From neutronics to nuclear scenarios

Joliot-Curie School - 2019

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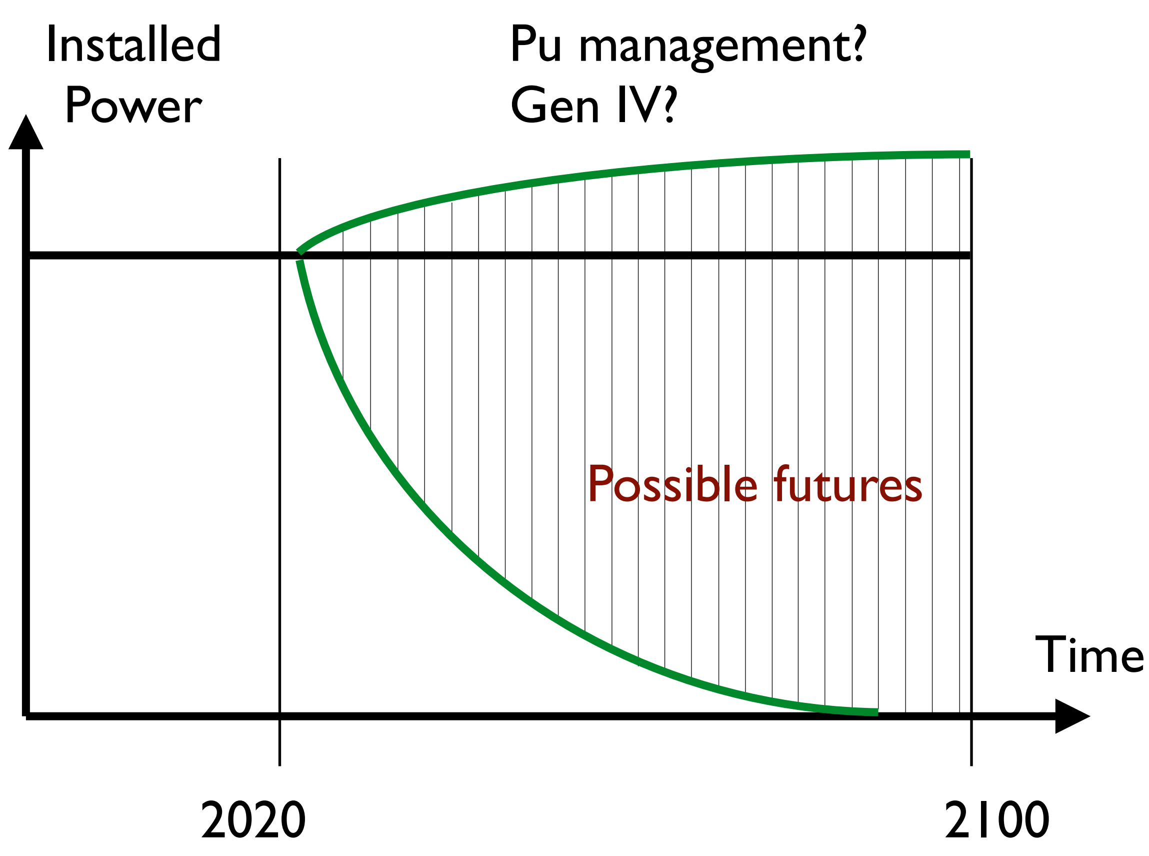
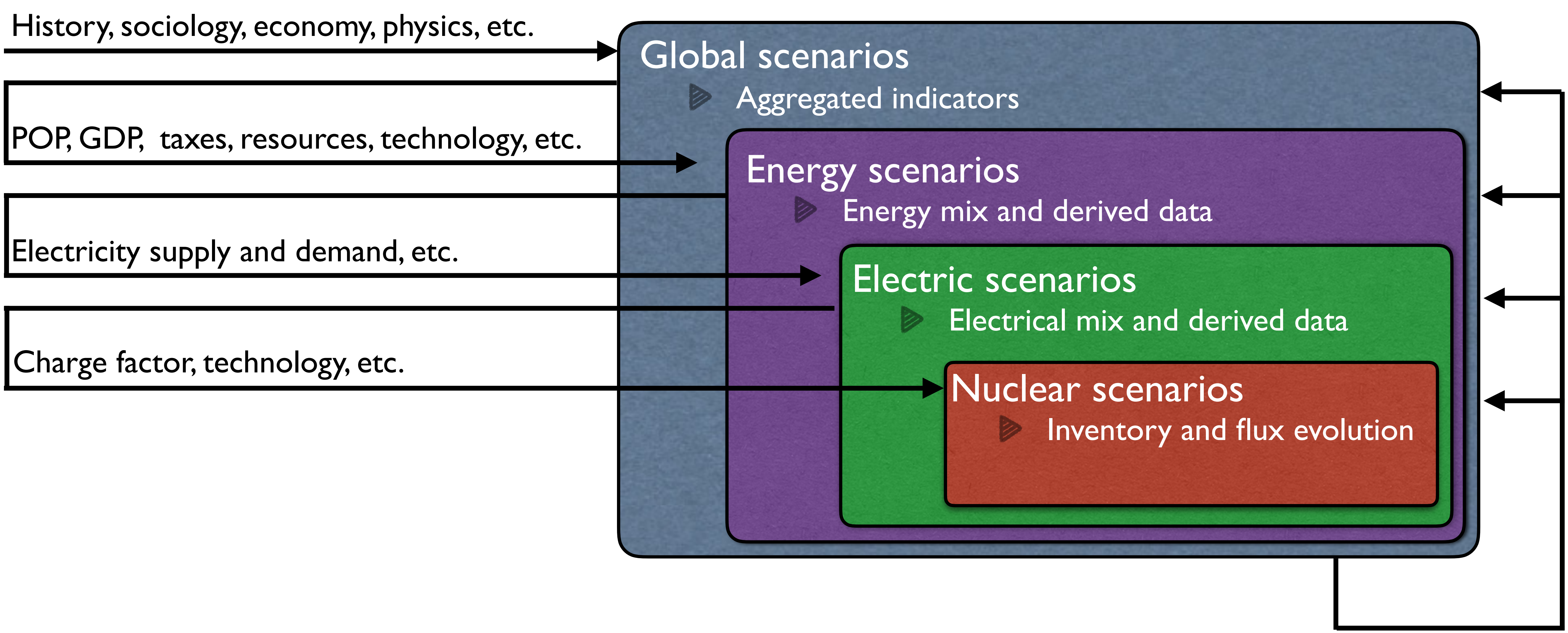
Introduction

- Several countries are involved in an « energy transition »
  - Low carbon emission energy sources
  - Low natural resources impact
  - Economic competitiveness
- In this framework, nuclear industry has to face important challenges
  - Ensure a very high safety level
  - Find a reliable solution for nuclear wastes
  - Show the feasibility of fleet dismantlement
  - Produce electricity at a competitive price
  - Support important required investments
- Context of high uncertainty
  - Role of anticipation
  - ➡ Scenario studies



- Scenarios are useful tool to assess/understand complex systems
  - Too much variables in interaction to build a formalized system
  - Building relations in a dynamic system is a complicated
  - Strong influence of human behavior or decisions (regulations, etc.)
    - ➡ Natural connexion with public/private decision making process



- What is the role of nuclear scenario in decision process?
  - ➡ Stéphanie Tillement's talk

- A fuel cycle simulation is based on operational hypothesis
  - Installations parameters (BU, cooling time, fuel fabrication time, etc.)
  - Spent fuel reprocessing strategy (LiFo, FiFo, etc.)
  - Time and duration for new technologies deployment
  - ➡ New methodology based on GSA formalism

- A fuel cycle simulation contains uncertainties that propagates
  - Nuclear data (PhD G. Krivtchik - CEA, 2014)
  - Reactor simulation simplifications (PhD A. Somaini - IPNO, 2017)
  - Fuel cycle simulations simplifications (FIT Project, etc.)

- Economic evaluations are built with high uncertainties
  - Individual costs dedicated to each operations
  - Calculation with the Levelized Cost of Electricity (LCOE)

Several scales in scenarios

Nuclear reaction : < 1s

Few reactor campaigns : around 10 years

One generation lifetime : around 50 years

One generation transition which suppose two or more generation involved : around 100 y.

Nuclear waste generated by reactors have typical lifetime that can reach several millions years

- Nuclear physics
- Neutronics
- Reactor physics

- Fuel cycle physics

time

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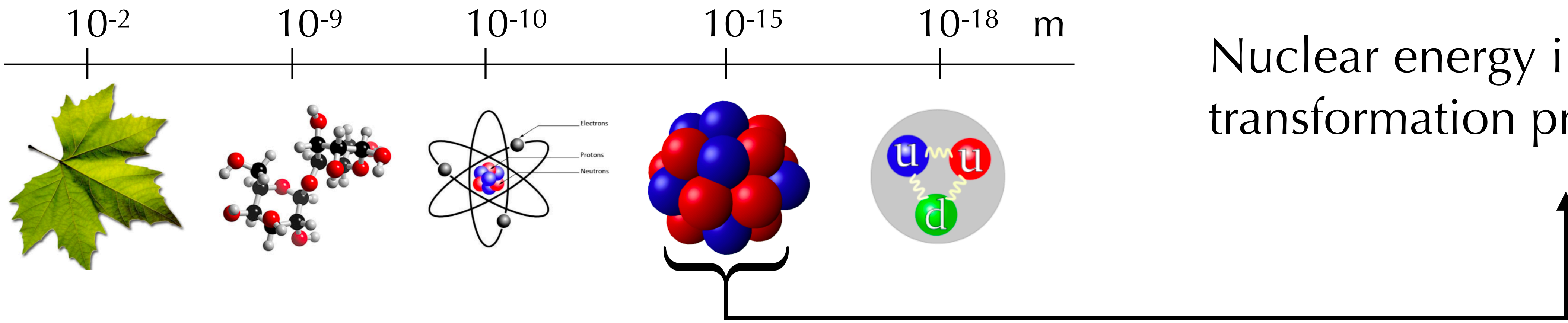
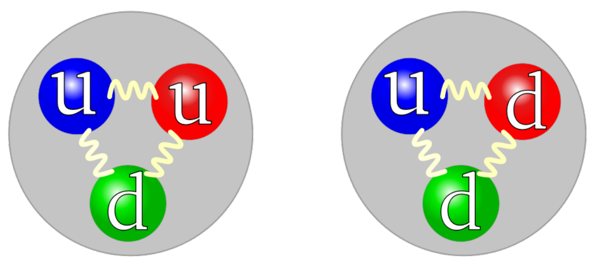
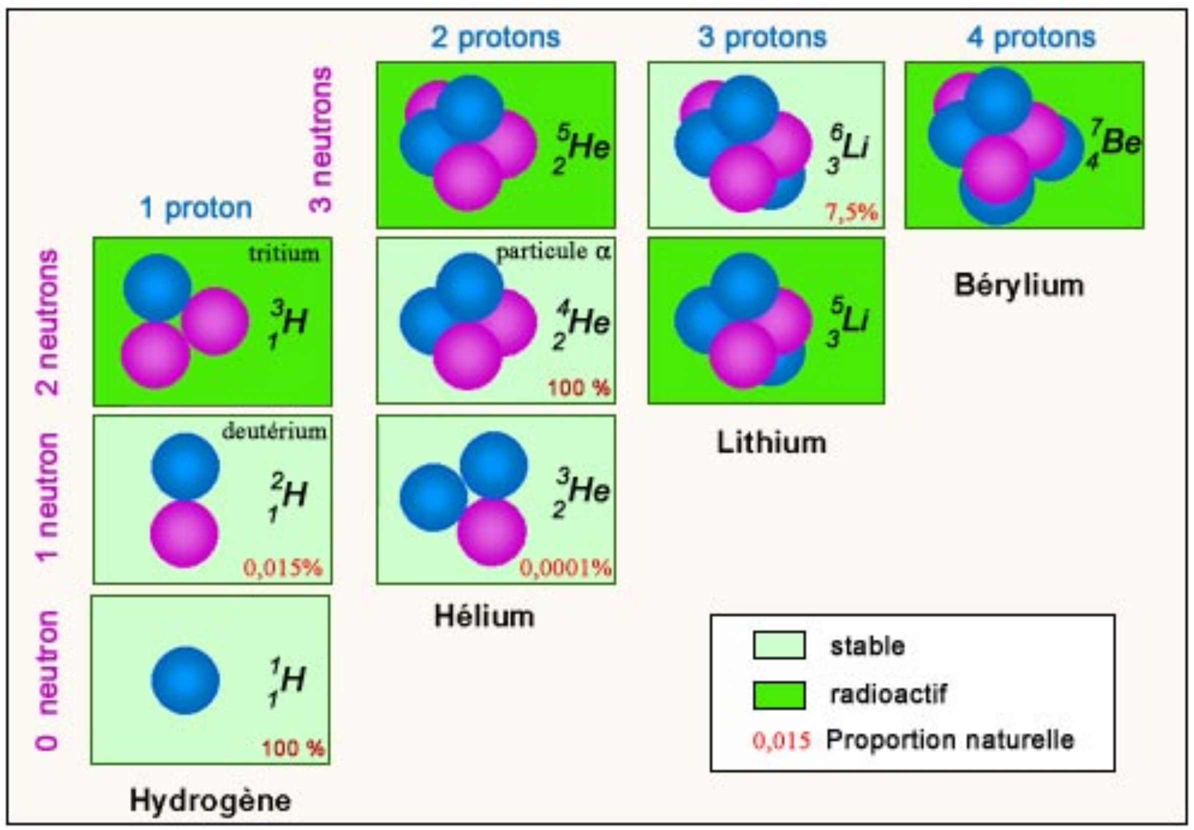
The atomic nucleus



Nuclear energy involves nucleus transformation processes

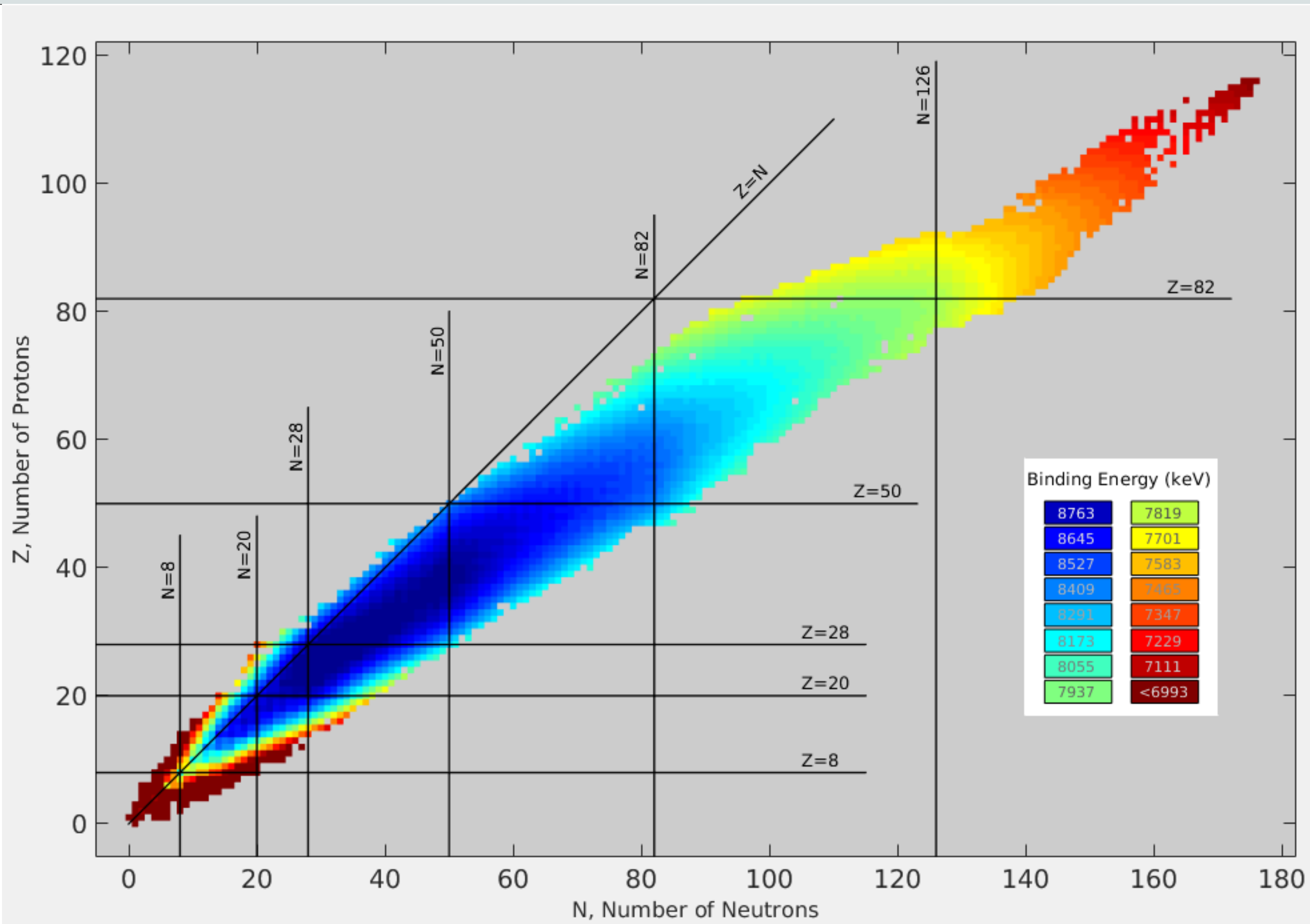
- Atomic nucleus is composed by two types of nucleons
  - The proton
  - The neutron

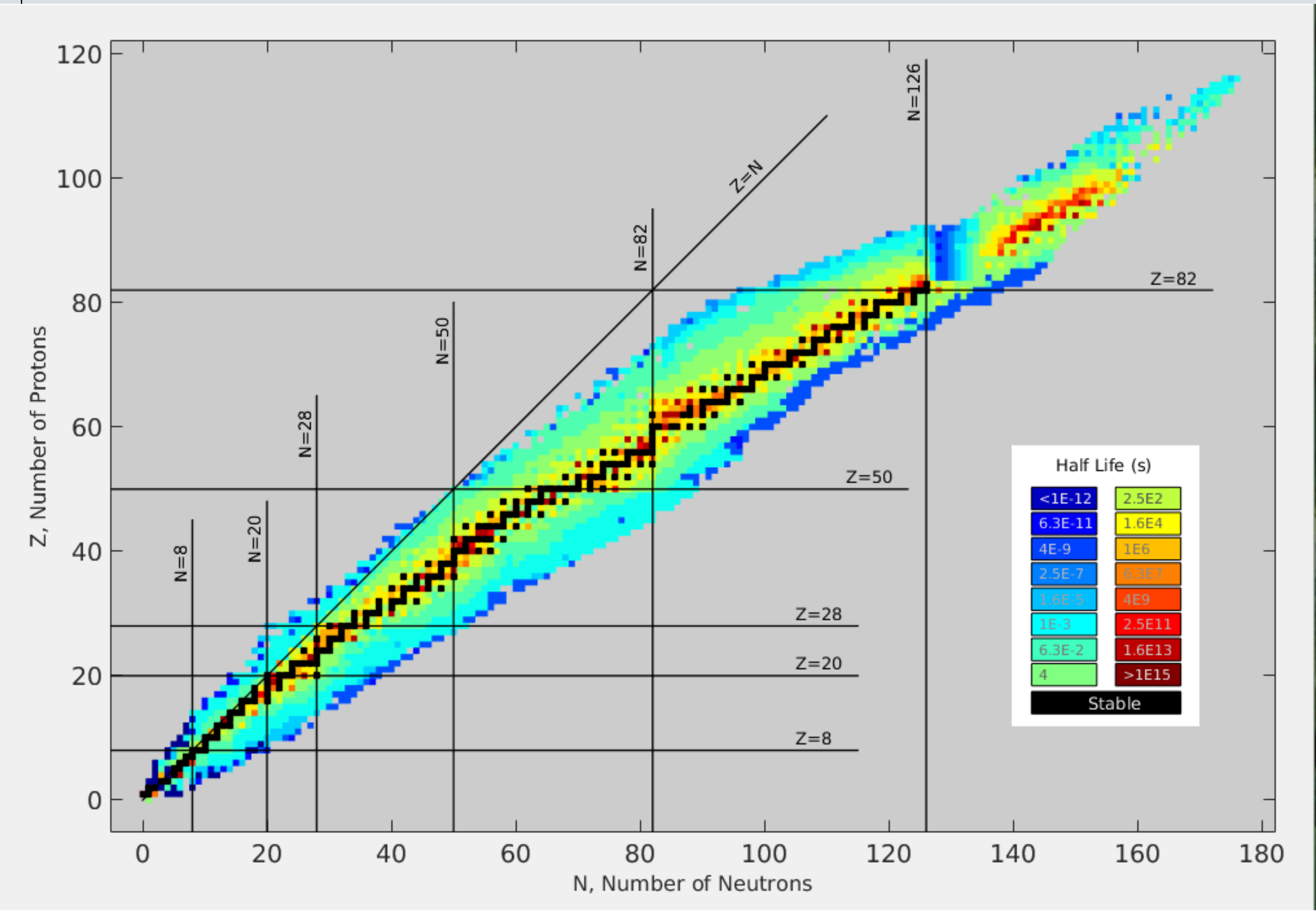




- Z is the atomic number
- A = Z + N is the mass number
- Stability depends on A and Z

Valley of stability



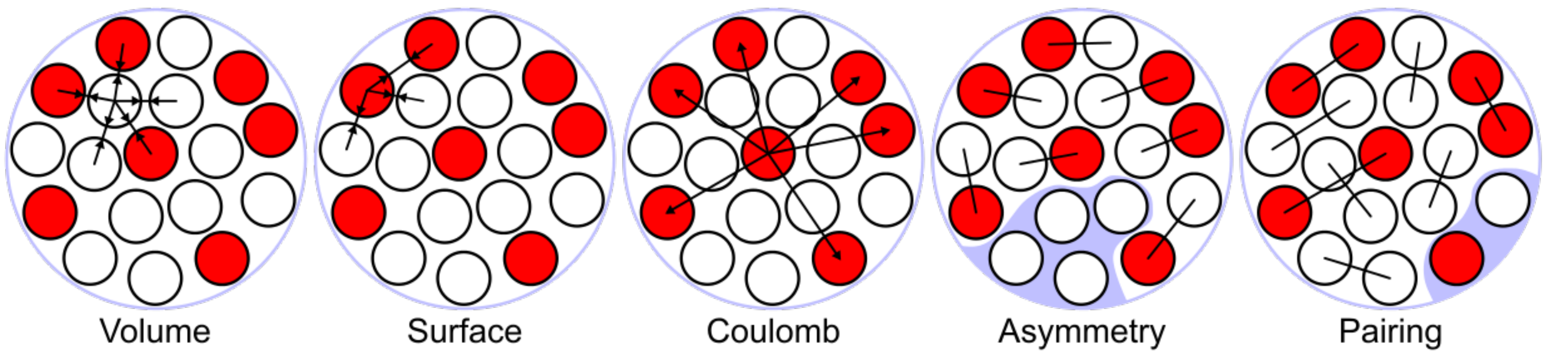


- The valley of stability is a representation of known isotopes on Z and N
- Isotopes binding energy study can explain the shape of valley of stability
- Binding energy is the energy required to separates a system of particles

$$-B(A,Z) = m(A,Z)c^2 - Zm_pc^2 - (A-Z)m_nc^2$$

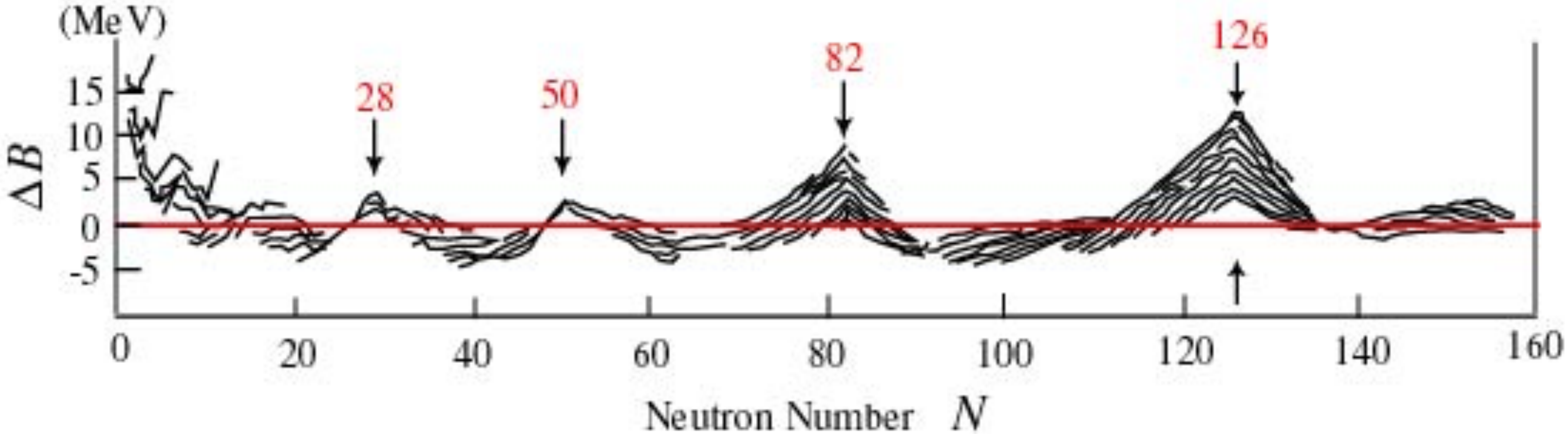
Liquid drop model

- A simple nucleus model to explain nucleus binding energy

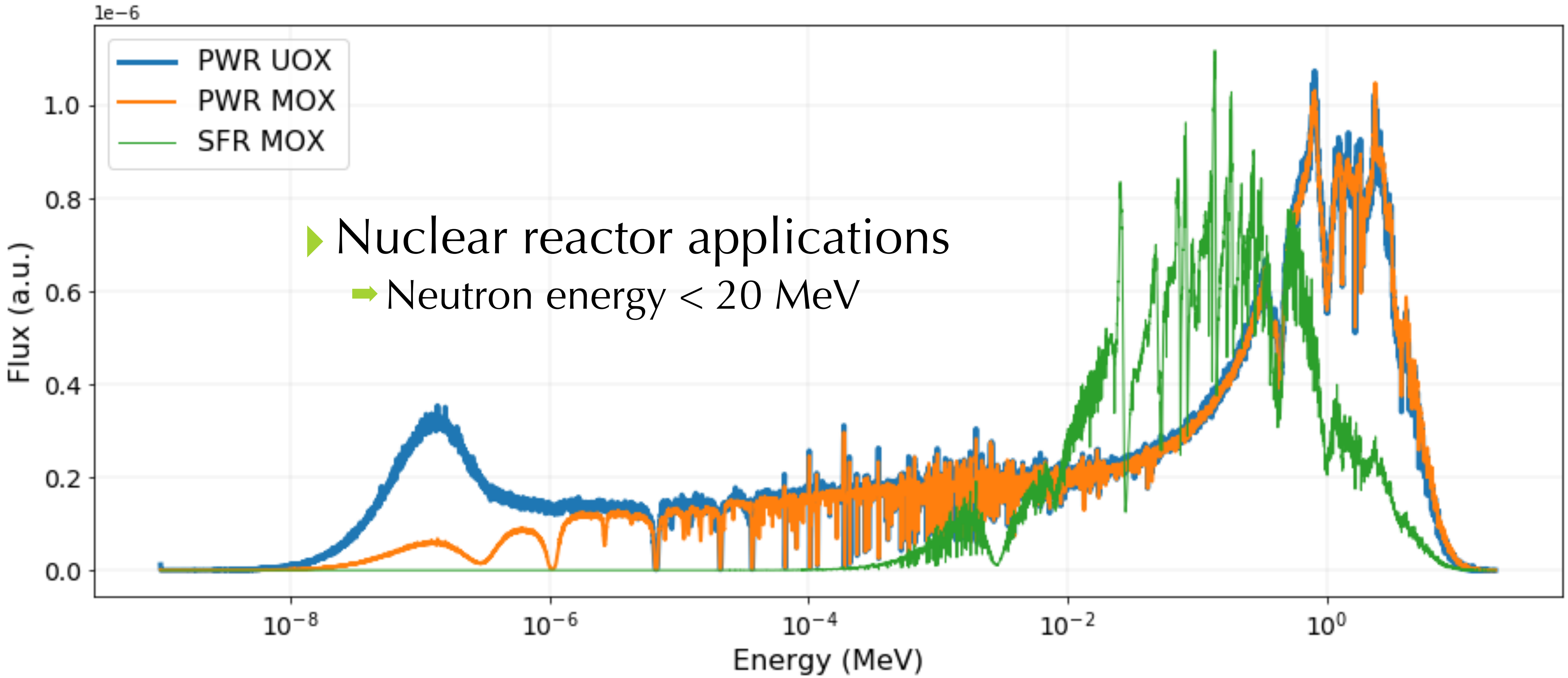


$$B(A,Z) = a_v A - a_s A^{2/3} - a_c \frac{Z^2}{A^{1/3}} - a_a \frac{(N-Z)^2}{A} + c$$

- But with some deviations with experimental data



Neutron main interactions @ $0 < E_n < 20$ MeV



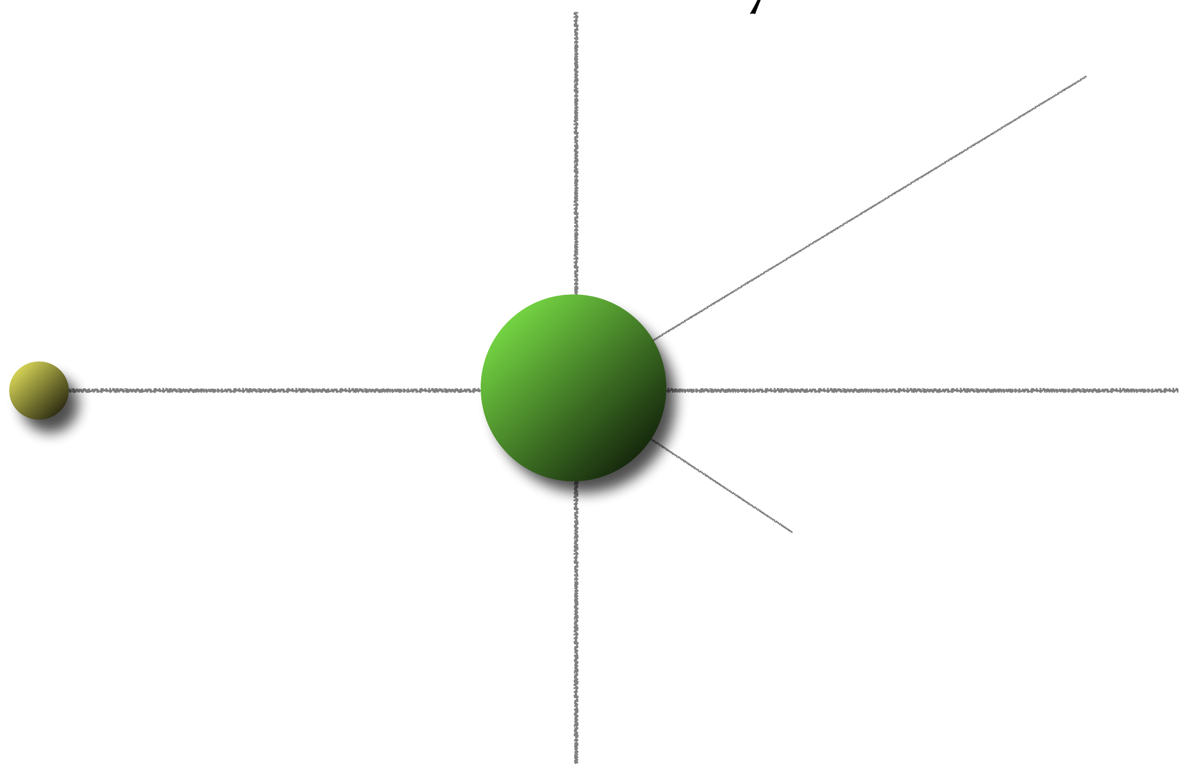
- Nuclear reactor applications
  - ➡ Neutron energy < 20 MeV

- Neutron absorption
  - Fission
  - Radiative capture
  - (n, xn) with n =2, 3, …

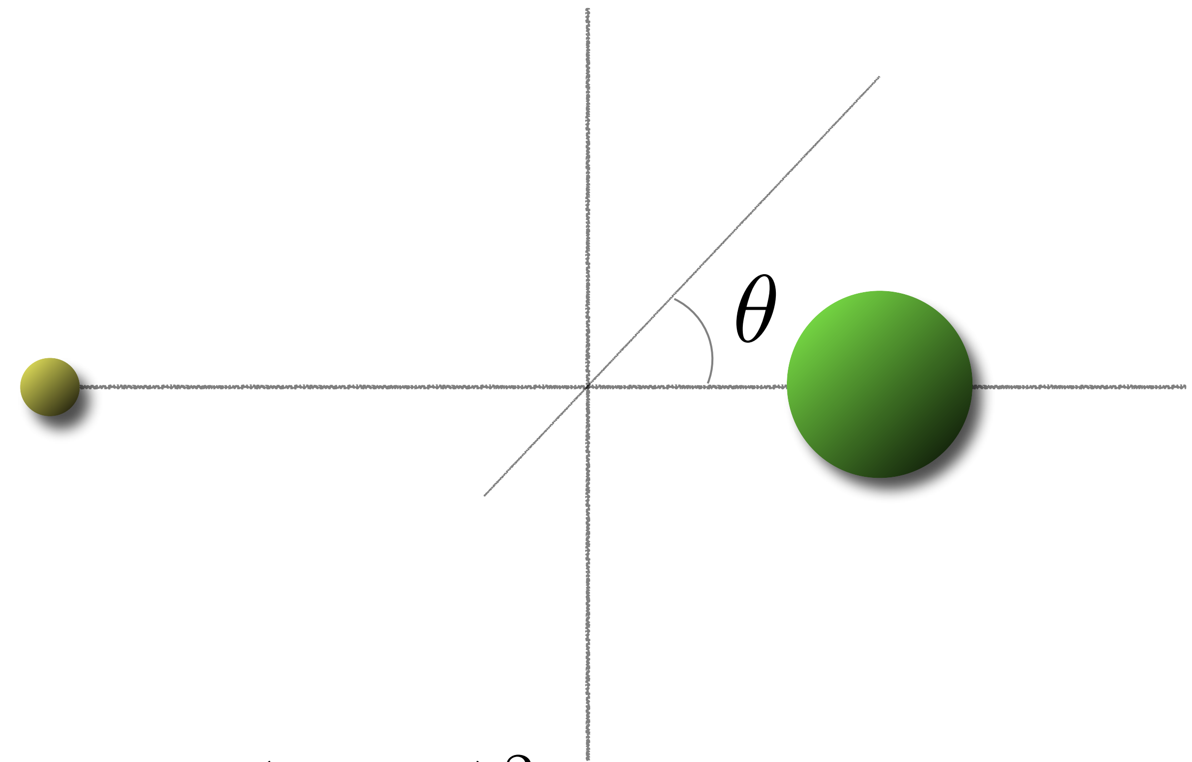
- Neutron scattering
  - Elastic
  - Inelastic

Neutron elastic scattering

- Laboratory



- Center of mass



$$E_{\mathrm{nf}} = \frac{E_{\mathrm{ni}}}{2}\left((1+\alpha) + (1-\alpha)\cos\theta\right)$$

$$\alpha = \frac{(1-A)^2}{(1+A)^2}$$

- Backscattering on hydrogen

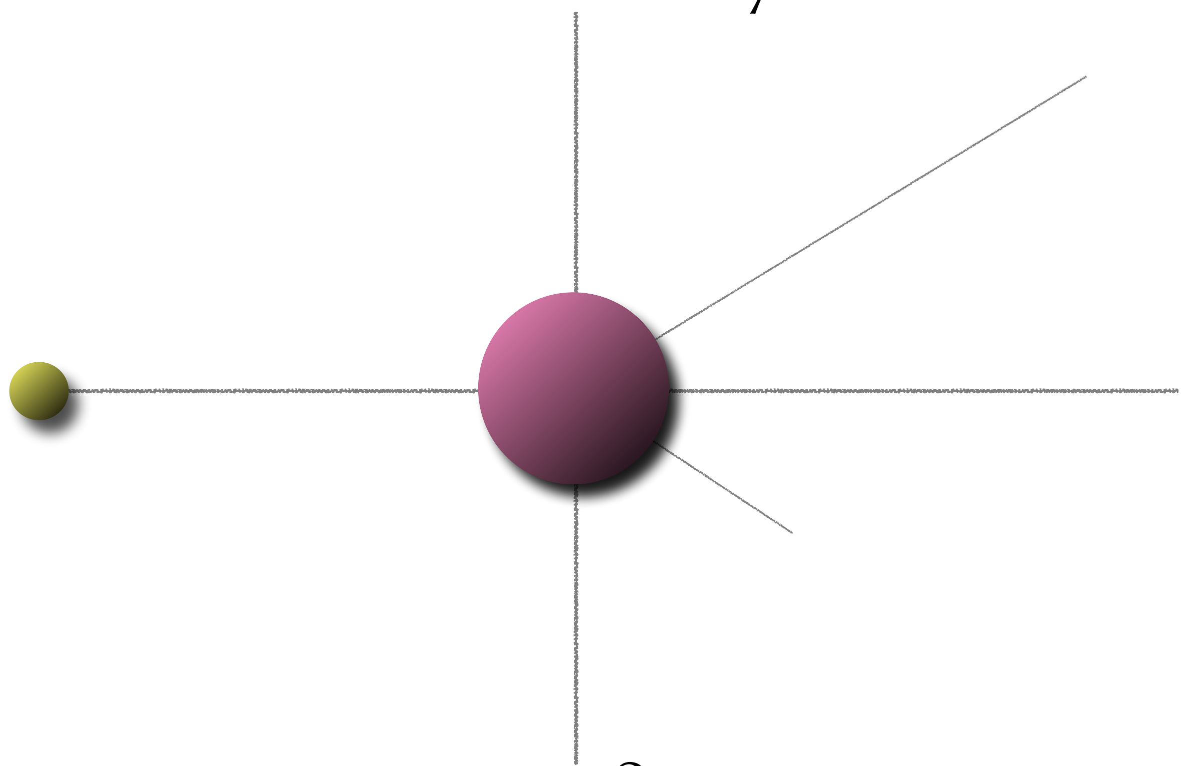
$$\alpha = 0$$
$$\cos(\theta) = -1$$
$$E_{\mathrm{nf}} = 0$$

- Backscattering on lead

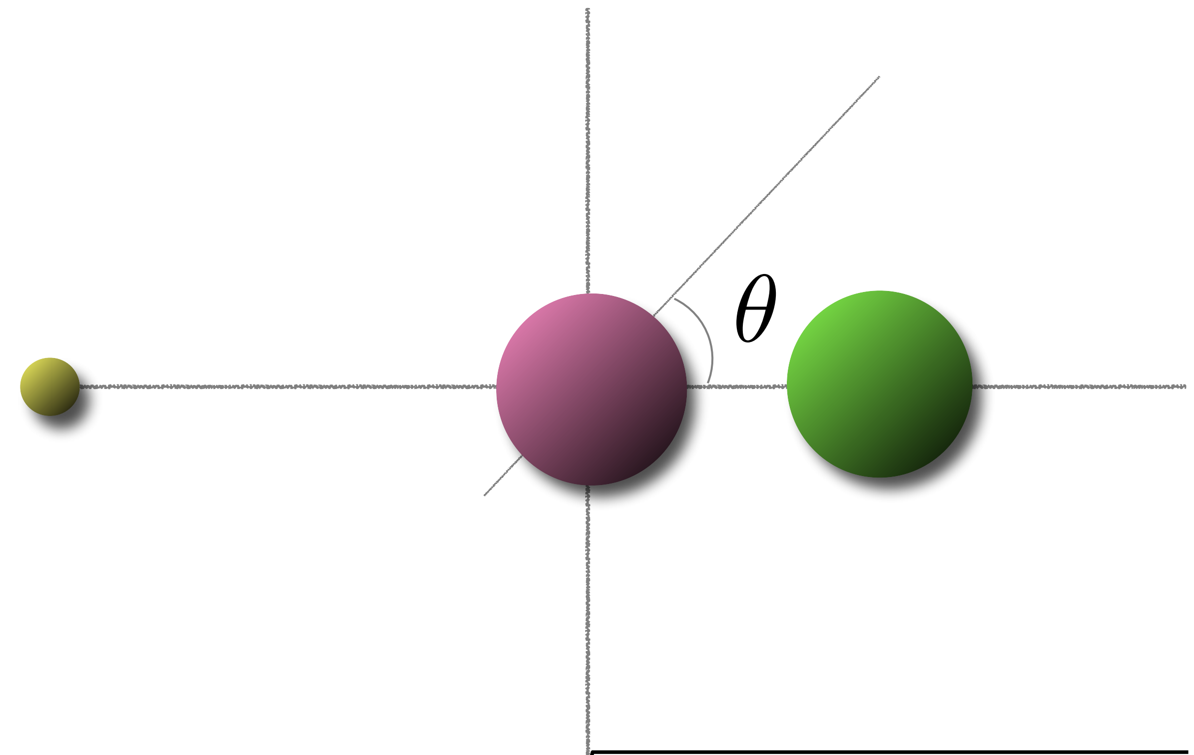
$$\alpha = 0.98$$
$$\cos(\theta) = -1$$
$$E_{\mathrm{nf}} = 0.99\ E_{\mathrm{ni}}$$

Neutron inelastic scattering

- Laboratory



- Center of mass



$$E_{\mathrm{nf}} = E_{\mathrm{ni}} \frac{1+\gamma^2+2\gamma\cos(\theta)}{(1+A)^2}$$

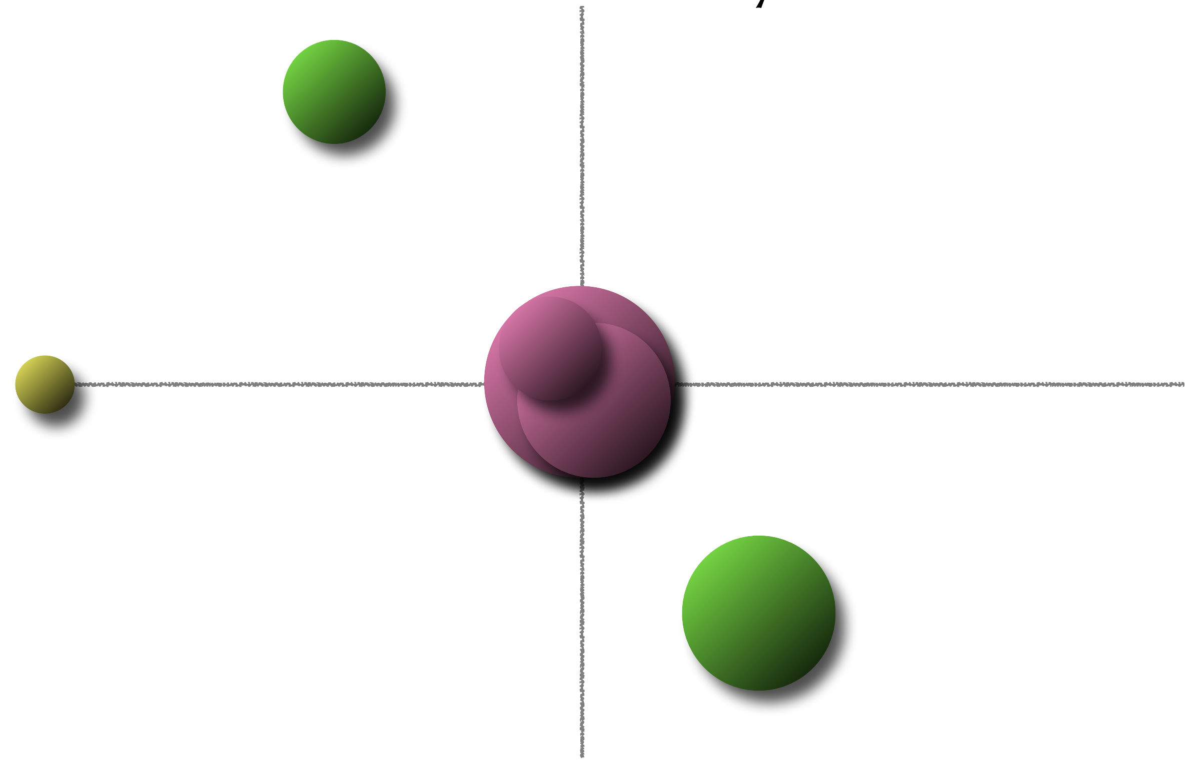
$$\gamma = A\sqrt{1-\frac{A+1}{A}\cdot\frac{Q}{E_0}}$$

- Q is the excitation energy of the target nucleus
- Inelastic scattering is a threshold reaction

$$E_{\mathrm{threshold}} = \frac{A+1}{A}\cdot Q$$

Neutron induces fission

- Laboratory



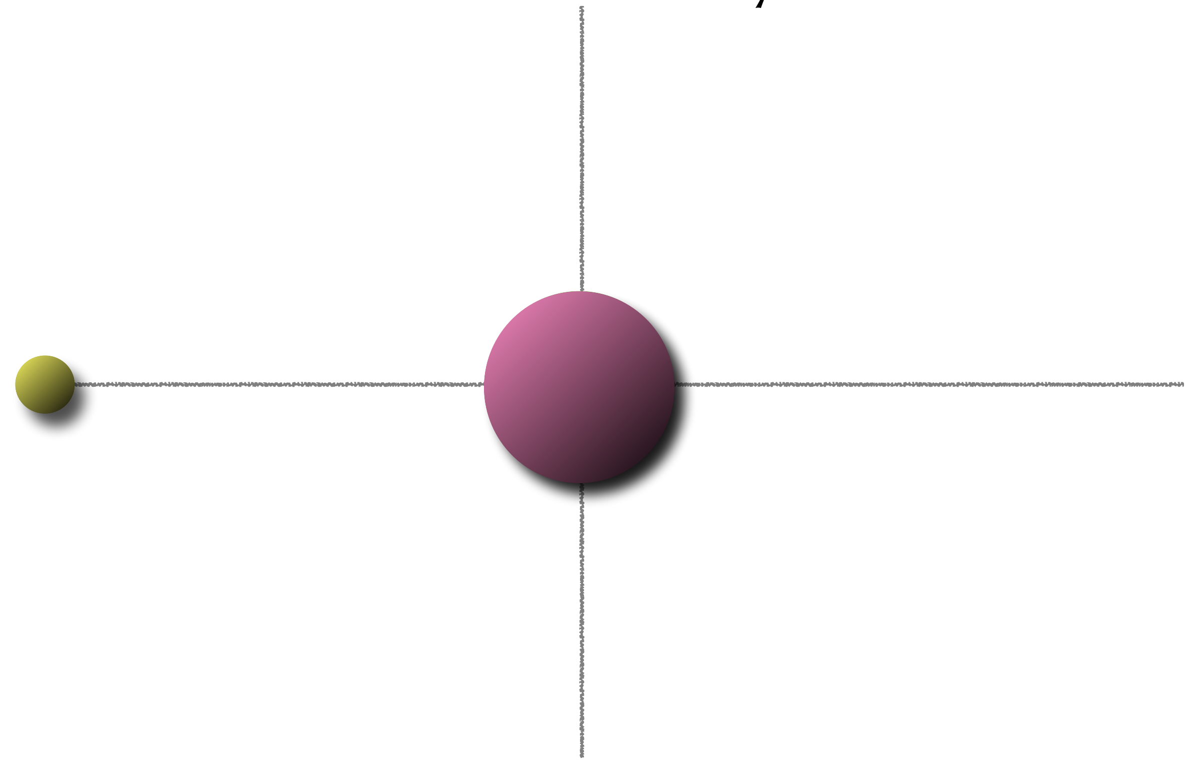
- Compound nucleus
- Prompt emission
  - Fission products
  - Prompt neutrons (~3 / fission)
  - Gamma rays
- Delayed emission
  - $\beta^- + \bar{\nu}$
  - Delayed neutrons (~0.01 / fission)

Energy released (thermal neutron on ^{235}U)

