Investigation of Protective Properties of Newly Designed Chromite Ore Dust Filled Glass Fiber Reinforced Epoxy Composite for Neutron Radiation Applications

Bünyamin AYGÜN*, Abdülhalik KARABULUT

Abstract - In this study, chromite dust-filled glass fiber reinforced epoxy matrix composites were designed and produced by a hand-laying technique by varying the chromite dust content from 0% by weight to 15% by weight. To understand the capacity of these composites to reduce fast neutrons, such as effective removal cross-section, half value layer, mean free path, and neutron transmission factor (NTF) was determined theoretically using the Monte Carlo simulation GEANT4 code. All the results found were compared with paraffin, a neutron shielding material. The addition of chromite dust was found to increase both the mechanical and radiation absorption capacities of glass fiber epoxy composites. It was observed that these composite materials exhibited absorption performance at values close to paraffin. In particular, it was determined that the sample containing 15% chromite dust absorbed higher neutron radiation than the other samples. For this reason, these composite materials can be used as shielding materials to prevent neutron radiation leakage that may occur in nuclear reactors, transport, and storage of used radioactive waste, boron neutron treatment units, and activation analysis applications.

Index Terms — Composite; neutron shielding; Geant 4

1 INTRODUCTION

Neutron particles, which are indirect ionizing radiation with high energy, are widely used in industry, nuclear reactors, radiotherapy, nuclear and high energy physics experiments, oil and natural gas exploration, archeology, agriculture, biological studies, detecting defects that may occur in metal castings [1].

Neutron particles, which do not have an electrical charge, have an energetic and highly penetrating property because they are not affected by the coulomb force of the atom. Exposure to neutron radiation can harm human life, the environment, and all living and non-living things. Moreover, direct exposure to this radiation can cause radiation sickness, damage to organs and tissues, DNA mutations, and various cancers in humans [2].

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In order to protect against the harmful effects of radiation, shielding is done against radiation. To make radiation use safe, suitable, effective, and permanent shielding materials are needed [3]. In order to protect the environment and life from unwanted radiation leaks, researchers have designed and produced new protective materials that will stop these leaks or reduce them to the optimum level. These studies continue with increasing interest as radiation gains new usage areas. The materials to be used for radiological shielding have high density, good radiation absorption or attenuation abilities, and low cost. Many materials such as heavy concretes [4], [5], [6], [7] for building materials, glasses [8], [9], [10], [3] , stainless steels [11], [12], [13], [14], for high-temperature environments, alloys [15], [16], [17], with high corrosion resistance, and molecules that can be used for drugs [18], [19], [20], [21], [22], [23], [24] have been designed and produced to provide radiation protection.

Polymers and their composites have been widely used in radiation shielding fields in recent years, as they have superior physical, mechanical, and radiation shielding properties in addition to their resistance to corrosion and heat, lightness, and flexibility. [25]. Moreover, polymers can be used in high ratios with materials with a large atomic number to increase the radiation-shielding capacity of composites. Many polymer-containing composite materials have been developed and used in neutron shielding applications. An epoxy-based (0.5%, 10%, 12%, and 15%) boron-added composite material with a thickness of 0.5 cm, containing different stages, was produced. This content was brought to a thickness of 2 cm and the shielding of thermal neutrons was made. Then, it was determined that the fast neutrons stopped when the composition contained 12% boron. As a result, cadmium was added to the combination (0.10%, 15%, 20%, and 25%) to stop the fast neutrons [26]. Neutron and gamma radiations can occur together as a result of most nuclear reactions, so materials are needed to stop both types of radiation. Boron and tungsten reinforced polyethylene (PE) based composite materials, which have the ability to absorb both of these types of radiation, have been designed and produced [27]. New polymer materials have been developed for gamma radiation shielding applications with a high content of polypropylene with varying concentrations of tincal. It has been determined that increasing the tincal ratio in the composition also increases the gamma radiation absorption capacity of the material [28]. New types of composite materials with densities ranging from 1.06 to 1.59 g/cm3 were produced by adding colemanite at different rates of 5, 15, 25, and 35 wt% into polypropylene. Gamma rays absorption parameters were determined. According to the results, it was determined that the colemanite mineral composite increased the radiation absorption ability at certain rates [29]. In this study, unlike other studies in the literature, chrome powder filled glass fiber reinforced epoxy matrix composites with high neutron radiation shielding capacity were designed and produced. To understand the neutron attenuation capacity of these composites, neutron radiation attenuation parameters were theoretically calculated using the Monte Carlo simulation GEANT4 code.

