Neutron Shielding Materials

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INTRODUCTION

Understanding the fundamentals of the interactions of neutrons with matter is an important step in working safely in areas where neutron radiation may be encountered. The following monograph will cover the basic principles and terminology of neutron sources and shielding. A more detailed treatment of each topic may be found in references [1-5].

DEFINITION OF COMMON TERMS

Barn – The standard unit for neutron cross sections, equal to 10^{-24} cm^2, denoted by the symbol b. Typical cross section values range between 0.001 and 1000 barns(b).

Compound Nucleus – An unstable nucleus formed during neutron capture. The neutron combines with the target nucleus, adding its kinetic energy and binding energy, forming a new nucleus in an excited state. If sufficient energy is present, one or more nucleons from the compound nucleus may overcome nuclear binding forces and escape the nucleus in a process called evaporation. Excess energy may also be released in the form of gamma radiation. The typical lifetime of a compound nucleus is 10^{-14} to 10^{-20} seconds.

Cross Section – A value which describes the probability that a nuclear reaction will occur. For neutrons, the cross section is related to the geometric cross section of the target nucleus. Therefore, cross sections are typically expressed in cm^2 or barns(b) = 10^{-24} cm^2. The cross section for a given nuclear interaction is also dependent on other factors, such as the speed of the neutron, the type of interaction (scattering, capture, etc..), and the stability of the target nucleus. Cross sections are typically defined for specific nuclear reactions and for the overall probability of nuclear reaction with a target nucleus. The term cross section may refer to the **microscopic cross section**, or the interaction of neutron(s) with a single target nucleus, or **macroscopic cross section** interaction of neutron(s) with a thick layer of material. Cross sections may be further divided into cross section values for individual types of interaction (scattering, absorption).

Elastic Scattering – A scattering interaction in which the neutron-target system has essentially the same kinetic energy before and after interaction. Elastic scattering of neutrons may alter the direction and speed of the neutron, but will not alter the identity of the neutron or the target or cause nuclear excitation in the target. Relatively small kinetic energy losses in the neutron-target system may occur through atomic or molecular excitations.

Epithermal Neutrons – Neutrons of higher energy than thermal neutrons, typically ~0.1eV and 1keV. Some resources may characterize epithermal neutrons with slightly different energy ranges.

Fast Neutrons – Neutrons with energy >0.1 MeV. Some references may characterize fast neutrons with slightly different energy ranges.

Fission – A nuclear reaction or decay process (spontaneous fission) in which the nucleus of an atom splits into lighter parts. Fission may produce lighter nuclides as well as additional neutrons, photons and large amounts of energy. Fission is one possible outcome from neutron capture reactions.

Inelastic Scattering – A scattering interaction in which a neutron transfers energy to a target nucleus causing nuclear excitation. The excited target nucleus then returns to a non-excited state through the emission of gamma radiation. The identity of the neutron and target nucleus are not altered. However, there is a net loss in kinetic energy of the neutron-target system.

Macroscopic Cross Section – A value (Σ), with units of cm^{-1}, that describes the probability of interaction of a neutron with a thick layer of target material.

Mean Free Path – The average distance traveled by a moving particle (neutron) in a target medium between interactions with the target material.

Microscopic Cross Section – A value (σ), typically in cm^{-2} or barns, which describes the probability of the interaction of a neutron with a single target nucleus.

Moderating Power – A value relating the effectiveness of a material in the lowering of neutron energy through scattering reactions (moderation/thermalization). The moderating power is defined as the product of macroscopic scattering cross section (Σ) and the average logarithmic energy loss upon scattering (ξ). As moderating power increases, less material is required to achieve the same degree of moderation.

Moderating Ratio – A value relating the effectiveness of a material in lowering the neutron energy through scattering reactions, which also takes into account neutron capture cross sections. The moderating ratio is defined as the moderating power divided by the macroscopic neutron capture cross section for the material. Materials with large moderating ratios are good neutron moderators and poor neutron absorbers.

