Indonesian Journal of Physics and Nuclear Applications
Volume 2, Number 2, June 2017, p. 83-90
ISSN 2549-046X, © FSM UKSW Publication

A Conceptual Design Optimization of Collimator With ^{181}Ta As Neutron Source for Boron Neutron Capture Therapy Based Cyclotron Using Computer Simulation Program Monte Carlo N Particle Extended

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Abstract- The optimization of collimator with 30 MeV cyclotron as neutron source and ^{181}Ta as its proton target. cyclotron assumed work at 30 MeV power with 1 mA and 30 kW operation condition. Criteria of design based on IAEA's recommendation. Using MCNPX as simulator, the result indicated that with using ^{181}Ta as target material with 0.55 cm thickness and 19 cm diameter, 25 cm and 45 cm PbF_2 as reflector and back reflector, 30 cm ^{32}S as a moderator, 20 cm ^{60}Ni as fast neutron filter, 2 cm ^{209}Bi as gamma filter, 1 cm $^{6}Li_2CO_3$- polyethylenes as thermal neutron filter, and 23 cm diameter of aperture, an epithermal neutron beam with intensity 4.37 x 10^9 n.cm^{-2}.s^{-1}, fast neutron and gamma doses per epithermal neutron of 1.86 x 10^{-16} Gy.cm^2.n^{-1} and 1.93 x 10^{-13}Gy.cm^2.n^{-1}, minimum thermal neutron per epithermal neutron ratio of 0.003, and maximum directionality 0,728, respectively could be produced. The results have passed all the IAEA's criteria.

Keywords : Conceptual design, collimator, BNCT, MCNPX, *Cyclotron*, ^{181}Ta, IAEA criteria.

INTRODUCTION

Cancer becomes leading cause of death by 13% (7.6 million deaths) of the total causes of death in the world in 2008. Lung cancer, stomach, liver and breast cancer as the most is the cause of death. Liver cancer accounts for 695 thousand as a cause of death after lung cancer and stomach cancer. In Indonesia as many as 165 people out of 100,000 population die from cancer each year. The prevalence of cancer in 2013 the total number of cancer cases that occurred in 2013 in a region compared to the overall number of cases of disease that occurred in 2013 in the region. Cancer prevalence is highest in Yogyakarta (4.1 ‰), followed by Central Java (2.1 ‰), Bali (2 ‰), Bengkulu and Jakarta respectively 1.9 per mil. Data were collected by interview at any age, according to the doctor's diagnosis. The prevalence of cancer is somewhat higher in infants (0.3 ‰) and the increase in aged> = 15 years, and the highest at age> = 75 years (5 ‰). The prevalence of cancer in the city tend to be higher than in the village [1,2,3].

Several attempts treatment against cancer that has been intensified by surgery, chemotherapy, and radiotherapy. Radiotherapy is a type of therapy that typically use X-rays, gamma rays, and charged particles to destroy cancer cells and shrink tumors. Radiation uses high-energy dose that can cause ionization in the surrounding normal cells, and also the radiation is usually effective only when the characteristics of Linear Energy Transfer (LET) (53 keV.µm-1 or smaller) [4,5,6].

Boron Neutron Capture Therapy (BNCT) is a type of therapy for cancer using radiation as a source of neutron radiation. A therapy is said to be good when the therapy is capable of destroying cancer cells without causing any harmful side effects to the surrounding normal cells. In the boron treatment process will be put into the patient's body with the help of boron delivery and then irradiation field to be irradiated with epithermal neutrons, and neutrons will be captured by the isotope ^{10}B and then this reaction will produce charged particles are particles of α (alpha) and the nucleus ^{7}Li. Both of these particles have the same LED high (≥75 keV.µm-1) and a short-range light (about 4.5 to 10 m) [7], so that the light is confined in the cells [8]. Neutron sources were fired into isotope Boron-10 is derived from the material circular accelerator (Cyclotron). There is one vital component of this Cyclotron systems, proton-targets. This component serves as a neutron-generating system for later ditumbukkan components Boron-10 inside the patient. In addition, these components must be able to withstand very high heat 30kW, certainly needed a material that has a high melting point. While this used in Cyclotron result of the cooperation between the Kyoto University Research Reactor Institute (Kurri) and Sumitomo Heavy Industries, Ltd. (SHI) is a pure Beryllium material [9]. Despite Beryllium is a good material to withstand high heat, this material is toxic. As an alternative, will be modeled material for the proton-targets of material ^{181}Ta.

