Bypass Flow and Hot Spot Analysis for PMR200 Block-Core Design with Core Restraint Mechanism

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1. Introduction

The accurate prediction of local hot spot during normal operation is important to ensure core thermal margin in a very high temperature gas-cooled reactor because of production of its high temperature output. The active cooling of the reactor core determining local hot spot is strongly affected by core bypass flows through the inter-column gaps between graphite blocks and the cross gaps between two stacked fuel blocks. The bypass gap sizes vary during core life cycle by the thermal expansion at the elevated temperature and the shrinkage/swelling by fast neutron irradiation.

This study is to investigate the impacts of the variation of bypass gaps during core life cycle as well as core restraint mechanism on the amount of bypass flow and thus maximum fuel temperature. The core thermo-fluid analysis is performed using the GAMMA+ code for the PMR200 block-core design [1]. For the analysis not only are some modeling features, developed for solid conduction and bypass flow, are implemented into the GAMMA+ code but also non-uniform bypass gap distribution taken from a tool calculating the thermal expansion and the shrinkage/swell of graphite during core life cycle under the design options with and without core restraint mechanism is used.

2. Modeling Features

The implementation of modeling features to get the detailed temperature and flow distributions and the method to determine the bypass gap size distribution are briefly described below:

2.1 Code Improvements

- The heat conduction model to calculate the temperature profile in TRISO particles dispersed within a fuel matrix is implemented. The implicit coupling method is used to consider heat exchange between a fuel matrix and coated TRISO particles.
- The multi-D heat conduction model is further improved to consider submeshes within hexagonal fuel and reflector blocks. The effect of cross diffusion by non-orthogonal geometry also is considered.
- The friction and heat transfer correlations for a rectangular flow channel are implemented and used for lateral and cross bypass flow paths.
- The correlation for cross flow loss factor taken from the Kaburaki's experiment [2] is added.

$$K = 1.555\left(0.78/\delta \mathrm{Re} + 1.70\right) \text{ for parallel gap}$$

Where Re is Reynolds number based on hydraulic diameter and δ is the largest gap opening size (m).

2.2 Core Analysis Model

The PMR200 block-core design has the thermal power of 200 MW, the system pressure of 70 bar and the coolant inlet and exit temperatures of 490°C and 950°C, respectively. Fig. 1 shows the 1/6 core analysis model consisting of the nodal schemes for the coolant and bypass flow channels, the lateral and cross flow networks defined at the middle point of graphite block and the interface between graphite blocks, and the fuel and solid sub-meshings.

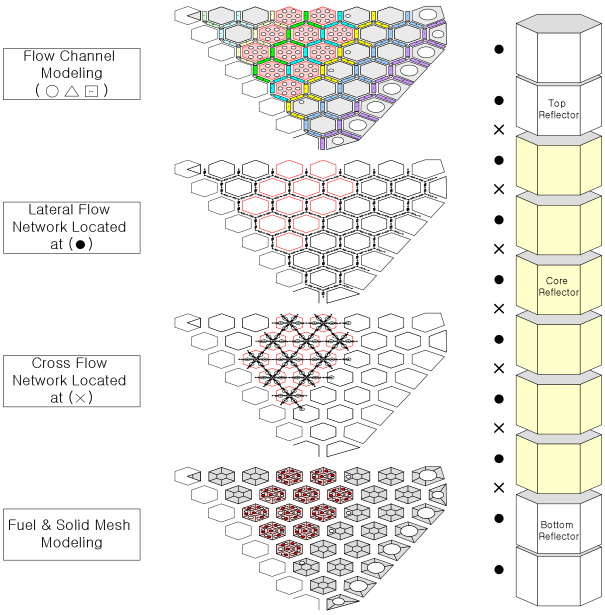

Figure 1. GAMMA+ 1/6-Core Analysis Model

In the analysis the parallel cross-gap only is considered for the cross flows because it is worse than the wedge-shaped cross-gap with about three times higher loss factor. The core power distribution is taken from the end of core (EOC) condition with fuel-block shuffling scheme.

2.3 Bypass Gap Distribution

The non-uniform bypass gap distribution is taken from a tool developed to calculate local bypass gap distribution in a prismatic core [3]. Fig. 2 shows the bypass gap distributions for the design options with and without core restraint mechanism. The bypass gap size for the unconstraint case is determined by three parts:

(1) initial gap by block-installing tolerance, (2) thermal expansion of core support plate and (3) net effect of graphite expansion/swell and shrinkage. In the constraint case the gap changes by the first and second effects disappear by core restraint. Therefore there are no gap openings at the central and side reflector regions as well as the top and bottom ends of the active core.

