GEN IV International Forum

MOX FUEL FOR ADVANCED REACTORS

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Meet the Presenter

GEN IV International Forum℠

Nathalie Chauvin is working at CEA Cadarache IRESNE in the fuel Studies Department International Expert on fuels for fast reactors. She worked for a long time on the minor Actinides transmutation program, participating to the optimization of the fuel design, the irradiation experiments and the synthesis reports. Then she was project manager for the development of very innovative fuels for the Gas cooled Fast Reactor with oxide/carbide fuels, refractory cladding including ceramic composites one for pin or plate type fuel element. She is now in charge of international cooperations devoted to fast reactor fuels development as 1) Chair of the Working Party on the Fuel Cycle at OECD/Nuclear Science Committee; 2) Chair of the Expert Group on Innovative Fuel at OECD/NSC/WPFC; 3) GIF French representative in the GFR system – Fuels & material; 4) Project manager of PUMMA (Plutonium Management for More Agility at EURATOM); 5) Leader of fuel properties workpackage in the project ESFR-SMART; 6) French representative in the CRP on Fuels and Materials for Fast Reactors at the IAEA. She is also participating in several activities in different scientific committees of international conferences (IEMPT, FR GLOBAL), and she is the CEA counterpart in several bilateral collaborations with other international scientific organizations devoted to MOX fuel.

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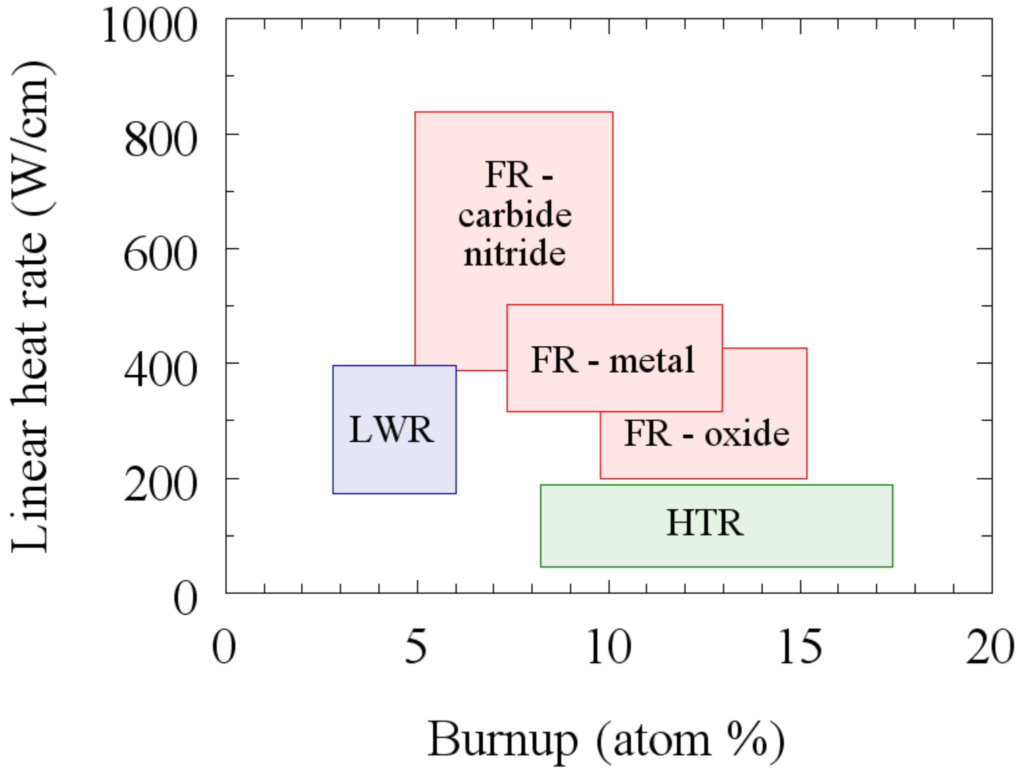
Nuclear Materials for Gen IV Reactors

	Gen II/III LWR	*SCWR*	**SFR**	**LFR**	**ADS**	**GFR**	**VHTR**	**MSR**
Fuel +MA	*UO_2, MOX Th-MOX*	*UO_2, MOX Th-MOX*	**$UPuO_2$ UPuZr UPuN UPuC**	**$UPuO_2$ UPuN**	U free fuel, Inert Matrix Fuel	**$UPuO_2$ UPuC**	UO_2,UCO PuO2 $(Zr,Y,Pu)O_2$	$LiF-ThF_4-UF_4$
Cladding	*Zr alloy*	*F/M steel*	**15/15Ticw T91 ODS**	**T91**	T91	**SiC-SiCf**	iPyC/SiC/oPyC	
Liner	-	-	**-**	**-**	-	**W W/Re**	Buf Carbon	Structures
Fuel form	*Pellet*	*Pellet*	**Pellet (Sphere Pac)**	**Pellet (Sphere Pac)**	Pellet (Sphere Pac)	**Plate Pin**	Coated Particle	Fluid
Coolant	*Water*	*Water*	**Na**	**Pb**	Pb or Pb/Bi	**He**	He	$NaF-NaBF_4$

Operating Conditions in Fast Reactors

- High linear heat rate: 400 to 500 W/cm max
- High fuel temperature 600 to 2400°C for $(U,Pu)O_2$
  ~1000°C for (U,Pu)Zr
- High burnup 130 GWd/t or 15 at %
- Residence time >800 days or >130 dpa



GIF (2019), GEN-IV International Forum, « GIF R&D Outlook for Generation IV Nuclear Energy Systems : 2018 Update, » Version 1.1, June 2019.

Criteria for Choice of Fuel Materials

- Material properties
  - High density of fissile atoms
  - High thermal conductivity and high melting point + high thermal stability
    → High margin to melt
    → No phase transition, no dissociation,
  - High mechanical stability
    - Isotropic expansion, radiation resistant
  - Acceptable chemical compatibility with cladding and coolant: no strong reaction

- Performances for evaluation
  - High burn-up and flexibility towards operation conditions
  - Behaviour during transients & accidents
  - Fuel Cycle :
    - Flexibility towards fuel cycle options (Pu and Minor Actinides management)
    - Cost of fabrication and reprocessing

CONTENT

- **Main features of mixed oxide fuel for advanced reactors**
  - Characteristics of the material
  - Fuel properties
  - Comparison of $(U,Pu)O_2$ properties under irradiation with the others fuels
  - Fuel element design with $(U,Pu)O_2$
- **Fuel behaviour under irradiation**
  - Main features of fuel behaviour
  - Evolution of fuel microstructure and composition
  - Thermo-mechanical behaviour
  - Behaviour during accident
- **Fuel element performances, design and qualification**
  - Fuel element performances
  - Improvement in the design and qualification of MOX pins
  - Qualification of fuel performance codes
- **Synthesis & Conclusion**

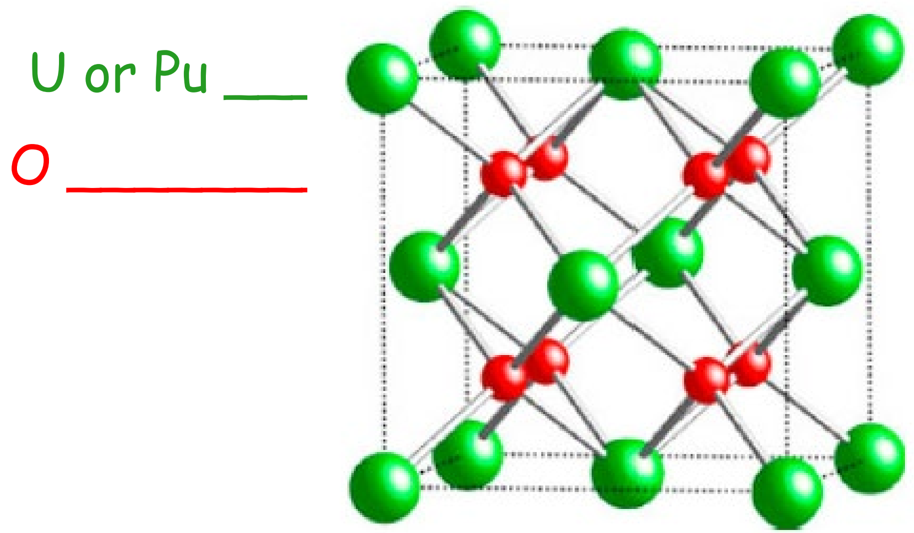
PART 1 : Main Features of Mixed Oxide Fuel for Advanced Reactors

Cadarache facilities:

LEFCA

LECA

Structure of mixed oxides $(U_{1-y}Pu_y)O_{2\pm x}$ Face Centred Cubic (fcc) : fluorite type



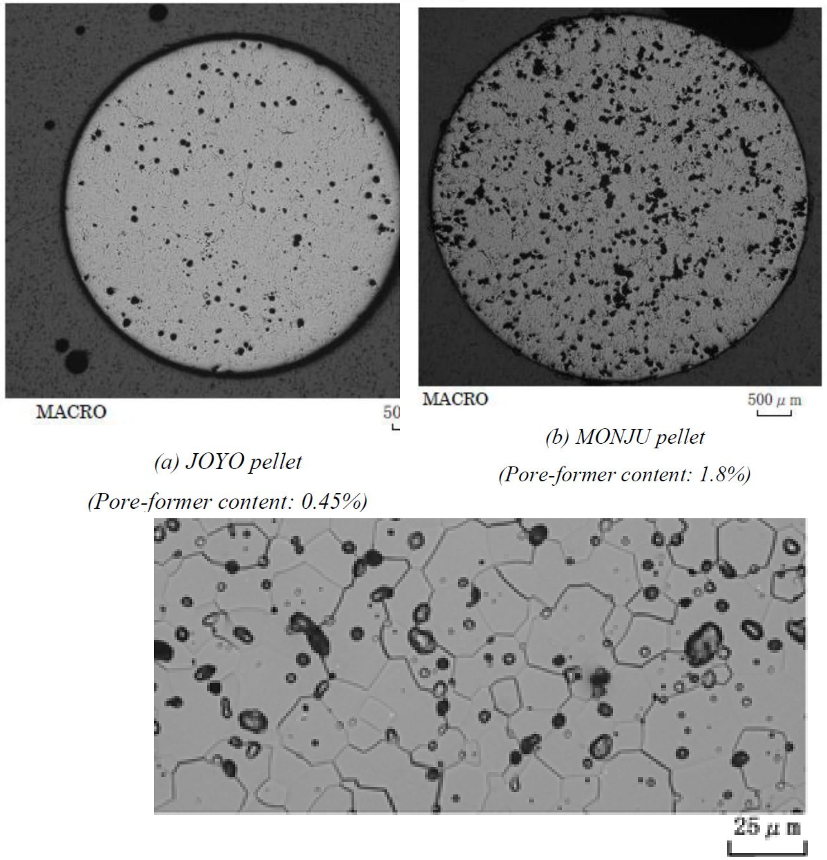
Lattice parameter depends on x and y

- U - Pu substitutions : from 0% to 100% (theoretical)

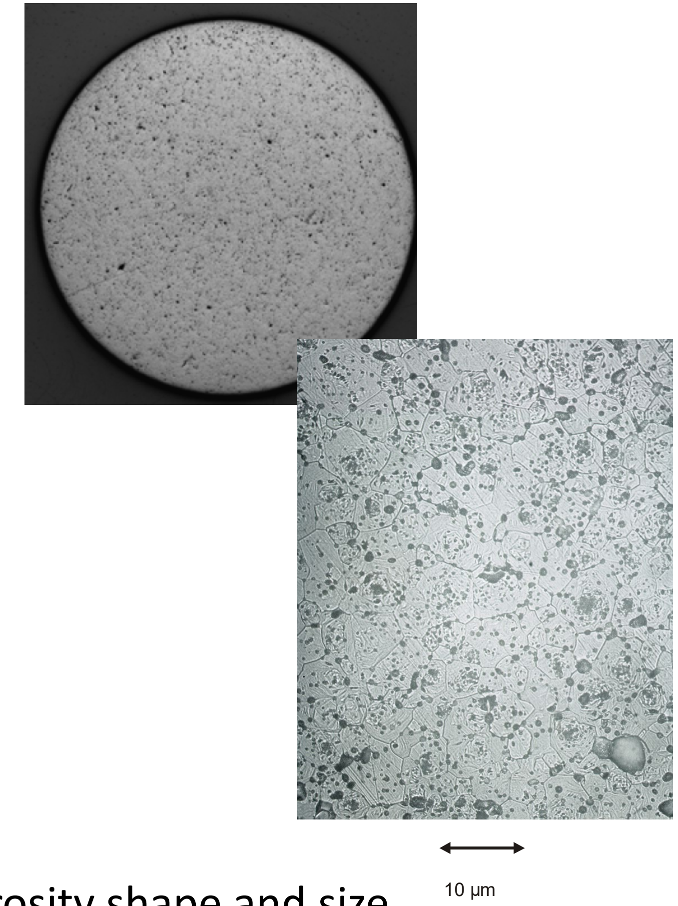
- Non stoichiometry in actinide oxides
  - $x < 0$: O vacancies or An interstitials (or mixture)
  - $x > 0$: O interstitials or An vacancies (or mixture)

