



In2p3

Future scenarios for fission based reactors?

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Summary and focus on nuclear power

3rd part

Simplified construction of an energetic world in 2050 Motivations, hypothesis, approach and results

S. Bouneau



ANNALES DE PHYSIQUE

Construction d'un monde énergétique en 2050

S. Bouneau, S. David, J.-M. Loiseaux, O. Méplan

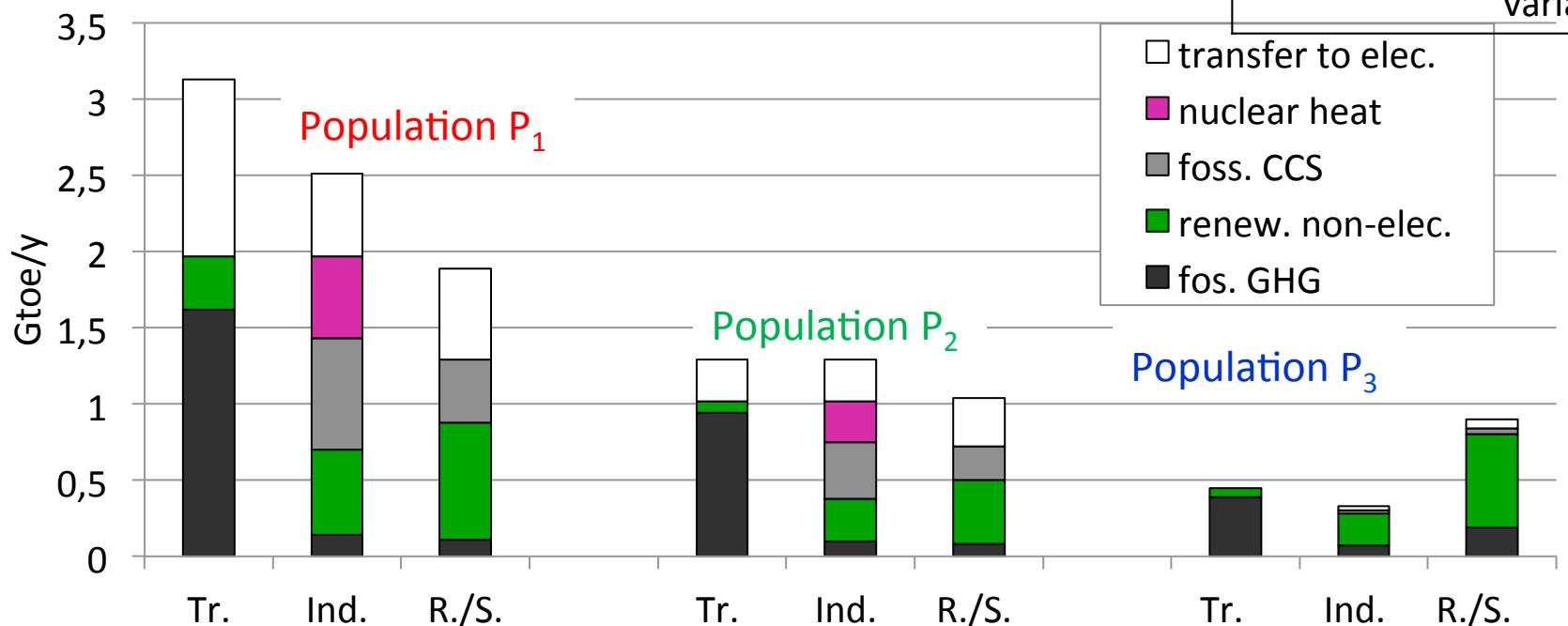


method to construct energy mix in 2050

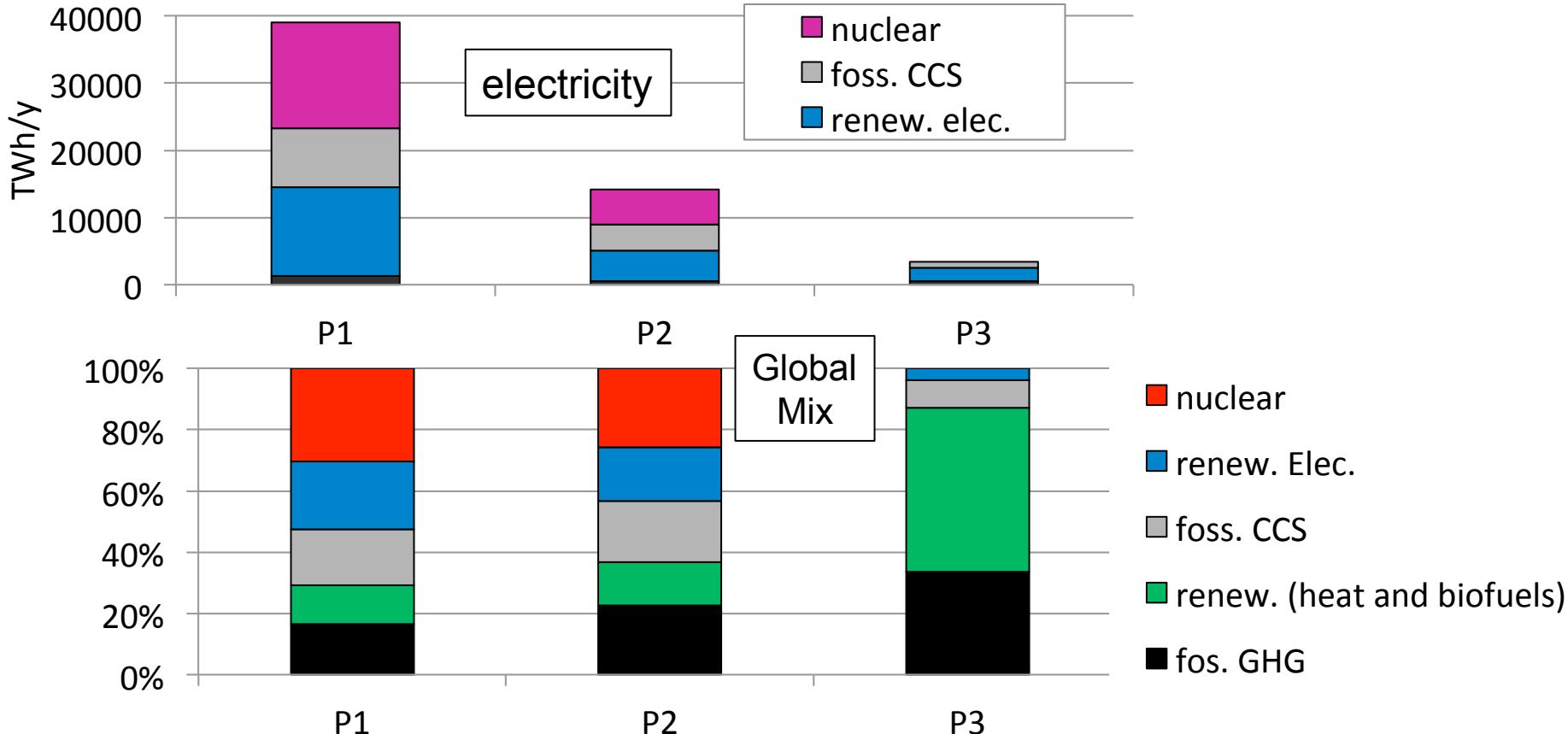
- Needs are known for each sector and for each population
- Fossile fuel essentially for transport, biomass essentially for rurals
- Heat production with renewable energy and fossil fuels with CCS
- Potential of renew. sources and CCS fixed to optimistic values
- Missing energy tranferred to electricity
- Electricity mix filled with renewable, CCS
- **nuclear energy is used as final adjusting variable**

Fixed potential

Sources	Gtoe/y
Renew. elec	3,5
Renew. heat	3,5
Biofuels	0,5
CCS	3,7
Nuclear	Free variable



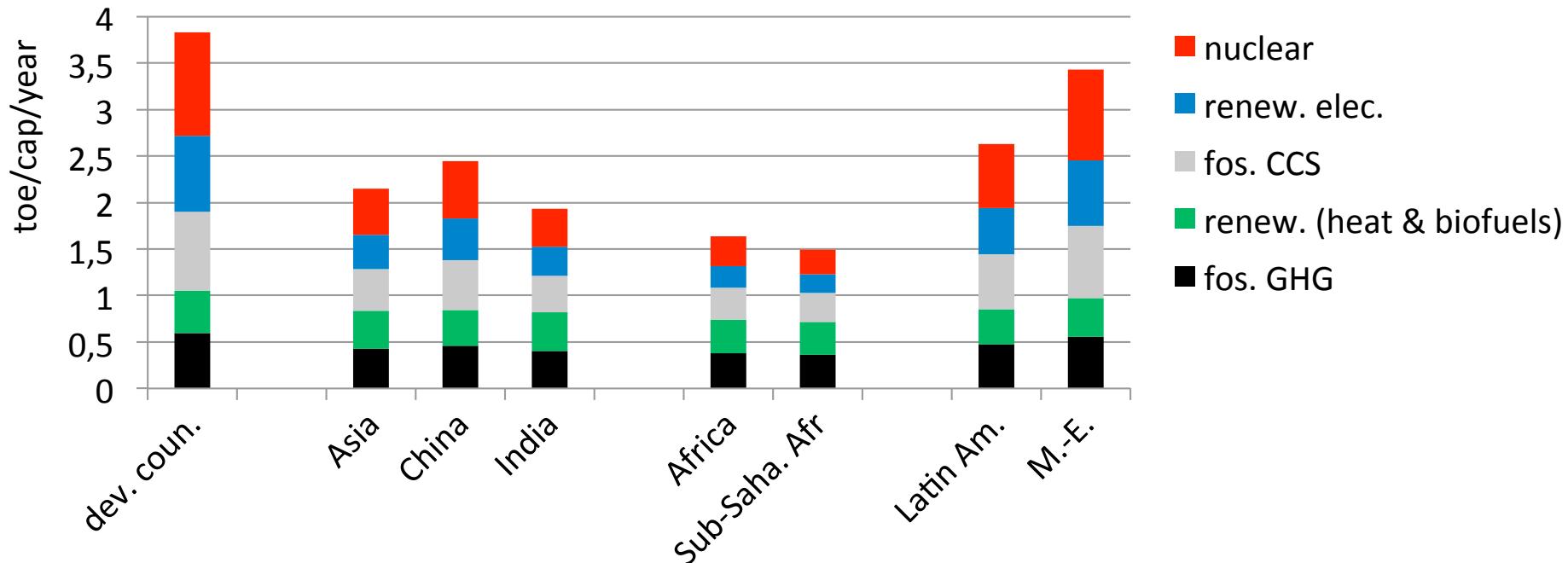
□ electric mix after tranfer of missing heat and transport needs



Nuclear energy :

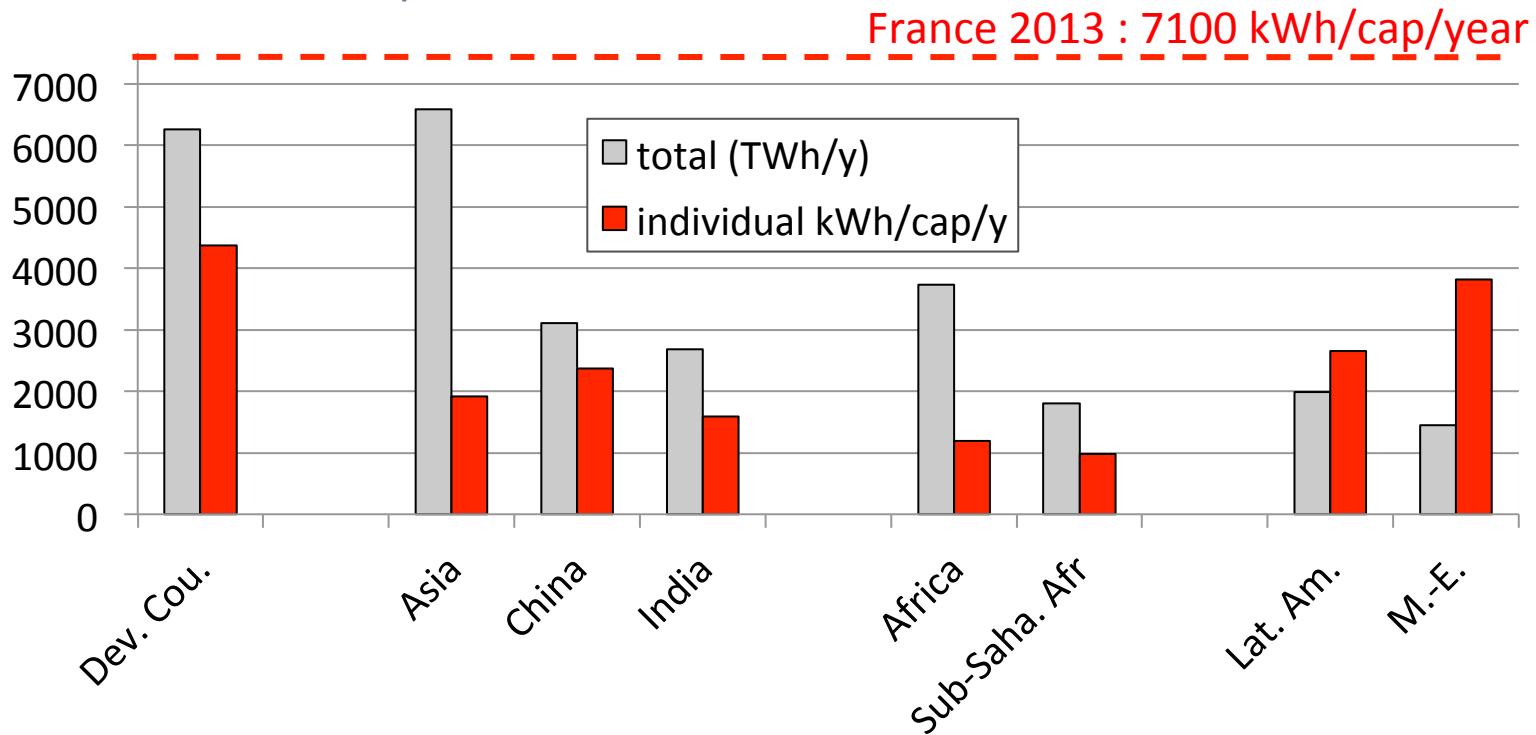
Only use for P1 and P2 population, urban population, centralized production, and high technical level

□ energy mix of different geographical regions



- the use of fossil fuels are leveled around 0,5 toe/cap/year except in poorest regions
⇒ CO₂ emissions ~ 1,6 tCO₂/cap/year
- CCS technology is developed in all regions
- homogenous distribution of renewable energy sources for heat and transport needs
- part of CCS technology, renewable energy sources for electricity generation and nuclear energy increase with the fraction of population P₁

focus on nuclear power



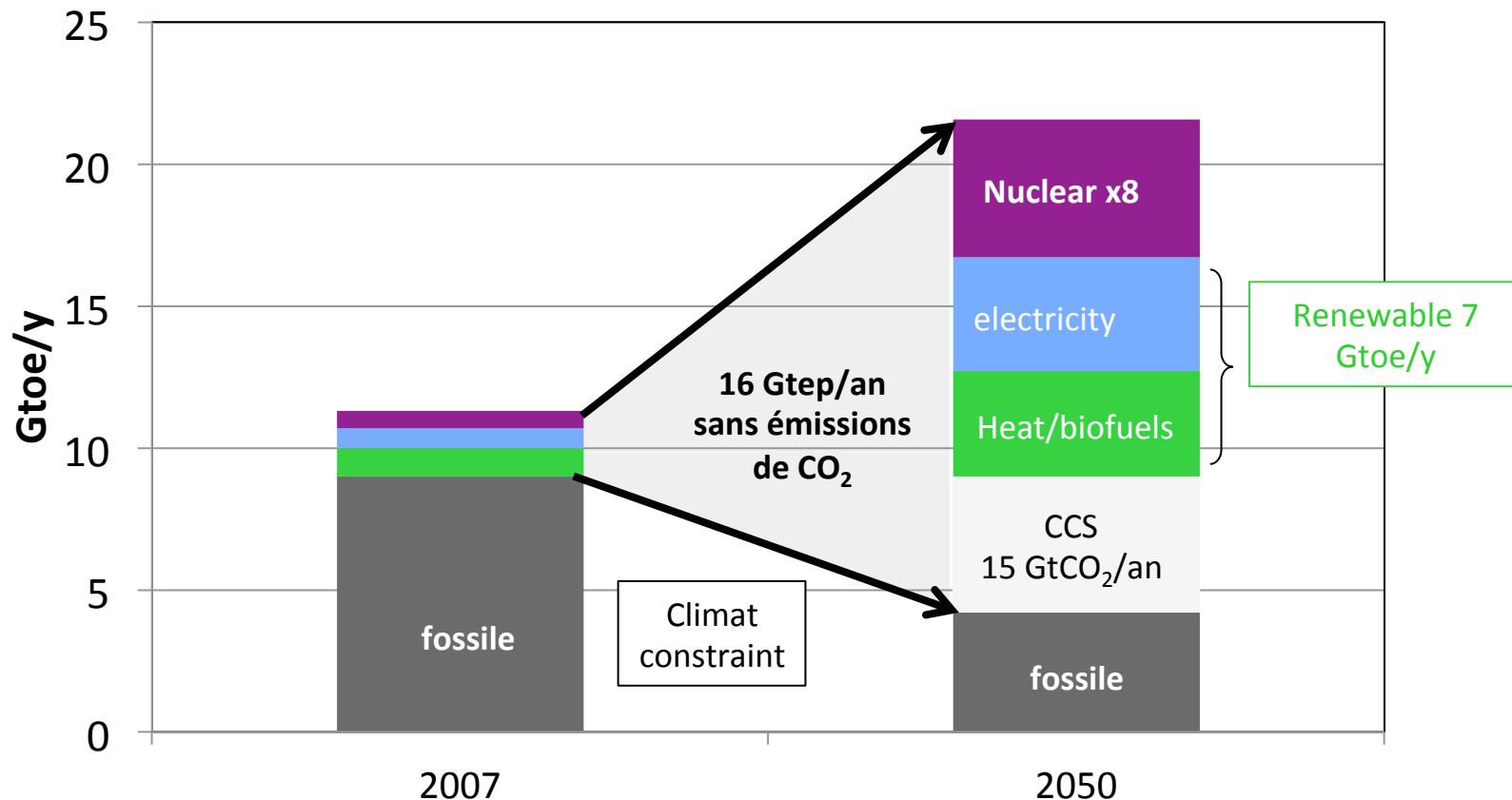
- nuclear heat ~ 0,8 Gtoe/year
- nuclear electricity ~ 21 000 TWh (~ 37% of the total electricity consumption)
- ⇒ X 9 / today (5,4 Gtoe/year)
- mainly developed in Asia
- even if nuclear power is multiplied by ~10, the world remains much less nuclearized than France today

Scenario nuclear x~10 : taken as a « maximizing » hypothesis to test the ability of the different reactor technologies to provide a long term nuclear power

A possible scenario for 2050

Simplified construction of a energetic world in 2050 (cf Sandra Bouneau)

Possible scenario with a significant increase of the nuclear energy production : 5 Gtoe/y, ie 8 times more than today



Energy context

Fossile fuels dominate the world energy production

source	Gtoe/y (2008)
Fossile fuels	9.1
biomass	1.2
hydropower	0.7
Nuclear	0.6
New renewable	0.02
Total	11,62

Nuclear power dedicated to electricity generation

< 6% of the world primary energy production

15% of the world electricity generation, decreasing

35% de l'électricité en Europe

438 reactors, in 31 countries

installed power 375 GW

Production 2009 = 2560 TWh = 292 GW effective, load factor ~80%

Future scenarios for fission based reactors?

Reminders about the basics principle of present nuclear reactors

Uranium consumption

Breeding : basic principles, technologies, transition Gen3/Gen4

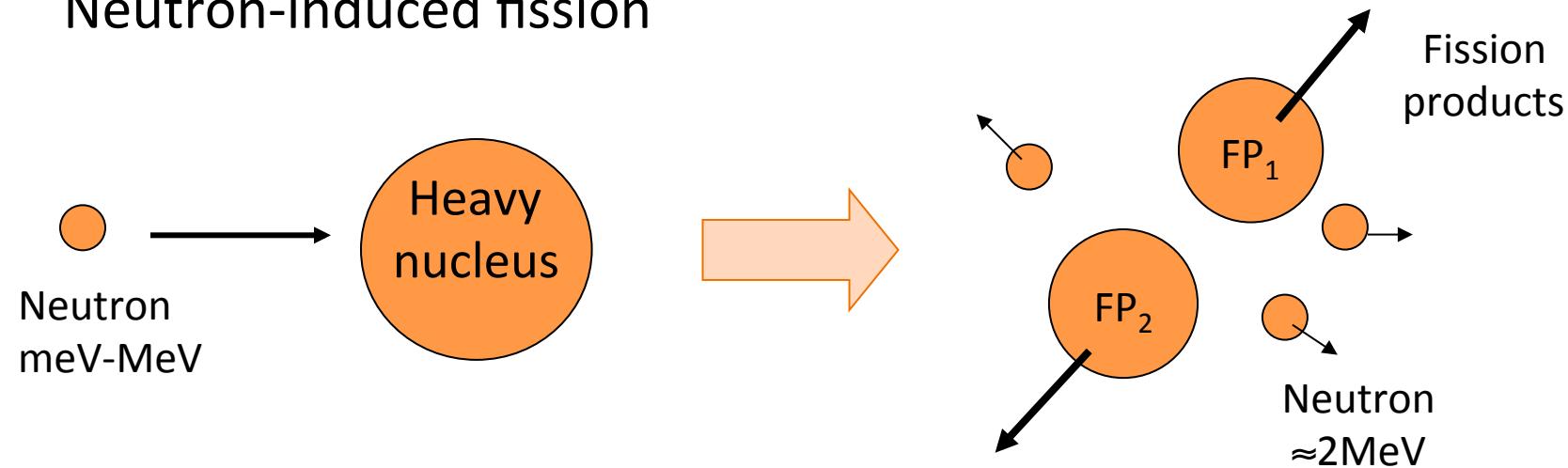
The question of Plutonium

And :

Waste transmutation

A few words on thorium cycle

Neutron-induced fission



$$M > M_1 + M_2 + 3M_n$$

$$\Delta M = 200 \text{ MeV}/c^2$$



1 reactor, power 1GWe, thermal efficiency 33%

1 ton of heavy nuclei fissioned per year

European electricity consumption = 1000 W
~ 1g/y per capita

reminders

Only the heavy nuclei (actinides, Z>90) can fission
 Only the « even-neutron » nuclei can fission « easily »

3 natural actinides ^{238}U , ^{235}U et ^{232}Th

Z	0	131	132	133	134	135	136	137	138	139	140	141	142	143	144	145	146	147	148	
94																				
93																				
92																				
91	Pa 222	Pa 223	Pa 224	Pa 225	Pa 226	Pa 227	Pa 228	Pa 229	Pa 230	Pa 231	Pa 232	Pa 233	Pa 234	Pa 235	Pa 236	Pa 237	Pa 238	Pa 239	Pa 240	
90	Th 221	Th 222	Th 223	Th 224	Th 225	Th 226	Th 227	Th 228	Th 229	Th 230	Th 231	Th 232	Th 233	Th 234	Th 235	Th 236	Th 237	Th 238	Th 239	
	α 5.7ms	α ~6.5ms	α ~0.95s	α 1.8s	α 1.8m	α 38.3m	e capture 26h	e capture 1.4d	e capture 17.4d	α 72y	α 3.25×10^4 y	β^- 1.32d	β^- 27.0d	β^- 1.40×10^{10} y	β^- 6.67h	β^- 24.1m	β^- 9.1m	β^- 8.7m	β^- 2.3m	β^- 14.1h
	α 1.7ms	α 2.9ms	α 0.66s	α 1.04s	α 8m	α 31m	α 18.72d	α 1.913y	α 7340y	α 7.7×10^4 y	β^- 25.52h	100%	α 1.40×10^{10} y	β^- 22.2m	β^- 24.1d	β^- 6.9m				

Only ^{235}U is a fissile nucleus (even number of neutrons but represent only 0.72% of natural uranium)

Why neutrons have to be slowed down ?

