Characterization of neutron calibration fields at the TINT's 50 Ci americium-241/beryllium neutron irradiator

T Liamsuwan[1], **J Channuie and W Ratanatongchai**
Nuclear Research and Development Division, Thailand Institute of Nuclear Technology (Public Organization), 9/9 Moo 7 Saimoon, Ongkharak, Nakorn Nayok, 26120 Thailand

E-mail: thiansinl@tint.or.th

Abstract. Reliable measurement of neutron radiation is important for monitoring and protection in workplace where neutrons are present. Although Thailand has been familiar with applications of neutron sources and neutron beams for many decades, there is no calibration facility dedicated to neutron measuring devices available in the country. Recently, Thailand Institute of Nuclear Technology (TINT) has set up a multi-purpose irradiation facility equipped with a 50 Ci americium-241/beryllium neutron irradiator. The facility is planned to be used for research, nuclear analytical techniques and, among other applications, calibration of neutron measuring devices. In this work, the neutron calibration fields were investigated in terms of neutron energy spectra and dose equivalent rates using Monte Carlo simulations, an in-house developed neutron spectrometer and commercial survey meters. The characterized neutron fields can generate neutron dose equivalent rates ranging from 156 μSv/h to 3.5 mSv/h with nearly 100% of dose contributed by neutrons of energies larger than 0.01 MeV. The gamma contamination was less than 4.2-7.5% depending on the irradiation configuration. It is possible to use the described neutron fields for calibration test and routine quality assurance of neutron dose rate meters and passive dosemeters commonly used in radiation protection dosimetry.

1. Introduction

Reliable measurement of neutron radiation is important for monitoring and protection in workplace where neutrons are present. In Thailand, neutrons are used in research, non-destructive testing, gemstone irradiation, and neutron activation analysis. Available neutron sources are radioactive sources, neutron generators and the Thai Research Reactor.

Despite a variety and increasing interest of neutron source applications, there is no standard calibration facility for neutron measuring devices available in Thailand and Southeast Asia. Therefore, neutron area monitoring devices need to be sent to Japan, Europe or the USA for annual calibration, while personal dosemeters for neutrons are calibrated once by manufacturers upon purchase and the received dose is routinely evaluated using computer software.

Recently, Thailand Institute of Nuclear Technology (TINT) has set up a multi-purpose neutron irradiation facility equipped with a 50 Ci ^{241}AmBe neutron source. Among other applications, the facility was developed for generating neutron calibration fields for radiation protection devices. Unlike recommended calibration facilities for neutron measuring devices [1, 2], the facility at TINT is not a

[1] To whom any correspondence should be addressed.

low-scatter type and the room size is not as large as most of the existing facilities. Moreover, since it is a multi-purpose facility, the neutron source is kept inside a cask and neutron irradiation is done through a collimator.

Due to scattering and absorption of neutrons by the facility's components, a simple set of correction factors for the determination of neutron dose is not achievable. Instead, Monte Carlo simulations, accompanied by measurements of neutron energy spectra and neutron dose, are required for characterization of the neutron calibration fields.

In this paper, the design of the neutron irradiation facility is described. The neutron calibration fields were characterized using Monte Carlo simulations accompanied by measurements with an in-house developed neutron spectrometer and commercial dose rate meters. The computational and experimental approaches served as validation tools for each other.

2. Material and Methods

2.1. Neutron irradiation facility

Figure 1 shows the layout of the TINT's neutron irradiation facility. The room is 6.5 m in width, 8.5 m in length, and 3.5 m in height. The neutron irradiator consists of the source storage container and the collimator enclosure, both sitting on an iron table. The collimator aperture is centered at 0.9 m above the concrete floor. To irradiate samples, a long handle is used for pulling the source from the storage container towards the collimator aperture. A warning light is activated as soon as the source is moved away from the centre of the storage container.

The storage container is a cylindrical tank with a diameter of 68.6 cm and a length of 94.8 cm. It is filled with neutron shielding materials. The 50 Ci ^{241}AmBe neutron source is housed in a cavity of 7.6 cm in diameter inside the storage container. The source has a cylindrical double encapsulation made of stainless steel. The outer dimensions are 5 cm in diameter and 8.52 cm in length. The wall thickness of each encapsulation is 0.2 cm. The neutron emission rate is 10^8 n/s (±10%) according to the manufacturer (QSA Global Inc.).

The collimator enclosure is a stainless steel box filled with a mixture of paraffin and boric acid. The dimensions of the enclosure are 69 cm in width, 90 cm in length, and 66 cm in height. The collimator duct of a rectangular pyramid shape sits inside the enclosure with the aperture of 10×10 cm^2 inwards and 15×15 cm^2 outwards. It is possible to place a neutron absorbing material inside the collimator duct to reduce the neutron flux. For that purpose, a graphite block of 14 cm thickness was constructed to fit the collimator duct.

