Determination Of 4.5 Mev Neutron Removal Cross Section For Some Shielding Materials

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Abstract

On the basis of the fast neutron attenuation, the total removal cross section at some shielding materials such as Al,Fe,Cu, and Pb and the compounds paraffin wax, polyethylene, borated polyethylene and concrete have been determined,using neutrons with energy 4.5 MeV emitted from ^{241}Am-Be neutron source.

Keyword: Fast neutron attenuation,removal cross section, shielding, Materials, neutron cross section, Neutron Sources.

حساب المقطع العرضي العياني لازالة نيوترونات ذات طاقة 4.5 MeV كدروع واقية من الاشعاع لبعض المواد المستخدمة

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الخلاصة

في البحث الحالي تمت دراسة وتحديد المقطع العرضي لازالة النيوترونات السريعة ذات الطاقة 4.5 MeV المنبعثة من المصدر النيوتروني النظائري ^{241}Am-Be لمواد مختلفة مثل الالمنيوم ,الحديد,النحاس و الرصاص وتراكيب من شمع البرافين , البولي اثيلين , البولي اثيلين الممزوج بالبورون بالاضافة الى الكونكريت لاستخدامها كدروع الواقية من النيوترونات السريعة على اساس عامل التوهين لهذه المواد .

INTRODUCTION:

The concept of the removal cross section is useful in describing the attenuation of fast primary neutrons in shielding materials.

Attenuation of neutrons in a shield takes place by elastic and inelastic scattering and capture of neutrons. Capture cross sections are large only at thermal and near- thermal energies, and degradation by scattering is, therefore, necessary before the fast neutrons can be captured. Inelastic scattering usuallyresults in a large energy loss by the scattered neutrons but, the process is possible only for the fast neutrons (E_n> 0.5 MeV). Heavy elements are the most effective scatterers. Hydrogen has no excited states and therefore produces no elastic scattering. Elastic scattering is necessary to degrade neutrons to the thermal region. In this process kinetic energy is transferred from the neutron to the nucleus, and the average fractional neutron-gamma loss is ($2A/ (A+1)^2$) where A is the nuclear mass number. Only light nuclei particularly,hydrogen, are effective. The hydrogen cross section increases rapidly as the neutron energy decreases. Elastic scattering complements the effect of inelastic scattering by the heavy elements. Since the highest energy neutrons are the most penetrating, the flux of the degraded neutrons comes into equilibrium with that of the source neutrons at the some depth in the shield [1].

The cross section σ_r for a particular process, which applies to a single nucleus, is frequently called themicroscopic removal crosssection to distinguish it from ($N\sigma_r$), called the macroscopic removal cross section of the materials for that process. Thus, representing the latter by $\sum_r$ definition is

$\Sigma_r = N\sigma_r cm^{-1}$ … … … … (1)

Where (N) is the number of the nuclei per cm^3 it is consequently the total cross section of the nuclei in ($1cm^3$) of the material. It will be noted that the macroscopic removal cross section has the dimensions of a reciprocal length. Replacing ($N\sigma_r$) by Σ_r in accordance with $\Sigma_r = N\sigma_r cm^{-1}$ it seen that,

$$-\frac{dI}{I} = \Sigma_r dx \dots \dots \dots \dots (2)$$

Since($-\frac{dI}{I}$) is the fraction of the neutrons absorbed in the path (dx), it is evident that ($\Sigma_r dx$) is the probability that neutrons will be absorbed in the path (dx). If (ρ) is the density of the absorbing material in grams per cm^3,and (A) it's atomic weight,of an element, then ($\frac{\rho}{A}$) is the number of gram atoms per cm^3. The number of atomic nuclei per cm^3 is then obtained upon multiplying by (N_A) the Avogadro number ($6.02x10^{23}$) , which gives the number of individual atoms (or nuclei) per gram atom [2] ; thus

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

And hence,

$$\Sigma_r = \left(\frac{\rho}{A} N_A\right) \sigma_r \dots \dots \dots \dots (3)$$

If the material under consideration contains several nuclear species, then the macroscopic removal cross section is given by

─ 0₅ Œ ₅ E 0₆ Œ ₆ E 0₇ Œ ₇ E fi E 0 Œ E fi

Where, in general,(N_i) isthe number of nuclei per cm^3 of the (ith) kind present and Œ is the microscopic removal cross section for the given process. By introducing - 0 Œ ?I ? 5into

