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Design of a novel nondestructive portable mobile neutron activation system

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ABSTRACT

A new non-destructive portable mobile system was designed to be used for neutron activation assay according to the Safeguards purposes. The sample is irradiated in a neutron field and its composition is determined by identifying the characteristic induced gamma radiation emitted by the fission and activation products. For the design of a neutron activation system, many aspects must be considered such as choosing the best materials to assure the protection against radiation and estimating the thickness required to shield the neutron source based on its flux or strength. The polyethylene and paraffin wax were selected as neutron shielding material, lead being chosen as gamma shielding material. Several calculations were performed using the Monte Carlo code MCNP5 to estimate the required shield thickness and distribution of thermal neutron flux at different irradiation position inside the shield. The available ^{252}Cf neutron source with 12 µCi of activity was used. The novel neutron activation analysis system was proposed, designed and fabricated to meet the maximum flexibility for neutron activation analysis, at minimum cost and with adequate optimized shielding and portability.

1. Introduction

Neutron Activation Analysis (NAA) is a non-destructive analytical method, capable of rapid and simultaneous multielement analysis involving the entire Periodic Table, from hydrogen to uranium [1].

Neutron activation analysis (NAA) is a nuclear process used for the determination of the elements concentration in materials. It is very useful as sensitive analytical technique for performing both qualitative and quantitative multi-elemental analysis [2]. This method is based on conversion of stable atomic nuclei into radioactive nuclei by irradiation with neutrons and subsequent detection of the radiation emitted by the radioactive nuclei and its identification. The basic essentials requirements to carry out an analysis of samples by NAA are:

- Available source of neutrons.
- Instrumentation suitable for detecting gamma rays.
- Detailed knowledge of the neutrons reactions and an appropriate soft to process the experimental gamma-ray spectra as well as to extract both energies and intensities.

Radiation shielding purpose is to place a shielding material between the ionizing radiations source and the worker or the environment [3].

Neutron source is an important element in neutron activation technique. The most common sources of neutrons are:

1. (α,n) AmLi neutron sources,
2. ^{252}Cf neutron source (Spontaneous fission neutron source).

Isotopic (α, n) neutron source has the disadvantage of higher gamma-ray dose rates than that of ^{252}Cf. ^{252}Cf is the most common spontaneous fission neutron source. Its half-life of 2.65 years is long enough to be reasonably convenient, and the ^{252}Cf isotope is one of the most widely produced of all the transuranics. The neutron yield is 0.116 n/s per Bq, where the activity is the combined alpha and spontaneous fission decay rate [4].

Compared with the other isotopic neutron sources, ^{252}Cf sources involve very small amounts of active material (normally of the order of micrograms) and can therefore be made in very small sizes, depending only by the encapsulation requirements [4].

MCNP is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code. It can be used in several transport modes: neutron only, photon only, electron only, combined neutron/photon transport where the photons are produced by neutron interactions, neutron/photon/electron, photon/electron, or electron/photon [5].

The main target of the present work was to design a portable mobile neutron activation system using isotopic ^{252}Cf neutron source with maximum flexibility for neutron activation analysis of safeguardability purpose, at minimum cost and with adequate shielding and portability.

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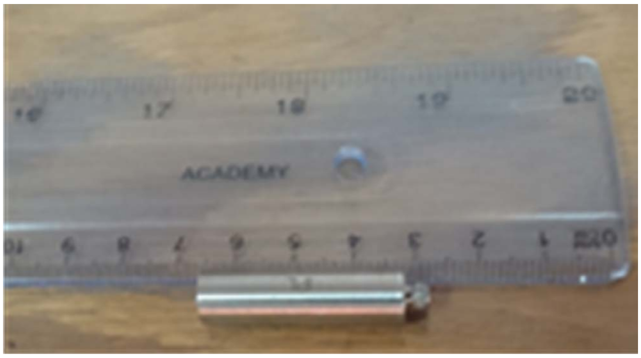

Fig. 1. ^{252}Cf neutron source.

2. Experimental

For the design of a neutron activation system, it is highly desired to choose the best materials for assuring the protection against radiations and to estimate the thickness required for shielding the neutron source based on its flux or strength. The transmission of directly and indirectly ionizing radiation through matter and its interaction with the matter is fundamental for the radiation shielding design and analysis. Design and analysis are effectively the two sides of the same coin. In design, the source intensity and permissible radiation dose or dose rate at some location are specified, and the task is to determine the type and configuration of shielding that is needed.

