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NEUTRON FLUX DISTRIBUTION IN AN Am-Be NEUTRON IRRADIATOR

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ABSTRACT

A neutron irradiator has been assembled at IPEN facilities to perform qualitative-quantitative analysis of many materials using thermal and fast neutrons outside the reactor premises. To establish the prototype specifications, the neutron flux distribution was calculated using the Monte Carlo technique. These theoretical predictions then allow one to discuss the irradiator's performance.

Keywords: neutron, Am-Be, irradiator, MCNP, NAA

I. INTRODUCTION

The neutron flux measurement and the neutron energy spectrum can be obtained through neutron-induced count rate using gamma spectrometry. The advantage in using gamma radiation it is the elimination of corrections related with self-absorption in the activation detectors. Usually, the thermal neutron flux is measured by gold activation foils (both bare and cadmium-covered) and the fast neutron flux by indium activation foils.

The neutron activation analysis (NAA) can be divided into steps: first, one exposes the material to be analyzed (sample) and a standard to the thermal neutron flux of a nuclear reactor; afterwards, one measures the induced activity in the sample by the comparative method. The use of the neutron irradiator presents the advantage of supplying a stable neutron flux for a long period, thus eliminating the need for using standard material. This way the analyzing process became agile, practical and economic.

In this work the description of the neutron irradiator is presented and, according to those specifications, the neutron flux distribution is calculated using the MCNP code [1].

II. NEUTRON IRRADIATOR DESIGN

This prototype consists of an aluminum cylinder of 5mm thickness with 1200mm length and 985mm diameter, filled with paraffin, and two perpendicular cylindrical cavities (B and C), with the same diameter (~ 80mm), which cross the prototype's geometric center. In the metallic cavity B (also of 5-mm-thick aluminum) a ruler passes through the longitudinal direction, where the material to be irradiated can be put in different positions. In the cylindrical hole (cavity C), the two neutron sources are positioned symmetrically, at the same distance from the geometric center, face to face. Details about the axes configuration are shown in Fig. 1.

The Americium-Beryllium sources were obtained commercially and both have the following specifications: 600GBq ^{241}Am-^{9}Be (α,n) neutron source, with cylindrical design (40mm diameter by 70mm long) made of corrosion-resistant alloy with neutron emission rate of 3.9×10^7n/s, each. Two different configurations, related with the neutron sources arrangements, can be explored:

a) Thermal neutrons prevalence. In this situation, a polyethylene cylinder around 50mm long is placed between each neutron source and the sample in order to thermalize the emitted neutrons;

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b) Fast neutrons prevalence. In this situation, the neutron sources are positioned at 35mm of the prototype's geometrical center.

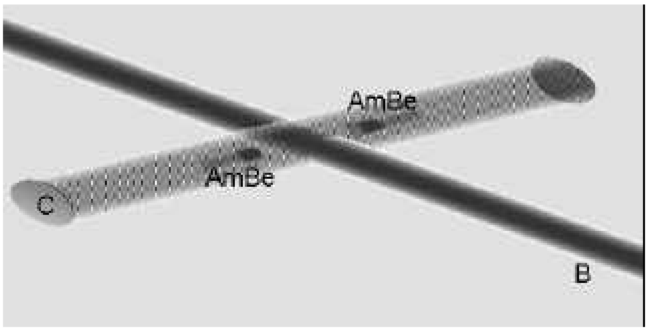

Figure 1. Detail of the Axes Configuration.

III. MONTE CARLO CALCULATIONS

The MCNP code developed in Los Alamos [1] carries out the radiation transport, relating to neutrons, photons and electrons with energetic and temporal dependence in a three-dimensional geometry by using the Monte Carlo method. In general, it is based in the probability distribution function for developing the random sampling of each event and performing the evolution of the particular phenomena being studied by means of convenient statistical techniques. The capabilities of this code involve the correct simulation of the physical problem and the geometrical configuration.

In this work the MCNP-4C code was used to estimate the flux in two different configurations related with neutron sources arrangements. The energy ranges considered were: thermal below the Cadmium cut-off energy (0.5eV), epithermal (between 0.5eV and 0.5MeV) and fast neutrons (above 0.5MeV). The results for the fast-prevalence configuration are shown in Fig. 2 and the ones for the thermal-prevalence one are in Fig. 3.

