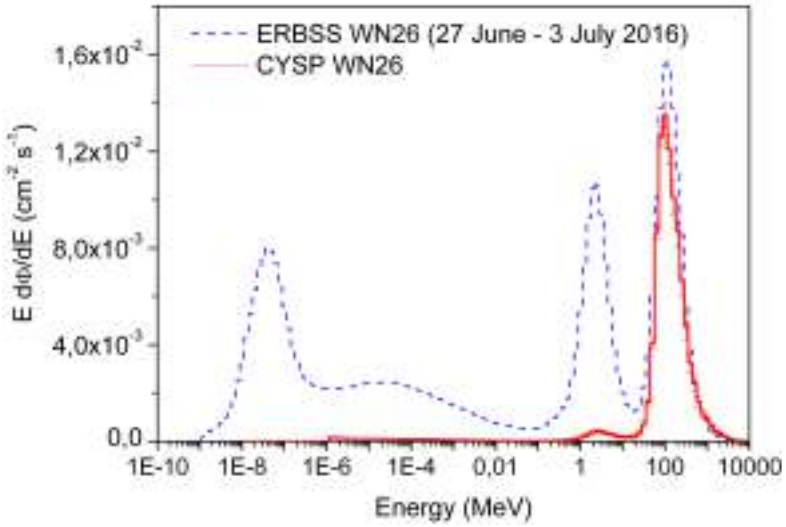
Experimental techniques for neutron detection



Roberto Bedogni

INFN-LNF Frascati

Experimental techniques for neutron detection

(L1) Introduction on neutron interaction and 2011 ICRU-recommended quantities (2 h)

- Quantities and Units (ICRU 85)
- Neutron Interaction
- Examples and videogames

(L2) Neutron Measuring Instruments (2 h)

- Detectors
- Spectrometers
- Dosemeters

(L3) Calibration fundamentals (2 h)

- Calibration?
- Calibration fields
- Workplace and calibration fields

(L4) Case study (2 h)

- CS intro
- Neutron source term
- Instruments and measurement points

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Quantities and units
ICRU85a – 2011

The fundamental concepts, quantities and units for ionizing radiation, radiation protection and spectrometry ionizing radiation may be found in

ICRU
(International Commission on Radiation Units and Measurements)
reports n.33, n. 39, n. 47, n.51, n.57, n. 60, n. 66,
n. 85
FUNDAMENTAL QUANTITIES AND UNITS FOR IONIZING RADIATION

ICRP (International Commission on Radiological Proteciton)
n. 74
Conversion Coefficients for use in Radiological Protection against External Radiation
n. 103
The 2007 Recommendations of the International Commission on Radiological Protection

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ICRU 85 System of quantities

Radiometry

Interaction quantities

Dosimetry

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Radiometry

Two classes of quantities are used in the characterization of a radiation field, referring either to the **number of particles** or to the **energy transported** by them.

The particle number **N** unit: 1

is the number of particles that are emitted, transferred, or received.
Unit: 1

The radiant energy **R** unit: J

is the energy (excluding rest energy) of the particles that are emitted, transferred or received.

The flux $\dot{N} = \frac{dN}{dt}$. unit: s^{-1}

is the quotient of dN by dt, where dN is the increment of the particle number in the time interval dt.

The energy flux $\dot{R} = \frac{dR}{dt}$. unit: W

is the quotient of dR by dt, where dR is the increment of radiant energy in the time interval dt.

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Fluence $\Phi = \frac{dN}{da}$ unit: m^{-2}

Φ is the quotient between the number of particle ***dN*** incident on the elemental sphere having cross sectional area *da,* and the cross sectional area ***da***; unit cm^{-2}.

In the definition of Fluence, all particles are equally weighted, independently on their direction.

The definition implies that an isotropic response is needed to accurately measure the quantity. Devices with prominent directional response, such as (for neutrons) activation foils or plastic detectors, should be used only in mono-directional fields.
Otherwise, the variation of their response with the angle of incidence and the directional distribution of the field should be known in advance.

In transport calculations (as those employed in the Monte Carlo codes like MCNP) the fluence in a cell having volume ΔV is calculated as

$$\Phi = \frac{\sum \ell_i}{\Delta V}$$

Where l_i are the particle path lengths in the volume ΔV.

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In a stationary field composed of particles with velocity v, the Fluence can be calculated from the **particle density** n (particles per unit volume) as

$$\Phi = nvt$$

Energy Fluence

$$\Psi = \frac{\mathrm{d}R}{\mathrm{d}a}$$

unit: J m^{-2}

In the definition of energy fluence, all energy contributions are equally weighted, independently on their direction.

The relation between the two distributions is **$\Psi(E) = E\,\Phi(E)$**

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The distributions, Φ_E and Ψ_E, of the fluence and energy-fluence with respect to energy

$$\Phi_E(E) = \frac{d\Phi}{dE}$$

$m^{-2}\ J^{-1}$, *more frequently* $cm^{-2}\ MeV^{-1}$

$$\Psi_E(E) = \frac{d\Psi}{dE}$$

m^{-2}, *more frequently* cm^{-2}

The relation between the two distributions is **$\Psi(E) = E\ \Phi(E)$**

Fluence rate and energy-fluence rate,

$$\dot{\Phi} = \frac{d\Phi}{dt}$$

$m^{-2}\ s^{-1}$

For a radiation field composed of particles of velocity **v**, the fluence rate is given by ***nv***

$$\dot{\Psi} = \frac{d\Psi}{dt}$$

$J\ m^{-2}\ s^{-1}$

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Particle radiance and energy radiance

$$\dot{\Phi}_\Omega = \frac{\mathrm{d}\dot{\Phi}}{\mathrm{d}\Omega}$$

$m^{-2}\ s^{-1}\ sr^{-1}$, *more frequently* $cm^{-2}\ s^{-1}\ sr^{-1}$

$$\dot{\Psi}_\Omega = \frac{\mathrm{d}\dot{\Psi}}{\mathrm{d}\Omega}$$

$W\ m^{-2}\ sr^{-1}$

The specification of a direction requires two variables. In a spherical coordinate system with polar angle θ and azimuthal angle φ, $d\Omega = \sin\theta\, d\theta\, d\varphi$.

Distribution of Particle radiance (or energy radiance) with respect to energy

$$\dot{\Phi}_{\Omega,E} = \frac{\mathrm{d}\dot{\Phi}_\Omega}{\mathrm{d}E} = \frac{d^3\Phi}{dt\ dE\ d\Omega}$$

$$\dot{\Psi}_{\Omega,E} = \frac{\mathrm{d}\dot{\Psi}_\Omega}{\mathrm{d}E}$$

The field of any radiation of a given particle type is completely specified by the distribution of the particle radiance with respect to particle energy, as this defines number, energy, local density, and arrival rate of particles propagating in a given direction. This quantity, as well as the distribution of the energy radiance with respect to energy, can be considered as basic in radiometry

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If the energy distribution of the particle radiance, $\Phi_{E,\Omega}$ (E,Ω), is known from calculations or measurements, then the fluence Φ may be derived from

$$\Phi_E(E) = \int \Phi_{E,\Omega}(E,\Omega)\, \mathrm{d}\Omega = \int_0^{2\pi} \mathrm{d}\varphi \int_0^{\pi} \mathrm{d}\vartheta \sin\vartheta\, \Phi_{E,\Omega}(E,\varphi,\vartheta)$$

$$\Phi = \int_0^{\infty} \Phi_E(E)\, \mathrm{d}E.$$

A spectrometer with isotropic response will directly provide the energy distribution of the fluence $\Phi_E(E)$, so that the first step is implicitly done in the instrument.

Whilst a number of photon and neutron spectrometers have been developed (and are used in practice) to measure $\Phi_E(E)$, very few instruments have been developed to determine the direction distribution of the fluence, and their use is very complex and time consuming.

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Interaction quantities / interaction coefficients

Interaction processes occur between radiation and matter. In an interaction, the energy or the direction (or both) of the incident particle is altered or the particle is absorbed.
The interaction might be followed by the emission of one or several secondary particles. **The likelihood of such interactions is characterized by interaction coefficients**. They refer to a specific interaction process, type and energy of radiation, and target or material.

The cross section $\sigma = N/\Phi$
unit: m^2
special unit: barn, b: 1 b = 10^{-28} m^{-2}

The cross section of a target entity (e.g. a nucleus), for a particular interaction produced by incident charged or uncharged particles of a given type and energy, **is the quotient of N by Φ, where N is the mean number of such interactions per target entity subjected to the particle fluence Φ.**

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μ

is the linear attenuation coefficient: the **probability that at normal incidence an uncharged particle undergoes an interaction in a material layer of thickness dl is μdl.**

The mass attenuation coefficient
unit: $m^2\ kg^{-1}$

$$\frac{\mu}{\rho} = \frac{1}{\rho}\frac{dN}{dl}\frac{1}{N}$$

The mass attenuation coefficient of a material, for **uncharged particles** of a given type and energy, is the quotient of dN/N by $\rho \bullet$dl, where dN/N is the mean fraction of the particles that experience interactions in traversing a distance dl in the material of density ρ.

The reciprocal of μ is called the **mean free path** of an uncharged particle.

The linear attenuation coefficient, μ, depends on the density, ρ, of the material. This dependence is largely attenuated by using the mass attenuation coefficient, μ/ρ.

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Demonstrate: $\mu = \sigma n$

Let **n** the atom density (cm^{-3})

In a volume with area **A** and thickness **dl**, the number of atoms is **n A dl.**

The number of interactions is therefore $\sigma \Phi$ **(n A dl).**

The number of incident particles is Φ **A**

If μ is the interaction probability per unit pathlenght

μ = 1/dl * number of interations / number of incident particles

$\mu = 1/dl * (\sigma \Phi$ **n A dl**$) / (\Phi$ **A**$) = \sigma$ **n**

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The mass attenuation coefficient can be expressed in terms of the total cross section, σ. The mass attenuation coefficient is the product of σ and $\mathbf{N_A/M}$, where $\mathbf{N_A}$ is the Avogadro constant, and **M** is the molar mass of the target material:

$$\frac{\mu}{\rho} = \frac{N_A}{M}\sigma = \frac{N_A}{M}\sum_J \sigma_J$$

where σ_J is the component cross section relating to an interaction of type J.

The mass attenuation coefficient of a compound material is usually treated as if the latter consisted of independent atoms.

$$\frac{\mu}{\rho} = \frac{1}{\rho}\sum_L (n_t)_L \sigma_L = \frac{1}{\rho}\sum_L (n_t)_L \sum_J \sigma_{L,J}$$

where $(n_t)_L$ is the number density of target entities of type L, σ_L the total cross section for an entity L, and $\sigma_{L,J}$ the cross section of an interaction of type J for a single target entity of type L.

Effects of molecular, chemical, or crystalline environment are ignored.

This is justified in most cases, but can occasionally lead to errors, for example, in the interaction of slow neutrons with molecules, particularly those containing hydrogen.