2.1. Neutron source

The available ^{252}Cf neutron source with 12 μCi of activity (as of March 1st, 2010) was used; it is shown in Fig. 1.

Table 1 shows the characteristics and the isotopic composition of the ^{252}Cf neutron source [6].

2.2. Shielding materials

Shielding should be light enough to allow for easy movement of the device from one location to other. The polyethylene and paraffin wax were selected as neutron shielding material due to its enrichment in hydrogen and easy availability.

Polyethylene was used to fabricate the system internal part (system core) because of several features, as follows: it is enriched with hydrogen; is in solid state; is easy to machine; its neutron scattering properties are well-known; has a wide availability.

Polyethylene is available in different shapes (plate, cylinder ,…) from the local market. The cylindrical shape was selected for our work purposes, since it was very suitable for our design as long as the system core looks like a motor, containing two parts (rotor and stator).

The maximum polyethylene cylinders radius available from the local market was only 10 cm. Therefore the paraffin wax has been selected to externally surround the polyethylene parts in order to get the required thickness for shielding; the paraffin wax it is enriched with hydrogen and easily available.

Table 1
Characteristics of a ^{252}Cf neutron source.

Nuclide	^{252}Cf		
Activity	12 μCi (144 kBq) (as of March 1st, 2010)		
Capsule type	*A3014*		
Active diameter/mass	*1.57 mm (0.062")*		
Cover	*Stainless steel*		
Half life	*2.645 year*		
Isotopic composition	Nuclide	Mass%	Activity%
	Cf-249	*9.748*	*0.145*
	Cf-250	*30.286*	*11.944*
	Cf-251	*14.765*	*0.0848*
	Cf-252	*45.202*	*87.826*
Recommended working life	*15 years*		

Table 2
Shielding materials specifications preparation.

Material	Density (g/cm^3)	Chemical formula
Polyethylene	0.94	$(C_2H_4)_n$
Paraffin wax	0.9	C_nH_{2n+2}
Lead	11.38	Pb

Gamma-ray shielding materials should be characterized by high density and atomic numbers so that they have a high total linear attenuation coefficient and a high photoelectric absorption probability. In this respect, the most appropriate shielding material is the lead - it is readily available, has a density of 11.35 g/cm^3 and an atomic number of 82, and is quite inexpensive. Lead can be molded into many shapes; however, because of its high ductility it cannot be machined easily. Its large atomic number leads to reduced slowing down of neutrons per scattering. It has an (n, 2n) threshold of about 11 MeV [7].

The polyethylene, paraffin wax and lead specifications are presented in Table 2.

2.3. Theoretical calculation for shielding of the system

To estimate the required shield thickness, the MCNP5 code was used, neutron and gamma dose rates from the neutron source being calculated. For simplification, it was postulated that the source used in the code transport simulation symmetrically emits radiation and it was treated in the model as a point source. The energy of a neutron emitted from ^{252}Cf is often modeled as either a Maxwellian or Watt fission spectrum. MCNP5 manual recommends using of a Watt fission spectrum in estimating the fission spectrum of ^{252}Cf [8].

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

The coefficients *a* and *b* vary weakly from one isotope to another. For spontaneous fission of ^{252}Cf,

$a = 1.025 \quad b = 2.926$

MCNP code provides seven standard neutrons, six standard photons, and four standard electron final results (tallies), all normalized to be per starting particle. Some tallies in criticality calculations are normalized differently [9].

For solving our problem F2 card (Tally type) was used. The problem to be estimated (calculated) has a surface flux tally.

MCNP tallies can be modified in many different ways. The DE and DF cards allow modeling of an energy-dependent dose function, which is a continuous function of energy from a table whose data points need not coincide with the tally energy bin structure (E card). An example of such a dose function is the flux-to-radiation dose conversion factor given in Appendix H of MCNP5 manual [10]. The flux was converted into dose equivalent rates using the dose factors of the International Commission on Radiological Protection (ICRP) Publication 21 (ICRP 1973) [11].

For description of the tallies, it is useful to consider a reference sphere whose origin is at the same position with that of the source.

For our problem, we used only the history cutoff (NPS) card. The amount of histories considered in each performed run was large enough to reach a Monte Carlo calculation uncertainty less than 1%.

The MCNP5 code calculations were performed using an Intel(R) Core(TM)2 Duo CPU (E6750 @ 2.66 GHz 2.66 GHz) with 2 GB RAM under the Windows 7 Operating System.