Interaction of Neutrons With Matter

- Neutrons interactions depends on **energies**: from > 100 MeV to < 1 eV

- Neutrons are **uncharged particles**:
  ⇒ **No** interaction with atomic electrons of material
  ⇒ interaction with the nuclei of these atoms

- The **nuclear force**, leading to these interactions, is very short ranged
  ⇒ neutrons have to pass close to a nucleus to be able to interact
  ≈ 10^{-13} cm (nucleus radius)

- Because of small size of the nucleus in relation to the atom, neutrons have low probability of interaction
  ⇒ **long travelling distances in matter**

See Krane, Segre', …

While bound neutrons in stable nuclei are stable, free neutrons are unstable; they undergo beta decay with a lifetime of just under 15 minutes

$$n \rightarrow p + e^- + \bar{\nu}$$

$$\boxed{\tau_n = 885.7 \pm 0.8 \text{ s}}$$

Long life times $\Rightarrow$ before decaying possibility to interact

$\Rightarrow$ n physics …

- Free neutrons are produced in nuclear **fission** and **fusion**

- Dedicated neutron sources like research reactors and spallation sources produce free neutrons for the use in irradiation neutron scattering exp.

*N.B. Vita media del protone: τ_p > 1.6*10^{33} anni*
età dell'universo: (13.72 ± 0,12) × 10^9 anni.

beta decay $p \rightarrow n + e^+ + \nu$ *can only occur with bound protons*

Neutron sources

These devices find use in a diverse array of applications in areas of physics, engineering, medicine, nuclear weapons, petroleum exploration, biology, chemistry, nuclear power and other industries…

There are several kinds of neutron sources:

Small size

Medium size

Large size

Small Size devices

Radioisotopes which undergo spontaneous fission

^{252}Cf and all other spontaneous fission neutron sources are produced by

irradiating uranium or another transuranic element in a nuclear reactor;

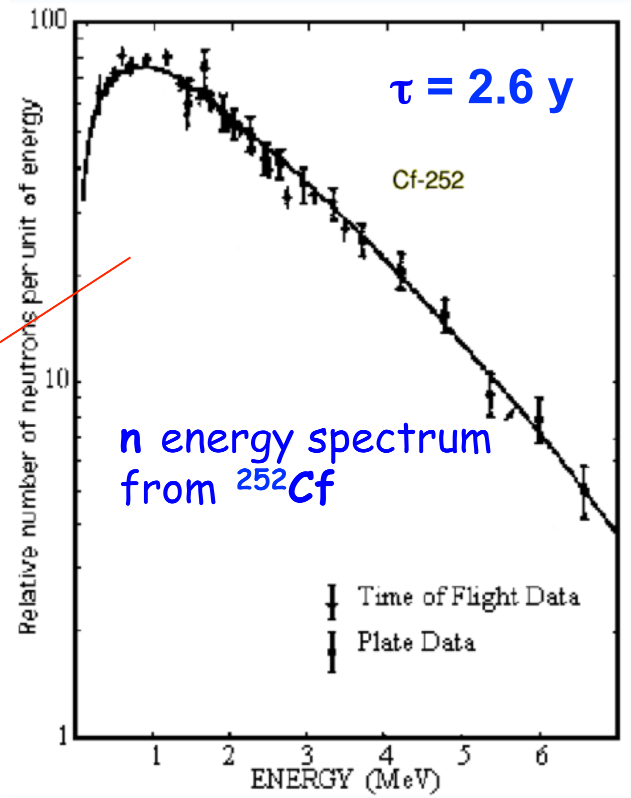
neutrons are absorbed in the starting material and its subsequent reaction products, transmuting the starting material into the SF isotope.

When purchased new typical ^{252}Cf neutron sources emit between 1×10^7 to 1×10^9 neutrons per second but with a half life of **2.6 years**.

Maxwellian distribution

$$\frac{dN}{dE} = \sqrt{E}\exp\left(\frac{-E}{T}\right)$$

T = 1.3 MeV



τ = 2.6 y

n energy spectrum from ^{252}Cf

Radioisotopes which decay with alpha particles packed in a low-Z elemental matrix

(α,n) reactions

Neutrons are produced when alpha particles impinge upon any of several low atomic weight isotopes including isotopes of **beryllium, carbon and oxygen**.

This nuclear reaction can be used to construct a neutron source by intermixing a radioisotope that emits alpha particles such as radium or polonium with a low atomic weight isotope, usually in the form of a mixture of powders of the two materials. Typical emission rates for alpha reaction neutron sources range from 1×10^6 to 1×10^8 neutrons per second.

$$\alpha + {}^{9}Be \rightarrow {}^{13}C^{*} \rightarrow \begin{cases} {}^{12}C^{*} + n \\ {}^{8}Be + \alpha + n \\ 3\alpha + n \end{cases}$$

γ = 4.4 MeV

from ^{241}Am, ^{238}Pu, ^{242}Cm, ^{226}Ra, ^{227}Ac...

As an example, a representative **alpha-beryllium** neutron source can be expected to produce approximately **30 neutrons for every one million alpha particles**.

The useful lifetime for these types of sources is highly variable, depending upon the half life of the radioisotope that emits the alpha particles. The size and cost of these neutron sources are also comparable to spontaneous fission sources.

Usual combinations of materials are plutonium-beryllium (**PuBe**), americium-beryllium (**AmBe**), or americium-lithium (**AmLi**).

Radioisotopes which decay with high energy photons co-located with beryllium or deuterium

(γ,n) reactions

Gamma radiation with an energy exceeding the neutron binding energy of a nucleus can eject a neutron.

Two examples and their decay products:

- $^{9}Be + \gamma\ (>1.7\ MeV) \rightarrow 1\ n + 2\ ^{4}He$
- ^{2}H (deuterium) $+ \gamma\ (>2.26\ MeV) \rightarrow 1n + ^{1}H$

Advantage over (α,n) reaction:
emittion of monoenergetic n's, since γ's are not slowed down as in the case of the α's.

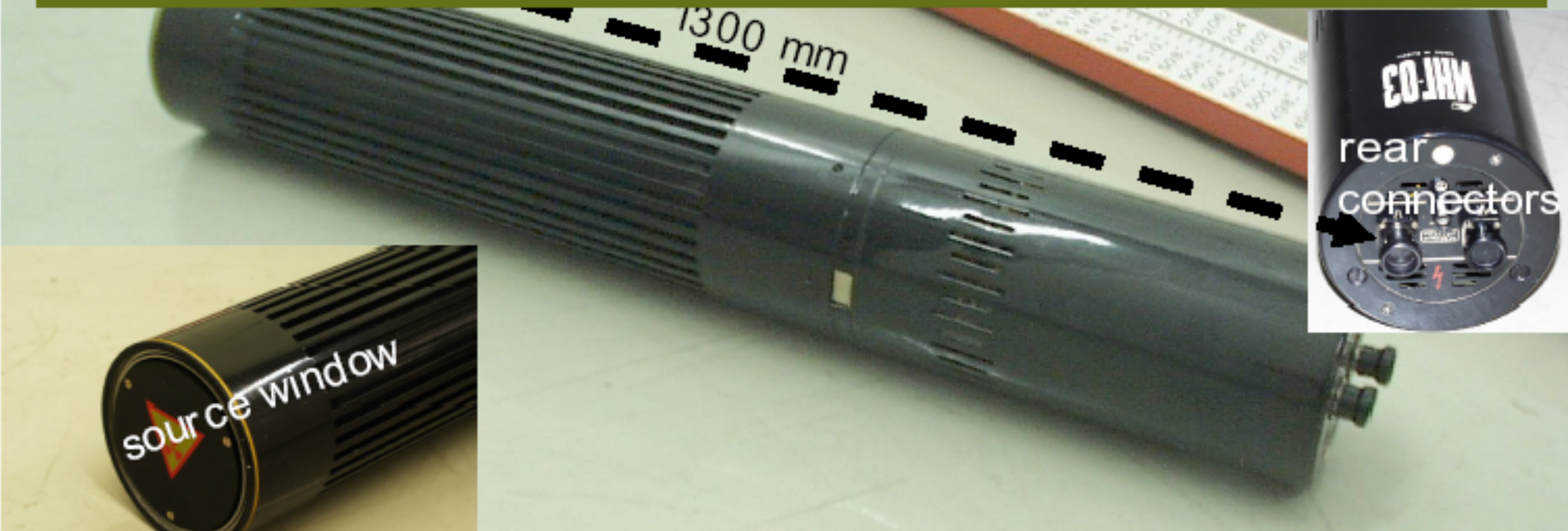
Sealed tube neutron generators

(d,n) reactions on D or T based targets

Some accelerator-based neutron generators exist that work by inducing fusion between beams of deuterium and/or tritium ions and metal hydride targets which also contain these isotopes.

Commercial Neutron Generator ING-03

Neutrons can be produced in a variety of reactions, e.g., in nuclear fission reactors or by the $D(d,n)^3He$ or $T(d,n)^4He$ reactions



Specs: $< 3\cdot10^{10}$ $D(d,n)^3He$ neutrons/s Total yield $2\cdot10^{16}$ neutrons

Pulse frequency 1-100Hz Pulse width >0.8 μs, Power 500 W

Alternative option:
$T(d,n)^4He$, $E_n \approx 15$ MeV

Source: All-Russian Research Institute of Automatics **VNIIA**

Medium size devices

Plasma focus and plasma pinch devices

The plasma focus neutron source (see Plasma focus, not to be confused with the so-called Farnsworth-Hirsch fusor) produces controlled nuclear fusion by creating a dense plasma within which ionized deuterium and/or tritium gas is heated to temperatures sufficient for creating fusion.

Light ion accelerators

Traditional particle accelerators with hydrogen (H), deuterium (D), or tritium (T) ion sources may be used to produce neutrons using targets of deuterium, tritium, lithium, beryllium, and other low-Z materials.

Typically these accelerators operate with voltages in the > 1 MeV range.

High energy bremsstrahlung photoneutron/photofission systems

Neutrons (so called) are produced when photons above the nuclear binding energy of a substance are incident on that substance, causing it to undergo giant dipole resonance after which it either emits a neutron or undergoes fission. The number of neutrons released by each fission event is dependent on the substance. Typically photons begin to produce neutrons on interaction with normal matter at energies of about 7 to 40 MeV, which means that megavoltage photon radiotherapy facilities may produce neutrons as well, and require special shielding for them. In addition, electrons of energy over about 50 MeV may induce giant dipole resonance in nuclides by a mechanism which is the inverse of internal conversion, and thus produce neutrons by a mechanism similar to that of photoneutrons.

Large sized devices

Nuclear fission reactors

Nuclear fission which takes place within in a reactor produces very large quantities of neutrons and can be used for a variety of purposes including power generation and experiments.

Nuclear fusion systems

Nuclear fusion, the combining of the heavy isotopes of hydrogen, also has the potential to produces large quantities of neutrons. Small scale fusion systems exist for research purposes at many universities and laboratories around the world. A small number of large scale nuclear fusion systems also exist including the National Ignition Facility in the USA, JET in the UK, and soon the recently started ITER experiment in France.

High energy particle accelerators

A spallation source is a high-flux source in which protons that have been accelerated to high energies hit a target material, prompting the emission of neutrons.

Neutron flux

For most applications, a higher neutron flux is always better (since it reduces the time required to conduct the experiment, acquire the image, etc.). Amateur fusion devices, like the fusor, generate only about 300 000 neutrons per second. Commercial fusor devices can generate on the order of 10^9 neutrons per second, which corresponds to a usable flux of less than 10^5 n/(cm^2 s). Large neutron beamlines around the world achieve much greater flux. Reactor-based sources now produce 10^{15} n/(cm^2 s),AND spallation sources generate greater than 10^{17} n/(cm^2 s).

Nuclear Interactions of Neutrons

No electric charge → no direct atomic ionization → only collisions and reactions with nuclei → 10^{-6} x weaker absorption than charged particles

Processes depend on available n energy E_n:

$E_n \sim 1/40$ eV (= $k_B T$) Slow diffusion, capture by nuclei

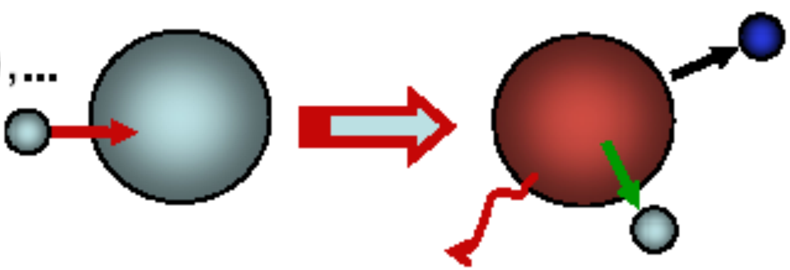
E_n < 10 MeV Elastic scattering, capture, nucl. excitation

E_n > 10 MeV Elastic+inel. scattering, various nuclear reactions, secondary charged reaction products

Characteristic secondary nuclear radiation/products:

*Always **heavy charged particles** (NOT electrons as in the case of γ interaction)*

1. γ-rays (n, γ)
2. charged particles (n,p), (n, α),...
3. neutrons (n,n'), (n,2n'),...
4. fission fragments (n,f)

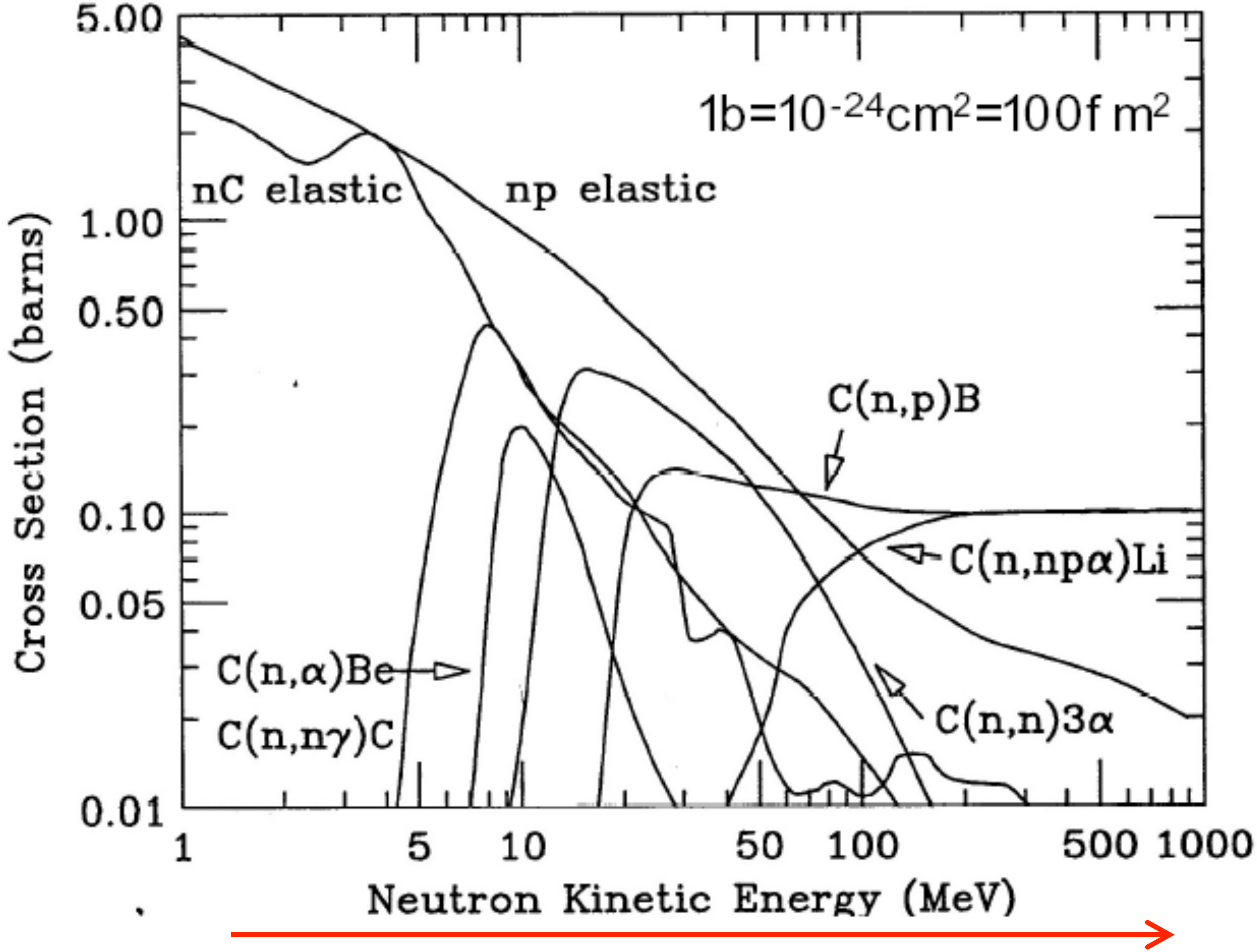


5. Spallation Reactions: high energy neutrons (> 150 MeV) may strike a nucleus producing a <u>shower</u> of secondary particles (very harmful …)

