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Design of Irradiation Channels in Radium-Beryllium ^{226}Ra-Be Neutron Irradiation Facility Usmba-Fsdm-Fez Morocco

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ABSTRACT

Using MCNP6 to determine axial and radial neutron fluxes of the seven channels irradiation sites of 3 mCi of radium is tended for physics experiments on artificial radioactivity and irradiation facility in nuclear physics laboratory at (FSDM-USMBA) Fez Morocco. Ra-Be source in the interior of the shielded laboratory is sealed emanation-tight into a brazed nickel and it's eccentrically off the axis of the lead cylinder. Lead thickness and distance shield are so designed that the gamma-dose rate from the source remains small enough in all directions. The seven irradiation channels are arranged vertically within the paraffin block.

Introduction

Neutron activation analysis (NAA) is a sensitive non-destructive method for determining the elemental composition of a simple. Nuclear reactors are frequently used for NAA because of their high neutron flux compared to the neutron source isotopes. The development of nuclear instrumentation to perform in neutron activation analysis using thermal, epithermal and fast neutrons, can be useful for studies elements in a variety of simples matrices outside the reactor premises. Installation of the neutron source ^{226}Ra-Be facility installed at department of physics Faculty of Science University of sidi Mohamed ben Abdellah fez, Morocco "Figs.1" , by company LEYBOLD-HEREUS GMBH, Federal Republic of Germany by license No.NW 8/66[1]. The seven irradiation channels (each 30 cm long, 3 cm of diameter) are arranged vertically within the paraffin block Figure 1. The sheet-metal pot with source, lead cylinder and paraffin wax is surrounded by a distance shield of 49.5 cm diameter joined to in by a sold steel construction irradiation channels Nos. 1, 2, 3 and 4 are in a circle of 7 cm radius about the Ra-Be source, irradiation channels Nos. 5 and 6 are 14 cm away the source and irradiation channel No. 7 is about 20 cm from the preparation "Figs.2, 3".

The shielded source of neutrons containing a radium-beryllium source is intended for physics and chemistry experiments on artificial radioactivity for students bachelors master's and PhD students nuclear study , the shield easy to portable besides producing stable neutron flux as their size design.

The objective of the work is to determine the neutron flux distribution axial and radial flux in seven irradiation sites to ensure increased utilization of the facility for research and education of nuclear physics for Moroccan students and outside.

Methodology

Neutron sources are characterized by a number of factors which include intensity (number of neutron emitted per seconds), energy distribution, Angular distribution, neutron polarization and mode of emission (continuous or pulsed), Neutron source are used in nuclear physics research and in practical applications such as neutron logging and neutron diffraction analysis, and classified into three groups nuclear fission reactors, radioisotopes, and particle accelerators.

Figure 1. picture of Source Ra-Be in USMBA-FSDM-Fez Morocco of Ra-Be neutron source assembly

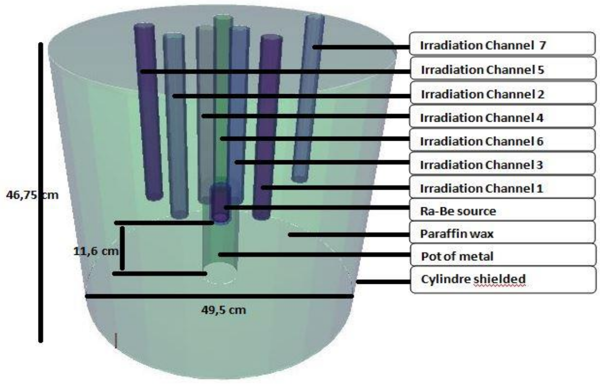

Figure 2. Schematic of Ra-Be neutron source assembly

There are two main types of radioisotope neutron sources, direct and indirect. Direct refers to radioisotope sources that emit neutrons in their natural decay processes. Californium-252 is the most widely used direct radioisotope neutron source. ^{252}Cf

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has a half-life of 2.645 years, and 3.09 % of decays are by spontaneous fission [3]. . Indirect radioisotope neutron sources refer to sources that rely on a charged particle emitting radionuclide and a stable target nuclide to produce neutrons through a nuclear reaction.

