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

**RTE ECO\textsubscript{2}mix app**

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

Map showing energy flow:
- Exports: 10749 MW at 4h00, 8405 MW at 8h30
- Imports: 1360 MW at 4h00, 3524 MW at 8h30
- Flow changes between 4h00 and 8h30:
  - 2000 MW from 4h00 to 8h30
  - 3524 MW import at 8h30
PCI: the balance

- $P_L$, $T_c$
- Pellet; expansion and diabolo shape
- $\sigma_\theta$ 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

- 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

I induced SCC after power transient

100 μm

cladding

fuel
Pellet Cladding Interaction

Iodine induced Stress Corrosion Cracking
- Intergranular cracking

3 points are needed:
1 sensitive 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

![Graph showing local LHGR vs. local burnup (GWd/t)](image)

- **Paramètres de fonctionnement**
  - Plin max, DPlin, vitesse

- **Paramètres du crayon**
  - Bu, gaine, combustible

- **Direction de l’Energie Nucléaire**

- **Rupture threshold**
  - UO$_2$ Zy4 cladding

- **Zy-4 PCI Threshold**
Accidental conditions LOCA
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

Phase 1
Ballooning- rupture
(fill in the channels)

Phase 2
High temperature oxidation

Phase 3
Reflooding

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
microstructurals and mechanical consequences:

Accelerated oxidation $\Rightarrow$ oxygen diffusion inside the metal $\Rightarrow$ brittleness

Residual ductility of the cladding depend on:

1. Ex-$\beta$ phase thickness
2. C(O) in the Ex-$\beta$ phase after quenching

**ductile $\Leftrightarrow$ brittle** transition

à $[O]$ critical $\sim 0.4\text{wt.\%}$
post oxidation /quench mechanical tests

Low oxidation level: ductile material

High oxidation level: fragile material
Current criteria:

- $T_{cladding}$
  $PCT < 1204^\circ C$

- $ECR < 17\%$
  (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 ~ $2.10^{21} \text{ n/cm}^2$)
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
**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**

Members of the Generation IV International Forum:
- USA
- France
- Canada
- Brazil
- South Africa
- South Korea
- Japan
- Switzerland
- EU
- United Kingdom
- China
- Russia

Direction de l’Energie Nucléaire
New challenges for materials!

Generations II-III

Gas Fast Reactor

Lead Fast Reactor

Supercritical Water-cooled Reactor

Very High Temperature Reactor

Molten Salt Reactor

Sodium Fast Reactor

Displacement per atom (dpa)

Temperature (°C)

Supercritical Water-cooled Reactor

Very High Temperature Reactor

Generations II-III

Molten Salt Reactor

Sodium Fast Reactor

Gas Fast Reactor

Lead 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

- **Corrosion resistance** *(primary coolant, power conversion, H₂ production)*
  - Corrosion and stress-corrosion cracking *(IGSCC, IASCC, hydrogen cracking & chemical compatibility…)*
Material requirements in future nuclear systems

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)

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.
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 glasses 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
- 600 MWe – $T_{He} \approx 480 \, ^\circ C$
- Potential for integral recycling of Actinides

High irradiation doses on cladding Corrosion

ELS Y
EUROTRANS
in EU FP6

Euratom countries

Japan

South Korea

U.S.A.

LFR Steering Committee

✔ 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)

System Arrangement SCWR signed Nov. 30, 2006

HPLWR in EU FP6

Euratom countries

Canada

Japan

South Korea

EAC
Corrosion

SCWR Steering Committee
Molten Salt Reactor (MSR)

Characteristics
- Fuel is liquid fluorides of U and Pu with Li, Be, Na and other fluorides
- 700–800°C 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
Sodium Fast Reactor (SFR)

A new generation of sodium cooled Fast Reactors

Reduced investment cost
Simplified design, system innovations
(Pool/Loop design, ISIR – SC CO₂ PCS)

Towards more passive safety features
+ Better management of severe accidents

Integral recycling of actinides?
→ Remote fabrication of TRU fuel

High irradiation doses on cladding and wrapper tubes
ECS fabrication

2008 +

Russia

France

South Korea

U.S.A.

Japan

Euratom countries

SFR Steering Committee

Direction de l’Energie Nucléaire
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 – $T_{\text{He}} \sim 850$ °C - Cogeneration of electricity, H₂, synfuel, process heat

- Safe management of cooling accidents

- Potential for integral recycling of Actinides

High temperatures in accidental conditions

GFR Steering Committee

- System Arrangement GFR signed Nov. 30 Nov., 2006
Development of fast reactors with a closed cycle

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

**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

**Core Sub-assemblies**
- 400 – 650°C
- Irradiation

**Core**
- 30 → 60 ans
- **Life time to design**

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

**Steam Generators, Heat Exchangers**
- 350 – 525°C
- Aging, Welds, Compatibility
- Avoid Na - H₂O

**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

**1000 MWe**
- Pool type
- **Modular SG**
- AREVA design
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)

![Graph showing temperature and dose distribution for cladding and wrapper](image)
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-PuO₂ 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
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 $4.7 \times 10^{21}$ ncm$^{-2}$ ($\times 80,000$).

Optimization to improve swelling resistance

Phenix experience:

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

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

**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)

ODS stainless steels

Ferritic/martensitic stainless steels (Fe - 9/18% Cr) reinforced by an homogeneous dispersion of nano-sized oxides particles (YTi$_2$O$_7$).

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

ODS Programme: manufacturing

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

Additive manufacturing?

AIM2 and ODS tested in BOR 60
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

Loading of the cladding in operation:
- Pressure: fission gas, primary coolant
- Strain controlled: dilatation, swelling, pellet cladding interaction
- Irradiation impact

Materials challenge:
- Refractory metals: Mo, W, ...
- SiCf/SiC

Paramètre de fragilité [Pampuch, JECS 98]
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)

- Gastightness 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**

- SiC/SiC cladding: refractory & resistant to irradiation... but prone to micro-cracking
- Leak-tightness 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)

**Demonstration was made by He permeation measurements during tensile test**

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

**Leak-tight domain with present-day CMC**

**CEA sandwich cladding**

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

**Elongation [%]**

- Metallic liner only ensures tightness up to failure limit

**Stress [MPa]**

- Composite ensures mechanical resistance

**Process is simple and reproducible**

- End-plugging solutions (welding)

**Noise level of leak measurement**

- Leak-tightness up to failure limit of structure

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

Braiding angle effect:

Macroscopic behavior

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.

## Structural materials for Innovative Reactor Systems

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

- **Other components**: 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, H₂…)
  - 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