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**ARTICLE**

# Conceptual Neutronic Design of a Research Pin-in-Block Type HTGR

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The core neutronic design was performed for a research High Temperature Gas-cooled Reactor (HTGR) as a part of the research cooperation with Polish National Centre for Nuclear Research (NCBJ). This research HTGR is designed to supply high temperature steam for industrial use and has 30 MW thermal power. The design policy of the research HTGR is to construct it without any additional demonstration tests, i.e., using only proven technology in Japanese HTGR development, with the aim of deploying it in the 2030s. In the core neutronic design of the research HTGR, we have challenged to reduce the reactor core height, the number of fuel enrichments and the number of control rods compared with the High Temperature Engineering Test Reactor (HTTR) in Japan for the purpose of cost reduction. We have confirmed that the design feasibility parameters related to reactor shutdown margin, temperature coefficient, xenon spatial oscillation, maximum fuel temperature and kernel migration distance satisfy the prescribed limits. As a result, the core neutronic design has been completed successfully by shortening the reactor core height by one fuel block, reducing the number of fuel enrichments from twelve to three, and reducing the number of control rod from sixteen pairs to seven pairs.

***KEYWORDS: HTGR, HTTR, design, advance reactor, neutronic calculation, research reactor***

## I. Introduction

A High Temperature Gas-cooled Reactor (HTGR) has been receiving particular attention as one of the Generation IV nuclear reactor systems in the world, because of its excellence in safety, economic efficiency and nuclear proliferation resistance, and applicability of nuclear power as a heat source.[1)] For example, Poland has an HTGR plan in which a construction of a research HTGR is started on 2020s.[2,3)] Poland aims to reduce $CO_2$ emissions by replacing steam boilers using coal fuel for industrial heat supply with HTGRs in the future. Meanwhile, in Japan, Japan Atomic Energy Agency (JAEA) has the High Temperature Engineering Test Reactor (HTTR),[4)] which is the prismatic type HTGR with 30 MW thermal power. The HTTR succeeded in the 50-day continuous operation test with the full power and reactor outlet coolant temperature of 950ºC in 2010,[5)] and the test results demonstrated that the HTTR can supply stable high temperature heat. Making the most of these experiences, JAEA has been performing versatile design studies.

JAEA signed the “Implementing Arrangement for R&D Cooperation in the HTGR Technology Field” with the Polish National Centre for Nuclear Research (NCBJ) on September 20, 2019. Furthermore, the implementing arrangement was amended on November 22, 2022, adding an item of the R&D Cooperation related to the basic design work for a research HTGR. Under the implementing arrangement, JAEA has been performing design study cooperation for HTGRs simulation, fuel/material research, and safety research on nuclear heat applications. As a part of the research cooperation, JAEA has been performing a core neutronic design for the research HTGR. This paper describes a core neutronic design of a research HTGR with 30 MW thermal power upgraded from the HTTR design as one of the reference designs for the Polish research HTGR proposed by JAEA.

## II. Outline of the Reactor Core

**Table 1** shows the major specifications of the research HTGR whose core neutronic design is performed in this study. Cost reduction is one of the most important issues in core design work. In general, the cost that can be reduced in core design is mainly classified into engineering cost and construction cost. Engineering cost pertains to design changes, including the design work itself, as well as the additional costs for demonstration tests or building new equipment, facility, etc. required to newly produce the components used in the design. The design uses the same components as the HTTR design in order to reduce engineering costs. Furthermore, construction costs are reduced by upgrading the proven design technology in the HTTR. The following are considered as upgrades from the HTTR design:

- Reduction in the reactor core size,
- Reduction in the number of kinds of fuel enrichments and
- Reduction in number of the control rod.

It should be noted that these improvements are achievable because this design has lower reactor outlet temperature than that of the HTTR design, and the HTTR design has too much shutdown margin (approx. 40%△k/k under one-rod stacking).

