

MONTE CARLO SIMULATIONS ABOUT NEUTRON AND GAMMA INTERACTIONS AND SHIELDING PARAMETERS OF ALGODONITE MINERAL

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Abstract

Algodonite is a copper-arsenide type mineral. Its name comes from its discovery locality Algodones Silver Mine, Coquimbo, Chile. In this paper, neutron and gamma interactions with Algodonite were simulated using by FLUKA Monte Carlo code. Gamma linear attenuation coefficients and neutron total macroscopic cross sections were calculated for different primary energies. Results were compared to most commonly neutron shielding materials. Also several interaction parameters as absorbed dose distribution, secondary radiation yield and radioisotope production were given. As a result, gamma and neutron shielding capacity of Algodonite is not bed all..

Keywords

Algodonite, neutron particles, gamma radiation, FLUKA Monte Carlo Code

1.Introduction

Algodones was first discovered in 1858 [1]. Algodonite is a mineral including copper and arsenide elements. Such as other copper-arsenide minerals (Domeykite and Whitneyite) Algodonite is most commonly located in hydrothermal deposits. Algodonite has high density as 8.55 gram per cubic centimeter and its chemical formula is Cu_6As . It is in Hexagonal crystal system and Dihexagonal-Dipyramidal class. It has no radioactive and magnetic property. This mineral is situated on the territory in 14 different countries (Argentina, Austria, Bolivia, Chile, Czech Republic, France, Germany, Iran, Namibia, New Zealand, Sweden, Switzerland, UK and USA).

An x-ray diffraction study on three Cu-As minerals (Dommeykite, Algodonite and Whitneyite) was formed [2]. Stabilities and compositions of

Algodonite were determined. In this study, Cu and As percentages were determined for Algodonite samples taken from different locations.

Also, three trace elements (Ni, Fe and Ag) were detected in samples taken from Cashin and Mohawk mines. [3]. In UK, Algodonite mine was discovered and a study was conducted to determine its stoichiometric formula. [4]. Koenig attempted to produce Algodonite mineral in laboratory conditions [5].

Neutrons are uncharged particles. For this reason, they are capable of traveling long distances within the material. Today, several heavy metals, hydrogen-containing materials as polymers, composites as heavy concrete and boron bearing compounds and minerals are used in neutron shielding processes. In literature, there are several studies about mineral based neutron shielding materials. Magnetite mineral was added in ordinary concrete for the purpose of fast neutron shielding [6]. Hematite, limonite and magnetite mineral were tested in terms of usage of radiation shielding material as concrete aggregates [7]. For slow neutron shielding process boron containing ores were used together with epoxy resin as a new composite material [8]. Hematite effect on radiation shielding property of concrete was simulated by using MCNP Monte Carlo code [9]. In our previously two studies, we used several boron ores against ^{241}Am -Be neutrons by experiments and FLUKA Monte Carlo simulations [10, 11]. In this paper, we simulated neutron and gamma shielding and interactions parameters for Algodonite mineral by using Monte Carlo simulations.

2. Methodology

Nowadays Monte Carlo simulation method has many application areas as physical science, engineering, medicinal physics, space physics, finance and telecommunication. Especially, in nuclear physics, sometimes this method is preferred instead of experiments because of radiation damages on human body. Radiation interactions with matter were simulated easily using different Monte Carlo packages called FLUKA, GEANT4, MCNP, MARS, and PHITS etc. Monte Carlo simulations about nuclear interactions are often found in literature [12-14].

