Journal of the Korean Nuclear Society
Vol. 3, No. 3, September, 1971

≪Technical Report≫

Average and Effective Energies, and Fluence-Dose Equivalent Conversion Factors for ^{239}Pu-Be, ^{241}Am-Li and ^{241}Am-F Neutron Sources

Seung Gy Ro and Young Soo Yoo

Health Physics Division, Atomic Energy Research Institute, Seoul, Korea

(Received June 15, 1971)

Abstract

Average and effective energies for ^{239}Pu-Be, ^{241}Am-Li and ^{241}Am-F neutron sources have been calculated from a number of published data for the neutron spectra and for the dose equivalent as a function of neutron energies by a numerical method. Also a calculation of the dose equivalent conversion factors, i.e., the first collision dose equivalent and the surface (or multicollision) dose equivalent that equals the product of surface-absorbed dose and a corresponding quality factor, per unit fluence of neutrons from these sources has been carried out in the same way as before.

The results are as follows:

1. for average energies
   4.07±0.33, 0.42 and 1.41 MeV;
2. for effective energies based on the concept of the first collision process in the human body
   4.45±0.344, 0.51 and 1.47 MeV;
3. for effective energies based on the concept of the multi-collision process in the human body
   4.50±0.36, 0.50 and 1.45 MeV;
4. for fluence-first collision dose equivalent conversion factors
   $(2.74\pm0.07)10^{-8}$, 1.58×10^{-8} and 2.34×10^{-8} rems/(n/cm^2); and
5. for fluence-surface dose equivalent conversion factors
   $(3.55\pm0.09)10^{-8}$, 2.19×10^{-8} and 2.82×10^{-8} rems/(n/cm^2); respectively.

요 약

중성자 검출기의 교정을 위한 기초자료를 제공할 목적으로 당 연구실에서 보유하고 있는 ^{239}Pu-Be, ^{241}Am-Li 및 ^{241}Am-F 중성자선원에 대한 평균 및 유효에너지와 중성자당 등가선량 환산인자를 계산하였다. 이들은 중성자 스펙트럼 및 중성자의 에너지 함수로서 주어진 등가선량에 대한 여러보고자료에 따라 수치계산법을 도입하여 구해졌다.

그 계산결과는 ^{239}Pu-Be, ^{241}Am-Li 및 ^{241}Am-F 순서로 각각 다음과 같이 주어진다

1. 평균에너지
4.07±0.33, 0.42 및 1.41 MeV;
2. 중성자의 단일충돌과정에 의하여 인체가 받게되는 선량의 개념에 따라 구한 유효에너지
4.45±0.344, 0.51 및 1.47 MeV;
3. 중성자의 다중충돌과정에 의하여 인체가 받게되는 선량의 개념에 따라 구한 유효에너지
4.50±0.36, 0.50 및 1.45 MeV;
4. 중성자당 단일충돌 등가선량 환산인자
$(2.74\pm0.07)10^{-8}$, 1.58×10^{-8} 및 2.34×10^{-8} rems/(n/cm²);
5. 중성자당 다중충돌 등가선량 환산인자
$(3.55\pm0.09)10^{-8}$, 2.19×10^{-8} 및 2.82×10^{-8} rems/(n/cm²).

1. Introduction

Various radioactive neutron sources are extensively used in health physics to calibrate neutron detectors and dosimeters.[1] If these neutron sources are employed for calibration, the neutron fluence-dose equivalent conversion factors as well as the average and effective energies may be of great interest.

Some laboratory neutron sources such as ^{239}Pu-Be, ^{241}Am-Li and ^{241}Am-F are available for neutron dosimetry in our laboratory. In the present work, for making them useful as a standard system of neutron dosimetry a calculational program was conducted determining the average and effective energies, and the dose equivalent conversion factors per unit fluence of neutrons from these sources. By means of a numerical method they were calculated from a number of published data for the neutron spectra[2-9] and for the differential dose equivalent.[10, 13]

2. Calculation

2—1. Average Energies

The average neutron energy $\bar{E}$ of a spectrum is defined by the relationship

$$\bar{E}=\frac{\int_{E_{min}}^{E_{max}}N(E)EdE}{\int_{E_{min}}^{E_{max}}N(E)dE} \quad \cdots\cdots\cdots\cdots(1)$$

Here N(E) dE is the number of neutrons with energies between E and E+dE.