2 MATERIAL and METHOD

2.1 Materials and sample preparation

In the study, composite materials with different chemical content were designed and produced by using epoxy-based chromite, powder, and glass fiber. Epoxy and glass fiber were purchased commercially, and chromite ore was procured from a chrome mine in Kayseri Province, Yahyalı District (Turkey). Glass fiber used chemical content is given in Table 1. The particle size of the chromite was prepared with a maximum grain size of approximately 250 micrometers (μm), it is given analysis results in Table 2. The weight mass percent ratios of all new composite samples were calculated using the Monte Carlo simulation program Geant4 code.

The mixing ratios of the composite samples by weight are designed to have a high neutron radiation absorption capacity. The contents of which are given in Table 3, epoxy and chromite powder in the determined proportions were mixed for 10 minutes by using a mixer until the mixture became homogeneous, and the hardener of the epoxy was added at the rate of 1/3 and the mixing process was continued. When the mixture reached the desired viscosity and homogeneity, it was prepared as glass fiber + homogeneous mixture + glass fiber + homogeneous mixture + glass fiber using the hand-laying method in previously prepared 5x2 cm molds, compressed under a certain pressure and left to dry at room temperature.

TABLE 2

ρ=2.2 g/cm^3	SiO_2	Al_2O_3	B_2O_3	CaO	MgO	Na_2O	TiO_2	Fe_2O_3	F
	52	15	10	18	4	1	0.5	0.4	0.7

GLASS FIBRE CHEMICAL COMPOSITION (%)

TABLE 2

CHROME ORE ANALYSIS RESULT

ρ= (4.5 g/cm^3)	Cr_2O_3	MgO	Al_2O_3	Cr	SiO_2	S	P	Fe
Weight fraction	53.190	16.800	11.150	12.560	2.720	0.007	0.005	3.568

TABLE 3

WEIGHT FRACTIONS OF THE CONCRETE TYPES

Sample	CS1 (ρ=1.16 g/cm^3)	CS2 (ρ=1.78 g/cm^3)	CS3 (ρ=2.04 g/cm^3)	CS4 (ϱ=2.10 g/cm3)
Epoxy	60	55	50	45
Chromite dust($FeCr_2O_3$)	0	5	10	15
Glass fibre	40	40	40	40

CS: Composite sample

1. Neutron radiation transmission

The fast neutron removal attenuation cross section (ΣR) gives the probability of the initial collision of neutrons with a certain energy with the target material [30]. The removal cross section also separates neutrons that have a fast or fission energy level from the group of neutrons that can collide with the material from the group of neutrons that can pass without collision. If the target material with which the neutron interacts is in a mixture structure, each element forming the mixture has an effect on this cross section. Considering this important situation, the effective removal cross section value can be calculated as follows.

$$\sum R = \sum \left(\frac{\sum R}{\rho}\right) i \quad (1)$$

It is desirable that the number of neutrons passing through the neutron shielding material is low. The number of neutrons that pass through the material with which the neutrons interact and cannot pass is expressed by the Neutron transmission factor (NTF). This parameter is generally used in shielding material selections and this data can be calculated with the following equation [31], [32].

$$NTF = \frac{I}{I_0} \quad (4)$$

I is the number of neutrons passing through the interaction material, and I_0 is the number of neutrons coming into the interaction material. thickness (HVL), and this value is calculated using the equation below.

$$HVL = \frac{ln2}{\sum R} \quad (2)$$

The mean free path (λ) indicates the average distance that neutrons can travel between two successive collisions in a material with which they interact. This parameter gives information about the shielding capacity of the material in shielding applications and this value can be calculated with the equation as following.

