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A multi-detector neutron spectrometer with nearly isotropic response for environmental and workplace monitoring

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ABSTRACT

This communication describes an improved design for a neutron spectrometer consisting of ^{6}Li thermoluminescent dosemeters located at selected positions within a single moderating polyethylene sphere. The spatial arrangement of the dosemeters has been designed using the MCNPX Monte Carlo code to calculate the response matrix for 56 log-equidistant energies from 10^{-9} to 100 MeV, looking for a configuration that permits to obtain a nearly isotropic response for neutrons in the energy range from thermal to 20 MeV. The feasibility of the proposed spectrometer and the isotropy of its response have been evaluated by simulating exposures to different reference and workplace neutron fields. The FRUIT code has been used for unfolding purposes. The results of the simulations as well as the experimental tests confirm the suitability of the prototype for environmental and workplace monitoring applications.

1. Introduction

Survey instruments for workplace monitoring around neutron generating facilities should be designed to measure the operational quantity ambient dose equivalent, $H^*(10)$ [1,2]. This quantity is related to the neutron fluence by a set of conversion coefficients which are calculated through radiation transport methods [3]. Therefore, the energy dependence of the reading per unit fluence of a neutron area monitor should be ideally proportional to the fluence-to-ambient dose equivalent conversion coefficients. These conversion coefficients are available in publications ICRP 74 [3] and ICRU 57 [4]. Nevertheless, this requirement results impossible to fulfill in practice because of the very broad range of energies over which the neutron spectrum extends and the strong dependence of the conversion coefficients on neutron energy, specially in the range between 10 keV and 1 MeV, where their values increase by a factor up to 40 [2–4].

These problems arising from the poor energy response of currently available survey instruments can be overcome if the neutron spectrum is measured and the ambient dose equivalent is directly determined through the conversion coefficients [3,4]. Once the energy distribution of fluence, Φ_E, is known, this information can be used to calculate any fluence-related dosimetric quantity by using the corresponding energy dependent conversion coefficients [2].

For more than three decades, the Bonner spheres spectrometer (BSS) has been widely employed as the main reference instrument for assessing the neutron spectra [5]. Basically, the BSS consists of a thermal neutron detector placed at the center of several moderating spheres of increasing diameters. The neutron spectrum is determined through the response matrix of the BSS system for unfolding the set of experimental data from different spheres. Therefore, the procedure requires successive exposures and long irradiation sessions. A reference instrument should be employed when the neutron field varies with time.

More recently, alternative designs based on multiple dosemeters appropriately arranged within a single moderating sphere have been proposed and discussed [6–9]. In particular, a neutron spectrometer based on thermoluminiscent dosemeters (TLD) allocated in a 30 cm polyethylene sphere was designed at CIEMAT[1] for radiation protection applications. This spectrometer is capable to respond to neutrons in the energy range 1 keV–20 MeV [7]. Nevertheless, such a device showed a significant directional dependence of the response [10] that was not adequate for the determination of the total fluence and ambient dose equivalent, both requiring an isotropic response [2].

The purpose of this paper is to describe the design of a new spectrometer for workplace monitoring, with a nearly isotropic response from thermal to fast neutron energies. The response matrix has been calculated with the MCNPX [11] Monte Carlo code. The feasibility of the device has been evaluated by

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simulating the exposure to different neutron spectra with MCNPX [11] and the FRUIT unfolding code [12].

2. Calculations

Calculations of the response functions and simulation of exposures have been performed with MCNPX and the ENDF-60 cross-section library. Room temperature cross-section tables, S(α,β), have been used for modelling neutron scattering in polyethylene. Photonuclear (γ,n) reactions and the production of secondary neutrons have been also considered. No variance reduction technique was employed and a cut-off in the number of histories has been applied to obtain statistical uncertainties lower than 3% (1 s.d.) in all the cases.

The moderator material was standard polyethylene $(CH2)_n$, 0.927 g/cm^3 in density. The dosemeters were standard TLD-600 (LiF:Ti,Mg) chips $0.3\times0.3\times0.09$ cm^3 in volume and 2.64 g/cm^3 in density. The ^{6}Li enrichment was 94.9%, the remainder being ^{7}Li. The response of the TL dosemeters has been calculated through the track-length scoring option for the fluence (F4 tally). In particular, the TLD response is equal to the number of ^{6}Li(n,α)^{3}H reactions occurring within the dosemeter volume, normalized per incident neutron fluence [13,14], i.e.

$$N_{(n,\alpha)} = \int dE\Phi_E V\Sigma_{(n,\alpha)} = \int dE\Phi_E V\rho_{at}\sigma_{(n,\alpha)} \tag{1}$$

where $\sigma_{(n,\alpha)}$, $\Sigma_{(n,\alpha)}$ are, respectively, the microscopic and macroscopic cross-sections for (n,α) reactions, V is the volume, Φ_E is the energy distribution of neutron fluence, ρ_{at} is the atomic density (i.e. atoms per unit volume) and $N_{(n,\alpha)}$ is the number of (n,α) reactions in the considered volume[2].

