14.8 MeV NEUTRON REMOVAL CROSS SECTION CALCULATIONS USING THE MONTE CARLO METHOD

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الملخص

تضمنت هذه الورقة دراسة عمليات التوهين التى تحدث للنيوترونات السريعة بطاقة 14.8م.أ.ف (MeV). و كما هو معروف بإن عمليات التوهين يمكن تمثيلها بعلاقات أسيّة تعتمد على المقاطع المستعرضة (∑R) والتي تختلف باختلاف نوع المادة التى تمر بها هذه النيوترونات. باستخدام تقنية مونتى- كارلو تم حساب هذه المقاطع لكل من الجرافيت والألومنيوم وشمع البرافين والفولاذ والرصاص ومن تم مقارنة النتائج المتحصل عليها بنتائج معملية [1] والتى أظهرت توافق كبير معها. كما تم دراسة تغير أشكال الآطياف النيوترونية مع سماكة هذه المواد، إضافة إلى ذلك تم حساب المقاطع المستعرضة لمواد أخرى للنيوترونات بطاقات (2 MeV, 10 MeV).

ABSTRACT

The attenuation of fast neutrons (14.8 MeV) has been studied using a Monte Carlo based Code (MCNP-4C). The attenuation of primary flux in different materials can be well described by a simple exponential relation based on the removal cross section. The macroscopic removal cross section (Σ_R) for graphite, aluminum, paraffin, steel and lead are calculated and then compared with experimental data which has been determined using the ^{63}Cu (n, 2n) ^{62}Cu threshold reaction for the detection of the primary neutrons [1]. The dependence of secondary neutron spectrum on the thickness of slabs has also been determined by MCNP Code for the same materials. Construction of data library for Σ_R for other materials such as Beryllium, Water, Cadmium and Copper at different neutron energies (2 MeV, 10 MeV) has been done.

KEYWORDS: Removal cross section; 14.8 MeV neutrons; Attenuation of neutrons; Monte Carlo.

INTRODUCTION

It is known that the high-energy neutron intensities obtained from neutron generators are of the order of $10^9$and 10^{11} neutron\sec from the D-D and D-T reactions, respectively [2]. In view of the maximum permissible level of 10 neutron/cm^2-sec (14 MeV) the reduction of high-energy neutron flux by a factor of about 10^9 must be achieved by the choice of the type of biological shield and by its thickness. The degree of penetration of neutrons through shielding materials depends on the neutron energy and the composition of shield material. The attenuation takes place by elastic collisions and inelastic scattering, as well as by neutron capture reactions. The presence of heavy elements in the shield is most effective in reducing neutron energy by inelastic scattering. Elastic scattering is necessary to reduce the neutron energy to thermal energy and only the nuclei with low atomic number can contribute significantly to neutron slowing down by the elastic scattering process.

Because the highest energy neutrons are most penetrating the degraded neutron flux reaches equilibrium with that of the source neutrons at some depth in the shield material. The total flux then falls off with a ″relaxation″ length characteristic of the incident energy. The relaxation length has the meaning of the distance in the shield necessary for the exponential absorption of neutrons. The macroscopic removal cross section is defined as the inverse of the relaxation length, and is the effective probability for the removal of neutrons from the incident energy range. The concept of removal cross section so useful for calculating the attenuation of fast primary neutrons in shielding materials. The attenuated neutron flux traversed a shield with total thickness of x is given by [3]:

$$\Phi(x) = \Phi_0 e^{-\Sigma_R X}$$

Where Σ_R is the macroscopic removal cross section and Φ_0 the primary neutron flux incident on the shield.

Integral measurements of neutron spectra with good resolution for simple geometries and compositions, together with investigation of the attenuation of 14 MeV neutrons in various media permit the checking of neutron transport calculations for shielding, effects of nuclear weapons and biological effects of neutrons. In the past most of the measurements were limited to fission neutrons as required by reactor developments.

Using the Monte- Carlo techniques for simulation of attenuated set ups, logarithms of neutron flux ratio $(\Phi(x) \backslash \Phi_0)$ plotted against the distance x results in a straight line with a slope of Σ_R from which the microscopic cross section for different geometries and materials can be determined.

