PAGE — E-261

MEASUREMENTS AND CALCULATIONS OF FAST NEUTRON ENERGY SPECTRA EMERGING FROM LAMINATED MATERIALS

P. R. P. Coelho

Instituto de Pesquisas Energéticas e Nucleares, IPEN/CNEN-SP
Caixa Postal 11049 (Pinheiros)
05422-970, São Paulo, Brasil
E-mail: prcoelho@net.ipen.br

ABSTRACT

Fast neutron energy spectra emerging from assembles of laminated materials were measured and compared with calculated computational code results in order to evaluate calculational method and related library performance. The experiment was modeled computationaly using the Monte Carlo method, MCNP transport code, and the neutron energy spectra emerging from the laminated material are been calculated for the several NE-213 scintillator measurement positions. Results of comparisons between measured and calculated neutron energy spectra have been showed good agreement in the spectrum shape and, deviation of about 2% between integrated spectrum from 4.1 to 17.3 MeV.

INTRODUCTION

Experimental data are needed to evaluate calculational method and related libraries performances and, experimental facilities have been used in all the world with this aim. In this paper is presented an experiment, conduct in a facility of IPEN/CNEN-SP, in which were measured fast neutron energy spectra emerging from assembles of laminated materials. This experiment was simulated with the Monte Carlo transport code MCNP and comparison between experimental and calculated results is presented and discussed.

DESCRIPTION OF THE EXPERIMENT

The experimental facility, showed in a schematic view in figure 1, is comprised of a Van de Graaff accelerator used to accelerate deuterium with a voltage up to 170 kV to produce 14 MeV neutrons in a tritium target due to a $^3H(d,n)^4He$ reaction. These neutrons are incident in a laminated assemble composed by steel, polyethylene, lead and steel, squared plates with 60 cm of side and respective thickness of 2.2 cm, 15 cm, 10 cm and 2.2 cm. The fast neutron energy spectrum emerging from this assemble was measured using a fast neutron spectrometer, composed of a NE-213 organic liquid scintillator, type recoil protons detector, glass encapsuled, 5.08 cm height by 3.81 cm of diameter, optically joint by a light guide to a RCA 8850 fast photomultiplier which is connected to a pulse shape electronic system for neutron and gamma discrimination [1]. The neutron interactions with the several materials produce γ-rays by (n,γ) reactions and, as the NE-213 detector is sensible to γ-rays and neutrons, it is necessary this neutron and gamma discrimination to be possible the fast neutron spectrometry.

The accelerator target and the test section ($60x60x60$ cm^3) are located inside a water tank, used to obtain a well defined model for computational purpose. The absolute neutron yield in the target is measured by associated alpha particle counting technique, using a surface

5051

barrier detector located outside the tank in a evacuated tube perpendicular to the accelerator tube. As this detector is damaged by radiation, a thermal neutron detector, a BF_3 inside the tank, intercalibrated with the surface barrier detector was used for the neutron yield measurements. The neutron production was determined with a precision better or equal to 1.6% [2].