For radiation safety reasons, the shielding wall of 150 cm height consisting of 5 cm thick borated polyethylene sheets and 40 cm thick heavy concrete was constructed behind and at one end of the irradiator. The shielding wall was designed to minimize the total dose equivalent rates to comply with the dose limit of 10 μSv/h outside the irradiation area.

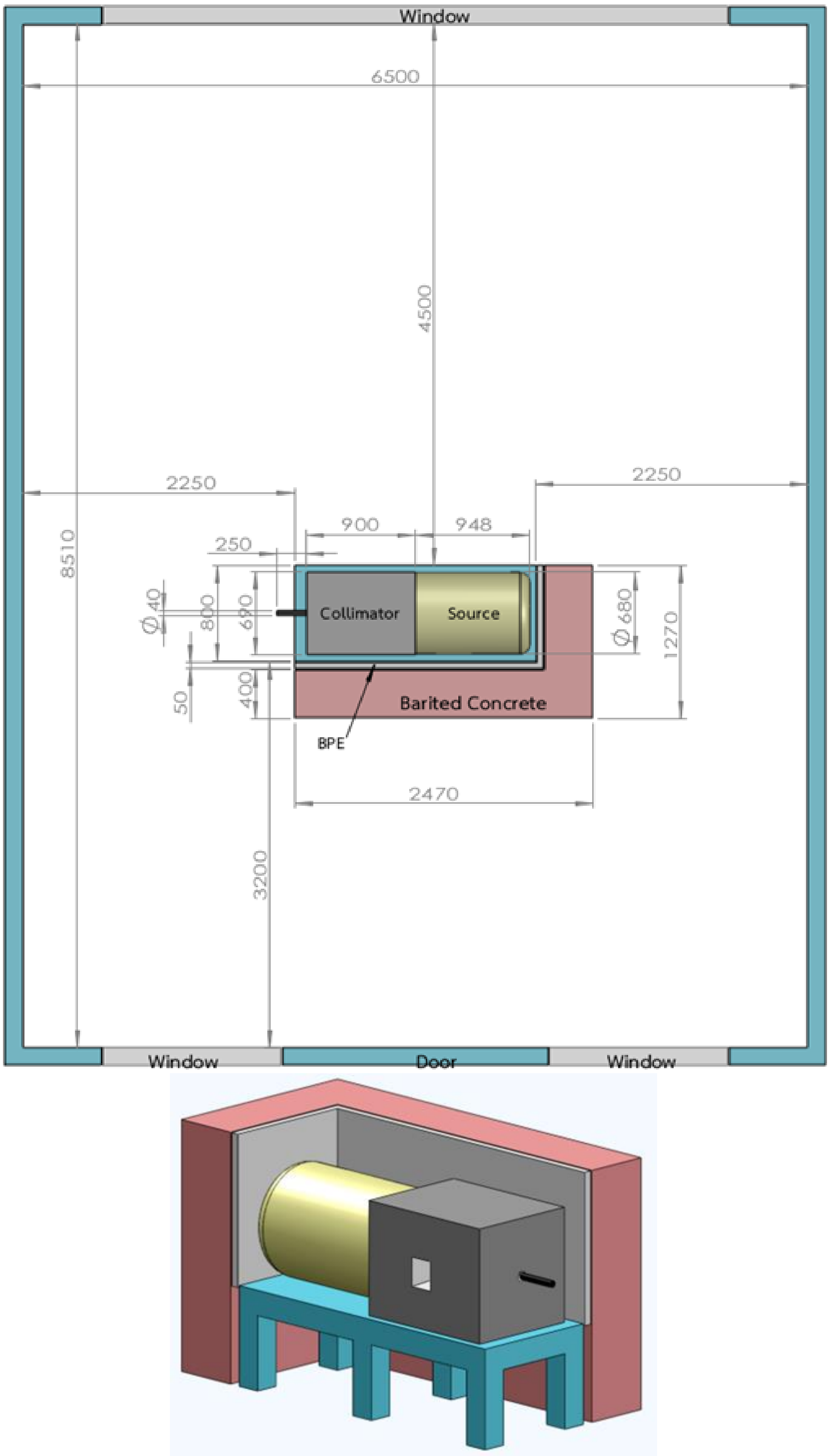


Figure 1. The layout of the TINT's neutron irradiation facility. The upper figure shows the dimensions (unit: mm) of the room and the neutron irradiator. The lower figure displays the setup of the neutron irradiator, which consists of the storage container and the collimator enclosure with the aperture of 15 × 15 cm^2, both sitting on the iron table. It is possible to insert a graphite absorber inside the collimator duct to reduce the neutron flux. The neutron irradiator is surrounded by the shielding wall, which is made of 5 cm thick borated polyethylene sheets and 40 cm thick heavy concrete.