MOX fuel : microstructure & fabrication

Powder metallurgy
Pore former process
JAEA

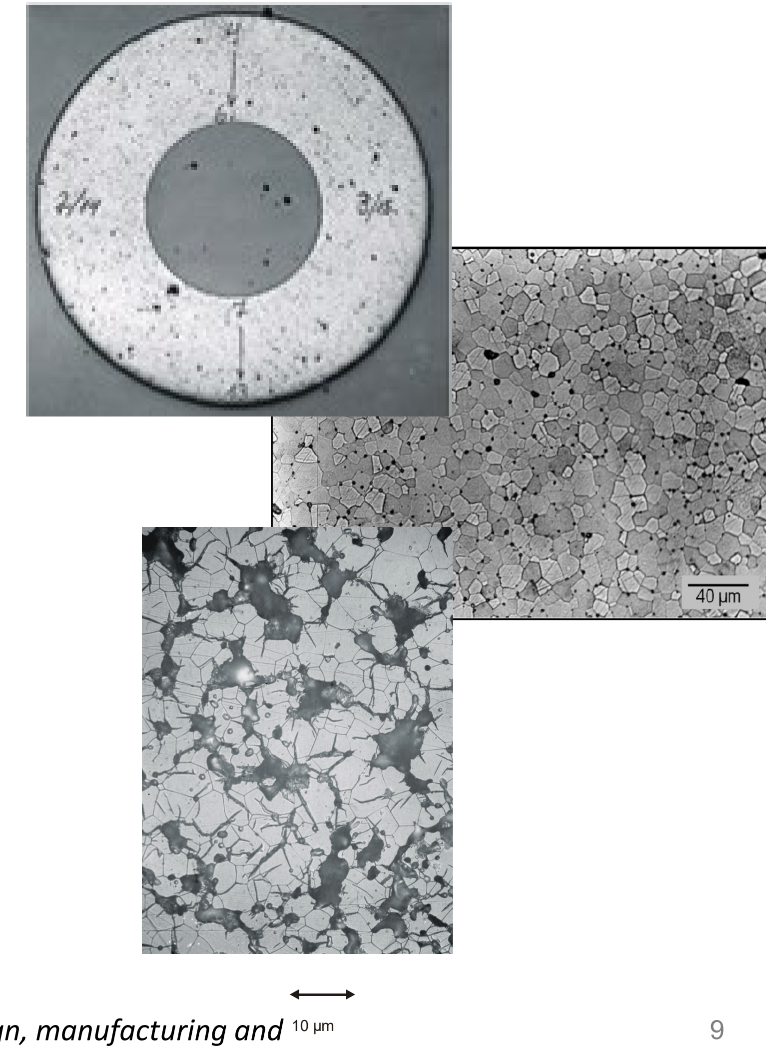


Powder metallurgy
COCA process
CEA



SOLGEL process

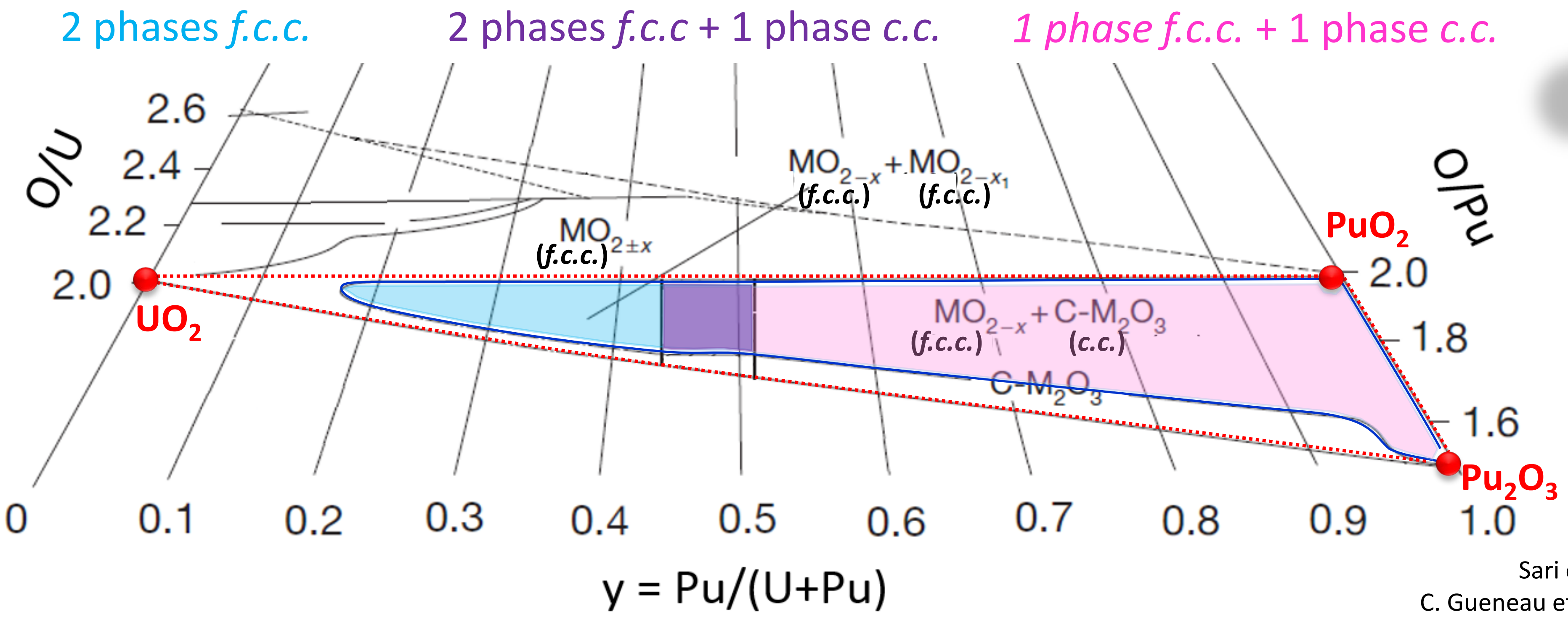
JRC - Karlsruhe

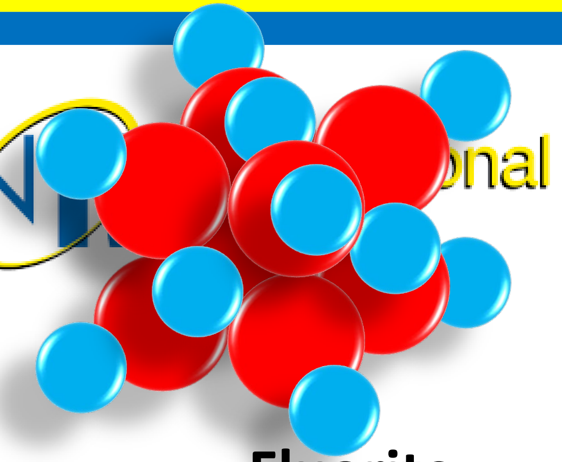


- Microstructure : grain size, density, porosity shape and size
- Microstructure depends on fabrication process

AIEA –TECDOC n°1689 "Design, manufacturing and irradiation behavior of fast reactor fuel".

Mixed oxides $(U_{1-y}Pu_y)O_{2\pm x}$





Fluorite (*f.c.c.*)

298 K

Sari et al., J. Nucl. Mater.35, 267-277, 1970
C. Gueneau et al. J. Nucl. Mater. 419, 145-167, 2011

- O/M < 1,98 and T<1100K and Pu>18% with possible phases : $(U,Pu)O_2$, $(U,Pu)O_{2\pm x}$, $(U,Pu)_2O_3$,
- 1,98 < O/M < 2,0 or T>1100K : fcc solid solution

MOX properties : NEEDS FOR FUEL PERFORMANCE CODES

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$(U\text{-}Pu)O_2$ properties / models of interest	Parameters of influence / (Range of interest)						
	Temperature (293 – boiling)	Pu/M ratio (15 – 35%)	O/M ratio (1.94 – 2.00)	Fract. porosity (0 – 40%)	Grain size (4 – 30 µm)	Stress (1 – 100 MPa)	Burn up (0-125 GWd/t)
Lattice parameter	X	X	X				X
Thermal conductivity	X	X	X	X			X
Melting point		X	X				X
Specific heat capacity	X	X	X				X
Enthalpy of fusion		X	X				X
Emissivity	X	X	X				X
Theoretical density	X	X	X				
Thermal expansion	X	X	X				X
Elastic constants	X	X	X	X			
Brittle-to-ductile transition temperature		X	X	X			
Yield stress, ultimate stress	X	X	X	X			
Thermal creep	X	X	X	X	X	X	X
Diffusion / migration of pores, of fission gas, of oxygen, of U, of Pu	X	X	X				
Oxygen potential	X	X	X				X
Grain growth	X			X	X		

New measurements expected

Comparison of fuel properties during irradiation

Properties	$(U_{0.8}Pu_{0.2})O_2$	$(U_{0.8}Pu_{0.2})C$	$(U_{0.8}Pu_{0.2})N$	U-19Pu-10Zr
Theoretical density, g·cc	11.04	13.58	14.32	15.73
Melting point, K	3083	2750	3070	1400
Thermal conductivity, $(W \cdot m^{-1} \cdot K^{-1})$ at 1000–2000 K	2.6–2.4	18.8–21.2	15.8–20.1	40–40
Crystal structure	Fluoride	Nacl	Nacl	Alfa
Breeding ratio	1.1–1.15	1.2–1.25	1.2–1.25	1.35–1.4
Swelling	Moderate	High	Moderate	High
Handling	Easy	Pyrophoric	Inert	Inert
Compatibility: clad	Average	Carburisation	Good	Eutectics
Compatibility: coolant	Average	Good	Good	Good
Dissolution and reprocessing	Good	Demonstrated	Risk of C14	Amenable for pyro reprocessing
Fabrication/irradiation experience	Large and good	Limited	Very little	Limited

GIF – "Advanced Sodium Fast Reactor (SFR) Fuel Comparison », March 2009.

Metal fuel

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Properties	(U0.8Pu0.2)O2	(U0.8 Pu0.2)C	(U0.8Pu0.2)N	U-19Pu-10Zr
Theoretical density, g·cc	11.04	13.58	14.32	15.73
Melting point, K	3083	2750	3070	1400
Thermal conductivity, $(W \cdot m^{-1} \cdot K^{-1})$ at 1000–2000 K	[illegible]	[illegible]	8–20.1	40–40
Crystal structure	[illegible]	[illegible]	Nacl	Alfa
Breeding ratio	[illegible]	[illegible]	2–1.25	1.35–1.4
Swelling	Moderate	High	Moderate	High
Handling	Easy	Pyrophoric	Inert	Inert
Compatibility: clad	Average	Carburisation	Good	Eutectics
Compatibility: coolant	Average	Good	Good	Good
Dissolution and reprocessing	Good	Demonstrated	Risk of C14	Amenable for pyro reprocessing
Fabrication/irradiation experience	Large and good	Limited	Very little	Limited

Low melting temperature
High thermal conductivity
High swelling : large gap + metal bond
Eutectic with clad

Carbide fuel

Properties	(U0.8Pu0.2)O2	(U0.8 Pu0.2)C	(U0.8Pu0.2)N	U-19Pu-10Zr
Theoretical density, g·cc	11.04	13.58		
Melting point, K	3083	2750		
Thermal conductivity, (W·m^{-1}·K^{-1}) at 1000–2000 K	2.6–2.4	18.8–21.2		
Crystal structure	Fluoride	Nacl		
Breeding ratio	1.1–1.15	1.2–1.25		
Swelling	Moderate	High		
Handling	Easy	Pyrophoric		
Compatibility: clad	Average	Carburisation	Good	Eutectics
Compatibility: coolant	Average	Good	Good	Good
Dissolution and reprocessing	Good	Demonstrated	Risk of C14	Amenable for pyro reprocessing
Fabrication/irradiation experience	Large and good	Limited	Very little	Limited

High melting temperature + high thermal conductivity:
High margin to melt
Moderate thermal creep
High swelling (to be managed with Na bond or low thermal level or reduced Burn Up)
Fabrication complex, costly

Nitride fuel

Properties	(U0.8Pu0.2)O2	(U0.8 Pu0.2)C	(U0.8Pu0.2)N	U-19Pu-10Zr
Theoretical density, g·cc	11.04	13.58	14.32	15.73
Melting point, K	[illegible]	[illegible]	3070	1400
Thermal conduct (W·m^{-1}·K^{-1}) at 1000–2000 K	[illegible]	[illegible]	15.8–20.1	40–40
Crystal structure	[illegible]	[illegible]	Nacl	Alfa
Breeding ratio	[illegible]	[illegible]	1.2–1.25	1.35–1.4
Swelling	[illegible]	[illegible]	Moderate	High
Handling	Easy	Pyrophoric	Inert	Inert
Compatibility: clad	Average	Carburisation	Good	Eutectics
Compatibility: coolant	Average	Good	Good	Good
Dissolution and reprocessing	Good	Demonstrated	Risk of C14	Amenable for pyro reprocessing
Fabrication/irradiation experience	Large and good	Limited	Very little	Limited

High melting temperature
High margin to melt but possible dissociation at 1800K
High thermal creep (low mechanical interaction with clad)
Moderate swelling

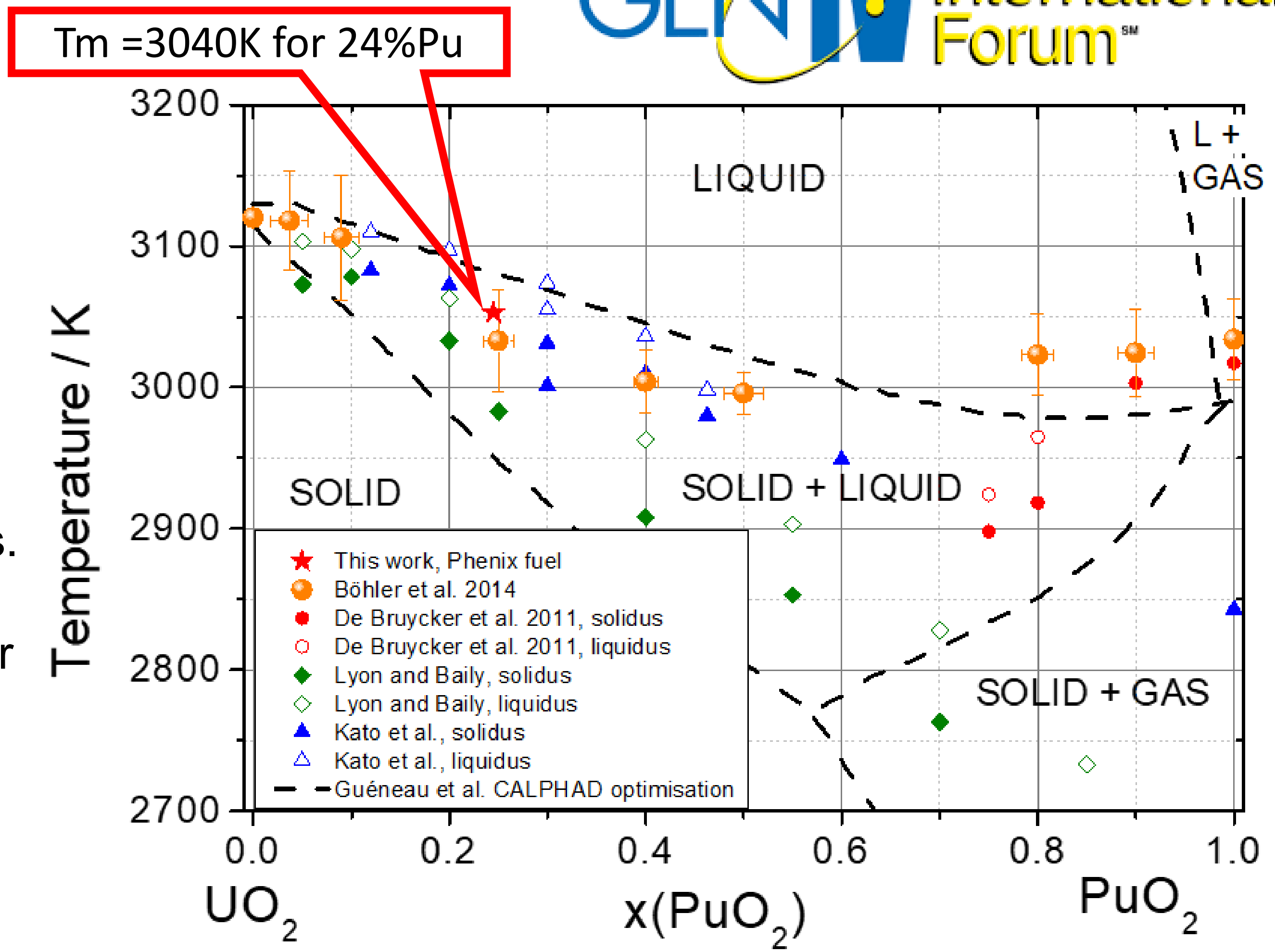
Oxide fuel

Properties	(U0.8Pu0.2)O2	(U0.8 Pu0.2)C	(U0.8Pu0.2)N	U-19Pu-10Zr
Theoretical density, g·cc	11.04	13.58	14.32	15.73
Melting point, K	3083	2		
Thermal conductivity, ($W \cdot m^{-1} \cdot K^{-1}$) at 1000–2000 K	2.6–2.4	18.		
Crystal structure	Fluoride	N		
Breeding ratio	1.1–1.15	1.2		
Swelling	Moderate	H		
Handling	Easy	Pyrophoric	Inert	Inert
Compatibility: clad	Average	Carburisation	Good	Eutectics
Compatibility: coolant	Average	Good	Good	Good
Dissolution and reprocessing	Good	Demonstrated	Risk of C14	Amenable for pyro reprocessing
Fabrication/irradiation experience	Large and good	Limited	Very little	Limited

