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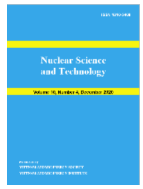

Development of a neutron detector for radiation protection monitoring

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Abstract: The paper presents the results of the development of a neutron detector for radiation protection purposes. Monte Carlo simulations, using MCNP5 code, were performed to optimize the configuration of the neutron detector. The developed detector consists of a ^{3}He proportional counter embedded in a multi-layer moderator made of high-density polyethylene (HDPE) and Cadmium. The characteristics of the developed neutron detector including neutron fluence response and ambient dose equivalent response were calculated, analyzed and compared with those from other neutron survey meters. The simulation model and computed results were assessed through experimental measurements at the Secondary Standards Dosimetry Laboratory of the Institute for Nuclear Science and Technology (INST). A good agreement between the simulated and experimental results was observed within 9.3% for ^{241}Am-Be source and four simulated workplace neutron fields.

Keywords: *developed neutron detector, neutron fluence response, ambient dose equivalent response.*

I. INTRODUCTION

In recent years, with the increase in the use of radioactive sources, especially the applications of neutrons in industry and medicine, radiation protection monitoring continues to be a subject of increasing general interest. Apart from the main sources of neutrons such as sealed radio-isotopic sources, nuclear reactors, and neutron generators, neutrons are also encountered in the high energy particle accelerator systems. The radiation field outside the shielding of accelerators is frequently dominated by the neutron component, which has complex and non-uniform energy distributions [1]. Exposure to free neutrons can be hazardous since the interaction of neutrons with molecules in the body can lead to chromosome damage and adversely affects human health [2]. According to the recommendation of the International Commission on Radiological Protection (ICRP) for area monitoring, the ambient dose equivalent, H*(10), is used as an approximation of the protection quantity in radiation measurements of external exposure [3]. The mensuration of the ambient dose equivalent will furnish the necessary information for controlling the radiation at workplaces and definition of controlled or forbidden areas.

Neutron survey meters are employed to measure the neutron ambient dose equivalent rate. They are designed based on single or multiple detectors enclosed by the single or

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multi-layer moderator to have a response function that is independent of the energy and direction of incident radiation [4, 5]. Depending on the measured energy range, neutron survey meters are classified in two main groups: conventional neutron survey meters (e.g., LB 6411 [6], TPS 451C [5]) and extended neutron survey meters (e.g., LINUS [7], WENDI [8], LUPIN [9], PRESCILA [10]). In the first group, a combination of Cadmium (or/and Boron) and high hydrogen concentration materials are used as filters and moderators to measure neutrons in the range from thermal to 15 MeV. Meanwhile, by adding heavy metals such as lead, copper or tungsten to the moderator, extended neutron survey meters are capable of measuring the ambient dose equivalent rate for a neutron field of energy up to several hundred MeV through spallation reactions (n, xn). Although in reality most of the instruments encounter many difficulties at energies between 1 keV and 200 keV, where over-estimations by up to a factor of ten are rather the normal case, however, they are still widely used for radiation monitoring due to their advantage capability of measuring and displaying the neutron ambient dose equivalent rate in real-time.

In this study, the neutron detector was designed by following the principle of conventional neutron survey meters. Monte Carlo N-Particle simulations (using MCNP5 code [11]) were applied to optimize the configuration of the neutron detector. The fluence response function of the developed neutron detector was calculated and validated by experimental measurements at the Secondary Standards Dosimetry Laboratory of the Institute for Nuclear Science and Technology (INST). The H*(10) response was also investigated in comparison with those from other neutron survey meters to confirm the proper operation of the developed neutron detector.

II. MATERIAL AND METHOD

A. Helium proportional counter

In this work, a cylindrical ^{3}He proportional counter was used. The counter, supplied by the Centronic UK Ltd, has dimensions of 2.54 cm in diameter and 15 cm in length. It is filled with ^{3}He gas at 2 atm, corresponding to 5.02 x 10^{19} atoms/cm^3 at room temperature. The cathode shell is made of stainless steel with a thickness of 1 mm. Figure 1 shows the sketch of the ^{3}He proportional counter.

