Joint ICTP-IAEA-MAMBA School on Materials Irradiation: from Basics to Applications

Nuclear fission reactors

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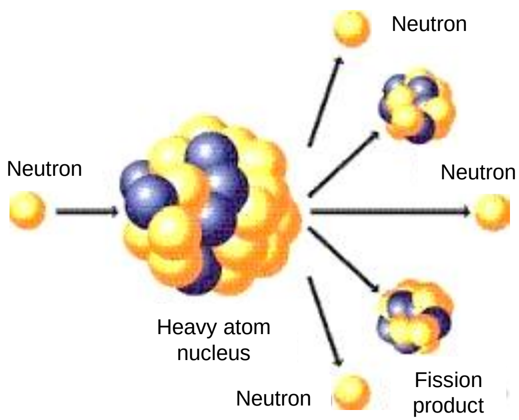
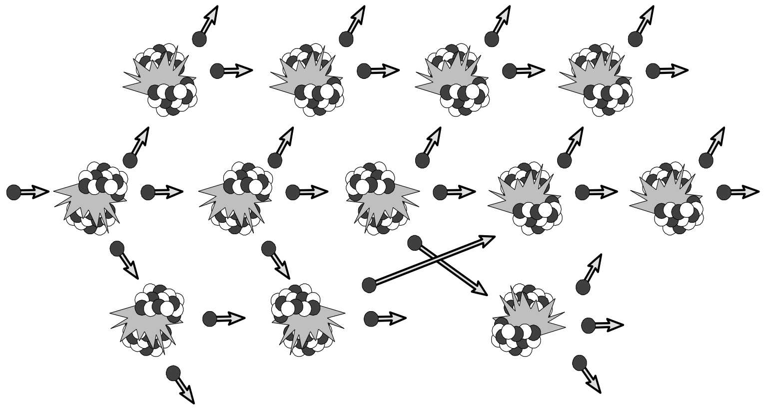
Nuclear fission

The operation of a **nuclear power plant** is the same that the operation of other power station (coal, oil, gas). Fossil or nuclear fuels produce heat, which is used to boil water to make steam, which is delivered to a turbine.

In the current nuclear reactors, this heat is created by means of **fission reactions** produced in the nuclear fuel: the fission generates heat and other particles (neutrons) with high kinetic energy that can be transmitted to the surrounding media and used to create more fission reactions.





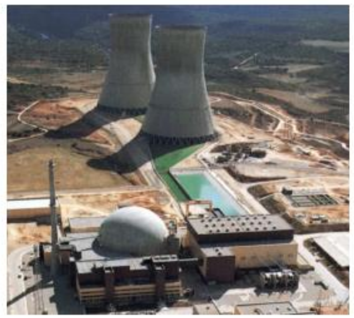
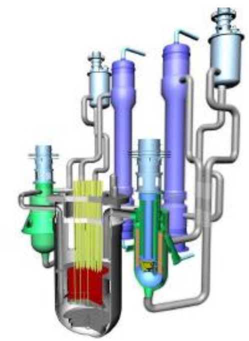
Evolution of Nuclear Energy

The development and evolution that nuclear energy has experienced in an **international and national framework** can be established in the following stages, defined by the historical facts and the successive reactions to them:

[Origin: discovery of radioactivity at end of XIX century
and nuclear fission in 1938 …]

1st stage (1939-1953): Secret and militarization of all the matters related to nuclear energy; First nuclear reactor in the frame of the Manhattan Project (CP-I,1942)
2nd stage (1953-1979): Demilitarization and promotion of the civil uses of nuclear energy in search of its commercialization;
1953 United Nations: “Atoms for Peace” declaration
3rd stage (1979-nowadays): Revision of the nuclear matters under the view of the nuclear safety; TMI-2 accident -> Concern on nuclear safety and radiologic protection

Evolution of Nuclear Reactors

Generation I	Generation II	Generation III+	Generation IV
Early prototype reactors	Commercial power reactors	Evolutionary/advanced LWRs	Innovative reactors
			
•Shippingport •Dresden, Fermi-I •Magnox	•LWR: PWR, BWR •CANDU, AGR •VVER/RBMK	• EPR • AP1000 • ABWR • ESBWR	• VHTR • SFR • GCR • SCWR • LFR • MSR

1950 1960 1970 1980 1990 2000 2010 2020 2030 2040

Obninsk, 1954

Fermi-I, 1963

Soviet Alpha class submarines, 70s

Akademik Lomonosov, 2019

HTR-PM, Shidao Bay, 2023

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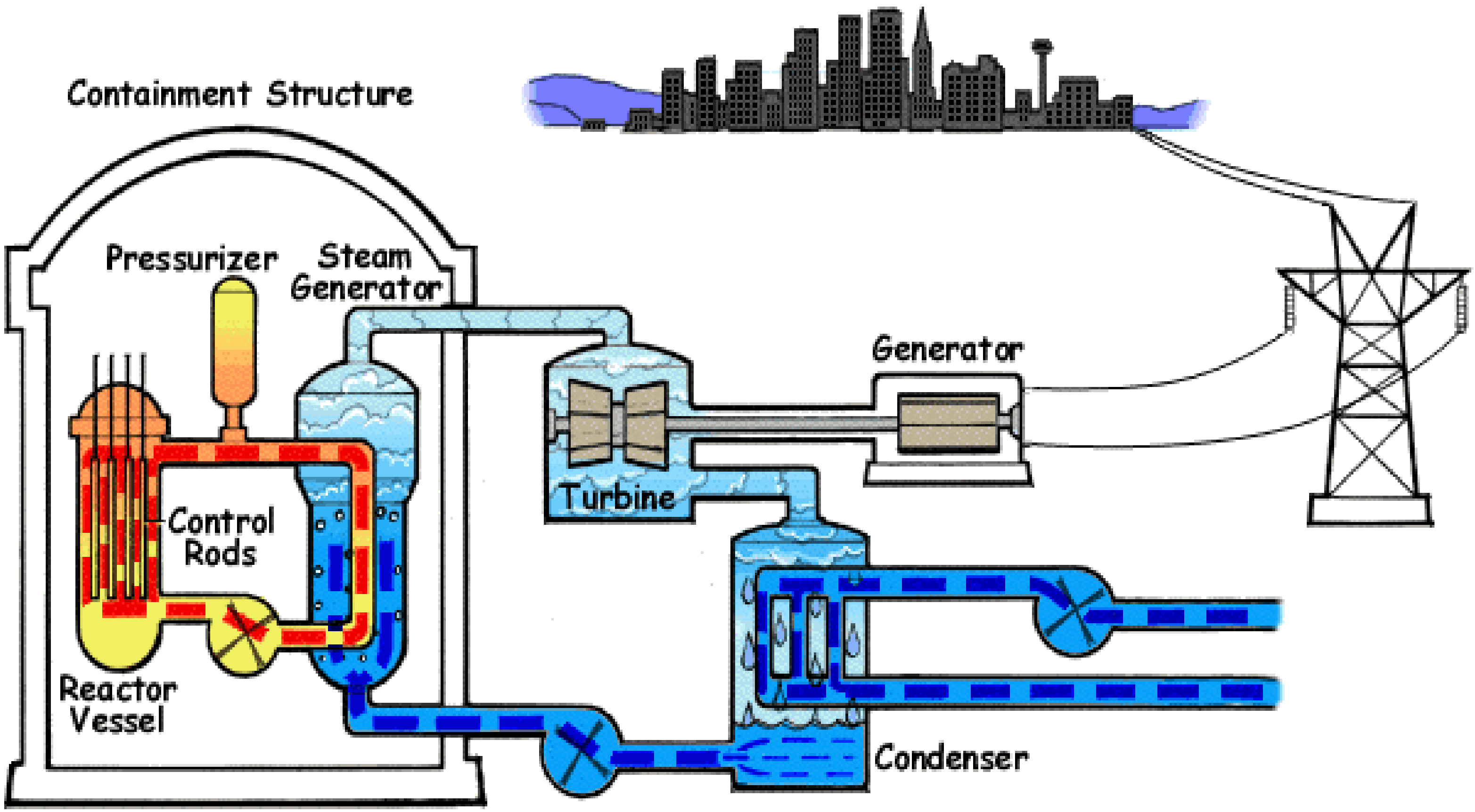
Nuclear reactor

General scheme of a Pressurized Water Reactor (PWR)



Reactor design

The three decisions with utmost importance in nuclear reactor design are:

- **Fuel** (natural U, enriched U, MOX, ...)
- **Moderator** (water, graphite, ...)
- **Coolant** (water, sodium, lead, gas, ...)