+ L + A? ˙ ° :v;

It follows that,

+ L + A? °

Or

u/u L A? ° :w;

The quantity (u/u) is the fraction of incident neutrons which succeed in penetrating the thickness (x) of material without undergoing the reaction being considered. Since the probability that reaction will occur between (x) and ($x+dx$) is given by -dx, the average distance ( ) a neutron will travel before being absorbed is given by [3]

ª L ∫₄^¶ T- A? ° @T s / ∫₄^¶ - A? ° @T - ?I :x;

Thisintegral in numerator and denominator being standard form. The average distance ( ) calculated above is called the (mean free path) for the given nuclear reaction it has, of course, the dimension of the length, since (-) is a reciprocal length, replacing (-) by (−5) the result is,

u/u L A?5° :y;

If (x) is set equal to ( ),then,

u/u L A?5 :z;

So, that ( ) may also be regarded as the distance in which all but a fraction (−5) of the incidence neutrons are absorbed. Relaxation length of the neutrons in the given medium it is the distance in which the intensity of the neutron beam is reduced to a fraction ($\frac{1}{e}$) of its initial value due to absorption of neutrons in the medium were no scattering.

To calculate microscopic removal cross section $\sigma_r$there is a relation withthe macroscopic removal cross section is given by, [4]

$\Sigma_r = \frac{0.602\sigma_r\rho}{A} cm^{-1}$... ... ... (9)

Where σ_r is the microscopic removal cross section (barns)ρIs the density (g/cm$^{3)}$ and A is the atomic weight

The removal cross section per atom $\Sigma_r\ cm^{-1}$ cannot be expressed in a direct relation with the microscopic neutron cross sections of nuclear physics .However, as a first approximation:

$\sigma_r \cong 0.6\sigma_t$... ... ... (10)

Where σ_t is the total cross section of the nuclei in 1 cm^2 of the material. For high energy neutrons ($\geq 3\ MeV$) and Z < 11

$\sigma_r \cong 0.75\sigma_t$... ... ... (11)

For Hydrogen $\sigma_r \cong \sigma_t$. If the shield contains enough hydrogen, scattered neutrons have a good chance of being further slowed and thermalized and the build-up factor of the degraded neutron is small .The macroscopic removal cross section for a material of several elements is obtained by simple summation over its constituents.[5]

$$\Sigma_{r\,compound} = (\frac{\Sigma_r}{\rho})_1\rho_1 + (\frac{\Sigma_r}{\rho})_2\rho_2 + (\frac{\Sigma_r}{\rho})\ \ _3\rho_3 + \cdots$$

The aim of the present work is the measurement of the primary and removal cross section for different shielding materials and different combinations of these materials for the purpose of providing more detailed and improved data for neutron attenuation and providing the necessary information which helps in deciding the suitable material and the thickness needed.

EXPERIMENTAL:

Neutron Source:

Am-Be source (2.283 x 10^6 n/s in 11/1985, half-life 432 years).Americium-Beryllium (Am-Be), which produces neutrons via the ^{9}Be (a, n)^{12}C reaction (^{241}Am has a half-life of 458 years). Isotopic neutron sources have the advantage of having a long useful life and producing a relatively constant flux of neutrons. Physical form of ^{241} Am-Be neutron source is compacted mixture ofamericium oxide with beryllium metal. Alpha particles sending from ^{241}Am have approximately 5.5 MeV maximum energy [5].

Materials:

Shielding materials (such as Al , Fe ,Cu and Pb) and the compounds(paraffin Wax , polyethylene , 5% borated polyethylene and concrete) have been used in this study . Sample dimensions were 30cmx30cmwith different thickness.

Counting:

The neutron counting systems consist of two BF3(35 cm length, 2.5 cm in diameter)long counter neutron detectors. One of the BF3 long counter detector has been used as a monitor (M) which was placed at 90^o to the beam direction of neutrons, and the other BF3(R) neutron detector have been used for the total neutron detection and placed in front of the Neutronbeam,Fig (1a) .Both detectors connected to spectroscopy system (preamplifier, amplifier, HV, Timer scalar and MCA).

