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Burnup computation for HTR-10 using MCNPX as the function of radius and fuel enrichment

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Abstract. The HTR-10 is a gas-cooled high-temperature reactor with spherical fuel and moderator called pebble bed and operated on 10 MW thermal power. Until now, a lot of research and development on the HTR-10 in terms of neutronic computational modeling has been carried out, one of which is burnup analysis. Fuel depletion or burnup analysis is an analysis related to fuel control and reactor output power. This analysis is needed because changes in fuel composition will affect the neutronic parameter values of a reactor. The 10 MW HTR fuel depletion study has been conducted by modeling the reactor using MCNPX (Monte Carlo N-Particle e-Xtended) to analyze the effect of radius and enrichment of reactor fuel to the left value and the burnup rate. The study is conducted by modeling a 10 MW HTR with three sorts of fuel kernel radiuses using MCNPX. First, 250 μm radius, second, 275 μm radius, third, 300 μm radius and all radiuses are variated with 10-15% fuel enrichment. TRISO particles are dispersed using simple cubic (SC) lattice in the fuel zone. The fuel balls are allocated in the reactor core along with moderator balls using a body-centered cubic. To perform all the calculations, MCNPX utilizes continuous energy data library ENDF/B-VI. From the calculation result using MCNPX can be concluded that the kernels with the radius of 275 μm and 300 μm at 14% and 13% fuel enrichment respectively were more able to maintain a critical condition within a year of operation than the kernels a 250 μm radius and the fuel with 250 μm radius has the highest burnup rate that is 52,07 GWd/MTU.

1. Introduction

High-Temperature Gas-Cooled Reactor (HTGR) is expected to compete with the other generation IV nuclear reactors for its notorious core design concepts, prismatic block, and pebble-bed, and also for its advantages that cannot be obtained by previous designs. The first construction and operation of the 10 MW High-Temperature Gas-Cooled Reactor-Test module (HTR-10) were conducted by the Institute of Nuclear and New Energy Technology (INET) of Tsinghua University, China [1]. Its design represents modular HTGR (High-Temperature Gas-Cooled Reactor) design features, so it aims for verifying and demonstrating the feature of the modular HTGR and also for developing the utilizations of high temperature from the nuclear process [2].

HTGR is a co-generation nuclear power plant which means that the produced heat can be used for other systems and applications besides producing electricity such as producing hydrogen, coal liquefaction, and water desalination. HTR-10 is primarily characterized by inherent safety features [3]. Some other advantages of the reactor include the refueling process that can be done online without stopping the electricity production process which can greatly reduce the outage times, increase burnup in spent fuels, and more efficient utilization of natural uranium. This reactor also has spherical fuel which can move down, so the fuel-moderator ratio decreases, increasing the reactivity of the reactor [4].

As a research power reactor, HTGR is the best reactor preference. Therefore in 2014, the National Nuclear Energy Agency of Indonesia (BATAN) thought of constructing an experimental power reactor using HTGR design which can be used for research, producing electricity and it is intended to be able to introduce nuclear reactor to the people. This reactor User Requirement Document (URD) only provides reactor design general specifications, but giving specific fuel design parameters in terms of ranges of the option [5], thus until now, the studies to obtain optimum design of the reactor are still continuing. The precise and accurate analysis is necessary to design a safe reactor and the neutronic analysis is also crucial in fuel management, so it can be used effectively and optimally. It is also important that the reactor must be able to maintain critical condition as long as possible, so the key to designing the reactor is fuel geometry.

The reactor neutronic analyzes are mostly performed computationally to solve the transport equation in the reactor. The widely-used Monte-Carlo computer code for neutronics computation, MCNP was employed in this study. The version of MCNP used in this study was MCNPX 2.6.0 which has the feature to calculate the fuel burnup. For all the calculations, MCNPX utilizes continuous energy data library ENDF/B-VI [6]. A lot of research on HTR-10 using MCNPX has been done so far. According to Zuhair (2012) [7], the fuel enrichment of HTR-10 affected in increasing the reactor lifetime and multiplication factor. The fuel burnup of the reactor could reach 72 GWd/MTU according to Wu (2014) [4].

