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The use of MNCP-5 particle transport program for calculation of flux radiation exposure in object surface

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Abstract. Calculation of particle or photon flux in three dimensions is needed to facilitate the observation and the real approach of radiation to the observed object. Calculation of the flux of radiation received on an object's surface can be done using the MCNP-5 particle transport program (Monte Carlo N particle-5). In this study, the radiation source is a neutron particle emitted from the center of the ball with a radius of 0.5 cm with 16 MeV of energy on an object's surface of an iron ball with a radius of 0.5 cm. The source of neutrons and irradiated objects is in a cube vessel, with a side length of 10 cm and 20 cm. This simulation can be useful as a model for calculating the impact of radiation exposure on the surface of objects in human organs. Flux and the magnitude of the radiation particle's track on the object's surface are calculated based on changes in media material, media size and number of radiation particles (NPS). The results showed that the flux and particle track received by the surface of an object were greatly influenced by the media material and media size and the NPS value of the radiation particles.

1. Introduction

Accurate diagnosis in detecting early cancer in the human body's organs [1] needs to be followed up with a proper treatment of the disease. Before conducting the treatment, it is necessary to do preliminary research on simulations of radiation doses received by body's organs.

Radiation from a source that hits a particular object will provide an absorbed dose. The amount of dose absorbed or received depends on the type of radiation. It also depends on energy, source strength, source distance, and time of irradiation [2]. The radiation dose received can be determined in two ways, namely calculation and measurement. Measurements can be made using dosimeter measuring devices, whereas calculations are performed using statistical analytical and probabilistic methods. One probabilistic method that can be used is Monte Carlo N Particles (MCNP) [3].

Monte Carlo is a statistical numerical method. This method simulates random numbers to solve problems that are not possible to be solved analytically [2,4]. One of the computer programs based on the Monte Carlo method is MCNP. MCNP can simulate how the trip or the traces of electron particles, photons and neutrons in various types of three-dimensional material.

The first simulation application using Monte Carlo was to calculate the criticality of a nuclear bomb in the Manhattan project in the 1940s [5]. Since calculations using the simulation method were highly dependent on computer capabilities, at that time the use of a Monte Carlo-based computer

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program was initially less desirable. Since the development of computer technology that is quite rapid, simulation methods using Monte Carlo have also developed.

Making the MCNP input in principle is simple by filling in what is called a "card". There are three cards in the MCNP input, namely cell card, surface card, and data card. The cell card and the surface card are the geometrical inputs of the object to be simulated, while the data card is information about the material of the simulated object, the definition of the particle source, and the tally or physical quantity to be calculated. The purpose of this study is to illustrate how MCNP-5 handles the case of calculating fluxes and particle tracks on an object's surface where the material and size of the exposure media and the number of radiation particles are changed to other specifications for better purposes.

2. Method and materials

Particle simulation with MCNP starts from "born" then interacts with material until it ends in "dead area" as shown in Figure 1. This figure shows a particle from the source interacting with a material. The first interaction (z_1) is in the form of scattering, then the next interaction (z_2) produces photons and electrons which are scattered into the dead area. The photons produced interact with material (x_1) to produce neutrons and scattered photons, then absorbed by material (x_2) . Furthermore, the neutrons that occur scattered (y_1) then enter the dead area [6].



Figure 1. The principle of particle simulation in MCNP [7].

In addition to being able to simulate electron, neutron and photon particles separately, MCNP can also simulate the three particles simultaneously, as shown in the example in Figure 1. From particle simulation, MCNP can provide output in the form of flux, energy, pulses, etc. To do a simulation using MCNP, there are three stages that are passed, namely making input, running, and interpreting the output [7]. In this study, the simulation was carried out using the MCNP-5 program.

2.1. Concept of cross-section

The quantitative description of nuclear interactions requires a known cross-section of the neutron, that is the area of the cross-section of the nucleus as seen in the approaching neutron. Transverse cross-section, their energy dependence, and the relative probability of a neutron will undergo scattering, capture or division are the basis of physical data that form the basis of the residual properties of a chain reaction [8]. The neutron reaction rate R [1/s cm²], in the target material proportional to the intensity of the incident neutron beam. I $[1/s \text{ cm}^2]$ or neutron flux in the case of the neutron field and the number of target atoms N [1/cm³]. The reaction rate unit implies that the proportionality constant must have a unit of area [cm²] and is marked as the probability that a reaction will occur and is called a nuclear cross section [9] so that the reaction rate is defined as: $R = \sigma \times I \times N$, the intensity of a neutron beam is represented by the number of neutrons in a volume unit (n) and the velocity (v) and $I = n \times v$. The number of neutrons colliding with the nuclei in the target material is proportional to the average intensity of the neutron beam and the number of nuclei in the target material [10] which is defined as the number of nuclei in the target = $N \times V$, which is the number of neutron collisions per second in the target material = $I \times N$. As an effective cross-sectional area, I is representing the number of colliding neutrons with a single target nucleus per unit time in the target [10,11].

