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IN-VESSEL THERMAL-HYDRAULIC ANALYSIS

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W. T. Sha, W. L. Baumann, H. M. Domanus, D. Mohr  
R. C. Schmitt, and J. E. Sullivan  
Argonne National Laboratory  
9700 South Cass Avenue  
Argonne, Illinois 60439

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## IN-VESSEL THERMAL-HYDRAULIC ANALYSIS

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In-vessel thermal-hydraulic analysis<sup>1</sup> is new, and it includes all components, i.e., reactor core, upper plenum, lower plenum, piping, etc., in a reactor vessel. In the past, the standard calculational procedure assumes that detailed core thermal-hydraulic analysis can be performed separately and independently from the upper and lower plenum. While this assumption is valid for reactors operated under high flow conditions, recent experimental evidence<sup>2,3,4</sup> as well as analyses<sup>1,5,6,7</sup> have shown that there is a strong coupling between reactor core and plena under mixed and natural-convection modes. In order to account for the coupling effects properly, in-vessel thermal hydraulic analysis must be carried out. Furthermore, one of the unique features of in-vessel analysis is that the boundary conditions used in the analysis are well defined.

This paper presents some recent results obtained from the COMMIX-1A code<sup>8</sup> for the EBR-II reactor transient test No. 10, Phase 2<sup>5</sup>. Figure 1 presents a schematic sketch of the EBR-II primary cooling system. Both the reactor vessel and the neutron shield assembly and assembly arrangement in the reactor core are shown in Figs. 2 and 3 respectively. The computational grid system used in COMMIX-1A can be found in Fig. 4. Reactor flow and power transients are shown in Fig. 5. Figures 6-9 present velocity and temperature distributions at steady state and  $t$  (time) = 60 sec. Finally, a comparison between the calculated results from COMMIX-1A and experimental measurements are presented in Figs. 10-12 for outlet temperatures for driver subassembly of XX08, top-of-core temperature for driver subassembly of XX08, and low-pressure plenum mass flow respectively.

Based on the comparison between the experimental data and calculational results, it is revealed that: (1) a strong coupling between the reactor components of the EBR-II primary system is taking place under mixed and natural convection conditions, and thus the reactor components inside a reactor vessel cannot be modeled separately from each other; and, (2) the reverse flow in the lower pressure plenum is caused by buoyancy within the core region and sustained by the heat capacity of the reactor structures.

### References

1. W. T. Sha, et al., "Three Dimensional Analysis of LMFBR Decay Heat Removal System," Proc. of the LMFBR Safety Topical Meeting, Lyon-Ecully, France (July 19-23, 1982).
2. R. M. Singer, D. Mohr, and J. L. Gillette, "Transition from Forced to Natural Convection Flow in Adverse Thermal Conditions," Proc. 7th Inter. Heat Transfer Conference, München, BRD, Paper NR-24 (September 6-10, 1982).

References (Contd.)

3. E. L. Gluekler, T. A. Shih, and S. S. Grewal, "Experimental and Analytical Investigation of the Reactor Flow Distribution during Shutdown Cooling," Proc. of the LMFBR Safety Topical Meeting, Lyon-Ecully, France (July 19-23, 1982).
4. T. R. Beaver, H. G. Johnson, and R. L. Stover, "Transient Testing of the FFTF for Decay Heat Removal by Natural Circulation," Proc. of the LMFBR Safety Topical Meeting, Lyon-Ecully, France (July 19-23, 1982).
5. W. L. Baumann, H. M. Domanus, D. Mohr, and W. T. Sha, "EBR-II In-Vessel Natural Circulation Analysis," NUREG/CR-2821, ANL-82-66 (September 1982).
6. S. P. Vanka, H. M. Domanus, and W. T. Sha, "COMMIX-1A Three-Dimensional In-Vessel Simulation of the FFTF Transient Thermal Hydraulics," NUREG/CR-2773, ANL-CT-82-14 (May 1982).
7. R. Webster, "Convection Flows during Low Power Natural Circulation Experiments on the PFR," Proc. of the LMFBR Safety Topical Meeting, Lyon-Ecully, France (July 19-23, 1982).
8. H. M. Domanus, W. T. Sha, R. C. Schmitt, and V. L. Shah, "COMMIX-1A: A Three-Dimensional Transient Single-Phase Computer Program for Thermal Hydraulic Analysis of Single and Multicomponent Systems," Draft Report (September 1982).

