) ΔT_1 @ PEAK	HIGH DECAY POWER NOMINAL PRESSURE DROP (CASE III) ΔT_2 @ PEAK	DESIGN PRESSURE DROP NOMINAL DECAY POWER (CASE II) ΔT_3 @ PEAK	DECAY HEAT UNCERTAINTY EFFECT $(\Delta T_2/\Delta T_1)_{max}$	PRESSURE DROP UNCERTAINTY EFFECT $(\Delta T_3/\Delta T_1)_{max}$
Row 2	100% P/100% F	401.7	447.9	457.4	1.12	1.14
	75% P/75% F	329.7	372.9	378.9	1.13	1.15
Row 6	100% P/100% F	350.2	408.7	397.6	1.17	1.14
	75% P/75% F	274.0	321.0	311.1	1.17	1.14

5.8 Qualitative Discussion of Other Uncertainties

This discussion for the 75% power/75% flow uncertainty which have not been included in Section 5.1 would be the same as that provided for the 100% power/100% flow initial conditions in Section 4.8.

As indicated in Section 4.8, even the nominal analyses (Case I) are conservative due to the described phenomenon which are known to be occurring, but are not included in the NCVP pre-test predictions. These effects were not modeled due to either their being defined after the analyses were completed or that they are not quantitatively defined at the current time. The effects will be evaluated in the post-test analyses phase.

The 75% power/75% flow analyses are even more conservative than those for the 100% power/100% flow initial conditions due to the assumed 25 hour initial operation at 75% power/75% flow in the analyses. The current test specification for the transient indicates that this period will be maximum of 4.5 hours. This results in a decay power that is in the order of 14% too large at the time maximum temperatures are attained (i.e., ~120 seconds after shutdown). As a result, a temperature over-prediction in the order of 25°F would be expected for the Rows 2 and 6 FOTA's.

6. CONCLUSIONS

The following conclusions can be made from the results obtained in this report:

A) The transients for the nominal cases studied are very mild since the expected peak transient ΔT's are less than or only slightly higher than the initial steady state temperature rises as shown by Case I in Figures 4-16, 4-17, 5-13 and 5-14 for the two different FOTA's. This is true for the natural circulation transient initiating from either 100% power/100% flow or 75% power/75% flow conditions.

B) The accuracy of the predictions is demonstrated in terms of hot channel factors as conventionally used in reactor design. By statistically considering 3σ uncertainties, the maximum temperatures of the two FOTA's are shown by Figures 4-19, 4-20, 5-16 and 5-17 for the 100% power/100% flow and 75% power/75% flow initial conditions. The probability of measured temperatures in the respective FOTA's exceeding these curves should therefore be less than 0.1%.

C) The CRBRP design approach predictions shown by Case IV on Figures 4-19, 4-20, 5-16 and 5-17 are very conservative with respect to the nominal temperatures of the FFTF FOTA's under natural circulation conditions.

D) Inter- and intra-assembly flow and heat redistribution effects within the core become significant under natural circulation conditions. The example shown in Section 4.4 during the 100% power/100% flow transient indicates for the Row 2 FOTA hot rod that:

. inter-assembly flow redistribution amounts to as much as 25% of the initial flow fraction;
. intra-assembly flow redistribution is as much as a 12% increase of the initial flow in addition to the inter-assembly flow redistribution;
. inter- and intra-assembly heat redistribution at the top coolant axial node position of the core amounts to 33% of the heat from the hot rod or 19% of the heat generated in the node at the time when the (Q_{ex}) reaches the maximum. The integrated value over the entire length of the fuel section is about 6% of the total heat generated at this time.

The above effects are beneficial in reducing the maximum temperature. They become even more beneficial for higher decay power cases.

E) As described in Sections 4.8 and 5.8, even the nominal analyses predicted herein are anticipated to be conservative due to many factors which have been identified. These effects will be quantitatively evaluated during the NCVP post-test analysis phase.

By demonstrating that the pre-test predictions provides: 1) a reasonable characterization of the phenomena controlling natural circulation flow and heat transfer in both the reactor and the heat transfer loops; and 2) a conservative estimate of the critical plant parameters, the CRBRP Project methodology used here will be verified as conservative and acceptable.

7.0 REFERENCES

1. FFTF TS-51-5A008, Rev. 4, "Specification for Transient Natural Circulation Test", November 1980.

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4. T. L. George, et al., "CØBRA-WC: A Version of CØBRA for Single-Phase Multi-Assembly Thermal-Hydraulic Transient Analysis", PNL-3259, July 1980.

5. J. V. Miller and R. D. Coffield, "FØRE-2M: A Modified Version of FØRE-II Computer Program for the Analysis of LMFBR Transients", CRBRP-ARD-0142, November 1976.

6. R. M. Singer and J. L. Gillette, "Measurements of Subassembly and Core Temperature Distributions in an LMFBR", AIChE Symposium Series, 73, p. 97 (1977).

7. A. K. Agrawal, et al., "Dynamic Simulation of LMFBR Plant Under Natural Circulation", 18th National Heat Transfer Conference, ASME Paper 79-HT-6, August 1979.

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9. "User's Guide for Irradiation Experiments in the FTR", HEDL-MG-22, Rev. 2, May 1978.

10. H.P. Planchon, W.R. Laster and R. Calvo, "DEMØ Pre-Test Predictions for the FFTF Transient Natural Circulation Tests", WARD-94000-00321, March 1980.

11. P.M. McConnell, "Results of Full Range Pressure Drop Tests Performed on Prototypic FFTF Shield Orifice Assemblies", TC-1608, May 1980.

12. Y.S. Tang, R.D. Coffield and R.A. Markley, "Thermal Analysis of Liquid Metal Fast Breeder Reactors", ANS Monograph, American Nuclear Society, La Grange Park, Illinois, p. 93 (1978).

13. H.G. Johnson, Personal Communication, January 1981.

APPENDIX C

FTR Natural Circulation Tests - Decay Power Distributions

C. A. McGinnis

APPENDIX C

FTR NATURAL CIRCULATION TESTS

DECAY POWER DISTRIBUTIONS

This appendix summarizes the analysis performed to generate FTR Natural Circulation Test decay power predictions.

All results are based on a numerical integration (S4M computer program) using ENDF/B-IV fission product decay energy release rate data (Ref. C-1) and the power/time history of the FTR operation prior to the natural circulation tests. Input reaction rate data was generated from BOEC fission rates provided in Reference C-2. It was assumed that reaction rates vary linearly with reactor power for the analysis. Using the fission rate data from Reference C-2 as values for 100% power conditions, the reaction rate versus time model followed the power/time histories provided for the 35, 75 and 100% power natural circulation tests (see Figures 18, 19 and 20 of the report).

Decay heat values were calculated at 144 times after shutdown for times up to 1000 seconds after shutdown. Nominal values and values including uncertainties were calculated. The uncertainties are only those associated with the decay energy release rate kernels used in S4M and are time-dependent. No uncertainty associated with the model, fission rate data, or individual assembly operating power is included. U^{238} capture rate data was generated using Reference C-2 U^{238} fission rate information and the ratio of the U^{238} capture rate to fission rate for an average fuel assembly as predicted in CRBR studies.

Six sets of results were calculated for the FTR Natural Circulation Tests. For each of the three power levels, both individual assembly data for the inner and outer driver assemblies and total decay heat (sum of all inner and outer driver assemblies) values were calculated. The summation of decay power for all fuel assemblies under natural circulation test conditions are listed in the attachment to this appendix.

REFERENCES:

C-1) F. Schmittroth and R.E. Schenter, "Uncertainties in Fission Product Decay Heat Calculations", Nucl. Sci. Eng., 63, 276, 1977.

C-2) MG-22, "FTR User's Guide", May 1979.

