



Belgian Nuclear Society Evening Lectures, Brussels, 27 March 2025

Nuclear fuel bearing minor actinides for the minimization of long-term radiotoxicity of high-level waste

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JRC directorate G: Nuclear Safety and Security

Implementing the EURATOM Research & Training Programme

Radioactive Waste Management

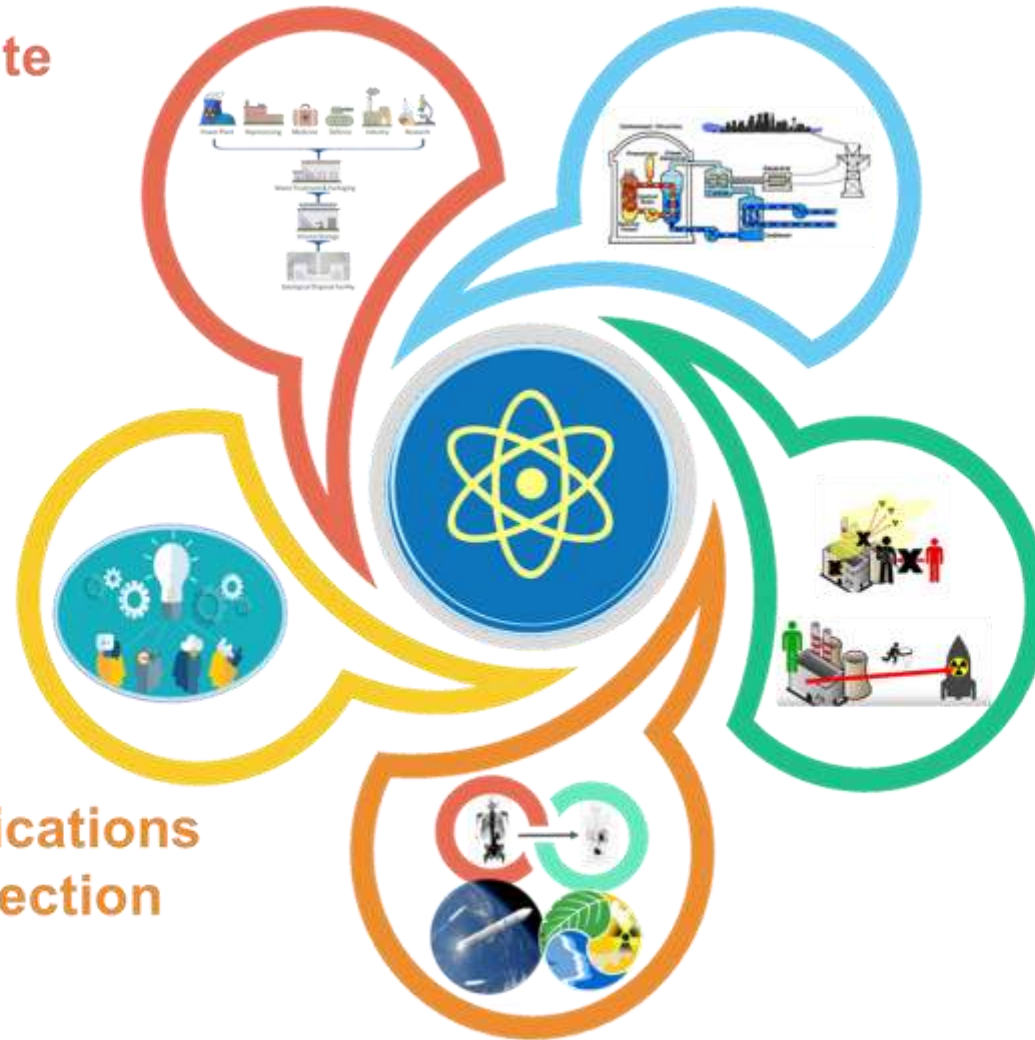
Deep Geological Disposal
Extended Interim Storage
New Waste Forms (ATF, SMR)
Regulatory framework
E&T, KM, Open Access

Nuclear Knowledge & Competence

Maintain Competence (E&T)
Human Resources Observatory
Support JRC Open Access
Reference Data & Standardization
Innovation & Technology
from Research to Industry

Non-power Applications & Radiation Protection

Medicine, Environment, Space
EU beating Cancer
Standardization
Accelerators
Open access, E&T



Nuclear Safety of Nuclear Power Plants

Nuclear reactor safety
Update of safety regulations
LTO, SMR, Gen-IV
Innovative materials
Fuel development and testing
Infrastructures: JHR, HFR and Open Access
Emergency Preparedness

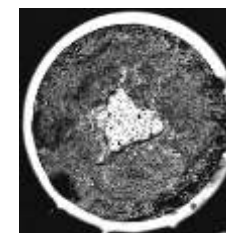
Nuclear Safeguards, non Proliferation, Security

EU and international safeguards obligations
Synergies with Security Union & Defense
Strategic trade control
Nuclear Security Training Centre

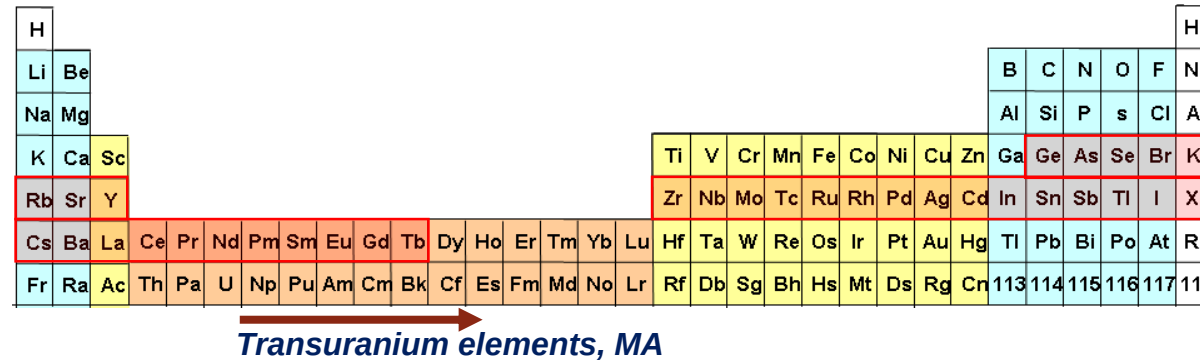
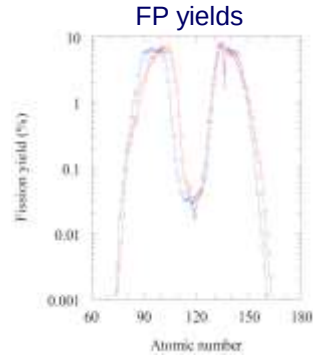


Outline

- Spent fuel, minor actinides and the nuclear fuel cycle
 - Partitioning and transmutation fuels
 - Projects and initiatives (examples)
 - Supporting research activities
 - Other types of minor actinide fuel
 - Summary and perspectives
- } *focus on oxides*



Fission Products (FP) and Minor Actinides (MA)



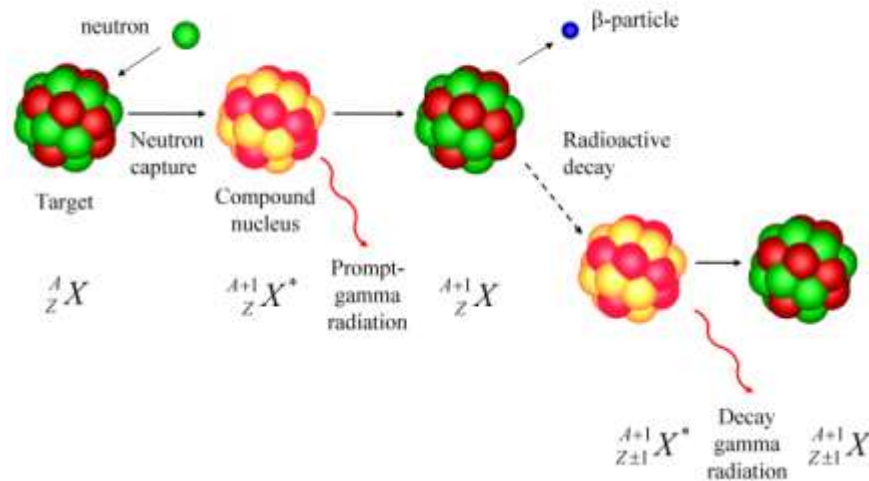
FP Yields

Rare Earths, Y: 24%
 Tc, Ru, Rh, Pd: 16%
 Kr, Xe: 15%
 Zr, Nb: 14%
 Mo: 13%
 Rb, Te, I, Cs: 11%
 Sr, Ba: 7%

In addition to products, neutron capture by actinides followed by beta-decay events causes the formation of heavier elements

Fissile Pu and the Minor Actinides (MA) Np, Am, Cm are formed

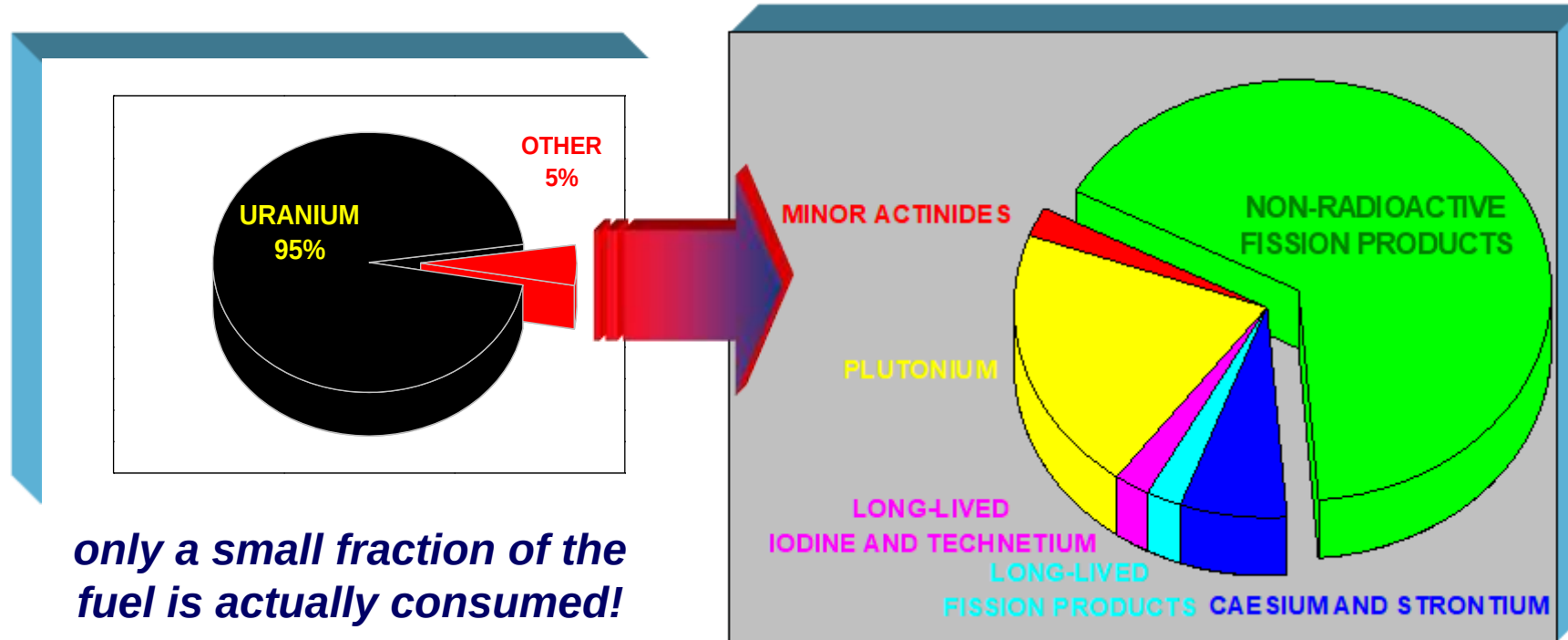
MA contribute significantly to the long term radiotoxicity of spent fuel or high level waste



MA	Nuclide $t_{1/2}, y$	
	${}^{239}\text{Pu}$	$2.4 \cdot 10^4$
	${}^{237}\text{Np}$	$2.1 \cdot 10^6$
	${}^{241}\text{Am}$	433
	${}^{243}\text{Am}$	7370
	${}^{244}\text{Cm}$	18.1

