PWR Internal Structures

PWR Cladding

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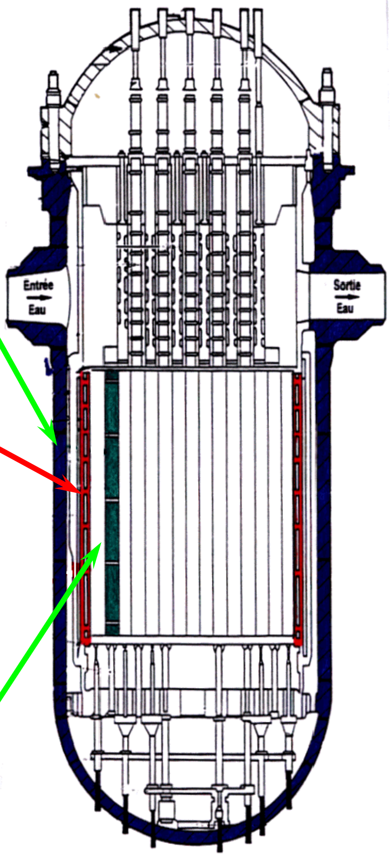
Internals

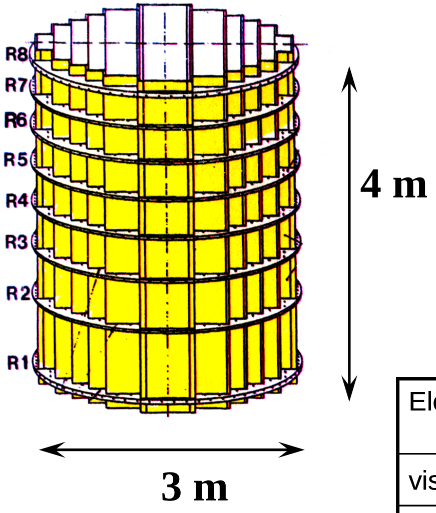
Pressure vessel

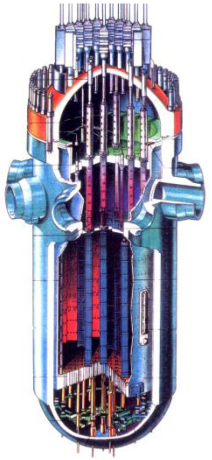
Internals

- barrier
- baffle
- baffle bolts

Fuel assembly

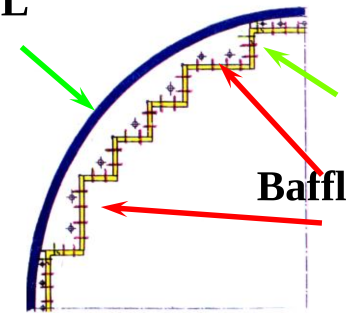






Elément	Matériau	Dose (dpa) après 40 ans
visserie	Cold worked	10 à 80
cloisons	Quench annealed	10 à 80
renfort	Quench annealed	5 à 60

Core shroud (envelop) 304L weld 308L

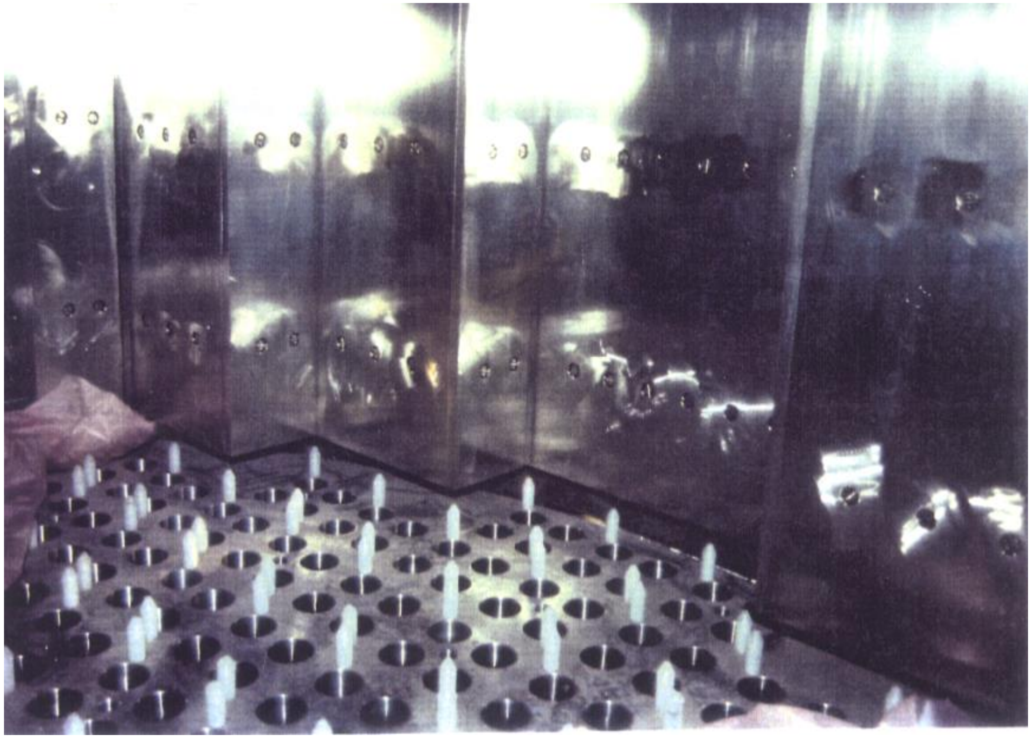


Baffle 304L

Baffle bolts 316 / 316

960 /reactor

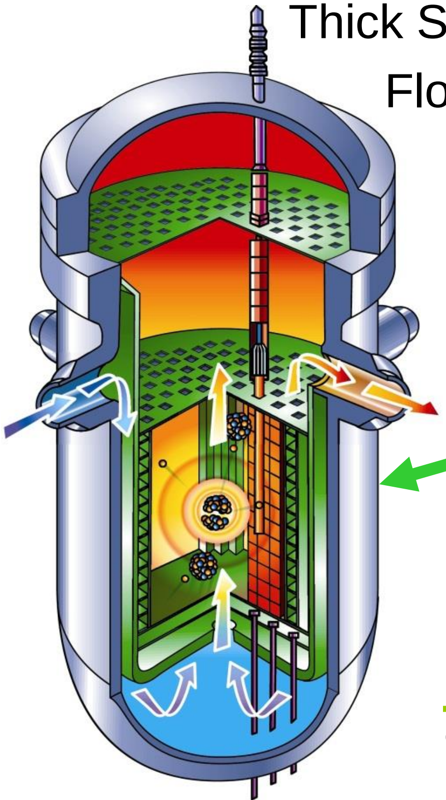
Internals N4 (PWR 1350MW)



Role of internals for PWR reactors

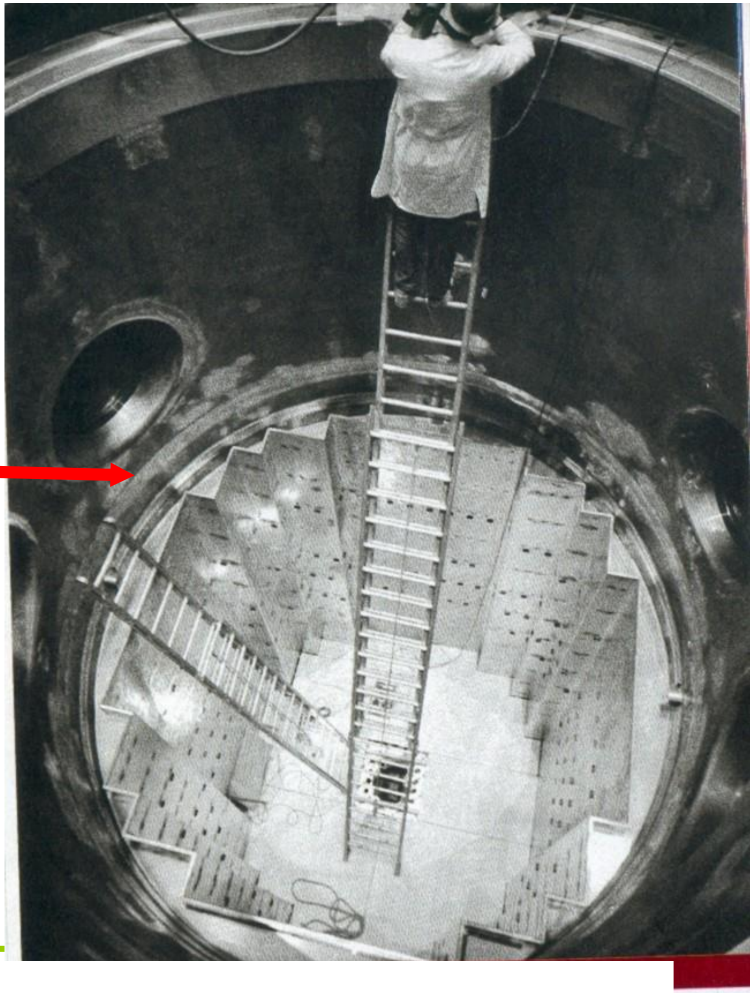
core internal structures (core mechanical support, hydraulic, neutronic protection of the vessel)

- Baffle
  - Junction between pressure vessel cylinder and poly-square type fuel assembly core
- Thick SS plates screwed together
  - Flow control

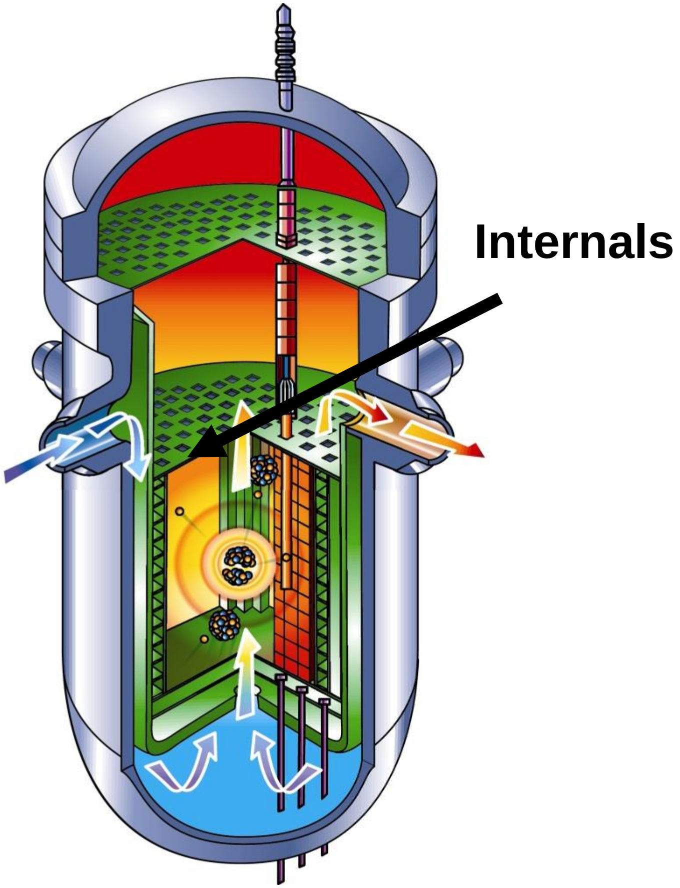


Internals (temperature, up to 380°C)

