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GAMMA DOSE RATE ANALYSIS IN BIOLOGICAL SHIELDING OF HTGR-10 MWth PEBBLE-BED REACTOR

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ABSTRACT.

GAMMA DOSE RATE ANALYSIS IN BIOLOGICAL SHIELDING OF HTGR-10 MWth PEBBLE BED REACTOR. HTGR-10 MWth is a high-temperature gas-cooled reactor. The fuel and moderator are pebble shaped with a radius of 3 cm. One fuel pebble consists of thousands of UO_2 kernels with a density of 10.4 gram/cc and the enrichment rate of 17%. The core of HTGR-10 MWth is the center of origin of neutrons and gamma radiation resulting from the interaction of neutrons with pebble fuel, moderator and biological shield. The various types of radiations generated from such nuclear reactions should be monitored to ensure the safety of radiation workers. This research was conducted using MCNP-6 Program package with the aim to calculate and analyze gamma radiation dose in biological shield of HTGR-10 MWth. In this study, the biological shield is divided into 10 equal segments. The first step of the research is to benchmark the created program against the critical height of HTR-10. The results of the benchmarking show an error rate of ± 1.1327%, while the critical core height of HTGR 10 MWth for the ratio of pebble fuel and pebble moderator (F:M) of 52: 48 occurs at a height of 134 cm. The rate of gamma dose at the core is 3.0052E + 05 mSv/hr. On the biological shield made of regular concrete with a density of 2.3 grams/cc, the rate of gamma dose decreases according to an equation $y = 0.0042\ e^{-0.03x}$. Referring to Perka Bapeten no 4 of 2013, the safe limits for workers and radiation protection officers will be achieved if the minimum thickness of biological shield is 115 cm with gamma dose rate of 0 mSv/hour.

Keywords: Gamma dose rate, HTGR 10 MWth, biological shield, pebble

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ABSTRAK

ANALISA LAJU DOSIS GAMMA PADA PERISAI BIOLOGIS REAKTOR PEBBLE-BED HTGR 10 MWTH. *HTGR 10 MWth merupakan reaktor temperatur tinggi berpendingin gas dengan bahan bakar dan moderator berbentuk pebble beradius 3 cm. Satu bola bahan bakar terdiri dari ribuan kernel* UO_2 *berdensitas padatan 10,4 g/cm*^{3} *dengan tingkat pengkayaan uranium sebesar 17% (17%-235U). Teras HTGR-10 MWth merupakan pusat asal radiasi neutron maupun gamma hasil dari interaksi neutron dengan bahan bakar pebble, moderator maupun perisai biologis. Radiasi yang dihasilkan dari reaksi nuklir tersebut harus terpantau untuk menjamin keselamatan pekerja radiasi. Penelitian ini dilakukan dengan menggunakan paket Program MCNP6 dan bertujuan untuk menghitung dan menganalis dosis radiasi gamma yang terjadi pada perisai biologis HTGR-10 MWth. erisai biologis terbagi dalam 10 segmen yang sama. Langkah awal penelitian adalah melakukan bechmark program yang dibuat terhadap tinggi kritis HTR-10. Hasil benchmark menunjukkan tingkat kesalahan sebesar ±1,1327%. Sedangkan tinggi teras kritis HTGR 10 MWth untuk rasio bahan bakar pebble terhadap moderatur pebble (F:M) sebesar 52:48 terjadi pada ketinggian 134 cm. Laju dosis gamma yang terjadi pada teras adalah sebesar 3.0052E+05 mSv/jam. Pada perisai biologis yang terbuat dari beton reguler berdensitas 2,3 g/cm*^{3}*, laju dosis gamma menurun mengikuti persamaan* $y= 0,0042\ e^{-0,03x}$*. Dengan merujuk Perka Bapeten no 4 tahun 2013 maka batas aman untuk pekerja dan petugas proteksi radiasi akan tercapai bila minimal ketebalan perisai biologis sebesar 115 cm dengan nilai laju dosis gamma 0 mSv/jam.*