In 1950 - 1994 files that are used for facilities BNCT is a neutron thermal, but the beam of neutrons thermal memilliki penetration is low and dose distributions are bad, then in 1994 neutron epithermal used the first time in the USA (Mitr-II and BMRR) [10]. Based on IAEA criteria for BNCT procedure epithermal neutron beam intensity is suitably more than $1E+10^9$ n.cm-2.s-1 or more, but the epithermal neutron flux for $5E+10^8$ $n.cm^{-2}.s^{-1}$ they can be used even with a longer irradiation time [9]. Epithermal neutrons in order to produce the desired flux, we need a method to filter and moderate the fast neutrons resulting from the reaction between beryllium targets with energy 30 MeV protons accelerated by a cyclotron HM-30. Collimator is a system that is designed to be able to filter and moderate the fast neutrons so that the output beam of neutrons produced in accordance with the criteria of the IAEA.

In designing the collimator, the characteristics of the output beam of concern there are two, namely the intensity and quality. The intensity of the beam will be the determining factor for how long therapy. Quality relates to the type, energy and other radiation intensity present apart epithermal neutrons. The following epithermal neutron beam parameters recommended by the IAEA.

Tabel 1. Beam Parameters.[9]

Parameters	Value
Epithermal beam intensity Φ_{epi} **(n.cm^{-2}.s^{-1})**	$>1,0\times10^9$
Fast neutron dose per epithermal neutron $\dot{D}_f / \Phi_{epi}$ **(Gy.cm^2.n^{-1})**	$<2,0\times10^{-13}$
Foton dose per epithermal neutron $\dot{D}_\gamma / \Phi_{epi}$ **(Gy.cm^2.n^{-1})**	$<2,0\times10^{-13}$
Thermal neutron flux per epithermal neutron flux ratio Φ_{th} / Φ_{epi}	<0,05
Neutron current per total neutron flux ratio J / Φ_{total}	>0,7

To get the criteria neutron beam for BNCT procedure as described above, then in designing the collimator needs to be optimized in dimensions and material selection right. A collimator is usually made up of five elements, namely

1) *Reflector*

Wall collimator which could either keep the neutrons remain in the moderation space and does not leak to the outside. Materials commonly used as wall collimator is Pb, Bi, and PbF2.

2) *Moderator*

Fast neutron moderation which can best be achieved with a material that has a low atomic mass. Moderator or the selected filter material should not decompose at a high radiation field, nor produce moisture. All products of neutron activation should have a short life span. Candidates suitable material is Al, AlF_3, D_2O, S, Pb, graphite.

3) Gamma shielding

Materials such as Pb and Bi can be used to reduce the gamma rays from the reaction protons smashing into beryllium, however the addition of gamma filter can also reduce the intensity of neutrons. Bismuth is almost as good as the lead for shielding gamma and neutron epithermal passed more. Despite that needs more attention in handling bismuth irradiated neutrons due to the formation of ^{210}Po, emitting alpha formed from the reaction of beta decay is accompanied catches ^{209}Bi ^{210}Bi. Encapsulation of bismuth is strongly recommended.

