GEN IV systems

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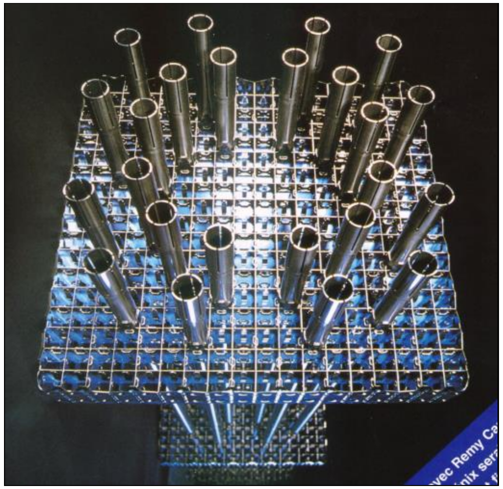
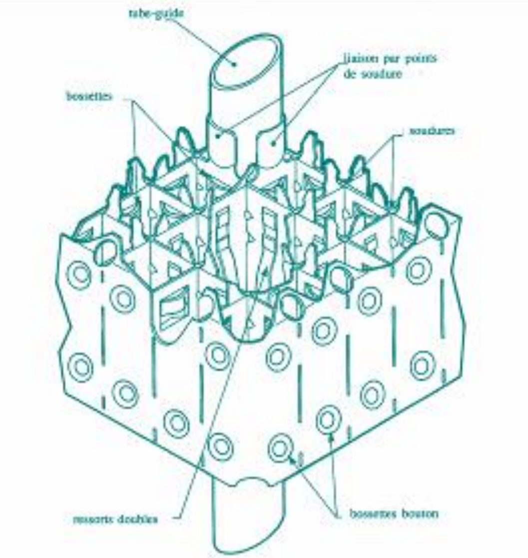
Specific issues of structural materials



Beware of growth

Beware of relaxation of dimples and springs

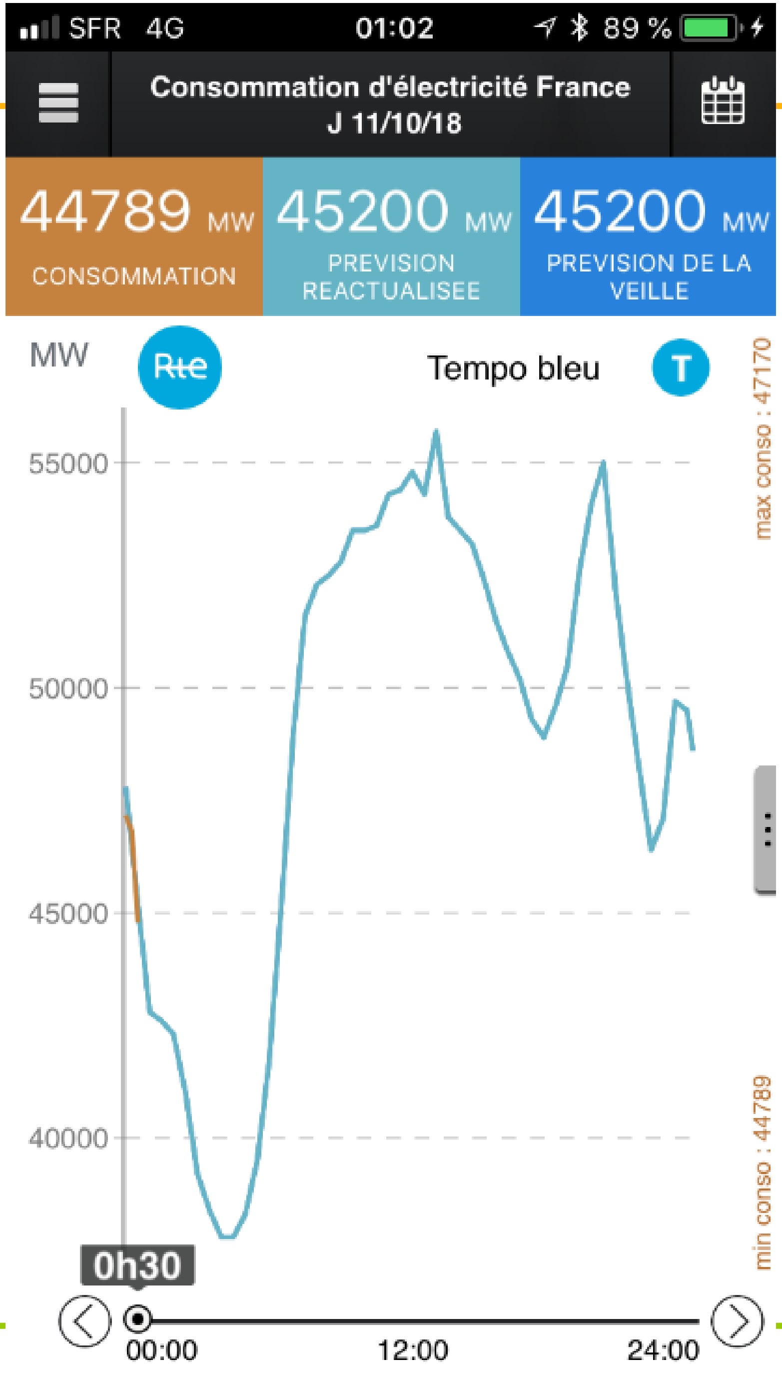


Need for flexibility

Why do we need to have variations of power in the French plants?

What are the impacts on the cladding?

Daily variation



SFR 4G 01:02 89 %

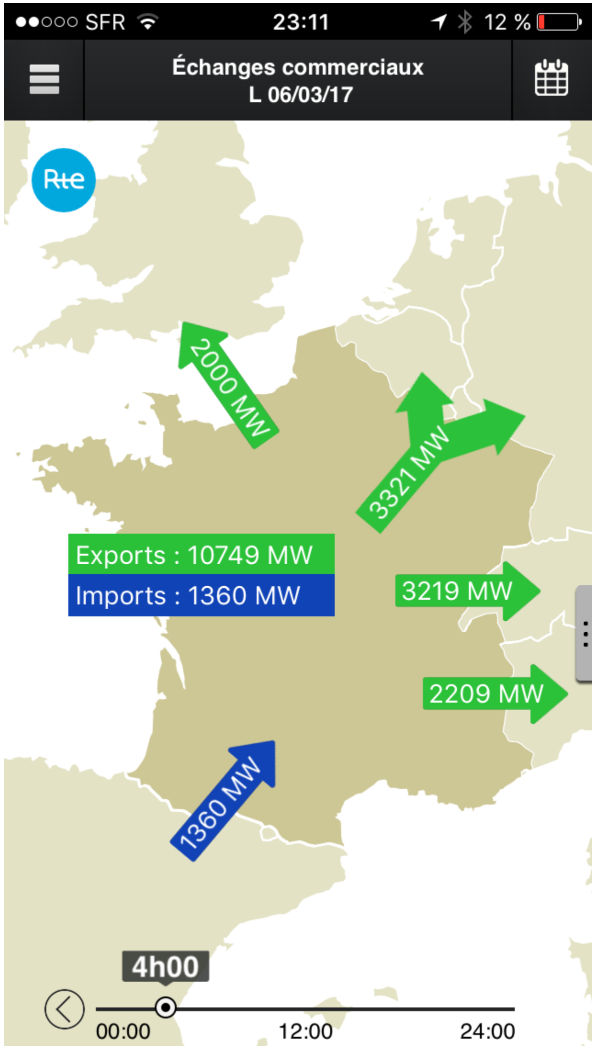
Consommation d'électricité France
J 11/10/18

44789 MW	45200 MW	45200 MW
CONSOMMATION	PREVISION REACTUALISEE	PREVISION DE LA VEILLE

RTE ECO$_2$mix
app

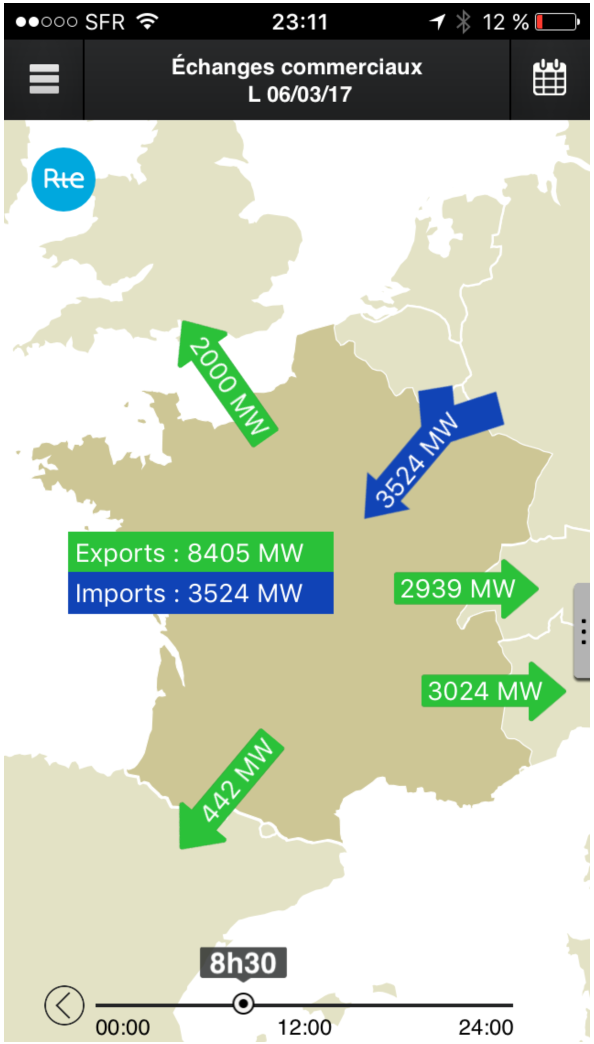
For electricity data look also for the **electricitymap** app

Interconnection



Exports : 10749 MW

Imports : 1360 MW

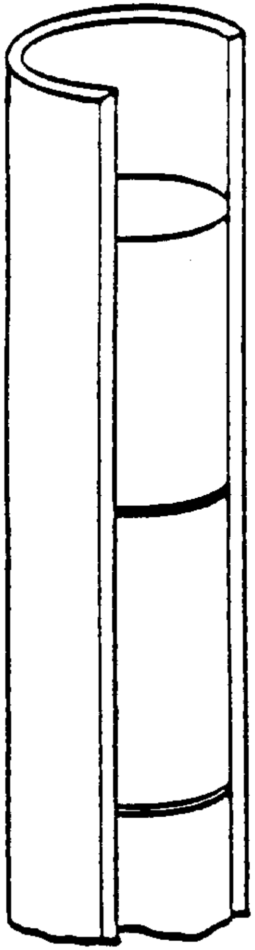


Exports : 8405 MW

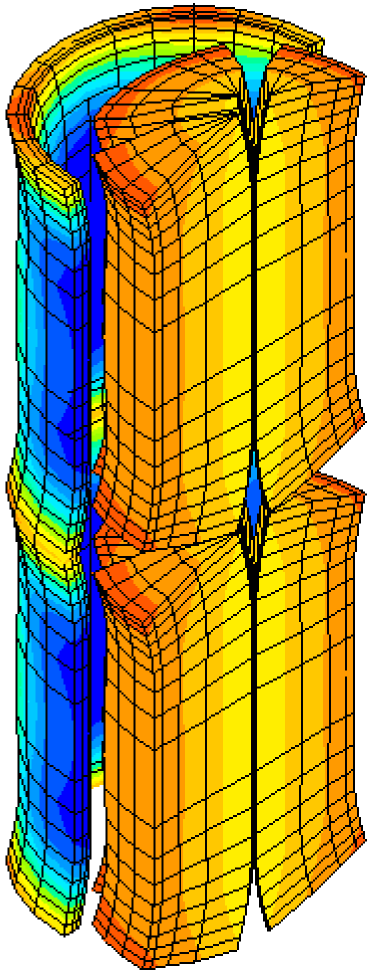
Imports : 3524 MW

PCI: the balance

- P_L ↗ , T_c ↗
- Pellet; expansion and diabolo shape
- ☐ σ_θ inside the cladding and between two pellets
- Release of FP
  - Of which iodine
- SCC inside the cladding
- Potential rupture



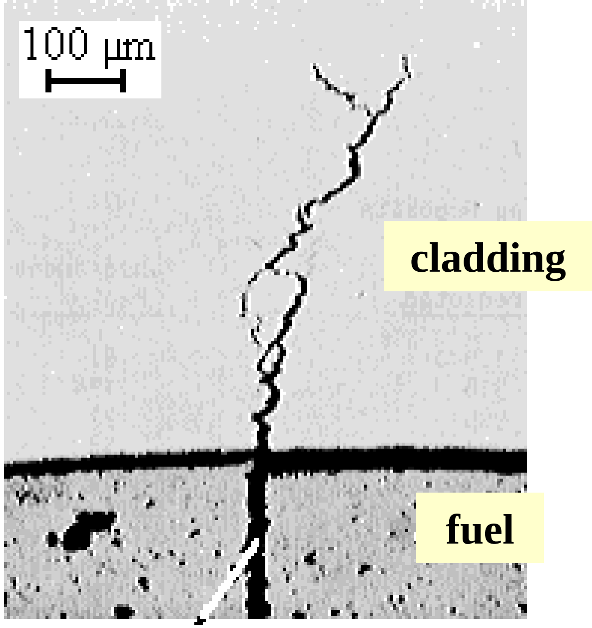
170 W.cm^{-1}



380 W.cm^{-1}

Pellet Cladding Interaction

I induced SCC
after power transient



- Risk of cracks in cladding and rod failure
  - Transient> 420 W/cm

- Cracks appears after some mn at high power
  - Located at inter-pellet
  - In front of pellet cracks

    Where stresses are maximum

    And I escapes and condenses

- PCI risk seems to be maximum at the end of second cycle

Pellet Cladding Interaction

Iodine induced Stress Corrosion Cracking

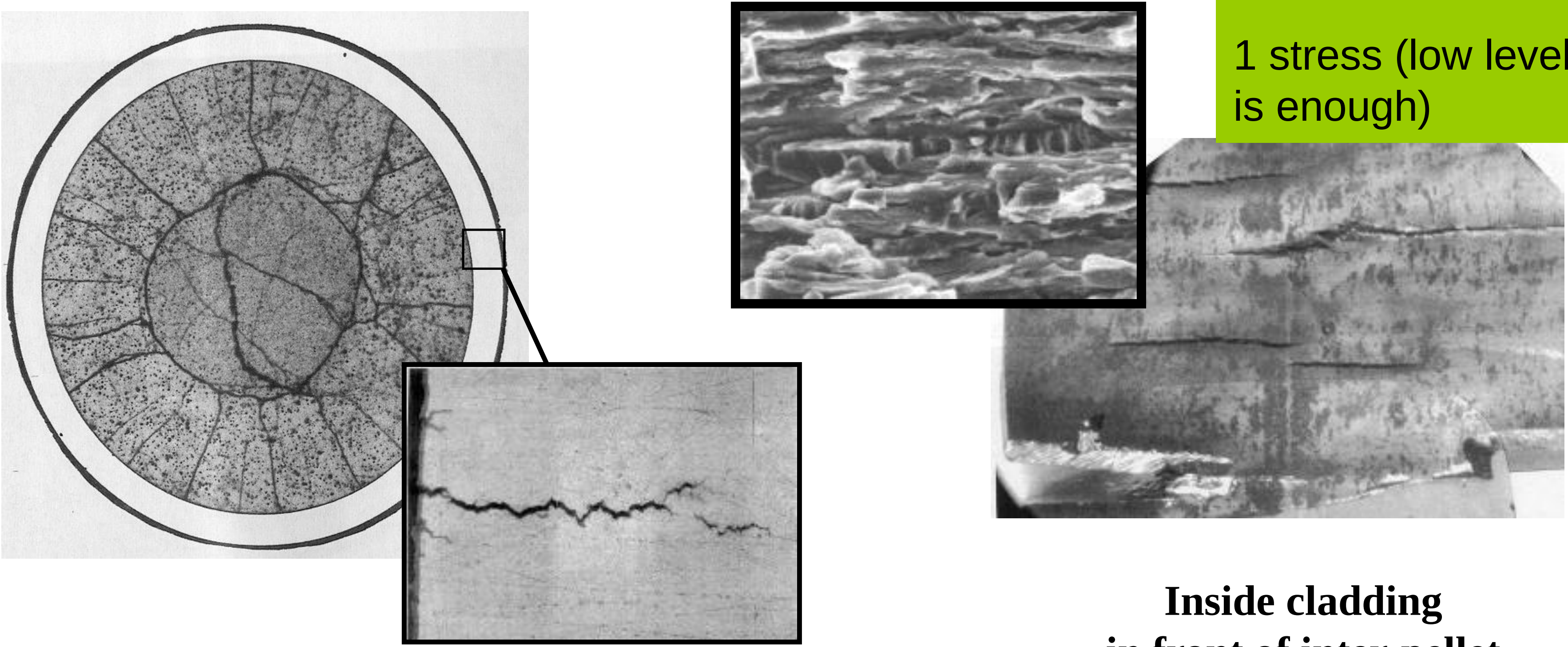
- Intergranular cracking

3 points are needed:

1sensitive material

1 environment

1 stress (low level is enough)