ATTACHMENT TO APPENDIX C

Summation of Decay Power for All Fuel
Assemblies Under N/C; Initiated from
Three Different Power Conditions

TABLE I - All Fuel Assemblies for the Case Initiated from 35% Power

TABLE II - All Fuel Assemblies for the Case Initiated from 75% Power

TABLE III - All Fuel Assemblies for the Case Initiated from 100% Power

TABLE I

SUMMARY OF DECAY POWER CALCULATION RESULTS (WMCMOPI) Dated 5/5/81

SUM OF ALL ASSEMBLIES FOR FFTF 35 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINT
S (SEC)			(MEGAWATTS)				(O/O)
0.	6.317E+00	2.191E-01	6.536E+00	8.610E+00	2.191E-01	8.829E+00	35.07
1.0000E+00	5.884E+00	2.191E-01	6.103E+00	7.758E+00	2.191E-01	7.977E+00	30.69
2.0000E+00	5.624E+00	2.190E-01	5.843E+00	7.313E+00	2.190E-01	7.532E+00	28.91
3.0000E+00	5.435E+00	2.189E-01	5.654E+00	7.002E+00	2.189E-01	7.221E+00	27.71
4.0000E+00	5.286E+00	2.188E-01	5.505E+00	6.763E+00	2.188E-01	6.982E+00	26.82
5.0000E+00	5.162E+00	2.188E-01	5.381E+00	6.567E+00	2.188E-01	6.785E+00	26.11
6.0000E+00	5.054E+00	2.187E-01	5.273E+00	6.400E+00	2.187E-01	6.618E+00	25.51
7.0000E+00	4.960E+00	2.186E-01	5.179E+00	6.255E+00	2.186E-01	6.474E+00	25.00
8.0000E+00	4.876E+00	2.185E-01	5.095E+00	6.127E+00	2.185E-01	6.346E+00	24.56
9.0000E+00	4.800E+00	2.185E-01	5.018E+00	6.013E+00	2.185E-01	6.231E+00	24.17
1.0000E+01	4.730E+00	2.184E-01	4.948E+00	5.909E+00	2.184E-01	6.127E+00	23.82
1.1000E+01	4.665E+00	2.183E-01	4.884E+00	5.813E+00	2.183E-01	6.031E+00	23.50
1.2000E+01	4.605E+00	2.182E-01	4.824E+00	5.725E+00	2.182E-01	5.943E+00	23.21
1.3000E+01	4.550E+00	2.182E-01	4.768E+00	5.643E+00	2.182E-01	5.861E+00	22.94
1.4000E+01	4.497E+00	2.181E-01	4.715E+00	5.567E+00	2.181E-01	5.785E+00	22.68
1.5000E+01	4.448E+00	2.180E-01	4.666E+00	5.495E+00	2.180E-01	5.713E+00	22.44
1.6000E+01	4.401E+00	2.179E-01	4.619E+00	5.427E+00	2.179E-01	5.645E+00	22.22

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 35 PERCENT CASE

TIME	********NOMINAL VALUE***********			****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINTY
(SEC)			(MEGAWATTS)				(O/O)
1.7000E+01	4.357E+00	2.179E-01	4.575E+00	5.364E+00	2.179E-01	5.582E+00	22.00
1.8000E+01	4.315E+00	2.178E-01	4.533E+00	5.304E+00	2.178E-01	5.521E+00	21.80
1.9000E+01	4.275E+00	2.177E-01	4.493E+00	5.246E+00	2.177E-01	5.464E+00	21.61
2.0000E+01	4.237E+00	2.177E-01	4.455E+00	5.192E+00	2.177E-01	5.410E+00	21.43
2.2000E+01	4.166E+00	2.175E-01	4.384E+00	5.090E+00	2.175E-01	5.308E+00	21.08
2.4000E+01	4.100E+00	2.174E-01	4.318E+00	4.997E+00	2.174E-01	5.215E+00	20.77
2.6000E+01	4.040E+00	2.172E-01	4.257E+00	4.911E+00	2.172E-01	5.129E+00	20.48
2.8000E+01	3.983E+00	2.171E-01	4.200E+00	4.832E+00	2.171E-01	5.049E+00	20.21
3.0000E+01	3.930E+00	2.169E-01	4.147E+00	4.758E+00	2.169E-01	4.975E+00	19.96
3.2000E+01	3.880E+00	2.168E-01	4.097E+00	4.688E+00	2.168E-01	4.905E+00	19.72
3.4000E+01	3.833E+00	2.166E-01	4.050E+00	4.623E+00	2.166E-01	4.840E+00	19.50
3.6000E+01	3.789E+00	2.165E-01	4.005E+00	4.562E+00	2.165E-01	4.778E+00	19.29
3.8000E+01	3.747E+00	2.163E-01	3.963E+00	4.503E+00	2.163E-01	4.720E+00	19.09
4.0000E+01	3.707E+00	2.162E-01	3.923E+00	4.448E+00	2.162E-01	4.664E+00	18.91
4.2000E+01	3.668E+00	2.160E-01	3.884E+00	4.396E+00	2.160E-01	4.612E+00	18.73
4.4000E+01	3.632E+00	2.159E-01	3.848E+00	4.346E+00	2.159E-01	4.562E+00	18.56
4.6000E+01	3.597E+00	2.157E-01	3.813E+00	4.298E+00	2.157E-01	4.514E+00	18.39

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 35 PERCENT CASE

TIME	********NOMINAL VALUE**********			****VALUE w/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINT
(SEC.)			(MEGAWATTS)				(O/O)
4.8000E+01	3.563E+00	2.156E-01	3.779E+00	4.253E+00	2.156E-01	4.468E+00	18.24
5.0000E+01	3.531E+00	2.154E-01	3.747E+00	4.209E+00	2.154E-01	4.424E+00	18.09
5.2000E+01	3.500E+00	2.153E-01	3.716E+00	4.167E+00	2.153E-01	4.382E+00	17.94
5.4000E+01	3.471E+00	2.151E-01	3.686E+00	4.127E+00	2.151E-01	4.342E+00	17.80
5.6000E+01	3.442E+00	2.150E-01	3.657E+00	4.088E+00	2.150E-01	4.303E+00	17.67
5.8000E+01	3.414E+00	2.149E-01	3.629E+00	4.051E+00	2.149E-01	4.266E+00	17.54
6.0000E+01	3.387E+00	2.147E-01	3.602E+00	4.015E+00	2.147E-01	4.230E+00	17.42
6.2000E+01	3.362E+00	2.146E-01	3.576E+00	3.980E+00	2.146E-01	4.195E+00	17.30
6.4000E+01	3.337E+00	2.144E-01	3.551E+00	3.947E+00	2.144E-01	4.161E+00	17.18
6.6000E+01	3.312E+00	2.143E-01	3.527E+00	3.914E+00	2.143E-01	4.129E+00	17.07
6.8000E+01	3.289E+00	2.141E-01	3.503E+00	3.883E+00	2.141E-01	4.097E+00	16.96
7.0000E+01	3.266E+00	2.140E-01	3.480E+00	3.853E+00	2.140E-01	4.067E+00	16.85
7.2000E+01	3.244E+00	2.138E-01	3.458E+00	3.823E+00	2.138E-01	4.037E+00	16.75
7.4000E+01	3.223E+00	2.137E-01	3.436E+00	3.795E+00	2.137E-01	4.009E+00	16.65
7.6000E+01	3.202E+00	2.136E-01	3.415E+00	3.767E+00	2.136E-01	3.981E+00	16.56
7.8000E+01	3.181E+00	2.134E-01	3.395E+00	3.740E+00	2.134E-01	3.954E+00	16.47
8.0000E+01	3.162E+00	2.133E-01	3.375E+00	3.714E+00	2.133E-01	3.928E+00	16.38

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 35 PERCENT CASE

TIME	********NOMINAL VALUE********		***********	****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINTY
S (SEC)			(MEGAWATTS)				(O/O)
8.2000E+01	3.143E+00	2.131E-01	3.356E+00	3.689E+00	2.131E-01	3.902E+00	16.29
8.4000E+01	3.124E+00	2.130E-01	3.337E+00	3.665E+00	2.130E-01	3.878E+00	16.20
8.6000E+01	3.106E+00	2.128E-01	3.318E+00	3.641E+00	2.128E-01	3.853E+00	16.12
8.8000E+01	3.088E+00	2.127E-01	3.301E+00	3.617E+00	2.127E-01	3.830E+00	16.04
9.0000E+01	3.071E+00	2.125E-01	3.283E+00	3.595E+00	2.125E-01	3.807E+00	15.96
9.2000E+01	3.054E+00	2.124E-01	3.266E+00	3.573E+00	2.124E-01	3.785E+00	15.88
9.4000E+01	3.037E+00	2.123E-01	3.250E+00	3.551E+00	2.123E-01	3.763E+00	15.81
9.6000E+01	3.021E+00	2.121E-01	3.233E+00	3.530E+00	2.121E-01	3.742E+00	15.74
9.8000E+01	3.006E+00	2.120E-01	3.218E+00	3.510E+00	2.120E-01	3.722E+00	15.67
1.0000E+02	2.990E+00	2.118E-01	3.202E+00	3.490E+00	2.118E-01	3.702E+00	15.60
1.0200E+02	2.975E+00	2.117E-01	3.187E+00	3.470E+00	2.117E-01	3.682E+00	15.53
1.0400E+02	2.961E+00	2.115E-01	3.172E+00	3.451E+00	2.115E-01	3.663E+00	15.46
1.0600E+02	2.946E+00	2.114E-01	3.158E+00	3.433E+00	2.114E-01	3.644E+00	15.40
1.0800E+02	2.932E+00	2.113E-01	3.144E+00	3.415E+00	2.113E-01	3.626E+00	15.34
1.1000E+02	2.919E+00	2.111E-01	3.130E+00	3.397E+00	2.111E-01	3.608E+00	15.28
1.1200E+02	2.905E+00	2.110E-01	3.116E+00	3.380E+00	2.110E-01	3.591E+00	15.22
1.1400E+02	2.892E+00	2.108E-01	3.103E+00	3.363E+00	2.108E-01	3.574E+00	15.17

