The Abdus Salam
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International Atomic Energy Agency

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School on Physics, Technology and Applications of Accelerator Driven Systems (ADS)

19 - 30 November 2007

ADS Fuels for Partitioning & Transmutation

Frodo KLAASSEN

NRG Fuels, Actinides & Isotopes
1755 ZG Petten
The Netherlands

ADS fuels for Partitioning & Transmutation

Frodo Klaassen
Nuclear Research and consultancy Group (NRG)
Westerduinweg 3 / P.O. Box 25
1755 ZG PETTEN, the Netherlands
klaassen@nrg-nl.com

Sylvie Pillon
Commissariat à l’Énergie Atomique (CEA)
Centre du Cadarache

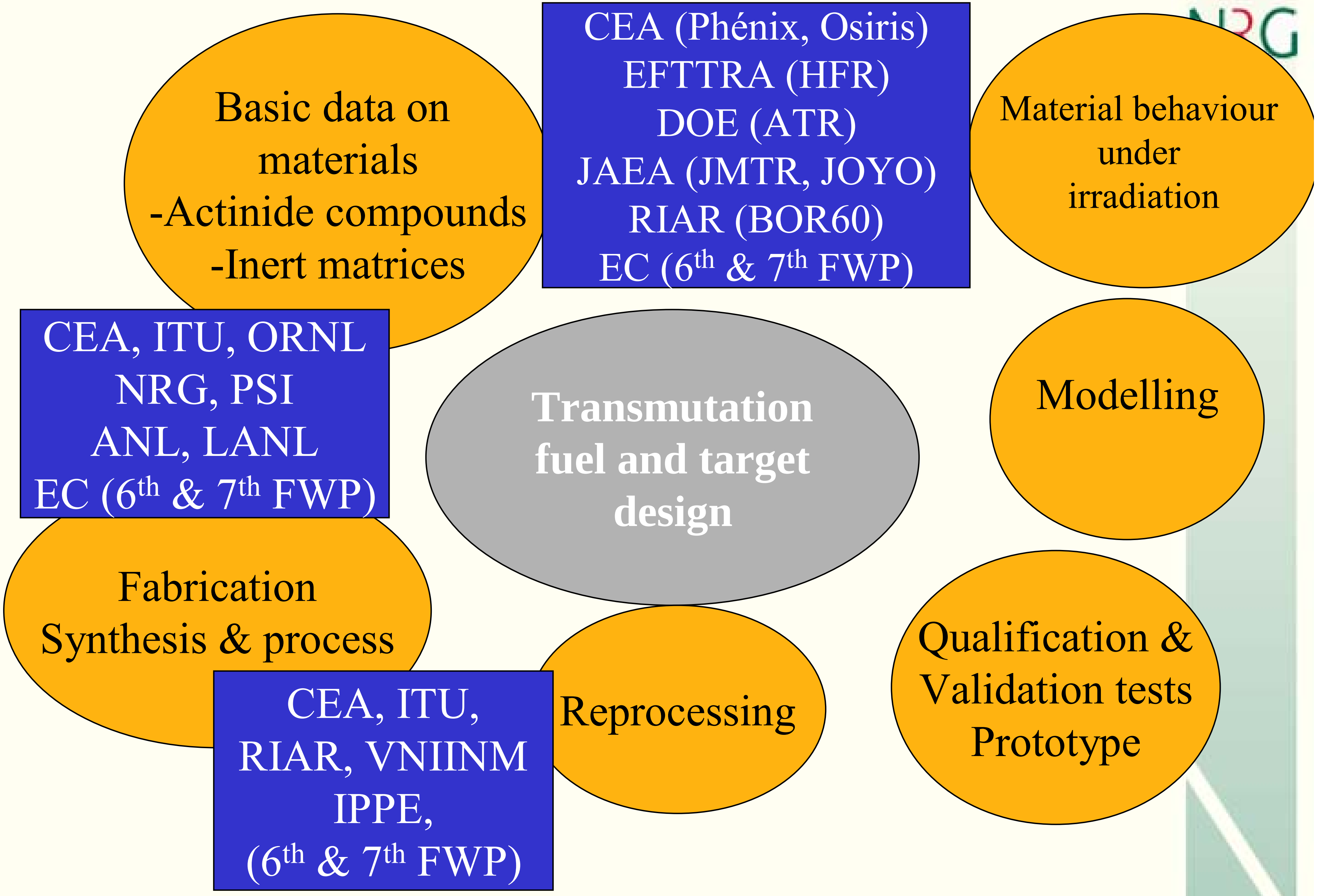
School on Physics, Technology and Applications of ADS,
ICTP, Trieste, 30 November 2007

Content

1. Fuel development : needs and organisations
2. ADS Fuel specification : what is an ADS fuel?
    - Fuel requirements
    - Fuel types : oxides, nitrides, metal alloys
    - EUROTRANS selection
3. ADS Fuel behaviour under irradiation
    - Current knowledge on ADS fuels (and targets)
    - Comparison with MOX fuel containing Am and Np
    - Experiments in HFR and Phénix

BREAK

Content



Transmutation fuel and target design

Basic data on materials
-Actinide compounds
-Inert matrices

CEA (Phénix, Osiris)
EFTTRA (HFR)
DOE (ATR)
JAEA (JMTR, JOYO)
RIAR (BOR60)
EC (6th & 7th FWP)

Material behaviour under irradiation

CEA, ITU, ORNL
NRG, PSI
ANL, LANL
EC (6th & 7th FWP)

Modelling

Fabrication
Synthesis & process

CEA, ITU,
RIAR, VNIINM
IPPE,
(6th & 7th FWP)

Reprocessing

Qualification &
Validation tests
Prototype

ADS fuel specification: strategy

- * Double strata strategy (MA burning)
  - – U-free fuels or TRU fuels
    - – Typically 30 to 50% Pu
    - – 70% Am and 30% Cm
    - – Np included or not (depending on the P&T scenario)
  - – High burn-up
- * Integrated ADS fuel cycle (TRU burner)
  - – Typically 4 to 6 t TRU/year
  - – Reprocessing and TRU separation
    - – Pyrochemistry or pyro/hydro mixed system
  - – Fuel refabrication

ADS fuel specification: requirements

- Fuel compatibility with coolant
  - EUROTRANS : Pb coolant (Pb/Bi for the demonstrator)
  - Options are:
    - He coolant (gas)
    - Na coolant
    - Pb or Pb/Bi

- Fuel compatibility with cladding
  - Na : austenitic steels : 15-15Ti (AIM1)
    - Peak cladding temperature : 650°C
  - Pb, Pb/Bi : ferritic-martensitic steels T91 (9Cr-1Mo)
    - Peak cladding temperature : 550°C
  - He : ceramic composite based on SiC
    - Peak cladding temperature : 1100°C

Actinide compounds	Melting point or temperature of decomposition (°C)	Density (g/cm^3)	Thermal conductivity at 1273K (W.m^{-1}.K^{-1})	ΔG_f^0 (298K) (kJ/mol)
Pu-Am-Cm-40Zr	1327	9,61	22	-
NpO_2	2550	11,14		-720
PuO_2, Pu_2O_3	2390, 2360	11,46		-730
AmO_2, Am_2O_3	2175, 2205	11,71		-620,
Cm_2O_3	377		-	-
NpN	2830 (12 atm N_2)		17	-
PuN	2675 (1 atm N_2)		13	-150
AmN	2570 (1 atm N2)		-	-
CmN	-	-	-	-
NpC_{1-x}	>1400			-90
PuC_{1-x}, Pu_2C_3	1654, 2050	13,6		-70, -100
Am_2C_3	-	-		-70
NpS	-	-		-342
PuS	2347	-		-360
AmS	-	-	-	-
CmS	>1550	-	-	-

The best thermal conductivity and TRU density

The best thermal-dynamical stability

The highest melting points and good cond.

Intermediate properties

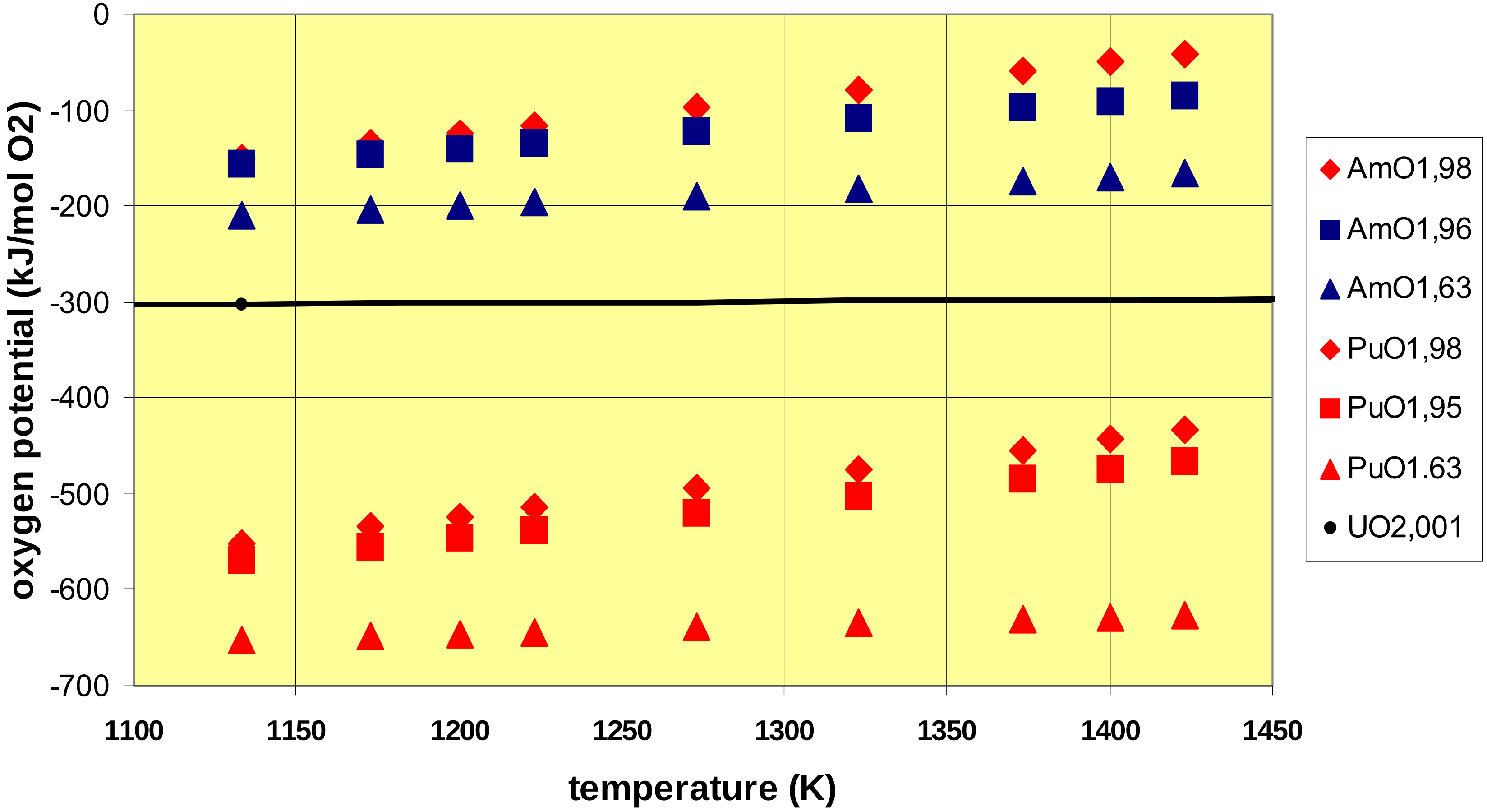
3 main candidates: metal, oxide and nitride

Fuels: past experience (FBR)

	Mixed Oxide $(U,Pu)O_2$	Mixed Nitride (U,Pu)N	Metal alloys U-Pu-Zr
Compatibility with Pb or Pb/Bi	yes	yes	**no**
Compatibility with clad	yes	yes	yes
Fuel temperature limit	melting point (solidus : ~2800°C)	nitride decomposition **(~1700°C, influence of the atmopshere)**	eutectic reaction with clad : **725°C** with U-Pu-Zr : **Na bond**
Behaviour under irradiation	moderate swelling accommodated by the clad in FBR temperature conditions 25% achieved : limitation by the clad damage dose	high swelling accommodated by the free volume 15 at% achieved : limitation not yet known	high swelling accommodated by the clad 20 at% BU achieved : limitation by the fuel/clad chemical interaction
Fabricability	Easy (wet and dry process)	**Complex** due to the first nitridation phase from oxides	**Pyrophoricity**
Solubility in molten salts	yes	yes	yes
Solubility in nitric acid	yes	yes	yes
Application to ADS	- **inert matrix** to improve the thermal behaviour	-**inert matrix** to improve the stability at high temperature -high AmN volatility (fabrication issue) -C14 production from the N14 : (n, p) reaction	-only for Na-cooled ADS - Na bonding required - high Zr content to improve the melting point - high Am volatility (fabrication & irradiation issue)

Chemical stability of (ADS) oxide fuels

- U-free fuels have poorer thermal & thermodynamic properties compared to U-based fuels
  - Lower performance in pile



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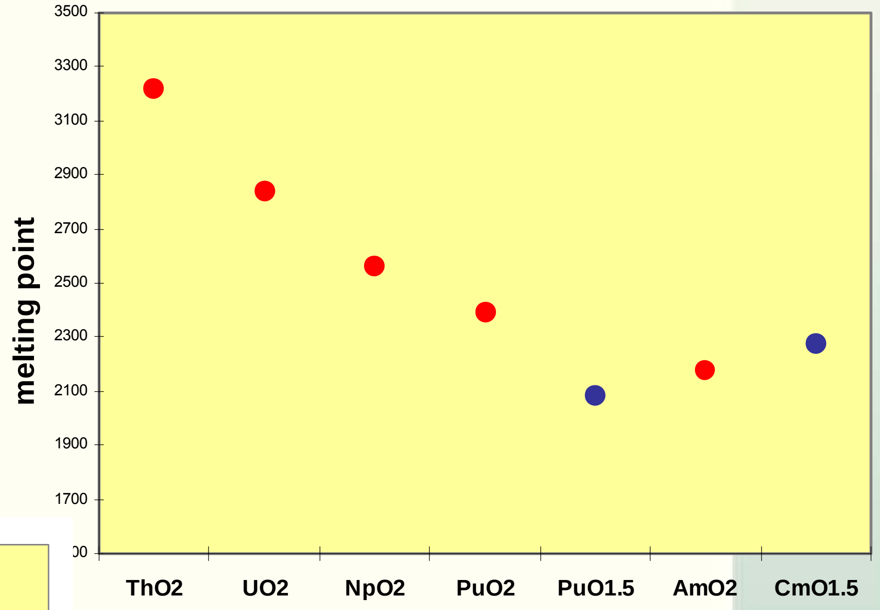

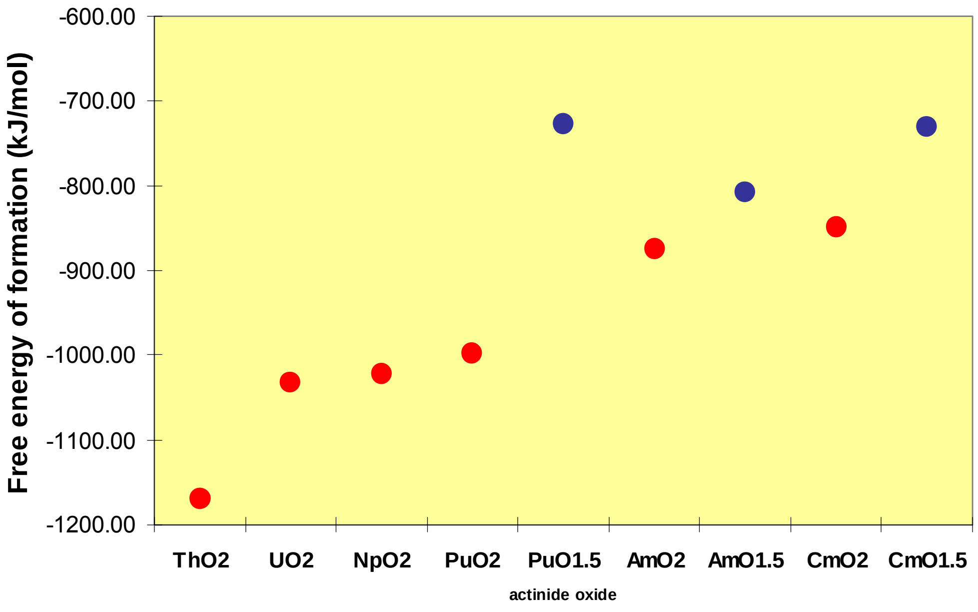

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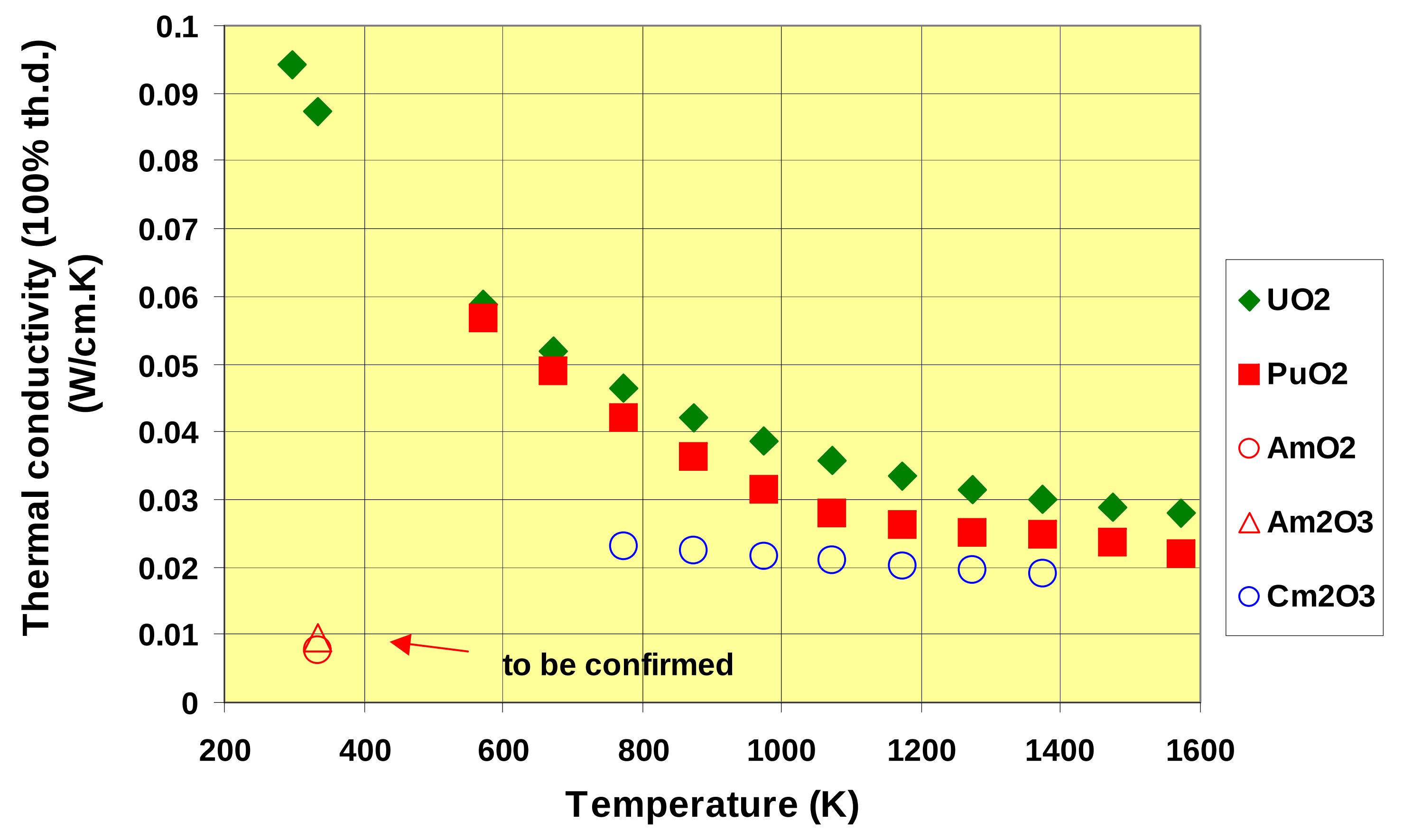


ADS fuel specification: main fuel types

- * TRU oxides
  - – Low thermal conductivity
    - – Improvement with inert matrix of high conductivity

→ EUROTRANS development

- * TRU nitrides
  - – Low stability at high temperature
    - – AmN is especially volatile
  - – ^{14}C production from (n,p) reaction on ^{14}N
    - – Improvement with inert matrix of high stability
    - – ^{15}N enrichment

→ Considered in Europe / Japan / (US)

- * TRU metal alloy
  - – Low melting point
    - – Increase of Zr content
    - – Na bond

→ US development

Inert matrices fuels (IMF) for ADS

- * Inert matrix provides a fertile-free (Uranium-free) diluent in which the (minor) actinide phase is incorporated

- * Inert Matrix Fuels improve the fuel properties
  - Thermal conductivity higher than that of UO_2
  - Thermal stability higher than that of AmN
  - High melting point (higher than the Actinide compound)
- * Inert Matrix Fuels improve the fuel performance
  - Clad corrosion (fission product retention)
  - High burnup (local)
  - Stability under irradiation

ADS fuels: many requirements !

- Good thermal properties
  - High melting point
  - High thermal conductivity
  - High thermal stabilty
- Good chemical properties
  - Compatible with coolant, clad and compound of MA
- Resistant to neutron, FP and alpha radiation
  - Cubic crystal structure (isotropic deformation)
- Low activation
  - Waste reduction
- Ability to be reprocessed and recycled
  - Aqueous process
  - Molten salts process
- Fabricability
- Compatibility with glasses (final storage)

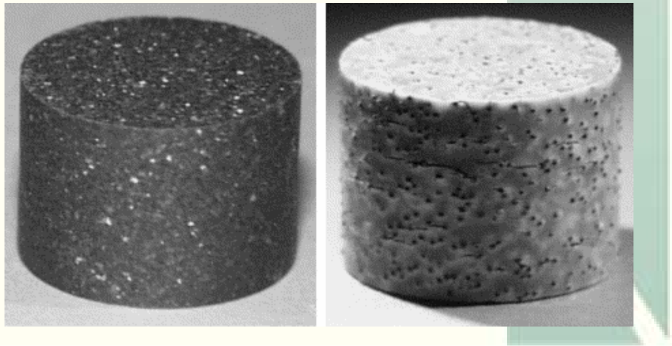
Inert Matrices for ADS: selection for EUROTRANS

	Inert matrix	Crystal lattice	Thermal properties	neutron activation	Chemical compatibility with					
					AnO_2	AnN	Zry	Acier	Na	H_2O
	st-ZrO_2	+	-	+	+ (solid solution)		+	+	+	+
oxide	MgO	+	+	+	- (eutectic)		?	+	+	-
	Al_2O_3	- (hexagonal)	+	+	- (eutectic)		+	+	+	+
	Y_2O_3	+	-	+	+ (solid solution)		?	+	-	-
	CeO_{2-x}	+	-	+	+ (solid solution)		-	+	-	+
	$MgAl_2O_4$	- (monoclinic)	+	+	- (eutectic)		+	+	+	+
	$Y_3Al_5O_{12}$	+	+	+	+		+	+	+	+
	$CePO_4$	- (monoclinic)	-	+	+		?	?	-	+
	$ZrSiO_4$	- (tetragonal)	- (decomposition)	+	+		?	?	-	+
carbide, nitride	SiC	-(hex./cubic)	+	+	-	-	-	-	+	+
	$^{11}B_4C$	- (trigonal)	- (vapor pressure)	+		-		-	+	+
	^{11}BN	-(hex./cubic)	- (vapor pressure)	- (^{14}C)					+	+
	Si_3N_4	- (hexagonal)	- (vapor pressure)	- (^{14}C)				+	+	+
	CrN		- (vapor pressure)	- (^{14}C)						
	TiN	+	+	- (^{14}C)	+	+		?	+	?
	YN	+	+	- (^{14}C)		+ (solid solution)		?	?	- (hygroscopic)
	ZrN	+	+	- (^{14}C)	-	+ (solid solution)		-	+	+
	CeN	+	+	- (^{14}C)		+ (solid solution)		?	?	-
	AlN	- (hexagonal)	- (vapor pressure)	- (^{14}C)	- (?)	+		+	+	-
metal	Cr	+	+	+	- (eutectic ?)	- (eutectic)	+	+	+	-(H_2 hazard)
	V	+	+	+	- (eutectic ?)	- (eutectic)	+	+	+	-(H_2 hazard)
	W	+	+	+	+	+	+	+	+	-(H_2 hazard)
	Mo	+	+	- (^{95}Mo)	+	+	+	+	+	-(H_2 hazard)
	Nb	+	+	- (^{94}Nb)			-		+	-(H_2 hazard)
	Acier	+	-	+	- (eutectic)	- (eutectic)	+	+	+	-(H_2 hazard)

ADS Inert Matrix Fuels (IMF)

* Composition of IMF is a compromise between:

- Volume of inert matrix higher than 55-60%
  - To optimise the composite microstructure
    - Inert matrix surrounding fissile particles
    - Improve the quality of the pellet surface
    - Reduce the matrix cracking
  - To keep good thermal properties when mixed with fissile particles
  - For an easier fabrication process
- Volume of inert matrix as low as possible
  - To increase the transmutation rate
  - To reduce the core size
  - Easier reactivity control



Inert Matrix Fuels: selection for EUROTRANS

1. CERCER fuels (composite CERamic-CERamic)
   - (Pu, Am,-*Np,Cm*-)$O_{2\pm x}$ + MgO or ZrO_2
     - $MgAl_2O_4$, the most studied matrix under irradiation (good stability under neutron), was not retained
     - Idem for YAG ($Y_3Al_5O_{12}$)

2. CERMET fuels (composite CERamic-METal)
   - (Pu,Am,-*Np,Cm*-) $O_{2\pm x}$ + Mo
     - Cr, V not retained : non compatible with TRU oxide
     - W not retained : too absorbent

Inert Matrix Fuels: selection for EUROTRANS

1. CERCER fuels (composite CERamic-CERamic)
   - (Pu, Am,-*Np,Cm*-)$O_{2\pm x}$ + MgO or ZrO_2
2. CERMET fuels (composite CERamic-METal)
   - (Pu,Am,-*Np,Cm*-) $O_{2\pm x}$ + Mo
3. Nitride fuel (solid solution)
   - (Pu,Am,-Np,Cm-)N + ZrN

- TiN not retained (bi-phased composite: no impact on the AmN stability)
- YN not retained (hygroscopic)

4 major candidates!