High melting temperature
Low thermal conductivity
High margin to melt
High thermal creep (low mechanical interaction with clad)
Low swelling : pin design easier

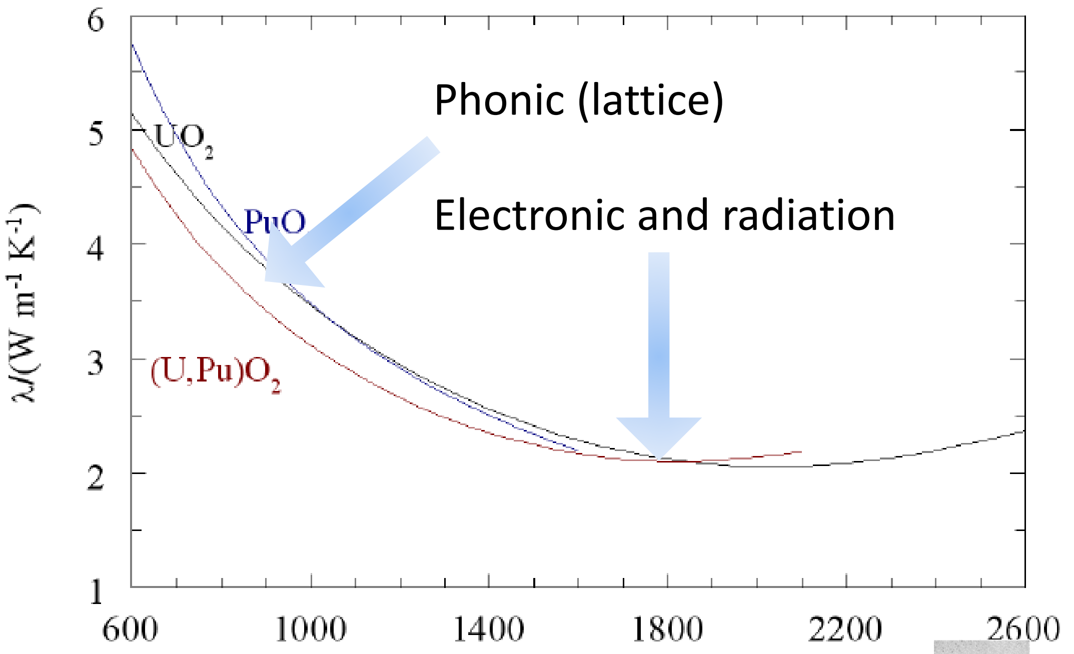
Melting temperature of (U,Pu)O2

- Existence of a minimum around 60-70% Pu content to be confirmed
- Disparity of measurements above 60% of Pu (200 K deviation for PuO_2)
- O / M impact to be evaluated
  - CALPHAD evaluation under estimates solidus.
  - The existing law for melting temperature should be revised following all these and other recent results.
- Needs for additional measurements :
  - →high Pu content
  - →Effect of O/M
  - →Pu% for the lowest $T_{melting}$ (safety analysis)

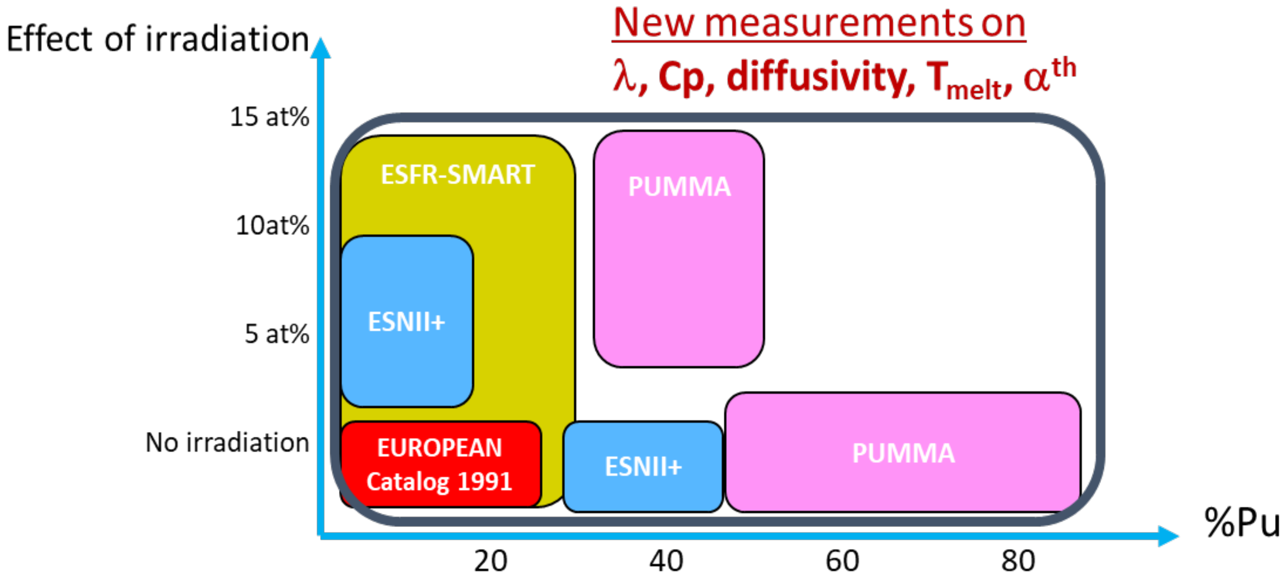


Thermal conductivity of $(U,Pu)O_2$

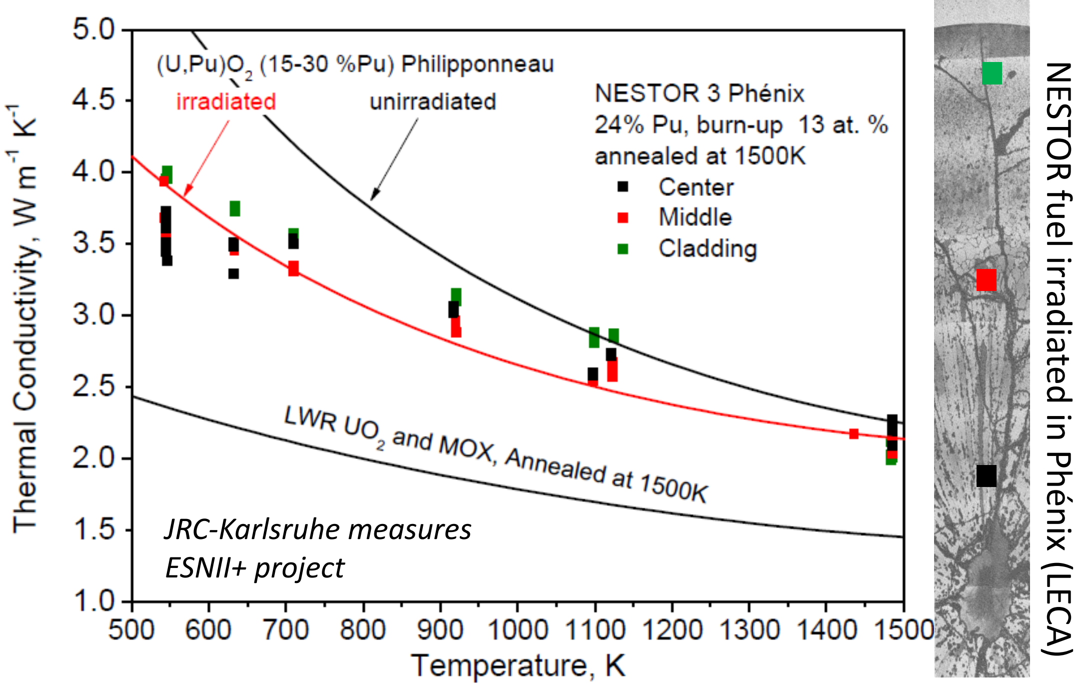
- Strong effect of temperature, O/M, Pu content, density, irradiation :
  - → Discrepancy between the laws of λ
  - → main source of uncertainty on the fuel temperature



- Intensive European experimental programme:



ESNII+ (2013-2017)	ESFR-SMART (2017-2021)	PUMMA (2020-2024)



Potential fuel designs

Fuel composition
Metal
Oxide
Nitride
Carbide
Fluoride (salt)

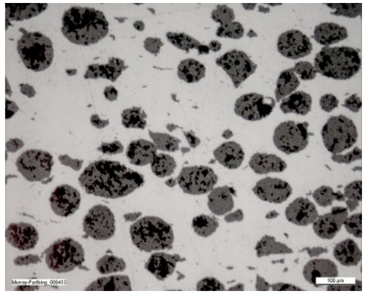
Fuel form
Single phase
Solid solution
Composite

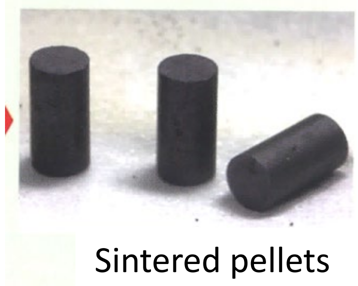
Fuel packing
Pellet
Particle
Liquid

e.g. UO_2

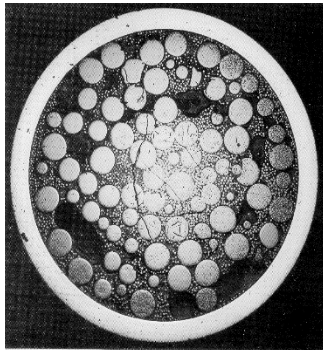
e.g. $(U,Pu)O_2$

e.g. $Mo+UO_2$





Sintered pellets

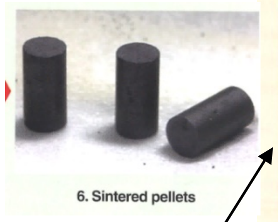


Sphere pack

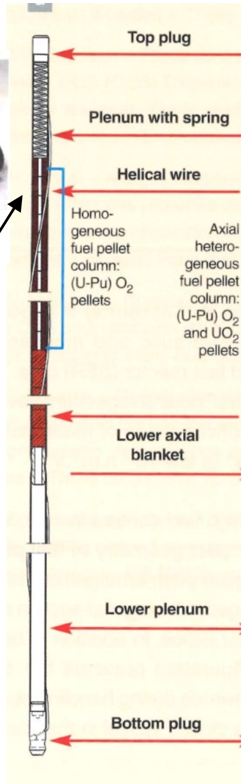
Y. Guerin International School on Nuclear Engineering , 2009
R. Konings, FJOH 2009
J. Somers, FJOH 2013
T. Wiss, FJOH 2016

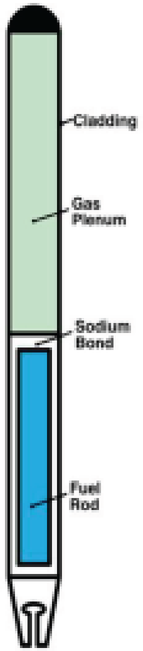
Potential fuel elements (1/2)

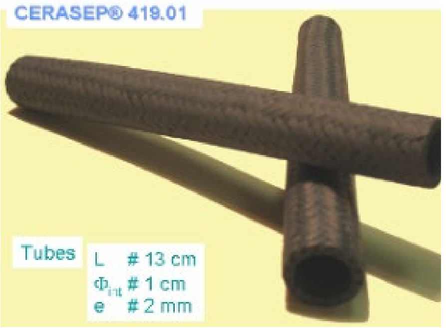
Fuel elements		
Standard pin With oxide pellets	Standard pin with metal slug	Pin with innovative clad : composite SiC-SiC_{fiber}

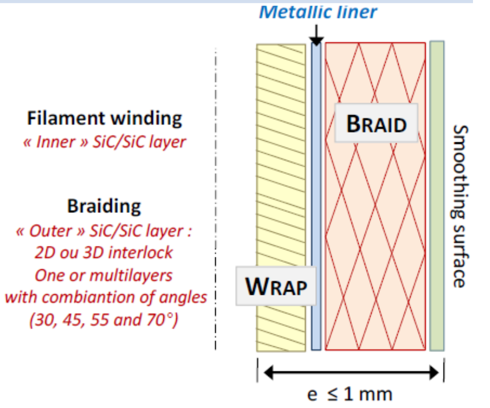


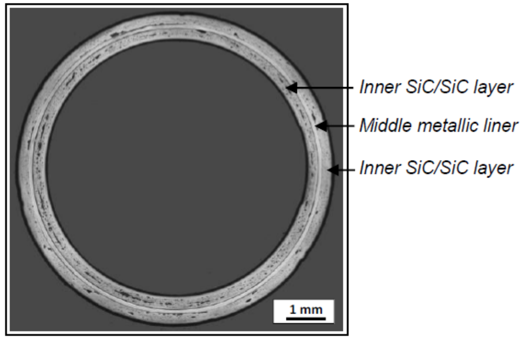
Fuel pellet column : $UPuO_2$ pellets







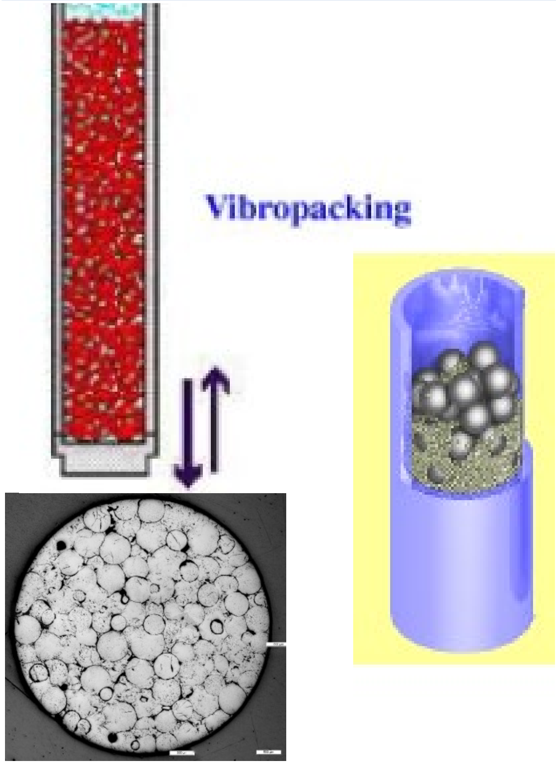


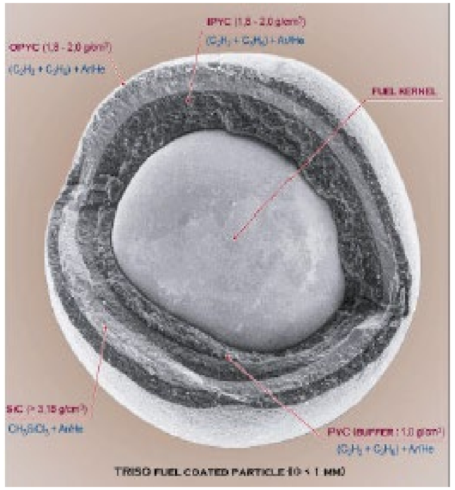


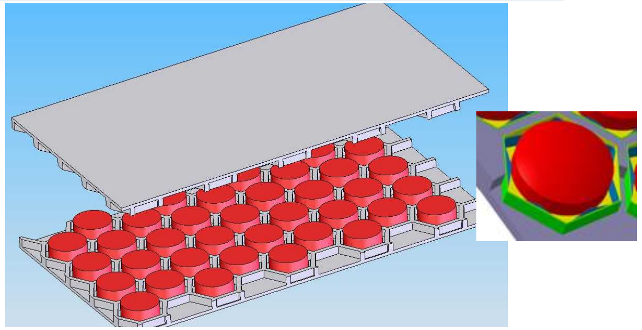
(b)