Fuel

The optimum use of fuel is sought:

- Long burn-up
- Low generation of waste
- Management of Pu and minor actinides

Fuel design

- Pellets
- Spherical fuel (TRISO)
- Alternatives (hexagonal, plates)
- Oxide/Carbide/Metallic, U, U+Pu, etc.
- Molten salt

Coolant

Main characteristics

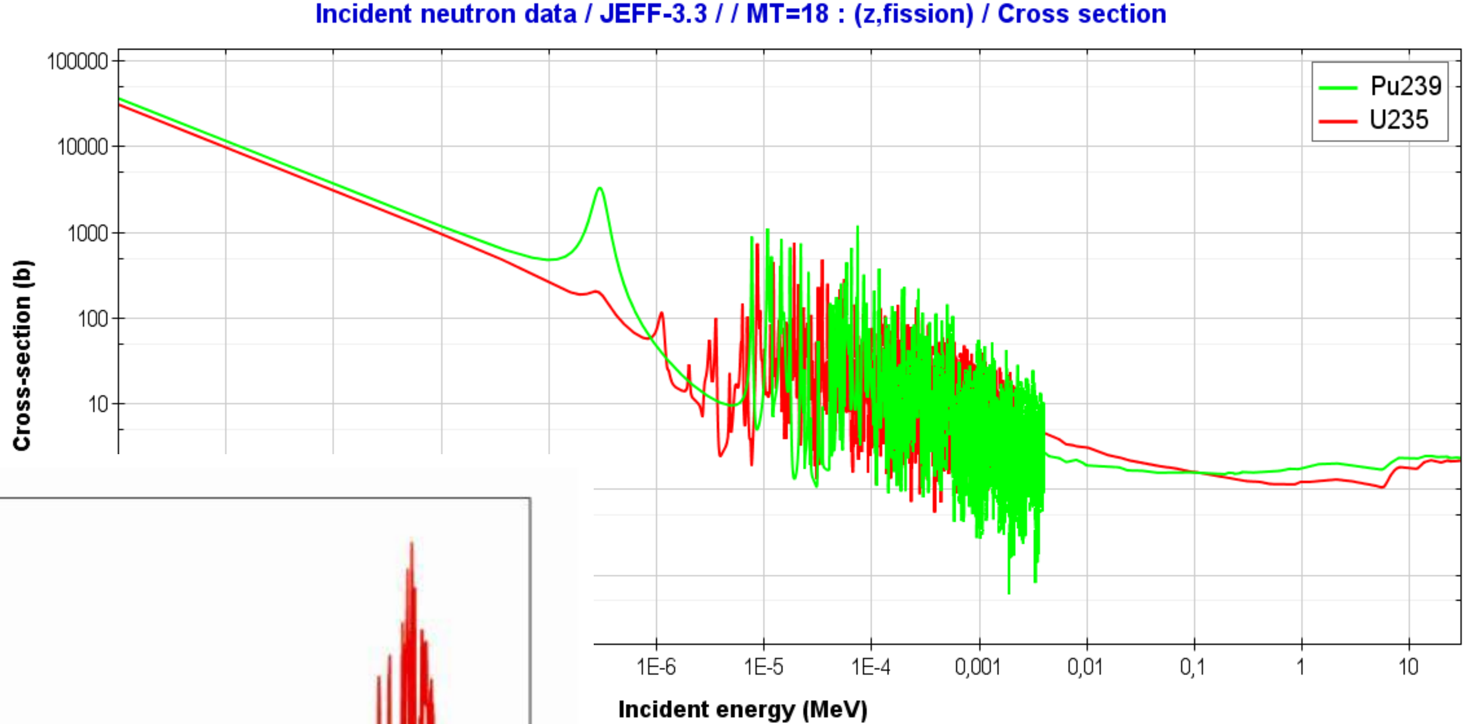
- Operation temperature
- Heat capacity
- Thermal conductivity
- Density
- Neutronic aspects
- Corrosion
- Technology for the adequate operation (airtightness, pressure, in-service inspection, ...)
- Safety
- Chemical inertia
- Opacity

Neutron moderation

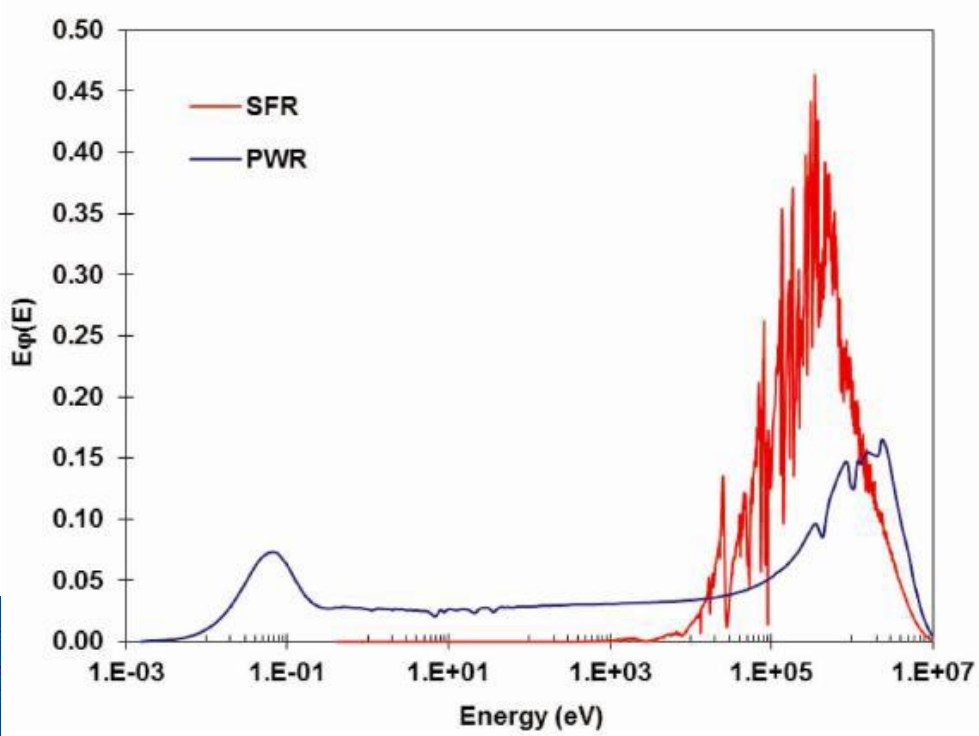
- Presence of moderating elements if required
  - Water in liquid state and organic compounds (C or H)
  - Gases (He, CO_2), liquid metals

