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E. U. Khan
T. L. George
D. R. Rector

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Pacific Northwest Laboratory
Richland, Washington 99352

COBRA-WC PRETEST PREDICTIONS AND POST-TEST ANALYSIS OF THE FOTA TEMPERATURE DISTRIBUTION DURING FFTF NATURAL CIRCULATION TRANSIENTS ξ

E. U. Khan*, T. L. George, D. R. Rector*

The natural circulation tests of the Fast Flux Test Facility (FFTF) demonstrated a safe and stable transition from forced convection to natural convection and showed that natural convection may adequately remove decay heat from the reactor core. The COBRA-WC computer code (1, 2) was developed by the Pacific Northwest Laboratory (PNL) to account for buoyancy-induced coolant flow redistribution and interassembly heat transfer, effects that become important in mitigating temperature gradients and reducing reactor core temperatures when coolant flow rate in the core is low. This report presents work sponsored by the U.S. Department of Energy (DOE) with the objective of checking the validity of COBRA-WC during the first 220 seconds (sec) of the FFTF natural-circulation (plant-startup) tests using recorded data from two instrumented Fuel Open Test Assemblies (FOTAs). Comparison of COBRA-WC predictions of the FOTA data is a part of the final confirmation of the COBRA-WC methodology for core natural-convection analysis. COBRA-WC <u>pretest</u> predictions of the FOTA temperature distributions were made using expected operating parameters; then, using the same thermal-hydraulic models, COBRA-WC <u>post-test</u> calculations of the FOTA temperature distributions were made using actual operating parameters measured during the FFTF tests. The COBRA-WC analyses were made for natural-circulation transients starting from three conditions: 35% reactor power/75% coolant flow (35/75), 75% power/75% flow (75/75), and 100% power/100% flow (100/100).

A natural-circulation transient (NCT) is assumed to occur when the reactor is operating at power and loss of all offsite and onsite A.C. power results in a reactor scram and pump coastdown without additional pumping power from the auxiliary pony motors (3). A typical flow and power reduction rate during such an event is shown in Figure 1. The flow reduces to about 10% of steady-state, full-flow rate in about 40 sec and continues to drop off to about 2.5% at about 100 sec before it increases again. The power-to-flow ratio continuously changes. After the pumps have stopped at about 100 to

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120 sec, the driving force for coolant flow through the core is predominantly the difference in the gravitational head between the hot and cold legs of the primary heat transport system.

In the present study a plant code, DEMO(4), provided the boundary conditions for COBRA-WC. The DEMO code provided the core delayed neutron powers, dynamic average region temperatures, and inlet flow and average core pressure drops. Utilizing DEMO boundary conditions, a detailed whole-core flow and heat redistribution analysis of all the parallel core assemblies and bypass regions was performed by the COBRA code. This was done for a sector of the reactor core that included the bypass flow. The nodal representation distant from the region of interest was less detailed than in the area of interest, i.e., near the FOTA.

Figure 1 shows the sector of the FFTF core modeled. Figure 14 in reference 5 shows the various flow paths in the FFTF vessel. Two-step calculations were used with the COBRA-WC code. The first step utilized a single-channel node for each core assembly in the modeled sector of the core. The inter-assembly heat transfer and flow redistribution effects were modeled. A simplified upper plenum model was included to account for heat transfer between the upper chimneys and the plenum. The first step yielded assembly flow information and the temporal pressure drop across the fuel assembly for providing boundary conditions to a second step calculation that contained more detailed FOTA analysis. The second step calculation used a seven-assembly cluster, the central assembly being the FOTA assembly. The FOTA assemblies were modeled by COBRA-WC using 37 subchannels each, by lumping 25 standard subchannels and 12 rods into one interior channel and one rod. The surrounding driver assemblies adjacent to the row 6 FOTA were modeled by 19 flow channels.

Figure 2 shows a typical plot of comparison of the row 2 FOTA data and COBRA-WC predictions. The peak coolant temperature as well as the coolant temperature history was invariably predicted well by COBRA-WC [within the accuracies-of-code prediction (Ref. 5) and of temperature measurement (18^{o}F) (Ref. 6)]. There was generally a small delay (15-30 sec) in predicting the time of occurence of peak coolant temperature. The post-test calculated temperatures match the data better than the pretest predictions primarily because of improvements in the pressure drop boundary condition provided by the DEMO code and because actual reactor operating conditions and power history was available for the post-test analysis. In addition improved estimates were available for pressure drop loss coefficients for the various components within the reactor vessel.

