Paul Scherrer Institut

Wir schaffen Wissen – heute für morgen

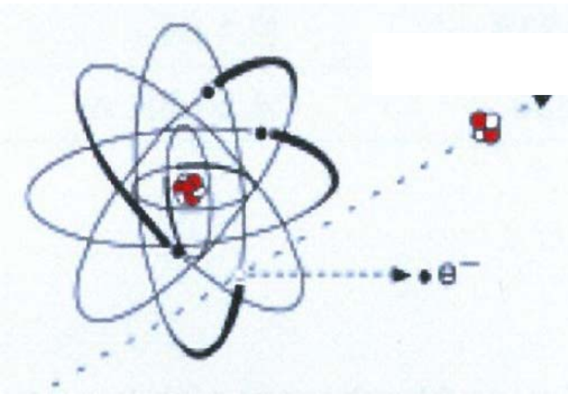
Radiation & Radiation protection

Daniela Kiselev

Content:

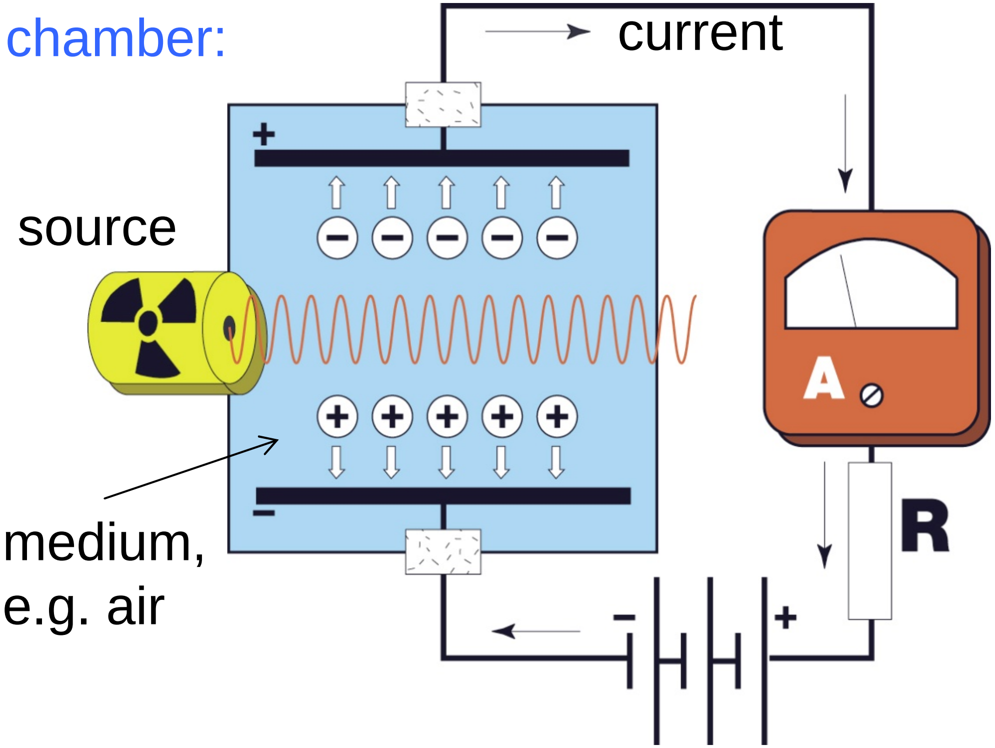
- Definition of radioactive radiation
- Prompt radiation
- Shielding against photons & neutrons
- Residual radiation
- α,β,γ radiation: ranges and shielding
- Quantifying damage:
  - Ionizing density & Linear Energy Transfer (LET)
  - Quality factor & radiation weighting factor
  - Dose equivalent, organ dose, effective dose
- The measured dose and personal dosimeters
- Time evolution of activation
- Activity A → dose rate
- The law of distance: point source and extended source
- Legal dose limits and „natural“ exposure
- Direct and indirect damage of DNA
- Repairment mechanism & prediction of cancer mortality
- Summary of Radiation & Radioprotection rules

Ionizing radiation



ionizing = direct production of charge (electrons, ions)
particles: α,β, γ, e, p, π+ .....

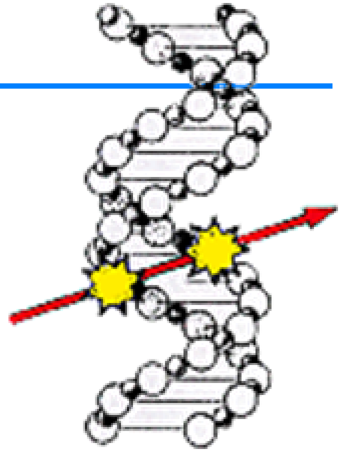
Ionization chamber:



amperemeter
= measurement,
proportional to radiation

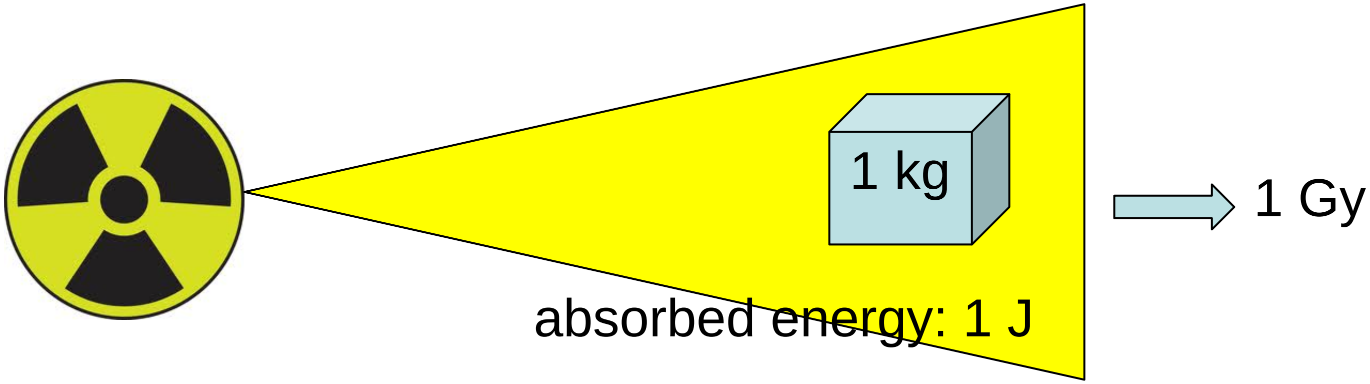
Exposure („ion dose“) D_i = produced charge/mass in AIR, unit: [A s/kg] = [C/kg],
old unit: „Röntgen“ = R = 2.58 x 10^{-4} C/kg
= basic quantity and definition of radioactivity in the past

Radiation: the ability to damage



(significant) energy is absorbed by tissue, bones (and other material)

→ damage



Energy dose: $$D_E = \frac{\Delta E}{\Delta M}$$

↑ material dependent !

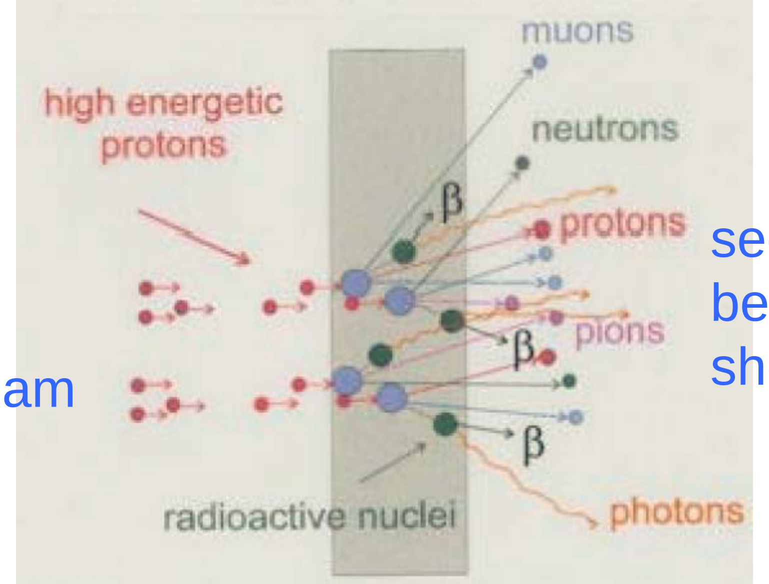
unit: $\frac{J}{kg}$ = 1 Gray [Gy], 1 Gy = 100 rd (rad)

works also for neutrons!

Radiation at accelerators

Prompt radiation: ***during*** accelerator operation

beam on:



primary beam

secondary beam/particles, shower

material

beam off:

Residual radiation: ***after*** accelerator operation,
important during maintenance,
depends on irradiation before and time passed
↑
cooling time

Prompt radiation

- production of almost any particle (dependent on energy of primary beam),
- lots of nuclear reactions,
- huge number of particles produced

Radioprotection:

- control of entries (closed doors): security chain

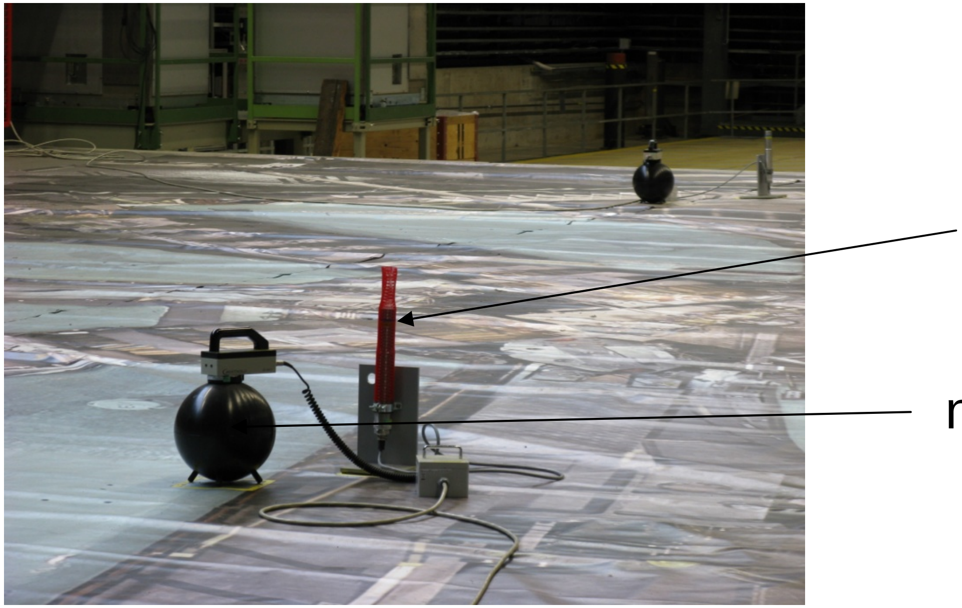
beam permit:
security chain closed

controlled access:
on notice, using keys

free access

Measurement of prompt radiation

- inside the accelerator facility:

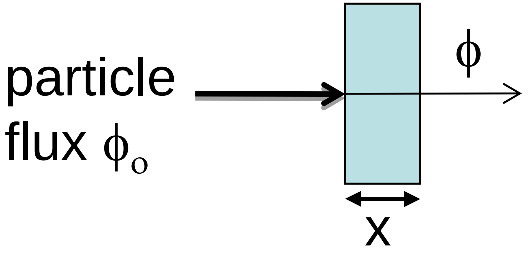


active controlling

- outside of the accelerator facility:

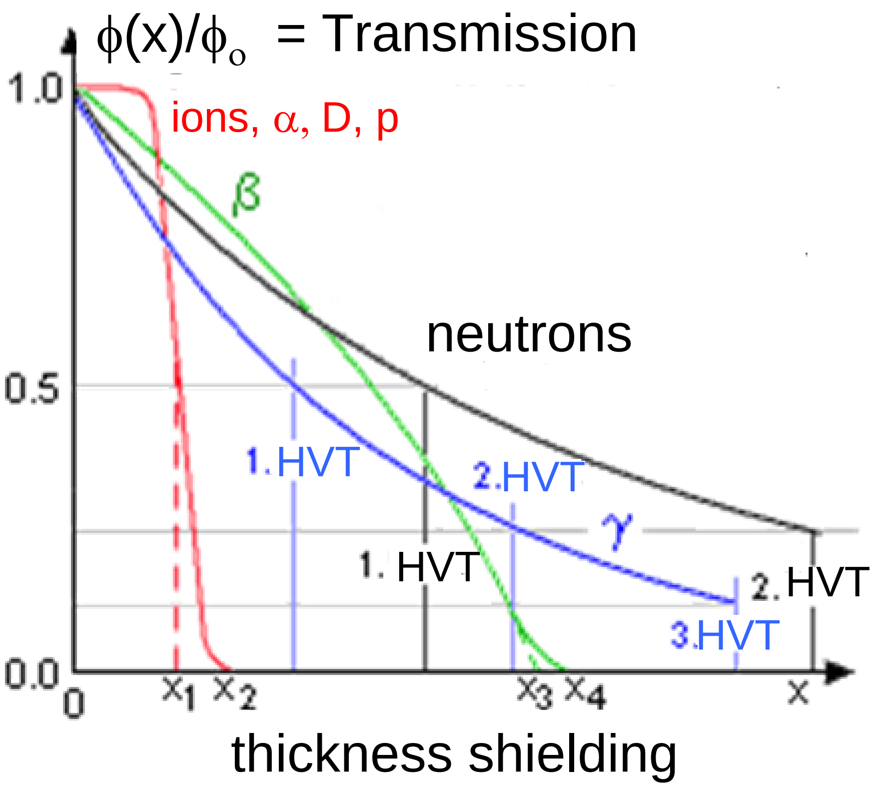
passive controlling

Radioprotection: Shielding



half value thickness: HVT

ϕ (x= HVT) $/\phi_o = 0.5$



x_1, x_3 averaged range

x_2, x_4 maximum range

charged particles are stopped earlier,

→ neutrons, photons are left

only qualitative picture:
Details depend on energy and shielding materials!

Consequences due to the long range of n & γ

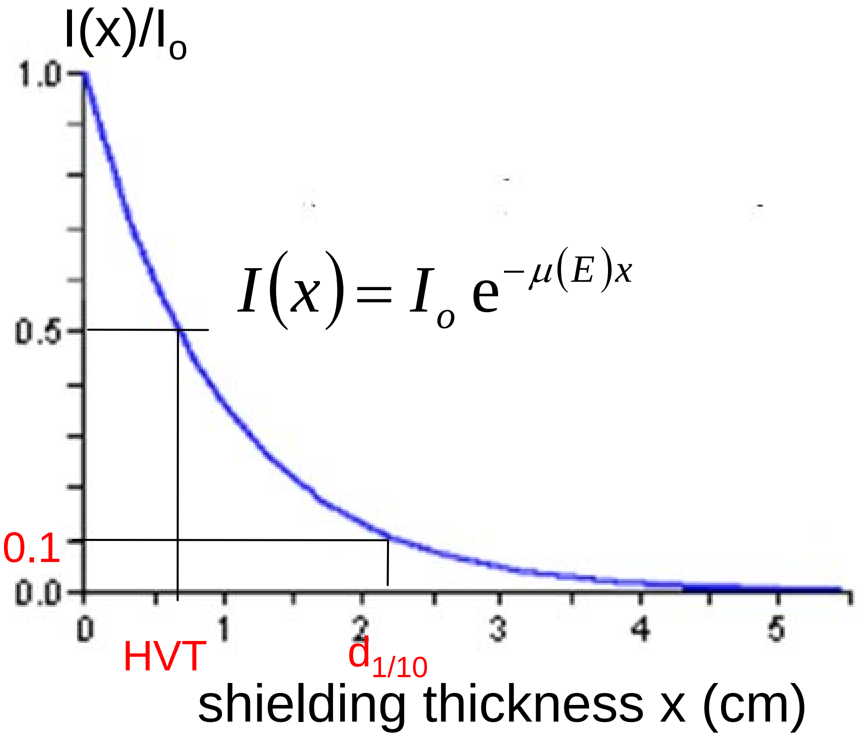
- high energetic n needs large, massive shielding (iron, conrete)
  → expensive, needs space

- low energetic n like to backscatter from conrete walls
→ unwanted n background
→(sometimes) leakage to radiologically uncontrolled areas
(→ increase shielding!)