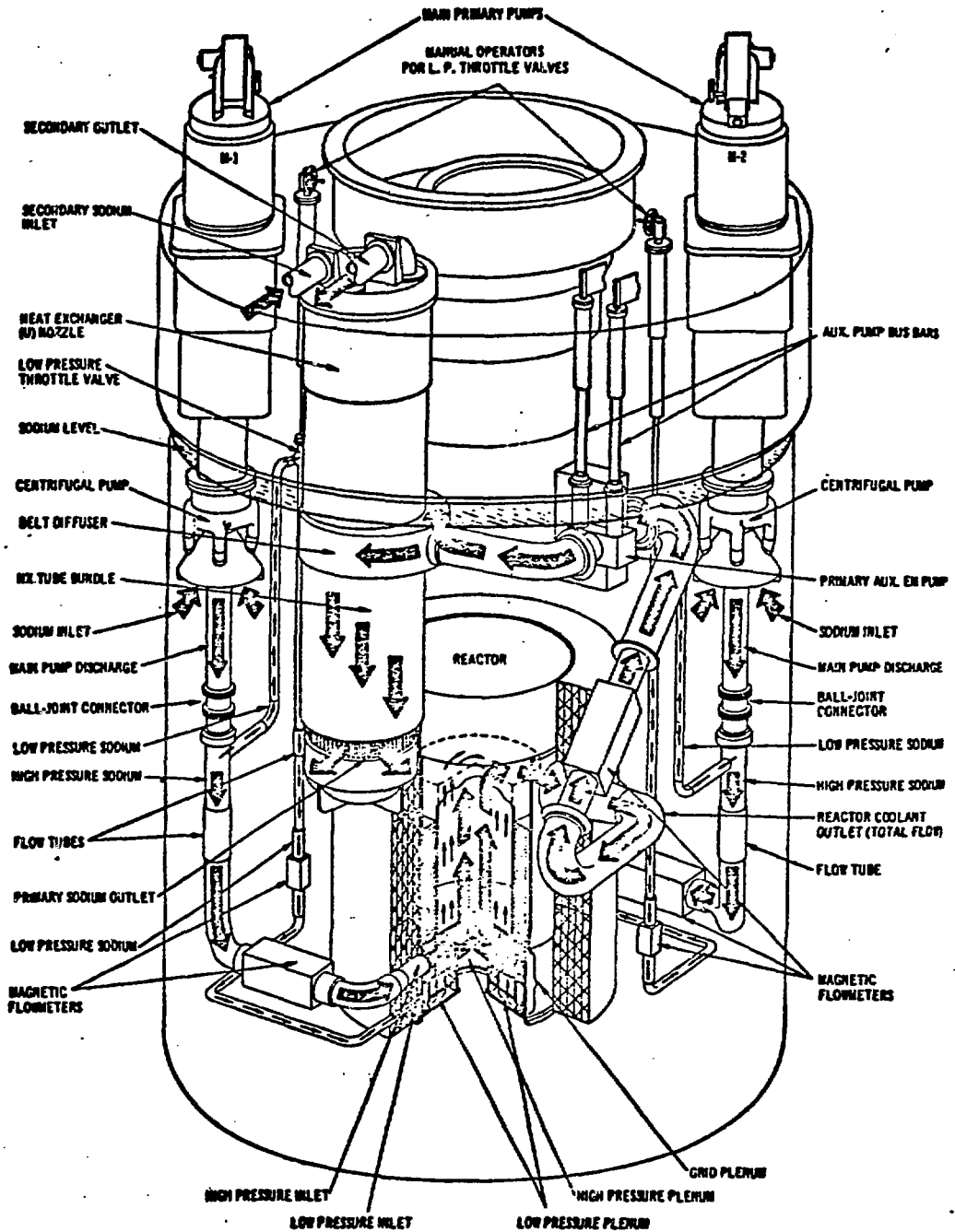


Fig. 1. EBR-II primary cooling system

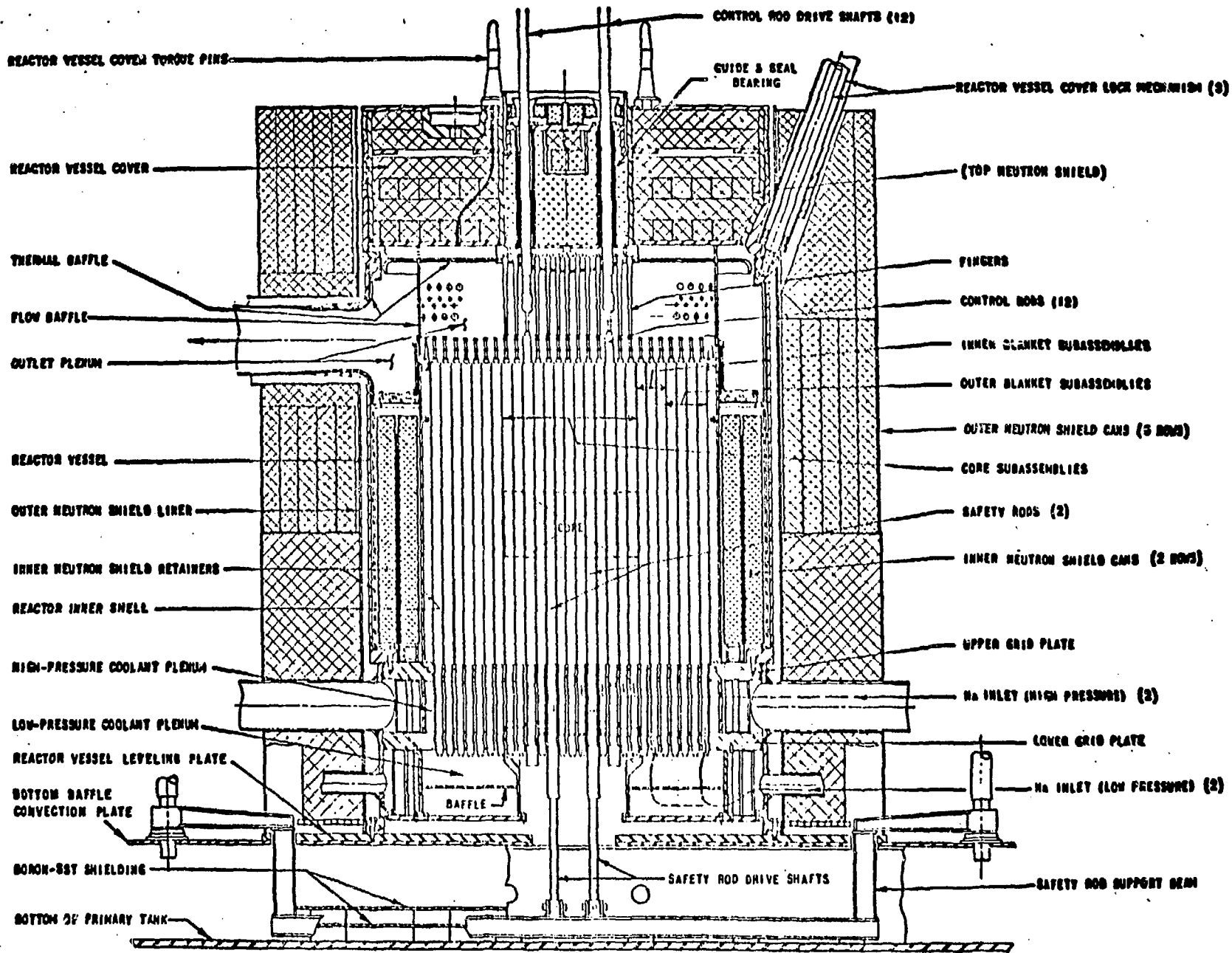
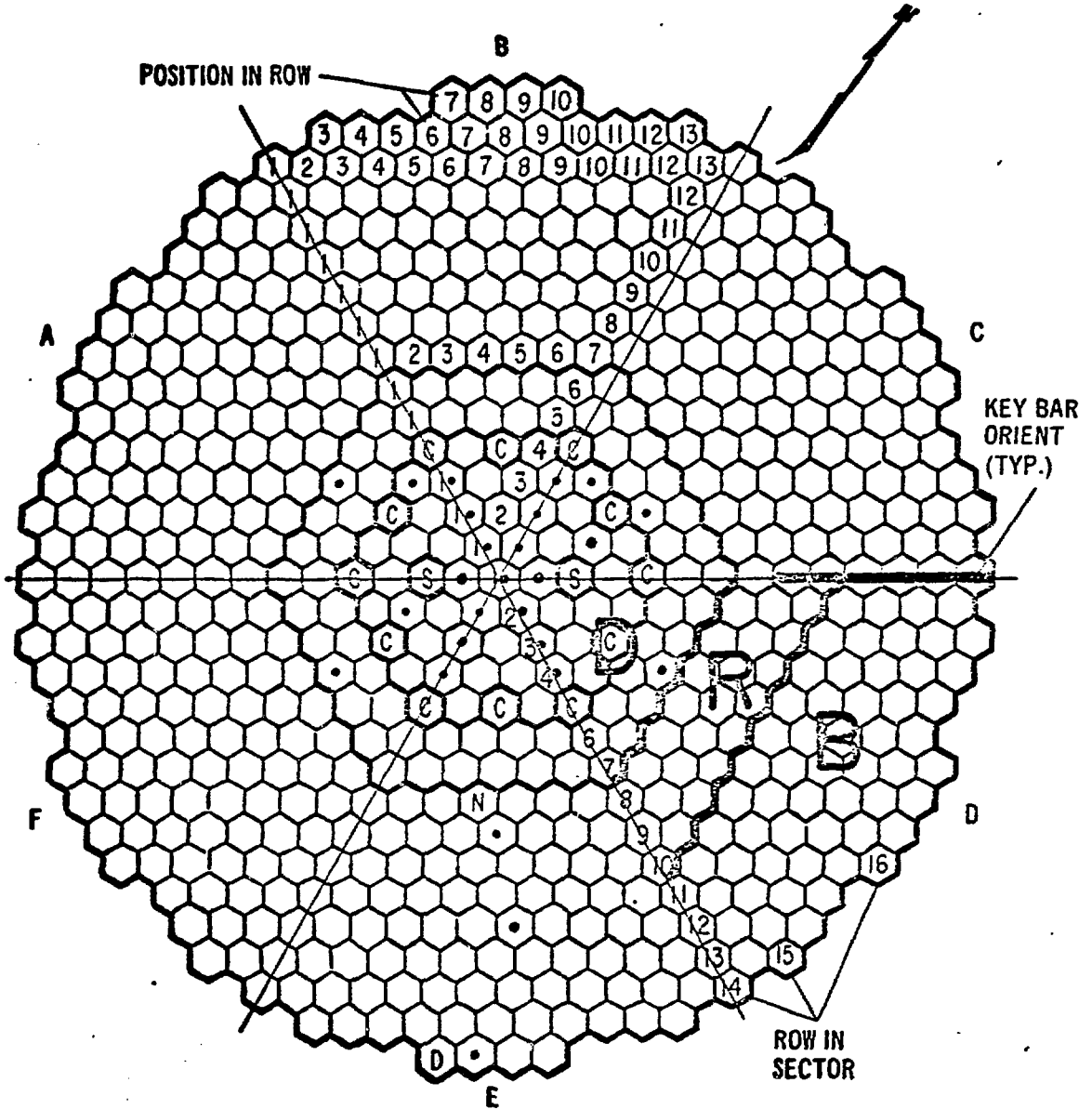


