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A Monte Carlo Simulation for Estimating of the Flux in a Novel Neutron Activation System using ^{252}Cf Source

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Abstract: *A new neutron activation system design for nuclear material verification is being established. Neutron activation analysis is a nuclear technique for the analysis of sample composition whereby the sample is irradiated in a neutron field and composition is determined by identifying characteristic induced gamma radiation emitted by the activation products. Calculation of neutron flux at the irradiation positions of a neutron activation system is essential for the design and evaluation of experiments involving material irradiations. A computational model of the new neutron activation system was developed using the Monte Carlo code MCNP5. The model included detailed geometrical representation of the new neutron activation system. The MCNP5 model was applied to determine the best location for the irradiation positions. A ^{252}Cf neutron source with 12μCi of activity was used in the calculations. The MCNP5 estimated neutron fluxes were compared with physical measurements using activation foils method. A good agreement between calculated and experimental results was observed and discussed.*

Keywords: *Neutron Activation, Neutron flux, Irradiation positions, Cf-252 neutron source, Monte Carlo.*

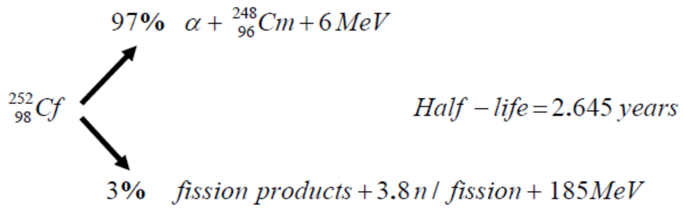
I. Introduction

Among more than 150 different instrumental analytical methods, the neutron based techniques have kept their leading positions in solving special problems and are used in many different applications [1- 4].

Neutron flux is a very important data for neutron source. Many studies focus their eyes on neutron flux [5, 6]. Neutron detection by foil activation is based on the formation of a radioisotope by neutron capture, and subsequent counting of the radiation emitted by that radioisotope. Because neutrons are difficult to detect, neutron activation is used to produce γ -rays and β-particles, which are proportional to the neutron flux and are easier to detect [7].

The initial step in neutron activation analysis is irradiating a sample with neutrons in a nuclear reactor or sometimes in other neutron sources. A common source of neutrons is the spontaneous fission of ^{252}Cf. Neutrons are emitted directly in the fission process, at a rate of about 3.8 neutrons per fission. The effective half-life of ^{252}Cf is 2.65 years. Spontaneous fission occurs in about 3% of the decays, α-decay accounts for the rest, and the neutron production rate is $2.3\text{x}10^{12}$ neutrons/s per gram of ^{252}Cf. The neutron energies are characteristic of fission, a continuous Maxwellian distribution with an average energy of 2 MeV. A summary of radioactive decay properties is presented below:

Half-life (effective):	2.65 years
Half-life (spontaneous fission):	85.5 years
Half-life (α-decay):	2.73 years
Neutron emission:	$2.3\cdot10^{12}$ neutrons/s per gram
Average neutron energy:	2 MeV
Specific activity:	$2.0\cdot10^{13}$ Bq per gram



Spontaneous fission neutron sources have many useful applications due to their simplicity, reliability and small size. ^{252}Cf is convenient since it may provide moderate neutron intensity over a sufficiently long half-life. Low rate of heat emission, γ-radiation, and helium production allow fabrication of simple and small ^{252}Cf neutron sources. Fig.1. shows an energy spectrum of the neutrons emitted during the spontaneous fission of ^{252}Cf.The mean energy is 2 MeV. The spectrum depends on many variables such as fission fragment excitation energy and average total fission energy release, but can be approximated by a Maxwellian distribution N(E) where N(E) varies as $(E)^{1/2}$ exp (-E/l.43 MeV). This spectrum is, proportional to $(E)^{1/2}$ at low energies; it then falls exponentially at high energies [8].

