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Neutron calibration field at Institute for Nuclear Science and Technology

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Abstract: In this study, a neutron calibration field of a ^{241}Am-Be standard source at the Institute for Nuclear Science and Technology is characterized in terms of neutron spectral fluxes and neutron ambient dose equivalent (nDE) rates based on ISO 8529 recommended generalized fit method together with measurements using a Bonner Sphere Spectrometer and MAXED unfolding techniques. The total, direct and scattered neutron components as well as nDE rates and neutron spectral fluxes are separated from each other. The direct nDE rates and neutron spectral fluxes in the free field as functions of distances from the source are also theoretically calculated based on the neutron source strength. The comparison between experimental data and theoretical calculations of the nDE rates and neutron spectral fluxes shows good agreement within 3%, this has confirmed the reliability of the field characterization process and its applicability in the practical calibration works for neutron survey meters.

Keywords: *neutron calibration field, neutron spectral fluxes, nDE rates, generalized-fit method*

I. INTRODUCTION

The Institute for Nuclear Science and Technology (INST), a branch of the Vietnam Atomic Energy Institute, possesses a unique secondary standard dosimetry laboratory in Vietnam (SSDL-VN) on ionizing radiation dosimetry and calibration [1]. However, SSDL-VN has not yet been able to perform calibrations for neutron measuring devices. Recently, INST is equipped with a neutron calibration facility using a ^{241}Am-Be neutron source. To put the calibration facility into operation, the characterization of the neutron calibration field is crucially needed to ensure the reliability of radiation measurements and safety assessment.

In principle, a neutron measuring device is calibrated by being placed in a neutron reference field, in which the calibrated quantity is well known, to determine its calibration factor which is a unique property of the device and the neutron source, and should not be a function of the characteristics of the calibration facility [2].

In fact, the total neutron field measured by a neutron measuring device consists of two components: a direct component of neutrons coming to the devices directly from the source without any interaction, and a scattered component coming to the devices after interactions with surrounding objects in the calibration room, such as the air and concrete walls [1]. In the present work, the characterization of the neutron calibration field is characterized in terms of neutron spectral fluxes and nDE rates using BSS measurements together with MAXED unfolding code. The

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direct and scattered components of the neutron field are also separated from the total one using the ISO 8529 recommended generalized fit method (GFM) [2]. Alternatively, the direct neutron fluxes and appropriate direct nDE rates in the free field (FF) as a function of distances from the source are theoretically calculated based on the neutron source strength. Comparison between results (direct nDE rates and direct neutron spectral fluxes) obtained from the experiments and in FF is made to confirm the reliability of the characterization process.

II. MATERIALS AND METHOD

A. Neutron calibration source and instruments

A ^{241}Am-Be neutron calibration source of X14 type capsulation supplied by Hopewell Designs, Inc., USA with the neutron source strength of 1.299 × 10^7 s^{-1} at Jan.-23, 2015, which is traceable to the NIST, USA, was installed in a container at the central floor of the neutron room which has the inner dimensions of 700 cm × 700 cm × 700 cm with the concrete wall thickness as shown in Fig. 1. When performing the calibration, the neutron source is pumped up to the center of the calibration room by a pneumatic system through a cylindrical aluminum pipe which has the wall thickness of 0.5 cm.

The BSS system is widely used for neutron spectrometry relating to radiation protection assessment. The system consists of a thermal neutron sensitive detector of ^{6}LiI(Eu) accompanying with a set of six 0.95 g/cm^3 polyethylene spheres with the diameters of 2, 3, 5, 8, 10, and 12 inches (see Fig. 2). The cylindrical ^{6}LiI(Eu) detector with 96% of ^{6}Li has the dimensions of 0.4 cm in height and the same size in diameter. The system configuration allows getting spectral information of neutrons from thermal energy to 20 MeV.

The thermal neutrons are detected based on the reaction ^{6}Li(n,α)^{3}H (Q value = 4.78 MeV) then the Ludlum 2200 scaler counts the electronic pulses from a photomultiplier. The BSS system is basically not sensitive to photons.

In this work, the BSS was set up on a half diagonal of the room central plane which is parallel to the floor and the ceiling traversing through the source center (see Fig. 1.a). The BSS measurements are done using each sphere every 10 cm in the range of 60 cm to 250 cm from the source.