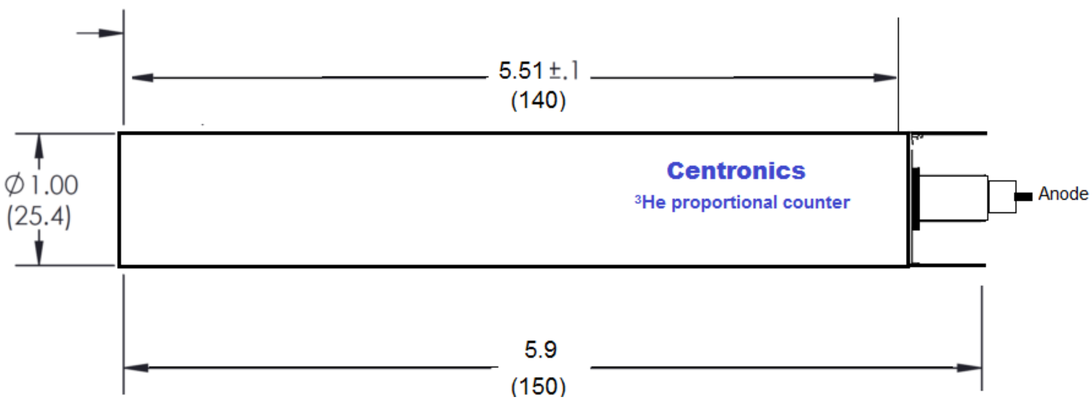

Fig. 1. The structure of the ^{3}He proportional counter (dimensions in mm)

B. Design principle

The conventional true value of neutron ambient dose equivalent rate, $H^*(10)$ (in pSv/h) measured by neutron survey meters, is expressed as Eq. (1).

$$H^*(10) = \int \phi(E).h(E).dE \qquad (1)$$

Where, $\phi(E)$ is the incident spectral neutron fluence rate (in $h^{-1}.cm^{-2}$); $h(E)$ is the neutron fluence-to-dose equivalent conversion factor (in $pSv.cm^2$), given by ICRP 74 [3].

The ambient dose equivalent rate can be calculated by using the Eq. (1) if the neutron energy spectrum is known. A typical method using the Bonner sphere spectrometer can be applied for measuring the neutron spectrum, but it requires long measuring time and a complex deconvolution algorithm [12]. An alternative approach to be applied to most neutron survey meters is to design the neutron detector to have a fluence response function with a similar form as the neutron fluence-to-dose equivalent conversion function given by ICRP 74. This method allows determining the neutron ambient dose equivalent rate through the reading number recorded without regard to the energy of incident radiation.

Suppose that: M and $R(E)$ are the reading number (count rate) and the fluence response (in cm^2) of the neutron detector, respectively. The interrelation between these quantities and the incident spectral neutron fluence rate is expressed by Eq. (2):

$$M = \int \phi(E).R(E).dE \qquad (2)$$

From Eq. (1) and Eq. (2), it is obvious that if the fluence responses, $R(E)$, have the same energy dependence as the neutron fluence-to-dose equivalent conversion factors, $h(E)$, then the reading of a neutron detector can be calibrated based on a conventional true value of $H^*(10)$ with a calibration factor of c as given by Eq. (3).

$$H^*(10) = c.M \qquad (3)$$

To achieve detector configuration according to the above principle, first of all, simulations were performed to evaluate the fluence response of the ^{3}He proportional counter with various moderator configurations using the high-density polyethylene (HDPE). The thickness of the moderator was gradually increased from 1 cm to 14 cm. Obviously, with the increase of polyethylene thickness, the fluence response of the neutron detector is significantly improved. However, it should be noted that the large thickness of the moderator makes it difficult to use in the workplace. In addition, the helium proportional counter is extremely sensitive to thermal neutrons, leading to a drastic over-estimation of the H*(10) at intermediate energies. To reduce their effect, the helium proportional counter must be wrapped by a perforated cadmium shell. The parameters including the diameter of the moderator, position and perforated fraction of the cadmium shell were adjusted iteratively until a best fluence response function is achieved. To enhance the accuracy and reliability of simulation results, measurements were also carried out to check the density and the purity of materials. The densities of polyethylene and cadmium were determined to be 0.95 g/cm^3 and 8.72 g/cm^3, respectively. These values have been obtained from measurements of cylindrical samples taken from the same batch of polyethylene and cadmium used for design. The purity of the materials is confirmed by the X-Ray fluorescence (XRF) method, giving a result greater than 99 % as stated by the manufacturer.

In MCNP5 simulations, the fluence response, $R(E)$, is determined through the number of capture reactions, $^{3}He(n,p)t$, generated in the active volume of the ^{3}He proportional counter. This number is obtained from F4 tally and the associated FM card by using the appropriate multiplication factor. The neutron cross-section was taken from the Evaluated Nuclear Data Files library (i.e., ENDF/B-VI [13]). The S(α,β) treatment, poly.01t [11], was used to account for the carbon and hydrogen chemical binding at room temperature. For each configuration of the neutron detector, simulations were performed with 29 mono-energetic sources ranging from 10^{-9} MeV to 15 MeV. In order to minimize statistical errors of MCNP5 outputs, the variance reduction techniques based on the geometry splitting method were applied [14]. The source particle histories were also considered to keep statistical uncertainties below 3%.