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**Table 1** Major specifications

| | Values | |
|---|---|---|
| | This study | HTTR (Reference) |
| Power (MWt) | 30 | 30 |
| Outlet coolant temp. (ºC) | 750 | 850/950 |
| Inlet coolant temp. (ºC) | 325 | 395 |
| Core diameter (m) | 2.3 | 2.3 |
| Core height (m) | 2.3 | 2.9 |
| Power density (MW/m$^3$) | 3.1 | 2.5 |
| Burn-up period (days) | 730 | 660 |
| Uranium enrichment (wt%) | 5.5-9.2 | 3.3-9.9 |
| Number of fuel enrichment | 3 | 12 |
| Number of pairs of CRs | 7 | 16 |

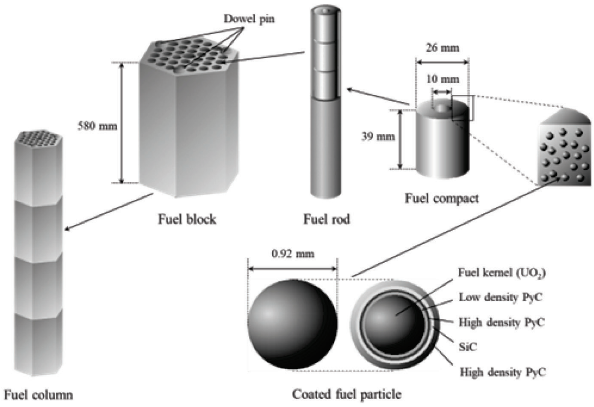


**Fig. 2** Fuel structure of the research HTGR

**Figure 1** shows the horizontal and vertical views of the reactor core. The core is constructed by stacking three kinds of hexagonal blocks, which are fuel blocks, control rod (CR) guide blocks and replaceable reflector blocks. They are surrounded by permanent reflectors made of graphite. All these hexagonal blocks are made of high-purity graphite (IG-110), and are the same size: 360 mm in across flats and 580 mm in height. The fuel region of the core is configured by 30 fuel blocks, which are composed by stacking four fuel blocks. The fuel region is divided into four in the radial direction according to the difference in the distance from the center of the reactor core. In the core neutronic design, the core size is reduced by one fuel block height by increasing the power density from 2.5 MW/m$^3$ of HTTR to 3.1 MW/m$^3$.

**Figure 2** shows the fuel structure of the research HTGR. The fuel specifications are the same as the HTTR, except for enrichment. The fuel compact is in the form of a hollow cylinder with an inner diameter of 10 mm, an outer diameter of 26 mm, and a height of 39 mm. The packing fraction of coated fuel particles, which have the function of preventing the release of fission products from the fuels to the coolant, is approximately 30%, and each compact contains 13,000 coated fuel particles. The fuel rod consists of 14 fuel compacts enclosed in a graphite sleeve made of IG-110. The fuel blocks in the fuel region number 1 and number 2 include 33 fuel rods. The fuel blocks in the fuel region number 3 and number 4 include 31 fuel rods. In the HTTR design, 12 kinds of fuel enrichment are used to optimize the power distribution in the reactor core so that the fuel temperature during normal operation is lower than the limit of 1,495ºC. In the research HTGR design, the power distribution is optimized using 3 kinds of fuel enrichment as shown in **Table 2**. In the axial direction, higher enrichment fuel is arranged in the upper fuel region so that the power density is higher in the upper fuel region where the coolant temperature is low.[6)] In the radial direction, higher enrichment fuel is arranged in the outer fuel region so that the power density is uniform.

A rod-type burnable poison (BP) made of $B_4C$/C composite is used to keep the optimized power distribution shape that is similar to HTTR during the burn-up period. The rod-type BPs are inserted into the vertical holes under the dowel pins for every fuel blocks. In the core neutronic design, B-nat. concentration of the rod type BP is 2.5wt%, which is uniform in the reactor core. By changing the diameter, the excess reactivity is adjusted during the burn-up period. The alignment of BPs is shown in **Table 3**.

The CRs are inserted into the core from the upper region of the reactor core to the bottom of the fuel region through vertical holes placed in the CR guide blocks. The HTTR has 16 pairs of 32 CRs consisting of one pair of central CR (C-CR), 6 pairs in the first ring (R1-CRs), 6 pairs in the second ring (R2-CRs), and 3 pairs in the third ring (R3-CRs). The