Informations on neutron spectra for the sources of interest were gathered from a number of articles.[2-9] Several papers[2-7] have been reported on the spectrum for ^{239}Pu-Be neutron source, but published spectra differ one from another. Also in some works[3-7] the data on the spectrum are given for neutrons higher than certain energies. In this case, the data below these energies were taken from curves which were smoothly drawn according to the experimental values of Stewart.[2] On spectra for ^{241}Am-Li and ^{241}Am-F neutron saurces the available experimental data are very scanty. For ^{241}Am-Li source the only one obtained by Bennett[8] was available. No published papers for ^{241}Am-F source were found in the course of literature survey. So for this neutron source the use substituted the spectrum of ^{210}Po-F neutrons, assuming that a ^{241}Am-F neutron source may have the same spectral form as that from a ^{210}Po-F neutron source. The spectrum for ^{241}Am-F source should not differ greatly, in fact, from that for ^{210}Po-F source since the alpha energies of ^{241}Am and ^{210}Po differ by only about 0.18 MeV.[11] The errors given for N(E), ^{239}Pu-Be neutron spectra, in the literature[2-7] are about ±10% for all the cases while the errors are not included for ^{241}Am-Li and ^{241}Am-F neutron spectra.[8, 9]

In order to determine the average energy $\bar{E}$

the continuous N(E) curve is thought to consist of a histogram with equal intervals of say 0.5 MeV($\triangle$E) for ^{239}Pu-Be neutron spectrum and 0.05 MeV($\triangle E$) for ^{241}Am-Li and ^{241}Am-F neutron spectra. Eq. (1) can then be written as a summation:

$$\bar{E} = \frac{\sum_{i=1}^{n} N(E_i)E_i\triangle E}{\sum_{i=1}^{n} N(E_i)\triangle E} \quad \cdots\cdots(2)$$

The energy axes of the plotted spectra were divided into a energy intervals with ($n \geq 22$). The individual integrals were obtained as areas by a simple histogram method.

2—2. Effective Energies

The effective energy E_k defined by Nachtigall[12] is expressed by the form

$$E_k = \frac{\int_{E_{min}}^{E_{max}} N(E)d_e(E)EdE}{\int_{E_{min}}^{E_{max}} N(E)d_e(E)dE} \quad \cdots\cdots(3)$$

where d_e(E) is dose equivalent as a function of neutron energies. Uncertainties in the differential dose equivalent assigned by authors[17] are about 10%. As implied in Eq. (3), the spectrum can be treated as if it is a monoenergetic spectrum with the effective energy E_K. This may find its application when such quantities as dose or dose equivalent are calculated for continuous neutron spectrum.

Differential dose equivalent d_e(E) up to 10 MeV of neutron energies was mainly extracted from the data which are graphically given by Snyder and Neufeld,[10] with the exception of the autogamma dose from the ^{1}H(n, γ)^{2}D reactions in the human body. For the autogamma dose the use was made from the data of Smith and Boot,[13] and their values are low by as much as 25% of those of Snyder and Neufeld. For neutrons with energies ranging from 10 to 11 MeV around which there may exist the upper limit of the ^{239}Pu-Be neutron spectrum, no appropriate data on the differential dose equivalent are available yet. So in this region the data on it were obtained by extrapolation of the remaining data of Snyder and Neudfeld.

In the same way as before, Eq. (3) is approximated by a summation:

$$E_k = \frac{\sum_{i=1}^{n} N(E_i)d_e(E_i)E_i\triangle E}{\sum_{i=1}^{n} N(E_i)d_e(E_i)\triangle E} \quad \cdots\cdots(4)$$

In this equation the factor $d_e(E_i)$ is considered as weighting factor and now taken equal to corresponding factor for the neutron spectrum of interest. The integral was obtained by the same method as in the above. They were computed with some conventions on the neutron dose equivalent, that is, first- and multi-collision dose equivalents. The term "multi-collision dose equivalent" is to be used with the same meaning as "surface dose equivalent" in this report. In view of the radiological protective measures, it seems more reliable to adopt the concept of multicollision dose equivalent than that of the first collision dose equivalent. Still the later concept, however, is accepted in many countries. Therefore, data obtained on the concept of the first collision dose equivalent are included in this paper.