- For Generation-IV reactors: Use of coolants compatible with low moderation as liquid metals at room temperature (mercury, sodium, sodium-potassium alloy, lithium, lead, lead-bismuth eutectic)

Neutron spectrum



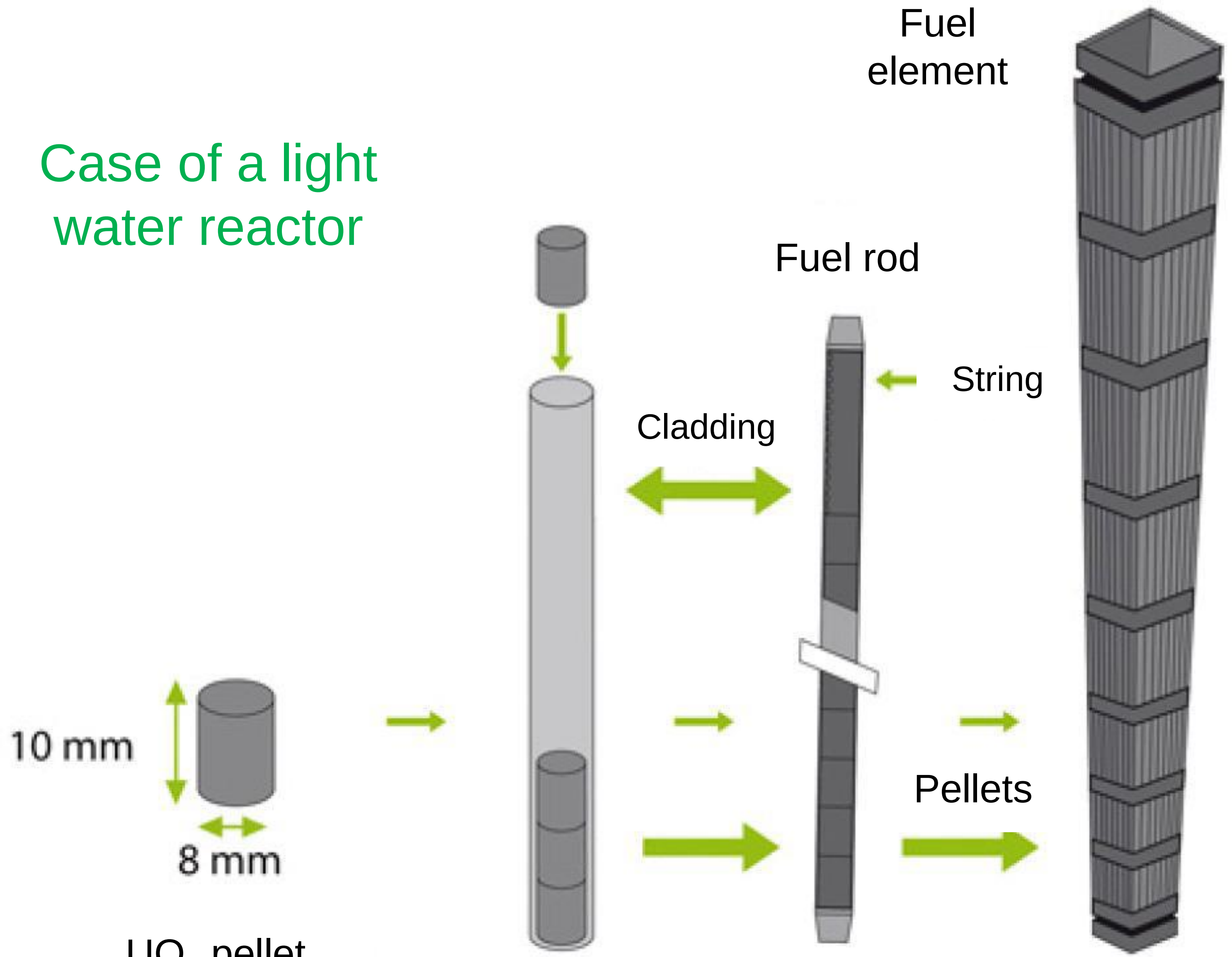
Fuente: ***JANIS database***, NEA.



Fuente: ***Fast Reactor Physics and Computational Methods***. W.S. Yang. *Nuclear Engineering and Technology, 44(2), 2012*.

Fuel assembly

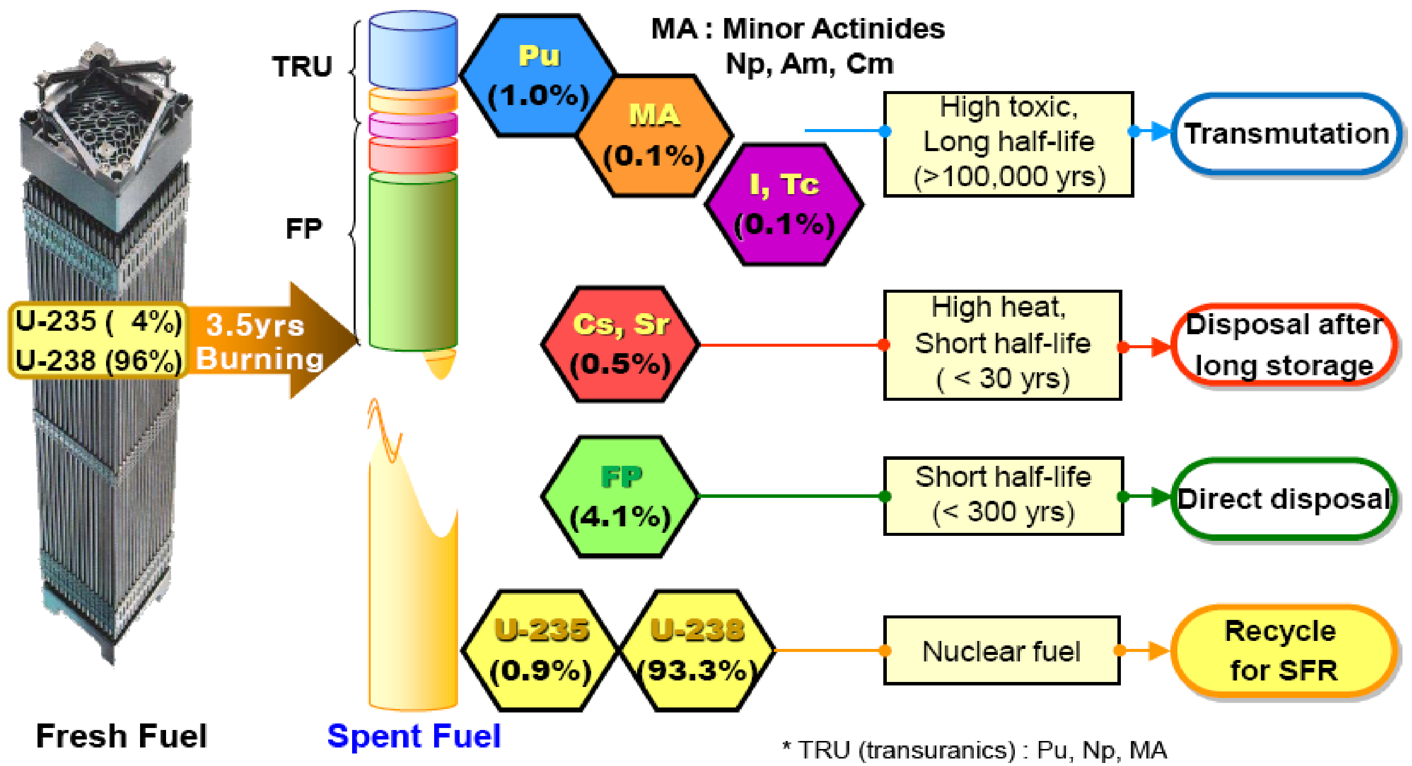
Case of a light water reactor



UO_2 pellet

Generation of waste

- In addition to generating energy, the irradiation creates non-desired isotopes or elements.