Potential fuel elements (2/2)

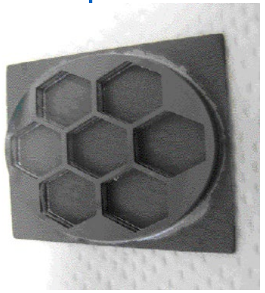
Fuel elements		
Standard pin With vipac fuel or spherepac fuel	Coated particles	Plate fuel



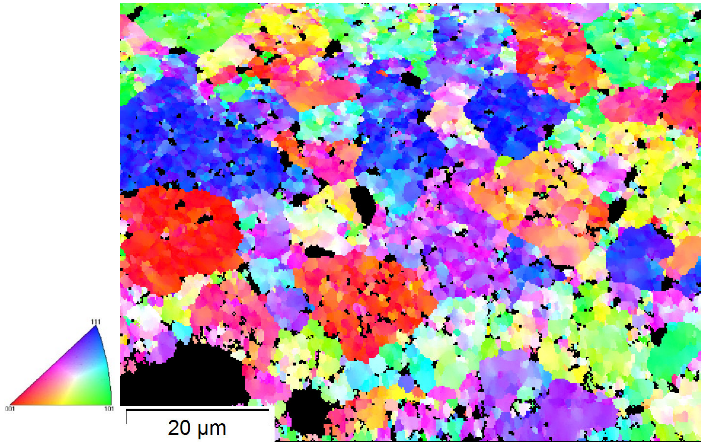




Two ceramic plates close a honeycomb structure containing cylindrical fuel pellets

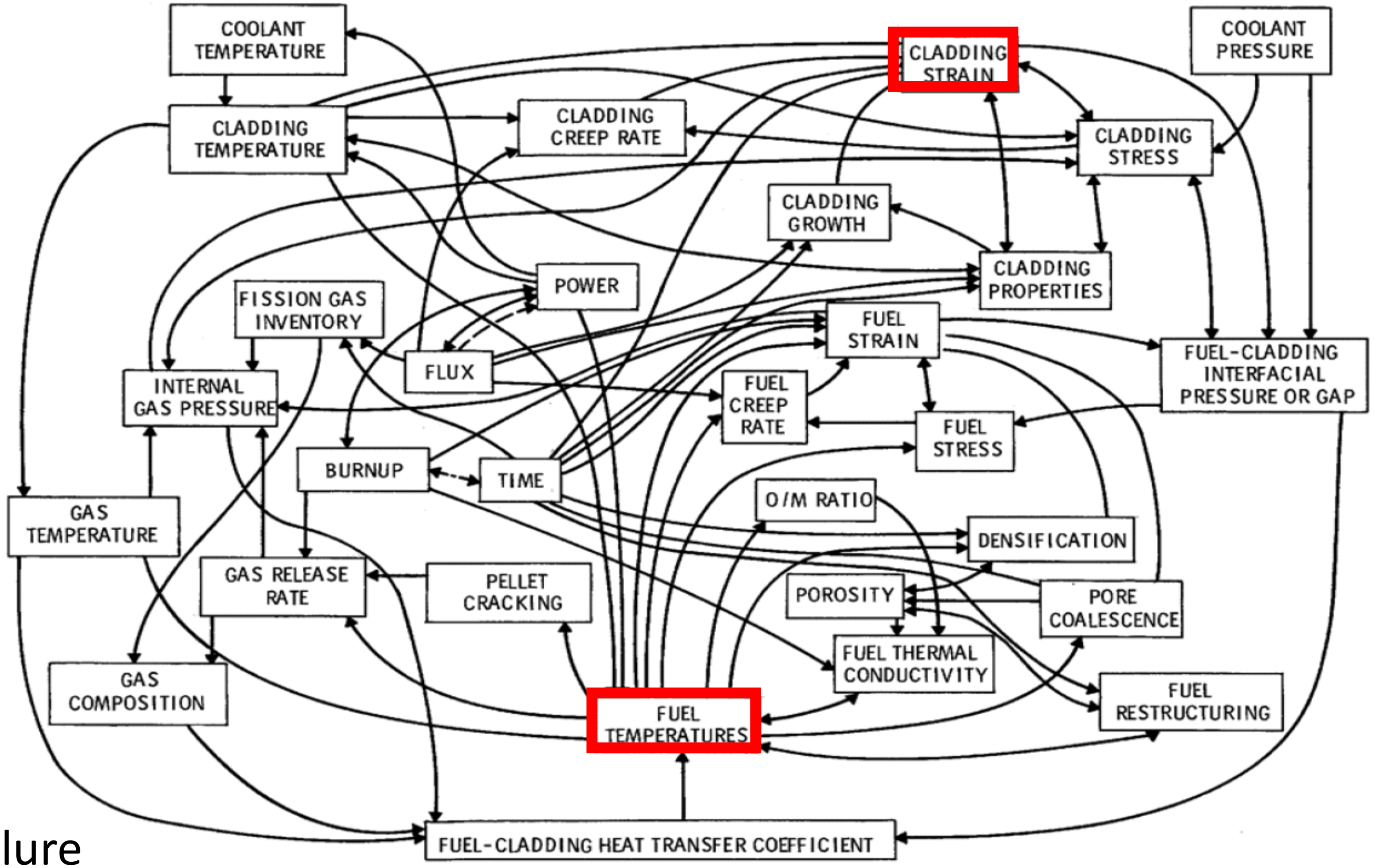


PART 2 : Fuel Behaviour Under Irradiation



Fuel behaviour : real life

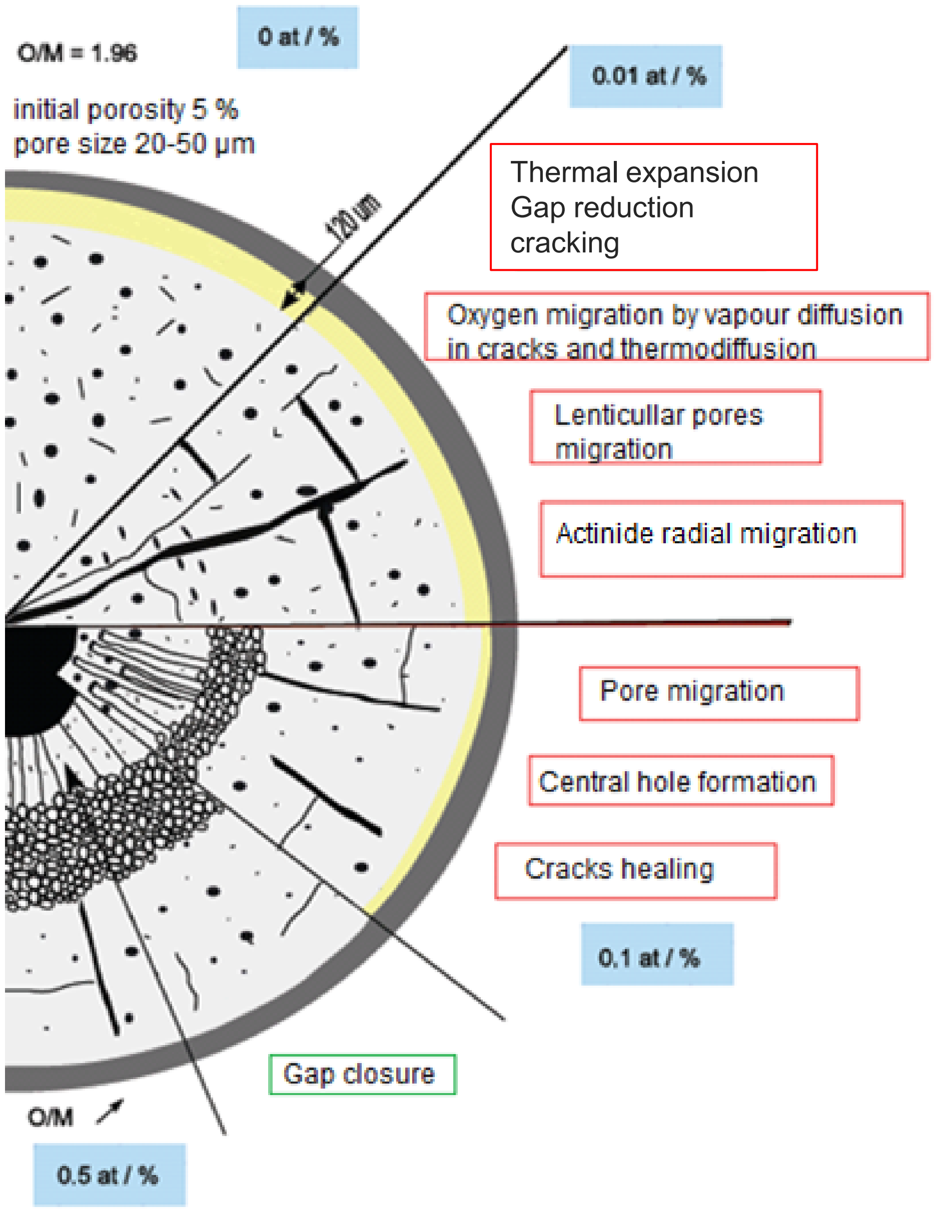
(Complete) view of how phenomenons interact
Complex system with coupled effects

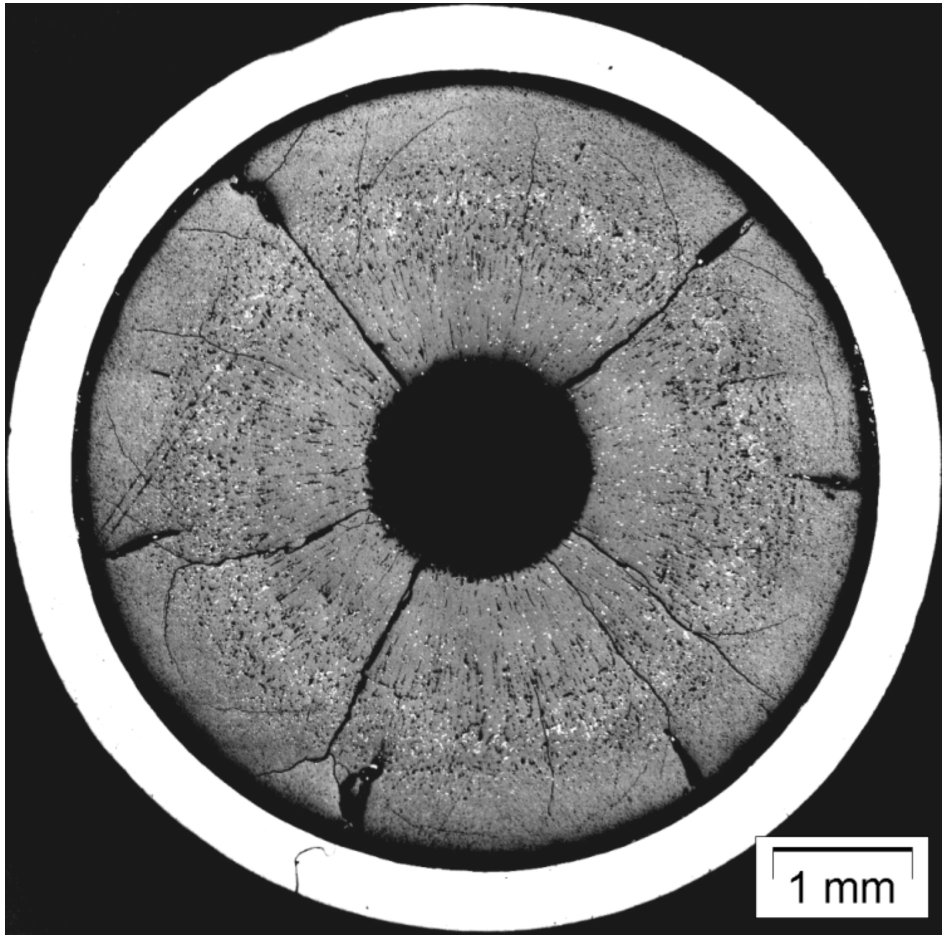


- Safety evaluation
  - Margin to clad failure
  - Margin to fuel melt

•Ref. Manning et al., MMSNF conference

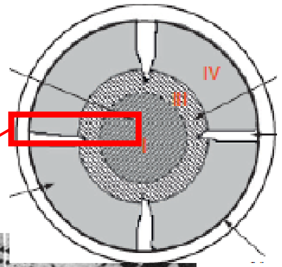
MOX behaviour : microstructure & composition evolution (1/2)



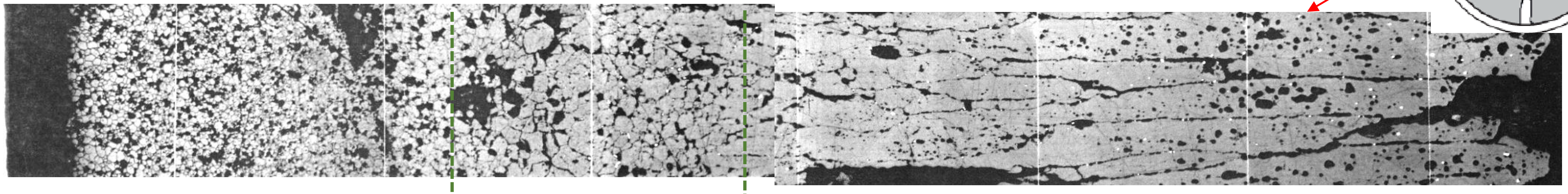


CEA-LECA

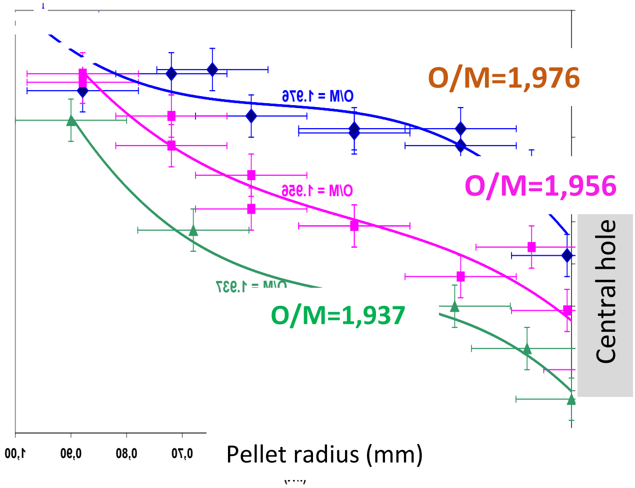
MOX behaviour : microstructure & composition evolution (2/2)



