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A NEUTRON IRRADIATOR TO PERFORM NUCLEAR ACTIVATION

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ABSTRACT

A neutron irradiator has been assembled at IPEN to perform qualitative-quantitative analysis of many materials using both thermal and fast neutrons outside the reactor premises. To standardize the prototype specifications, the neutron flux distribution and the neutron dose rate were calculated using the Monte Carlo technique. These results make possible to discuss its performance.

INTRODUCTION

The development of appropriate nuclear instrumentation to perform nuclear activation can be useful to investigate materials like: biological, geological, metallic and ceramic samples; also can be used to test and calibrate detectors, to check the quality control and many other applications in different areas. Basically, the technique consists of the irradiation of the material to be investigated, using nuclear reactor or particle accelerators, to produce radioisotopes products that provide information about its elemental composition as well as impurities present in the sample. The main advantage of this irradiator is its very stable neutron flux eliminating the use of standard material (to measure of the induced activity in the sample by the comparative method). This way the analysis became agile, practical and economic.

DESIGN CONFIGURATION

A view of the neutron irradiator prototype is shown in Figure 1A. Basically, this prototype consists of an aluminum cylinder of 5 mm thickness with 1200 mm of length and 985 mm of diameter, filled with paraffin, and two perpendicular cylindrical cavities (B and C) , with the same diameter (~ 80 mm), which cross the prototype's geometric center. In the metallic cavity B (also of aluminum with 5mm of thickness) a graduate ruler (scale in mm) passes through the longitudinal direction, where the material to be irradiated can be put at different positions, at a radial direction along the rule. In the cylindrical hole (cavity C), the two neutrons sources are positioned symmetrically, face to face, each at 5mm from the irradiator' s geometric center. Details are shoed in Figure 1B.

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The Americium-Beryllium sources were obtained commercially, both having the same specifications: 600 GBq $^{241}Am\ (\alpha, n)$ neutron source type NSR-F, with cylindrical design (40 mm diameter by 70 mm long) made of corrosion- resistant alloy MP- 5N, i.e. cobalt (2.5%), nickel (6%) and chromium (20%) and neutron emission rate of .9 x 10^7 n/s, each one.