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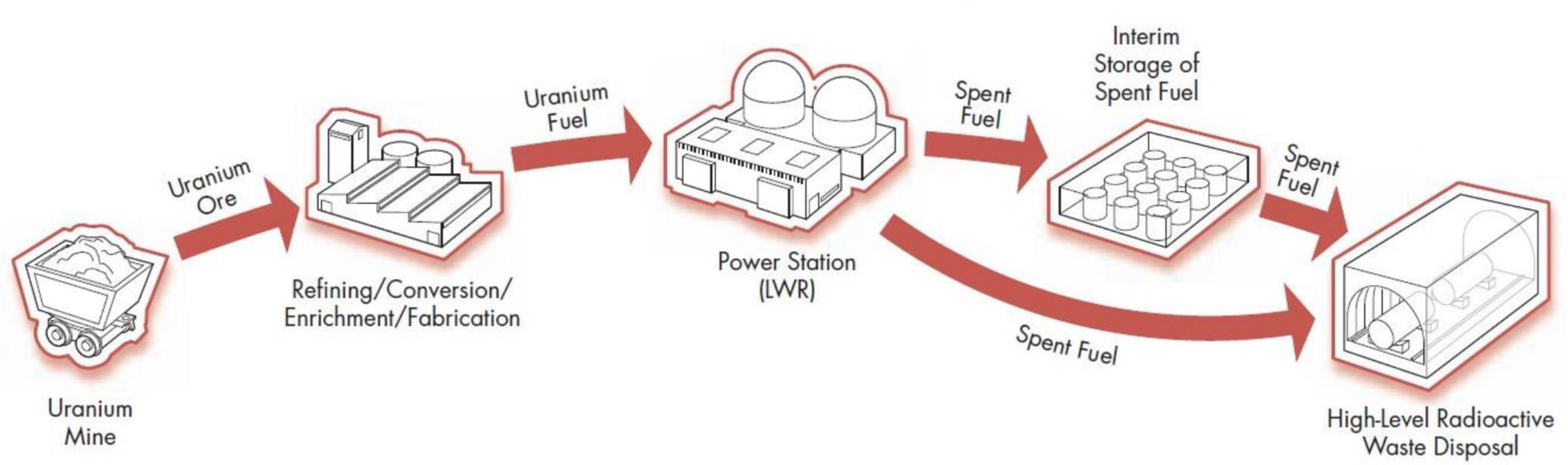
Nuclear fuel cycle

Open cycle (once-through)

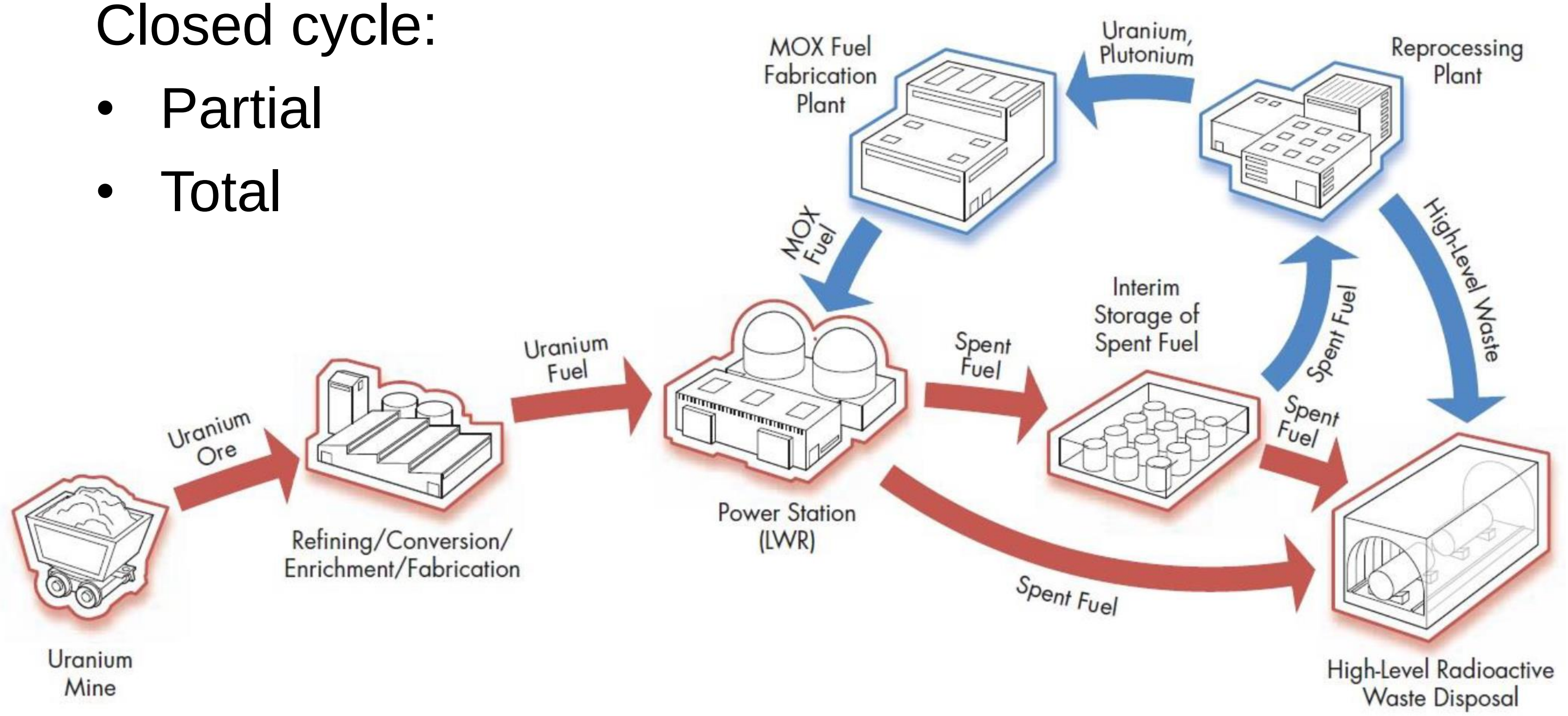


Source: A. Clamp. Toward an Integrated Nuclear Fuel Cycle. EPRI Journal, 2008 Spring

Nuclear fuel cycle

Closed cycle:

- Partial
- Total



Source*:* A. Clamp. Toward an Integrated Nuclear Fuel Cycle. EPRI Journal, 2008 Spring

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Modelling a nuclear reactor: Introduction

Reactor physics can be described by compact analytical equations and even some problems can be solved in an analytical form, with different degree of approximation. However most practical problems require the use of computerized numerical methods. This apply to all kind of problems in reactor physics:

1) Neutronics (Flux and power distributions, shielding, rad. protection, criticality,…)