FLUKA is a fully integrated particle physics Monte Carlo simulation package [15, 16]. It is a code based on the FORTRAN programming language. Areas of applications of FLUKA code are as follows:

- High energy experimental physics and engineering
- Shielding
- Detector and telescope design
- Cosmic ray studies
- Dosimetry
- Medical physics and radio-biology

FLUKA describes the interactions between matter and radiations. An input file prepares to run a simulation. There are many options in input file as beam properties, geometry, material definition, physical settings, score options and primary radiation weight. Each option is represented by an input card and the sequential order of these cards is important. This sequence should be in form beam properties, irradiation geometry, material definition physical settings, and score options. We used FLUKA Monte Carlo code to obtain gamma linear attenuation coefficients, neutron total macroscopic cross sections, absorbed dose distributions, secondary radiation yields and radioisotope productions after irradiation for Algodonite mineral. Beam energies were entered in input file as three different neutron energies (4.5 MeV by $^{241}\text{Am-Be}$, 2.16 MeV by ^{252}Cf and 14.1 MeV by reactors) and four different gamma energies (6 MeV by medical LINAC, 1.25 MeV by ^{60}Co , 662 keV by ^{137}Cs , 356 keV by ^{133}Ba). And then mass percentages (Cu - 83.58% and As - 16.42%) and density (8.55 g/cm³) of Algodonite in input. These values were taken webmineral webpage. Mineral used in the simulation have been entered in the form of the disk sample (diameter: 2cm and thickness: 1cm). NEW-DEFA card was used as

physical settings. Transport of neutron particles and gamma rays is activated in this way. Four different score cards used in input to obtain transmission factor (T), absorbed dose (D), double differential secondary particle yield and produced radioisotopes as USRBDX, USRBIN, USRYIELD and RESNUCLEI, respectively. Finally simulation was run by beam emission of source.

After simulations obtained transmission values (T, attenuation factors) were used in Eq-1 and in this way gamma linear attenuation coefficients (μ) and neutron total macroscopic cross sections (Σ) were calculated. In this equation x is sample thickness as 1 cm.

$$(\Sigma, \mu) = \ln T/x \quad (1)$$

Absorbed dose distribution, secondary radiation yield and radioisotope production values were shown as contour and line plots by using FLAIR [17]. FLAIR is a data analysis interface compatible with FLUKA.

3. Results and discussion

3.1. Neutron Interactions

Neutron interactions with Algodonite for three different neutron sources were simulated to report attenuation factors, total macroscopic cross sections, neutron fluencies, absorbed dose distributions, secondary neutron and photon yields and radioisotope production yields. Attenuation factors and macroscopic cross sections are shown in Table 1.

Table 1. Neutron Attenuation Factors and Total Macroscopic Cross Sections

Source	Attenuation Factor	Σ (cm ⁻¹)
$^{241}\text{Am-Be}$	0,2257 ± 0,032 %	0,2977
^{252}Cf	0,2945 ± 0,024 %	0,2445
Reactor	0,3075 ± 0,015 %	0,2359

If founded values compare to several most used neutron shielding material, it can be said that they are acceptable. For example, ordinary concrete ($\Sigma_{\text{Cf-252}} = 0,149$, $\Sigma_{\text{Am-241/Be}} = 0,148$ and $\Sigma_{\text{Reactor}} = 0,120$ by FLUKA) has lesser macroscopic cross sections than Algodonite almost two times. Another example is a comparison to stainless steel ($\Sigma_{\text{Cf-252}} = 0,277$, $\Sigma_{\text{Am-241/Be}} = 0,327$ and

$\Sigma_{\text{Reactor}} = 0,223$ by FLUKA). Fig.1, Fig.2 and Fig.3 shows neutron fluencies.

As can be seen these figures, most neutron counts are observed in the range of $0.1 \div 10$ MeV and thermal region is empty. Also there are several epithermal peaks. Absorbed dose (D) distributions as a function of simulation geometry are illustrated in Fig.4, Fig.5 and Fig.6 as 2D contour plots.

