Transmutacja Jądrowa w Reaktorach Prędkich i Systemach Podkrytycznych Sterowanych Akceleratorami

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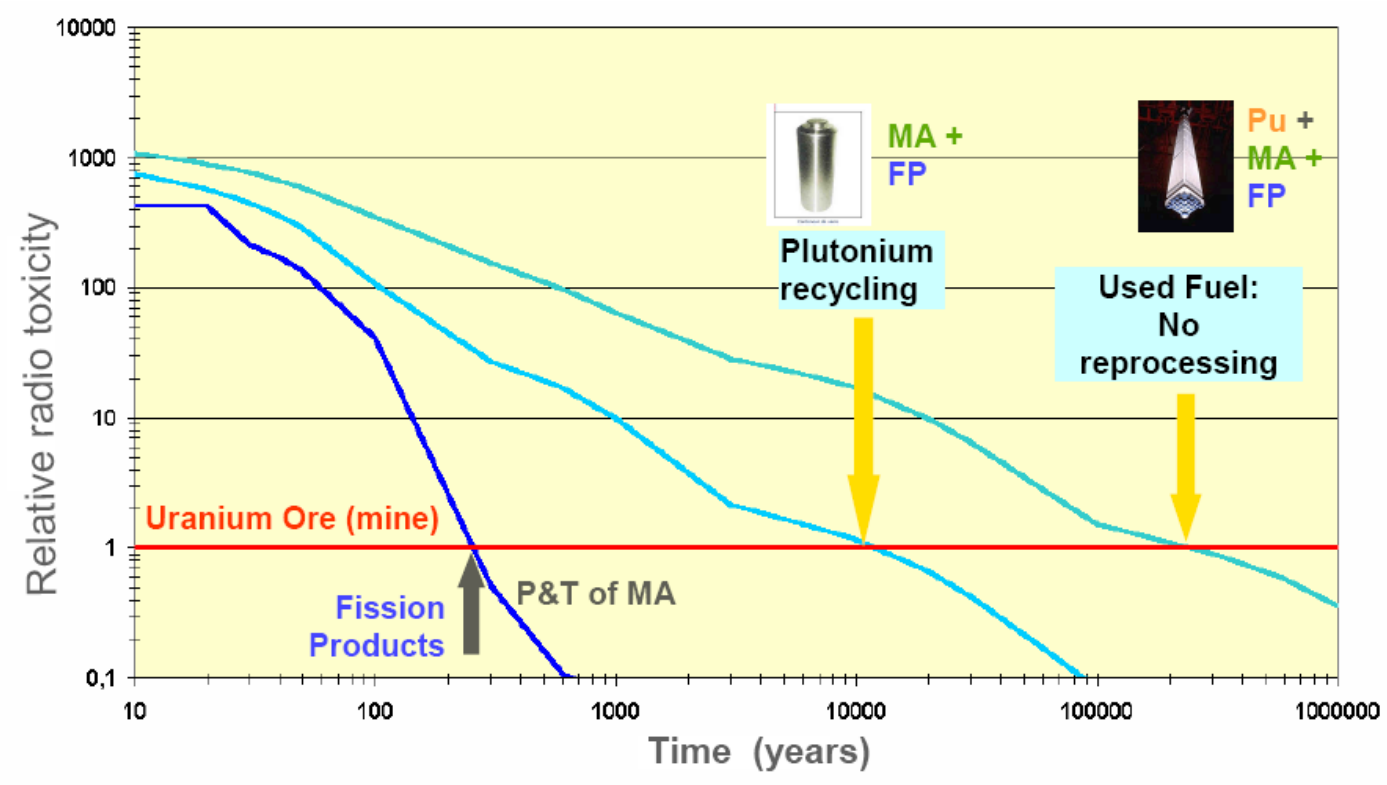
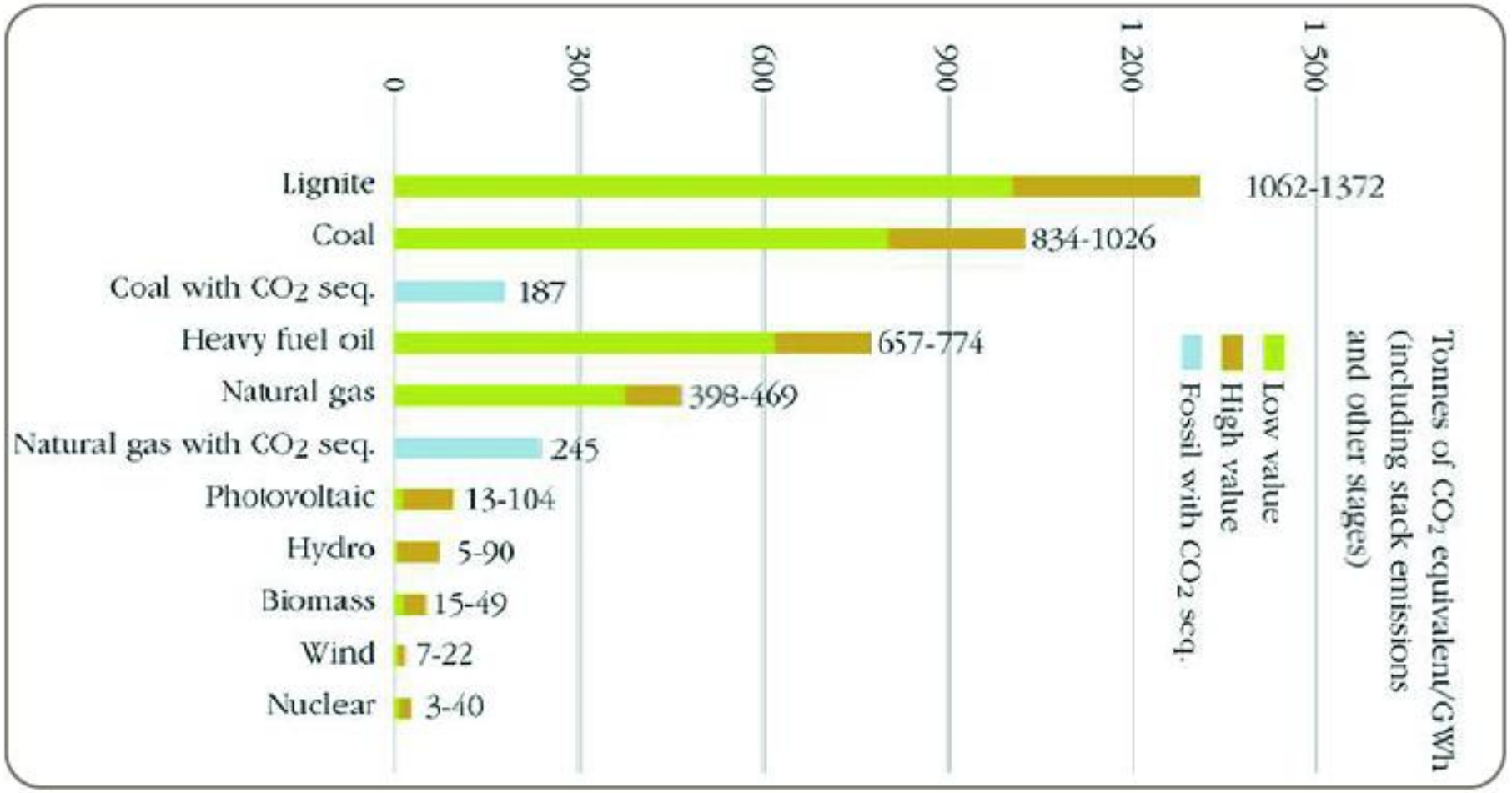
Introduction

Impact on repository potential radiotoxicity

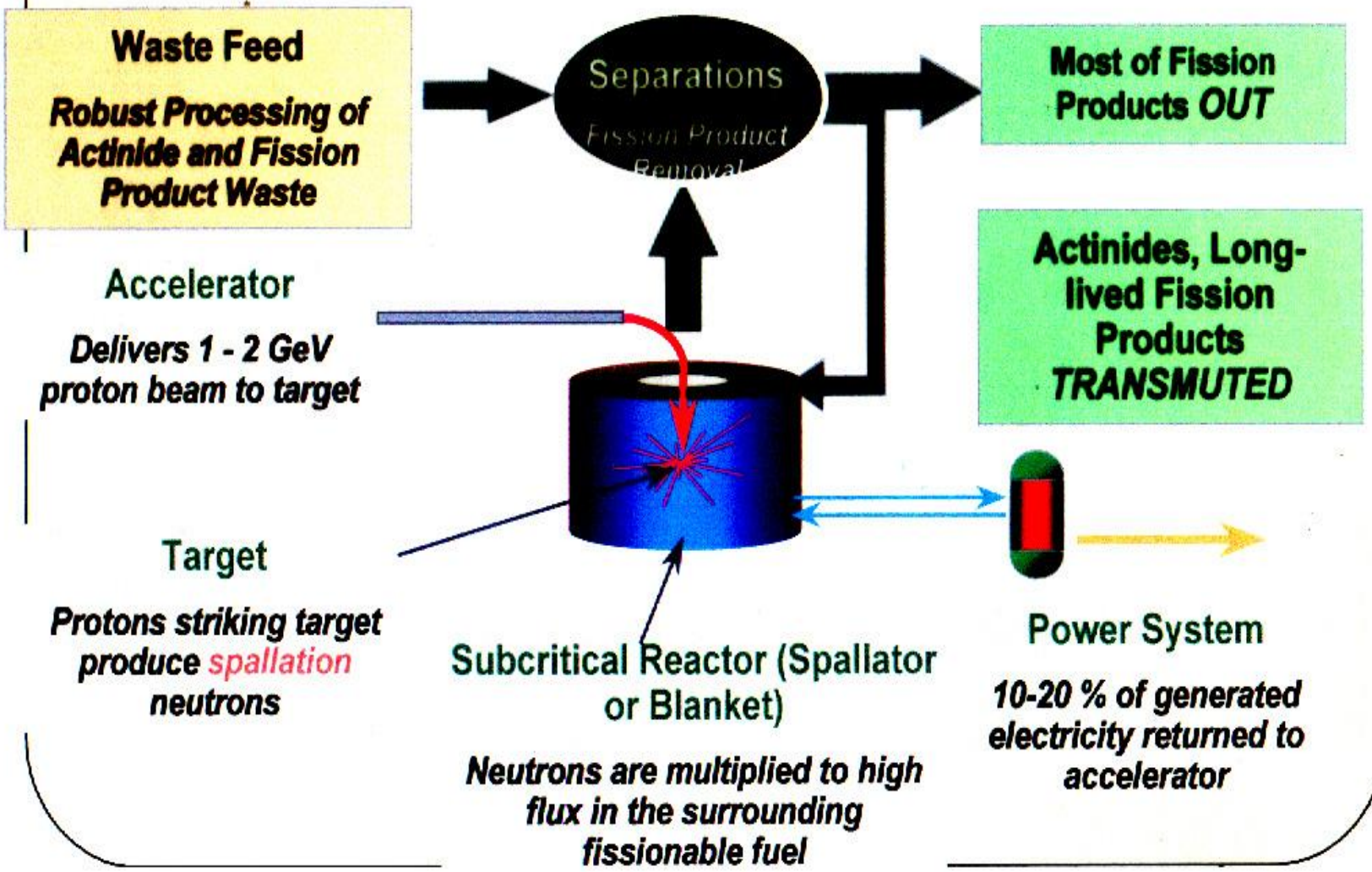


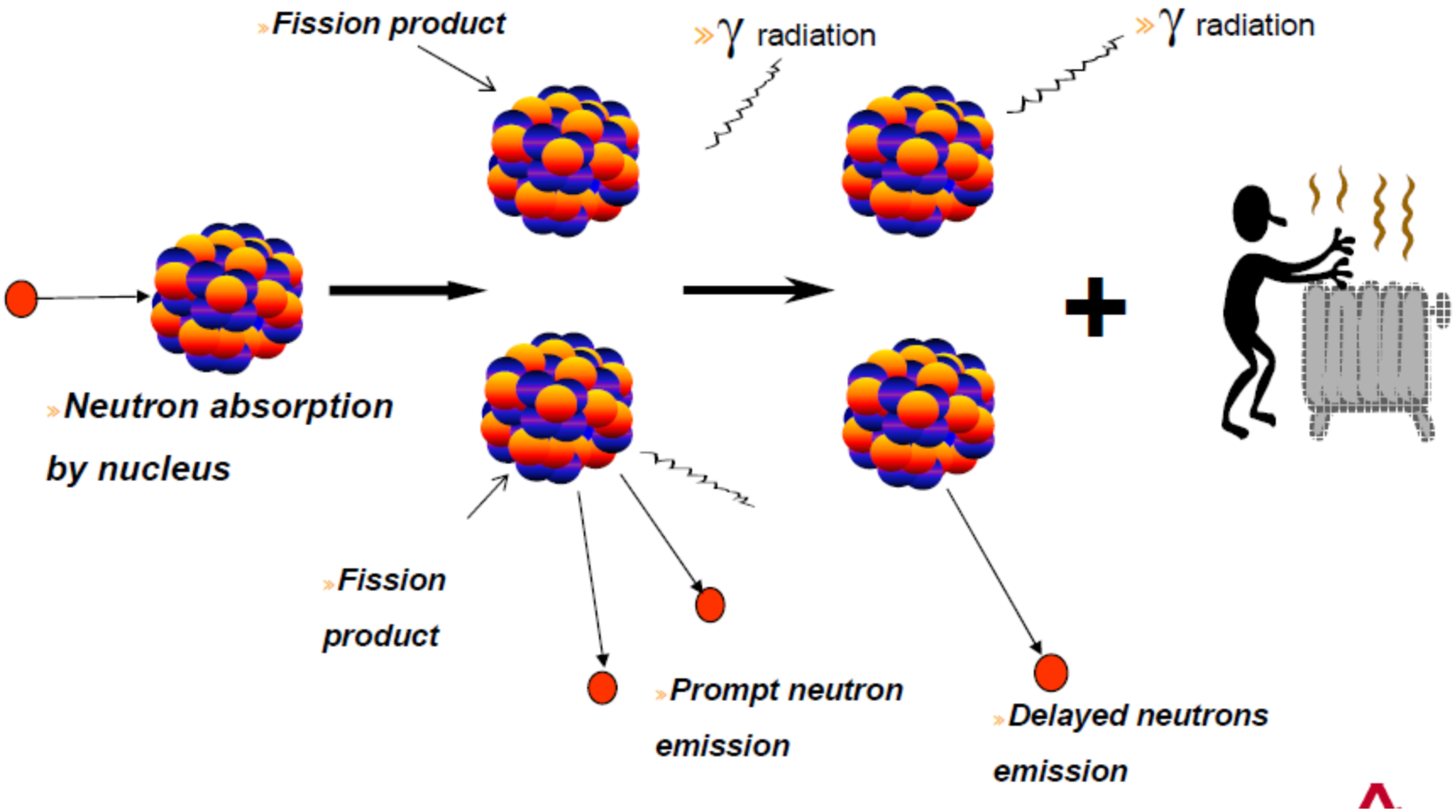
1st contributor: Pu ; 2nd contributor: Minor Actinides (MA)

Greenhouse gas emissions (in tonnes of CO_2-equivalent) per GWh for different electricity production means

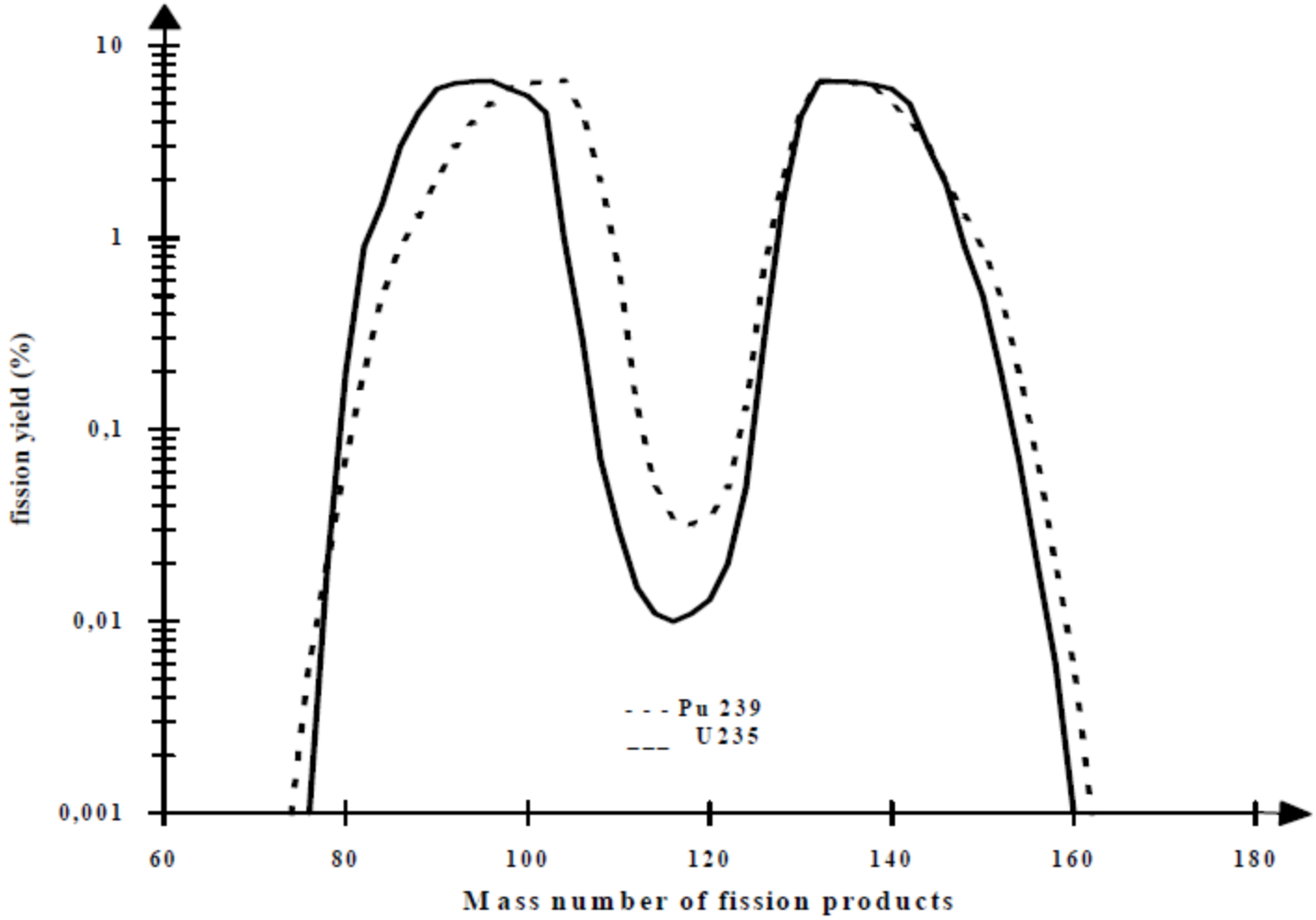


What is Accelerator-Driven Transmutation?
ATW Principal Components

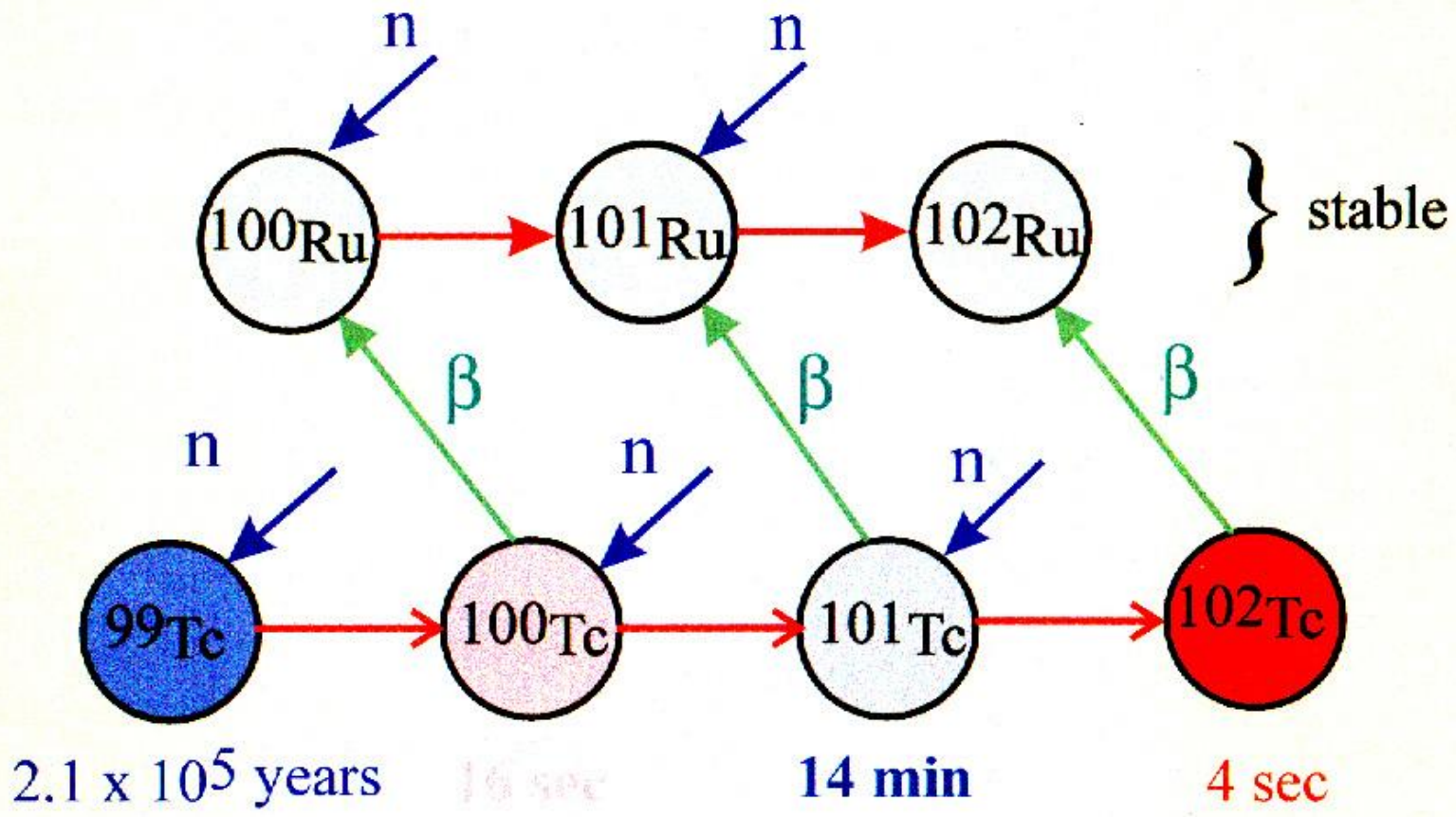



Fission product
γ radiation
γ radiation
Neutron absorption by nucleus
Fission product
Prompt neutron emission
Delayed neutrons emission

Fission products emitted and classified by their mass number



Transmutation of technetium



Nuclear waste composition (1 GW_e LWR)

$^{244, 245}$Cm
1.5 Kg/yr

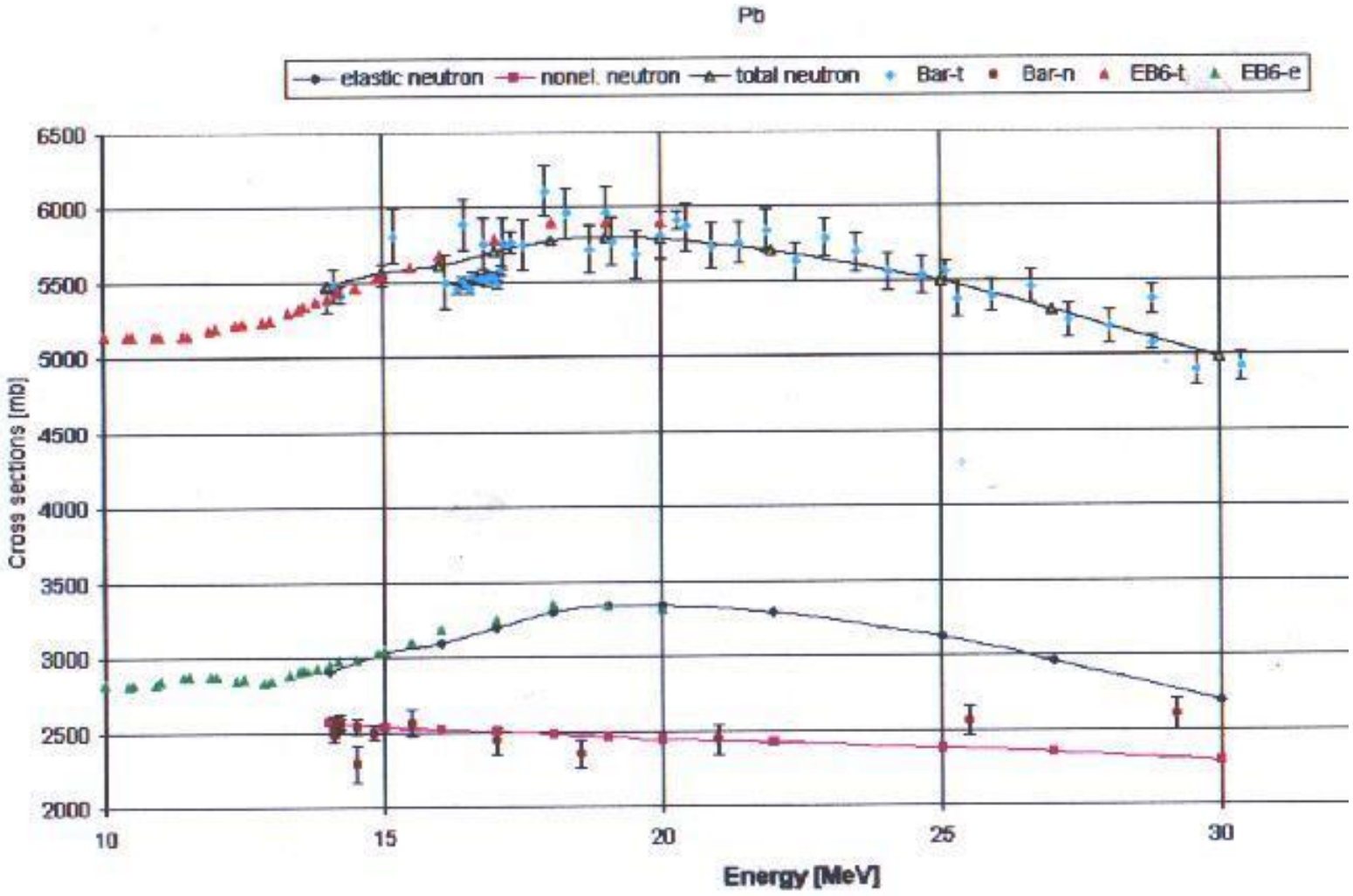
^{241}Am:11.6 Kg/yr
^{243}Am: 4.8 Kg/yr

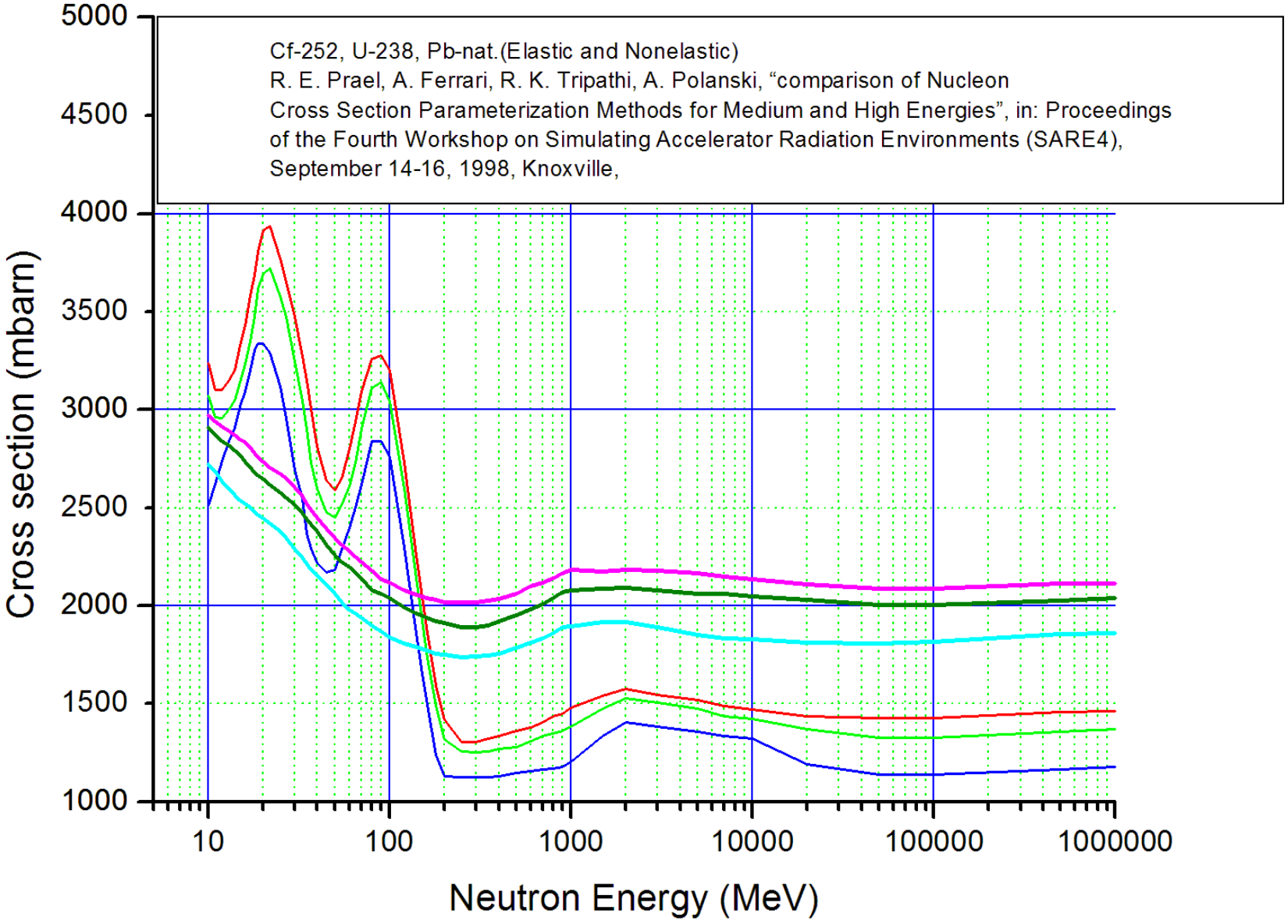
^{239}Pu: 125 Kg/yr

^{237}Np: 16 Kg/yr

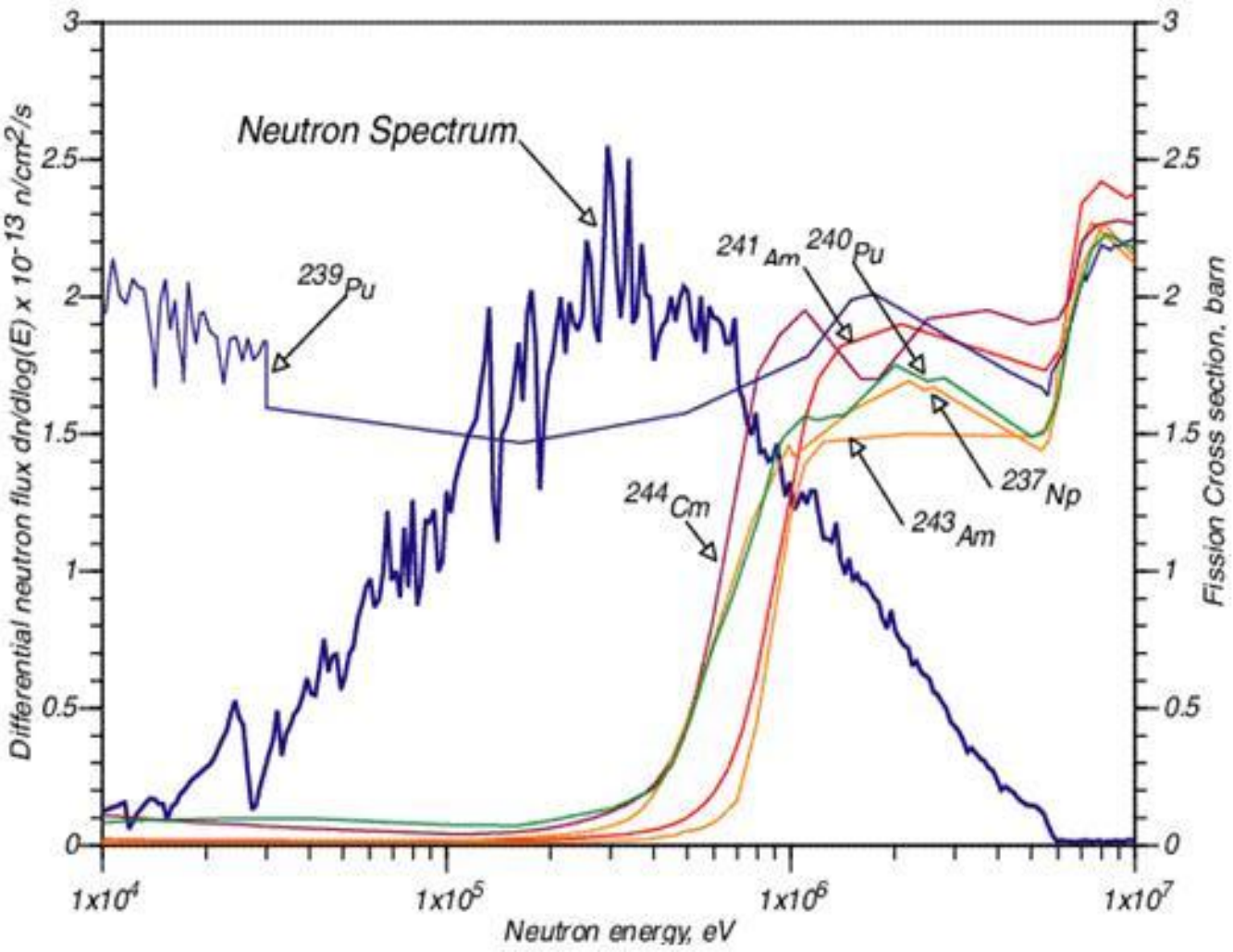
LLFP
76.2 Kg/yr

Neutron Cross Sections




Cf-252, U-238, Pb-nat.(Elastic and Nonelastic)
R. E. Prael, A. Ferrari, R. K. Tripathi, A. Polanski, “comparison of Nucleon Cross Section Parameterization Methods for Medium and High Energies”, in: Proceedings of the Fourth Workshop on Simulating Accelerator Radiation Environments (SARE4), September 14-16, 1998, Knoxville,
Cross section (mbarn)
5000
4500
4000
3500
3000
2500
2000
1500
1000
10
100
1000
10000
100000
1000000
Neutron Energy (MeV)

Fission Cross Sections



The fission rate: some orders of magnitude of cross sections for a thermal neutron fission

	σ_a (barn)	σ_f (barn)	ν	η
^{233}U	578.8	531.1	2.492	2.278 (2.31 rap)
^{235}U	680.8	582.2	2.418	2.068 (2.18 rap)
^{239}Pu	1011.3	742.5	2.871	2.108 (2.74 rap)
^{241}Pu	1377	1009	2.927	2.145
^{232}Th	7.56			
^{238}U	2.73			0 (1.07 rap)
^{240}Pu	286			
^{242}Pu	30			

Fission and capture cross sections

- α ratio = capture/fission non favorable in thermal spectrum

Isotope	ϕ : Low neutrons reactor (PWR)			ϕ : Fast neutrons reactor (FR)		
	σ_f	σ_c	α	σ_f	σ_c	α
^{235}U	38,8	8,7	0,22	1,98	0,57	0,29
^{238}U	0,103	0,86	8,3	0,04	0,30	7,5
^{239}Pu	102	58,7	0,58	1,86	0,56	0,3
^{240}Pu	0,53	210,2	396,6	0,36	0,57	1,6
^{241}Pu	102,2	40,9	0,40	2,49	0,47	0,19
^{242}Pu	0,44	28,8	65,5	0,24	0,44	1,8
^{237}Np	0,52	33	63	0,32	1,7	5,3
^{241}Am	1,1	110	100	0,27	2,0	7,4
^{243}Am	0,44	49	111	0,21	1,8	8,6
^{244}Cm	1,0	16	16	0,42	0,6	1,4
^{245}Cm	116	17	0,15	5,1	0,9	0,18

Models and Codes for Particle Transport

Models

- Optical , microscopic
- Quantum molecular dynamic
- Cascade-evaporation
- Dual Parton and Glauber

Transport Codes

- GEANT (CERN)
- Cascade (Dubna)
- MCNPX (USA)
- FLUKA (Italy)
- TRIPOLI(France)

Proton (ion) ransport is described by the classical linearBoltzmann or transport equation:

$$\Omega \Delta \psi(\vec{r}, E, \Omega) = J_{nucl}[\psi] + J_{elec}[\psi]$$

Where $\psi(\boldsymbol{r}, E, \boldsymbol{\Omega})$ is the ion angular flux, and $J_{nucl}[\psi]$ and $J_{elec}[\psi]$ are collision integrals describing, respectively, nuclear and electronic interactions.

$$J_{nucl}[\psi] = \int dE' \int_{4\pi} d\Omega' \sigma_s(E' \rightarrow E, \Omega', \Omega)\psi(\vec{r}, E', \Omega') - \sigma\ (E)\psi(\vec{r}, E, \Omega)$$

$\sigma_s(E' \rightarrow E, \boldsymbol{\Omega'.\Omega})$ is the ion/target atom double differential scattering cross section and $\sigma(E)$ the total cross section.

Fokker-Planck approximation to $J_{elec}[\psi]$

$$J_{elec}[\psi] = \frac{\partial}{\partial E}[S\ (E)\psi] + \frac{1}{2}\frac{\partial^2[T(E)\psi]}{\partial E^2}$$

Where $S(E)$ is the stopping power and $T(E)$ is the straggling coefficient.

MCNP/MCNPX

- MCNP is a code developed by LANL to simulate the transport of neutrons, gamma rays and electrons by the Monte Carlo method. It simulates a coupled transport, i.e., it also accounts for transport of secondary particle resulting the interaction of primary particles.
- MCNPX extends the capacilties of MCNP to other particles (e.g. charged particles, heavy ions, pions etc.)