Therminology

1) High Energy neutrons:	$E_n > 100$ MeV
2) Fast Neutrons:	100 keV $< E_n <$ 10 MeV
3) Epithermal Neutrons:	0.1 eV $< E_n <$ 100 keV
4) Thermal/Slow Neutron:	$E_n = 1/40$ eV
5) Cold/Ultracold Neutrons:	$E_n <$ meV … μeV

Neutron Cross Sections



Cross Section versus Neutron Energy

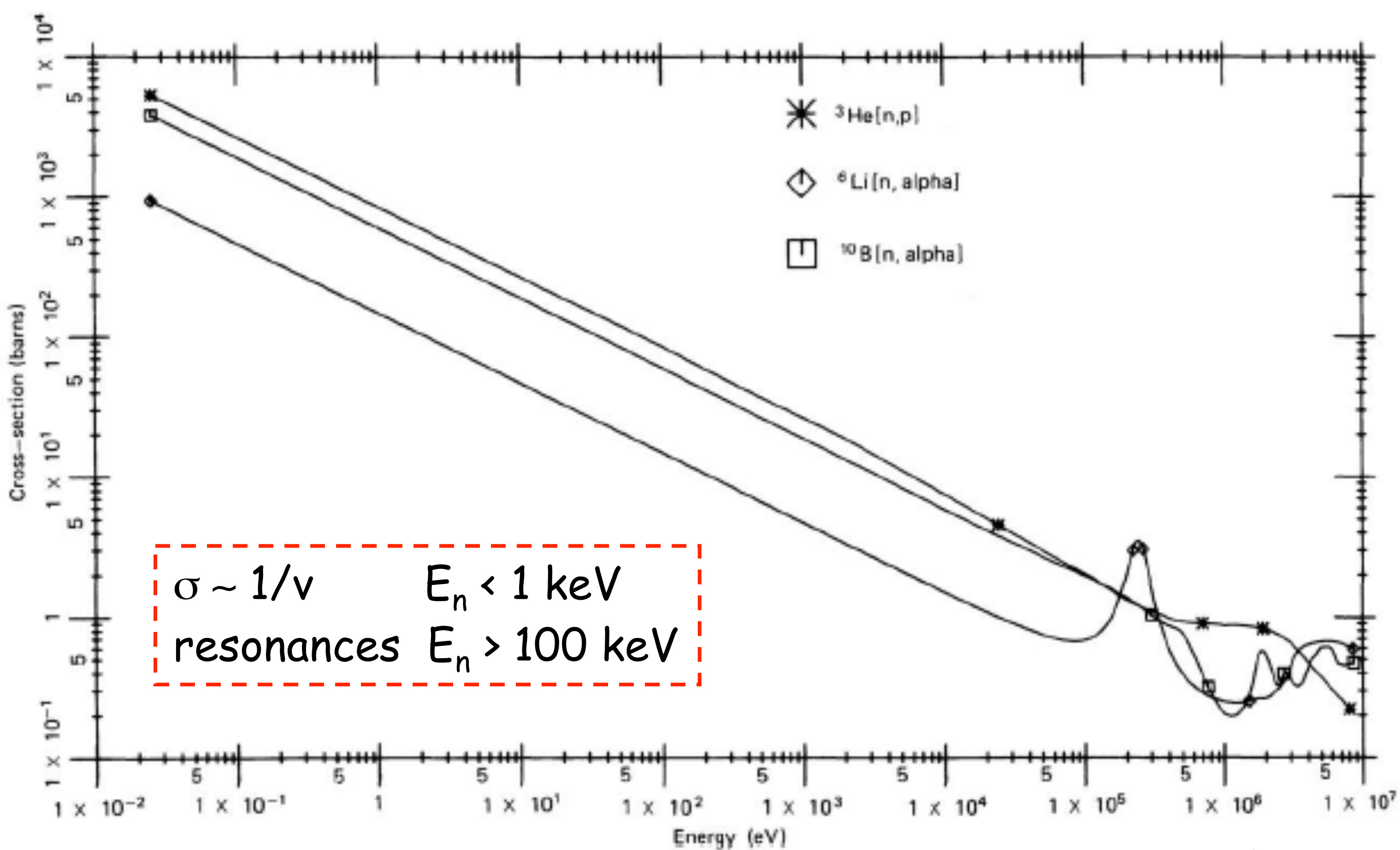


Figure 14.1 Cross section versus neutron energy for some reactions of interest in neutron detection.

Principle of neutron detection:

- **Conversion** of incident neutron into secondary charged particles
- **Direct detection** of charged particle

Relative probabilities of different interaction changes rapidly with E_n

Cross section are sizable only at very low energy (slow and thermal n)

slow neutrons (E_n < 0.5 eV):

- elastic scattering (NOT favorite for detection of scattered nucleus:
  $\Rightarrow$ little energy given to the nucleus to be detected)
- neutron-induced reactions creating secondary radation with sufficient energy
  for example radiative capture (n,γ) or (n,α), (n,p), (n, fission)

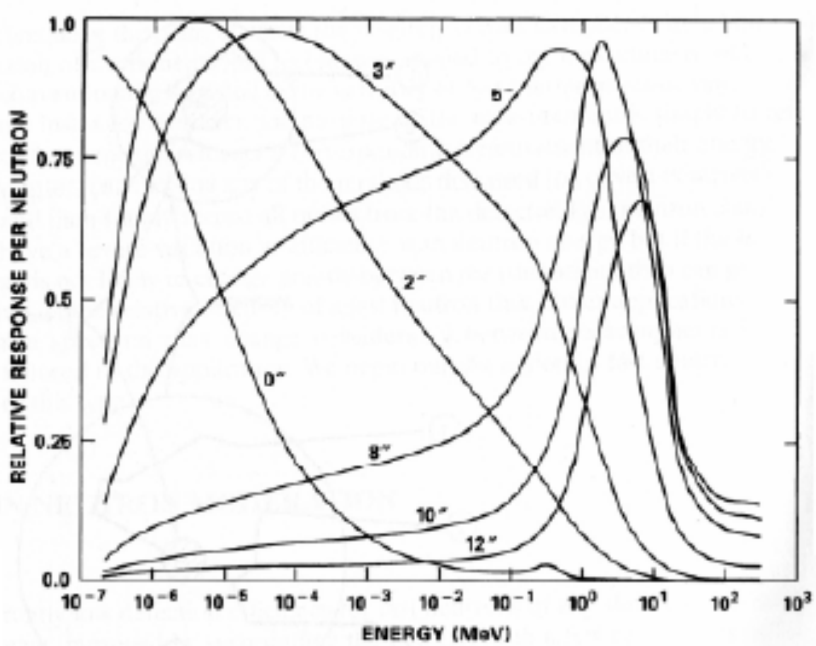
fast neutrons:

- scattering probability becomes greater: large energy transfer in one collision
  neutron loses energy and is moderated/slowed to lower energy

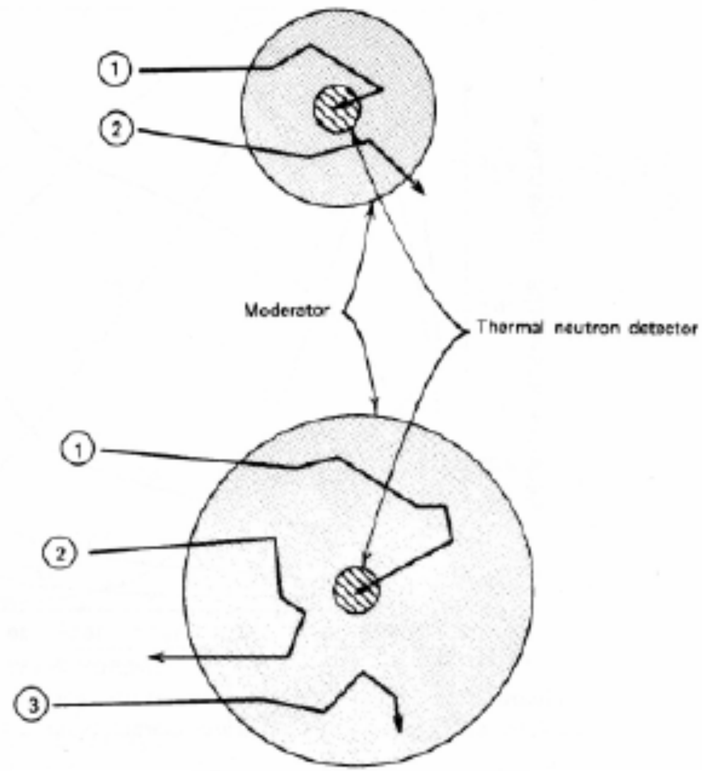
Best moderator is hydrogen
[it can get all n-energy in one collision !]

Counters Based on Neutron Moderation

- Moderate neutrons to increase efficiency in conventional slow-neutron detectors
- Moderation with hydrogenous materials such as polyethylene or paraffin
- Optimum thickness between few cm to tens of cm for energies of keV to MeV
- Trade-off between sufficient slow down and detection cross section



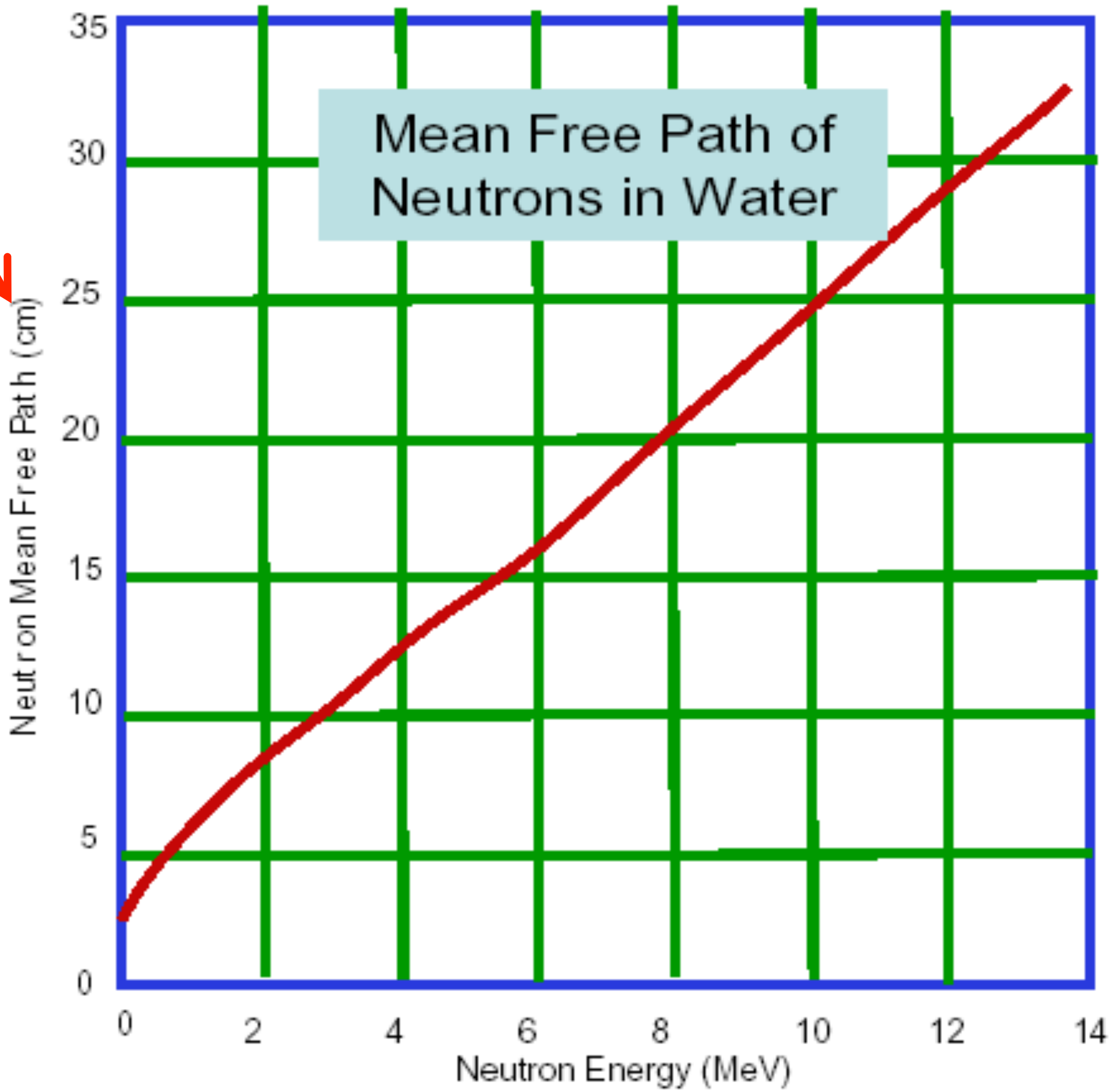
Energy dependence of relative efficiency for different diameter spheres



Bonner sphere

Average distance between successive collisions

Neutron Mean Free Path



$$N(x) = N(0)e^{-x/\lambda}$$

$$\lambda = \frac{1}{\mu} = \frac{1}{\rho\sigma} \quad mfp$$

ρ: atomic density (# atoms/volume)

σ: cross section

λ = average path length in medium between 2 collisions

$\sigma = \sigma_{TOT} =$
$\sigma_{ELASTIC} +$
$\sigma_{INELASTIC} +$
$\sigma_{CAPTURE} + \ldots$