Spent Nuclear Fuel

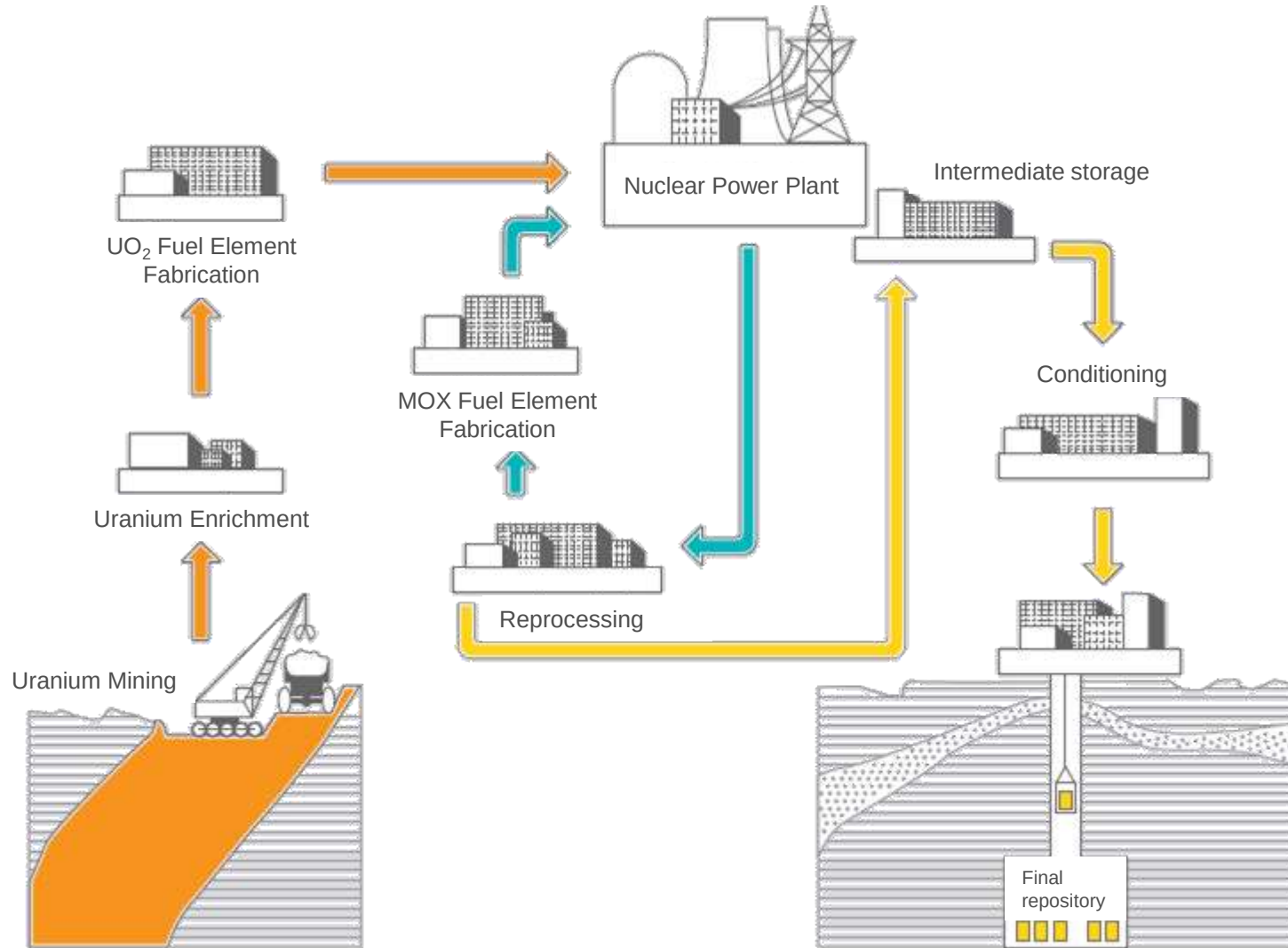
Waste (?) from nuclear energy production



After e.g. 3 y of irradiation in a LWR, 1000 kg of U (3.25% ^{235}U) have produced:

- ~ 36 kg fission products (FP)
- ~ 10 kg Pu
- ~ 0.8 kg minor actinides (MA)

The Nuclear Fuel Cycle Today



Currently two disposal options

- Direct disposal
- Reprocessing of Pu into MOX



Open or (partially) closed cycle?

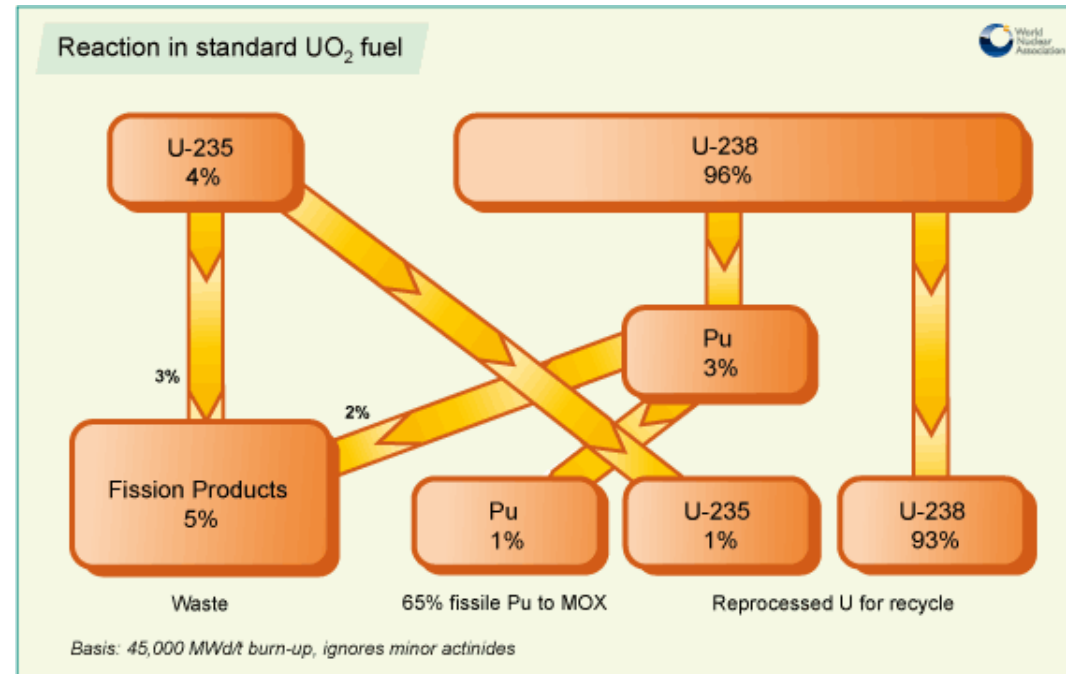
*What to do with the
~300000 t of spent fuel produced by nuclear power for civilian use?*

Direct disposal

- Fuel cycle back-end simplified
- Reduced proliferation risks
- Fuel matrix acts as containment for radionuclides



Geological repository



Reprocessing

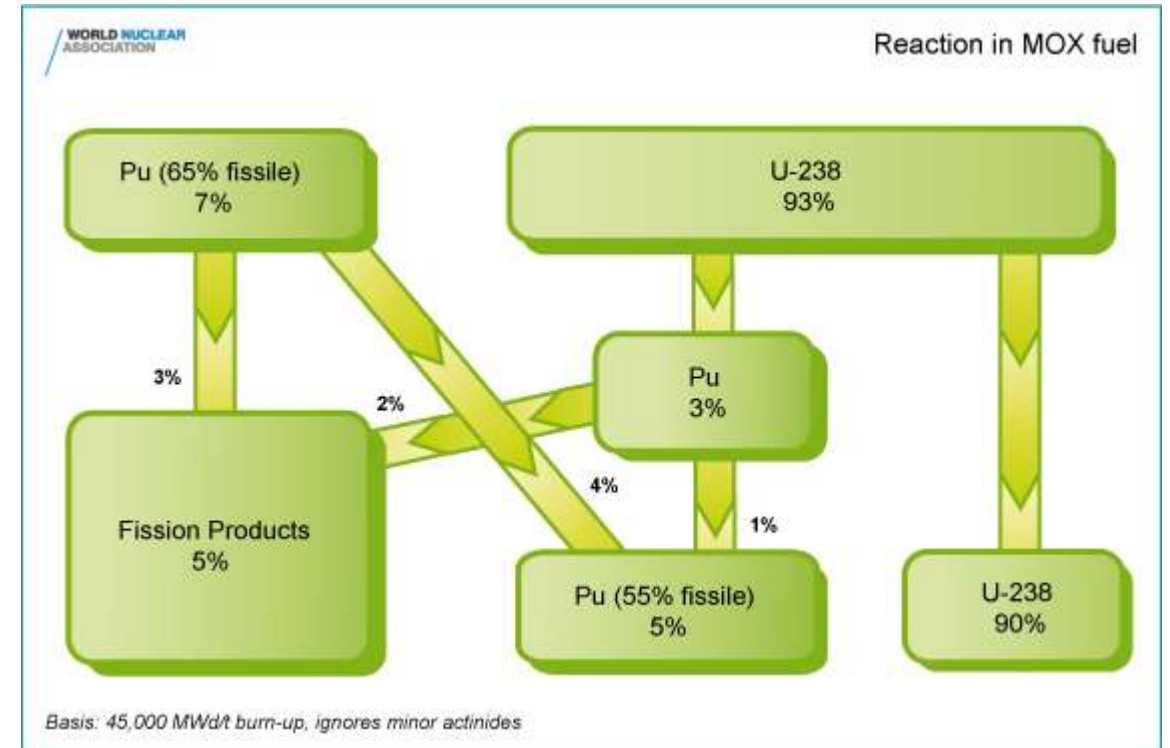
- Pu single recycling: gaining 25-30% energy; improved *sustainability*
- Reduction of high-level waste volume

Multiple Pu recycling

Typically ~1% of spent UO_2 fuel is Pu, and 2/3 of Pu is fissile (50% Pu-239, 15% Pu-241).

Fissile fraction decreases with further irradiation cycles (37% Pu-239, 16% Pu-241 in MOX with 45 GWd/t).

Current LWR allow only single recycling of Pu. Multi-recycling fuel configurations are under development.



Pu-241 ($t_{1/2} = 14$ years) decays to Am-241, strong gamma emitter, whose level in stored Pu increases ~0.5%/y. Therefore it is not convenient to store Pu for long time.

Fast neutrons are well suited for multiple recycling of plutonium.

Closing the fuel cycle

What to do with the >300000 t of spent fuel produced by nuclear power for civilian use?

Closed cycle advantages

Partitioning & Transmutation (P&T):

MA (mainly Am, Np) and LLFP extracted from spent fuel, then destroyed in FR or in “burners” (e.g. ADS)

Multi-recycling in fast spectrum

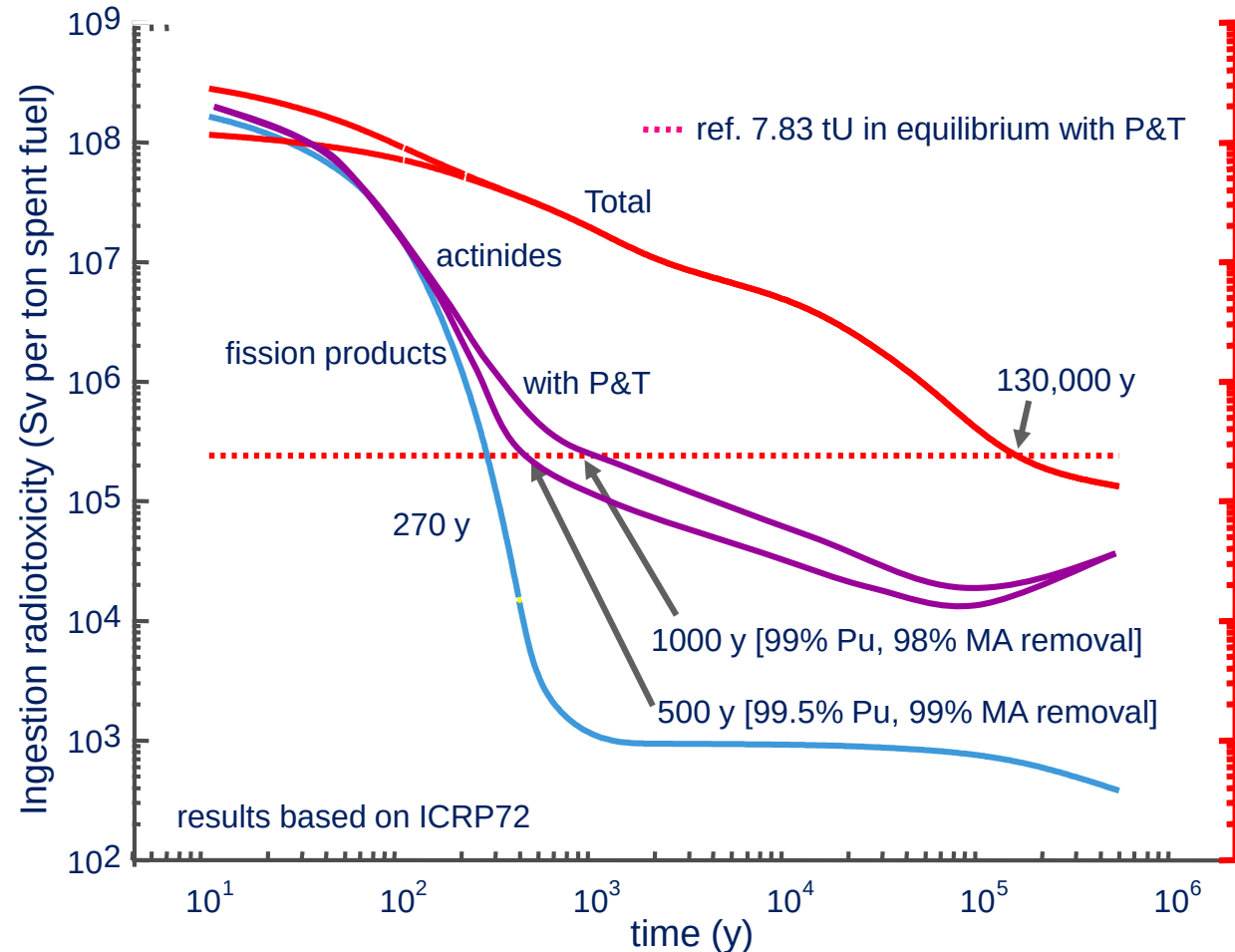
→ *minimize long term radiotoxicity of High Level Waste (HLW)*

(Breeding → *full sustainability*)