Internal structures of PWR



PWR Internals



18%Cr and 8-10%Ni (18-08/10) : 304 et 316

2 to 100 dpa : end of life dose

Irradiation effects: Aggregation of points defaults: loops, voids : **hardening**

- Depending on the temperature
  Hardening, reduction of ductility,
  Quick increase then saturation

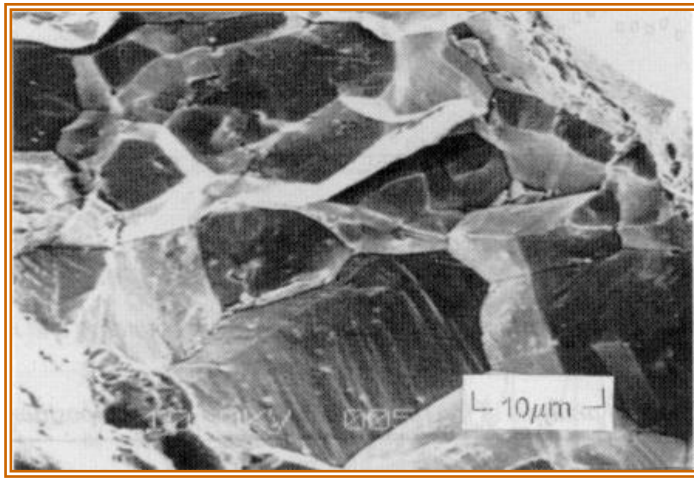
- Swelling is possible at high doses

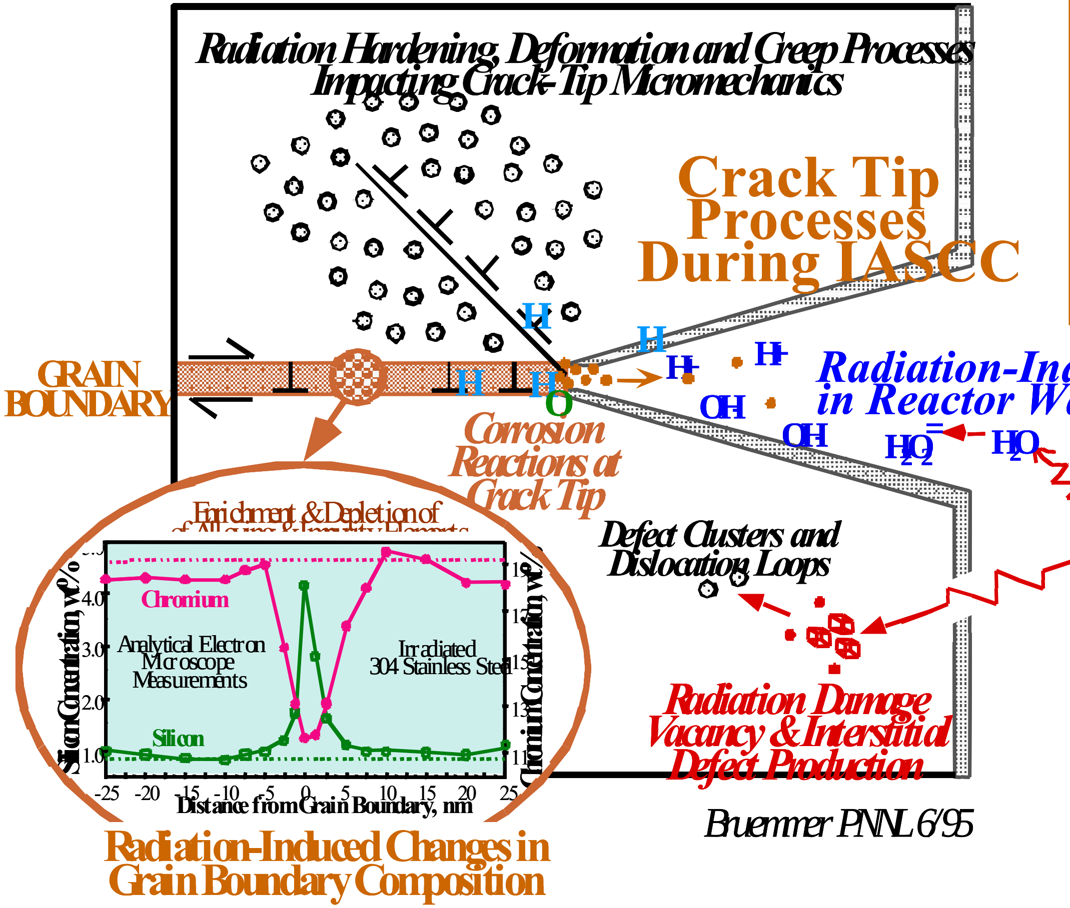
- He formation *in situ* (hardening)

^{58}Ni (n,γ) ^{59}Ni then ^{59}Ni (n,α) ^{56}Fe

For PWR : 0.5 to 1 ppm He/an

Irradiation assisted stress corrosion cracking





Radiation-Induced Changes in Reactor Water Chemistry

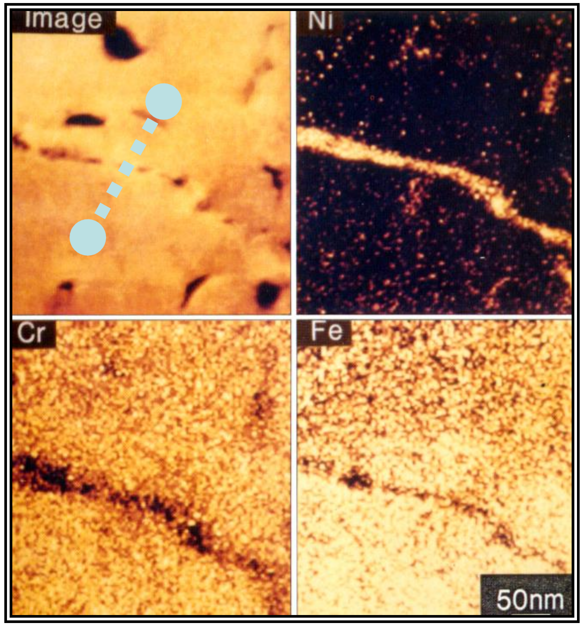
n

Bruemmer PNNL 6/95

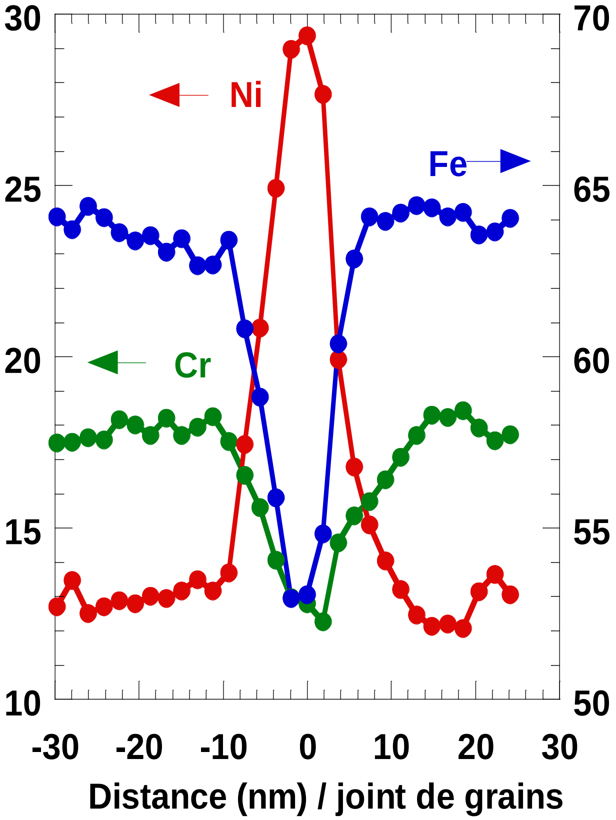
Radiation-Induced Changes in Grain Boundary Composition

Grain Boundary Segregation

Typical GB chemistry

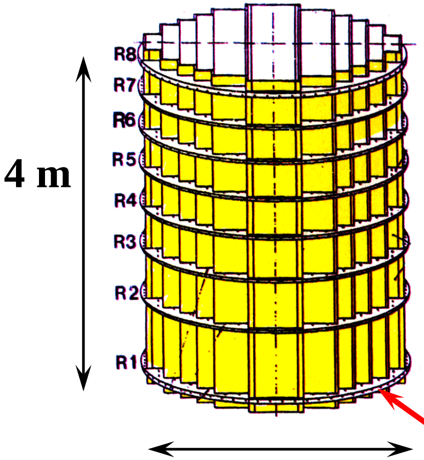


Dose : 3.7 10^{21} n.cm^{-2} , i.e. ≈ 0.1 dpa



One can observe lowering of Cr content at GB which favor the intergranular rupture. Especially if mechanical loading appears (stress corrosion cracking). This phenomenon is easier when hardening due to irradiation occurs and the corrosive environment which can be confined