$$\lambda = 1/\sum R \quad (3)$$

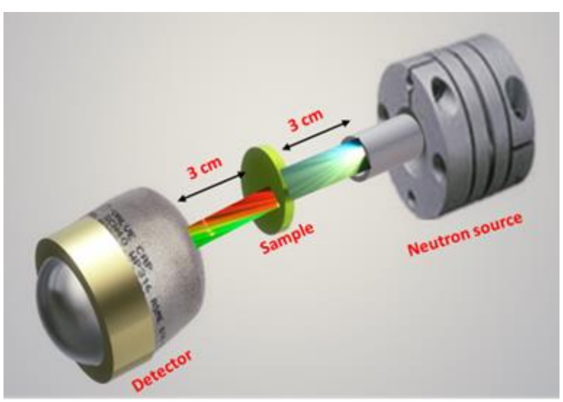

Fig. 1. GEANT4 simulation geometry

$\sum R$; is the fast neutron effective removal cross-section of the interaction material and ρ; stands for density.

The thickness of the shielding material, which halves the energy or number of neutrons when they pass through an shielding material, is expressed by the half value

3. RESULTS AND DISCUSSION

3.1 Neutron attenuation properties

Effective removal cross section values have been calculated of all samples for Fast neutron 3-14 MeV energy area by Geant4 code and obtained results given in Figure 2.

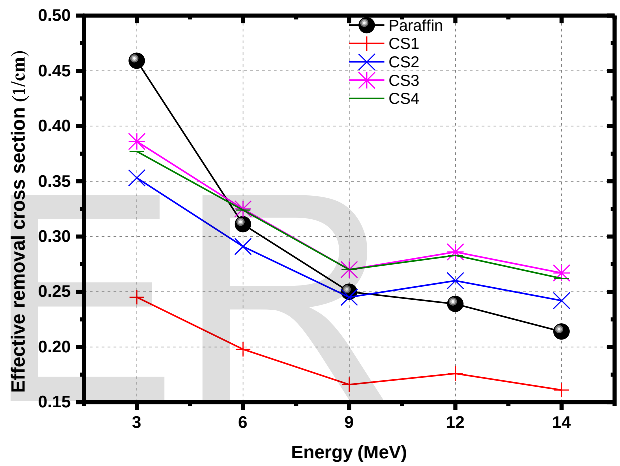

Fig. 2. Comparison of effective removal cross section values of all samples

When Table 4 and Fig 2 are examined, it can be seeing all CS samples have good neutron shielding capacity in 3-14 MeV energy levels. If effective removal cross section value of any material is big, this sample has good neutron absorption ability. According to, reference sample paraffin has big removal cross section value in every energy than CS1 sample, but it is low from other samples. According to effective removal cross section value, CS2, CS3, and CS4 samples have high absorption value than paraffin.

It is the average mean free path (MFP) at which energy of a known neutron can travel without any interaction in the shielding material. This is parameter important for neutron shielding applications.

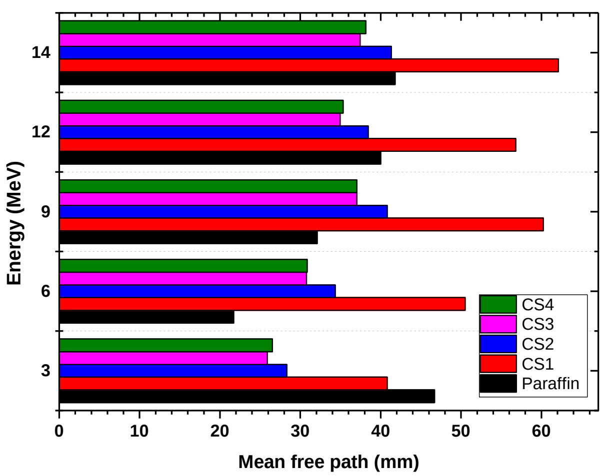

Fig. 3. Comparison of mean free path values of all samples

If the any materials have low mean free path value, it is have neutron radiation shielding capability is high [33]. As shown in Table 4 and Fig.3. MFP of the CS2, CS3 and CS4 samples are low than the paraffin in all study energies.

