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Characterization of an ^{241}Am-Be neutron irradiation facility at Institute for Nuclear Science and Technology

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Abstract: An automated panoramic irradiator with a ^{241}Am-Be neutron source of 5 Ci is installed in a bunker-type medium room at the Institute for Nuclear Science and Technology (INST) for calibration of neutron devices. Bonner Sphere Spectrometer (BSS) formed by 6 spheres plus bare detector, with cylindrical, almost point like, ^{6}LiI(Eu) scintillator and 2 different spectral unfolding FRUIT and BUNKIUT codes are used to characterize the neutron field in different measurement points along the irradiation bench. The neutron field is also simulated by MCNP5 software and compared with measurements performed by the BSS. The paper shows the main results obtained in terms of neutron spectra at fixed distances from the source as well as their neutron fluence rate (total and direct) and ambient dose equivalent rate. These values measured by the BSS with two unfolding FRUIT and BUNKIUT codes are in good agreement with that of simulated by MCNP5 within 10%.

Keywords: *Bonner Sphere Spectrometer; unfolding code, neutron fluence rate, neutron ambient dose rate.*

I. INTRODUCTION

In order to calibrate the neutron device, the Institute for Nuclear Science and Technology (INST) has established a secondary standard laboratory for neutron dosimetry. An automated panoramic irradiator with a ^{241}Am-Be neutron source of 185 GBq (5 Ci) is installed in a bunker-type medium room (7 m long, 7 m width and 7 m high) at the Secondary Standard Dosimetry Laboratory (SSDL) of the INST. The calibration room layout is shown in Fig.1. It was prepared to install a metrology bench, which is placed on the mid-floor and can be easily moved in the range of 0.5 m to 3.8 m from the source. When carrying out the calibration, the ^{241}Am-Be neutron source is pumped up to the center of the calibration room by a pneumatic source transfer system. The ^{241}Am-Be calibration source of X14 type capsulation was calibrated by the NIST, USA on January 23, 2015. Its strength is 1.299×10^{7} s^{-1} with the expanded uncertainty of ± 2.9% (2σ). Ideally, this source should be free in air to comply with ISO-8529 [1] recommendations, requiring a well-known spectrum, fluence rate and the device response or calibration factor should be independent of calibration facility. So it is essential to carefully characterize the neutron fields in different measurement points along the irradiation bench to put the calibration facility into operation.

The neutron fluence, Φ is the recommended physical quantity used for investigating and establishing the reference

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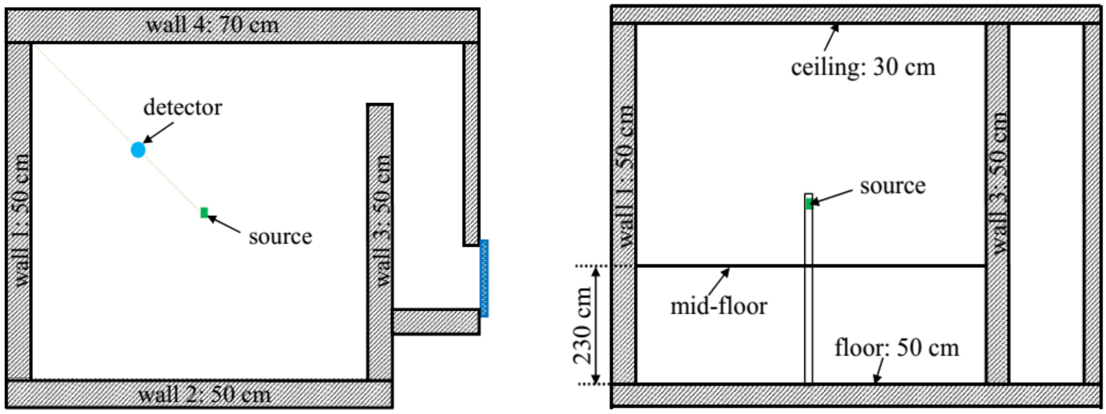
neutron field. Ambient dose equivalent, H^* $H^*(10)$ is an operational quantity used for calibrating the environmental dose meters. Personal dose equivalent, $H_p(d)$ is an operational quantity for calibrating the personal dosimeter on the phantom. The relationship between the physical quantity and the operational quantities used in calibration of neutron dose is shown in Fig. 2.



