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Internal Dose Analysis for Radiation Worker in Cancer Therapy Based on Boron Neutron Capture Therapy with Neutron Source Cyclotron 30 MeV Using Monte Carlo N Particle Extended Simulator

Aulia Setyo Wicaksono[1], Andang Widiharto[2], Yohannes Sardjono[3]
[1,2] Departemen Teknik Nuklir Teknik Fisika FT UGM
Jln. Grafika 2 Yogyakarta 55281 INDONESIA
[3]Pusat Sains dan Teknologi Akselerator (PSTA-BATAN)
Jln. Babarsari Kotak Pos 6101 ykbb Yogyakarta 55281
[1]aulia.setyo.w@mail.ugm.ac.id
[2]andang.widiharto@mail.ugm.ac.id
[3]sardjono.batan@gmail.com

Abstract Based Studies were carried out to analyze internal dose for radiation worker at Boron Neutron Capture Therapy (BNCT) facility base on Cyclotron 30 MeV with BSA and room that actually design before. This internal dose analyze include interaction between neutron and air. The air contains N2 (72%), O_2 (20%), Ar (0.93%), CO_2, Neon, Kripton, Xenon, Helium and Methane. That internal dose to the worker should be bellow limit dose for radiation worker amount of 20 mSv/years. From the particle that are present in the air, only Nitrogen and Argon can change into radioactive element. Nitrogen-14 activated to Carbon-14, Nitrogen-15 activated to Nitrogen-16, and Argon-40 activated to Argon-41. Calculation using tally facility in Monte Carlo N Particle Version Extended (MCNPX) program for calculated flux Neutron in the air $3{,}16 \times 10^7$ Neutron/cm^2s. room design in cancer facility have a measurement of length 200 cm, width 200 cm and high 166,40 cm. flux neutron can be used to calculated the reaction rate which is $80{,}1 \times 10^{-2}$ reaction/cm^3s for carbon-14 and $8{,}75 \times 10^{-5}$ reaction/cm^3s. Internal dose exposed to the radiation worker is 9.08E-9 µSv.

Keywords : Internal Dose, Radiation Worker, BNCT, MCNPX, BSA.

INTRODUCTION

A. Background

Radiation is radiant energy. Radiation divided into two types, that is ionizing radiation and non-ionizing radiation. Ionizing radiation is radiation that can ionize atoms or molecules at the area in their path. While non-ionizing radiation can not ionize the area in their path. Non-ionizing radiation has a wavelength of about 10 nm or more, an example of this radiation is radio waves, microwaves, visible light, and ultraviolet light. Ionizing radiation has wavelengths below 10 nm, for example, is the X-rays and gamma rays [1].

Someone can receive radiation dose from natural sources or from man-made radiation. In a report published in 2000, UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation), expressed as the average man will receive a dose of 2.8 mSv (280 mrem) per year. Approximately 14.5% of the dose received about a man comes from artificial radiation. Approximately 43% of the total dose received by a person coming from radon radionuclides present in the house. Radiation from radon gas is the main source of radiation that we receive everyday. This happens because 222Rn added to the air we breathe. Then, the radon gas that emits alpha radiation can irradiated lungs that would increase the risk of cancer. The average effective dose from radon gas is about 1.2 mSv (120 mrem) per year. [2].

Cancer is a cell that is not needed by the body and grow uncontrolled. There are more than 100 different types of cancer, and are classified based on the type of cell carriers. Cancer detrimental to the body because as the cells divide uncontrollably to form lumps or so-called tumors. Tumors can grow and interfere with the digestive system, the nervous, the circulatory and release hormones that alter body function [3]. The World Health Organization (World Health Organization) states that 8.2 million people died from cancer in 2012. Lung cancer is a contributor to deaths in the amount of 1.6 million [4]. In addition, data generated from the registration Jakarta Cancer shows that the top ten cancers, especially in women is breast cancer in 2005-2007 (18.6 cases many of the 100,000 population), cervical cancer (9.25 out of 2.1 million), ovarian cancer (4.27 100,000), colorectal cancer (3.15 100,000), cancer of the bronchus and lung (2.40 100,000), thyroid cancer (2.21 100,000), corpus cancer utery (1.76 from 100,000), cancer pharingeal (1.72 100,000), leukemia (1.61 100,000) and liver cancer (1.41 100,000) [5].