Summary Fuel selection for ADS

- Uranium-based (fertile) fuels
  - MOX: $(U,Pu)O_2$ with minor actinides
  - Nitrides (U,Pu)N etc.

- Inert Matrix Fuels provide an attractive alternative (more efficient TRU burning, option to tailor properties, etc.)

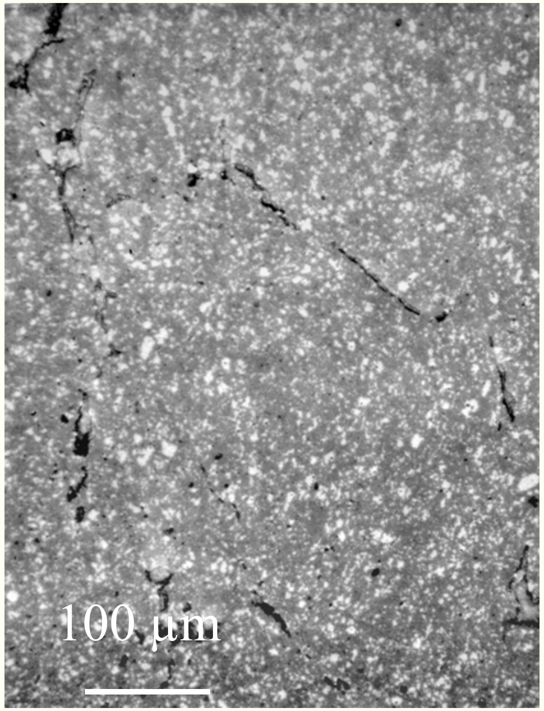
- Inert Matrix fuels:
  - CERCER fuels, based on MgO or ZrO_2
  - CERMET fuels, based on Mo
  - Nitrides, based on ZrN

Spinel as Inert Matrix

- * Magnesium aluminate spinel was, for a long time, thought to be the ideal inert matrix
- * $MgAl_2O_4$ was intensively investigated
  - – Good radiation resistance (neutrons / alpha)
  - – Good chemical stability
  - – Relatively high thermal conductivity > UO_2
  - – Ability to accommodate displacements (dpa)

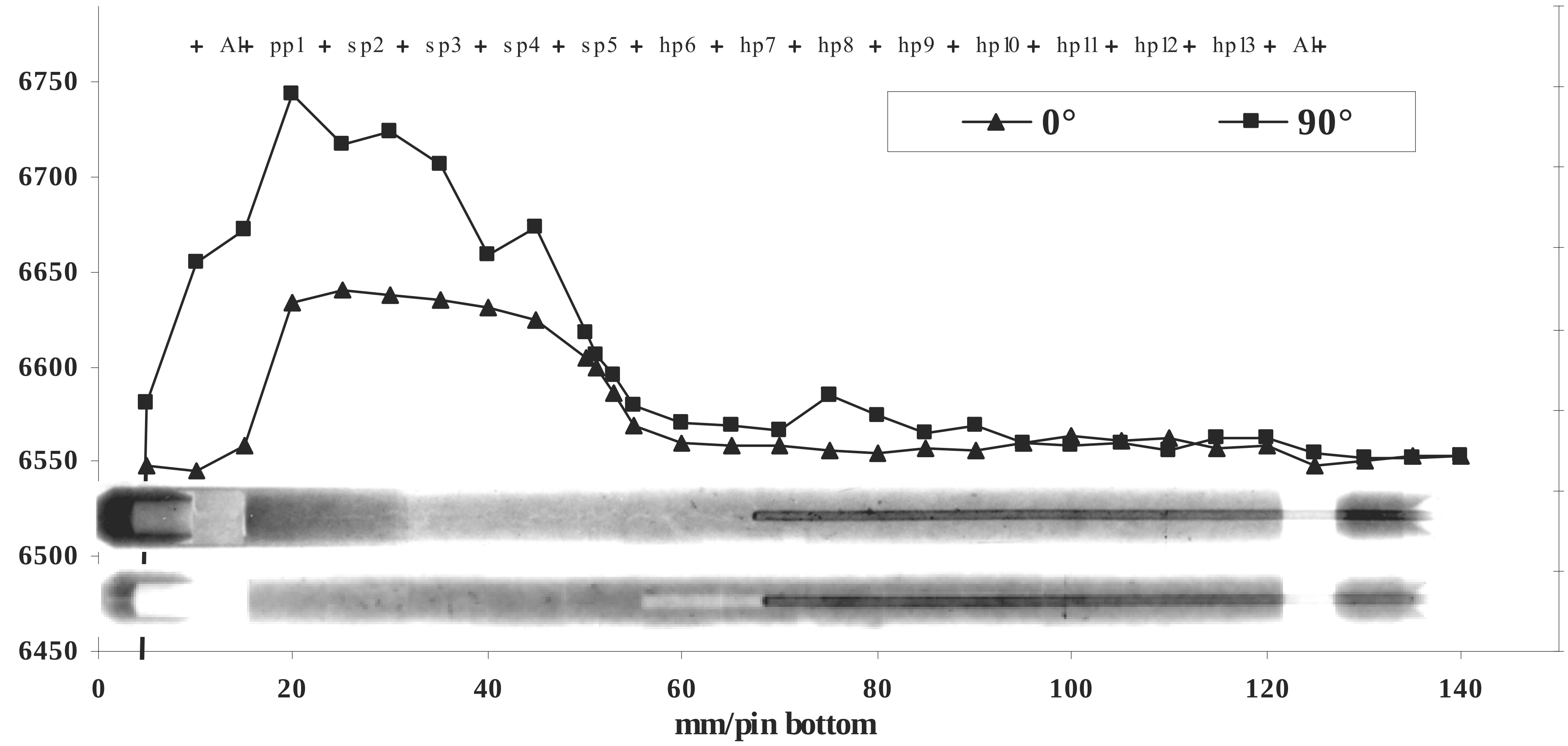
* In the end spinel was discarded as Inert Matrix, but we have learned a lot from the experiments on spinel

IMF behaviour under irradiation: spinel-based CERCER with U or Pu



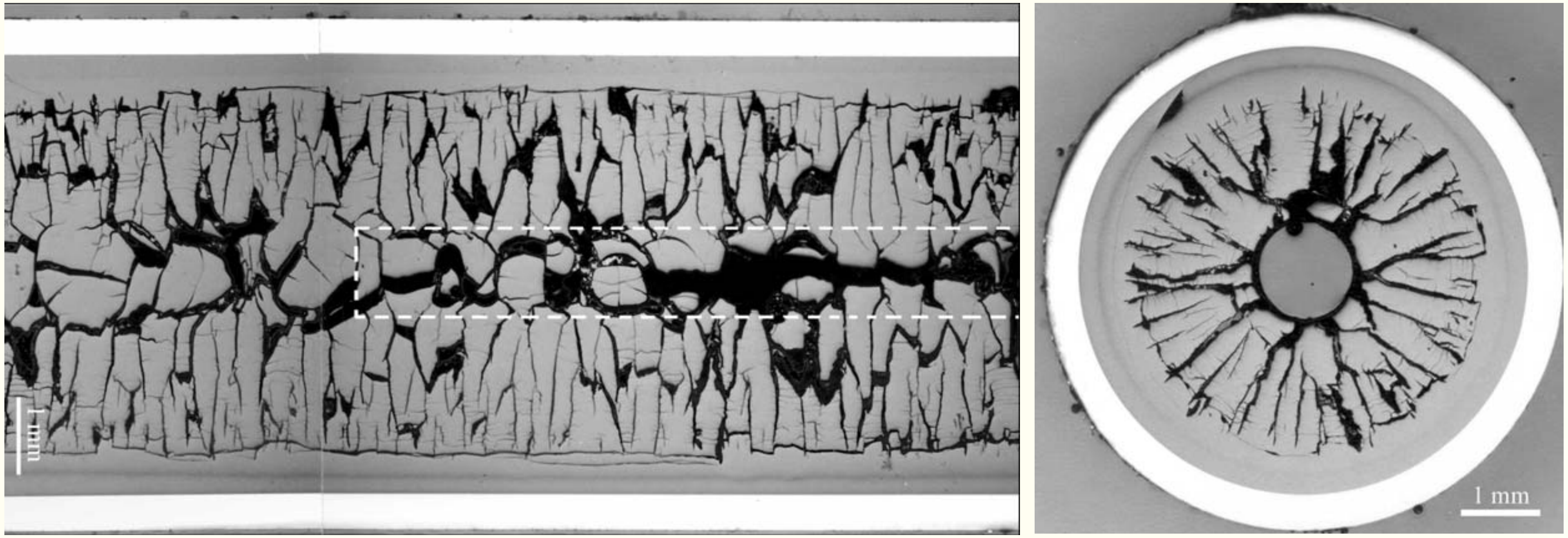
- * THERMHET (1997) in Siloe (CEA, France)
  - – UO_2 (40 wt%) + $MgAl_2O_4$ at low burn-up (1.3 at%)
  - – Prepared by powder mixing of UO_2 and spinel

THERMHET irradiation



Pin profilometry and Neutronography

THERMHET irradiation



- * Spinel, known to be resistant to neutron, is not resistant to FP damaging!
  - – Fracturation after strong swelling
    - – Void filling (Thermocouple hole, fuel/clad gap, fuel porosity)
  - – High swelling estimated at ~15 vol%

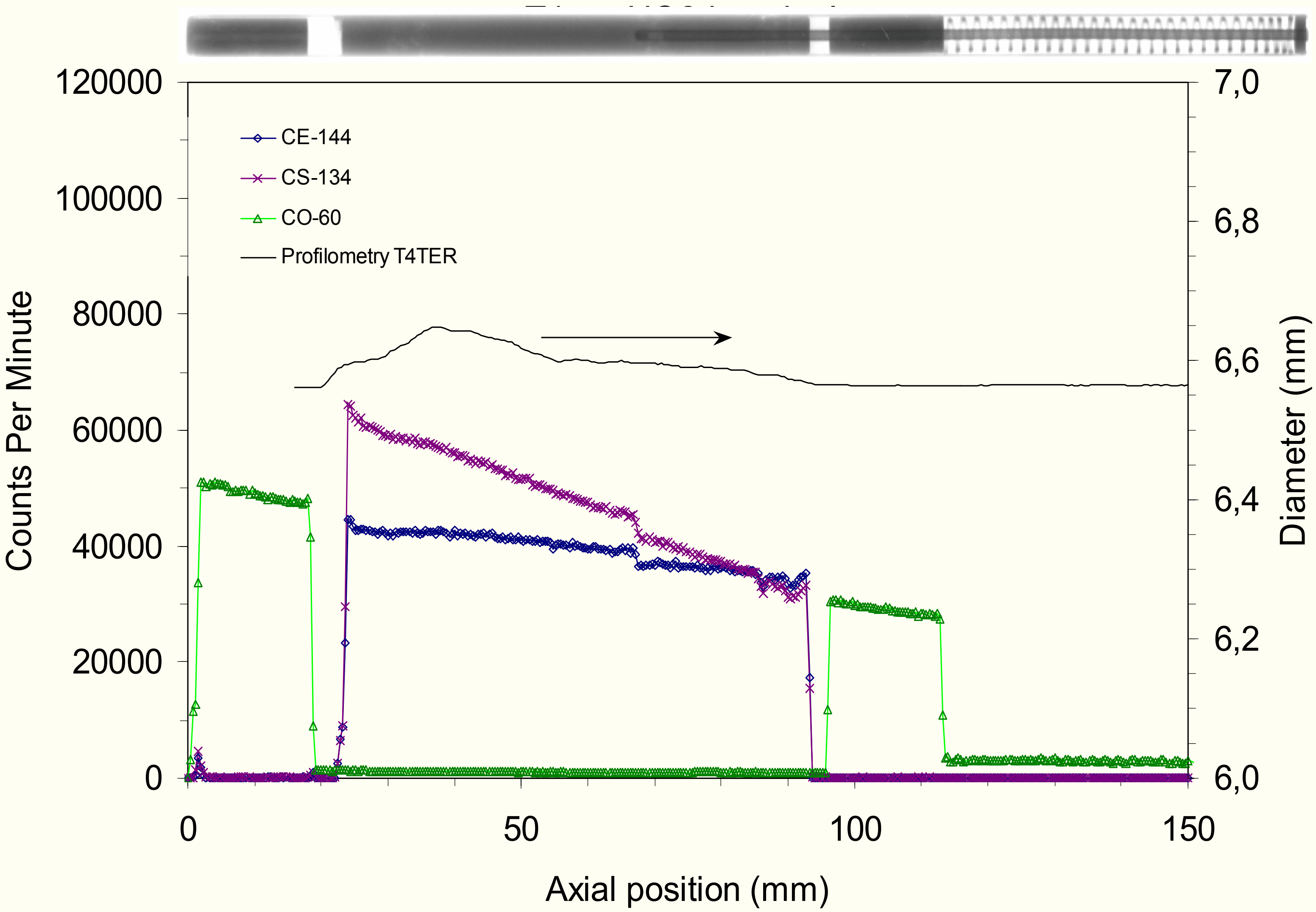
Results EFTTRA T4ter irradiation in HFR

* EFTTRA T4ter:

- 25 wt% UO_2 (20% enriched) mixed with spinel
- Irradiation of 652 full power days in HFR
- Burn-up ~ 150 GWd/m^3

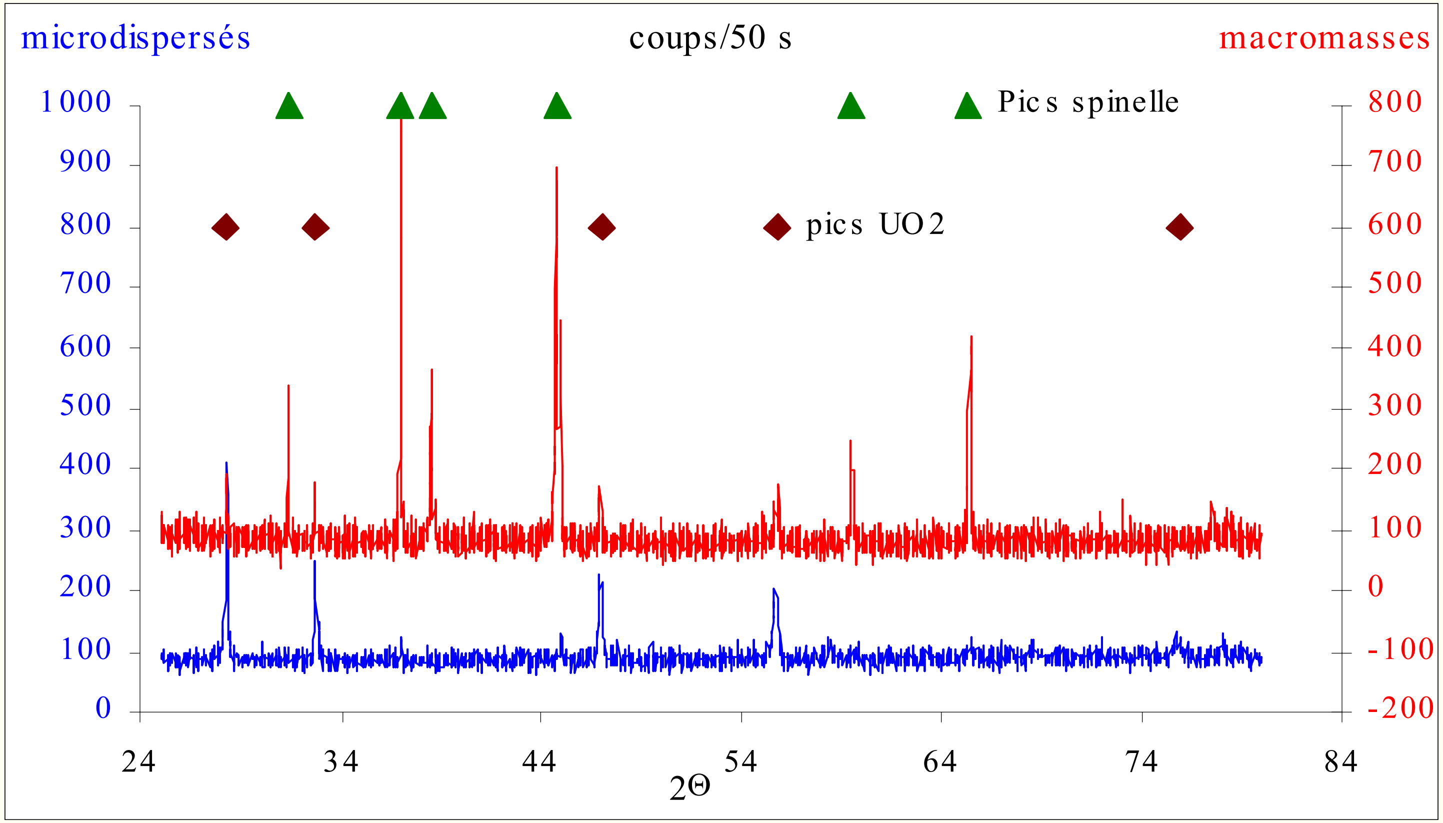
- Similar results as THERMHET:
  - Volume swelling of 11 vol%
  - Increase of the cladding diameter due to swelling

EFTTRA-T4ter shows similar results

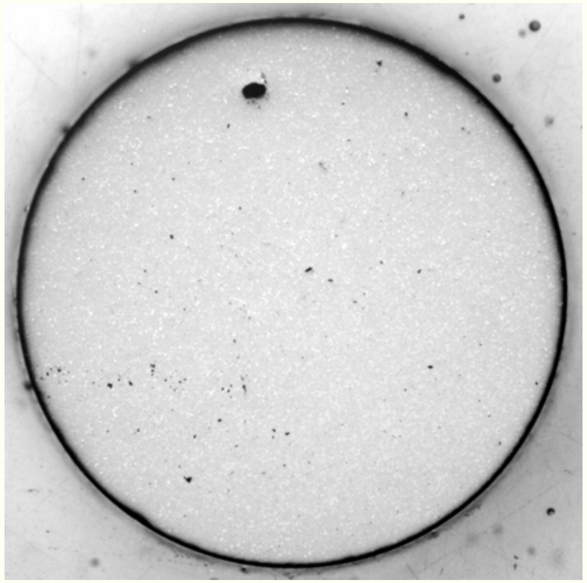


X-ray diffraction shows amorphisation of spinel

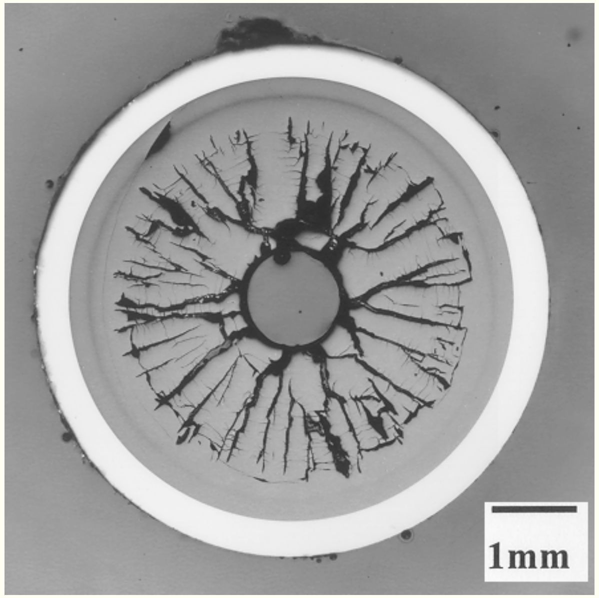
- Consequences of the spinel damaged by FP
  - Amorphisation of the crystal structure



Annealing of fission product damage



MATINA1
T≈1673K
BU = 1.27 at%



THERMHET
T<1373K
BU = 1.3 at%

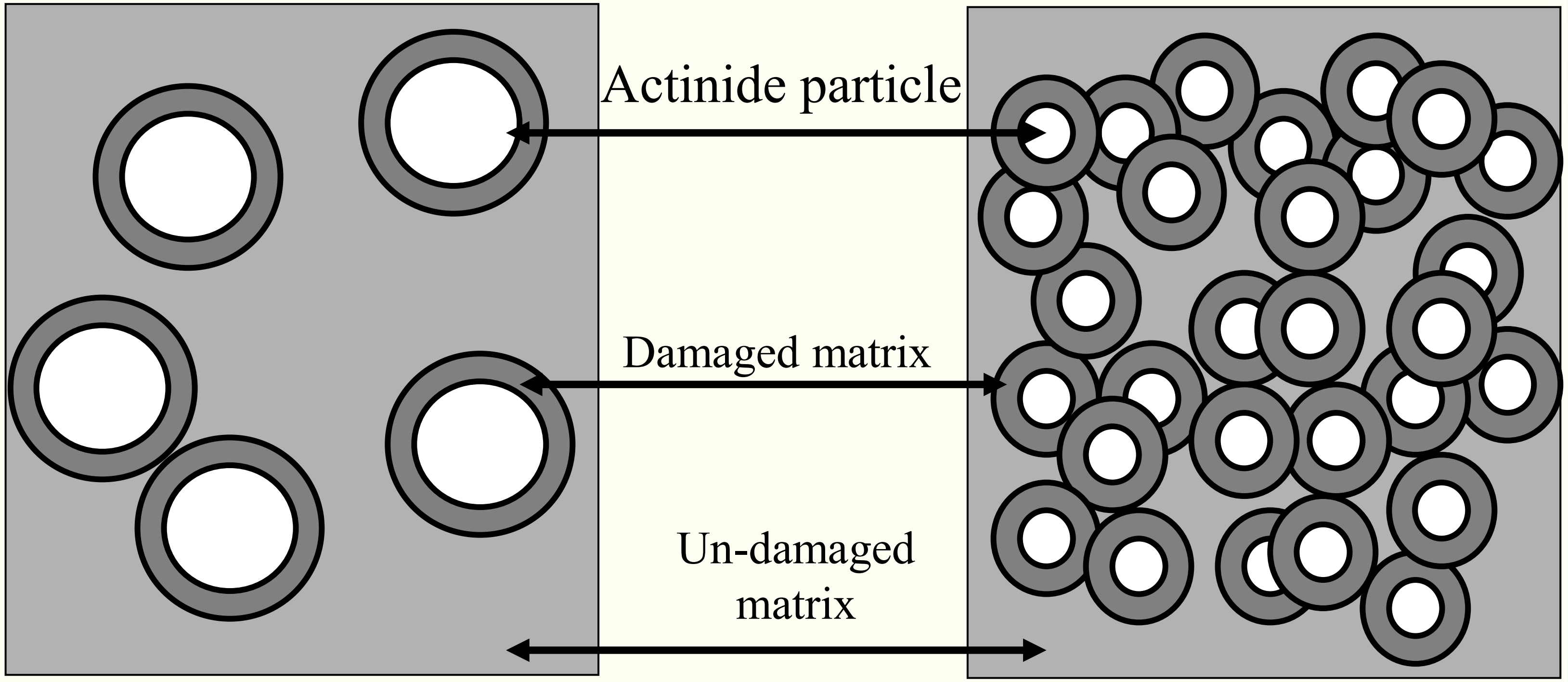
* FP damages annealed at temperature above 1100°C

Conclusions for spinel

- * Micro-dispersed actinide phases in spinel
  - – Spinel is very sensitive to high energy fission products
  - – Large swelling due to amorphisation of spinel
    - – THERMHET irradiation, J. Noirot et al., J. Nucl. Mater. **320** (2003), 117
    - – EFTTRA T4ter, Klaassen et al. J. Nucl. Mater. **319** (2003), 108
  - – At high irradiation temperatures, FP damage may be annealed
  - – Helium production (in transmutation) gives high porosity and additional swelling

- * Macro-dispersed actinide phases in spinel may offer better irradiation performance

Dispersion of the actinide phase



Actinide particle

Damaged matrix

Un-damaged matrix

MACRO

Fission product damage confined to the fissile particle

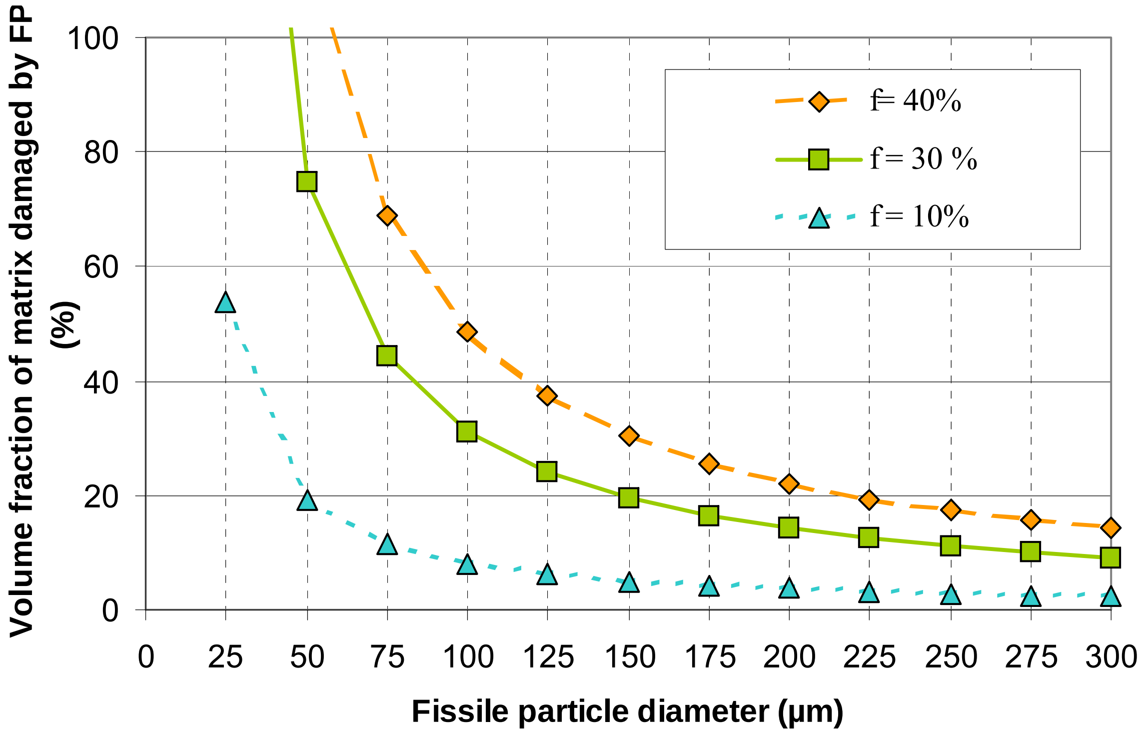
MICRO

Fission product damage in the inert matrix as well.