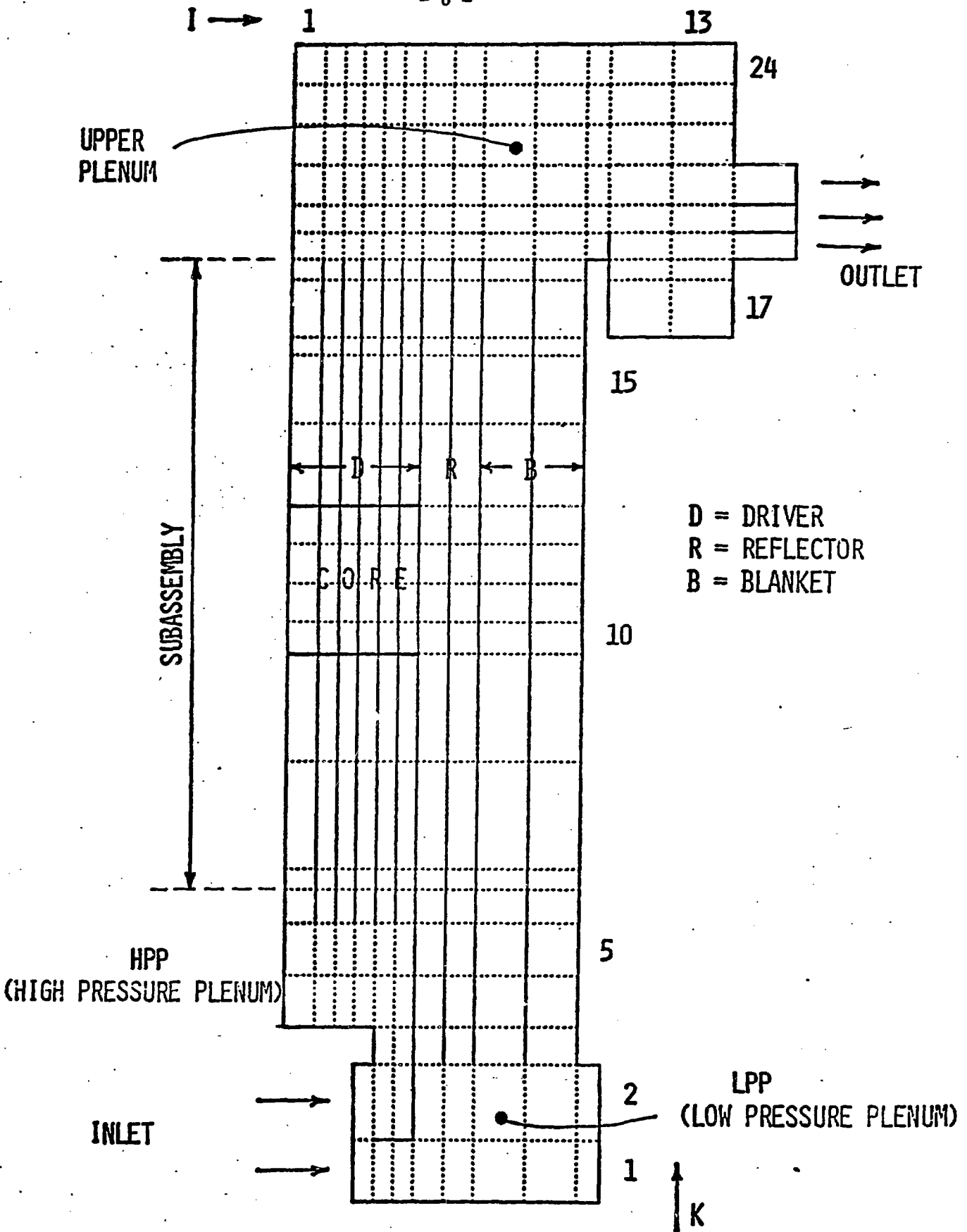
Fig. 2. Reactor Vessel and Neutron Shield Assembly