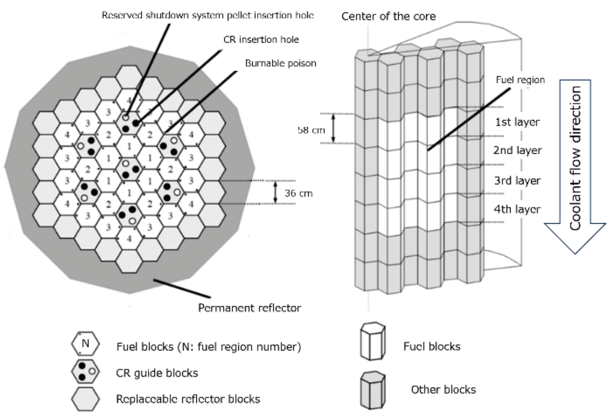


**Fig. 1** Horizontal and vertical cross-sectional views of the research HTGR

**Table 2** Alignment of fuel enrichment (wt%)

| Layer | Fuel region number | | | |
|---|---|---|---|---|
| | 1 | 2 | 3 | 4 |
| 1st | 7.2 | 7.2 | 9.2 | 9.2 |
| 2nd | 7.2 | 7.2 | 9.2 | 9.2 |
| 3rd | 5.5 | 5.5 | 7.2 | 7.2 |
| 4th | 5.5 | 5.5 | 7.2 | 7.2 |

**Table 3** Rod-type BP diameter (mm)

| Layer | Fuel region number | | | |
|---|---|---|---|---|
| | 1 | 2 | 3 | 4 |
| 1st | 14 | 14 | 14 | 14 |
| 2nd | 19 | 19 | 19 | 19 |
| 3rd | 14 | 14 | 14 | 14 |
| 4th | 14 | 14 | 14 | 14 |

layout of the CRs is illustrated in **Fig. 3**. The HTTR has a large shutdown margin of more than 40%Δk/k,[7] because it was the first HTGR in Japan and was designed with large conservativeness. In the core neutronic design of the research HTGR, 9 pairs of control rods corresponding to R2-CR and R3-CR of the HTTR core are reduced and the number of CRs loaded into the research HTGR core is 7 pairs.

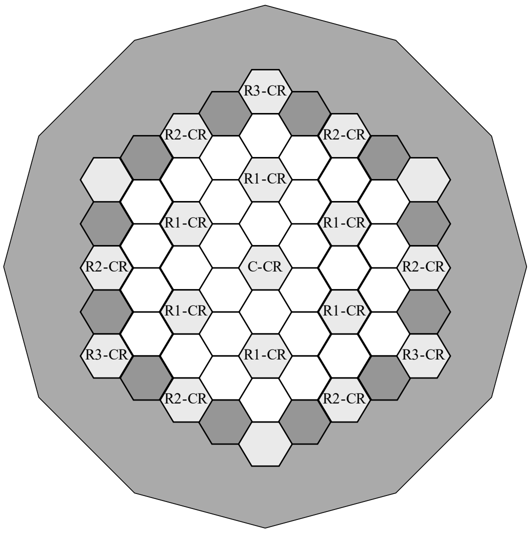


**Fig. 3** Layout of control rods of HTTR

## III. Design

### 1. Design Requirement

The feasibility of the core neutronic design is confirmed by showing that the calculation results of the reactor shutdown margin, temperature coefficient of reactivity, xenon spatial oscillation, maximum fuel temperature under normal operation conditions and fuel kernel migration distance satisfy the design requirements.

The shutdown margin must be more than 1.0%Δk/k when the temperature is 300 K, even if the pair of CRs with the largest reactivity value is completely withdrawn and cannot be inserted into the reactor core.

The temperature coefficient of reactivity must be a negative value during the burn-up period to provide negative reactivity feedback characteristics. The temperature coefficient of reactivity is calculated by following equation,

$$TC = \frac{\Delta\rho}{\Delta T}. \tag{1}$$

Where TC is the temperature coefficient of reactivity. The $\Delta\rho$ represents the change in reactivity when the temperature of the reactor core increases uniformly by $\Delta T$.

Spatial oscillations in the power density distribution can be caused by the feedback effects between the power distribution and xenon production rate distribution, and depend on neutron movability and physical core dimensions. This so called "xenon spatial oscillation," can be suppressed under the following conditions.[4]

- Core dimension is small.
- Power density and the flux level are low.
- Power coefficient is negative.

In this core design, the xenon spatial oscillation must not be occurred.

The maximum fuel temperature must not exceed 1600ºC to prevent fuel failure under any anticipated operational occurrence conditions.[4] To satisfy this requirement, the fuel temperature limit for normal operation condition is specified as 1,493ºC that does not exceed the fuel integrity limitation temperature of 1,600ºC during accidents. This is calculated in the same way as for the HTTR.

