Neutrons for analytical applications

FYS-KJM 3506: Nukleære målemetoder og instrumentering
17. oktober 2026

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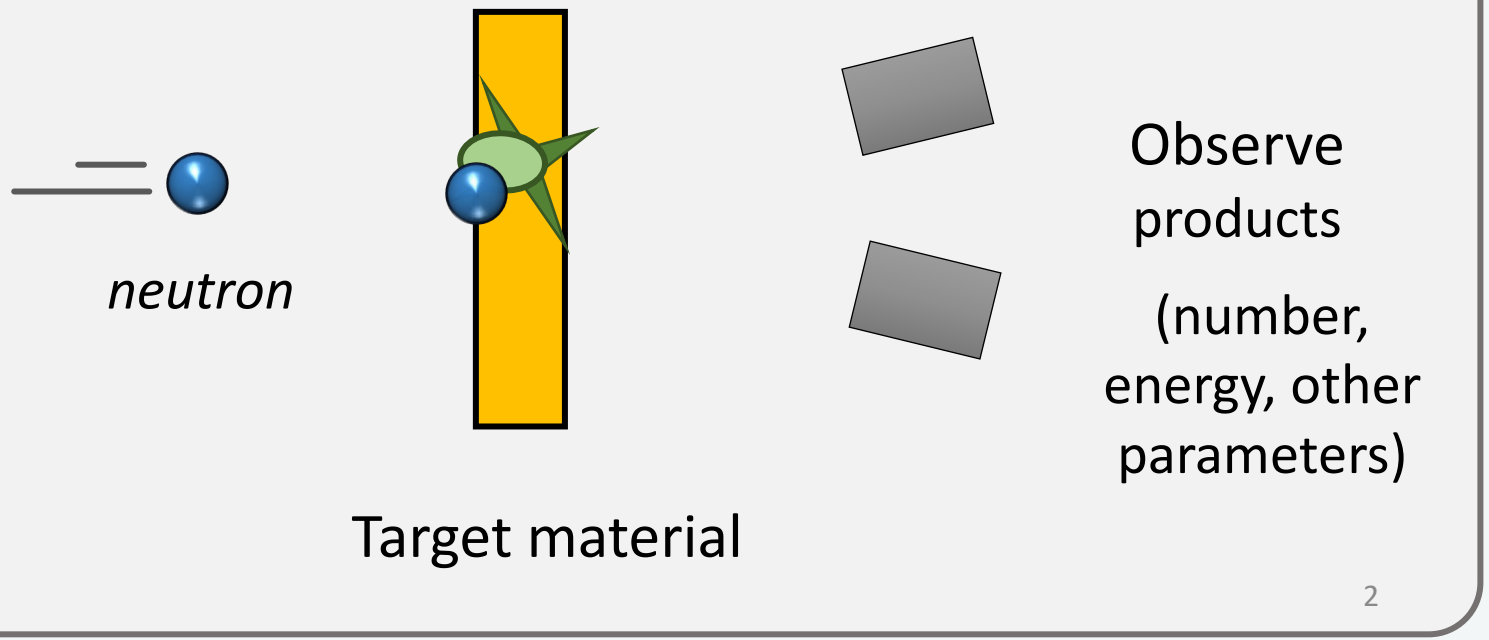
Neutrons
can be used as a probe/tool
to give us information
about a system

HOW?

Neutron Activation Analysis

Scattering,
diffraction,
imaging…

Other uses of neutrons:
- Energy
- Production of radionuclides



Neutron activation analysis, NAA, is the topic for this week's laboratory exercise

NAA is a sensitive analytical technique which allows us to determine elements present in a sample

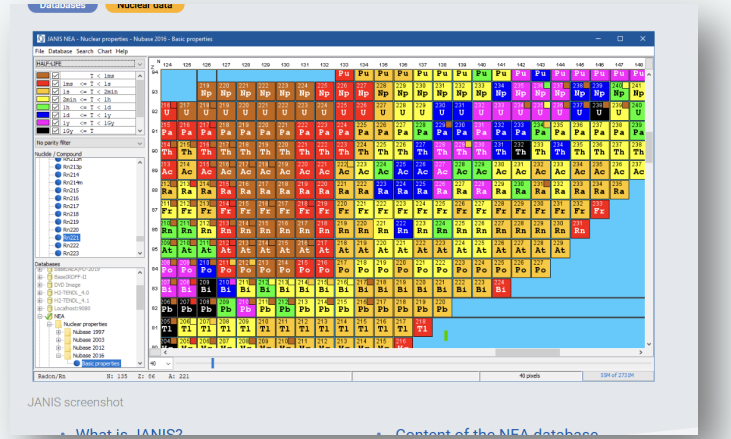
- Quantitative and qualitative analysis
- Non-destructive
- Due to the way neutrons interact with matter, technique is more or less matrix independent
- Offers very low detection limits for some elements
- High accuracy and low uncertainty
- About 70% element have properties suitable for NAA

1 Neutron interactions, basic NAA

	N-2	N-1	N	N+1
Z	(n, 3n)	(n, 2n)	Original nuklide (n, n')	(n, γ)
Z-1	(n, p3n)	(n, p2n)	(n, pn)	(n, p)
Z-2	(n, αn)	(n, α)	(n, ^{3}He)	
(n, *f*)				

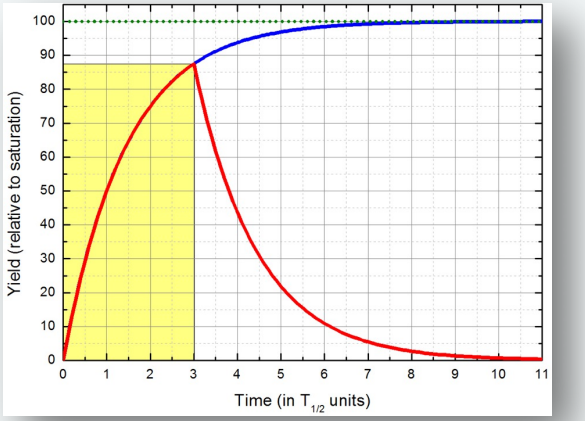


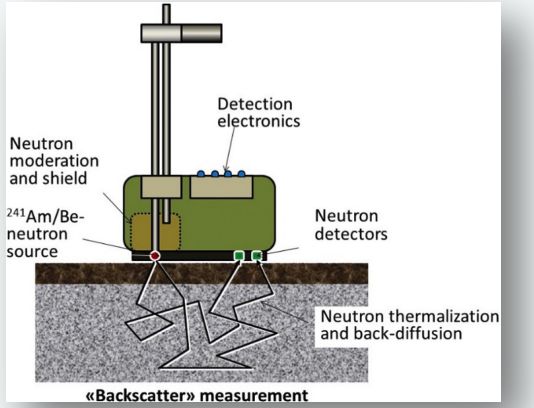
2 Neutron sources



3 What is being made?

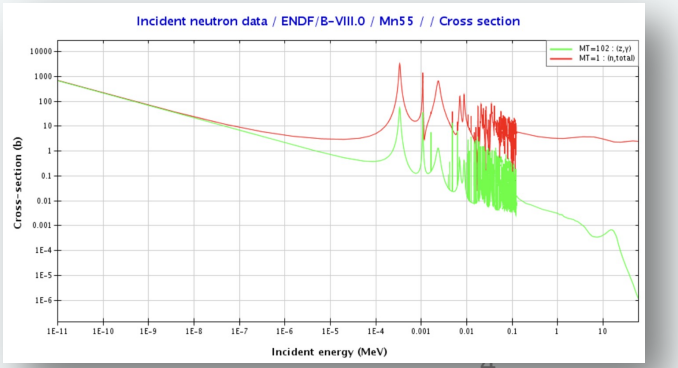
4 Quantitative analysis DGNNA





5 Some examples of macro-object applications

6 Laboratory exercise



1 Neutron interactions, basic NAA

The neutron has no charge is a small magnetic moment: interaction with matter is though nuclear forces and not EM interactions

Cold neutrons: $E_n < 12$ meV

Thermal neutrons:
12 meV $< E_n < 100$ meV

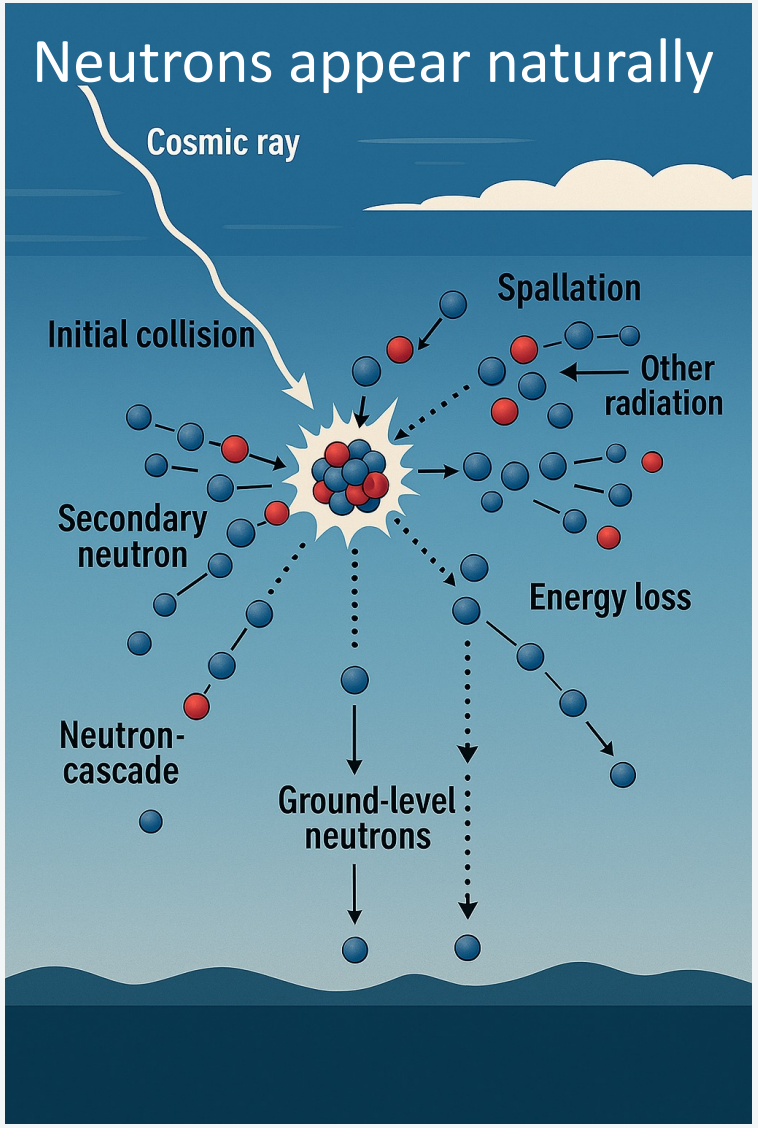
Epithermal neutrons:
100 meV $< E_n < 100$ keV

Fast neutrons:
$E_n > 1$ MeV

High energy neutrons: $E_n > 20$ MeV

Primary cosmic rays

Secondary cosmic rays



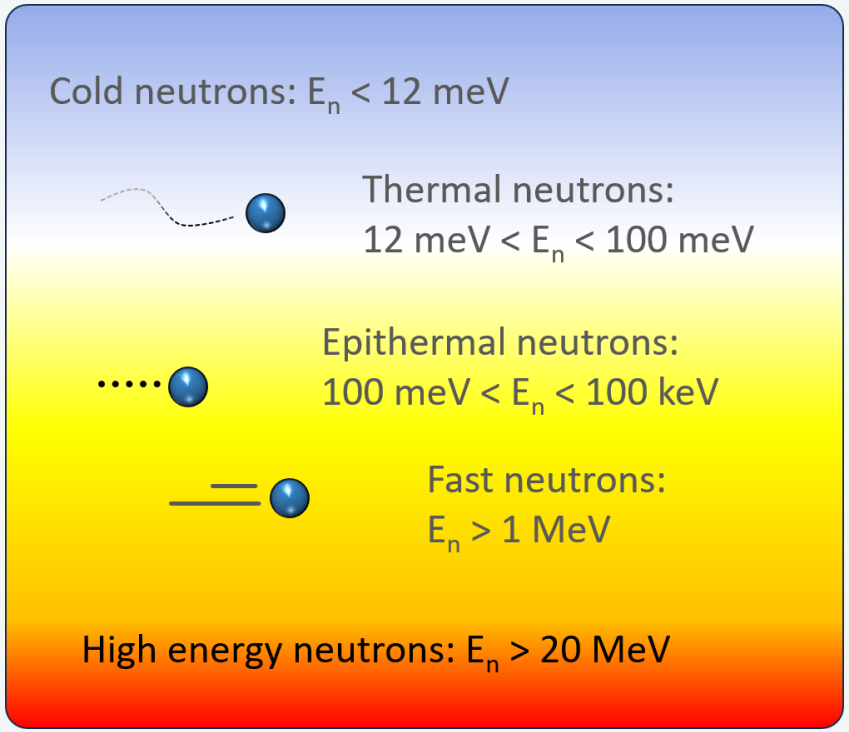
Interaction with matter depends on the energy of the neutron and the intensity of the source

ENERGY

Determines what the neutron can do when it interacts with matter

INTENSITY

Determines how 'well' or how fast



Activation analysis; many elements can be identified using thermal neutrons, but some elements are only identifiable with fast neutrons

Production of radionuclides for medical applications require high intensity and often thermal neutrons

In material characterization fast and thermal neutrons are applied depending on the technology

Possible neutron interactions

(1) Scattering reactions

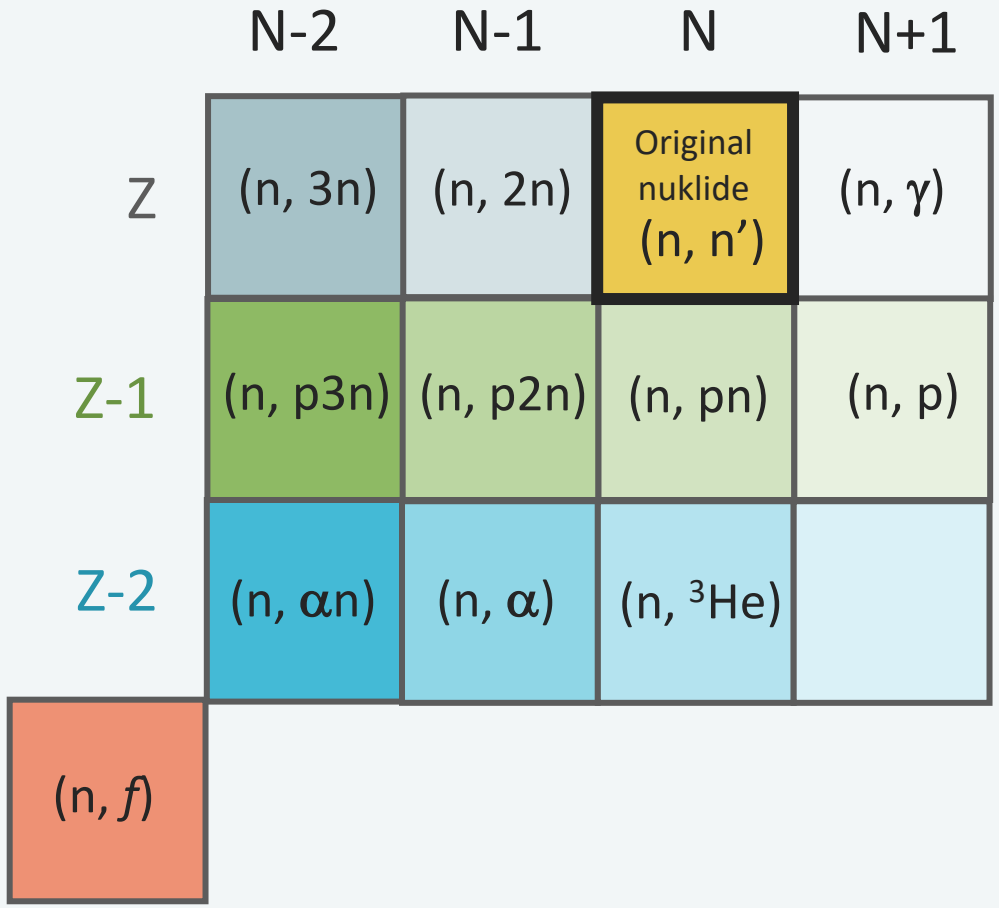
- Elastic scattering
- Inelastic scattering

(2) Absorption reactions

- Radiative capture
- Fission

(3) Transfer reactions

- Charged particle production
- Neutron production reaction



Possible neutron interactions

(1) Scattering reactions

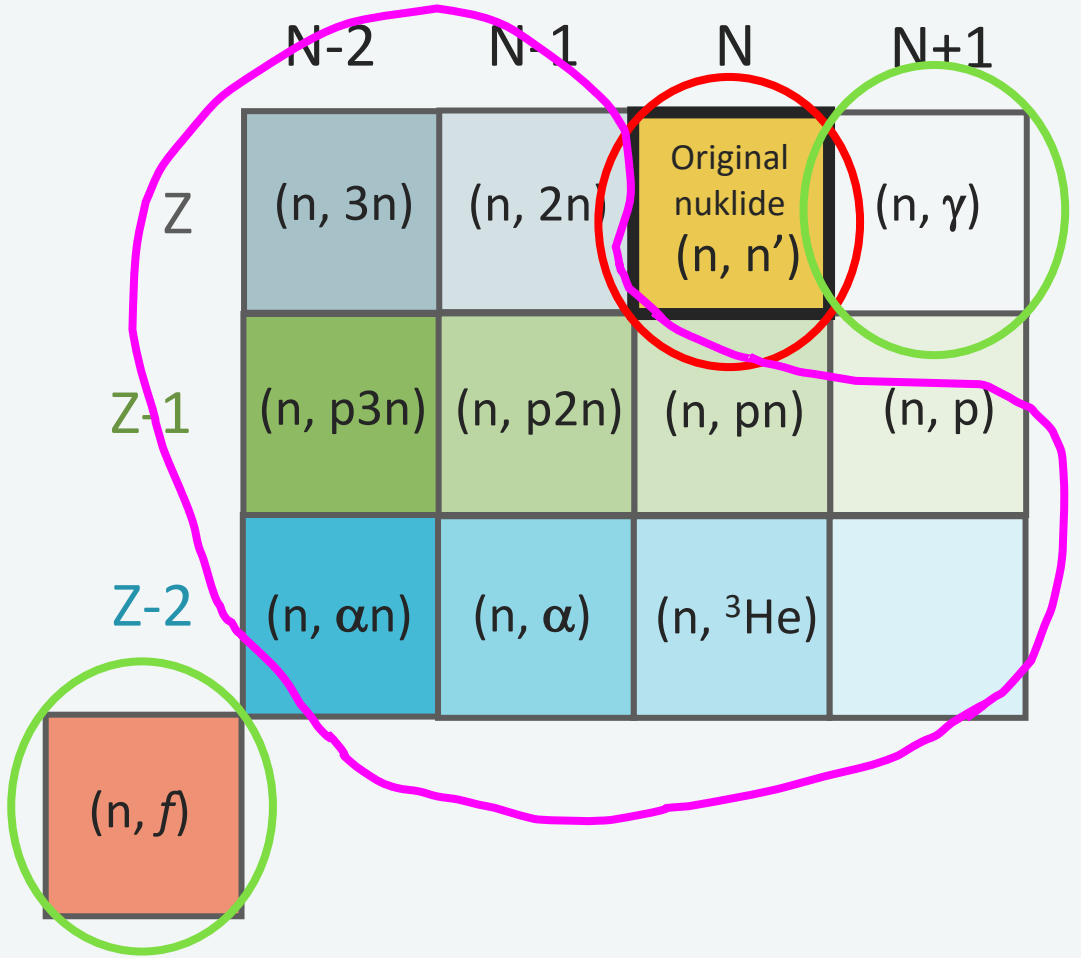
- Elastic scattering
- Inelastic scattering