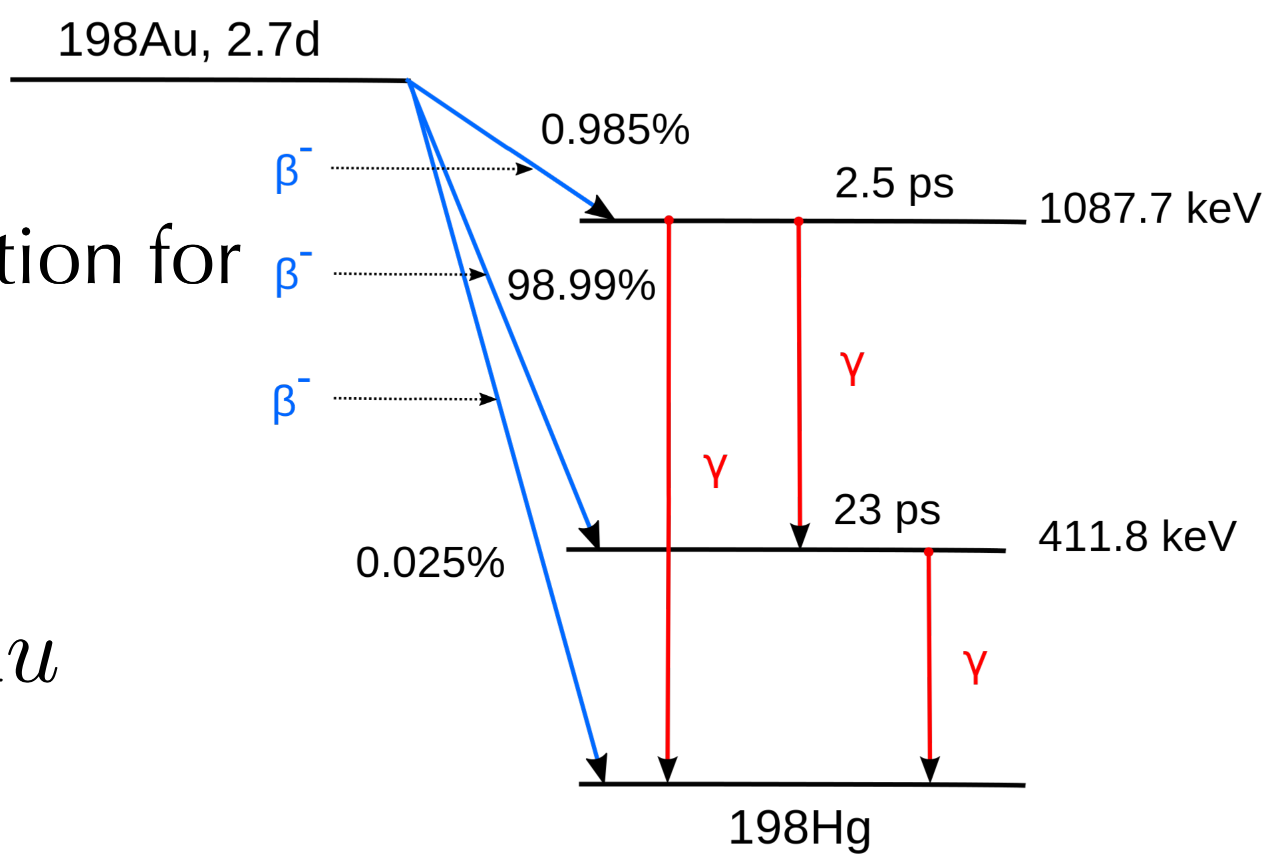
Prompt	Energy (MeV)	Delayed	Energy (MeV)
Fission Products	169	Beta -	6.4
Neutrons	4.8	Neutrons	0.010
Gammas	7.0	Gammas	6.2
		Neutrinos	10.0
Total	180.8	Total	22.610
TOTAL		203.410	

Radiative capture

- Laboratory



- Compound nucleus
- Prompt emission
  - Gamma rays

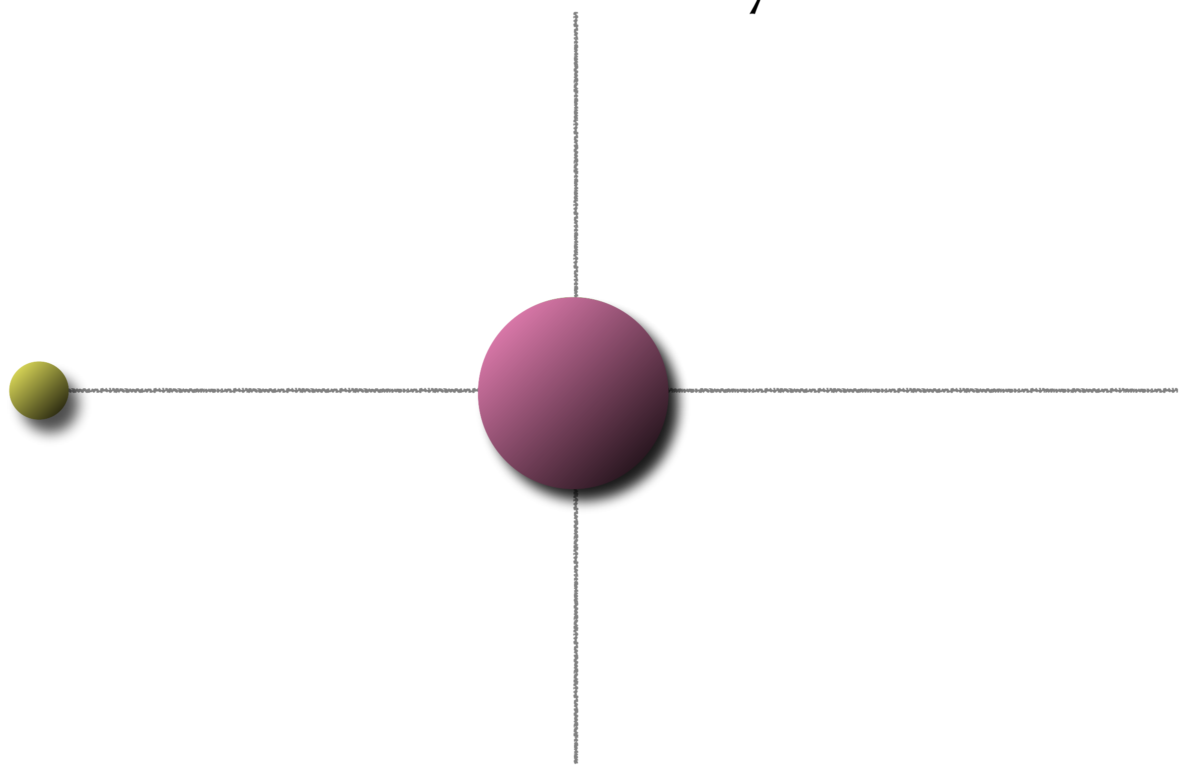


- Neutron capture is an important reaction for
  - Nuclear reactors reactivity
  - High mass nucleus nucleosynthesis
- Example :

$$^{197}Au\,(n,\gamma)\,^{198}Au$$

n,xn

- Laboratory



- Compound nucleus
- Prompt emission
  - 2 or more neutrons
  - Gamma rays

- (n, xn) are threshold reactions

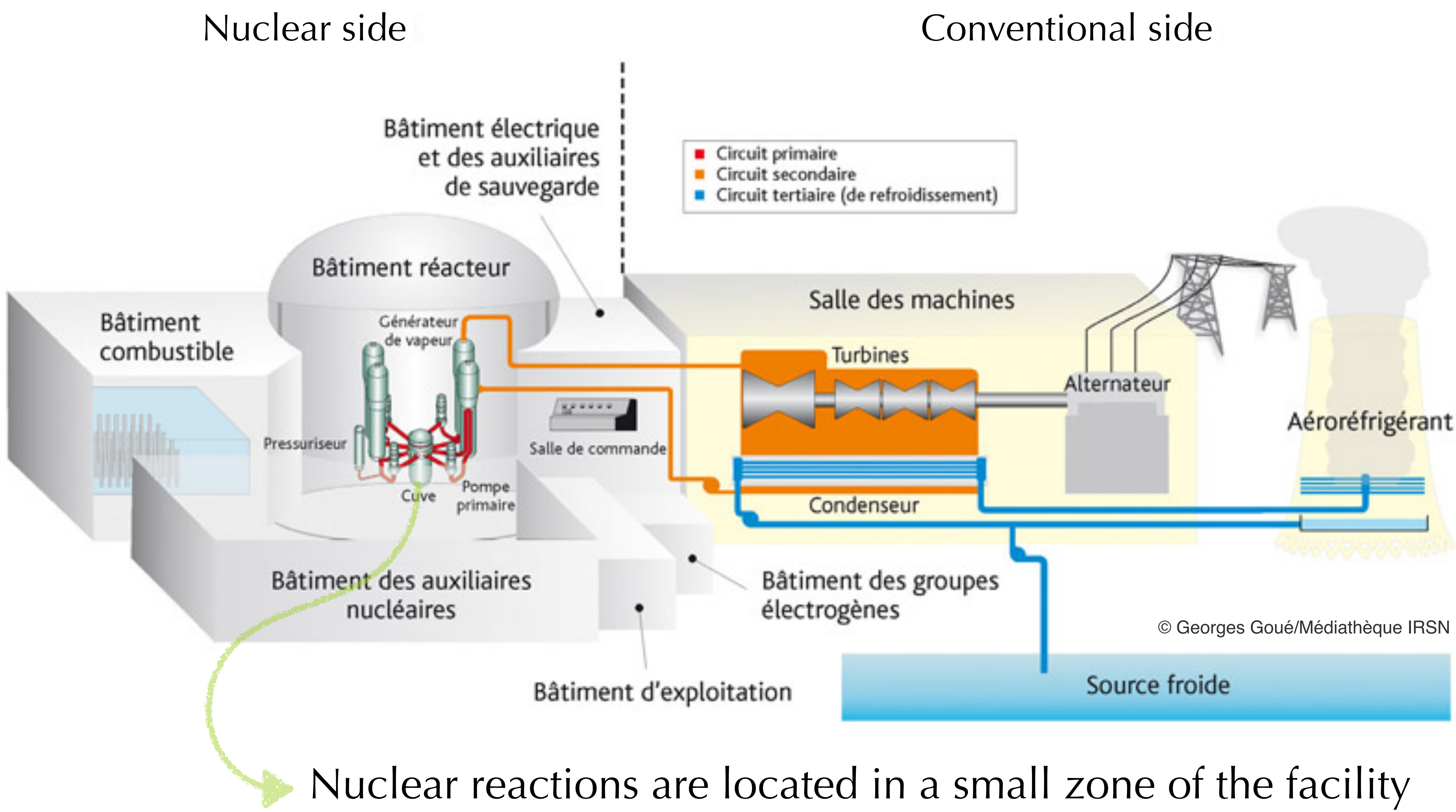
$$E_{\text{threshold}} = \frac{A+1}{A} E_l$$

- Example

$$n +^{238} U \longrightarrow^{237} U + n' + n''$$

$$^{237}U \longrightarrow^{237} Np + \beta^- + \bar{\nu}$$

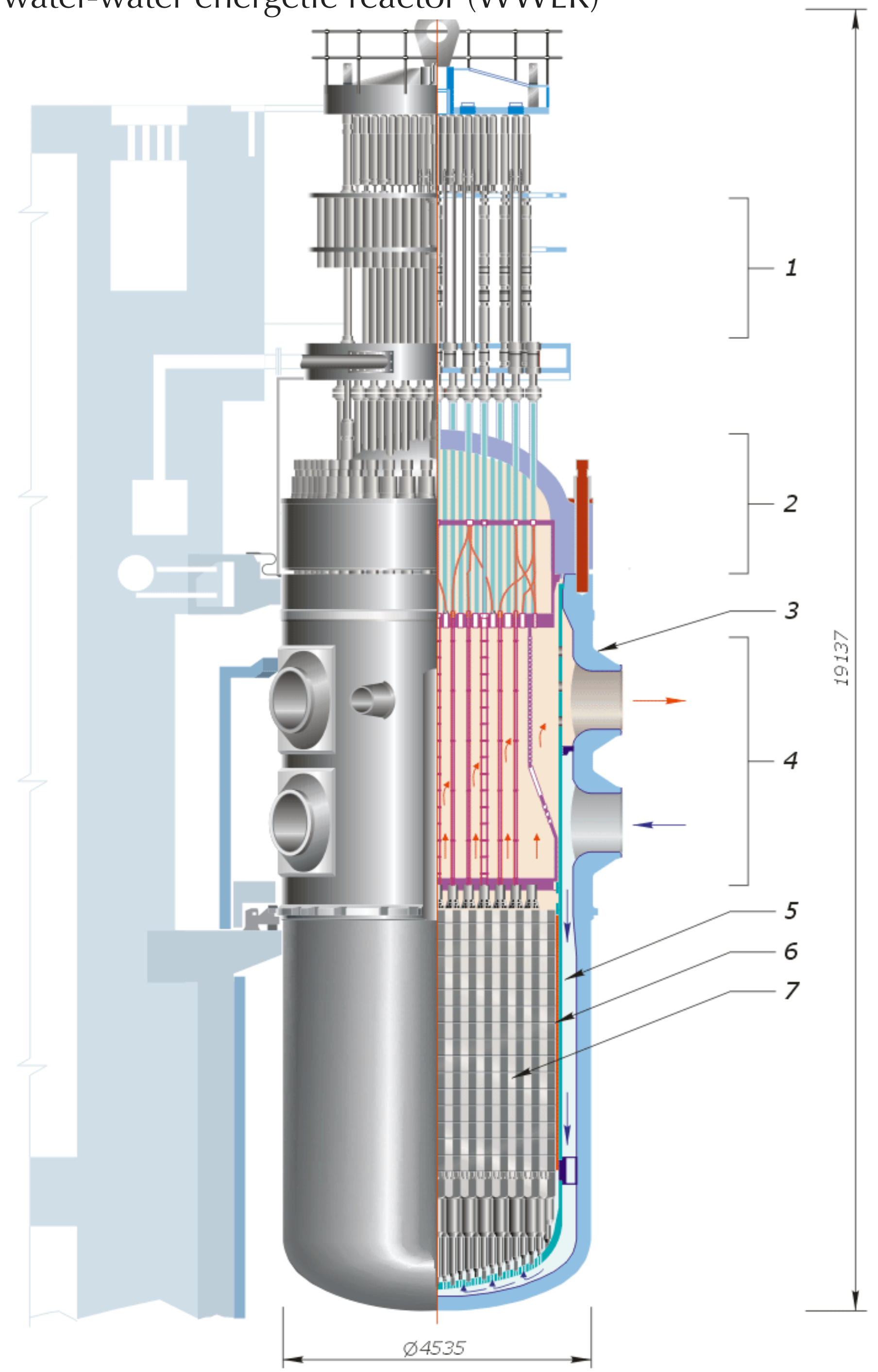
A nuclear reactor example

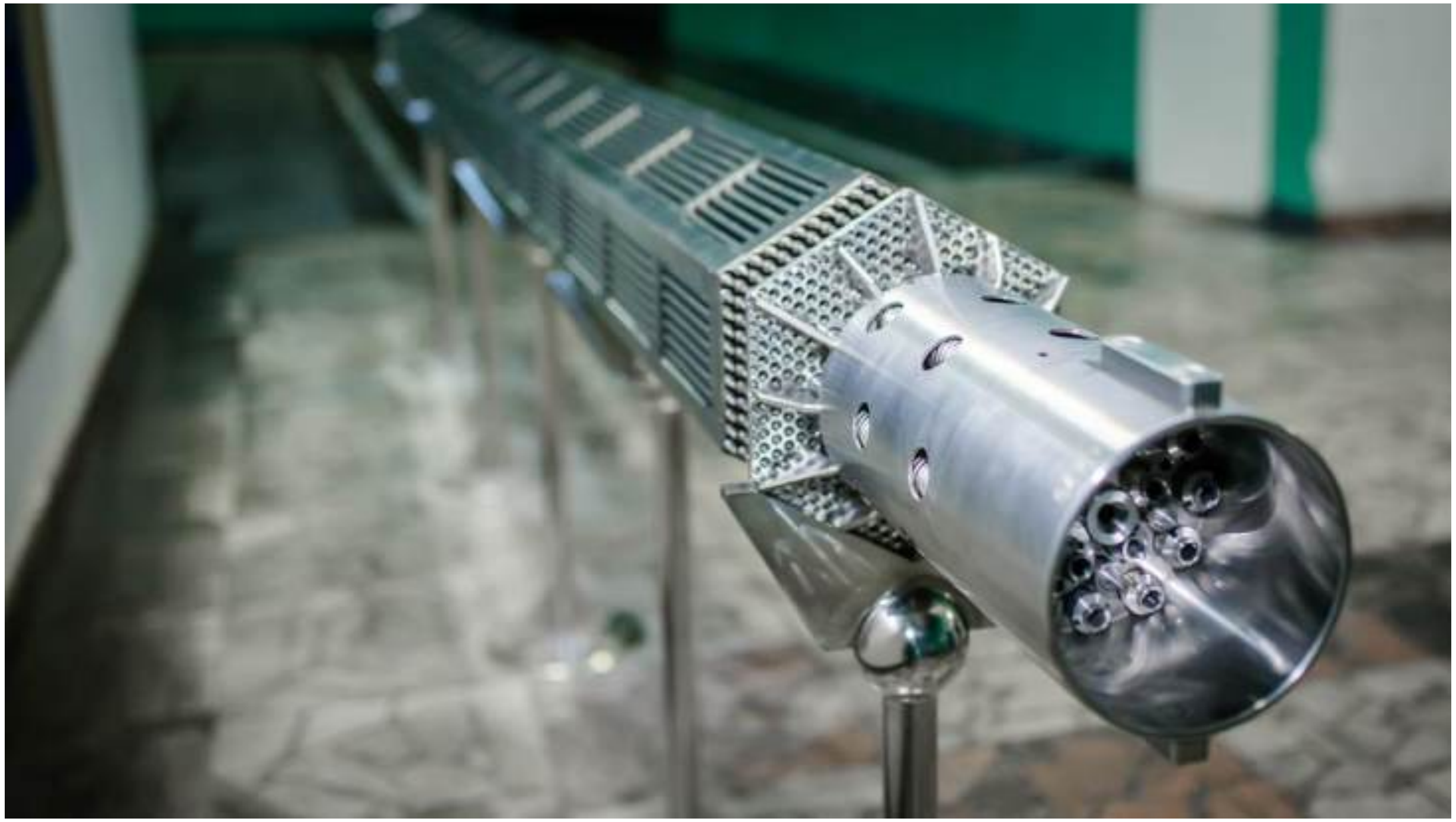


Nuclear reactions are located in a small zone of the facility

A nuclear reactor core example

water-water energetic reactor (WWER)





The modified fuel for use at Paks and Loviisa (Image: Rosatom)

By ORNL - originally ornl.gov now [1],
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Photo: Reuters

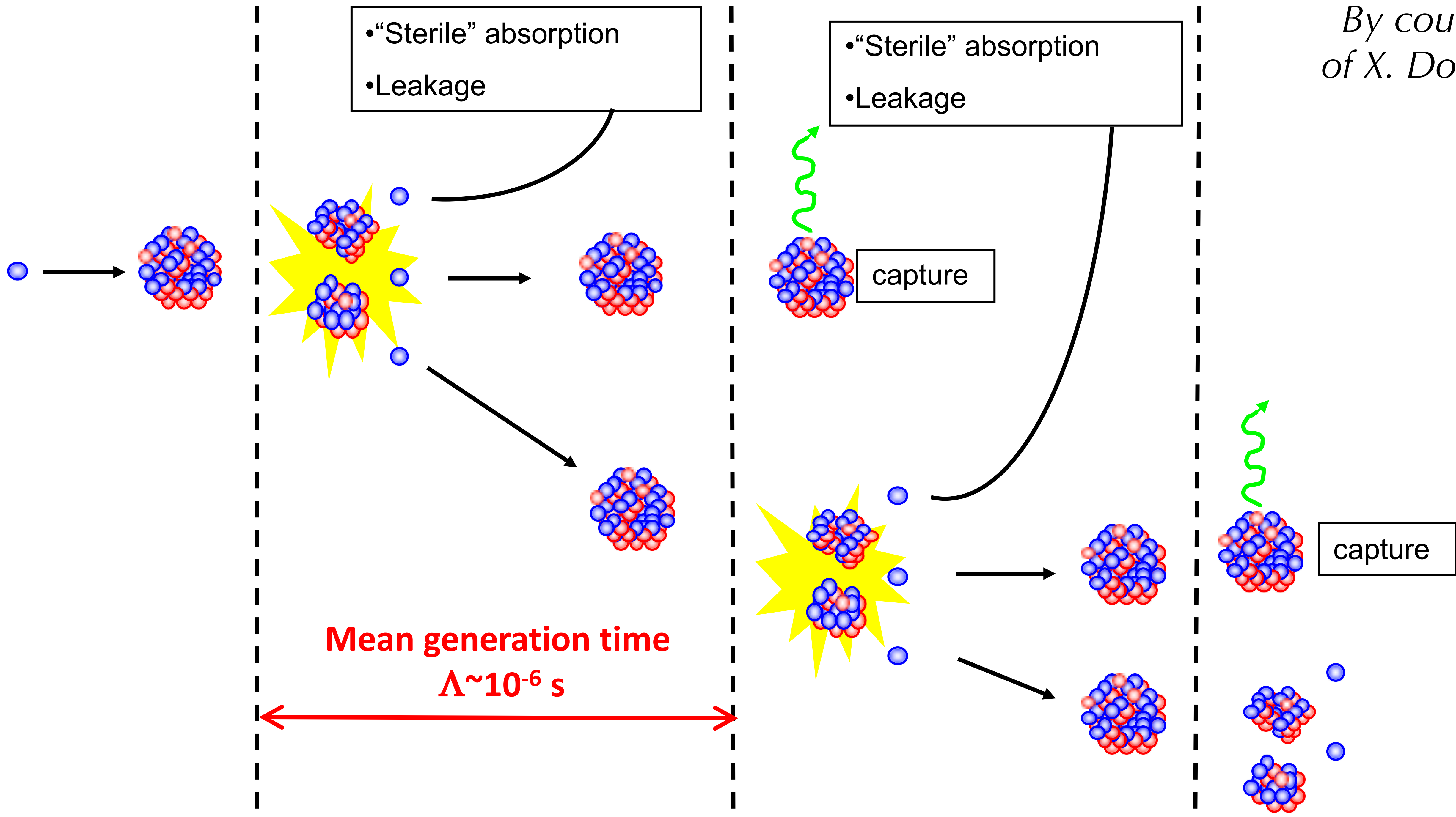
Chain Reaction example

For instance, a uranium-235 fission induces 2 fission products and is followed by the emission of 2 or 3 neutrons that can then hit other nucleus and so on… This is a « chain reaction".

suite

The controlled chain reaction

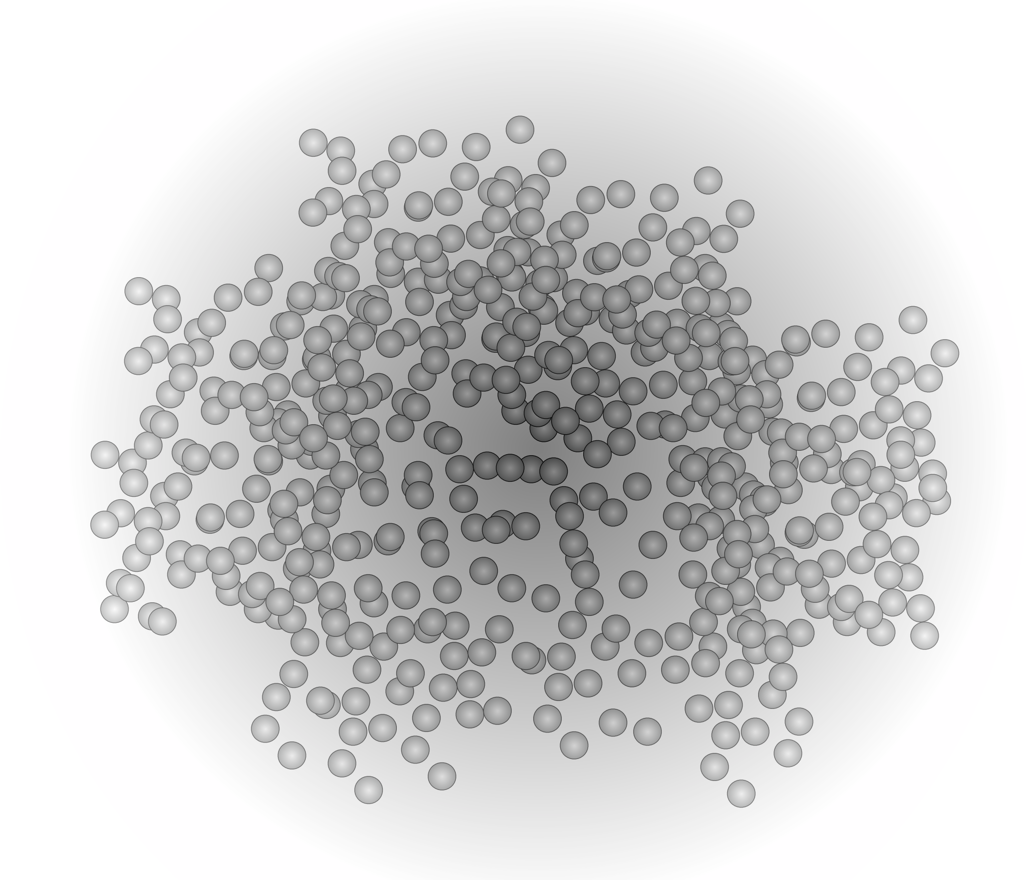
By courtesy of X. Doligez



- If material composition is almost fixed, fission rate is constant over time

Neutron multiplication factor

Infinite medium



$$k_{\text{inf}} = \frac{\text{number of created neutrons}}{\text{number of absorbed neutrons}}$$

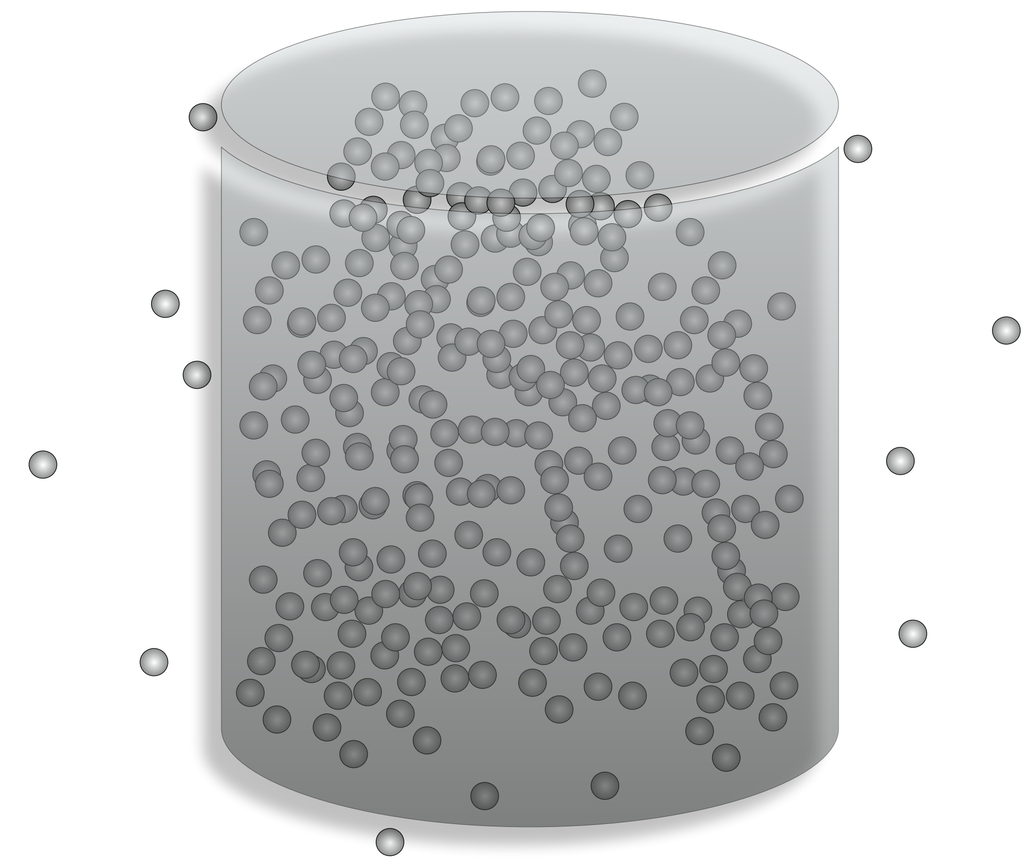
$$k_{\text{inf}} = \frac{\boxed{\nu}\,\boxed{\Gamma_f}}{\boxed{\Gamma_a}}$$

Number of emitted neutron per fission

Fission rate [s-1]

Neutron absorption rate [s-1]

Finite medium



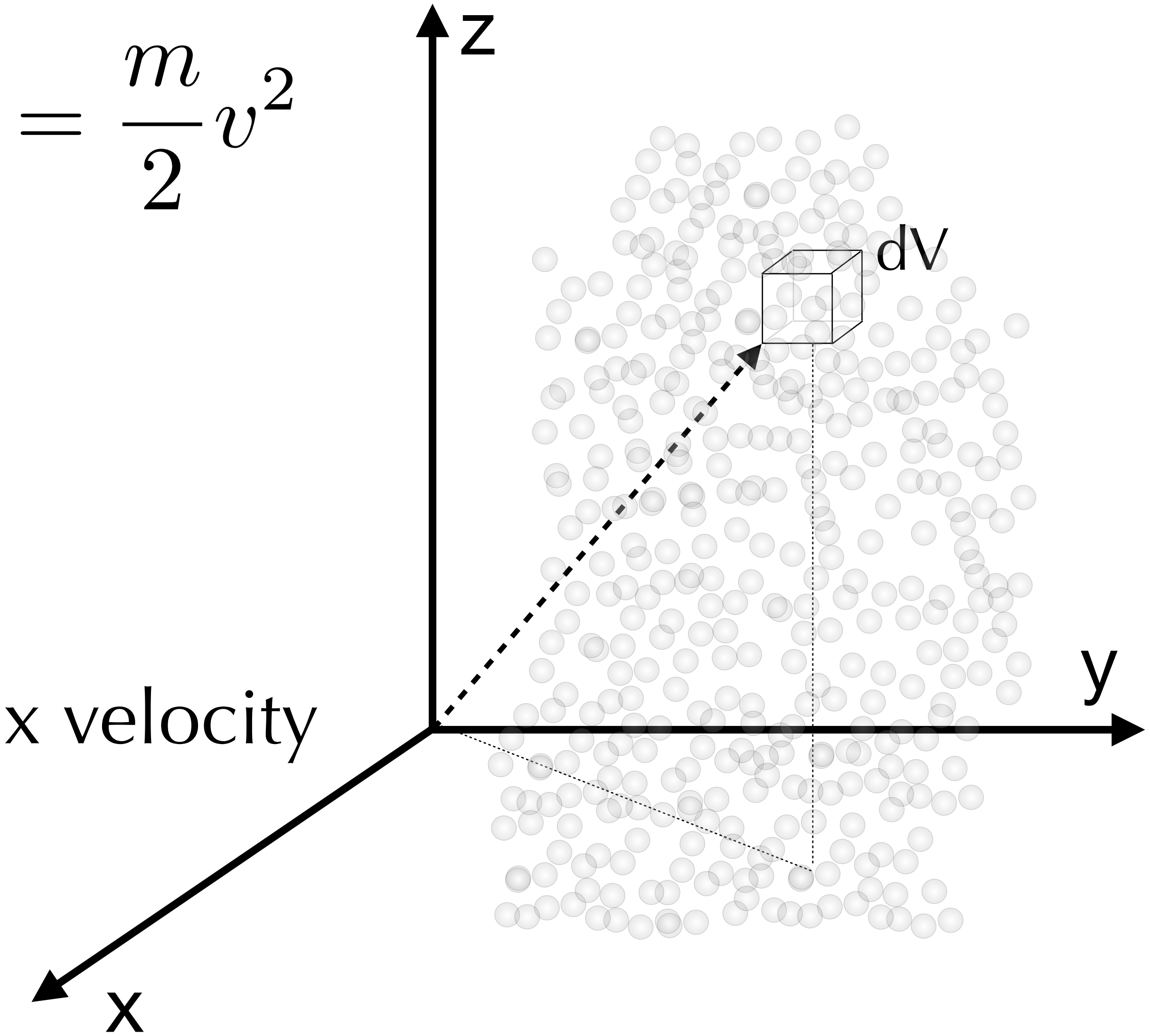
$$k_{\text{eff}} = k_{\text{inf}} \cdot \boxed{P_{\text{nl}}}$$

Probability for « non-leakage »
=
(1 - leakage probability)

Neutron population and flux

- Number of neutron [-]
  - with energy between E and E+dE
  - in the volume dV at the position r
  - for neutron with direction in dΩ
  - at a time t

$$E = \frac{m}{2}v^2$$



$$\longrightarrow n(E, \vec{r}, \vec{\Omega}, t)dEdVd\vec{\Omega}$$

- Neutron flux in V = number / volume x velocity

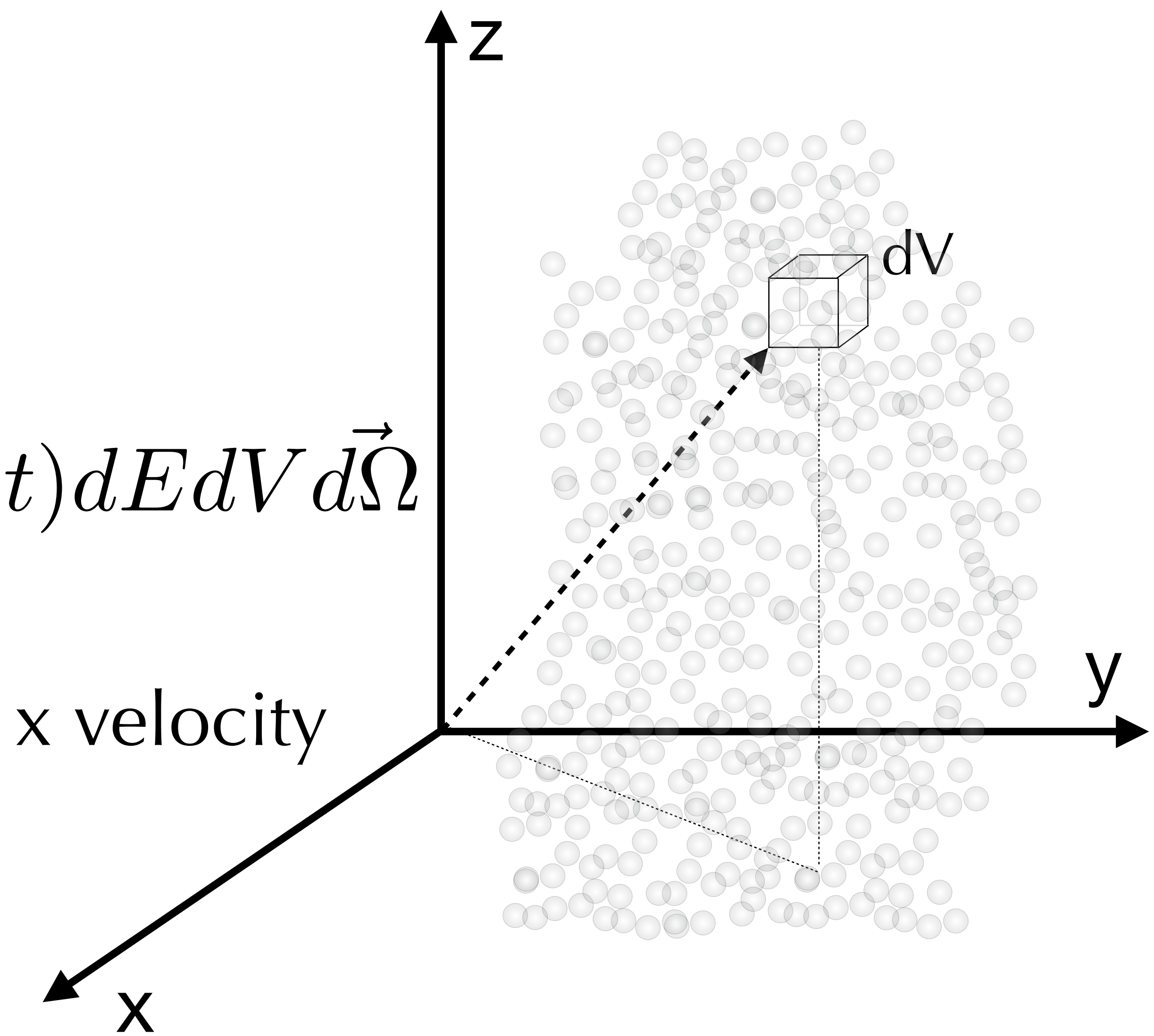
- Neutron flux [$cm^{-2}\ s^{-1}$]
  - with energy between E and E+dE
  - in the volume dV at the position r
  - for neutron with direction in dΩ
  - at a time t

$$\longrightarrow \varphi_{dV}(E, \vec{r}, \vec{\Omega}, t)dEd\vec{\Omega} = n(E, \vec{r}, \vec{\Omega}, t)dEd\vec{\Omega} \cdot v$$

Neutron population and flux

- Number of neutron [-]
  - with energy between E and E+dE
  - in the volume dV at the position r
  - at a time t

$$n(E,\vec{r},t)dEdV = \int_{\vec{\Omega}} n(E,\vec{r},\vec{\Omega},t)dEdV\,d\vec{\Omega}$$



- Neutron flux in V = number / volume x velocity

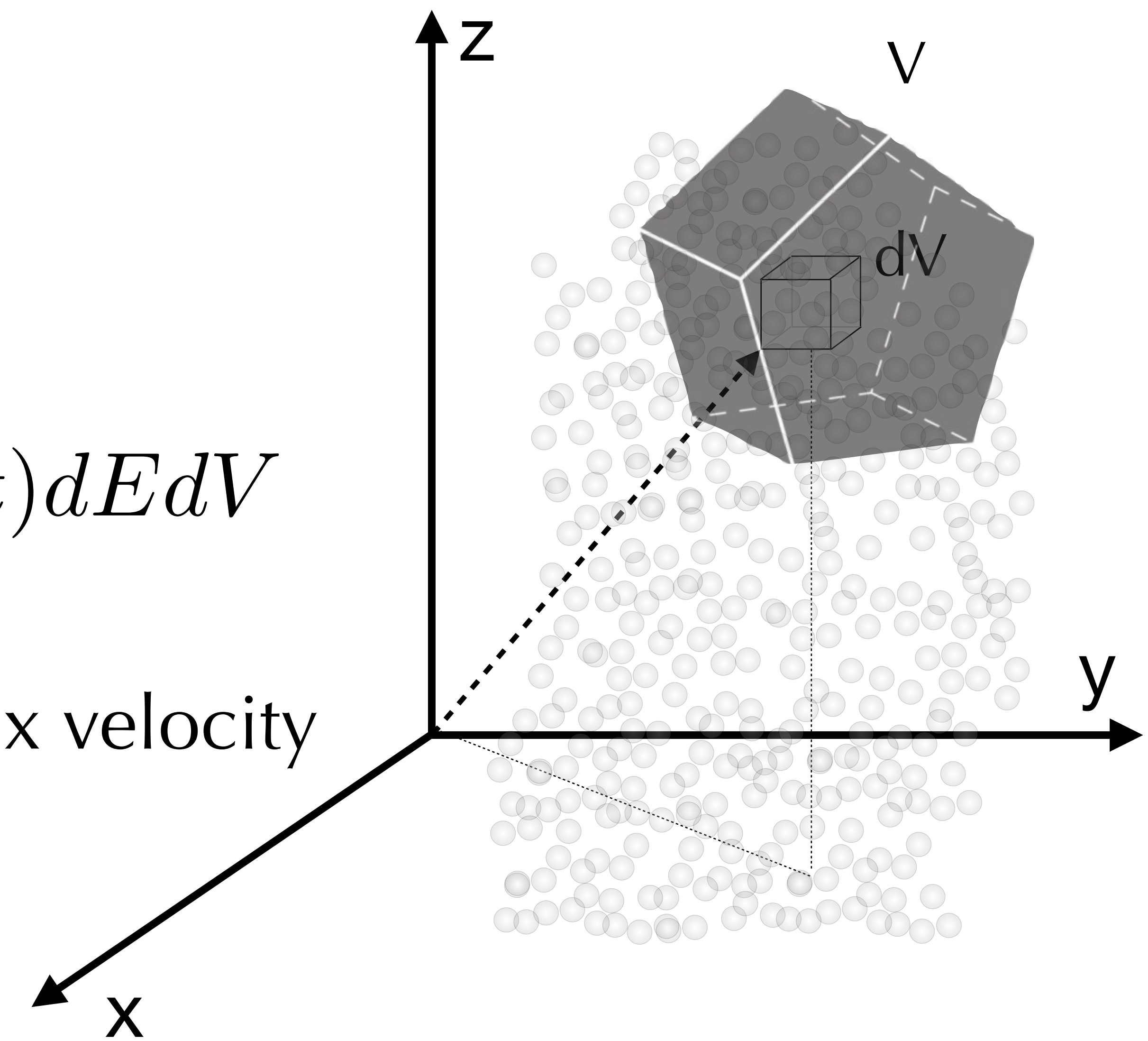
- Neutron flux [$cm^{-2}\ s^{-1}$]
  - with energy between E and E+dE
  - in the volume dV at the position r
  - at a time t

$$\varphi_{dV}(E,\vec{r},t)dE = n(E,\vec{r},t)dE \cdot v$$

Neutron population and flux

- Number of neutron [-]
  - with energy between E and E+dE
  - in the volume V
  - at a time t

$$\longrightarrow \quad N_V(E,t)dE = \int_V n(E,\vec{r},t)dEdV$$



- Neutron flux in V = number / volume x velocity

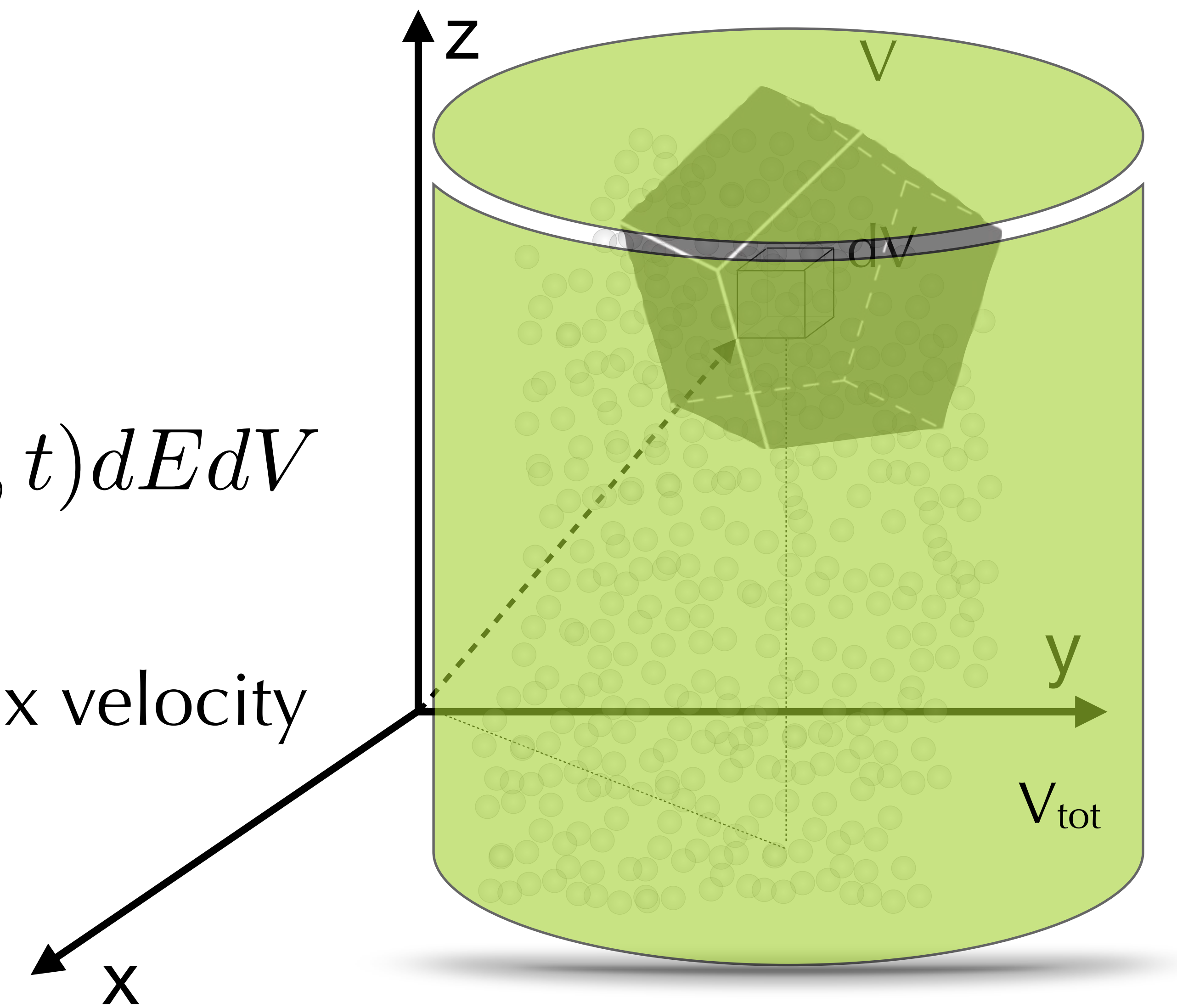
- Neutron flux [cm-2 s-1]
  - with energy between E and E+dE
  - in the volume V
  - at a time t

$$\longrightarrow \quad \varphi_V(E,\vec{r},t)dE = \frac{N_V(E,t)dE}{V} \cdot v$$

Neutron population and flux

- Number of neutron [-]
  - with energy between E and E+dE
  - in the volume V_{tot}
  - at a time t

$$\longrightarrow N_{V_{tot}}(E,t)dE = \int_{V_{tot}} n(E,\vec{r},t)dEdV$$



- Neutron flux in V = number / volume x velocity

- Neutron flux [cm^{-2} s^{-1}]
  - with energy between E and E+dE
  - in the volume V_{tot}
  - at a time t

$$\longrightarrow \varphi_{V_{tot}}(E,t)dE = \frac{n_{tot}(E,t)dE}{V_{tot}} \cdot v$$

Neutron population and flux

- Neutron flux [cm^{-2} s^{-1}]
  - with energy between E and E+dE
  - in the volume V_{tot}
  - for neutron with direction in dΩ
  - at a time t

$$\longrightarrow \quad \varphi_{V_{tot}}(E, \vec{\Omega}, t)dEd\vec{\Omega}$$

- Neutron flux [cm^{-2} s^{-1}]
  - with energy between E and E+dE
  - in the volume V_{tot}
  - for neutron with direction in dΩ
  - <u>for a stationary state</u>

$$\longrightarrow \quad \varphi_{V_{tot}}(E, \vec{\Omega})dEd\vec{\Omega}$$

- Integration over all directions

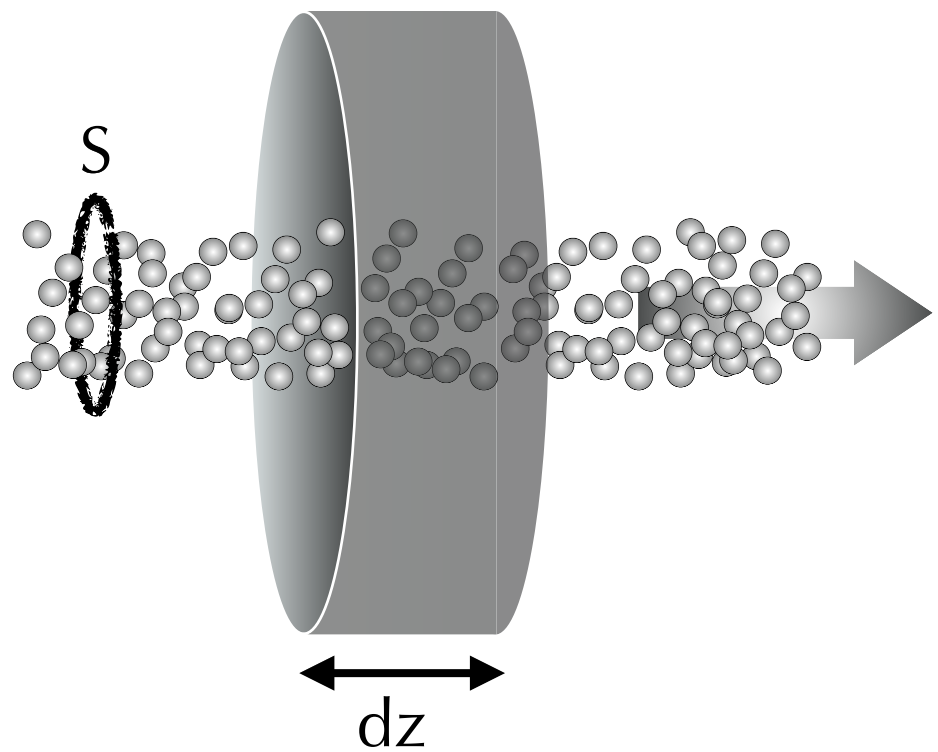
$$\longrightarrow \quad \varphi_{V_{tot}}(E)dE = \int_{d\vec{\Omega}} \varphi_{V_{tot}}(E, \vec{\Omega})dEd\vec{\Omega}$$