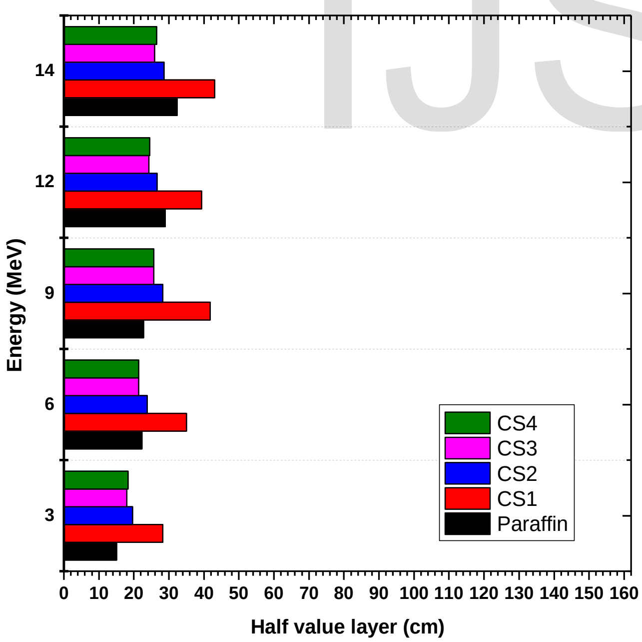

Fig. 4. Comparison of half value layer values of all samples

Half value layer (HVL) was calculated for all samples and the found results were presented in Table 4 and Fig. 4. The half value layer is the thickness of the shielding material required to half reduce the intensity of the neutron particles from passing shielding material.

Half value layer is low indicating that any shielding material has good neutron radiation absorption properties [34]. According to Table 4, and Fig. 4. HVL of every CS sample has a low value. But the reference sample paraffin has low value than CS1 and at 3MeV energy other CS samples.

The neutron transmission factor is useful a parameter in the study of neutron shielding research. That is desired low this factor value property for in research effect shielding sample. When carefully examined Table 4, it can be seen that, CS2, CS3, and Cs4 samples have low transmission factor than the reference sample paraffin except for 3 MeV energy. According to these samples have the super shielding ability of paraffin. But only the CS1 sample has low shielding ability than paraffin.

TABLE 4
COMPARISON OF FAST NEUTRON ATTENUATION PARAMETERS FOR 2 CM SAMPLE THICK AND 10^5 NEUTRONS

Sample code	Dose Energy (MeV)	Half Value Layer (cm)	Mean Free Path λ (mm)	Neutron Transmission Factor (I/I_0)	Effective Removal cross section (cm^{-1})
P	**3**	15.098±0.120	21.7±0.201	0.63013	0.459
	6	22.282±0.200	32.1 ±0.315	0.73199	0.311
	9	22.720±0.201	40.0 ±4.457	0.77873	0.250
	12	28.999±0.580	41.8 ±0.176	0.78675	0.239
	14	32.383±0.321	46.7 ±0.581	0.80661	0.214
CS1	**3**	28.225±0.208	40.81 ±0.480	0.78239	0.245
	6	35.00±0.350	50.50±0.575	0.82013	0.198
	9	41.746±0.417	60.24±0.602	0.84622	0.166
	12	**39.375±0.397**	**56.81±0.568**	**0.83857**	**0.176**
	14	43.043±0.430	62.11±0.601	0.85076	0.161
CS2	**3**	19.631±0.190	28.32±0.281	0.70239	0.353
	6	23.814±0.234	34.36±0.345	0.74702	0.291
	9	28.285±0.282	40.81±0.401	0.78197	0.245
	12	**26.653±0.266**	**38.46±0.382**	**0.77069**	**0.260**
	14	28.636±0.286	41.32±0.417	0.78472	0.242
CS3	**3**	17.953±0.178	25.90±0.259	0.67952	0.386
	6	21.323±0.212	30.76±0.305	0.72226	0.325
	9	25.666±0.256	37.03±0.370	0.76323	0.270
	12	**24.230±0.230**	**34.96±0.348**	**0.75118**	**0.286**
	14	25.955±0.295	37.45±0.378	0.76535	0.267
CS4	**3**	18.381±0.185	26.52±0.267	0.68589	0.377
	6	21.388±0.219	30.86±0.304	0.72261	0.324
	9	25.666±0.254	37.03±0.371	0.76298	0.270
	12	**24.487±0.241**	**35.33±0.356**	**0.75319**	**0.283**
	14	26.450±0.265	38.16±0.380	0.76928	0.262

