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Comparison of conversion factor methods in spectrometric dosimetry systems based on NaI(Tl) scintillation detectors

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HIGHLIGHTS

- Dosimetry methods have been developed based on the analysis of detector output data.
- Three different techniques are studied to obtain the conversion coefficients.
- The subdivision of measured spectrum method is provided to modify and improve the evaluated conversion coefficients.
- The dose quantity calculation error percentage reduces by utilizing the improved coefficients.

ABSTRACT

To monitor personal safety in the fields of biomedical and health physics, it is necessary to be aware of radiation doses to protect the health and safety of persons. Radiation protection quantities such as air kerma, ambient dose equivalent, and exposure dose rate are obtained by the measured spectrum to determine energy-dependent conversion coefficients/factors. This study aims to obtain and compare an ambient dose equivalent to $H^*(10)$ from the measured gamma-ray spectra by the NaI(Tl) scintillation detector using two various methods. The first method, which is based on the detector response function to find the conversion function, is called the $G(E)$ method. The second method is subdividing the measured gamma-ray spectra into the multiple energy bins, and then obtaining the ambient dose equivalent by using conversion coefficient functions ($\omega(E)$), which were determined by the conversion coefficients (ω_i) of each energy bin for three energy intervals of $\leq$ 185 keV, 185 to 850 keV, and $\geq$ 850 keV. To calculate the detector response matrix and the conversion coefficients of each region of energy, the Monte Carlo simulation code was used for the quasi-mono energetic gamma radiation sources and the synthetic spectra. The results indicate that using the technique based on subdividing the measured spectrum into multiple energy bins helps to avoid the inverse detector response matrix dimension limitations that occur in the $G(E)$ method and also have a lower error percentage in the dose quantity calculation. Consequently, NaI(Tl) scintillation detector has an excellent potential to replace the classical dose rate instruments, i.e. Geiger-Muller, for the early warning of environmental radiation monitoring.

KEYWORDS

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Monte Carlo simulation
Ambient dose
Spectro-Dosimeter method
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1 Introduction

In radiation protection and health physics, to evaluate and minimize the risk of radiation-induced effects on radiation workers, it is increasingly important to know the dose value in a real-time manner. Awareness of dose values is necessary for imaging procedures for diagnosis, treatment, and disease prevention. These procedures, particularly in Iran, are generally conducted with the use of radioactive materials, such as the planar (gamma camera) and single-photon emission computed tomography (SPECT). The thermoluminescence dosimeter (TLD) and photographic dosimeter, known as integrating dosimeter (Attix, 2008a), can not directly represent the real-time dose values, and they are not convenient for emergency scenarios. On the other hand, pulse-mode dosimeters such as the Geiger-Muller tube, silicon p-i-n photodiode (Si-PIN), silicon photomultiplier (SiPM), and scintillation detectors such as cerium chloride doped with lanthanum bromide ($LaBr_3(Ce)$), thallium doped with cesium iodide (CsI(Tl)), thallium doped with sodium iodide (NaI(Tl)), and high-purity germanium (HPGe) can

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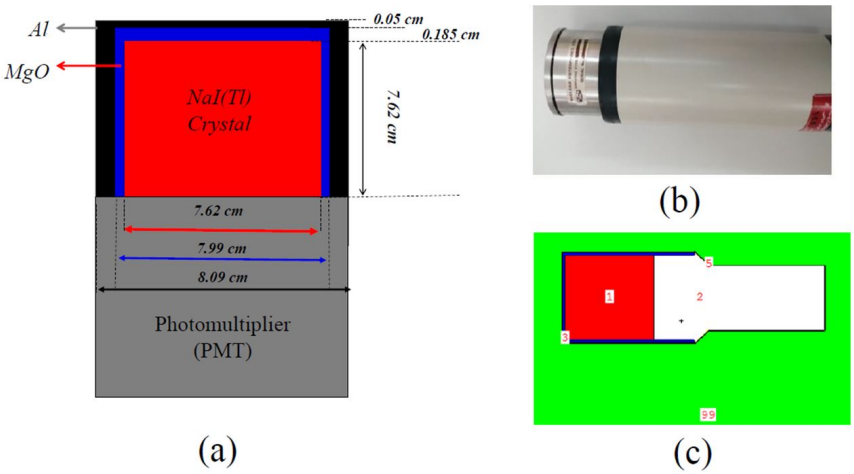