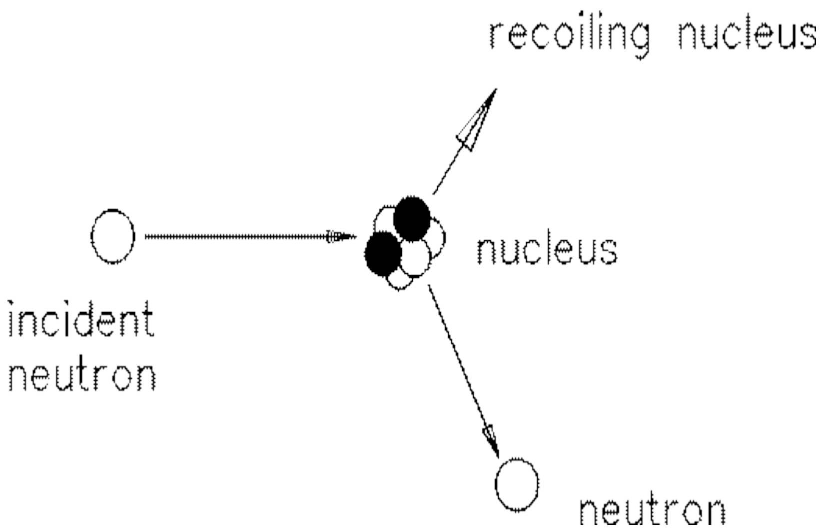
$$\Gamma_{FWHM} = 2.35 \cdot \sigma$$

Elastic Scattering (mostly for $E_n < 10$ MeV)

- billiard ball type of collision;
- The neutron collides with a nucleus and scatters in a different direction
- The energy the neutron loses is gained by the target nucleus which moves away at an increased speed.
- If target nucleus is massive, neutron scatters with same speed little energy loss

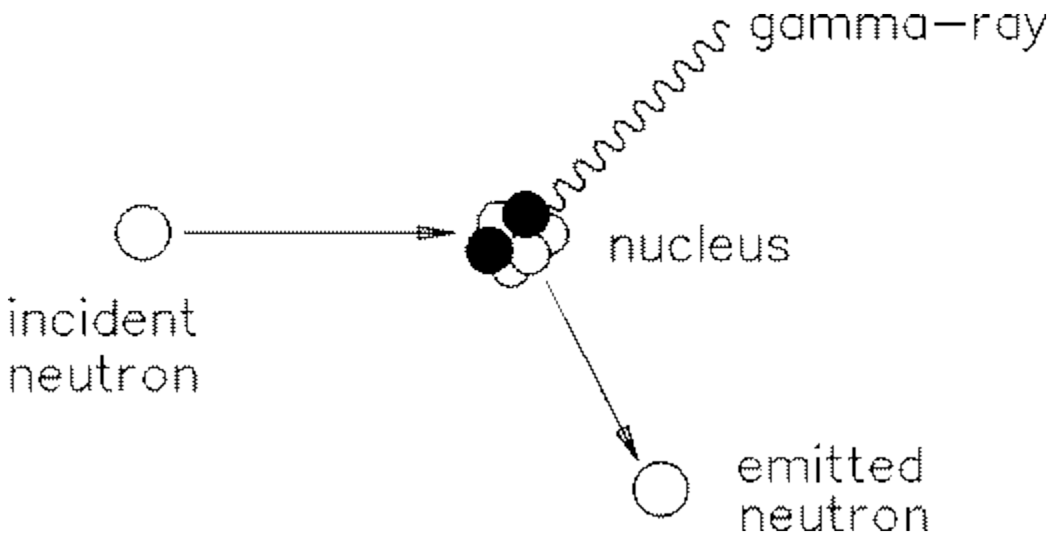
$M_T \gg M_n, \Rightarrow v'_n = v_n$ and $E_{loss} \sim 0$

- If target nucleus is light, neutron loses much energy $\Rightarrow$ very effective slowing down process
- Elastic scattering is not effective in slowing down neutrons with very high energy (above 150 MeV)



Inelastic Scattering (mostly for $E_n \geq 10$ MeV, heavy material)

- The neutron strikes a nucleus and form a compound nucleus
- The nucleus is unstable: emission of n and γ



Nuclear Reaction
Radiative Capture [slow neutrons]

This is one of the most common neutron reactions . The neutron is again captured by a nucleus which emits only a gamma photon . This reaction, which occurs in most materials, is the most important one for neutrons with very low energy. The product nuclei of (n,γ) reactions are usually radioactive and are beta and gamma emitters .

Compound nucleus (CN) formation
with exc. energy 6-10 MeV (close to binding energy)
Decay time of CN : 10^{-16} s
Prompt γ-emission in 10^{-9}-10^{-12} s until ground state is reached.
Delayed radiation if ground state is not stable (β, α, …)

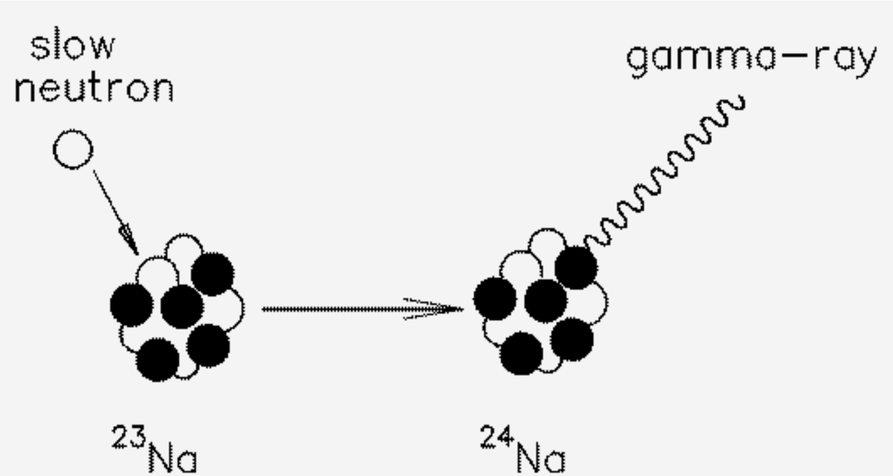
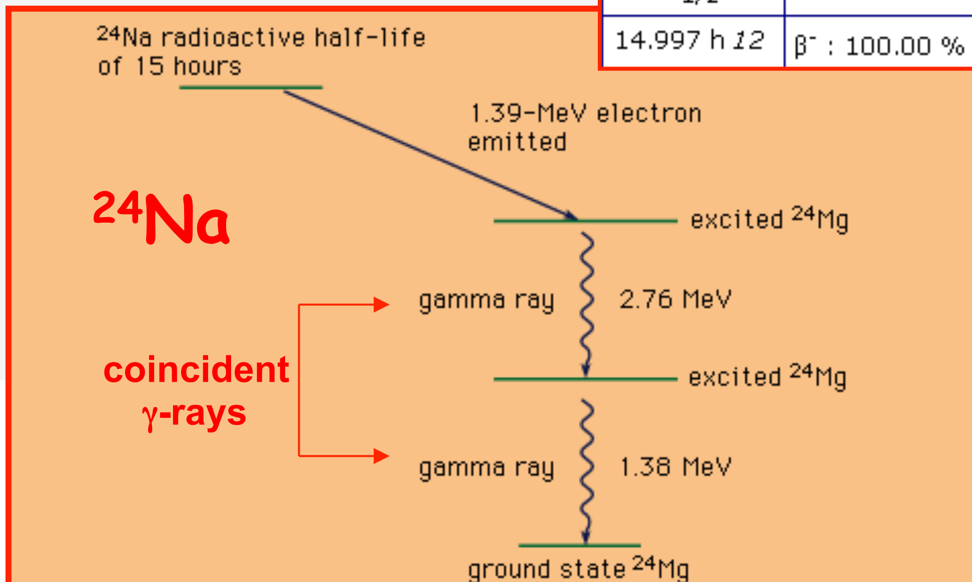


Figure 2.15: Neutron capture in $_{23}$Na

$T_{1/2}$	Decay Modes
14.997 h *12*	β^- : 100.00 %



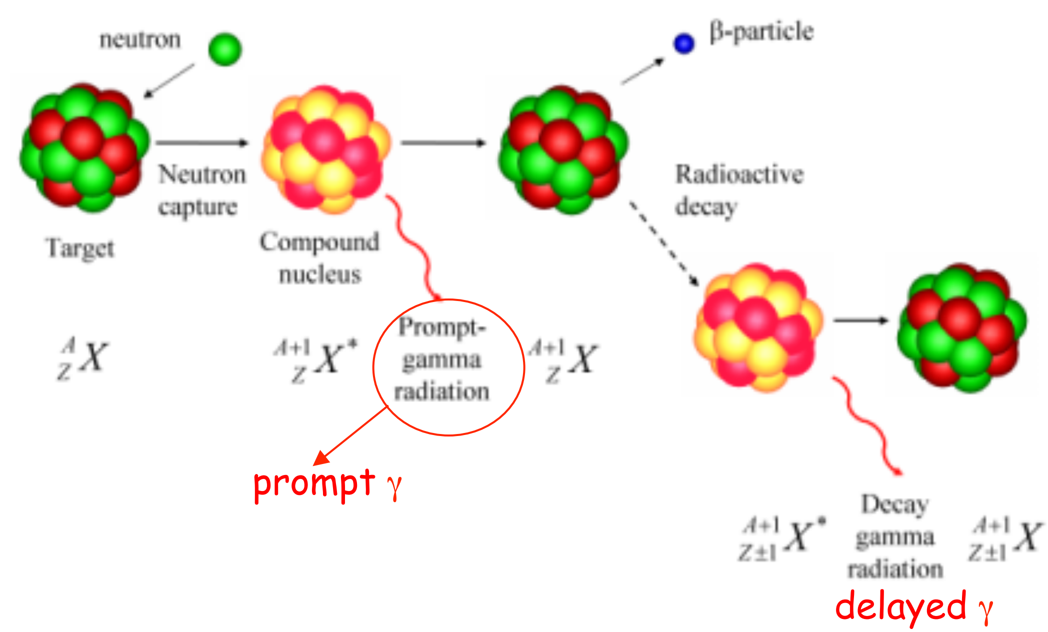
With a half-life of 15 hours, the sodium-24 nucleus decays by beta decay to an excited magnesium-24 (^{24}Mg) nucleus. Two gamma rays are rapidly emitted and the excitation energy is carried off, whereby the stable ground state of magnesium-24 is reached.

Uses of neutron capture reactions:

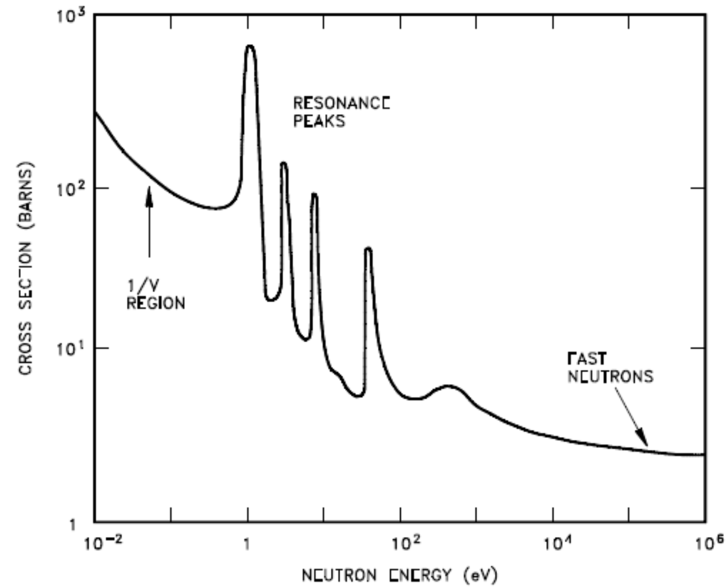
Neutron activation analysis can be used to remotely detect the **chemical composition** of materials:

- different elements release different characteristic radiation when they absorb neutrons.

- very useful in many fields related to **mineral exploration and security**



Typical neutron capture cross sections



1/v law of neutron capture + resonances

Neutrons have no charge:

- Neutron capture cross sections are proportional to the inverse neutron velocity ($E_n < 1$ eV)

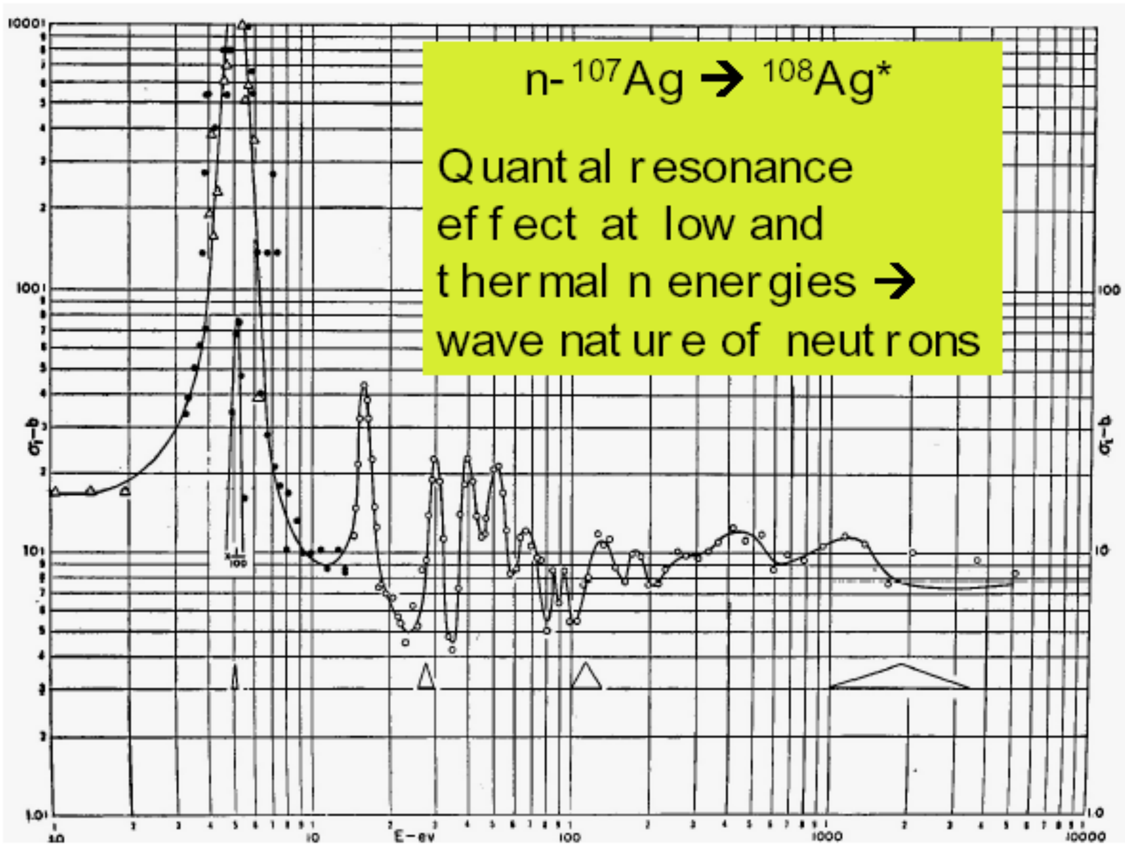
- Resonance region:
  "resonance peaks" for neutrons of certain energies
  neutron energy closely matchs the discrete, quantum energy level of the CN:

$$E^*_{CN} = B_n + E_{kin}$$

It happens when the ***binding energy*** of a neutron plus the ***kinetic energy*** of the neutron are exactly equal to the amount required to raise a compound nucleus from its ground state to a quantum level.