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

Fig. 1. Layout of neutron calibration room

Reference neutron field

↓

Physical quantity used to characterize the Reference neutron field

Neutron fluence, $\Phi(E,\Omega)$

Absorbed dose, D

↓

Operational quantities used for calibration (derived from physical quantity)

Ambient dose equivalent, $H^*(d)$

Personal dose equivalent, $H_p(d)$, (on phantom)

Fig. 2. Relationship between the physical quantity and the operational quantities used in calibration of neutron dose

In this work, the neutron calibration field is characterized in terms of neutron spectral fluences, ambient dose equivalent rates and personal dose equivalent ones at the different distances from the ^{241}Am-Be neutron source.

II. MATERIALS AND METHOD

The fluence spectra and dose equivalent rates at different positions in the ^{241}Am-Be reference field are determined by both methods: Monte-Carlo simulation and experimental measurements using a BSS.

A. Simulation of the fluence spectra and dose rate from the ^{241}Am-Be source using MCNP

MCNP5 (Monte Carlo N-Particle, Version 5) simulation software is used to simulate the neutron fluence spectra and calculate the neutron dose rate [2].

The geometry of the neutron calibration room is described in detail in the MCNP5 program's input file with the following objects:

+ ^{241}Am-Be standard source with X14 capsule;

+ Aluminum tube for the movement of the source to the central position of room;

+ Aluminum mid-floor;

+ Concrete walls;

+ The radiation shielding door.

The geometry of the calibration room is illustrated in Fig.1.

The materials in the simulation are taken from reliable international references on compound composition, mass ratio, and material density [2-6]. The basic materials used in the simulation include: concrete walls, aluminum, iron, polyethylene, stainless steel, air, etc.

Information on standard source is derived from current international recommendations [5,6]. The material distribution of the source is assumed to be homogeneous.

At each position, the total neutron field consists of two components: a direct component of neutrons directly reaching this position without any interaction, and a scattered component of neutron reaching this position after interactions with air, the walls, floor and ceiling of the calibration room. The total neutron fluence rate and direct neutron fluence one are recorded at positions of 50 cm to 255 cm from the source. Energy bins are divided into appropriate intervals according to the recommendations of ICRP 74 [7] to facilitate subsequent calculations.

B. Measurement of the fluence spectra and dose rate from ^{241}Am-Be source using BSS.

A Ludlum BSS with 5 spheres (2”, 3”, 5”, 8”, 10”and 12” diameters) and the bare detector (4 mm × 4 mm ^{6}LiI(Eu) scintillator) (Fig.3) are used to measure the neutron fluence rates. The polyethylene spheres have a density of 0.96 ± 0.01 g.cm^{-3}. 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). The BSS measurements are done using each sphere every 10 cm in the range of 60 cm to 250 cm from the source.

The generalized-fit method [1] is used to estimate the components of scattering neutrons and direct ones from the radiation source.

Fig. 3. Ludlum BSS

Two unfolding methods are used. The first unfolding method utilized is an iterative procedure with the SPUNIT algorithm [8,9] of the NSDUAZ code [10] with the response matrix UTA-4, with 32 energy bins. NSDUAZ (Neutron Spectrometry and Dosimetry from The Universidad Autónoma de Zacatecas) is a user friendly neutron unfolding package for

BSS with ^{6}LiI(Eu) developed under LabView® environment. Unfolding is carried out using a recursive iterative procedure with the SPUNIT algorithm, where the starting spectrum is obtained from a library initial guess spectrum to start the iterations, the package includes a statistical procedure based on the count rates relative to the count rate in the 8 inches-diameter sphere to select the initial spectrum. Neutron spectrum is unfolded in 32 energy groups ranging from 10^{-8} MeV up to 231.2 MeV.

The second unfolding code used by INST is FRUIT (Frascati Unfolding Interactive Tool) Ver. 4.0 in "parametric mode" [11]. It is an unfolding code that models a generic neutron spectrum as the superposition of up to four components (thermal, epithermal, fast and high energy), fully defined by up to seven positive parameters. Different physical models are available to unfold the sphere counts, covering the majority of the neutron spectra encountered in workplaces. The iterative algorithm uses Monte-Carlo method to vary the parameters and derive the final spectrum as the limit of a succession of spectra fulfilling the established convergence criteria. Uncertainties in the final results are evaluated with taking into consideration the different sources of uncertainty affecting the input data.

