Neutron dose rate analysis on HTGR-10 reactor using Monte Carlo code

Suwoto, H. Adrial, A. Hamzah, Zuhair, S. Bakhri, G. R. Sunaryo

Center for Nuclear Reactor Technology and Safety – National Nuclear Energy Agency of Indonesia (BATAN), PUPSPIPETEK Complex, Office Building No. 80, Serpong, Tangerang Selatan 15310, Indonesia, Telp. (021)756-0912, Fax. (021)756-0913, E-mail: suwoto@batan.go.id

Abstract. The HTGR-10 reactor is cylinder-shaped core fuelled with kernel TRISO coated fuel particles in the spherical pebble with helium cooling system. The outlet helium gas coolant temperature outputted from the reactor core is designed to 700 °C. One advantage HTGR type reactor is capable of co-generation, as an addition to generating electricity, the reactor was designed to produce heat at high temperature can be used for other processes. The spherical fuel pebble contains 8335 TRISO UO 2 kernel coated particles with enrichment of 10% and 17% are dispersed in a graphite matrix. The main purpose of this study was to analysis the distribution of neutron dose rates generated from HTGR-10 reactors. The calculation and analysis result of neutron dose rate in the HTGR-10 reactor core was performed using Monte Carlo MCNP5v1.6 code. The problems of double heterogeneity in kernel fuel coated particles TRISO and spherical fuel pebble in the HTGR-10 core are modelled well with MCNP5v1.6 code. The neutron flux to dose conversion factors taken from the International Commission on Radiological Protection (ICRP-74) was used to determine the dose rate that passes through the active core, reflectors, core barrel, reactor pressure vessel (RPV) and a biological shield. The calculated results of neutron dose rate with MCNP5v1.6 code using a conversion factor of ICRP-74 (2009) for radiation workers in the radial direction on the outside of the RPV (radial position = 220 cm from the center of the patio HTGR-10) provides the respective value of 9.22E-4 μSv/h and 9.58E-4 μSv/h for enrichment 10% and 17%, respectively. The calculated values of neutron dose rates are compliant with BAPETEN Chairman's Regulation Number 4 Year 2013 on Radiation Protection and Safety in Nuclear Energy Utilization which sets the limit value for the average effective dose for radiation workers 20 mSv/year or 10μSv/h. Thus the protection and safety for radiation workers to be safe from the radiation source has been fulfilled. From the result analysis, it can be concluded that the model of calculation result of neutron dose rate for HTGR-10 core has met the required radiation safety standards.

Keywords: HTGR-10, neutron dose rate, MCNP5v1.6, ICRP-74.

1. Introduction
Nuclear energy is seen as the best candidate in the future as a potential energy sources to meet energy demand. Safety is one of the problems facing the nuclear reactor technology that is most crucial at this
time, especially after the Fukushima incident. Fear of accidents due to human error or due to natural disasters cannot be avoided, but the high temperature gas cooled reactor (HTGR) as one type of Generation IV reactor is designed to have a mechanism of passive safety in its operations, and able to cover the problems of safety concern.

The Indonesia Experimental Power Reactor (HTGR-10)[1] designed by RENUKO (Rekayasa Engineering-Nukem-Kogas) is one type of HTGR that will soon be built in PUSPIPTEK area, Serpong Indonesia. The HTGR-10 refers and adopt the design of HTGR technology coming from Germany which applied to Chinese HTR-10 reactor[2,3,4]. The fuel of HTGR-10 reactor is designed using UO$_2$ TRISO coated fuel particles[5] which formed in the spherical fuel ball called pebble. Theoretically, the HTGR-10 reactor core can use the various fuel of kernel TRISO coated fuel particles such as uranium dioxide (UO$_2$)[6,7], thorium oxide (ThO$_2$/UO$_2$)[8,9,10] or plutonium dioxide (PuO$_2$)[11,12] without core shape and geometry changing. However, in this study only discusses fuel of uranium dioxide with two differences enrichment i.e. 10% and 17% of U$_{235}$. The outputted thermal power of HTGR-10 reactor is about 10 MW and outlet temperatures around 700°C and expected to generate electricity up to 3 MWe. In addition, the HTGR-10 reactor expected can be used as co-generation plant such as sea water desalination and another industrial processes using high temperature application like hydrogen production using thermo-chemical water splitting system and coal liquefaction.