4) *Neutron filter*

There are two types of filters that are simulated in the optimization of collimator designed. Both types of filters are filters fast neutron and thermal filter. In this research fast neutron filter taken from previous research that ^{60}Ni, ^{60}Ni because of fast neutrons is able to filter out so well that the author simply varying the thickness of ^{60}Ni filter in order to get optimal results. While the thermal filter authors also took on earlier work in which the material used for the thermal filter is $^{6}Li_2CO_3$-polyethylene, because $^{6}Li_2CO_3$- polyethylene is able to filter out the thermal neutron flux so well that the author simply varying the filter $^{6}Li_2CO_3$-polyethylene thickness in order to obtain optimal results.

5) Aperture

An aperture is usually made of a material that has a high absorption, the aperture serves to beam out from the right place, so as not to jeopardize the current procedure, therapy [10].

MATERIALS AND METHODS

A. Neutron Source Modelling

Neutron sources modeled in this study is 1 mA proton energy 30 MeV striking the target material ^{181}Ta diameter of 19 cm and 0.55 cm thick. The reaction (^{181}Ta (p, n) ^{181}W) generate high-energy neutrons will then be moderated in the collimator for the purposes of BNCT.

B. *Simulator*

MCNPX used for simulation in this study, in the writing code MCNPX input, input consists of three parts, namely surface card, cell card, and the data card. The contents of the card surface is a surface type and dimensions. Making the card followed by cell surface cell card card that contains specifications covering the space between the surface density of the material, material number, importance cell, and the name of each cell. In the data card contains the definition of the radiation source used, the number of iterations, the number of sampling, material definition, and continued with the calculation tally.

Tabel 2. Types of *Tally* in MCNPX.[11]

Mnemonic	Tally descriptions	Units
F1:<pl>	Particle current on surface	particle
F2:<pl>	Averaged flux on surface	particle.cm^{-2}
F4:<pl>	Averaged flux on cell	particle .cm^{-2}
F5a:N, F5a:P	Flux on certain point or ring shaped detector	particle.cm^{-2}
6:<pl>	Averaged energy deposition on cell	MeV.g^{-1}
+F6	Heating due to collision	MeV.g^{-1}
F7:N	Averaged fission energy on cell	MeV.g^{-1}
F8:<pl>	Energy distribution from pulse formed in detector due to radiation	pulse
+ F8:<pl>	Deposition	charge

Type tally is used is chosen according to the information to be obtained. In this study tally F1 and F2 are used to evaluate the value of the output collimator designed with the parameters that have been recommended by the IAEA.

At the time of writing the code normalizing factor or multiplier needed for the unit flux of calculation MCNPX tally different from the units used by the IAEA. It takes a powerful source in units of particles per second produced 30 MeV cyclotron with 1mA current strength can be obtained by the following calculation.

$$\frac{1\times 10^{-3}\ C/s}{1{,}6022\ \times\ 10^{-19}\ C/p} = 6{,}2414\times 10^{15}\ p/s$$

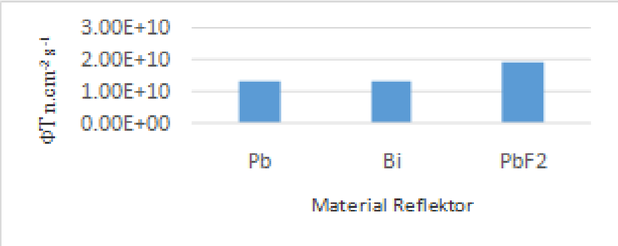
In the calculation of the dose based on the energy released by the radiation beam of neutrons and gamma photons of the material, the reference used is a table of coefficients Kerma issued in the Dosimetry System 2002 (DS02) on ICRU report 63. For the calculation of the neutron flux is required limits for the classification of energy neutron flux can be distinguished for thermal neutrons, epithermal and fast. In MCNP we can enter the upper limit of the neutron energy. Epithermal neutron energy range used in this study is 4 eV - 40 keV, neutrons with energies below 4 eV is a thermal neutron, whereas neutrons with energies above 40 keV are fast neutrons

RESULTS AND DISCUSSION

1. Reflector.

Three candidates were selected reflector material is Pb, PbF_2, and Bi. To get the best reflector material, each simulated reflector material with the same thickness and seen the value of the intensity of neutrons that exists in the It can be seen that the simulation results are presented in Figure 1.