(2) Absorption reactions

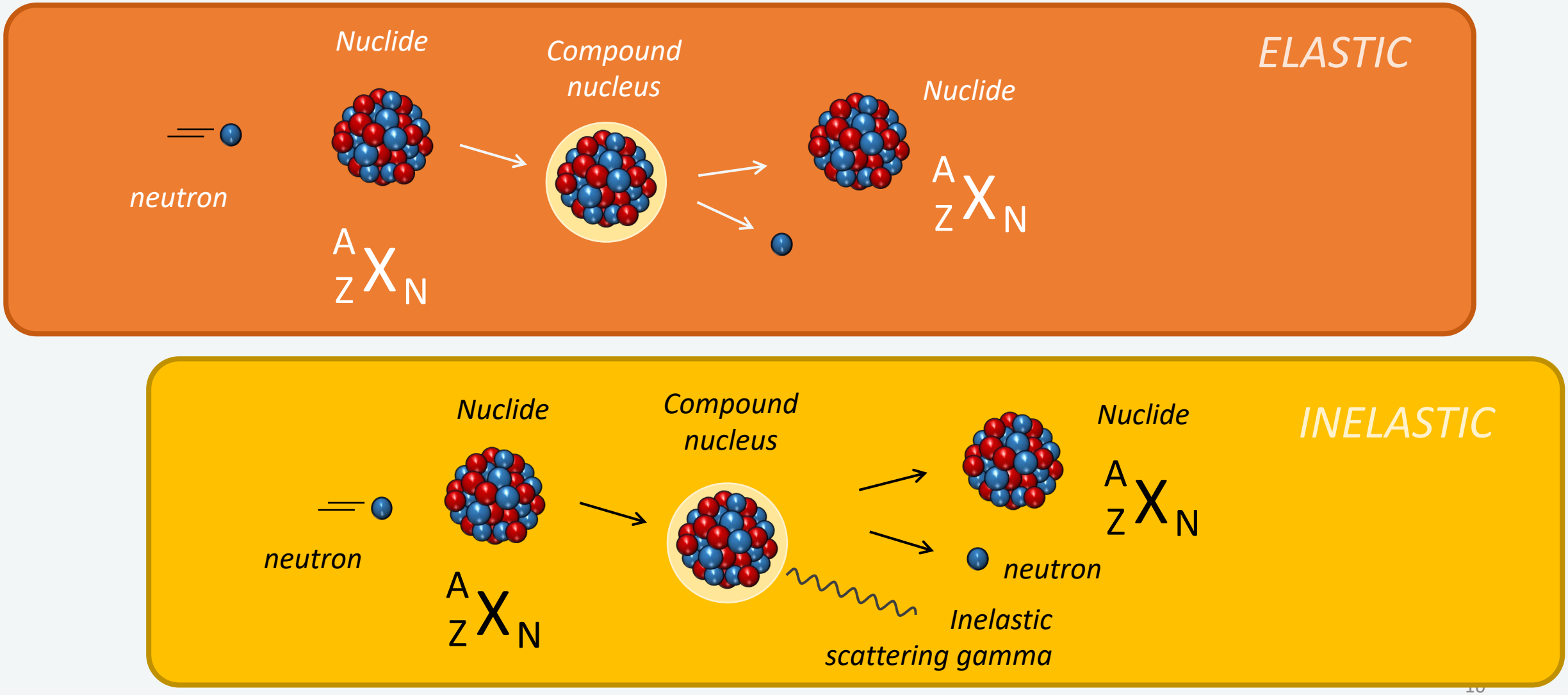
- Radiative capture
- Fission

(3) Transfer reactions

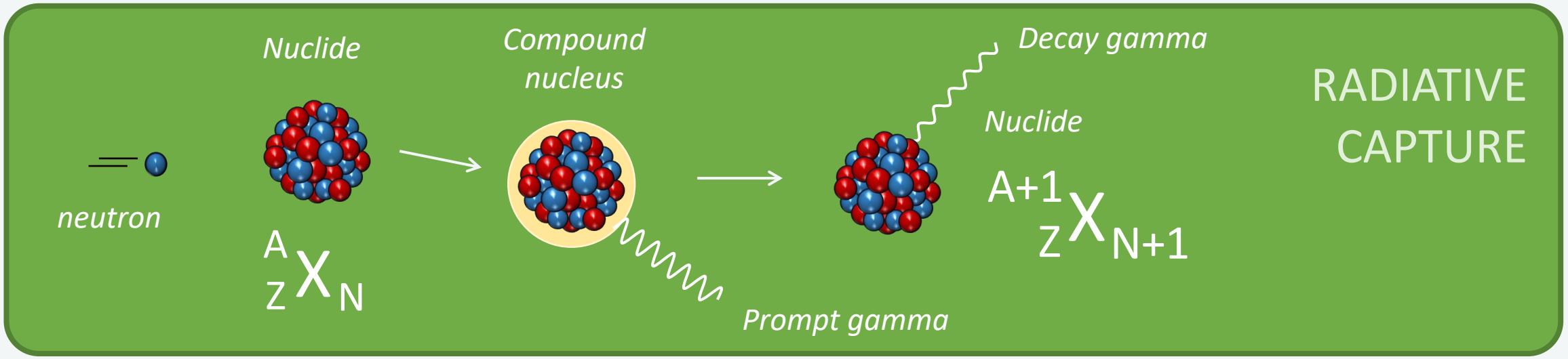
- Charged particle production
- Neutron production reaction

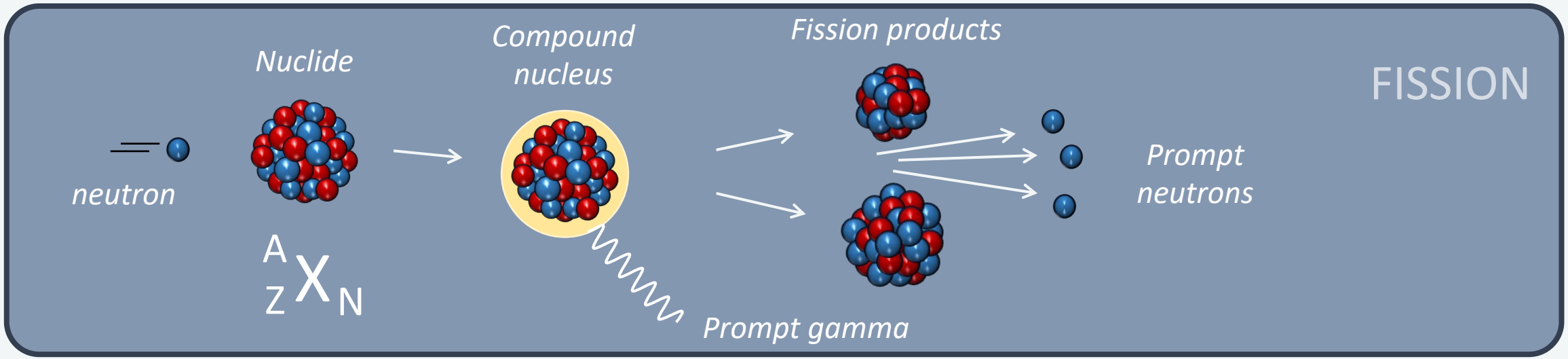


Scattering can either be elastic or inelastic

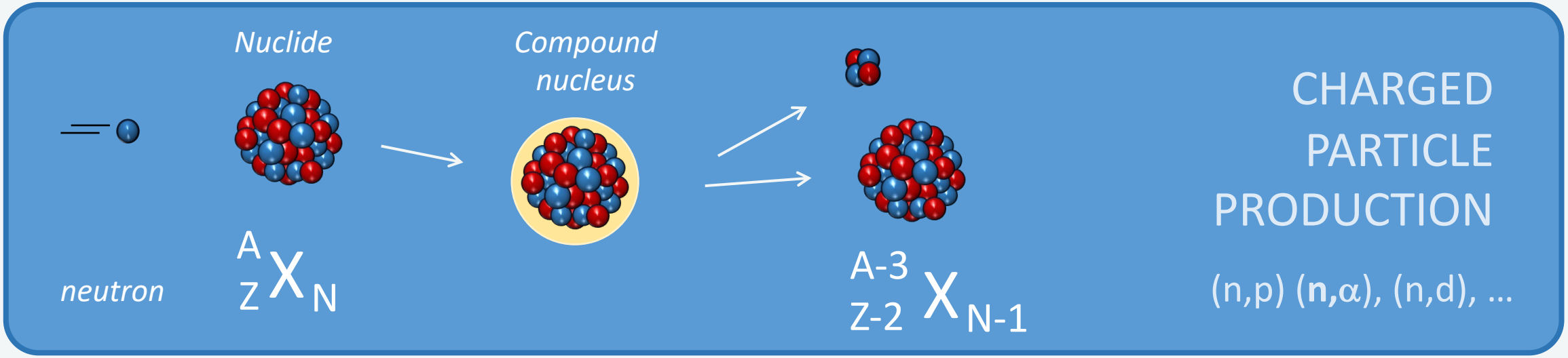


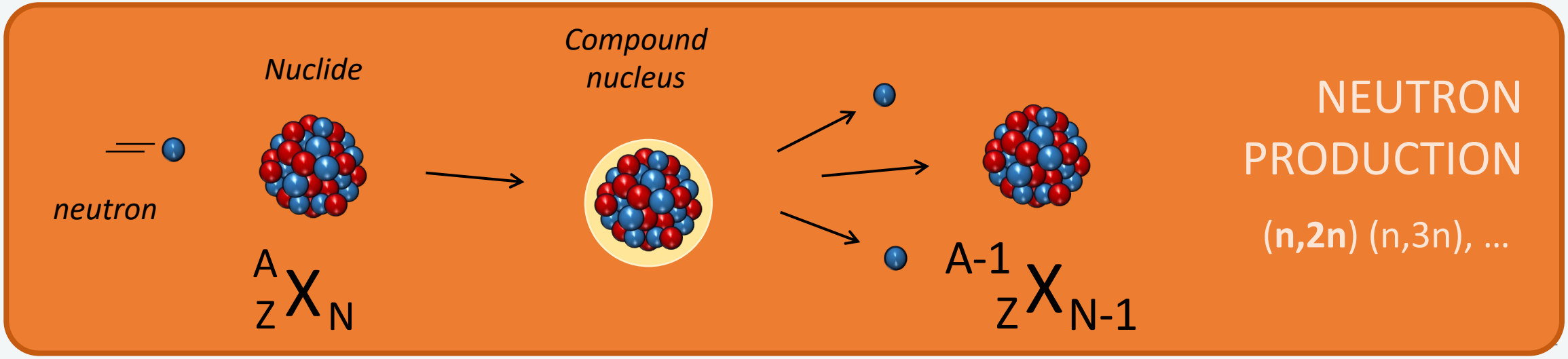
Absorption reactions:
the neutron can be absorbed or lead to fission





Transfer reactions leads to charged-particle or neutron production





Scattering reactions

- Elastic scattering – can occur at any incident neutron energy
- Inelastic scattering – threshold energy: $E_{th} = E^* [(A+1)/A]$

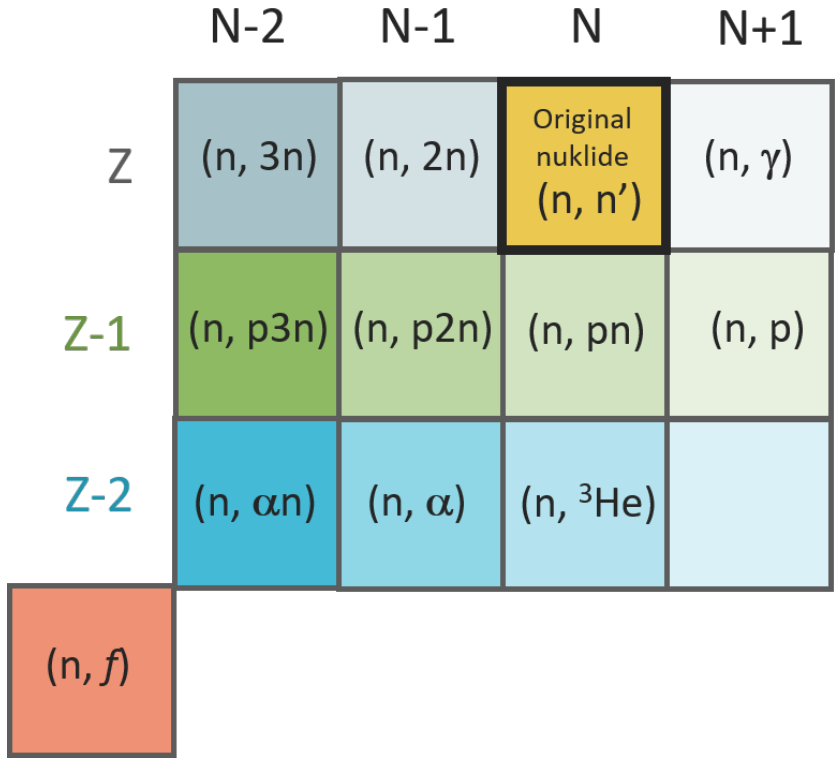
Absorption reactions

- Radiative capture – typically highest for thermal neutron energies, some has high likelihood in resonance region
- Fission – some heavy nuclides have very high likelihood for fission by low energy neutrons

Transfer reactions

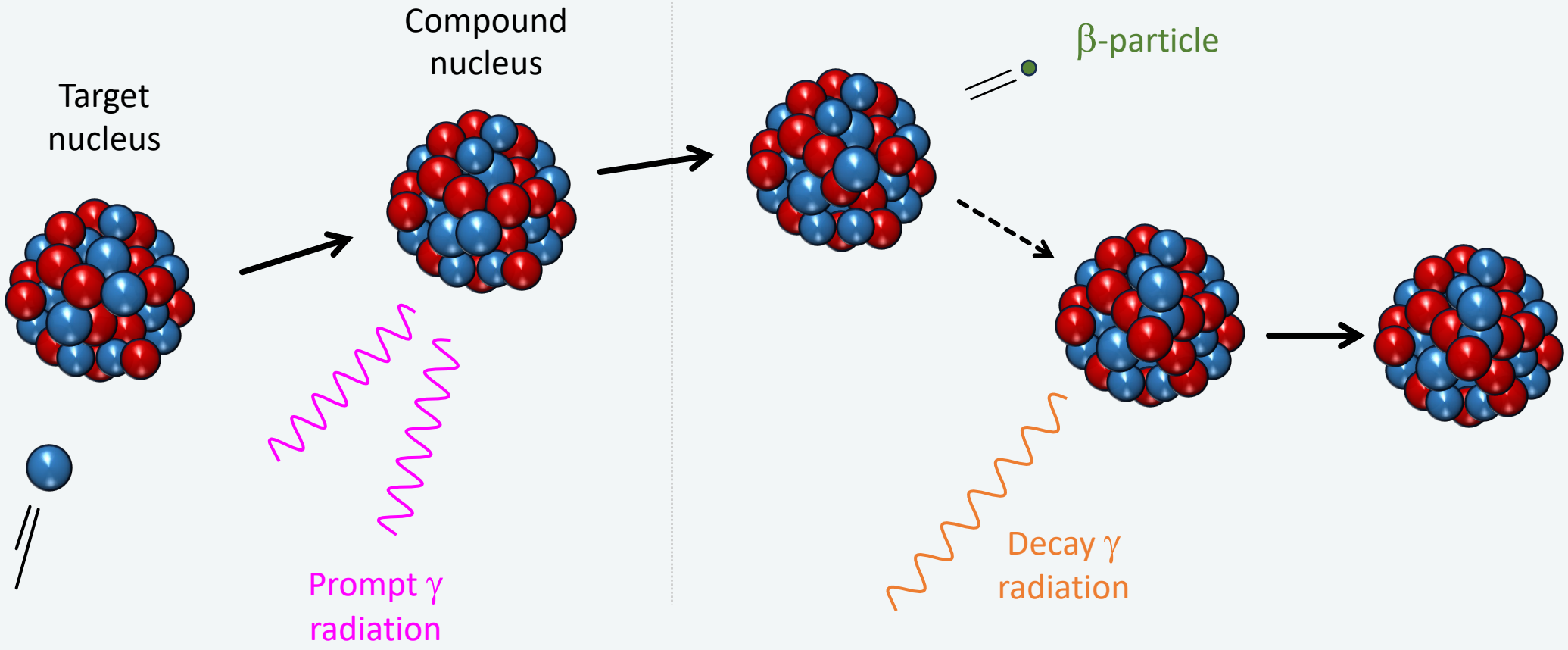
- Charged particle production – threshold energy, must be over Coulomb barrier of outgoing particle
- Neutron production reaction – threshold energy related to the neutron binding energy in target nuclide

If we want to identify an element – what mechanisms are relevant and happening at the energies of our neutron source?