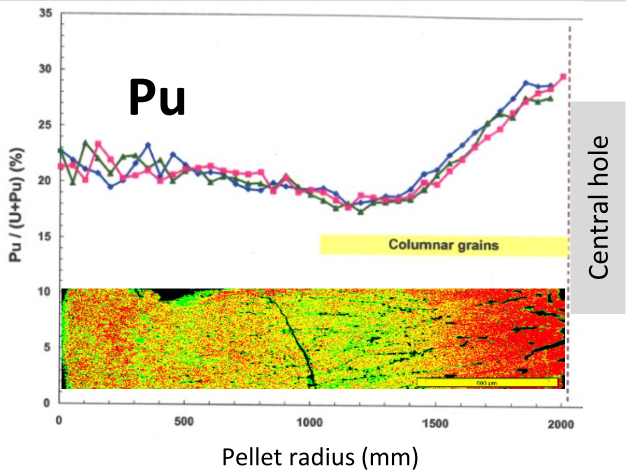
MICROSTRUCTURE RADIAL EVOLUTION



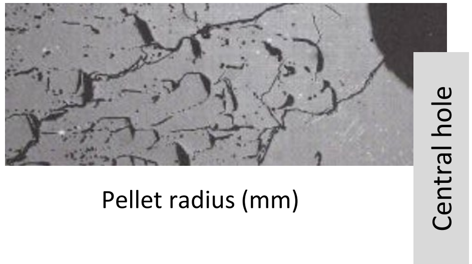
OXYGEN REDISTRIBUTION



ACTINIDES REDISTRIBUTION



PORE MIGRATION



- Atomic diffusion (thermal and athermal) of O, U and Pu (bulk diffusion)
- Grain boundary and surface diffusion
- Vaporisation condensation (pore diffusion)

Chemical state of fission products in irradiated oxide fuel

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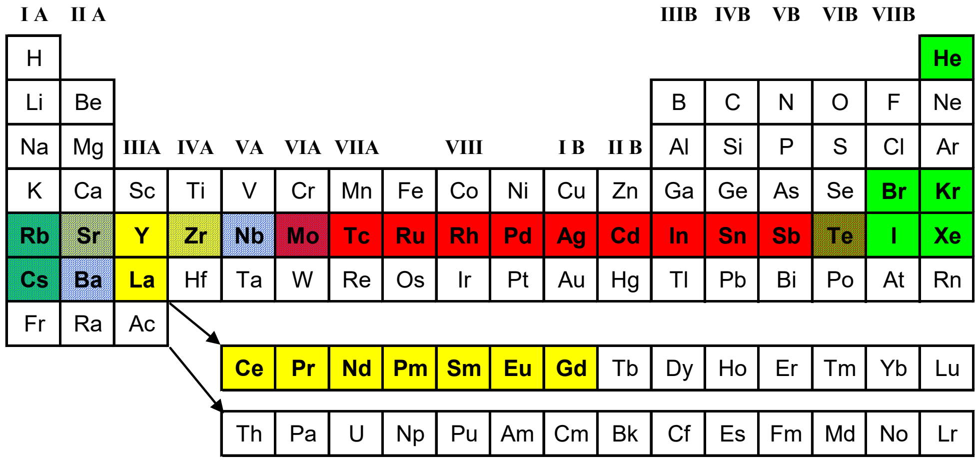
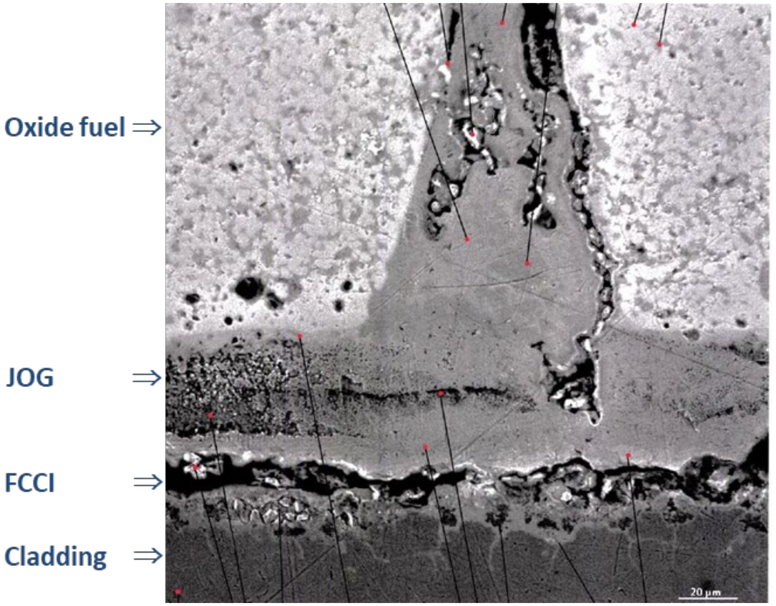
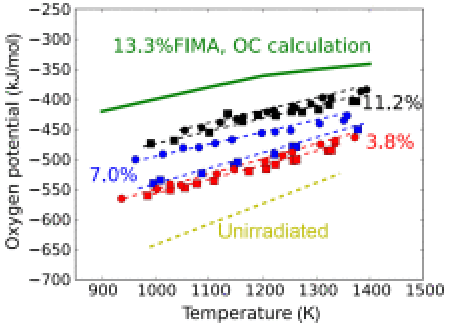
- Each fission creates 2 FPs:

FPs in solid solution
FPs forming metallic precipitates
FPs forming oxide precipitates
Gases and volatile FPs





- Chemical state of the fuel depends strongly of the oxygen chemical potential of $(U_{1-y}Pu_y)O_{2-x}$ that increases during irradiation. Fission is oxidizing.



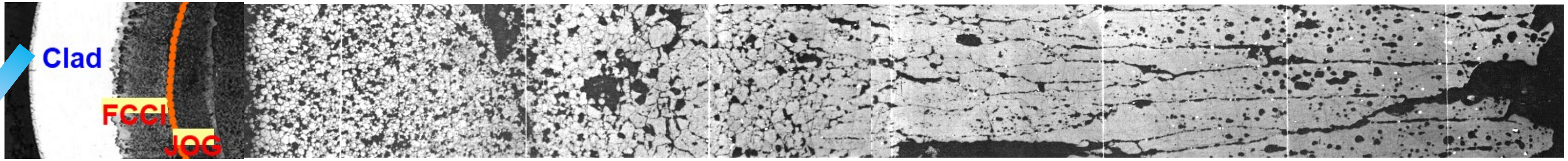
- Modification of physical and chemical properties of the irradiated fuel (**FP in solution, oxides precipitates, metallic precipitates**)

- Formation of :

→ **JOG** (oxide/clad joint) : Cs_2MoO_4 + others compounds
→ **FCCI** (Fuel Clad Chemical Interaction) or corrosion: **Te**, **I**, **Cs** reacts with clad (Fe, Ni, Cr): Cs_2CrO_4, $FeTe_{0,9}$, $NiTe_{0,6}$

MOX behaviour : effects of the irradiation

Clad and fuel evolution



CLAD SWELLING *neutrons*

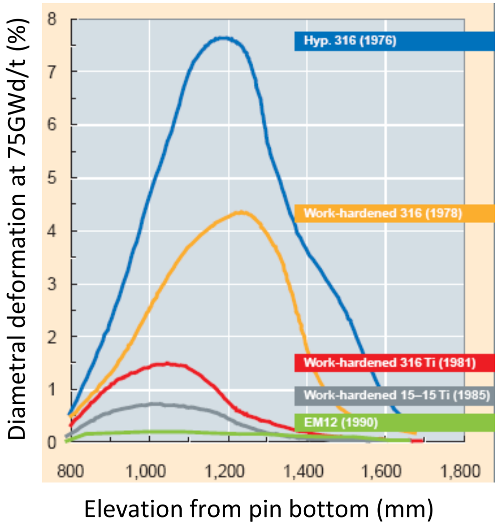
FUEL-CLAD CHEMICAL INTERACTION (FCCI or corrosion) Cs, Te, I

FISSION PRODUCTS JOINT (JOG) Pd, Mo, Te, Cs, I, O + Rb, Cd, Sn, …

FUEL GASEOUS SWELLING & GAS RELEASE Xe, Kr

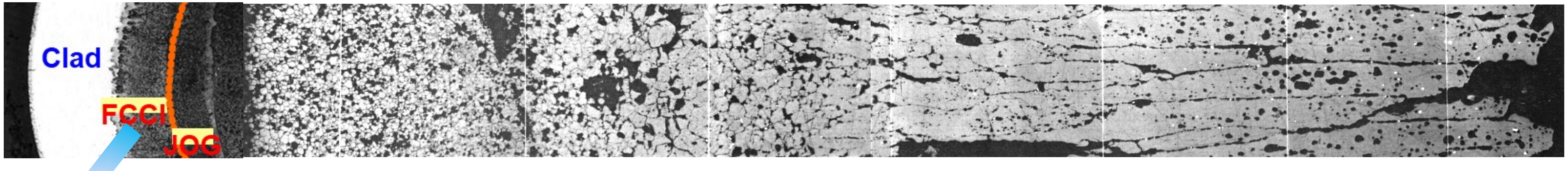
FUEL SOLID SWELLING Sr, Zr, La, Ce, Nd

FUEL PROPERTIES EVOLUTION all FP + fuel damage



MOX behaviour : effects of the irradiation

Clad and fuel evolution



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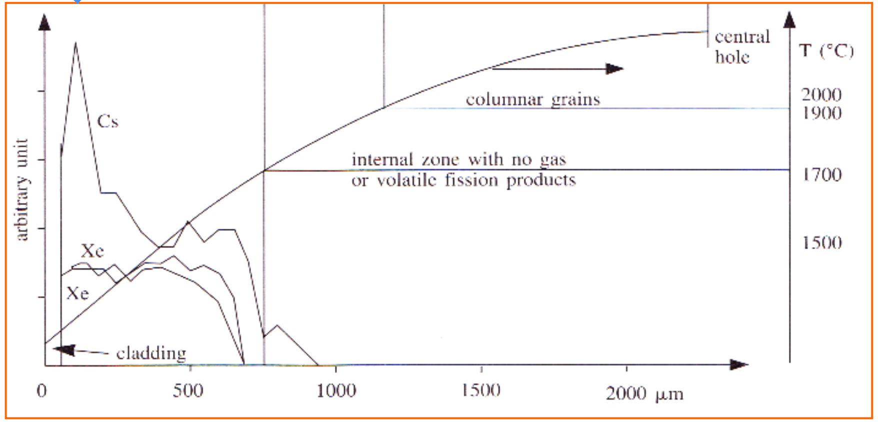
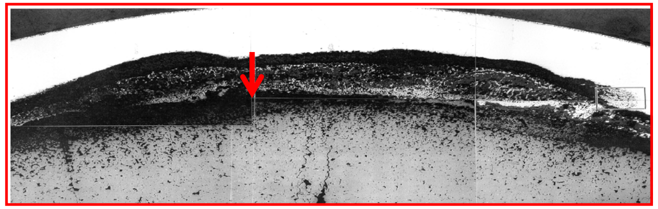
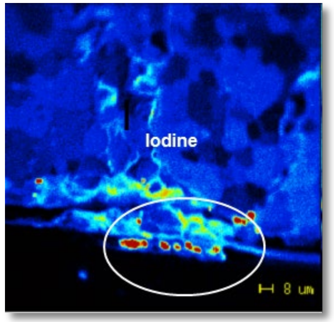


Corrosion thickness : until 200 µm

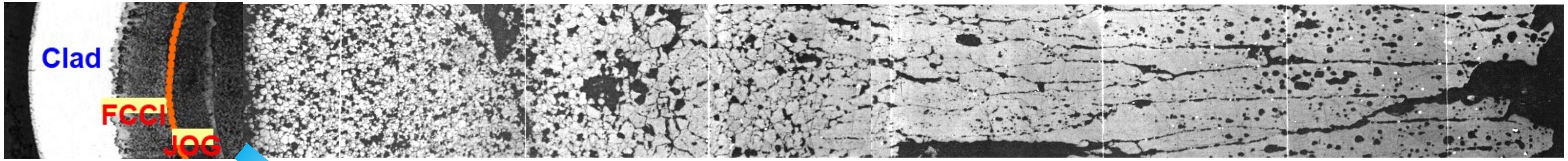


CEA-LECA



MOX behaviour : effects of the irradiation

Clad and fuel evolution



CLAD SWELLING *neutrons*

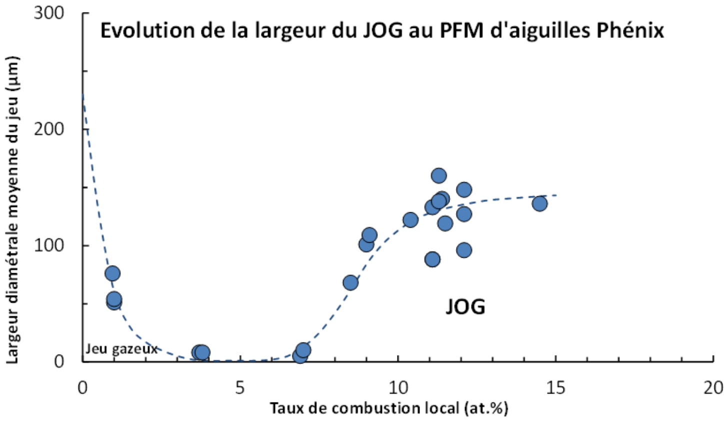
FUEL-CLAD CHEMICAL INTERACTION (FCCI or corrosion) Cs, Te, I

FISSION PRODUCTS JOINT (JOG) Pd, Mo, Te, Cs, I, O + Rb, Cd, Sn, …

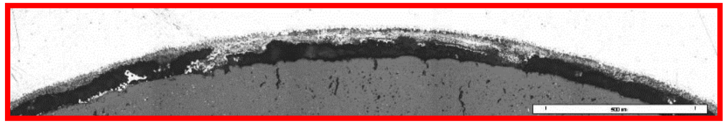
FUEL GASEOUS SWELLING & GAS RELEASE Xe, Kr