Radiotherapy use X-rays directly on target to destroy cancer cells and minimize the impact of radiation on healthy cells. Old radiotherapy treatment depends on several factors, such as location, type and stage of cancer, and whether radiotherapy is the treatment of single or combined with other types of cancer treatment, such as chemotherapy or surgery. Radiotherapy can be applied to treat cancer in many parts of the body. Radiotherapy treatment depending on the purpose, some people only receive one radiotherapy, while others may receive radiotherapy daily for one to eight weeks. Radiotherapy can be done once a day (but not always), five times a week and will take a few minutes each time therapy [6].

BNCT research conducted utilizing a 30 MeV Cyclotron neutron source with a beam shaping assembly (BSA). BSA mounted within a collimator for directing radiation to a room that is used for in vitro assays and in vivo. This room requires an analysis of internal dose for radiation workers in accordance with the applicable provisions that is provisions Perka BAPETEN No. 4 of 2013 on the value of the dose limit for radiation workers.

Research on analysis of internal dose in cancer therapy facility with a source of energy 30 MeV neutrons are held due to the lack of research on internal dose calculations beforehand. BNCT research conducted utilizing a 30 MeV Cyclotron neutron source with a BSA. BSA mounted within a collimator for directing radiation to a room that is used for in vitro assays and in vivo. This room requires an analysis of internal dose for radiation workers in accordance with the applicable provisions that is provisions Perka BAPETEN No. 4 of 2013 on the value of the dose limit for radiation workers.

B. Literature review

Boron Neutron Capture Therapy (BNCT)

Boron Neutron Capture Therapy (BNCT) is a technique for cancer therapy that has been designed to achieve selective basis. This technique was introduced by G.L. Locher in 1936 shortly after the discovery of the neutron by Chadwick in 1932 and described the reaction 10B (n, α) 7Li by Taylor and Goldhaber in 1935. [7]

In 1951 the first to demonstrate that certain boron compounds will allow a higher boron concentration in human brain tumor cells as compared to normal brain tissue [8]. During 1950-1960 in the Brookhaven Medical Research Reactor [9, 10] and the Massachusetts Institute of Technology Research Reactor [11] conducted the first clinical trials. Unfortunately, this trial failed to show evidence of the effectiveness of this method of therapy.

BNCT offer new possibilities for more effective cancer treatment. BNCT is a form of radiation therapy using non-radioactive nuclide 10B to capture neutrons that produce nuclear reaction 10B (n, α) 7Li. The products of this reaction has a characteristic linear energy transfer (LET) is high (approximately 150 keVμm particles of α-1, about 175 keVμm 7Li nucleus-1). The length of the trail of these particles in the air or tissue in the range from 4.5 to 10 μm; so the energy deposition is limited to the diameter of a single cell. Theoretically possible to mengiradiasi tumor cells selectively which already contains a number of 10B without the surrounding healthy cells [12]. Nuclear reactions taking place can be seen in Figure 2.1

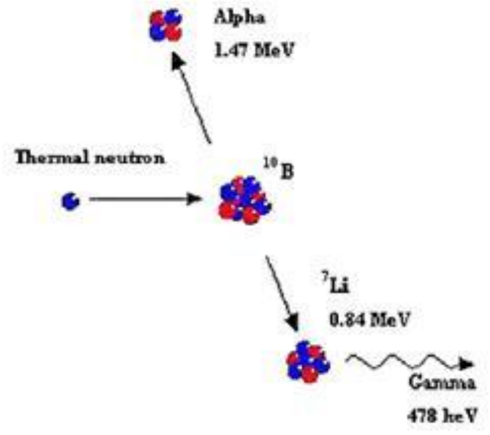

Figure 2.1. Schematic interaction between Boron and Neutron [20].

Arief Fauzi has done modeling radiation shields for the facility Boron Neutron Capture Therapy (BNCT) based 30 MeV Cyclotron with pre-designed BSA. Modeling include the selection of materials and determining the thickness of the shield. The shield is required to be able to withstand the radiation emanating from the room so that the dose of radiation leak is below the threshold dose for radiation workers is 20 mSv / year. The materials tested were paraffin, barite concrete, and borated polyethylene. Calculations using the facilities in the program tally Monte Carlo N Particle Extended version (MCNPX) to determine the leak with a threshold dose rate 10 μSv / h. Room modeling in this study is used as a reference for the calculation of internal dose [13].

MATERIAL AND METHOD

Radiation Interaction with Matter

Radiation is an emission and the propagation of energy through matter or space in the form of electromagnetic waves or particles decay of a radioactive substance emits charged particles. Charged particle radiation can be detected by utilizing the interaction of radiation with matter [14].