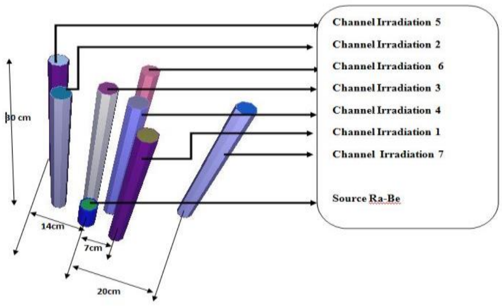

Figure 3. Schematic of Ra-Be neutron source assembly

The most common indirect radioisotope neutron source is the Ra-Be source. There, ^{226}Ra with a half-life of 1,599 years has two major alpha-particles located at 4.784 MeV (94.5 %) and 4.601 MeV (5.5 %) [4], the neutron producing reaction is:

$$^{4}_{2}\alpha + {}^{9}_{4}Be \rightarrow {}^{12}_{6}C + {}^{1}_{0}n$$

^{226}Ra (α, n) ^{9}Be which is a mechanical mixture in weight of beryllium to radium with energy spectrum up to about 13 MeV shows is not a mono energetic source [5] . In this work, the purpose is the modeled neutron irradiation facility and determining the neutron flux profile of the seven irradiation sites using Monte Carlo N-Particle (MCNP-6) code[6]. In this manner, the viability of the irradiation facility can be controlled to increase performance and use.

Ra-Be activity of 3 mCi of ^{226}Ra, neutron yield of the present investigation $5\ 10^{4}$ ns^{-1}, Ra powder is compressed into double emanations sealed in a hard-soldered nickel and welded in a cover fraction of weight composition is : C (0.004 %); Mn (1.59 %); P (0.011 %); S (0.008 %); Si (0.37 %); Cr (16,96 %); Ni (3.61 %); Mo (2.29 %); Fe (65.16 %) [2] “Fig. 3”, It is located in the interior of a compact lead cylinder, which is closed by a bolted and sealed lead plug, The remaining inside the lead cylinder space is filled with activated carbon[1] “Fig.4”.

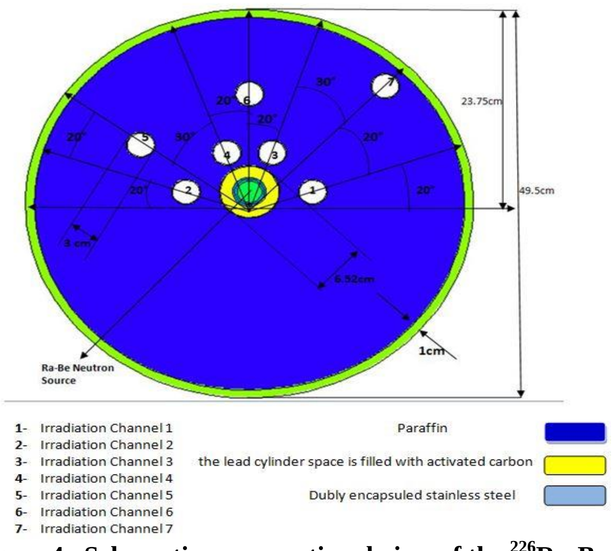

Figure 4. Schematic cross-sectional view of the ^{226}Ra-Be neutron source assembly

Channels irradiation of Aluminum each 30 cm long, 3 cm of diameter the following atomic fraction composition: 0.999999819 aluminum, 1.005 10^{-7} iron, 0.700 10^{-7} silicon, 0.025 10^{-7} copper, 0.015 10^{-7} zinc, 0.015 10^{-7} manganese, 0.025 10^{-7} magnesium, 0.025 10^{-7} titanium [7] “Figs.2.3”. MCNP-6 products by Los Alamos National Laboratory is a general-purpose, continuous-energy, generalized-geometry, time dependent, Monte Carlo radiation-transport code designed to track many particle types over broad ranges of energies. is the result of five years of effort by the MCNP-5 and MCNP-X code development teams. These groups of people, residing in the Los Alamos National Laboratory's (Initial MCNP-6 Release Overview MCNP-6 Version 1.0, Los Alamos National Laboratory report [6] .