**LEGEND**

1. SECTORS	A to F	
2. CONTROL RODS (12)	C	D = Driver
3. SAFETY RODS (2)	S	R = Reflector
4. THERMOCOUPLES (26)	.	B = Blanket
5. FIXED DUMMY	D	
6. NEUTRON SOURCE	N	
7. GRID POSITIONS		
CORE	61	
INNER BLANKET	66	
OUTER BLANKET	510	
TOTAL	<u>637</u>	

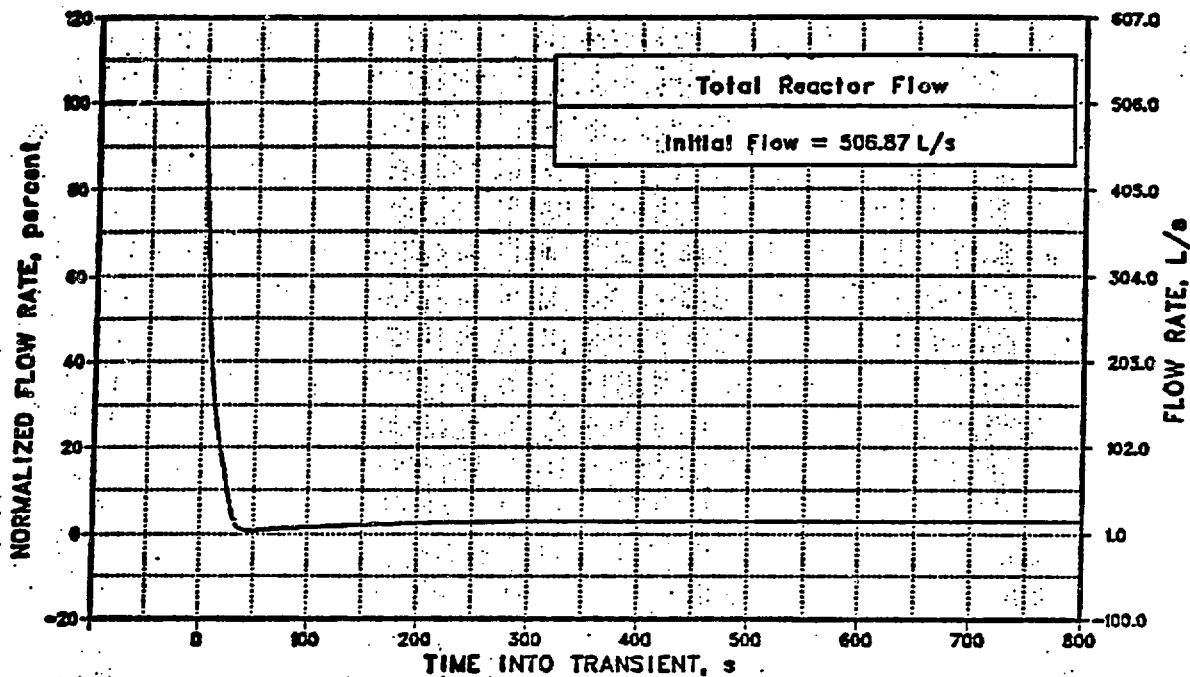
**Fig. 3. Subassembly Arrangement in the Reactor**