Nucleon – A particle that makes up an atomic nucleus. (neutrons or protons)

Neutron Capture – A nuclear reaction in which a neutron and target nucleus collide and merge, forming a heavier nucleus (**compound nucleus**). Depending on factors, including the energy of the incident neutron and the nuclear properties of the target nucleus, the neutron capture may be followed by the emission of gamma radiation or atomic particles and/or in the fission of the compound nucleus.

Neutron Excitation Function – A plot of cross section vs neutron energy for a given neutron-target system.

Neutron Fluence – The neutron flux integrated over a period of time with units of neutrons/cm^2.

Neutron Flux – A measure of the intensity of neutron radiation, expressed in neutrons/cm^2/sec, corresponding to the rate of flow of neutrons.

Neutron Moderation – See **Neutron Thermalization**.

Neutron Scattering – An interaction between a neutron and matter which results in a change in velocity of the neutron. Scattering may be elastic or inelastic.

Neutron Thermalization – The process of neutron energy reduction (**moderation**) to thermal values (~0.025eV) through scattering reactions.

Resonance Neutrons – Neutrons which are strongly captured in the resonance of U-238 and some commonly used detector materials (In, Au), typcially 1-300eV.

Resonance Peaks – Sharp peaks in the plot of cross section vs neutron energy (**neutron excitation function**) for a given neutron-target system. The peaks correspond to nuclear energy level spacings in the target material. Neutrons at resonance energies exhibit increased probability of interaction with the target (higher cross section).

Thermal Neutrons – Neutrons in thermal equilibrium with their surroundings, typically **~**0.025eV.

NEUTRON SOURCES

Neutron emission is typically associated with the fission of uranium or plutonium fuel in a nuclear reactor. However, there are many other potential sources of neutrons that may be encountered. A list of some common neutron sources is presented in Table 1.

Table 1. Neutron Sources

Type	Examples	Comment
Induced Fission	neutron bombardment of ^{235}U	Yields average of 2.5 neutrons with an average energy of 2MeV.
Spontaneous Fission	^{252}Cf	3.1% branch of ^{252}Cf decay. Yields 3.7 neutrons per fission, average energy 2.3MeV.
Delayed Neutron Emission	$^{87}Br \rightarrow ^{87}Kr + \beta^- \rightarrow ^{86}Kr + n$ $^{11}Li \rightarrow ^{11}Be + \beta^- \rightarrow ^{9}Be + 2n$	Neutron-rich nuclides in excited states may emit neutrons. Important for fission products.
(α,n) Sources	PuBe, AmBe, AmLi	Mixture of alpha emitter and low atomic number element. ~30 neutrons per one million alpha emissions. Neutron energy from 0.5-4MeV depending on alpha energy and target.
Spent Nuclear Fuel		Major sources are spontaneous fission of ^{244}Cm and ^{242}Cm and (α, n) reactions on oxygen
(γ,n) Sources	^{88}Y-^{9}Be, ^{124}Sb-^{9}Be	Gamma radiation in excess of nuclear binding energy causes neutron emission from target
Light Ion Accelerators	$D + T \rightarrow n + ^{4}He$ $D + D \rightarrow n + ^{3}He$	Deuterium or tritium ions are accelerated into deuterium or tritium hydride targets, producing neutrons from fusion.
Spallation Sources	ISIS neutron source SNS at ORNL	High energy protons impacting on depleted uranium, tugsten or tantalum target strip (spall) neutrons

Neutron Interactions with Matter

Like gamma radiation, neutrons undergo scattering and absorption interactions with matter. These interactions form the basis for methods used to shield and measure neutron radiation. However, unlike gamma radiation, which interacts primarily with the electrons in matter, neutrons interact primarily with the nucleus. Consequently, the types of materials favored for neutron shielding are quite different than the dense, high atomic number absorbers which are most effective in the attenuation of gamma radiation. Additionally, whereas isotopes of an element will have essentially identical gamma attenuation properties, isotopes of an element often have significantly different neutron attenuation properties.

