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Galena with B4C high density heavy concrete for shielding nuclear reactors

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Abstract
Shielding of biological type belong to nuclear reactor is considered as one of the main issues and lower complexity and expense of these installations is of important interest. In this paper, Galena mineral and Boron Carbide (B4C) were used to produce a high density heavy concrete. Galena minerals that present in the most regions of Iran were regarded to be applied in the concrete mix design. Boron Carbide (B4C) is regarded as a ceramic material that is efficient in order to absorb thermal neutron as a consequence of wide neutron absorption cross section. Neutron shielding characteristics of samples could explain the cross section in matter and neutron capture. Neutron cross section measurements of samples have done by using a source of 14.1MeV neutrons. By using Geant 4 Monte Carlo code, cross section and neutron capture of each samples could be calculated through it. As a consequence, cross section value of concrete can be raise by growing boron carbide (B4C) concentration and lower neutron capture value of samples and boost the attributes of shielding.

Keywords: Galena, B4C, neutron cross section, neutron capture, Geant 4 Monte Carlo, heavy concrete, shielding

Introduction

Because of its cheaper, easier molded into compound shape, good structural and appropriate as neutron shielding materials in comparison to other shielding materials, concrete is considered a multi-user material and it is usually accustomed as a radiation shielding material. Y. Abdoullah *et al.* [1] investigated that frequently concretes are composites material include aggregate, sand, water and cement. The radiation shielding type belong to nuclear reactor is an expensive and very complex method. Pavlenko VI *et al.* [2] have recognized that a nuclear reactor usually requires two shields one of which is a shield to guard the walls of the reactor from radiation harm and simultaneously, reflect neutrons back into core; and the other, a biological shield that guards people and the environment. The biological type shield lowers the rank of Gamma radiation and neutrons to current dose limits. The biological shield is marked by many centimeters of very high density concrete. S.M.J Mortazavi *et al.* [3] proposed that in nuclear type reactors, neutron radiation is the main hrd to shield and hydrogen is regarded as the major effective element in decelerating (thermalizing) neutrons over the whole energy spectrum. It is thought that the greatest hydrogen in concrete normally show in the model of water in which hydrated within cement curing and collects setting and free water streaming in the porousness of concrete. T. Korkut *et al.* [4] explored that Boron is an important chemical component to be used in neutron absorption mechanism. It is important in shielding technology because of its flawless shielding characteristics. Baştürk M *et al.* [5] have recognized that it is an effective absorber that can be used in neutron shielding materials. There are various research carried out about radiation shielding by boron mixtures [6-10]. Concrete has many advantages and is very effective material to be used in shielding reactors. The high density concrete, the higher linear gamma and neutron attenuation properties in comparison to regular concrete. Sun H. *et al.* [11] explored that concrete that is made up of Portland cement, sand aggregate and water and is considered as one of the main conventional materials used in the structure of commercial buildings. Nowadays, normal concrete (density about 2350kg/m3) is mostly advantageous to be applied in superficial and orthovoltage radiotherapy rooms [12] Galena (PbS) is the basic lead mineral [13].

Galena also have cerusssite ($PbCO_3$), plattenerite (PbO_2) and anglesite ($PbSO_4$) in its combination. Galena is a very condensed material, and has a density equal to 7400-7600 kg/m3, so it is nearly as dense as iron. The chemical combination and physical properties of Galena are summarized in table 1.

In nuclear type reactor, a specific composition of Portland, cement and sand was used to implement radiation shielding, while as Atsuhiko *et al.* (2004) confirm, boron carbide (B_4C) was doped with Portland cement to construct concrete as thermal neutron absorber and lower radioactivity through thermal neutron.

Table 1: Physical properties of the Galena used in this study

Mineral Properties	Galena
Chemical composition	Lead Sulfide (PbS)
Molecular weight	239.26g
Lead content	86.59% Pb 13.40% S
B_2O_3 content	---
Stiffness	2.5
Density (g/cm3)	7.0-7.5
Color	Gray

The main goals of this paper are to reach to neutron cross section through Geant4 Monte Carlo code for samples. Cross section according B4C percentage for Galena is shown in table 2.

Material and Methods

To start wit, the materials included gravel, sand, cement, water, micro siliceous and Boron Carbide powder. Galena minerals were applied for production of a high density concrete. Concrete must include a large amount of water in order to be used as a shield in nuclear reactors. Higher water content cause to be concrete more efficient than any regular concrete. In this paper, two types of concrete mixes were produced. First, regular concrete mixes were composed of gravel, sand, cement, water and micro siliceous. Second, GaB4C concrete Galena and B4C were applied to completely replace sand concrete mixture. In table 3, concentration of Galena and Boron Carbide (B4C) in concretes is exhibited. Cross section according neutron capture is shown in fig.1 and cross section according density is shown in fig.2.