Nuclear reaction

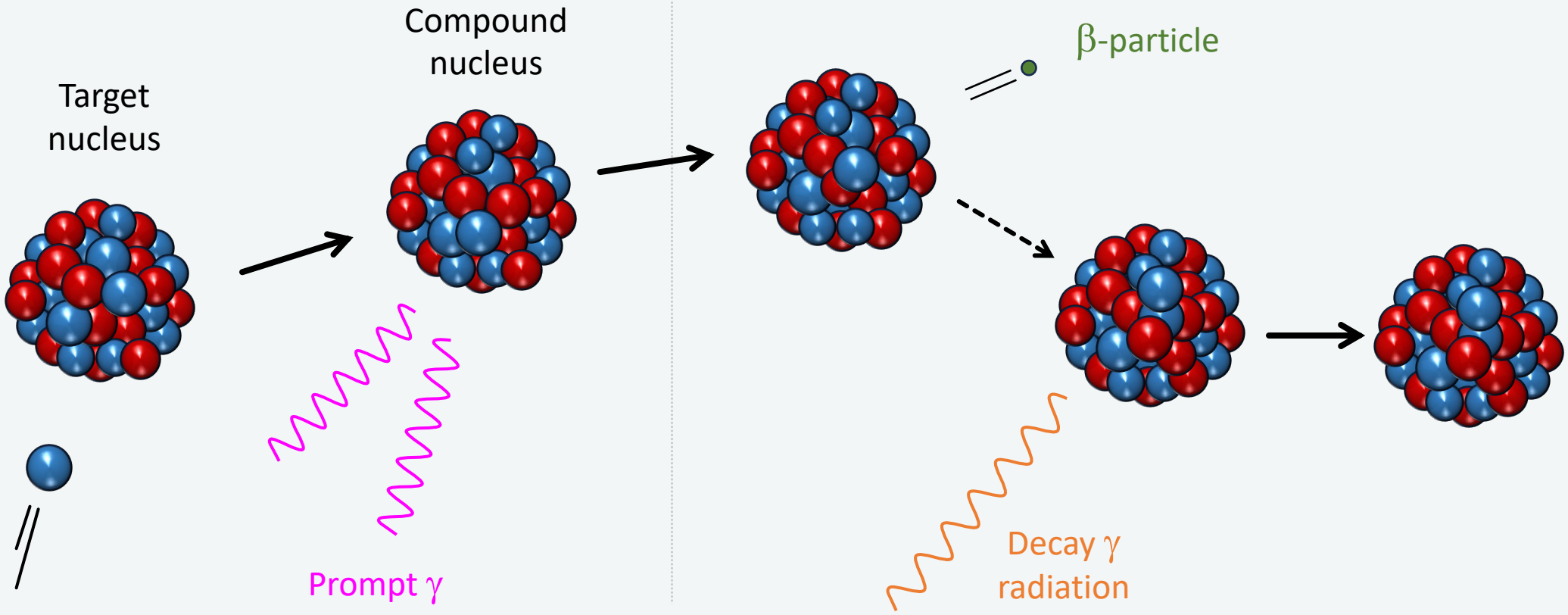
Radioactive decay



Prompt y – high energy photons emitted within nanoseconds of the capture
These are characteristic photon!

Nuclear reaction

Radioactive decay



Prompt Gamma Neutron Activation Analysis
PGNNA 10^{-16} – 10^{-12} s

Delayed Gamma Neutron Activation Analysis
DGNNA 10^{-6} – 10^{3} s

DGNNA
Delayed Gamma Neutron Activation Analysis

- INAA – Instrumental NAA
- RNAA – Radiochemical NAA
- FNAA – Fast NAA

Conventional: most common

Possible to measure many different elements in same sample without chemical processing

If chemical separation is applied to remove interferences or concentrate radionuclide of interest procedure become labor intensive and costly.

PGNNA
Prompt Gamma Neutron Activation Analysis

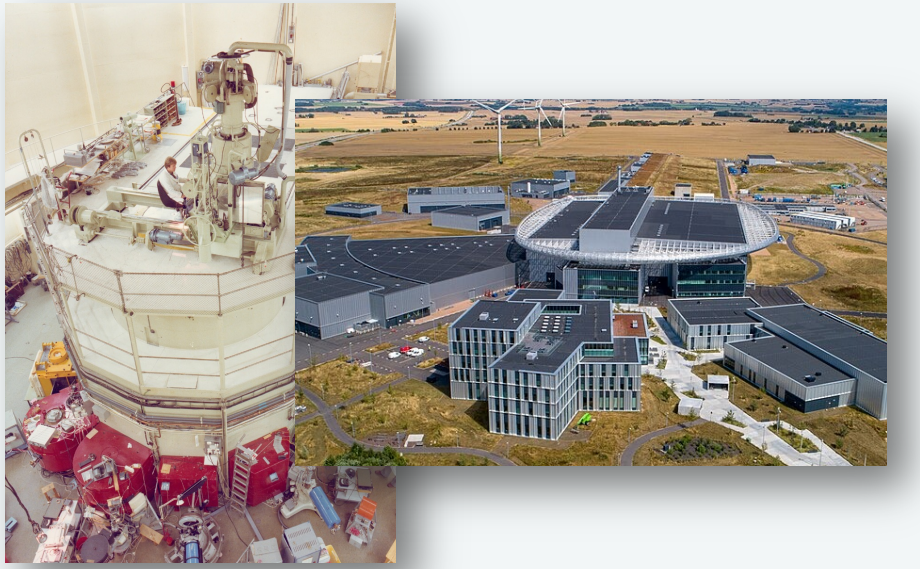
- TNC PGNNA
  – Thermal Captuere PGNAA
- NIS PGNNA
  – Neutron Inelastic Scattering PGNAA

Set-up might be challenging - detectors need to be protected from the neutrons

Useful when half-life are too short for conventional, the produced nuclides are stable or just have to weak/no γ-lines

2 Neutron sources

Reactors and High Energy Accelerators



$10^{14} – 10^{19}$ n/s

High intensity
Large infrastructure

Compact Neutron Generators



$10^{10} – 10^{13}$ n/s

Moderate intensity
~Room size

Portable Neutron Generators and Radionuclide Sources



$10^{5} – 10^{9}$ n/s

Limited

Licensing, cost and regulatory requirements…

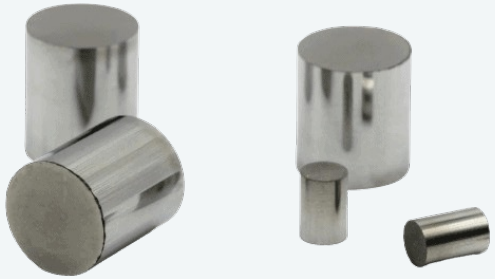
There are three types of neutrons sources based on radioactive materials:

(1) Neutrons from a spontaneously fissioning nuclide, ^{252}Cf ($T_{1/2}$ = 2.65 y)



(2) Alpha-particle emitter mixed with beryllium, α – n: ^{241}Am:Be, $^{238,239}Pu$:Be, ^{226}Ra:Be, ^{210}Po:Be

(3) Photon-induced, γ – n: ^{124}Sb:Be



https://frontier-cf252.com/products/antimony-beryllium/

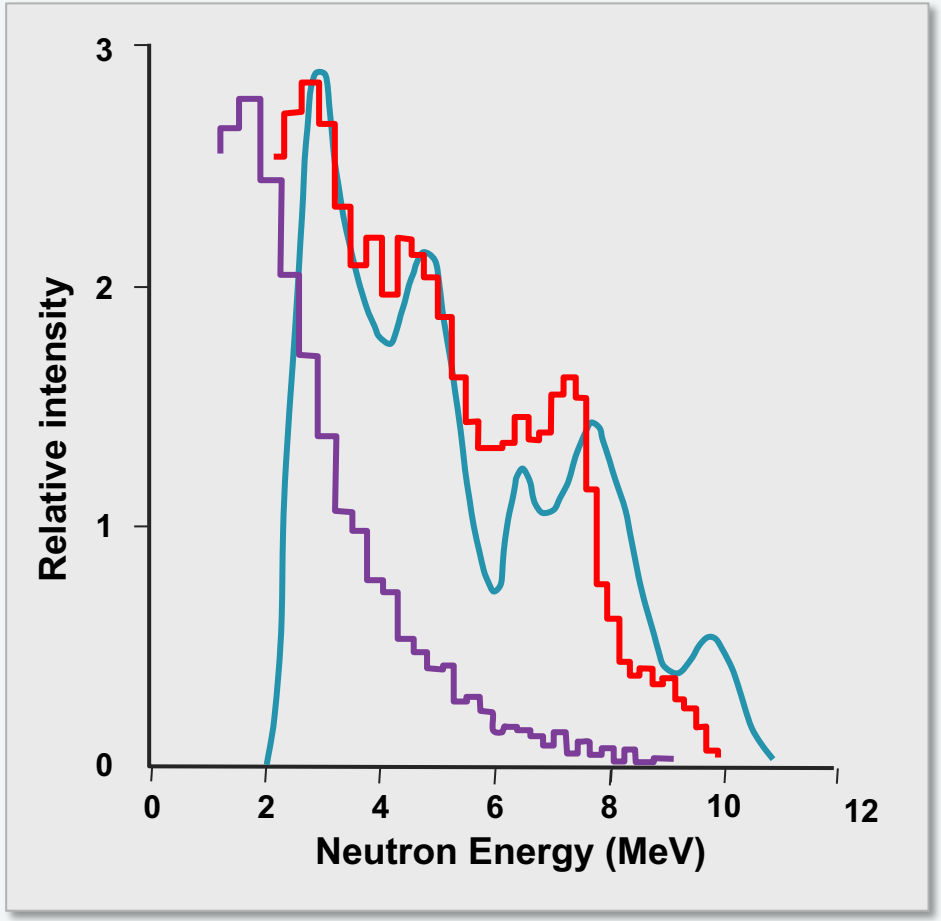


- Neutron energy spectrum similar to reactor fission spectrum
- High neutron yield per unit mass (allow for small sized sources)

^{252}Cf

- Activity of source depends on α-emitter activity, and how the materials are mixed/ constructed
- Be by far the most common, but is also possible with F or Li
- there is also a γ-dose depending on nuclide
- long term: potential build up of pressure due to helium being produced

$\alpha - n$



- Pro: emitted neutrons are close to mono-energetic
- Con: considerable γ-doserate

$\gamma - n$

Reactions in Be:

$^{9}Be + \alpha \longrightarrow {}^{12}C + n + \gamma$ (4.44 MeV)

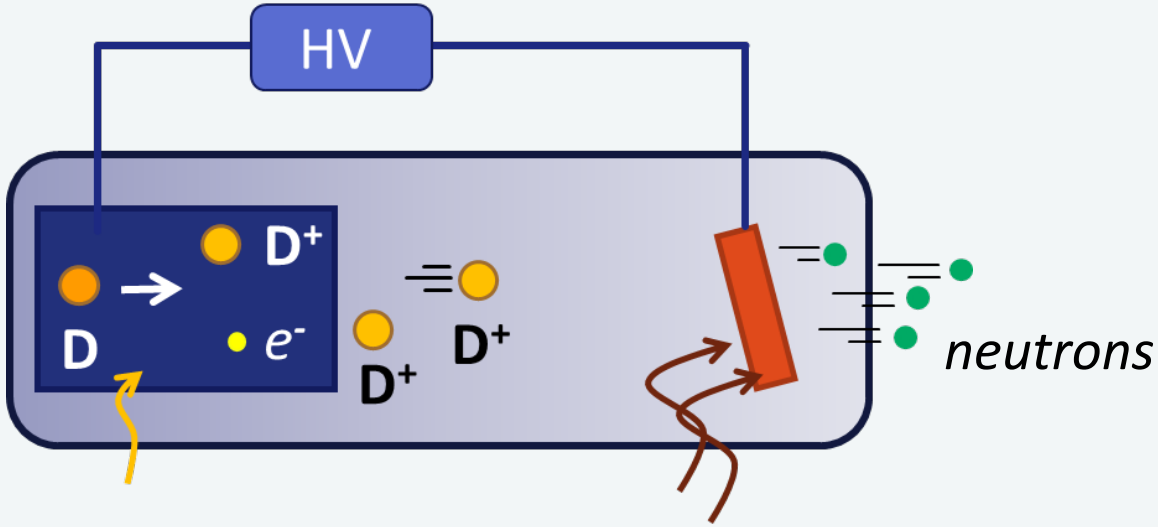
$^{9}Be + \gamma \longrightarrow {}^{8}Be + n$

^{252}Cf

^{238}Pu:Be

^{241}Am:Be

Neutron generators are accelerators based on reactions between hydrogen ions:



Target: D eller T
solid/gas

Neutrons
"on-demand"

Positive ion, constant voltage accelerator

DD GENERATOR

$^{2}H + {}^{2}H \longrightarrow$

^{3}He (0.82 MeV) + **n** (2.45 MeV) (50%)

^{3}H (1.01 MeV) + **p** (3.03 MeV) (50%)

DT GENERATOR

$^{2}H + {}^{3}H \longrightarrow$

^{4}He (3.51 MeV) + **n** (14.1 MeV)

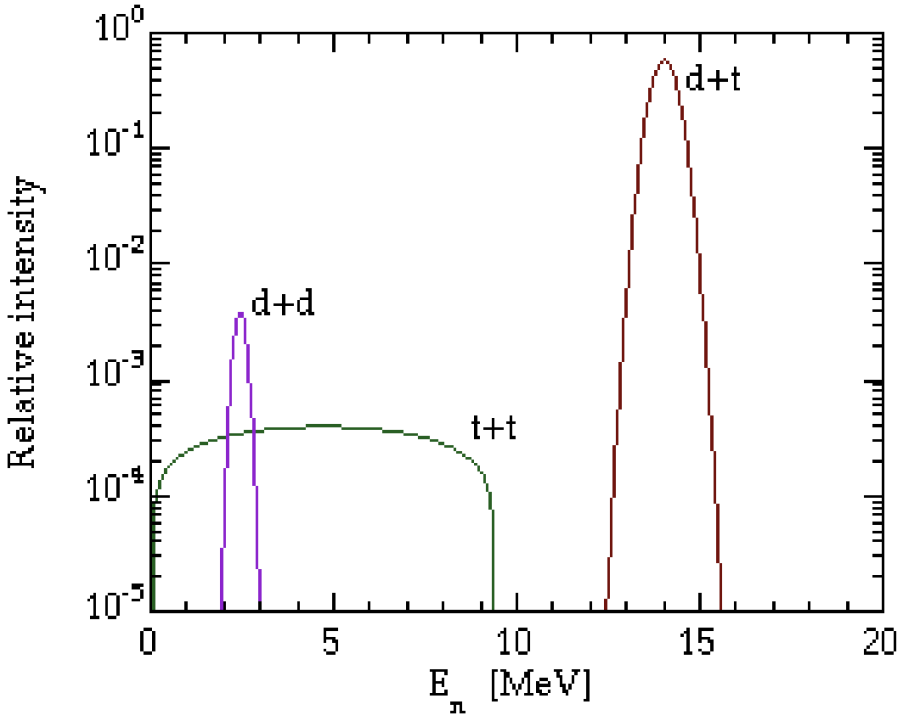


Figure 4. Fusion neutron energy spectra

COMPACT NEUTRON GENERATOR DEVELOPMENT AND APPLICATIONS

K.-N. Leung[1,2], J. Reijonen[1], F. Gicquel[1], S. Hahto[1], and T.P. Lou[1]
[1] Lawrence Berkeley National Laboratory, Berkeley, CA., USA
[2] Department of Nuclear Engineering, University of California, Berkeley, CA., USA





Parts:
Accelerator, High-voltage supply, and control console

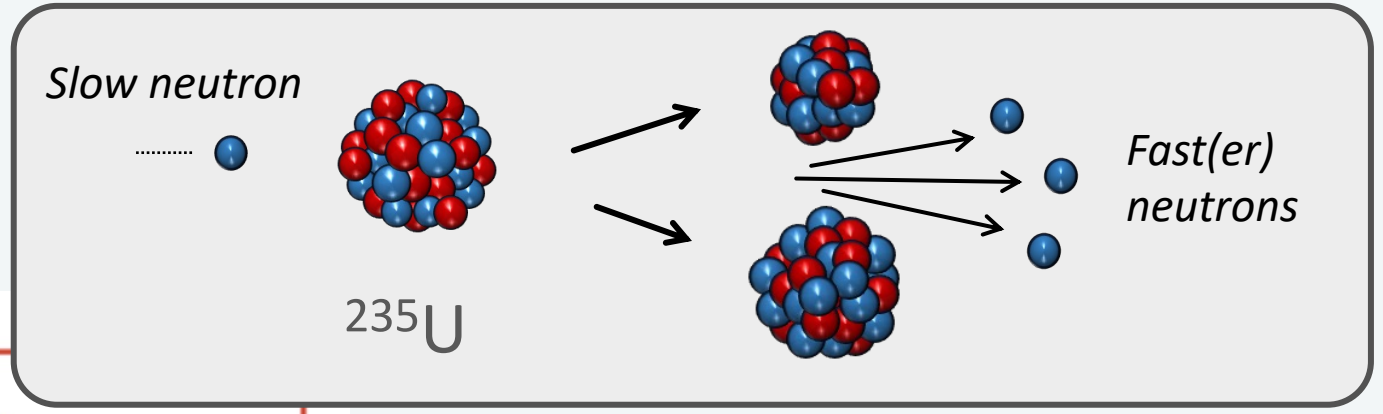
- Requires power supply of 100 – 500 keV to accelerate the deuterons
- Pulsed or continuous emission possible
- Ion-source, optics elements and target are all in vacuum tight enclosure