In general, for fast (high energy) neutrons, scattering interactions are more likely than capture interactions. As the energy of neutrons is reduced through scattering interactions (**neutron thermalization/moderation)**, all neutron interactions increase in probability and neutron capture interactions become more important. Scattering interactions for neutrons can be divided into elastic (kinetic energy of neutron-target system conserved) and inelastic scattering (kinetic energy of the neutron-target system is lowered through excitation of the target nucleus and subsequent gamma emission).

In neutron capture, a neutron and target nucleus collide and merge, forming a heavier nucleus (compound nucleus). Depending on the energy of the incident neutron and the nuclear properties of the target nucleus, the neutron capture may be followed by the emission of gamma radiation, atomic particles, and/or in the fission of the compound nucleus. Neutron capture reactions are denoted by X(n,a)Y, where X is the target nucleus, n is the incident neutron, a is the ejected particle(s) or gamma ray, and Y is the nucleus after absorption of the neutron and particle or gamma emission. Table 2 lists several types of neutron capture reactions.

Table 2. Common Neutron Capture Reactions

Reaction	Neutron Energy
(n,γ)	0-500 keV
(n,p)	0.5-50 MeV
(n,α)	0.5-50 MeV
(n,2n)	1-50 MeV
(n,np)	1-50 MeV
(n,2p)	1-50 MeV
Fission	Thermal to Fast

Neutron Cross Sections

The probability that neutron-target interactions will occur is expressed using cross sections, denoted by the symbol σ. Neutron cross sections are related to the geometric cross section of the target nucleus. Therefore, cross sections are typically expressed in cm^2 or barns(b) = 10^{-24} cm^2. The cross section for a given nuclear interaction is also dependent on other factors, such as the speed of the neutron, the type of interaction, and the stability of the target nucleus. Cross sections are typically defined for specific nuclear reactions and for the overall probability of nuclear reaction (σ_t) with a target nucleus.

$$\sigma_t = \sigma_{ei} + \sigma_i + \sigma_c + \sigma_f + \ldots\ldots$$

σ_{el} = elastic scattering cross section

σ_i = inelastic scattering cross section

σ_c = capture cross section (may be split into individual capture reactions)

σ_f = fission cross section

Cross section values typically range from 0.0001 to 1000 barns. Plots of cross section vs neutron energy for a given target are called **neutron excitation functions**. A typical neutron excitation function is depicted in Figure 1. In general, the cross section decreases with increasing neutron energy. However, at some neutron energies sharp increases in cross section may be observed. These sharp peaks are known as resonance peaks and correspond to nuclear energy level spacings in the target material.

The cross section discussion to this point has focused on the interaction of neutron(s) with a single target nucleus, or the **microscopic cross section (σ).** The microscopic cross section is useful for understanding the fundamental interaction processes for neutrons and matter. However, for the attenuation of neutrons in shielding material, it is better to discuss the interaction of neutron(s) with a thick layer of material, or the **macroscopic cross section (Σ)**. The total macroscopic cross section (Σ_t) can be defined as:

$$\Sigma_t = N\sigma_t$$

where N = the atom density of the target material σ_t is the total microscopic cross section. The intensity of a neutron beam, I(x), passing through a target material of thickness x can be expressed as:

$$I(x) = I_o e^{-N\sigma t x} \qquad \text{-or-} \qquad I(x) = I_o e^{-\Sigma x}$$

This function allows the calculation of the fraction of neutrons at a given energy that will pass through a thickness (x) of a given target or shielding material without undergoing any type of scattering or capture interaction. Calculations for composite materials can be performed by using the sum of the macroscopic cross sections of each individual element:

$$\Sigma = \Sigma_1 + \Sigma_2 + \Sigma_3 \ldots..$$

where the atom density of each element (N_i) is:

$$N_i = \rho N_A n_i/M$$

where ρ is the density of the composite material, M is the molecular or unit weight of the composite material, N_A is Avagadro's number, and n_i is the number of atoms of the element in the molecule or composite unit.