B. Generalized-fit method

The reading of a device, $M'_T(l)$, due to the total radiation field (direct neutrons plus scattered neutrons) can be expressed as Eq. (1) [2].

$$M'_T(l) = \frac{k}{l^2} \times F_L \times \left\{ \frac{F_1(l)}{F_A(l)} + F'_2(l) - 1 \right\} \quad (1)$$

where, l is the distance between the centre of the source and the reference point; k is the characteristic constant; $F_1(l)$ is the geometry factor; $F_A(l)$ is the air attenuation (air out-scatter) correction; $F'_2(l)$ is the correction function which describes the additional contribution from in scattered neutrons; and F_L is the linearity correction factor [2].

The linearity correction, F_L, is assumed as 1 in this work, so that $M'_T(l)$ in Eq. (1) can be replaced with $M_T(l)$ in Eq. (2).

$$M_T(l) = \frac{M'_T(l)}{F_L} \quad (2)$$

When performing measurements at the distance, l, from the source center to the detector center greater than detector radius, r_D, two times (i.e. $l/r_D \geq 2$), the geometry correction, $F_1(l)$, can be considered as Eq. (3):

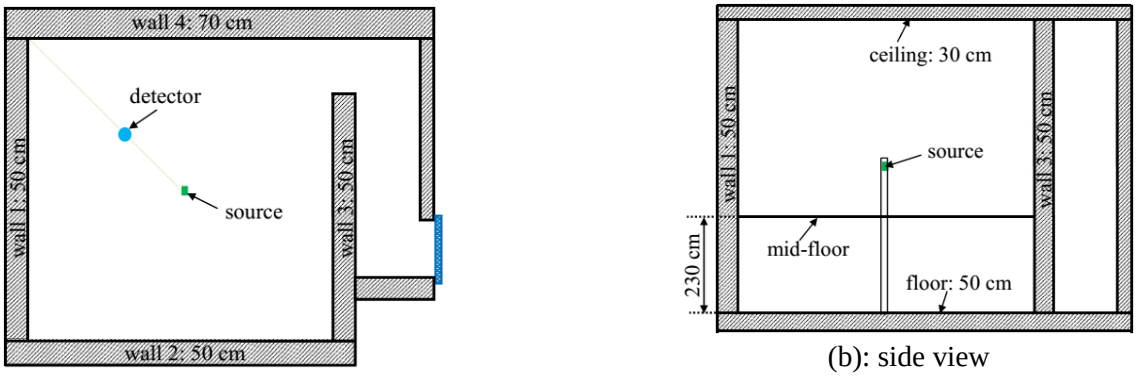
$$F_1(l) = 1 + \delta \times \left(\frac{r_D}{2 \times l} \right)^2 \quad (3)$$

where δ is the neutron effectiveness parameter whose value is recommended as 0.5 ± 0.1 for all cases [2].

The air attenuation correction, $F_A(l)$, is given by Eq. (4):

$$F_A(l) = e^{\overline{\Sigma}(E)\times l} \approx 1 + \overline{\Sigma}(E) \times l \quad (4)$$

where $\overline{\Sigma}(E)$ is the linear attenuation coefficient obtained by averaging the total neutron cross-sections for nitrogen and oxygen over the spectral neutron distribution of the source, which has a value of 890×10^{-7} cm^{-1} for ^{241}Am-Be neutron source [2].



(a): top view (b): side view

Fig. 1. Neutron calibration room

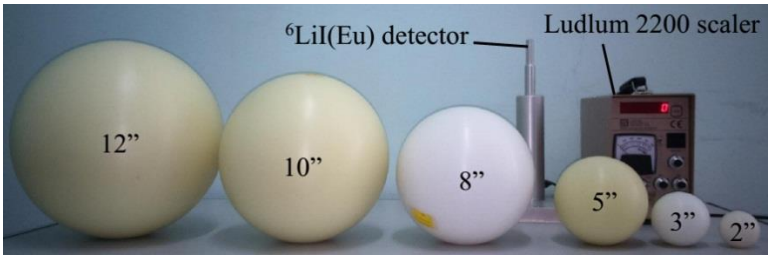


Fig. 2. BSS system with spheres 2”, 3”, 5”, 8”, 10”, 12”, ^{6}LiI(Eu) detector and Ludlum 2200 scaler

In GFM, all equations from (1) to (4) are used together with the assumption that the correction factor for inscattered neutrons, $F_2'(l)$, can be represented as Eq. (5) [2]:

$$F_2'(l) = 1 + A' \times l + s \times l^2 \quad (5)$$

where A', s are free parameters.