Dispersion of the actinide phase

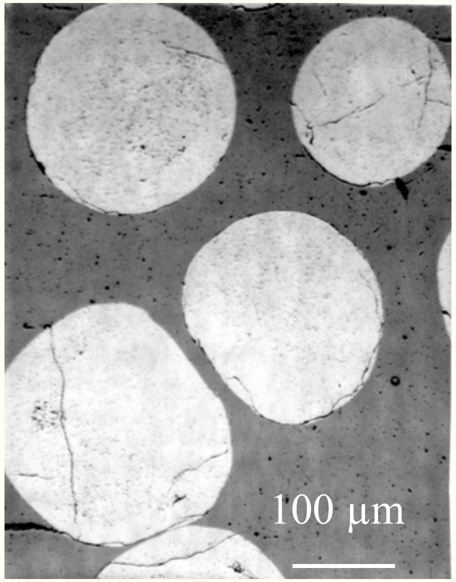
- Damage by FP is localised within a shell of 10µm around the fissile particle
  - Large fissile particles limits the matrix damaging

f=volume fraction of fissile particles in the matrix

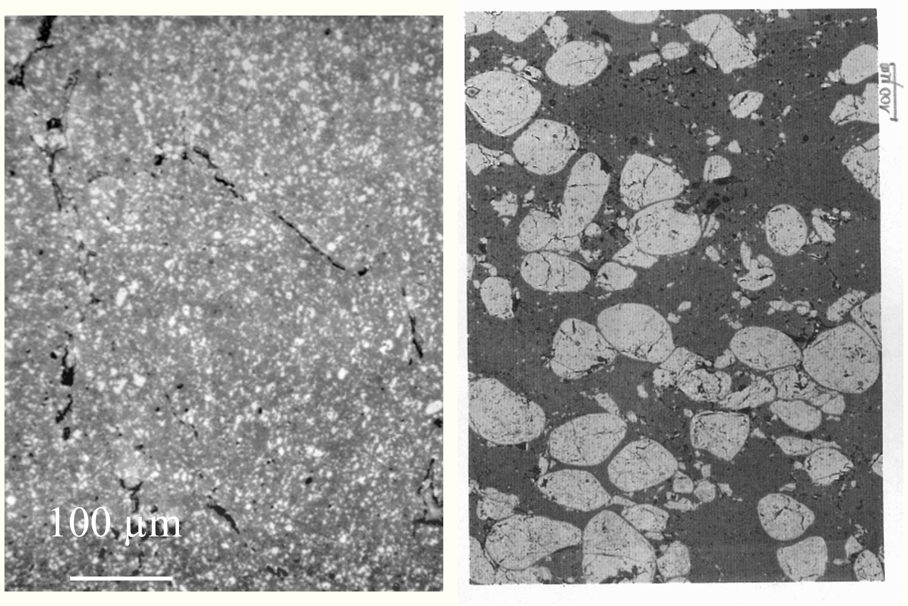


Micro- vs Macro-dispersion of the actinide phase

- * A strict control of the distribution and size of the fissile particles is required
  - – Impact on the fabrication process not negligible



No fuel swelling



Swelling of ~15 vol%

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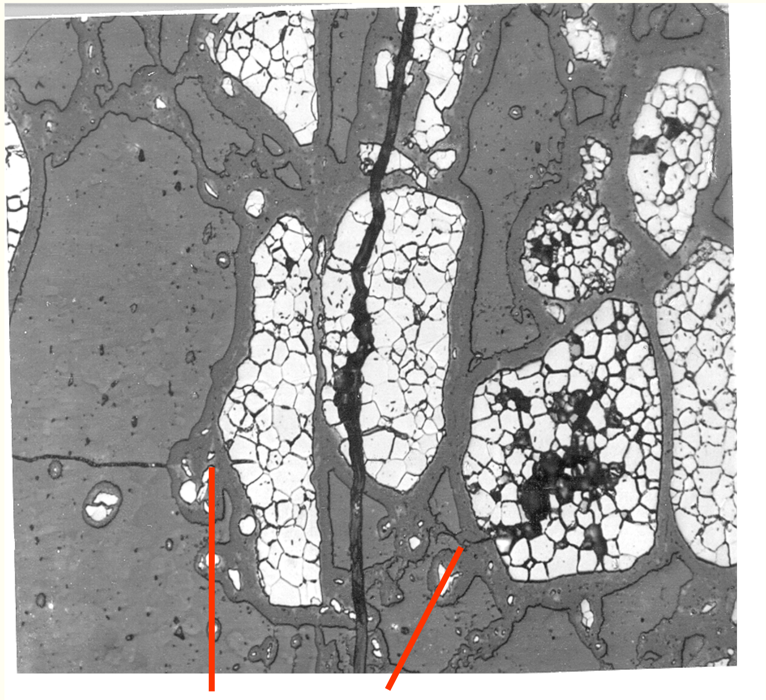
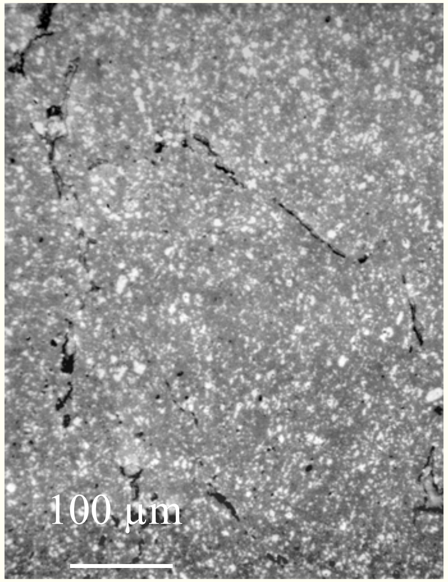
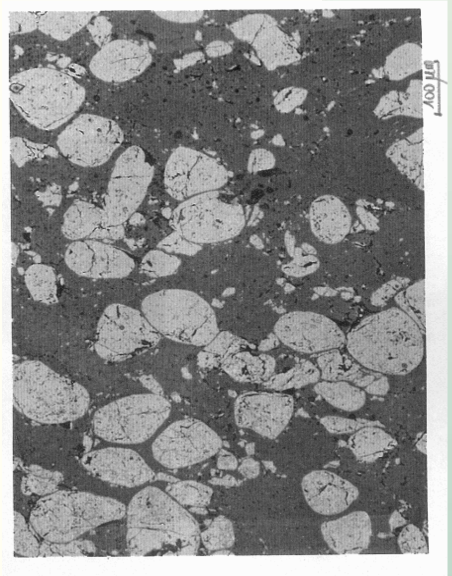


Matrix damaging around macromasses





Swelling of ~15 vol%

OTTO = Once Through incineration

* Geo-chemically stable inert matrices for Pu-burning

- Insoluble in nitric acid
- Suitable for final storage

* Zirconia-based IMF

- Cubic phase, i.e. single phase system (Zr,Y,Pu,Er)O_{2-x}
- Excellent irradiation performance
- Low thermal conductivity

* Spinel matrix with (Zr,Y,Pu,Er)O_{2-x} compound

- Two phase system with (Zr,Y,Pu,Er)O_{2-x}
- Micro-dispersed (fissile particles < 25 μm) and micro-dispersion tested (fissile particles ~ 200-250 μm)

OTTO: Joint project NRG / PSI / JAERI *

Cap sule	Composition	Dispersion	Fissile phase	Pu_{fis} (BOI)	Supplier	Instrum entation
			vol%	g/cm^3		
1	$(Zr,Y,Pu,Er)O_2$	Solid solution	single phase	0.37	PSI	-
2	$(Zr,Y,Pu,U)O_2$	Solid solution	single phase	0.34	JAERI	-
3	$(Zr,Y,Pu,Er)O_2+MgAl_2O_4$	<25 µm	20.8	0.32	NRG	TC
4	$(Zr,Y,Pu,U)O_2+MgAl_2O_4$	<25 µm	20.7	0.31	JAERI	TC
5	$(Zr,Y,Pu,Er)O_2+MgAl_2O_4$	200-250 µm	19.7	0.31	NRG	TC
6	$(Zr,Y,Pu,U)O_2+MgAl_2O_4$	200-250 µm	19.7	0.30	JAERI	-
7	$(Pu,U)O_2$	Solid solution	single phase	0.39	PSI	TC

OTTO studies:

Solid solution Micro-dispersed Macro-dispersed

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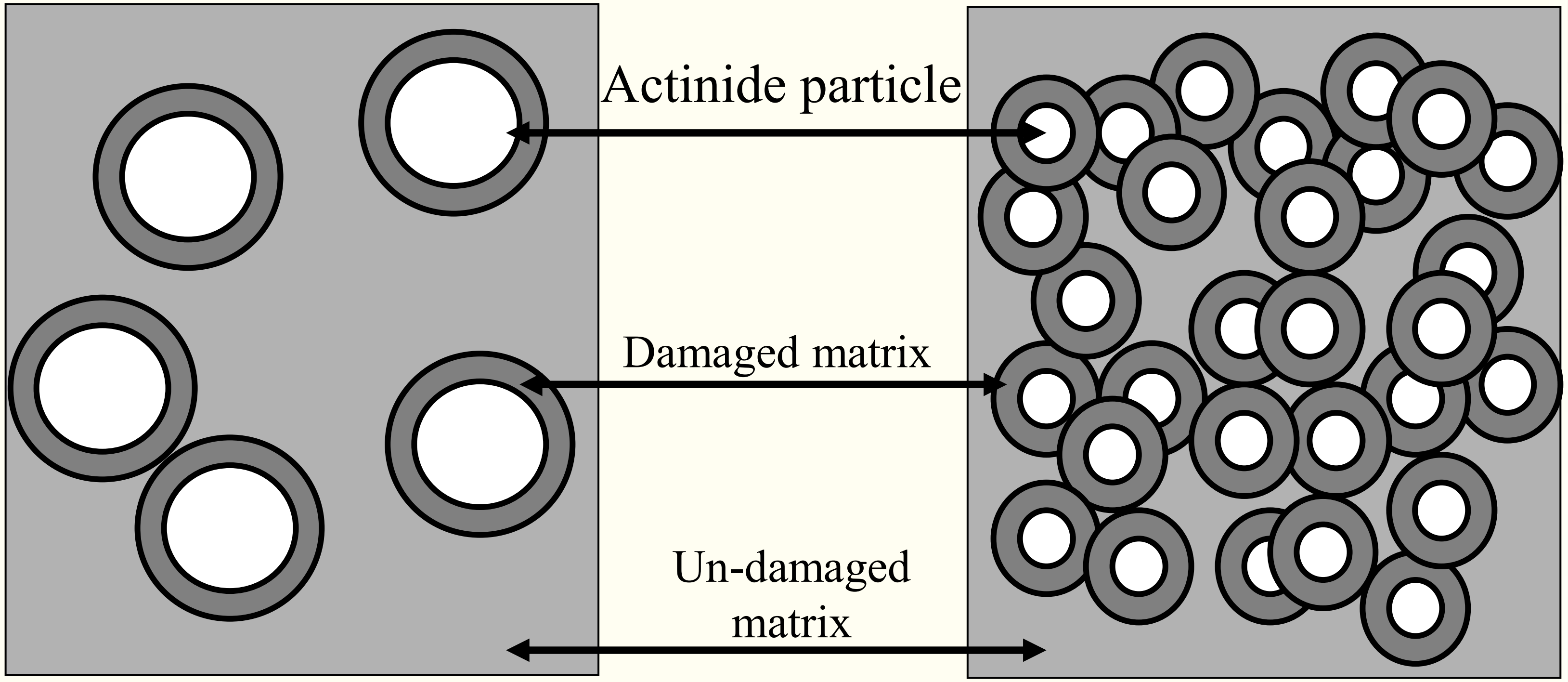
* Now JAEA

Dispersion of the actinide phase



Actinide particle

Damaged matrix

Un-damaged matrix

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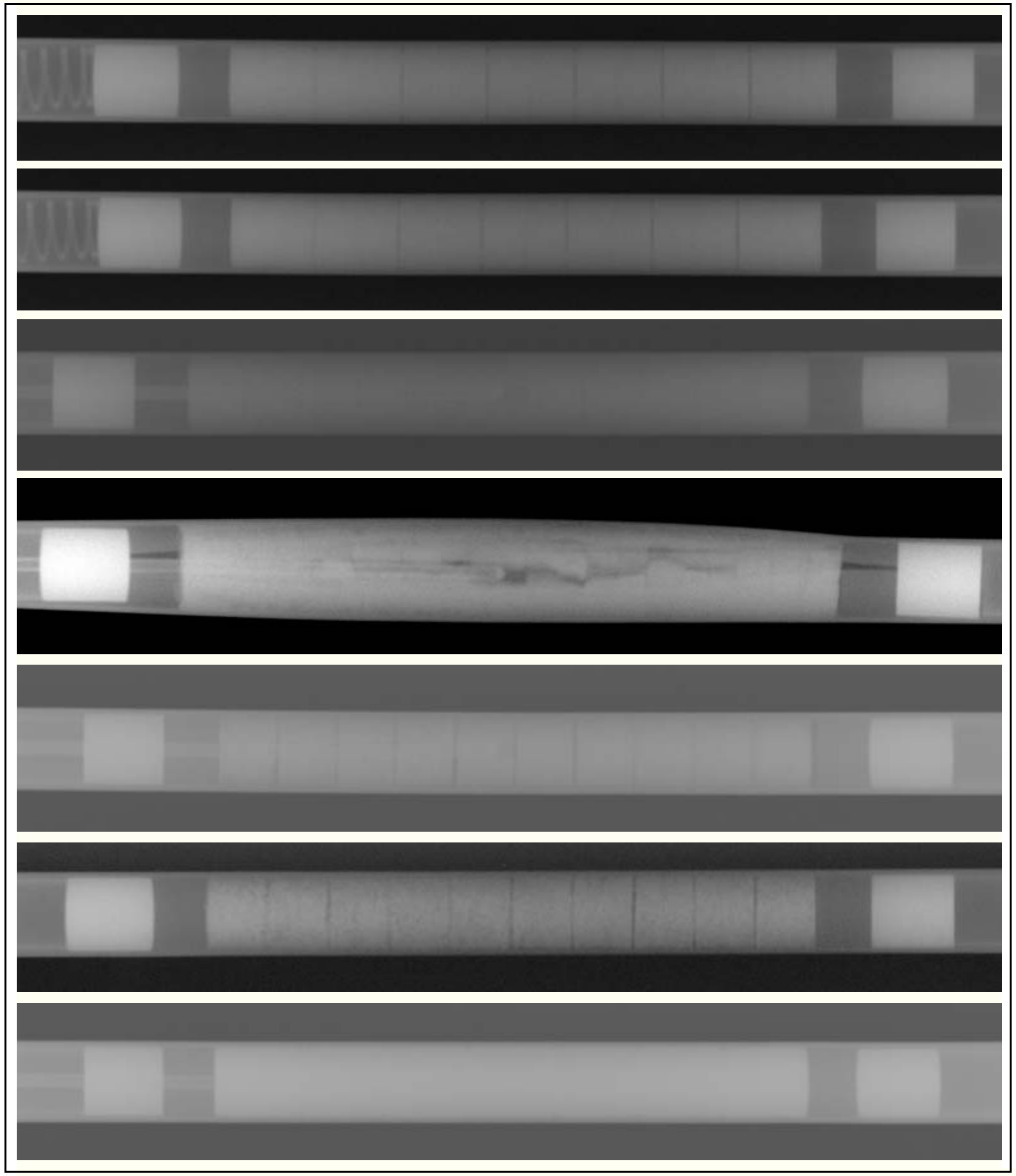
X-ray images after irradiation



Zirconia-based IMF

No swelling

Behaviour like UO_2

Spinel-based IMF

Micro-dispersed

Swelling due to amorphisation

Spinel-based IMF

Macro-dispersed

No swelling in spinel

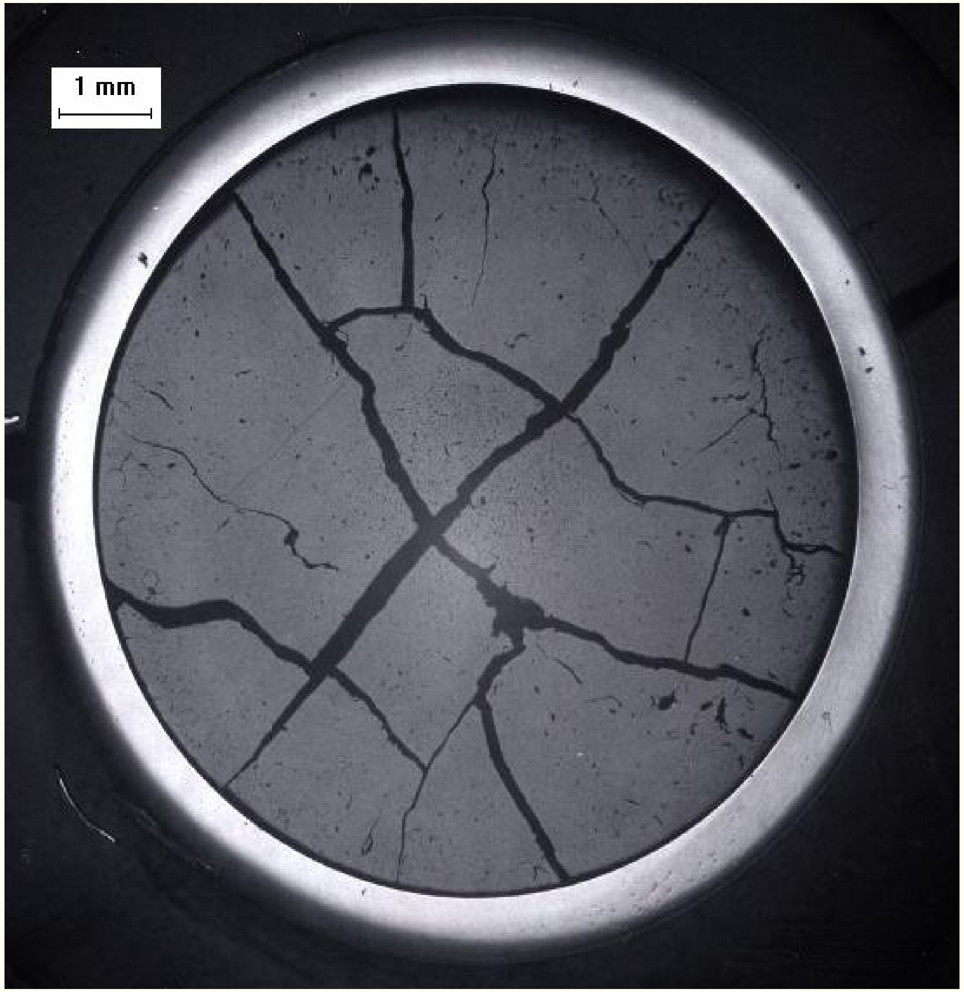
MOX-reference

$(Zr,Y,Pu,Er)O_{2-x}$ is a good IMF

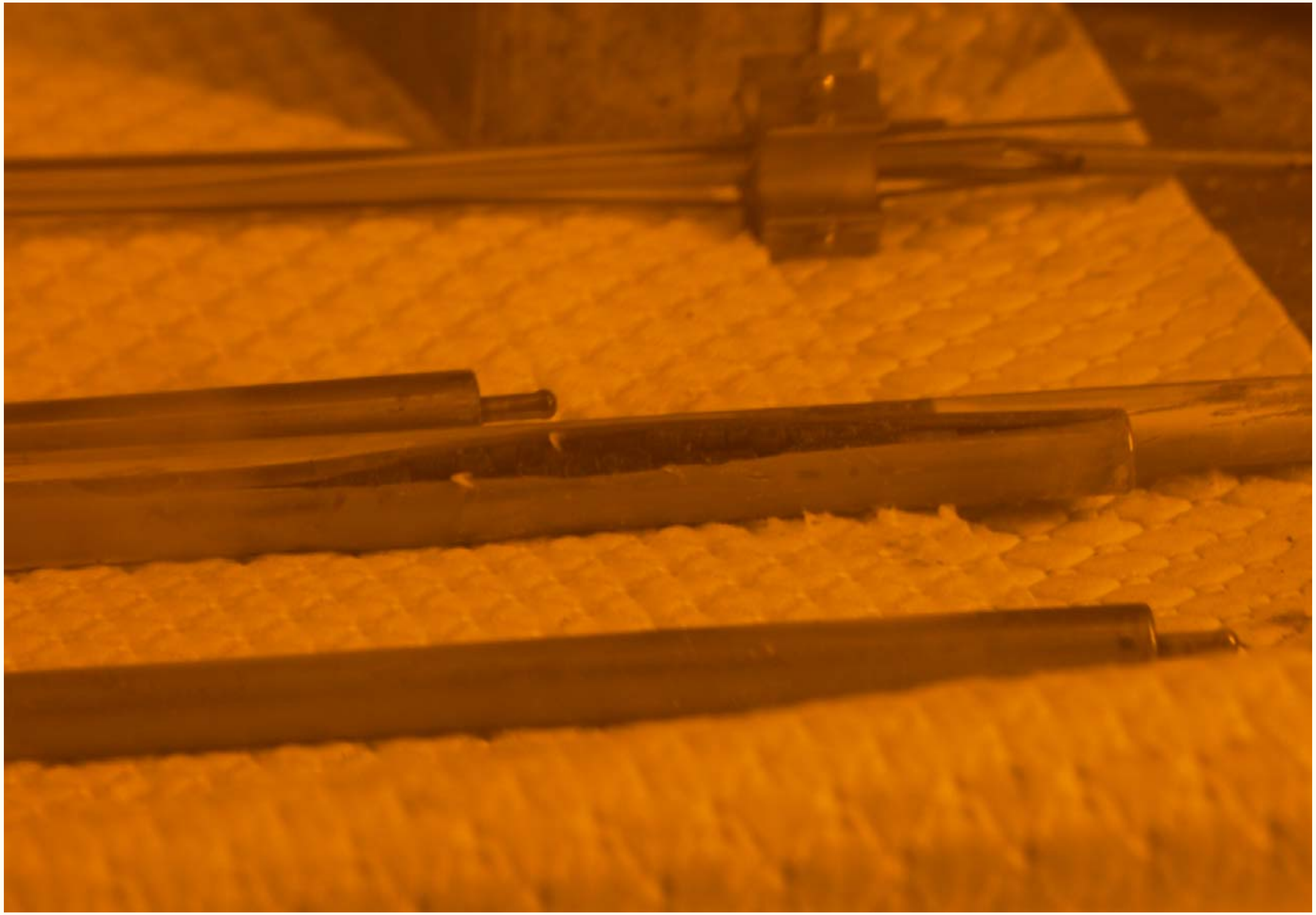
No swelling

Behaviour similar to UO_2

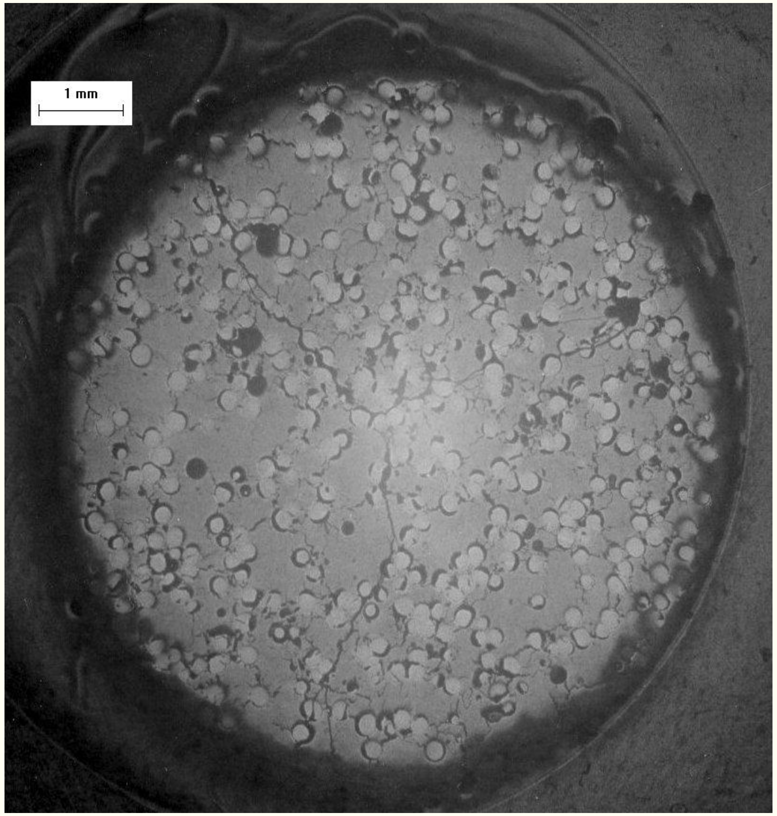
Rather high fission gas release:
6–9 % at a burn-up ~ 20 GWd/tU (eq.)