The data from BF_3(R) neutron detector have been monitored with respect to the data obtained by BF_3 (M)monitor. The source to detector and source to sample distances and theequipment's arrangement showsin Fig. (1b).

Results and Discussion:

The BF_3 long counter has been used for the differentmaterials 4.5 MeV neutron total removal cross section determination. These materials are used in construction of the building of neutron sources. The results obtained for lead Iron, Copper and aluminum are show in Fig. (2). The results obtained for Compound materials shown in Fig.(3).The data obtained by these experiments which are used for construction materials removal cross section determination are presented in Table(1) .

The4.5 MeV neutron removal cross section values shows a relatively good agreements with the published removal cross section values [6-9]. This applies for all material accept for concrete due to variation of water content which effects the results strongly, other reason is the chemical composition. The values of microscopic removal cross section have been calculated using equation (2) and presented in Table (1). One can see that $\frac{\Sigma r}{\rho}$ is a quantity depends only on the microscopic nuclear properties which is smoothly varying factor as a function of atomic weight Fig (4). The $\frac{\Sigma r}{\rho}$ values have been fitted with the theoretically calculated values and a good agreements was obtained between them as shown in the fitted curve. The relaxation (attenuation) length $\lambda\, cm$ have been determined for materialsused inthis studyand show in Fig.(5).The values obtained for relaxation (attenuation) length are given in Table (2).

Table (1):summarized values for several materials used in this study.

Materials	Atomic weight	Density g.cm^{-3}	Macro C.S $\Sigma_r cm^{-1}$	$\frac{\Sigma_r}{\rho}$	Micro C.S. σ_r $barn$
Aluminum	27	2.68	0.1879	0.070112	1.38
Iron	55.85	6.70	0.3259	0.048642	7.54
Copper	63.54	8.70	0.4156	0.047770	2.21
Lead	207.19	11.23	0.2064	0.018379	2.77
Concrete	-	2.31	0.1052	0.045541	-
Paraffin Wax	-	0.880	0.1831	0.208068	-
Polyethylene	-	1.08	0.1804	0.167037	-
Polyethylene-Bor.	-	1.44	0.2154	0.149583	-

Table (2): Macroscopic removal cross sections and Relaxationlength for several Materials used in this study.

Materials	Macro C.S $\Sigma_r cm^{-1}$	λ cm
Aluminum	0.1879	5.322
Iron	0.3259	3.068
Copper	0.4156	2.406
Lead	0.2064	4.844
Concrete	0.1052	9.505
Paraffin wax	0.1831	5.461
Polyethylene	0.1804	5.543
Polyethylene-Bor.	0.2154	4.642

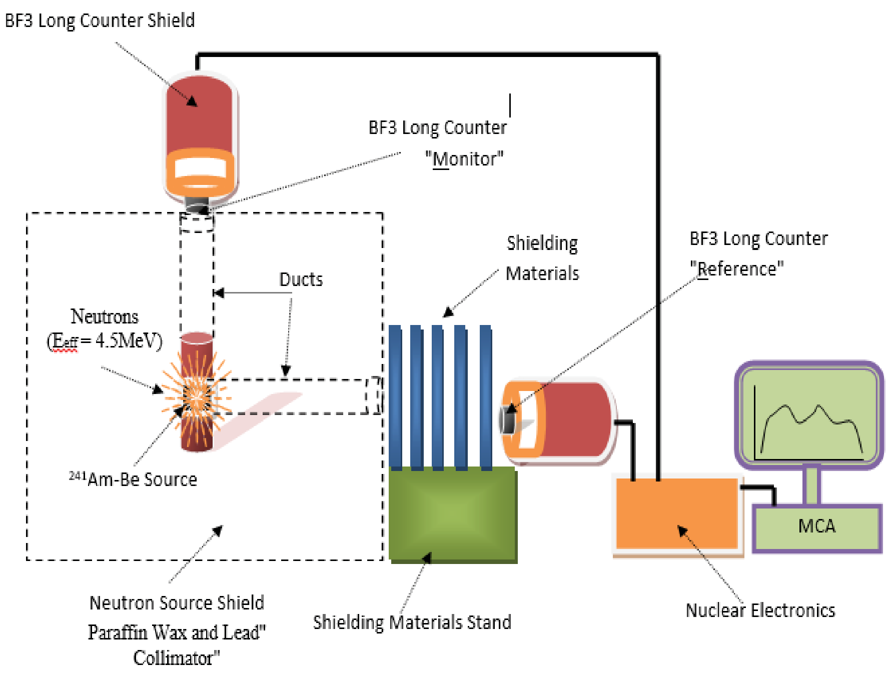

Fig. (1a): Experimental set-up for determining 4.5 MeV neutrons removal cross Section.