By using MCNP version 4C, Oktajianto (2015) [8] discovered the optimum design of HTR-10. There are three designs that achieved critical condition. Those are in the fuel enrichment of 12-15% at a 250 μm kernel radius, fuel enrichment of 11-14% at 275 μm kernel radius and fuel enrichment of 10-13% at a 300 μm kernel radius. Further investigation can be performed by using these design parameters to do the burnup calculation to find out the fuel depletion that is related to fuel control and reactor output power.

2. HTR-10 pebble-bed description

The HTR-10 is used as a research reactor which operated on 10 MW thermal power. It is a cylinder-shaped with cone-shaped at its lower end. The core is filled by sphere-shape fuel elements and moderators which can usually be described as pebble-bed. HTR-10 pebble-bed uses graphite as neutron moderator and helium as a coolant. The smallest fuel elements used in the reactor core are tri-isotropic (TRISO) particles. The individual TRISO particle is loaded with 5 g heavy metal (uranium in this case). This article is made of UO_2 kernel with 17% enrichment and 10.4 g/cm^3 density coated by layers of carbon. From the inside, there is carbon buffer made of pyrolytic carbon, inner pyrolytic carbon (IPyC), silicon carbide (SiC) and outer pyrolytic carbon (OPyC) which thicknesses are 0.009, 0.004, 0.0035 and 0.004 cm respectively. The TRISO particles are spread in a graphite matrix inside a fuel ball which has a radius of 3 cm (2.5 cm in full region radius). There are graphite balls alongside the fuel balls filling the reactor core-forming pebble-bed [9]. Fuel element and moderator characteristics of HTR-10 pebble-bed [9] are: Density of coating layer: PyC 1.1 g/cm^3, IPyC 1.9 g/cm^3, SiC 3.18 g/cm^3 and OPyC 1.9 g/cm^3. Boron impurity content in uranium is 4 ppm, in graphite 1.3 ppm, in moderator is 1.3 ppm.

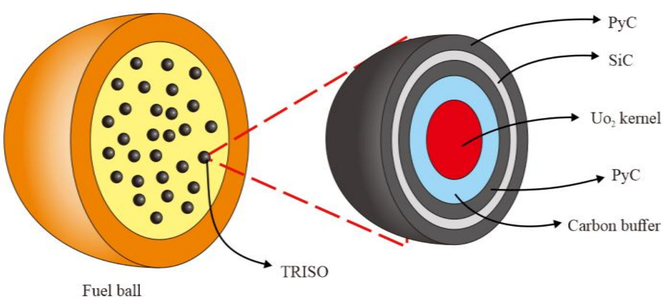

Figure 1. HTR-10 fuel element and TRISO particle

The active core has a dimension of 180 cm in diameter and 197 cm in height. There are approximately 27,000 fuel balls are randomly distributed inside the core forming a ratio of fuel and graphite balls, which is 57:43 [10]. Reactor's active core is also filled with fuel and graphite pebbles surrounded by graphite reflectors and the graphite reflectors are surrounded by layers of boronated carbon bricks. In the side reflector around the active core, there are twenty borings (130 mm diameter each). The borings are for control rods (ten borings), small absorber balls (seven borings) and irradiation channels (three borings). There are also twenty flow channels (80 mm diameter each) in the side reflector for helium channels. The active core has a dimension of 180 cm in diameter and approximately 197 cm in height. There are approximately 27,000 fuel balls randomly distributed inside the core forming a ratio of fuel and graphite balls, which is 57:43 [11].

