Nuclear data for nuclear technologies and applications

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In the course

ARIEL-H2020 1st international on-line school on nuclear data: the path from the detector to the reactor calculation – NuDataPath 2022

Index of content:

Scope of nuclear technologies and applications

Reactor design, safety and operation.
Including advanced nuclear reactors (Gen II, Gen III, Gen IV FR, ADS)
Nuclear data for transport calculations

Depletion calculations during burnup and for disposal
Nuclear data for depletion calculations
Advanced fuel cycles, Partition and Transmutation and Nuclear Data

EU research for nuclear data: SANDA

ND for nuclear technologies and applications

- Nuclear reactors
- Nuclear fuel cycle facilities and processes
- Naval applications
- Space applications
- Radioisotope power systems
- Health diagnostic and treatment
- Isotope production
- (n, γ,..) Radiography
- Radiation protection
- Neutron interrogation (prompt activation gammas)

Taking into account irradiation and tracing with radioactive isotopes

+ 8/17 UN Sustainable Development Goals

Plague control, quality verification, tracing materials and transport, origin of materials, decontamination, pollutants, non destructive tests, contaminants, soil quality, crops, insect sterilization, marine food chain,…

Uses of the ND for the design of nuclear reactors

- Simulation of the transport of n, γ, e and heavy ions
- Criticality (transport with high precision data need)
- Moderation and thermalization (elastic, inelastic, molecular effects)
- Evaluation of neutron sources (fission, (n,xn), (α,n), delayed-n,…)
- Energy deposition / Heat sources (fission, capture, decay,…)
- Residual heat after shut down (decay,..)
- Shielding for n, γ, e and heavy ions
- Materials activation (n-capture, (n,X), …)
- Fuel composition evolution with burn-up (capture,(n,X),decay,…)
- Reloading plans
- Control rod worth
- Reactivity coefficients
- Material damage (Kerma, PKS, dpa)
- Personnel radioprotection
- Reaction rate
- Interpretation of detector measurements
- …

Neutron transport equation

$$\frac{1}{v}\frac{\partial\varphi}{\partial t} + \widehat{\Omega}\cdot\vec{\nabla}\varphi + \Sigma_T(\vec{r},E)\varphi(\vec{r},E,\widehat{\Omega},t)$$
$$= \int_{4\pi} d\hat{\Omega}' \int_0^\infty dE'\Sigma_s(\vec{r},E'\to E,\widehat{\Omega}'\to\widehat{\Omega})\varphi(r,E',\hat{\Omega}',t) + s(\vec{r},E,\widehat{\Omega},t)$$

$$(s = s_F + s_{ext}) \quad s_F = \int_{4\pi} d\hat{\Omega}' \int_0^\infty dE' \frac{\chi(E,E')}{4\pi}\nu(E')\Sigma_F(\vec{r},E')\varphi(r,E',\hat{\Omega}',t)$$

Data appearing in the equation:

Σ_s Scattering cross sections including elastic, inelastic, and eventually (n,n+X) like (n,2n) or (n,np)

Σ_T Total cross sections also $\Sigma_T = \Sigma_s + \Sigma_{Abs} = \Sigma_s + \Sigma_F + \Sigma_C + \cdots$

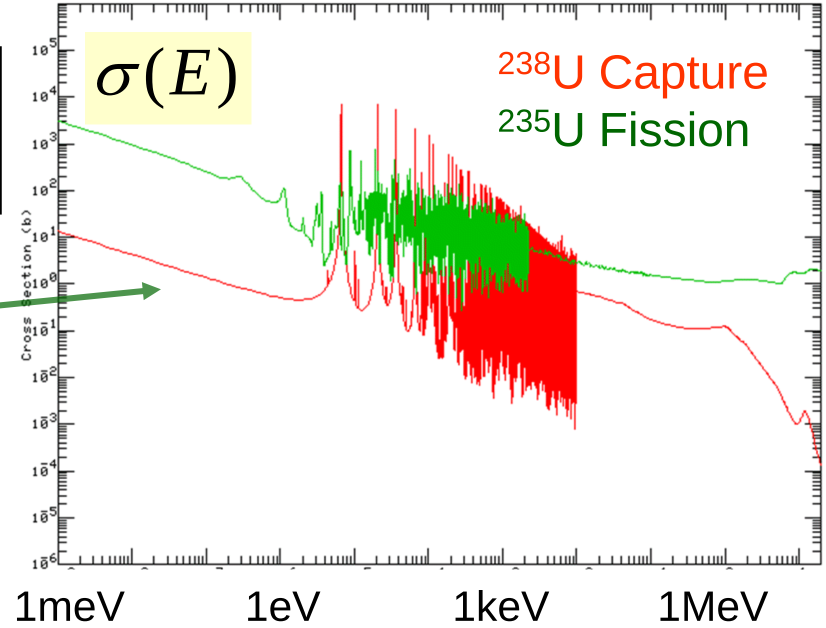
Σ_F Fission cross sections, could include other n-sources reactions like (n,2n),…

$\nu(E)$ Fission neutron multiplicity

$\chi(E,E')$ Fission neutron energy (prompt or prompt + delayed)

$$Reaction\ Rate = \int n\,\sigma(E)\,\phi(E)\,dE$$

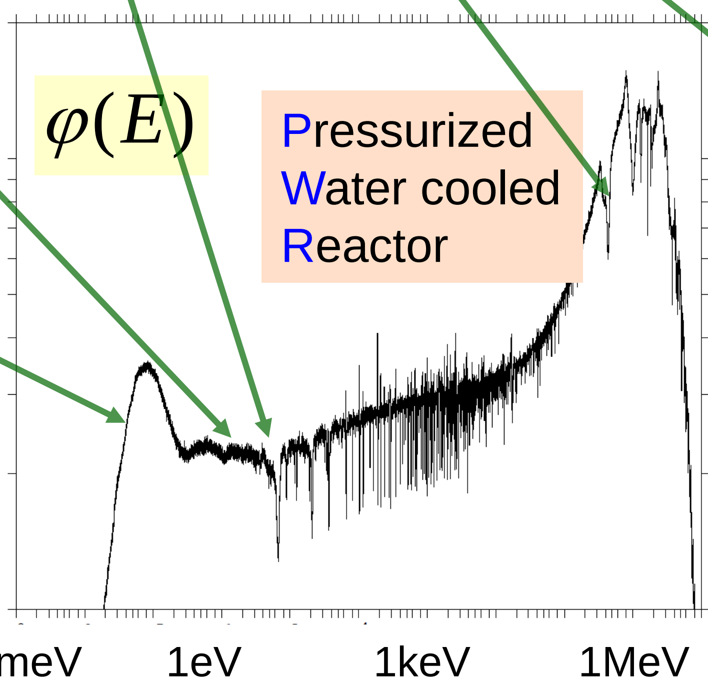
- n- induced fission
- neutron capture

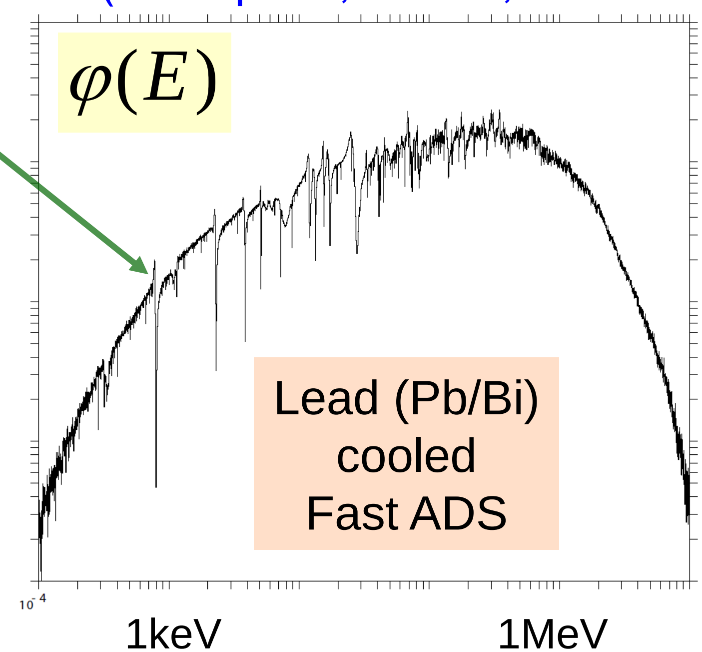


- elastic and inelastic neutron scattering

Resonances (absorption, elastic, inelastic,…)

Thermalization, Moderation





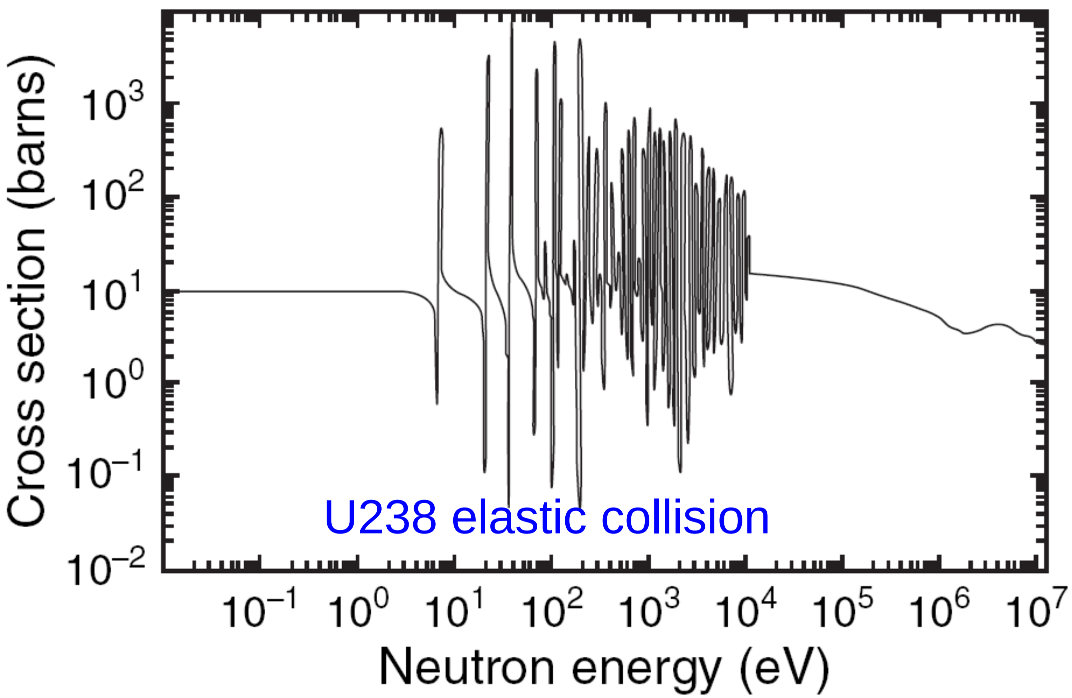
Scattering cross sections

Scattering in all materials (fuel, coolant, moderator, shielding, structural materials, neutron absorbers and control rods)

Elastic cross section: all the energy range from thermal to several MeV

Point wise, potential and resonances (positive or negative interference)

$$\sigma_n(E) = \sigma_o \frac{\Gamma_n}{\Gamma} \frac{1}{1+4(E-E_r)^2/\Gamma^2} + \sigma_o \frac{2R}{\lambda_o} \frac{2(E-E_r)/\Gamma}{1+4(E-E_r)^2/\Gamma^2} + 4\pi R^2.$$



Anisotropy at high energies

Inelastic cross sections (n, n') to different levels.

Molecular effects at thermal energies **S(α,β)**

(n, xn) is sometimes treated as scattering + neutron multiplication vs. absorption + neutron source

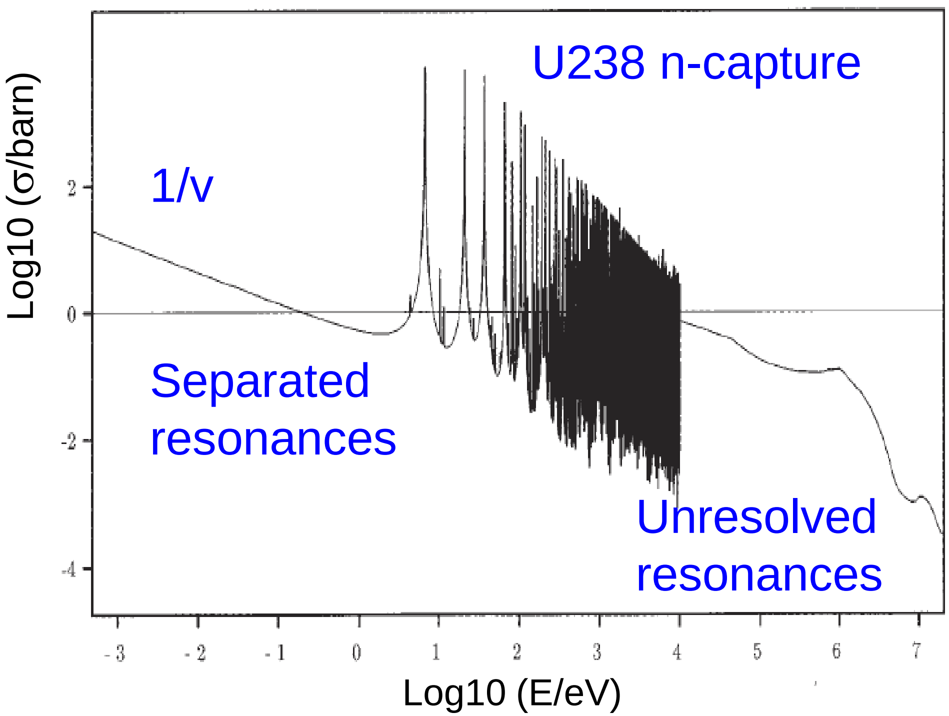
Absorption cross sections: Capture

Capture in all materials (fuel, coolant, moderator, shielding, structural materials, neutron absorbers and control rods)

Capture cross section: all the energy range from thermal to several MeV

Point wise and resonances

$$\sigma_\gamma(E) = \sigma_o \frac{\Gamma_\gamma}{\Gamma} \left(\frac{E_r}{E}\right)^{1/2} \frac{1}{1 + 4(E - E_r)^2/\Gamma^2}$$



Energy difference dissipated by gammas- **Total energy deposited**

Detailed level feeding and decay cascades needed for some applications

Isomers production branching ratio *vs. E*

Large contribution from **thermal** and epithermal energies for thermal reactors (Gen II and Gen III)

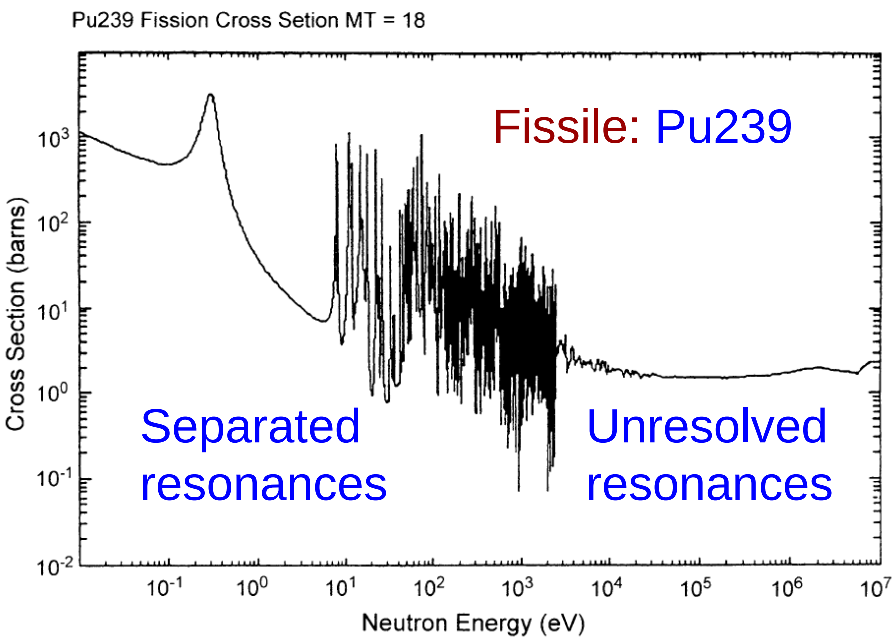
Capture in actinides contribute to HLW and in general to activations of materials

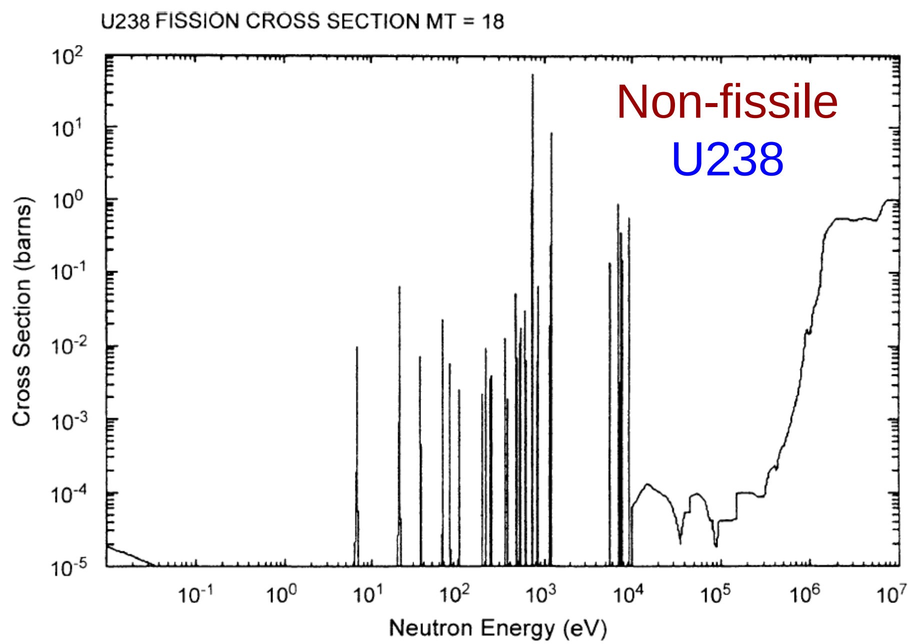
Fission: Absorption cross section + neutron source + energy + wastes

Fission only for **actinides** (and heavy isotopes) (Fuel, coolant, spallation target)

Fission cross section for <u>fissile</u> isotopes: all the energy range from thermal to several MeV: Point wise and resonances

Fission cross section for <u>non-fissile</u> isotopes: a threshold in the MeV range with pointwise data plus some small subthreshold resonances.