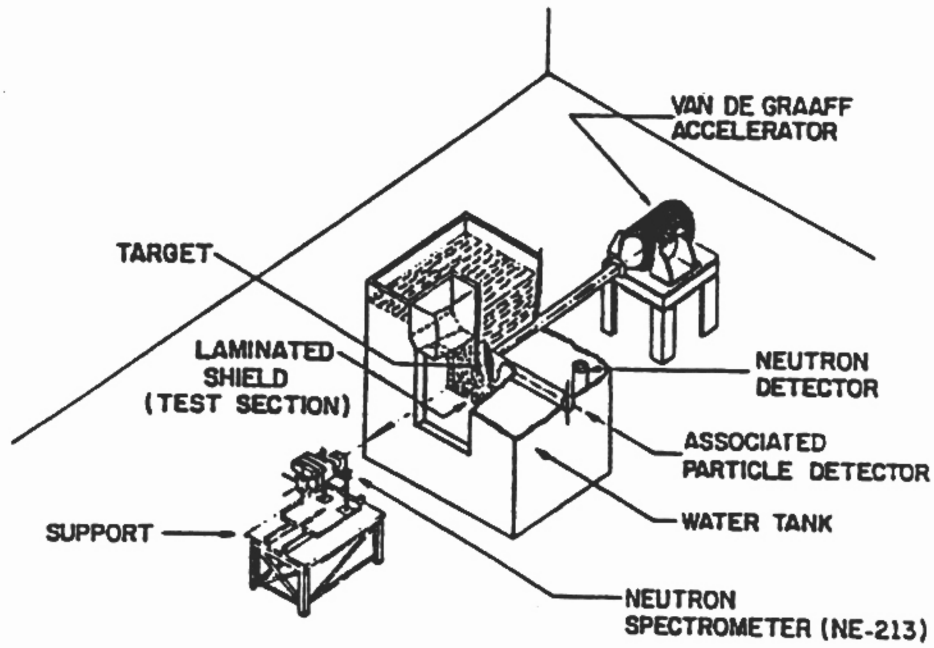

Fig.1. Experimental Facility (Schematic View).

The measured protons-recoil pulse height spectra ($M(E_p)$) from neutron interactions with the scintillator material is related to the neutron energy spectrum incident in the detector by the equation

$$M(E_p) = \int_{E_m}^{\infty} R(E_p, E_n)\phi(E_n)dE_n \tag{1}$$

where $R(E_p, E_n)$ is the detector response function and E_m is the minimum neutron energy to produce protons with energy greater then the smaller value of measured E_p. The spectrum unfolding, solution of this equation, is obtained using numerics algorithim. The data were unfolded using the FANTI code [3], which applies the matrix inversion method [4] to obtain the neutron energy spectrum. The espectrometer linearity verification and energy calibration were made with gamma ray sources and its performance was evaluated measuring spectra from standard neutron sources of Am-Be and ^{252}Cf [1]. The measured spectra shows good agreement with the spectra published in the literature. This spectrometer was calibrated to give neutron spectrum in the range of 2 to 18 MeV, with 6% intrinsic efficiency and a resolution between 4% and 11%.

Two sets of measurements were made in the experimental facility:

a) measurements changing the material composition and keeping fixed the detector NE-213 position with the objective of study the influence of the materials in the energy spectrum and neutron intensity on detector position.

b) measurements using a fix material composition and positioning the detector in several different places to obtain a map of the neutron energetic and spacial distribution.

CALCULATIONAL METHODOLOGY

The experiment was modeled computationaly using the Monte Carlo method, MCNP transport code [5], and the neutron energy spectra emerging from the laminated material are been calculated for the several NE-213 scintillator measurement positions. Figure 2 shows a Y-Z view of the geometric configuration used in this paper, where all the experimental room was modeled with 31 cells and 53 surfaces. The cells 3, 5, 6, 11, 12, 13, 14 and 15 are cylinder and the other are boxes. Table I lists the materials of the several cells. The neutron source (target) is located on the surface between cells 3 and 15. A ring detector tally in the center of cell 13 was used to determine the flux in the detector position.

To increase the efficiency of MCNP program process there are, in the MCNP, several variance redution technics and it is difficult, prior to employ, to establish the assemble of technics to be used to obtain the best results, so the MCNP code was processed several times, changing the assemble of technics, to get it.