The accurate calculation of total dose reduction for a given thickness of shielding is much more difficult due to the possibility of multiple scattering and capture reactions and the production of secondary radiation (gamma emission, etc.). The calculation of dose reduction in neutron systems must take into account contributions from scattered neutrons with lower energy and secondary radiation, and therefore requires more sophisticated calculation techniques, such as Monte Carlo N-Particle Transport Code (MCNP) [6], employing extensive libraries of cross sections. Appendix I. contains neutron attenuation data for a wide range of materials calculated using MCPN6[6].

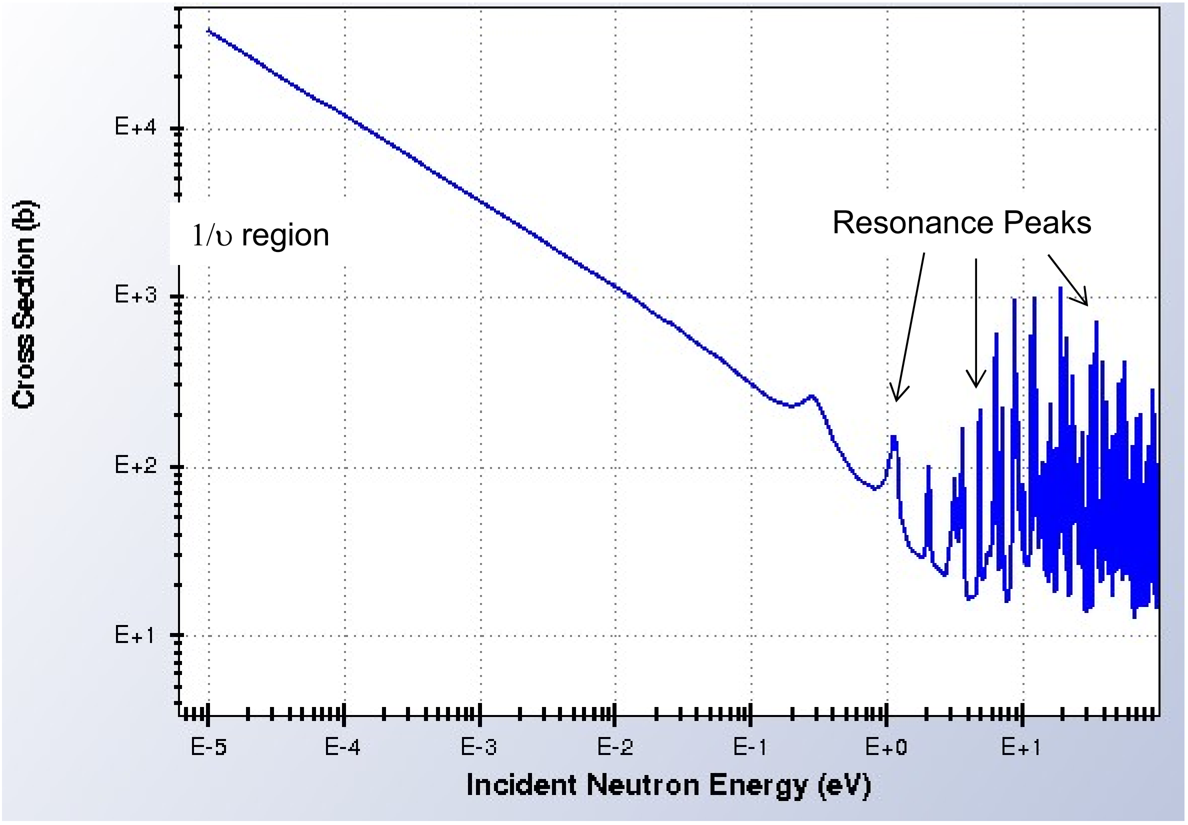


Figure 1. Neutron Excitation Function

Common Neutron Shielding Materials

Neutron shielding materials are typically constructed from low atomic number elements (hydrogen, carbon, and oxygen) with high scattering cross sections that can effectively moderate or thermalize incident neutrons. Shielding for small sources is often constructed from polyethylene or paraffin, while shielding for larger sources is made from concrete or large pools/tanks of water.