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 35 PERCENT CASE

TIME (SEC)	*******NOMINAL VALUE************* F.P.	U239/NP239	TOTAL	****VALUE w/ UNCERTAINTY********* F.P.	U239/NP239	TOTAL	TOTAL UNCERTAINTY (O/O)
			(MEGAWATTS)				
1.1600E+02	2.879E+00	2.107E-01	3.090E+00	3.346E+00	2.107E-01	3.557E+00	15.11
1.1800E+02	2.867E+00	2.105E-01	3.077E+00	3.330E+00	2.105E-01	3.541E+00	15.06
1.2000E+02	2.854E+00	2.104E-01	3.065E+00	3.314E+00	2.104E-01	3.525E+00	15.01
1.2200E+02	2.842E+00	2.103E-01	3.052E+00	3.299E+00	2.103E-01	3.509E+00	14.96
1.2400E+02	2.830E+00	2.101E-01	3.040E+00	3.284E+00	2.101E-01	3.494E+00	14.92
1.2600E+02	2.819E+00	2.100E-01	3.029E+00	3.269E+00	2.100E-01	3.479E+00	14.87
1.2800E+02	2.807E+00	2.098E-01	3.017E+00	3.255E+00	2.098E-01	3.464E+00	14.83
1.3000E+02	2.796E+00	2.097E-01	3.006E+00	3.240E+00	2.097E-01	3.450E+00	14.78
1.3200E+02	2.785E+00	2.096E-01	2.995E+00	3.227E+00	2.096E-01	3.436E+00	14.74
1.3400E+02	2.774E+00	2.094E-01	2.984E+00	3.213E+00	2.094E-01	3.422E+00	14.70
1.3600E+02	2.764E+00	2.093E-01	2.973E+00	3.200E+00	2.093E-01	3.409E+00	14.66
1.3800E+02	2.753E+00	2.091E-01	2.963E+00	3.187E+00	2.091E-01	3.396E+00	14.62
1.4000E+02	2.743E+00	2.090E-01	2.952E+00	3.174E+00	2.090E-01	3.383E+00	14.58
1.4200E+02	2.733E+00	2.089E-01	2.942E+00	3.161E+00	2.089E-01	3.370E+00	14.55
1.4400E+02	2.723E+00	2.087E-01	2.932E+00	3.149E+00	2.087E-01	3.358E+00	14.51
1.4600E+02	2.714E+00	2.086E-01	2.922E+00	3.137E+00	2.086E-01	3.345E+00	14.48
1.4800E+02	2.704E+00	2.084E-01	2.913E+00	3.125E+00	2.084E-01	3.333E+00	14.44

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 35 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL UNCERTAINTY
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	
(SEC)	(MEGAWATTS)						(O/O)
1.5000E+02	2.695E+00	2.083E-01	2.903E+00	3.113E+00	2.083E-01	3.321E+00	14.41
1.5200E+02	2.686E+00	2.082E-01	2.894E+00	3.102E+00	2.082E-01	3.310E+00	14.38
1.5400E+02	2.676E+00	2.080E-01	2.885E+00	3.090E+00	2.080E-01	3.298E+00	14.35
1.5600E+02	2.668E+00	2.079E-01	2.875E+00	3.079E+00	2.079E-01	3.287E+00	14.32
1.5800E+02	2.659E+00	2.077E-01	2.867E+00	3.069E+00	2.077E-01	3.276E+00	14.29
1.6000E+02	2.650E+00	2.076E-01	2.858E+00	3.058E+00	2.076E-01	3.265E+00	14.26
1.6200E+02	2.642E+00	2.075E-01	2.849E+00	3.047E+00	2.075E-01	3.255E+00	14.24
1.6400E+02	2.633E+00	2.073E-01	2.841E+00	3.037E+00	2.073E-01	3.244E+00	14.21
1.6600E+02	2.625E+00	2.072E-01	2.832E+00	3.027E+00	2.072E-01	3.234E+00	14.19
1.6800E+02	2.617E+00	2.071E-01	2.824E+00	3.017E+00	2.071E-01	3.224E+00	14.16
1.7000E+02	2.609E+00	2.069E-01	2.816E+00	3.007E+00	2.069E-01	3.214E+00	14.14
1.7200E+02	2.601E+00	2.068E-01	2.808E+00	2.997E+00	2.068E-01	3.204E+00	14.11
1.7400E+02	2.593E+00	2.066E-01	2.800E+00	2.988E+00	2.066E-01	3.195E+00	14.09
1.7600E+02	2.586E+00	2.065E-01	2.792E+00	2.979E+00	2.065E-01	3.185E+00	14.07
1.7800E+02	2.578E+00	2.064E-01	2.785E+00	2.969E+00	2.064E-01	3.176E+00	14.05
1.8000E+02	2.571E+00	2.062E-01	2.777E+00	2.960E+00	2.062E-01	3.166E+00	14.03
1.8200E+02	2.563E+00	2.061E-01	2.770E+00	2.951E+00	2.061E-01	3.157E+00	14.00

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 35 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINTY
(SEC)			(MEGAWATTS)				(O/O)
1.8400E+02	2.556E+00	2.060E-01	2.762E+00	2.942E+00	2.060E-01	3.148E+00	13.99
1.8600E+02	2.549E+00	2.058E-01	2.755E+00	2.934E+00	2.058E-01	3.140E+00	13.97
1.8800E+02	2.542E+00	2.057E-01	2.748E+00	2.925E+00	2.057E-01	3.131E+00	13.95
1.9000E+02	2.535E+00	2.055E-01	2.741E+00	2.917E+00	2.055E-01	3.122E+00	13.93
1.9200E+02	2.528E+00	2.054E-01	2.734E+00	2.909E+00	2.054E-01	3.114E+00	13.91
1.9400E+02	2.522E+00	2.053E-01	2.727E+00	2.900E+00	2.053E-01	3.106E+00	13.89
1.9600E+02	2.515E+00	2.051E-01	2.720E+00	2.892E+00	2.051E-01	3.097E+00	13.88
1.9800E+02	2.508E+00	2.050E-01	2.713E+00	2.884E+00	2.050E-01	3.089E+00	13.86
2.0000E+02	2.502E+00	2.049E-01	2.707E+00	2.877E+00	2.049E-01	3.081E+00	13.85
2.0200E+02	2.495E+00	2.047E-01	2.700E+00	2.869E+00	2.047E-01	3.074E+00	13.83
2.0400E+02	2.489E+00	2.046E-01	2.694E+00	2.861E+00	2.046E-01	3.066E+00	13.82
2.0600E+02	2.483E+00	2.045E-01	2.687E+00	2.854E+00	2.045E-01	3.058E+00	13.80
2.0800E+02	2.477E+00	2.043E-01	2.681E+00	2.846E+00	2.043E-01	3.051E+00	13.79
2.1000E+02	2.471E+00	2.042E-01	2.675E+00	2.839E+00	2.042E-01	3.043E+00	13.78
2.1200E+02	2.465E+00	2.040E-01	2.669E+00	2.832E+00	2.040E-01	3.036E+00	13.76
2.1400E+02	2.459E+00	2.039E-01	2.662E+00	2.825E+00	2.039E-01	3.029E+00	13.75
2.1600E+02	2.453E+00	2.038E-01	2.656E+00	2.818E+00	2.038E-01	3.021E+00	13.74

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 35 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINTY
(SEC.)			(MEGAWATTS)				(O/O)
2.1800E+02	2.447E+00	2.036E-01	2.651E+00	2.811E+00	2.036E-01	3.014E+00	13.73
2.2000E+02	2.441E+00	2.035E-01	2.645E+00	2.804E+00	2.035E-01	3.007E+00	13.71
2.2200E+02	2.435E+00	2.034E-01	2.639E+00	2.797E+00	2.034E-01	3.000E+00	13.70
2.2400E+02	2.430E+00	2.032E-01	2.633E+00	2.790E+00	2.032E-01	2.994E+00	13.69
2.2600E+02	2.424E+00	2.031E-01	2.627E+00	2.784E+00	2.031E-01	2.987E+00	13.68
2.2800E+02	2.419E+00	2.030E-01	2.622E+00	2.777E+00	2.030E-01	2.980E+00	13.67
2.3000E+02	2.413E+00	2.028E-01	2.616E+00	2.771E+00	2.028E-01	2.974E+00	13.66
2.3200E+02	2.408E+00	2.027E-01	2.611E+00	2.764E+00	2.027E-01	2.967E+00	13.65
2.3400E+02	2.403E+00	2.026E-01	2.605E+00	2.758E+00	2.026E-01	2.961E+00	13.64
2.3600E+02	2.397E+00	2.024E-01	2.600E+00	2.752E+00	2.024E-01	2.954E+00	13.63
2.3800E+02	2.392E+00	2.023E-01	2.595E+00	2.746E+00	2.023E-01	2.948E+00	13.63
2.4000E+02	2.387E+00	2.022E-01	2.589E+00	2.740E+00	2.022E-01	2.942E+00	13.62
2.4200E+02	2.382E+00	2.020E-01	2.584E+00	2.734E+00	2.020E-01	2.936E+00	13.61
2.4400E+02	2.377E+00	2.019E-01	2.579E+00	2.728E+00	2.019E-01	2.930E+00	13.60
2.4600E+02	2.372E+00	2.018E-01	2.574E+00	2.722E+00	2.018E-01	2.924E+00	13.59
2.4800E+02	2.367E+00	2.016E-01	2.569E+00	2.716E+00	2.016E-01	2.918E+00	13.59
2.5000E+02	2.362E+00	2.015E-01	2.564E+00	2.710E+00	2.015E-01	2.912E+00	13.58