The fuel kernel migration is a phenomenon in which the fuel kernel penetrates into the coating layer along the direction of the temperature gradient under irradiation conditions. The fuel kernel migration distance is calculated using the fuel kernel migration rate expressed as

$$KMR = 2 \times 10^{-6} \exp\left(-\frac{1.48\times10^{4}}{T}\right)\frac{1}{T^{2}}\frac{dT}{dr}. \tag{2}$$

where KMR is the fuel kernel migration rate (m/s), T is the temperature of fuel compact and r is radial location of fuel compact.[4] The fuel kernel migration distance at end of cycle (EOC) must be lower than 55 μm taking into consideration of manufacturing tolerance of coating layer thickness. The limitation value can be calculated by the following equations with deterministic approach:

$$\mathrm{KMD_{limit}} = \mathrm{th}_{1+2} - \sqrt{(3\sigma_1)^2 + (3\sigma_2)^2}\,. \tag{3}$$

Where the $\mathrm{th}_{1+2}$ is the sum of the thicknesses of the buffer layer and the inner PyC layer in a coated fuel particle, the $\sigma_1$ is the tolerance of a thickness in the buffer layer, and the $\sigma_2$ is the tolerance of a thickness in the inner PyC layer.[8] The buffer layer is made of low density PyC as well as the inner PyC layer.

### 2. Methodology

**Figure 4** shows the procedure of the core neutronic design.[9] The power distribution is optimized using three kinds of fuel enrichment. The optimized power distribution shape is maintained during the burn-up period by keeping the excess reactivity small and keeping the CR insertion depth into the reactor core shallow by means of the rod-type BPs. The excess reactivity, shutdown margin, temperature coefficient of reactivity and power distribution are calculated by performing the whole core burn-up calculations with cell burn-up calculation code SRAC/PIJ[10] and whole core calculation code SRAC/COREBN[10] based on diffusion theory and the Japanese Evaluated Nuclear Data Library version 4.0 JENDL-4.0.[11] In this calculation, it should be noted that there is a slight difference from the original version of the SRAC/PIJ code regarding the treatment of double heterogeneity.[12]

A three-dimensional triangular mesh is used for the whole core burn-up calculations, as shown in **Fig. 5**. The coarse group effective cross section set for each mesh is generated with SRAC/PIJ based on the collision probability method.

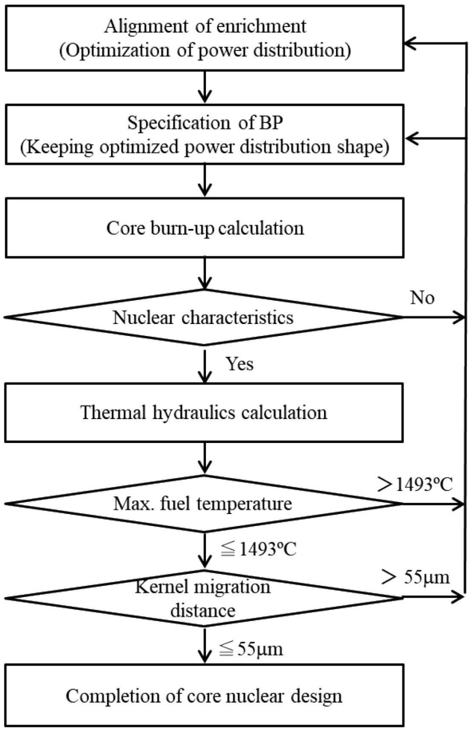


**Fig. 4** Procedure of the core neutronic design

Here, as with the HTTR design, the six energy group structure is adopted for the whole core burn-up calculations, which has seven energy partitions: 10 MeV, 8.6517 MeV, 0.96112 keV, 2.3824 eV, 0.68256 eV, 0.10963 eV, 0.01 meV. Each hexagonal block is divided into six triangular meshes horizontally and fourteen meshes vertically. The whole core burn-up calculations are performed by adjusting the CR position at a critical stage in each burn-up step.

According to the conservative evaluation method proposed by Randall, et al.,[4,13)] the threshold for xenon spatial oscillations is given by the ratio of active core height (H) or active core diameter (D) to the neutron migration length (M) and the thermal neutron flux for a cylindrical reactor under the condition where the power coefficient is zero. The neutron migration length is evaluated for an average fuel block cell by evaluating the diffusion coefficient and macroscopic absorption cross section of one group neutron energy.

