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MCNP6 CALCULATION OF NEUTRON FLUX MAP IN THE HTTR DURING NORMAL OPERATION

Detailed neutron flux distribution is important to understand the neutronic behavior during operation as well as to precise the core optimization and safety analysis of a reactor. In the literature, no calculations have been performed to show the detailed neutron flux map for the high temperature engineering test reactor (HTTR) because of the limitation of the old neutronic codes and the low performance of the computing system. HTTR is a prismatic type reactor, helium gas-cooled, and graphite-moderated, providing 30 MWth power and up to 950 °C outlet temperature. The present work deals with MCNP6 Monte-Carlo calculation to determine the detailed neutron flux map in the HTTR during normal operation. At first, the calculation of neutron flux at several positions in the reactor was validated by comparing the corresponding reaction rate between the calculation and measurement. After that detailed neutron flux with the small cells of 1cm×1cm×10cm was obtained for the entire reactor core using the fmesh tally of MCNP6 code.

Keywords: HTTR, HTGR, MCNP6, neutron flux, fmesh.

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MCNP6 әдісімен қалыпты жұмыс кезінде ЖТСР-ғы нейтрондар ағынының картасын есептеу

Нейтрондар ағынының егжей-тегжейлі таралуы жұмыс кезінде нейтрондардың әрекетін түсіну үшін, сондай-ақ ядроны дәл оңтайландыру және реактор қауіпсіздігін талдау үшін маңызды. Нейтронды - физикалық ескі кодтарының шектеулеріне және есептеу жүйесінің нашар өнімділігіне байланысты жоғары температуралық сынақ реакторы (ЖТСР) үшін егжей-тегжейлі нейтрон ағынының картасын көрсету үшін осыған дейін әдебиеттерде ешқандай есептеулер жүргізілмеген. ЖТСР призматикалық типті реактор, гелий газымен салқындатылған және графитпен модерацияланған, 30 МВт қуат пен 950 °С шығыс температурасын қамтамасыз етеді. Бұл жұмыс қалыпты жұмыс кезінде ЖТСР - дағы нейтрон ағынының егжей - тегжейлі картасын анықтау үшін Монте-Карло MCNP6 (Monte Carlo N - Particle Transport Code 6) есебіне арналған. Біріншіден, реактордың бірнеше нүктелеріндегі нейтрондар ағынының есебі есептеу мен өлшеу арасындағы сәйкес реакция жылдамдығын салыстыру арқылы расталды. Осыдан кейін MCNP 6 fmesh tally коды арқылы реактордың барлық өзегі үшін 1 см × 1 см × 10 см шағын ұяшықтары бар егжей - тегжейлі нейтрон ағыны алынды.

Түйін сөздер: ЖТСР, ЖТГР, MCNP6, нейтрон ағыны, fmesh.

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Расчет карты потока нейтронов в ВТТР при нормальной эксплуатации методом MCNP6

Подробное распределение потока нейтронов важно для понимания поведения нейтронов во время работы, а также для точной оптимизации активной зоны и анализа безопасности реактора. В литературе не проводились расчеты для отображения подробной карты потока нейтронов для высокотемпературного тестового реактора (ВТТР) из-за ограничений старых нейтронно - физических кодов и низкой производительности вычислительной системы. ВТТР представляет собой реактор призматического типа с гелиевым газовым охлаждением и графитовым замедлителем, обеспечивающий мощность 30 МВтт и температуру на выходе до 950 °С. Настоящая работа посвящена расчету методом Монте-Карло MCNP6 (Monte Carlo N - Particle Transport Code 6) для определения детальной карты потока нейтронов в ВТТР при нормальной работе. Сначала расчет потока нейтронов в нескольких точках реактора был подтвержден путем сравнения соответствующей скорости реакции между расчетом и измерением. После этого нами был получен подробный поток нейтронов с мелкими ячейками 1 см × 1 см × 10см для всей активной зоны реактора с использованием fmesh tally кода MCNP6.

Ключевые слова: ВТТР, ВТГР, MCNP6, нейтронный поток, fmesh.