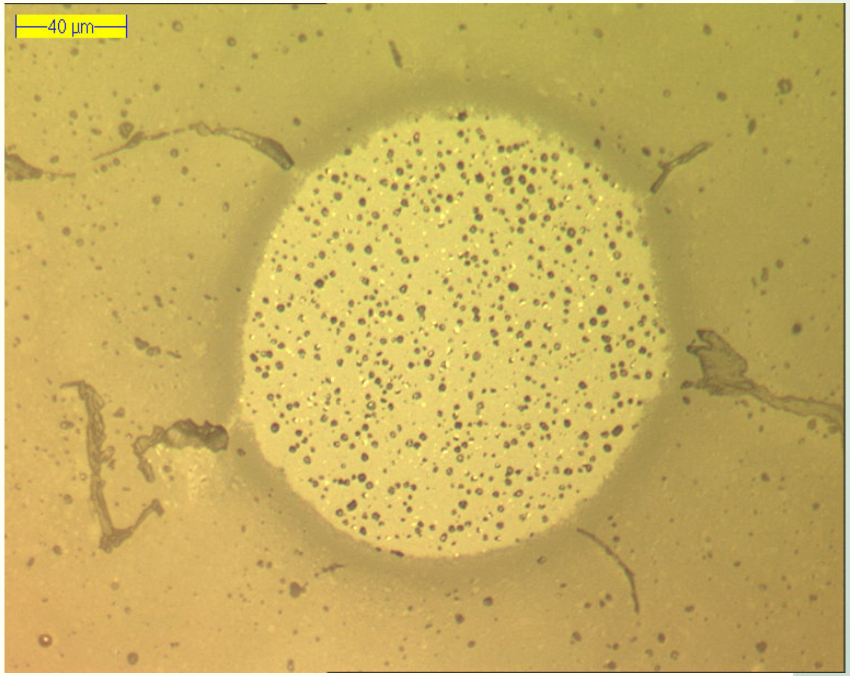


Capsule 4 cladding failure



Macro-dispersion in spinel

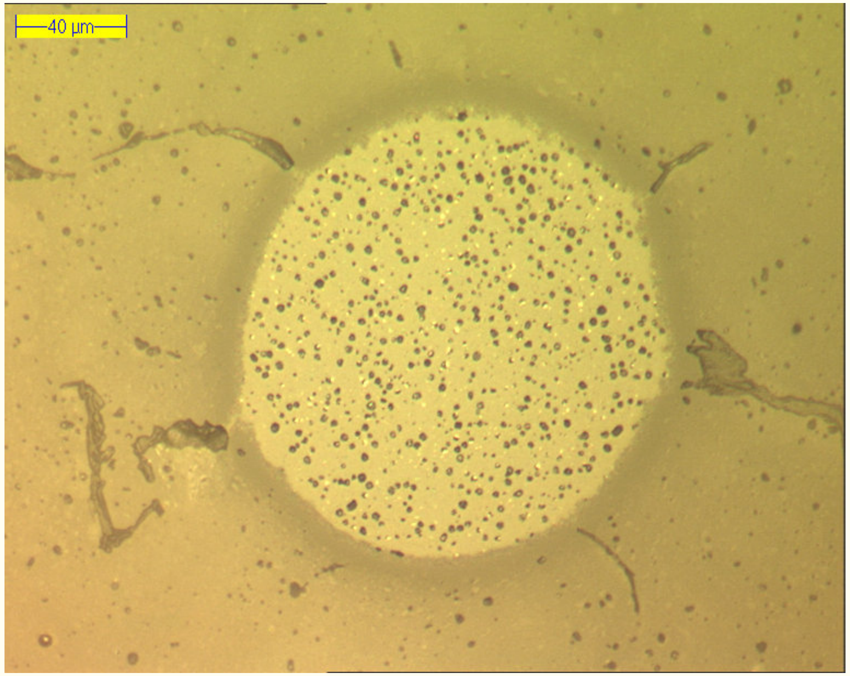




Halo around the fissile particle
due to f.p. damage in spinel

Macro-dispersed zirconia in spinel

Macro-dispersion limits irradiation damage from fission products



Halo around the fissile particle
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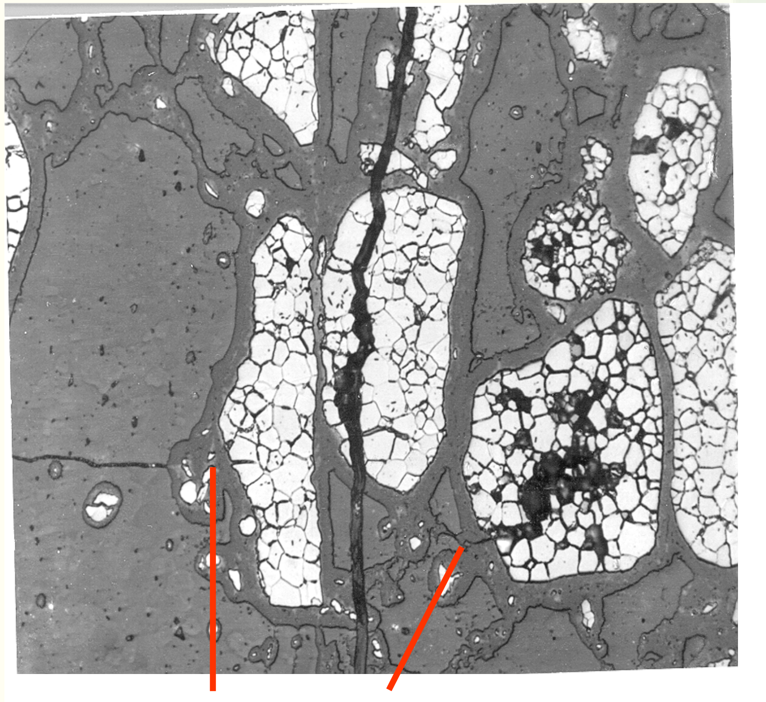


Matrix damaging
around
macromasses

OTTO irradiation, conclusions

* The OTTO irradiation was a successful test of Inert Matrix Materials (zirconia, spinel) to burn plutonium

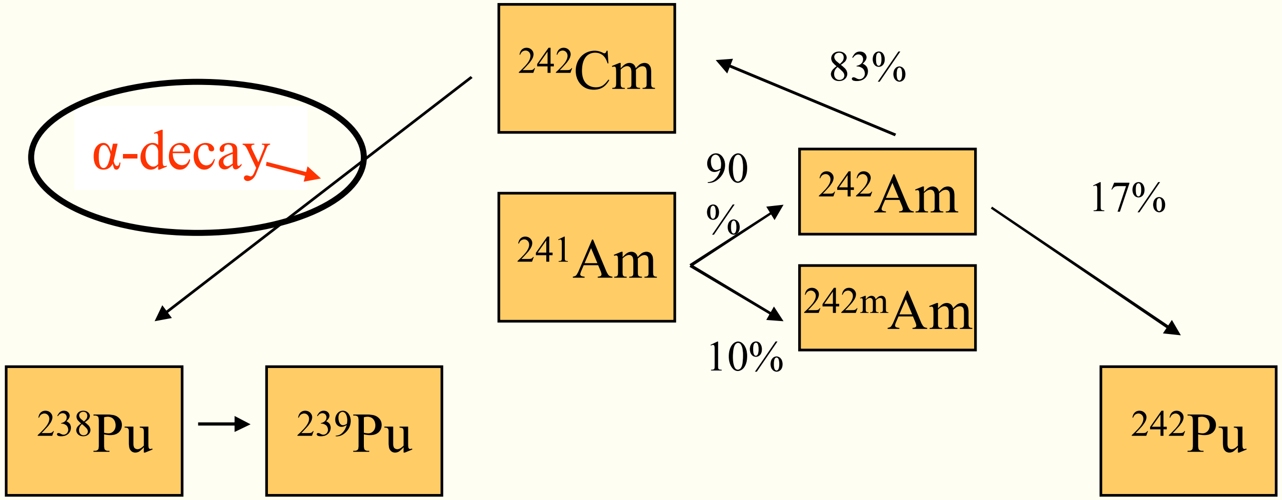
* An actinide depletion of 50% and a Pu transmutation of 60% was reached

* The performance of the inert matrices can be rated as follows:

Zirconia-based inert matrix fuels	+	(solid solution)
Spinel-based inert matrix fuels:	+/-	(macro-dispersed)
	-	(micro-dispersed)

Am-241 transmutation

* Transmutation scheme for Am-241:
requires on average 2.5 neutrons
produces 0.75 He-atom per initial Am-atom



The helium production due to ^{242}Cm decay is one order of magnitude larger than fission gas production

Contributes to the fuel swelling and pin pressurisation

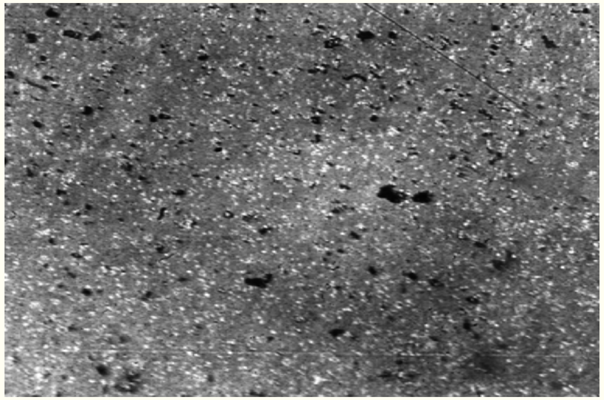
EFTTRA T4 & T4bis; two irradiations to study Am transmutation

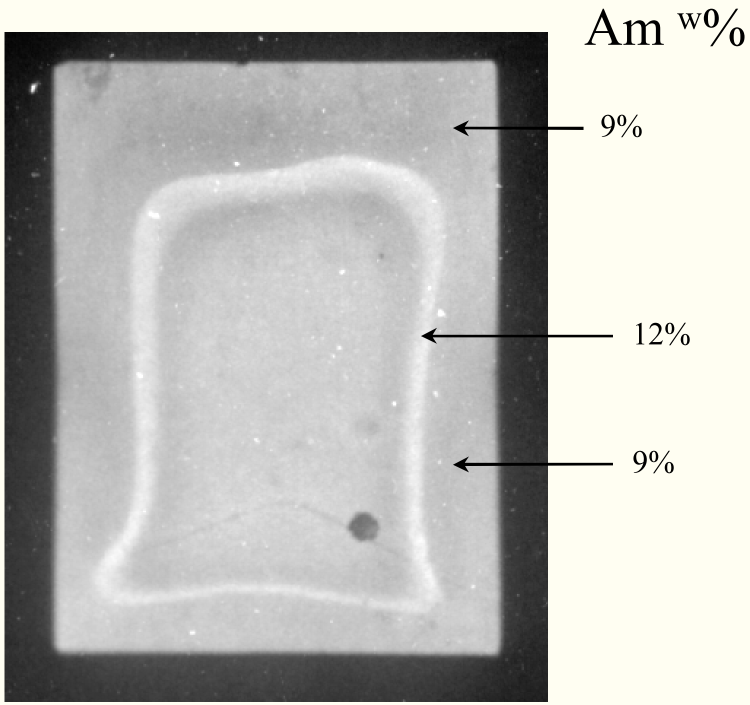
- * EFFTRA-T4 and T4bis experiments (1999) in HFR (EFTTRA group)
  - First irradiation tests of AmO_2 (11 wt%)+$MgAl_2O_4$
  - Prepared by infiltration of a porous spinel skeleton (ITU)
  - Homogeneous (micro)-dispersion of the fissile phase

    (3 μm-sized Am-particles)
  - Pellet diameter = 5.4 millimetre
  - Transmutation up to 99.8 % of Am-241 !!

EFTTRA T4 & T4bis; two irradiations to study Am transmutation

- * EFFTRA-T4 / T4bis pellets



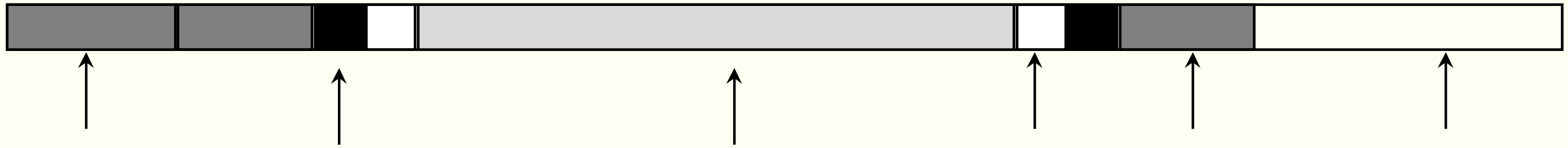


Alpha autoradiography

EFTTRA T4 & T4bis; two irradiations to study Am transmutation

	EFTTRA T4	EFTTRA T4 bis
Reactor	HFR Petten	HFR Petten
Irradiation time	358.4 FPD	652.6 FPD
Fissile phase	AmO_x	AmO_x
Fissile content	12.5 wt% (11.2 wt% ^{241}Am)	12.5 wt% (11.2 wt% ^{241}Am)
Thermal fluence ($E < 0.7$ eV)	$5.8 \cdot 10^{25}$ m^{-2}	$9.7 \cdot 10^{25}$ m^{-2}
Fast fluence ($E > 0.1$ MeV)	$16.8 \cdot 10^{25}$ m^{-2}	$21.3 \cdot 10^{25}$ m^{-2}
Actinide depletion	~ 28 %	~ 50 %
Burn-up	96 % (^{241}Am)	99.8 % (^{241}Am)
Power (W/cm)	40 / 70 (min/max)	40 / 65 (min/max)
Fuel central temperature	~ 700ºC	~ 700ºC

EFTTRA T4 pin after irradiation



bottom plug

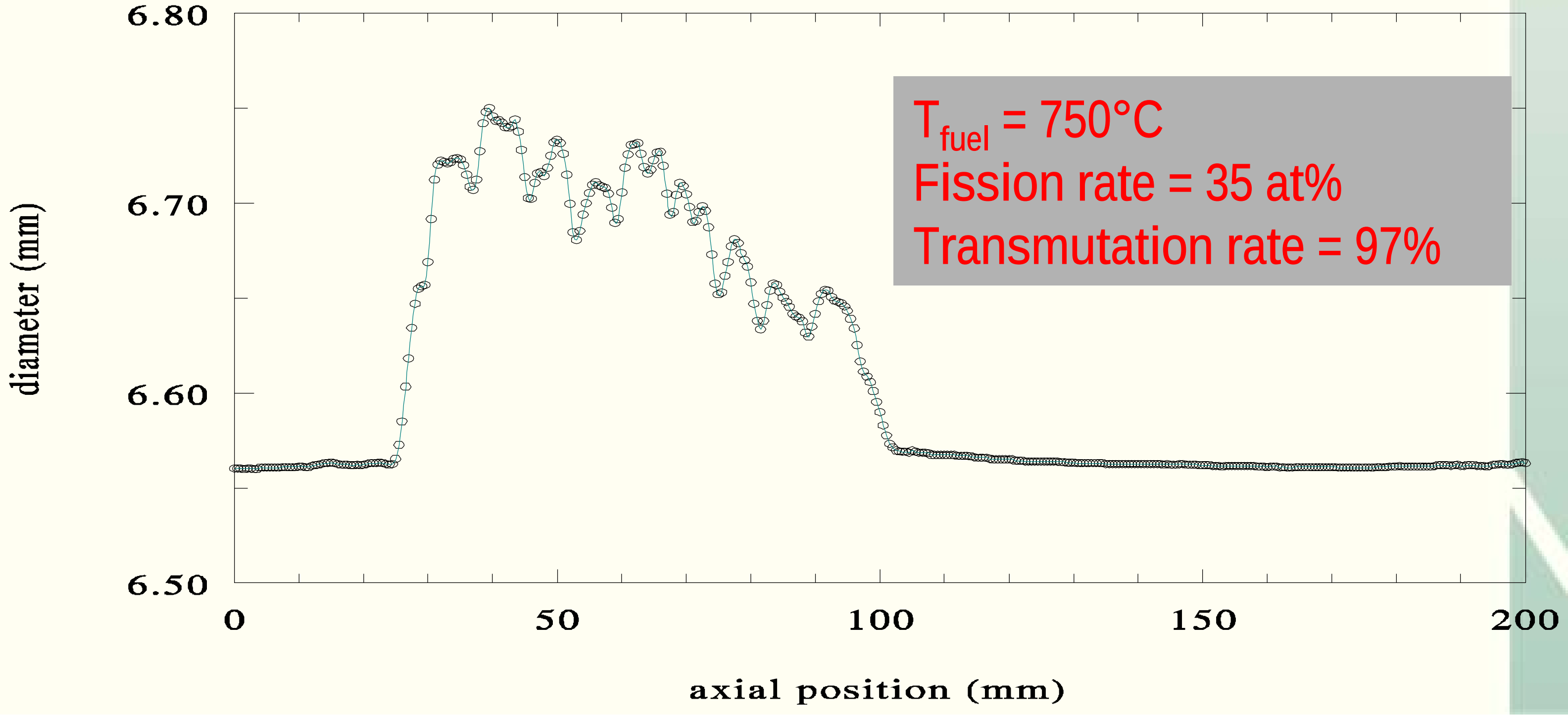
Hf pellet

americium pellets

spinel pellet

steel cilinder

spring in plenum



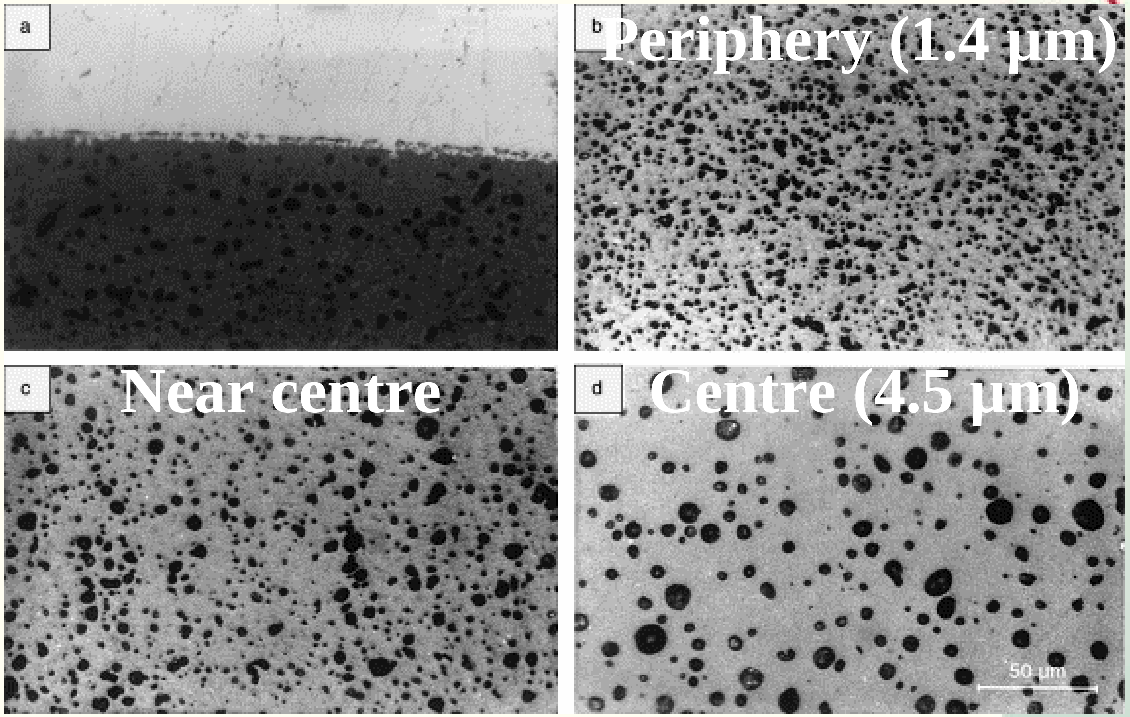
T_{fuel} = 750°C

Fission rate = 35 at%

Transmutation rate = 97%

Ceramographies of EFTTRA T4

T4
28 at% BU
Sw.=18 vol%
0,4g Am/cm^3
Released He = 20%
Released FG = 5%

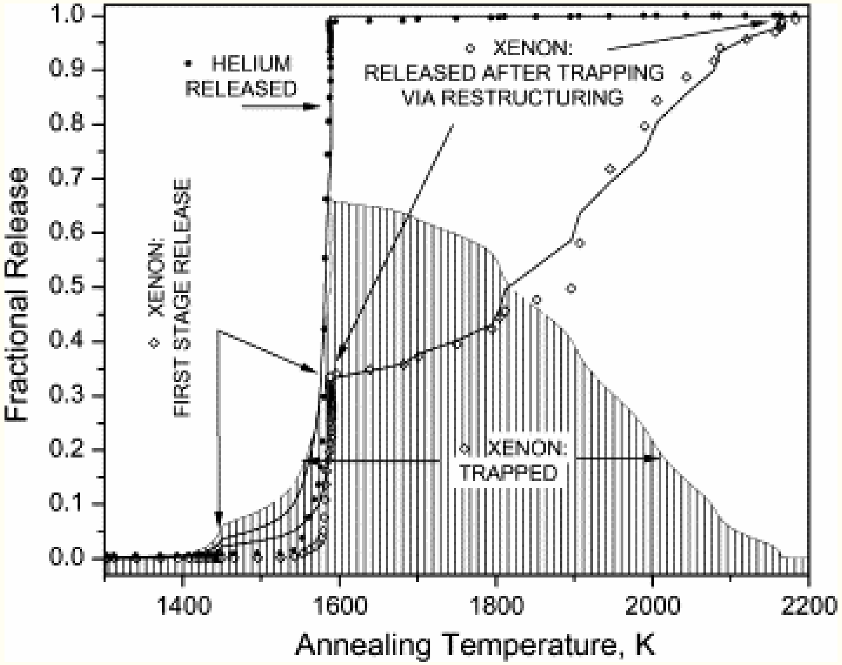


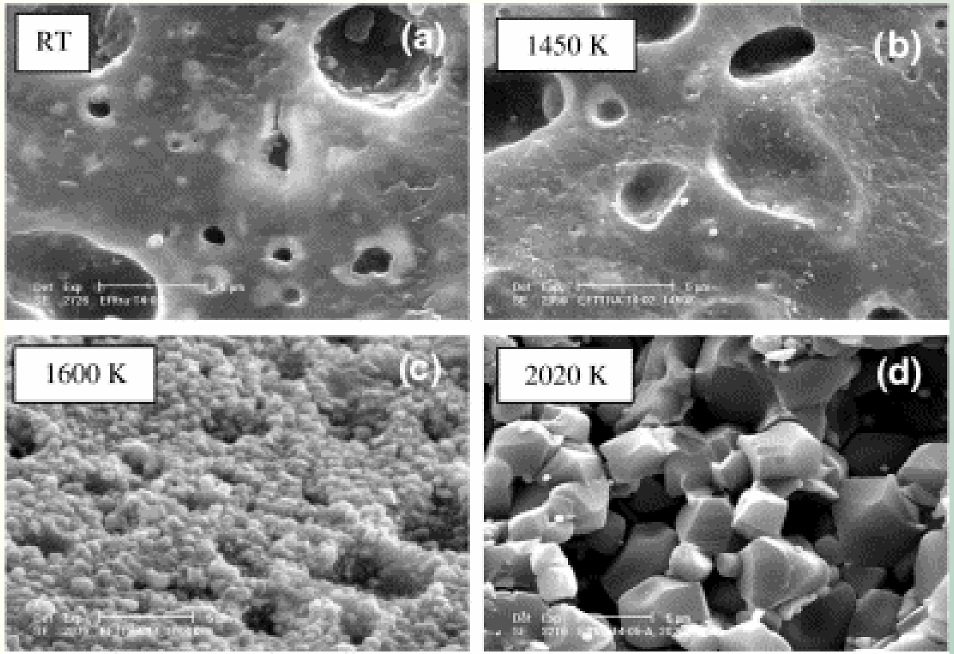
* The accumulation of He in gas bubbles seems to be the major cause of swelling (> #30vol%)

- This swelling rate should have favored the gas release thanks to the pore interconnection (creation of release paths)
  - As observed on nitride, metal, carbides and UOX or MOX fuels
- The abnormal retention could be due to the amorphisation/nanocristallisation of the crystal structure

Release of helium at higher temperatures

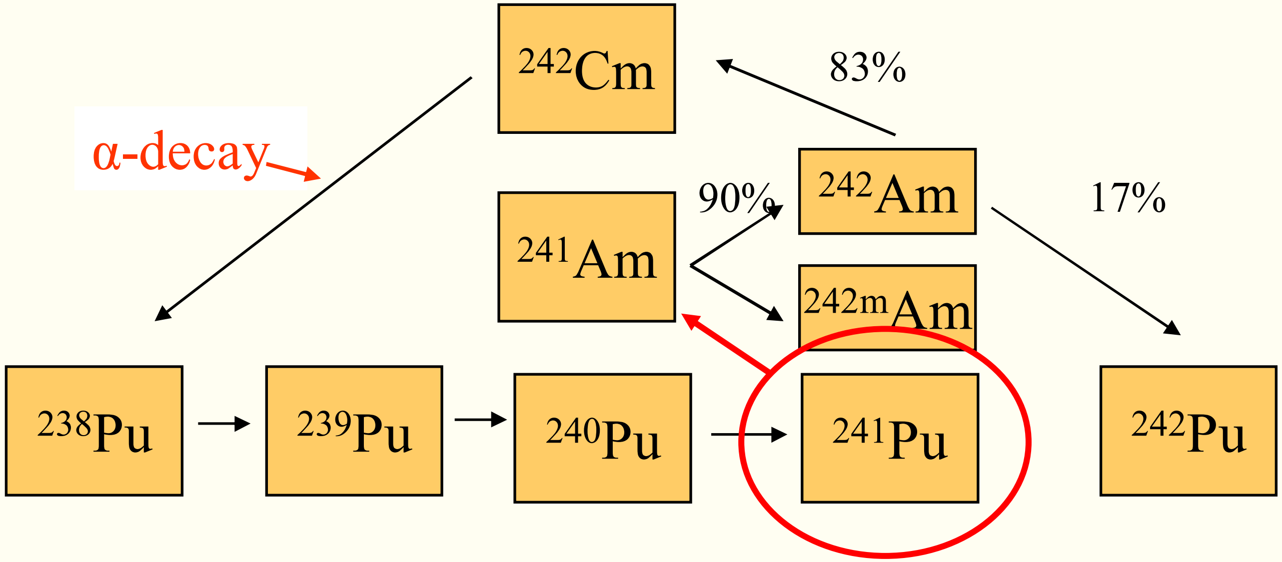
* Significant gas release is observed when IMF is heated at temperature around 1600K (out-of-pile observation)
* Correlation with the fuel restructuring (similar to the RIM effect)





Am-241 transmutation loop

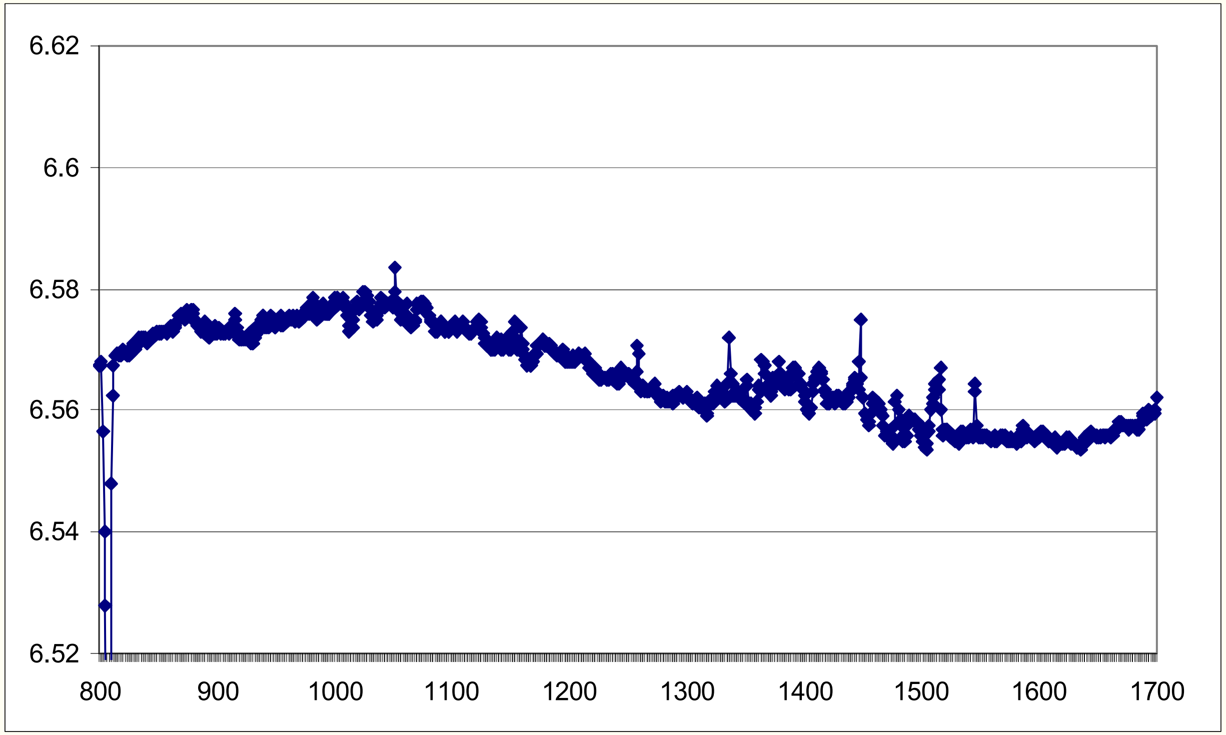
- In EFTTRA T4bis, Am-241 was transmuted for 99.8%
- Now the Am-concentration is increasing again
  - Decay of Pu-241 !!