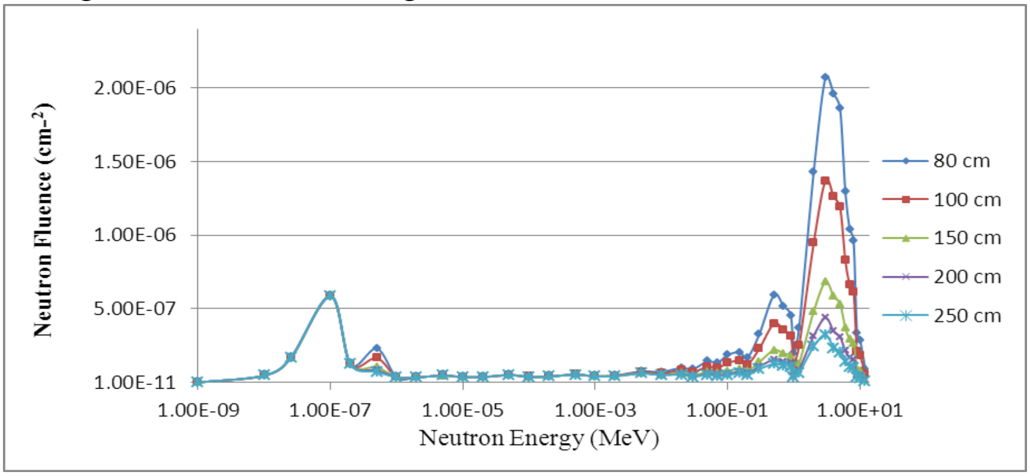


Fig. 4. Neutron fluence spectra at the different distances from the source

III. RESULTS AND DISCUSSION

A. Neutron fluence spectra simulated by MCNP

The results of neutron spectral fluences in the range of 50 cm to 255 cm from the radiation source are calculated by Tally 5 with the statistical uncertainty within 2%. Fig. 4 illustrates the neutron fluence at some distances. From Fig. 4. it is clear that the neutron fluence spectra at different distances in the air have big changes in the high energy region and little variation in the low energy region, which means that in the range of the investigated distances the scattering component is little changed. The values of total neutron fluence and direct neutron one from the source (excluding scattering neutron) at five distances calculated by MCNP5 are given in Table I.

Table I. also shows the values of neutron fluence rate coming directly from the source calculated from the strength of the source using the following formula:

$$\varphi = \frac{B.F_1.F_0}{4.\pi.l^2} \tag{1}$$

where, φ is the neutron fluence rate at distance l from the radiation source, B is the strength of the source, F_1 is the source anisotropy correction factor, F_0 is out scatter correction factor, $F_0 = e^{-\Sigma l}$, where Σ is the average linear attenuation coefficient of neutrons in the air. For ^{241}Am-Be source of type X14, F_1 = 1.04, Σ= 890.10^{-7} cm^{-1}, B = 1.299.10^{7} s^{-1}. Uncertainties of neutron fluence rate are about 3% (2σ). From Table I, it was found that all fluence rate values in columns 3 and 4 have a difference less than 0.5% so that the computation of neutron fluence simulations using MCNP5 can be confirmed as reliable and valid. These values can be considered the reference values for the calibration of neutron devices at INST's SSDL.