Boltzmann or neutron transport equation

$$\frac{1}{v}\frac{\partial \Phi(\vec{r},E,t)}{\partial t} + \vec{\Omega}\cdot\vec{\nabla}\Phi(\vec{r},E,t) = S(\vec{r},E,t) - \Sigma_T(\vec{r},E,t)\Phi(\vec{r},E,t) + \int_0^\infty \Sigma_s(\vec{r},E',t')f(\vec{r},E'\to E,t)\Phi(\vec{r},E',t)dE' + \int_0^\infty s(E')\Sigma_f(\vec{r},E',t')\nu(E',t)\Phi(\vec{r},E',t)dE'$$

$$\Sigma_T(\vec{r},E,t) = \sum_i \sigma_{Ti}(E)n_i(\vec{r},t),$$

$$n_i(\vec{r},t) \approx n_i(\Phi,t)$$

Modelling a nuclear reactor: Introduction

Reactor physics can be described by compact analytical equations and even some problems can be solved in an analytical form, with different degree of approximation. However most practical problems require the use of computerized numerical methods. This apply to all kind of problems in reactor physics:

1) Neutronics (Flux and power distributions, shielding, rad. protection, criticality,…)
2) Isotopic evolution of irradiated materials

Bateman or isotopic evolution equation

$$\frac{dn_i(\vec{r},t)}{dt} = \left[\sum_{j \neq i} n_j(\vec{r},t) \left(\lambda_{ji} + \int \sigma_{ji}(\vec{r},E) \Phi(\vec{r},E,t) dE \right) \right] - \left[n_i(\vec{r},t) \left(\lambda_i + \int \sigma_{abs}(\vec{r},E) \Phi(\vec{r},E,t) dE \right) \right], \text{ with } i=1,M \text{ isotopes.}$$

$$\Phi = \frac{P}{\sum_i n_i \sigma_{fi} Q_i}$$

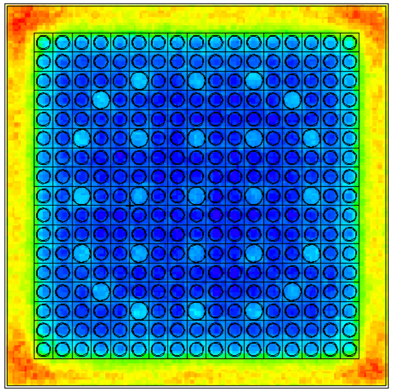
Modelling a nuclear reactor: Introduction

Reactor physics can be described by compact analytical equations and even some problems can be solved in an analytical form, with different degree of approximation. However most practical problems require the use of computerized numerical methods. This apply to all kind of problems in reactor physics:

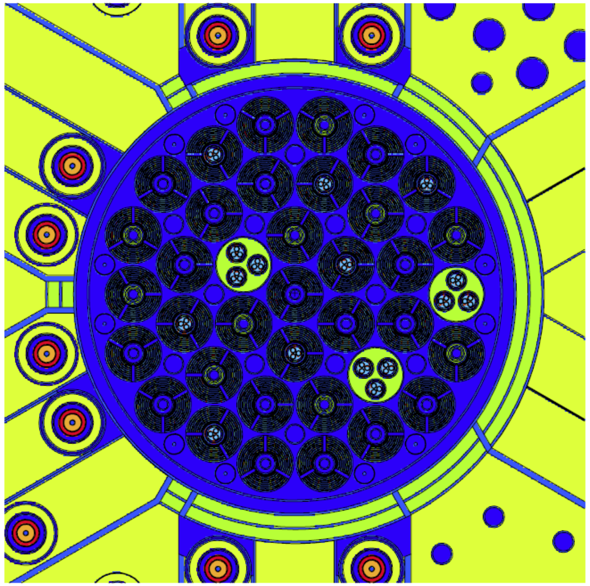
1) Neutronics (Flux and power distributions, shielding, rad. protection, criticality,…)
2) Isotopic evolution of irradiated materials
3) Thermal-hydraulics
4) Mechanical Structures and materials behavior
5) Reactor dynamics and control
6) Fuel loading optimization,
7) …

Why Numerical tools for neutron transport and neutronic problems

- Complex dependence of physical parameters on their variables
  - Capture, Fission and other Cross sections vs. energy
  - Angular and energy distributions and correlation of secondary particles
- Problem geometry
  - Shapes of different pieces
  - Relative position and orientation
  - Material distribution and evolution



- Boundary conditions
  - Sources: distribution, energy spectrum, angular distribution
  - Temperatures: thermal expansion, effective cross sections
  - Radiation protection
- Detector response
  - To measure flux – detect ^{235}U fissions or $B(n,\alpha)$.
  - Cross section of detector targets
  - Same for other magnitudes: power, dose, …



Type of neutronic problems to solve

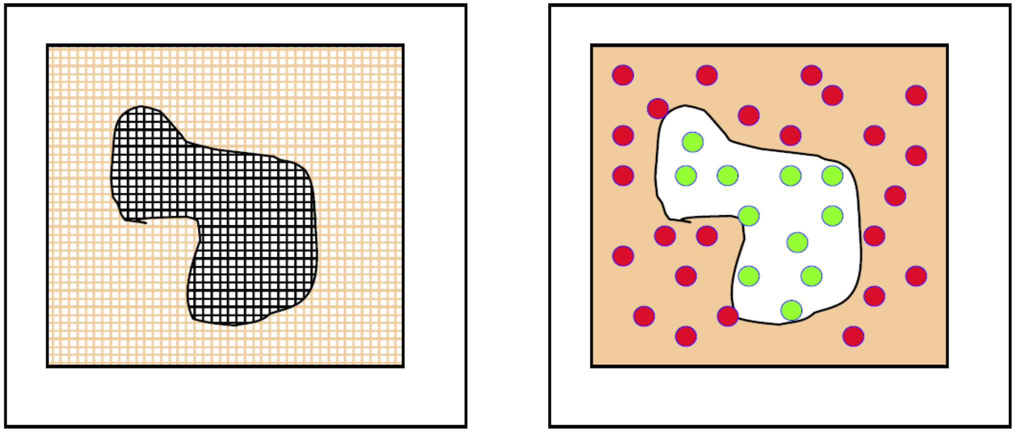
- Transport of neutrons, photons and other particles from a source through a system
  - Map of the neutron (photon, proton,…) flux from a source at a given point
  - Fraction of a radiation crossing a shielding element (dose, probability, damage,…)
  - Neutron energy spectra, dependence vs. time of the neutron flux at a point,…
- Criticality
  - Neutron multiplication in a nuclear system, moderation and thermalization
  - Determination of critical configurations
  - Optimization of geometry, material (fuel, absorbers) composition, reactivity margin
  - Effect of material evolution, control rods, consumable absorbers, temperature on k_{eff}
- Reaction rates
  - Fission rates and power: Total, in each fuel pin, maximum fission rate density
  - Capture rate and activation, Neutron production rate and radiation protection
  - Detection rates

Types of methods for resolution

- Finite difference
- Finite elements
- Generic deterministic codes
- Monte Carlo methods

The Monte Carlo method

- Definition: Solving problems by using properties of random variables and stochastic theory
- Simple Example: Surface calculation

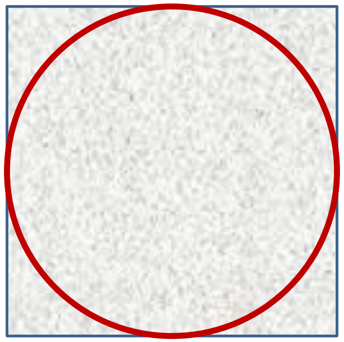


- Analog and non-analog MC solutions
- Some properties of MC method
  - Solution is not exact but have a statistical uncertainty that is reduced with the square root of the number of trials or of the computer time.