By exposing to neutron source 241Am-Be (number of events processed 100000) radiation test was carried out.

Monte carlo simulation

Described and defined by S for neutron. Inverse length is units of the linear attenuation coefficient that commonly pointed out by cm-1. The microscopic range about neutron interaction describes the cross section (s). Cross section illustrates the effective cross sectional region to neutrons displayed by each nucleus of attenuating materials. The units are normally the barn in which The Geant4 program is considered 1 barn is equivalent to 10cm.

As functional simulation tools for several applications in high energy physics. The interaction and propagation of neutrons in matter in shielding design with Geant4 program can be simulated. It is yielded that cross section and neutron capture could be obtained through Geant4 Monte Carlo code. In the first stage, we entered atomic stoichiometric and densities of sample. In second stage, simulation was started for 100000 primary neutron particles.

Results and Discussion

The cross section and neutron capture are regarded as effective elements in order to define neutron shielding characteristics of sample. Does not exist easy scaling rule for neutron linear attenuation coefficient S. But the cross section is the neutron cross section has been computed by a neutron detector. From Geant4 Monte Carlo code, we calculated the cross section and neutron capture. The calculated contents of cross section and neutron capture by using Geant4 Monte Carlo code are represented as a function of the percentage of the Boron Carbide (B4C) in table2 and Cross section according neutron capture is shown in fig.1.

As can be seen in table2, as cross section increases, the percentage of Boron Carbide (B4C) in the samples increases. It is seen successfully that the neutron cross section is strongly dependent on the Boron Carbide (B4C) intensity in the matter and as it can be seen in fig.1, 50% B4C + 50%Galena with density of 5.06 have high cross section value and so it has high neutron shielding properties in comparison to other samples.

Furthermore, the calculated contents of cross section and neutron

As demonstrated above, Boron Carbide percentage is effective on neutron shielding capability of matter. Thus, as can be seen in table2 and fig2, because of high cross section, %50 Galena + %50 Boron Carbide is more effective shielding material. Also cross section, Neutron capture and Density of Galena, Concrete and Boron Carbide listed in table 4.

As it could be inferred from table 2, we can say that neutron capture increases as the density increases and also we can see in fig2 that 50% B4C + 50%Galena with density of 5.06 have high cross section value and have high neutron shielding properties in comparison to other samples.

Conclusions

We have explored in present research, rapid neutron shielding capture by using Geant4 Monte Carlo code are represented as a function of the density in table2 and in fig.2, cross section is shown according Density.

Characteristics of Galena (PbS), Boron Carbide (B4C), different percentage of Galena with Boron Carbide samples by using experiment and simulation process in. The results of the present study do considered as a new explanation about the cross section of fast neutron through materials containing different percentage of Boron Carbide. Neutron cross section and neutron capture are mainly in relation on the value of Boron Carbide in our samples. Because of the high cross section and good neutron capture of our samples, %50 Galena + %50 Boron Carbide is a more acceptable shield than other samples. These materials can be very advantageous for building walls of nuclear energy centrals, as moderator for nuclear reactors, in nuclear medicine departments and nuclear research centers, etc., to keep safe harms from neutron particle.

Table 2: Cross section, neutron capture and density of concretes according B4C percentage and Galena

Material	Cross section(cm-1)	Neutron Capture	density (g/cm3)
5% B4C+95% PBS	0.206962671	20	7.6
10% B4C+90PBS	0.221739537	18	7.092
15% B4C+85% PBS	0.236877431	12	6.838
20% B4C+80%PBS	0.249168019	13	6.584
25% B4C+75% PBS	0.259999463	14	6.33
30% B4C+70% PBS	0.269149532	10	6.076
35% B4C+65% PBS	0.276289877	4	5.822
40% B4C+60% PBS	0.282124213	11	5.568
45% B4C+55% PBS	0.286514754	9	5.314
50% B4C+50% PBS	0.289220589	8	5.06
55% B4C+45% PBS	0.288880123	5	4.806
60% B4C+40% PBS	0.28829426	9	4.552
65% B4C+35% PBS	0.28662663	5	4.298
70% B4C+30% PBS	0.282943982	4	4.044
75% B4C+25% PBS	0.277399124	4	3.79
80% B4C+20% PBS	0.27010283	1	3.536
85% B4C+15% PBS	0.260706539	2	3.282
90% B4C+10% PBS	0.249846934	2	3.028
95% B4+C5% PBS	0.23731474	1	2.774