Figure 1: The schematic of an NaI scintillator: a) A 2D representation, b) experimental, and c) MCNPX model.

be used in the real-time measuring of the air kerma rate or the ambient dose equivalent rate $H^*(10)$ (Wang et al., 2015). The major advantage of the above-mentioned nuclear radiation detectors is that they measure the energy of charged particles and gamma-rays, in addition to dose quantity. It means that nuclear radiation detectors as spectra-dosimeter instruments can easily be used for the qualitative and quantitative analysis of radioisotopes. However, there are some risks associated with these procedures. Improving these measurement instruments with the best equipment or new software approaches can facilitate the better measurement of the radiation field and dosage simultaneously (Moriuchi and Miyanaga, 1966; Casanovas et al., 2016; Dizaji et al., 2014).

Since scintillation detectors have higher average atomic numbers than the tissue, it is impossible to determine the dose values directly from the detector output. Hence, the photon energy response of scintillation detectors must be modified. For this purpose, there exist various methods, e.g., coupling p-i-n diode or silicon photomultiplier to CsI(Tl) scintillation detector (Buzhan et al., 2018; Park et al., 2020), using software methods such as unfolding (Rahman and Cho, 2010), and deconvolution (stripping method) methods (Camp and Vargas, 2014).

Portable detectors such as NaI(Tl) scintillator can be used as a spectrometric system to measure the gamma-ray energy spectra. The energy spectrum measured with this detector provides information on photon energy for any gamma radiation sources; however, this energy spectrum includes partial energy deposition events due to the Compton scattering and gamma-ray escape. Therefore, by correcting and modifying the output of the NaI(Tl) scintillation detector, the measurement of the essential dosimeter quantities such as air kerma or ambient dose equivalent $H^*(10)$ is possible.

In the present work, the methodology for calculating air kerma and the ambient dose equivalent from the measured pulse-height distribution using the NaI(Tl) scintillation detector based on two software methods is described and compared. The first method, which is frequently called the $G(E)$ method, is based on the detector response function to find the conversion function. The second method is based on subdividing the measured gamma-ray spectra into multiple-energy bins to identify the conversion coefficients (ω_i) for each energy bin. We intend to obtain $H^*(10)$ from the measured gamma-ray spectra by the NaI(Tl) scintillator using the two above-mentioned methods. As a result, we see that the subdivision of the measured spectrum into multiple energy bins to avoid the dimension limitations of the inverse detector response matrix, which occurred in the $G(E)$ method, is more favorable. The calculated error percentage in the subdivision method is much less than that in the $G(E)$ method.

2 Simulation and Method

2.1 Monte Carlo (MC) simulation

MC simulation is the most accurate technique for tracking particle transport in any material. Due to the limited availability of quasi-mono-energetic gamma-ray sources to construct a spectrum database, it is necessary to precisely simulate the detector and its responses to the radiation sources. The MCNP4C code uses the track structure MC simulation, which provides valuable information for understanding the initial pattern of energy depositions in materials, detector response function, and pulse-height spectrum (Briesmeister, 2000). The gamma spectrum of the scintillation detector is the combination of partial (Compton scattering) and full energy deposition (photoelectric effect and escape peaks) events.

In the present work, the MCNP4C code is used to obtain the pulse-height spectra of the NaI(Tl) detector when it is exposed to gamma-ray sources. The detector model used in the simulation is the same as the scintillation detector used in a laboratory for experimental studies, which is a cylindrical NaI(Tl) (7.62×7.62 cm^2) as a sensitive volume covered by a MgO and Al layer. The inner structure and the density of this detector material are shown in Fig. 1 and Table 1, respectively.