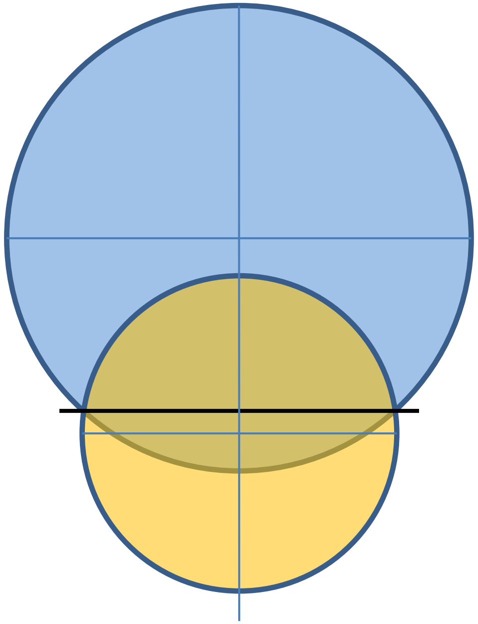
$$\varepsilon = \sqrt{c / N_t} = \sqrt{c' / CPU\_time}$$

The Monte Carlo method

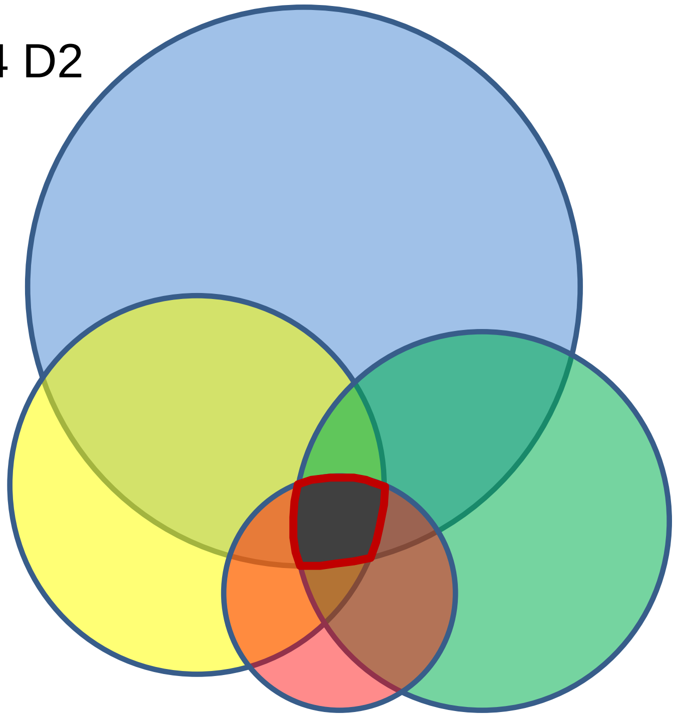
Level 1 D2



Level 2 D2

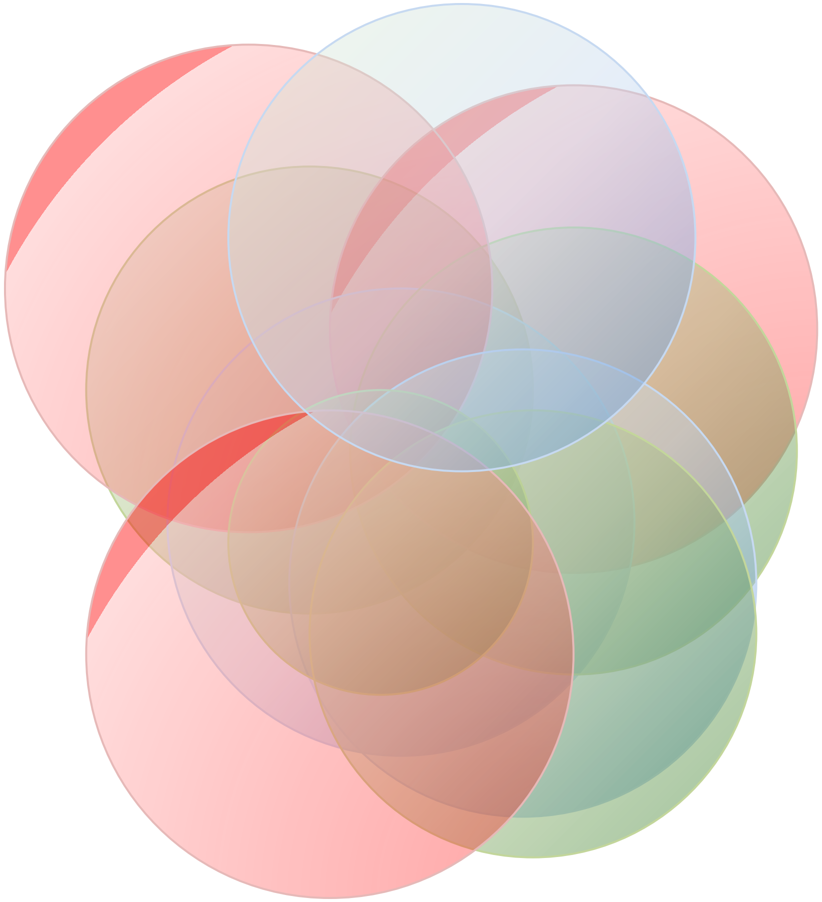


Level 4 D2



The Monte Carlo method

Level 10 D3



Generic solution MC: Dimension N_Dims × N_spheres

```
side = …..
N_inside = 0
Do m= 1, N_trials
     inside = true
     do k=1,N_spheres
          do j=1,N_Dims
             if(k== 1) x(j) = random() * side
             if ( (x(j)-csph(k,j))^2 > rsph(k)^2 ) inside = false
             if (inside == false) break
          end do
          if (inside == false) break
     end do
      if (inside) N_inside = N_inside +1
End do
Volume = (side^N_Dims) * N_inside /N_trials
```

Neutron transport by Monte Carlo

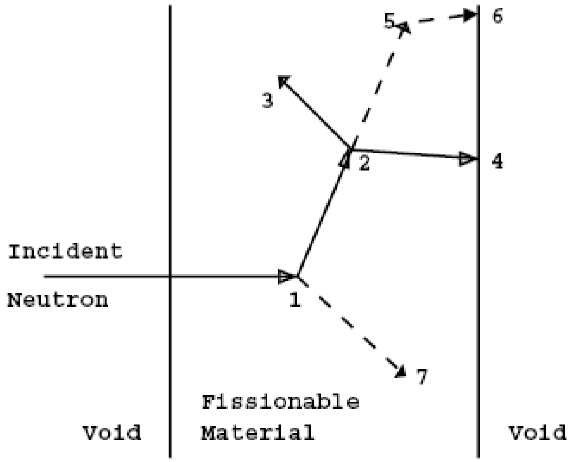
Concept of neutron transport by MC

- Neutronics of single neutrons described as particles (not fields) following the exact physics of neutrons in a medium (at least in a probabilistic sense).
- Individual, independent, random, different histories for each neutron
- Results obtained by averaging / statistical analysis of the different neutron histories

