ATTACHMENT IV

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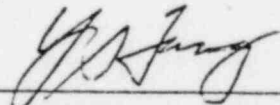INITIATED AT 35% POWER AND 75% FLOW CONDITIONS

CRBRP NCVP PRE-TEST PREDICTION OF
FOTA TEMPERATURES FOR FFTF NATURAL CIRCULATION TEST
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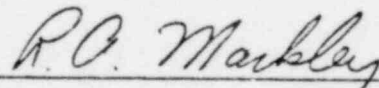
by

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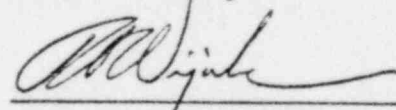Y. S. Tang

November 1980

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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, $Q_c(x,\tau)$ as

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

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 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 $G(x,\tau)$ is also input from CØBRA-WC analyses. Both $Q_{ex}(x,\tau)$ and $G(x,\tau)$ are based on nominal conditions in the CØBRA-WC code. It is conservative to use $Q_{ex}(x,\tau)$ and $G(x,\tau)$ 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/ØRE-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
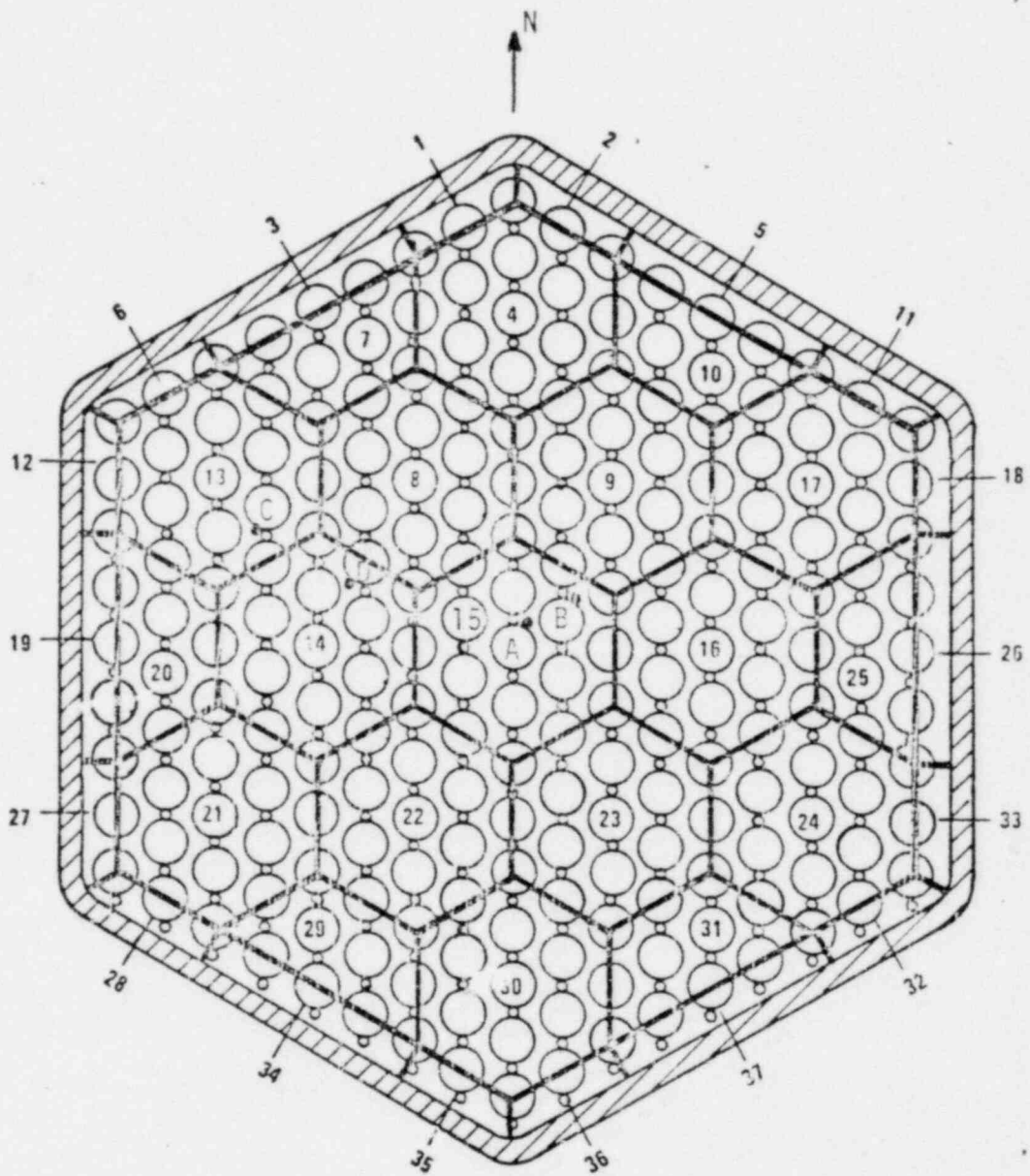
NOTES: Location A for Row 2 FOTA on Rod (9,9) FFTF Designation
Location B for Row 2 FOTA on Rod (8,8) FFTF Designation
Location C for Row 6 FOTA on Rod (4,4) FFTF Designation
Location D for Row 6 FOTA on Rod (6,6) FFTF Designation

