Study on detection geometry and detector shielding for portable PGNAA system using PHITS

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Abstract. Prompt gamma-ray neutron activation analysis (PGNAA) measurements require efficient detectors for gamma-ray detection. Apart from experimental studies, the Monte Carlo (MC) method has become one of the most popular tools in detector studies. The absolute efficiency for a 2 × 2 inch cylindrical Sodium Iodide (NaI) detector has been modelled using the PHITS software and compared with previous studies in literature. In the present work, PHITS code is used for optimization of portable PGNAA system using the validated NaI detector. The detection geometry is optimized by moving the detector along the sample to find the highest intensity of the prompt gamma generated from the sample. Shielding material for the validated NaI detector is also studied to find the best option for the PGNAA system setup. The result shows the optimum distance for detector is on the surface of the sample and around 15 cm from the source. The results specify that this process can be followed to determine the best setup for PGNAA system for a different sample size and detector type. It can be concluded that data from PHITS code is a strong tool not only for efficiency studies but also for optimization of PGNAA system.

1. Introduction

Neutron-capture gamma ray has been introduced as a method of elemental analysis years ago. This method is particularly useful for determining nondestructively elements which absorb neutrons but do not produce radioactive products [1]. The design of prompt gamma neutron activation analysis (PGNAA) has been optimized multiple times for high gamma detection efficiency and low background. Scintillation-based Sodium Iodide, NaI detectors have been used widely for radiation spectroscopy and radioisotope-based applications including medical and industrial areas [2]. These types of detectors have shown reliable results for low level radioactive source measurements due to its higher detection efficiency and suitable features for operating detection at room temperature. However, the accuracy of measurement depends strongly on some detection properties of scintillation detectors such as detection geometry and detection shielding. Experimenting with the geometry is an effective method to evaluate the efficiency of NaI detectors for optimizing portable prompt gamma neutron activation analysis.
For the past years, the Monte Carlo (MC) method has been widely utilized for efficiency determination studies [3-6]. Absolute efficiency is defined as the ratio of the number of gamma rays emitted by the source (in all directions). Some experimental and calculation studies have also reported on detection efficiency determination [1]. This study is performed by calculating the efficiency of 2x2 inch NaI scintillation detector and using the same detector model for geometry and shielding study for portable PGNAA system. Optimisation of photon detection for portable PGNAA system is important as the neutron source use such as Carlsfornium-252 emitting fast neutron thus the intensity of the prompt gamma is very low comparing to reactor based PGNAA. Current systems were using high-purity Germanium detector HPGe and optimisation were ran by experimentally, using this simulation study limit the exposure time for the workers in order to setup the best geometry for the portable PGNAA system measurement. The aim of this study is to simulate the portable PGNAA system and to find the best detection geometry with the best shielding configuration for the system.

2. Methodology

The PGNAA system presented in this paper was simulated and optimized using Particle and Heavy Ion Transport code System (PHITS) software [7]. In this study, PHITS has been used for detector efficiency calculation, geometry design and optimization. First the absolute efficiency of the modeled detector is calculated. Absolute efficiency must be known especially in radioactivity measurements. The definition of the absolute efficiency is shown in:

$$\varepsilon_{abs} = \frac{N_c}{N_g}$$  \hspace{1cm} (1)

where, $N_c$ is defined as the number of counts recorded by detector and $N_g$ is the number of radiations emitted by the source $d$ (all directions such as isotropic source) [1]. Efficiency of the NaI detector is calculated using model shown in figure 1. The efficiency of the detector is measured for different energy of the sealed source such as Na-22, Mn-54, Cs-137 and Co-60.

![Figure 1. 2x2” NaI detector model.](image1.png)

![Figure 2. Energy vs Efficiency graph.](image2.png)

Figure 2 shows the graph of the efficiency for NaI detector simulated using PHITS. The trend for the efficiency graph from this detector model is in agreement with previous study [8][9] where at the higher energy of photon the detection efficiency is reduced. This is due to higher energy gamma-ray photon has high penetrating power and less photon imparted its energy towards the detector.

Using the modeled NaI detector, the geometry of the PGNAA system is modeled according to experimental setup (figure 3). The neutron source is placed inside neutron howitzer so that the neutron is shielded and collimated to only one direction. Energy distribution for neutron source is model according to Watt fission spectrum. The sample is assumed to be silicon, Si cuboid and the space between the neutron source and sample is the polyethylene moderator. The detector crystals arranged 90 degree from the neutron source so that no direct hit by passing neutron. The detector crystal is
design to be shielded by lead so that only gamma rays from sample are detected. The result that we obtained from this simulation is reflecting the capability of the detector chosen to produce the intensity of prompt gamma counted and hence producing the full spectrum of intensity of gamma ray versus the energy of gamma ray.

Figure 3. (a) Portable PGNAA system setup and (b) spectrum of Cf-252 source used

In details of this work, the dimensions used for the moderator is 10 cm in length and has a thick of 8 cm. The target or known as sample is a Silicon (Si) cuboid with a dimension of 10 cm × 10 cm × 30 cm is bombarded with neutron beam with an energy distribution shown in figure 3(b) at 180 degree angle from the neutron source. The axis of cylindrical-shaped detector, NaI is located perpendicular or 90 degree from the position of the neutron generator. For the first objective, position of the detector is varied along the sample surface to find maximum prompt gamma intensity detected. Afterwards, using optimized distance of detector placement, shielding effect on the detector is studied. Two geometries setup is present in this study to find the effect towards the detection of the prompt gamma that is, with lead shield and without lead shielding referring to figure 3(a).