CS: Composite sample, ***P:*** Paraffin

1 CONCLUSION

In this study, four different content new shielding composite samples (CS) were developed and produced for fast neutron. Neutron shielding parameters such as effective removal cross-section, mean free path, half value layer, and transmission factors were calculated for all samples with the GEANT4 code. Obtained all results were compared with reference sample paraffin to determine the shielding capacity of new composite samples. It is found that except CS1 sample other CS2, CS3, and CS4 samples have high shielding capacities from reference sample paraffin in the 6-14 MeV energy area. According to these results, all-new composite samples can be used as shielding material against fast neutron leaks, in nuclear reactors, in the used of radioactive waste storage, in radiotherapy rooms, and in nuclear shelters.

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References

[1] B. Aygün, Improving Neutron and Gamma Radiation Shielding Properties of Polysiloxane /Cr_2O_3-Fe_2O_3 Added Composite Material Bünyamin AYGÜN, (2019). http://www.ijser.org (accessed October 23, 2022).

[2] B. Aygün, A. Akıncıoğlu, M.I. Sayyed, A. Karabulut, Investigation of some drug active substances able to protect against radiation damage with experimental and Monte Carlo calculations, Radiation Physics and Chemistry. 191 (2022) 109850. doi:10.1016/J.RADPHYSCHEM.2021.109850.

[3] Y. Al-Hadeethi, M.I. Sayyed, A.Z. Barasheed, M. Ahmed, M. Elsafi, Fabrication of Lead Free Borate Glasses Modified by Bismuth Oxide for Gamma Ray Protection Applications, Materials 2022, Vol. 15, Page 789. 15 (2022) 789. doi:10.3390/MA15030789.

[4] B. Aygün, E. Şakar, O. Agar, M.I. Sayyed, A. Karabulut, V.P. Singh, Development of new heavy concretes containing chrome-ore for nuclear radiation shielding applications, Progress in Nuclear Energy. 133 (2021).

doi:10.1016/j.pnucene.2021.103645.

[5] B. AYGÜN, Developed and Produced New Laterite Refractory Brick Samples Protective for Gamma and Neutron Radiation Using GEANT4 Code, Gümüşhane Üniversitesi Fen Bilimleri Dergisi. 10 (2020) 1–6. doi:10.17714/GUMUSFENBIL.571726.

[6] D. Sariyer, R. Küçer, Effect of Different Materials to Concrete as Neutron Shielding Application, 137 (2020). doi:10.12693/APhysPolA.137.477.

[7] Eastern Anatolian Journal of Science » Submission » Development and Production of High Heat Resistant Heavy Concrete Shielding Materials for Neutron and Gamma Radiation, (n.d.). https://dergipark.org.tr/en/pub/eajs/issue/40091/467325 (accessed July 13, 2022).

[8] B. Aygün, E. Şakar, A. Karabulut, B. Alım, M.I. Sayyed, V.P. Singh, N.Y. Yorgun, Ö.F. Özpolat, Development of SiO2 based doped with LiF, Cr2O3, CoO4 and B2O3 glasses for gamma and fast neutron shielding, Radiochimica Acta. (2020). doi:10.1515/ract-2020-0067.

[9] V.P. Singh, N.M. Badiger, Shielding efficiency of lead borate and nickel borate glasses for gamma rays and neutrons, Glass Physics and Chemistry. (2015). doi:10.1134/S1087659615030177.