In conclusion, the reading of a device due to the total radiation field, $M_T(l)$, can be fitted as a function of distances from the source center to the detector center, l, and detector radius, r_D, based on Eq. (6):

$$M_T(l) = \left(\frac{k}{l^2}\right)\left[\frac{1+\delta\times\left(\frac{r_D}{2\times l}\right)^2}{1+\overline{\Sigma}(E)\times l} + A' \times l + s \times l^2\right] \quad (6)$$

where $\delta = 0.5$, $\overline{\Sigma} = 890 \times 10^{-7}$ cm^{-1} and k, A', s are free parameters obtained by fitting. A' takes into account the air in-scatter component and s is the factor accounting for contributions of any other in-scattered neutrons.

C. Unfolding code

In this work, MAXED code was used to unfold the neutron flux spectra. The code is based on maximum entropy principle in the inverse problem of spectrum unfolding [3,4]. In general, the reading, C_i, of a detector with number i of a set n detectors is an integral of the energy response function, $R_{ib}(E)$, of the detector i in the energy range with b bins and the energy spectrum, $\Phi_b(E)$. The relationship can be written as Eq. (7):

$$C_i = \int_{b=1}^{\infty} R_{ib}(E) \times \Phi_b(E) \; (i = 1 \div n) \quad (7)$$

In most of cases, the number of energy bins, b, is much greater than the number of detectors, n (i.e. Eq. (7) has infinite solutions).

The inverse problem of unfolding process is choosing a particular solution spectrum, $\Phi_b(E)$, from the infinite number ones. Therefore, we need a priori information of solution spectrum out of other boundary conditions (i.e.: the reading of the detector i, C_i; and the energy response function, $R_{ib}(E)$ of each detector). If our priori guess spectrum is good we expect our final solution to be close to the true spectrum. In this work, the authors paid much care for preparing the MCNP5 input files in order to estimate the best priori guess spectrum. In addition, the authors took the energy response functions, $R_{ib}(E)$, from the IAEA compendium [5].

III. RESULTS AND DISCUSSION

A. Fitting result

The total count rates due to the total neutron field were measured by BSS using each sphere every 10 cm in the range from 60 cm to 250 cm apart from the source. The measured data are fitted by a non-linear function as Eq. (6) using the instrumental weighting method in the Levenberg Marquardt iteration algorithm [6]. Fig. 3 shows the relationship of the count rates due to the total neutron field as functions of distances from the source, l; and different size BSS spheres.

The goodness of fitting is confirmed by the adjusted R square when it is close to 1. In all fitting cases, these values are as close as 1 which show the reliability of fitting. By this fitting method, ones can separate the direct neutron component from the total one.

In all fitting curves, the deviation of the characteristic constants show the uncertainty less than 5%. The uncertainties of characteristic constants in the fitting curves must be taken into account when applying them in the spectrum unfolding process of direct neutron component.