III. RESULT AND DISCUSSION

A. The developed neutron detector

Based on the simulation results, the configuration of the neutron detector with an appropriate response function as the recommendation of IEC 61005-2014 [15] was selected for designing the neutron detector. Figure 2 depicts the structure of the developed neutron detector, in which the ^{3}He proportional counter is surrounded by a 6 cm thick outer polyethylene moderator, a 2.7 cm thick inner polyethylene moderator, and a 3 mm thick cadmium shell. In order to extend the response to thermal neutrons, the cadmium shell is perforated so that the opening accounts for 10% of the surface area of the ^{3}He counter. The self-developed neutron detector is about 6 kg weigh with the outer dimensions of 20.5 cm in diameter and 24.5 cm in length.

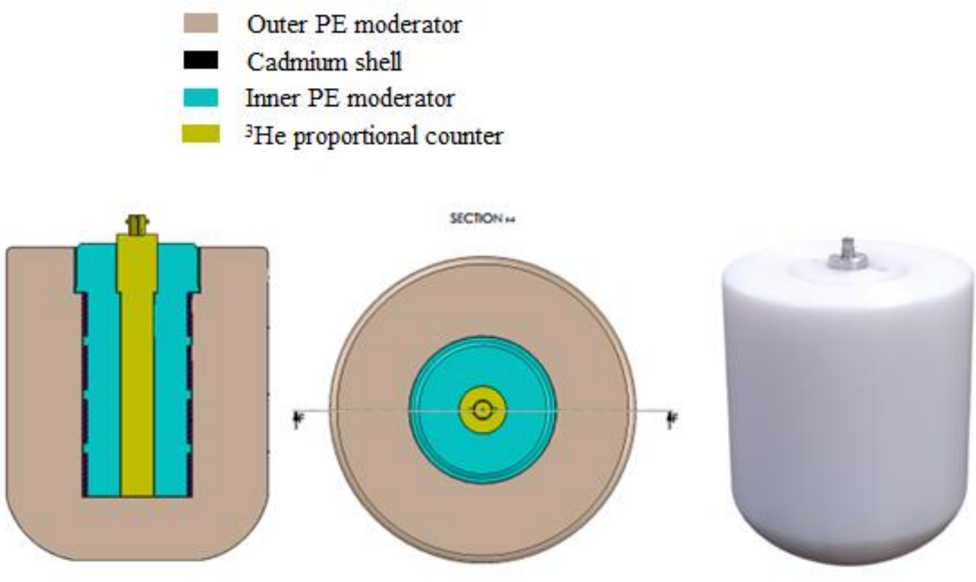

Fig. 2. The developed neutron detector configuration (from the left to the right): (1) side view, (2) top view, and (3) outer view

- ***Neutron fluence response***

The simulation results are given in Figure 3, which shows the variation of the fluence response as a function of incident neutron energy. In general, fluence responses of the developed detector closely follow the shape of the neutron fluence-to-dose equivalent conversion curve, $h(E)$ [3, 15], in the energy range from 50 keV to 10 MeV. At energies between 10 eV and 50 keV and above 10 MeV, there is a significant difference between the shape of the neutron fluence response function and the recommended ICRP 74 curve. However, it should be noted that this behavior is a common feature of conventional neutron survey meters.

- ***Ambient dose equivalent response***

To investigate the variation of ambient dose equivalent due to incident neutron energy, computed fluence responses were converted to $H^*(10)$ responses and then normalized to the ^{241}Am-Be source. The $H^*(10)$ response is defined as the ratio between the fluence response and the corresponding neutron fluence-to-ambient dose equivalent conversion factor. As a result, Figure 4 shows the relative H*(10) response function of the developed neutron detector (normalized to $^{241}Am - Be$) together with those obtained from other neutron survey meters (i.e. the Aloka TPS-451C, Hitachi Co, Ltd.; the NDN1, Fuji Electric Co, Ltd; and the NSN1, Fuji Electric Co, Ltd [5]).

The developed neutron detector over-estimates the neutron ambient dose equivalent rate by a factor between 1.2 and 5.3 in the energy region from 10 eV to 50 keV and under-estimates the neutron ambient dose equivalent rate from 30% to 50% at energies above 10 MeV. It can be seen that the relative $H^*(10)$ response of the developed neutron detector has a similar tendency with those of other commercial neutron survey meters. These values also comply with the recommendations of the international standard IEC 61005 [15]. That means the performance of the developed neutron detector over the range from thermal to 15 MeV is comparable to that of commercial neutron survey meters.