Half-life 14.4 years

Comparison with MOX fuel containing MA
SUPERFACT experiment (1988) in Phénix

Fuel	P_{lin} (W/cm)		BU_{max} (at%)	T_{clad} (°C)	
	BOL	EOL	EOL	BOL	EOL
2% Np	**385**	**350**	**6.6**	**613**	**604**
2% Am	**379**	**346**	**6.48**	**612**	**603**
20%Am+20%Np	**174**	**286**	**4.1**	**603**	**595**
45% Np	**197**	**302**	**4.56**	**603**	**595**
$(U, Pu)O_2$	**402**	**353**	**7.38**	**618**	**609**

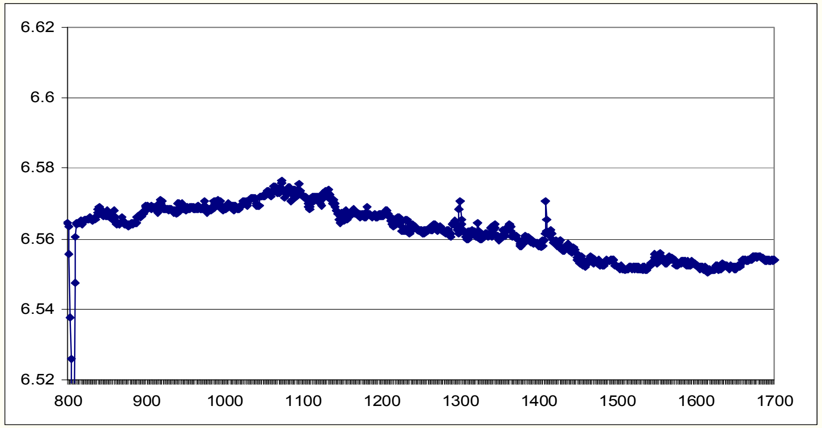


2% Np

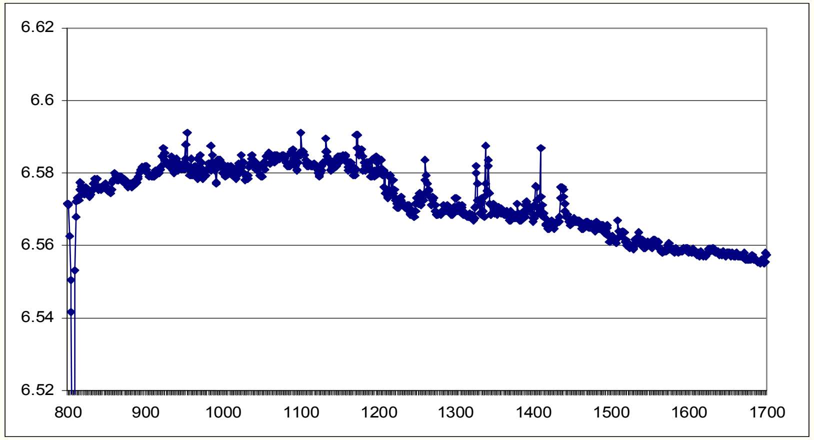
Profilometry

Comparison with MOX fuel containing MA
SUPERFACT experiment (1988) in Phénix

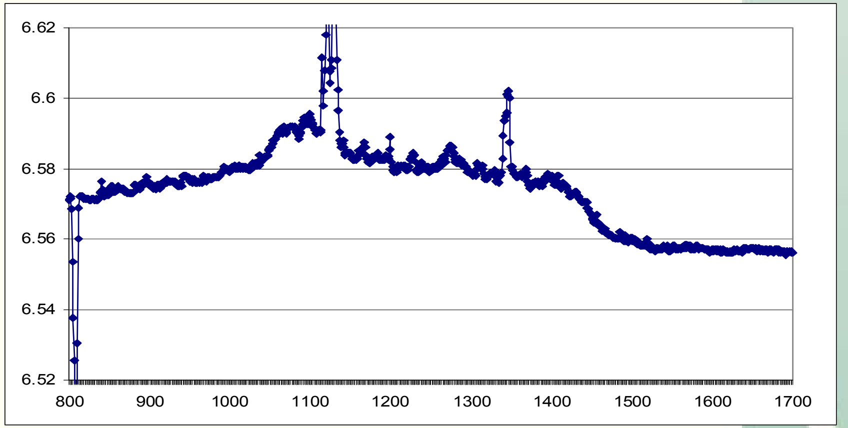
Profilometry



45% Np



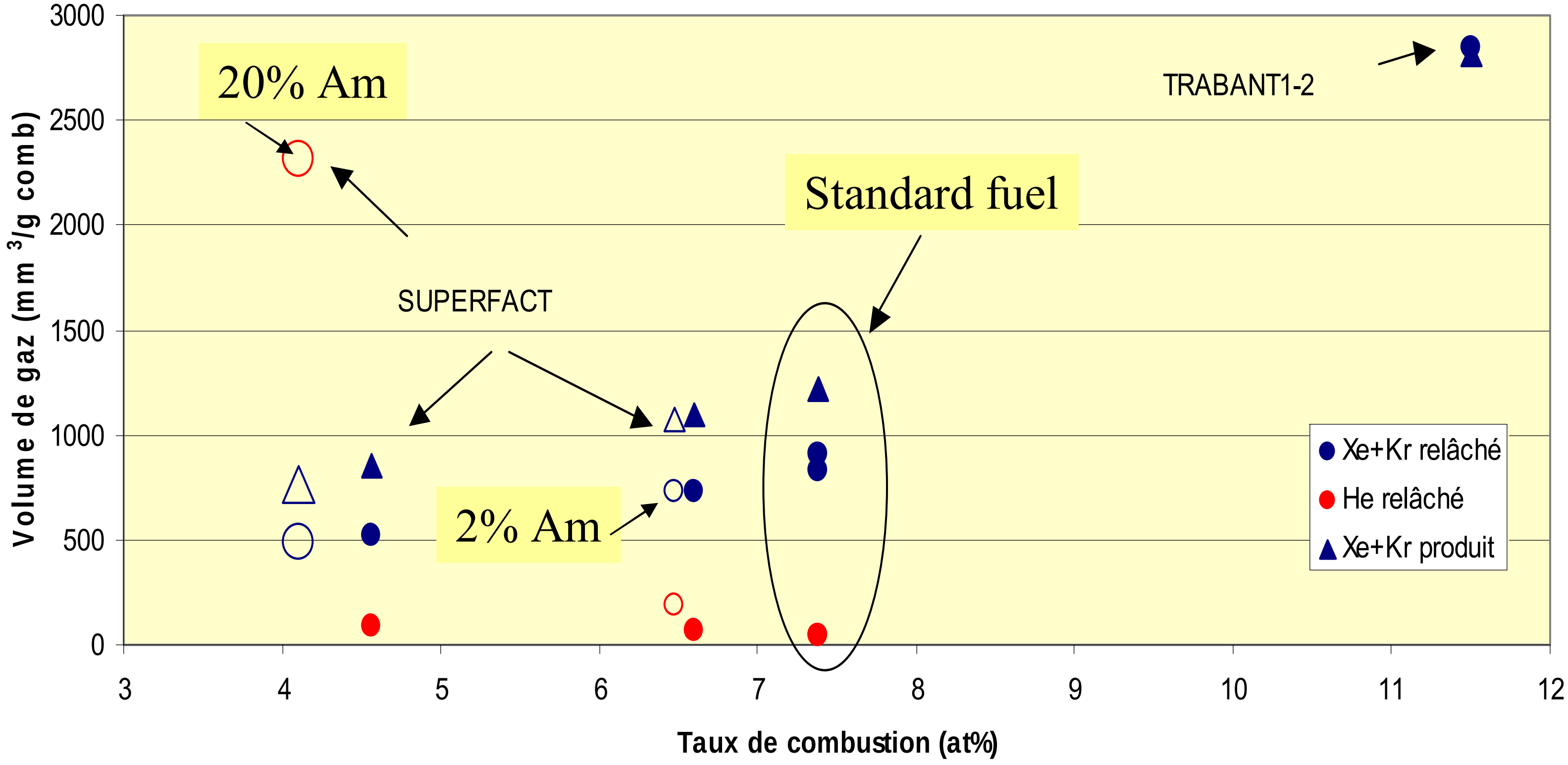
2% Am



20% Np+20% Am

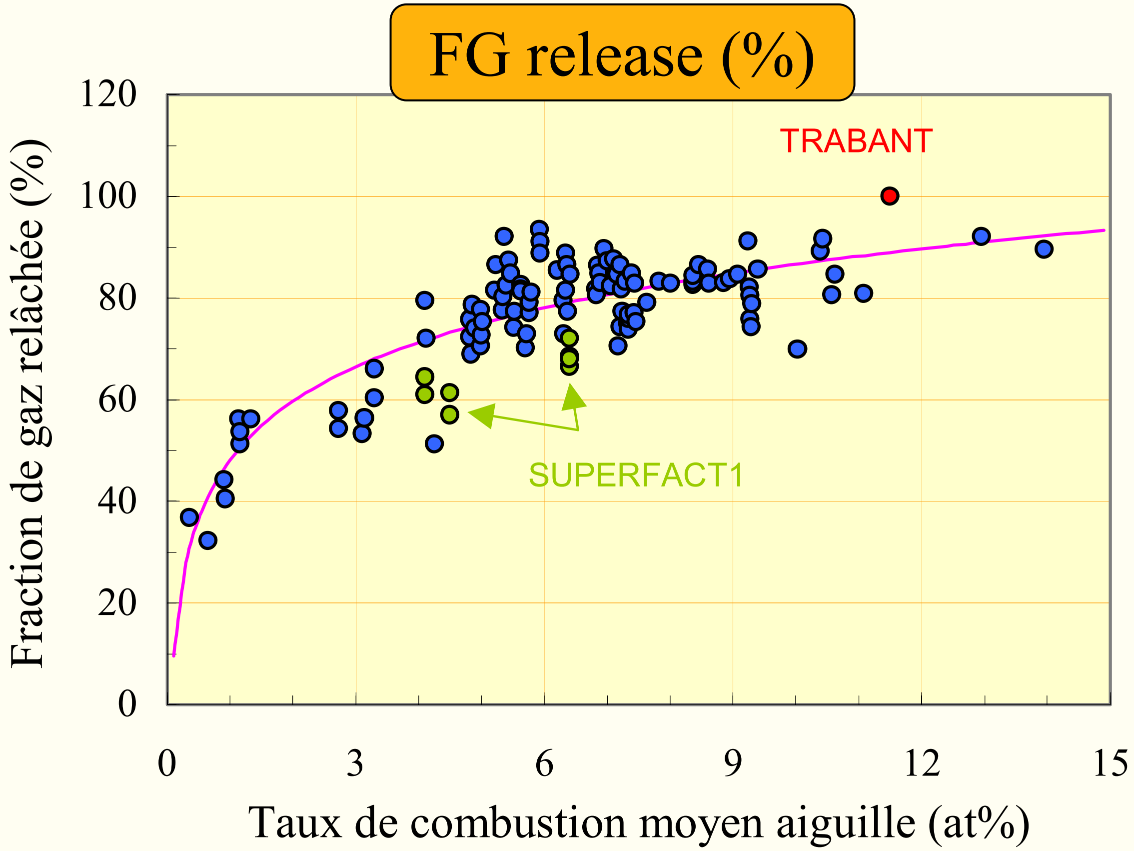
Comparison with MOX fuel containing Am and Np

mm³ of gas/g of fuel



- Fission gas release very similar to that of MOX fuel

Comparison with MOX fuel containing Am and Np



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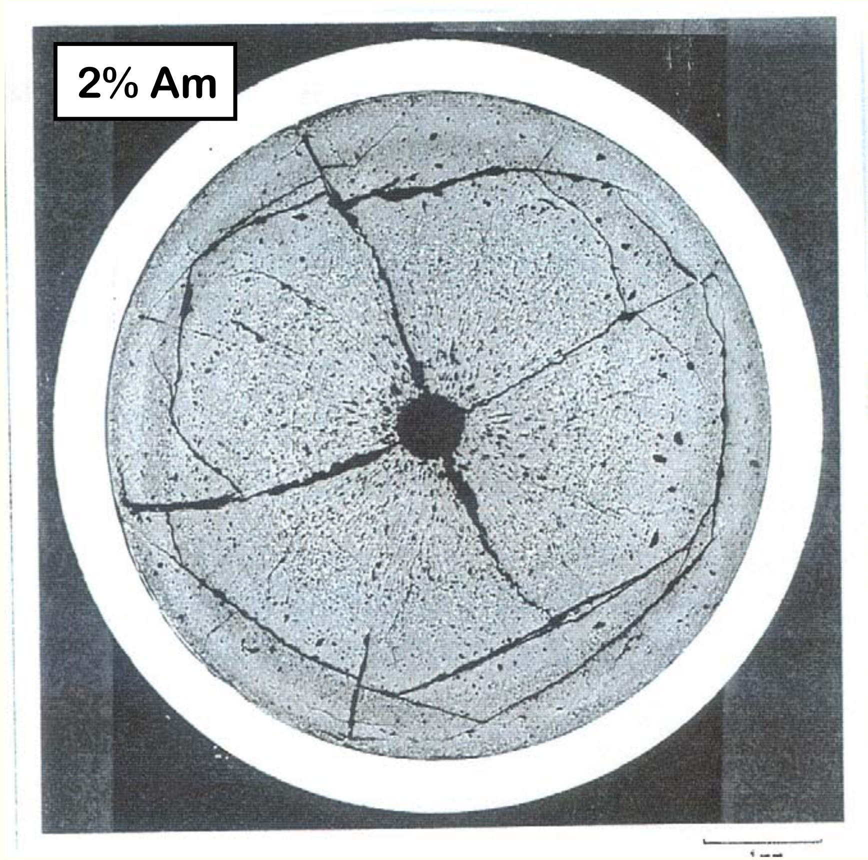
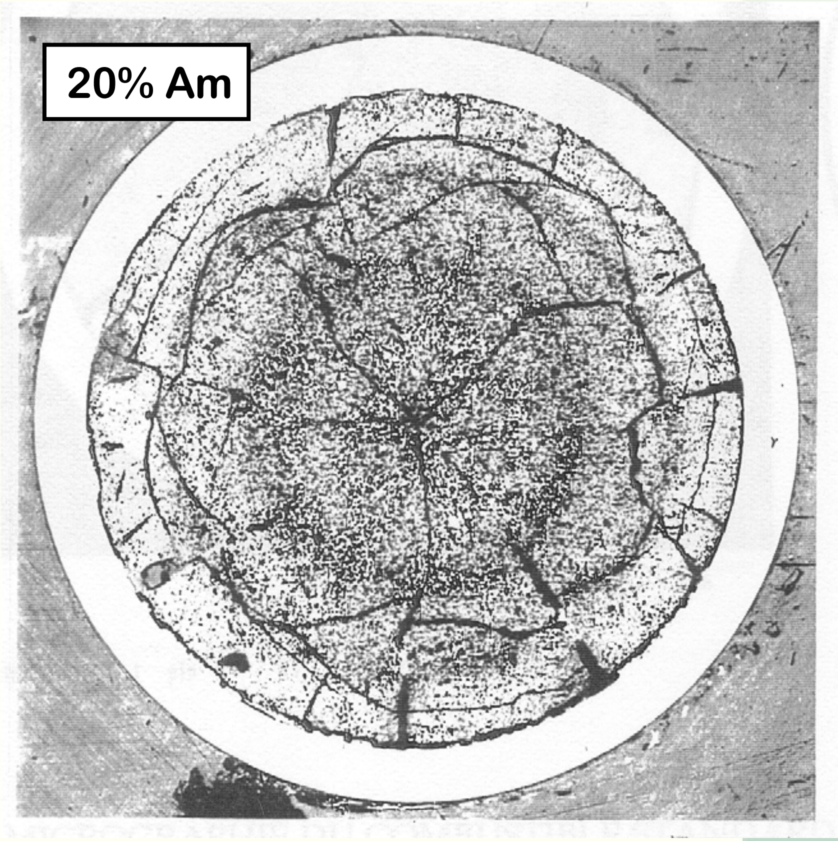


Low He production



High He production

Intermediate conclusion

* Swelling (>~15 vol%) due to the amorphisation of the crystal structure

- (TANOX, THERMHET, OTTO, EFTTRA T4)
- Sensitive matrices: **YAG ($Y_3Al_5O_{12}$) and spinel**
- Gas retention enhancement

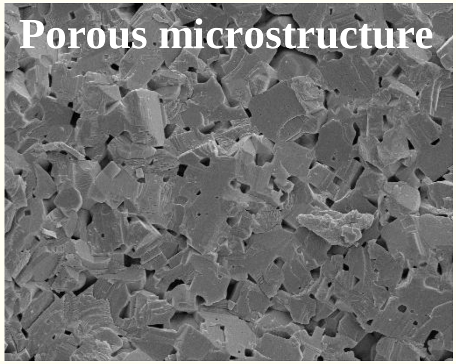
* Macrodispersion (dispersion of MA compounds >100 µm) limits the damaged volume of inert matrix (THERMHET, ROX, OTTO, TANOX)

- Strict control of the particle dispersion and particle size
- More complex fabrication process
- Content of particles limited

* Damage annealing above 1200°C (MATINA1)

- Swelling limited to 5 vol%

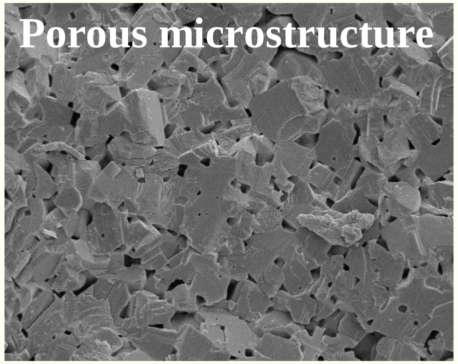
Intermediate conclusion



(Zr, Y, Pu, Er)O_2 in
HFR
50 at% BU
C. HELLWIG, U. KASEMEYER
Journal of Nuclear Materials 319 (2003) 87-94

* Objectives of the current R&D:
- Optimise the transmutation rate and capacity of the TRU-fuels
  - Reach higher fission rate (up to 90% in moderated flux)
  - Increase the MA content (from 0.7 to ~5 g/cm^3)

* Directions:
- Enhance the He release during irradiation
  - By providing release paths as soon as the irradiation starts
  - New IMF microstructure with open porosity
  - By increasing temperature with Pu addition
    - (Pu,Am)O_2 + inert matrix



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  - Increase the MA content (from 0.7 to ~5 g/cm^3)

* Directions:
- Enhance the He release during irradiation
- Select inert matrices more resistant to the FP damage at temperatures < 1000-1100°C
  - MgO, ZrO_2, Mo
- Test new Am compounds to reduce oxygen potential and limit the clad corrosion
  - $Am_2Zr_2O_7$ (pyrochlore)

Fabrication processes for ADS fuels

- A key issue
  - High neutron source (of ^{238}Pu and ^{244}Cm)
  - High thermal constraint (heat release of ^{238}Pu and Cm isotopes)

Isotop	Period (year)	Activity ($x10^{10}$ Bq/g)	Neutron source ($n.g^{-1}.s^{-1}$)	Heat release (W/g)	Ingestion dose factor ($x10^{3}$ Sv/g)
Np 237	$2,14x10^{6}$	0,00261	0	0	0,028
Pu 238	87,7	**63,4**	**2511**	0,568	**630**
Pu 239	24390	0,227	0,02	0,002	2,6
Pu 240	6540	0,843	**905,6**	0,007	9,8
Pu 241	14,3	**382**	0	0,003	90
Pu 242	$3,87x10^{5}$	0,0141	**1678**	0	0,15
Am241	432	**12,7**	1,2	0,115	**150**
Am 242m	152	**36,0**	150	0,004	**410**
Am 243	7380	0,736	0	0,007	8,8
Cm 243	28,5	**170**	0	1,685	**1300**
Cm 244	18,1	**299**	**5892**	2,825	**1800**
Cm 245	8530	0,633	**5448**	0,006	7,6

Fabrication processes for ADS fuels

No industrial experience is available today!

- * Minor actinides in solution
  - – 250 g $^{243+241}$Am /l : #25 W/l
    - – Typical concentration for a chemical operation of precipitation
  - – 250 g ^{244}Cm/l : #700 W/l
  - – Industrial experience : 10 W/l (FP concentrates at La Hague)
- * Minor actinides in solid state
  - – 1 m^2 of powder bed kept below 80°C must not exceed
    - ⇒ 0.5 mm of thickness with natural cooling (air, room temperature)
    - ⇒ 5 mm of thickness with forced cooling

Fabrication processes for ADS fuels

* Implementation in hot cells with heavy shielding and remote handling

- Reduction of the volume for cost reduction (process compaction)
  - Simple process
  - Limitation of the control steps
  - Limitation of liquid and solid waste treatment
- Development of robust processes
  - Limitation of the maintenance
  - Simple process

* Avoid powder handling

- More difficult to cool
- Risk of dust retention and accumulation

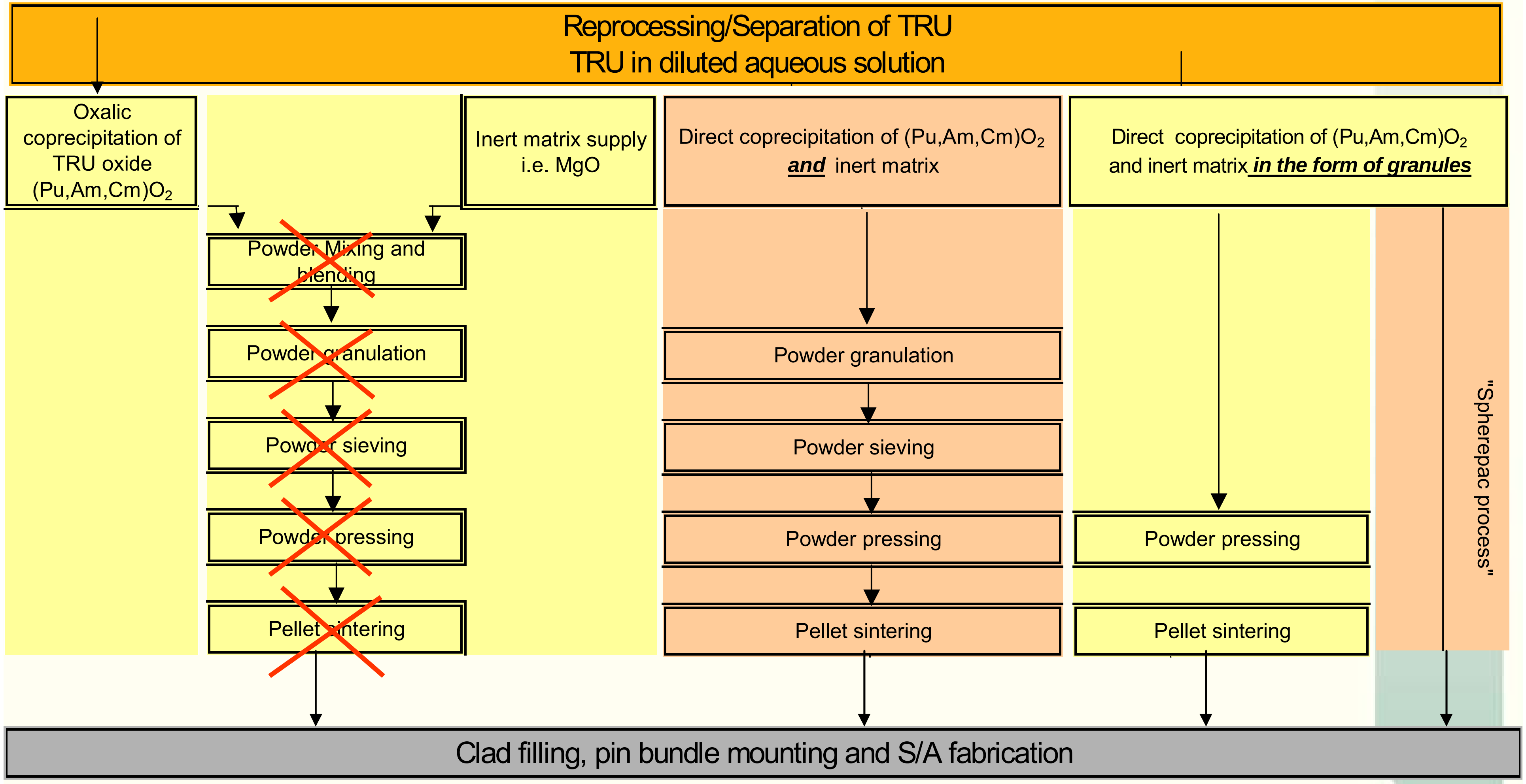
Fabrication processes for ADS fuels

- R&D limited by the hot lab availability
  - In Europe, only 2 labs (CEA, ITU)
  - In Japan, 1 lab (JAEA)
  - In USA, 3 labs (INL, LANL, ORNL)

- R&D limited by the hot lab capacity

	ATALANTE	MA-LAB	TRU-HITEC
^{241}Am	1700 g	2000 g	10 g
^{242}Am	2 g	0,1 g	-
^{243}Am	30 kg	34 g	0,2 g
^{243}Cm	110 g	0,14 g	-
^{244}Cm	70 g	5 g	0,02 g
^{237}Np	-	9600 g	10 g

Fabrication processes for ADS fuels



Reprocessing/Separation of TRU
TRU in diluted aqueous solution

Oxalic coprecipitation of TRU oxide $(Pu,Am,Cm)O_2$

Inert matrix supply i.e. MgO

~~Powder Mixing and blending~~

~~Powder granulation~~

~~Powder sieving~~

~~Powder pressing~~

~~Pellet sintering~~

Direct coprecipitation of $(Pu,Am,Cm)O_2$ ***and*** inert matrix

Powder granulation

Powder sieving

Powder pressing

Pellet sintering

Direct coprecipitation of $(Pu,Am,Cm)O_2$ and inert matrix ***in the form of granules***

Powder pressing

Pellet sintering

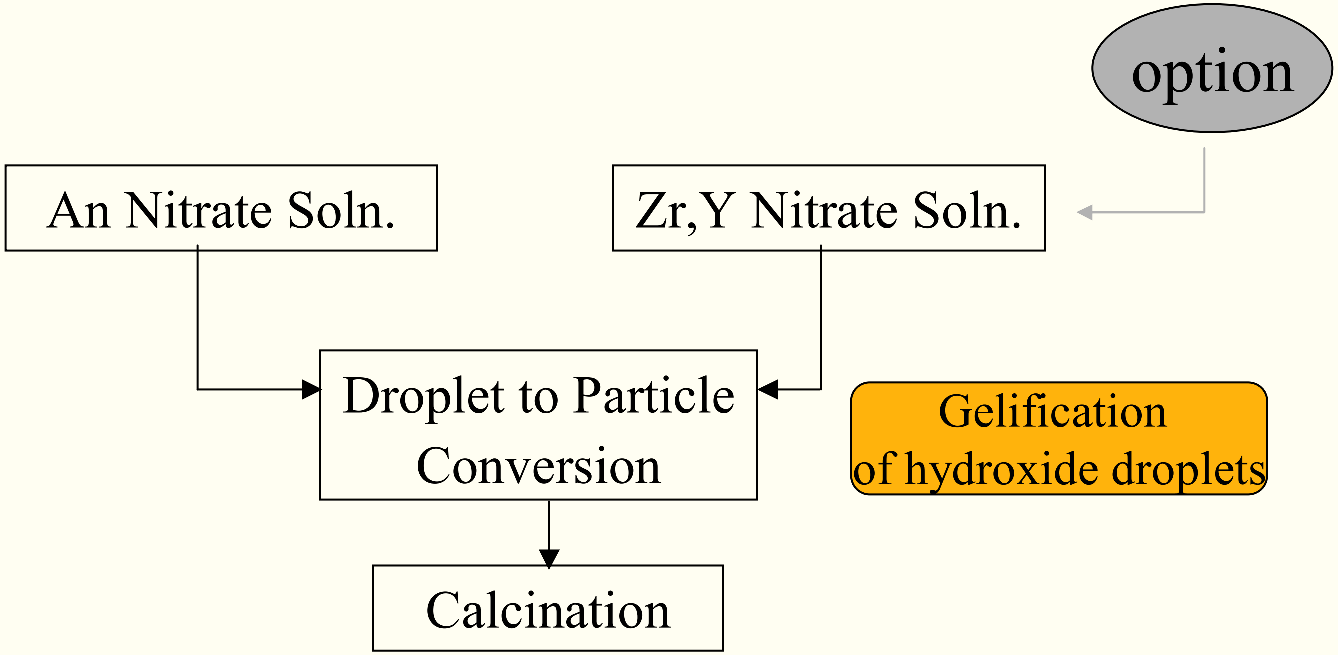
"Spherepac process"

Clad filling, pin bundle mounting and S/A fabrication

Fabrication processes for ADS fuels

- * Wet processes for [liquid ==> solid] conversion have great potential
  - – SOL-GEL (internal or external gelation process)
    - – direct synthesis of flowable, dust free MA particles (>100 µm)
      - – clean installation, easier maintenance after material removing (operator intervention)
      - – suppression of dirty operation of mechanical granulation
    - – but concentrated solution : 200 to 360 g/l
      - – continuous cooling throughout process (tanks, pipes, transfer tubing, vessels, etc...)
      - – Risk of hydrolysis, radiolysis of organic compounds used for the gelification

Fabrication processes for ADS fuels



option

An Nitrate Soln.