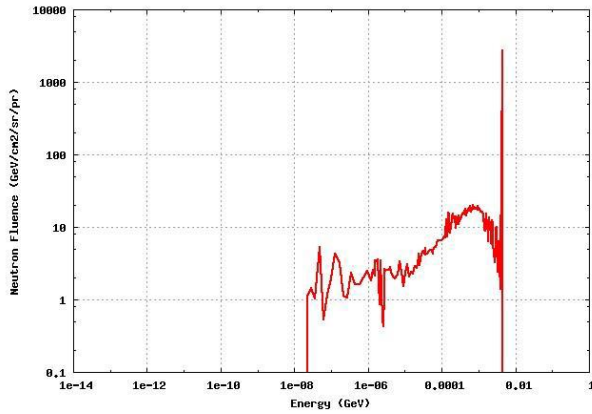


Figure 1. Neutron fluence as a function of energy for $^{241}\text{Am-Be}$ source

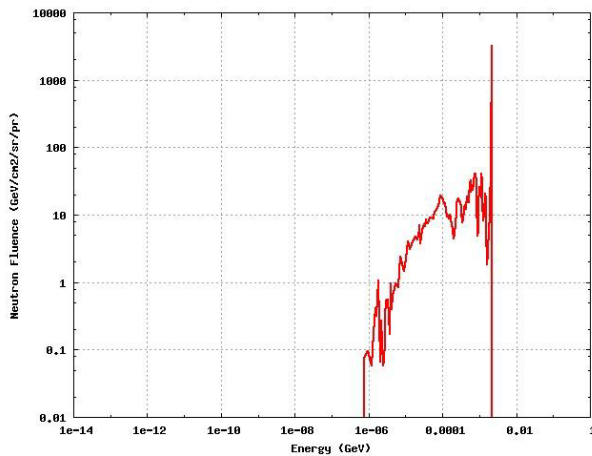


Figure 2. Neutron fluence as a function of energy for ^{252}Cf source

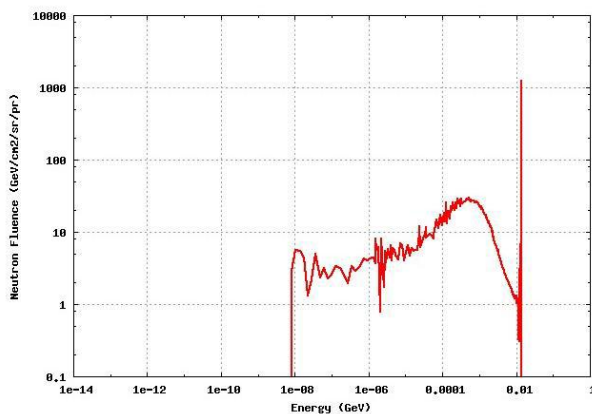


Figure 3. Neutron fluence as a function of energy for reactor neutrons

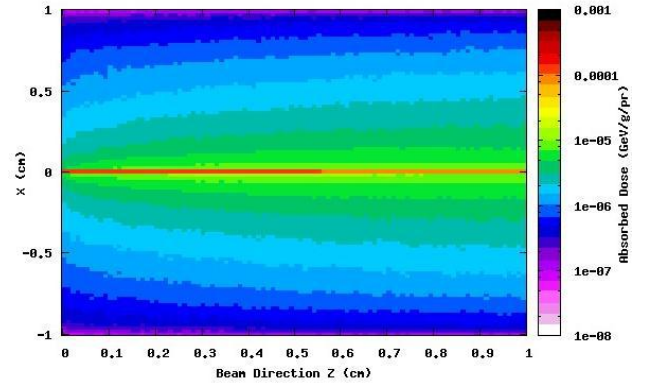


Figure 4. Absorbed Dose distribution for $^{241}\text{Am-Be}$ source

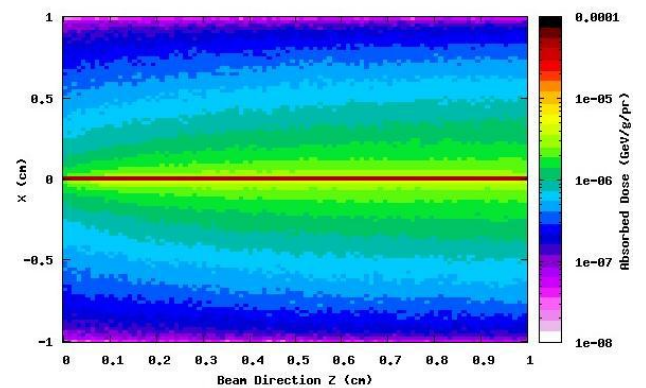


Figure 5. Absorbed Dose distribution for ^{252}Cf source

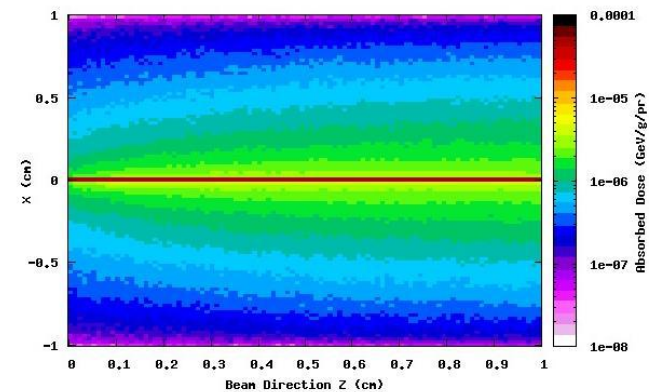


Figure 6. Absorbed Dose distribution for reactor neutrons

Neutrons and photons as secondary radiation emerged from neutron interactions with Algodonite. In Fig.7, Fig.8 and Fig.9 secondary photon and neutron double differential yields are given. The highest energy photons were produced by reactor neutrons. Photon and neutron yields are close for three neutron sources.