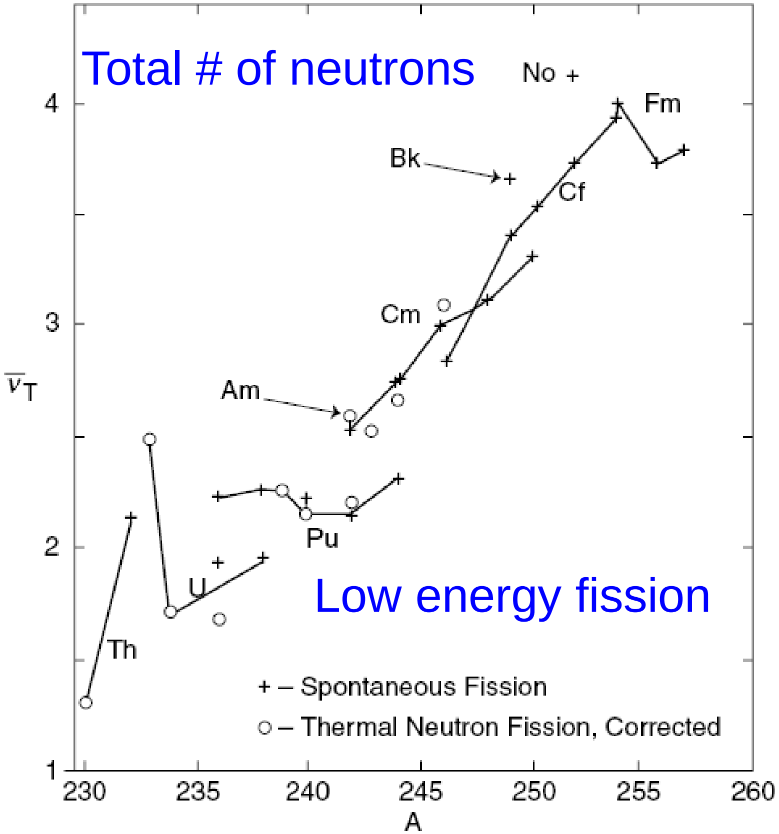


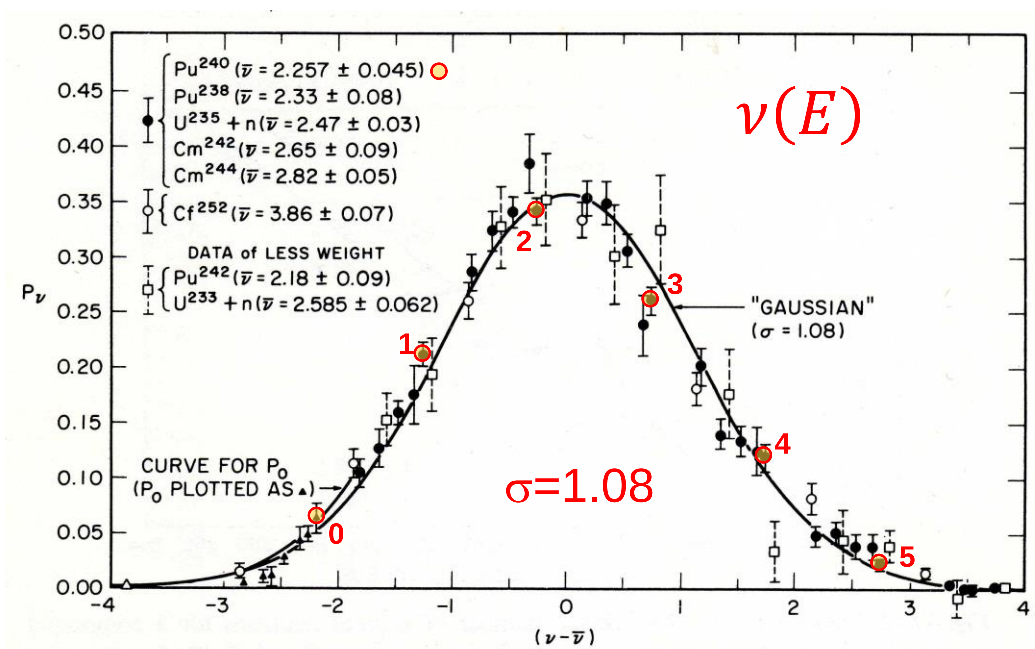


Fission: Absorption cross section + neutron source + energy + wastes

Multiplicity of prompt neutrons after fission:

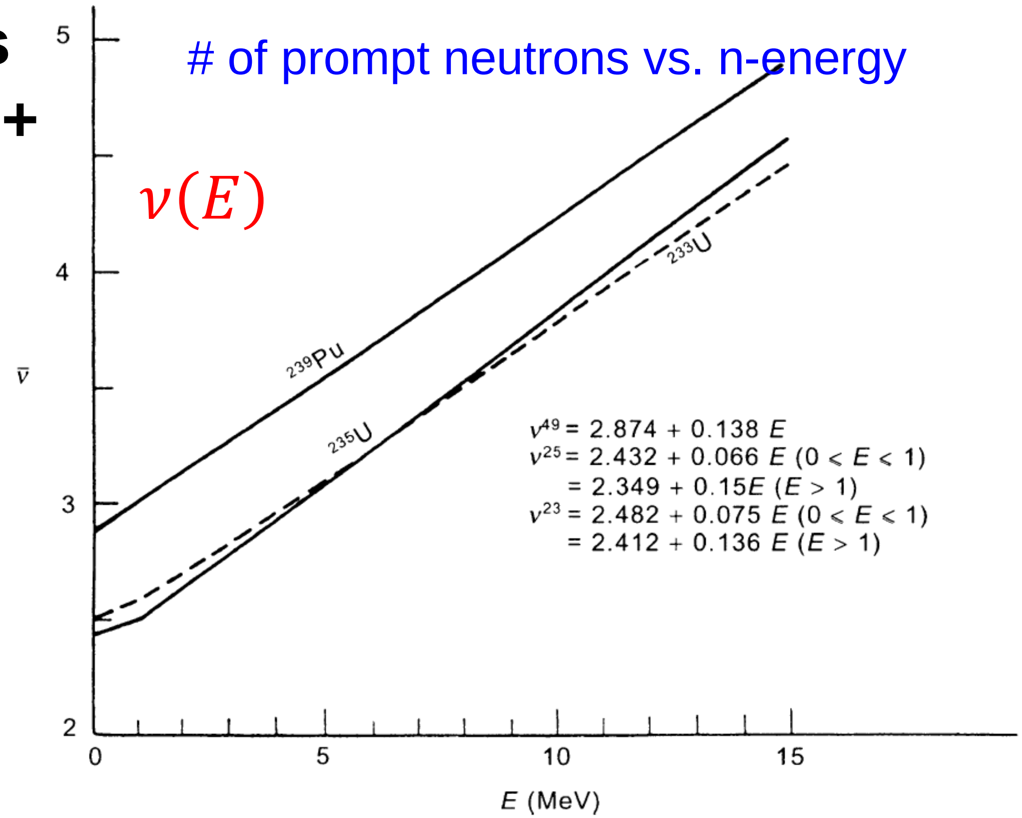
- Number of neutrons after each fission follows a random distribution
- Average number of neutrons depends on isotope and on n energy





Distribution of the number of prompt neutrons

Fission: Absorption cross section + neutron source + energy + wastes



Multiplicity of prompt neutrons after fission increases with E for fast neutrons

Delayed neutrons:
- By precursors decay
- Added by time groups

Delayed neutrons

$$^{87}Br \Rightarrow {}^{87}Kr^* \Rightarrow {}^{86}Kr + {}^{1}n$$

(54.5 s)

$$^{137}I \Rightarrow {}^{137}Xe \Rightarrow {}^{136}Xe + {}^{1}n$$

(21.8 s)

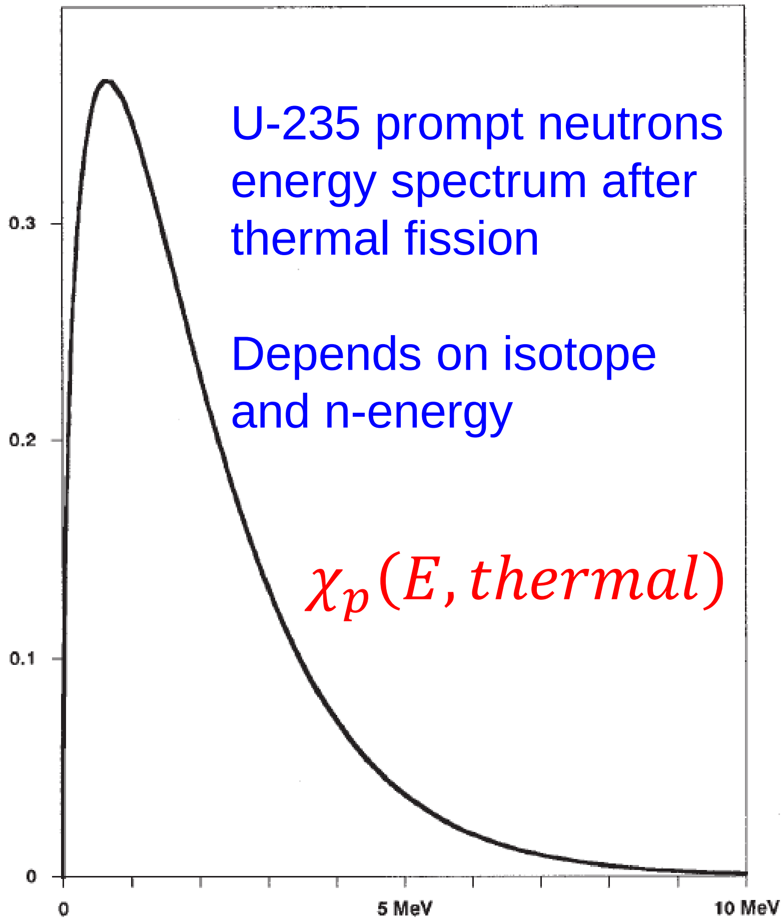
U235

Group	β (pcm)	T (s)	τ (s)
1	24	54.5	78.6
2	123	21.8	31.5
3	117	5.98	8.62
4	262	2.23	3.22
5	108	0.495	0.714
6	45	0.179	0.258
Average	679	7.84	11.31

	En	β(pcm)
Th232	F	2433
U233	Th	296
U235	Th	679
U238	F	1828
Pu239	Th	224
Pu240	F	292
Pu241	Th	535

Fission: Absorption cross section + neutron source + energy + wastes

Emitted neutrons energy spectrum :
- Parametrization by Maxwell, Watt, Cranberg or evaporation for prompt neutrons
- Specific delayed neutron spectra for each precursor or time-group

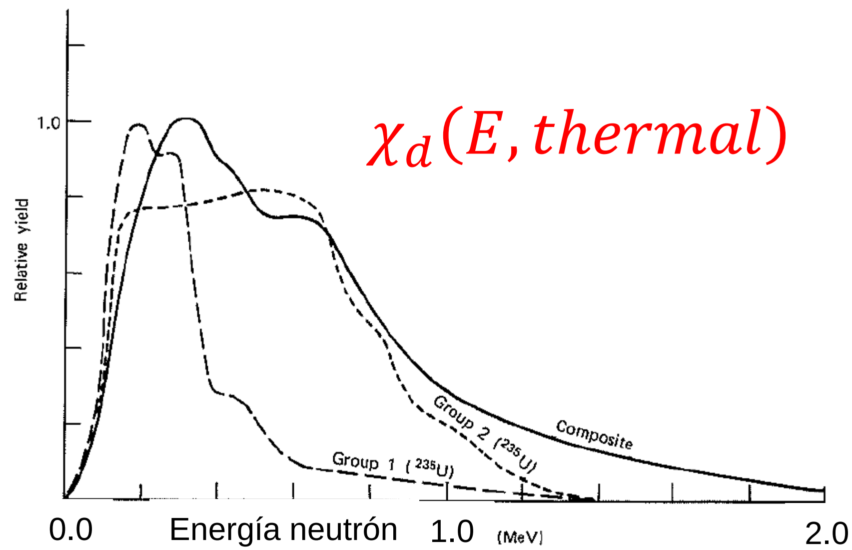


U-235 prompt neutrons energy spectrum after thermal fission

Depends on isotope and n-energy

$\chi_p(E, thermal)$

Delayed neutrons softer spectrum and specific of each precursor or time-group



$\chi_d(E, thermal)$

Other neutron sources

In addition to fission there are other neutron sources for nuclear reactors

Neutron sources from the n transport

- **(n,2n), (n, xn), (n, n'+X):** They are threshold reactions for fast neutrons in heavy and some light materials (fuel, spallation target, heavy coolant, struct. mat.)

Other particles (external sources or radioactivity of fuel and materials)

- **(α,n)** in the fuel (particularly Pu and M.A.)
- **(γ,n) (p,n) Spallation reactions**: typically in heavy and some light materials (fuel, spallation target, heavy coolant, struct. mat.)

Radioactive neutron sources

Data needed:

Probability/cross section of the reaction
Distribution of the # of neutrons per reaction
Energy distribution of produced neutrons
Angular distribution of neutrons and correlation with source particle direction

Fission: Absorption cross section + neutron source + energy + wastes

Fission energy balance details depends on isotope and on n energy

Generated energy (MeV/fission)	
• Fission Fragments:	166.2
• Instantaneous gammas:	8.0
• Neutrons:	4.8
• Beta radiation (electrons):	7.0
• Antineutrinos from beta-decay:	9.6
• Gammas from beta-decay:	7.2
Total:	202.8

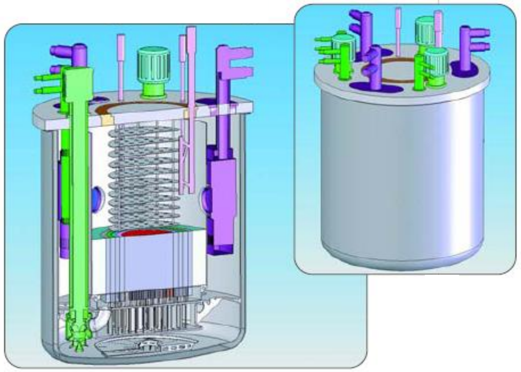
Recoverable Energy (local heat)	
• Generated energy:	202.8
• Antineutrinos from beta-decay:	-9.6
• Gammas from neutron capture:	+8.4
Total:	201.7

Data as integrated local heat deposition per fission as a function of isotope and n (energy),

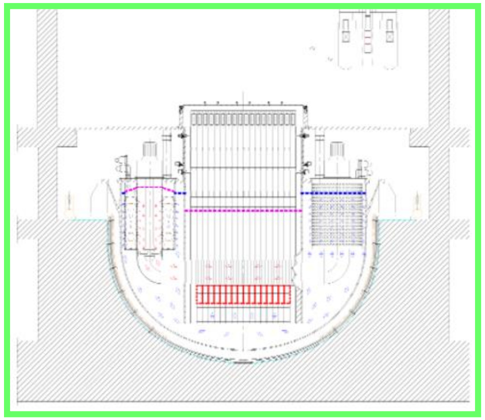
or

Data on number and energy of instantaneous gammas, kinetic energy of FF, plus transport of γ, n, e-, and decay of FF,…

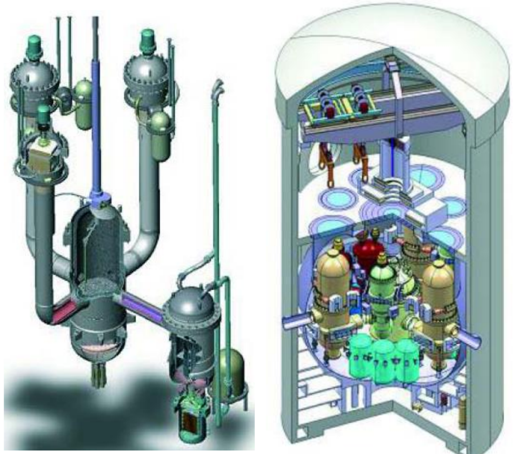
Gen IV Fast reactors



Liquid Sodium : ASTRID
CP-ESFR, SFR-SMART

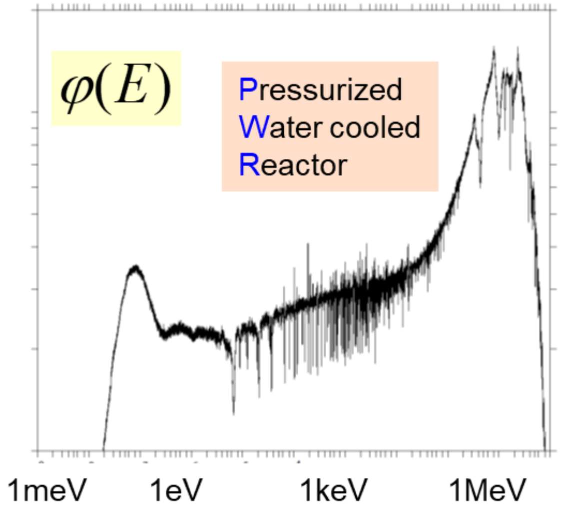


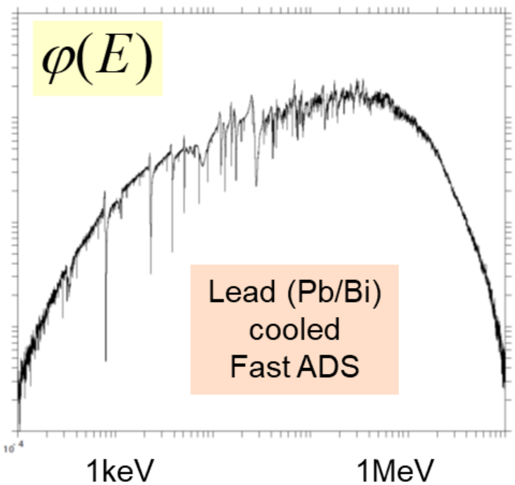
Liquid Lead: ALFRED,
EURATOM ELSYLEADER, CDT, HeLiMnet, MYRRHA



Gas: ALLEGRO
ALLEGRO, GoFastR

Different technology and different functions results in different neutron spectra, different fuel and different materials. In consequence, different data needs and relevance of energy ranges.

