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Improving Neutron and Gamma Radiation Shielding Properties of Polysiloxane /Cr_2O_3 – Fe_2O_3 Added Composite Material

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Abstract-Polysiloxanes ($C_8H_{24}O_4Si_4$) are used commonly due to low and high temperature performance, electrical insulation or in high resistance to weathering and chemical corrosion in area such as textile industry, cosmetics, medical/pharmaceutical preparations, paper coatings, defoamers, paints, coatings, waxes, mechanical fluids. The polysiloxanes to use in the radiation shielding field, in this study, eight different composite shield materials have been developed and produced. To increasing radiation shielding capacities of composite materials certain proportions Cr_2O_3 and Fe_2O_3 have been add. Mixing ratios of composite materials have been determined by using the GEANT4 code of the Monte Carlo simulation program. Neutron shielding parameter the total macroscopic cross section have been theoretically calculated by using GEANT4 simulation code. In addition to, experimental absorbed dose measurements have been carried out by using 4.5 MeV energy ^{241}Am-^{9}Be neutron source and BF_3 gas detector. To determined gamma radiation shielding parameters, the mass, linear attenuation coefficient and half-value layer (HVL) have been calculated by using VinXCom software. Obtained results have been compared with paraffin, conventional concrete. The results show that both absorbed ability radiation of new composites much better than these reference materials and with the the increment of Cr_2O_3 and Fe_2O_3 added, shielding properties increase.

Index Terms—Heavy concrete; gamma; neutron cross section; Geant 4 Monte Carlo code.

1 INTRODUCTION

THE use of radiation is increasing in areas such as industry, energy, medicine, agriculture, military and space research application. A radiation leak may occur at the these applications and this quite harmful for people health. Effective shielding material must used to protect from these damages. Therefore, there is a need for new protective materials which are non-toxic, lightweight, chemical and environmental corrosion resistant and have high radiation absorption capability [1]. Polymer-based lightweight, nontoxcit new composite materials such as polypropylene, polystyrene, polyethylene, polystyrene, and rubber are widely used for the radiation shielding. New radiation shielding materials was developed by adding different content materials to the polymers [2-3]. Polysiloxanes ($C_8H_{24}O_4Si_4$) or silicones are a type of polymer consisting of silicon-oxygen atoms with organic and methyl groups attached to silicon atoms.

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Because of their biocompatibility and bio-resistance, polysiloxanes are widely used in health care [4]. They are widely used as elestometric materials but it is need to be reinforced with different materials to enhance some properties. Polysiloxanes can be exhibit different properties wide range of temperatures. Because of thermal, chemical and biological properties these polymers are used in the varied technological applications [5]. To avoid radiation leaks shielding material such as heavy concretes, lead glass and stainless steel are generally used in the radiation applications. But they are neither aesthetic nor very durable to corrosion so Polysiloxanes based new composite materials must developed for radiation shielding. Polysiloxanes type added composites has become to be recently used due to its excellent chemical and mechanical durable properties as a shielding material in the nuclear applications. In the treatment of some head and neck lesions with high dosage radiation at the thin radiochromic film strips and soft-tissue-simulating plastic (Polysiloxanes) shielding materials have been used at the spectrophotometer and metal powders such as (Ag–Cu and Sn–Sb) were added to this composite [6]. Nowadays, silicone (Polysiloxanes) rubber is used in many varying indüstrial applications, because it has low density, excellent flexibility, thermal and corrosion resistance [7-8]. Bismuth silicone (Polysiloxanes) rubber materials have been developed for gamma radiation shielding. It has been observed, radiation shielding ability increases with bismuth material in silicone (Polysiloxanes) rubber [9]. Therefore, in this study, different content new polymer composite materials have been developed and designed for radiation shielding using Polysiloxanese as a polymer matrix and chromium oxide (Cr_2O_3) and hematite (Fe_2O_3) which has a high cross section number.