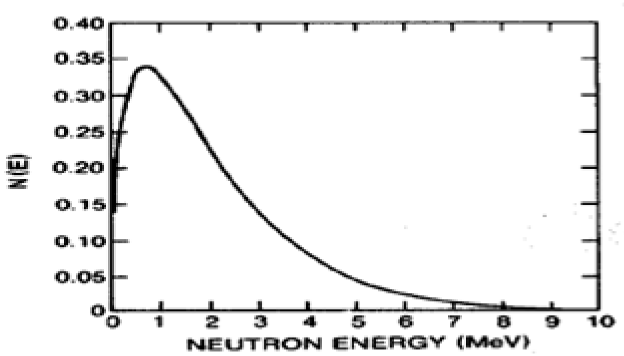


Fig.1. Prompt neutron spectrum from the spontaneous fission a Maxwellian distribution with "temperature" T = 1.43MeV.

Modern Monte Carlo algorithms are all run on computers. The software package currently employed the most for such calculations is MCNP, which stands for Monte Carlo N-Particle. It is capable of tracking 34 particle types of nucleons, photons, light ions, and 2000+ heavy ions at nearly all energies. It uses standard evaluated data libraries mixed with physics models where libraries are not available.

A new neutron activation cell design for nuclear material verification is going to be established. It will compose of a paraffin wax sphere with a 20 cm radius surrounded by a 1 cm thickness of lead. Fig.2 show simplified scheme of the system geometry.

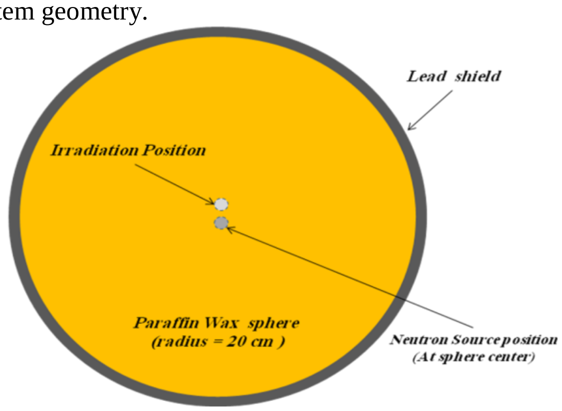


Fig.2 Simplified scheme of the system geometry

The objective of this work is to determine the thermal neutron flux at a certain distance from the neutron source (^{252}Cf) inside the neutron activation cell to choose the best position for irradiation. This is done by using MCNP5 code simulation and to validate the precision of the results calculated with this model, it compared with those obtained by physical measurement using the foil activation method (using gold foil).

II. Materials and methods

Our design for a new neutron activation cell is based on the various considerations of reflection and moderating effects. Paraffin wax was used as reflector and moderator which is shaped into a hemisphere with a radius of 20 cm and surrounded by a 1 cm thickness of lead. To calculate the flux, the neutron activation cell geometry simulated using MCNP5 code and the irradiation cavity was simulated at various distances ranging from 0 to 20 cm from the source position. Fig.3 show a schematic diagram for the system geometry used in MCNP5 model.

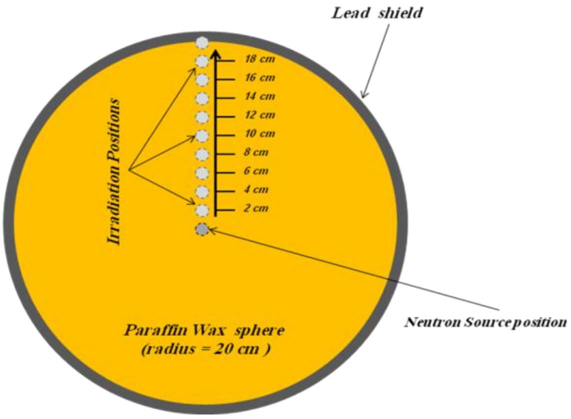


Fig.3 Show a schematic diagram for the system geometry used in MCNP5 model.

We have used ^{252}Cf spontaneous fission neutron source for the simulation, where the neutron energy spectrum represents a fission spectrum that can be described by Watt distribution. The formula is given below [9]:

$$\mathbf{f\,(E) = C\,exp(-E/a)\,sinh(bE)^{1/2}} \quad \ldots\ldots\ldots\ldots\ (1)$$

E is the neutron energy, a= 1.025, b=2.926, c=0.3. The fission neutron energy spectrum of ^{252}Cf can be obtained by MCNP5 simulation.

