A Novel Neutron Dosimeter

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Abstract

Background: Neutron contamination in our environment can cause harmful biological effects on human body. Therefore, many efforts have been made to construct a neutron dosimeter to estimate the received dose.

Objective: To design a simple neutron dosimeter.

Methods: The primary dosimeter had ^3He as a spherical thermal neutron detector encircled by paraffin 10 cm in radius. Then, the paraffin sphere was replaced with ICRU that contains soft tissue and dose equivalent determined as a desired output. Finally, an appropriate relation between counts and dose equivalent was found.

Results: Results on the energies below 1 MeV demonstrated the similarity of changing process of these two quantities, so they could relate to each other with an adequate factor. To find the best fit, different factors considered and the smallest χ^2 (goodness of fit) was 1.17×10^5 At the next step, two covers of cadmium and gadolinium, separately, put around the detector to improve χ^2, which was 2.51×10^4 for cadmium cover and 6.33×10^3 for gadolinium cover. As we see, gadolinium cover fits the curves of counts and equivalent dose in a better way.

Conclusion: Applying this simple dosimeter lead us to estimate whole body dose equivalent.

Keywords

Neutrons; Dosimetry; Dose Equivalent; Cadmium; Gadolinium.

Introduction

Human body is continuously exposed to radiation contamination. One of the hazardous radiation contaminations is exposure to neutrons which are difficult to detect. A single exposure to a high-level radiation delivered to the whole body over a very short period may cause serious health risks [1] such as malignant diseases and genetic aberration. After the International Commission on Radiological Protection has issued its new recommendations on radiation protection quantities (ICRP60) [2], there is now increasing interest in commercially available instruments optimized and calibrated for the measurement of neutron ambient dose equivalent [3]. Monitoring of ambient dose equivalent is necessary at the boundaries of radiation-controlled areas and nuclear facilities [4]. Dosimeters can be used as stationary monitors, attached to walls, ceilings, *etc*., in facilities such as particle accelerators and the nuclear reactor. They are able to detect neutrons over a very wide energy range, including thermal energies, with good efficiency. Although there are numerous neutron dosimeters, there is still a critical need to create a more accurate dosimeter [5]. The objective of this study was to design a novel and simple neutron dosimeter that is more suitable

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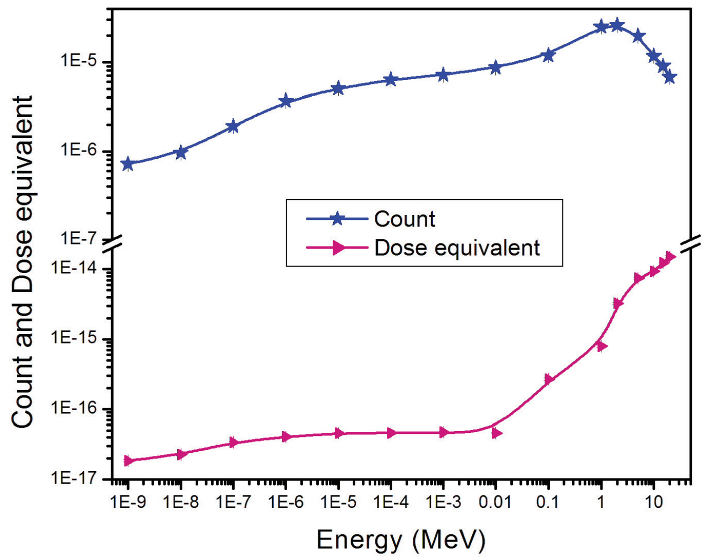

Figure 1: Total dose equivalent and count *vs* energy

for radiation protection purpose.

Materials and Methods

Common neutron dosimeters are sphere-shaped, because in this case the direction of neutron's entrance is not important. There is a detector at the center of the sphere [6]. Different detectors with various sensitivities to energy can be used. Dosimeters count the number of neutrons. Therefore, to assess the radiation dose it is necessity to find an appropriate relation between neutron counts and dose. The desired dose output is "dose equivalent," which gives an estimation of effective dose at a certain point. Therefore, we used simulation with MCNP4C for a beam of monoenergetic neutrons; we obtained two sets of data and found a reliable relationship between count and dose equivalent.