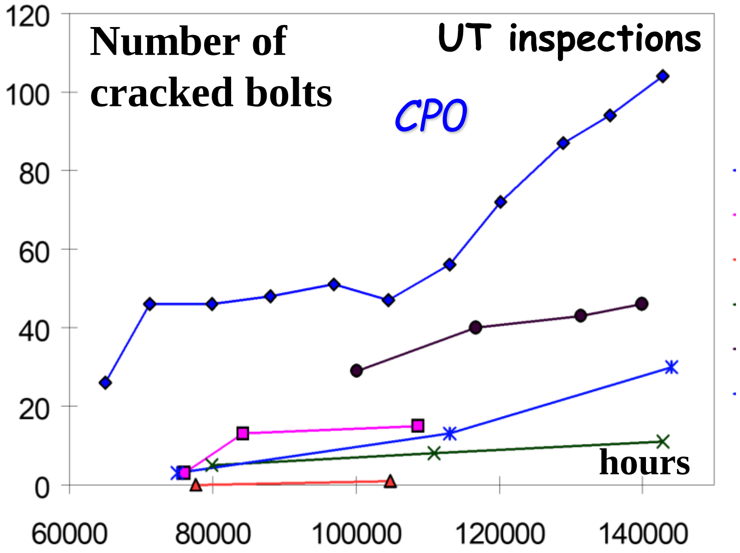
Irradiation effects on Core Internals



Cracked bolts

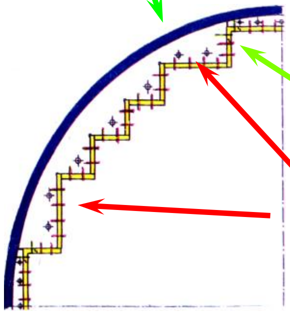
most irradiated

most stressed



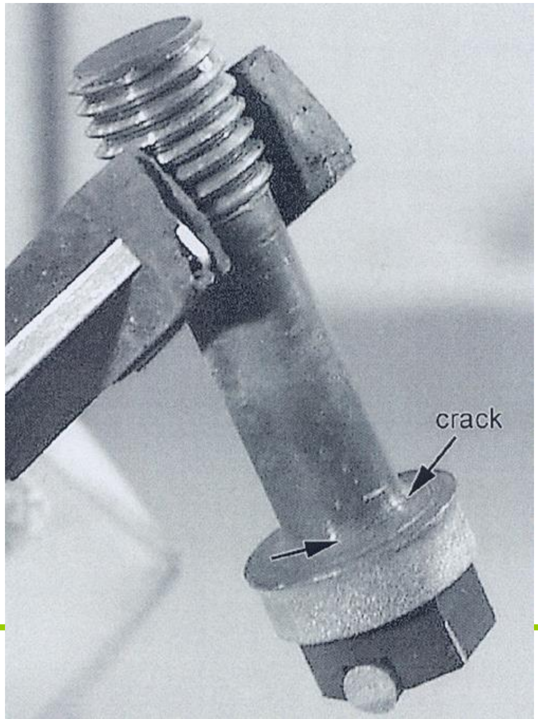
Former
SA 304L

Core barrel 304L
Welds 308L

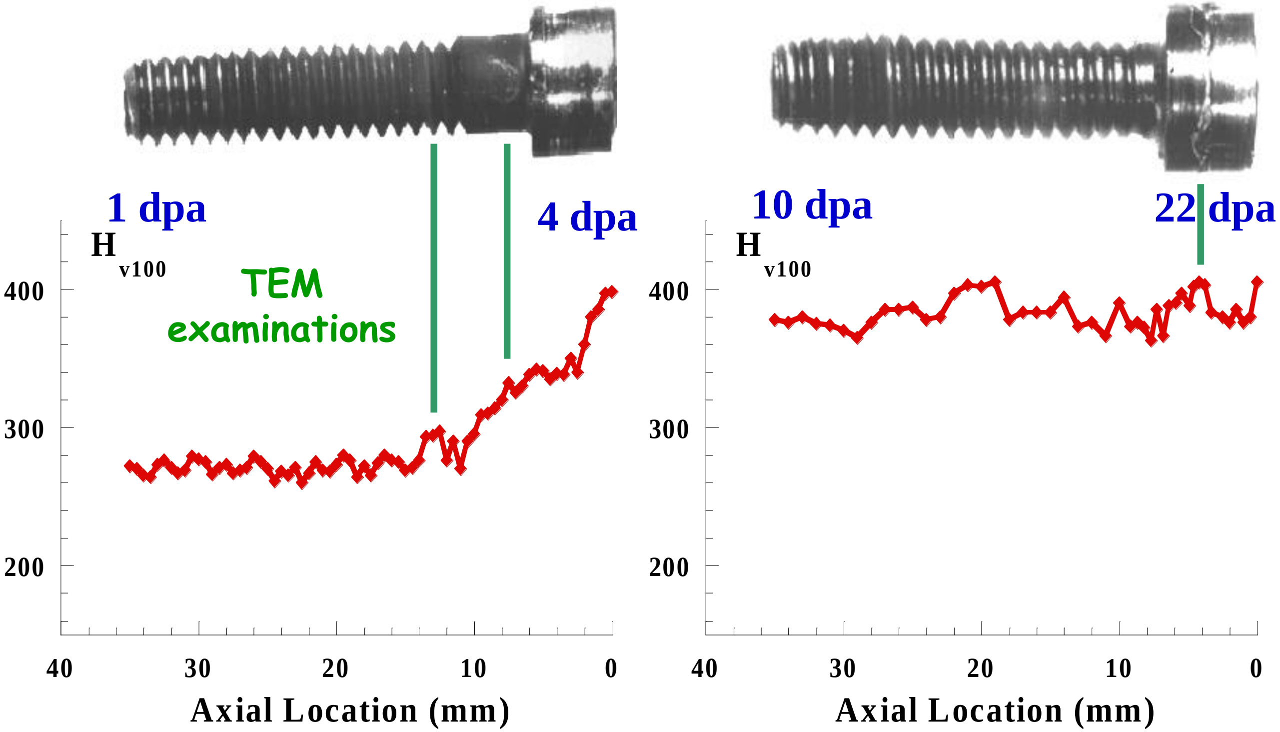


Baffle plate
SA 304L

Baffle bolts
CW 316 / 316L
960 /reactor



Hardness of internal baffle bolts for PWR



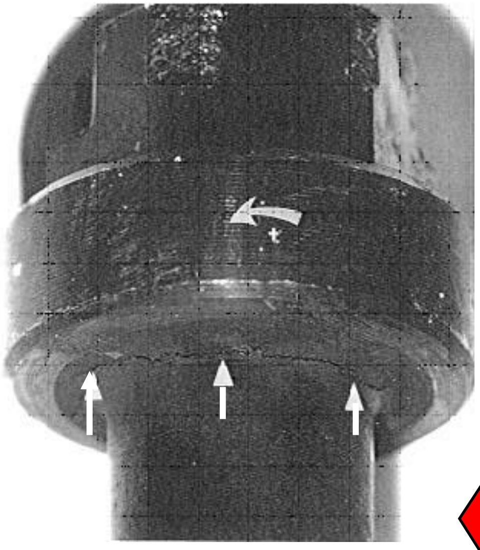
-> Saturation of the hardness around 5 dpa

High hardening

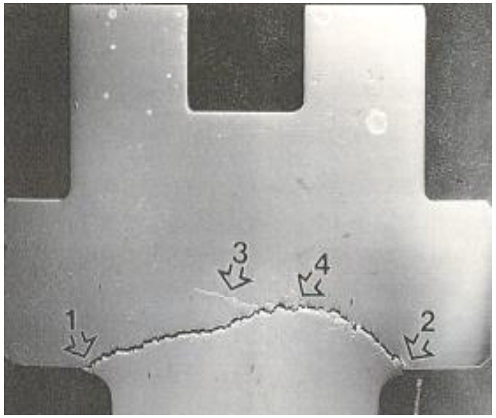
Cracking of baffle bolts for PWR internals IASCC

(Irradiation Assisted Stress Corrosion Cracking)

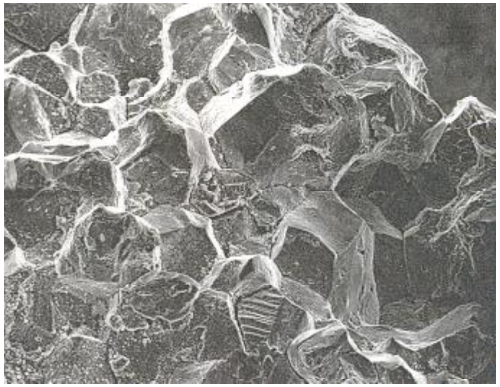
Irradiation dose : 5 - 10 dpa



5 mm



5 mm



25 µm

Cracking at the junction between: head - deformation area

Intergranular rupture (out of weld area !)

Stresses, atmosphere(+ irradiation), sensitive material

Solution : reduced loading in the bolts

-Modification of the working conditions of PWR (water flux=>thermohydraulic loading)

-Bolts design: stress concentration

-Replacing some bolts at mid life (at a cost ! To replace 150 bolts, the reactor needs a 3 week outing)

-Thinking about the improvement of the material (for all internals : search for low activation or rapid deactivation materials)

Outline

Fuel generalities

Fabrication

- Cladding and structural materials

Behavior under operating conditions

- Cladding and structural materials

Incidental conditions

Accidental conditions (design)

Transport and long term interim storage

Nuclear Fuel

Rule n°1 in France :

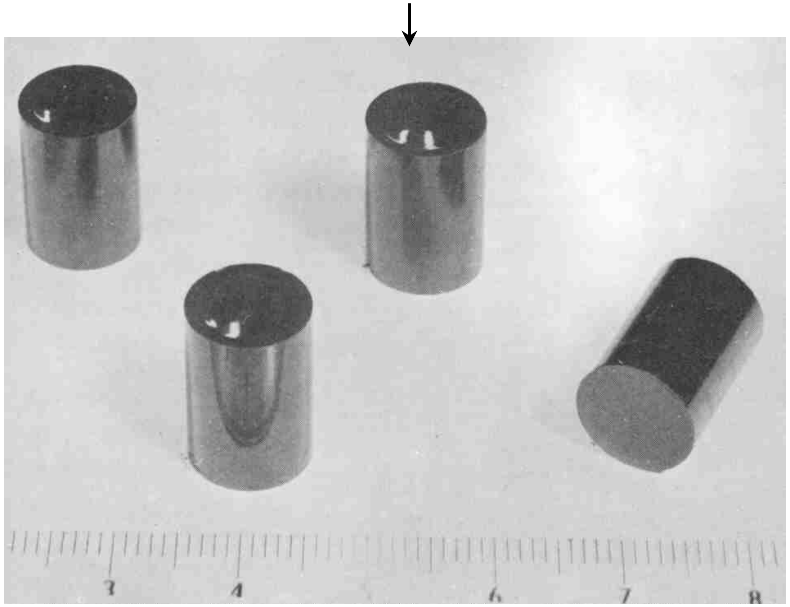
Containment of radioactive materials (FP, fission gases, fissile materials) by 3 leak tight barriers :

fuel pin + vessel + reactor boundary

Cladding : 1st barrier

Fuel pellet

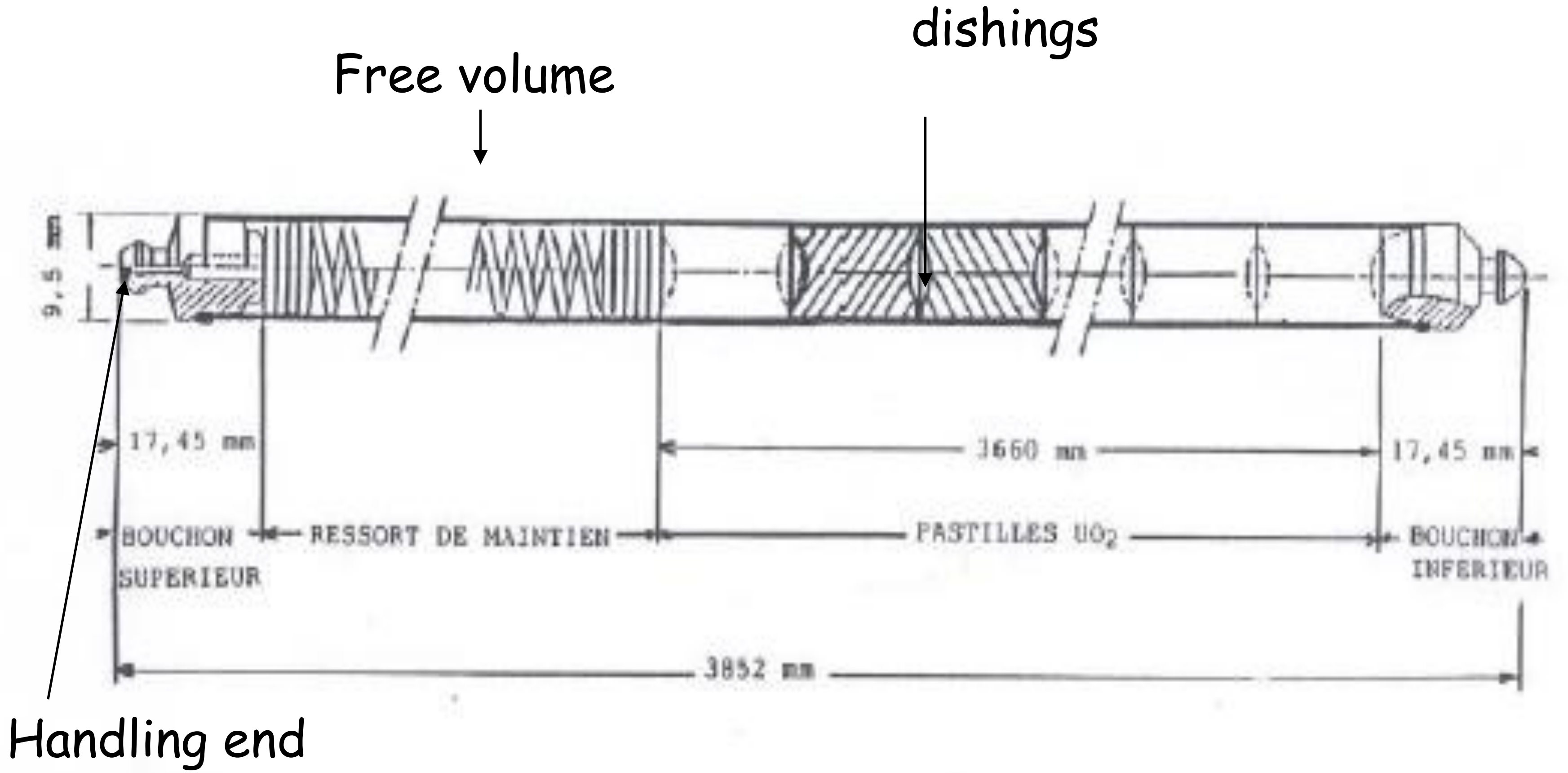
Fissile material : *Solid containing actinides (in PWR UO_2 or MOX pellets)*



