Experimental and MCNP Studies of Paraffin and Polyethylene in Neutron Moderation and BF_3 Detector Efficiency

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***Abstract* — The Nuclear and Energy Research Institute – IPEN, offers post-graduate programs, namely: Nuclear Technology - Applications (TNA), Nuclear Technology - Materials (TNM), Nuclear Technology - Reactors (TNR). The Institute programs mission is to form expert technicians and engineers with a strong knowledge in their discipline to work in the nuclear area.**

The course: "Theoretical Fundaments and Practices of the Instrumentation used in Nuclear Data Acquisition" covers the use of laboratory nuclear instrumentation and the accomplishment of experiments to obtain nuclear parameters.

One of these experiments is object of this work: "Experimental and MCNP Studies of Paraffin and Polyethylene Neutron Moderation and BF_3 Detector Efficiency".

Neutrons are uncharged particles and, therefore, cannot be detected by Coulomb interactions. Thus, the detector assembly used must contain some kind of material with high cross section for interaction with neutrons, called converters. A boron trifluoride (BF_3) detector was used in this experiment to detect neutron in real time.

However, the response of this arrangement varies according to the energy range of incident neutrons. Their efficiency for thermal neutrons is above 90%, but, this result decreases, significantly, for neutrons of energy greater than 0.5 eV. The neutron moderation and, consequently, its energy variation were obtained by interposing different thicknesses of moderator material (Paraffin or Polyethylene) between the source and the detector.

The detector efficiency and the optimal thickness of the moderators were determined experimentally and through computer simulations using the MCNP code. This code uses the Monte Carlo method to simulate radiation transport in matter.

***Keywords*: neutron irradiator, decay constant, nuclear course**

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I. Introduction

Boron Trifluoride (BF_3) nuclear detector are used to real time neutron flux measurements, being the internal detector walls lined with boron.

$$^{10}B + ^{1}n \rightarrow \left[^{11}B\right] \rightarrow ^{4}He + ^{7}Li + \gamma(0.48 MeV) + 2.31 MeV \quad (1)$$

$$^{10}B + ^{1}n \rightarrow \left[^{11}B\right] \rightarrow ^{4}He + ^{7}Li + 2.79 MeV \quad (2)$$

Figure 1 shows the detector response (spectrum), there are two peaks corresponding to equations (1) and (2) with probabilities of 94% and 6% respectively [1,2].

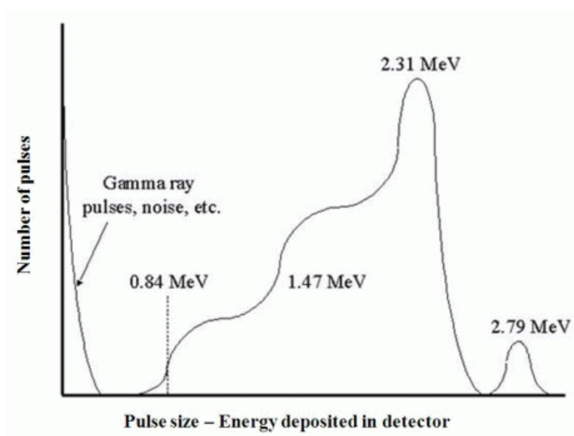

Fig. 1. Energies in detector

BF_3 detectors have an efficiency of over 90% for neutrons of energy less than 0.5 eV [2]. However, their response is significantly less sensitive for higher energy neutrons. So, a more detailed study is needed for detectors response when exposed to neutron fields with heterogeneous energy, because despite presenting a less intense response, more energetic neutrons cannot be neglected in the interpretation of data.

Computer simulations were performed using the MCNP code [3] resulting in a good estimate of the neutron flux reaching the detector. The MCNP code is a Monte Carlo method to simulate the transport of radiation. Such simulations are statistical in nature, assuming that the interaction of radiation with matter leads to a successive stochastic events that occur according to a probability distribution. [4]

I. Objectives

The aim of this work is studies of paraffin and polyethylene as neutron moderators determine the optimum thickness of thermalization, performance of the BF_3 detector efficiency during the thermalization of neutrons from an AmBe source and compare the experimental and simulated results.