Neutron Interaction with Matter

Neutron interactions with matter can be broadly divided into two, that is scattering or absorption. A more detailed description of the various reactions is described as follows [15].

A. Scattering

Scattering divided into elastic scattering and inelastic scattering. Neutron strikes an atomic nucleus that is almost always in a state of quiescent or ground state. That is currently has the lowest energy of atoms in normal circumstances. Neutron out of the core and the core in a state leaving its ground state (not experiencing excitation). The Neutron has undergone elastic scattering by the atomic nuclei because of the state of the system unchanged. Notation nuclear reaction on this interaction can be written with the symbol (n, n).

Whereas inelastic scattering is almost the same as the elastic scattering, the difference in inelastic scattering of atomic nuclei that have been pounded by neutrons are in an excited state, the atom has more energy than the energy during its ground state. The collision energy is stored in the nucleus so that this interaction endothermic. Scattering inelatis symbolized by (n, n ').

B. Capture / Radiation Absorption

Catch reaction/ absorption is the interaction of atomic nuclei which absorb neutrons then transmit one or more so-called capture gamma rays γ-rays. This event is symbolized by the exothermic interactions (n, γ). For radiation absorption symbolized (n,α).

Gamma Interaction with Matter

γ-rays or can be called photons produced by the source of unstable nuclides. There are many possibilities of interaction of photons with matter, but only three important interactions that need to be considered in the measurement of radiation is the photoelectric effect, Compton scattering and pair production [1].

A. Photoelectric Effect

The photoelectric effect is the interaction of photons with a strong bound electrons in atoms that is electrons in the inner shell of an atom, usually leather K or L. Photon will pound these electrons and the electrons that bind strongly because the electrons will absorb the entire energy photons. As a result electrons are emitted out of the atom with kinetic energy equal to the difference energy photons and electrons connective power [16]

B. Compton Scattering

Compton scattering occurs when X-rays unit of energy in the event of light deflected from its original path by an interaction with an electron. The electrons are rejected, excluded from the position of the orbital and the amount of energy depends on the corners or corner in not spread and the nature of the medium that spreads, because the unit energy X-rays in a diffuse light has less energy, and wavelengths longer and fewer penetration compared to the unit of energy in light.

C. Pair Production

Pair Production an interaction between a photon and an atomic nucleus. Pair production can only occur if the incident photon energy is worth 1.022 MeV or more. Energy photon is absorbed completely, resulting photons turn into electron-positron pair. The rest mass of each particle electron and positron equivalent to 0.511

MeV. Positron engaged in medium continuously lose energy due to collisions with atomic electrons. At the end of its trajectory, the positron will recombine with electrons and both suffered annihilasi, then the two photons emerge with total energy $2mc^2$ [1].

Neutron Activation by Material

Neutron activation (fast) to 14N into C14 with t1 / 2 = 5568 years and emits γ-radiation with an energy of 0.6 MeV [22]. Fast neutrons in the thermal column is so small that the possibility of neutron activation to 14N is very low. Neutron activation to oxygen, producing nitrogen-16 with a half-life is very short that is 7.1 seconds so it will be quickly exhausted. An abundance of argon in the air 40 Ar 18 (99.6%), neutron activation will occur to produce radioactive isotopes 41Ar 18, emits radiation-γ rays with energies 1.2936 MeV and a half-life of 1.83 hours [21].

MCNPX Program (Monte Carlo N-Particle Extended)

MCNPX program is a particle transport simulation techniques included in the theoretical experimentation developed at Los Alamos National Laboratory (US). In the field of science kenukliran, the Monte Carlo method is packaged in a computer code, one of the most widely used is the Monte Carlo N-Particle (MCNP). Computer code Monte Carlo N-Particle (MCNP) is a code of transport of particles by the three-dimensional geometry and capability modeling of sources that can be applied on reactor physics, shield criticality, cleanup of nuclear waste in the environment, medical imaging, and some other related fields [17].

Monte Carlo methods have kemamapuan to perform simulations in various methods of neutron, photon, neutron-photon, neutron-electron, or neutron-photon-electron formulated in the format of the input code. While MCNPX an MCNP code designed to simulate particles with a wide energy range. MCNPX 2.6.0 has new capabilities, especially the phenomenon of transmutation, burnup and kasep particle production. Some new tally sourcing options and variance-reduction has also been added. Improvement of the physical aspect is the new version of Cascade-exciton model (CEM), the addition of a large selection of Los Alamos gluon String Model (LAQGSM) and a substantial increase into a muon physics [18].