According to energy calibration and resolution calibration, the Full Width at Half Maximum (FWHM) func-

Table 1: The material compositions and geometry of the NaI(Tl) detector.

Component	Density (g.cm^{-3})	Size (cm)
NaI(Tl) (Crystal)	3.67	7.62×7.62
Al (Layer)	2.70	0.05
MgO (Layer)	3.58	0.185
Photomultiplier tube (PMT) (to be treated as a vacuum aluminum cylinder)	2.70	8.32×23

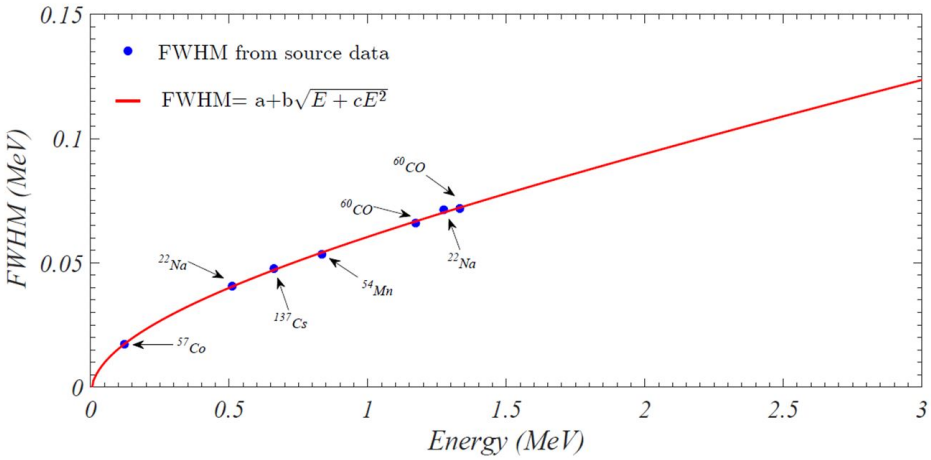

Figure 2: FWHM variation curve as a nonlinear function of energy.

tion is considered to recover the statistical factors for broadening the photopeak with the energy axis or applying the Gaussian Energy Bordering (GEB) with a Gaussian function. The photopeaks in the measured spectra are Gaussian shaped and are not sharp as in the simulations. Therefore, in MC simulations, the GEB is used to broaden the photopeaks. The measured peaks are often represented by FWHM, where the best fitting function of FWHM can be written as (Salgado et al., 2012):

$$FWHM = a + b\sqrt{E + cE^2} \tag{1}$$

where a, b, and c are the fitting parameters. For this work, the values of these parameters for the NaI(Tl) detector are $a = -0.0033$ MeV, $b = 0.0584$ MeV$^{1/2}$, and $c = 0.1911$ MeV^{-1}. With these coefficients, the FWHM variation curve is a nonlinear function of energy. By considering the energy and resolution calibration of the NaI(Tl) detector, the simulation approach can be compared with the experimental results. Figure 2 shows a graph of the FWHM function for an NaI(Tl) scintillation detector.

In gamma-ray spectroscopy, the relation between the measured spectrum N and the fluence of the incident gamma-rays $\overrightarrow{\phi}$ can be expressed as:

$$\overrightarrow{N} = R.\overrightarrow{\phi} \tag{2}$$

where R is the detector response matrix or the library of the detector responses to monoenergetic gamma-rays. The response matrix R was calculated by using the F8 tally of the MCNP4C code. Setting the number of primary particles at 10^8 keeps the uncertainties well below 0.2% for parallel-beam mono-energetic gamma-rays with energies ranging from 0.03 to 3 MeV.