Neutron generators can be turn on/off,
but radionuclide sources are always ‘on’…

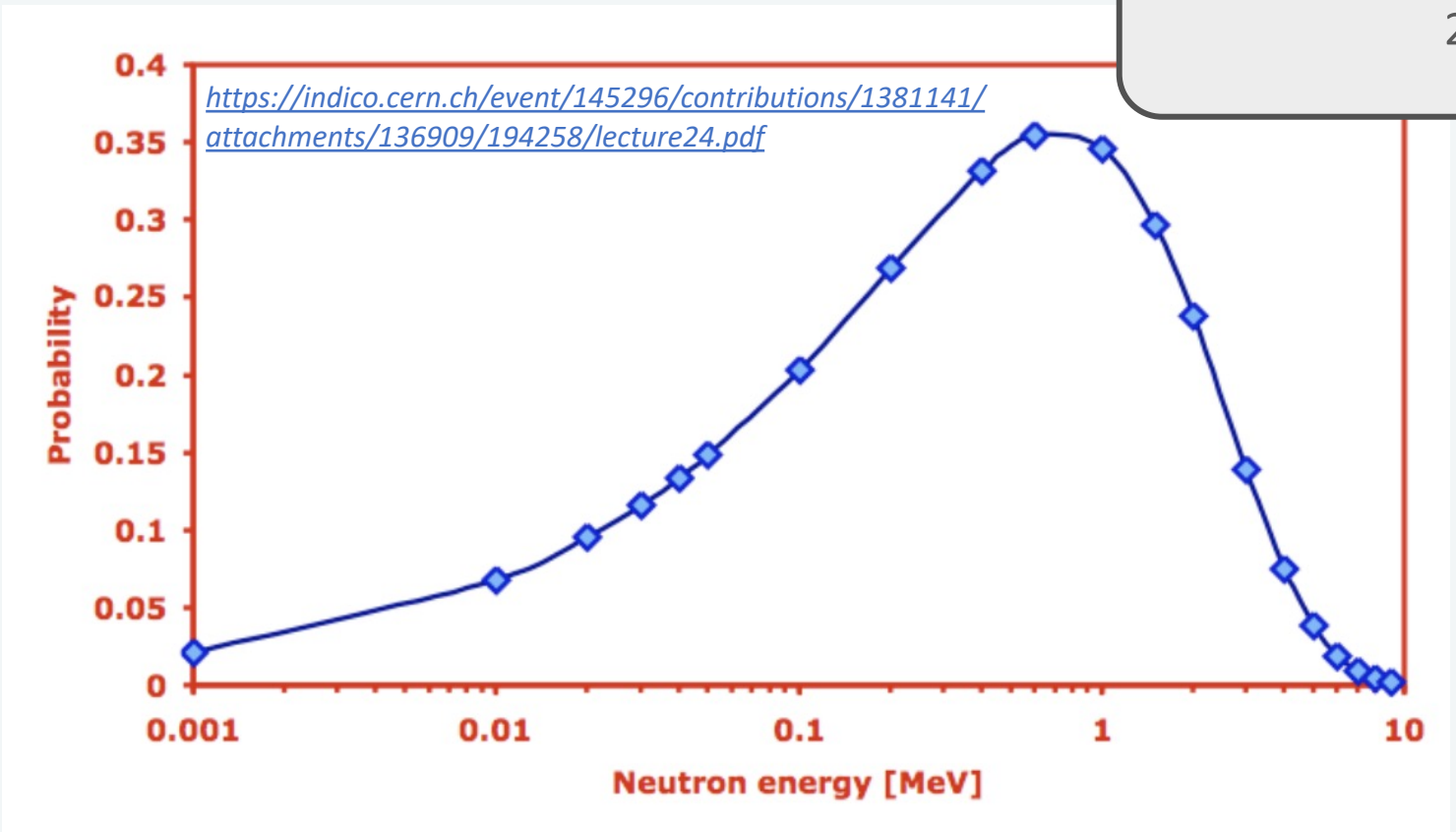
Neutron output =
total number of neutrons per second in 4π

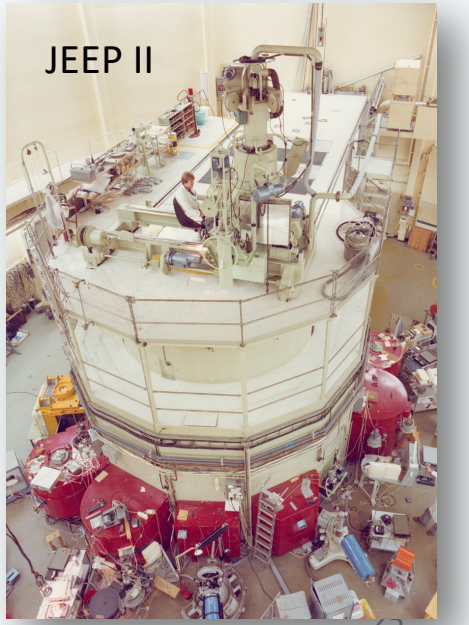
Flux is lower:
neutrons per cm^2 second

Research reactor often have beam ports that allow neutrons to escape reactor core, - consequently samples may be inserted to core for irradiation

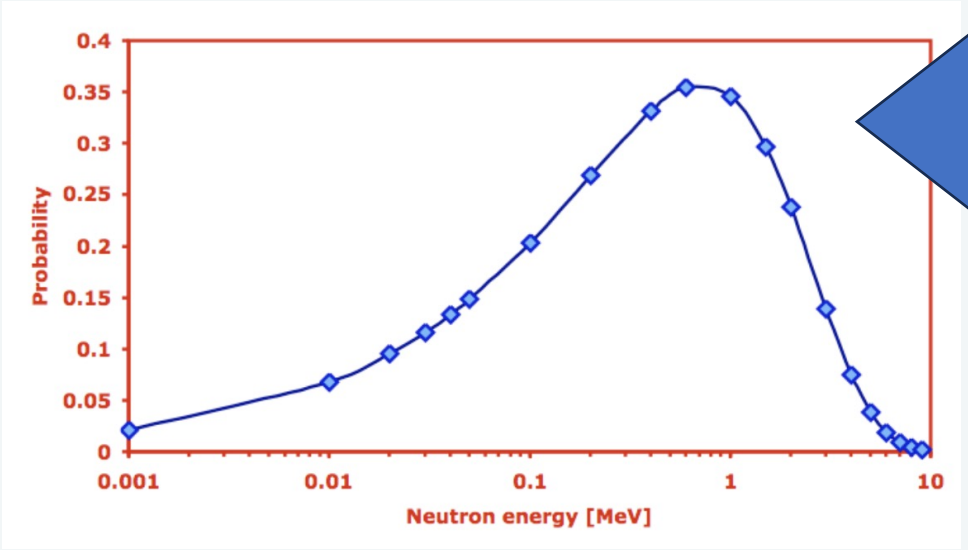


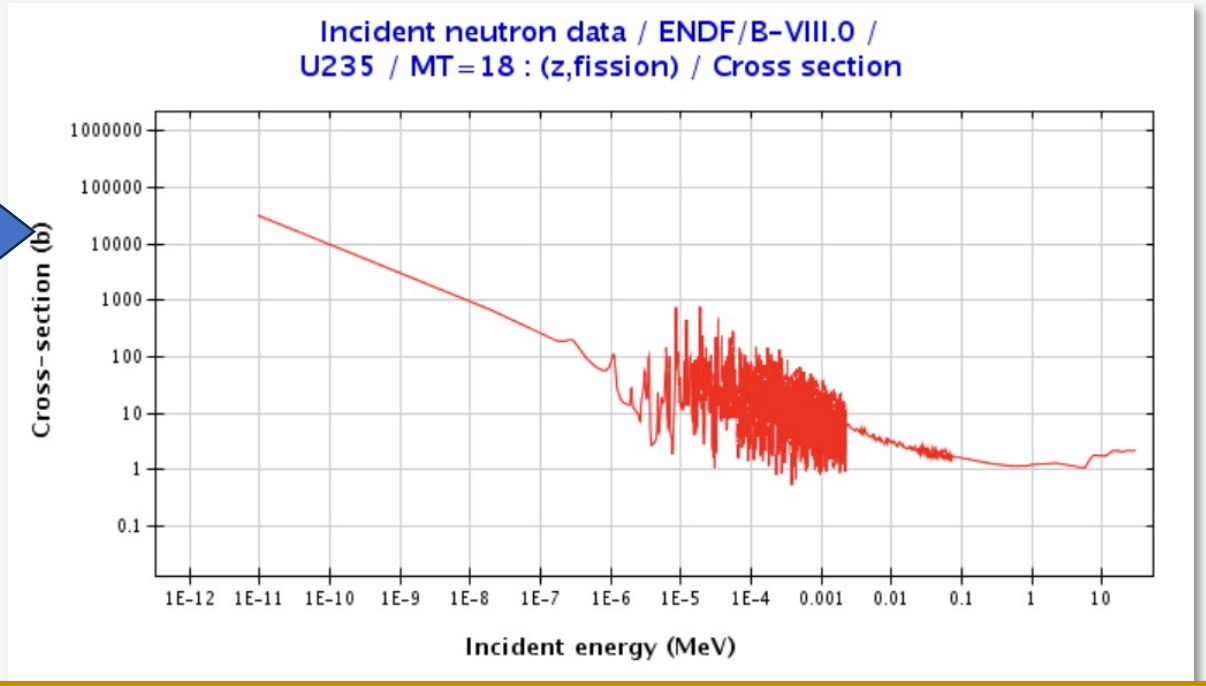
Fission Reactor Spectrum





The fission reactor neutrons needs to be moderated to sustain the fission reaction

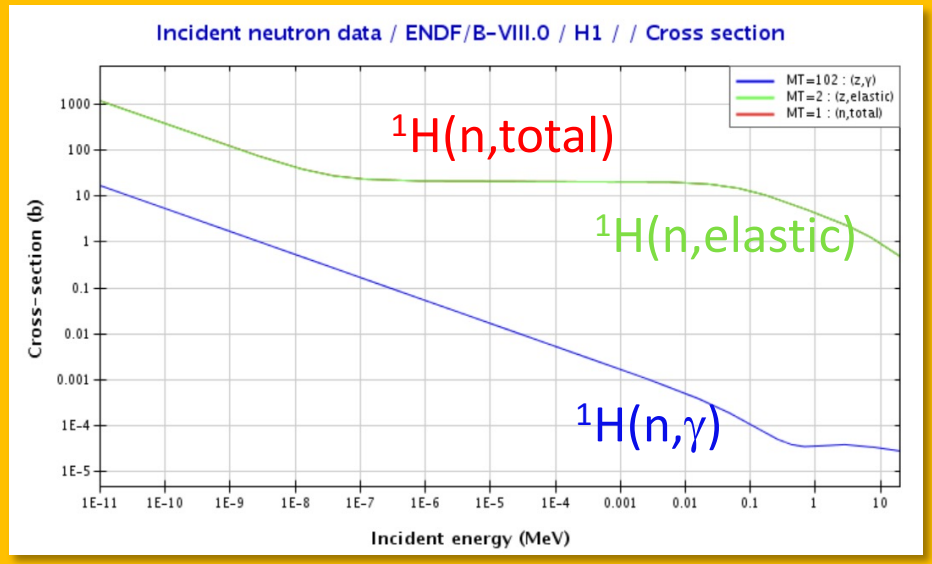




MODERATING THE NEUTRONS

Material with:

- Large scattering cross section
- Large energy loss per collision
- Small absorption cross section



Research reactor often have beam ports that allow neutrons to escape reactor core, - consequently samples may be inserted to core for irradiation

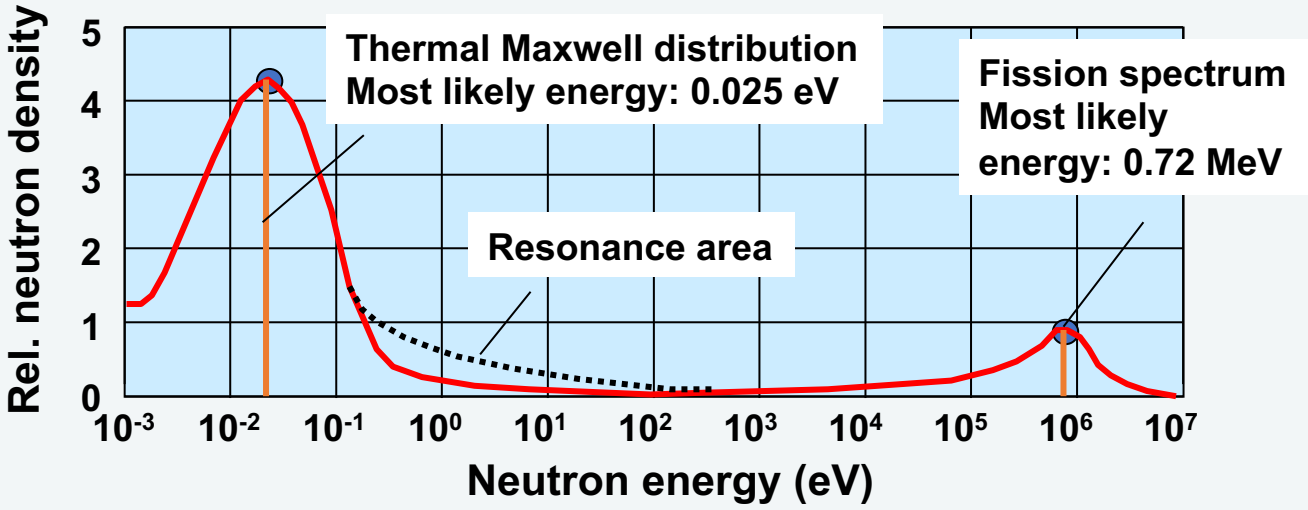
Reactors offer very large fluxes: 10^{12} – 10^{15} n /cm^2 s

Typically:
10^{13} n/cm^2 s
per 1 MW

The neutron energy distribution may vary greatly

- for different positions in a reactor
- for different reactors
- due to the materials used for moderation of the neutron energy

Energy distribution in an irradiation position may look like:



3 What is being made?

What elements are suitable for NAA?

H 1																	He
Li	Be											B 1	C	N 500	O 500	F 1000	Ne 100
Na 1.0	Mg 100											Al 10	Si 1000	P 1000	S	Cl 1.0	Ar 0.1
K 10	Ca 1000	Sc 0.1	Ti 100	V 1.0	Cr 10	Mn 0.1	Fe 100	Co 1.0	Ni 100	Cu 10	Zn 10	Ga 1.0	Ge 100	As 0.01	Se 0.1	Br 0.1	Kr 1.0
Rb 10	Sr 100	Y 100	Zr 100	Nb 1000	Mo 10	Tc	Ru 10	Rh 100	Pd 10	Ag 1.0	Cd 10	In 0.01	Sn 10	Sb 0.1	Te 0.1	I 0.1	Xe 1.0
Cs 1.0	Ba 10	La 0.1	Hf 0.1	Ta 1.0	W 0.1	Re 1.0	Os 100	Ir 0.1	Pt 10	Au 0.01	Hg 10	Tl	Pb	Bi	Po	At	Rn
Fr	Ra	Ac															
			Ce 10	Pr 1.0	Nd 100	Pm	Sm 0.1	Eu 1.0	Gd 10	Tb 1.0	Dy 0.1	Ho 1.0	Er 10	Tm 1	Yb 0.1	Lu 0.01	
			Th	Pa	U	Np	Pu	Am	Cm	Bk	Cf	Es	Md	Fm	No	Lr	

Fast Neutron Activation Analysis
Prompt Gamma Activation Analysis
Thermal Neutron Activation Analysis
Not done by NAA
*Numbers represent ppm

https://www.nist.gov/laboratories/tools-instruments/instrumental-neutron-activation-analysis-inaa

https://www.oecd-nea.org/jcms/pl_39910/janis

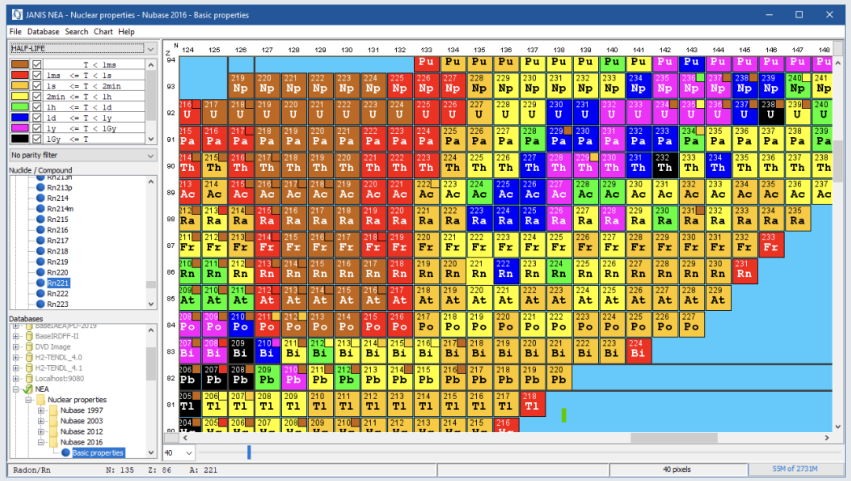
https://www.oecd-nea.org/jcms/pl_39910/janis

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Comparison of experimental and evaluated cross-sections.

n - neutron-induced

γ - gamma-induced

p - proton-induced

d - deuteron-induced

t - triton-induced

h -helion-induced

α - alpha-induced

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ENDF search

Criteria

Incident particle: N

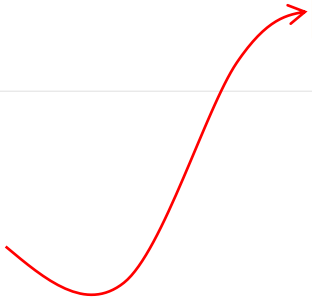
Library: ENDF/B-VIII.0

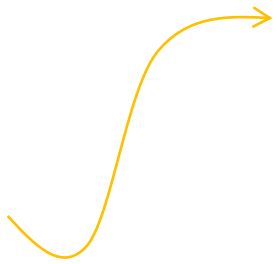
Target: Z: 1 (H) Hydrogen | A: 2 | State:

MF: MF=3 : Reaction cross sections

MT:

Search | Reset





Choose which ones to plot

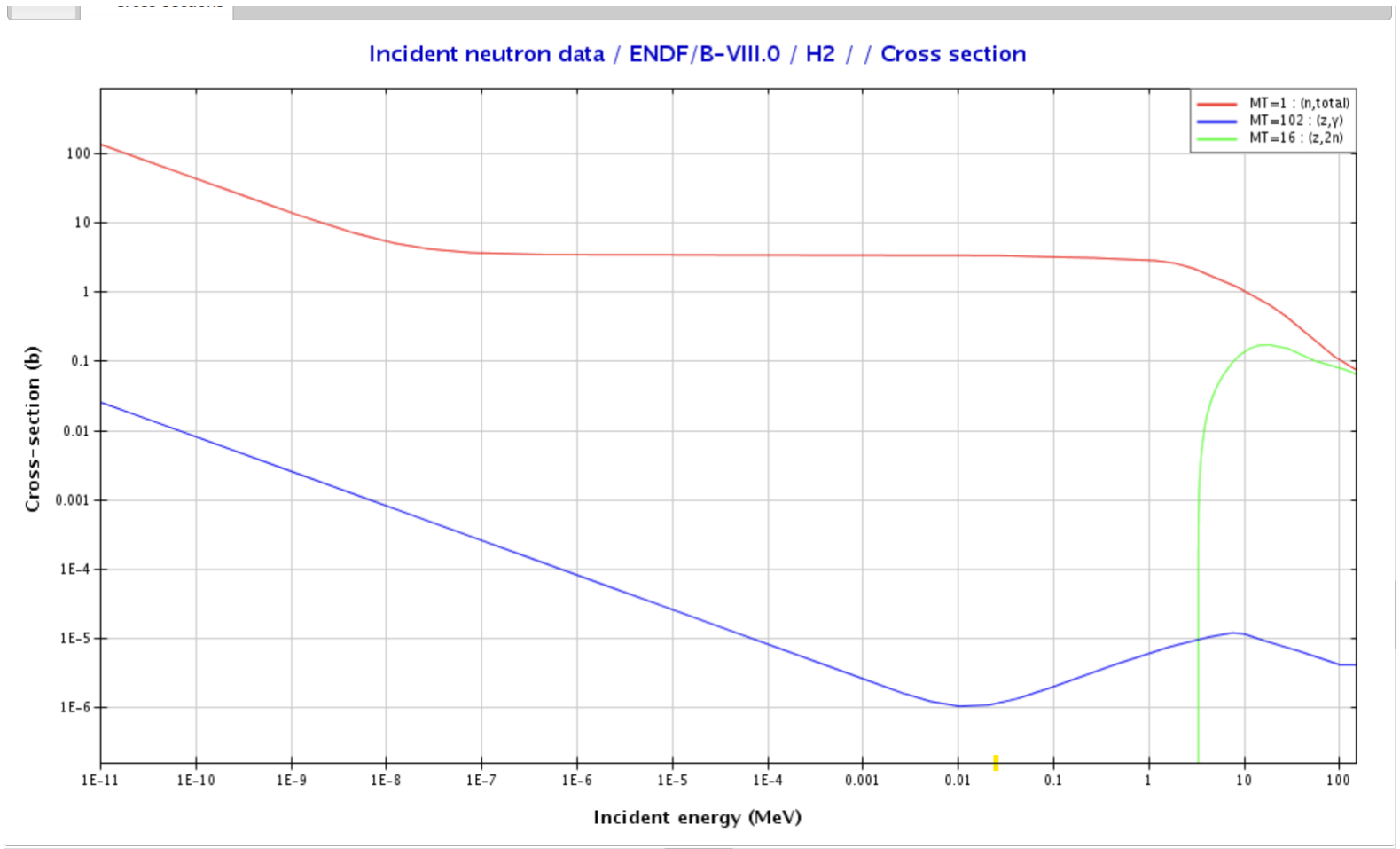
ENDF search

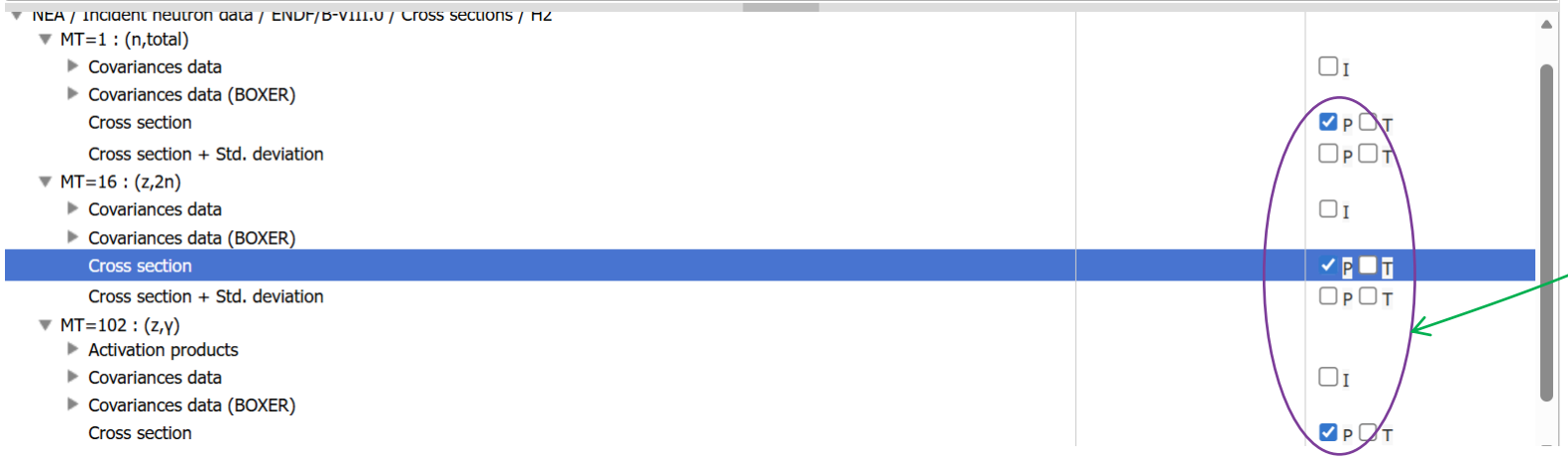
Criteria Z=1 A=2, Library=ENDF/B-VIII.0, Particle=N, MF=3

19 results

Prev. Next [1]

	Incident particle	Evaluation	Material	MF	MT
☑	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=1 : (n,total)
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=2 : (z,elastic)
☑	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=16 : (z,2n)
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=102 : (z,γ)
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=203 : (z,Xp)
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=205 : (z,Xt)
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=221 : free-gas scattering
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=301 : KERMA total (energy balance)
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=302 : KERMA elastic
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=303 : KERMA non-elastic
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=304 : KERMA inelastic
☐	Incident neutron data	ENDF/B-VIII.0	H2	3	MT=318 : KERMA fission





Select for plotting

PGNAA:

- Energies from 45 keV to 12 MeV
- Poisson statistics during measurement

DGNAA:

- Energies from 15 keV to 2.8 MeV
- Non-Poisson statistics during measurement, due to decay

BOTH:

- Baseline increase toward low energies
- Peak position to identify element
- Peak area to determine quantity

https://www-nds.iaea.org/pgaa/pgaa7/index.html

International Atomic Energy Agency

PGAA

Prompt Gamma-ray Neutron Activation Analysis

This database has been developed as part of a Coordinated Research Project for the Development of a Database for Prompt Gamma-ray Neutron Activation Analysis sponsored by the International Atomic Energy Agency (IAEA). The data are derived from the isotopic measurements compiled in the Evaluated Nuclear Structure Data File (ENSDF), updated when necessary, and from elemental measurements performed at the Budapest Reactor Centre. The data were received from Richard B. Firestone (Lawrence Berkeley National Laboratory, USA). The program "Viewer" was developed by Viktor Zerkin (IAEA, NDS).

PGAA: Target Elements and Isotopes

Selected Element

6-Carbon (13)
C-12 (6)
C-13 (7)

1 H																	2 He
3 Li	4 Be												5 B	6 C	7 N	8 O	9 F	10 Ne
11 Na	12 Mg												13 Al	14 Si	15 P	16 S	17 Cl	18 Ar
19 K	20 Ca	21 Sc	22 Ti	23 V	24 Cr	25 Mn	26 Fe	27 Co	28 Ni	29 Cu	30 Zn	31 Ga	32 Ge	33 As	34 Se	35 Br	36 Kr
37 Rb	38 Sr	39 Y	40 Zr	41 Nb	42 Mo	43 Tc	44 Ru	45 Rh	46 Pd	47 Ag	48 Cd	49 In	50 Sn	51 Sb	52 Te	53 I	54 Xe
55 Cs	56 Ba	57* La	72 Hf	73 Ta	74 W	75 Re	76 Os	77 Ir	78 Pt	79 Au	80 Hg	81 Tl	82 Pb	83 Bi	84 Po	85 At	86 Rn
87 Fr	88 Ra	89** Ac	104 Rf	105 Db	106 Sg	107 Bh	108 Hs	109 Mt	110 *	111 *	112 *						
		* Lanthanides		58 Ce	59 Pr	60 Nd	61 Pm	62 Sm	63 Eu	64 Gd	65 Tb	66 Dy	67 Ho	68 Er	69 Tm	70 Yb	71 Lu
		** Actinides		90 Th	91 Pa	92 U	93 Np	94 Pu	95 Am	96 Cm	97 Bk	98 Cf	99 Es	100 Fm	101 Md	102 No	103 Lr

4 Quantitative analysis DGNNA

Given a stable nuclide in a sample is exposed to neutrons

$$ {}^{A}_{Z}X + n \longrightarrow {}^{A+1}_{Z}X + \gamma $$



Each nuclide may capture a neutron and produce a radioactive isotope of the element…



The reaction rate is given by:

$$R = \sigma \cdot \phi \cdot N_t$$

σ – cross section [cm^2]

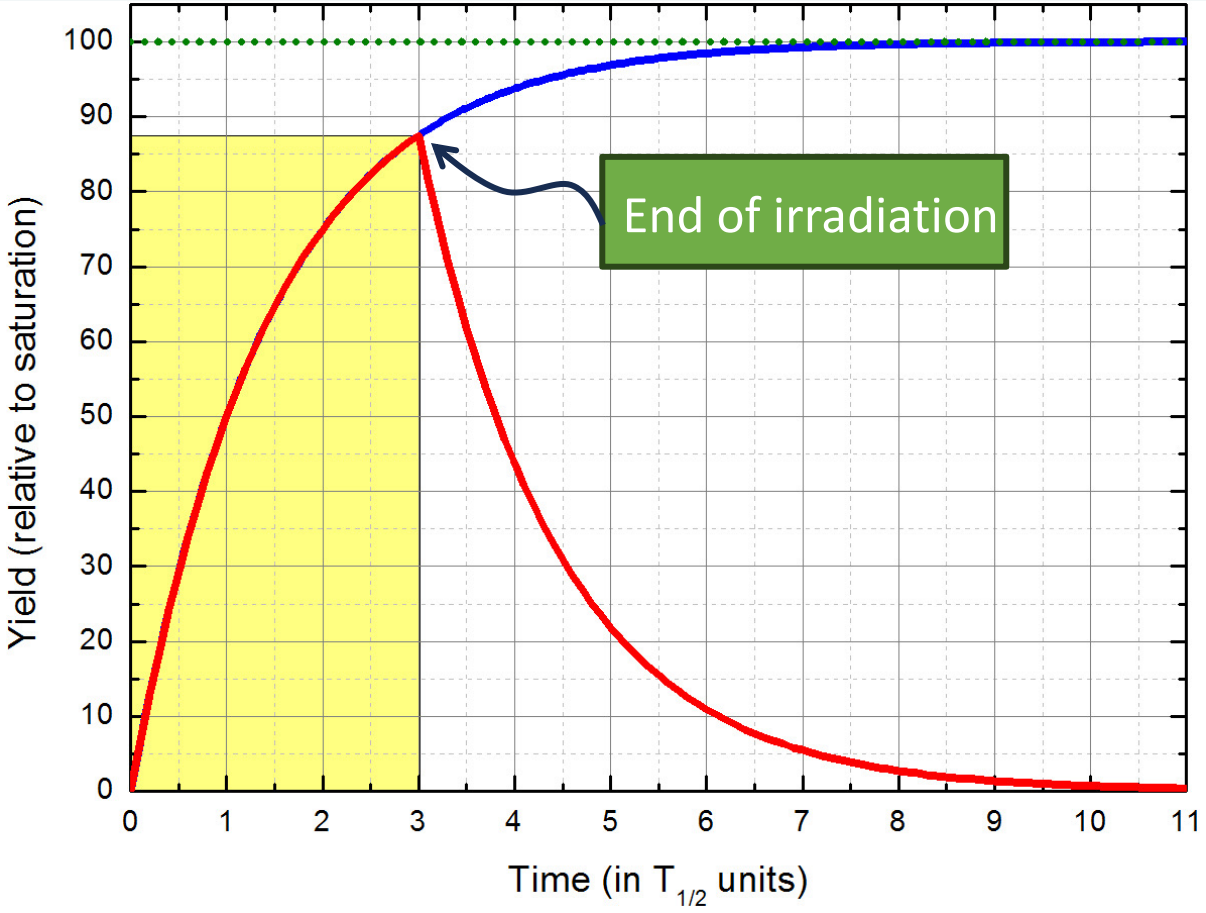
ϕ – particle flux [particles per cm^2 per s]

N_T – number of target atoms [atoms]

Dimension R: [atoms /s]

Note: The target is ***smaller*** than the area filled by the incident particles

Reaction yield = Rate of production – Rate of decay



$$\frac{dN^*}{dt} = N\phi\sigma - \lambda N^*$$

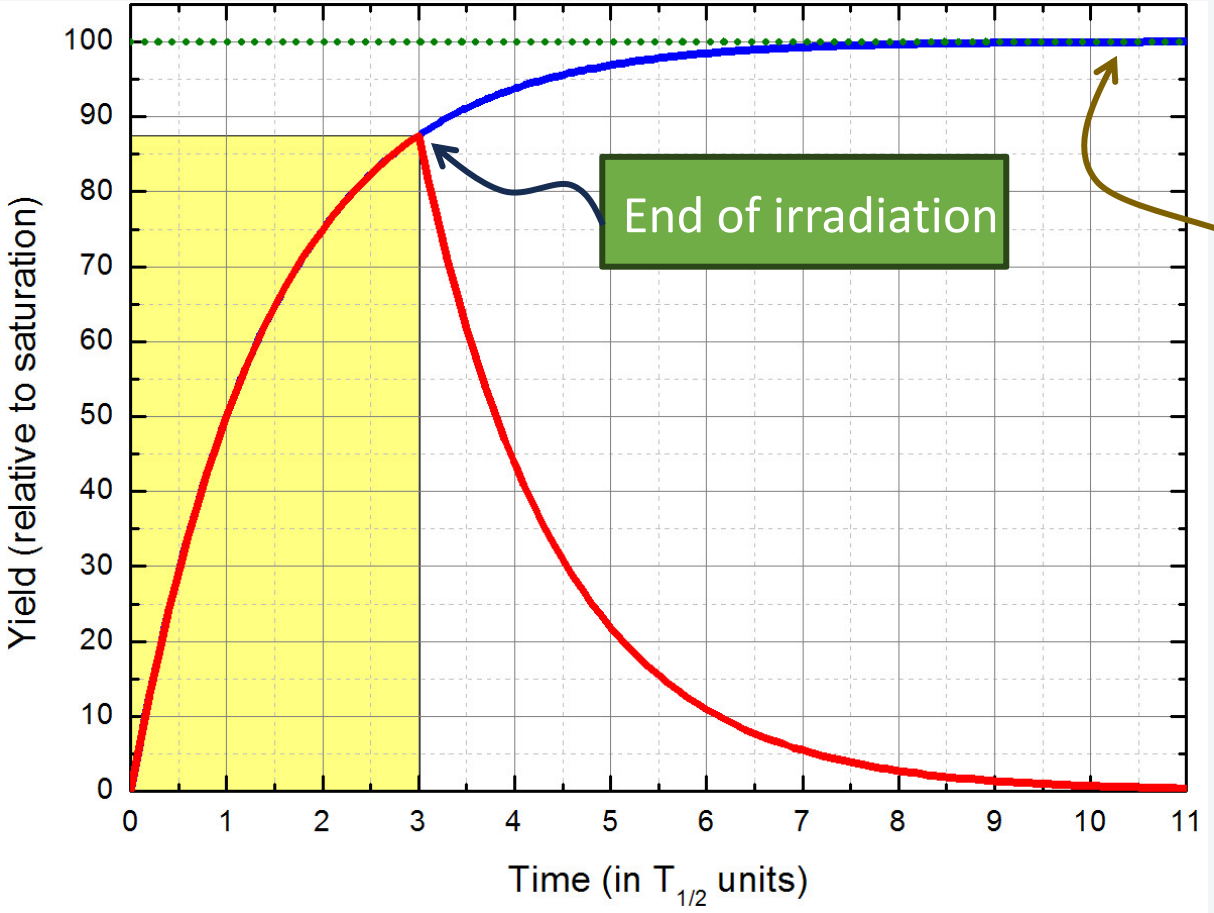
At t = 0, N = 0*

$$N^*(t) = \frac{N\phi\sigma}{\lambda}(1 - \exp(-\lambda t))$$

Activity at time, t_{irr}, is given by
$A(t_{irr}) = \lambda N(t_{irr})$

$$A(t_{irr}) = N\phi\sigma\,(1 - \exp(-\lambda t_{irr}))$$

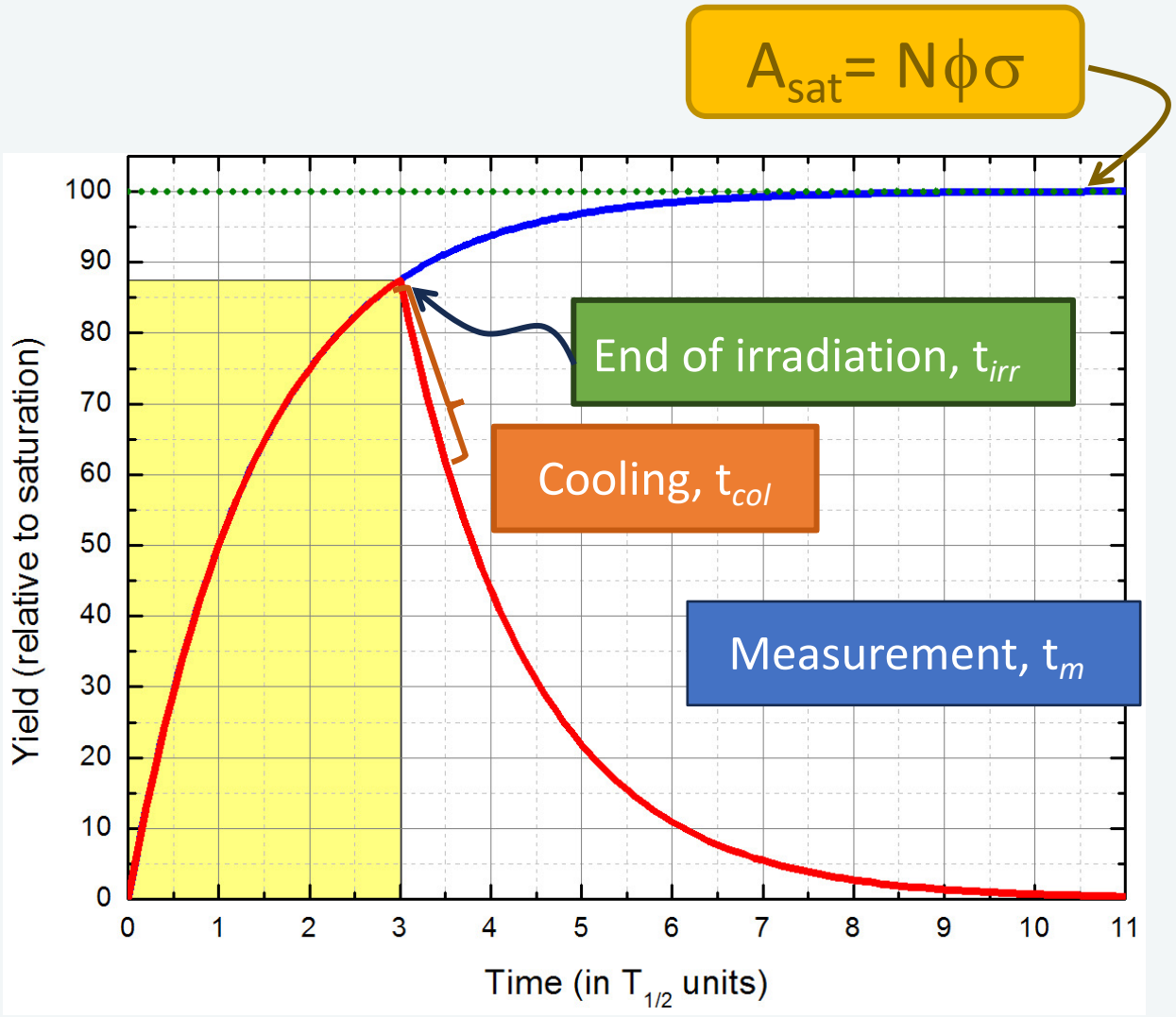
Reaction yield = Rate of production – Rate of decay



$$A(t_{irr}) = N\phi\sigma\,(1 - \exp(-\lambda t_{irr}))$$

$$t_{irr} \rightarrow \infty \Rightarrow A(t) = N\phi\sigma = A_{sat}$$

$$A_{sat} = N\phi\sigma$$



Cooling: $A(t_{col}) = A(t_{irr})\ \exp(-\lambda t_{col})$

Activity at start of measurement:

$$A(t_{col}) = N\phi\sigma\ (1 - \exp(-\lambda t_{irr}))\cdot\exp(-\lambda t_{col})$$

