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Photoneutron spectrum estimation and its experimental validation using neutron REM (Roentgen Equivalent in Man) detector

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ABSTRACT

MCNP model of photoneutron source has been developed. The leakage neutron spectrum from the beryllium target of thickness 2 cm and 9 cm diameter is estimated by Monte Carlo calculation using MCNP code. The NCRP-38 flux to dose conversion factors has been used for neutron dose calculation. The calculated neutron dose at 42 cm from the source is 5.29E−04 ± 5.9% Sv/h. The experimentally measured neutron dose using the Neutron rem detector is 5.52E−04 ± 3.9% Sv/h.

1. Introduction

The Variable Energy Microtron at Mangalore University facilitates acceleration of electrons up to 9.0 MeV with RF cavity type I and 12 MeV with RF cavity type II. Bremsstrahlung photons of peak energy corresponding to incident electron energy are produced using tantalum target. Recently photoneutron source has been installed at Microtron Centre. The photoneutrons are produced by using Beryllium target irradiated by the microtron based bremsstrahlung radiation. We made detailed calculations for estimating the photon yield and spectrum from tantalum target using EGS4 (Neilson et al., 1985) simulation technique and photoneutron yield (2×10^9 n/s) from beryllium target using MCNP (Briesmeister, 1993) code. Details of these calculations are discussed in our earlier work (Eshwarappa et al., 2005, 2007).

With the wide spread applications of neutron sources, the fundamental importance of shielding and dosimetry have increased. A dosimeter that provides correct results in any neutron field does not exist. Readouts determined by the neutron monitor are greatly dependent on the neutron spectrum at the work place. Therefore, measurement of neutron energy spectra becomes essential. The threshold detectors could be used to measure neutron spectrum. However, EGS4 simulation results of the estimation of bremsstrahlung spectrum of tantalum (Eshwarappa et al., 2005) shows that the intensity of the higher energy photons is too small, so also the intensity of higher energy photoneutrons. For those threshold detectors which are having higher threshold energy for a particular reaction, this higher energy neutron flux is not enough to build minimum activity even for long duration of irradiation. Therefore, using threshold activation detectors the measurement of higher energy part of the photoneutron spectra is difficult. Hence, for validating neutron rem monitor readout and quick assessment of neutron spectrum; we have approached the Monte Carlo calculation method for estimation of photoneutron spectrum.

2. Methodology

The incident bremsstrahlung photon flux is transported inside the Be target to estimate photon flux and spectrum. The source term for photon transport is determined by EGS4 simulation and is introduced into SDEF card of MCNP input file which has the goal of calculating photon track length distribution. The results of photon transport are used for the neutron yield ($n\sigma\phi$) estimation using the photonuclear cross section (shown in Fig. 1) data supplied by IAEA (EXFOR, 2003; Goryachev et al., 1992). These neutrons are transported inside the beryllium to estimate the neutron leakage spectrum. The source term for neutron transport is determined from the photon transport results and is introduced into the SDEF

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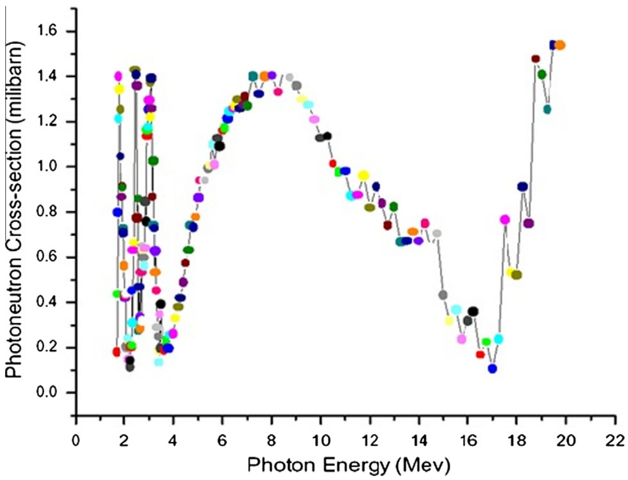

Fig. 1. The $Be^9(\gamma, n)Be^8$ reaction cross section in the energy range from threshold to 20 MeV.

card of input file which has the specific goal of calculating neutron leakage spectra. SDEF referred as general source card. The source is distributed uniformly in volume throughout cell (beryllium solid cylinder). The cell is enclosed by a sampling volume centered at 0, 0, 0. The axis of the sampling volume is the line through 0, 0, 0 in the direction 0, 0, 1. The inner and outer radii of the sampling volume are 0 and 4.5 cm, and it extends along 0, 0, 1 for a distance 3 cm from 0, 0, 0. The energies of the source particles are sampled using the results of neutron yield calculation (Eshwarappa et al., 2005) where the photon track length distribution in beryllium was estimated in various energy bins. The neutron dose is calculated by using leakage photoneutron spectrum and neutron flux to dose conversion factors (extracted from the appendix-H-1 of MCNP user manual).