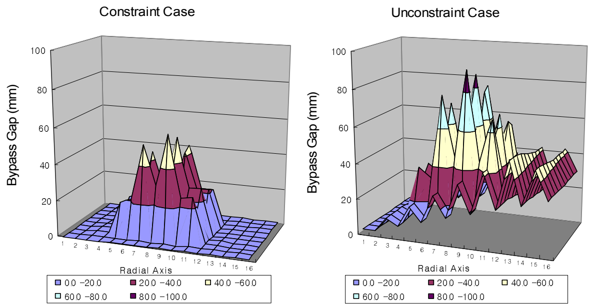

Figure 2. Bypass Gap Size Distributions for Cy7

3. Results and Discussions

The calculation results for core bypass flow and hot spot analysis are summarized at Table 1. By core restraint, the total bypass gap reduces to less than half and hence the core bypass flow reduces about 20%. Therefore core restraint acts a great role of reducing maximum kernel temperature by not less than 100^{o}C.

Table 1 Analysis Cases and Calculated Major Parameters

Core Life Cycle	Core-average Total Bypass Gap (mm)	Helium Coolant Exit Temp. (^{o}C)	Max. Kernel Temp. in Core (^{o}C)	Core-average Gap/Total Bypass Flow (%)
Cy07: Constraint	17	954	1201	6/12
Cy07: Unconstraint	52	971	1330	26/31
Cy14: Constraint	27	957	1222	10/15
Cy14: Unconstraint	63	974	1372	30/35

Note: Cy14 corresponds to about 20 years of reactor operation which is the end of the lifetime of replaceable graphite.

The bypass flow distribution shown at Fig. 4 resembles the bypass gap distribution (Fig. 2), but the amount of bypass flow increases toward the core bottom since the bottom ends of CR/RSC holes are plugged.

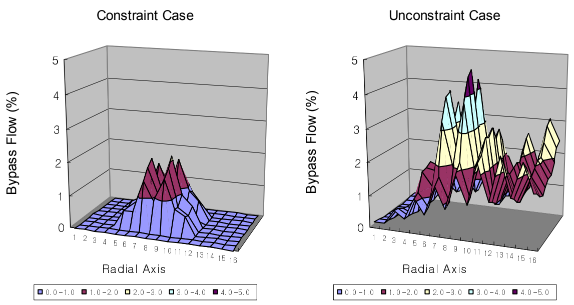

Figure 4. Bypass Flow Distributions for Cy7

The maximum kernel temperature profile shown at Fig. 5 is proportional to the power-to-flow ratio. That means the profile is strongly affected by bypass flow distribution. Therefore the uniform bypass gap case is different from other cases. Particularly the fuel blocks with CR/RSC holes show the highest kernel temperatures because the coolant flow passing through the centre region is reduced by the bypass flow through CR/RSC hole.

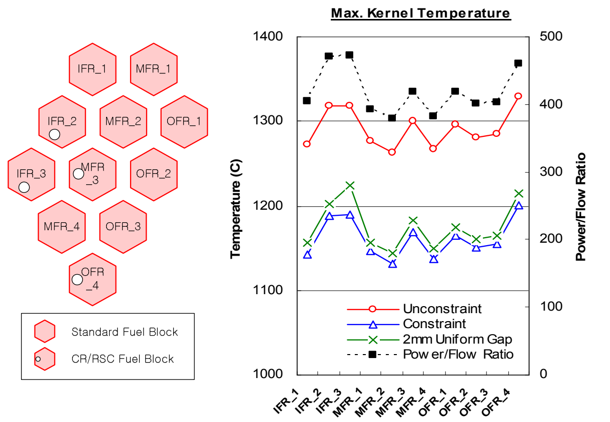

Figure 5. Maximum Kernel Temperature Profiles

4. Conclusions

We investigate the impacts of the bypass gap variation during core life cycle as well as core restraint mechanism on core bypass flow and thus maximum fuel temperature.

- It is clearly observed that core restraint mechanism reduces about 20% of core bypass flow and thus not less than 100^{o}C of maximum UO_2 kernel temperature.

- The maximum kernel temperature profile is proportional to the power-to-flow ratio and hence is strongly affected by bypass flow distribution.

- Therefore the hot spot is located at the center of CR/RSC fuel block instead of standard fuel block.

ACKNOWLEDGEMENTS

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REFERENCES

[1] J.H. CHANG, et al., "A Study of A Nuclear Hydrogen Production Demonstration Plant," Nuclear Engineering and technology, Vol.39, no.2, pp.111-122, April 2007.

[2] H. KABURAKI, T. TAKIZUKA, "Cross Flow Characteristics of HTGR Fuel Blocks," Nuclear Engineering and Design, Vol.120, pp.425-434, 1990.

[3] M.H. KIM, et al., "Preliminary Estimation of Local Bypass Flow Gap Sizes for A Prismatic VHTR Core," Transactions of the Korea Nuclear Society Spring Meeting, Jeju, Korea, May 21-22, 2009.