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A cube of carbon containing carbon with 10 cm side. Inside the vessel there are two balls each with a diameter of 1 cm with coordinate position of ball 1 (0, -4 -2.5) and ball 2 (0, 4, 4) (Figure 2). Ball 1 contains oxygen, and in it there is a radiation source with 16 MeV energy. The radiation source is right at the center of the ball. Ball 2 contains iron material which is also contained in the same cube. Radiation exposure from ball 1 to ball 2. In ball 2 the magnitude of flux and the track of flux on the surface of ball 2 are calculated. In this model the material of the media cube is changed from carbon to oxygen, the size of the cube is also changed from the side of 10 cm to 20 cm and the number of particles to be simulated (NPS) is changed from 10^5 to 10^6 .



Figure 2. The Computational results of physical modeling with MCNP-5

3. Results and Discussions

3.1. Parameter of cubes side size

The results of simulation calculations between particle energy and the average flux size (Tally 2) are shown in Figure 3. In this simulation the cube side parameters are changed from 10 cm to 20 cm. For a side size of 10 cm a simulation result such as Figure 3(a) is obtained and a side size of 20 cm a simulation result such as Figure 3(b) is obtained. The difference between the two graphs of the simulation results is clear, when the size of the cube is reduced, the average value of the flux across the surface of the ball 2 per 1 cm² area increases. While the value of the particle track to the surface of the ball 2, due to changes in the value of the side of the cube from 10 cm to 20 cm, it showed an increase of the value of the particle track as seen in Figure 4.

3.2. Parameter of cube material

Simulation results of calculations between the average energy of particles with a length of particle track every 1 cm^2 on the surface of the ball 2 are shown in Figures 5 and 6. In this case, the contents of the cube material or media material are changed from carbon to oxygen. In Figure 5, the amount of flux is lower and there are fewer types of energy levels on the surface of ball 2 when the cube contains oxygen [Figure 5(b)] compared to Figure 5(a) where the cube contains Carbon.

For the number of particle tracks, the cube containing oxygen has a lower number of particle tracks [Figure 6 (b)] and there are fewer kinds of energy levels on the surface of ball 2 than the cube containing Carbon [Figure 6 (a)].

3.3. Parameter of number of particles probability

The simulation results of the calculation between the particle energy with the average flux size (Tally 2) are shown in Figure 7. The simulation results with NPS = 100,000 are indicated in the graph of Figure 7(a), and for the NPS = 1000,000 is illustrated in a graph of Figure 7(b). Based on these two graphs, it can be seen that the increment of NPS value causes significant increment of the value of the average flux of particles.

When the NPS increases, particle fluxes appear with more energy levels such as Figure 8(b), and the particle tracks appears larger at NPS=1000,000 [Figure 8(b)] compared to NPS = 100,000 [Figure 8(a)].



Figure 3. The average flux size of the particles due to the parameter of cubes side size.







(b). The cube side size of 20 cm

Figure 4. The average track of particle flux due to parameter of cubes side size.



(a). The media material containing carbon

(b). The media material containing oxygen

Figure 5. The average flux size of the particles due to the parameter of cube/media material.



(a). The media material containing carbon

(b). The media material containing oxygen

Figure 6. The average track of particle flux due to parameter of cube/media material.



Figure 7. The average flux size of the particles due to the parameter of NPS.



Figure 8. The average track of particle flux due to parameter of NPS.

4. Conclusion

The MCNP-5 program can simulate the changes in the parameters of media size, media material and the number of radiation particles to the average amount of flux and the track of particles received by the object surface. The average flux size and the track of the particles received by the surface of an object are greatly influenced by the media material, media size and NPS value of the radiation particles.

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References

- [1] Pandiangan, T., Bali, I. and Silalahi, A.R.J., 2019 Atom Indonesia 45-1-9.
- [2] Neural Network IAEA, 2014 Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, Interim Edition. General Safety Requirements Part 3. No. GSR Part 3. (IAEA, Vienna).
- [3] Hughes, H.G. and James, M.R., 2011 International Conference on Nuclear Criticality (Edinburgh, Scotland).
- [4] X-5 MONTE CARLO TEAM, 2003 MCNP-A General Monte Carlo N-Particle Transport Code, Version 5. Volume 1: Overview and Theory, LA-UR-03-1987, (Los Alamos National Laboratory, New Mexico).
- [5] X-5 Monte Carlo Team, 2003 MCNP-A General Monte Carlo N-Particle Transport Code, Version 5. Volume II: User's Guide, LA-UR-03-1987, (Los Alamos National Laboratory, New Mexico).
- [6] David, P. and Gierga, B.S., 1996 *Electron Photon Calculations using MCNP* (Nuclear Engineering Rensselaer Polytechnic Institute).
- [7] X-5 Monte C. Team: Briesmeister F.J. (Hrsg.) 2008 *General Monte Carlo N-Particle Transport Code*, Version 5. (Los Alamos National Laboratory, New Mexico).
- [8] Lewis, E.E., 2008 Fundamentals of Nuclear Reactor Physics (Academic Press).
- [9] Duderstadt, J. J. and Hamilton, L.J., 1976 Nuclear Reactor Analysis (John Wiley & Sons).
- [10] Lewis, E.E., 1993 Computational Methods of Neutron Transport (Wiley-Interscience).
- [11] Shultis, J.K. and Faw, R.E., 2002 *Fundamentals of Nuclear Science and Engineering* (Kansas State University).