The HTGR-10 is 10 MWth reactor should be met with the operation standard safety limits of a nuclear reactor, so that the research on calculation of neutron dose rate is very important related to design of radiation shielding. The technical safety limit parameter of HTGR reactor should be met with the operation safety limit is also intended to safety limit of radiological aspect[13]. Radiological safety limit values are required for the HTGR-10 reactor include the dose rate of neutron radiation produced by the reactor core. The safety limit values should not be exceeded for safety purposes for radiation workers, communities and the surrounding environment.

The kernel of TRISO coated fuel particles, spherical fuel balls (pebble) and modeling of reactor core include reflector, radiation and radiological shielding of HTGR-10 core are fully 3-D modeled using Monte Carlo MCNP5v1.6 code. Tally card utilization on the Monte Carlo MCNP5v1.6 code in modeling of radiation sources, the modification to the desired output by adding normalized tally of neutron source strength and energy tally (DE, Dose Energy card) is performed. Tally is a language MCNP to mention the physical desired quantities. The addition of this tally will provide the output of the energy spectrum of the radiation source. All calculations are done on initial core of HTGR-10 that use fresh fuel. Status of previous studies have been calculated the neutron dose rate on the reactor type of pebble-bed HTGR 200 MWth (RGTT200K)[14] and analysis of neutron source strength and neutron dose rate calculation of RDE initial core[15] with results in accordance with BAPETEN Chairman's Regulation Number 4 Year 2013 on Radiation Protection and Safety in Nuclear Energy Utilization.

Monte Carlo MCNP5v1.6 code is a powerful probabilistic method code to simulate particles trip of neutrons, photons, electrons and others. This code has been developed by Monte Carlo X-5 team from LANL (Los Alamos National Laboratory) - United States. To perform the analysis of neutron dose rate calculation on the HTGR-10 reactor, Monte Carlo MCNP5v1.6 code with continuous neutron cross-section energy library (ACE files) taken from ENDF/B-VII file was used extensively in this research. The neutron energy group structure (EGS) in accordance with HTGR type reactor spectrum that can be made using EGS99304 utility code. The coated fuel particle with TRISO layered, spherical pebble fuel, the reactor core, control rod channel, cooling helium gas channel and a reflector and radiation and biological shielding[16] dimension and geometry are inputted as an input parameter.

The neutron dose rate calculation of using Tally Flux Detector (F5) combining of Tally Dose (DE) and Tally Energy Df(E) which is used to convert from the flux to dose with certain factors in accordance with the amount of conversion factors derived from the reference of ICRP-74 (2009)[17].
The addition of this tally will provide the output of the dose rate (Sv/h or rem/h) generated at a certain position on the geometry of the HTGR-10 core.

The main purpose of this research was to determine neutron dose rates distribution on HTGR-10 reactors. Furthermore the results of this research can be used to design of the radiation and the biological shielding thickness. Therefore the distribution of neutron radiation that spread to the whole area around the reactor can be well predicted. To meet the radiation safety requirements, especially radiation workers, then around the reactor core is installed safe radiation and biological shielding. The reactor operation meet of the radiation safety standards required by applicable law is fulfilled.

2. Description of Fuel And Reactor Core

Brief illustration of the fuel and reactor core of HTGR-10 is follow the geometry of the HTR-10 reactor in China. Schematic and technical data of the TRISO coated fuel particle of uranium dioxide (UO$_2$) as given in figure 1 and table 1.

