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Research Article

# Improving Neutron Shielding Capacities of Datolite and Galena by Boron Carbide Additive for Nuclear Reactor Biological Shielding

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## ABSTRACT

Biological shielding of nuclear reactor is an important interest and diminishing the intricacy and cost of theseinstallations is important interest. In this paper, we used Datolite and Galena minerals and boron carbide for nuclear reactor shielding aim. Datolite and Galena minerals that exist in many parts of world were used in the concrete mix design. Boron carbide is an important material for neutron absorption processes. The cross section in matter and neutron capture is utilizable causes to explain neutron shielding characteristics of samples. Neutron cross section simulations of samples have done by using a source of 14.1 MeV reactor neutrons. Cross section and number of captured neutrons of each sample were estimated by using Geant4 Monte Carlo code. As a result, growing boron carbide concentration can raise cross section value of Datolite and galena.

**Keywords:** Datolite; Galena; Boron carbide; Neutron cross section; Geant4 montecarlo; Shielding

## INTRODUCTION

Biological shield is a mass of absorbing material placed around a reactor or radioactive source to reduce the radiation to a level safe for humans. The effectiveness of a material as a biological shield is related to its cross section for scattering and absorption. Geant4 Monte Carlo toolkit can calculate all of the neutron cross sections as elastic scattering, inelastic scattering and capture. The common neutron absorbers and moderators are compounds of boron, cadmium, carbon and hydrogen. Also paraffin wax and heavy concrete have been widely used effectively for this aim. Recently cheaper and more lightweight alternative materials for nuclear reactor shielding have been investigated. Datolite and galena minerals are known as high radiation shielding performance. High performance heavy concrete including colemanite and galena as a biological shield against photons and neutrons was developed. This colemanite-galena concrete has a 10% greater neutron absorption compared with ordinary concrete [1]. In another study ulexite and galena minerals were used to produce radiation shielding intermediate-weight concrete. This new concrete has a 7.22 times higher neutron attenuation capacity compared with ordinary concrete [2]. Datolite based heavy concrete as nuclear reactor shield and medical radiation therapy room structural material was produced. This concrete has higher compressive strength, neutron and photon shielding capacity than normal concrete [3]. A Monte Carlo design study by MCNP code about shielding of radiation therapy rooms was performed. Datolite-Galena concrete was modeled and radiation shielding parameters were estimated in addition to magnetite, limonite, and serpentine concretes. Also photo-neutron production capacities of these samples were evaluated [4]. In a different study a new type of neutron and photon shielding material based on 40% galena, 55% polyethylene and 5% boric acid was suggested using by MCNP Monte Carlo code [5]. In this paper, to enhance neutron shielding capacities of Datolite and galena minerals boron carbide additive was evaluated by Geant4 Monte Carlo methodology.

## MONTE-CARLO DESIGN

Monte Carlo methods are based on the use of random numbers and probability statistics to investigate nuclear interactions. The most common Monte Carlo simulation tools in nuclear area are MCNP, FLUKA, PHITS, MARS and GEANT4. These codes widely use in radiation interactions with matter [6-8], radiation detectors [9], cosmic ray physics [10], medical physics [11] etc. The Geant4 program [12-14] is useful simulation device for

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multitude applications in high energy physics (http://geant4.web.cern.ch/geant4/). Geant4 can simulate the interaction and propagation in matter of neutrons in shielding plan. We obtained cross section and number of captured neutrons via Geant4 Monte Carlo code. QGSP-BIC-HP hadronic process was applied in the simulations to describe neutron interactions with samples. Sample dimension was selected as 1 × 1 × 1 cm. Mass compositions and densities of samples have been entered (Table 1). In second place simulation has been started for 100000 primary neutron particles. As output, neutron total macroscopic cross sections and number of captured neutrons were taken.

**Table 1:** Composition and Densities of samples.

| S.No | Material | Density(g/cm3) | Material | Density(g/cm3) |
|---|---|---|---|---|
| 1 | 95% Datolite+5% B4C | 2.881 | 5% B4C+95% PBS | 7.6 |
| 2 | 90% Datolite+10% B4C | 2.862 | 10% B4C+90% PBS | 7.092 |
| 3 | 85% Datolite+15% B4C | 2.843 | 15% B4C+85% PBS | 6.838 |
| 4 | 80% Datolite+20% B4C | 2.824 | 20% B4C+80% PBS | 6.584 |
| 5 | 75% Datolite+25% B4C | 2.805 | 25% B4C+75% PBS | 6.33 |
| 6 | 70% Datolite+30% B4C | 2.786 | 30% B4C+70% PBS | 6.076 |
| 7 | 65% Datolite+35% B4C | 2.767 | 35% B4C+65% PBS | 5.822 |
| 8 | 60% Datolite+40% B4C | 2.748 | 40% B4C+60% PBS | 5.568 |
| 9 | 55% Datolite+45% B4C | 2.729 | 45% B4C+55% PBS | 5.314 |
| 10 | 50% Datolite+50% B4C | 2.71 | 50% B4C+50% PBS | 5.06 |
| 11 | 45% Datolite+55% B4C | 2.691 | 55% B4C+45% PBS | 4.806 |
| 12 | 40% Datolite+60% B4C | 2.672 | 60% B4C+40% PBS | 4.552 |
| 13 | 35% Datolite+65% B4C | 2.653 | 65% B4C+35% PBS | 4.298 |
| 14 | 30% Datolite+70% B4C | 2.634 | 70% B4C+30% PBS | 4.044 |
| 15 | 25% Datolite+75% B4C | 2.615 | 75% B4C+25% PBS | 3.79 |
| 16 | 20% Datolite+80% B4C | 2.596 | 80% B4C+20% PBS | 3.536 |
| 17 | 15% Datolite+85% B4C | 2.577 | 85% B4C+15% PBS | 3.282 |
| 18 | 10% Datolite+90% B4C | 2.558 | 90% B4C+10% PBS | 3.028 |
| 19 | 5% Datolite+95% B4C | 2.539 | 95% B4C+5% PBS | 2.774 |