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   1.035

*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 $Q_{ex}(x,\tau)$ and $G(x,\tau)$. 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 out-let 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 pro-vided 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 signi-ficant 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; correc-tions were made for the recent shield/orifice assembly pressure loss co-efficient 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; correc-tion 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 irradia-
  tion 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 redistri-
  bution 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
  Apperdix 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 distribu-
tion 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 +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 tempera-
tures 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, depend-
ing 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 dis-
cussed 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 uncer-
tainties 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 calcu-
lations 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.
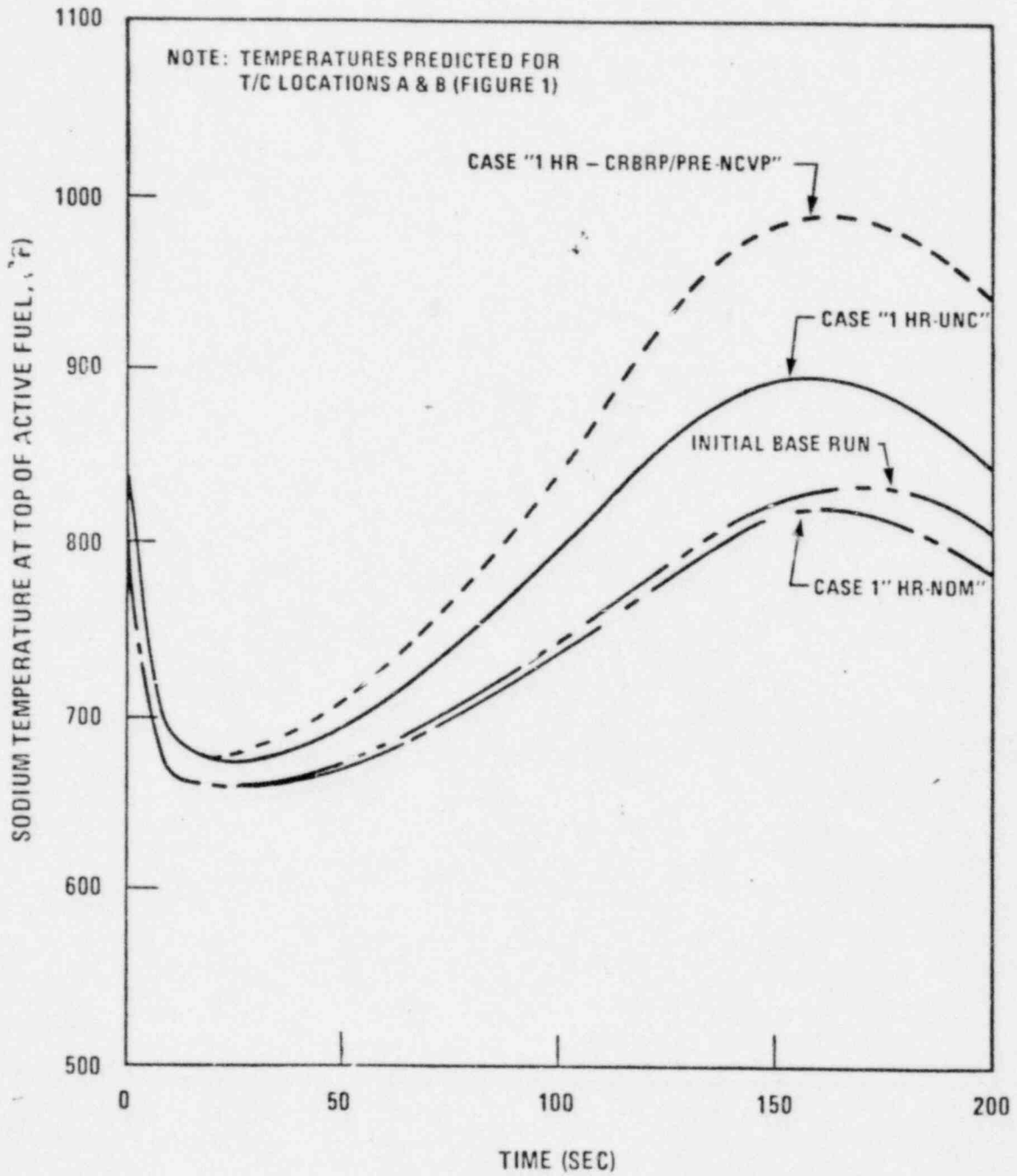
Figure 2. Sodium Temperatures at Top of Active Core Axial Position for Row 2 FOTA with 1 Hour Irradiation History (Test Initiated from 35% Power 75% Flow)