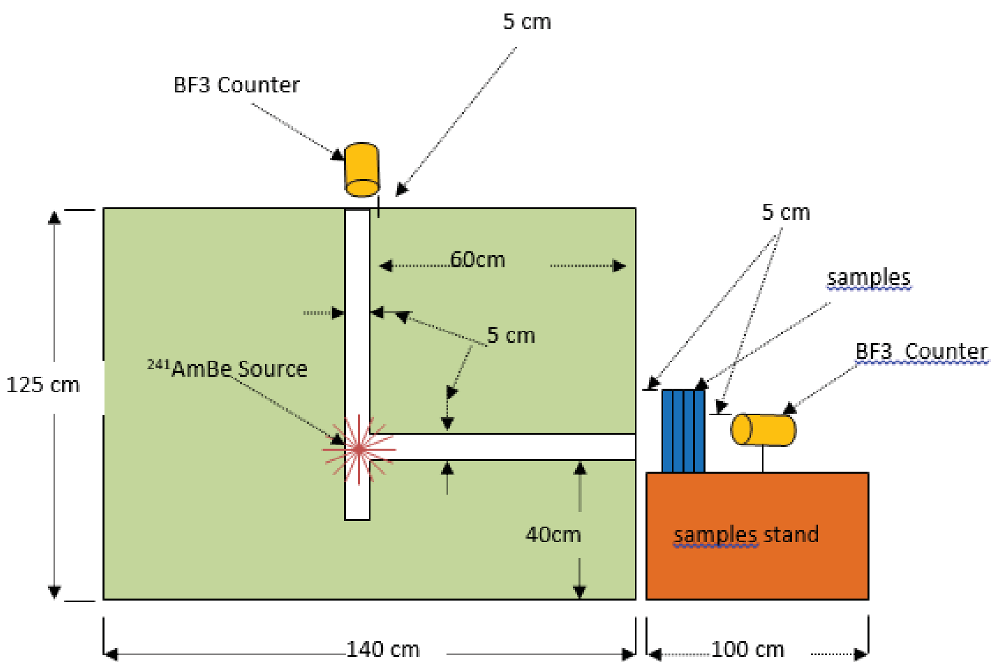

Fig. (1b): Experimental set-up for determining fast neutron removal cross section (Sample-source-detector dimension).

