EXPERIMENTAL AND MCNP STUDIES OF NEUTRON MODERATORS AND BF DETECTOR EFFICIENCY

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RESUMEN

El Instituto de Investigaciones Energeticas y Nucleares - IPEN, ofrece diversos programas de postgrado, a saber: Tecnología Nuclear - Aplicaciones (TNA), Tecnología Nuclear - Materiales (TNM), Tecnología Nuclear - Reactores (TNR). Estos programas tienen como misión la formación de técnicos, expertos e ingenieros con un profundo conocimiento en su disciplina para trabajar en el área nuclear.
El curso "Fundamentos Teóricos y Prácticos de la Instrumentación Utilizada en la Adquisición de Datos Nucleares" cubre el uso de laboratorio de instrumentación nuclear y la realización de experimentos para obtener parámetros nucleares.
Uno de estos experimentos es objeto de este trabajo: "Los estudios experimentales y de MCNP de moderadores de Neutrones y la eficiencia del detector BF_3".
Los neutrones son partículas sin carga y, por lo tanto, no pueden ser detectados por las interacciones de Coulomb. Por lo tanto, el montaje del detector utilizado debe contener algún tipo de material con alta sección eficaz para la interacción con neutrones, llamado convertidor. Un detector de trifluoruro de boro (BF3) se utilizó en este experimento para detectar neutrones en tiempo real. Sin embargo, la respuesta de esta disposición varía de acuerdo con el rango de energía de los neutrones incidentes. Su eficiencia para neutrones térmicos es superior a 90%, pero, este resultado se reduce, de manera significativa, para los neutrones de energía superior a 0,5 eV. La moderación de neutrones y, en consecuencia, su variación de energía se obtuvieron mediante la interposición de diferentes espesores de material moderador (parafina o polietileno) entre la fuente emisora (AmBe) y el detector.
La eficiencia (respuesta) del detector y el espesor óptimo de los moderadores se determinaron experimentalmente y mediante simulaciones por ordenador utilizando el código MCNP para una comparación de los resultados. Este código utiliza el método de Monte Carlo para simular el transporte de radiación en la materia. Este trabajo representa un informe de los estudiantes del curso basado en la adquisición de datos obtenidos durante el ejercicio experimental, los datos de análisis y una comparación con estos datos experimentales utilizando una simulación con código MCNP.

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 Neutron Moderators 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, these are 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 for results comparison. This code uses the Monte Carlo method to simulate radiation transport in matter. This work represents a report from the course students based on the data acquisition obtained during the experimental exercise, data analyzes and a comparison with this experimental data using MCNP code simulation.

1. INTRODUCTION

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

In the detector response (spectrum), there are two peaks corresponding to equations (1) and (2) with probabilities of 94% and 6% respectively, as shown in Fig. 1 [1,2].

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

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

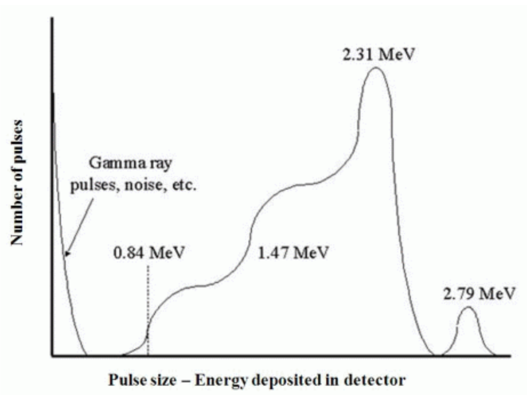

Figure 1: Energies in detector [2].

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].

2. OBJETIVES

The aim of this work represents a report description of an experimental exercise developed at the CENF (Nuclear Engineering Center) laboratory: study of paraffin and polyethylene as neutron moderators determining the optimum thickness of thermalization, performance of the BF_3 detector efficiency during the thermalization of neutrons from an AmBe source and a comparison between the experimental and simulated results (response curves). This experimental exercise does not intend to establish new values for paraffin and polyethylene moderators, but rather to train students in nuclear properties of elements.

3. 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 moderators (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.

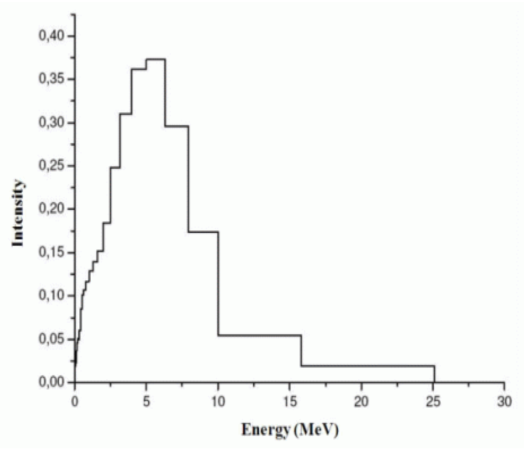

Figure 2: AmBe neutron energy spectrum [6].