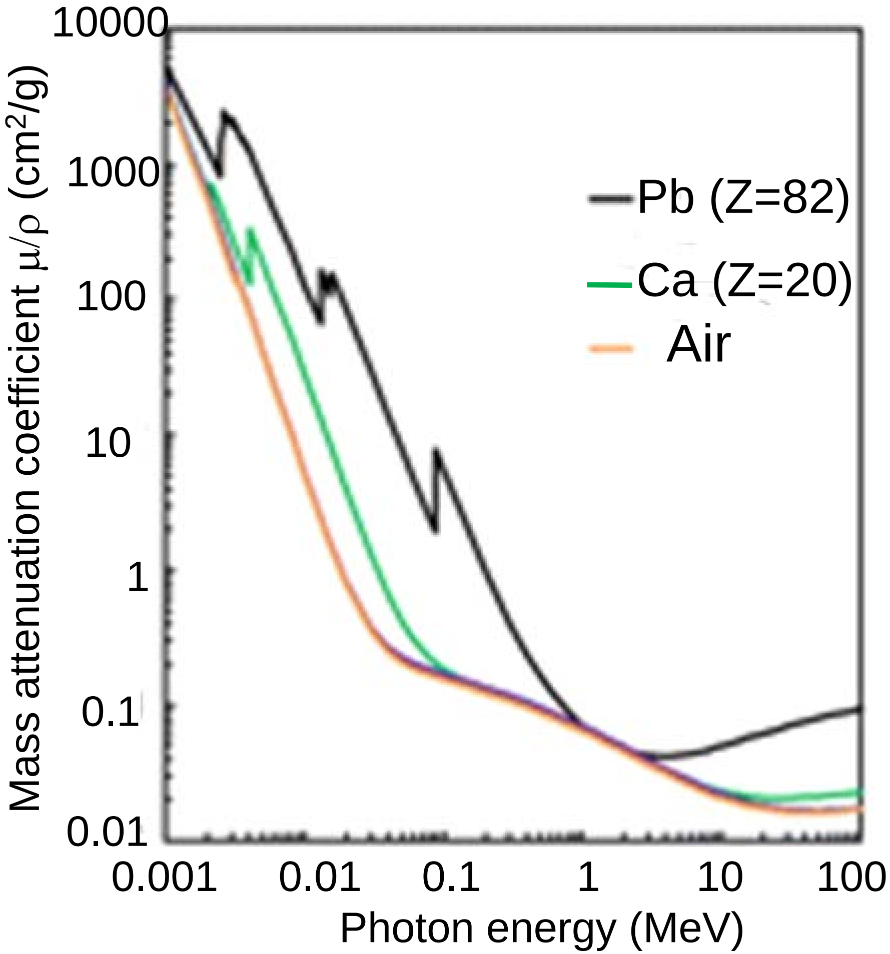
- n & γ produce damage at electronics,
  → extra shielding
  around sensitive devices

lead mats as shielding
against γ-radiation

Shielding of photons (the theory)



$$I(x) = I_o\, e^{-\mu(E)x}$$



μ: linear attenuation coefficient

μ/ρ : mass attenuation coefficient,
~ Z^2 for E > 10 MeV

Example: E= 100 MeV
HVT for concrete: 12.5 cm, Iron: 2.1 cm, Pb: 0.64 cm

high Z material works best

Shielding of photons (the praxis)

Attenuation law $I(x) = I_o\, e^{-\mu(E)x}$ (*)

only valid for monoenergetic photons and pencil beam

Corrections:

1) build-up of lower energetic photons

high energetic photons are not absorbed but produce a shower of lower energetic photons (and electrons)

→ continuous energy loss of photons + angular spread

→ larger shielding required

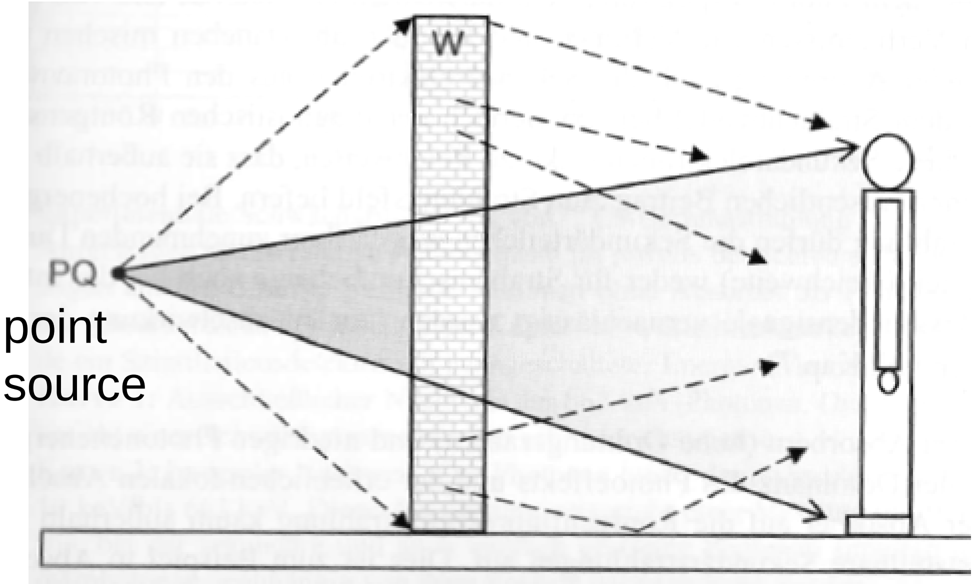
Example: 14 cm Iron

→ shielding factor 40 instead of 350 after (*)

2) Contribution of scattered photons to primary beam

→correction by geometric factor

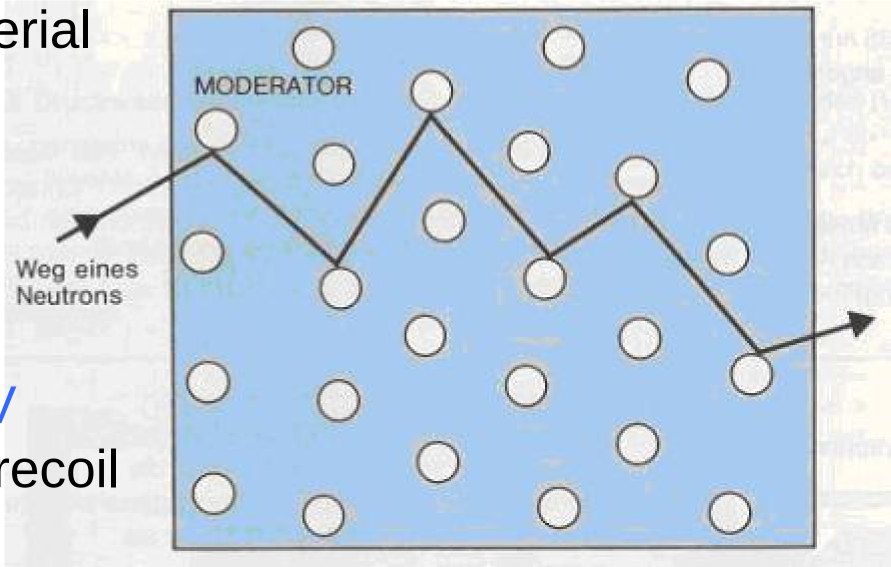
shielding



Shielding of neutrons (1)

For monoenergetic neutrons: $I(x) = I_o \, e^{-\mu(E)x}$

μ strongly depends on energy and material



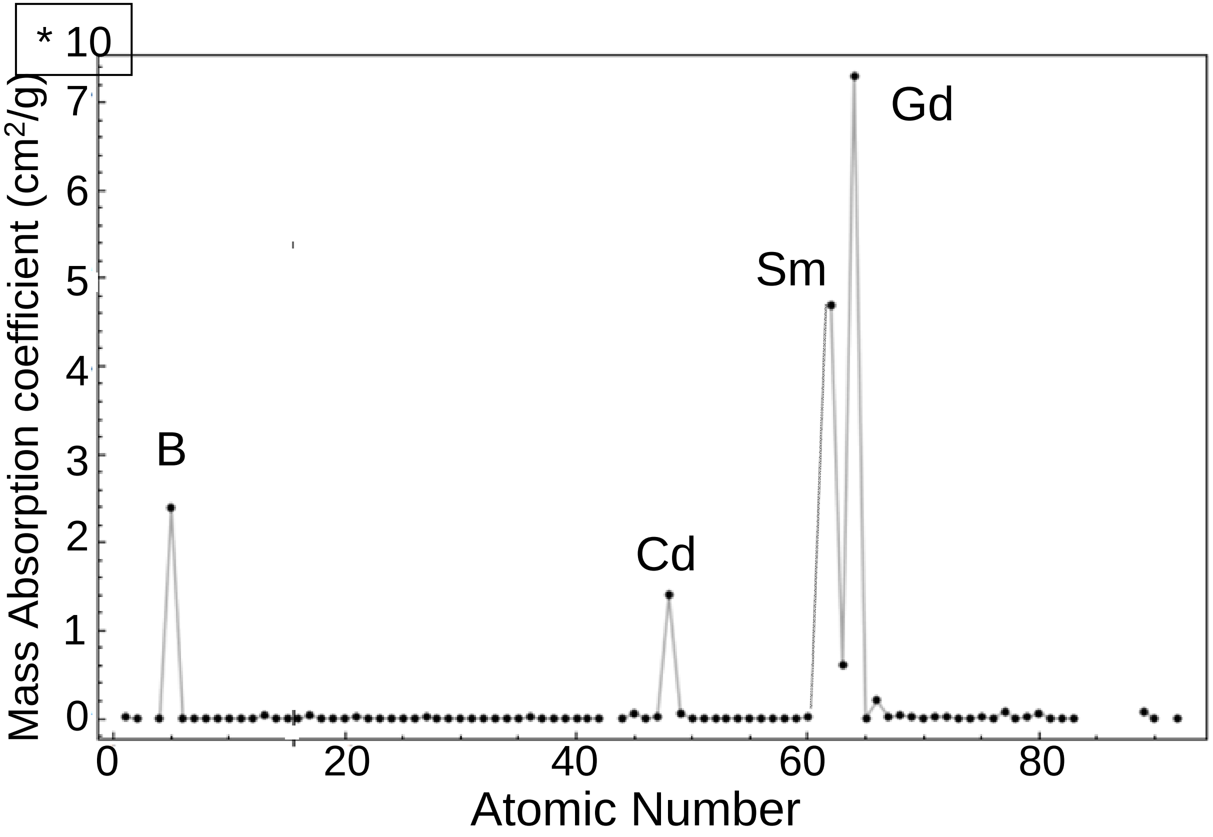
High energetic neutrons: > 20 MeV
energy loss by scattering
→ dense material like iron

Medium energy neutrons: ~ 1 – 20 MeV
most energy loss at light atoms due to recoil
→ moderation
number of collisions needed for 2 MeV n → 0.025 eV (thermal):
at H: 18
at C: 114
at U: 2172
→ Concrete, Polyethylen contains lots of H !
(HVT ~ 7 cm for 1 MeV neutrons in concrete)

Shielding of neutrons (2)

Low energetic neutrons: ~ 0.025 eV (thermal)

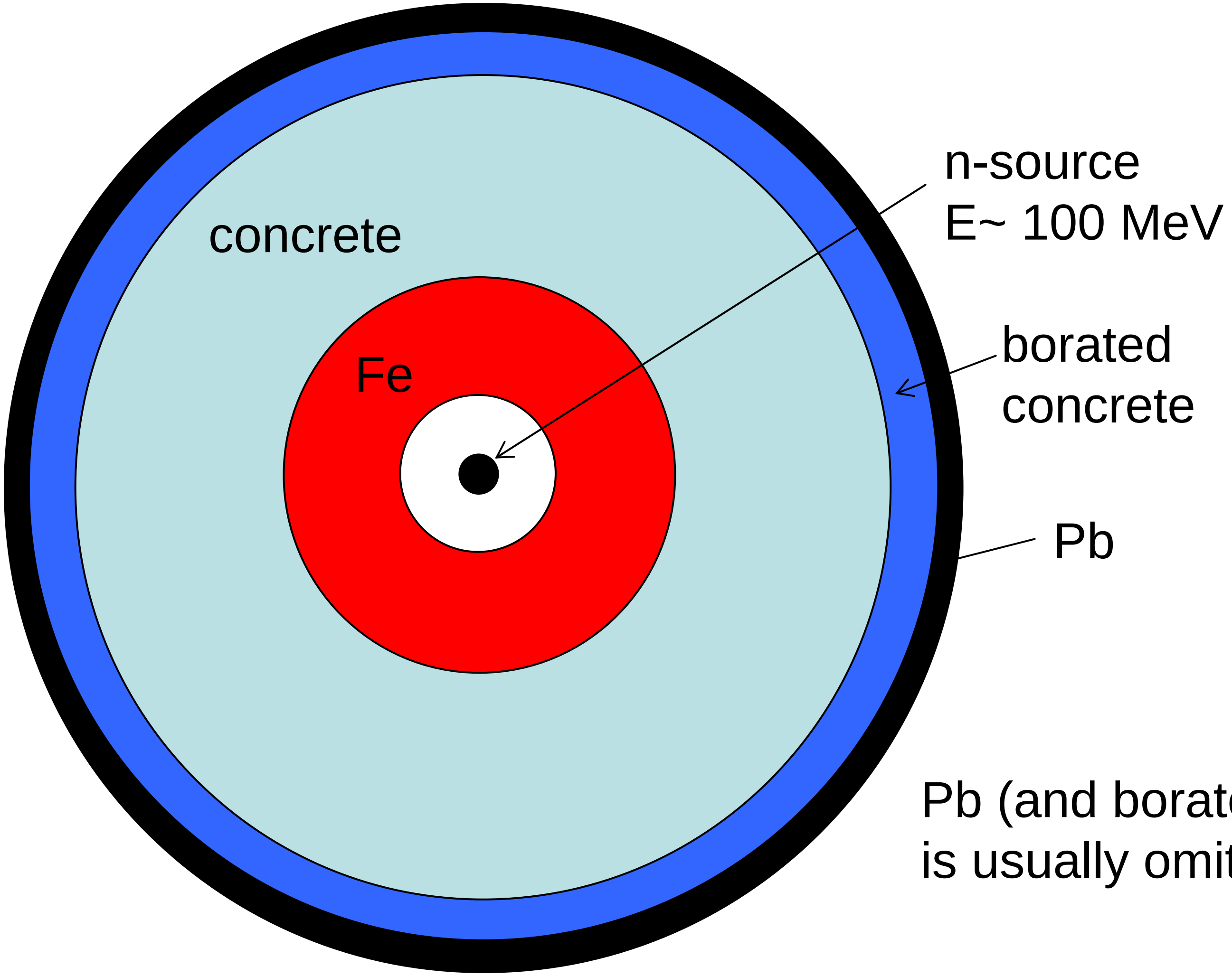
absorption, often (n,γ)-reaction, called capture



neutron absorption often increases γ-background

Example: Shielding for high-energetic neutrons

Sandwich: stacking of different materials

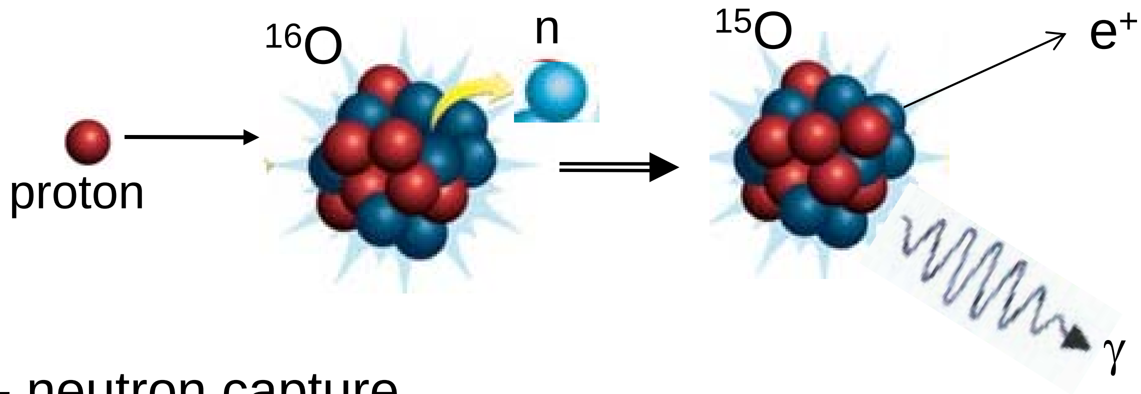


Pb (and borated concrete)
is usually omitted

Residual activation

Production of radioactive nuclei due to prompt radiation, e.g.:

- knock-out of neutrons, protons

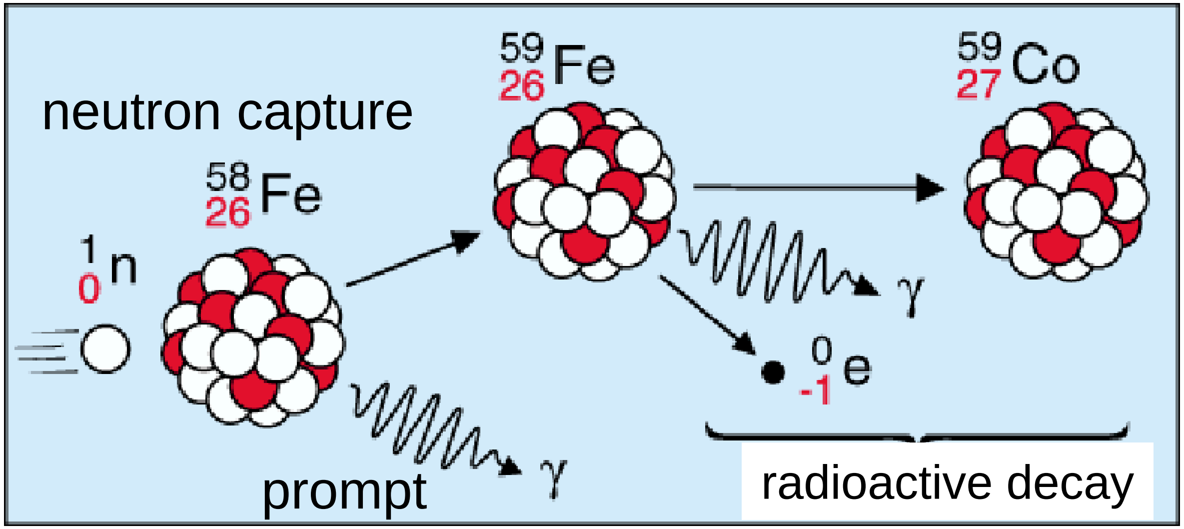


β⁺ decay:

$^{15}O \longrightarrow {}^{15}N + e^+ + \nu_e$

$(p \rightarrow n + e^+ + \nu_e)$

- neutron capture



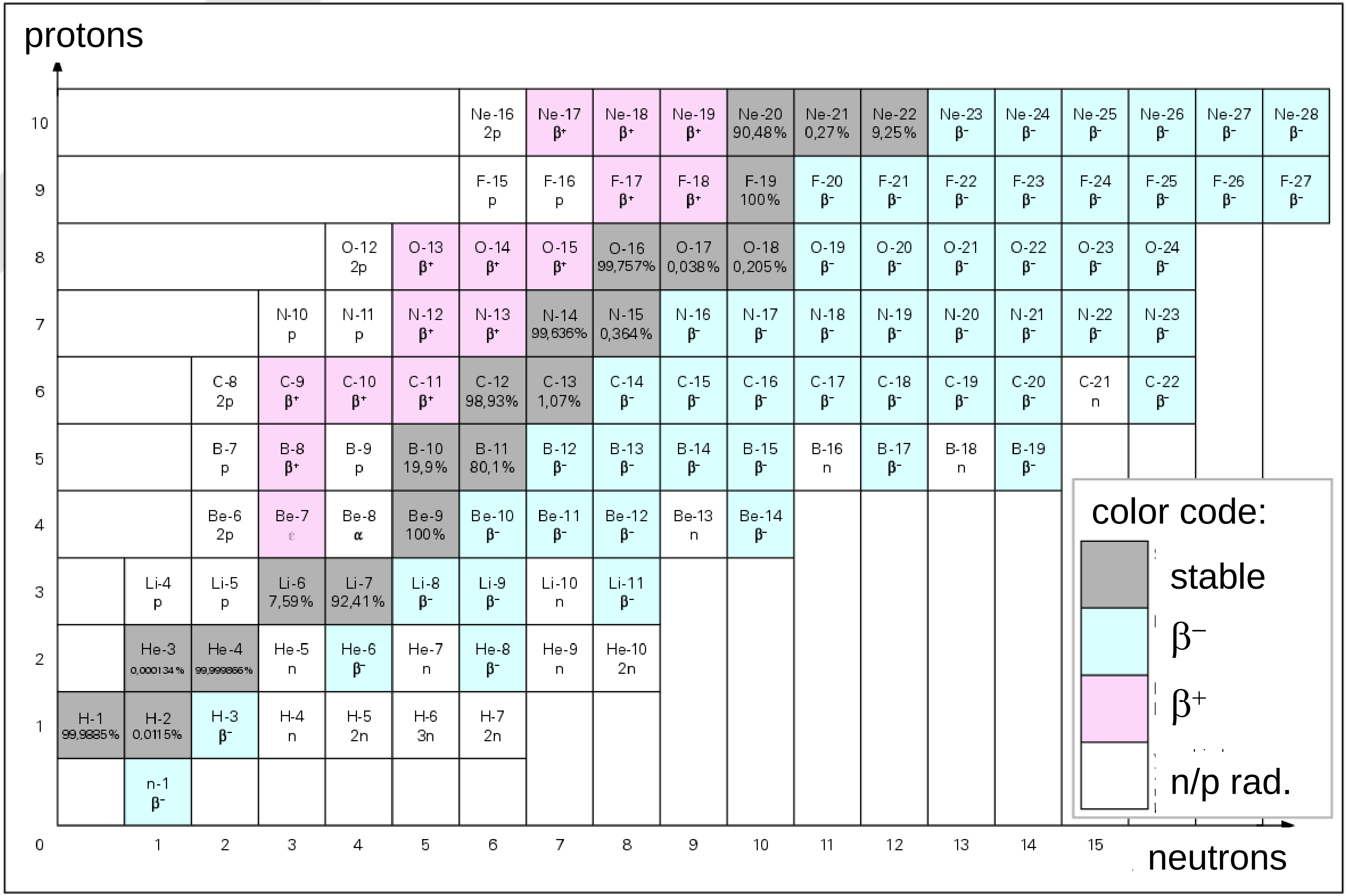
β⁻ decay:

$^{59}Fe \longrightarrow {}^{59}Co + e^- + \overline{\nu}_e$

$(n \rightarrow p + e^- + \overline{\nu}_e)$

often accompanied by γ-radiation

Nuclide chart

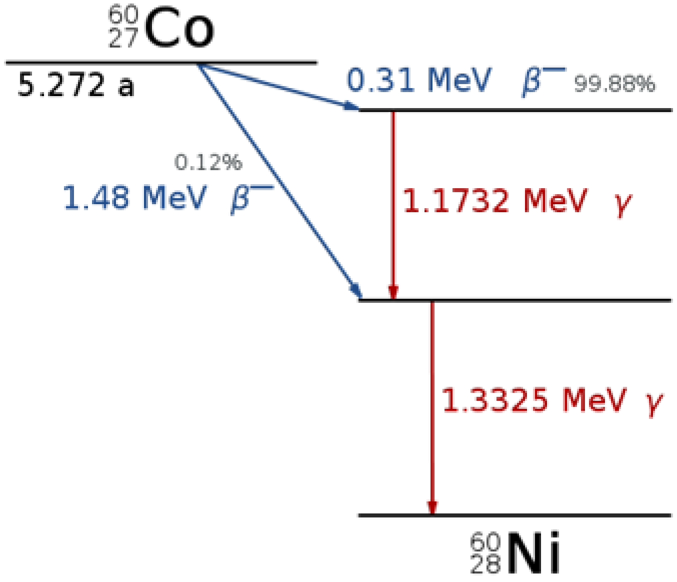


protons	0	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15			
10							Ne-16 2p	Ne-17 β^+	Ne-18 β^+	Ne-19 β^+	Ne-20 90,48%	Ne-21 0,27%	Ne-22 9,25%	Ne-23 β^-	Ne-24 β^-	Ne-25 β^-	Ne-26 β^-	Ne-27 β^-	Ne-28 β^-
9							F-15 p	F-16 p	F-17 β^+	F-18 β^+	F-19 100%	F-20 β^-	F-21 β^-	F-22 β^-	F-23 β^-	F-24 β^-	F-25 β^-	F-26 β^-	F-27 β^-
8					O-12 2p	O-13 β^+	O-14 β^+	O-15 β^+	O-16 99,757%	O-17 0,038%	O-18 0,205%	O-19 β^-	O-20 β^-	O-21 β^-	O-22 β^-	O-23 β^-	O-24 β^-		
7				N-10 p	N-11 p	N-12 β^+	N-13 β^+	N-14 99,636%	N-15 0,364%	N-16 β^-	N-17 β^-	N-18 β^-	N-19 β^-	N-20 β^-	N-21 β^-	N-22 β^-	N-23 β^-		
6			C-8 2p	C-9 β^+	C-10 β^+	C-11 β^+	C-12 98,93%	C-13 1,07%	C-14 β^-	C-15 β^-	C-16 β^-	C-17 β^-	C-18 β^-	C-19 β^-	C-20 β^-	C-21 n	C-22 β^-		
5			B-7 p	B-8 β^+	B-9 p	B-10 19,9%	B-11 80,1%	B-12 β^-	B-13 β^-	B-14 β^-	B-15 β^-	B-16 n	B-17 β^-	B-18 n	B-19 β^-				
4			Be-6 2p	Be-7 ε	Be-8 α	Be-9 100%	Be-10 β^-	Be-11 β^-	Be-12 β^-	Be-13 n	Be-14 β^-								
3		Li-4 p	Li-5 p	Li-6 7,59%	Li-7 92,41%	Li-8 β^-	Li-9 β^-	Li-10 n	Li-11 β^-										
2		He-3 0,000134%	He-4 99,999866%	He-5 n	He-6 β^-	He-7 n	He-8 β^-	He-9 n	He-10 2n										
1	H-1 99,9885%	H-2 0,0115%	H-3 β^-	H-4 n	H-5 2n	H-6 3n	H-7 2n												
0		n-1 β^-																	

color code:

- stable
- β^-
- β^+
- n/p rad.

γ-Emission



$$^{60}\text{Co} \xrightarrow{\beta^-} {}^{60}\text{Ni}^* \xrightarrow{2\,\gamma} {}^{60}\text{Ni}$$

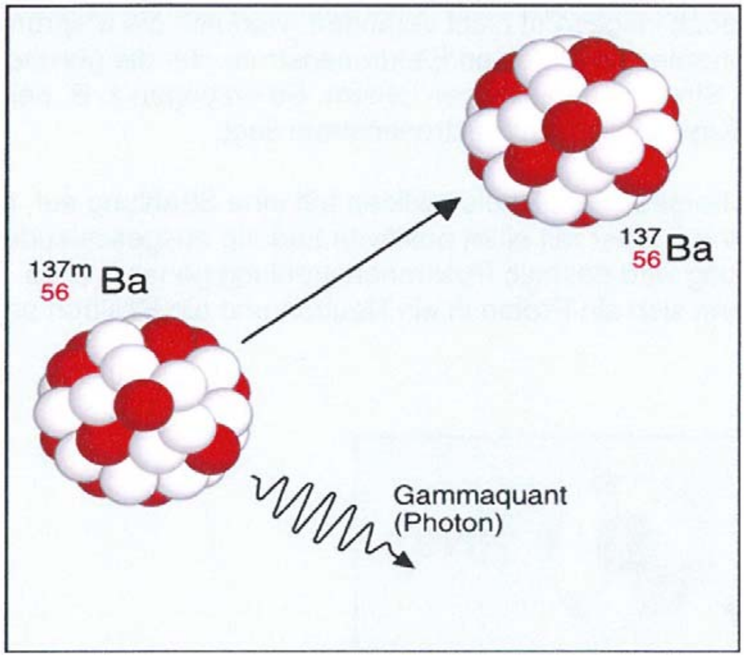
^{60}Ni*: excited state

^{60}Ni: ground state

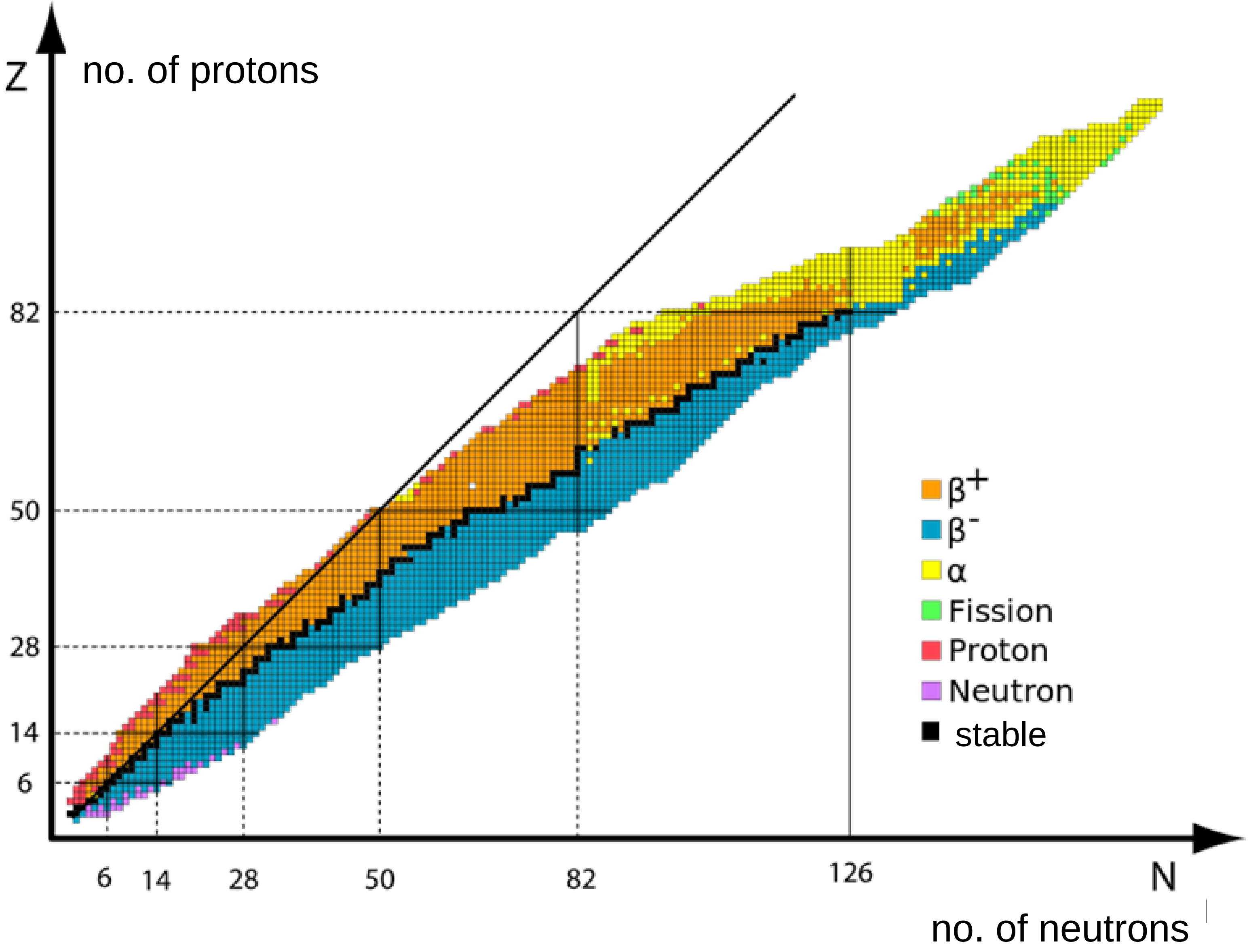
typical $E_\gamma \sim 1\text{–}2$ MeV

$$^{137m}\text{Ba} \xrightarrow{\gamma} {}^{137}\text{Ba}$$

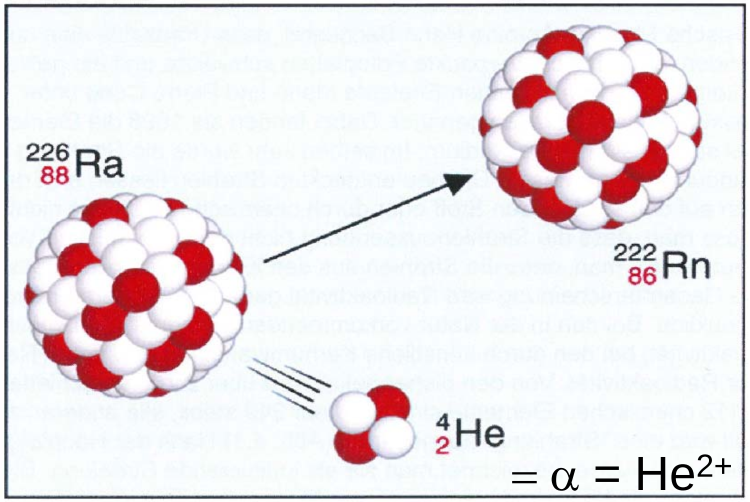
^{137m}Ba: metastable (Isomer):
excited state,
long life time