II. Materials and Methods

The experiment was developed to determine the response of a detector (BF_3 proportional counter) for different neutron fluxes. These fluxes were obtained through some kind of moderator (paraffin and polyethylene), keeping the AmBe source and detector at a fixed distance with different moderator material thicknesses between them.

BF_3 detector used was model S3179 from Reuter Stokes with 1.6cm in diameter and 2.4cm in length with a sensitive volume about 4.83 cm^3. Standard NIM electronic modules from ORTEC were used for this experiment: a preamplifier model 142 with input coupled to BF_3 detector, connected to a spectroscopy amplifier 572 and a Model 919 Multichannel Spectrum Master. Americium-Beryllium (Am-Be) source is a sealed cylindrical capsule with 2.4cm in diameter and 4.1cm in height with an activity of 37 GBq (1Ci, August 1970) [5]. The emission spectrum of the neutrons is shown in Fig. 2.

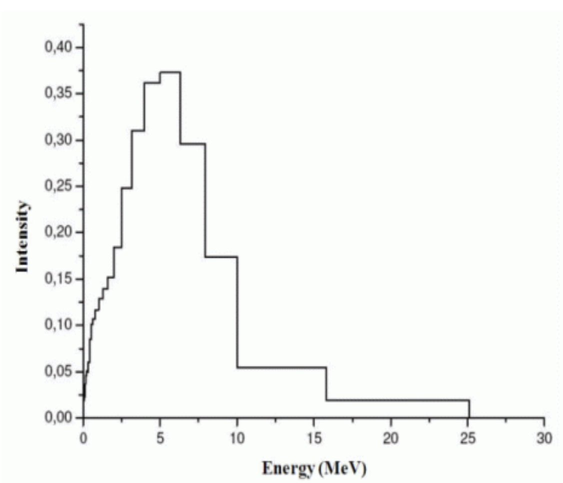

Fig. 2. AmBe neutron energy spectrum

To keep the source and detector at a fixed distance an aluminum frame keeping them separate was used. Experimental arrangement was implemented on a wooden table and wrapped in polyethylene and paraffin blocks aimed at shielding the neutrons emitted by the source (radiological shielding). To shield the detector from external interference during the experiment, it was partially surrounded by a cadmium (Cd) sheet, so that only neutrons from the source reached the detector.

The material moderator plates (10x10x1 cm) were placed between source and detector to modify the thickness from 0 to12 cm, performing a sample of 300 seconds for each measurement. Two sets of measurements were carried out for the experiment: a) with paraffin and b) polyethylene as moderator material. Experimental setup is shown in Fig. 3.

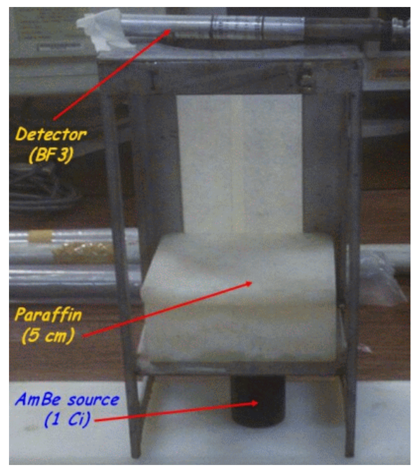

Fig. 3. Experimental detection set up. BF_3 detector, paraffin moderator and AmBe source.

The experimental procedure was also performed using the dummy code MCNP5 developed for the transport of radiation [1].

The Monte Carlo method simulates a mathematical problem stochastically It has been often used to simulate processes involving random behavior and to quantify physical parameters that are difficult or even impossible to calculate by means of experimental measurements. Monte Carlo techniques have become popular in many areas, such as reactor physics and medical physics, due to the stochastic nature of radiation transport, emission and detection processes [5].

For each thickness of moderator material a computer simulation was performed determining (with statistical error of less than 2%) the neutron flux reaching the detector. This flux is divided into 3 bands of energy fixed to the sensitivity of the detector.

The simulation provides additional data to those obtained experimentally, since the information are not limited to the sensitivity of the detector used, BF_3. Fig. 4 shows the simulation geometry. An aluminum structure to maintain source and detector separated was used, and can be neglected in the simulation session due to the low aluminum cross section for interactions.

