

CFD Analysis of Hot Spot Fuel Temperature in the Control Fuel Block Assembly of a VHTR core

Min-Hwan Kim*, Nam-il Tak, Jae Man Noh

Korea Atomic Energy Research Institute, P.O.Box 105, Yuseong, Daejeon, Rep. of Korea, 305-350

*Corresponding author: mhkim@kaeri.re.kr

1. Introduction

The Very High Temperature Reactor (VHTR) dedicated for efficient hydrogen production requires core outlet temperatures of more than 950°C. As the outlet temperature increases, the thermal margin of the core decreases, which highlights the need for a detailed analysis to reduce its uncertainty. Tak et al. [1] performed CFD analysis for a 1/12 fuel assembly model and compared the result with a simple unit-cell model in order to emphasize the need of a detailed CFD analysis for the prediction of hot spot fuel temperatures. Their CFD model, however, was focused on the standard fuel assembly but not on the control fuel assembly in which a considerable amount of bypass flow is expected to occur through the control rod passages.

In this study, a CFD model for the control fuel block assembly is developed and applied for the hot spot analyses of PMR200 core. [2] Not only the bypass flow but also the cross flow is considered in the analyses.

2. Analysis Model

Configuration and key dimensions of the control fuel block are shown in Fig. 1. A symmetric half block was selected for the analysis. The size of bypass gap between the control fuel block and other blocks around was assumed as 2mm. There was one control rod channel located eccentrically. A total of 186 fuels were inserted into the control fuel block. There was a small gap between the fuel and graphite regions filled with a stagnant coolant. The height of the block is 793 mm including the fuel region in the middle and the graphite region of the graphite plugs at the top and fuel seats at the bottom.

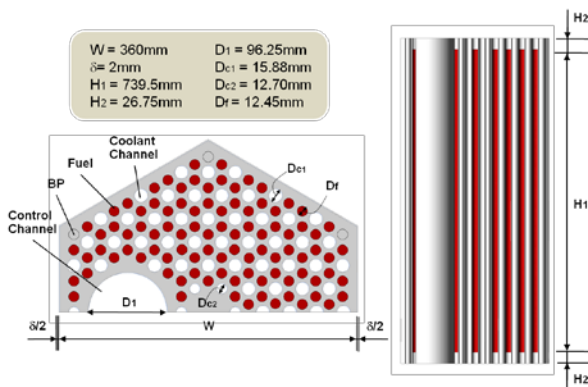


Fig. 1 Configuration of control fuel block

Fig. 2 shows the CFD model for the hot spot analysis

of a control fuel block assembly. PMR200 was selected as a reference reactor, the inlet and outlet temperatures of which are 490°C and 950°C, respectively. The computational domain consisted of the active core region of 6-layered control fuel blocks and two reflector blocks for both the top and bottom sides. Mass flow rate was fixed at the inlet and a constant static pressure condition was applied at the outlet. All the side boundaries were assumed as symmetry conditions. The tiny gap between the fuel and graphite region was modeled by effective conductivity. The k-ε model with scalable wall function was used for the turbulence closure. Key parameters for the boundary conditions are listed in Table 1. The fraction of mass flow at the inlet boundary was selected from the GAMMA+ analysis. [4]

Two kinds of heat sources were applied for the fuel region. One was constant power density and the other was axial power distribution at the hottest fuel assembly obtained from the core neutronics analysis [3] as shown in Fig. 3. By considering a cross gap between the blocks on top of another, 4 cases of the hot spot analysis were determined as shown in Table 2.

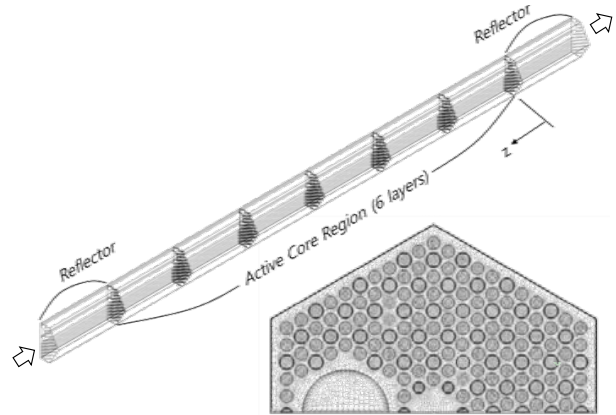


Fig. 2 A CFD model for the 1/2 control fuel assembly

Table 1 Parameters for the boundary conditions

Inlet	Coolant holes	88.7%
	Bypass	3.1%
	CR channel	8.2%
Fuel	Average power density	$2.84 \times 10^7 \text{ W/m}^3$

Table 2. Test matrix for hot spot analyses

Power Profile	No Cross Gap	Cross Gap (1mm)
Mean Power	AVG-CG0	AVG-CG1
Axial profile at EOC	EOC-CG0	EOC-CG1

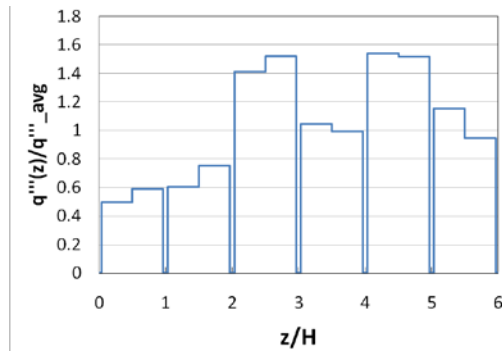


Fig. 3 Axial power profile of the hottest fuel assembly at EOC

3. Results

Temperature distributions are compared for the cases of EOC-CG0 and EOC-CG1 in Fig. 4. The fuel temperature is higher in the EOC-CG1 case considering the cross gap of 1mm. Due to the cross flow, the coolant temperature in the control channel is higher than the EOC-CG0 case.