To increase reaction probability:
⇒ need for "**moderating**" neutrons to lower "**thermal**" velocities.
The best neutron moderators are light mass materials because there is large energy transfer in scattering:
Graphite: easy to make from carbon, **Heavy Water**

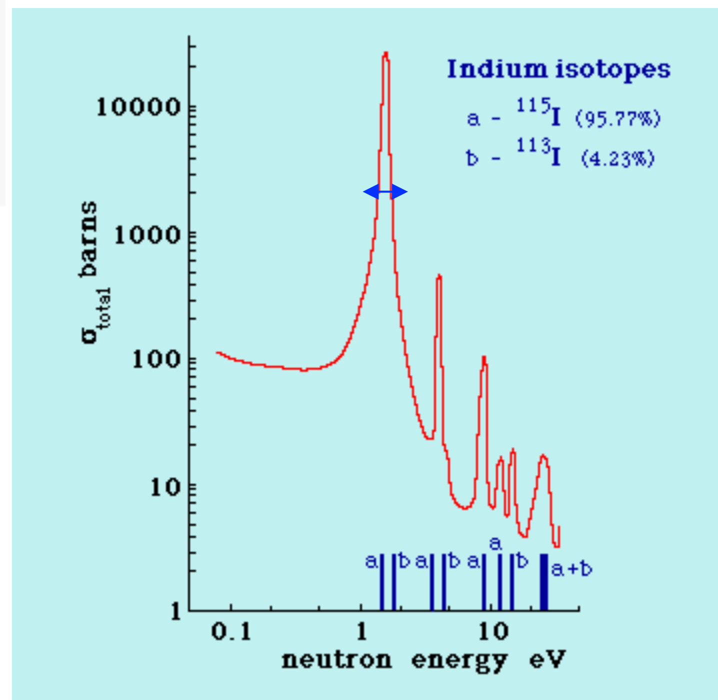
Neutron Resonance Capture



CN reactions

$$E^*_{CN} = B_n + E_{kin} \sim 8 \text{ MeV}$$

$$\tau_{CN} = \eta/\Gamma \sim 10^{-16} \text{ s}$$



- Resonances are very closely spaced due to the high level density of CN at 8 MeV.
- Slowly varying E_{kin} (~ few eV) one excites CN states one by one
- resonance spacings give a direct measurement of level density

See neutron resonance physics papers

J.B.Garg et al., PRC25(1982)1808, M.S.Pandey et al., PRC15(1977)600, F.G.P.Seidl et al., PR95(1954)476

- **n-TOF-techniques** and **Transmission measurements** (ORELA Facility Oak Ridge:Electron Accelerator producing n by bremsstrahlung on Ta target)
- **Crystal Spectrometers**

$\Rightarrow$ *both allow 0.1 eV resolution*

Isolated resonance

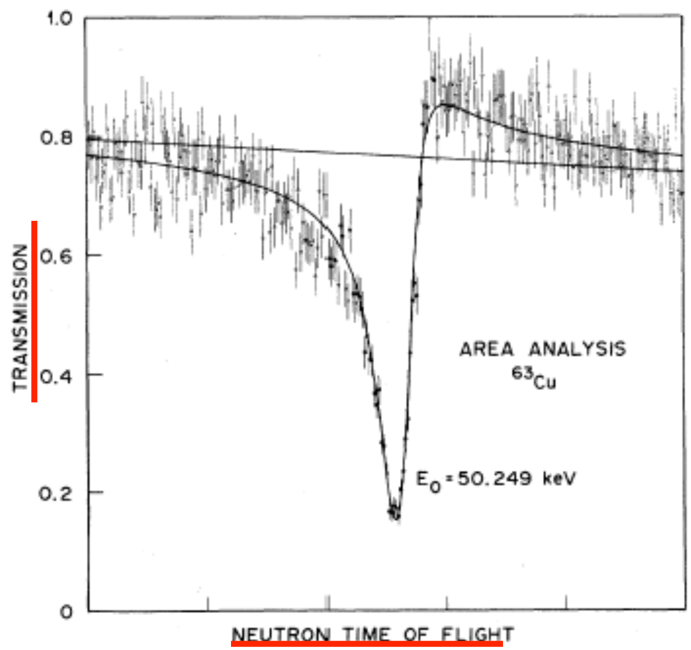
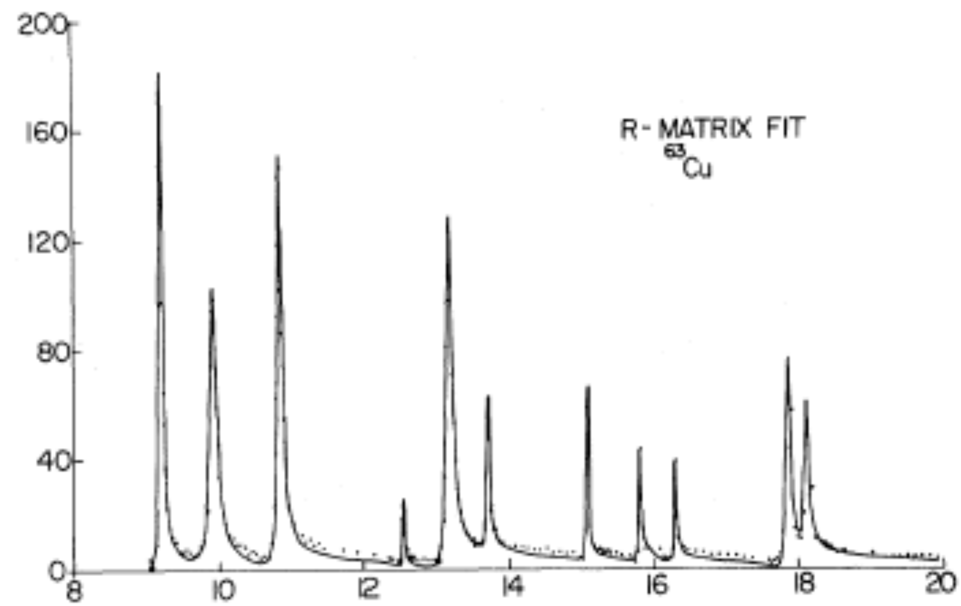


FIG. 1. An example of the area program fit to the transmission data of an isolated resonance in ^{63}Cu at $E_0 = 50.249$ keV.



The transmission of neutrons as a function of energy is computed by the simple relation $T = (N_s - BG_s)/(N_o - BG_o)$, where N_s and N_o are the neutron counting rate with the sample in position and the open beam, respectively, and the BG's are the corresponding backgrounds. The true ex-

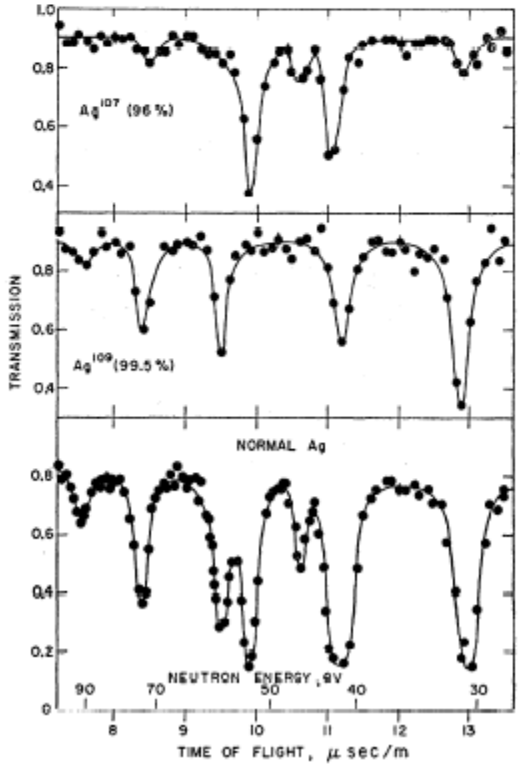
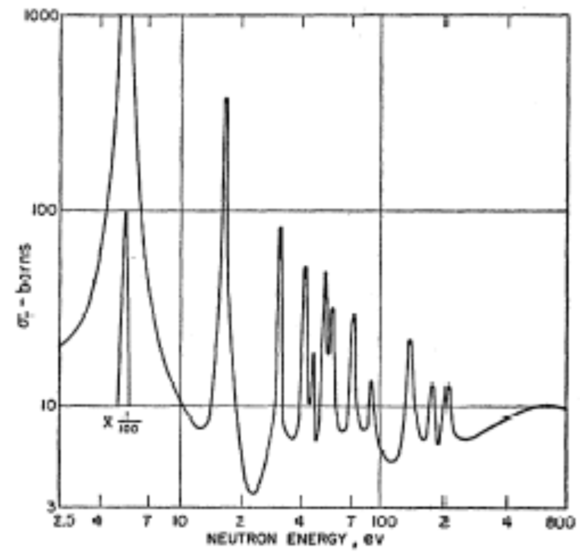


FIG. 22. The transmission as a function of neutron time of flight for samples of isotopically enriched silver and of normal silver. The Ag^{107} sample contained 1.20 g/cm² of Ag^{107}, the Ag^{109} sample 0.82 g/cm² of Ag^{109}, and the sample of normal silver 3.4 g/cm² of Ag^{107} and 3.2 g/cm² of Ag^{109}. The calculated contribution of the 4 percent Ag^{109} to the transmission of the Ag^{107} sample agreed with the dips observed at 31, 57, and 72 ev. The 42-ev "level" appeared to be too wide in the normal Ag sample, but in the enriched samples it was resolved into two levels (one in each isotope) at somewhat different energies.

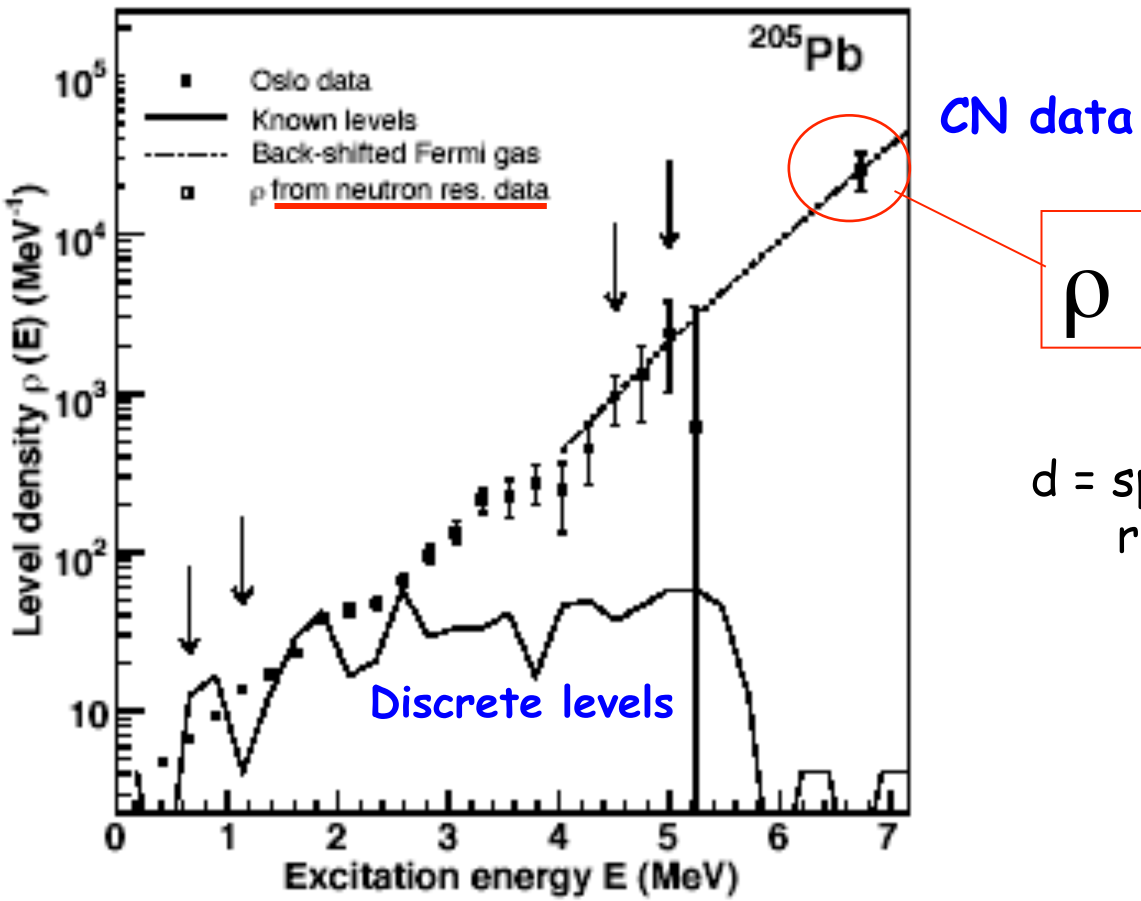
The total cross section σ_T, which is the sum of the absorption and scattering cross sections, determines the change in transmission observed as a resonance dip, the transmission T being given at each energy by the relationship

$$T = I/I_0 = \exp(-n\sigma_T), \tag{4}$$

where I and I_0 are the counting rates observed with and without the sample, containing n atoms per cm². If the



From Resonance spacing CN level density is deduced



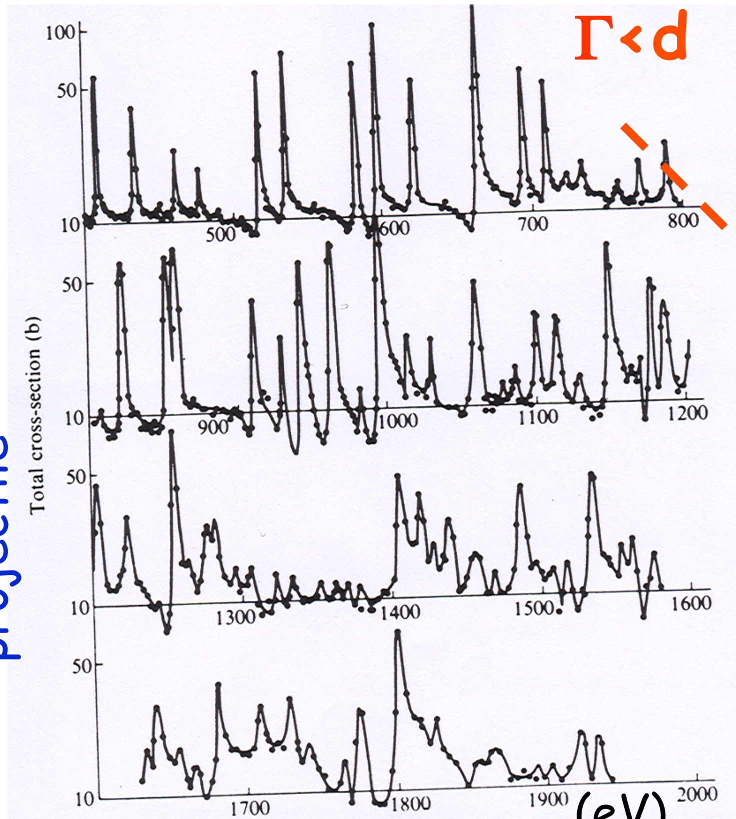
$\rho \sim 1/d$

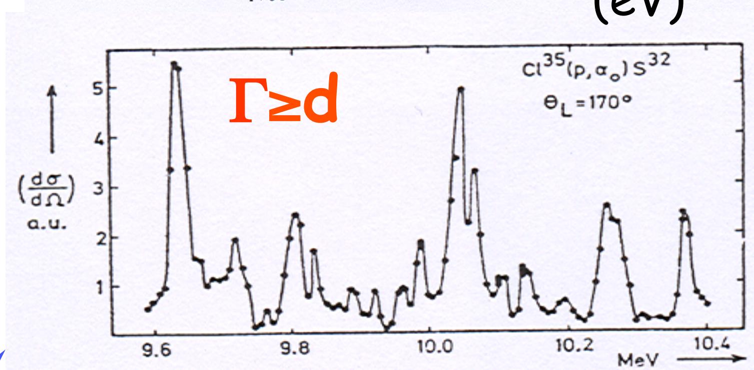
d = spacing between resonances

From n resonances: Statistical Properties of CN states

n resonances in ^{238}U

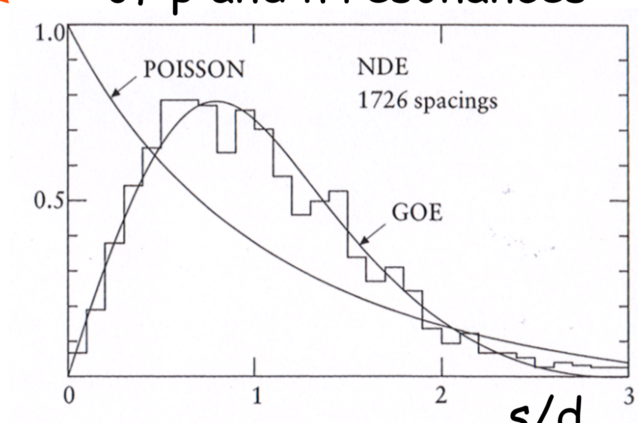
$E_{projectile}$





NON overlapping resonances:
isolated resonances
behaving stocastically
(chaotic system)