![Illustration of fuel and core used in the HTGR-10 reactor](image)

**Figure 1.** Illustration of fuel and core used in the HTGR-10 reactor[18]

**Table 1.** Core and fuel data parameter of HTGR-10 reactor[1,18,19]

| PARAMETER                                              | Value |
|--------------------------------------------------------|-------|
| REACTOR CORE                                           |       |
| Thermal Reactor Power                                  | 10    |
| Equivalent Core Diameter, cm                           | 180   |
| Equivalent Core High, cm                               | 197   |
| Reactor Core Volume, cm$^3$                            | 5.00  |
| Volumetric filling fraction of balls in the core       | 0.61  |
| Height of the empty cavity above the pebble bed, cm    | 41.7  |
| Diameter of fuel discharging tube, cm                  | 50    |
| FUEL ELEMENT                                           |       |
| SPHERICAL FUEL PEBBLE                                  |       |
| Diameter of ball, cm                                   | 6.00  |
- Diameter of fuel zone, cm.
  5.00
- Density of graphite in matrix and outer shell
  0.50
- Heavy metal (uranium) loading (weight) per ball, g
  5.00
- Density of graphite in matrix and outer shell, g.
  1.73

**TRISO Coated Fuel Particle**
- Fuel kernel material
  UO$_2$
- Kernel diameter, cm
  0.05
- Enrichment, % (U-235)
  17
- Kernel density, g/cm$^3$
  10.40

**Coating layer**
- Coating layer materials (starting from kernel)
  C/IPyC/SiC/OPyC
- Coating layer thickness, cm
  0.095/0.040/0.035/0.04
- Coating layer density (g/cm$^3$)
  1.1/1.90/1.38/1.90
- Kernel total diameter + TRISO coating layer, cm
  0.092

**SIDE REFLECTOR**
- Equivalent thickness of the side reflector graphite, cm
  77.8
- Equivalent thickness of the side boronated carbon brick, cm
  22.2
- Number of control rod channels
  10
- Diameter of the control rod channel & radial coordinate
  13 & 102
- Number of cold helium flow channels
  20
- Diameter of the helium flow channel & radial coordinate
  8 & 144.6

**CORE BARREL**
- Inner radius of core barrel, cm
  190
- Thickness of core barrel, cm
  5
- Base Stainless Steel Material
  SA516-70

**RPV (Reactor Pressure Vessel)**
- Inner radius of RPV, cm
  210
- Thickness of RPV, cm
  10
- Base Stainless Steel Material
  SA516-70

**BIOLOGICAL SHIELD[1]**
- Inner radius of biological shield, cm
  320
- Thickness of biological shield, cm
  250
- Base Material
  Concrete

---

**Table 2. Concrete wall specifications used in biological shielding[20]**

| Nuclide | MCNP-ID | Weight fraction | Atomic fraction | Atomic Density (atom/barn.cm$^2$) |
|---------|---------|-----------------|-----------------|----------------------------------|
| H       | 1001    | 0.022100        | 0.305330        | 0.030369                         |
| C       | 6000    | 0.002484        | 0.002880        | 0.000286                         |
| O       | 8016    | 0.574930        | 0.500407        | 0.049773                         |
| Na      | 11023   | 0.015208        | 0.009212        | 0.000916                         |
| Mg      | 12000   | 0.001266        | 0.000725        | 0.000072                         |
| Al      | 13027   | 0.019953        | 0.010298        | 0.001024                         |
| Si      | 14000   | 0.304627        | 0.151042        | 0.015023                         |
| K       | 19000   | 0.010045        | 0.003578        | 0.000356                         |
| Ca      | 20000   | 0.042951        | 0.014924        | 0.001484                         |
| Fe      | 26000   | 0.006435        | 0.001605        | 0.000160                         |
| Amount  |         | 0.999999        | 1.000000        | 9.9464E-02                       |
Material composition data in the biological shielding of 250 cm concrete thickness used in the research was taken from the National Institute of Standards and Technology (NIST) - the USA, as presented in table 2.