2 SHIELDING THEORY

2.1 Total Macroscopic Cross Section

Neutrons interaction with any matter may be make either scattering or absorption form, if this scattering is inelastic scattering, there may be a change in the energy of the neutron. The possibility of these interactions is both dependent on the energy of the neutrons and on the properties of the target material nucleus with which it is interacting. The area of interaction of a single target nucleus where a neutron particle interacts is expressed by a microscopic cross-section (σ). If these interactions of neutrons occur with heavy materials such as concrete, the probability of interaction is expressed by total macroscopic cross section (∑) and the relationship between the two is given by the formulas below.

$$\sum = N\sigma \tag{1}$$

$$N = \frac{\rho}{A} N_A \tag{2}$$

Where, N; atomic density of the interaction material, ϱ; density of the interaction material and NA; the number of Avogadro is refers to. Total macroscopic cross section is calculated with equation 3.

$$\Sigma \text{ Total} = \Sigma \text{ scattering} + \Sigma \text{ absorption} + \Sigma \text{ capture} + \Sigma \text{ fission} + \ldots \tag{3}$$

If the total macroscopic cross section high value of the target material, it means that the neutron stopping power is good [10].

2.2 The Linear and Mass Attenuation Coefficient

The linear attenuation coefficient (μ, cm^{-1}) indicates the amount of x-ray or gamma ray absorbed or scattered per unit thickness of the target material. This parameter displays the probability of a photon scattered or absorbed from the nucleus or electron of an atom in a volume of a cm^3 target material. A linear attenuation coefficient is calculated based on the density of a material, but the mass attenuation coefficient (μm, cm^2/g) is more commonly used and it is independent of the density of the material.

The mass attenuationcoefficient of a material is given in μ / ϱ and ϱ indicates the density of the target material.

Beer-Lambert exponential attenuation law are used in the shielding studies to gamma ray mass attenuation coefficient of the material as follows.

$$I = I_0 e^{-(\mu/\rho)t} \tag{4}$$

linear attenuation coefficient (μ) through the following relation: Where is, I and I_0 are the

incaming and transmitted beam intensities, (cm^{-1}) and

Half Value Layer (HVL)

The half value layer (HVL) is the thickness that reduces the intensity of the gamma ray coming on the material in half and it is commonly used in the shielding calculates. Half value layer (HVL) is used for the predict of the shielding material thickness [11-12-13]. This parameter inversely proportional to the linear attenuation coefficient and can be calculated as follow:

$$HVL = \frac{\ln(2)}{\mu} = \frac{0.639}{\mu} \tag{5}$$

3. MATERIAL and METHOD

3.1 Monte Carlo Simulation Method (Geant4 Code)

pre-experimental events are provided great eases in nuclear and particle physics. Monte Carlo (MC) simulation Geant 4 code is commonly used for this purpose. Geant4 is a general purpose Monte Carlo simulation tool, it is used to estimate the interaction of particles and radiation with matter. It is used in variety areas such as high energy and nuclear physics, space engineering, medical applications, material science,radiation protection and security. Detailed information can be found at www.Geant4.org.

3.2 Sample Mix Design and Preparation

In the enrichment of polysiloxanes chromium oxide (Cr_2O_3) and hematite (Fe_2O_3). Mixing ratios of the materials to be incorporated into the polysiloxanes were determined by the Monte Carlo simulations and the composite materials were modelled. The polysiloxanes and additives the admixture were mixed until homogeneous. The obtained homogeneous mixture was poured into molds of 10 X10 X 4 cm. After casting, the molds were subjected to vibration in a vibration table and the air trapped in the mold was removed. The samples were then dried at 23^0C ±3^0C for 7 days. The chemical contents of the produced new composite samples are shown in Table 1.