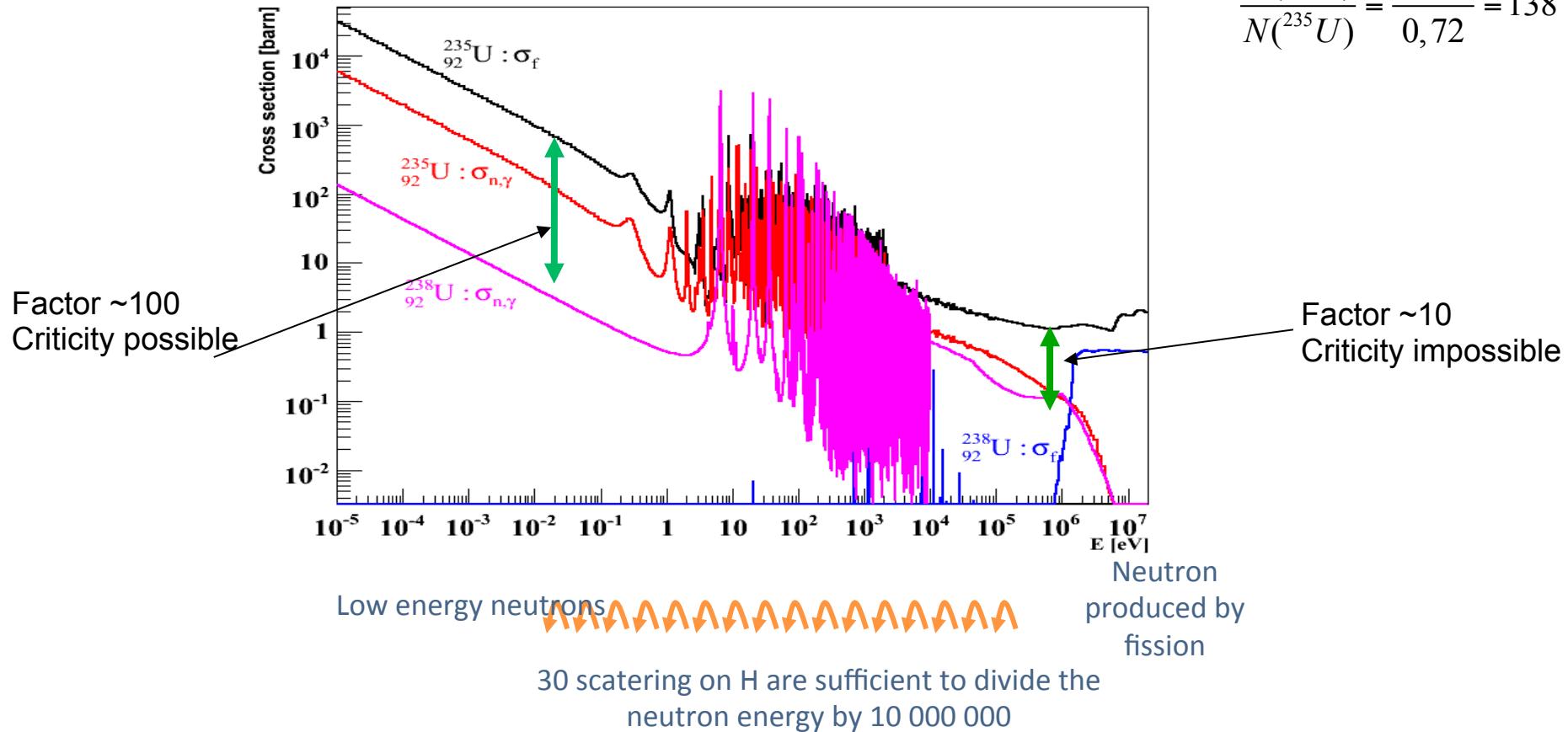
Using ^{235}U in natural uranium requires to slow down the neutrons

Neutrons are produced by the fissions at $\sim 2\text{MeV}$

At low energy, fission cross section of ^{235}U is so high that it compensates the absorption of ^{238}U

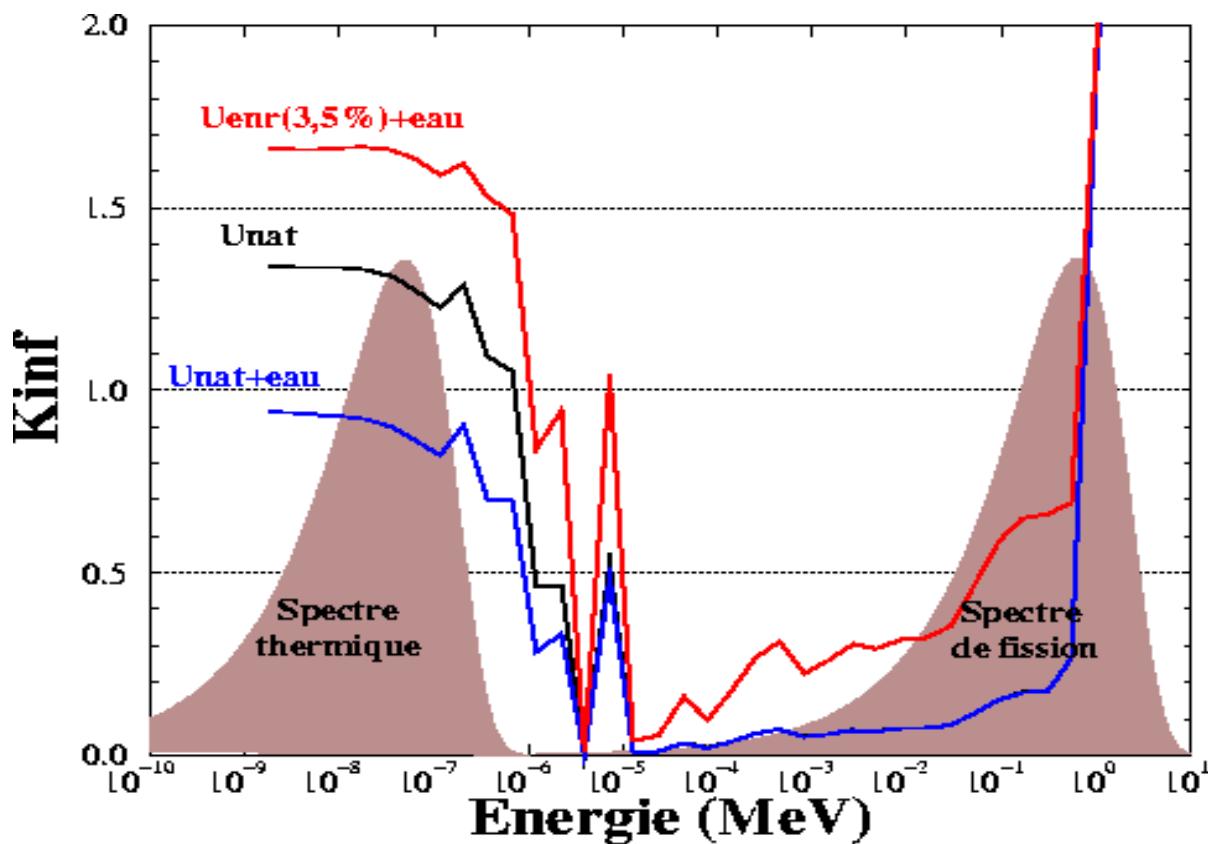
Chain reaction can be maintained

$$\frac{N(^{238}\text{U})}{N(^{235}\text{U})} = \frac{99,28}{0,72} = 138$$



Multiplication factor

$$k = \frac{\text{Produced neutrons}}{\text{Absorbed neutrons}} = \frac{\nu N_{\text{U-235}} \sigma^{\text{fission}} \phi}{N_{\text{U-238}} \sigma^{\text{capture}} \phi + N_{\text{U-235}} (\sigma^{\text{fission}} + \sigma^{\text{capture}}) \phi}$$

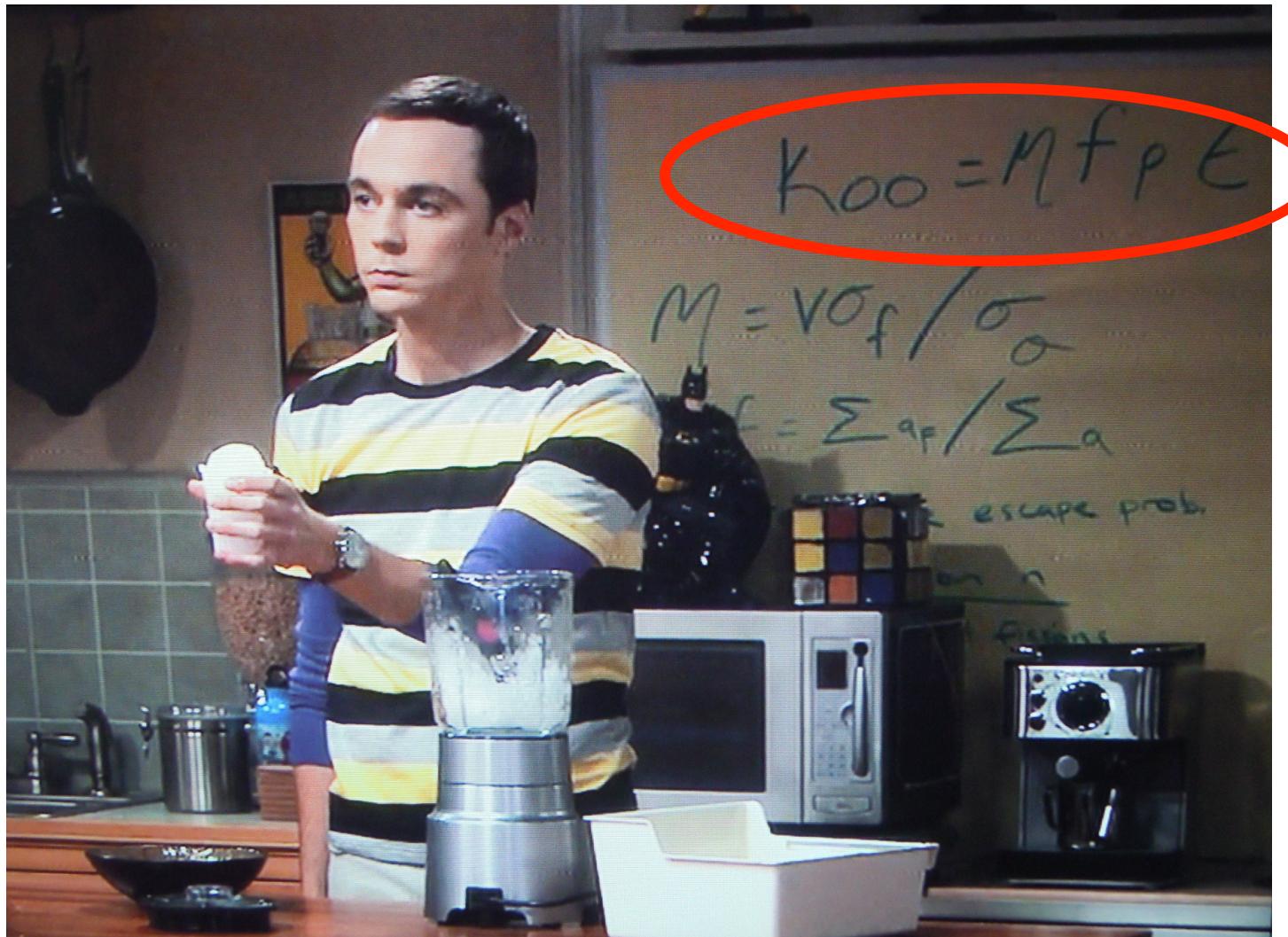


Unat + Heavy Water
(CANDU)

Uenr + Light Water
(PWR, BWR)

Other references for criticality calculation

Multiplication factor : 4 factor formula



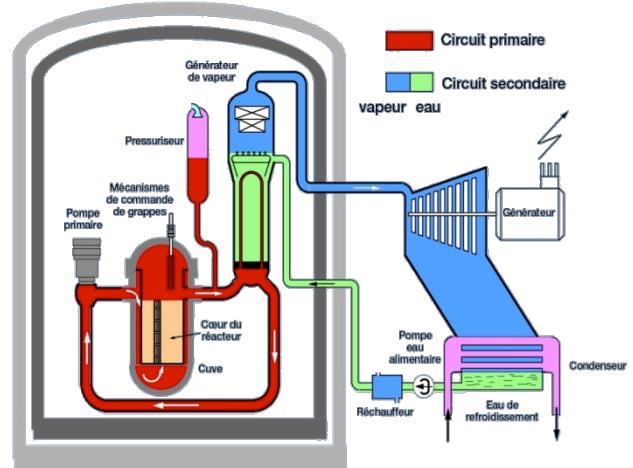
Basic principle of standard nuclear reactors

Production of thermal energy by fission in the reactor core

Water : moderator and coolant

fuel : enriched Uranium (3,5% ^{235}U)

PWR, Gen2



Temperatures

Fuel $\sim 1500^\circ\text{C}$

Primary circuit

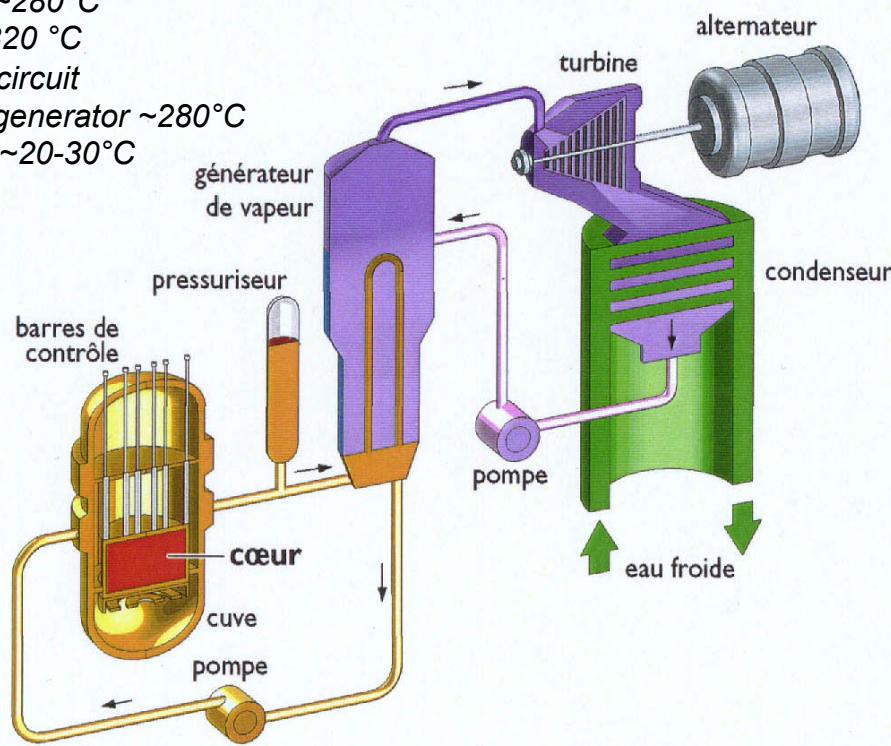
Core entry $\sim 280^\circ\text{C}$

Core out $\sim 320^\circ\text{C}$

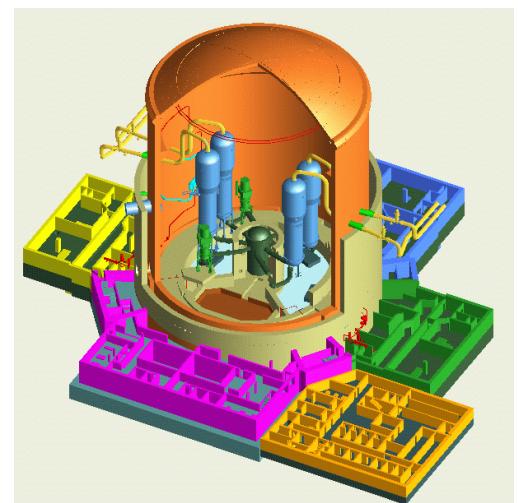
Secondary circuit

Out vapor generator $\sim 280^\circ\text{C}$

Condenser $\sim 20-30^\circ\text{C}$

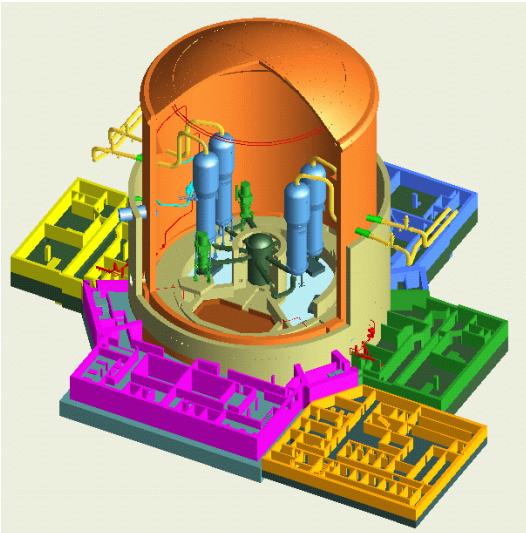


EPR, Gen3

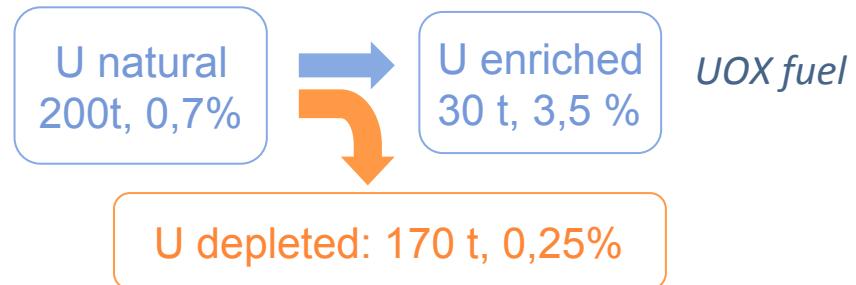


Present water technologies

Technology PWR (EPR, AP100, VVER, ...)



Uranium consumption
Fission $200 \text{ MeV} = 1 \text{ t / GWe.y}$
This requires more natural uranium

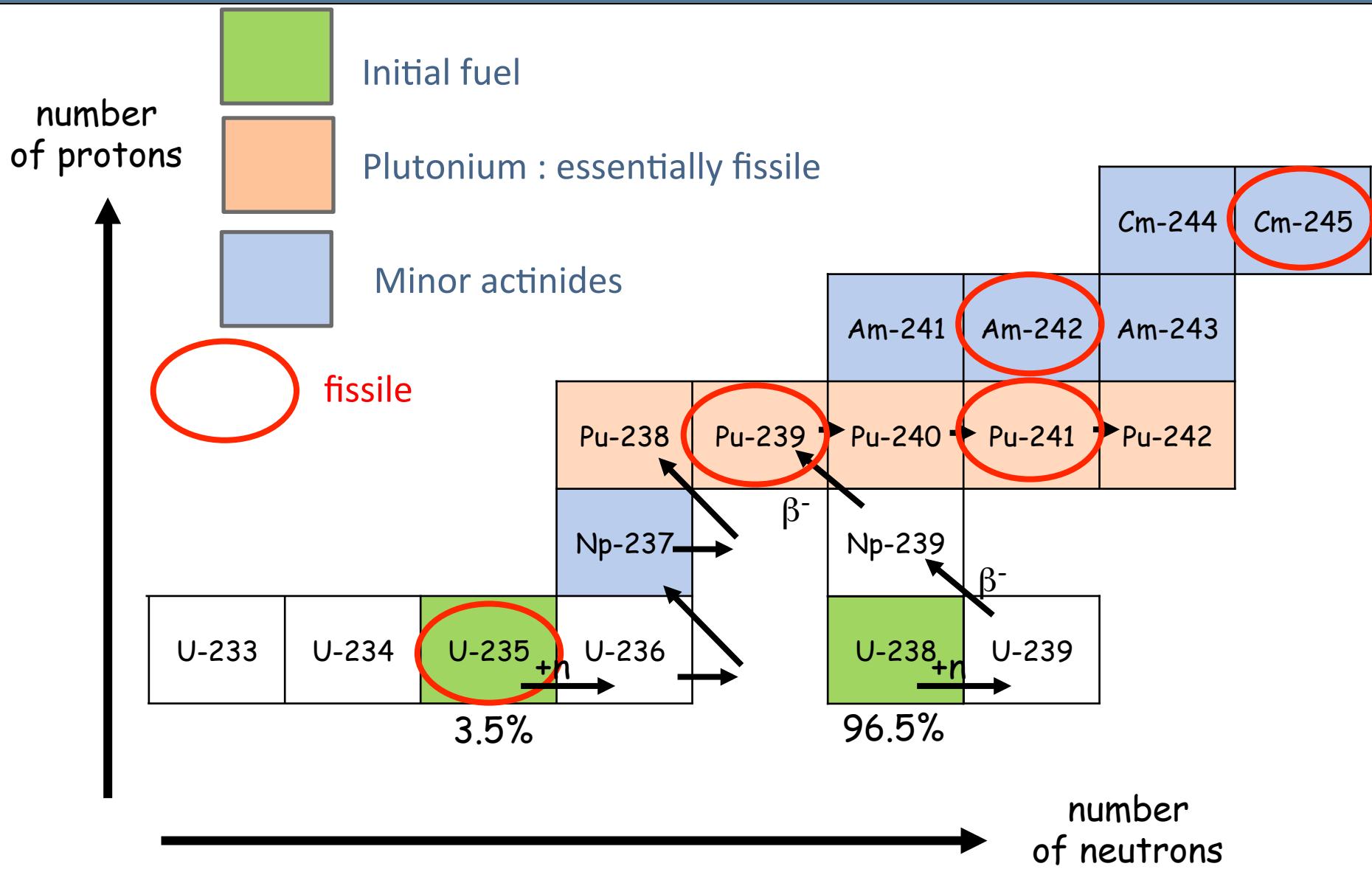


UOX spent fuel

USA : waste

France : reprocessing

U : still 1% of ^{235}U ($1200 \rightarrow 350 \text{ kg}$)
Pu : ~ 1%, 330 kg, fissile material (capture on ^{238}U)
Minor actinides : Am, Cm, Np, 25 kg
Fission products : 1000 kg and 100 kg are radioactive



French reprocessing strategy

UOX reprocessing
Simplified scheme

UOX spent fuel

Uranium

Plutonium

Fission products
Minor actinides

Structures
(clading, ...)