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Theory

The neutron fluxes investigated in this work are thermal, epithermal and fast according to the MCNP input file prepared, the thermal neutron flux energy ranges from (0 to 6.25 10^{-1}) MeV, the epithermal neutron flux energy ranges from (6.25 10^{-7} to 8.21 10^{-1}) MeV and the fast neutron flux ranges from (8.21 10^{-1} to 20.00) MeV. In this work, making 50 10^{6} particle histories monitored in the MCNP6 simulation. Radial and axial tallies were created in the seven irradiation channels for the source design. The axial neutron flux was determined by dividing the irradiation channels into 10 at an interval of 3 cm for each channel 1 to 7. The radial neutron flux of the seven irradiation channels was determined by creating four cylindrical tubes along of the diameter. The radii of the cylinders are 0.26 cm for each channel. Point wise cross-section data were used for neutrons; all reactions given in a particular cross-section evaluation (such as ENDF/B-VI/B-VII.0/VI.8) were accounted for. Thermal neutrons were described by both the free gas and S (α,β) models (X-6 Monte Carlo Team, 2013)[6]. The average neutron flux in the volume of each of the division of the channels was determined by using F4: N tally card or cell flux tally card as described in the MCNP6 manual (X-6 Monte Carlo Team, 2013) [6].

Results and discussions

The results performed during the calculated with the MCNP6 code from the tallies retrieved from the output file are normalized with formula (1) using the MS-Excel and interpreted into graphs as shown in figures (5 to 11 axial flux and 12 to 18 radial flux) :

$$\emptyset = \frac{\text{Tally} \times \text{Source strength}}{\text{Volume of select cell}} \quad \left(\frac{n}{cm^{2}s}\right) \tag{1}$$

The MCNP-6 simulation was run for the neutron flux profile in the whole irradiation channel. The thermal, epithermal and fast neutron fluxes were determined including their uncertainties as evaluated by MCNP-6 code.

Profit of flux axial

The flux thermal, epithermal and fast growing exponentially axially from the bottom of the source with a maximum increase in the center of the source, and starts to pour down from the exponentially source to very low values along of irradiation channel. The channels (1, 2, 3 and 4) the neutron flux more intense and the maximum intensity of the flux thermal, epithermal and fast respectively 2300, 1760 and 1280 n/cm^{2}s^{-1} are almost equal in each channels irradiation 1 to 4, and then the tubes 5 and 6 the maximum intensity of the flux thermal, epithermal and fast respectively 1200, 300 and 80 n/cm^{2}s^{-1}, then the neutron flux decreases almost half the channel 7 at a flow rate near zero fast neutrons respectively 150, 25 and 5 n/cm^{2}s^{-1}.

The average thermal, epithermal and fast fluxes recorded in the whole of the irradiation site 1, 2, 3,4,5,6 and 7 are shown in

"Table 1" and percentage of neutron fluxes in each channel shown in "Table 2". The average thermal flux in the irradiation site 1, 2, 3 and 4 is tow times that of irradiation sites 5 and 6 and 12 times that of irradiation 7. This indicates that as the distance from the source increases the thermal neutron flux decreases as a result of leakage and absorption. The average epithermal flux in the irradiation site 1, 2, 3 and 4 is four times that of the irradiation site 5 and 6 and big values in irradiation channel 7. The average fast flux in site 1, 2, 3 and 4 is 10 times that of irradiation channel 5 and 6.

The neutron flux distribution along of the irradiation channels are as shown Figs. 5,6,7,8,9,10 and 11 for irradiation site 1, 2, 3,4,5,6 and 7 respectively.

This validate the fact that as distance from the source in the paraffin increases the percentage thermal neutron flux increases whereas the percentage epithermal and fast neutron fluxes decreases accordingly as shown in Table 2.