TABLE 1
CHEMICAL COMPOSITIONS of COMPOSITE SAMPLES (MASS %)

Sample Code	Dencity (g/cm3)	Polysiloxanes (C8H24O4Si4)	Chromium Oxide (Cr2O3)	Hematite (Fe2O3)
CM1	3.098	50	10	40
CM2	3.097	50	15	35
CM3	3.096	50	20	30
CM4	3.095	50	25	25
CM5	3.094	50	30	20
CM6	3.093	50	35	15
CM7	3.092	50	40	10

CM; Composite material

241Am/Be neutron source (10 mCi) which average emits 2–11 MeV neutron radiation and Canberra brand neutron detector were used for the neutron measurements. Experimental design is shown in Fig. 1

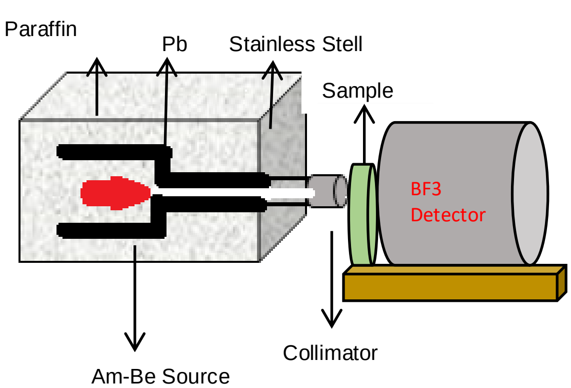

Fig 1. Experimental designee.

4. Results and Discussion

4.1 Neutron Shielding Properties

In this study, seven different types of polysiloxanes based composite samples are designed and produced; the chemical compositions are shown in Table 1. This type polysiloxanes based composite materials were content chromium 10-40 wt. %. and hematite It has been seen that by adding chromium oxide, hematite the composite materials structure has increased both heat, mechanical resistance and radiation shielding ability.

Shown in the Table 2 neutron Total Macroscopic Cross Section were theorical calculated with Geant4 code for all studied materials and obtained results were compared with paraffin and conventional concrete which are used in the shielding radiation studies generally. According to paraffin and conventional concrete, composite samples have high total macroscopic cross sections were determined.

TABLE 2

TOTAL MACROSCOPIC CROSS SECTIONS VALUES of THE SAMPLES 1CM THIC

Sample code	Total Macroscopic Cross Section (cm^{-1})
C.C	0.1671
P	0.2014
Pol	0.1359
CM1	0.2961
CM2	0.2915
CM3	0.2912
CM4	0.2928
CM5	0.2921
CM6	0.2944
CM7	0.2946

CC:Conventinal concrete, P: Paraffin, Pol: Polysiloxanes

Total macroscopic section values of the all samples have been calculated for the 4.5 MeV energy neutrons using the Monte Carlo simulation Geant4 code. As shown in Table 2 and Fig.2. calculated results have been compared with conventional concrete and paraffin which, they are usually used as a shielding material in nuclear applications.

The calculations also showed that, added additives doubled the shielding capacity of pure polysiloxanes.

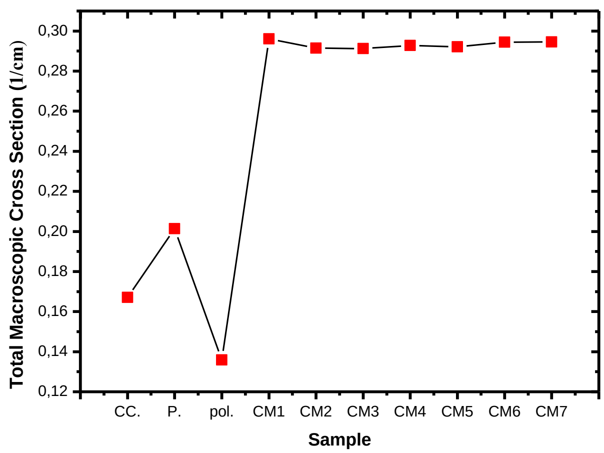

Fig. 2. Total Macroscopic Cross Section (cm^{-1})

Neutron absorption dose measurement experiment has been carried out. Absorbed dose results have been calculated and these values have been compared with paraffin and conventional concrete. Obtained results are shown that Table 3.