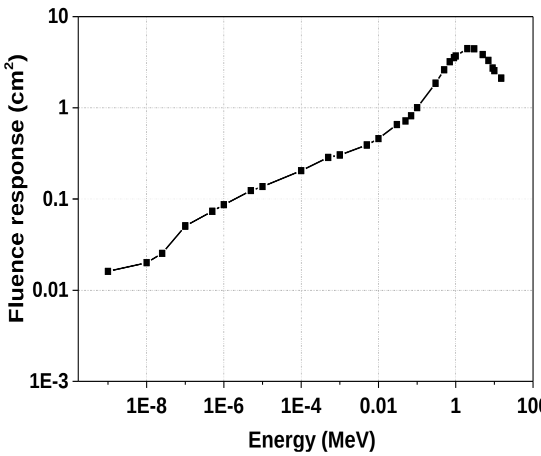

Fig 3. The variation of the fluence response as a function of neutron energy

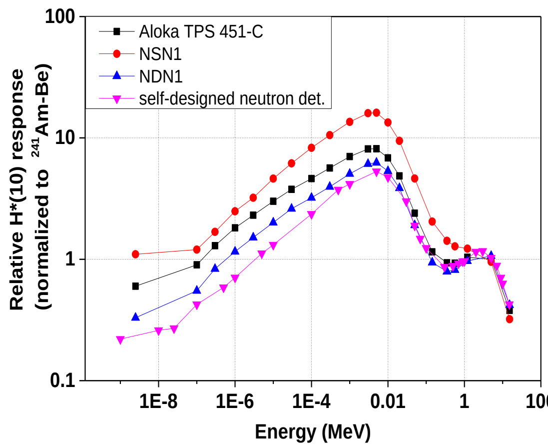

Fig. 4. H*(10) responses of the developed neutron detector and three commercial neutron survey meters (J.Saegusa et al., [5]). The results are normalized at ^{241}Am-Be response.

B. Experimental validation

In order to validate the simulated results, measurements are usually performed in standardized mono-energetic neutron fields as recommendations of IEC 61005-2014 [15]. However, it presents many difficulties for testing in a wide range of energy due to the limitation of experimental conditions in Vietnam. To achieve meaningful results, the simulation model was corroborated via experimental measurements at the Secondary

Table I. The fluence response obtained from MCNP calculations and measurements for ^{241}Am-Be source and four simulated workplace neutron fields

Neutron source	Neutron fluence response (cm^2)		Relative deviation (%)
	MCNP Calculation	Measurement	
^{241}Am-Be	2.55 ± 0.07	2.81 ± 0.09	9.25
20 cm Mod	1.87 ± 0.03	2.04 ± 0.06	8.33
25 cm Mod.	1.71 ± 0.04	1.88 ± 0.05	9.04
30 cm Mod.	1.66 ± 0.03	1.81 ± 0.06	8.29
35 cm Mod.	1.63 ± 0.02	1.72 ± 0.06	5.23

Standards Dosimetry Laboratory of INST. A comparison of the neutron fluence response was performed between MCNP5 simulations and experiments for ^{241}Am-Be source and four simulated workplace neutron fields. The emission rate of the ^{241}Am-Be source was 1.287 x 10^7 (n/s) with an expanded uncertainty of about 2.9% (k=2). Simulated workplace neutron fields are formed by moderating ^{241}Am-Be source with 4 polyethylene spheres with diameters of 20 cm, 25 cm, 30 cm, and 35 cm. The characteristics of neutron calibration fields were described in detail in the Ref. [16, 17]. In experiments, the developed neutron detector was placed at 150 cm from the source and connected to a read-out system including pre-amplifier, shaping amplifier, and single-channel analyzer (SCA*)*. In order to eliminate the gamma interference and overall electronic noise, the lower-level discrimination of SCA is applied by an appropriate value of about 300 mV. The measurement data are transmitted to a personal computer via RS-232 communication.

The fluence response of the neutron detector is determined by the ratio between the count rate and the incident neutron fluence rate, as given by Eq. (2). The obtained results are presented in Table I, which shows the relative deviation between simulations and experiments within ± 9.3 %. The consistency between these results implies that the calculation model and simulated results given in section III.1 are reliable.

IV. CONCLUSIONS

A neutron detector, consists of a ^3He proportional counter and a multi-layer moderator, has been developed to measure neutrons in the energy range from thermal to 15 MeV. The instrument has dimensions of 20.5 x 24.5 cm and weighs about 6 kg. The characteristics of the neutron fluence response and the ambient dose equivalent response were investigated using MCNP5 simulations and validated by experiments. The results show that the neutron detector is suitable for neutron measurements and the radiation safety assessment.

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