Nuclide chart



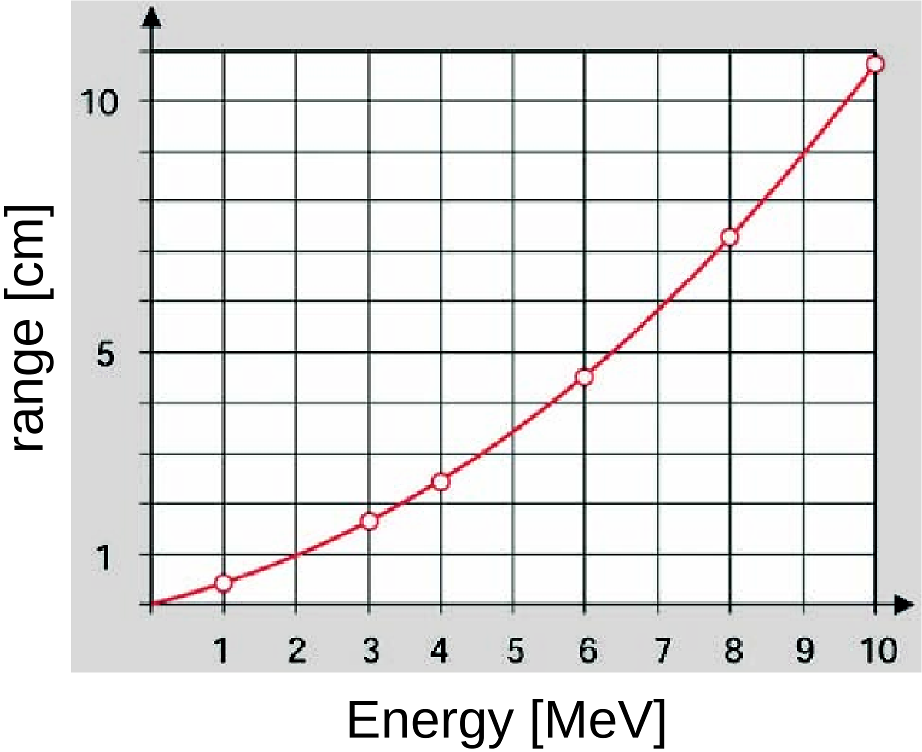
α-Decay



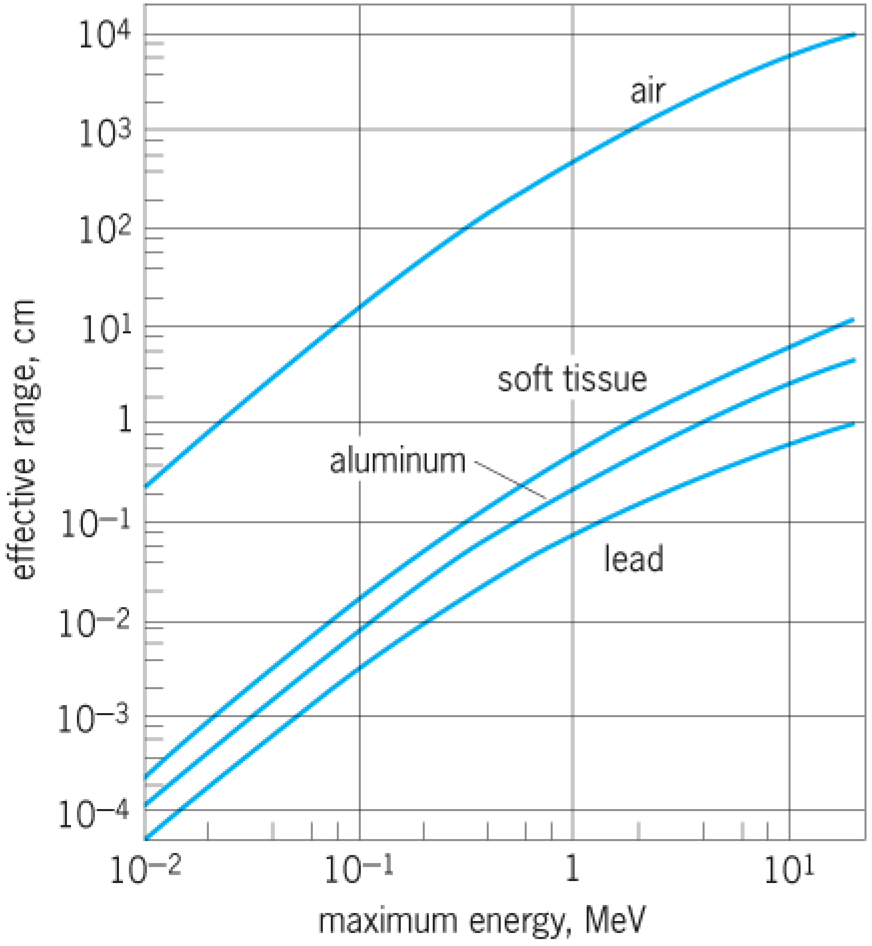
heavy nuclei are often α-Emitters,
typical $E_\alpha \sim 4 – 8$ MeV

R = 2.5 cm for $E_\alpha \sim 4$ MeV
→ no radiation hazard
except for incorporation
(no eating in radiological
controlled areas!)

Range of α-particles in air



Range of β-particles (e-, e+)



↑

typical β-energy

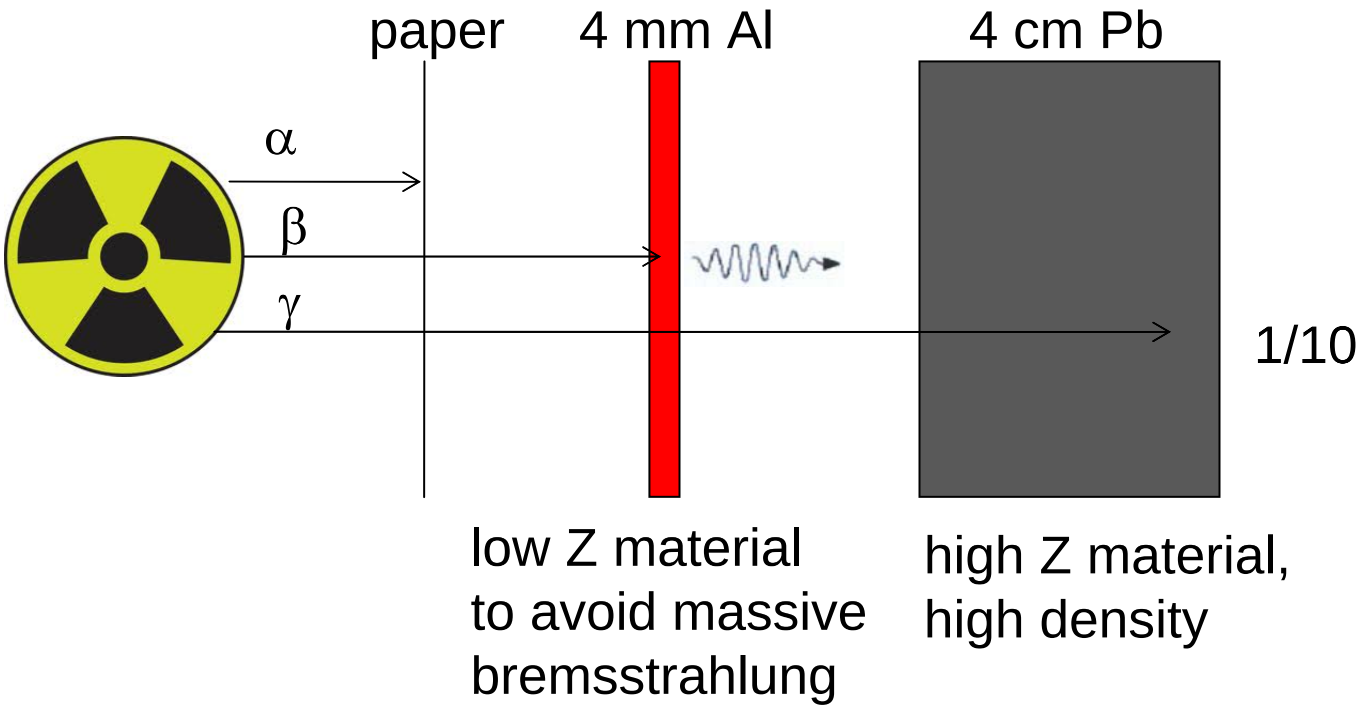
β-radiation:

- ~ 1 cm into tissue (= skin)
- hazard to eyes → use glasses !

High Z materials like Pb
stop β´s effectively
but produce bremsstrahlung
(= low energetic photons)

Shielding against residual activation

Particles like γ and electrons have much lower energy:
γ: < 3 MeV
e^-/ e^+: 1-2 MeV (max. β-energy)
α: ~ 4 – 8 MeV



low Z material to avoid massive bremsstrahlung

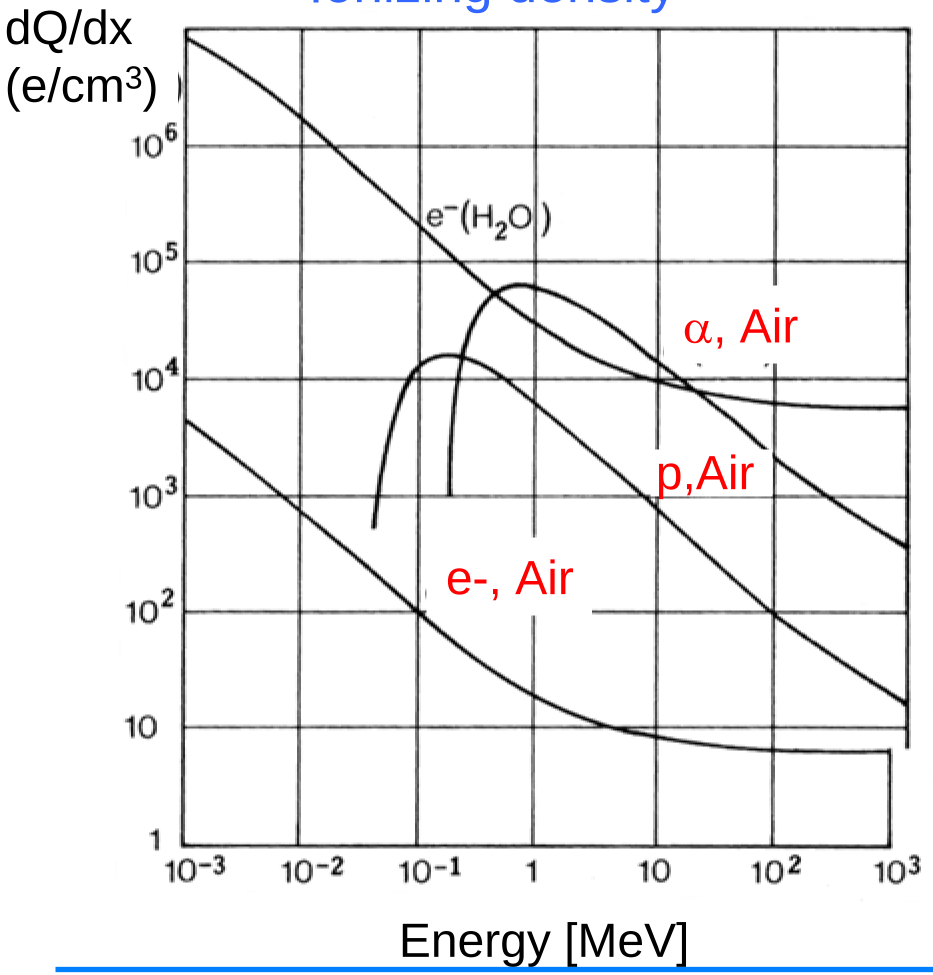
high Z material, high density

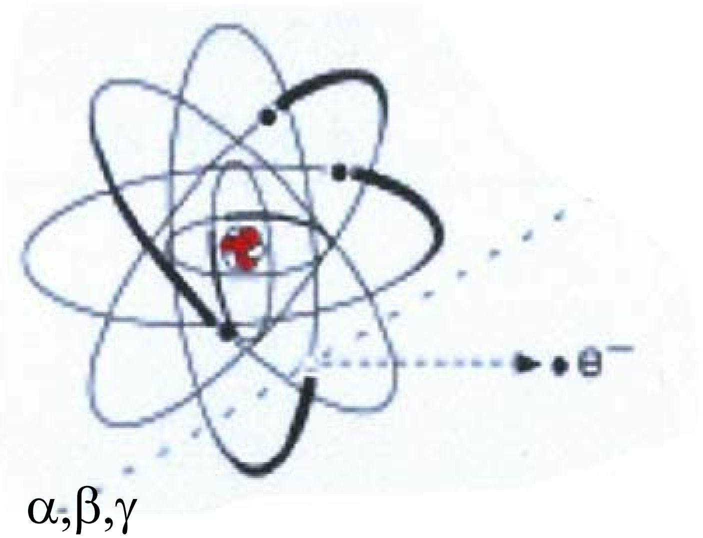
Ionizing density

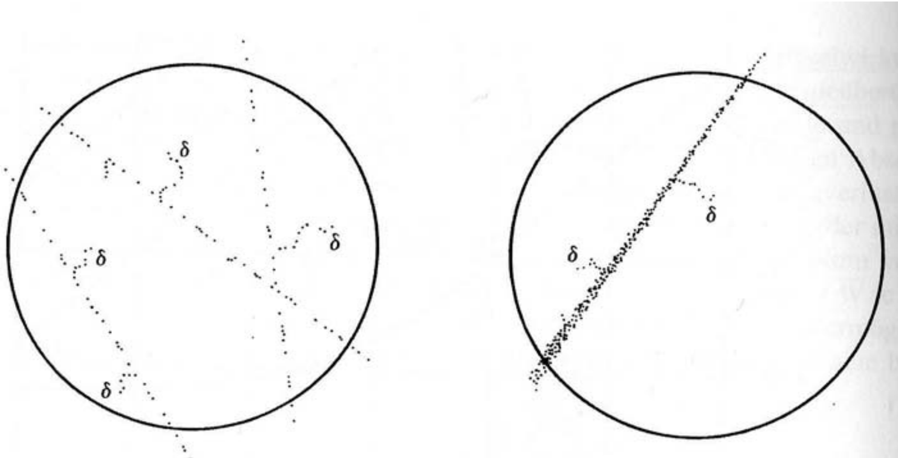
Charged particles & γ´s ionize atoms

→ exposure as measure

Ionizing density







low

high density
→ more damage

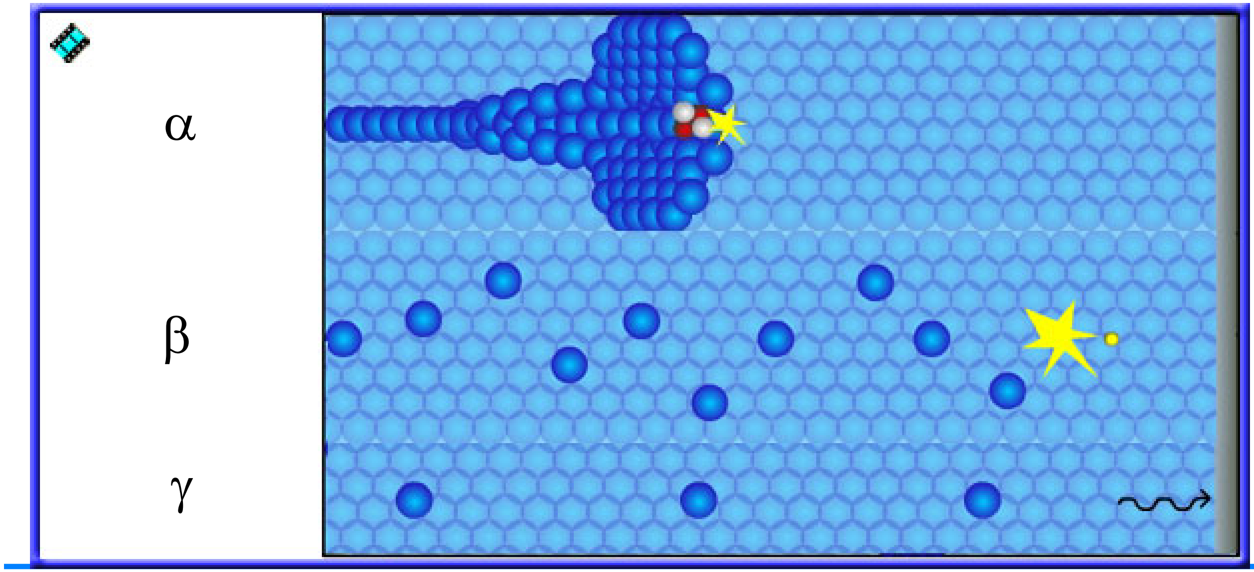
More general: Energy transfer

only ~ 50 % of the energy is used for ionization,
depends on energy and particle type

neutrons: no ionization at all but energy transfer by capture/absorption

→ averaged energy transferred per unit length to tissue

LET: Linear Energy transfer: $LET = \frac{\overline{dE}}{dl}$ unit: $\left[\frac{keV}{\mu m}\right]$