TABLE 3
ABSORBED DOSE VALUES OF SAMPLES

Sample code	Absorbed equivalent dose rates (µSv/h) by the samples	Absorbed dose percentage of the samples (%)
Backround	1.2805	-
C.C	0.2701	21.09
P	0.3019	23.57
pol	0.2071	16.17
CM1	0.3758	29.34
CM2	0.3605	28.15
CM3	0.3970	31.00
CM4	0.4018	31.37
CM5	0.4067	31.76
CM6	0.4086	31.90
CM7	0.4099	31.01

As shown in Table 3 it was found that the dose of 1.2805 (µSv/h) emitted from the source was absorbed conventional concrete (29.00 %), paraffin (23.57 %), polysiloxanes (29.34 %), CM1 (29.34 %), CM2 (28.15 %) CM3 (31.00 %), CM4 (31.37 %), CM5 (31.76 %), CM6 (31.90 %) and CM7 (31.01 %) of the dose.

According to the results, all composite samples show perfect shielding ability for neutron radiation, thus all composite samples having high corrosion and oxidation resistance can be used instead of paraffin and conventional concrete, in nuclear applications.

4.2 Gamma-Ray Shielding Properties

Theoretical gamma-ray shielding parameters mass, attenuation coefficient (MAC), linear attenuation coefficient (LAC), and half value layer (HVL) of the samples have been calculated by using the Beer-Lambert law and Win-XCOM software in the continuous energy range from 0.015 to15 MeV [14-15]. Obtained results were compared with conventional concrete and paraffin it shown respectively Table 4-5-6. According to Table 4 when MAC values are smaller than paraffin of composite samples in the at the low energy region. But MAC values are bigger than conventional concrete all energy regions. As shown in Table 5, Linear attenuation coefficient values of all composite samples are higher than both paraffin and conventional concrete. These Half Value Layer values are given in Table 6. Accordingly, HVL values of all samples are smaller than paraffin and conventional concrete. These results are proof that the higher the gamma radiation attenuation capacity of the samples. Radiation retention of all samples is better than paraffin and conventional concrete.

TABLE 4

MASS ATTENUATION COEFFICIENT (cm2/g) of THE SAMPLES

Energy (MeV)	CM1	CM2	CM3	CM4	CM5	CM6	CM7	Conventional Concrete	Paraffin
1.50E-02	21.679	21.248	20.817	20.386	19.955	19.524	19.094	7.054	0.721
2.00E-02	9.736	9.536	9.336	9.135	8.935	8.735	8.534	3.105	0.427
3.00E-02	3.154	3.088	3.023	2.957	2.892	2.826	2.761	1.048	0.276
4.00E-02	1.458	1.429	1.400	1.371	1.342	1.313	1.284	0.541	0.235
5.00E-02	0.836	0.821	0.806	0.790	0.775	0.760	0.745	0.358	0.217
6.00E-02	0.555	0.546	0.537	0.528	0.519	0.510	0.501	0.275	0.206
8.00E-02	0.325	0.321	0.317	0.313	0.309	0.305	0.301	0.204	0.191
1.00E-01	0.237	0.235	0.233	0.231	0.229	0.227	0.225	0.175	0.180
1.50E-01	0.163	0.162	0.161	0.161	0.160	0.159	0.159	0.143	0.161
2.00E-01	0.136	0.136	0.136	0.135	0.135	0.135	0.135	0.127	0.147
3.00E-01	0.112	0.112	0.112	0.112	0.112	0.112	0.112	0.109	0.128
4.00E-01	0.099	0.099	0.099	0.099	0.099	0.099	0.098	0.097	0.114
5.00E-01	0.090	0.090	0.090	0.089	0.089	0.089	0.089	0.088	0.104
6.00E-01	0.083	0.083	0.083	0.082	0.082	0.082	0.082	0.081	0.097
8.00E-01	0.072	0.072	0.072	0.072	0.072	0.072	0.072	0.071	0.085
1.00E+00	0.065	0.065	0.065	0.065	0.065	0.065	0.065	0.064	0.076
1.50E+00	0.053	0.053	0.053	0.053	0.053	0.053	0.053	0.052	0.062
2.00E+00	0.046	0.046	0.046	0.046	0.046	0.046	0.046	0.045	0.053
3.00E+00	0.037	0.037	0.037	0.037	0.037	0.037	0.037	0.037	0.042
4.00E+00	0.033	0.033	0.033	0.033	0.033	0.033	0.033	0.032	0.036
5.00E+00	0.030	0.030	0.030	0.030	0.030	0.030	0.030	0.029	0.032
6.00E+00	0.028	0.028	0.028	0.028	0.028	0.028	0.028	0.027	0.029
8.00E+00	0.026	0.026	0.026	0.026	0.026	0.026	0.026	0.024	0.025
1.00E+01	0.025	0.025	0.025	0.025	0.025	0.025	0.025	0.023	0.022
1.50E+01	0.024	0.024	0.024	0.024	0.023	0.023	0.023	0.021	0.019