- Integration over all energies

$$\longrightarrow \quad \varphi_{V_{tot}} = \int_{E} \varphi_{V_{tot}}(E)dE$$

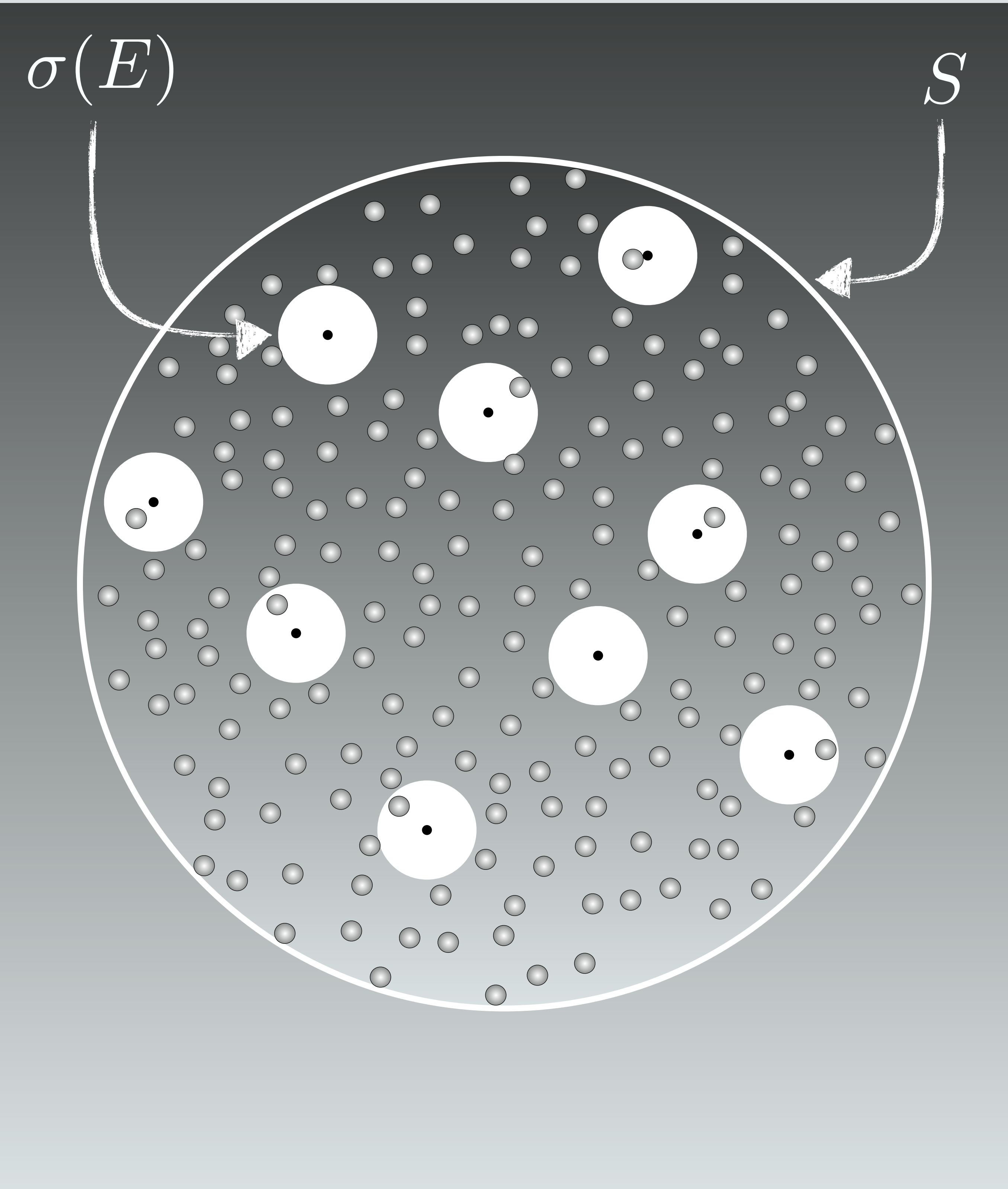
Reaction rate

- Mono-energetic neutron beam and stationary state
- Infinitely thick target so the interaction probability is very small
  - A neutron can cross the target without interaction
  - A neutron can undergo one unique interaction
  - There is no more than one interaction per neutron
- Characteristics of the experiment
  - Neutron flux is φ(E) constrained in the surface *S*
  - Target thickness is *dz*
  - Target is composed of *N* nucleus per unit of volume
- Number of reaction between t and t+dt in Sdz :



$$N(t+dt)\cdot Sdz - N(t)\cdot Sdz = dN\cdot Sdz$$

Reaction rate and cross section



- Number of target nuclides in Sdz

$$N \cdot S \cdot dz$$

- Neutron flux crossing the surface S

$$\varphi_V(E) \cdot S$$

- Interaction probability on a nuclide

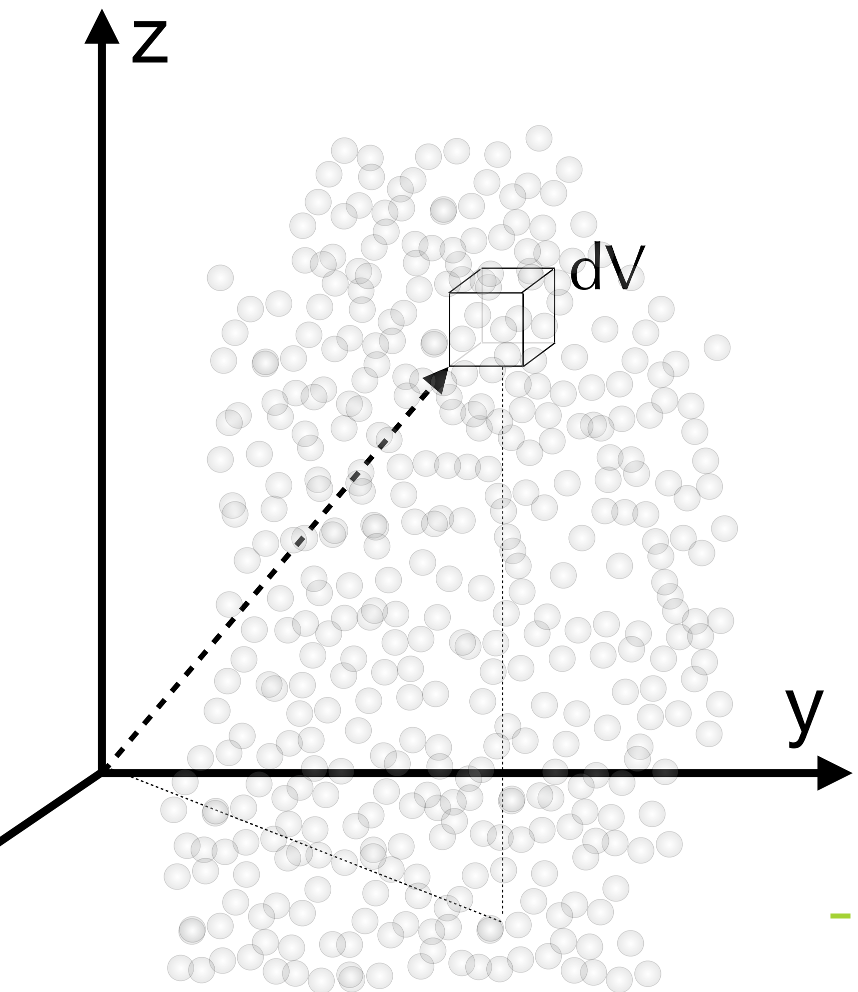
$$\frac{\sigma(E)}{S}$$

- Measurement duration dt

▸ Reaction rate :

$$\frac{dN}{dt} = -N \cdot \sigma(E) \cdot \varphi_V(E)$$

Boltzmann equation



- We consider a volume dV inside a volume V with diffusing neutrons

- The variation of the number of neutron in dV, with energy between E and E+dE, with a direction in dΩ, at a time between t and t+dt is

- [+] Ingoing neutrons in dV by diffusion during dt
- [-] Outgoing neutrons in dV by diffusion during dt

- [-] Neutron production in dE and dΩ by scattering
- [+] Neutron disparition of dE and dΩ by any nuclear reaction

- [+] Neutron produced by fission

Boltzmann equation

$$\frac{1}{v}\frac{d\phi(\vec{r},E,\vec{\Omega},t)}{dt}$$
$$= -div(\phi(\vec{r},E,\vec{\Omega},t).\vec{\Omega}) - \boxed{\Sigma_{total}(r,E)}.\phi(\vec{r},E,\vec{\Omega},t)$$
$$+ \iint_{4\pi} d^2\overrightarrow{\Omega'} \int_0^{\infty} dE'\, \Sigma(\vec{r},E')p(E' \to E,\overrightarrow{\Omega'} \to \vec{\Omega})\phi(\vec{r},E',\overrightarrow{\Omega'},t)$$
$$+ \frac{1}{4\pi}\boxed{\chi(E)} \iint_{4\pi} d^2\overrightarrow{\Omega'} \int_0^{\infty} dE'\, \boxed{\Sigma_{fiss}(\vec{r},E',\vec{\Omega},t)}.\phi(\vec{r},E',\overrightarrow{\Omega'},t).\boxed{\nu(E')} + \boxed{S(\vec{r},E',\overrightarrow{\Omega'},t)}$$

fission neutron spectrum
Total macroscopique cross section
Macroscopique fission cross section
Number of neutron emitted by fission
Source of neutrons

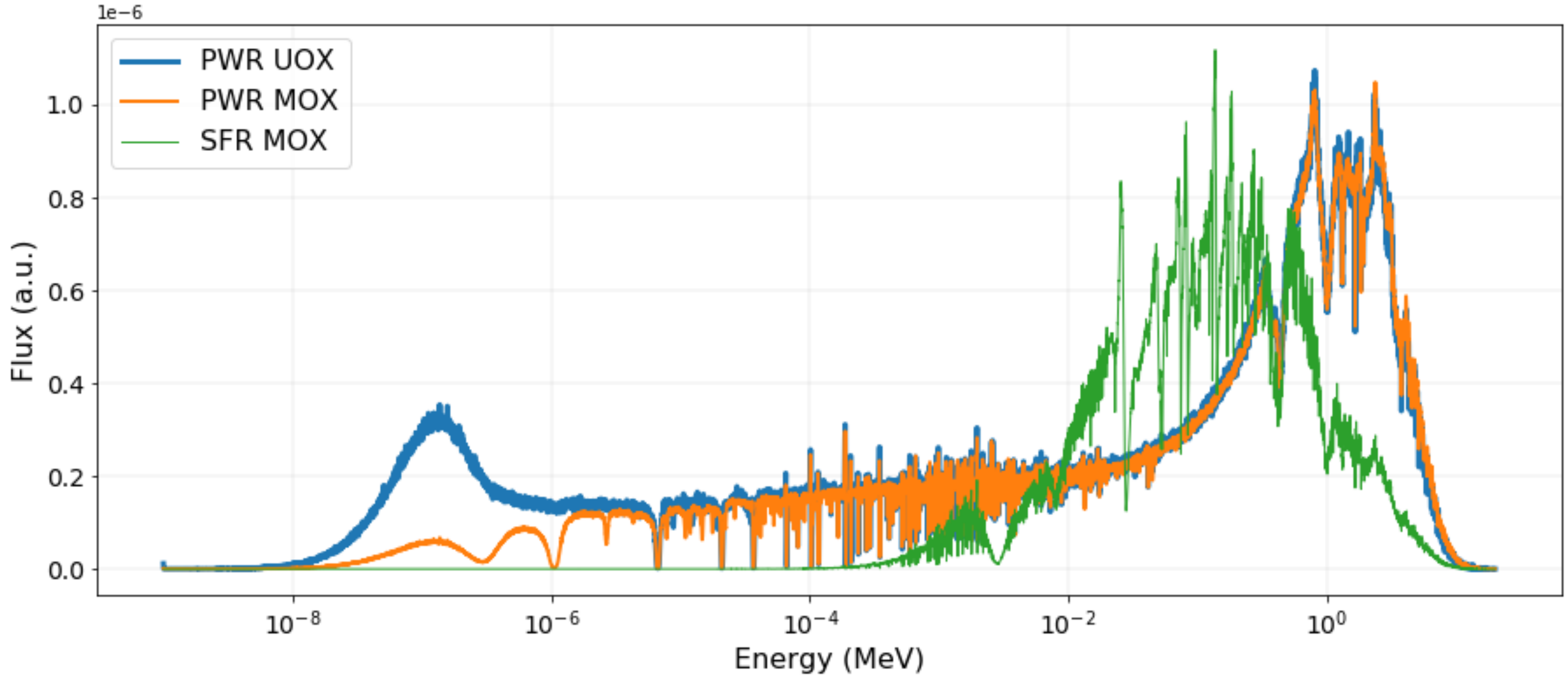
- This equation cannot be directly solved for realistic systems

Several possible spectra

- Total neutron flux

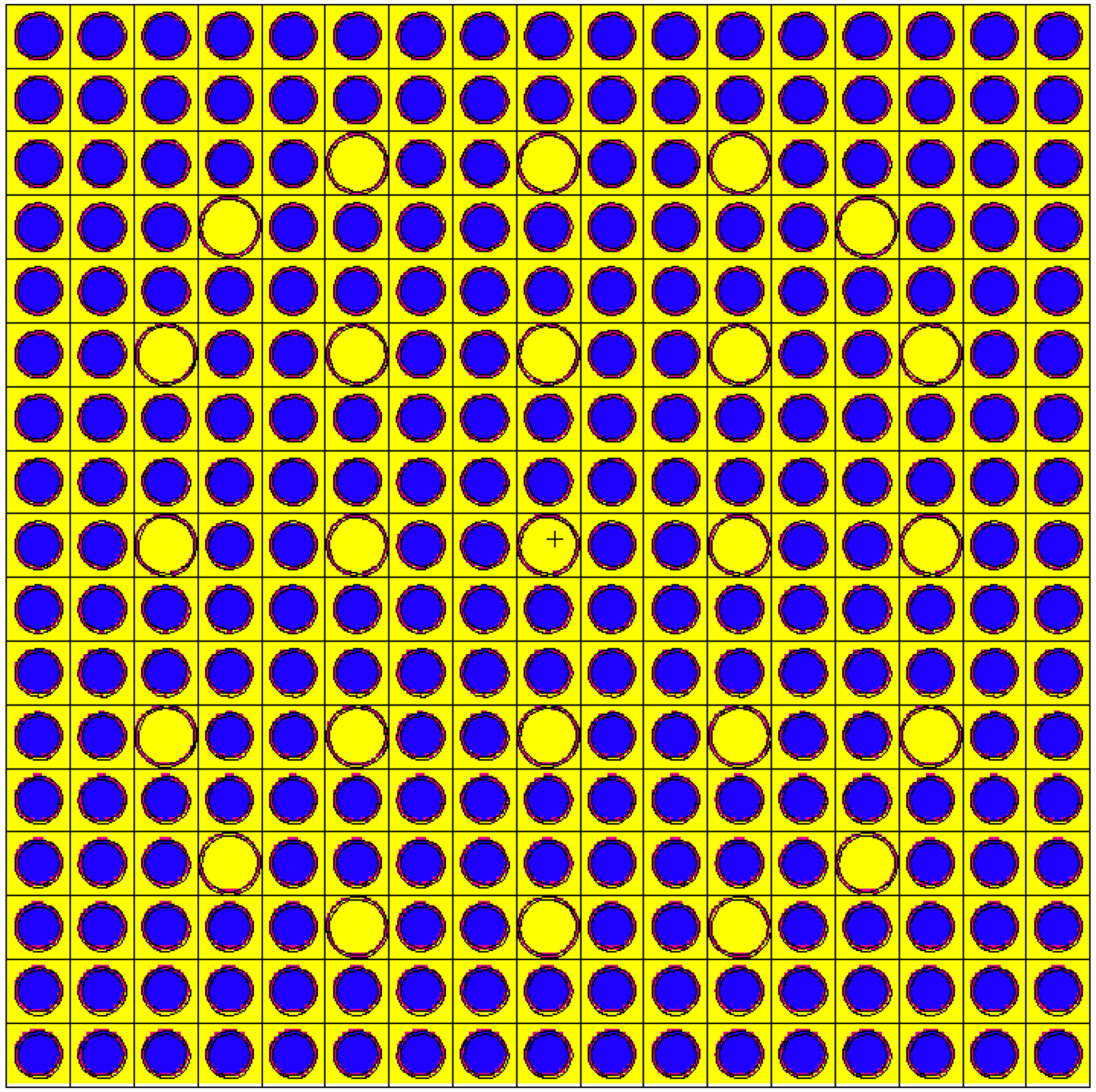
$$\Phi_V(t) = \int_E \varphi_V(E,t)dE$$

- Neutron spectrum
  - Fast spectrum
  - Epi-thermal spectrum
  - Thermal spectrum



Simple simulation

- Let's try to understand simply a neutron spectrum
  - MCNP simulation - PWR UOX infinite assembly

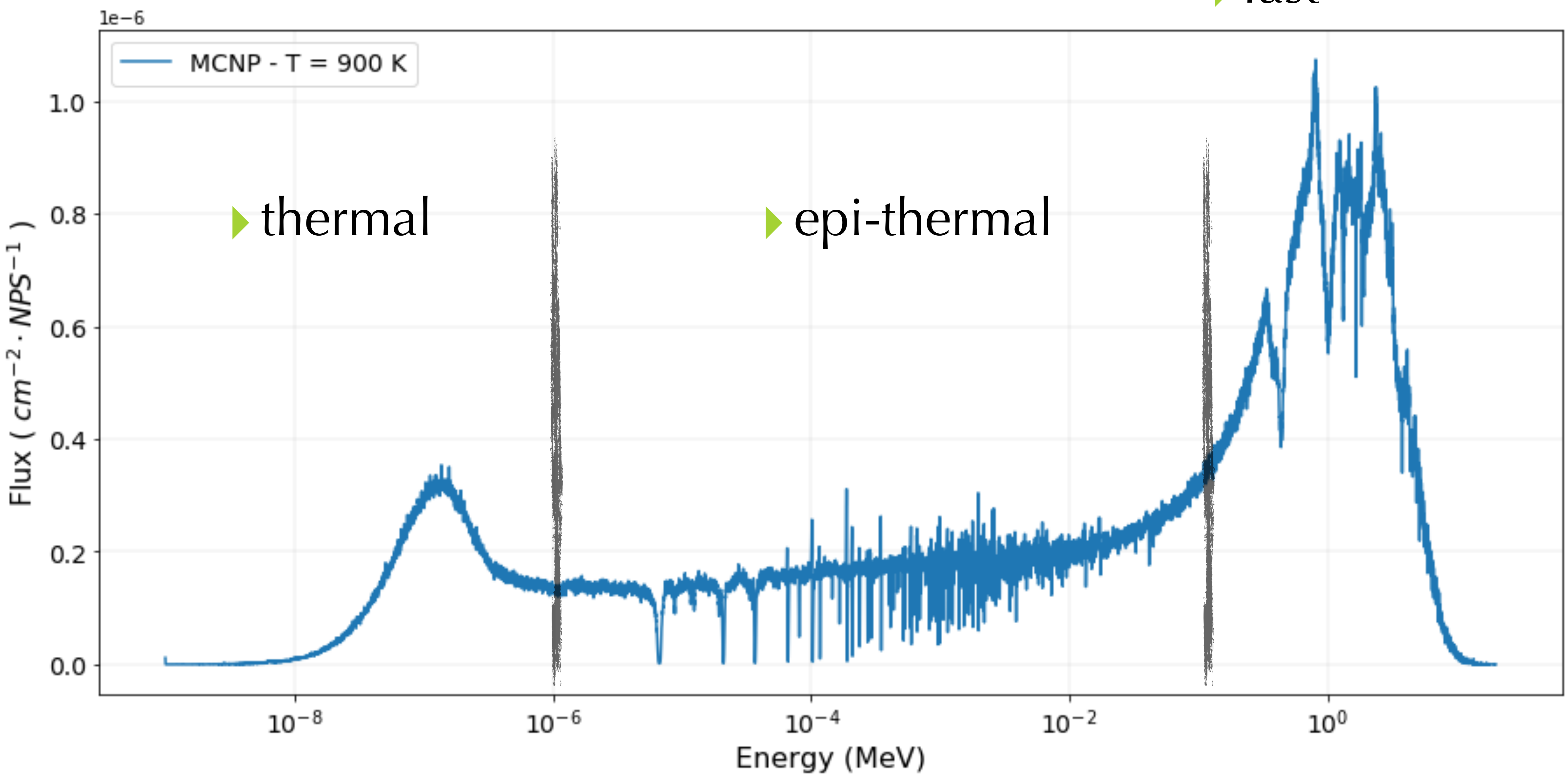


- Fuel @ 900K
  - 96% of ^{238}U
  - 4% of ^{235}U
- Coolant @ 600K
  - Water
- Structures @ 700K
  - Zircaloy

Spectrum decomposition

- Neutron Spectrum could be divided into three parts
  - Fast spectrum
  - Epi-thermal spectrum
  - Thermal spectrum

fast



The Watt distribution

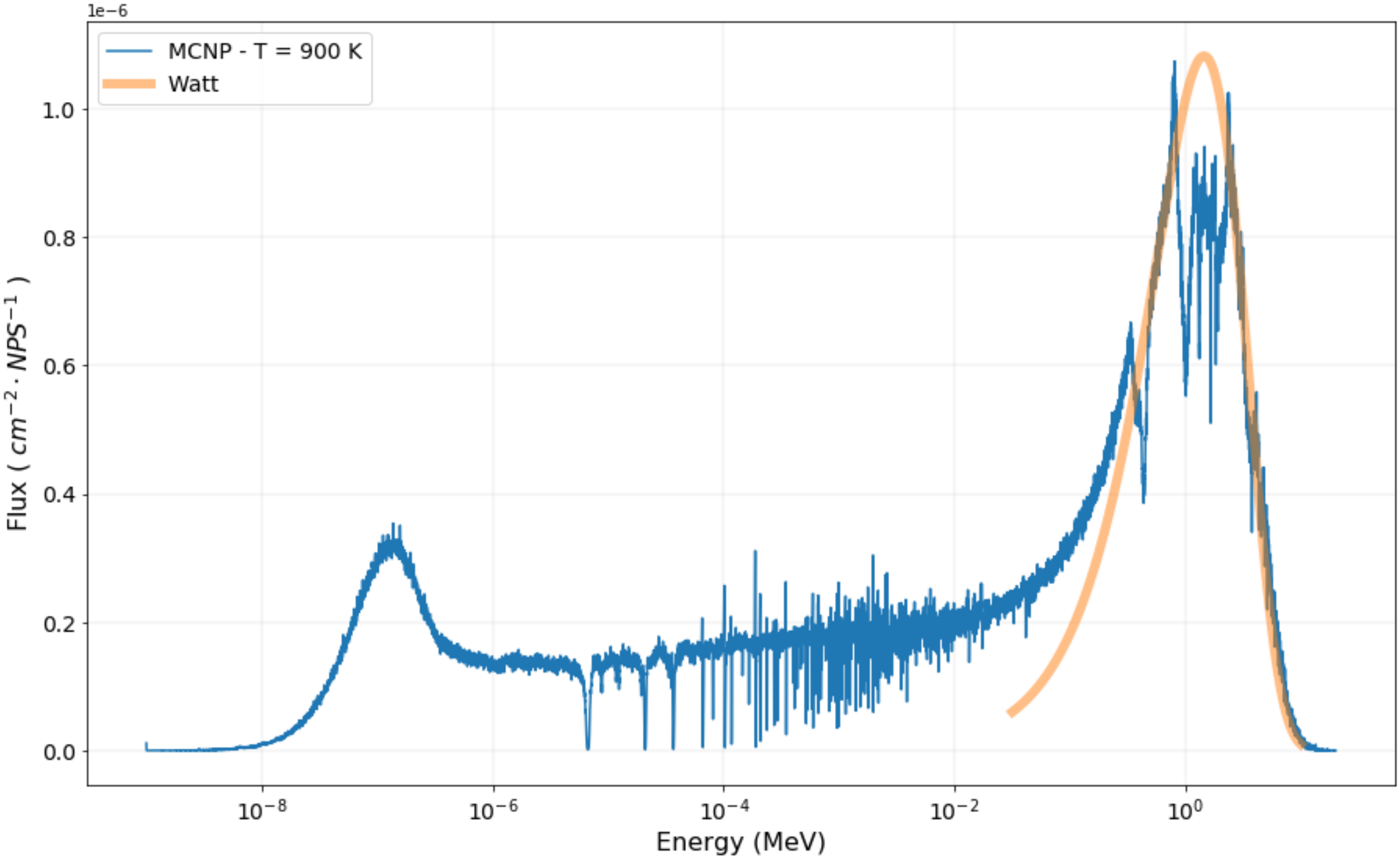
- In a PWR loaded with UOx, neutrons are generated by fissions
- Typical fission neutrons energy is around the MeV

- Usually, transport codes samples fission neutron energies from a Watt spectrum

$$f(E) = \frac{2\exp\left(-ab/4\right)}{\sqrt{\pi a^3 b}} \cdot \exp\left(-E/a\right) \cdot \sinh\left(\sqrt{bE}\right)$$

- The Watt spectrum is not coming from theory, it's just a common function that fits well the fission neutron distribution probability

Fast spectrum and Watt function



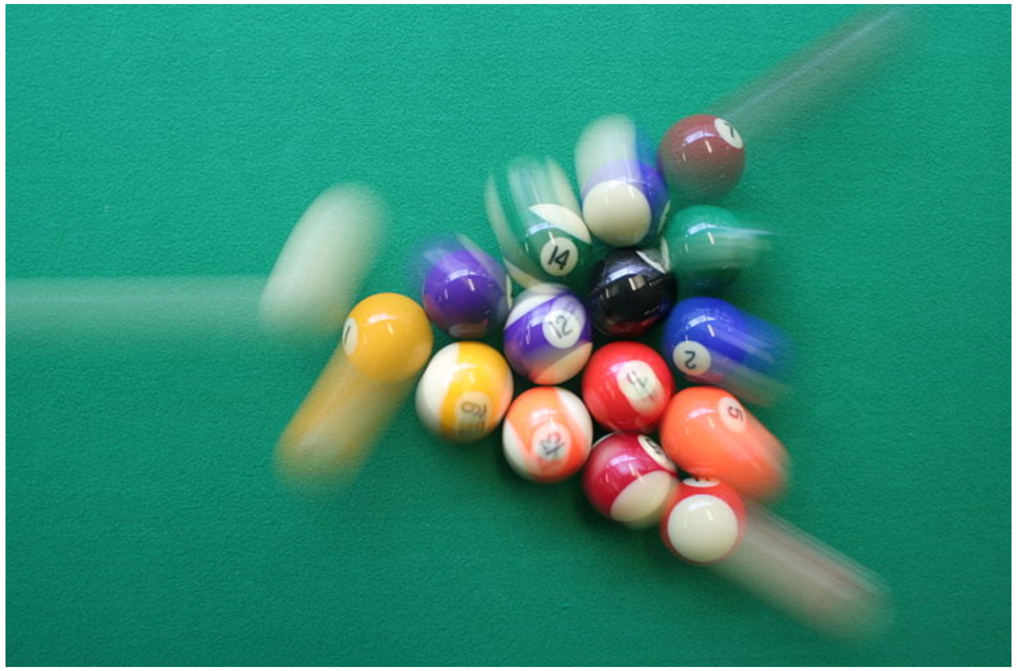
Slowing-down process

- Once fast fission neutrons are generated, here comes the slowing-down process mainly based on elastic scattering

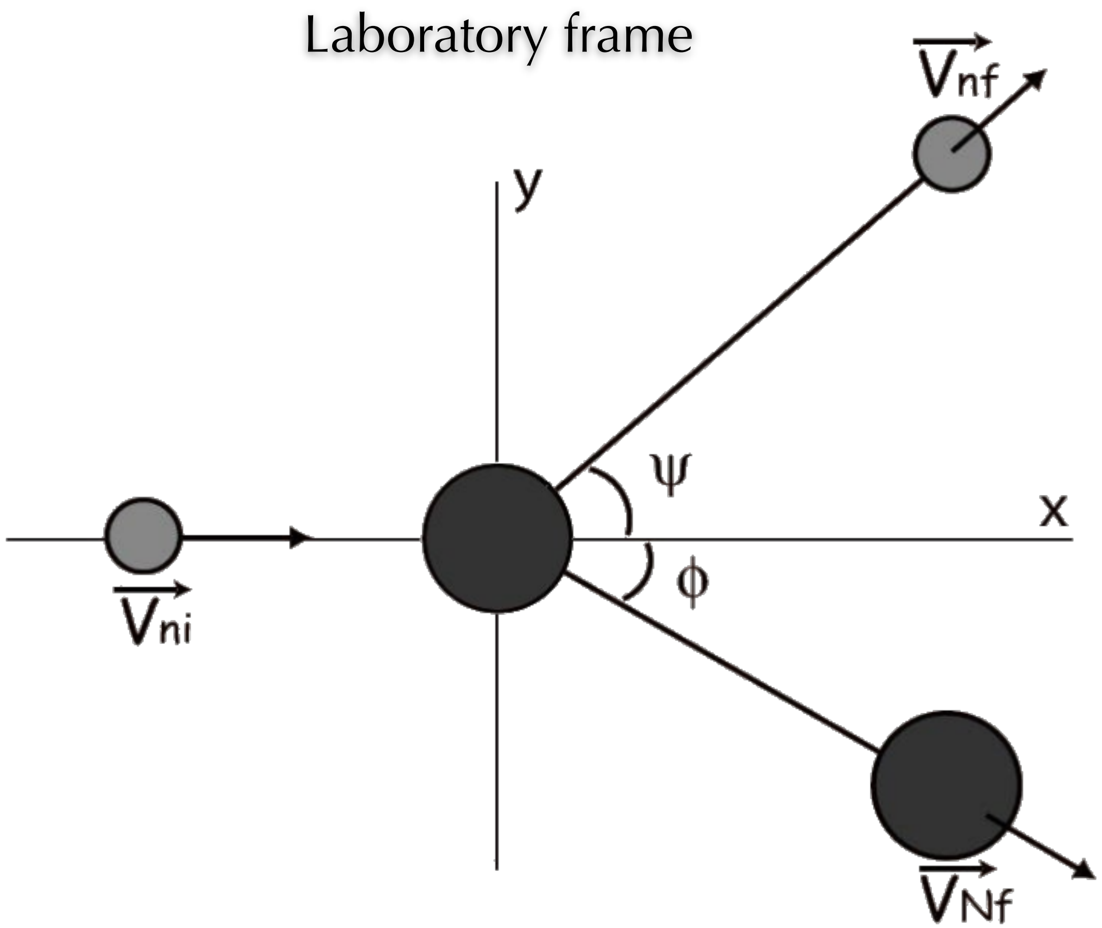
- Elastic scattering is considered if following conditions are respected:
  - Conservation of kinetic energy
  - Conservation of momentum
  - Conservation of the number of particles

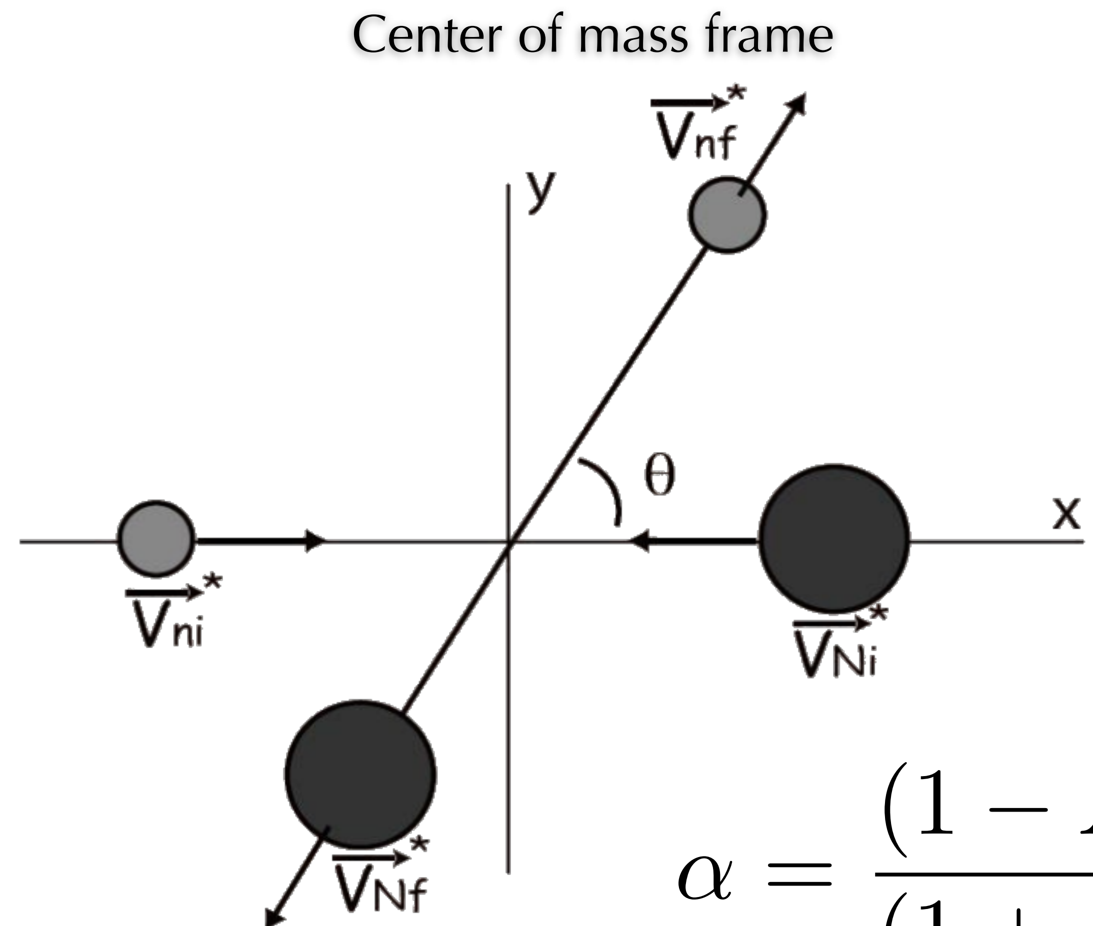
- Elastic scattering can be considered in a classical point of view in which the neutron is a perfect rigid billiard ball

- With E in [few eV - 20 MeV], energy of target is neglected



Elastic scattering kinematics



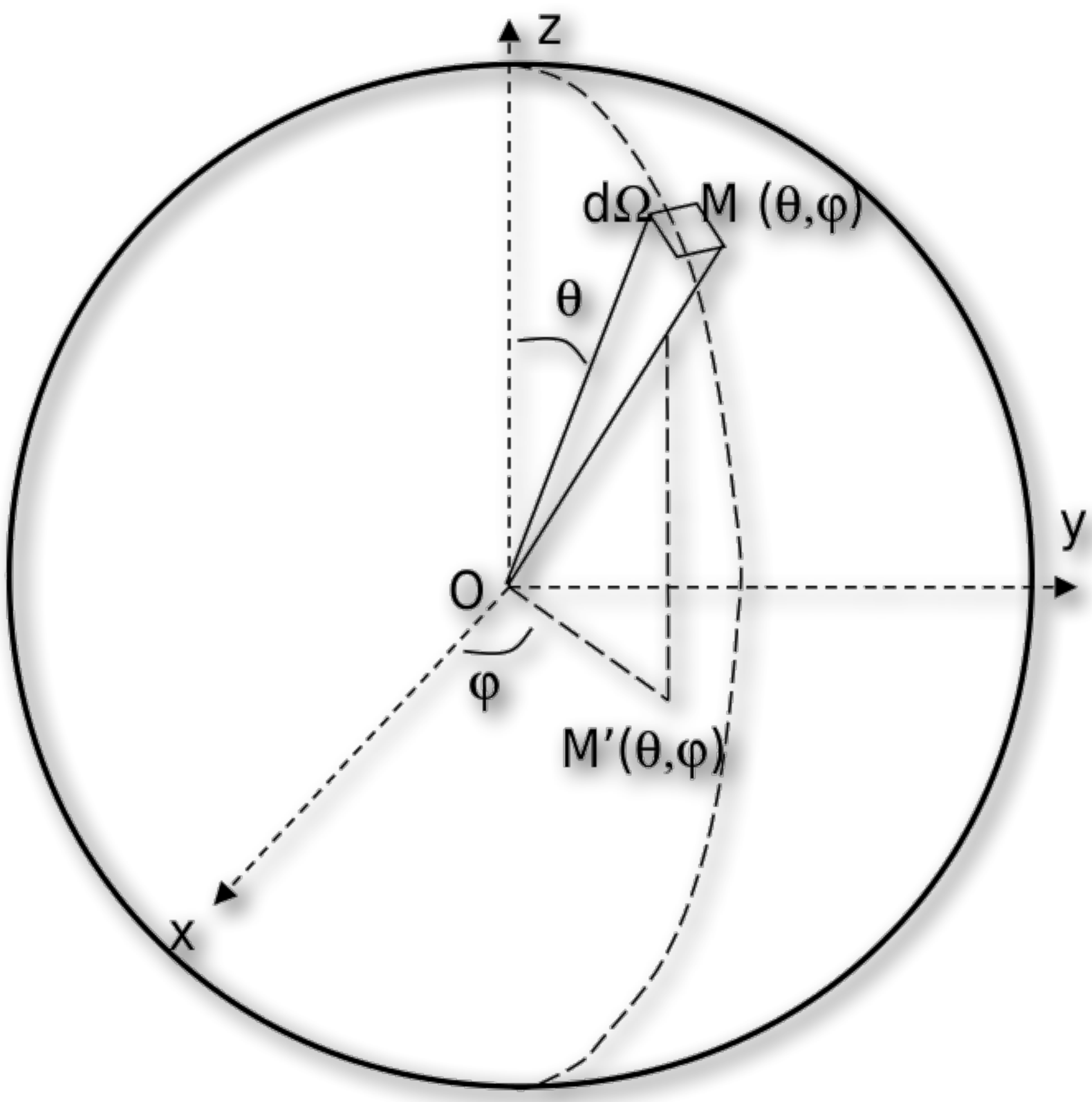


$$\alpha = \frac{(1-A)^2}{(1+A)^2}$$

- Laboratory to center of mass frame formula
- Conservation of kinetic energy and momentum in the center of mass frame

$$E_{nf} = \frac{E_{ni}}{2}\left((1+\alpha) + (1-\alpha)\cos\theta\right) \qquad \alpha E_{ni} \leq E_{nf} \leq E_{ni}$$

Isotropic collision hypothesis



- Radius of the sphere is 1
- A point M is located with:
  - θ angle / Oz - ($\theta \in [0, \pi]$)
  - φ is OM projection on the plan Oxy - ($\varphi \in [0, 2\pi]$)
- A solid angle at M direction is defined as : $d\Omega = \sin\theta \, d\theta \, d\varphi$

$$\boxed{\text{Isotropic collision} = P(\theta,\varphi)\, d\Omega \propto d\Omega}$$

$$\rightarrow P(\theta, \varphi)d\Omega = \frac{d\Omega}{4\pi}$$

- Oz is the particule incident direction

$$\rightarrow P(\theta)d\theta = \frac{\sin\theta}{2}d\theta$$

Mean energy after a collision

- From probability laws, we can calculate average values for energy, lethargy and angle θ or μ = cos(θ).

$$\langle x \rangle = \int_{x_{min}}^{x_{max}} x P(x) dx \quad \text{with} \quad \int_{x_{min}}^{x_{max}} P(x) dx = 1$$

$$\langle E_{\mathrm{nf}} \rangle = \frac{1+\alpha}{2} E_{\mathrm{ni}}$$

- Hydrogen $\langle E_{\mathrm{nf}} \rangle = 0.5\ E_{\mathrm{ni}}$ — nc ~ 20
- Carbon $\langle E_{\mathrm{nf}} \rangle = 0.86\ E_{\mathrm{ni}}$ — nc ~ 80
- Sodium $\langle E_{\mathrm{nf}} \rangle = 0.92\ E_{\mathrm{ni}}$ — nc ~ 120
- Lead $\langle E_{\mathrm{nf}} \rangle = 0.99\ E_{\mathrm{ni}}$ — nc ~ 1600

- The parameter α is very important in the slowing down capacity of a material with mass number A

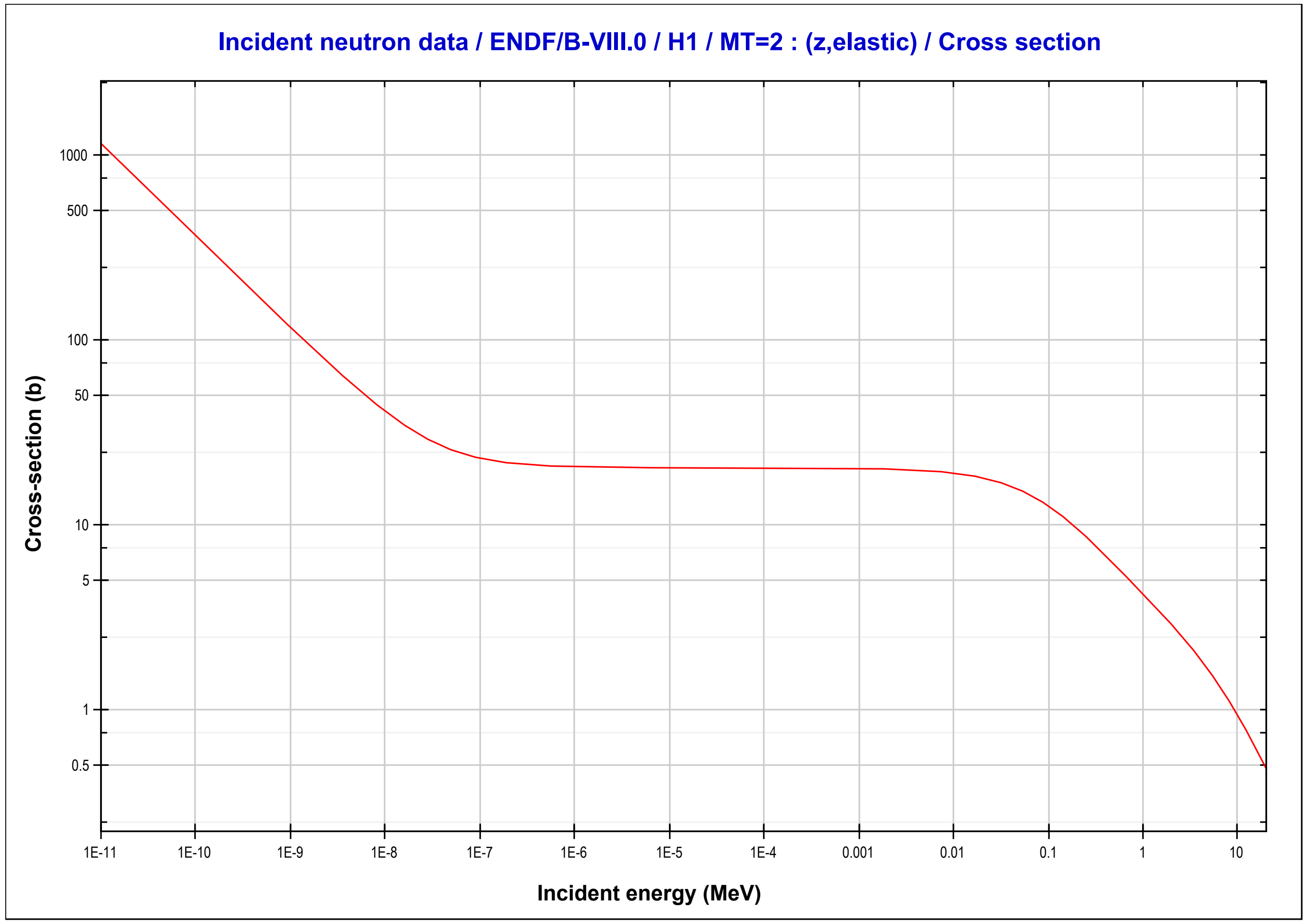
Flux in hydrogen

- The resolution of slowing down equation in following condition:
  - Medium thermal energy is negligible compared to neutron energy
  - Infinite isotropic and homogeneous medium of hydrogen
  - Isotropic elastic scattering in the center of mass
  - No absorption
- leads to following analytic solution for the neutron flux

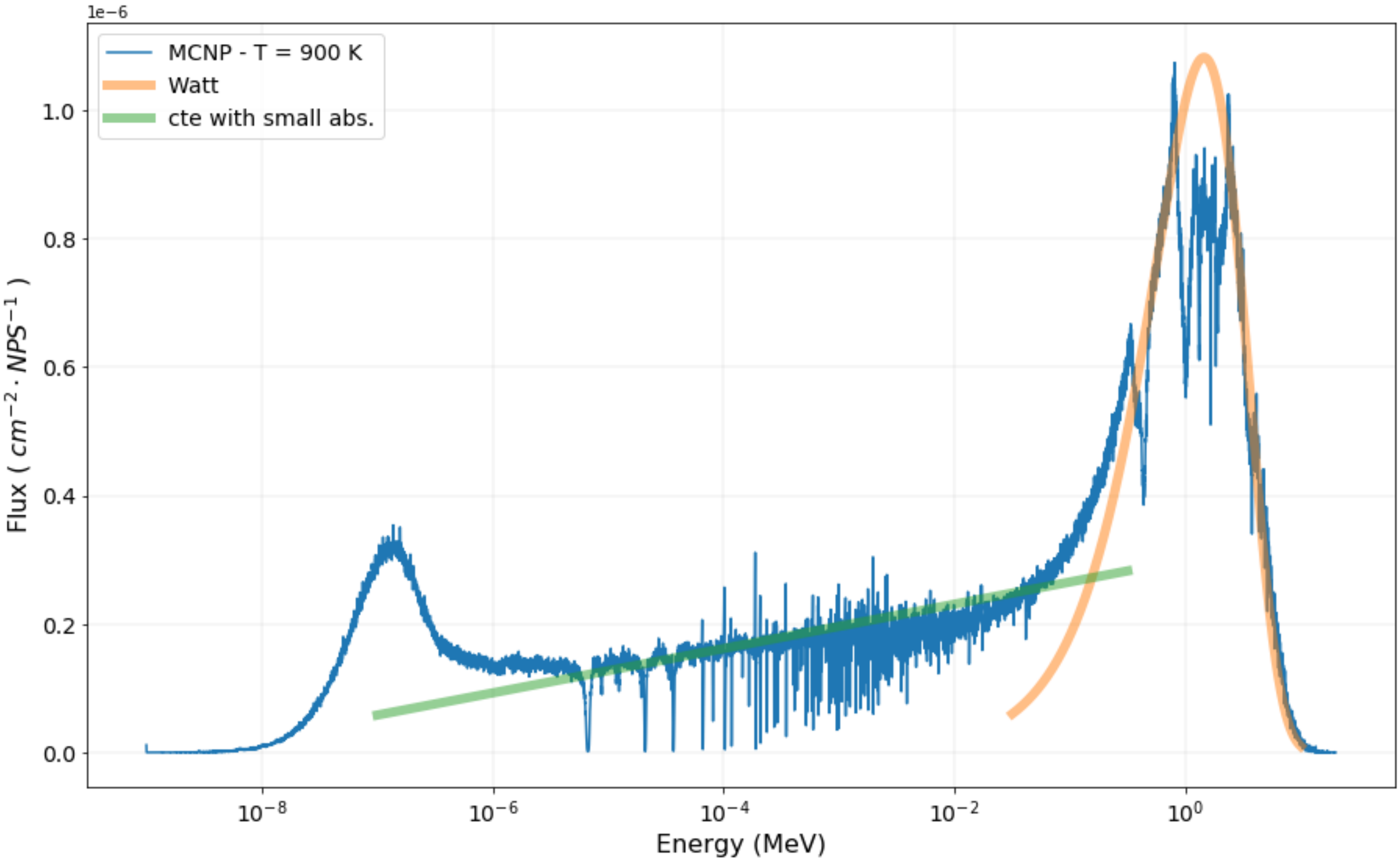
$$\varphi(E) = \frac{S}{E\Sigma_s(E)}$$

$$\text{with } U = \ln \frac{E_{\text{ref}}}{E}$$

$$\varphi(U) = \frac{S}{\Sigma_s(U)}$$



Epithermal model



Maxwell Boltzmann distribution

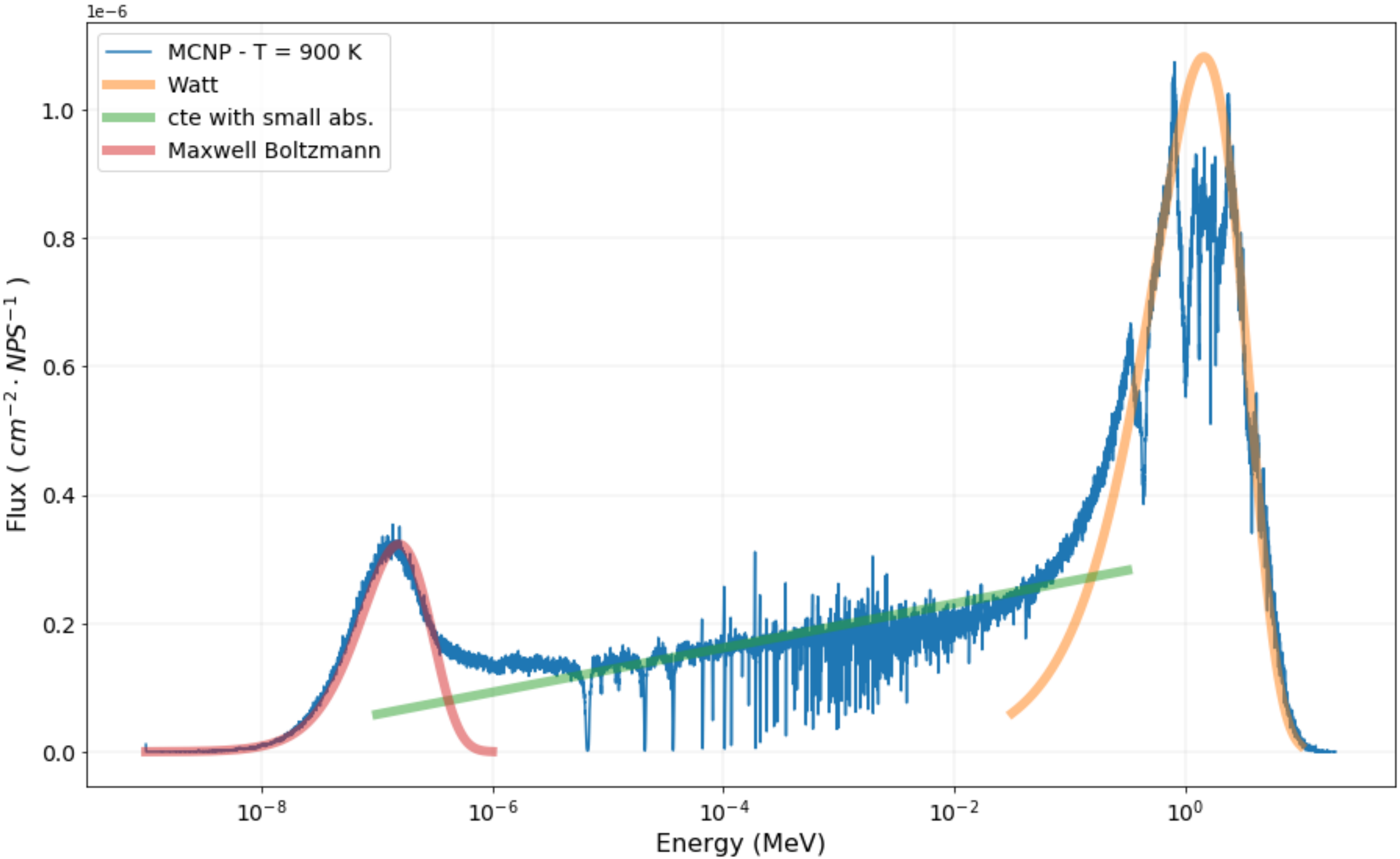
- Physics formalism depends on neutron energy compared to target energy
  - Target energy neglected : slowing down process
  - Target energy non negligible : thermalization process

- A neutron population that diffuses in a medium at the temperature T can be view as a neutron gaz at equilibrium (without absorption)

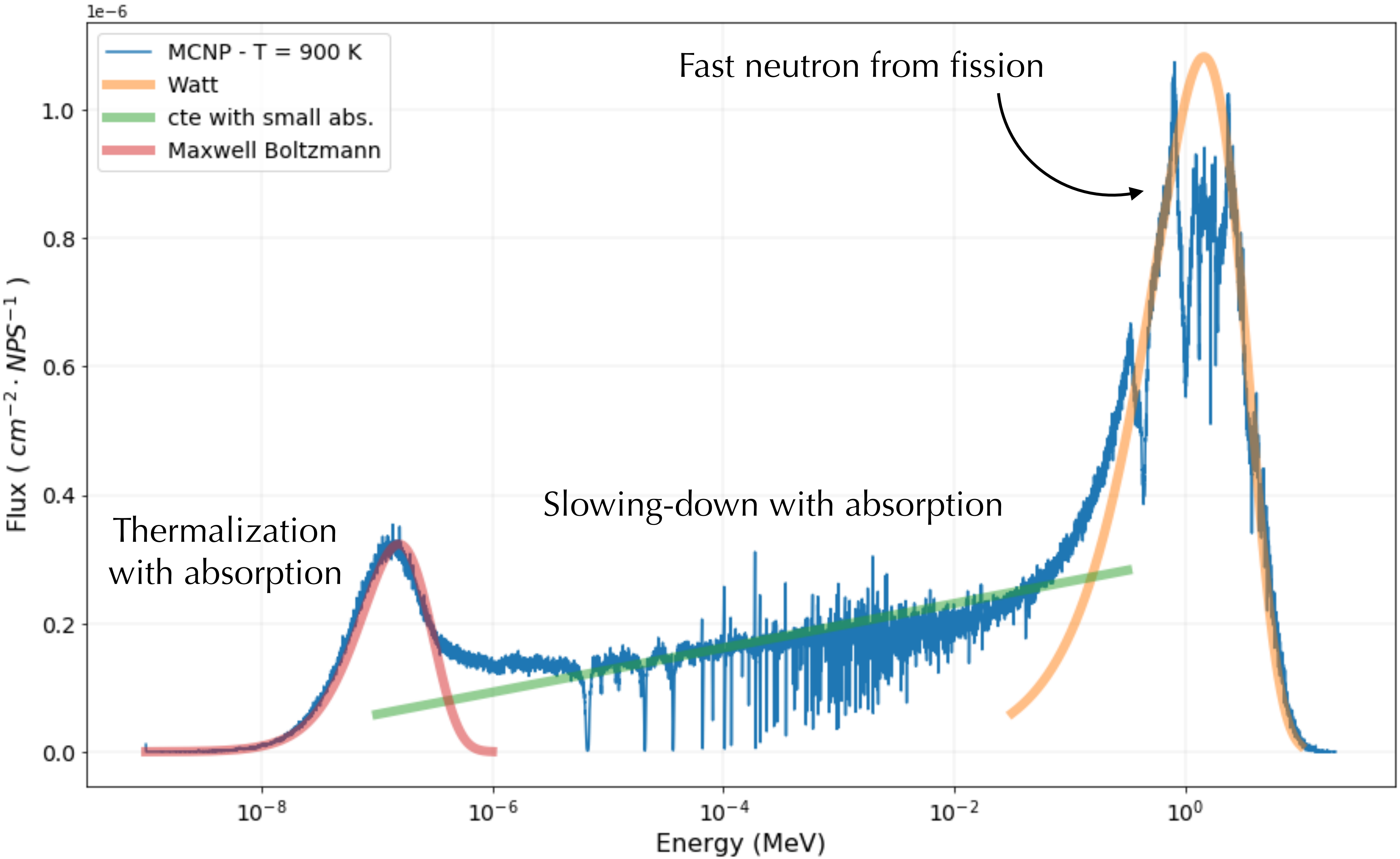
- In this case, energy distribution is constant and the probability to have a neutron between E and E+dE is given by the Maxwell-Boltzmann distribution :

$$P(E)dE = \frac{2}{\sqrt{\pi}}\sqrt{\frac{E}{kT}} \cdot e^{-\frac{E}{kT}} \frac{dE}{kT}$$

Maxwell Boltzmann distribution



Maxwell Boltzmann distribution



Transport codes

- The analytic resolution of Boltzmann equation is usually not possible
- Numerical resolution from neutron transport simulation are widely used

Monte Carlo

- Based on random number samplings
- Neutrons « real » propagation simulation
- Uncertainty decreases with cpu time
- Little required simplifications
- Used as reference calculations
- MCNP, SERPENT, Tripoli, etc.