MATERIAL AND METHOD

Monte Carlo Simulation

The Monte Carlo method is a technique that depends on the random sampling of probabilities distributions that model physical processes. A number of Monte Carlo based computer codes are used in simulation of particle transport phenomena. In this study the attenuated primary flux of 14.8 MeV neutrons traversing slabs of graphite, aluminum and lead and disks of paraffin and steel samples has been calculated by the MCNP-4C code [4] for the simulated arrangement demonstrated in Figure (1).

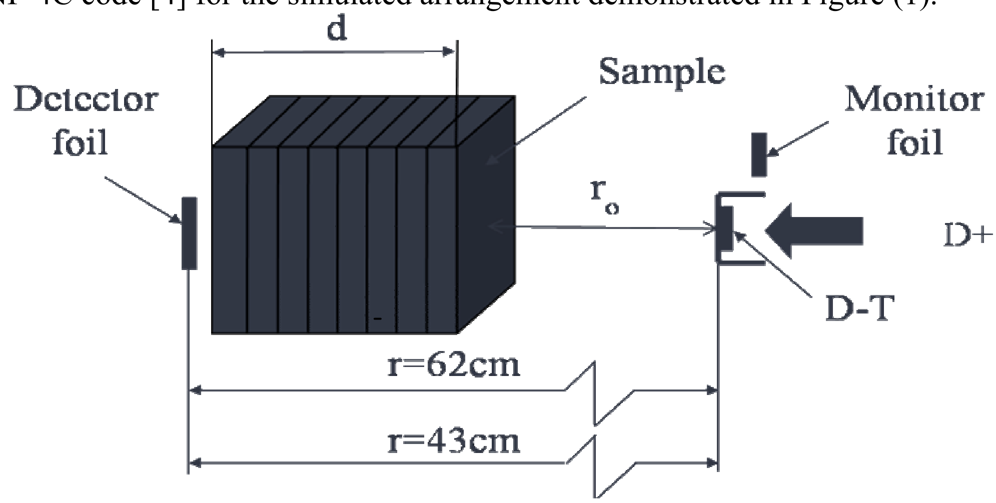

Figure 1: The geometry of the system

For the mentioned above configuration the neutrons are produced through the D-T reaction. The source is simulated as a beam of monoenergetic neutrons of 14.8 MeV energy with intensity of 10^{12} n/s. For each material the transmitted neutron flux is calculated for each increasing thickness steps. Figure (2) shows the code output for the simulated geometry for the attenuation of neutrons in paraffin slabs.

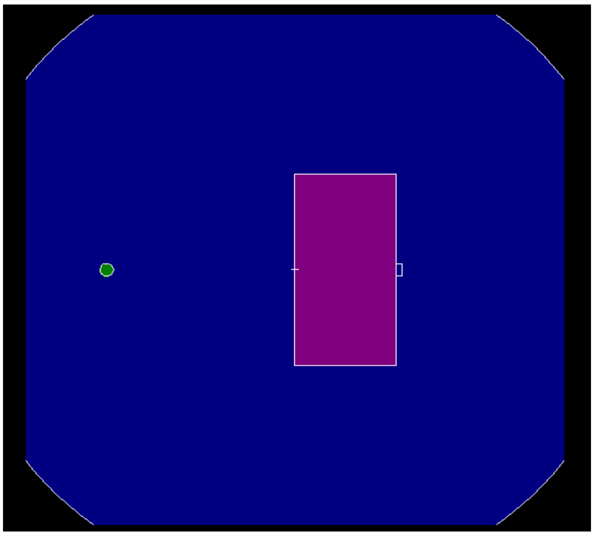

Figure 2: The code output for Paraffin shield of 20cm thickness

RESULTS AND DISCUSSION

Macroscopic Removal Cross section

Since the measurement cell in the MCNP Code is so small and facing the beam direction, so the expected neutrons which will be detected in this cell are the uncollided neutrons. This implies that the values of [ln (Φ/Φ_o)] as a function of shield thickness (d) can be well approximated by straight lines with a slope of Σ_t [5]. The removal cross section can be estimated as two third of Σ_t. The obtained Σ_R values for the materials used are shown in Table (1).