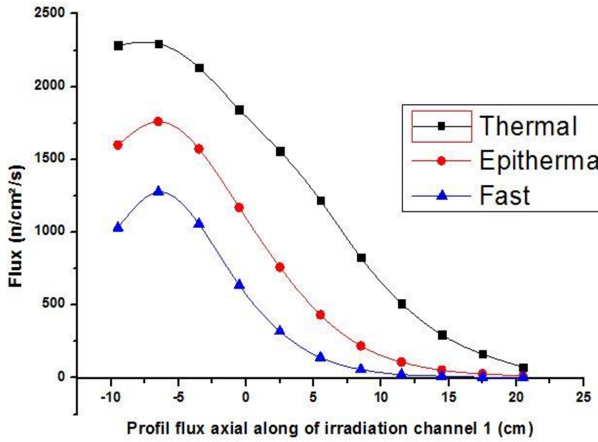

Figure 5. Axial Profile Flux distribution of channel 1

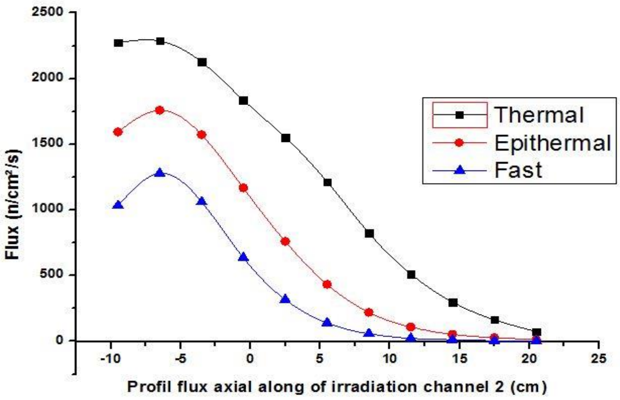

Figure 6. Axial Profile Flux distribution of channel 2

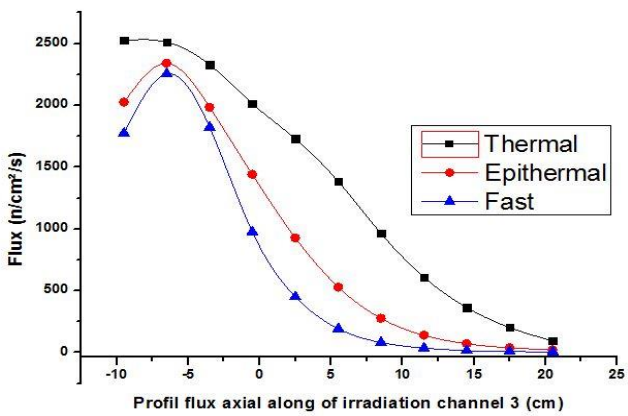

Figure 7. Axial Profile Flux distribution of channel 3

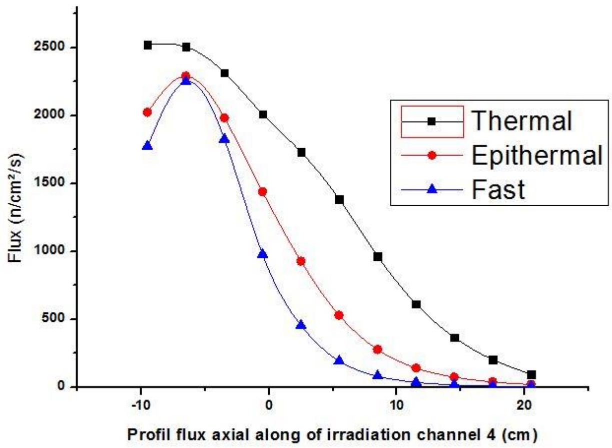

Figure 8. Axial Profile Flux distribution of channel 4

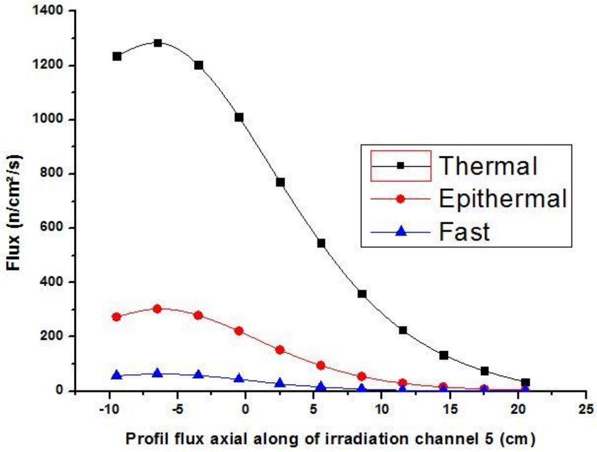

Figure 9. Axial Profile Flux distribution of channel 5

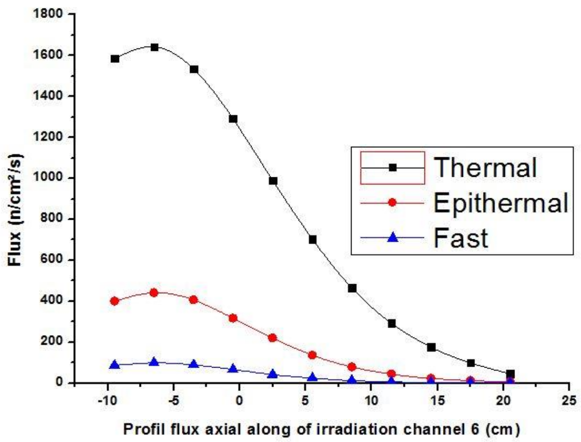

Figure 10. Axial Profile Flux distribution of channel 6

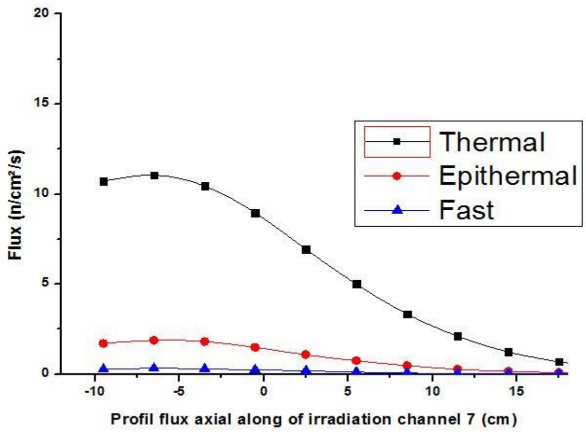

Figure 11. Axial Profile Flux distribution of channel 7

Table 1. The average thermal, epithermal and fast neutron in each irradiation site as obtained in this work

Irrad. Channel	Neutron flux (n/cm^2s^{-1})		
	Thermal	Epithermal	Fast
