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Estimation of Detection Efficiency for NaI detector using MCNP

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Abstract: *Detector efficiency is one of the main parameters in a radiation detection system. The detection efficiency of a 3" × 3" NaI detector was estimated experimentally and calculated by using MCNP method at energies of 661.7, 834.8, 1173.2, and 1332.5 keV obtained from ^{137}Cs, ^{54}Mn and ^{60}Co radioactive sources at different distances between the detector and source. The results obtained from MCNP calculations are compared with the experimental results. A good agreement was found between calculations and experimental values with average differences from -1.49 % to 1.87 %.*

Keywords: *NaI detector, efficiency, Experimental, MCNP.*

I. Introduction

Gamma detection techniques are widely used in γ- spectrometry for nuclear physics studies [1], especially scintillation detectors which are widely used in different fields [2].

The detection systems most widely used for γ- spectrometry are NaI(Tl) and HPGe based detectors [3]. One of the most important characteristics of a detector is the efficiency of the system. An important advantage of NaI(Tl) is its high detection efficiency[4].

The absolute efficiency of the gamma detector system used at Turkish Accelerator and Radiation Laboratory at Ankara (TARLA) was simulated using MCNPX code [5]. The results obtained for NaI(Tl) detector system were compared with the experimental results. A good agreement was found between calculation and experimental values. The Monte Carlo method has been used to calculate the photon detection efficiency and energy resolution curves for a 1.5" X 1" NaI(Tl) scintillation detector by C.M. Salgado et al [6]. The detector has been exposed to gamma rays in the energy range from 20 keV to 662 keV. The calculated results showed good agreement with the experimental data.

An efficient Monte Carlo computer program for simulation and calculation of the total and full energy peak efficiency (absolute and intrinsic) of a cylindrical NaI(Tl) detector has been described by A. B. Kadhem et al [7]. They concluded that the results show quite well agreement with experimental data and with other calculations within error rate less than 2% and the results can be used in γ- spectroscopy and determining the activity of sources.

The variation of the intrinsic efficiency of the NaI(Tl) detector against the source-detector distance has been calculated by A. A. Mowlavi et al [8] for different gamma ray energies by using MCNP code. The intrinsic efficiency depends not only on energy of photons but also on the geometry and configuration of source and detector. In this study, the absolute efficiency of Nal detector has been estimated experimentally and calculated by using the MCNP5 code for γ – rays of energies at 661.7, 834.8, 1173.2 and 1332.5 keV. The radioactive point sources that emitted the γ- rays were placed at different distances between the detector and source. The calculated results have been compared with experimental results.

II. Experimental work

Radiation sources (^{137}Cs , ^{54}Mn and ^{60}Co) that give 661.7, 834.8, 1173.2, and 1332.5 keV gamma ray energy respectively were placed at eleven different distances (0, 1,2, 3,4, 5,10, 15, 20, 25, and 30 cm) from the face of detector. Each measurement has been done for a period of 600 sec to obtain good statistics in the evaluation of each gamma peak. The data sheet states values of half-lives, photon energies and photon emission probabilities per decay for the all radionuclides used in this work are listed in Table (1).

Table (1). Half-life, photon energy and photon emission probability per decay for the used radionuclides [9].

Nuclide	Half-life (days)	Energy (keV)	Emission Probability %
Cs- 137	10975.55	661.7	85.1
Mn- 54	312.3	834.8	99.98
Co- 60	1924.061	1173.2	99.97
Co- 60	1924.061	1332.5	99.98

The gamma ray spectrometer used in this work is a portable scintillation detector assembly based on a Miniature Multi Channel Analyzer model (MCA-166) with a NaI (Tl) detector model (12S12-3.VD.PA.003) and serial number (2518.05.09). As provided by the manufactures, the detector has a NaI (Tl) crystal with dimensions (76.2 x76.2 mm) and an Aluminum housing of 1mm thickness. The experimental setup is shown in Fig. 1