Elements with high capture cross sections for thermal neutrons (boron, cadmium, and gadolinium) are often dispersed in the shielding material to capture moderated/thermalized neutrons. Borated polyethylene, layers of B_4C and aluminum, boron-aluminum alloys, and boric acid in water are examples of materials incorporating boron. Adding boron to neutron shielding materials reduces the dose from secondary gamma production from radiative capture (n,γ). Boron, specifically the ~20% naturally abundant boron-10, has a very high (n,α) capture reaction, which yields much lower energy gamma radiation than the radiative capture reactions (n,γ) of hydrogen, oxygen, or carbon.

Neutron shielding may also incorporate high atomic weight elements or layers of higher atomic weight shielding material to reduce dose from gamma radiation produced from neutron capture interactions (n, γ). Lead, bismuth, and tungsten are often used due to their high density, good gamma attenuation characteristics, and relatively benign activation products. However, the relatively high (n,γ) cross-section of tungsten should be considered when choosing shielding for areas with high neutron fields, due to the significant secondary gamma radiation dose that can be produced from the capture of neutrons escaping the primary neutron shield. Lead or bismuth may be better choices for shielding (n,γ), due to their much lower (n,γ) cross-sections.

Moderating power and **moderating ratio** are two measures commonly used to describe the moderating effectiveness of materials. Materials with high moderating power typically have high scattering cross sections and induce large energy losses in the neutron for each scattering interaction. Moderating ratio, also takes into account capture cross section for the material. Materials with high moderating ratios have high moderating power and low probability of neutron capture interactions.

Provided below (Tables 3, 4 and 5) are brief descriptions of the attenuation characteristics and physical properties of some materials commonly used in neutron shielding. Selection of the appropriate shielding material requires consideration of many factors including neutron moderation and capture properties, production of secondary radiation (n,γ), potential neutron activation reactions which can induce radioactivity in the shielding material, and chemical and physical compatibility of the shielding material with the environment in which it will be used.

References

1) Thomas E. Johnson and Brian K. Birky, Health Physics and Radiological Health, 4th ed., Wolters Kluwer/Lippincott Williams and Wilkins, 2012.
2) J. Kenneth Shultis and Richard E. Faw, "Radiation Shielding," American Nuclear Society, Inc. La Grange Park, IL, 2000.
3) William D. Ehmann and Diane E. Vance, Radiochemistry and Nuclear Methods of Analysis, John Wiley and Sons, Inc., New York, 1991.
4) Phillip M. Rinard, "Neutron Interactions with Matter," http://www.fas.org/sgp/othergov/doe/lanl/lib-www/la-pubs/00326407.pdf
5) O.W. Herman and C.W. Alexander, "A Review of Spent-Fuel Photon and Neutron Source Spectra," ORNL/CS/TM-205, DE86-006764. https://www.orau.org/PTP/PTP%20Library/library/Subject/Neutrons/spectra.pdf
6) "MCNP6 User's Manual," version 1.0, May 2013, LA-CP-13-00634, Rev. 0
7) Neutron Scattering Lengths and Cross Sections, NIST Center for Neutron Research, http://www.ncnr.nist.gov/resources/n-lengths/

Table 3. Half Thickness for Po-Be neutrons (4MeV)

Material	Half Thickness, cm
Parrafin	6.6
Water	5.4
12% Boric Acid-Water	5.3
Aluminum	7.8
Steel	4.9
Lead	6.8

*From reference [1].