1kg ^{235}U ~2000 T fuel or 3000 T coal

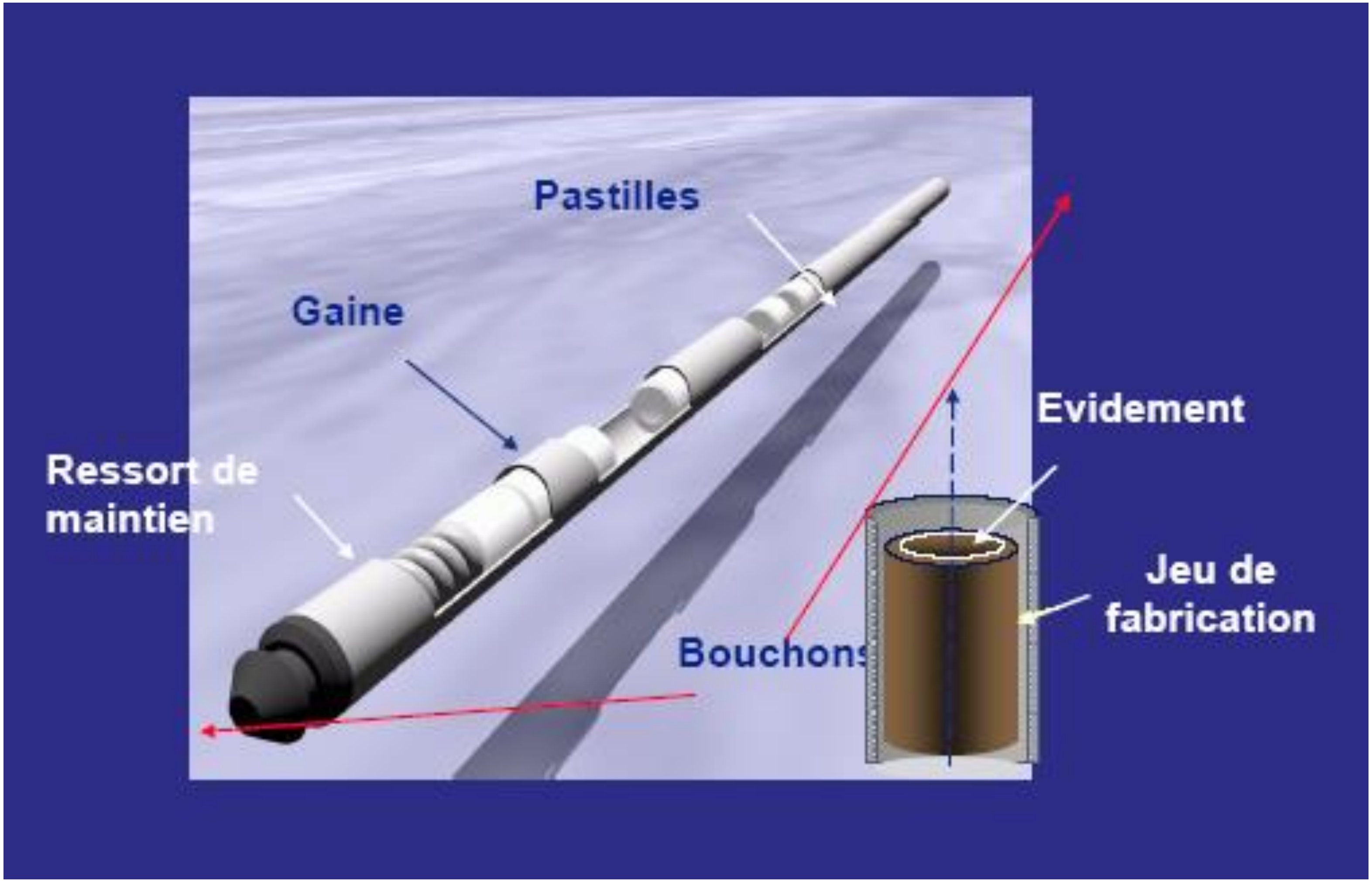
1 g Pu ~ 1 to 2 T of petrol

PWR fuel pin



CRAYON COMBUSTIBLE

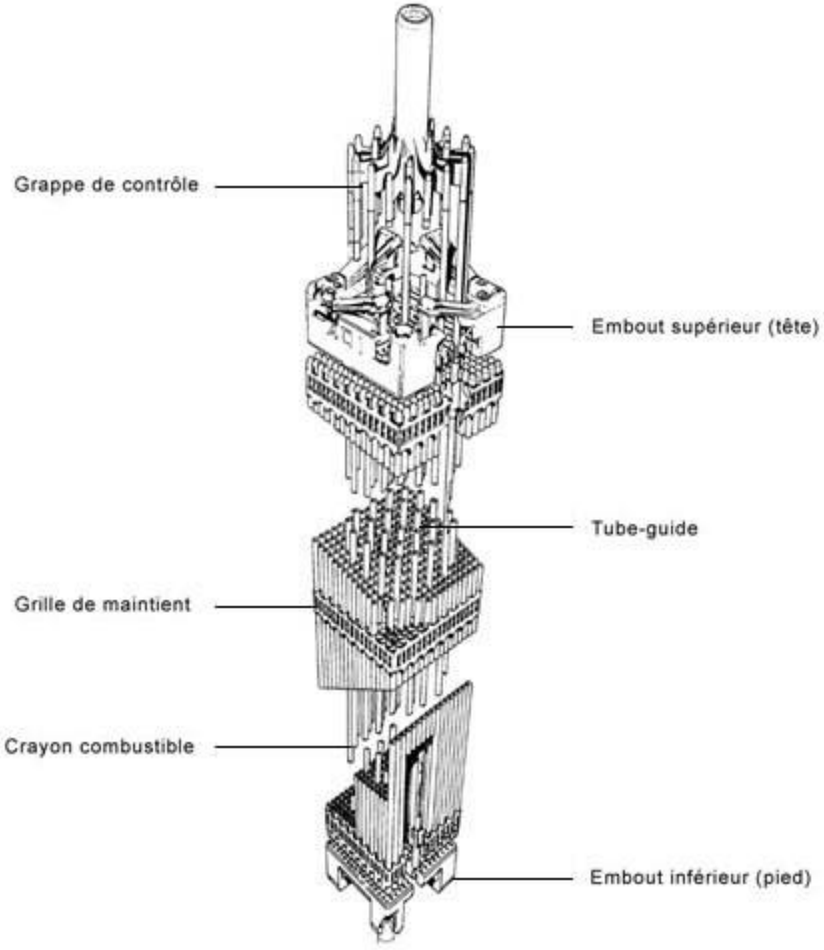
PWR fuel pin


Pastilles
Gaine
Evidement
Ressort de maintien
Jeu de fabrication
Bouchons

PWR fuel assembly

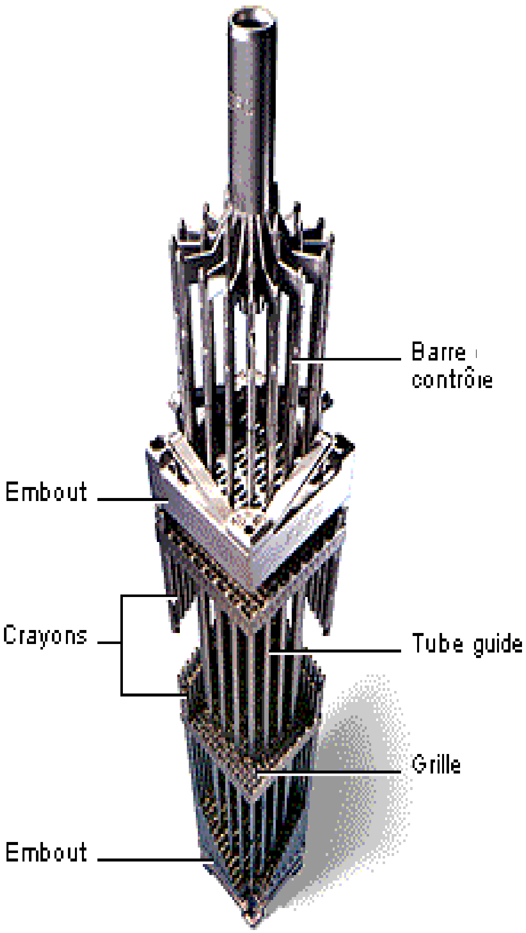
PWR

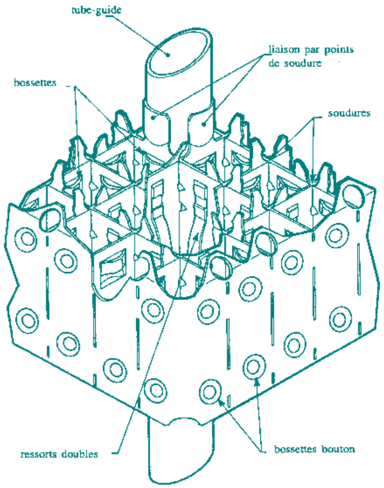
- **Square grid of 17x17, with 24 guides (for control rods) and an instrumentation tube**
- **Top and bottom plates** *(304 steel)*
- **8 or 10 spacing grids** *(Inconel replaced by Zr alloy)*



PWR fuel assembly

(17X17 = 289 - 24 (GT) - 1 (IT) =)