3. Modeling and calculation using MCNPX

Before modeling HTR-10 pebble-bed using MCNPX, the first thing to do is to calculate the density of the components of the reactor as inputs in MCNPX. Density in MCNPX indicates the number of particles which form a cell card. There are boron, carbon, and uranium densities to be counted. Modeling HTR-10 pebbled-bed begins with creating a TRISO particle that contained UO_2 kernel coated with carbon layers as shown in figure 2. The TRISO particles are distributed in the fuel zone using simple cubic (SC) lattice. The fuel ball has a radius of 2.5 cm and layered by graphite which thickness of 0.5 cm. To maintain a constant ratio of uranium-graphite, the packing fraction of 0.5 is made constant in each kernel radius. The Fuel ball that contained by TRISO particles is shown in figure 3.

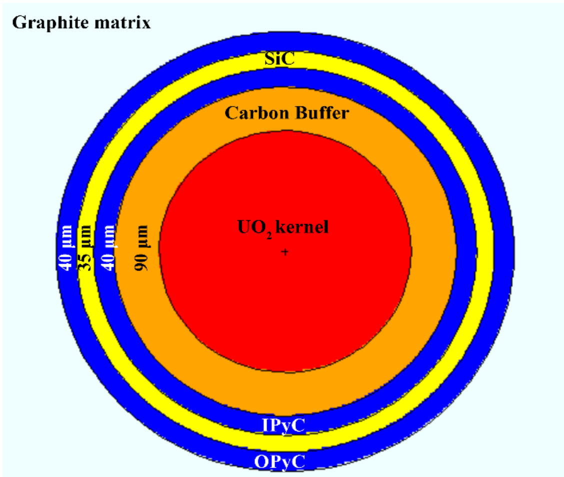

Figure 2. Representation of the TRISO particle model used in MCNP.

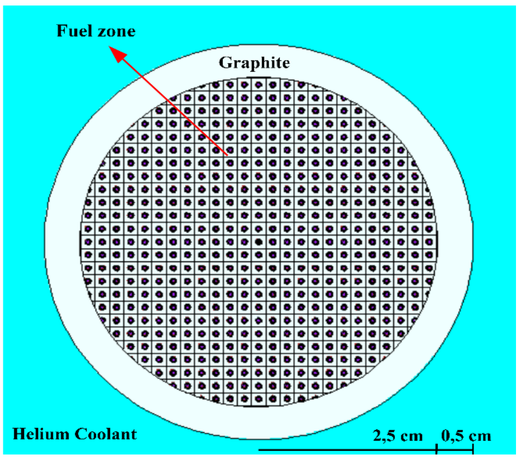

Figure 3. Representation of the fuel ball model used in MCNP.

Fuel ball and moderator ball are distributed in the reactor core using a body-centered cubic (BCC) lattice with the packing fraction and the ratio of fuel and moderator ball of 0.61 and 57:43 respectively. The fuel ball is placed at the center of the lattice while one-eighth of the moderator ball at the corner. After finishing all the fuel region of the reactor, the entire reactor structure can be created. The cross-sectional view of the modeled structure is shown in figures 4 and 5.

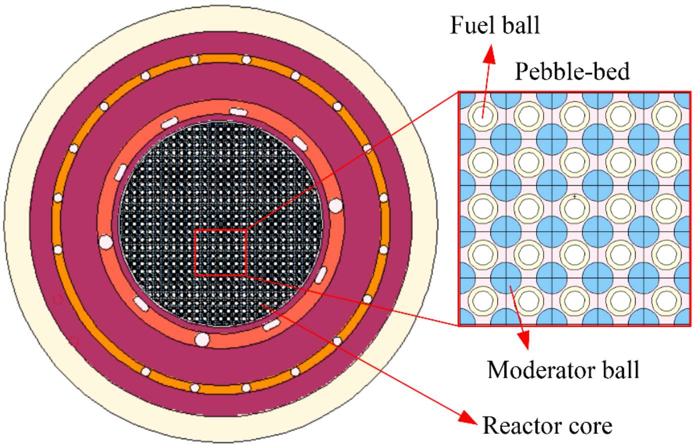

Figure 4. Representation of the horizontal cross-section of the HTR-10 model used in MCNP.