Deterministic

- Based on Boltzmann equation resolution
- Spatial simplifications
- Energy are treated as multi-group
- Resonance auto-protection
- Faster compared to Monte Carlo
- Dragon, Eranos, Apollo, etc.

- Evaluation and experimental nuclear data are fundamentals !
  - Evaluated nuclear data file are used by Monte-Carlo and deterministic codes
  - Several database are used from many countries in the world
  - ENDF, JEFF, JENDL, etc.

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Subatech CNRS IN2P3 Les deux infinis IMT Atlantique Bretagne-Pays de la Loire École Mines-Télécom UNIVERSITÉ DE NANTES

The burn-up

- Irradiation could be represented as a function of time or burn-up
- Burn-Up is the released energy per initial heavy mass [GWd/tHM]
- Usual burn-up is around 40 GWd/tHM (100 GWd/tHM) for a PWR (SFR)

- Burn-up definition $$BU(t) = \frac{\Gamma_f \cdot t \cdot E_f}{M_{HN}}$$

- Burn-up as a function of fissioned mass $$BU(t) = \frac{N_a \cdot E_f}{A} \cdot \frac{M_f(t)}{M_{HN}}$$

- Burn-up after numerical application $$BU(t) = 9.3 \cdot 100 \frac{M_f(t)}{M_{HN}}$$

- Fraction in per-cent of the irradiated mass

Main form

- Bateman equations are the differential equations that describe the evolution under irradiation in a reactor core.
- It contains for each nuclide loss and creation by decay and nuclear reaction

$$\boxed{\frac{dN_i}{dt}} = \boxed{-(\lambda_i + \sigma_i \phi) N_i} + \boxed{\sum_{j \neq i} (\lambda_{j \to i} + \sigma_{j \to i} \phi) N_j}$$

- Variation rate of the nuclide i

- Loss rate by decay
- Loss rate by nuclear reaction

- Creation rate by decay
- Creation rate by nuclear reaction

Matrix form

$$\frac{d\vec{N}}{dt} = \begin{pmatrix} -(\lambda_0 + \sigma_0 \phi) & & & & \\ & \ddots & & (\lambda_{j \to k} + \sigma_{j \to k} \phi) & \\ & & -(\lambda_i + \sigma_i \phi) & & \\ & (\lambda_{l \to m} + \sigma_{l \to m} \phi) & & \ddots & \\ & & & & -(\lambda_N + \sigma_N \phi) \end{pmatrix} \vec{N}$$

Mean cross section

- Reaction rate defined in Bateman equations:

$$N\sigma\phi = \int N\sigma(E)\varphi(E)dE$$

Normalized flux in E and E+dE
=> [cm^{-2} · s^{-1} · MeV^{-1}]

- Mean cross section simplifications

$$N\sigma\phi = N\langle\sigma\rangle\phi$$

with

$$\langle\sigma\rangle = \frac{\int \sigma(E)\varphi(E)dE}{\int \varphi(E)dE}$$

$$\phi = \int \varphi(E)dE$$

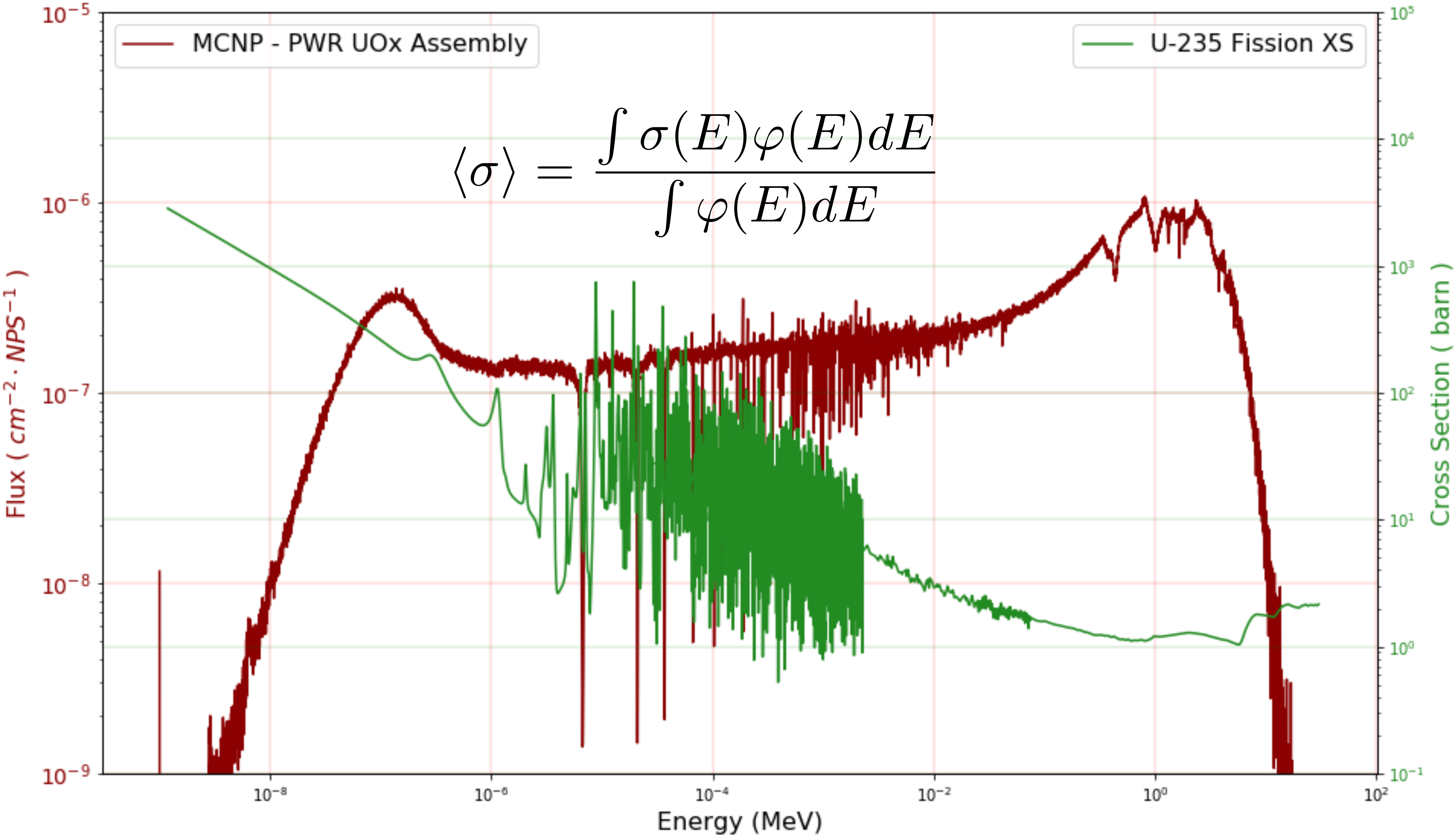
Total flux in the system
=> [cm^{-2} · s^{-1}]

- A mean cross section aggregates the cross section in a unique number

- The total flux is then used to normalize

Mean cross section

$$\langle\sigma\rangle = \frac{\int \sigma(E)\varphi(E)dE}{\int \varphi(E)dE}$$



Reaction rate vs mean XS

- In practice, it is possible to calculate

Reaction rates

- Reaction rates are calculated at each steps of the transport code calculation as the multiplication of the flux and cross sections.

- Number of reaction rate to calculate
  - ➡ Number of isotopes x Number of reactions

Mean cross sections

- Mean cross sections are calculated at the end of the transport calculation

- One observable to asses
  - ➡ Neutron flux

$$\varphi(E)$$

- A very high precision is required
  - ➡ More than 10 000 energy groups

- For a complex calculation inducing a high number of isotopes, mean cross sections based evolution decrease the CPU time by a factor ~30 (SMURE team)

Normalization

- Neutron spectrum is included inside mean cross sections
- Bateman equation depends on neutron total flux

$$\frac{dN_i}{dt} = -(\lambda_i + \langle\sigma_i\rangle\phi)N_i + \sum_{j\neq i}(\lambda_{j\to i} + \langle\sigma_{j\to i}\rangle\phi)N_j$$

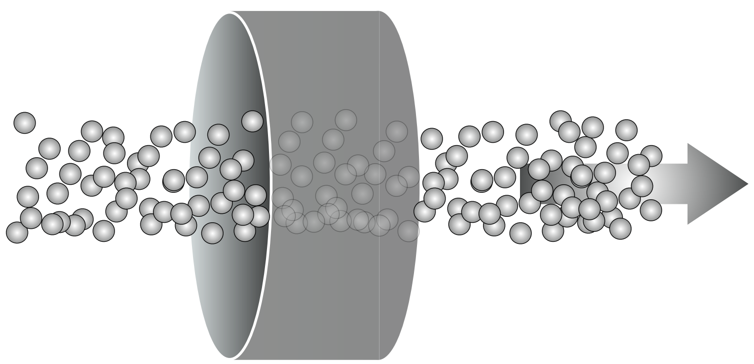
- Neutron flux in a reactor is not precisely known and depends on:
  - Thermal power of the reactor core
  - Burn-up of the fuel
  - Control rod position
  - Neutronic poisons amount (boron, Gd, etc.)
  - etc.

- In practice, user calculates/imposes the neutron flux according to the power

Normalization

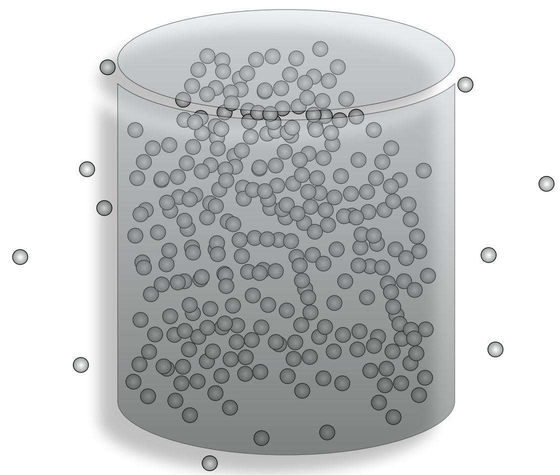
- A simple solution could be to consider a constant flux during irradiation

Target irradiation by neutrons



- Possible experimental flux : 10^6 cm^{-2} s^{-1}
- Number of atoms in the target : 10^{24}
- Reaction rate : 10^6 absorption s^{-1}
- Irradiation time : 10^5 s
  - ➡ Composition modification is small
  - ➡ Spectrum modification is small
  - ➡ Neutron Flux is stationary

Nuclear power plant core irradiation

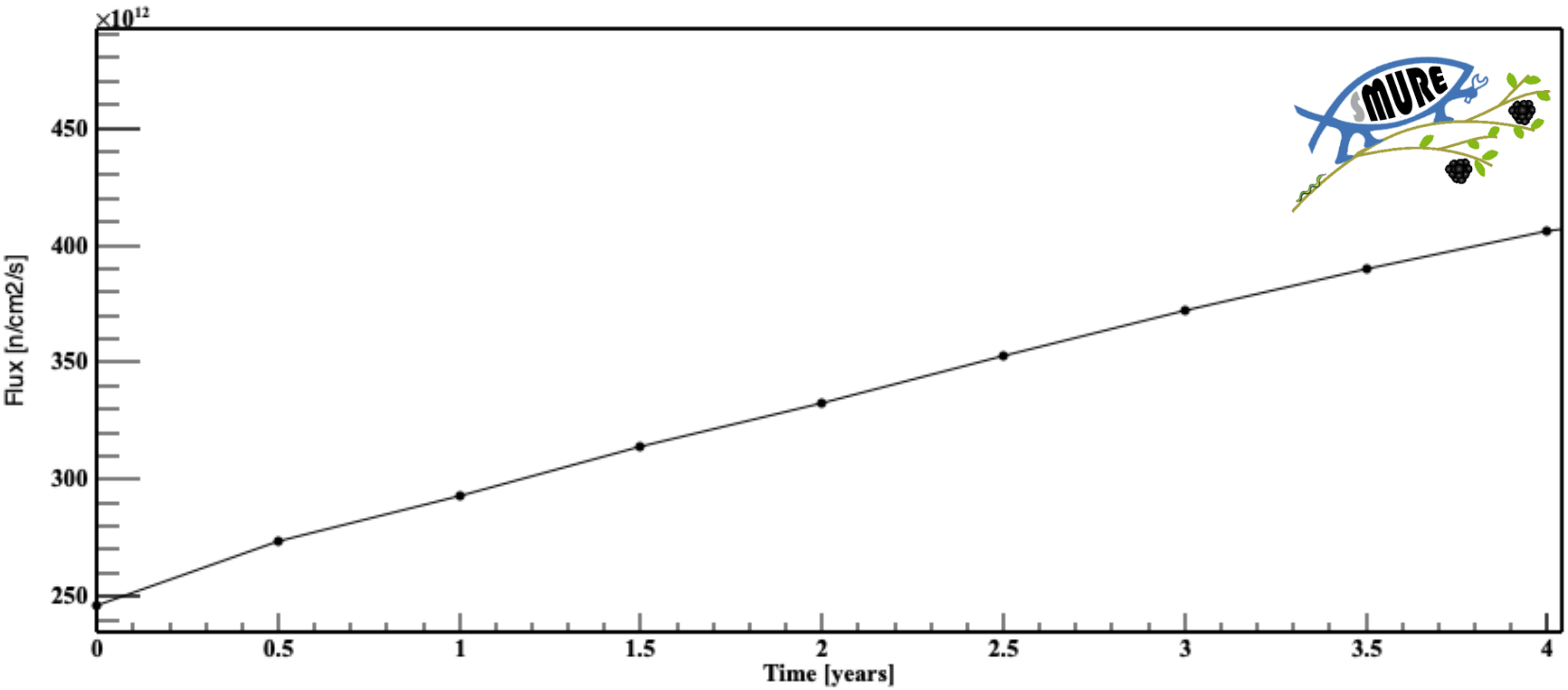


- Possible experimental flux : 10^{14} cm^{-2} s^{-1}
- Number of atoms in the target : 10^{29}
- Reaction rate : 10^{20} absorption s^{-1}
- Irradiation time : 10^8 s
  - ➡ Composition modification is significant
  - ➡ Spectrum modification is significant
  - ➡ Neutron Flux is not constant

- Flux normalization is possible for experiments with low neutron flux
- For a reactor evolution, the flux can't be considered as constant

Neutron flux evolution

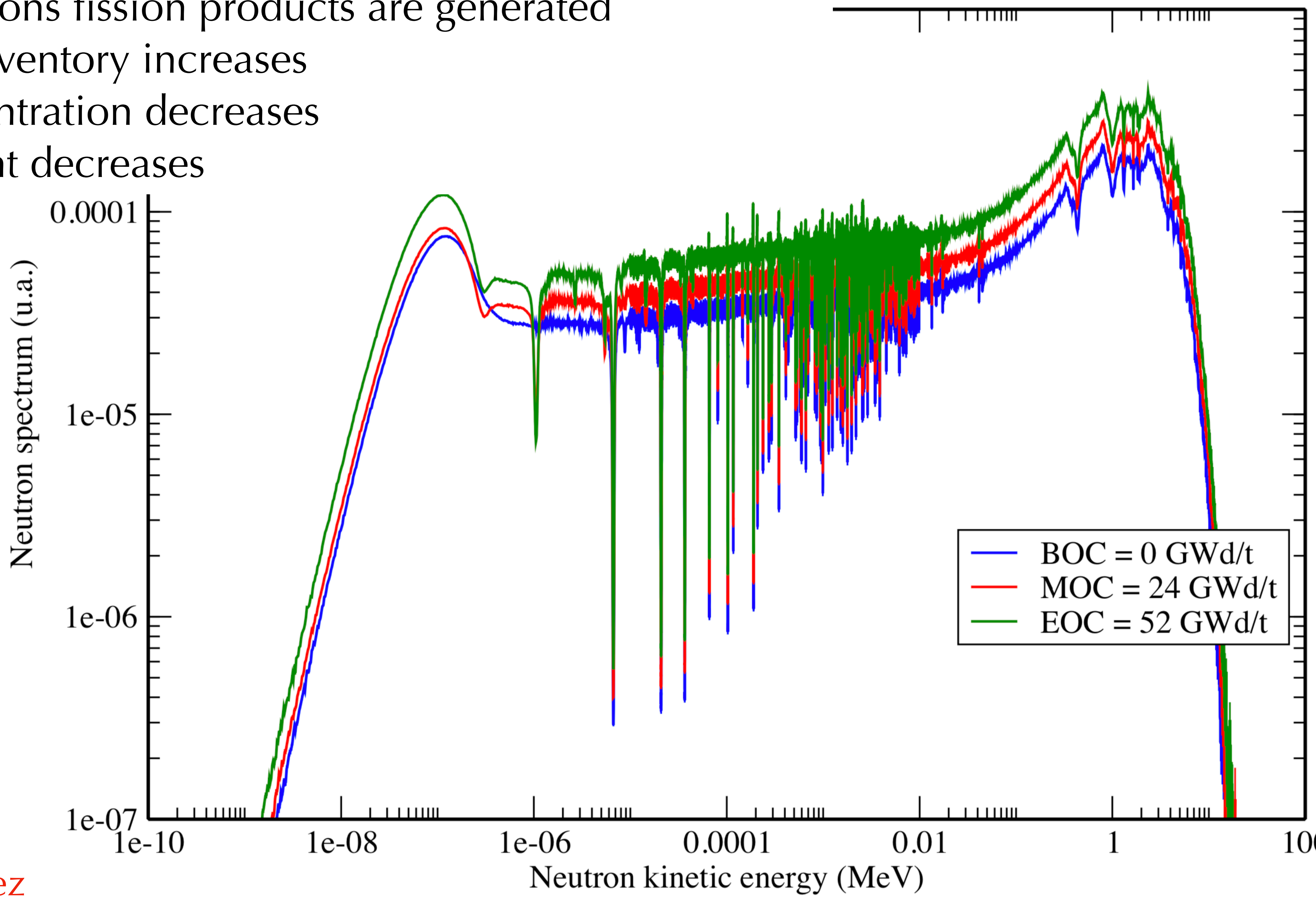
- Infinite assembly PWR simulation with SMURE - UOx (3.7% ^{235}U)
- Constant thermal power during irradiation



- During irradiation, some neutron absorbent are generated
- Neutron flux increase in order to have a constant thermal power

Spectra evolution

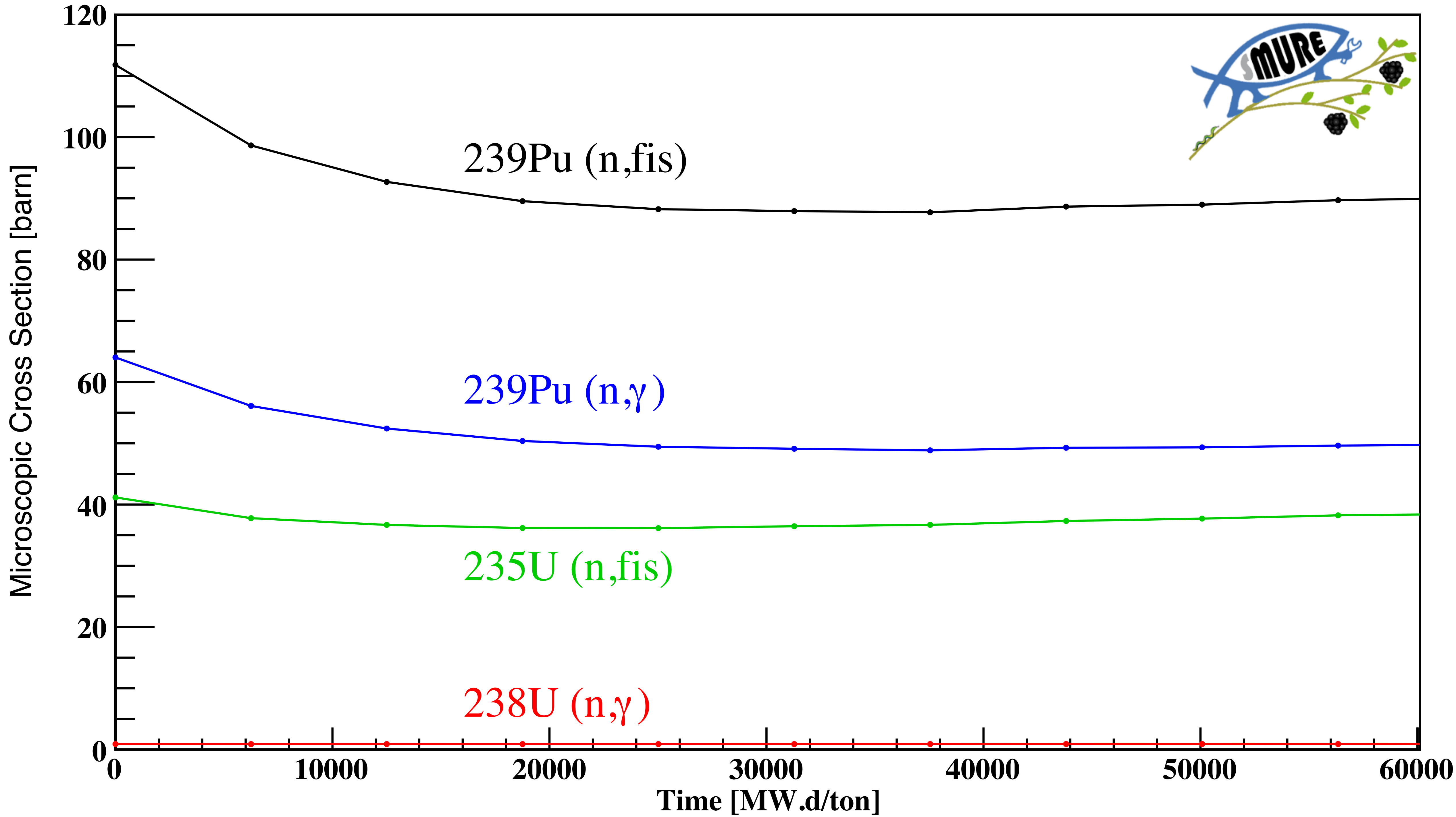
- During the burnup, the fuel is significantly impacted
  - Neutron poisons fission products are generated
  - Plutonium inventory increases
  - Boron concentration decreases
  - Fissile content decreases



In courtesy of X. Doligez

Mean XS evolution

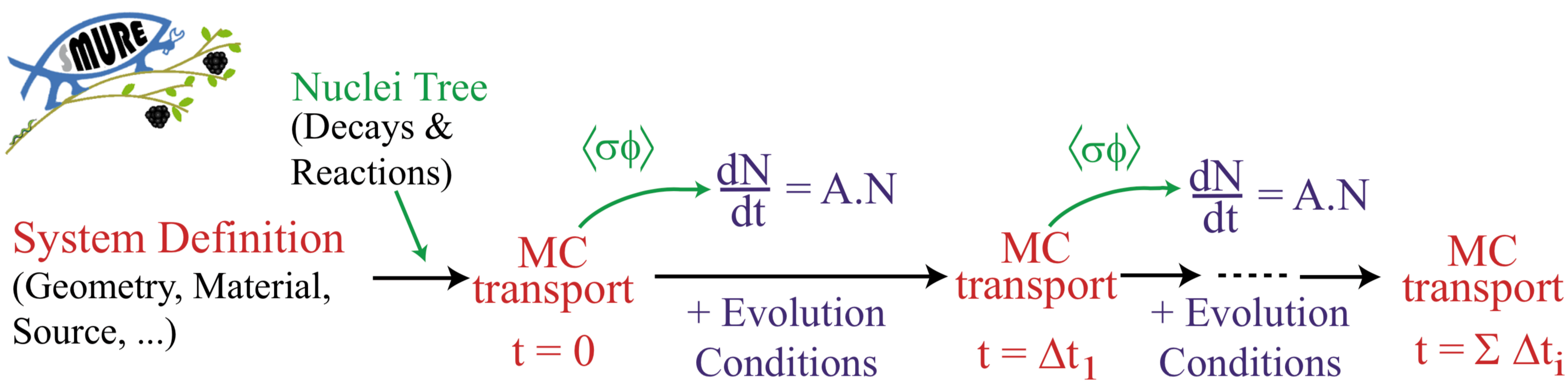
- Mean cross sections are strongly impacted during the burnup



In practise

- During neutron irradiation, a lot of parameters are evolving
  - Neutron spectrum and neutron total flux
  - Mean cross sections

- Neutronic data need to be updated a lot of time to avoid biases
  - Coupled transport calculation / evolution calculation
  - Compromise between calculation time and required precision



Energy released by fission

- Thermal power and neutron flux relation is given by:

$$P_{th} = \sum_i N_i \int \sigma_i^{fis}(E)\phi(E)\epsilon_i(E)dE$$

  - N_i is the number of fissile nucleus
  - σ_i is the cross section of nucleus i
  - Φ is the neutron flux
  - ε_i is energy released by the nucleus i

Energy released by a fission on ^{235}U by a thermal neutron

Prompt	Energy (MeV)	Delayed	Energy (MeV)
Fission Products	169	Beta -	6.4
Neutrons	4.8	Neutrons	0.010
Gammas	7.0	Gammas	6.2
		Neutrinos	10.0
Total	180.8	Total	22.610
TOTAL		203.410	

Resolution

- Simple method but not really used
  - Not precise
  - Not stable

System to be solved:

$$\frac{dy}{dt} = f(y,t) \qquad y(0) = y_0$$

simple example

$$\frac{dN_x}{dt} = -\lambda_x N_x(t)$$
$$N_x(0) = N_0$$

$$y = N_x$$
$$f(y,t) = -\lambda_x N_x(t)$$

Time discretization and first order development:

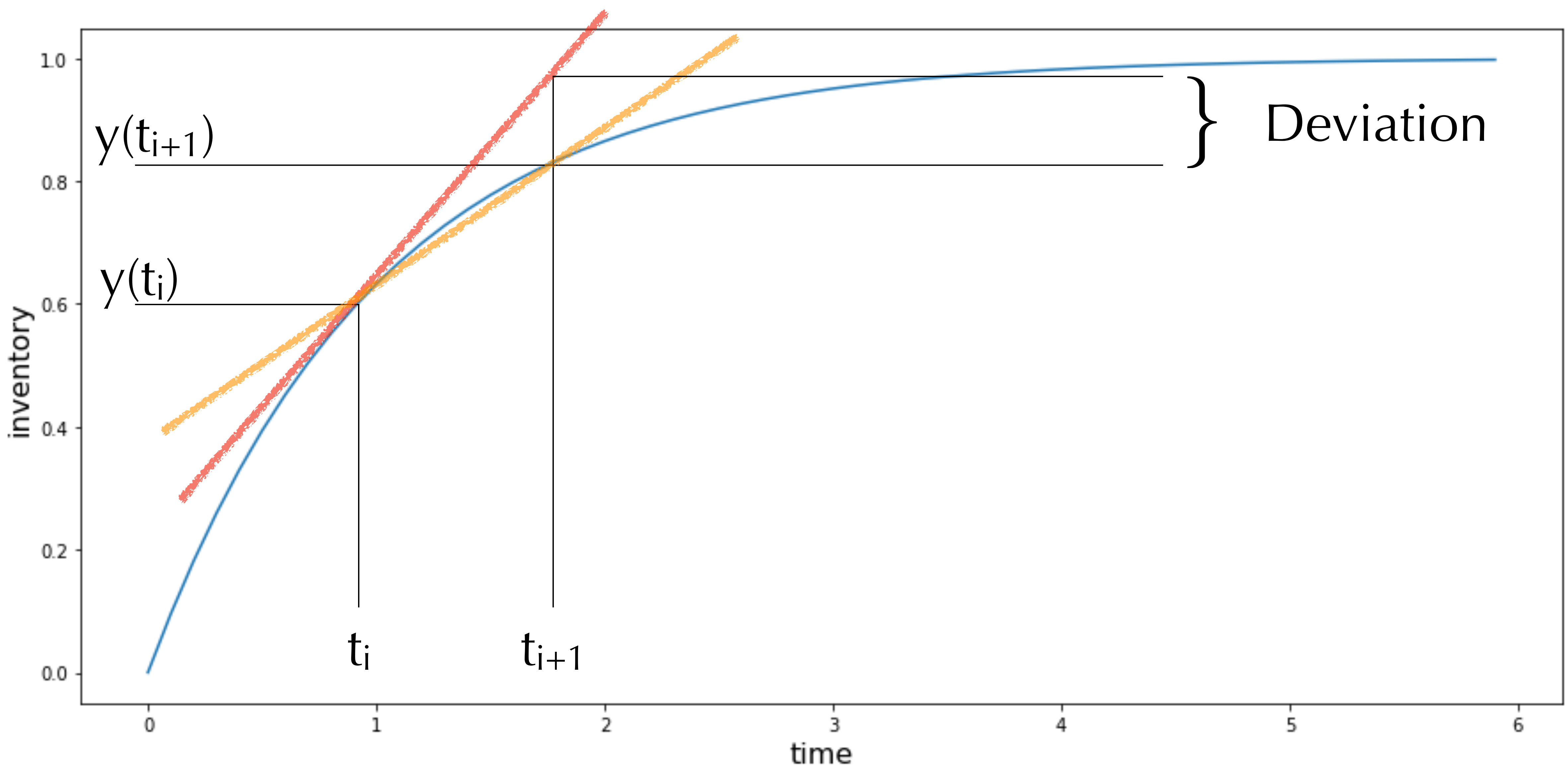
$$y(t_i + \Delta t) = y(t_i) + \Delta t \left(\frac{dy}{dt}\right)_{t_i} + O(\Delta t^2)$$

$$\longrightarrow \quad y_{i+1} = y_i + \Delta t f(y_i, t_i) + O(\Delta t^2)$$

Resolution

$$y(t_i + \Delta t) = y(t_i) + \Delta t \left(\frac{dy}{dt}\right)_{t_i} + O(\Delta t^2)$$

$$\longrightarrow y_{i+1} = y_i + \Delta t f(y_i, t_i) + O(\Delta t^2)$$



Runge-Kutta

- Runge-Kutta methods are widely used to solve evolution equations
- Advantages:
  - Easy to program and to use
  - Usually stable
  - Easy to modify the time binning
  - Initial conditions allows integration
- Drawbacks:
  - High calculation time

Runge-Kutta 2

$$r_{1i} = f(y_i, t_i)$$

$$r_{2i} = f(y_i + \frac{\Delta t}{2} r_{1i}, t_i + \frac{\Delta t}{2})$$

$$\longrightarrow y_{i+1} = y_i + \Delta t r_2$$

Runge-Kutta 4

$$r_{1i} = f(y_i, t_i)$$

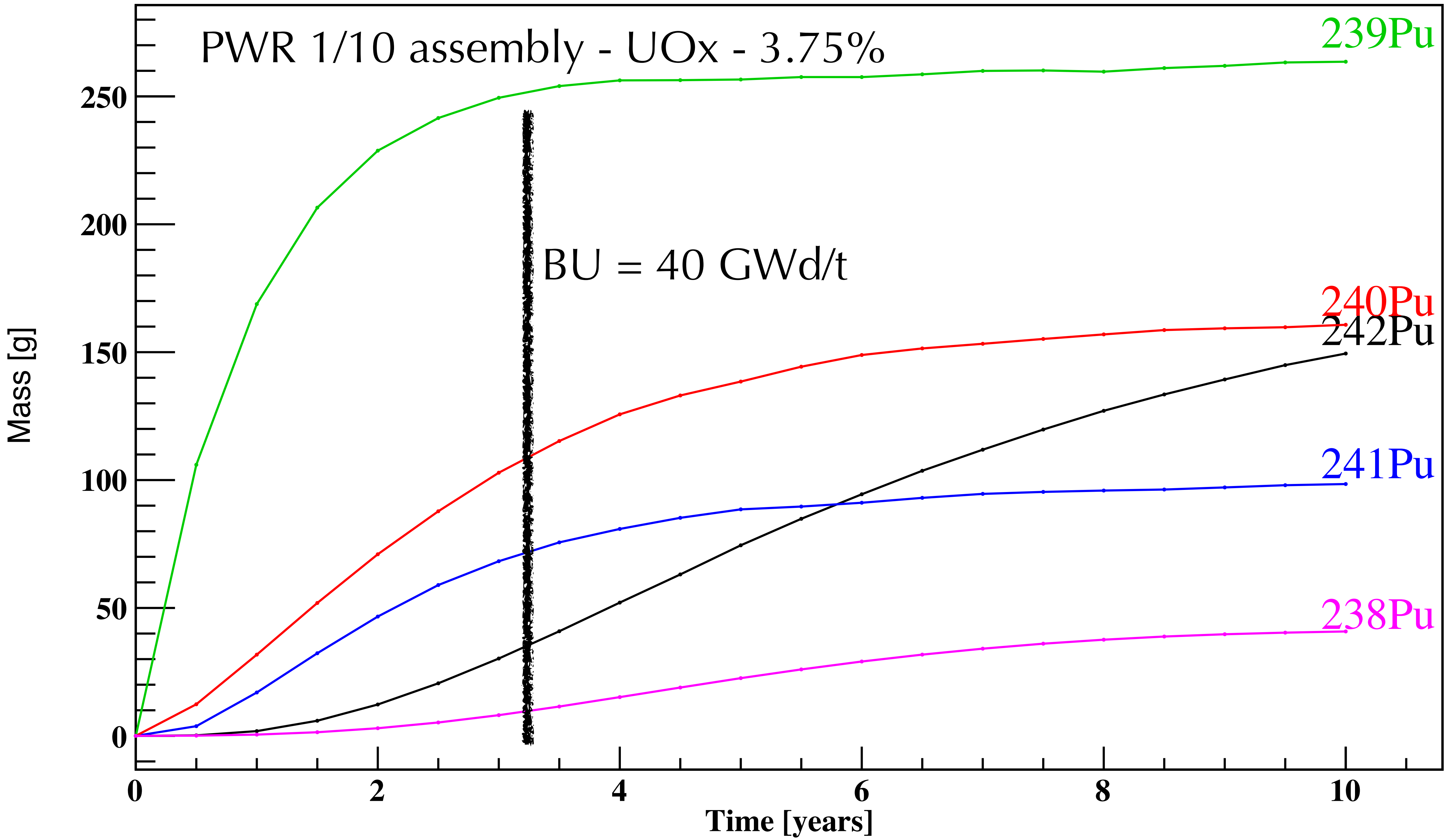
$$r_{2i} = f(y_i + \frac{\Delta t}{2} r_{1i}, t_i + \frac{\Delta t}{2})$$

$$r_{3i} = f(y_i + \frac{\Delta t}{2} r_{2i}, t_i + \frac{\Delta t}{2})$$

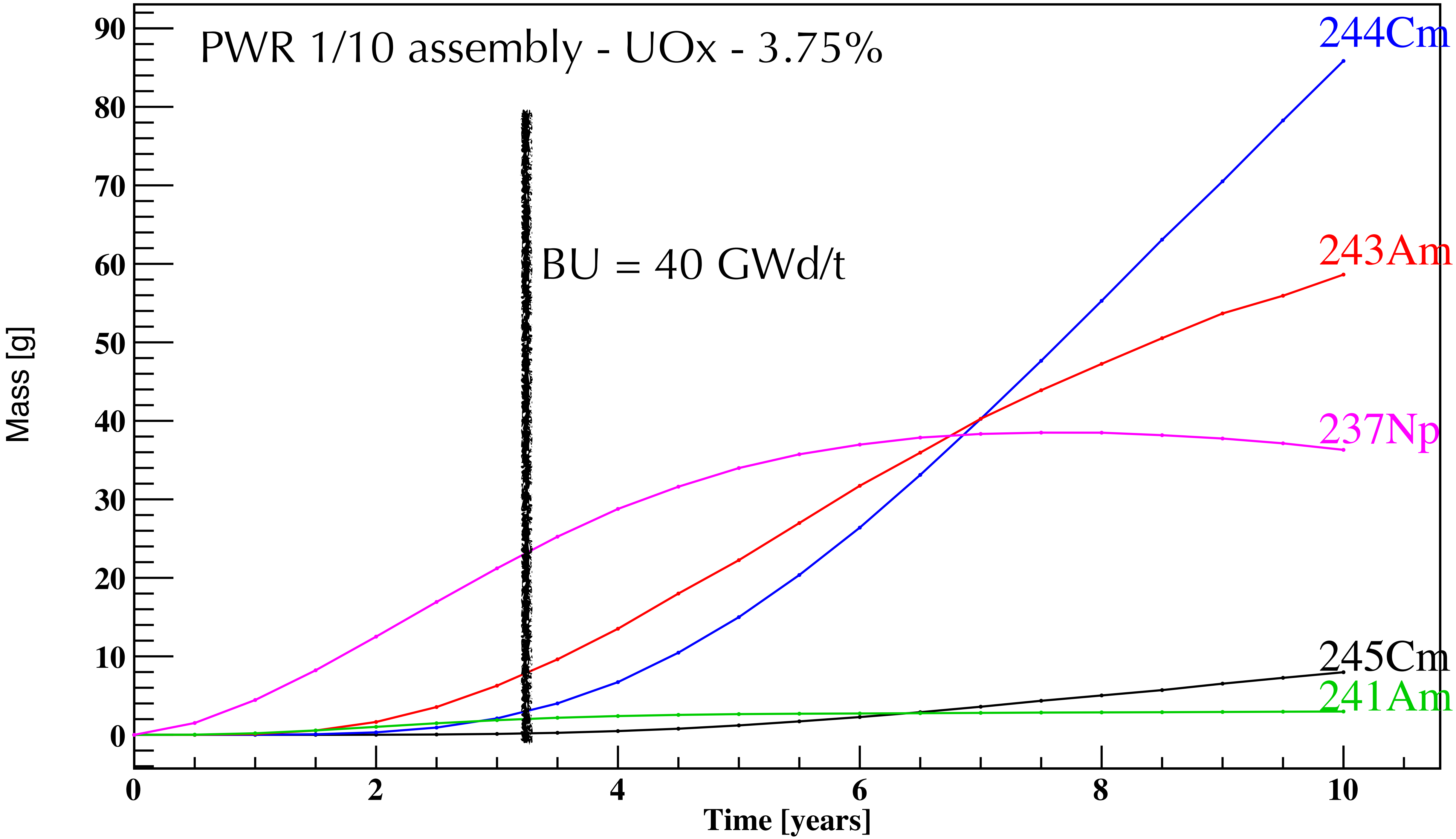
$$r_{4i} = f(y_i + \Delta t\, r_{3i}, t_i + \Delta t)$$

$$\longrightarrow y_{i+1} = y_i + \frac{\Delta t}{6}(r_{1i} + 2r_{2i} + 2r_{3i} + r_{4i})$$

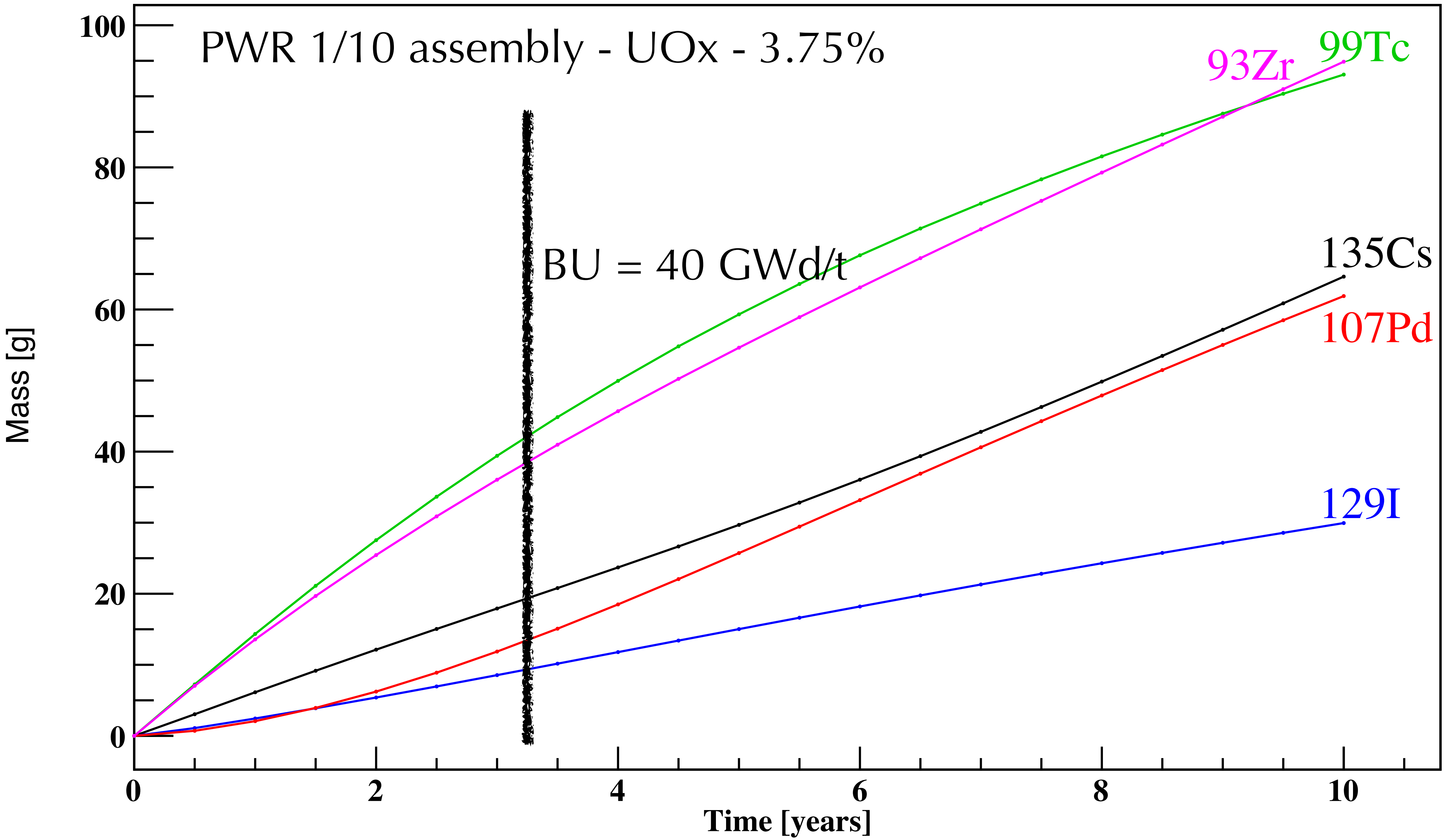
Example - PWR UOx



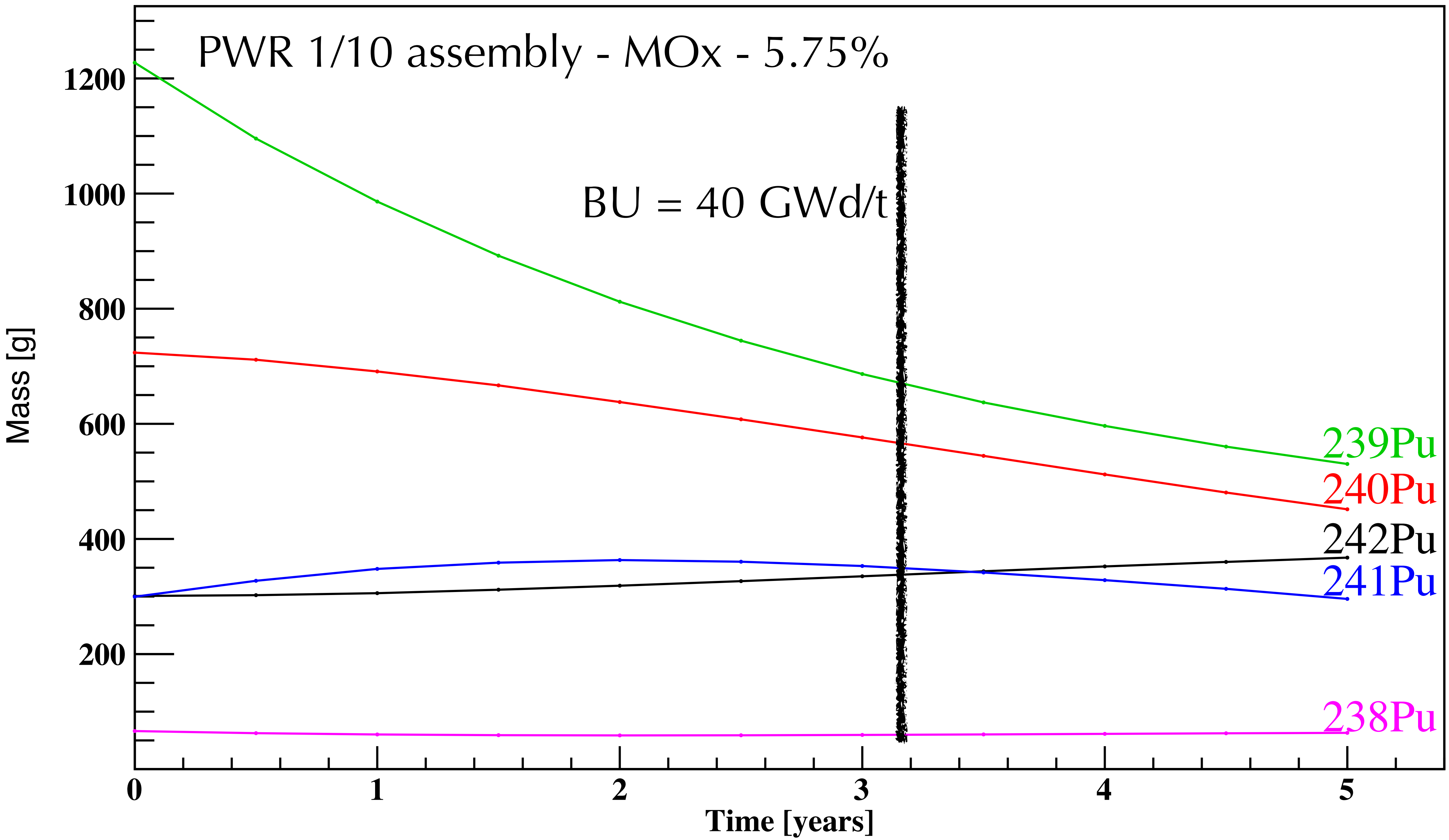
Example - PWR UOx



Example - PWR UOx

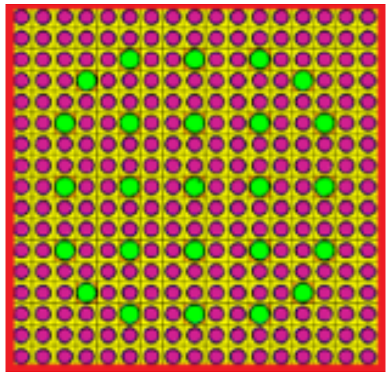


Example - PWR MOx



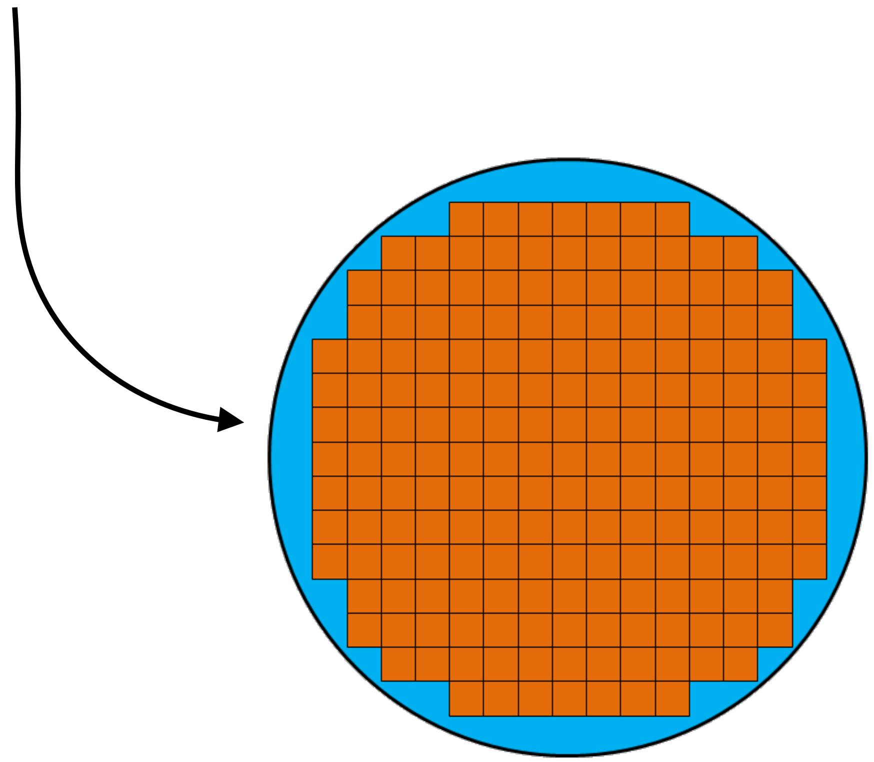
Scheme calculation

- Numerical resolution of a full core calculation is usually not possible because the calculation time is too high (deterministic or Monte-Carlo)
- In practice, a full core calculation is divided in two steps