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 surrounded by

polyethylene and paraffin sheeting in order to shield the neutrons emitted by the source to the ambience (radiological shielding). Provide that 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 moderators materials. Experimental setup (outside the experiment bench) is shown in Fig. 3.

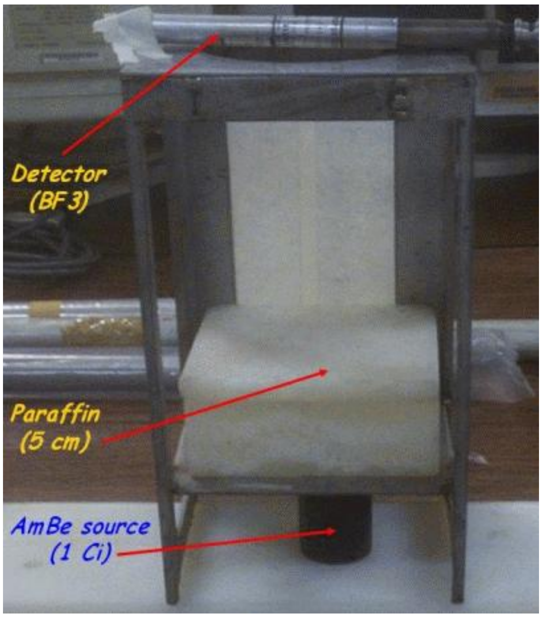

Figure 3: Experimental detection set up (outside the experiment bench). BF_3 detector, paraffin moderator and simulated AmBe source.

Maestro software was calibrated with standard energy sources 60Co and 137Cs that have the initial activities and the known energy spectra. 300 seconds counting were performed (live time - real time minus dead time) for each time 1 cm thick moderator (paraffin or polyethylene) was added, and the maximum thickness was 10 cm. For each neutron field one data acquisition was carried out and a spectra was obtained for each one, as shown in Fig. 4. These spectra were acquired using the ROI function of Maestro software (channel selection) to obtain the integral counts or “Gross Areas”. These ROIs were made from the first level due to Li's wall effect in the excited state to the end of the lower peak which is the Li energy storage in the ground state and the alpha particle.

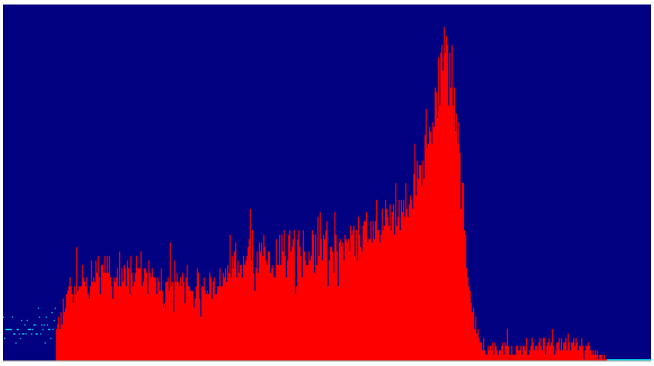

Figure 4: Neutron spectra detected by the BF3 tube and analised by the Maestro software.

A comparison with the experimental procedure (data acquisition) was made using the dummy code MCNP5 developed for the transport of radiation [3].

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 [4].

For each thickness of moderator material (1 to 12 centimeters, instead of 10 cm in the experiment)) 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 [7]. Fig. 5 shows the simulation geometry. During the experimental procedure an aluminum structure to maintain source and detector separated was used, and this can be neglected in the simulation session due to the low aluminum cross section for interactions.

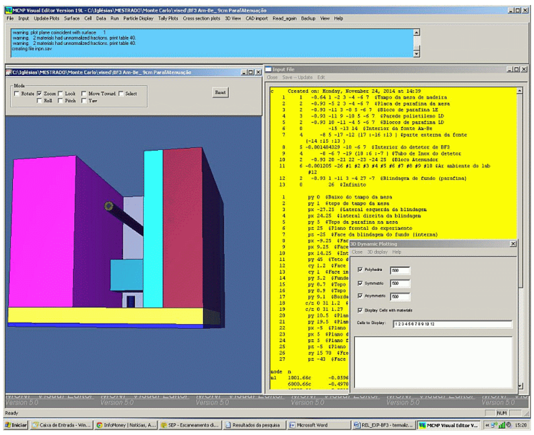

Figure 5: Simulated geometry for MCNP.

4. RESULTS AND DISCUSSION

Table 1 shows the data acquisition considering the Gross Areas for each measurement.

