Preliminary analysis of dose rates distribution of experimental power reactor 10 MW using MCNP

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Abstract. The reaktor daya eksperimental (RDE) is 10 MWth experimental power reactor HTR Pebble-bed type. Comparing with water pool research reactors, the distance between the core and the reactor vessel is very close while the medium mainly is graphite. Thus, the main role of the radiation shield relies on biological shields to obstruct radiation exposure coming from the reactor core. Meanwhile, the RDE reactor should be met with the operation standard safety limits of a nuclear reactor. Therefore, the purpose of this study is to estimate the radiation dose rates in the working area and outside the RDE reactor building. By analyzing the distribution of radiation dose rates can be estimated radiation safety levels for workers and society and the surrounding environment. The radiation dose rates calculation was done using MCNP. By determining the tally position, it can be determined the radiation dose rates inside and outside the RDE reactor building. The results show that the maximum dose rates on the outer surface of the biological shielding (working area) under normal operating conditions are $8 \ \mu$Sv/h. That dose rates are below the limit value determined by BAPETEN showing the radiation shielding design make the RDE safe for workers and the community and the surrounding environment from radiation hazards.

Keywords : RDE, shielding, radiation safety, dose rates.

1. Introduction

It has been being done various studies in facing the development of reaktor daya eksperimental (RDE) of high temperature gas cooled reactor Pebble-bed reactor with a nominal power of 10 MWth. The RDE design referred to the Germany HTGR technology which was applied to Chinese HTR-10 reactor [1,2]. The RDE fuel are thousands of 17% enriched UO2 particles coated with TRISO [3,4] in the 6 cm diameter graphite sphere (pebbles). The RDE reactor categorized as a Generation IV reactor because it offers many advances compared to previous generation. The RDE reactor should be met with the operation standard safety limits of a nuclear reactor. The dose rate analysis of the RDE reactor is one of the most important studies in safety concern.

By estimating the characteristic of the radiation dose rate, it can be determined that the limits of duration and space for workers in a nuclear reactor to ensure radiation safety following ALARA principles. Therefore, the objective of this study is to estimate the dose rates in the working area of the RDE reactor.

The calculation of the distribution of dose rates inside and through the outside of RDE reactor building is done using the MCNP [5] computer program. The MCNP program is a reliable computer program and has been widely used for dose rate analysis by many domestic and overseas researchers.
The calculation of the dose rate is carried out under normal operating and shut-down conditions. Under normal operating condition, all radiation sources in the reactor core such as radiative capture, spontaneous fission, and radioactive decay contribute to the dose rates calculation. While at the shut-down condition, the calculation of dose rate only comes from gamma decay source of radioactive material in reactor core.

The ENDF/B-VII cross section library was applied in this MCNP calculations. However, the output of MCNP is in normalized value which have to convert into absolute dose rates using a conversion factor. The magnitude of the conversion factor is derived from the core source strength taken from the ORIGEN2.1 [11-14] calculation and the analytical calculations.

2. Methodology

2.1. RDE Modeling

The model of RDE HTR pebble-bed reactor including radiation shielding was made based on the data of core geometry and the composition of the reactor core and other components. The reactor core is modeled as the RDE working core with height 197 cm. Modeling was done using VisEd software [5] as part of MCNP package. Modeling is an important step for the next analysis that can be done correctly.

2.2. Calculation of RDE Core Source Strength

Based on the core composition and other data then the next step has is source strength calculation of the of RDE reactor core in operating condition. There are 3 (three) types of gamma radiation sources and also the neutron source located inside the reactor core. The source of the gamma reactor core is derived from radiative capture reactions, spontaneous fission reactions, and radioactive decay processes. All types of radiation sources of the reactor core occur simultaneously and mixed together. Theoretically, all types of radiation sources can be separated from each other. The radiative capture and spontaneous fission sources strength were calculated analytically. The source strength calculation of the decay of radioactive material is performed using the ORIGEN2.1.