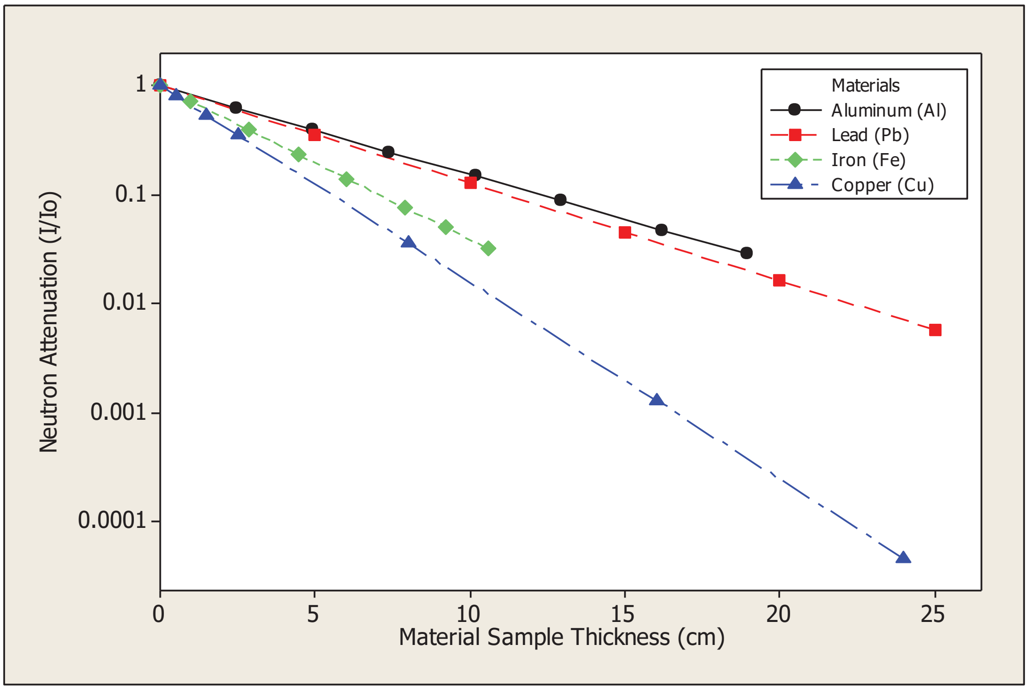

Fig. (2):4.5 MeV Neutrons Attenuation (I/Io) VS Material Sample Thickness (cm).

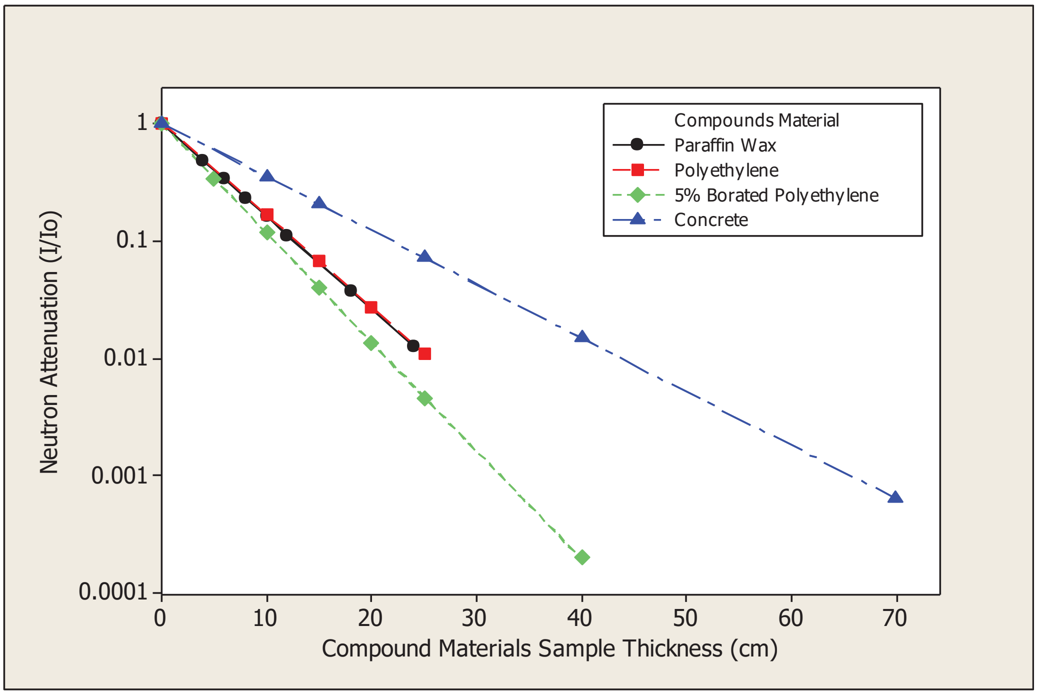

Fig.(3): 4.5 MeV neutrons Attenuation (I/Io) VS Compound Materials Sample Thickness (cm).