Particularly, for biological samples the knowledge of neutron dose rate is essential and this estimation can also be done using this code. The neutron absorbed dose rates for both configurations were calculated using the MCNP-4C code, and the results are shown in Fig. 4 and Fig. 5.

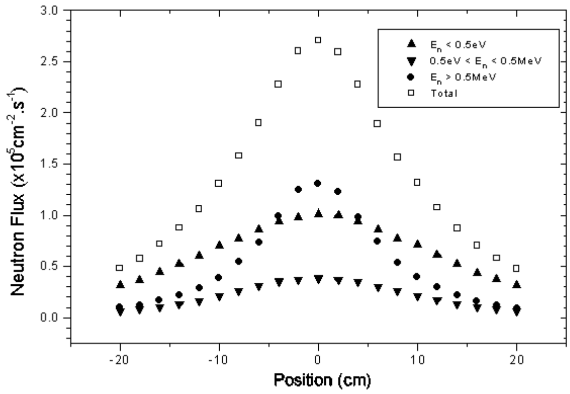

Figure 2. Neutron Flux Distribution in the Fast Configuration

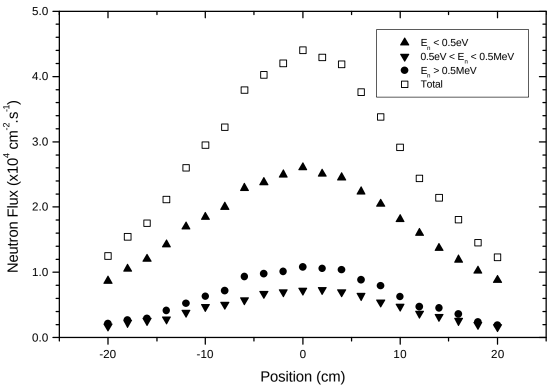

Figure 3. Neutron Flux Distribution in the Thermal Configuration.

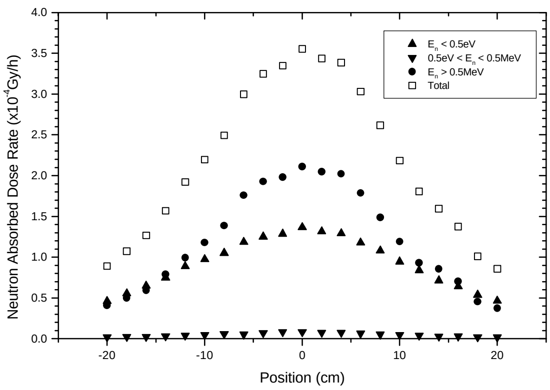

Figure 4. Neutron Absorbed Dose Rate in the Thermal Prevalence Configuration

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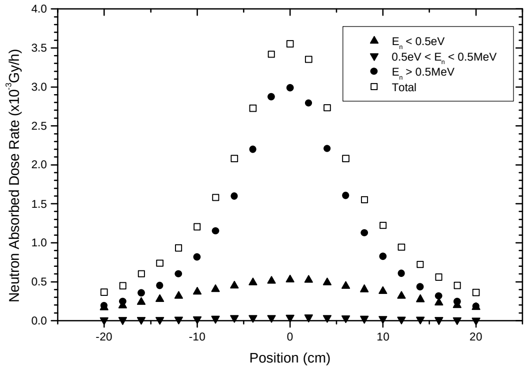

Figure 5. Neutron Absorbed Dose Rate in the Fast Prevalence Configuration

IV. DISCUSSION

The results presented in this work are an important information for a better knowledge of neutron flux distribution in the prototype. According to theoretical simulations, the prototype can be useful to investigate biological, geological, metallic and ceramic samples. It can also be used to test detectors and to do quality control check using NAA.

However, two disadvantages must also be considered when addressing to the prototype: the requirement of personnel experienced with radiological protection to perform the analysis outside the reactor premises; and the impossibility of investigating material with a low microscopic neutron cross section, due to the low neutron flux available.

REFERENCE

[1] Briesmeiter, J. F. (Ed.), **MCNP-A General Monte Carlo N- particle transport Code, Version 4C**, LA-13709 M, 2000.