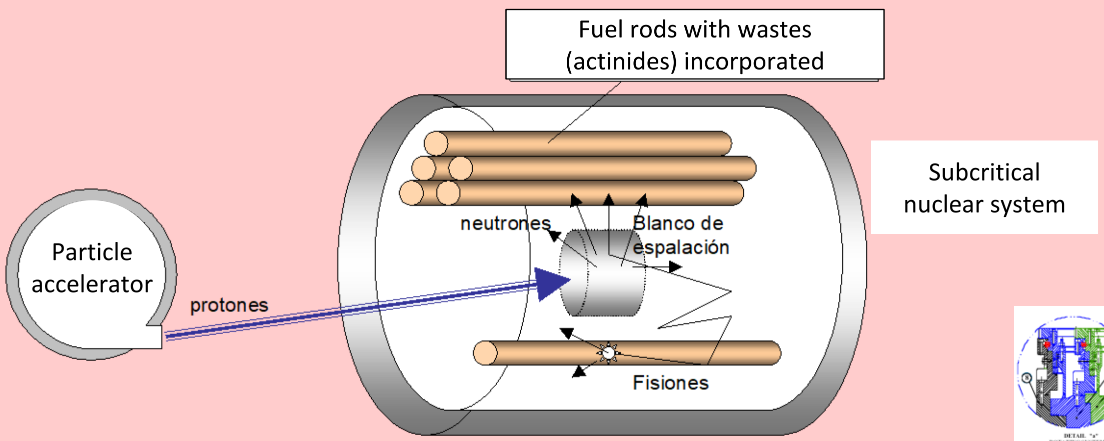


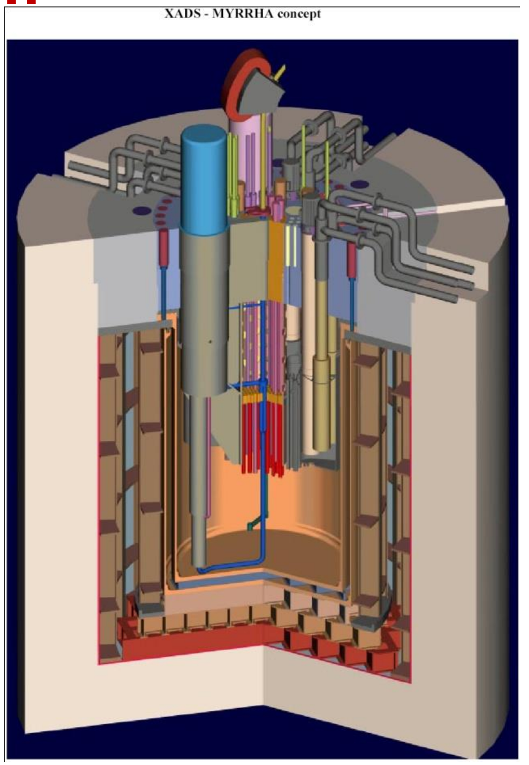


ADS = Accelerator Driven Subcritical System

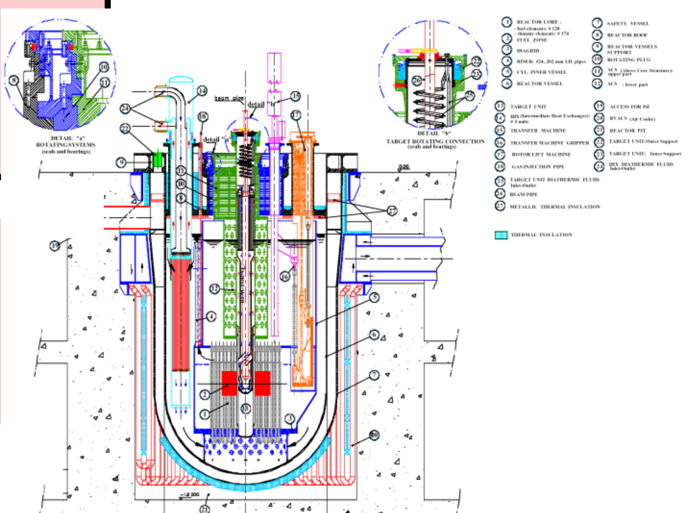
An ADS is a subcritical nuclear system (Keff = 0.95-0.98) in which the power is maintained by a high intensity external source of neutrons, typically produced by spallation induced by high energy protons (near 1 GeV) in heavy nuclei.

In this way, the ADS can operate in a safe way with fuels including high content of Pu and Minor Actinides even in a fast neutron spectrum and with very high burn-up.

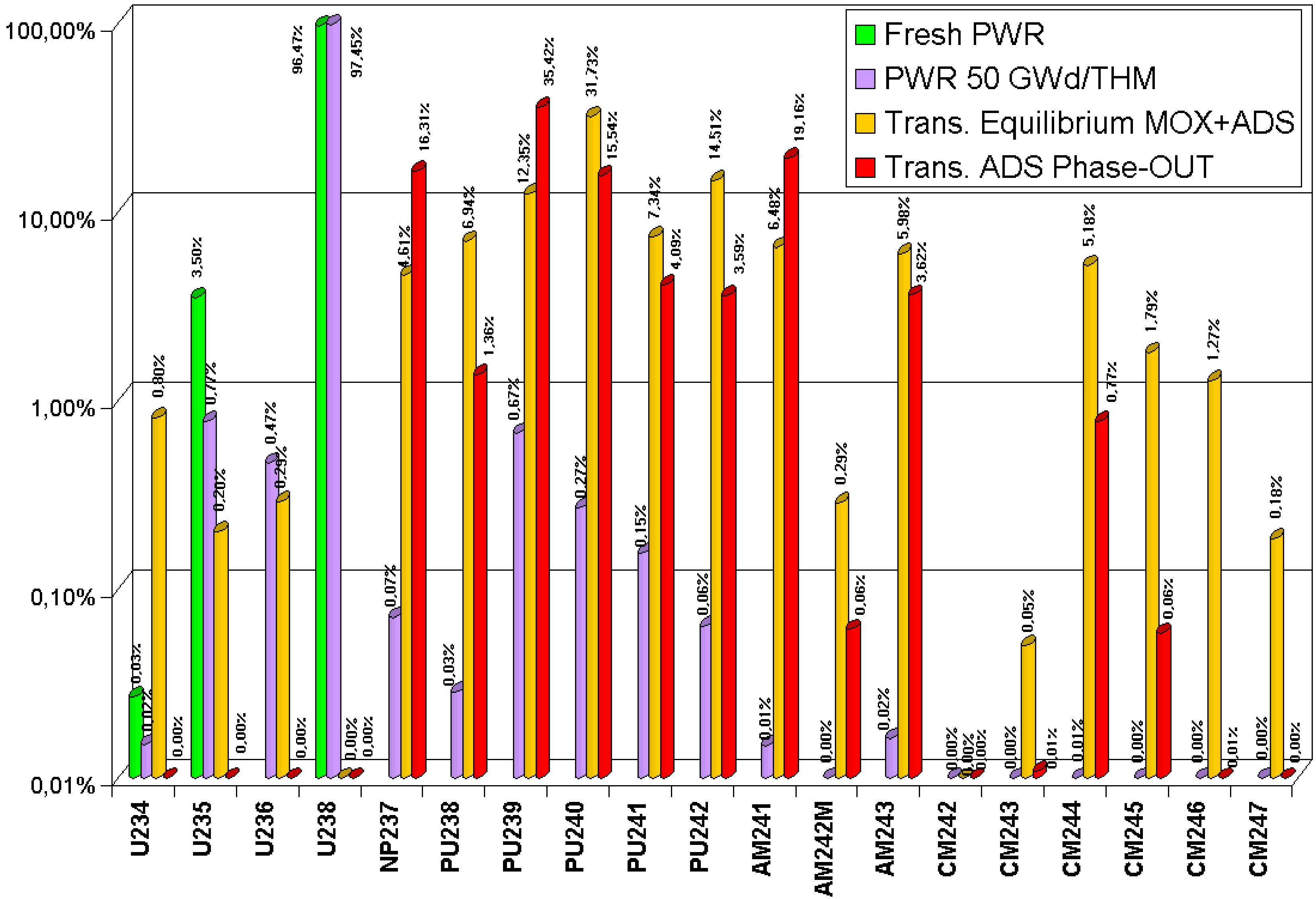


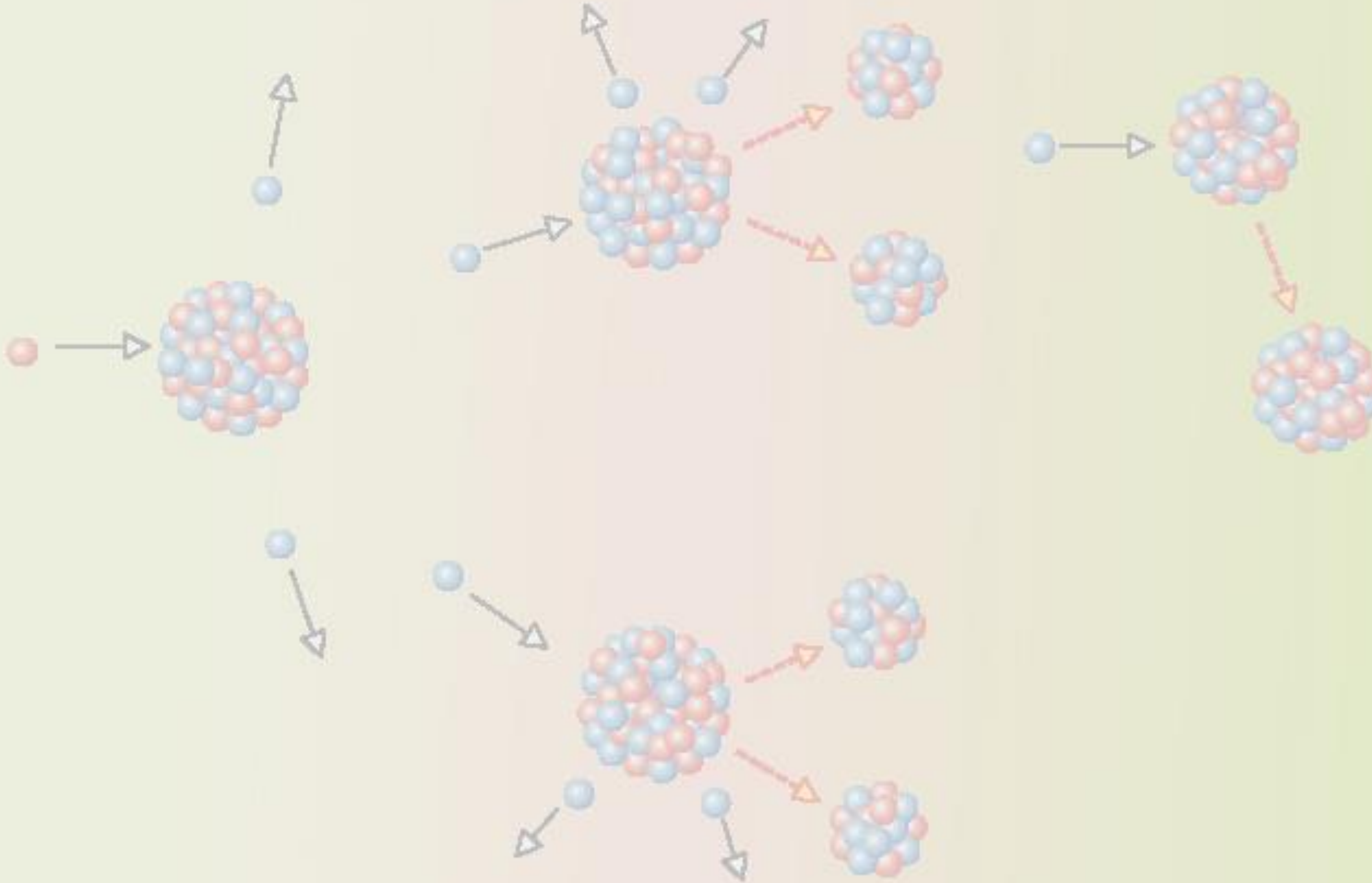


Simulation of safety and performance of ADS requires to extend the nuclear data and models to the high energy range, up to the energies of the spallation neutrons (several hundreds MeV).



Fuel composition for different types of reactors



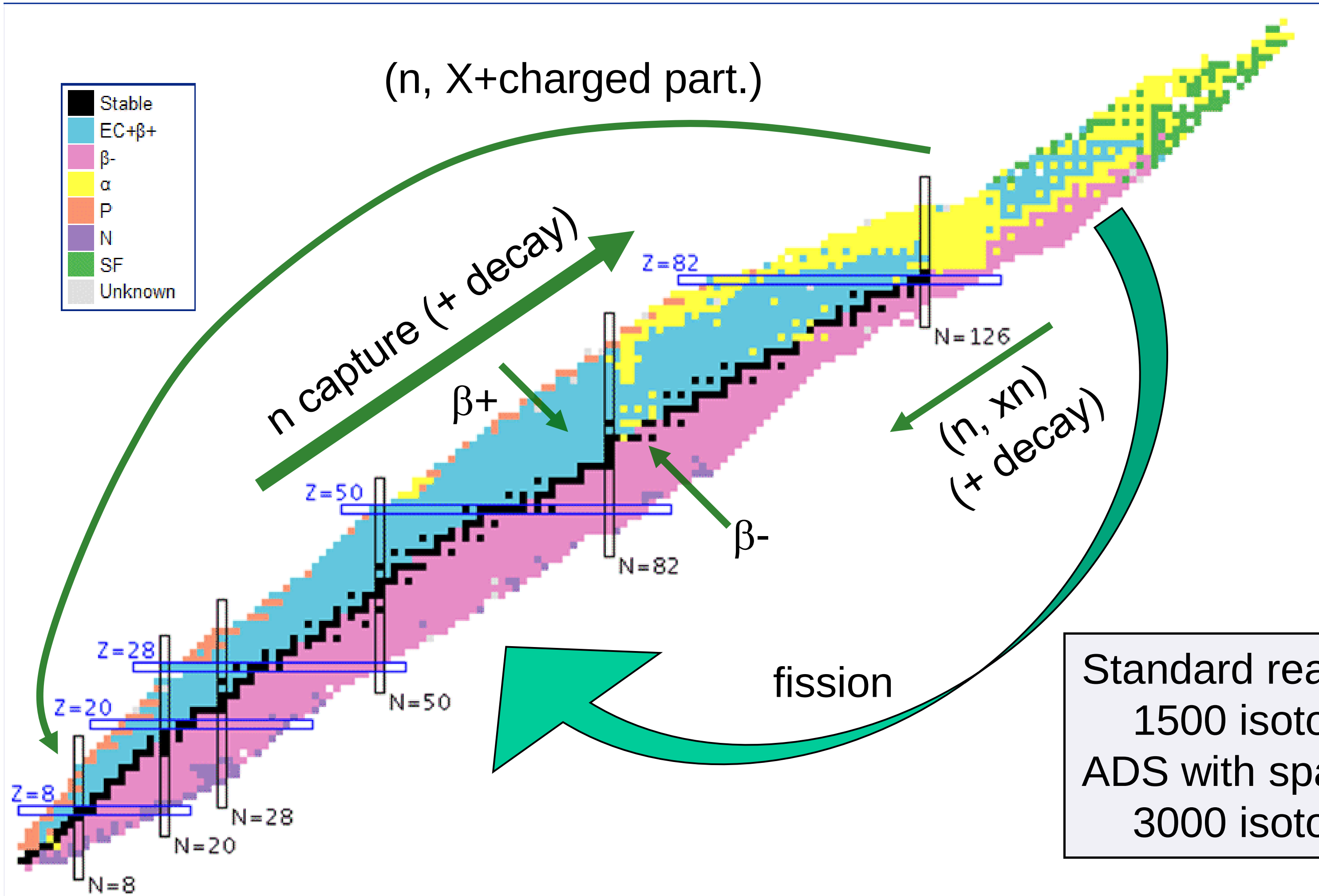


Uses of the ND for the nuclear fuel cycle

- Evolution of fuel composition during and after irradiation
- Evolution of the reactor characteristics with burn-up
- Nuclear waste (fission fragments, actinide activation, structural material activation, impurities activation)
- Interim storage
- Reprocessing (fissile material, neutron poisons, heat, n, radiation…)
- Partition and Transmutation
- Reprocessed fuel fabrication conditions
- Loading of reactors with reprocessed fuel
- New fuel concepts and new wastes
- Final disposal (Deep Geological Repository)
- Effects of advanced fuel cycles on
  - Radiotoxicity inventory
  - Decay heat
  - Gallery length
  - Dose at surface
- …

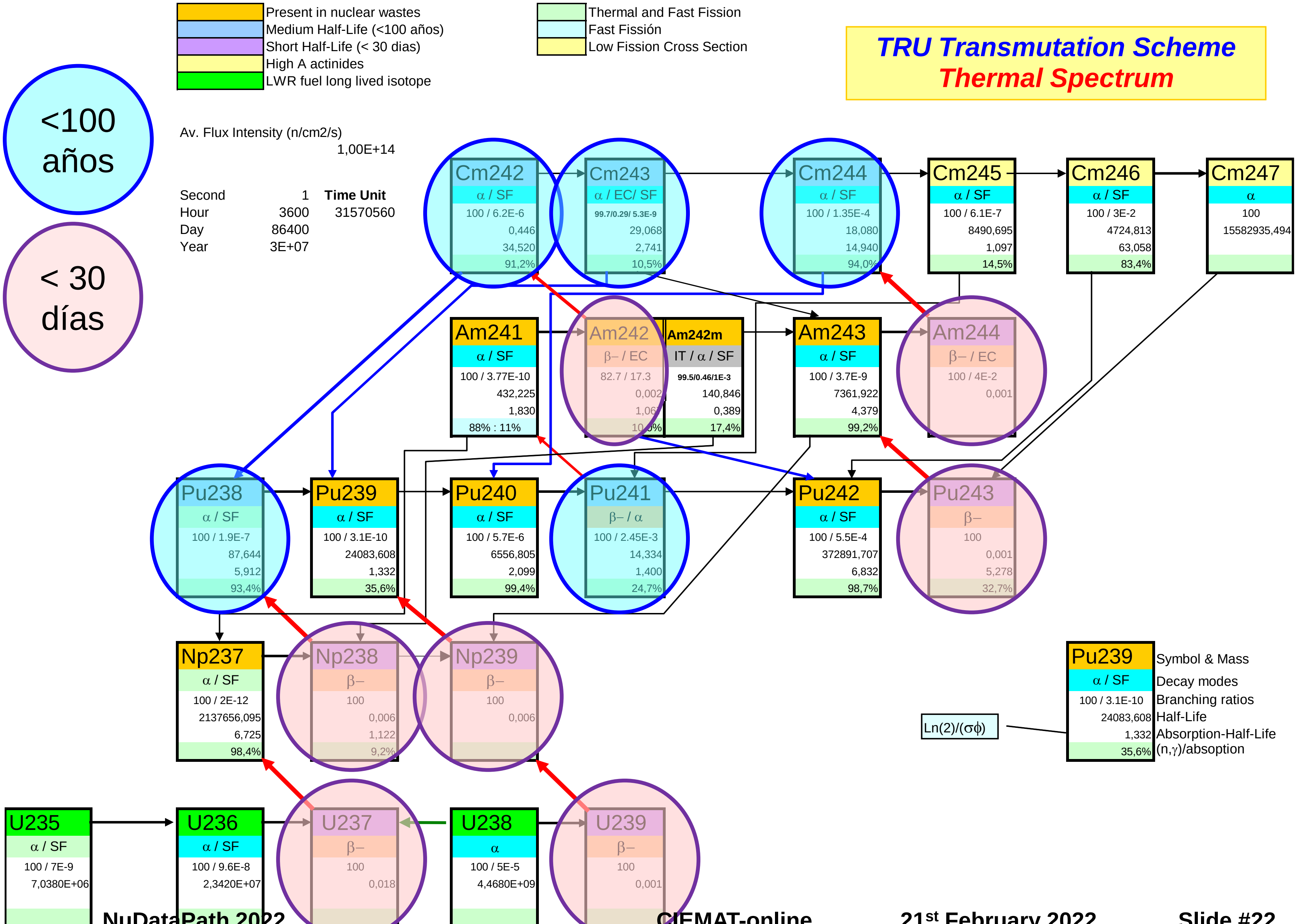
Main reactions in a nuclear reactor or transmutation device