Table1: Measured and calculated macroscopic removal cross sections

Shielding material	Form and dimensions	$\Sigma_R \left[\frac{1}{cm}\right]$ measured	$\Sigma_t \left[\frac{1}{cm}\right]$ calculated	$\Sigma_R = \frac{2}{3}\Sigma_t$	$\frac{\Sigma_{Rexp}}{\Sigma_{Rcal}}$
Graphite (1.69 g/cm^3)	Slabs 20x60 cm^2	0.0584± 0.003	0.112	0.0746	0.0782
Paraffin (0.93 g/cm^3)	Disk Ø 30 cm	0.0715±.0006	0.100	0.066	1.080
Aluminum (2.7 g/cm^3)	Slabs 30x30 cm^2	0.0626±0.003	0.097	0.0646	0.969
Steel (7.82 g/cm^3)	Disk Ø 30 cm	0.1200±0.0010	0.177	0.118	1.010
Lead (11.34 g/cm^3)	Slabs 30x30 cm^2	0.0878±0.003	0.137	0.0913	0.962

The results were compared with experimental results in which those who performed the experiment used samples with rectangular shape of 30 cm×30 cm and disks with a diameter of 30 cm. The attenuated primary neutron flux was measured at distances of 62 cm as well as at 43 cm with copper detector foils via the ^{63}Cu (n, 2n) ^{62}Cu reaction [1]. As it can be noticed the calculated results are in a good agreement with the measured results.

The attenuation of the primary neutron flux as a function of shield thickness for Graphite, Aluminum, Paraffin, Steel and lead are demonstrated through Figures (3, 4, 5, 6 and 7) respectively.