The gamma source of the radiative capture is the gamma radiation that occurs because the nuclei of the atom (nuclide) undergo a neutron capture reaction. Sources strength of radiative capture is calculated on the basis of equation (1) [15,16].

\[ K_c = \sum_{m=1}^{M} \sum_{i=1}^{N} Q_i \times p_{im} \]  

with:
- \( Q_i \) = reaction rates of radiative capture = \( \sigma_i \times N \times \phi \), \( \sigma_i \) = cross section of radiative capture of \( i^{th} \)-nuclide (cm\(^2\)), \( N \) = number of \( i^{th} \)-nuclides and \( \phi \) = neutron flux,
- \( p_{im} \) = the probability of gamma rays being emitted in any radiative capture of the \( i^{th} \)-nuclide in the \( m^{th} \)-energy range,
- \( M \) = number of energy groups: 7 groups: 0 – 1 MeV, 1 – 2 MeV, 2 – 3 MeV, 3 – 4 MeV, 4 – 6 MeV, 6 – 8 MeV and 8 – 11 MeV,
- \( n \) = the number of nuclides undergoing radiative capture reactions.

The gamma source of spontaneous fission is the gamma radiation that emits at the time of the fission reaction of 235U atom. Gamma sources strength of spontaneous fission is calculated by equation (2) [15,16].

\[ K_f = K_i \times f_r \]  

with:
- \( K_i = \frac{1}{(E_{i+1} - E_i)^{1/2}} \int_{E_i}^{E_{i+1}} E \times \eta(E) \, dE \)  

\( \eta(E) = 26.0 \, e^{-2.3E} \) (0.3 < \( E \) < 1)
\( \eta(E) = 8.0 \, e^{-1.1E} \) (1 < \( E \) < 7)
and

\[ f_r = 6.25 \times 10^{18} \frac{P}{E_n} \text{ fisi/detik} \]  

(4)

with:

- \( E_i \) = gamma energy in \( i^{th} \)-group,
- \( P \) = reactor power,
- \( E_n \) = gamma energy emitted when a fission reaction occurs (~200 MeV).

Radioactive substances that emit gamma radiation are the result of fission, activation and other reactions inside the reactor core during operation. This gamma radiation emission is the result of the decay process of these radioactive substances. This gamma decay source strength is calculated using the ORIGEN2.1 by simulating fuel burn-up in accordant to the reactor operation pattern. The ORIGEN2.1 calculations are based on the depletion theory and decay reaction of radioactive substances.

2.3. Calculation of Dose Rates Distribution of RDE Reactor

Based on the model that has been made and the result of source strength calculation, the next step is the calculation of dose rate distribution of RDE reactor. The calculation of gamma dose rates using MCNP has taken into account all types of radiation sources located within the reactor core when the reactor in operating mode. This MCNP simulation by utilizing of continuous energy cross-section data taken from ENDF/B-VII file [17-19]. The main inputs required are the composition and size of the geometry of reactor core, reflector, reactor vessel, biological shield and reactor building. By determining the tally position, it can be calculated the dose rate of the RDE reactor. Calculation of dose rate with MCNP applying kcode card option. Because still in the model improvement, the calculation is still applying a relatively rough approach that is with as many as 1000 particles per cycle done as much as 250 cycles so it still provides relatively small deviation.

The output of the MCNP program is a gamma spectrum that is in normalized value. So it needs a quantity as a conversion factor in order to obtain absolute source strength. The magnitude of the neutrons source strength for the average of cell unit volume can be determined by equation (5) [20]. So that the total gamma source strength of the RDE core is the cell source strength multiply by the volume of the reactor core.

\[ FM = 3.15E10 * P * \frac{\eta}{k_{eff}} * \phi_{tally} \]  

(5)

with:

- \( P \) = reactor power = 10 MWth,
- \( \eta \) = the number of neutrons produced in a fission reaction,
- \( k_{eff} \) = the effective multiplication factor of the reactor core,
- \( \phi_{tally} \) = output tally at the position specified on the MCNP input.