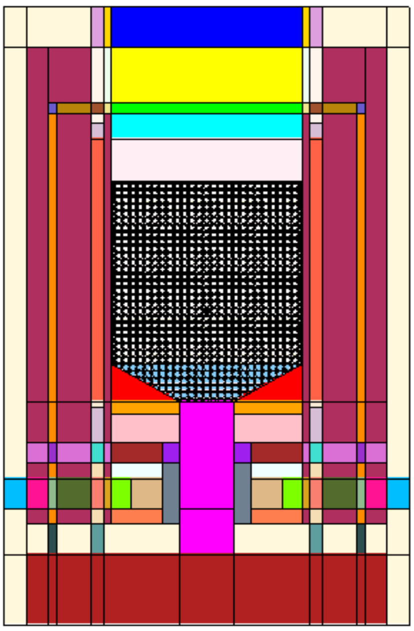

Figure 5. Representation of the vertical cross-section of the HTR-10 model used in MCNP.

The MCNPX (Monte Carlo N-Particle e-Xtended) version has the new built-in capabilities compared to the previous versions, it is to calculate fuel burnup due to its integration with CINDER90 depletion code. CINDER90 generally utilizes one group cross sections tallied by MNCPX [12]. By adding the burn card in MCNPX input, a burnup calculation can be performed. Some parameters must be specified to be added in the data card and burn the card to provide precise calculations. There will be 110 cycles and 1000 neutrons generated per cycle. An initial guess for k_{eff} is set to 1.0 so the accumulated calculations of k_{eff} are expected to approach the critical condition of a reactor. Skipping 10 cycles before data accumulation needs to be performed to avoid convergence of the source and the fission source can be stable before averaging the k_{eff}. The burnup period is set to 360 days with 30 days of burn step which means that every 30 days, the calculation will be accumulated. The power of 10 MW (thermal) with a fraction of 1.0 is applied to every burning step. Since this study only analyzes what related to fuel management, only the UO_2 is selected to be analyzed in the *burnup* process. All the calculations are done using continuous energy data library ENDF/B-VI.

4. Results and Discussions

If the k_{eff} value is equal to 1.0 (k_{eff} =1, a reactor can be described in critical condition. The k_{eff} value of 1.0 must be maintained so that it is more than equal to 1.0 ($k_{eff} \geq 1$), therefore the reactor can continue to operate and chain fission reactions can also continue to carry out. The radius and the fuel enrichment are varied in this study, thus the optimum combination of radius and fuel enrichment can be obtained from the design. The calculation results of different radius and fuel enrichment deliver different results of k_{eff} values as shown in figures 6,7 and 8.

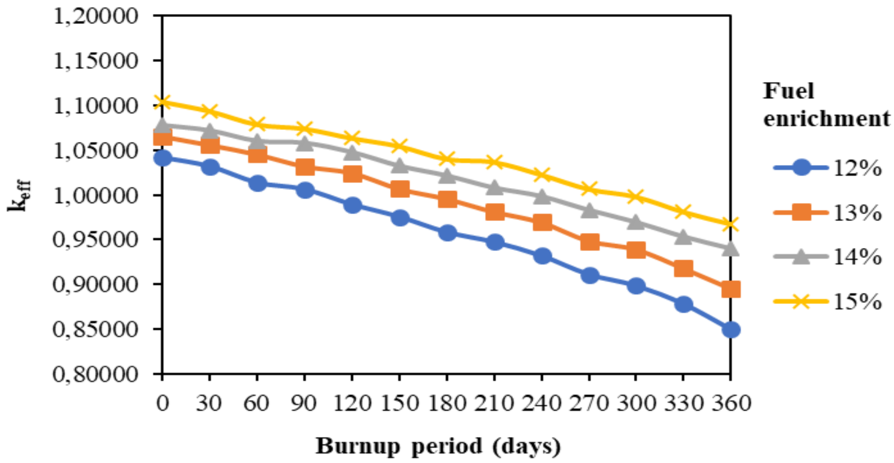

Figure 6. k_{eff} value as the function of fuel enrichment in a 250 µm kernel radius.