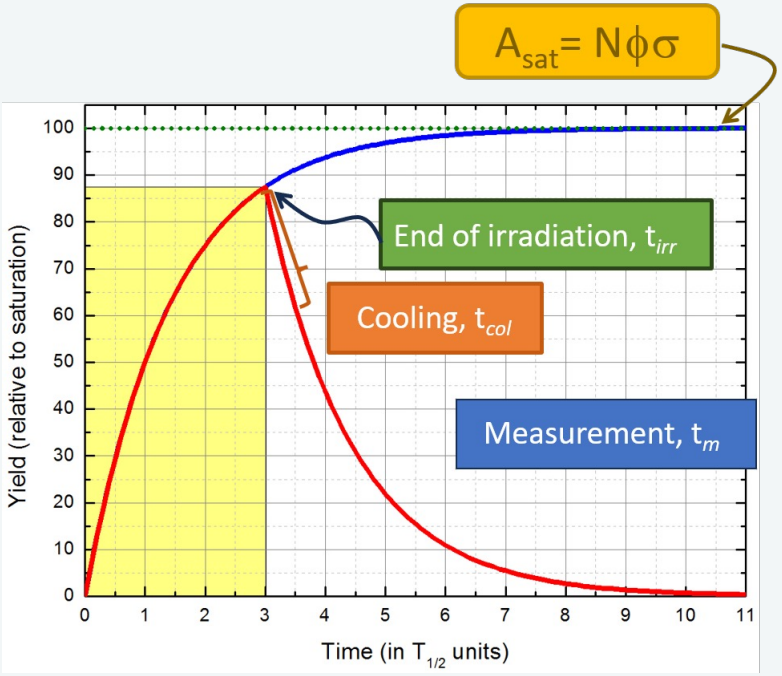
Activity decay during measurement time, t_m:

$$A_{int} = \int_0^{t_m} A(t_{col})\ \exp(-\lambda t)\ dt$$

$$A_{int} = \int_0^{t_m} A(t_{col}) \exp(-\lambda t)\, dt$$

$$A_{int} = N\phi\sigma\,(1 - \exp(-\lambda t_{irr})) \cdot \exp(-\lambda t_{col}) \int_0^{t_m} \exp(-\lambda t)\, dt$$

$$A = N\phi\sigma\,(1 - \exp(-\lambda t_{irr})) \cdot \exp(-\lambda t_{col}) \cdot \frac{1 - \exp(-\lambda t_m)}{\lambda}$$



We can then relate what we measure on our gamma-ray detector to this activity

Measured net count, N_p: $N_p = A_{int} \cdot \varepsilon \cdot I_\gamma$

ε – detector efficiency at γ-energy

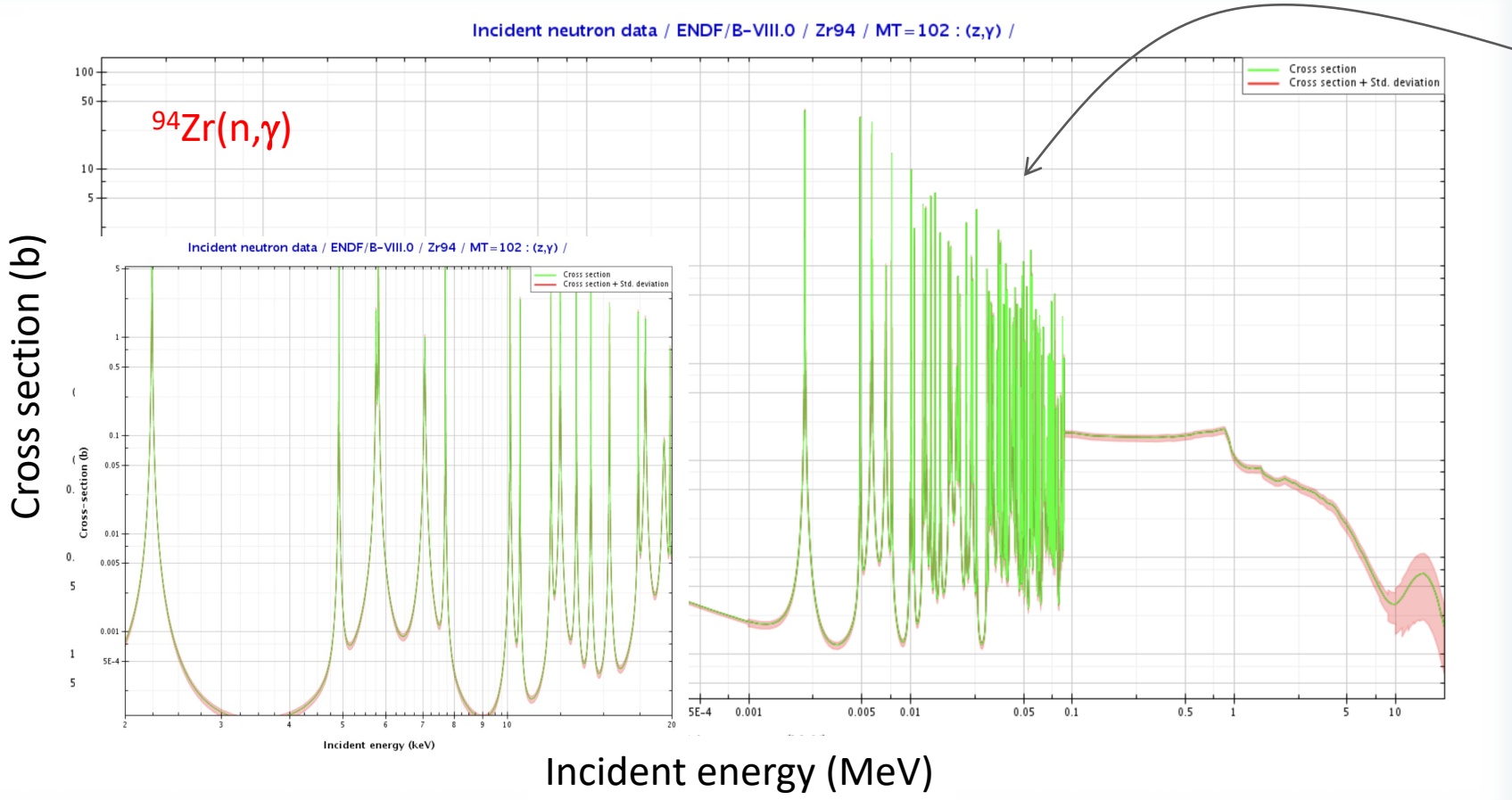
I_γ – γ-line emission probability

$$N_p = N\phi\sigma\,(1 - \exp(-\lambda t_{irr})) \cdot \exp(-\lambda t_{col}) \cdot \frac{1 - \exp(-\lambda t_m)}{\lambda} \cdot \varepsilon \cdot I_\gamma$$

$A_{sat} = N\phi\sigma$ So, we can solve for A_{sat}:

$$A_{sat} = \frac{N_p \lambda}{(1 - \exp(-\lambda t_{irr})) \cdot \exp(-\lambda t_{col}) \cdot \varepsilon \cdot I_\gamma \cdot (1 - \exp(-\lambda t_m))}$$

Absolute determination of the concentration/amount of an element of interest is challenging due to large uncertainties in ϕ and σ.



Large variations in the cross sections

How well do we know the cross sections?

How well is the neutron energy spectrum of the source described?

Absolute determination of the concentration/amount of an element of interest is challenging due to large uncertainties in ϕ and σ.

Relative standardization

Irradiation of calibration standard together with unknown sample

The calibration standard contains a known amount of the element of interest

Comparison to a certified reference material, e.g. gold (Au)

Experimentally determined k- factors

$$k_0 = \frac{M_{Au} \cdot \gamma_a \cdot \theta_a \cdot \sigma_a}{M_a \cdot \gamma_{Au} \cdot \theta_{Au} \cdot \sigma_{Au}}$$

M = *atomic mass*
γ = *g-line probability*
θ = *isotopic abundance*
σ = *cross section*

Singel comparator, k_0 method

I have many elements of interest!!!

5 Some examples of macro-object applications

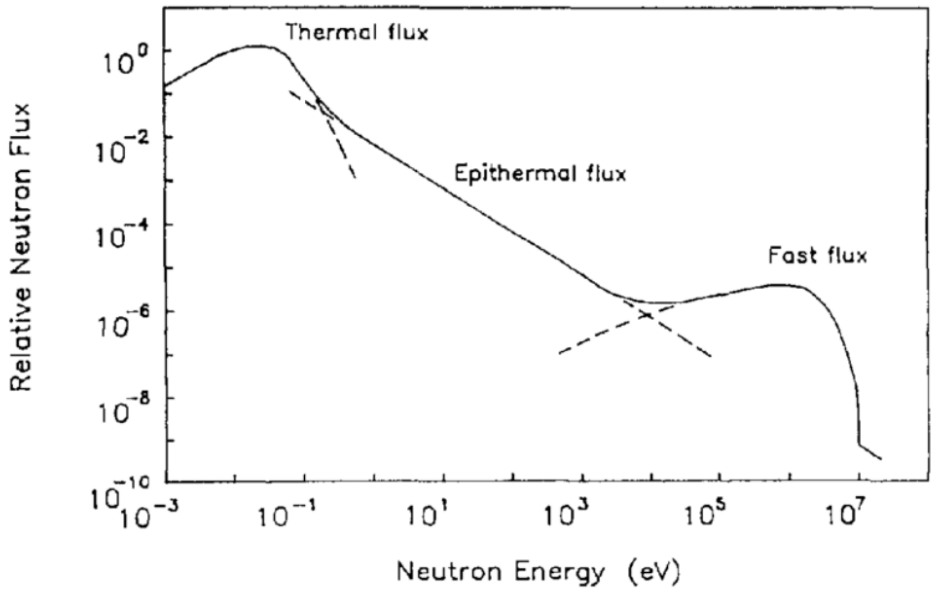
A complex matrix, and I am interested in a trace element, what to do?

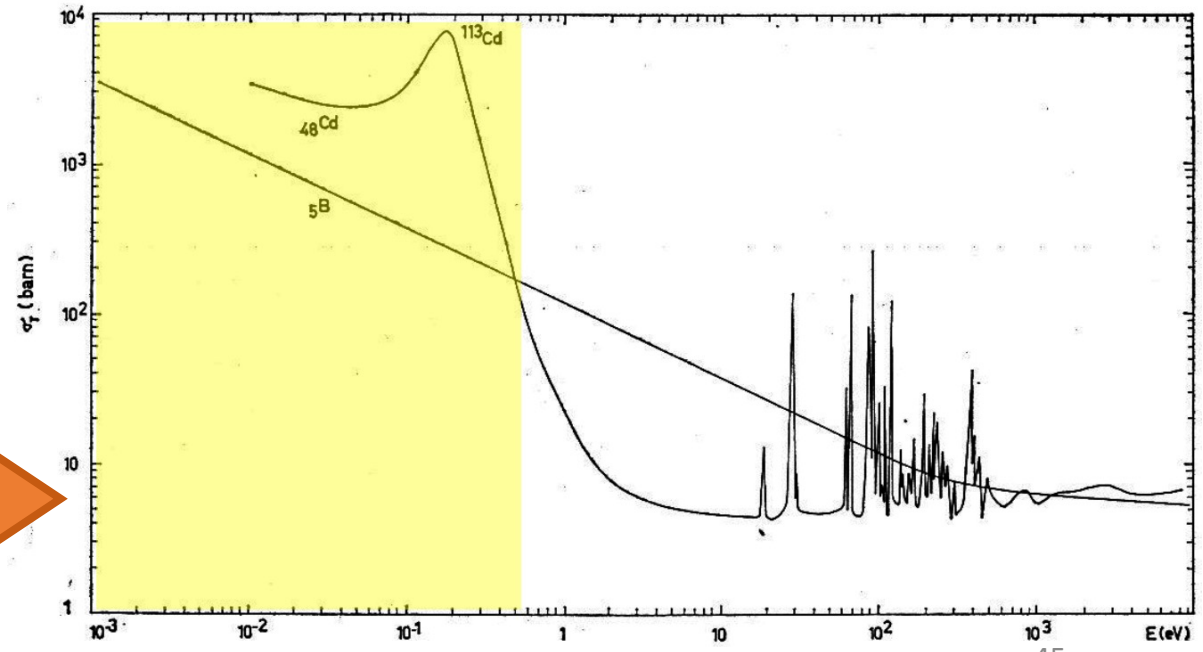
Epithermal NAA

Eliminate the response to thermal neutrons:

when samples contain lot of elements which have strong thermal response and overshadows the response from trace elements of interest!

Cadmium (Cd) and boron (B) have large absorption of thermal neutrons and small cross section for epithermal



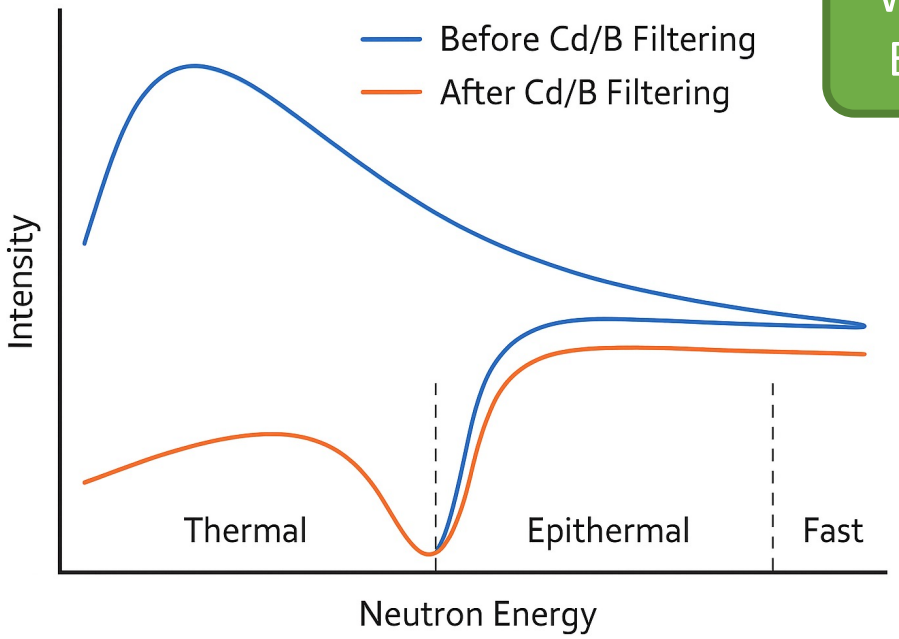


Some elements have large resonance integrals: easier to detect when thermal neutrons are filtered out

https://www.osti.gov/etdeweb/servlets/purl/20332542

The resonance integral, I_0 is the reaction probability integrated over epithermal range (0.5 eV – 10 keV)

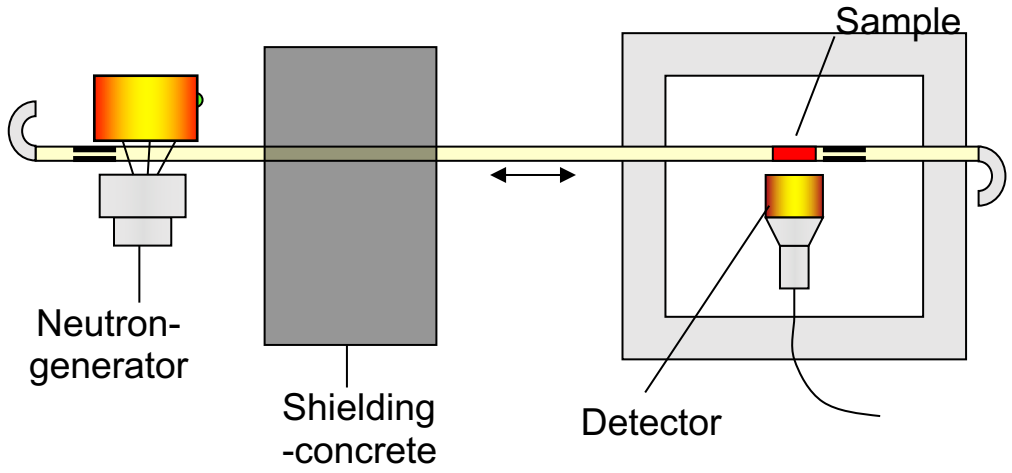
When $I_0 >> \sigma_{th}$ ENAA is useful!



Nuclide	Thermal cross section (barn)	Resonance Integral (barn)
^{79}Br	10.3	127
^{81}Br	2.4	50
^{121}Sb	5.9	205
^{123}Sb	4.0	127
^{75}As	4.2	61
^{127}I	6.2	150
^{136}Ce	6.5	77
^{197}Au	99	1550
^{232}Th	7.4	85
^{238}U	2.7	277

I want to analyze some elements not suitable for conventional NAA!
– Cyclic irradiation with fast neutrons

Cyclic activation for *short half-lives* (s, min) for acceptable count statistics. Number of cycles and their lengths must be carefully chosen: - over time the background will increase.