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Demonstrate that **mean free path = 1/μ**

Mean free path = average distance before a collision occur

Because of the definition of μ, for a beam traversing an infinitesimal thickness dl the number of interactions will be proportional to the initial number, N, and the thicnkess dl through a constant, μ (linear attenuation coefficient)

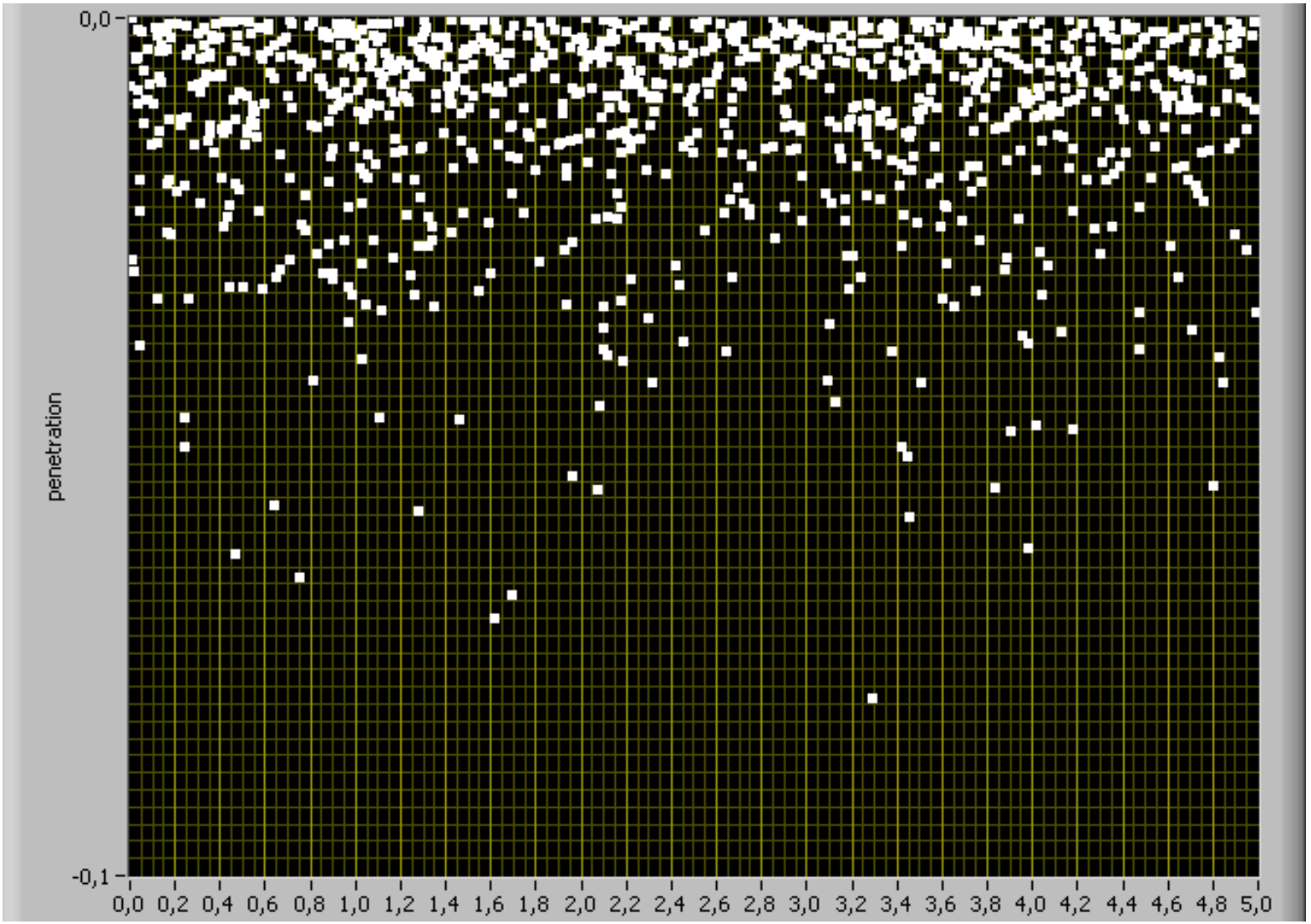
$$dN = - \mu N \, dl$$

By integrating dN/N for a finite thickness l: $N(l) = N_o e^{-\mu \cdot l}$

$N(l)/N_o = e^{-\mu \cdot l}$ represents the survival probability for a thickness l.

The average travelled distance is therefore

$$m.f.p = \langle l \rangle = \frac{\int_0^\infty l \, e^{-\mu \cdot l} dl}{\int_0^\infty e^{-\mu \cdot l} dl} = \frac{1/\mu^2}{1/\mu} = 1/\mu$$

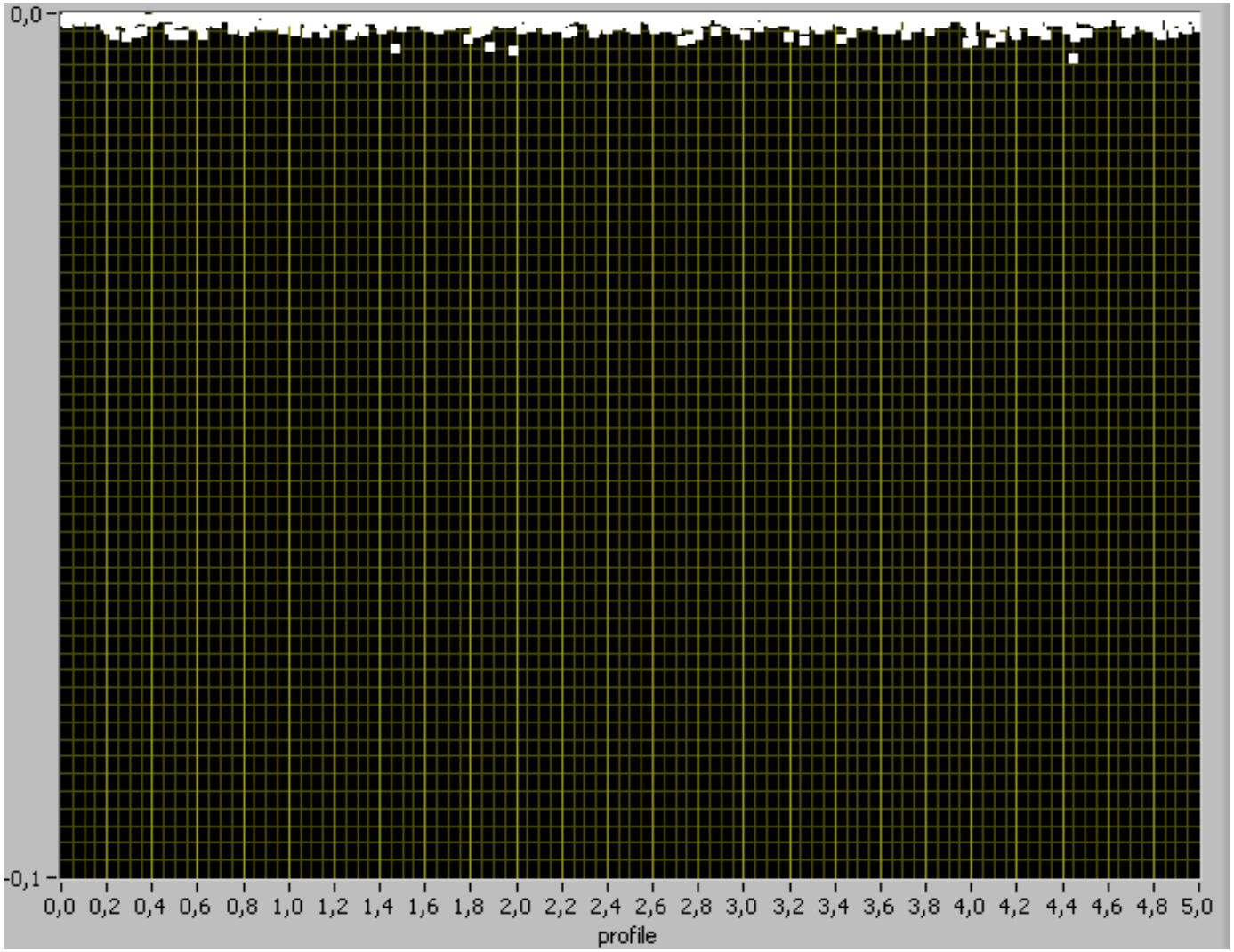


C: \ L1 \ Quantities and Units (ICRU 85)

Thermal neutrons on Boron Carbide (λ = 0.01 cm)

C: \ L1 \ Quantities and Units (ICRU 85)

Thermal neutrons on Gadolinum (λ = 7E-4 cm)



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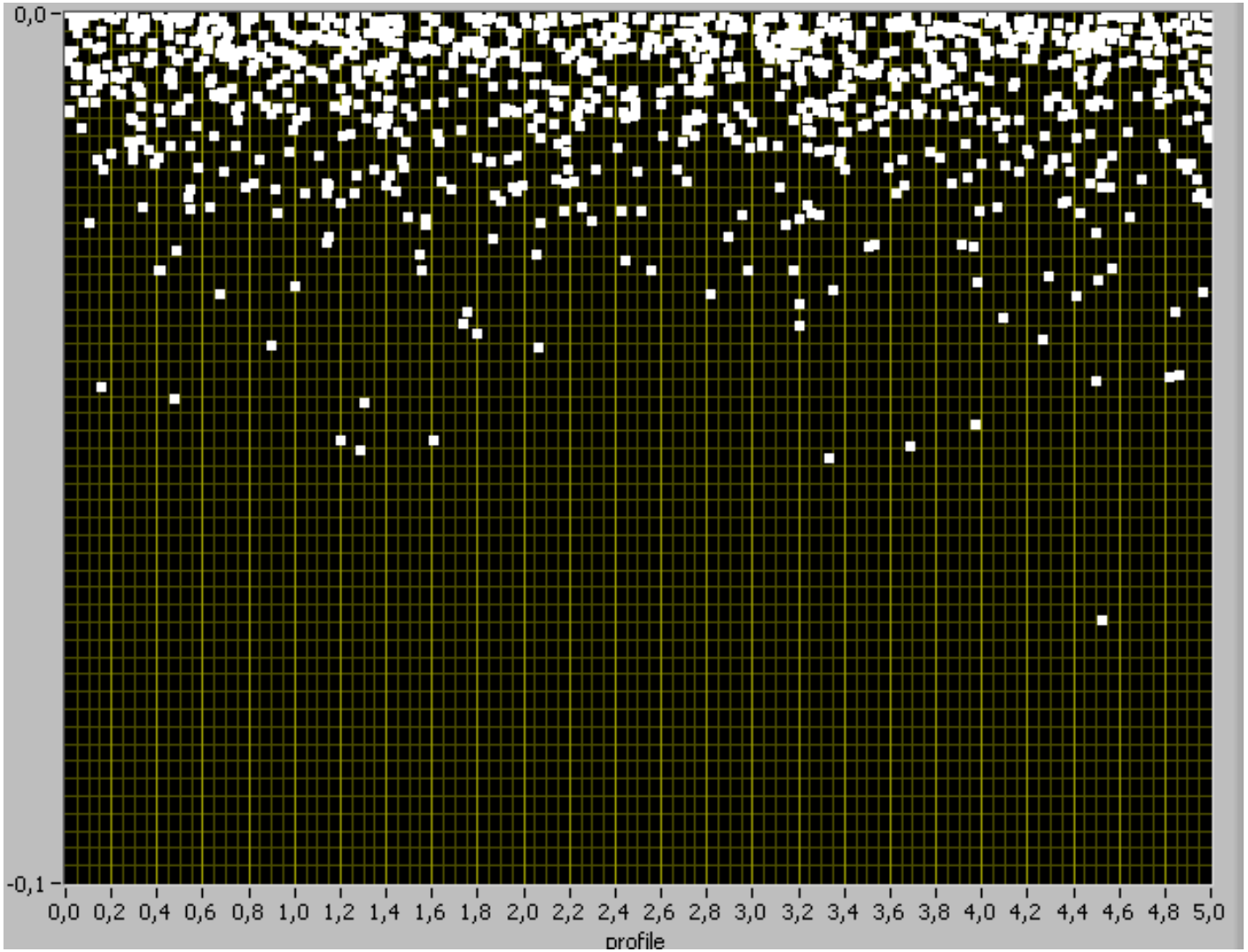
Thermal neutrons on CADMIUM (λ = 0.009 cm)