The average active height and diameter of HTGR-10 cylindrical reactor core are 197 cm and 180 cm, respectively. The reactor core divide into 10 zones in axial directions and 10 zones in axial directions, as presented in figure 2. The lower axial part in the core is zone number: 900, 901, 902, 903, 904, 905, 906, 907, 908 and 909. The center axial part in the core is zone number: 950, 951, 952, 953, 954, 955, 956, 957, 958 and 959. The top axial part in the core is zone number: 990, 991, 992, 993, 994, 995, 996, 997, 998 and 999.

![Figure 2. The geometry modelling of HTGR-10 core with MCNP5v1.6 code](image)

### 3. Calculation Methodology

TRISO kernel coated fuel particles and pebble fuel geometry modelling are involved packing fraction consideration[21]. Determination of the number of the group structure of the HTGR-10 reactor spectrum is done using EGS99304 utility code. MCNP5v1.6 code was used extensively in this research by utilizing of continuous energy cross-sectional data taken from ENDF/B-VII file.

Modelling of kernel TRISO coated fuel particles is shown in figure 3. TRISO coated fuel particles in the Monte Carlo MCNP/MCNPX code do with fully heterogeneity model on kernel TRISO i.e. kernel UO$_2$, high porosity carbon buffer layer, inner pyrolytic carbon layer, silicon carbide layer, outer pyrolytic carbon layer and graphite matrix. Spherical fuel zone containing of 8335 UO$_2$ kernel TRISO coated fuel particles with diameter 2.5cm using packing fraction of 5%. The spherical fuel zone covered with graphite outer shell of 0.5 cm thickness, thus finally, pebble fuel formed with a diameter of 6 cm is shown in figure 4.
Figure 3. Modelling of kernel TRISO coated fuel particle in Monte Carlo MCNP5v1.6 code
(a). Heterogeneous cells model of kernel TRISO coated fuel particles
(b). Kernel cell model at the center of a simple cubic (SC)

Figure 4. Heterogeneous modelling of kernel TRISO in the formation of spherical fuel pebble using Monte Carlo MCNP5v1.6 code

Spherical fuel (pebble) is then inserted into the reactor core with lattice model BCC (Body Centered Cubic), such that the packing fraction obtained by 61% (optimal value for HTGR with pebble fuel). The calculation of the radiation field by proper tallies of MCNP5v1.6 code, was carried out by calculating the neutron track length estimate over cell flux along the radial of the reactor core, using the F5 tally (particle flux at a ring detector). The detectors position in the reactor system are placed at the radius of 28.4605cm, 40.2492cm, 49.2950cm, 56.9210, 63.6396cm, 69.2950cm, 75.2994cm, 80.4984cm, 85.3815cm, 90cm, 95.60cm, 108.60cm, 148.60cm, 167.79cm, 190cm, 195cm, 210cm, 220cm, 320cm, 330cm, 340cm, 350cm, 360cm, 370cm, 380cm, 400cm, 410cm, 420cm, 430cm, 440cm, 450cm, 460cm, 470cm, 480cm, 490cm, 500cm, 510cm, 520cm, 530cm, 540cm, 550cm, 560cm and 570cm from the reactor center.

The calculation of neutron average neutron fluxes generated reactor core using the following equation:

\[
\Phi \left( \frac{\text{neutron}}{cm^2 \cdot sec} \right) = \text{Tally F5} \left( \Phi_{F5} \right) \left(1/cm^2\right) \times \text{NSS (neutron/sec)} \times \frac{1}{k_{\text{eff}}} \quad (1)
\]

where:
- \( \Phi \) is the result of the calculation output Monte Carlo MCNP code, \( (1/cm^2) \),
- \( k_{\text{eff}} \) is a constant value of neutron effective multiplication factor,
- NSS is Neutron Source Strength, neutron/sec.