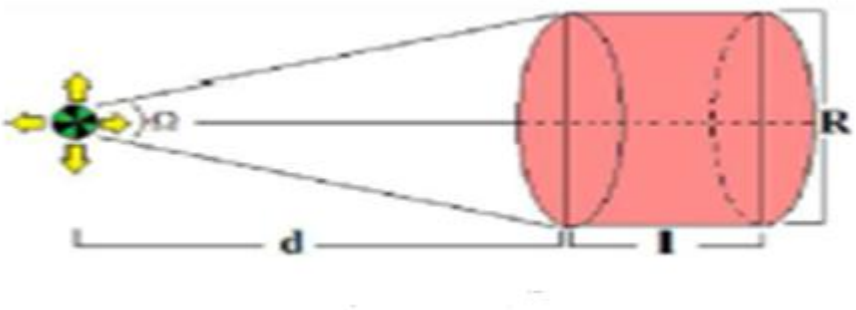

Fig.1 Representation the experimental setup. [(S) the point source, (d) source- detector distance, (l) detector length and (R) the detector diameter].

III. MCNP simulation

Determination of detector efficiency is very important in various scientific. Because the experimental work is tedious and even difficult for ex-tended sources, many research work have been focused on the development of computational techniques to estimate the detector efficiency. There are three methods used in this field, the semi-empirical, the Monte Carlo and the direct mathematical methods.

The MCNP is a computerized calculation technique that is useful especially to solve complicated three-dimensional problems. The input file created by the user is subsequently read by MCNP. This file comprises information about the material specifications, the characteristic of geometry, the location and features of the photon, electron or neutron source, the kind of answers or tallies desired and any variance reduction methods used to increase efficiency [10]. In this work, MCNP5 was used to simulate NaI detector efficiency. The efficiency was obtained by using F8 tally. F8 is the pulse height tally without any variance reduction. The detector geometry definition about cells and surfaces are to be given in MCNP5 input file.

The computed efficiency of the detector could vary depending on several factors such as distance between source and detector and the energy of gamma ray line for the different radiation sources (^{137}Cs, ^{54}Mn and ^{60}Co) placed at eleven different distances from the detector as in the experimental work at 661.7, 834.848, 1173.2, and 1332.5 keV gamma ray energies. Simulations have been done for the period of suitable time for every energy line at all of the specified distances in the MCNP calculation to reduce error rate.

IV. Results and discussion

The absolute efficiency of the NaI has been estimated experimentally for each gamma ray energy emitted from ^{137}Cs , ^{54}Mn and ^{60}Co radioactive isotopes. As the detection efficiency of the NaI detector may vary with the distance source- to- the detector face, the efficiency values have been obtained for 11 different distances from the detector at 4 different energies.

MCNP5 input files have to contain detailed characteristics of the detector as well as the experimental setup and configuration. The MCNP5 code has been used for modeling the detector response, since it contains a tally, F8, which is specific for detector pulse height estimation. The fraction of gamma-rays with certain energy absorbed in the detector active volume represents its absolute full energy peak efficiency at that energy.

MCNP5 input files are designed for every energy line at all measured distances to perform the calculations. The number of histories was selected to keep the relative standard deviation due to MCNP calculations less than 2%. Each run of calculation was performed using 10×10^6 number of histories.

The obtained results from experimental work have been compared with the calculation results obtained from using MCNP simulation. The comparisons of four energies have been displayed in Fig.2