2.2 The $G(E)$ function method

2.2.1 The $G(E)$ function by using the response matrix

In the spectro-dosimetry method, the dosimetric quantities, e.g., ambient dose equivalent, air kerma, or exposure dose, can be calculated from the pulse-height spectrum of any spectrometric detectors. By knowing the ambient dose equivalent response function (fluence to ambient dose equivalent conversion factors), which was introduced and defined by the ICRP 47 document (ICRP, 1996), dose quantities can be calculated. Furthermore, the ambient dose equivalent for each given fluence energy can be described as:

$$H^*(10)(E) = \overrightarrow{C}(E).\overrightarrow{\phi}(E) \tag{3}$$

where $\overrightarrow{C}(E)$ is the fluence to ambient dose equivalent conversion factor. By substituting Eq (2) into Eq (3), the ambient dose equivalent can be presented as:

$$H^*(10) = \overrightarrow{C}.(R^{-1}.\overrightarrow{N}) = (\overrightarrow{C}.R^{-1}).\overrightarrow{N} \tag{4}$$

Correspondingly, the dose conversion factor $G(E)$ can be determined as:

$$G(E) = \overrightarrow{C}.R^{-1} \tag{5}$$

It should be noted that the $G(E)$ function is dependent on both the detector geometry and the direction of incident irradiation. Moreover, the low energy resolution of the NaI(Tl) detector affects the calculation of the inverse detector response matrix. Due to the zero determination of the square response matrix in the inverse calculation of the matrix, the dimensions of the response matrix were

reduced from 1024 × 1024 to 50 × 50. The detector response matrix was assumed to be a 50 × 50 square matrix with involved energies ranging from 0.03 to 3 MeV with energy intervals of 60 keV. To reduce the dose value errors, it is recommended to use a polynomial function that has the power of the natural logarithm of the energy with a gradient-descent method, least-square method, and adaptive moment estimation method (Park et al., 2020).

2.2.2 The $G(E)$ function by using the least-square estimation

In general, the relation between the measured spectrum $N(E)$ and the fluence of the incident gamma spectrum $\phi(E_0)$ can be expressed as (Tsoulfanidis and Landsberger, 2021):

$$N(E) = \int_{E_{min}}^{E_{max}} R(E, E_0)\, \phi(E_0)\, \mathrm{d}E_0 \tag{6}$$

where $R(E, E_0)$ is the response function, which represents the gamma rays of energy E_0 that is the deposited energy in the NaI(Tl) scintillation detector. By using the ambient dose equivalent $H^*(10)(E_0)$ generated at a given fluence rate $\phi(E_0)$ from Eq. (3), the fluence-to-ambient dose equivalent conversion factors for specific gamma-ray energy can be described as (Attix, 2008b):

$$\begin{aligned} C(E_0) &= \frac{H^*(10)(E_0)}{\phi(E_0)} \\ &= \int_{E_{min}}^{E_{max}} R(E, E_0)\, G(E)\, \mathrm{d}E \end{aligned} \tag{7}$$

Here, $G(E)$ is a spectrum to dose conversion function. The $G(E)$ function can be expressed as a polynomial function with the power of the natural logarithm of the energy approximated (Huang, 2018; Tsuda and Saito, 2017):

$$G(E) = \sum_{K=1}^{KMAX} A(K) \times \left(\log_{10}(E)\right)^{K-M-1} \tag{8}$$

where $KMAX$ is the maximum term number of the polynomial equation, M is a constant, and $A(K)$ refers to the coefficient for the K^{th} term in the $G(E)$ equation, which was calculated by applying the least-square method. Consequently, the ambient dose equivalent can be written as:

$$H^*(10) = \sum_{i=1}^{k} N(E_i)\, G(E_i) \tag{9}$$

in which k is the number of channels in spectrometric systems. Figure 3 shows the $G(E)$ function curve in Eq. (5) and Eq. (8) for $KMAX = 8$.