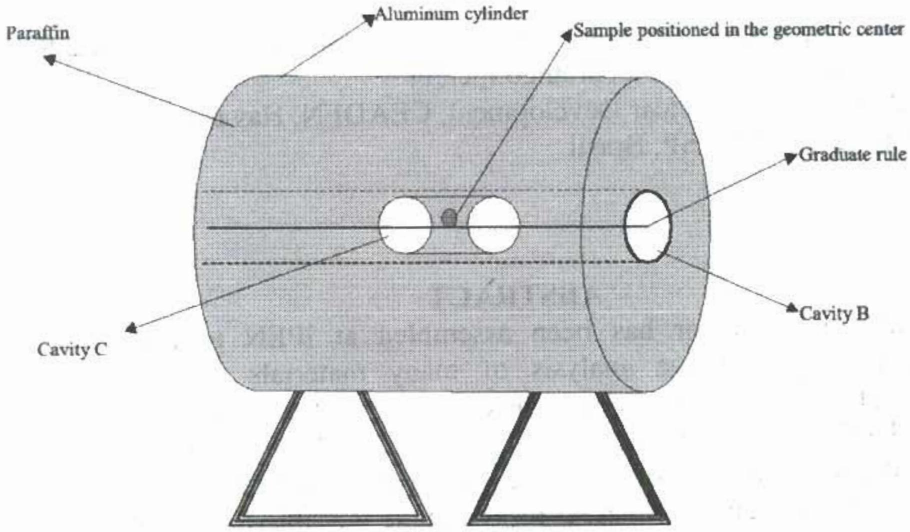

Figure 1A. Overview of the Neutron Irradiator Prototype

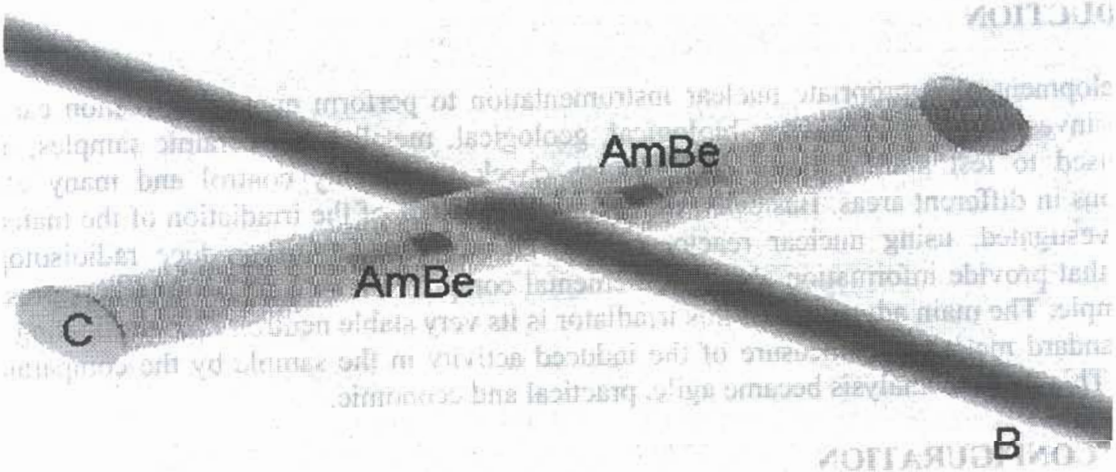

Figure 1B. Details of the axis configuration.

THEORETICAL SIMULATION

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. This is an stochastic method for physical problem simulations. 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 neutron flux and dose rates in different energy ranges and several position within the geometry of the irradiator. The energy ranges considered were: thermal neutrons (En<0.5eV), epithermal neutrons (0.5eV<En<0.5MeV) and fast neutrons (En>0.5MeV). The simulation results are shown in Figure 2 .

EXPERIMENTAL PROCEDURE AND RESULTS

Experimentally the thermal neutron flux distribution has been measured using NAA and the results compared with the estimate from MCNP-4C code [1].
For this purpose, the flux distribution had been measured for two energy ranges: below the Cd cut off energy and above the Cd cut off energy. A metallic foil of gold (2.54 diameter and 0.5g) was used to identify and quantify the radioactive nuclide of ^{198}Au ($T_{1/2}$=2,7 d, $E\gamma$ = 411 keV). The experimental procedure consists of gold irradiation for hours, in different positions (i.e. cavity C) of the prototype's irradiator. The same procedure was repeated using the gold sample covered with metallic cadmium. After each irradiation, the gamma activity of the samples were counted during 1 hour, using a hyper-pure Ge detector connected to Adcam multichannel (Ortec) and to a PC computer, and the area of the select gamma-ray peak was obtained by using the IDF program [2]. The solid angle correction have been done using the FCAS program [ ]. The flux distribution results for thermal neutron and the Monte Carlo simulation for thermal, epithermal and fast neutron flux (using the MCNP-4C code) are shown in Figure 2. The experimental result for the cadmium factor is presented in Figure .

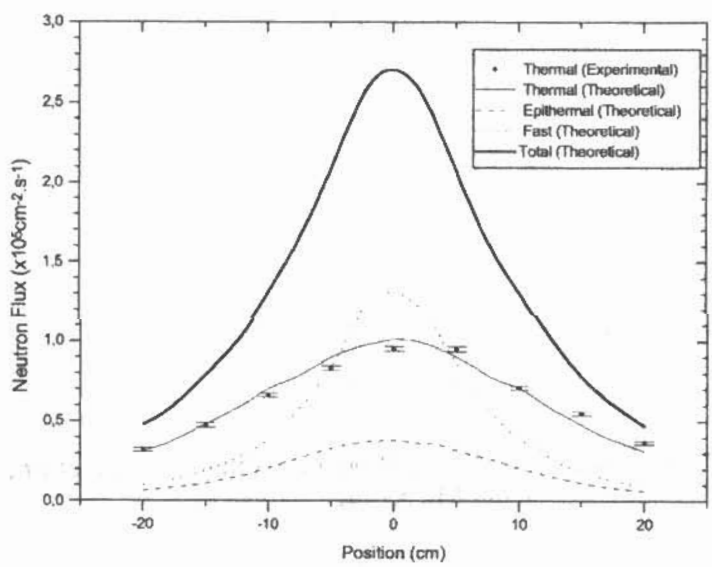

Figure 2. Neutron Flux distribution in different positions at a radial direction with regard to the AmBe sources. Thermal, epithermal and fast neutron flux were calculated using MCNP-4C Code and thermal neutron flux was measured using gold activation analysis.

