Detection Efficiency of a Compton Suppression HPGe Spectrometer Using Experimental Validated Monte Carlo Simulation

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1. Introduction

One of the most important tasks in the neutron activation analysis (NAA) is a detection efficiency calibration, determination of the peak-efficiency curves versus the energy. It must be achieved for various combinations of sample-detector geometries to quantify the radioactivity in the radioactive samples.

In this paper, the MCNPX 2.5 code [1] was adopted to determine the peak-efficiency curves of the HPGe detector of the CSS-Compton Suppression Gamma-ray Spectrometer at KAERI Neutron Activation Analysis laboratory. We also investigated the changes of the efficiency curves by changing a distance between source and detector in the energy range from 40 keV to 1835 keV. The ratios of the experimental efficiencies to simulated ones are consistent within 9%. It shows that our simulation program based on MCNPX code is good enough for later studies on our Compton Suppression Spectrometer (CSS).

2. Experimental

2.1 Detector system

The n-type closed-end HPGe detector was manufactured by EG&G Ortec (Model No. GMX40-76). A relative efficiency and energy resolution at 1.33 MeV are 40% and 1.95 keV, respectively. Peak-to-Compton ratio with ^{60}Co is given as 59:1.

Different from conventional gamma spectrometer for INAA, our system is a CSS including an n-type HPGe. It also includes an annular BGO detector (SAINT-GOBAIN CRYSTALS, serial number A/C 127 YPE 152/BGO), which gives timing signals for anticoincidence or coincidence function in several spectrometer modes. The shapes and dimensions of the n-type HPGe and BGO detectors are shown in Fig. 1 and Fig. 2, respectively.

2.2 MCNPX Simulation

We used MCNPX version 2.5e for efficiency calculations. In these calculations, the F8 tally card was used to get the pulse height of photons. With the variance reduction, electron transport was turned off by using the mode card (mode: p). Furthermore, the Energy Physics Cutoff Card (phys: p) and Cutoffs Card (cut: p) was also applied to edit the MCNPX input file deck. The statistical behavior of the result and assurance of valid confidence intervals for each tally bin were assessed for each run by checking the associated tables in the tally fluctuation chart bin.

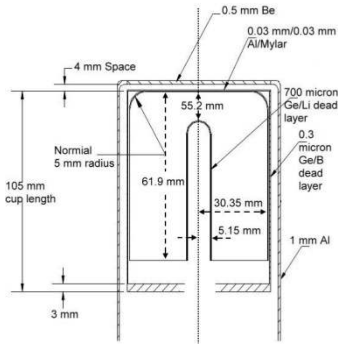

Fig. 1. Cross-section of HPGe detector crystal and aluminum casings. All dimensions are given in millimeters.

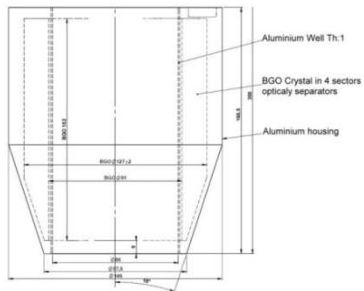

Fig. 2. Cross-section of BGO detector crystal and aluminum casings. All dimensions are given in millimeters.

2.3 Multi-nuclide standard source

Calculated results were experimentally validated by placing a multi-nuclide standard source at several distances along the co-axial detector axis, which are distant enough to avoid coincidence-summing effects. The multi-nuclide source consists of nine sealed radionuclides whose photon energies cover the range under investigation.

2.3 Modeling of the detector and standard source

Based on the dimensions (Section 2.1) provided by the manufacturer, the geometry of detector and source was modeled using the MCNPX code. The shape and magnitude of the peak-efficiency curve depend on the detector dimension, source shape and geometry of this system including BGO detector and lead shielding and so on. Especially, some previous studies by J. Rodenas et al [2] and Huy et al [3] have presented the lower efficiency of the real detector compared with those from the Monte Carlo simulation. The main reason for this difference is believed due to a dead layer thicker than stated by the manufacturer.

The dead layer was estimated to be about twice the nominal value. Moreover, we must design this detector to have a curve-beveled edge at the top corner and a cylindrical inner hole. Model for the detector and source was created in detail by using a Visual Editor tool [4] that is an easy design tool included in the MCNP code. The layout geometry are given in Fig. 3.