Introduction

High temperature engineering test reactor (HTTR) is the first Japanese high-temperature gas-cooled reactor (HTGR) located in the Oarai Research and Development Institute of Japan Atomic Energy Agency (JAEA) [1-6]. HTTR is a prismatic type reactor, helium gas-cooled, and graphite-moderated, providing 30 MWth power and up to 950ºC outlet temperature.

Many neutronic calculations for the HTTR have been performed to show various neutronic properties of the reactor such as neutron flux, multiplication factor, excess reactivity, temperature reactivity coefficient, shutdown margin, etc. [7-14]. However, the detailed neutron flux map was not be obtained in previous studies because of the old nuclear codes as well as the low computing performance. The detailed neutron flux distribution would help to calculate detailed power distribution and detailed depletion of the fuel and therefore precise the core optimization and safety analysis. Therefore, the purpose of this study is to provide the neutron flux map of the HTTR during normal operation using the Monte-Carlo MCNP6 code [15]. In order to obtain the detailed neutron flux map, the fmesh tally divided the core into about 16 million rectangular cells with dimensions of 1cm×1cm×10cm in x, y, and z directions, respectively.

Methodology

The overview of HTTR is shown in Figure 1. The HTTR is a prismatic type of high temperature gas cooled reactor (HTGR) with thermal power of 30MW and outlet temperature of 850/950oC. There are 150 hexagonal fuel blocks in the core region stacking in 30 fuel columns. Each fuel block contains 31 or 33 fuel rods and there are 14 fuel compacts in each fuel rod. The fuel compact comprises approximately 13,000 coated fuel particles in an annular graphite matrix [16]. More detailed fuel design of the HTTR can be seen in Figure 2.

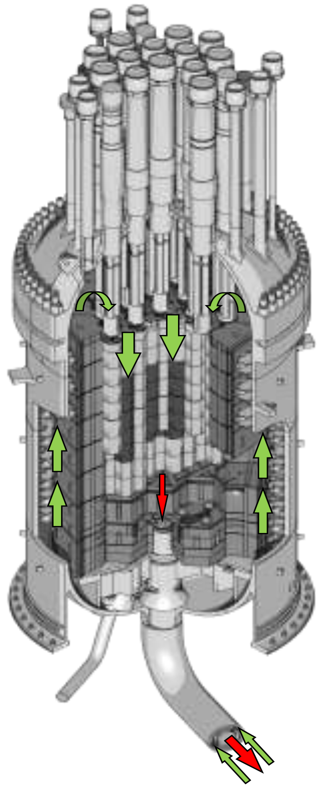

Figure 1 – Overview of HTTR

The MCNP6 model for the HTTR has been developed as much detail as possible. This HTTR model with MCNP6 has been validated in previous studies [12-14]. The neutronic calculation was carried out with ENDF/B.VII-1 nuclear library [17]. The number of neutrons per cycle and the number of active cycles were 20,000 and 1000 (excluding 50 skip cycles), respectively.

It is of interest that MCNP6 code provides the fmesh tally, which allows users to achieve the tallies in a very fine mesh [18-20]. There are two types of mesh that could be used with fmesh tally including Cartesian (XYZ) and Cylindrical (RZT) mesh. The geometries of the XYZ and RZT meshes are shown in Figure 3. This study chooses the XYZ meshes because of the convenience during post-plotting.