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The energy-transfer coefficient μ_{tr}

unit: m^{-1}

$$\mu_{tr} = \frac{1}{dl} \cdot \frac{dR_{tr}}{R}$$

The energy-transfer coefficient, μ_{tr} of a material for uncharged particles of a given type and energy, is

the fraction of radiant energy transferred to kinetic energy of charged particles by interactions of the uncharged particles of incident radiant energy R, per unit pathlength

If ***f*** is the average fraction of energy transferred per collision, then

$$\mu_{tr} = f\,\mu$$

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The mass energy-transfer coefficient

unit: $m^2\ kg^{-1}$

$$\frac{\mu_{tr}}{\rho} = \frac{1}{\rho dl}\frac{dR_{tr}}{R}$$

If incident uncharged particles of a given type and energy can produce several types of independent interactions in a target entity, the mass energy-transfer coefficient can be expressed in terms of the component cross sections, σ_J, by the relationship

$$\frac{\mu_{tr}}{\rho} = \frac{N_A}{M}\sum_J f_J \sigma_J$$

where $\mathbf{f_J}$ is the **quotient of the mean energy transferred to kinetic energy of charged particles in an interaction of type J by the kinetic energy**.

Also,

$$\frac{\mu_{tr}}{\rho} = \frac{\mu}{\rho} f$$

$$f = \frac{\sum_J f_J \sigma_J}{\sum_J \sigma_J}$$

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The mass energy-absorption coefficient

unit: $m^2\ kg^{-1}$

A fraction g of the kinetic energy transferred to charged particles can be subsequently lost in radiative processes (bremsstrahlung, in-flight annihilation, and fluorescence radiations) as the charged particles slow to rest in the material, and this fraction **g (average)** is specific to the material and energy.

$$\frac{\mu_{en}}{\rho} = \frac{\mu_{tr}}{\rho}(1-g)$$

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Dosimetry

The effects of radiation on matter depend on the radiation field, as specified by the radiometric quantities, and on the interactions between radiation and matter, as characterized by the interaction quantities.

Dosimetric quantities, which are selected to provide a physical measure to correlate with actual or potential effects, are products of radiometric quantities and interaction coefficients.

In calculations, the values of these quantities and coefficients must be known, while measurements might not require this information.

1st step: estimating the energy transferred to the secondary cp

The quantity kerma

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Conversion of energy - **Kerma**

The kerma, K, for **ionizing uncharged particles**, is the quotient of dEtr by dm, where dEtr is the mean sum of the initial kinetic energies of all the charged particles liberated in a mass dm of a material by the uncharged particles incident on dm, thus

$$K = \frac{dE_{tr}}{dm}$$

Unit: J kg^{-1} special name: gray, Gy

The quantity d**E**tr includes the kinetic energy of the charged particles emitted in the decay of excited atoms/molecules or in nuclear de-excitation or disintegration.

Demonstrate: $\mathbf{K = \Phi E \mu_{tr}/\rho = \Psi \mu_{tr}/\rho}$ (monoenergetic beam, E)

Suppose a cylinder-like volume element with base a and height dl, and normally incident particles of fluence N/A

$\mathbf{dE_{tr} = - R\ dl\ \mu_{tr}}$

$\mathbf{dEtr / dm = (R\ dl\ \mu_{tr}) / (\rho\ A\ dl)}$

$\mathbf{= (N\ E\ dl\ \mu_{tr}) / (\rho\ A\ dl)}$

$\mathbf{= (N/A)\ E\ \mu_{tr} / \rho}$

$\mathbf{= \Phi\ E\ \mu_{tr}/\rho}$

$\mathbf{= \Psi\ \mu_{tr}/\rho}$

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Conversion of energy - **Kerma**

The kerma per fluence, $K/\Phi = E\ \mu_{tr}/\rho$, is termed the **kerma coefficient** for uncharged particles of energy **E** in a specified material. The term kerma coefficient is used in preference to the older term kerma factor, as the word coefficient implies a physical dimension whereas the word factor does not.

For a generic energy distribution of the particle fluence, $\Phi_E(E)$:

$$K = \int \Phi_E E \frac{\mu_{tr}}{\rho} \, dE = \int \Psi_E \frac{\mu_{tr}}{\rho} dE$$

The expression makes it clear that one can refer to a value of kerma or kerma rate for a specified material at a point in free space, or inside a different material. Thus, one can speak, for example, of the air kerma at a point inside a water phantom.

Kerma rate

$$\dot{K} = \frac{dK}{dt}$$

Unit: Gy/s

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Mean energy imparted

$$\bar{\varepsilon} = R_{\text{in}} - R_{\text{out}} + \sum Q$$

unit: J, also eV

The mean energy imparted to the matter in a given volume equals the

mean radiant energy $\mathbf{R}_{\text{in}}$ of charged and uncharged particles that **enter the volume**

minus

mean radiant energy $\mathbf{R}_{\text{out}}$ of charged and uncharged particles that **leave the volume**

*plus the mean sum of all changes of the rest energy of nuclei and elementary particles that occur in the volume (**Q** > 0: decrease of rest energy; **Q** < 0: increase of rest energy).*

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Absorbed dose

(Non-stochastic quantity)

$$D = \frac{\mathrm{d}\bar{\varepsilon}}{\mathrm{d}m}$$

unit: J kg^{-1} = Gy

D is the mean energy imparted by ionizing radiation to matter of mass dm.

The absorbed dose, D, is considered a point quantity, but it should be recognized that the physical process does not allow dm to approach zero in the mathematical sense.

The absorbed dose should not be identified with the organ absorbed dose (which is an extension of the absorbed dose)

Absorbed-Dose rate

$$\dot{D} = \frac{\mathrm{d}D}{\mathrm{d}t}$$

unit: J kg^{-1} s^{-1} = Gy s^{-1}

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NEUTRON INTERACTION

Production – Absorption - Leakage = 0

Source
(n,f)
(n, 2n)
.....

(n, p)
(n, alpha)
(n, gamma)
(n, t)
.....

Scattering
.....

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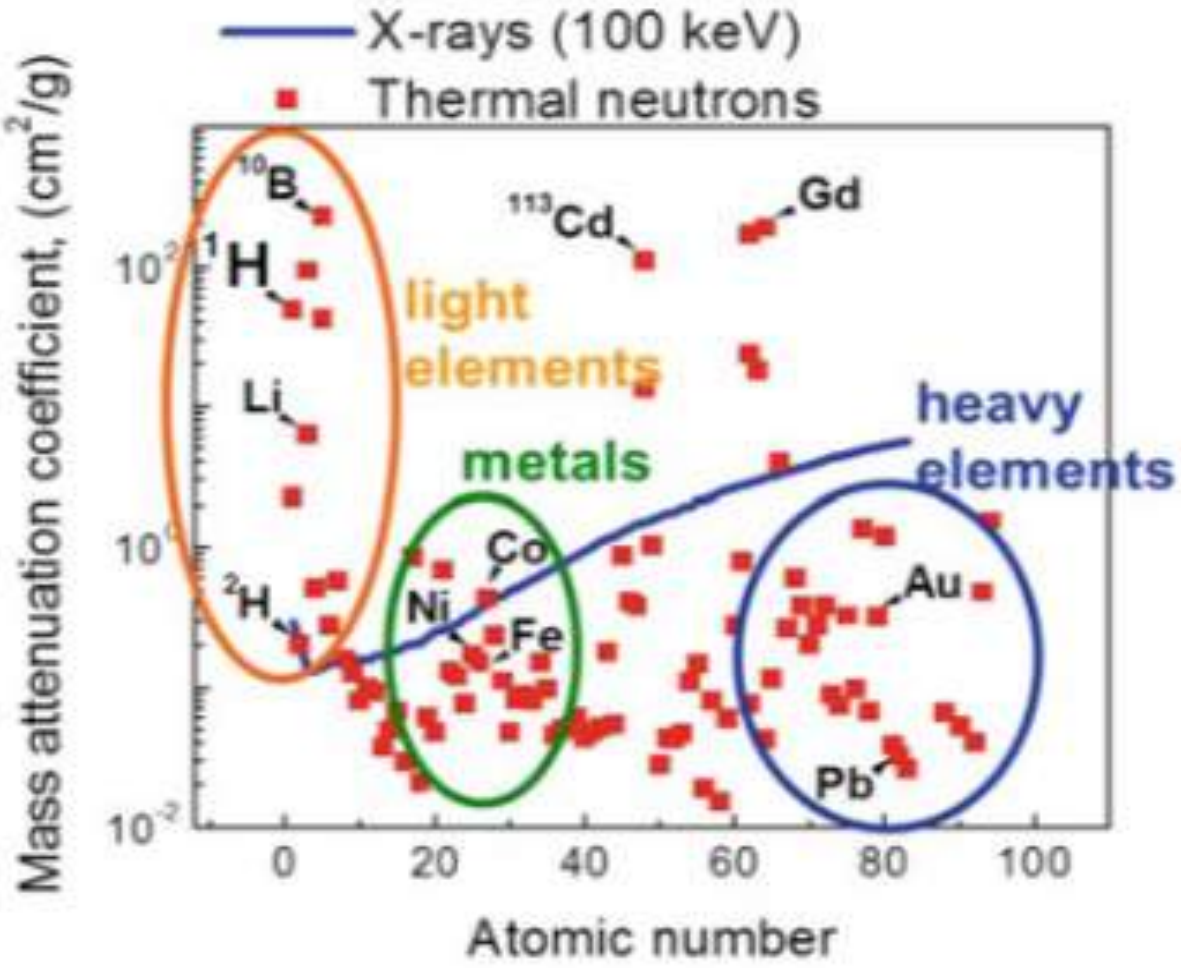
No simple dependence exists from interaction probability and A or Z.