D = DRIVER  
 R = REFLECTOR  
 B = BLANKET

Fig. 4. GRID SYSTEM

1 RALPH TEST10-02



25 RALPH TEST10-02

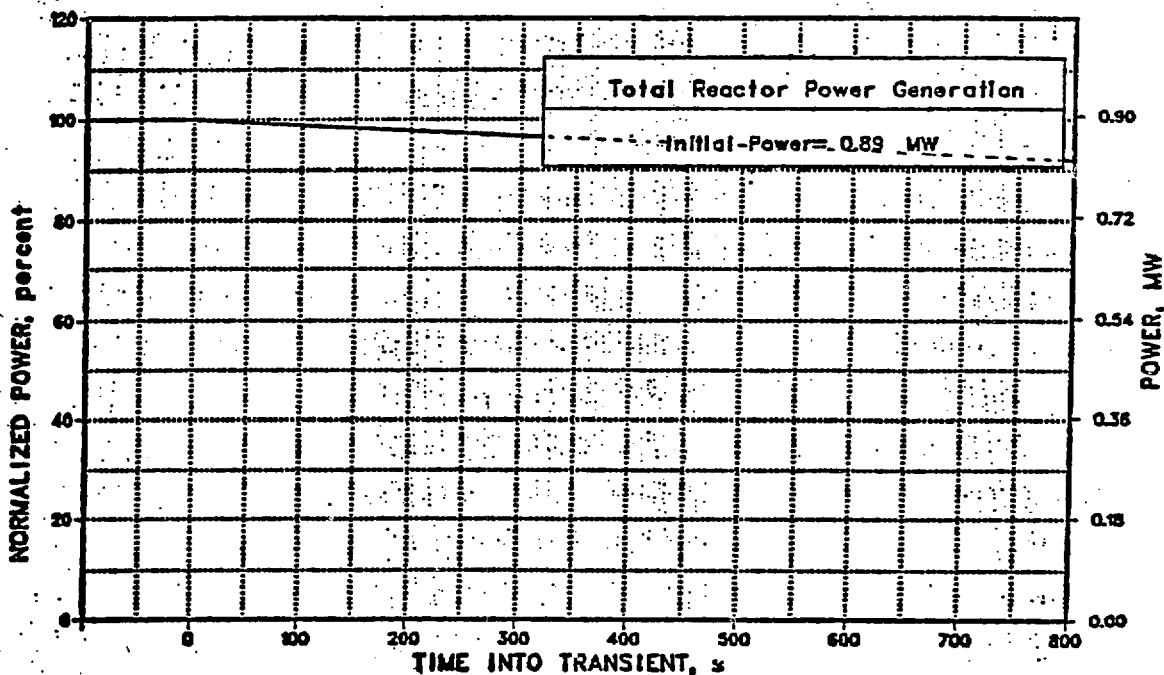


Fig. 5. Transient Functions for Total Reactor Flow and Total Reactor Power



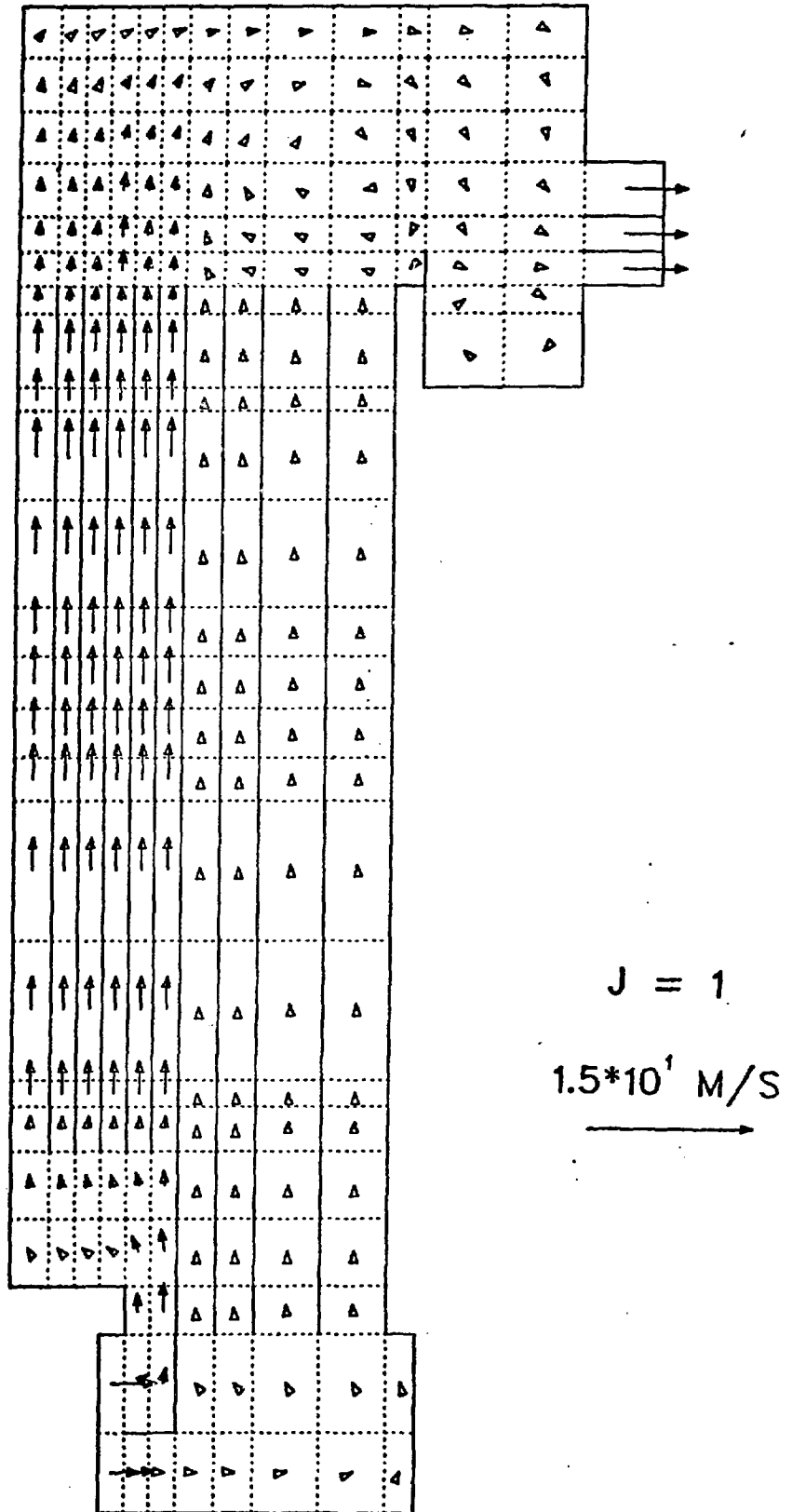