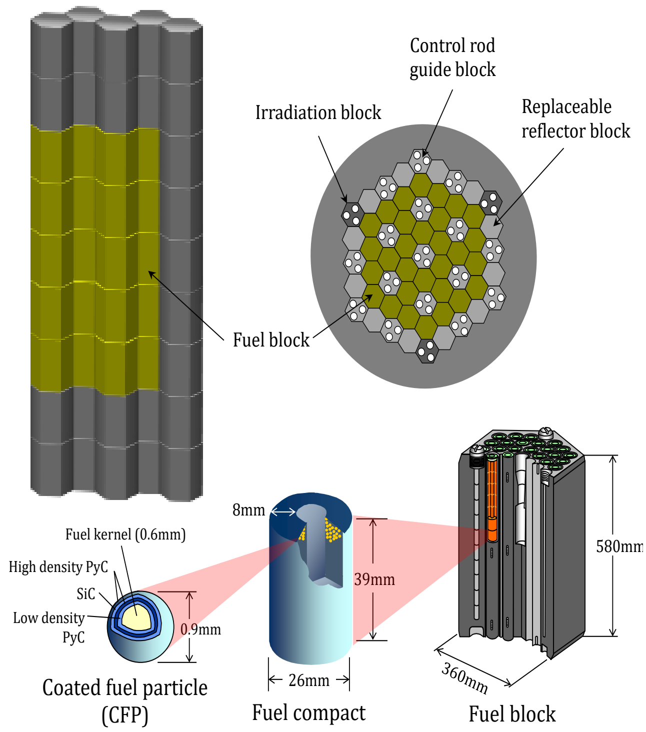

Figure 2 – HTTR fuel design

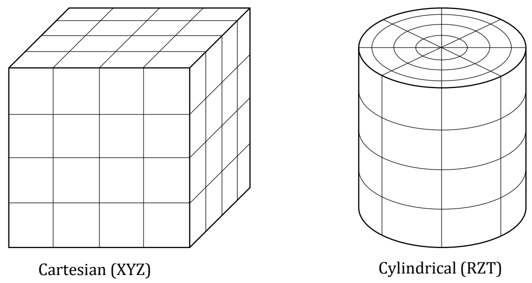

Figure 3 – There are 2 types of fmesh in MCNP6

Results

In the first step, the neutron flux calculation was verified by comparing the calculated neutron flux with the measurement neutron flux at zero power operation. The reactor operated at almost zero power in the experiment so that it is difficult to determine the absolute value of neutron flux. Therefore, the equivalent reaction rate was used for the comparison. Figure 4 shows that the calculated reaction rate appears to be in good agreement with the measured value.

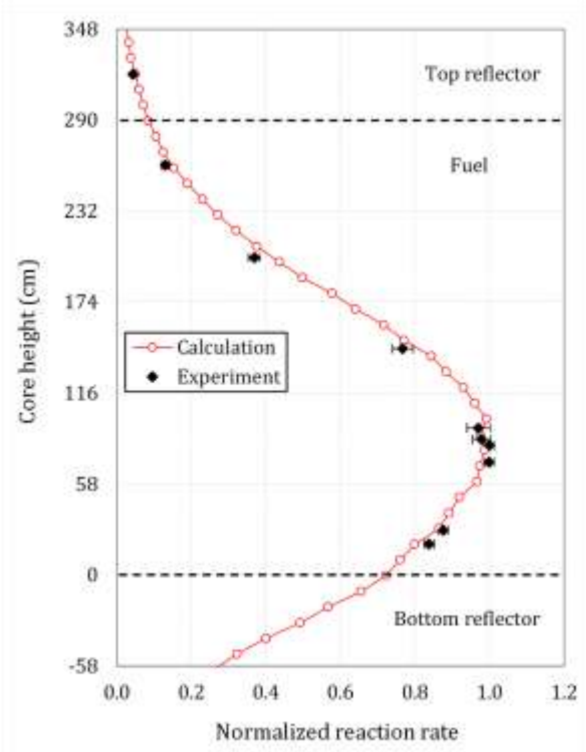

Figure 4 – Verification of reaction rate at zero power operation

After verifying the neutron flux calculation method, the detailed neutron flux map was calculated using the fmesh tally. With 20×106 neutron history, the neutron flux could be achieved with a relative error of about 1%. The thermal and fast neutron fluxes in the radial direction are shown in Figures 5 and 6, respectively. It can be seen in Figure 5 that the highest thermal neutron flux appears at the center control rod block, followed by at the replaceable blocks at side reflector. The depth gaps of thermal neutron flux caused by control rods and burnable poison rods are also clearly seen in Figure 5.

According to Fig. 6, the fast neutron flux is only significant in the fuel region. It decreases remarkably at the side reflector region because of the good moderation of graphite moderator. In contrast to thermal neutrons, which are mostly appeared at non-fuel blocks, the fast neutrons are mainly distributed in the fuel blocks because the fission neutrons are fast neutrons.