Detector description

Radiation instruments used as survey monitors are either gas filled detectors or solid-state detectors. In our research, we used a ^{3}He proportional counter tube [7]; under a high voltage the detector could create an electrical pulse when a neutron radiation interacts with the gas in the tube [8]. The absorption of a neutron in the nucleus of ^{3}He causes the prompt emission of a proton. These charged particles can then cause ionization of the gas that is produced an electrical pulse. These neutron-measuring proportional counters should be kept in large amounts of hydrogenous material (*e.g.,* paraffin) to slow the neutron to thermal energies. Other surrounding filters allow an appropriate number of neutrons to be detected and thus provide a flat-energy response with respect to dose equivalent. The design and characteristics of these devices are such that the amount of secondary charge collected is proportional to the degree of primary ions produced by the radiation.

Dose equivalent determination

Dose equivalent was first introduced in ICRP21 [9], which is a quantity used in radiation safety. Expressed in Sivert, dose equivalent reflects the physical damage that the radiation may produce. According to ICRU report 51 [10], dose equivalent is calculated as the product of the absorbed dose at a certain point in tissue multiplied by a quality factor

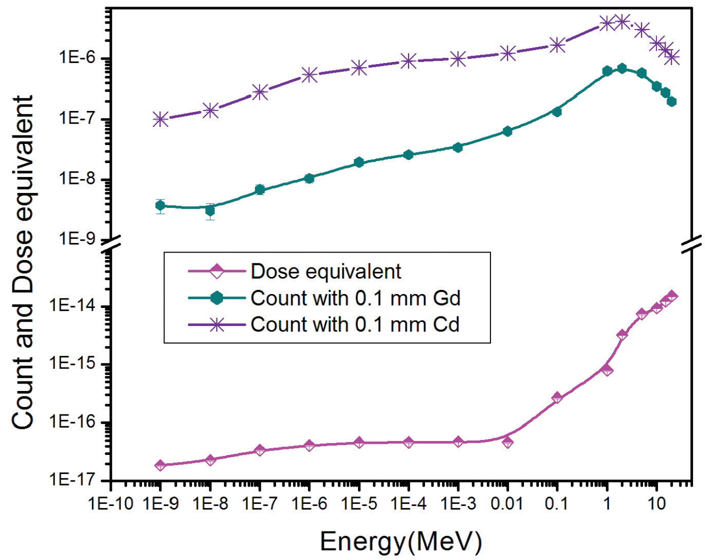

Figure 2: Total dose equivalent and counts *vs* energy in the optimized dosimeter

[11]. In addition, it could not be measured directly, so for its estimation, one should specify its relation with a well measurable quantity, say counts. Therefore, finding the best factor is imperative.

χ^2 test (goodness of fit)

Dose equivalent and the count were calculated in the energy region between 1×10^{-9} MeV and 20 MeV. Then, χ^2 test was used to determine the best coefficient for fitting the two curves that defines the deviation of count from the product of dose equivalent and coefficient according to the following equation:

$$\chi^2 = \sum_{i=1}^{15} \frac{(C_i - deq_i)^2}{\sigma^2(C_i) + \sigma^2(deq_i)}$$

where C is the count, deq is dose equivalent, $\sigma^2(C_i)$ is count's relative error and $\sigma^2(deq_i)$ is dose equivalent's error. The smaller the χ^2, the better will be the fit [12]. In this equation, summation is over energy.

Data acquisition

The primary dosimeter

To determine the neutron counts, a spherical ^3He proportional counter tube with a radius of 0.5 cm was simulated in the center of a paraffin-moderating sphere with a diameter of 10 cm. The required tally for count measurement was F4 [13]. Then, by replacing paraffin by soft tissue we calculated dose equivalent using F6 as the desired tally.