Fig. 6. Steady Velocity Distribution in the Azimuthal Plane  $J = 1$

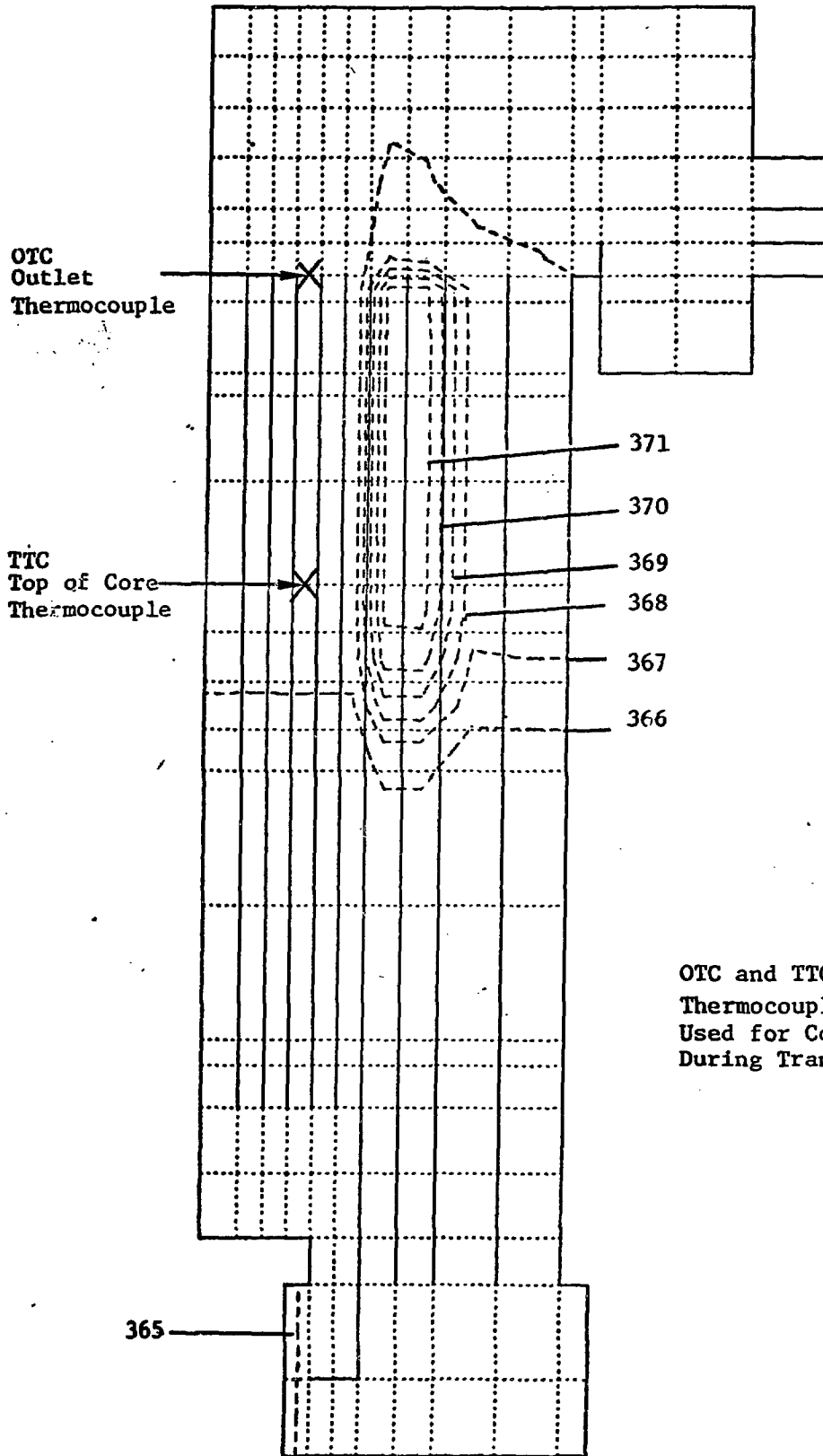
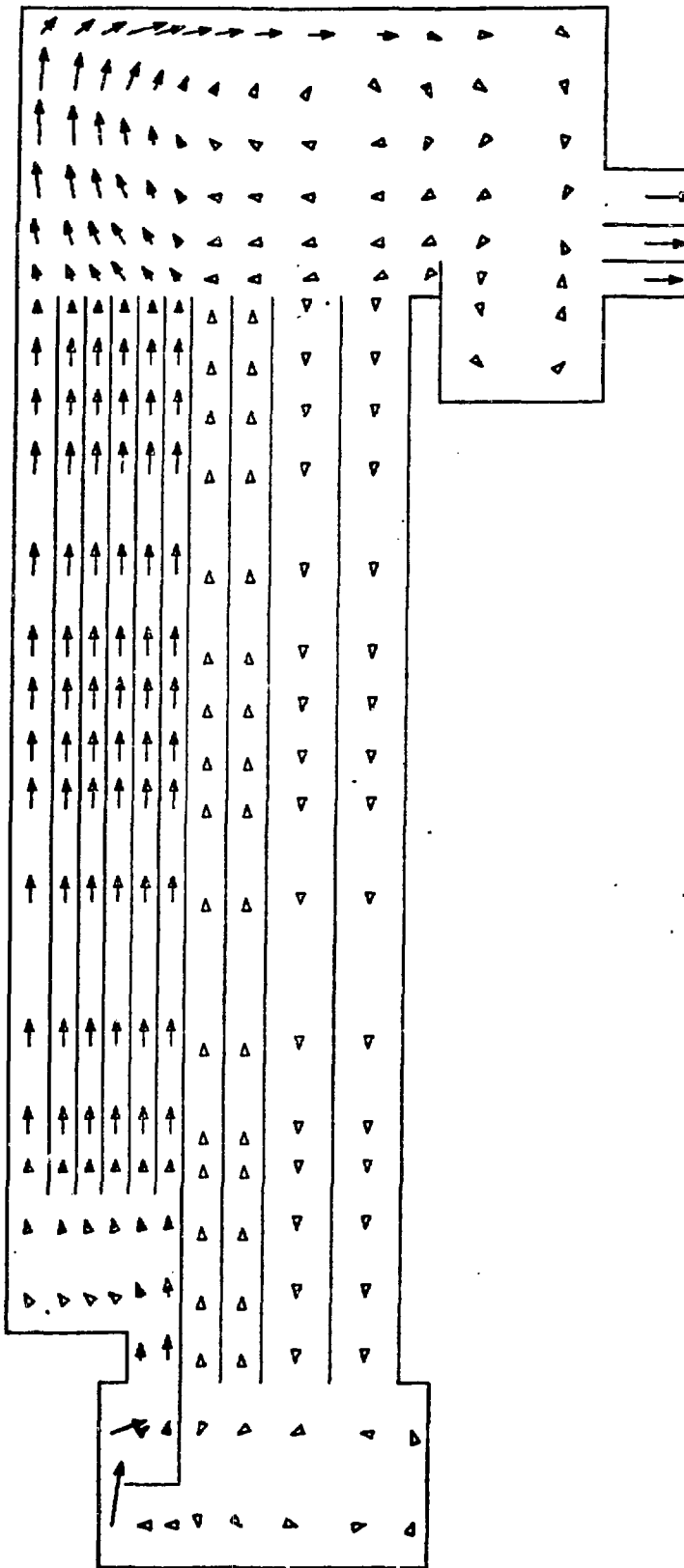


Fig. 7. -- Steady Isotherm Plot Showing Temperature in the Azimuthal Plane  $J = 1$  (Temp. in  $^{\circ}\text{C}$ )