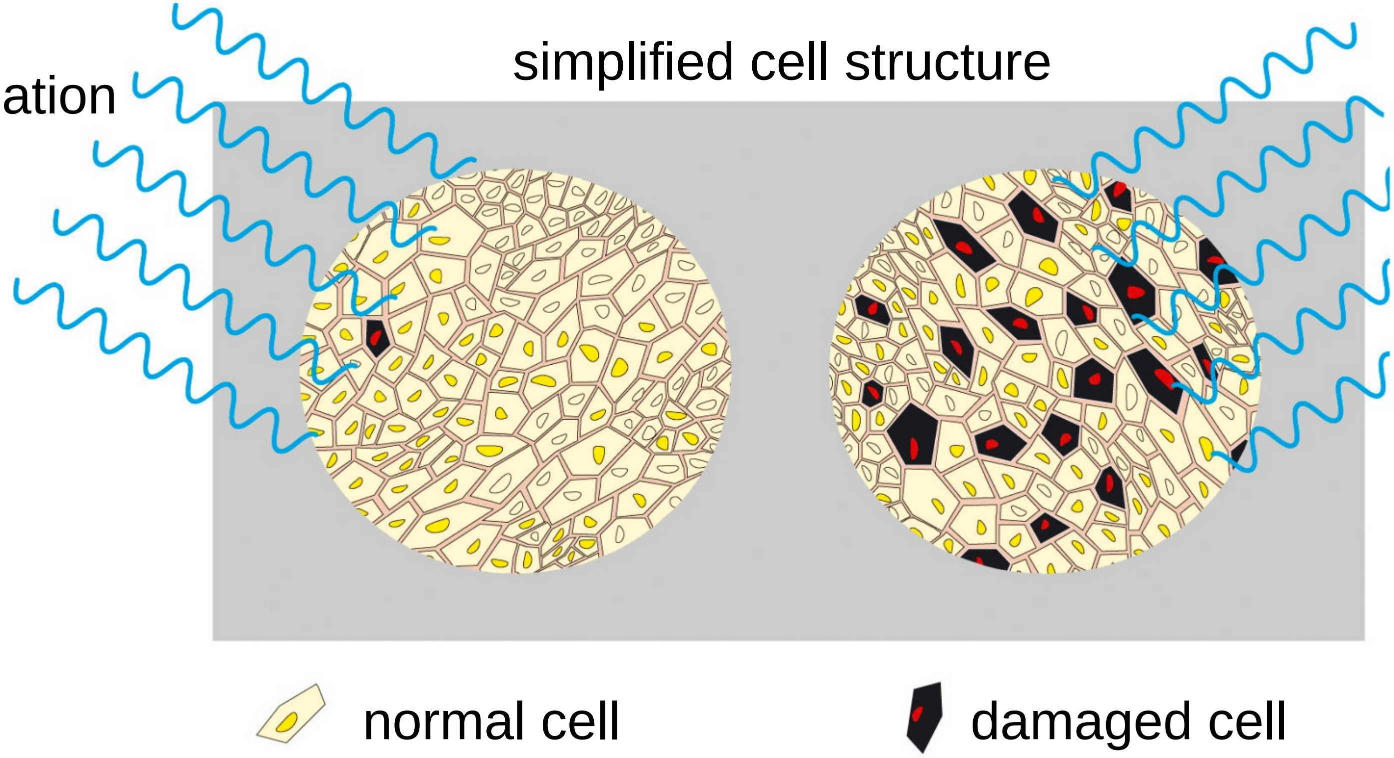
LET ~ 200

→ High LET radiation

LET < 3.5

→ Low LET radiation

Damage to the tissue



β, γ radiation

simplified cell structure

α-particles

normal cell

damaged cell

Empirical observation:

at identical absorbed doses (e.g., 2 Gy) α-particles show a 20-times more damaging biological effect than β, γ.

→ introduction of a quality factor

Quality factor

based on LET (nowadays):

LET (keV/cm) in H_2O	Q(L)
< 10	1
10 - 100	0.32 LET – 2.2
> 100	$300/\sqrt{LET}$

ICRP60, 1991

Dose equivalent: $H = Q\, D_T$

unit: Sv, 1 Sv = 100 rem

energy dose
absorbed by tissue

For a mixed particle or particle energy field, Q is an average:

$$\overline{Q} = \frac{1}{D}\int Q(L)\frac{dD}{dL}dL$$

Dose equivalent is a more theoretical quantity!

More practical: The organ dose H_T

(also called equivalent dose („Äquivalentdosis“))

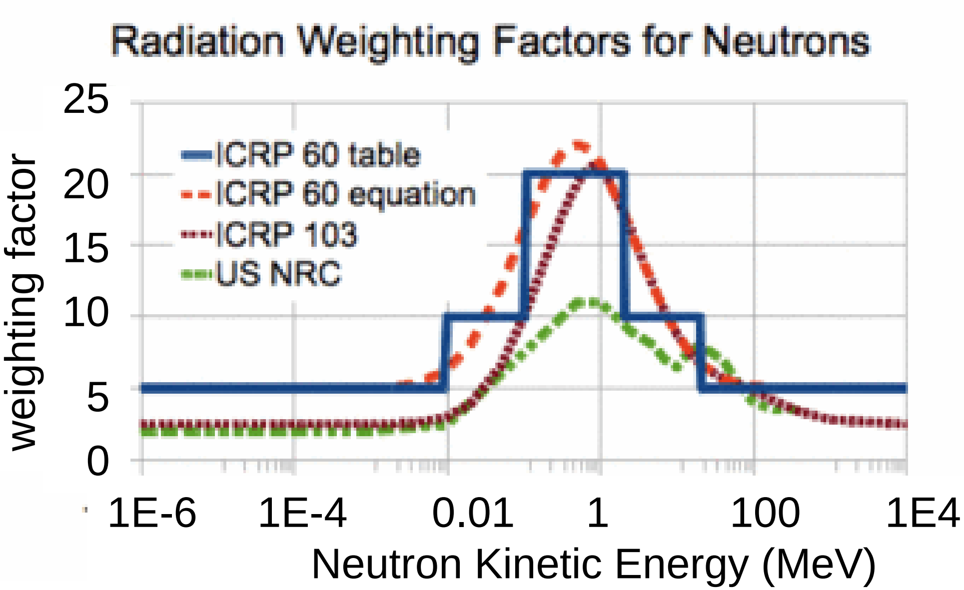
$H_T = w_R D_T$ for the judgement of biological damage

D_T : energy dose averaged over organs, tissue

w_R: radiation weighting factor ≈ quality factor averaged used to represent relative biological effectiveness (RBE)

Radiation type	**W_R**
Photons, all energies	1
Electrons, muons, all energies	1
Protons and charged pions	2
Alpha particles, fission fragments, heavy ions	20

according to ICRP103 (2007)



Organ Dependent Effects – Effective Dose E

Various organs react differently when irradiated with the same H_T.

→ Effective Dose $E = w_T H_T$

w_T = tissue weighting factors for different body parts:

gonads	0.08
breast, bone marrow, lung	0.12
colon, stomach	0.12
thyroid, liver	0.04
bone surface, skin, brain	0.01
sum of all organs (whole body)	1

ICRP103

If more than one organ is affected, sum over all contributions.

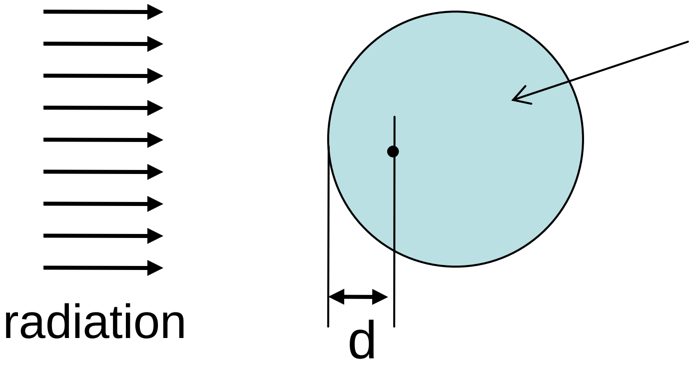
Example: Incorporation of radioactive iodine → thyroid gets H_T = 100 mSv.

E = 100 mSv x 0.04 = 4 mSv

i.e. same effect as if the whole body were irradiated with 4 mSv.

The measurement of dose

1) at a location (e.g. experimental hall, accelerator)
Dosimeters are calibrated on the ICRU-sphere ϕ 30 cm, $\rho = 1\ \mathrm{g/cm^3}$



equivalent
to muscles & soft tissue
(76% O_2, 11% C,
10% H, 3% N_2)

- for high energetic radiation:
  measurement independent of orientation of radiation

→ Ambient dose equivalent: H*(d), d = 10 mm unit: Sv

(H*(d) and H can differ by 50% due to back scattering
& production of secondary particles in medium)

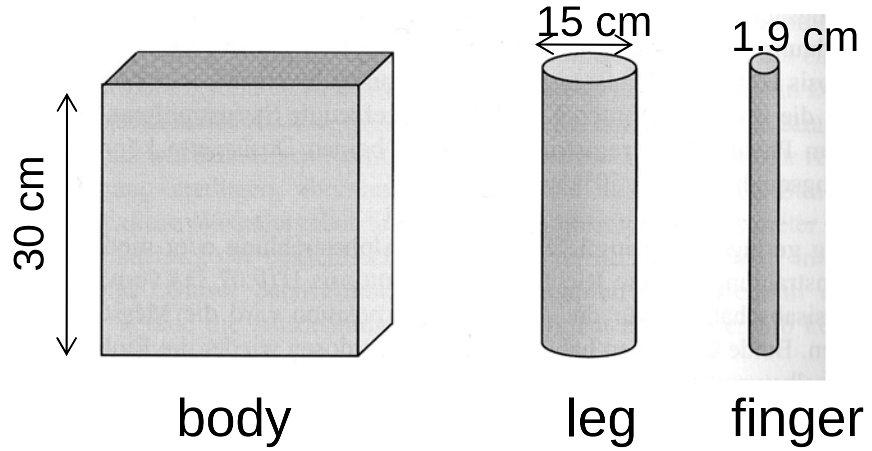
The measurement of dose

- for low energetic particles: $\beta < 2$ MeV, $\gamma < 15$ keV, α
  measurement depends on angle relative to incident radiation
  → directional dose equivalent: $H'(d,\Omega)$, d = 0.07 mm (surface)

Measured doses are an estimate for doses @
- human body (d = 10 mm)
- skin (d = 0.07 mm)
- eyes (d = 3 mm)

2) at persons:

dosimeters are calibrated at phantoms



finger dosimeter

measures $H_p'(0.07)$

Examples for personal dosimeters

β & γ radiation: $H_p(10)$, $H_p(0.07)$
Detector: 3 LiF (TLD700) (with different covers)
(stores excitations = thermoluminescence)
$H_p(10)$: 0.1 mSv bis 5 Sv
$H_p(0.07)$: 1 mSv bis 5 Sv
γ energy: 20 keV -3 MeV

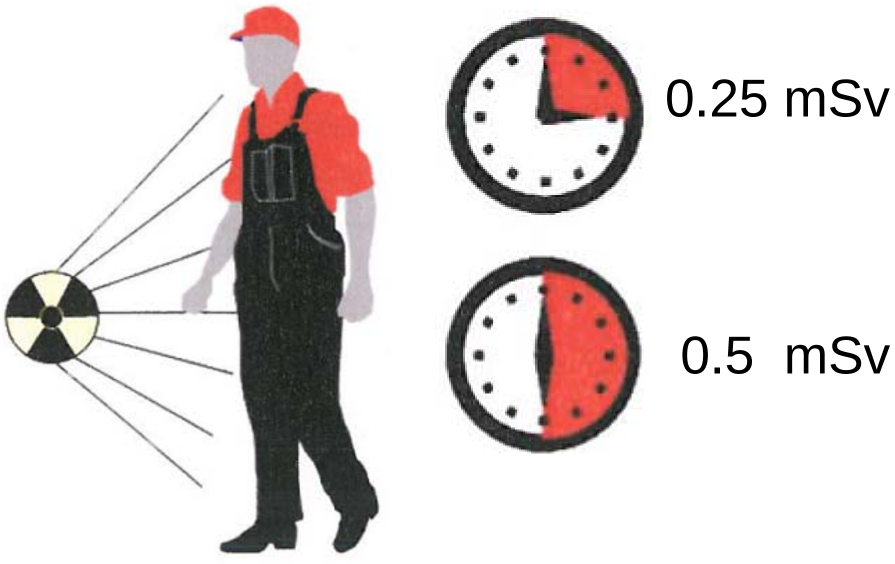
Neutron radiation:
1 CR39-Detektor (plastic polymer)
+ 3 converters: PE, PE(Li), Al
Measurement of tracks produced by neutrons in converters
$H_p(10)$: 0.5 mSv bis 5 Sv
Energy: thermal and 200 keV - 15 MeV

remarks: both types are also used for monitoring prompt radiation at PSI: ~ 100 distributed in the accelerator areal

Dose rate ***D***

Dose rate ***D*** = dose/time
often used unit: [Sv/h]
naming: officially $\dot{D}$

For all kind of doses
D_T, H_T, E, H, H*(10), H*(0.07)
one gets the average dose rate by
dividing by exposure time



H*(10) = 1 mSv/h

Rules of Radioprotection: To reduce the dose
→ **be fast!**
→ work planning and preparation might help
(praxis at nuclear power plants: training at a mock-up)
→ Distribution of work to several persons