Internal Dose

Internal dose calculations performed by summing the radiation is absorbed in various target tissues, organs derived from a number of sources with a significant number of radionuclides. MIRD scheme is the most common method used for the calculation of internal dose estimates.

The Regulation of BAPETEN

Dose constraints used as a comparison radiation doses received by workers based on the Regulation of the Nuclear Energy Supervisory Agency (BAPETEN) number 4 of 2013 Article 15 [19].

IMPLEMENTATION OF RESEARCH

This study begins by gathering information and data related to the issues to be discussed that is looking for internal dose through library of papers, theses, books, and from the internet.In solving the problems, this study is divided into several sections, that is;

Usage in the Manufacture Code MCNPX

The use MCNPX program used in this study to determine the neutron flux in the air. Powerful source of protons in the BSA of 1 mA to calculate the neutron dose used tally F4. A strong source of proton is converted to particles per second by means multiplied by $1\,x\,10^{-3}\,c/s$ so the multiplayer factors Be neutron target for dose calculation.

$$6{,}24\,x\,10^{15}\,p/s \qquad (4.3)$$

1. Identify the Problem

The aim of this study as a support of the radiation safety system. The paper considers the worst that will happen when all the existing safety system can't function properly.

2. Plan Analysis Results

Use of MCPX room made with a length of 200 cm, width 200 cm, height 166.4 cm. The design of this room is tailored to the needs of BATAN for in vitro assays and in vivo. Then the results obtained neutron flux in the air. Having obtained the neutron flux we can calculate the rate of reaction by calculating the cross-sectional and the density of atoms. Final results are expected in this study is that we were able to calculate how long the radiation workers to be in the treatment room.

RESULTS AND DISCUSSION

Figure 5.2. below to clarify the description above :

Beam Shaping Assembly Modeling

Beam shaping assembly (BSA) were used in this study is modeling the BSA is being designed by Prayoga with such specifications can be seen in Figure 5.1.

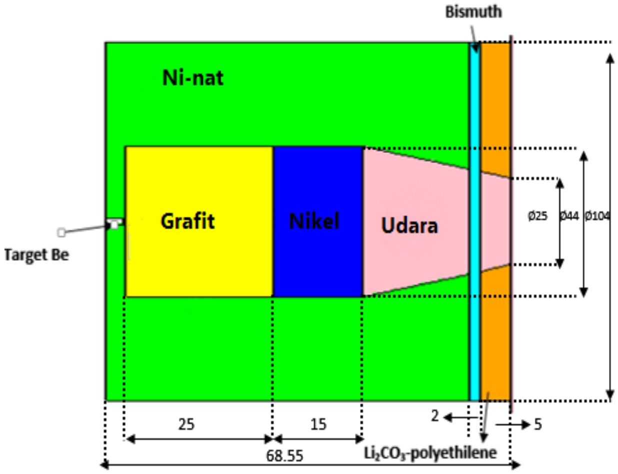

Figure 5.1. Beam shaping assembly

Collimator above using nickel wall 30 cm, 25 cm moderator carbon, filter fast and thermal neutrons than nickel 15 cm, gamma deduction of bismuth 2 cm thick. Collimator aperture diameter of 25 cm.

Room Modelling

The room is made with a length of 200 cm, width 200 cm, height 166.4 cm that is tailored to the needs of BATAN for in vitro assays and in vivo. The rooms were simulated just a bare room containing air. Dose calculations using a material contains as in soft tissue (healthy tissue) that is placed around the room for isotropic radiation that is spread in all directions

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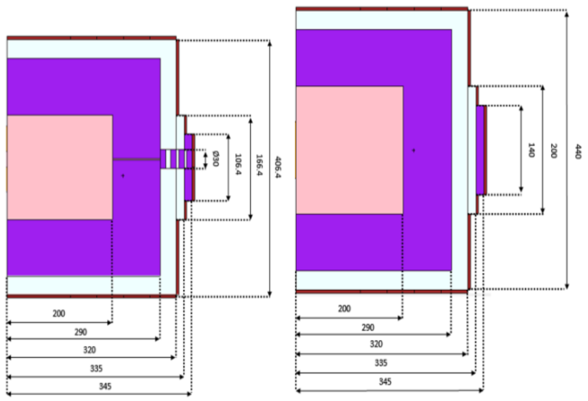


(a) (b)

Description :

▨ □ 🟪 🟥

Figure 5.2. Design with Radiation Shielding Materials and Concrete Polyethylene Borated barite: (a) Side View (b) Front (in cm) [13]

Neutron flux

The results of the simulation are MCNP neutron flux in the treatment room is as follows.