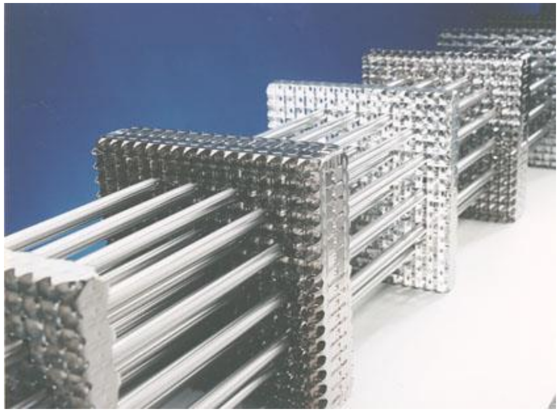
264 pins (17x17)
4,00 (4.80) m
mass 670 (765) kg

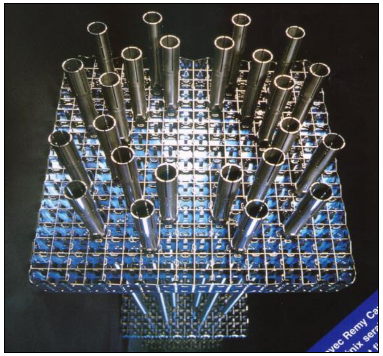
Nbr ass. / PWR : 157 (193)
3,66 (4,27) m of fuel/pin
152 (217) km of fuel pellets
Weight 72 (102) T

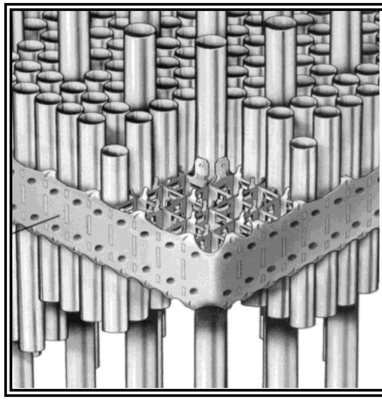
900 MWe(1300 Mwe)

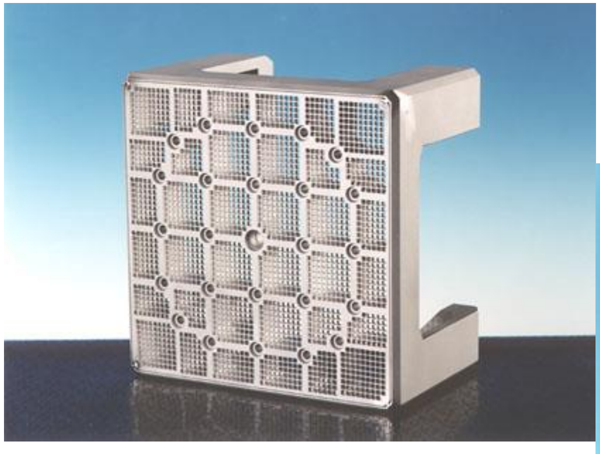
Rigidity ensured by the guide tubes
Turbulence generated by the dimples

The fuel assembly geometry

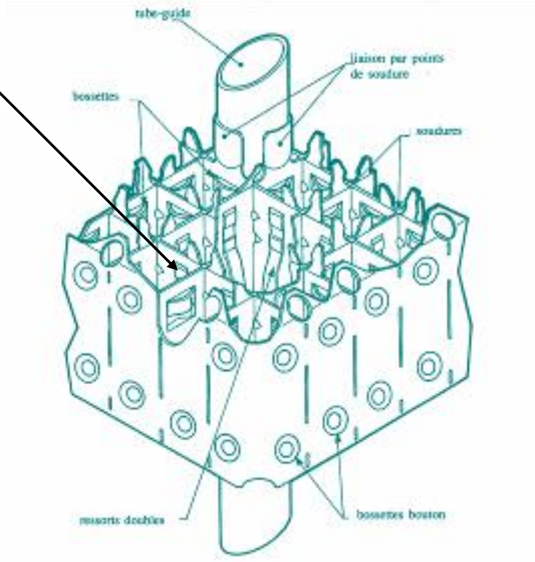








Spacer grids



Classes of operation of reactors

- **Class1 : normal operation**
  - **Base operation** : global power constant
  - **Load variation** (normal transients) :
- **Class 2 : incidents of limited occurence (0.01 < f < 1)**
  - Large local power variation (power ramp ⇔ **PCI**, Pellet Cladding Interaction)

- **Criteria for classes 1 and 2**
  - **Leak tightness of pins must be guaranteed**

  - **Pin must be cooled correctly**

  - **Oxyde must not melt**

Classes of operation of reactors

- **Class 3 : low frequency accidents ($10^{-4} < f < 10^{-2}$)**
  - Event causing damage to at least one barrier
- **Class 4 : serious and improbable accidents ($10^{-6} < f < 10^{-4}$)**
    - LOCA : Loss of Coolant Accident
    - RIA : Reactivity Initiated Accident
  - The reactor can be brought back to a safe and sub critical state
  - **Geometry allows for cool down**

Competitiveness to be found in fuel and its management !!!

To improve competitiveness

- increasing **Burnup** : produce more electricity with the same fuel assemblies
- increasing **linear power** : produce the same energy with less fuel assemblies
- increasing the **outlet temperature of water** : improvement of efficiency
- increasing **length of cycles** (12 → 18 → 24 months) : increase of disponibility of reactors

Limiting Phenomena

Will of utilities to increase burn ups

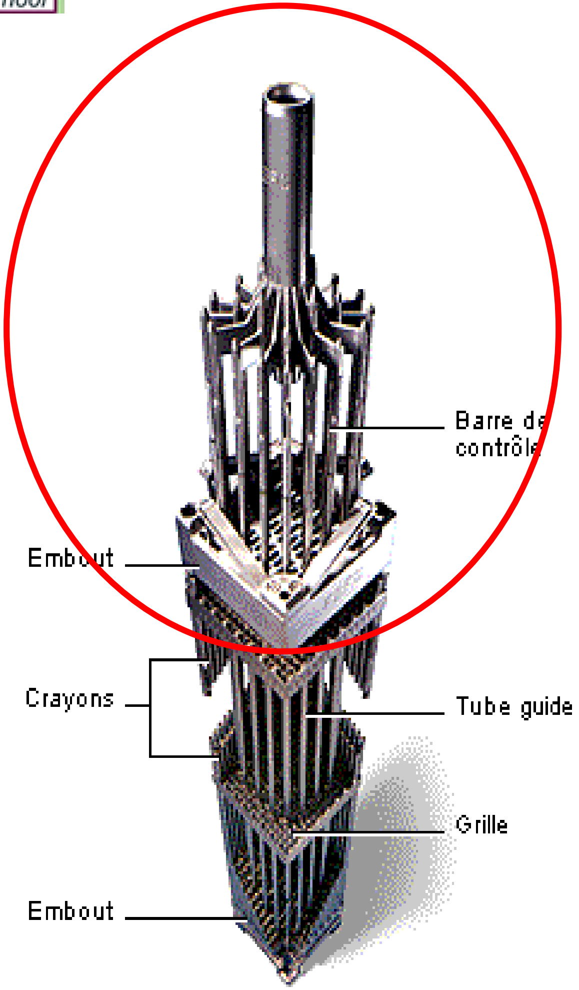
Economical interest (fabrication, back end, waste)

UO_2 52 ➔ 62 (➔ 70 GWj/t)
MOX 40 ➔ 52 (➔ 55 - 60 GWj/t)

Average burnup of most loaded assembly

- **Behavior of assembly**
  - Growth and bending of pins and guide tubes
  - Wear of cladding
- **Behavior of fuel pins**
  - Internal pressure and fission gas release
  - Corrosion and hydriding of cladding
- **Control of reactor : boron content, gadolinium**
- **Cycle : enrichment in ^{235}U (limit at 5 %)**

Control rod bundles



- **Emergency stop**
  - **S rods AIC/B_4C**

  (AIC = Ag - In - Cd)

- **Control rods**
  - **« black »** rods AIC or AIC/B_4C (low exposure)
  - **« grey »** rods AIC/steel (high exposure)
- **Temperature regulation**
  - **R Rods AIC/B_4C**

Stainless steel cladding

Selection criteria for cladding materials ?

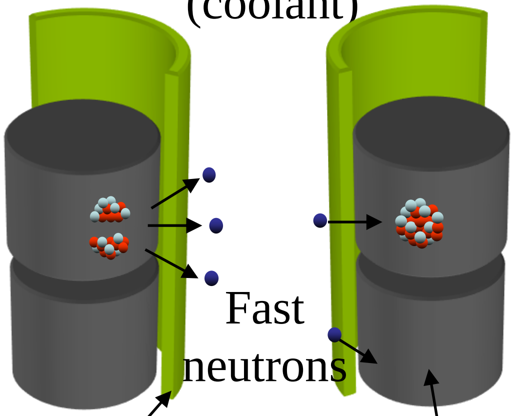
Selection criteria for cladding materials

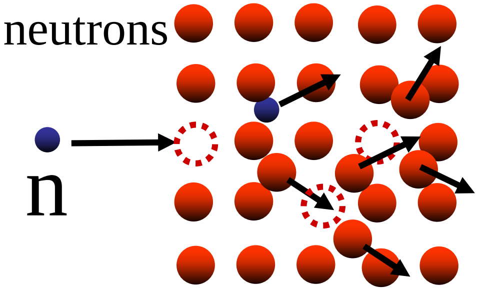
- Interaction with neutrons
  - Low capture cross section
- Easy processing
- General properties
  - Mechanical properties
    - strength, creep
  - Operating properties
    - Corrosion
    - Compatibility with coolant and fuel
      including fission products (and also reprocessing media)

Choice of Zr alloys / properties

Cladding = First containment barrier of the nuclear fuel

Used in light water reactors





Displacement cascades

1 *- High transparency to thermal neutrons,*

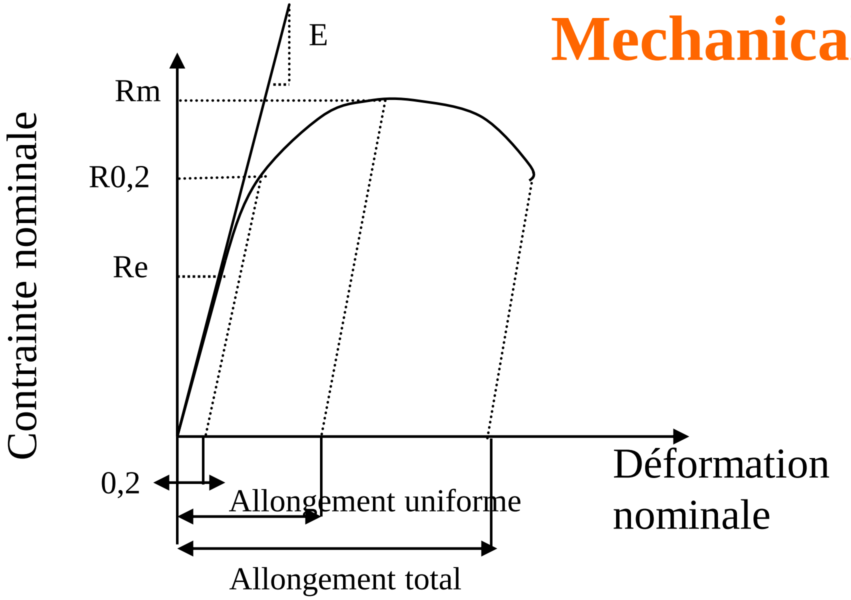
2 ***- Good mechanical properties up to 400°C (easy to process)***