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 35 PERCENT CASE

TIME (SEC)	NOMINAL VALUE F.P.	NOMINAL VALUE U239/NP239	NOMINAL VALUE TOTAL	VALUE W/ UNCERTAINTY F.P.	VALUE W/ UNCERTAINTY U239/NP239	VALUE W/ UNCERTAINTY TOTAL	TOTAL UNCERTAINTY (O/O)
	(MEGAWATTS)						
3.0000E+02	2.254E+00	1.982E-01	2.452E+00	2.584E+00	1.982E-01	2.783E+00	13.48
4.0000E+02	2.091E+00	1.919E-01	2.283E+00	2.397E+00	1.919E-01	2.589E+00	13.41
5.0000E+02	1.968E+00	1.859E-01	2.154E+00	2.256E+00	1.859E-01	2.442E+00	13.37
6.0000E+02	1.869E+00	1.802E-01	2.049E+00	2.143E+00	1.802E-01	2.323E+00	13.34
7.0000E+02	1.785E+00	1.747E-01	1.960E+00	2.046E+00	1.747E-01	2.221E+00	13.33
8.0000E+02	1.712E+00	1.695E-01	1.881E+00	1.962E+00	1.695E-01	2.131E+00	13.31
9.0000E+02	1.646E+00	1.646E-01	1.811E+00	1.887E+00	1.646E-01	2.052E+00	13.30
1.0000E+03	1.588E+00	1.599E-01	1.748E+00	1.820E+00	1.599E-01	1.980E+00	13.30

END OF PROBLEM

TABLE II

SUMMARY OF DECAY POWER CALCULATION RESULTS (WMCM08X) Dated 5/11/81

SUM OF ALL ASSEMBLIES FOR FFTF 75 PERCENT CASE

TIME	*******NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL UNCERTAINTY
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	
(SEC.)	(MEGAWATTS)						(0/0)
0.	1.295E+01	3.857E-01	1.333E+01	1.774E+01	3.857E-01	1.812E+01	35.93
1.0000E+00	1.203E+01	3.855E-01	1.242E+01	1.594E+01	3.855E-01	1.632E+01	31.45
2.0000E+00	1.148E+01	3.854E-01	1.187E+01	1.500E+01	3.854E-01	1.538E+01	29.63
3.0000E+00	1.108E+01	3.852E-01	1.147E+01	1.434E+01	3.852E-01	1.473E+01	28.41
4.0000E+00	1.077E+01	3.851E-01	1.116E+01	1.384E+01	3.851E-01	1.422E+01	27.50
5.0000E+00	1.051E+01	3.849E-01	1.089E+01	1.342E+01	3.849E-01	1.381E+01	26.77
6.0000E+00	1.028E+01	3.848E-01	1.067E+01	1.307E+01	3.848E-01	1.346E+01	26.16
7.0000E+00	1.008E+01	3.846E-01	1.047E+01	1.277E+01	3.846E-01	1.315E+01	25.64
8.0000E+00	9.904E+00	3.845E-01	1.029E+01	1.250E+01	3.845E-01	1.288E+01	25.19
9.0000E+00	9.743E+00	3.843E-01	1.013E+01	1.225E+01	3.843E-01	1.264E+01	24.80
1.0000E+01	9.595E+00	3.842E-01	9.980E+00	1.203E+01	3.842E-01	1.242E+01	24.44
1.1000E+01	9.459E+00	3.840E-01	9.843E+00	1.183E+01	3.840E-01	1.222E+01	24.11
1.2000E+01	9.333E+00	3.838E-01	9.717E+00	1.165E+01	3.838E-01	1.203E+01	23.81
1.3000E+01	9.215E+00	3.837E-01	9.599E+00	1.147E+01	3.837E-01	1.186E+01	23.53
1.4000E+01	9.105E+00	3.835E-01	9.488E+00	1.131E+01	3.835E-01	1.170E+01	23.27
1.5000E+01	9.001E+00	3.834E-01	9.384E+00	1.116E+01	3.834E-01	1.154E+01	23.03
1.6000E+01	8.902E+00	3.832E-01	9.286E+00	1.102E+01	3.832E-01	1.140E+01	22.80

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 75 PERCENT CASE

TIME (SEC.)	********NOMINAL VALUE************ F.P.	U239/NP239	TOTAL	****VALUE W/ UNCERTAINTY********* F.P.	U239/NP239	TOTAL	TOTAL UNCERTAINTY (0/0)
			(MEGAWATTS)				
1.7000E+01	8.809E+00	3.831E-01	9.193E+00	1.089E+01	3.831E-01	1.127E+01	22.58
1.8000E+01	8.721E+00	3.829E-01	9.104E+00	1.076E+01	3.829E-01	1.114E+01	22.37
1.9000E+01	8.637E+00	3.828E-01	9.020E+00	1.064E+01	3.828E-01	1.102E+01	22.17
2.0000E+01	8.557E+00	3.826E-01	8.939E+00	1.052E+01	3.826E-01	1.091E+01	21.99
2.2000E+01	8.407E+00	3.823E-01	8.789E+00	1.031E+01	3.823E-01	1.069E+01	21.64
2.4000E+01	8.268E+00	3.820E-01	8.650E+00	1.011E+01	3.820E-01	1.049E+01	21.32
2.6000E+01	8.140E+00	3.817E-01	8.522E+00	9.931E+00	3.817E-01	1.031E+01	21.02
2.8000E+01	8.021E+00	3.814E-01	8.402E+00	9.763E+00	3.814E-01	1.014E+01	20.74
3.0000E+01	7.909E+00	3.811E-01	8.290E+00	9.607E+00	3.811E-01	9.988E+00	20.48
3.2000E+01	7.804E+00	3.808E-01	8.185E+00	9.461E+00	3.808E-01	9.842E+00	20.24
3.4000E+01	7.705E+00	3.805E-01	8.086E+00	9.323E+00	3.805E-01	9.704E+00	20.01
3.6000E+01	7.612E+00	3.802E-01	7.992E+00	9.194E+00	3.802E-01	9.574E+00	19.80
3.8000E+01	7.523E+00	3.799E-01	7.903E+00	9.071E+00	3.799E-01	9.451E+00	19.59
4.0000E+01	7.438E+00	3.796E-01	7.818E+00	8.955E+00	3.796E-01	9.335E+00	19.40
4.2000E+01	7.358E+00	3.793E-01	7.737E+00	8.845E+00	3.793E-01	9.224E+00	19.21
4.4000E+01	7.281E+00	3.790E-01	7.660E+00	8.739E+00	3.790E-01	9.118E+00	19.04
4.6000E+01	7.207E+00	3.787E-01	7.586E+00	8.639E+00	3.787E-01	9.018E+00	18.87

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SUMMARY OF DELAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 75 PERCENT CASE

TIME	*******NOMINAL VALUE***********			****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINTY
(SEC.)			(MEGAWATTS)				(0/0)
4.8000E+01	7.137E+00	3.784E-01	7.515E+00	8.543E+00	3.784E-01	8.921E+00	18.71
5.0000E+01	7.069E+00	3.781E-01	7.447E+00	8.451E+00	3.781E-01	8.829E+00	18.55
5.2000E+01	7.004E+00	3.778E-01	7.382E+00	8.363E+00	3.778E-01	8.740E+00	18.40
5.4000E+01	6.941E+00	3.775E-01	7.319E+00	8.278E+00	3.775E-01	8.655E+00	18.26
5.6000E+01	6.881E+00	3.772E-01	7.258E+00	8.196E+00	3.772E-01	8.574E+00	18.12
5.8000E+01	6.823E+00	3.769E-01	7.200E+00	8.118E+00	3.769E-01	8.495E+00	17.99
6.0000E+01	6.766E+00	3.766E-01	7.143E+00	8.042E+00	3.766E-01	8.419E+00	17.86
6.2000E+01	6.712E+00	3.763E-01	7.088E+00	7.969E+00	3.763E-01	8.346E+00	17.74
6.4000E+01	6.659E+00	3.760E-01	7.035E+00	7.899E+00	3.760E-01	8.275E+00	17.62
6.6000E+01	6.609E+00	3.757E-01	6.984E+00	7.831E+00	3.757E-01	8.206E+00	17.50
6.8000E+01	6.559E+00	3.754E-01	6.935E+00	7.765E+00	3.754E-01	8.140E+00	17.39
7.0000E+01	6.511E+00	3.751E-01	6.886E+00	7.701E+00	3.751E-01	8.076E+00	17.28
7.2000E+01	6.465E+00	3.748E-01	6.840E+00	7.639E+00	3.748E-01	8.014E+00	17.17
7.4000E+01	6.420E+00	3.745E-01	6.794E+00	7.579E+00	3.745E-01	7.954E+00	17.07
7.6000E+01	6.376E+00	3.742E-01	6.750E+00	7.521E+00	3.742E-01	7.895E+00	16.97
7.8000E+01	6.333E+00	3.739E-01	6.707E+00	7.465E+00	3.739E-01	7.839E+00	16.87
8.0000E+01	6.292E+00	3.736E-01	6.665E+00	7.410E+00	3.736E-01	7.784E+00	16.78