Kata Kunci *:Laju Dosis Gamma, HTGR 10 MWth, Perisai biologis, Pebble*

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INTRODUCTION

The HTGR 10 MWth pebble Bed Reactor is one model of the fourth generation reactors designed to have passive safety inherent. One of application from the inherent safety concept on HTGR10 MWth is the concept of negative reactivity. In the concept of negative reactivity reactor, the process of fission reactions on the HTGR 10 MWth reactor is always in controlled conditions so that the core and reactor fuel will never melt in the event of an accident[1-8]. This condition occurs because core construction materials and reactor fuels of HTGR 10 MWth pebble bed reactor are dominated by graphite elements which can absorb neutrons, so the reactor core is always in a state of negative reactivity. However, research on radiation dose comes from the nuclear reactor is still needed to support reactor safety and radiation workers.

In the reactor core, the interaction of neutrons with the fuel produces several types of radiation that one form of gamma radiation (photons). Gamma Radiation dose calculated is the result of neutron and gamma radiation reaction in the fuel inside the reactor core. Consequently Gamma Source Strength (S) will greatly affect the number of neutrons produced in each fission event (v) to produce more power per watt. To calculate strength source gamma radiation need to do a conversion from watt-s to fission through of conversion equation (1).

$$CF = \left(\frac{1^{J}/_{s}}{Watt}\right)\left(\frac{1\,MeV}{1.602x10^{-13}J}\right)\left(\frac{fisi}{180\,MeV}\right)$$
$$= 3.47\;x\;10^{10}\;fission/Watt.s \quad (1)$$

While the Gamma Source Strength described in equation (2).

$$S = 3.47\;x\;10^{10}\left(\frac{fission}{Watt.s}\right)x\;P\;(Watt)x\;v \quad (2)$$

ν value can be obtained directly from MCNP 6.1 output in the criticality calculations, so that the actual flux can be calculated by the equation [21]:

$$\phi_{actual} = \phi_{F4}\left(\frac{1}{k_{eff}}\right) S \quad (3)$$

Φ_{actual}: The gamma flux obtained from the MCNP 6.1 calculation to tally F4

While the absorption process of gamma doses by the material shielding is calculated through equation (4).

$$D_t = D_0\, e^{-vx} \quad (4)$$

Where Dt is gamma dose rate after through the material shielding and Do is gamma dose rate before through the material shielding and x is the thickness of the shielding material which the passed of the gamma dose

The HTGR 10 MWth reactor core is a source of neutron and gamma radiation that interacts with a biological radiation shield. On biological radiation shields, neutron and gamma radiation will be absorbed according to the thickness and type of biological shield used. The radiations must be continuously monitored to ensure the safety of radiation workers. Research on radiation doses on nuclear reactors and biological shields using MCNP5 program packages with Tally F2 and F5 has been widely done [9-13], similarly the research regarding calculations inventory analysis of HTGR fuel using MCNPX program package has also been done [14-16]. However, the current study will use the package program MCNP6 with Tally F4 to analyze the dose of gamma radiation from the HTGR core.

The purpose of this study is to analyze the dose of gamma radiation coming out from the biological shield model of the HTGR 10 MWth Pebble Bed Reactor in normal operation. The result would be essential to make the required action of radiation worker. Therefore the radiation dose received by the worker will not exceed the allowed limit. The study was conducted

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using the program package MCNP 6 version 1.0 by using Tally F4. While Visual Editor (Vised) version 24E used to serves as a visualization program to see the results that have been made.

HTGR-10 MWth

HTGR-10 MWth pebble bed reactor is a high-temperature reactor using inert helium gas as the cooling medium. This reactor is designed to operate at 10 MWth power. HTGR-10 MWth fuel pebble-shaped (solid ball) has a diameter of 6 cm. Each pebble fuel has a graphite matrix 5.0 cm in diameter. While in the graphite matrix there are 8335 kernels. Every kernel contains UO_2 with enrichment of 17%. Each kernel is coated with a TRISO layer. In the pebble, kernels are arranged in the form of a Simple Cubic (SC) lattice. The graphite matrix material is then coated with graphite as thick as 0.5 cm. The moderator of HTGR-10 MWth pebble bed reactor also form a solid ball with 3 cm in diameter, made of graphite material. In the reactor core, the ratio of fuel and moderator pebble in the core is 52:48. Pebble fuel and moderator are arranged in a lattice form BCC (Body-Centered Cubic)[17].