$^{18}O(n,p)^{16}N$

$T_{1/2}$ = 7,13 s

E_γ = 6,1 og 7,1 MeV

n_{th} 0,14 mb



$^{207}Pb(n,n')^{207m}Pb$

$T_{1/2}$ = 0,803 s

E_γ = 569,7 og 1064 keV

n_{th} = 0.54 b

n_{th} = 0,23 mb



Neutron backscatter (back diffusion) – some uses with portable sources

Hydrogen are efficient in slowing down neutrons. Thermalization in water is in the order of μs

(1) Content of water in soil

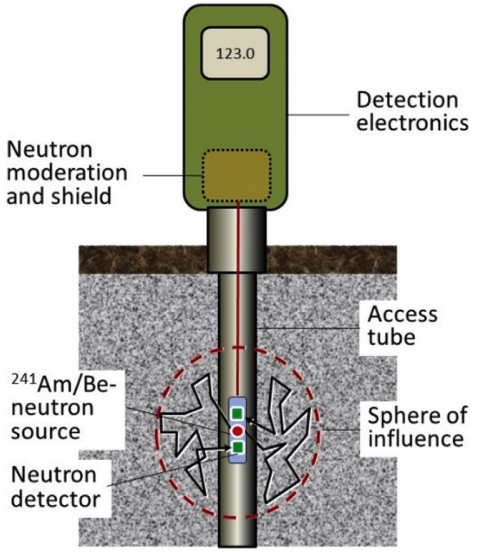
(2) Road integrity

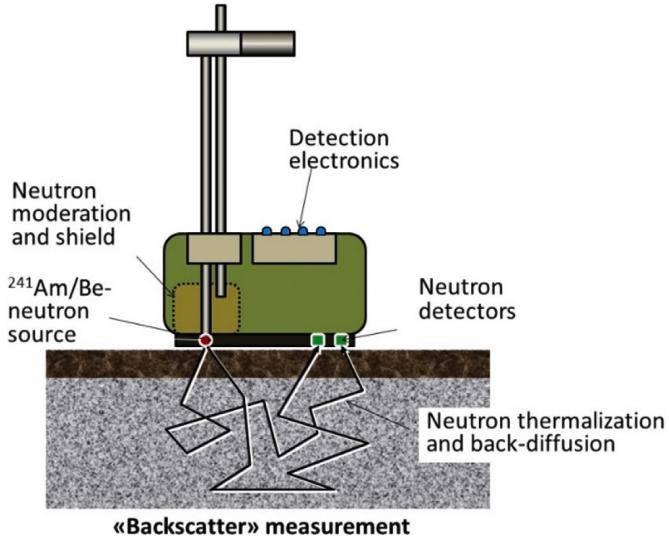
(3) Well logging (oil)

Equipped with detectors sensitive for thermal neutrons, ^{3}He, BF_3

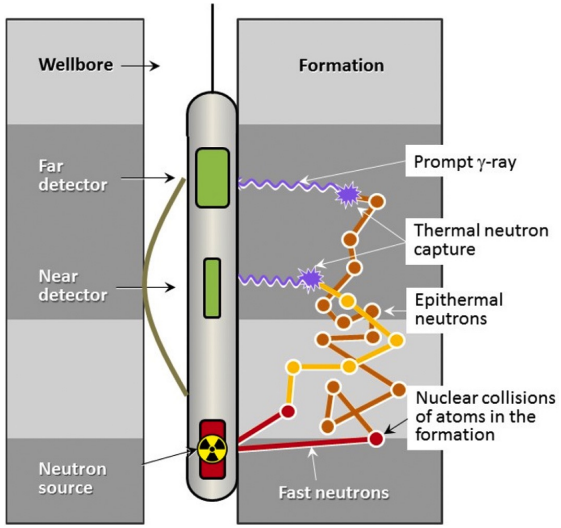
$$^{3}He + n_{th} \rightarrow {}^{3}H\ (191\ keV) + p\ (573\ keV)$$

$$^{10}B + n_{th} \rightarrow {}^{4}He\ (1.47\ MeV) + {}^{7}Li\ (0.84\ MeV) + \gamma\ (0.48\ MeV)$$





«Backscatter» measurement



Si and Al - bulk analysis of red mud wastes

X-ray techniques like EDXRF are often chosen for bulk analysis.

Not good for light elements.

Si, Al, Na, Mg: self absorption and peak overlap with background

Fast NAA

$^{27}Al(n,p)^{27}Mg$

$T_{1/2}$ = 9.44 min
E_γ = 0.84 and 1.01 MeV

$^{28}Si(n,p)^{28}Al$

$T_{1/2}$ = 2.25 min
E_γ = 1.78 MeV

$^{29}Si(n,p)^{29}Al$

$T_{1/2}$ = 6.56 min
E_γ = 1.28 MeV

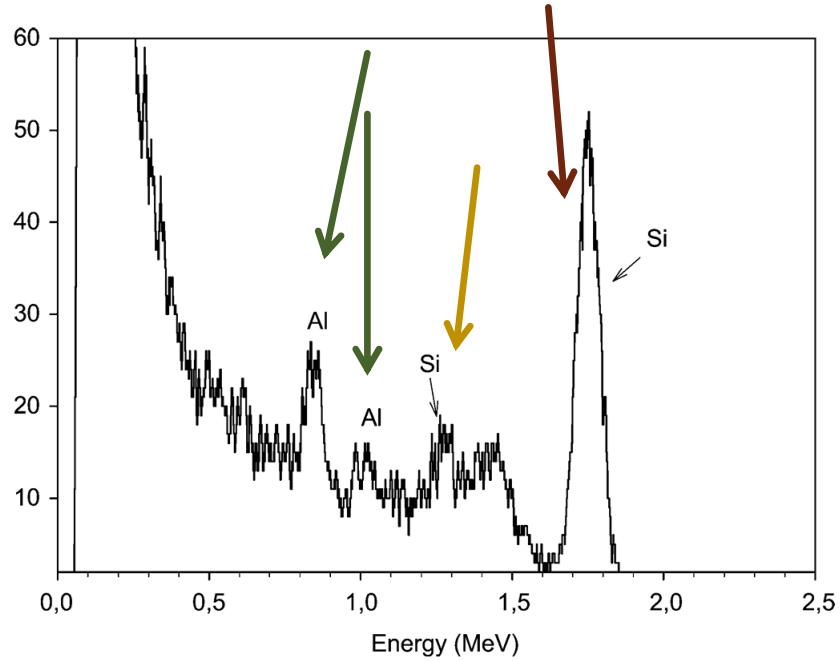


Fig. 3. Gamma-ray spectrum of RM obtained by the FNAA. Ordinate presents counts.

Gamma-ray spectrum after 1-h irradiation of 50 g red mud with 14 MeV 10^7 n/s

From: OBHONDAS et. al. 2011
doi:10.1109/ANIMMA.2011.6172957

PGNAA with fast neutrons for identification of explosives and illicit materials

Neutron generator

Projectile with mustard gas. Airtight steel container



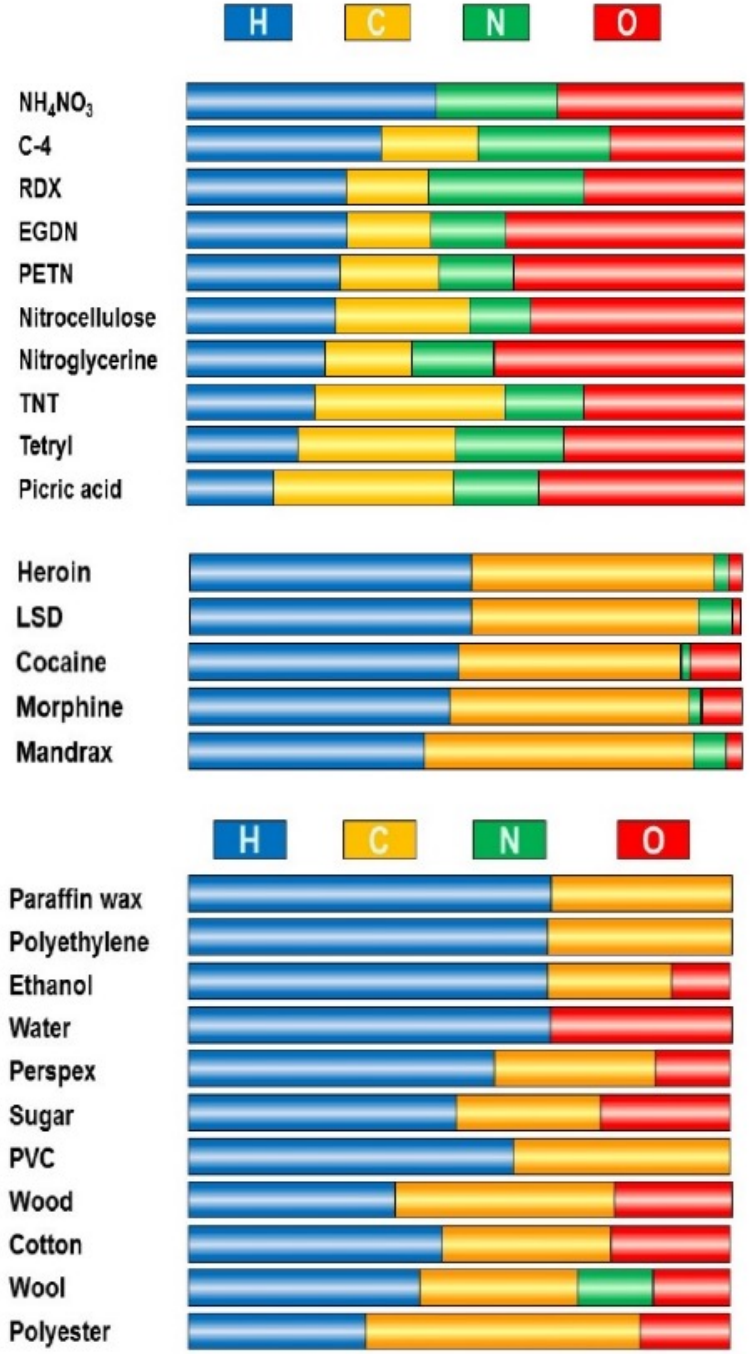
Gamma-ray detector

Evaluation of the H/C/N/O ratio:

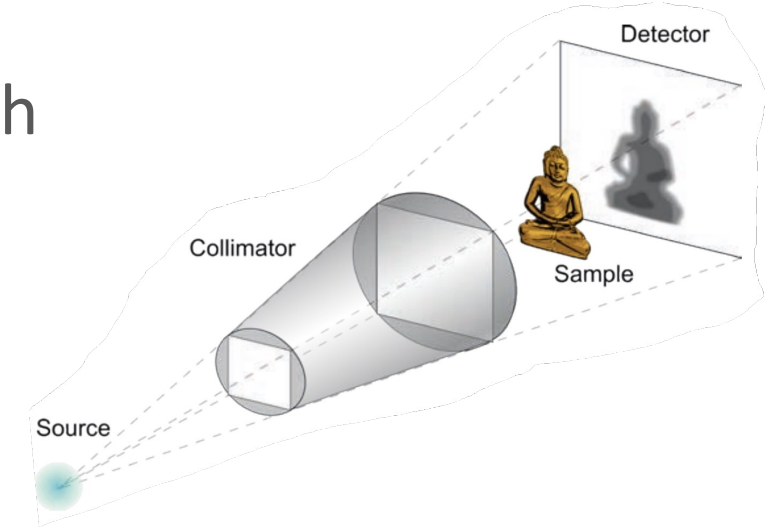
Rection	Eγ (MeV)	σ (barn)
$^{12}C(n,n'\gamma)$	4.44	0.42
$^{14}N(n,n'\gamma)$	2.3 & 5.1	0.39
$^{16}O(n,n'\gamma)$	6.13	0.31

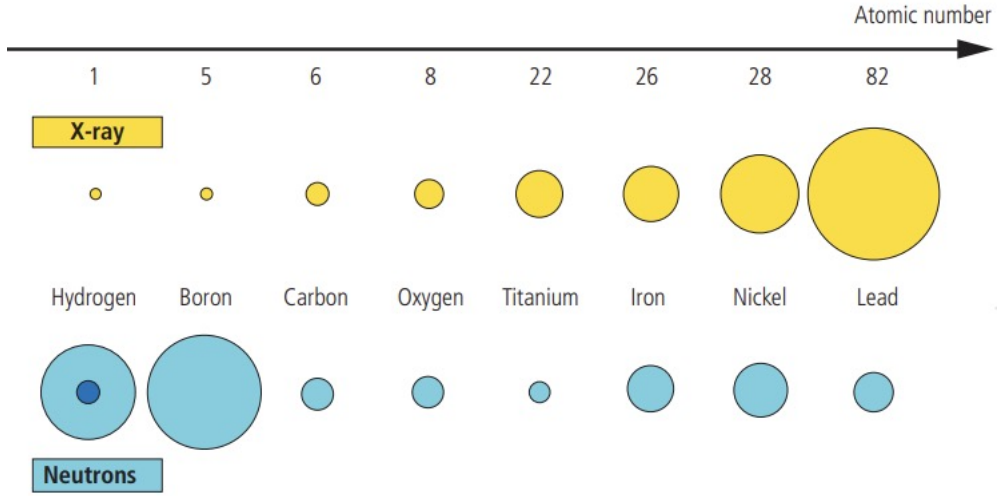
U.S. Army Dugway Proving Ground, 2003,

T. R. Twomey et al, INL/CON-07-12304, 2007



Imaging with neutron







X-ray image



Neutron image

Attenuation coefficient Neutrons (25meV) [cm^{-1}]

1a	2a	3b	4b	5b	6b	7b	8			1b	2b	3a	4a	5a	6a	7a	0
H 3.44																	He 0.02
Li 3.30	Be 0.79											B 101.60	C 0.56	N 0.43	O 0.17	F 0.20	Ne 0.10
Na 0.09	Mg 0.15											Al 0.10	Si 0.11	P 0.12	S 0.06	Cl 1.33	Ar 0.03
K 0.06	Ca 0.08	Sc 2.00	Ti 0.60	V 0.72	Cr 0.54	Mn 1.21	Fe 1.19	Co 3.92	Ni 2.05	Cu 1.07	Zn 0.35	Ga 0.49	Ge 0.47	As 0.67	Se 0.73	Br 0.24	Kr 0.61
Rb 0.08	Sr 0.14	Y 0.27	Zr 0.29	Nb 0.40	Mo 0.52	Tc 1.76	Ru 0.58	Rh 10.88	Pd 0.78	Ag 4.04	Cd 115.11	In 7.58	Sn 0.21	Sb 0.30	Te 0.25	I 0.23	Xe 0.43
Cs 0.29	Ba 0.07	La 0.52	Hf 4.99	Ta 1.49	W 1.47	Re 6.85	Os 2.24	Ir 30.46	Pt 1.46	Au 6.23	Hg 16.21	Tl 0.47	Pb 0.38	Bi 0.27	Po	At	Rn
Fr	Ra 0.34	Ac	Rf	Ha													

*Lanthanides	Ce 0.14	Pr 0.41	Nd 1.87	Pm 5.72	Sm 171.47	Eu 94.58	Gd 1479.04	Tb 0.93	Dy 32.42	Ho 2.25	Er 5.48	Tm 3.53	Yb 1.40	Lu 2.75
**Actinides	Th 0.59	Pa 8.46	U 0.82	Np 9.80	Pu 50.20	Am 2.86	Cm	Bk	Cf	Es	Fm	Md	No	Lr neut.

Legend

$$\text{Attenuation coefficient [cm??]} = \frac{\sigma\text{-total} * \text{sp.gr.} * 0.6023}{\text{at.wt.}}$$

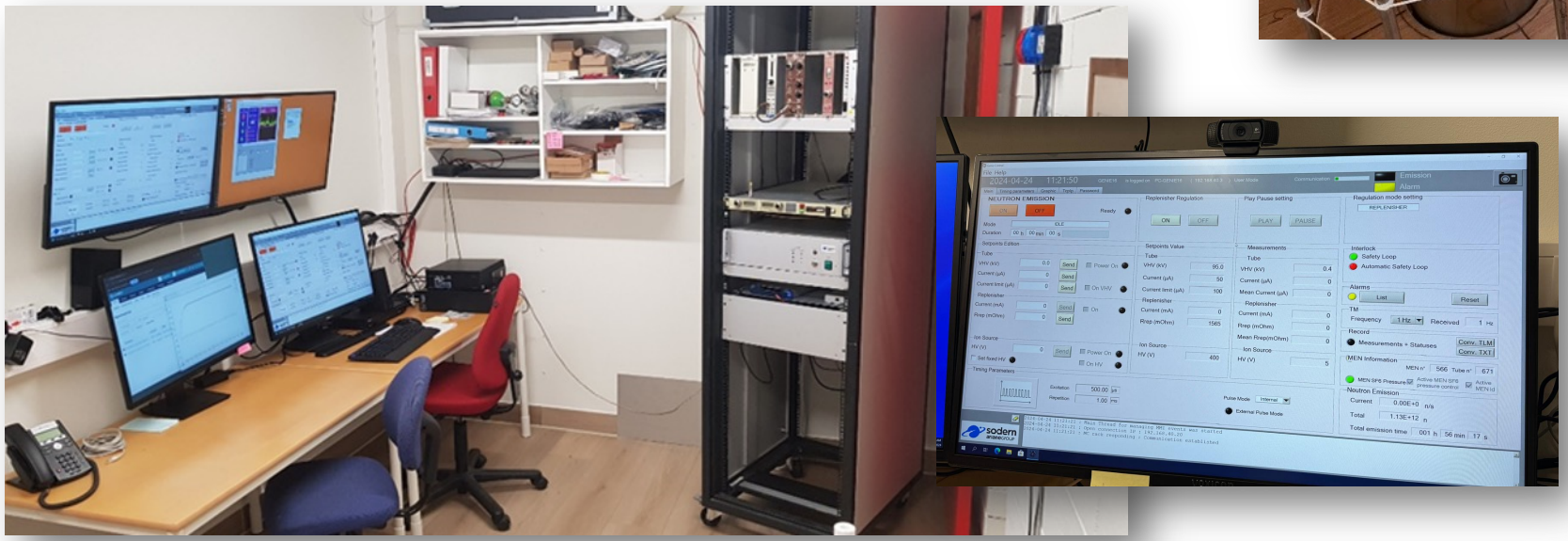
σ-total: JEF Report 14, TABLE OF SIMPLE INTEGRAL NEUTRON CROSS SECTION DATA FROM JEF-2.2, ENDF/B-VI, JENDL-3.2, BROND-2 AND CENDL-2, AEN NEA, 1994.
and Special Feature: Neutron scattering lengths and cross sections, Varley F. Sears, AECL Research, Chalk River Laboratories Chalk River, Ontario, Canada KOJ 1JO, Neutron News, Vol. 3, 1992, http://www.ncnr.nist.gov/resources/n-lengths/list.html.
sp.gr.: Handbook of Chemistry and Physics, 56th Edition 1975-1976.
at.wt.: Handbook of Chemistry and Physics, 56th Edition 1975-1976.

