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Development and Validation of an MCNP model for a Broad-Energy Germanium Detector

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Abstract

An MCNP model was developed for a Broad-Energy Germanium Detector (Model BE2830). This model was constructed based on the manufacturer specifications provided for the detector. In order to develop the complete MCNP model, the activity of three standard radioactive sources (Co-57, Co-60 and Cs-137) was calculated by estimating the detector absolute full energy peak efficiency at different gamma-ray energy lines, using the MCNP model, and measuring count rates due to those energy lines, using the BEGe detector, with different setup configurations. The obtained results were in agreement with certified values with a relative difference ranging from -1.84 % to 1.57 %. As an application for the constructed MCNP model, ^{235}U mass contents for five Standard Nuclear material (SNM) samples were estimated. The obtained results indicate that this model could be used effectively for nuclear safeguards purposes.

Keywords: MCNP model, BEGe detector; MCNP efficiencies; activity; ^{235}U mass contents; standard NM sample.

1. Introduction

Germanium (Ge) detectors are the most common tool to assay nuclear materials in all sizes, shapes and composition nondestructively [1, 2].

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Efficiency calibration is a central aspect of accurate nuclear material quantification. Generally, the efficiency is dependent on the gamma-ray energy, the entire setup geometry, and the composition of any material in the path of the sample and the detector [1, 2].A relative method is the best way to get the most accurate results, however, radioactive standards are not always available for assayed material. Therefore, semi-absolute or absolute methods can be considered. This requires the determination of the parameters and factors affecting the measurements [3]. The most important factor affecting the measurements is the details of the detector characteristics. Specification of the detector design provided by the manufacturer is not always adequate for developing an accurate model. As a result, effort must be spent to obtain accurate detector characterization. Monte Carlo (MC) and semi-analytical methods that simulate radiation transport provide a tool to address virtually any aspect of the NDA measurement that departs from the ideal case [4]. Canberra Broad Energy Ge (BEGe) Detector covers the energy range of 3 keV to 3 MeV like no other. The resolution at low energies is equivalent to that of Low Energy Ge (LEGe) Detector and the resolution at high energy is comparable to that of good quality coaxial (SEGe) detectors [5]. Most importantly, the BEGe has a short fat shape which greatly enhances the efficiency below 1 MeV for typical sample geometries. The main purpose of this paper is to develop an MCNP model for Broad-Energy Germanium Detector (Model BE2830) in order to validate the physical dimensions of the detector provided by the manufacturer.

2. Method and experiment

2.1. Detector Specifications

Commercial Canberra portable Radionuclide Identifier (Falcon 5000®) with BEGe detector (Model BE2830) [5], is shown in Figure1. The germanium crystal has a diameter of 60.80 mm and a height of 30.90 mm. The crystal is held by an aluminum cup in a 1.5-mm-thick aluminum endcap and placed 13.2 mm from the front window. The front window is made of 1.2 mm-thick aluminum. The recommended bias voltage is -3300 V. The data acquisition system in this work involves a pre-amplifier (Model PSC823C) and the Genie-2000 software. The system comes with a full version of Canberra's industry leading gamma analysis software, Genie 2000. The full power of Genie 2000 analysis is available in the Falcon 5000 [5].

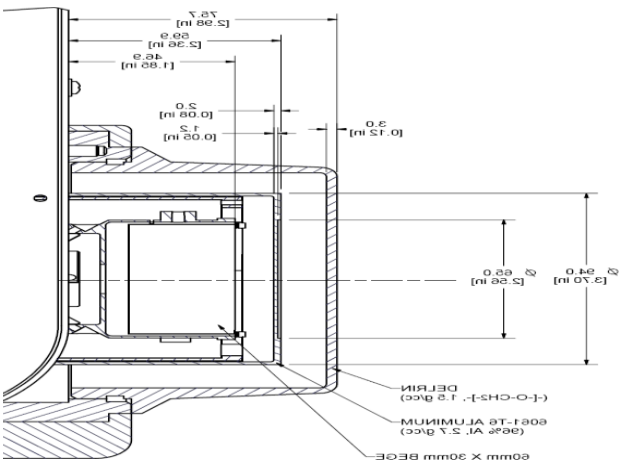

Figure 1: Schematic view of the BEGe detector

2.2. Monte Carlo Model

MC input file has been constructed for the broad-energy germanium using the data provided by the detector manufacturer. The detector was modeled using the MCNPX code.

Tally F8, which is specific for detector pulse height determination [6], was used to estimate the detector absolute full energy peak efficiency at different gamma energy lines. Detector geometry was modeled as shown in Figure 1. The detector dimensions, its Al-cap, Al-holder and the distance from the detector crystal to the front of the detector cap are those of the manufacturer [7]. Figure 2 shows the characteristics of the simulated BEGe detector as drawn by MCNPX visual editor. Determination of peak efficiency from MCNPX results was performed.