Table 5. 1. Neutron Flux

Energy (MeV)	Flux Neutron ($n.cm^{-2}.second^{-1}$)
5,00E-01	3,00E+07
1,00E+00	1,03E+06
2,50E+00	2,98E+05
2,00E+01	2,43E+05
Total	3,16E+07

Simulation results using MCNPX table shows the value of the neutron flux in the room

Reaction rate calculation

The calculation of the reaction by using BNCT treatment begins with finding the data cross-sectional uptake in software Janis, Flux Neutron, and Density Atom. The calculation of the rate of reaction can be performed using equation 3.7.

Absorption cross section derived from Avogadro's number that is 6.023 x 1023 $atom.mol^{-1}$ multiplied by σ σ used is conditioned by 10^{-3} $g.cm^{-3}$ and then multiplied by the number of nitrogen-14 in nature amounted to 0.996 and shared with the masses atoms of nitrogen-14, which is 14. The results of these calculations yield values for nitrogen absorption cross section-14 of 3.38 x 10^{19} $atom.cm^{-3}$ or can be expressed by equation 5.1.

$$\mathrm{Nx} = \frac{N_A \times \rho \times f}{M_r} \qquad (5.1)$$

Description :

N_x = Atom Density ($atom.cm^{-3}$)

N_A = Avogadro number ($atom.mol^{-1}$)

f = Volume nitrogen-14 fraction

ρ = The density of the air (cm^3)

M_r = Massa Atom ($g.mol^{-1}$)

The result of the calculation of the absorption cross section is then used to calculate the rate of the reaction of nitrogen-14. The result of the calculation of the reaction rate or rate of formation of carbon-14 is 80,106 $reaction.cm^{-3}.second^{-1}$.

Breathe in the rate calculation

The rate of inhaled obtained by using multiplication of the respiratory rate multiplied by the rate of reaction where in one minute one can inhale L. 6-8 in this study authors used 7 L per minute. Respiratory rate is converted to $cm^3.second^{-1}$ to 0.42 $cm^3.second^{-1}$ calculation results obtained breathe rate amounted to 1,046,462,976 $Bq.year^{-1}$.

$\lambda\lambda_{eff}$ Calculation

λeff magnitude is an effective decay constant obtained from λf with λb. λ obtained from the half-life of physical and biological half-life. Generally, biological half-life has a half shorter than the physical half-life time, due to the excretory system in the body is able to shed the entry element that is not needed by the body quickly. The half-physical to nitrogen-14 very long years is 5730, while the biological half-life much different from the physical half-life of 12 days. The conclusion result λeff for nitrogen-14 is 1.21 x 10-4 year.

The calculation of the rate of input (Mm)

The rate of input obtained from the rate multiplied by λf breathe. The rate of input is used to determine the activity dose of radioactive elements per unit time. The rate of input to the carbon-14 is 126,622.02 Bq.

Calculation Activities

Activities calculations obtained by using equation 5.2

$$\frac{d}{dt} A_B = M_m - \lambda_{\text{eff}} . A_B \qquad (5.2)$$

Equation 5.2 is derived using Laplace transforms that into the equation 5.3 as follows

$$A_B(t) = \frac{M_m}{\lambda_{\text{eff}}}(1 - e^{-\lambda E t}) \qquad (5.3)$$

Description :

A_B = Activities in the body (Bq)