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UNLIKE X-rays / gammas (attenuation and detection)

Neutrons are typically MORE sensitive to some light elements like H, Li, B

The distribution of attenuation coefficients for neutrons is independent of Z which helps to possibly distinguish neighbouring elements, while for X-rays one finds an approximately exponential increase with the atomic number;

The neutrons easily penetrate thick layers of metals like Pb, Fe and Cu where X-rays or gammas fail;

Material	Au	Ag	Cu	Sn	Bronze	Pb	Zn	Fe
Σ (cm^{-1})	6.28	3.99	0.99	0.20	0.87	0.37	0.34	1.20

The neutrons can distinguish between isotopes (for instance 1H and 2H) which is not the case for X-rays.

Instead of THREE main effects, many nuclear reactions are to be cosidered

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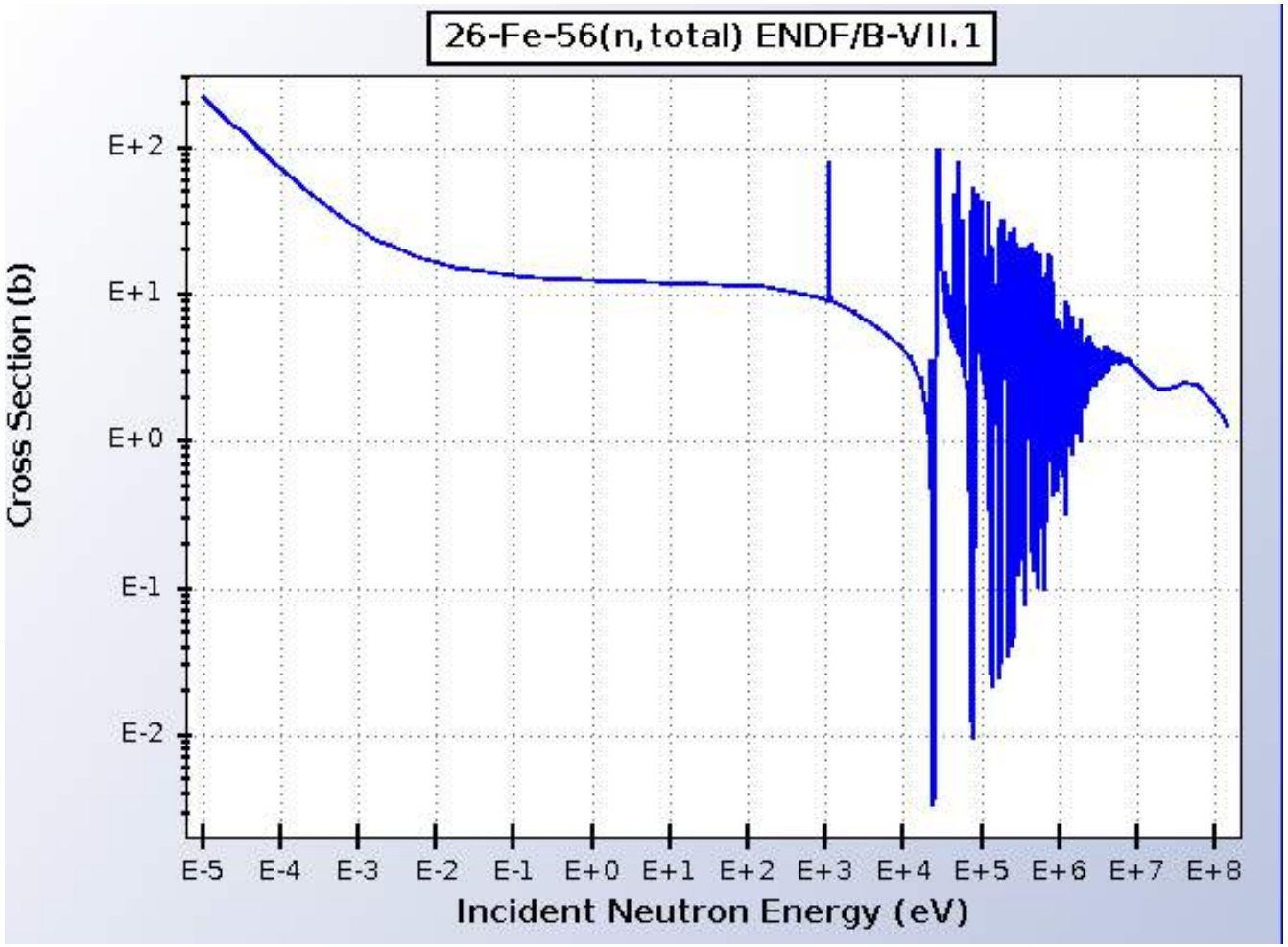
Cross Section and Energy

Slow neutrons (includes thermal): 1/V REGION
Epithermal or intermediate (< MeV), A>40: RESONANCE REGION
Fast neutron (> MeV): Large res/ INELASTIC



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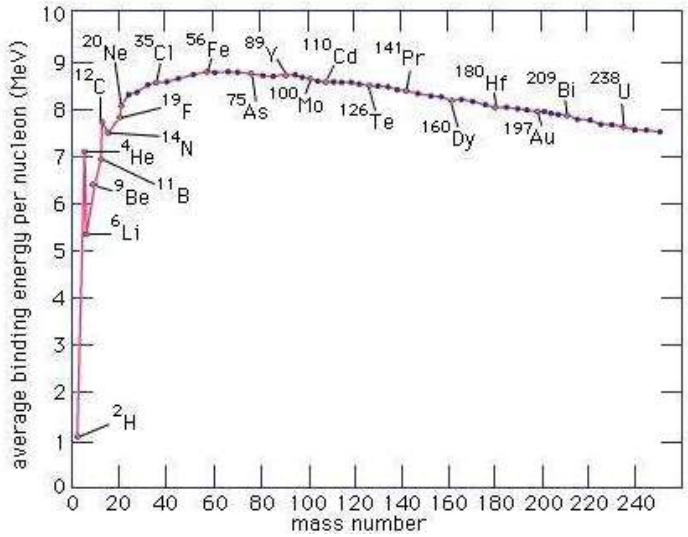
Interaction through compound nucleus ($\tau \approx 10^{-14}$ s / n transit time 10^{-21} s)
When E < < Binding Energy / nucleon.
Breit-Wigner –like

Level spacing: for 150 < A < 240: 0.1 MeV near fundamental,
1-10 eV near 1st excited (+ 8 MeV)

Light nuclei 1 MeV near fundamental,
10 keV near 1st excited (+ 8 MeV)



Interaction with single nucleons (small groups)
Independent particles
When E > > BE/n.

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If the levels were not affected by "uncertainty" (in Heisenberg sense), the Xs plot would differ from zero only in correspondence of the excited states (sharp resonances).

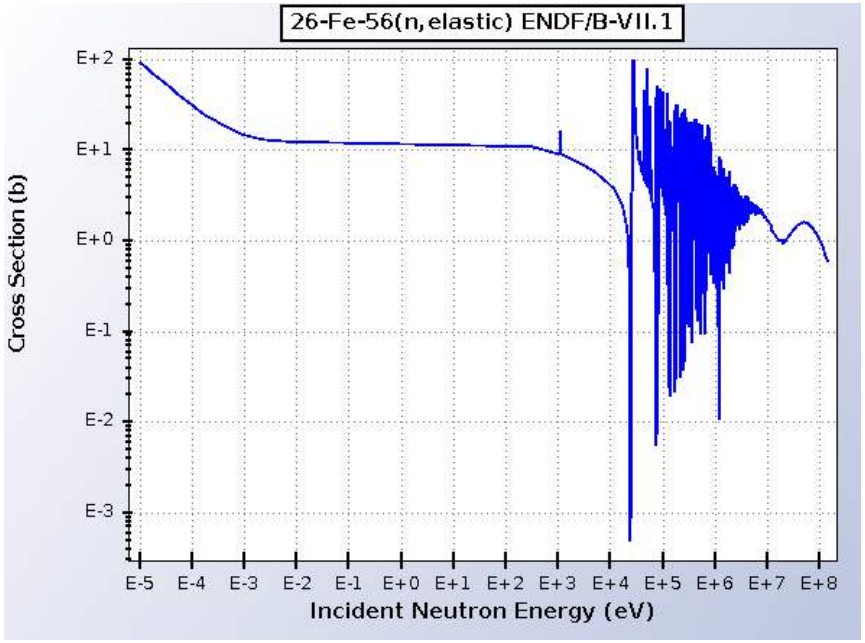
The width of a resonance (delta E ≈ Γ) is given by the uncertainty principle, where delta t (≈ τ , or half life) is the time constant of the excited state.

In case of the competition between scattering and radiative capture (n, γ) in the intermediate energy region,

$$\Gamma = \Gamma_\gamma + \Gamma_n$$

Γ_γ / Γ probability for compound nucleus to gamma-decay

Γ_n / Γ probability for compound nucleus to re-emit a neutron (scattering)



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SLOW neutrons tend to interact via

ELASTIC SCATTERING: In ES too little energy can be transferred for a convenient use in detectors

BUT Elastic scattering with light nuclei is very probable and helps to bring the slow neutron into thermal equilibrium with the medium. At thermal energies (25 meV) a number of convenient (n, charged particle) positive-Q capture reactions can be used for detection

RADIATIVE CAPTURE (n,g) is the most probable in many materials: good for shielding but less good for detections (efficiently catch the gamma!).
However, activation foils are based on (n,g)

MUCH BETTER are (n,alpha), (n,p), (n, fission) but the set of materials is restricted

BEST Slow neutron absorbers
^{3}He (n, alpha), ^{10}B (n, alpha), ^{6}Li (n, alpha)
Gadolinium, Dysprosium, Indium, Cadmium (n, g)

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RELEVANT THERMAL CAPTURE Xs

^{3}He	5333 barn
^{6}Li	940
^{10}B	3835
^{113}Cd	20600
^{115}In	202
^{164}Dy	2840