Predicted interassembly flow redistribution increased the fraction of total flow into the row 2 FOTA from the steady-state value by as much as 25 to 30% during the natural circulation transient (Fig. 2). Intrasssembly flow redistribution caused an increase in flow to the hotter regions of the bundle by as much as 12% of the initial flow, in addition to the interassembly flow redistribution. For natural convection, the predicted total vessel flow was very sensitive to small changes in the total nozzle-to-nozzle pressure drop.

The comparison of pretest predictions with data verifies that a prior performance prediction, based on estimated boundary conditions, can conservatively predict peak coolant temperatures and natural-convection flows for transients initiating from different power levels and power-to-flow rates. The comparison of post-test prediction with data validates the calculation scheme.

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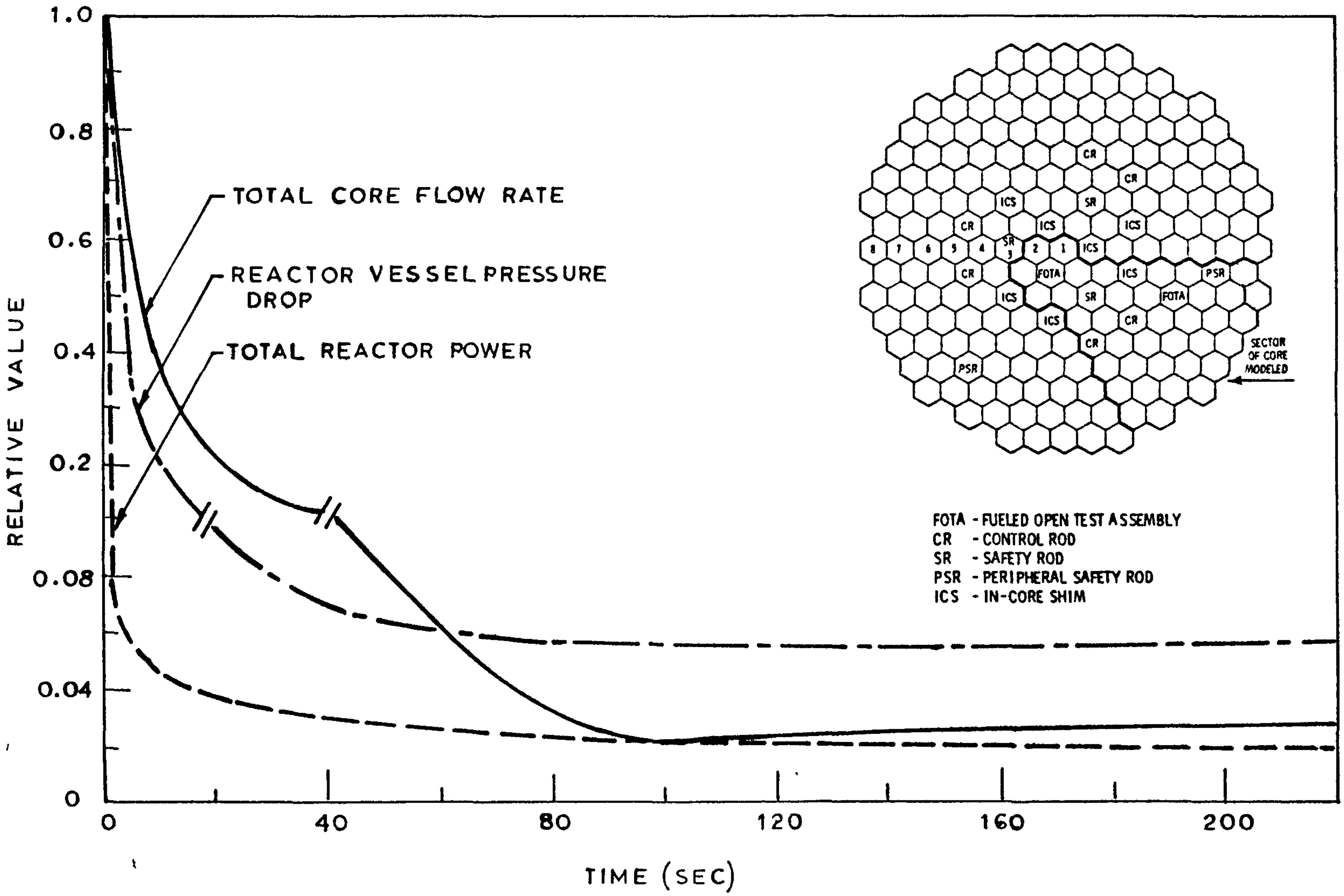

Figure 1. Flow and Power Fractions vs. time For Natural Convection Transient. Also shown is sector of core modeled for COBRA Analysis.