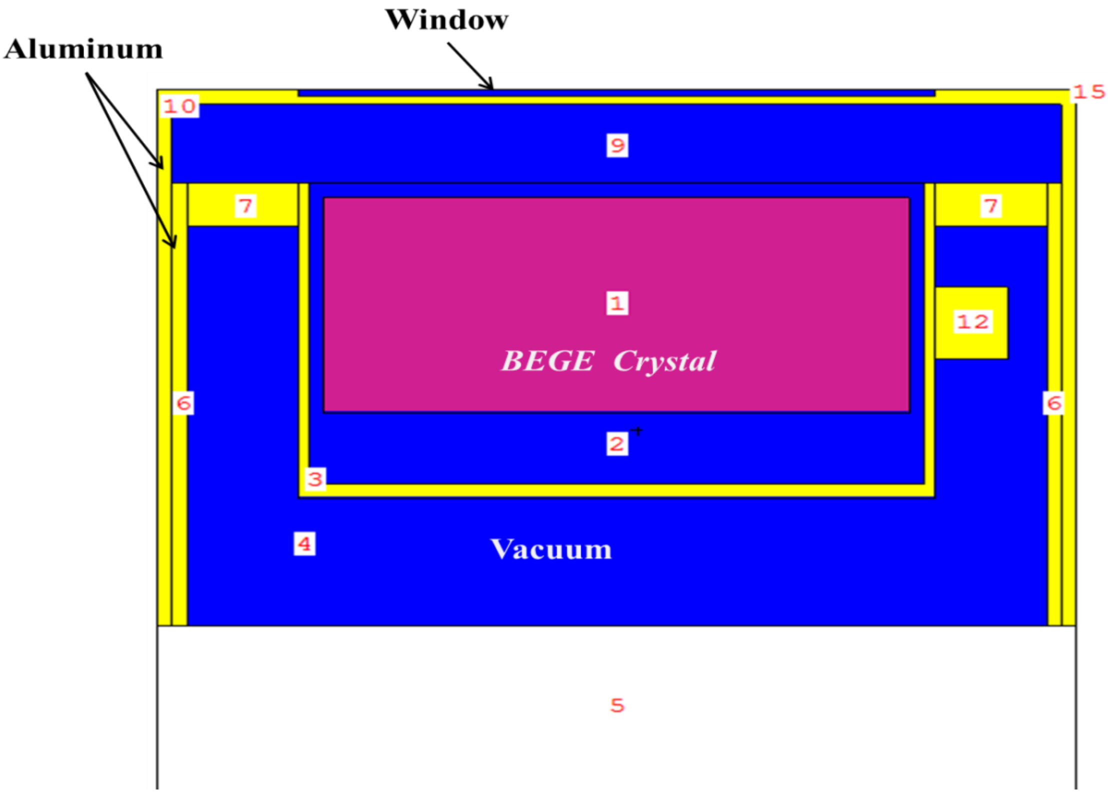

Figure 2: Detector model as drawn by MCNPX visual editor

Histories number (nps card) was chosen to keep the relative standard deviation due to MC calculations less than 1%. MC calculations were performed on a 2.66 GHz processor. The calculation time was approximately 4 minutes (10^7 histories).

2.3. Experimental Setup

Three standard gamma-ray point sources (^{137}Cs, ^{60}Co and ^{57}Co) were used to complete, refine and validate the MCNPX model. The parameters of each source are listed in Table (1).

Table 1: Specification of the certified point sources

Source	Activity (µci)	Production date	E (KeV)	Iγ % [8]
Co-57	10.57	15/7/2007	122.1	85.6
			136.5	10.68
Co-60	4.430		1173.23	99.86
			1332.5	99.98
Cs-137	5.002		661.7	85.1

Initially, ^{57}Co source was placed at 5 cm distance from the detector window. Count rates due to ^{57}Co energy lines were measured using the BEGe detector. Then, the experiment was conducted with source angles of 0^o, 45^o, 90^o, 135^o and 180^o from the detector window. Figure 3 shows the experimental setup arrangement to measure the count rate with different setup configurations.

Both ^{60}Co and ^{137}Cs were placed at a distance equal to 15 cm from the detector window and count rates due to energy lines were measured using the BEGe detector at different angles.