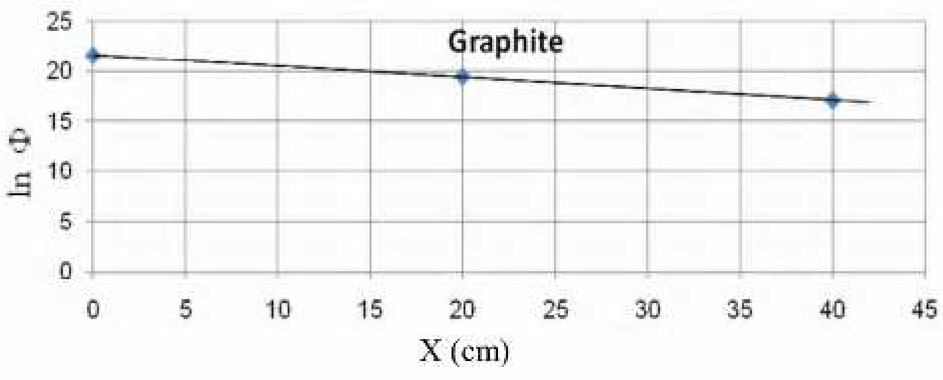

Figure 3: The values of ln Φ as a function of shield thickness for graphite

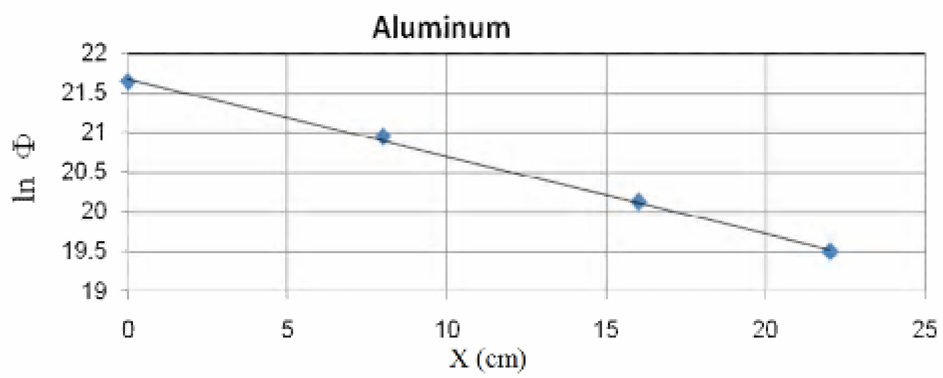

Figure 4: The values of ln Φ as a function of shield thickness for aluminum

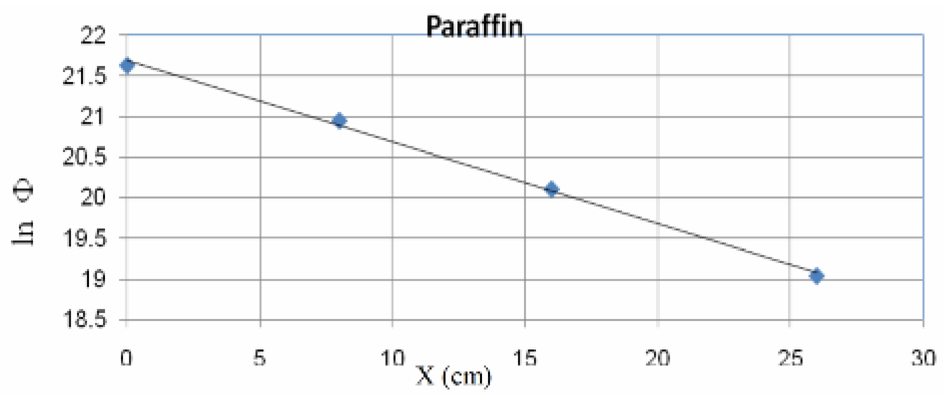

Figure 5: The values of ln Φ as a function of shield thickness for paraffin

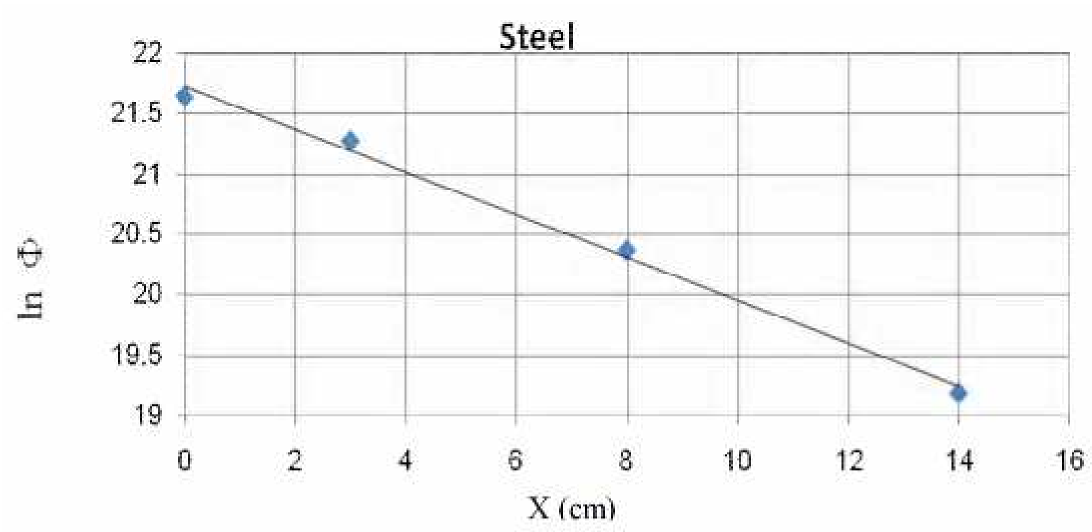

Figure 6: The values of ln Φ as a function of shield thickness for steel

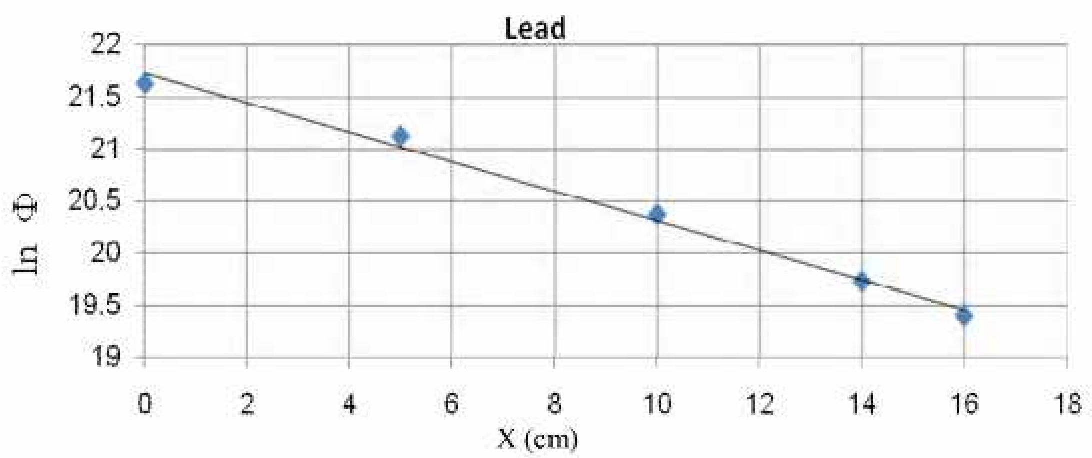

Figure 7: The values of ln Φ as a function of shield thickness for lead

Neutron Spectra

By using MCNP code the high energy part of the secondary neutron spectra have also been calculated in different (d) depth of the samples. The (d) distance was measured from the front