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• *Technical innovation*

Designing an Am-Be miniature neutron source

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Background: Miniature neutron sources with high neutron flux have abundant applications in medicine, industry and researches. The most important general characteristic of miniature neutron sources is their diameter which is 3mm in average. In this research, we have surveyed and designed an Am-Be miniature neutron source fabrication. ***Materials and Methods:*** This investigation resulted in creation of an Am-Be neutron source, using beryllium metal powder with 98% carat and 100-200 µm mesh and Americium source with activity of about 200 µCi. Neutron source designing was performed under safety and protective factors. The system was designed in two different forms based on the fluent yield of neutron or cut off neutron yield. ***Results:*** The mean neutron flux of miniature neutron source was measured as 1.14 (n/sec.cm^2), and it was calculated as 2.56 (n/sec.cm^2) by MCNP (4C) code. Due to purity and mesh of beryllium, which were not calculated by MCNP code, the calculated flux via Monte Carlo method was approximately 2 times larger than neutron flux from fabricated miniature neutron sources. ***Conclusion:*** In order to fabricate the miniature neutron sources Am-Be with high efficiency, the americium sources with high activity and the target material (Be) in different forms are required. **Iran. J. Radiat. Res., 2007; 5 (1): 41-44**

Keywords: *Miniature neutron source, MCNP (4C) Code, BF3 detector, neutron flux, activity.*

INTRODUCTION

Neutron with different energies is required in nuclear medicine, radiotherapy and industry. For example, fast neutron is used in radiobiological research and radiotherapy, epithermal neutron in boron neutron capture therapy (BNCT), and thermal neutron in neutron activation analysis [1, 2]. Neutrons can be produced from different sources such as nuclear reactors, particle accelerators and isotopic neutron sources. Due to their simplicity of installation, operation and low price, comparing to other neutron sources, isotopic neutron sources have many applications. However, these neutron sources have deficiencies such as low neutron yield and short half life [3]. Isotopic neutron sources usually were fabricated in the form of capsules with equal height and diameter (about centimeters), while miniature neutron sources diameters are less than 3 mm. By decreasing the capsules, diameter, the achievement of miniature neutron source will become possible. Traditional radiation treatment in radiotherapy makes use of gamma rays or X-rays. Neutrons can be more effective than gamma and X-rays, due to the fact that they can deposit more concentrated energy at the sub-cellular level, yet, the neutron will damage surrounding normal tissue unfortunately. Miniature neutron sources enable physicians to insert miniature neutron source into the body of the patient with specific devices. At that rate neutrons are slightly emitted into region directly without any damage to the surrounding healthy tissues [4].

Oak ridge national laboratory (ORNL) and Isotron, Inc., have co-developed ^{252}Cf miniature neutron sources suitable for interstitial and intracavitary HDR NBT [5]. One of these ^{252}Cf miniature neutron sources contains 30 µg of ^{252}Cf in a 2.8mm diameter by 23mm long capsule [6]. The interaction between particles can be simulated with Monte Carlo method. In the present study, the experimental results have been

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simulated with Monte Carlo method, MCNP (4C) code.

MATERIALS AND METHODS

In order to have high neutron flux beam at necessary time, as shown in figure 1, we have calculated 1mm gap between target and americium source by Srim code [7]. In this method, the neutron beam was under control, and one could switch off the source by using a suitable sheet between ^{241}Am and beryllium. The distance between americium and beryllium was dependant on some important factors such as the energy of alpha radiation and cross section of ^{9}Be (α,n) ^{12}C reaction.

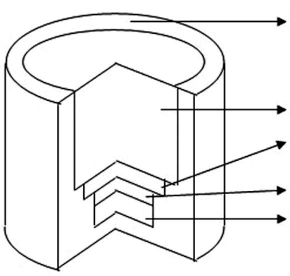

Figure 1. Designed Am-Be neutron source.

When the alpha particle pass through the air, its energy decreases, so, in certain energy and distance, the cross section of above reaction increases and the resonance happens, therefore, with considering technical restriction and facilities, we optimized the distance to get high alpha radiation flux. The attenuation energy of alpha particle in air and their range in beryllium were estimated by Srim code. Regarding effective thickness of target (beryllium) for generating neutron with suitable flux, the beryllium metal powder with 98% carat, and 100-200 µm mesh and americium source with about 200 µCi activity were used to fabricate Am-Be neutron source; the angular distribution of neutron flux was measured by BF_3 detector (VICTOREEN) with diameter and height of 25.37mm and 162.61mm, respectively. The pressure of BF3 gas was 23 cm-Hg, and due to symmetric position, it was the same in all directions. As shown in figure 1, the (α,n) reactions happened in a plexi-glass chamber, in form of cylinder. In geometry of chamber, the results were the same in all direction.

Regarding the available alpha radiation source (^{241}Am) with 14 mm diameter, for using maximum alpha radiation flux to achieve high neutron flux, beryllium target diameter was considered 12mm.

RESULTS

The stopping power of alpha particles in beryllium and air was calculated by Srim code. The partial of results are shown in table 1. The alpha particle energy of americium source is 5.48 MeV and its energy is deducing about 0.08 MeV after crossing 1mm in air. Therefore, alpha particle were entered into the target with energy about 5.4 MeV. To determine the range of projectiles in beryllium target, the stopping power of alpha particles in beryllium was calculated.

