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ARTICLE

# Calculation of Neutron Fluxes and Radiation Doses for Neutron Irradiator $^{226}$Ra-Be Using the MCNP5 Code

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**Abstract:** The irradiator $^{226}$Ra-Be unit available at the Physics Department of the Sciences Faculty, Damascus University, was simulated by using the MCNP code. Calculations of neutron fluxes and radiation doses were carried out.

Simulation results showed that neutron fluxes, in the energy ranges: thermal ($10^{-9}$-$10^{-6}$ MeV), fast (0.11-12.0 MeV) and medium ($10^{-5}$-$10^{-1}$ MeV), had approximately the same magnitudes in some channels (See text for the definition of these channels), where the flux percent was about (thermal 70.0%, fast 18.0 % and medium 12.0%). On the other hand, the flux percent in one particular channel was about (thermal 40.0%, fast 39.0% and medium 21.0%) and in another one (thermal %60.0, fast %26.0 and medium %14.0) with the presence of a plate of Cadmium whose thickness is 2 mm.

Absorbed radiation doses, in two channels, were calculated by using MCNP5 code and then compared with those measured experimentally by using thermoluminescent dosimeters (TLD). A satisfactory agreement between calculated and measured results was found. The relative differences were about 3.8% and 7.2% in these two channels, respectively.

**Keywords:** Neutron irradiator, $^{226}$Ra-Be, Thermoluminescent dosimeter, MCNP5 code.

## Introduction

Neutron irradiation following $^{9}Be(\alpha, n)^{12}C$ reaction (Q=5.7MeV) is the most commonly used irradiation, as it gives the highest neutron yield. For many years, Radium 226, with its decay products, has been used as an alpha emitter when long-lived sources are needed. More recently, the availability of isotopes, such as $^{239}$Pu, $^{227}$Ac and $^{241}$Am, has made it possible to produce neutron sources which have certain advantages, in particular a less intense gamma emission [1].

Radium 226 was the first alpha emitter used, because it was well studied as a radioactive source and it was relatively plentiful compared with other high energy alpha emitting isotopes. Polonium-210, which itself is a decay product of $^{226}Ra$, was also used as an isotope in early neutron sources. Another early radioactive material used in isotopic neutron sources was Actinium 227 ($^{227}Ac$), but because of its relative scarcity, this source was rarely used.

It was found that beryllium gave the best neutron yield of light elements; therefore, nearly all isotopic neutron sources after the 1950s were a combination of an alpha emitter and beryllium. However, some isotopic neutron sources used fluorine, boron or lithium instead of beryllium [2].

Neutron sources have been used in many areas, including research as well as nuclear and military industries, but the political situation and security concerns in recent years have led to the pursuit of reducing the use of neutron sources. Isotopic neutron sources have been replaced by sources of the type $(d,t)$ and accelerators as sources of neutrons, but in spite of the benefits

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of neutron sources of these two types, they are disadvantageous compared with isotopic neutron sources which give high-energy neutrons, in addition to large-scale electrical energy needed to operate neutron sources of the two mentioned neutron source types [1].

The aim of this work is to calculate the neutron fluxes in the channels of $^{226}$Ra-Be irradiator unit by using MCNP5 code, as well as to calculate the radiation doses in channels 5 and 12 and compare the resulting values with the values measured by using a thermoluminescent dosimeter.

## Materials and Methods

### Description of $^{226}$Ra-Be Irradiator Unit

The $^{226}$Ra-Be irradiator unit (PHYWE-Bedienungsanleitung-neutronenquelle 3.5 mCi-09080.01) is available at the Physics Department of the Sciences Faculty, Damascus University. It consists of:

**The Container:** It is made of steel (thickness = 4mm) in the form of a parallelepiped of dimensions 50×50×60cm$^3$ and covered with a rectangular steel cover. The container contains a moderator of paraffin (density = 0.904 g/cm$^3$). There are ten cylindrical channels for irradiation (for each channel: thickness = 1mm, diameter = 2.2 cm). There is also a gap to insert a Cadmium plate, which is used as an absorber of thermal neutrons.

Five channels (4,5,6,7,8) are distributed on the circumference of a circle of 10 cm radius around the source ($^{226}$Ra–Be). The other five channels are located differently away from the source at 15 cm (channel 9), 20 cm (channels 11 and 12) and 25 cm (channels 10 and 13), as shown in Table 1 and Fig. 1 [3].

