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Transient Analysis of the FFTF LOFWOS Test using GAMMA+ Code

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1. Introduction

Fast Flux Test Facility (FFTF) was a 400 MW thermal powered, oxide-fueled, liquid sodium cooled test reactor designed by the Westinghouse Electric Corporation for the U.S. Department of Energy (DOE) (Fig. 1). In 1986, a series of unprotected transients were performed in FFTF as part of the passive safety demonstration program. Among those tests, the experimental results of the loss of flow without scram (LOFWOS) test #13 are currently being analyzed by the participants of an IAEA coordinated research project (CRP) [1].

After the GAMMA+ code was recently updated for the SFR application at KAERI, a simulation for the transient of the FFTF LOFWOS test #13 using GAMMA+ is in progress as an integral effect test validation for SFR [2]. In this proceeding, the FFTF modeling by GAMMA+ is described and the steady-state and transient results are presented.

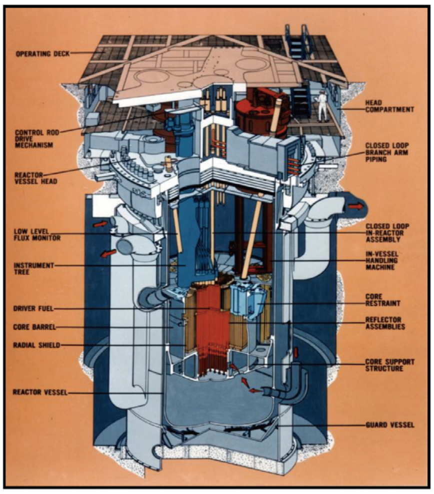

Fig. 1. FFTF reactor vessel overview.

2. FFTF Plant Model

2.1 Overview of FFTF Plant Model

A FFTF plant model was developed according to IAEA CRP benchmark specification for FFTF LOFWOS test # 13 [1]. As neutronics calculations for FFTF core were not performed by KAERI, the results of ANL's neutronics calculations were used for FFTF model. Figs. 2 and 3 show an overview of the FFTF coolant system and the nodalization of thermal-fluidic modeling for FFTF plant. Eleven flow channels which consist of two proximity instrumented open test assemblies (PIOTA), driver fuel assemblies for each flow zone, non-fuel assemblies, and leakage flow were formed for core flow modeling. Three primary loops and three secondary loops were modeled but the calculation model for the secondary pumps and air dump heat exchangers (DHX) were omitted for the sake of simplicity in modeling. The temperatures and flow rates of the secondary cold legs were determined by the boundary conditions at the outlets of DHXs.

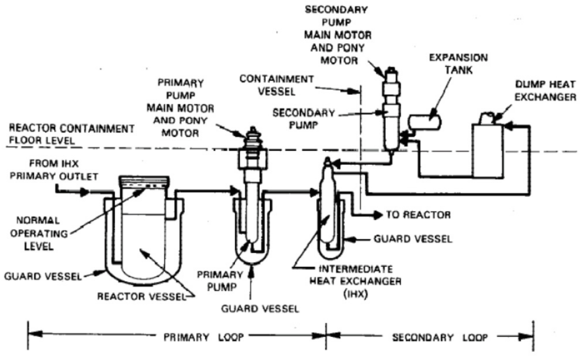

Fig. 2. FFTF coolant system overview.

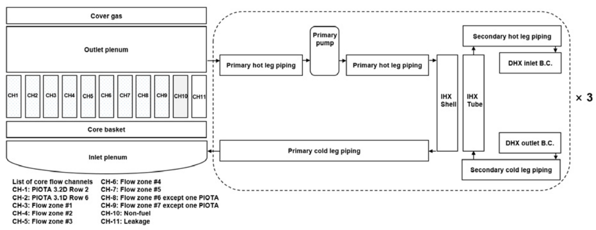

Fig. 3. Nodalization for FFTF plant.