## RESULTS AND DISCUSSION

Cross section per volume (elastic scattering+inelastic scattering +capture) and number of captured neutrons values obtained by Geant4 output are influential factors to determine neutron shielding characteristics of samples. Simulations were performed for datolite with B4C and for galena with B4C as 5% step intervals (5% B4C+95%

Datolite, 10% B4C+90% Datolite).Cross section per volume values versus B4C content for datolite can be seen in Figure 1. Looking at this figure cross section per volume value is directly proportional (R2=0.9975) with B4C%. In other words, neutron shielding capacity of datolite increases with increasing B4C additive. Figure 2 shows number of captured neutrons against B4C% in datolite. Neutron capture property decreases with increasing B4C content as linear (R2=0.8936). So, the addition of B4C to datolite was reduced neutron absorption. Cross section per volume values with B4C content for datolite can be seen in Figure 3. This curve is a quadratic polynomial distribution with R2=0.9998. So, there is a peak value at 50% B4C point. After this point, cross section per volume values decrease with increasing B4C additive. In other words, optimal B4C additive proportion in galena is 50%. Figure 4 shows number of captured neutrons against B4C% in galena. Neutron capture ability of galena decreases with increasing B4C content as linear (R2=0.9178) similar to datolite. The addition of B4C to galena was reduced neutron capture capability.

Fig.1.

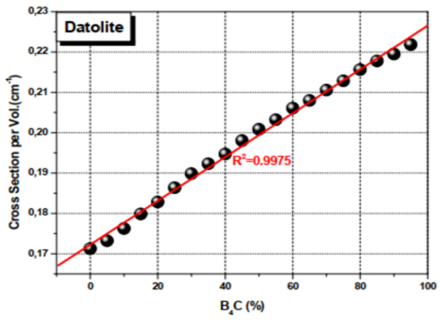


**Figure 1:** Cross Section per Volumes for Datolite-B4C.

Fig.2.

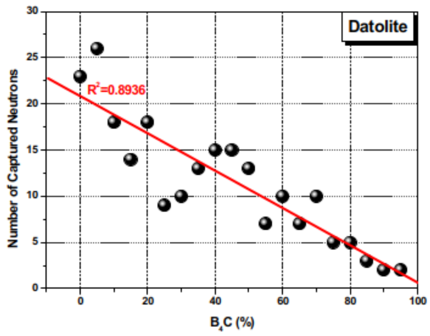


**Figure 2:**Number of Captured Neutrons for Datolite-B4C.

Fig.3.

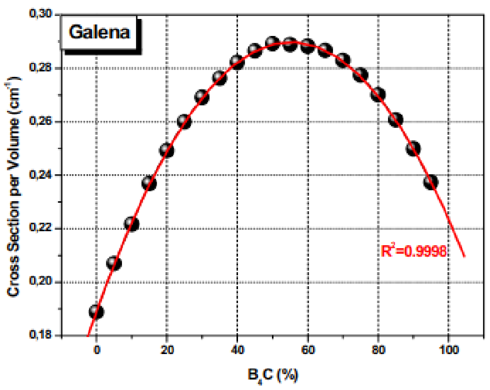


**Figure 3:**Cross Section per Volumes for Galena-B4C.

Fig.4.

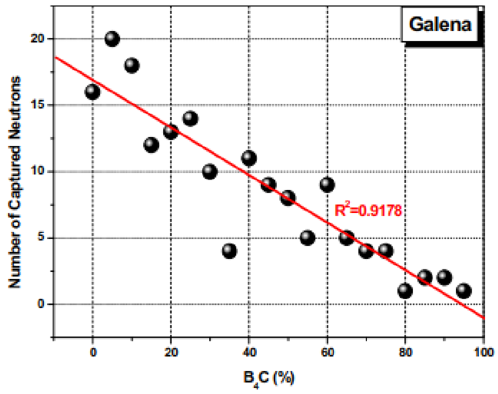


**Figure 4:**Number of Captured Neutrons for Galena-B4C.

## CONCLUSION

Cross section per volume (elastic scattering+inelastic scattering +capture) and number of captured neutron values were estimated for interactions between 14.1 MeV reactor neutrons and new materials including different mass compositions of datolite-boron carbide and galena-boron carbide materials by Geant4 Monte Carlo methodology.

Cross section per volume value was increased by the addition of B4C from 0.17 cm-1 to 0.22 cm-1 for datolite and from 0.19 cm-1 to 0.29 cm-1 for galena. These cross section values are higher than cross section value of concrete (0.16 cm-1). So, these new materials may be used as shielding designs in the nuclear industries and the nuclear

reactor investigations. The second part of this research, simulated samples will be produced and experimental evaluation will be made.

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