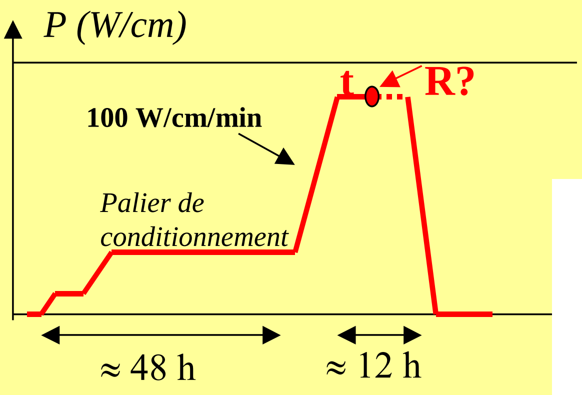


Inside cladding
in front of inter-pellet

Technological limit deduced from experiments

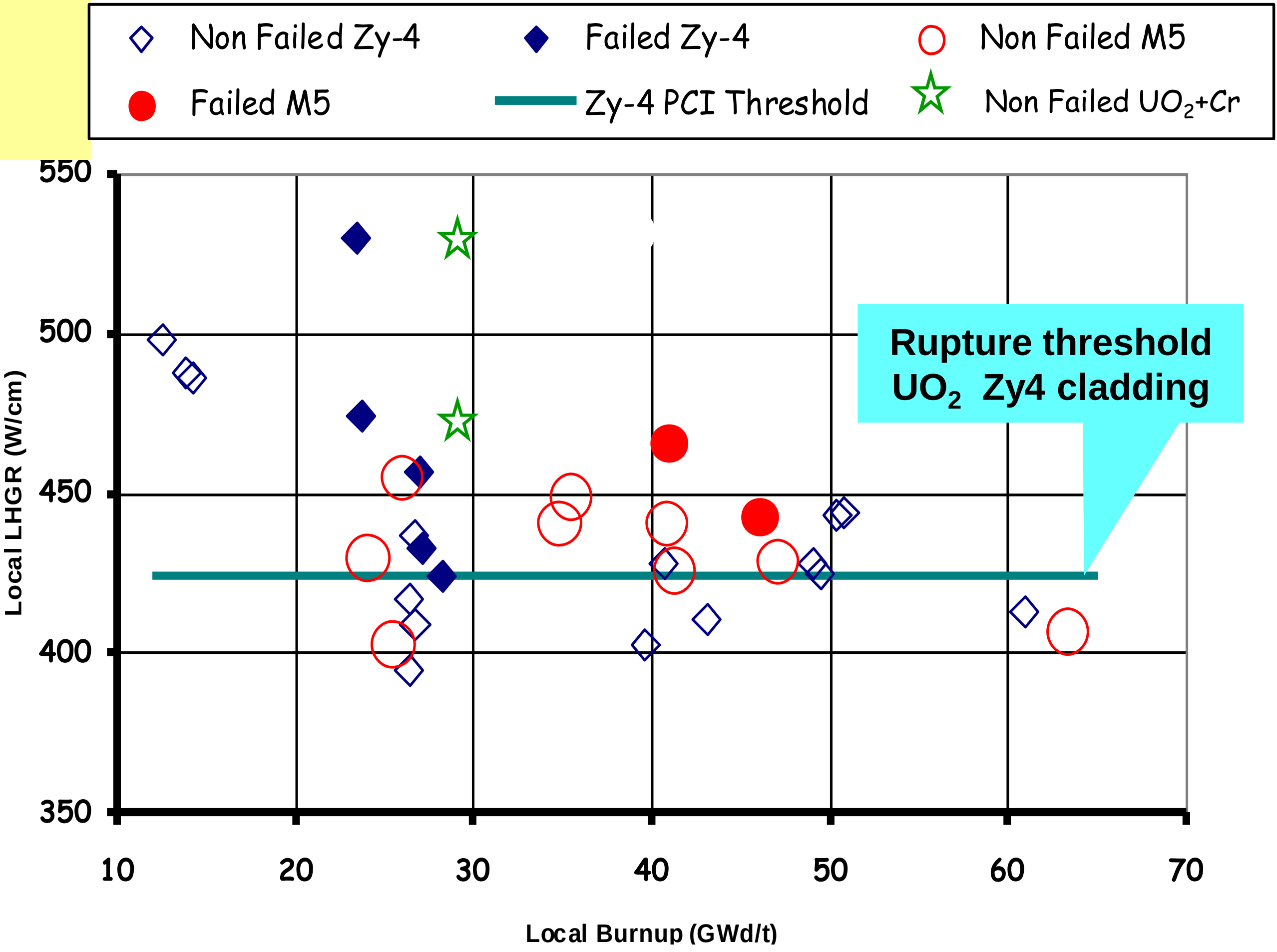
Power ramps

- **Determination of failure criteria by PCI/SCC during class 2 transient**
- **Ramp protocol**

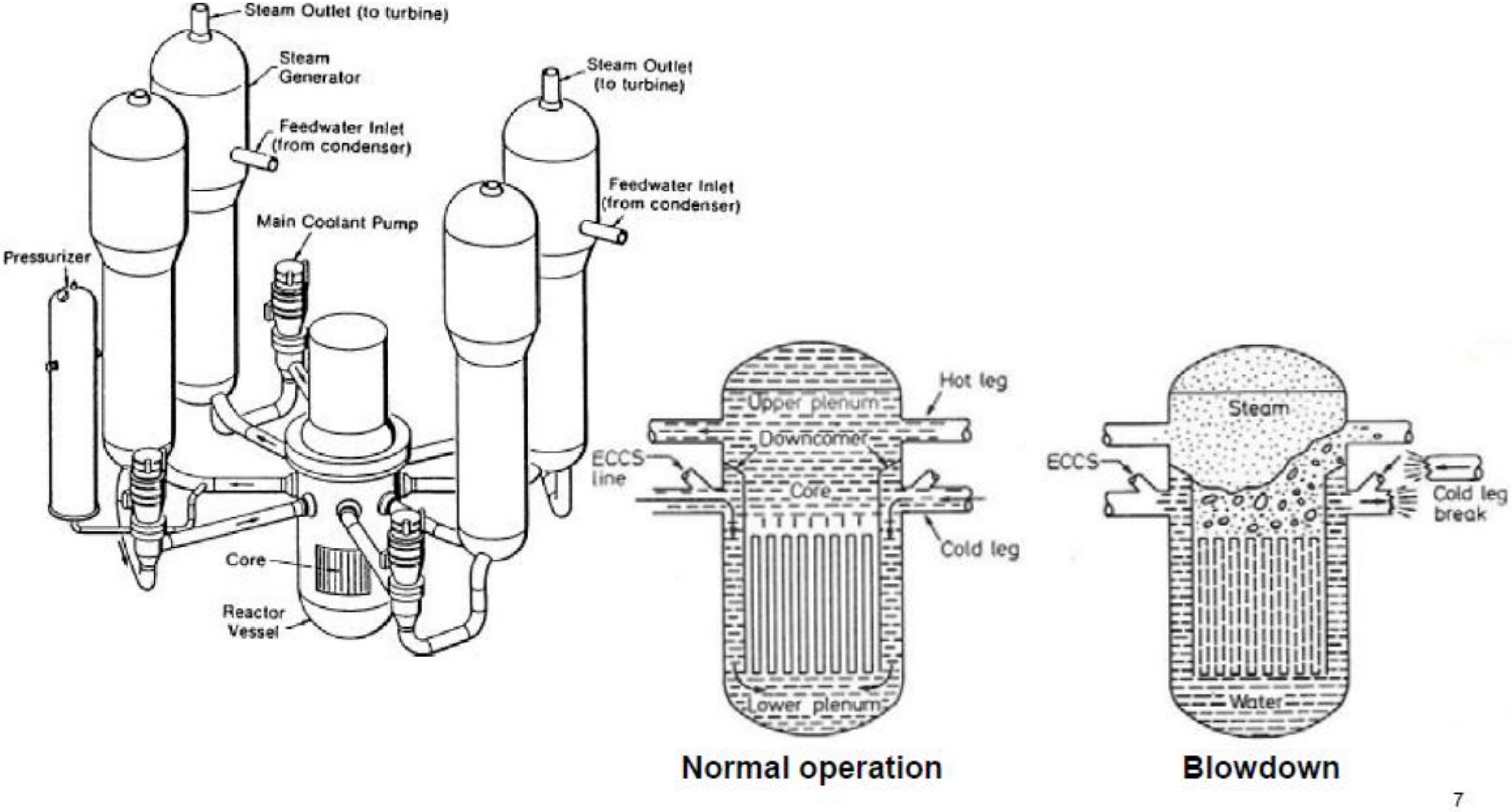


Paramètres de fonctionnement
Plin max, DPlin, vitesse

Paramètres du crayon
Bu, gaine, combustible



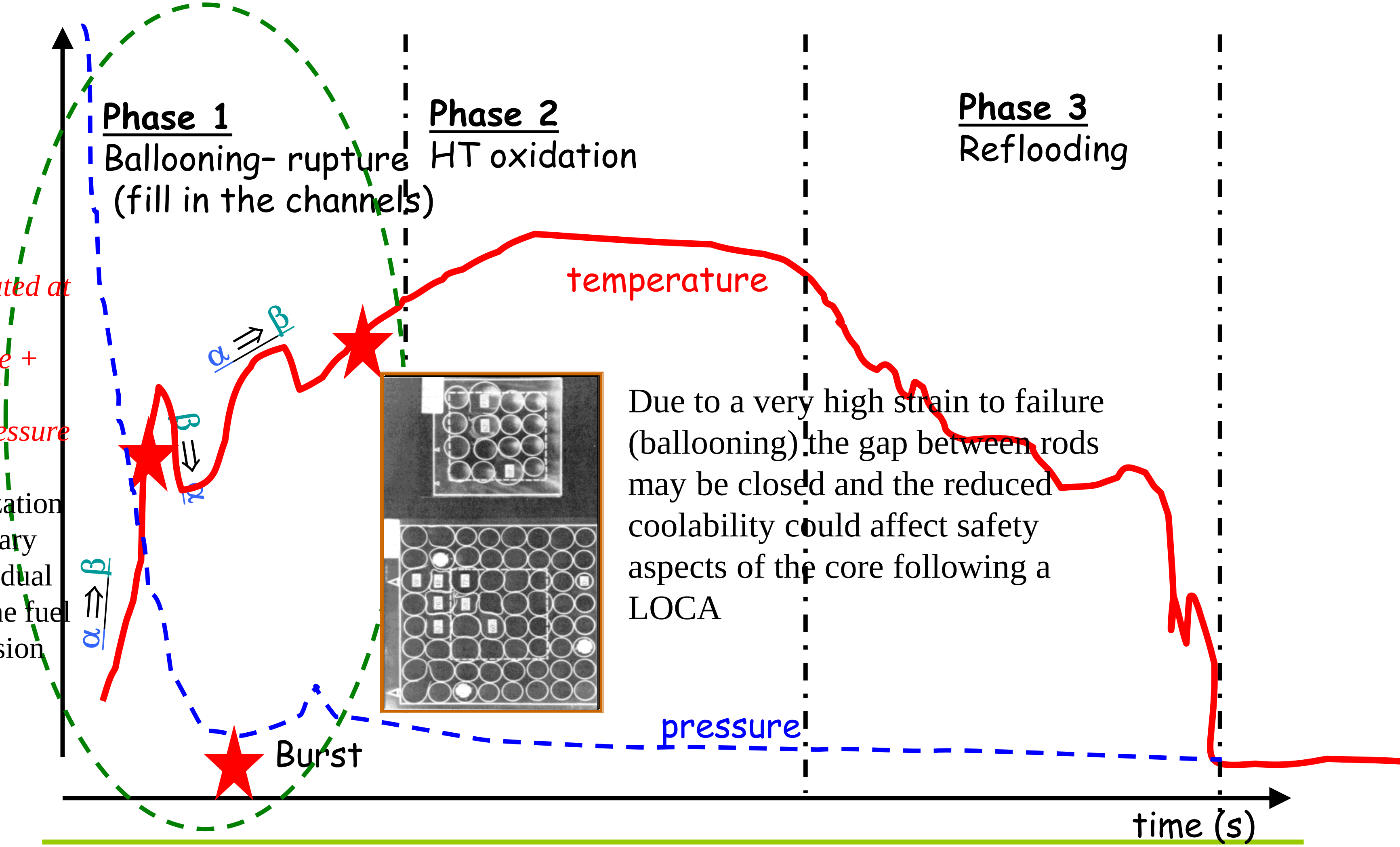
Accidental conditions LOCA



Behaviour for accidental conditions (high temperature)

First phase of LOCA - Cladding ballooning and recoolability

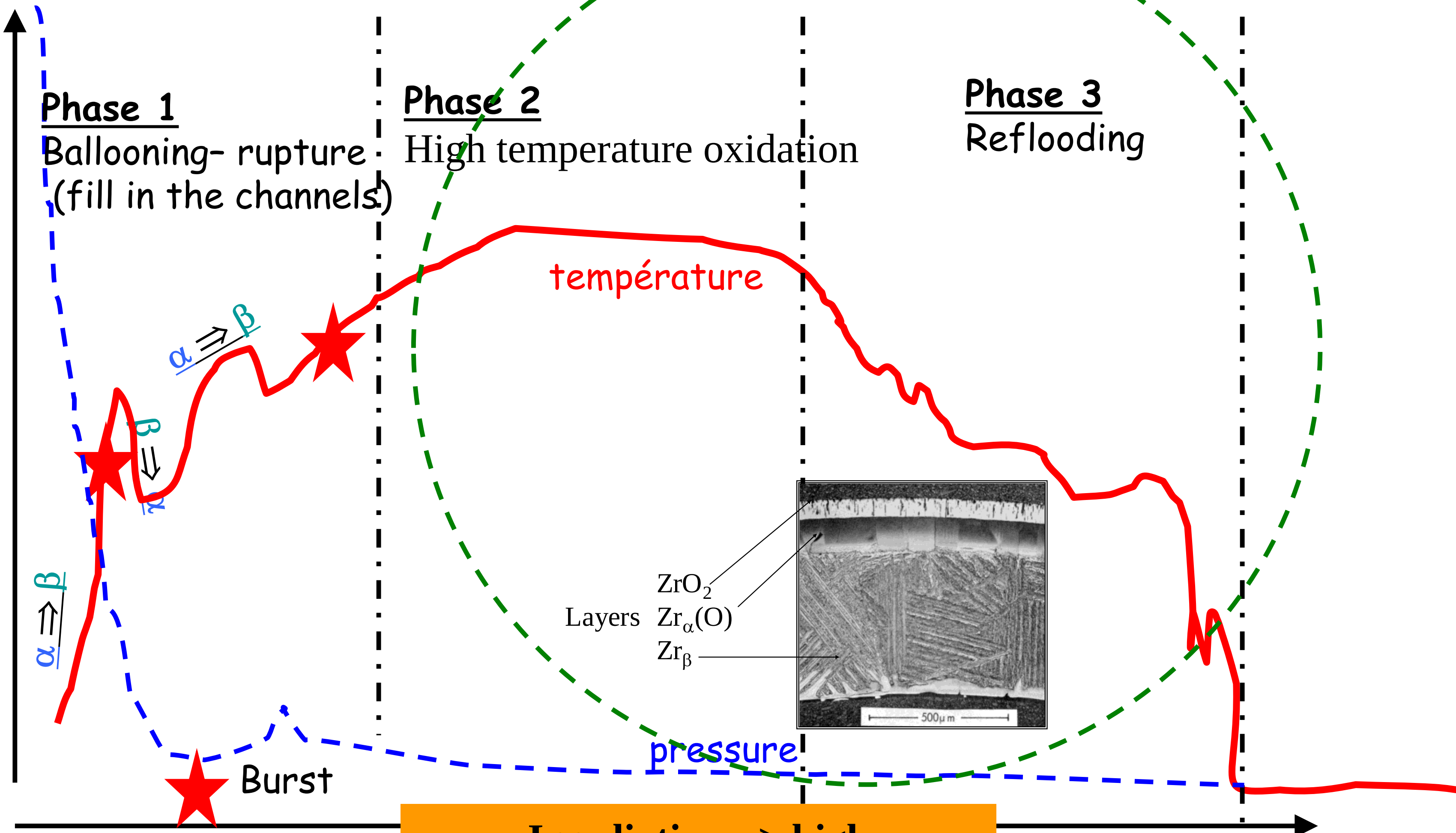
During accidental conditions LOCA, the cladding is *quickly heated at high temperature + increase of internal pressure* due to the depressurization of the primary circuit, residual power of the fuel and gas fission release



Due to a very high strain to failure (ballooning) the gap between rods may be closed and the reduced coolability could affect safety aspects of the core following a LOCA