Irrad. 1	1200±7	702±07	415±8
Irrad. 2	1200±7	701±7	415±8
Irrad. 3	1340±7	890±7	692±8
Irrad. 4	1340±7	885±7	692±7
Irrad. 5	626±0.2)	131±2	25.5±0.7
Irrad. 6	803±2	189±2	39.1±0.8
Irrad. 7	74.7±0.12	11.9±0.09	1.9±0.12

Table 2. Percentage of neutron fluxes in each irradiation channel (%)

Irrad. Channel	Percentage of neutron fluxes (%)		
	Thermal	Epithermal	Fast
Irrad. 1	51.81	30.28	17.89
Irrad. 2	51.7	30.3	18.01
Irrad. 3	51.69	30.31	18.01
Irrad. 4	51.81	30.28	17.89
Irrad. 5	80.00	16.70	3.30
Irrad. 6	77.80	18.40	3.80
Irrad. 7	84.40	13.50	2.14

Profile radial flux

The four cylindrical tubes modeled along the diameter of the irradiation sites 1, 2, 3, 4, 5, 6 and 7 was simulated for the average neutron flux in each volume of each cylinder. The results were normalized and shown in figs. 12, 13, 14, 15, 16, 17 and 18 for irradiation sites 1, 2, 3, 4, 5, 6 and 7, respectively.