Level spacing distribution
of p and n resonances



$J^{\pi} = 1/2^{+}$
$U \approx$ 8-10MeV
$d \approx$ 10 eV
$\Gamma \approx$ 1 eV

strongly overlapping resonances:
Ericson fluctuations
(chaotic system)

The CN is CHAOTIC

Nuclear Reaction
Trasmutation [1 MeV to > 10 MeV]

When neutrons , protons , or other secondary particles produced by spallation strike a nucleus and form a compound nucleus which then ejects a different particle, a transmutation is said to have occurred. This is because the target nucleus is changed from one element to another. These nuclear reactions are most likely to occur when the energy of the incident particle is between a few MeV and several tens of MeV. An example is the neutron-proton (n,p) reaction with oxygen described below.

$_{16}$O captures a high energy neutron and emits a proton to form $_{16}$N. The product, $_{16}$N is radioactive with a half-life of 7.2 seconds; it is a beta emitter and also emits very hard gamma-rays which have energies of 6 or 7 MeV.

$$^{16}_{8}O + ^{1}_{0}n \Rightarrow ^{16}_{7}N + ^{1}_{1}p$$

This may be written in the shorter form shown below:

$$^{16}_{8}O(n,p)^{16}_{7}N$$

In addition to (n,p) reactions, many other reactions are possible such as (p,n), (p,2n), (n,2n), (π,n), and (n, α). Most of these reactions result in nuclei which are deficient in neutrons and therefore decay by positron emission or electron capture .

The (n,α) reaction in boron is an important reaction which is used for the detection of slow neutrons . Slow neutrons are captured by $_{10}$B causing the following reaction:

$$^{10}_{5}B + ^{1}_{0}n \Rightarrow ^{7}_{3}Li + \alpha$$

or:

$$^{10}_{5}B(n,\alpha)^{7}_{3}Li$$

Exotermic reaction: it occurs also for very low n energy

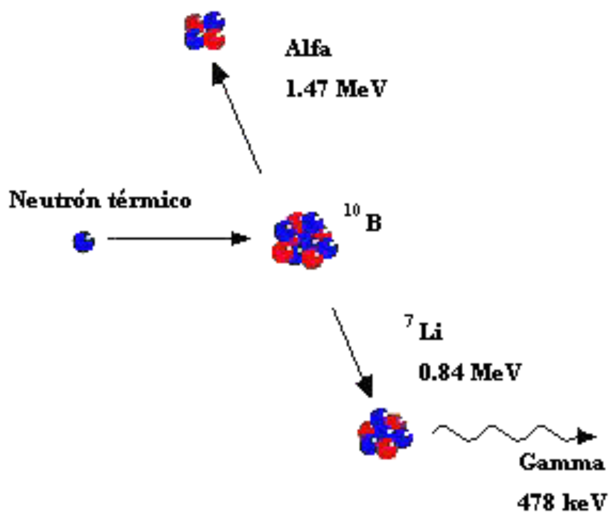
For E_n = 0.025 eV, σ = 3770 b !!
$\Rightarrow$ ideal for n detection
$\Rightarrow$ useful for measuring the flux of thermal n

Application:Boron-Neutron Capture Therapy (BNCT)

Boron Neutron Capture Therapy (BNCT) is a binary technique that involves the concurrent presence of a flux of neutrons of adequate energy and a ^{10}B capture compound that accumulates mainly in tumor cells. Their interaction generates heavy particles that damage tumor cells. The neutron and boron components do not produce significant damage to tissues when they are present separately.

After neutron capture, the following reaction occurs:

$$^{10}B + n \rightarrow {}^{7}Li + {}^{4}He + 2.79\ MeV$$



94% of the time, a 478 keV photon is emitted in the nuclear decay of ^{7}Li. The range in tissue of ^{7}Li and ^{4}He (alpha particle) is approximately 5 μm and 8 μm respectively, i.e. about the diameter of a tumor cell (~ 10 μm).

A few alpha particles suffice to destroy a tumor cell. The killing effect of the capture reaction would occur mainly in those cancer cells that have selectively accumulated boron. The normal cells that have not incorporated important amounts of boron will not suffer significant damage.

Clinical applications: head and neck tumor (in Catania, eye tumor…)
liver tumor, …

Danger of n to human body

Interaction of Neutrons with Tissue

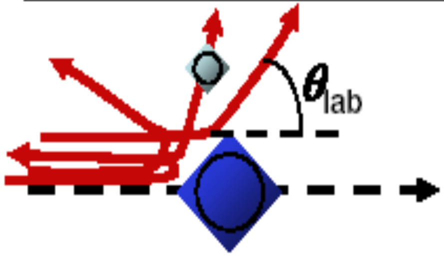
The human body is composed largely of water, about 60% by weight, which contains many hydrogen nuclei. Elastic scattering of the neutrons with the hydrogen nuclei will cause the protons to recoil violently. Similarly elastic collisions of neutrons with carbon, oxygen or other heavier nuclei will cause these to recoil. Because the mass of protons and the other recoiling nuclei is much greater than that of electrons, they generate a much denser ion path resulting in more damage to the tissue. Once neutrons have been slowed down by elastic collisions to thermal energy, 0.025 **eV**, they are readily captured by some of the reactions described above.

A very common reaction is the (n,γ) reaction, particularly with hydrogen. The gamma photon produced in this reaction always has an energy of 2.2 MeV and will cause indirect ionisation as described previously. When neutrons are absorbed by an (n, γ) reaction in the body the tissues will be further damaged by gamma radiation in addition to damage which they receive in slowing down the neutrons.

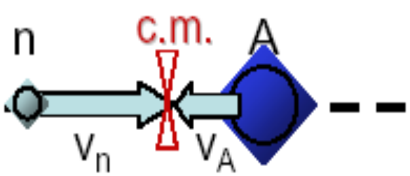
Other radionuclides may be formed in the body by the interaction of slow neutrons with stable nuclei. However, the dose contributed by these radionuclides is usually insignificant compared to the dose the neutrons themselves contribute.

1. elastic scattering with hydrogen: violent recoil of p
2. (n,γ) reaction with hydrogen: ionization of 2.2 MeV γ-ray
3. radionuclides production (minor effect)

Energy Transfer in Elastic Scattering



Neutron with lab velocity v, energy E, scatters randomly off target nucleus of mass number A at rest in lab.

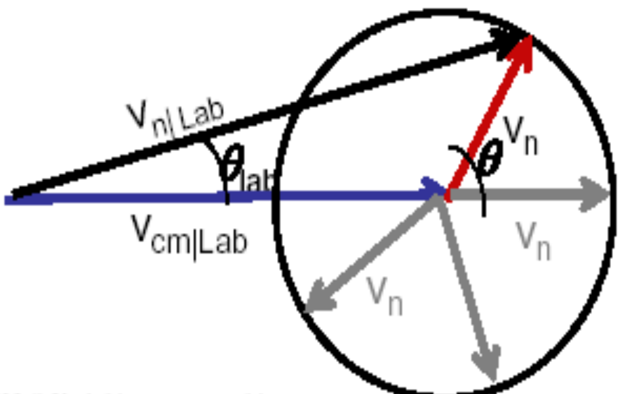


$$c.m.: \; p_n = -p_A \quad p_n + p_A = 0$$

$$v_n = v\frac{A}{A+1} \qquad v_A = -v\frac{1}{A+1}$$

Particle's velocity in CM before the collision

$$v_{cm|Lab} = v\frac{1}{A+1}$$ lab velocity of center of gravity $= m_1v/(m_1+m_2)$



$$\max, \min : v_{n|Lab} = v_{cm|Lab} \pm v_n$$

$$\max : E_{n|Lab} = E \qquad \boxed{\theta = 0}$$

$$\max : v_{n|Lab} = v$$

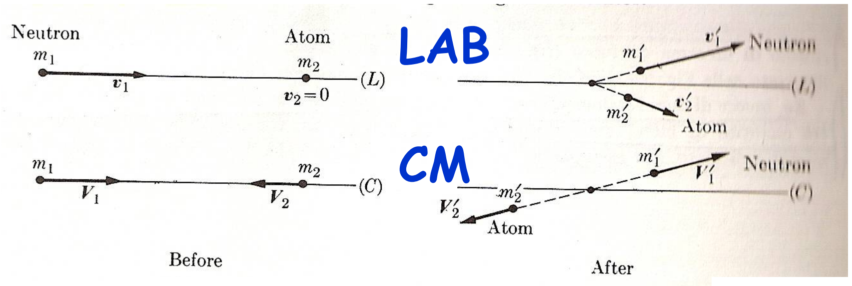
$$\min : v_{n|Lab} = v\frac{A-1}{A+1}$$

$$\min : E_{n|Lab} = E\frac{(A-1)^2}{(A+1)^2} \qquad \boxed{\theta = \pi}$$

W. Udo Schröder, 2003

$$\boxed{\frac{E'}{E} = \frac{A^2 + 1 + 2A\cos\theta}{(A+1)^2}}$$

Elastic collision neutron-atom at rest



Before collision

$$\vec{v}_{CM\backslash lab} = \frac{m_1\vec{v}_1 + m_2\vec{v}_2}{m_1 + m_2} = \frac{\vec{v}_1}{1+A}$$

v of CM in Lab Frame

$$\vec{V}_1 = \vec{v}_1 - \vec{v}_{CM|lab} = \frac{A\vec{v}_1}{1+A}$$

v neutron in CM

$$\vec{V}_2 = 0 - \vec{v}_{CM|lab} = -\frac{\vec{v}_1}{1+A}$$

v nucleus in CM

Particles retain
their speed in CM
($p_{CM} = 0$)

$$V_1 = V_1'$$

$$V_2 = V_2'$$

Magnitude velocity in CM
is the same
before and after the collision

After collision

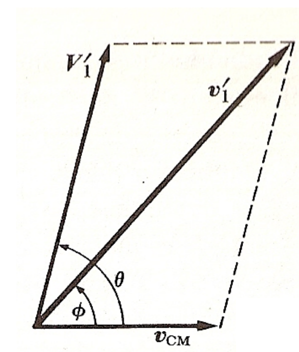
Direction of motion
may be different in CM

$$\vec{v}_1' = \vec{V}_1' + \vec{v}_{CM\backslash lab}$$

$$(v_1')^2 = V_1^2 + v^2_{CM\backslash lab} + 2\vec{V}_1' \cdot \vec{v}_{CM\backslash lab}$$

$$= V_1^2 + v^2_{CM\backslash lab} + 2V_1' v_{CM\backslash lab} \cos\theta$$

Cosine rule of triangle



$$(v_1')^2 = v_1^2 \frac{A^2 + 2A\cos\theta + 1}{(A+1)^2}$$

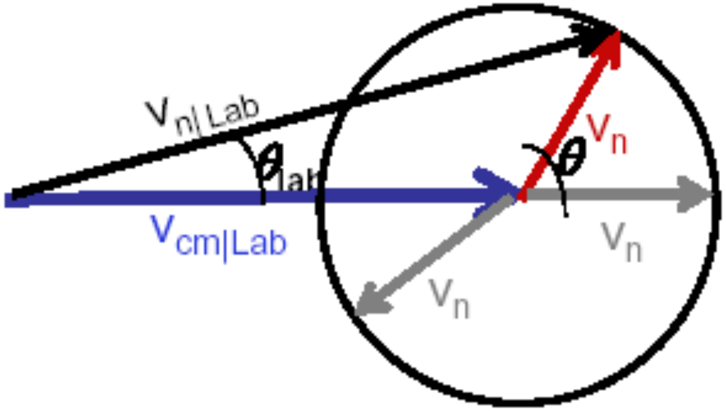
$$\frac{E_1'}{E_1} = \frac{(v_1')^2}{v_1^2} = \frac{A^2 + 2A\cos\theta + 1}{(A+1)^2} \Rightarrow$$

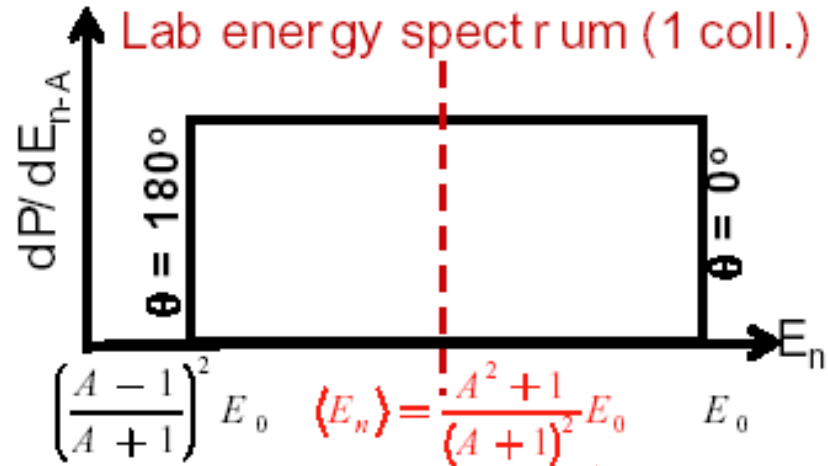
$E_1' = E_1$ $\quad \theta = 0$ No change in direction

$E_1' = \left(\frac{A-1}{A+1}\right)^2 E_1$ $\quad \theta = \pi$ Head-on

Neutron Energy Spectrum

Neutron with energy E_0 scatters off target nucleus A at rest in lab.