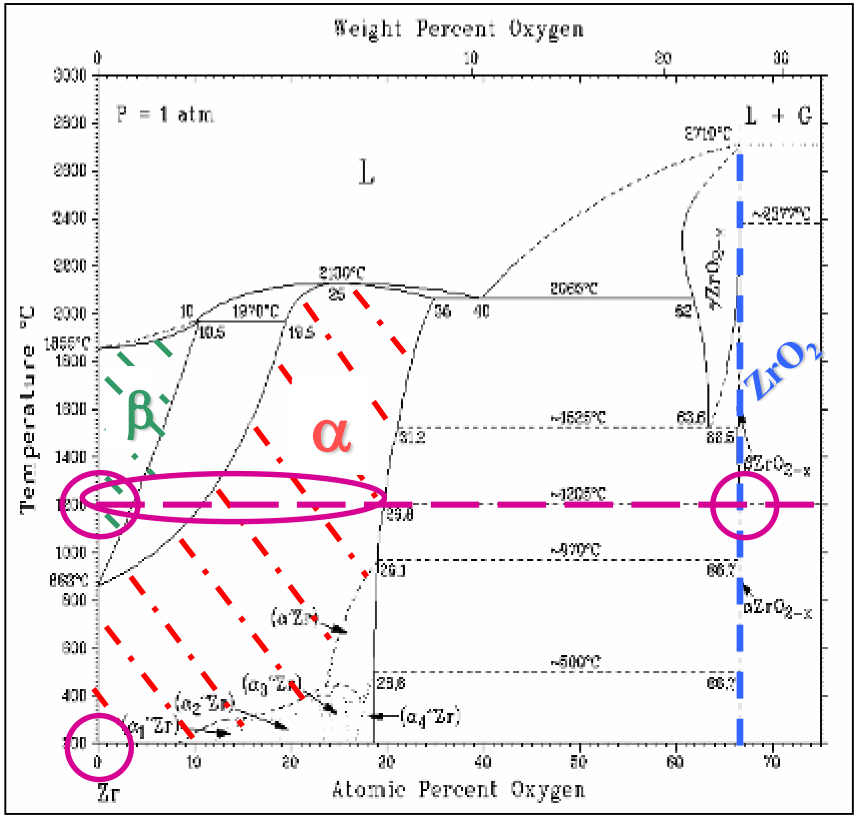
Ballooning test

Second phase of LOCA ***– High temperature oxidation, « quenching» and « post-quenching » behavior : mechanical properties , material handling***

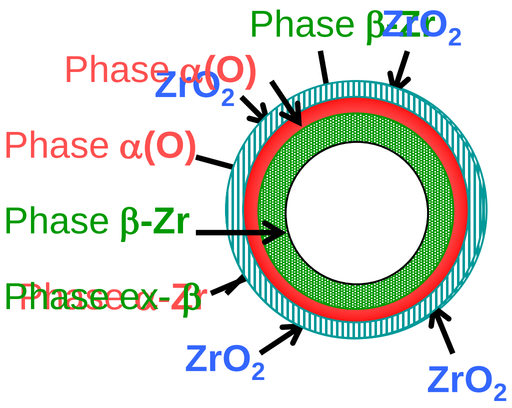


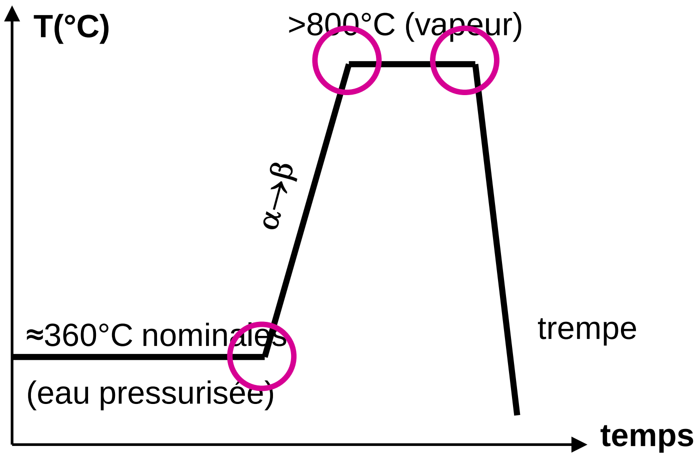
Irradiation => high temperature = annealing of PD

After HT oxydation, complex partition of alloying elements and O between the different phases => **large consequences on residual mechanical properties**



Binary diagram Zr-O

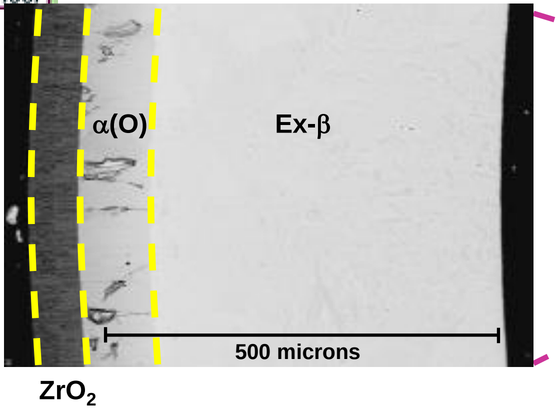


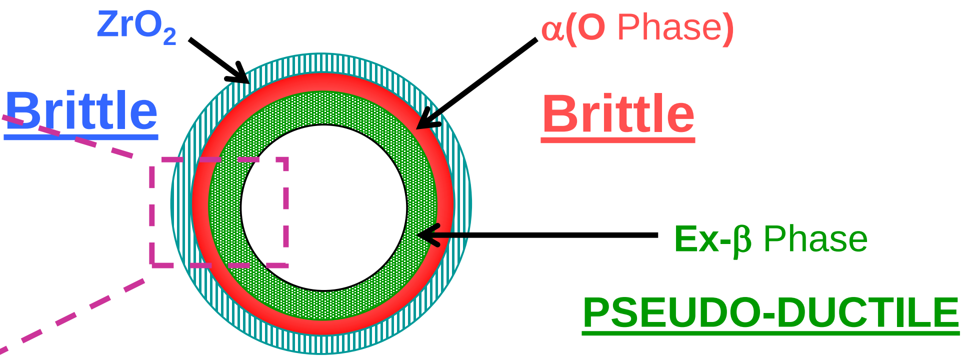


Transitoire de trempe simulant les conditions hypothétiques accidentelles (APRP)

microstructurals and mechanical consequences :

Accelerated oxidation ⇒ oxygen diffusion inside the metal ⇒ brittleness





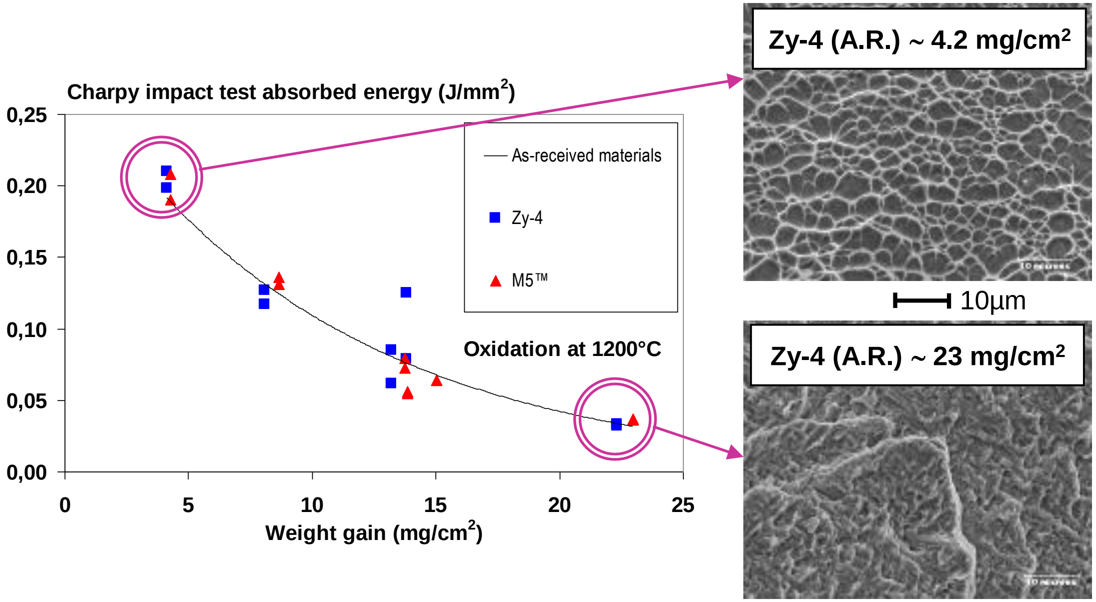
Residual ductility of the cladding depend on :

(1) **Ex-β** phase thickness

(2) C(O) in the Ex-β phase after **quenching**

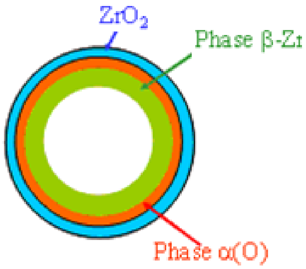
ductile ⇔ brittle transition

à [O] critical ~ 0.4wt.%



post oxidation /quench mechanical tests

← Low oxidation level: ductile material



High oxidation level: fragile material →

LOCA accident

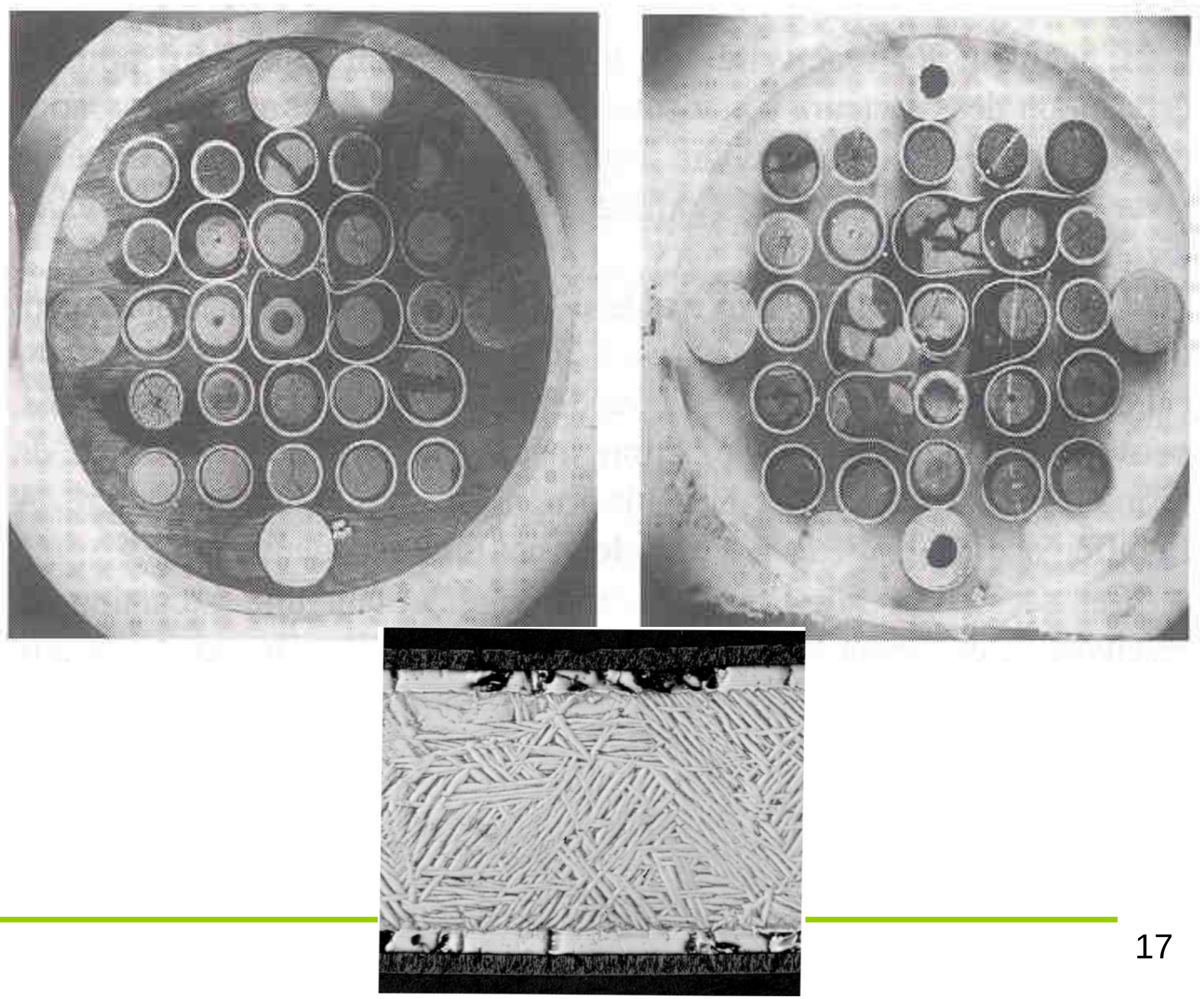
Test in PHEBUS reactor

Current criteria :

- T cladding
PCT < 1204°C

- ECR < 17 %
(Equivalent Cladding Reaction)

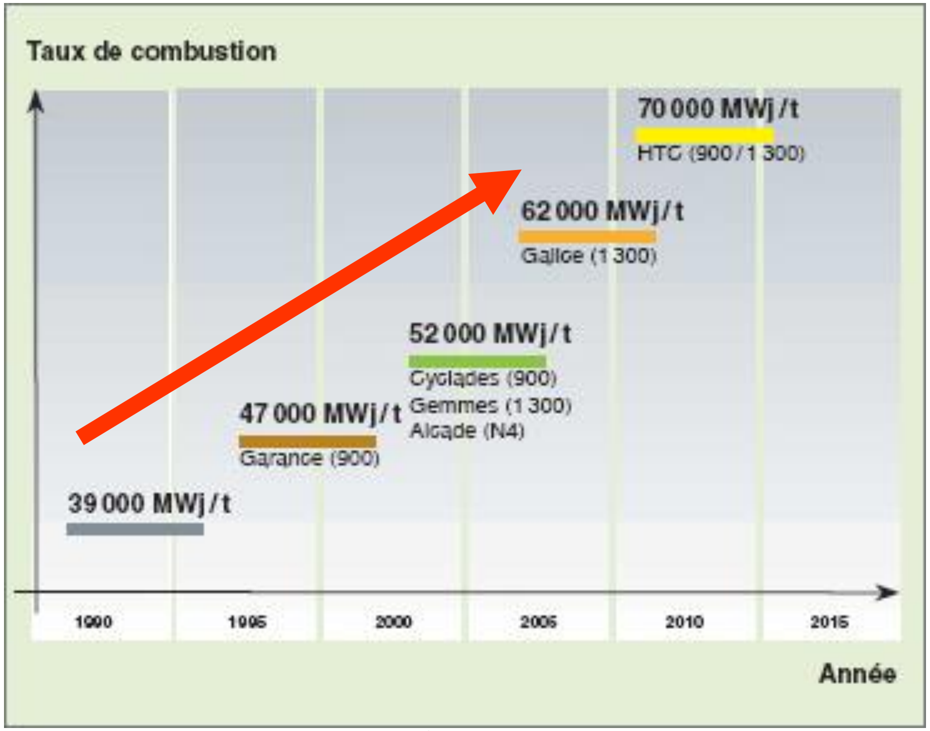
New criteria being discussed



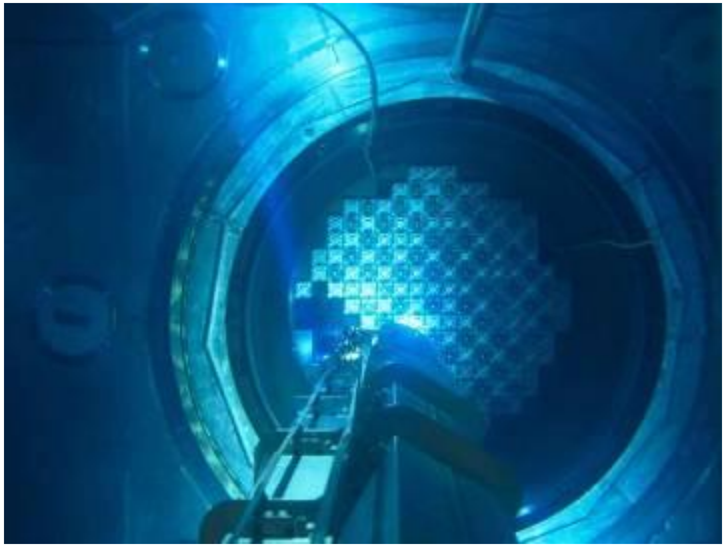
Conclusions

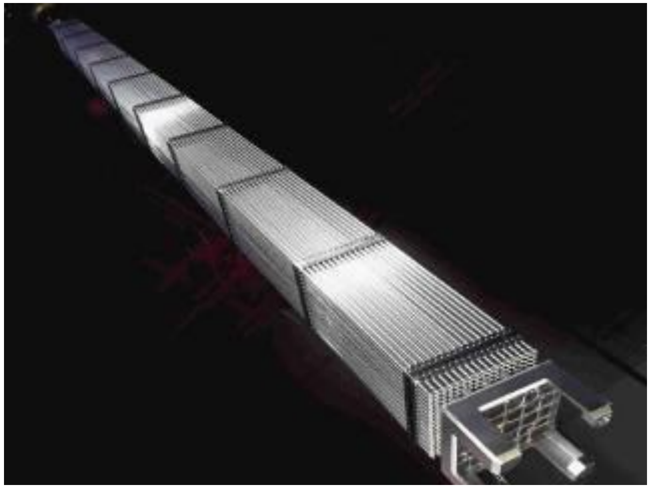
Huge gains in improving the fuel performances