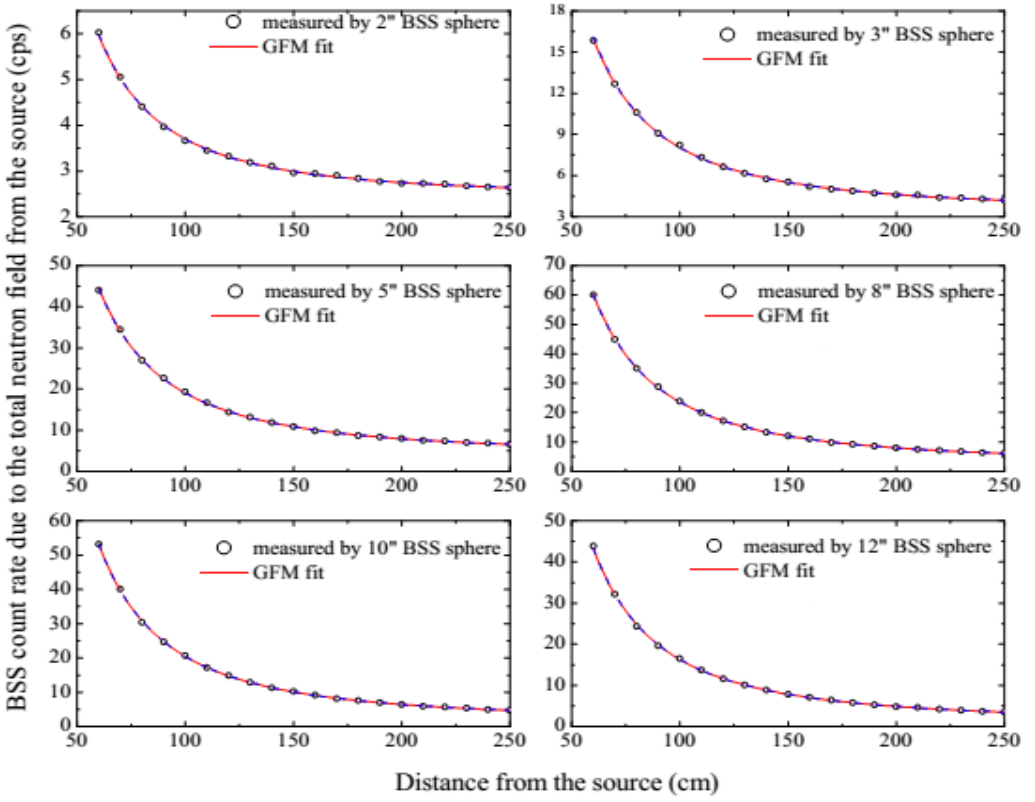


Fig. 3. Count rates due to total neutron field as functions of distances from the source

B. Unfolding of total and direct neutron flux spectra by MAXED code

The count rates due to the total neutron field measured by six BSS spheres at the same distance were compiled in MAXED input file to determine the total neutron flux spectrum at that distance. The neutron spectral flux uncertainty in an individual energy bin between unfolding results and measuring data (i.e. input data) is not concerned in MAXED code but the integrated uncertainty of the output solution is concerned in the code. The input data of neutron spectral fluxes were only chosen to put into the unfolding input file when those gave the neutron spectral flux outputs consistent with the input data within an integral uncertainty of 3% (k=1). The neutron spectral fluxes obtained from BSS measurements together with fitting and unfolding techniques as a function of distances from the source, l, are listed in Table I.

Table I. Neutron fluxes of different components as functions of distances from the source.

l (cm)	Neutron spectral flux of different components (cm^{-2}/ unit lethargy)			
	Direct (FF)[a]	Total (UF)[b]	Direct (GFM-UF)[c]	Scattered (GFM-UF)[d]
60	296	328	290	38
70	217	249	213	35
80	166	193	163	30
90	132	160	129	31
100	107	137	105	32
110	88	117	86	30
120	74	101	73	29
130	63	91	62	29
140	54	81	53	28
150	47	75	46	28
160	42	69	41	28
170	37	63	36	27
180	33	59	32	27
190	30	56	29	27
200	27	53	26	26
210	24	50	24	27
220	22	48	22	27
230	20	47	20	27
240	18	45	18	27
250	17	43	17	26

[a]: Free field
[b]: Unfolding by MAXED code
[c]:Unfolding by MAXED code with the input data obtained from the Generalized-Fit Method
[d]: Total (UF) - Direct (GFM-UF)

As the results summarized in the Table I, the count rates due to the direct neutron component of the field can be deduced from the characteristic constants of fitting curves then those six direct count rates of all BSS spheres at the same distance are compiled in MAXED input file to determine the direct neutron flux spectrum at that distance. The scattered neutron spectral flux is determined by subtracting the direct component from the total one. The direct neutron spectral fluxes in the free field are also calculated based on the neutron source strength.

Ones can see that the direct neutron spectral fluxes obtained from experiment (GFM-UF) and in FF are consistent to each other within 3%. This implies that the fitting methods used in this work can be applied for the practical calibration process.