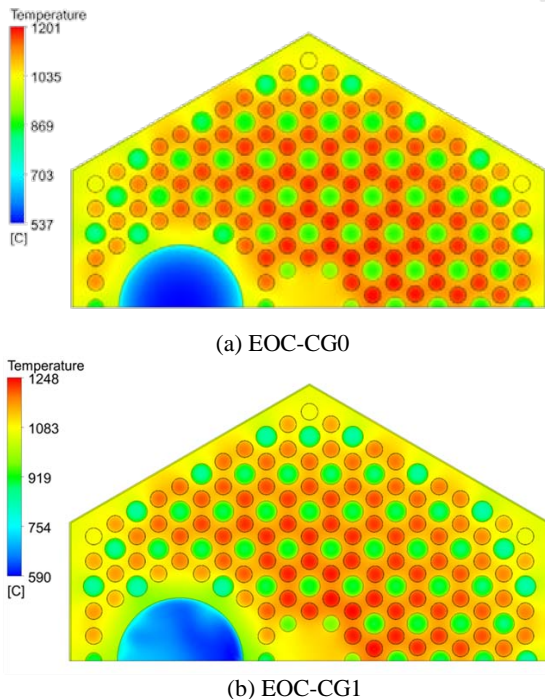


Fig. 4. Temperature distribution at T_{max} plane

Mass flow distribution at each block layer is illustrated in Fig. 5. The coolant flow decreases as it approaches the outlet. The change of the bypass flow is so minute that most of the cross flow occurs between the coolant hole to the control channel.

Maximum fuel temperature for each case is provided in Table 3. The influence of power profile on the hot spot fuel temperature is very small. The cross flow, however, results in an increase of the maximum fuel temperature by about 50°C. The temperature for the AVG-CG1 case is above the normal operation limit of 1250°C, which implies that the integrity of fuel cannot be assured. Although obtained from a simple

assumption of opening condition at the outlet of coolant channel, the results indicate that the cross flow should be carefully considered in the hot spot analysis.

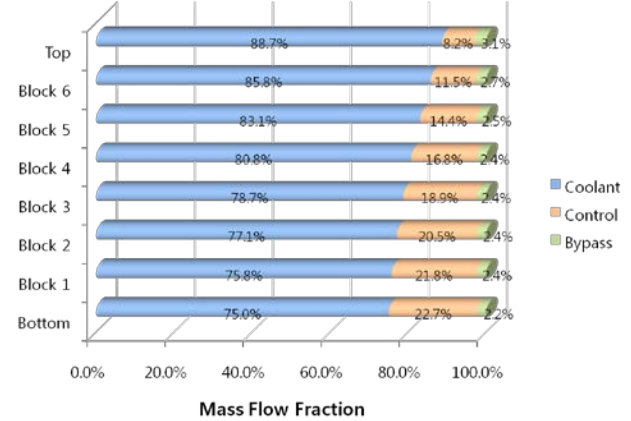


Fig. 5 Mass flow distribution for the EOC-CG1 case

Table 3. Maximum fuel temperatures

Cases	Maximum Fuel T. (°C)
AVG-CG0	1206
EOC-CG0	1201
AVG-CG1	1265
EOC-CG1	1248

4. Conclusions

CFD analyses were performed to assess the hot spot fuel temperatures in the control fuel block assembly for the PMR200. The detailed CFD model made it possible to investigate the main factor affecting the hot spot fuel temperatures. The results indicated that the influence of cross flow was so large that the integrity of fuel performance would be deteriorated. In order for a more detailed assessment, a configuration of the outlet of the control channel should be identified and considered in the future analysis.

Acknowledgements

This work was supported by Nuclear R&D Program of the National Research Foundation of Korea (NRF) grant funded by the Korean government (MEST). (Grant code: 2009-006258)

REFERENCES

- [1] N.I. Tak et al., "Numerical investigation of a heat transfer within the prismatic fuel assembly of a very high temperature reactor," *Annals of Nuclear Energy*, Vol. 25, 1892-1899, 2008.
- [2] J. Chang et al., "A Study of a Nuclear Hydrogen Production Demonstration Plant," *Nuclear Engineering and Technology*, Vol. 39, No.2, April 2007.
- [3] C.K. Jo, H.S. Lim and J.M. Noh, "Preconceptual Designs of the 200MWth Prism and Pebble-bed Type VHTR Cores," *International Conference on the Physics of Reactors*, Interlaken, Switzerland, Sep. 14-19
- [4] J.S. Chun, Thermo-fluid Analysis of PMR 200MWth using GMMA+ code, KAERI Calculation Note, NHDD-RD-CA-09-004, 2009.