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 75 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL UNCERTAINTY
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	
S (SEC.)			(MEGAWATTS)				(O/O)
8.2000E+01	6.251E+00	3.733E-01	6.625E+00	7.357E+00	3.733E-01	7.730E+00	16.69
8.4000E+01	6.212E+00	3.730E-01	6.585E+00	7.305E+00	3.730E-01	7.678E+00	16.60
8.6000E+01	6.174E+00	3.727E-01	6.547E+00	7.255E+00	3.727E-01	7.628E+00	16.51
8.8000E+01	6.137E+00	3.724E-01	6.509E+00	7.206E+00	3.724E-01	7.578E+00	16.43
9.0000E+01	6.100E+00	3.721E-01	6.472E+00	7.158E+00	3.721E-01	7.530E+00	16.35
9.2000E+01	6.065E+00	3.718E-01	6.437E+00	7.112E+00	3.718E-01	7.484E+00	16.27
9.4000E+01	6.030E+00	3.715E-01	6.402E+00	7.067E+00	3.715E-01	7.438E+00	16.19
9.6000E+01	5.997E+00	3.712E-01	6.368E+00	7.023E+00	3.712E-01	7.394E+00	16.11
9.8000E+01	5.964E+00	3.710E-01	6.335E+00	6.980E+00	3.710E-01	7.351E+00	16.04
1.0000E+02	5.931E+00	3.707E-01	6.302E+00	6.938E+00	3.707E-01	7.308E+00	15.97
1.0200E+02	5.900E+00	3.704E-01	6.270E+00	6.897E+00	3.704E-01	7.267E+00	15.90
1.0400E+02	5.869E+00	3.701E-01	6.239E+00	6.857E+00	3.701E-01	7.227E+00	15.83
1.0600E+02	5.839E+00	3.698E-01	6.209E+00	6.818E+00	3.698E-01	7.188E+00	15.76
1.0800E+02	5.810E+00	3.695E-01	6.179E+00	6.780E+00	3.695E-01	7.149E+00	15.70
1.1000E+02	5.781E+00	3.692E-01	6.150E+00	6.743E+00	3.692E-01	7.112E+00	15.64
1.1200E+02	5.753E+00	3.689E-01	6.122E+00	6.706E+00	3.689E-01	7.075E+00	15.58
1.1400E+02	5.725E+00	3.686E-01	6.094E+00	6.671E+00	3.686E-01	7.040E+00	15.52

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 75 PERCENT CASE

TIME (SEC.)	********NOMINAL VALUE************ F.P.	U239/NP239	TOTAL	****VALUE W/ UNCERTAINTY********* F.P.	U239/NP239	TOTAL	TOTAL UNCERTAINTY
	(MEGAWATTS)						(0/0)
1.1600E+02	5.698E+00	3.683E-01	6.067E+00	6.636E+00	3.683E-01	7.005E+00	15.46
1.1800E+02	5.672E+00	3.680E-01	6.040E+00	6.602E+00	3.680E-01	6.970E+00	15.41
1.2000E+02	5.646E+00	3.677E-01	6.014E+00	6.569E+00	3.677E-01	6.937E+00	15.35
1.2200E+02	5.621E+00	3.674E-01	5.988E+00	6.537E+00	3.674E-01	6.904E+00	15.30
1.2400E+02	5.596E+00	3.672E-01	5.963E+00	6.505E+00	3.672E-01	6.872E+00	15.25
1.2600E+02	5.571E+00	3.669E-01	5.938E+00	6.474E+00	3.669E-01	6.841E+00	15.21
1.2800E+02	5.547E+00	3.666E-01	5.914E+00	6.444E+00	3.666E-01	6.811E+00	15.16
1.3000E+02	5.524E+00	3.663E-01	5.890E+00	6.414E+00	3.663E-01	6.780E+00	15.11
1.3200E+02	5.501E+00	3.660E-01	5.867E+00	6.385E+00	3.660E-01	6.751E+00	15.07
1.3400E+02	5.478E+00	3.657E-01	5.844E+00	6.357E+00	3.657E-01	6.722E+00	15.03
1.3600E+02	5.456E+00	3.654E-01	5.822E+00	6.329E+00	3.654E-01	6.694E+00	14.99
1.3800E+02	5.434E+00	3.651E-01	5.799E+00	6.301E+00	3.651E-01	6.666E+00	14.95
1.4000E+02	5.413E+00	3.648E-01	5.778E+00	6.274E+00	3.648E-01	6.639E+00	14.91
1.4200E+02	5.392E+00	3.646E-01	5.756E+00	6.248E+00	3.646E-01	6.612E+00	14.87
1.4400E+02	5.371E+00	3.643E-01	5.736E+00	6.222E+00	3.643E-01	6.586E+00	14.83
1.4600E+02	5.351E+00	3.640E-01	5.715E+00	6.197E+00	3.640E-01	6.561E+00	14.80
1.4800E+02	5.331E+00	3.637E-01	5.695E+00	6.172E+00	3.637E-01	6.535E+00	14.76

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 75 PERCENT CASE

TIME (SEC)	NOMINAL VALUE F.P.	NOMINAL VALUE U239/NP239	NOMINAL VALUE TOTAL	VALUE W/ UNCERTAINTY F.P.	VALUE W/ UNCERTAINTY U239/NP239	VALUE W/ UNCERTAINTY TOTAL	TOTAL UNCERTAINTY
	(MEGAWATTS)						(0/0)
1.5000E+02	5.311E+00	3.634E-01	5.675E+00	6.147E+00	3.634E-01	6.511E+00	14.73
1.5200E+02	5.292E+00	3.631E-01	5.655E+00	6.123E+00	3.631E-01	6.486E+00	14.70
1.5400E+02	5.273E+00	3.628E-01	5.636E+00	6.099E+00	3.628E-01	6.462E+00	14.66
1.5600E+02	5.254E+00	3.626E-01	5.617E+00	6.076E+00	3.626E-01	6.439E+00	14.63
1.5800E+02	5.236E+00	3.623E-01	5.598E+00	6.054E+00	3.623E-01	6.416E+00	14.60
1.6000E+02	5.218E+00	3.620E-01	5.580E+00	6.031E+00	3.620E-01	6.393E+00	14.57
1.6200E+02	5.200E+00	3.617E-01	5.562E+00	6.009E+00	3.617E-01	6.371E+00	14.55
1.6400E+02	5.183E+00	3.614E-01	5.544E+00	5.988E+00	3.614E-01	6.349E+00	14.52
1.6600E+02	5.165E+00	3.611E-01	5.527E+00	5.966E+00	3.611E-01	6.327E+00	14.49
1.6800E+02	5.148E+00	3.608E-01	5.509E+00	5.945E+00	3.608E-01	6.306E+00	14.47
1.7000E+02	5.132E+00	3.606E-01	5.492E+00	5.925E+00	3.606E-01	6.285E+00	14.44
1.7200E+02	5.115E+00	3.603E-01	5.475E+00	5.904E+00	3.603E-01	6.265E+00	14.42
1.7400E+02	5.099E+00	3.600E-01	5.459E+00	5.885E+00	3.600E-01	6.245E+00	14.39
1.7600E+02	5.083E+00	3.597E-01	5.443E+00	5.865E+00	3.597E-01	6.225E+00	14.37
1.7800E+02	5.067E+00	3.594E-01	5.427E+00	5.846E+00	3.594E-01	6.205E+00	14.35
1.8000E+02	5.052E+00	3.592E-01	5.411E+00	5.827E+00	3.592E-01	6.186E+00	14.32
1.8200E+02	5.036E+00	3.589E-01	5.395E+00	5.808E+00	3.589E-01	6.167E+00	14.30