LLFP = long-lived fission products

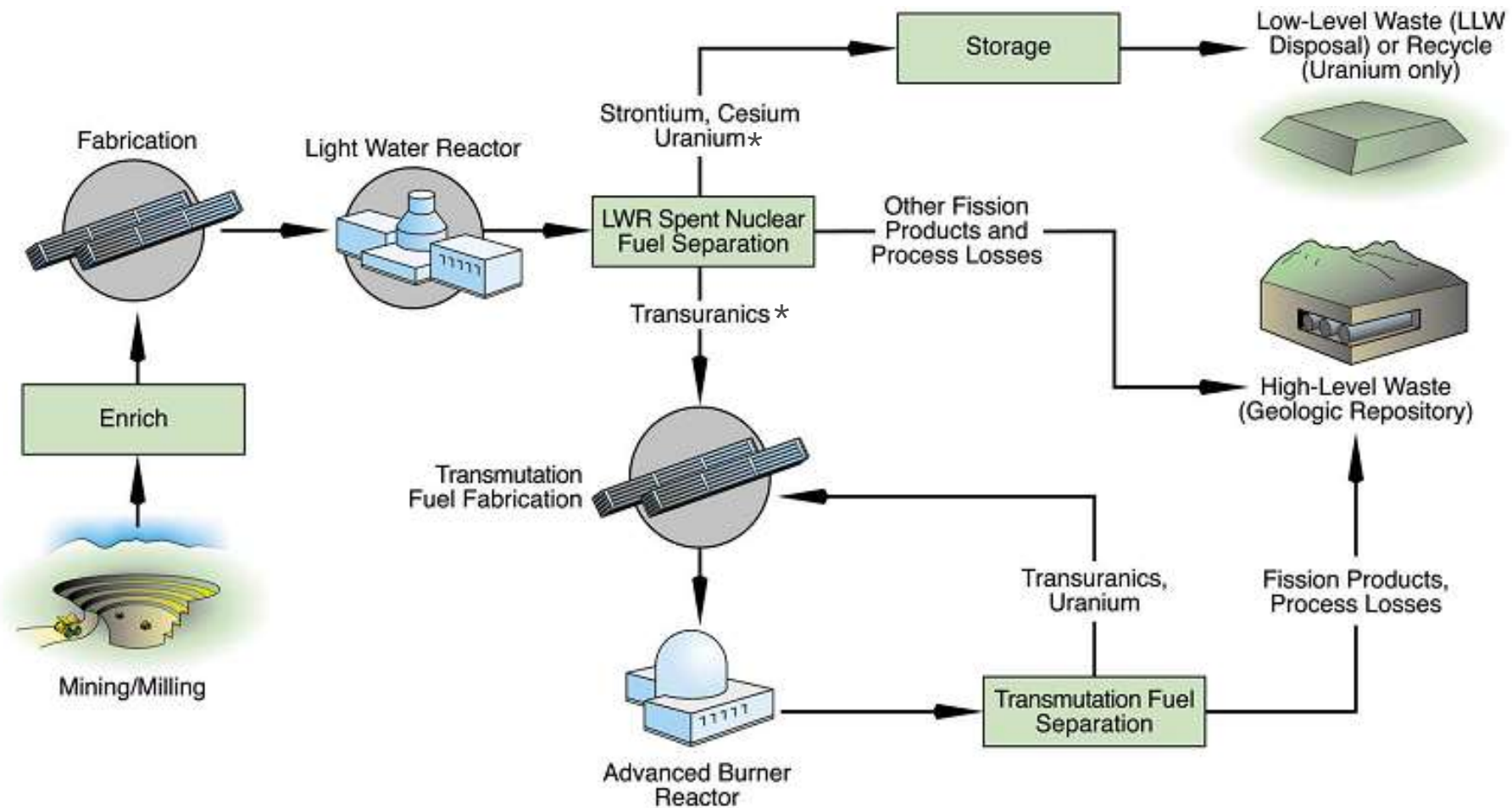
FR = fast reactor

ADS = accelerator driven system



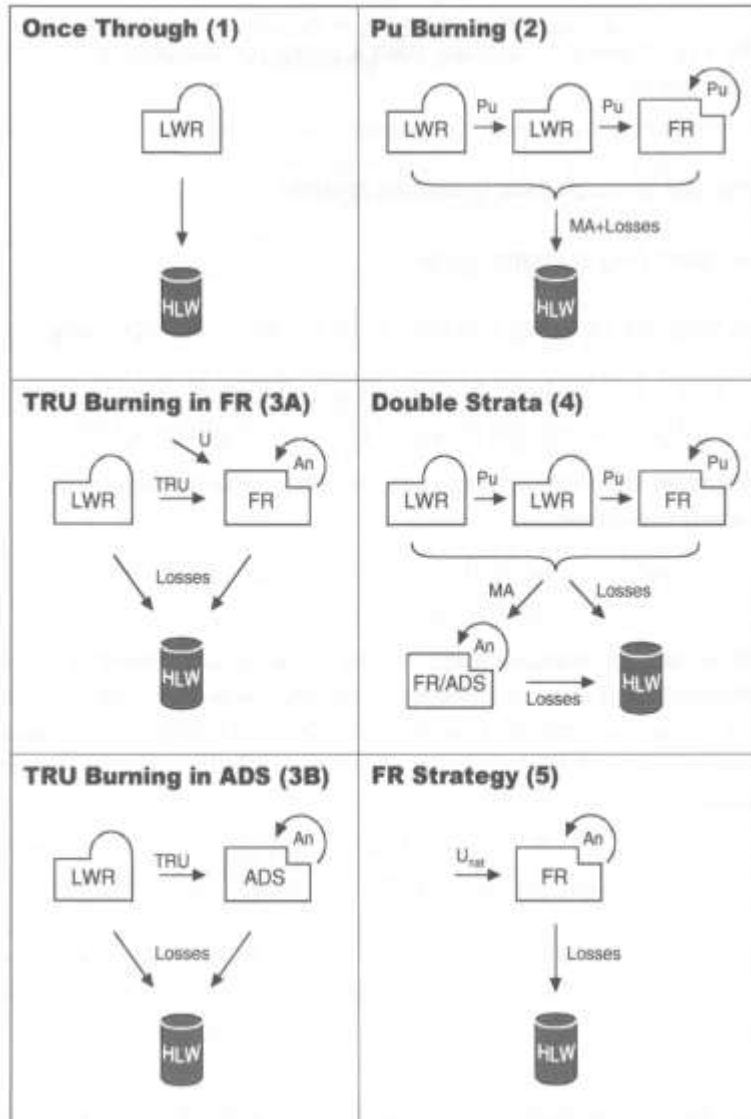
<http://www.nucleonica.net>

The Closed Nuclear Fuel Cycle (example)



*Uranium could join the transuranics

Closed fuel cycle for sustainability and waste radiotoxicity minimization



Various options have been proposed to integrate or replace once-through cycle

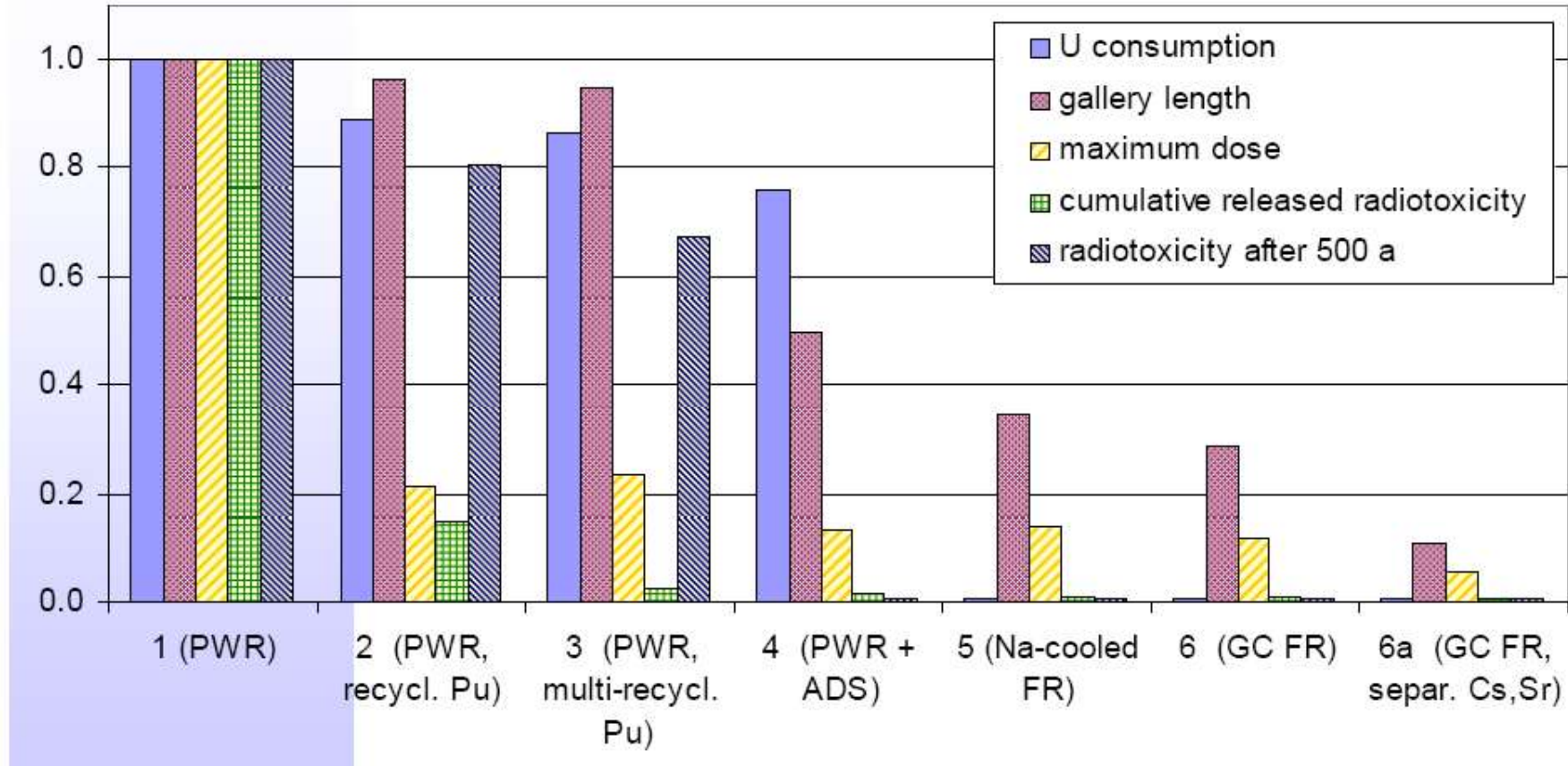
- integration or replacement of thermal with fast neutron systems

- “normal” vs. dedicated cores (burners): single vs. double strata, Accelerator Driven Systems (ADS)

LWR = light water reactor
ADS = accelerator driven system
FR = fast reactor

Impact of different fuel cycles on repository

data normalized to PWR spent fuel (open cycle); **REDIMPACT** project



PWR = pressurized water reactor

ADS = accelerator driven system

FR = fast reactor

GC = gas cooled

Partitioning and transmutation fuels

Minor actinides cross sections in LWR, FR

Comparison of thermal and fast neutron spectrum

	PWR UOX			FR (EFR)		
Isotope	σ_f	σ_c	$\alpha = \sigma_c / \sigma_f$	σ_f	σ_c	$\alpha = \sigma_c / \sigma_f$
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
²⁴² Cm	1.14	4.5	3.9	0.58	1.0	1.7
²⁴³ Cm	88	14	0.16	7.2	1.0	0.14
²⁴⁴ Cm	1.0	16	16	0.42	0.6	1.4
²⁴⁵ Cm	116	17	0.15	5.1	0.9	0.18
⁹⁹ Tc	/	9	/	/	0.5	/

α is the cross section ratio capture/fission: lower is better!

→ recycling minor actinides in LWR is not effective

Advanced reprocessing and partitioning

All partially or fully closed cycles include a step in which **spent fuel is dissolved and separated, or partitioned, in its main components**

today's reprocessing separates U and Pu only

advanced (closed) cycles envisage partitioning fissile and fertile fractions, MA (mainly Am, Np) and possibly also Long-Lived Fission Products (LLFP), e.g. ^{135}Cs , ^{129}I , ^{99}Tc ,...

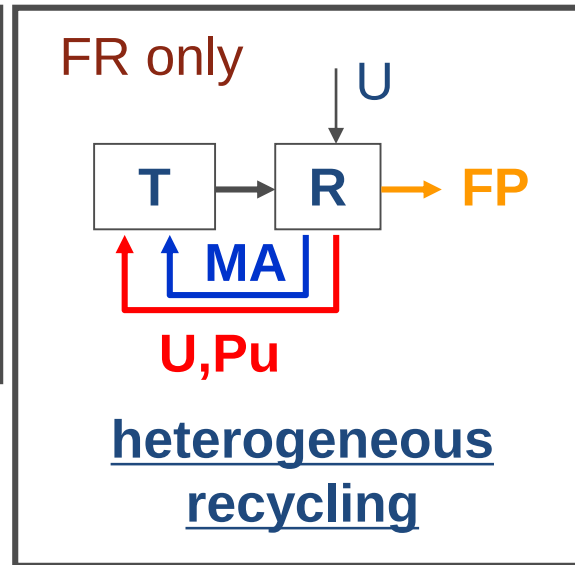
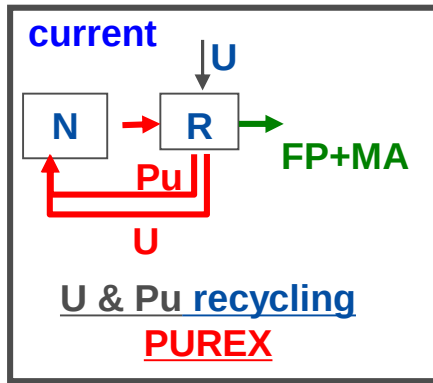
many options and variants: once SF is dissolved, groups of elements could be separated together; **co-extraction and grouping** of fissile material together with other species would also limit proliferation risks

separated elements are subjected to *ad-hoc* treatments to

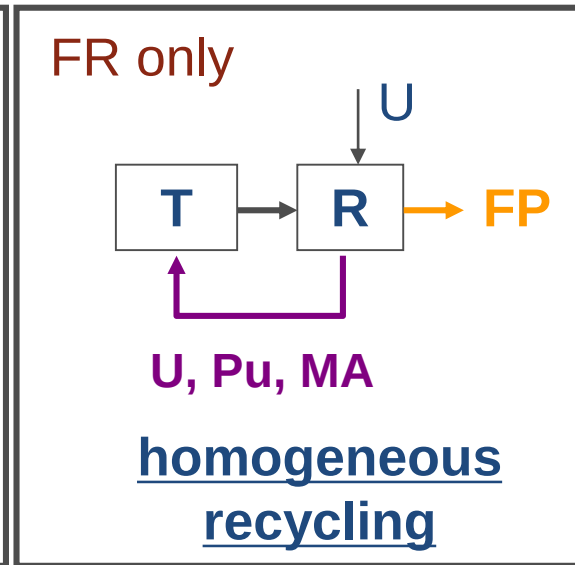
- residual fission products: immobilize in **waste forms** in view of disposal
- **MA (and LLFP): incorporate into fuel** for transmutation (burning).

