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The Preliminary GAMMA Code Thermal Hydraulic Analysis of the HTR-10 Steady State Initial Core

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1. Introduction

This paper describes the preliminary thermal hydraulic analysis of the HTR-10[1] steady state full power initial core to provide a benchmark calculation for the VHTR(Very High-Temperature Gas-Cooled Reactors) system analysis code of GAMMA(GAs Multicomponent Mixture Analysis)[2]. The input data of the GAMMA code is produced for the models of a fluid block, wall block, radiation heat transfer and each components material properties in the HTR-10 reactor. The temperature and flow distributions of the HTR-10 steady state 10 MW_{th} full power initial core are calculated by the GAMMA code with boundary conditions of the total reactor inlet flow rate of 4.32 kg/s, inlet temperature of 250 ^{o}C, inlet pressure of 3 MPa, outlet pressure of 2.992 MPa and a fixed temperature at the RCCS(Reactor Cavity Cooling System) water cooling tube of 50 ^{o}C. The calculation results are compared with the measured solid material temperatures[3] at 22 fixed instrumentation positions in HTR-10.

2. Analysis Method

The fluid system of HTR-10 consists of the helium gas cooling system for cooling the reactor core pumping the helium into the RPV(Reactor Pressure Vessel) by the circulator, and the RCCS for cooling the air cavity between the RPV and the concrete wall. As shown in Figure 1, the calculation of the helium cooling system is modeled up to the horizontal coaxial hot gas duct connected by the RPV and the steam generator. The heat generated in the reactor core is cooled by a conduction and radiation heat transfer of the reactor structures as well as by a convective heat transfer of the fluid system.

In the GAMMA code, the solid wall conduction and fluid contact heat transfer are calculated by setting the wall block region and then providing the input data of the geometrical information, heat generation, material properties and boundary conditions for each region. The radiation heat transfer is only considered on the specific wall block by providing input data of the emissivity and the view factor. It is calculated by using an irradiation/radiosity method, which assumes that the radiation exchange between surfaces is gray and diffuses, and the fluid is a non-participating one.

It is difficult to predict the main flow rate and the bypassing flow rate through the actual flow path due to the occurrence of a clearance among the graphite blocks. An accurate prediction of the flow rate needs a detailed flow resistance of the fluid block although the leakage is not considered. This calculation uses the following assumptions of the core flow rate, bypassing flow and leakages. Total reactor inlet flow rate(TFLOW) is 4.32 kg/s. 87% of TFLOW is used for an effective cooling to the pebble bed zone, and 2% of TFLOW goes through the control rod channels[3]. Up-flow through the fuel discharging tube to the pebble bed zone[3] is expected to be 1% of TFLOW, but it is not considered. It is assumed that 9.2% of TFLOW goes through the gap leaking zone of the carbon bricks at the bottom of the reactor into the hot outlet plenum, and 1.8% of TFLOW is a non-effective cooling flow through the vertical annular flow path at the reactor bottom support zone into the reactor outlet boundary volume. These leakages are used to adjust the reactor outlet temperature to 700 ^{o}C with boundary conditions of the inlet temperature of 250 ^{o}C, inlet pressure of 3 MPa, outlet pressure of 2.992 MPa and a fixed temperature at the RCCS water cooling tube of 50 ^{o}C.

3. Results and Discussions

Figure 2 shows the overall wall temperature distribution of the HTR-10 reactor. The wall temperature distribution in the pebble bed core shows that the minimum temperature of 358 ^{o}C is located at the upper core, a higher temperature zone than 829 ℃ is located at the inner region of a 0.45 m radius at the bottom of the core centre, and the maximum wall temperature is 897℃. The wall temperature linearly decreases at the radially and axially farthest side from the bottom of the core centre. The maximum temperature of RPV is

230 °C, and the maximum values of the fuel average temperature and TRISO centerline temperature are 907 °C and 929 °C, respectively and they are much lower than the fuel temperature limitation of 1230 °C.

