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SIMULATION OF THE RESPONSE OF THE FORT ST. VRAIN HIGH TEMPERATURE GAS COOLED REACTOR SYSTEM TO A POSTULATED ROD WITHDRAWAL ACCIDENT*

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SUMMARY

Transients resulting from the accidental withdrawal of a control rod pair from the Fort St. Vrain (FSV) reactor core from 100% power conditions have been analyzed with the ORTAP[1] nuclear steam supply system simulation. This analysis was done as part of an ongoing effort to obtain an independent assessment of the HTGR system response to several postulated accidents. For a rod pair withdrawal transient it is assumed that the rod pair of maximum worth is withdrawn at the maximum rate from the core at full power conditions. The neutron flux controller and the rod withdrawal prohibit are assumed inoperative. The worst case results when the rods are withdrawn from the fully inserted position at the end of the equilibrium cycle when the delayed neutron fraction is lowest and temperature coefficients of reactivity are least negative. The FSV plant protection system is designed to initiate an automatic scram action as a result of either of the following

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DISCLAIMER

conditions:

a. an increase in core power to 140% rated, and

b. an increase of the measured reheat steam temperature to 75°F above rated.

In order to simulate transients affecting the HTGR, the FSV nuclear steam supply system simulation has been developed at ORNL. Sufficient detail has been retained in the development of component models to allow the simulation of a wide range of normal operational and postulated accident transients.

The operating core is modeled with CORTAP[2] which couples the heat transfer equations for the fuel, the graphite moderator and the helium coolant for a typical channel with the neutron kinetics equations. An alternate core model (ORECA[3]) is used for transients involving decay power and low-flow cases. This model includes calculation of three-dimensional temperature distributions throughout the entire core, accounts for the varying flow distribution among the individual refueling regions, and accommodates flow reversals.

The reheater and once through steam generator are simulated with BLAST[4], which utilizes a multinode, fixed-boundary homogeneous flow model. The time-dependent conservation of energy, mass, and momentum equations for both the helium side and the water/steam side are solved by an implicit integration technique.

Models of the turbine generator plant and the major plant control loops are also included in ORTAP. This is necessary in order to accurately predict the primary system response because of the close coupling between the primary and secondary systems.

For the postulated transient involving accidental withdrawal of a control rod pair, ORTAP results indicate that if the reactor does not scram at the 140% power level but rather at a measured reheat steam temperature 75°F above rated, power levels significantly above 140% are reached. This is due to the action of the plant control systems, the relatively slow response of the reheat steam temperature sensor, and the large heat capacity of the core graphite.

The response of several plant parameters during this transient is shown in Figs. 1-a and 1-b. The action of the major plant control systems has a significant effect during the transient. For example, the main steam temperature controller reduces the circulator speed, and therefore the primary coolant flow, prior to the scram in an attempt to maintain main steam temperature at its setpoint. This reduction in coolant flow results in higher fuel temperatures and delays the trip signal since it reduces the rate of increase of the reheat steam temperature.

The transient fuel temperatures obtained from this analysis are in good agreement with those calculated by General Atomic and reported in ref. 5. These temperature predictions have been used by General Atomic in determining that the amount of fission product release from fuel particles failing due to the temperature transient would be less than design limits.

ORTAP is also being used to analyze a primary system depressurization, a sudden reduction of feedwater flow, and a loss of main loop cooling accident. Plans have also been made to compare ORTAP predictions with