$$\boxed{\frac{E'}{E} = \frac{A^2+1+2A\cos\theta}{(A+1)^2}}$$

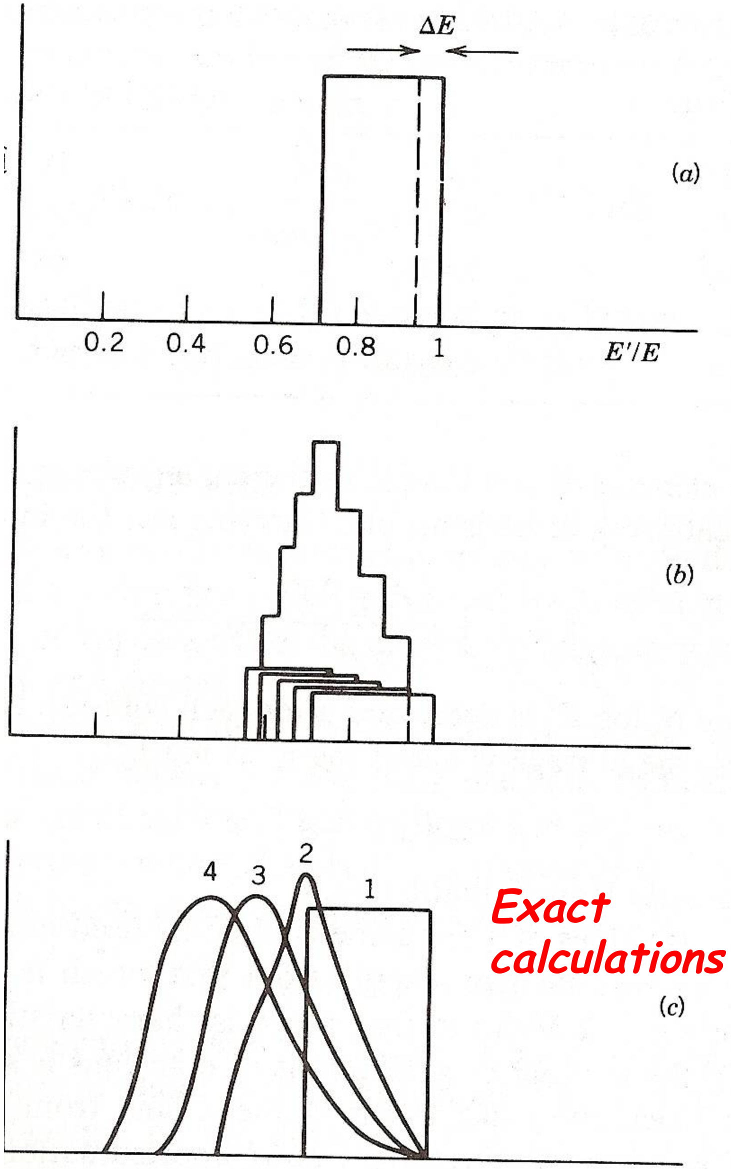
Average
energy distribution
$\langle E_n \rangle = (E_{min}+E_{max})/2$

After 1 collision the neutron energy is
<u>**uniform**</u> between $E_{min} = (A-1)^2/(A+1)^2 \times E_0$ and $E_{max} = E_0$

BUT: each neutron scatters <u>**many times**</u>

$\Rightarrow$ we need to calculate ***repetedly*** the energy loss
from a <u>**NON monoenergetic**</u> source …

$n + {}^{12}C$

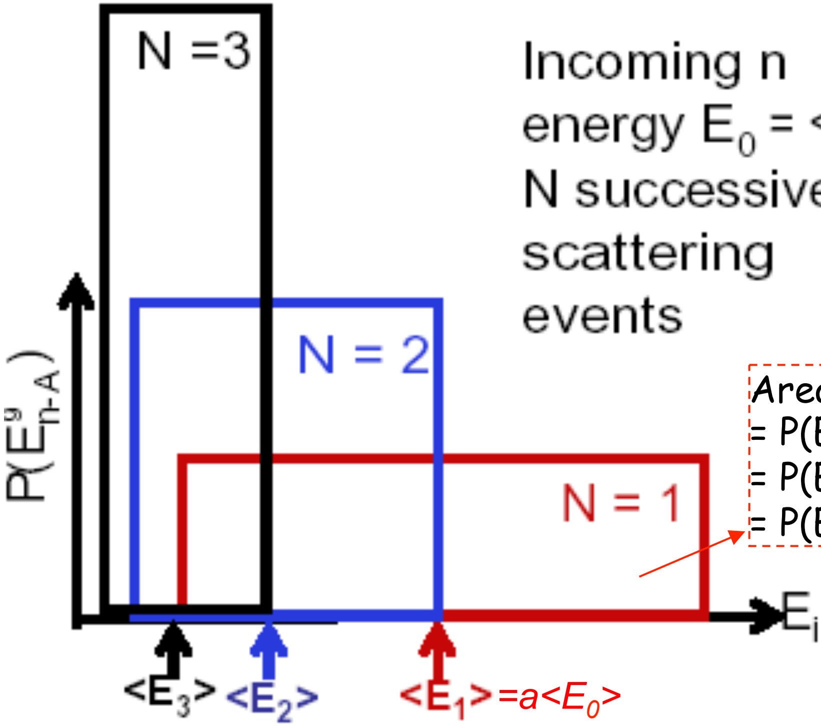


1. consider an interval ΔE to be a new generation of <u>monoenergetic</u> neutrons

2. resulting SECOND generation of n-energy distribution

3. Continuing this process one obtains the succeeding generations of energy distributions.

More quantitative calculations for
Multiple n- A Scattering



Incoming n energy $E_0 = \langle E_0\rangle$, N successive scattering events

Area = 1
= $P(E_1) \times (E_{max} - \langle E_{min}\rangle)$
= $P(E_1) \times 2A/(A+1)^2 E_0$
= $P(E_1) \times (1-a)E_0$

$$\Rightarrow \boxed{P(E_1) = \frac{1}{(1-a)E_0}}$$

$$\langle E_1\rangle = \left[\frac{A^2+1}{(A+1)^2}\right]^1 E_0 = a^1\langle E_0\rangle$$

$$\langle E_2\rangle = a\langle E_1\rangle = a^2\langle E_0\rangle$$

....

$$\langle E_N\rangle = a\langle E_{N-1}\rangle = a^N\langle E_0\rangle$$

average n energy <u>decreases</u>

"Logarithmic Decrement" ξ

$$\xi :=_{def} \langle \ln(E_0/E_1)\rangle = \int_{aE_0}^{E_0} dE \ln\left(\frac{E_0}{E}\right) P(E) = \frac{1}{(1-a)}\int_1^a dE \ln E =$$

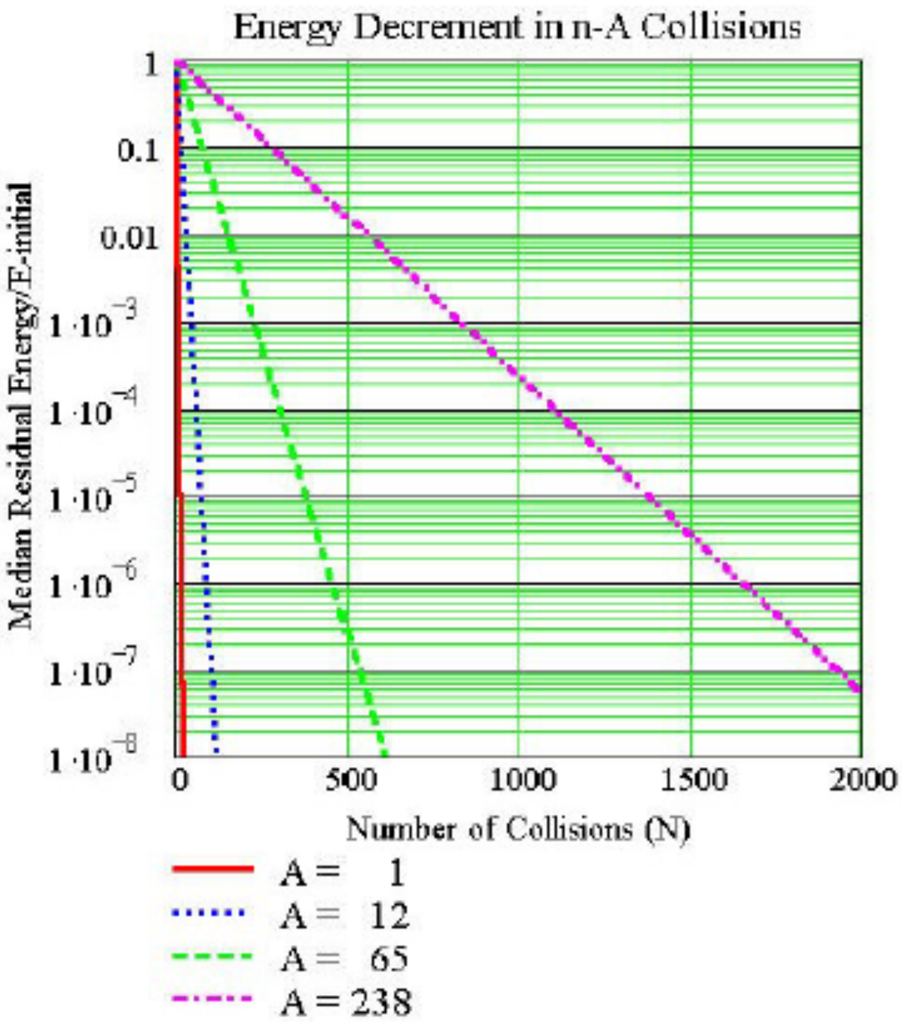
$$= \frac{[x\ln x - x]_1^a}{(1-a)} = 1 + \frac{a\ln a}{(1-a)} = 1 + \frac{A^2+1}{2A}\ln\left(\frac{A^2+1}{(A+1)^2}\right) \xrightarrow{A>10} \frac{2}{A+2/3} \neq f(E_0)$$

<u>Constant value</u> !!!

$$\langle \ln E_N\rangle = \langle \ln E_0\rangle - N\xi$$

after each collision logE decreases by a constant value ξ

Thermalization Through Scattering



$$\langle \ln E_N \rangle = \langle \ln E_0 \rangle - nx$$

$$Define\ \tilde{E}\ as\ median\ (<mean)$$

$$\ln \tilde{E} := \langle \ln E_N \rangle = \langle \ln E_0 \rangle - Nx$$

$$\boxed{\tilde{E}(N) = E_0 \cdot e^{-Nx}}$$

1/40eV

N-therm: E_0=2 MeV → 0.025eV

thermal neutron

A	ξ	N- therm
1	1.0000	18
12	0.1578	115
65	0.0305	597
238	0.0084	2172

Protons are the best moderators !

⇒ purified water and heavy water (enriched with hydrogen and deuterium)

Graphite: easy to make from carbon !

Elastic Scattering

Basic previous assumption:

Neutrons scatter on **Atoms at rest**
→ ***Good approx for MeV neutrons***
→ ***For thermal n***: thermal motion of the atoms comparable with v_n

Scatterig is best analyzed by statistical mechanics:
n are in thermal equilibrium with moderator

$$f(v)\,dv = 4\pi n\left(\frac{m}{2\pi kT}\right)^{3/2} v^2\, e^{-mv^2/2kT}\, dv \qquad (12.9)$$

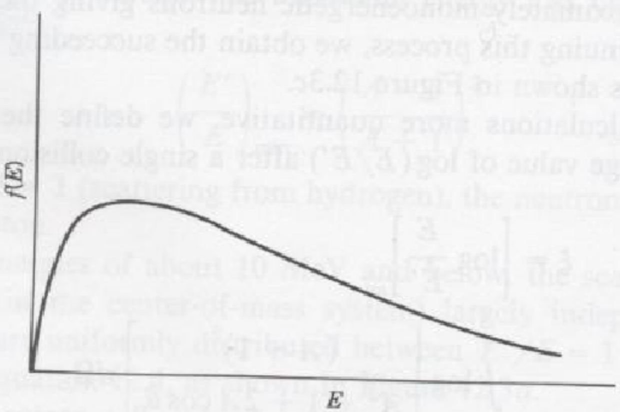
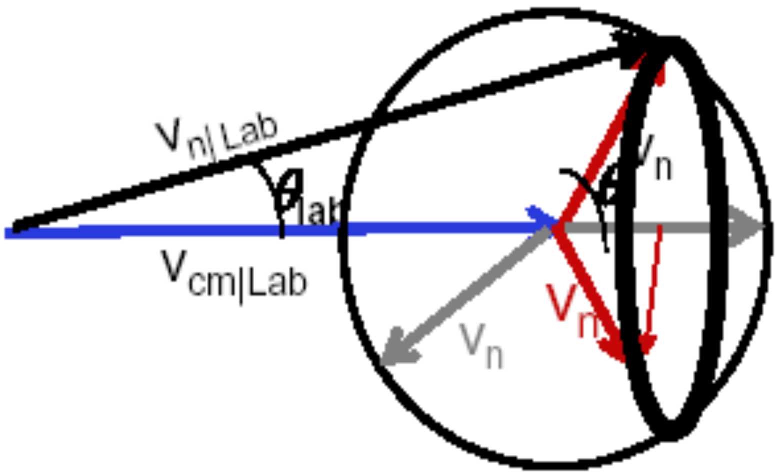


Figure 12.4 Maxwellian energy distribution, a representation of the neutron energy spectrum after many scatterings.

n Angular Distribution



before collision

$$v_{n|CM} = \frac{v}{A+1}A \qquad v_{cm|Lab} = \frac{v}{A+1}$$

after collision

$$\vec{v}^{\,2}_{n|Lab} = (\vec{v}\,'_{n|CM} + \vec{v}_{cm|Lab})^2 =$$

$v_{n,CM}=v'_{n,CM}$
$= V_1=V_1'$ in
previous slides

$$= \frac{v^2}{(A+1)^2}\left[A^2 + 1 + 2A\cos\theta\right]$$

$$v'^{\,2}_{n|CM} = v^2_{n|Lab} + v^2_{cm|Lab} - 2v_{n|Lab}v_{cm|Lab}\cos\theta_{Lab} \qquad \Big| : v^2\big/(A+1)^2$$

$$A^2 = \left[A^2 + 1 + 2A\cos\theta\right] + 1 - 2\sqrt{A^2 + 1 + 2A\cos\theta}\,\cos\theta_{Lab}$$

$$\boxed{\cos\theta_{Lab} = \frac{1 + A\cos\theta}{\sqrt{A^2 + 1 + 2A\cos\theta}}}$$

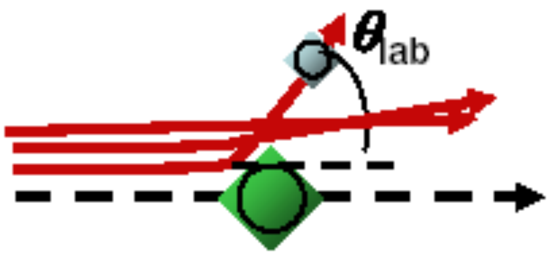
$$\langle\cos\theta_{Lab}\rangle = \frac{1}{2}\int_{-1}^{+1} d\cos\theta\,\frac{1 + A\cos\theta}{\sqrt{A^2 + 1 + 2A\cos\theta}} = \frac{2}{3A}\ \left\{\ >0 \rightarrow \text{forward scattering}\right.$$