- n- induced fission (energy + wastes)
- neutron capture (activation + breeding)
- elastic and inelastic neutron scattering
- radioactive decay
- (n,xn), (n, charged particle), …

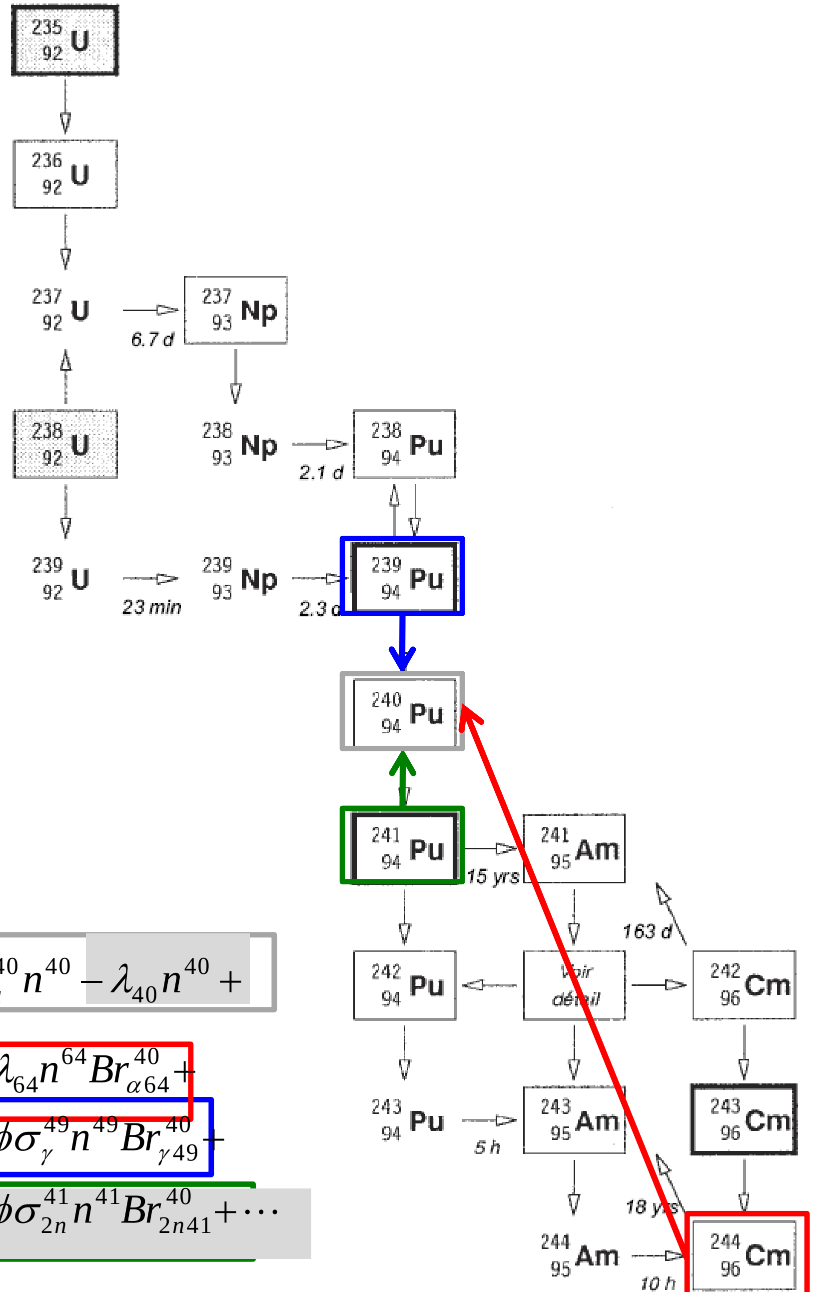


Standard reactor
1500 isotopes
ADS with spallation
3000 isotopes

Fuel evolution with burnup: Actinides



Fuel evolution with burnup and after irradiation (ϕ=0)



$$\frac{dn^{40}}{dt} = -\phi\sigma_a^{40} n^{40} - \lambda_{40} n^{40} + \lambda_{64} n^{64} Br_{\alpha 64}^{40} + \phi\sigma_\gamma^{49} n^{49} Br_{\gamma 49}^{40} + \phi\sigma_{2n}^{41} n^{41} Br_{2n41}^{40} + \cdots$$

Actinides

$$\frac{dn^k}{dt} = -\left(\phi\sigma_a^k + \lambda_k\right) n^k + \sum_{j \neq k} \phi\sigma_a^j Br_{aj}^k n^j + \sum_{j \neq k} \lambda_j Br_{dj}^k n^j$$

$$\frac{d\vec{n}}{dt} = [A] \bullet \vec{n}$$

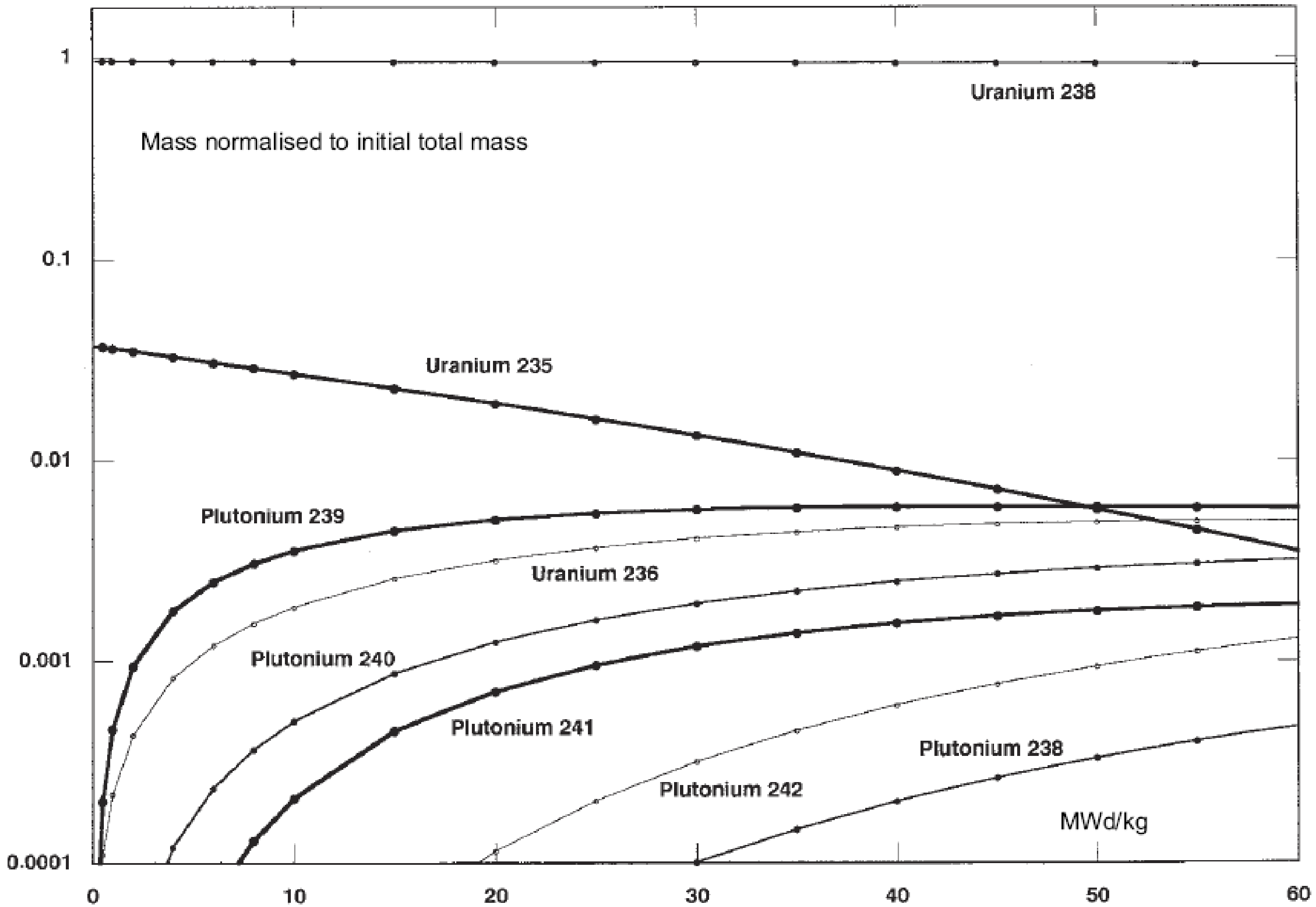
$$\gamma_i \Sigma_f \varphi = \int \gamma_i(E)\, \Sigma_f(E)\, \varphi(E)\, dE$$

$$\varphi\sigma_a^j Br_{aj}^i = \int \varphi(E)\, \sigma_a^j(E)\, Br_{aj}^i(E)\, dE$$

Fission fragments

$$\frac{dN_i}{dt} = -\left(\phi\sigma_a^i + \lambda_i\right) N_i + \gamma_i \Sigma_f \phi + \sum_{j \neq k} \phi\sigma_a^j Br_{aj}^i N_j + \sum_{j \neq k} \lambda_j Br_{dj}^i N_j$$

Fuel evolution with burnup: Actinides



Depletion and decay (Bateman) equation

$$\boxed{\frac{dN_i}{dt} = -\left(\varphi\sigma_a^i + \lambda_i\right)N_i + \gamma_i\Sigma_f\varphi + \sum_{j\neq k}\varphi\sigma_a^j Br_{aj}^i\, N_j + \sum_{j\neq k}\lambda_j Br_{dj}^i N_j}$$

$$\varphi\sigma_a^j Br_{aj}^i = \int \varphi(E)\,\sigma_a^j(E)\,Br_{aj}^i(E)\,dE \quad ; \quad \gamma_i\Sigma_f\varphi = \int \gamma_i(E)\,\Sigma_f(E)\,\varphi(E)\,dE$$

Data appearing in the equation:

λ_i Decay constant of isotope/isomer i

σ_a^i Absorption cross sections of isotope i also $\sigma_a = \sigma_F + \sigma_{Capture} + \cdots$

Σ_F Fission cross sections (Macroscopic cross section $\Sigma_F = \sum_j \sigma_F^j N_j$)

γ_i Fission yield of isotope (FF) i

Br_{aj}^i Branching ratio to isotope/isomer i after n absorption in isotope j

Br_{dj}^i Branching ratio to isotope/isomer i after decay of isotope j

Depletion/decay calculation

Decay reactions

- **β-, α, E.C., S.F.,** The main decay channels for fission fragments and actinides
- **β- n** Important for delayed neutrons
- **β+** Rare for this application

Activation reactions

- **(n,γ)** In some cases it can produce long lived isomers.
  Important mechanism for the build-up of heavy Pu +minor actinides
- **(n,2n)** Important to open channels to produce isotopes not in the (n,γ) chain
- **(n, ch. part.)** Finally all channels can be important for waste issues

For monitoring and detector interpretation it is also important to reconstruct the levels feeding of the child isotope and in general de (γ, e-) decay cascades.

Fission yields (FY):

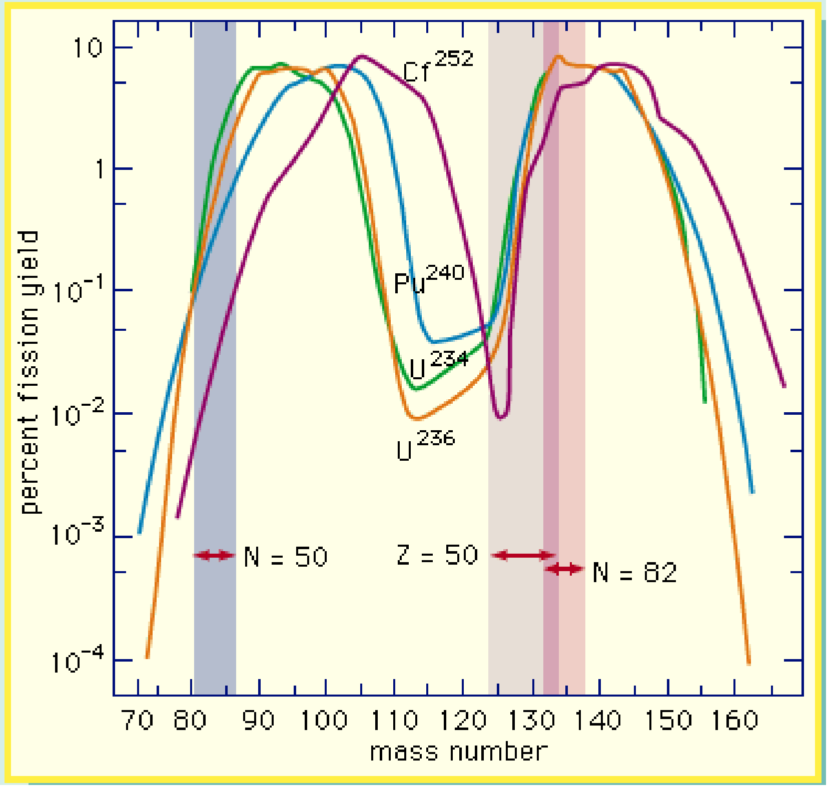
Fission yields can be available as individual isotopes FY or as **cumulative FY** for a long lived FF, adding all the FY of the precursors of that isotope.

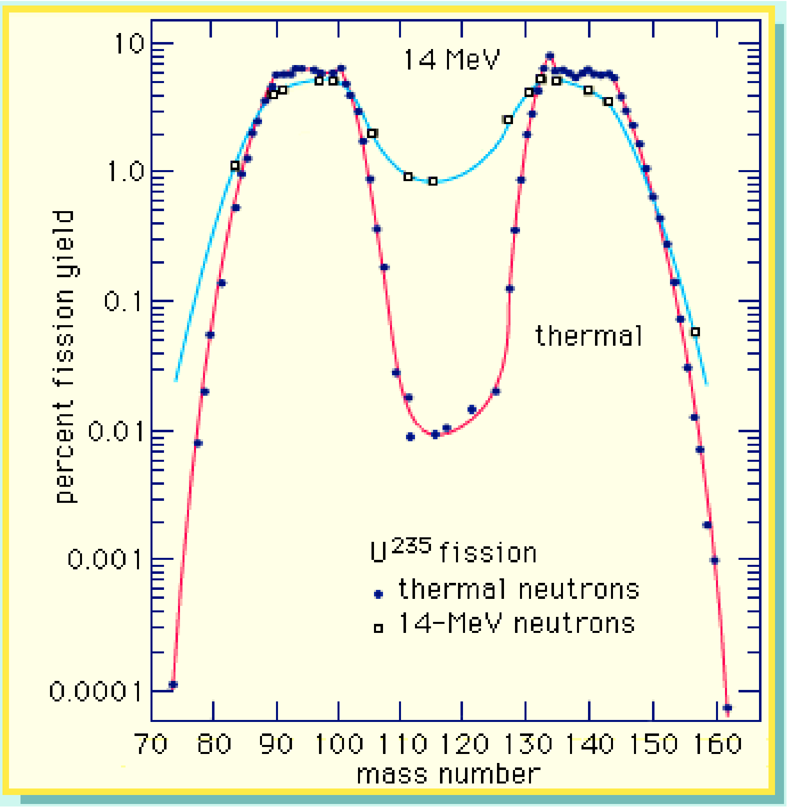
Individual FY of short lived isotopes can be high but difficult to measure for fission at low excitation energies (thermal fission).

Cumulative FY are often easier to measure and enough for many problems.