TABLE 1. Some general properties of $^{226}$Ra-Be neutron irradiator

| Name | Shape | Dimensions | Notices |
|---|---|---|---|
| Steel container | Cubic | 60×70×50 cm$^3$ | Contains paraffin moderator |
| Cadmium | Plate | 40×10×0.2 cm$^3$ | On axis Ox |
| $^{226}$Ra-Be source | Cylindrical tube | 7×2 cm$^2$ | - |

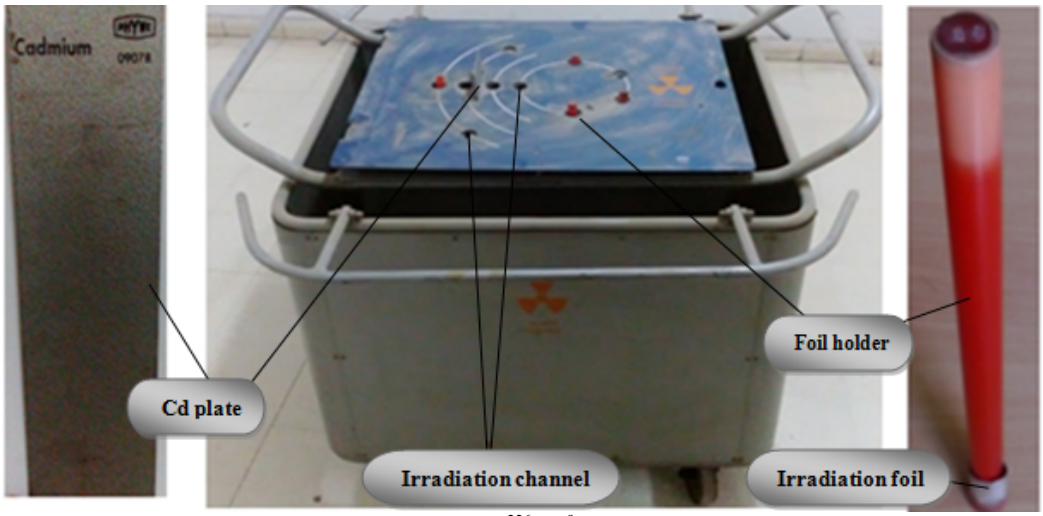



FIG. 1. Neutron irradiator ($^{226}$Ra-$^{9}$Be) unit (PHYWE).

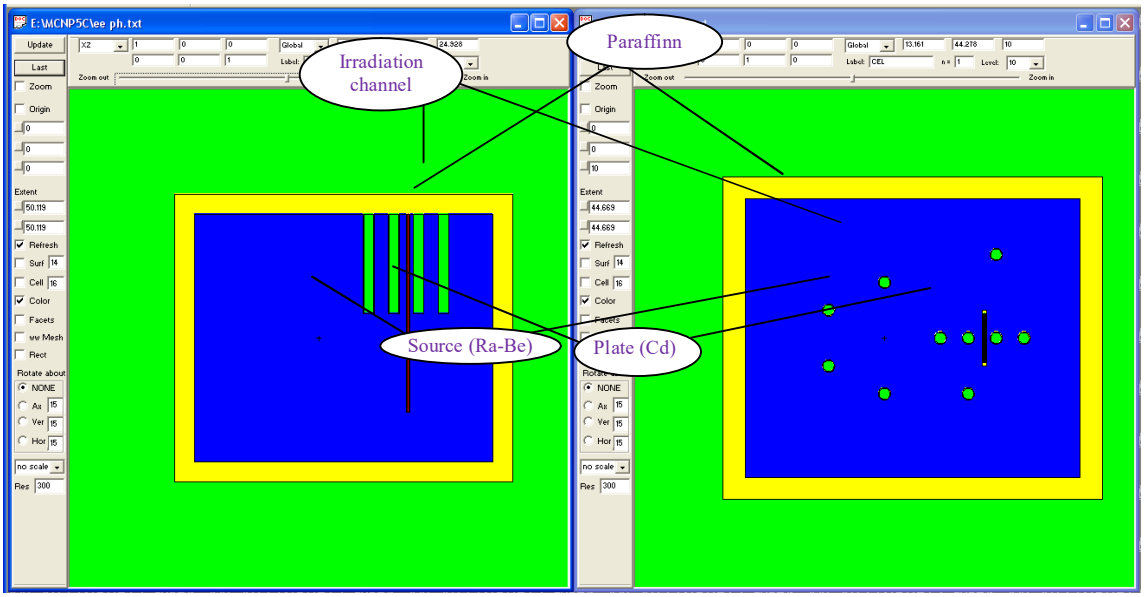
**Radium-Beryllium Source:** The $^{226}$Ra-Be neutron source is a homogeneous mechanical mixture of an $\alpha$-emitting nuclide $^{226}$Ra with the light element $^{9}$Be. The mixture ratio Be to Ra is 1:5. The mass of the mixture is 3.5 mg. It is enclosed in a two-wall cylindrical tube from nickel then from steel, with a length of 7 cm and an outer diameter of 2 cm. The two-wall cylinder is placed in a cylinder from lead with a length of 7 cm and a diameter of 4 cm. The flux rate of this source is up to 9.09x10$^4$n/s, in view of the fact that the decay of $^{226}$Ra leads to the alpha-emitting progeny $^{222}$Rn and $^{210}$Po, producing alpha particles which contribute to the total product of neutrons by a ratio of 6/7≈0.86. The yield of the source $^{226}$Ra–Be may reach 2.0x10$^7$n/s for each 1 Ci from Radium. The source $^{226}$Ra–Be can be distinguished by continuous spectra of neutrons with an average ranging from 4 to 5 MeV.