- **Old fuel assemblies: 3 PWR cycles ~30 GWj/t**
- **New fuel assemblies: 5 PWR cycles ~50 GWj/t**



(1 PWR cycle ~ 2.10^{21} n/cm²)





Outline

Motivations of the Generation IV International Forum

Presentation of the systems and their respective challenges

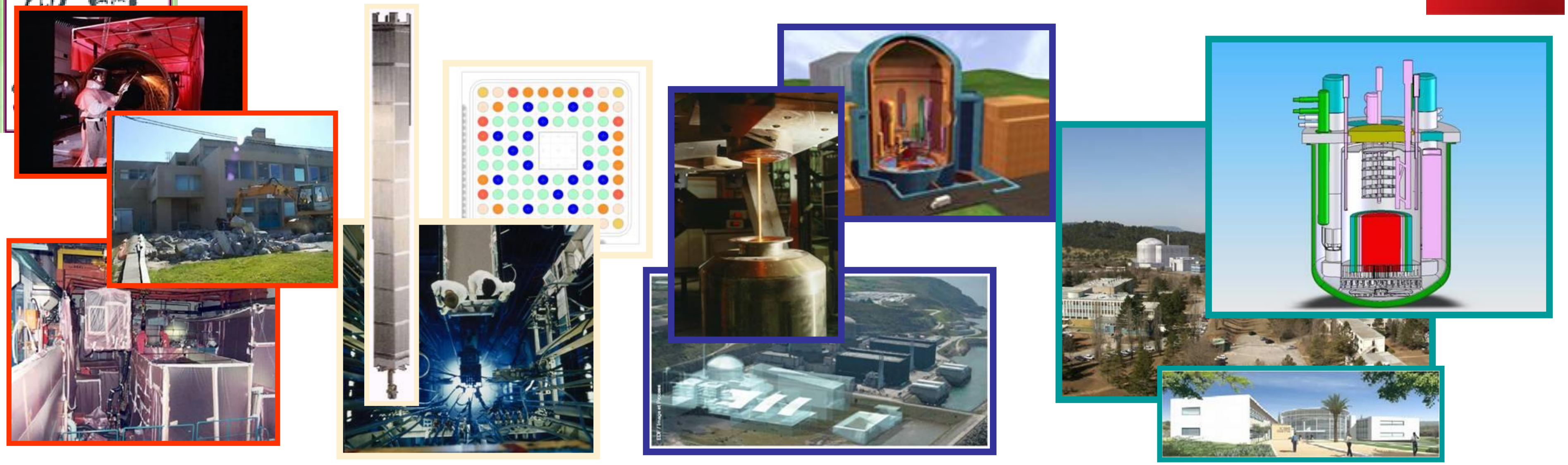
The French strategy

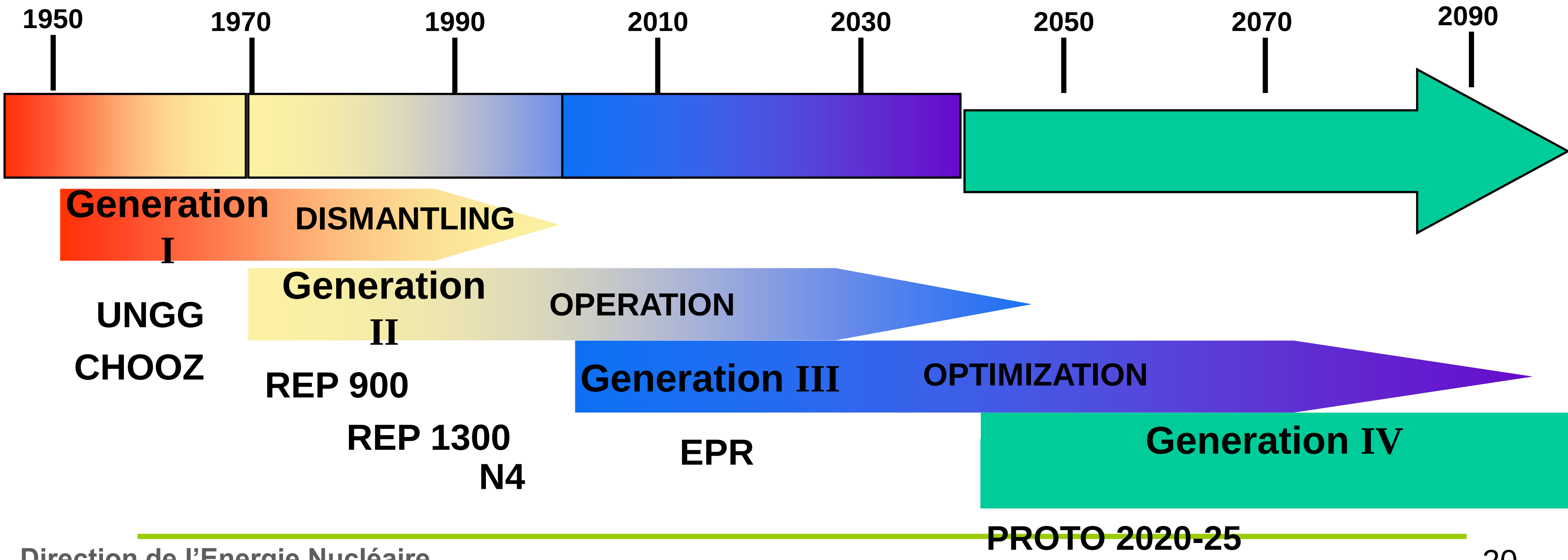
2 examples of development

SFR cladding materials

GFR cladding materials

Generations of Nuclear Power Systems





1950
1970
1990
2010
2030
2050
2070
2090

Generation I DISMANTLING

UNGG
CHOOZ

Generation II OPERATION

REP 900
REP 1300
N4

Generation III OPTIMIZATION

EPR

Generation IV

PROTO 2020-25

Generation IV Nuclear Systems

New goals for sustainable nuclear energy

Continuous progress:

- ✓ Economically competitive
- ✓ Safe and reliable

Break-throughs:

- ✓Natural resources conservation
- ✓Waste minimisation
- ✓Proliferation resistance

Systems marketable from 2040 onwards

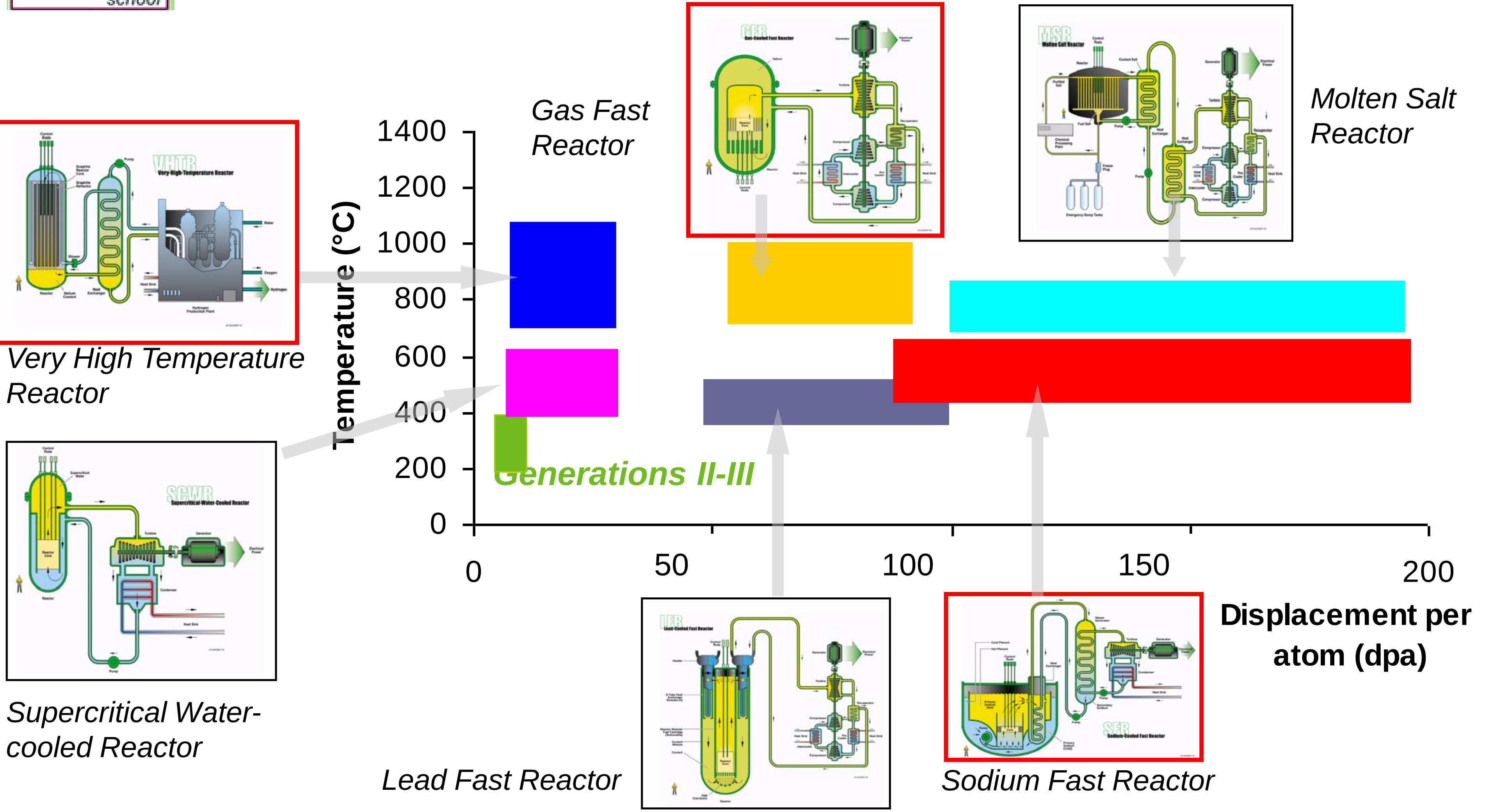
A closed fuel cycle

True potential for new Applications: *Hydrogen, Syn-fuel, Desalinated water, Process heat*

Internationally shared R&D

Generation IV Forum: selection of six nuclear systems

☞ **New challenges** for materials !



Gas Fast Reactor

Molten Salt Reactor

Very High Temperature Reactor

Supercritical Water-cooled Reactor

Lead Fast Reactor

Sodium Fast Reactor

Material requirements in future nuclear systems

Technical challenges & Leading physical phenomena

- ***60-year lifetime***

- ***Fast neutron damage*** *(fuel and core materials)*
  - → *Effect of irradiation on microstructure, phase instability, precipitation*
  - → *Swelling growth, hardening, embrittlement*
  - → *Effect on tensile properties (yield strength, UTS, elongation…)*
  - → *Irradiation creep and creep rupture properties*
  - → *Hydrogen and helium embrittlement*

- ***High temperature resistance*** *(SFR > 550°C, V/HTR > 850-950°C)*
  - → *Effect on tensile properties (yield strength, UTS, elongation…)*
  - → *High temperature embrittlement*
  - → *Effect on creep rupture properties*
  - → *Creep fatigue interaction*
  - → *Fracture toughness*

Incidental and accidental

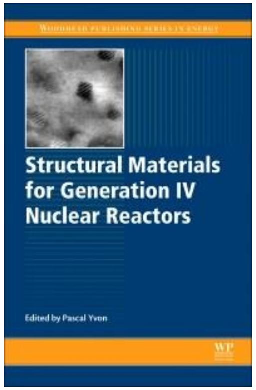
- ***Corrosion resistance*** *(primary coolant, power conversion, H_2 production)*
  - → *Corrosion and stress-corrosion cracking (IGSCC, IASCC, hydrogen cracking & chemical compatibility…)*

Additional requirements

- ***Material availability and cost***
- ***Fabricability, joining technology***
- ***In service inspection***
  - → *Non destructive examination techniques*
- ***Safety approach and licensing***
  - → ***Codes and design methods***
  - → ***R&D effort needed to establish or complement mechanical design rules and standards***
- ***Decommissioning and waste management***

Advanced irradiation resistant materials

(See Zinkle chapter in Gen IV book)



Structural Materials for Generation IV Nuclear Reactors

Edited by Pascal Yvon

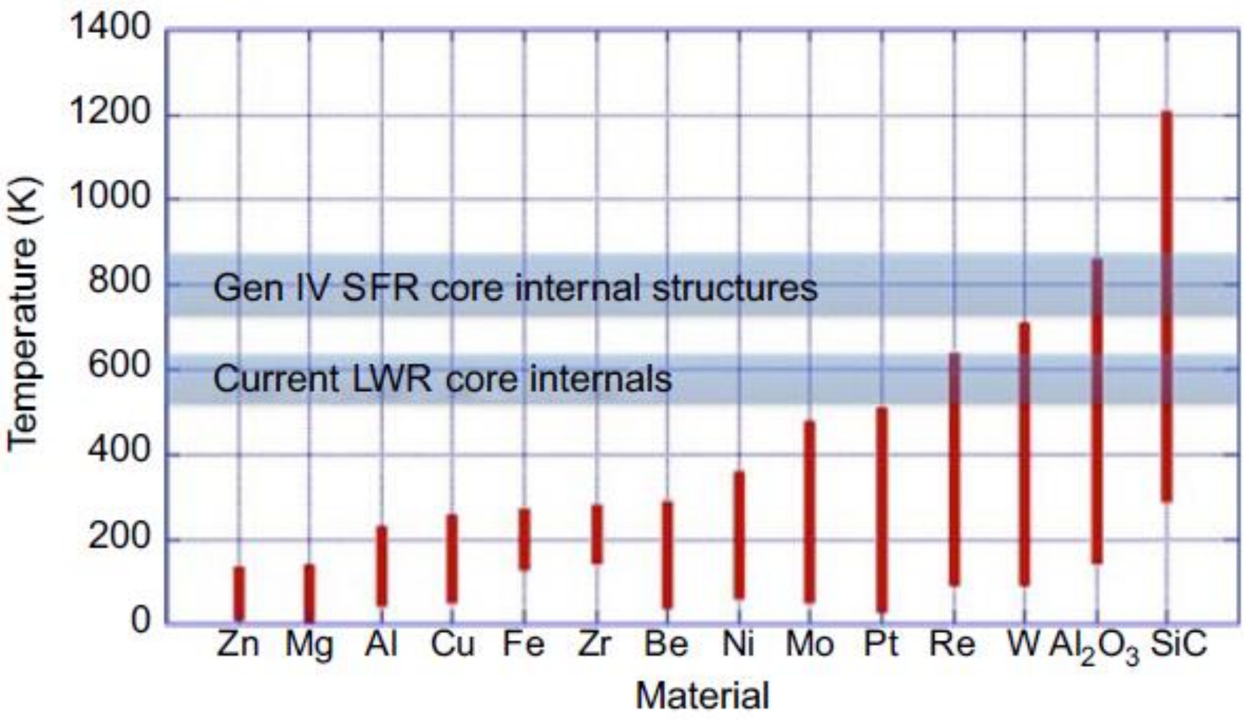
How to design such materials?

- *High point defects sink strength*
  - ***These defect recombination sinks can be dense dislocation arrays, finely dispersed precipitates, nanoscale grain dimensions, or nanoscale multilayer interfaces. Introduction of high concentrations of precipitates or nanoscale interfaces (grain boundaries or multilayer interfaces).***

Advanced irradiation resistant materials

- *Low vacancy mobility*

Select temperatures where the interstitial is mobile but the vacancy is immobile. Under these conditions, the immobile vacancies can serve as built-in interstitial recombination centers produced as a by-product of neutron irradiation

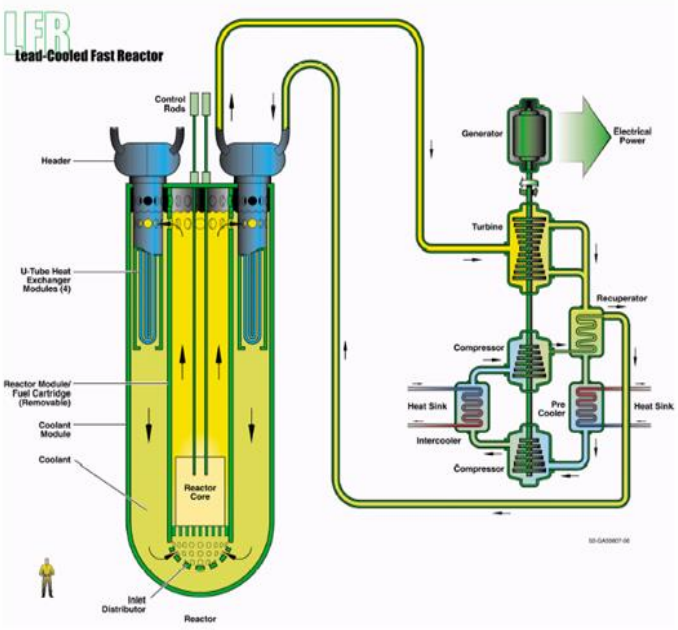


Advanced irradiation resistant materials

- *Radiation resistant matrix phase*
  - ***A third general method to design radiation tolerance is to select material compositions or phases that have intrinsically low radiation defect accumulation. Utilization of body centered cubic (BCC) phase materials such as ferritic/martensitic steels (vs. austenitic steels) or vanadium alloys is the most widely studied example of this approach. Although the primary defect production rate (per unit of displacement damage) for BCC metals is comparable to that for face centered cubic (FCC) or hexagonal close packed (HCP) metals the spatial distribution and defect clustering characteristics within individual energetic displacement cascades facilitates more efficient defect recombination processes during subsequent cascade evolution.One also use bulk metallic galsses or high entropy alloys***

Lead Fast Reactor *(LFR)*

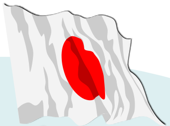
- **An alternative Liquid Metal cooled Fast Reactor:**
  - ***→ thermal management of lead***
  - ***→ in service inspection and repair***
- **Weight of primary system** *(seismic behaviour…)*
- **Prevention of corrosion of 1^{ry} system structures**
- **600 MWe – T_{He} ~ 480 °C**
- **Potential for integral recycling of Actin**



High irradiation doses on cladding
Corrosion

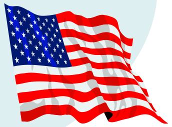
ELSY
EUROTRANS
in EU FP6


Euratom countries


Japan


South Korea

LFR Steering Committee


U.S.A.