Time evolution: Build-up and Decay of activation

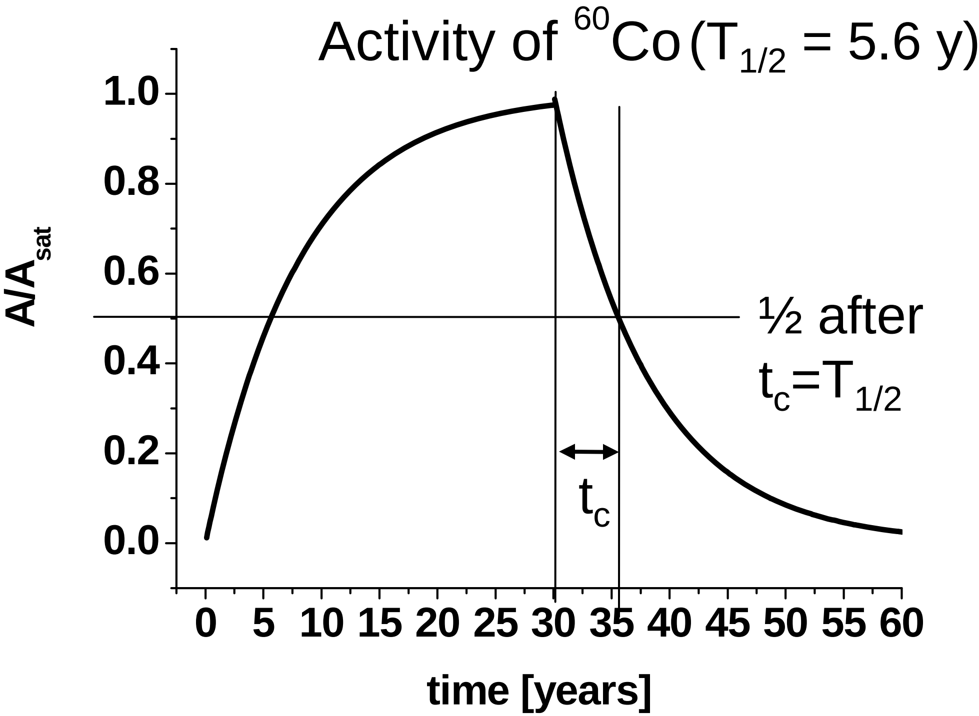
Law of the radioactive decay:

$$N(t) = N(t_o)\ e^{-\lambda(t-t_o)}$$

no. of nuclei, which are in the sample at time t

λ: decay constant $\lambda = \dfrac{\ln 2}{T_{1/2}}$

$T_{1/2}$: half-life



Definition of the activity: unit Bq

$$A(t) = -\frac{dN(t)}{dt} = \lambda N(t)\ = \lambda\ \mathrm{N}_o e^{-\lambda t} = A_o e^{-\lambda t}$$

Specific activity: A/m [Bq/g]

example: A(^{60}Co) < 1Bq/g → below the value of free release (LE value)

Build-up of activity

The simple case: 1 radioisotope

Rate equation:

$$\frac{dN(t)}{dt} = P - \lambda N(t)$$

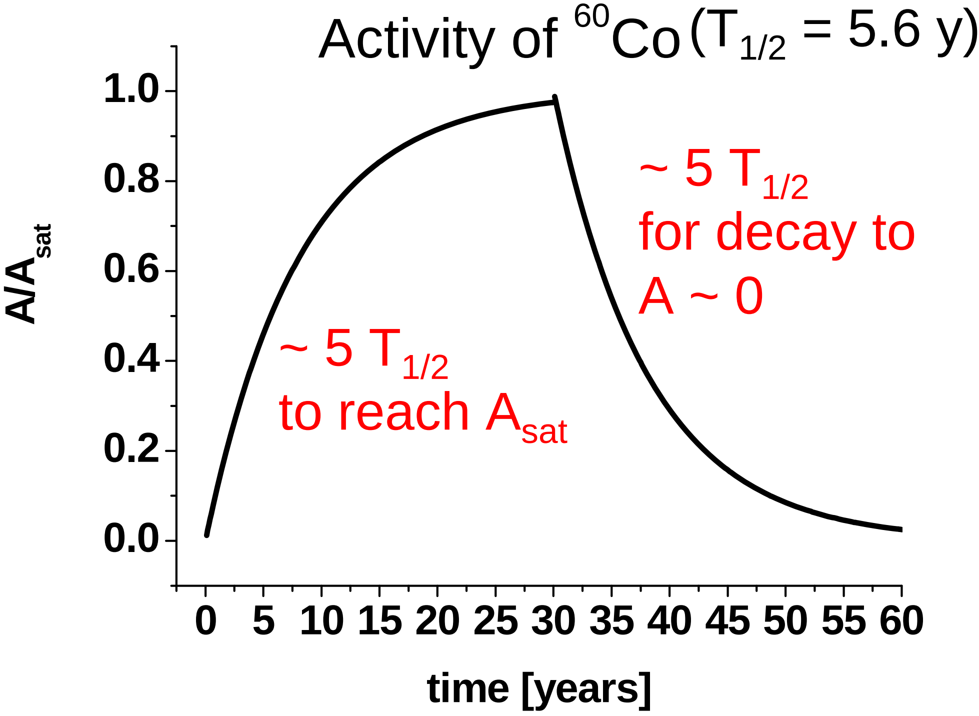
P: (constant) production rate

Solution of the differential equation

$$A(t) = P\left(1 - e^{-\lambda t}\right)$$

$$t \rightarrow \infty : A \rightarrow P = A_{sat}$$

A_{sat}: saturation activity


Activity of ^{60}Co ($T_{1/2}$ = 5.6 y)

~ 5 $T_{1/2}$ to reach A_{sat}

~ 5 $T_{1/2}$ for decay to A ~ 0

For the same production rate P (i.e. same beam current):
It is not possible to produce an activity > A_{sat}

Special case: short irradiation periode

i.e. $t_{irr} << T_{1/2}$ t_{irr}: irradiation time

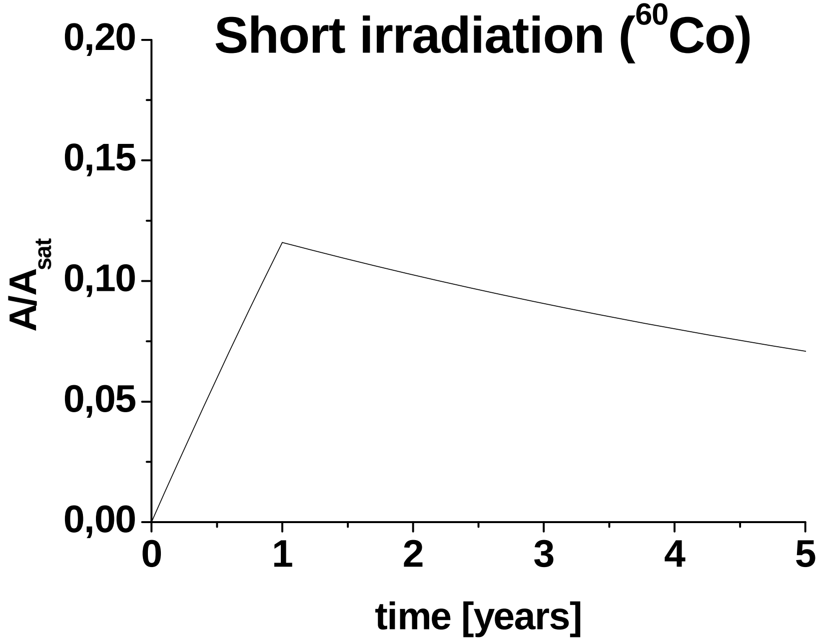
$$A(t_{irr}) = P\left(1 - e^{-\lambda t_{irr}}\right) \cong P\left(1 - \left[1 - \lambda t_{irr}\right]\right) = P\lambda t_{irr}$$

→ activity increases linear with irradiation time $t_{irr,}$
→ activity much smaller than A_{sat} for long-lived isotopes

Decay of $A_{irr}=A(t_{irr})$:

$$A(t_c) = A_{irr} e^{-\lambda t_c} \underset{\uparrow}{\sim} \frac{t_{irr}}{T_{1/2}}\left[1 - \frac{t_c}{T_{1/2}}\right]$$

$t_c << T_{1/2}$

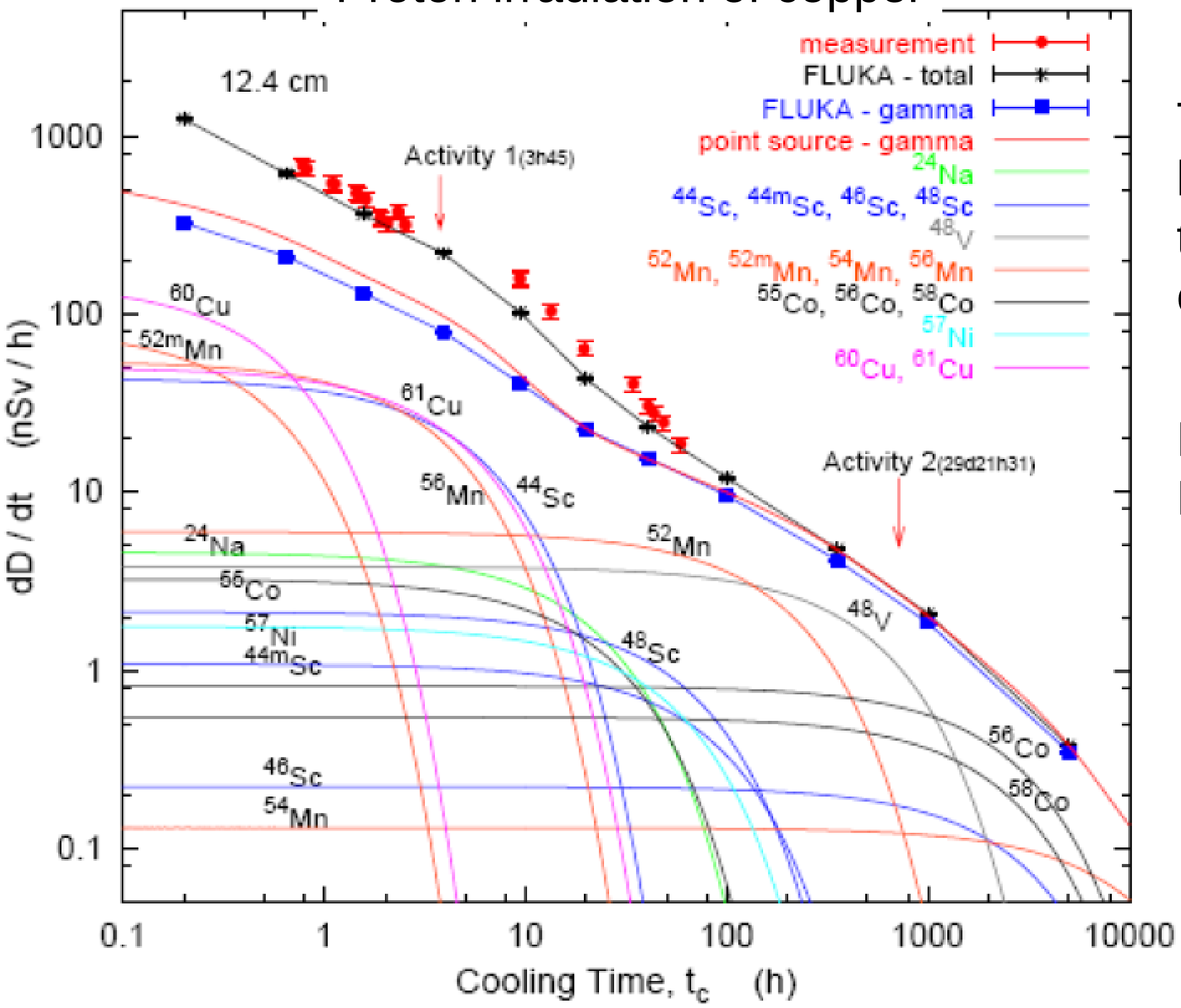


t_c: cooling time

It takes much longer for decay, but

short-lived isotopes, i.e $T_{1/2}=t_c/5$ are completely decayed

Relevant isotopes



Proton irradiation of copper

The isotope with the largest contribution to the γ-dose rate changes with time.

Nuclide inventory Depends also on the Irradiation time

talk from L. Ulrici, CERN

Activity A → dose rate

For point source
(e.g. a hot spot):

$$\dot{H}^* = \Gamma_{H^*} \frac{A}{r^2}$$

γ-dose
conversion factor

Example: A(^{60}Co) = 10^6 Bq

r = 10 cm
→ $\boldsymbol{H^*}$ = 0.036 mSv/h
→ = 36 μSv/h

Rule of radioprotection:
Keep distance!

γ-dose conversion factor Γ_{H^*}
(from the Swiss radioprotection regulation):

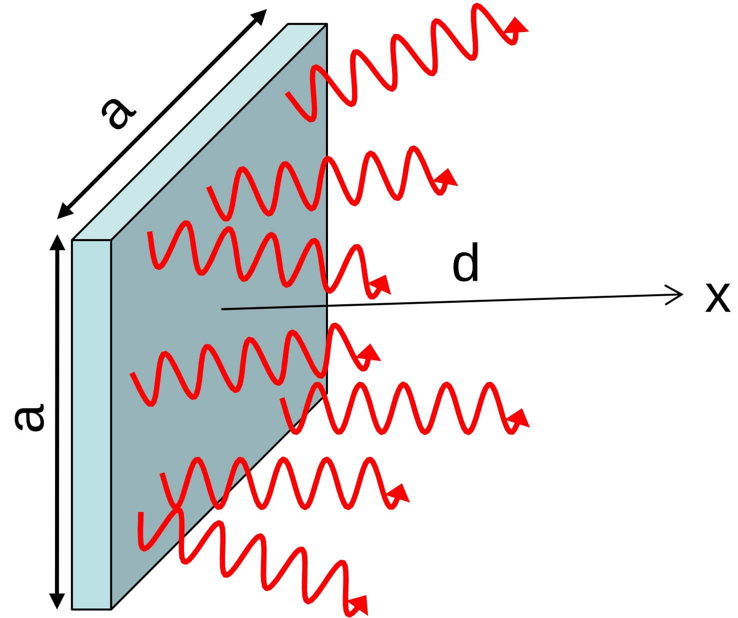
Nuclide	Γ_{H^*}(mSv/h/GBq m^2)	E_γ (MeV)
^{60}Co	0.366	1.2, 1.3
^{22}Na	0.330	1.3
^{54}Mn	0.126	0.83
^{7}Be	0.008	0.48
^{152}Eu	0.179	many
^{154}Eu	0.185	1.3 (35%), 1.0 (30%)

values depend on energy of γ
(and sometimes on follow-up products)

Dose rates at extended objects

only very simple geometries can be solved analytically
in general: very complicated!
use ray trace programms or Monte Carlo simulations

Simple example: large plane



γ–flux at surface:
ϕ_o (1/cm²/s)