The relatively long-lived Radium 226, with its decay products, form a group of five $\alpha$-emitting isotopes with energies ranging from 4.8 to 7.7 MeV and an average energy of 5.8 MeV. These energies are enough for surmounting the potential barrier of a beryllium nucleus which is approximately 4.0 MeV. However, alpha particles interact with the atomic electrons of beryllium, so that they lose a part of their energy and slow down to below 4.0 MeV. Therefore, not all alpha particles can excite the nuclear interaction $(\alpha, n)$ in beryllium; only (1-1.5)×$10^4$ particles can penetrate beryllium nuclei [4].

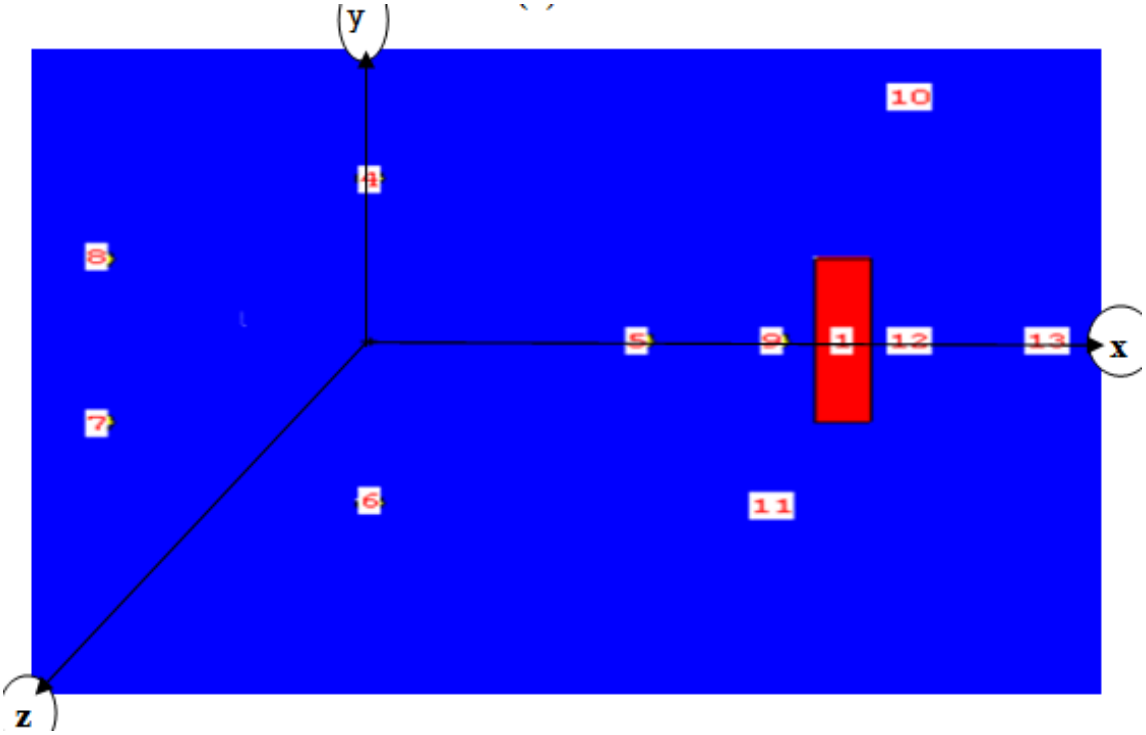
## Simulation of $^{226}$Ra-Be Unit Using MCNP Code

MCNP5 is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron or coupled neutron, photon and electron transport. It can also be used for three-dimensional, time-dependent general geometry. This code is widely used around the world for many radiation protection and shielding applications [5]. MCNP, V5 has various new user features. These include: improved photon physics, neutral particle radiography, enhancements and additions to variance reduction methods, new source options, improved parallelism support (PVM, MPI, Open MP) and new nuclear and atomic data libraries [6].

The neutron source $^{226}$Ra-Be was simulated as a point source located at the center of coordinates (0,0,0). The source definition card (SDEF) was used to describe the source $^{226}$Ra-Be. The neutron spectrum of $^{226}$Ra-Be was used from literature [1]. The container, around the source, was filled with paraffin except in the defined channels (Fig. 2). In addition, Table 2 shows the physical properties of the irradiator as used in the simulation by MCNP5 code.