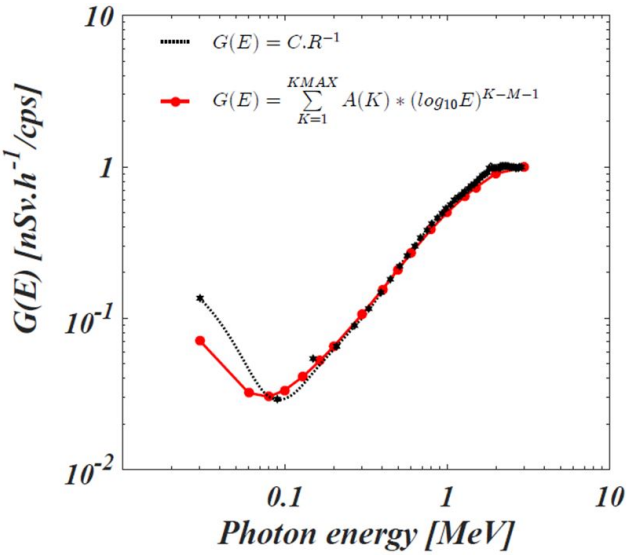

Figure 3: The conversion factors ($G(E)$ function) for the response spectrum of a NaI(Tl) scintillation detector.

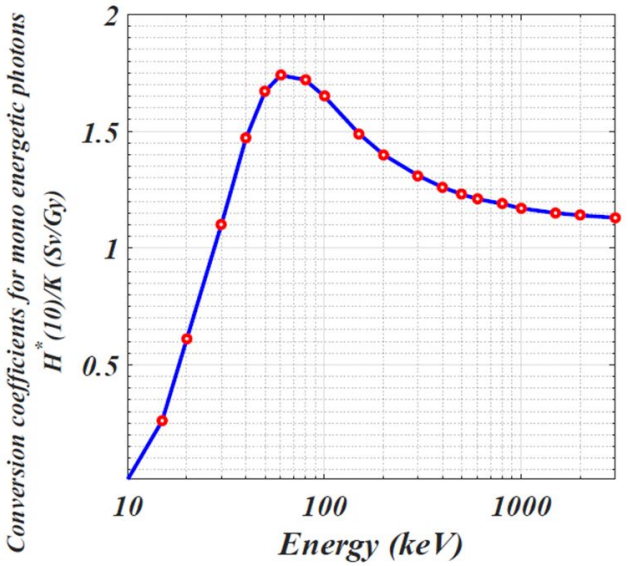

Figure 4: The conversion coefficients (ICRP, 1996) for monoenergetic photons relating air kerma to ambient dose equivalent.

2.3 *The subdivision of the measured gamma-ray spectrum method*

The second method is subdividing the measured gamma-ray spectra into multiple energy bins to obtain the energy-dependent conversion coefficient, instead of using the $G(E)$ function method. In this method, the total energy deposited in the NaI(Tl) detector is well correlated with the air kerma values, which are measured by ionization chambers. There is a relationship between the number of counts in the measured spectra and the air kerma responsible for these counts (Lowder et al., 1964) due to the ratio of mass energy-absorption coefficients between air and crystal in the energy range of 0.4 to 3 MeV which is relatively constant. Therefore, the total energy deposited in the

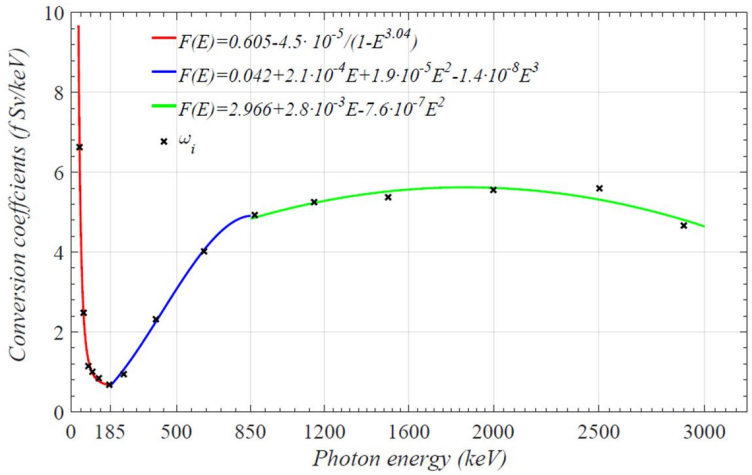

Figure 5: The energy-dependent conversion coefficient functions ($\omega(E)$) for a NaI(Tl) scintillation detector.