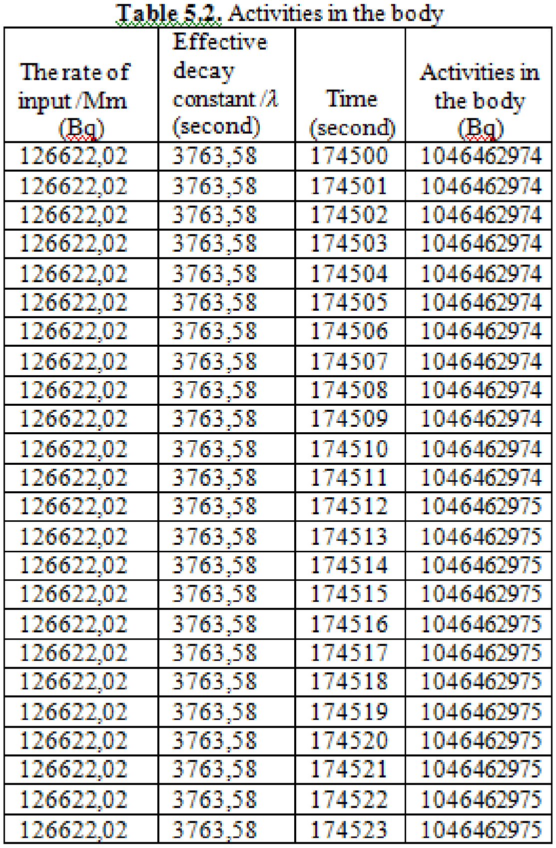
M_m = Rate of input ($Bq.second^{-1}$)

λ_{eff} = Effective decay constant (second)

t = Time (second)

Results calculated from equation 5.2 can be described using Table 5.2 as follows

Table 5.2. Activities in the body



The rate of input /Mm (Bq)	Effective decay constant /λ (second)	Time (second)	Activities in the body (Bq)
126622,02	3763,58	174500	1046462974
126622,02	3763,58	174501	1046462974
126622,02	3763,58	174502	1046462974
126622,02	3763,58	174503	1046462974
126622,02	3763,58	174504	1046462974
126622,02	3763,58	174505	1046462974
126622,02	3763,58	174506	1046462974
126622,02	3763,58	174507	1046462974
126622,02	3763,58	174508	1046462974
126622,02	3763,58	174509	1046462974
126622,02	3763,58	174510	1046462974
126622,02	3763,58	174511	1046462974
126622,02	3763,58	174512	1046462975
126622,02	3763,58	174513	1046462975
126622,02	3763,58	174514	1046462975
126622,02	3763,58	174515	1046462975
126622,02	3763,58	174516	1046462975
126622,02	3763,58	174517	1046462975
126622,02	3763,58	174518	1046462975
126622,02	3763,58	174519	1046462975
126622,02	3763,58	174520	1046462975
126622,02	3763,58	174521	1046462975
126622,02	3763,58	174522	1046462975
126622,02	3763,58	174523	1046462975

Results from Table 5.1 can be used to create a graph of time and activity of carbon-14, ie, in Figure 5.3

Figure 5.3 Graph of time against activity

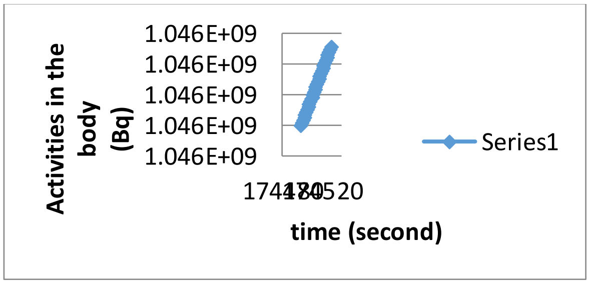


Figure 5.3 shows that in order to reach saturation activity takes 5.90 x 1010 second, the longer the time, the activity will be saturated. Activities saturation of carbon-14 is 1046462975 Bq.

Activities will be converted into an equivalent dose (Sv) to determine the dose that would be received by radiation workers. Dose obtained from activity multiplied by eg in annex 1 Regulation of the Nuclear Energy Supervisory Agency Number 4 Year 2013 About Protection and Radiation Safety in Nuclear Power Utilization for 8.3E-09 $Sv.Bq^{-1}$. The result of this calculation is 5,893 x 10^{-14} $Sv.year^{-1}$

CONCLUSION

After doing research on the analysis of the dose of radiation workers BNCT facility with 30 MeV Cyclotron neutron source, it can be concluded :

1. Internal radiation dose in BNCT facility with 30 MeV Cyclotron neutron source is 5,893 x 10^{-11} $mSv.year^{-1}$.
2. Based BAPETEN Chief Regulation No. 4 Year 2013 On Protection and Utilization of Radiation Safety in Nuclear Power as saying that the maximum dose of radiation worker 20 $mSv.year^{-1}$, then the internal dose for radiation workers safe.

SUGGESTION

1. Undertake analysis of on the external dose treatment room in order to get the total dose obtained by radiation workers.
2. It is recommended to do further research to verify the real value of doses that have been obtained.

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