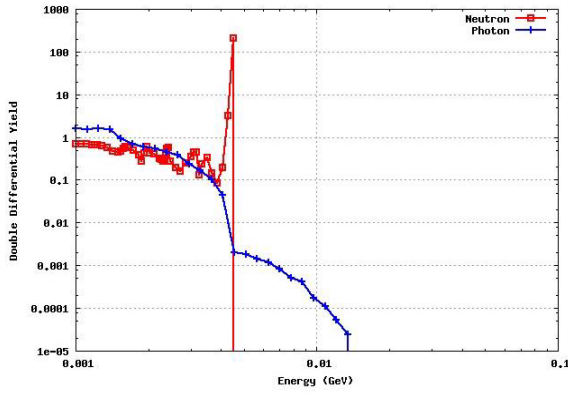


Figure 7. Double differential secondary neutron and photon yields for $^{241}\text{Am-Be}$ source

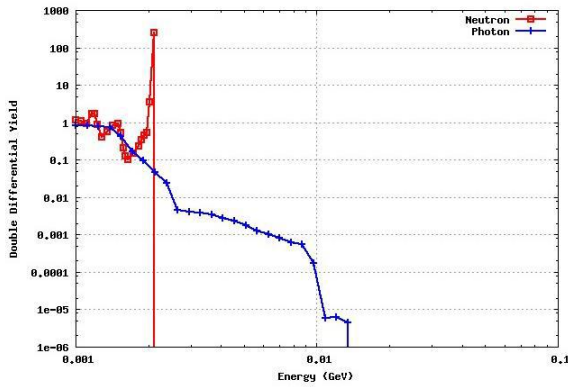


Figure 8. Double differential secondary neutron and photon yields for ^{252}Cf source

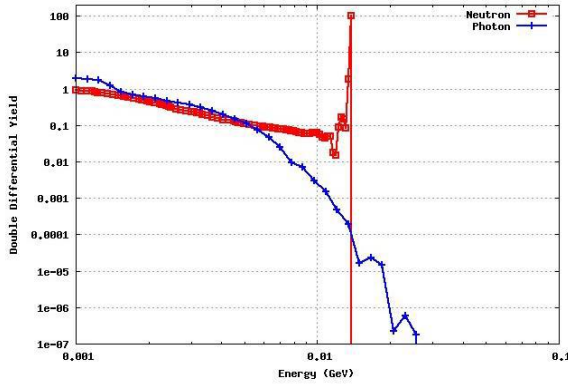


Figure 9. Double differential secondary neutron and photon yields for reactor neutrons

Radioisotope yields after neutron irradiation as a function of mass and atomic number are shown in Tables 2 - 4, for ^{252}Cf , $^{241}\text{Am-Be}$ and reactor source respectively. After neutron interactions with Algodonite nine, eleven and sixteen different isotopes occur for ^{252}Cf , $^{241}\text{Am-Be}$ and reactor source respectively.