Fission: Absorption cross section + neutron source + energy + wastes+ fuel evolution

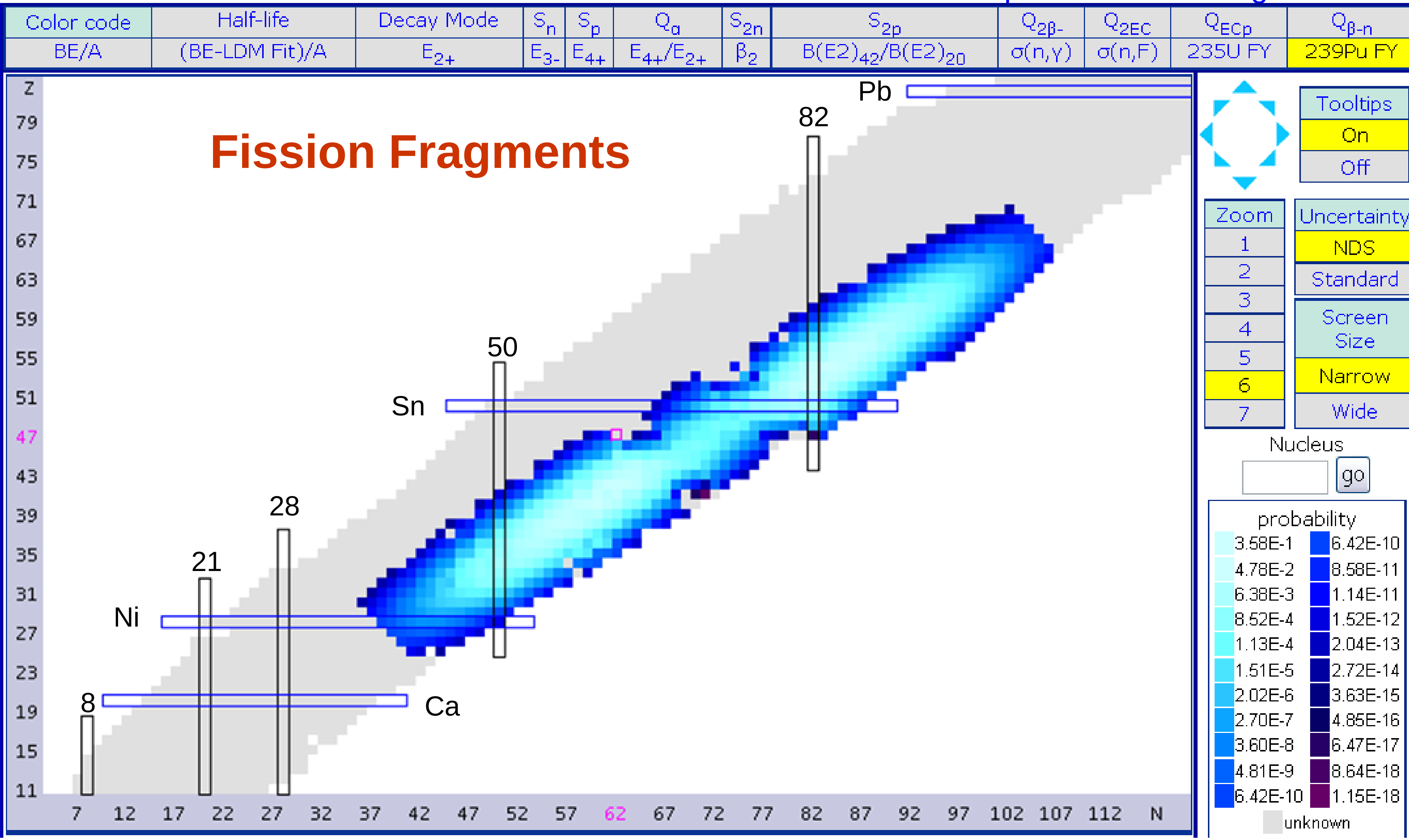
Fission fragments yields:
- Depend of target and n energy
- Contribute to fuel evolution: neutron poisons, radioactive wastes



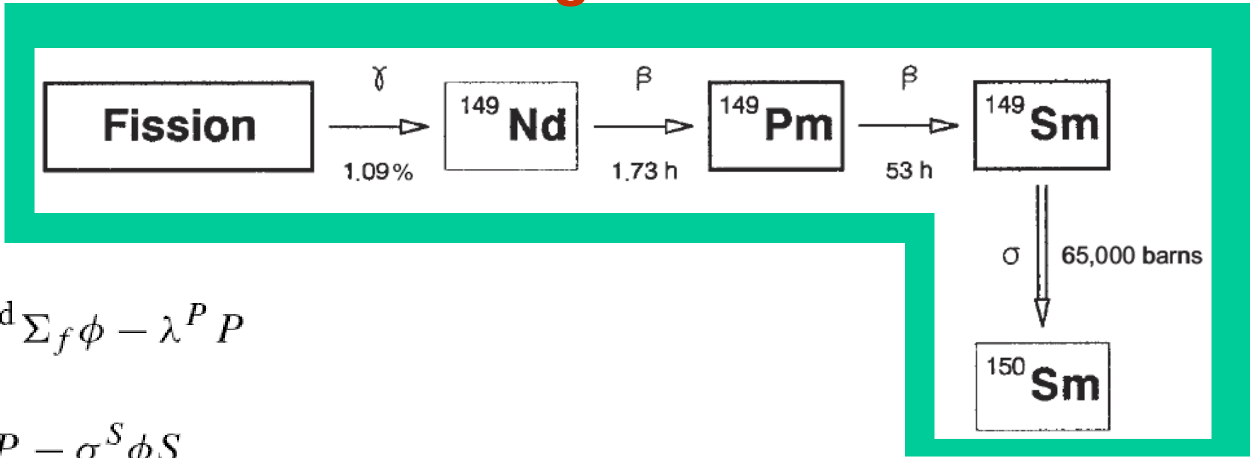


Fission: Absorption cross section + neutron source + energy + wastes+ fuel evolution

http://www.nndc.bnl.gov/chart/

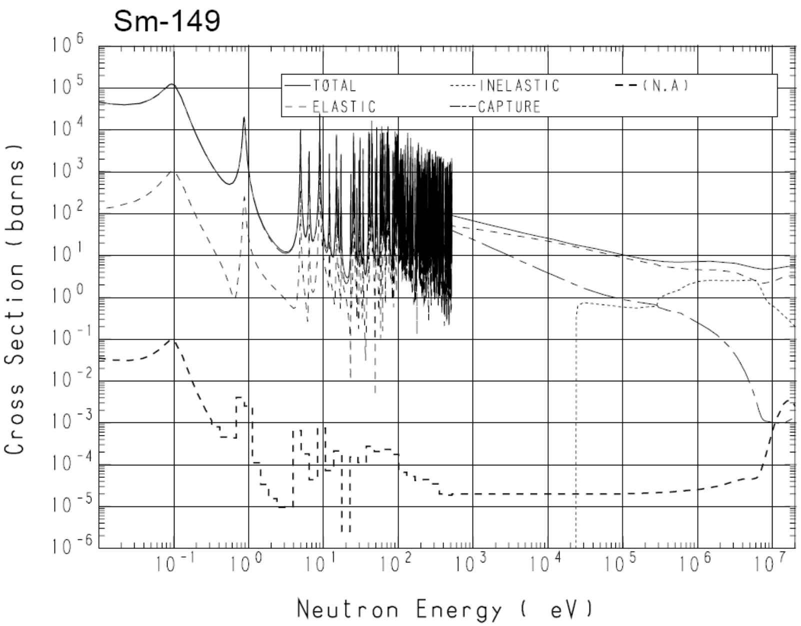


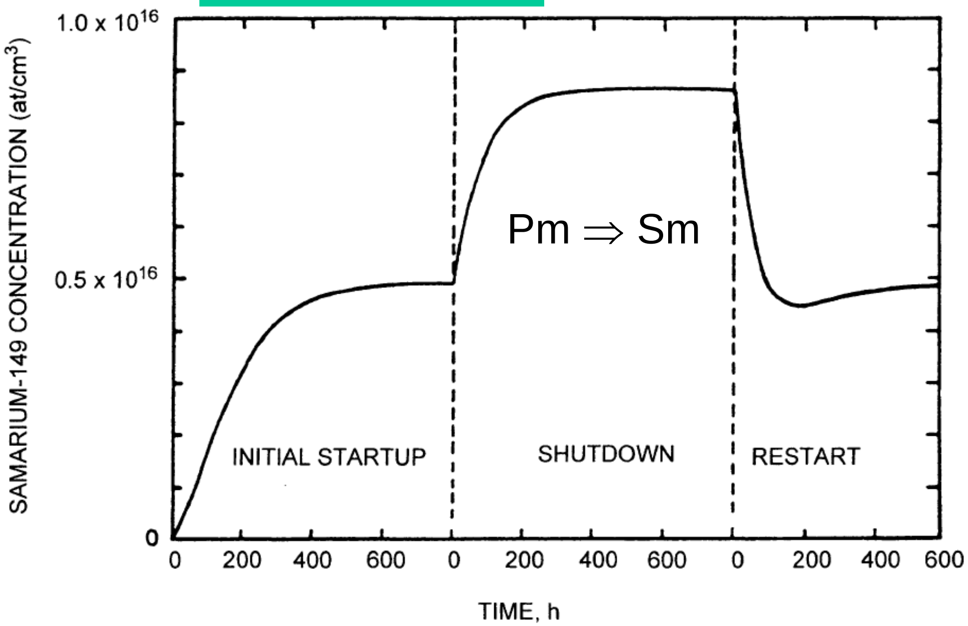
Fission Fragments: ^{149}Sm effect



$$\frac{dP}{dt} = \gamma^{\mathrm{Nd}} \Sigma_f \phi - \lambda^P P$$

$$\frac{dS}{dt} = \lambda^P P - \sigma_a^S \phi S$$





$$\pi_{S,\mathrm{eq}} = -\frac{\gamma^{\mathrm{Nd}} \Sigma_f}{\sigma_a^S} \frac{\sigma_a^S}{\Sigma_a} = -\gamma^{\mathrm{Nd}} \frac{\Sigma_f}{\Sigma_a} \approx 0.0045$$

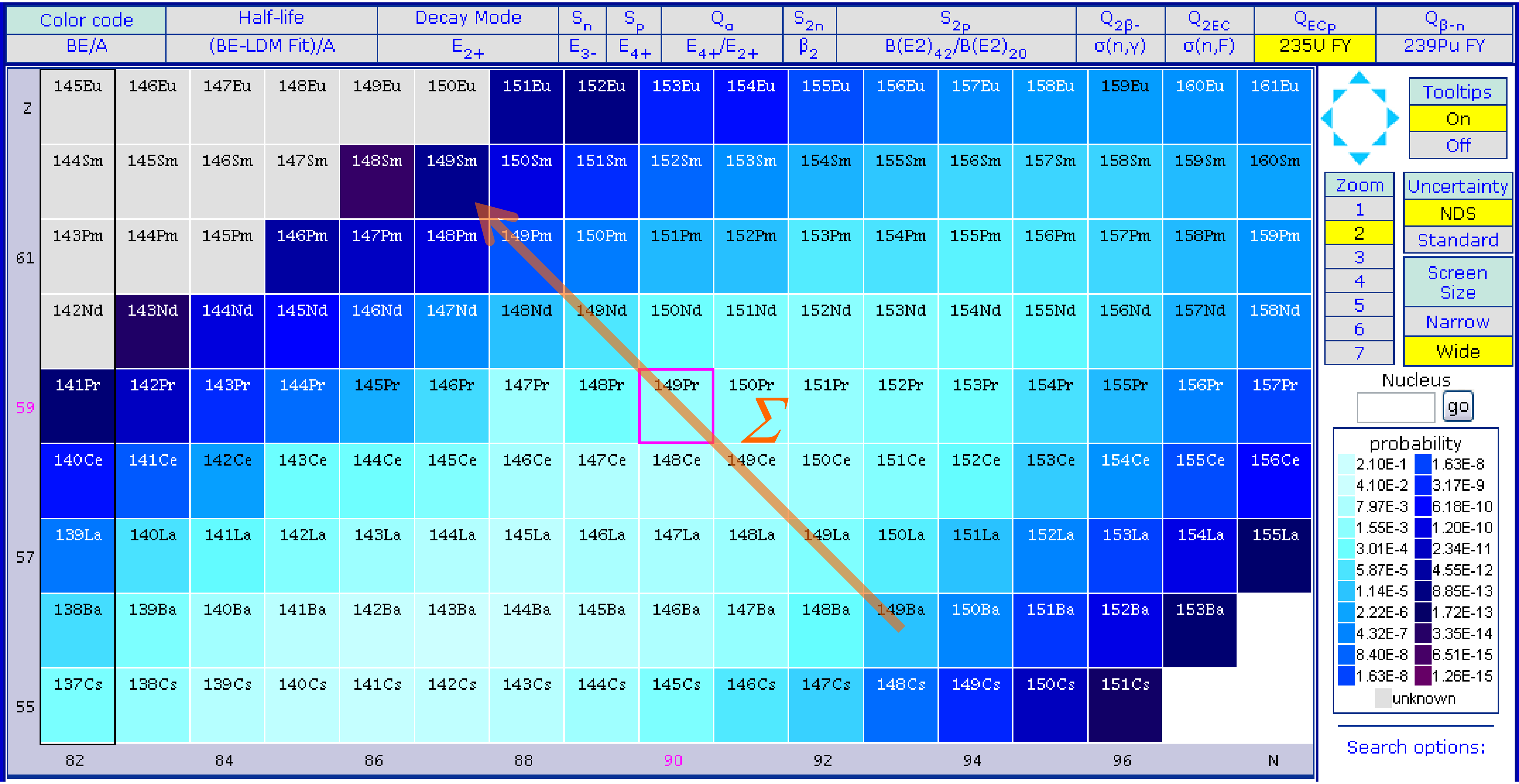
$$P_{\mathrm{eq}} = \frac{\gamma^{\mathrm{Nd}} \Sigma_f \phi}{\lambda^P}, \qquad S_{\mathrm{eq}} = \frac{\gamma^{\mathrm{Nd}} \Sigma_f}{\sigma_a^S}$$