✓***System Arrangement LFR to be signed***

Supercritical Water Cooled Reactor *(SCWR)*

- ***Open cycle & thermal / closed cycle & fast spectrum***
- ***High pressure, High temperature (>22.1 Mpa, 374 °C)***

→ ***Highly ranked in economics (thermal efficiency, plant simplification)***

→ ***Electricity production (and others)***

EAC
Corrosion

HPLWR in EU FP6

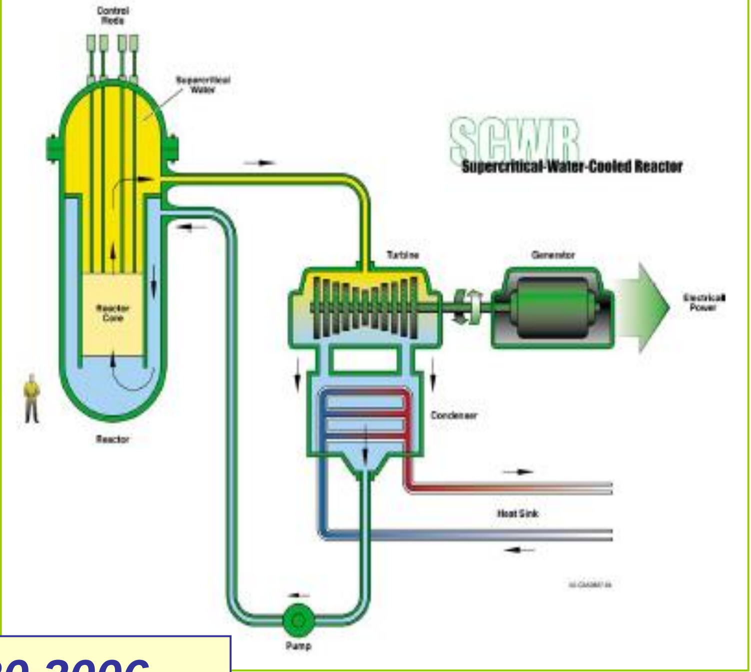
Euratom countries

Canada

Japan

South Korea

SCWR Steering Committee

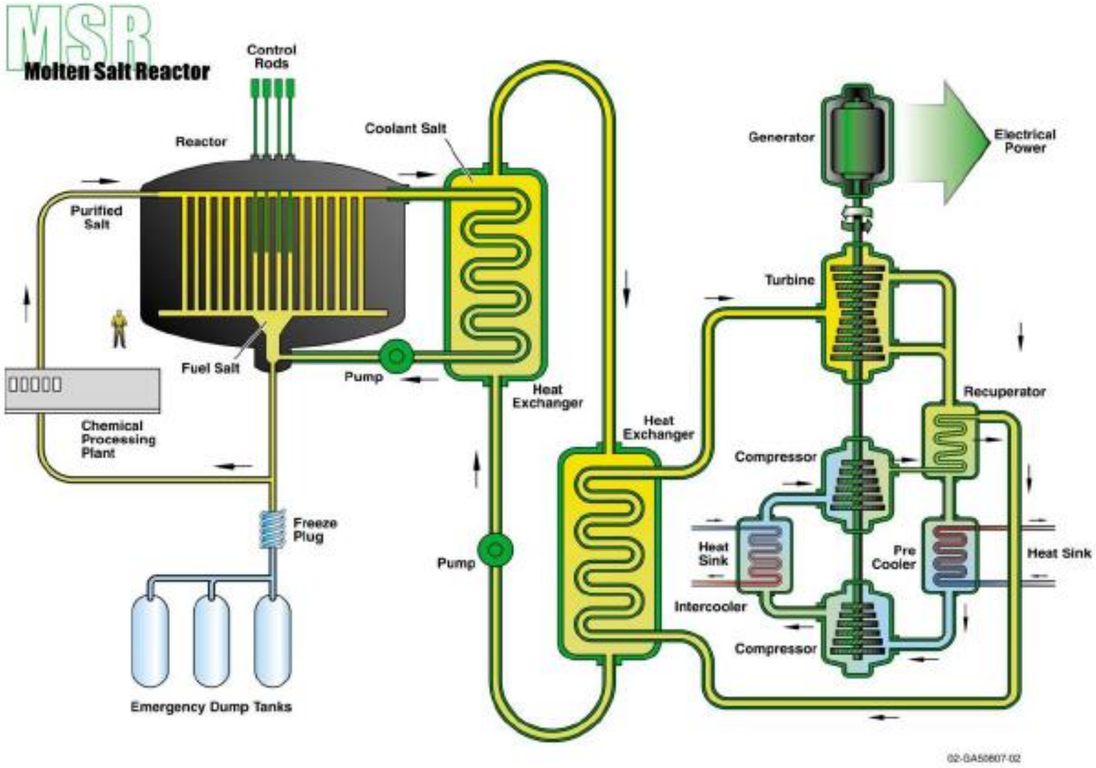


✓***System Arrangement SCWR signed Nov. 30,2006***

Molten Salt Reactor *(MSR)*

Characteristics

- Fuel is liquid fluorides of U and Pu with Li, Be, Na and other fluorides
- 700–800C outlet temperature
- 1000 MWe
- Low pressure (<0.5 MPa)



Benefits

- Waste minimization
- Avoids fuel development
- Proliferation resistance through low fissile material inventory

Corrosion,
Fuel,
reprocessing

Very High Temperature Reactor *(V/HTR)*

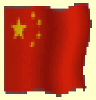
- **A nuclear system dedicated to the production of high temperature process heat for the industry and hydrogen**
- **600 MWth - T_{He} >1000 °C**
  Thermal neutrons
  Block or pebble core concept
- **Passive safety features**
- **I-S Cycle or HT Electrolysis for H_2**

IHX material depending on T and secondary

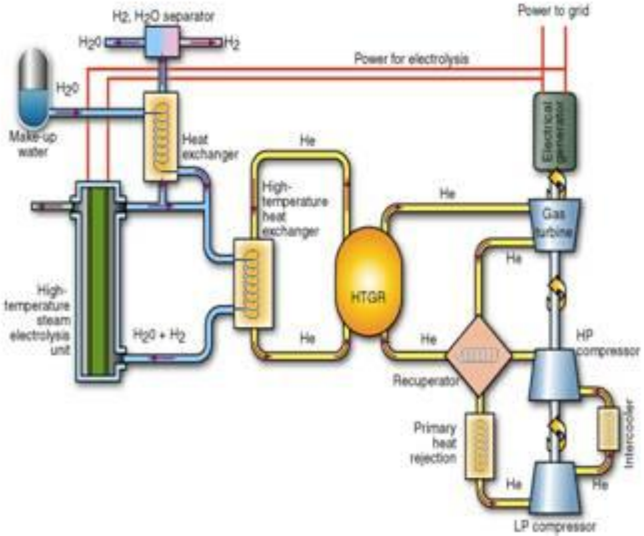
2008

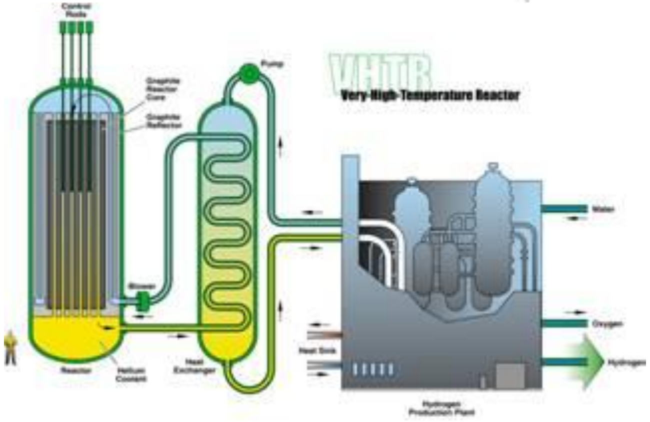
South Africa

+



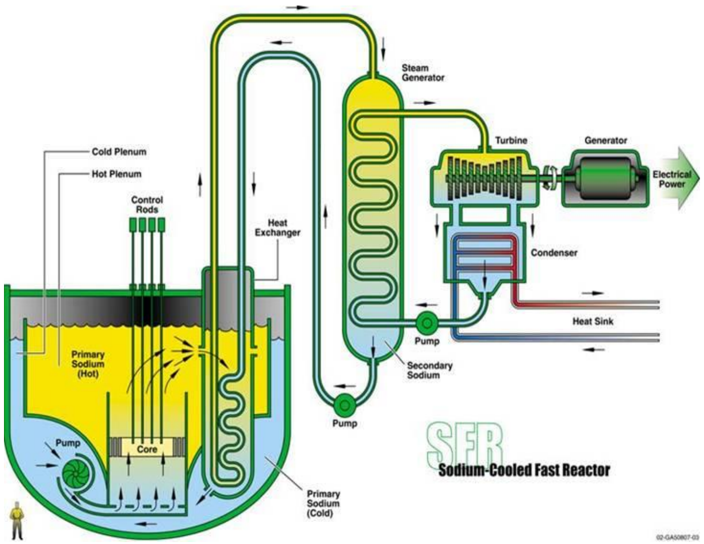
China





Sodium Fast Reactor *(SFR)*

- **A new generation of sodium cooled Fast Reactors**
- **Reduced investment cost**
  Simplified design, system innovations
  (Pool/Loop design, ISIR – SC CO_2 PCS)
- **Towards more passive safety features**
  + Better managt of severe accidents
- **Integral recycling of actinides?**
  → Remote fabrication of TRU fuel

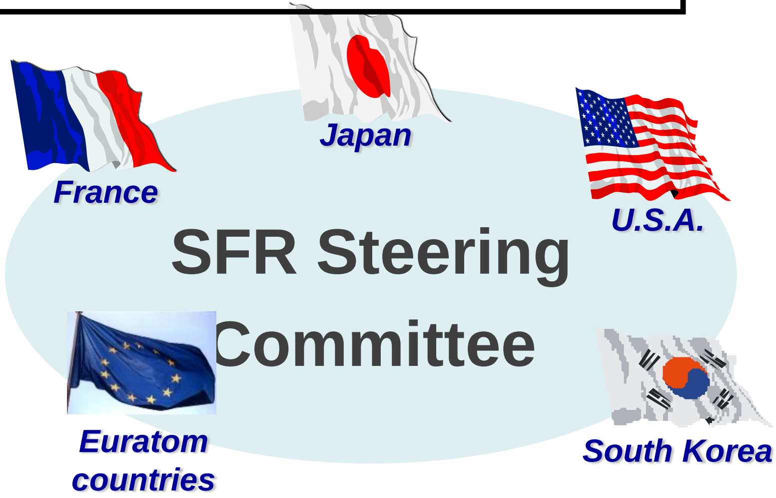


High irradiation doses on cladding and wrapper tubes
ECS fabrication

2008
+

Russia

China



France

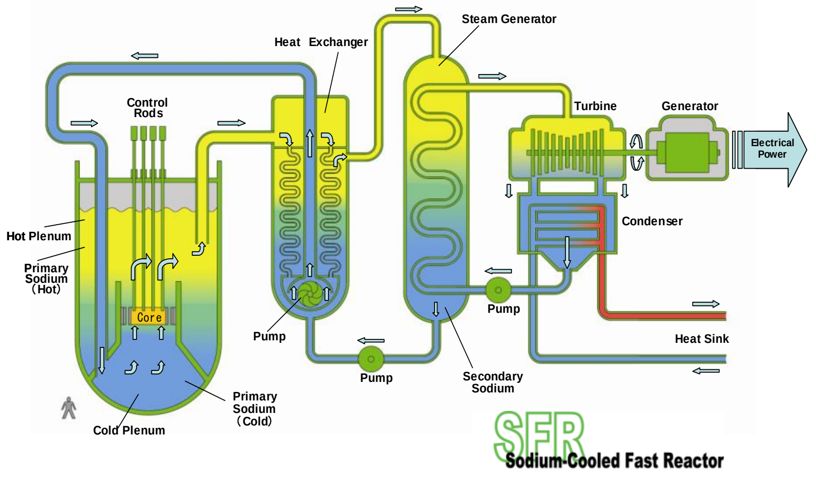
Japan

U.S.A.

SFR Steering Committee

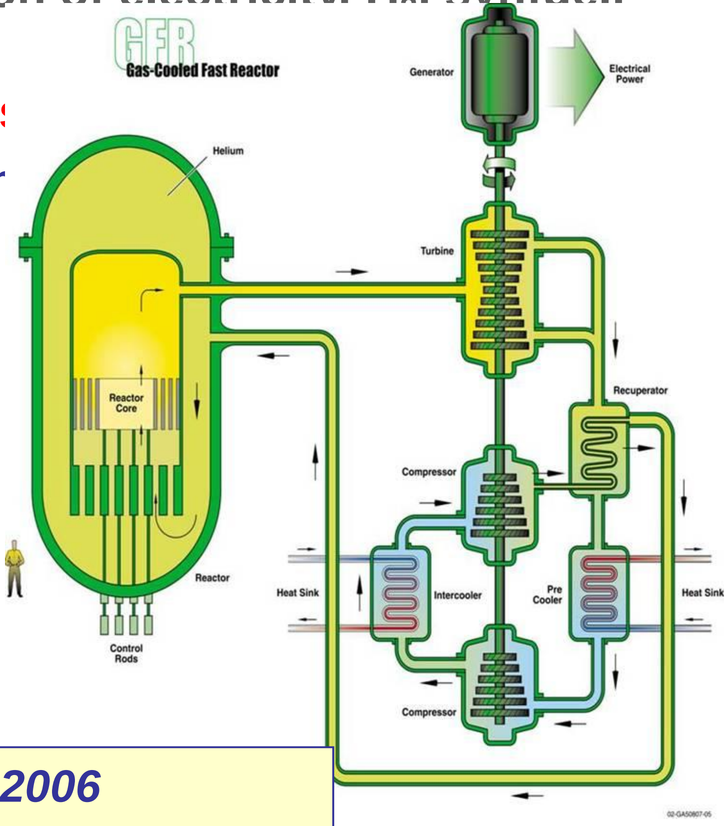
Euratom countries

South Korea



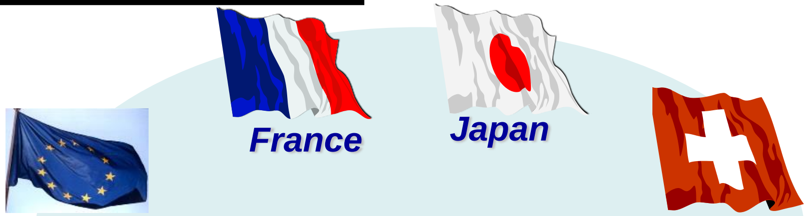
Gas Fast Reactor *(GFR)*

- **A novel type of Gas-cooled Fast Reactor:**
  - *→ an alternative to the Sodium Fast Reactor, and*
  - *→ a sustainable version of the VHTR*
- **Robust heat resisting fuel *(<1600°C)***
- **1200 MWe – T_{He} ~ 850 °C - Cogeneration of electricity, H_2, synfuel, process heat**
- **Safe management of cooling accident**
- **Potential for integral recycling of Acti**



High temperatures in accidental conditions

GCFR 5-6 EU PCRD



Euratom countries

France

Japan

Switzerland

GFR Steering Committee

✓System Arrangement GFR signed Nov. 30 Nov.,2006

✓Project Arrangements "Fuel " & "Design-Safety-Integration" in 2009

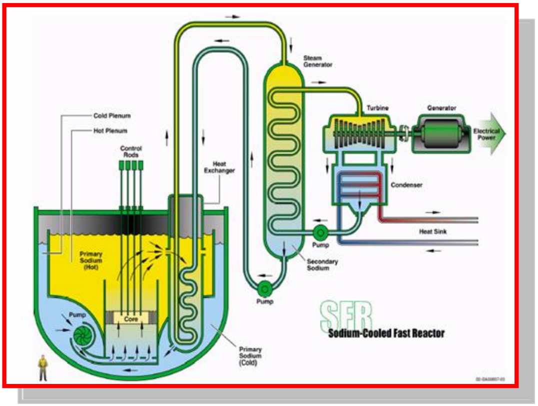
U.S.A.