Table 2. Isotope yields as a function of A and Z for ^{252}Cf

A	Isotope Yield (nuclei/cm ³ /pr)	Error (%)
76	6.95E-04	1.42E-01
75	7.41E-02	2.71E-02
66	5.10E-04	1.86E-01
65	7.35E-02	2.37E-02
64	1.36E-03	1.05E-01
63	1.94E-01	1.72E-02
60	6.04E-08	2.19E+01
4	6.52E-08	1.26E+01
1	3.54E-03	2.66E-02
Z	Isotope Yield (nuclei/cm ³ /pr)	Error (%)
33	7.48E-02	2.60E-02
29	2.66E-01	1.71E-02
28	3.54E-03	5.74E-02
27	6.04E-08	2.19E+01
2	6.52E-08	1.26E+01
1	3.54E-03	2.66E-02

Table 3. Isotope yields as a function of A and Z for $^{241}\text{Am-Be}$

A	Isotope Yield (nuclei/cm ³ /pr)	Error (%)
76	4.79E-04	2.70E-01
75	9.12E-02	1.53E-02
66	2.87E-04	1.38E-01
65	1.21E-01	6.67E-03
64	1.05E-03	1.25E-01
63	2.70E-01	4.93E-03
62	2.61E-04	1.45E-01
60	1.15E-04	1.12E-01
4	1.14E-04	4.42E-01
2	2.62E-04	2.10E-01
1	1.08E-02	7.76E-02
Z	Isotope Yield (nuclei/cm ³ /pr)	Error (%)
33	9.17E-02	1.48E-02
29	3.81E-01	4.88E-03
28	1.11E-02	5.39E-02
27	1.15E-04	1.10E-01
2	1.14E-04	4.42E-01
1	1.11E-02	7.99E-02

Table 4. Isotope yields as a function of A and Z for reactor source

A	Isotope Yield (nuclei/cm ³ /pr)	Error (%)
76	3.69E-04	1.70E-01
75	3.55E-02	2.34E-02
74	4.22E-02	3.16E-02
71	5.55E-05	8.06E-01
66	1.58E-04	2.93E-01
65	5.82E-02	1.90E-02
64	7.14E-02	9.11E-03
63	1.39E-01	7.34E-03
62	1.24E-01	9.70E-03
61	5.59E-05	4.93E-01
60	7.48E-03	1.03E-01
59	2.08E-03	1.35E-01
4	1.04E-02	4.86E-02
3	3.39E-06	4.90E+00
2	2.44E-03	1.63E-01
1	5.99E-02	2.15E-02
Z	Isotope Yield (nuclei/cm ³ /pr)	Error (%)
33	7.65E-02	1.18E-02
32	1.60E-03	1.56E-01
31	5.55E-05	8.06E-01
29	3.32E-01	4.37E-03
28	6.07E-02	2.67E-02
27	1.04E-02	9.87E-02
2	1.04E-02	4.86E-02
1	6.24E-02	2.39E-02

3.2. Photon Interactions

For photon interactions with Algodonite, linear attenuation coefficients, attenuation factors, and absorbed dose distributions are given in the present paper. Linear attenuation coefficient (μ) is an important parameter to define good photon shielding materials. Gamma ray shielding capability of materials increases with increasing linear attenuation coefficient value. Linear attenuation coefficients and attenuation factors for four different primary photon energies were given in Table 5.

Table 5. Photon Attenuation Factors and Linear Attenuation Coefficients

Source	Attenuation Factor	$\mu(\text{cm}^{-1})$
LINAC	0,2652± 0,059%	0,2655
⁶⁰ Co	0,1092 ± 0,073%	0,443
¹³⁷ Cs	0,0485 ± 0,118%	0,6054
¹³³ Ba	0,0164 ± 0,447%	0,8224

As a comparison, linear attenuation coefficients of concrete were simulated by FLUKA. These values are 0.193, 0.217, 0.230 and 0.243 for LINAC, ⁶⁰Co, ¹³⁷Cs, ¹³³Ba sources respectively. As can be seen these comparison, it can be said that Algodonite has fine gamma ray shielding capability. Absorbed dose distributions can be seen in Fig.10, Fig.11, Fig.12 and Fig.13. Average absorbed doses are D_{LINAC}=16.747, D_{Co-60}=9.1514, D_{Cs-137}=6.6076 and D_{Ba-133}=7.0875 in GeV per gram unit.