FUEL SOLID SWELLING Sr, Zr, La, Ce, Nd

FUEL PROPERTIES EVOLUTION all FP + fuel damage



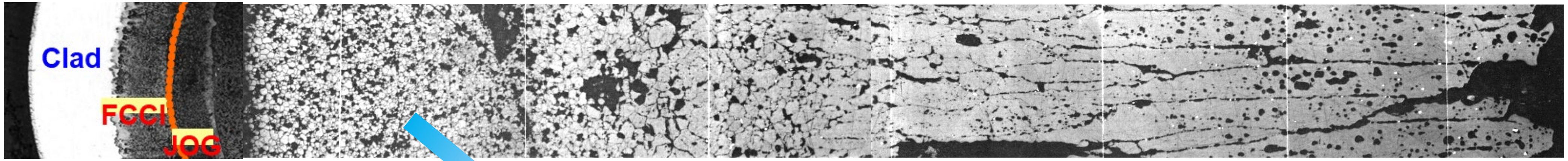
CEA-LECA



JOG thickness : until 160 µm

MOX behaviour : effects of the irradiation

Clad and fuel evolution



CLAD SWELLING *neutrons*

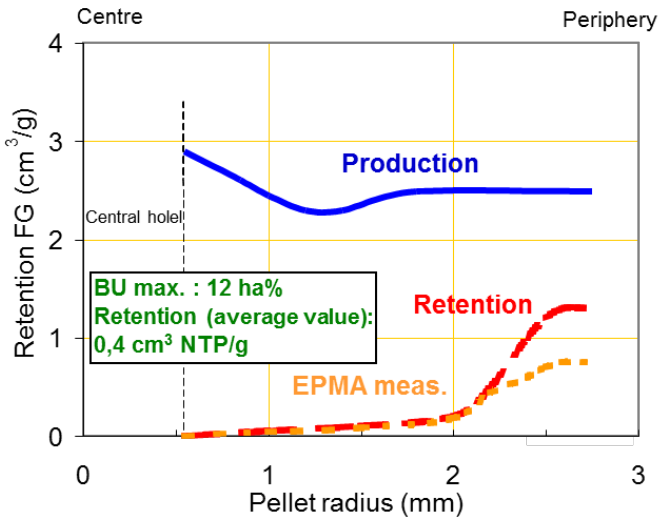
FUEL-CLAD CHEMICAL INTERACTION (FCCI or corrosion) Cs, Te, I

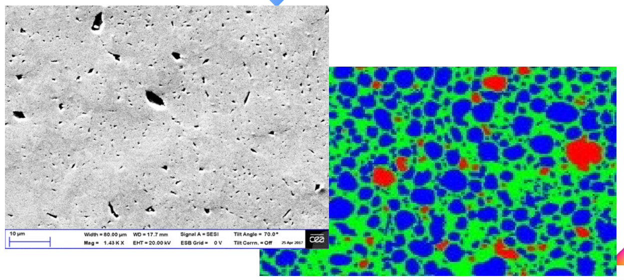
FISSION PRODUCTS JOINT (JOG) Pd, Mo, Te, Cs, I, O + Rb, Cd, Sn, …

FUEL GASEOUS SWELLING & GAS RELEASE Xe, Kr

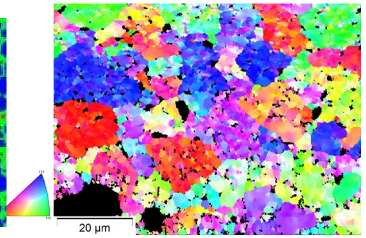
FUEL SOLID SWELLING Sr, Zr, La, Ce, Nd

FUEL PROPERTIES EVOLUTION all FP + fuel damage

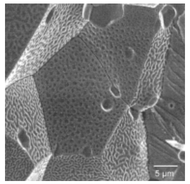




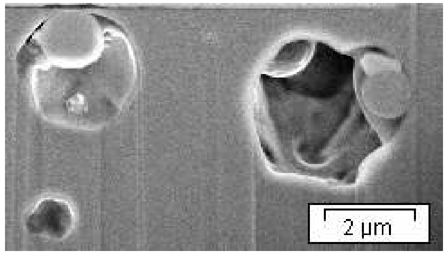
EPMA



EBSD



TEM



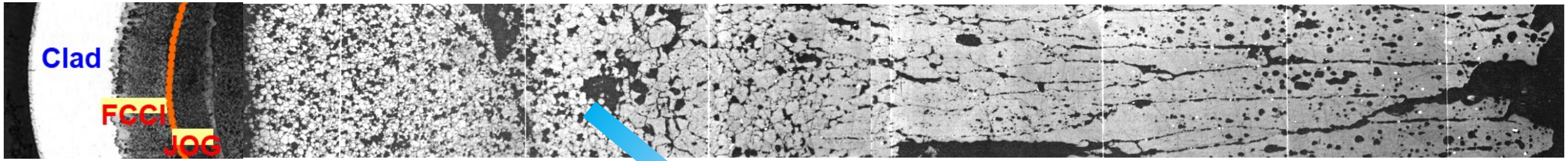
MEB-FIB

CEA-LECA

Noirot et al, Nuc. Eng. Tech., 50, 2018

MOX behaviour : effects of the irradiation

Clad and fuel evolution



CLAD SWELLING *neutrons*

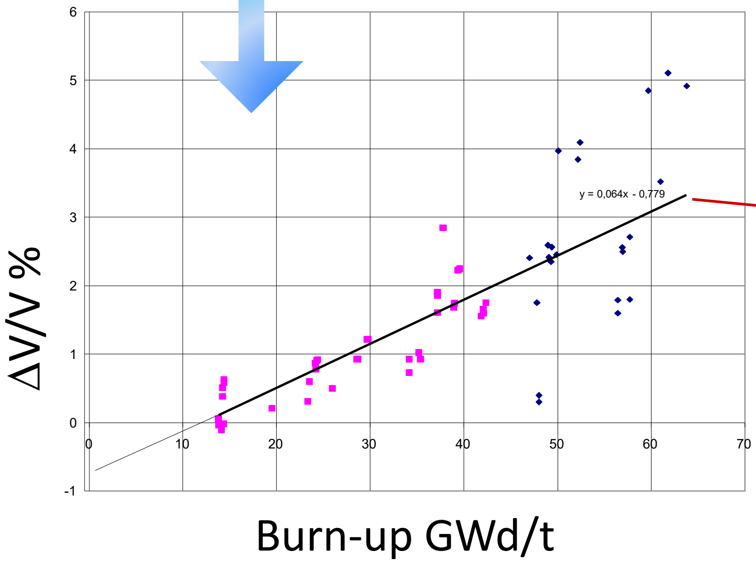
FUEL-CLAD CHEMICAL INTERACTION (FCCI or corrosion) Cs, Te, I

FISSION PRODUCTS JOINT (JOG) Pd, Mo, Te, Cs, I, O + Rb, Cd, Sn, …

FUEL GASEOUS SWELLING & GAS RELEASE Xe, Kr

FUEL SOLID SWELLING Sr, Zr, La, Ce, Nd

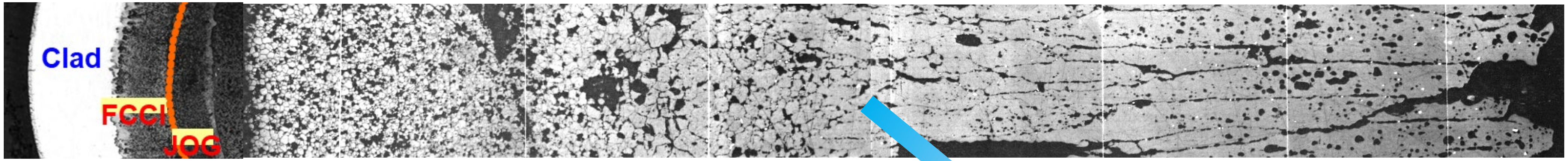
FUEL PROPERTIES EVOLUTION all FP + fuel damage



$\Delta V/V \sim 0{,}6$ % / at%

MOX behaviour : effects of the irradiation

Clad and fuel evolution



CLAD SWELLING *neutrons*

FUEL-CLAD CHEMICAL INTERACTION (FCCI or corrosion) Cs, Te, I

FISSION PRODUCTS JOINT (JOG) Pd, Mo, Te, Cs, I, O + Rb, Cd, Sn, …

FUEL GASEOUS SWELLING & GAS RELEASE Xe, Kr

FUEL SOLID SWELLING Sr, Zr, [illegible], Ce, Nd

FUEL PROPERTIES EVOLUTION all FP + fuel damage

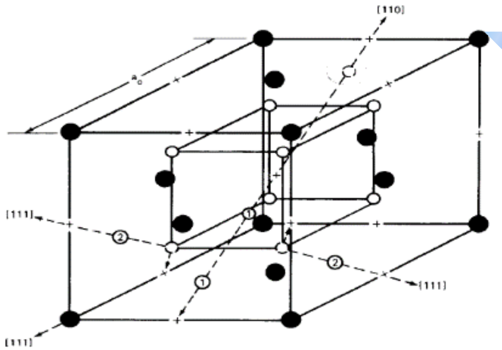
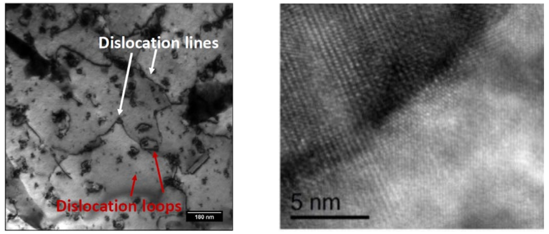


Fig. 11.5 Defect complex in UO_2. ●, uranium ions. ○, normal oxygen. (1), type 1 interstitial oxygen. (2), type 2 interstitial oxygen. ( ), vacancy in normal oxygen site. +, interstice at center of cube formed by eight normal oxygen sites.

D.R. Olander (1976)



TEM

CEA-LECA

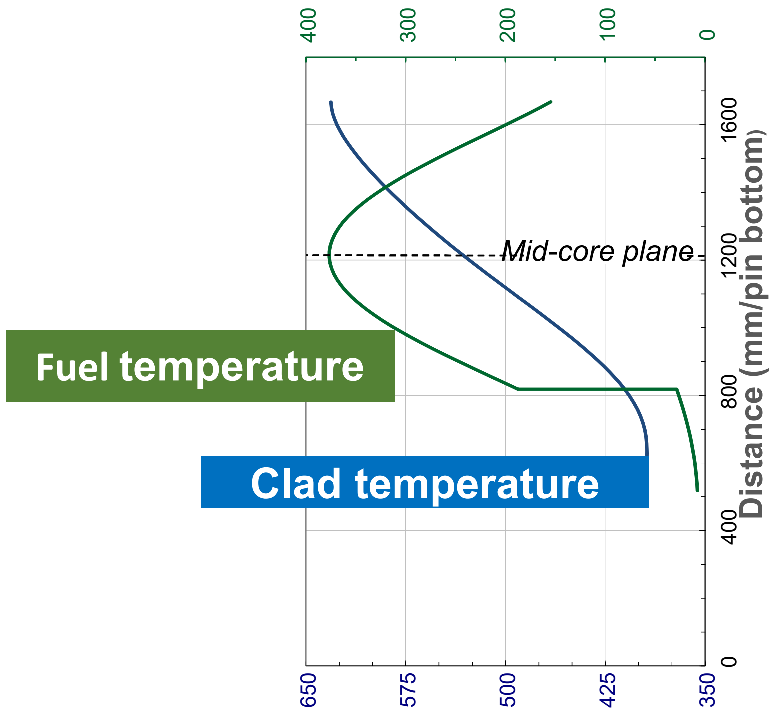
Onofri et al, J. Nuc. Mat., 482, 2016

Thermal behaviour of fuel element

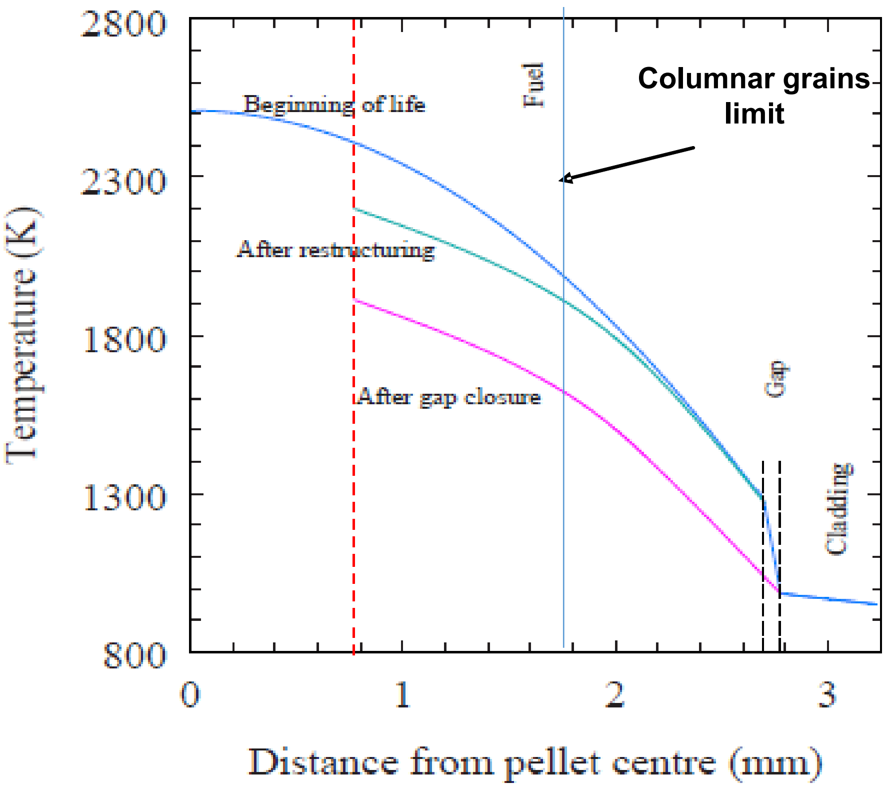
- **Objectives :**
  - Predict the temperature of clad and fuel with an evaluation of margin to melt

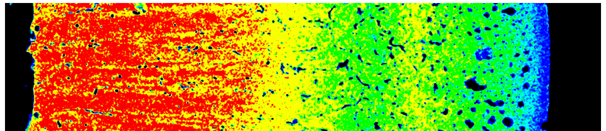
- **Thermal profile in the pin**

Axial



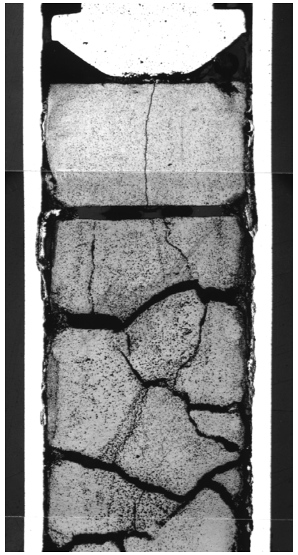
Radial





Factors affecting the temperature profile
- Thermal conductivity degradation
- Fuel swelling
- Fuel restructuring
- Pellet-cladding interaction

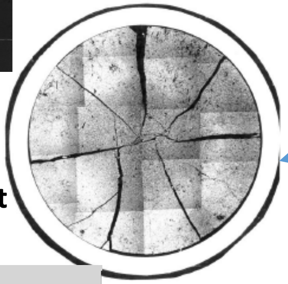
Mechanical behaviour of fuel element (1/2)