Zr,Y Nitrate Soln.

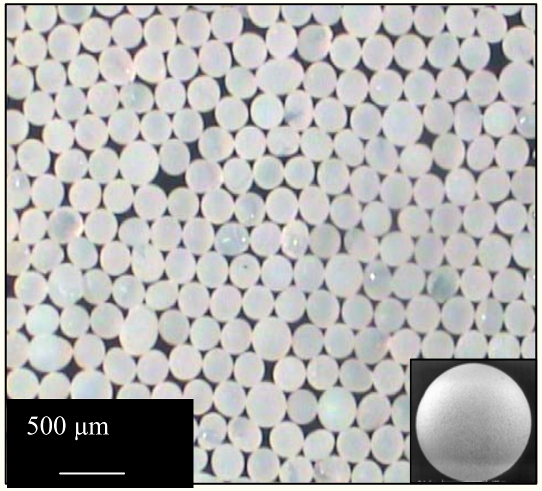
Droplet to Particle Conversion

Gelification of hydroxide droplets

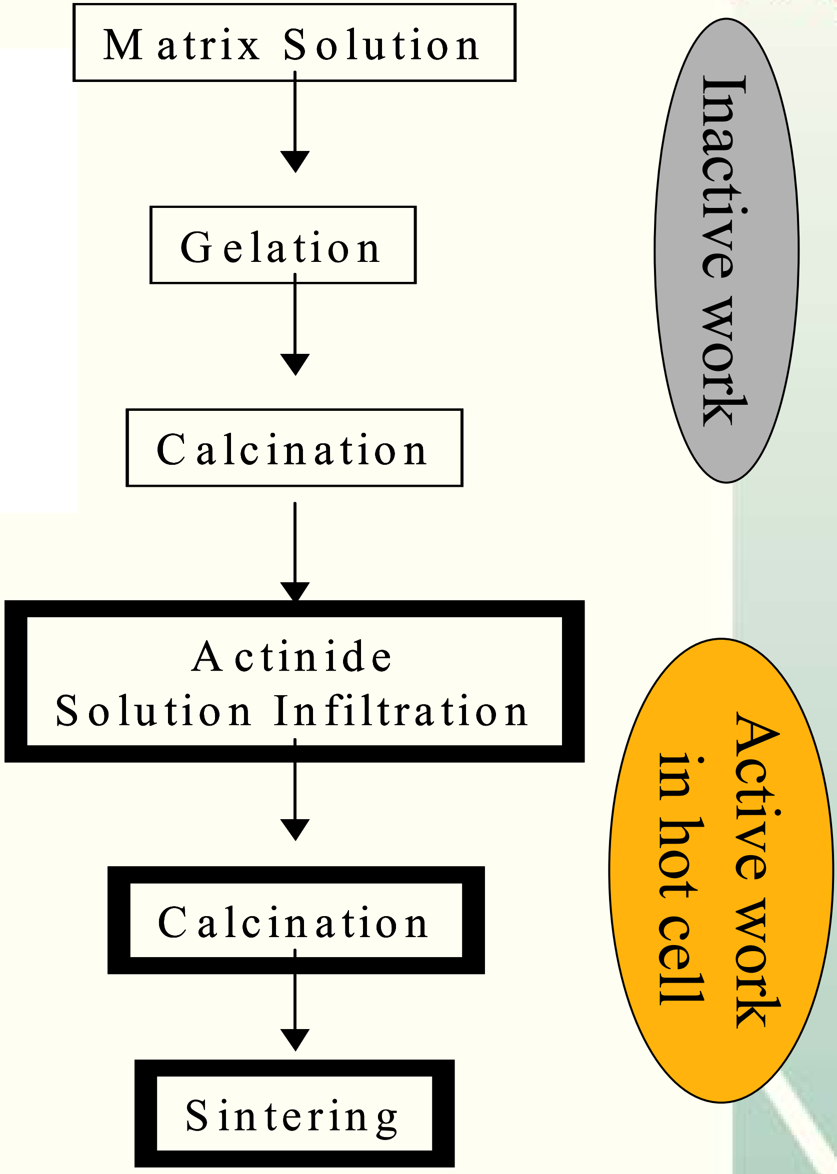
Calcination

Fabrication processes for ADS fuels

- * Other wet process
  - – Infiltration process
    - – High An concentration
    - – Limited An content
    - – No organic compounds



Porous beads of inert matrix before calcination



Matrix Solution

Gelation

Calcination

Actinide Solution Infiltration

Calcination

Sintering

Inactive work

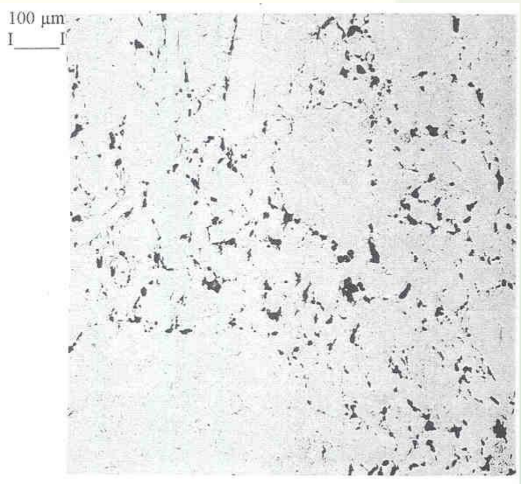
Active work in hot cell

Fabrication processes for ADS fuels

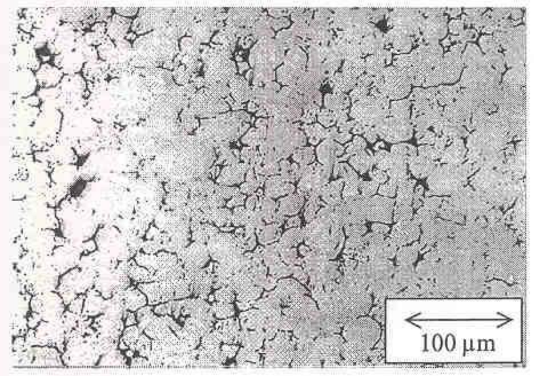
* Pressing & sintering
  - Use of organic lubricant is necessary for an industrial scale
  - Radiolysis and thermolysis issue
    - quick degradation of the lubricant property
    - further de-cohesion of the green pellets

Innovative development required (such as automatic lubrication of press dies)

Before aging



0,3 % StZn et 0,5 % AZB



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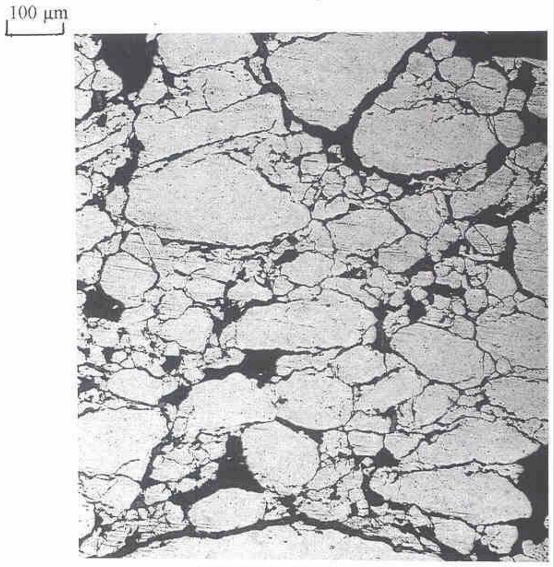

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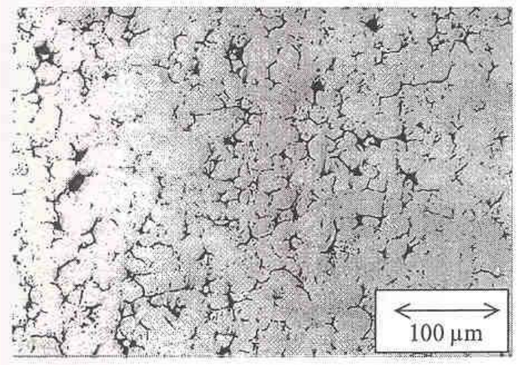

Innovative development required

(like automatic lubrication of press dies)

Assembly, mounting and storage

- * Neutron radiation ==> implementation necessarily in hot cell with remote handling
- * Heat control ==>continuous cooling (blowing engine)
  - very large cells to enable S/A handling (EFR S/A overall length : 4.8 m)
  - ancillary equipment (crane, etc...) in hot cells

S/A mounting, transport and storage in shielded cells

Simplified fabrication process: SpherePAC

* SPHEREPAC (called VIPAC when molten salts processes)

- Direct filling by vibropacking of fuel granules into the pin
- Avoid granulation, pressing and sintering steps
  - the simplest, the most robust and compact process
- Large experience existing in Russia
  - less intensive controls and only on welded pins (cleanness)
    - density and actinide distribution in pins
- Select inert matrices able to be co-precipitated with actinides :
  - MgO, ZrO_2

Issues for SpherePAC fuels

- * Optimisation of the filling to reach smeared density of ~ 85% th.d.
    - Selection of convenient granulometry classes
- * Segregation issue to be addressed
    - when mixing inert matrix and actinide compounds
- * Assess the impact of the fuel performance
    - Initial load of MA, fission rate, corrosion, …
    - Thermal conductivity

- * Irradiation tests required !!

Fabrication processes for nitride fuels

* Nitride fuel fabrication is a more complex process : additional step of oxide nitridation by carbothermic reduction

1. Fabrication of (Pu, Am, Zr)O_2 + *free C* as starting material
    - Powder metallurgy ➔ powder
    - Sol-gel process ➔ beads
    - Am is alternatively infiltrated into porous beads

2. Carbothermic reduction from oxides:

3. Shaping

Fabrication processes for nitride fuels

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1. Fabrication of (Pu, Am, Zr)O_2 + *free C* as starting material

2. Carbothermic reduction from oxides

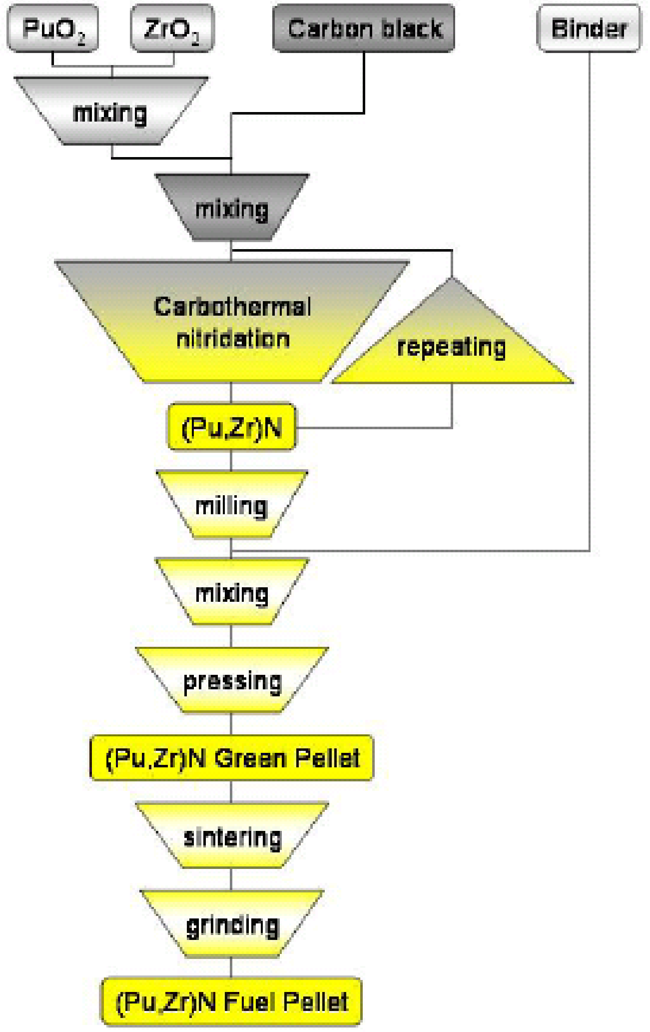
   (Pu, Am, Zr)O_{x+u} + xC +1/2N_2 +uH_2 ➔ (Pu, Am, Zr)N + uH_2O+ xCO

   - Intermediate products : carboxide, carbonitrides, oxide, carbide
   - Nitridation temperature limited by the AmN vaporisation at low temperature (~1500°C)
   - Product of good quality difficult to synthesize (incomplete reaction)

3. Shaping (grinding, sieving, pressing, sintering)

Fabrication processes for nitride fuels

- (Pu,Zr)N for CONFIRM fabricated by this fabrication route
- Density of $(Pu_{0.3},Zr_{0.7})N$ $\rho \sim 80\%$ T.D.



PuO_2

ZrO_2

Carbon black

Binder

mixing

mixing

Carbothermal nitridation

repeating

(Pu,Zr)N

milling

mixing

pressing

(Pu,Zr)N Green Pellet

sintering

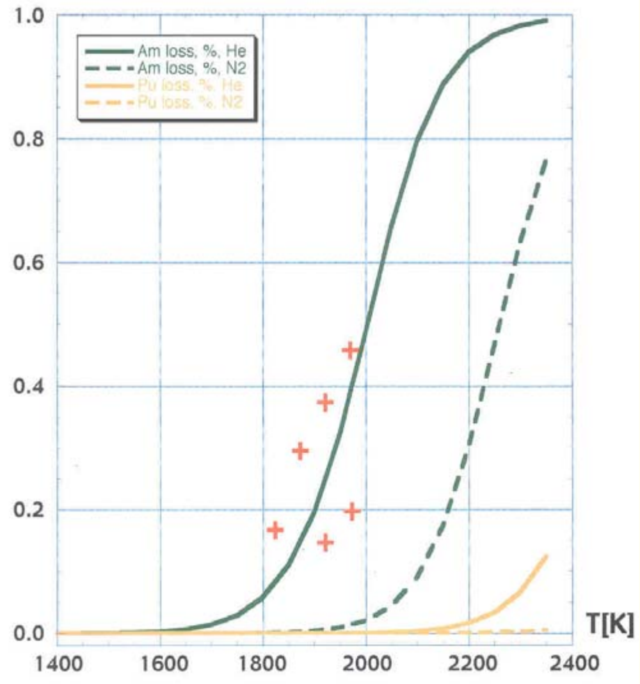
grinding

(Pu,Zr)N Fuel Pellet

Stability of AmN

- * AmN shows volatile behaviour at relative low temperatures of ~ 1800 K

- * This may severely limit the application of minor actinide nitride fuels



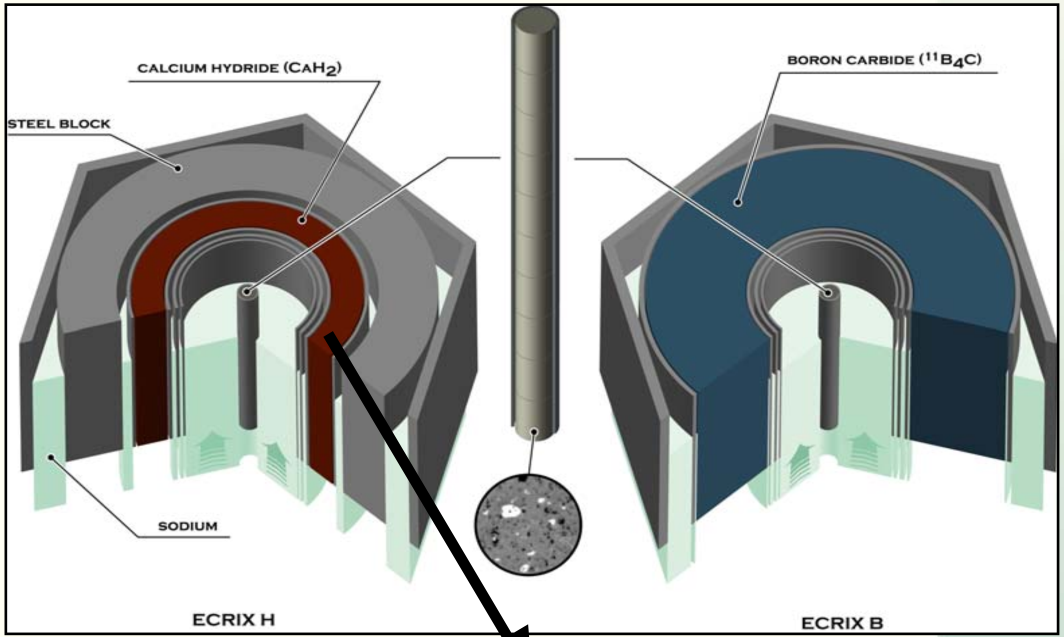
Conclusions and summary : fuel fabrication

- * Wet processes for shaping particles or granulates
  - – To avoid powder handling and dust dissemination (cleanness)
  - – To simplify the process
    - – VIPAC, SPHEREPAC process is an objective for the future
- * Impact of the fabrication process on the fuel performance process must still to be assessed
  - – A compromise between the simplification of the fabrication process and the increase of the fuel performance will have to be found

Experiments in Phénix: ECRIX

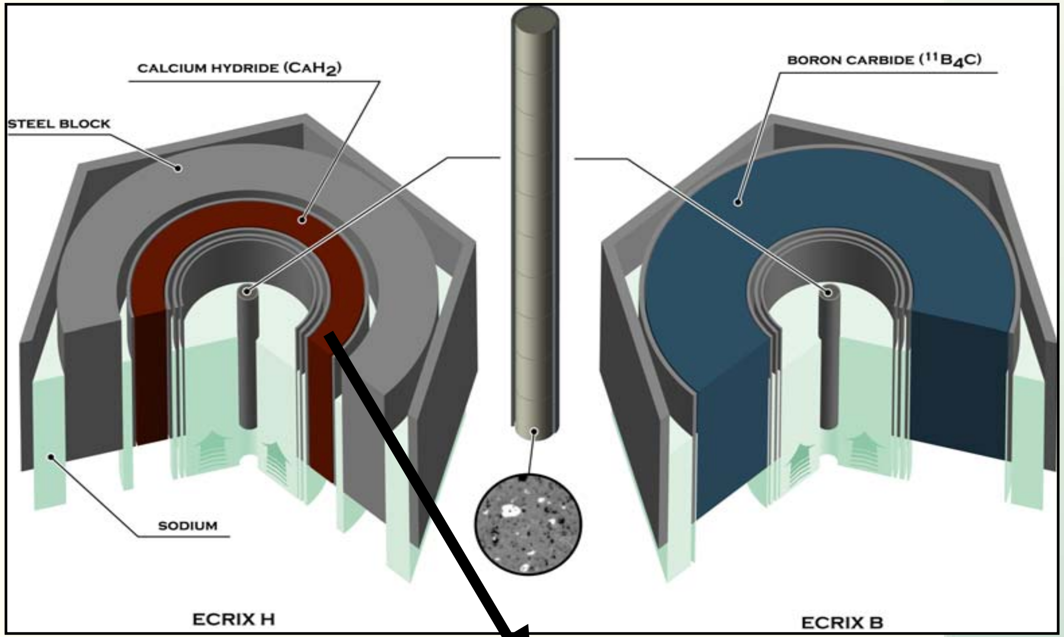
- ECRIX : $AmO_{1.6}$+ MgO
  - micro-dispersed
  - 0.7 g Am/cm^3
  - 200 mm (≈ 2,75 g of Am with 95 % of ^{241}Am)

- Objective
  - Test MgO
  - Fission rate = 30 at%
  - Role of the flux moderation



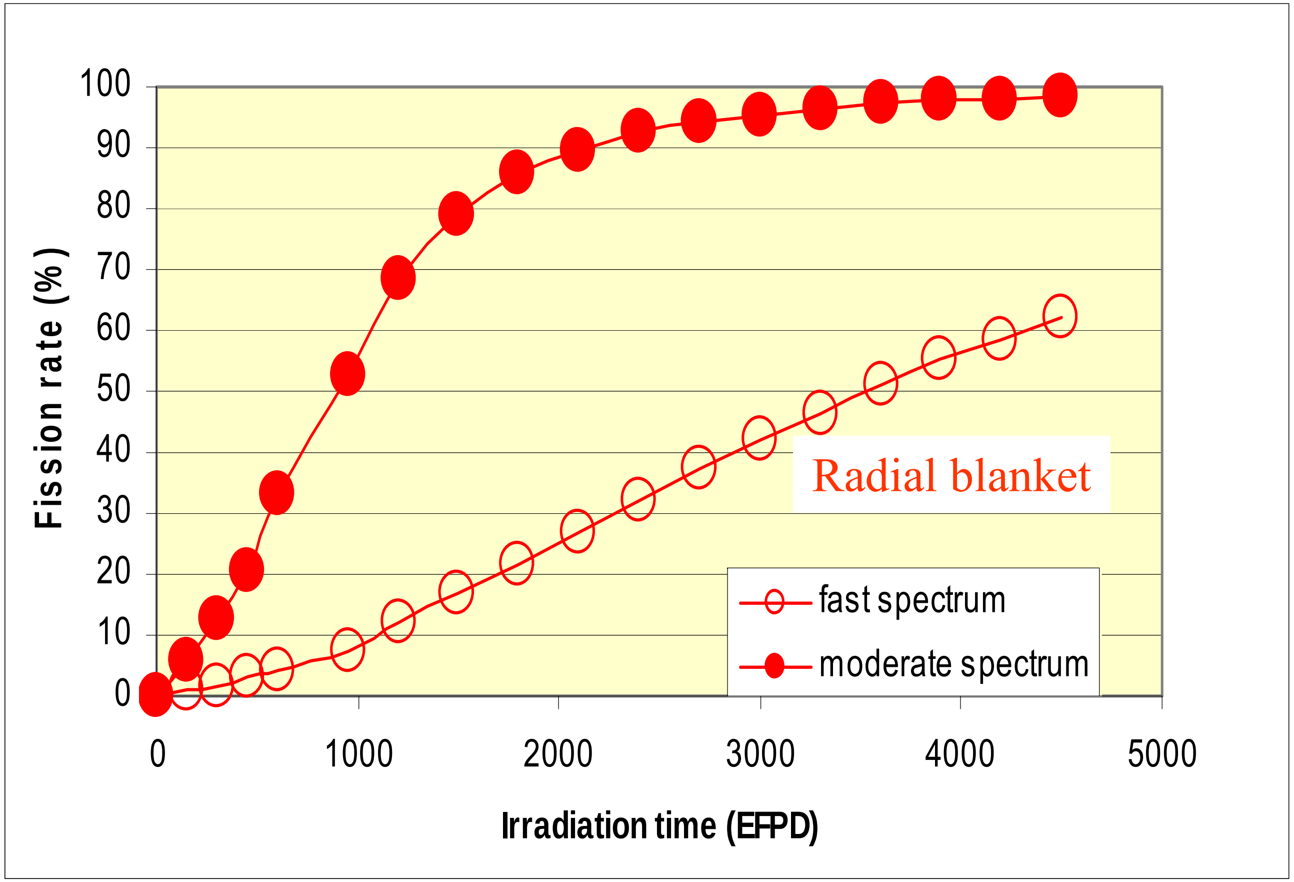
Experiments in Phénix: ECRIX

- * Role of the flux moderation
  - ECRIX B ($^{11}B_4C$)
    - Irradiation time = 670 EFPD
    - Transmutation rate = 80 %
    - End of irradiation : December 2007
  - ECRIX H (CaH_2)
    - Irradiation time = 340 EFPD
    - Transmutation rate = 90%
    - End of irradiation : October 2005