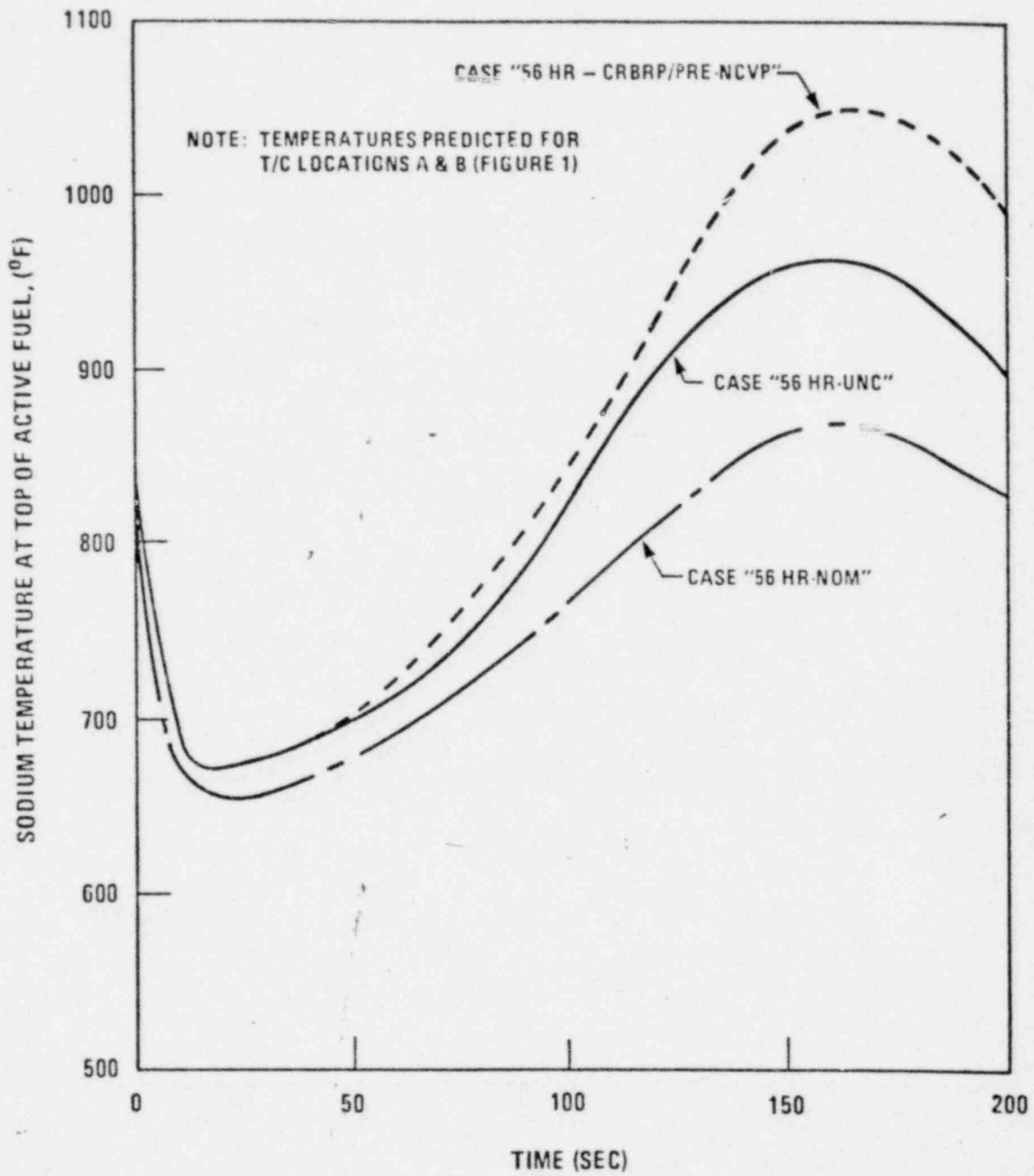
Figure 3. Sodium Temperatures at Top of Active Core Axial Position for Row 2 FOTA with 56 Hour Irradiation History (Test Initiated from 35% Power/75% Flow)
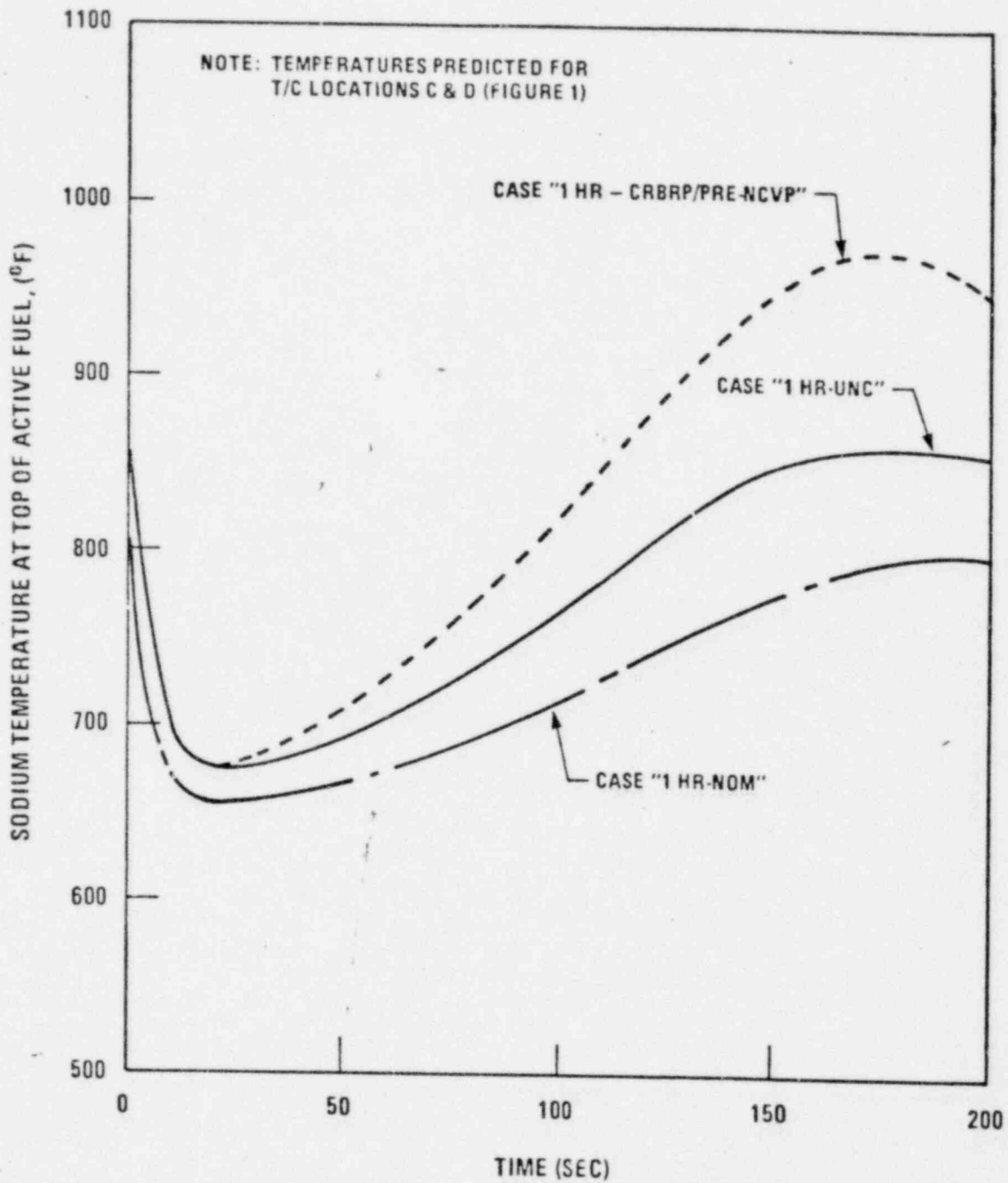
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Figure 4. Sodium Temperatures at Top of Active Core Axial Position for Row 6 FOTA with 1 Hour Irradiation History (Test Initiated from 35% Power/75% Flow)
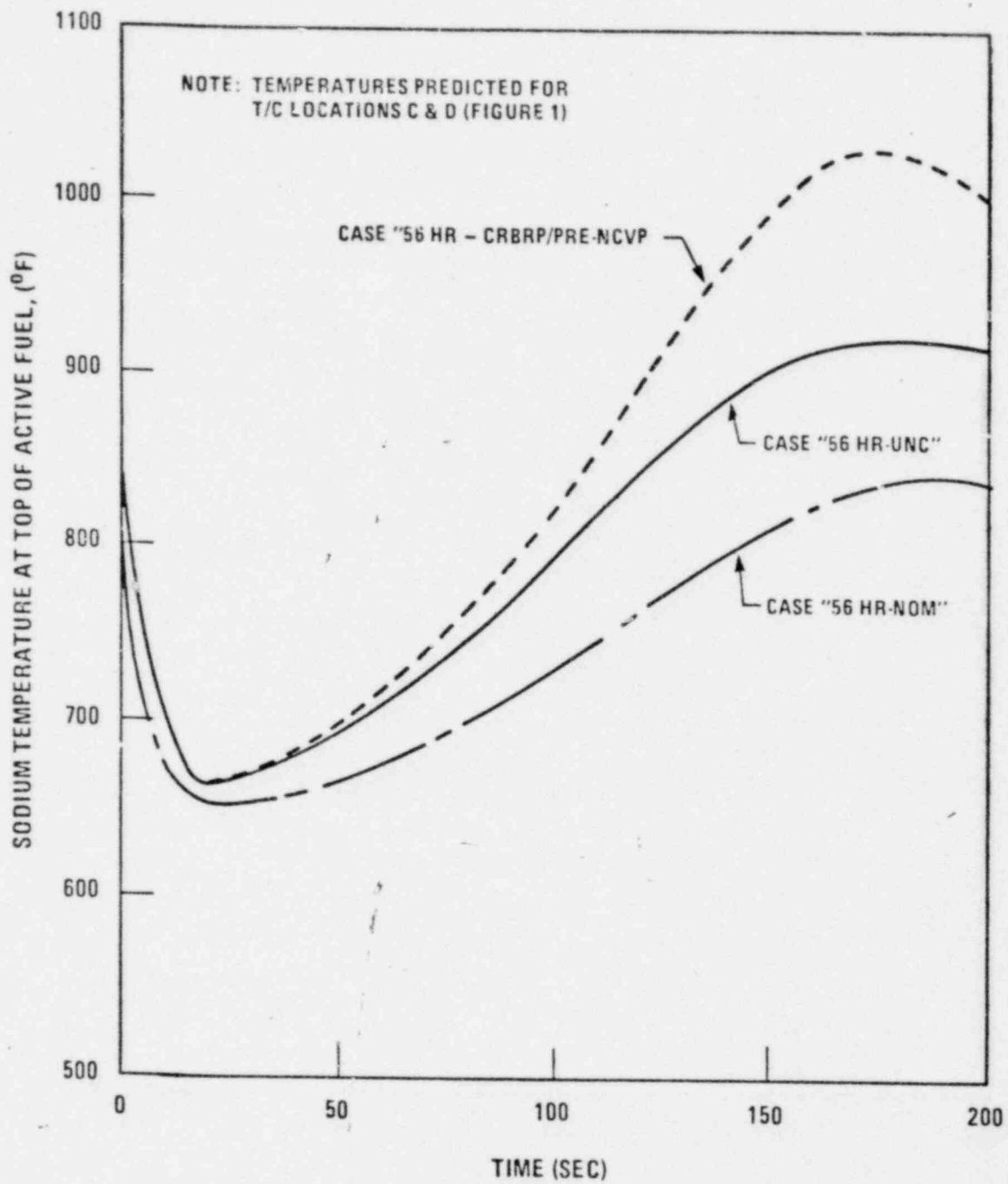
5005-3