A Dependence of Angular Distribution

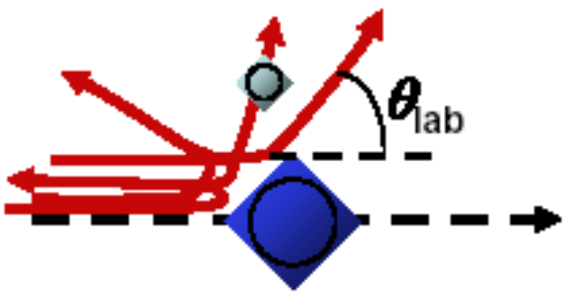
Properties of n scattering depends on the sample mass number A

➔ Measure time-correlated flux of transmitted or reflected neutrons

Light Nucleus



Heavy Nucleus



$$\langle \cos\theta_{lab} \rangle = 2/(3A) \propto A^{-1} \quad average$$

$$\langle \ln E_N \rangle = \langle \ln E_0 \rangle - N\xi$$

$\xi = 2/(A+2/3)$

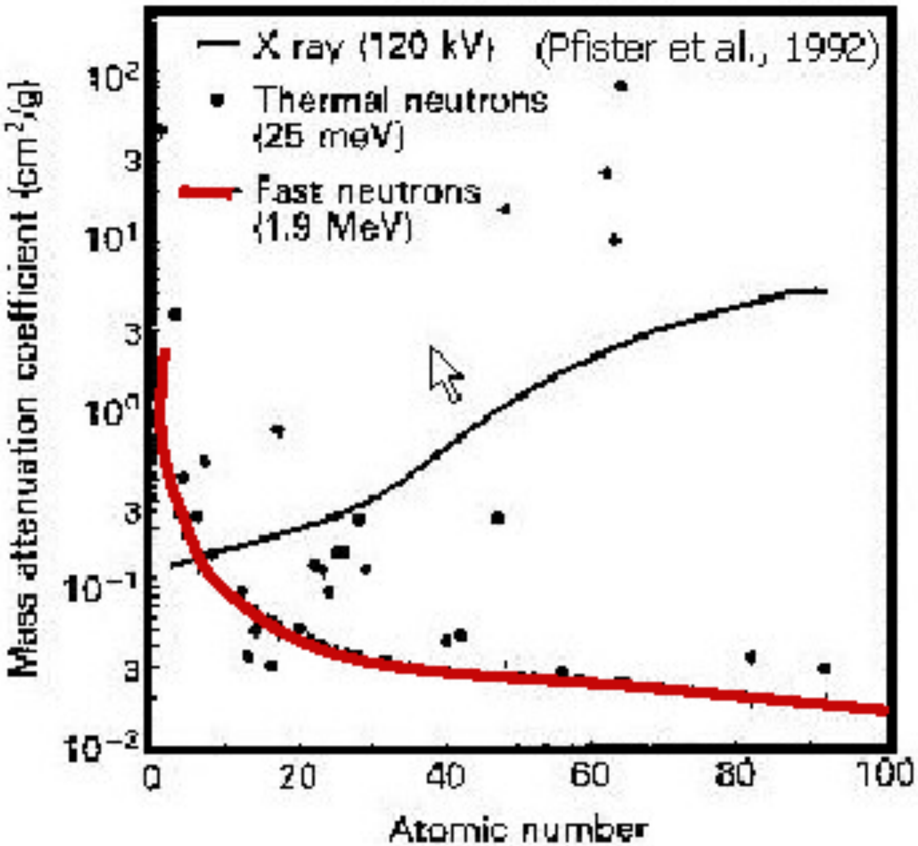
$$\tilde{E}(N) \approx E_0 \cdot \exp\left\{\frac{-2N}{(A+2/3)}\right\}$$

$$(After \ N \ collisions)$$

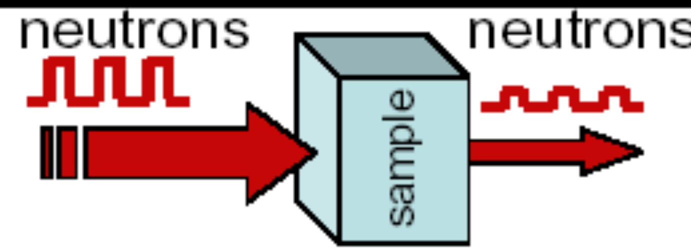
Light nuclei: slowing-down and diffusion of neutron flux

Heavy nuclei: neutrons lose less energy, high reflection/transmission

Principle of Fast-Neutron Imaging



Transmission decreases exponentially (reflectivity increases) with thickness and density of sample, for most materials.



$\phi(0)$ $\rightarrow$ δ $\leftarrow$ $\phi(\delta)$

incoming transmitted

$$\phi(\delta) = \phi(0) \cdot T(\delta)$$

$$T(\delta) = e^{-\Sigma \cdot d} \quad Transmission$$

$$\Sigma = \mu \cdot \rho = N_A \cdot \sigma_A \cdot \rho$$

$$\mu = atten.coeff$$

$$\rho = material\ density$$

NEUTRON detectors

Detection of secondary events produced in

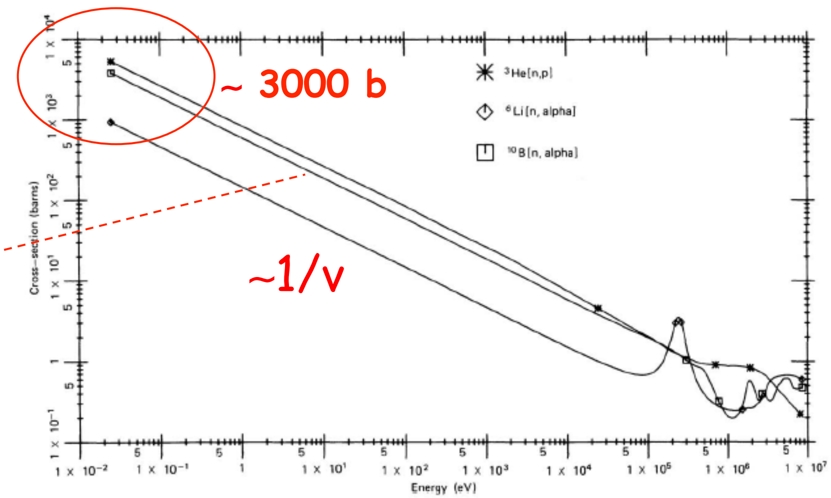
- <u>nuclear reactions</u>: (n.p), (n,α), (n,γ), (n,fission)
- <u>nuclear scattering</u> from light charged particles, than detected

For slow and thermal n:

(n,p) and (n,α) mostly …

Best reaction: $^{10}B + n \rightarrow {}^{7}Li^{*} + \alpha$

γ = 0.48 MeV



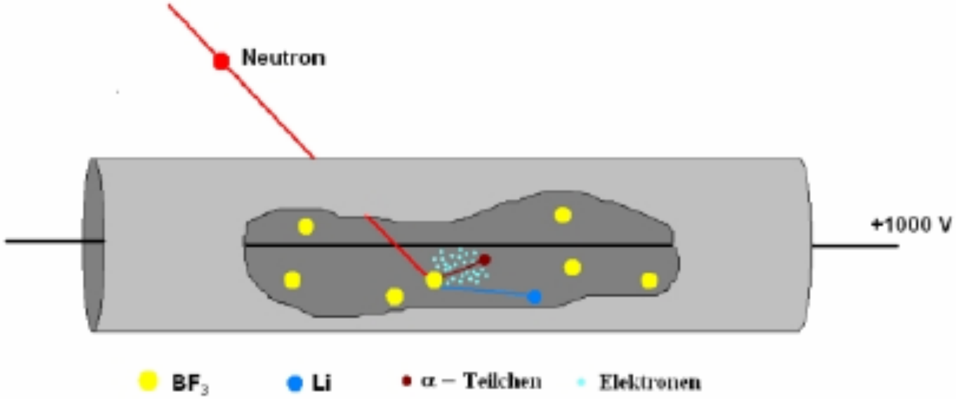
• <u>Natural abundance</u> of ^{10}B is high: 20%
 ⇒ material enriched with ^{10}B increase the efficiency

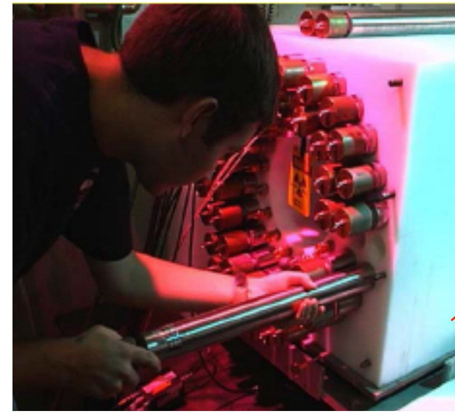
• <u>Exotermic reaction</u>:
 it occurs also for very low n energy

 For E_n = 0.025 eV, σ = 3770 b !!
 ⇒ ideal for n detection
 ⇒ useful for <u>measuring the flux of thermal n</u>

Ionization chamber/proportional counters filled with BF_3 gas

[typical sized tubes: e.g., 2 - 5 cm diameter]





Paraffine as moderator

$$n + {}^{10}B \rightarrow ({}^{11}B)^*$$

$$\rightarrow {}^{7}Li^* + {}^{4}He,\ Q = 2.31\ MeV\quad [94\%]$$

K.E. 0.84 MeV + 1.47 MeV

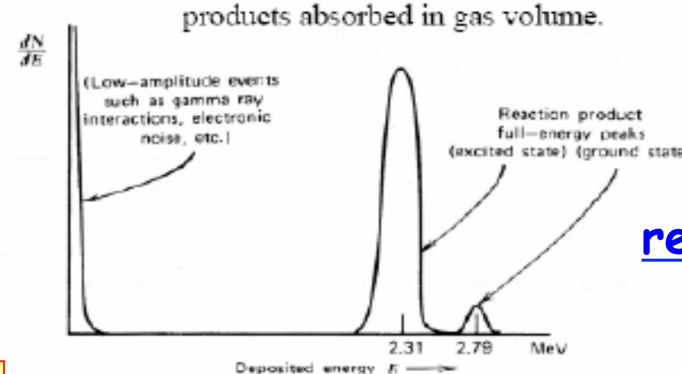
$$\rightarrow {}^{7}Li + {}^{4}He,\ Q = 2.79\ MeV\quad [6\%]$$

- BF_3 gas, enriched to >90% of ^{10}B
- Operated as proportional or G-M counter

Only determination of interaction, NOT energy

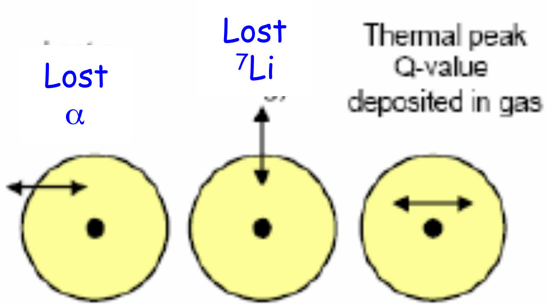
Wall effect prevents precise energy measurements

"Ideal" response: large tube, all reaction products absorbed in gas volume.

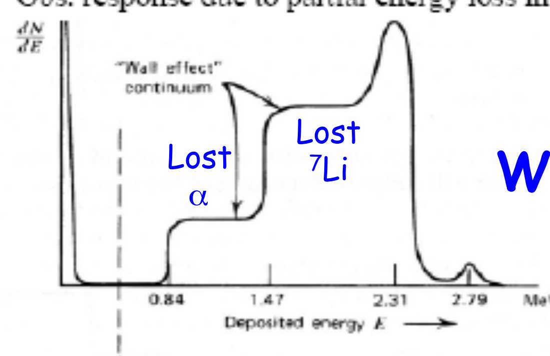


Ideal response

The wall effect



Obs. response due to partial energy loss in tube walls



Real Response

Wall effect

Due to partial energy deposition of lost ion

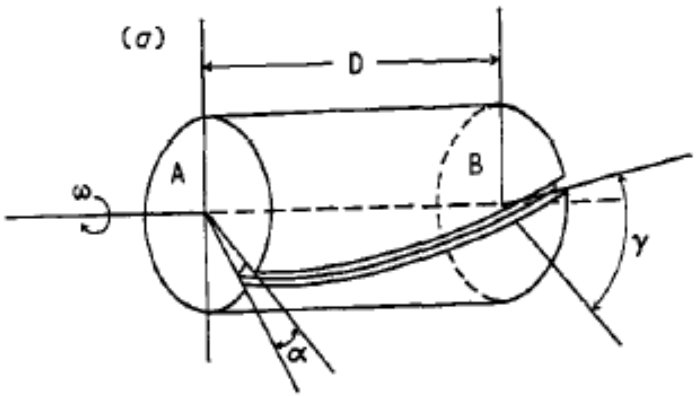
Neutron Energy Determination

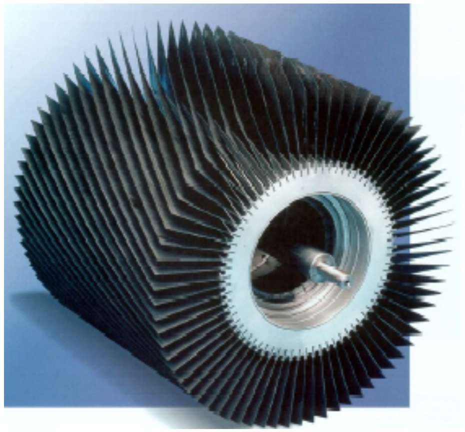
Velocity selector

rotating cylinder with one or more **helical slots** cut into the surface;
cylinder made with **highly absorbing material** (Cd, …)

only n with $t = \mathbf{D/v} = \gamma/\omega$ will pass ($\gamma = \omega T$)

changing angular speed ω permits selection of neutron velocity





Practical for ***thermal energy*** only:
$v \sim 2200$ m/s $\Rightarrow$ ~7000 rpm

Chopper for neutron pulses (variant)

Continuous stream of neutrons enters
from left and a pulse of neutrons emerges at right
if the rotor slits line up with the entrance slits