In the Monte Carlo simulation, the tally F4: N was used for the neutron flux calculations. This tally allows calculations of the flux average over a cell (particles/cm2).

The energy spectrum of a neutron can be broken down into four main regions: thermal, intermediate, fast, and relativistic. Thermal neutrons have an energy of less than 0.5 eV, with the most probable energy being 0.025 eV. Intermediate neutrons have energies between 0.5 eV and 10.0 keV. The energy of a fast neutrons lies between 10.0 keV and 10.0 MeV. Finally, relativistic neutrons have energies above 10.0 MeV. In general, the dividing line between "fast" and "slow neutrons is approximately 0.5 eV, or about the energy of the abrupt drop in absorption cross section in cadmium, also known as the cadmium cutoff. The energy ranges considered in this model were: (a) thermal below the cadmium cut-off energy (0.5 eV) and (b) fast neutrons (above 0.5 eV) cadmium cut-off energy. For each simulation, 10,000,000 histories were run in order to obtain a sufficiently low degree of error and pass all statistical tests.

Neutron activation foils are a widely used and extremely reliable type of neutron detector. Depending upon the composition of the foil, they can be used over a wide variety of both flux levels and neutron energies. In order to validate the precision of the results calculated with this model neutron flux verification measurements were performed using gold foils with and without cadmium covers to obtain the thermal and epithermal neutron flux distribution in irradiation position. The foils were placed at the position of measurement. For the neutron flux measurements the foil is irradiated and counted at a calibrated high-purity germanium semiconductor detector. The activity of the foils was determined and corrected for absolute detector efficiency and decay.

When a thin foil of the target atoms is placed in a neutron flux its activity increases exponentially until the activity reaches a saturation value. If the activity does not reach saturation, the activity, A_0, of the foil after the irradiation time (t_i) is given by

$$A_0 = \lambda N(t) = A_s\left(1 - e^{-\lambda t_i}\right)$$

where As is the saturation activity and λ is the decay constant. If a foil is irradiated nine times of its half life, with 0.4% error, the activity will be equal to saturation activity. Neutron flux in units of neutrons $\cdot cm^{-2}\cdot s^{-1}$ was obtained [10], which can be calculated by

$$\phi = \frac{P_k\, A_i\, \lambda_{i+1}}{\varepsilon(E_k) e_k\ \ a_i\, N_A\, \sigma_i\, m(1 - e^{-\lambda_{i+1} t_0})(e^{-\lambda_{i+1} t_1} - \ e^{-\lambda_{i+1} t_2})}$$

Where P_k = net number of counts under the peak

$\varepsilon(E_k)$ = absolute full-energy peak detector efficiency at energy $\mathbf{E_k}$

e_k = probability that a photon of energy $\mathbf{E_k}$ is emitted per decay of the isotope (also known as intensity of this gamma)

$t_2 - t_1$ = counting time = T

N_A = the Avogadro's number
m = the mass of the specimen and
A_i = the atomic mass.

III. Results and discussion

Fig. 4 shows the Monte Carlo calculation results for the neutron flux per each neutron emitted by the source at each distance from the neutron source position. As shown in Fig.4, we observe that the flux is reduced with the increase in thickness of paraffin,