Recycling and
re-enrichment
fuel URE :30%
stored : 70%

recycled
comb. MOX
 U_{dep} / Pu

*MOX spent fuel are
not reprocessed*

Vitrification
Long lived,
High level waste
HA-VL

Compaction
Mid level waste

To be stored in a geological disposal

Maximal potential for existing technologies (no innovation)

Plutonium:

Mono-recycling in standard MOX fuels : 10% of the power of the fleet

Uranium:

Mono recycling and re-enrichement : 10% of the power of the fleet

→Uranium savings ~20%
 $200 \text{ t/(GWe.y)} \rightarrow 160 \text{ t/(GWe.y)}$

Reducing 235U in depleted uranium (enrichment process) : 20% of U savings

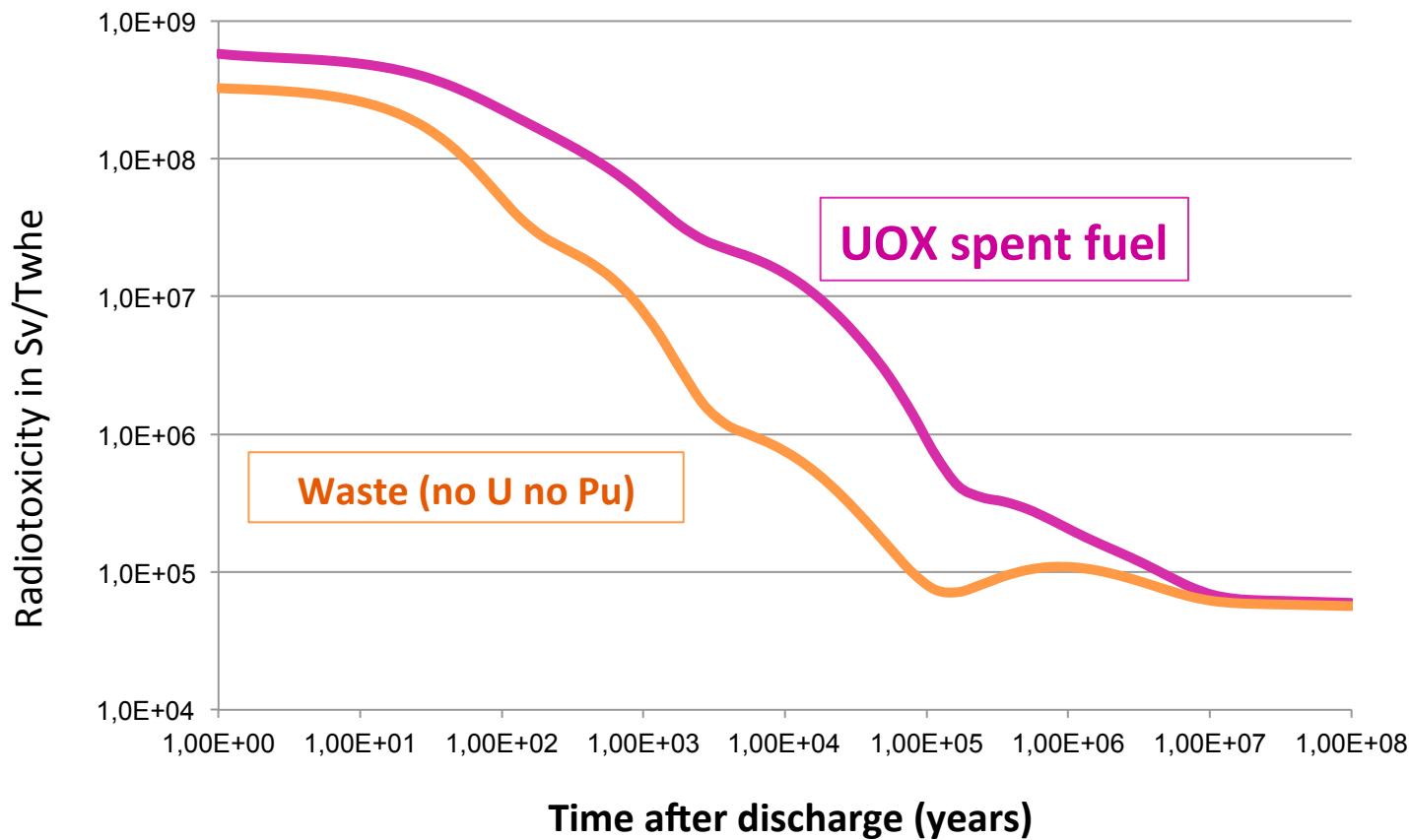
Optimizing the thermal efficiency of the thermodynamic cycle

$160 \text{ t/(GWe.y)} \rightarrow 130 \text{ t/(GWe.y)}$

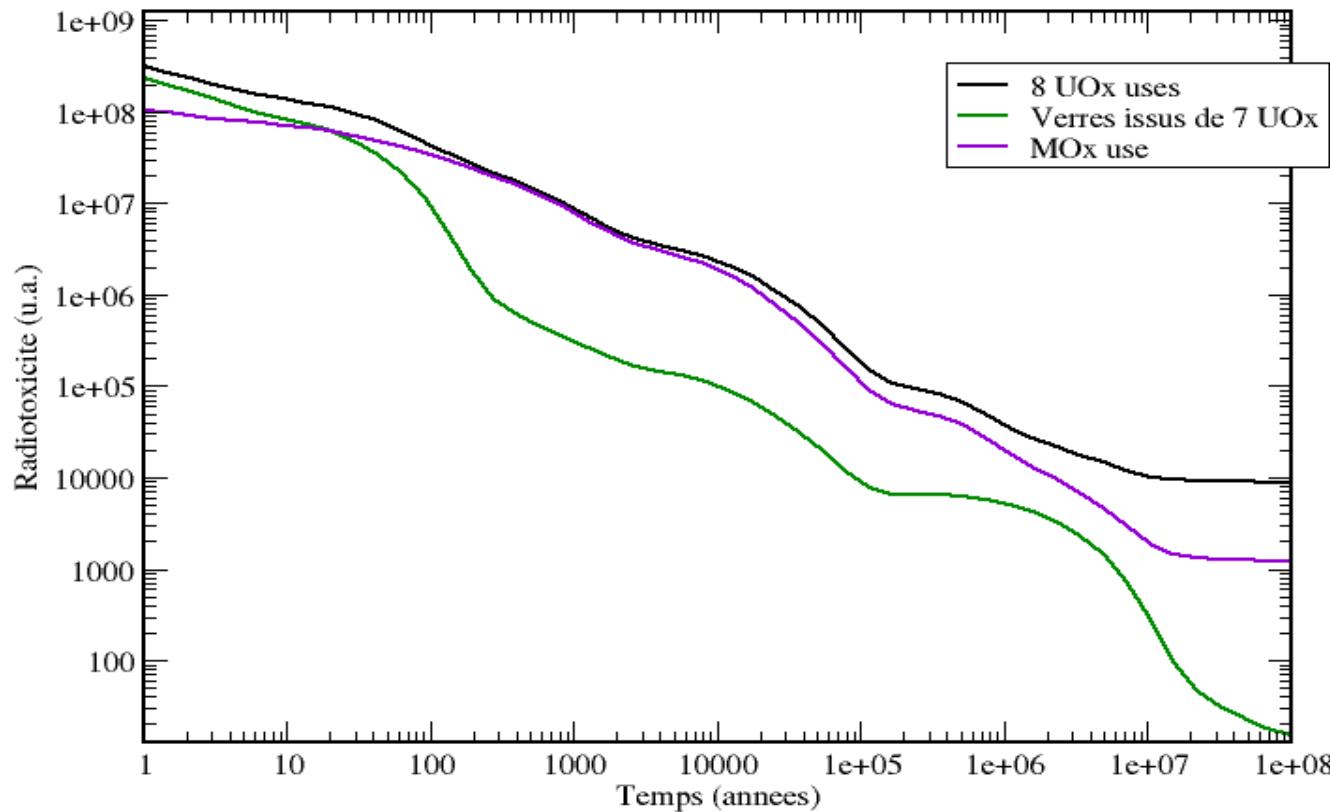
It will be difficult to go beyond this limit with standard technologies

Radiotoxicity of spent fuel

Pu dominates radiotoxicity of the spent fuel



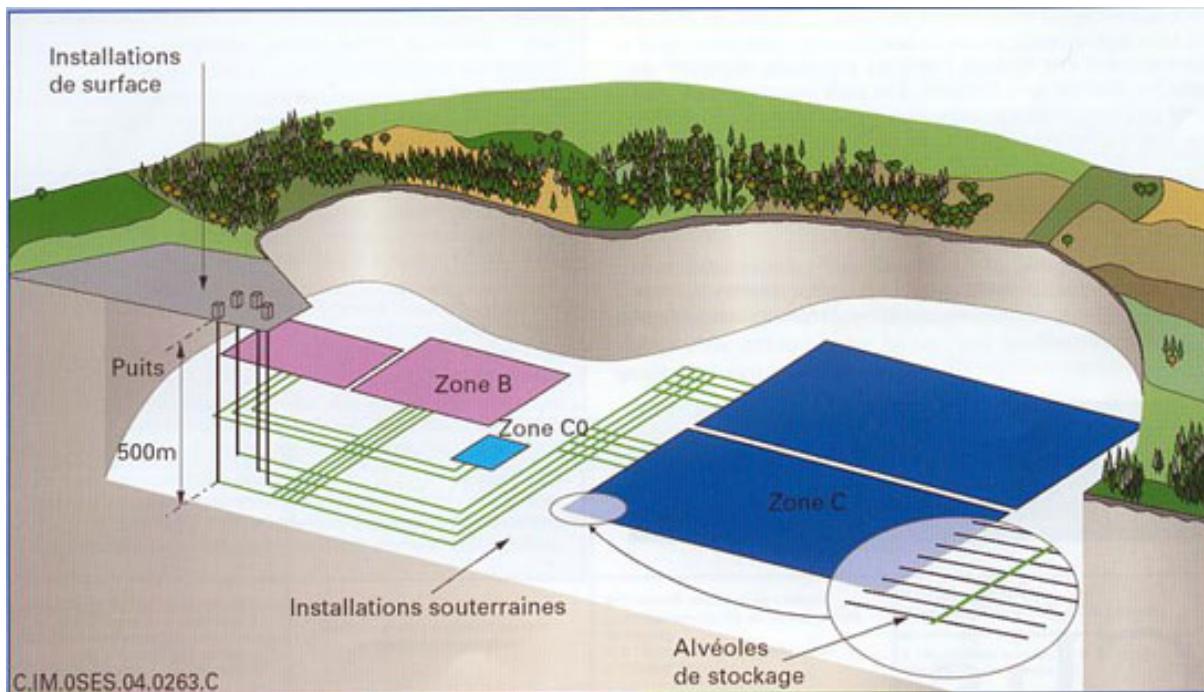
Globally, once through cycle does not change the total radiotoxicity



Ultimate waste : storage

- vitrification is considered as irreversible
- french option : storage in clay, 500m in the ground
- The idea of the storage is to be irreversible
- Surface 300 ha pour les verres HAVL (évacuation de la chaleur résiduelle)
- Reference option for France (loi de 2006), **decision 2013 ?**
- In France, only glasses are concerned, not the MOX spent fuel

Site de Bure



1 ha on the ground (in La Hague)
↓

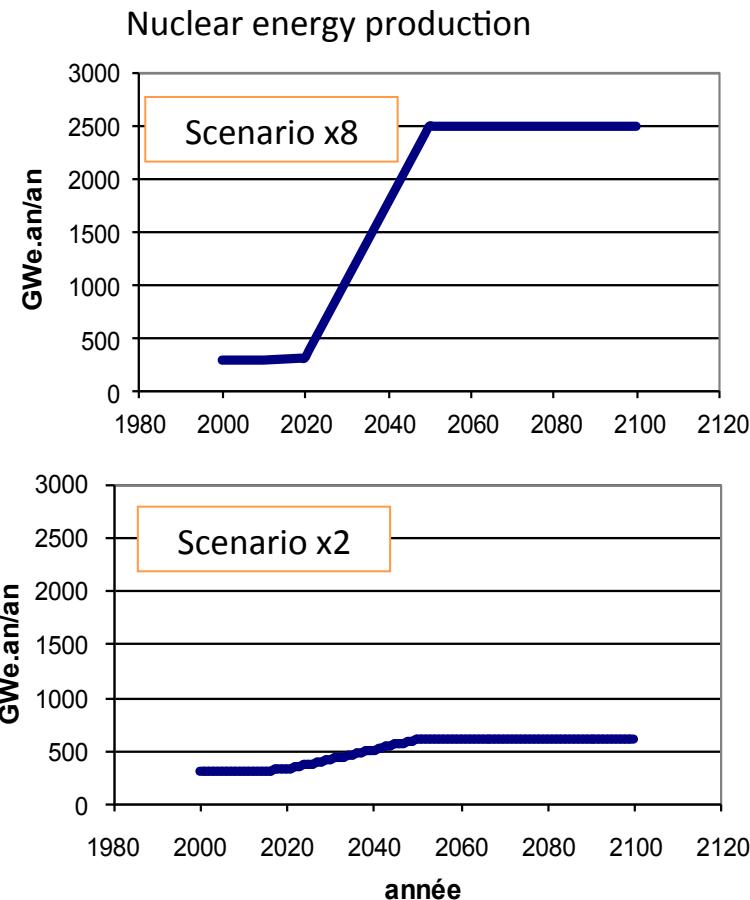
>300 ha in geological disposal
in order to limit the temperature increase

Perhaps a paradox...

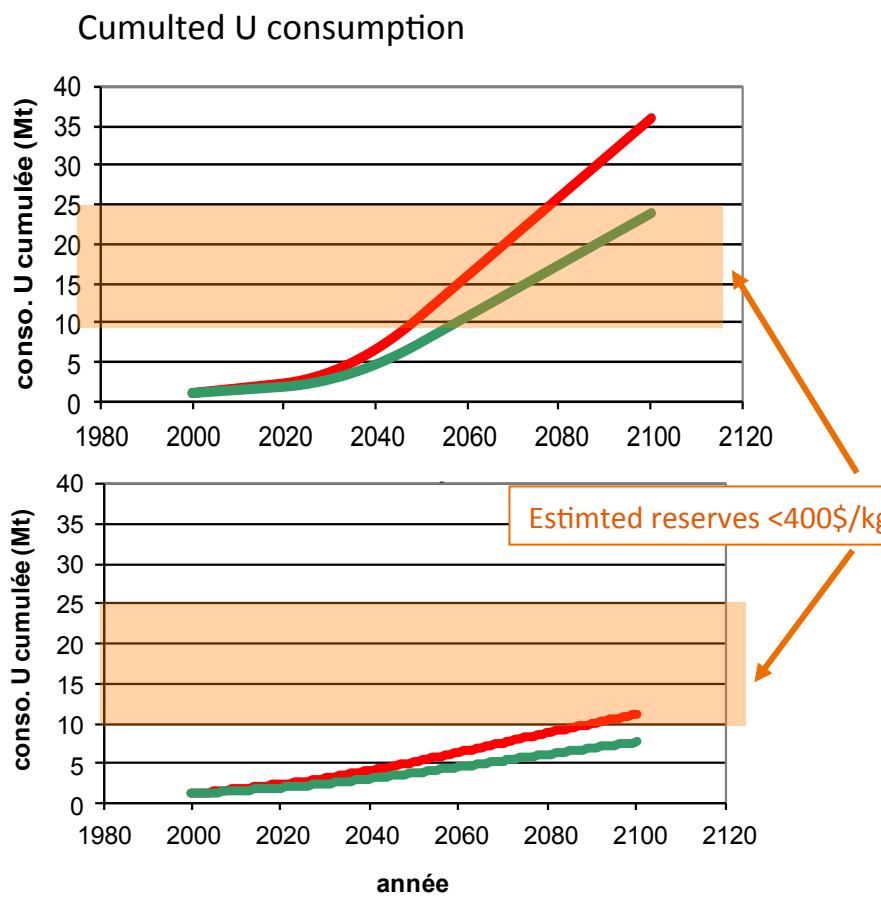
	Trust into society	
	YES	NO
<u>Favourable to nuclear power</u>		
- Favourable to irreversible storage		✓
- Favourable to nuclear deployment	✓	
<u>Anti-nuclear</u>		
- Favourable to reversible disposal	✓	
- Do not trust the society enough to manage the nuclear reactors and insure safety		✓

Production potential of standard reactors

Scénario 100% Gen2 + Gen3



— 200t/(GWe.y)
— 130t/(GWe.y)
↓ Tails ^{235}U dans U_{app} , recycling U, Pu



Nuclear for the future

Present technologies

Based on ^{235}U = 0,7% of U nat

Only fissile nucleus in nature

Consumption for 1 GWe.y

fissioned 1 t

U enriched 4% 27t

U natural 200 t

Possibility to reduce to 130t/GWe.an

reduce ^{235}U in U depleted

recycling U

recycling Pu

No technology innovation

Breeder reactors



Fertile
nucleus

Fissile
nucleus

Pu consuled is bred : Mass Pu = cte

Only ^{238}U is consumed : 1 t / (GWe.y)

French depleted U in 2010 : 300000 t
→ 5000 y of electricity generation

Mass of Pu to provide to start the reactor ~20 t / GWe

French case : 1100 t of Pu needed for 60 GWe

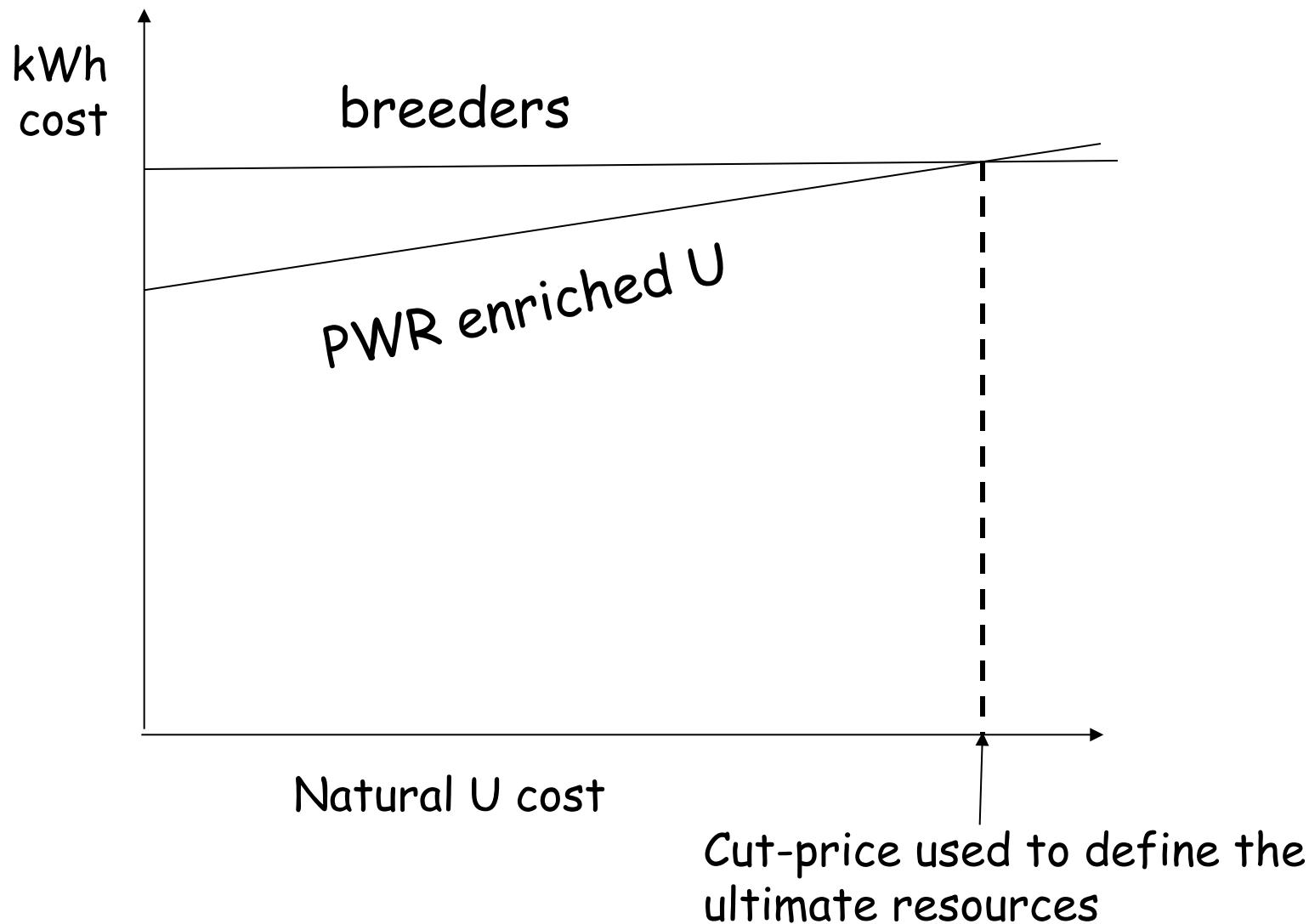
Pu inventory available in 2010 ~ 300t

Technologie plus complexe (donc plus chère)
Réacteurs à neutrons rapides

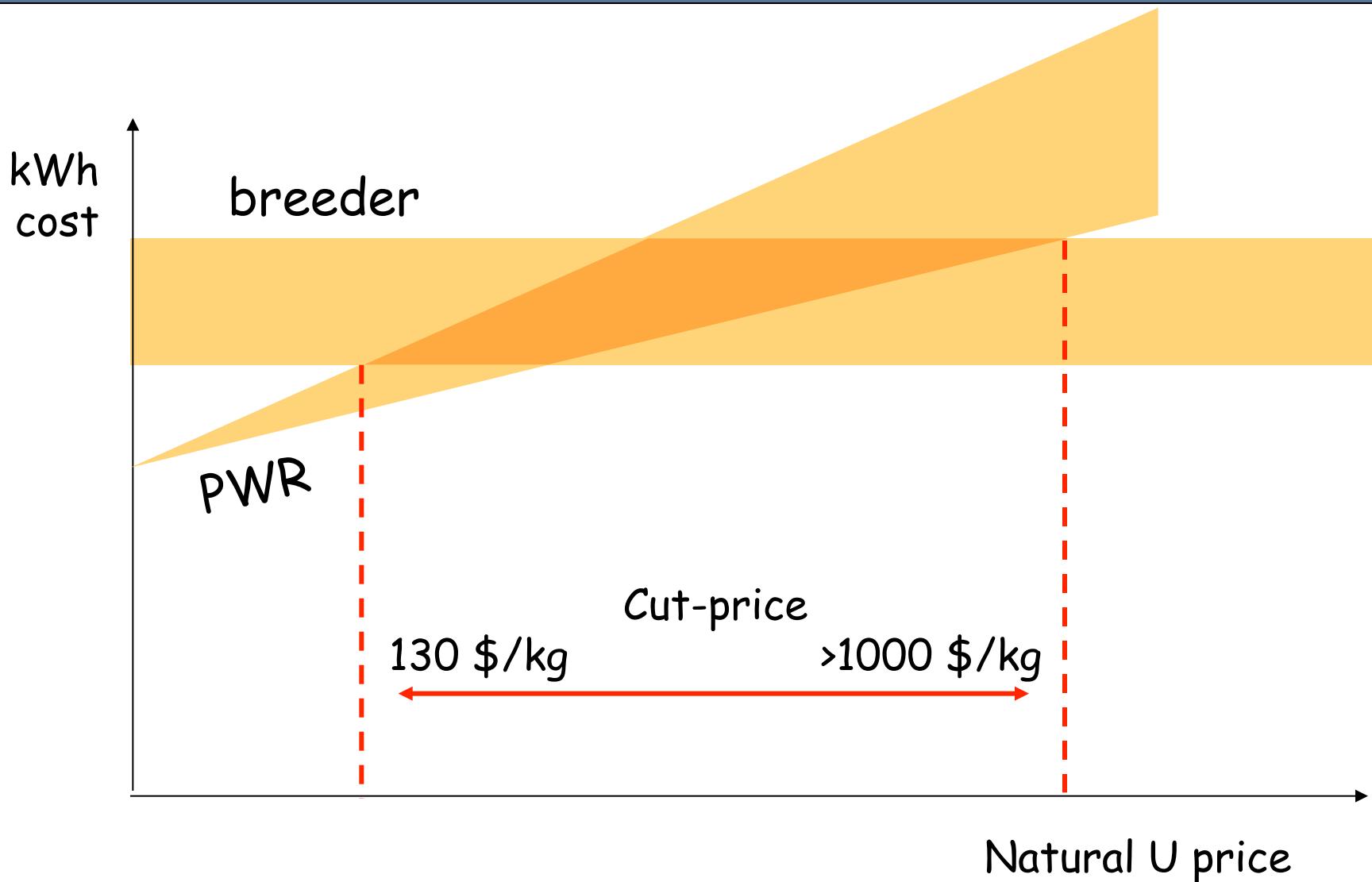
Thorium cycle possible



Economic competition gen3/gen4 « naive scheme »



With uncertainties...