fabrication lines involving MA are heavily shielded and safeguarded

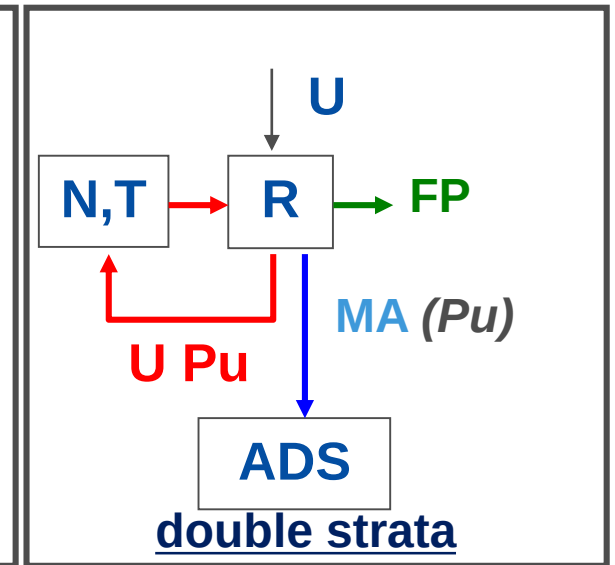
Recycling schemes



(U,MA)O_{2-x} or Inert Matrix
MA only in blanket (MABB)
Compatible with PUREX
Specific MA fabrication line
 $x_{MA} = 10\% - 20\%$



(U,Pu,MA)O_{2-x}
MA in whole core (MABD)
Impact on fuel fabrication
 $x_{MA} < 5\%^*$



ADS: first core potentially
(U,Pu)-based

R: Reprocessing plant

N: conventional reactor

T: Transmutation reactor

ADS: Accelerator Driven System (sub-critical)

Inert Matrix: MA (eventually Pu) incorporated in a non-fertile matrix to ensure no new MA are formed

Cm not envisaged for P&T

- high radioactivity, n-emission – difficult handling
- relatively short half-life
- relatively low transmutation efficiency

Transmutation fuels

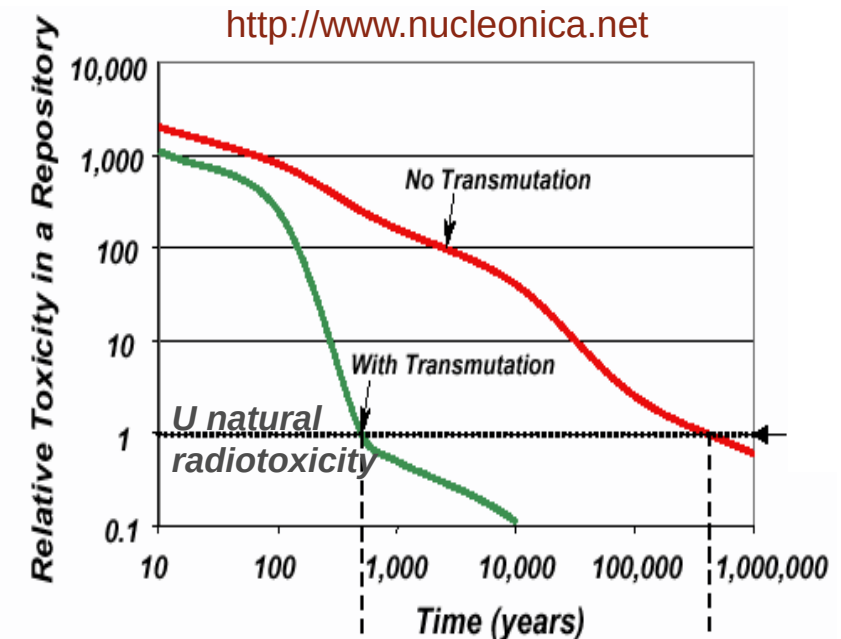
Fabrication routes affected by radioactivity of MA → powderless routes in remotely operated facilities are envisaged

He generation by alpha-decay may cause additional swelling during irradiation

Proposed transmutation fuels

- Oxides (U,Pu,MA)O₂ 
- Metallic (U,Zr,Pu,MA) 
- Carbides (U,Pu,MA)C 
- Nitrides (Zr,Pu,MA)N 
- Molten salts (fluorides, chlorides)

	nuclide	$t_{1/2}, y$
MA	²³⁹ Pu	$2.4 \cdot 10^4$
	²³⁷ Np	$2.1 \cdot 10^6$
	²⁴¹ Am	433
	²⁴³ Am	7370
	²⁴⁴ Cm	18.1
	⁹⁹ Tc	$2.1 \cdot 10^5$
	¹²⁹ I	$1.6 \cdot 10^7$
	¹³⁵ Cs	$2 \cdot 10^6$



LLFP = long-lived fission products

Transmutation fuel - basic properties

	Metal (U,Pu)Zr	Oxide (U,Pu)O₂	Nitride (U,Pu)N	Carbide (U,Pu)C
Heavy metal density (g cm ⁻³)	14.0	9.7	13.5	12.9
Melting point (°C)	1070	2730	2720	2320
Thermal conductivity λ (W m ⁻¹ K ⁻¹)	35 ^b	2.9 ^c	19.8 ^c	19.6 ^c
T centreline (°C)	700	2000	1500	1500
C_p (J g ⁻¹ K ⁻¹)	17	34	26	26
Thermal expansion α (10 ⁻⁶ K ⁻¹)	16.5	12.6	10.0	12.4
Fuel swelling (vol.% at.% ⁻¹)	~10	0.6-0.7	1.1	1.6

^a JNM, 441, 2013 ^b at 500 °C ^c at 1000 °C

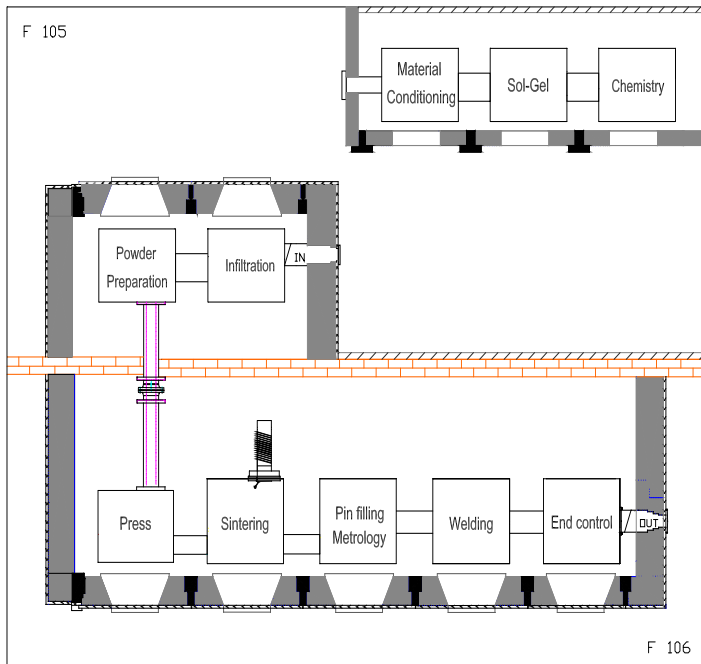
Synthesis of minor actinide fuel

Isotope	Limiting mass (g) ^a	Criterion
²³¹ Pa	10	Shielding
²³⁷ Np	- ^b	
²⁴¹ Am	50	License ^c
^{242m} Am	0.1	Shielding
²⁴³ Am	65	License ^c
²⁴⁴ Cm	5	Shielding & licence ^c

^a to yield max 2 μSv/h at 1 metre

^b no practical limit

^c corresponding to the dosis equivalent of 200 g Pu in the form of powder (oxide)



Minor Actinide laboratory (MA-lab) at JRC



wet routes
minimising dust and waste



Projects and initiatives (examples)

European partitioning projects

FP3
1990-1994

HLLW partitioning by means of completely incinerable extractants

FP 4
1994-1998

New partitioning techniques for minor actinides NEWPART

FP5
1998-2002

Pyrometallurgical reprocessing PYROREP; New Solvent Extraction Processes for minor actinides PARTNEW

FP6
2002-2006

Partitioning of MA from high active wastes EUROPART

FP7
2007-2013

*Actinide recycling by separation and transmutation ACSEPT;
Safety of ACTinide Separation proceSSes SACSESS*

FP8
2014-2020

GEN IV Integrated Oxide fuels recycling strategies GENIORS

FP9
2021-2027

*MultI-recycling strategies of LWR SNF focusing on MOlten SAlT technology
MIMOSA*



MIMOSA



European
Commission

Research projects and initiatives on P&T

Oxides irradiation

	Irradiation test	Irradiation date	Participants	Reactor	Pu, MA content	Burn-up
(U,Pu,MA)O ₂ Fuels	SUPERFACT	80's	CEA/JRC	PHENIX	24% Pu 2% Am 2% Np	~7 at%
	AM1	2008	JAEA	JOYO	29% Pu 2-5%Am 0-5%Np	10 min & 24h
	AFC-2C&2D	2008-2010	US DOE-INL	ATR	20% Pu 3% Am 2% Np	10-24 at%
	SPHERE	2013-2015	FAIRFUELS PELGRIMM	HFR	21% Pu 3% Am	5 at%
(U,MA)O ₂ fuels	SUPERFACT	80's	CEA/JRC	PHENIX	20%Am 20%Np	~6 at%
	MARIOS	2011-2012	FAIRFUELS PELGRIMM	HFR	15%Am	~1- 1.5 at%
	DIAMINO	2014-2015	CEA	OSIRIS	7.5%- 15%Am	~1 - 1.8 at%
	MARINE	2016-2017	PELGRIMM*	HFR	13%Am	~5 at%

*Gallais-During *et al.*, JNM 512 (2018) 214-226

Euratom projects

Framework Programs 6,7; Horizon 2020; Horizon EU



Addressing relevant aspects of MA recycling



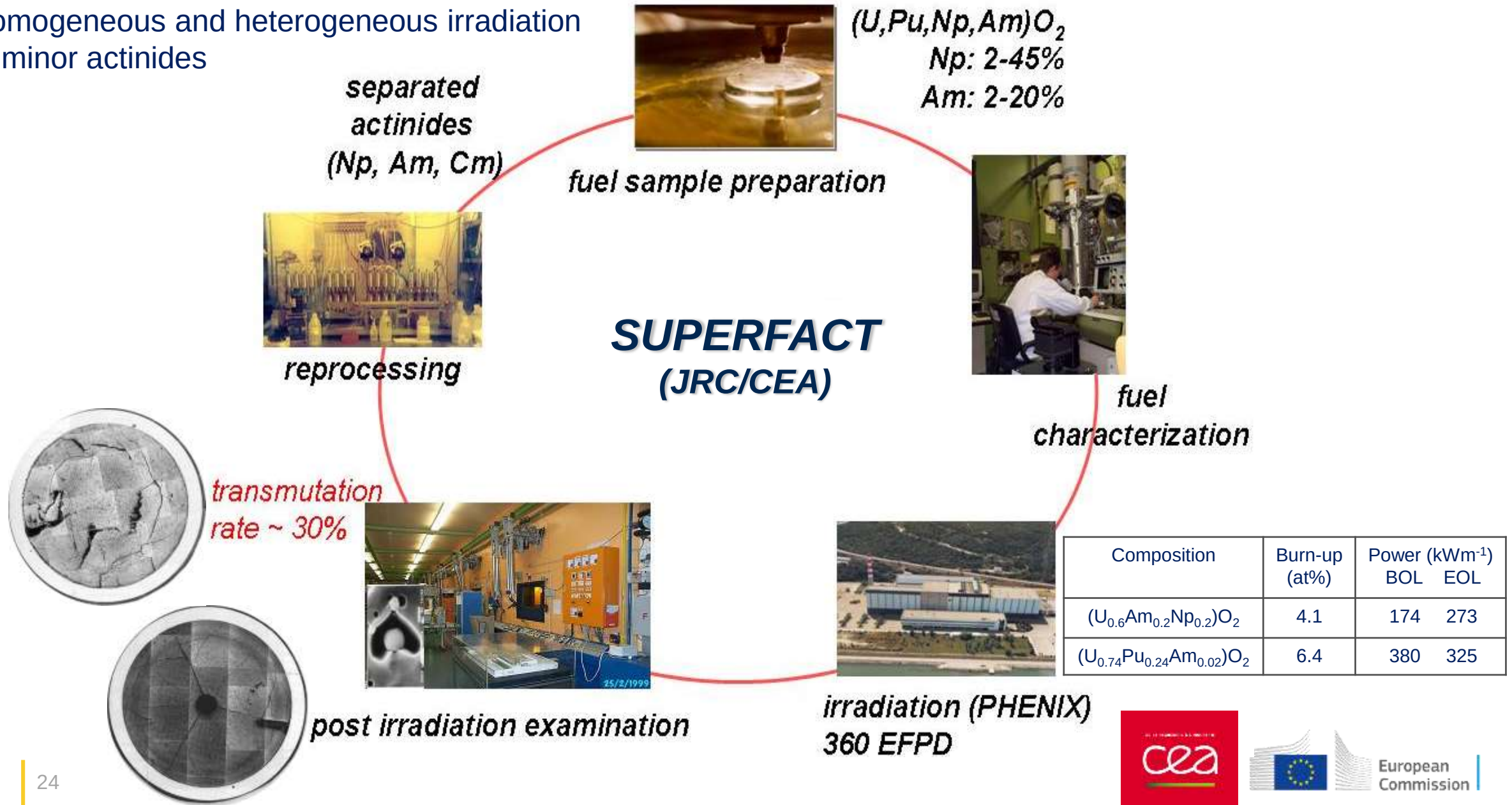
★ MYRRHA-related

Transmutation Projects (not only oxides)