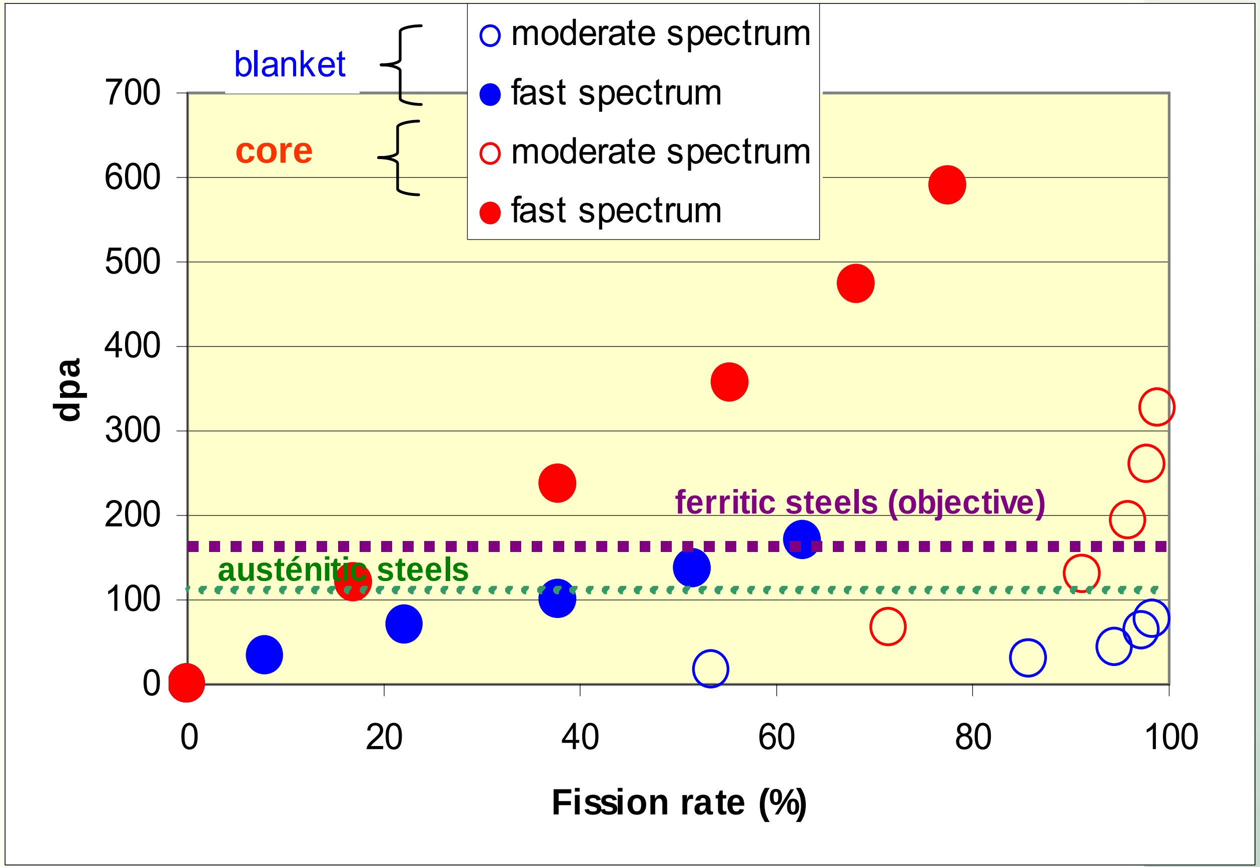
Experiments in Phénix: ECRIX

- * Role of the moderator
  - – Reach higher burnup by limiting the irradiation time



Experiments in Phénix: ECRIX

AmO_2 (20 wt%) + MgO
sub-assemblies
in FBR



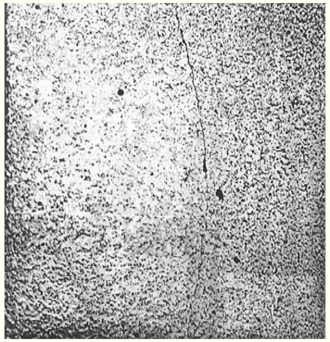
* Role of the moderator: to reach a higher burn-up by limiting the damage dose on the clad

Experiments in Phénix: CAMIX/COCHIX

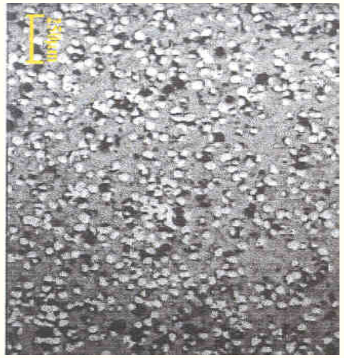
* Objective:

- Test new Am compounds, microstructures, and MgO
- Fission rate = 30 at% ⇨ irradiation time = 340 EFPD

⇨ Transmutation rate = 90 %

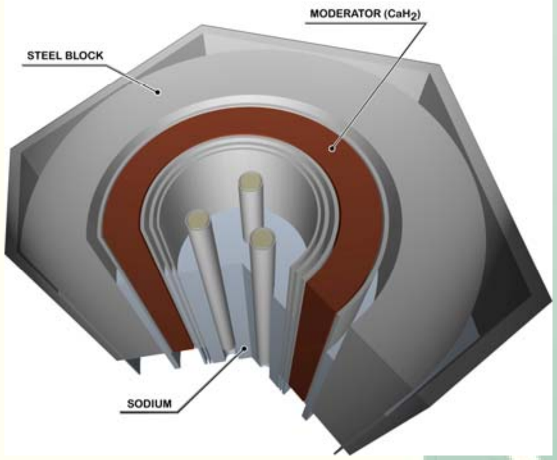


Sol. Solution



Micro-dispersed

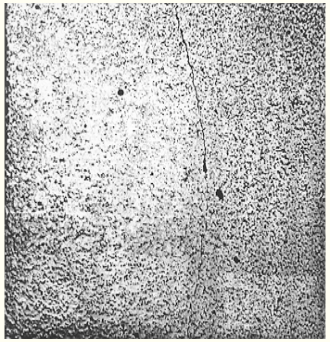
Macro-dispersed



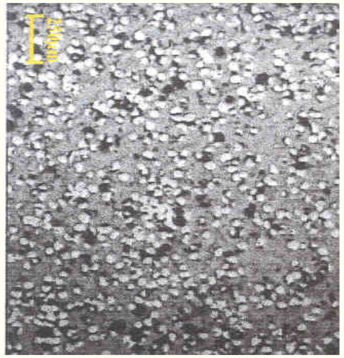
Experiments in Phénix: CAMIX/COCHIX

* CAMIX/COCHIX

- 0,7 g Am/cm^3, height of the column = 100 mm
- CAMIX 1 : Sol. Sol. $(Am_{0.06},Zr_{0.78},Y_{0.16})O_{2-x}$
- CAMIX 2 : CERCER micro. $(Am_{0.2},Zr_{0.66},Y_{0.14})O_{2-x}$ in MgO particule size $40 < \phi < 60$ μm
- COCHIX 3 : CERCER macro. $(Am_{0.2},Zr_{0.66},Y_{0.14})O_{2-x}$ in MgO particule size $100 < \phi < 125$ μm (30% vol of the matrix damaged)

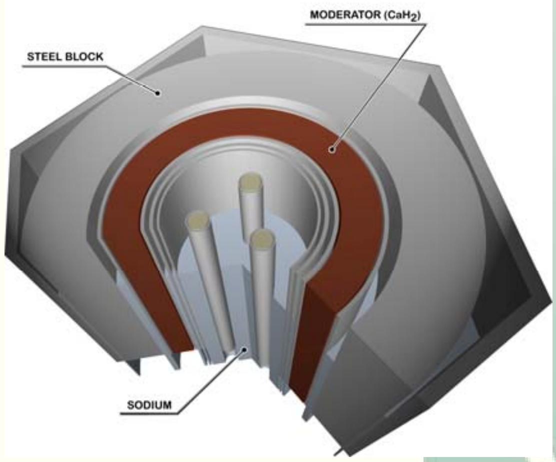


Sol. Solution

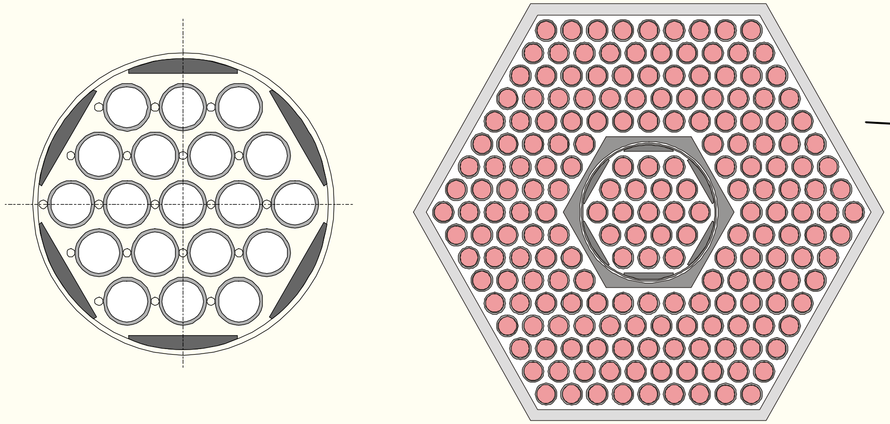


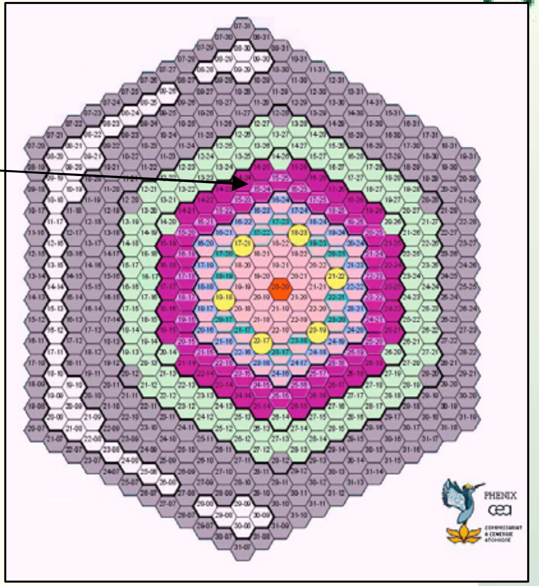
Micro-dispersed

Macro-dispersed



Experiments in Phénix: FUTURIX FTA





- * Standard Phénix capsule
  - 19 pins (length : 1.80 m, ϕ : 5.65-6.55 mm)
    - 8 experimental transmutation fuel pins
    - Fuel column : 10 cm
- * 2 irradiation cycles
  - half 2007 ➔ beginning 2009
  - 240 EFPD, 4th ring in the core, flux 4.10^{15} $n.cm^2.s^{-1}$

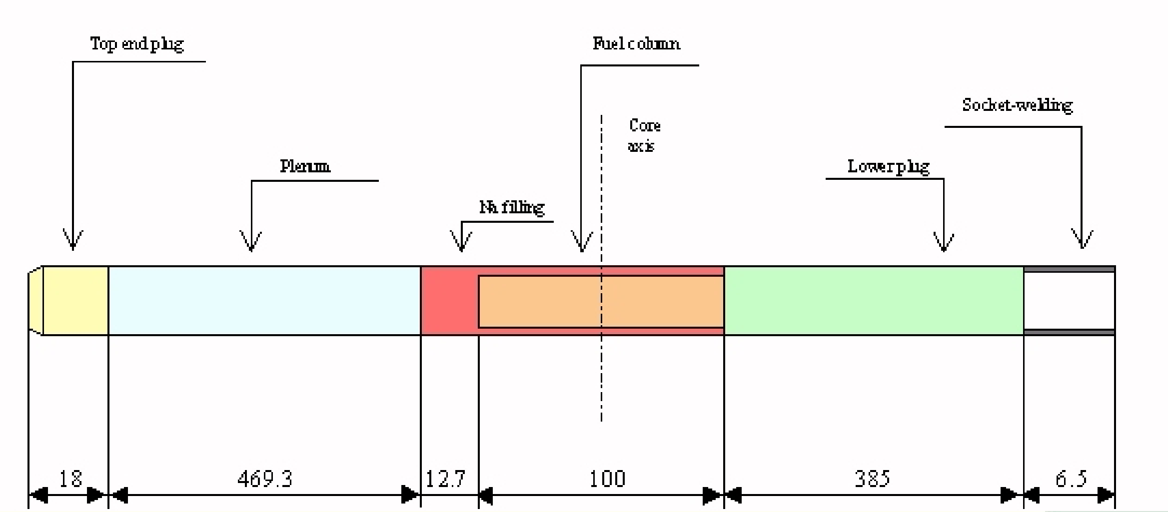
Experiments in Phénix: FUTURIX FTA

fuels	Composition	Am content (g/cm^3)	TRU content (g/cm^3)
U-free	48Pu-12Am-40Zr (in wt%)	1.2	5.8
	$(Pu_{0.50},Am_{0.50})N$ + 36wt% ZrN	2.7	5.3
	$(Pu_{0.2},Am_{0.8})O_2$ + 65 vol% MgO	2.7	3.4
	$(Pu_{0.5},Am_{0.5})O_2$ + 70 vol% MgO	1.45	2.9
	$(Pu_{0.23},Am_{0.25},Zr_{0.52})O_2$ + 50 vol% Mo^{92}	1.1	2.1
	$(Pu_{0.8}, Am_{0.2})O_2$ + 50 vol% Mo^{92}	0.9	4.6
Low fertile	35U-29Pu-4Am-2Np-30Zr (in wt%)	0.5	4.0
	$(U_{0.50},Pu_{0.25},Am_{0.15},Np_{0.10})N$	1.7	5.7

- Composition of the FUTURIX-FTA device
- Nitride pins to be irradiated later

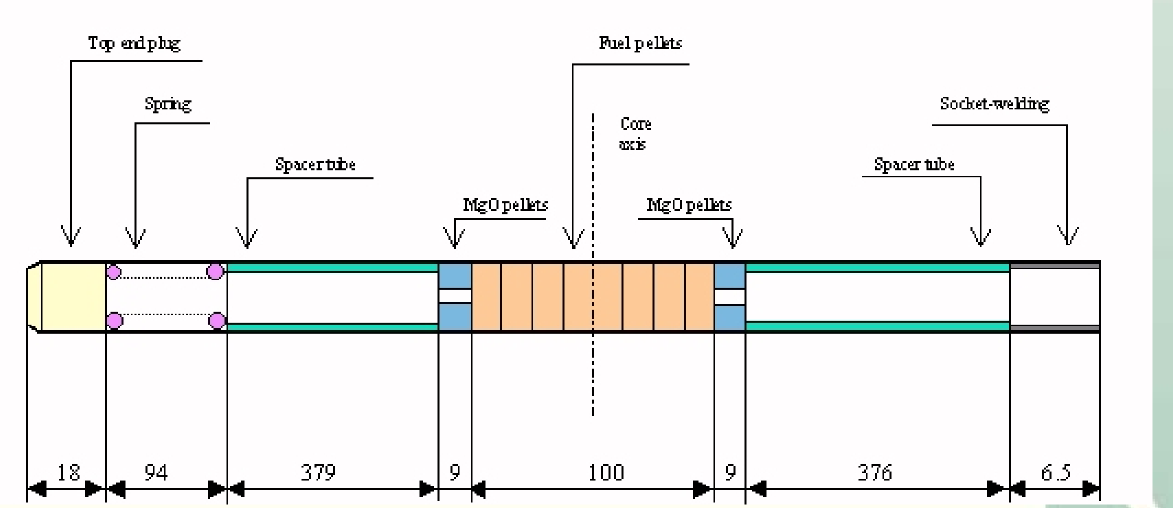
Futurix FTA in Phénix

Na-bonded pin



FUTURIX-FTA
pin
design

He-bonded pin

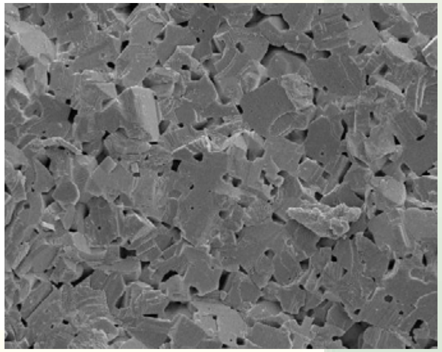


HELIOS planned in HFR

* Five pins containing 0.7 grams ^{241}Am /cm^3

* Objectives

- Test pyrochlore compound of Am
- Test Mo matrix
- Confirm the effect of temperature on the He release
- Test porous inert matrix to favour the He release



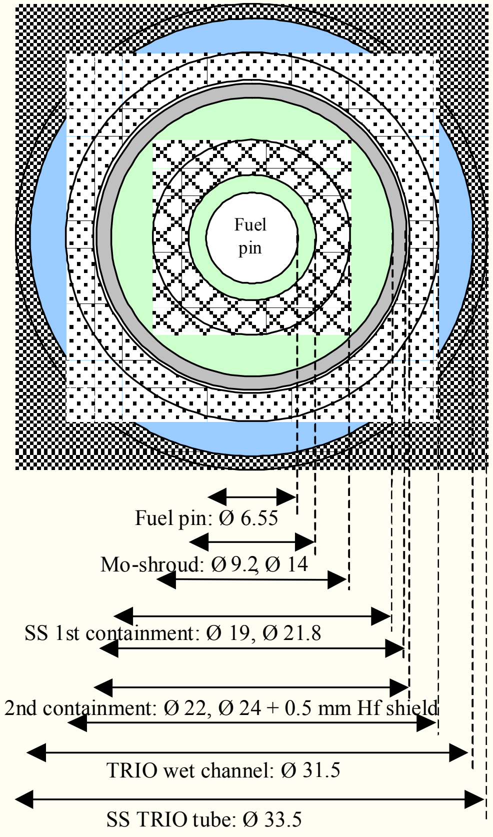
Helios	Composition	Fuel length (mm)	Fuel diameter (mm)	Volume (cm3)	Actual Density (g.cm-3)	Stack weight (g)	%Pu	%Am
1	Am2Zr2O7-MgO	62.0	5.458	1.451	3.96	5.748	0.00%	17.84%
2	ZrYAmO2	60.7	5.43	1.406	5.92	7.497	0.00%	12.17%
3	ZrYPuAmO2	60.5	5.43	1.401	5.85	7.555	7.85%	12.12%
4	ZrYAmO2/Mo	59.5	5.44	1.383	8.85	12.110	0.00%	7.88%
5	PuAmO2/Mo	63.1	5.41	1.450	10.00	14.246	12.50%	2.97%

Nitride experiment in HFR

- CONFIRM
  - Irradiation of 2 nitride pins (Pu,Zr)N as precursor of ADS nitride fuels
    - $(Pu_{0.3},Zr_{0.7})N$ irradiated
    - $(Pu_{0.2},Zr_{0.8})N$ also fabricated
  - Started on 25 November 2007
    - Linear heat rate = 450 W/cm
    - 6 cycles (150 EFPD)

CONFIRM

- Twee pins with (Pu,Zr)N fuel
- Linear heat rate ~ 450 W/cm
- Hafnium shield with a thickness of van 0.5 mm



▩ **Stainless steel TRIO tube**

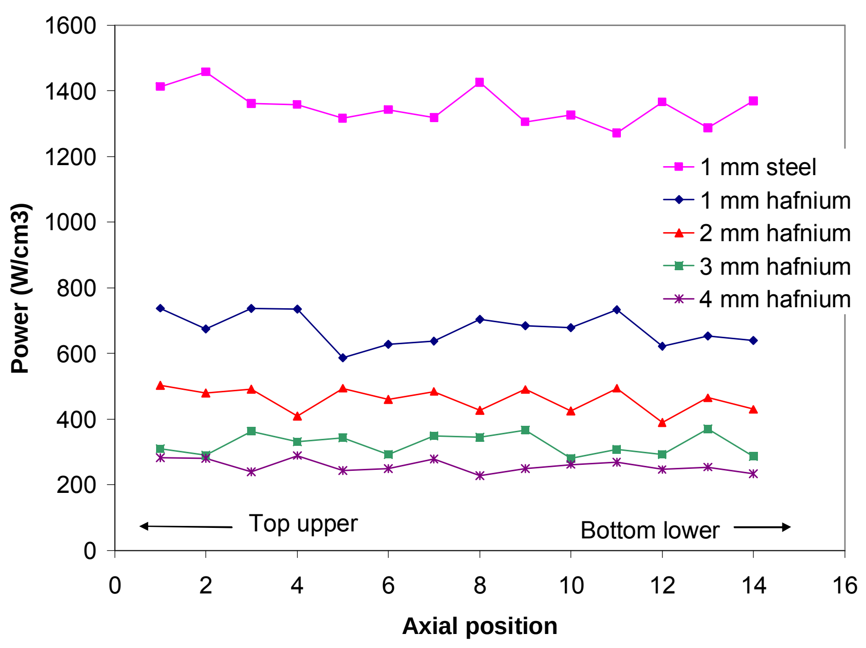
🟦 **Water in wet TRIO channel**

⬚ **Stainless Steel 2nd containment with 0.5 mm Hf shield**

▇ **Stainless Steel 1st containment**

🟩 **Na**

⊠ **Mo**



EUROTRANS

- Eurotrans is a 6th European Framework Programme
  - From 2005 – 2009, budget ~ 50 M€
  - Dedicated to the design & performance of an ADS demonstrator

  - Various core designs are studied, and two inert matrix fuel types are considered for MA burning

– MgO CerCer	Slightly better transmutation rate
– Mo CerMet	Better safety characteristics

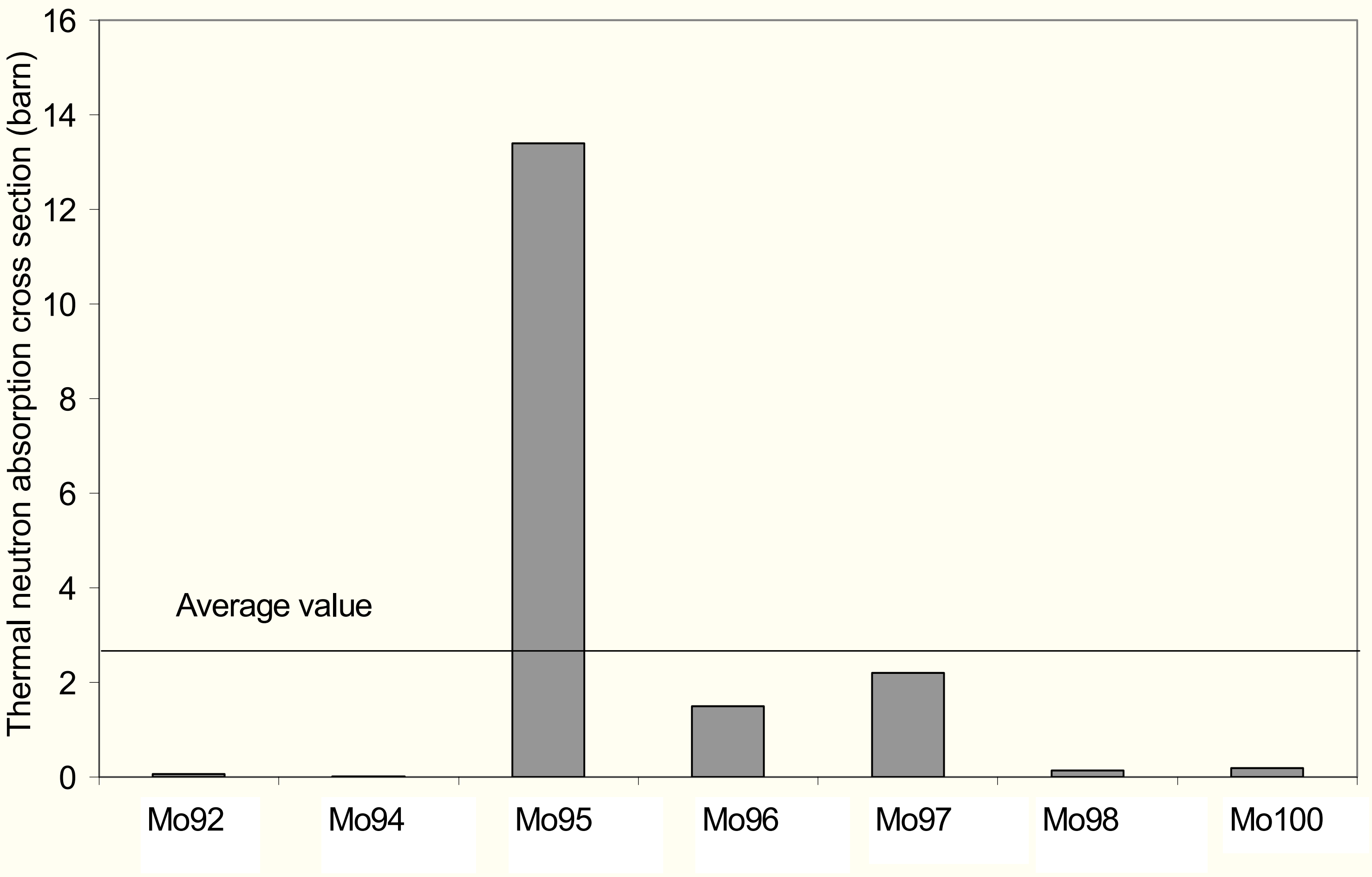
Why Molybdenum metal ?

