SIMULATION OF OPTIMUM THICKNESS AND CONFIGURATION OF 10 MeV CYCLOTRON SHIELD

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Abstract

Baby Cyclotrons that made in Self-shield type have been employed for use in Medical centers for the diagnosis of cancer diseases by positron emission tomography (PET) system. Conceptual design studies and construction of a 10 MeV cyclotron have been done at the Amirkabir University of Technology. Here in we have done a discussion on simulation of gamma and neutron dose rates at a distance of one meter outside of the cyclotron shielding. This shield consist of Lead, polyethylene (10% B) layers from inside to outside respectively. With increasing the thickness of lead and polyethylene we will see a decrease in the gamma and neutron dose which received by the water phantom at a distance of one meter outside from the surface of the shield of the cyclotron. Note that the gamma and neutron dose at the beginning (without any shielding) was on the order of several thousand μSv per hour that by achieve to a certain amount of thickness of the shield, the dose was reduced to below of the limited level. In this study, the MCNPX Code has been used. In MCNPX Code that used the variance reduction techniques for decreasing relative errors of calculation which was a good method for this case study.

INTRODUCTION

With the development of cyclotron in the 1930s, radioisotopes have been produced for medicine, industry, agriculture and research significantly [1]. Cyclotron accelerators have many applications in the industrial and medical fields.

Today, fluorine is used in radiopharmaceuticals and plays an essential part in the oncology. The cyclotron accelerator is applied in medicine to produce radioisotopes for PET device using for the detection of cancerous diseases. PET is one of the ways to determine the physiological and chemical processes in the body by a quantitative method. Some radioisotopes produced by Cyclotron are ^{11}C, ^{15}O, ^{14}N, and ^{18}F which their half-lives are 20, 2, 10, and 110 minutes respectively. Operation of accelerators will produce gamma and neutron radiation. These radiations can have damaging effects to accelerator operator and those referring to accelerators department. In order to reduce the effects of this radiation, there are different ways that must comply with the principle of ALARA. One of these ways is using of a radiation shield. Depending on the kind of application, the type of radiation shielding will be important. In determining the type of radiation shield, the location and atmosphere dedicated to the cyclotron room is effective.

If the large space will be available, the cyclotron vault model can be used. Because cyclotrons have medical applications and are often used in medical centers and hospitals, there is space limitation for it. On the other hand, since short half-life radioisotopes are produced (approximately 2 hours), it is required to use them in a cyclotron nearby. These items create a situation that a self-Shield is used for radiation shielding.

WORK METHOD

In this study, we simulate a self-shield type for protection. In fact, instead of cyclotron room, a shield that attached to the cyclotron is used. It can be used anywhere. Because of producing gamma and neutron radiation in the cyclotron accelerator, each of them require their appropriate shielding. In this situation, combined shield is needed. For gamma-ray, high atomic number materials such as lead shielding are used. And for neutron radiation, low atomic number materials such as concrete, polyethylene and boron that is neutron capture also, lithium and cadmium are used.

Neutron absorption cross section of these materials are several thousands Barn [2]. This order of number is good for absorbing neutrons.

Although cadmium has a higher neutron absorption cross section than the other two materials, but it leads to strong secondary gamma-ray production and therefore, that is required to utilize thicker layer of lead for attenuation of gamma rays which is not appropriate. So it is better to use boron and lithium [3]. In this study that is based on a 10MeV cyclotron accelerator, Negative hydrogen ion beam has a current of 150 Aμ and is accelerated to 10MeV. In this model, particles are accelerated horizontally. The outer dimensions of the accelerator, which includes its height and diameter, are 1767 mm and 1760 mm respectively. Conceptual design studies and construction of a 10 MeV cyclotron have been done at the Amirkabir University of Technology [4].

Target of cyclotron is usually considered as the main source of radiation. In fact, more than 90% of radiation comes from the target, although there are reactions with other components of the cyclotron, such as the collision of proton beams and secondary radiation with the accelerator body.

A cylindrical target of 1.5 mm height, 1.2 mm inner diameter, and 3 mm outer diameter has been studied. The thickness of cylinder base, which calculated with SRIM code is 1μm and it is bombarded by proton beams. Target foil is regarded neodymium. To produce FDG, target

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material is enriched water with oxygen-18 (95% enrichment).