Figures 7 and 8 show the thermal and fast neutron flux in the axial direction at center cross-section of the core. It can be seen in Figure 7 that the thermal flux deceases significantly at the top reflector region because of the existence of control rods here. In Figure 8, the fast flux is also dominant in the fuel region as same as in the radial direction.

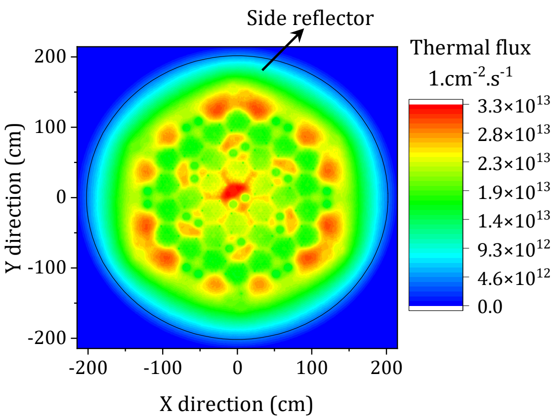

Figure 5 – Radial thermal neutron flux at the top fuel layer

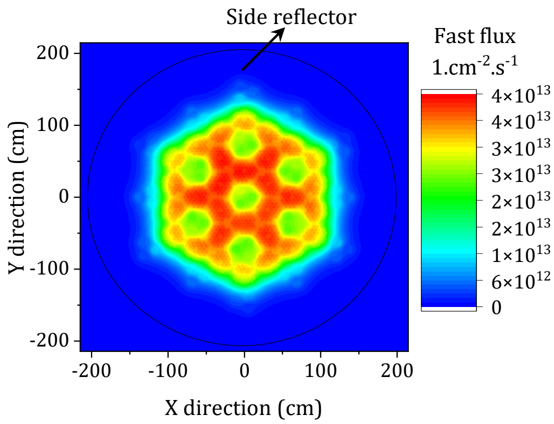

Figure 6 – Radial fast neutron flux at the top fuel layer

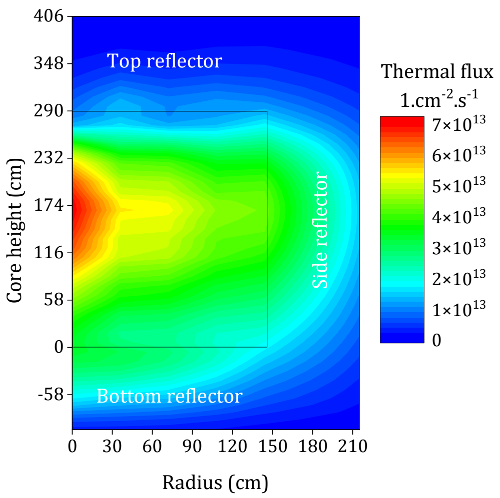

Figure 7 – Axial thermal neutron flux at the core center

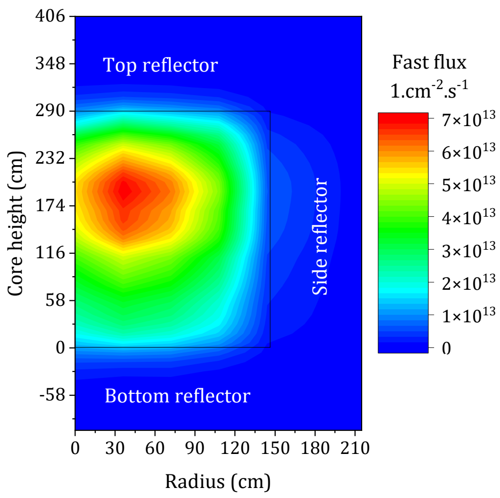

Figure 8 – Axial fast neutron flux at the core center

Conclusions

This study calculated and constructed the detailed thermal/fast neutron flux map for the HTTR using MCNP6 code. The flux could be obtained in detail even for every fuel rod. This result is useful for understanding the neutronic behavior as well as for future core optimization and safety analysis of the HTTR.

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