3. Result and Discussion

PGNAA technique has been simulated using PHITS software which applied the Monte-Carlo code to find the probability of neutron activation of photon flux produced to identify element exist in a sample. Thermal neutron flux in prompt-gamma neutron activation analysis setting is the key factor to determine a portable PGNAA performance. In this study 8 cm polyethylene block is simulated as neutron moderator. The bigger the size of the moderator between the source and the sample gives higher thermal neutron produce hence higher gamma ray intensities. This is due to the increasing of thickness space of the moderator in between source and sample. A moderator in the system carried a function to reduce the initial kinetic energy of the free neutron from the source. In this process, the energy released is conserved due to the neutron kinetic energy passing through the material of the moderator. The speed of the neutron released undergoes reduction, as a product of decreasing kinetic energy.

Figure 4. Normalize photon count with varying detector distance from source.
The first objective is to study the effect of distance of detector from the source to observe the intensity of gamma ray produced. The higher yield of gamma ray, the higher the sensitivity of the system in which PGNAA system can optimized to construct a better non-destructive technique for a bigger sample size. Graph in figure 4 shows the highest intensity of photon counting is around 15 cm thus suggesting it is the optimum distance for detector placement. This can be explained as near the source the neutron had minimum interaction with the sample. The interaction of thermal neutron is at maximum until the neutrons penetrate around 15 cm. After 15 cm the graph shows decreasing of photon counting as most of the thermal neutron had imparted their energy to the sample.

The other objective is to study the effect of detector shielding using lead cylinder for gamma-ray shielding and polyethylene block for neutron shielding. Figure 5 shows the result from the simulation with two different geometries. The red spectrum on the graph represents setup with no lead shielding. While the blue spectrum which is plotted on top of red spectrum represent setup with 5 cm lead shielding. Table on figure 5 shows photon flux at the detector for five different peaks for the two geometry setup. Without the present of lead shielding overall photon counting were higher than with the present of lead shielding. This is due to the isotropic nature of detection at the detector crystal. When lead shielding is present, overall photon counting is lower which is not good for prompt gamma detection.

![Figure 5. Prompt gamma spectrum from SiO₂ sample with two different arrangements; Geometry 1: No-lead shielding (red spectrum); Geometry 2: 5cm lead shielding (blue spectrum).](image)

Analysing the result for peak to peak counting we found that lead shielding functioning as shielding for unwanted prompt gamma peak from Hydrogen at 2.23 MeV and Carbon at 4.44 MeV which activated from the moderator. Another observation made is the present of Lead peak at 7.37 MeV when using the lead as detector shielding. This cannot be dismissed as it is the prompt gamma which is directly activated from the shielding material itself. For the interested peak which is Si peak, there is not much difference as the small difference is only due to the collimated nature of the detection when the shielding is present.

4. Conclusion
A portable PGNAA system has been modelled and simulated using PHITS software. The simulated system used Cf-252 neutron source with silicon, Si sample and NaI detector. The prompt gamma spectrum for the sample was successfully detected. The effect of moderator towards the quality of the spectrum detected in the simulation was studied. The result showed that in the present of the moderator, more prompt gamma was detected. This is due to the presence of more thermal neutron in the sample. On the other hand the noise or background signal was higher with the use of moderator. For this, shielding effect was studied and it is found that lead shielding functioning as shielding for unwanted prompt gamma peak from moderator material. On the other hand the present of shielding also activated the prompt gamma from lead. The results also shows the detector needed to arrange
close to the sample so the detection of prompt gamma rays is increased. For an optimum system, the best distance is recorded to be at 15 cm. The simulation have confirm the two parameters studies in this work and it can be used for the setup of real experimental of portable prompt gamma activation analysis (PGNAA) system in future.

5. References

[1] Richard M L 1993 Prompt-Gamma Activation Analysis J. of Research of the National Institute of Standards and Technology 98(1) 127-133

[2] Yi C Y and Hah S H 2012 Monte Carlo calculation of response functions to gamma-ray point sources for a spherical NaI(Tl) detector Appl. Radiation and Isotopes 70(9) 2133–2136

[3] Baccouche S, Al-Azmi D, Karunakara N, and Trabelsi A 2012 Application of the Monte Carlo method for the efficiency calibration of CsI and NaI detectors for gamma-ray measurements from terrestrial samples Applied Radiation and Isotopes, 70(1) 227–232

[4] Moreira M C F, Conti C C and Schirru R 2010 A new NaI(Tl) four-detector layout for field contamination assessment using artificial neural networks and the Monte Carlo method for system calibration Nuc. Instr. and Meth. in Phys. Research Section: A 621(1–3) 302–309

[5] Vegors Jr. S H, Marsden L L and Heath R L 1958 (U.S. Atomic Energy Commission (USAEC) Report 17 IDO-16370).

[6] Miller W F and Snow 1961 W J NaI and CSI efficiencies and photo fractions for gamma-ray detection Nucleonics 19(11) 174

[7] Sato T, Niita K, Matsuda N, Hashimoto S, Iwamoto Y, Noda S, Ogawa T, Iwase H, Nakashima H, Fukahori T, Okumura K, Kai T, Chiba S, Furuta T and Silver L 2013 Particle and heavy ion transport code system PHITS Version 2.52 J. Nucl. Sci. Technol. 50(9) 913-923

[8] Akkurt I, Tekin H O, Mesbahi A 2015 Calculation of detection efficiency for gamma detector using MCNPX Acta Physica Polonica A, 128(28)

[9] Akkurt I, Gunoglu K, Arda S S 2014 Detection efficiency of NaI(Tl) detector in 511-1332keV Energy Range Sci. and Tech. of Nuclear Installation, 5 186798

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