Figure 1. The graph of the reflector material to neutron fluxmoderation.



The simulation result shows that the material PbF_2 more superior than the other two materials. This is due to the scattering cross section of neutrons for materials PbF_2 higher than the other two materials and also has a cross-section of material PbF_2 very low neutron absorption. As a result, neutrons tend to be scattered back into space moderation and a few neutrons are absorbed. To that end, PbF2 material used as the reflector material on this collimator design.

Then, the material PbF_2 simulated by varying the thickness of the reflector in two directions (side reflector and a back reflector) to get the value of the thickness of the reflector is the most optimal. The simulation results side wall thickness variation reflector and a back reflector is presented in Figure 2 and Figure 3.

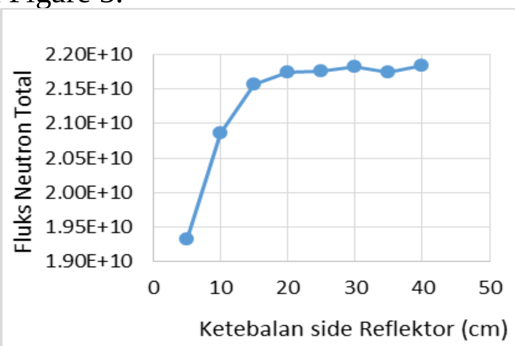


Figure 2 The graph of the neutron flux to the thickness of the side reflector

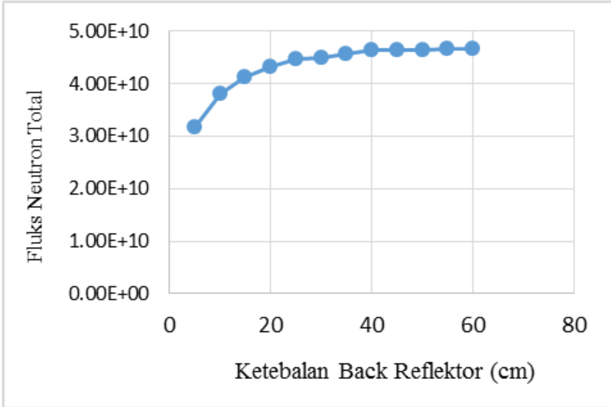


Figure 3 The graph of the neutron flux to the back reflector

From the simulation results can be seen that the neutron flux increases concurrently increasing the thickness of the reflector. The optimum thickness for the side reflector and a back reflector of each reflector achieved when the thickness of 25 cm and 45 cm.

2. Moderator

In this design the moderator election that will be used to have three criteria, namely where epitermalnya high flux, decreasing the thermal flux per epithermal neutron small, fast neutron flux decline per large epithermal neutrons so that the classification process focused on fast neutron flux alone. Materials tested in material selection moderator is Al, AlF_3, D_2O, S, Pb, graphite.

Here are some of the moderator material selection is used, based on criteria defined type of material suitable as a moderator in this study are Sulfur, Figure 4 depicts a comparison of materials against each flux neutron moderator

Election moderator thickness based on three criteria, namely where epitermalnya high flux, decreasing the thermal flux per epithermal neutron small, fast neutron flux decline per large epithermal neutrons but does not exceed the value of the epithermal flux. Based on the simulation results that has been done the data is obtained, as shown in Figure 5 where the most appropriate thickness obtained at a thickness of 30 cm.