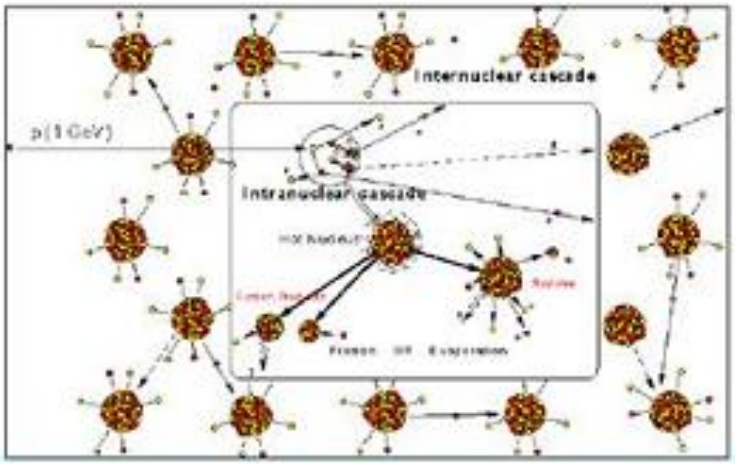
Problems That Can Be Solved/fields of Application

- **Reactor calculations**
- Burnup
- Criticality
- Current
- Fluence
- Energy deposition
- Medical application
- Detector
- High energy physics (MCNPX)
- Space applications

MCNPX

- Structure of the input file
  - Cell
  - Surfaces
  - Materials (XS)
  - Source definition
  - Tally
  - Additional options
- Output files
  - Binary
  - Out file
  - Other files

Simulation of Spallation



Neutrons produced by a **series of nuclear reactions** (intranuclear cascade, preequilibrium, evaporation, etc…)

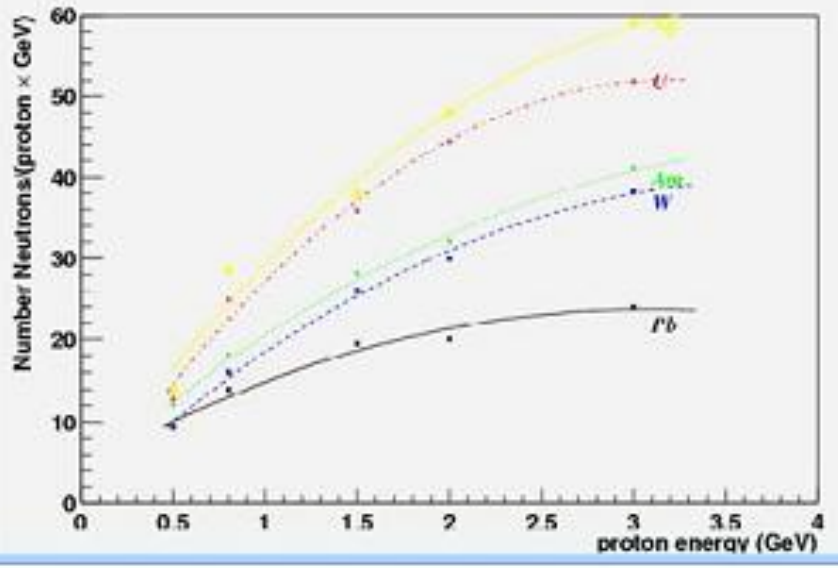
Need **high-energy proton** beams

Large volume spallation targets

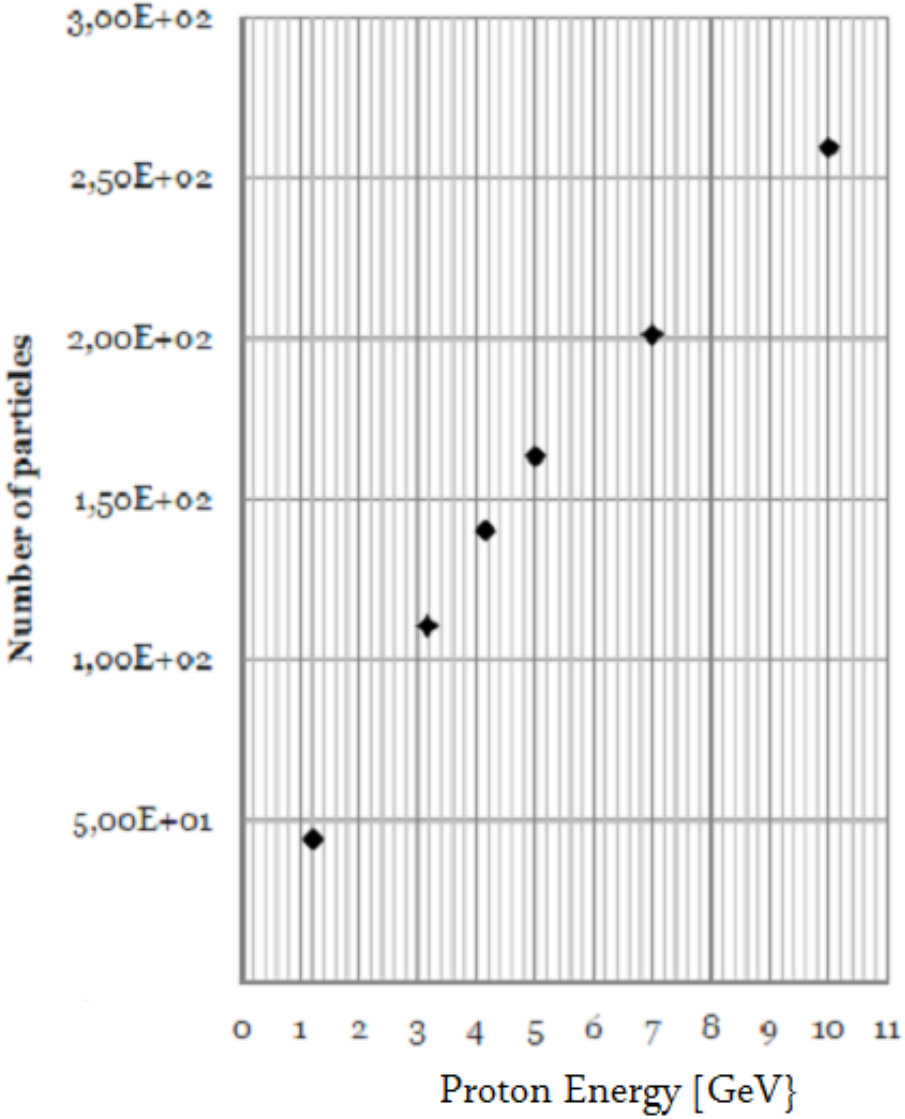
Neutron production depends on **atomic weight** and **density** of the material ➡ high-Z material needed

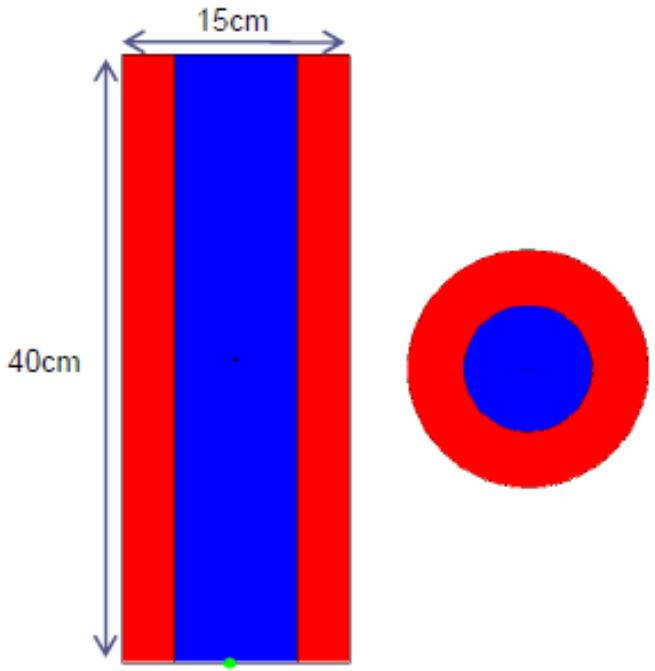
Depends on **proton energy**

Choice of **target** depends also on other factors (radiation resistance, costs, etc…)

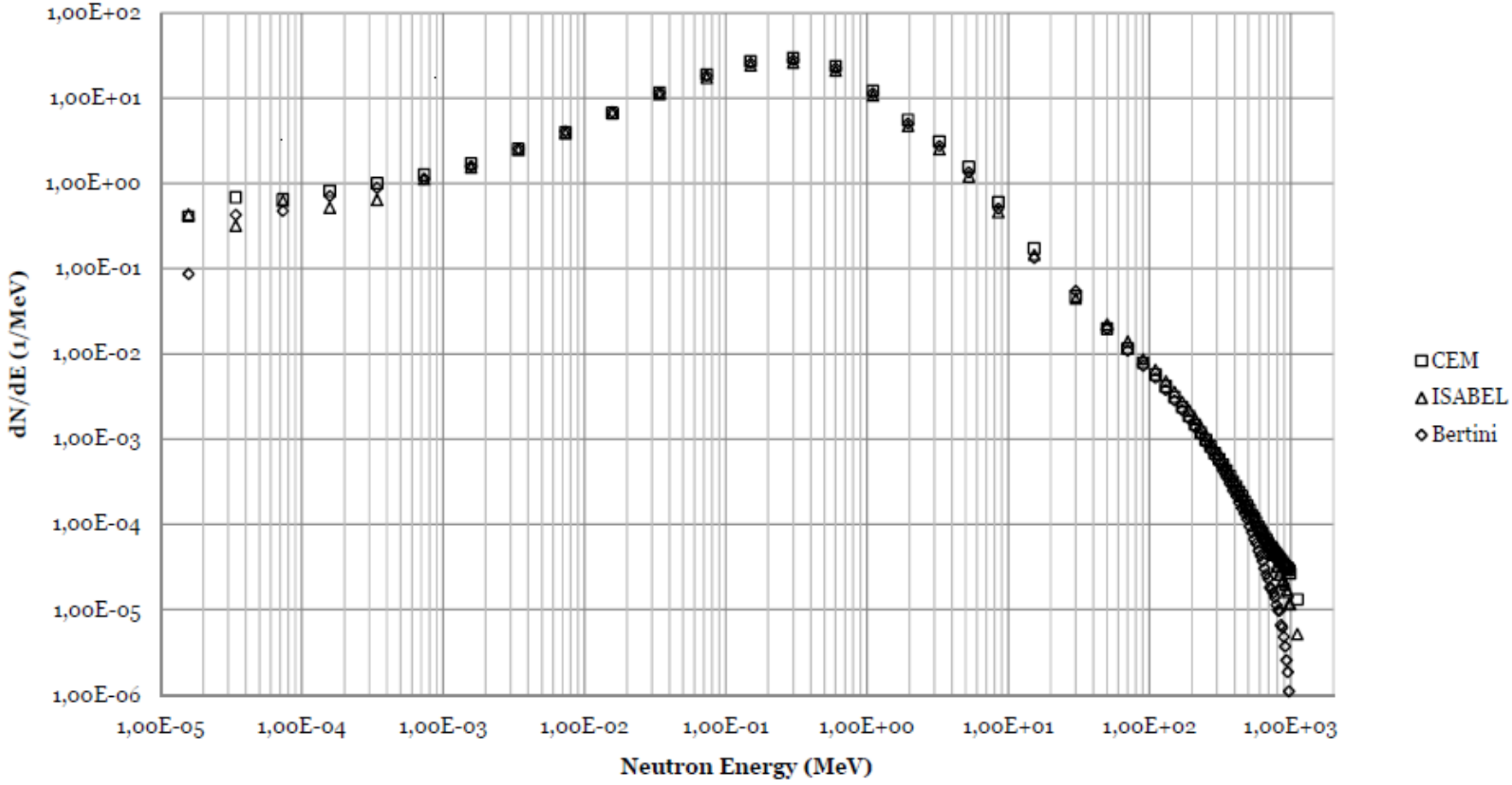


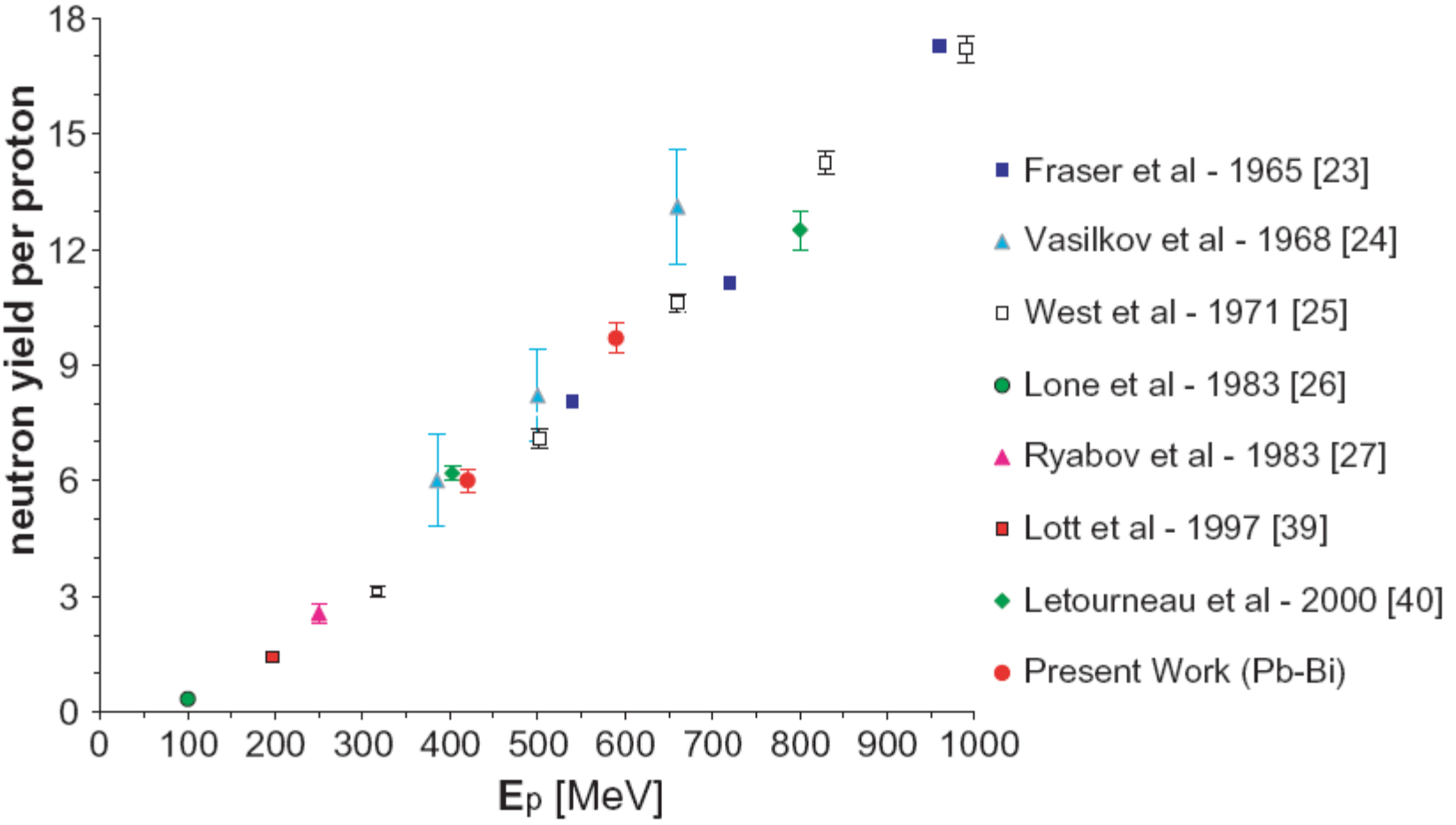
Spalation neutrons from uranium target



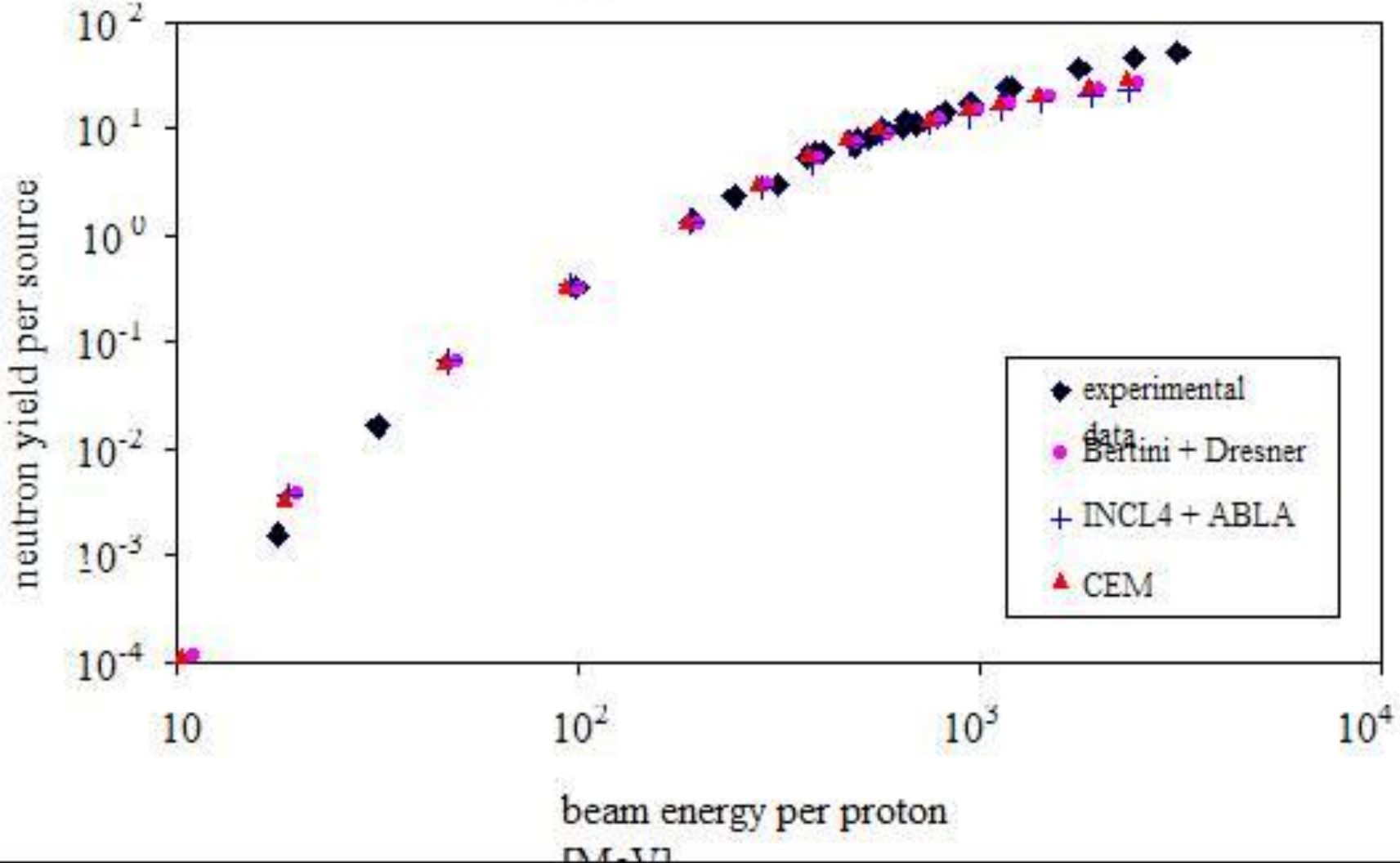


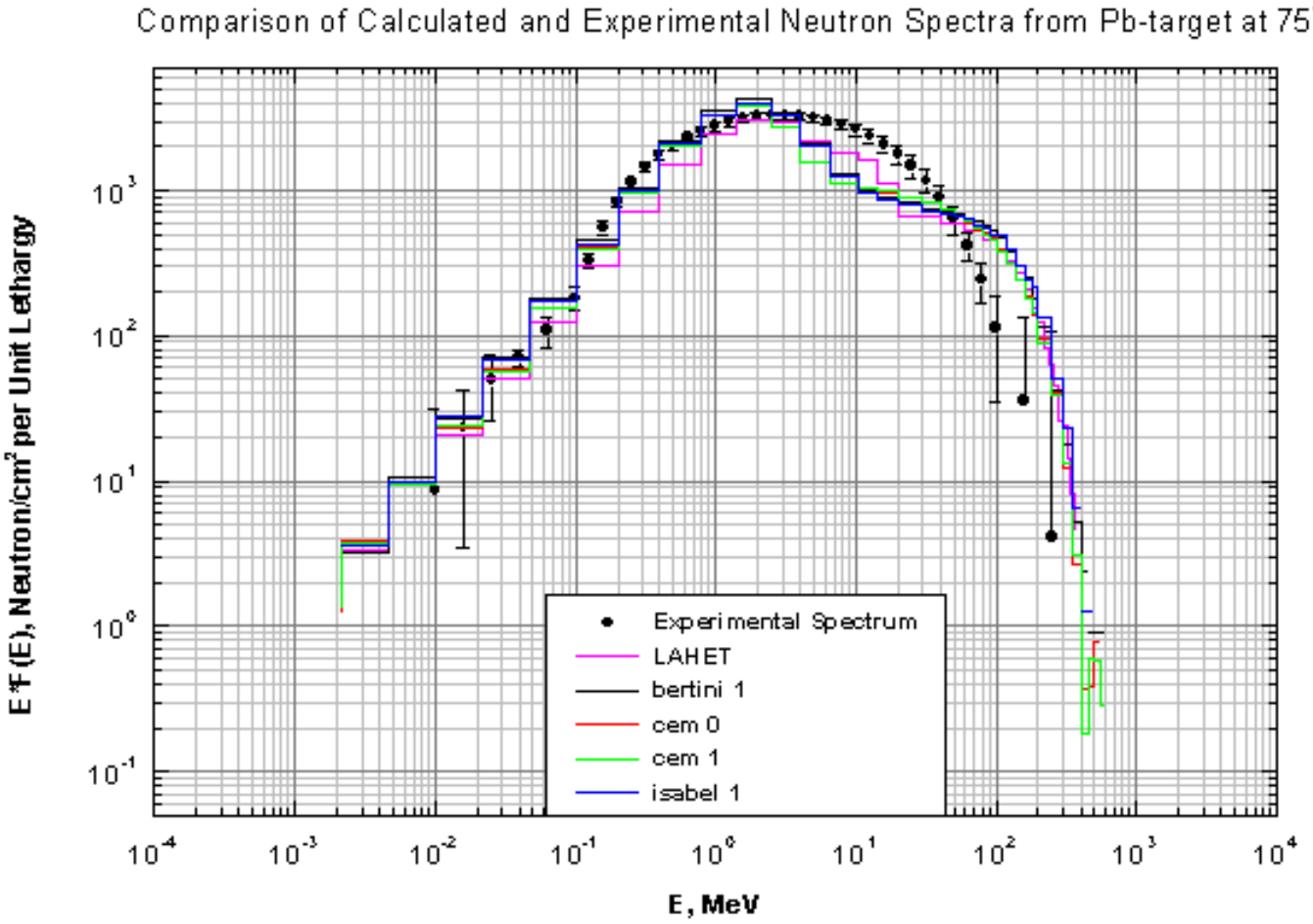
Neutron energy spectrum for differents model, proton kinetic energy 1,22 GeV



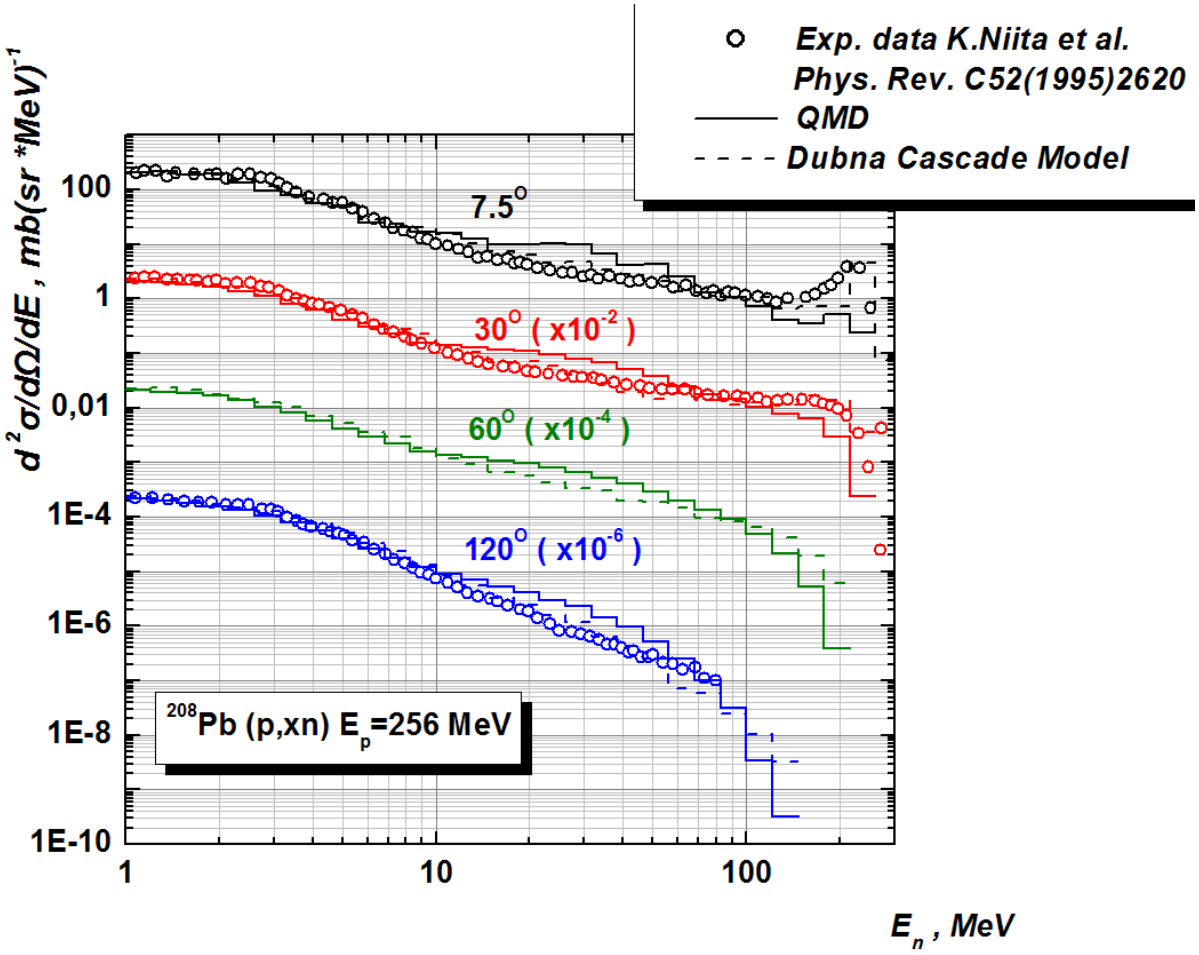


Compilation of thick-target n/p values for p + Pb and Pb/Bi measured to date at intermediate-E_p.

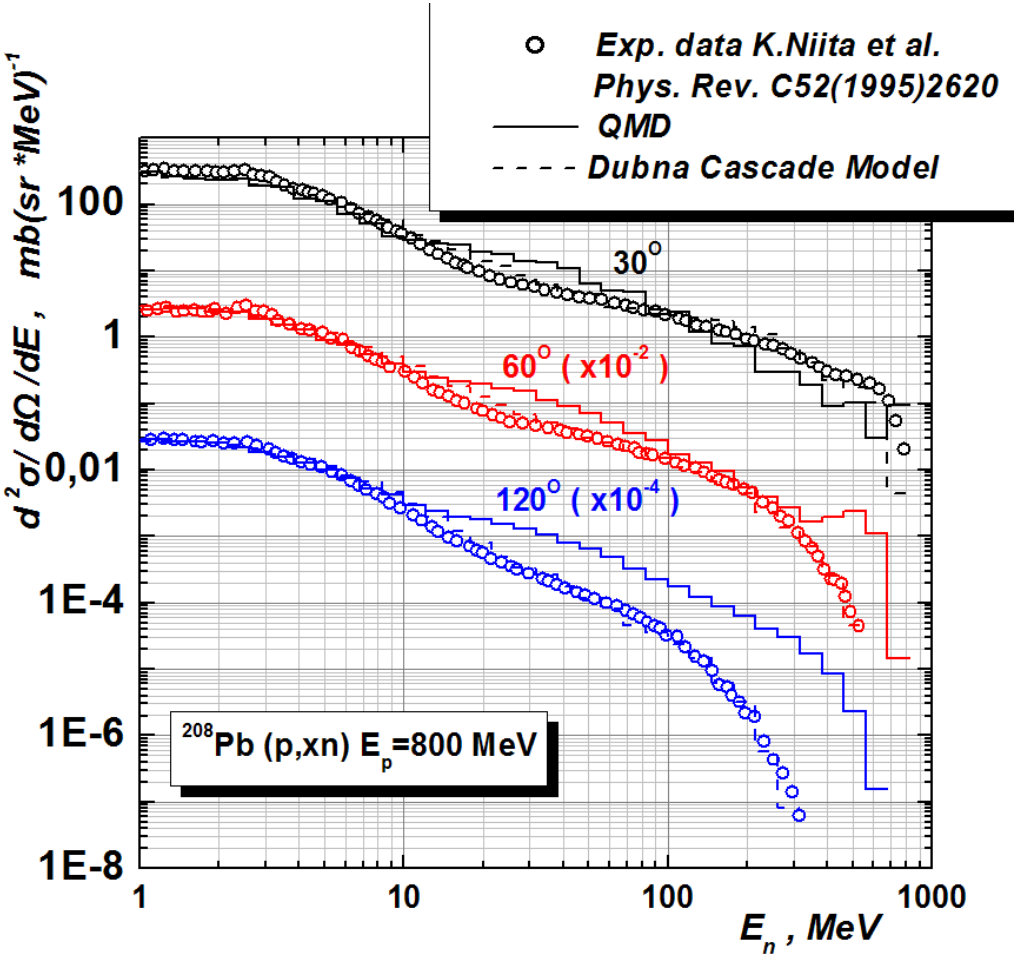

Neutron yield (log scale)
neutron yield per source
10^{2}
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10
10^{2}
10^{3}
10^{4}
beam energy per proton
experimental data
Bertini + Dresner
INCL4 + ABLA
CEM

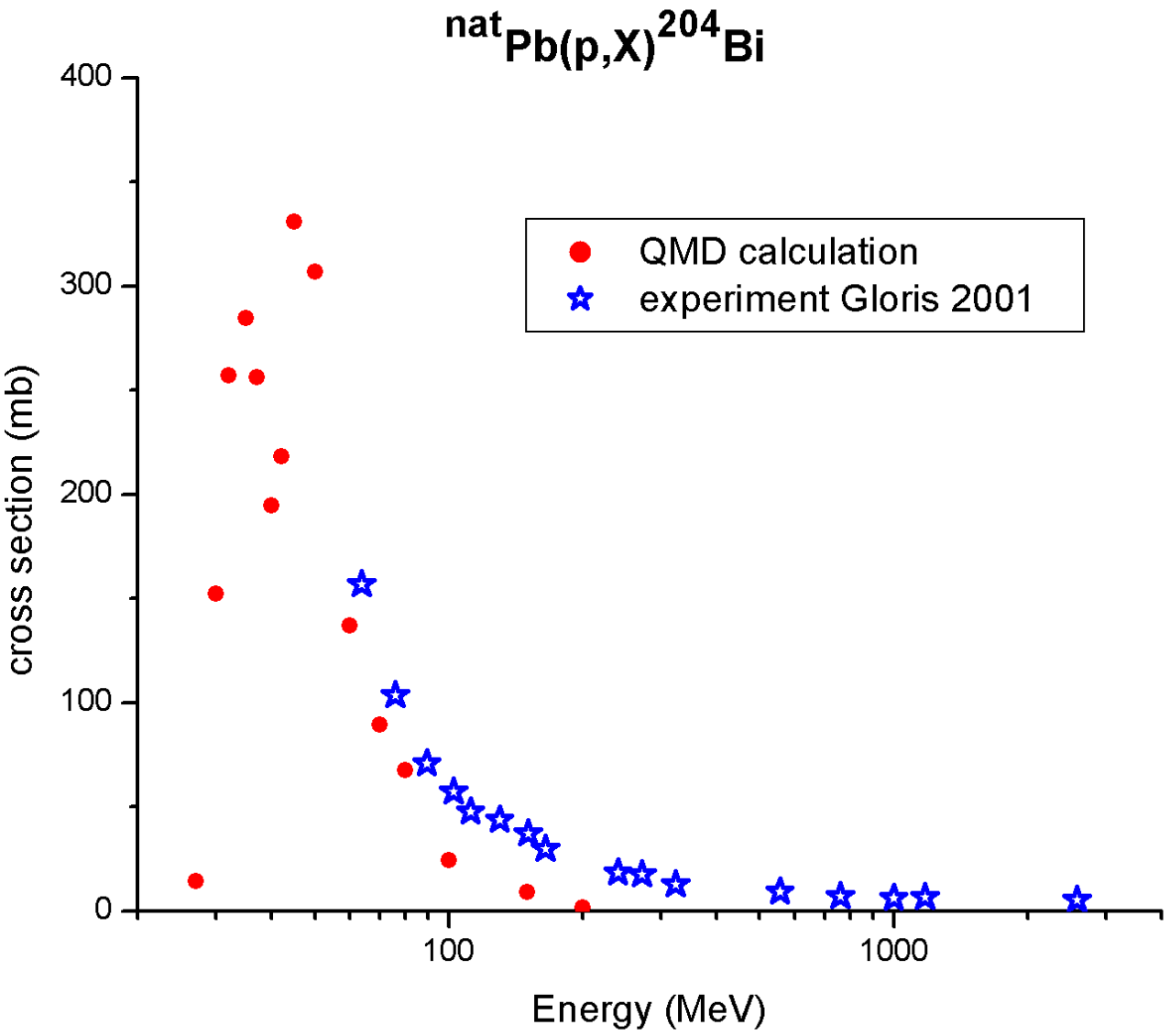

Comparison of Calculated and Experimental Neutron Spectra from Pb-target at 75°
E*F(E), Neutron/cm² per Unit Lethargy
E, MeV
Experimental Spectrum
LAHET
bertini 1
cem 0
cem 1
isabel 1

Cross sections

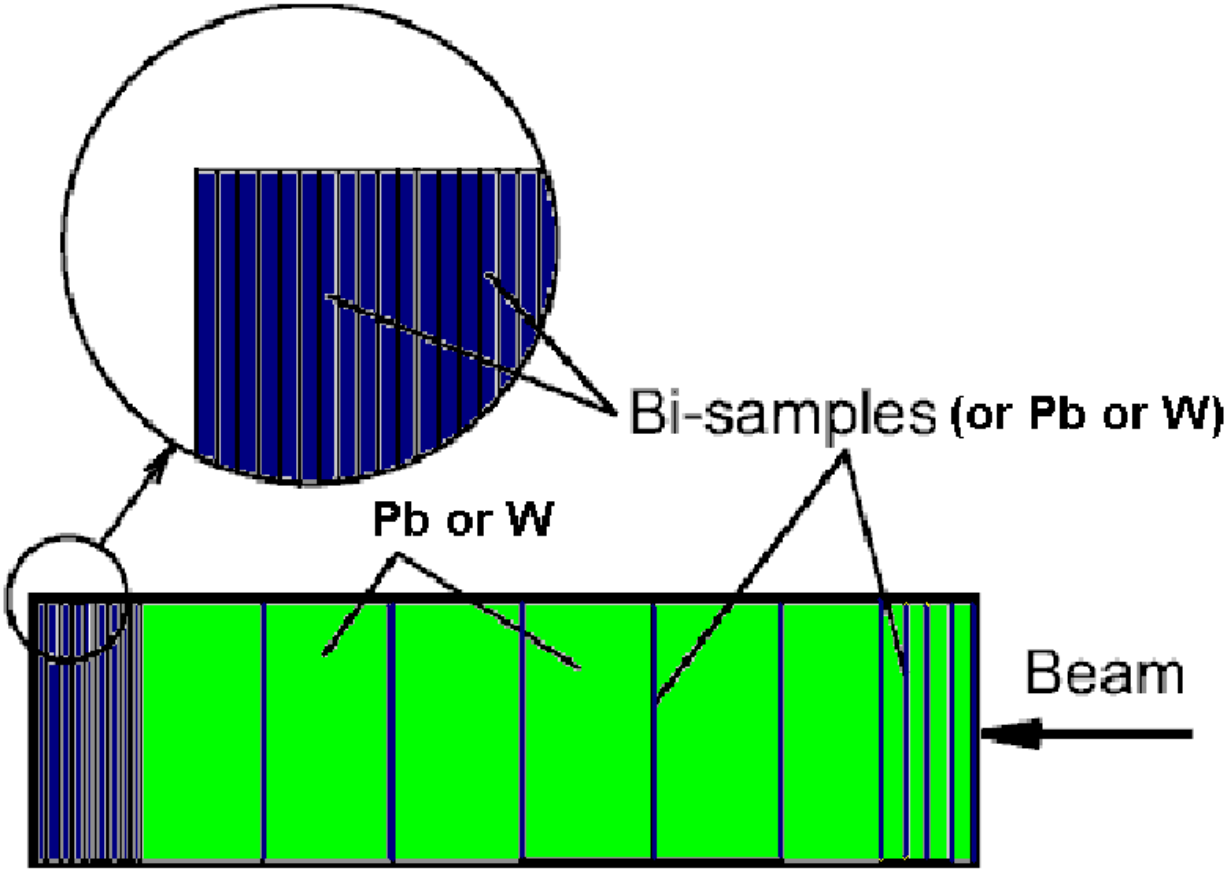


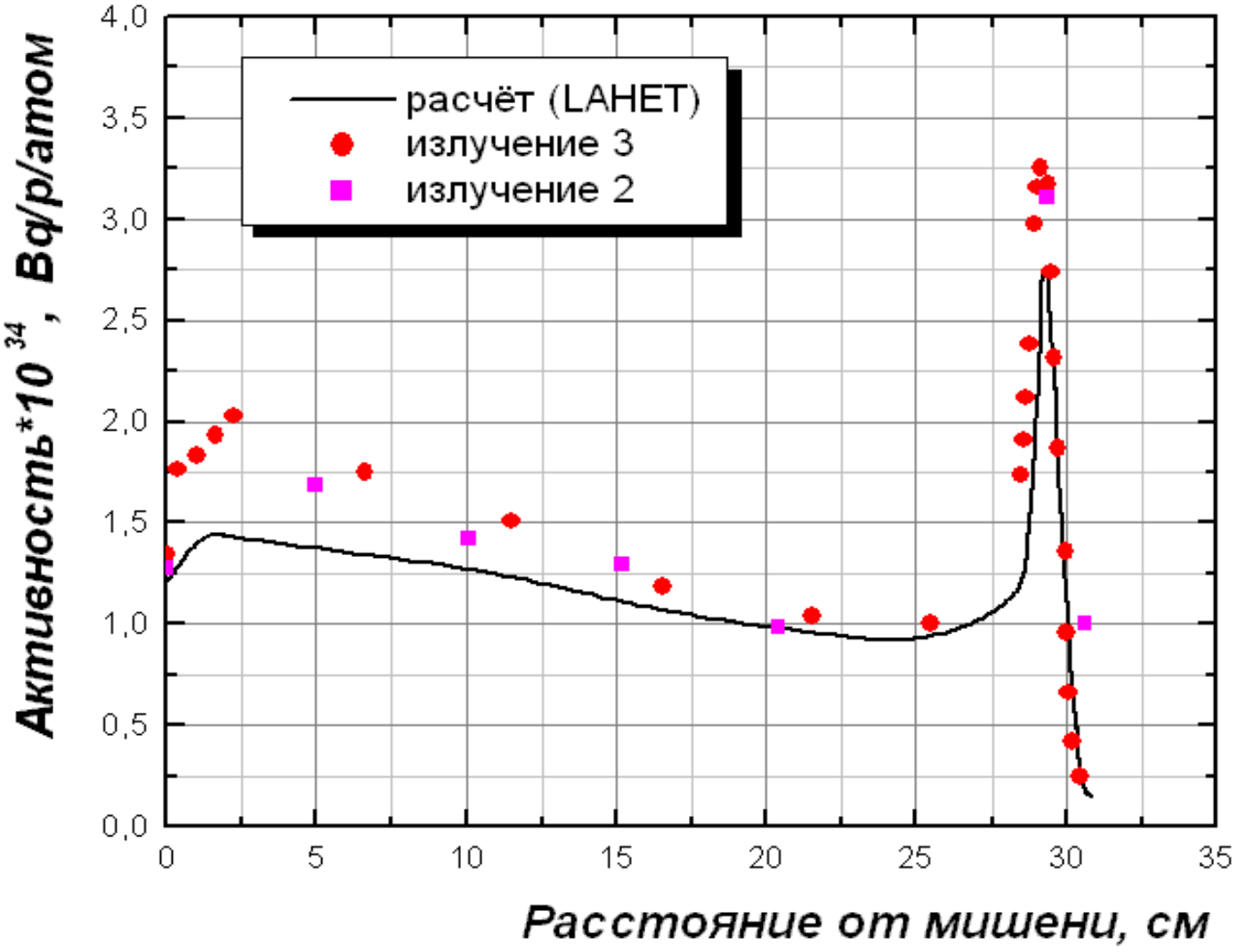
Cross sections



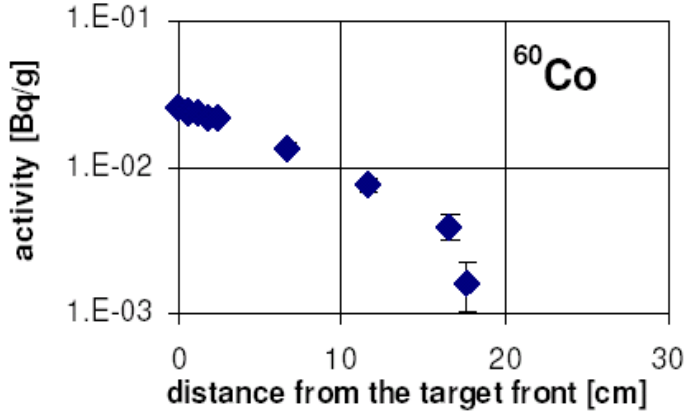

$^{nat}Pb(p,X)^{204}Bi$
QMD calculation
experiment Gloris 2001
cross section (mb)
Energy (MeV)
0
100
200
300
400
100
1000

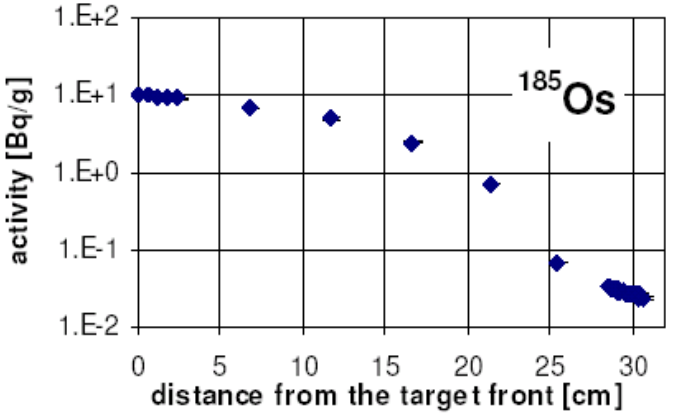
Measurement of generation and distribution of bismuth isotopes in the volume of lead target

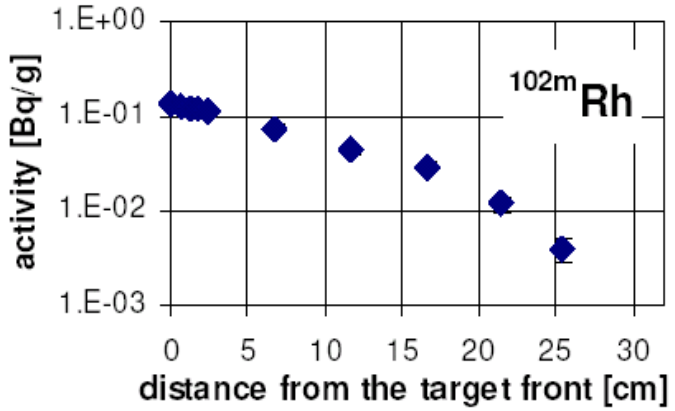


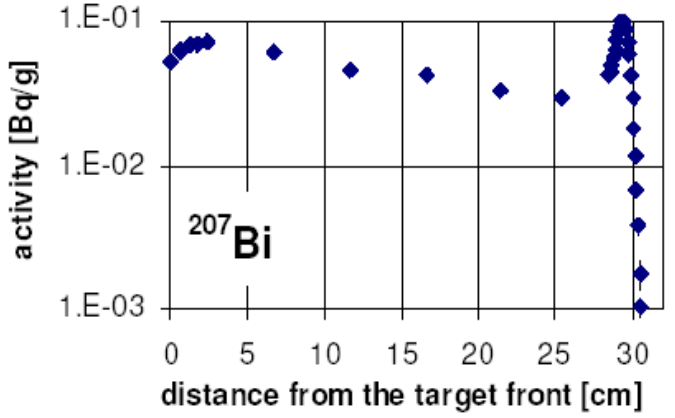

расчёт (LAHET)
излучение 3
излучение 2
Активность*10^34, Bq/p/atom
Расстояние от мишени, см
0,0
0,5
1,0
1,5
2,0
2,5
3,0
3,5
4,0
0
5
10
15
20
25
30
35

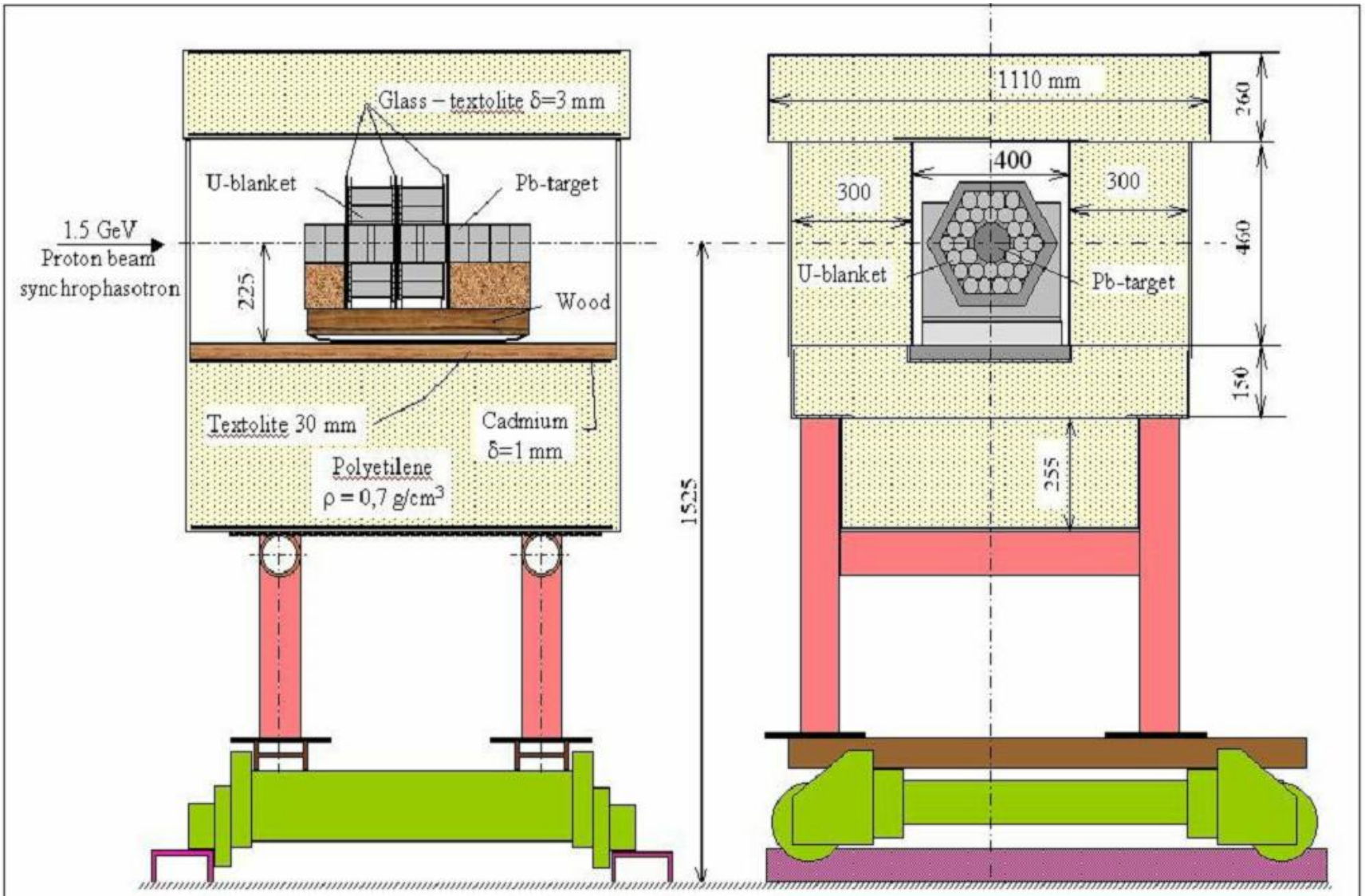
Distributions of the specific activity of radionuclides along the Pb target irradiated with 660 MeV protons.