NaI(Tl) detector may be proportionate to air kerma with a unique conversion coefficient. However, at low gamma-ray energies, due to the photoelectric effect, the ratio of mass energy absorption is not constant (Grasty et al., 1992). As a result, we can not use a unique conversion coefficient between the deposited energy in the detector and kerma in the air. To solve this problem, the spectrum is divided into several energy regions ν, and for each one, a calibrated conversion coefficient is applied. The total air kerma resulting from the sum of ν particle values of kerma can be calculated from each energy region. The relationship between the air kerma and energy deposited in the detector for each region can be written as:

$$K = \sum_{i=1}^{\nu} \omega_i \, n_i \, E_i \tag{10}$$

where ω_i is the energy-dependent conversion coefficient of the energy region i, and the value $n_i E_i$ corresponds to the energy deposited in the region i. the conversion coefficients that should be used to convert air kerma to ambient dose equivalent $H^*(10)$ have been given in the ICRP 47 document (ICRP, 1996). Also, these conversion coefficients $H^*(10)/K$ for monoenergetic photons are displayed in Fig. 4.

If the energy-dependent conversion coefficient ω_i is obtained by the MC simulation, the deposited energy in the detector can be directly related to the ambient dose equivalent, $H^*(10)$. Then, Eq. (10) is changed to:

$$H^*(10) = \sum_{i=1}^{\nu} \omega_i \, n_i \, E_i \tag{11}$$

To solve Eq. (11), which is a system of linear equations, the Cholesky decomposition method is used (Madar, 2015). Thereby, the general solution obtained is:

$$\omega_i = \frac{H^*(10)_i - \sum_{j=1}^{i-1} n_{ij} \, E_j \, \omega_j}{n_{ii} \, E_i} \tag{12}$$

The selected position of these desired energy boundaries in Eq. (12), including most of the energy in the photopeak deposited in the associated window selected energy regions in the MC simulation together with the obtained conversion coefficients, are summarized in Table 2. It should be mentioned that in each energy region, there must be at least one strong energy line, and the number of conversion coefficients is equal to the number of sources used. Clearly, the number of these conversion factors for measuring the ambient dose value is less than the conversion factors in the $G(E)$ method.

Figure 5 shows three different fitting functions defined in the energy ranges $\leq$ 185 keV, 185 to 850 keV, and $\geq$ 850 keV to be used for generating the values of the energy-dependent conversion coefficients ω_i for the NaI(Tl) detector. By using these fitting functions, Eq. (11) can be replaced by:

$$H^*(10) = \sum_{i=1}^{\nu} \omega_i(E_i) \, n_i \, E_i \tag{13}$$

Table 2: Conversion coefficients ω_i calculated for the NaI(Tl) detector.

γ peaks (keV)	Energy regions (keV)	Conversion coefficient (pSv.keV^{-1})
40	25 - 55	6.63×10^{-3}
60	55 - 65	2.47×10^{-3}
80	65 - 95	1.14×10^{-3}
100	95 - 105	1.01×10^{-3}
130	105 - 155	8.39×10^{-4}
180	155 - 205	6.76×10^{-4}
250	205 - 295	9.48×10^{-4}
400	295 - 505	2.32×10^{-3}
630	505 - 755	4.03×10^{-3}
870	755 - 985	4.92×10^{-3}
1150	985 - 1315	5.26×10^{-3}
1500	1315 - 1685	5.37×10^{-3}
2000	1685 - 2315	5.55×10^{-3}
2500	2315 - 2685	5.59×10^{-3}
2900	2685 - 3115	4.66×10^{-3}