(a)




(b)

FIG. 2. a) Diagram for the $^{226}$Ra–Be irradiator unit using Vized program. b) Irradiation channels and Cadmium plate [3].

TABLE 2. Physical properties of irradiator as used in the simulation by MCNP5 code

| Name | Shape | Position relative to $^{226}$Ra-Be (cm) | Notice |
|---|---|---|---|
| Channel number | Irradiator channel with a diameter of 2cm (according to its number in the simulation) | | |
| 4 | Cylinder | 10 | On axis Oy |
| 5 | Cylinder | 10 | On axis Ox |
| 6 | Cylinder | -10 | On axis Oy |
| 7 | Cylinder | -10 | On axis xOy |
| 8 | Cylinder | -10 | On axis xOy |
| 9 | Cylinder | 15 | On axis Ox |
| 10 | Cylinder | 25 | In plane xOy |
| 11 | Cylinder | 20 | In plane xOy |
| 12 | Cylinder | 20 | On axis Ox |
| 13 | Cylinder | 25 | On axis Ox |
| Material carrier | Cylinder | radius (0.5 cm) and length (30 cm) | The length inside paraffin = 20 cm |
| Source $^{226}$Ra-Be | | | |
| $^{226}$Ra-Be Mixture | Cylinder | The mixture is in a pair-wall cylindrical tube from nickel then from steel, length = 7 cm and diameter = 2 cm. Ratio of $^{226}$Ra to $^{9}$Be = 1/5, mass= 3.5 mg | |
| Cylinder of steel | Cylinder | Length = 7 cm, diameter = 2 cm | |
| Source carrier | Cylinder | Length = 18 cm, diameter = 4 cm | |

## Results and Discussion

### Calculation of the Neutron Flux inside the Irradiator Channels

Using the F4 (flux averaged over a cell (particles/cm$^2$)) card in MCNP5, the neutron flux was calculated in different channels, where the neutron flux is proportional to path grand total K with a length $L_k$ for neutrons with energy $E_j$ across the channel volume as illustrated in Eq. (1):

$$\Phi_j \propto \frac{1}{V}\sum_{k=1}^{K} L_k(E_j). \qquad (1)$$

The neutron flux $\Phi_j(cm^{-2})$ is expressed using F4 card, as in Eq. (2) [5]:

$$F_4 = \int_V \int_t \int_E \Phi(\vec{r},E,t)\,dE\,dt\,\frac{dV}{V}. \qquad (2)$$

Fig. 3 shows the composition of the neutron flux (thermal, medium and fast) at the channels of the irradiator unit as obtained from MCNP5 code. Obviously, the neutron flux decreases away from the source. In addition, the thermal neutron flux dominates at all channels, except for those which contain Cadmium plates.

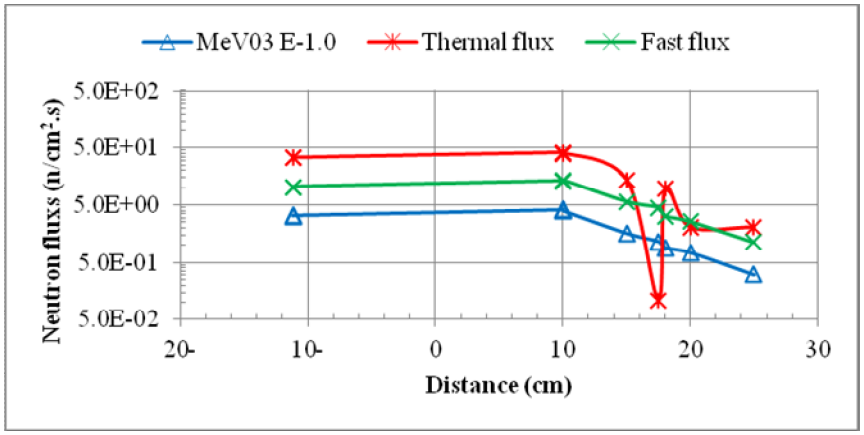



FIG. 3. Neutron flux at irradiator channels according to their distance from the source.

The ratio "thermal to fast":

- equals, in average, 3.17±0.03 at channels 4,5,6,7 and 8;
- decreases to 0.024±0.01 at Cadmium plate;
- decreases to 0.8±0.02 and 1.83±0.01 in channels 12 and 13, respectively, as these channels are located behind the Cadmium plate,
- equals 2.41±0.03, 3.01±0.04 and 2.95±0.02 at channels 9,10 and 11, respectively.

Figs. 4 to 8 show the changes of neutron flux versus the neutron energy at the channels and the Cadmium plate.