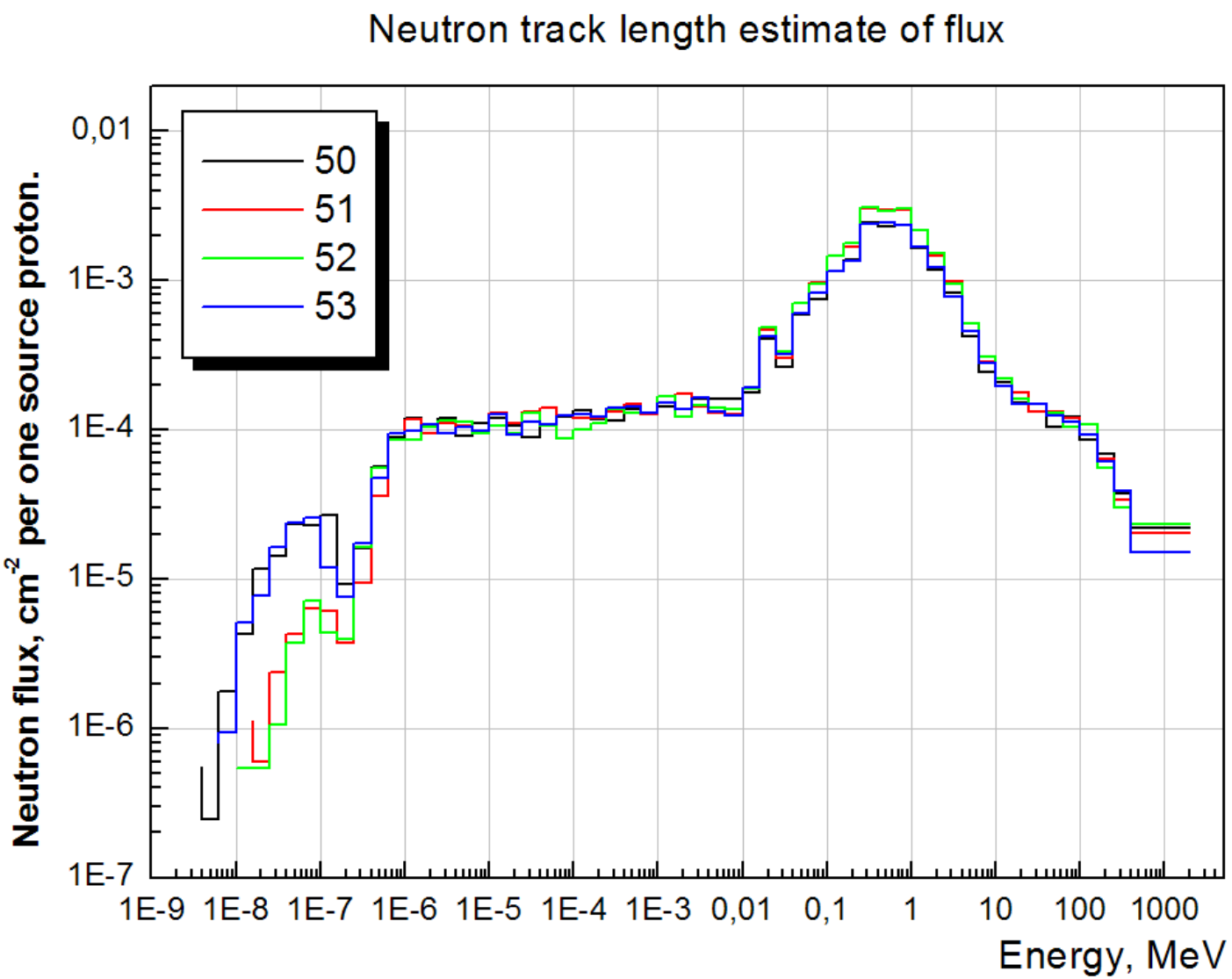




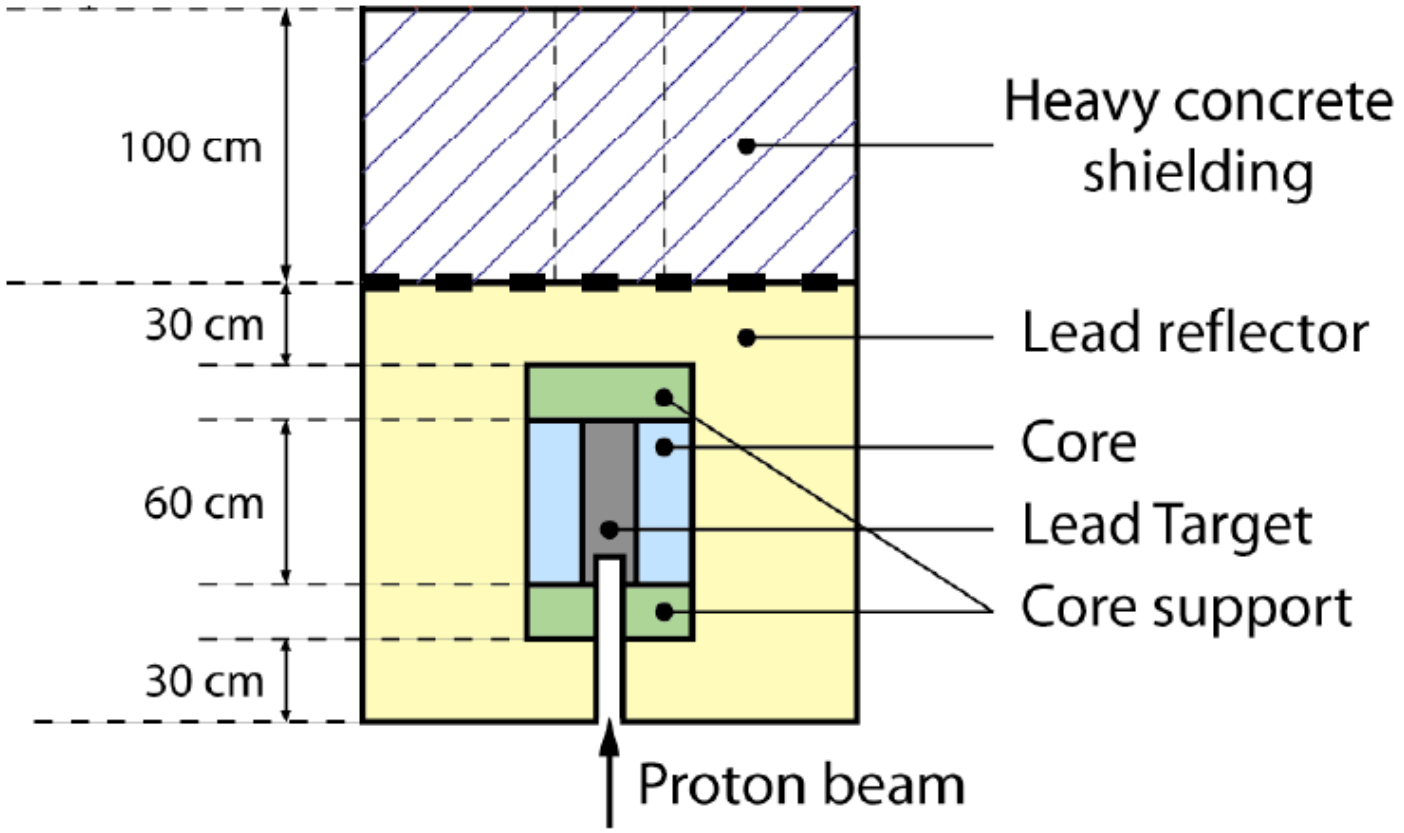




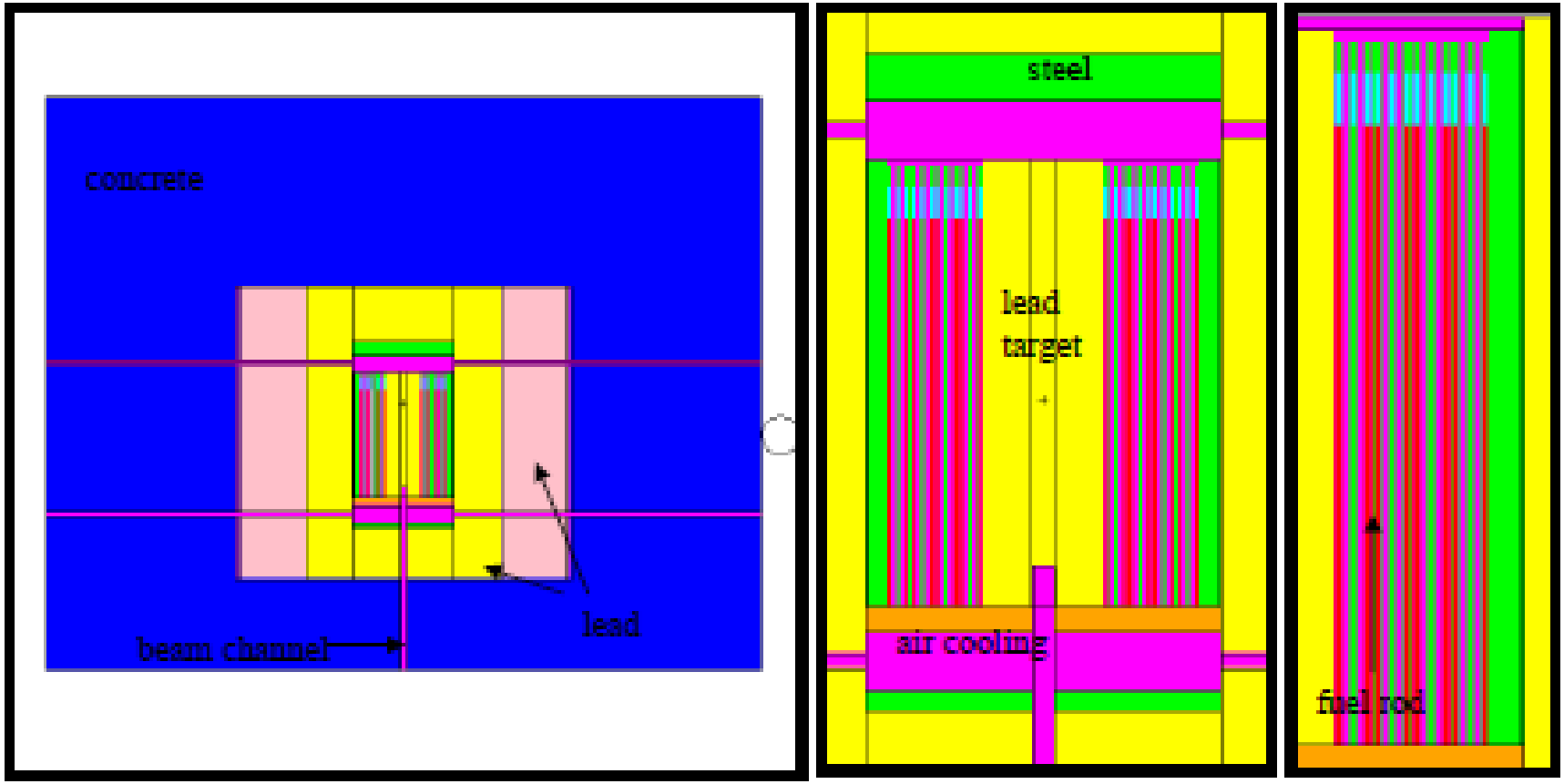
A scheme of the prototype electronuclear installation "Energy plus Transmutation" to study the model U/Pb assembly using the proton beam from the synchrophasotron (JINR, Dubna) at 1.5 GeV (RUN -November 1999)


Neutron track length estimate of flux
Neutron flux, cm-2 per one source proton.
0,01
1E-3
1E-4
1E-5
1E-6
1E-7
50
51
52
53
1E-9
1E-8
1E-7
1E-6
1E-5
1E-4
1E-3
0,01
0,1
1
10
100
1000
Energy, MeV

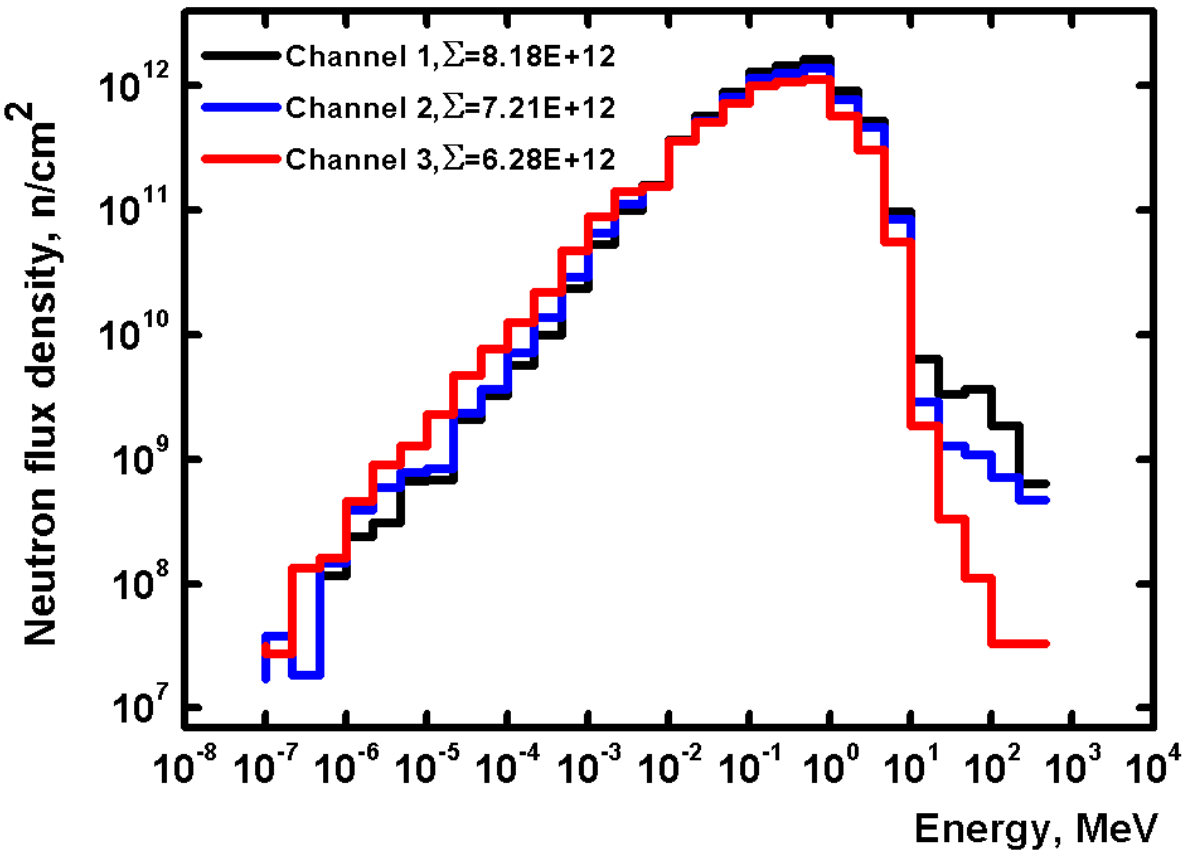
Experimental Subcritical Reactor



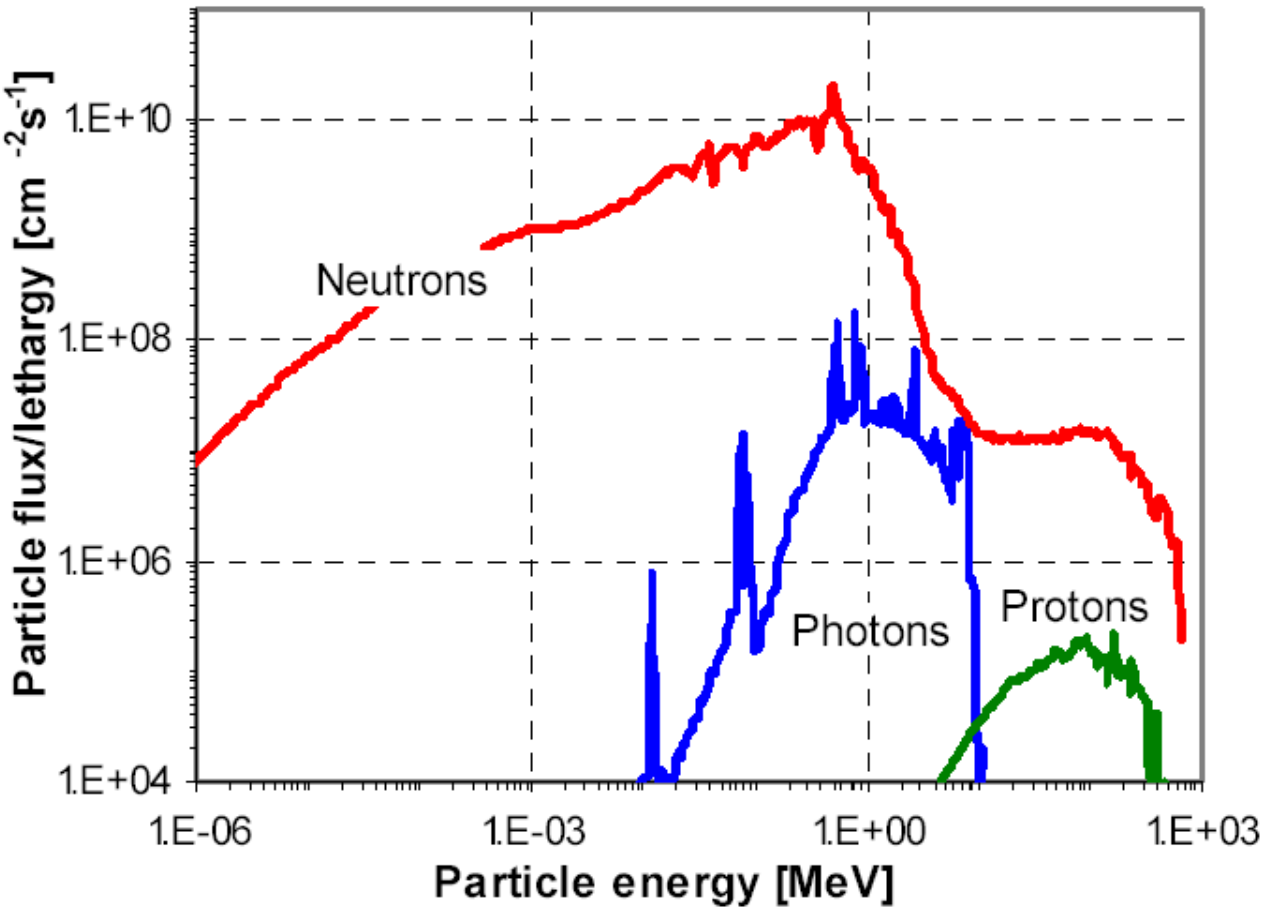
Subcritical Reactor Geometry



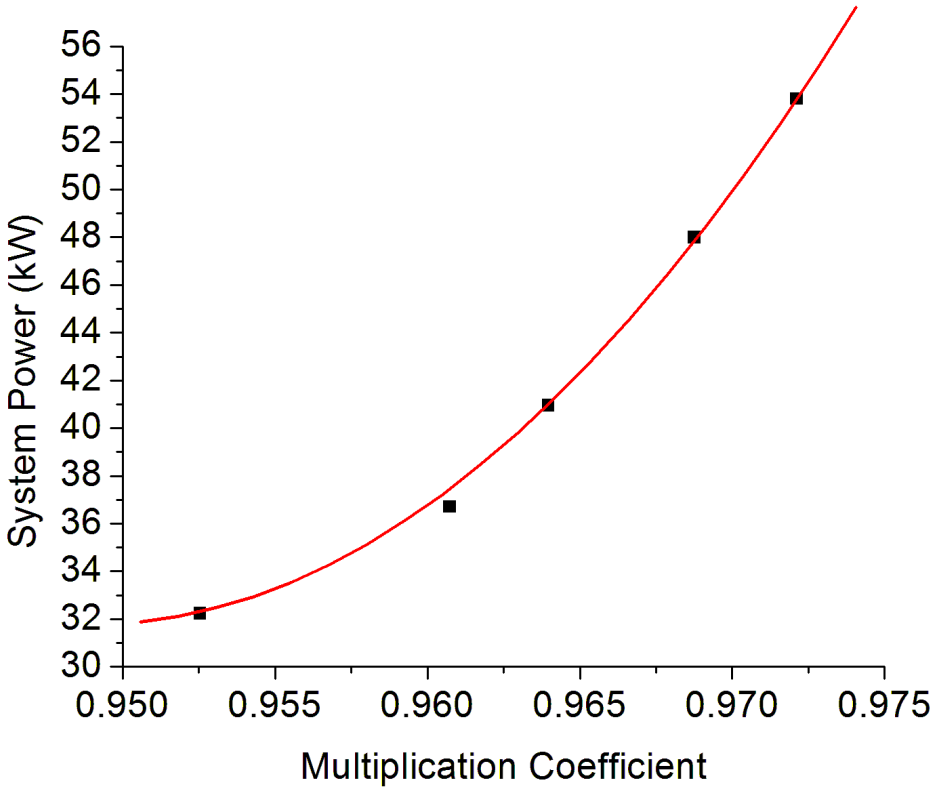
Neutron spectra in the centers of vertical experimental channels 1-3 for subcritical assembly



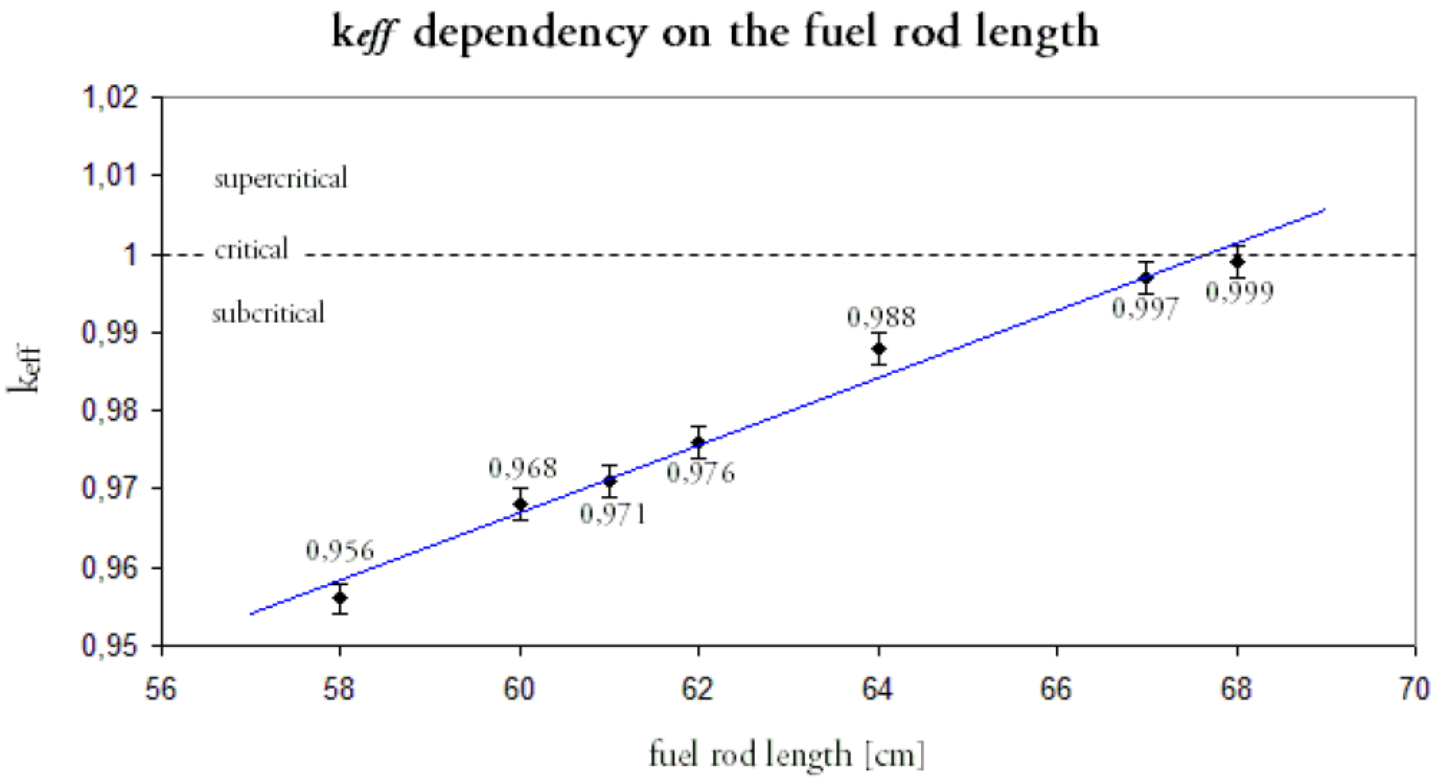
Spectrum behind concrete. Effective dose is 190 μSv/h



System Power Vs Multiplication Factor for 1 kW Proton Beam Power



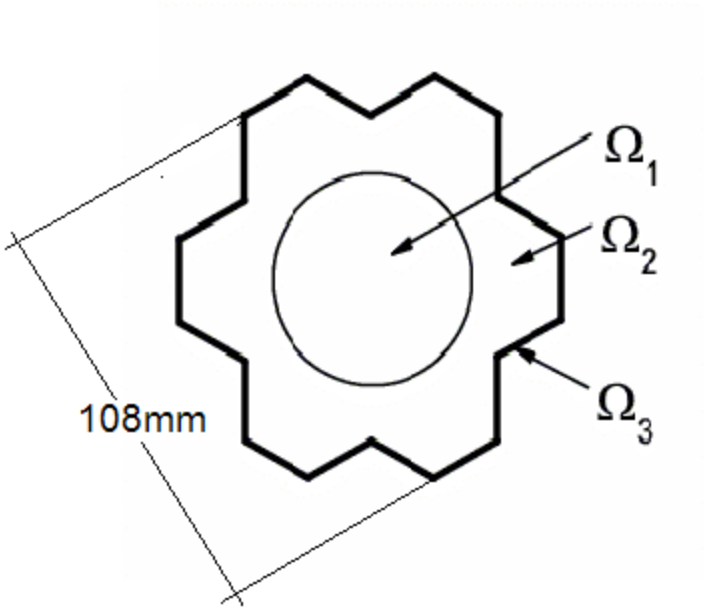
K-eff


keff dependency on the fuel rod length
1,02
1,01
1
0,99
0,98
0,97
0,96
0,95
keff
supercritical
critical
subcritical
0,956
0,968
0,971
0,976
0,988
0,997
0,999
56
58
60
62
64
66
68
70
fuel rod length [cm]

Activity Evolution in SAD Target

- Calculations of formation of radioactive isotopes in each area of a target. (mcnpx)
- Calculations of activity of isotopes at the interesting moment of time in each area of a target. (TEA; JEF 2.2)
- Calculations of gamma ray spectrum at the interesting moment of time. (JEF 2.2)
- Calculations of the contribution in an equivalent dose in meter from a target from each area of a target (mcnp).

Subdivision zones (Ω_1, Ω_2, Ω_3) used for modeling.



The target was subdivided into three zones: central cylindrical area with diameter 58 mm and length 179 mm (Ω_1), second layer (Ω_2) – includes the remaining lead and the target casing made of two-millimeter stainless steel which comprises the external third layer (Ω_3). Total mass of the target is 52 kilograms; mass of the lead block is 7.7 kg.

Calculation of Activity Evolution

$$\begin{cases} \frac{dN_i(t)}{dt} = \Lambda_i^t + \sum\limits_{k \neq i} \lambda_{ik}^r N_k(t) - \lambda_i^d N_i(t), \quad i \in [1, I], \\ N_i(t_0) = N_i^0, \quad i \in [1, I], \end{cases}$$

To determine density function and activity for each isotope at any arbitrary moment of time for the chosen operation mode it is necessary to solve the Cauchy problem for the set of kinetic equations (*i*=1, 2, … *I*) at the finite time segment t [t_{min}, t_{max}].
Where: I – number of isotopes considered in the task.
$N_i(t)$ – number of nuclei of the *i*-th type at the moment of time *t*,
Λ_i^t - generation rate for the *i*-th nuclide in nuclear reactions,

λ_i^d - probability of nuclear decay for such nucleus per unit time;

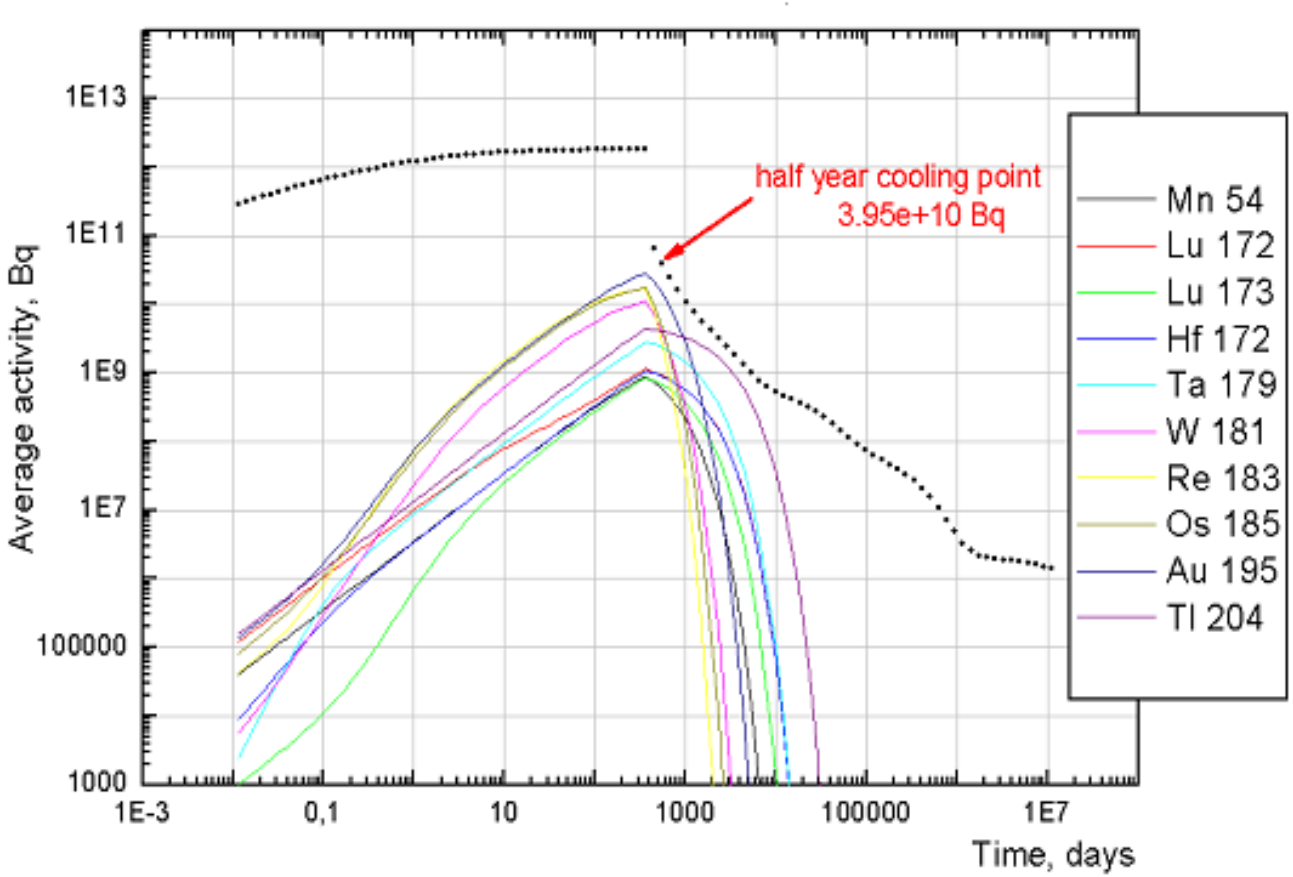
λ_{ik}^r -probability of a nuclear decay for a nucleus of the *k*-th nuclide when it is converted into the nucleus of the *i*-th nuclide;

Methods used for calculations.

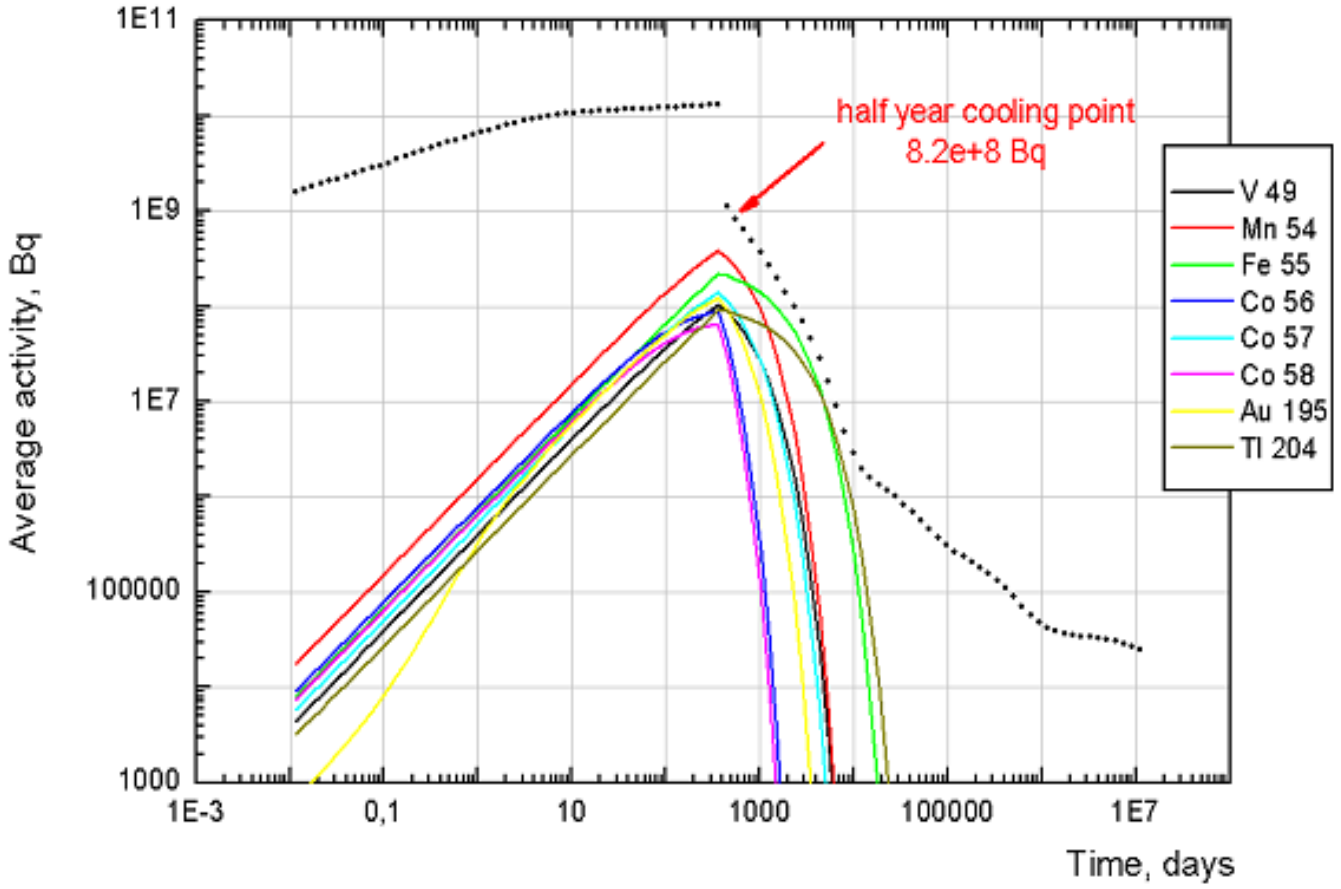
To solve the task in the central zone (Ω_1) we had to consider 1311 radioactive isotopes with non-zero at start time. During the calculation the number of isotopes increases up to 1328. The program calculates activity of the system during the time interval from t_1=1000 seconds to $t_{max}\approx$ 27 years. As a result of calculations we received 100 activity points along the time variable equidistant in logarithmic scale.]. Data library for radioactive decay JEF 2.2 was used in calculations.

Results of calculations of activity

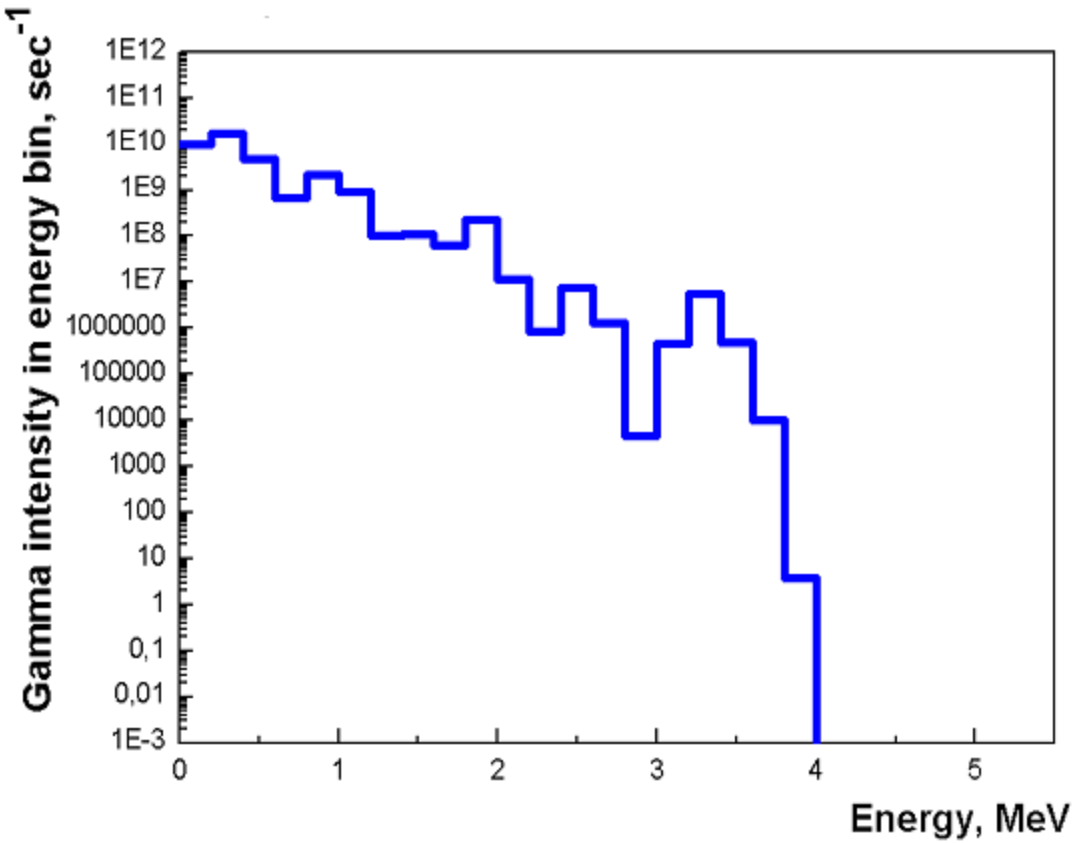
Total activity of the lead target



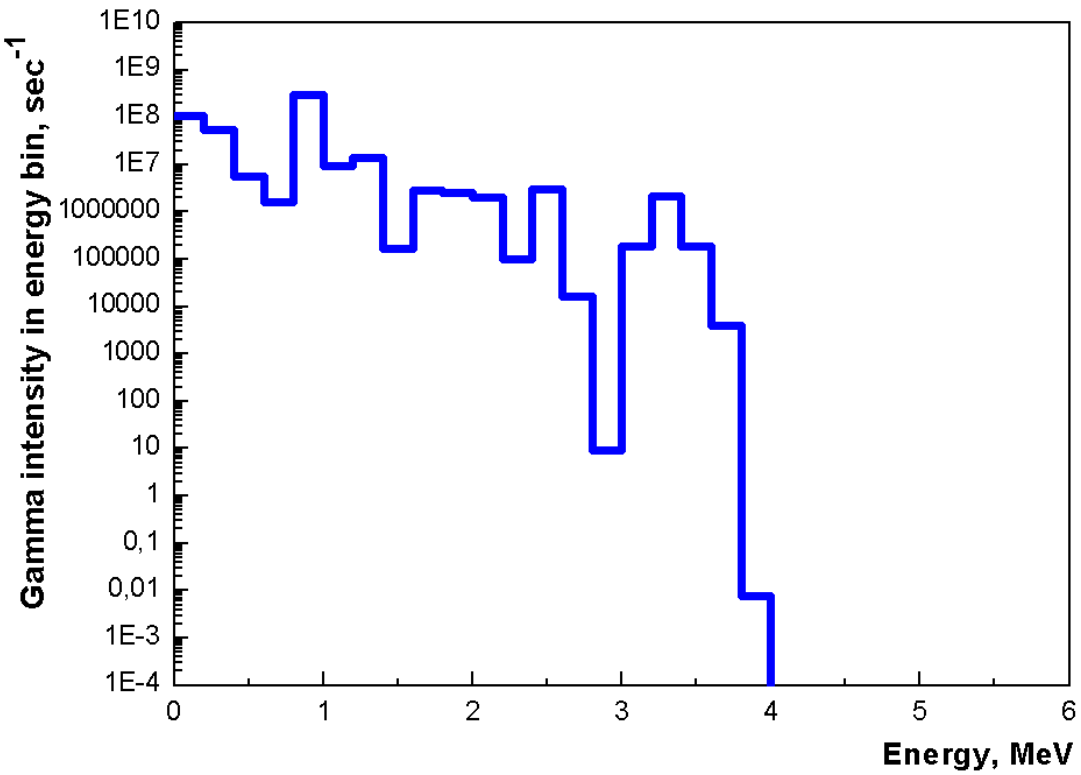
Activity of the lead blocks surrounding the central part of target



Gamma intensity in the energy bin (gamma quanta with energy in the limits of one bin of the histogram) in the target



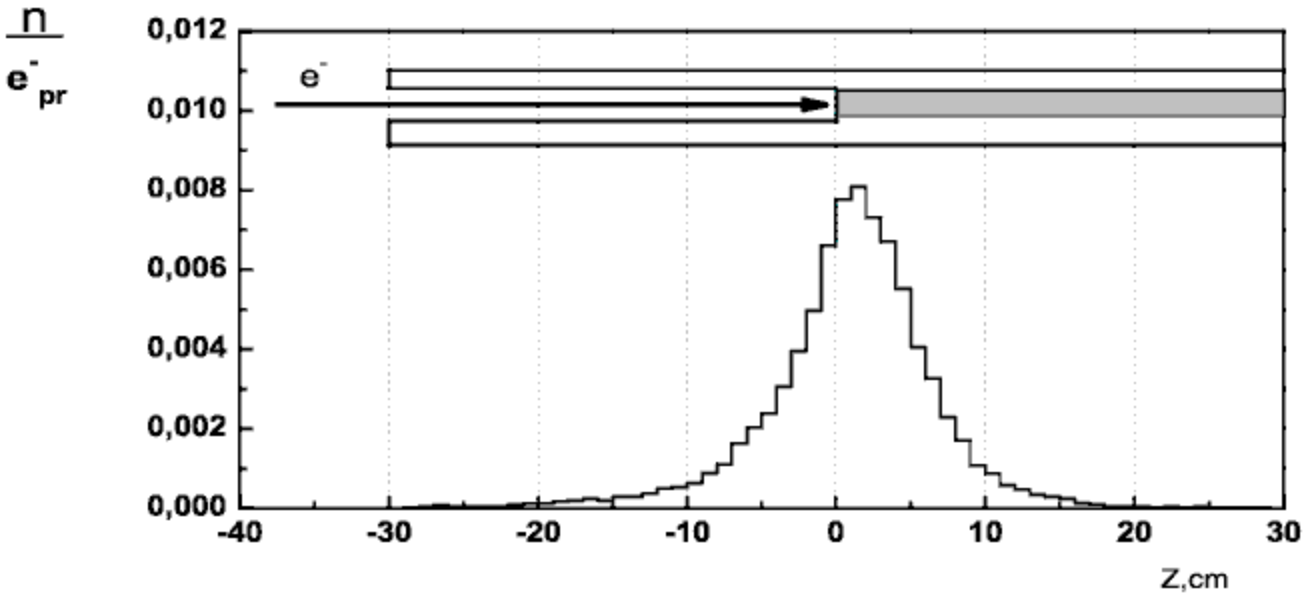
Gamma intensity in the energy bin (gamma quanta with energy in the limits of one bin of the histogram) in one of the lead blocks.