2.2 Core Model

Flow through the core was simplified with a flow channel model. Sixteen flow zones of FFTF core were grouped into eleven flow channels (Fig. 4). The channels no. 1 – 9 have the same node structure because they are related with the driver fuel assemblies (Fig. 5). Each flow channel is represented by an average single pin model, consisting of thermal-fluidic nodes and heat structure nodes. While the thermal-fluidic nodes have only one-dimensional axial volumes, the heat structure nodes have both axial and radial mesh points with cylindrical symmetry. The axial heat conduction within the heat structure nodes was not considered in the simulation. Active core region was modeled with six axial heat structure nodes and seven radial mesh points in each axial heat structure. Gap gas between the fuel and cladding was assumed to be helium gas only. There are various kinds of non-fuel assemblies such as control rods, reflectors, etc. which have different geometries in flow zones 8 – 16. The radial reflector row 8A was chosen as a representative geometry to model the non-fuel channel. Leakage channel was modeled with only thermal-fluidic nodes. Flow resistance coefficients at the inlet junction of each flow channel were adjusted to meet the core flow rate distribution. Axial power distributions of all the assemblies calculated by ANL were merged according to the flow channel grouping. In the case of the non-fuel channel, it was assumed that its axial power distribution follows the average power distribution of all the reflectors. Modified Schad correlation was used for the fuel pin bundle sodium heat transfer.

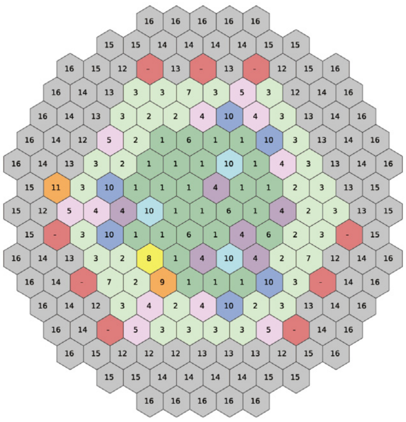

Fig. 4. Flow zones in FFTF core.

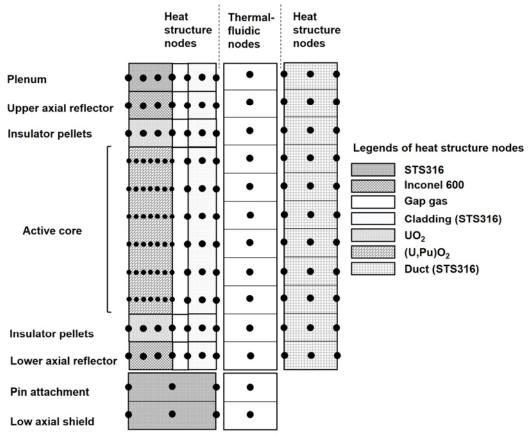

Fig. 5. Nodalization for driver fuel assemblies.

2.3 Reactivity Feedback Model

A point kinetic model was used to simulate the core power of FFTF based on ANL's neutronics calculation data. It should be noted that the reactivity feedback effects are inherently space-dependent. This is not only due to the fact that the temperatures vary spatially, but even for the same temperatures the magnitude of the effect will depend on the location within the reactor. Since the point-kinetic equations suppress any spatial dependence, an appropriately weighted spatial integration of the evaluated local reactivity feedback effects was performed. Various reactivity feedback models related with the fuel Doppler effect, sodium density, axial fuel expansion, radial core expansion, and control rod drive line (CRDL) / reactor vessel (RV) expansion were taken into account in the FFTF plant model. Variation of gas expansion modules (GEM) reactivity due to the change of the sodium level provided in the ANL's neutronics data was used as a transient boundary condition.

2.4 IHX Model

Three intermediate heat exchangers (IHX) for three heat transport loops were modeled. 4.204 meter long active heat transfer regions for the shell-side, tube wall (STS304), and tube-side were discretized with twenty equally spaced axial nodes (Fig. 6). Four radial mesh points were formed in each axial heat structure node for the tube wall. Graber-Rieger and Lyon-Martinelli correlations were used for the shell- and tube-sides, respectively.

