แบบรับรองการใช้ประโยชน์จากผลงานวิจัยหรืองานสร้างสรรค์
มหาวิทยาลัยราชภัฏหมู่บ้านจอมบึง

ชื่องานวิจัยหรืองานสร้างสรรค์ การพัฒนาตัวกลางเลเซอร์ในช่วงอินฟาเรดจากแก้วบอเรตโดยใช้เทคโนโลยีในประเทศ

ผู้วิจัยหรือผู้สร้างสรรค์ รองศาสตราจารย์ ดร.สมิต อินทร์ศิริพงษ์

วัน/เดือน/ปี ที่ทำวิจัยหรือสร้างสรรค์เสร็จสมบูรณ์ 30 กันยายน พ.ศ. 2561

วัตถุประสงค์ของงานวิจัยหรืองานสร้างสรรค์ (โปรดระบุวัตถุประสงค์งานวิจัยข้อที่นำไปใช้ประโยชน์)

เพื่อเป็นแนวทางพัฒนาตัวกลางเลเซอร์ในช่วงอินฟาเรดจากแก้วบอเรตราคาประหยัด

ประเภทของการใช้ประโยชน์จากงานวิจัยหรืองานสร้างสรรค์ (ตอบได้มากกว่า 1 ข้อ)

( ) การใช้ประโยชน์ในเชิงสาธารณะ (✓) การใช้ประโยชน์ในเชิงนโยบาย

( ) การใช้ประโยชน์ในเชิงพาณิชย์ (✓) การใช้ประโยชน์ทางอ้อมของงานสร้างสรรค์

ผลที่เกิดขึ้นอย่างเป็นรูปธรรมจากการนำงานวิจัยหรืองานสร้างสรรค์ไปใช้ประโยชน์ตามวัตถุประสงค์

จากการศึกษาแก้วบอเรตในระบบ K2O3– CaO–B2O3 ที่เติม Er^{3+} พบพีคการเปล่งแสงที่ประมาณ 1.5 ไมโครเมตรซึ่งอยู่ในช่วงอินหาเรด และจากการวิเคราะห์ JO analysis พบว่ามีค่าภาคตัดขวางของการกระตุ้นให้เปล่งแสง เท่ากับ 0.919×10^{-20} cm^2 และสัดส่วนการเปล่งแสง (β) ของแก้ว เติมด้วย Er_2O_3 สูงมากจึงคาดว่าน่าจะใช้เป็นตัวกลางเลเซอร์เพื่อทดแทนการนำเข้าจากต่างประเทศได้อย่างดี

และขอรับรองว่า (หน่วยงาน) บริษัท เซิร์นเทค จำกัด ได้นำงานวิจัยหรืองานสร้างสรรค์ ไปใช้ประโยชน์ในจริง และสามารถนำไปสู่การพัฒนาได้อย่างมีประสิทธิภาพเพิ่มขึ้น

ลงชื่อ .....

(อนุ ทองเกิด)

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Monte Carlo Design and Experiments on the Neutron Shielding Performances of B_2O_3–ZnO–Bi_2O_3 Glass System[1]

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Abstract—Neutron shielding properties of $(90 - x)B_2O_3$–10ZnO–xBi_2O_3 glass systems (where $x = 15$, 20, 25 and 30 mol %) were investigated by Monte Carlo simulations (FLUKA and GEANT4) and experiments. Neutron mass removal cross sections, number of inelastic scattering, elastic scattering, and capture interactions were estimated by simulations. ^{241}Am/Be neutron source was used for the neutron equivalent dose rate measurements. As a result, produced glass samples have fine neutron shielding capacity.

Keywords: Neutron shielding, bismuth-zinc-borate glass, Monte Carlo simulation

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INTRODUCTION

Glass materials are widely used in many industries. So, investigations of radiation shielding properties of glass materials are a coveted issue nowadays. Bismuth borate glasses are technologically important materials because of their high densities, low melting temperatures, excellent optical properties and high attenuation coefficients for X-ray and gamma rays. In addition to these fine properties, neutron shielding performances of these glass systems are an important issue.