Uranium resources and nuclear deployment

World nuclear production	285 Gwe (full power equivalent)
Natural U consumption	60000 t/y
U reserves (proved and speculative)	10-23 millions of tons
Production	
- at present rate (and present use)	200 – 400 years
- scenario « nuclear x 10 » et optimisation of U in PWR	≈ 40 – 80 years

- breeding is really needed before 2050 only if nuclear power production increases by a factor 5 or 10

- if nuclear power increases slowly, standard water reactors will dominate

-But we have to manage the « long characteristic time » of nuclear industry

- developement of a new technology : decades

-Lifetime of reactors : more than 50 years

- accumulation of Pu to start the breeders : decades

- anticipation is needed !

Resources vs flux

Capacity of the mining industry to make the resources available ?

Today

Natural Uranium production 45000 t/y
9 mines only with a capacity > 1000 t /an
50% of the world production (26000 t/an)

Ex : Cigar Lake

- operation delayed from 2007 to 2013?
- 550000 t U₃O₈

2050

IIASA « bas » et « haut » : 216000 t/an à 336000 t/an

Even if resources are present, mining industry could be uncapable of providing natural uranium fast enough

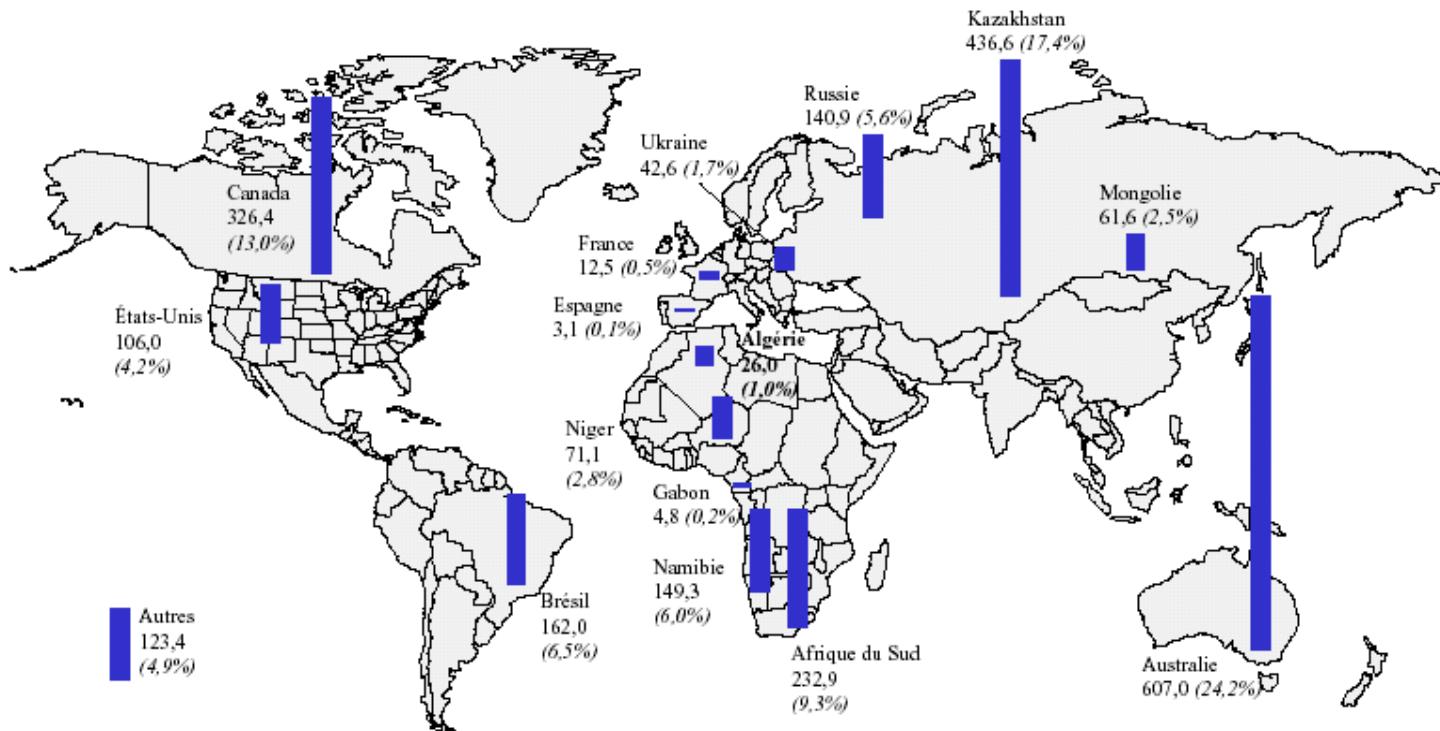
Mine	Production 2005 (t/y)
Mac Arthur River	7 200
Ranger	4 589
Olympic Dam	3 388
Krasnokamensk	3 037
Rossing	2 583
Arlit	1 750
Rabbit Lake	1 544
Akouta	1 403
Akdala	1 000

Réf M. Delpech, CEA/DEN

⇒ deployment of Gen4 possible even if nuclear power does not increase so much

Réserves mondiales prouvées d 'uranium* (1. 1. 1999)

Unité : Milliard de tep



Total monde : 2 506,2 milliers de tonnes (hors Chili et Chine)

(*) ressources raisonnablement assurées récupérables à moins de 80\$/kg U

Source : Observatoire de l'énergie d'après CEA/DSE et AIE/OCDE

Breeding principle

Breeding : every fissile nucleus consumed must be replaced



The mass of fissile material is constant, only the fertile material is consumed
(1 ton per year for 1 GWe reactor, 200 t today in PWR)

But breeding needs neutrons

For one fission



ν

neutrons are produced



1

neutron used to induce one new fission

α

neutrons captured = $\sigma^{\text{cap}} / \sigma^{\text{fis}}$ on fissile nucleus

$1 + \alpha$

neutrons captured to replace the fissile lost

$$\nu - 2(1 + \alpha)$$

$$> 0$$

\Rightarrow breeding possible

$$< 0$$

\Rightarrow breeding impossible

Breeding principle

In a first approximation, the breeding condition depends only on neutronic characteristics of the fissile nucleus

$$\alpha = \frac{\sigma_{\text{fissile}}^{\text{capture}}}{\sigma_{\text{fissile}}^{\text{fission}}}$$

$$Nd(E) = v - 2(1 + \alpha(E))$$

Thermal spectrum

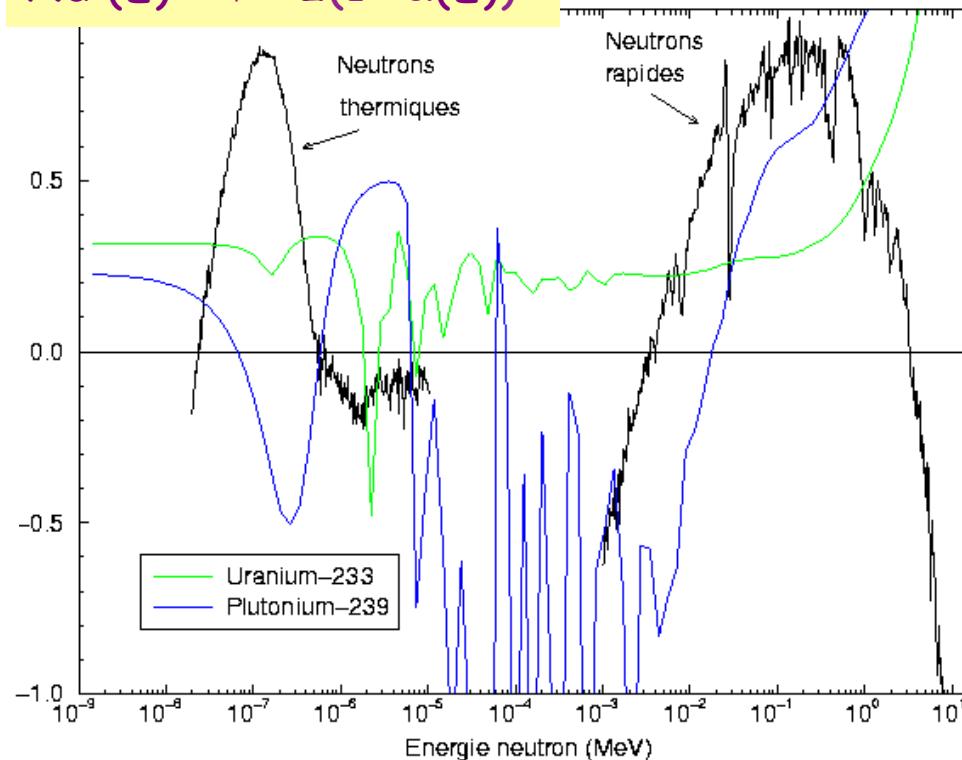
Th/U Nd > 0

U/Pu Nd < 0

Fast spectrum

Th/U Nd > 0

U/Pu Nd > 0



Uranium cycle requires fast neutrons to reach the breeding condition

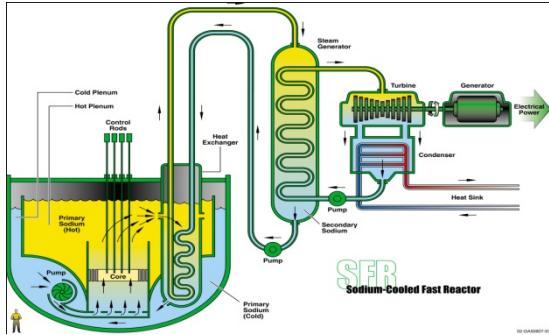
Sustainable nuclear power : breeding principle

Mean values on thermal or fast spectra

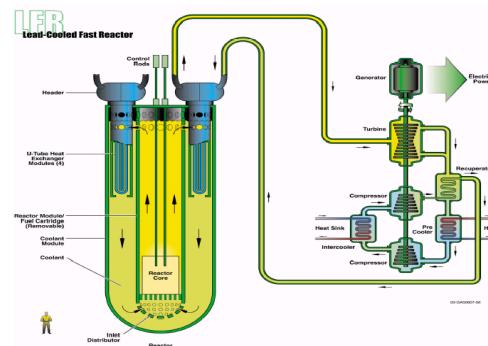
	Thermique		rapide	
Cycle	U/Pu	Th/U	U/Pu	Th/U
Fissile	^{239}Pu	^{233}U	^{239}Pu	^{233}U
σ_{fis} fissile	90	50	1,85	2,7
σ_{cap} fissile	50	6	0,5	0,27
ν	2,9	2,5	2,9	2,5
$N_d = \nu - 2(1 + \alpha)$	-0,2	0,3	0,36	0,3

Breeder technologies

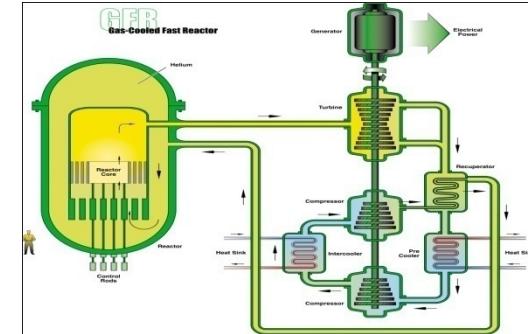
Na



Pb



He



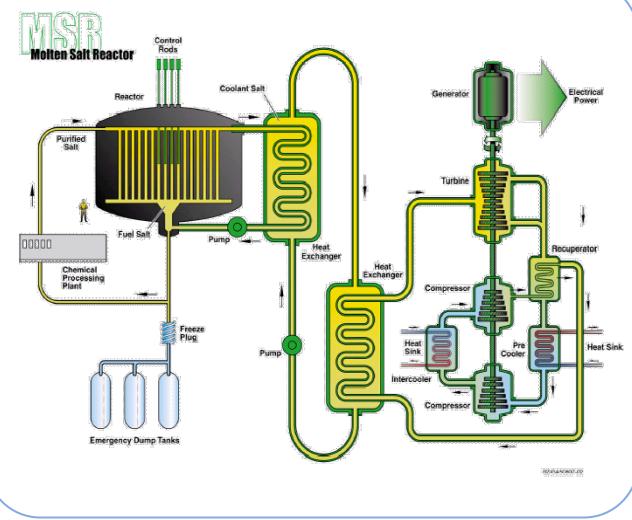
Water cannot be used anymore

Heat must be transferred with a heavy fluid or a gas

SFR : sodium fast reactor

Main difficulties :

- positive void coefficient
- chemical instability of sodium
- industrialization, safety, cost



ASTRID prototype

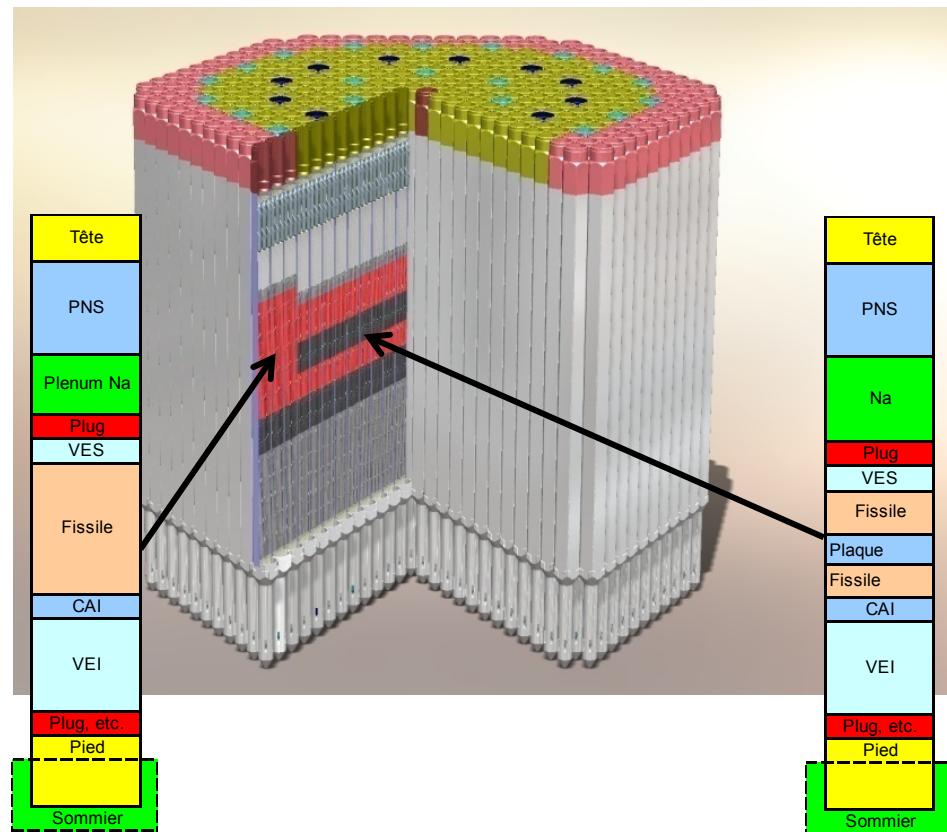
ASTRID french SFR Prototype 600 MWe
2020-2025 ?

Main innovation as compared to Superphenix concept

Heterogeneous core

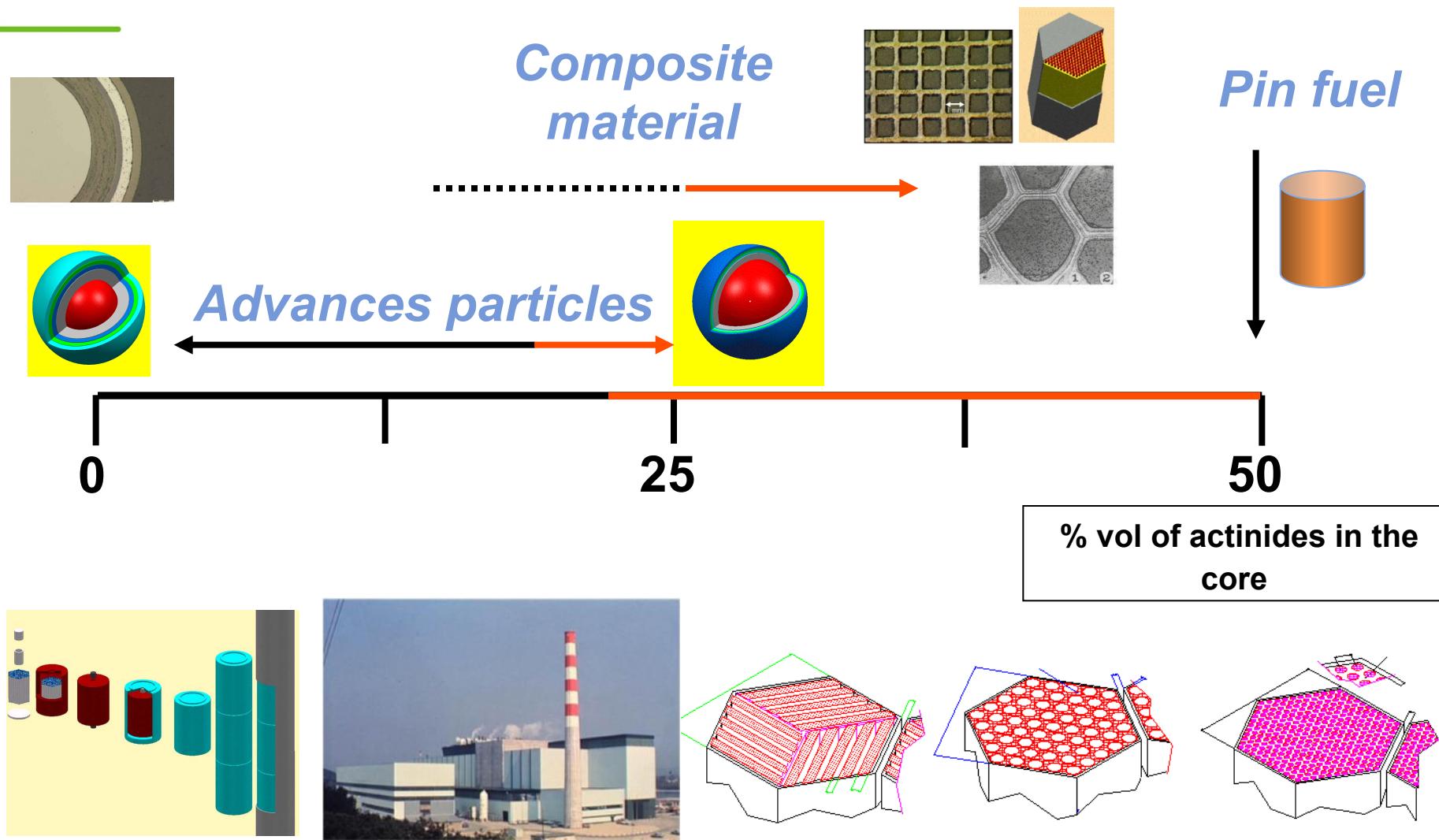
Reduce the positive void effect

Decision planed for 2012, but Fukushima, politics... decision delayed after 2017...



Innovative fuel for fast gas reactors

Main challenge : the fuel



Fast breeder reactor : capacity of deployment

key point : How many Pu do we need to start a fast breeder reactor?

Once started, a breeder reactor consume only fertile material (^{238}U)

Mass Pu of Pu constant

But Pu inventory has to be provided to start the reactor

This « long term » question helps to understand why the Pu management today can be so different from a country to another

US : Pu = waste

France : Pu = valuable material

In both cases, Pu dominates the risks of nuclear materials : radiotoxicity, criticity, proliferation, etc.

Fast breeder reactor : capacity of deployment

key point : How many Pu do we need to start a fast breeder reactor?

Once started, a breeder reactor consume only fertile material (^{238}U)

Mass Pu of Pu constant

But Pu inventory has to be provided to start the reactor

Example for a fast breeder reactor (Superphenix type)

$\rho_{\text{th}} = 500 \text{ Wth/cm}^3$ (of fuel)

$d = 10 \text{ g / cm}^3$ (Oxide fuel)

$\eta_{\text{th}} = 40 \%$

$\approx 50 \text{ tons of fuel (U+Pu) for 1 GWe core}$

Simple calculation for a fast breeder reactor (Superphenix type)

1/ Mass of total fuel U+Pu in a core

$\rho_{th} = 450 \text{ Wth/cm}^3$ of fuel $(\text{U+Pu})\text{O}_2$

$d = 10 \text{ g / cm}^3$ (Oxide fuel)

$\eta_{th} = 40 \%$

$\approx 50 \text{ tons of fuel (U+Pu) for 1 GWe core}$

2/ Mass of Pu

The concentration of fissile material is determined by the iso-generation condition
capture on fertile = (capture+fission) of fissile

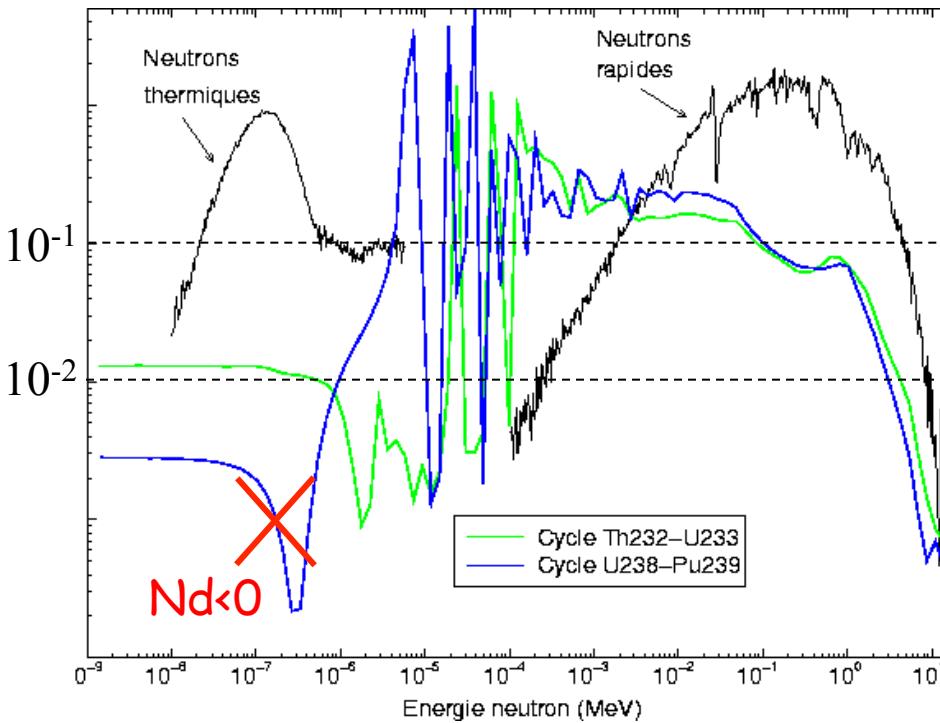
$$N_{\text{fertile}} \sigma_{\text{fertile}}^{\text{capture}} = N_{\text{fissile}} \sigma_{\text{fission}}^{\text{capture+fission}}$$

$$\frac{N_{\text{fissile}}}{N_{\text{fertile}} + N_{\text{fissile}}} = \frac{\sigma_{\text{fertile}}^{\text{capture}}}{\sigma_{\text{fission}}^{\text{capture+fission}} + \sigma_{\text{fertile}}^{\text{capture}}}$$

Pu concentration for breeding

Like $N_d(E)$, this concentration can be seen as a function of the energy of the neutron

$$C_{\text{fissile}} = \frac{N_{\text{fissile}}}{N_{\text{fissile}} + N_{\text{fertile}}} = \frac{\sigma_{\text{fertile}}^{\text{cap}}}{\sigma_{\text{fertile}}^{\text{cap}} + \sigma_{\text{fissile}}^{\text{cap+fis}}}$$



Th/U thermal

$$C_{\text{fissile}} \approx 2\%$$

U/Pu fast

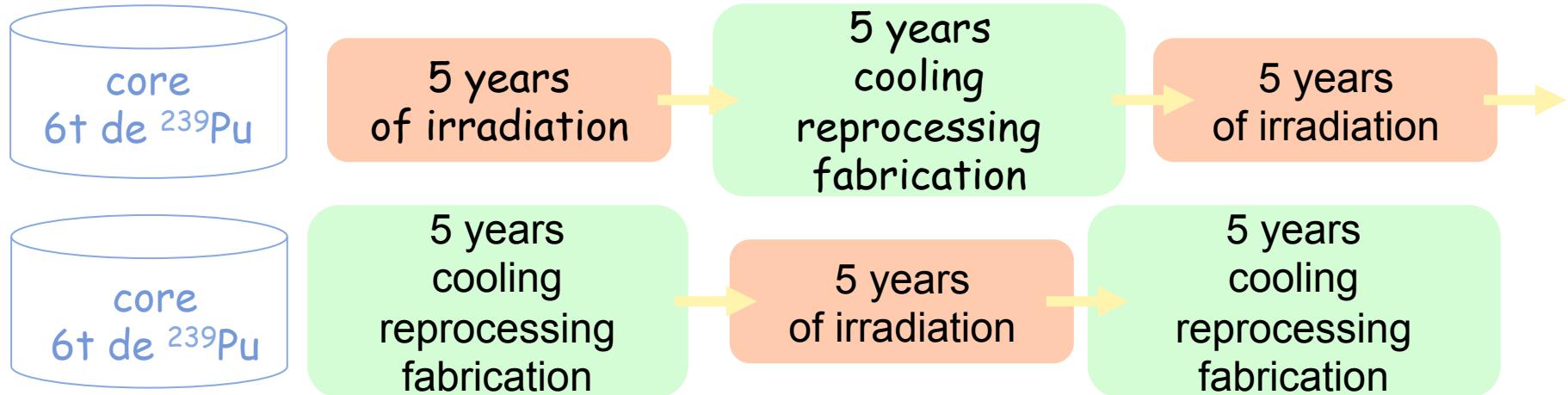
$$C_{\text{fissile}} \approx 12\%$$

For U/Pu cycle, in fast spectrum
Mass of the core = 50 tons / GWe

	σ^{cap} fertile	σ^{cap} fissile	σ^{fis} fissile	c (%)	M_{core}
U/Pu fast	0.3 b	0.5 b	1.85 b	12,5 b	6,5 tons

Here we took only into account ^{239}Pu , but in a real reactor, we have also ^{240}Pu , ^{241}Pu , ^{242}Pu , ...