C. The source anisotropy correction factor

The anisotropy correction factor was investigated at 100 cm far from the neutron source (this is the common distance that the neutron measuring devices being calibrated) [7]. Therefore, the 100 cm radius sphere with the source at the center was segmented with a solid angle of $\pi/45$ which has the symmetric axis perpendicular to the source axis at its center. The MCNP5 simulations with the “F2” tally accompanying with FS2 cards were performed to calculate the average neutron flux (i.e. angular source strength) crossing the solid

angle surface in two cases: 1) the real source at the sphere center, and 2) the point source at the sphere center. The angular source strength in the first case was then normalized to that in the latter case to deduce the source anisotropy factor. The anisotropy correction factor at the angle of π/2 was obtained as 1.030 at 100 cm apart from the ^{241}AmBe source in X14 capsulation type.

D. Calculation of Neutron Ambient Dose Equivalent Rate

The total and direct nDE rates (denoted as nDE_{tot} and nDE_{dir}, respectively) can be alternatively calculated by Eq. (8) and Eq. (9). nDE_{FF} in the free field can be calculated by Eq. (10):

$$nDE_{tot} = \sum_{b=1}^{n} \Phi_{tot-b} \times h_{\Phi-b} \quad (8)$$

$$nDE_{dir} = B_{UF} \times h_{\Phi} \quad (9)$$

$$nDE_{FF} = \frac{B \times F_1(\theta)}{4\pi \times l^2} \times h_{\Phi} \quad (10)$$

where, Φ_{tot-b} is the neutron flux obtained from MAXED unfolding of total neutron spectra in the energy bin b caused by the neutron source strength B (cm^{-2}); $h_{\Phi-b}$ is the conversion factor from unit neutron flux to neutron ambient dose equivalent in the energy bin b ($pSv.cm^2$) [8]; B_{UF} is the direct neutron spectral flux obtained from unfolding of direct neutron spectra (cm^{-2}); h_{Φ} is the conversion factor from unit neutron flux to neutron ambient dose equivalent which is 391 $pSv.cm^2$ for ^{241}Am-Be source [9]; l is the distance between the centre of the source and the reference point (cm). Table 2 shows the values of nDE rates of different components as functions of distances from the source. The results of direct nDE rates obtained by Eq. (9) and Eq. (10) are in good agreement within 3% implies that the reliable characterization process can be applied in the practical calibration for neutron survey meters.

Table II. Neutron ambient dose equivalent (nDE) rates of different components and direct nDE rates in the free field as functions of distances from the source.

l (cm)	Neutron ambient dose equivalent rate of different components (μSv/h)			
	Total	Scattered	Direct	
	UF	GFM-UF	GFM-UF	FF
60	415.1	6.5	408.7	416.5
70	308.9	8.7	300.3	306.0
80	235.2	5.3	229.9	234.3
90	191.1	9.5	181.7	185.1
100	159.7	12.6	147.1	149.9
110	133.0	11.4	121.6	123.9
120	113.4	11.2	102.2	104.1
130	98.8	11.7	87.1	88.7
140	86.7	11.7	75.1	76.5
150	77.6	12.2	65.4	66.6
160	70.1	12.6	57.5	58.6
170	62.7	11.8	50.9	51.9
180	57.7	12.3	45.4	46.3
190	52.9	12.2	40.8	41.5
200	48.8	12.0	36.8	37.5
210	45.5	12.1	33.4	34.0
220	42.9	12.5	30.4	31.0
230	40.3	12.5	27.8	28.3
240	37.8	12.2	25.5	26.0
250	36.1	12.6	23.5	24.0
Notations: Refer to Table I				

IV. CONCLUSIONS

In this work, the neutron spectral fluxes of the total, direct and scattered components have been characterized using ISO recommended generalized fit method together with the BSS measurements and MAXED unfolding code. The neutron ambient dose equivalent rates of the total, direct and scattered components have also been determined.

The direct neutron ambient dose equivalent rates and neutron spectral fluxes in the free field have also theoretically calculated

which are consistent with those from the BSS experiments within 3%. It is obvious that the reliable characterization process in this work can be applied in the practical calibration works for neutron survey meters.

V. ACKNOWLEDGEMENTS

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