Table I. Total and direct neutron fluence rate at different distances

Distances from radiation source (cm)	Neutron total fluence rate (cm^{-2}s^{-1}) (by MCNP5)	Neutron direct fluence rate (cm^{-2}s^{-1}) (by MCNP5)	Neutron direct fluence rate (cm^{-2}s^{-1}) (according to formula (1))
50	495	430	429
70	270	218	218
75	241	191	190
80	216	167	167
90	176	132	132
95	161	119	118
100	148	107	107
110	128	88.0	88.1
115	121	80.8	80.5
120	113	73,9	73.9
130	101	62.9	62.9
135	95.6	58.6	58.3
140	90.9	54.2	54.2
150	83.6	47.3	47.2
155	80.0	44.3	44.2
160	77.7	41.4	41.4
170	72.1	36.6	36.7
175	70.2	34.7	34.6
180	68.1	32.6	32.7
190	64.8	29.3	29.3
195	63.8	27.9	27.8
200	61.0	26.4	26.4
210	59.0	23.9	23.9
215	58.0	22.9	22.8
220	56.2	21.8	21.8
230	53.2	19.9	19.9
235	52.5	19.1	19.1
240	51.6	18.2	18.3
250	50.3	16.8	16.8
255	50.2	16.2	16.2

B. Ambient dose equivalent rate determined by MCNP

From the spectra of neutron fluence at different distances, the ambient dose equivalent rate of $\dot{H}^*(10)$ and the personal dose equivalent rate of $\dot{H}p(10)$ at each distance are calculated according to the conversion coefficients in ICRP 74.

Neutron total ambient dose equivalent rate is calculated according to the following formula:

$$\dot{H}^*(10) = \left[\sum_{i=1}^{n}\left(\Phi_{n-p}\right)_i.\left(h_{\Phi}\right)_i\right].B \qquad (2)$$

Where, $(\Phi_{n-p})_i$ is fluence of component p in the energy bin i; $\dot{H}^*(10)$ is the ambient dose equivalent rate; $(h_\Phi)_i$ is the conversion factor.

Table II. Ambient dose equivalent rate at different distances from the radiation source

Distance (cm)	Ambient dose equivalent rate (μSv/h)	
	$\dot{H}^*(10)_{total}$	$\dot{H}^*(10)_{direct}$
50	665	603
70	348	307
75	305	269
80	270	235
90	217	185
95	197	167
100	180	151
110	151	124
115	140	114
120	129	104
130	113	88.5
135	106	82.5
140	99.3	76.2
150	87.6	66.4
155	84.5	62.4
160	79.9	58.3
170	72.7	51.6
175	70.0	48.8
180	66.7	45.9
190	61.6	41.2
195	59.7	32.2
200	57.3	37.1
210	53.6	33.6
215	52.3	32.2
220	50.3	30.6
230	47.5	28.0
235	46,6	26,9
240	45.1	25.7
250	43.0	23.7
255	42.5	22.9

from the neutron fluence to the ambient dose equivalent of the energy bin i in ICRP 74. B is the neutron source intensity. The results of the calculated total and direct (without scattering) neutron dose equivalent rates from the source at the sites of interest are summarized in Table II.

C. Neutron fluence spectra by BSS measurement

The count rates due to the total neutron field measured by six BSS spheres at each reference point on the irradiation bench (at the fixed distance from the radiation source) were compiled in FRUIT and NSDUAZ input files to determine the total neutron fluence rate at that distance. The count rates due to the direct neutron component (derived from the fitting constants of the generalized-fit method) at each distance were also compiled in FRUIT and NSDUAZ input files to determine the direct neutron fluence rate at that distance. Uncertainties presented about 3%, include all relevant causes of uncertainty: counting, overall response matrix uncertainty, source anisotropy, calibration factor and unfolding procedure. Fig. 5 and Fig. 6 illustrate the obtained neutron spectra including the total neutron spectrum at 75 cm and 150 cm, the direct neutron spectrum at 75 cm and 150 cm from the neutron source in linear scale and logarithm scale correspondingly.

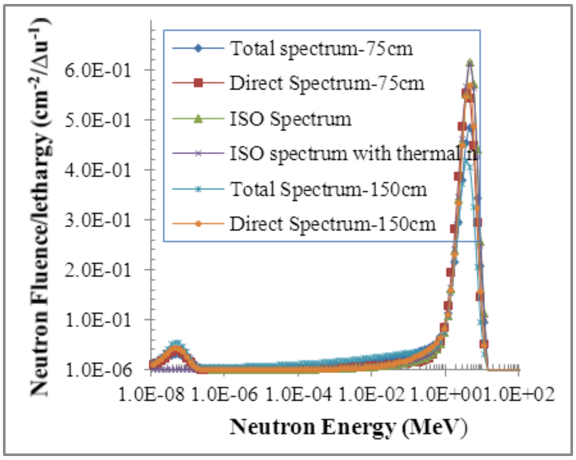


Fig. 5. Comparison of neutron spectra (in linear scale)