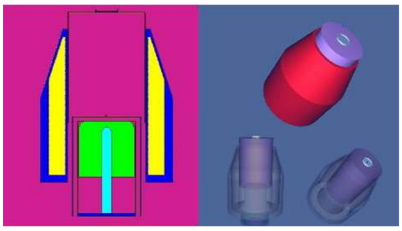

Fig. 3. The layout geometry of the CSS model

3. Result of simulation and measurement

3.1 The simulated spectrum of multi-nuclide source

Based on parameters of the multi-nuclide source was given in the Table 1, we carried out simulation for generating a gamma-ray spectrum in the detector. In the spectrum, twelve full energy absorption peaks are prominent on the continuum region, which are involved in the experimental assessment. To decrease the consume time for calculation, 1 keV photon energy cutoff has been used. The Gauss broaden peak with GEB option was not necessarily in this case. The number of histories has been set to 10^8 histories in order to obtain a relative error less than 1% at every peak centroid. The simulated spectrum with distance 125 mm from multi-nuclide source to detector was shown in Fig 4.

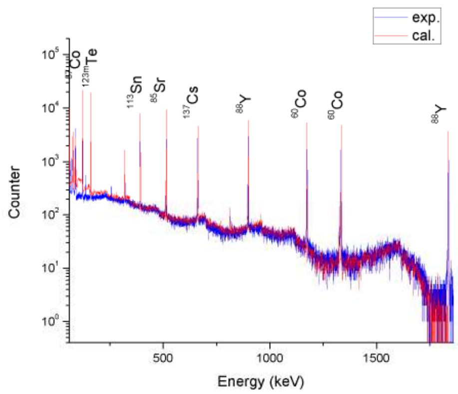

Fig. 4. The comparison of the calculated and experimental spectra of multi-nuclide standard source

3.2 Comparison of calculated and experimental efficiencies for several distances between source and HPGe detector

For achieve the calculated absolute efficiency, the peak area after subtract background was divided by the number of histories gave the absolute peak efficiency. Due to using the multi-nuclide, therefore we have to multiply the absolute efficiency by the factor that is probability for each nuclide in the normalized.

In measure the experimental efficiency, the detector was calibrated using multi-nuclide source that was introduced in section 2.3. The MAESTRO-32 software (Ortec) [5] was used to collect the gamma-ray spectrum.

In the energy region below 100 keV, the discrepancy between calculated and experimental efficiency became quite larger. This discrepancy is typical and is caused by the uncertainty in the detector dimensions and the increase of the dead layer thickness. To correct this problem, the detector geometry was fine-tuned from the manufacture's specification.

The calculated efficiency curves for these distances are represented versus photon energy in the Fig. 5. Efficiency decreases as expected according to the distance between source and detector surface.

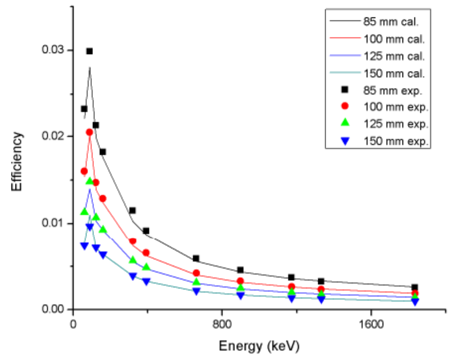

Fig. 5. The calculated and experimental efficiency curves for four distances.

4. Conclusions

MCNPX code is a powerful and useful tool in investigating the absolute efficiency of an HPGe detector. Deviations of about 9% between the calculated efficiency and the experimental one for the detector has obtained for most of the energies investigated. The agreement between simulated and experimental efficiencies is good enough to validate the model. Thus, the simulation method by using MCNPX code may assist authors in constructing the efficiency calibration curves in case experimental method is not available. That is one of some advantages of simulation method.

REFERENCES

[1] D. B. Pelowitz, ed., "MCNPX User's Manual Version 2.5.0," Los Alamos National Laboratory report, in press (February 2005).

[2] J. Rodenas, A. Pascual, I. Zarza, V. Serradell, J. Ortiz, L. Ballesteros, Analysis of the influence of germanium dead layer on detector calibration simulation for environmental radioactive samples using the Monte Carlo method, Nucl. Instr. and Meth, A 496 (2003) 390–399.

[3] Huy N.Q., Binh D.Q., An V.X. (2007), "Study on the increase of inactive germanium layer in a high purity germanium detector after a long time operation applying MCNP code", Nucl. Instrum. Methods, A 573 384-388.

[4] http://www.mcnpvised.com

[5] http://www.ortec-online.com/Solutions/applications-software.aspx