The surface dose equivalent in place of the maximum dose equivalent commonly adopted was used in this study. Unless otherwise specified, the term "surface dose equivalent" refers to "total surface dose equivalent" as will be described later. It is, of course, very advisable to use the concept of the maximum dose equivalent. Some practical difficulties, however, are often encountered in interpreting the experimental data obtained by a personnel dosimeter by which tissue dose is usually measured at the surface of the human body. It is hardly possible to get the maximum dose equivalent if the measurements are made in the radiation field where incident radiation is not free from the scattered radiation flux in

the human body. In fact, there is no significant difference between values obtained on these two concepts in the energy region of interest as can seen in the reference.[10] Furthermore, there is an advantage[14] that the dose equivalent obtained on the former concept is consistent with the routine interpretation of the gamma-ray film dosimetry. Also it has a merit that the dose equivalent from the external gamma-ray and from the $^1H(n, \gamma)^2D$ process giving a 2.2 MeV gamma-ray in the human body can be readily obtained by the usual film dosimeter.

For some purposes it is desirable to separate the effective energies for the charged particles production reactions in the human body from those for the $^1H(n, \gamma)^2D$ reaction from which dose equivalent can be easily measured by the usual film dosimetry as aforementioned. For all the nuclear processes (charged particles plus autogamma production reactions) they are, however, nearly the same as those for the former process because the contribution by the later is negligible compared to that by the former process. In the case of first collision concept, the contribution by the $^1H(n, \gamma)^2D$ reaction to the total dose equivalent is less than by several orders of magnitude. The effective energies for the former process only were thus computed.

2—3. Average Fluence-Dose Equivalent Conversion Factors

The average fluence-dose equivalent conversion factor $\bar{d}_e$ may be written by

$$\bar{d}_e = \frac{\int_{E_{min}}^{E_{max}} N(E) d_e(E) dE}{\int_{E_{min}}^{E_{max}} N(E) dE} \quad \cdots\cdots(5)$$

where all the symbols have the same meaning as given elsewhere in this report.

The $\bar{d}_e$ was obtained in the same way as in the above. Fluence-First collision charged partcle, -multi-collision charged particle and-multi-collision autogamma dose equivalent conversion factors were computed.

3. Results and Discussion

The results calculated, for average and effective energies, and fluence-dose equivalent conversion factors, are summarized in Table 1. In this table the symbols $\bar{E}$ and K_t refer to average and effective energies, respectively. The subscripts f and m mean first- and multi-collision processes of neutrons in the human body while c, g and t imply charged particles, autogamma production and total nuclear reactions (charged particles plus autogamma production reactions). $\bar{d}_e$ means average fluence-dose equivalent conversion factor, and the suffices in the symbol have the same meaning as described above.

As shown in Table 1, the concept of the first collision process gives dose equivalents 20~40% less than that of the multi-collision process in the human body in the energy region of interest.

For ^{239}Pu-Be neutron source, some discrepancies between the values calculated from one to another literature quoted on the spectrum are noted. It is not possible to decide which data should be considered from the divergent values. One may therefore use the averaged values as a compromise. In the table the averaged values are listed and the standard deviations in these values are given as the square root of the average value of the square of the individual deviations from the mean value. For comparison, some data obtained by Nachtigall[12] are given in parenthesis among which 3.52×10^{-8} rems/(n/cm^2) in the second column from the righthand side of the table was obtained as a fluence-maximum dose equivalent conversion factor by him. He calculated them in the same way as in this study. It is clear that there is actually no

Table 1. Average and Effective Energies, and Fluence-Dose Equivalent Conversion Factors for ^{239}Pu-Be, ^{241}Am-Li and ^{241}Am-F Neutron Soures