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 75 PERCENT CASE

TIME (SEC.)	********NOMINAL VALUE************ F.P.	U235/NP239	TOTAL	****VALUE W/ UNCERTAINTY********* F.P.	U235/NP239	TOTAL	TOTAL UNCERTAINTY (0/0)
			(MEGAWATTS)				
1.8400E+02	5.021E+00	3.586E-01	5.380E+00	5.790E+00	3.586E-01	6.148E+00	14.28
1.8600E+02	5.006E+00	3.583E-01	5.365E+00	5.771E+00	3.583E-01	6.13[illegible]E+00	14.26
1.8800E+02	4.992E+00	3.580E-01	5.350E+00	5.754E+00	3.580E-01	6.112E+00	14.24
1.9000E+02	4.977E+00	3.577E-01	5.335E+00	5.736E+00	3.577E-01	6.094E+00	14.22
1.9200E+02	4.963E+00	3.575E-01	5.320E+00	5.719E+00	3.575E-01	6.076E+00	14.21
1.9400E+02	4.949E+00	3.572E-01	5.306E+00	5.702E+00	3.572E-01	6.059E+00	14.19
1.9600E+02	4.935E+00	3.569E-01	5.292E+00	5.685E+00	3.569E-01	6.042E+00	14.17
1.9800E+02	4.921E+00	3.566E-01	5.278E+00	5.668E+00	3.566E-01	6.025E+00	14.16
2.0000E+02	4.907E+00	3.563E-01	5.264E+00	5.652E+00	3.563E-01	6.008E+00	14.14
2.0200E+02	4.894E+00	3.561E-01	5.250E+00	5.636E+00	3.561E-01	5.992E+00	14.12
2.0400E+02	4.881E+00	3.558E-01	5.237E+00	5.620E+00	3.558E-01	5.975E+00	14.11
2.0600E+02	4.868E+00	3.555E-01	5.223E+00	5.604E+00	3.555E-01	5.959E+00	14.09
2.0800E+02	4.855E+00	3.552E-01	5.210E+00	5.588E+00	3.552E-01	5.944E+00	14.08
2.1000E+02	4.842E+00	3.550E-01	5.197E+00	5.573E+00	3.550E-01	5.928E+00	14.07
2.1200E+02	4.829E+00	3.547E-01	5.184E+00	5.558E+00	3.547E-01	5.913E+00	14.05
2.1400E+02	4.817E+00	3.544E-01	5.171E+00	5.543E+00	3.544E-01	5.897E+00	14.04
2.1600E+02	4.805E+00	3.541E-01	5.159E+00	5.528E+00	3.541E-01	5.882E+00	14.03

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR FFTF 75 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL UNCERTAINTY
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	
(SEC.)			(MEGAWATTS)				(0/0)
2.1800E+02	4.793E+00	3.538E-01	5.146E+00	5.514E+00	3.538E-01	5.868E+00	14.01
2.2000E+02	4.781E+00	3.536E-01	5.134E+00	5.500E+00	3.536E-01	5.853E+00	14.00
2.2200E+02	4.769E+00	3.533E-01	5.122E+00	5.485E+00	3.533E-01	5.839E+00	13.99
2.2400E+02	4.757E+00	3.530E-01	5.110E+00	5.471E+00	3.530E-01	5.824E+00	13.98
2.2600E+02	4.745E+00	3.527E-01	5.098E+00	5.458E+00	3.527E-01	5.810E+00	13.97
2.2800E+02	4.734E+00	3.525E-01	5.086E+00	5.444E+00	3.525E-01	5.796E+00	13.96
2.3000E+02	4.723E+00	3.522E-01	5.075E+00	5.431E+00	3.522E-01	5.783E+00	13.95
2.3200E+02	4.711E+00	3.519E-01	5.063E+00	5.417E+00	3.519E-01	5.769E+00	13.94
2.3400E+02	4.700E+00	3.516E-01	5.052E+00	5.404E+00	3.516E-01	5.756E+00	13.93
2.3600E+02	4.689E+00	3.514E-01	5.041E+00	5.391E+00	3.514E-01	5.742E+00	13.92
2.3800E+02	4.679E+00	3.511E-01	5.030E+00	5.378E+00	3.511E-01	5.729E+00	13.91
2.4000E+02	4.668E+00	3.508E-01	5.019E+00	5.366E+00	3.508E-01	5.716E+00	13.90
2.4200E+02	4.657E+00	3.505E-01	5.008E+00	5.353E+00	3.505E-01	5.704E+00	13.90
2.4400E+02	4.647E+00	3.503E-01	4.997E+00	5.341E+00	3.503E-01	5.691E+00	13.89
2.4600E+02	4.636E+00	3.500E-01	4.986E+00	5.328E+00	3.500E-01	5.678E+00	13.88
2.4800E+02	4.626E+00	3.497E-01	4.976E+00	5.316E+00	3.497E-01	5.666E+00	13.87
2.5000E+02	4.616E+00	3.494E-01	4.965E+00	5.304E+00	3.494E-01	5.654E+00	13.87

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR THE 75 PERCENT CASE

TIME	******NOMINAL VALUE****** F.P.	U235/NP239	TOTAL	****VALUE W/ UNCERTAINTY********* F.P.	U235/NP239	TOTAL	TOTAL UNCERTAINTY
(SEC.)			(MEGAWATTS)				(%)
3.0000E+02	[illegible]	3.427E-01	4.733E+00	5.042E+00	3.427E-01	5.385E+00	13.77
4.0000E+02	4.052E+00	3.297E-01	4.381E+00	4.653E+00	3.297E-01	4.983E+00	13.72
5.0000E+02	3.798E+00	3.174E-01	4.115E+00	4.362E+00	3.174E-01	4.679E+00	13.70
6.0000E+02	3.593E+00	3.056E-01	3.899E+00	4.127E+00	3.056E-01	4.432E+00	13.69
7.0000E+02	3.420E+00	2.944E-01	3.714E+00	3.928E+00	2.944E-01	4.222E+00	13.68
8.0000E+02	3.269E+00	2.837E-01	3.553E+00	3.755E+00	2.837E-01	4.039E+00	13.69
9.0000E+02	3.135E+00	2.736E-01	3.409E+00	3.602E+00	2.736E-01	3.876E+00	13.69
1.0000E+03	3.015E+00	2.639E-01	3.279E+00	3.465E+00	2.639E-01	3.729E+00	13.70

END OF PROBLEM

TABLE III

SUMMARY OF DECAY POWER CALCULATION RESULTS (WMCM08V) Dated 5/11/81

SUM OF ALL ASSEMBLIES FOR THE FFIF 100 PERCENT CASE

TIME (Sec.)	*******NOMINAL VALUE************ F.P.	U239/NP239	TOTAL	****VALUE W/ UNCERTAINTY********* F.P.	U239/NP239	TOTAL	TOTAL UNCERTAINT (0/0)
			(MEGAWATTS)				
0.	1.831E+01	6.981E-01	1.901E+01	2.485E+01	6.981E-01	2.554E+01	34.39
1.0000E+00	1.708E+01	6.979E-01	1.778E+01	2.244E+01	6.979E-01	2.314E+01	30.10
2.0000E+00	1.635E+01	6.977E-01	1.705E+01	2.118E+01	6.977E-01	2.188E+01	28.35
3.0000E+00	1.582E+01	6.974E-01	1.651E+01	2.030E+01	6.974E-01	2.100E+01	27.18
4.0000E+00	1.540E+01	6.972E-01	1.609E+01	1.963E+01	6.972E-01	2.033E+01	26.30
5.0000E+00	1.504E+01	6.970E-01	1.574E+01	1.907E+01	6.970E-01	1.977E+01	25.60
6.0000E+00	1.474E+01	6.968E-01	1.544E+01	1.860E+01	6.968E-01	1.930E+01	25.01
7.0000E+00	1.447E+01	6.966E-01	1.517E+01	1.819E+01	6.966E-01	1.889E+01	24.51
8.0000E+00	1.424E+01	6.964E-01	1.493E+01	1.783E+01	6.964E-01	1.853E+01	24.08
9.0000E+00	1.402E+01	6.962E-01	1.472E+01	1.751E+01	6.962E-01	1.821E+01	23.70
1.0000E+01	1.382E+01	6.959E-01	1.452E+01	1.721E+01	6.959E-01	1.791E+01	23.36
1.1000E+01	1.384E+01	6.957E-01	1.454E+01	1.695E+01	6.957E-01	1.764E+01	23.04
1.2000E+01	1.347E+01	6.955E-01	1.417E+01	1.670E+01	6.955E-01	1.739E+01	22.76
1.3000E+01	1.331E+01	6.953E-01	1.401E+01	1.646E+01	6.953E-01	1.716E+01	22.49
1.4000E+01	1.317E+01	6.951E-01	1.386E+01	1.625E+01	6.951E-01	1.694E+01	22.24
1.5000E+01	1.303E+01	6.949E-01	1.372E+01	1.605E+01	6.949E-01	1.674E+01	22.00
1.6000E+01	1.290E+01	6.947E-01	1.359E+01	1.586E+01	6.947E-01	1.655E+01	21.78