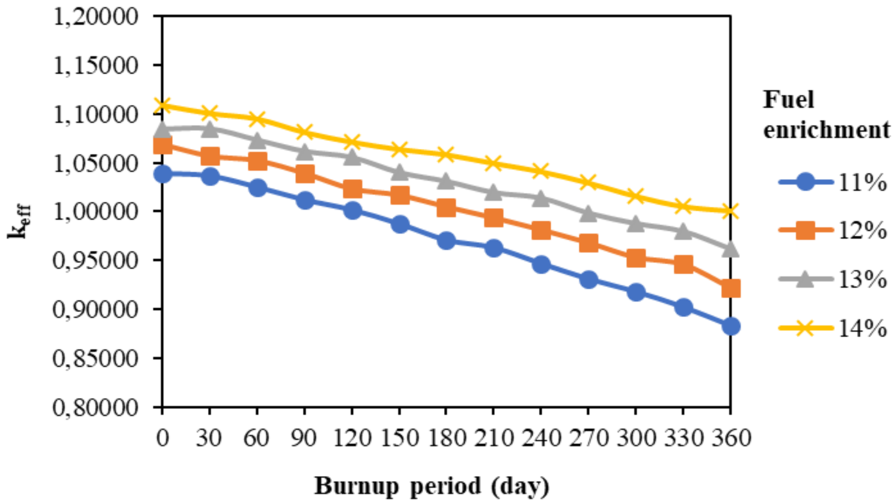

Figure 7. k_{eff} value as the function of fuel enrichment in a 275 µm kernel radius.

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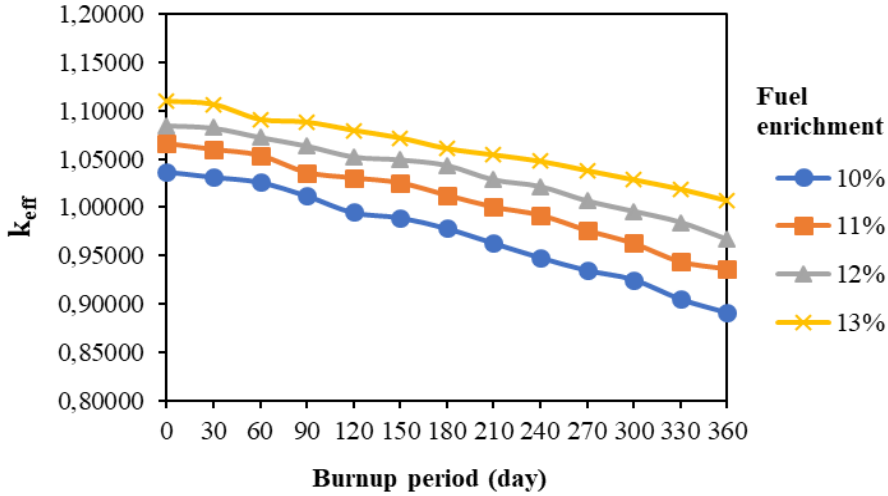

Figure 8. k_{eff} value as the function of fuel enrichment in a 300 μm kernel radius.

From figures 6, 7 and 8, we can analyze the effects of variations in kernel radius and enrichment on the k_{eff} value that operating for one year. Based on the obtained value of k_{eff}, the higher fuel enrichment results in a higher k_{eff} value and is more likely to maintain a critical condition for a year of operation. Since the increasing number of U^{235} contained in the fuel particles, thus the thermal neutrons are produced to induce more and more chain fission reactions. A large radius of fuel kernel will have an impact on the volume of fuel which is also large, it means that the number of uranium atoms will increase in the fuel particles. This will result in increased fission in the core, therefore the k_{eff} value is also high.