HTGR-10 MWth reactors pebble-bed has a radius core of 90 cm surrounded by a reflector made of graphite material with 100 cm in thickness and reactor pressure vessel (tank pressure) is made from stainless steel with a thickness of 8 cm. In the reflector area contain there are 10 holes locations control rod, 7 holes absorber balls, 3 irradiation holes and 20 holes with helium cooling. The reactor was then surrounded by Reactor Cavity Cooling System (RCCS) contain dry air that serves to cool the reactor pressure tank, then coated biological radiation shield which made of regular concrete with a density of 2.3 g/cm^3 and the initial thickness for calculation model is 200 cm. The height of active core HTGR-10 MWth pebble bed reactor is 197 cm with the input temperature of helium coolant is 250 °C while of the output temperature is 750 °C and an operating pressure of 3 Mpa. In this part of the active high-top core there is a void with a thickness of 41.698 cm and a reflector on this part. The cooling medium composed of the inert helium gas and control rods made of B_4C having a density of 10.4 g/cm^3. In this paper, description of HTGR 10MWth was made following design of the HTR-10. To clarify the understanding of HTGR-10 MWth, than the reactor core parameter and fuel of HTGR-10 MWth can be seen Figure 1. Modelling HTGR-10 MWth is created using MCNP6 program packages and visualized with VISED program (Visual Editor).

METHODOLOGY

Calculation of the gamma dose rate on biological shielding radiation of HTGR-10 MWth pebble-bed reactor in this research is conducted using MCNP-6.1 program package. MCNP (Monte Carlo Particle Transport Code) is a computer program developed by Monte Carlo X-5 since 2003 by Los Alamos National Laboratory [18-20].

MCNP using a stochastic approach to simulate problems of physics and mathematics. Monte Carlo method used to simulate individual particle counting and recording some aspect of average behavior of the particles. The problems are difficult to solve with analytic approach would be simpler by using this method. MCNP is the code (program package) that is used to calculate the transport phenomena in neutrons, photons, electrons, or a combination there of MCNP has very broad applications include radiation protection and dosimetry, radiation shielding, radiography, medical physics, nuclear criticality safety, detector design and analysis, oil well logging, accelerator, fission and fusion reactor design, decontamination and decommissioning.