surface of the samples. In all cases the thickness of the samples was equal to the maximum value used in the removal cross section measurements. In Figure (8) the calculated spectra are presented for the investigated materials.

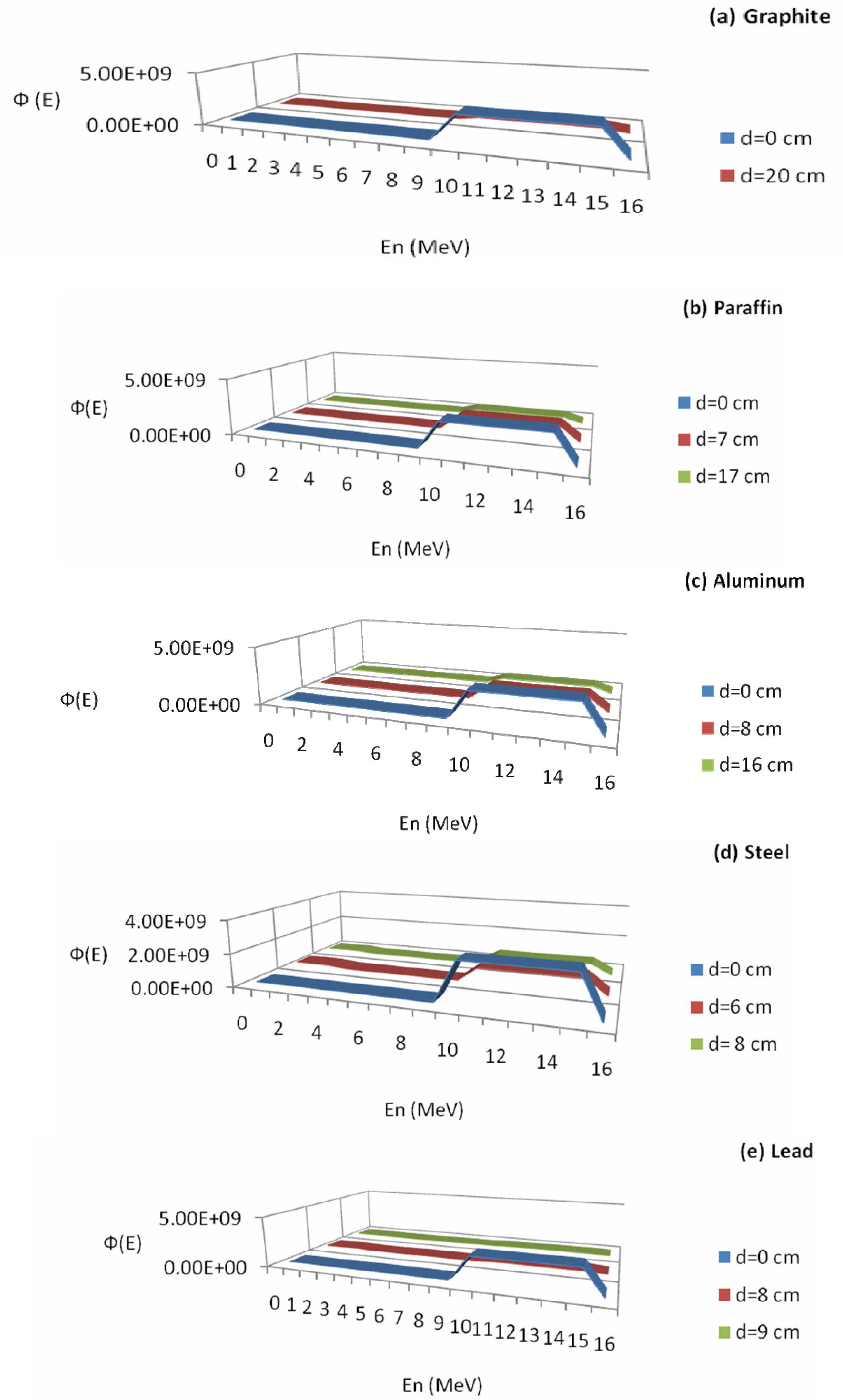
(a) Graphite
5.00E+09
Φ (E)
0.00E+00
0 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16
En (MeV)
d=0 cm
d=20 cm
(b) Paraffin
5.00E+09
Φ(E)
0.00E+00
0 2 4 6 8 10 12 14 16
En (MeV)
d=0 cm
d=7 cm
d=17 cm
(c) Aluminum
5.00E+09
Φ(E)
0.00E+00
0 2 4 6 8 10 12 14 16
En (MeV)
d=0 cm
d=8 cm
d=16 cm
(d) Steel
4.00E+09
Φ(E)
2.00E+09
0.00E+00
0 2 4 6 8 10 12 14 16
En (MeV)
d=0 cm
d=6 cm
d= 8 cm
(e) Lead
5.00E+09
Φ(E)
0.00E+00
0 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16
En (MeV)
d=0 cm
d=8 cm
d=9 cm