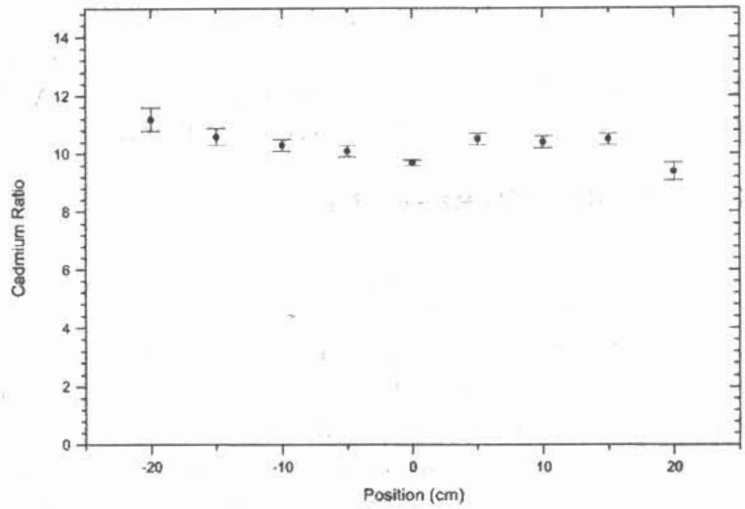

Figure . The experimental result of Cadmium factor measured by NAA

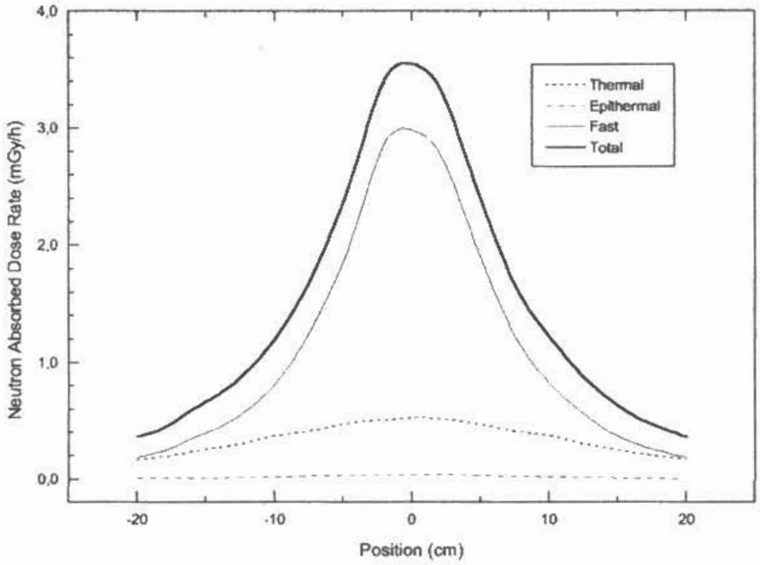

Figure 4. Neutron Absorbed dose rate in different positions in radial direction, calculated by using MCNP-4C.

According to Figure 2 the experimental results of thermal flux distribution compared to those obtained by MCNP-A code showed good agreement. These results are an important support to perform irradiation in the prototype besides the knowledge of dose rate is essential for studying biological materials.

The simplicity of this apparatus makes this a good choice to perform NAA outside the reactor premises. Also, this prototype offers the additional advantage of low cost. Regarding the disadvantage two points must be considered: the need of technician experienced with radiological protection to perform the analysis outside the reactor premises and the difficulty or the impossibility of investigating material that has a low microscopic neutron cross section, however this problem can be solved it is if great quantity of material can be used.

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