6 Laboratory exercise

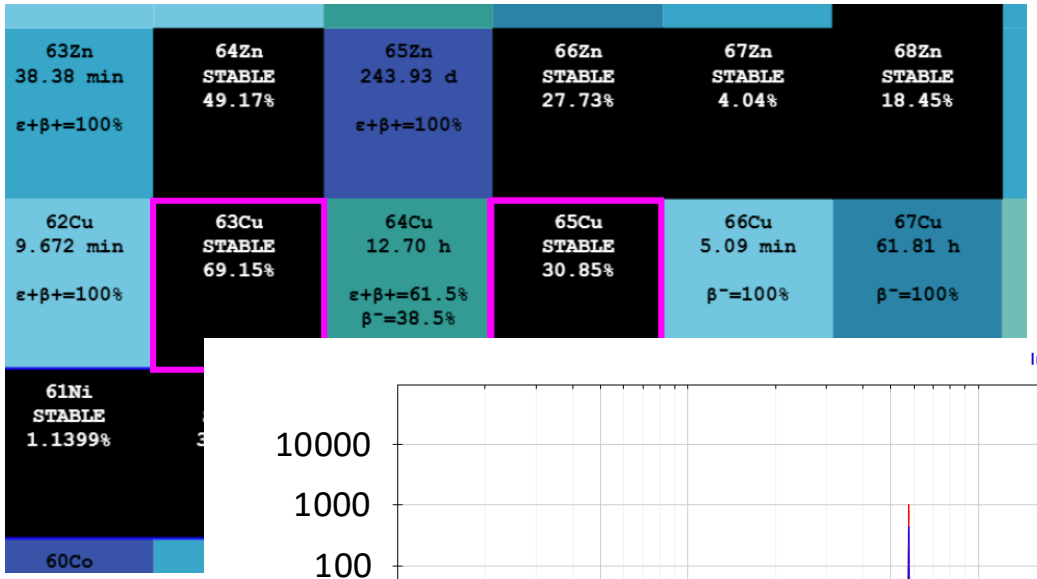
LABORATORY EXERCISE

<u>Qualitative and quantitative analysis:</u>
Copper reference disks and identification of unknown

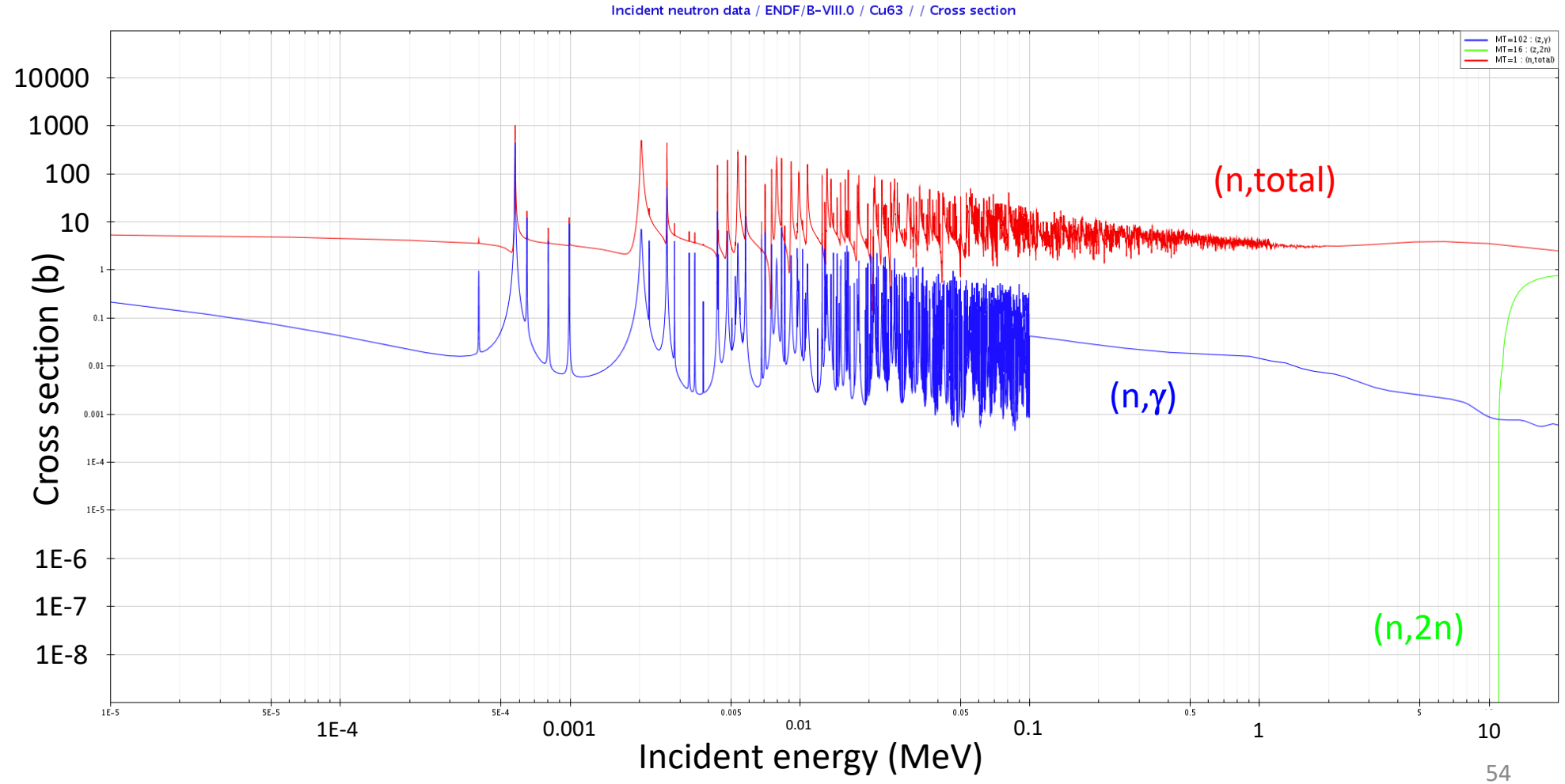


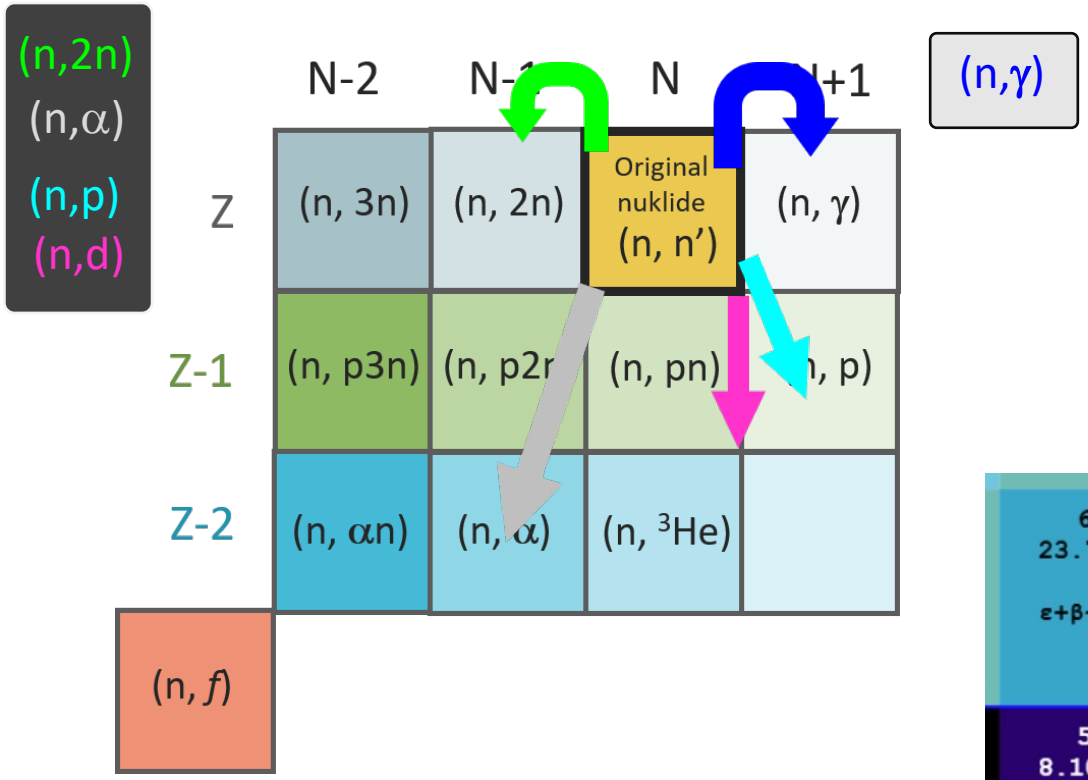




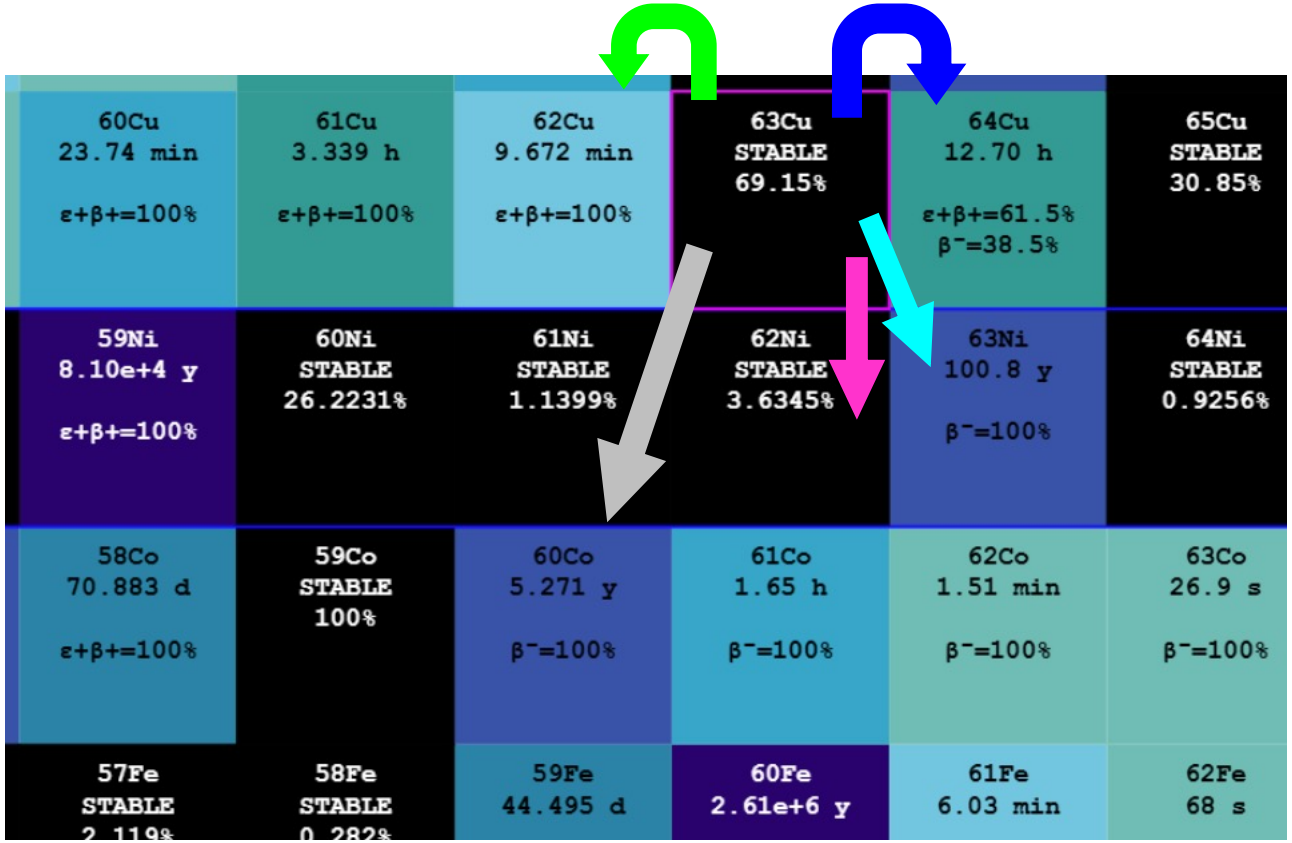

63Zn
38.38 min
ε+β+=100%
64Zn
STABLE
49.17%
65Zn
243.93 d
ε+β+=100%
66Zn
STABLE
27.73%
67Zn
STABLE
4.04%
68Zn
STABLE
18.45%
62Cu
9.672 min
ε+β+=100%
63Cu
STABLE
69.15%
64Cu
12.70 h
ε+β+=61.5%
β⁻=38.5%
65Cu
STABLE
30.85%
66Cu
5.09 min
β⁻=100%
67Cu
61.81 h
β⁻=100%
61Ni
STABLE
1.1399%
60Co

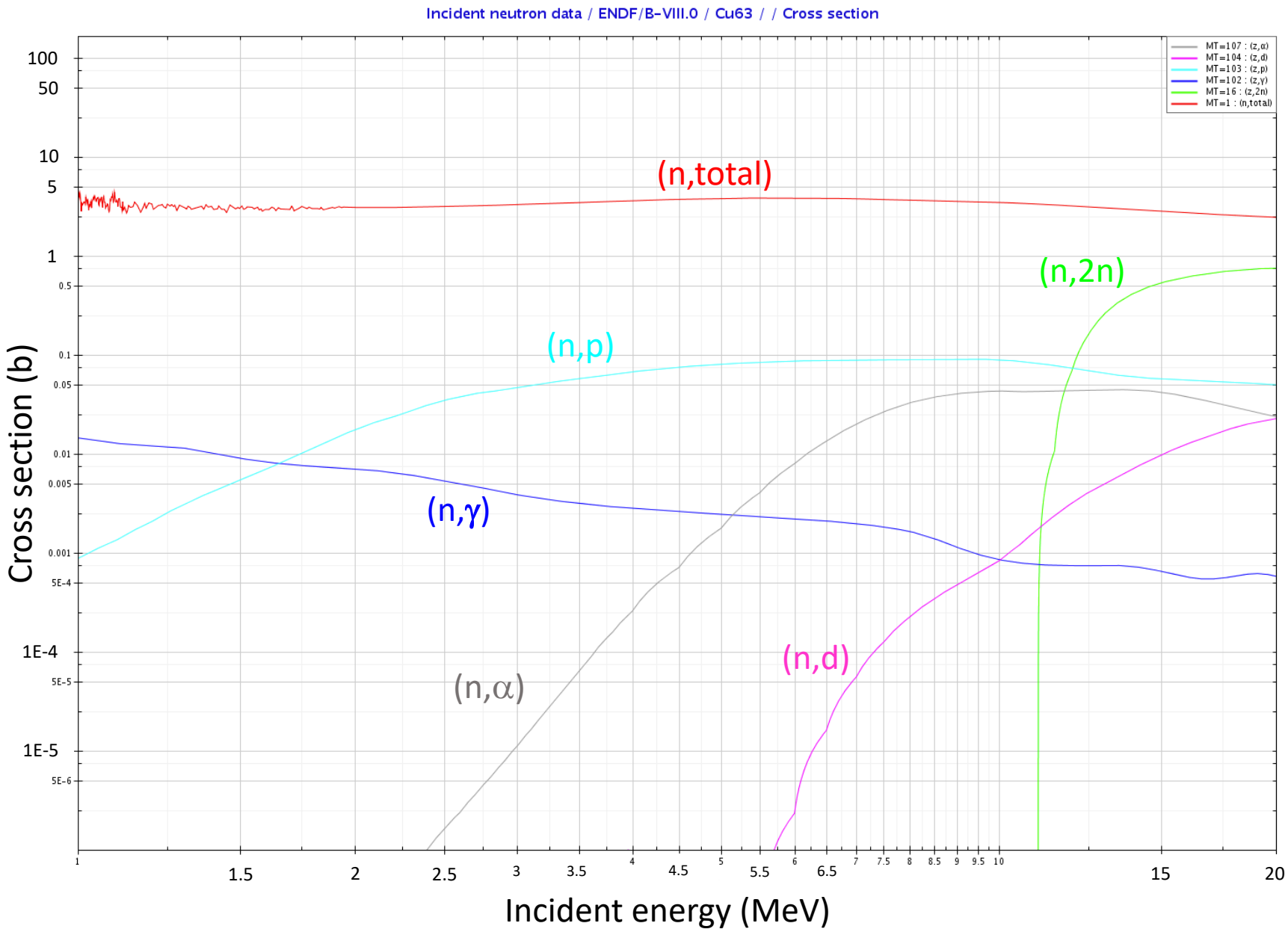
^{63}Cu


Incident neutron data / ENDF/B-VIII.0 / Cu63 / / Cross section
MT=102 : (z,γ)
MT=16 : (z,2n)
MT=1 : (n,total)
(n,total)
(n,γ)
(n,2n)
Cross section (b)
10000
1000
100
10
1
0.1
0.01
0.001
1E-4
1E-5
1E-6
1E-7
1E-8
1E-5
5E-5
1E-4
5E-4
0.001
0.005
0.01
0.05
0.1
0.5
1
5
10
Incident energy (MeV)


(n,2n)
(n,α)
(n,p)
(n,d)
N-2
N-1
N
N+1
Z
Z-1
Z-2
(n, 3n)
(n, 2n)
Original nuklide
(n, n')
(n, γ)
(n, p3n)
(n, p2n)
(n, pn)
(n, p)
(n, αn)
(n, α)
(n, 3He)
(n, f)

(n,γ)


60Cu
23.74 min
ε+β+=100%
61Cu
3.339 h
ε+β+=100%
62Cu
9.672 min
ε+β+=100%
63Cu
STABLE
69.15%
64Cu
12.70 h
ε+β+=61.5%
β⁻=38.5%
65Cu
STABLE
30.85%
59Ni
8.10e+4 y
ε+β+=100%
60Ni
STABLE
26.2231%
61Ni
STABLE
1.1399%
62Ni
STABLE
3.6345%
63Ni
100.8 y
β⁻=100%
64Ni
STABLE
0.9256%
58Co
70.883 d
ε+β+=100%
59Co
STABLE
100%
60Co
5.271 y
β⁻=100%
61Co
1.65 h
β⁻=100%
62Co
1.51 min
β⁻=100%
63Co
26.9 s
β⁻=100%
57Fe
STABLE
58Fe
STABLE
59Fe
44.495 d
60Fe
2.61e+6 y
61Fe
6.03 min
62Fe
68 s


Incident neutron data / ENDF/B-VIII.0 / Cu63 / / Cross section
MT=107 : (z,α)
MT=104 : (z,d)
MT=103 : (z,p)
MT=102 : (z,γ)
MT=16 : (z,2n)
MT=1 : (n,total)
(n,total)
(n,2n)
(n,p)
(n,γ)
(n,α)
(n,d)
Cross section (b)
Incident energy (MeV)