Table 1. Results of SRIM code (stopping unit= MeV/mm) for different ion energies.

Ion Energy	dE/dx Elec.	dE/dx Nuclear	Projected Range
100.00 keV	2.32E+02	2.17E+00	6310 A
300.00 keV	3.22E+02	8.95E-01	1.32 um
400.00 keV	3.33E+02	7.05E-01	1.62 um
500.00 keV	3.34E+02	5.85E-01	1.92 um
600.00 keV	3.32E+02	5.01E-01	2.22 um
700.00 keV	3.26E+02	4.40E-01	2.52 um
800.00 keV	3.19E+02	3.93E-01	2.83 um
900.00 keV	3.11E+02	3.55E-01	3.15 um
1.00 MeV	3.03E+02	3.25E-01	3.48 um
2.00 MeV	2.32E+02	1.78E-01	7.27 um
3.00 MeV	1.87E+02	1.25E-01	12.09 um
4.00 MeV	1.58E+02	9.72E-02	17.92 um
5.00 MeV	1.37E+02	7.98E-02	24.72 um
5.40 MeV	1.31E+02	7.46E-02	27.70 um
5.48 MeV	1.29E+02	7.36E-02	28.32 um

The neutron yield for Am-Be standard source is adapted from literature, as 2.7×10^6 n/sec.Ci, so, the neutron yield was calculated for present ^{241}Am activity (194.625 µCi) as follows:

(2.7×10^6 n/sec.Ci) × (^{241}Am activity)

As, it is shown in figure 1, there has been a gap between the ^{241}Am and beryllium target, so the detector efficiency (ε = 0.01) and solid angle correction (Ω) were considered to determine the correct value of neutron rate in the present investigation, as following (8):

$$S = N\frac{4\pi}{\varepsilon\Omega} \quad (1)$$

N is the number of counted neutrons with detector (N=1.6). The neutron rates per second were measured, at various directions, by BF_3 detector, as 320 neutrons per second.

The Am-Be source was considered as a circular disk with "R" radius. The neutron flux was determined by the relation as follow (9):

$$\varphi = \frac{S}{4\pi R^2}\mathrm{Ln}\{\frac{1}{2z^2}[z^2+R^2-d^2+\sqrt{R^4+2R^2(z^2-d^2)+(z^2+d^2)^2}]\} \quad (2)$$

Where, the parameters of "z" and "d" are the horizontal distance from source axis and the vertical distance from Am-Be source, respectively. From the above equation the neutron flux was measured as φ = 1.14 (n/sec.cm^2). The measurements of flux have done by three cylindrical chambers with the same radius and different height as 1, 1.5 and 2 millimeters. The results of these experiments at various heights were same.

The neutron flux was calculated at various thickness of beryllium target by MCNP code. The results of these calculations are tabulated in table 2.

Table 2. Results of MCNP code (Relative error≤0.0009) for different target thickness.

Beryllium thickness (µm)	MCNP code result (1/cm^2)	Neutron flux (n/sec.cm^2)
2000	4.89×10^{-3}	2569.6×10^{-3}
1500	5.085×10^{-3}	26722×10^{-3}
1000	5.3098×10^{-3}	279028×10^{-3}
500	5.562×10^{-3}	2923.03×10^{-3}
400	5.617×10^{-3}	295209×10^{-3}
300	5.674×10^{-3}	2981.52×10^{-3}
200	5.729×10^{-3}	3010.62×10^{-3}
100	5.787×10^{-3}	3041.35×10^{-3}
80	5.799×10^{-3}	3047.35×10^{-3}
60	5.81×10^{-3}	3053.57×10^{-3}
40	5.822×10^{-3}	3059.69×10^{-3}
27.7	5.829×10^{-3}	3063.39×10^{-3}
25	5.82995×10^{-3}	3063.58×10^{-3}
20	5.82986×10^{-3}	3063.56×10^{-3}
15	5.82988×10^{-3}	3063.54×10^{-3}
10	5.82993×10^{-3}	3063.53×10^{-3}
5	5.82946×10^{-3}	3063.32×10^{-3}

DISCUSSION

As shown in table 2, due to the absorption of neutron by beryllium, the flux of neutron was increasing with increasing beryllium thickness. As a result of some experimental restrictions in the present investigation, the flux of neutron was measured only at 1, 1.5 and 2 millimeter thickness of beryllium target. Our experimental flux at 2 mm thickness of target seemed to be in better agreement with MCNP results, since, in the calculation by MCNP code in the present study, beryllium target was not considered as a powder form with 98% carat and 100-200 µm mesh. As it is shown in the table 1, the α-particle (E_α=5.4 MeV) range in the beryllium target is 27.7 µm, as indicated in the table 2, by increasing the thickness of target over than α-particle range, the flux of neutron was constant. This result is in agreement with the experimental flux values at 1, 1.5 and 2 millimeter thickness of beryllium target.

Being able to have a number of beryllium target plates with smaller α-particle range thickness, then, we would be able to determine the optimum thickness of beryllium target for Am-Be miniature neutron source.

By using of ^{241}Am source with high activity, neutron yield increases for various applications.

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