Fuel	Project	Fuel composition	Reactor	Irradiation	Burn-up
Oxide	SUPERFACT	$(U_{0.74}Pu_{0.24}Np_{0.02})O_2$ $(U_{0.74}Pu_{0.24}Am_{0.02})O_2$ $(U_{0.55}Np_{0.45})O_2$ $(U_{0.6}Am_{0.2}Np_{0.2})O_2$	Phénix	1986 – 1988	6.4 %FIHMA 6.4 %FIHMA 4.5 %FIHMA 4.1 %FIHMA
	SPHERE	$(U_{0.76}Pu_{0.21}Am_{0.03})O_2$	HFR	2013 - 2014	~ 5 %FIHMA
	MARIOS	$(U_{0.85}Am_{0.15})O_{1.94}$	HFR	2011 - 2012	~ 1.5 %FIHMA
	MARINE	$(U_{0.87}Am_{0.13})O_2$	HFR	2016 - 2017	1.57 & 1.98 %FIHMA
IMF	FUTURIX	$Pu_{0.8}Am_{0.2}O_{2-x}$ - 86% Mo $(Pu_{0.23}Am_{0.24}Zr_{0.53})O_{2-x}$ - 60% Mo	Phénix	2007 – 2009	17.8 %FIHMA 12.9 %FIHMA
	HELIOS	$(AmZr,Y)O_2$ $(Am,Pu,Zr,Y)O_2$	HFR	2010 – 2011	3.50 %FIHMA 13.0 %FIHMA
	EFTRRA-T3	Diverse IM + UO_2	HFR		17 – 20 %FIHMA
	EFTRRA-T4, T4bis, T4ter	$MgAl_2O_4$ - $AmAlO_3$	HFR	1996 – 1998	96 %Am, 99.8 %Am
	CAMIX	$(Zr,Y,Am)O_2$	Phénix	- 2009	~ 25 at%
	COCHIX	MgO - $(Zr,Y,Am)O_2$	Phénix	- 2009	~ 25 at%
	METAPHIX	U-Pu-Zr 2% MA - 2% RE U-Pu-Zr 5% MA U-Pu-Zr 5% MA – 5% RE	Phénix	2003 – 2004	2.5 at%, 7at%, 10 at%

R&D demonstration of closed cycle

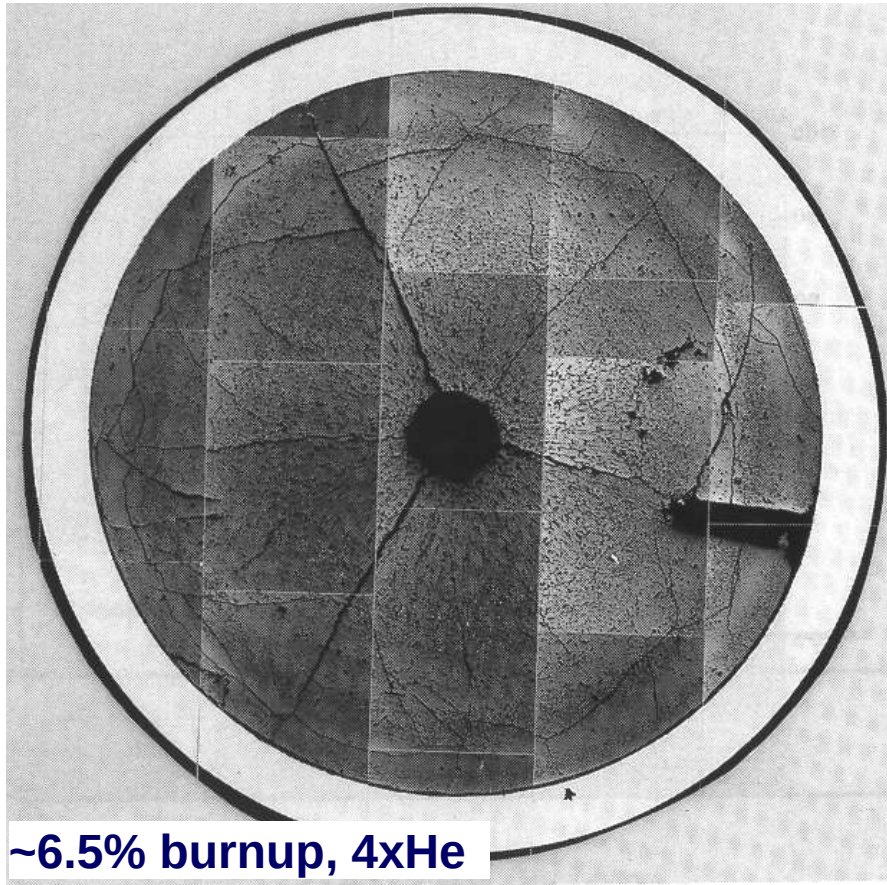
Homogeneous and heterogeneous irradiation of minor actinides



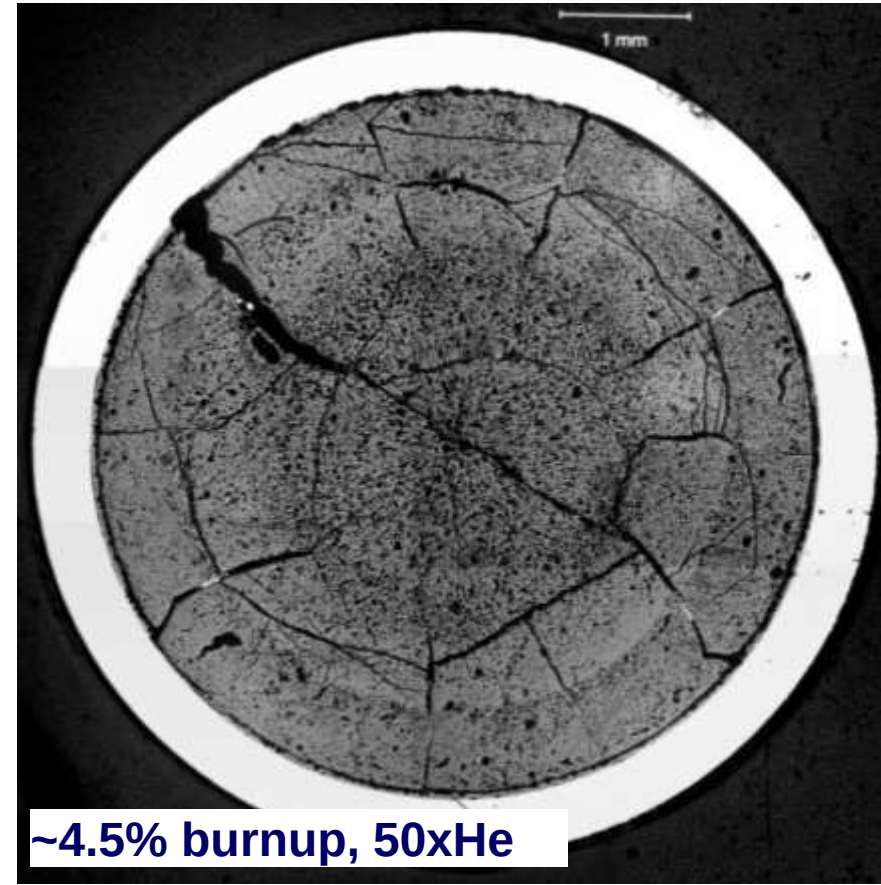
SUPERFACT

SUPERFACT irradiation - fast reactor MOX containing MA - ~30% transmutation
(Phenix reactor, collaboration with CEA)

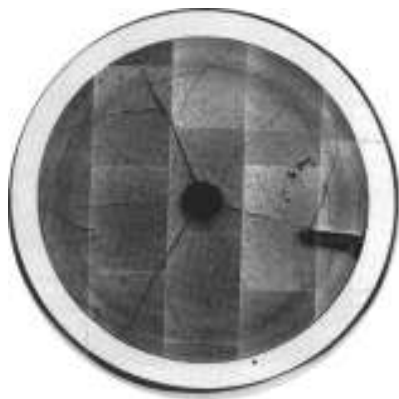
$\text{MOX } (\text{U}_{0.74}\text{Pu}_{0.24}\text{Am}_{0.02})\text{O}_{1.973}$
homogeneous recycling, higher power rating



$(\text{U}_{0.60}\text{Np}_{0.20}\text{Am}_{0.20})\text{O}_2$
heterogeneous recycling, lower power rating



SUPERFACT



Homogeneous recycling

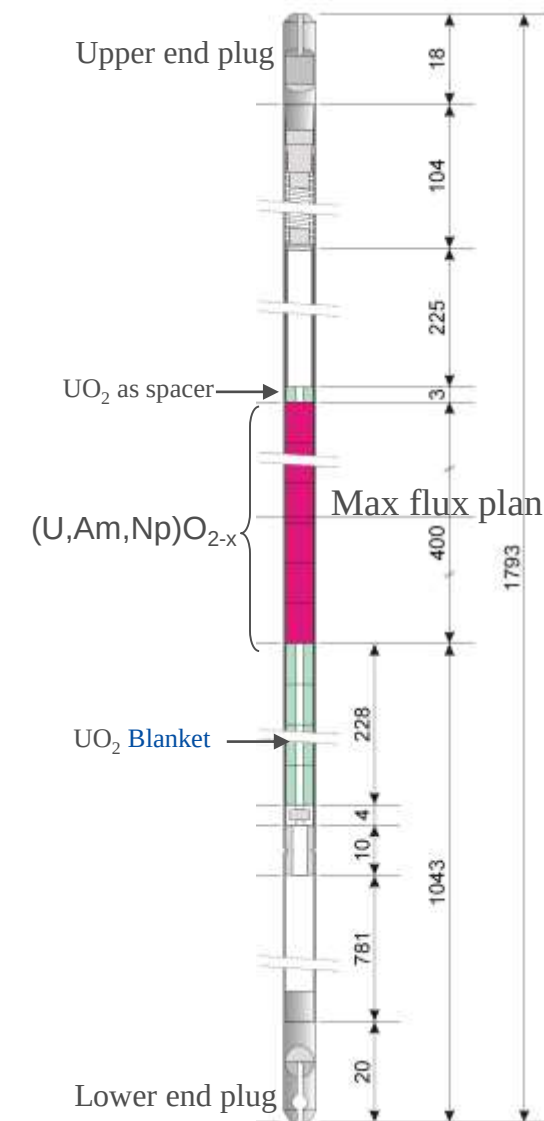
- Restructuring and clad corrosion similar to standard Phénix pins
- He 4 times higher than standard Phénix pins
- In SF16 about 90% of the burn-up resulted from the fission of Pu



Heterogeneous recycling

- Transmutation: 31% Am, 27% Np
- He release: 100% (He produced during storage: 50% release)
- Swelling: axial 2.3%, radial 3.3%
- FCMI, ~60μm layer
- He production: 225 cm³ / 100 g fuel vs MOX: 10 cm³/ 205 g fuel

FG release ~60% as predicted
Reprocessing by aqueous method was successful.



Some conclusions from irradiation campaigns and PIE

- **SUPERFACT-1:** the behaviour of homogeneous recycling fuels (MADF) is similar to FR fuel, whereas heterogenous fuel (MABB) generates high amount of He
- More complex phase transitions with actinides addition
- Under vacuum, ceramic samples exhibit Pu and Am losses starting in the temperature range 1900–2100 K, with a rate sharply increasing with temperature. Irradiation temperature should be kept below 1900 K
- **Transmutation feasibility of Am-241 and Np-237 in fast reactor demonstrated. Irradiated fuel was reprocessed successfully → multi-recycling possible.**
- The **MABB concept has reached a key milestone with the separate-effect irradiation MARIOS ($\text{Am}_{0.15}\text{U}_{0.85}\text{O}_{1.94}$)**, which has shown **no significant swelling** in the range of ~1000C-1300C.
- **FUTURIX** showed that the high thermal conductivity of CERMETS leads to lower fuel operating temperature.
- A significant difference between pellet and spherepac fuels lies in the presence of FCMI for the pelletized pin, which seems **absent for the spherepac pin (PELGRIMM)**.



Supporting R&D

Neutron data for P&T



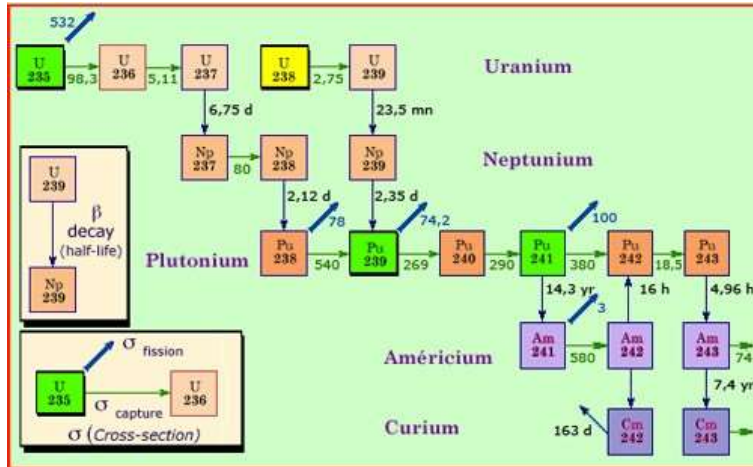
Minor actinides and fission products

Tc ,I, Am, Np, Cm → cross sections and evaluations

Systems

FR → inelastic cross sections of fuel, coolant and structural materials, e.g. U, Na, Pb, Fe,...