3 ***- Good corrosion resistance in primary water up to ~350°C,***

4 ***- Relative stability of properties under irradiation (low activation)***

Choice of Zr alloys / properties

Mechanical properties



Pure Zr : low mechanical properties, at 20°C Re = 140MPa
Rm = 300 MPa

Need to reinforce Zr

Zr alloy (Zy-4, Zy-2, Zr-Nb-O), à 20°C Re = 240 MPa
Rm = 410 MPa, Young modulus at 20°C ~ 98000 MPa

Experimental/commercial alloys

Alloy	Typical Weight %							Commercial Applications
	Sn	Nb	Fe	Cr	Ni	O	Mo	
Zry-1	2.5							
Zry-2	1.5		0.13	0.10	0.05	0.11		BWR Clad, Structures
Zry-3A	0.25		0.25					
Zry-3B	0.5		0.4					
Zry-3C	0.5		0.2		0.2			
Zry-4	1.5		0.21	0.10		0.13		PWR Clad, Structures
ELS 0.8	0.8		0.3	0.2				PWR duplex cladding
ZIRLO®	1.0	1.0	0.1					
NSF 0.5	1.0	1.0	0.5			0.10		
NSF 0.2	1.0	1.0	0.20			0.10		
Valloy			0.15	1.2				
Ozhennite-0.5	0.2	0.1	0.1		0.1			
Scanuk	0.06	0.6	0.04	0.32			0.22	
Excel	3.5	0.8					0.8	
E110 Zr-1%Nb		1				0.06		WWER, RBMK Clad
Zr-2.5% Nb		2.5				0.12		CANDU P/T*
E 635	1.2	1	0.4			0.06		
E 125		2.5				0.06		RBMK P/T*

M5 1 0.025 0.1250 PWR (AREVA)

Zircaloy's (Sn)

Zr+Sn (solid solution) :

Zr – Nb

Zr+Nb (solid solution + precipitates) :

*(current Zy-4 : Sn ~ 1,3 % 1300 ppm O)

Where can Zr be found ?

- Large quantity in the sun, stars and moon
- On earth : 2.5 10^{-4} ≈ Zn, Cr,V
- Origin of the name "*zagrun*" : « gold color »
  - *zircon* (**si***lex* *ci***rcon***ius)*
- Zircon : $ZrSiO_4$, principal ore coming from Australia, Republic of South Africa and US
- Always mixed with Hafnium 1 - 3 %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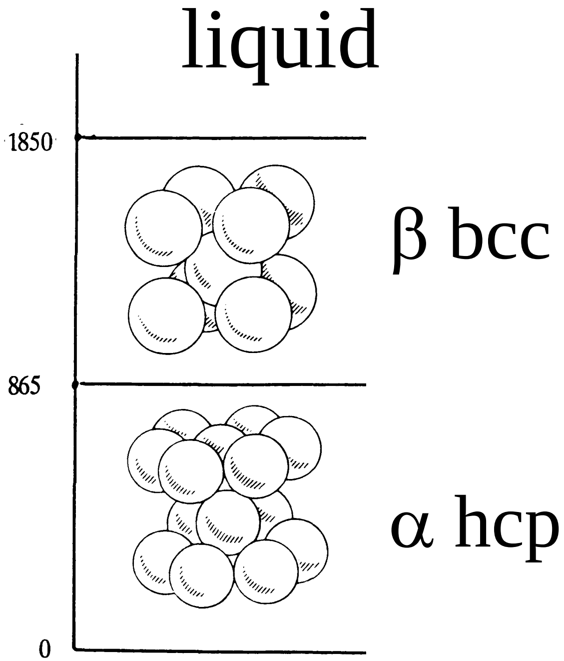
Physical Properties

- Density : 6.5 g.cm^{-3} (RT)
- Atomic weight : 91.22
- Atomic number: 40
- Transition metal 4d (IV A, Mendeleev classification)
- Melting temperature : 1852 °C
- Isotopic distribution (w %) :

Isotope	*Abundance*	σ_{abs} *(barn)*
90	51.46	0.1
91	11.23	1.58
		Incentive for isotope separation ?
92	17.11	0.25
94	17.4	0.08
96	2.8	0.1

Crystallography of pure Zr

- Room temperature
  - α hcp ($\rho_{at.}$ = 74 %)
- 865 - 1852 °C
  - β bcc ($\rho_{at.}$ = 68 %)
- 1852 °C : liquid

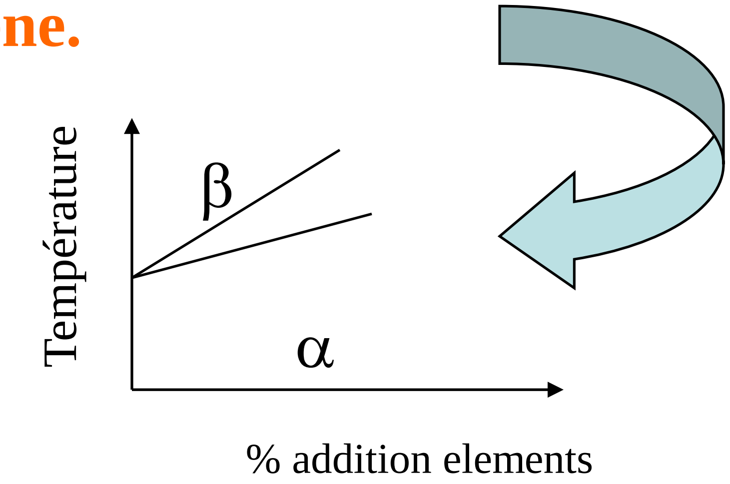


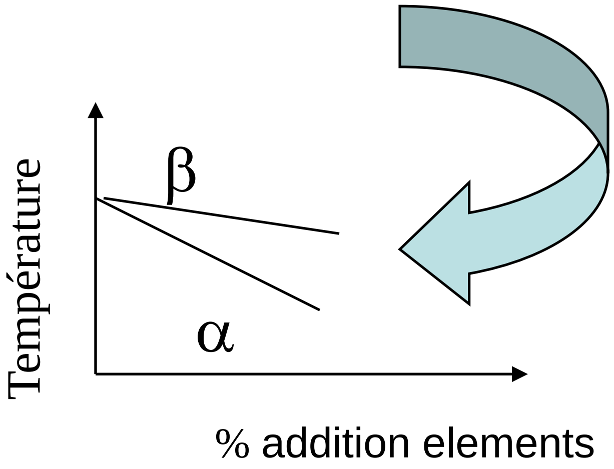
Alloying elements and phase transformations

Principal elements: iron, chromium, tin, oxygen, niobium
Two opposite effects on the phase diagram

Sn et **O**, are soluble in the α phase, increase the α phase domain and also the transformation temperatures α⇒β. We call these elements **α-gene.**

Nb, Fe, Cr, H soluble in the β phase, stabilizing the β phase and lower the phases transformation temperatures. We call these elements **β-genes.**

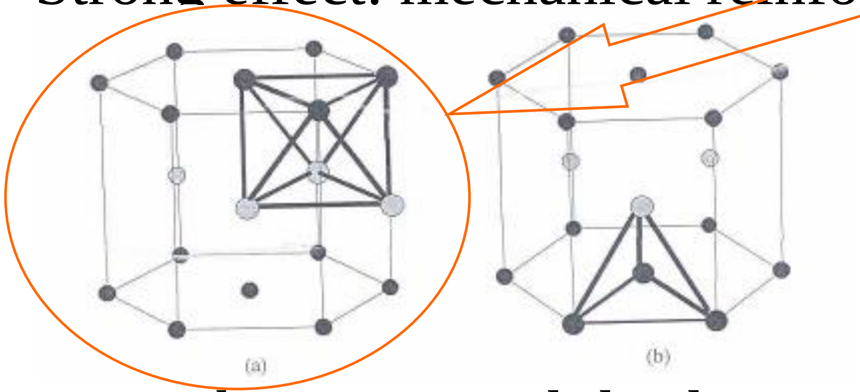




Focus on two alloys: Zy-4 and M5™ (Zr-1%Nb-O)

Role of alloying elements

O : interstitial, solid solution in the matrix (octahedral position)
Strong effect: mechanical reinforcement (interaction with dislocations)

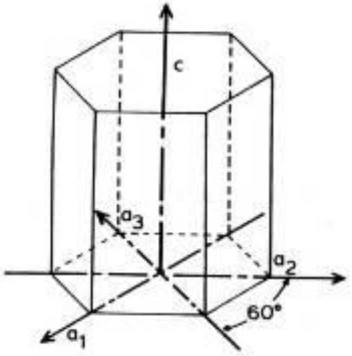


Nb : substitutional, solid solution in the matrix. Large improvement of the corrosion behavior, creep also

Sn : substitution, solid solution in the matrix. Improvement of the mechanical properties in temperature (creep) and corrosion (1,2-1,8%).

Fe et Cr : formation of precipitates (very low solubility limit, ~ 150 ppm at 850°C) intermetallics such as Zr $(Fe, Cr)_2$ et Zr(Nb, Fe) which improve the corrosion behavior and limit the grains size

About the geometry and structure of crystal



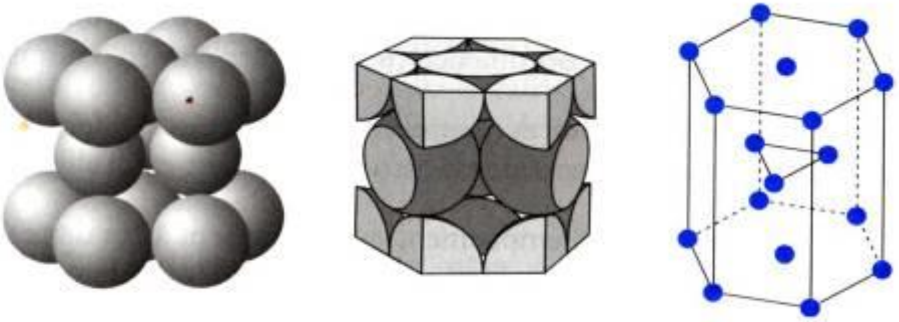
Crystal reference

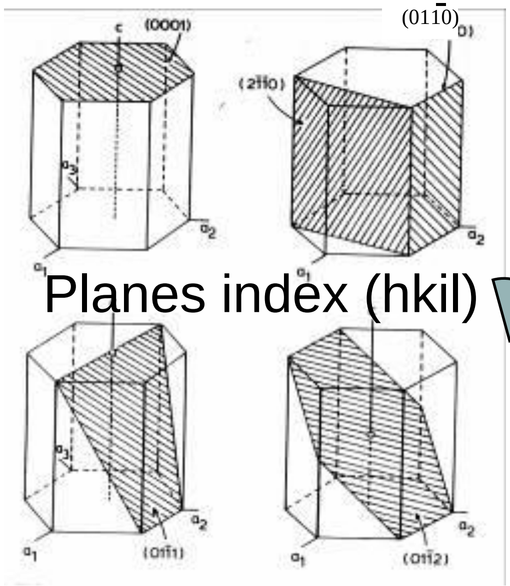
(the four-index scheme)