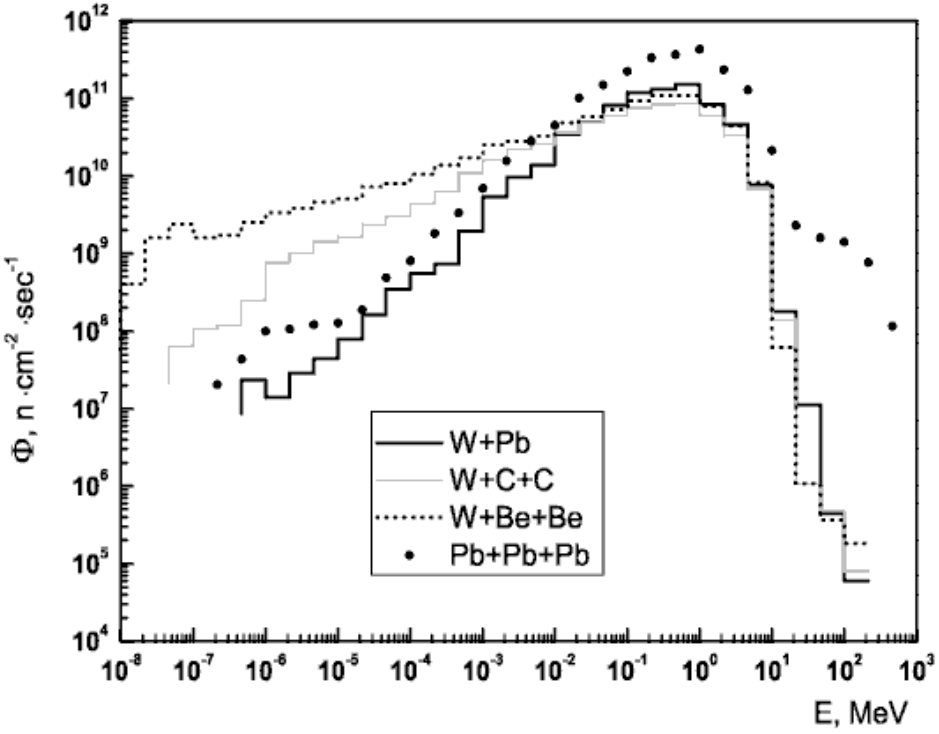
Doses from activated target

- Calculation of gamma-spectra at 1 meter distance from the target has been carried out using the program MCNPX.
- Intensity of equivalent dose is calculated by means of conversion coefficients from the fluence to the dose in case of external radiation of the whole body with a parallel beam in front and back geometry
- Thus, intensity of the dose at one meter distance from the target extracted from the SAD installation after half a year since the shut down was equal to 94 μSv/hr; for one hexagonal lead block is equal to 28.3 μSv/hr.
- Reloading of the target will be accomplished by means of a cylindrical container with wall thickness 140 mm (layer of lead between steel sidewalls). In this case intensity of the dose from the target will be equal to ≤ 0.2 μSv/hr, therefore the design level of dose intensity for the maintenance personnel established at the level 6 μSv/hr is ensured.

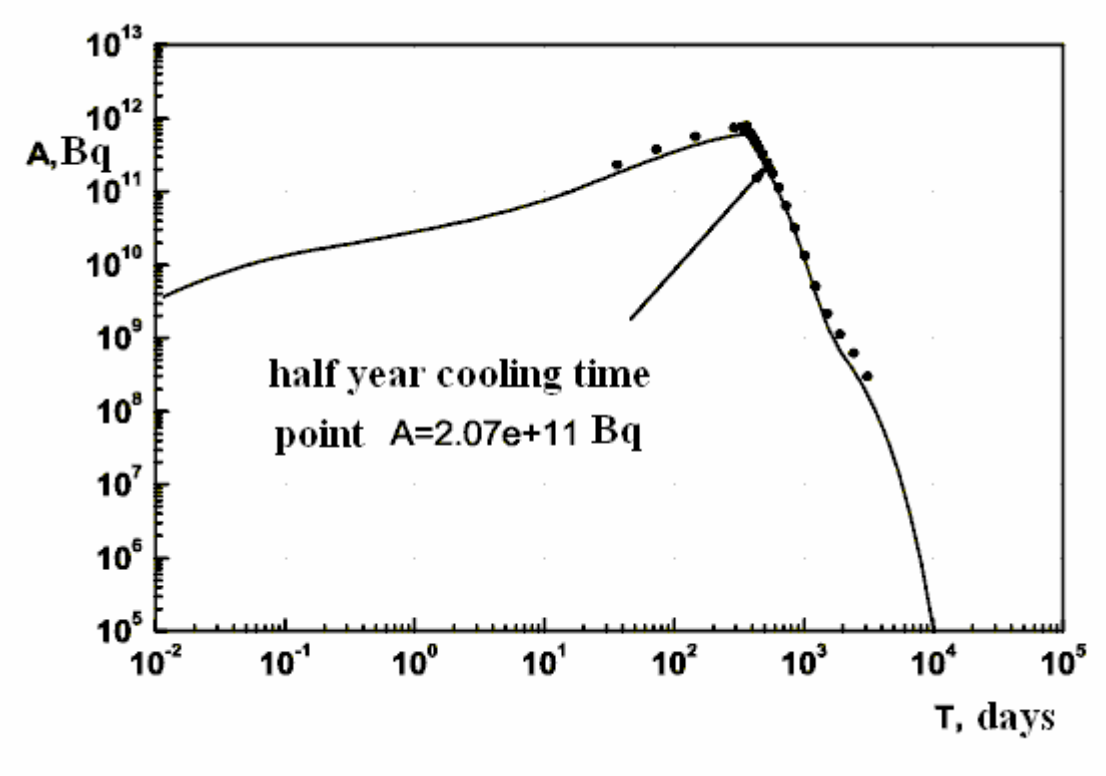
Neutron emission through the side surface of the target with the tungsten insert depending on the coordinate along the target axis.



Neutron Spectra in Different Modifications



The Activity Evolution of Tungsten Target Irradiated by 200 MeV Electron Beam.



ADS Conclusions

Neutron flux·density about 10^{17} n·cm-^{2}·sec^{-1}

- Fast neutrons for transmutation
- K-eff=0.98
- Thermal power of system 1.2 GW. Beam power 10 MW. Energy of protons 1.2 GeV
- Lead target, lead reflector and helium cooling system of sub-critical reactor
- Electrical power 0.4 * 1200 MW – 40 MW=440MW

Fuel Cycle Options

Nuclear energy requires nuclear REACTORS + nuclear FUELS

The industrial stages necessary to transform the natural nuclear fuel (uranium, thorium) into a fuel assembly suitable to reactors (front end) and, later on, stages to process the spent fuel (backend) are called the NUCLEAR FUEL CYCLE

Typical feature of the nuclear fuel:

unlike most other energy sources, a significant part of the nuclear « spent » fuel can be recycled

Subsequently, the nuclear fuel cycle can either be:

OPEN: when spent fuel is declared a waste and is globally disposed of,

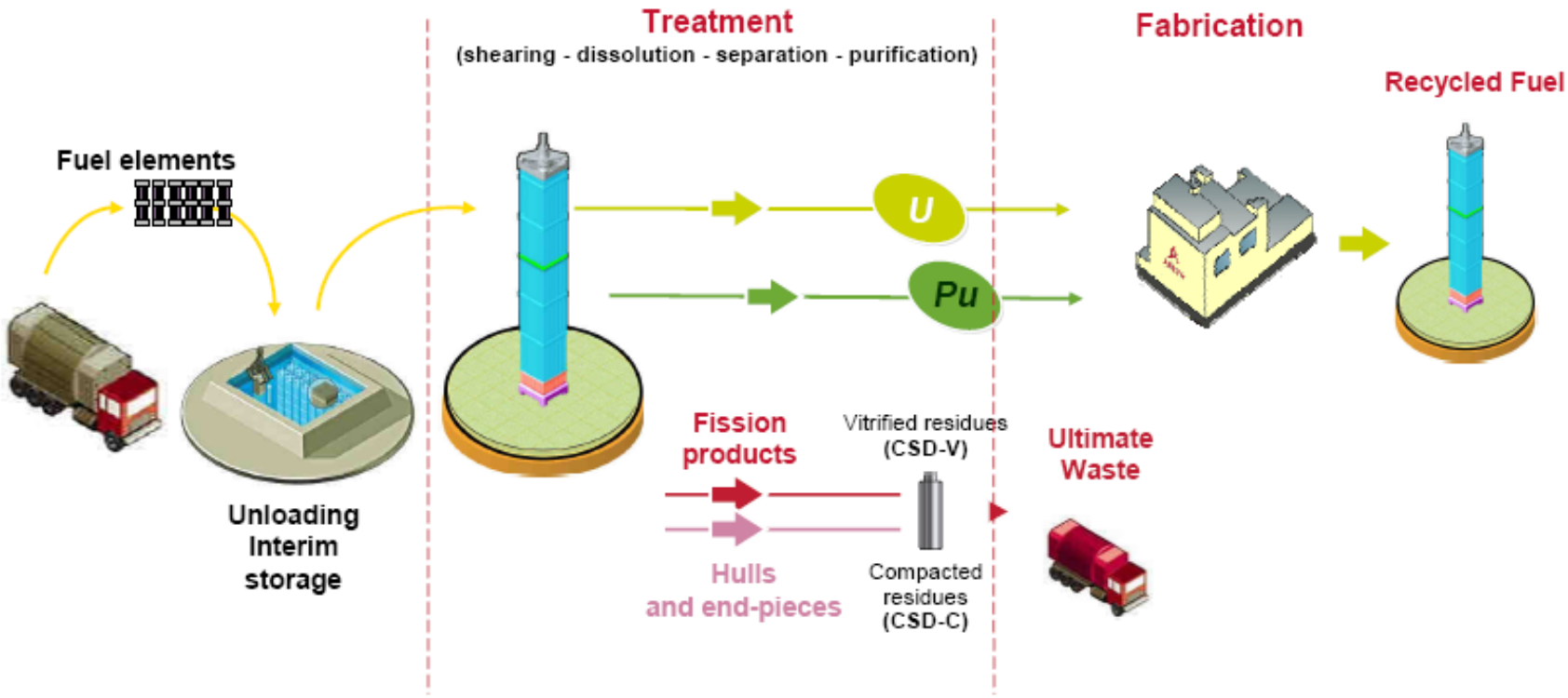
CLOSED: when spent fuel is reprocessed and its fissile/fertile part recycled for further reactor fueling

Why recycling ?

Recycles	**96%**	of the content of spent nuclear fuel
Saves	**25%**	of our natural resources
Represents less than	**6%**	of the cost of the kWh (about the same as direct disposal)
Divides waste volumes by	**5**	
Divides waste toxicity by	**10**	

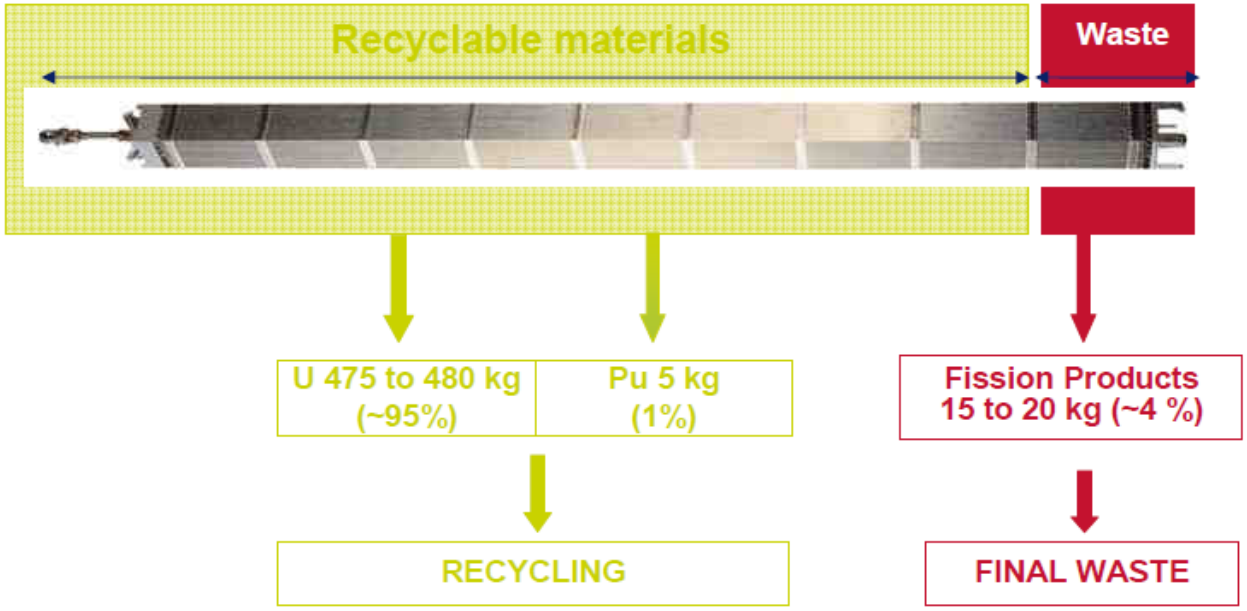
Recycling technologies produce few waste packages that remain stable for tens of thousands of years

The main stages in recycling



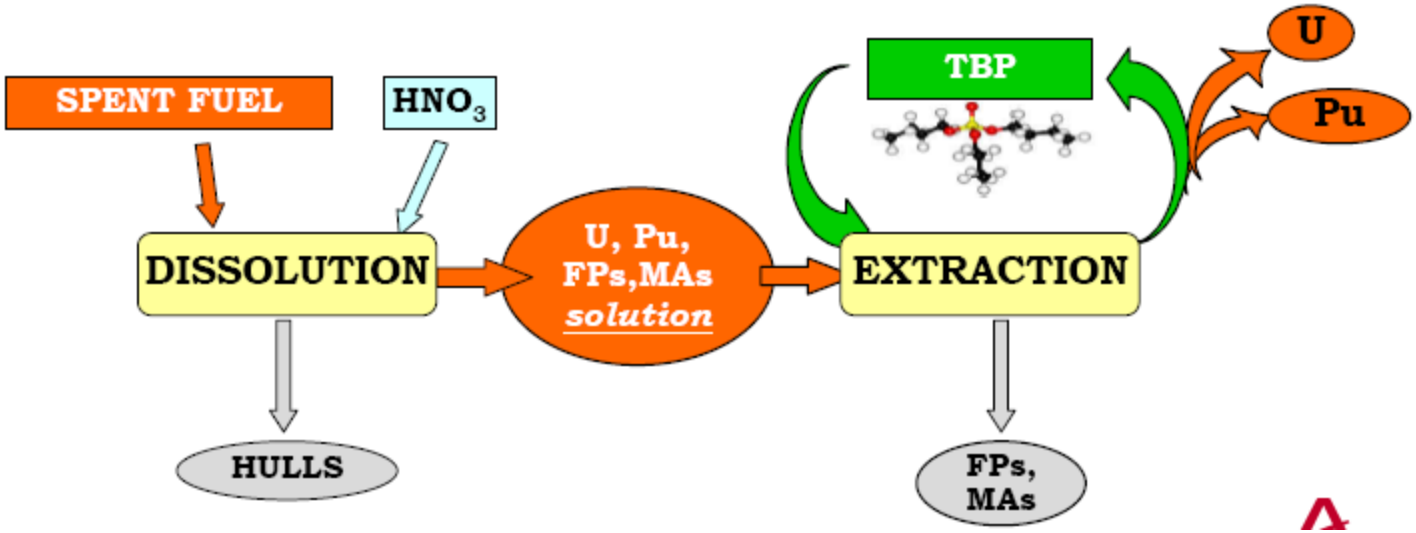
Recycling – the heart of the Back End: 96% of Used Fuel is Recyclable

- Typical composition of a Light Water Reactor Fuel
  - Before irradiation: ~ 500 kg of Uranium (PWR)
  - After irradiation

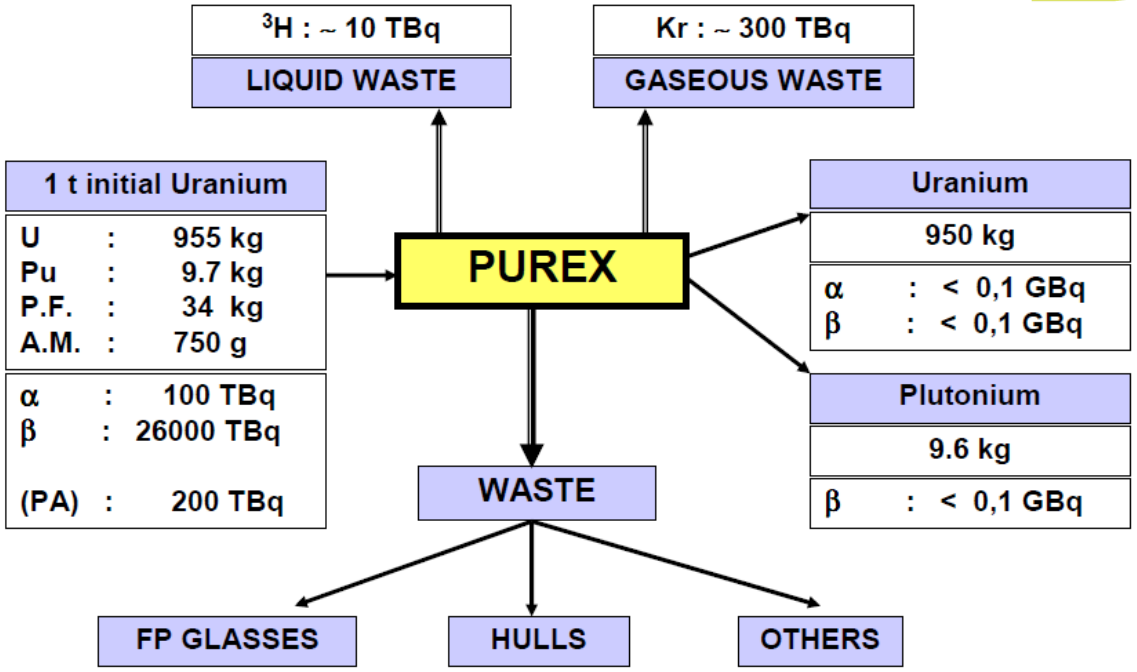


Spent fuel recycling

The principle of spent fuel recycling generally adopted throughout the world is based on the separation of the different components by liquid/liquid extraction in tributyl phosphate (TBP) diluted in an alkane after the fuel has been dissolved in nitric acid.

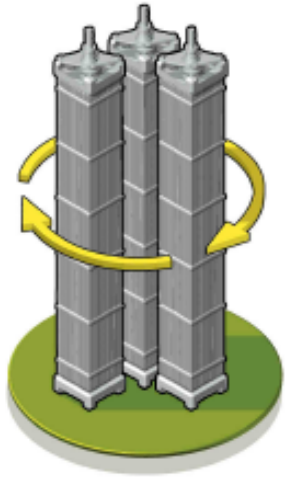


PUREX: global balance



Recycled fuels

- MOX & Reprocessed Uranium fuels can accommodate all Light Water Reactors (80% of worldwide nuclear capacity)
- For utilities, recycled fuels have outstanding performance
  - Excellent reliability track record
  - Equal energy performance compared to natural uranium
  - MOX in-core behavior similar to Uranium fuel under normal and accidental conditions
- EPR designed to accommodate up to 100% MOX



Fission products

The non reusable materials are conditioned into a stable and compact form suitable for transport, storage and final disposal

- Encapsulation of Fission Products in a stable, homogeneous, and durable glass matrix with a long-term predictable behaviour

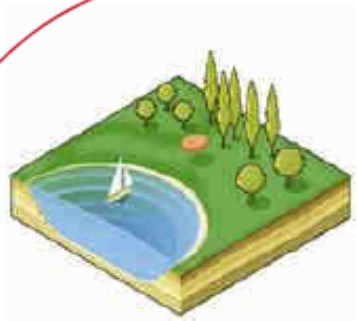
- Compaction of structural pieces (hulls and end-pieces)

- Both the glass matrix and compacted waste are encased in a standard “Universal Canister”

Recycling major benefits

- **Natural resources savings**
  - **Used fuel contains 96% of reusable materials**
  - **Up to 25% natural uranium savings**

- Improved ultimate waste management
  - Volume of ultimate waste divided by 5
  - Waste toxicity substantially reduced
  - Standard, durable, specifically designed waste forms and containers

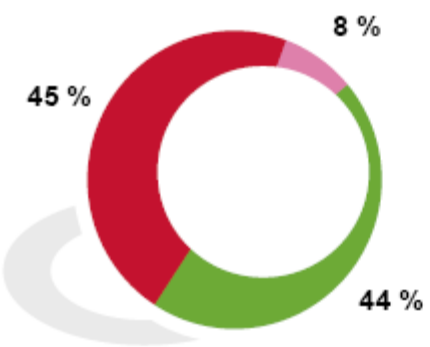


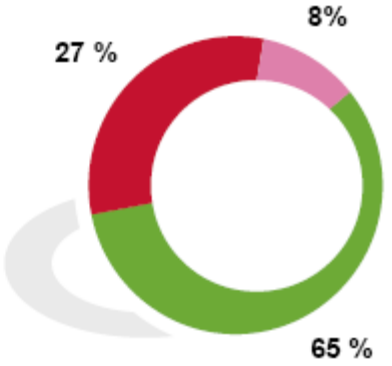
- Reinforced economic interest of recycling
  - Demonstrated competitiveness vs. once-through strategy
  - Ability to control overall back-end costs based on proven 40-year industrial track record
- While ensuring Health, Safety and Environmental protection

Recycling Increases Nuclear Acceptance

Level of acceptance of nuclear energy by European Union citizens

Are you in favour of energy production by nuclear power stations?



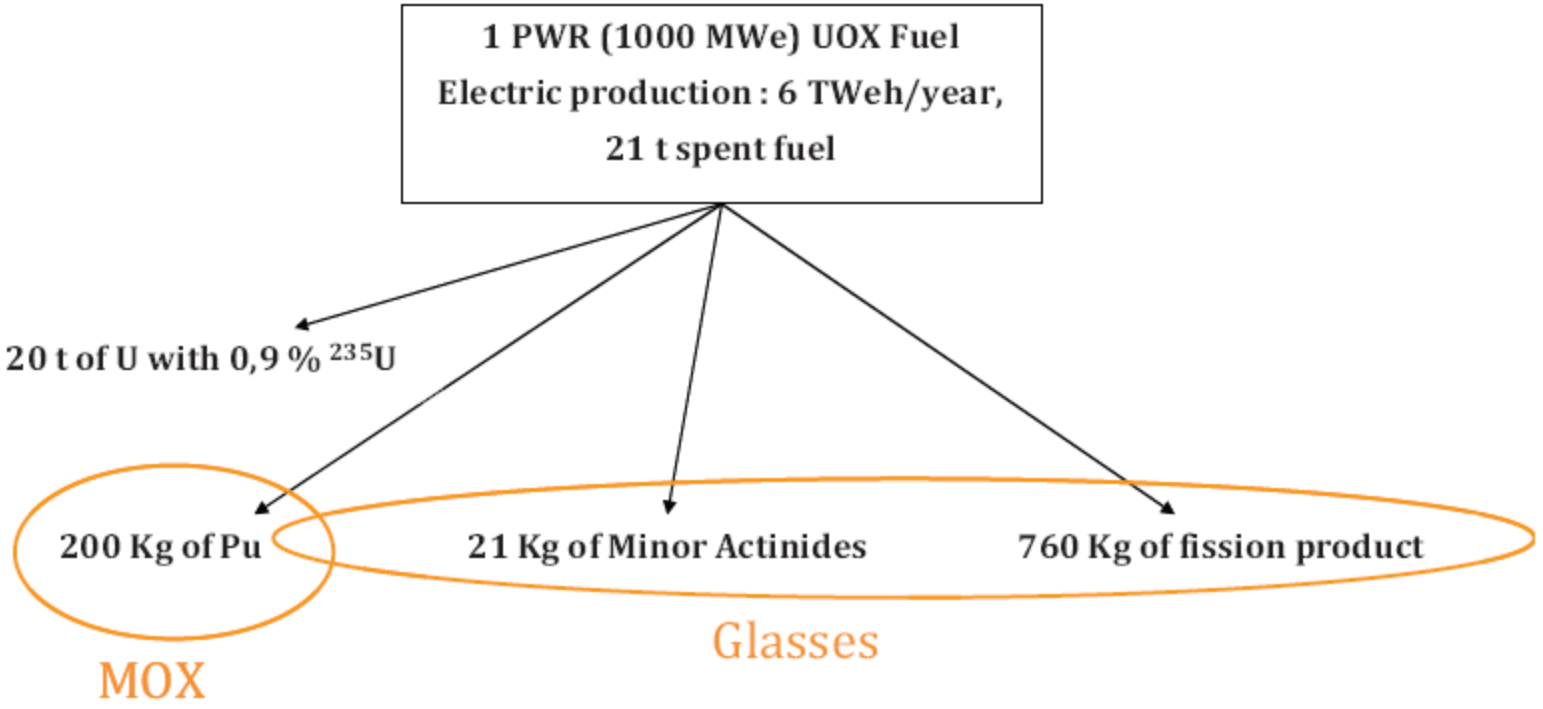


«with a solution for the management of radioactive waste

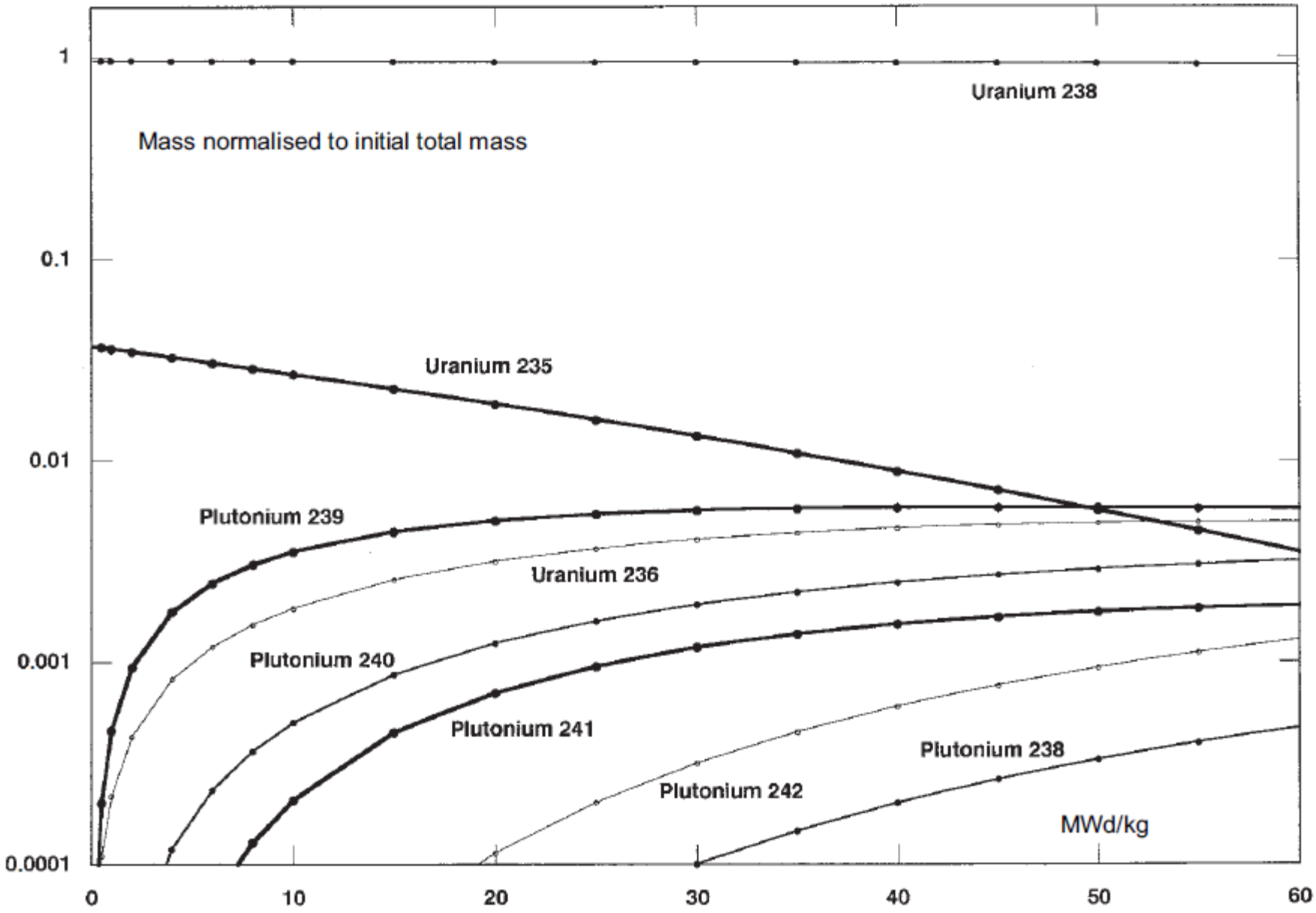
- *Favorable*
- *Unfavorable*
- *No opinion*

Source : EUROBAROMETER 2008

Waste Production

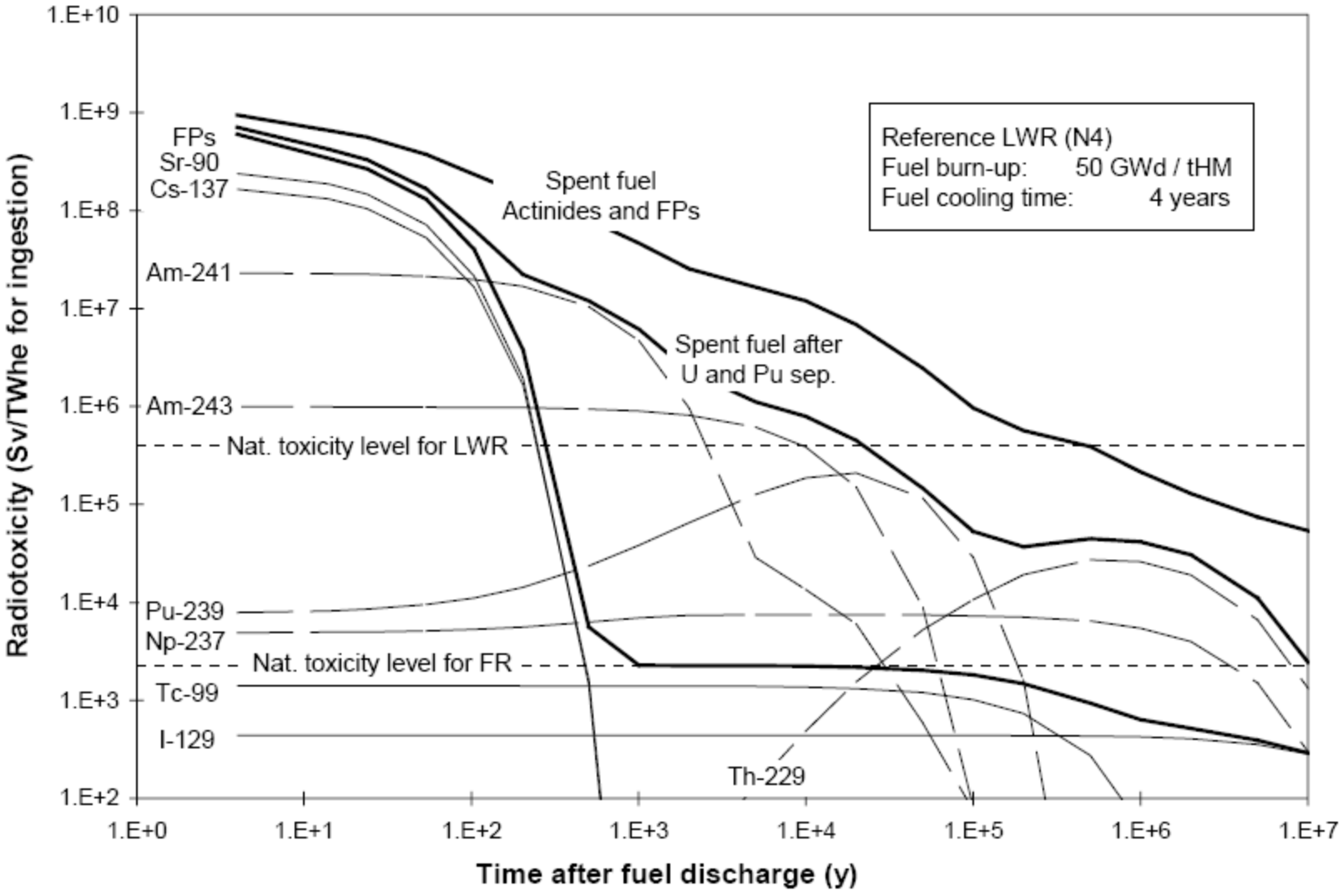


French fleet (58 reactors) : 400 TWeh
Production per year ~12 t Pu ; 1,2 t M.A. ; 3,5 t FP

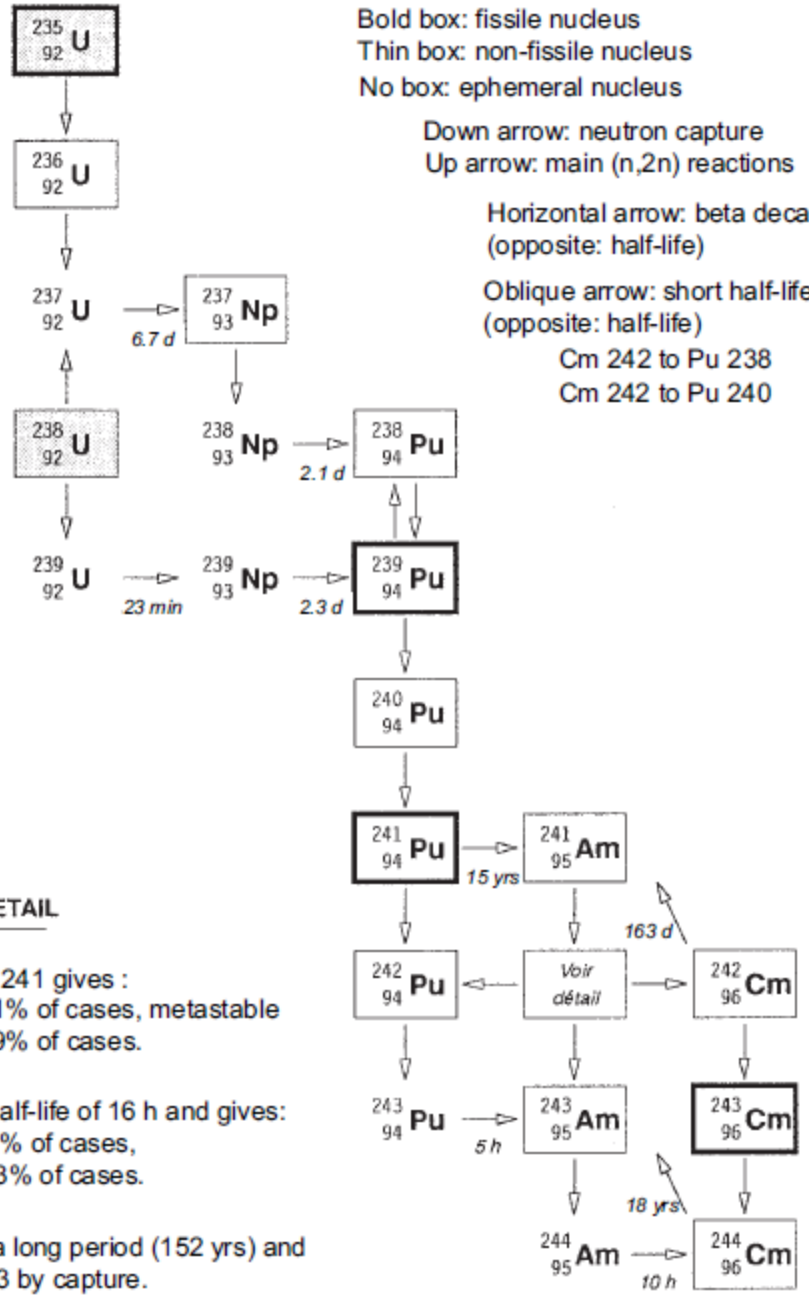


Evolution of mass the main uranium and plutonium isotopes for a pressurised water reactor with respect to burn-up (the ordinate axis is on a log scale).

Radiotoxity of LWR Spent Fuel



Actinides production



DETAIL

By capture, Am 241 gives :
Am 242 in 11% of cases, metastable
Am 242 in 89% of cases.

Am 242 has a half-life of 16 h and gives:
Pu 242 in 17% of cases,
Cm 242 in 83% of cases.

Am 242 m has a long period (152 yrs) and gives Am 243 by capture.