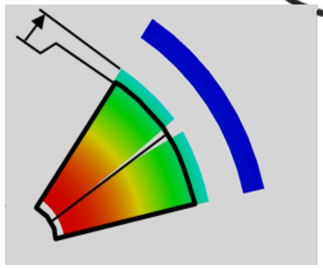
- **Objectives :**
  - Predict dimensional changes: clad strain (5-10% of max. strain), gap closure
  - Predict the risk of clad failure during nominal conditions or power increases

- **Mechanical phenomenons**
  - Fuel :

    swelling, creep, mechanical properties evolution, cracking due to very high thermal gradient (5000K/cm) → differential expansion leading to fracturing of the pellets, relocation of the fragments

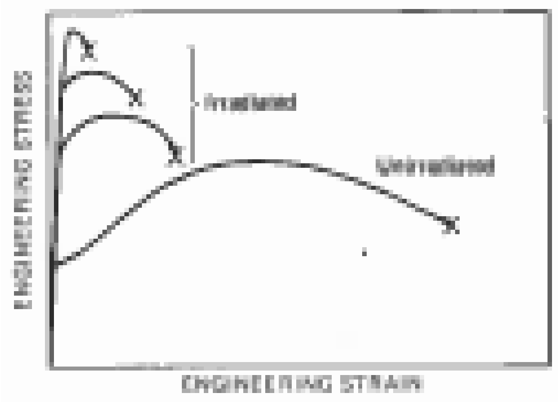


Relocation displacement induced by pellet fragmentation



  - Clad:

    swelling at high dose, creep (causing damage), embrittlement, loss of properties during irradiation



Mechanical behaviour of fuel element (2/2)

✓Fuel / cladding mechanical interaction (FCMI)

In fast oxide fuels, this benign effect of FCMI during steady-state operation is a consequence of the following points:

- Low swelling rate(~0,6%/at%) of oxide as compared to carbide or metal fuels
- High fuel temperature that allows high creep rates. Even in the outer part of pellets, thermal creep and irradiation creep in oxide fuels relieve the FCMI stresses induced by fuel swelling.
- JOG formation that acts as an elastic joint (composition Cs, Mo, O, mainly Cs_2MoO_4)
- Design of fuel element in order to avoid FCMI through low smear density and with plenum to avoid primary stresses (pressure due to fission gas release)

✓Fuel /cladding chemical interaction (FCCI)

- Reduce clad thickness and thereby limiting fuel burn-up
- FCCI may be evaluated with fission products diffusion, thermodynamics of fuel/clad interface

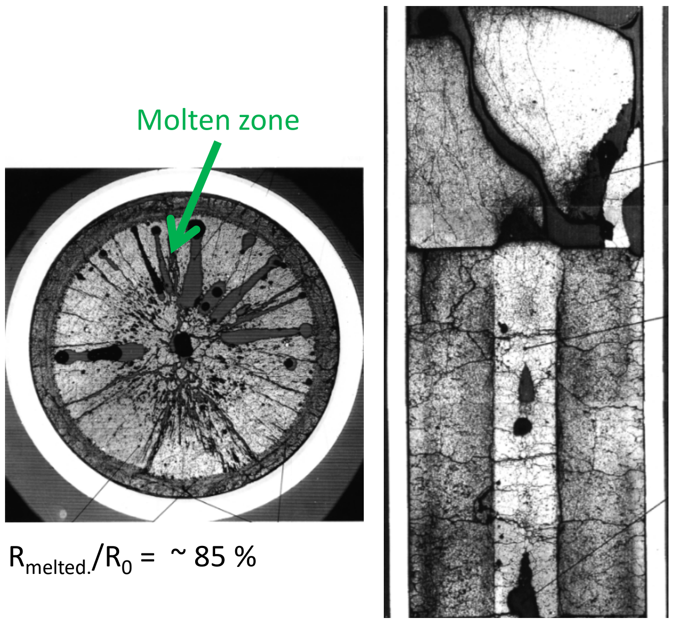
MOX behaviour during accidents

➢Unexpected Control Rod Withdrawal Accident (CRWA) : slow/limited power transient (SA prevention domain: demo of pin integrity)
➢Unprotected Transients : transient of power or loss of flow (UTOP/ULOF) : fast/very large power transient (SA mitigation domain: demo of fuel ejection phenomena)
➢Total Instantaneous assembly inlet Blockage (TIB) : (SA mitigation domain: study of heat transfer and propagation phenomena)

Associated R&D aimed at:

- Data acquisition from separate effect or integral tests (e.g. single rod in CABRI reactor or with fuel bundle in SCARABEE : from 19 to 37 pins). Feedback exists (CABRI, TREAT, …) but still low and expensive, so a large panel of scenarii and uncertainties still exist.
- Development of simulation codes validated on tests with complex interactions between a lot of phenomena.

	Programme	Objectives/Comments
1971-1974	SCARABEE 1st phase	Na ebullition, fuel melt; 16 single-pin tests, 8 bundle tests (7 pins)
1976-1986	CABRI-1	CDA primary phase, 32 tests, full pellets, 316 ε steel, burnup: 0-1-2, 9-4.8 at%
1983-1990	SCARABEE-N	TIB/APL (10 tests), CDA (3 tests), fuel bundle tests (19 and 37 pins)
1986-1994	CABRI-2	CDA (9 tests) and CRWA (3 tests) Industrial fuel, full/annular pellets; 15-15 Ti clads
1992-1997	CABRI FAST	Annular pellets; CDA (5 tests), CRWA (2 tests)
1996-2001	CABRI RAFT	CDA transition phase (5 tests), fuel ejection during CRWA (2 tests)



Fuel partly melt during (CABRI – E5 test)

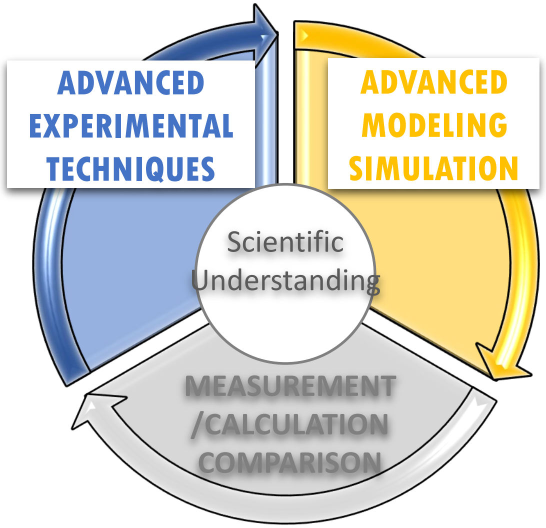
FJOH 2016, N. Girault
ISNE 2020 C. Struzik, V. Blanc
Y. Fukano et al. Journal of NUCLEAR SCIENCE and TECHNOLOGY, Vol. 46, No. 11, p. 1049–1058 (2009)
I. Sato et al. NUCLEAR TECHNOLOGY VOL. 145 JAN. 2004

SFR MOX driver fuel : main features of the irradiation behaviour

- Microstructure, composition and properties evolution
  - Microstructure evolution early during the irradiation
  - Species transport (actinides and O migration) and oxygen potential evolution
  - Fuel gaseous and solid swelling, coupled with creep
  - Fuel properties evolution depending on : composition, density, microstructure, temperature, burn-up
  - Clad properties : creep, swelling at high burn-up/temperature, embrittlement, loss of mechanical properties under irradiation

- Thermomechanical and thermochemical behaviour
  - Fission gas release : effect of all parameters (T, Burn-up, fuel microstructure,…)
  - Fuel to clad gap closure and Heat transfer in the fuel-clad gap
  - JOG formation and axial transfert
  - Pellet cracking and re-location of the fragments
  - FCMI: threshold of over-power or over-temperature during transients
  - Burn up linked phenomena: FCCI (clad corrosion), JOG composition

PART 3 : Fuel Element Performances, Design and Qualification



Performances of MOX Pins in SFR

- Long experience:
  - Started in BR-5 in 1957 in Russia, Rapsodie in 1967 in France, SEFOR in USA,
  - Then EBR-II and FFTF in the USA, BR-10, BOR-60, BN-350, BN-600 and BN-800 in Russia, the prototype fast reactor (PFR) in the UK, Phenix and Superphenix in France, KNK and SNR-300 in Germany, JOYO and Monju in Japan, FBTR in India and Experimental Fast Reactor (CEFR) in China.

- Performances in recent SFR :

  →more than 20 at% (200 GWd.t-1)

  →155 dpa

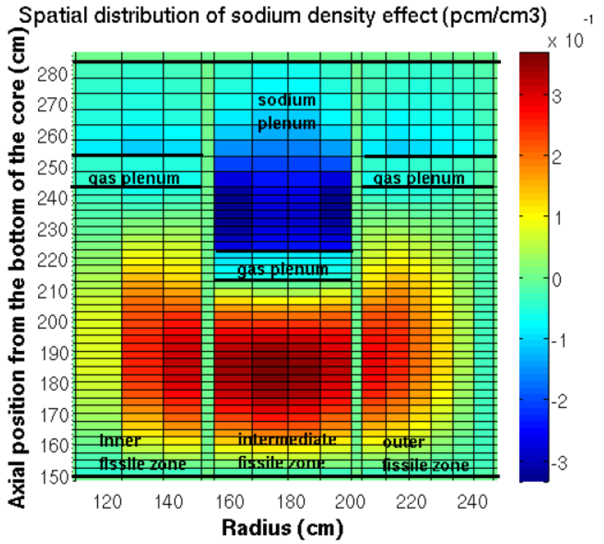
  → 550 W/cm max

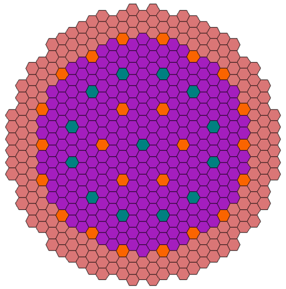
Standard MOX fuel		Experimental fuel		
No. of pins irradiated	Burnup reached $MW \cdot d \cdot t^{-1}$	Max. burnup $MW \cdot d \cdot t^{-1}$	Main reactors	Type of fuel pellets
265000	135000	200000	PHÉNIX, PFR, KNK-II	Solid&annular
64000	130000	200000	FFTF	Leading pins
50000	100000	120000	JOYO	Solid
13000	135000	240000	BOR-60	Vibro-pac
1800	100000	-	BN-350	Solid&annular
1500	100000	-	BN-600	Solid&annular

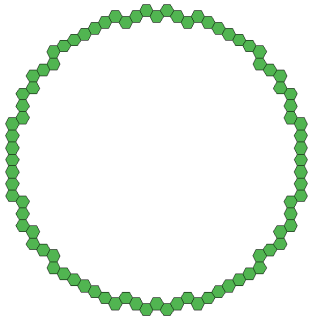
- Fast reactor in operation :
  - BOR60, BN600, BN800 : UOx and MOX, pellet or vi-pack
  - JOYO : MOX, pellet
  - FBTR : carbide and MOX experimental pins and *PFBR : MOX, pellet*
  - CEFR : UOX, pellet

Improvements in the Fuel Element Design

- Improvements on the geometry:
  - Annular pellet (for safety improvement : BOL & transients)
  - large pin diameter
  - Axially heterogeneous pin (safety improvement)
- Large range of composition :
  - Uranium: natural, depleted, reprocessed
  - Plutonium: 15 to 45%, several grades (ex spent LWR-MOX)
  - Minor actinides : transmutation with different ways $(U,Pu,Am)O_{2-x}$, $(U, Am)O_{2-x}$, $(MA,O)_{2-x}$+ inert matrix
- Specifications:
  - Mastered by several processes
  - Adapted to industrial fabrication
  - Responding to safety issues







Fuel Element Qualification

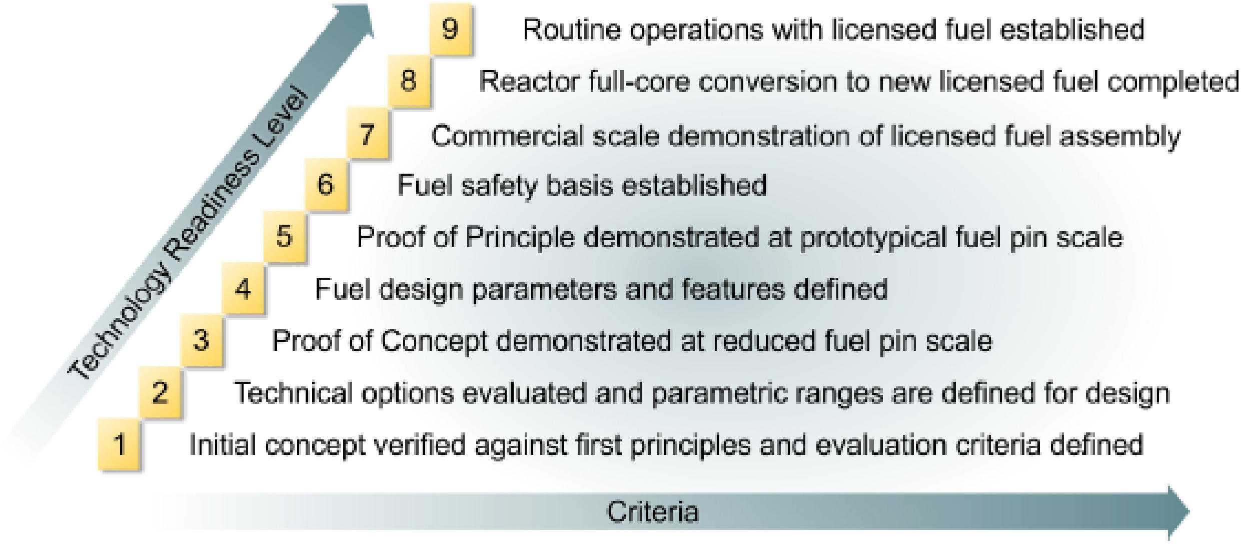
- Objectives :
  - Licensing : authorisation of safety authorities for fuel loading in NPP

- Requirements:
  - regulatory guidance :
    - higher level safety objectives,
    - qualification of computational tools,
    - uncertainties consistent with Safety Margins
  - regulatory criteria/limits :
    - maintaining cladding integrity, coolable geometry, and limiting radiological consequences
    - fuel failure and degradation mechanisms are identified & controlled

Fuel Element Qualification

- An essential part of fuel qualification is to define a test envelope to cover expected operating, transient, and accident conditions to assess fuel performance and validate fuel performance codes.