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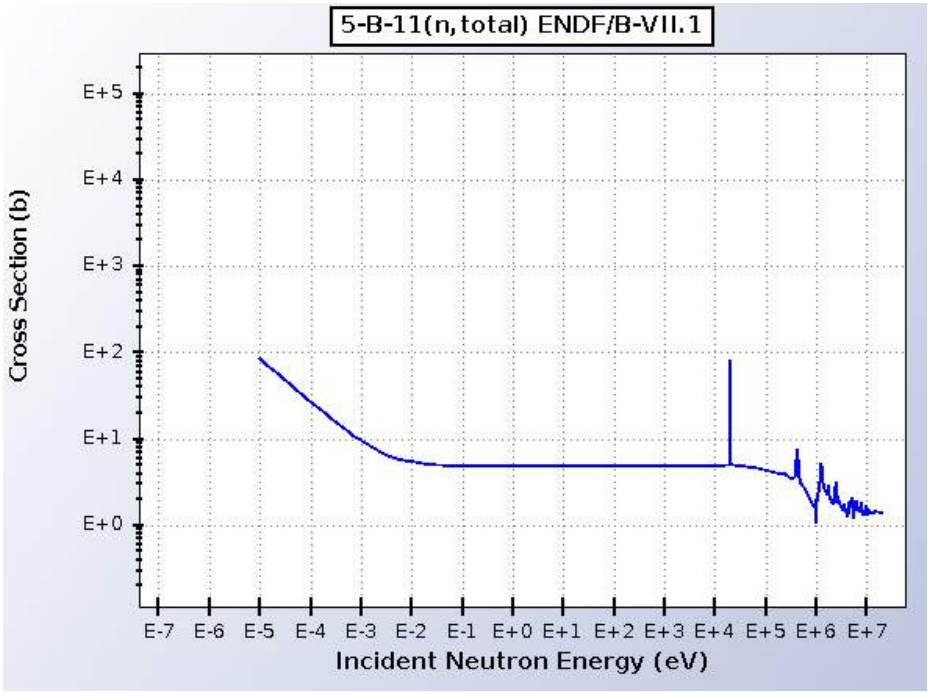
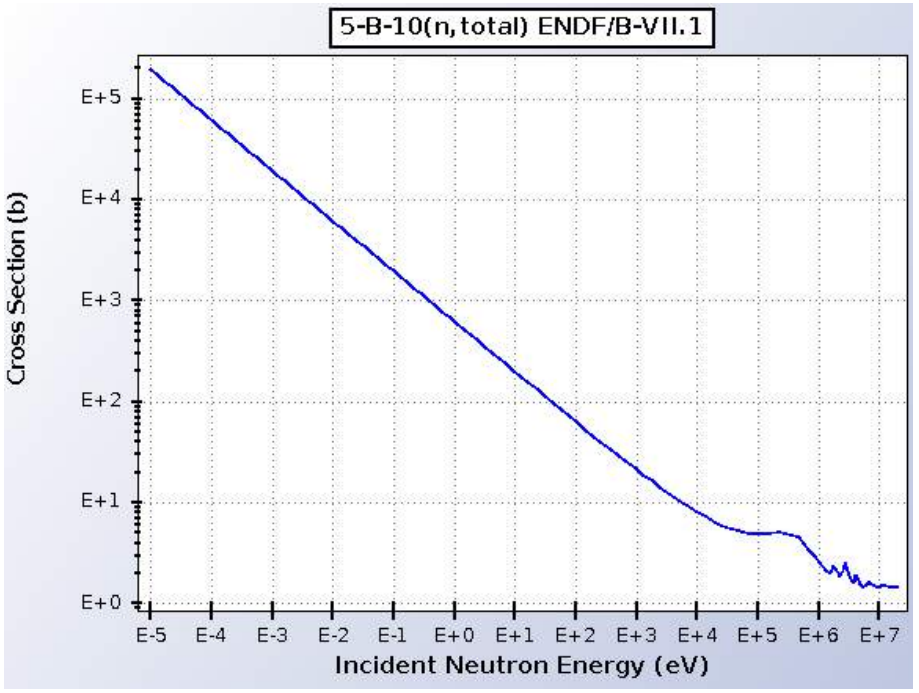
In slow and epithermal region, capture Xs **tends** to follow $E^{-1/2}$





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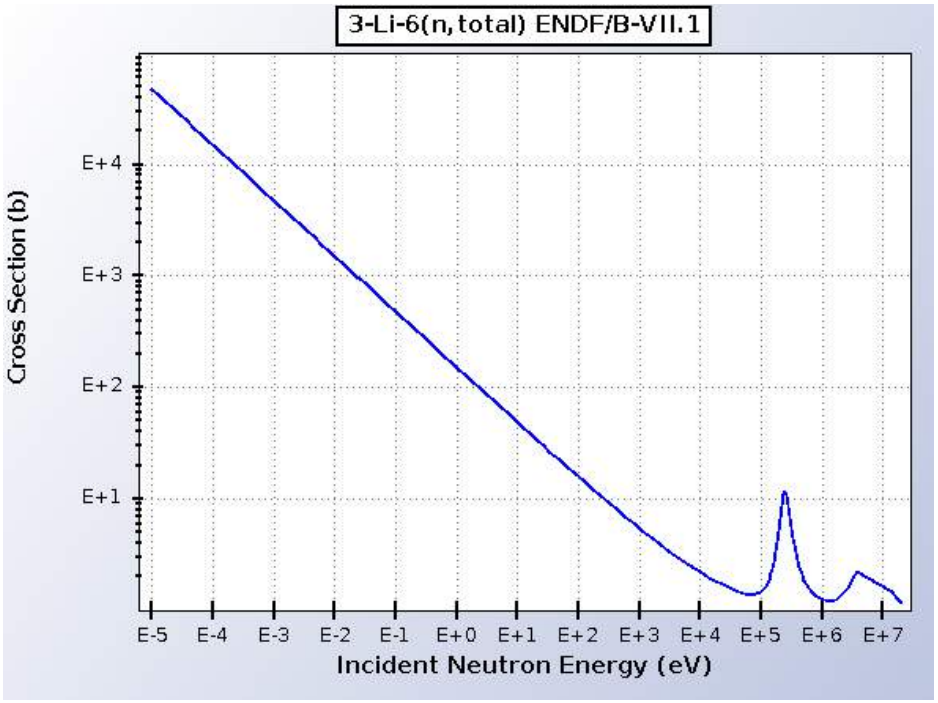
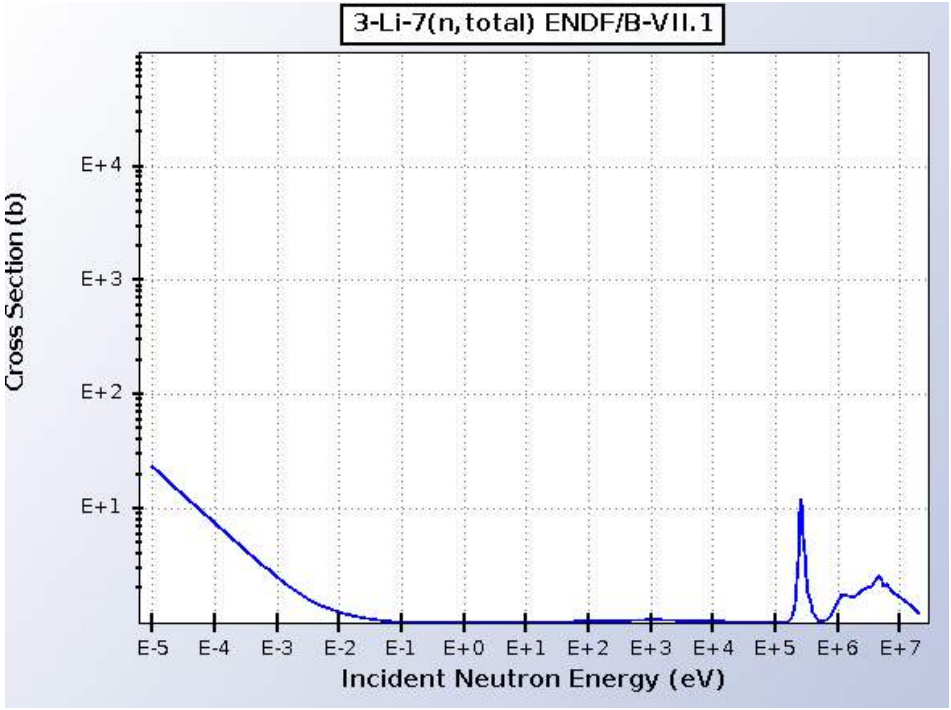
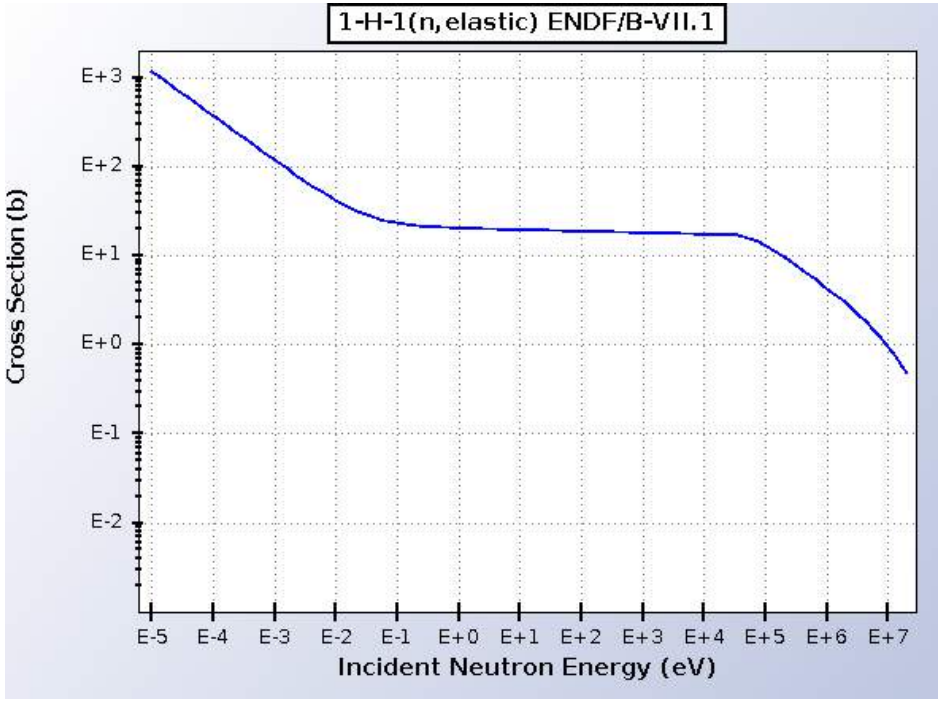
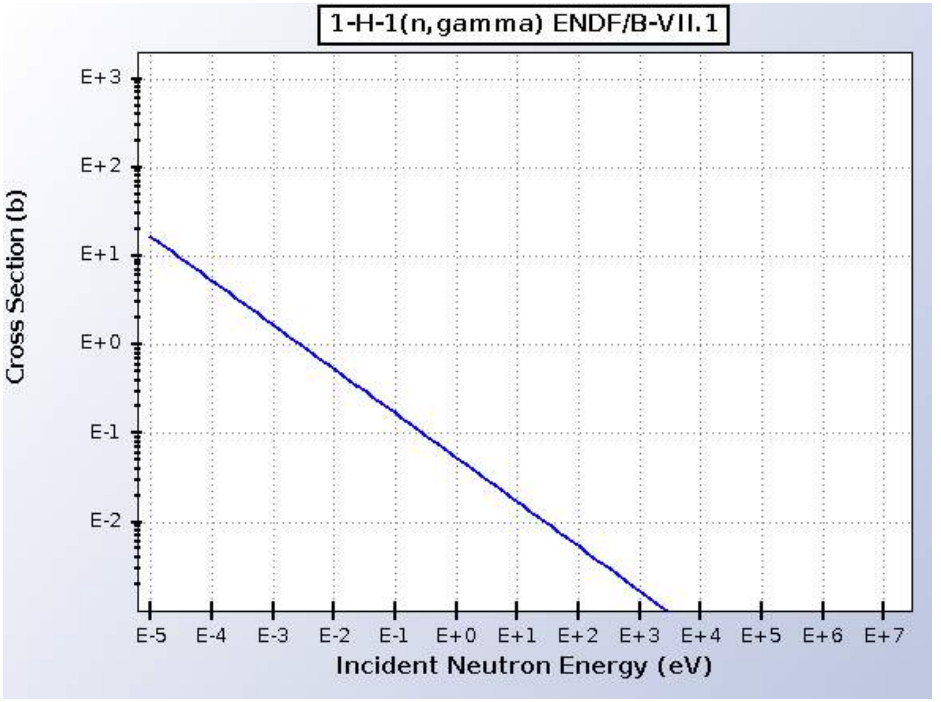