The comparison between the GAMMA code predictions and the measured temperature data shows that the calculation results are very close to the measured values in the top and side reflector region, but a great difference appears in the bottom reflector region. Some measured data is abnormally high in the bottom reflector region, and so a confirmation of the data is necessary in the future. Figure 3 shows the prediction error of the GAMMA code for the measured wall temperatures[3] at 22 fixed instrumentation positions in HTR-10. Fifteen of the twenty two data have a relative error less than a ±20% and six of them only show good predictions with less than a ±10% error. For a more accurate calculation of GAMMA code in future, the application of the cone-shaped core model, a consideration of the heat generation and the fluid block in the fuel discharging tube and the application of the improved geometrical model in the bottom reflector zone are suggested.

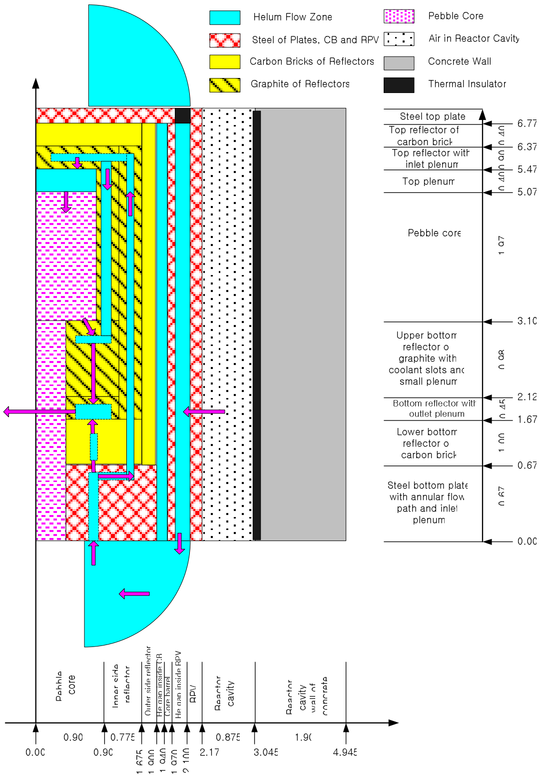

Fig.1 Flow Pass Model of HTR-10 Reactor for GAMMA Code

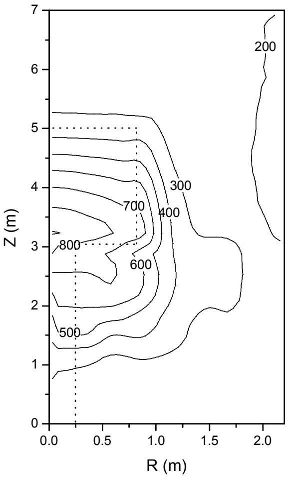

Fig.2 Overall Wall Temperature Distribution

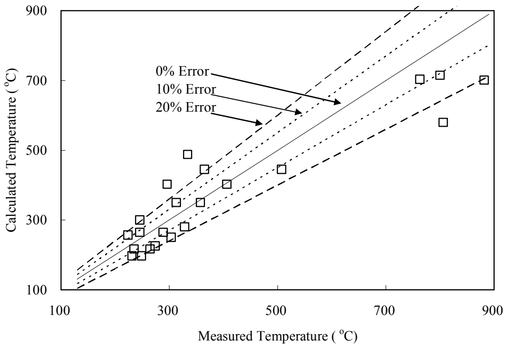

Fig.3 Wall Temperature Prediction Error

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REFERENCES

[1] Zongxin Wu, Dengcai Lini, Daxin Zhong, “The Design Features of the HTR-10”, Nuclear Engineering and Design 218 (2002) 25-32, 2002

[2] Hong Sik Lim, Hee Cheon No, “GAMMA Multidimensional Multicomponent Mixture Analysis to Predict Air Ingress Phenomena in an HTGR”, Nuclear Science and Engineering 152 (2006) 1-11, 2006

[3] IAEA, “Current Status and Future Development of Modular High Temperature Gas Cooled Reactor Technology”, IAEA-TECDOC-1198, 2001