Minor Actinides

- **Neptunium :**
  - Np^{237} activity and neutron source low ($T_{1/2}$=2.14 10^6 ans), this isotope doesn't need specific protection at targets reprocessing and fabrication
  - In nuclear reactor, It appears by neutronic capture on d'U^{236} and β^- decay or by (n,2n) reaction on l'U^{238}.
- **Americium :**
  - Am^{241} mostly from β^- decay of Pu^{241} ($T_{1/2}$=14,4 ans),
  - Power and neutronic emission important (α,n), in reactor, production of Pu^{238} and Cm^{242}.
  - Am^{243} is produce by neutronic capture on Pu^{242} . Epithermal poison, γ hard (issued by Np^{239}), increase of its dose flow
- **Curium :**
  - ^{242}Cm and ^{244}Cm have short periods and important activities, power and neutronic emission are very important

Principle of Transmutation

«Disappearance » or transmutation rate T : disappearance rate of isotope i by nuclear reaction r after Δt under flux level ϕ

$$T_i = 1 - e^{-\bar{\phi}\Delta t \bar{\sigma}_r^i}$$

If r = absorption we call it ***transmutation*** rate

>

If r = fission we call it ***fission*** rate

Consumption C : masse balance relative to associated energy production

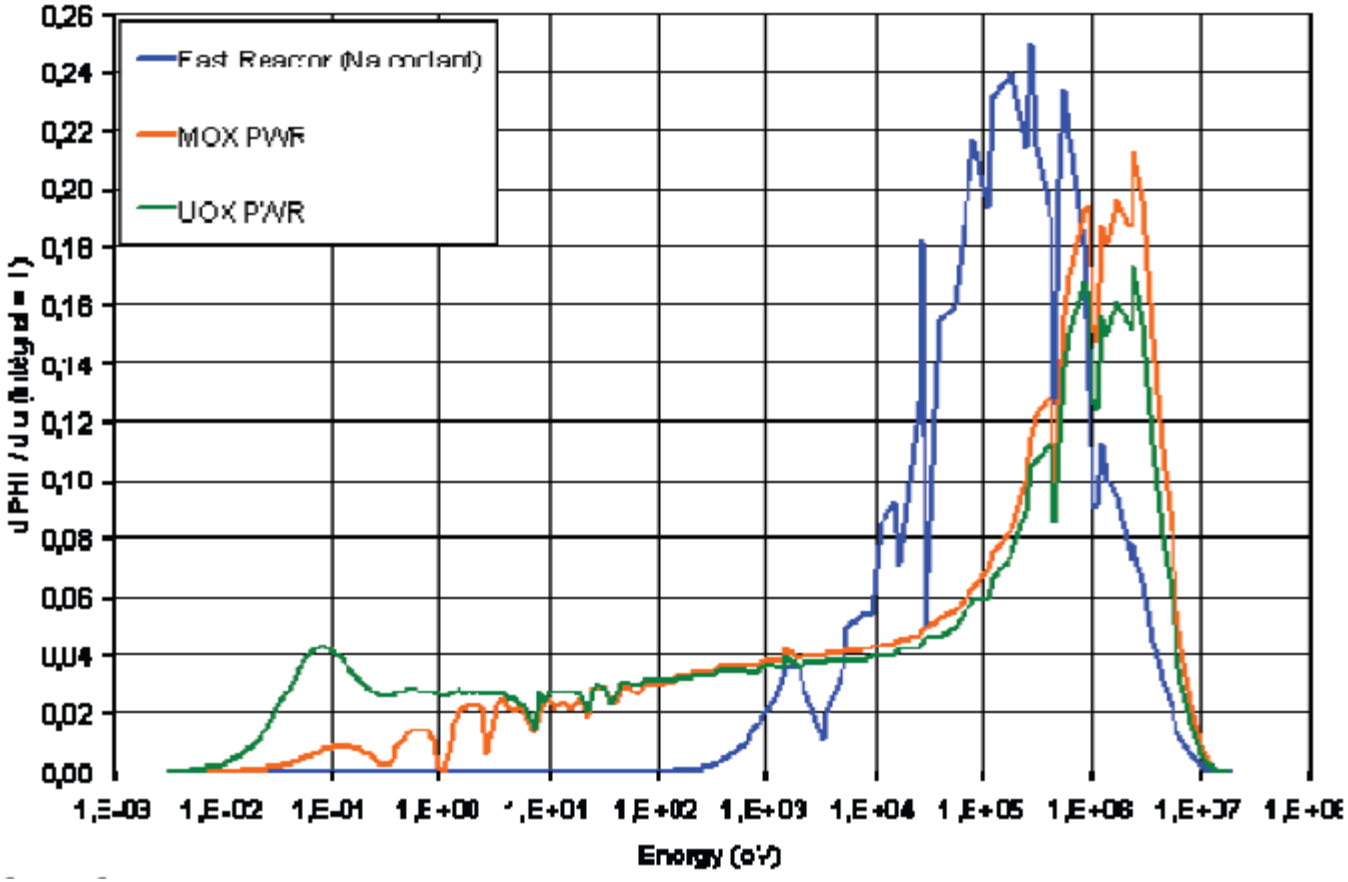
$$C_i(\Delta t) = \frac{m_i(t=0)T_i(\Delta t)}{P \times \Delta t} \qquad \text{(kg/TWeh)}$$

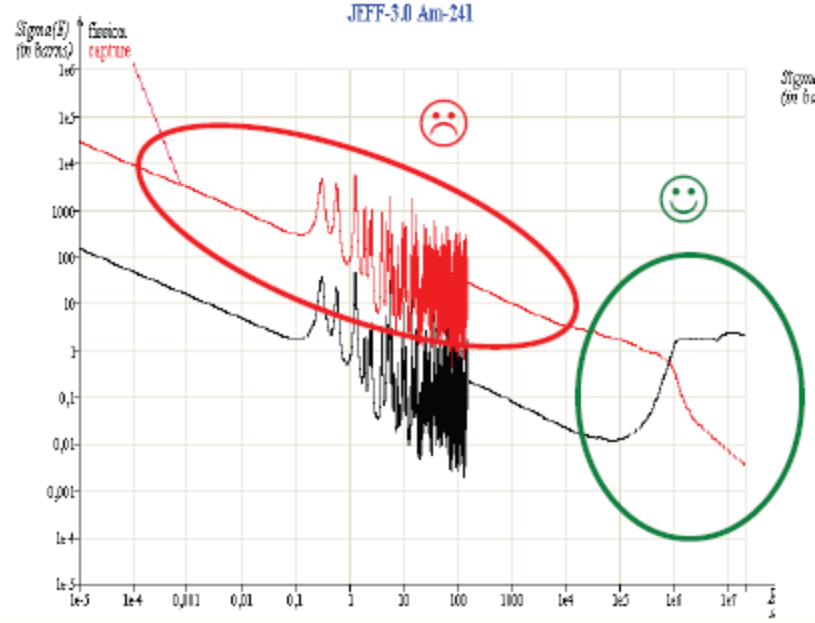
Fission and capture cross sections

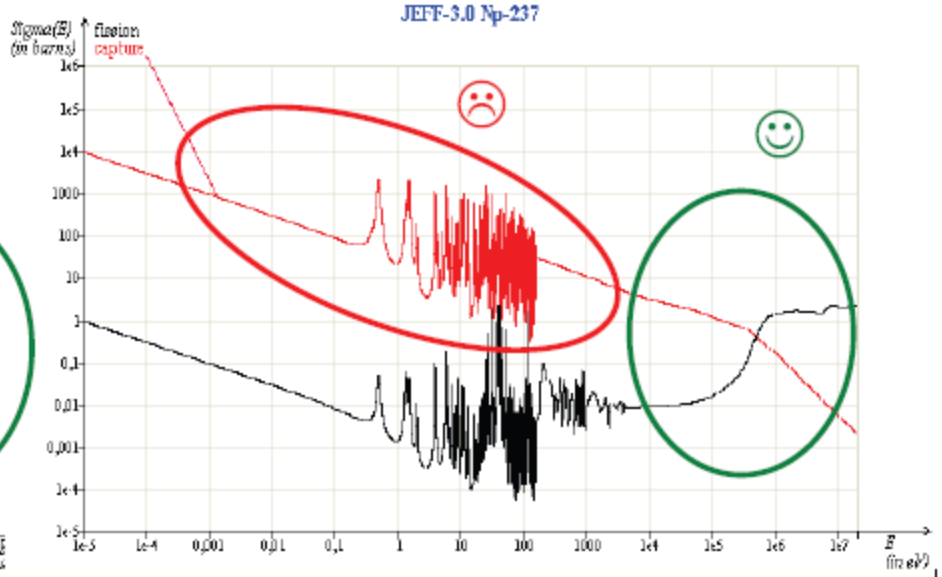
- α ratio = capture/fission non favorable in thermal spectrum

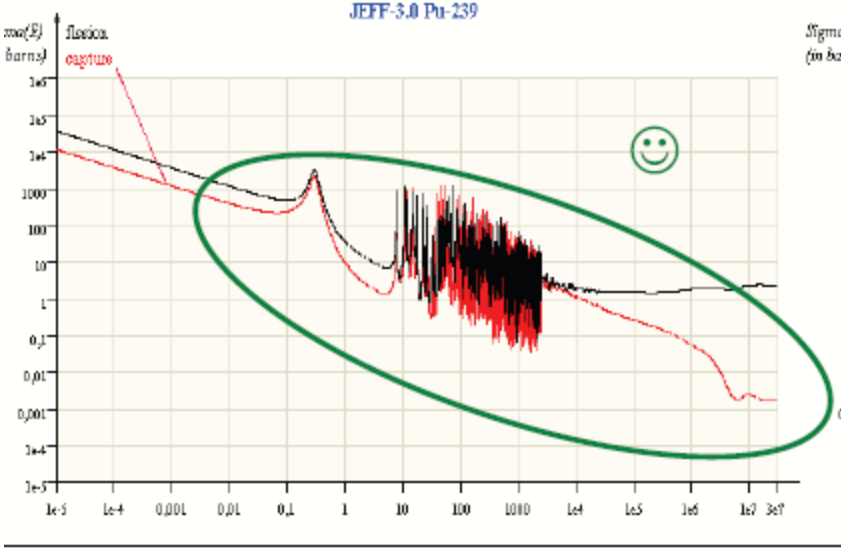
Isotope	ϕ : Low neutrons reactor (PWR)			ϕ : Fast neutrons reactor (FR)		
	σ_f	σ_c	α	σ_f	σ_c	α
^{235}U	38,8	8,7	0,22	1,98	0,57	0,29
^{238}U	0,103	0,86	8,3	0,04	0,30	7,5
^{239}Pu	102	58,7	0,58	1,86	0,56	0,3
^{240}Pu	0,53	210,2	396,6	0,36	0,57	1,6
^{241}Pu	102,2	40,9	0,40	2,49	0,47	0,19
^{242}Pu	0,44	28,8	65,5	0,24	0,44	1,8
^{237}Np	0,52	33	63	0,32	1,7	5,3
^{241}Am	1,1	110	100	0,27	2,0	7,4
^{243}Am	0,44	49	111	0,21	1,8	8,6
^{244}Cm	1,0	16	16	0,42	0,6	1,4
^{245}Cm	116	17	0,15	5,1	0,9	0,18

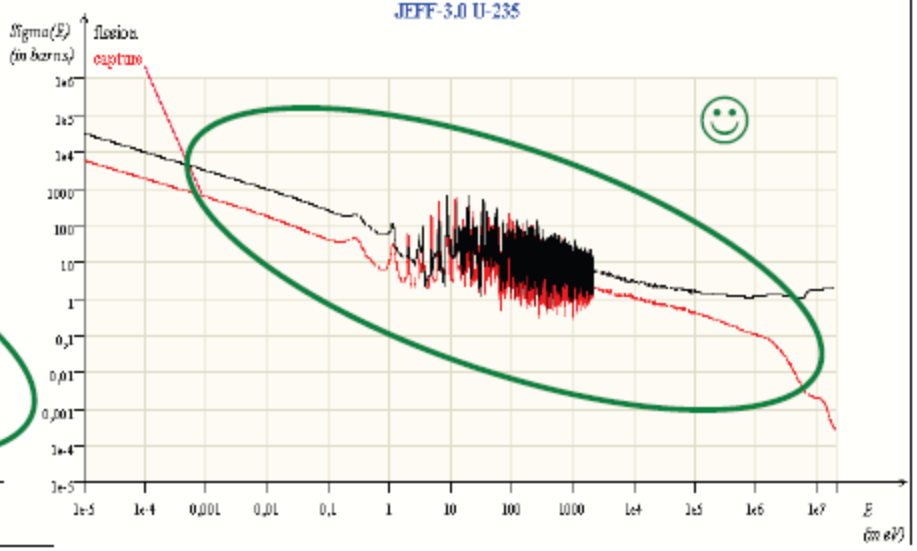
Neutron Spectrum




JEFF-3.0 Am-241
Sigma(E) (in barns)
fission
capture
E (in eV)


JEFF-3.0 Np-237
Sigma(E) (in barns)
fission
capture
E (in eV)

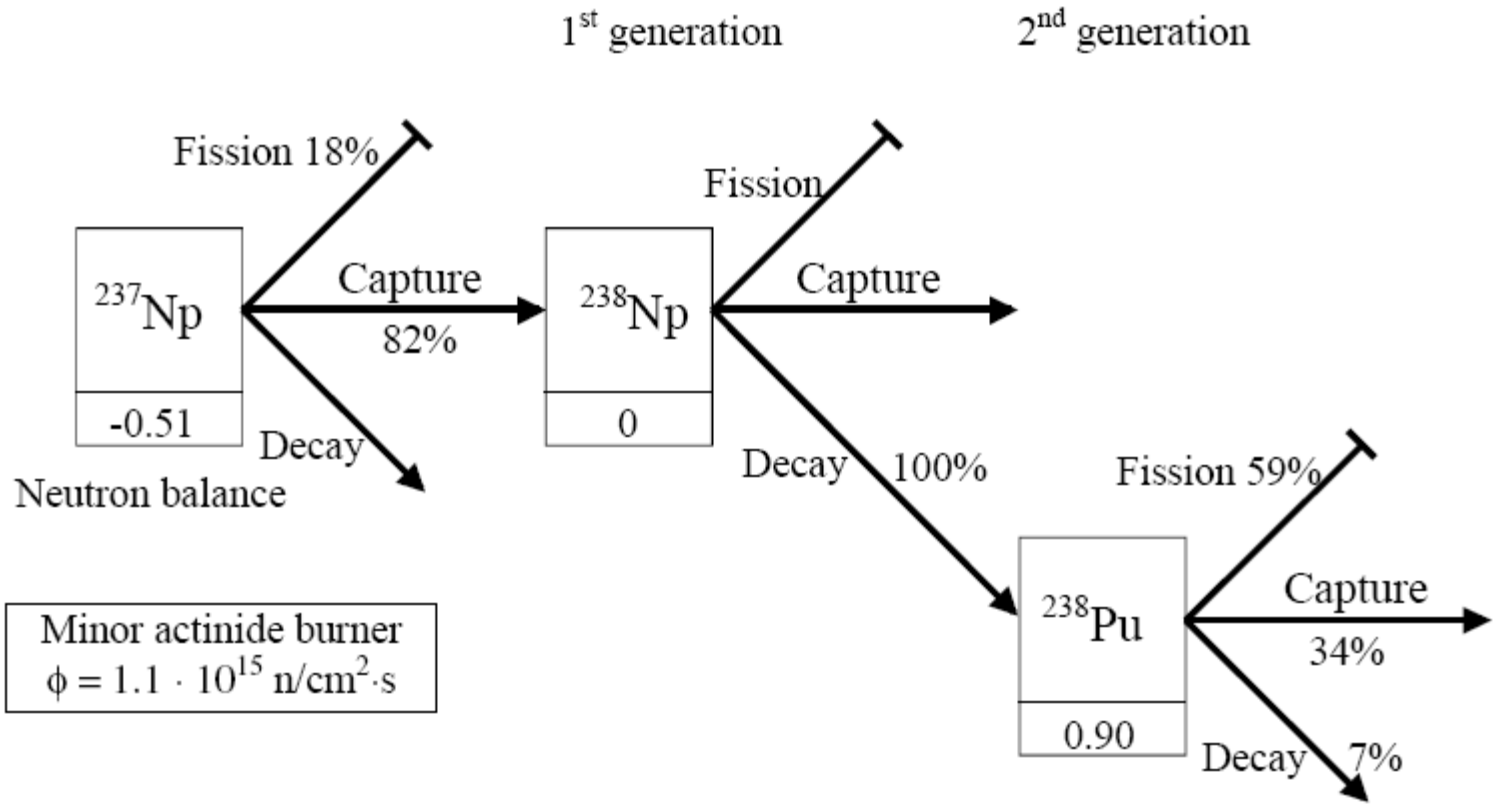

JEFF-3.0 Pu-239
fission
capture


JEFF-3.0 U-235
Sigma(E) (in barns)
fission
capture
E (in eV)

Consumption

Concept	PWR MOX Φ = 2.5 10^{14} n/cm^2.s Ts : 1500 EFPD		FR Φ = 34 10^{14} n/cm^2.s Ts : 1700 EFPD	
Isotopes	transmutation	fission part	transmutation	fission part
^{237}Np	46%	4%	63%	24%
^{241}Am	70%	10%	69%	24%
^{243}Am	65%	6%	63%	15%
^{244}Cm	44%	16%	50%	27%

Transmutation of Neptunium

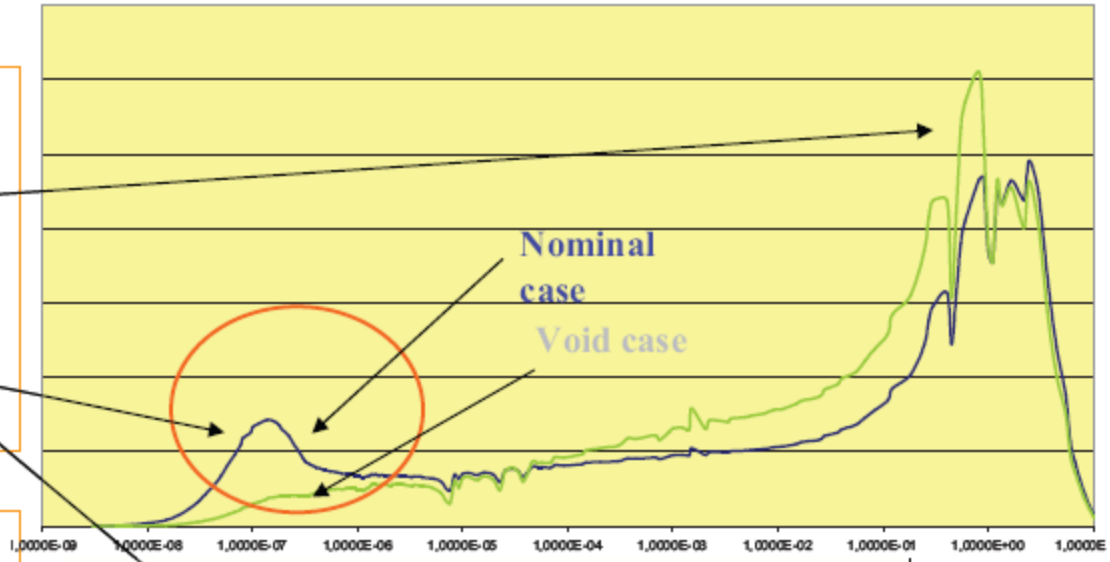


Transmutation of MA in PWRs.Coolant Void Impact

Physical effect :

During the coolant voiding, the thermal contribution disappear (absence of moderator)

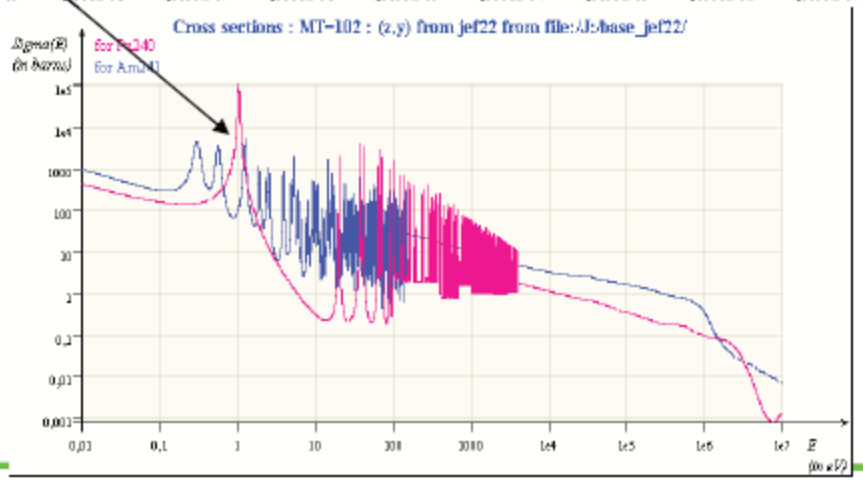
the contributions of captures in thermal and epithermal resonances disappear



Reactiviy Increasing : Pu^{239} (content > 10%), Pu^{240} et Pu^{242}, Np et Am

Reactivity decreasing : Pu^{241}

Pu^{238} sligthy effect (few resonances)



Pu limitation ~ 12%

Pu and MA ~9% to 7%

Plutonium management

Burn-up (GWd/t)	Initial Enricht.	Pu quality		Am 241	Am 243	Cm 244	Cm 245
42	3.70 %	66%		0.58	0.49	0.14	0.01
55	4.50 %	63%		0.47	0.56	0.22	0.02
65	4.95 %	60%		0.48	0.71	0.34	0.03

Transmutation Ways

Two ways for transmutation have been investigated :

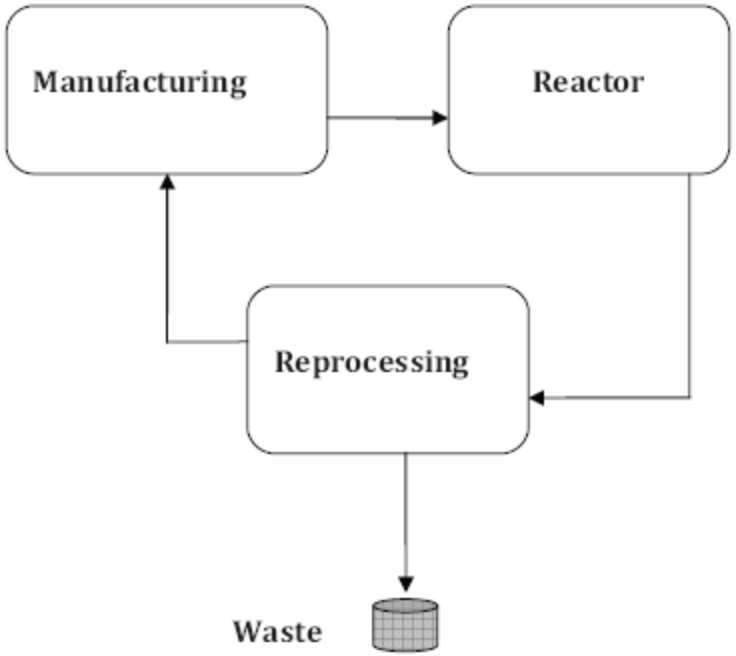
- The homogeneous mode where the minor actinides to be transmuted are directly mixed with "standard" fuel of the reactor,
- The heterogeneous way for which the actinides to be transmuted are separated from the fuel itself, in limited number of S/A (targets) devoted to actinides transmutation.

Actinides Management

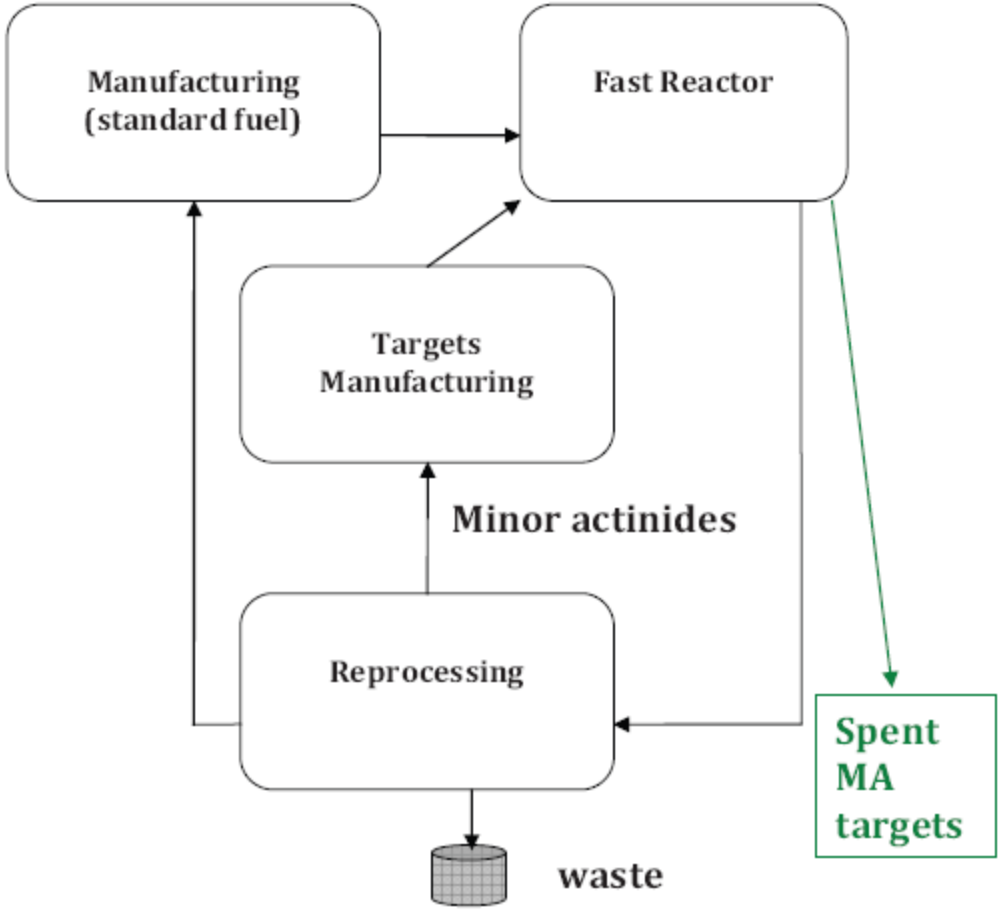
Two ways for actinides management have been also studied :

- The multi-recycling : in this case whole or part of minor actinides and plutonium at the end of each reactor cycle is sent back in the following cycle. In that way, only reprocessing losses go to the waste,
- The once-through way : in this case the minor actinides are transmuted in targets where very high burn up is reached

homogeneous way (multirecycling)



heterogenous way (multirecycling)/ once-through



Transmutation of MAs in FRs:homogeneous Way

- The *homogeneous* mode in fast neutron reactors appears to be the *easiest solution* for transmutation (for neutronic behavior) given the following advantages:
  - homogeneous distribution in the core *does not disrupt power distributions*
  - the high fluence level and the excess neutrons accommodate minor actinides easily
  - the overall balance of the reactor, which initially works with plutonium, is slightly disrupted
  - The acceptable limit is approximately a level of *2.5%* by mass of minor actinides for EFR type reactors (*which represents in this case a ton of minor actinides in the core*)
  - the reached fission rates shows that it necessary to *multirecycling MA*

Transmutation of MAs in FRs:heterogeneous way

- In a fast neutron reactor, a substantial neutron flux escapes from the core and can be used to:

  - *create plutonium* by placing depleted uranium in the peripheral areas known as blankets (breeding);
  - *consume minor actinides* by positioning targets in areas of the core's periphery, the advantage of which is that the safety factors are slightly changed.

Transmutation of MAs in FRs:heterogeneous way

- Multirecycling : targets are placed in the first row of radial blanket (with a standard S/A's design) and are reprocessed after irradiation.
  - The waste quantity mainly depends on partitioning performances.

- In the once-through approach the targets are special S/A with moderator (YH_2) and inert matrix (MgO).
  - They are placed in core and in the first row of the radial blanket.
  - The objective in this case is to reach a very high fission rate (90%) for MA (Am,cm) under respect of design criteria.
  - the neutron spectrum is locally changed so as to increase the transmutation rate and speed up the process.
  - In comparison to multi-recycling, the obvious advantage of this solution is the minimization of minor actinide fluxes in the nuclear cycle.

Heterogeneous way in FRs

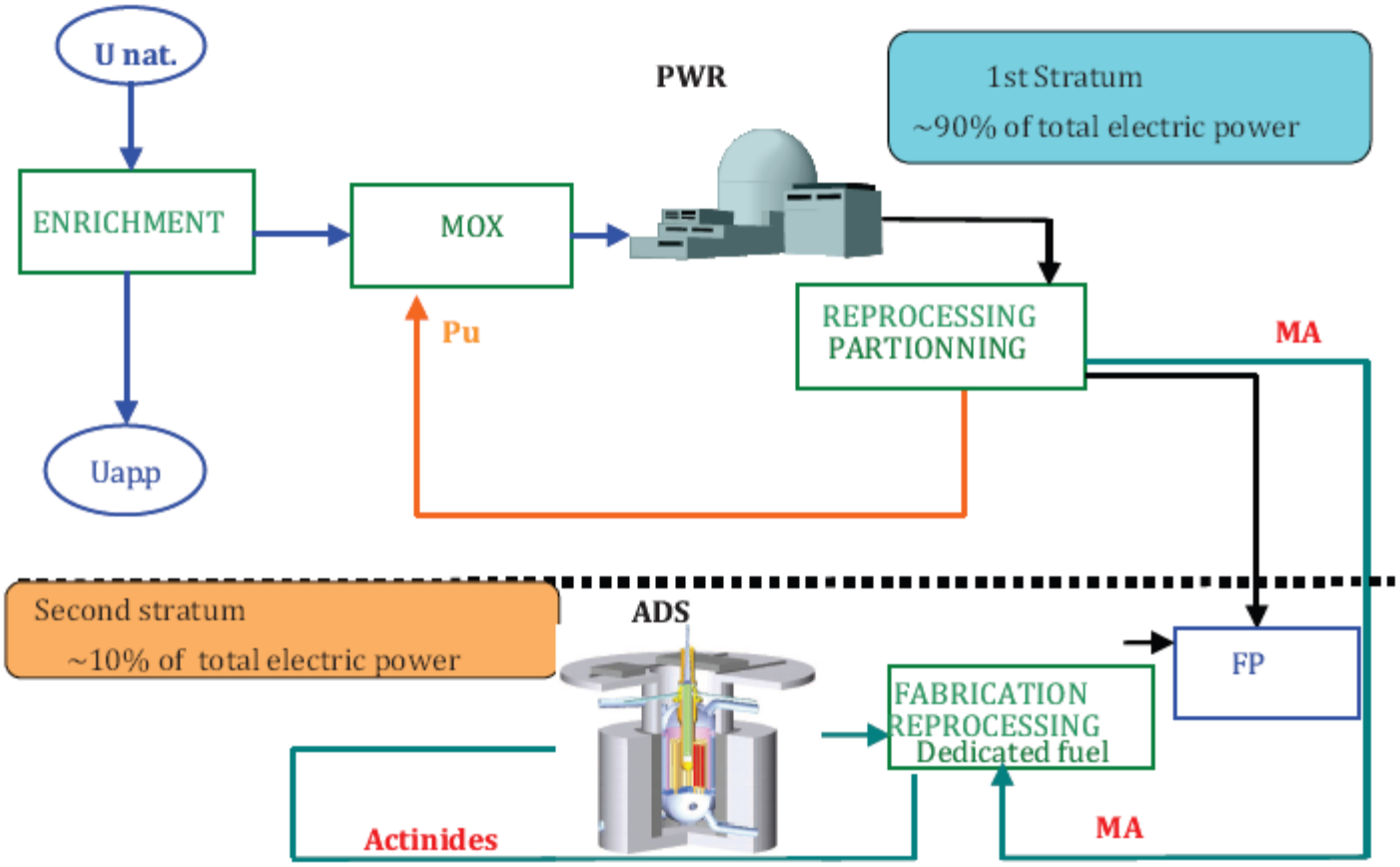
Transmutation way	Multi.	Once-through
% FR in the fleet	100	55
Number of S/A Core Radial blanket	 - 78	 48 78
Matrix	Al_2O_3	MgO
MA content (%) Am Cm	 40 -	 19 1.7
Consumption (kg/Tweh)	14	7.4
Fission rate (%)	22	90

ADS performances for transmutation

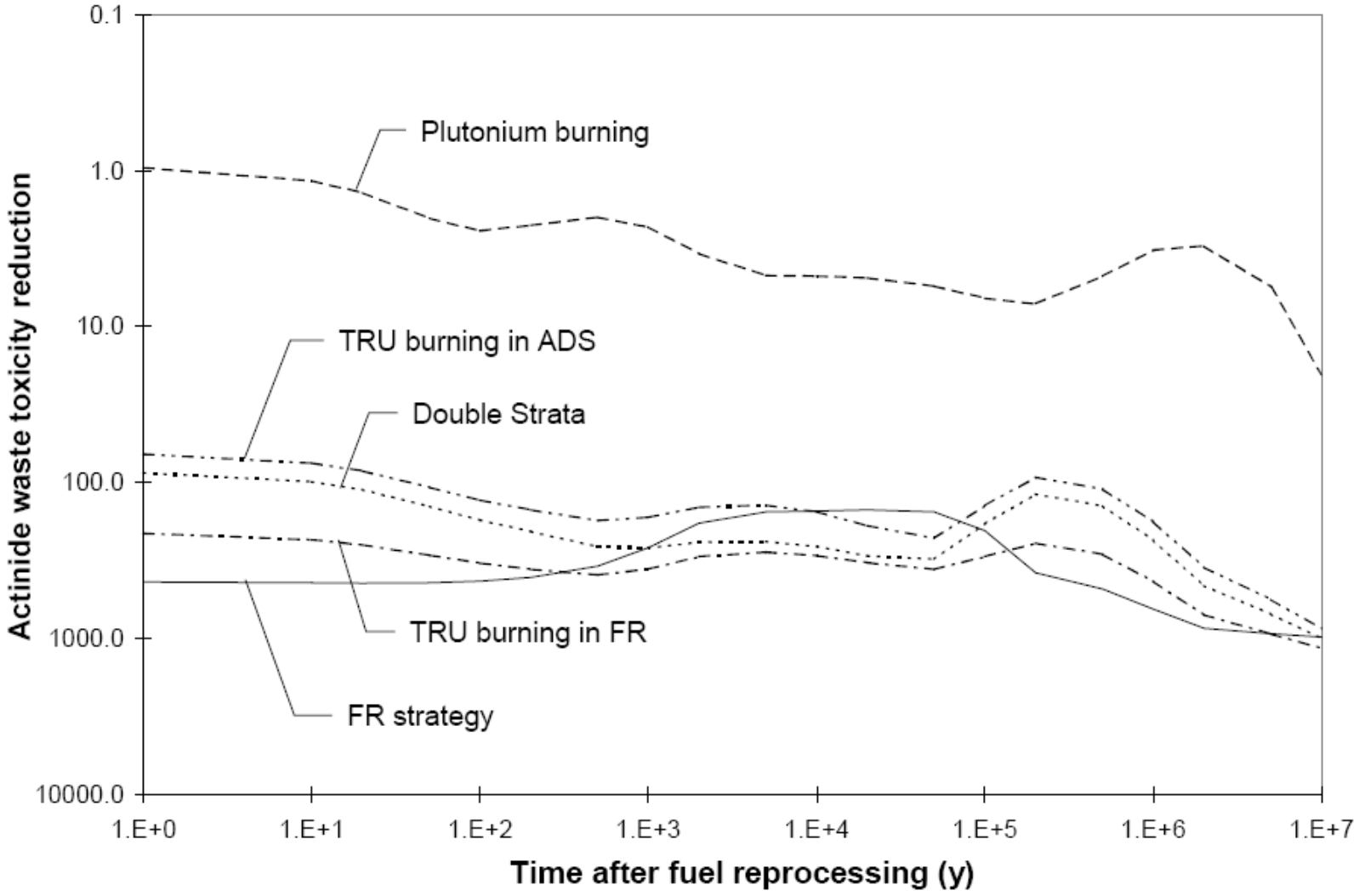
- ADT (Gas cooled) results

Electric Power	**160 MW**
Thermal Power	**400 MW**
Irradiation length	**3 x 320 EFPD**
H. N. mass	**9.3 tons**
Fuel cooling time/ aging time	**1 y / 2 ys**
Average transmutation rate	**Np : - 22 kg/TWeh** **Am : - 83 kg/TWeh** **Cm : - 4 kg/TWeh**

Double strata scenario with ADS



Radiotoxicity Reduction



Fission Product Transmutation

Fission product	$T_{1/2}^{Decay}$ (y)	$T_{1/2}^{Trans}$ (y)[1]	Isotopic separation	Transmutable (yes/no)
^{129}I	$1.6 \cdot 10^{7}$	51	no	yes[2]
^{135}Cs	$2.3 \cdot 10^{6}$	170	yes	no
^{99}Tc	$2.1 \cdot 10^{5}$	51	no	yes
^{126}Sn	$1.0 \cdot 10^{5}$	$4.4 \cdot 10^{3}$	yes	no
^{79}Se[3]	$6.5 \cdot 10^{4}$	$2.2 \cdot 10^{3}$	yes	no

1. Thermal flux: 10^{14} n/cm^2·s.
2. R&D necessary to improve the iodine separation yield and the stability of the target material.
3. Half-lives for ^{79}Se around $6.5 \cdot 10^{4}$ years have been used widely in waste inventory and repository performance assessments. Recent nuclear data studies, however, indicate a much longer half-live for this nuclide (see http://nucleardata.nuclear.lu.se/nucleardata/).

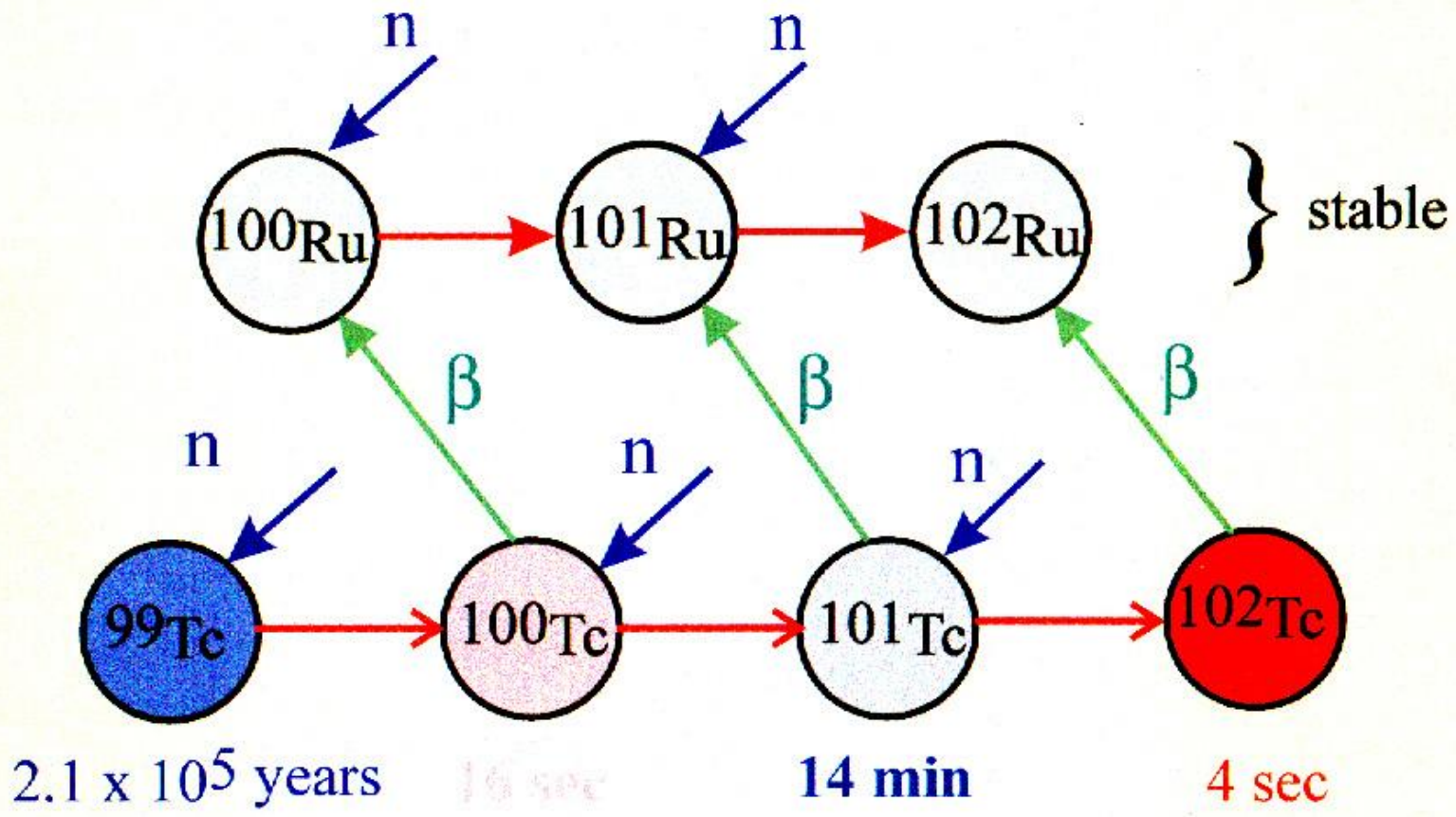
$$T_J^{transm} = \frac{\ln 2}{\sigma_{n,\gamma}^{J} \phi \times 3.15 \times 10^{7}} \text{ years}$$

Physical parameters of the major LLFP

Isotope	Period (y)	Decay mode	Thermal power (W/Bq)	Dose (ingestion) (Sv/Bq)	Fraction in an irradiated fuel (g/t)[(a)]
^{14}C	5.7×10^{3}	β	1.6×10^{-14}	5.7×10^{-10}	1.3×10^{-1}
^{36}Cl	3.0×10^{5}	β^-, β^+	4.4×10^{-14}	8.2×10^{-10}	1.6×10^{0}
^{79}Se	6.5×10^{4}	β	6.5×10^{-15}	2.3×10^{-9}	4.7×10^{0}
^{90}Sr	2.9×10^{1}	β	2.8×10^{-14}	3.9×10^{-8}	5.0×10^{2}
^{90}Y	7.3×10^{-3}	β	1.5×10^{-13}		1.3×10^{1}
^{93}Zr	1.5×10^{6}	β	2.6×10^{-15}	4.2×10^{-10}	9.8×10^{1}
^{99}Tc	2.1×10^{5}	β	1.4×10^{-14}	3.4×10^{-10}	8.2×10^{2}
^{107}Pd	6.5×10^{6}	β	1.4×10^{-15}	3.7×10^{-11}	2×10^{2}
^{126}Sn	1×10^{5}	β	4.2×10^{-14}	5.1×10^{-9}	2.0×10^{1}
^{126}Sb	3.4×10^{-2}	β	5.0×10^{-13}		6.9×10^{-6}
^{129}I	1.6×10^{7}	β	1.3×10^{-14}	7.4×10^{-8}	1.7×10^{2}
^{135}Cs	2.3×10^{6}	β	9×10^{-15}	1.9×10^{-9}	1.3×10^{3}
^{137}Cs	3.0×10^{1}	β	3.2×10^{-14}	1.4×10^{-8}	1.1×10^{3}
^{137m}Ba	4.9×10^{-6}	β	1.1×10^{-13}		1.7×10^{-4}
^{151}Sm	9.0×10^{1}	β	3.2×10^{-15}	9.1×10^{-11}	1.6×10^{1}

(a) UOX from a PWR (3.5%, 33 GWd/t).

Transmutation of technetium



Parameters of LLFP to be eventually transmuted in a fast (E_n (neutron energy) = 0.2 MeV) and thermal (E_n = 1 eV) spectra with standard flux levels: $\Phi = 10^{15}$ (n/cm^2·s) and $\Phi = 10^{14}$ (n/cm^2·s) respectively

Isotopes, J	$\sigma^J_{n,\gamma}$ (barn)		$T_{1/2}$ (year)	T^{transm} (year)		Recommendation from neutronic viewpoint
	Fast spectrum	Thermal spectrum		Fast spectrum	Thermal spectrum	
^{79}Se	0.03	0.1	6.5×10^4	7.3×10^2	2.2×10^3	*Questionable*
^{90}Sr	0.01	0.14	29	2.2×10^3	1.6×10^3	*Non-transmutable*
^{93}Zr	0.03	0.28	1.5×10^6	730	790	*Transmutable*
^{94}Nb	0.04	2.2	2.0×10^4	5.5×10^2	1×10^2	*Questionable or transmutable*
^{99}Tc	0.2	4.3	2.1×10^5	110	51	*Transmutable*
^{107}Pd	0.5	0.3	6.5×10^6	44	730	*Transmutable*
^{126}Sn	0.005	0.05	1×10^5	4.4×10^3	4.4×10^3	*Questionable*
^{129}I	0.14	4.3	1.6×10^7	160	51	*Transmutable*
^{135}Cs	0.07	1.3	2.3×10^6	310	170	*Transmutable*
^{137}Cs	0.01	0.02	30	2.2×10^3	1.1×10^4	*Non-transmutable*
^{151}Sm	0.7	700	89	31	0.3	*Non-transmutable or questionable*

Transmutation of ^{99}Tc or ^{129}I

	Masses (kg/TWhe)			σ_c	Φ	$T_{1/2}$ (transm.)
	Loaded	Transmuted	%	Barns	10^{15} n/cm^2/s	Years
In fast spectrum						
^{99}Tc	60.5	6.13	10.1	0.32	0.758	91
^{129}I	(26.0)	(3.96)	15.2	0.31	1.21	59
In moderated spectrum.						
1% ^{99}Tc	1.17	1.08	92	11.9	0.493	3.7
5% ^{99}Tc	5.86	3.62	62	5.85	0.372	10
10% ^{99}Tc	11.73	4.88	42	3.46	0.353	18
20% ^{99}Tc	23.46	6.19	26	2.19	0.317	32
To be compared to productions:			≈ 3 kg/TWhe (^{99}Tc) ≈ 0.7 kg/TWhe (^{129}I)			

Conclusions

- Excess neutrons produced by critical and sub-critical burners can, in principle, be utilised to transmute fission products. With the neutron fluxes available in these systems, it is theoretically possible to transmute the long-lived fission products; the transmutation of the more abundant short-lived fission products, however, is impracticable due to insufficient transmutation rates. In practice, the necessity of isotopic separations and difficulties in the preparation of targets present difficult obstacles for the fission product transmutation which currently reduce the number of candidate nuclides to only one or two, i.e. ^{99}Tc, and, possibly, ^{129}I.