C: \ L1 \ Neutron Interaction

In thermal and epithermal region, capture Xs **tends** to follow $E^{-1/2}$





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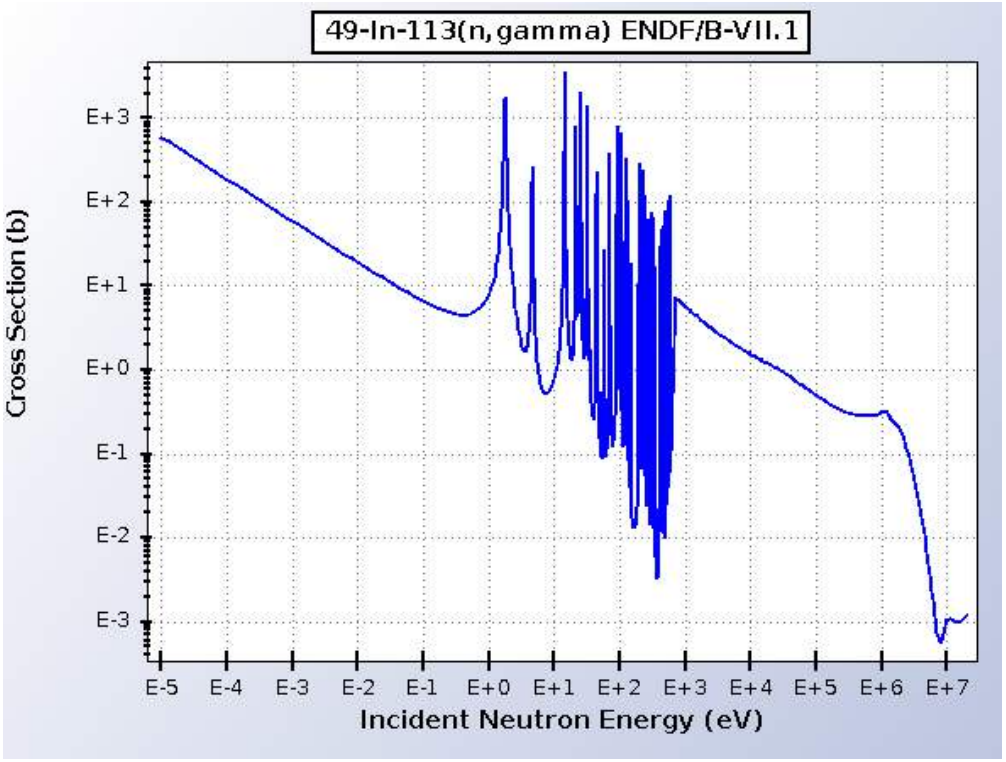
For medium-weight nuclei (A > 40), "signature-like" **resonances** appear from 1 eV and 100 keV
USED IN THE ACTIVATION FOIL TECHNIQUE

Breit-Wigner like modelling for compound nucleus



$$\sigma_{n\gamma}(E) = \pi \cdot \lambda_{rid}^2 \frac{\Gamma_n \cdot \Gamma_\gamma}{(E - E_R)^2 + (\Gamma/2)^2}$$

Γ = FWHM of the resonance

$\Gamma_\gamma \approx 0.1$ eV, proportional to gamma emission probability

$\Gamma_n \approx E^{1/2}$

If $E << E_R$ and $\Gamma << E_R$

$$\sigma_{n\gamma}(E) \propto \frac{1}{\sqrt{E}}$$

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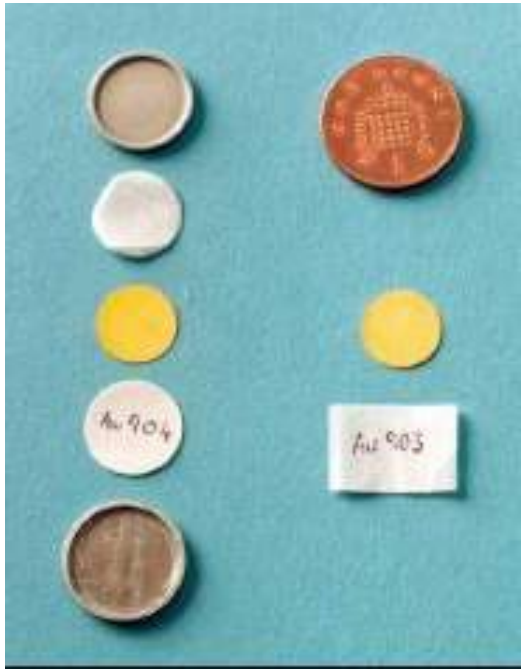
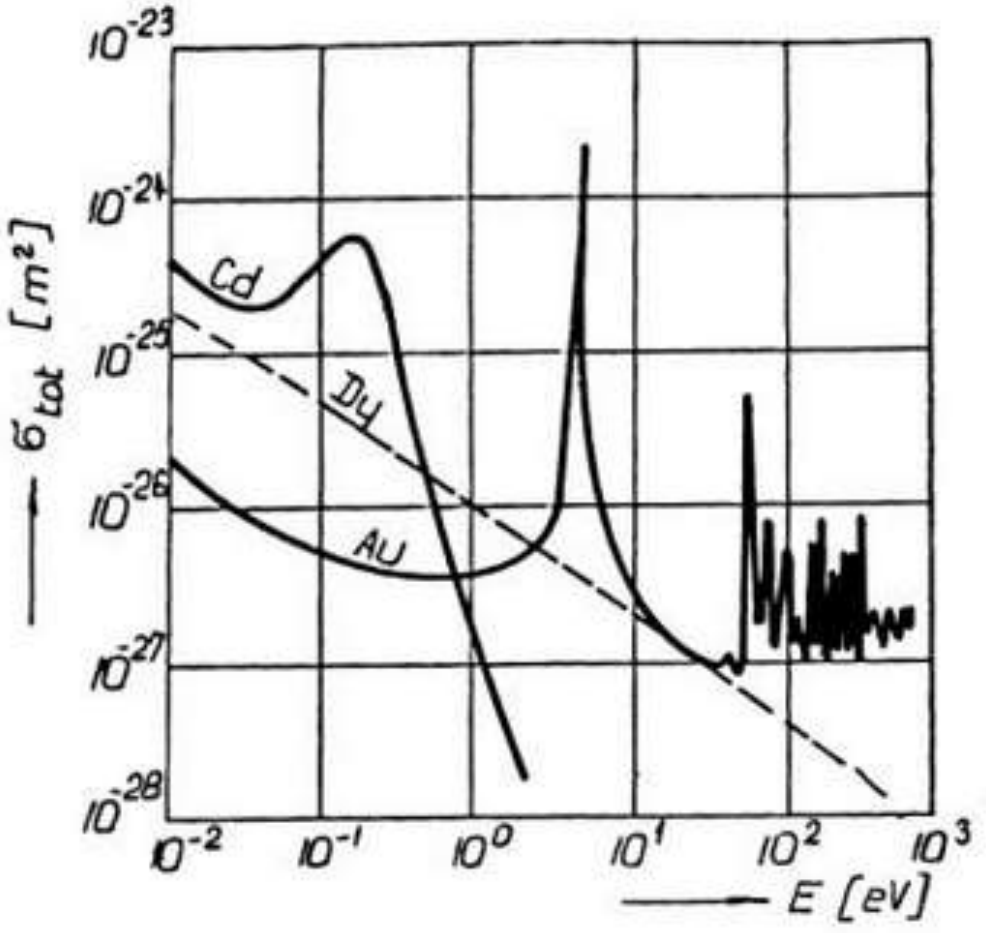
Possibly, "signature-like" **resonances** appear from 1 eV and 100 keV (excited states)
USED IN THE ACTIVATION FOIL TECHNIQUE





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FAST neutrons tend to interact via

ELASTIC SCATTERING: As the transferred energy can be seen by CP detectors, ES can be convenient used in detection.

Secondary particles are RECOIL NUCLEI who received significant fraction of neutron energy (for H, even 100%. H is the most efficient moderator).
Good for detection & for attenuation

Slowing down consists of **MULTIPLE scattering** in a material, each dispersing a significant fraction of the neutron.

Heavier nuclei gradually become less efficient for slowing down neutrons.

Above given thresholds (material dependant), **INELASTIC** reactions can take place. The target nucleus is promoted to an **excited state** (thus subtracting MORE energy from the neutron with respect to corresponding ES). De-excitation is accompanied by gamma emission.
Good for attenuation, not really for measurement.