Inter-planar spacing is a function of cell parameters

$$\frac{1}{d^2} = \frac{4}{3}\left(\frac{h^2 + hk + k^2}{a^2}\right) + \frac{l^2}{c^2}$$

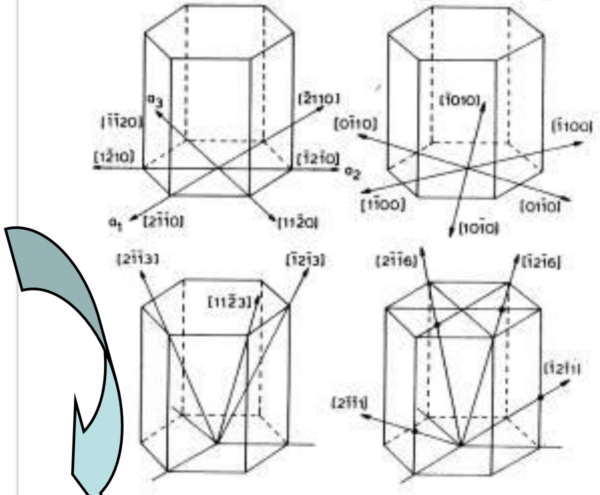
Hexagonal crystal lattice





Planes index (hkil)

Directions index [uvtw]



***i* = −*h* − *k*.**

Here h, k and l are identical to the Miller index, and i is a redundant index.

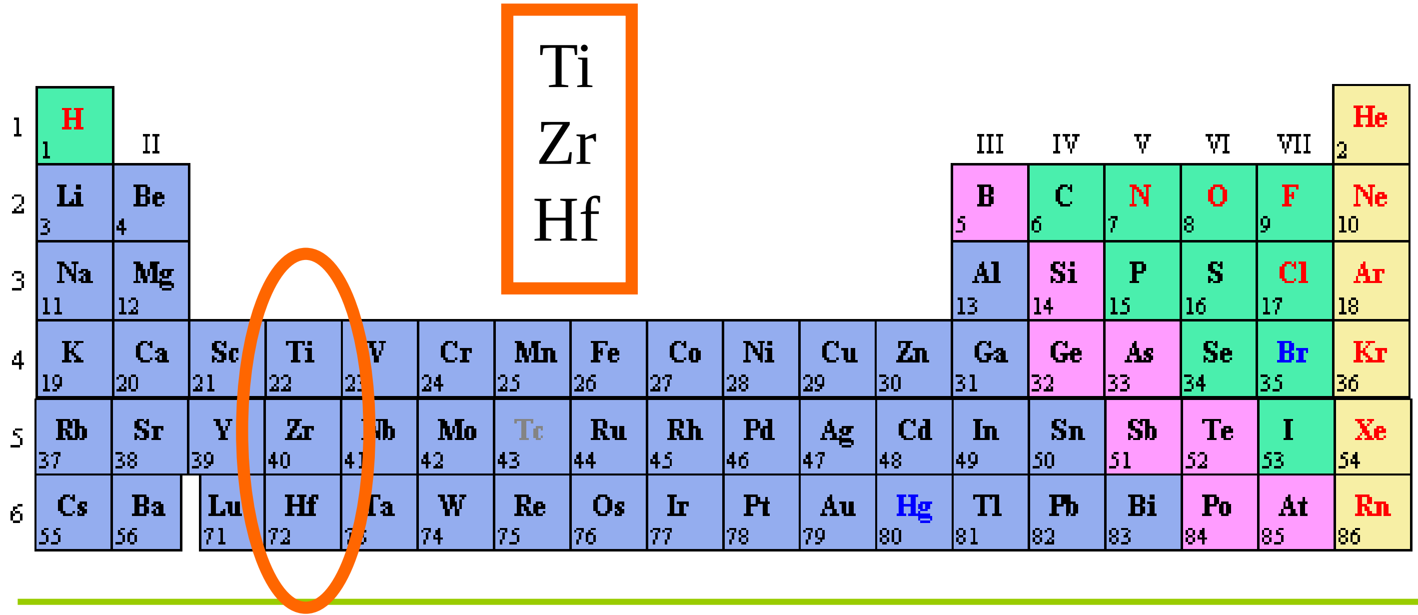
Zr Elastic anisotropy

- The elastic constants are dependent of crystallographic orientation, but are only weakly anisotropic.
- $E_{[10.0]}$= 99100 MPa
- $E_{[0001]}$= 125300 MPa
- Minimum at 52° from [0001] : 80 000 MPa
- Standard plates and tubes, E_{macro} = 96 000 MPa ± 2%

Also anisotropic thermal expansion of Zr (higher in <c> direction)

Zr nuclear properties

- Very small absorption cross section of thermal neutrons :
  - σ_{abs} : 0.185 barn (10^{-24} cm^2) at 25 meV
  - σ_{scat} : 8 barn *(elastic diffusion)*
- Hf chemically very near to Zr
- But Hafnium : σ_{abs} : 105 barn !!!



Ti
Zr
Hf

		II											III	IV	V	VI	VII	
1	H 1																	He 2
2	Li 3	Be 4											B 5	C 6	N 7	O 8	F 9	Ne 10
3	Na 11	Mg 12											Al 13	Si 14	P 15	S 16	Cl 17	Ar 18
4	K 19	Ca 20	Sc 21	Ti 22	V 2	Cr 24	Mn 25	Fe 26	Co 27	Ni 28	Cu 29	Zn 30	Ga 31	Ge 32	As 33	Se 34	Br 35	Kr 36
5	Rb 37	Sr 38	Y 39	Zr 40	Nb 41	Mo 42	Tc 43	Ru 44	Rh 45	Pd 46	Ag 47	Cd 48	In 49	Sn 50	Sb 51	Te 52	I 53	Xe 54
6	Cs 55	Ba 56	Lu 71	Hf 72	Ta 7	W 74	Re 75	Os 76	Ir 77	Pt 78	Au 79	Hg 80	Tl 81	Pb 82	Bi 83	Po 84	At 85	Rn 86

Consequence: avoid Hf

- Separation of Hf/Zr is imperative for every nuclear reactor practical application
- Not required to use Zr without Hf for industrial chemical application
- To avoid any error, Zr alloys manufacturers propose Zr without Hf whatever application is considered

Zr chemical purification

First step Zr chloration

- $ZrCl_4$ fabrication

  $ZrSiO_4$ (+ HfO_2+SiO_2) + 2C + $2Cl_2$ => $ZrCl_4$ ($HfCl_4$ + $SiCl_4$) + 2 CO

  - T = 1200 °C (current process)
    - Fluidized bed by Cl_2

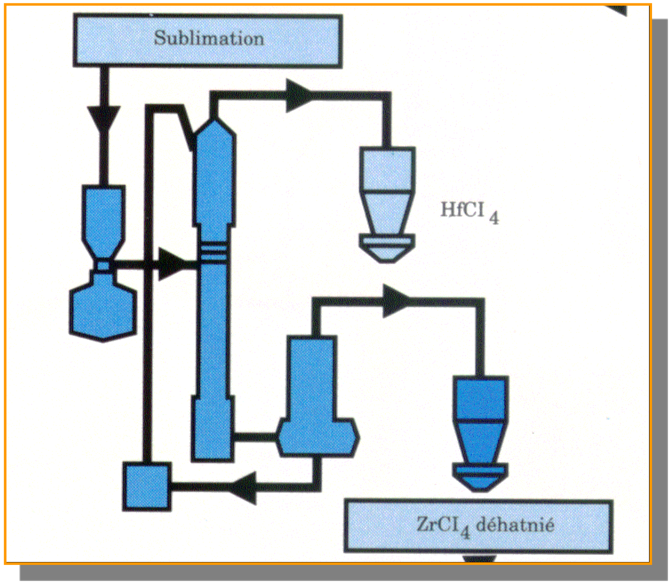
      separation of $(Zr,Hf)Cl_4$ from other chlorides by two step condensation

- $(Zr,Hf)Cl_4$ compound for hafnium separation

Hf purification process

Second step separation ZrCl4/HfCl4

- Extractive distillation within a mixture of KCl and $AlCl_3$
  - – upper level : $HfCl_4$
  - – lower level : $ZrCl_4$
- P = 1 atm, T = 350°C
- 50 m height tower (Jarrie, close to Grenoble)
- 6 800 t/y $ZrCl_4$ for 2000 t/y sponge Zr
- Low waste levels (250 kg/t_{Zr})
- [Hf] < 50 ppm



CEZUS AREVA

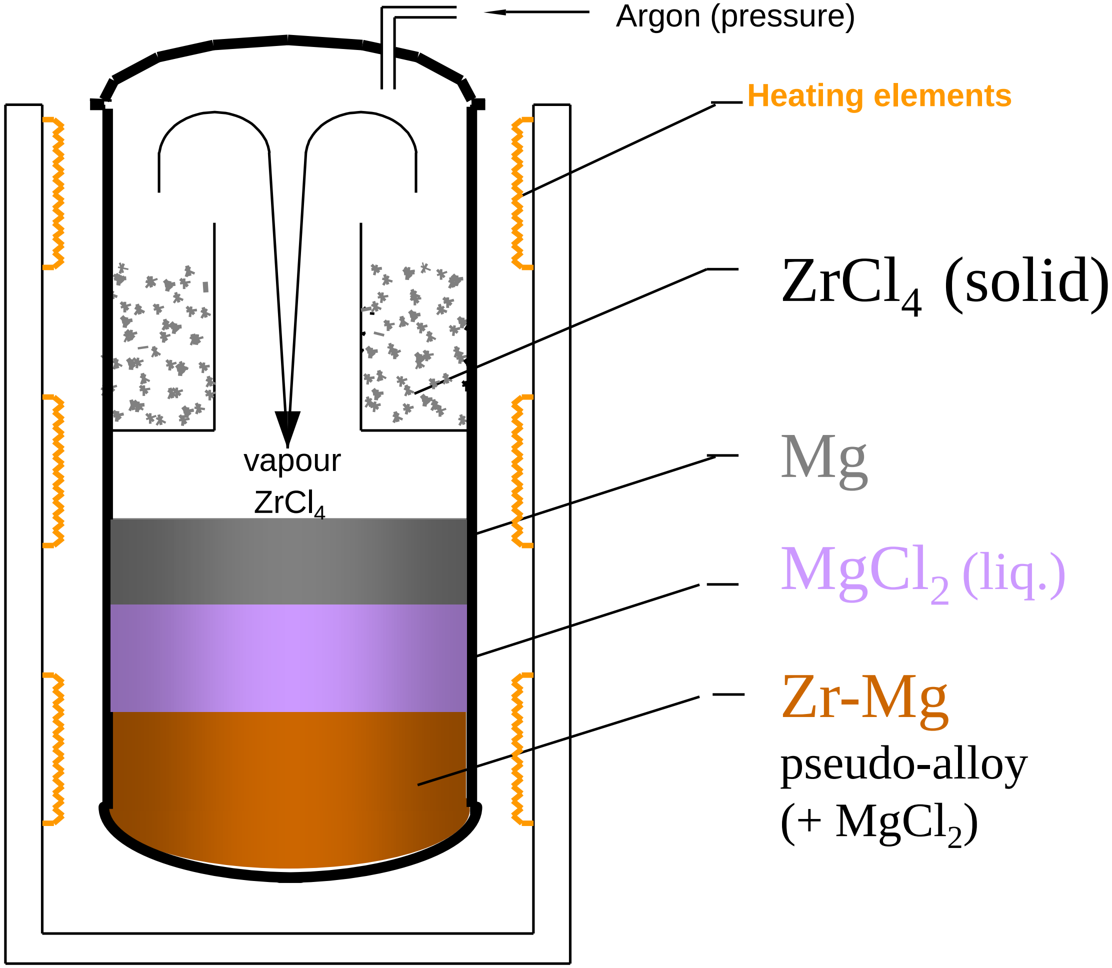
Reduction to the metal

Third step: extractive distillation and reduction

- Reduction of gaseous $ZrCl_4$ by liquid Mg
- $ZrCl_4 + 2\ Mg \Rightarrow 2\ MgCl_2 + Zr$ (+ 39 kcal/mol)
- Chemical reaction with liquid Mg and formation of Zr metal
- Residual Mg is confined in Zr sponge
- Distillation to remove remaining chlorides from the sponge (1225K)
- Crunching Zr sponge