Type of source	Ref. quoted in spectrum	$\bar{E}$ (MeV)	First collision		Multi-collision					
			$\bar{d}_{fce}$ rems/(n/cm^2)	E_{kfc} (MeV)	$\bar{d}_{mce}$ rems/(n/cm^2)	E_{kmc} (MeV)	$\bar{d}_{mge}$ rems/(n/cm^2)	E_{kmg} (MeV)	$\bar{d}_{mte}$ rems/(n/cm^2)	K_{kmt} (MeV)
^{239}Pu-Be	2	4.44	2.82×10^{-8}	4.79	3.67×10^{-8}	4.87	2.34×10^{-10}	4.17	3.69×10^{-8}	4.87
〃	3	3.98	2.71	4.45	3.51	4.47	2.37	3.72	3.53	4.47
〃	4	4.20	2.78	4.61	3.56	4.64	2.35	3.92	3.58	4.64
〃	5	3.85	2.69	4.30	3.46	4.34	2.37	3.57	3.48	4.34
〃	6	4.42	2.81	4.78	3.58	4.84	2.33	4.14	3.60	4.84
〃	7	3.51	2.62	3.79	3.39	3.81	2.41	3.26	3.41	3.81
	Average	4.07 ±0.33 (4.1)	2.74 ±0.07	4.45 ±0.344	3.53 ±0.09	4.50 ±0.36	2.36 ±0.03	3.80 ±0.32	3.55 ±0.09 (3.52)	4.50 ±0.36
^{241}Am-Li	8	0.42	1.58	0.51	2.16	0.50	2.64	0.41	2.19	0.50
^{241}Am-F	9	1.41	2.34	1.47	2.79	1.45	2.50	1.40	2.82	1.45

difference between the values based on the different concepts, namely, maximum and surface dose equivalents. Also the values calculated here are in good agreement with those obtained by Nachtigall.

As can be seen in the table, these neutron sources are likely to be very useful in calibrating neutron detectors and dosimeters because they cover the most interest region with average energies 0.42—4.1 MeV usually encountered in the radiation safety control like around reactors.

Unfortunately ^{239}Pu-Be neutron source is inadequate for the use as a constant flux source because neutron emission rate increases with time in the useful lifetime of the source [15, 16] It is known that the presence of ^{241}Pu in Pu at the time of source fabrication gives chiefly rise to the increase in the neutron yield. ^{241}Pu decays to ^{241}Am, which produces neutrons by the ^{241}Am-Be(α, n) reaction. According to the work of Jordan et al. [15] the increasing rates are of the order of 0.11 to about 7% a year but vary depending upon the isotopic composition of the plutonium source material used. It may thus be difficult to estimate the growth rate if an information on the exact composition of the source is not known. In conclusion, it is not very recommendable to use plutonium-beryllium neutron source as a standard in the precise work where high accuracy is required.

Acknowledgement

The authors are indebted to Dr. Chae-shik Rho for his criticism throughout this work.

References

1) S. Block, J. Bryan, C. Prevo and D. Mortan, Health Phys. **13**, 1025(1967).
2) L. Stewart, Phys. Rev. **98**, 740(1955).
3) M. E. Anderson, MLM-1422, 1967.

4) R. Huber, Z. Lewandewski, R. Plattner, C. Poppelbaum and R. Wagner, Helv. Phys. Acta **35**, 557 (1962).
5) H. Gredel and M. Säbel, Neutron Monitoring, IAEA, Vienna, 167(1967).
6) A. Rubbino, D. Zubke and C. Meixner, Il Nuovo Cimento **10**(44B), 178(1966).
7) E. Tochilin, Neutron Dosimetry, Vol. 1, IAEA, Vienna, 553(1963).
8) E. F. Bennett, ANL-7110, 337(1965).
9) J. De Pangher, Nucl. Instrum. Meth. **5**, 61 (1959).
10) U. S. National Bureau of Standards, Handbook 63, Protection against Neutron Radiation up to 30 Million Electron Volts, 1957.
11) D. Strominger, J. M. Hollander and G. T. Seaborg, Table of Isotopes, Rev. Mod. Phys. **30**, No. 2, Part 2, 585(1958).
12) D. Nachtigall, Health Phys. **13**, 213 (1967).
13) J. M. Smith and S. J. Boot, AERE-R3696, 1961.
14) J. A. Dennis, AERE-R4365, 1964.
15) E. D. Jordan, W. H. Bedenbaugh and T. E. Carew, Nucl. Sci. Eng. **30**, 308 (1967).
16) R. S. Caswell, R. H. Noyce, S. B. Garfinkel and E. R. Mosburg Jr., Neutron Dosimetry, Vol. 2, IAEA, Vienna, 547 (1963).
17) W. S. Snyder and J. Neufeld, ORNL-1872, 1955.