In practice, the measured TL readout (glow curve area) would be actually proportional to the number of ^{6}Li(n,α)^{3}H reactions calculated according to Eq. (1), so a calibration by exposing the detector to a known fluence would be required. Moreover, the gamma contribution to TLD-600 need to be subtracted, for instance using pairs of ^{6}Li/^{7}Li based dosimeters (TLD-600/TLD-700) [15].

3. Spectrometer design

A previous study [8] discussed the spectral capability of a neutron spectrometer based on ^{6}LiF TL dosemeters located along the diameter of a single moderating sphere in polyethylene. Nevertheless, such a device showed a significant directional dependence of the response [10] that could be inconvenient for those applications when directional information is not required, and where the spectral fluence (angular independent), Φ_E, (or a related quantity, e.g. the ambient dose equivalent, $H^*(10)$) has to be determined.

In general, the measurement obtained for a dosemeter located at a position, $\vec{r}$, referred to the center of the sphere, is given by

$$M(\vec{r}) = \int\int\int dE\, d^2\Omega \Phi_{E,\Omega} R(E,\Omega,\vec{r}_i). \tag{2}$$

In all those exposure situations where the angular dependence cannot be neglected, a method to reduce the directional dependence could be to consider the average response of a given number of dosemeters, N, symmetrically distributed at the same distance, d, from the center, i.e.

$$\frac{1}{N}\sum_{i=1}^{N} M(\vec{r}_i) = \int\int\int dE\, d^2\Omega \Phi_{E,\Omega} \frac{1}{N}\sum_{i=1}^{N} R(E,\Omega,\vec{r}_i) \tag{3}$$

with $|\vec{r}_i| = d$, for $i=1, 2 \ldots N$.

In particular, if the detector is isotropically irradiated, Eq. (2) will become

$$\frac{1}{N}\sum_{i=1}^{N} M^{(ISO)}(\vec{r}_i) = \int\int\int dE\, d^2\Omega \Phi_{E,\Omega} \frac{1}{N}\sum_{i=1}^{N} R^{(ISO)}(E,\vec{r}_i) \tag{4}$$

that can be formally written in terms of Φ_E as

$$M_d^{(ISO)} = \int dE\Phi_E R_d^{(ISO)}(E) \tag{5}$$

with

$$M_d^{(ISO)} = \frac{1}{N}\sum_{i=1}^{N} M^{(ISO)}(\vec{r}_i) \tag{6}$$

$$R_d^{(ISO)}(E) = \frac{1}{N}\sum_{i=1}^{N} R^{(ISO)}(E,\vec{r}_i) \tag{7}$$

In a theoretical case of a continuous distribution of infinite point dosemeters, uniformly distributed on a spherical surface of radius d, Eqs. (5)–(7) will be also valid for any angular distribution of the incident neutrons, because of the spherical symmetry. In practice if the number of dosemeters is large enough, the angular dependence of $R_d(E)$ could be neglected and Eqs. (5)–(7) would become

$$M_d \approx \int dE\Phi_E R_d(E) \tag{8}$$

$$R_d(E) = R_d^{(ISO)}(E) + \Delta R_d \tag{9}$$

where M_d is the average readout of the dosemeters located at a distance d from the center, $R_d(E)$ is the averaged response function for such dosemeters and ΔR_d is the discrepancy between $R_d(E)$ and the response matrix calculated for isotropic irradiation, $R_d^{(ISO)}(E)$. In this case, the response functions will depend only on the neutron energy, E, and the radial distance, d.

Two spatial distributions have been evaluated in order to determine the minimum number of dosemeters required to obtain a nearly isotropic response. The first one, shown in Fig. 1a, consists of six positions at different distances d from the center, along three perpendicular axis that will be referred in the following as (1 0 0), (0 1 0) and (0 0 1) [i.e., $(\pm d\,0\,0)$, $(0\pm d\,0)$, $(0\,0\pm d)$]. The second configuration, shown in Fig. 1b, includes eight additional positions for the maximum radial distance, d_S, thus located in the corners of a cube at coordinates

$$\left\{\frac{1}{\sqrt{3}}(\pm d_S,\ \pm d_S,\ \pm d_S)\right\}. \tag{10}$$

Taking into account that previous calculations [8] for a 30 cm diameter polyethylene sphere showed a good energy resolution up to 20 MeV, the same size has been considered for the new design. The number of radial positions have been increased to obtain a better energy resolution and to increase the robustness of the device. The latter aspect should be considered carefully because during operational measurements the loss of a TLD reading it is not infrequent. Thus, after some preliminary simulations, a set of eight radial distances has been selected: 0, 3, 6, 9, 10.5, 12, 13 and 14 cm, any of them with six dosemeters located along the three perpendicular axis, and eight additional dosemeters for the shallowest position (14 cm).

[2] This has been done applying a track-length tally multiplier card, FM4, to a pure ^{6}Li material (e.g. labelled 99) specifically declared for tallying, in this case FM4 5.0796E-4 99 105, where 105 identify the (n,α) microscopic cross-section (with units of barns) within the MCNPX cross-section library and $V\rho_{at}=5.796\times10^{-4}$ barn^{-1} cm^2.