Fuel reprocessing needed : extract the fission products, add fresh ^{238}U (consumed)



In this case, two cores are needed, and work alternately in the reactor. The total inventory of fissile material for one reactor is thus the double of the core inventory

In this case, 12 tons of ^{239}Pu are needed to start a fast breeder reactor (1 GWe)
Taking into account ^{240}Pu , ^{241}Pu and ^{242}Pu ,
we need ~20 tons of Pu for 1 GWe fast breeder reactor

French case

Plutonium inventory for a FBR fleet

Constant power

$$60 \text{ GWe} \rightarrow 60 \times 20 = 1200 \text{ tonnes de Pu}$$

situation in 2010

Total : 300 tons of Plutonium in UOX and MOX spent fuels

= only 25% of the inventory of 60 GWe FBR fleet

In this case, Pu cannot be considered as a waste, but as a valuable material to start the future breeder reactors, even if we are not sure to start them one day...

Another way to see the transition PWR → FBR

1 PWR UOX produces ~ 250 kg de Pu/(GWe.an)
= FBR Pu inventory in 50 years of operation = life time of PWR
Transition PWR → FBR is possible at constant power

Scénario Pu mono-reprocessing (MOX)

French case : 80% des UOX reprocessed (MOX)
20% des UOX stored
MOX used fuel are not reprocessed

Consumption of 40% of the Pu during the MOX irradiation

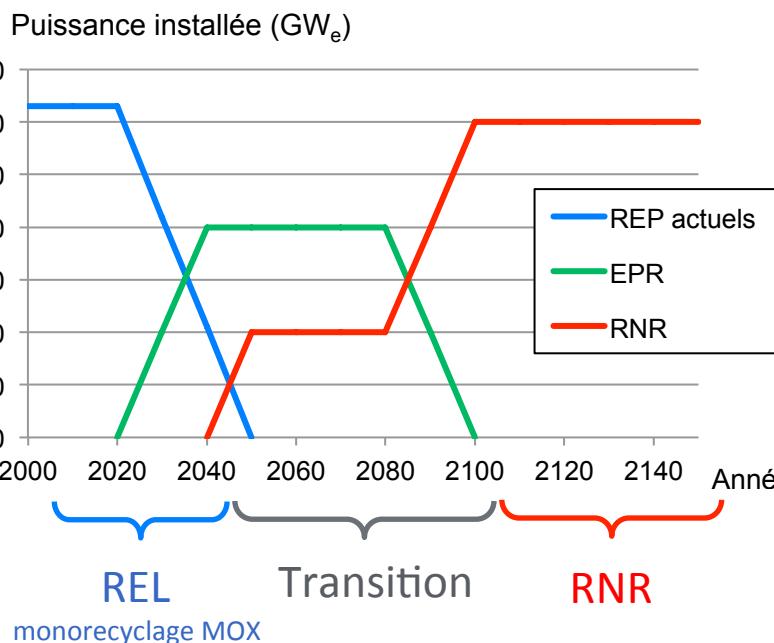
mean production of Pu /(GWe.an) in France = 150 kg
inventory of FBR needs 80 years of operation

The simplified calculations are consistent with detailed calculations concerning transition scenarios PWR → FBR

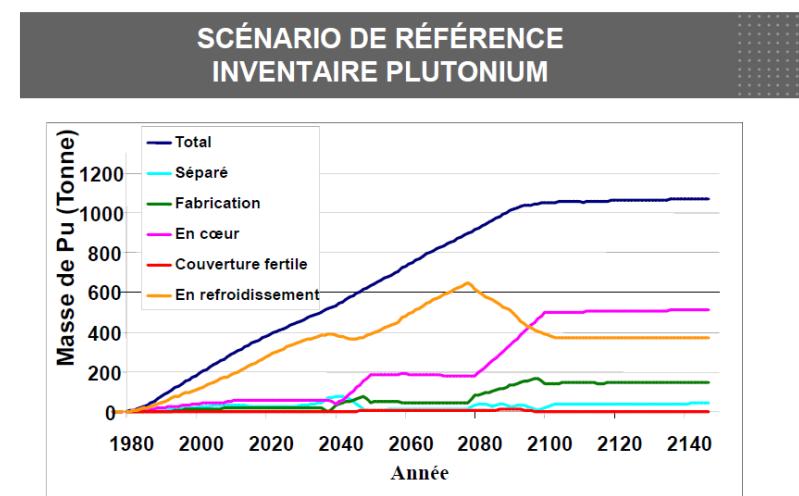
The transition can be finished in 2100, with no Pu excess

Pu inventory in fast reactors

Typical french scenario of a « fast »
FBR deployment



~1000 tons of Pu at equilibrium
in the french park (60 Gwe)



Inventaire Pu total ~1050 tonnes à l'équilibre,
avec une teneur Pu de 16.5% assurant l'iso-génération

Audition CNE du 5 janvier 2011 : Cycle des matières pour Astrid



The deployment of Gen4 systems is limited by the Pu needed to start them
In this type of scenario, Pu produced today cannot be considered as a waste !
It has to be kept during decades to allow the future deployment of breeder...

Doubling time of a reactor

- Once the breeder started, it can surgenerate fissile material
- This material can be accumulated outside the reactor, in order to start another breeder reactor
- The time needed to accumulate enough fissile material is called « doubling time » T_d
- The mass surgenerated depends on N_a . If no neutron is lost on structures and by leakage, the surgenerated mass is $M_s = N_a F$ avec F = mass de fissionned material per year

$$T_d = \frac{I_{\text{tot}}}{M_{\text{sur}}} = \frac{I_{\text{tot}}}{N_a F}$$

Typical value of what is possible in a real reactor

$N_a = 0.3$; $F = 1000 \text{ tons/y}$; $I_{\text{tot}} = 20 \text{ tons}$

$T_d = 67 \text{ years}$

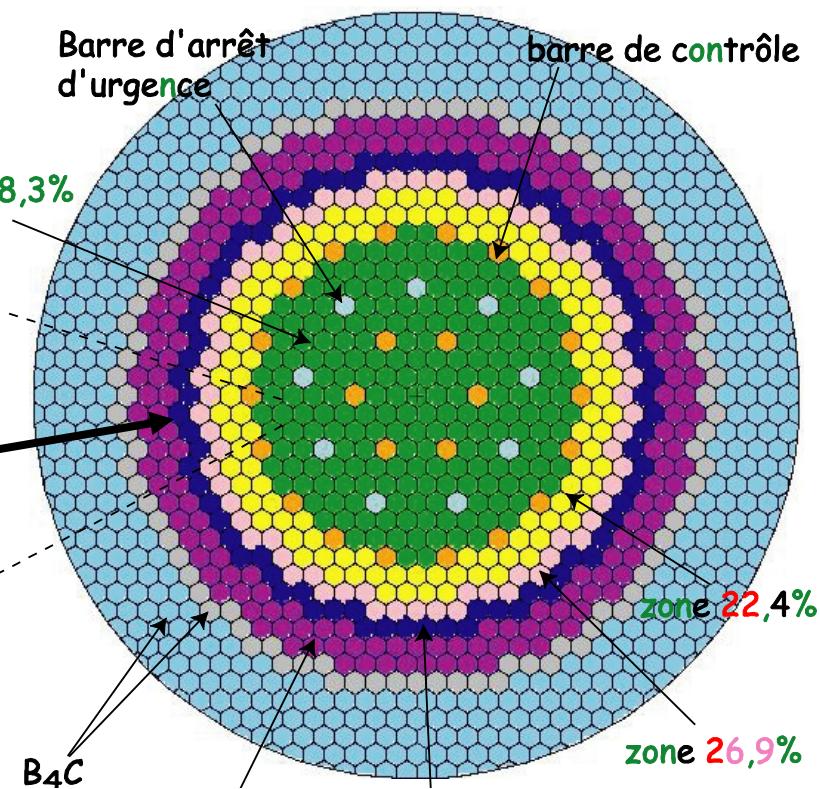
In order to optimize the use of the available neutrons, the core can be surrounded by a fertile blanket. The leakage are minimized, and the breeding is optimal.

Radial blanket can be managed independently from the core

But axial blanket must be reprocessed with the fuel

Example:
Na-FBR
EFR or
SFR type

Blanket of
fertile
(U_{app})



For a fleet containing a large number of reactors, the doubling time can be reduced

$$dN_r = x N_r(t) \delta t$$

x = fraction of mass
surgenerated per year $x =$
 $M_{\text{sur}} / I_{\text{tot}}$

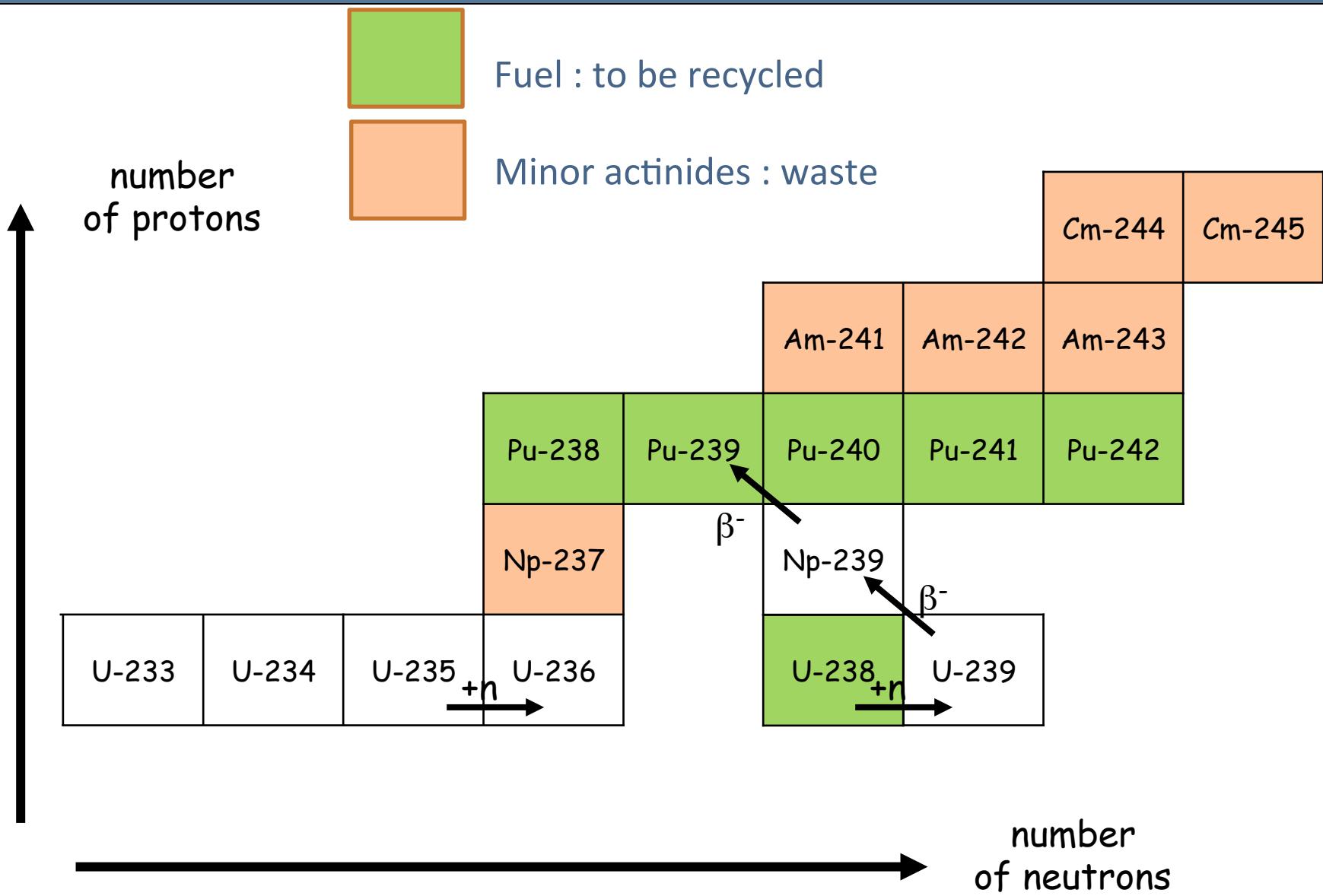
Number of
reactor at
the time t

We have $N_r(t) = N_r^0 \exp(xt)$ et $T_d^\infty = \frac{\ln(2)}{x} = \ln(2) T_d^0$

The real doubling time is between
 $0,7 T_d^0$ et T_d^0

The minimal doubling time for FBR-Na is around 25 ans

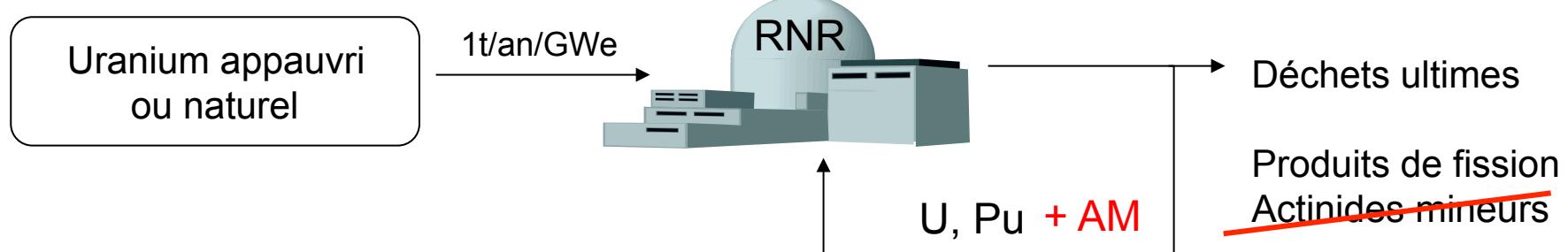
Transmutation of minor actinides



Fast reactors : minor actinides transmutation

Fast reactors are able to recycle their own minor actinides

Different strategies are possible



recyclage homogène : U,Pu, Np, Am, Cm recyclés ensemble, tout le cycle est affecté par la présence des AM

- recyclage hétérogène : AM recyclés en couverture fertile (20 à 40% AM), une partie du cycle est affecté par les AM
- double strate : les AM sont transmutés dans des réacteurs dédiés (ADS) : découplage production d'électricité / transmutation AM

- Transmutation des actinides mineurs

⇒ Recyclage U+Pu+AM, déchets ultimes minimisés

Transmutation of M.A.

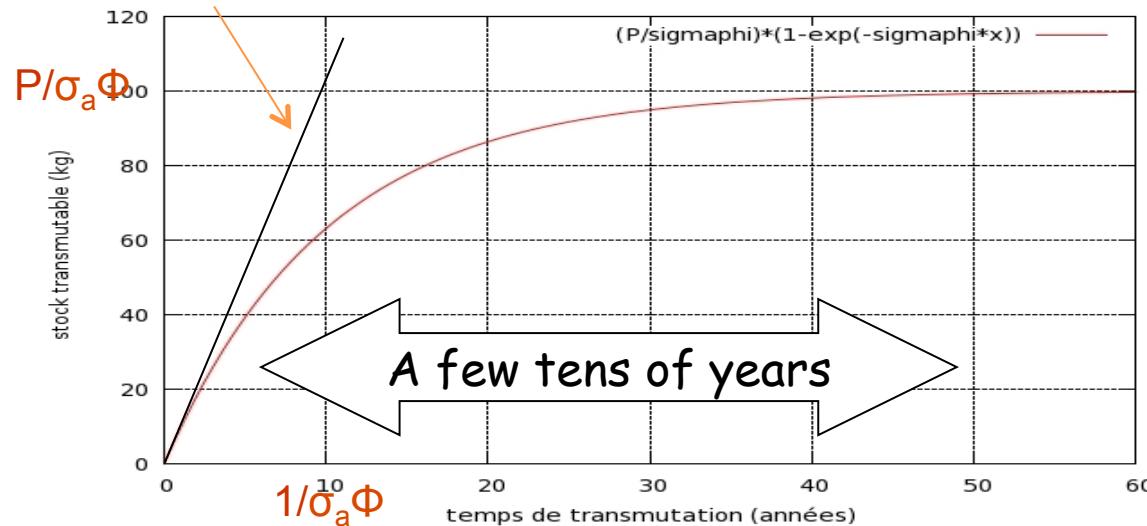
1/ transmutation of M.A. can be considered only together with Pu multireprocessing strategies

2/ transmutation of M.A. consists on separating M.A. from spent fuel and transmute them in reactor

3/ naive scheme (but not so false)

$$\frac{dN}{dt} = A - \sigma\phi N \rightarrow N(t) = \frac{P}{\sigma_a\phi} (1 - e^{-\sigma_a\phi t})$$

Without transmutation



$1/\sigma_a\Phi$ = time needed to reach the equilibrium

But also :

- Equilibrium inventory = production at $t = 1/\sigma_a\Phi$ without transmutation
- For Am and Cm, $1/\sigma_a\Phi = 5-20$ years in fast spectrum

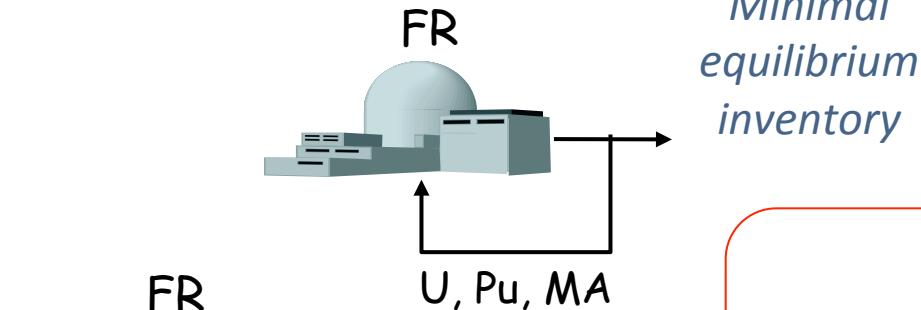
Take into account time for cooling down and reprocessing : real $1/\sigma_a\Phi$ multiplied by 2 or 3

Transmutation strategies must last several decades or centuries

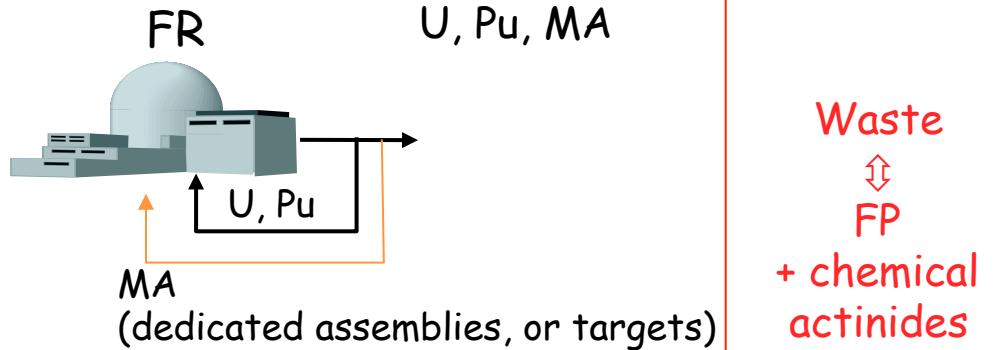
M.A. transmutation

Main strategies of M.A. considered

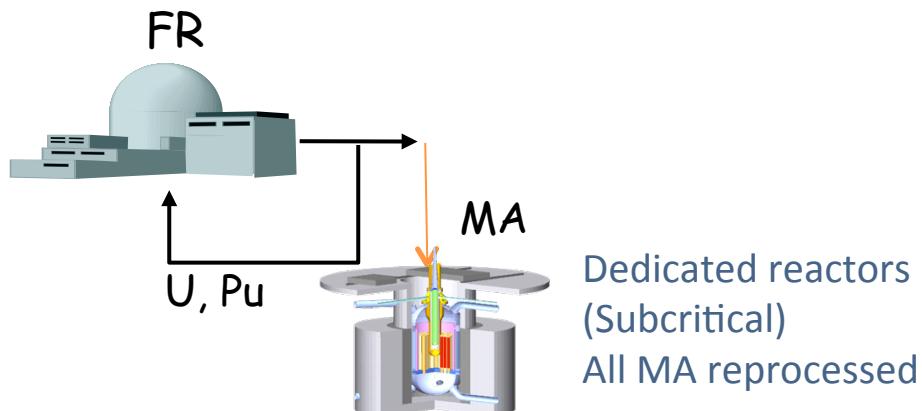
Homogeneous transmutation
(no U/Pu/MA separation possible)