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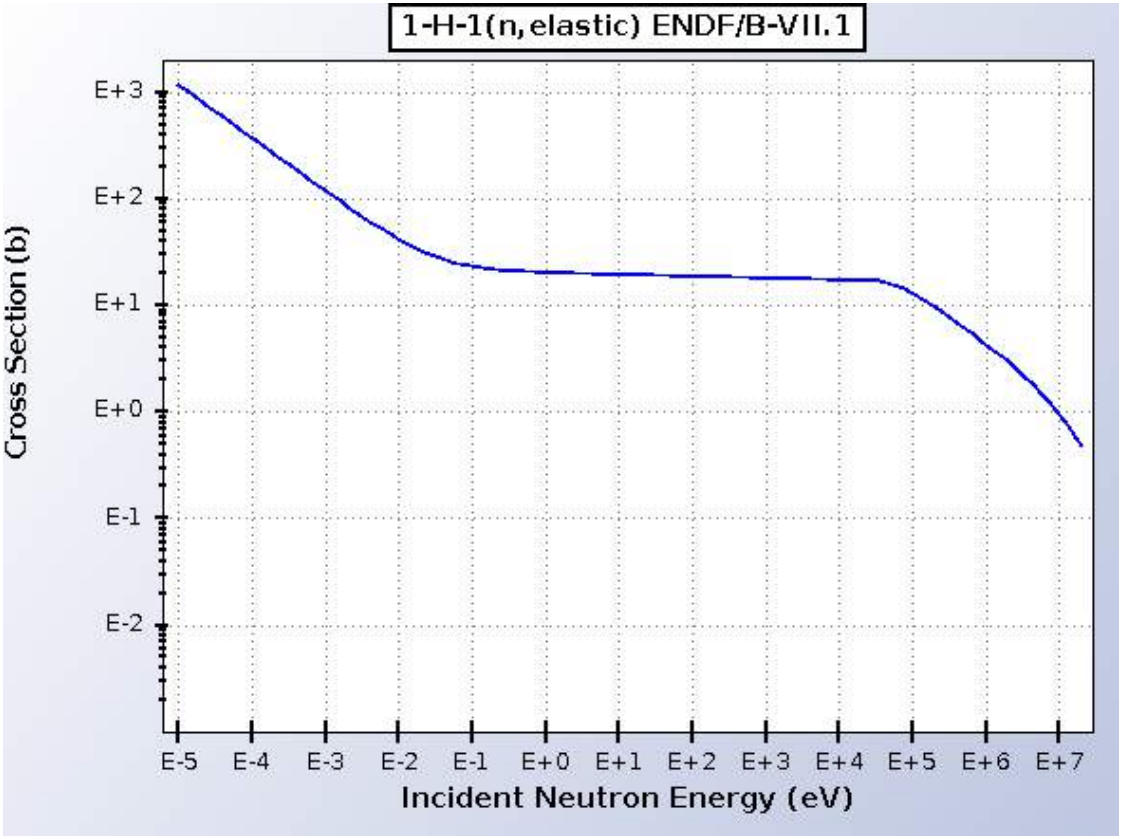
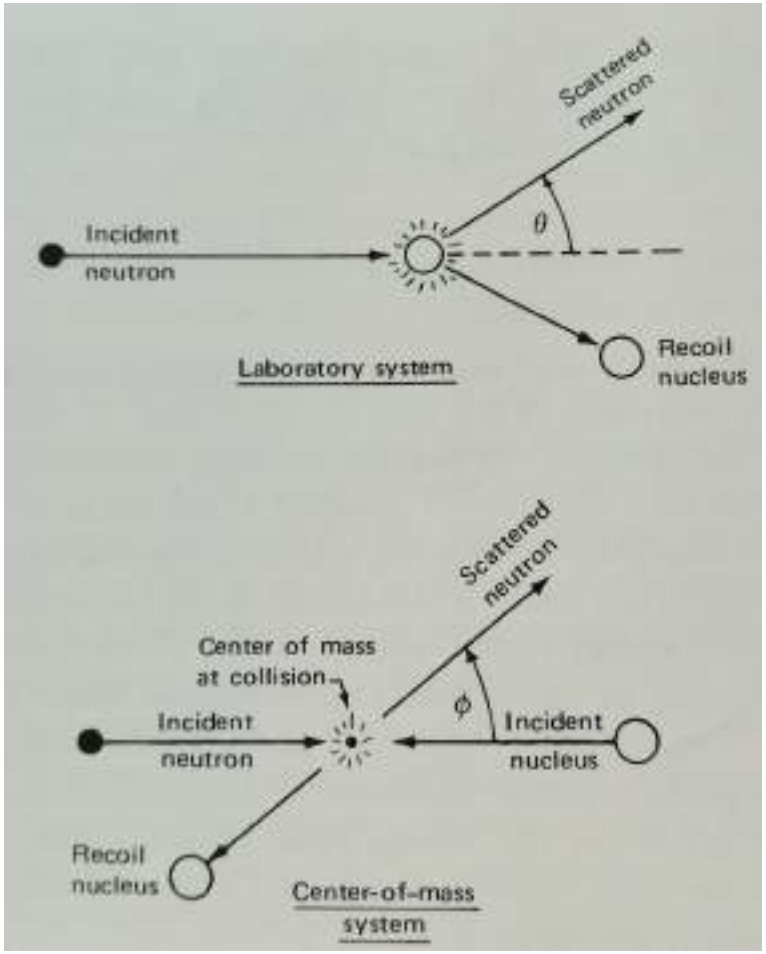
ELASTIC SCATTERING





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ELASTIC SCATTERING

In CM system, E/Eo is:

$$\frac{E}{E_0} = \frac{A^2 + 2A\cos\phi + 1}{(A+1)^2}$$

ϕ And θ are thus related:

$$\cos\theta = \frac{A\cos\phi + 1}{(A^2 + 2A\cos\phi + 1)^{1/2}}$$

In "backscattering" $\theta=\pi$ condition:

$$\frac{E_{min}}{E_0} = \left(\frac{A-1}{A+1}\right)^2$$

Below 0.1 MeV neutron scattering is
- isotropic in CM
- Forward peaked in LAB. Tends to be isotropic in LAB for heavy nuclei

$$\cos\bar{\theta} = \frac{2}{3A}$$

Above 0.1 MeV ES is ANISOTROPIC (except lightest nuclei)

$$\bar{\theta} = \frac{200}{(E_0)^{1/2}A^{1/3}} \text{ degrees}$$

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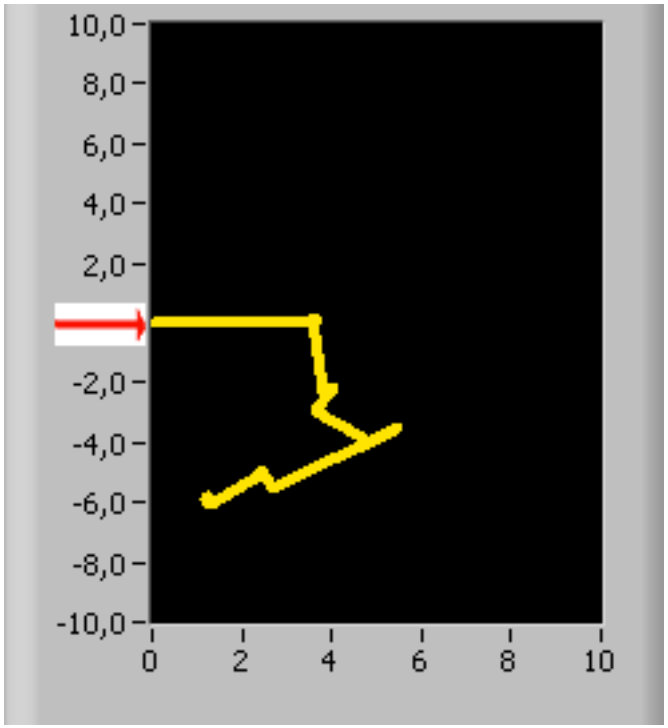
ELASTIC SCATTERING
Energy loss

The typical energy decrease of a neutron during multiple elastic scattering suggests an EXPONENTIAL LAW (example 1 MeV in PE)



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ELASTIC SCATTERING

Energy loss

The average logarithmic energy decrease per collision, < ln E / E' > is related to A

$$\xi = 1 + \frac{\alpha}{1-\alpha}\ln\alpha \qquad \alpha = \left(\frac{A-1}{A+1}\right)^2$$

If Eo = 1 MeV, in with Hydrogen

<n coll > = ln (1 / 2.53E-8) = 17.5

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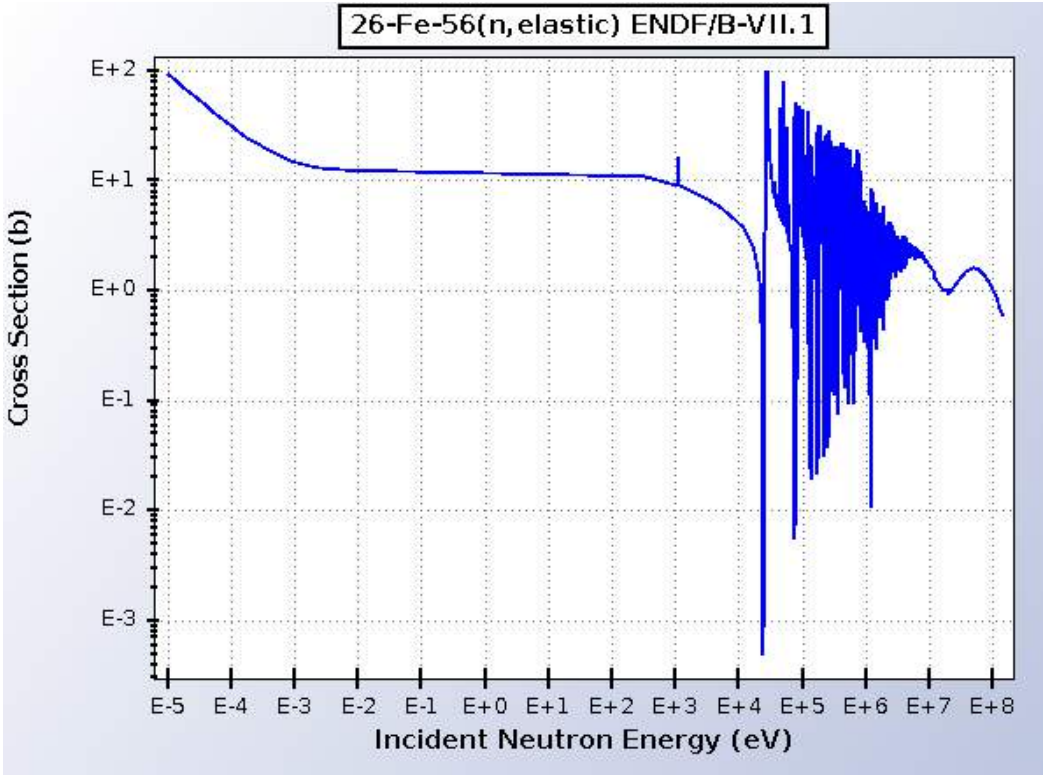
ELASTIC AND RESONANT SCATTERING

Corpuscular / Wave aspect



In intermediate-heavy nuclei (such as Fe)

Classical elastic scattering dominates the eV-keV

Resonant scattering (Breit-Wigner like) dominates E>100 keV

In addition: "interference" effects

$$\sigma_s(E) = \frac{a}{(E-E_R)^2 + (\Gamma/2)^2} + b + \frac{c \cdot (E-E_R)}{(E-E_R)^2 + (\Gamma/2)^2}$$

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INELASTIC SCATTERING

In IS, portion of the incident energy appears as excitation of the target, which de-excites emitting a gamma (IMPORTANT secondary source of gammas).

Inelastically scattered neutrons tend to preserve original direction but with much reduced energy (Good to attenuate fast neutrons)

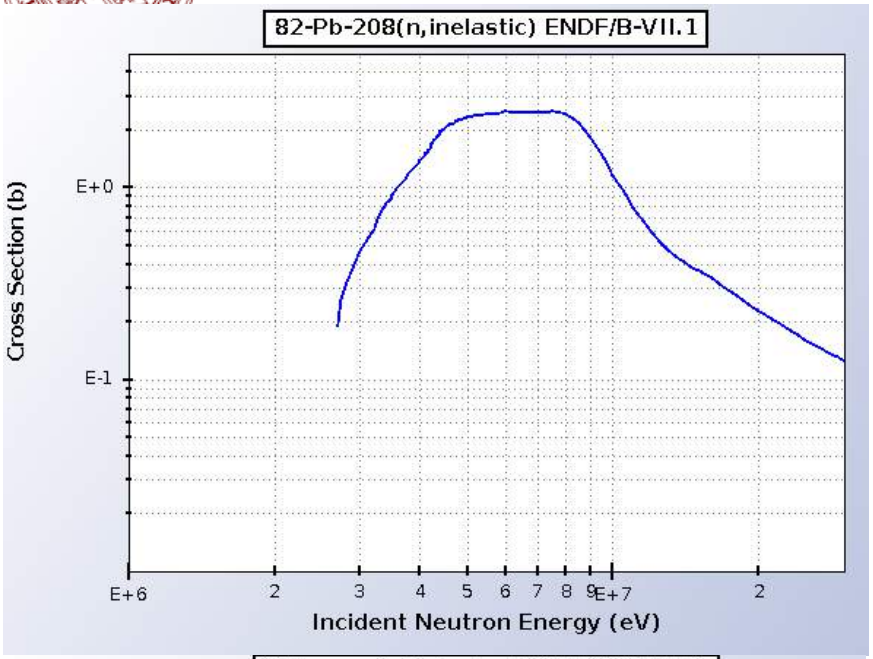
IS scattering CANNOT occur unless the neutron energy exceeds 1st excited state of the target (material dependent, from 0.1 to few MeV)
Higher excited level can be reached as the Neutron energy increases.