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR THE FFTF 100 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINTY
(SEC.)			(MEGAWATTS)				(O/O)
1.7000E+01	1.277E+01	6.944E-01	1.347E+01	1.568E+01	6.944E-01	1.637E+01	21.57
1.8000E+01	1.265E+01	6.942E-01	1.335E+01	1.551E+01	6.942E-01	1.620E+01	21.38
1.9000E+01	1.254E+01	6.940E-01	1.323E+01	1.534E+01	6.940E-01	1.604E+01	21.19
2.0000E+01	1.243E+01	6.938E-01	1.313E+01	1.519E+01	6.938E-01	1.588E+01	21.01
2.2000E+01	1.223E+01	6.934E-01	1.293E+01	1.490E+01	6.934E-01	1.560E+01	20.67
2.4000E+01	1.205E+01	6.930E-01	1.274E+01	1.464E+01	6.930E-01	1.533E+01	20.37
2.6000E+01	1.187E+01	6.925E-01	1.257E+01	1.440E+01	6.925E-01	1.509E+01	20.09
2.8000E+01	1.171E+01	6.921E-01	1.241E+01	1.417E+01	6.921E-01	1.487E+01	19.82
3.0000E+01	1.157E+01	6.917E-01	1.226E+01	1.396E+01	6.917E-01	1.466E+01	19.58
3.2000E+01	1.142E+01	6.913E-01	1.212E+01	1.377E+01	6.913E-01	1.446E+01	19.35
3.4000E+01	1.129E+01	6.908E-01	1.198E+01	1.358E+01	6.908E-01	1.428E+01	19.13
3.6000E+01	1.117E+01	6.904E-01	1.186E+01	1.341E+01	6.904E-01	1.410E+01	18.93
3.8000E+01	1.105E+01	6.900E-01	1.174E+01	1.325E+01	6.900E-01	1.394E+01	18.74
4.0000E+01	1.092E+01	6.896E-01	1.162E+01	1.309E+01	6.896E-01	1.378E+01	18.55
4.2000E+01	1.083E+01	6.891E-01	1.152E+01	1.294E+01	6.891E-01	1.363E+01	18.38
4.4000E+01	1.072E+01	6.887E-01	1.141E+01	1.280E+01	6.887E-01	1.349E+01	18.21
4.6000E+01	1.062E+01	6.883E-01	1.131E+01	1.267E+01	6.883E-01	1.336E+01	18.06

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR THE FFTF 100 PERCENT CASE

TIME	********NOMINAL VALUE************ F.P.	U239/NP239	TOTAL	****VALUE W/ UNCERTAINTY********* F.P.	U239/NP239	TOTAL	TOTAL UNCERTAINTY
(SEC.)			(MEGAWATTS)				(O/O)
4.8000E+01	1.053E+01	6.879E-01	1.122E+01	1.254E+01	6.879E-01	1.323E+01	17.90
5.0000E+01	1.044E+01	6.875E-01	1.113E+01	1.242E+01	6.875E-01	1.310E+01	17.76
5.2000E+01	1.035E+01	6.870E-01	1.104E+01	1.230E+01	6.870E-01	1.298E+01	17.62
5.4000E+01	1.027E+01	6.866E-01	1.095E+01	1.218E+01	6.866E-01	1.287E+01	17.49
5.6000E+01	1.019E+01	6.862E-01	1.087E+01	1.207E+01	6.862E-01	1.276E+01	17.36
5.8000E+01	1.011E+01	6.858E-01	1.079E+01	1.197E+01	6.858E-01	1.265E+01	17.23
6.0000E+01	1.003E+01	6.854E-01	1.072E+01	1.187E+01	6.854E-01	1.255E+01	17.11
6.2000E+01	9.960E+00	6.849E-01	1.064E+01	1.177E+01	6.849E-01	1.245E+01	17.00
6.4000E+01	9.889E+00	6.845E-01	1.057E+01	1.167E+01	6.845E-01	1.236E+01	16.88
6.6000E+01	9.821E+00	6.841E-01	1.051E+01	1.158E+01	6.841E-01	1.227E+01	16.78
6.8000E+01	9.755E+00	6.837E-01	1.044E+01	1.149E+01	6.837E-01	1.218E+01	16.67
7.0000E+01	9.690E+00	6.833E-01	1.037E+01	1.141E+01	6.833E-01	1.209E+01	16.57
7.2000E+01	9.628E+00	6.829E-01	1.031E+01	1.133E+01	6.829E-01	1.201E+01	16.47
7.4000E+01	9.567E+00	6.824E-01	1.025E+01	1.125E+01	6.824E-01	1.193E+01	16.38
7.6000E+01	9.508E+00	6.820E-01	1.019E+01	1.117E+01	6.820E-01	1.185E+01	16.28
7.8000E+01	9.451E+00	6.816E-01	1.013E+01	1.109E+01	6.816E-01	1.177E+01	16.19
8.0000E+01	9.396E+00	6.812E-01	1.008E+01	1.102E+01	6.812E-01	1.170E+01	16.11

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR THE FFTF 100 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY**********			TOTAL UNCERTAINTY
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	
(SEC.)			(MEGAWATTS)				(0/0)
8.2000E+01	9.341E+00	6.808E-01	1.002E+01	1.095E+01	6.808E-01	1.163E+01	16.02
8.4000E+01	9.289E+00	6.804E-01	9.969E+00	1.088E+01	6.804E-01	1.156E+01	15.94
8.6000E+01	9.237E+00	6.800E-01	9.917E+00	1.081E+01	6.800E-01	1.149E+01	15.86
8.8000E+01	9.187E+00	6.795E-01	9.867E+00	1.074E+01	6.795E-01	1.142E+01	15.78
9.0000E+01	9.138E+00	6.791E-01	9.817E+00	1.068E+01	6.791E-01	1.136E+01	15.71
9.2000E+01	9.091E+00	6.787E-01	9.769E+00	1.062E+01	6.787E-01	1.130E+01	15.64
9.4000E+01	9.044E+00	6.783E-01	9.723E+00	1.056E+01	6.783E-01	1.124E+01	15.56
9.6000E+01	8.999E+00	6.779E-01	9.677E+00	1.050E+01	6.779E-01	1.118E+01	15.49
9.8000E+01	8.955E+00	6.775E-01	9.632E+00	1.044E+01	6.775E-01	1.112E+01	15.43
1.0000E+02	8.911E+00	6.771E-01	9.588E+00	1.038E+01	6.771E-01	1.106E+01	15.36
1.0200E+02	8.869E+00	6.767E-01	9.545E+00	1.033E+01	6.767E-01	1.101E+01	15.29
1.0400E+02	8.828E+00	6.763E-01	9.504E+00	1.028E+01	6.763E-01	1.095E+01	15.23
1.0600E+02	8.787E+00	6.758E-01	9.463E+00	1.022E+01	6.758E-01	1.090E+01	15.17
1.0800E+02	8.748E+00	6.754E-01	9.423E+00	1.017E+01	6.754E-01	1.085E+01	15.[illegible]
1.1000E+02	8.709E+00	6.750E-01	9.384E+00	1.012E+01	6.750E-01	1.080E+01	15.06
1.1200E+02	8.671E+00	6.746E-01	9.346E+00	1.007E+01	6.746E-01	1.075E+01	15.00
1.1400E+02	8.634E+00	6.742E-01	9.308E+00	1.003E+01	6.742E-01	1.070E+01	14.95

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR THE FFTF 100 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINT'
(SEC)			(MEGAWATTS)				(0/0)
1.1600E+02	8.598E+00	6.738E-01	9.272E+00	9.979E+00	6.738E-01	1.065E+01	14.90
1.1800E+02	8.562E+00	6.734E-01	9.236E+00	9.933E+00	6.734E-01	1.061E+01	14.85
1.2000E+02	8.527E+00	6.730E-01	9.200E+00	9.889E+00	6.730E-01	1.056E+01	14.80
1.2200E+02	8.493E+00	6.726E-01	9.166E+00	9.845E+00	6.726E-01	1.052E+01	14.75
1.2400E+02	8.460E+00	6.722E-01	9.132E+00	9.803E+00	6.722E-01	1.047E+01	14.71
1.2600E+02	8.427E+00	6.718E-01	9.099E+00	9.761E+00	6.718E-01	1.043E+01	14.66
1.2800E+02	8.395E+00	6.714E-01	9.066E+00	9.720E+00	6.714E-01	1.039E+01	14.62
1.3000E+02	8.363E+00	6.710E-01	9.034E+00	9.680E+00	6.710E-01	1.035E+01	14.58
1.3200E+02	8.332E+00	6.706E-01	9.003E+00	9.641E+00	6.706E-01	1.031E+01	14.54
1.3400E+02	8.302E+00	6.702E-01	8.972E+00	9.602E+00	6.702E-01	1.027E+01	14.50
1.3600E+02	8.272E+00	6.698E-01	8.941E+00	9.565E+00	6.698E-01	1.023E+01	14.46
1.3800E+02	8.242E+00	6.694E-01	8.912E+00	9.528E+00	6.694E-01	1.020E+01	14.43
1.4000E+02	8.213E+00	6.690E-01	8.882E+00	9.492E+00	6.690E-01	1.016E+01	14.39
1.4200E+02	8.185E+00	6.686E-01	8.854E+00	9.456E+00	6.686E-01	1.012E+01	14.36
1.4400E+02	8.157E+00	6.682E-01	8.825E+00	9.421E+00	6.682E-01	1.009E+01	14.32
1.4600E+02	8.130E+00	6.678E-01	8.798E+00	9.387E+00	6.678E-01	1.005E+01	14.29
1.4800E+02	8.103E+00	6.674E-01	8.770E+00	9.353E+00	6.674E-01	1.002E+01	14.26