Calculation of the dose rate is also done in shut-down conditions. While at the shut-down condition, the calculation of dose rate only comes from decay gamma source of radioactive material. This is because of the process of the radiative capture and spontaneous fission stopped in a shut-down condition. The conservative decay gamma source strength was gotten from the RDE reactor operation simulation to the maximum burn-up of 80 GWD/THM using ORIGEN2.1. The dose rates calculation was done using MCNP in SDEF source mode with appropriate particles statistic.

3. Results And Discussion

3.1. RDE Model

The RDE reactor core consists of around 27,000 fuel pebbles. The uranium fuel load rate is 5 grams per pebble with 17% U-235 enrichment. The core diameter and height of the first criticality are 180
and 123.06 cm [21] while the operational core height is 197 cm. The RDE reactor model can be seen in Figure 1.

![Figure 1. The RDE reactor model.](image)

This modeling was made based on geometry data and RDE reactor core composition and other components. As shown in Figure 1, the reactor core containing fuel is modeled in a homogeneous form in working core condition with height 197 cm. The reactor core is surrounded by neutron reflector made of graphite material. Outside the reflector was covered by a stainless steel barrel. All of the core components are incorporated into stainless steel reactor vessel. The regular concrete of 2.3 g/cm³ density was used as biological shielding and reactor building.

The result of effective neutron multiplication factor (k-eff) calculation of RDE 10 MWth HTR Pebble-bed at BOC and EOC conditions are 1.08586 ± 0.00018 and 1.04257 ± 0.00017 respectively. This result can be reasonably believed because the calculation of k-eff in critical condition with the core height of 123.06 cm is 0.99673 ± 0.00039 which is very close to the measurement (k-eff = 1) while the benchmark result for HTR-10 uses an SRAC program is 0.9973 for a 120 cm core height [21]. The results of all the above calculations still assume that all control rods are pulled out from the reactor core. Thus it can be seen that the RDE core model that has been made is already proper.

### 3.2. RDE Core Source Strength

The results of the RDE core source strength calculation of radiative capture, spontaneous fission, and radioactive decay were plotted in Figure 2 together with the gamma flux to dose conversion factor. Gamma radiation source strength coming from the radiative capture reaction is the highest source strength and emits high gamma energy. In spontaneous fission and radioactive decay are also emit high-energy gamma but in low intensity, so their contributions are very low, even though the gamma flux to dose conversion factor has a high value at high energy. These three types of core radiation sources occur simultaneously and are mixed together when the reactor in operating condition.
3.3. Dose Rates Distribution of RDE Reactor

Dose rates calculation of RDE Reactor was done based on the model that has been made as can be seen in Figure 1. The results of gamma dose rates distribution calculations in normal operation conditions in the RDE reactor building are plotted in Figure 3.

![Gamma source strength RDE core](image)

**Figure 2.** Gamma source strength RDE core.

![Gamma dose rates distribution at RDE reactor](image)

**Figure 3.** Gamma dose rates distribution at RDE reactor.

It is seen in Figure 4 that gamma dose rates in the working area that on the outer surface of the biological shield in normal operation condition is 8 μSv/hr. These dose rates outside the biological shield are still below the dose rate of 10 μSv/h (20 mSv/year dose) i.e. the dose limit value (DLV) for workers determined by BAPETEN [21]. Thus 2 m biological shield thickness made of ordinary concrete with a density of 2.3 g/cm³ is quite safe to protect workers from radiation hazards. Thus the Pebble-bed HTR reactor under normal operation conditions is safe for workers.

This estimation of the RDE radiation dose rate characteristic can be used to determine the limits of duration and space for workers to ensure radiation safety following ALARA principles. The application of the ALARA principles, it can increase the ensuring of radiation safety for workers.
4. Conclusion
It can be concluded that the calculation results of the gamma dose rates of RDE Pebble-bed HTR reactor under normal operating conditions in the working area are below the limit value determined by BAPETEN. This result indicates that it is safe for workers from radiation hazards. The application of the ALARA principles, it can increase the ensuring of radiation safety for workers.

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