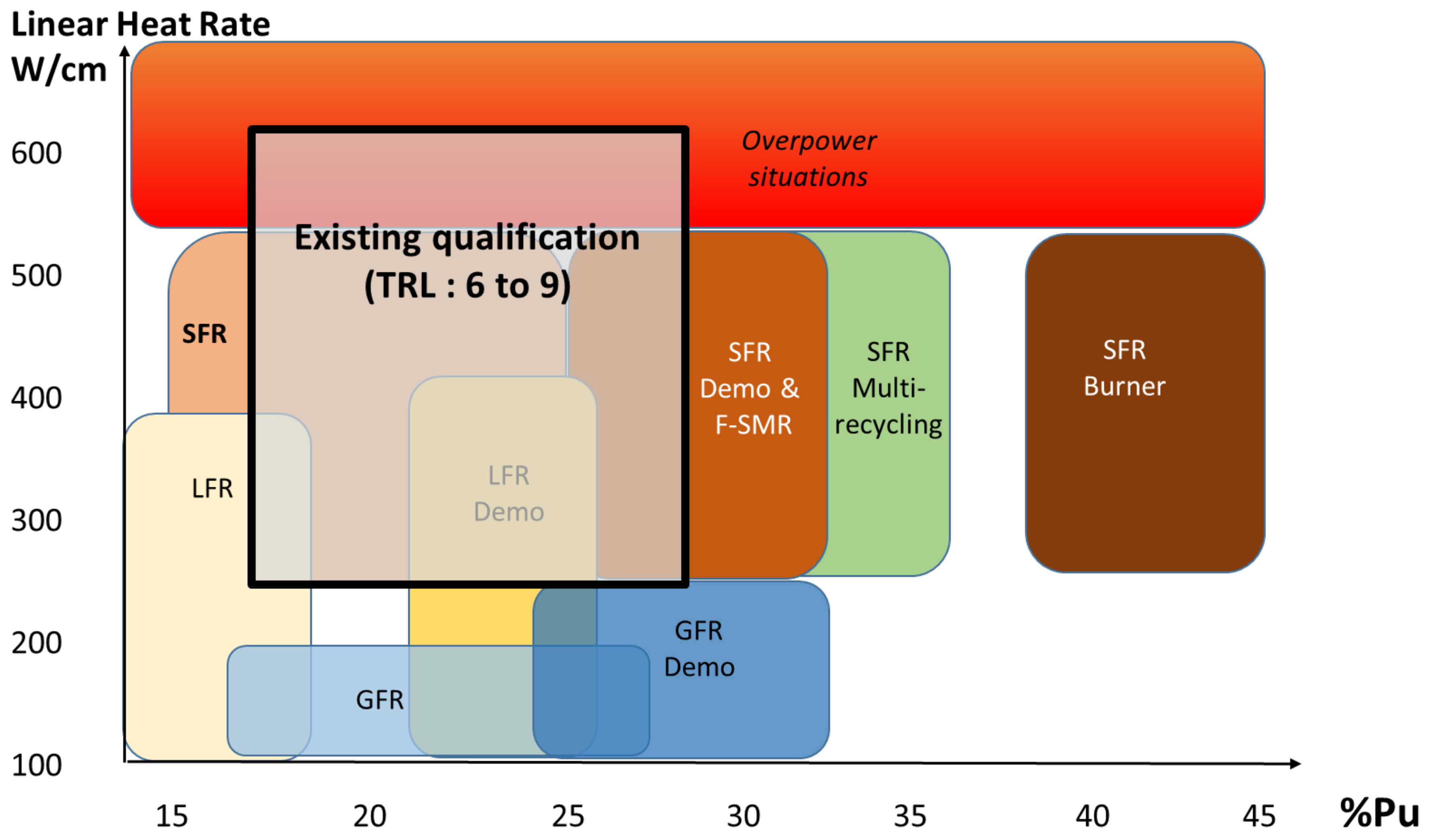
	Fundamental Property Measurements	Out-of-Pile Testing	In-Pile Testing with Intermediate conditions (< target burnup)	In-Pile Testing with Prototypic conditions (> traget burnup)	Transient Testing	Reactor Operations
Multiple Assemblies (Core Loads)						TRL 8 - 9
Bundle / Sub-assemblies				TRL 6 - 7	TRL* 7 - 8	TRL 8
Pins		TRL 4	TRL 5	TRL 6	TRL 6 - 7	
Samples & Rodlets	TRL 4	TRL 4	TRL 4	TRL 5	TRL 5 - 6	



TRL 9 for Phenix type pins (Phenix, SNR300, Joyo) & same geometry with central hole (EBR2, PFR, BN800), SPX type
TRL 6-7 for others concepts

D. Crawford et al. , Journal of Nuclear Materials, 371 (2007) 232-242.
OECD – report No. 6895 Dec. 2014. “State-of-the-art Report on Innovative Fuels for Advanced Nuclear Systems”.

QUALIFICATION OF MOX FUEL FOR GENIV SYSTEMS



Fuel performance code qualification : ex. platform PLEIADES

with GERMINAL for fast reactor

Material evolution → **Material damage & properties**

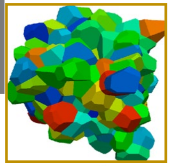
Thermal analysis → **Fuel temperature distribution**

Mechanical calculations → **Mechanical fields**

Physico-chemistry analysis → **Fission gas Location/Migration/ Release**

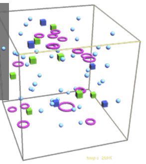
Thermochemistry → **Fission product speciation Thermo-migration Source term**

VER

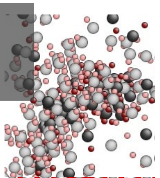


Dislocation Dynamics

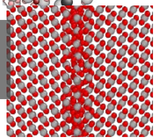
Cluster dynamics

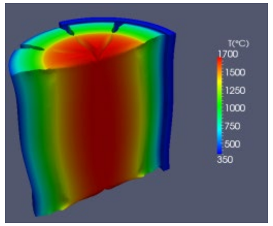


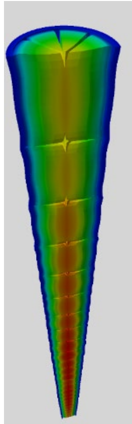
Empirical Potential MD

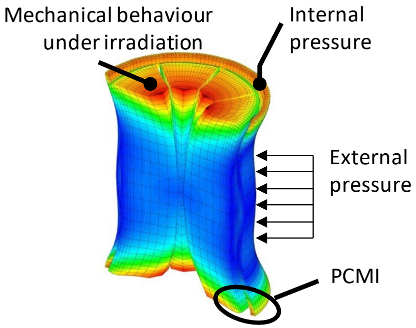


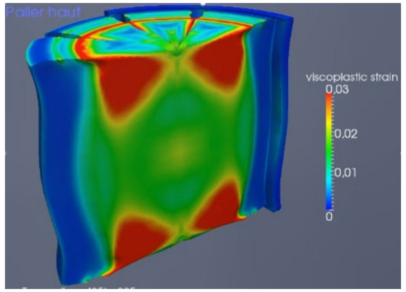
First Principles DFT

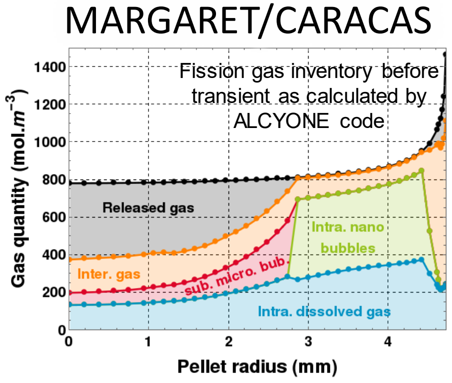


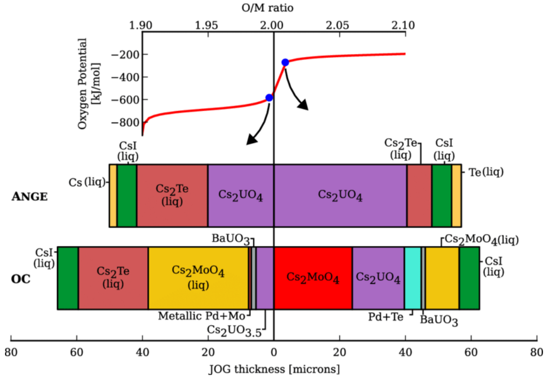












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Portelette et al, J. Nuc. Mat., 510, 2018
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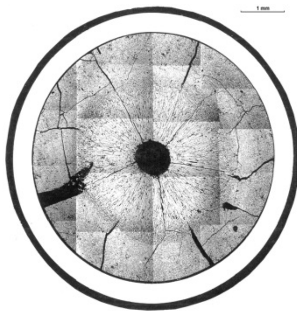
Synthesis & Conclusion

Fuel for SFR : the feed back

- Oxide fuels
  - Demonstrated a good stability and behaviour under irradiation up to very high burn-ups (20 at.%), limitation is due to clad and wrapper deformation.
  - High creep rate (high temperature) and optimised pin design (smear density) to avoid FCMI
  - Low thermal conductivity compensated by high melting point
  - Compared to metal fuel, lower fuel swelling under irradiation
  - Na reaction to be managed as well as clad corrosion at high burn-up
  - Compatibility with stainless steel cladding
  - Large feed back of safety tests
  - Fuel Performance Codes : numerous and qualified on a set of reliable exp. tests
  - Manufacturing and reprocessing processes similar to the Light Water Reactors (LWR) fuel industrial processes, taking advantage of LWR experience and existing facilities
  - Fuel cycle : well known fabrication process and large experience on reprocessing
  - Scenario : flexible towards Pu management (% and grade), Uranium use and high capabilities for Minor Actinides transmutation



Phénix - CEA

Perspectives for R&D on MOX Pins

- Fuel material:
  - Properties measurements and recommandation for uncertainties reduction
  - Different fuel compositions for an enhanced flexibility towards fuel cycle options

- Clad material :
  - Development of several candidates for high neutron doses (>150dpa) with low swelling and FCCI resistance

- Fuel element :
  - Improvement design :
    - Annular pellets (→ enhance thermal behaviour and transient consequences)
    - Large pin diameter (→ reducing Na volume and structural materials)
  - Assesment of pin behaviour during incidental and accidental scenarios
  - Modelling – Simulation : thermochemical, thermomechanical, 3D, multiscale approaches

SOURCES

- **Bibliography**
  - Comprehensive Nuclear materials, Konings, R. J. M., Ed. Elsevier 2012.
  - Advances in Nuclear Chemistry, M.H.A. Piro, Elsevier 2020.
  - The nuclear fuel of pressurized water reactors and fast neutron reactors, H. Bailly, et al. 1999, Pergamon Press. 660.
  - H. Matzke, Science of Advance LMFBR Fuels. Amsterdam: North-Holland (1986)
  - Dedicated reports or **TECDOC from IAEA** and **Sate Of the Art Reports from OECD/NEA**
  - Publications mainly from ***Journal of Nuclear Materials***, ***Nuclear Engineering and Design*** and ***Nuclear Technology***

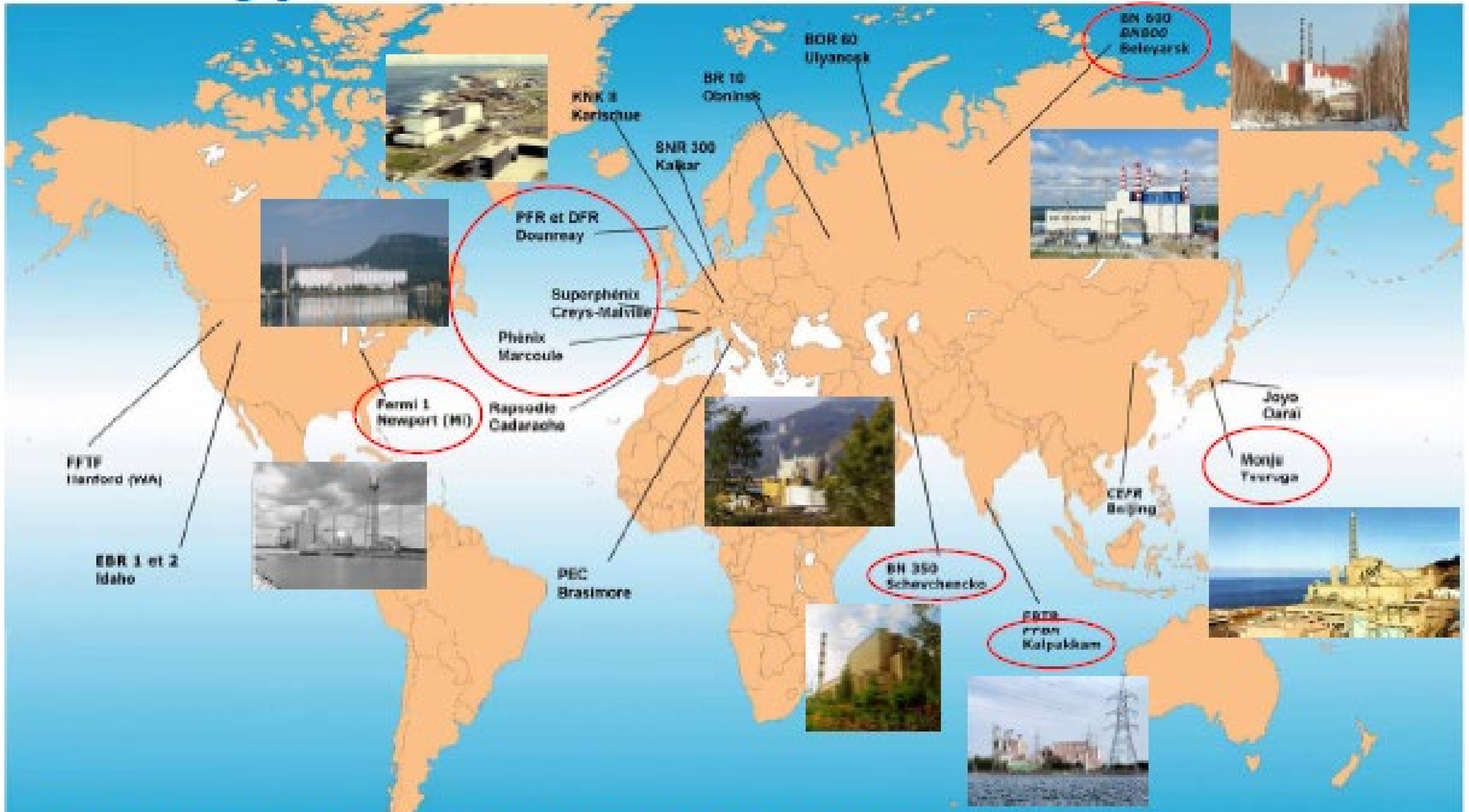
- **Courses/school** : FJOH, ISNE, ENEN

- **Conferences** ; FR09, 13, 17, GLOBAL 1993 →2019, IEMPT, NUMAT, ATALANTE

- **Current International activities on oxide and others fuels for advanced reactors :**
  - OECD: expert group on innovative fuel elements (NSC/WPFC/EGIF)
  - AIEA : CRP fuels and materials for fast reactors
  - GIF : Advanced fuel project management board in the SFR, GFR and VHTR systems
  - EUROPEAN PROJECTS : ESFR-SMART, INSPYRE, PUMMA

- **International data bases:**
  - http://therpro.hanyang.ac.kr/search/search_map.jsp
  - https://www.oecd-nea.org/science/taf-id/taf-id-public/
  - https://www.oecd-nea.org/science/wprs/fuel/ifpelst-request.html

Prototypes & industrial SFRs



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25 March 2021	Introducing new Plant Systems Design (PSD) Code	Dr. Prinja Nawal, Jacobs, UK
22 April 2021	Experience of HTTR licensing for Japan's New Nuclear Regulation	Mr. Etsuo ISHITSUKA, JAEA, Japan

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