2.2. Neutron field characterization

2.2.1. Monte Carlo simulations. The general purpose Monte Carlo Particle and Heavy Ion Transport code System (PHITS) version 2.64 [3] was used for simulating the neutron irradiation facility. The simulation took into account the source encapsulation, storage container, collimator, iron table, concrete floor, ceiling and shielding materials. Interactions of neutrons, photons and charged particles were considered. The Event-Generator mode implemented in PHITS was used for transporting neutrons with energy below 20 MeV. The cut-offs for radiation transport were set at 1 keV for scattered photons and 1 keV/u for secondary ions. The neutron energy spectrum of the bare source (without encapsulation) was taken from the ISO recommendation [4].

Spherical virtual detectors with the diameter of 1 cm filled with air were used for tallying neutrons and gammas, which were incident at selected positions of the room. Neutron dose equivalent (H_N) was calculated using the ICRP conversion method [5] according to

$$H_N = \sum_i k_i(E)\phi_{i,N}(E) \tag{1}$$

where $k_i(E)$ is the ICRP's ambient dose equivalent per unit neutron fluence ($H^*(10)/\phi$) (Figure 2) [5], E is the neutron energy, and $\phi_{i,N}(E)$ is the calculated neutron fluence in the energy interval between E and $E+\Delta E$. For scattered gammas, the kerma approximation implemented in PHITS was used for determination of scattered photon dose. The simulation results were normalized to the neutron emission rate (10^8 neutrons/s).

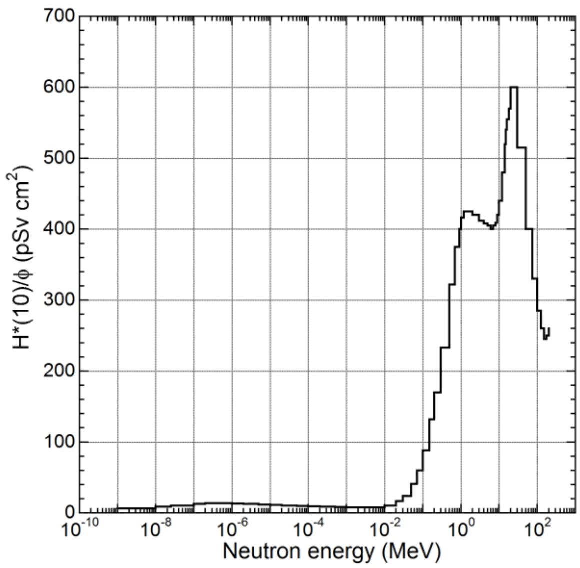


Figure 2. The ICRP's ambient dose equivalent per unit neutron fluence [5].

2.2.2. Radiation measurements. Measurements of neutron and gamma dose rates and neutron energy spectra were carried out to accompany the Monte Carlo calculations. Commercial survey meters were used for measurements of neutron and gamma dose equivalent rates (Ludlum model 12-4 of Ludlum Measurements, Inc. and Radiagem™ 2000 of Canberra Industries, Inc., respectively). For neutron spectrometry, a neutron spectrometer was developed in-house with a similar design as the Nested Neutron Spectrometer of Dubeau et al [6] (patented and commercialized by Detec) and the same working principles as well-known Bonner sphere spectrometers [7].

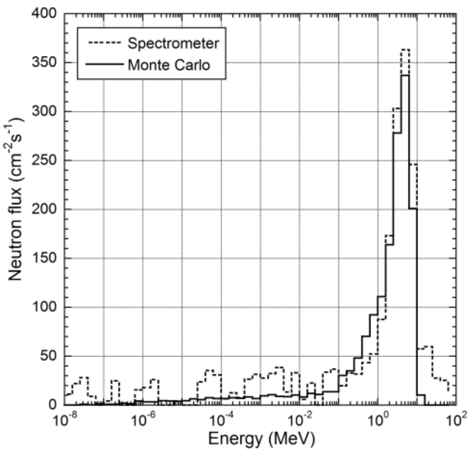
The neutron spectrometer consists of a small cylindrical ^{3}He thermal neutron detector of the model LND251 (LND, Inc.) and 7 cylindrical moderator shells made of high density polyethylene. The outer diameters of the moderator shells range from 7.5 to 20 cm. The smallest shell has an inner cavity that fits the neutron detector. Each shell was designed to fit into the next larger shell. By sequentially placing each shell to the detector and to the smaller shells, the neutron count rates associated with eight moderator thicknesses were measured. The neutron count rates (C) were unfolded to obtain the neutron energy spectra (ψ) for 50 energy bins by solving a system of linear equations:

$$C = R\psi \tag{2}$$

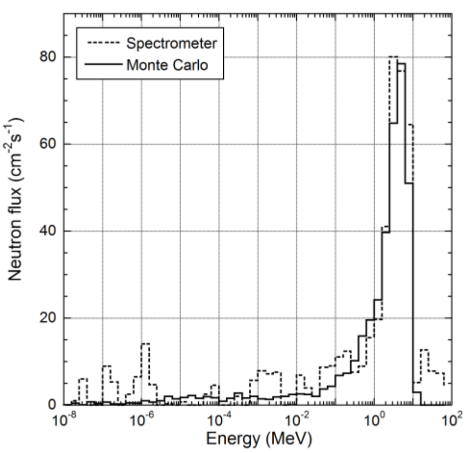
where R is the spectrometer's response matrix obtained using Monte Carlo simulations. Equation (2) was solved by the Levenberg-Marquardt method implemented in MATLAB (The MathWorks, Inc.).

3. Results and Discussion

Figure 3 shows the calculated and measured neutron energy spectra at the source-to-detector distances of 100 and 200 cm, without the graphite absorber inside the collimator duct. The Monte Carlo simulation and the spectrometer measurement showed good agreement in the neutron energy range of 0.1-10 MeV, covering the energy region in which the ICRP fluence-to-dose conversion factors are relatively large (Figure 2). Discrepancies between the two data sets are observed at lower energies, but neutrons of this energy range have a small contribution to the total neutron dose equivalent. At energies above 10 MeV, the spectrometer overestimated the simulation results. Since the ICRP conversion factor maximizes in this energy range (Figure 2), the neutron dose equivalent determined using the spectrometer are expected to be higher than those obtained using the Monte Carlo simulations. Comparing the neutron energy spectra at 100 and 200 cm source-to-detector distances, the magnitude of the neutron flux at the closer distance is about 4-4.5 times as high as the neutron flux at the larger distance, in close agreement to the inverse square law.



(a)



(b)

Figure 3. The neutron energy spectra at the source-to-detector distances of (a) 100 and (b) 200 cm, without the graphite absorber inside the collimator duct. The solid lines were obtained using the Monte Carlo simulations and the dashed lines were determined by the measurement with the in-house developed neutron spectrometer.

Table 1 and Table 2 show the evaluated neutron and gamma dose equivalent rates at the source-to-detector distances 75, 100 and 200 cm, with and without the graphite absorber inside the collimator duct, respectively. At closer distances (<75 cm) the field size would be too small (smaller than 20 × 20 cm^2), while at larger distances (>200 cm) the detector would be too close to the concrete wall and the room scatter contribution appreciably increased. The final dose equivalent rates were averaged over the results obtained by the Monte Carlo simulations, the survey meters and, in some cases, also the neutron spectrometer. For neutrons, the standard deviations of the means were relatively small without the graphite absorber (from less than 1% up to 18%), and increased up to 30% when the graphite absorber was used. The calculated and measured gamma dose rates differed by 1-8.4 folds, but in all cases the gamma contributions were less than 4.2-7.5% of the neutron dose equivalent rates.

The maximum neutron dose equivalent rate was about 3.5 mSv/h without the graphite absorber. By placing the graphite absorber inside the collimator duct, it was possible to reduce the neutron dose equivalent rate to 156 μSv/h at the source-to-detector distance of 200 cm. In all cases, nearly 100% of the neutron dose equivalent rates were contributed by neutrons with energies larger than 0.01 MeV. Despite the small contribution of thermal neutrons to the total neutron dose, the thermal component can be important for calibration of detectors that are sensitive to thermal neutrons e.g. TLD, BF_3 and ^3He detectors.

Table 1. Neutron and gamma dose equivalent rates at different source-to-detector distances (without the graphite moderator).

Distance (cm)	Neutron dose equivalent rates (μSv/h)				Gamma dose equivalent rates (μSv/h)		
	MC	Survey meter	Spectrometer	Average	MC	Survey meter	Average
75	3,537	3,500	-	3,518.7	75	148	111.7
100	1,896	2,000	2,243	2,046.3	59	85	72.1
200	445	375	534	451.3	5	21	13.0

Table 2. Neutron and gamma dose equivalent rates at different source-to-detector distances (with the graphite moderator).