[10] M.I. Sayyed, G. Lakshminarayana, I. V. Kityk, M.A. Mahdi, Evaluation of shielding parameters for heavy metal fluoride based tellurite-rich glasses for gamma ray shielding applications, Radiation Physics and Chemistry. (2017). doi:10.1016/j.radphyschem.2017.05.013.

[11] B. Aygün, T. Korkut, A. Karabulut, High Performance Shielding Material to Prevent Radiation Leaks, (n.d.).

[12] B. Aygün, E. Şakar, T. Korkut, M.I. Sayyed, A. Karabulut, M.H.M. Zaid, Fabrication of Ni, Cr, W reinforced new high alloyed stainless steels for radiation shielding applications, Results in Physics. 12 (2019). doi:10.1016/j.rinp.2018.11.038.

[13] B. Alım, E. Şakar, İ. Han, M.I. Sayyed, Evaluation the gamma, charged particle and fast neutron shielding performances of some important AISI-coded stainless steels: Part II, Radiation Physics and Chemistry. (2020). doi:10.1016/j.radphyschem.2019.108454.

[14] B. Aygün, High alloyed new stainless steel shielding material for gamma and fast neutron radiation, Nuclear Engineering and Technology. (2020). doi:10.1016/j.net.2019.08.017.

[15] T. Korkut, B. Aygün, Ö. Bayram, A. Karabulut, Study of neutron attenuation properties of super alloys with added rhenium, Journal of Radioanalytical and Nuclear Chemistry. (2015). doi:10.1007/s10967-015-4063-z.

[16] N. Ekinci, F.I. El-Agawany, K.A. Mahmoud, A. Karabulut, B. Aygün, E. Yousef, Y.S. Rammah, Synthesis, physical properties, and gamma–ray shielding capacity of different Ni-based super alloys, Radiation Physics and Chemistry. 186 (2021). doi:10.1016/j.radphyschem.2021.109483.

[17] B. Aygün, Neutron and gamma radiation shielding Ni based new type super alloys development and production by Monte Carlo Simulation technique, Radiation Physics and Chemistry. 188 (2021) 109630. doi:10.1016/J.RADPHYSCHEM.2021.109630.

[18] B. Aygün, B. Alaylar, K. Turhan, E. Şakar, M. Karadayı, M.I.A. Al-Sayyed, E. Pelit, M. Güllüce, A. Karabulut, Z. Turgut, B. Alım, Investigation of neutron and gamma radiation protective characteristics of synthesized quinoline derivatives, Https://Doi.Org/10.1080/09553002.2020.1811421. 96 (2020) 1423–1434. doi:10.1080/09553002.2020.1811421.

[19] B. Alaylar, B. Aygün, K. Turhan, G. Karadayi, E. Şakar, V.P. Singh, M.I. Sayyed, E. Pelit, A. Karabulut, M. Güllüce, Z. Turgut, M. Isaoglu, Characterization of gamma-ray and neutron radiation absorption properties of synthesized quinoline derivatives and their genotoxic potential, Radiation Physics and Chemistry. 184 (2021) 109471. doi:10.1016/J.RADPHYSCHEM.2021.109471.

[20] M.I. Sayyed, S.A.M. Issa, S.H. Auda, Assessment of radio-protective properties of some anti-inflammatory drugs, Progress in Nuclear Energy. (2017). doi:10.1016/j.pnucene.2017.07.003.

[21] B. Alaylar, B. Aygün, K. Turhan, M. Karadayı, E. Cinan, Z. Turgut, G. Karadayı, M.I.A. Al-Sayyed, M. Güllüce, A. Karabulut, Investigation of radiation protective features of azadispiro derivatives and their genotoxic potential with Ames/Salmonella test system, Https://Doi.Org/10.1080/09553002.2022.2087930. (2022). doi:10.1080/09553002.2022.2087930.

[22] B. Aygün, B. Alaylar, A. Aklncloğlu, B. Allm, E.T. Kocaman, M. Karadayl, M.I. Abu Al-Sayyed, S. Aksu, E. Sakar, Ö.F. Özpolat, S. Göksu, M. Güllüce, A. Karabulut, Determination of radioprotective and genotoxic properties of sulfamide derivatives, Radiochimica Acta. 109 (2021) 891–904. doi:10.1515/RACT-2021-1088/MACHINEREADABLECITATION/RIS.