Table 3: Concentration of Galena and Boron Carbide (B4C) in concretes

Material Concrete	Galena	Boron Carbide
1	%95	%5
2	%90	%10
3	%85	%15
4	%80	%20
5	%75	%25
6	%70	%30
7	%65	%35
8	%60	%40
9	%55	%45
10	%50	%50
11	%45	%55
12	%40	%60
13	%35	%65
14	%30	%70
15	%25	%75
16	%20	%80
17	%15	%85
18	%10	%90
19	%5	%95

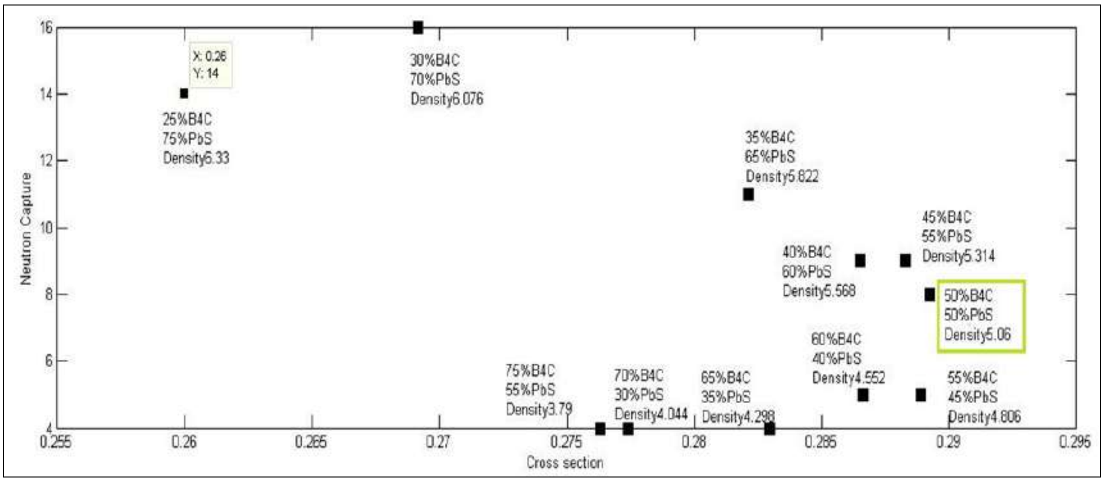

Fig 1: The measured value of Cross section as a function of Neutron Capture

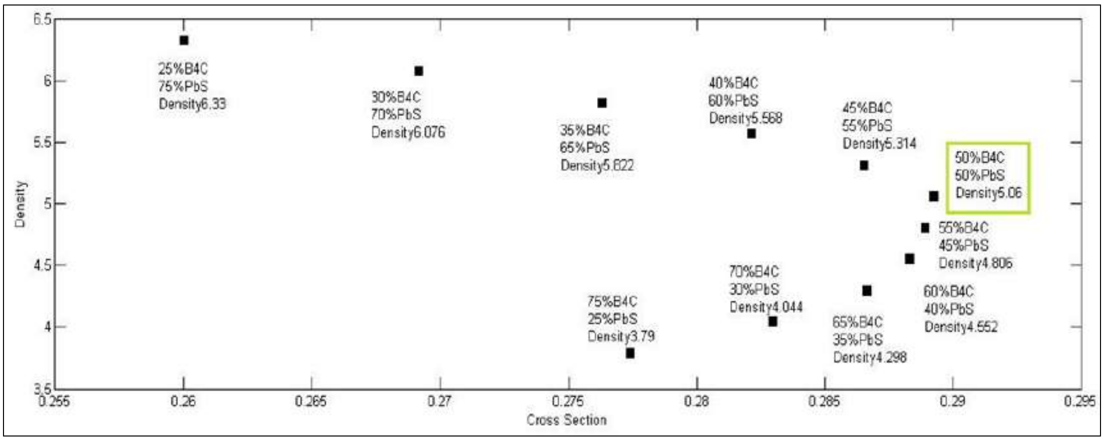

Fig 2: The measured value of cross section as a function of Density

Table 4: Shielding properties of Galena, Concrete and Born Carbide

Material	Cross Section	Neutron Capture
PbS	0.188831501	16
Concrete	0.163818597	20
%100 B4C	0.2251179	-

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