As the result of reaction between Target and proton beams, Fluorine is produced. Fluorine emits beta radiation which is used in PET systems. In this reaction, neutrons and gamma rays are produced. To simulate the shielding, Lead and Polyethylene Borated is used. In this study, the optimal thickness of each material is calculated by MCNPX. In simulate shielding we need to have a high penetration in material. However the outputs must have an acceptable accuracy, it is needed to apply variance reduction techniques. For this simulation, weight window method has been used. Weight window method can be implemented in several ways in MCNP code. Here, the implemented is reducing density technique. We reduce the density of the materials to an untrue amount that can be obtained an output for Tally. So at each stage material density has increased toward its real value. At the end we should arrive to the actual appropriate output values with permissible error.

Simulations

We have used polyethylene borated (10% boron) for neutron shielding and lead for gamma shielding. Shielding geometry is cylindrical. First polyethylene and then lead stayed back of it. We wanted out of the shield the dose rate of neutron and gamma reach to a limit level. In MCNPX code, neutron and gamma dose are calculated by a water phantom out of shielding.

In this section the dose attenuated for neutron and gamma are illustrated.

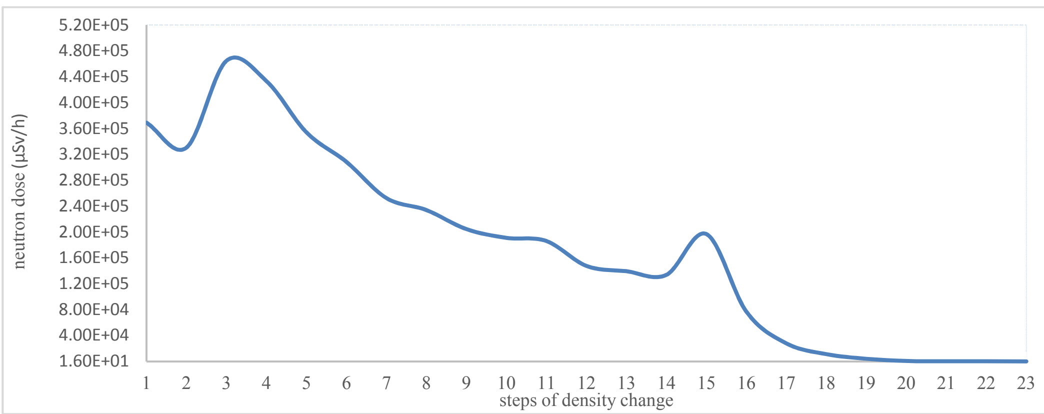

Figure 1: Neutron dose with density change technique of variance reduction.

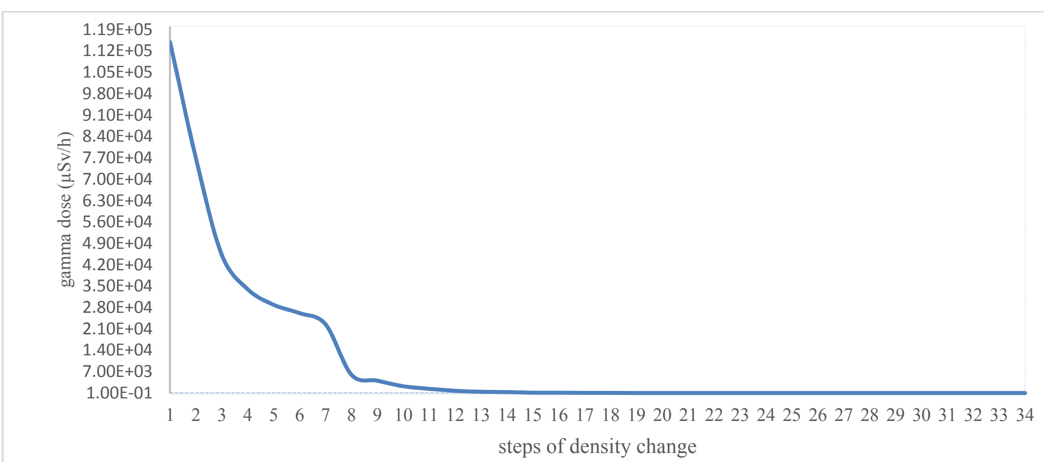

Figure 2: Neutron dose with density change technique of variance reduction.

According to Figs. 1 and 2 neutron and gamma dose rate is less than the permissible limit and was obtained 16 µsv/hr and 0.1 µsv/hr respectively. Table1 and Table2 show density values of materials at each stage of simulation. It should be noted that the density of the material during the simulation are shown in the tables.