Fission Fragments: ^{149}Sm effect

Decay mode http://www.nndc.bnl.gov/chart/

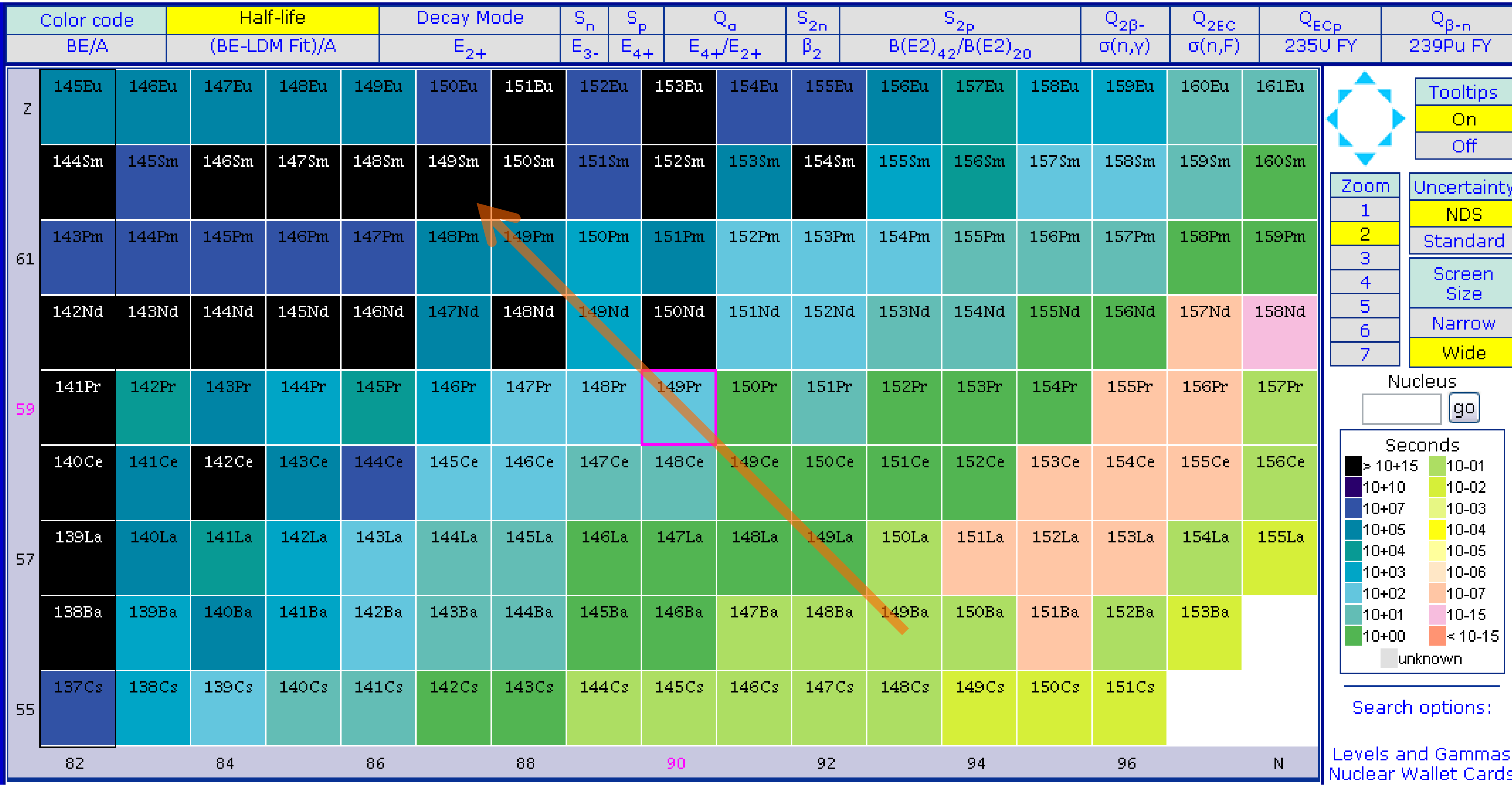
Fission Fragments: ^{149}Sm effect

Fission yields http://www.nndc.bnl.gov/chart/

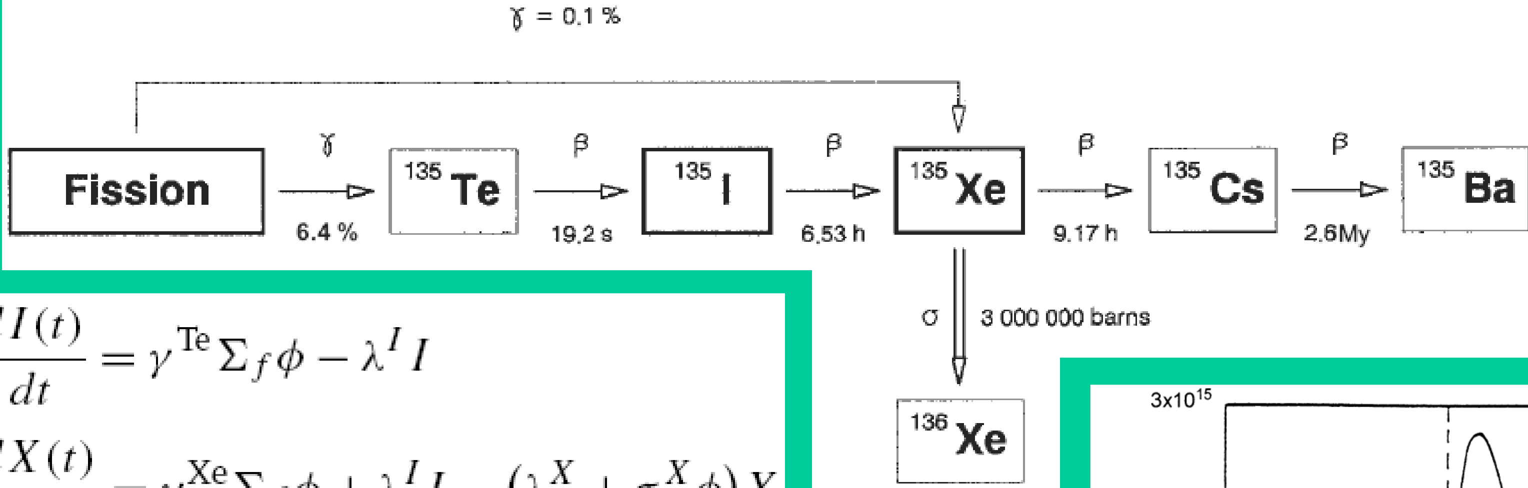


Fission Fragments: ^{149}Sm effect

Half-life http://www.nndc.bnl.gov/chart/

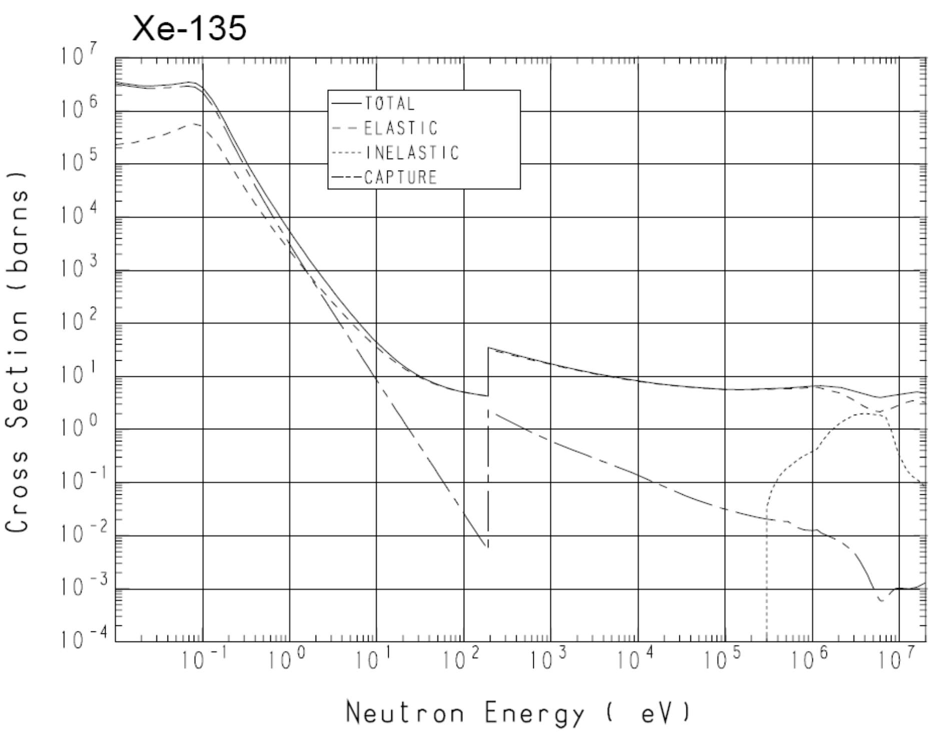


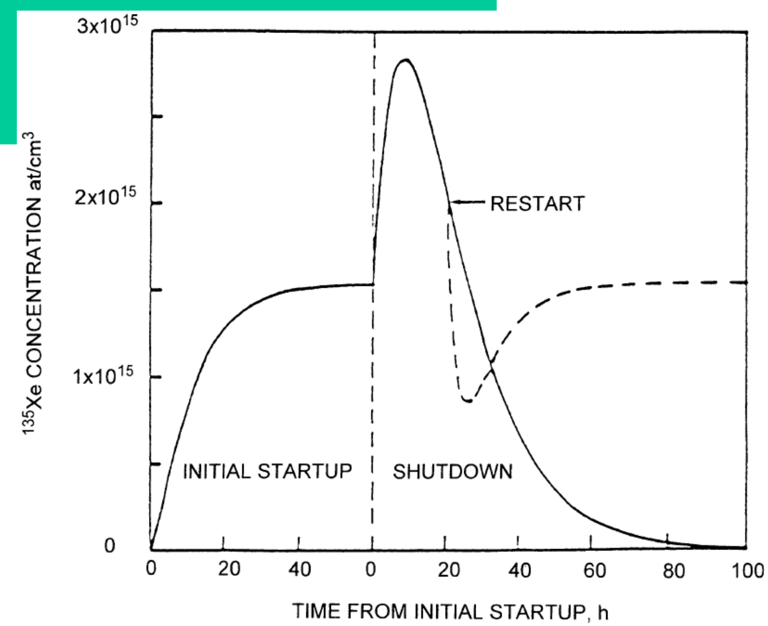
Fission fragments: ^{135}Xe effect



$$\frac{dI(t)}{dt} = \gamma^{\mathrm{Te}}\Sigma_f\phi - \lambda^I I$$

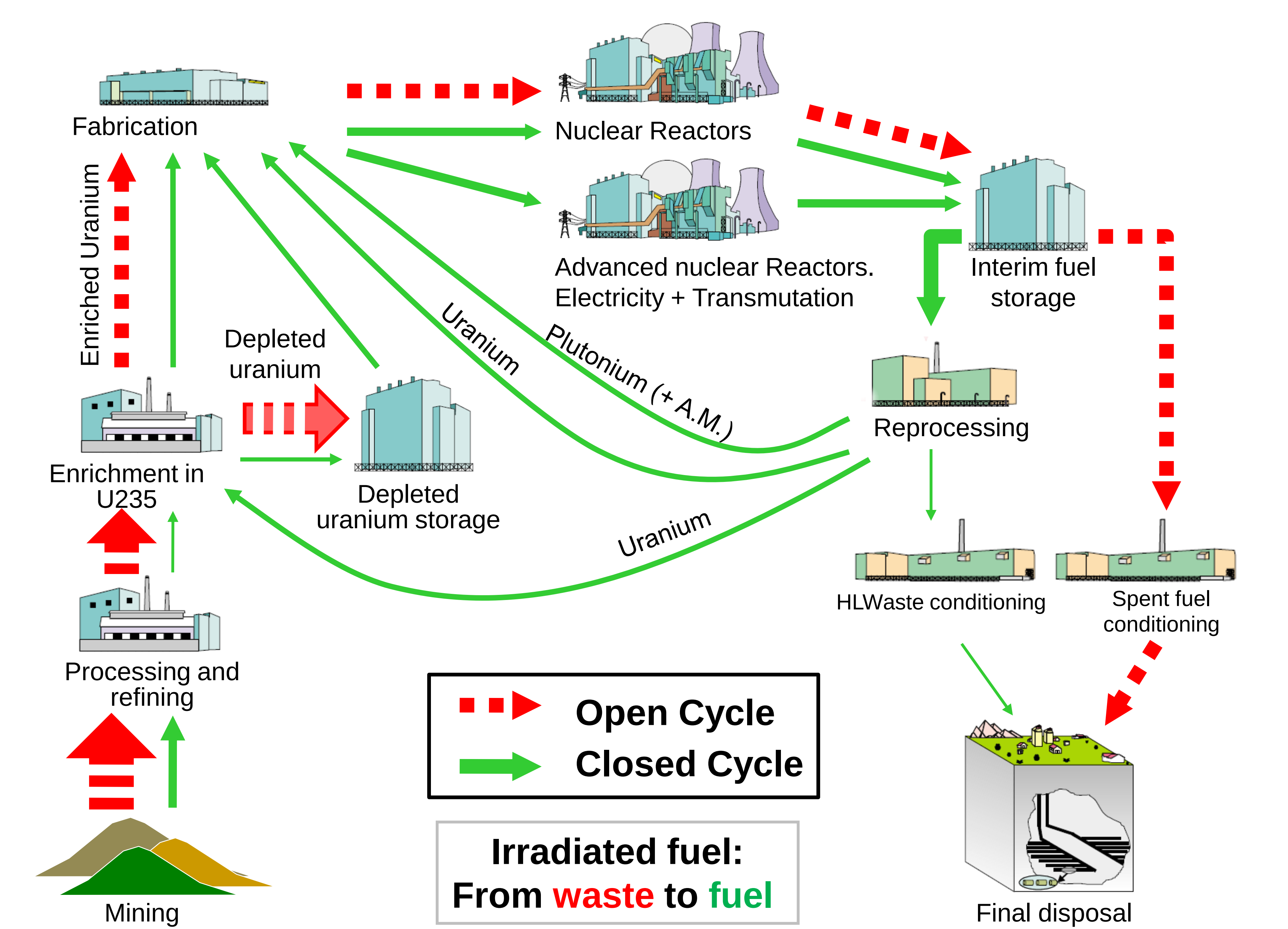
$$\frac{dX(t)}{dt} = \gamma^{\mathrm{Xe}}\Sigma_f\phi + \lambda^I I - (\lambda^X + \sigma_a^X\phi)X$$

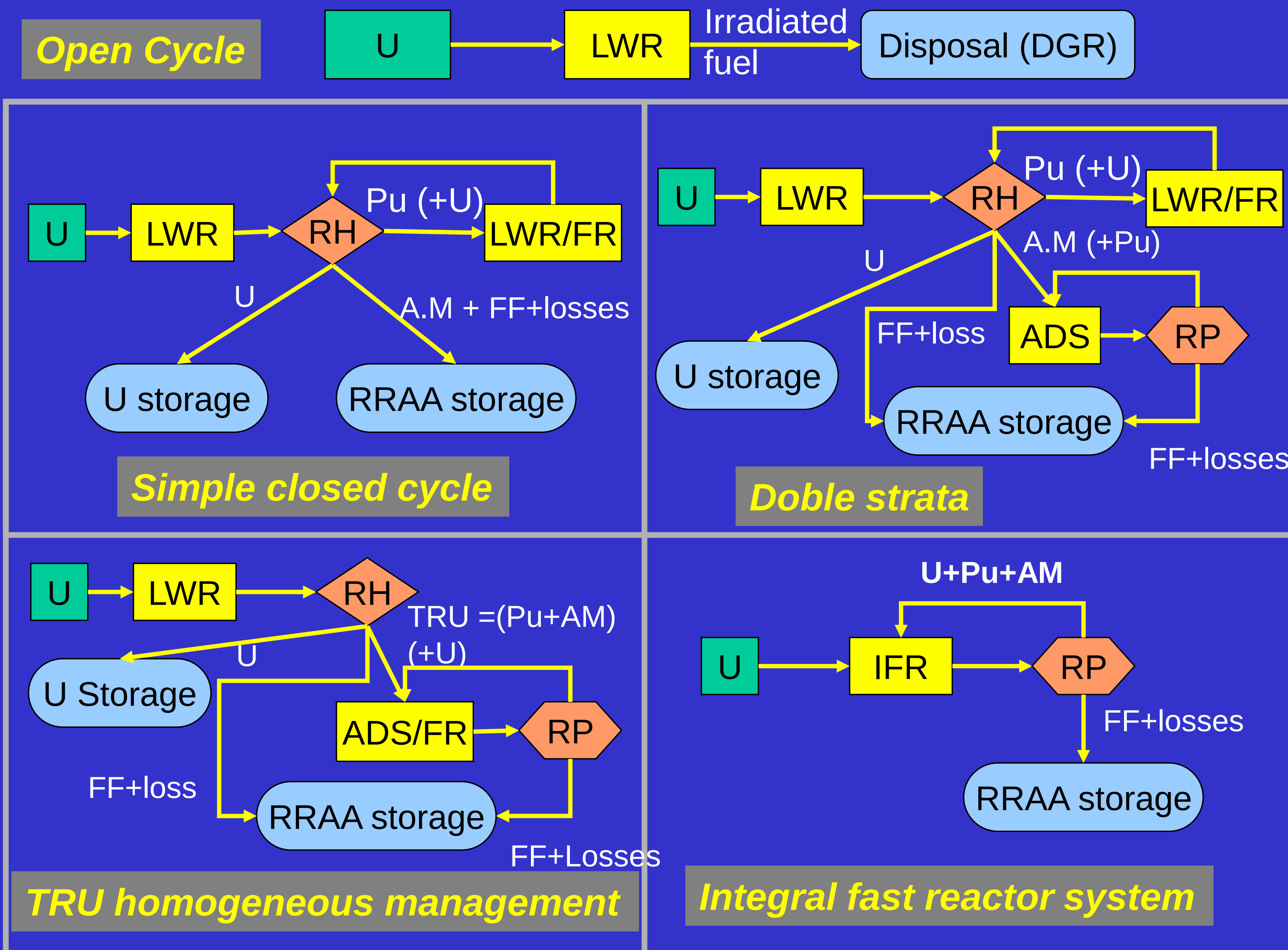




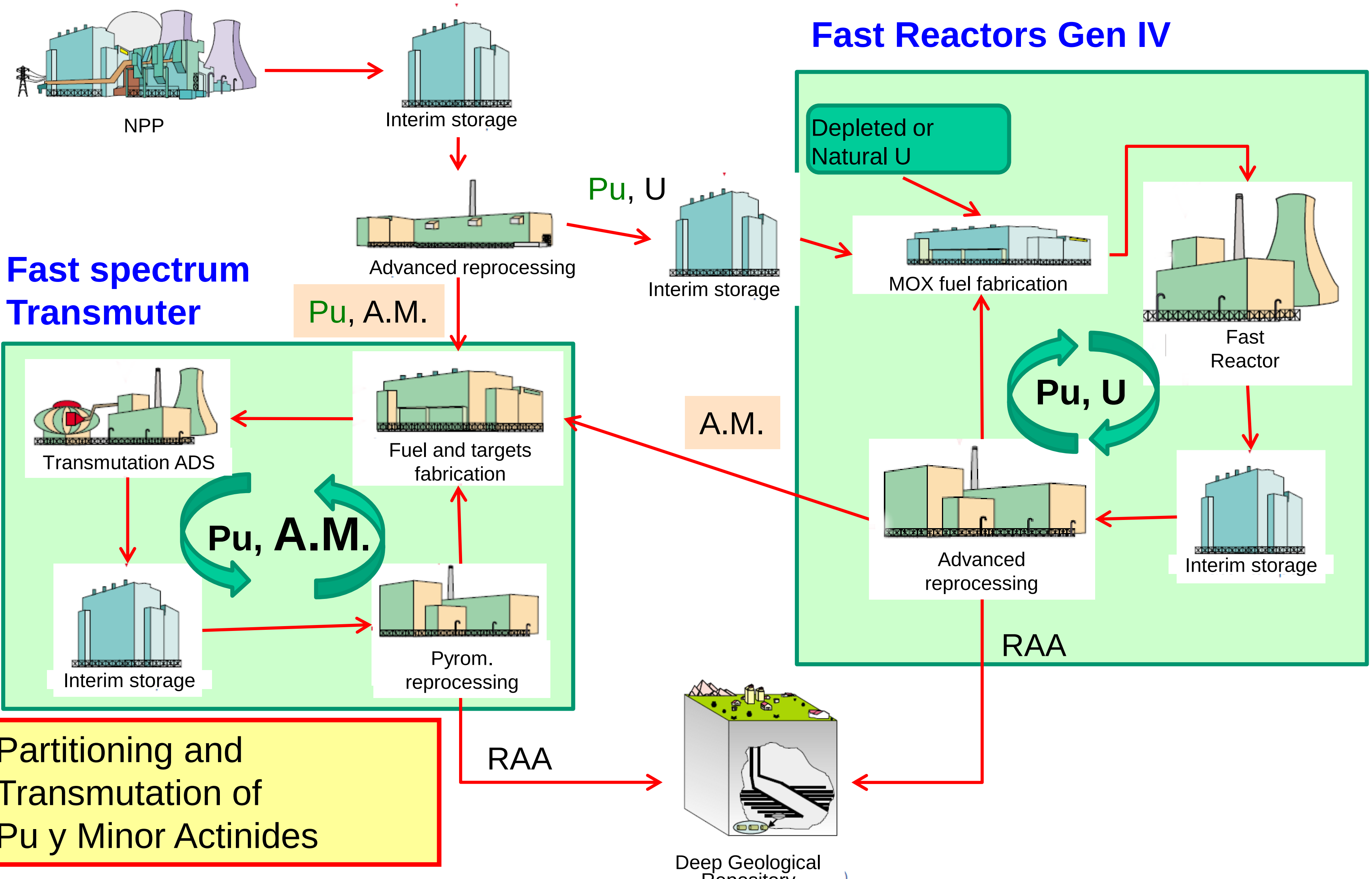
$$\pi_{X,eq} = (\gamma_I + \gamma_X)\frac{\sigma_X\Phi}{\lambda_X + \sigma_X\Phi}\left[\frac{\Sigma_f}{\Sigma_a}\right]_f$$

$$= -\frac{0.026}{1 + (0.756\times 10^{13})/\phi} \cong -3000 \text{ pcm}$$


Fabrication
Nuclear Reactors
Advanced nuclear Reactors.
Electricity + Transmutation
Interim fuel storage
Enriched Uranium
Depleted uranium
Uranium
Plutonium (+ A.M.)
Reprocessing
Enrichment in U235
Depleted uranium storage
Uranium
HLWaste conditioning
Spent fuel conditioning
Processing and refining
Open Cycle
Closed Cycle
Irradiated fuel:
From waste to fuel
Mining
Final disposal


Open Cycle
U
LWR
Irradiated fuel
Disposal (DGR)
U
LWR
RH
Pu (+U)
LWR/FR
U
A.M + FF+losses
U storage
RRAA storage
Simple closed cycle
U
LWR
RH
Pu (+U)
LWR/FR
A.M (+Pu)
U
FF+loss
ADS
RP
U storage
RRAA storage
FF+losses
Doble strata
U
LWR
RH
TRU =(Pu+AM) (+U)
U
U Storage
ADS/FR
RP
FF+loss
RRAA storage
FF+Losses
TRU homogeneous management
U+Pu+AM
U
IFR
RP
FF+losses
RRAA storage
Integral fast reactor system