French strategy *(until last Atomic Energy Committee)*

Decided by Atomic Energy Committees

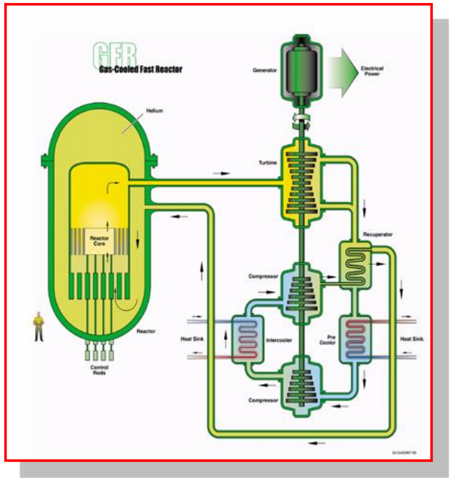
Development of fast reactors with a closed cycle

- ➢*Sodium Fast Reactor (SFR)*
- ➢ *Gas Fast Reactor (GFR)*
- ➢ *New processes for recycling of used fuel*



SFR

GFR



The reference is the SFR : *ASTRID is the prototype*

More mature option

In collaboration with French industrials EDF and AREVA

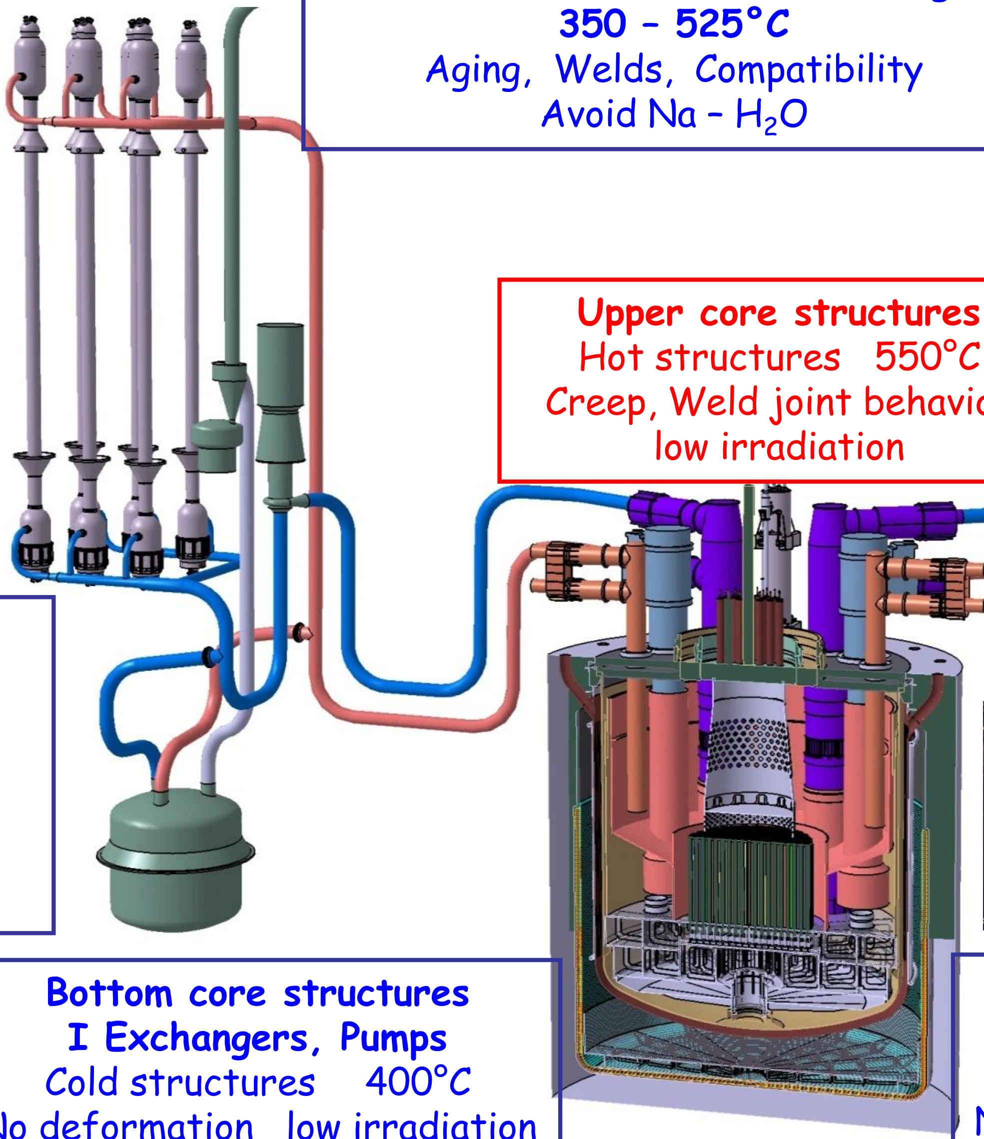
Alternative and long term option : the GFR : *ALLEGRO is the first experimental GFR (V4G4)*

Why a fast neutron reactor ?

- **Full recycling of fuel**
- **Preservation of the uranium resource**
- **Acceptation of nuclear in the public opinion →** **Separation/transmutation** of minor actinides in fast reactors- law from June 28th 2006 on Sustainable management of radioactive waste
- **A large scale nuclear project and a major asset to maintain the skills**

→ Development of reactors and associated fuel cycle

SFR – Main components



Steam Generators, Heat Exchangers
350 – 525°C
Aging, Welds, Compatibility
Avoid Na – H_2O

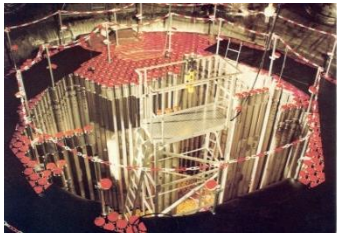
5 10^5 h

Life time
to design
30 → **60 ans**

1000 MWe
Pool type
Modular SG
AREVA design

Upper core structures
Hot structures 550°C
Creep, Weld joint behavior
low irradiation

Core
Sub-assemblies
400 – 650°C
Irradiation



Circuits - Pipes
350 – 550°C
Creep, fatigue,
creep-fatigue,
thermal fatigue,…
Aging
Welds

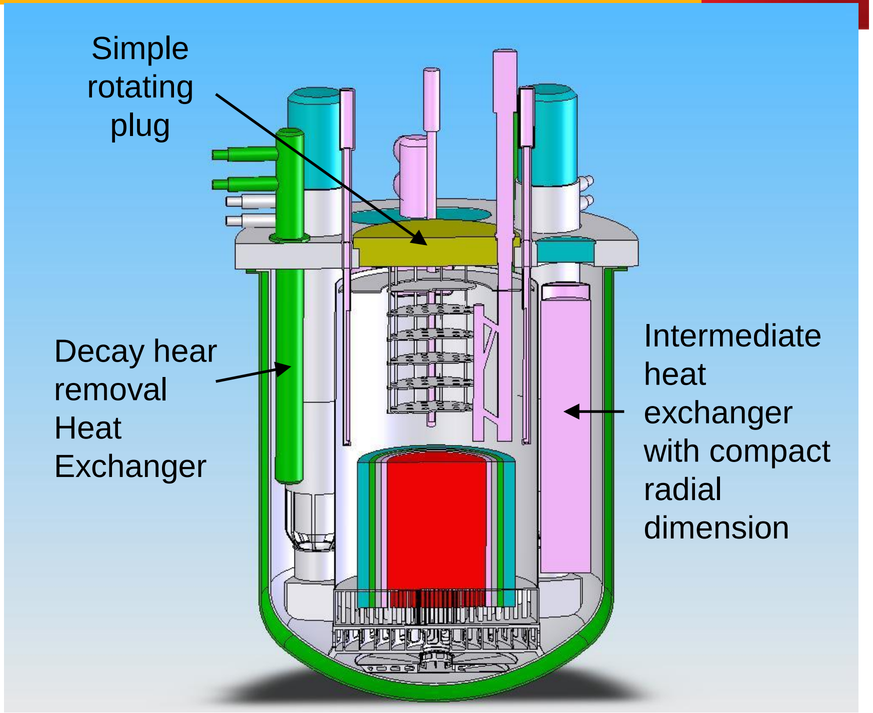
Bottom core structures
I Exchangers, Pumps
Cold structures 400°C
No deformation low irradiation

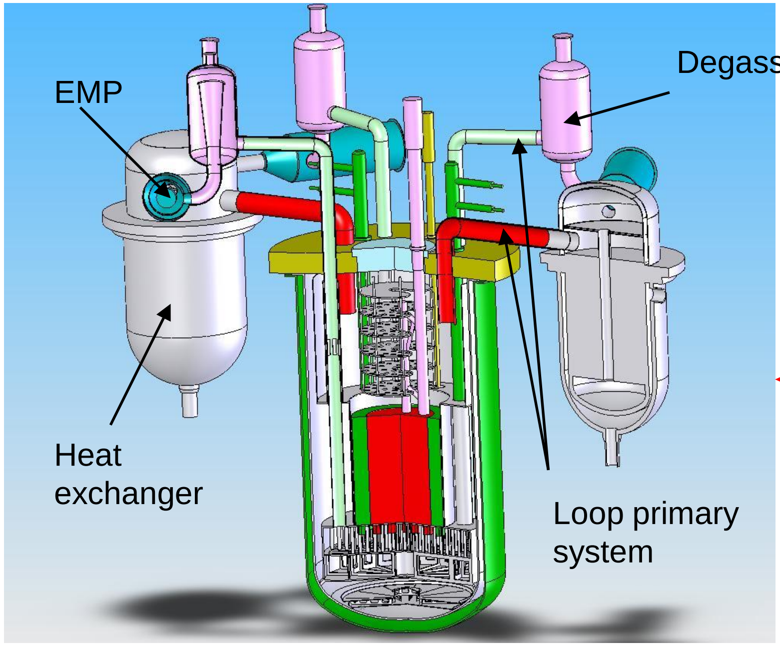
Vessel
400°C
No deformation
Negligeable creep

Sodium Fast Reactor *(SFR)*

Large pool type

1500 MWe optimized size



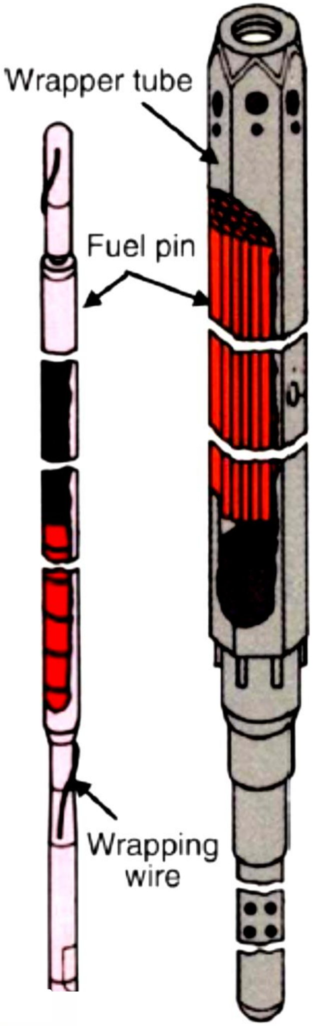


Modular concept with gas conversion system

SFR Fuel assembly

Cladding main functions (400-650°C)

- 1st containment barrier for MOX fuel
  - To contain the fuel
  - To contain the fission products
- To transfer heat
- Wrapping wire = to ensure separation between pins and to allow the Na coolant flow



→ Design of radiation-stable components:
to limit change in microstructure, in dimensions and in functional properties

SFR Fuel assembly : operating conditions

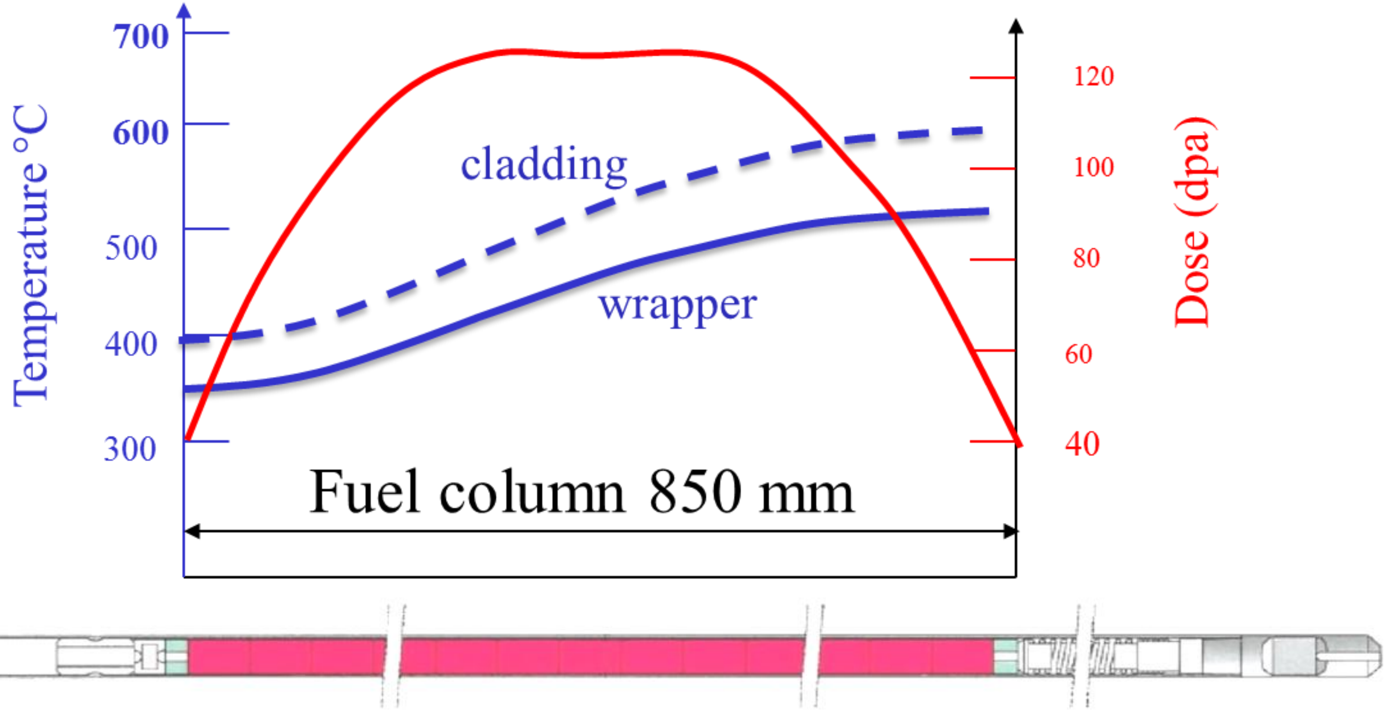
Cladding

- Temperature : 400 – 650°C
- Hoop Stress < 100 MPa (fission gas)

Wrapper

- Temperature : 350 – 550°C
- Small level of stress

Irradiation maximum Dose : ~120 dpa (~3 years)



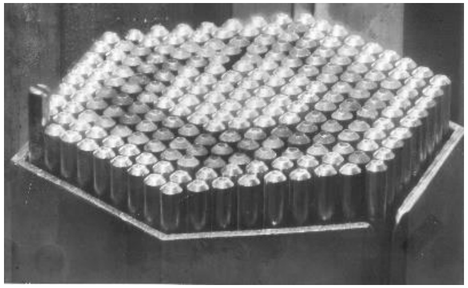
Fuel assembly : material requirements

- Transparency to fast neutrons
- Dimensional stability → minimize deformation
- Maintain good mechanical properties (strength, ductility, and fracture toughness) → no cladding break
- Physicochemical compatibility with the flowing sodium and the U-PuO_2 fuel (+ fission products)
- Weldability (cladding –plug for instance)
- Reprocessing issues (resistance to dissolution in nitric acid)
- Industrial fabrication at reasonable cost

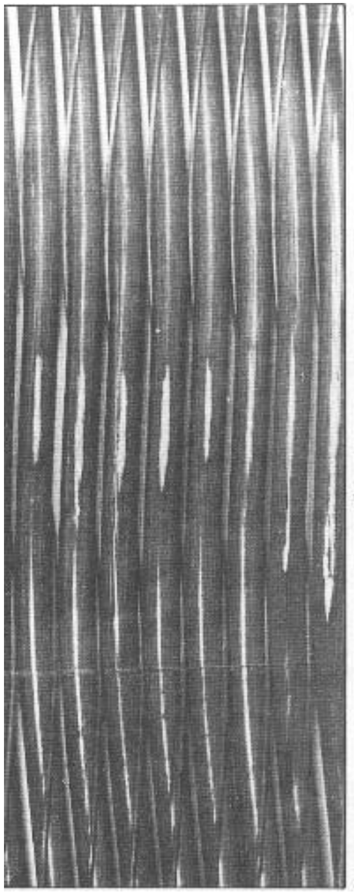
Creep resistance
strain < 1%
650°C – 100 MPa