- * Molybdenum has excellent prospects
  - – TANOX irradiation (40 wt% UO_2 in Mo) has shown good irradiation behaviour
    P. Dehaudt et al., *IAEA Technical Comittee Meeting on Research of Fuel aimed at Low Fission Gas Release*. Oct. 1-4, 1996. Moscow, Russia, IAEA TECDOC 970 (1997)
  - – High melting point (~ 3000 K),
  - – High thermal conductivity (~ 100 W/mK at 1500 K)
  - – Reprocessible
  - – Easy fabrication compared to ceramic IMF

- * But molybdenum has adverse neutronics properties
  - – Relatively high absorption cross section
  - – Neutron capture in Mo-98 leads to the production of long-lived Tc-99

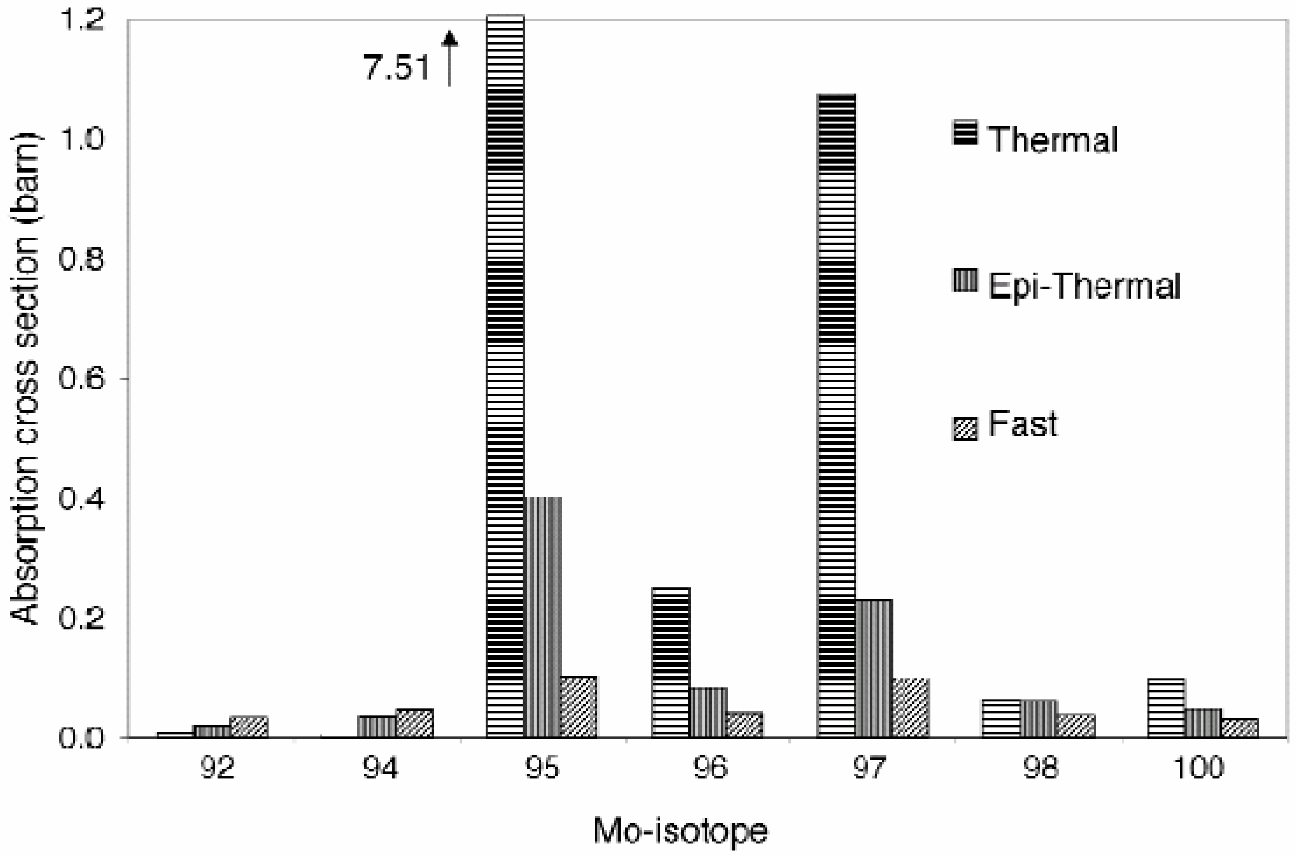
Molybdenum has seven stable isotopes

* Only Mo-95 has a high cross section for thermal neutrons



Molybdenum has seven stable isotopes

- For fast neutrons, the absorption cross sections are quite similar for all isotopes
- The production of Tc-99 becomes a more important issue



Molybdenum depletion in ^{95}Mo

* Neutronic properties can be improved by depletion in Mo-95 and / or in Mo-98

* Urenco & NRG studied depletion of Mo in ^{95}Mo *
- Mo depletion (<1% ^{95}Mo) can be achieved
- Costs of depletion are relatively low
- Thermal cross section of Mo can be reduced significantly

* In a similar way, depletion in Mo-98 can be achieved

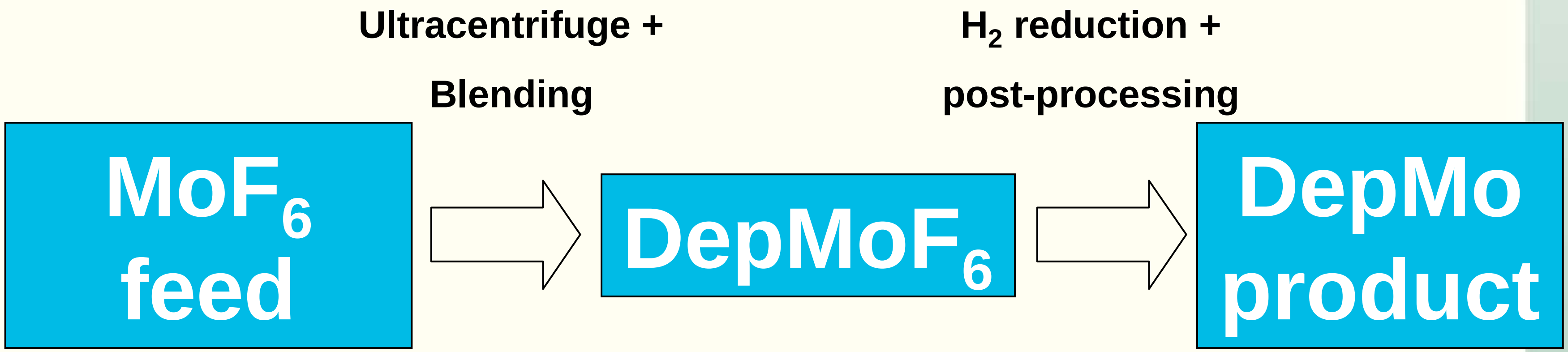
* K. Bakker *et al.*, Nucl. Technology **146** (2004), 325

Molybdenum depletion in ^{95}Mo

- * For the following procedure was designed for the production of Depleted Mo (DepMo)
- * Separation into three fractions
  - – Light fraction (enriched in Mo-92, Mo-94)
  - – Heavy fraction (enriched in Mo-98, Mo-100)
  - – Middle fraction(containing the Mo-95) ➔ Tails
- * Light and heavy fractions are mixed to achieve the lowest Mo-95 content
- * Urenco Stable Isotopes has optimised for this scenario

URENCO process for depleting Mo

- The enrichment (depletion) process for Mo is similar to the enrichment of uranium
- Conversion is achieved by reduction to Mo-metal



Ultracentrifuge + Blending

H_2 reduction + post-processing

MoF_6 feed

$DepMoF_6$

DepMo product

Final isotopic composition

- After depletion and conversion to metal, a typical blend was made with Mo-95 < 1%
- But in principle, any isotopic composition can be achieved

	Natural	Light	Heavy	Typical blend
Isotope	at%	at%	at%	at%
92	14.84	~ 90 %		14.0
94	9.25			1.4
95	15.92			0.9
96	16.68			1.9
97	9.55			5.3
98	24.13		} ~ 90 %	42.5
100	9.63			34.0

Molybdenum issues

* Now we have produced DepMo and established a successful fabrication route for DepMo-based CerMets

* Question:

What is the optimum isotopic composition of DepMo?

* Three criteria have to be balanced:
  - Parasitic absorption of neutrons by DepMo
  - Limit Tc-99 production
  - Minimise costs of depletion of Mo

Three criteria

- Assessment of:
  - Parasitic absorption — σ_{abs}
  - Tc-99 production — ^{99}Tc (capture) / ^{99}Tc (fission)
  - Costs of depletion — SWU needed / material loss

- For different reactor types:
  - Light Water Reactor
  - Gas-Cooled Fast Reactor
  - Accelerator Driven Systems

- For various DepMo mixtures
  - Light fraction only (~ 90% Mo-92)
  - Produced DepMo with Mo-95 ~ 1%
  - DepMo with Mo-98 ~ 2 % (low Tc-99 production)

Results: σ_{abs} and Tc-99 production

Energy averaged absorption cross section σ_{abs} (barn)

	Natural Mo	Light fraction ~ 90% Mo-92	LWR DepMo ~ 1% Mo-95	FR DepMo ~ 2% Mo-98
LWR	0.983	0.217	0.252	0.870
GCFR	0.203	0.087	0.151	0.152
ADS	0.129	0.066	0.091	0.106

Ratio Tc-99 production: ^{99}Tc (capture in Mo) / ^{99}Tc (fission An)

	Natural Mo	Light fraction ~ 90% Mo-92	LWR DepMo ~ 1% Mo-95	FR DepMo ~ 2% Mo-98
LWR	0.148	0.001	0.200	0.012
GCFR	2.385	0.008	3.226	0.196
ADS	1.680	0.006	2.273	0.138

For LWR, Tc-99 production increases by ~ 20 %
For fast reactor systems, depletion in Mo-98 is necessary

Conclusions for Molybdenum

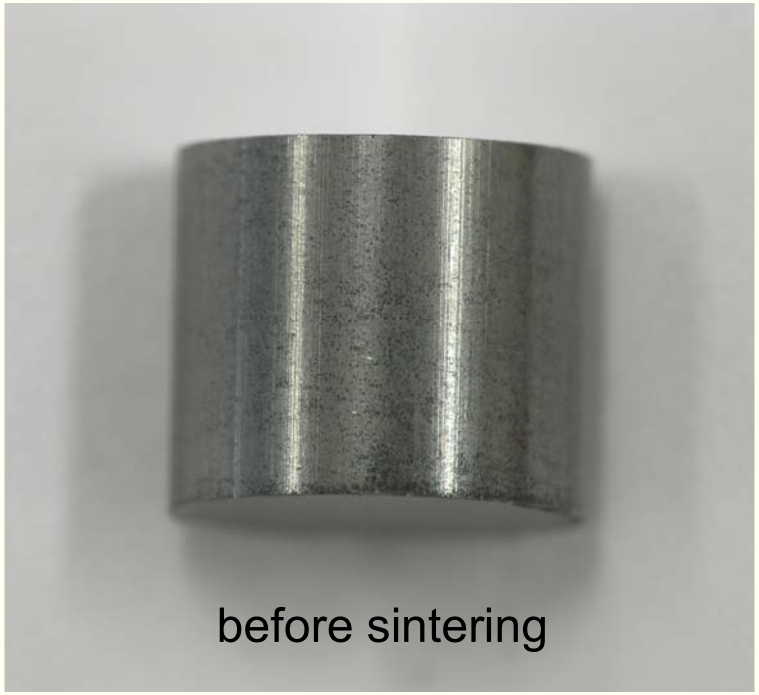
- Molybdenum is a very promising inert matrix material

- Adverse neutronic properties can resolved by depletion in Mo-95 / Mo-98
  - Process developed by URENCO
  - Costs are reasonable in relation to potential benefits
  - DepMo can be used also for U-Mo MTR fuel

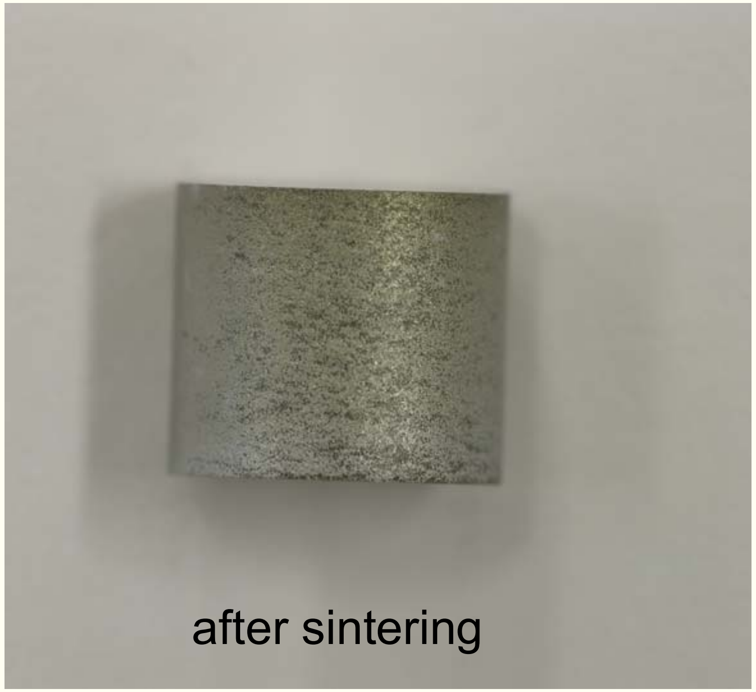
- Fabrication of CERMETs has been sucessfully achieved

- We are eagerly waiting for the test irradiations
  - FUTURIX FTA in Phenix
  - HELIOS in High Flux Reactor (HFR)
  - DepMo / PuO_2 irradiation in BR-2 (LWR-Deputy programme)

Last, but not least: Fabrication is easy!



before sintering



after sintering

Composites, 30 vol.% microspheres

Leading ideas for MA-IMF design

From transmutation experiments (such as EFTTRA T4bis, we have optained a good insight in the most important isssues and how to influence them

Try to limit volume swelling by:

- Promoting Helium release:
  - Providing open porosity
  - By increasing the temperature
- Alternatively other inert matrices with better retention properties are also considered, such as metals

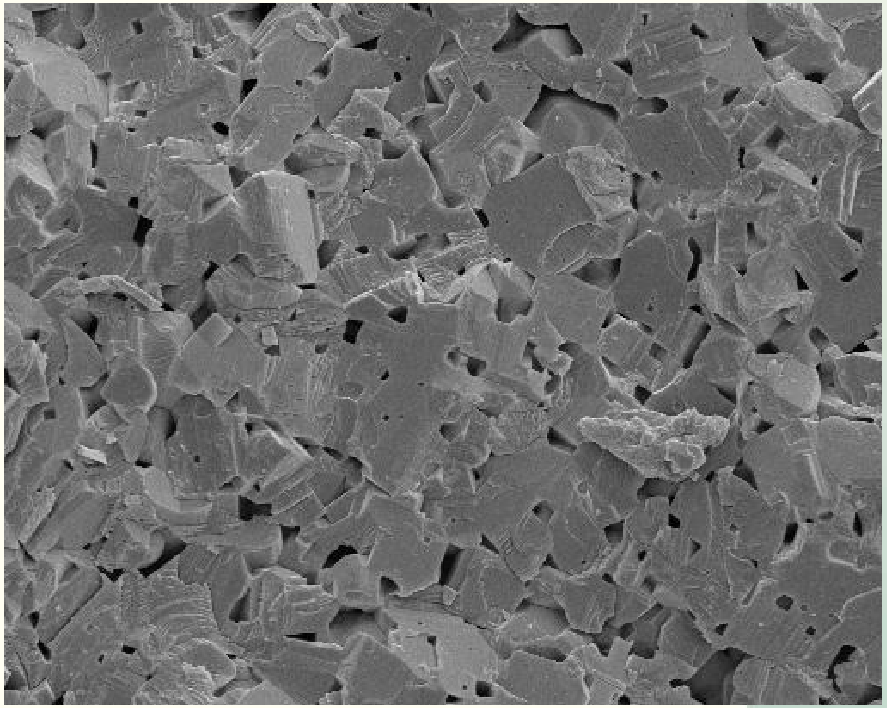
Providing Open porosity

Open porosity, up

to 50% of total porosity,

is obtained in MgO matrix:

- sintering under nitrogen pressure
- double pressing before sintering



Experiments performed by CEA

Density is 85-90%T.D. while open porosity remains high

Increase of temperature

- Implantation experiments of Helium in spinel and zirconia indicate that:
  - onset of Helium release: about 800 Kelvin
  - completion of release : about 1600 Kelvin

Fuel temperatures must be higher than about 1400 C

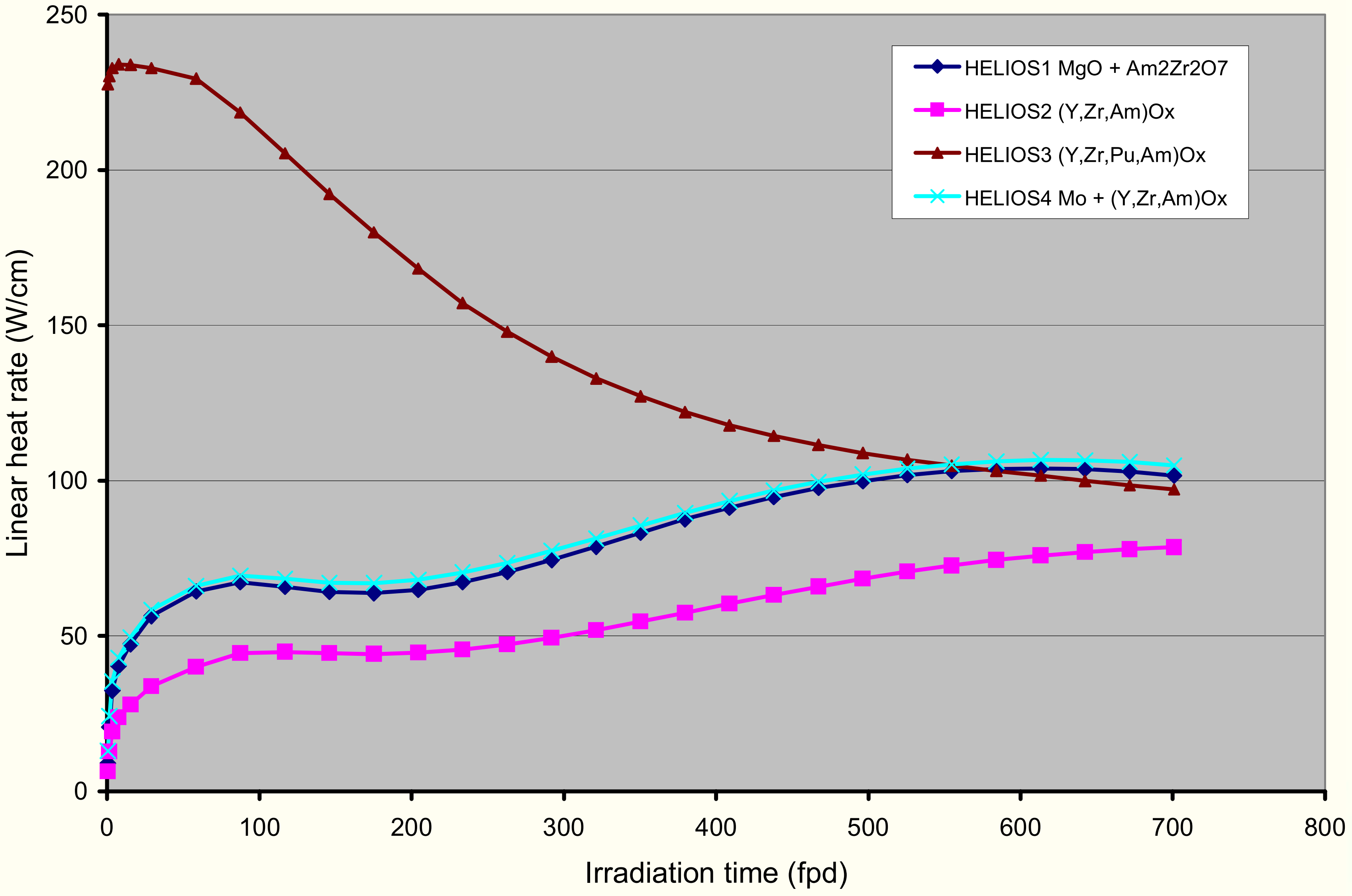
- Fuel temperature increase:
  - Addition of Plutonium in order to study the effect of temperature on release and swelling
  - Adjustment of thermal design (increase of initial gap)

Next experiment: Helios

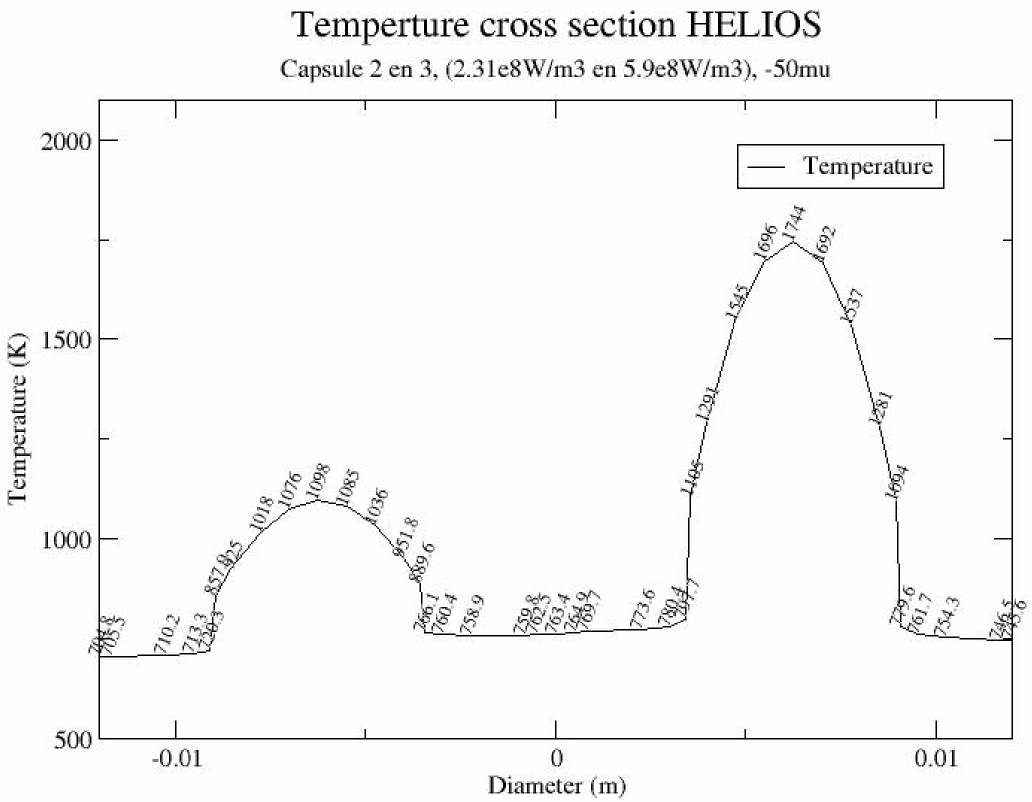
- In HELIOS, many of the lessons learned will be used
  - Irradiation in HFR, ~ 250 – 300 full power days
  - MgO with open porosity
  - Zirconia with / without Pu to address helium release mechanism
  - Testing of Mo as metal matrix

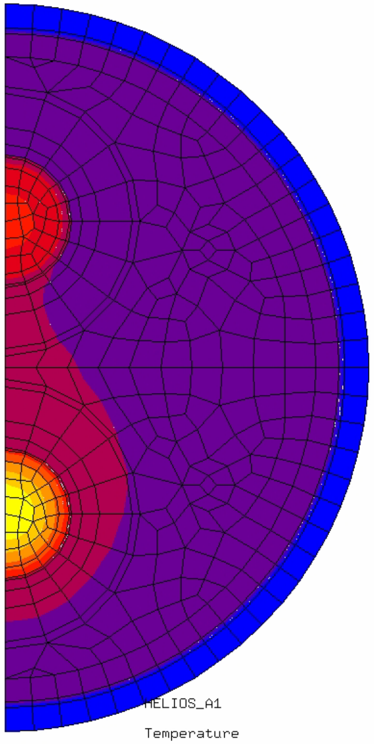
Pin nr.	Composition	Target Am density	Target Pu density	Expected T.D.	Specified act. dens. (based on 100% T.D.)
		g/cm^3	g/cm^3	%	g/cm^3
1	$Am_2Zr_2O_7$+MgO	0.7		90%	0.78
2	$(Am,Zr,Y)O_2$	0.7		90%	0.78
3	$(Am,Pu,Zr,Y)O_2$	0.7	0.25	90%	0.78 / 0.28
4	$(Zr,Am,Y)O_2$+Mo	0.7		95%	0.74
5	$(Pu,Am)O_2$+Mo	0.3	1.2	95%	0.74 / 1.26

Power densities in HELIOS



Addition of plutonium: increase of temperature to promote helium release





Summary HELIOS

- HELIOS provides an irradiation test to study the release options for helium
  - To optimise the transmutation of Am

- Study of CerCer and CerMet matrices

- Two methods are being tested:
  - Providing open porosity (in MgO)
  - Fuel temperature increase by addition of Plutonium

- Five Am-containing inert targets will be irradiated, starting from 2008 in HFR

Conclusions and summary : fuel composition

* ADS fuels are composed of:

- High content of TRU
  - Degraded thermal properties
  - Degraded fuel performance in pile compared to U+Pu-based fuels
- 55-60 vol% of inert matrix
  - To improve the thermal properties of the fuel
    - Melting point, thermal conductivity, stability at high temperature,…
  - To improve the fuel quality

Conclusions and summary : fuel composition

- * ADS fuel can be:
  - – An oxide composite :
    - – Reference: (Pu,-Np-,Am,Cm)O_2+MgO, Mo and ZrO_2 (He bond or Pb bond)
  - – A nitride solid solution
    - – Reference: (Pu,-Np-,Am,Cm)N + ZrN (He, Pb or Na bond)
  - – A metal alloy
    - – Reference: Pu-Np-Am-Cm-40Zr (Na bond)
  - – Knowledge on oxide fuels is the largest in Europe
    - – Several Irradiation tests now completed,
    - – Several irradiation tests to be completed
    - – Others in preparation and started in 2007
  - – New PIE results on nitride should be available soon

Conclusions and summary: fuel performance

- Inert matrices have been selected for
  - Their good thermal properties (higher than that of UO_2)
  - Their compatibility with the TRU compound, the coolant, the clad
  - Their resistance to neutron, FP recoil damage
    - MgO, ZrO_2 for TRU oxide
    - ZrN for TRU nitride
- ADS fuels must be designed to accommodate
  - The swelling due to the inert matrix damage (FP recoil)
    - Depends on the inert matrix (sensitivity to amorphisation)
    - Beneficial effect of the irradiation temperature (>1100°C) and macrostructure
  - The swelling due to the He production
    - Beneficial effect of the irradiation temperature and microstructure (open porosity)

Conclusions and summary: fuel performance

- ADS fuel performance (Linear power & BU) is limited
  - By the melting of (Pu,-Np-Am,Cm)O_2 or the eutectic reaction between (Pu,-Np-Am,Cm)O_2 and the inert matrix
  - By the vaporisation of (Pu,-Np-,Am,Cm, Zr)N
  - The cladding damage and clad corrosion

- Not so many hot labs in Europe to handle TRUs
  - CEA (ATALANTE), ITU (MA-lab), *NRG (Act. Lab)*
  - Only small quantities of Am and Cm
- Simplification, automation, compaction, robustness
  - heavy shielding, remote handling & cooling throughout process
    - from conversion to S/A storage and transport
  - limited fabrication possibilities

Conclusions and summary : fuel fabrication

- * Wet processes for shaping particles or granulates
  - – To avoid powder handling and dust dissemination (cleanness)
  - – To simplify the process
    - – VIPAC, SPHEREPAC process is an objective for the future
- * Impact of the fabrication process on the fuel performance process must still to be assessed
  - – A compromise between the simplification of the fabrication process and the increase of the fuel performance will have to be found