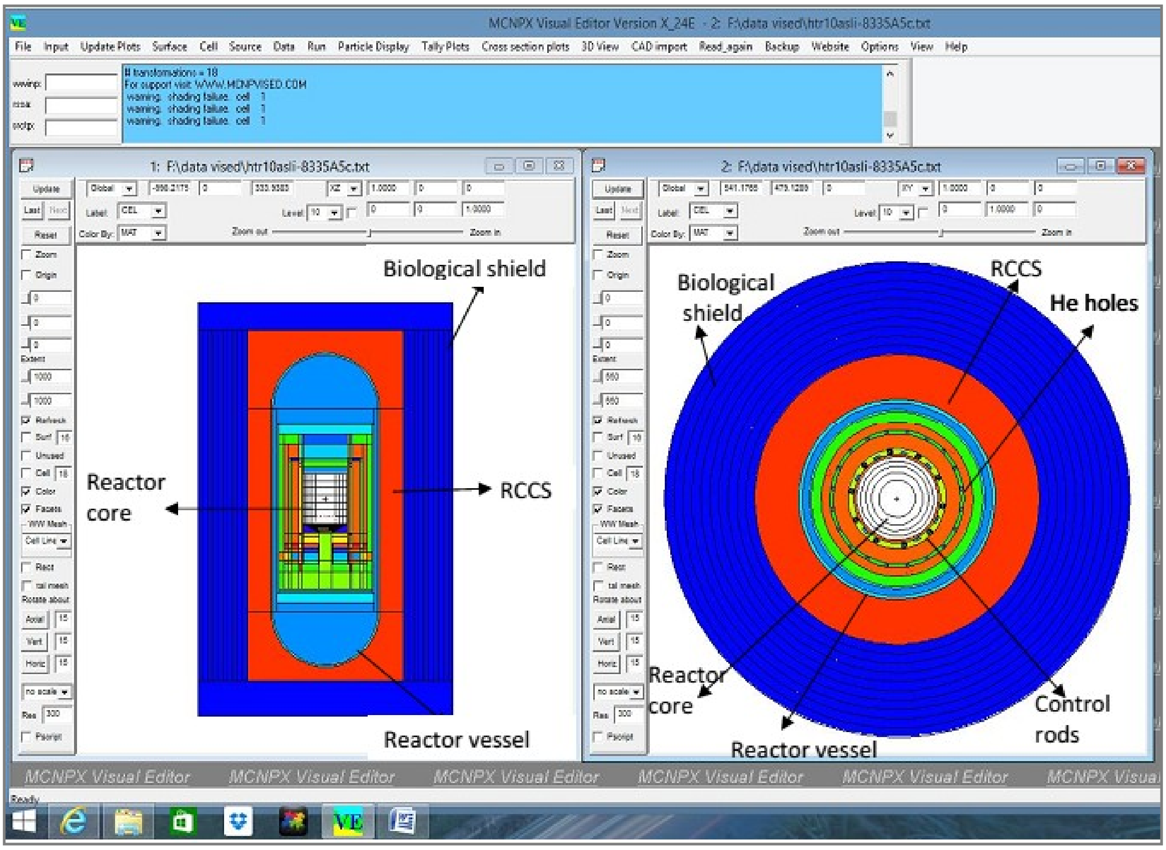

Figure1. Modeling of the HTGR-10 MWth viewed from the side and top of the program package VISED

Gamma dose rate analysis methodology for HTGR-10MWth reactor using MCNP6 started with modeling kernel-coated particles TRISO and modeling pebble mixture of fuel and moderator pebble. TRISO-coated kernels are modeled with simple cubic (SC, Simple Cubic) lattice cell dispersed in a graphite matrix. The SC lattice cell at monte carlo program MCNP6 made utilizing LAT option and FILL. In each pebble fuel contain of the 8335 kernel in active zone with diameter of 5 cm. Active zone on the fuel coated with a graphite shell 0.5 cm thick so as the diameter fuel pebble to 6 cm. While the ball is modeled as graphite moderator with diameter 6 cm. The next step is to insert a mixture of fuel and moderator pebble in the form of a BCC (Body Centered Cubic) lattice into the HTGR-10 MWth reactor core with a volume ratio of 52:48 using the MCNP6 program. Pebble packing fraction in core of HTGR-10 MWth is 61%, meaning 61% fuel pebble filled on core and 39% helium gas filled on core that serves as a coolant. Furthermore, also done modeling of the reactor structure components such as reflectors, cooling canals, cavities and the control rods. Visualization modeling kernel-coated TRISO particles and modeling of pebble fuel can be seen in Figure 2 and Figure 3.

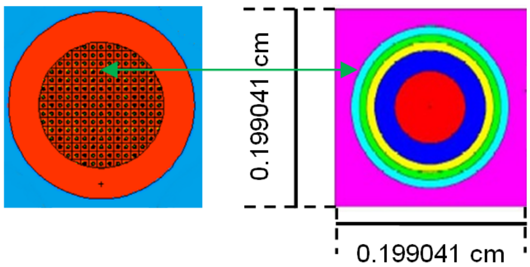

Figure 2. Visualization of kernel plated TRISO in the form of SC lattice on pebble fuel using Vis Ed program

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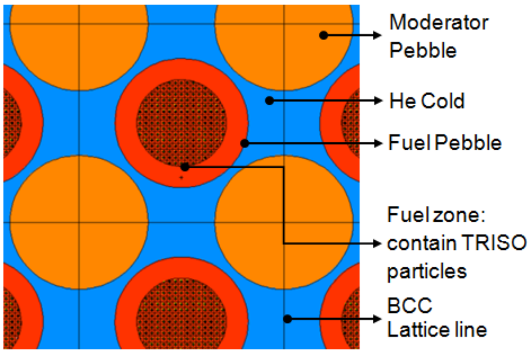

Figure 3. Modeling pebble mixture of fuel and moderator in the BCC lattice.