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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1. R.D. Coffield, et al., "LMFBR Natural Circulation Verification Program (NCVP) Review of Experimental Facilities and Testing Recommendations", WARD-NC-3045-1, July 1977.

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

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9. 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.

10. T.L. George, K.L. Basehore and W.A. Prather, "CØBRA-WC Model and Predictions for a Fast Reactor Natural Circulation Transient", AIChE Symposium Series, 76, No. 199, p. 205 (1980).

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

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

APPENDIX 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.
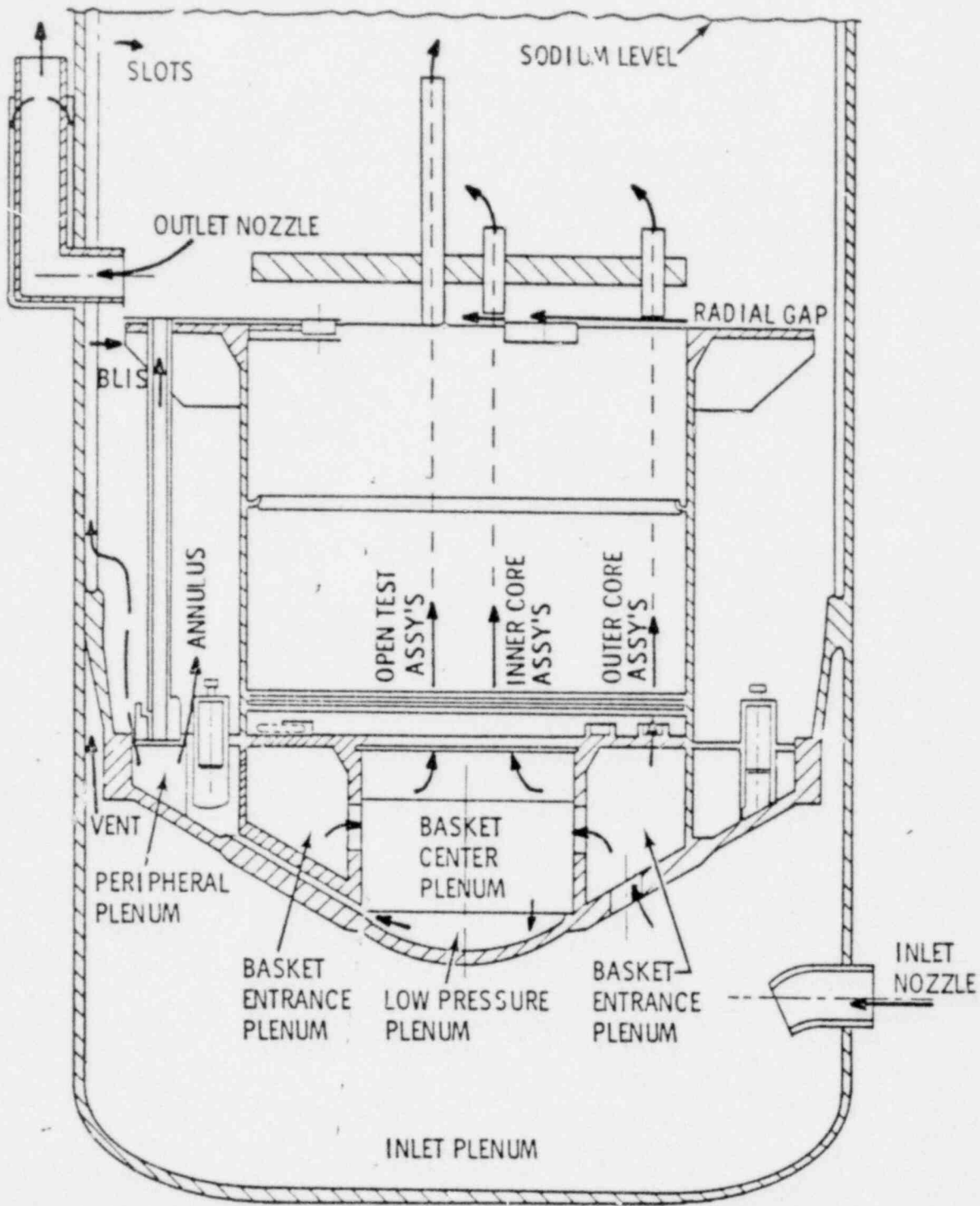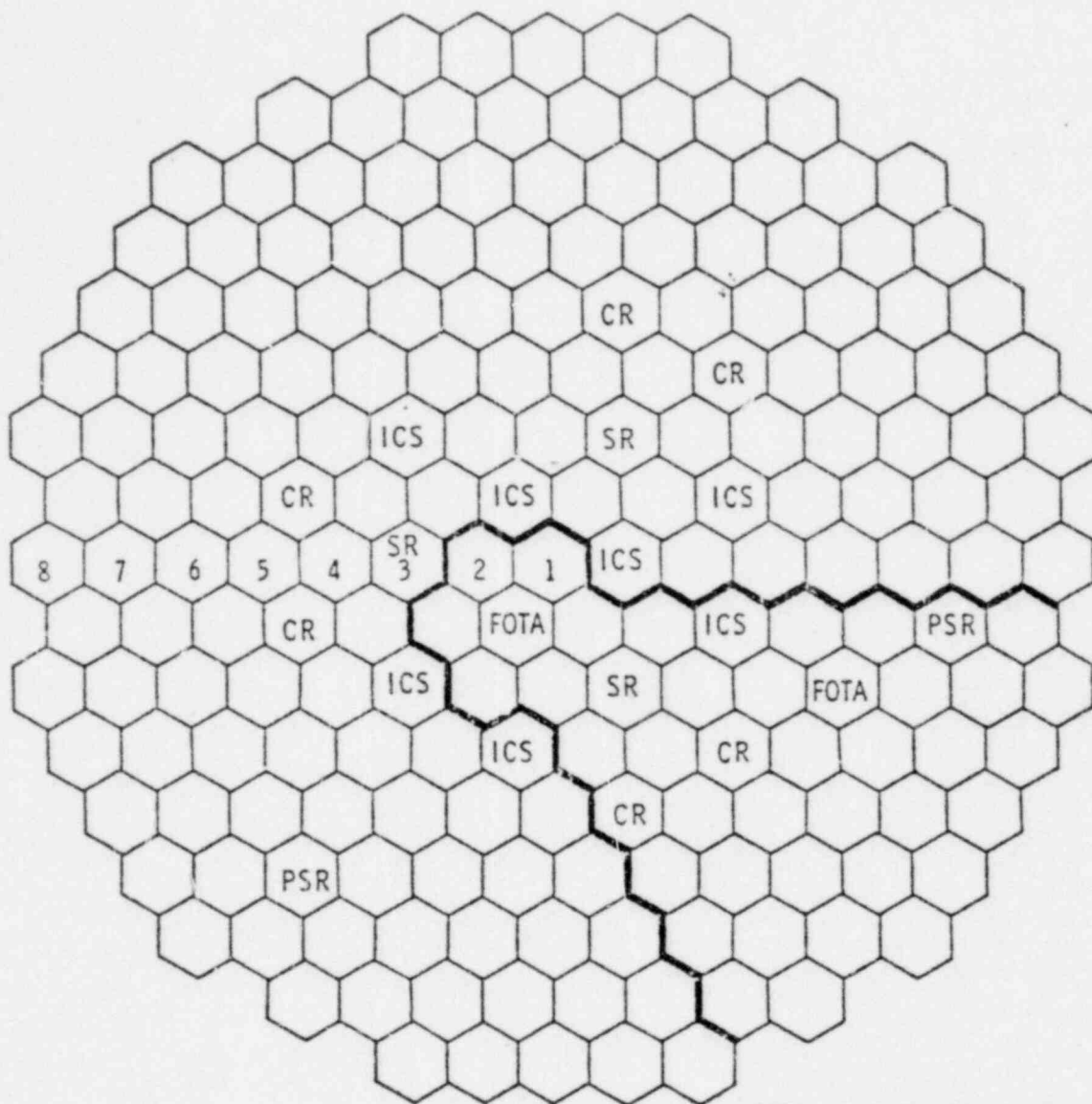
FIGURE A-1

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 operating power.

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.

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ATTACHMENT VI

MICROFICHE OF THE DEMØ CODE (FFTF MODEL) -
PROGRAM MODIFICATIONS, INPUT AND OUTPUT DATA

MICROFICHE OF THE DEMØ CODE (FFTF MODEL) - PROGRAM MODIFICATIONS MADE
SINCE SUBMITTAL BY THE REFERENCE LETTER, EDITED INPUT AND OUTPUT DATA

Reference: Letter S:L:890, P. S. Van Nort to R. P. Denise, "Trans-
mittal of DEMØ Computer Code, Revision 4," dated April 2,
1976.

Three sets of microfiche related to WARD-94000-00321 report are enclosed:

(1) 35% power, 75% flow best estimate case (2 pages)

(2) 35% power, 75% flow design case with 125% nominal decay heat (2 pages)

(3) 35% power, 75% flow design case with 75% nominal decay heat (2 pages)