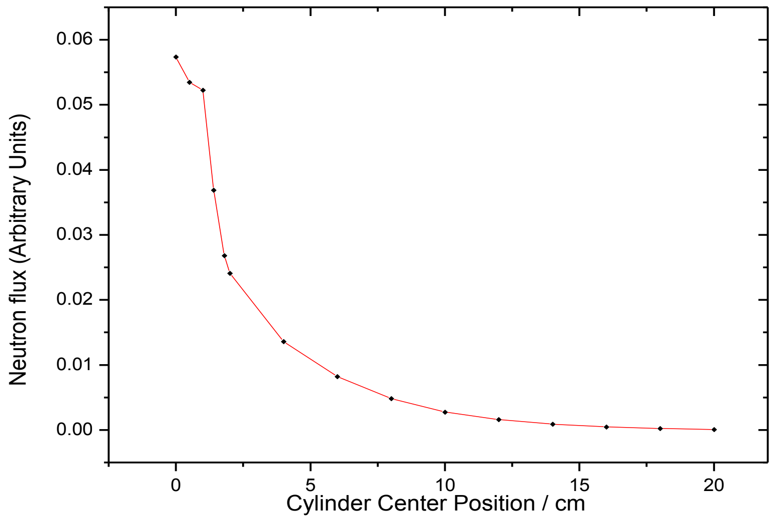


Fig.4. Neutron flux at different distances from the neutron source

Fig. 5 & 6 shows the Monte Carlo calculation results for the thermal neutron flux and fast neutron flux per each neutron emitted by the source at each distance from the neutron source position.

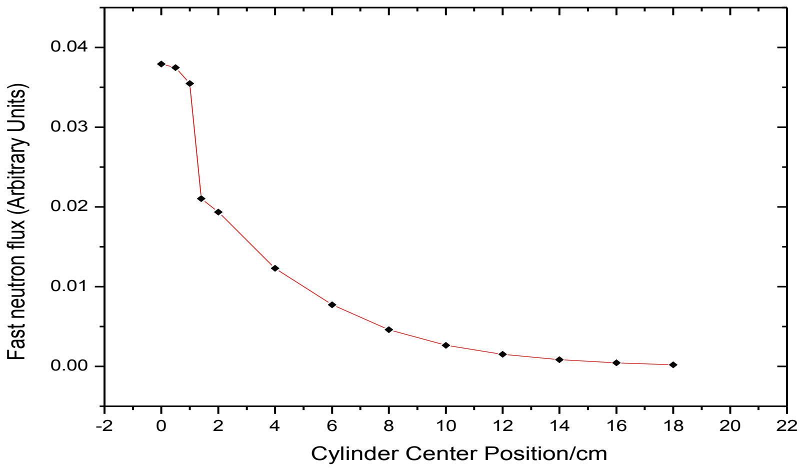


Fig.5. Fast neutron flux at different distances from the neutron source

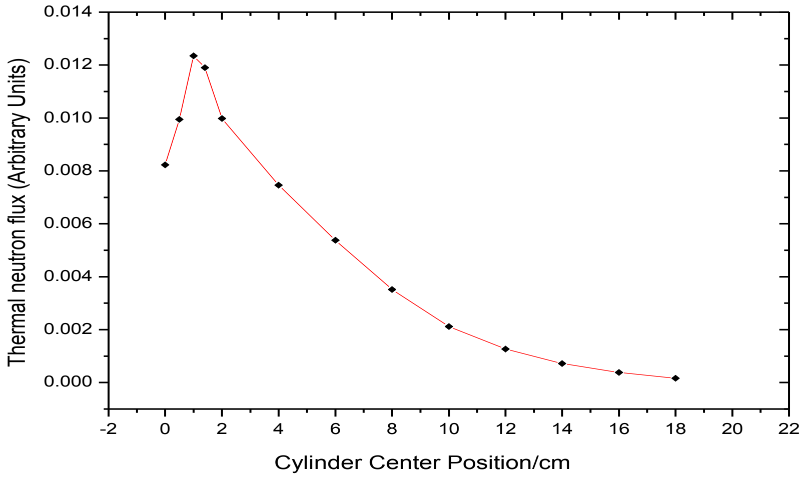


Fig.6. Thermal neutron flux at different distances from the neutron source

It is clear from Fig. 5 & 6 that with an increase in the paraffin thickness, the thermal neutron flux increases, while fast neutron flux decreases. The thermal neutron flux rate initially increased with the moderator radius with a maximum value achieved for 1-2cm thickness moderator followed by a subsequent decrease in the thermal neutron flux rate. The initial increase in the thermal neutrons yield may be due to the neutrons scattering cross section, which is greater than the absorption cross section. The subsequent decrease in thermal neutron yield with further increase in the moderator thickness may be attributed to the absorption cross section. The same reason leads to the fast neutrons flux rate reducing linearly with the increase of the paraffin thickness. It can be seen in Fig. 6 that an appropriate amount of thermal and epithermal neutrons were present in 2 cm locations.