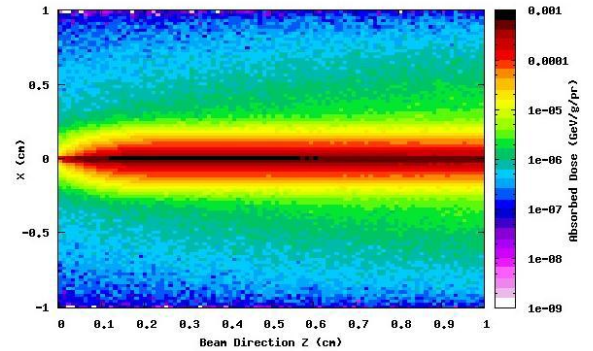


Figure 10. Absorbed Dose distribution for medical LINAC source

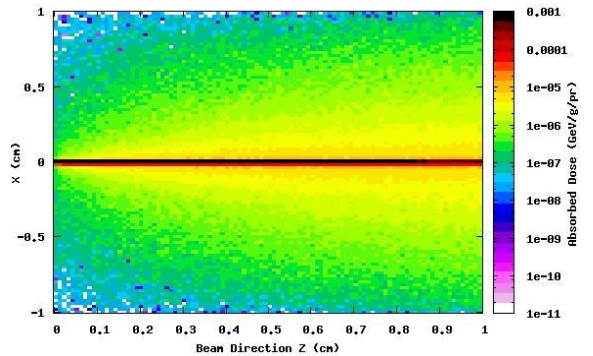


Figure 11. Absorbed Dose distribution for ⁶⁰Co source

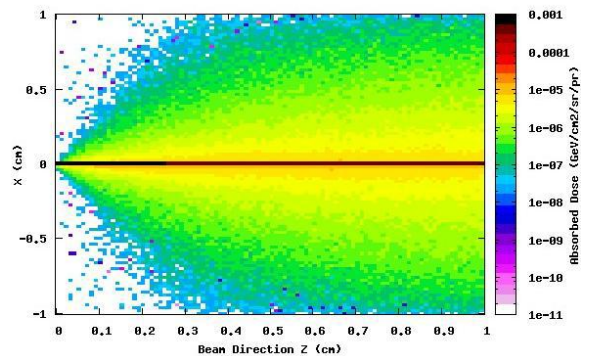


Figure 12. Absorbed Dose distribution for ¹³⁷Cs source

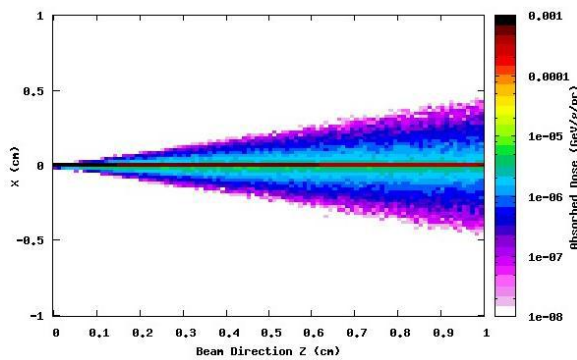


Figure 13. Absorbed Dose distribution for ^{133}Ba source

4. Conclusions

We are simulated neutron and photon interactions with Algodonite copper-arsenide type mineral. We suggested an alternative gamma and neutron shielding material. It can be added concrete, lead or different base materials to prevent against radiation damages. Also we gave information about absorbed doses, primary neutron fluencies, secondary radiation yields and isotope production yield. This information may be useful researchers about nuclear engineering, mineral processing, radiation physics etc.

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