ADS → Pb, Bi



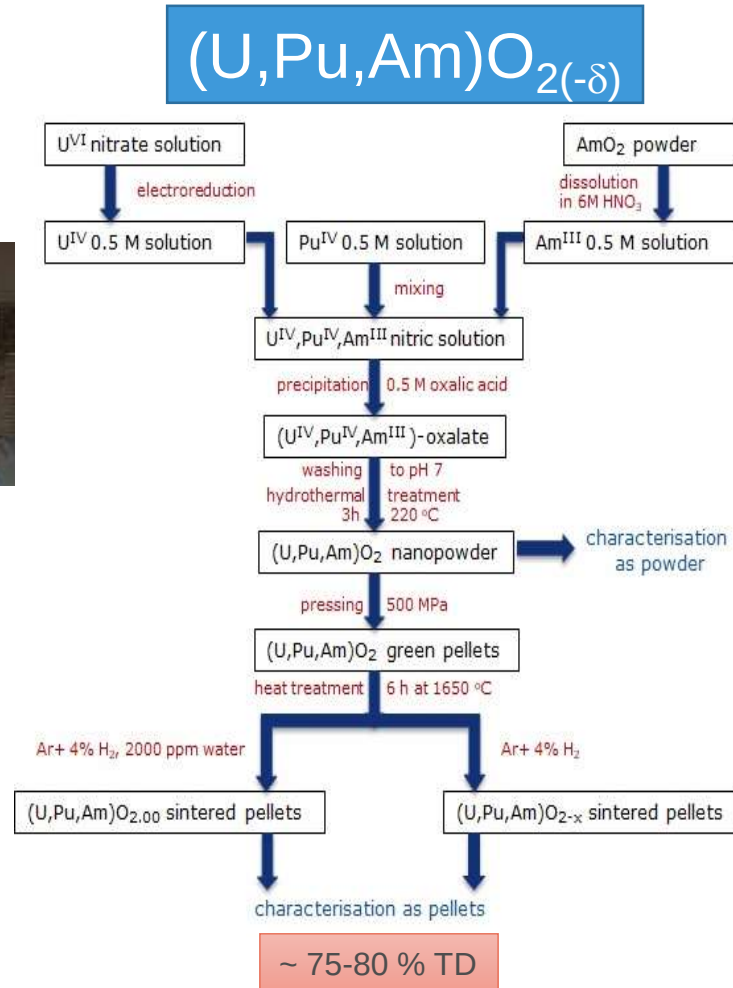
Contribution of experimental data and evaluations to the Joint Evaluated Fusion and Fission File (JEFF)



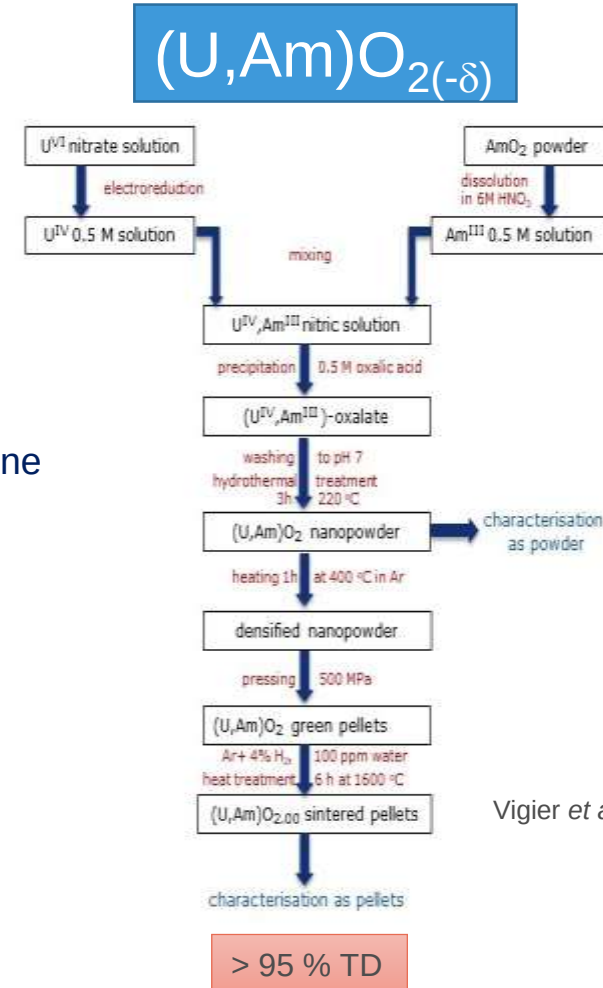
Nuclear Data High Priority Request List

Hydrothermal conversion of An -oxalates to produce AnO_2

Aliovalent Am^{3+} can be incorporated in AnO_2 solid solutions of nc - and sintered forms.



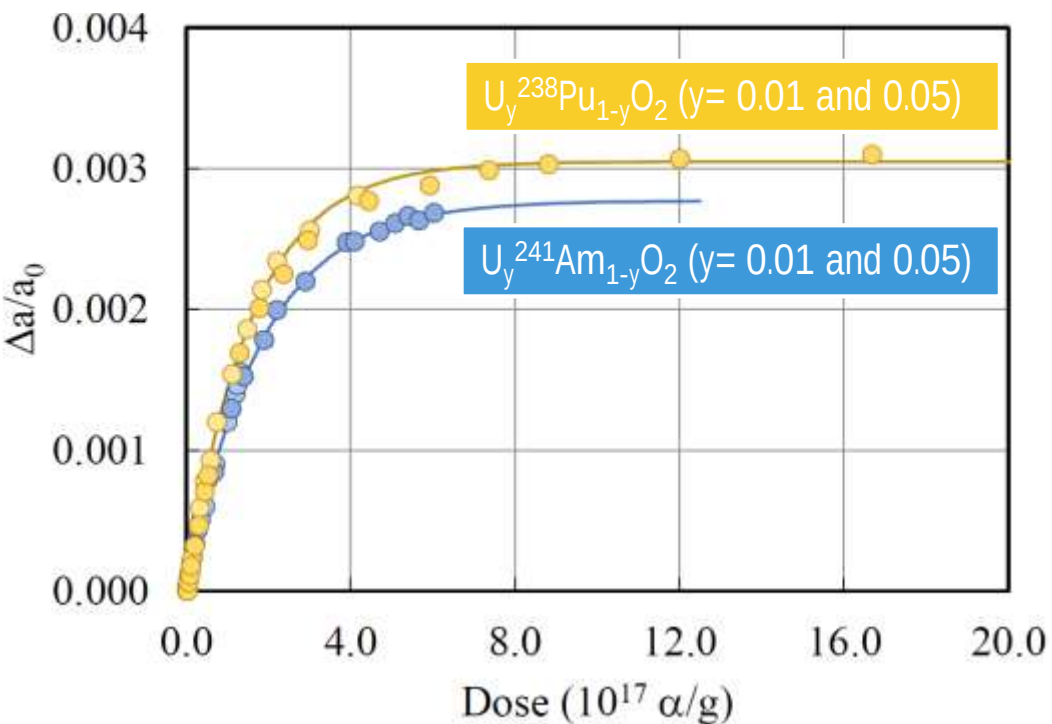
nc = nano-crystalline



Vigier et al., CrystEngComm 24 (2022) 6338-6348

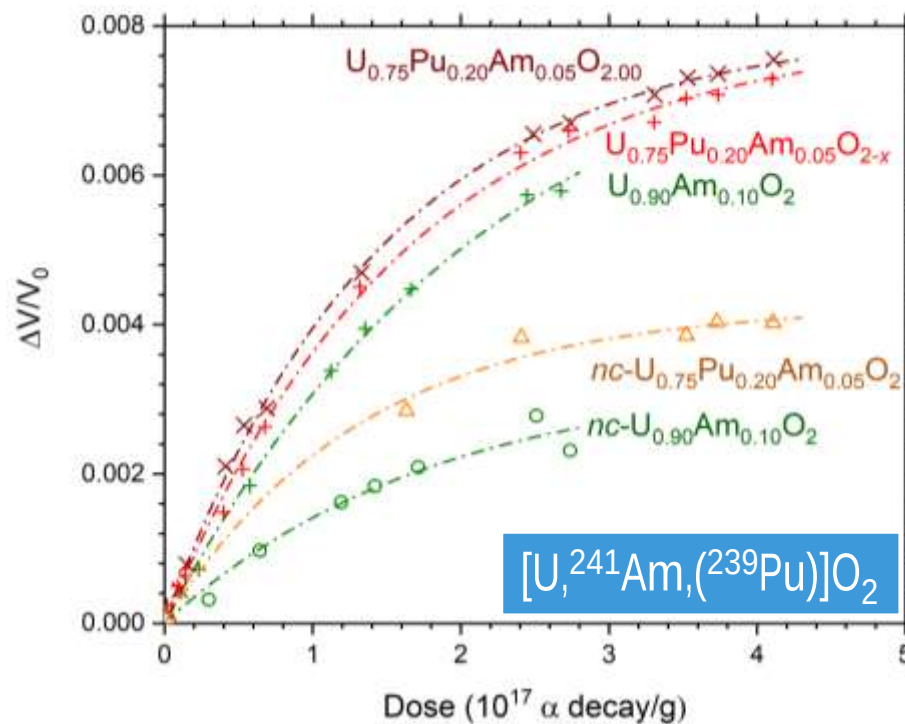
Swelling of (U,(Pu),Am)O₂

The presence of the dopant influences the radiation-induced expansion of the fluorite (-like) structures