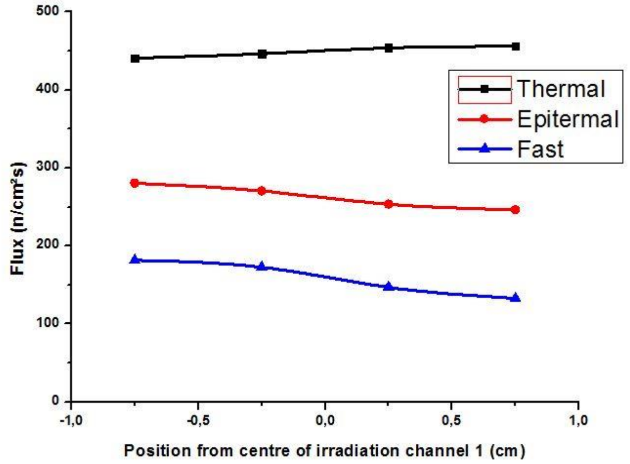

Figure 11. Radial Profile Flux distribution of channel 1

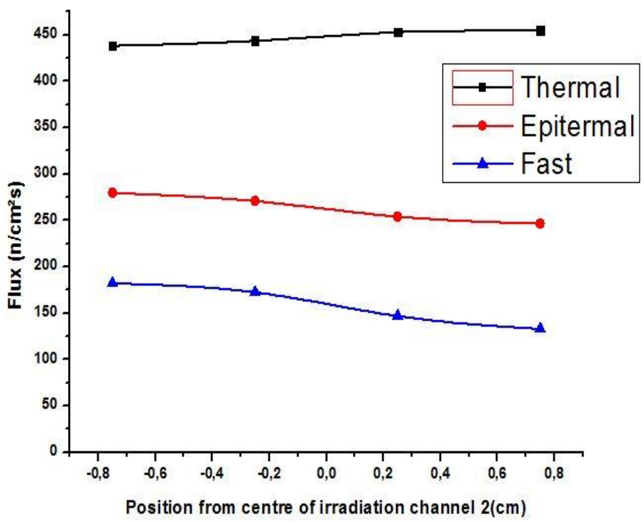

Figure 12. Radial Profile Flux distribution of channel 2

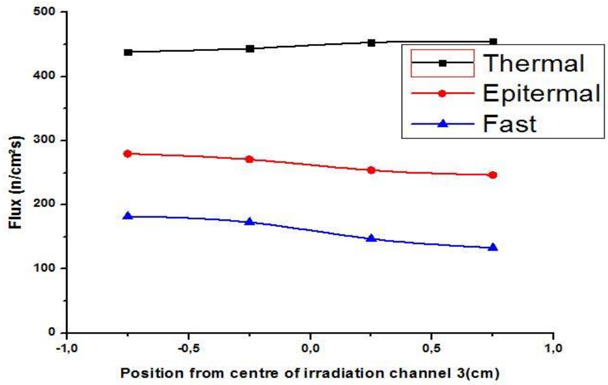

Figure 13. Radial Profile Flux distribution of channel 3

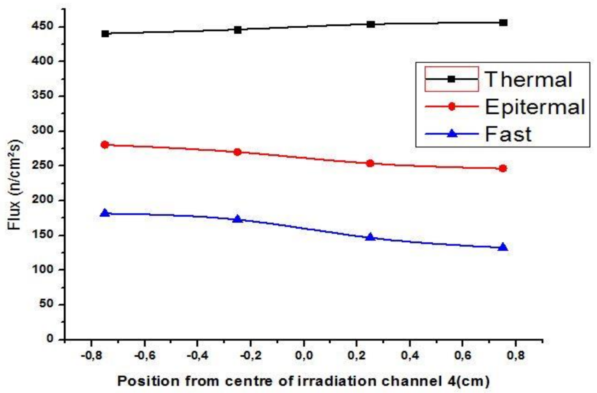

Figure 14. Radial Profile Flux distribution of channel 4

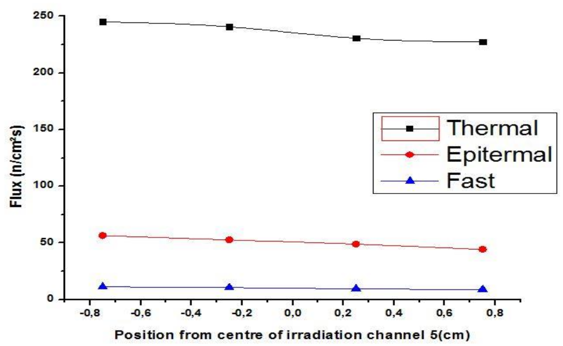

Figure 15. Radial Profile Flux distribution of channel 5

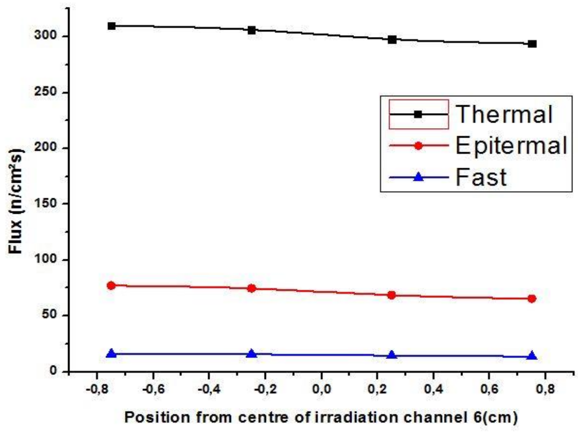

Figure 16. Radial Profile Flux distribution of channel 6

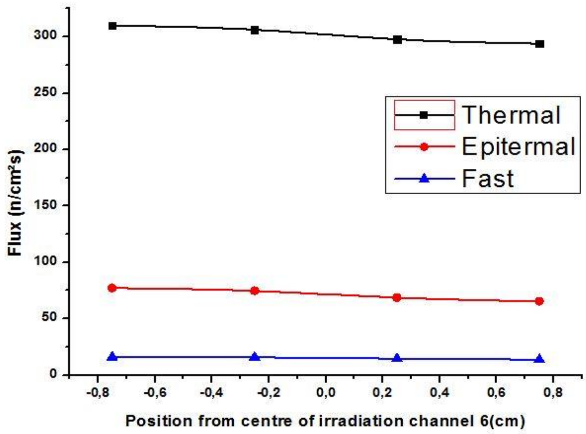

Figure 17. Radial Profile Flux distribution of channel 7

It can be observed that the fluxes of the irradiation site 1, 2, 3 and 4 are almost uniform, and decrease at 22 n/cm^{2}s^{-1} along the diameter for the site 1, 2, 3 and 4 in the irradiation site 5, 6 the rate of decrease was 10 n/cm^{2}s^{-1} along the diameter, and decrease at 1, 51 n/cm^{2}s^{-1} along the diameter for the site 7. The rate of neutron flux decrease between irradiation channels 1, 2, 3, 4 and 5, 6 is (2.2) times and 14, 56 times in channel 7.

Conclusions

The installation of the neutron source Ra-Be modeled and simulated using the code MCNP6, the profile of axial and radial neutron flux was determined in seven irradiation channels, levels of flux and a maximum of four very close channels of the source, and then average the two channels (5 and 6), but in channel 7 is almost at very low values. It can be concluded that as the distance from the source increases the neutron fluxes levels decreases and as the distance from the source increases the percentage of the average thermal flux increases, but that of the average epithermal and fast neutron flux decreases as a result of moderation by the paraffin. The results are compared very well with similar work using the same method and experimental done with Mr. Asamoah [2] and Zevallos-Chavez and Zamboni[8] which gives a validation for calculating made in this work. Ra-Be source used for the practical work of neutron activation analysis installed at the Faculty of Science University of Sidi Mohammed Ben Abdelahh Fez, cannot be the monopoly University fez but also for all national universities and outside for research and education.

Acknowledgment

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