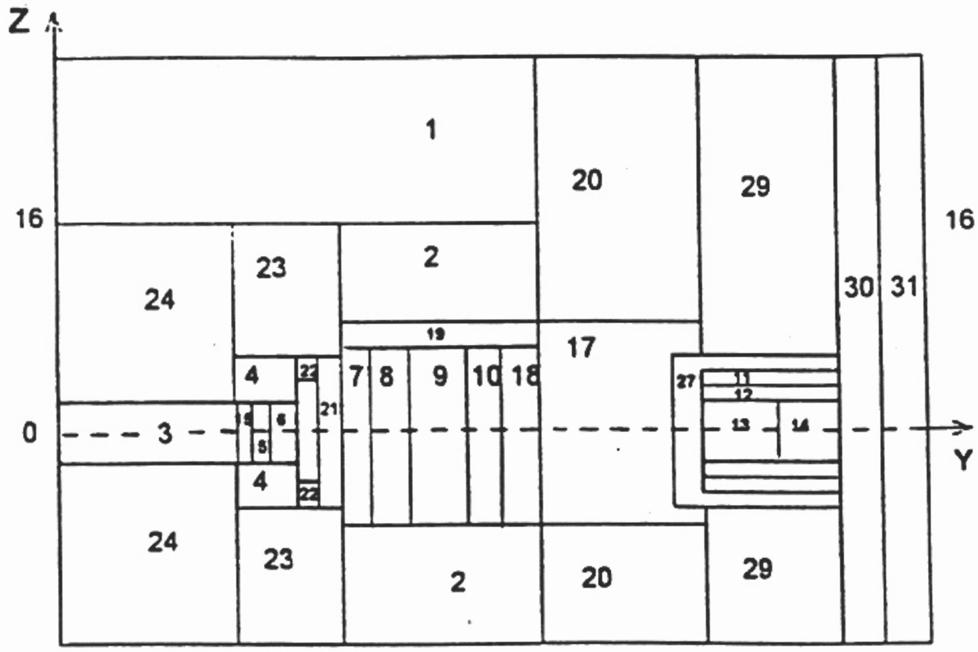

Fig. 2. Calculational model of the experimental configuration.

TABLE I
Composition of the 31 cells.

MATERIAL	CELLS
air	1,4,12,17,18,19,20,21,22,26,27,28,29,30,31
water	2,5,23,24,25
copper	6,15
steel	7,10
polyethylene	8
lead	9,11
NE-213	13
light guide	14

RESULTS

The results showed a good symmetry around the deuteron beam axis (Y axis) indicating that the contribution of neutrons scattered in the ground to detector direction is insignificant.

The effects of neutron flux moderation and attenuation can be observed in the measurements with variation of the materials' composition as well as in the measurements where the NE-213 detector is moved from the Y axis, as shown in figure 3.

We have some statistical flutuation in the experimental results due mainly to low neutron production obtained in the target (~10^7 n/s). The register produced by the detector for neutrons with energy greater than 15.68 MeV (maximum source neutron energy), presented in figure 4, does not have any physical meaning. This effect is due to the unfolding of pulse height spectrum measured using response function considering a gaussian resolution for the NE-213 detector.

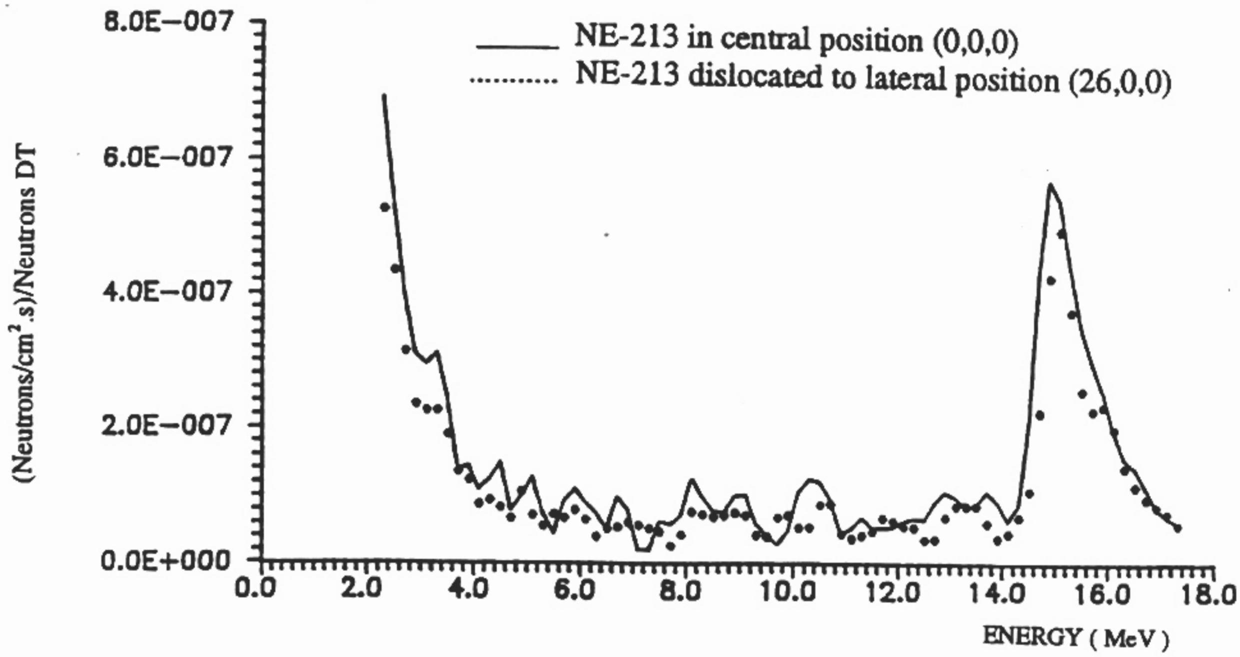

Fig. 3. Measured neutron energy spectra.