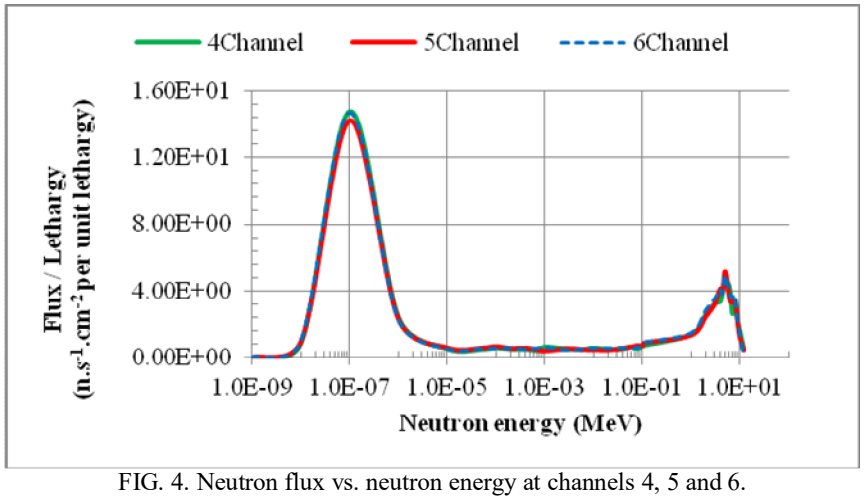



FIG. 4. Neutron flux vs. neutron energy at channels 4, 5 and 6.

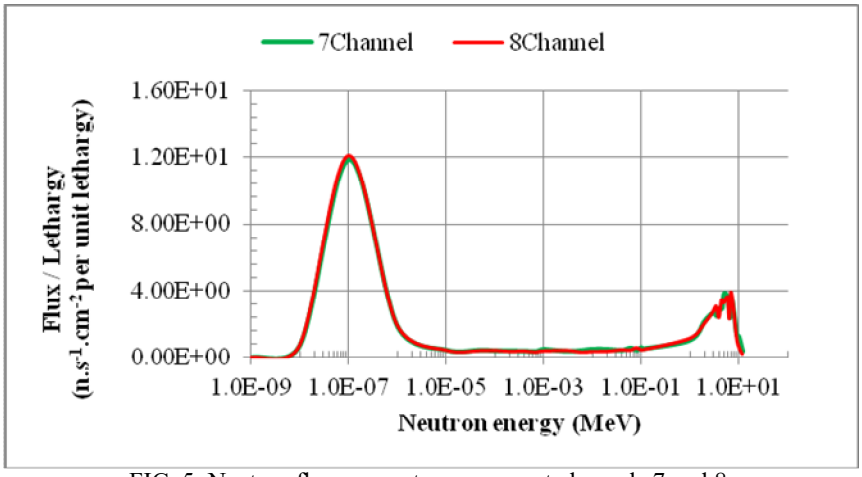



FIG. 5. Neutron flux vs. neutron energy at channels 7 and 8.

TABLE 3. The neutron flux (thermal, medium, and fast) at channels 4, 5, 6, 7 and 8.

| Channel number | ($10^{-9}$ to $10^{-6}$) MeV | ($10^{-5}$ to $10^{-1}$) MeV | (0.11 to 12) MeV |
|---|---|---|---|
| 4, 5, 6 | 71.0% | 10.7% | 18.3% |
| 7, 8 | 72.4% | 10.4% | 17.9% |

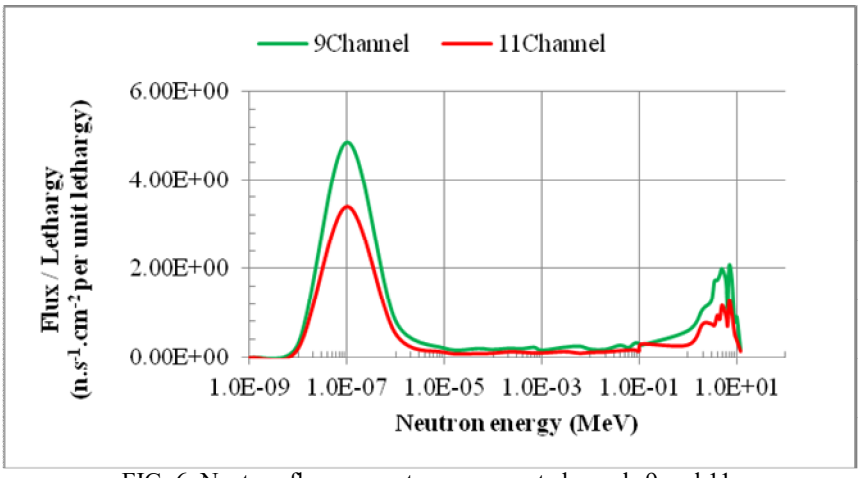



FIG. 6. Neutron flux vs. neutron energy at channels 9 and 11.