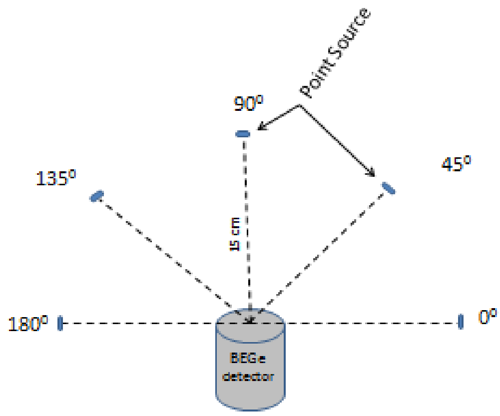

Figure 3: Experimental setup arrangements to measure the count rate with different angles for point source

The absolute photo-peak efficiency, ε, relates the number of detector pulses to the number of gamma-rays emitted by the source and can be specified as follows:

$$\varepsilon = \frac{N}{A\, I_{\gamma}\, t} \qquad (1)$$

Where:

ε is the absolute efficiency value at energy E,

N is the area of the photopeak of energy E, A is the activity (disintegration per second) of the gamma source,

Iγ is the gamma emission probability,

t is the life time of the counting, in second.

To validate the model for nuclear safeguards verification purposes, a set of five cylindrical-shaped Standard Nuclear Material (SNM) samples with different enrichment percentages were used to perform some experimental measurements. The specifications and characteristics of the SNM samples are given in Table (2).

Table 2: Specifications of the certified NM standards

Sample	Enr.%	U-235 mass (g)	U-238 mass (g)
1	4.46	7.572	162.109
2	2.95	5.004	164.677
3	1.94	3.295	166.386
4	0.71	1.208	168.473
5	0.31	0.537	169.144

The samples were placed in front of the detector as shown in Figure 4. The samples-to-Al cap of the detector distances were adjusted and optimized in such a way to obtain the maximum count rate. Meanwhile, the counting losses due to pile up and dead time were minimized.

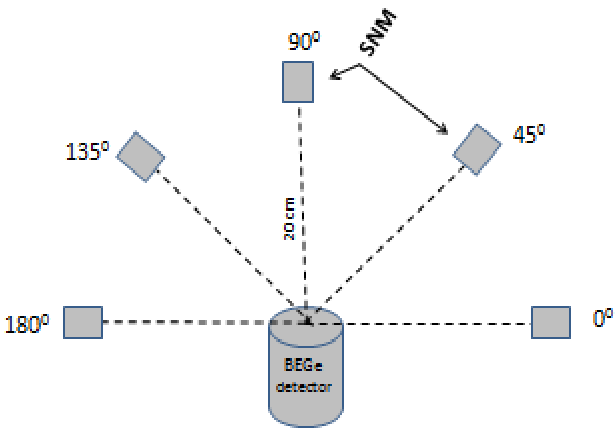

Figure 4: Experimental setup arrangements to measure the count rate with different angle for SNM

Specific isotope mass in a certain radioactive sample measured by a detector could be estimated as follows [3]:

$$m_i = \frac{C_r}{P_i \times F} \qquad (2)$$

Where:

m_i : mass of the assayed isotope "i" [g];

C_r: count rate of the detector of a certain gamma ray due to isotope "*i*" [s^{-1}];

F : fraction of the specific gamma ray absorbed in the active detector material estimated using MC calculations;

P_i: physical constant for the specific gamma energy (specific activity of the assayed isotope and the branching ratio of the gamma ray) [$g.s^{-1}$].

3. Results and discussion

3.1. Activity

Table (3) presents the determined activities estimated for the standard gamma-ray point sources. The measured count rates at gamma energy lines, the calculated absolute full energy peak efficiency at the same energies, and the gamma emission probability of the measured gamma energy line were substituted into Eq. (1) to obtain the activity.

Table 3: Estimated activity for the point sources

point source	Energy Line (KeV)	Degree	Estimated Activity based on MCNPX (kBq)	Reference Activity (kBq)	Different
Cs-137	661.7	0	126.6181	128.465	-1.8469
		45	128.8434		0.3784
		90	129.5622		1.0972
		135	129.3520		0.887
		180	128.7405		0.2755
Co-60	1173.23	0	51.39001	50.9353	0.45471
		45	51.63766		0.70236
		90	51.20632		0.27102
		135	50.83298		-0.10232
		180	51.38088		0.44558
	1332.5	0	51.72663	50.9965	0.73013
		45	52.57596		1.57946
		90	50.56649		-0.43001
		135	49.34335		-1.65315
		180	51.48266		0.48616
Co-57	122.1	0	0.084302	0.0855865	-0.00128
		45	0.083389		-0.0022
		90	0.084657		-0.00093
		135	0.085798		0.000212
		180	0.087644		0.002058

Figure 5 shows the estimated activity for Cs-137 point source. It is clear that the estimated masses using both methods are in agreement within the uncertainties.