CM; composite material

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TABLE 5
LINEAR ATTENUATION COEFFICIENT (cm^{-1}) VALUES OF SAMPLES

Energy (MeV)	CM1	CM2	CM3	CM4	CM5	CM6	CM7	Conventional Concrete	Paraffin
1.50E-02	67.16	65.81	64.45	63.1	61.74	60.39	59.04	16.22	0.65
2.00E-02	30.16	29.53	28.9	28.27	27.64	27.02	26.39	7.14	0.38
3.00E-02	9.77	9.56	9.36	9.15	8.95	8.74	8.54	2.41	0.25
4.00E-02	4.52	4.43	4.34	4.24	4.15	4.06	3.97	1.25	0.21
5.00E-02	2.59	2.54	2.49	2.45	2.4	2.35	2.3	0.82	0.2
6.00E-02	1.72	1.69	1.66	1.63	1.61	1.58	1.55	0.63	0.19
8.00E-02	1.01	0.99	0.98	0.97	0.96	0.94	0.93	0.47	0.17
1.00E-01	0.73	0.73	0.72	0.72	0.71	0.7	0.7	0.4	0.16
1.50E-01	0.5	0.5	0.5	0.5	0.49	0.49	0.49	0.33	0.14
2.00E-01	0.42	0.42	0.42	0.42	0.42	0.42	0.42	0.29	0.13
3.00E-01	0.35	0.35	0.35	0.35	0.35	0.35	0.34	0.25	0.11
4.00E-01	0.31	0.31	0.31	0.31	0.31	0.3	0.3	0.22	0.1
5.00E-01	0.28	0.28	0.28	0.28	0.28	0.28	0.28	0.2	0.09
6.00E-01	0.26	0.26	0.26	0.26	0.26	0.25	0.25	0.19	0.09
8.00E-01	0.22	0.22	0.22	0.22	0.22	0.22	0.22	0.16	0.08
1.00E+00	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.15	0.07
1.50E+00	0.16	0.16	0.16	0.16	0.16	0.16	0.16	0.12	0.06
2.00E+00	0.14	0.14	0.14	0.14	0.14	0.14	0.14	0.1	0.05
3.00E+00	0.12	0.12	0.12	0.12	0.12	0.12	0.12	0.08	0.04
4.00E+00	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.07	0.03
5.00E+00	0.09	0.09	0.09	0.09	0.09	0.09	0.09	0.07	0.03
6.00E+00	0.09	0.09	0.09	0.09	0.09	0.09	0.09	0.06	0.03
8.00E+00	0.08	0.08	0.08	0.08	0.08	0.08	0.08	0.06	0.02
1.00E+01	0.08	0.08	0.08	0.08	0.08	0.08	0.08	0.05	0.02
1.50E+01	0.07	0.07	0.07	0.07	0.07	0.07	0.07	0.05	0.02