1) Determine γ-flux $\phi_P(x)$:

$$\Phi_P(x) = \frac{\Phi_o}{4} \ln\left(1 + \frac{a^2}{4d^2}\right)$$

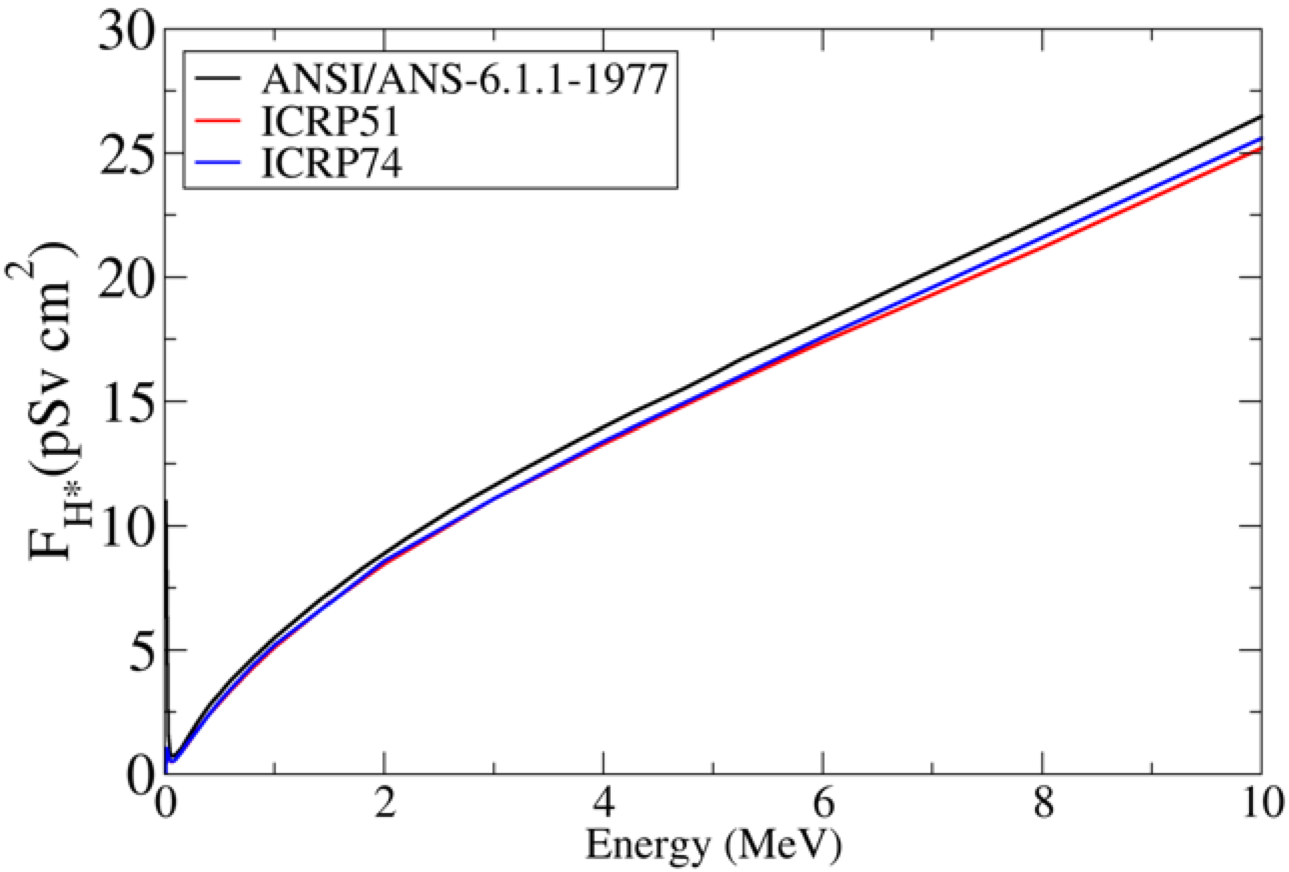
2) $\phi_P \rightarrow \boldsymbol{H^*}$

$$\boldsymbol{H^*} = F_{H^*}\,\phi_P$$

flux-to-dose conversion factors

PAUL SCHERRER INSTITUT

Flux-to-dose conversion factors F_{H*}

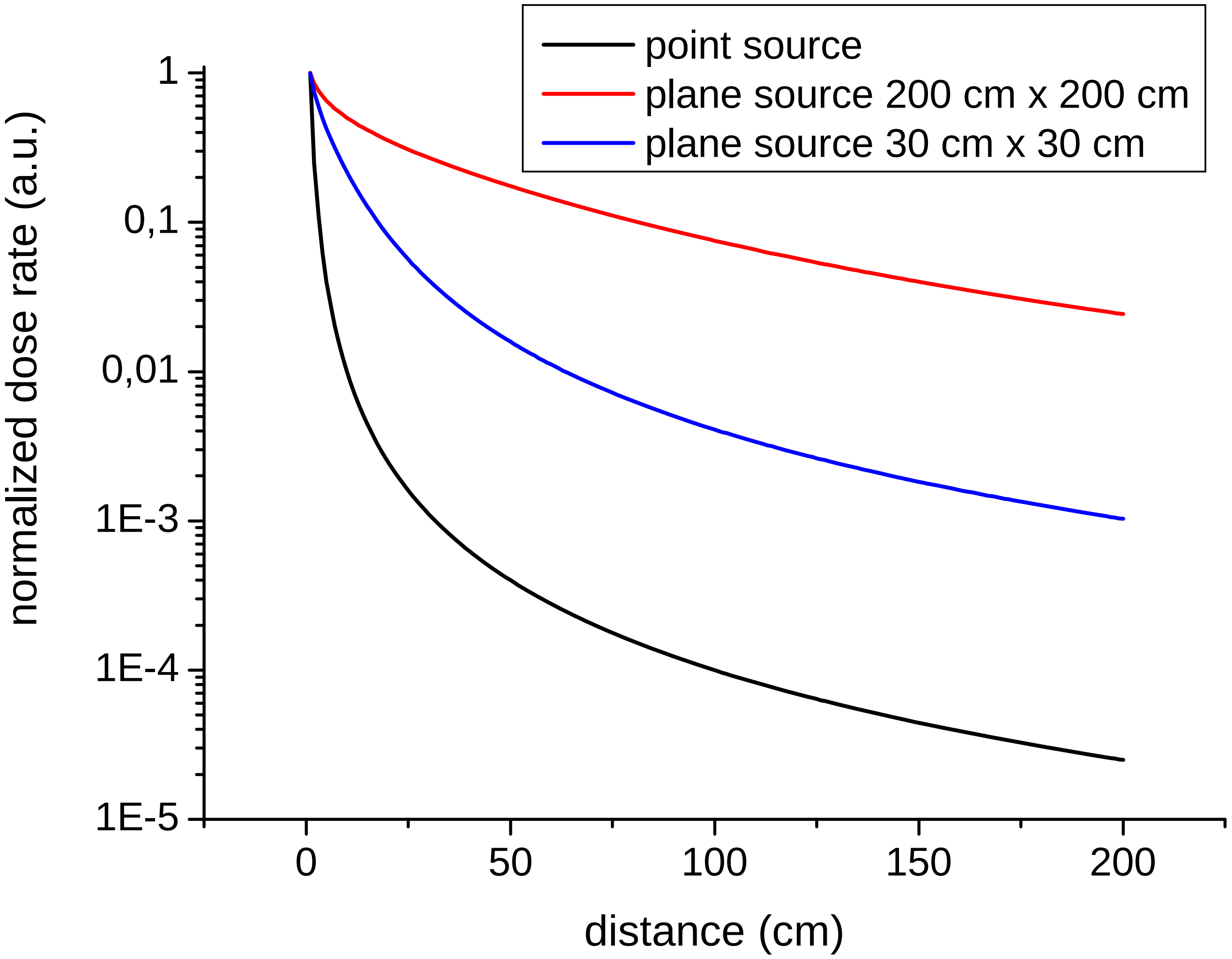


The harder the radiation, the larger the dose (more damage)

$$\boldsymbol{H^*} = F_{H*}\,\phi_P$$

works also for prompt radiation!

Comparison point source – plane source



in 1 m distance:

point:	0.0001
30x30:	0.004
200x200:	0.075

At larger distance (d > 2 a) from the plane source it behaves like $1/d^2$.

Examples for activation

Short-time irradiation: 20 days 100 nA protons of 72 MeV on

316L: 40 mSv/h

in 10 cm distance,
12 h cooling

Co content: 0.16%

→ Activation of steel increases with Co-content !

Long-time irradiation: 20 years 100 nA protons of 72 MeV on

Cu:	24 mSv/h
Magnet-Fe:	140 mSv/h
Graphite:	1.3 mSv/h

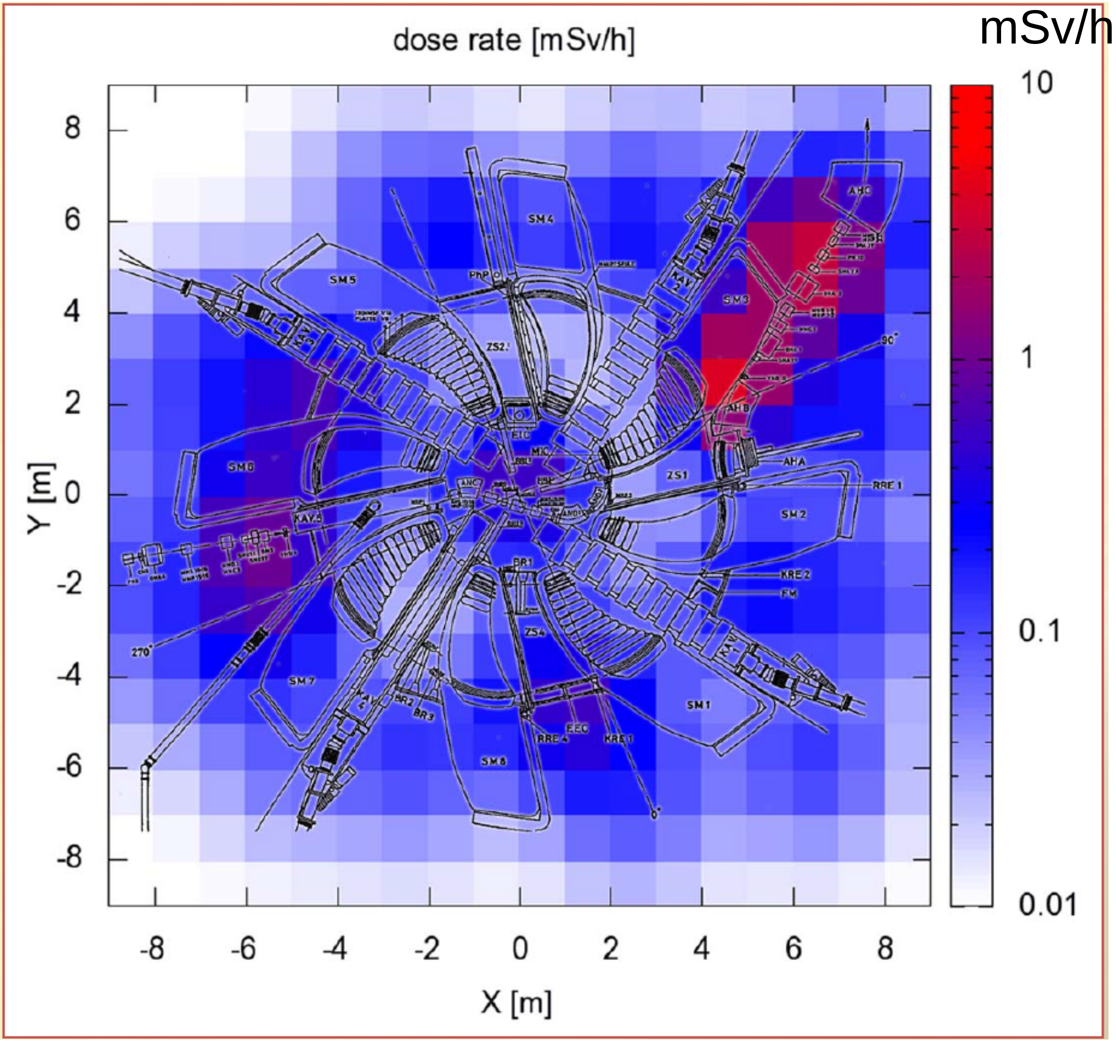
in 10 cm distance
after 64 h cooling

→ It is important to consider later activation already in the design phase.

→ Choose materials with lower activation, if possible.

Doses are calculated with MCNPX/Cinder90,
typically within a factor 2 agreement to the measurement

Examples for activation: Ring cyclotron at PSI



compiled by M. Seidel from 30 data points

For maintenance/repairment:
good planning of work procedure
→ 50 -300 μSv per mission
and person

In 2011/2012 shutdown:
total collected dose: 40.5 mSv
participating people: 149
→ in average: 0.27 mSv

highest personal dose: 3.2 mSv

When is a material radioactive?

1. Ambient dose equivalent:

$H^*(10) > 0.1\ \mu\text{Sv/h}$

OR

2. Activity: decays/sec, unit: Bq

$$\sum_i \frac{A_i}{R_i} > 1$$

(Sum rule)

A_i: specific activity [Bq/g]
R_i: exemption limit
given in the radioprotection regulation

OR

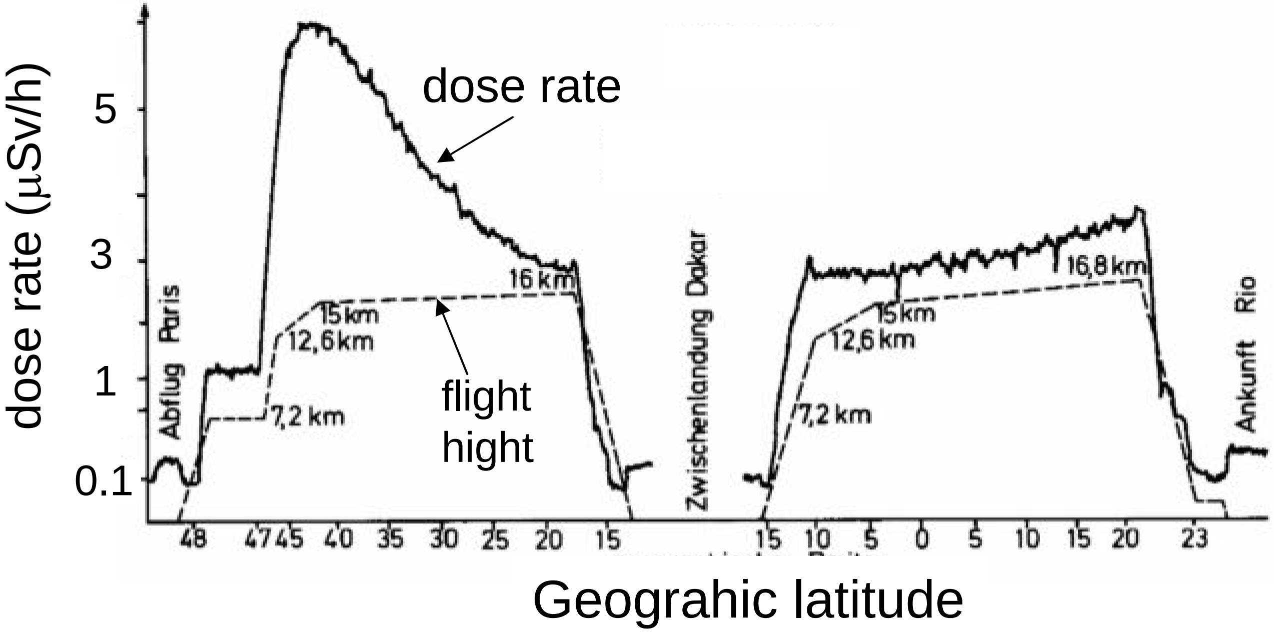
3. Surface contamination:

- $> 1\ \text{Bq/cm}^2$ in case of unidentified β- and γ-emitters
- $> 0.1\ \text{Bq/cm}^2$ in case of unidentified α-emitters
- > CS-value (given in regulation) for specific isotope

Legal dose limits

for workers: 20 mSv/year
150 mSv for eye lense
500 mSv for skin (on 1cm^2), hands, feet
for public: 1 mSv/year

Flight from Paris to Rio de Janeiro and back: ~ 0.1 mSv/h



aircraft personal: get ~ 3 mSv/year

Effective doses for the population

Terrestical source:

Cosmic rays,
food,
natural isotopes in soil and air

Worldwide average: 2.4 mSv

Kanada:	1.8
Great Britain:	2.2
Germany:	2.2
USA:	3.0
Switzerland:	4.3
India (Cerala Coast):	12.5
China (YangJiang):	6.3
Worldwide range:	1 – 10