SFR first cladding material 316

Swelling of SFR cladding

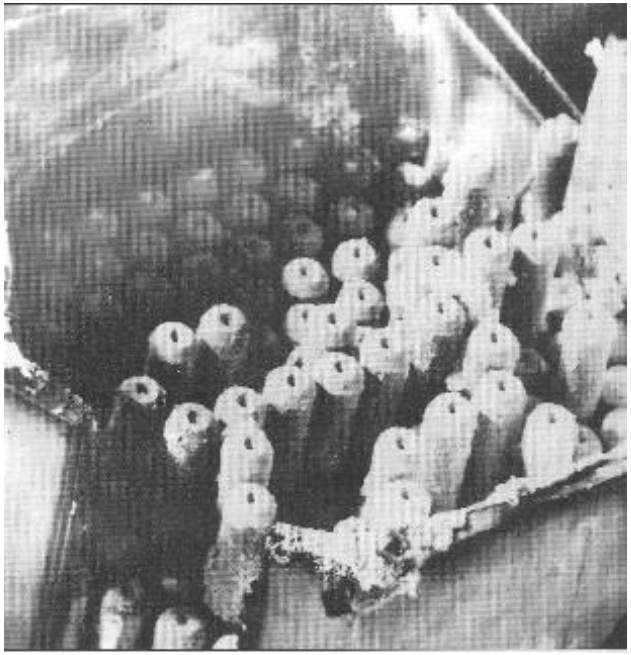


316 before irradiation



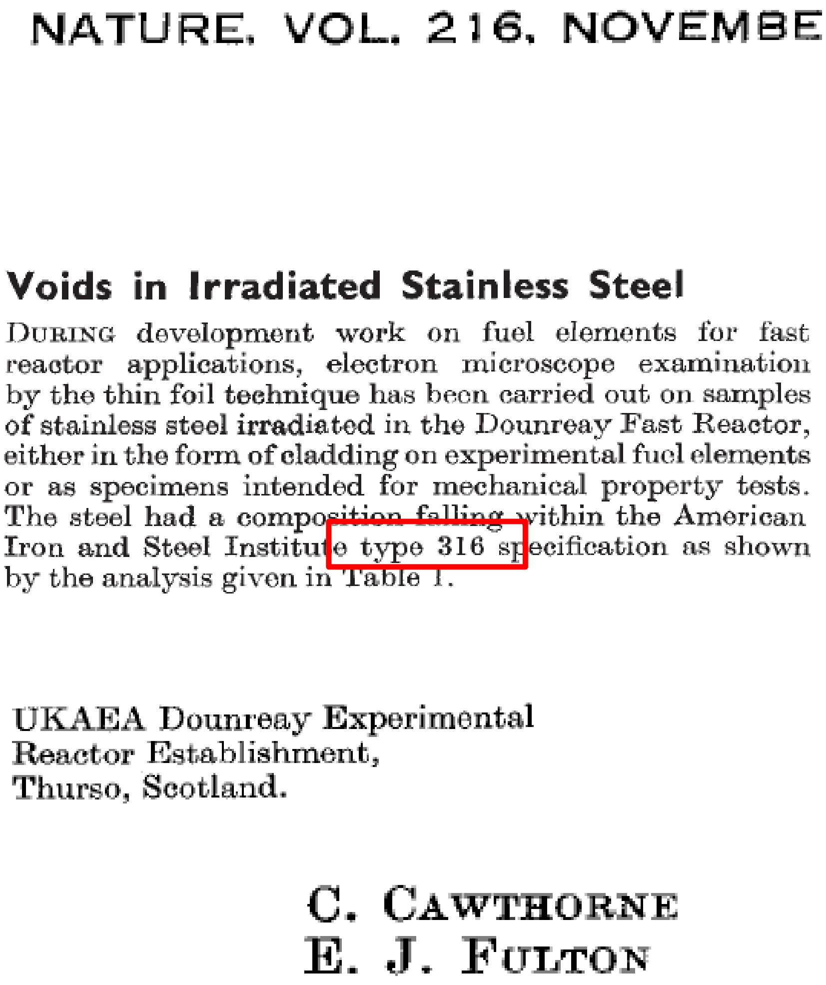
316 after irradiation

Phenix



Rapsodie

Void swelling : swiss cheese microstructure



NATURE, VOL. 216, NOVEMBER 11, 1967

Voids in Irradiated Stainless Steel

DURING development work on fuel elements for fast reactor applications, electron microscope examination by the thin foil technique has been carried out on samples of stainless steel irradiated in the Dounreay Fast Reactor, either in the form of cladding on experimental fuel elements or as specimens intended for mechanical property tests. The steel had a composition falling within the American Iron and Steel Institute type 316 specification as shown by the analysis given in Table 1.

UKAEA Dounreay Experimental
Reactor Establishment,
Thurso, Scotland.

C. CAWTHORNE
E. J. FULTON

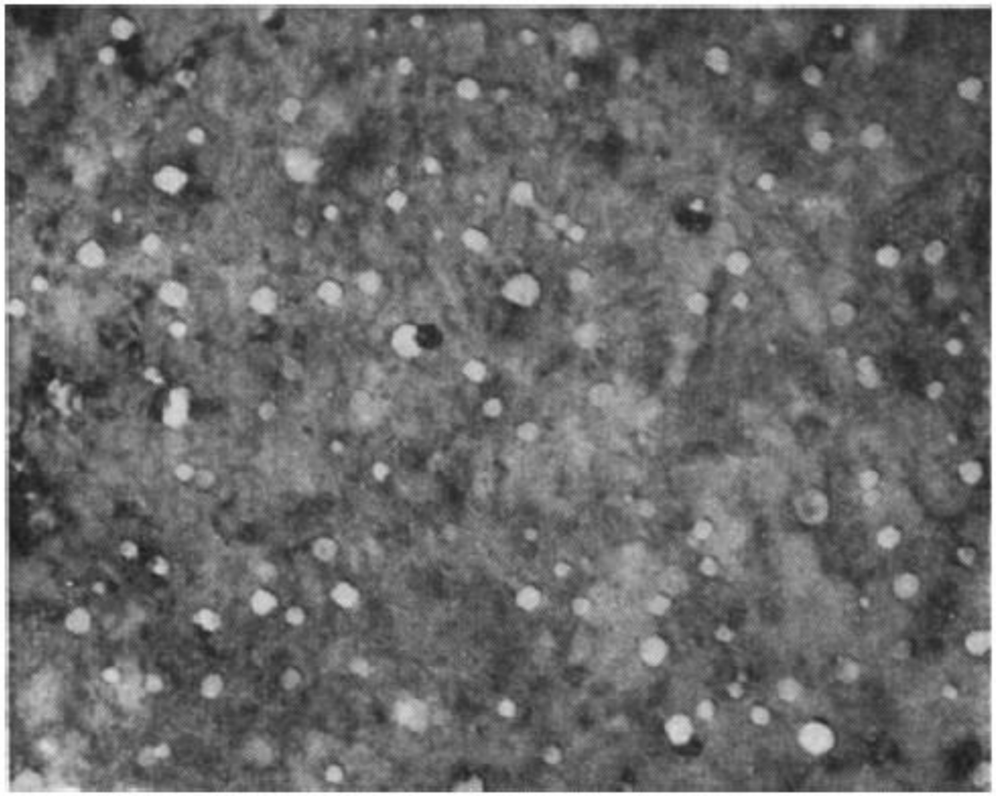
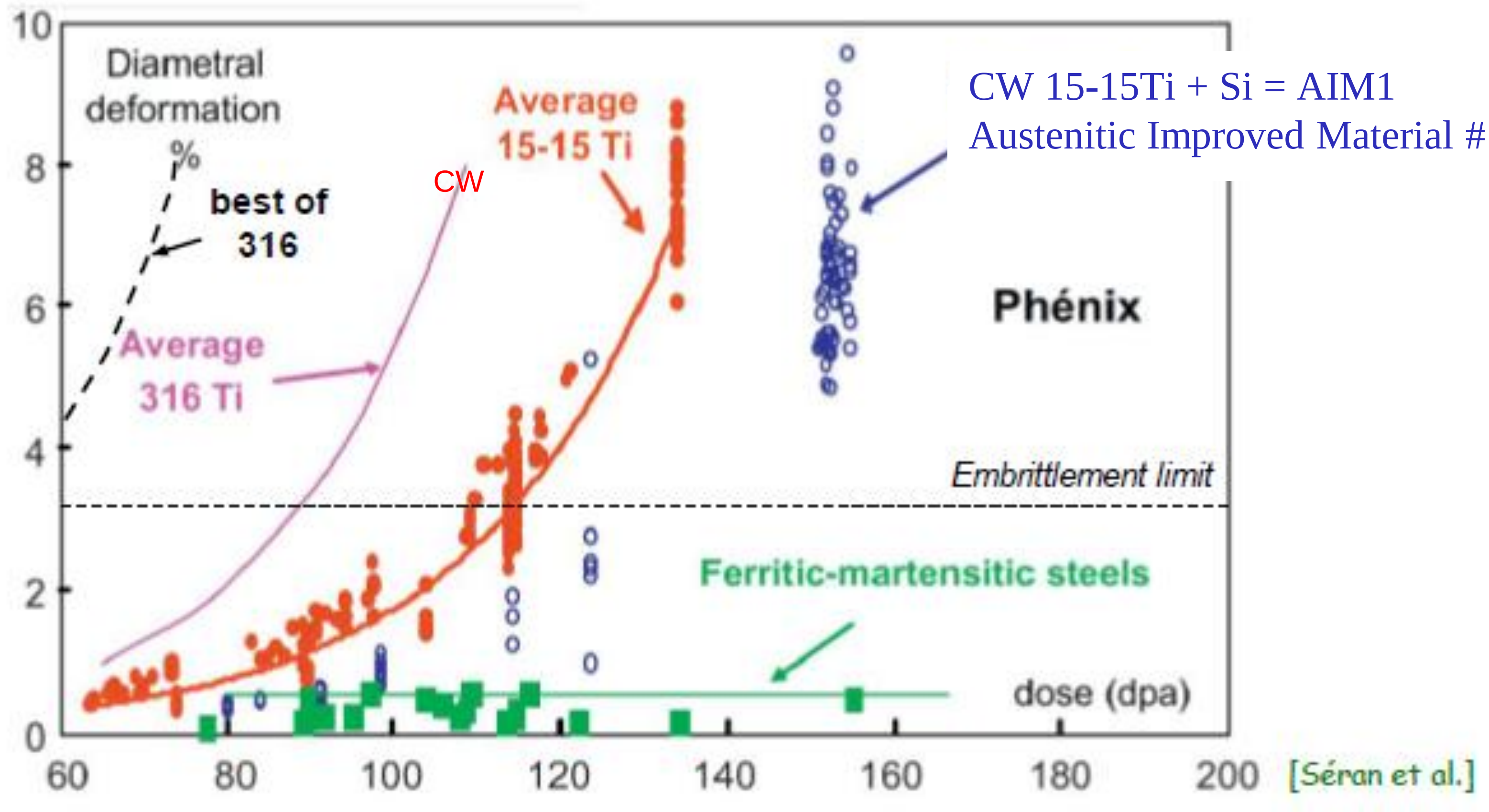


Fig. 1. Sample of fuel element cladding irradiated at 510° C to a neutron dose of $4{\cdot}7 \times 10^{22}$ ncm^{-2} (× 60,000).

C. Caxthorne et al., Voids in Irradiated Stainless Steel, Nature, vol. 216, 575-576 (1967).

Optimization to improve swelling resistance

- Phenix experience:



CW 15-15Ti + Si = AIM1
Austenitic Improved Material #1

P. Yvon, J.L. Séran et al., Nuclear Engineering and Design, volume 294, 161-169 (2015).

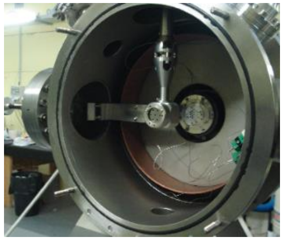
R&D on advanced austenitic steels AIM2

Subject: Study of the bi-stabilization effect (Ti, Nb) on swelling resistance of 15-15Ti steels for further AIM2 recommendations.

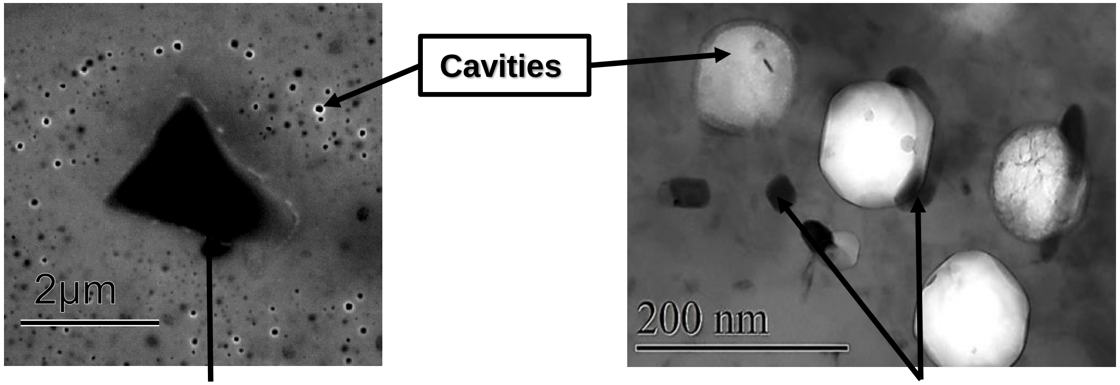
Experimental procedure : Elaboration of model alloys (Fe-15Cr-15Ni + Ti, Nb…), microstructural characterization, study of the behavior under ion irradiations performed at JANNus-Saclay.

Temperatures: 500°C, 550°C, 600°C
Doses: from 3 to 130 dpaKP
Ions: Fe^{3+} ; 2MeV

JANNUS

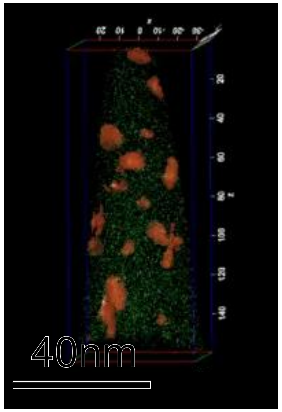


Results: Multiscale study of microstructural effects on swelling.



Primary precipitate

Secondary precipitates



Nanocarbides induced by ion irradiation (APT)

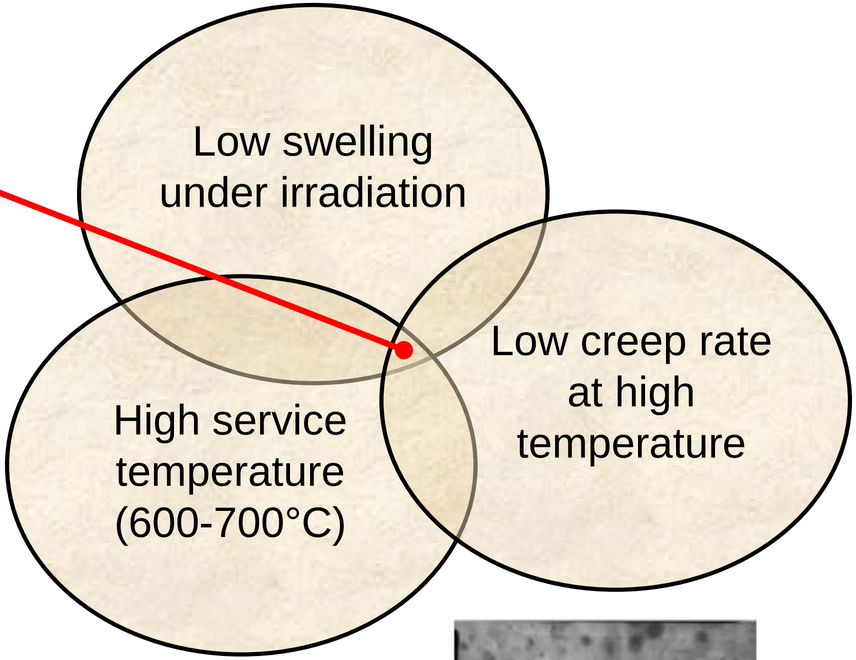
Outlooks:
- Impact of irradiation temperature and nanocarbides precipitation on swelling resistance
- Simulation of microstructural evolution by clusters dynamics (CRESCENDO code)

A.Vaugude et al., *Étude paramétrique du gonflement sous irradiation d'aciers austénitiques 15Cr/15Ni par dynamique d'amas*, Matériaux 2018-799 (2018). o

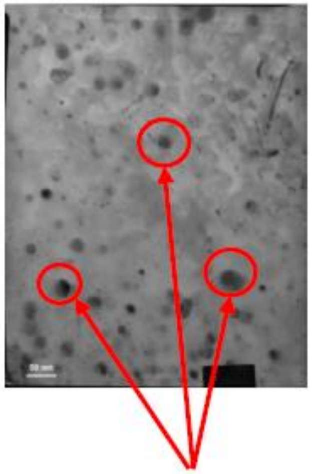
ODS stainless steels programme

ODS stainless steels

Ferritic/martensitic stainless steels (Fe - 9/18% Cr) reinforced by an homogeneous dispersion of nano-sized oxides particules (YTi_2O_7).