The observed difference between neutron energy spectra measured and calculated below 4 MeV is due to the neutrons from DD reaction produced in the accelerator target and not considered in the neutron source spectrum for the MCNP code because it is difficult to estimate the contribution of these DD neutrons.

Results of comparisons between measured and calculated neutron energy spectra have been showed good agreement in the spectrum shape (figure 4) and, deviation of about 2% between integrated spectrum from 4.1 to 17.3 MeV.

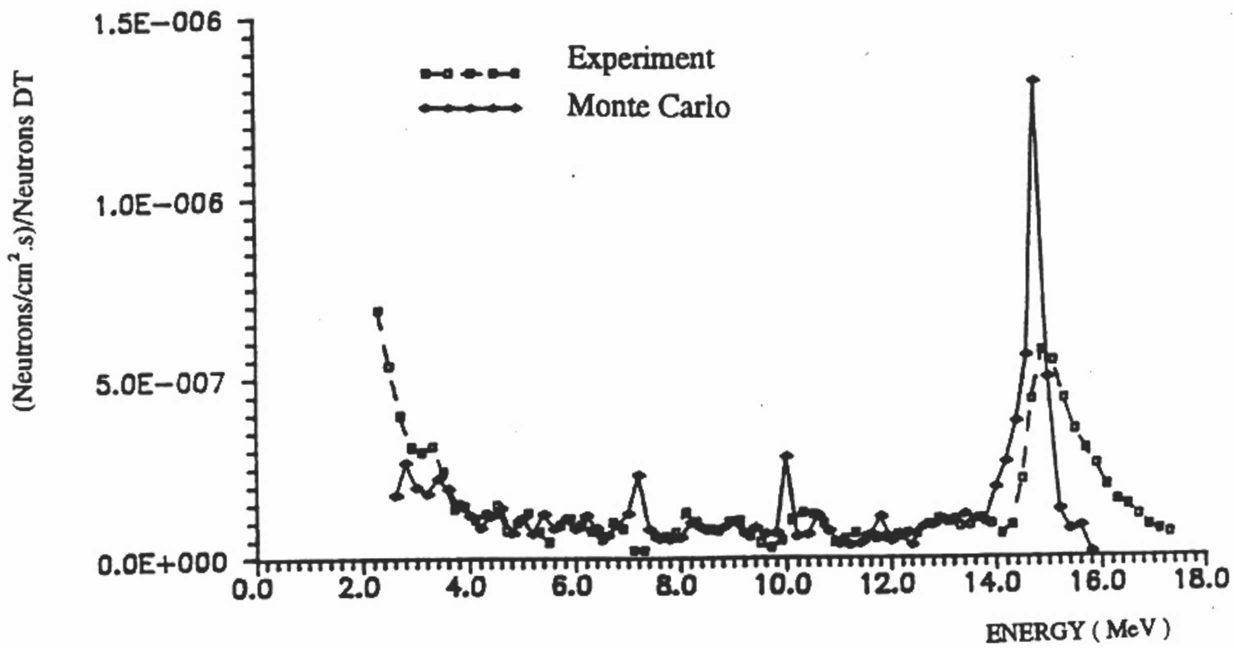

Fig. 4. Comparison between measured and calculated neutron energy spectrum.

REFERENCES

1. P. R. P. COELHO, A. A. DA SILVA AND J. R. MAIORINO, "Neutron Energy Spectrum Measurements of Neutron Sources with an NE-213 Spectrometer," *Nucl. Instr. and Meth. in Phys. Res. A280,* 270 (1989).

2 P. R. P. COELHO, "Medida da Produção de Nêutrons na Reação DT," *V CGEN - Congresso Geral de Energia Nuclear*, Rio de Janeiro (1994), p 101.

3. L. J. ANTUNES, G. BORKER, H. KLEIN AND G. BULSKI, "Unfolding of NE 213 Scintillation Spectra Compared with Neutron Time-of-Flight Measurements," *Radiat. Effects* 96, 33 (1986).

4. G. F. KNOLL,. *Radiation detection and measurement,* John Wiley & Sons, Inc, New York (1979).

5. J. F. BRIESMEISTER, "MCNP- A General Monte Carlo Code for Neutron and Photon Transport- Version 3A," LA-7396-M, Los Alamos National Laboratory (1986).