3. MCNP simulation

MCNP simulation is carried out for beryllium target of solid cylindrical having a dimension of 9 cm diameter and 6 cm thickness. The plot of the geometry of MCNP model is shown in Fig. 2. As many as 1,000,000 histories were run in order to produce reliable confidence intervals and surface flux tally F2 (unit $1/cm^2$) was used. The estimated spectrum was multiplied with the NCRP flux dose conversion factors ($(Sv/h)/n/cm^2/s$). Finally resultant product was multiplied with the total neutron yield (n/s). The above said steps of converting the calculation results to dose rate are formulated in the form of dimensionally balanced equation as shown below.

$$\frac{1}{cm^2} * \frac{Sv/hr}{n/cm^2/s} * \frac{n}{s} = Sv/h$$

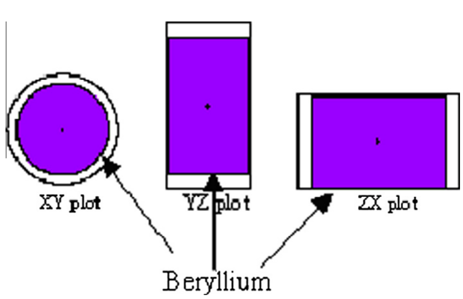

Fig. 2. MCNP plot of the geometry of photoneutron target.

The calculated leakage spectrum is shown in Fig. 3.

4. Experimental details

The neutron rem detector (Procured from Thermo Electron Corporation, USA) together with Eberline's ASP-2e rate meter is used for neutron dose measurement. The Eberline Neutron Rem Detector (NRD) sphere is a portable instrument for the detection and measurement of the dose equivalent rate from neutron radiation. Neutron rem detector ("A modified-sphere neutron detector" reported by Hankins (1967) is a nine-inch-diameter, cadmium-loaded polyethylene sphere with a boron trifluoride (BF3) detector in the centre consists of a cylindrical aluminum proportional tube filled with a BF3 fill gas at a pressure of 0.5–1.0 atmospheres. The energy response of this detector for the neutrons of the energy range from 0.025 eV to about 10 MeV was reported by Hankins (1967). Similar types of rem detectors are reported by (Olsher et al., 2004, 2000). The BF3 tube allows excellent gamma rejection. Eberline's ASP-2e is a general-purpose microprocessor based portable radiation meter. Non-volatile storage is available for one complete set of probe parameters with power supply range of 300–2500 V and detection thresholds of 0.5–60 mV. The detector is shown to be having gamma rejection up to 500 R/h (5 Sv/h) (Hankins, 1967). Linearity checks have been made by placing the detector in three neutron fields: 27 mRem/h, 100 mRem/h and 200 mRem/h (100 mRem/h value was used as a calibration value). The observed readings are shown in Table 1.

4.1. Photon dose estimation

The Monte Carlo code EGS4 (Neilson et al., 1985) was used to obtain angular distribution of bremsstrahlung photons for Ta target. The electron beam (8.75 MeV) is incident on tantalum foil of

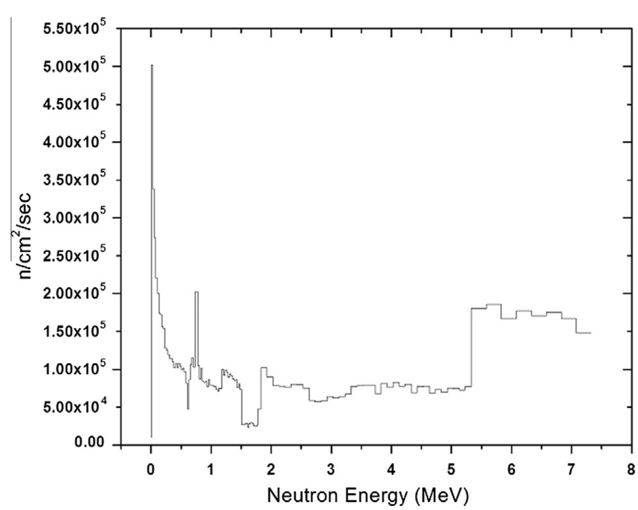

Fig. 3. Calculated photoneutron spectrum of Beryllium irradiated by the 8.75 MeV microtron based bremsstrahlung radiation.

Table 1
Neutron rem detector linearity check results(100 mRem/h value was used as a calibration value).

Field (mRem/h)	Results (mRem/h)	Difference (%)
27	26.1	3.3
100	100	–
200	197	1.5