Furthermore, calculation of multiplication factors was carried out for the height of the active core from 80 cm to 197 cm (full core) with the gradually Then the biological shield modeling was carried out with a thickness of 200 cm with materials made of ordinary concrete with a density of 2.3 g/cm^3. Biological shield is divided into 10 segments that have the same thickness. Description of the nature of physical properties of regular concrete can be seen in Table 1.

Table 1. Physical properties of regular concrete [22]

Regular Concrete						
Density (g/cm^3)	2.3					
Total atomic Density	8.178E-02					
Element	Neutron ZA	Photon ZA	Weight Fraction	Atomic Fraction	Atomic Density	
H	1001	1000	0.01	0.168038	0,013742	
O	8016	8000	0.532	0.563183	0.046056	
Na	11023	11000	0.029	0.021365	0.001747	
Al	13027	13000	0.034	0.021343	0.001745	
Si	14000	14000	0.337	0.203231	0.01662	
Ca	20000	20000	0.044	0.018595	0.001521	
Fe	26000	26000	0.014	0.004246	0.000347	
Total			1.0	1.0	0.081778	
MCNP Form	Weight Fraction		Atomic Fraction		Atomic Density	
Neutrons	1001	-0.01	1001	0.168038	1001	0.013742
	8016	-0.532	8016	0.563183	8016	0.046056
	11023	-0.029	11023	0.021365	11023	0.001747
	13027	-0.034	13027	0.021343	13027	0.001745
	14000	-0.337	14000	0.203231	14000	0.01662
	20000	-0.044	20000	0.018595	20000	0.001521
	26000	-0.014	26000	0.004246	26000	0.000347
Photons	1000	-0.01	1000	0.168038	1000	0.013742
	8000	-0.532	8000	0.563183	8000	0.046056
	11000	-0.029	11000	0.021365	11000	0.001747
	13000	-0.034	13000	0.021343	13000	0.001745
	14000	-0.337	14000	0.203231	14000	0.01662
	20000	-0.044	20000	0.018595	20000	0.001521
	26000	-0.014	26000	0.004246	26000	0.000347

Biological shield that is at the outer circumference of HTRG-10MWth modeled using the MCNP6 program package based on the physical properties contained in Table 1. Visualization of the biological shield model viewed using Vis Ed program package which presented in Figure 4.

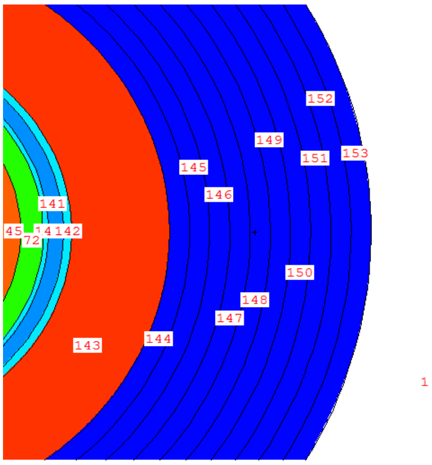

Figure 4. Visualization biological shield modeling by using MCNP6

The final stage of this study calculating the gamma dose rate on core and biological shields using option tally flux (tally F4) and conversion factors from Flux to Dose multiplied by Gamma Source Strenght (S) as the tally multiplier (FM card), so that the distribution of the dose rate of gamma will be obtained on the core and HTGR-10 MWth biological shield.

All calculations regarding the criticality, flux and gamma dose rate is done using monte carlo program MCNP6 under normal operating conditions by using the cards option KCODE 2500 1.0 10 110 and KSRC 0.0 0.0 0.0 and Mode n p. With the KCODE, KRSC and Mode options like that, then the MCNP will perform a critical calculation with a nominal source size of 2500 particles, estimate keff 1.0, skip 10 cycles before the average keff and run a total of 110 cycles with the initial source location is (0, 0, 0), which is determined by the card KSRC . The n p mode shows the influential particles in the calculation are neutrons and photons (gamma).