TABLE 6
HALF VALUE LAYER (cm) VALUES OF SAMPLES

Energy (MeV)	CM1	CM2	CM3	CM4	CM5	CM6	CM7	Conventional Concrete	Paraffin
1.50E-02	0.01	0.011	0.011	0.011	0.011	0.011	0.012	0.043	1.069
2.00E-02	0.023	0.023	0.024	0.025	0.025	0.026	0.026	0.097	1.802
3.00E-02	0.071	0.072	0.074	0.076	0.077	0.079	0.081	0.288	2.786
4.00E-02	0.153	0.157	0.16	0.163	0.167	0.171	0.175	0.557	3.271
5.00E-02	0.268	0.273	0.278	0.283	0.289	0.295	0.301	0.841	3.55
6.00E-02	0.403	0.41	0.417	0.424	0.432	0.439	0.448	1.095	3.745
8.00E-02	0.689	0.698	0.707	0.716	0.725	0.735	0.745	1.475	4.037
1.00E-01	0.943	0.952	0.96	0.969	0.978	0.987	0.997	1.727	4.276
1.50E-01	1.376	1.382	1.388	1.394	1.4	1.407	1.413	2.111	4.786
2.00E-01	1.641	1.645	1.649	1.653	1.657	1.662	1.666	2.37	5.237
3.00E-01	1.994	1.997	2	2.002	2.005	2.008	2.01	2.775	6.032
4.00E-01	2.264	2.266	2.268	2.27	2.273	2.275	2.277	3.114	6.736
5.00E-01	2.497	2.499	2.501	2.503	2.505	2.507	2.509	3.418	7.378
6.00E-01	2.709	2.711	2.713	2.715	2.717	2.719	2.721	3.701	7.978
8.00E-01	3.096	3.098	3.1	3.102	3.104	3.106	3.108	4.218	9.084
1.00E+00	3.449	3.451	3.453	3.455	3.457	3.459	3.462	4.694	10.104
1.50E+00	4.236	4.238	4.241	4.243	4.246	4.249	4.251	5.765	12.417
2.00E+00	4.901	4.904	4.908	4.911	4.914	4.918	4.921	6.686	14.496
3.00E+00	5.972	5.977	5.983	5.988	5.994	5.999	6.004	8.217	18.173
4.00E+00	6.786	6.794	6.802	6.81	6.818	6.826	6.834	9.431	21.38
5.00E+00	7.416	7.426	7.437	7.448	7.458	7.469	7.48	10.411	24.226
6.00E+00	7.902	7.915	7.928	7.942	7.955	7.968	7.982	11.201	26.769
8.00E+00	8.572	8.591	8.609	8.628	8.646	8.665	8.684	12.371	31.123
1.00E+01	8.979	9.002	9.025	9.047	9.07	9.093	9.116	13.156	34.695
1.50E+01	9.427	9.457	9.487	9.518	9.549	9.58	9.611	14.203	41.242

5. CONCLUSION

The polysiloxanes based composite samples were designed and produced at the seven different content. Mass attenuation coefficients (cm^2/g) Linear attenuation coefficients (cm^{-1}) and half value layer values were calculated all of the samples for the energy range of 0.1 MeV to 100 GeV, gamma radiation. Obtained results compared with paraffin and conventional concrete, it is seen that, all produced composite samples have gamma–ray shielding capacity higher than paraffin and conventional concrete. The absorbed dose was experimental measured for fast neutron radiation. 4.5 MeV fast neutron radiation shielding parameter total macroscopic cross section was determined. It was founded both the total cross section values and absorption ability for fast neutron particles of the composite samples better than the paraffin and conventional concrete. Chromium oxide and hematite filled polysiloxanes composite samples have very high-quality shielding ability for both neutron and gamma radiation studies fields in measured energies. This materials can be used in nuclear applications for radiation protection studies.