Optimized sustainable system with Partition and Transmutación of all actinides



Partitioning and Transmutation of Pu y Minor Actinides

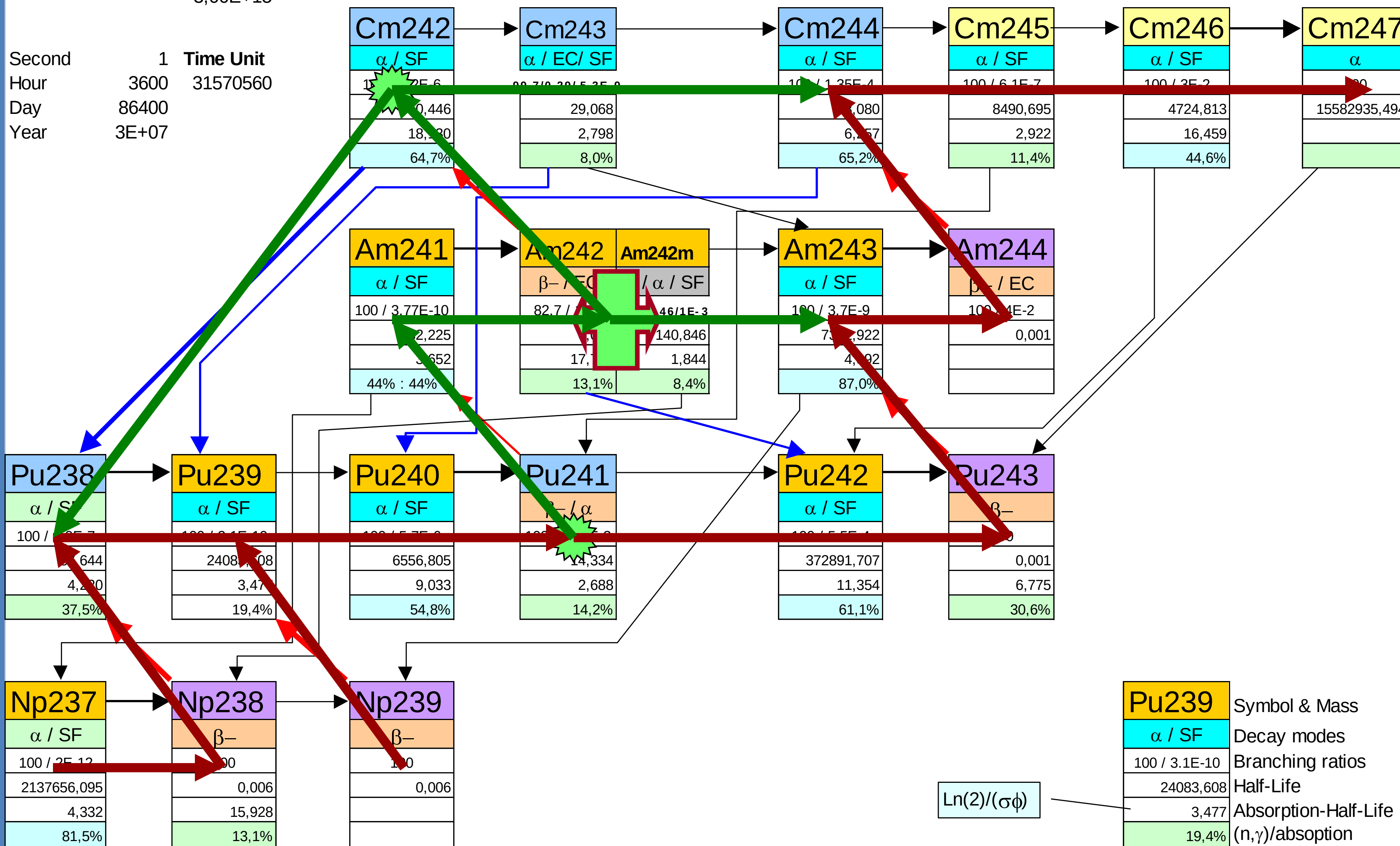
TRU Transmutation Scheme

Fast Spectrum

Present in nuclear wastes
Medium Half-Life (<100 años)
Short Half-Life (< 30 dias)
High A actinides
Thermal and Fast Fission
Fast Fissión
Low Fission Cross Section

Av. Flux Intensity (n/cm2/s)
3,00E+15

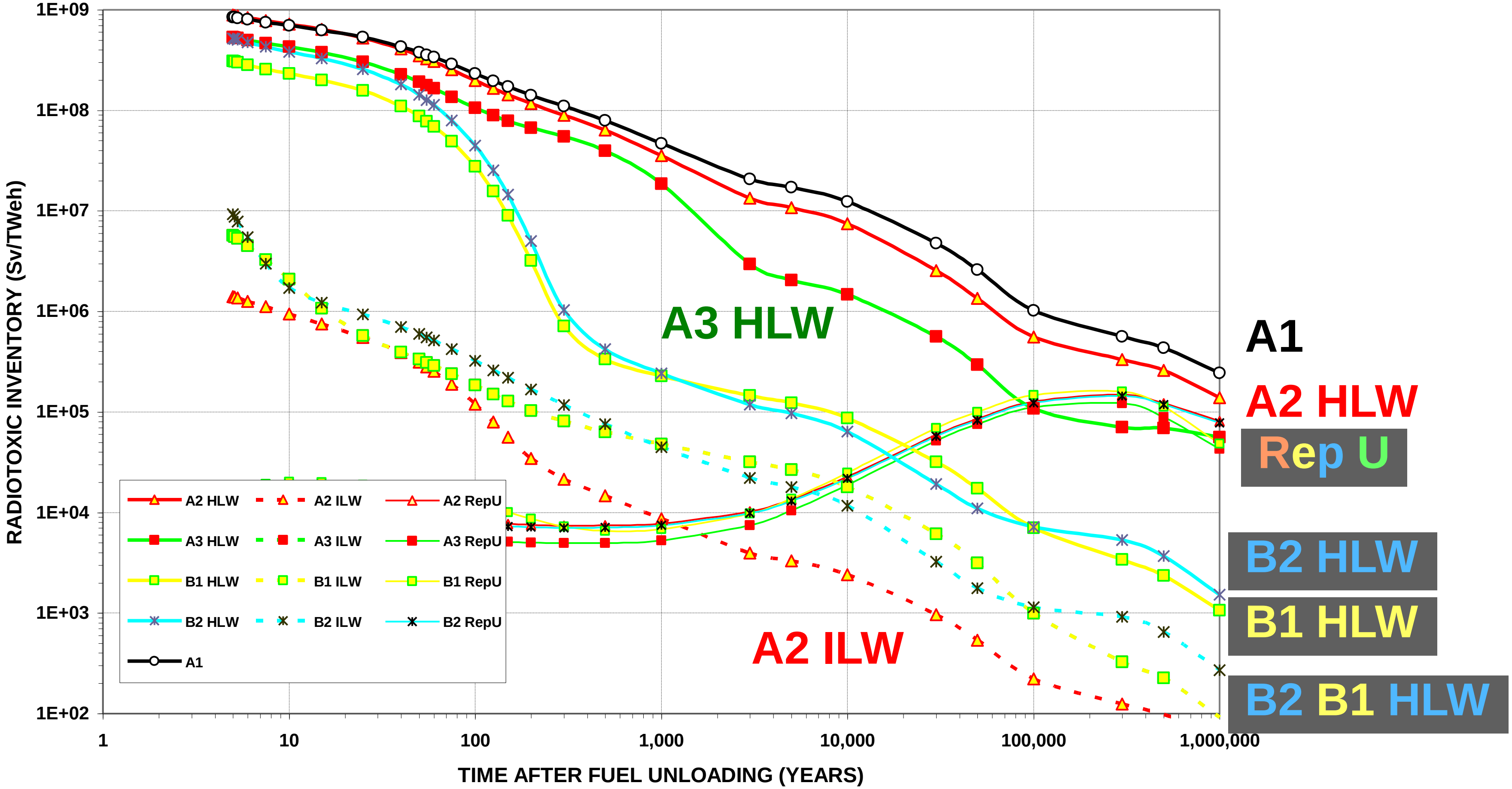
Second	1	**Time Unit**
Hour	3600	31570560
Day	86400	
Year	3E+07	



HLW + ILW: Radiotoxicity

Red-Impact HLWs and ILWs radiotoxicity evolution with time:
Reduction 1/10 if only Pu recycled and 1/100 when Pu+MA transmuted

- Small contribution from ILW
- Small but dominant contribution from Rep U at very long time

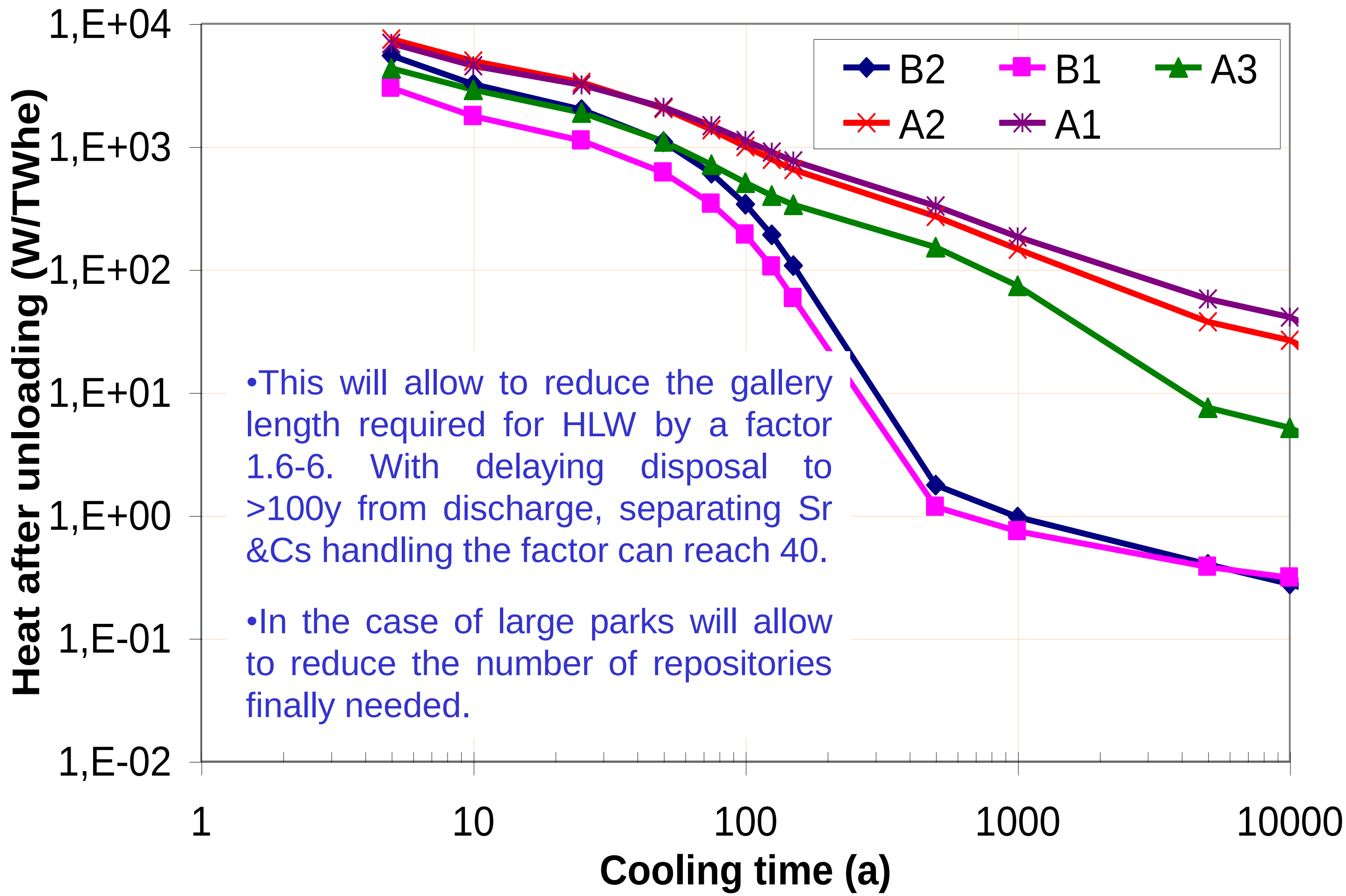


Derived magnitudes: Heat Load

- Scenario
- Mass flow
- Waste Strs: HLW, ILW,…
- Isotopic V.
- **Heat, RdTx, RdAc,**
- Packages
- Repository C
- Volume
- P.A.: Doses

Variants & Time Dependent Scenarios

Red-Impact HLWs (total) thermal power evolution with time:



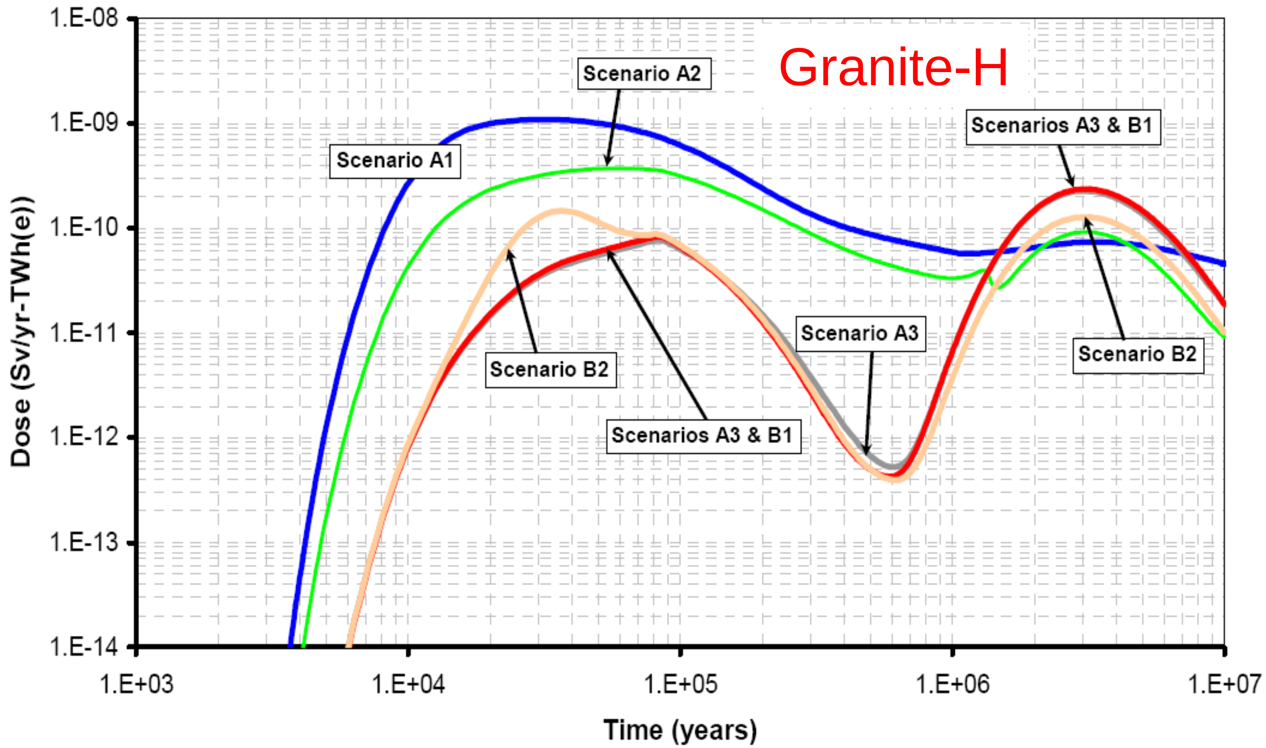
- This will allow to reduce the gallery length required for HLW by a factor 1.6-6. With delaying disposal to >100y from discharge, separating Sr &Cs handling the factor can reach 40.
- In the case of large parks will allow to reduce the number of repositories finally needed.

Large effect from M.A. recycling after 100 yr

Individual Dose from repository: HLW

Scenario
Mass flow
Waste Strs: HLW, ILW,…
Isotopic V.
Heat, RdTx, RdAc,
Packages
Repository C
Volume
P.A.: Doses

Variants & Time Dependent Scenarios



Small effect as RdTX mainly from F.F. and activation prod.

Matrix lifetime has a strong effect on doses due to the mobile fission and activation products such as ^{129}I, ^{14}C and ^{36}Cl.

The decrease in doses from these elements in the scenarios with reprocessing is a consequence of the reduced inventory in the HLW, as they are released (under control) as effluents .

Advanced fuel cycle

Parameters for optimization (to be simulated)

- **Inventory** of different isotopes/isomers
- **Radiotoxicity/Radioactivity Inventory**
- **Heat production**
- **neutron source**
- **Gallery length (from heat source)**
- **Dose at surface**

Nuclear data involved

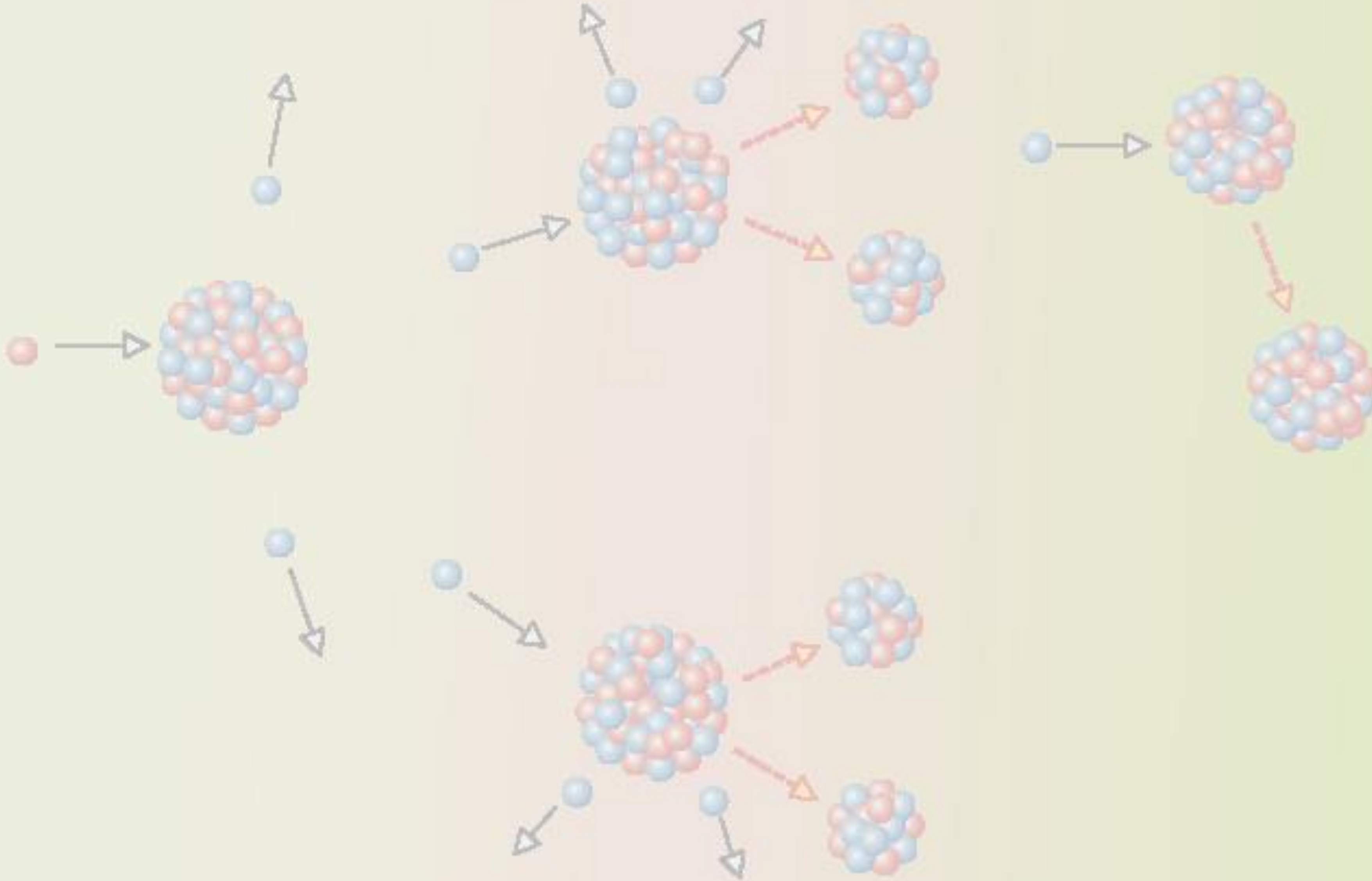
- **Same than for depletion/burnup:** Inventory at discharge of different reactors.
- **Energy per decay of different reactions (Q of reaction)**: Some times it is difficult to be able to reconstruct the decay heat from the individual contributions. Integral phenomenological models could be available.
- **S.F. and (α,n)** detailed data as **main neutron source** after irradiation
- **Delayed neutron precursors:** identification of all precursors and the detailed values of their decay need improvements to be able to compute it from individual contributions.
- **Dose coefficients for mobile isotopes in geologic formations**.