Heterogeneous transmutation



Double-strata strategy



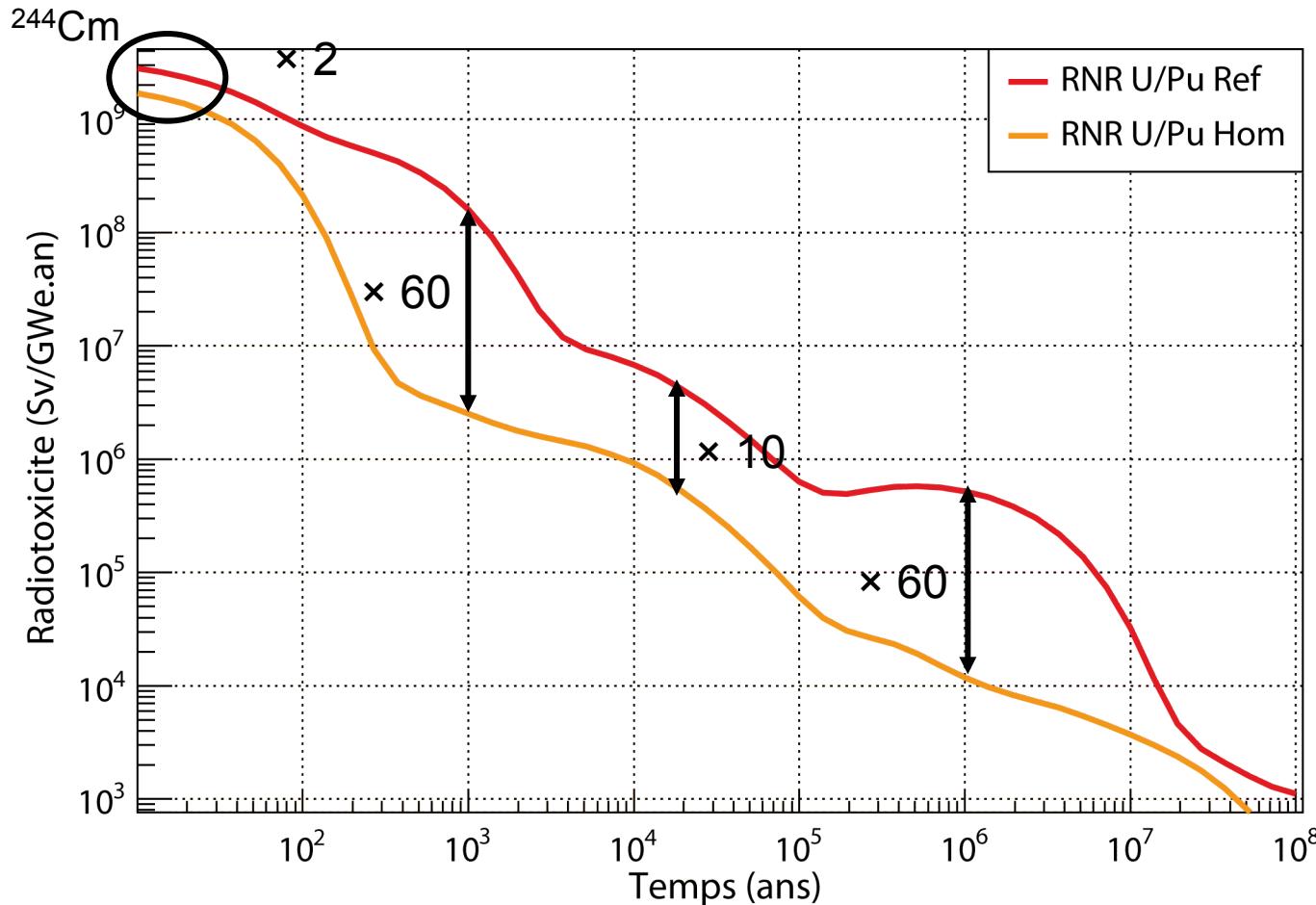
Minimal equilibrium inventory

Waste
↔
FP
+ chemical actinides losses (~0,1%)

Dedicated reactors
(Subcritical)
All MA reprocessed

Waste comparison

Comparison Fast Breeder Reactor (Na) with and without MA transmutation



Transmutation scenarios for the French fleet

C1

Replacement of present fleet

50% EPR 2020 - 2035

50% FBR 2035 - 2050

No transmutation of MA

C2

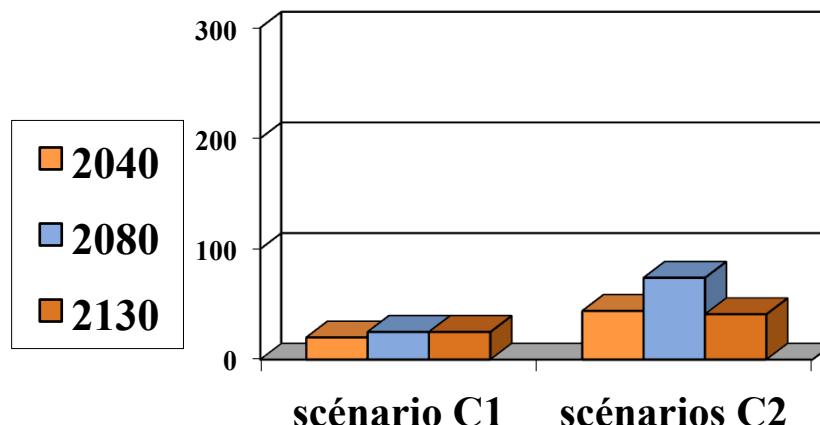
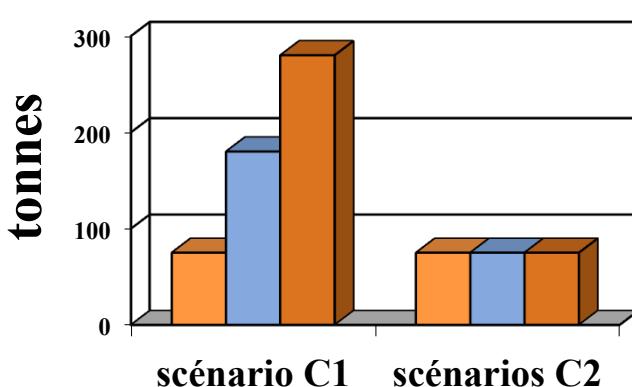
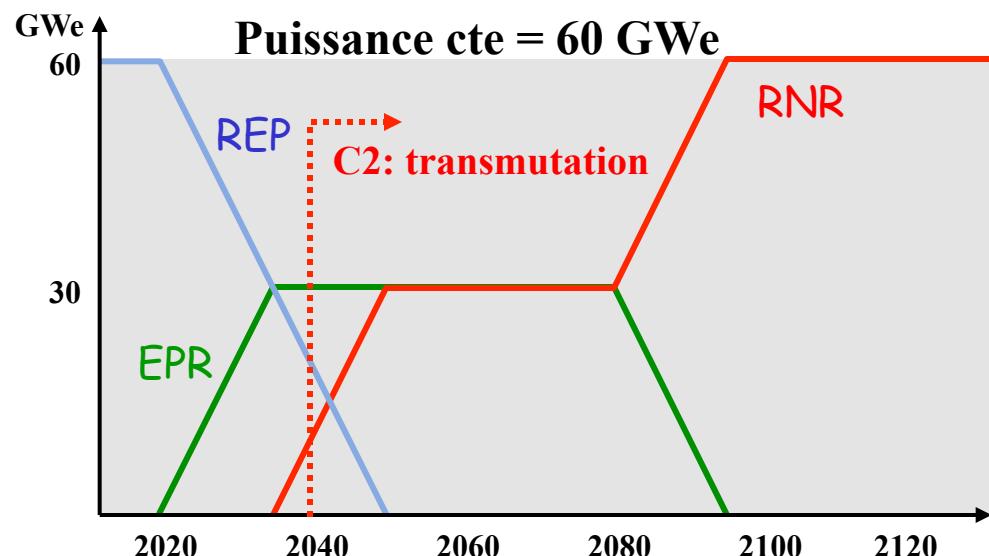
Replacement of present fleet

50% EPR 2020 - 2035

50% FBR 2035 - 2050

+ Transmutation of MA

In FBR in 2040



Calculation made by CEA, french public debate,
2005

Homogeneous transmutation in FBR vs double strata strategy

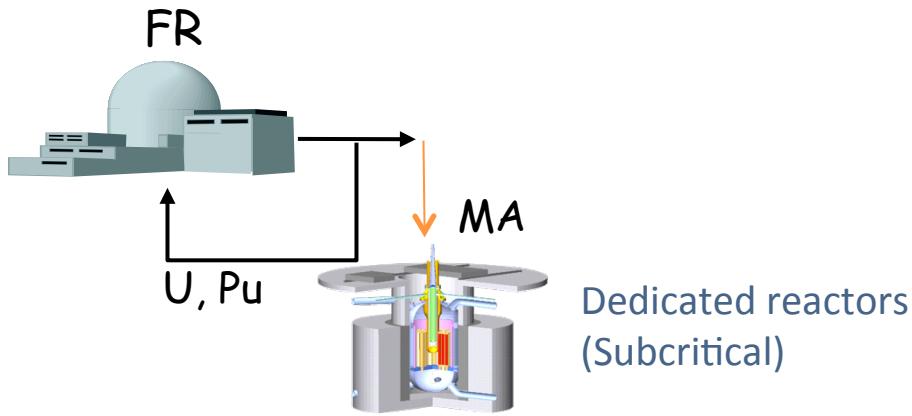
~Same performances in term of waste transmutation, waste production, ...

The difference concerns essentially the industrial / political approach

MA Inventories more important with ADS strategies, due to low power ADS

	Hom.	Double strata	
	MOX (ref.)	FBR Pu+AM	FBR Only Pu
Reprocessing			
Residual power (W/g/tHM)	1	x 2	x 1
			x 70
Neutron source (n/ s/tHM)	1	x 2	x 1
			x 200
Fabrication			
Residual power (W/g/tHM)	1	x 2.5	x 0.5
			x 90
Neutron source (n/ s/tHM)	1	x 150	x 1
			x 20000
t/an	820	~400	9

ADS : impact of low power systems and associated cycle, EFIT 400 MWth



Mass of MA to be transmuted → determine the power of the second stratum

FBR scenario : MA produced = 45 kg/(GWe.an)

$E_{\text{fission}} \sim 200 \text{ MeV} \rightarrow P_{\text{th}} (\text{ADS}) = 0,1 \text{ GWth} = 0,04 \text{ GWe} = 4\% \text{ from first strata}$

French case, 60 GWe (load factor=80%) = ADS 5,28 GWth ~16 EFIT (for 50 FBR)

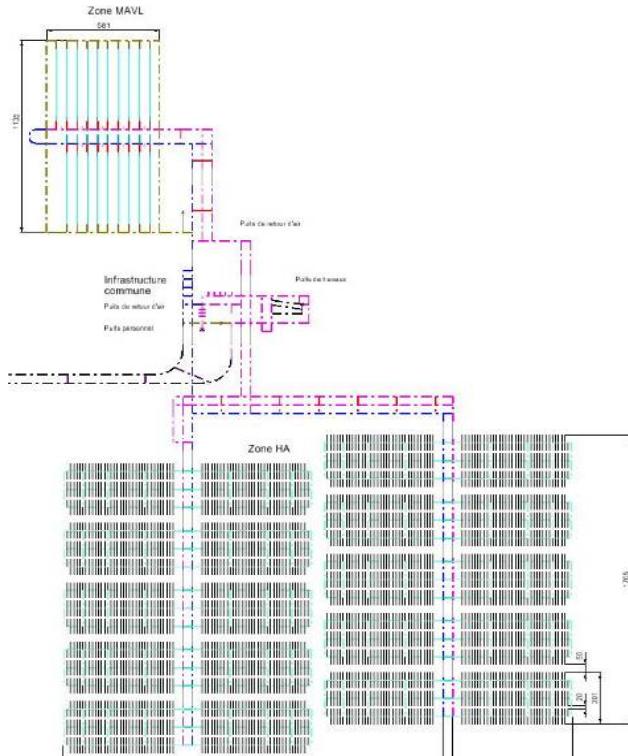
Increasing the power of ADS is a key point :

- Intensity accelerator → >50 mA !
- Management of the radial power distribution

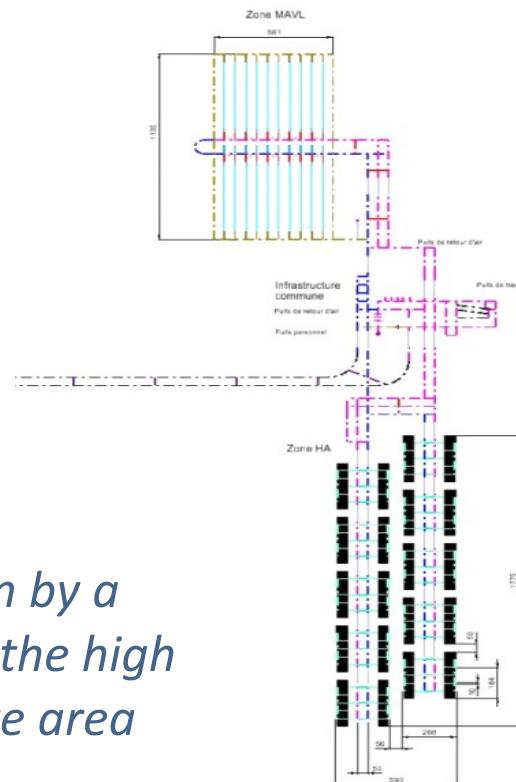
Decay heat and impact for geological disposal

French study (ANDRA, CEA 2012)

Without MA transmutation
Intermediate disposal 70y



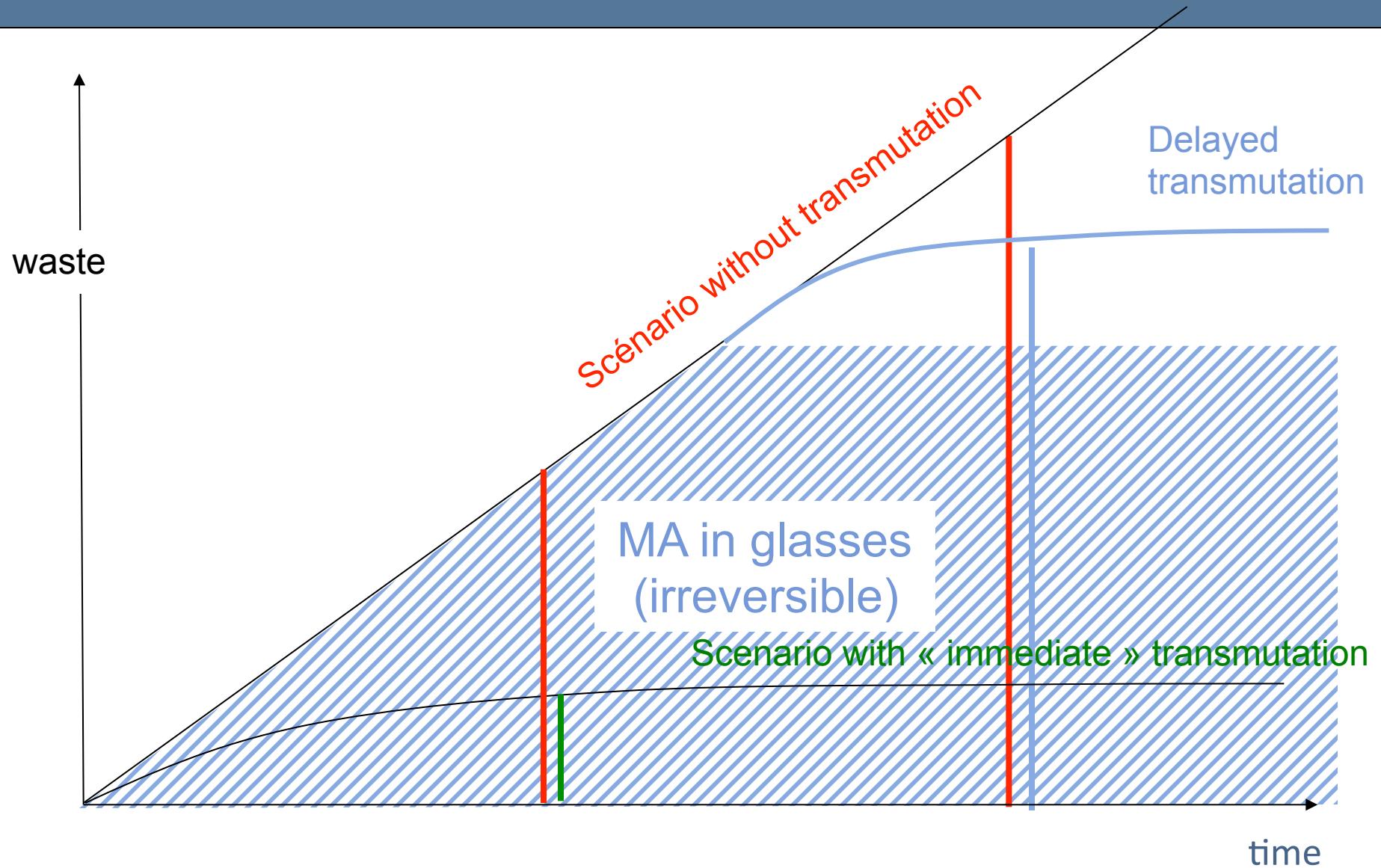
With MA transmutation
Intermediate disposal 120 y



Reduction by a factor 5 for the high level waste area

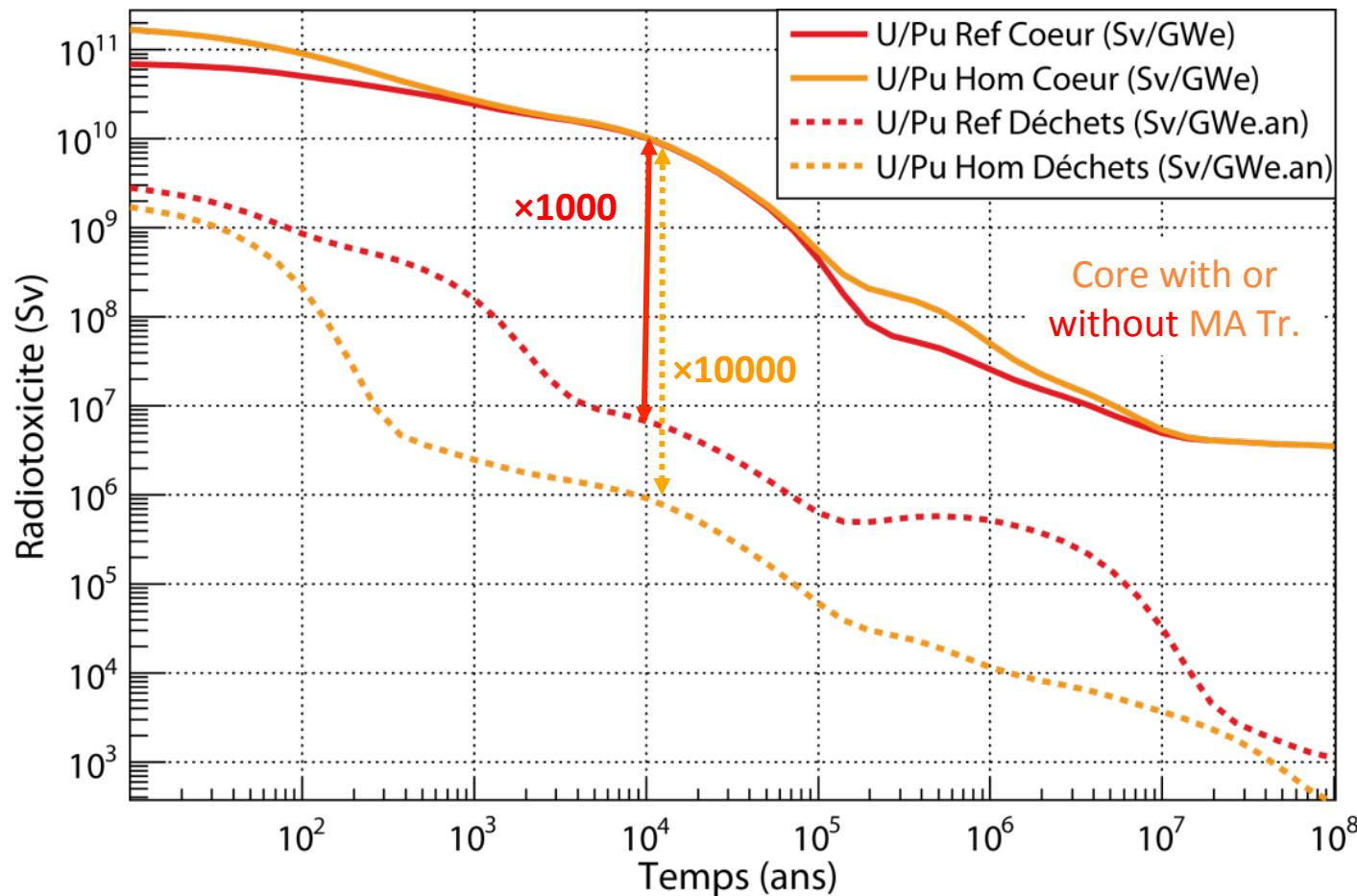
Ref: CEA, dossier 2012

If we look backward...



If we look forward...

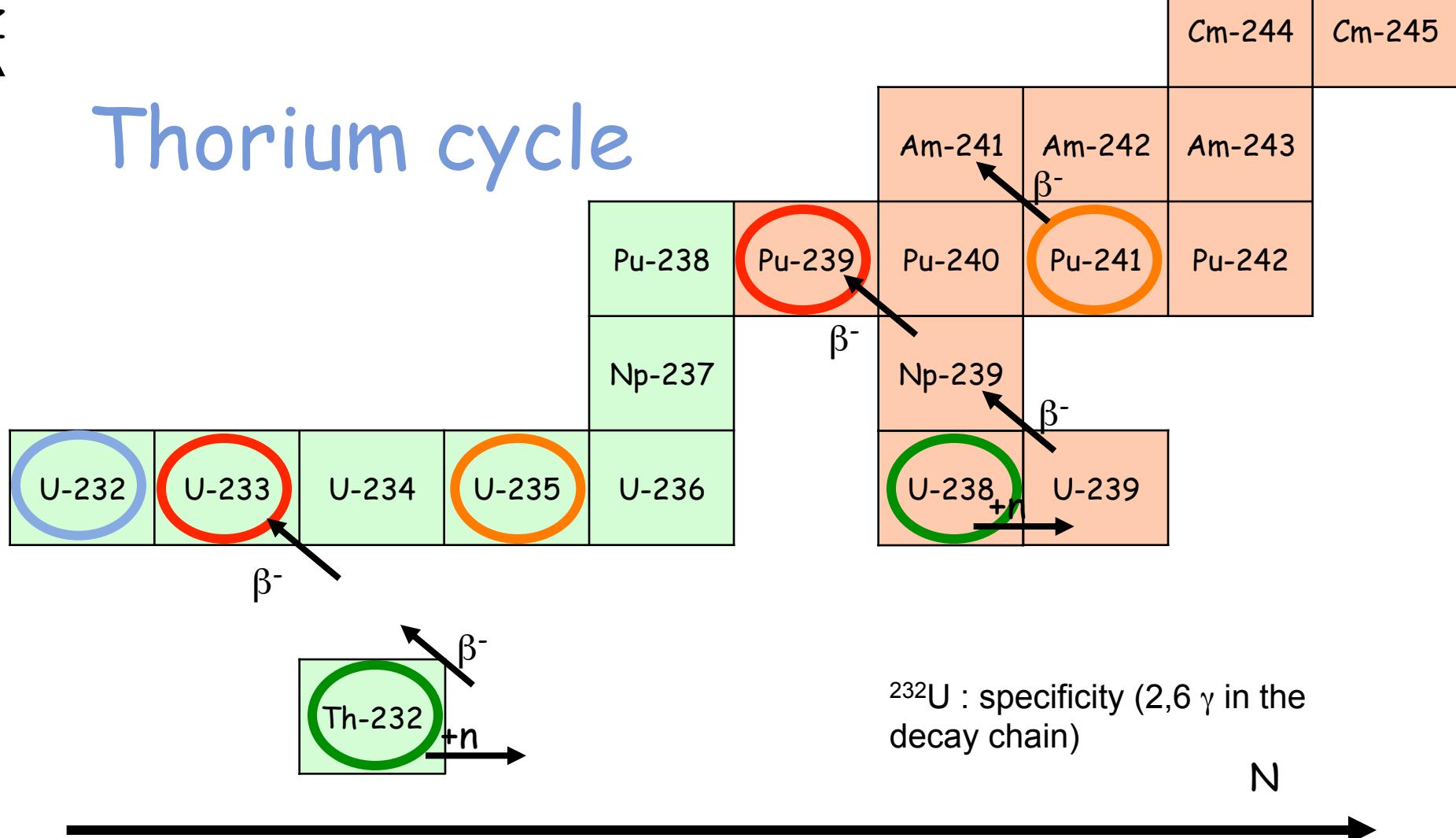
Finally, the Pu remains will remain the most radiotoxic material for a long time, even if it is used in Fast Breeder Reactors...