Basic processes

- Generation of neutrons from sources (position, energy, direction)
- Straight, uniform (constant speed) and free movement of neutrons between collisions
- Crossing of borders between regions with different materials, densities, temperatures, … (geometry)
- Elastic collisions with change of direction and neutron energy moderation
- Inelastic collisions (capture, fission, (n,xn), (n,n’), …) absorption and generation of secondary particles
- Transport of secondary particles till the end of active particles
- Recording and scoring of information for:

  every collision, track segment, secondary particle creation



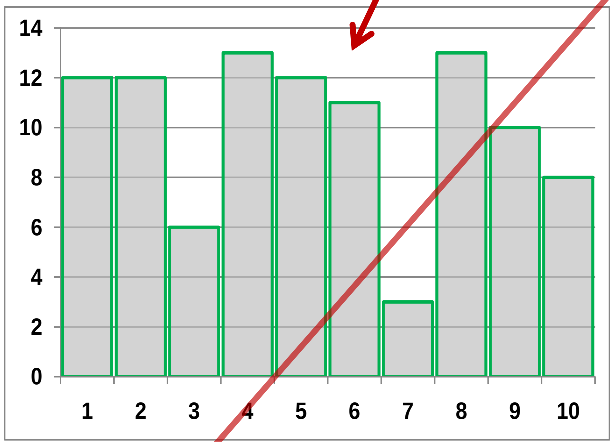
Special techniques

- Criticality calculations
- Variance reduction

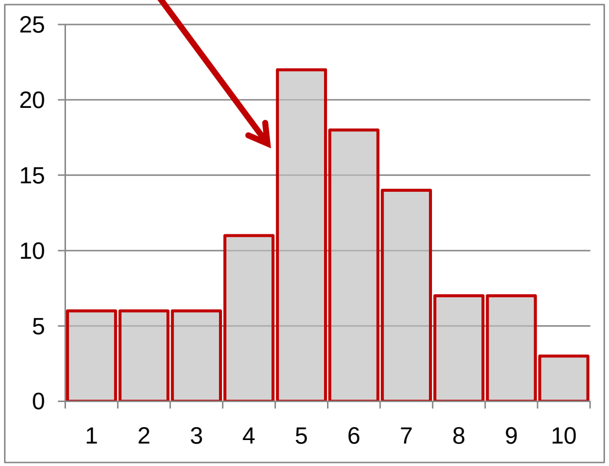
Results provided by Monte Carlo

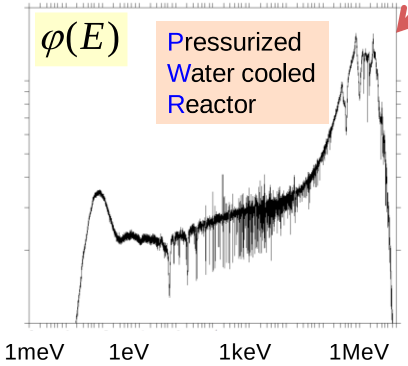
Tally: Histograms of Frequencies: How many events within each given interval of possible values (between 1- 2,…) If there are many small intervals it may look like a function.

1.375		
0.554	1	12
6.286	2	12
4.017	3	6
3.349	4	13
4.590	5	12
8.183	6	11
5.394	7	3
2.205	8	13
3.052	9	10
3.008	10	8
4.097		0
7.503		
0.768		100
1.828		
1.897		
8.272		
4.964		
5.670		
8.622		
0.722		
9.918		

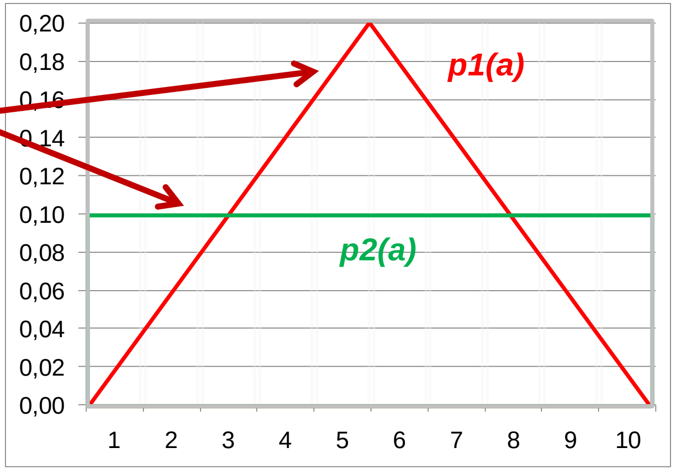


0.405		
5.512	1	6
0.685	2	6
4.748	3	6
4.639	4	11
4.349	5	22
4.762	6	18
7.304	7	14
6.016	8	7
5.501	9	7
5.183	10	3
0.988		0
1.675		
6.635		100
9.163		
5.367		
6.817		
6.446		
4.822		
8.553		
3.194		
5.629		





Probability density function p(E): the probability of an event between E and E+dE is p(E)•dE. It can be estimated from a histogram of frequencies with enough statistics after normalization as $\int p(E)\,dE = 1$



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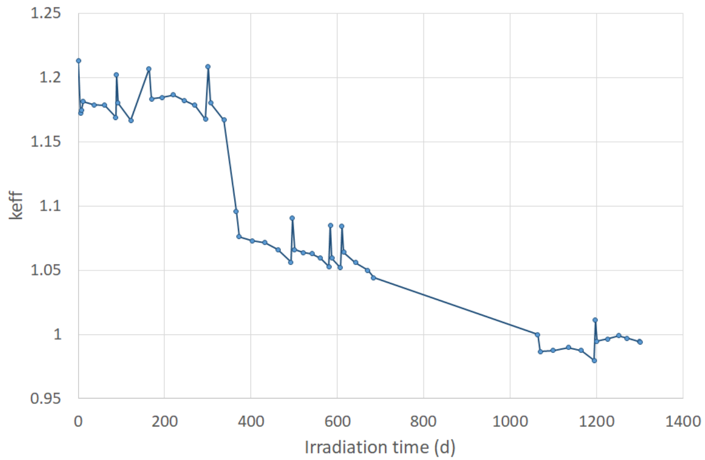
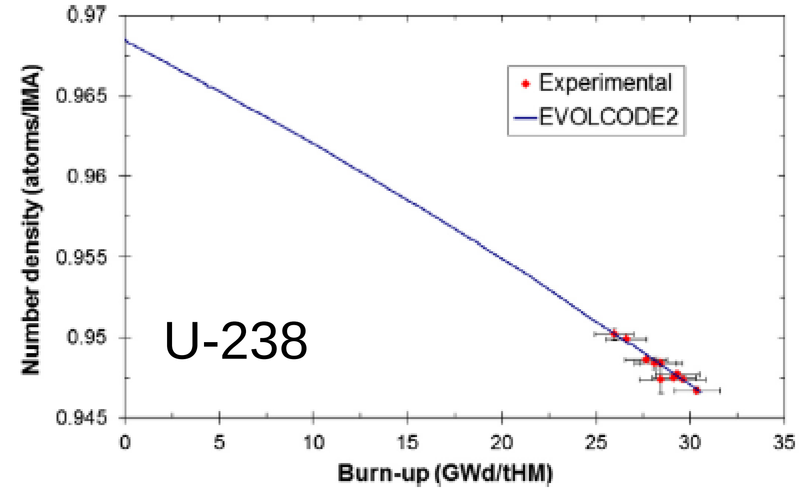
Burn-up code results