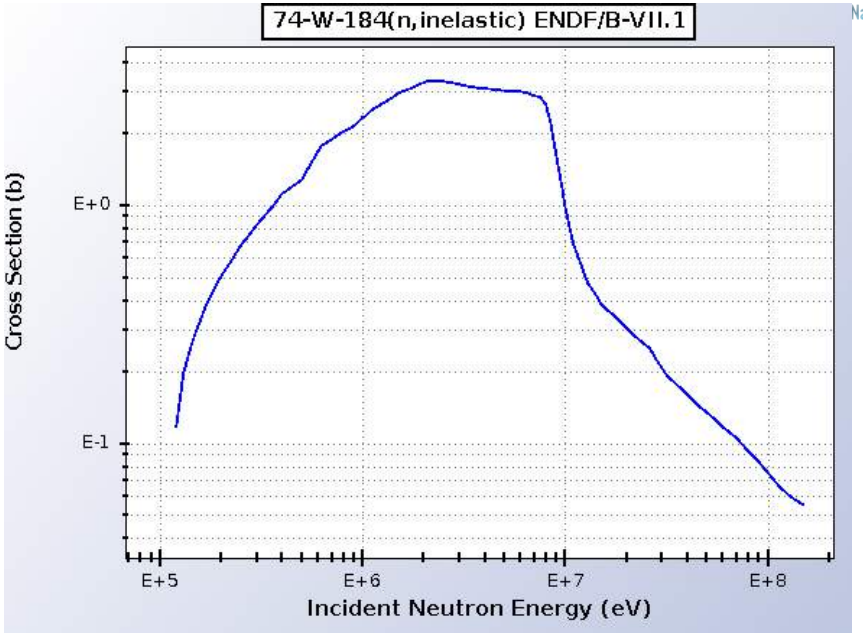
As the excitation Energy increases, levels become **CONTINUOUS**

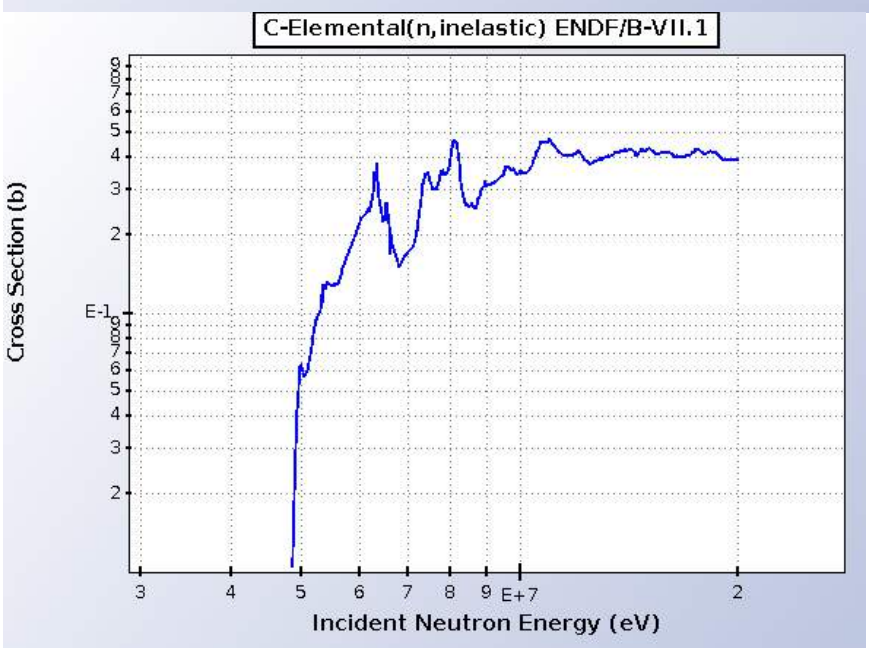
For all BUT the first level, **de-excitation** can occur:
- By emitting a single photon down to the ground state
- By emitting a cascade of photons, as a result of intermediate levels.

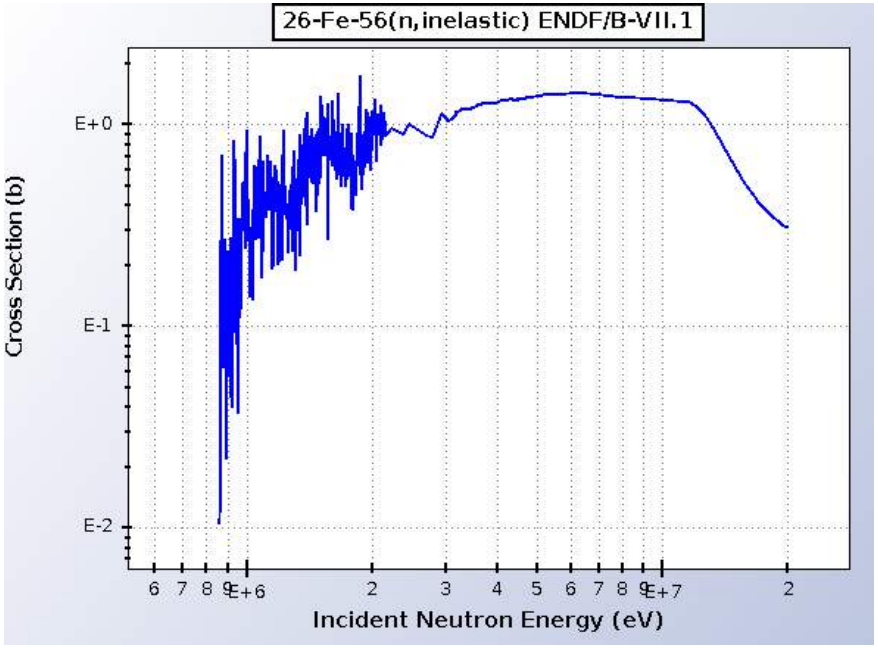
INELASTIC Xs tables are very complex. **Roughly:**
Xs increases with increasing A and E
Threshold decreases with increasing Z

C: \ L1 \ Neutron Interaction









C: \ L1 \ Examples

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Demonstrate: $\mu = \sigma n$

Let **n** the atom density (cm^{-3})

In a volume with area **A** and thickness **dl**, the number of atoms is **n A dl.**

The number of interactions is therefore $\sigma \Phi$ **(n A dl).**

The number of incident particles is Φ **A**

If μ is the interaction probability per unit pathlenght

μ = 1/dl * number of interations / number of incident particles

$\mu = 1/dl * (\sigma \Phi$ **n A dl**$) / (\Phi$ **A**$) = \sigma$ **n**

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Demonstrate that **mean free path = 1/μ**

Mean free path = average distance before a collision occur

Because of the definition of μ, for a beam traversing an infinitesimal thickness dl the number of interactions will be proportional to the initial number, N, and the thicnkess dl through a constant, μ (linear attenuation coefficient)

$$dN = - \mu N \, dl$$

By integrating dN/N for a finite thickness l: $N(l) = N_o e^{-\mu \cdot l}$

$N(l)/N_o = e^{-\mu \cdot l}$ represents the survival probability for a thickness l.

The average travelled distance is therefore

$$m.f.p = \langle l \rangle = \frac{\int_0^\infty l \, e^{-\mu \cdot l} dl}{\int_0^\infty e^{-\mu \cdot l} dl} = \frac{1/\mu^2}{1/\mu} = 1/\mu$$

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Demonstrate: $\mathbf{K = \Phi E \mu_{tr}/\rho = \Psi \mu_{tr}/\rho}$ (monoenergetic beam, E)

Suppose a cylinder-like volume element with base a and height dl, and normally incident particles of fluence N/A

$$\mathbf{dE_{tr} = - R\, dl\, \mu_{tr}}$$

$$\mathbf{dEtr / dm = (R\, dl\, \mu_{tr}) / (\rho\, A\, dl)}$$

$$\mathbf{= (N\, E\, dl\, \mu_{tr}) / (\rho\, A\, dl)}$$

$$\mathbf{= (N/A)\, E\, \mu_{tr} / \rho}$$

$$\mathbf{= \Phi\, E\, \mu_{tr}/\rho}$$

$$\mathbf{= \Psi\, \mu_{tr}/\rho}$$

C: \ L1 \ Videogames

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POLYETHYLENE SLAB

INPUT
- Monoenergetic neutrons of variable energy
- Variable PE thickness
- Variable ^{10}B %

EFFECTS INCLUDED
- Elastic scattering in H
- Absorption in H
- Absorption in B

OUTPUT
- Particle tracks
- Energy loss
- Average penetration length (La)
- Average number of collisions before absorption (Na)
- % backscattered n
- % absorbed n
- % transmitted n

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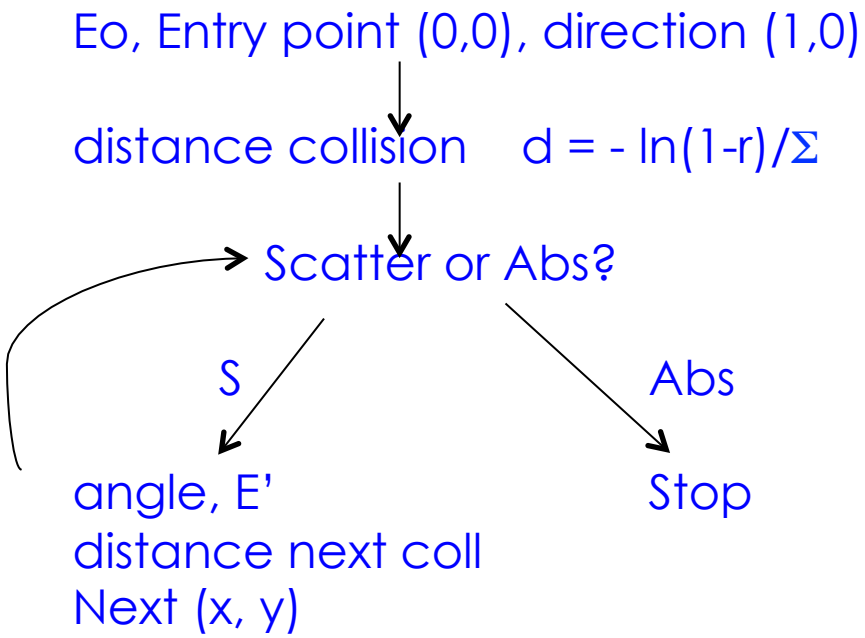
POLYETHYLENE SLAB

Flowchart



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POLYETHYLENE SLAB **Useful games**

"Infinite" block:
- Lp(E)
- Np(E)

At 1 eV:
- Effect of %B on Lp
- Effect of %B on Np

At 1 keV
- Effect of %B on Lp
- Effect of %B on Np

At 1 MeV
- Effect of %B on Lp
- Effect of %B on Np

Finite block: use of %B to reduce transmission
1 eV, 1 keV, 1 MeV