C.D. Clark et al. J.Sci.Instrum. 43 (1966)1

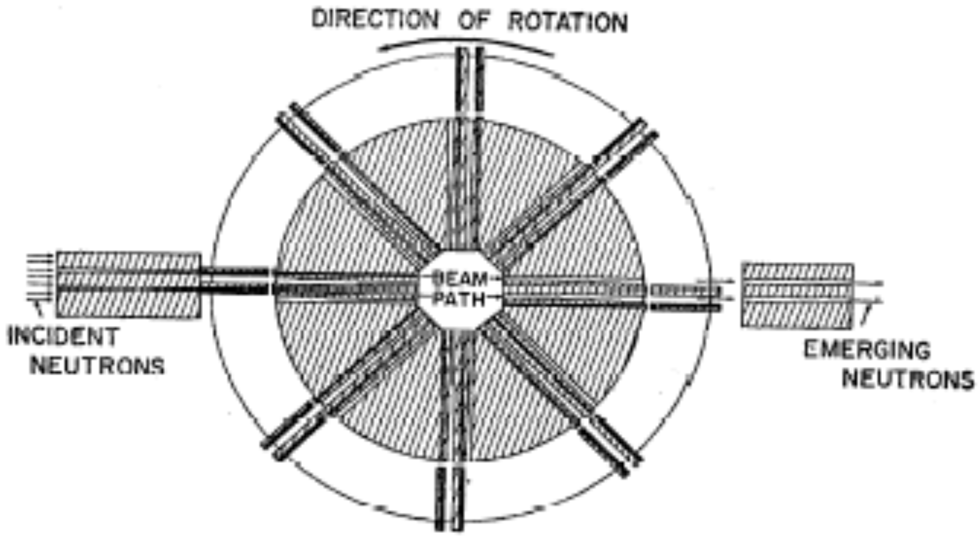
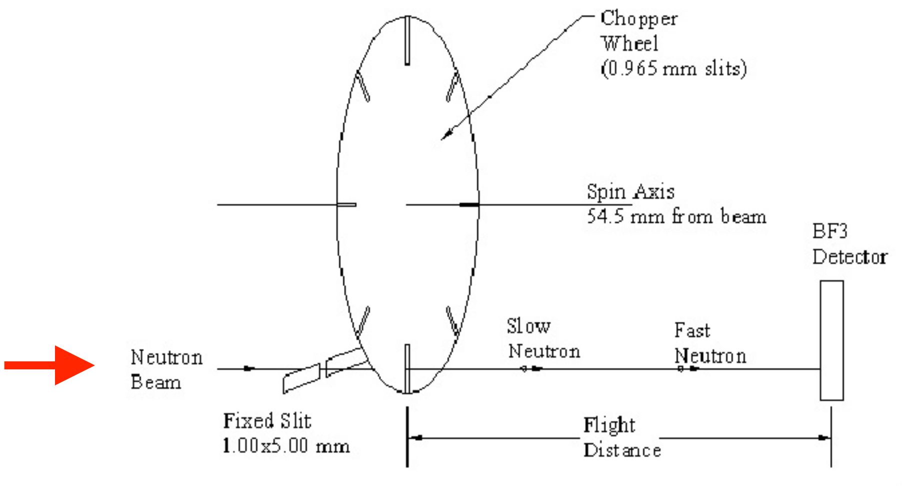
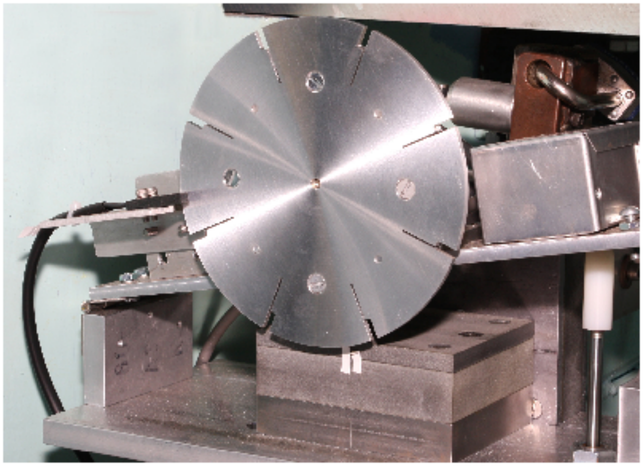


FIG. 2. A schematic plan view of the fast chopper rotor and entrance and exit stators. The shaded areas indicate plastic that is opaque to neutrons. The shape of the slit system is such that each pair of rotor slits opens only once per revolution.



igure 2. Schematic diagram of the time-of-flight spectrometer showing beam chopper and the BF3 Detector

The chopper is a slotted disk of neutron-absorbing cadmium which rotates about an axis to the neutron beam line, as illustrated below.



Time of flight techniques

In a TOF measurement, one determines the speed of the neutron v from the time t it takes to travel a flight path of length L. The kinetic energy of the neutron is given by

$$E = Mc^2\left(\frac{1}{\sqrt{1-\beta^2}} - 1\right) = Mc^2\left(\frac{1}{\sqrt{1-L^2/c^2t^2}} - 1\right) \qquad (14.54)$$

where

$$Mc^2 = 939.552 \text{ MeV} = \text{rest mass energy of the neutron}$$

The nonrelativistic equation is the familiar one,

n kinetic energy: NON-rel. limit

$$E_{NR} = \frac{1}{2}Mv^2 = \frac{1}{2}M\frac{L^2}{t^2} \qquad (14.55)$$

In neutron TOF experiments, the neutron source is a burst of neutrons generated either by a velocity selector (chopper) or by an ion beam, as explained later in this section. The time-of-flight t is the difference between the time of production of the neutron burst and the time of neutron detection.

The uncertainty Δt consists of three parts.

1. Δt_s is the uncertainty in the time of neutron emission; it is essentially equal to the width of the neutron burst and ranges from a few hundred nanoseconds to less than a hundred picoseconds.
2. Δt_d is the uncertainty in the time of neutron detection; it depends on the pulse risetime, since it is the pulse risetime that signals the time of detection. Neutron detectors used today have a pulse risetime equal to 5 ns or less.[48]
3. Δt_m is the uncertainty in neutron slowing-down time if the source is surrounded by a moderator.

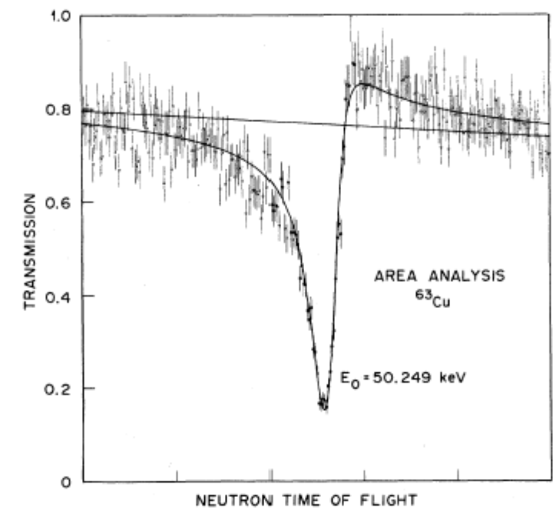


FIG. 1. An example of the area program fit to the transmission data of an isolated resonance in ^{63}Cu at $E_0 = 50.249$ keV.

$$\frac{\Delta E}{E} = 2\sqrt{\left(\frac{\Delta L}{L}\right)^2 + \left(\frac{\Delta t}{t}\right)^2}$$

The uncertainty ΔL is due to the finite thicknesses of the neutron-producing target and the neutron detector. The uncertainty in the measurement of L itself can be made negligible. The longer the flight path is, the smaller the uncertainty $\Delta L/L$ becomes. As the length L increases, however, the intensity of the source should increase, and by a greater factor, so that the counting rate in the detector stays the same.

PRECISE Energy measurements in THERMAL region:

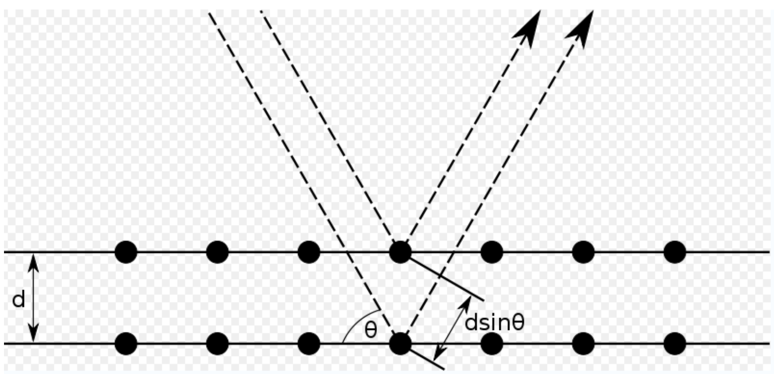
Crystal Diffraction

$\lambda_n \sim 0.1$ nm similar to spacing **d** between atoms in a crystal lattice

Bragg condition for interference maxima

→ *[wave nature of neutron beam]*

$$n\lambda = 2d\sin\theta$$



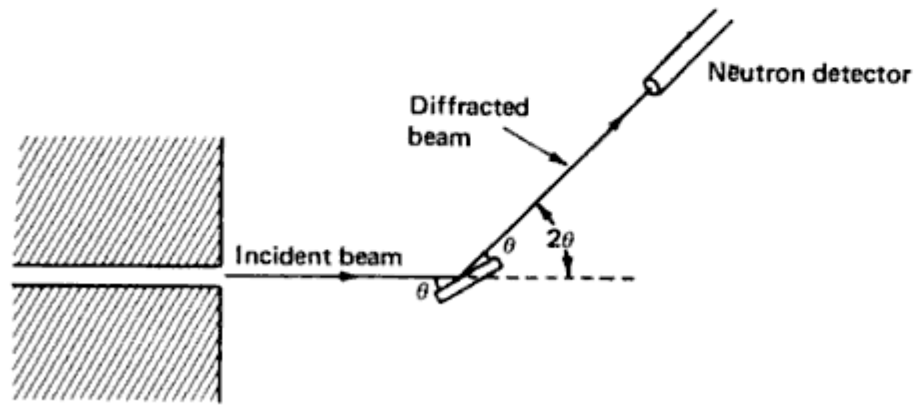


Figure 14.19 The arrangement of a neutron diffraction spectrometer.

Technique used to study
crystalline properties/atomic spacing of materials

Monochromator:

How to use Bragg Condition to produce Monoenergetic neutron beam

- High energy neutrons generated from nuclear reactor source of spallation source are moderated by light water (H2O, ~300K) or liquid hydrogen (H2, ~20K) and its velocity (or energy) is distributed accroding to the Maxwellian distribution. By selecting necessary energy (or wavelength) range of neutron beam, which is called 'monochromatization', various versatile experiment could be done.
  Monochromatization means the process of diffraction toward the special direction (2) with characteristic wavelength( ) determined by the spacing of lattice plane and the Bragg condition (2d sin = n). The crystal used for this process is called as 'monochromator'.

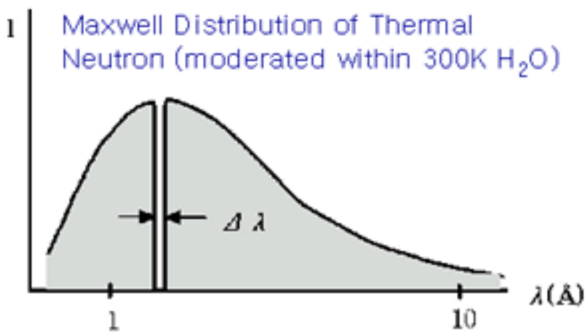


Fig. 1 Neutron spectrum moderated by light water

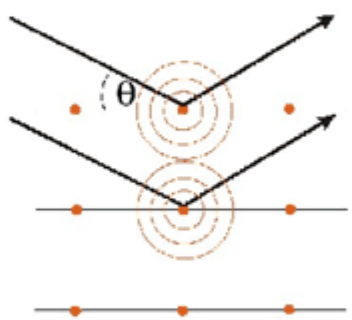
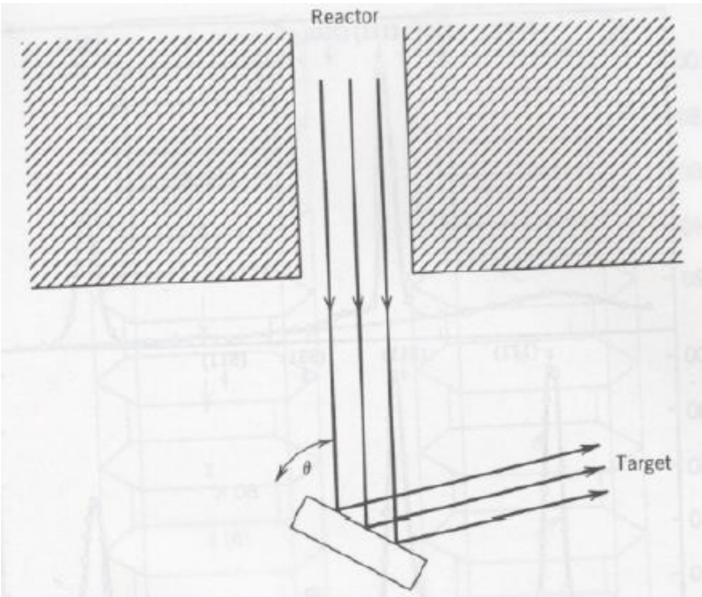


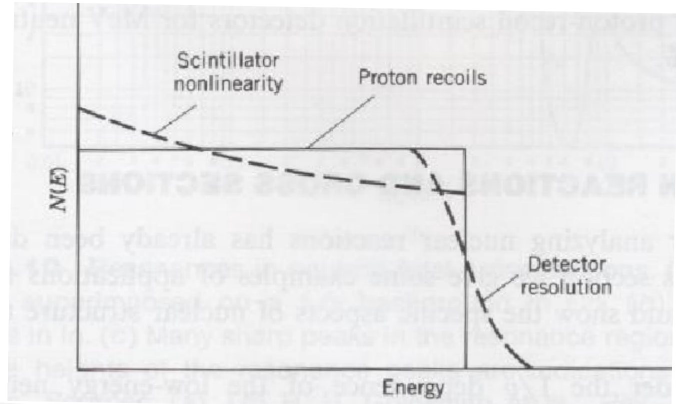
Fig. 2 Bragg reflection by crystal plane



Energy measurements of FAST neutrons

Energy of recoil in elastic scattering on light targets

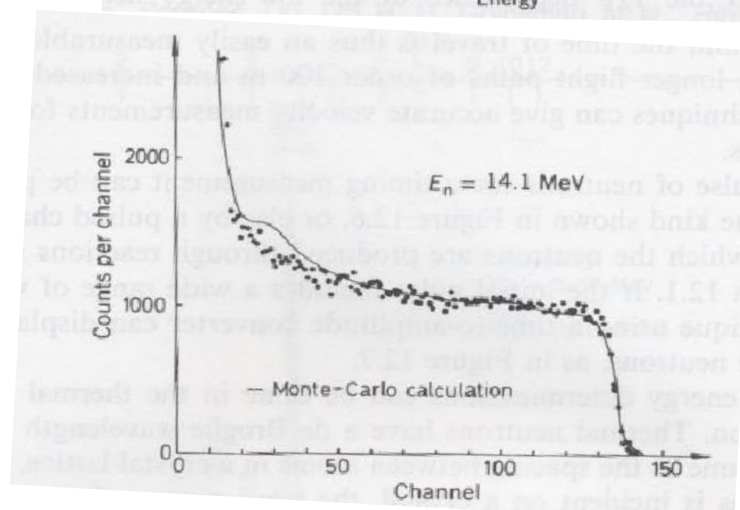
H, ^{2}H, ^{3}He, ^{4}He, …



$$E_R = E - E'$$

$$(E_R)_{max} = E - (E')_{min} = E\frac{4A}{(A+1)^2}$$

[E_{min}= $(A-1)^2/(A+1)^2 \times E$ and E_{max} = E]



in H $(E_R)_{max}$ = E
in ^{3}He $(E_R)_{max}$ = 0.75 E

Proton recoil signal is normally observed in scintillator materials rich of H, such as plastic/organic liquid

⇒ scintillator serves as target (for neutrons) and detector (recoil protons)

Typical efficiency proton-recoil from MeV neutrons is ~50%

Review paper on n measurements

Neutron Flux and Energy Measurements

J. A. DENNIS et al,
PHYS. MED. BIOL. 1966, VOL, 11, NO. 1, 1-14