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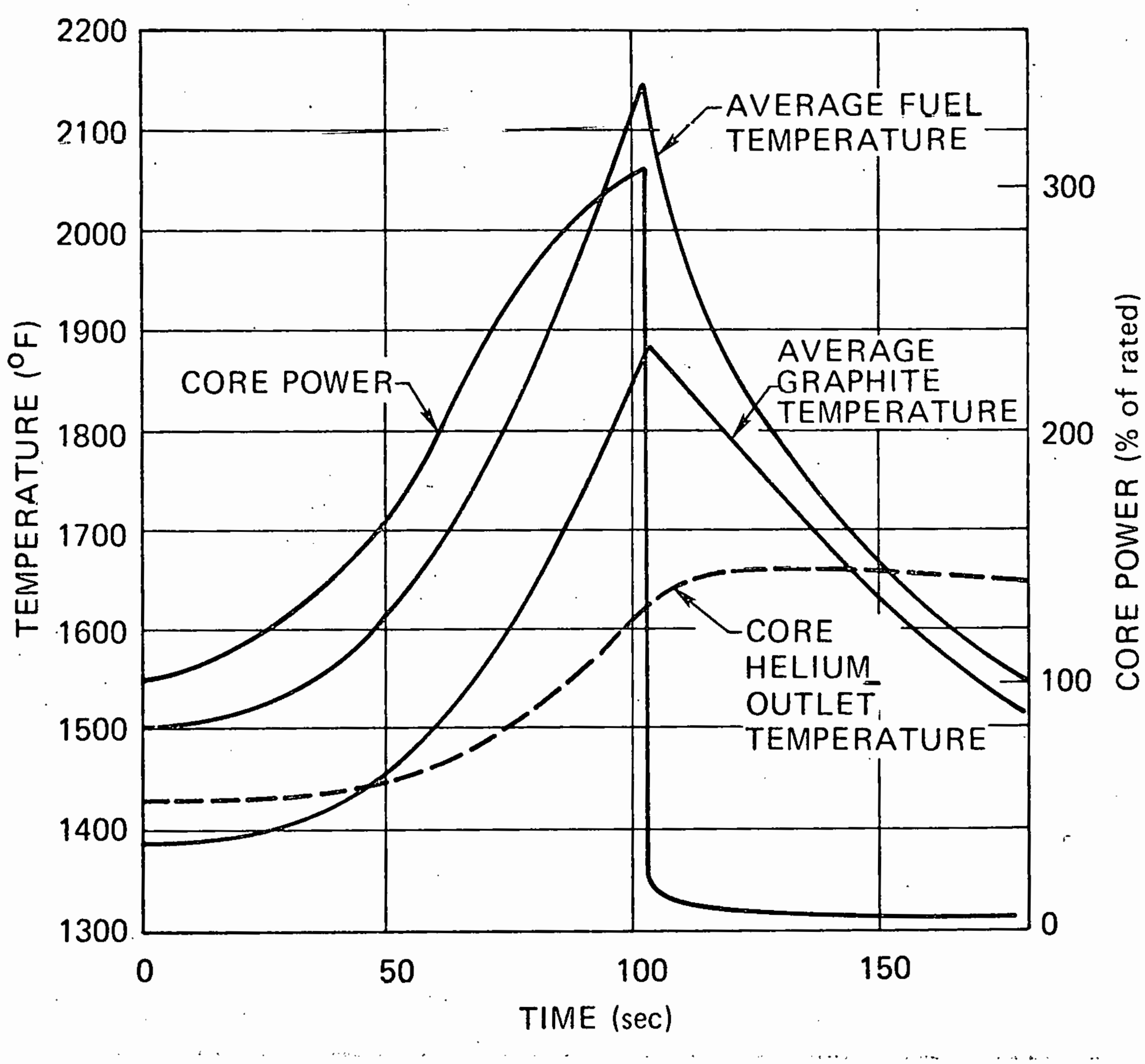

Fig. 1-a. FSV rod-pair-withdrawal transient. Initial rod position, fully inserted; trip signal, 75°F increase in measured reheat steam temperature.