Table 4. Common Neutron Shielding Materials

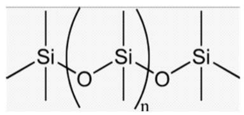
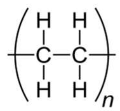
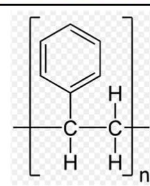
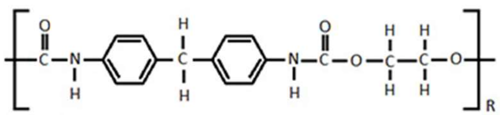
Product	Description	(g/cm^3) Density	mass% Hydrogen Content	H-atoms/ cm^3	Max Temp. (^oF)	Gamma Shielding	Notes
Masonite™	Compressed wood	1.3	6	$5x10^{22}$		poor	can be sawed/cut into shapes
Boral™	Aluminum clad B_4C	2.5	0	0		poor	expensive, difficult to cut/shape
Permali-NH™	Beechwood laminant w/ phenolic resin		6			poor	can be sawed/cut into shapes
Permali-JN™	Beechwood laminant-Borated (3%)		6			poor	can be sawed/cut into shapes
Silicone		1.1-1.2	7-10	$5.6x10^{22}$	205	poor	Can be doped with W/Bi/Fe to improve gamma shielding
Polyethylene		1.0	7.7	$8.1x10^{22}$	melts at 150-160	poor	can be sawed/cut into shapes
Borated Polyethylene	5-10% Borated	1.0	7	$8.1x10^{22}$	melts at 275	poor	can be sawed/cut into shapes
Polystyrene		1.0	7.7	$4.6x10^{22}$	melts at 130	poor	can be sawed/cut into shapes
Polyurethane		1.2	6.1	$3.7x10^{22}$	100	poor	can be sawed/cut into shapes, gives off hydrogen cyanide when it burns
Water	Water in pools or Steel/Plastic Tanks	1.0	11	$6.7x10^{22}$		poor	
Aqua-Gel™	Solid Hydrated Gel					poor	
Concrete		2.4-4.0	~1.6	$1.6x10^{22}$		moderate	density can be increased with Fe/Pb/Ba, very high temperature resistance, cheap
Shieldwerx-237	1% Borated Silicone	1.6	7.4	$4.5x10^{22}$	400	poor	
Shieldwerx-259	0.9% Boron in castable aggregate	1.2	12	$7.1x10^{22}$	150	poor	Not compatible with water or aluminum
Shieldwerx-277	Concrete w/ 1.6% B and 3x more hydrogen content than ordinary concrete	1.7	4.7	4.8×10^{22}	350F	moderate	

Table 5. Neutron Attenuation Properties of Selected Materials*

Element	Atomic Number	cross section (b) Thermal Scattering	cross section (b) Thermal Capture	Average Capture Gamma Emission (keV)	Notes
H	1	82.02	0.3326	2223	High scattering cross section. Energetic gamma upon capture.
Li	3	1.37	70.5	2094	Relatively high capture cross section. Energetic gamma upon capture.
B	5	5.24	0.103	7005	High capture cross section and relatively low energy gamma emmision for (n,a) reaction.
B (n,a)	5	5.24	767	478	
C	6	5.551	0.0035	4945	
O	8	4.232	0.00019	1469	
Al	13	1.503	0.231	3737	
Fe	26	11.62	2.56	4620	Difficult to shield activation products.
Cd	48	6.5	2520	1522	Very high capture cross section. Toxic metal. Difficult to shield activation products.
Gd	64	180	49700	1440	Very high capture cross section. Expensive.
W	74	4.6	18.3	1758	Good activation characteristics. Adds good gamma attenuation. Expensive.
Pb	82	11.118	0.171	7336	Adds good gamma attenuation. Low activation for pure grades.
Bi	83	9.156	0.0338	4118	Good activation characteristics Adds good gamma attenuation.
U-233	92	12.9	574 (fission = 531)		Very high fission cross section.
U-234	92	19.3	100 (fission = 67)		High fission cross section.
U-235	92	14	681 (fission = 585)		Very high fission cross section.
U-238 (depleted)	92	8.87	2.68 (fission = 1.68E-05)		Moderate capture cross section. Very low fission cross section. Good gamma attenuation.
Pu-239	94	7.7	1017 (fission = 747)		Very high fission cross section.

* Table compiled from data in references [1] and [7].

Appendix I.