Integral measurements for validation of nuclear data

In actual applications of nuclear data for simulations or interpretation of real life problems many isotopes and reactions are involved and it is difficult to exclude biased or closely correlated data that could result in wrong results. So its is important to validate data for different applications with integral experiments of similar complexity of the real applications. For transport and fuel cycle problems.

Especial cases

- **new design concepts:** Involve often new isotopes, new energy ranges or new reaction channels that should be validated (Gen-IV FR, ADS, MSR)
- **new fuels:** New fuel option, new matrix, ATF, Mo matrix for dense fuels, …
- **criticality applications:** Many already available but always important because the high precision required
- **shielding**: few available sets well documented – more needed to avoid bias from criticality experiments
- **depletion and burn-up calculations:** few available sets well documented – more needed to avoid bias from criticality experiments
- **reference spectra averaged cross section of minor/new isotopes**: important validation if the right spectrum is used. Validation of thin to thick samples results.



SANDA Basic data

H2020 Grant Agreement number: 847552

A project for the EURATOM WP2018 for NFRP-2018-4

Project Start date: 01/09/2019

Duration: 48 months

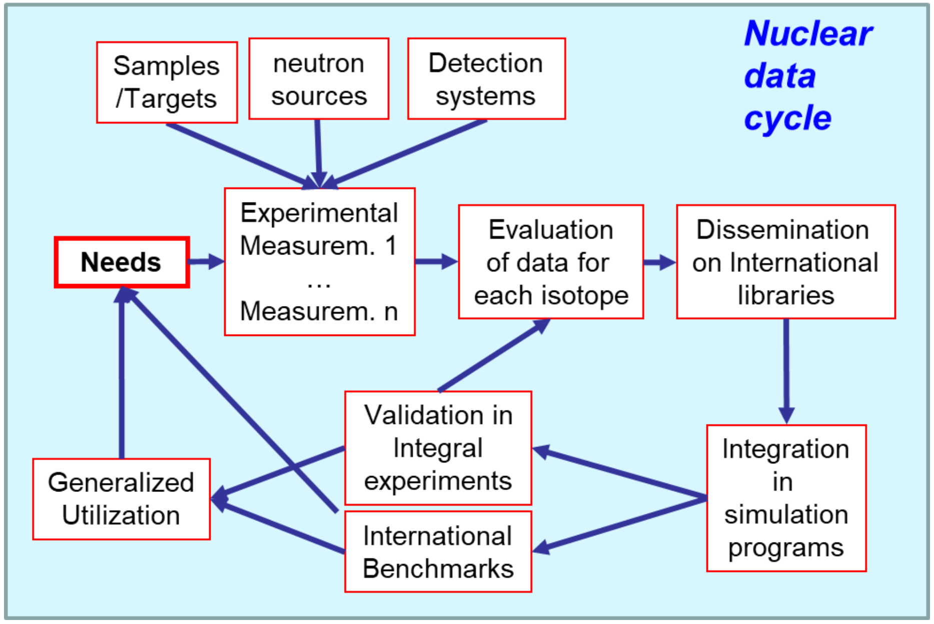
Requested contributions: 3.5 MEuros

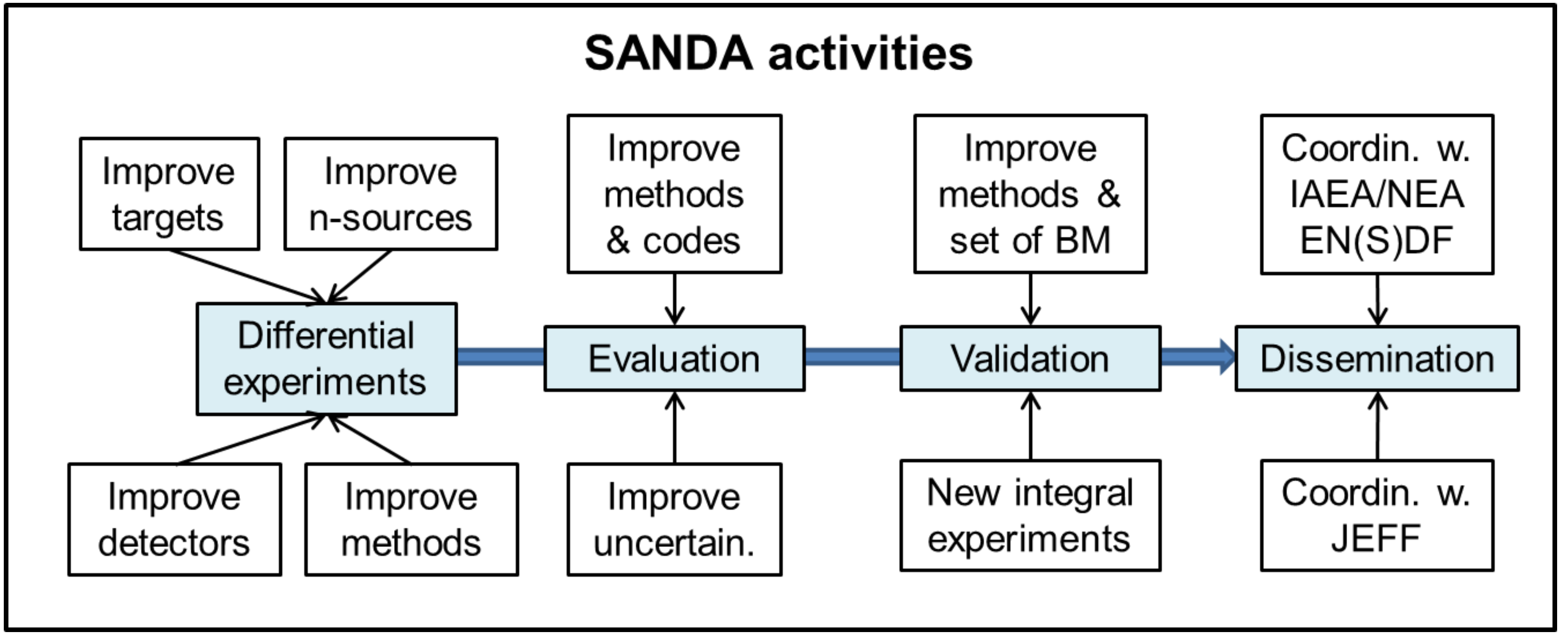
35 Partners: *CIEMAT*, Atomki, CEA, CERN, CNRS, CSIC, CVREZ, ENEA, HZDR, IFIN-HH, IRSN, IST-ID, JRC, JSI, JYU, KIT, NPI, NPL, NRG, NTUA, PSI, PTB, SCK-CEN, Sofia, TUW, UB, ULODZ, UMAINZ, UMANCH, UOI, UPC, UPM, USC, USE, UU.

from 18 EU-countries (Au, Be, Bu, Cz, Fi, Fr, Ge, Gr, Hu, It, Ne, Pol, Por, Ro, Sln, Sp, Sw, UK) + Switzerland

Coordinator: CIEMAT (E.M. Gonzalez-Romero)

SANDA objectives





SANDA WP1: Developments of new innovative detector devices

Task 1.1: Innovative devices from fission cross section to fission products decay studies

Subtask 1.1.1: fission cross section

GRPD - gas recoil proton detector and a MicroMegas-based time projection chamber.

Subtask 1.1.2: fission yields and decay data studies

Coupling the FALSTAFF spectrometer with the FIPPS gamma spectrometer at ILL.
A new gas cell with electric field guidance at IGISOL.
A new version of the BELEN detector (based on the Bonner sphere principle).
Upgrades on half-life and nuclear decay data measurement facilities to allow half-life measurements for radionuclides important to the nuclear medicine and nuclear industry.

Task 1.2: Innovative devices for neutron emission studies

Subtask 1.2.1 fast neutron spectrometer

A new compact broad band fast neutron spectrometer will be developed to characterize neutron flux.

Subtask 1.2.2 neutron detectors

SCONE setup based on plastic scintillator bars wrapped with a Gd loaded material to measure (n,xn).

Subtask 1.2.3 gamma detectors

A new HPGe prototype with adapted electronics will thus be designed and tested at CERN/n_TOF.

Task 1.3: innovative devices for capture cross section measurement on actinides: 2 for n_TOF EAR2

One based on the CLYC inorganic scintillator acting as γ-ray calorimeter or as total energy detector.
The second is based on total energy detectors with gamma-ray imaging capability (i-TED).

Task 1.4: detectors for non-energy application

Extending the techniques developed so far at n_TOF EAR1, to a measure DDX data for the neutron-induced emission of light charged particles from carbon, nitrogen and oxygen.

SANDA WP2: New nuclear data measurements for energy and non-energy applications 1/2

Task 2.1: Neutron induced fission and charged particle production cross sections

Task 2.1.1: Neutron induced fission cross sections

- Energy dependence of the nubar for the $^{235}U(n,f)$ cross section at JRC-Geel.
- Surrogate reaction excitation functions and cross sections for the $^{239}Pu(n,f)$, $^{241}Pu(n,\gamma)$ and $^{241}Pu(n,f)$.
- New measurements of $^{230}Th(n,f)$ and $^{241}Am(n,f)$ cross section at the CERN n_TOF EAR2
- $^{239}Pu(n,f)$ cross section measurement with the STEFF spectrometer

Task 2.1.2: Neutron induced charged particle production cross sections

- $^{16}O(n,\alpha)$ reaction in the energy range from the threshold up to 20 MeV
- $^{nat}C(n,lchp)$ reaction at NFS for improving cross section standards.
- New (n,chp) cross section data with a powerful array of hyper pure germanium detectors.
- Prompt fission neutron spectra above 10 MeV ($^{235}U(n,f)$) .

Task 2.2: Neutron capture cross sections

Subtask 2.2.1. Capture measurements of fissile isotopes

- Combined measurement of the $^{239}Pu(n,\gamma)$ and $^{239}Pu(n,f)$ cross sections at GELINA and n_TOF.

Subtask 2.2.2. Capture measurement of stable isotopes

- $^{92,94,95}Mo(n,\gamma)$ cross sections at GELINA and n_TOF .

Task 2.3: Neutron elastic and inelastic scattering and neutron multiplication cross sections:

- Neutron inelastic cross section measurements on ^{239}Pu, ^{233}U, ^{14}N and $^{35,37}Cl$.
- Branching ratio for ^{209}Bi, $^{208}Pb(n,tot)$ and $^{238}U(n,inel)$ cross sections at GELINA.

SANDA WP2: New nuclear data measurements for energy and non-energy applications 2/2

Task 2.4: Decay data measurements

Subtask 2.4.1. Beta decay measurements with TAGs

High precision decay data for fission products from major and minor actinides with DTAS detector.

Subtask 2.4.2. Beta delayed neutron measurements

New measurements with the BELEN detector.

Subtask 2.4.3. Measurement of half-live and γ-ray emission probabilities of beta emitters

Measurement of half-lives for : ^{106}Ru, ^{153}Sm, ^{166}Ho, ^{186}Re, ^{212}Pb, ^{225}Ac and ^{223}Ra

Accuracy measurements of high priority isotopes defined in the framework of NFRP-2018-6 for SNF.

Task 2.5: Fission yields measurements

Subtask 2.5.1. Fission yield studies in (n,f) reactions

^{235}U at ILL by coupling the first arm of FALSTAFF to the new FIPPS γ-ray spectrometer.

Fission yield studies with the LOHENGRIN spectrometer at ILL.

New method based on the PI-ICR technique for general fission product yield studies at JYFL.

Subtask 2.5.2. Fission yield studies in inverse kinematics

Test of (p,2p) as surrogate reactions for fission experiments (fission of ^{237}Pa).

Task 2.6: New measurements for non-energy applications:

Subtask 2.6.1. Spectrum averaged cross sections for dosimetry

Activity induced in foils by neutrons from a ^{252}Cf source via the ^{117}Sn(n,inl)^{117m}Sn and ^{60}Ni(n,p) .

Subtask 2.6.2. Measurement of cross sections relevant for hadron therapy

Measurement of double-differential charged-particle emission cross sections at n_TOF in the range from 20 MeV to 200 MeV.

Subtask 2.6.3. Measurement of beta+ emitters

High priorities of IAEA: ^{11}C, ^{13}N, ^{15}O, ^{30}P produced by protons <250 MeV (also ^{10}C, ^{12}N, ^{38m}K and ^{29}P).

SANDA WP3: Target Preparation for Improvement of Nuclear Data Measurements

Task 3.1: Intensification of the "producer – user – interaction"

A series of regular meetings of target makers with the users to better communicate the requirements from both sides. Support bilateral meetings and organize user workshops.

Task 3.2: Fostering the network of target makers

Sharing knowledge, equipment and resources as a key issue for efficient work in this high-cost and man-power intensive activity. Especially for producing radioactive targets, there are only a few laboratories in Europe, which are able and allowed to handle such material.

Task 3.3: Target production

A limited number of targets can be produced according to requests from collaboration members. Both PSI and JRC will be responsible for the manufacturing of the final target.

Task 3.4: Development of an isotope separator

Definition and development of the design of an isotope separator meeting the requirements for the special application as target production facility and the preparation of the site for the installation.
The final aim is to install at PSI a dedicated modern high efficiency, high transmission, high throughput mass separator designed for these special applications.

SANDA WP4: Nuclear data evaluation and uncertainties

Task 4.1: Nuclear reaction code developments and evaluations

Task 4.1.1: TALYS development

Test influence of theoretical parameters and improve prompt fission neutron and gamma observables.

Task 4.1.2: Nuclear reaction evaluation

Improve evaluation methodologies for nuclear data and the associated uncertainties, by making use of Bayesian inference and "model defect" methods. Provide *all evaluations with covariance* information. **New evaluations**: new (n,xnγ) for the main actinides, Cr, actinides (U235, U238, Pu239, and Am241), important fission products (Sm, Nd, Cs, Mo, Ru, Eu, Gd, Rh) and the Pu isotopic chain (Pu238-Pu244)

Task 4.2: Fission yields and nuclear structure and decay data evaluations

Task 4.2.1: Evaluation of Fission yields

Deeply test some model assumptions used in the fission yield evaluations using the measurements of kinetic energy dependency of yields, isomeric ratios or isotopic distributions.

Task 4.2.2: Evaluation of nuclear structure and decay data

Perform ENSDF (Evaluated Nuclear Structure Data File) evaluations: theoretical calculations, evaluations, modern evaluation tools and nuclear data library production, to improve the next version of the JEFF Radioactive Decay Data Library and the Evaluated Nuclear Structure Data File.

Task 4.3: Processing and sensitivity

Best processing parameters for processing CE libraries with the AMPX system + checking, processing and verification of evaluated nuclear data files. Sensitivity calculations for fission yields.

Task 4.4: Applications

Selection/classification of benchmarks for nuclear data sensitivities and validation of nuclear data.

Task 4.5: High-energy model uncertainties

Investigate if the present methodology, using the Bayesian framework developed at CHANDA, can be generalized to the whole set of parameters of INCL and extended to ABLA.

SANDA WP5: Nuclear data validation and integral experiments 1/2

Task 5.1: Impact studies, sensitivity analyses, and assessment of needs for various applications

Subtask 5.1.1: Impact studies and sensitivity analyses

Impact of (JEFF) nuclear data uncertainties and systematic errors on reactor engineering design and safety parameters will be evaluated in a quantitative manner for innovative nuclear systems (and fuel cycles): sodium-cooled fast reactors such as ASTRID or ESFR, lead-cooled fast reactors such as MYRRHA and ALFRED, and the JHR.

+ Nuclear data sensitivity/impact on Criticality-safety and Decommissioning and waste disposal.

Subtask 5.1.2: Assessment of (JEFF) nuclear data need

Update of the OECD/WPEC/SG26 report and Recommendations will be made as to which nuclear data are in need of improvement and what "performance" gains can be expected as a consequence.

Task 5.2: Validation studies (using existing experiments)

Subtask 5.2.1: Assessing correlations in integral experiments

Assessments of "missing correlations in integral experiments" problem plus Simulations will be made to estimate the correlations.

Subtask 5.2.2: C/E validation and trends

C/E validations and sensitivity/uncertainty analysis on reactor physics experiments (IRPhE), shielding benchmarks (SIMBAD), criticality benchmarks and pile oscillation experiments (MINERVE/CERES).

The results derived from these studies will be combined to validate nuclear data and analyze possible biases.

Gaps in the validation will be identified and discussed

SANDA WP5: Nuclear data validation and integral experiments 2/2

Task 5.3: New integral experiments

Subtask 5.3.1: Experiments at GELINA and MINERVE

Neutron transmission measurements at the JRC GELINA facility using the same samples used in the MINERVE/CERES measurements.

Samples are UO_2 matrix with a small admixture of a fission product: Sm, Nd, Cs, Mo, Ru, Eu, Gd, Rh

Combined analysis of MINERVE/GELINA to improve the fission product cross section in the resonances

Subtask 5.3.2: Experiments at LR-0

- Full characterization of a critical ^{235}U-fuelled configuration for an IRPhEP-quality type benchmark;
- Direct and indirect measurements of the ^{235}U prompt fission neutron spectrum;
- Measurements of spectrum-averaged cross sections in well-characterized neutron spectra (graphite)

Subtask 5.3.3: Experiments at TAPIRO

Measure minor actinide spectrum-averaged fission and capture cross sections.

SANDA WP6: Management, ND research coordination at EU level and Education and Training

Task 6.1: Management

Task 6.2: Sustainable framework for the coordination of the European nuclear data research

Task 6.3: Coordination of Education and training activities

Task 6.4: Coordination of Dissemination and Communication activities

Nuclear data for nuclear technologies and applications

E.M. Gonzalez-Romero (CIEMAT)

Questions?