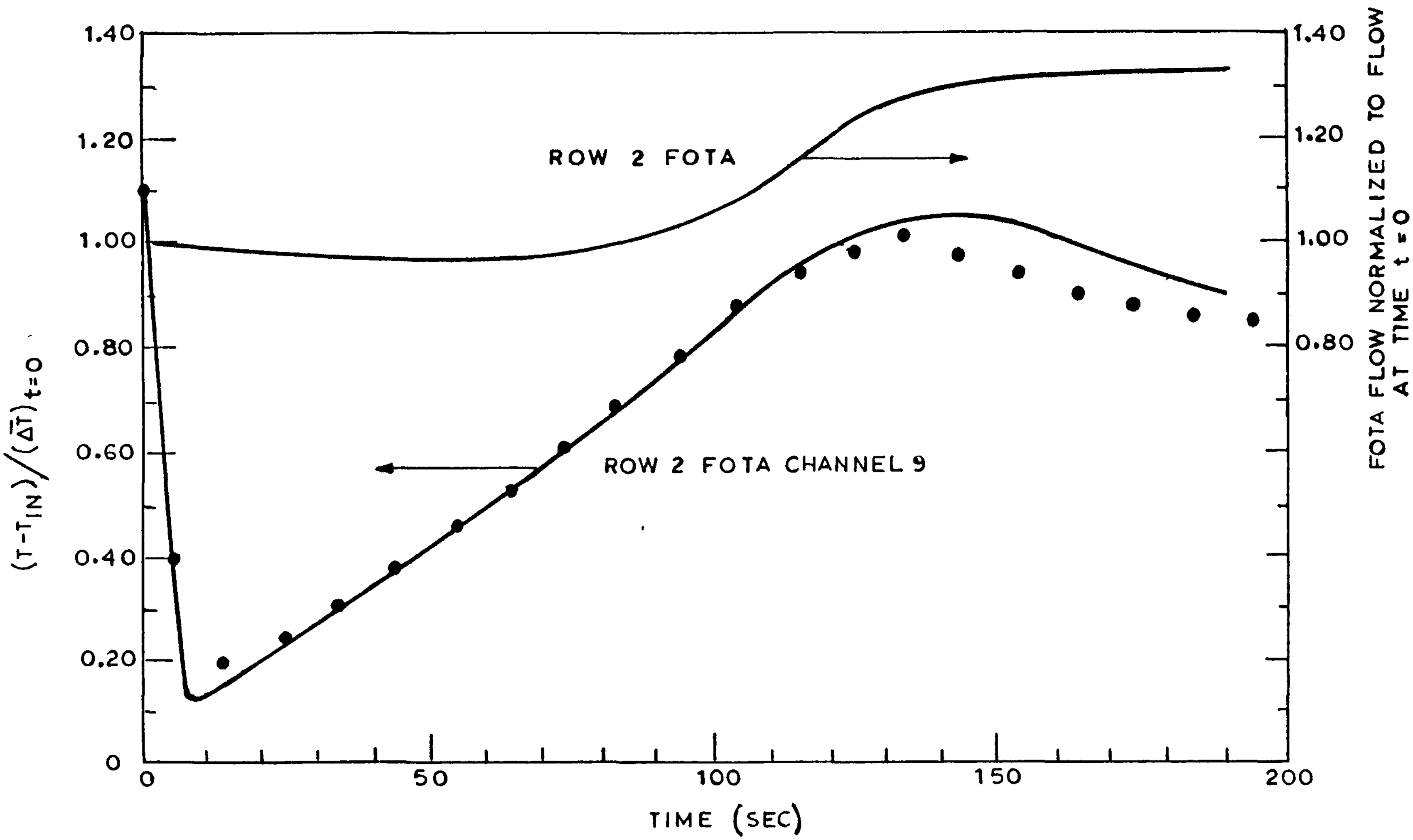

Figure 2. A comparison of the predicted and measured temperature history for Row 2 FOTA. Also shown is the increase in Row 2 FOTA flow with time.