Kernels with a radius of 275 μm and 300 μm at 14% and 13% fuel enrichment respectively were more able to maintain a critical condition within a year of operation than the kernels a 250 μm radius. The subcritical condition of a reactor (k_{eff} <1) can be overcome by performing online refueling. It means that refueling needed to be done when the reactor is still in critical condition so that the reactor can still produce electricity even though refueling is in progress. The supercritical state of the reactor (k_{eff} >1) can be overcome by using control rods. The insertion of the control rods at certain depths will affect the amount of neutron absorbed by the control rods. This will also affect the value of k_{eff}.

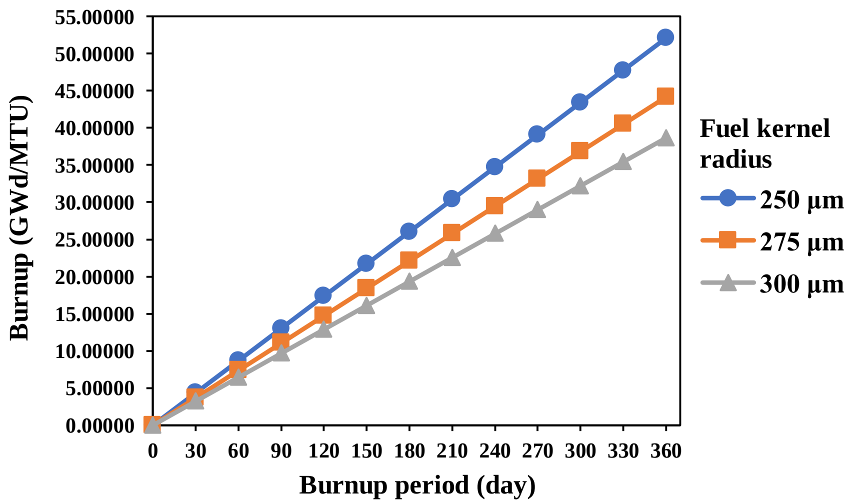

Figure 9. Burnup value as the function of radius versus burnup period.

Figure 9 presents the calculation result of the burnup rate value with the variation of fuel radius. The burnup rate states the amount of energy produced for every unit of weight of fuel in the reactor. The unit used in expressing the burnup rate is Gigawatt days per metric tonne of uranium (GWd/MTU), which is the mass of heavy metal uranium used to produced power per day. From figure 9, we can get different burnup rates for each radius. The greater the radius of the fuel used, causing the amount of uranium loaded in the increase of the fuel, therefore the number of fission materials is not proportional to the small number of graphite in the pebble bed.

Kernels with a 250 μm radius have the highest fuel burnup value of 52.07 GWd/MTU up to one year of operation compared to kernels with a radius of 275 μm and 300 μm which value of 44.21 and 38.69 GWd/MTU respectively. This indicates that the fuel in the reactor with a kernel radius of 250 μm burns optimally during one year of operation. These fuel burnup values didn't reach the limit (72 GWd/MTU) [4] due to its only one year of operation.

5. Conclusions

A burnup calculation on an HTR-10 pebble bed has been performed using MCNPX with three different parameters in radius and fuel enrichment. From the calculation results, we can conclude that kernels with a radius of 275 μm and 300 μm at 14% and 13% fuel enrichment respectively were more able to maintain a critical condition within a year of operation than the kernels a 250 μm radius. It means from the criticality aspect within a year, that those designs can be considered. While the aspect of fuel management, we can conclude that the fuel with a 250 μm radius has the highest burnup of 52.07 GWd/MTU.

Acknowledgments

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