Distance (cm)	Neutron dose equivalent rates (μSv/h)			Gamma dose equivalent rates (μSv/h)		
	MC	Survey meter	Average	MC	Survey meter	Average
75	1,463	1,000	1,231.6	80	85	82.5
100	786	525	655.5	49	49	48.9
200	177	135	156.0	1	12	6.5

Apart from generating neutron dose equivalent rates for calibration of dose rate meters, the investigated neutron fields can be used for irradiation of integrating dosemeters e.g. passive dosemeters. It is possible to adjust the irradiation time according to the required neutron dose, the source-to-detector distance and the configuration with or without the graphite absorber. Table 3 shows examples of the irradiation time required for neutron dose equivalents of 0.2-3 mSv at the source-to-detector distance of 100 cm, with and without the graphite absorber. This range of neutron dose equivalents is required for the routine quality assurance of the TLD and OSL personal dosemeters serviced by TINT.

For relatively low dose, it is possible to choose between the configurations with and without the graphite absorber. In both cases, the normalized neutron energy spectra were not significantly different. For applications that require the time uncertainty to be minimal, longer irradiation time (with the graphite absorber) is preferred. For relatively large dose (e.g. 2-3 mSv), due to time constraints, the configuration without the graphite absorber might be the only alternative.

4. Conclusions

The neutron fields at the TINT's multi-purpose neutron irradiation facility were characterized using Monte Carlo simulations, an in-house developed neutron spectrometer and commercial survey meters. The facility is equipped with a 50 Ci ^{241}AmBe neutron irradiator with the collimator of 15 × 15 cm^2 aperture. It is possible to place a neutron absorbing material inside the collimator duct to reduce the neutron flux. The investigated neutron fields generated neutron dose equivalent rates ranging from 156 μSv/h to 3.5 mSv/h. In all cases, the largest dose component was due to neutrons of energies above 0.01 MeV. Depending on the irradiation setup, the gamma contamination was at maximum 4.2-7.5% of the neutron dose. Irradiation time for integral dose measurement can be varied by changing the irradiation configuration such as the source-to-detector distance and the neutron absorbing material inside the collimator duct. The characterized neutron fields can facilitate calibration test and quality assurance of neutron dose rate meters and personal dosemeters commonly used in radiation protection dosimetry. The work has paved the way for setting up a calibration facility for neutron measuring devices in Thailand.

Table 3. Required neutron irradiation time and expected gamma dose equivalents at the source-to-detector distance of 100 cm**.**

Neutron dose equivalent (mSv)	Without graphite absorber		With graphite absorber	
	Irradiation time (min)	Expected gamma dose equivalent (mSv)	Irradiation time (min)	Expected gamma dose equivalent (mSv)
0.2	5.9	0.01	18.3	0.01
0.4	11.7	0.01	36.6	0.03
0.6	17.6	0.02	54.9	0.04
0.8	23.5	0.03	73.2	0.06
1.0	29.3	0.04	91.5	0.07
1.5	44.0	0.05	137.3	0.11
2.0	58.6	0.07	183.1	0.15
3.0	88.0	0.11	274.6	0.22

Acknowledgments

The authors wish to thank Department of Nuclear Engineering, Chulalongkorn University and Nuclear Technology Service Center and Safety Unit at TINT for supporting some of the experimental equipment.

References

[1] IAEA 1988 *Guidelines on Calibration of Neutron Measuring Devices (Technical Report Series No. 285)* (Vienna, Austria: International Atomic Energy Agency)
[2] ISO 2000 *Reference neutron radiations - Part 2: Calibration fundamentals of radiation protection devices related to the basic quantities characterizing the radiation field* (Geneva, Switzerland: International Organization for Standardization)
[3] Sato T *et al* 2013 Particle and Heavy Ion Transport code System, PHITS, version 2.52 *J. Nucl. Sci. Technol.* **50** 913–23
[4] ISO 2001 *Reference neutron radiations - Part 1: Characteristics and methods of production* (Geneva, Switzerland: International Organization for Standardization)
[5] ICRP 1996 Conversion Coefficients for use in Radiological Protection against External Radiation (ICRP Publication 74) *Ann. ICRP* **26** 3–4
[6] Dubeau J, Witharana S S H, Atanackovic J, Yonkeu A and Archambault J P 2012 A neutron spectrometer using nested moderators *Radiat. Prot. Dosimetry* **150** 217–22

[7] Thomas D J and Alevra A V 2002 Bonner sphere spectrometers—a critical review *Nucl. Instr. Meth. Phys. Res. A*. **476** 12–20