Determination of the neutron dose rates produced by reactor core are calculated using equation (1), then converted into neutron dose rates with specific conversion factor based on the ICRP-74 (2009) reference as presented in table 3. Doses are derived from neutron fluxes obtained with detector F5 tallies by means of the Dose Energy card (DEn), and the Dose Function card (DFn).
Table 3. Conversion factor from neutron fluxes into neutron dose rates[23]

| Neutron energy (MeV) | DF(E) (mrem/h)/(n/cm^2.s) | Neutron Energy (MeV) | DF(E) (mrem/h)/(n/cm^2.s) |
|----------------------|--------------------------|----------------------|--------------------------|
| 1.00E-09             | 2.3760E-03               | 0.1500               | 4.7520E-02               |
| 1.00E-08             | 3.2400E-03               | 0.2000               | 6.1200E-02               |
| 2.53E-08             | 3.8160E-03               | 0.3000               | 8.3880E-02               |
| 1.00E-07             | 4.6440E-03               | 0.5000               | 1.1592E-01               |
| 2.00E-07             | 4.8600E-03               | 0.7000               | 1.3500E-01               |
| 5.00E-07             | 4.8960E-03               | 0.9000               | 1.4400E-01               |
| 1.00E-06             | 4.7880E-03               | 1.0000               | 1.4976E-01               |
| 2.00E-06             | 4.6440E-03               | 1.2000               | 1.5300E-01               |
| 5.00E-06             | 4.3200E-03               | 2.0000               | 1.5120E-01               |
| 1.00E-05             | 4.0680E-03               | 3.0000               | 1.4832E-01               |
| 2.00E-05             | 3.8160E-03               | 4.0000               | 1.4688E-01               |
| 5.00E-05             | 3.5640E-03               | 5.0000               | 1.4580E-01               |
| 1.00E-04             | 3.3840E-03               | 6.0000               | 1.4400E-01               |
| 2.00E-04             | 3.2040E-03               | 7.0000               | 1.4580E-01               |
| 5.00E-04             | 2.9880E-03               | 8.0000               | 1.4724E-01               |
| 1.00E-03             | 2.8440E-03               | 9.0000               | 1.5120E-01               |
| 0.0020               | 2.7720E-03               | 10.0000              | 1.5840E-01               |
| 0.0050               | 2.8800E-03               | 12.0000              | 1.7280E-01               |
| 0.0100               | 3.7800E-03               | 14.0000              | 1.8720E-01               |
| 0.0200               | 5.9760E-03               | 15.0000              | 1.9440E-01               |
| 0.0300               | 8.5320E-03               | 16.0000              | 1.9440E-01               |
| 0.0500               | 1.4796E-02               | 18.0000              | 2.0520E-01               |
| 0.0700               | 2.1600E-02               | 20.0000              | 2.1600E-01               |
| 0.1000               | 3.1680E-02               |                      |                          |

4. Results and Discussion

The calculation of the neutron dose rates are done using a combination of a couple of Tally F5 (particle flux at a ring detector) by Tally DE (Dose Energy Card) and Tally DF (Dose Function Card) which is used to convert from the neutron flux to dose. The conversion factors from neutron fluxes to neutron dose rates in accordance with the value of the conversion factor was taken from International Commission on Radiation Protection (International Commission on Radiological Protection, ICRP), the ICRP-74. With the pair combined tally DE and DF will provide the output of the dose rate (Sv/h or rem/h) generated at a certain position on the geometry of HTGR-10 core with the normalization to value of neutron source strength has been derived from it (as tally FM).

All calculations of neutron dose rates generated by HTGR-10 core is performed with UO2 fuel enriched 235U at 10% and 17%. The conversion factor of neutron flux to dose rates based on ICRP-74 (2009) reference is fully used.

The calculated results of neutron dose rate using Monte Carlo MCNP5v1.6 code equipped with tally detector cards (Tally F5) and conversion factors Flux-to-Dose multiplied by neutron source strength has been obtained. Neutron dose rates distribution on HTGR-10 core as shown in figure 5 for the enrichment of 10% and figure 6 for the enrichment of 17% are clearly observed. All calculations using MCNP5v1.6 with the conversion factor of the neutron flux in the neutron dose of ICRP-74 (2009) are performed.

The latest regulations from BAPETEN Chairman's Regulation Number 4 Year 2013 on Radiation Protection and Safety in Nuclear Energy Utilization, average dose limit value effective for radiation workers are set at 20 mSv (2000 mrem) per year or equivalent to 10 μSv/h.