MCNP Data

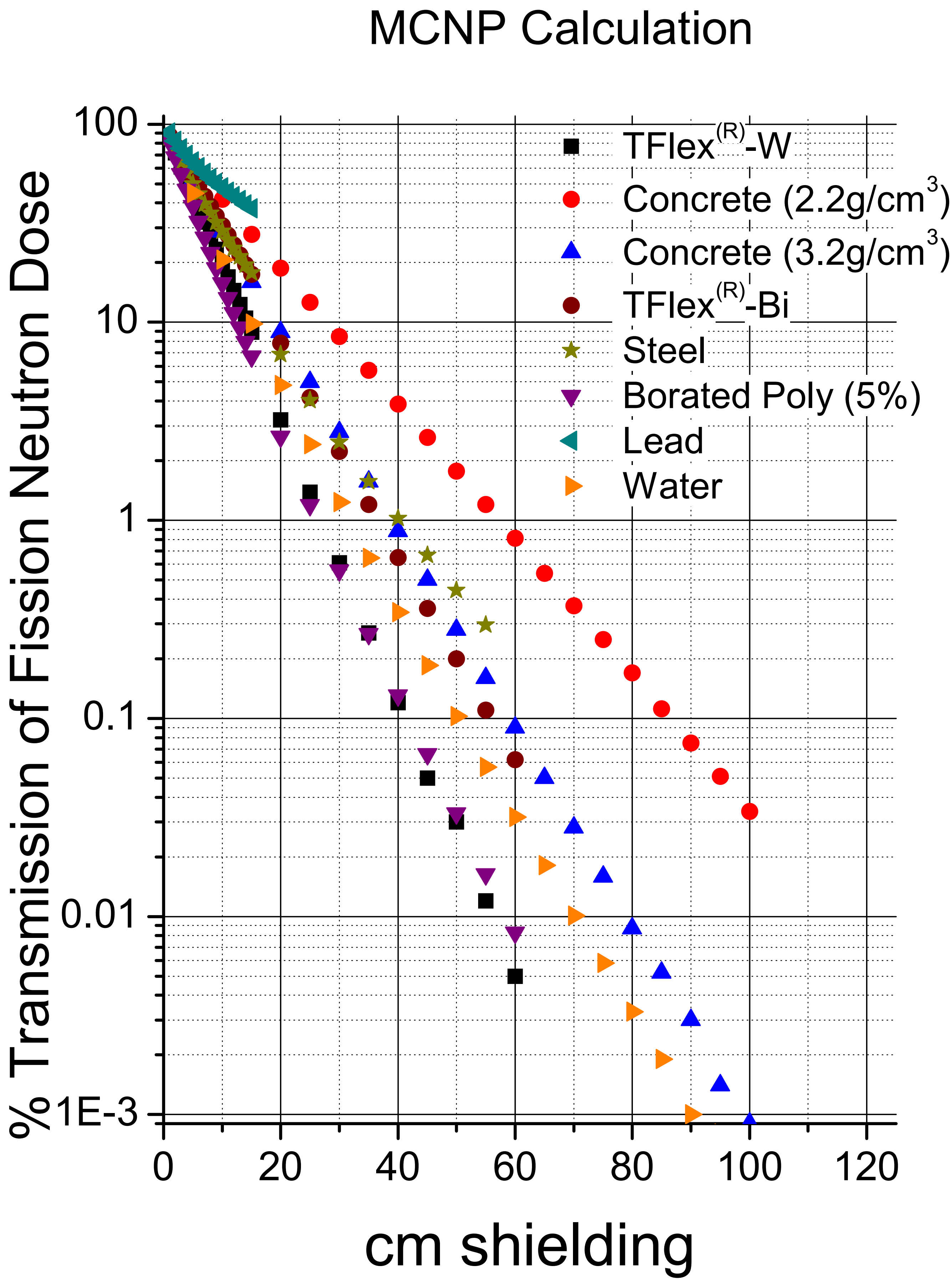


Figure 1. Attenuation of Neutron Dose. Plotted as transmission for clarity. Watt Fission Spectrum.