RESULTS AND DISCUSSION

The result from the calculation of the value of the multiplication to the core height from 80 cm to 197 cm (full core) gradually using the program package MCNP6 to HTGR-10 MWth reactor and HTR-10 as a benchmark, is visualized in Figure 5.

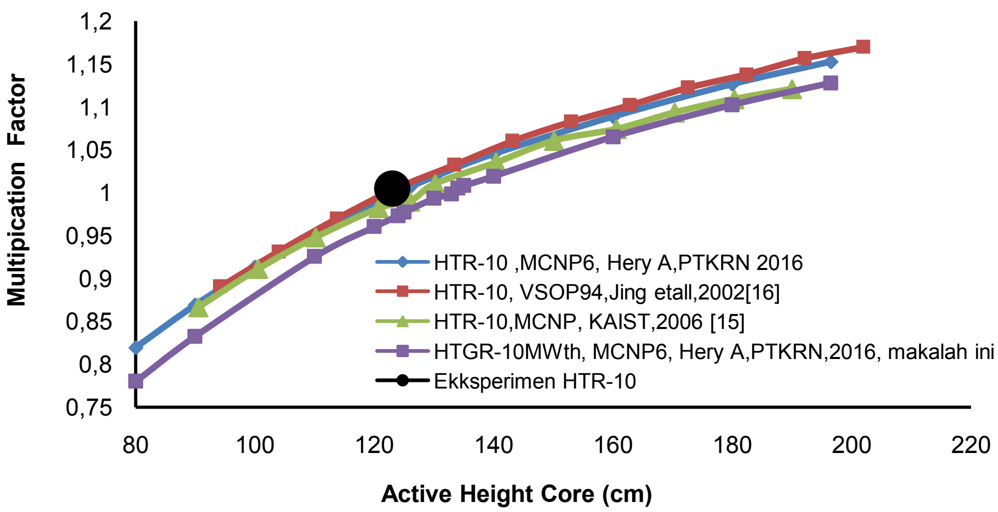

Figure 5. Calculation of core criticality on HTGR-10MWt and HTR-10

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Figure 5, it shows that for the benchmark HTR-10 charts the results obtained very close to the results produced by Jing et all [24] with VSOP'94 and KAIST [23] with MCNP. By comparing the calculation results presented in Figure 5, it is known that core height of initial critically of HTR-10 used MCNP6 by Hery A et al is 125 cm while the results of VSOP package created by Jing et al is 123.57 cm [24] and the core height of criticality from KAIST by MCNP is 130.15 cm [23]. The other hand the results of initial core height resulted from criticality experiment of HTR-10 is 123.06 [25]. The inaccuracy of the results generated by the initial core height criticality with MCNP6 against to expriment is 1.58%. Seeing very small the resulting error becomes the basis for applying programs made into HTGR-10MWth. It can be confirm that the program made good enough and give accurate results. Similarly to HTGR-10 MWth criticality calculations found that initial core height criticality of HTGR-10 MWth will occur at a core height of 134 cm with a value of multiplication factor of 1.00514. From these results it appears that in critical condition, core height of HTGR-10 MWth is higher than HTR-10. This situation is caused due to the amount of HTGR-10 MWth moderator more than HTR-10 in accordance with the ratio of fuel to the moderator pebble RDE is 52:48 while the HTR-10 is 57:43. Thus the neutrons are absorbed at HTGR-10 MWth more than HTR-10. At HTGR-10 MWth for the first full core with an active core height of 197.0 cm produced a multiplication factor of 1.12796.

In the calculation of neutron flux using MCNP6 with F4 and En Tally option. Tally remedy F4 function calculates the average flux in the cell while the tally En used to obtain the flux distribution obtained by the bin of energy the desired. Determination grid of the energy bin/number of group structure, that is made in this study using a package program EGS99304 with opsi vitamin C 36 group structure for neutron group energy structure with 36 group vitamin C. The distribution of flux on the core of HTGR-10 MWth generated by MNCP6 program packages are shown in Figure 6.