From the analysis, it can be seen on figure 5 and figure 6 that neutron dose rates on the axial direction of the center part of the core has higher than those in the bottom and top of HTGR-10 core.
Using the conversion factor of ICRP-74 (2009), at a radial distance of 168 cm from the center of HTGR-10 core (the outermost part of reflector zone), neutron dose rate is weakened of 2.08 μSv/h and 2.44 μSv/h for 10% and 17% enrichment, respectively. Those both values are greatly met with requirements of BAPETEN Chairman’s Regulation Number 4 Year 2013.

Figure 5. The calculated results of neutron dose rate distribution on HTGR-10 core with UO₂ fuel enrichment 10% (²³⁵U) using MCNP5v1.6 code

Figure 6. The calculated results of neutron dose rate distribution on HTGR-10 core with UO₂ fuel enrichment 17% (²³⁵U) using MCNP5v1.6 code
The MCNP5v1.6 calculation results of the neutron dose rates at a distance of 220 cm from the center of core (outer part of reactor pressure vessel, RPV), the neutron dose rates value of 9.22E-4 μSv/h and 9.58E-4 μSv/h are obtained for enrichment of 10% and 17%, respectively.

The both calculation results of neutron dose rate calculations performed on biological shield of 250 cm thickness (radial distance from the core center of 570 cm) are clearly safe. In accordance with the MCNP5v1.6 calculations results on 0 cm thickness of the biological shielding, the value of neutron dose rates have been dropped up to 1.18E-06 μSv/h (enrichment 10%) and 1.33E-06 μSv/h (enrichment 17%) by using ICRP-74 (2009) flux-to-dose conversion factor.

Using a conversion factor flux-to-dose ICRP-74 (2009) used in compliance with the applicable requirements are BAPETEN Chairman's Regulation Number 4 Year 2013. So it can be concluded that the design of modelled biological shielding thickness is sufficient and safe for radiation workers, so that the worker is protected from nuclear radiation and can work safely in a nuclear reactor environment.

5. Conclusion
Overall calculation results of neutron dose rates on HTGR-10 core are performed with Monte Carlo MCNP5v1.6 code by using a conversion factor of neutron fluxes-to-dose dose rates based on ICRP-74 (2009) are clearly observed. The calculation analysis results for the neutron dose rates on HTGR-10 core outside the radial position of 220 cm from the center of HTGR-10 reactor core (outer part of RPV) are giving each value of 9.22E-4 μSv/h and 9.58E-4 μSv/h for enrichment UO₂ 10% and 17%, respectively. Those both values are greatly met with the requirements specified by BAPETEN Chairman's Regulation Number 4 Year 2013, so that the radiation workers have safe and protected from neutron radiation sources produced reactor core.

Acknowledgment
The author would like to thank Syaiful Bakhri Ph.D. as Head of Reactor Physics and Technology Division and Dr. Pande Made Udiyani as research coordinator, on suggestion and improvement of this paper. This study was fully funded by DIPA PTKRN-BATAN 2016.