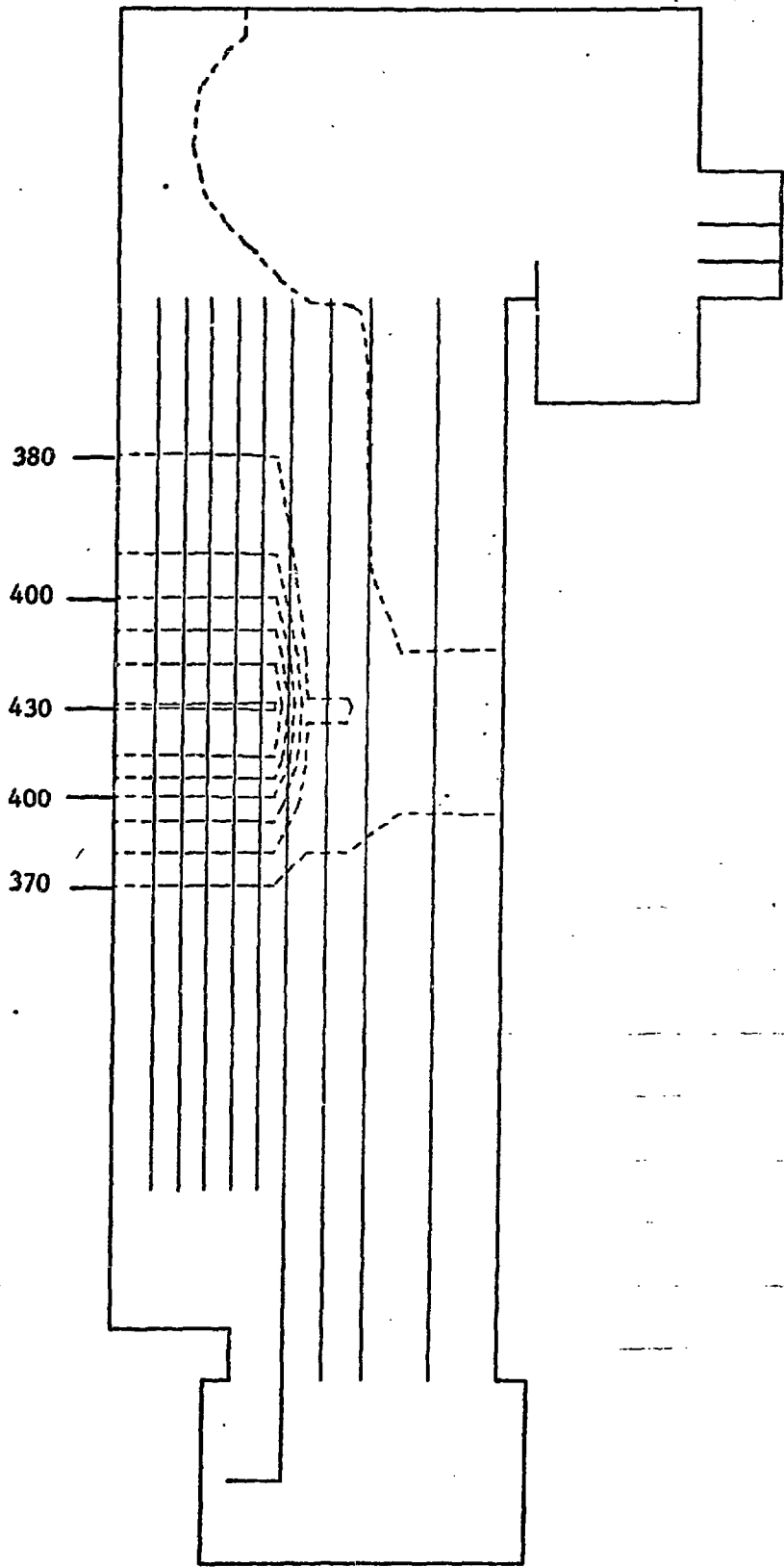
LOT 1 12.16.57 TUES 4 MAY, 1982 JOB-TEER2 , PRONNE DISSPLR VER 8.2



$J = 1$   
→ 0.19 M/S

TIME  
60.00 SEC.

Fig. 8. Velocity Distribution at Time = 60 s



$J = 1, 0$

TIME  
60.0 SEC.