TABLE 4. The neutron flux (thermal, medium and fast) at channels 9 and 11.

| Channel number | ($10^{-9}$ to $10^{-6}$) MeV | ($10^{-5}$ to $10^{-1}$) MeV | (0.11 to 12) MeV |
|---|---|---|---|
| 9 | 66.1% | 12.1% | 21.8% |
| 11 | 70.4% | 10.8% | 18.8% |

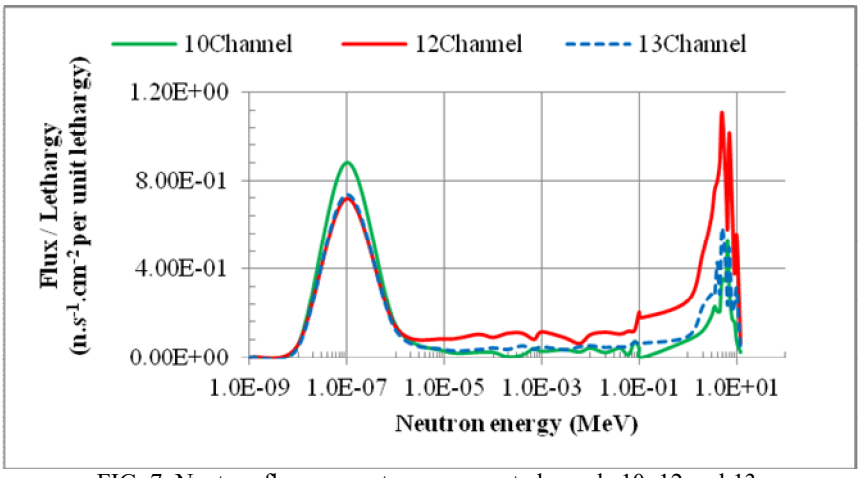



FIG. 7. Neutron flux vs. neutron energy at channels 10, 12 and 13.

TABLE 5. The neutron flux (thermal, medium and fast) at channels 10, 12 and 13.

| Channel number | ($10^{-9}$ to $10^{-6}$) MeV | ($10^{-5}$ to $10^{-1}$) MeV | (0.11 to 12) MeV |
|---|---|---|---|
| 10 | 72.0% | 10.0% | 18.0% |
| 12 | 40.0% | 21.0% | 39.0% |
| 13 | 60.0% | 14.0% | 26.0% |

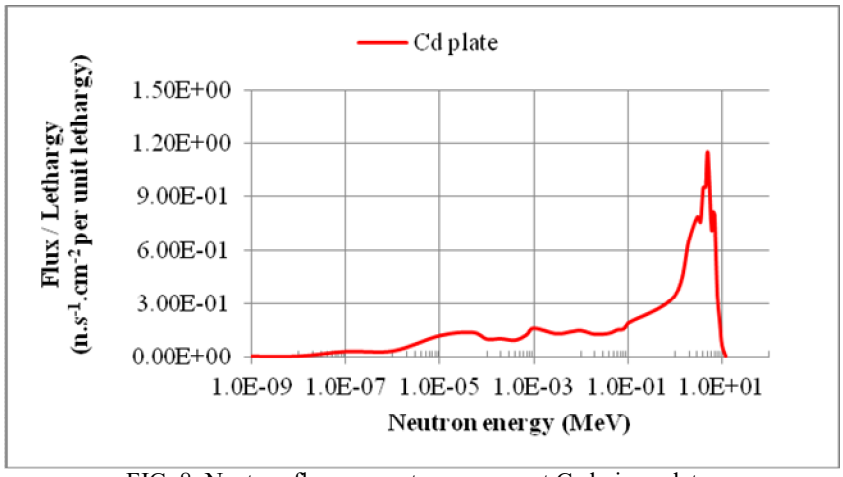



FIG. 8. Neutron flux vs. neutron energy at Cadmium plate.

Fig. 8 shows that the neutron flux at the Cadmium plate is composed generally from fast neutrons in the range (0.11-12.0) MeV with a ratio of 63.0% and in the energy range ($10^{-9}$ - $10^{-6}$) MeV with a ratio of 3.0%, while a ratio of 34.0% is formed by neutrons in the energy range ($10^{-5}$ -$10^{-1}$) MeV.

As shown in Figs. (9, 10 and 11), the Cadmium plate plays an effective role in reducing the total neutron flux from the Ra-Be source at channels 12 and 13 with a ratio of 40.0% and 17.0%, respectively. On the other hand, the absence of the Cadmium plate will lead to increase the thermal neutron flux ratio to 64.0% and 66.0% (in channels 12 and 13, respectively) and reduce the fast neutron flux to 24.0% and 22.0% (in channels 12 and 13, respectively). Also, the short distance between the source and some channels and the absorption in the moderator reduce the neutron flux in close channels.