References
[1] Tim Penyusun Dokumen Spesifikasi Teknis RDE-RENUKO, 2014. Dokumen Spesifikasi Teknis Reaktor Daya Eksperimental, Nomor Identifikasi Dokumen: DT.002.KRN.2014, Rev.0, tanggal: 20-Nov-2014, PTKRN-BATAN, Tangerang Selatan.
[2] Wang, M. J., et al., 2014. Criticality calculations of the HTR-10 pebble-bed reactor with SCALE6/CSAS6 and MCNP5, Annals of Nuclear Energy 64, 1–7.
[3] Chen, F., et al., 2009. Benchmark Calculation for the Steady-State Temperature Distribution of the HTR-10 under Full-Power Operation, Journal of Nuclear Science and Technology, Vol. 46, No. 6, p. 572–580.
[4] Rosales, J., et al., 2014. Computational Model for the Neutronic Simulation of Pebble Bed Reactor’s Core Using MCNPX, Hindawi Publishing Corporation, International Journal of Nuclear Energy, Volume 2014, Article ID 279073, 12 pages, http://dx.doi.org/10.1155/2014/279073.
[5] Kania, M. J., Nabielek, H. And Nickel, H., 2015. Coated Particle Fuels for High-Temperature Reactors, Materials Science and Technology. DOI: 10.1002/9783527603978.mst0449, p.1–183.
[6] Kania, M. J., Nabielek, H., Verfondern, K., Allelein, H.J., 2013. Testing of HTR UO₂ TRISO Fuels in AVR and in Material Test Reactors, Journal of Nuclear Materials 441, 545–562.
[7] Tran, H. N., Hoang, V. K., 2012. Neutronic characteristics of an OTTO refueling PBMR, Nuclear Engineering and Design 253, 269 – 276.
[8] Allelein, H.J., Kania, M. J., Nabielek, H., Verfondern, K., 2014. Thorium Fuel Performance Assessment in HTRs, Nuclear Engineering and Design 271, 166–170.
[9] Trinuruk, P., Obara, T., 2015. Particle-Type Burnable Poisons for Thorium-Based Fuel in HTGR, Energy Procedia 71, 22 – 32.
[10] Wols, F. J., Kloosterman, J. L., Lathouwers, D., van der Hagen, T.H.J.J, 2014. Core Design and Fuel Management Studies of a Thorium Breeder Pebble Bed High Temperature Reactor, Nucl. Technol., 186, pp. 1–16.
[11] Jonnet, J. J., Kloosterman, J. L., Boer, B., 2010. Performance Of TRISO Particles Fueled With Plutonium and Minor Actinides in A PBMR-400 Core Design, Nuclear Engineering and Design 240, 1320–1331.
[12] Acir, A., Coskun, H., 2012. Neutronic Analysis of The PBMR-400 Full Core Using Thorium Fuel Mixed With Plutonium or Minor Actinides Annals of Nuclear Energy 48, 45–50.
[13] Valentin, J. (Ed.), 2012. The 2007 Recommendations of the International Commission on Radiological Protection, p.98 (5.10. Dose Limits), Recommendation No. 244, Elsevier, submitted: March 29.
[14] S. Suwoto and Z. Zuhair, 2016. Analysis of Neutron Dose Rates on RGTT200K Core Using MCNP5, Jurnal Sains dan Teknologi Nuklir Indonesia 17 (2), 15.
[15] S. Suwoto, H. Adrial, Z. Zuhair. 2017. Analisys of Neutron Source Strength and Neutron Dose Rate Calculation of RDE Initial Core, Urania. Scientific Journal of Nuclear Fuel Cycle, 23(1), 33.
[16] Remetti, R., Andreoli, G., Keshishian, S., 2012. Monte Carlo Calculation of The Neutron Effective Dose Rate At The Outer Surface Of The Biological Shield Of HTR-10 Reactor, Nuclear Engineering and Design 243, 148–152.
[17] Ward, D. C., 2009. Impact of Switching To The ICRP-74 Neutron Flux-To-Dose Equivalent Rate Conversion Factors, at The Sandia National Laboratory Building 818 Neutron Source Range, SANDIA report, SAND2009-1144, March (2009).
[18] Seker V., Çolak U., 2003. HTR-10 full core first criticality analysis with MCNP, Nuclear Engineering and Design 222, 263–270.
[19] Terry, W. K., et al., 2006. Evaluation of the HTR-10 Reactor as a Benchmark for Physics Code QA, ANS Topical Meeting on Reactor Physics, PHYSOR.
[20] Mcconn, R.J., Jr., et al., 2011. Compendium of Material Composition Data for Radiation Transport Modeling, PIET-43741 -TM-963, PNNL-15870 Rev. 1, March 4.
[21] Türkmen, M., Çolak, U., 2012. Effect of Pebble Packing on Neutron Spectrum and the Isotopic Composition of HTGR Fuel, Annals of Nuclear Energy 46, 29–36.