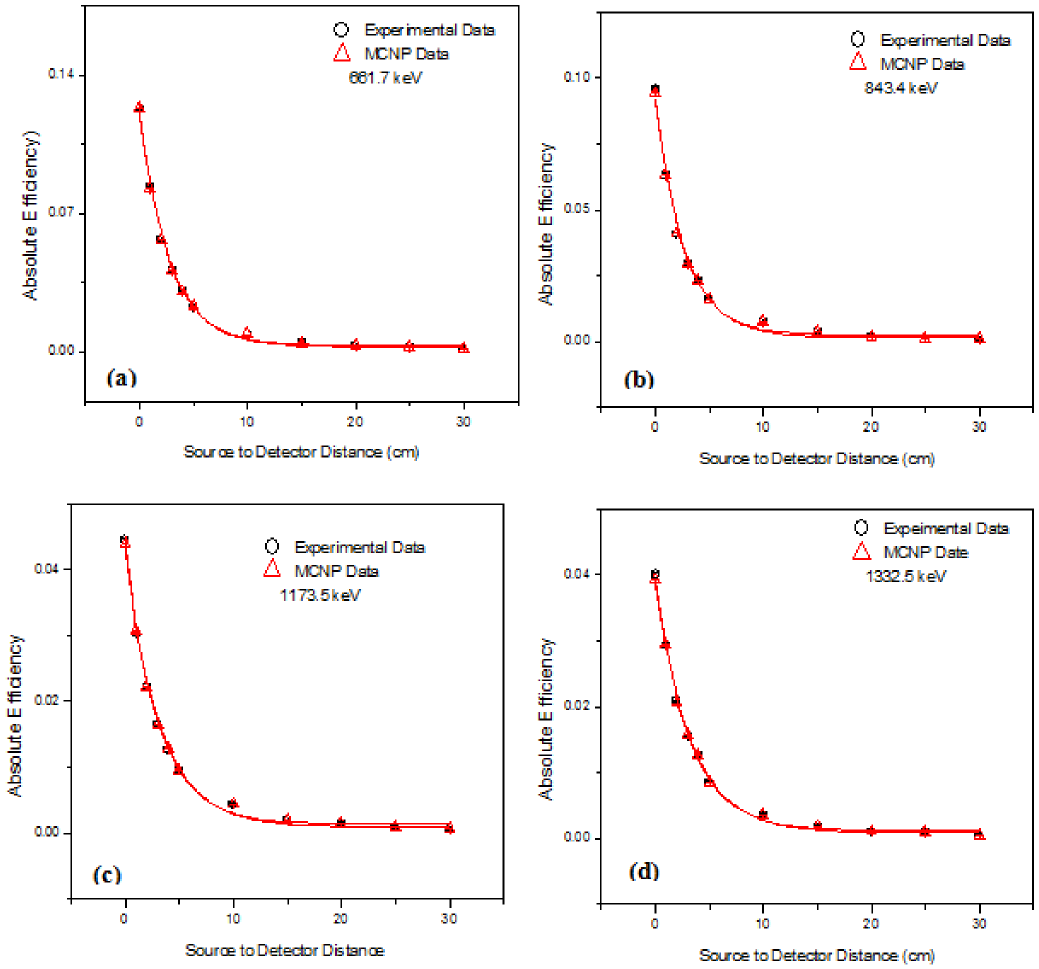

Figure 2. Absolute efficiency at different source to doctor distance in comparison with MCNP value

It can be seen from Figure 1 (a) that, the detection efficiency begin from experimental value (0.12282) and calculation value (0.12355) and decrease to become (0.00181) as experimental value and (0.00178) as calculation value in the ending.

The experimental value of the detection efficiency was (0.09559) and the calculation value was (0.09466) in the beginning from figure 1 (b), while the ending experimental value equal to (9.59065E-4) and the calculation value equal to (9.6824E-4).

From figure 1 (b and c), the values of detection efficiency begin from (0.04437 and 0.04004) as experimental values and (0.04398 and 0.03929) as calculation values respectively. The ending values of the experimental detection efficiency were (5.64055E-4 and 5.63999E-4) while the calculation values were (5.63463E-4 and 5.66160E-4) respectively.

It can be seen that, the detection efficiency has decreased with the increasing distance from detector face and the efficiency of the detector is higher at low source energies and decreases as the energy increases.

In this study it was found that, MCNP simulation method is very useful method for gamma ray detector modeling and efficiency calculations. It was found from this study that MCNP gives very consistent results by comparing with the experimental results. This would encourage to extend this study for other NM samples of different geometry without using reference NM samples for calibrating the measuring system to estimate the enrichment of sample for safeguards applications.

V. Conclusion

In the radiation measurement, one of the most important characteristics of a detector is the efficiency of the detector. The variation of the absolute efficiency of the NaI detector against the source-detector distance has been measured experimentally and calculated by using MCNP code for different gamma ray energies. The absolute efficiency for a NaI scintillation detector has been estimated experimentally and calculated by using MCNP simulation. The detection efficiency depending on gamma ray energy and also source- detector distance has been investigated. A good agreement was observed between experimental and calculated results by using

MCNP method. The method could also be applied to other detector systems in a simple manner since detector–source distance, detector and source dimensions and energy are all controllable input parameters.

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