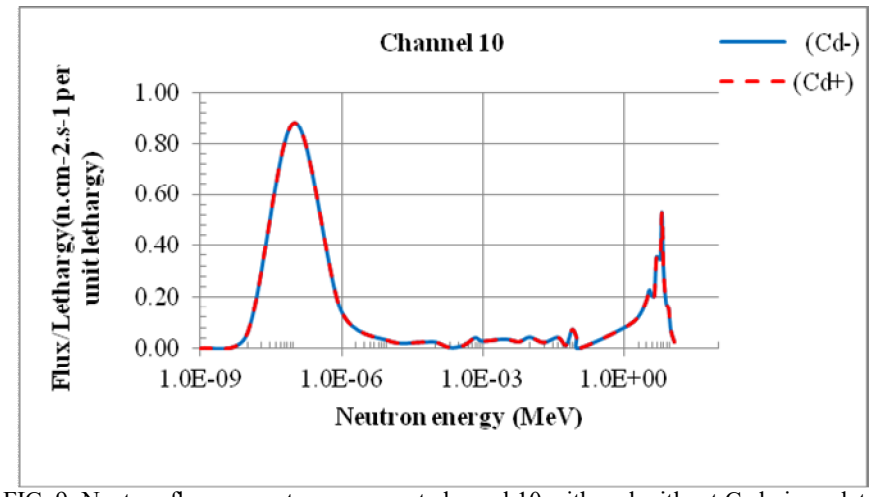



FIG. 9. Neutron flux vs. neutron energy at channel 10 with and without Cadmium plate.

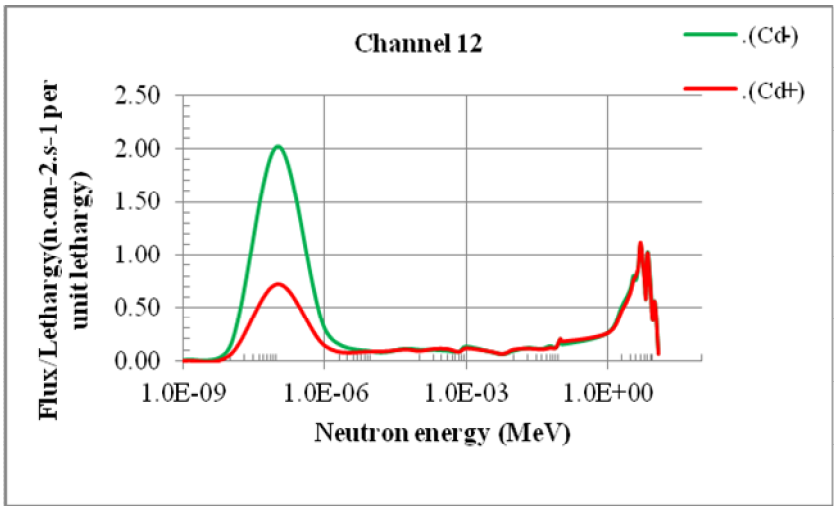



FIG. 10. Neutron flux vs. neutron energy at channel 12 with and without Cadmium plate.

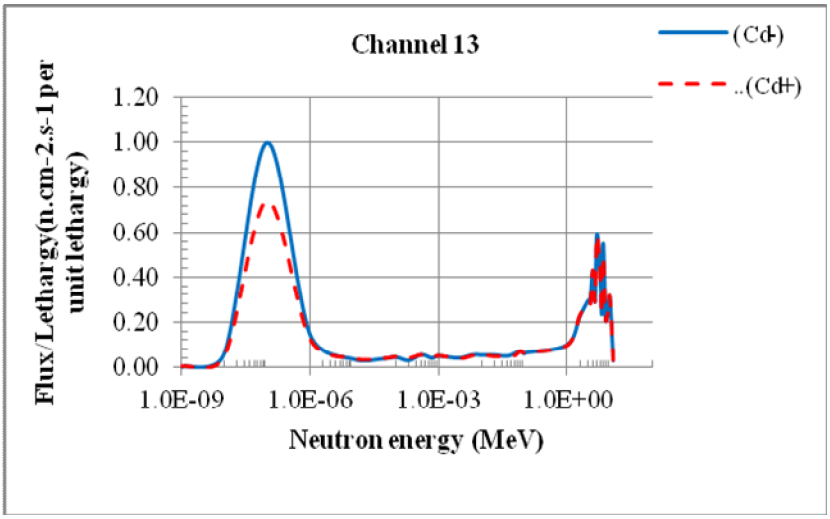



FIG. 11. Neutron flux vs. neutron energy at channel 13 with and without Cadmium plate.