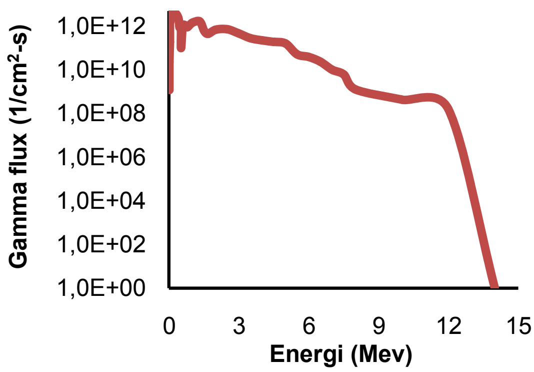

Figure 6. Calculation of core criticality on HTGR-10MWt and HTR-10.

Figure 6 shows the spontaneous gamma radiation emission (prompt fision gamma) which has an energy range of 0.3 Mev until to 9 Mev, while the gamma energy range >9 Mev is the gamma energy range due to decay [26-27]. Spontaneous gamma radiation is produced due to the occurrence of fission reaction on the reactor core so that the nucleus of fissile material divides into fissioned elements accompanied by spontaneous gamma radiation. Figure 6, also shows that gamma radiation decay decreases with increasing energy. The gamma dose rate calculations on the core and the biological shields made of regular concrete which the density of 2.3 g/cm^3 is shown in Tabel 2 and its visualization in Figure 7.

Table 2 and Figure 6 show that based on the calculation of MCNPX using the F4 option, the gamma dose rate at the center of the terrace is very high at 3,0052E+05 mSv/hour. After passing through the core of the reactor, the gamma radiation dose decreases to 8.9770E+04 mSv/hour. On the core area, the dose of gamma radiation has a small decrease in dose rate. This is because the core only contains pebble fuel, moderator and helium cooling media, where the total attenuation factor for all the material is still small. At the time of passing through the vessel and RCCS the gamma radiation dose decreased sharply to 5.0952 E+00 mSv/hour. This is because the area contains material that has a large attenuation factor. Based on Perka Bapeten No. 4 of 2013, which sets the limit of the gamma radiation dose rate allowed for radiation workers is 20 mSv/year equal to 0.010 mSv/hour [28]. Thus the dose rate coming out of the HTGR 10 MWth reactor vessel and RCCS is still very large and not safe for radiation workers. Therefore it is necessary to make biological shielding made of ordinary concrete with a density of 2.3 g/cm^3 to reduce gamma radiation so that it is safe for workers. The calculation results of gamma radiation dose to shielding thickness using MCNPX can also be seen in Figure 7. To clarify the dose that occurs in biological shielding, Figure 8 shows a decrease in the dose of gamma radiation on shielding radiation.

Tabel 2. Result value of gamma dose rate of HTGR-10 MWth with MCNPX

Position (cm)	Gamma dose rate (mSv/hour)	Item
-313	3.0052E+05	Core
-220.5	8.9770E+04	Reflector
0	5.0952E+00	Initial shielding
15	2.5900E+00	Shielding
35	1.0517E+00	Shielding
55	5.8104E-01	Shielding
75	3.5484E-01	Shielding
95	2.1196E-01	Shielding
115	0.0000E+00	Shielding
135	0.0000E+00	Shielding
155	0.0000E+00	Shielding
175	0.0000E+00	Shielding
195	0.0000E+00	Shielding

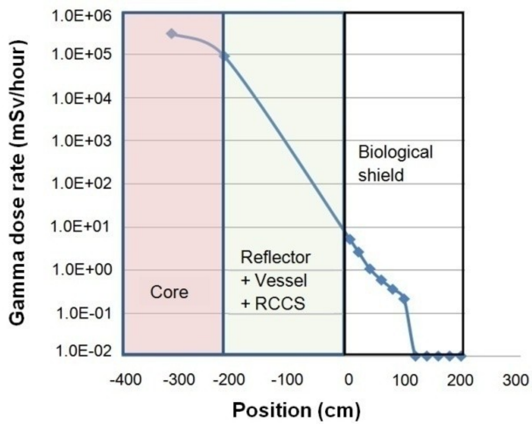

Figure 7. Relation of gamma dose rate in the core and the thickness of shielding the HTGR-10 MWth