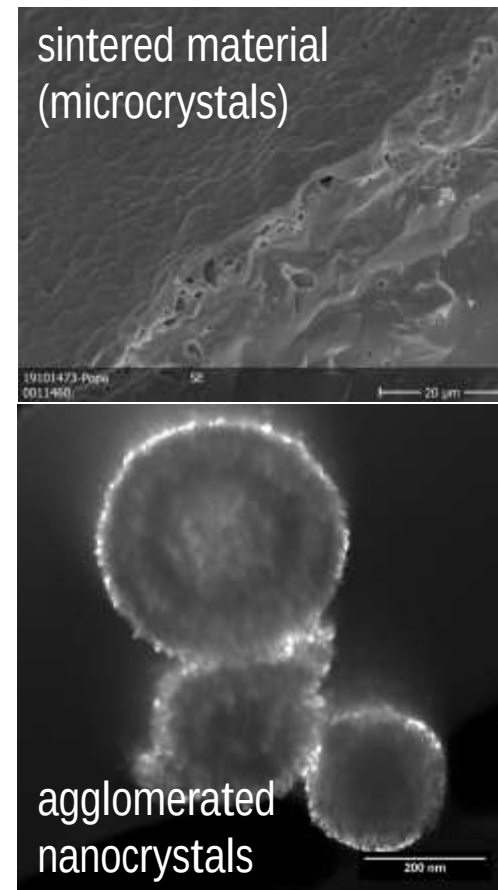


Griveau et al., J. Appl. Phys. **132** (2022) 185103

Reduced swelling for *nc*-form compared to sintered materials



shorter diffusion path to grain boundaries / surfaces

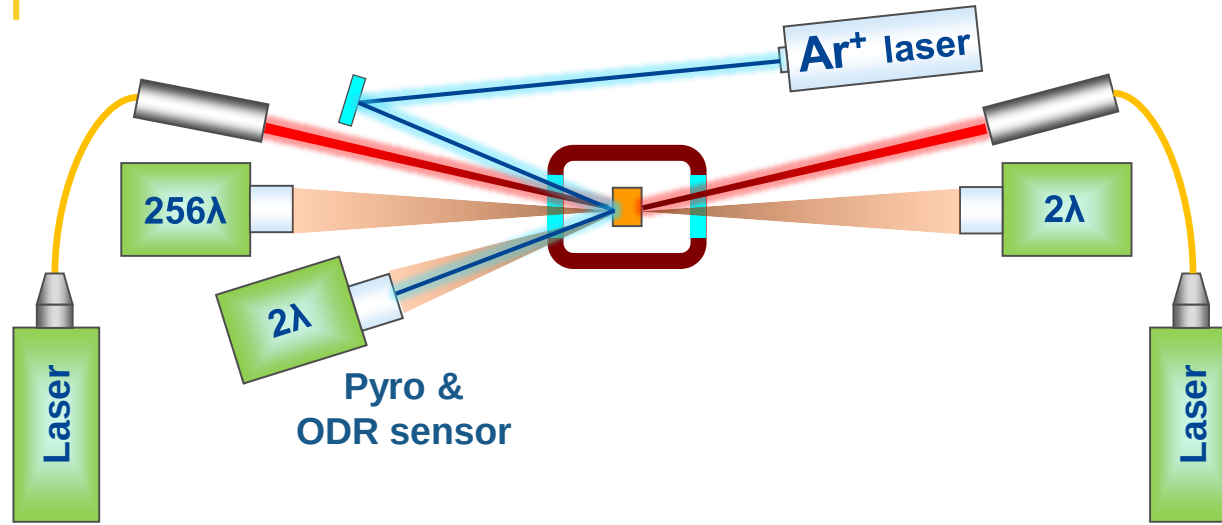


Vigier et al., CrystEngComm **24** (2022) 6338-6348

High temperature studies of mixed U-Am oxides

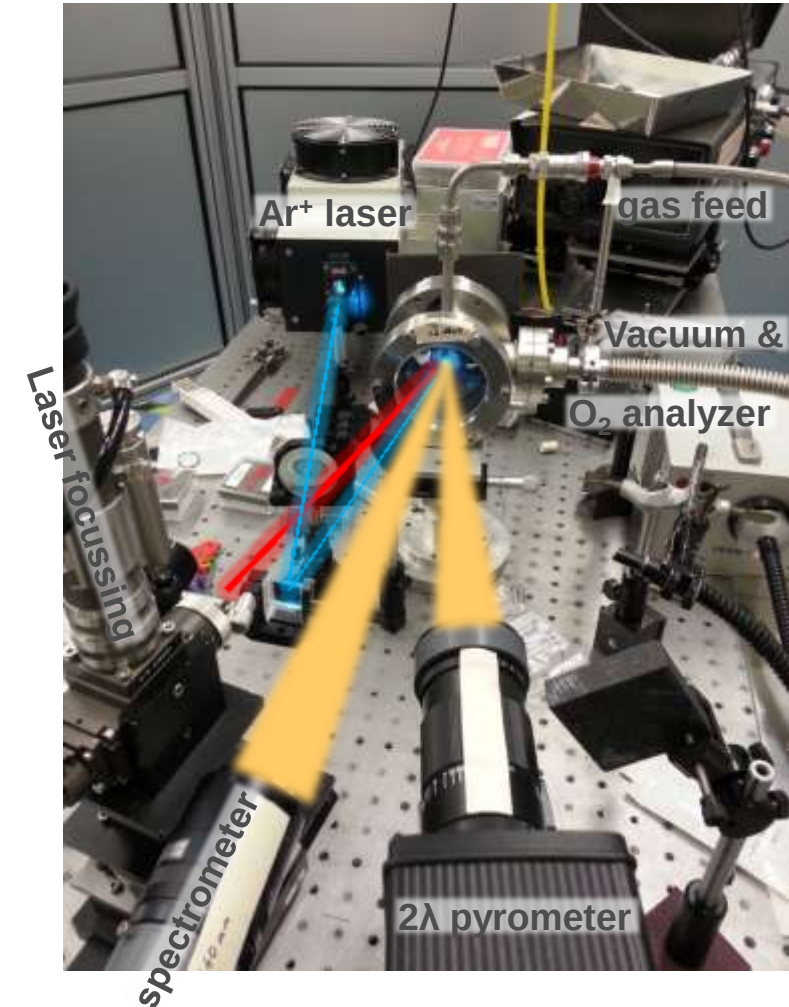
- In the **homogeneous transmutation scenario**, pellets with up to 5 mol% of Am would be distributed throughout the reactor core, where temperature can reach 2300 K
- In the **heterogeneous transmutation scenario** pellets of U-Am mixed oxides with up to 20 mol% of Am would be placed in the periphery of a fast neutron reactor core (blankets), where temperature can reach 1500 K
- The **melting temperature** and the thermal conductivity of the fuel are key parameters that determine the **safety margin**
- **High-temperature studies** of these materials are particularly challenging due to the intense radioactivity and the limited amount of Am available for laboratory investigations

Measurement setup

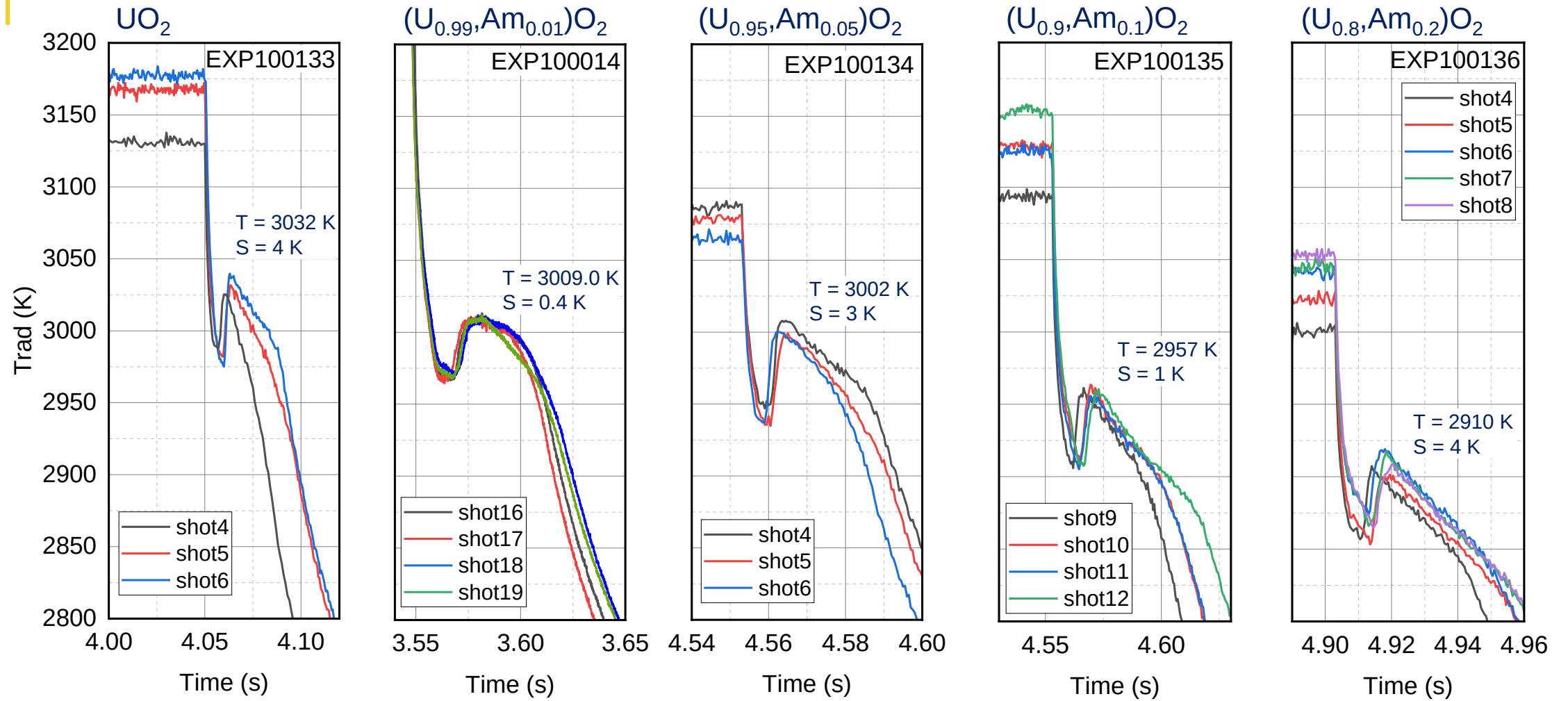


Measurement principle

- Heat sample to desired temperature, then let it cool down
- Plateaus and inflections in the recorded thermogram reveal phase transitions



T_{652} at solidification (radiance)

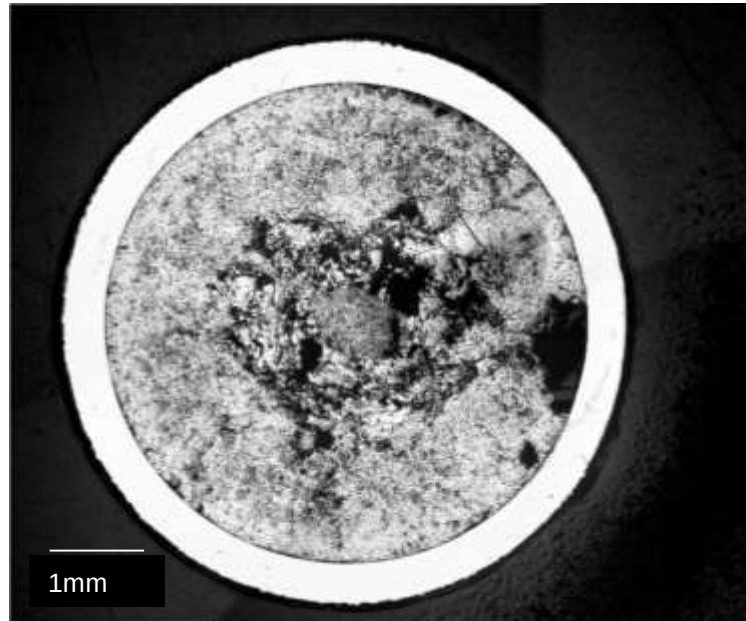


Other fuels

Demonstrations of closed cycle: METAPHIX

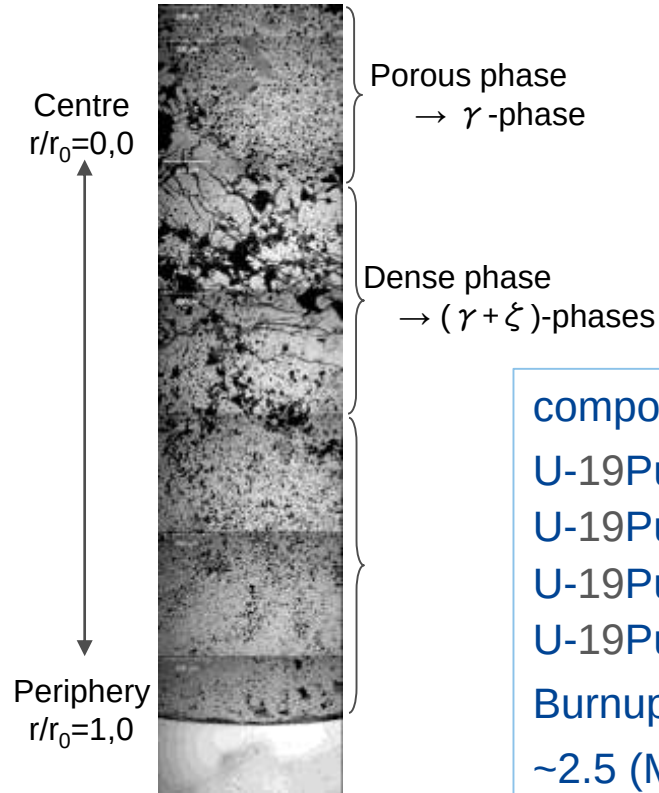
*safety of sustainable closed cycle –
FR metallic fuel*

(Phenix reactor - coll. CEA)



U-19Pu-10Zr-5MA-5RE METAPHIX-1

Optical micrograph 370mm from bottom of the fuel stack;
fuel outer $T = 505^{\circ}\text{C}$; estimated fuel center $T = 600\text{--}660^{\circ}\text{C}$.
Dark phase deposits along grain boundaries at $r/r_0=0.25\text{--}0.5$.
MA- and RE-containing precipitates at outer regions ($r/r_0=0.5\text{--}1.0$).



composition [wt%]:

U-19Pu-10Zr

U-19Pu-10Zr + 5MA

U-19Pu-10Zr + 2MA + 2RE

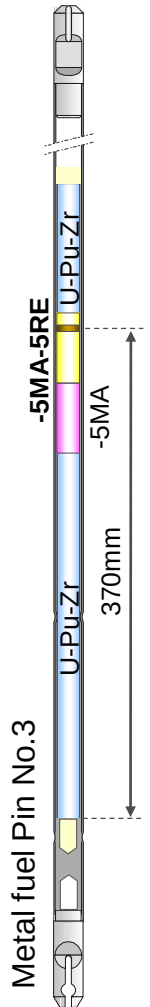
U-19Pu-10Zr + 5MA + 5RE

Burnup [at%]:

~2.5 (METAPHIX-1)

~7 (METAPHIX-2)

~10 (METAPHIX-3)

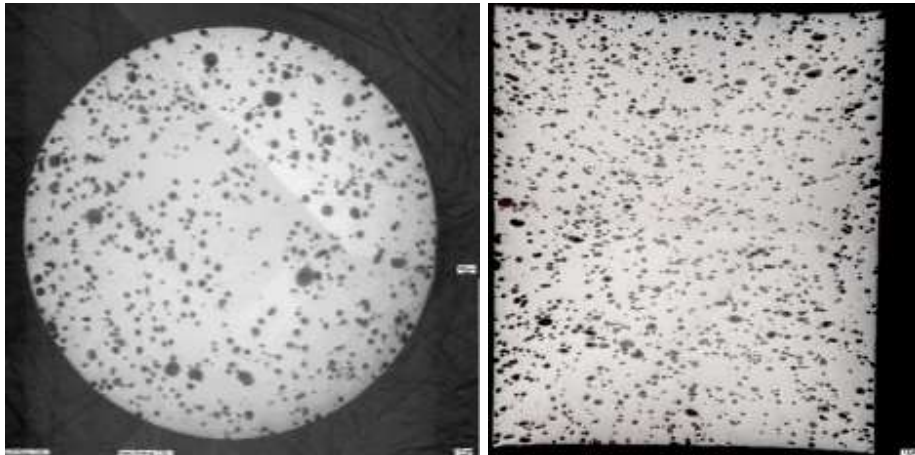
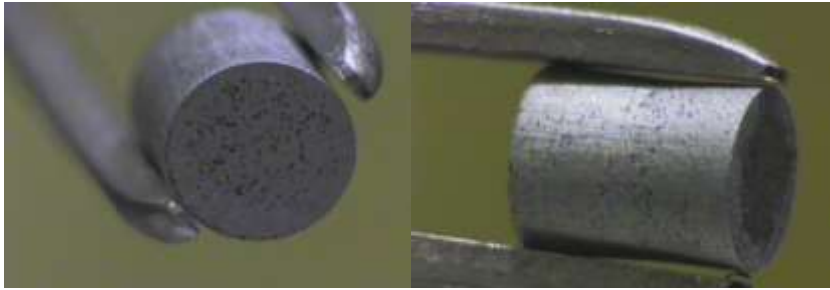


PIE confirms good irradiation behaviour compared to reference FR fuel (Phenix, EBR II)