### Calculation of the Dose at the Surface

The card F5 (flux at a point or ring detector (particles/cm$^2$)) was used to calculate the neutron and photon doses at points located on the upper surface of the irradiator unit. Fig. 12 shows the dose values at points located on the irradiator. It can be noticed that both neutron and photon doses have a maximum value near the point x=0 on axis Ox, then the dose decreases as the neutron flux decreases away from the source. The asymmetry of the dose distribution is due to the fact that the source is not in the center of the container.

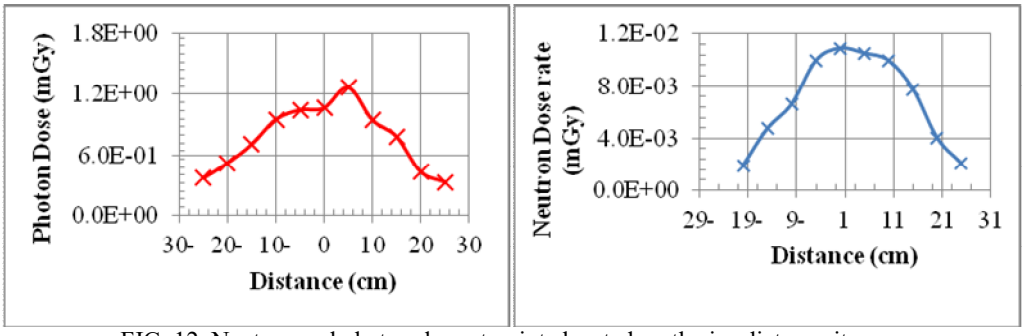



FIG. 12. Neutron and photon dose at points located on the irradiator unit.

The radiation (neutron and photon) dose on the surface of the irradiator unit was calculated by using MCNP5 code for radiation protection purposes. The results (Table 6) revealed that the dose received by workers, from this unit, is very low in comparison with the recommended annual dose limit for radiation workers (20 mSv/y).

TABLE 6. Neutron and photon doses on the upper surface of the irradiator ($^{226}$Ra-Be) unit by using MCNP5 code.

| Particles | Effective dose (μSv) |
|---|---|
| Neutrons | $3.43\times10^{-7}$ Rem/h=0.03 mSv =30μSv |
| Photons | $1.64\times10^{-5}$ Rem/h=0.6 mSv=60 μSv |

### Calculation of the Dose at Channels 5 and 12

It was supposed that the total radiation dose is composed of a neutron dose (scattering of fast neutrons on material moderator) and a photon dose (resulting from the absorption of thermal neutrons in hydrogen nuclei).

The card F6 (energy deposition averaged over a cell (MeV/g)) was used for the calculation of radiation doses at channels 5 and 12 only. Table 7 shows the calculated radiation doses at channels 5 and 12 by using MCNP5 code.

TABLE 7. Radiation doses at channels 5 and 12 calculated by using MCNP5 code.

| Channel number | Neutron dose | | Photon dose | | Sum (mGy) |
|---|---|---|---|---|---|
| | J/kg | mGy | J/kg | mGy | |
| 5 | 1.67×10-3 | 23.31±0.01 | 1.90×10-3 | 16.14±0.02 | 39.45 |
| 12 | 1.15×10-4 | 2.66±0.03 | 3.33×10-4 | 4.66±0.01 | 7.32 |

### Measurement of the Radiation Dose at Channels 5 and 12 Using Thermoluminescent Dosimeters (TLDs)

Thermoluminescent dosimeters, TLD-700 type, were used to measure the absorbed dose in channels 5 and 12. Four dosimeters were placed inside each channel, on a cylindrical plastic carrier, for one month. Two dosimeters were placed outside the irradiator to measure the background.

The calibration of TLD chips was carried out using a $^{137}$Cs source. The chips were read using Harshaw reader, Model 4500 [7]. The results at channels 5 and 12 are given in Table 8 to compare with those from simulation. As shown, there is a good agreement between both measured and simulated results.

TABLE 8. Measured radiation doses at channels 5 and 12 by using MCNP5 code and TLDs.

| Channel number | Total dose by TLD (mGy) | Total dose by MCNP(mGy) | Difference percentage |
|---|---|---|---|
| 5 | 41.00±0.05 | 39.45±0.02 | 3.8% |
| 12 | 7.87±0.8 | 7.32±0.03 | 7. 2% |

## Conclusions

The results presented in this work show that the calculation of neutron fluxes in $^{226}$Ra–Be unit is very important and useful to understand the neutron flux in each channel for neutron activation analysis. Also, the calculated doses on the surface of the irradiator unit prove that it is safe from the radiation protection point of view. The simulated radiation dose results were validated by carrying out measurements with TLDs, where good agreement was found between the measured and the simulated values for channels 5 and 12.

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