The fuel compact temperature and its gradient in the radial direction are calculated with the pin-in-block type HTGR fuel temperature calculation code FTCC[14,15)] with the calculation results of the power distributions in the reactor core for each burn-up step and the design flow rate distributions in **Table 4**. The temperature calculation is based on the multi-cylindrical model which consists of a central void or central helium gas, fuel compacts, a helium gap, graphite sleeves, helium gas in a main fuel cooling channel, and graphite blocks. The heating quantity distribution in fuel compacts and the neutron fluence distributions that affects thermal conductivity for graphite structures are input from neutronic calculation results. Here, the homogenized model is used for the fuel compacts, and the thermal conductivity for the fuel compacts is conservatively set to 0.1256 W/(cm･K). The thermal radiation heat transfer between the fuel compacts and the graphite sleeves and that between the graphite sleeves and the graphite blocks are considered. The boundary condition is conservative set to adiabatic. The details on the temperature distribution calculation are available in Ref. 14,15). To obtain maximum fuel temperature, we should not only obtain nominal temperature, by summing these temperature rises, but also take any uncertainties into account by using hot spot factor as follows,

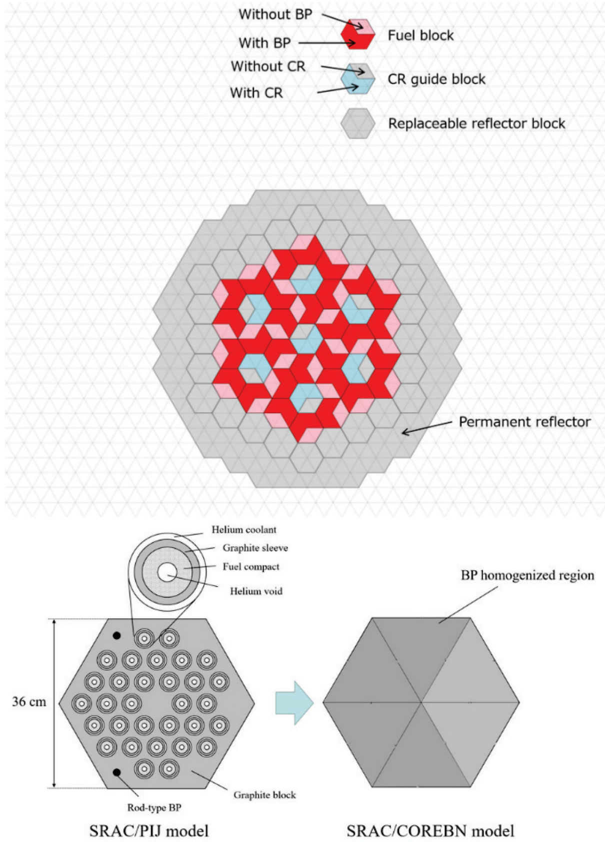


**Fig. 5** Calculation geometry of whole core calculation (Upper) and comparison of SRAC/PIJ model and SRAC/COREBN model for a fuel block (Lower)

**Table 4** Design flow rate distribution (kg/s)

| Layer | Fuel region number | | | |
|---|---|---|---|---|
| | 1 | 2 | 3 | 4 |
| 1st | 0.3835 | 0.3924 | 0.3680 | 0.3746 |
| 2nd | 0.3813 | 0.3924 | 0.3658 | 0.3724 |
| 3rd | 0.3835 | 0.3924 | 0.3680 | 0.3746 |
| 4th | 0.3857 | 0.3946 | 0.3746 | 0.3813 |

(Note) The coolant pressure is set to 4.0 MPa. The fuel region number 1 & 2 have 33 coolant holes, and the fuel region number 3 & 4 have 31 coolant holes.

$$T_f = T_{in} + \sum_{i=1}^{5}(F_i \cdot \Delta T_i^N)\,. \tag{4}$$

Where the $T_{in}$ is core inlet coolant temperature, the $F_i$ is overall hot spot factor for the $i$-th component, the $\Delta T_i^N$ is nominal temperature rise in the $i$-th component, the suffix $i$ indicate each component, coolant, film, graphite sleeve, gap, and fuel compact by from 1 to 5, respectively. The hot spot factor can treat systematic uncertainties and statistical uncertainties. The fuel kernel migration distance is calculated by integrating KMR, which can be obtained from the temperature distribution in each fuel compact as an output from the calculation with the FTCC, over the burn-up period of 730days.