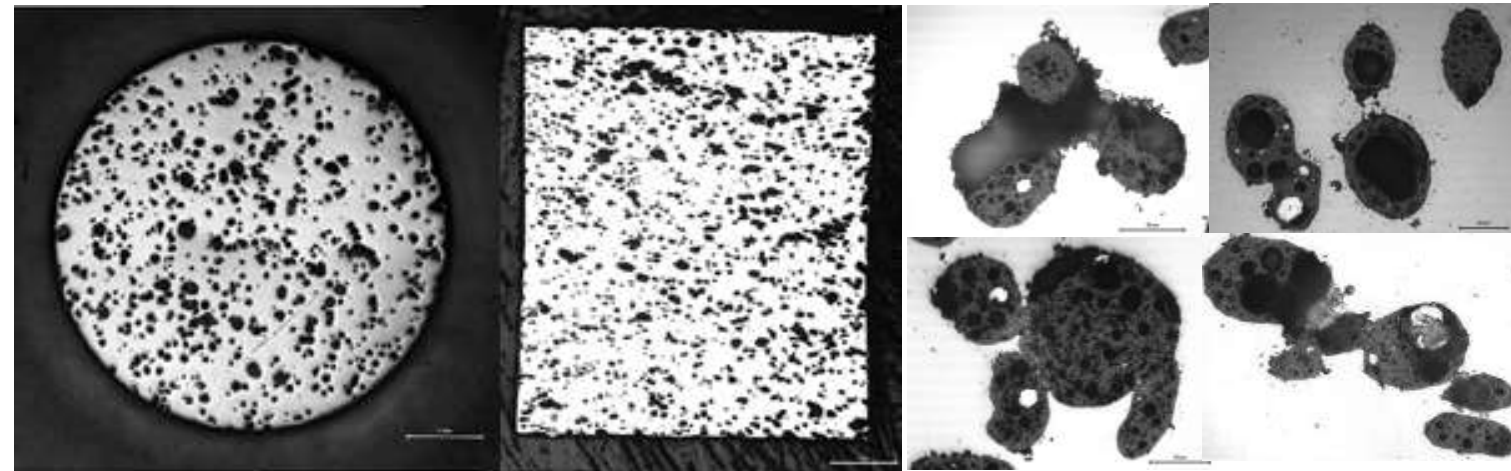
Characterization of $(\text{Pu}_{0.8},\text{Am}_{0.2})\text{O}_2 + 86 \text{ vol.}\% \text{ Mo}$

FUTURIX project – CERamic METal composite (CERMET)

Before irradiation



After irradiation



No cracks in matrix; metallic precipitates, FG bubbles in particles, morphology changes at interfaces

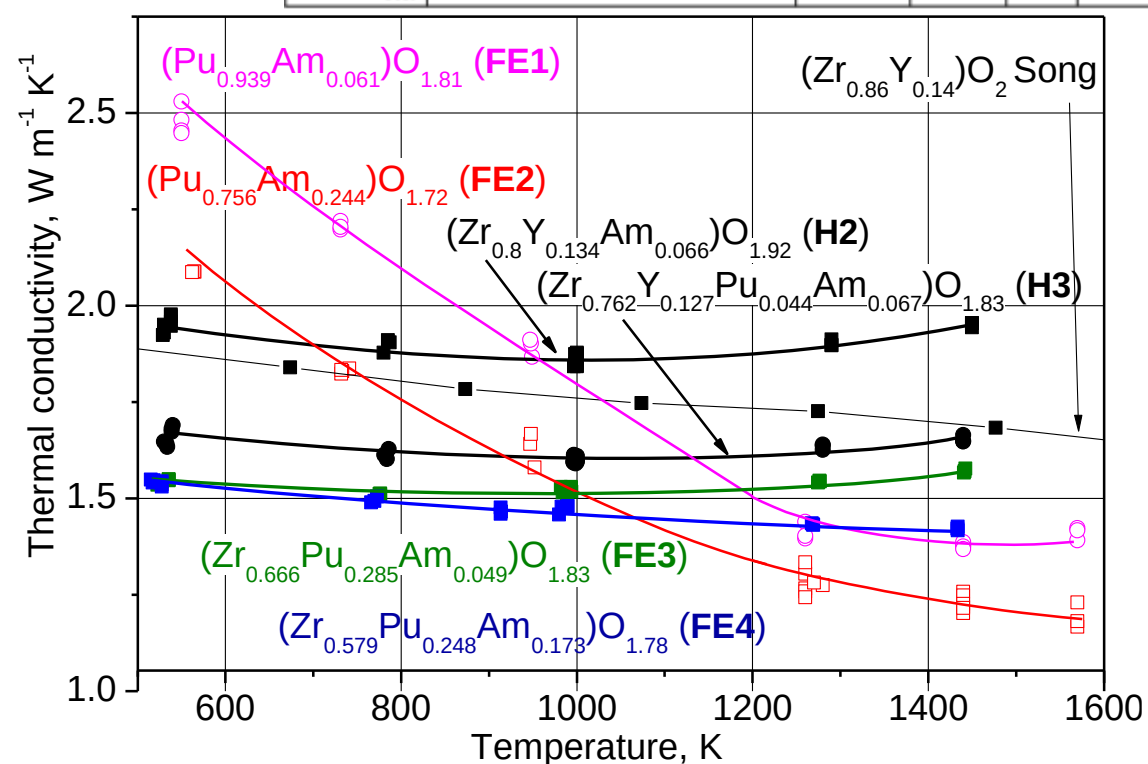
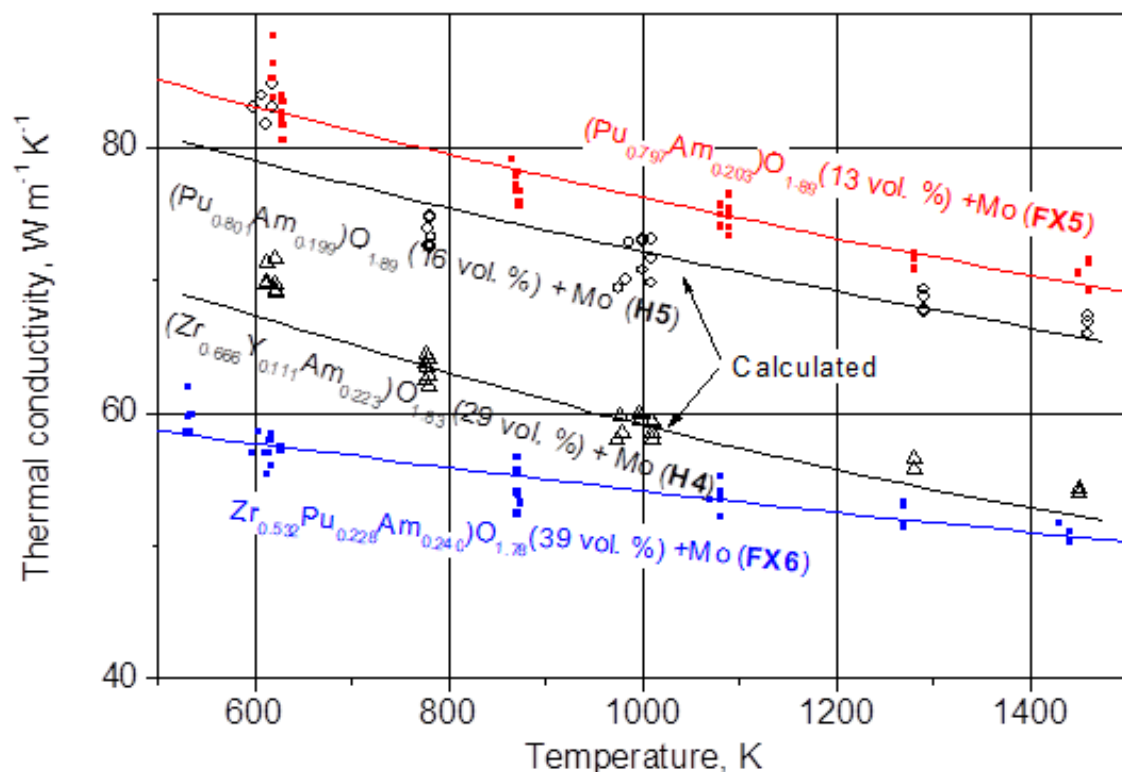
Burnup 17.8 at%; FGR ~31%; transmutation rate 27%

FGR = fission gas release

Thermal conductivity of Am, Pu compounds

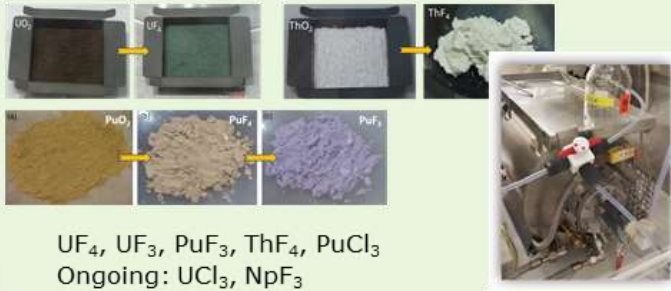
Laser flash measurements

Sample	Composition	Density g·cm ⁻³	%TD	TD g·cm ⁻³	Porosity Vol. %
FUTURE FE1	(Pu _{0.939} Am _{0.061})O _{1.81}	9.40	82.1	11.45	17.9
	(Pu _{0.756} Am _{0.244})O _{1.72}	10.41	90.4	11.51	9.6
	(Zr _{0.666} Pu _{0.285} Am _{0.049})O _{1.83}	7.17	92.2	7.78	7.8
	(Zr _{0.579} Pu _{0.248} Am _{0.173})O _{1.78}	7.87	91.3	8.62	8.7
HELIOS H2	(Zr _{0.8} Y _{0.134} Am _{0.066})O _{1.92}	5.91	92.6	6.389	7.4
	(Zr _{0.762} Y _{0.127} Pu _{0.044} Am _{0.067})O _{1.83}	6.14	94.1	6.523	5.9
	(Zr _{0.666} Y _{0.111} Am _{0.223})O _{1.83} (29 vol. %) + Mo	8.934	94.9	9.409	5.1
	(Pu _{0.801} Am _{0.199})O _{1.89} (16 vol. %) + Mo	10.076	96.8	10.406	3.2
FUTURIX FX5	(Pu _{0.797} Am _{0.203})O _{1.89} (13 vol. %) + Mo	10.37	95.1	11.50	4.9
	(Zr _{0.532} Pu _{0.228} Am _{0.240})O _{1.78} (39 vol. %) + Mo	9.497	94.8	8.48	5.2



R&D activities on molten salt fuels at JRC

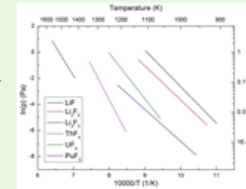
Synthesis & Electrochemistry



MSR Safety Studies

Thermo-physical Properties

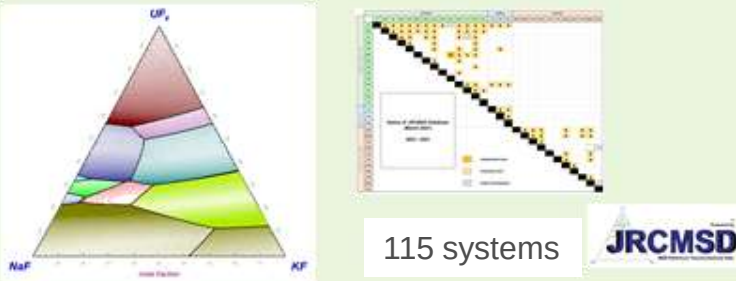
- Melting point
- Thermal conductivity
- Heat capacity
- Fission product behaviour
- Vapour pressure
- Transition enthalpy
- Phase diagrams
- Density, Viscosity
- Corrosion studies



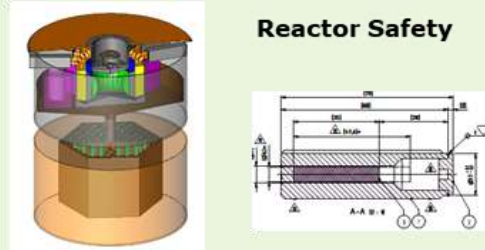
Irradiation of MS & PIE MS



Thermodynamic Database development



Reactor Safety



MIMOSA

Corrosion studies
Density
Heat capacity

Ongoing / experimental and modeling activities

PATRICIA

Characterization of MA-MOX, in particular Am-MOX: melting temperature, specific heat, thermal conductivity. Both experimental and modelling with TRANSURANUS.

Additional developments: Helium behaviour: SCIANTIX code coupled with TRANSURANUS and GERMINAL; fuel creep.

Interaction of fuel with coolant.

Interaction of MOX, MA-MOX with Pb-Bi eutectic (LBE) up to 800 C was tested in projects SEARCH, Myrte.

Extend interaction temperature between fresh MOX & LBE at 1200°C and 1650°C;

Test interaction of LBE with irradiated MOX at 500°C and 800°C.

Pre-design and pre-safety analysis of a transient irradiation test in a MTR. Study on fuel pin for transmutation in MYRRHA.

TRANSPARANT

Efficiency of Am separation from used fuel.

Experimental- and fuel performance code development work on the behaviour of Am-bearing fuel under irradiation.

Safety related research supporting the licensing process of MYRRHA.

Combination of experiments, theoretical studies and numerical simulations that builds upon the collaborative efforts initiated in the PATRICIA project.

Conclusions



Recycling of Pu, U in light water reactors is possible: MOX fuel is currently used

Closing the fuel cycle introducing partitioning and transmutation (P&T) allows destroying long-lived radiotoxic species, in particular minor actinides, reducing the radiotoxic lifetime of high level nuclear waste

The feasibility of P&T has been demonstrated; further research is needed to optimize the behaviour of fuel and the various steps of a closed nuclear fuel cycle, in particular scaling-up from laboratory to industrial demonstration.

The necessary tools (research infrastructure, competences, funding) must be available.

Collaborative efforts, including European projects under the research framework program, JRC contributions and international initiatives, continue to address critical aspects of Minor Actinides P&T to optimize efficiency and operational safety.

Key safety aspects are investigated: phase stability of the material; transient behaviour; fission gas and helium behaviour, swelling; thermal conductivity and operating temperature; fuel-cladding interaction.

Disposal of high-level waste in a geologic repository will remain a necessary step of the fuel cycle.

Thank you



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