CONSIDERABLE VARIATION

- 25 % < 1 mSv
- 65 % < 3 mSv
- 10 % > 3 mSv

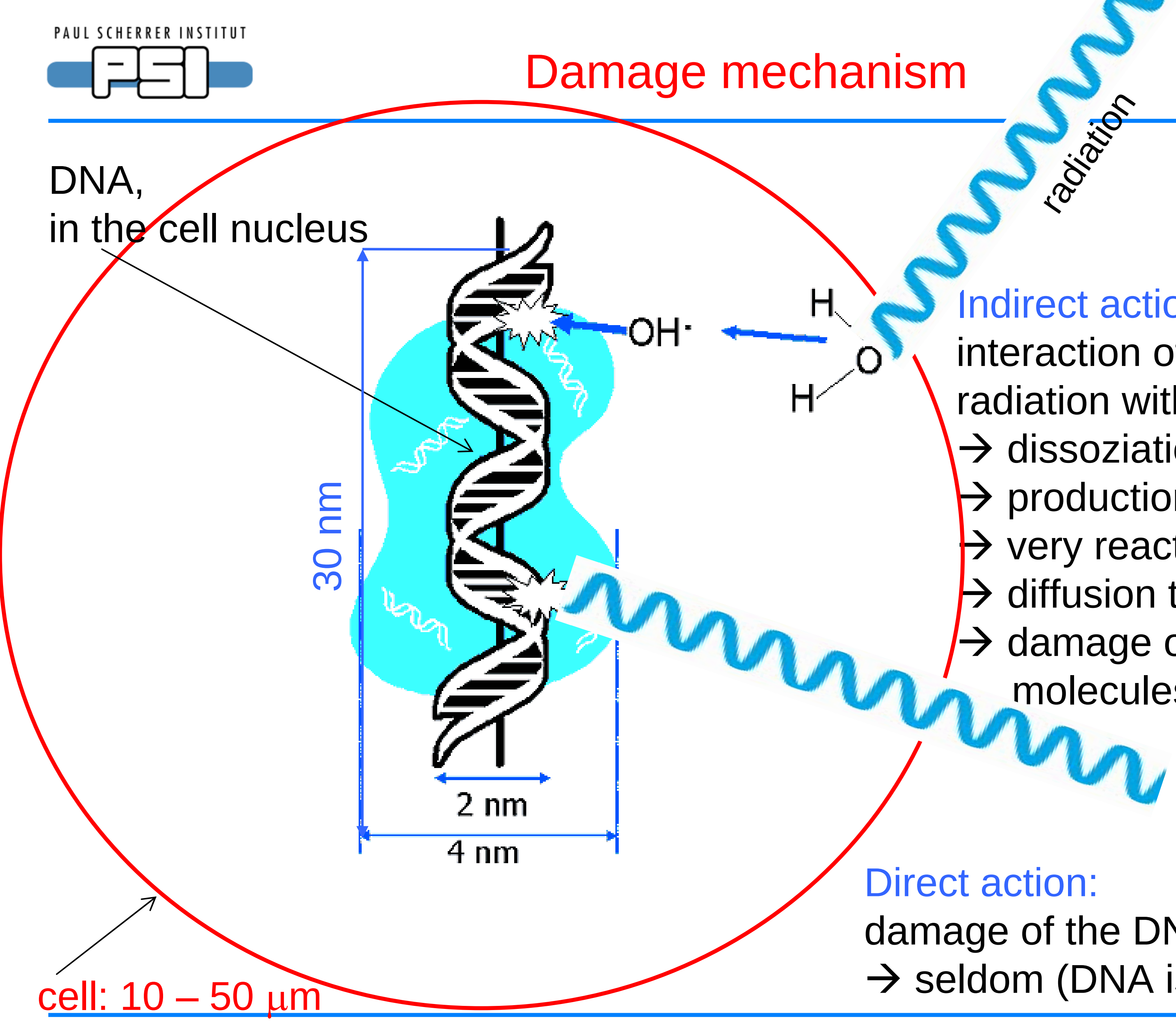
+ medical treatment: 0.5 – 3 mSv

Medical exposures:

examination	**mSv**
CT	
- Head	2
- Spine	6
- Chest	7
- Abdomen	15
x-ray	
- Dental	0.005
- Chest	0.1
- Spine	1.5

treatment of tumors:
20 – 60 Sv

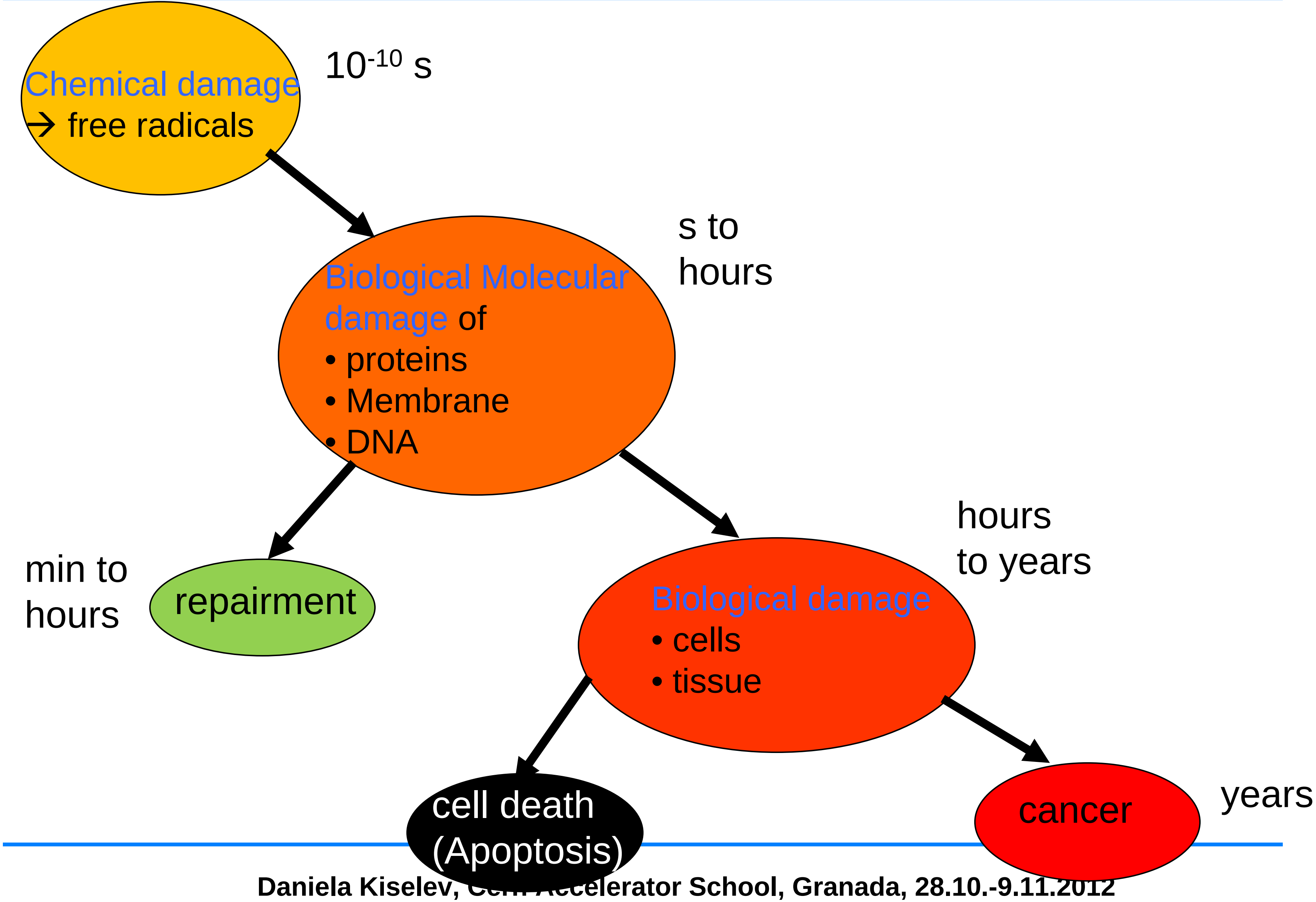
Damage mechanism



Indirect action:
interaction of the radiation with the cell water
→ dissoziation of H_2O
→ production of radicals
→ very reactiv
→ diffusion to DNA
→ damage of DNA & other molecules

Direct action:
damage of the DNA
→ seldom (DNA ist very small)

The time scale



Chemical damage
→ free radicals

10^{-10} s

Biological Molecular damage of
- proteins
- Membrane
- DNA

s to
hours

min to
hours

repairment

Biological damage
- cells
- tissue

hours
to years

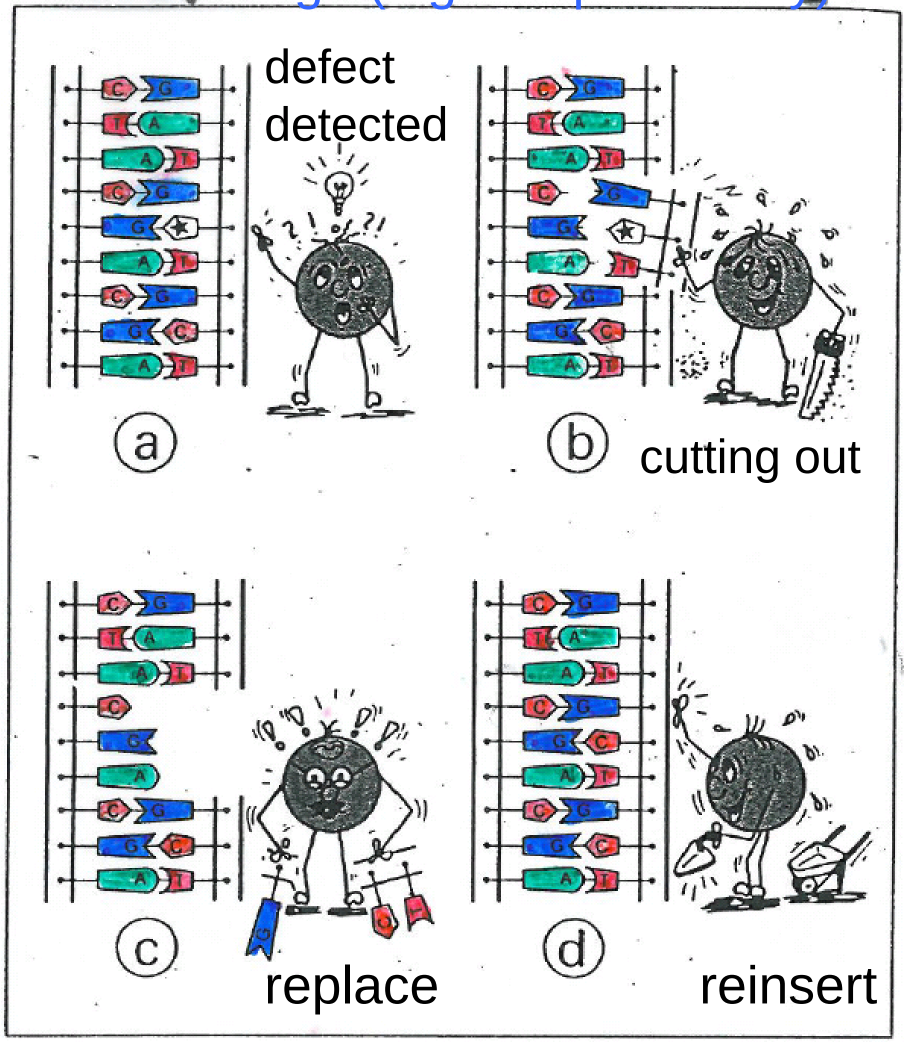
cell death
(Apoptosis)

cancer

years

DNA-Repairment (very simplified)

Base damage (highest probability):



Bases:

Cytosin Guanin is a pair

Thymin Adenin is a pair

- small parts are replaced in minutes

- large parts and double-strand repairment can take hours

Some numbers: Effective dose and cell defects

Average effective dose in CH:
5.6 mSv/year = 15 μSv/day
produces ~ 4 10^{14} radicals/day in the „standard“ human

Human has ~10^{13} cells → 40 defects/cell per day
most of them get repaired
10^{-2} -10^{-3}/cell faulty or unrepaired
→ in 2.5 to 25 days 1 cell has a persistent defect

Cell reproduction mechanism:

- 1 DNA/cell
- DNA has ~3 10^{9} base pairs
- replication of the DNA in 1-30 days
  → 1 -10 spontaneous mutations (important for the evolution of life)
  → in < 30 days 1 cell has a persistent defect

Damaged cells should declare cell death (Apoptosis).

Prediction of cancer mortality

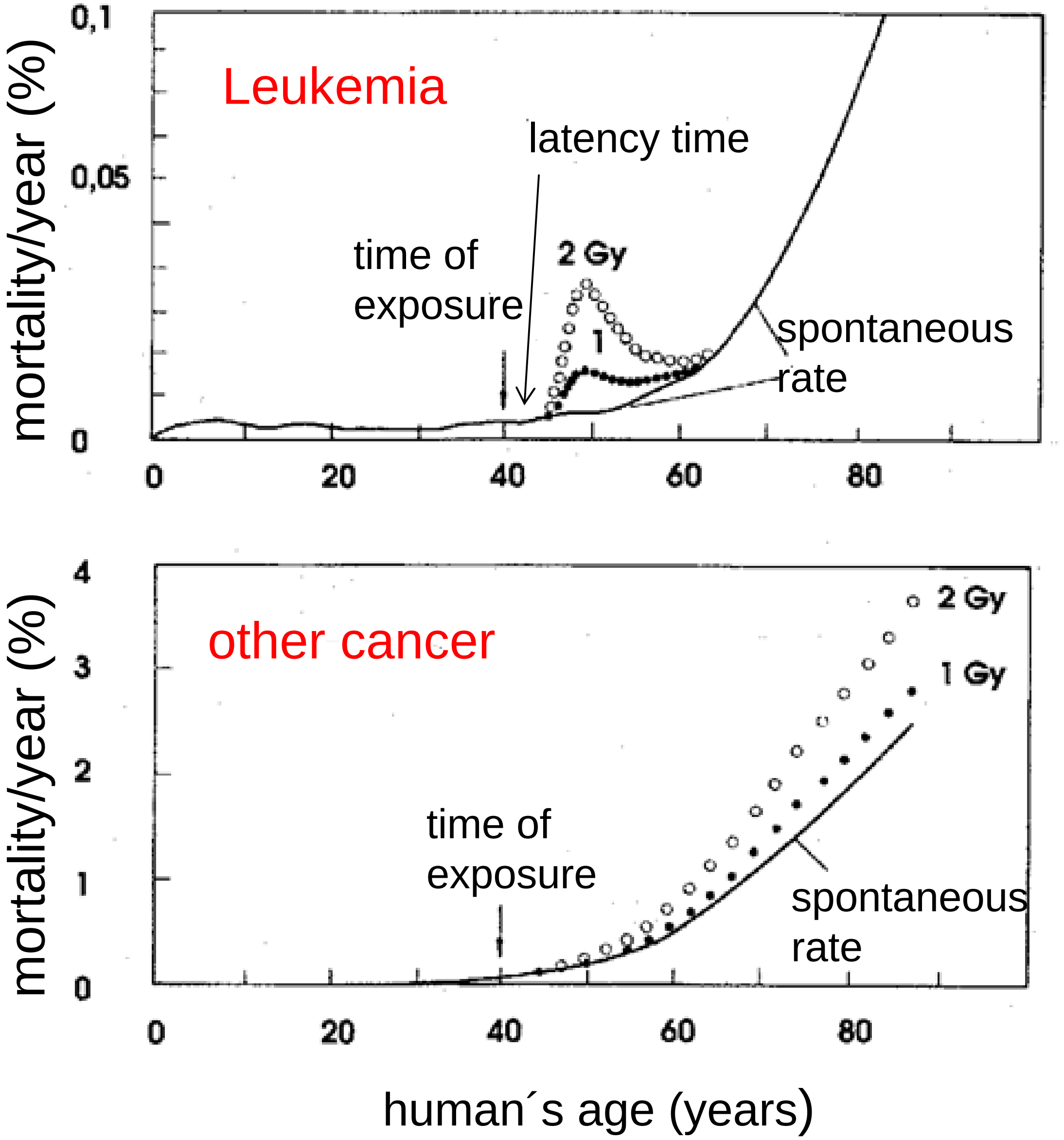
very difficult!

depends on many parameters:

- cell cycle
- kind of cell
- Chemicals can increase or decrease the effect
- Chemicals can prevent repairment mechanism
- O_2 content
- pH value in cell
- human´s age
- exposure time and pattern
- spatial distribution of dose

stochastic effect:

+5% cancer mortality/Sv within 40 years



Summary of Radiation

Prompt radiation: all kind of particles when beam on
important: photons & high energetic neutrons
shielding: high Z material against photons
neutrons: iron & concrete

Residual radiation: beam off,
buildup depends on the irradiation history
important: γ-radiation
dose decreases with time and distance

Definition of several doses: purpose of quantifying the damage
most damaging are heavy particles
The measured dose H*(10) should represent the dose in the human.

cancer mortality: +5% /Sv within 40 years
„natural“ risk: 25 %

Summary of Radioprotection rules: ALARA

As Low As Reasonable Achievable

If possible,

- use (additional) shielding
- use remotely operating devices
- reduce time with work planning
- enter after reasonable cooling time
- keep distance (particularly from hot spots)
- use protection clothes against contamination of skin
- use protective mask and breathing apparatus to avoid
  - inhalation (particularly tritium, when opening vacuum systems or entering closed rooms)
- exposures can be significantly reduced, if considered in design phase