3. Neutron filter

3.1 Fast neutron filter

In this study the material selected as the fast neutron filter is ^{60}Ni, this material is able to degrade fast neutrons and epithermal neutron interactions with power so as to improve the beam quality. This happens because nickel has a fast neutron absorption cross section

On testing fast neutron filter thickness variation is obtained which is attached in Figure 6

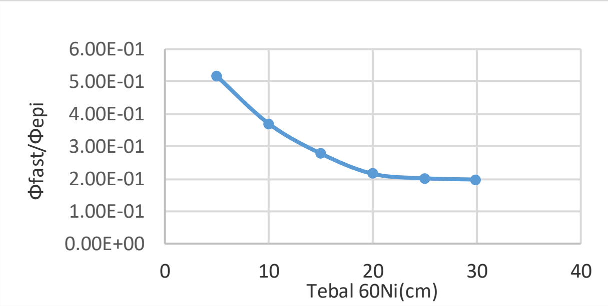


Figure 6. Graph of ^{60}Ni thickness to ratio of fast neutron flux per epithermal neutron flux

Based on the figure 6 it can be seen that the thickness of the filter material of fast neutrons in this case the thickness of Ni is inversely proportional to the neutron flux ratio of fast and epithermal neutron flux, which means that the thick Ni filter is used, the ratio of fast neutron flux and epithermal neutron flux will be smaller. To choose the thickness of the filter Nickel will be used

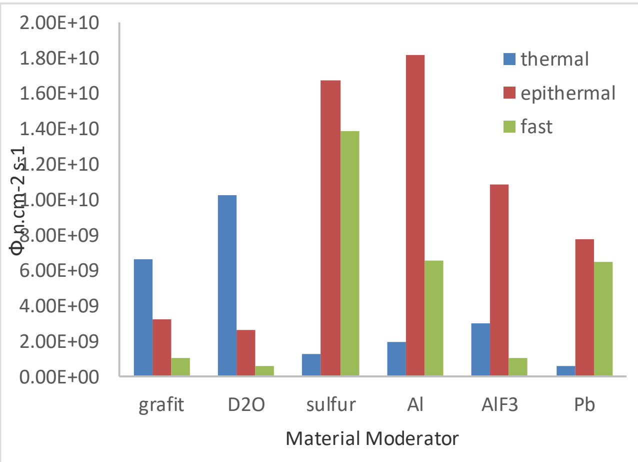


Figure 4. The graph of the neutron flux to the materials moderator

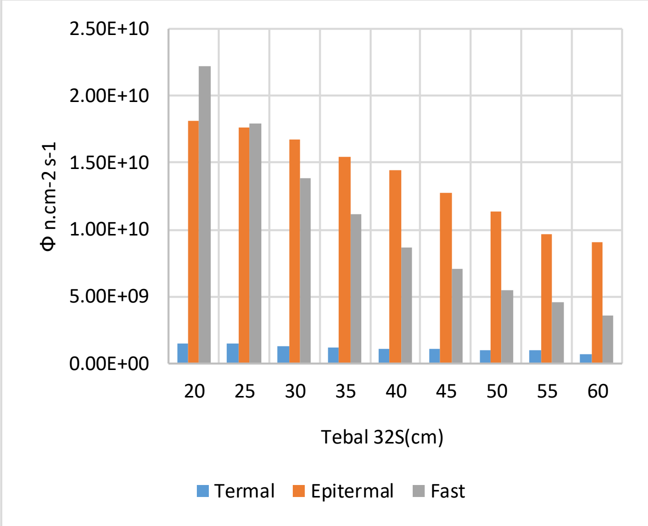


Figure 2. Graph of the thickness of the neutron flux Moderator

indicator used is its ability to lower spectrum fast neutron good and epithermal flux remains high, based on the results of the simulation has been done, the authors choose the thickness of the filter is 20 cm where the comparative value of the fast neutron flux and epithermal amounted to 0.215 because on the thickness of the comparative value is found fast and epithermal neutron flux is relatively small, it indicates that the thickness of the fast neutron flux is quite small but the epithermal neutron flux still likely to be high.