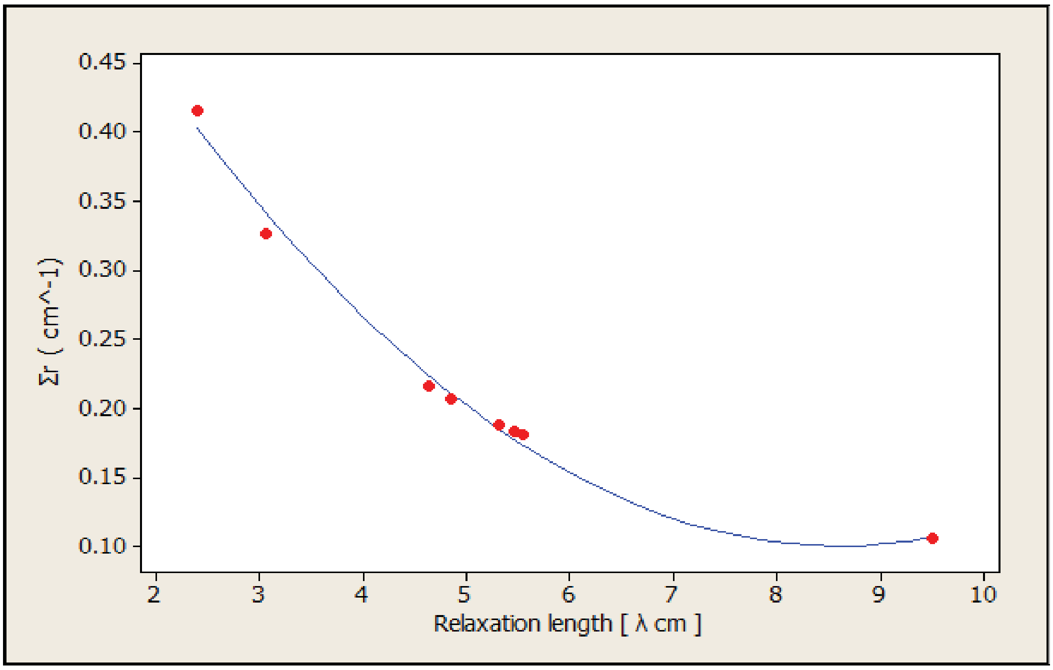

Fig.(4):Macroscopic neutron removal crosssections [$\Sigma_r cm^{-1}$]VS Relaxation length [$\lambda\ cm$].

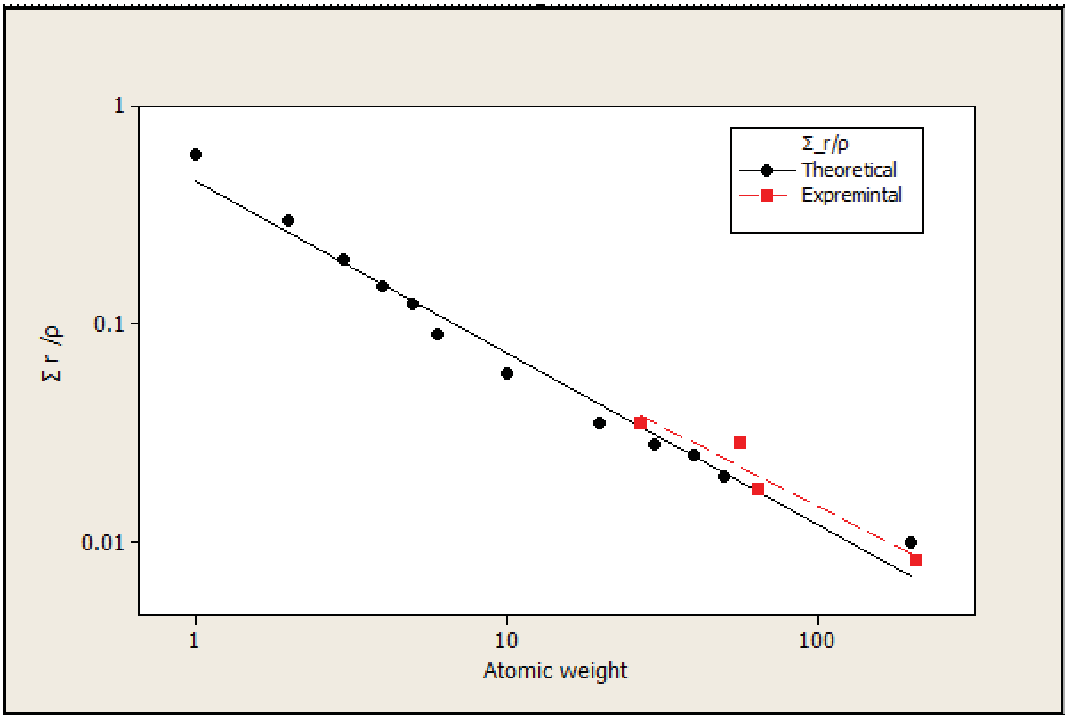

Fig.(5): The relationship between the [(Macroscopic removal cross section) /Density] ($\frac{\Sigma r}{\rho}$) and atomic weight (A).

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