Figure 8: Neutron spectra obtained for various materials, a) Graphite, b) Paraffin, c) Aluminum, d) Steel, e) Lead.

Estimation of Σ_R For Other Materials for different neutron Energies

For water, beryllium, cadmium and copper the macroscopic removal cross section for different source neutron energies (2, 10MeV) are calculated by the code. The results are presented in Table (2). It can be noticed that if the energy of neutrons decreases the Σ_R is increased.

Table 2: The calculated macroscopic removal cross section for various materials and neutrons energies

Shielding material	Form and dimensions	Σ_t En=2 MeV	Σ_R En=2 MeV	Σ_t En=10 MeV	Σ_R En=10 MeV	Σ_t En=14.8 MeV	Σ_R En=14.8MeV
Water(1g/cm^3)	Slabs 30x30cm^2	0.236	o.157	0.105	0.07	0.094	0.0626
Beryllium(1.85g/cm^3)	Slabs 30x30cm^2	0.188	0.125	0.177	0.118	0.156	0.104
Cadmium(8.65g/cm^3)	Slabs 30x30cm^2	0.22	0.146	0.174	0.116	0.172	0.1146
Copper(8.96g/cm^3)	Disks Ø30cm	0.230	0.153	0.239	0.153	0.199	0.1326

CONCLUSION

The attenuation of 14.8 MeV with intensity of 10^{11} neutron/sec traversing slabs of Graphite, Aluminum and Lead and discs of Paraffin and steel samples has been presented. The removal cross sections for these materials are calculated using the MCNP-4C code. Since the obtained results were satisfactory as compared with experimental and tabulated data and as an attempt for data library construction, the removal cross sections for Beryllium, Water, Cadmium and Copper at different neutron energies were determined.

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