- Minimising the fraction of specialised transmuters in the reactor park can result in an insufficient neutronic potential for transmuting the long-lived fission products of the entire park. The present study shows that critical and sub-critical TRU burners perform similarly in this respect. If the transmutation would be limited to ^{99}Tc and ^{129}I, all TRU burning strategies could, in principle, accomplish the task; however, a neutron shortage would not allow these fission products to be completely transmuted in the minor actinide burners of a double strata scheme.

Generation IV reactors

Definitions

- ADRIANA: "ADvanced Reactor Initiative And Network Arrangement": coordination action for mapping and gap analysis of research infrastructures for ESNII (http://adriana.ujv.cz)
- ADS: Accelerator Driven Systems
- ALFRED: Advanced Lead Fast Reactor European Demonstrator
- ALLEGRO: Experimental facility for the development and demonstration of GFR technologies
- ASTRID: Advance Sodium Technological Reactor for Industrial Demonstration
- ENEF: European Nuclear Energy Forum
- ESFRI: European Strategy Forum on Research Infrastructures
- Gen IV: 4th generation of nuclear reactors
- GFR: Gas-cooled Fast neutron Reactor
- GIF: Generation IV International Forum (www.gen-4.org)
- LFR: Lead-cooled Fast neutron Reactor
- MWe: Megawatt electrical power
- MWth: Megawatt thermal power
- MYRRHA: Multi-purpose hybrid research reactor for high-tech applications
- SET-Plan: The EU's Strategic Energy Technology Plan
- SETIS: SET-Plan Information System, managed by the EU Joint Research Centre
- SFR: Sodium-cooled Fast neutron Reactor
- SNETP: Sustainable Nuclear Energy Technology Platform (www.snetp.eu)
- SRA: SNETP's Strategic Research Agenda

Generation IV International Forum

Nuclear is a CO_2-free option for sustainable energy

New requirements for sustainable nuclear energy

Search innovative solutions for:

- Waste minimisation
- Natural resources conservation
- Proliferation resistance

Perform continuous progress on:

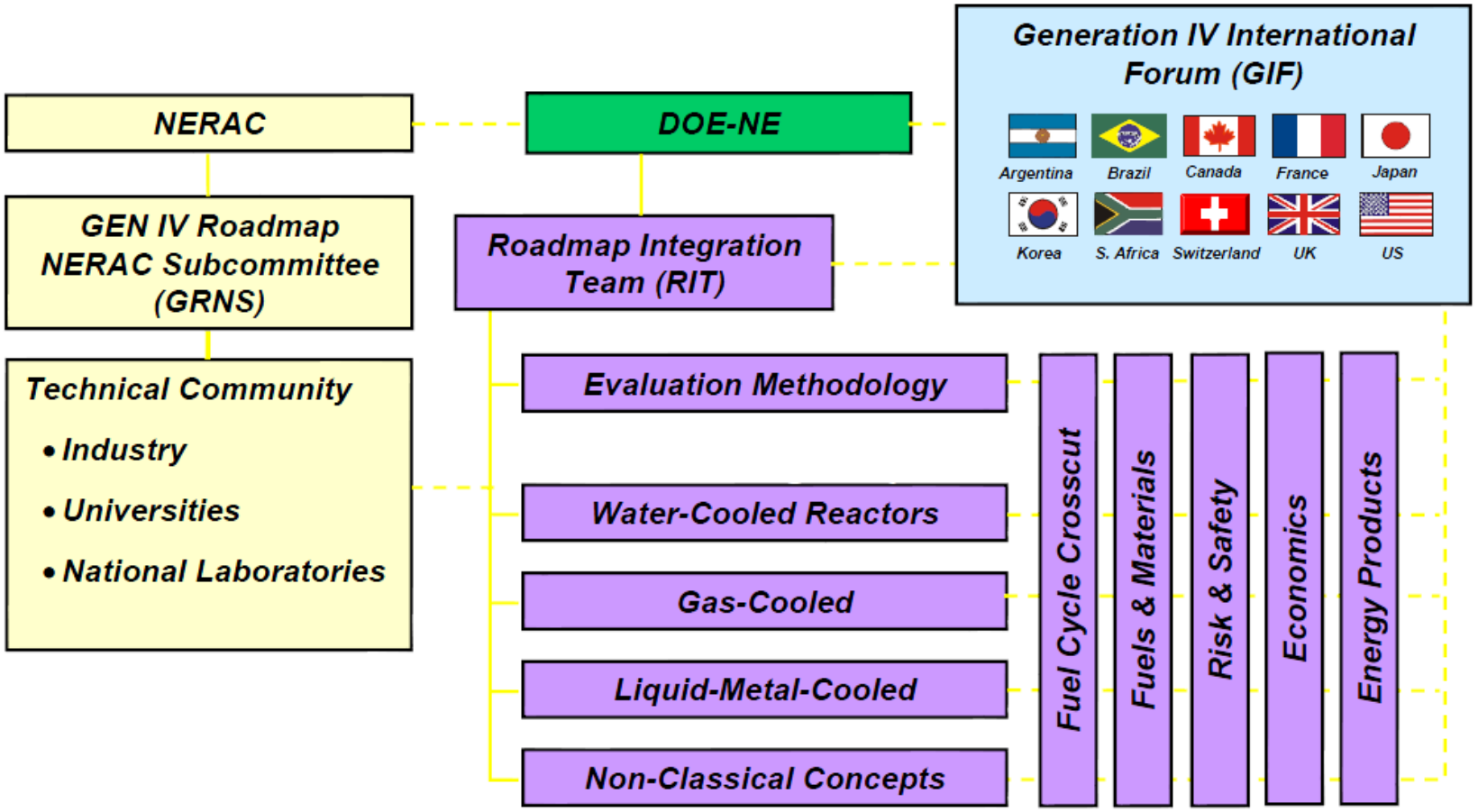
- Competitiveness
- Safety and reliability

Develop the potential for new applications:

hydrogen, syn-fuels, desalinated water, process heat

➔ ***Systems marketable from 2040 onwards***

Overall Generation IV roadmap organisation



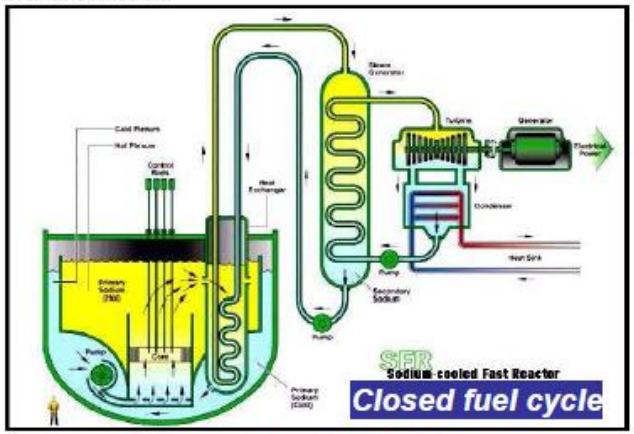
120 Gen IV concepts were proposed and classified in 19 homogeneous families

Nearly 100 technical experts contributed to the selection

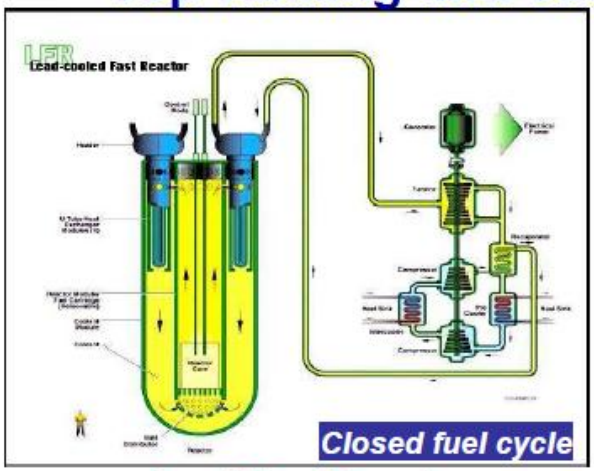
Japanese Chairmanship since end of 2009 (3 year term): Mr Yutaka Sagayama, from JAEA

VHTR	◆	◆	◆	◆	◆	◆	◆	◆	◆	
GFR		◆	◆	◆		◆				
SFR		◆	◆	◆	◆		◆	◆		◆
SCWR	◆	◆		◆						
LFR		◆		◆						
MSR		◆	◆							

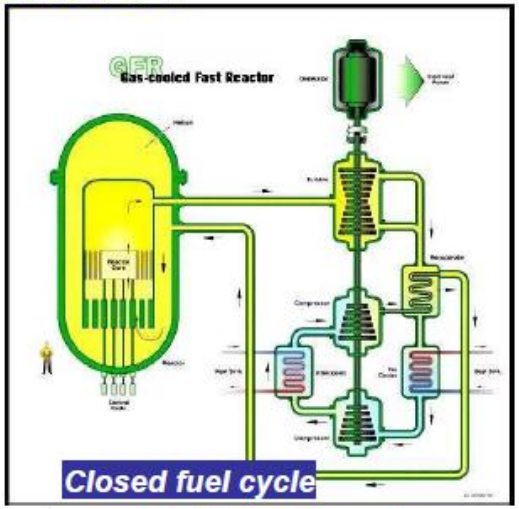
Top-ranking and outsiders ?



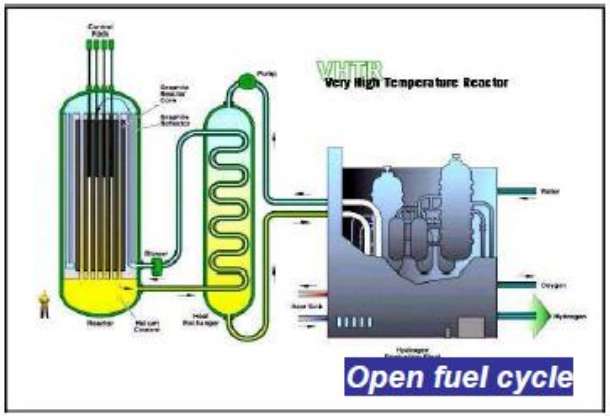
Sodium Fast Reactor



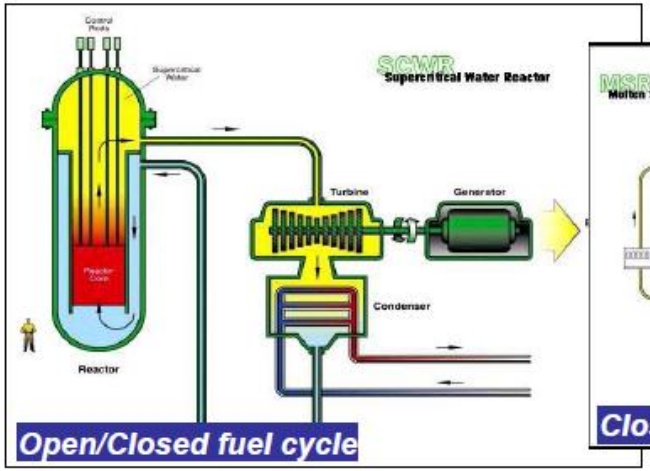
Lead Fast Reactor



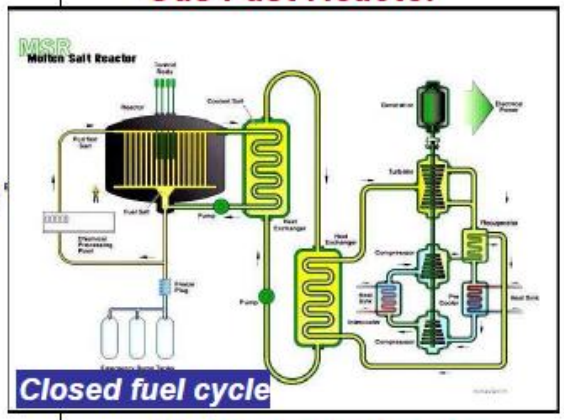
Gas Fast Reactor



Very High Temperature Reactor



Supercritical Water Reactor



Molten Salt Reactor

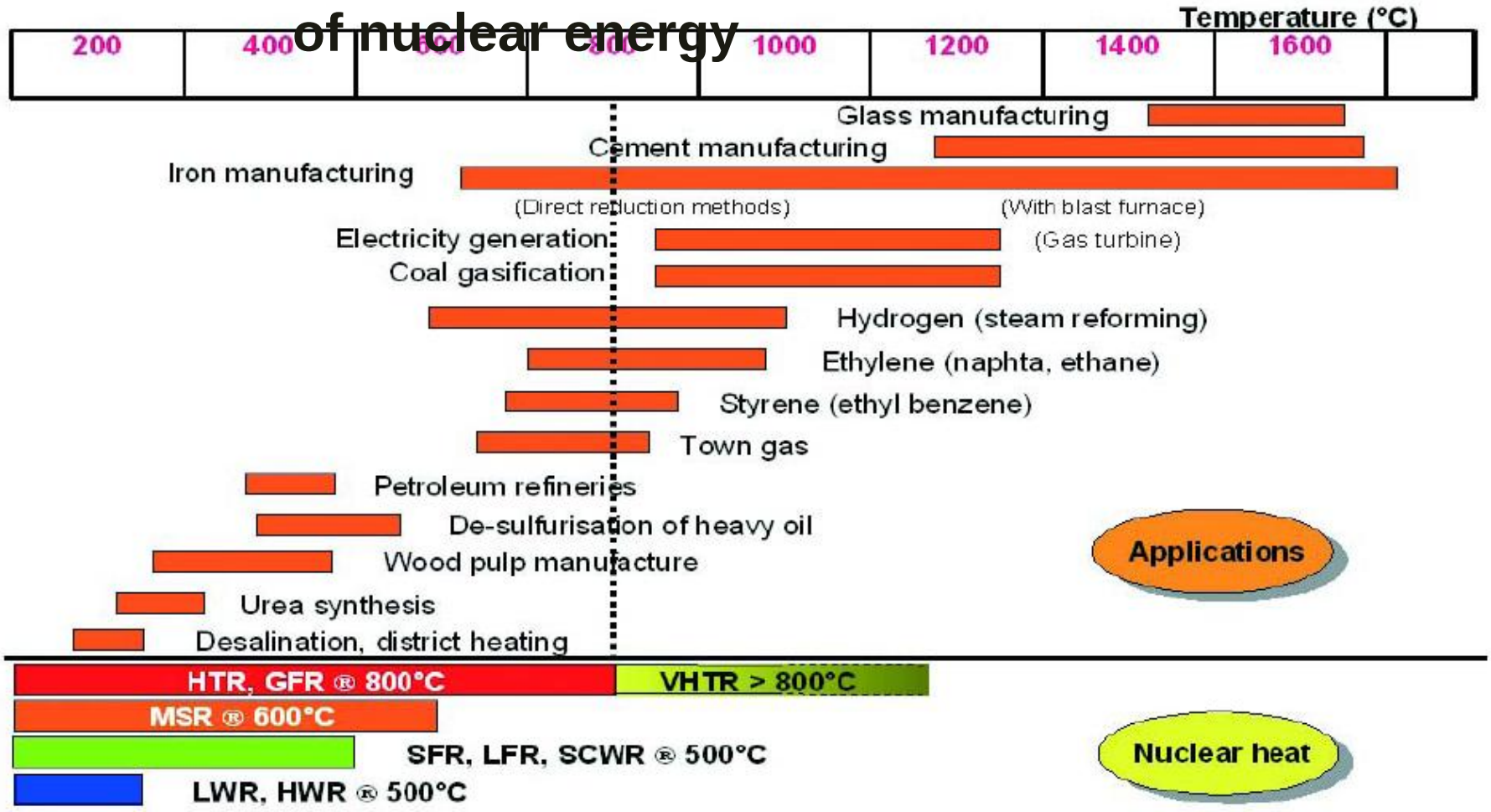
The recognition of the major potential of fast neutron systems with closed fuel cycle for breeding (fissil regeneration) and waste minimization (minor actinide burning)

	GFR	LFR	MSR	SFR	SCWR	VHTR
Coolant	helium	Pb or Pb-Bi	molten salt	sodium	supercritical water	helium
Spectrum (F fast, T thermal)	F	F	T	F	T/F	T
Thermodynamic efficiency (%)	48		44-50	42	44	50
Unit thermal power (MW)	2400	125-3600	~2000	350-1250 1500-4000	~3800	400-600
Core power density (MW/m^3)	50-100	10-150	22	350	100	6-10
Primary pressure (bar)	70-90	1-4	1-2	1-2	250	
Core outlet temperature (℃)	850	550-800	700	550	510	1000
Fuel type	Carbide or nitride	Nitride (or metal)	Molten salt (fluorides)	Oxide or metal	Oxide (UOx, MOX)	Carbide
Fuel burn-up (at%)	5	10-15	Continuous recycling	15-20	5	> 10
Fuel reprocessing	Pyro/Hydro	Pyro	Pyro	Hydro/Pyro	Hydro	

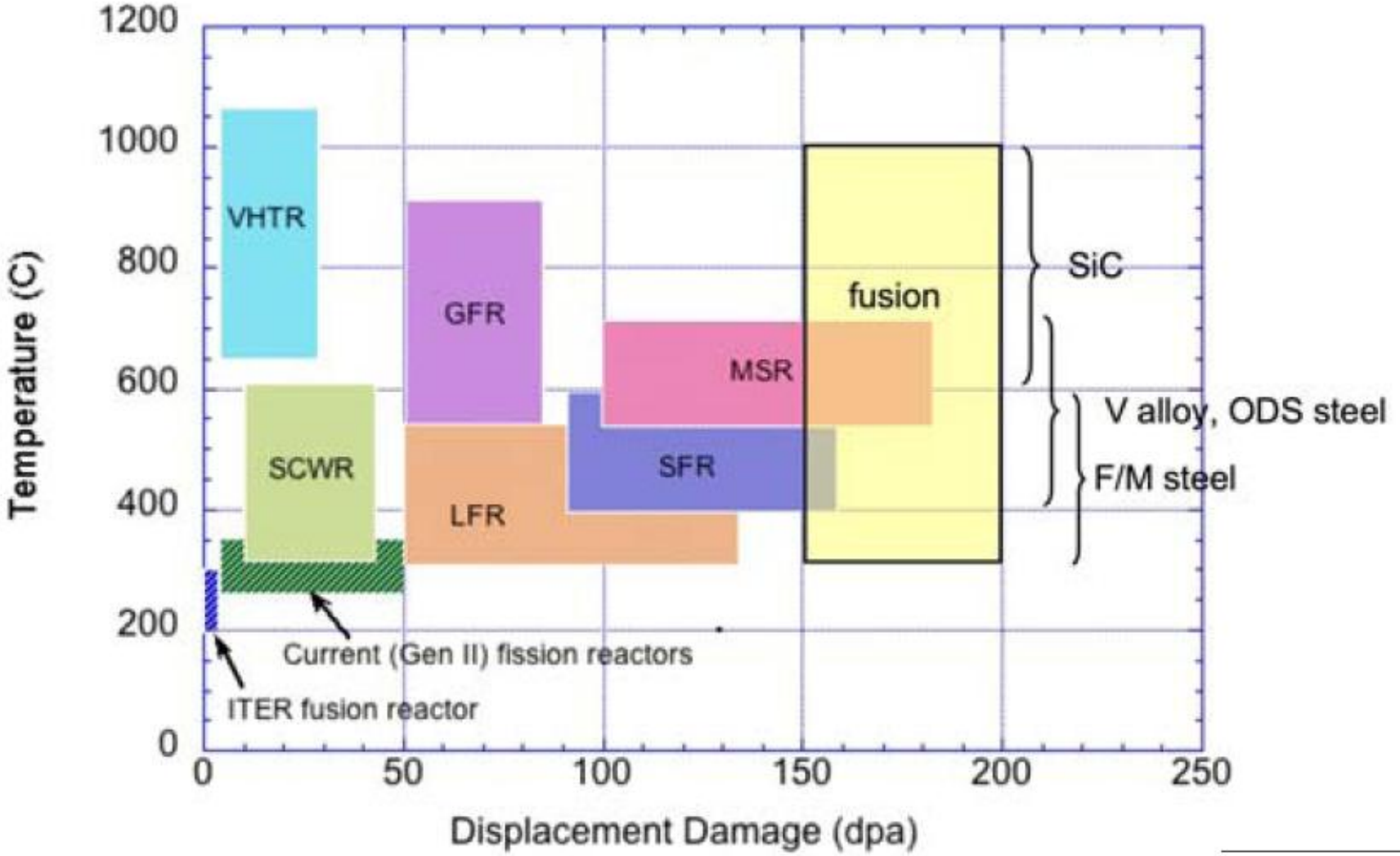
High temperature heat processes,

developing other applications

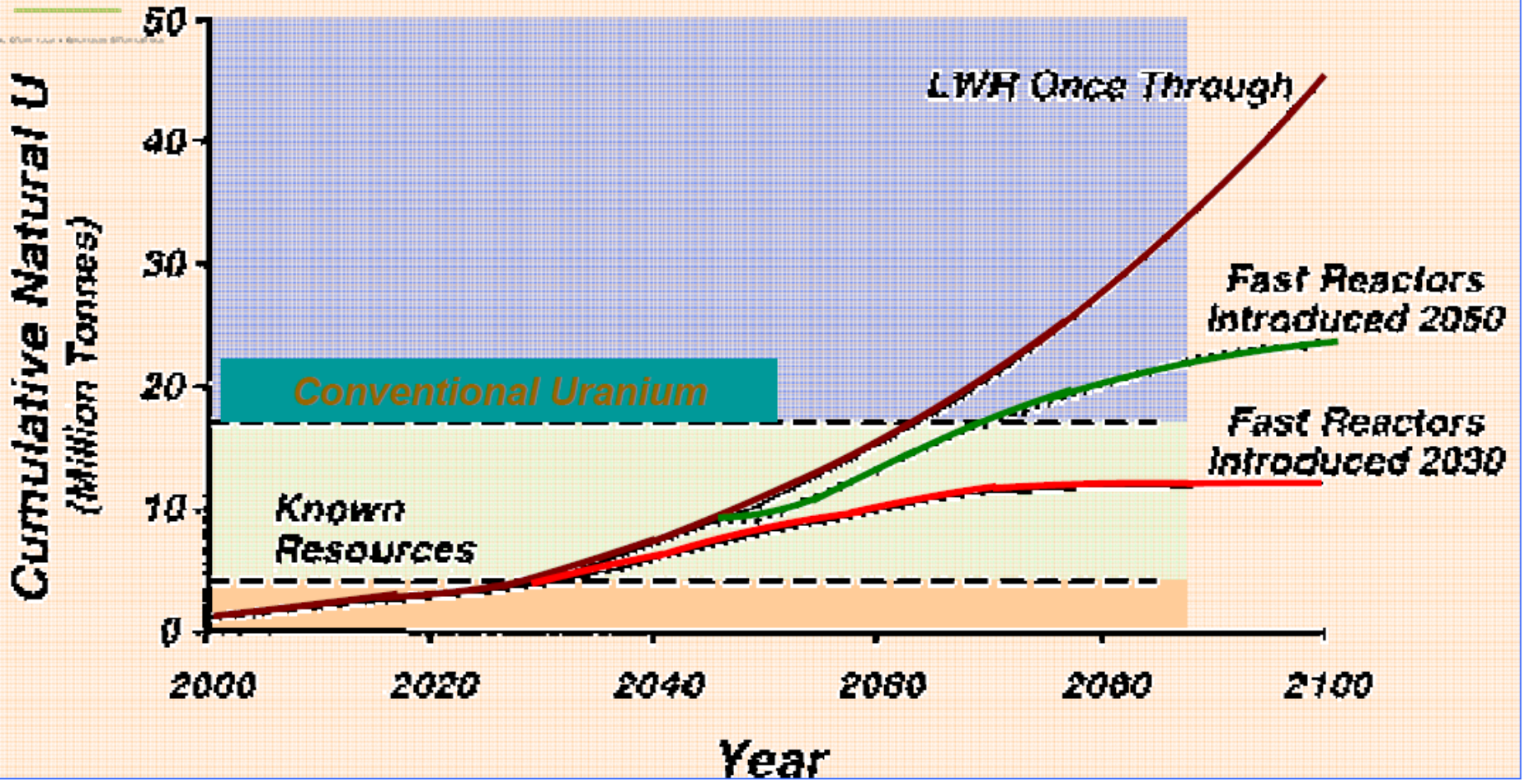
of nuclear energy



Operating temperature ranges and irradiation damage for different reactor concepts (dpa: displacement per atom)

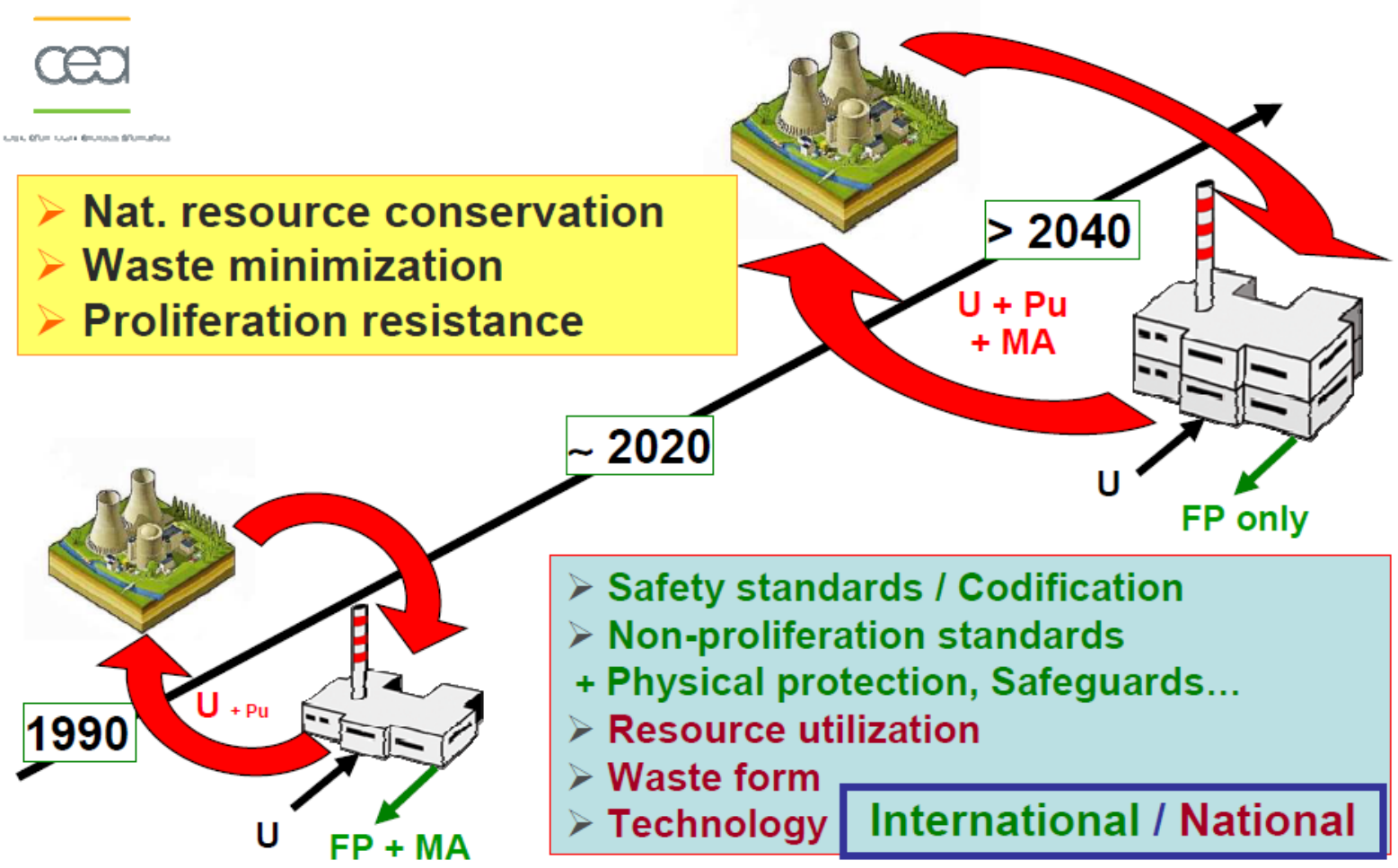


Worldwide Uranium Resource Utilization



GEN IV Fast Spectrum Systems with Closed Fuel Cycle (SFR, LFR, GFR, ADS)

Phased development of Fast Nuclear Energy Systems



Fast Reactors: Motivation and Principles

FRs : MAIN CHARACTERISTICS

- Fast neutron Reactors operate with a spectrum of neutron not slow down
- So they do not use any moderator
- Their first advantage is a capacity for breeding. This is this only reactor line that can use all the natural uranium
  - ^{238}U is transformed in ^{239}Pu by a neutron capture

Fast Reactors: Motivation and Principles

BREEDING CONDITIONS

- A nuclear reactor consumes fissile matter (C_F)
  - ^{3}U or ^{5}U or ^{9}Pu
- And produces it by transmuting fertile matter (P_F) :
  - $^{238}U \Rightarrow {}^{239}U \Rightarrow {}^{239}Np \Rightarrow {}^{239}Pu$
  - ou $^{232}Th \Rightarrow {}^{233}Th \Rightarrow {}^{233}Pa \Rightarrow {}^{233}U$

- $T_C = P_F/C_F$ is called breeding ratio. The breeding gain is $G = T_C - 1$

Only FRs can reach significant breeding gain

- One necessary condition for breeding is :

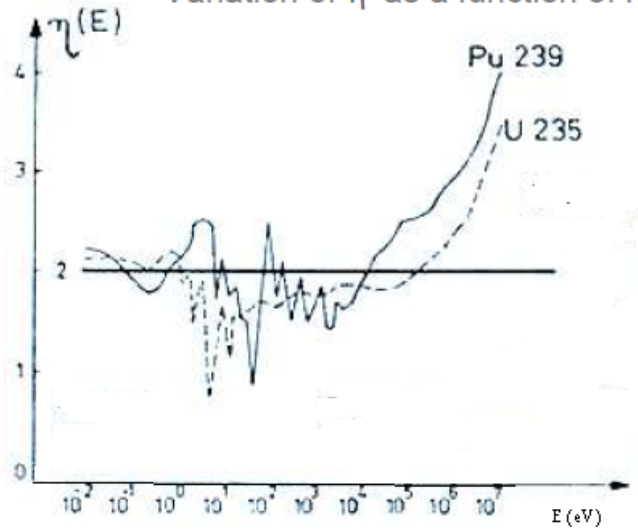
$$\eta > 2$$

η number of neutrons produced for each one absorbed

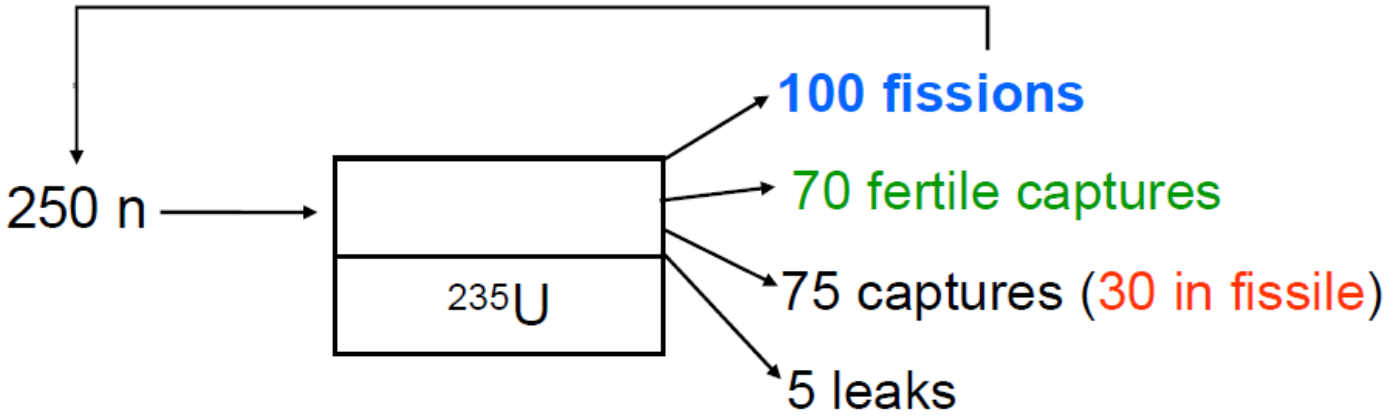
- ✓ 1 for a new fission,
- ✓ 1 to convert fertile into fissile
- ✓ + losses

- This condition is well obtained at high energy, mainly with ^{239}Pu

Variation of η as a function of neutron energy

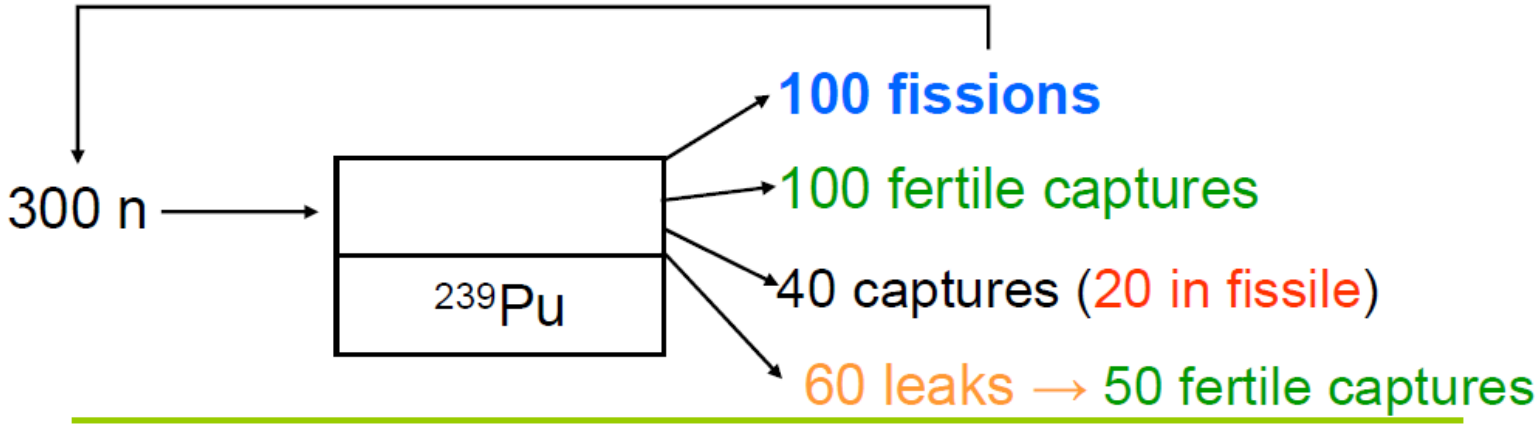


Simplified neutron balance of a PWR: Tc =Pf/Cf = 70/(100+30) ≈ 0,55



Simplified neutron balance of a FR: Tc interne= 100/(100+20) ≈ 0,85

Tc tot = (100+50)/(100+20) ≈ 1,25



Fast Reactors: Motivation and Principles

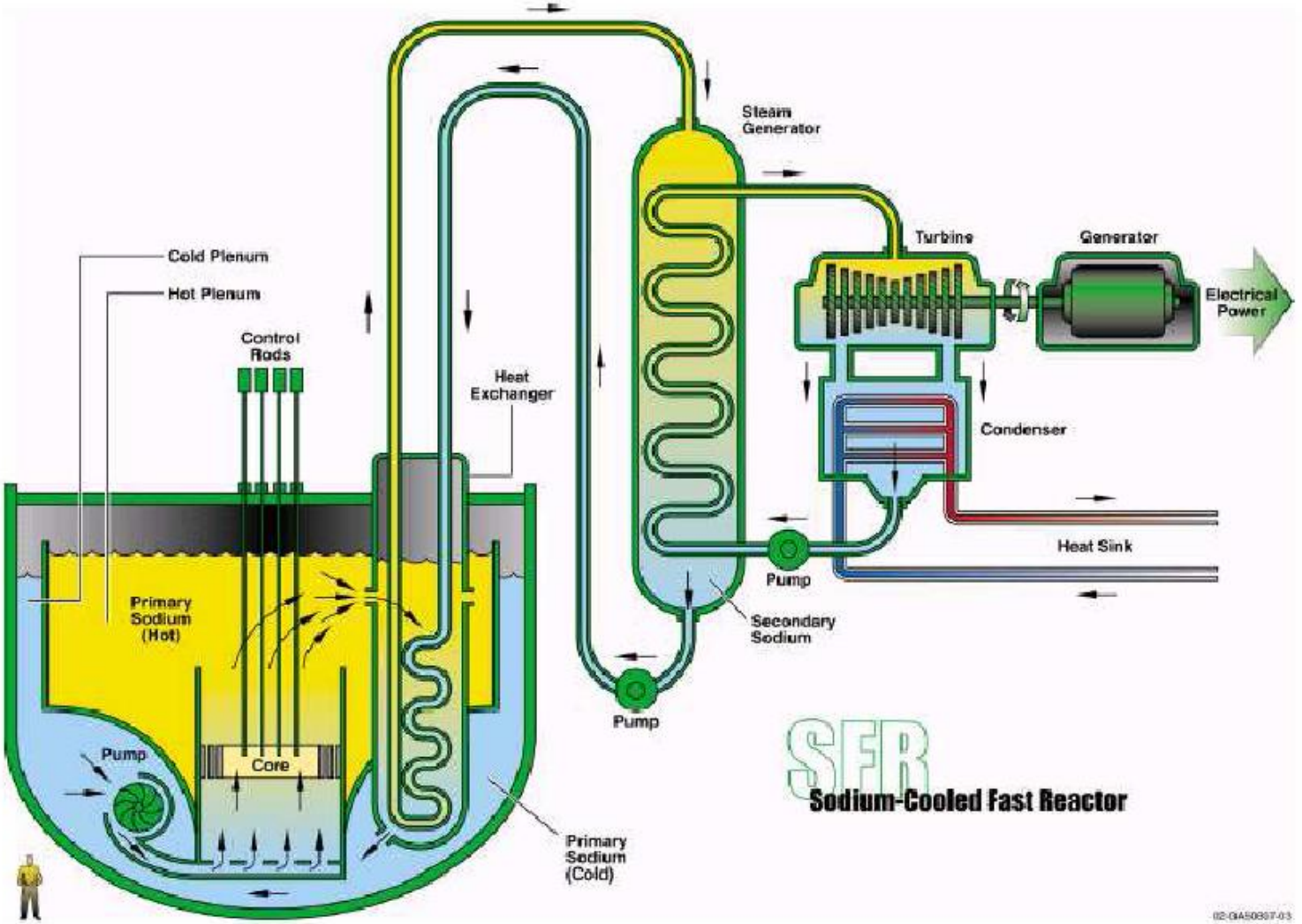
URANIUM USE IN A FR

- With a FR fleet
  - Recovered Pu by reprocessing is utilized to fabricate new fuel
  - Only natural U (mining or reprocessing) enters the cycle

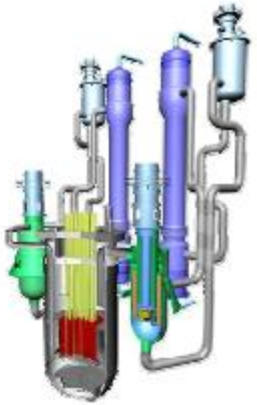
- All the Uranium can be consumed. Natural reserves are multiplied by a factor around **100**.