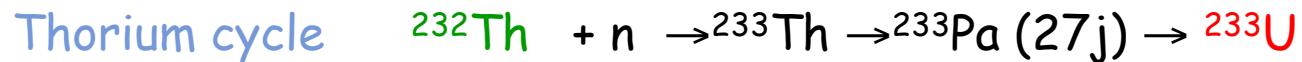
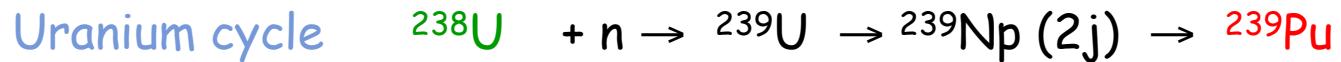


THE END

Thorium cycle

Sustainable nuclear power : thorium cycle

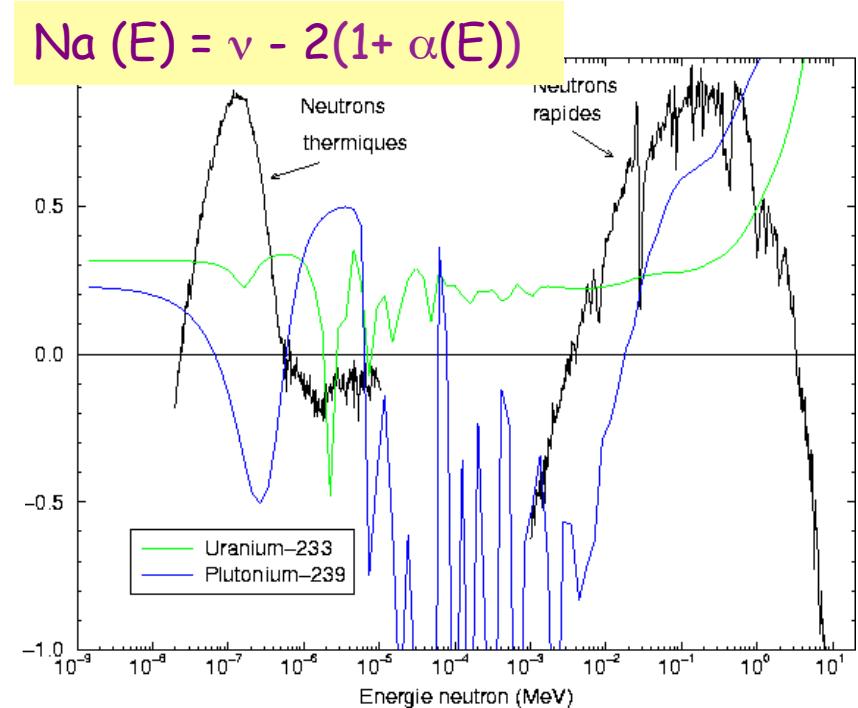




Fertile
nuclei

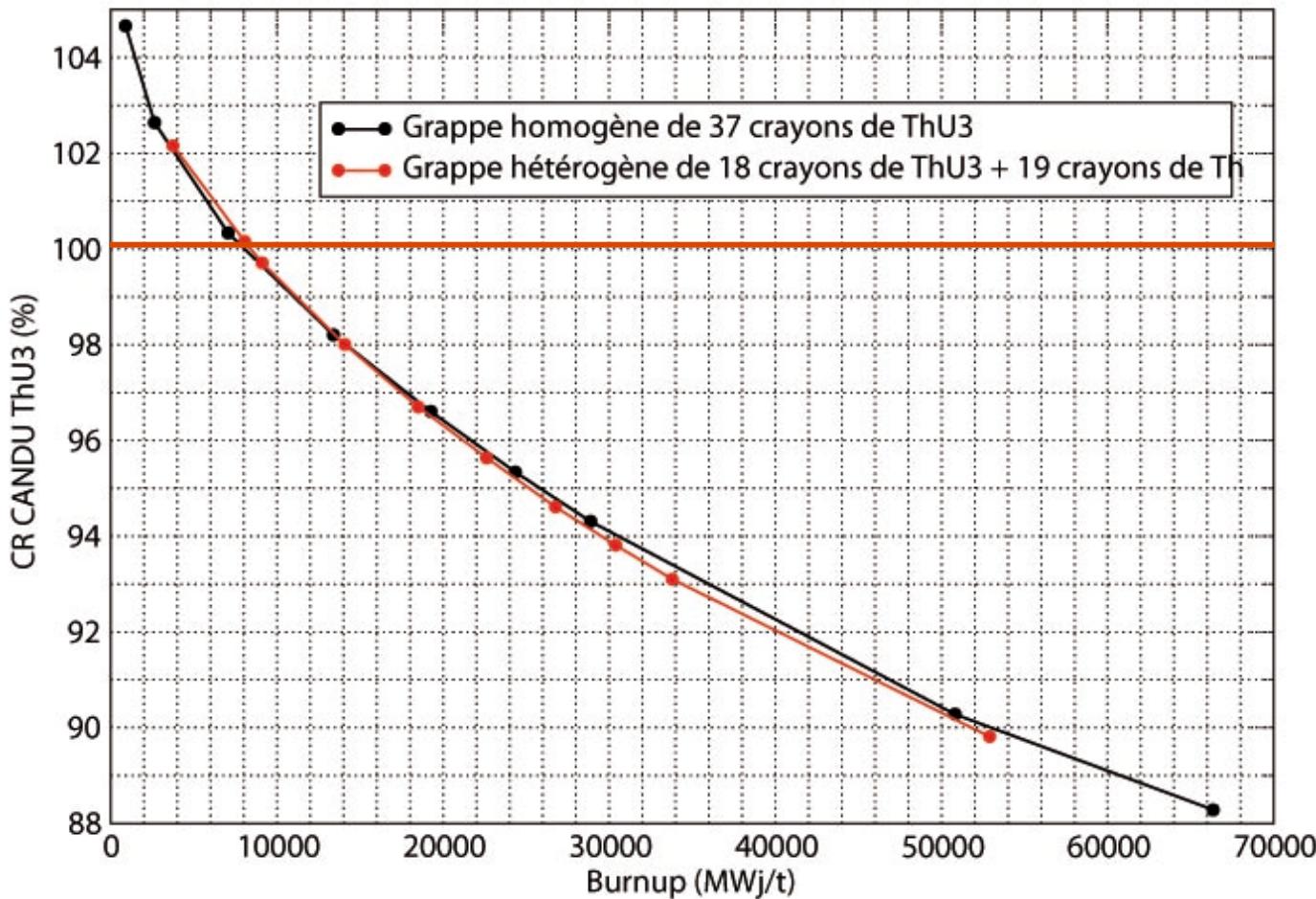
Fissile
nuclei

Breeding possible (theoritically) in a thermal spectrum



Optimized system with solid fuel : Candu reactor (heavy water, parasitic capture minimized)

CR de grappes de CANDU ThU3 en fonction du burnup et de la répartition du combustible



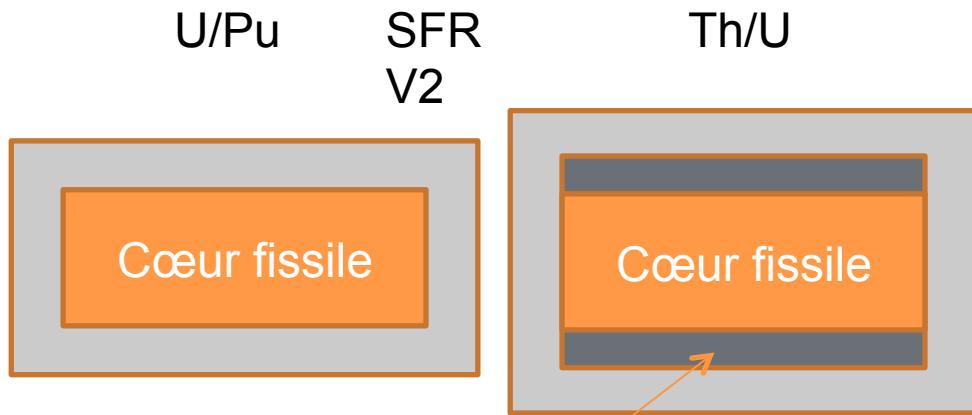
Breeding possible
for BU<10000
MWj/t

Spectre rapide : comparaison U/Pu et Th/U

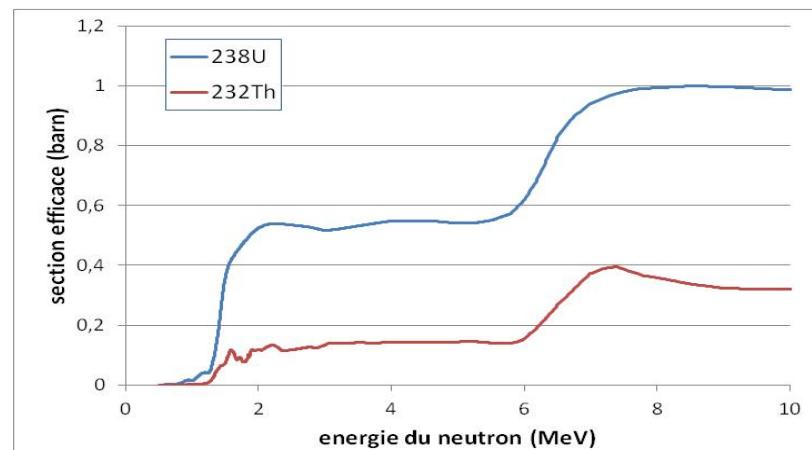
Fast spectrum : neutron balance better for U cycle

FBR-Na U/Pu : ^{238}U 15% of fissions

FBR-Na Th/U : ^{232}Th 3% of fissions



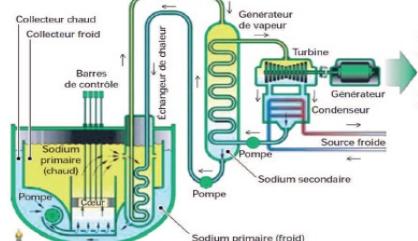
Fertile blanket (^{232}Th) necessary for breeding



Void coefficient, SFR-V2

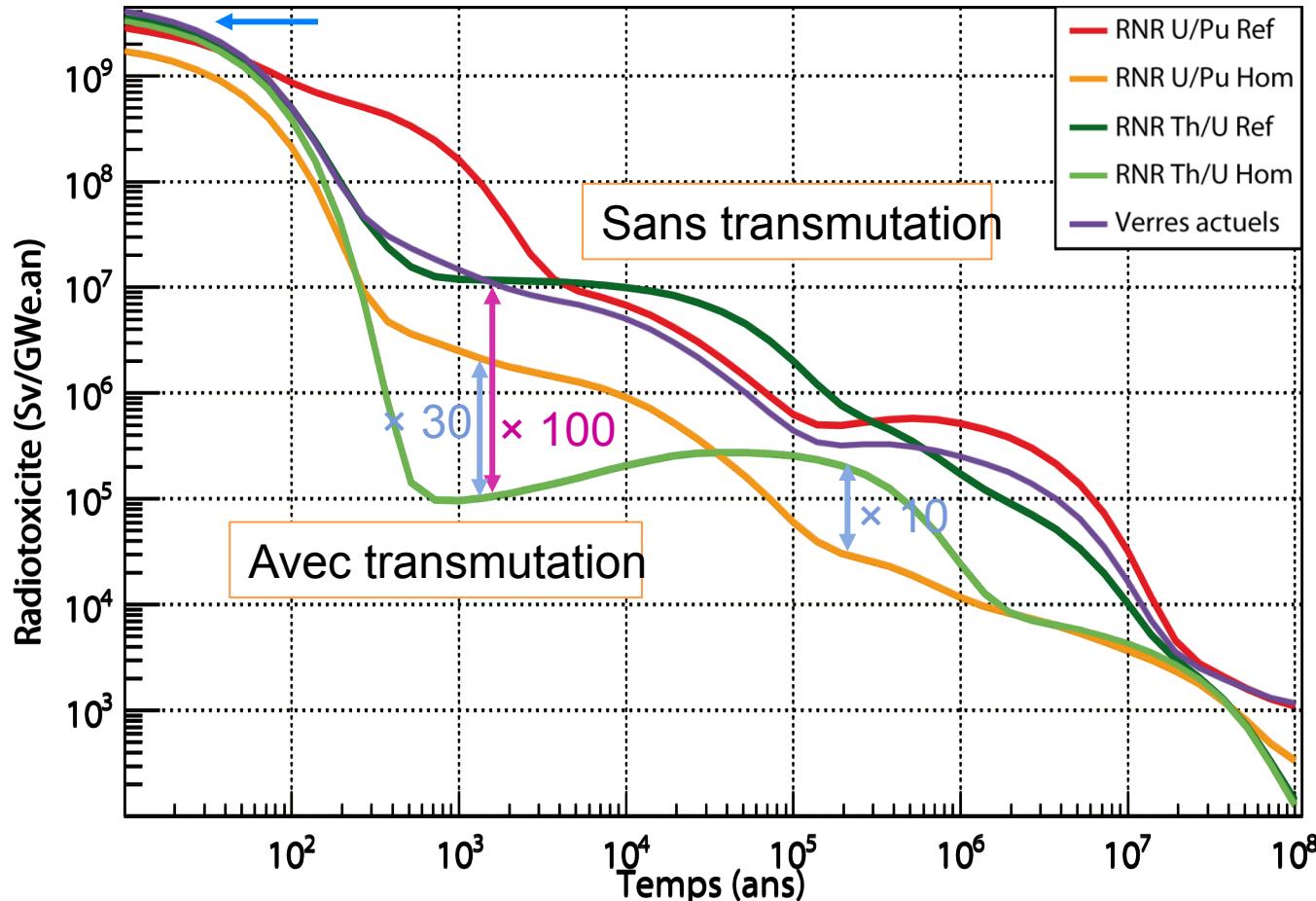
RNR U/Pu ~1750 pcm

RNR Th/U -200 pcm



Induced radiotoxicities for U and Th cycles

With or without minor actinides transmutation

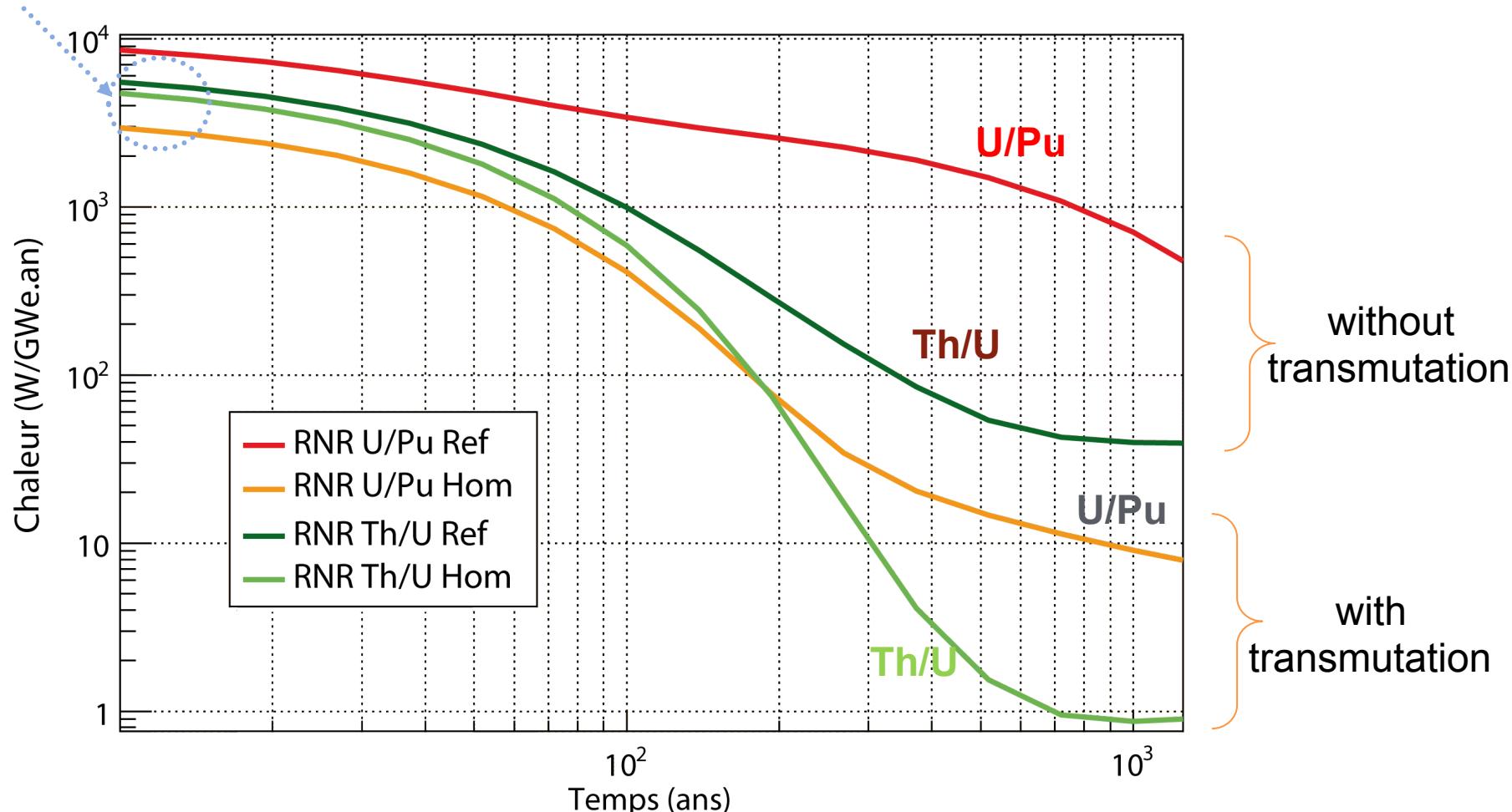


Residual power of waste

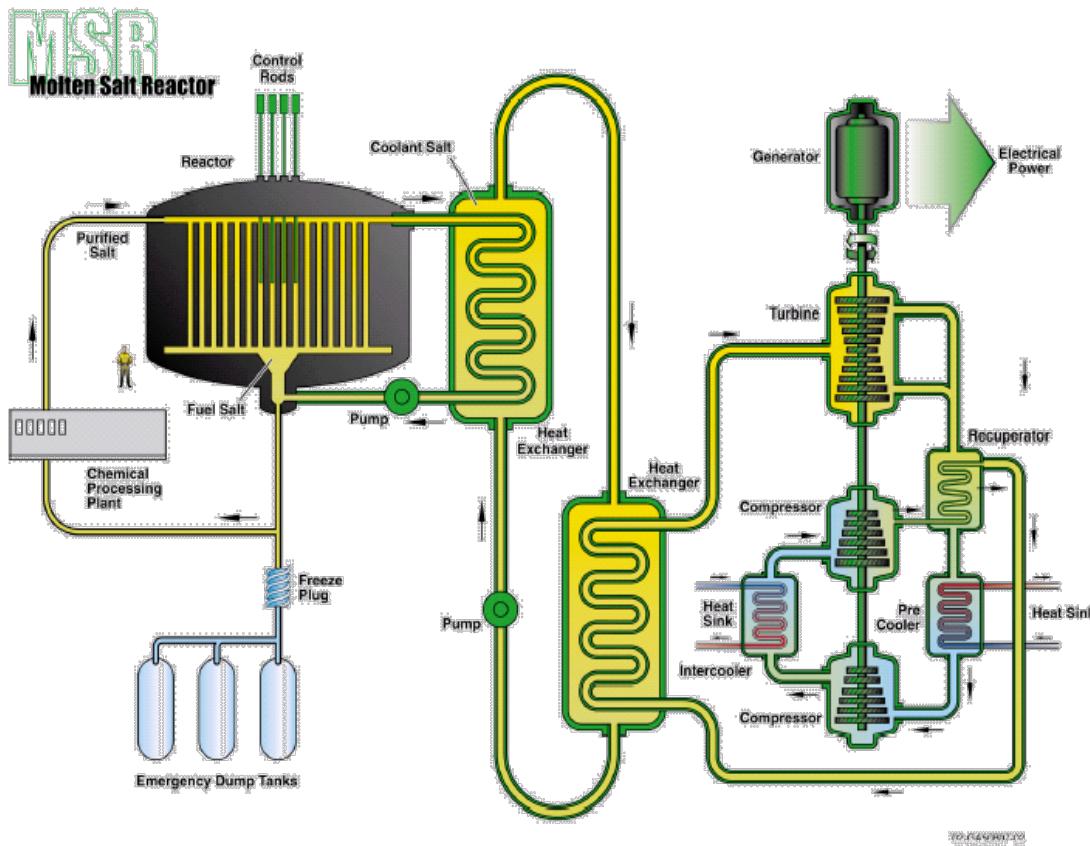
Cycle Th/U

^{137}Cs : + 30%

^{90}Sr : +300%



Molten Salt Reactor : liquid fuel, with or without moderator

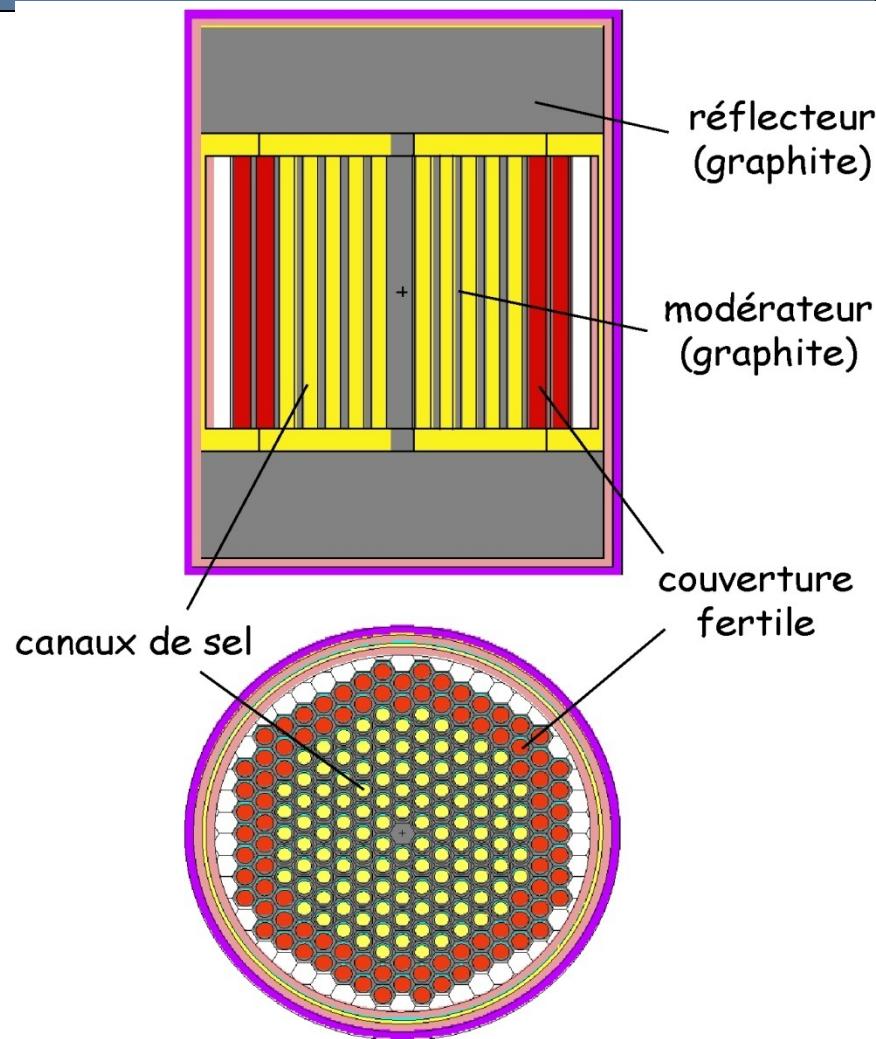


Thermal concept
Moderator = graphite
Salt : fluoride

The salt circulates in channels pierced in graphite, leaves the core, go through heat exchanger and come back to the core
A part of salt can derivated in a on-line reprocessing plant