[23] B. Aygün, M. Ş Entürk, E. Cinan, Ö. Ş Imsek, M.I. Abu Al-Sayyed, A. Karabulut, Determination of biological radioprotective characteristics of some natural organic compounds for radiation shielding applications, Radiochimica Acta. (2022). doi:10.1515/RACT-2022-0028/MACHINEREADABLECITATION/RIS.

[24] B. Aygün, A. Karabulut, Eastern Anatolian Journal of Science Cite this article as: B. Aygün and A. Karabulut, The Interaction of Neutron and Gamma Radiation With Some Cancer Drug Effect Ingredients: A Simulation Study, Eastern Anatolian Journal of Science. VI (2020) 35–43.

[25] C. V. More, H. Alavian, P.P. Pawar, Evaluation of gamma-ray attenuation characteristics of some thermoplastic polymers: Experimental, WinXCom and MCNPX studies, Journal of Non-Crystalline Solids. 546 (2020). doi:10.1016/j.jnoncrysol.2020.120277.

[26] M.A. Ibrahim, K.H. Mahdi, Design and manufacture of neutrons shield from composite materials, AIP Conference Proceedings. 2190 (2019) 020047. doi:10.1063/1.5138533.

[27] M. Arif Sazali, N. Khair Alang Md Rashid, K. Hamzah, F. Mohamad Idris, M. Syahir Sarkawi, N. Syazwani Mohd Ali, Polyethylene composite with boron and tungsten additives for mixed radiation shielding, IOP Conference Series: Materials Science and Engineering. 1231 (2022) 012010. doi:10.1088/1757-899X/1231/1/012010.

[28] N. Ekinci, K.A. Mahmoud, S. Sarıtaş, B. Aygün, M.M. Hessien, I. Bilici, Y.S. Rammah, Development of Tincal based polypropylene polymeric materials for radiation shielding applications: Experimental, theoretical, and Monte Carlo investigations, Materials Science in Semiconductor Processing. 146 (2022) 106696. doi:10.1016/J.MSSP.2022.106696.

[29] N. Ekinci, K.A. Mahmoud, B. Aygün, M.M. Hessien, Y.S. Rammah, Impacts of the colemanite on the enhancement of the radiation shielding capacity of polypropylene, Journal of Materials Science: Materials in Electronics 2022 33:25. 33 (2022) 20046–20055. doi:10.1007/S10854-022-08822-5.

[30] J.M. An, H. Lin, E.Y.B. Pun, D.S. Li, Evaluation of gamma and neutron shielding capacities of tellurite glass system with Phy-X simulation software, Physica B: Condensed Matter. 634 (2022). doi:10.1016/J.PHYSB.2021.413433.

[31] H. Schober, An introduction to the theory of nuclear neutron scattering in condensed matter, Journal of Neutron Research. (2014). doi:10.3233/JNR-140016.

[32] H. Sahadath, A.S. Mollah, K.A. Kabir, M. Fazlul Huq, Calculation of the different shielding properties of locally developed ilmenite-magnetite (I-M) concrete, Radioprotection. 50 (2015) 203–207. doi:10.1051/RADIOPRO/2015005.

[33] A Study on the Gamma Radiation Protection Effectiveness of Nano/Micro-MgO-Reinforced Novel Silicon Rubber for Medical Applications | Mendeley, (n.d.). https://www.mendeley.com/search/?query=A Study on the Gamma Radiation Protection Effectiveness of Nano/Micro-MgO-Reinforced Novel Silicon Rubber for Medical Applications (accessed November 6, 2022).

[34] W. Chaiphaksa, S. Yonphan, E. Kalkornsurapranee, A. Tuljittraporn, S. Kothan, S. Kaewjaeng, C. Kedkaew, J. Kaewkhao, Photon, charged particles, and neutron shielding properties of natural rubber/ SnO2 composites, Radiation Physics and Chemistry. (2022) 110622. doi:10.1016/J.RADPHYSCHEM.2022.110622.