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR THE FFTF 100 PERCENT CASE

TIME	********NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL UNCERTAINTY
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	
(SEC.)			(MEGAWATTS)				(O/O)
1.5000E+02	8.077E+00	6.669E-01	8.744E+00	9.321E+00	6.669E-01	9.987E+00	14.23
1.5200E+02	8.051E+00	6.665E-01	8.717E+00	9.288E+00	6.665E-01	9.955E+00	14.20
1.5400E+02	8.025E+00	6.661E-01	8.691E+00	9.256E+00	6.661E-01	9.922E+00	14.17
1.5600E+02	8.000E+00	6.658E-01	8.666E+00	9.225E+00	6.658E-01	9.891E+00	14.14
1.5800E+02	7.975E+00	6.654E-01	8.640E+00	9.194E+00	6.654E-01	9.860E+00	14.11
1.6000E+02	7.951E+00	6.650E-01	8.616E+00	9.164E+00	6.650E-01	9.829E+00	14.09
1.6200E+02	7.927E+00	6.646E-01	8.591E+00	9.135E+00	6.646E-01	9.799E+00	14.06
1.6400E+02	7.903E+00	6.642E-01	8.567E+00	9.105E+00	6.642E-01	9.770E+00	14.03
1.6600E+02	7.880E+00	6.638E-01	8.543E+00	9.077E+00	6.638E-01	9.740E+00	14.01
1.6800E+02	7.857E+00	6.634E-01	8.520E+00	9.048E+00	6.634E-01	9.712E+00	13.99
1.7000E+02	7.834E+00	6.630E-01	8.497E+00	9.021E+00	6.630E-01	9.684E+00	13.96
1.7200E+02	7.812E+00	6.626E-01	8.475E+00	8.993E+00	6.626E-01	9.656E+00	13.94
1.7400E+02	7.790E+00	6.622E-01	8.452E+00	8.967E+00	6.622E-01	9.629E+00	13.92
1.7600E+02	7.768E+00	6.618E-01	8.430E+00	8.940E+00	6.618E-01	9.602E+00	13.90
1.7800E+02	7.747E+00	6.614E-01	8.409E+00	8.914E+00	6.614E-01	9.575E+00	13.88
1.8000E+02	7.726E+00	6.610E-01	8.387E+00	8.888E+00	6.610E-01	9.549E+00	13.86
1.8200E+02	7.706E+00	6.606E-01	8.366E+00	8.863E+00	6.606E-01	9.524E+00	13.84

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SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR THE FFTF 100 PERCENT CASE

TIME	********NOMINAL VALUE***********			****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINTY
(SEC)			(MEGAWATTS)				(0/0)
1.8400E+02	7.685E+00	6.602E-01	8.345E+00	8.838E+00	6.602E-01	9.499E+00	13.82
1.8600E+02	7.665E+00	6.598E-01	8.325E+00	8.814E+00	6.598E-01	9.474E+00	13.80
1.8800E+02	7.645E+00	6.594E-01	8.305E+00	8.790E+00	6.594E-01	9.449E+00	13.78
1.9000E+02	7.626E+00	6.590E-01	8.285E+00	8.766E+00	6.590E-01	9.425E+00	13.77
1.9200E+02	7.606E+00	6.586E-01	8.265E+00	8.743E+00	6.586E-01	9.401E+00	13.75
1.9400E+02	7.587E+00	6.582E-01	8.245E+00	8.719E+00	6.582E-01	9.378E+00	13.73
1.9600E+02	7.568E+00	6.578E-01	8.226E+00	8.697E+00	6.578E-01	9.355E+00	13.72
1.9800E+02	7.550E+00	6.575E-01	8.207E+00	8.674E+00	6.575E-01	9.332E+00	13.70
2.0000E+02	7.531E+00	6.571E-01	8.188E+00	8.652E+00	6.571E-01	9.309E+00	13.69
2.0200E+02	7.513E+00	6.567E-01	8.170E+00	8.630E+00	6.567E-01	9.287E+00	13.67
2.0400E+02	7.495E+00	6.563E-01	8.152E+00	8.609E+00	6.563E-01	9.265E+00	13.66
2.0600E+02	7.478E+00	6.559E-01	8.134E+00	8.588E+00	6.559E-01	9.243E+00	13.64
2.0800E+02	7.460E+00	6.555E-01	8.116E+00	8.567E+00	6.555E-01	9.222E+00	13.63
2.1000E+02	7.443E+00	6.551E-01	8.098E+00	8.546E+00	6.551E-01	9.201E+00	13.62
2.1200E+02	7.426E+00	6.547E-01	8.081E+00	8.525E+00	6.547E-01	9.180E+00	13.61
2.1400E+02	7.409E+00	6.543E-01	8.064E+00	8.505E+00	6.543E-01	9.160E+00	13.59
2.1600E+02	7.393E+00	6.540E-01	8.047E+00	8.485E+00	6.540E-01	9.139E+00	13.58

SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR THE FFTF 100 PERCENT CASE

TIME (S)	********NOMINAL VALUE************* F.P.	U239/NP239	TOTAL	****VALUE W/ UNCERTAINTY********* F.P.	U239/NP239	TOTAL	TOTAL UNCERTAINTY (O/O)
			(MEGAWATTS)				
2.1800E+02	7.376E+00	6.536E-01	8.030E+00	8.466E+00	6.536E-01	9.119E+00	13.57
2.2000E+02	7.360E+00	6.532E-01	8.013E+00	8.446E+00	6.532E-01	9.100E+00	13.56
2.2200E+02	7.344E+00	6.528E-01	7.997E+00	8.427E+00	6.528E-01	9.080E+00	13.55
2.2400E+02	7.328E+00	6.524E-01	7.980E+00	8.408E+00	6.524E-01	9.061E+00	13.54
2.2600E+02	7.312E+00	6.520E-01	7.964E+00	8.390E+00	6.520E-01	9.042E+00	13.53
2.2800E+02	7.297E+00	6.516E-01	7.948E+00	8.371E+00	6.516E-01	9.023E+00	13.52
2.3000E+02	7.281E+00	6.512E-01	7.933E+00	8.353E+00	6.512E-01	9.004E+00	13.51
2.3200E+02	7.266E+00	6.509E-01	7.917E+00	8.335E+00	6.509E-01	8.986E+00	13.50
2.3400E+02	7.251E+00	6.505E-01	7.902E+00	8.317E+00	6.505E-01	8.968E+00	13.49
2.3600E+02	7.236E+00	6.501E-01	7.887E+00	8.300E+00	6.501E-01	8.950E+00	13.48
2.3800E+02	7.222E+00	6.497E-01	7.871E+00	8.282E+00	6.497E-01	8.932E+00	13.47
2.4000E+02	7.207E+00	6.493E-01	7.857E+00	8.265E+00	6.493E-01	8.914E+00	13.47
2.4200E+02	7.193E+00	6.489E-01	7.842E+00	8.248E+00	6.489E-01	8.897E+00	13.46
2.4400E+02	7.179E+00	6.486E-01	7.827E+00	8.231E+00	6.486E-01	8.880E+00	13.45
2.4600E+02	7.164E+00	6.482E-01	7.813E+00	8.215E+00	6.482E-01	8.863E+00	13.44
2.4800E+02	7.151E+00	6.478E-01	7.798E+00	8.198E+00	6.478E-01	8.846E+00	13.44
2.5000E+02	7.137E+00	6.474E-01	7.784E+00	8.182E+00	6.474E-01	8.830E+00	13.43

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SUMMARY OF DECAY POWER CALCULATION RESULTS

SUM OF ALL ASSEMBLIES FOR THE FFTF 100 PERCENT CASE

TIME	*******NOMINAL VALUE************			****VALUE W/ UNCERTAINTY*********			TOTAL
	F.P.	U239/NP239	TOTAL	F.P.	U239/NP239	TOTAL	UNCERTAINT
(S)			(MEGAWATTS)				(0/0)
3.0000E+02	6.831E+00	6.380E-01	7.469E+00	7.826E+00	6.380E-01	8.464E+00	13.33
4.0000E+02	6.370E+00	6.199E-01	6.989E+00	7.296E+00	6.199E-01	7.916E+00	13.26
5.0000E+02	6.023E+00	6.026E-01	6.625E+00	6.898E+00	6.026E-01	7.501E+00	13.21
6.0000E+02	5.741E+00	5.861E-01	6.328E+00	6.576E+00	5.861E-01	7.162E+00	13.18
7.0000E+02	5.503E+00	5.704E-01	6.073E+00	6.302E+00	5.704E-01	6.872E+00	13.16
8.0000E+02	5.295E+00	5.555E-01	5.850E+00	6.063E+00	5.555E-01	6.619E+00	13.14
9.0000E+02	5.110E+00	5.413E-01	5.651E+00	5.852E+00	5.413E-01	6.393E+00	13.13
1.0000E+03	4.943E+00	5.277E-01	5.471E+00	5.661E+00	5.277E-01	6.189E+00	13.12

END OF PROBLEM