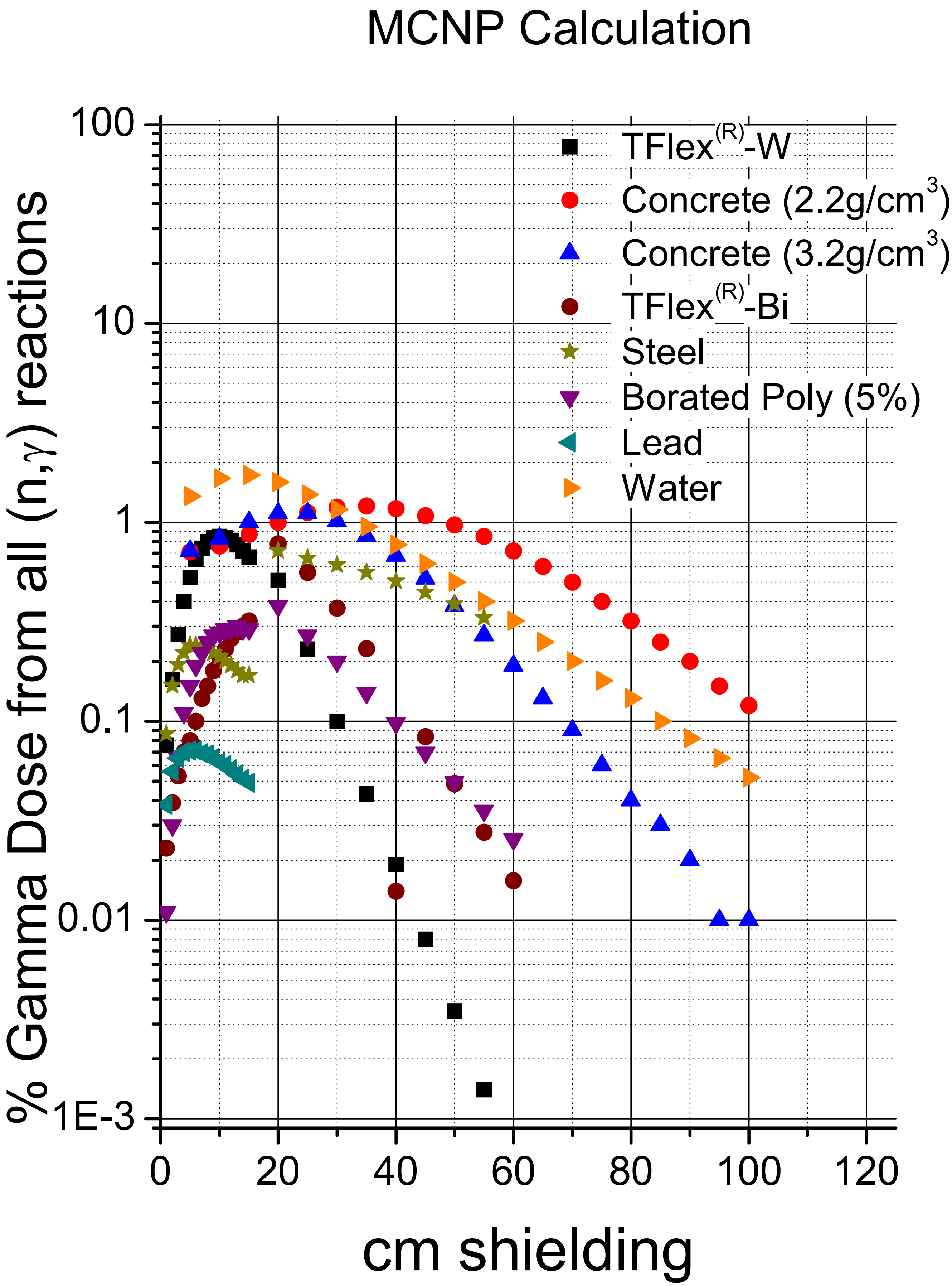


Figure 2. Secondary Gamma Dose (% of unshielded neutron dose) from all (n,γ) reactions, including radiative capture and scatter of Watt Fission Neutrons.