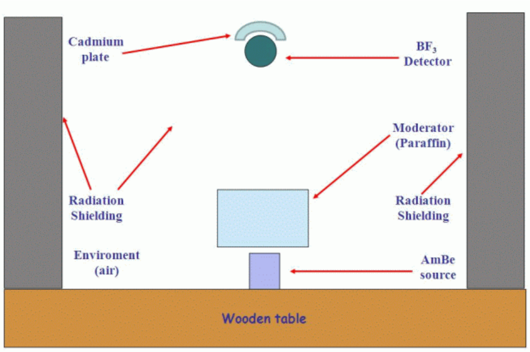

Fig. 4. Simulated geometry for MCNP

III. Results and Discussion

For better comparison, Fig. 5 shows the number of counts per second reaching the detector as a function of the thickness of moderator material, both paraffin and polyethylene.

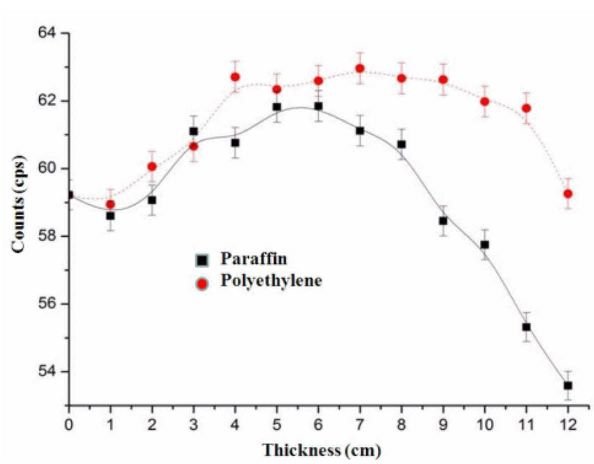

Fig. 5. Response to neutrons moderation

The Am-Be source (Fig. 2) emits fast neutrons priority, however, the BF_3 detector is more sensitive to thermal neutrons, thus, the response curve is due mainly by two factors: a growing factor corresponding to fast neutrons moderation that start to enter an energy region where the detector is more sensitive, and a decreasing factor related to the absorption of thermal neutrons [6]. Fig. 5 shows that neutrons moderated by polyethylene soon reach a maximum flow, and then show a less steep fall in comparison with paraffin.

In 2011, the same experiment was performed using only paraffin as a moderator, without the presence of cadmium sheet surrounding (partially) the sensor [7]. Fig. 6 shows the result obtained for the simulation of the neutron flux for the experimental arrangement, while Fig. 7 shows the simulation with cadmium sheet.

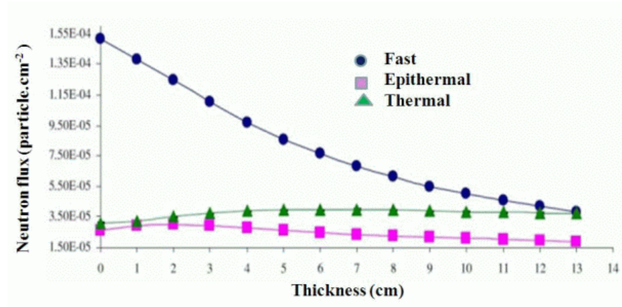

Fig. 6. Simulation without Cd sheet

Same energy ranges were adopted in simulations: **E1** from 0 to 0.5 eV (thermal), **E2** from 0.5 eV to 10keV (epithermal) and **E3** from 10keV to 16MeV (fast), there were no necessity to simulate higher energies because the source does not emit neutrons in this range.

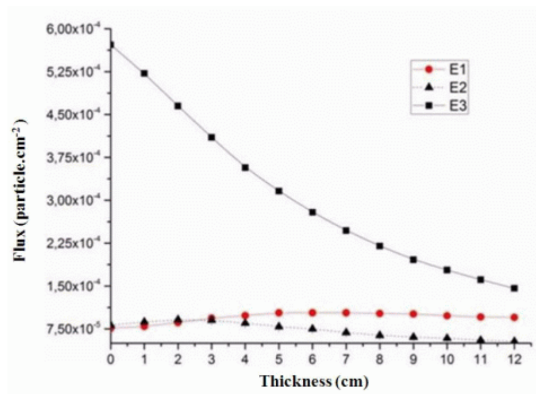

Fig. 7. Simulation with Cd sheet

Results showed that the behavior of the curves is almost the same, with the only difference in the curve of least energy (**E1**) which features an offset due to absorption of low neutrons by cadmium.