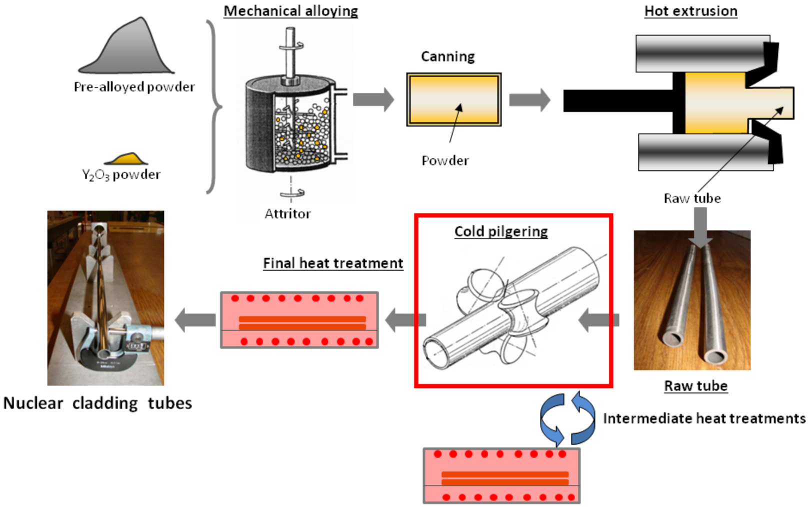
- Very low swelling (BCC structure + trapping action of nanoprecipitates)
- Good corrosion properties (%Cr)
- High service temperature
- Good creep behavior (precipitation)



Nano-oxides (radius ~ 2-4 nm)

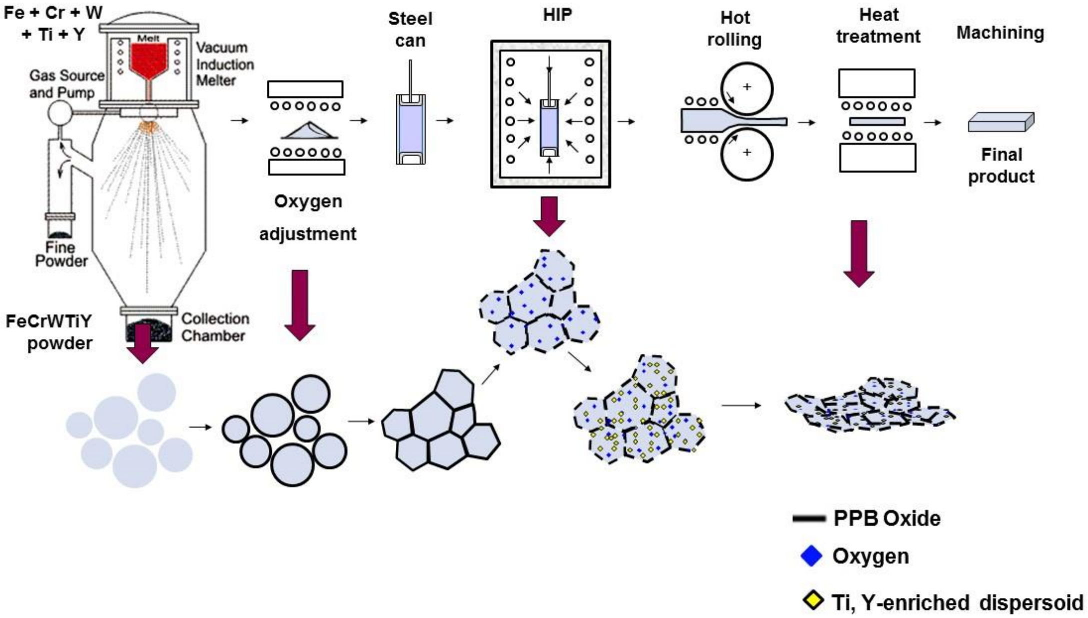
P. Yvon, F. Carré, *Structural materials challenges for advanced reactor systems*, JNM, 385 (2009) 217–222.

ODS Programme : manufacturing



AIM2 and ODS tested in BOR 60

MA



STARS ***(Surface Treatment of gas Atomized powder followed by Reactive Synthesis)***

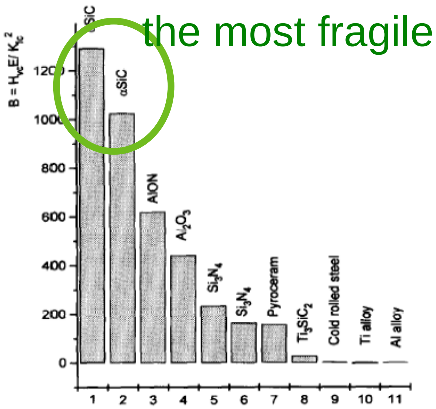
Additive manufacturing?

GFR Fuel design

Cladding Function :

- Leak-tightness barrier to the fission products
- Good mechanical behavior
    - ductility, fracture toughness
- Heat transfer exchange
- Chemical compatibility with fuel

SiC
the most fragile



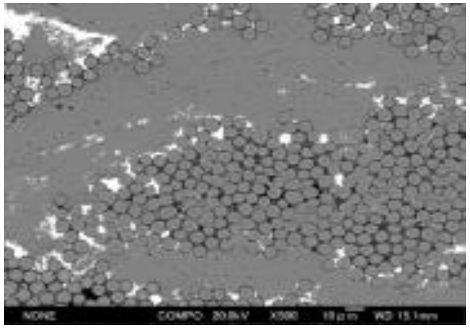
Paramètre de fragilité
[Pampuch, JECS 98]

Loading of the cladding in operation :

- Pressure fission gas, primary coolant
- Strain controlled
    - dilatation, swelling,
    - pellet cladding interaction
- Irradiation impact

~~Refractory metals Mo, W,…~~

SiC$_f$/SiC



a Materials challenge

SiC/SiC composites in GFR

Technological lock : find a structural material for fuel containment

Functional Requirements : (+ corrosion and leaktightness)

- ✓ neutronic compatibility
- ✓ keep good mechanical property and thermal conductivity under :
  - high operating (up to 1100°) and accidental (> 1600°C) temperature
  - fast neutron and high fluence operating conditions

⇒ **SiC is the best candidate from past experience**

But monolithic SiC does not possess adequate toughness and deformation capability

SiC/SiC composites have advantages :

- ✓ improved toughness (tolerance to damage via microcracks process)
- ✓ improved deformation capability (due to fiber reinforcement)

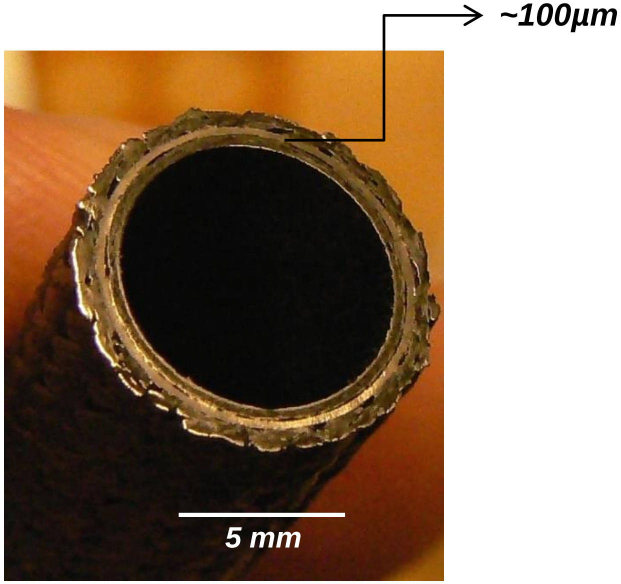
⇒ **SiC/SiC composites are the best candidates**

Matrix cracking – Impact on gas-tightness

- When cracks occur, there is a loss of leak-tightness (fission products) = « leak before break »
- Need of gas-tight systems abble to sustain strain in operating conditions

→ CEA sandwich concept with inner metallic liner

(CEA Patent)



Examples of sandwich cladding prototype
SiC/SiC + metallic liner + SiC/SiC

- Gastightess and mechanical properties have been demonstrated
- Good chemical compatibility between SiC and liner (Ta, Nb)

Gas tight pin cladding concepts

CEA "sandwich" SiC/SiC pin cladding concept (GFR)

Motivation & challenges

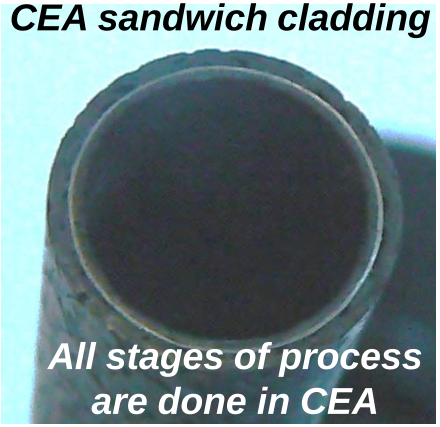
Patent WO 2013/017621 A1

- SiC/SiC cladding: refractory & resistant to irradiation… but prone to micro-cracking
- Leak-tighness is an issue ⇒ separate functions « resistance » / « containment »
- US-proposed "Duplex/Triplex" design (monolithic SiC inner layer) raises questions:

SiC failure beyond elastic limit & End-plug joining ⇒ long-term leak-tightness?

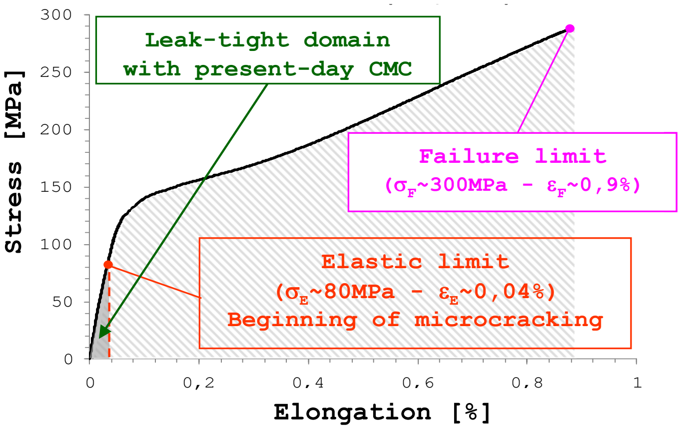
- Metallic liner: ductility & weldability… but raises compatibility issue (SiC & UPuC)

CEA sandwich cladding

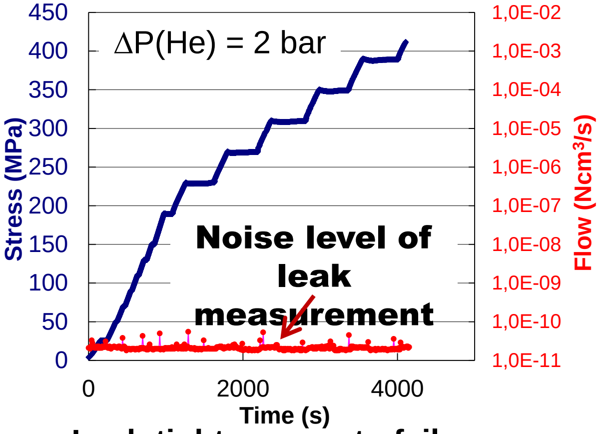


All stages of process are done in CEA

Inner SiC/SiC: e~0.3mm
liner Ta : e<0.1mm
Outer SiC/SiC: e~0.6mm



Demonstration was made by He permeation measurements during tensile test



Leak-tightness up to failure limit of structure

- **Metallic liner only ensures tightness up to failure limit**
- **Composite ensures mechanical resistance**
- **Process is simple and reproducible**
- **End-plugging solutions (welding)**

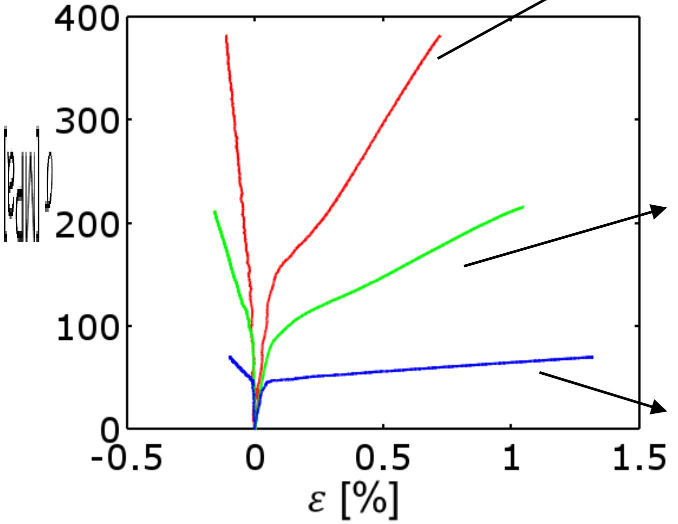
E-ATF !!!!

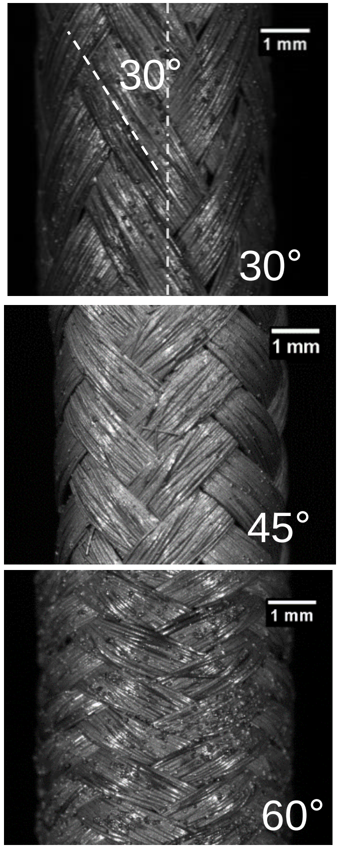
R&D on SiC/SiC composites

Subject: 3D investigation of damage in SiC/SiC composites; effect of the braiding angle.

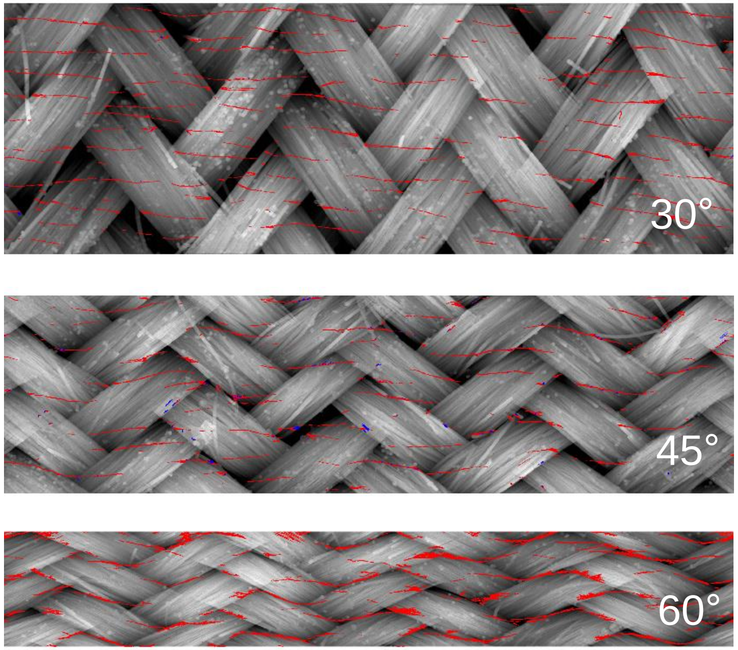
Braiding angle effect:

Macroscopic behavior





Damage mechanisms



Conclusion: - Crack initiation: understanding of the braiding angle effect on the onset of damage.
- Crack propagation: different types of cracks evidenced for different braiding angles.

Y. Chen, *Damage mechanisms of SiC/SiC composite tubes: three-dimensional analysis coupling tomography imaging and numerical simulation*, PhD Thesis, CEA, Université Paris-Est, 2017.

Structural materials for Innovative Reactor Systems

	SFR	GFR	LFR	VHTR	SCWR	MSR	Fusion
Coolant T (°C)	Liquid Na few bars	He, 70 bars 480-850	Lead alloys 550-800	He, 70 bars 600-1000	Water 280-550 24 MPa	Molten salt 500-720	He, 80 b 300-480 Pb-17Li 480-700
Core Structures	Wrapper *F/M steels* Cladding *Aust* *F/M ODS*	Fuel & core structures *SiCf-SiC composite*	Target, Window Cladding *F/M steels* *ODS*	Core *Graphite* Control rods *C/C* *SiC/SiC*	Cladding & core structures *Ni based Alloys &* *F/M steels*	Core structure *Graphite* *Hastelloy* *Ni based alloys*	First wall Blanket *F/M steels* *ODS* *SiC_f-SiC*
Temp. °C	*390-700*	*600-1200*	*350-480*	*600-1600*	*350-620*	*700-800*	*500-625*
Dose	Cladding *200 dpa*	*60/90 dpa*	Cladding *~100 dpa* ADS/Target *~100 dpa*	*7/25 dpa*	*7/30 dpa*	*up to 100 dpa*	*> 100 dpa* + 10 ppmHe/dpa + 45 ppmH/dpa
Other components		IHX or turbine *Ni alloys*		IHX or turbine *Ni alloys*			

Conclusions

- **Materials science and new materials are the key to meet the advanced nuclear systems objectives :**

- **Incremental progress and breakthroughs are sought on a wide span of structural materials for fuel claddings, core structures, reactor cooling systems & components** (RPV, IHX, SG…)**, power conversion systems** (electricity, H_2…)

- **Increased role of Materials science** (analytical research and modelling) **for a more predictive R&D towards aimed materials properties – need for multiscale modelling, experimental simulation and "smart" experiments in MTRs**