Criticality evolution

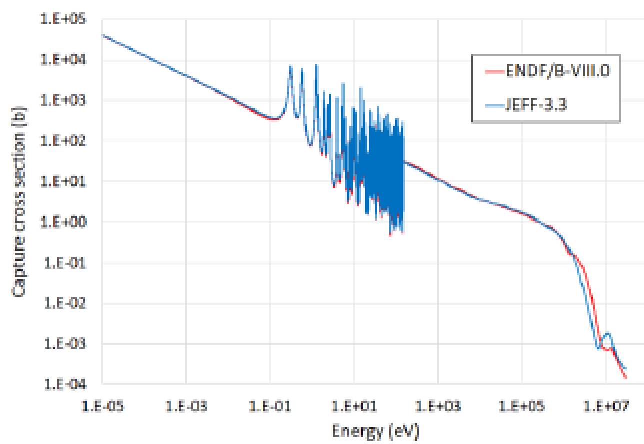


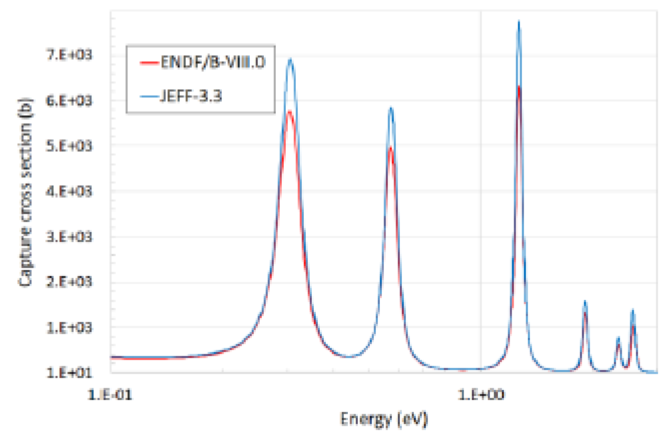
Evolution of isotope mass



Isotope	Case A
U-233	-2.3%
U-234	0.8%
U-235	-1.5%
U-236	-1.6%
U-238	0.1%
Np-237	-0.8%
Pu-238	-2.4%
Pu-239	-2.5%
Pu-240	-0.5%
Pu-241	-3.2%
Pu-242	-1.7%
Am-241	-9.7%
Am-242m	16.6%
Am-243	-2.2%
Cm-242	0.3%
Cm-243	-1.1%
Cm-244	-11.6%
Cm-245	-15.4%
Cm-246	-11.1%

Deep analyses





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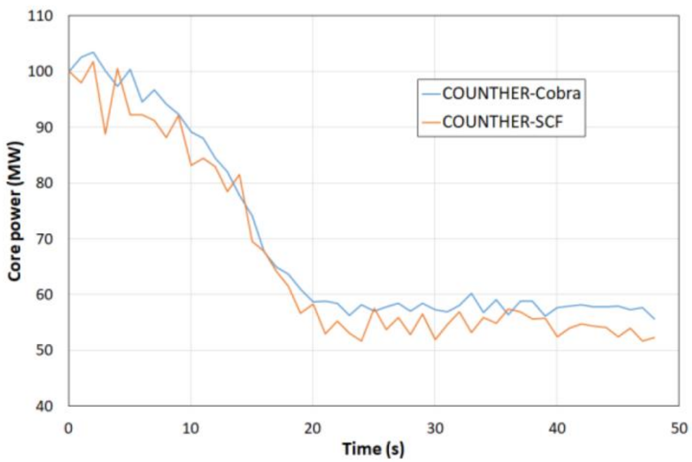
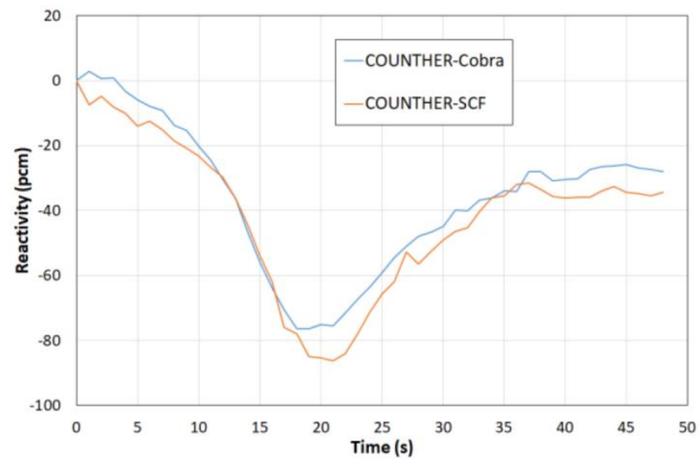
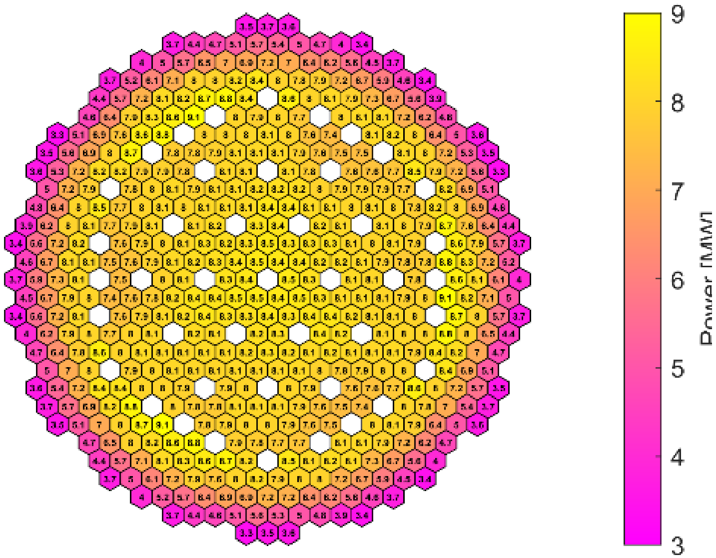
Fuel cycle code results

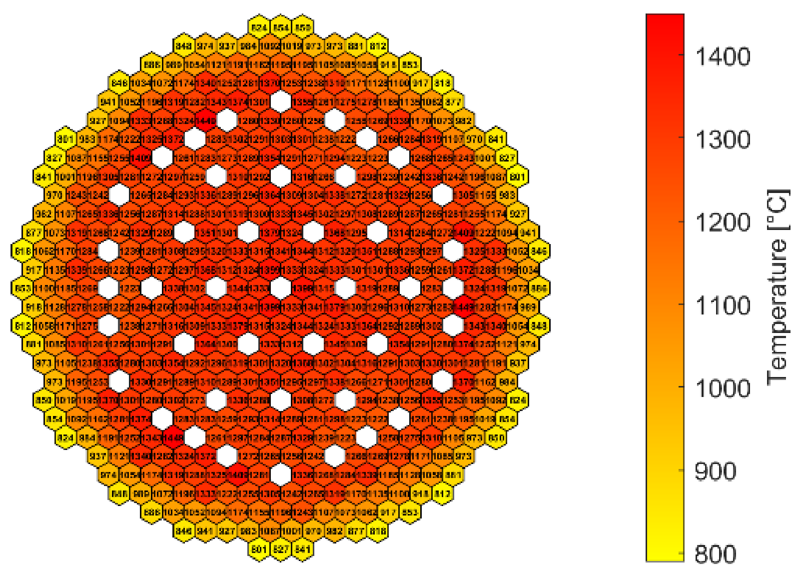
Evolution of main magnitudes in a ULOF transient scenario





Core map: power and temperature





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Conclusions

- Nuclear fission is a well-known technology
- Based on nuclear fission chain reaction
- Fuel, coolant and cladding are important for reactor design
- A nuclear reactor can only be understood in a fuel cycle strategy
- Complexity leads to the necessity of simulation
- Monte Carlo stochastic procedure is often the selected methodology for simulation

Safety first!

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Thank you for your attention

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