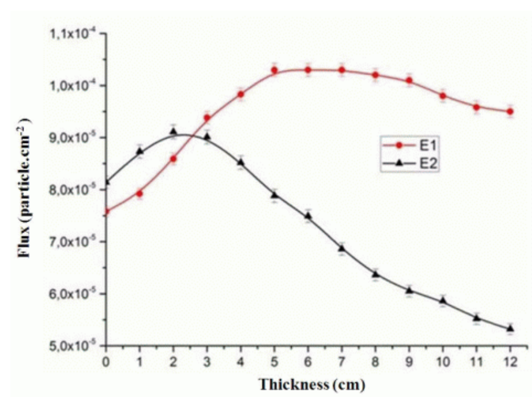

Fig. 8. Simulation with Cd sheet and low energy neutrons

Fig. 8 shows the curves **E1** and **E2** in the presence of cadmium sheet, thermal neutrons have a lower counting than epithermal neutrons for smaller moderator material thicknesses. This does not occur in the simulations without cadmium. By comparison the experimental detector response obtained with the simulated curves, we conclude that the standard response of the detector is due mainly by thermal neutrons, as expected.

When comparing the simulated thermal neutrons flux obtained using paraffin and polyethylene (Fig. 9), similar response curves are obtained, however, behaviors which do not agree with each other taking into account the experimental results.

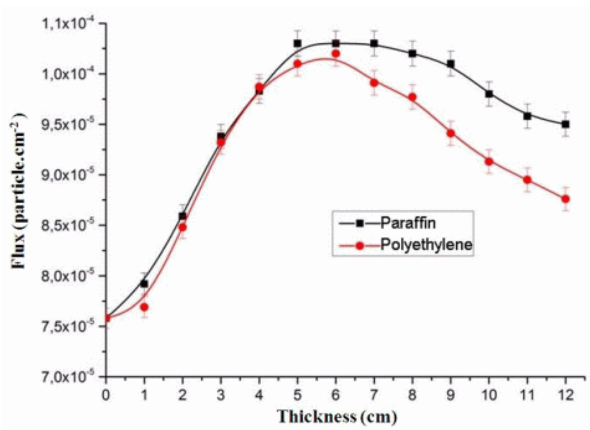

Fig. 9. Moderators comparison

Experimentally, the paraffin showed a moderation effect more intense than polyethylene. Then it was seen that the detector response is governed by thermal neutrons.

Therefore, it is expected that the thermal neutron flux simulated to provide a stronger attenuation in the paraffin than in the polyethylene. However, what occurs in the simulation contrast (Fig. 9), the polyethylene has a “steeper” (strong) attenuation.

The difference between the two materials is theoretically negligible. The materials used in the simulation have different density (0.88 g/cm3 to 0.94 and the paraffin and polyethylene respectively). Being denser, it would be natural to expect that the effect of neutron absorption in polyethylene were more intense, as the simulation. However, what has not been taken into account during the simulation is that the paraffin is a porous material, and non-homogeneity of the material enhances the scattering effect, so that fewer neutrons eventually reach the detector, thereby generating a lower count.

IV. Conclusions

Through the experiment, it was found that the response of the detector is derived mostly from neutrons of energy less than 0.5 eV. A computer simulation generated from the Monte Carlo method proved to be able to provide additional data to those obtained experimentally, as the power attenuation of the moderators for neutrons of all energies, not just thermal. The analysis of the simulated and experimental data allows to infer that the formation of inhomogeneous paraffin implies it is a better material to be used for neutron shielding than polyethylene.

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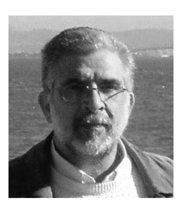

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