## IV. Results

The shutdown margin at 0EFPD is 2.7%Δk/k under one-rod stacking and room temperature conditions and satisfies the design requirement of more than 1.0%Δk/k.

**Figure 6** shows the calculation result of the temperature coefficient of reactivity, which satisfies the design requirement of being negative value during the burn-up period.

The threshold of the thermal neutron flux for oscillations is shown in **Fig. 7** for a cylindrical reactor with a zero-power coefficient. The neutron migration length of the research reactor is approximately 27 cm. The neutron migration length is evaluated with the SRAC/PIJ for an average fuel block cell by evaluating the diffusion coefficient and macroscopic absorption cross section of one group neutron energy. The ratios of H/M and D/M are 8.5 and 8.6, respectively. Since these do not exceed the threshold value for the maximum thermal neutron flux of $2\times10^{13}$ n/cm$^2$/sec (E<0.1 eV), a xenon spatial oscillation does not occur in the research reactor.

**Figure 8** shows the calculation result of the critical control rod positions corresponding to the CR positions where the each calculated effective multiplication factor is closest to unity. The control rods move in the upper region of the 1st layer throughout the burn-up period. This contributes to reducing the distortion of the power distribution brought by CR insertion. **Figures 9** to **11** show the calculation results of the power distribution and fuel temperature at the beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC), respectively. The ideal curve shown in Figs. 9 to 10 represents the ideal power distribution that minimizes fuel kernel migration distance [6]. In these power distributions, the dips that appear in each layer represent areas where no fuel rods are loaded at the joints between the fuel blocks. The shapes of the power distribution are close to the ideal curve. The maximum fuel temperature during the burn-up period is 1,451°C, which occurs at the 4th layer of fuel region 4 at 600EFPD and satisfies the design requirement.

The fuel kernel migration distance at the EOC is 52.4 µm, which occurs at the 3rd layer of fuel region number 3 and satisfies the design requirement.

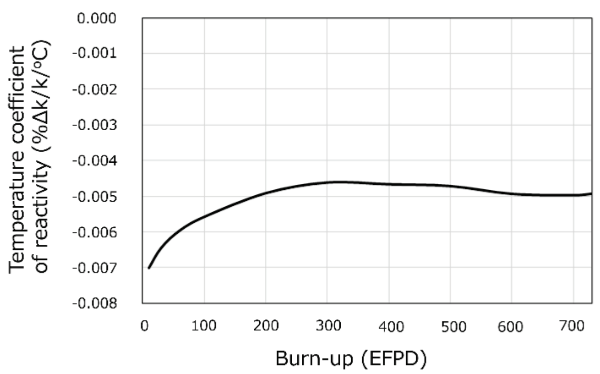


**Fig. 6** Calculation result of temperature coefficient of reactivity

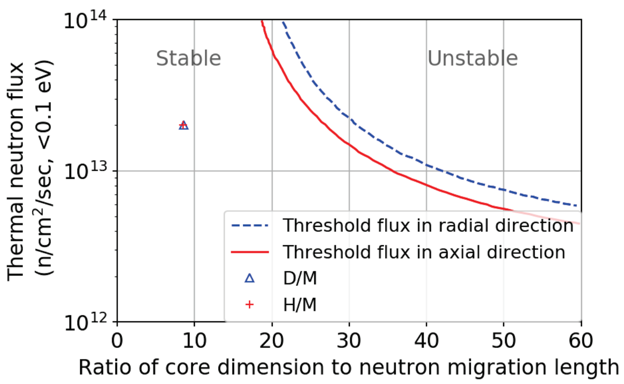


**Fig. 7** Threshold value of thermal neutron flux for instabilities caused by xenon spatial oscillation in cylindrical cores