The fission neutron energy spectrum of ^{252}Cf can be obtained by MCNP5 simulation. Fig. 7 shows an energy spectrum of the neutrons for the source position calculated using MCNP5 simulation model.

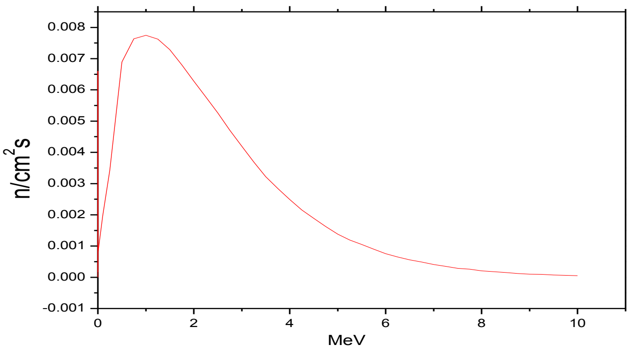


Fig. 7 The Flux vs. energy bin for the source calculated using MCNP5

The gold (Au) foils were chosen to be irradiated at the 2cm positions because it had been determined that an appropriate amount of thermal and epithermal neutrons were present in this location that would result in a measurable amount of activity after a irradiation period. The observed activity after the irradiation of Au is caused by a mixture of thermal and resonance energy neutrons. These contributions can be separated differentially through manipulation of the cadmium cutoff. A cadmium cover was used as a selective filter to block neutrons of energy <0.5 eV, but still allow higher energy neutrons to pass with little moderation. A good agreement between calculated and experimental results was observed.

IV. Conclusion

Several Monte Carlo calculations were carried out for the flux estimation and evaluation using MCNP5 code. The MCNP5 calculations were performed at various distances from the neutron source position inside the system ranging from 0 up to 20 cm in a paraffin wax sphere. A ^{252}Cf neutron source with 12µCi of activity was used to achieve the calculations. The results show that the flux level near to the source position is the best position.

In order to validate the precision of the results calculated with this model. The foil activation method was used. Neutron flux was determined in the irradiation cavity by counting the emitted gamma rays of gold foils with HPGe detector. The calculations performed by means of the Monte Carlo radiation transport code MCNP5 are compared with that carried out experimentally. The MCNP results show agreement with the measurements.

References

[1]. Yousif-Ali et al., 1995. Investigations of neutron fields used in elemental analysis of bulk samples. Nucl. Geophysics, 9. 203-217.

[2]. SPIE, 1996. Proc. Int. Conf. on Neutrons in Research and Industry, 9-15 June, Crete, SPIE Vol. 2867, Section 3 Nonintrusive Testing and Characteriyation p.173-242.

[3]. IAEA 1999. Application of nuclear techniques to anti-personnellandmines identification, IAEA/PS/RC-799, Vienna, Austria

[4]. Jonah S. A. 2001. Development of neutron-based Facilities for bulk H analysis of energy minerals, Bulk Hydrogen Analysis Using Neutrons, FI-RC-655.3, p. 104-109. IAEA, Vienna, Austria.

[5]. Yu V Stenkin et. al , Thermal neutron flux produced by EAS at various altitudes[J]. Chinese physics C.2013, 37(1): 015001-015001.

[6]. WANG Wen-Xin, et. al, Study on spatial resolution of micromegas as a neutron detector under condition of high neutron flux and γ ray background[J]. Chinese physics C.2009,33(2): 110-113.

[7]. S Vaughn, Investigation of a passive, temporal, neutron monitoring system that function within the confines of start I, M.S. thesis (Air Force Institute of Technology, 2003)

[8]. Doug Reilly, Norbert Ensslin, and Hastings Smith, Jr."Passive Nondestructive Assay of Nuclear Materials"NuREG/cR-5550 LA-UR-90-732 (1991)

[9]. MCNP — A General Monte Carlo N-Particle Transport Code, Version 5 manual- Volume I: Overview and Theory- APPENDIX H - Constants for the Watt Fission Spectrum.

[10]. PARISA AKHLAGHI et al, 'The measurements of thermal neutron flux distribution in a paraffin phantom', PRAMANA journal of physics, Indian Academy of Sciences Vol. 80, No. 5, May 2013