Fig. 9. Temperature Distribution at Time = 60 s

# XX08 OTC TEMP. TL 40118

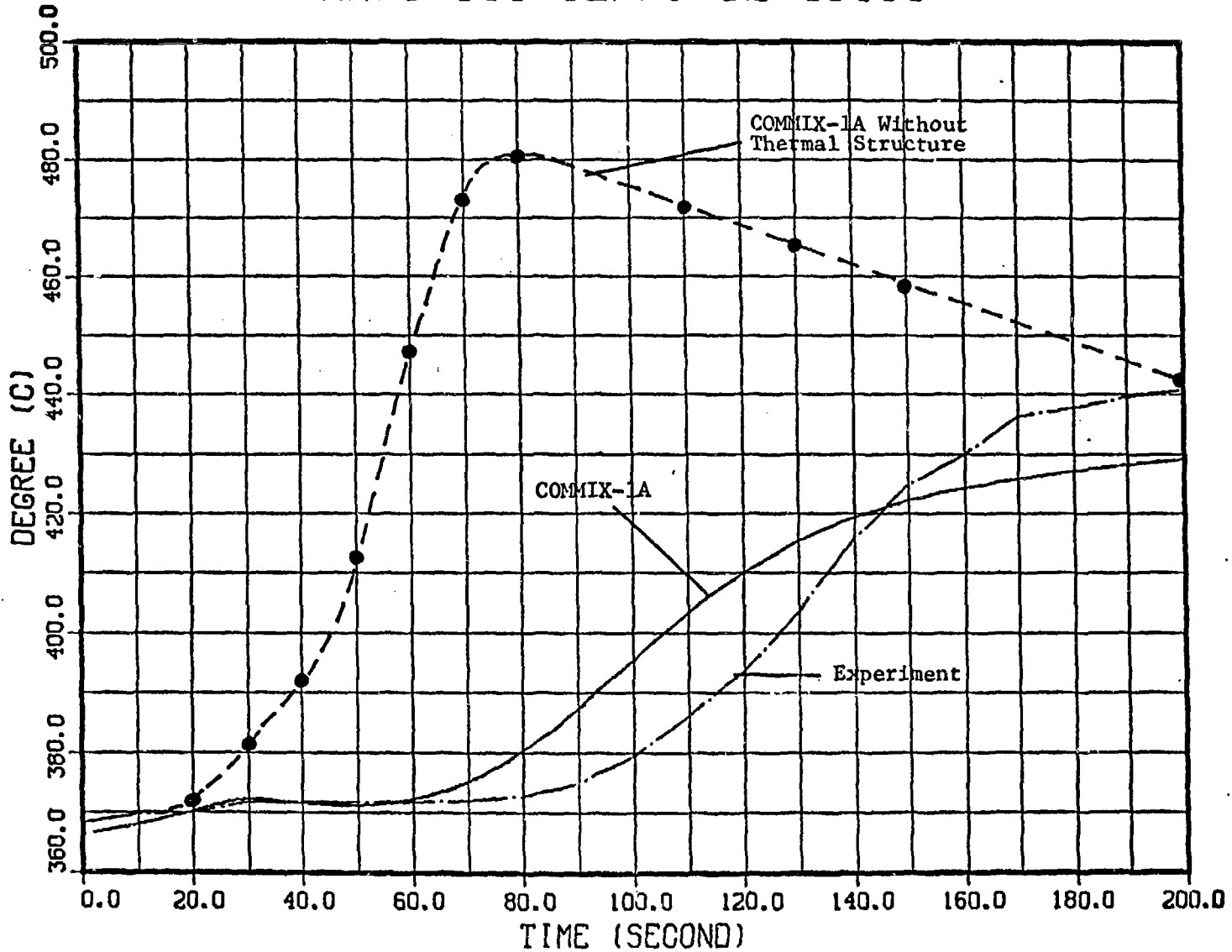


Fig. 10. Outlet Temperatures for Driver Subassembly XX08

# XX08 TTC TEMP. TL 40113

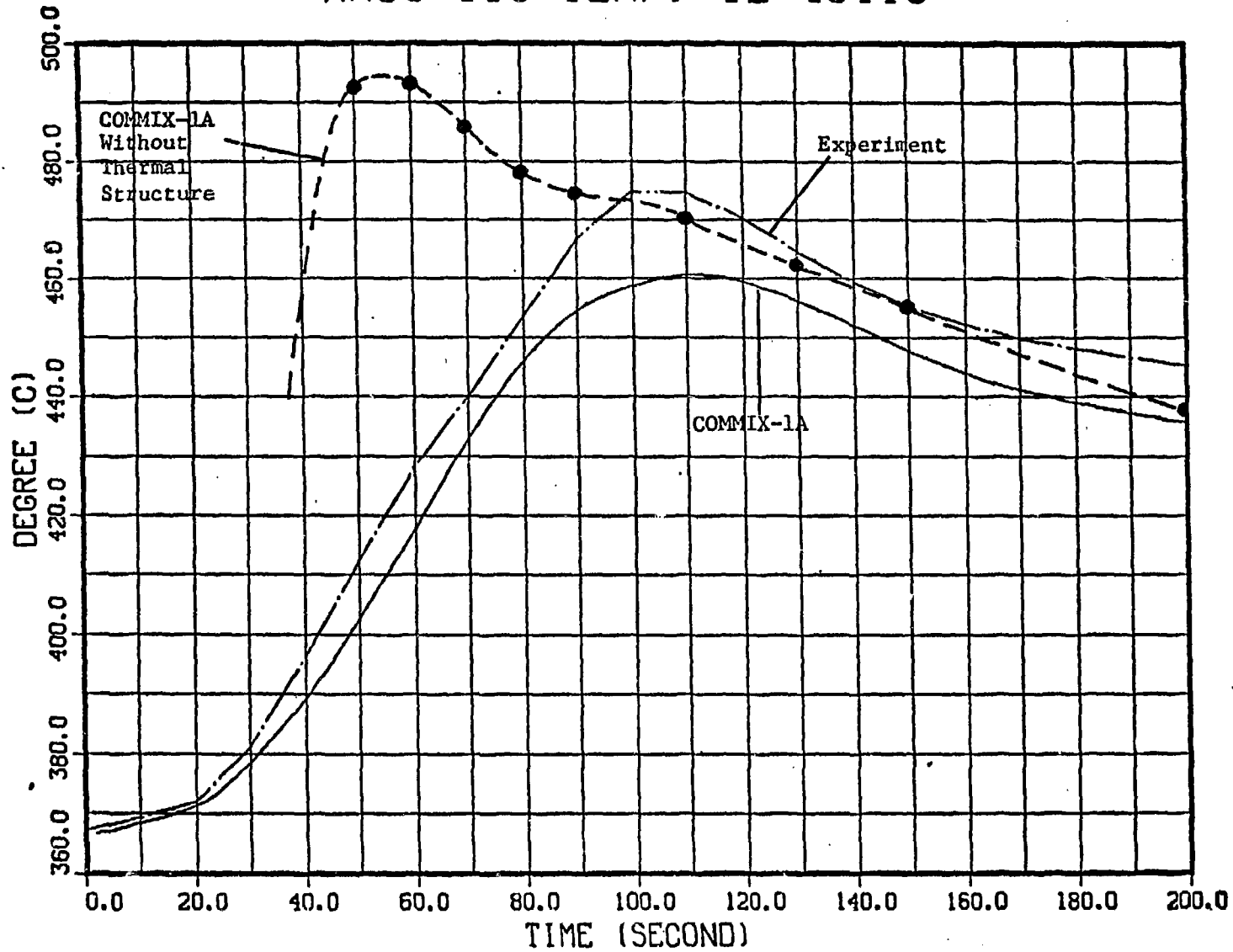


Fig. 11. Top-of-Core Temperatures for Driver Subassembly XX08

# NORMALIZED LPP-FLOW UL 40101

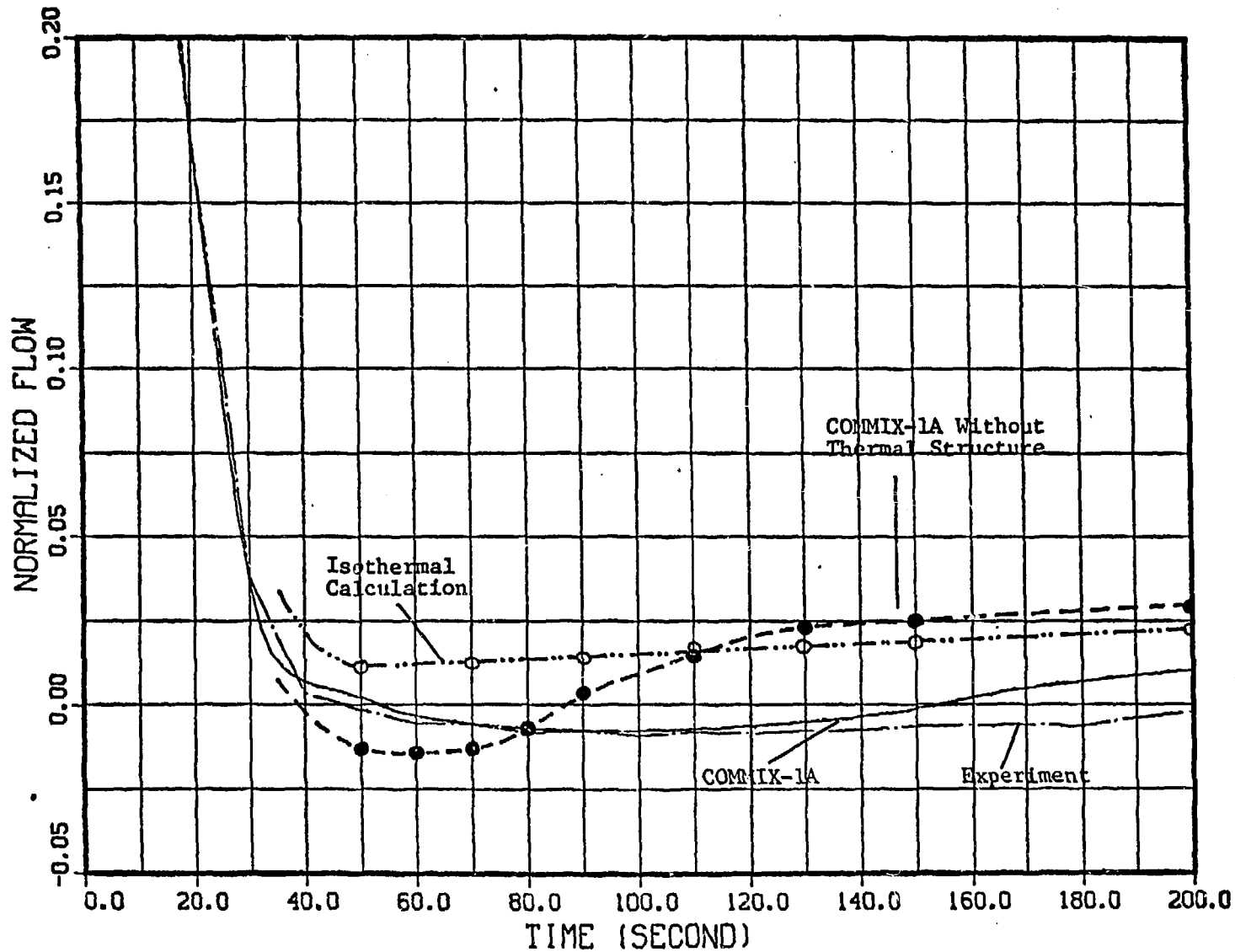


Fig. 12. Low Pressure Plenum Mass Flow