- The breeding ratio can be adjusted to the needs: Pu production where they increase; consumption at the opposite.
  - The reactor core can consume Pu: high Pu ratio and low U content

Sodium Fast Reactor



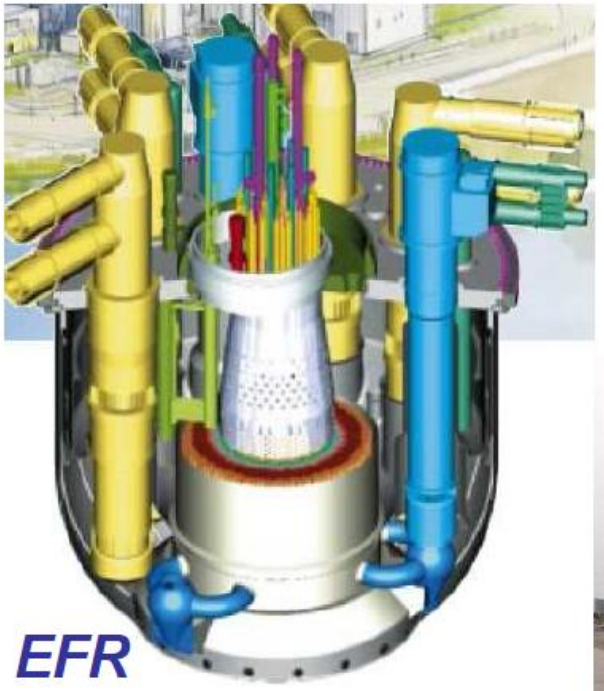
SFR system today



- Several projects
  - JSFR in Japan
  - ASTRID in France
  - ***BN800*** & BN1200 in Russia
  - ***CEFR in China and PFBR in India***
- An active collaboration
- Hard to switch from GEN III SFR to GEN IV SFR
- Competition between building soon and innovate
- Enhanced Safety is of first concern
  - A fast neutron core is not in its more reactive configuration
  - Sodium is reactive and opaque
- An incentive to share correctly R&D world wide

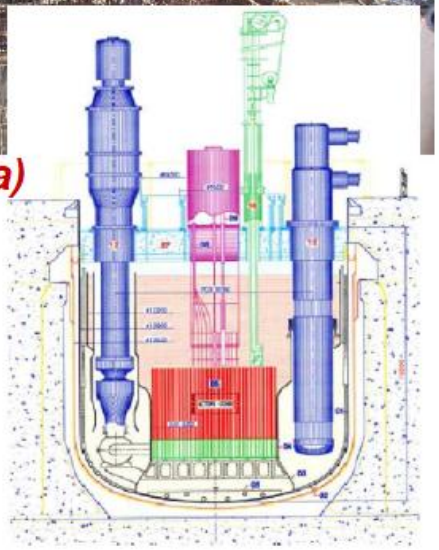
European Fast reactor (EFR) Project

- *1500 MWe (1988–1998)*
- *Pool type reactor design*



EFR

PFBR (India)



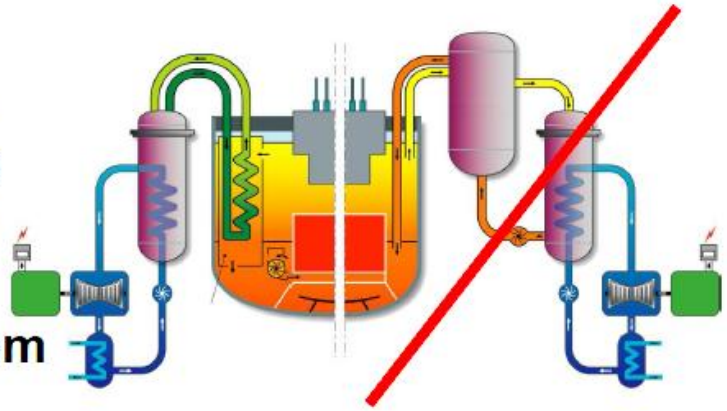
CEFR (China)

and BN 800 (Russia)

SFR-Astrid

Four tracks for innovation

– An attractive core with enhanced safety
– Improved resistance to severe accidents
– Energy conversion systems to minimize sodium risk
– Simplified and optimized plant and system design (economy, operability, ISIR)

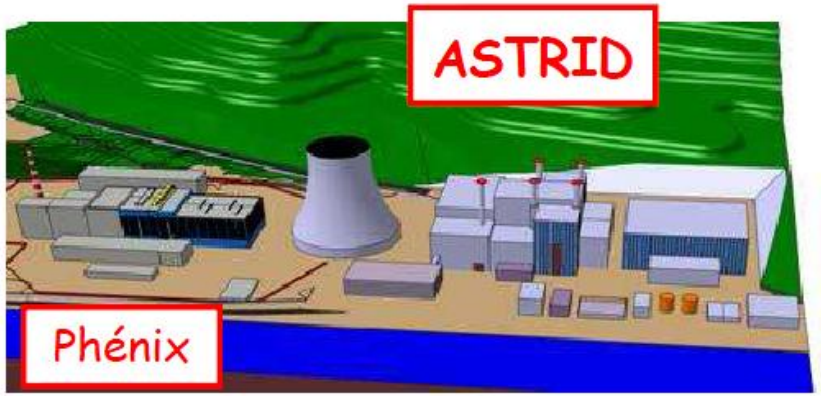


The ASTRID SFR prototype:

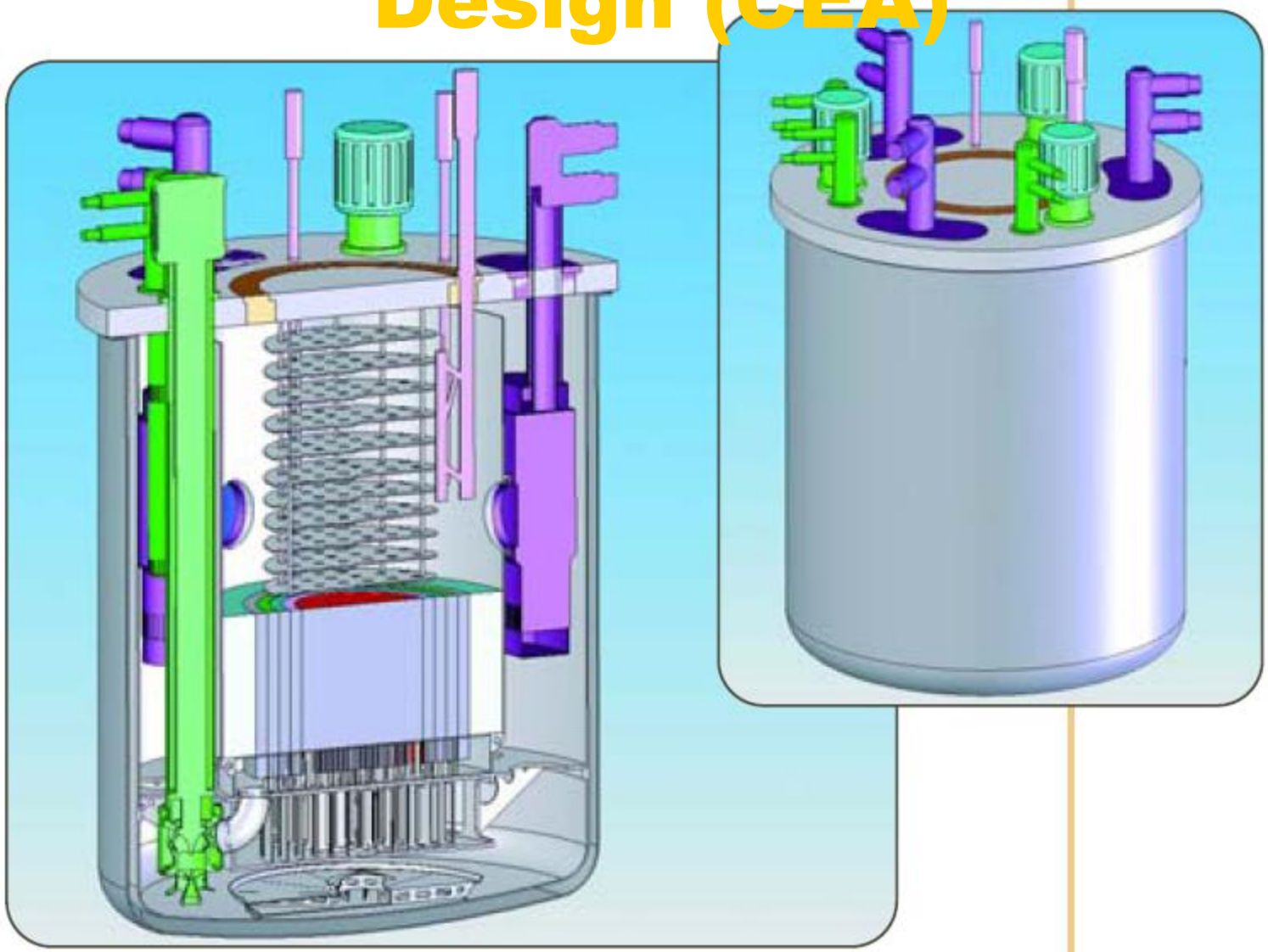
600-1500 MWt (250-600 MWe)

→ Precursor of a first of a kind

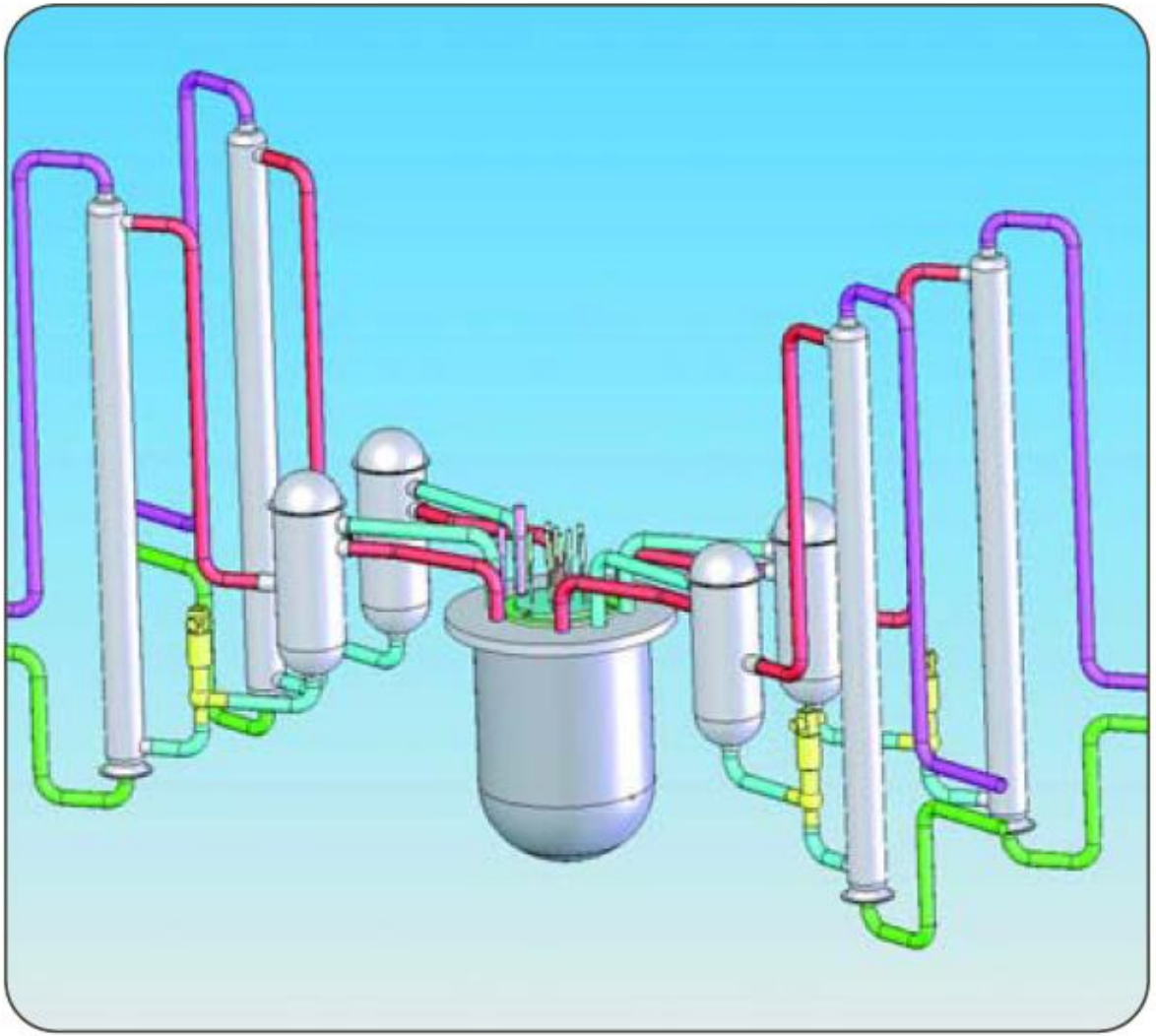
→ Irradiation tool



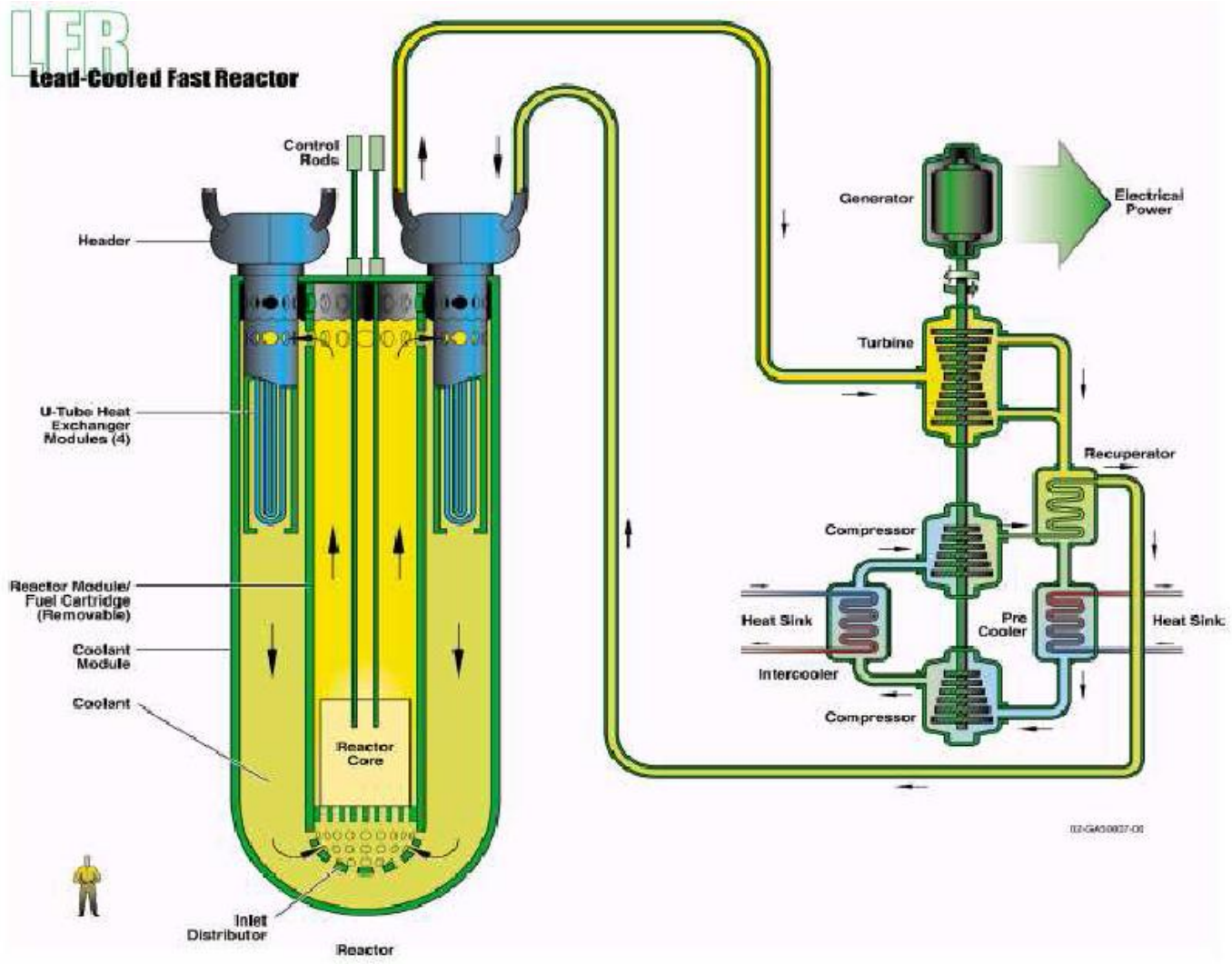
1500 MWe Innovative SFR Pool Design (CEA)



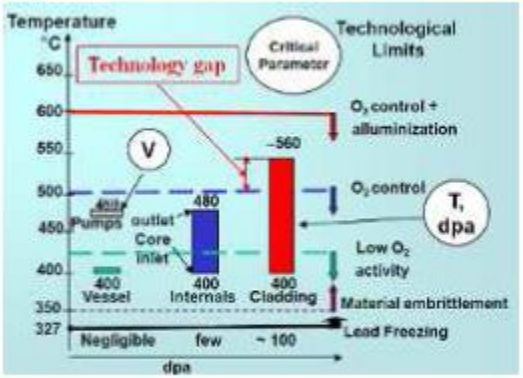
1500 MWe Innovative SFR Loop Design (CEA)

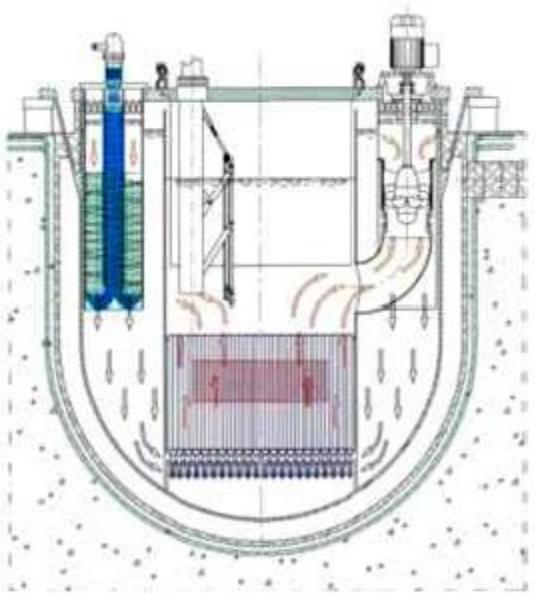


Lead Fast Reactor



LFR system today

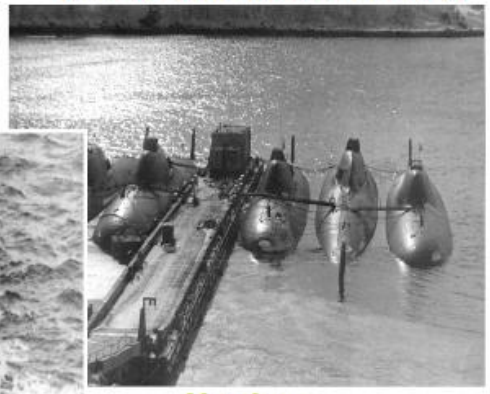




- Waiting for additional contributions
  - Russian program?
- Low progress on corrosion
  - The main challenge
- New design adapted to present know-how
  - But with reduced performance
  - Safety to be assessed
- Advantage upon SFR to be improved
- Often mixed with Accelerator Driven System development

No past experience on Pb-cooled reactors but a lineage of Pb-Bi cooled reactors in Russia :
→ 8 submarines (~ 80 reactor.years)

(Source IPPE, A. Zrodnikov)

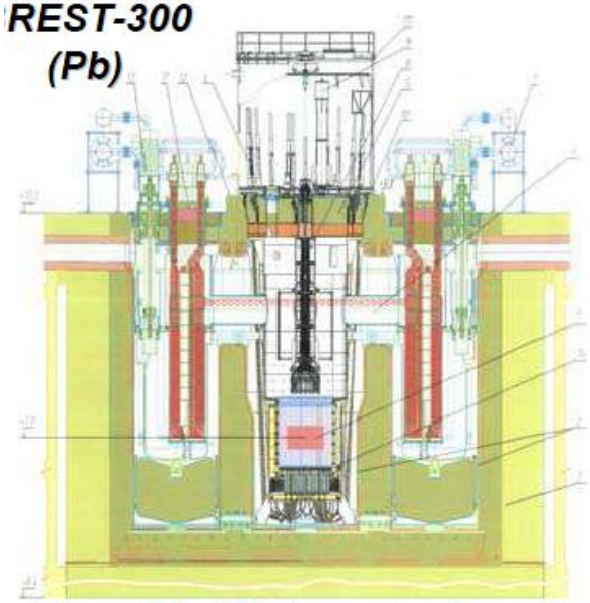


Disarmament of the last submarine in 1995

Nuclear submarine-705 serial (1976-1996)

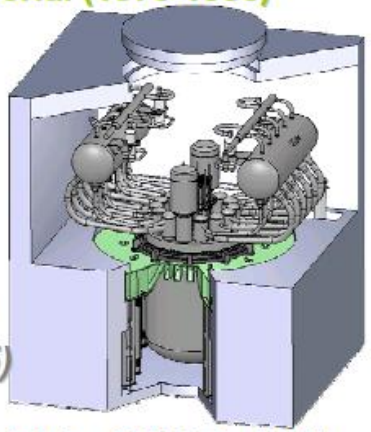
Nuclear submarine-705 demo (1971)

BREST-300 (Pb)



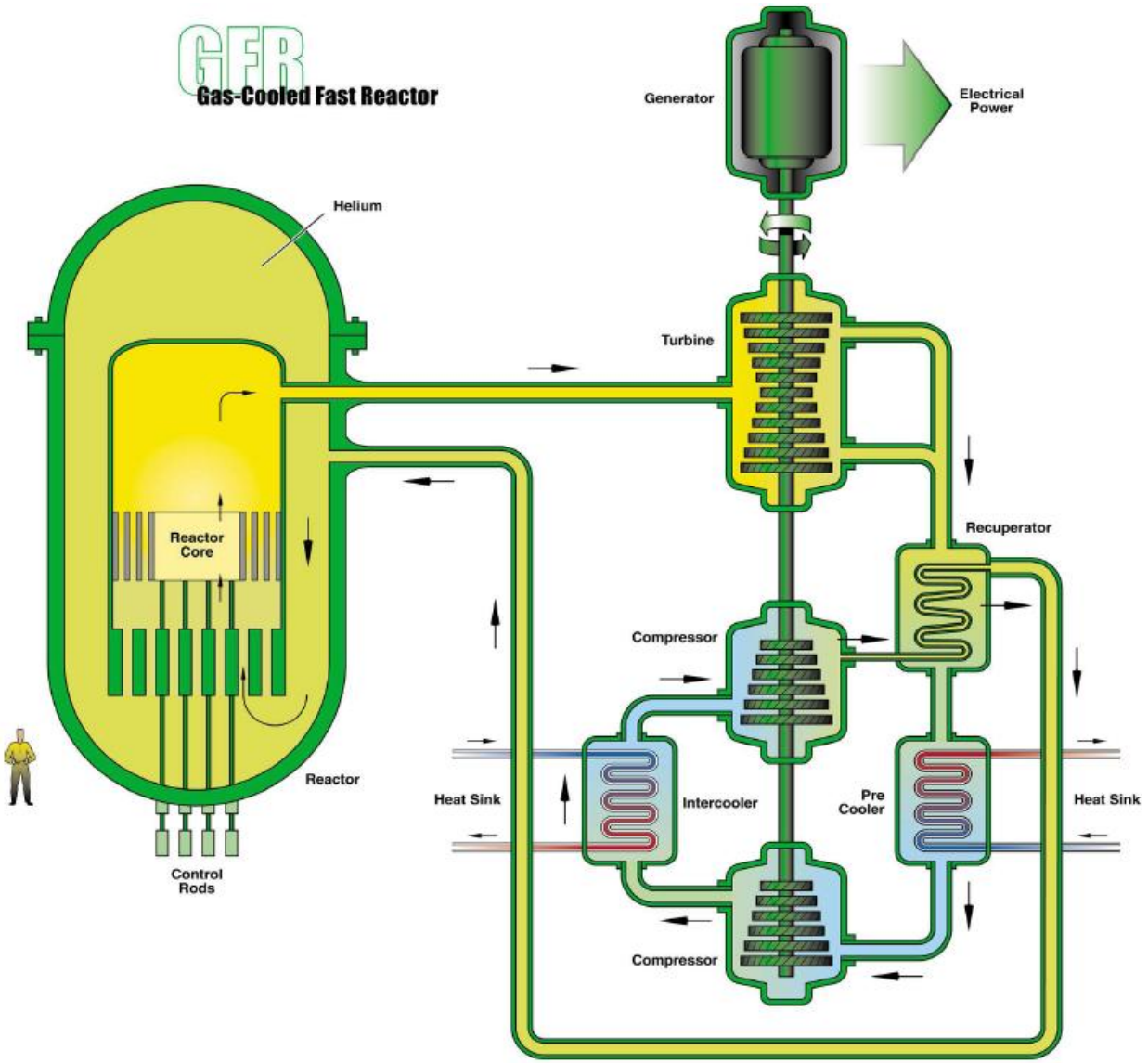
Prototype nuclear submarine Project 645 (1963)

SVBR-100 (Pb-Bi, 2015)



- **Renewed interest after the 90's : ADS, Russia (projects BREST-300, SVBR-100), SA (SSTAR) and Europe (ELSY, ALFRED demo)**

Gass Fast Reactor



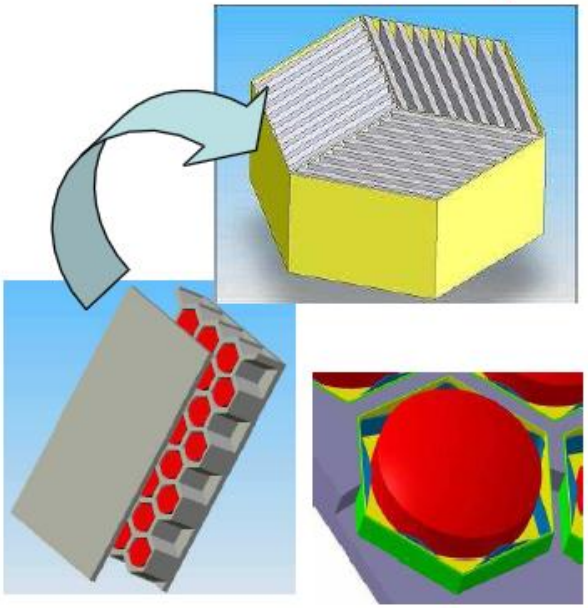
GFR system today

- Interest limited to few countries
- A new consistent design with attractive features
  - But a complex primary structure
  - Indirect conversion cycle
- Mainly paper studies
  - Technology to be assessed
- Ceramic clad for fuel is a big challenge
  - More than 15 year developments ?
  - Metallic alloy as a backup is also difficult
- Longer term deployment scheduled
- Tendency to decrease (optimize) performance for a first construction
  - T° < 850°C, breeding ratio ±0, no minor actinides …

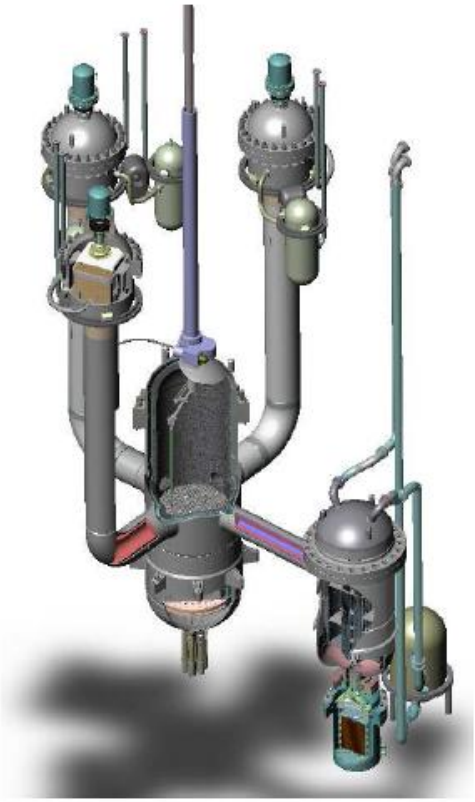
A first consistent design for a 2400 MWt GFR

Robust decay heat removal strategy (passive after 24hrs)

GFR preliminary feasibility report issued January 2008



Innovative fuel

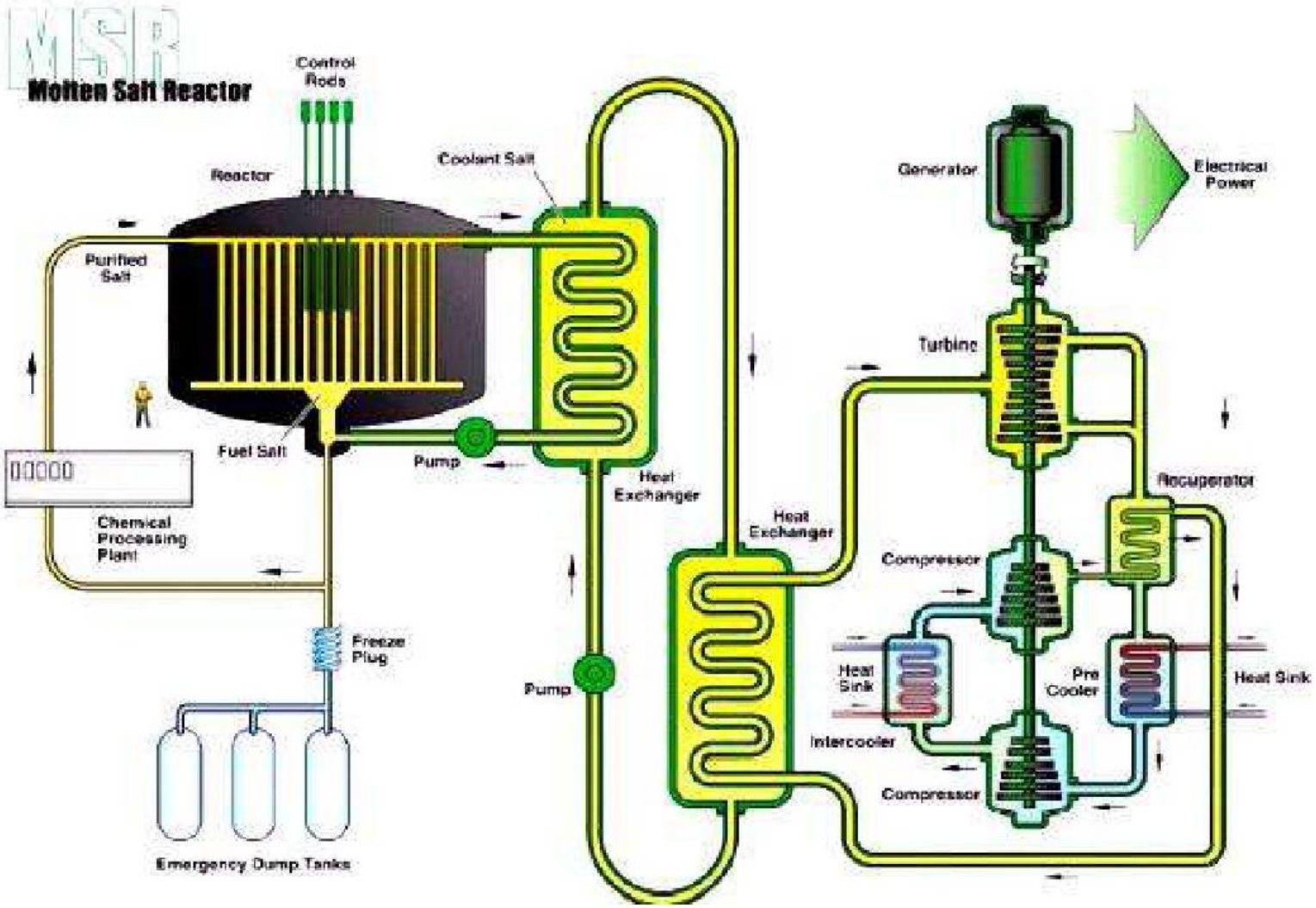


GFR 2400 MWt (1100 MWe) reference concept

ALLEGRO (50-70 MWt)

ALLEGRO

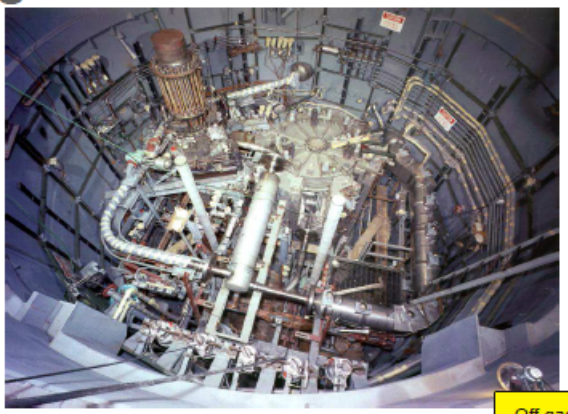
Molten Salt Reactor



MSR

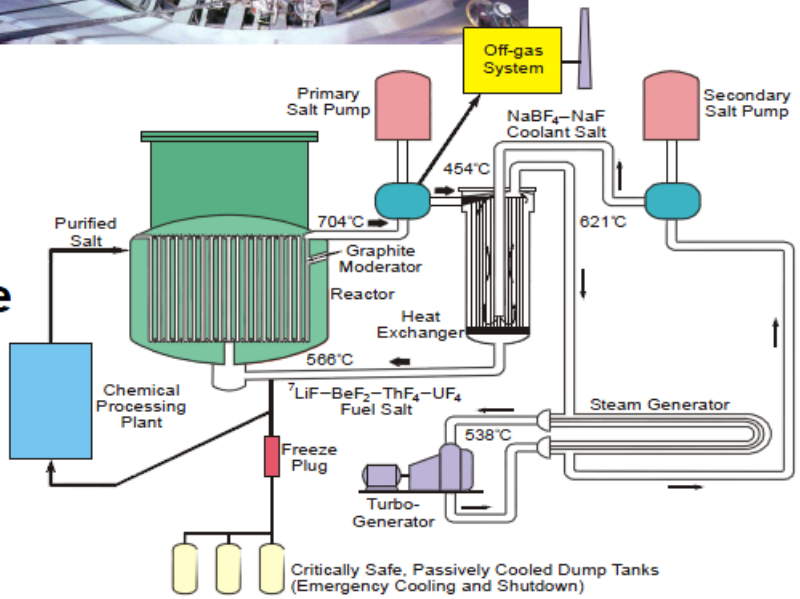
A technology deeply investigated in the 60's and 70's

- **The successful operation of the MSRE (Molten Salt Reactor Experiment) from 1965 to 1969**



3 fuel types:
- **Uranium enriched 30% with ^{235}U**
- **Pure ^{233}U**
- **^{239}Pu**

Fuel salt reprocessing not implemented in MSRE, except the off-gas system



- **The detailed design of a 1000 MWe breeder reactor in Th/U cycle, the MSBR**

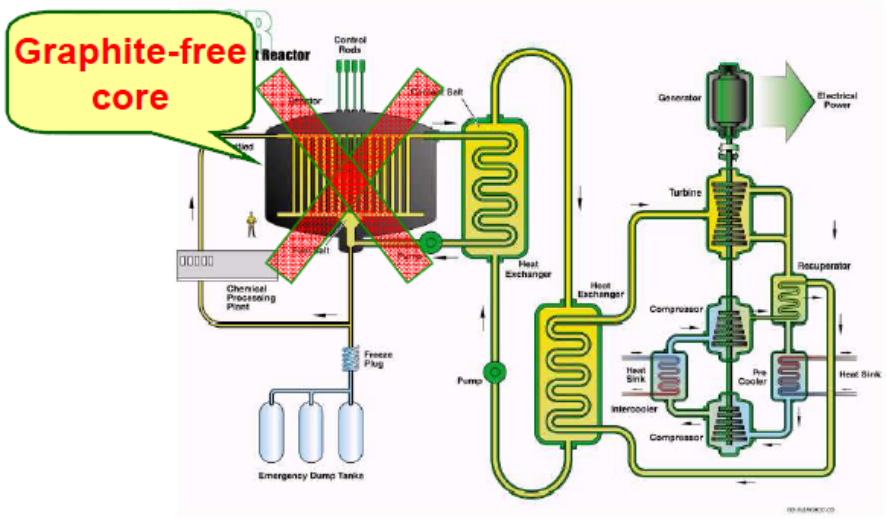
Fuel salt
71%LiF-16%BeF_2-12%ThF_4-0.3%UF_4

The termination of the program in 1975 left a number of open questions relative to the viability of MSRs

Molten Salt Fast Reactor

Recently, the interest has focused on fast neutron MSR (MSFR) as a long term alternative to solid-fuelled fast neutron reactors

cf. SNETP-SRA (June 2009): "*Molten salt reactors may represent a sustainable nuclear energy system in the long term…*"



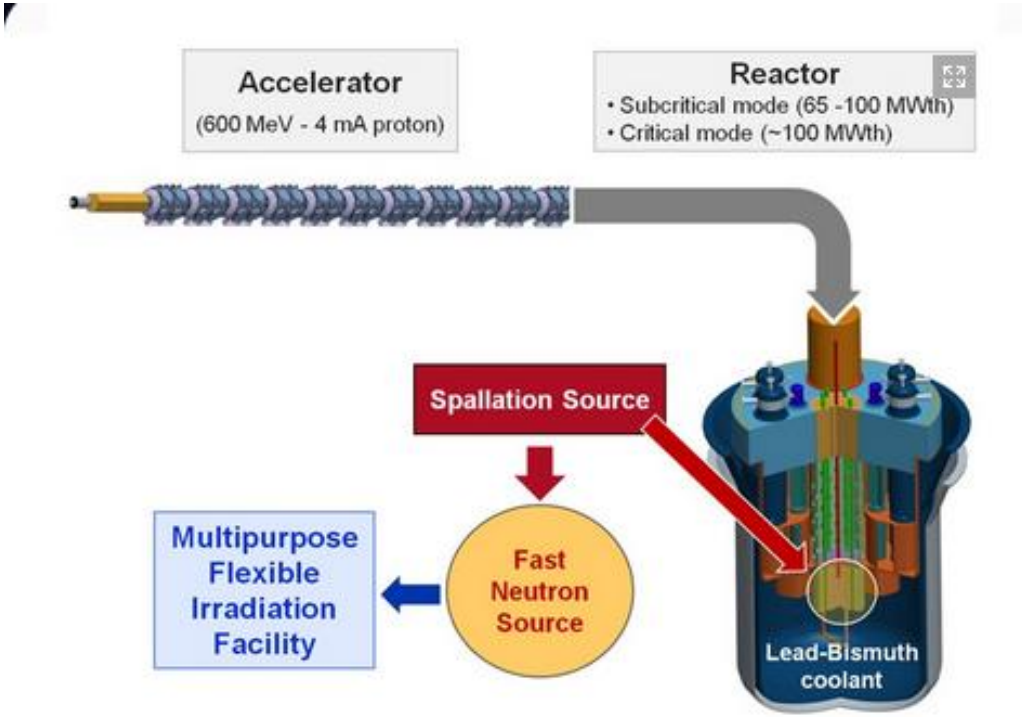
Attractive features (very negative feedback coefficients, simplified fuel cycle,…)

Technological challenges to be addressed and safety approach to be established

- **Reference MSFR concepts have been developed for breeding and for Pu and MA transmutation and recycling**
- **Opportunities are offered by liquid salts for heat transport in other systems (SFR, LFR, VHTR)**

MYRRHA PROJECT

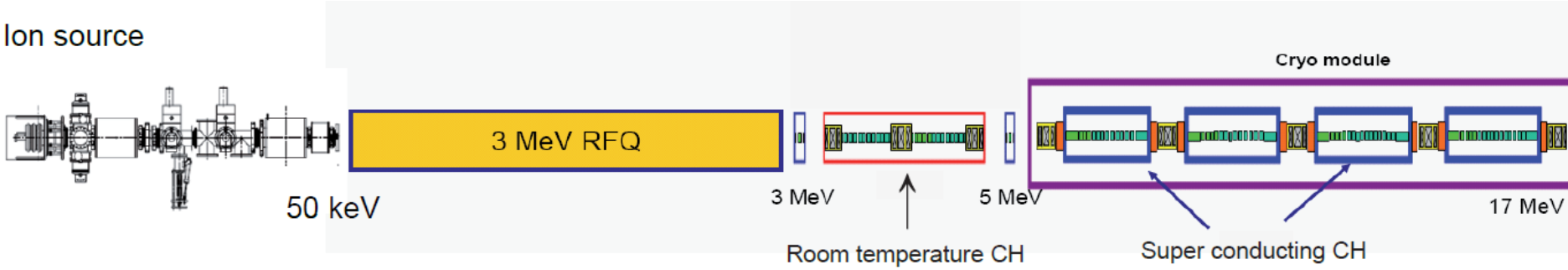
- MYRRHA, a flexible fast spectrum research reactor (50-100 MW_{th}) is conceived as an accelerator driven system (ADS), able to operate in sub-critical and critical modes. It contains a proton accelerator of 600 MeV, a spallation target and a multiplying core with MOX fuel, cooled by liquid lead-bismuth (Pb-Bi).

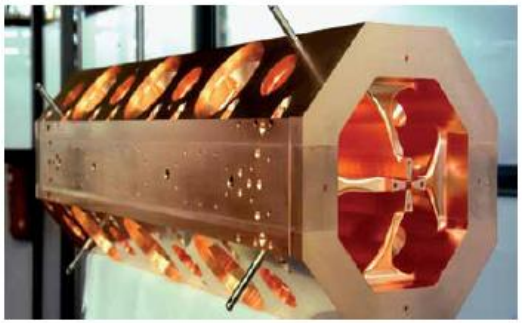


Nuclear Research Centre in Mol

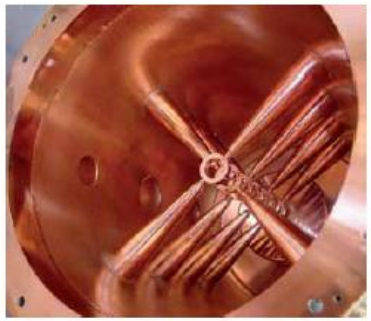
- SCK•CEN, the Belgian Nuclear Research Centre in Mol has been working for several years on the design of a multi-purpose irradiation facility in order to replace the ageing BR2 reactor, a multi-functional materials testing reactor (MTR), in operation since 1962.