Table 1: Material Density of Shielding for Calculating Neutron Dose in Each Stage of Simulation (Unit of Density is g/cm3)

Steps of Density Change	Lead	Poly with Boron
1	0.35	0.05
2	0.95	0.05
3	1.65	0.05
4	2.65	0.05
5	3.65	0.05
6	4.65	0.05
7	5.65	0.05
8	6.35	0.05
9	7.05	0.05
10	7.75	0.05
11	8.5	0.05
12	9.25	0.05
13	10.5	0.05
14	10.7	0.05
15	11.35	0.05
16	11.35	0.15
17	11.35	0.25
18	11.35	0.35
19	11.35	0.45
20	11.35	0.65
21	11.35	0.8
22	11.35	0.9
23	11.35	1

Table 2: Material Density of Shielding for Calculating Gamma Dose in Each Stage of Simulation (Unit of Density is g/cm3)

Steps of Density Change	Lead	Poly with Boron	Steps of Density Change	Lead	Poly with Boron
1	0/05	0/1	18	4/6	1
2	0/05	0/2	19	5/1	1
3	0/05	0/35	20	5/5	1
4	0/05	0/5	21	5/8	1
5	0/05	0/6	22	6/2	1
6	0/05	0/75	23	6/5	1
7	0/05	0/85	24	7	1
8	0/75	1	25	7/3	1
9	0/95	1	26	7/7	1
10	1/45	1	27	8/2	1
11	1/95	1	28	8/5	1
12	2/45	1	29	8/9	1
13	2/85	1	30	9/3	1
14	3/05	1	31	9/9	1
15	3/65	1	32	10/7	1
16	4	1	33	11	1
17	4/3	1	34	11/35	1

To ensure that the neutron dose does not leak out of the shield, neutron dose Map has been calculated by mesh tally. You can see neutron dose map in the Fig. 3. Dose map of neutron calculated than 20 cm inside of shield to 25 cm outside of shield. The point R=165cm is on outside surface of the shield. As seen in Fig. 3 in this point neutron dose is below than 16 µsv/hr.

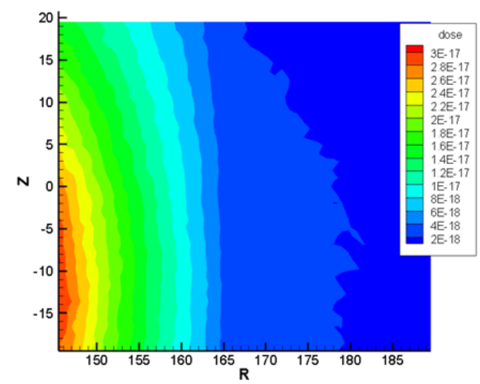

Figure 3: Dose map of neutron.

Note that the dose rate unit in the Fig. 3 is ((rem/hr)/(n/cm^2.s)). These instructions are a typical implementation of the requirements. As are visible in Figs. 4 and 5, cyclotron shield is designed in Solidworks. Also, according to simulations to calculate the dose in MCNP, designed geometry is achieved in the visual editor. Visual editor is a Sub-Program of MCNPX.

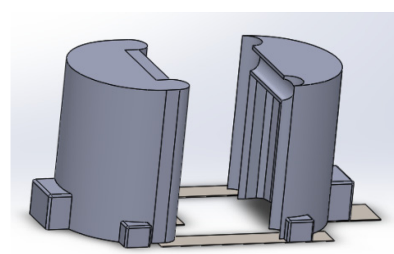

Figure 4: Geometry is designed in solidworks.

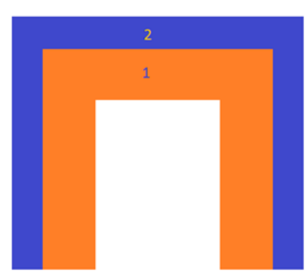

Figure 5: Geometry designed in MCNP also shows (side view) that the shielding materials. 1 is polyethylene borated and 2 is Lead.

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CONCLUSION

At the end of simulation concluded that 60 cm and 35 cm are the best thicknesses for polyethylene and lead respectively. Because according to our data in these thicknesses dose rate of neutron and gamma are 16 μsv/hr and 0.1 μsv/hr on the outside surface of shield that are lower than the limitation level for radiations in determine distance. While without shielding, neutron and gamma dose rates are a few thousand microsievert per hour.

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