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REFERENCES

[1] Krocher J.F., Browman R.E., ‘‘(Eds.) Reinhold’’ New York, 1984.

[2] Turgay Korkut, Osman Gencel, Erol Kam, and Witold Brostow, (2013) “X-Ray,Gamma, and Neutron Radiation Tests on Epoxy-Ferrochromium Slag Composites by Experiments and Monte Carlo Simulations”, International Journal of Polymer Analysis and Characterization, Vol. 18, No. 3, PP. 224-231.

[3] K. H. Mahdi, Z. S. Ahmed, and A. F. Mkhaiber'' Calculation and Study of Gamma ray Attenuation Coefficients for Different Composites''(2012) Ibn Al-Haitham Journal for Pure and Applied Science Vol. 25, No. 3, PP. 133-141.

[4] Colas A, Curtis J. Ch. 2.3 Silicone biomaterials: history and chemistry. In: Ratner B, Hoffman A, Schoen F, Lemons J (editors). Biomaterials Science: An Introduction to Materials in Medicine. 2nd Edition. New York: Elsevier; 2004. p. 80–6.

[5] Boileau S, Teyssié D. Synthesis and properties of liquid crystalline polysiloxanes. Journal of Inorganic and Organometallic Polymers. 1574-1443. 1991.

[6] M. Farahani, F. C. Eichmiller, W. L. McLaughlin. Metal–polysiloxane shields for radiation therapy of maxillo–facial tumors. American Association of Physicists in Medicine 17. 1991.

[7] C. Morari, I. Balan, J. Pintea, E. Chitanu, and I. Iordache, (2011) '' Electrical Conductivity and Electromagnetic Shielding Effectiveness of Silicone Rubber Filled With Ferrite and Graphite'' Progress In Electromagnetics Research M, Vol. 21, PP. 93-104, 201.

[8] V. Rajini, and K. Udaya kumar, (2009) “Degradation in Silicone Rubber Under AC or DC Voltages in Radiation Environment” IEEE Transactions on

[9] Dielectrics and Electrical Insulation. Vol. 16, No. 3, PP. 834-841.

[10] S. El-Fiki, S. U. El Kameesy, D. E. El.

[11] Nashar, M .A. Abou- Leila, M. K. El-Mansy, M. Ahmed. Influence of Bismuth Contents on Mechanical and Gamma Ray Attenuation Properties of Silicone Rubber Composite. International Journal of

[12] Advanced Research. (2015), Volume 3, Issue 6, 1035-1039.

[13] Mclane, V., Charles, L., Dunford, P., Rose, F., 1988. Neutron Cross Sections.’’ ISBN 978-0-12-484, 220-5.

[14] Gerward, G., L. Guilbert., Jensen, N., Leyring, K.B. H., 2004. WinXCom—a program for calculating X-ray attenuation coefficients, Radiat. Phys. Chem., 71-653.

[15] Singh, K.J., Singh, N., Kaundal, R.S., Singh, K., 2008. Gamma-ray shielding and structural properties of PbO-SiO2 glasses. Nuclear Instruments and Methods in Physics Research B 266, 944-948 (ELSEVIER).

[16] Sayyed, MI., Tekin, HO., Kılıcoglu, O., Agar, O., Zaid, MHM. Shielding features of concrete types containing sepiolite mineral: comprehensive study on experimental, XCOM and MCNPX results. Results in Physics. 11, 40-45.

[17] Berger, M.J., Hubbell, J.H. Comparative Study between Lead Oxide and Lead Nitrate Polymer as Gamma-Radiation Shielding Materials. Journal of Environmental Protection. 2016, 2-7.

[18] Singh, VP., Badiger, N.M., Gamma ray and neutron shielding properties of some alloy materials. Annals of Nuclear Energy. 2014, 64. 301-310.