There are several studies on bismuth borate and bismuth borosilicate glasses about their production techniques, optical properties and gamma radiation shielding capacities. In [1] bismuth-alumina-borosilicate glasses are produced using melt quenching technique and performed MAS-NMR spectroscopic measurements with thermal analysis [1]. Absorption spectroscopic studies on gamma irradiated bismuth borosilicate glasses were made and the effect of the heavy metal oxide Bi_2O_3 on the glass composition was also studied [2]. In another study, the composition of the glass is (mol %) $20Bi_2O_3$–$15Na_2O$–$50B_2O_3$–$15SiO_2$ was prepared by conventional melt quench method and effect of heavy ion irradiation on the structural properties of this glass system was studied. A significant decrease in the band gap is observed after irradiation which is indicative of the fact that radiation has caused compaction in the glass structure [3]. The ultrasonic velocity for different compositions of irradiated recycled heavy metal oxide (HMO) borosilicate glasses of the xBi_2O_3–50BaO–$(50 - x)B_2O_3$ glass system (where $x = 0$, 5, 10, 15, 20 mol %) were studied by using the pulse echo technique [4]. The influence of gamma-ray irradiation on the optical properties of Bi_2O_3–B_2O_3–SiO_2 glass system was studied in another paper. Broadband infrared (IR) emission at 1310 nm with a FWHM over 200 nm is observed in the gamma-ray irradiated glass [5]. Radiation shielding parameters of bismuth borosilicate glass from 1 keV to 100 GeV were studied. Gamma rays shielding properties of $(50 - x)$ SiO_2 : $15B_2O_3$: $2Al_2O_3$: 10CaO : $23Na_2O$: xBi_2O_3 glasses (where $x = 0$, 5, 10, 15 and 20 mol %) were evaluated by using XCOM software [6]. Calculation study of radiation shielding capacities of silicate and borate heavy metal oxide glasses were made [7].

In this paper, neutron shielding properties of Bi_2O_3–ZnO–B_2O_3 glasses were studied. GEANT4 and FLUKA Monte Carlo codes were used to estimate shielding parameters. Neutron equivalent dose rate measurements were performed to determine shielding performances of produced samples. Results were compared to ordinary glass at the same dimension with the produced glass samples.

[1] The article is published in the original.

Table 1. Codes of produced glass samples

Composition of glass samples, mol %	Code
$75B_2O_3$–$10ZnO$–$15Bi_2O_3$	B75Z10Bi15
$70B_2O_3$–$10ZnO$–$20Bi_2O_3$	B70Z10Bi20
$65B_2O_3$–$10ZnO$–$25Bi_2O_3$	B65Z10Bi25
$60B_2O_3$–$10ZnO$–$30Bi_2O_3$	B60Z10Bi30

Monte Carlo Codes

Monte Carlo is a method based on the random number seeds and the mathematical algorithms [8]. This technique can be applied for physical systems, especially in nuclear science. In this study we used FLUKA [9, 10] and GEANT4 [11, 12] as Monte Carlo simulators. FLUKA is a Monte Carlo package used in interactions between many subatomic particles and elements, compounds and mixtures. FLUKA has many advantages in terms of wide energy range. For example, it can simulate neutrons from thermal level to 20 TeV. Also it is useful in many scientific areas (high energy experimental physics and engineering, shielding, detector and telescope design, cosmic ray studies, dosimetry, medical physics and radio-biology). We used the FLUKA Monte Carlo code to simulate neutron and gamma radiation with different materials in our latest studies [13–17]. In some of these studies we compared FLUKA results to experiments. The FLUKA Monte Carlo Simulation code was used to calculate neutron mass removal cross sections of glass formulations. Detailed information can be found at its web page. GEANT4 is a toolkit for the simulation of the passage of particles through matter. Its areas of application include high energy, nuclear and accelerator physics, as well as studies in medical and space science. It has two hadronic interaction models such as the Quark-Gluon String Precompound (QGSP), the Bertini Model and the FRITIOF Precompound (FTFP). In the present study 4.9.4.p01 released version with QGSP-HP model was used to simulate interactions between neutrons and glass samples. GEANT4 Monte Carlo Simulation toolkit was used to obtain neutron mass removal cross sections and neutron scattering-capture information. Detailed information about GEANT4 can be found at web page.

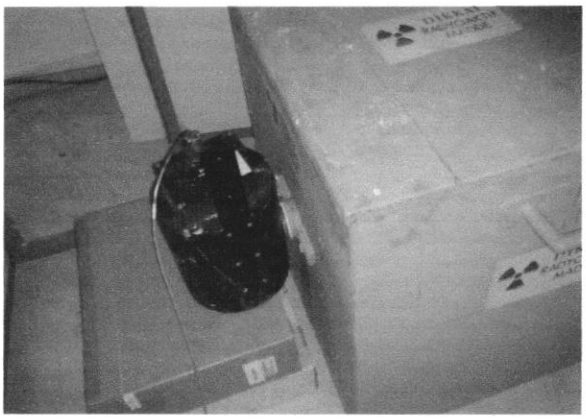

Fig. 1. Experimental design for neutron equivalent dose rate measurements.