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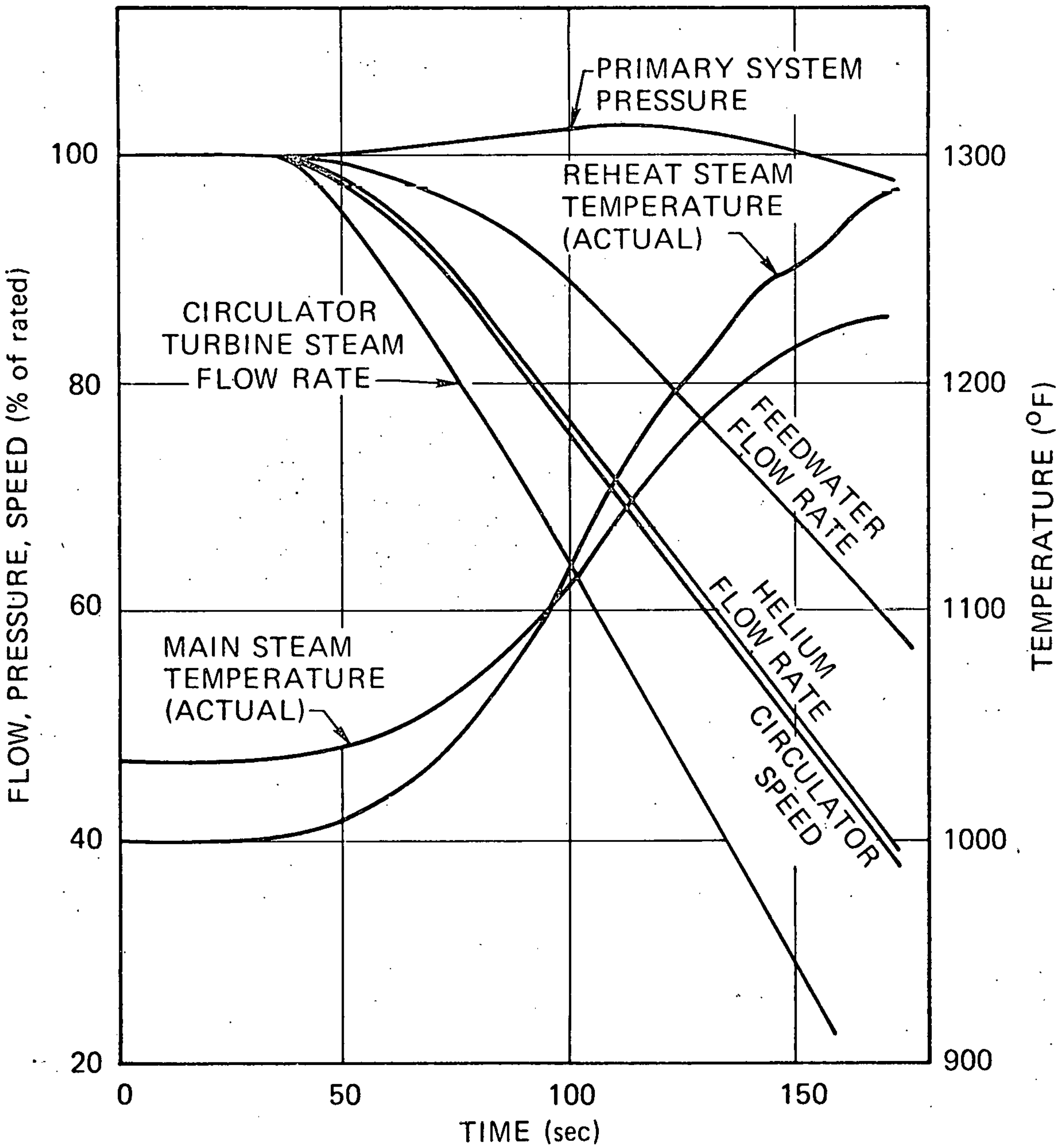

Fig. 1-b. FSV rod-pair-withdrawal transient. Initial rod position, fully inserted; trip signal, 75°F increase in measured reheat steam temperature.

dynamic test results from the FSV and to make appropriate improvements in the code once the comparisons are available.

References

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