Table 1. Data acquisition from Maestro software

Moderator thickness (cm)	Paraffin (Gross area)	Polyethylene Gross area)
0	20182 ± 142	20182 ± 142
1	20754 ± 144	20971 ± 145
2	21640 ± 147	21772 ± 148
3	22276 ± 149	22536 ± 150
4	22387 ± 150	22258 ± 149
5	22532 ± 150	22470 ± 150
6	22416 ± 150	22401 ± 150
7	22265 ± 149	21801 ± 148
8	21654 ± 147	20972 ± 145
9	21206 ± 146	20465 ± 143
10	20244 ± 142	19673 ± 140

Using the data acquisition obtained with ORTEC Multichannel Analyzer and Maestro software the curves responses (Table 1) were plotted using Origin software. Fig. 6 shows the number of counts per second reaching the detector as a function of the thickness of paraffin moderator material.

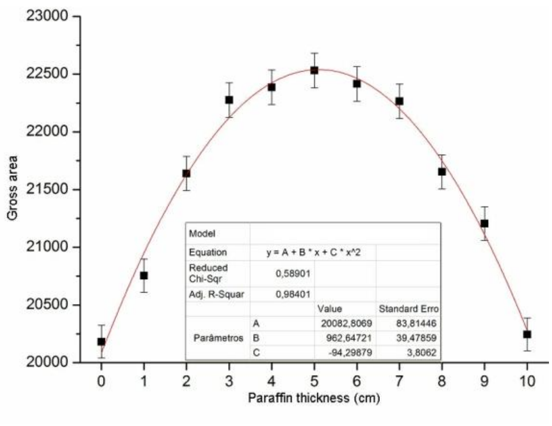

Figure 6: Response of paraffin moderator to fast neutrons.

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 [8]. Fig. 7 shows the response to fast neutrons moderated by polyethylene. This response soon reaches a maximum flow and then shows a less steep fall in comparison with paraffin.

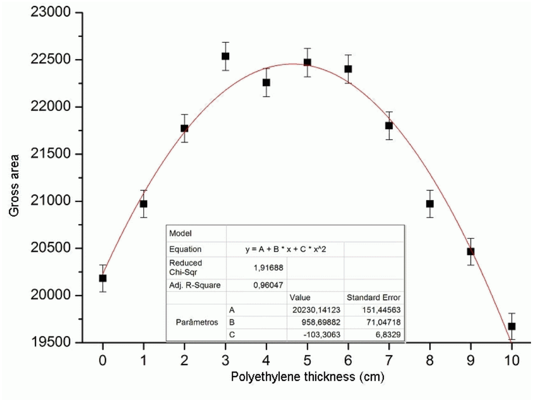

Figure 7: Response of polyethylene moderator to fast neutrons.

When comparing the simulated thermal neutrons flux obtained using paraffin and polyethylene (Fig. 8), similar response curves are obtained, however, behaviors which do not agree with each other taking into account the experimental results.
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. 8), the polyethylene has a “steeper” (strong) attenuation.

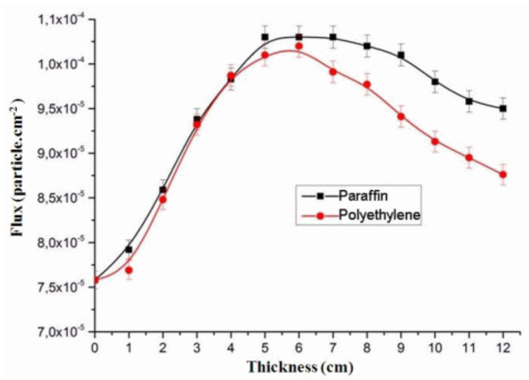

Figure 8: Simulated paraffin and polyethylene moderator’s comparison.

The difference between the two materials is theoretically negligible. The materials used in the simulation have different density (0.88 g/cm^3 to 0.94 for 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.

5. 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 inferring that the formation of inhomogeneous paraffin implies it is a better material to be used for neutron shielding than polyethylene.

REFERENCES

1. Frame, P. "Boron Triuoride (BF3) Neutron Detectors", Oak Ridge Associated Universities.
2. Knoll, G.F. "Radiation Detection and Measurement", John Wiley & Sons, Inc., New York (2000).
3. Briesmeister, F. “MCNP - A general Monte Carlo N-particle transport code, version 4C”, Los Alamos National Laboratory, LA-13, 709-M, (April, 2000).
4. Yoriyaz, H. “Monte Carlo method: Principles and Applications in Medical Physics", *Revista Brasileira de Física Médica*, 3, pp.141-149 (2009).
5. Madi, F.T. “Desenvolvimento de Detector de Nêutrons usando sensor tipo barreira de superfície com conversor (n,p) e conversor (n,a)", CNEN-IPEN, São Paulo (1999).
6. International Atomic Energy Agency (IAEA), “Compendium of Neutron Spectra and Detector Responses for radiation Protection Purpose”, Vienna (2001).
7. Castro, V.A., Cavalieri, T.A., Siqueira, P.T.D., Fedorenko, G.G., Coelho, P.R.P., Madi, F.T., "MCNP to study the BF3 detection efficiency", INAC, Rio de Janeiro 2011.
8. Suckling, E. "Neutron Detection and Spectroscopy," University of Surrey, Guildford (2005).