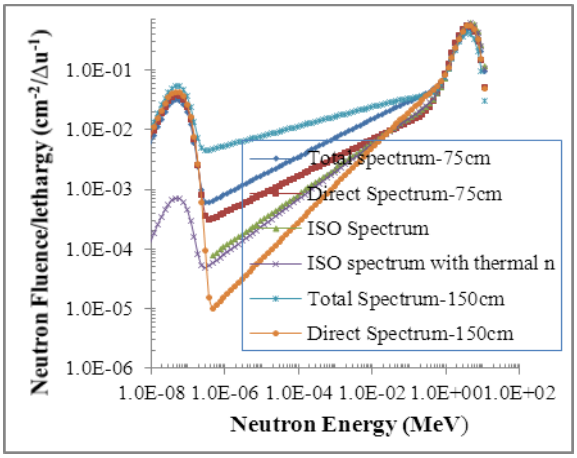


Fig. 6. Comparison of neutron spectra (in logarithm scale)

The spectra are expressed per unit lethargy. The spectra are expressed per unit lethargy Spectrum at 75 cm has a dominance of the fast region components in comparing with that of 150 cm and not so big different with the ISO 8259. So the spectrum at 75 cm from the source may be assumed as the free field for calibration of survey meters and TLDs.

D. Neutron fluence rate

The total neutron fluence rate can be calculated as the integral by the neutron energy of the neutron fluence rate from the spectral distribution of the neutron fluence rate. Table III. Summarizes the results obtained at each distance.

Table III. Neutron fluence rates obtained at four distances from the source ($cm^2.s^{-1}$)

Distance	70 cm		75 cm		100 cm		150 cm	
Unfolding Code	Direct	Total	Direct	Total	Direct	Total	Direct	Total
MCNP5	218	270	191	241	107	148	47.3	83.6
NSDUAZ	211	272	187	241	106	145	49.1	92.0
FRUIT	215	271	207	232	115	145	51.6	83.3

E. Ambient dose equivalent rate

Ambient dose equivalent can also be obtained from the spectral distribution of the neutron fluence rate, as

$$\dot{H}^*(10) = \int_E \varphi_E(E)\,h^*(10)dE \qquad (3)$$

where, $\dot{H}^*(10)$ is ambient dose equivalent rate, φ_E is fluence rate of neutron with energy E, *h*(10)* are the fluence to ambient dose equivalent conversion coefficients recommended in ICRP 74 [7].

The obtained values are indicated in Table IV. It is obvious that values of neutron fluence rates at every point measured by BSS with the help of NSDUAZ and FRUIT codes are agreed with that of calculated by MCNP5 within 5%; values of neutron ambient dose rates at every point measured by BSS with the help of NSDUAZ and FRUIT codes are agreed with that of calculated by MCNP5 within 10%.

Table IV. Ambient dose equivalent rates obtained at four distances from the source ($\mu Sv.h^{-1}$)

Distance	70 cm		75 cm		100 cm		150 cm	
Unfolding Code	Direct	Total	Direct	Total	Direct	Total	Direct	Total
MCNP5	307	348	269	305	151	180	66.4	87.6
NSDUAZ	277	312	249	267	141	166	63.2	75
FRUIT	278	326	267	280	149	167	66.4	81.8

So NSDUAZ and FRUIT codes can be used with BSS to characterize reliably the neutron field of INST's dosimetry calibration laboratory to calibrate accurately neutron dose rate meters and personal dosimeters.

IV. CONCLUSIONS

The study offered a good opportunity to compare results from two different unfolding tools as NSDUAZ, FRUIT and MCNP5.

In this work, the neutron spectral fluences of the total, direct and scattered components have been characterized using MCNP5 as well as ISO recommended generalized fit method together with the BSS measurements and two unfolding codes. Then, 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 fluence rates in the free field have also theoretically calculated which are very consistent with those simulated from MCNP5 (within 0.5%) and agreed with the BSS experiments within 10%. Those data are reliable reference values for calibration of neutron doserate meters.

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