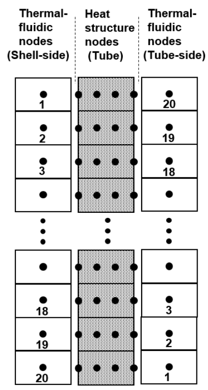

Fig. 6. Nodalization for IHX.

3. Results

3.1 Steady-state results

LOFWOS test #13 started from 50% power and full flow with the pump pony motors turned off. Table I shows the steady-state results at the initial condition obtained from the FFTF modeling. The GAMMA+ calculation results showed a good agreement with the experimental initial condition.

3.2 Transient results

LOFWOS test #13 was initiated at 0 s with the tripping of the three primary pumps simultaneously. The secondary loop sodium pumps remained operational throughout the test. The DHX fan speed was reduced approximately one minute before the test began, resulting in higher DHX sodium outlet temperatures. Accordingly, the transient of the FFTF was simulated from –120 to 900 s to capture the change of the DHX sodium outlet temperatures with the transient boundary condition.

Table I: Initial condition comparison

Parameter	Exp. Test	FFTF model
Reactor power (MW)	199.2	199.2
Core inlet temp (℃)	317.2	317.227
Flow in core channel #1 (kg/s)	24.815	24.818
Flow in core channel #2 (kg/s)	20.5	20.504
Flow in core channel #3 (kg/s)	558.602	558.656
Flow in core channel #4 (kg/s)	289.728	289.734
Flow in core channel #5 (kg/s)	458.062	458.099
Flow in core channel #6 (kg/s)	304.668	304.714
Flow in core channel #7 (kg/s)	99.84	99.8437
Flow in core channel #8 (kg/s)	74.445	74.4513
Flow in core channel #9 (kg/s)	41.0	41.0
Flow in core channel #10 (kg/s)	116.765	116.802
Flow in core channel #11 (kg/s)	213.81	213.773
Total core flow (kg/s)	2202.235	2202.39

Figs 7 – 10 show the representative transient results obtained by the GAMMA+ calculation. In Fig. 7, the total power is the sum of the fission power and decay power. To predict the decay power curve, ANS79 decay power model was used in the GAMMA+. Agreement between the experimental total power and the calculated total power was reasonably good. The calculated core flow rate also had a good agreement with the experimental data (Fig. 8), showing a pump coastdown and transition to natural circulation.

The calculated net reactivity was compared with the net reactivity obtained from the first three hundred seconds of the experimental test results (Fig. 9). The calculated and measured net reactivity showed a quite good agreement. At the beginning of the transient, the large decrease in the net reactivity was mainly caused by the negative GEM reactivity feedback. Another important reactivity feedback was the positive Doppler feedback. As the decrease in the net reactivity led to the drop of the fission power, the decrease in the fuel temperature caused the large positive Doppler reactivity feedback.

The outlet temperatures of the Row 2 PIOTA assembly were evaluated under the various calculation conditions (Fig. 10). At the beginning of the test, the core flow rate decreased faster than the total core power, which resulted in the increase in the power-to-flow ratio and the core outlet temperature. All the calculated results captured the initial increase in the outlet temperature and the subsequent drop by the negative GEM reactivity feedback. The second temperature peak at around 100 s was also well predicted. After the second peak, the outlet temperature decreased gradually during the transient.

The GAMMA+ simulation overestimated the temperature rise at the second peak at the first calculation. It is because the heat transfer between the assemblies was not taken into account. Considering the inter-assembly heat transfer, the second peak temperature dropped largely as the colder surrounding assemblies cooled down the Row 2 PIOTA assembly. Under the actual contact area condition (100%), the simulation result underestimated the outlet temperature to some degree. Moisseytsev and Sumner reported that the single-pin model would overestimate the heat transfer between two assemblies, unless some sort of correction is implemented to consider the effect of average pin locations [3]. They concluded the simulation results with the optimized correction factor of 0.5 applied to the heat transfer coefficient for the single pin model resulted in much improved agreement with the FFTF LOFWOS test. In this work, accordingly the contact area was reduced by a factor of 0.5 instead of reducing the heat transfer coefficient. Consequently, the second peak temperature showed a better agreement with the test result due to the 50% reduced contact area.