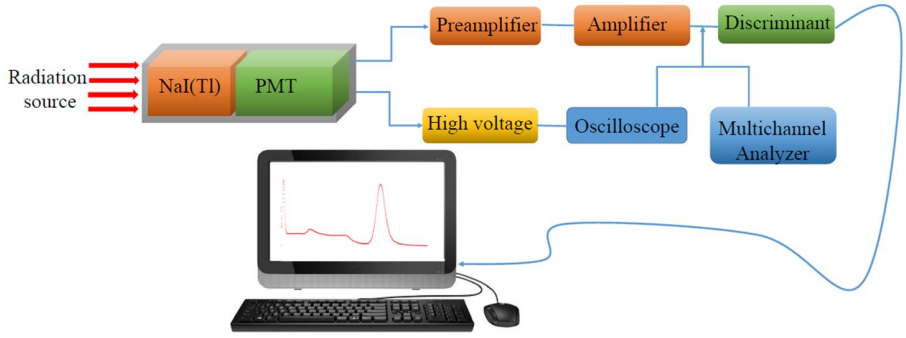

Figure 6: Demonstration of a scintillation detector system.

Table 3: Experimental and simulation dose values.

Source	Value	$H^*(10)(\mu Sv.h^{-1})$	$K(\mu Gy.h^{-1})$
Cs-137	Experimental: $\omega(E)$	0.309 ± 0.03	0.254 ± 0.02
	$G(E) = \overrightarrow{C}.R^{-1}$	0.288 ± 0.05	0.231 ± 0.08
	$G(E) = \sum_{K=1}^{KMAX} A(K) \times \left(\log_{10}(E)\right)^{K-M-1}$	0.297 ± 0.04	0.249 ± 0.05
	Theoretical	0.314	0.262
Co-60	Experimental: $\omega(E)$	0.208 ± 0.01	0.168 ± 0.02
	$G(E) = \overrightarrow{C}.R^{-1}$	0.172 ± 0.06	0.147 ± 0.04
	$G(E) = \sum_{K=1}^{KMAX} A(K) \times \left(\log_{10}(E)\right)^{K-M-1}$	0.189 ± 0.03	0.159 ± 0.06
	Theoretical	0.201	0.172

3 Results and discussion

To investigate the structure of this method in the dosimetry field, gamma radiation sources are placed at a distance of 10 cm from the detector surface, and the NaI(Tl) detector output signal is fed into a preamplifier. Then its output is connected to an amplifier, which is used to avoid shifting the baseline. The output signal of the amplifier is directly given to the multi-channel analyzer to obtain the energy spectrum of each gamma-ray source. The NaI(Tl) scintillation detector readouts for various gamma-ray sources are experimentally obtained using the geometry shown in Fig. 6. The main point for using the conversion coefficients to obtain the air kerma or ambient dose equivalent in the experimental process should subtract the background spectrum (the environmental spectrum without any gamma-ray sources) from the measured spectrum.

Table 3 summarizes the results of the measured and calculated air kerma rate and ambient dose equivalent rate of some standard gamma-radiation sources, which have been obtained in this work.

The results show that due to the low energy resolution of the NaI(Tl) detector (energy resolution of 7.1% at the energy peak of 662 keV) and also the limitation in taking an inverse of the detector response matrix, the probability of finding the conversion factor for measuring the error of air kerma and ambient dose equivalent can be reduced by using the subdivision technique. Uncertainty values, which are shown in Table 3, are less than 10%, which is acceptable for general guidance on uncertainty levels based on the ICRP recommendations (ICRU, 2016). To this end, by using the subdividing gamma-ray spectrum method, we can decrease the uncertainty of measuring the dose value with the NaI(Tl) detector in the dosimetry field.

4 Conclusions

In this study, we show that in addition to detecting radioisotope radiations emitted from natural radioactive sources by the NaI(Tl) detector, the dose rate values of the spectra can be calculated by determining the conversion factors. Two diverse methods including the $G(E)$ function and the subdivision of the energy spectra have been studied and compared. The results show that the subdivision method is more beneficial compared to the $G(E)$ function method since it uses a smaller number of conversion coefficients and avoids the limitation of the detector response function. Furthermore, the error percentage of measuring the dose rate using the subdivision method is less than that of using the $G(E)$ function method. For this purpose, the conventional dosimeter, e.g. the Geiger-Muller tube, can be replaced with these real-time spectrometric systems for emergency scenarios.

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