Fabrication of Glass Samples

The glasses with their chemical compositions $(90 - x)B_2O_3$–$10ZnO$–xBi_2O_3 (where x = 15, 20, 25 and 30 mol %) were prepared by the normal melt-quenching technique. All chemicals; ZnO, Bi_2O_3 and B_2O_3, used in the present work were of high purity. For each batch composition, 20 g of homogeneous mixture of starting chemicals were melted in high purity alumina crucibles by an electric furnace at a temperature of 1100°C for 3 hours. The melts were quenched by pouring into pre-heated stainless steel molds. The glasses were then annealed at 500°C for about 3 hours to remove thermal strains. Produced samples and their codes can be seen in Table 1.

Neutron Equivalent Dose Rate Measurements

For neutron equivalent dose rate measurement, we used a ^{241}Am-Be neutron source and a Canberra portable neutron detector equipment. ^{241}Am/Be source emits 2–11 MeV neutrons. The physical form of ^{241}Am/Be neutron source is a compacted mixture of AmO_2 with beryllium metal. The NP-100B detector provides us to detect slow and fast neutrons. Equivalent dose rates of the neutron field can be measured by it as μSv per hour. Equivalent dose rate results were read via the RADACS software on the system PC. Experimental design is shown in Fig. 1.

RESULTS AND DISCUSSION

Results of Monte Carlo Simulations

We calculated four different parameters by Monte Carlo simulations for four glass compositions. Neutron mass removal cross sections (Σ/ρ) simulated by FLUKA and GEANT4 codes can be seen in Fig. 2. As can be seen in this figure neutron mass removal cross sections decrease with decreasing B_2O_3 contents. Looking at the R-square values, simulation results are around the mean values (R^2 = 0.99978 for FLUKA and R^2 = 0.99934 for GEANT4). Also FLUKA and GEANT4 results are compatible. Table 2 shows the distribution of neutrons as number of inelastic scattered, elastic scattered, captured and transported neutrons after neutrons-glass interactions. In other words, this table shows partitions of 1000000 primary neutrons after neutron-glass interactions. Number of inelastic and elastic scattered neutrons increased with decreasing B_2O_3 percentage. Number of transported

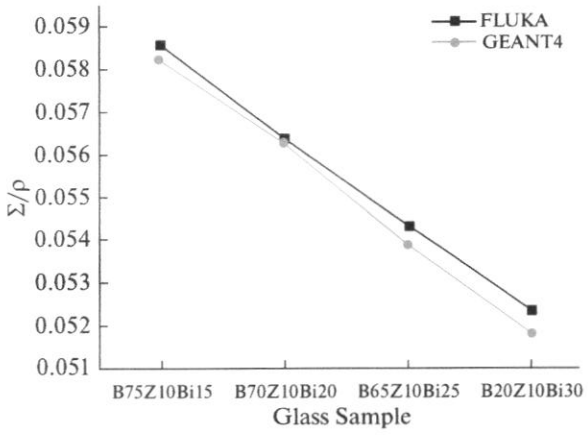

Fig. 2. Neutron mass removal cross sections by Monte Carlo.

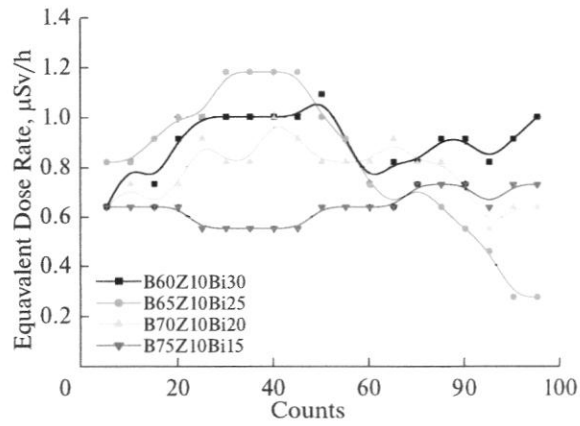

Fig. 3. Equivalent dose rates for glass samples.

neutrons decreased with decreasing B_2O_3 content. B70Z10Bi20 sample has the highest number of captured neutrons value.