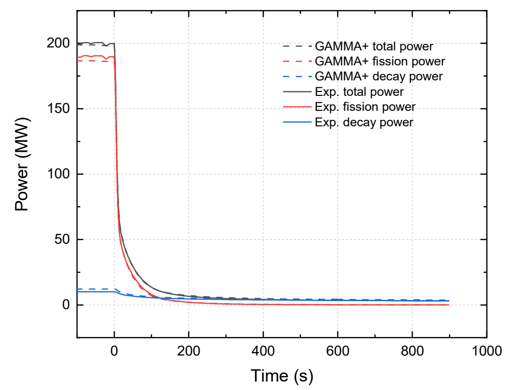

Fig. 7. Transient results: reactor power.

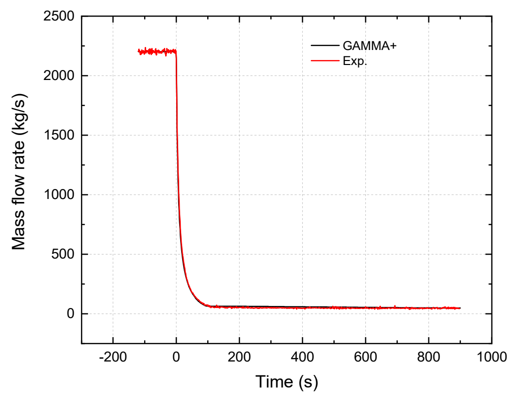

Fig. 8. Transient results: core flow rate.

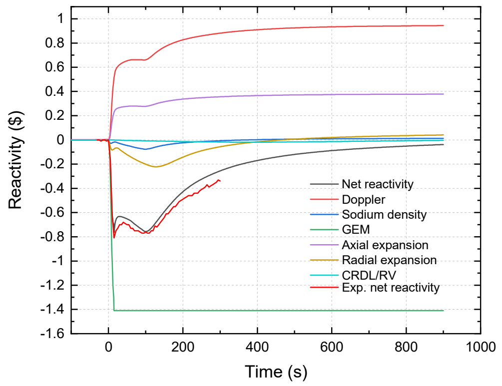

Fig. 9. Transient results: core reactivity.

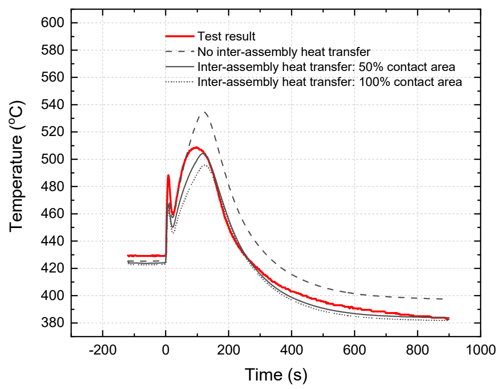

Fig. 10. Transient results: Row 2 PIOTA outlet temperature.

3. Conclusions

The GAMMA+ code was validated by simulating the FFTF LOFWOS test #13 as an integral effect test validation for SFR application. Transient simulation results such as the reactor power, core flow rate, net reactivity, and PIOTA outlet temperature showed a good agreement with the FFTF unprotected test results. Therefore, the GAMMA+ updates turned out to have a satisfactory simulation capability for the SFR transient events. As a further work, a multidimensional thermal-hydraulic analysis will be added in the current FFTF simulation model to capture a thermal stratification in the outlet plenum.

ACKNOWLEDGMENTS

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