TMSR thermal concept

- Salt: 78%LiF, 21.7%ThF₄, 0.3%UF₄
- Initial ²³³U inventory : 890 kg
- Salt volume : 20 m³
- Core radius 2.55 m
- Core height 5.35 m
- Graphite hexaone size 5 cm
- Channel radius 1.33 cm
- Temperature coefficient
 - Salt -2.4 pcm/K
 - Doppler -3.3 pcm/K
 - density +0.9 pcm/K
 - Graphite +1.6 pcm/K
 - Total -0.8 pcm/K
- Regeneration rate > 1
- He bubbling : on-line extraction of gaseous FP
- Mean time of fuel reprocessing (chemical) = 6 months



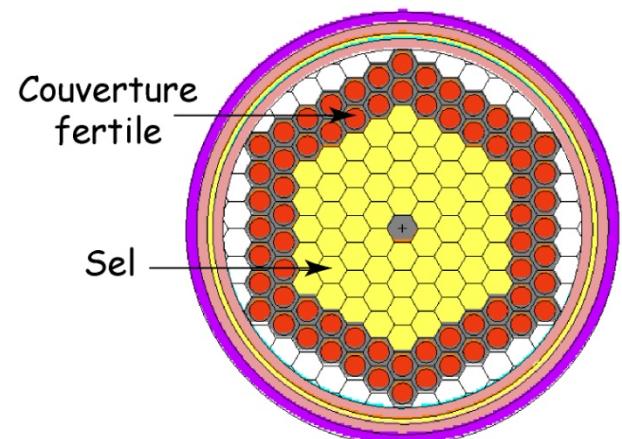
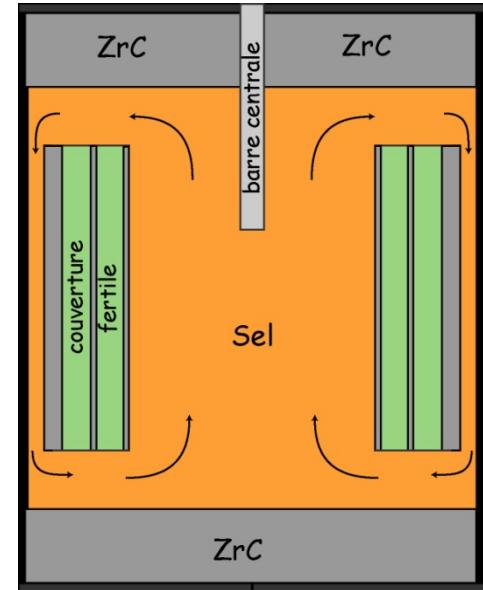
Fast Molten Salt Reactor

Fast / epithermal spectrum:

- No graphite (no problem of graphite life time)
- Neutron balance improved (FP poisoning minimized)
- use of Pu possible (very difficult in thermal spectrum, see Na)

But

^{233}U inventory increased up to 5 - 6 tons/GWe



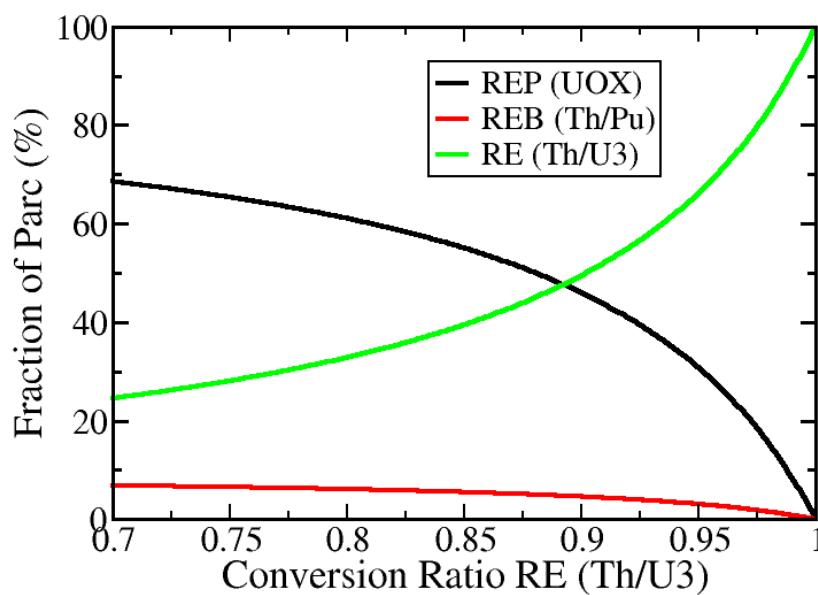
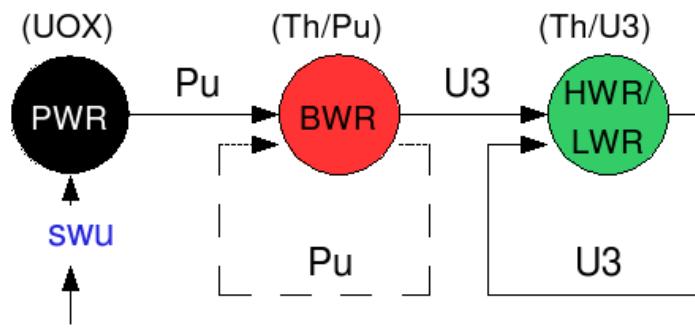
Exemple de parc mixte UOX + Th/Pu + Th/U

Interest of thorium cycle with “standard technologies” : LWR and HWR

Improve the neutronic balance, even if breeding cannot be reached

Minimize the production of minor actinides

Multi-strata scenarios



If one increases the conversion rate, one minimizes the need of natural uranium, but increases the cost of cycle (BU ↴)

Equilibrium between cycle cost and price of natural uranium

This strategy could perhaps make longer the use of water technologies and delay the use of FBR

Ideas of symbiotic fleet FR / LWR

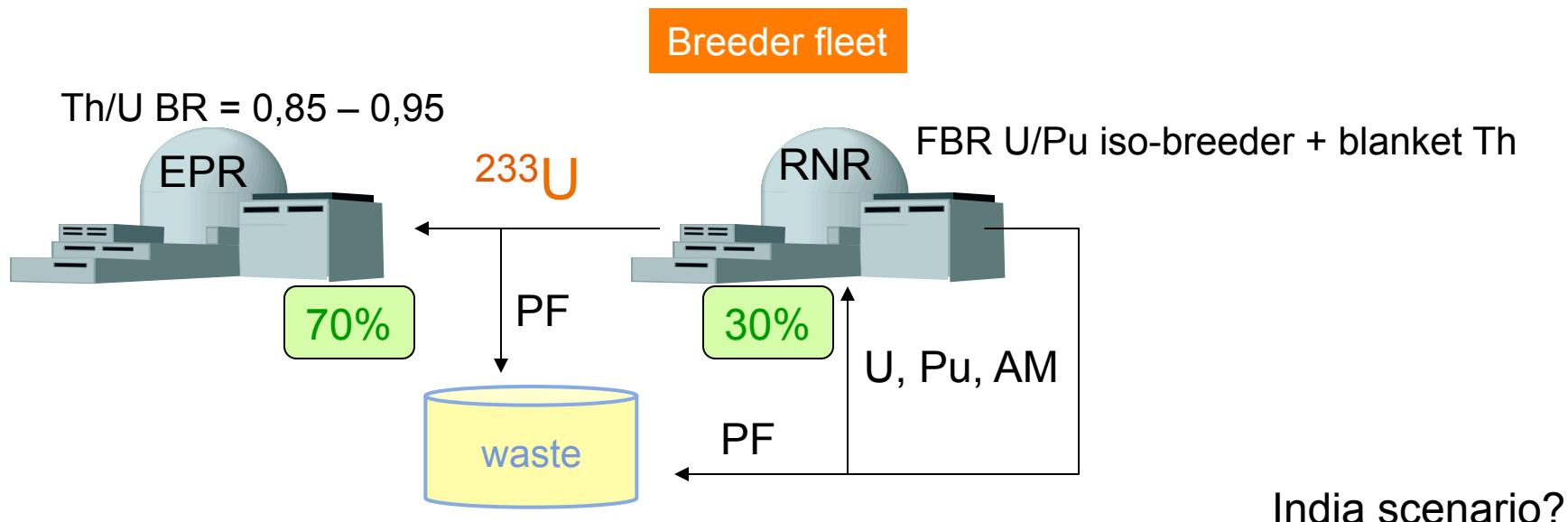
Use the good neutron balance of U/Pu in fast spectrum

And the good neutron balance of Th/U cycle in thermal spectrum

Example : 20% FR (surgeneration) + 80% LWR (no breeding)

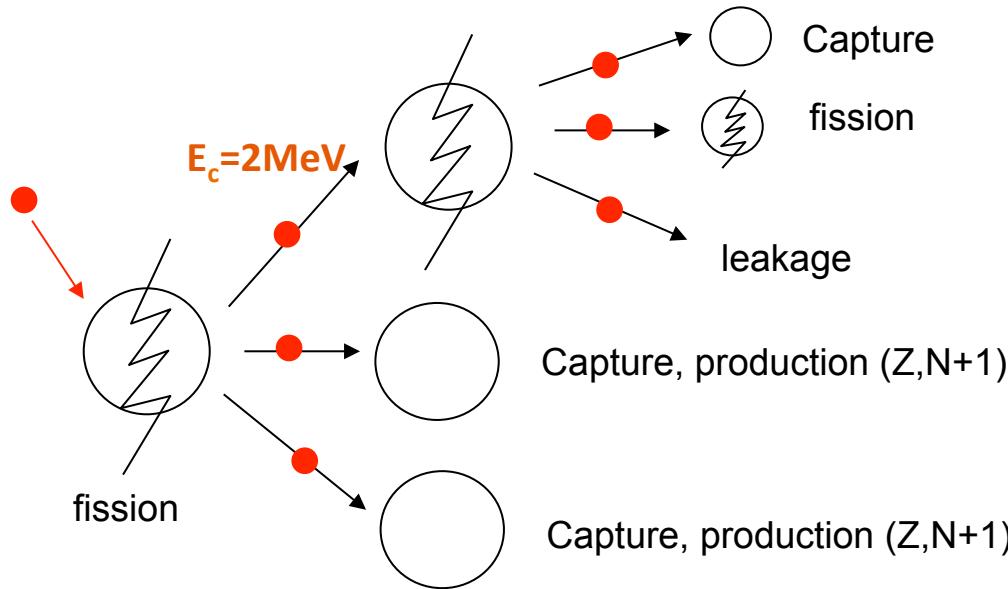
The global fleet is at equilibrium (breeding)

Advantage : minimization of waste thanks to FR as soon as possible, without forbidding a transition towards a breeding fleet using the same technologies (but not the same fuel cycle)



Back up

Fissions produce neutrons, which can induce new fissions



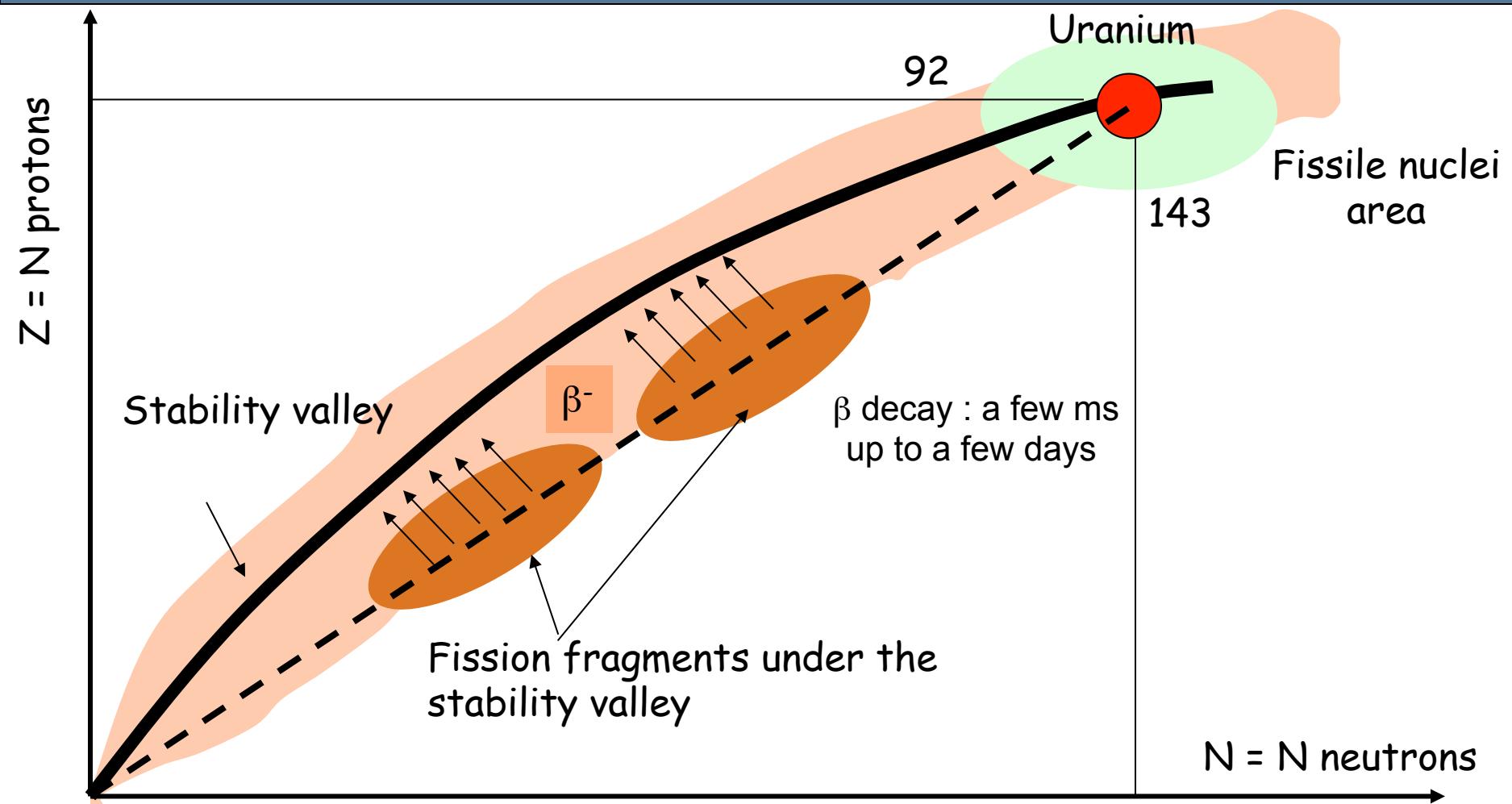
Chain reaction : 1 fission \rightarrow 1 fission \rightarrow 1 fission \rightarrow 1 fission $\rightarrow \dots$

Constant power (number of fission constant)

The chain reaction can be driven :

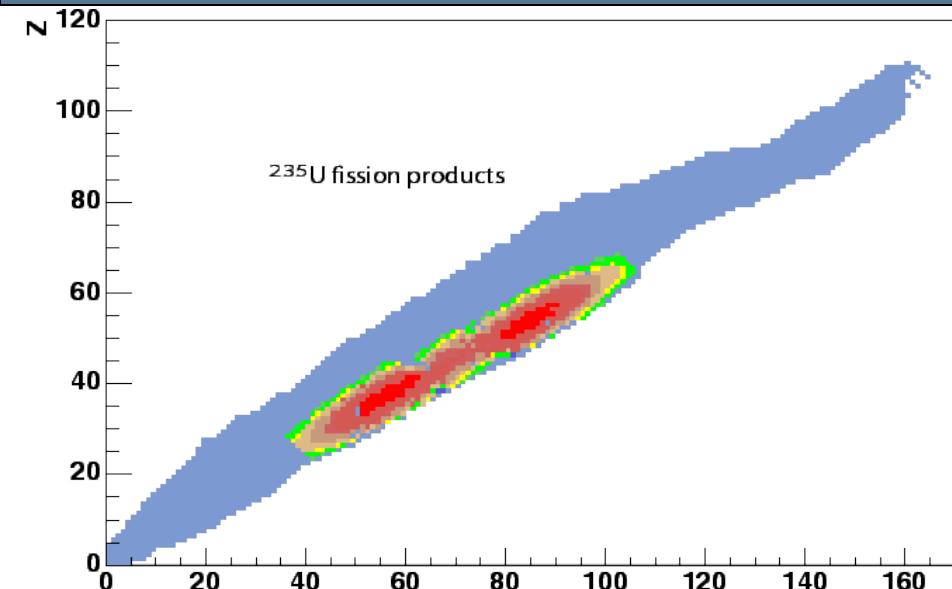
- some neutrons are produced with a delay of a few seconds after the fission : enough time to react !
- if more than 1 fission is induced by 1 fission
 - power increases → temperature increases → capture/fission probability increases → chain reaction stabilized → constant power naturally recovered

Fission products

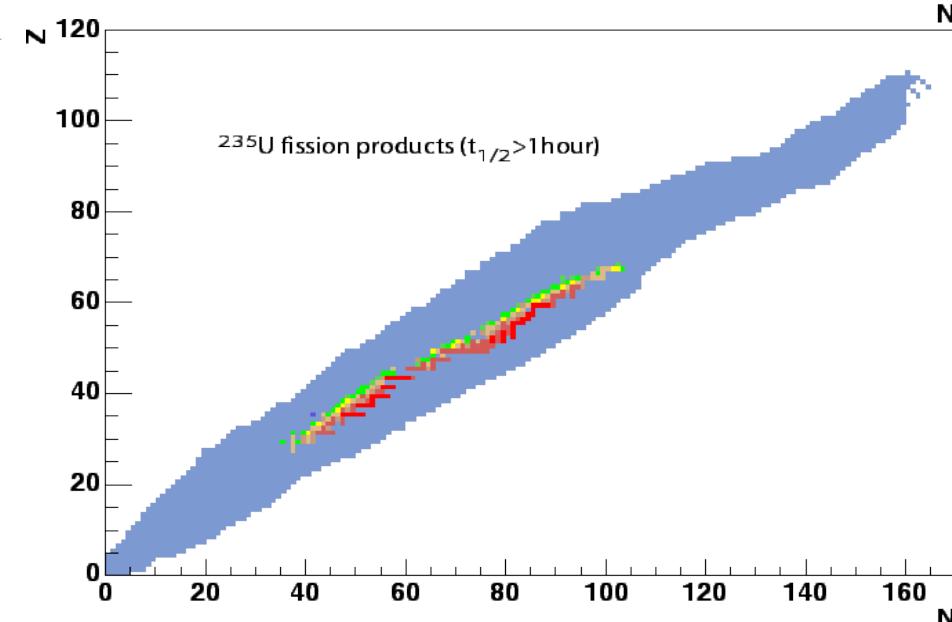


Two essential consequences

- ☺ Production of delayed neutrons after the first β decay which allow driving the chain reaction
- ☹ Residual power after the end of the chain reaction during days and months : safety issue



When the chain reaction stops,
10% of the power is still produced,
even if there is no more fissions in
the core

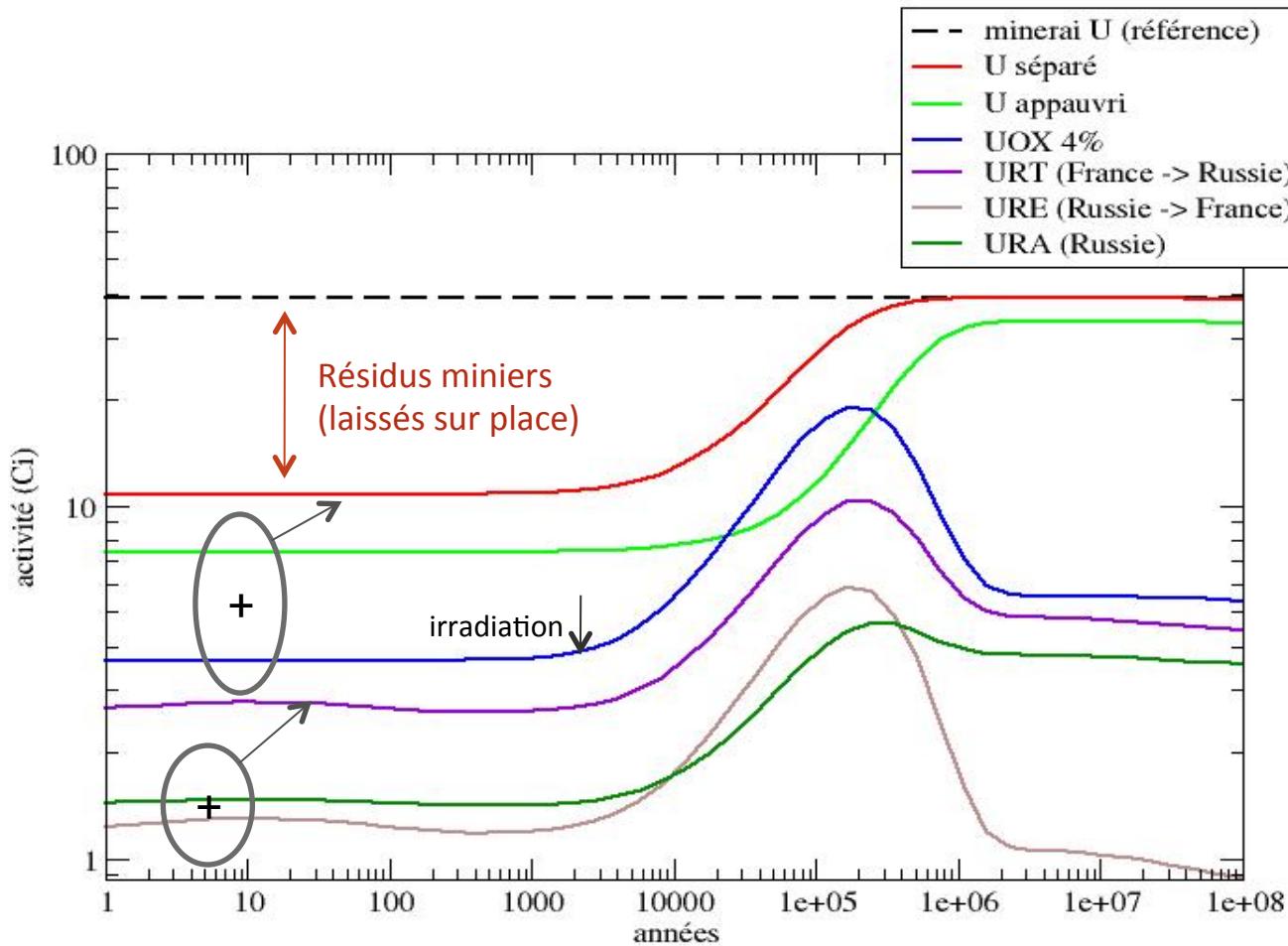


The core must be cooled down
during weeks to avoid a core
melting (TMI, Tchernobyl,
Fukushima)

Nuclear waste :
At the end of the decay chains,
sometimes, nuclei can have a long
period : a few decades up to
millions of years :
Long-term nuclear waste

Radioactivity of different uranium inventories

Normalized per ton of enriched U



Uraniums and spent-fuel

The main radioactivity of nuclear energy is contained in the spent-fuel, after irradiation, and not in the different uranium inventories

These spent fuels are transported on roads, rails, ships, ... (France / germany, japan, ...)