Ion Source of Accelerator of MYRRHA

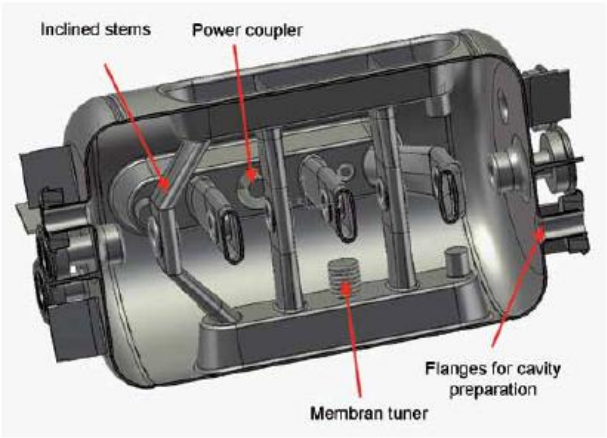




RFQ element

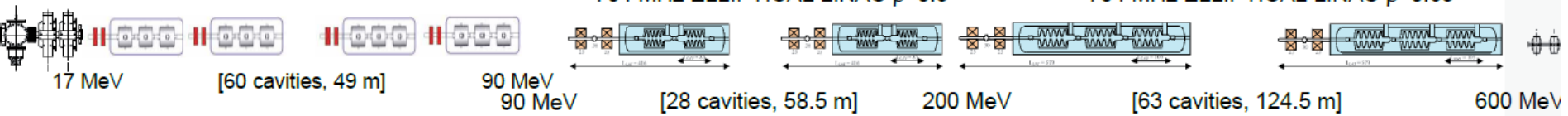


Copper CH RF cavity



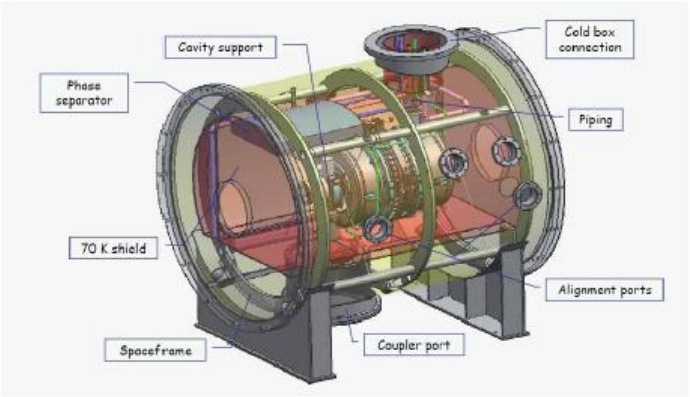
Superconducting CH cavity design

Accelerator

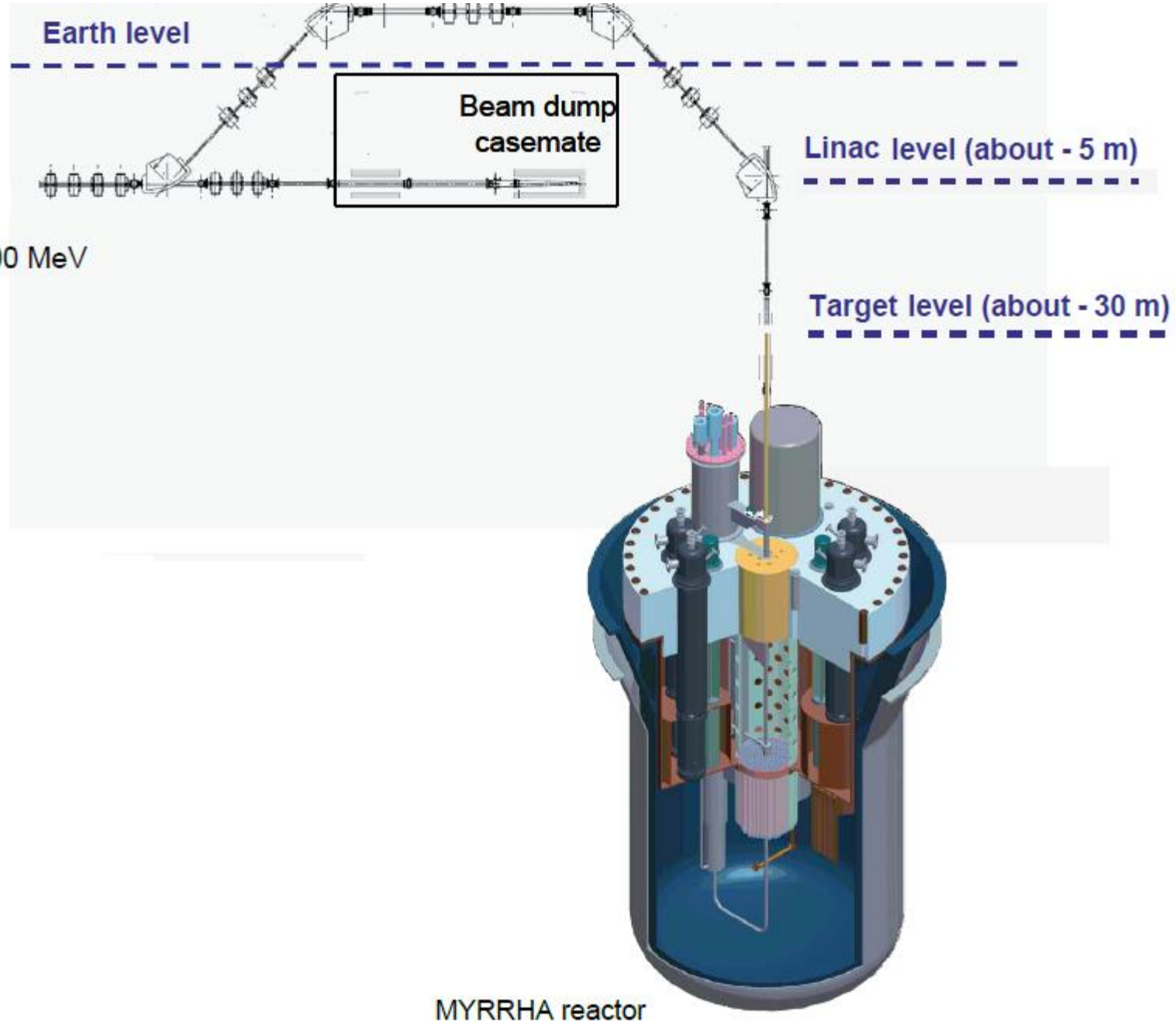


5 element elliptical cavity

Elliptical cavity envelope



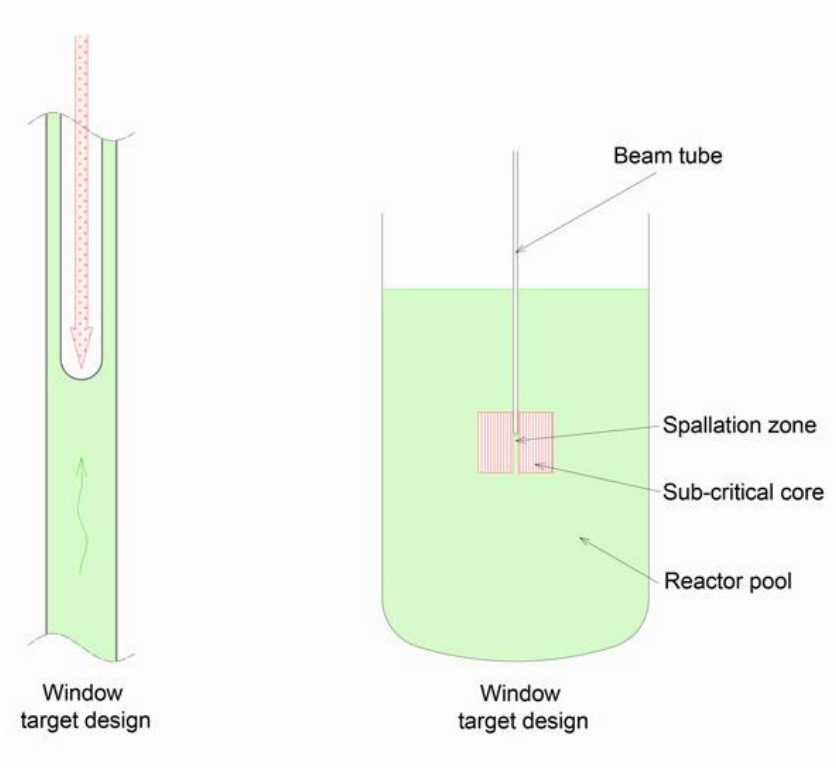
Cryomodule for the elliptical cavities



MYRRHA reactor

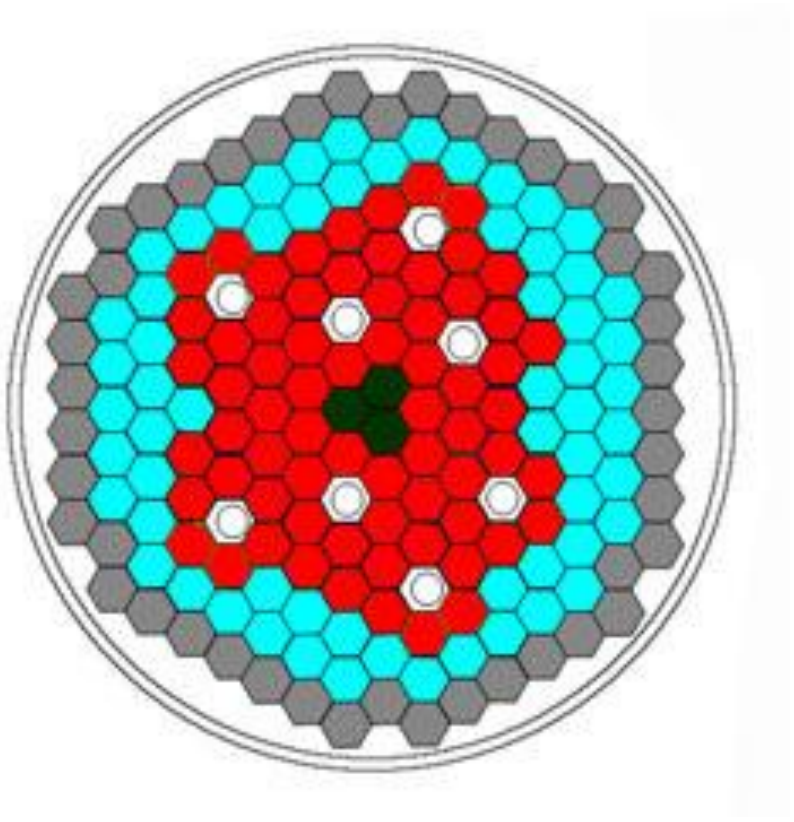
Proton beam

- Proton energy
- 600 MeV
- Max. beam intensity
- 4 mA in Continuous Wave mode
- Beam entry vertically from above
- Beam stability
- Energy: 1 %, Intensity: 2 %, Size: 10 %
- Beam footprint on target
- Circular, "doughnut" $\phi_{out} = 100$ mm, $\phi_{in} = 50$ mm



Reactor Core

- The fresh core of MYRRHA contains a lattice of 183 hexagonal channels of which 68 are loaded with fuel assemblies (this configuration considers a core composed of only fresh fuel). A space of 3 hexagons is cleared at the centre of the sub-critical core to house the spallation target module.



Reactivity effects
1.E5*(1-keff)/keff

Reactivity effects and safety parameters for the sub-critical mode operation
The adopted sub-criticality level (-4700 pcm for a k_{eff} of 0.955) assures a comfortable margin for safe operation taking into account foreseeable positive reactivity insertions due to fuel and coolant temperature effects, radial and axial expansion.

- The Doppler constant, Tdk/dT, is -370 pcm, which is lower but of the same order of magnitude as for a Na-cooled fast reactor.
- The LBE-coolant temperature coefficient, dk/dT, is small but negative, -2.4 pcm/K, over the temperature range from 150 C to 400 C.
- The effect of voiding the spallation target module has a negative reactivity contribution of about -1000 pcm. Voiding the group of 21 central assemblies for 30 cm of the active height has about +300 pcm positive reactivity contribution, while a full voiding has a zero or slightly negative reactivity contribution.
- Voiding all the assemblies in the core (both partial as total height) has a negative contribution of up to -2500 pcm. Flooding of the spallation target tube with lead-bismuth eutectic has a maximum of +144 pcm reactivity contribution.
- The combined effect of radial and axial expansion of the structures going from the cold state (150 C) to the hot state ($T_{average}$ = 350 C) amounts to -490 pcm

Reshuffling

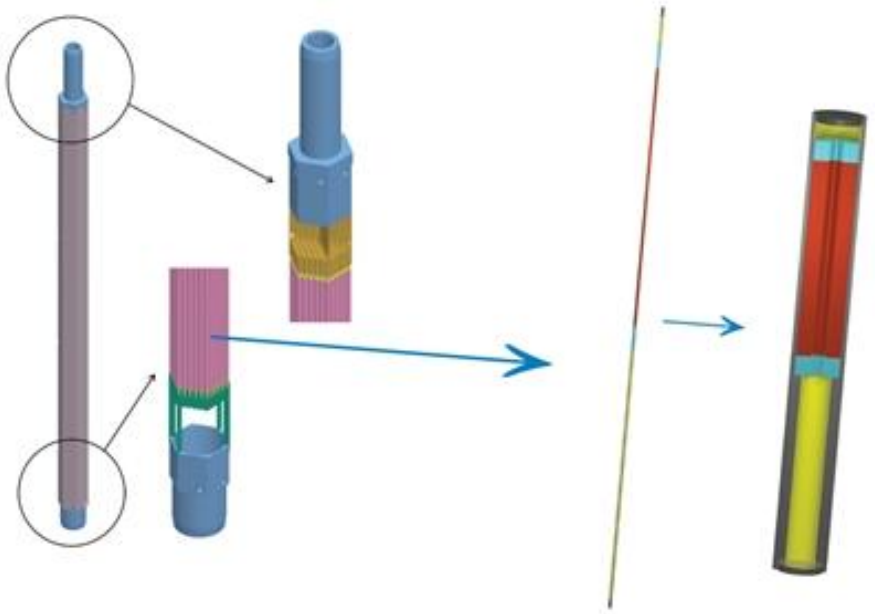
- A ten-step reshuffling scheme has been studied for a core power of 85 MW_{th} in sub-critical mode operation. The total number of fuel assemblies is determined at every beginning of 90-day operation as to set k_{eff} to 0.955.
- Three options are available to handle the reactor power drop during the operational cycle of 90 days.

1. In the first option, no measures are taken and one has to live with the power gradient over 90 days.
2. A second option would be gradually increase the beam current to compensate the power loss.
3. A third option would be to use a burnable poison to keep k_{eff} as constant as possible.

•

Fuel element pre-design

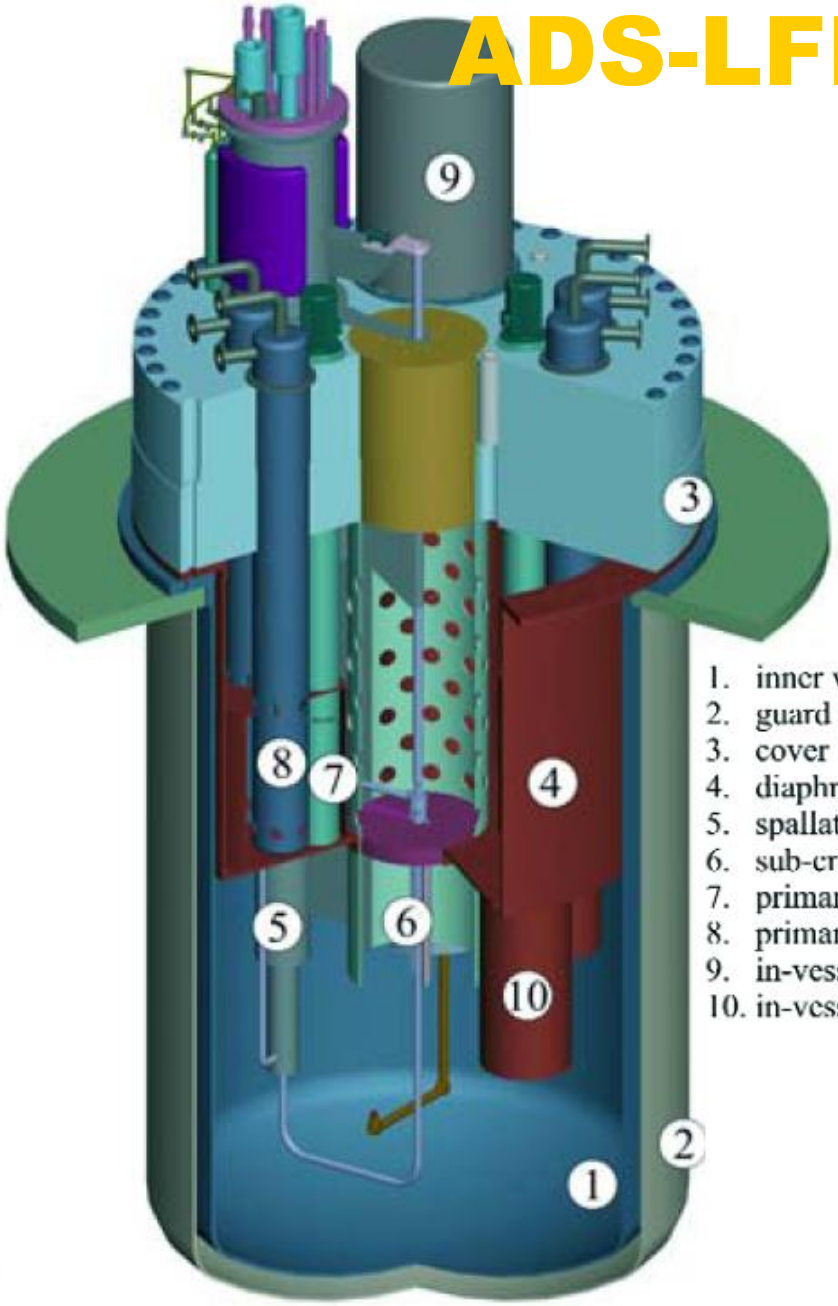
A mixed plutonium-uranium oxide fuel (MOX) was selected as the main candidate since the beginning of the MYRRHA project, taking into account a large experience in the MOX production in Europe and its better neutronic properties, compared to uranium dioxide, in the fast neutron spectrum. Based on past experience within the fast reactor programme in Europe, enrichment of 30 to 35 % of plutonium is envisaged.



Fuel assembly layout

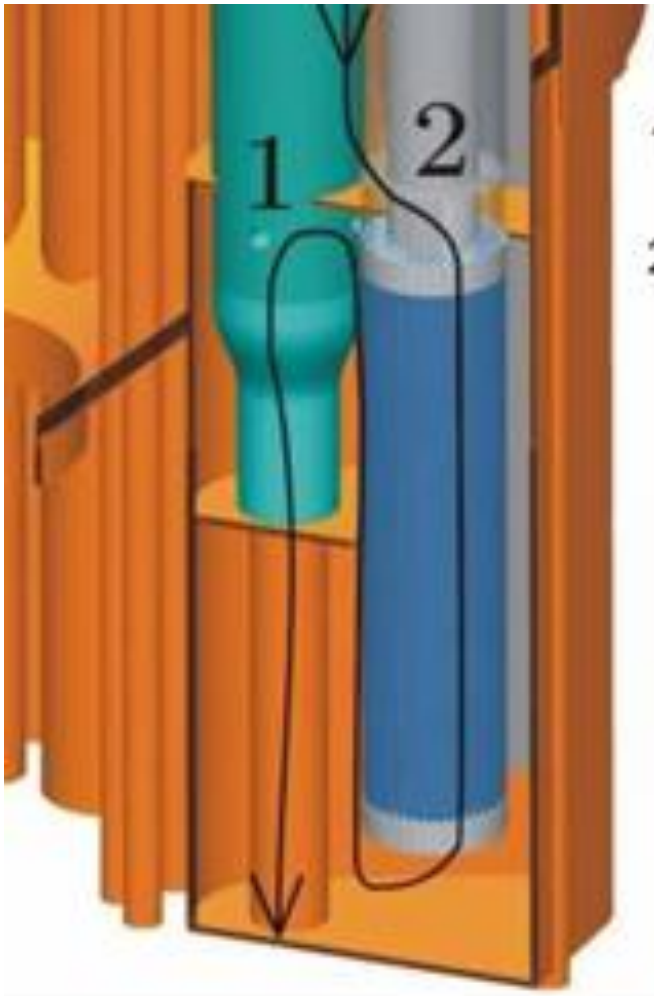
Fuel pin design

ADS-LFR Myrrha



1. inner vessel
2. guard vessel
3. cover
4. diaphragm
5. spallation loop
6. sub-critical core
7. primary pumps
8. primary heat exchangers
9. in-vessel fuel transfer machine
10. in-vessel fuel storage

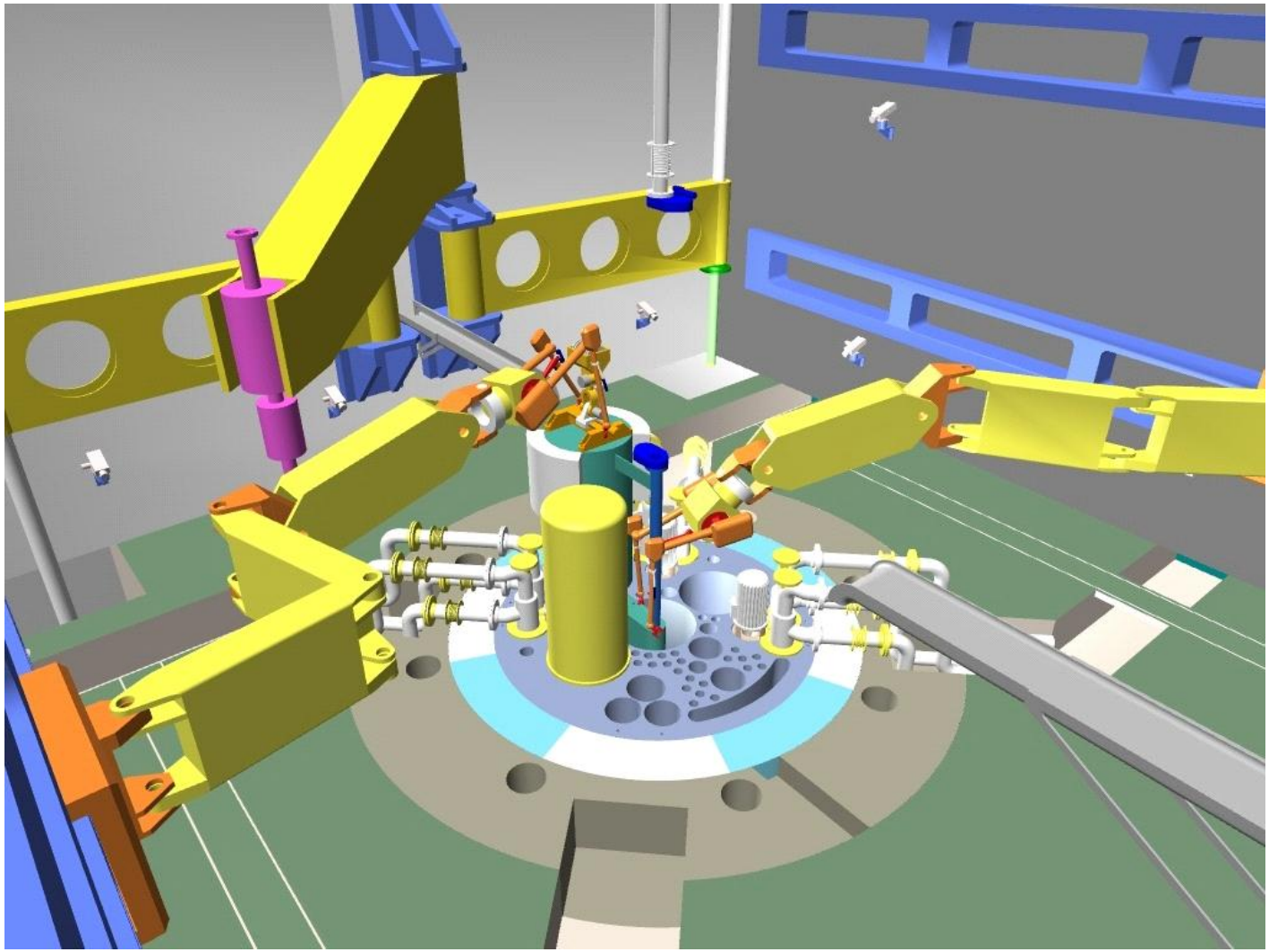
Heat Exchanger



1. Primary pump

2. Primary heat exchanger

The MYRRHA remote handling systems



Building

The present design of this building is quite large (100 m length, 30 m width, 40 m height; this height being subdivided, in the "reference" configuration, into 10 m above ground level and 30 m underground).

The R&D programme covers 5 different fields

Lead-Bismuth technology

The R&D programme on Lead-Bismuth technology is needed to sufficiently understand the behaviour of the heavy liquid metal in reactor conditions such as in MYRRHA.

Fuel qualification

The R&D programme for the qualification of the MOX fast reactor driver fuel for MYRRHA.

Material qualification

Since the development of innovative reactors such as MYRRHA is highly dependent on the material behaviour in reactor conditions, a dedicated material qualification programme is required.

Component qualification

The correct working of large innovative components needs to be assured through a dedicated R&D programme.

Reactor physics

The reactor physics of these innovative fast and sub-critical cores demands the development of new calculation tools and for code validation by integral experiments for which a representative zero power facility is a pre-requisite.

International Cooperation

The design of the accelerator is a European collaboration, started in FP5 (PDS-XADS) and pursued in FP6 (EUROTRANS).

- The main partners are CNRS (France), CEA (France), INFN (Italy), U. Frankfurt (Germany).
- Smaller contributors are Empresarios Agrupados (Spain), ITN (Portugal) and SCK•CEN (Belgium).

Animations

http://myrrha.sckcen.be/en/Media_gallery/MYRRHA_animations

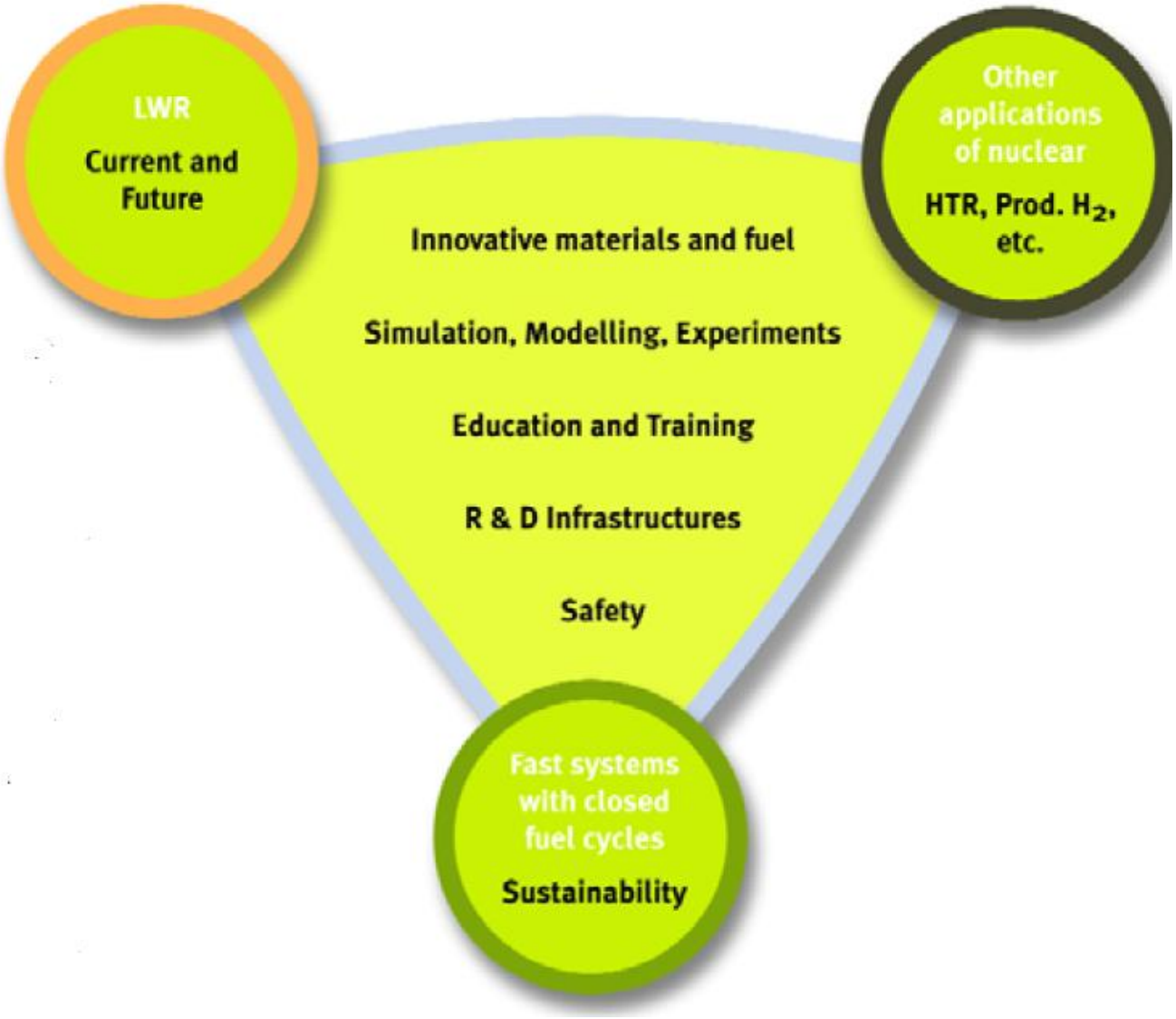
SNETP

The Sustainable Nuclear Energy Technology Platform (SNETP), launched in September 2007, gathers European stakeholders involved in nuclear fission: industry and services, research, academia, safety organisations, NGOs, and associations.

SNETP prepared a vision resting on three technology pillars, as means to achieving the SET-Plan goals:

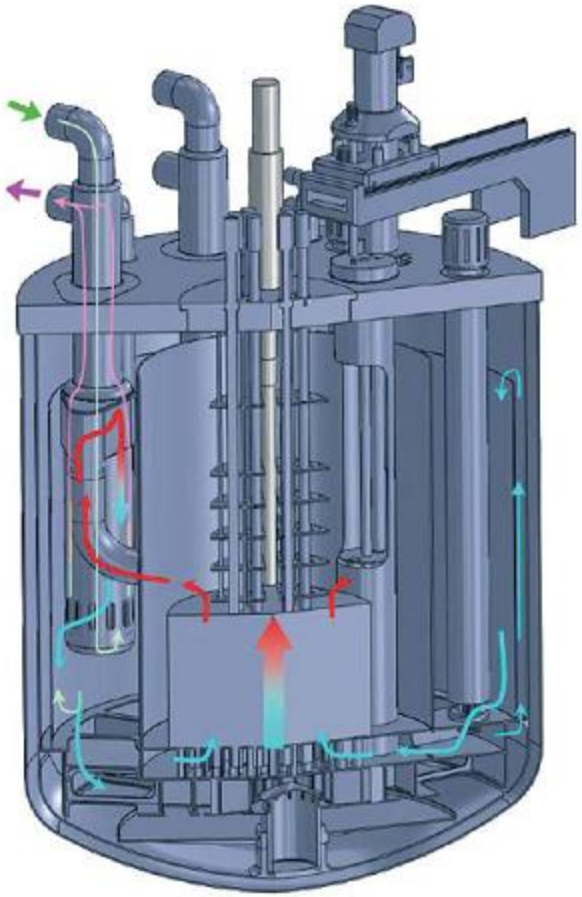
- Safety and competitiveness of today's fission technology,
- Fast neutron reactors with closed fuel cycles for increased sustainability,
- Cogeneration of process heat and power for industrial applications.

The SNETP vision

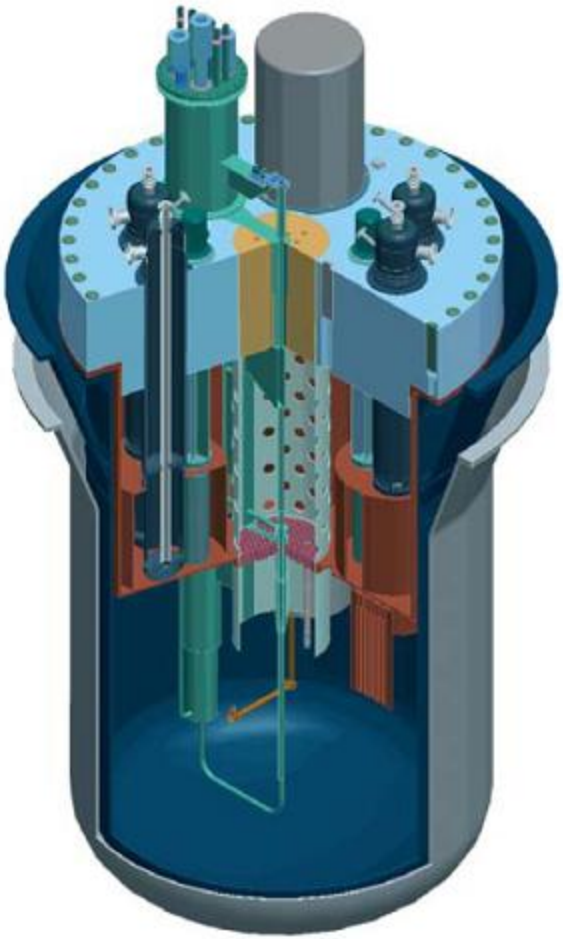


ASTRID(600-1500 MWt), MYRRHA,ALLEGRO(50-70 MWt)

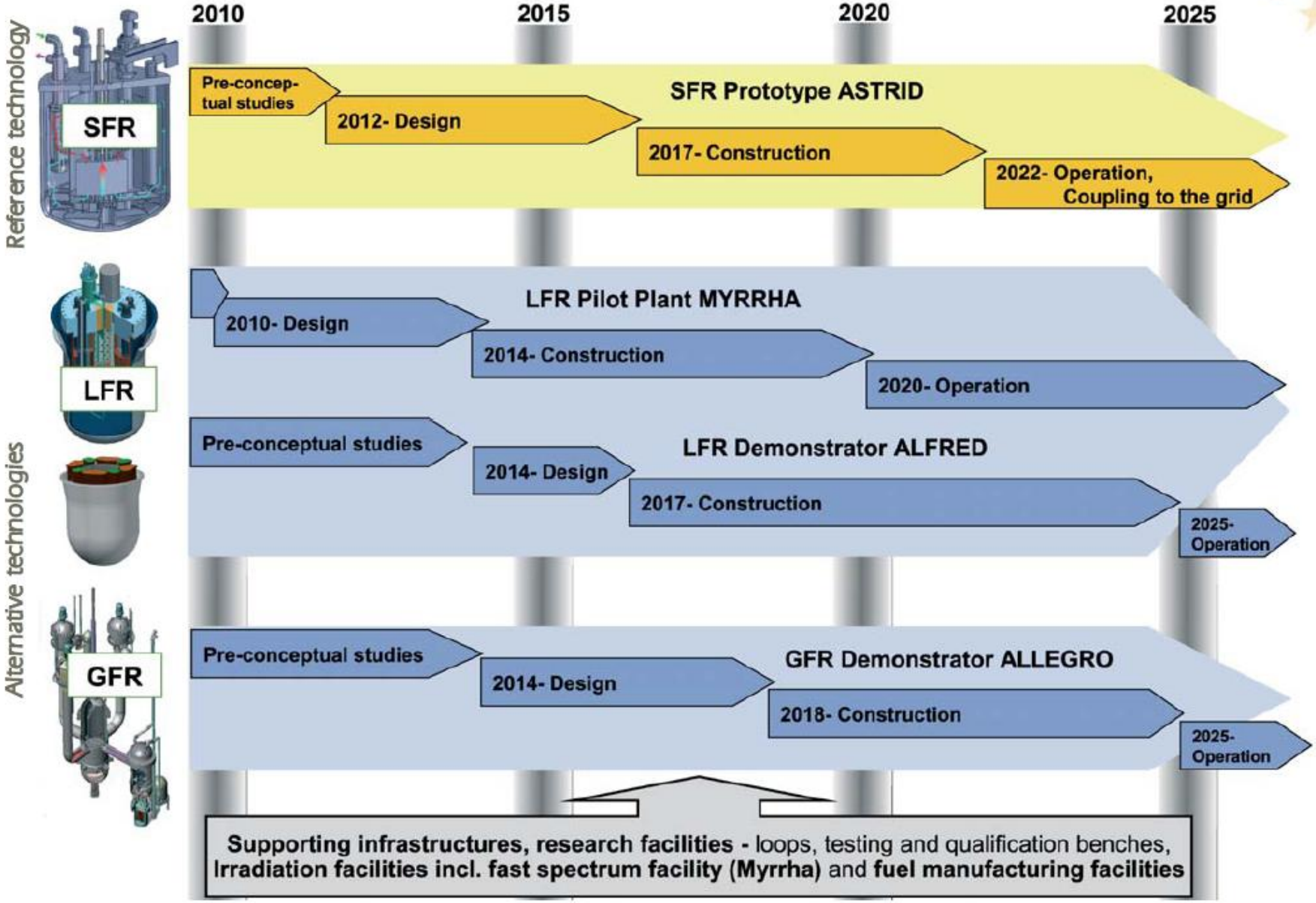
The SFR concept



The LFR/ADS concept



The GFR concept


2010
2015
2020
2025
Reference technology
Alternative technologies
SFR
LFR
GFR
Pre-concep-tual studies
2012- Design
SFR Prototype ASTRID
2017- Construction
2022- Operation, Coupling to the grid
2010- Design
LFR Pilot Plant MYRRHA
2014- Construction
2020- Operation
Pre-conceptual studies
2014- Design
LFR Demonstrator ALFRED
2017- Construction
2025- Operation
Pre-conceptual studies
2014- Design
GFR Demonstrator ALLEGRO
2018- Construction
2025- Operation
Supporting infrastructures, research facilities - loops, testing and qualification benches, Irradiation facilities incl. fast spectrum facility (Myrrha) and fuel manufacturing facilities

New European Nuclear Research Facilities

Priority topics for action

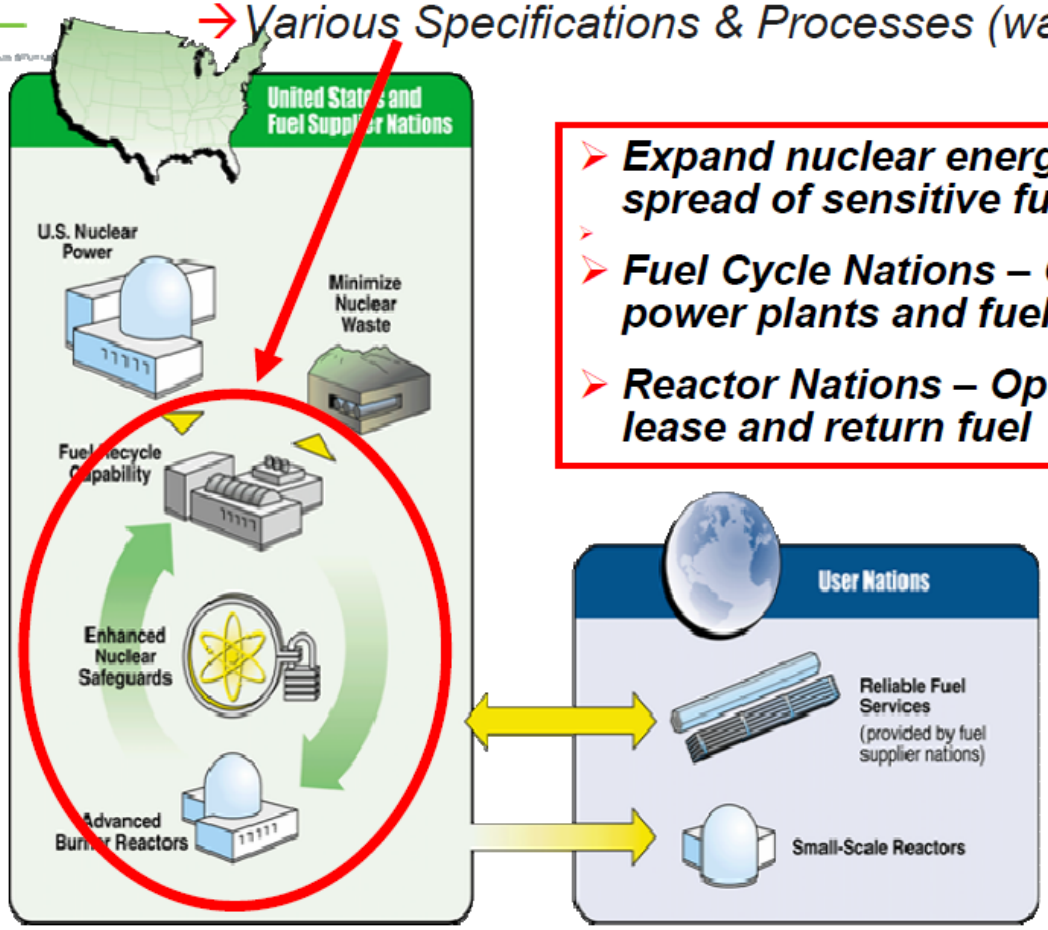
A. Properties of MA Fuel (2010-2012) with extension to 2020

- Fabrication of solid pellet, coated particle (nitride, carbide, oxide, fluoride) based on Th, U, Pu, MA. Determination of phases, melting point, heat capacity, thermal conductivity, and mechanical properties of those pellets or particles and viscosity for molten fuel forms.
- Experimental investigation of fuel interaction with: fission products and helium, coolant and cladding, supported by modelling for extrapolation to off–normal operating conditions.

International Centers for fuel services

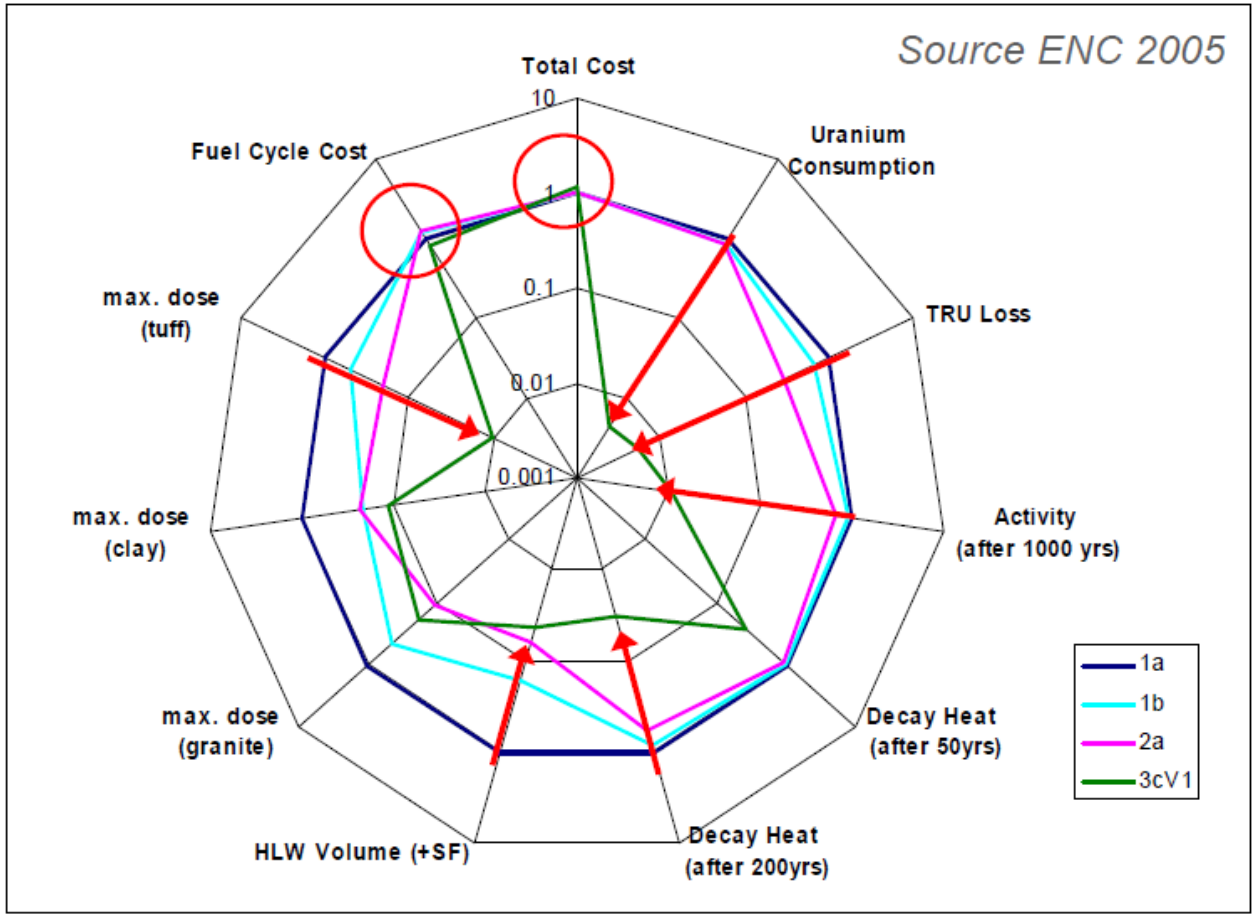
→ *International Standards for non-proliferation & safeguards*

→ *Various Specifications & Processes (waste, technologies…)*



- ***Expand nuclear energy while preventing spread of sensitive fuel cycle technology***
- ***Fuel Cycle Nations – Operate both nuclear power plants and fuel cycle facilities***
- ***Reactor Nations – Operate only reactors, lease and return fuel***

Generation IV and Partitioning & Transmutation impacts →



1a: ***Once-through cycle as reference***

1b: ***full LWR fleet, Pu re-used once***

2a: ***full LWR fleet, multiple re-use of Pu***

3cV1: ***full fast reactor fleet and fully closed fuel cycle (Gen IV)***

PWR