- Cell calculation
  - Stationary Boltzmann equation with no leakage
  - Energy condensation and space homogenization
  - => XS multigroup

- Core calculation
  - Diffusion equation



$$-D\Delta\Phi = \nu\Sigma_f\Phi - \Sigma_a\Phi$$

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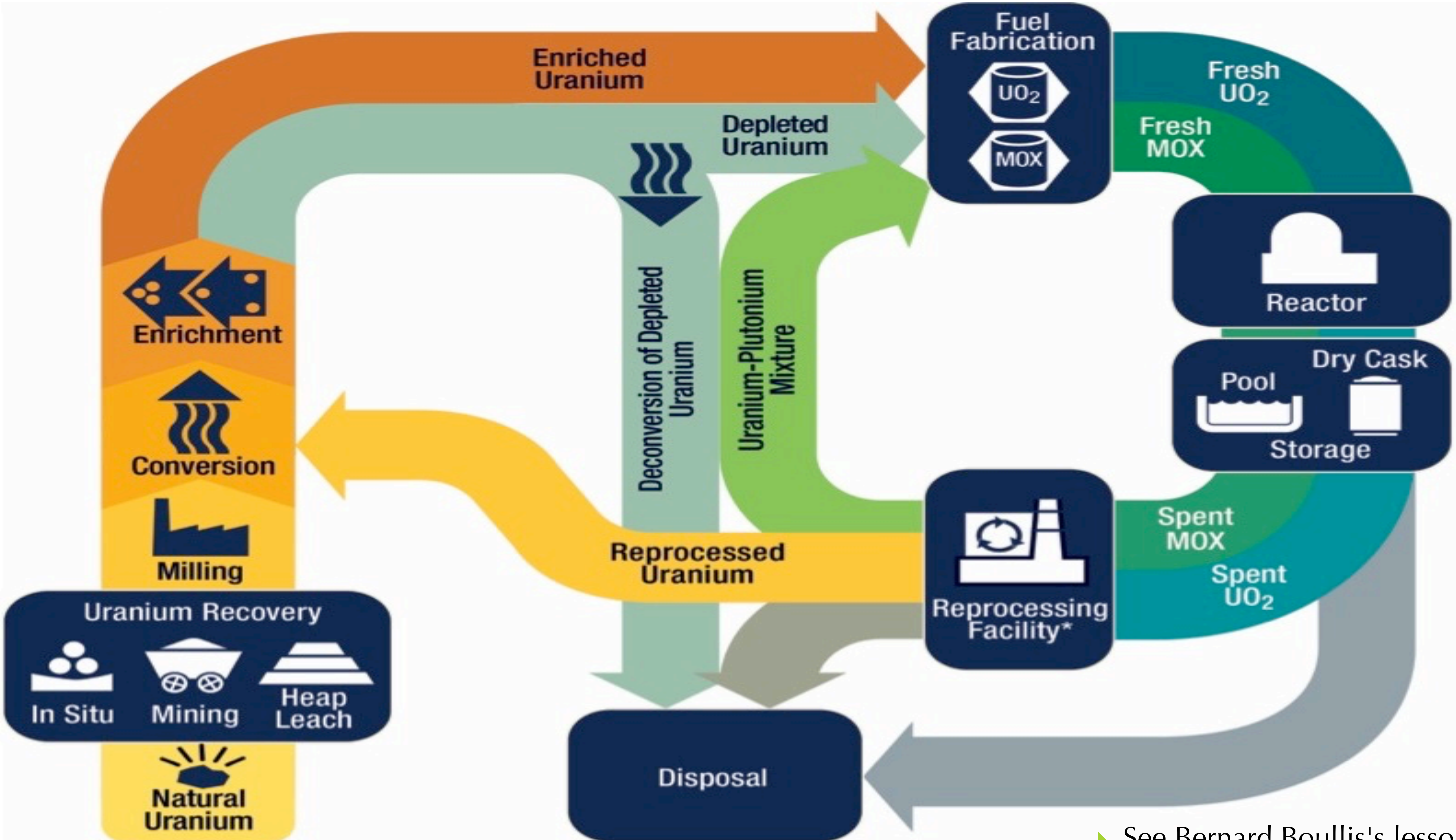
The fuel cycle



- See Bernard Boullis's lesson

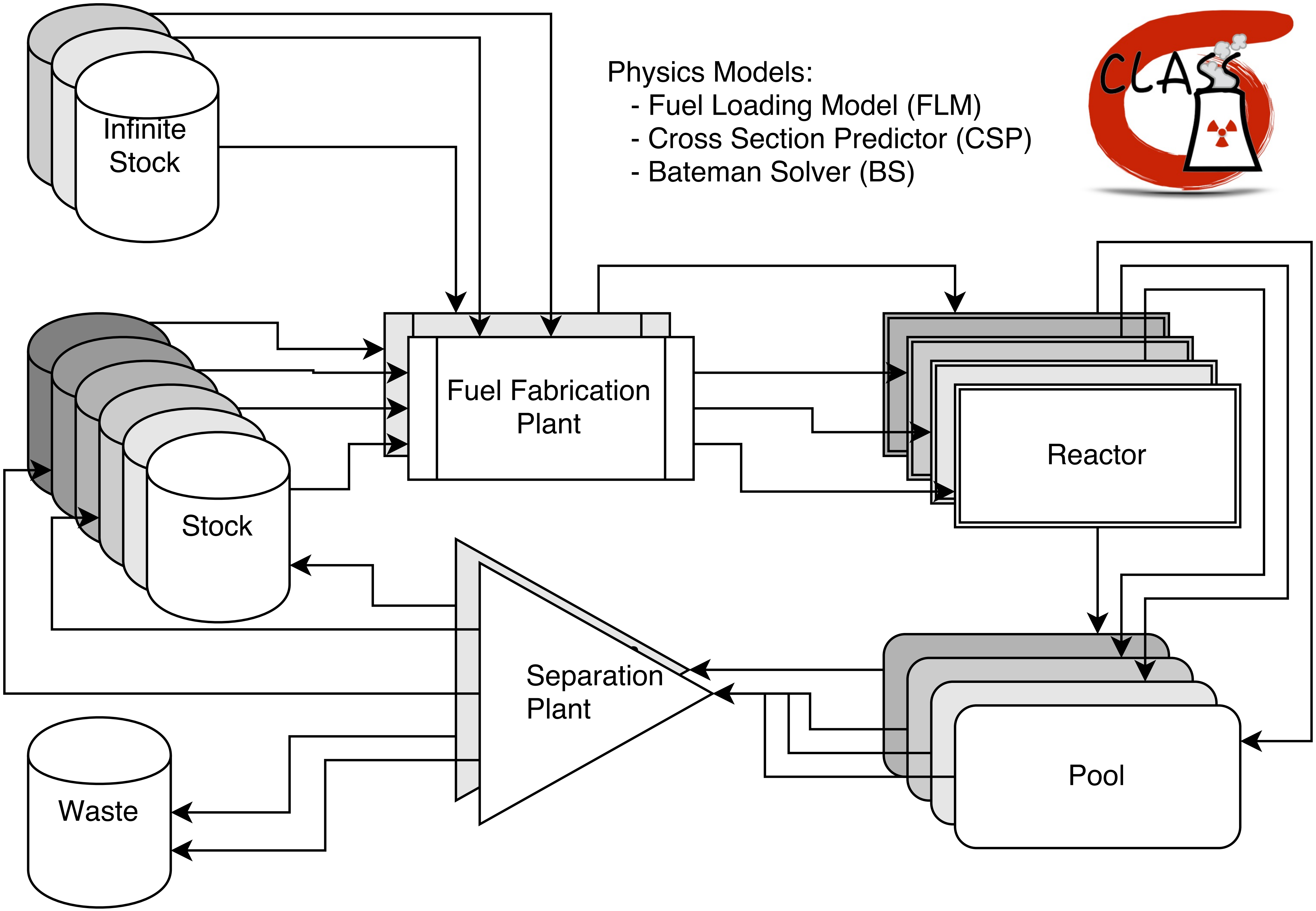
A lot of simulators

- A lot of fuel cycle simulators are developed worldwide

  - COSI
    - ➡ Developed by CEA since 80's to support french nuclear fleet management
    - ➡ Very detailed simulating framework with a lot of derived data
  - CLASS (Core Library for Advanced Scenario Simulation)
    - ➡ Developed by CNRS since 2010
    - ➡ Flexible C++ library connected to ROOT analysis framework
    - ➡ Used by IRSN for ASTRID scenario calculations
  - CYCLUS
    - ➡ Developed mainly by university of Madison-Wisconsin since few years
    - ➡ Agent-based fuel cycle simulator based on a powerful dynamic resource exchange solver
  - ORION (National Nuclear Laboratory, UK)
  - DYMOND (Argonne National Lab, US)
  - DANESS (Private company Nuclear-21)
  - Etc.

The code CLASS

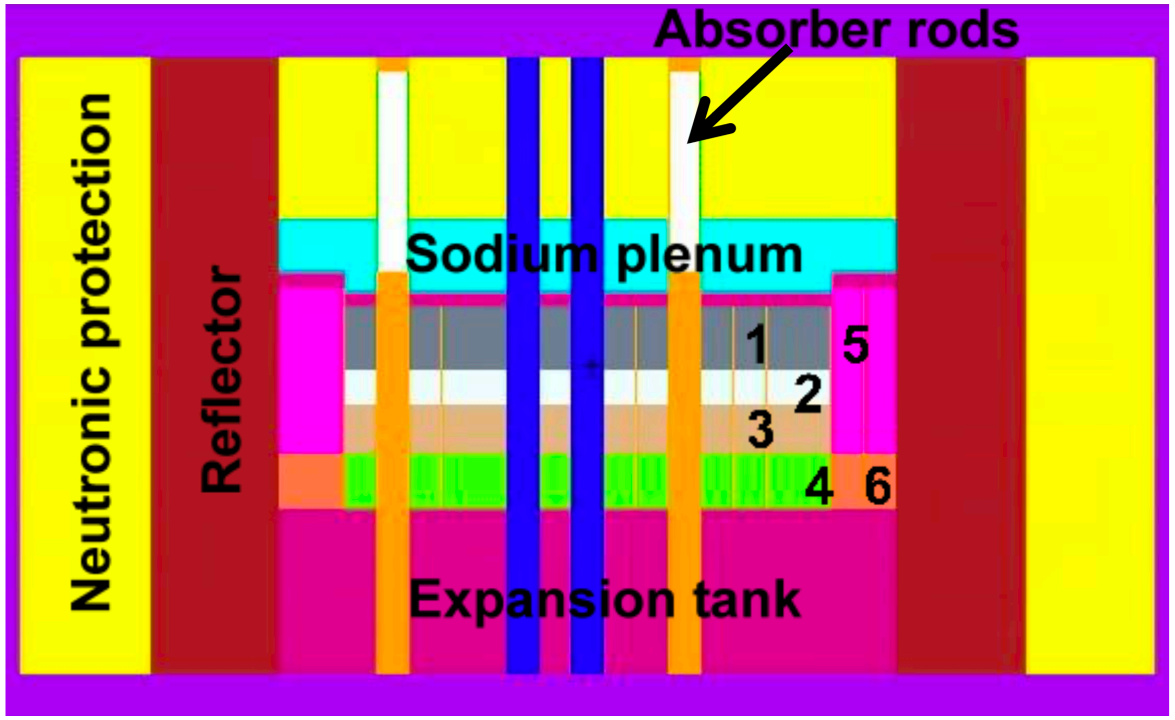
Physics Models:
- Fuel Loading Model (FLM)
- Cross Section Predictor (CSP)
- Bateman Solver (BS)



Reactor models

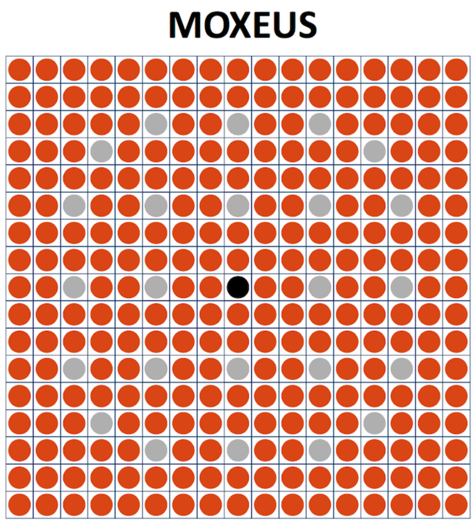
- Most important developments are related to physics models for reactors
  - Fuel Loading Model (FLM)
    - ➡ Neutronic data predictor
    - ➡ Fresh fuel loading algorithms
  - Cross Section Predictor (CSP)
  - Bateman Solver (BS)
- Up to now, several reactors have been implemented in CLASS
  - Pressurized Water Reactors
    - ➡ UOx
    - ➡ MOx
    - ➡ MOx-Am
    - ➡ MOx on enriched uranium support
  - Sodium fast reactors
    - ➡ ESFR
    - ➡ ASTRID

Léa Tillard PhD



An example of FLM

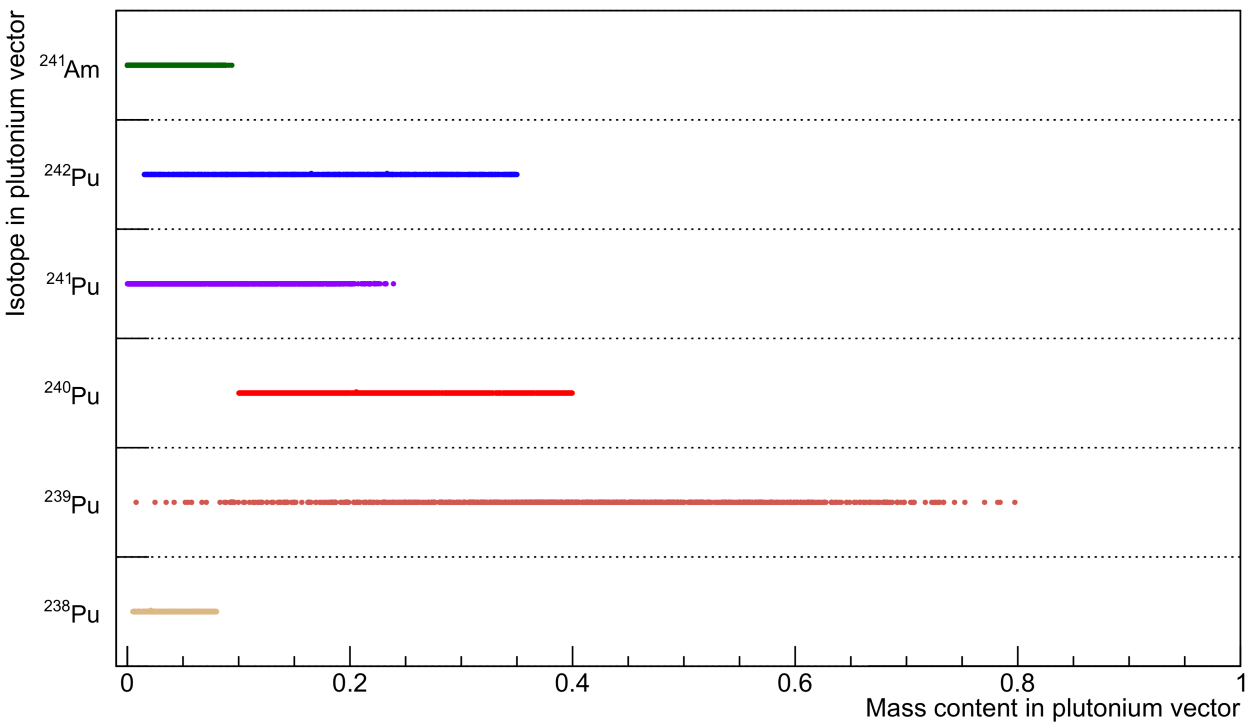
- Choice of the spatial scale
  - Pin
  - Assembly
  - Full core



- Choice of the transport code
  - Monte Carlo
    - Serpent
    - MCNP / SMURE
  - Deterministic
    - Donjon/Dragon
    - etc.

- Choice of the Pu vector boundaries

	^{235}U	w_{Pu}	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
min	0.25	0	0.5	10	10	0	1.5
max	5	16	8	80	40	25	35



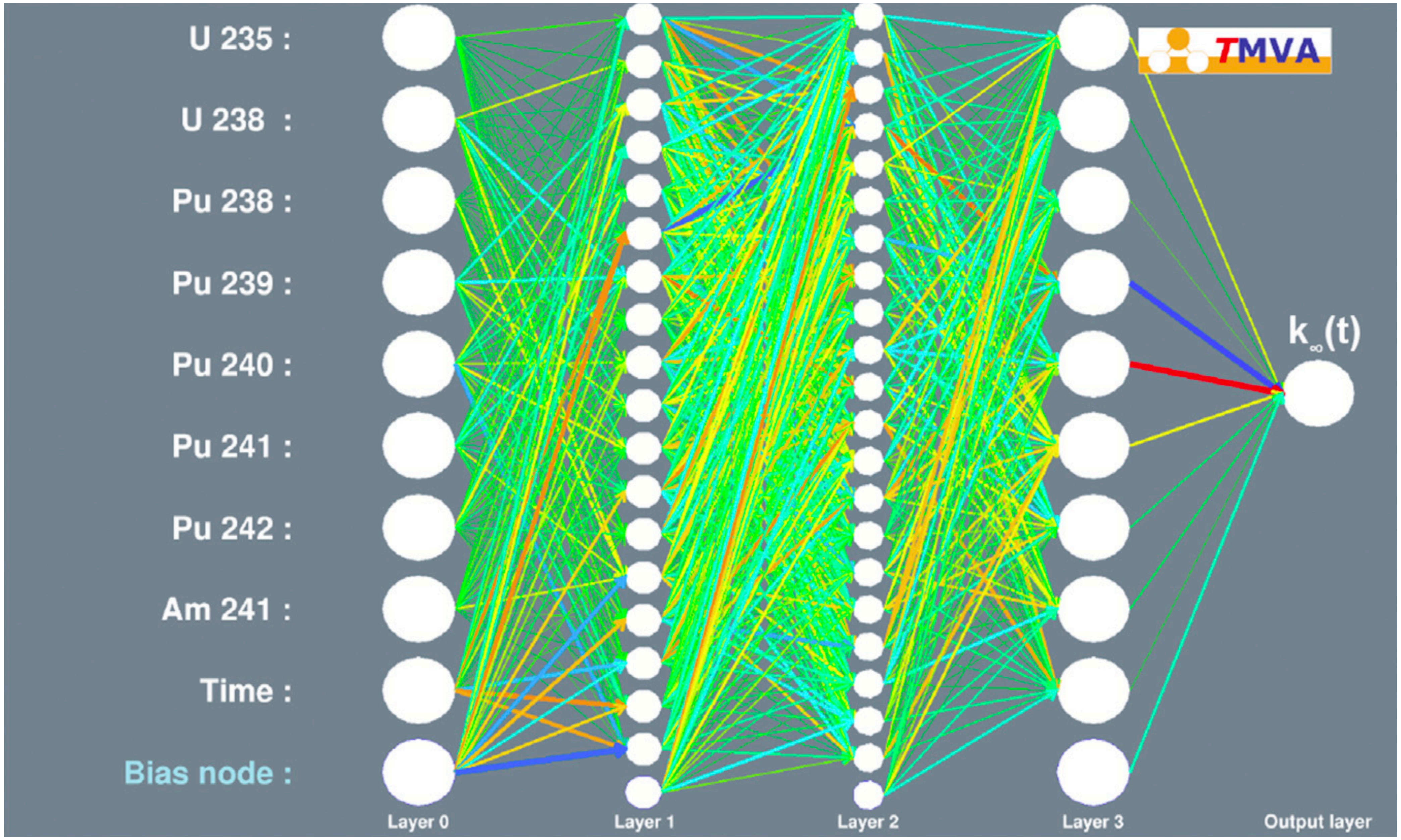
- Sampling on LHS with sum = 1
- Run of thousands simulations

F. Courtin, et al. Neutronic predictors for PWR fuelled with multi-recycled plutonium and applications with the fuel cycle simulation tool CLASS, Progress in Nuclear Energy, Volume 100, 2017.

An example of FLM

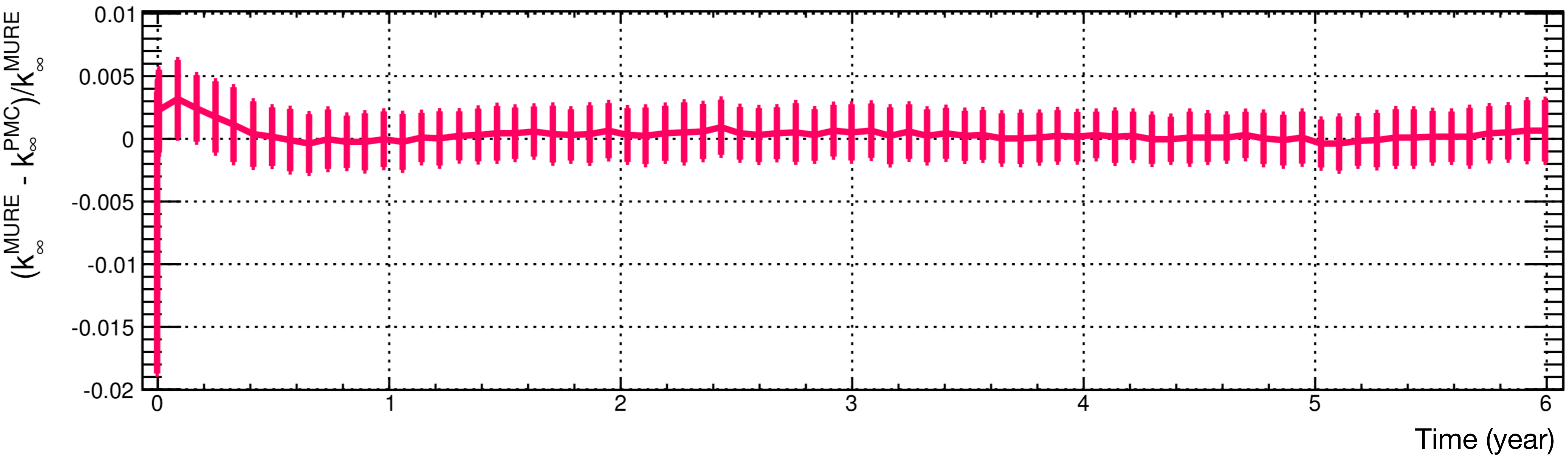
F. Courtin, et al. Neutronic predictors for PWR fuelled with multi-recycled plutonium and applications with the fuel cycle simulation tool CLASS, Progress in Nuclear Energy, Volume 100, 2017.

- Prediction of reactivity evolution
  - Polynoms
  - Neural networks



Precision of the predictor

- Precision of prediction from an independent set of data
  - 100 independant runs
  - Represented points are relative deviation mean values
  - Error bars are standard deviations



- The prediction precision is close to Monte-Carlo uncertainty
- An artefact at small irradiation time is coming from xenon effect

Fanny Courtin. Etude de l'incinération du plutonium en REP MOX sur support d'uranium enrichi avec le code de simulation dynamique du cycle CLASS. Physique Nucléaire Expérimentale [nucl-ex]. Ecole nationale supérieure Mines-Télécom Atlantique, 2017. Français. NNT : 2017IMTA0044 . tel- 01668610

Fuel batching

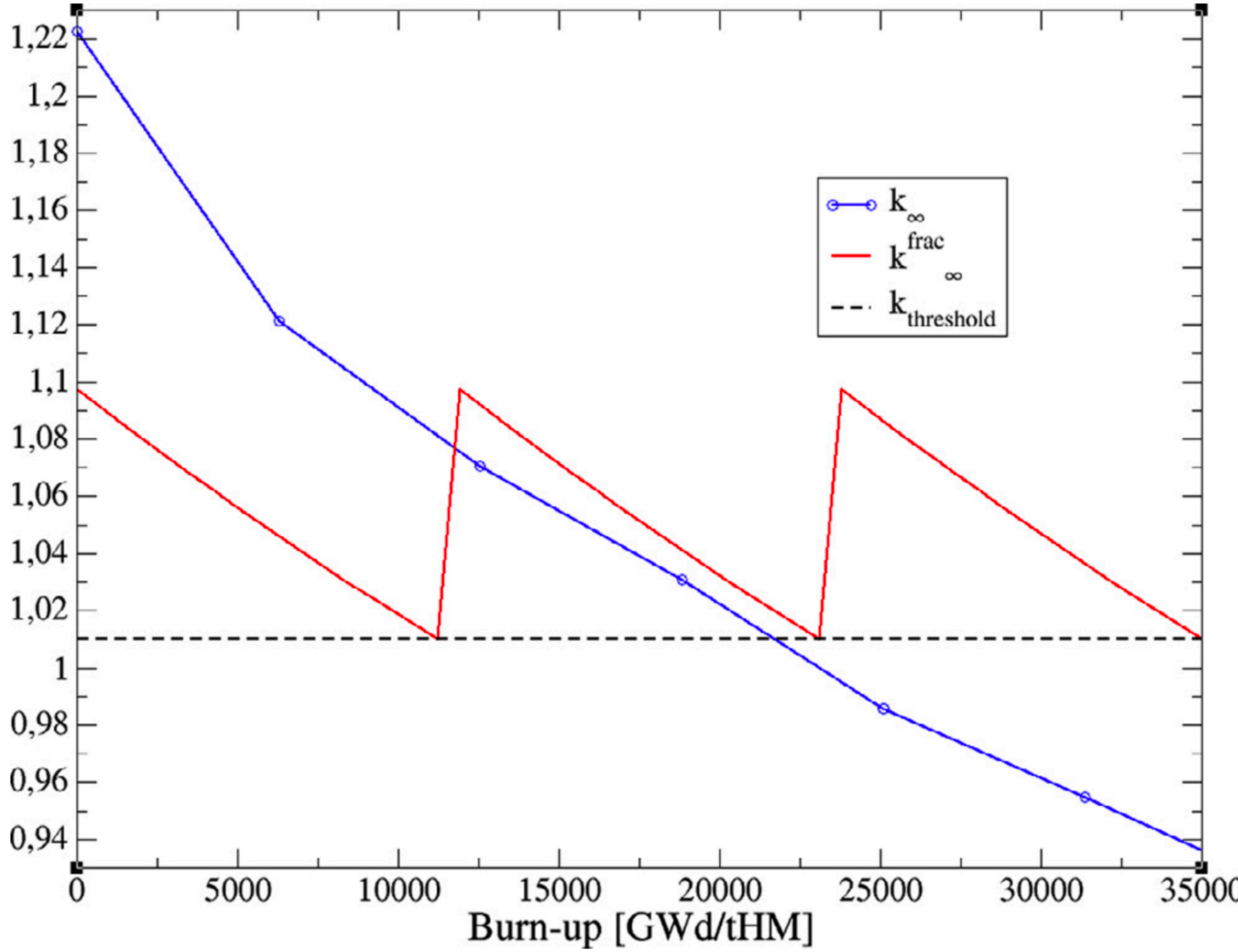


Fig. 2. k_∞^{frac}, k_∞ and $k_{threshold}$ of a PWR MOX with a maximum burnup achievable of 35 GWd/tHM and a 3 batches management.

- Reactivity with batches

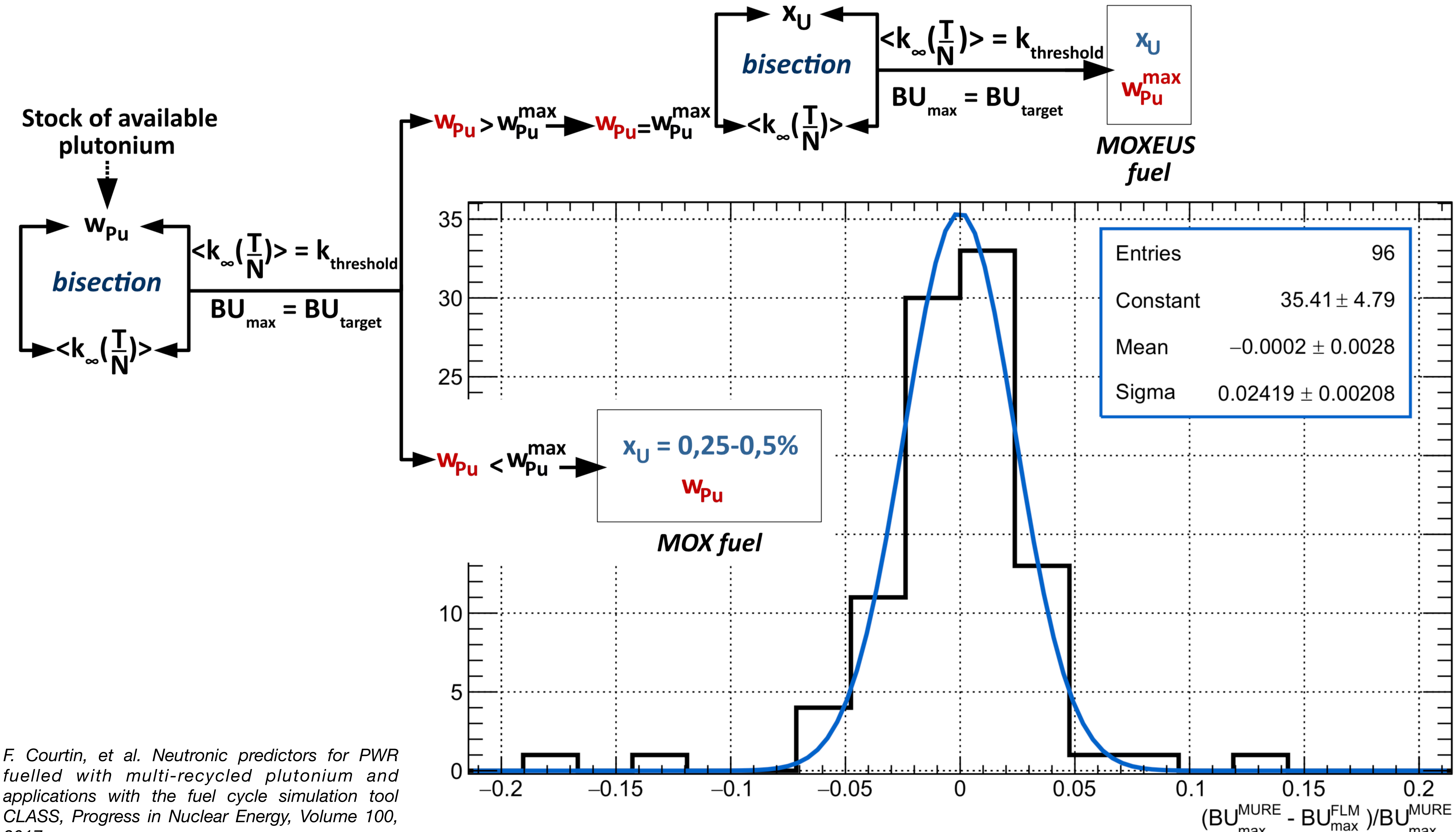
$$< k_\infty^C > (t) \sim \frac{1}{N}\sum_{i=0}^{N-1} k_\infty(t + \frac{iT}{N})$$

- Maximum burn-up

$$< k_\infty^C > (t = \frac{T}{N}) \sim 1 + F \sim k_{seuil}$$

Baptiste Leniau et al. A neural network approach for burn-up calculation and its application to the dynamic fuel cycle code CLASS.

Maximal burn-up

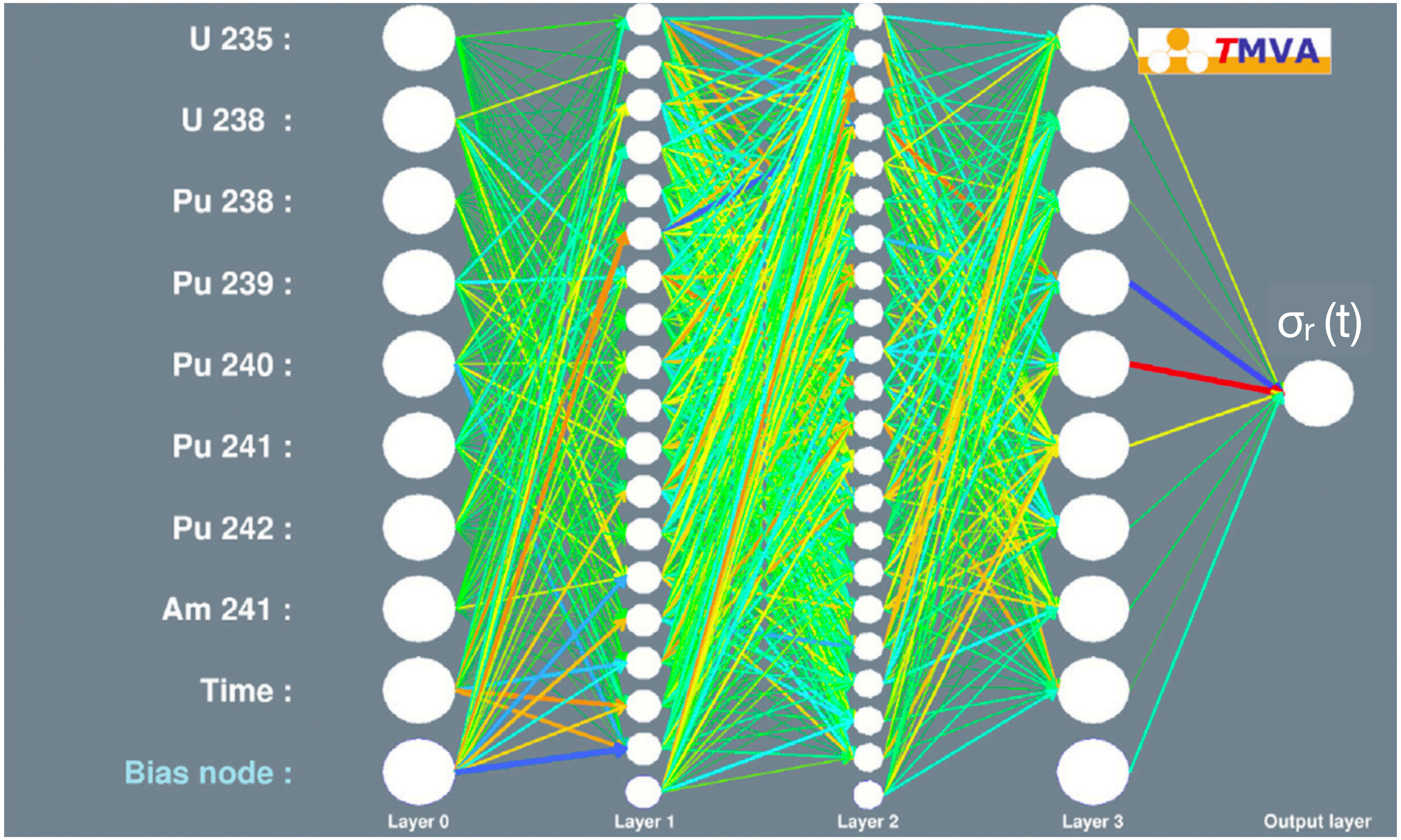


F. Courtin, et al. Neutronic predictors for PWR fuelled with multi-recycled plutonium and applications with the fuel cycle simulation tool CLASS, Progress in Nuclear Energy, Volume 100, 2017.

An example of XSM

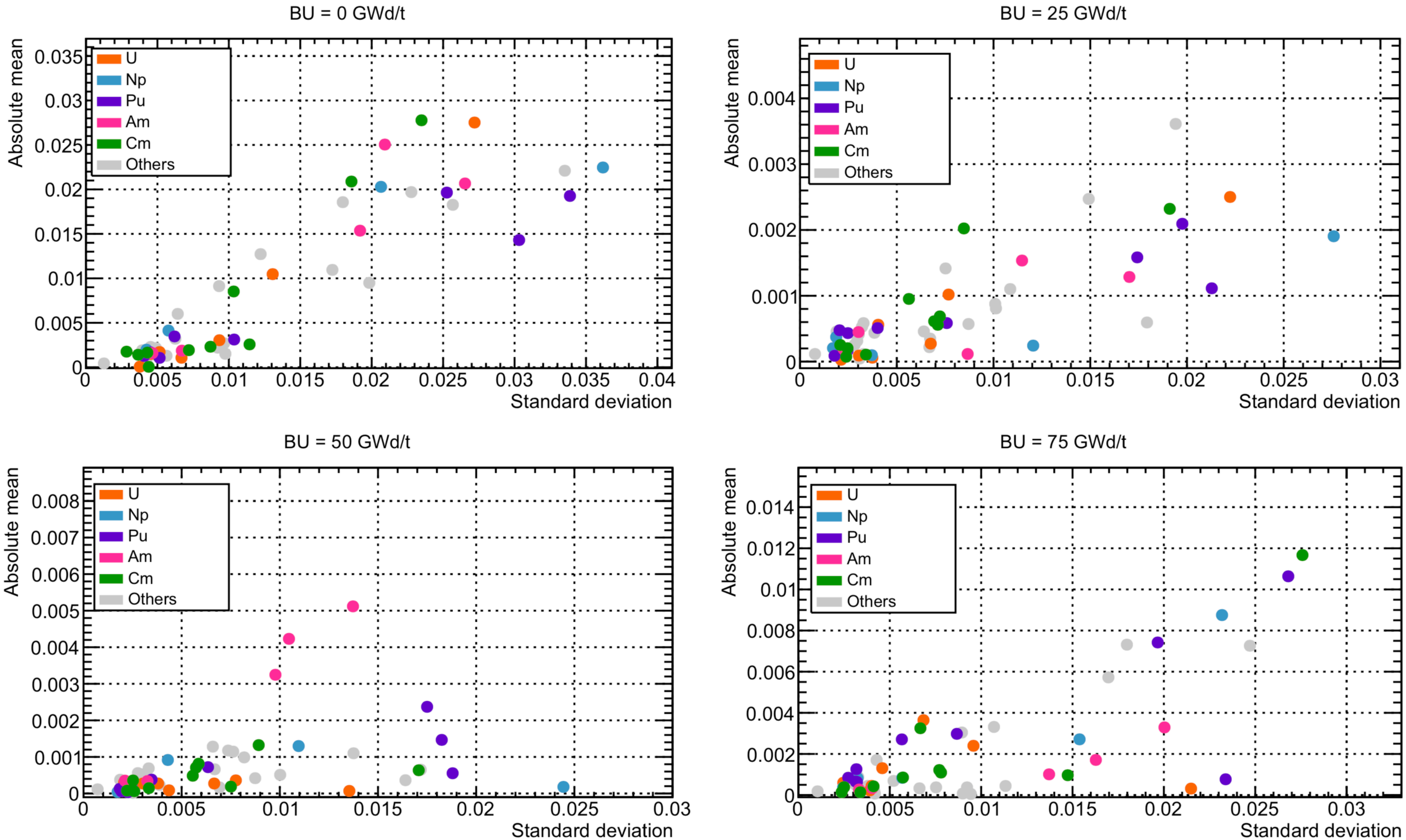
- Prediction of cross sections
  - Polynoms
  - This example : Neural networks

Fanny Courtin. Etude de l'incinération du plutonium en REP MOX sur support d'uranium enrichi avec le code de simulation dynamique du cycle CLASS. Physique Nucléaire Expérimentale [nucl-ex]. Ecole nationale supérieure Mines-Télécom Atlantique, 2017. Français. NNT : 2017IMTA0044 . tel-01668610



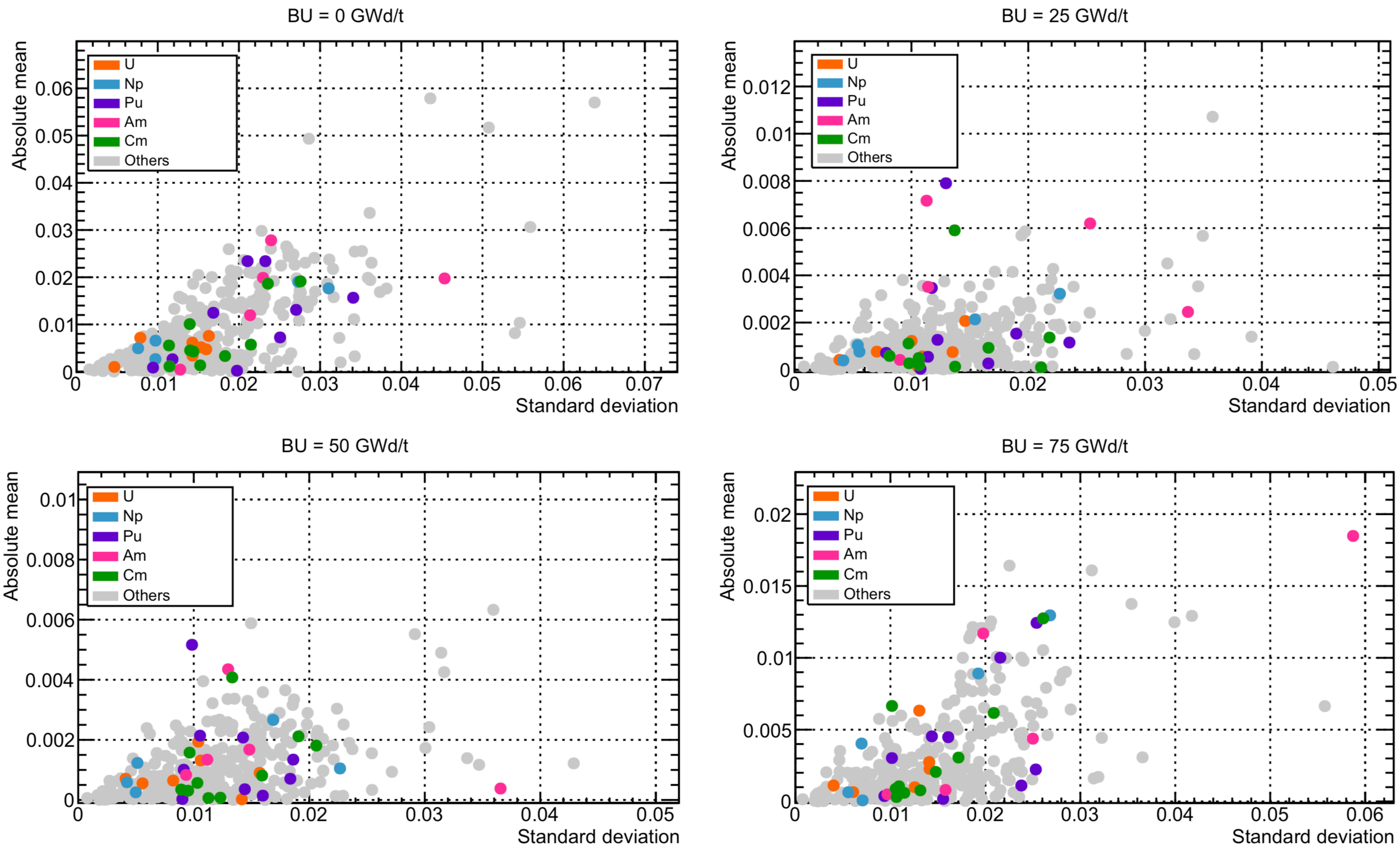
- ~700 neural networks
- 3 réactions
  - (n,f)
  - (n,g)
  - (n,2n)

Precision of (n,f)



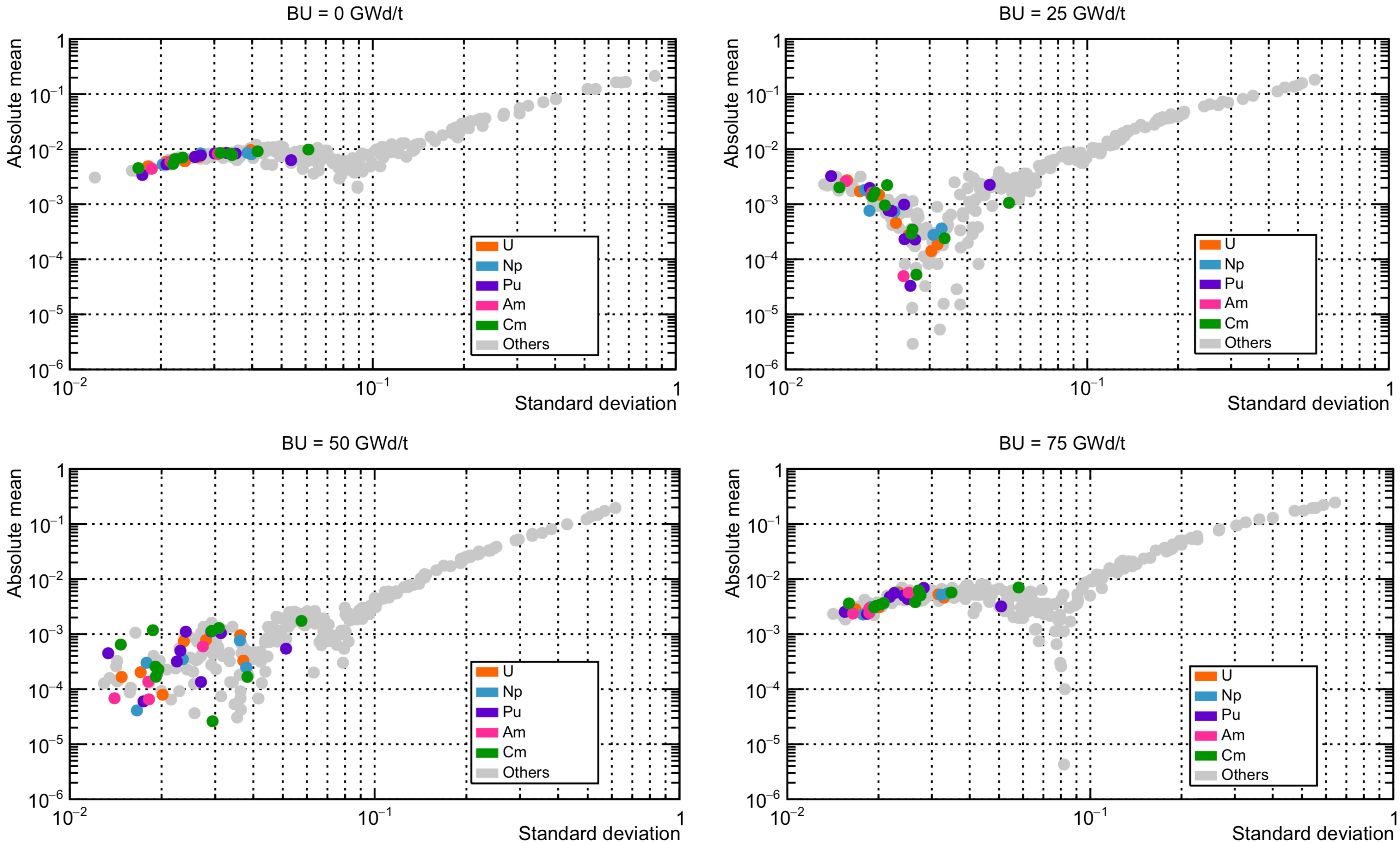
Fanny Courtin. Etude de l'incinération du plutonium en REP MOX sur support d'uranium enrichi avec le code de simulation dynamique du cycle CLASS. Physique Nucléaire Expérimentale [nucl-ex]. Ecole nationale supérieure Mines-Télécom Atlantique, 2017. Français. NNT : 2017IMTA0044 . tel- 01668610

Precision of (n,g)



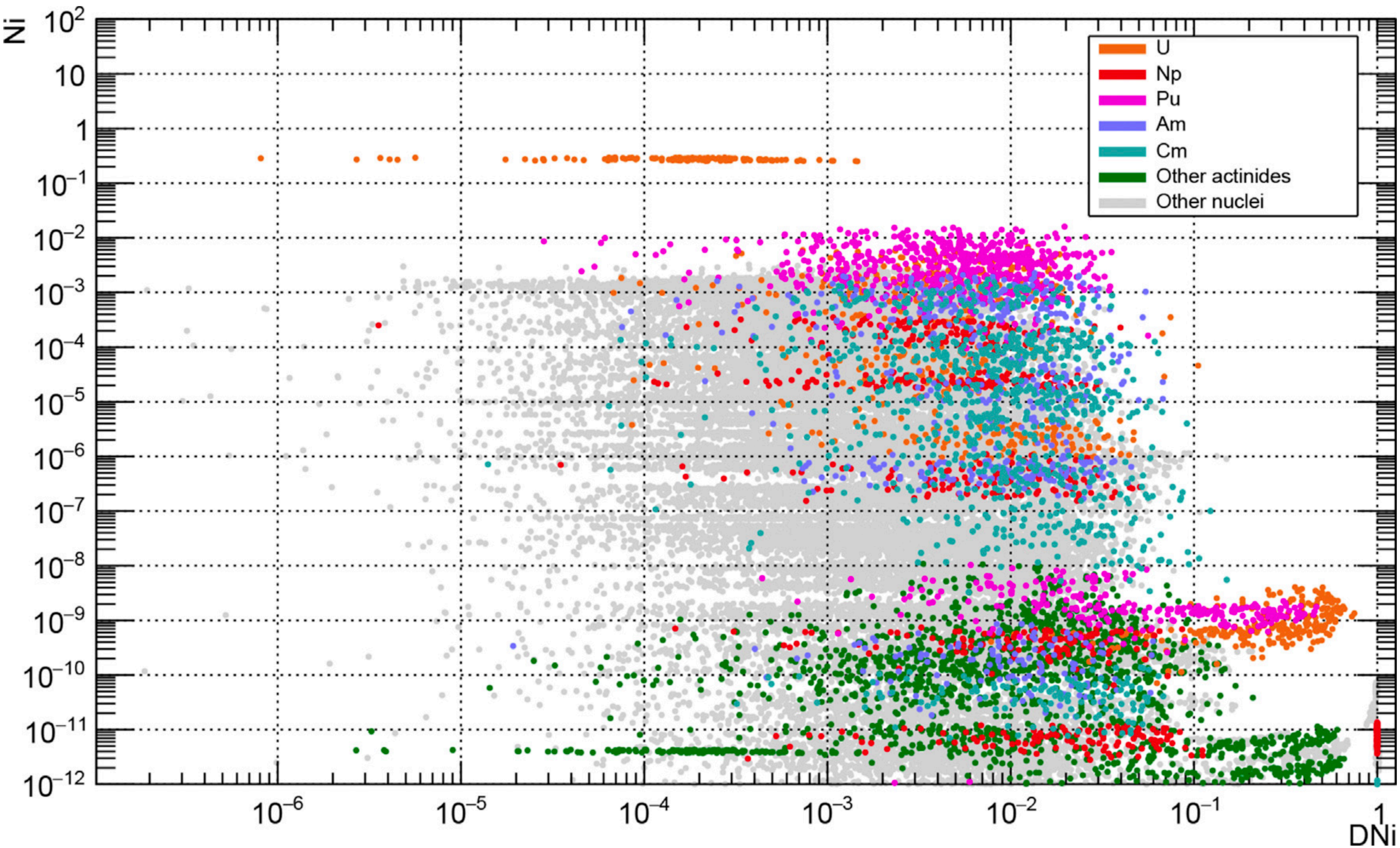
Fanny Courtin. Etude de l'incinération du plutonium en REP MOX sur support d'uranium enrichi avec le code de simulation dynamique du cycle CLASS. Physique Nucléaire Expérimentale [nucl-ex]. Ecole nationale supérieure Mines-Télécom Atlantique, 2017. Français. NNT : 2017IMTA0044 . tel- 01668610

Precision of (n,2n)



Fanny Courtin. Etude de l'incinération du plutonium en REP MOX sur support d'uranium enrichi avec le code de simulation dynamique du cycle CLASS. Physique Nucléaire Expérimentale [nucl-ex]. Ecole nationale supérieure Mines-Télécom Atlantique, 2017. Français. NNT : 2017IMTA0044 . tel- 01668610

Precision of evolution calculation



Scenarios methodology

Fanny Courtin PhD. IMT Atlantique 2017.