Results of Neutron Equivalent Dose Rate Measurements

Neutron equivalent dose rate measurements were performed for four glass samples and an ordinary glass sample which has the same dimensions with others. Figures 3 and 4, show equivalent dose rate values against number of count for this glass system and ordinary glass. For normal glass sample, equivalent dose rate values are varies between 0.46 and 1.27 μSv/*h*. Maximum and minimum equivalent dose rate values for B75Z10Bi15 are 0.55 and 0.73 μSv/*h* respectively. Other three glasses have equivalent dose rate values in the range of 1.18 and 0.27 μSv/*h*. Average dose rate values can be seen in Fig. 5. Looking at this figure, neutron shielding capacities of our samples are higher than ordinary glass sample. Also B75Z10Bi15 is the best neutron shielding glass with 36% difference from ordinary glass sample.

Looking at the results, it seems that there is a harmony between experimental and Monte Carlo results. As the B_2O_3 content increased in the sample, the experimental equivalent dose rate (absorbed by detector) values decreased (Fig. 3) and the mass removal coefficient values, obtained by simulations (Fig. 2), increased. These results support the increase in the number of elastic and inelastic scattered neutrons and the decrease in the number of transported neutrons estimated by Monte Carlo simulations (Table 2). The materials to which neutrons are scattered in high numbers are valuable in terms of neutron shielding quality. That means, neutron shielding performance has increased with increasing B_2O_3 content.

CONCLUSIONS

In this paper, neutron interactions with $(90-x)B_2O_3$–$10ZnO$–xBi_2O_3 (where x = 15, 20, 25 and 30 mol %) glass system were modeled by FLUKA and GEANT4 Monte Carlo codes. Also neutron equivalent dose rate experiments were performed for produced four samples. The experimental and simulation results show that produced glasses including B_2O_3, ZnO, and Bi_2O_3 have higher neutron shielding performance. B75Z10-Bi15 is the finest neutron shielding glass among others. Also neutron shielding performance increases with increasing B_2O_3 content has increased. So, it can be said that produced glasses have fine compact neutron shielding property. This information may be used to produce neutron shielding glass systems in different nuclear applications.

Table 2. Distribution of neutrons as number of inelastic scattered, elastic scattered, captured and transported neutrons after neutrons-glass interactions

	B75Z10Bi15	B70Z10Bi20	B65Z10Bi25	B60Z10Bi30
Inelastic(n, n')	19450	20996	22777	24750
Elastic(n, n)	179398	186114	190107	194050
Capture	20	21	20	17
Transport	801132	792869	787096	781183
Total	1000000	1000000	1000000	1000000

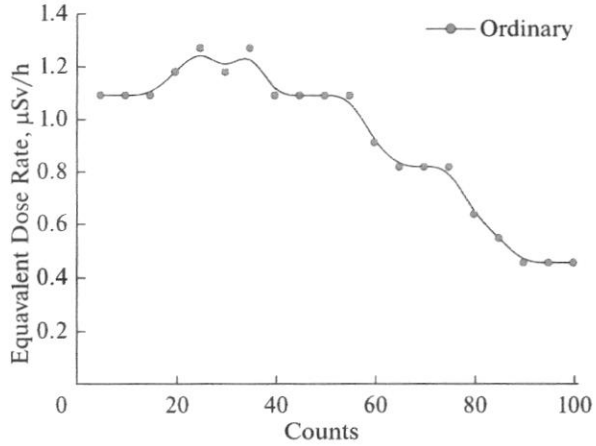

Fig. 4. Equivalent dose rates for ordinary glass.

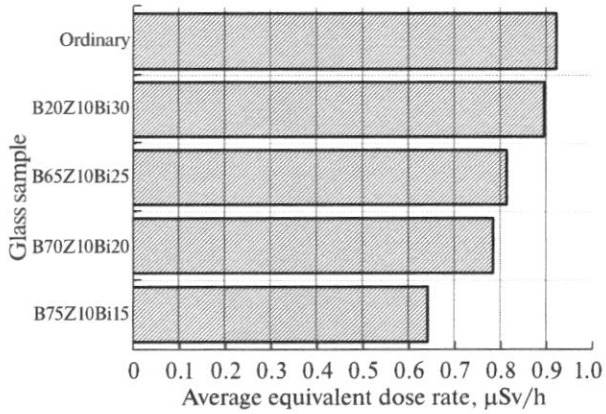

Fig. 5. Average equivalent dose rates for samples.

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