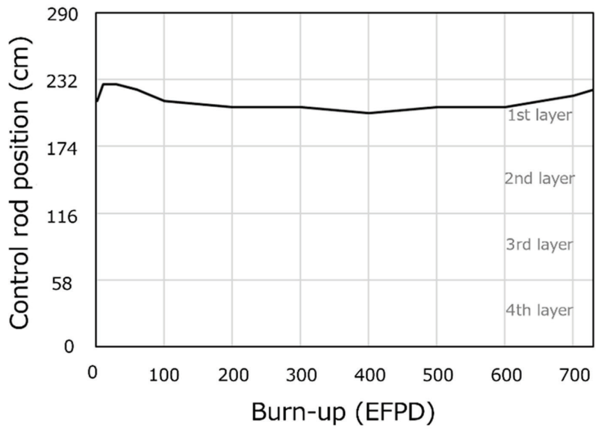


**Fig. 8** Calculation result of critical control rod position

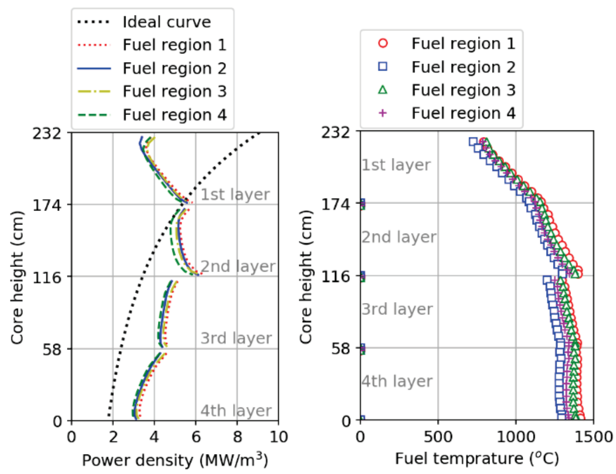


**Fig. 9** Calculation results of power density distribution (left) and fuel temperature (right) at BOC (10 EFPD)

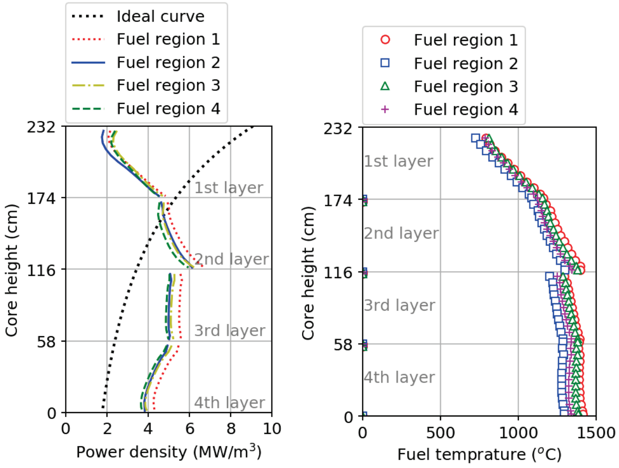


**Fig. 10** Calculation results of power density distribution (left) and fuel temperature (right) at MOC (400 EFPD)

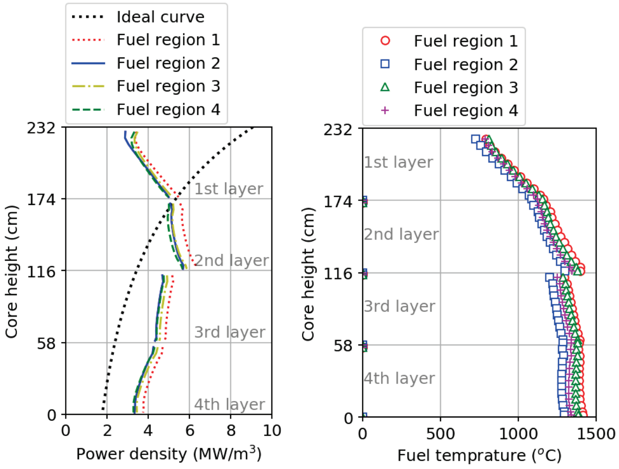


**Fig. 11** Calculation results of power density distribution (left) and fuel temperature (right) at EOC (730 EFPD)

## IV. Conclusions

The core neutronic design is performed by upgrading from the HTTR. Engineering costs are expected to be reduced since the main structures, systems and components used in this design are based on the HTTR design. Furthermore, construction costs are also expected to be reduced owing to reduction in the reactor core size, the number of kinds of fuel enrichments, and number of the CRs. The core size is reduced by one fuel block height by increasing the power density from 2.5 MW/m$^3$ of the HTTR design to 3.1 MW/m$^3$. The power distribution is optimized only with 3 kinds of fuel enrichment: 5.5wt%, 7.2wt% and 9.2wt%. The number of CRs can be reduced from 16 pairs of the HTTR design to 7 pairs, while maintaining the shutdown margin. These reductions are practicable without additional R&Ds.

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