- Fuel Cycle Simulators (FCS) are developed for many purposes
  - Study existing nuclear fleet in support for industrial operation optimization
  - Study and analysis of electro-nuclear future trajectories for prospective reflexions
  - Verification and/or assessment of nuclear fleets by safety authorities
  - Training and educational tool for the fuel cycle understanding

- FCS confidence outputs is a major issue

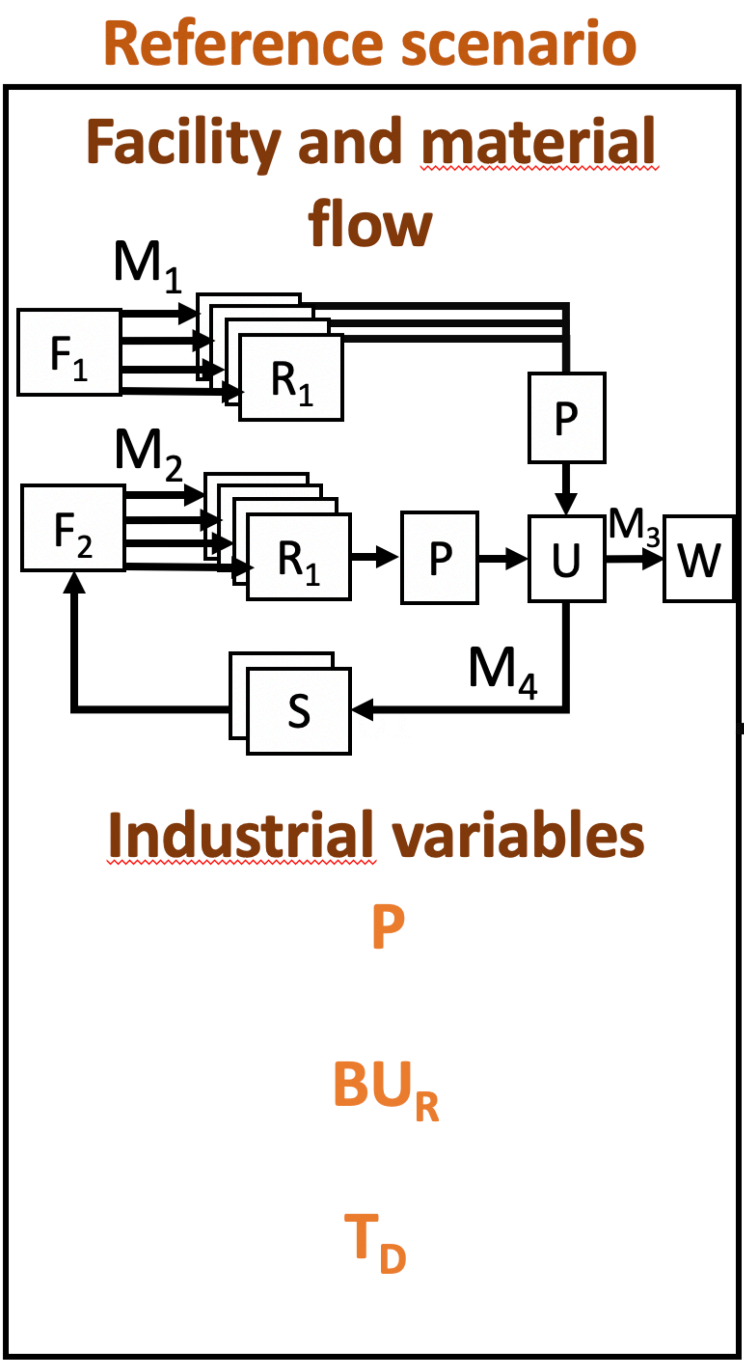
- FCS bias & uncertainty
  - Reactor simplifications
    - ➡ System simulations
    - ➡ Nuclear Data
  - Scenario simplifications
    - ➡ Technical parameters
    - ➡ Facility operating hypothesis
  - FCS use
    - ➡ Problem definition
    - ➡ Problem solving method

- Reactors studies
  - Code development and qualification
  - Benchmarks
  - Precise reactor coupling with FCS

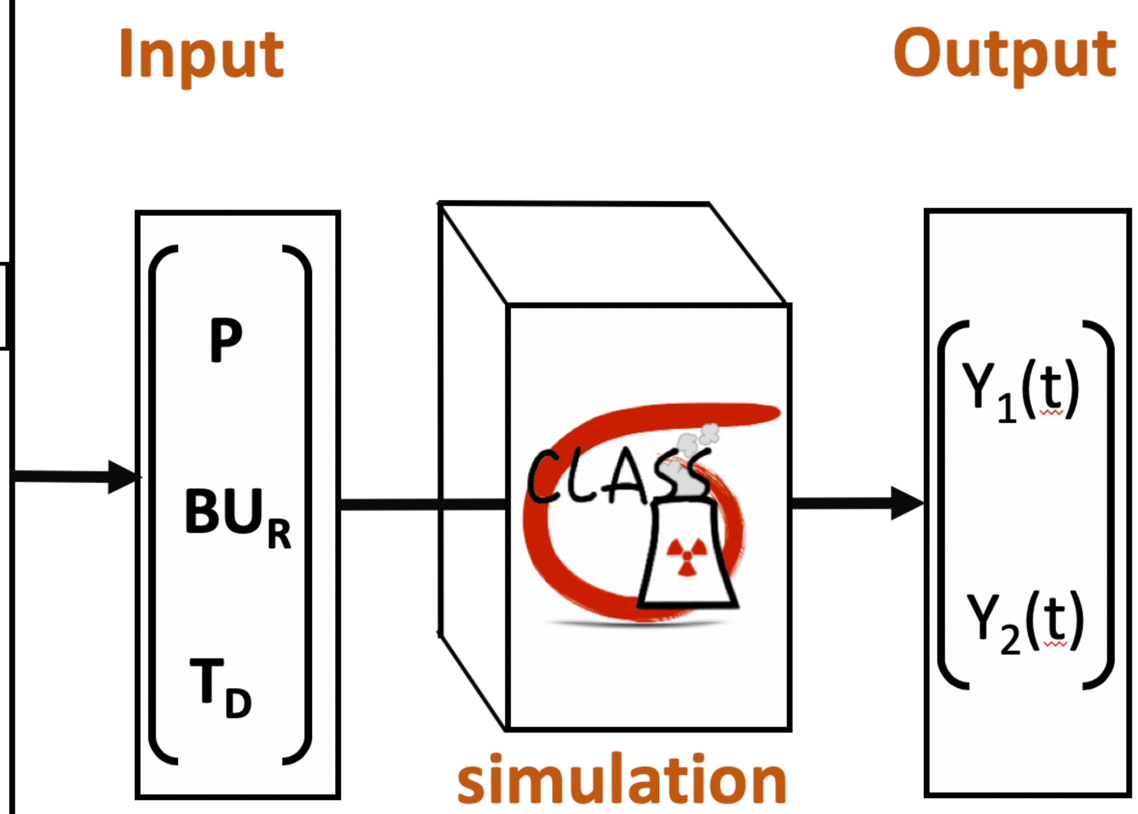
- Fuel cycle studies
  - Functionality testing
  - Global benchmarks

- FCS use
  - Sociology related question

- Historical methodology to build and run nuclear scenario



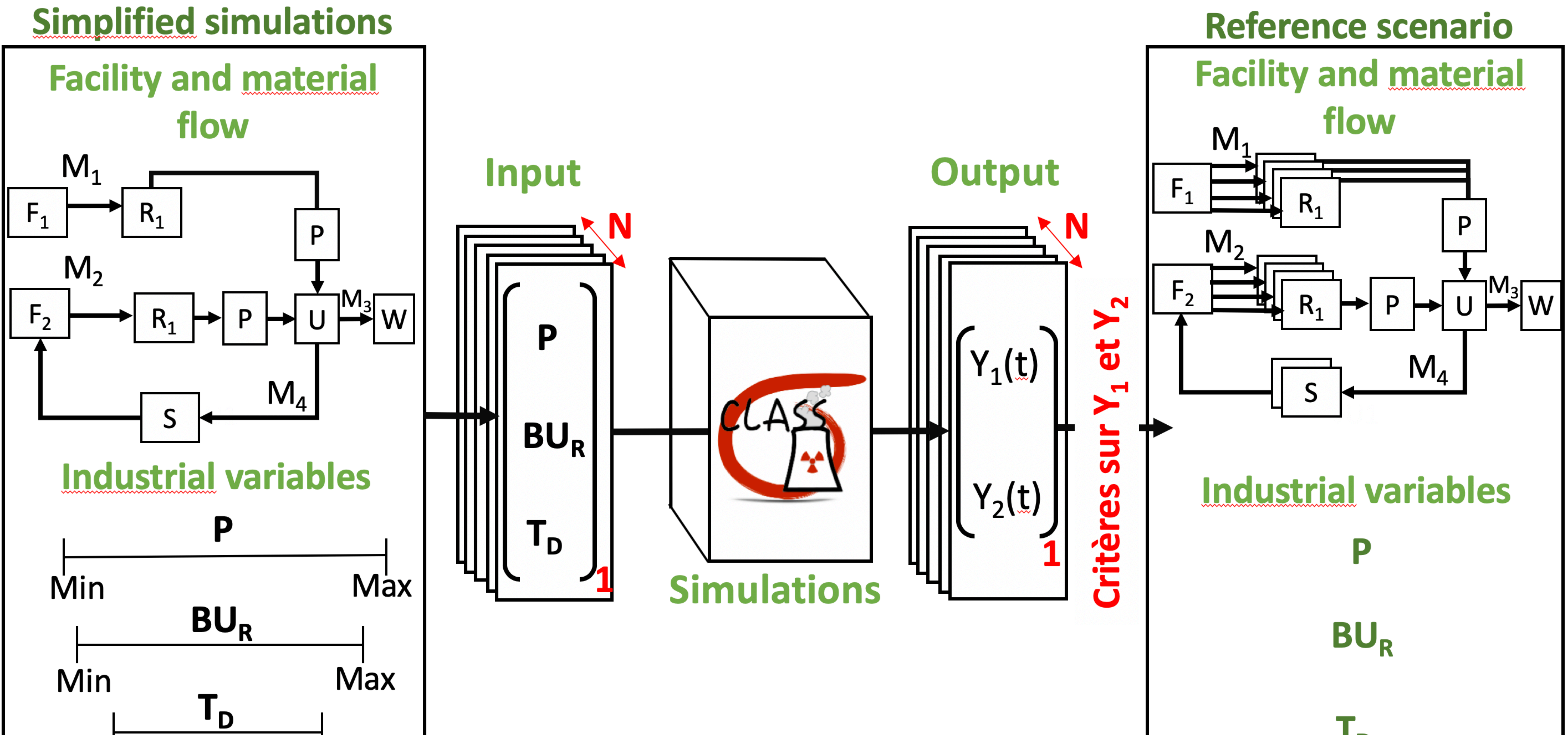
- Scenario simplifications
  - ➡ Technical parameters
  - ➡ Facility operating hypothesis



Fanny Courtin PhD. IMT Atlantique 2017.

Scenarios methodology

- New methodology is used to define reference scenarios



Fanny Courtin PhD. IMT Atlantique 2017.

- A fuel cycle simulator is using nuclear data for physics models :
  - Fuel loading models
  - Irradiation in reactors
  - Cooling in facilities

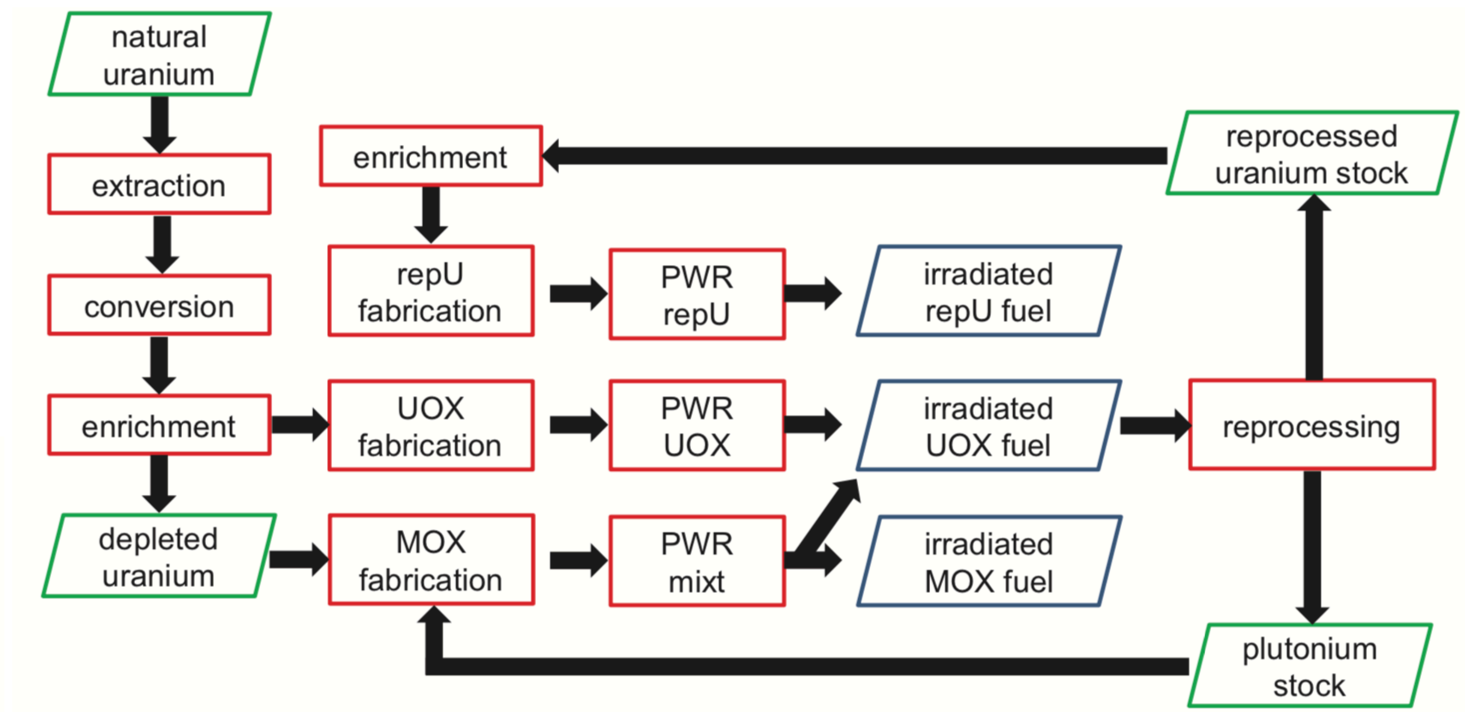


Figure 6.19: Scenario C: fuel cycle before 2038

- Propagation on scenarios :

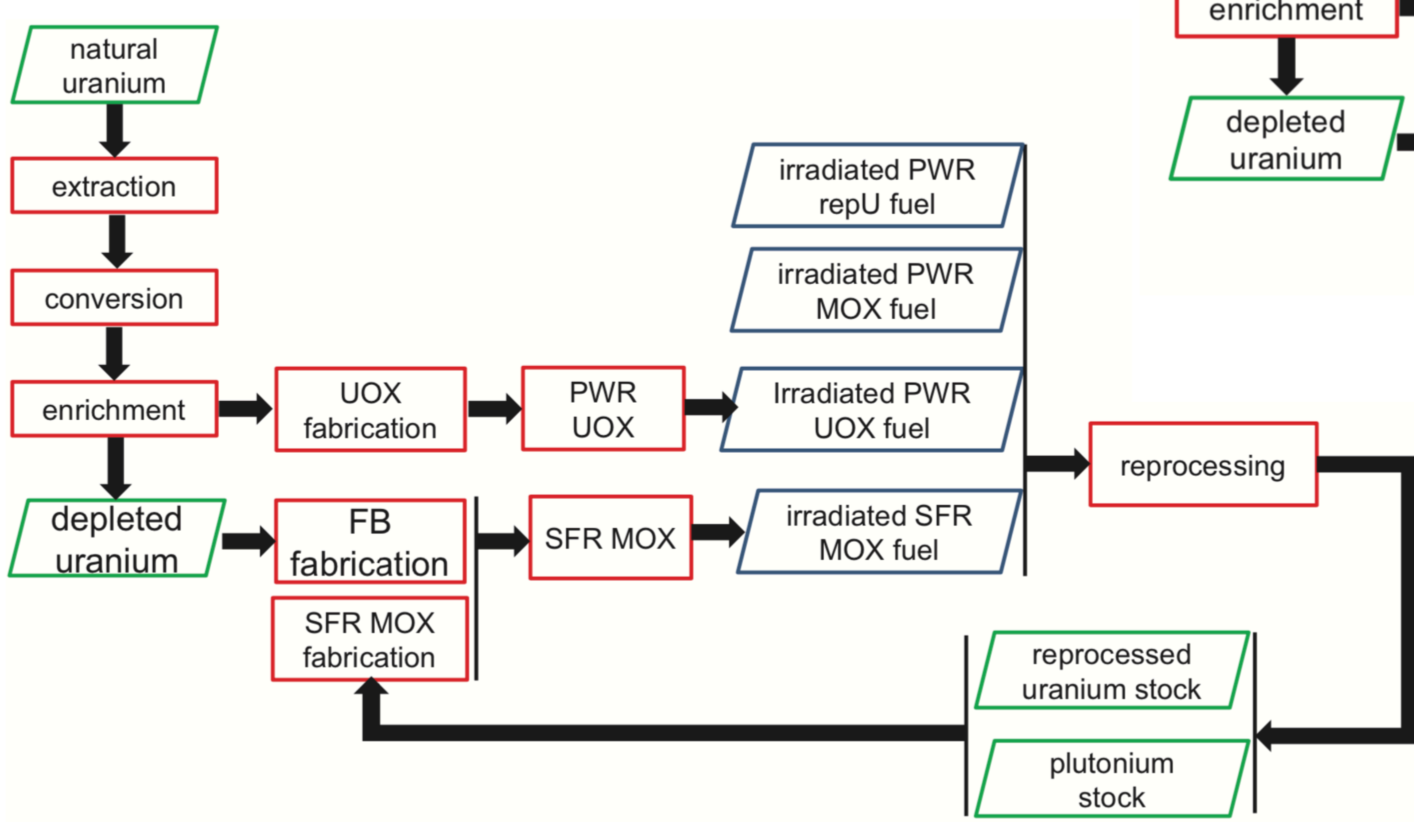


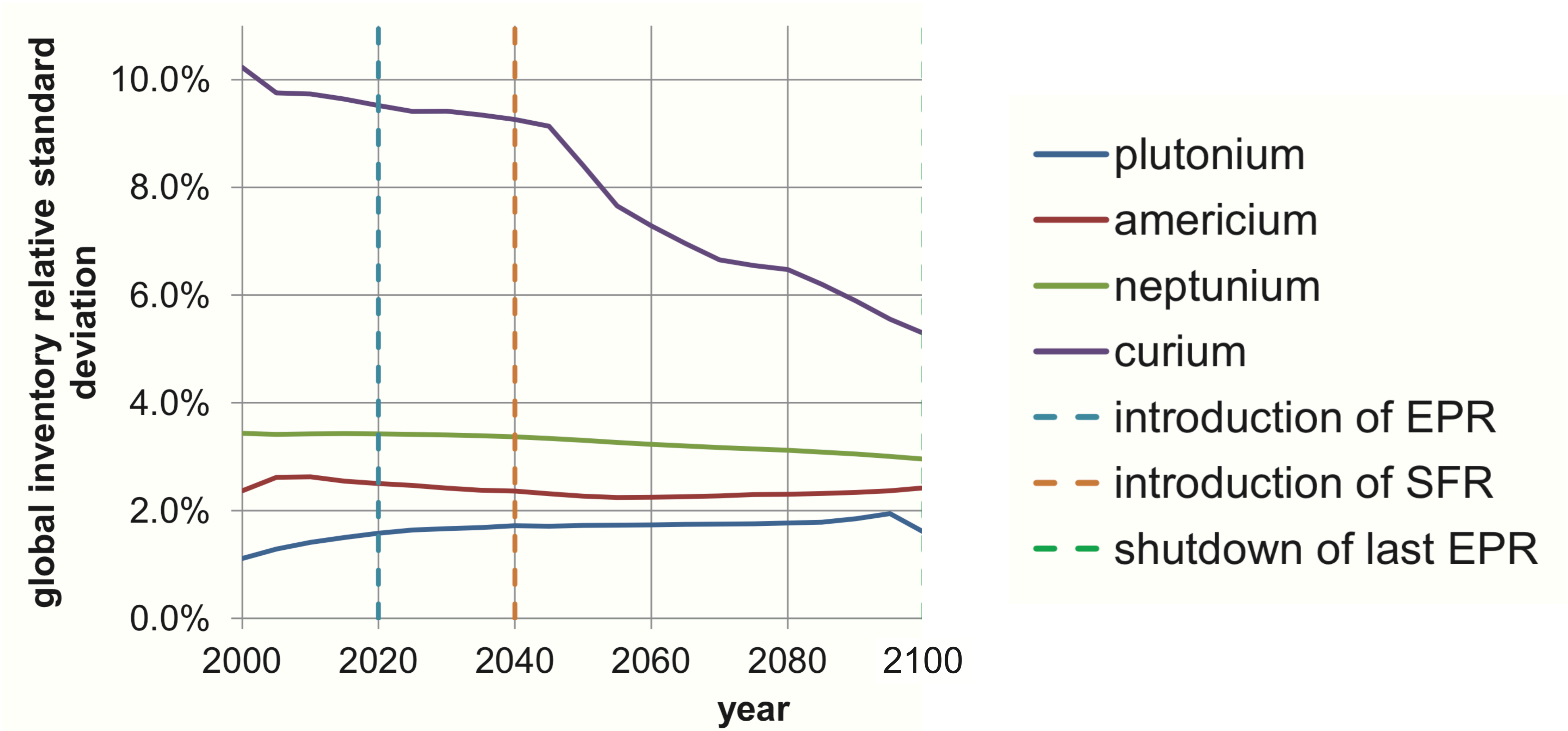
Figure 6.20: Scenario C: fuel cycle between 2038 and 2100

- Reactor simplifications
  ➡ Nuclear Data

G. Krivtchik. PhD, CEA 2014.

Nuclear data

- For all simulations, nuclear data are sampled with the correlations

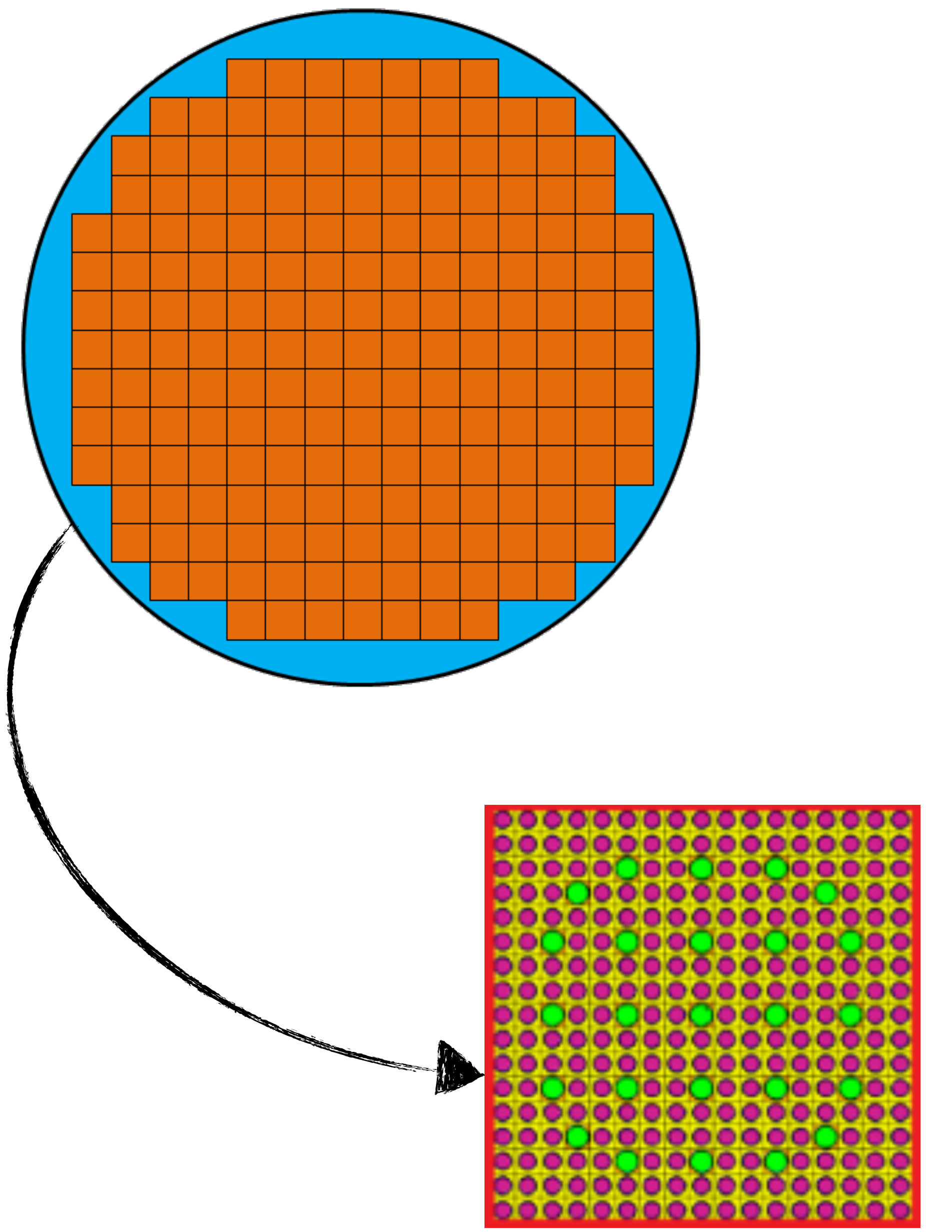


- Nuclear data have a significant impact on Pu and MA inventories
- But the effect is probably small compared to other uncertainty sources

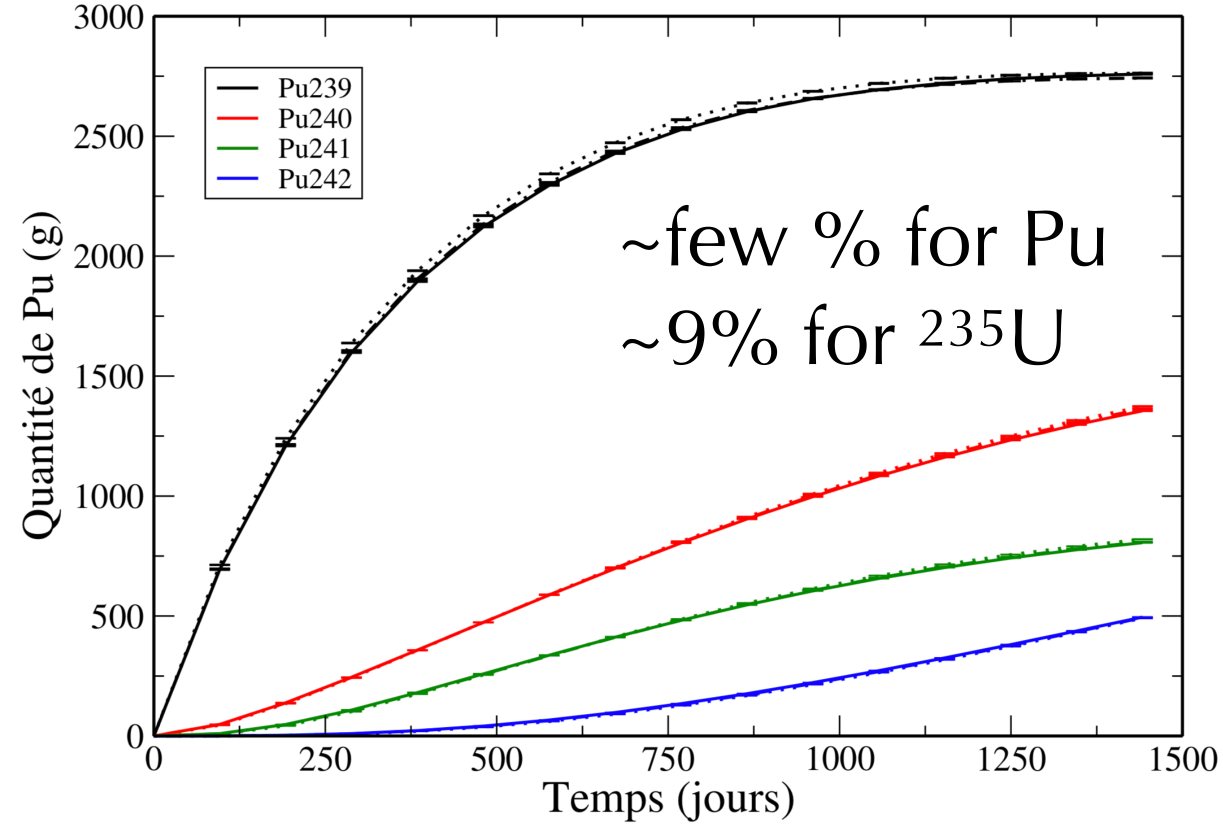
G. Krivtchik. PhD, CEA 2014.

Reactor simplifications

- A fuel cycle simulator is using neutronic predictors from reactor simulations



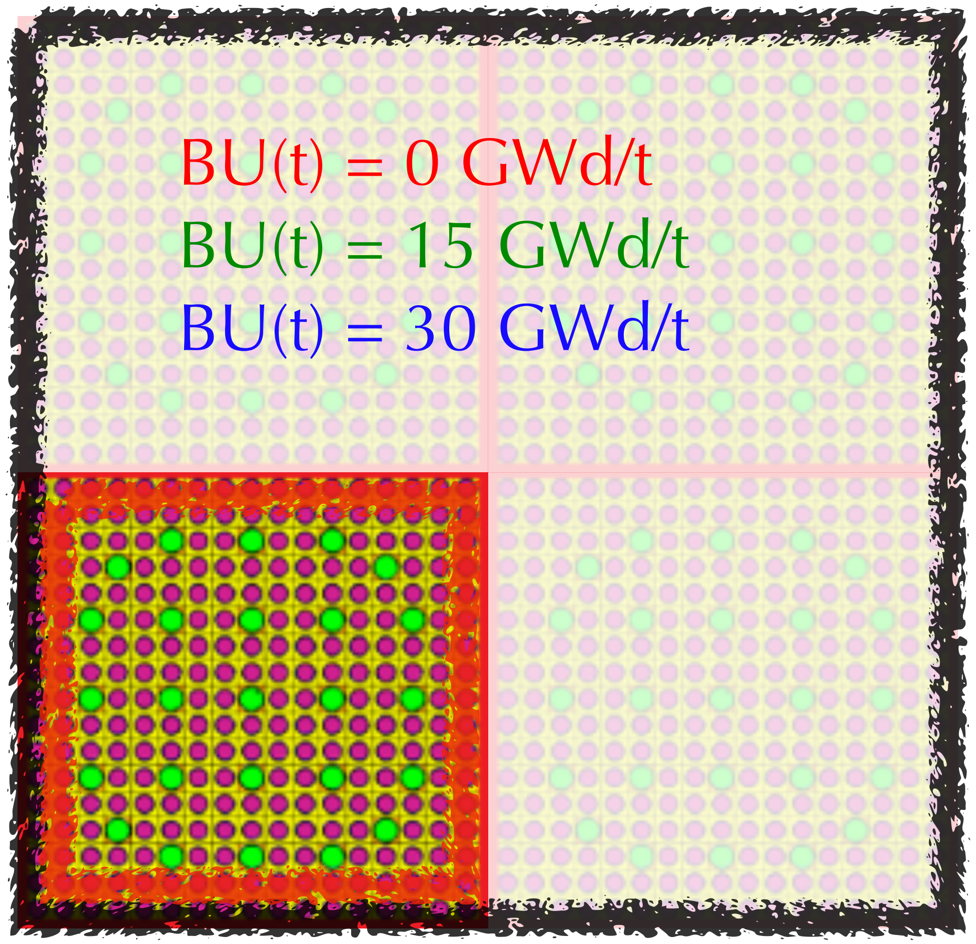
- Reactor simulations are very simplified
  - Full core calculation scheme is complex
  - A lot of calculation are required
    - ➡ Use of infinite assembly calculation
- Impact on neutrons axial leakage

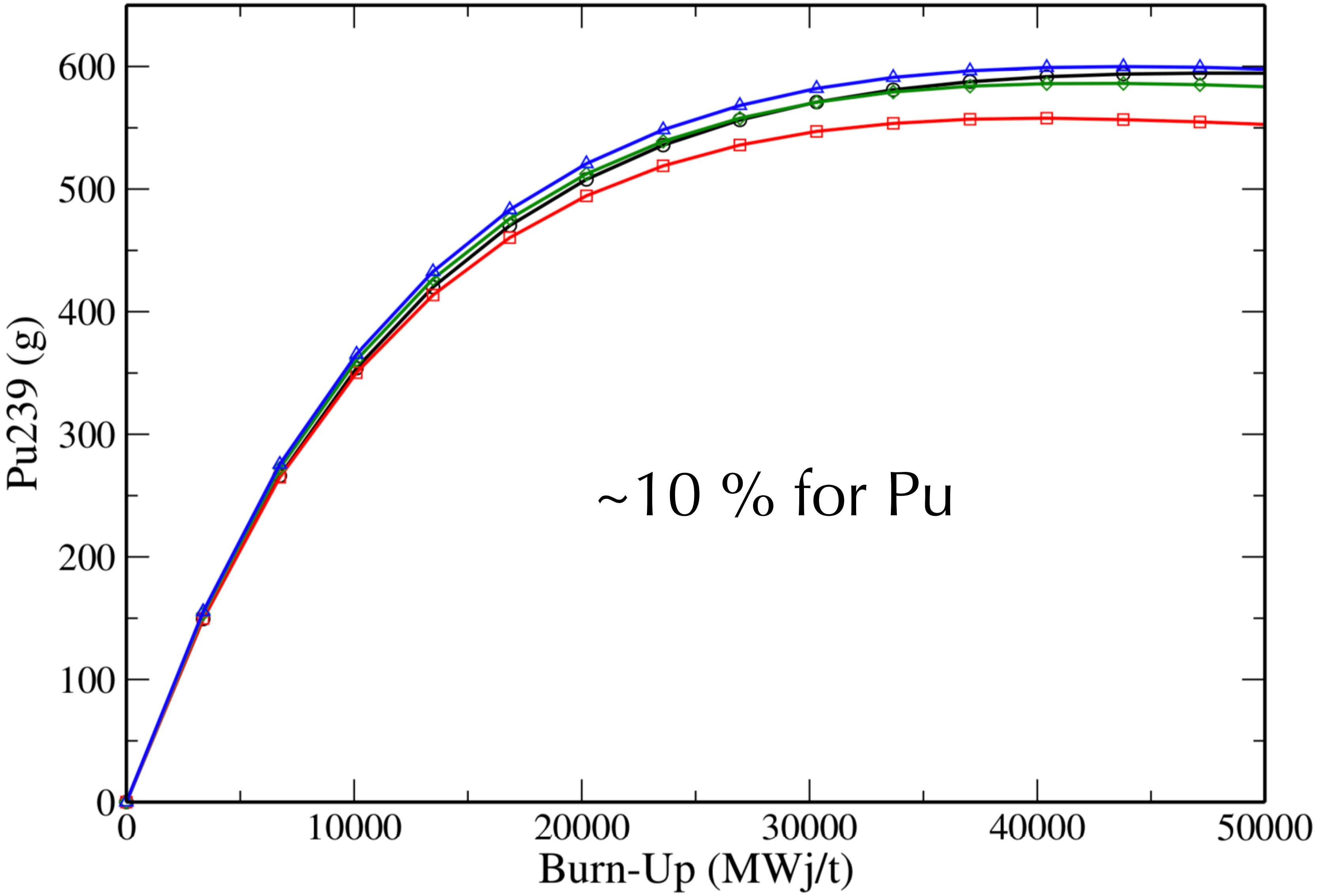


A. Somaini. PhD, IPNO. 2017

Reactor simplifications

- A fuel cycle simulator is using neutronic predictors from reactor simulations
- Infinite assembly
  - Impact of the assembly cross-talk

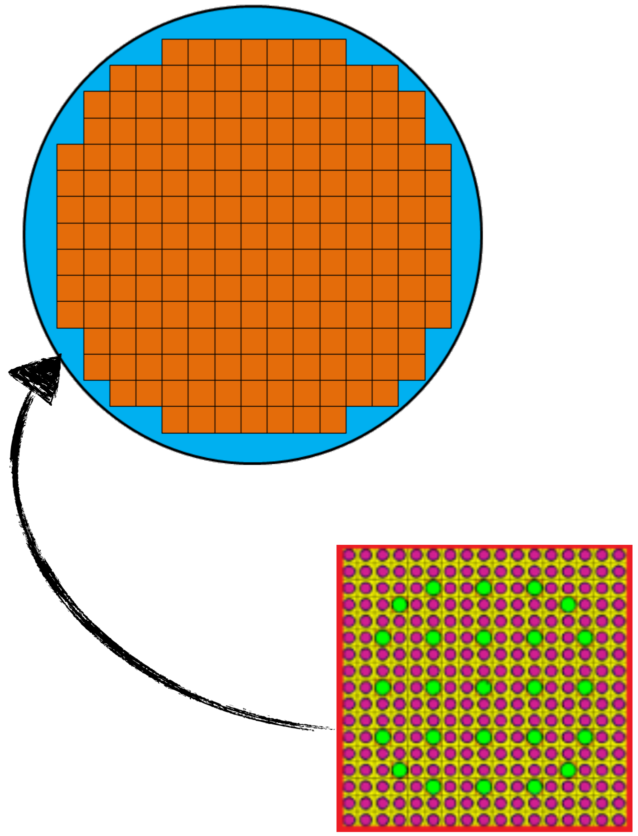


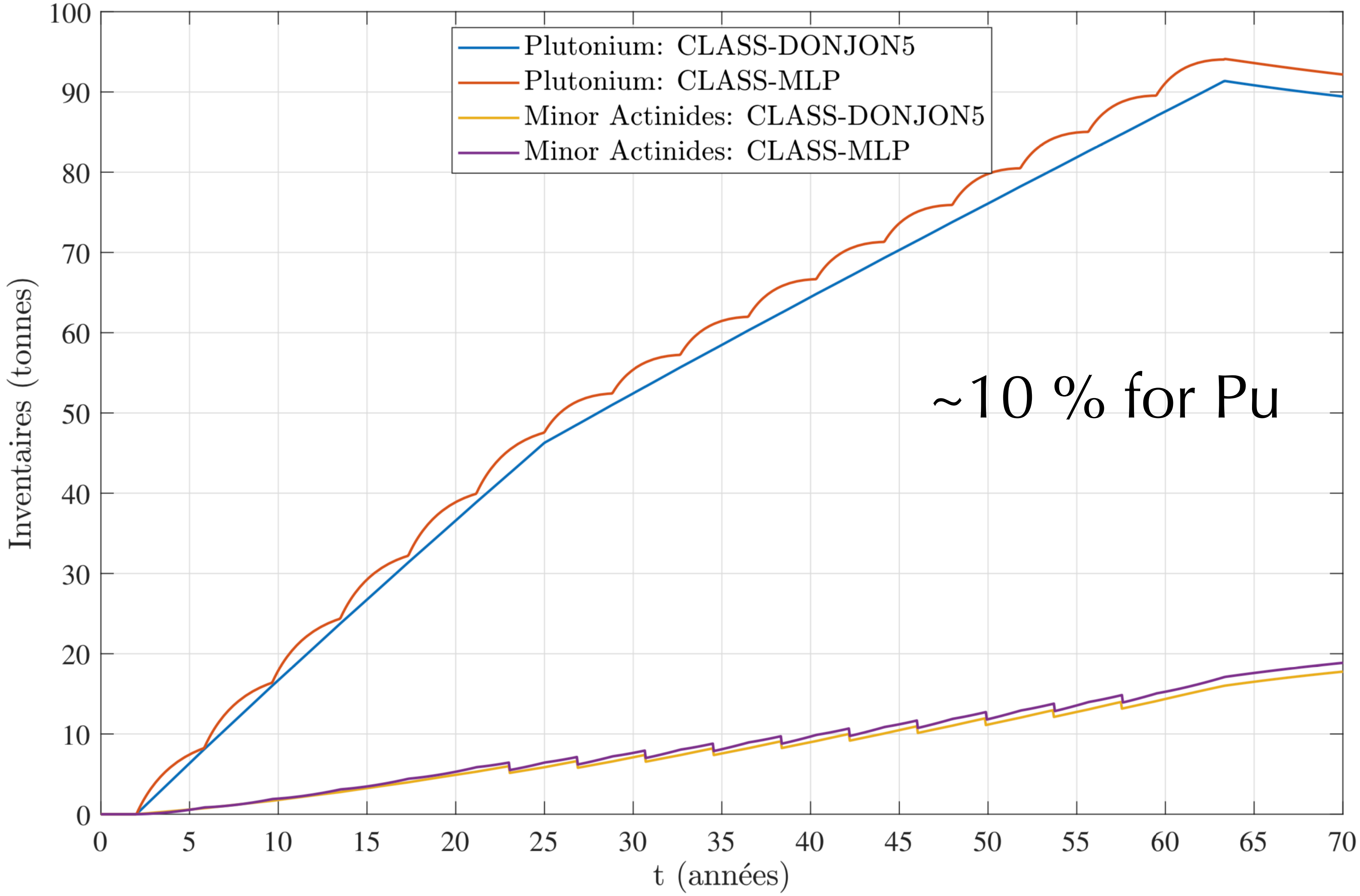


A. Somaini. PhD, IPNO. 2017

Reactor simplifications

- A fuel cycle simulator is using neutronic predictors from reactor simulations
- Core coupling with scenario calculation
  - Transport code : Dragon
  - Core calculation : Donjon
  - Cycle calculation : CLASS





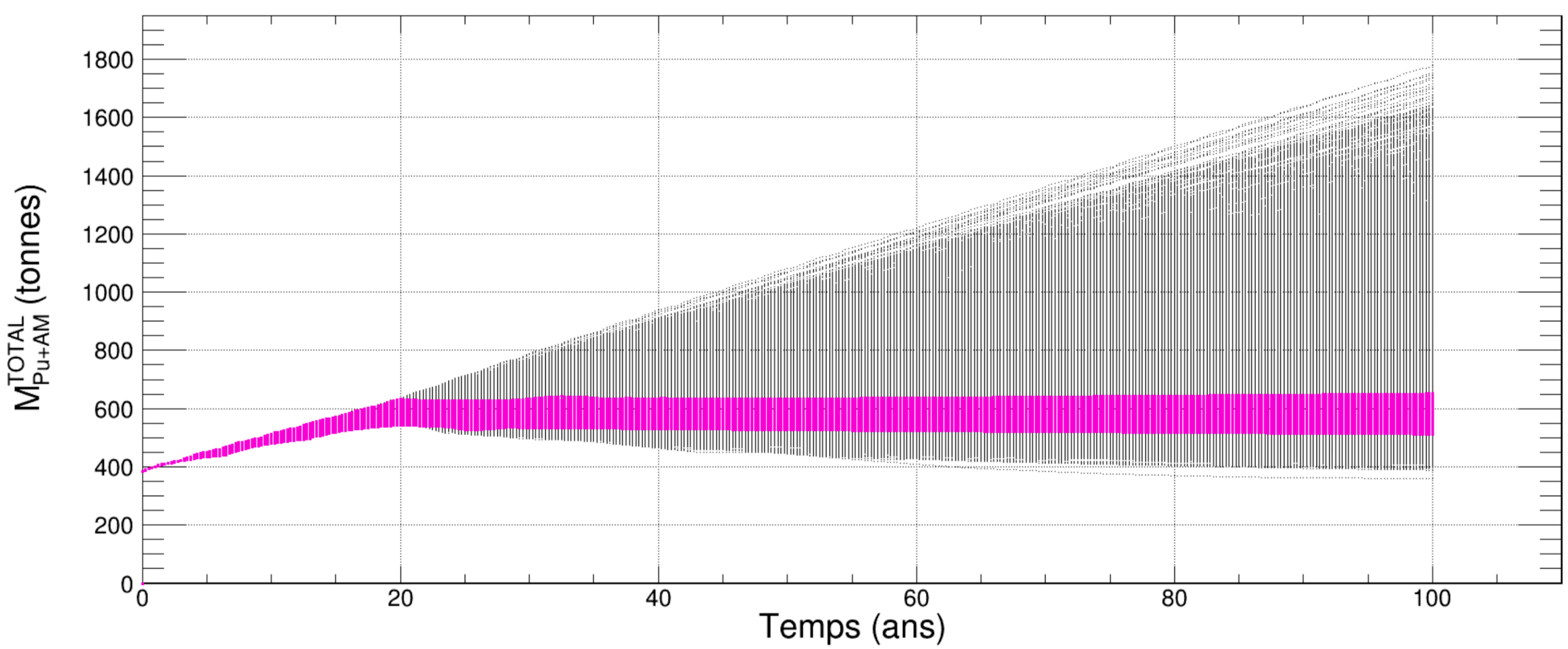
M. Guillet. Master thesis, Polytechnic Montreal, 2019.