3.2 Thermal neutron filter

At this stage the energy spectrum of the neutrons needed to be in the range of intermediate energy neutron flux in other words you want to get is the epithermal neutron flux, where the neutron filter is used to reduce the thermal neutron flux. In this study the authors chose material $^{6}Li_2CO_3$-polyethylene as thermal neutron filter material, based on the study of literature and research that has been conducted previously found that the ability of the material $^{6}Li_2CO_3$-polyethylene in terms of reducing the flux of epithermal pretty good, it also aims to improve the quality of radiation beams needed in the process of therapy, because radiation is good in that it has a high grade epithermal neutron flux while a low thermal neutron flux. $^6Li_2CO_3$-polyethylene material has a thermal neutron absorption cross section is high enough. Based on the authors try to simulate relating to use of the material as the thermal neutron filter. The simulation results obtained can be shown in Figure 7

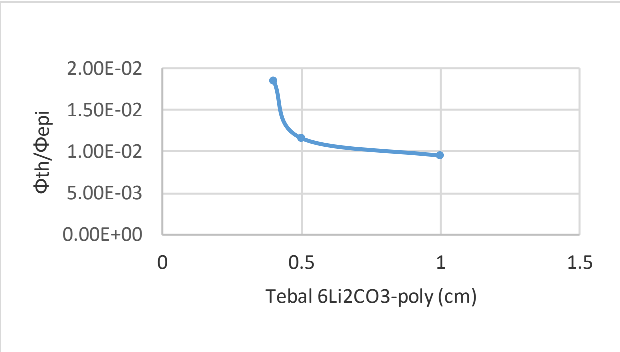


Figure 7. Graph neutron flux thermal / epithermal neutron flux to the thickness of the thermal filter

Based on Figure 7 can be seen that the use of materials $^6Li_2CO_3$-polyethylene tends to give the effect of decreasing the value of the ratio of thermal and epithermal neutron flux. Based on the function, the value of the ratio of the neutron flux of thermal and epithermal at this stage must be small, then based on Figure 7 can be seen that the comparative value of the neutron flux of thermal and epithermal will be enclosed with increasing thickness of the filter, but the author in this case takes the value of the thickness of the filter of 1 cm causes at this thickness value ratio of thermal and epithermal neutron flux has been relatively small in the amount of 0.00948. This means that in these conditions the thermal neutron flux value is already relatively small while the epithermal neutron flux is still likely to be high.

4. Gamma Filter

Shielding gamma is one component of collimator that serves as a shield against gamma-ray dosage, where the selection of material for shielding gamma based on the tanpang latitude total of the material, in this case the author using Bi as a shield gamma causes total interaction of the gamma great that the ability of absorption of the dose gamma rays are also great.

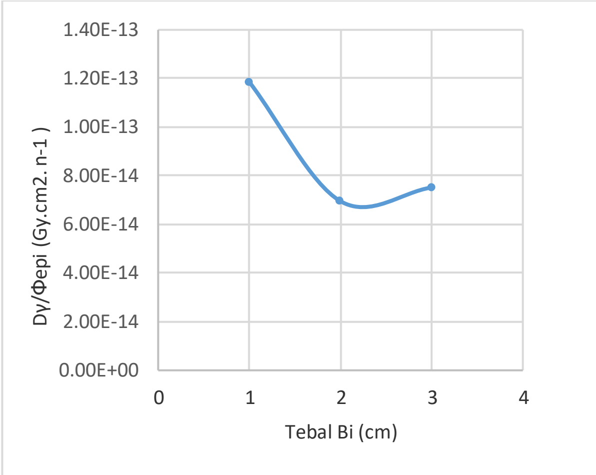


Figure 8. Bi thickness of the gamma dose / epithermal neutron flux.