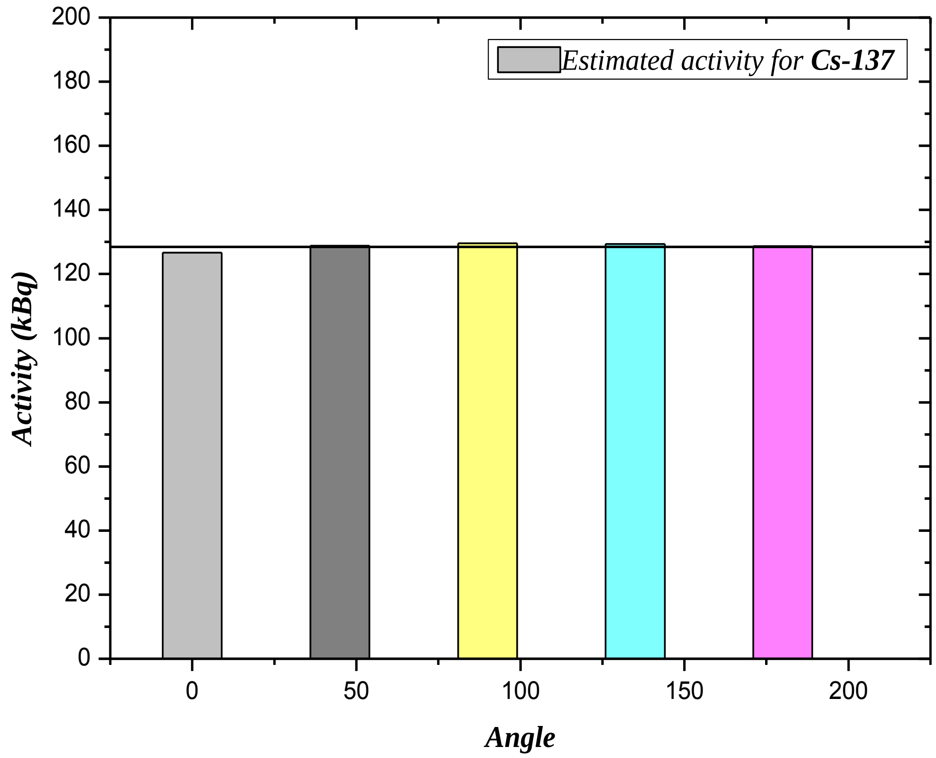

Figure 5: Estimated activity for Cs-137 point source

3.2. ^{235}U Mass Estimation

The measured count rates at 185.7 KeV gamma-ray energy line, the calculated absolute full energy peak efficiency at the same energy line, and the specific activities of the measured gamma energy line were substituted into Eq. (2) to obtain the ^{235}U mass contents in nuclear materials. Table (4) presents the ^{235}U masses estimated.

Table 3:^{235}U masses estimated by MCNPX with the associated uncertainties

Sample	Enr.%	U-235 mass (g)	Estimated U-235 mass (g) ± RSD	Different
1	4.46	7.572	7.565± 0.254	- 0.007
2	2.95	5.004	5.0037± 0.0968	- 0.0003
3	1.94	3.295	3.331± 0.0658	0.036
4	0.71	1.208	1.2033± 0.0262	- 0.0047
5	0.31	0.537	0.549± 0.0164	- 0.012

Figure 6 shows the estimated ^{235}U-mass content values with their uncertainties. It is clear that the estimated masses using both methods are in agreement within the uncertainties.

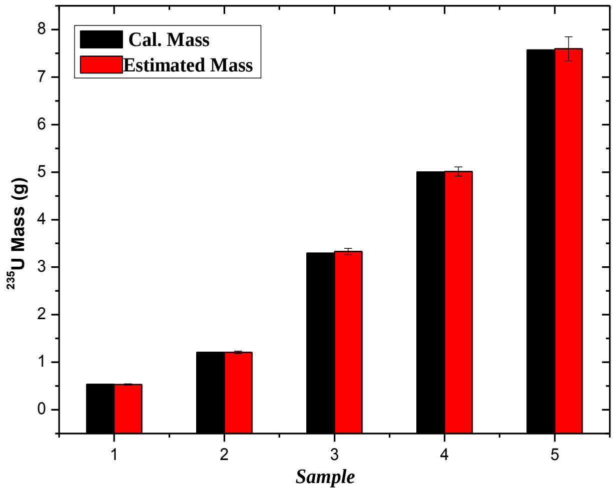

Figure 6: Estimated ^{235}U mass contents using MCNPX model and calculate-based methods

4. Conclusions

The objective of this work is to develop and validate the simulated model for a Broad-Energy Germanium Detector (Model BE2830) based on the physical dimensions of the detector given by the manufacturer. The most sensitive and accurate way to develop the complete MCNP model is by comparison with traceable source measurements so that the activity calculated based on MCNPX efficiencies for three point sources (Co-57, Co-60 and Cs-137) are compared against the certified values. The ^{235}U-mass content values for a set of standard NM samples, with cylindrical shapes, were estimated with their uncertainties. It is clear that the estimated activity and masses using both methods are in agreement within the uncertainties. The obtained results indicate that, this model could be used effectively for nuclear safeguards purposes.

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