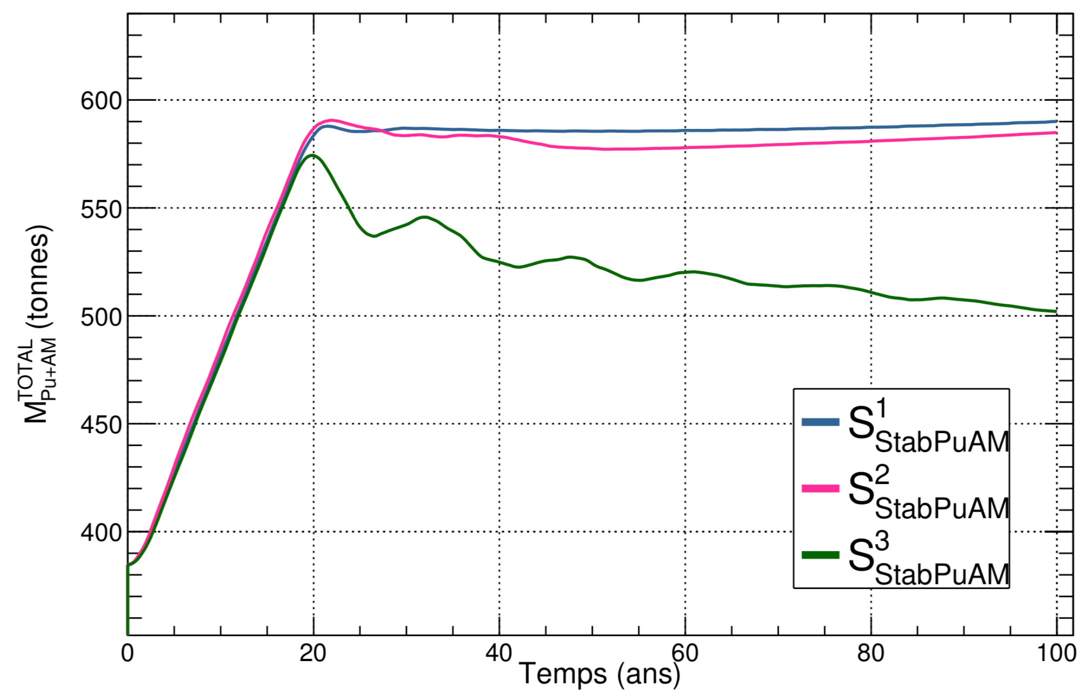
Lot of uncertainties

Fanny Courtin PhD. IMT Atlantique 2017.

- A complex nuclear fleet is composed by many reactors in interaction
  - Very detailed fleet

    VS
  - Few macro reactors





- For a complex french nuclear fleet simulation, detailed fleet has a similar behavior when compared to macro reactors fleet.
- Except for some scenarios which is explained by a lack of plutonium

The FIT project

- A fuel cycle code functionality is the translation into computer software language of a physical or technical process related to nuclear facilities.

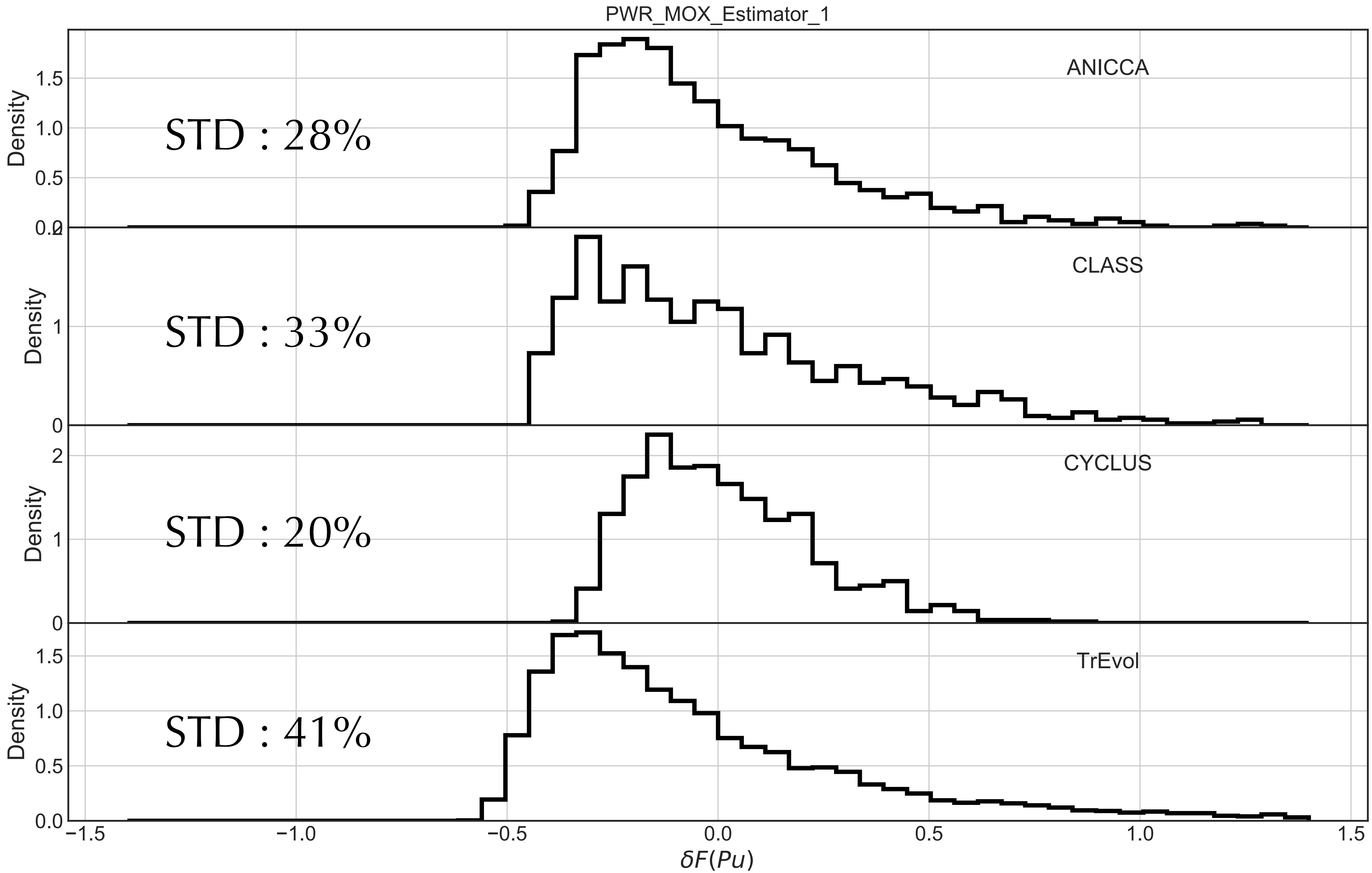
Reference	Functionality to develop
At each reactor loading, the reactor fresh fuel composition is constant	At each reactor loading, the reactor fresh fuel composition depends on available material isotopic composition
The reactor load factor is constant over the reactor lifetime	The reactor load factor takes into account precise industrial constraints, such as partial refueling
The mean cross sections used to perform the fuel evolution in reactor are calculated at BOC and kept constant during the cycle	The mean cross sections used to perform the fuel evolution in reactor are updated according to fuel composition
The reactor first cycles composition is not taken into account and is assumed to be the steady states composition	The exact reactor first cycles composition is used

Table 1: Examples of simplified and more complex functionalities.

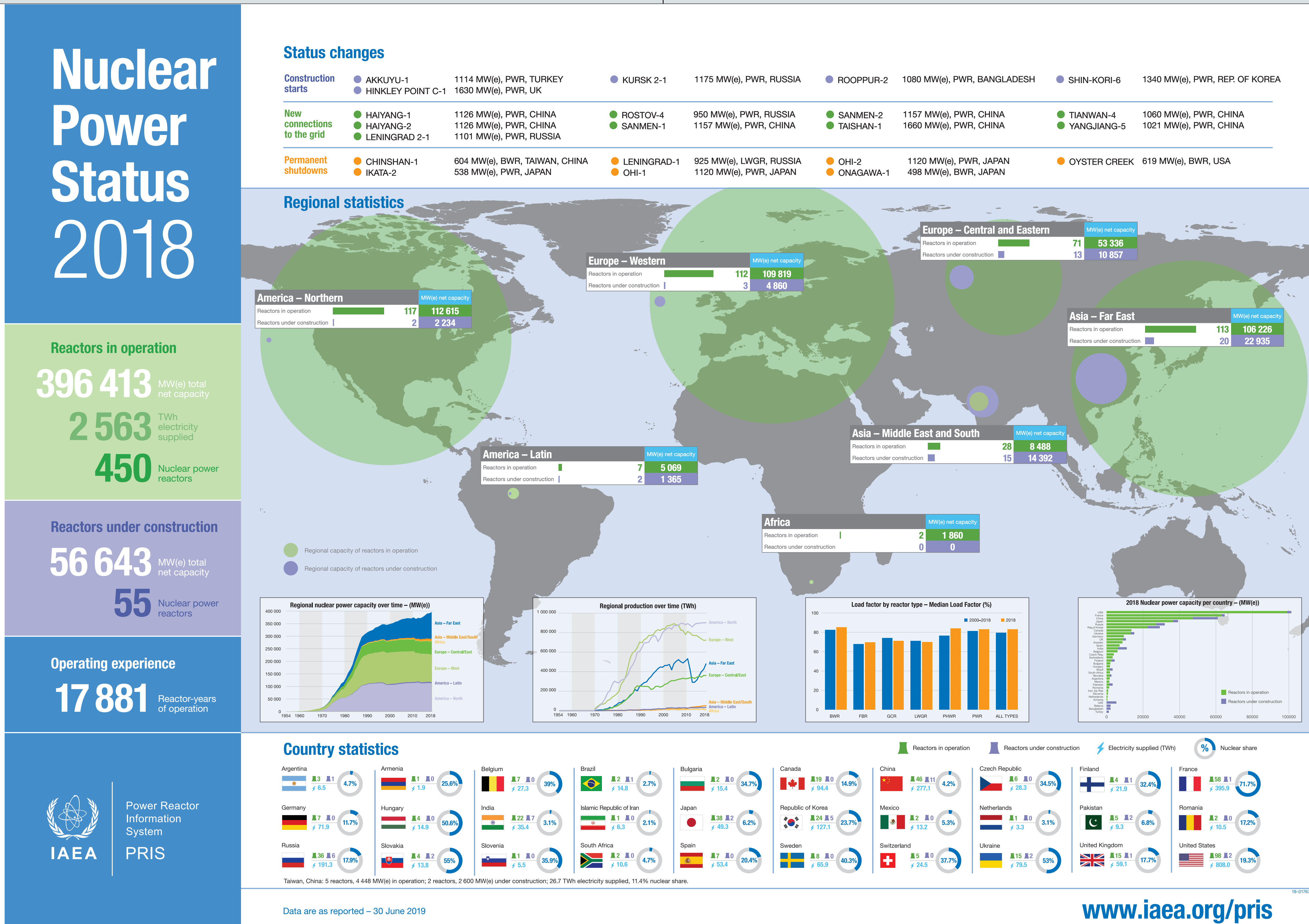
Fresh fuel @ B.O.C.

- FF (Fixed Fraction)
  - Fissile fraction is constant
- FLM (Fuel Loading Model)
  - Reactor requirements
  - Available isotopes stock

The FIT project



The PRIS Database



The PRIS Database

COUNTRIES
- Argentina
- Armenia
- Bangladesh
- Belarus
- Belgium
- Brazil
- Bulgaria
- Canada
- China
- Czech Republic
- Finland
- France
- Germany
- Hungary
- India
- Iran, Islamic Republic of
- Italy
- Japan
- Kazakhstan
- Korea, Republic of
- Lithuania
- Mexico
- Netherlands
- Pakistan

France

SUMMARY

Nuclear Power Reactors

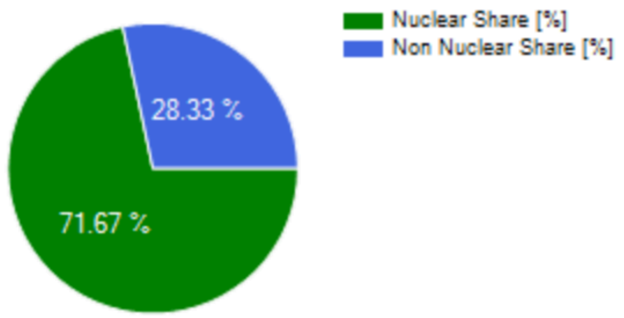
Under Construction	Operational	Long-Term Shutdown	Permanent Shutdown
1	58	0	12

Annual Electrical Power Production

Total Electricity Production (including Nuclear)
548600.00 GW.h
(Net, 2018)

Nuclear Electricity Production
393200.00 GW.h
(Net, 2018)

Electricity Production Share in 2018 Trend



REACTORS

Name	Type	Status	Location	Reference Unit Power [MW]	Gross Electrical Capacity [MW]	First Grid Connection
BELLEVILLE-1	PWR	Operational	LERE	1310	1363	1987-10-14
BELLEVILLE-2	PWR	Operational	LERE	1310	1363	1988-07-06
BLAYAIS-1	PWR	Operational	BRAUD ST.LOUIS	910	951	1981-06-12
BLAYAIS-2	PWR	Operational	BRAUD ST.LOUIS	910	951	1982-07-17
BLAYAIS-3	PWR	Operational	BRAUD ST.LOUIS	910	951	1983-08-17
BLAYAIS-4	PWR	Operational	BRAUD ST.LOUIS	910	951	1983-05-16

The PRIS Database

REACTOR DETAILS

Reactor Type	Model	Owner	Operator
PWR	**CP1**	**ELECTRICITE DE FRANCE**	**ELECTRICITE DE FRANCE**

Reference Unit Power (Net Capacity)	Design Net Capacity	Gross Capacity	Thermal Capacity
910 MW_e	**910 MW_e**	**951 MW_e**	**2785 MW_t**

Construction Start Date	First Criticality Date
01 Jan, 1977	**20 May, 1981**

First Grid Connection	Commercial Operation Date
12 Jun, 1981	**01 Dec, 1981**

LIFETIME PERFORMANCE

Electricity Supplied	Energy Availability Factor	Operation Factor	Energy Unavailability Factor	Load Factor
215.97 TW.h	**76.6 %**	**77.9 %**	**23.4 %**	**72.7 %**

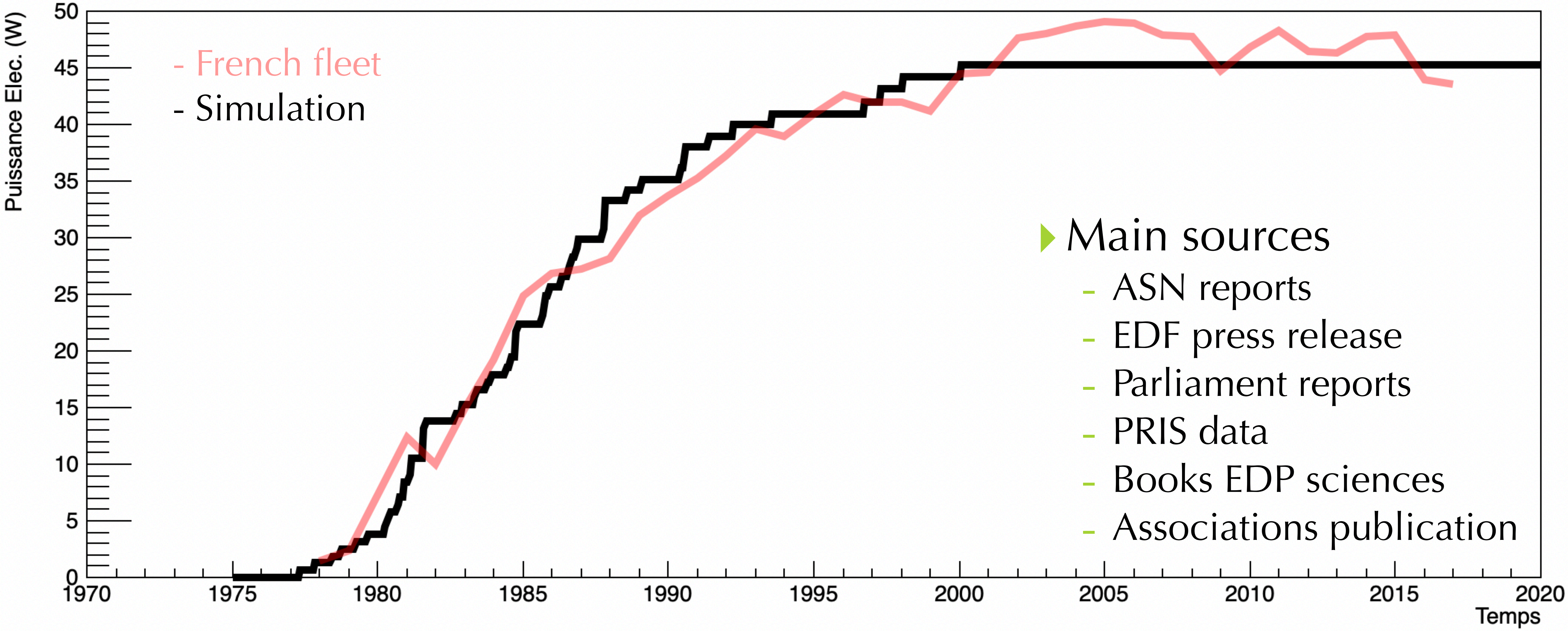
Lifetime performance calculated up to year 2018

OPERATING HISTORY

Year	Electricity Supplied [GW.h]	Reference Unit Power [MW]	Annual Time On Line [h]	Operation Factor [%]	Energy Availability Factor [%]		Load Factor [%]	
					Annual	Cumulative	Annual	Cumulative
1981	1636.200	920	2584	70.2	65.6	65.6	65.6	65.6
1982	6129.800	910	7588	86.6	81.5	80.2	76.9	76.0
1983	3453.000	910	4285	48.9	43.9	62.8	43.3	60.3

French fleet history

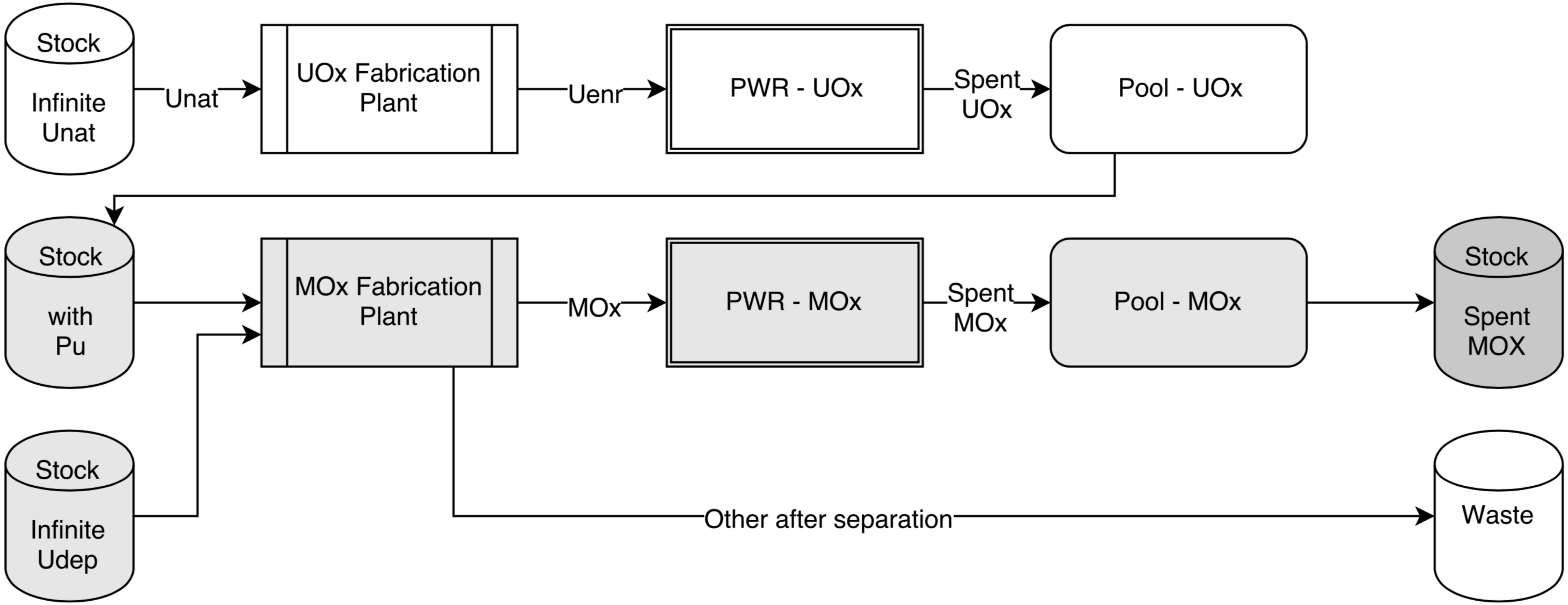
- Goals
  - Centralize available data
  - Get the initial conditions
  - Compare with available data
- Lack of data
  - Reactor deployment
  - First loadings
  - Discharge burn-up
  - Fuel management strategy



- Main sources
  - ASN reports
  - EDF press release
  - Parliament reports
  - PRIS data
  - Books EDP sciences
  - Associations publication

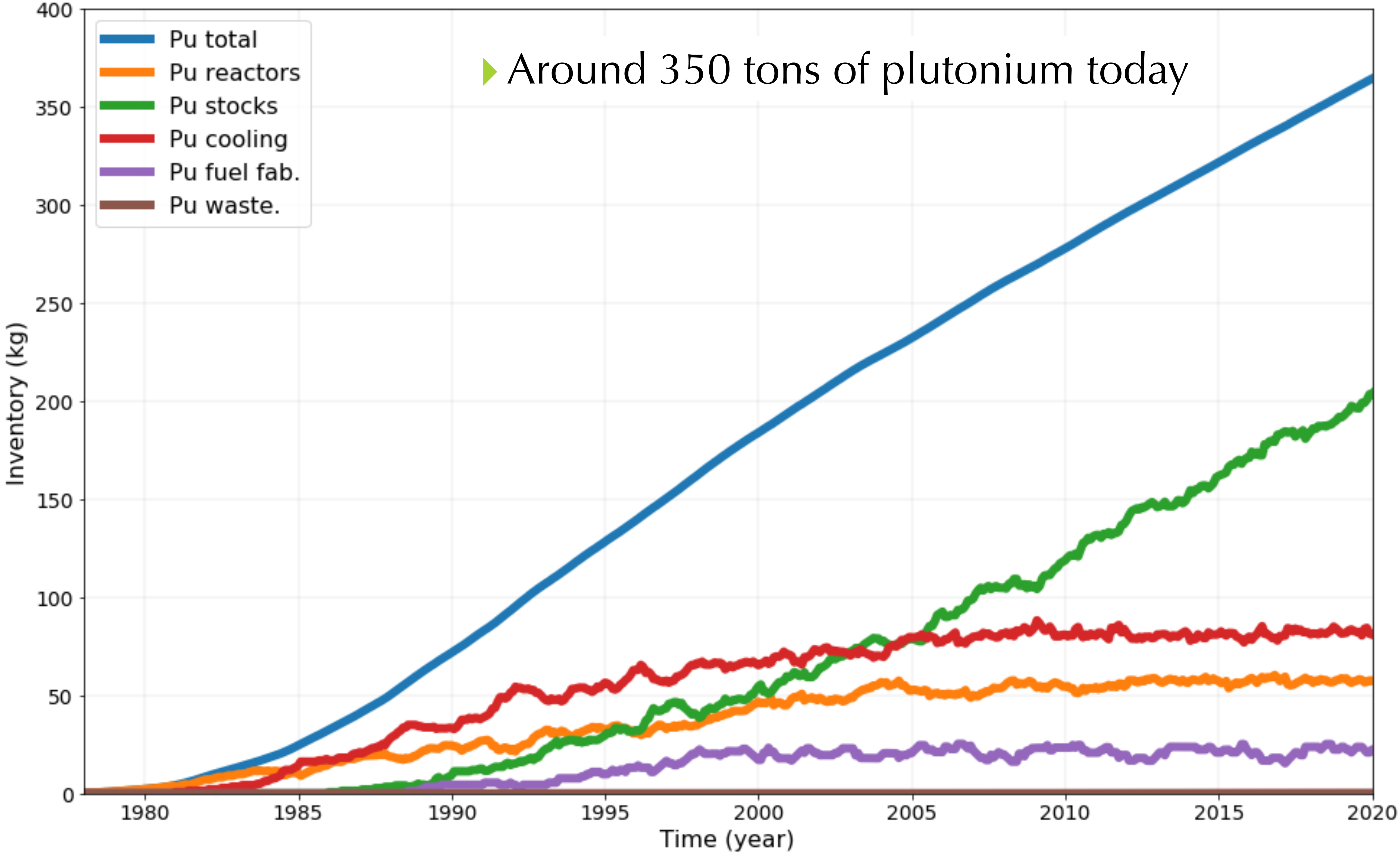
French fleet simulation

- French fleet schematic view with two fuel management stages
  - The UOX level
  - The MOX level
  - Nuclear waste

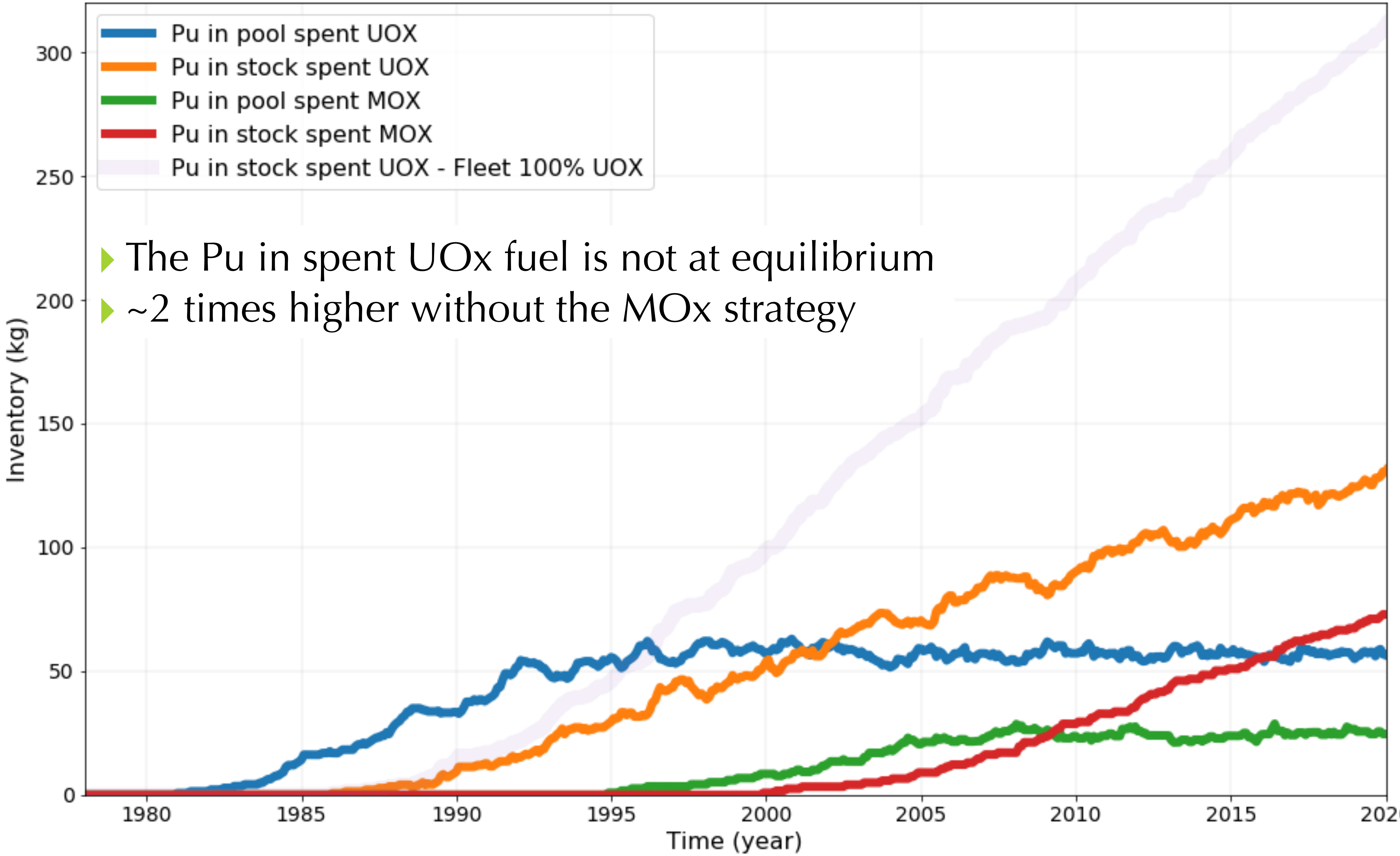


Plutonium inventory

- Around 350 tons of plutonium today

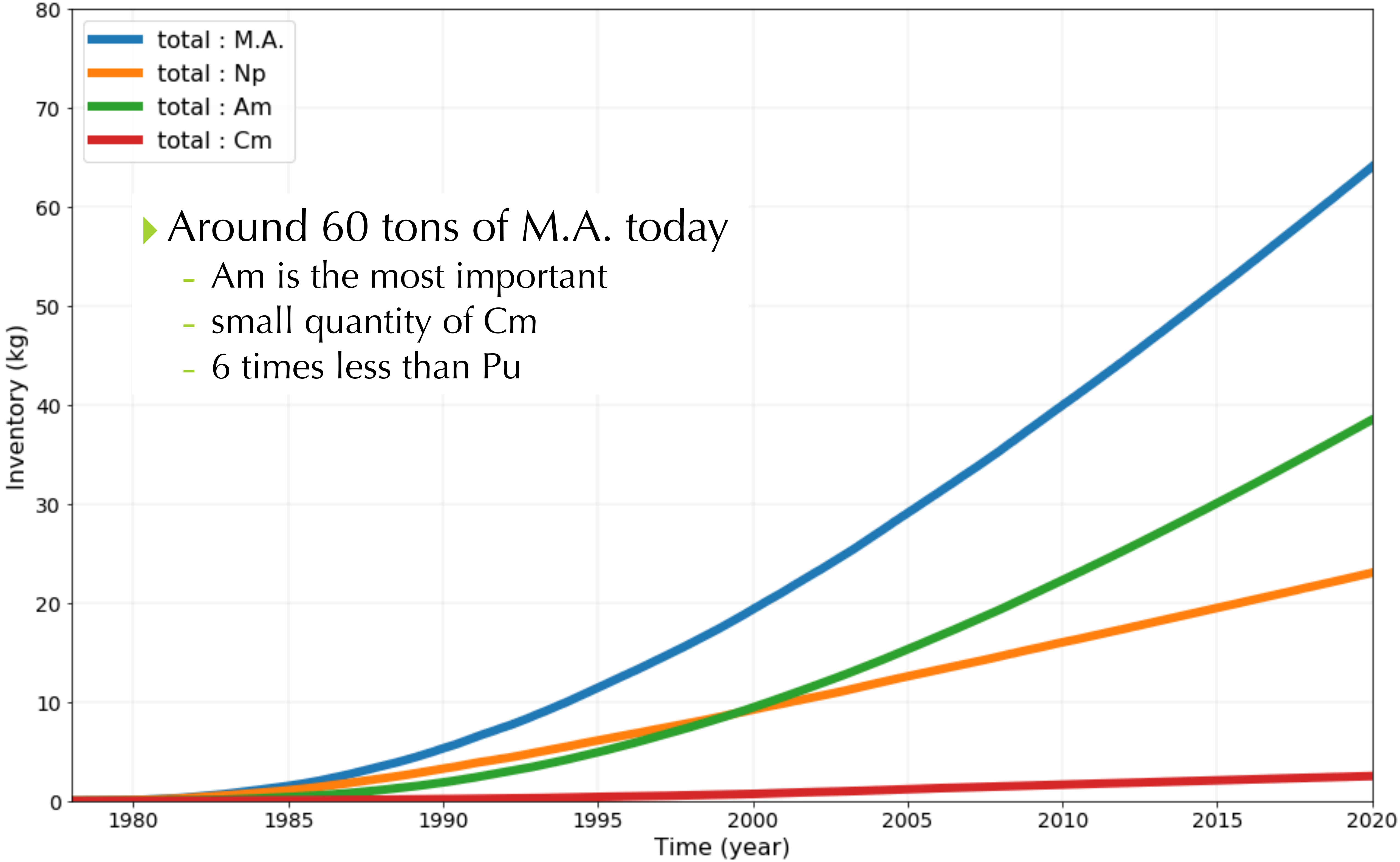


Pu stocks and pools



- The Pu in spent UOx fuel is not at equilibrium
- ~2 times higher without the MOx strategy

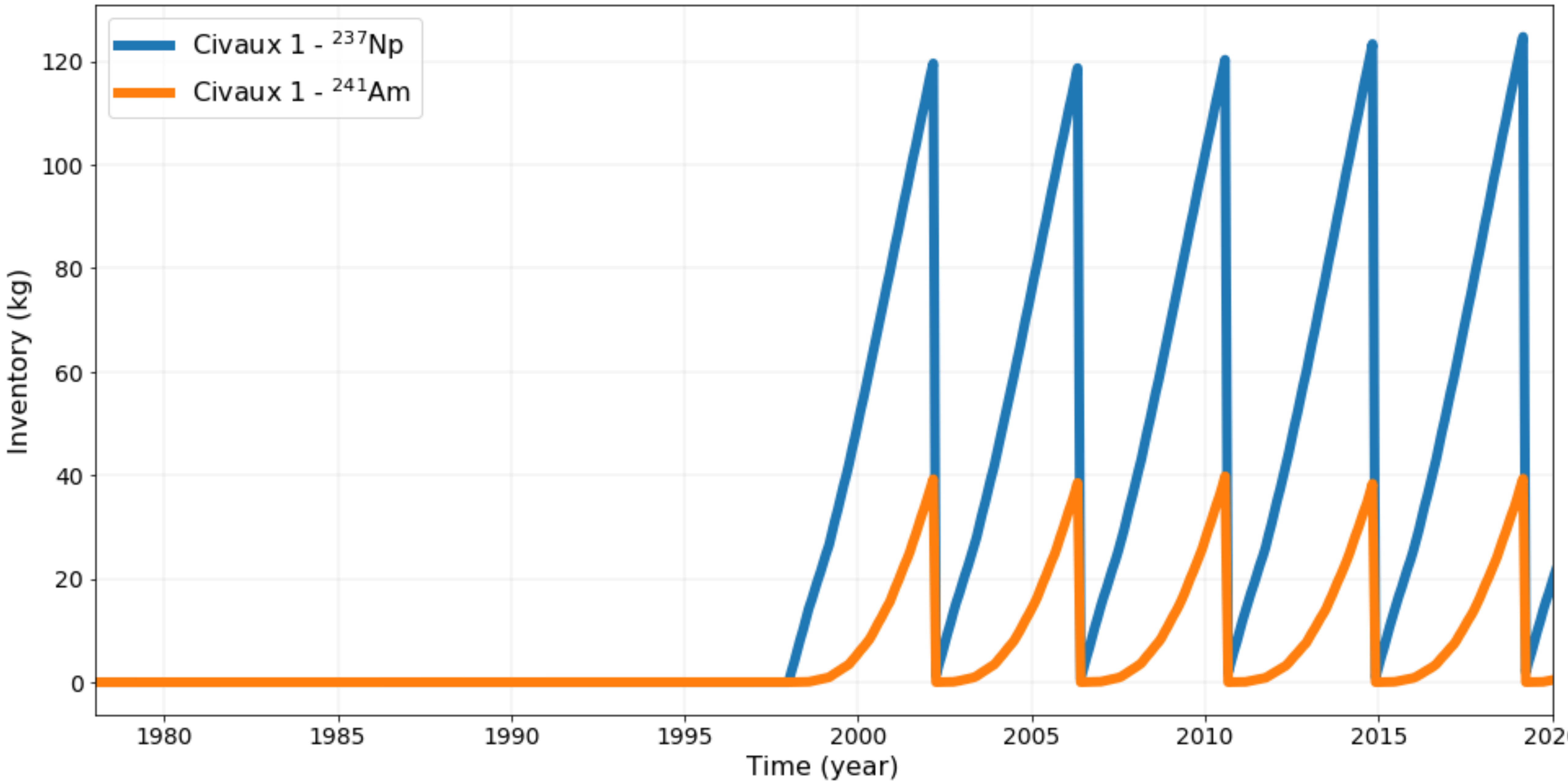
Minor Actinides inventory

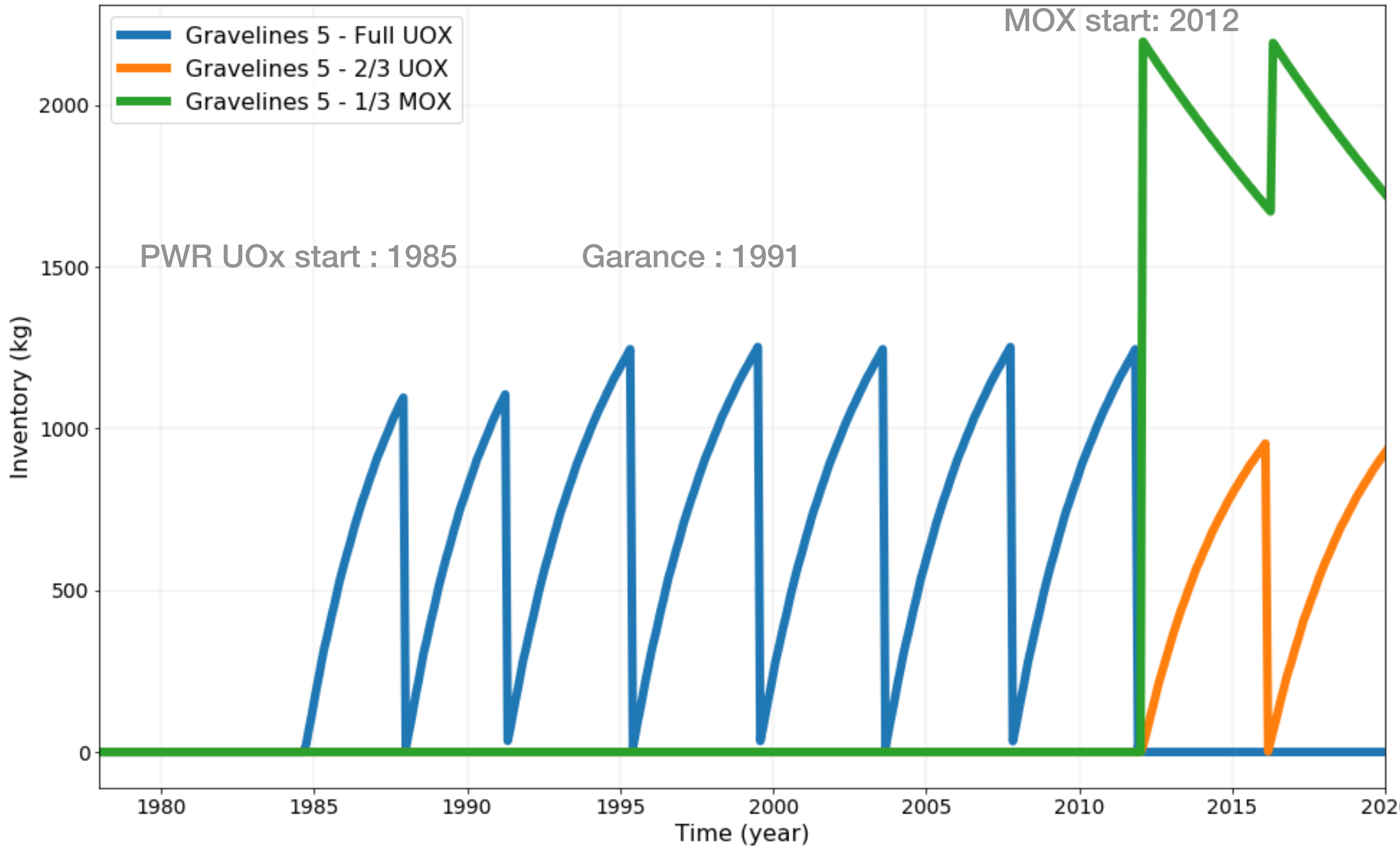


- Around 60 tons of M.A. today
  - Am is the most important
  - small quantity of Cm
  - 6 times less than Pu

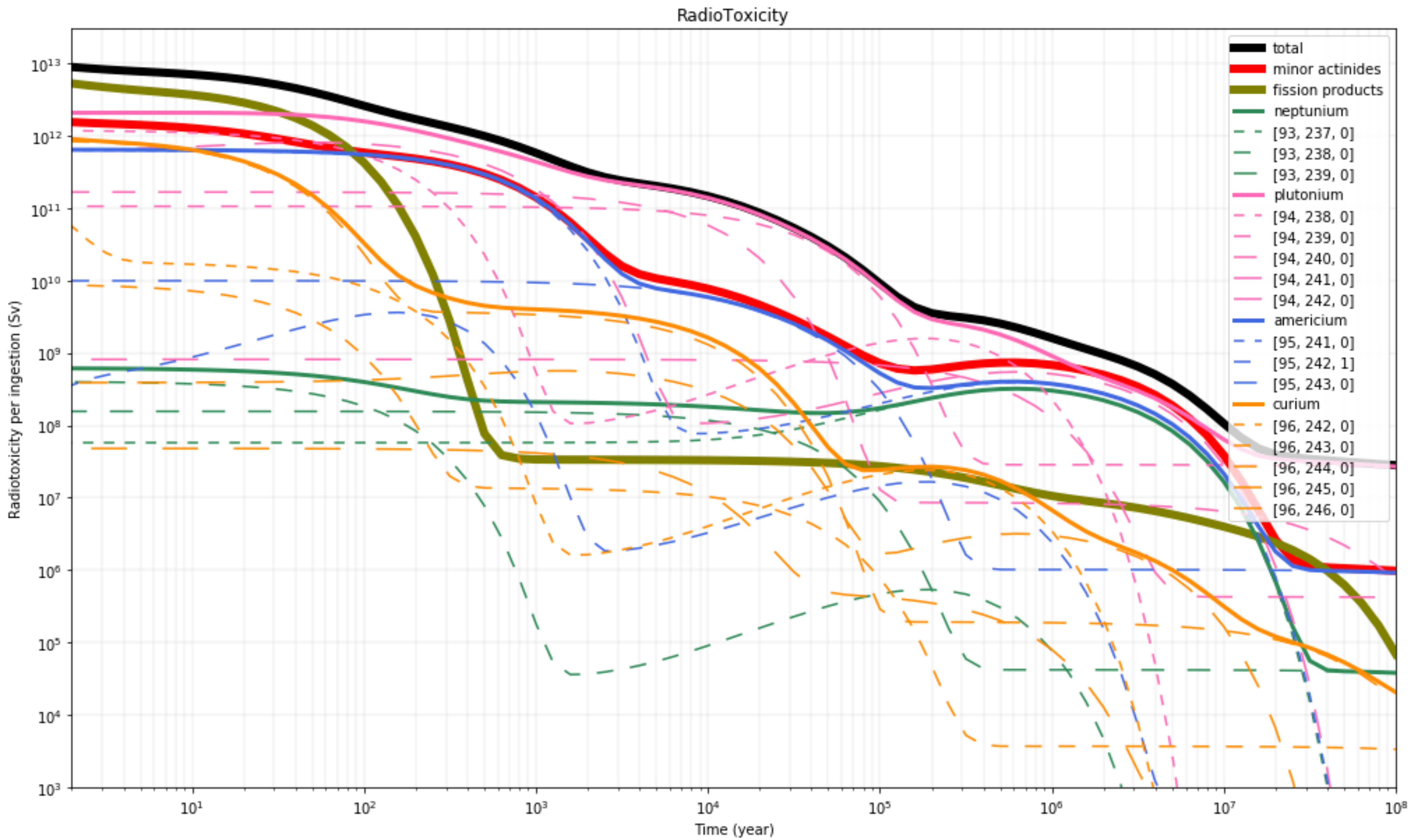
Comparison with Americium

- ^{241}Am production is smaller than ^{237}Np production in reactor
  - ^{241}Am is mainly produced outside from the reactor by ^{241}Pu decay
  - ^{241}Am increases during fuel cycle evolution




Gravelines 5 - Full UOX
Gravelines 5 - 2/3 UOX
Gravelines 5 - 1/3 MOX
MOX start: 2012
PWR UOx start : 1985
Garance : 1991
Inventory (kg)
Time (year)
0
500
1000
1500
2000
1980
1985
1990
1995
2000
2005
2010
2015
2020

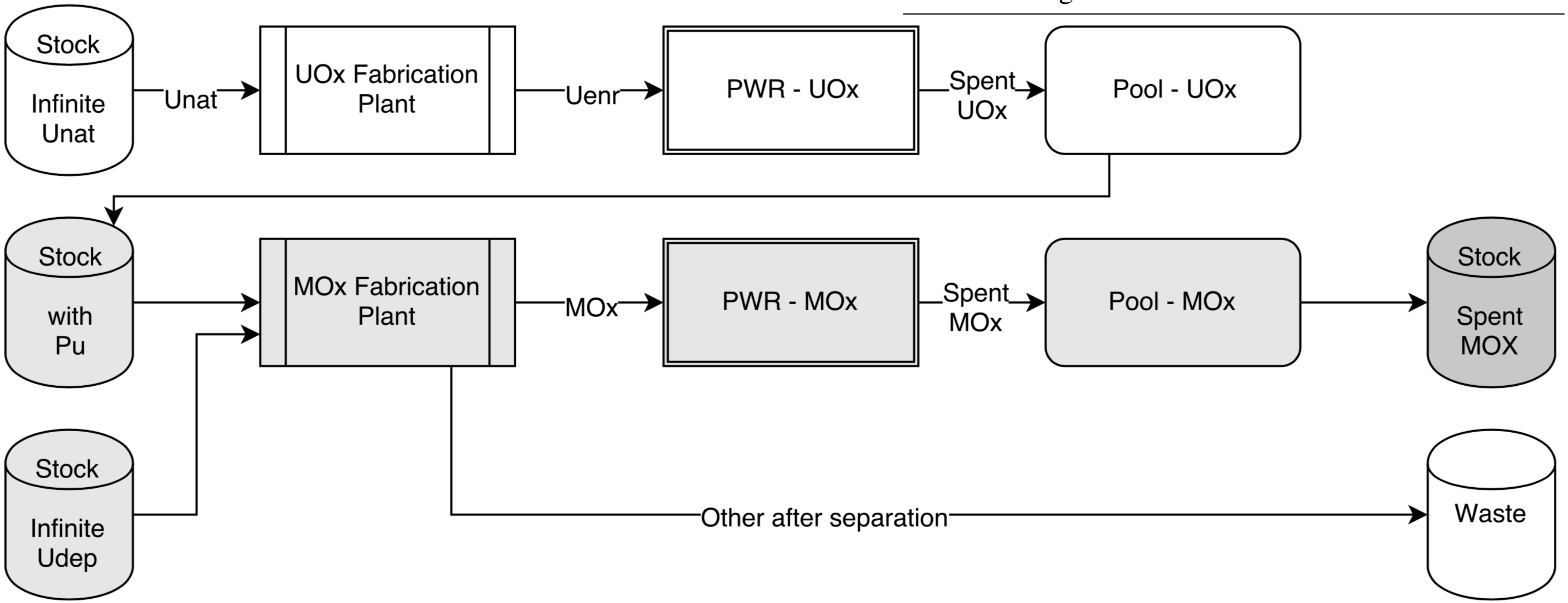
Radiotoxicity of total inventory @ 2015



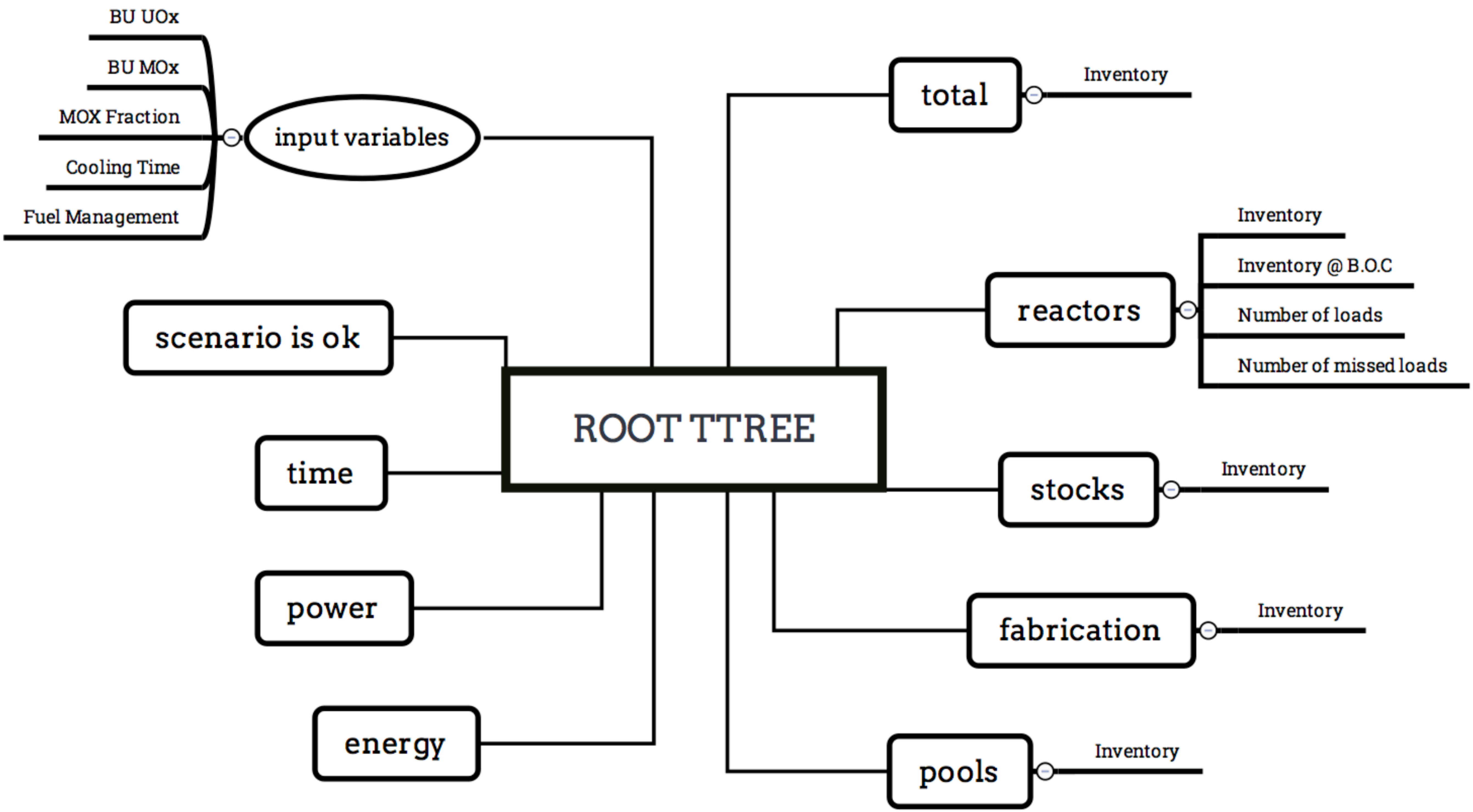
Wide Sweeping Methodology

- Global sensitivity analyse framework
  - Wide design of experiment
  - High number of simulations
  - Macro reactors simulation