From the results of the graph in Figure 8 shows that the actual use of shielding gamma form of material Bi provides a good ability in terms of lowering the dose of gamma but if the attention of some aspects and results of the simulation has been done, the authors choose the thickness of Bi as a shield gamma by 2 cm, with the comparative value of gamma dose and flux of epithermal 6.95 x 10^{-14} Gy.cm^2.n^{-1}. At this value at any dose gamma epithermal neutron flux already meets standards set by the IAEA.

5. Aperture

Based on the simulation results that has been done, it is to be obtained is a good aperture mechanism to be patient during therapy should not be a lot of movement and the targets to be subjected to a more targeted therapy so that the mechanism can be maximized. The simulation results, it can be seen in Figure 9

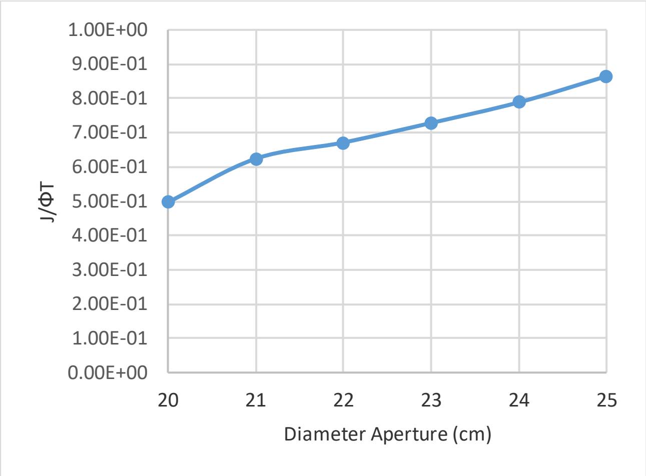


Figure 9. Graph of J / Φ_{total} to aperture diameter

Based on Figure 9 can be seen that the large diameter of the aperture is directly proportional to the value of the current ratio and total neutron flux. Where the value to be achieved for a good therapeutic mechanism is a comparison of current value and the total flux must be greater than 0.7 because the value has met the standards set by the IAEA, the author chose an aperture diameter of 23 cm, where the diameter is all parameters set IAE meets the criteria of the IAEA.

CONCLUSION

Based on the research that has been done, get the results of the design collimator configuration as follows,

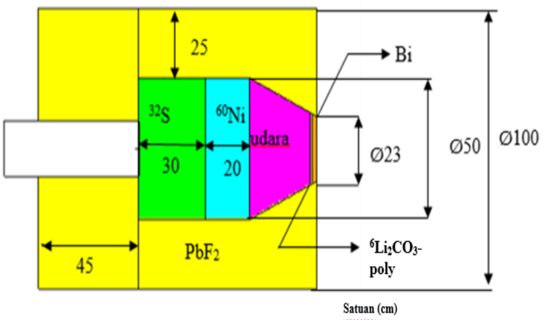


Figure 10. Configuration Results Collimator Design

1. Reflector: side reflector with a 25 cm thick, and a back reflector with a length of 45 cm.
2. Moderator material as thick as 30 cm of ^{32}S.
3. Filter material of fast neutrons with ^{60}Ni 20 cm thickness.
4. Shield gamma with ^{209}Bi material thickness of 2 cm
5. Shield of thermal neutrons with $^{6}Li_2CO_3$-polyethylene material as thick as 1 cm.

Table 3. Design result.

Unit	Design Value	IAEA Design Criteria
Φ_{epi} (n.cm^{-2}.s^{-1})	4.37 x 10^9	>1,00E+09
$\dot{D}_f$ / Φ_{epi} (Gy.cm^2.n^{-1})	1,86 x 10^{-16}	< 2,00 x 10^{-13}
$\dot{D}_\gamma$ / Φ_{epi} (Gy.cm^2.n^{-1})	1,93 x 10^{-13}	< 2,00 x 10^{-13}
Φ_{th} / Φ_{epi}	0,00298	< 0,05
J / Φ_{total}	0,728	> 0,7

From the results in Table 3 all parameter value has met the criteria of the IAEA

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