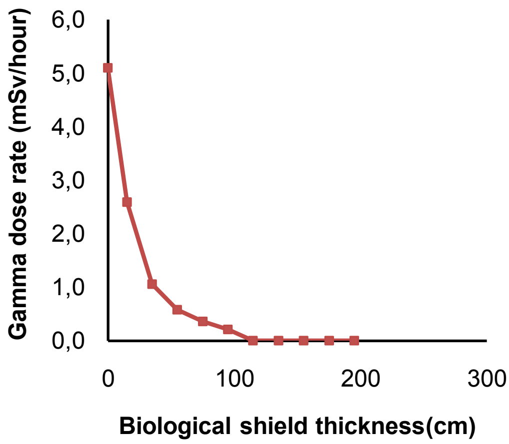

Figure 8. Relationship of gamma dose rate against shield the thickness of HTGR-10 MWth

From Figure 8, it can be seen that gamma doses along biological shielding decrease exponentially due to attenuation by shielding material. Based on the results of calculations using the MCNPX program package the gamma dose value becomes zero at a shielding thickness of 115 cm.

To simplify the calculation of the thickness of the biological shield needs to be made illustrations with semi-logaritmic scale that can be seen in Figure 9.

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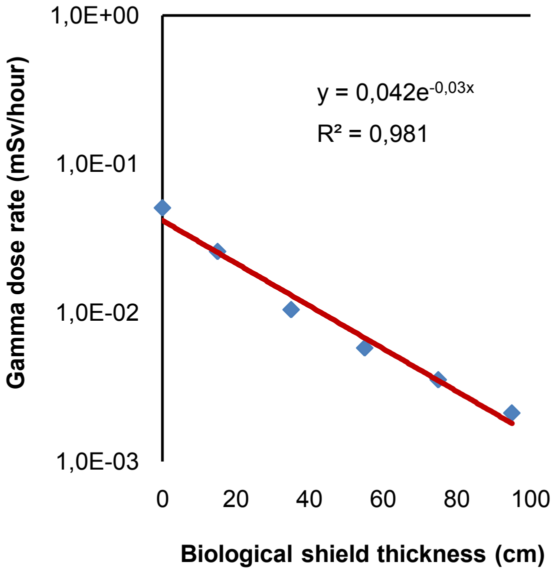

Figure 9. Semi-logarithmic relationship of gamma dose rate against biological shield thickness at HTGR-10 MWth

From the calculation of gamma radiation dose rate by using MCNP6 illustrated in Figure 7 it appears that at the center of the core HTGR-10 MWth, gamma radiation dose rate has a very large value that 3.0052E+05 mSv/h. This is due to the nuclear reaction in the central core will generating gamma radiation as a result of collisions of neutrons to the fuel pebble. By providing a radiation shield made of regular concrete density of 2.3 g/cm^3 gamma radiation dose rate has decreased following the exponential equation, namely 0.0042 e$^{-0,03x}$. The dose radiation reaches 0 mSv/h at 115 cm of a thickness of biological shielding. Thus the effective thickness of the biological shield to get the dose rate that is safe for workers is 115 cm. Based on Bapeten Chairman's Regulation No. 4 year of 2013 received a radiation dose permitted for radiation workers is 20 mSv per year which is equivalent to 10 μSv/h or 0,010 mSv/h. Compared with the results of the dose rate generated by MCNP6 then the condition will be achieved by applied a biological shield with a thickness ranging from 100 up to 115 cm.

CONCLUSION

HTGR-10MWth models created using MCNP6 program package worth continuing because its benchmarked with the HTR-10 have a very small error rate is ±1.1327%. for critically core height. The results of calculations performed for the MCNP6 high HTGR-10 MWth critical core with pebble fuel ratio to pebble moderator at 52:48 occurs at a height of 134 cm. At the center of the terrace has a gamma dose level of 3.0052E+05 mSv/h, while in the biological shield is made of ordinary concrete with a density of 2.3 g/cm^3 is at the outer circumference of the HTGR10 MWth core has a gamma dose level decreased following equation y = 0.0042$^{e-0.03x}$. By following Bapeten Perka No. 4 of 2013, the safety limit for workers will be achieved with a minimum thickness of a biological shield by 115 cm.

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