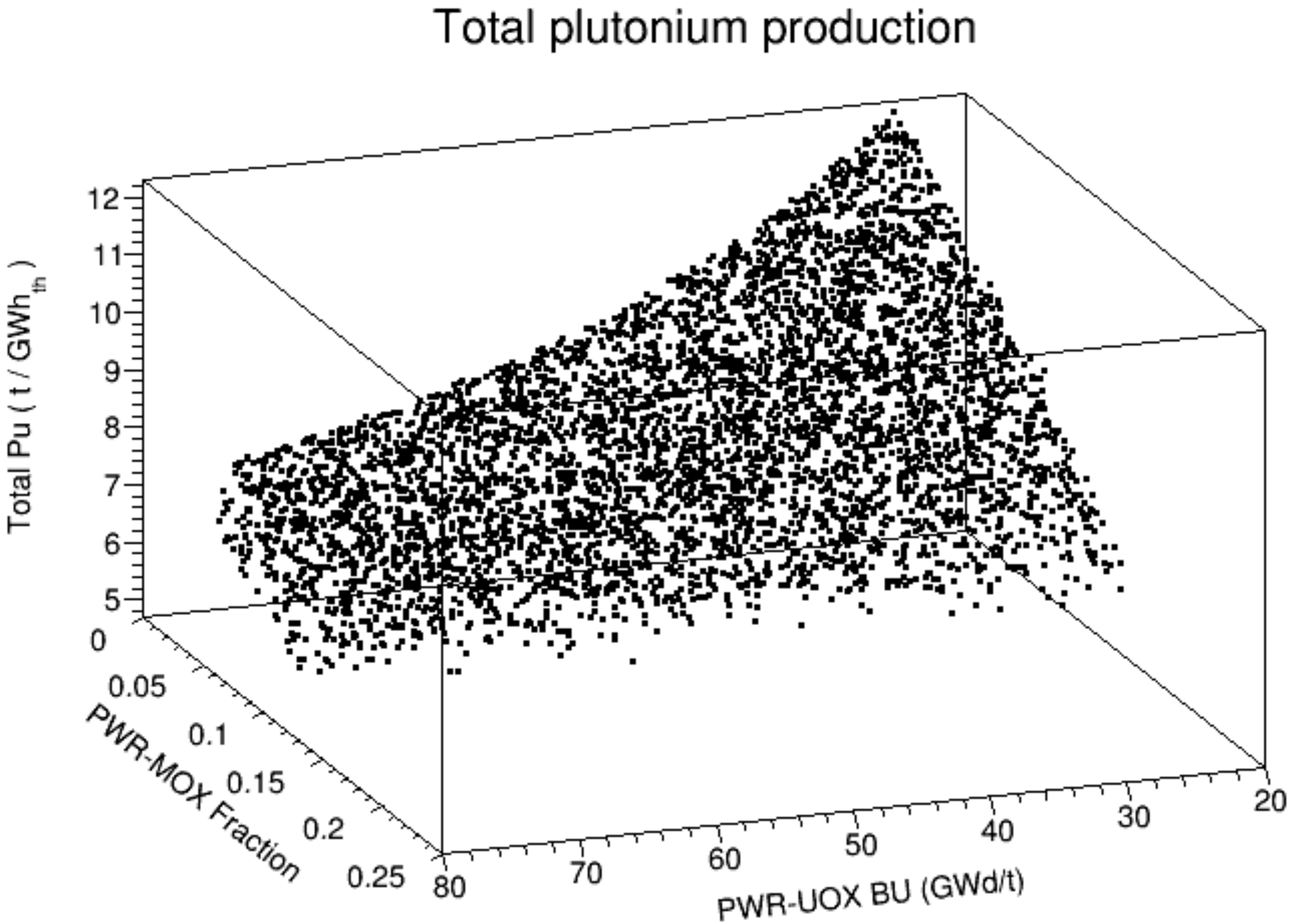
Input Data	Min. Value	Max. Value
PWR-UOx BU [GWd/t]	30	60
PWR-MOx BU [GWd/t]	30	60
PWR-MOx Fraction	0	0.20
Pool Cooling time (y)	0	20
Stock Management	FiFo / LiFo	



Data storage

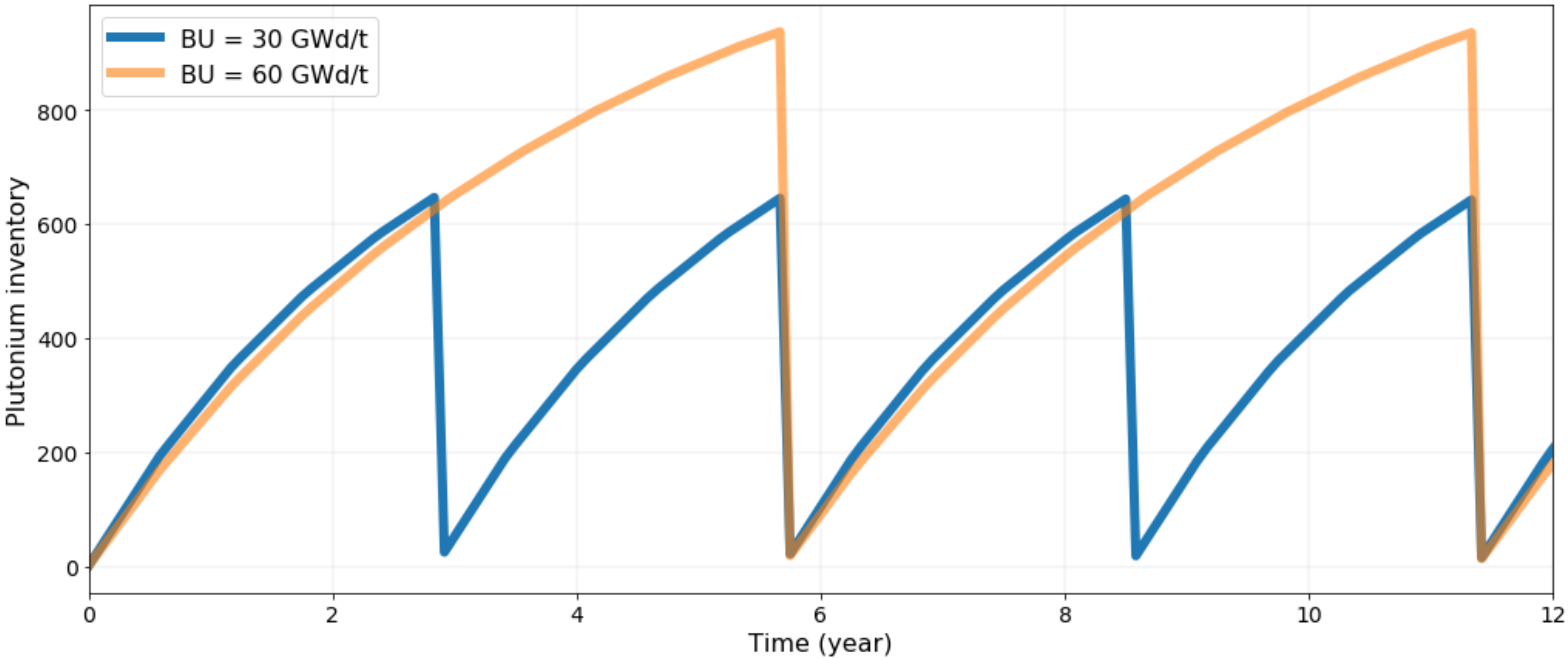


Plutonium inventory @ EOS



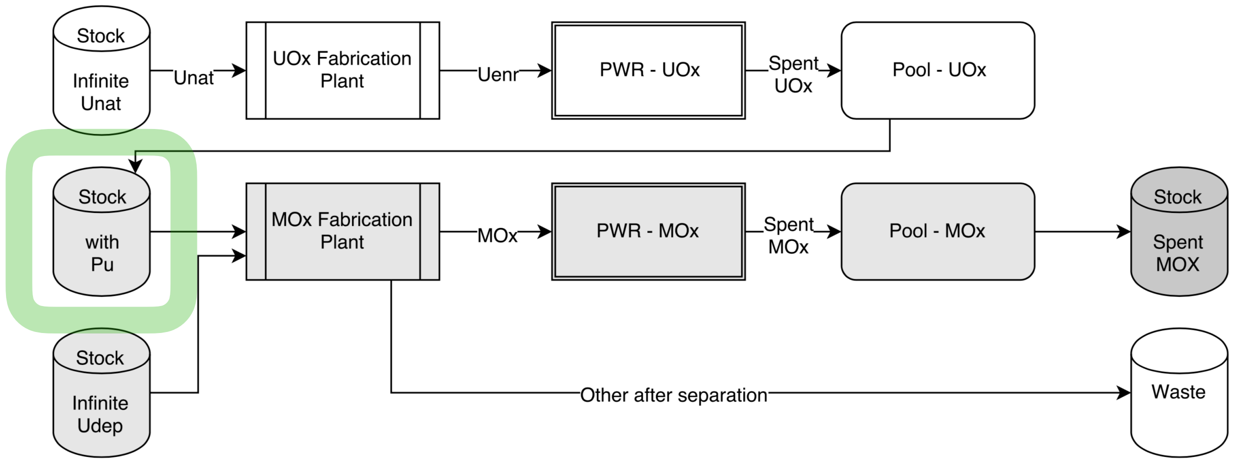
Pu dependency with UOX BU

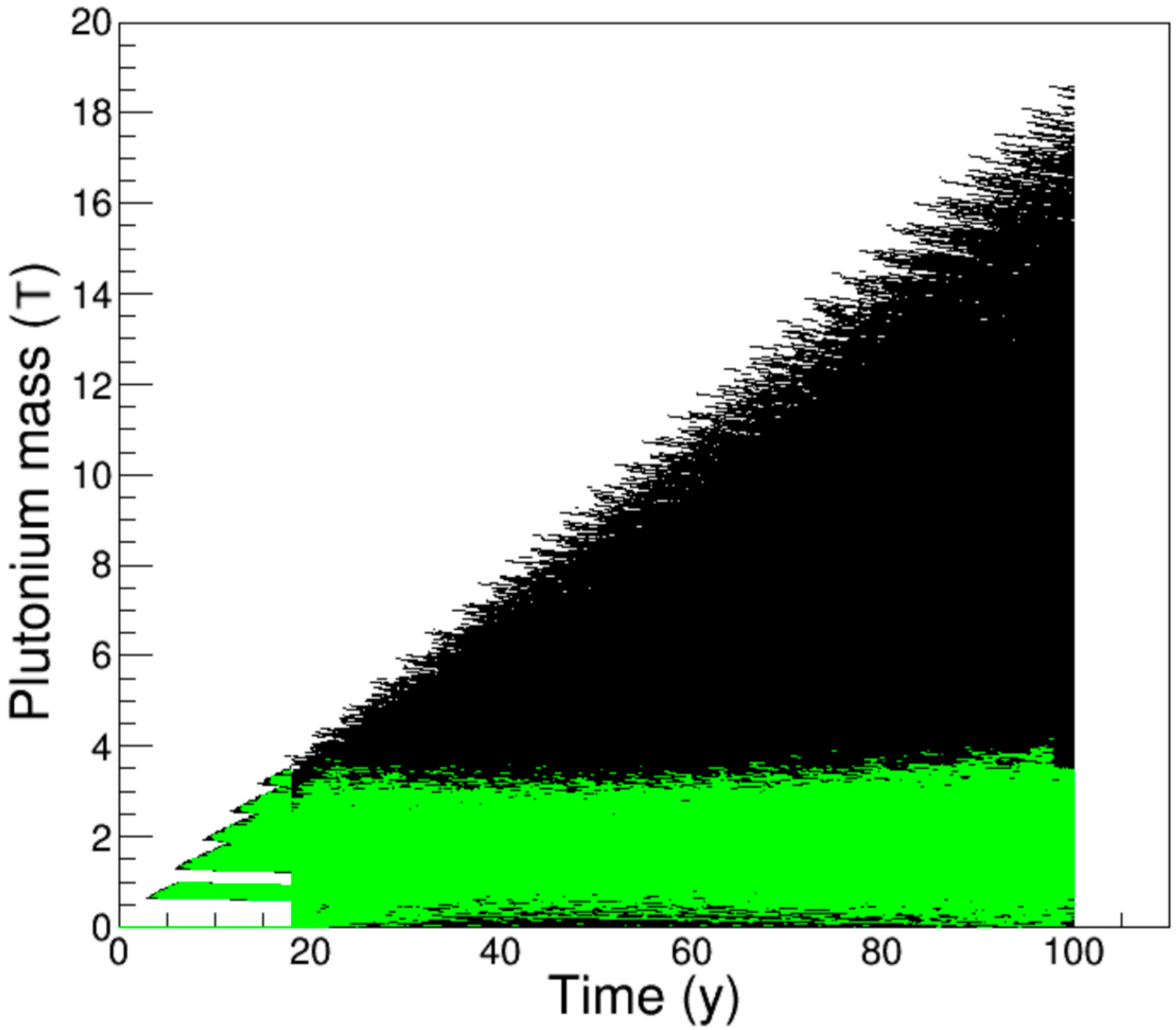
- Pu production is not linear with PWR UOX burn-up
  - High for small irradiation time
  - Smaller for high burn-up
- Pu production rate is higher for small PWR UOX burn-up

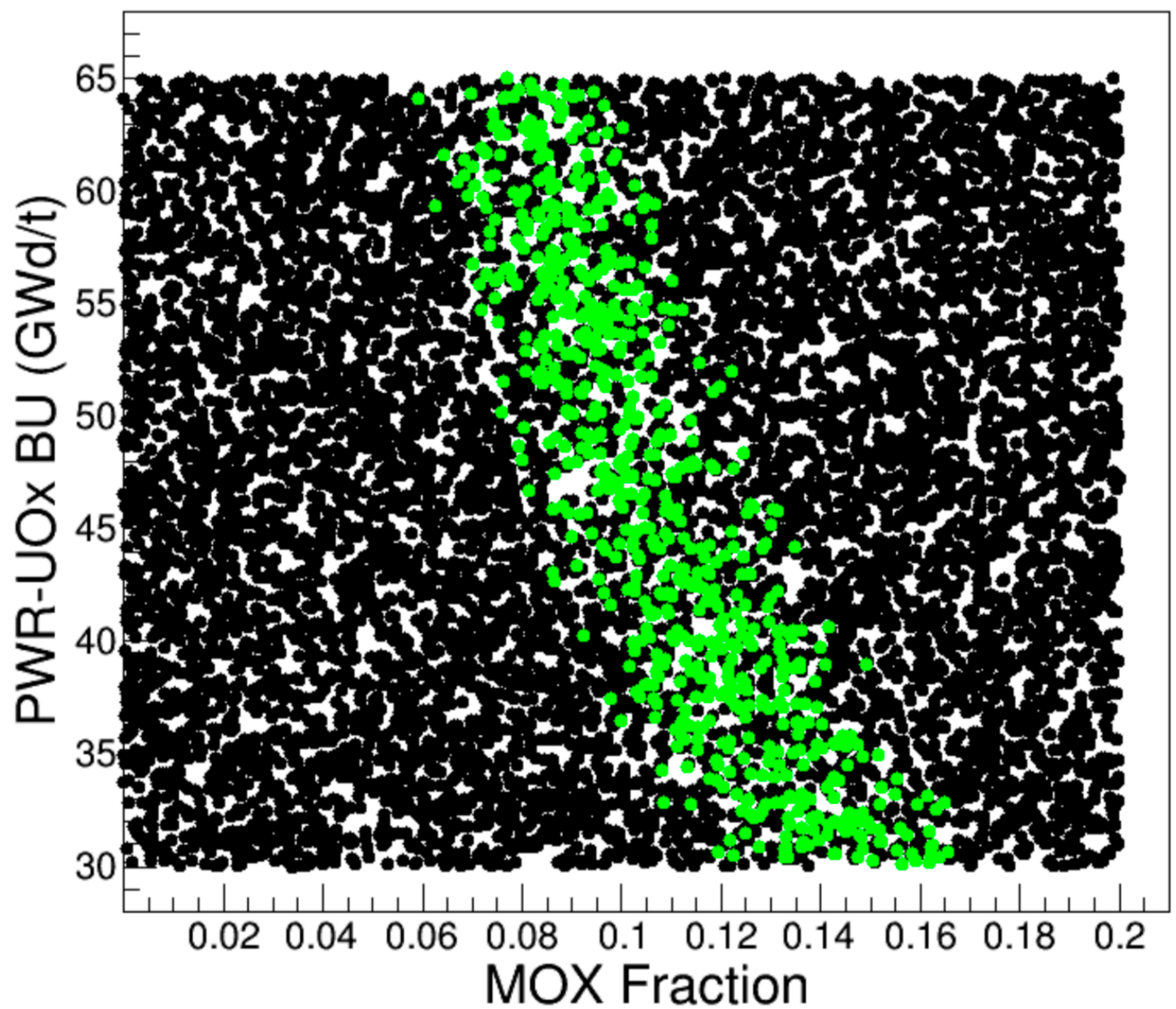


Plutonium equilibrium

- Limitation of a MonoMOx fuel cycle
  - Plutonium availability in stock
  - Plutonium accumulation in spent MOx

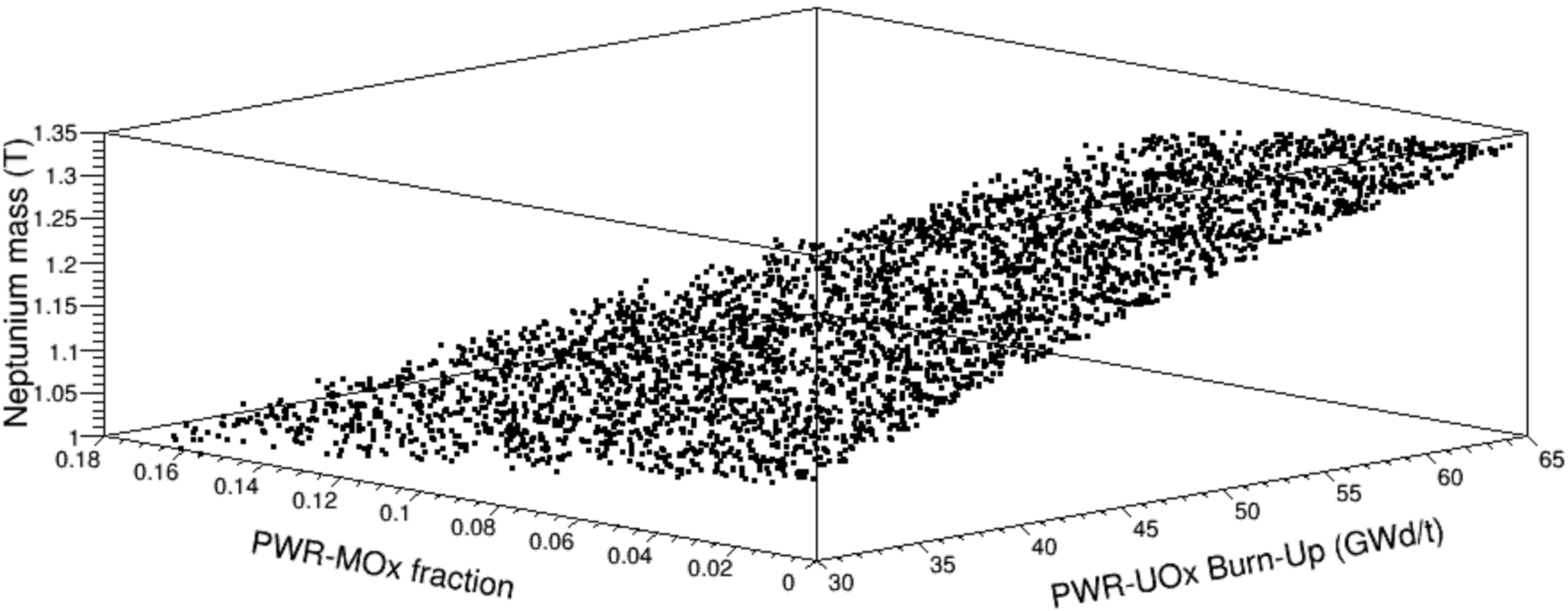






Neptunium production

- Neptunium production mainly depends on two variables
  - Production increases with PWR-UOx burn-up
  - Production decreases with PWR-MOx fraction

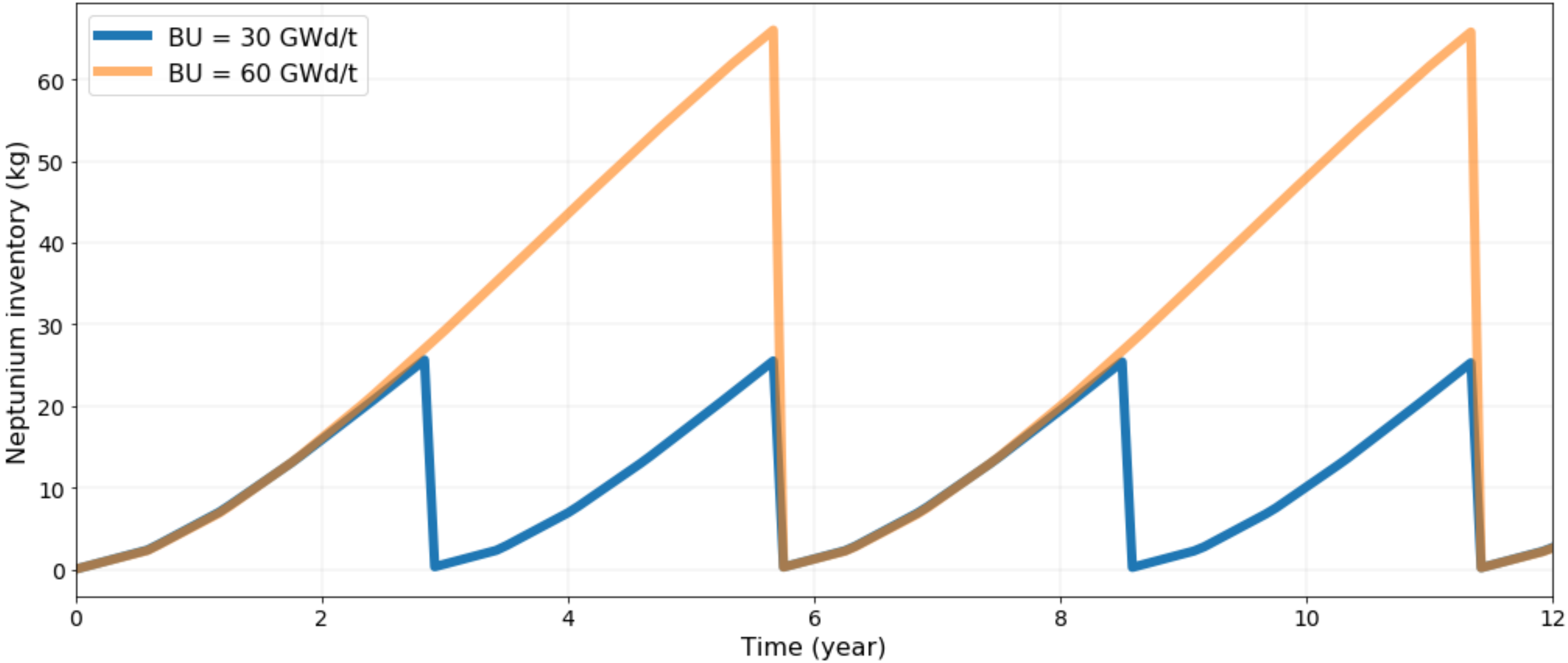


Neptunium production

$$^{235}U(n,\gamma) \rightarrow {}^{236}U(n,\gamma) \rightarrow {}^{237}U \xrightarrow[6.7d]{} {}^{237}Np$$

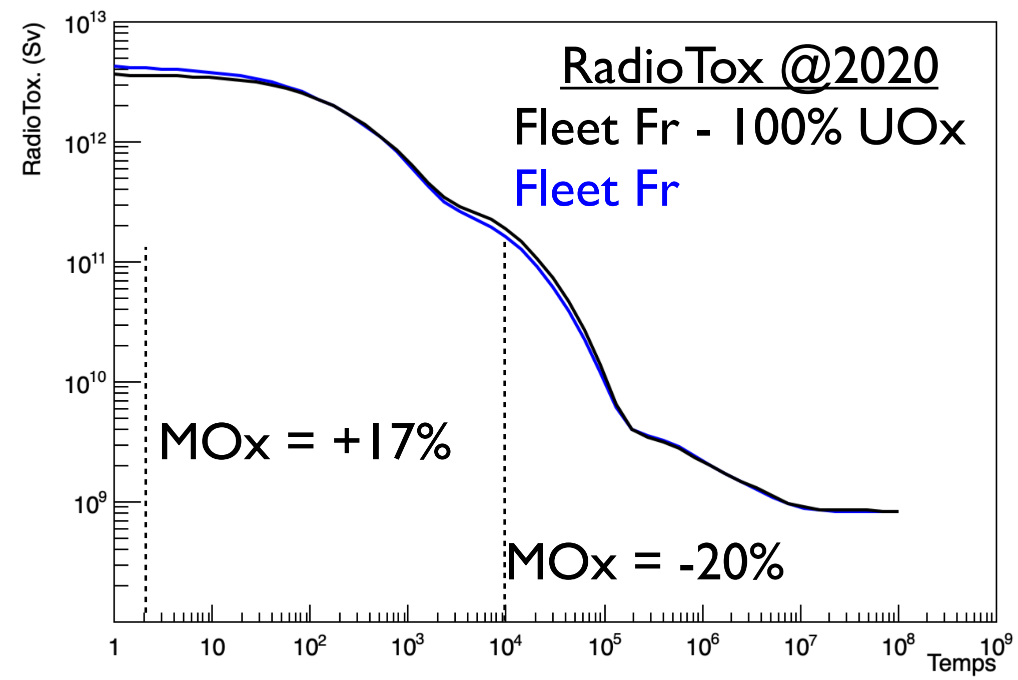
$$^{238}U(n,2n) \rightarrow {}^{237}U \xrightarrow[6.7d]{} {}^{237}Np$$

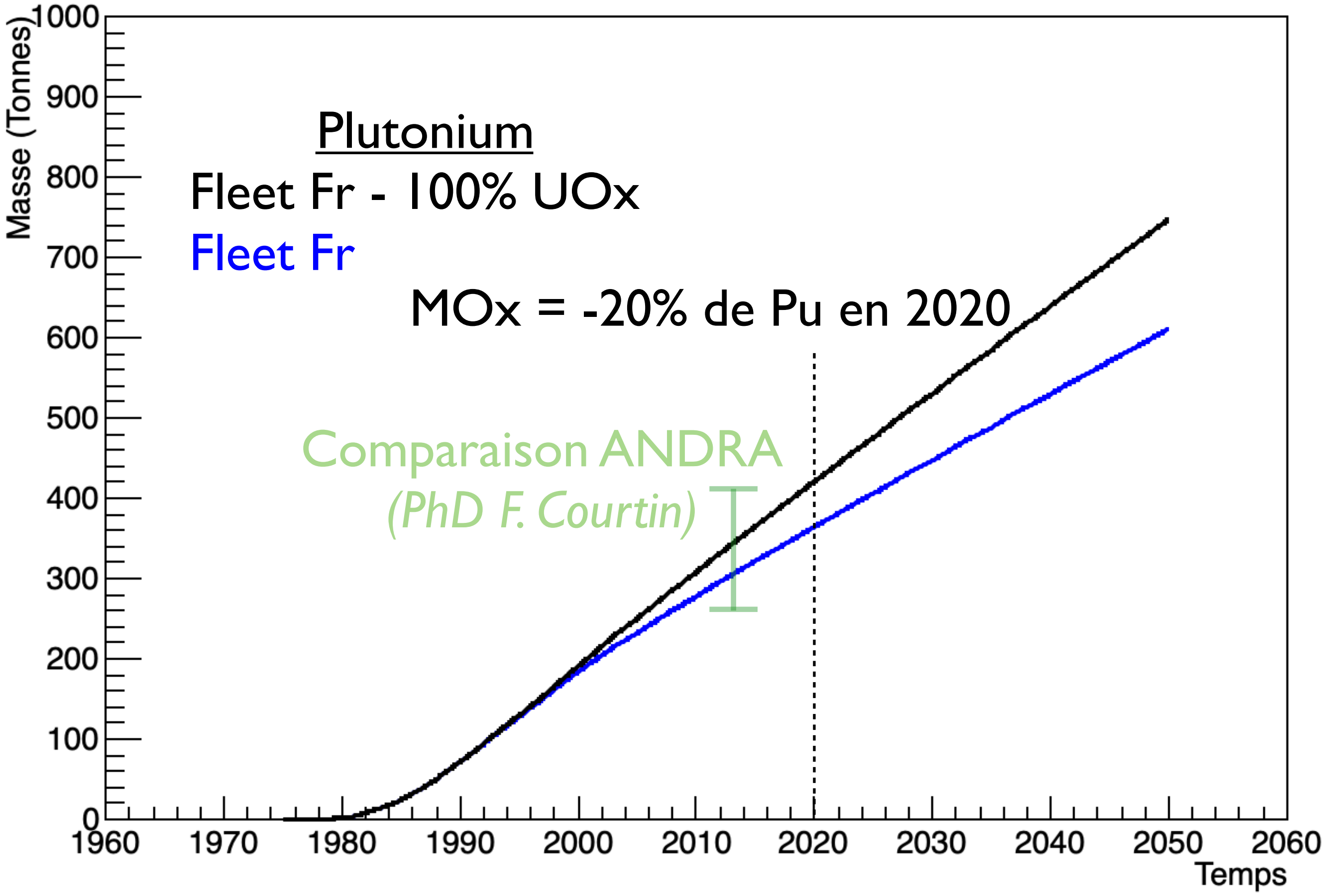
- 2 production pathways
- Production rate increases with BU

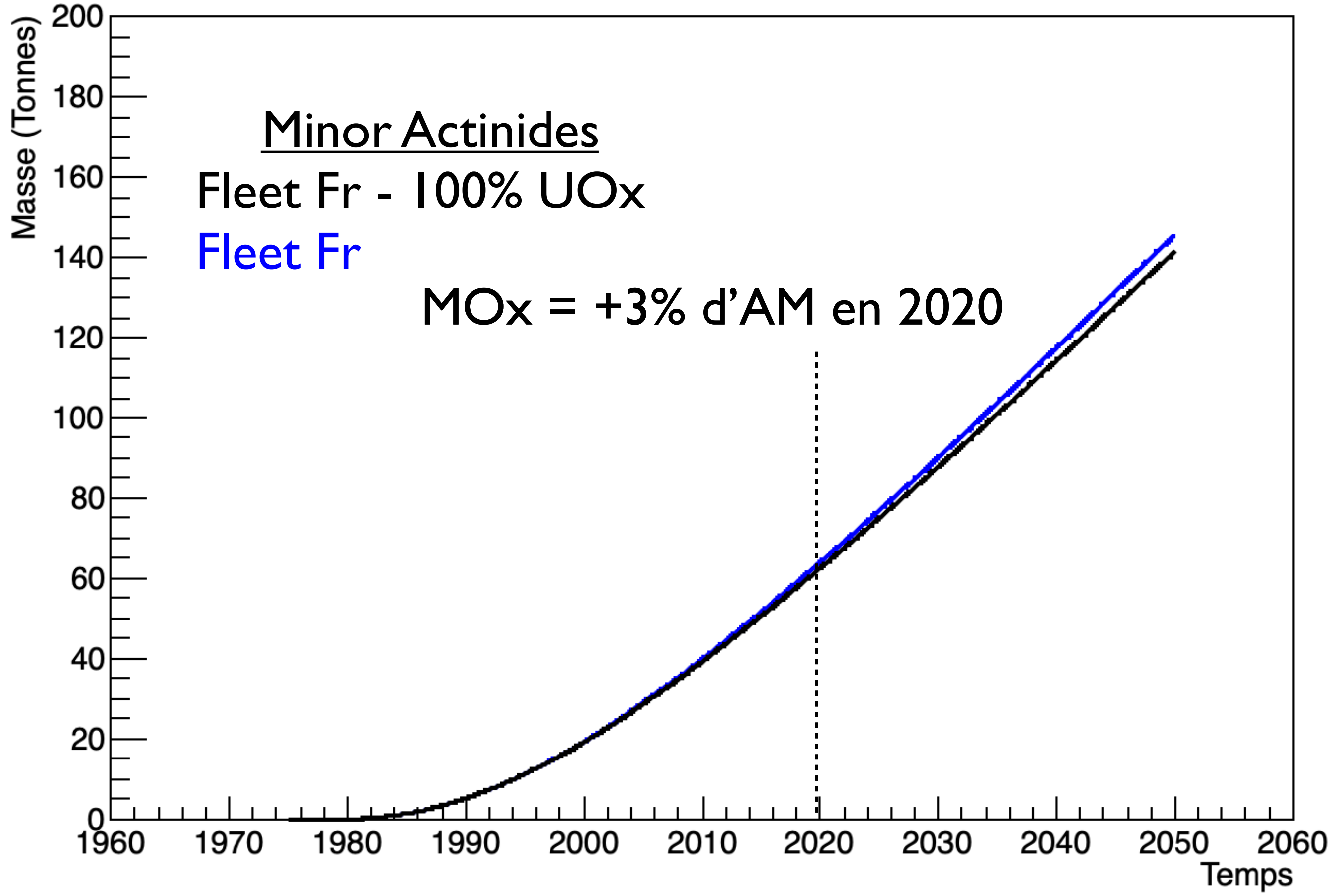


MOx / UOx comparison

- MOx strategy evaluation
  - Pu separation technical mastering
  - Pu concentration in spent fuel
  - Small resource economy
- Concerning materials inventory
  - Small Pu incineration
  - Small AM sur-production
  - Small impact on HN radiotoxicity

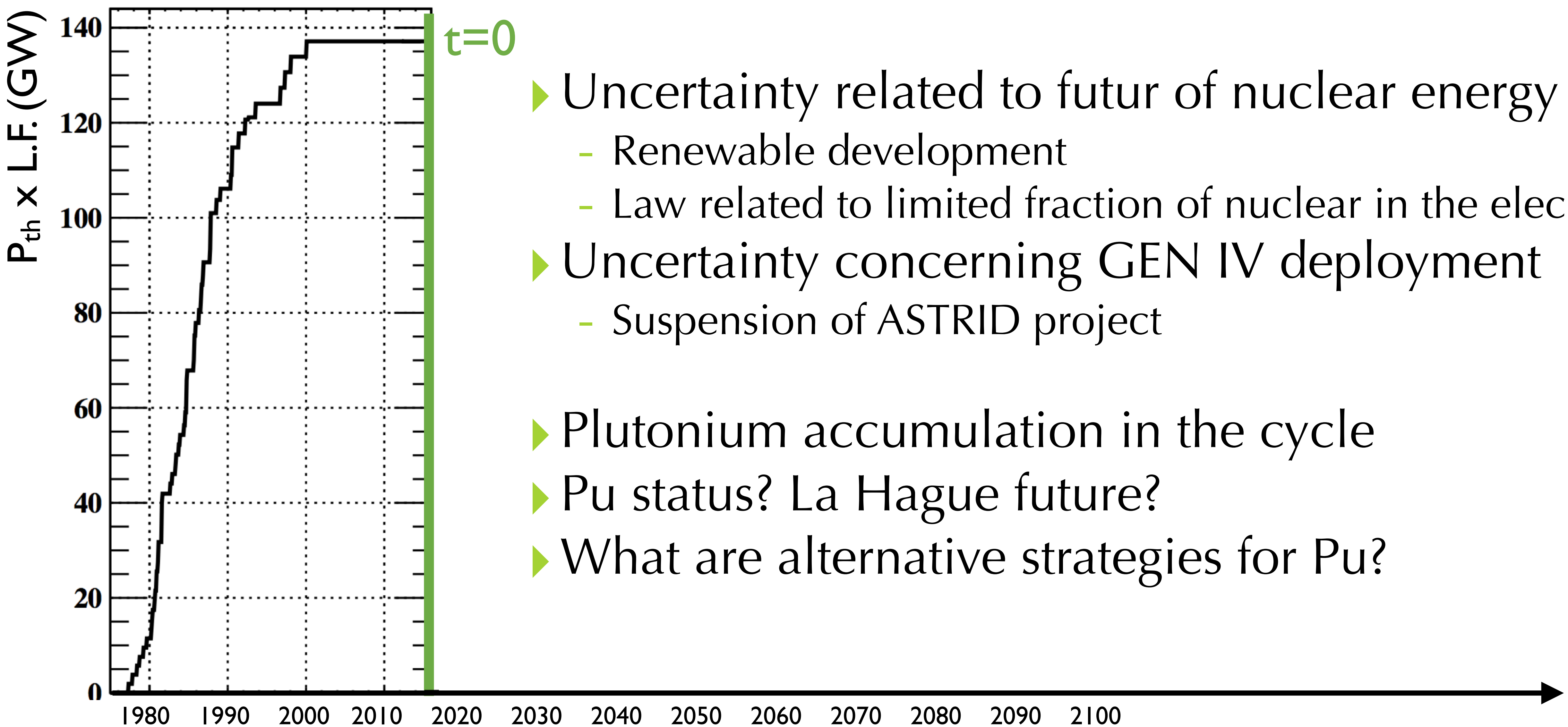






Multi-MOx in PWR

- French « official » nuclear scenarios were since the 80's very clear
  - Nuclear production was supposed to be constant during the century
  - Sodium fast reactors will be deployed to manage Pu and MA

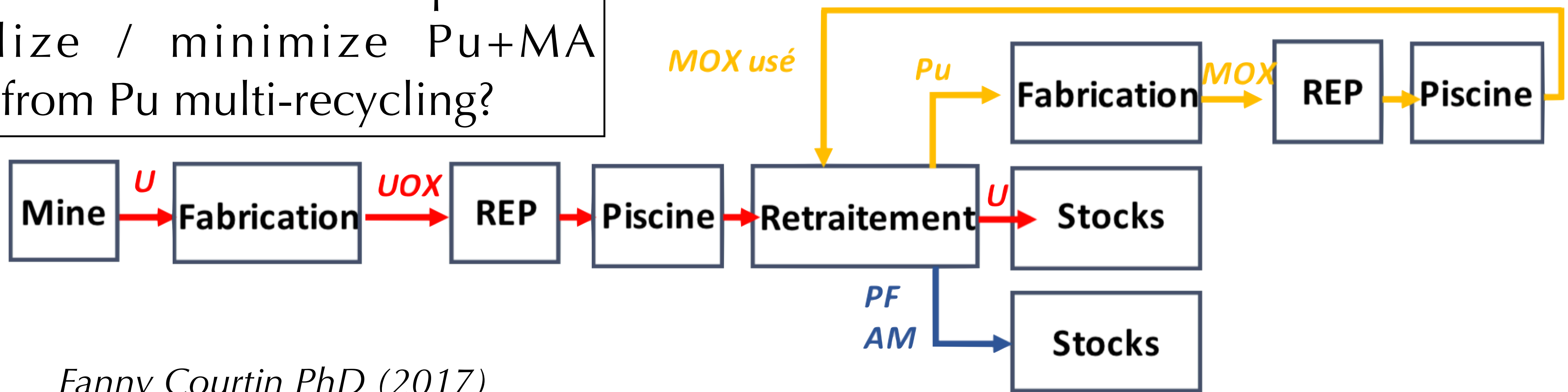


- Uncertainty related to futur of nuclear energy
  - Renewable development
  - Law related to limited fraction of nuclear in the elec. mix
- Uncertainty concerning GEN IV deployment
  - Suspension of ASTRID project

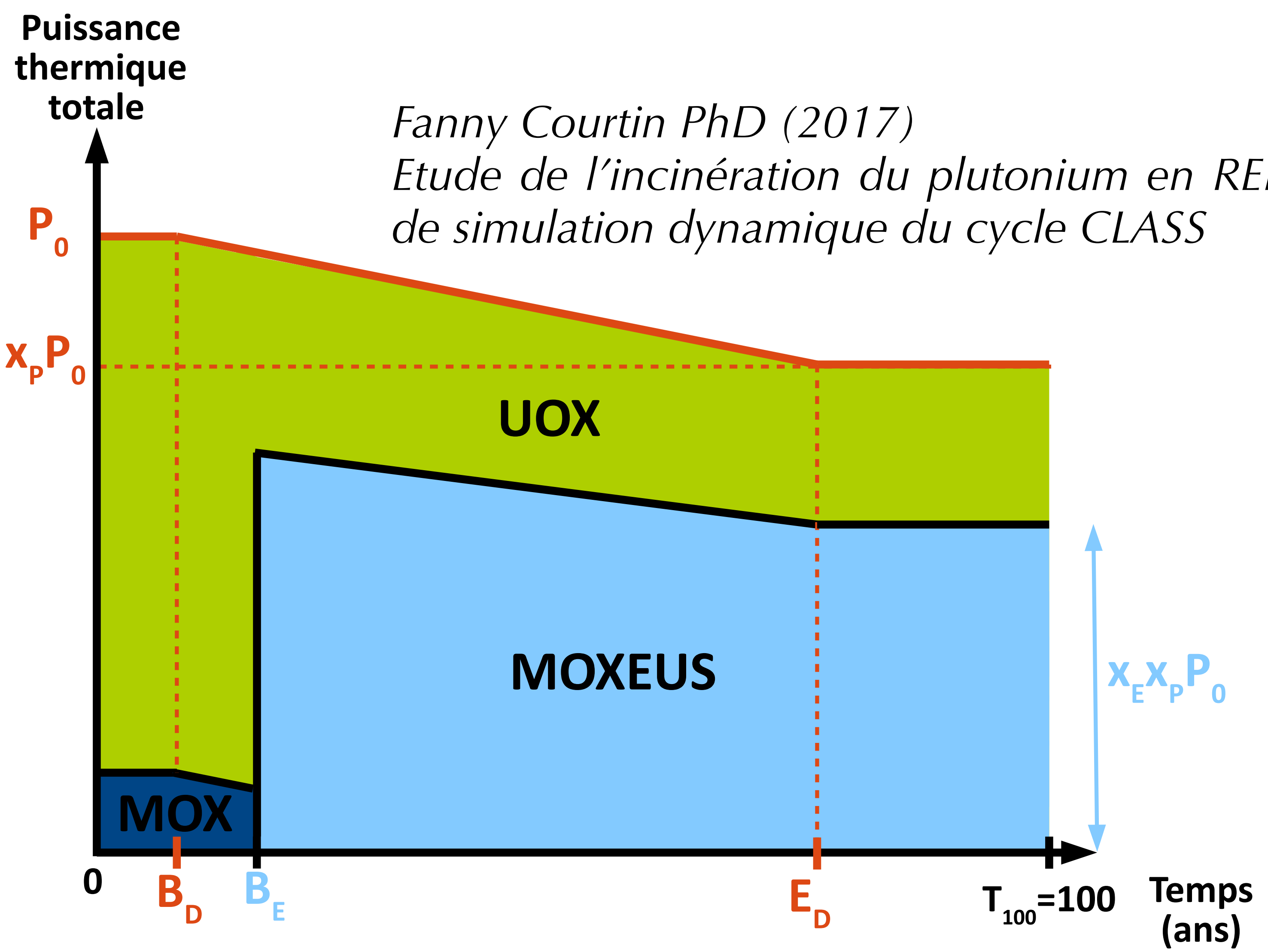
- Plutonium accumulation in the cycle
- Pu status? La Hague future?
- What are alternative strategies for Pu?

Design Of Experiment

Under which condition is this possible to stabilize / minimize Pu+MA inventory from Pu multi-recycling?

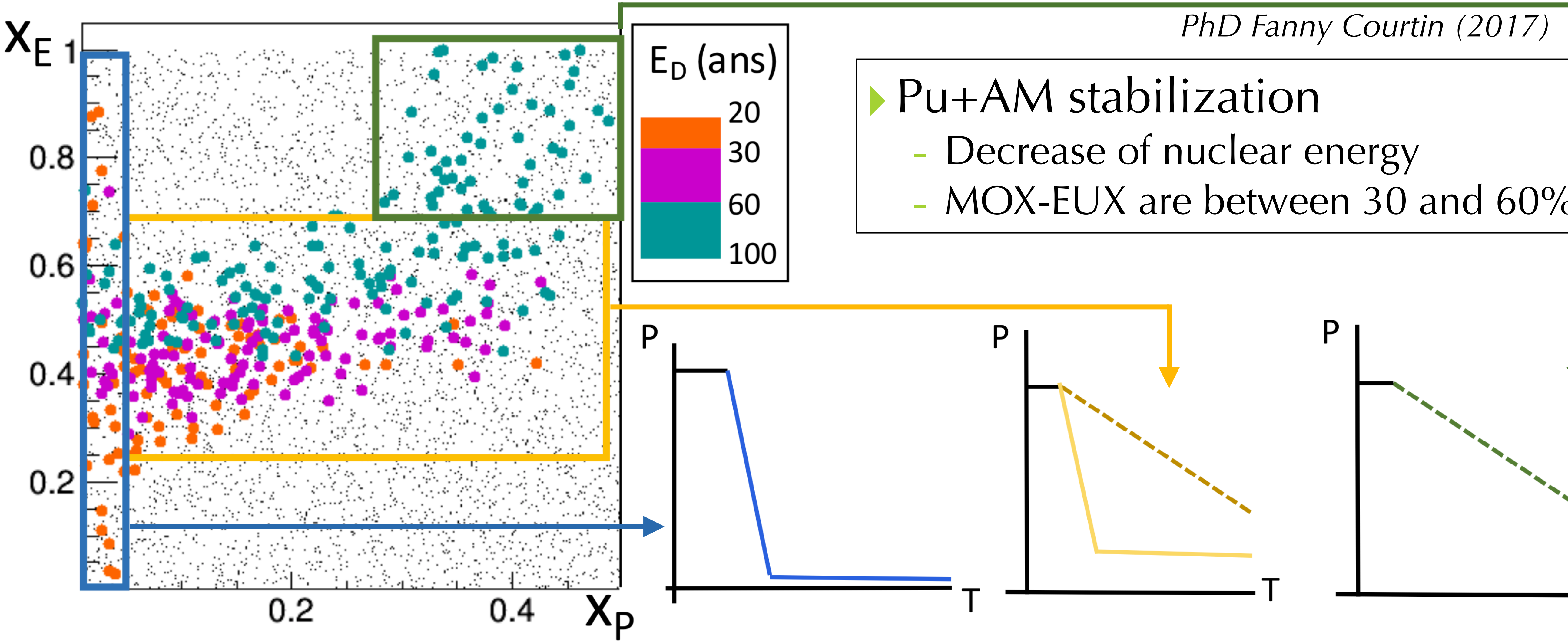


Fanny Courtin PhD (2017)
Etude de l'incinération du plutonium en REP MOX sur support d'uranium enrichi avec le code de simulation dynamique du cycle CLASS



BU UOX	[30 - 65] GWd/t
BU MOX	[30 - 65] GWd/t
BU MOXEUS	[30 - 65] GWd/t
X_p	[0 - 1]
E_D	[20 - 100] ans
X_E	[0 - 1]
Frac Max Pu	[8 - 13] %
TC_{UOX}	[3 - 10] y
TC_{MOX}	[3 - 10] y
TC_{MOXEUS}	[3 - 10] y
Strat. Fuel	LiFo, FiFo, Mix, Rand

Some results



- Pu+AM stabilization
  - Decrease of nuclear energy
  - MOX-EUX are between 30 and 60%

- Reference scenarios built with detailed fleet from GSA and macro reactors
  - are keeping the stabilization properties of Pu and MA,
  - are allowing to better quantify trajectories.

Many thanks !