Optimizing the dosimeter

To improve χ^2, we covered the ^3He detector of the primary dosimeter with materials that have high absorption cross sections [14]. Different thicknesses of cadmium, gadolinium, bismuth, and lead were tested. For these new dosimeters count *vs* energy were calculated.

Results

The count and dose equivalent for 15 neutron energies and their relative errors are given in Table 1. The total dose equivalent and count *vs* energy are shown in Figure 1. As we see in the low energies changes of two curves on energies are the same and in higher energy (more than 100 keV) are significantly large. At the next step, different coefficient were tested and with the best one which fit two curves more appropriate, a linear relationship between two

Table 1: Count and dose equivalent for 15 different neutron's energy (the data of the fourth column are the sum of dose equivalent created by neutrons and photons)

Neutron's energy	Counts	Counts' relative error	Dose equivalent	Dose equivalent's relative error
10^{-9}	7.17541×10^{-7}	0.0307	1.86×10^{-17}	0.0437
10^{-8}	9.58283×10^{-7}	0.026	2.29×10^{-17}	0.038
10^{-7}	1.91761×10^{-6}	0.0188	3.41×10^{-17}	0.0308
10^{-6}	3.69013×10^{-6}	0.0138	4.1×10^{-17}	0.0278
10^{-5}	5.0508×10^{-6}	0.0117	4.62×10^{-17}	0.0264
10^{-4}	6.34388×10^{-6}	0.011	4.65×10^{-17}	0.026
10^{-3}	7.17531×10^{-6}	0.0105	4.76×10^{-17}	0.0261
10^{-2}	8.69733×10^{-6}	0.0098	4.63×10^{-17}	0.0263
10^{-1}	1.1862×10^{-5}	0.0087	2.68×10^{-16}	0.0437
1	2.50271×10^{-5}	0.0064	7.93×10^{-16}	0.0368
2	2.60725×10^{-5}	0.0063	3.25×10^{-15}	0.0284
5	1.96509×10^{-5}	0.0073	7.44×10^{-15}	0.0305
10	1.17191×10^{-5}	0.0092	9.37×10^{-15}	0.0245
15	9.01748×10^{-6}	0.0106	1.23×10^{-14}	0.00249
20	6.77532×10^{-6}	0.012	1.51×10^{-14}	0.029

quantities was found.

For the primary dosimeter, the smallest χ^2 corresponding to a factor of 7.91×10^8 was 1.17×10^5. For the optimized dosimeter the best result was for 0.1-mm thick cadmium and 0.1-mm thick gadolinium cover; the χ^2 values decreased significantly (Table 2). Three sets of new data of optimized dosimeters are shown in Figure 2. At higher energies, we observed more similar results, especially for the curve of the count of dosimeter with gadolinium cover. By using gadolinium cover and multiplying dose equivalent curve by the corresponding coefficient, the two curves fit better.

Discussions

We found a better linear correlation between the count and dose equivalent in the optimized dosimeter than the primary one. The optimized neutron dosimeter with gadolinium cover could help us to reach a better estimation of dose equivalent with a higher accuracy. This is due the high absorption cross section of gadolinium [15], which leads to increased number of thermal neutrons. It converts fast neutrons to thermal ones and because of the sensitivity of ^3He to low-energy neutron, those fast particles that had not been counted before could be counted after they became thermal neutrons; in this way the number of detector counts enhanced. The new neutron dosimeter works more efficiently. Due to the differences observed in the shape of the curves in the two energy regions, in the next step we should try to use two different detectors for low-energy and high-energy neutrons to have more accurate results.

Table 2: The smallest χ^2 and its related coefficient at three different situations.

Geometry	Smallest χ^2	Best coefficient
Without cover	1.17×10^5	7.91×10^8
With 0.1 mm cadmium	2.51×10^4	1.24×10^8
With 0.1 mm gadolinium	6.33×10^3	2.3×10^7

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