Kroll's reaction

- T = 800 - 920 °C
- Protective atmosphere
- Distillation of remaining Mg and volatile products under vacuum at 1000°C
- Sponge cake crushing



Sponge Zr cake

Typical mass = 3 T

Mechanical Crushing

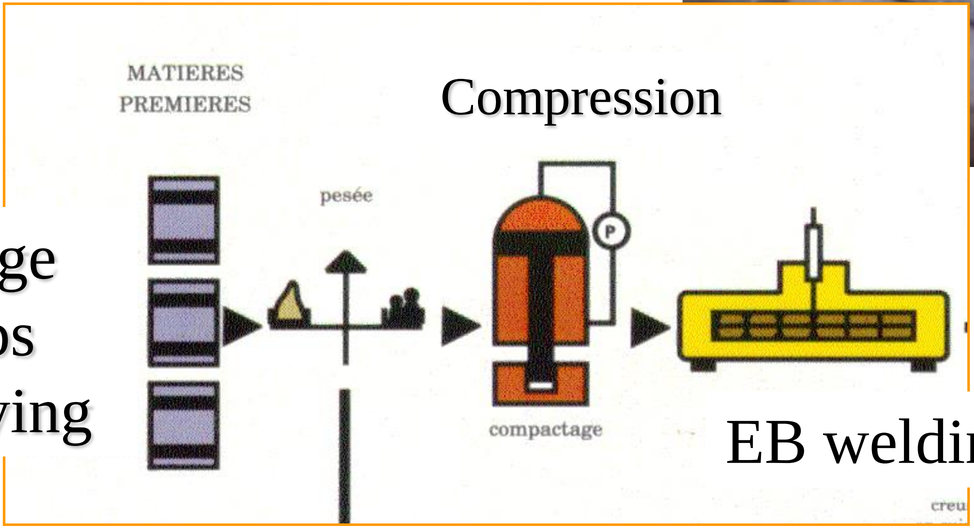
100 % manual selection of the all the pieces

Chemical analysis
ASTM B 349

Melting of the alloys

Compacting and alloy composition adjustments

Sn Fe, Cr, O
Nb, O



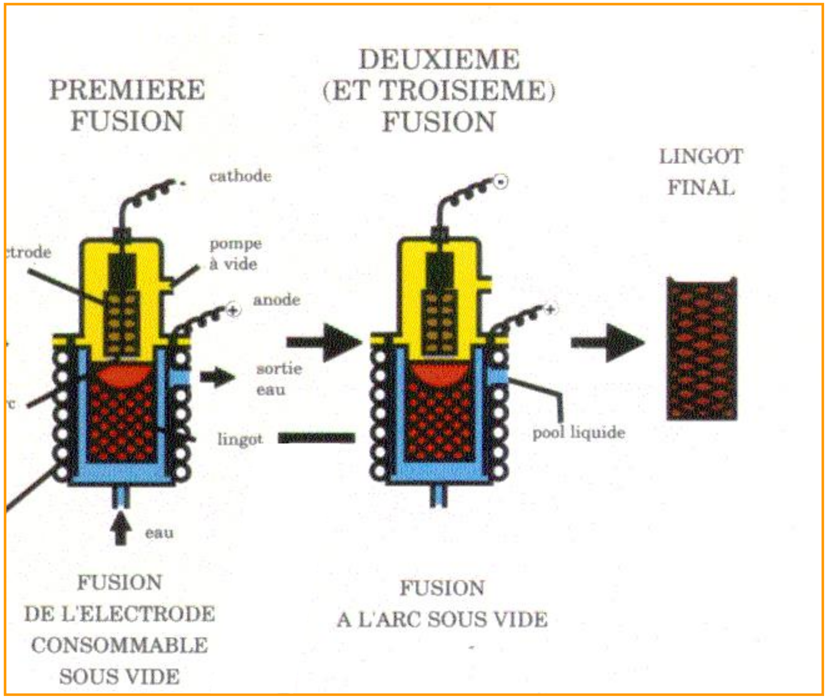
Compression

Sponge
Scraps
Alloying

EB welding

Vacuum melting consumable arc (3, 4 times)

Vacuum arc remelting : arc with a consumable electrode under vacuum, specific for reactive metals (Zr, Ti) and super alloys

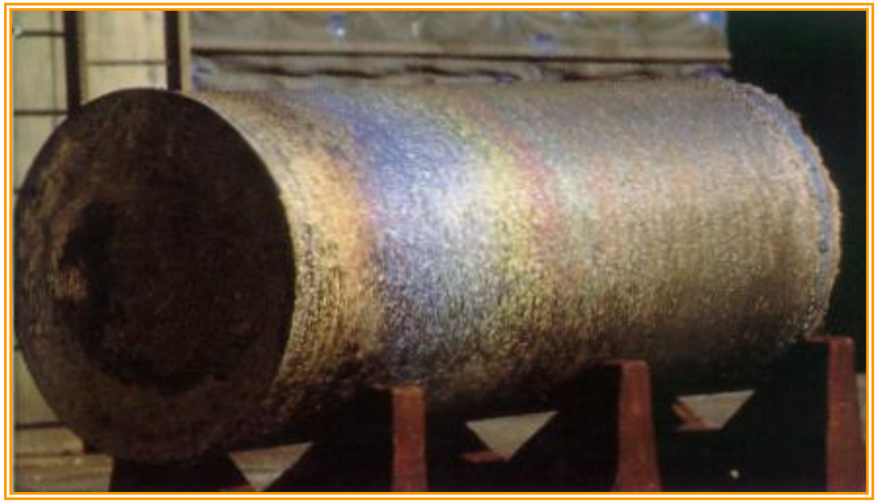


- Cu cooled crucible with water
- Ingots are used as electrodes for the following melts
- Final
  - ∅ ≈ 600 –800 mm, length 2-3 m
  - mass 4 - 8 t

Ugine, Savoie, France

Final ingot

- ASTM B 350 : Specification of Zr and Zr alloy ingots for nuclear applications
- Chemical analysis
- US testing for internal cracks



Hot Forging : 1000-1300K

Hydraulic press

Upper α or $\alpha + \beta$ or lower β range

Quenching:

Heating in the β phase (typically above 1273K)

Homogenizing the alloying elements

Partly erase the texture

Cooling to room temperature

Back to α phase

Controlling the cooling rate

Starting point for the SPP size control

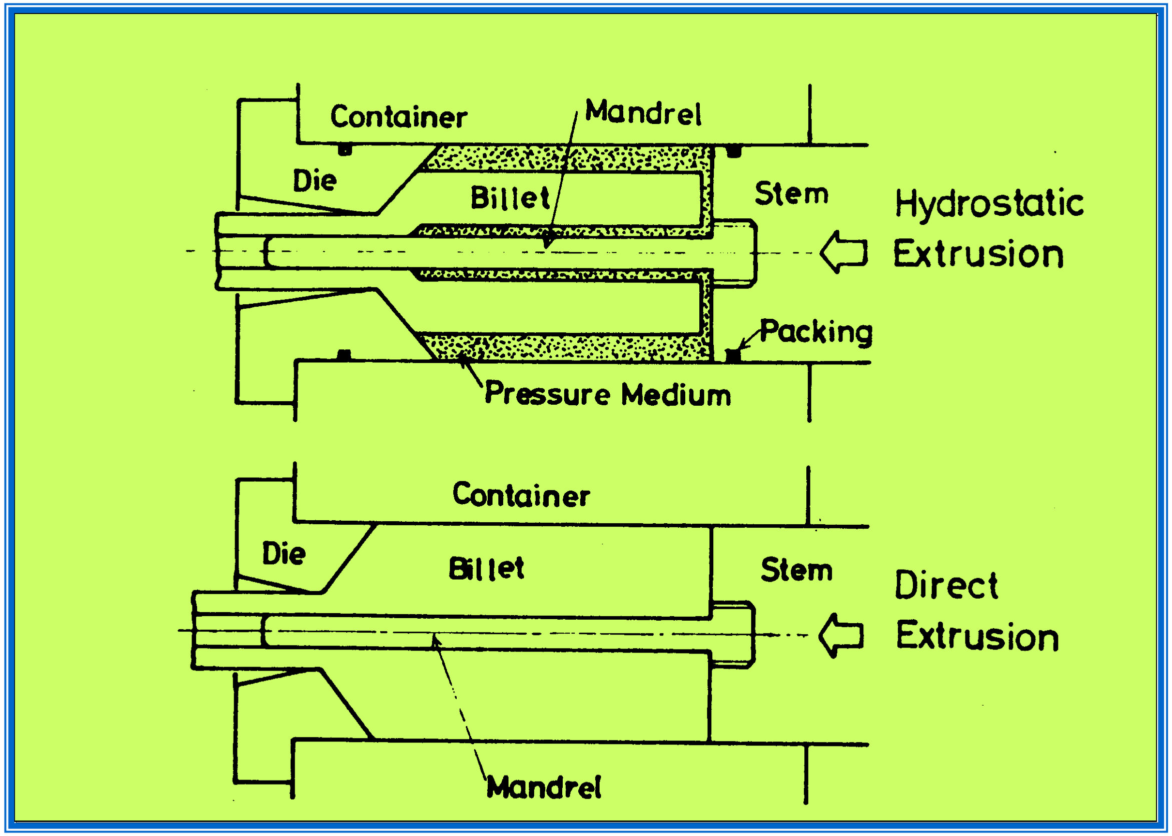
After the Quench, all the annealing in the α phase

Low strain rate: press forging

Final dimensions : diameter: 100-250 mm for billets, thickness 100 mm for slabs

Standard extrusion process

cladding and guide tube process



Extrusion: first step

Hole drilling

Cleaning, chamfering

Coating with lubricant or copper

Heating at 850-1000K

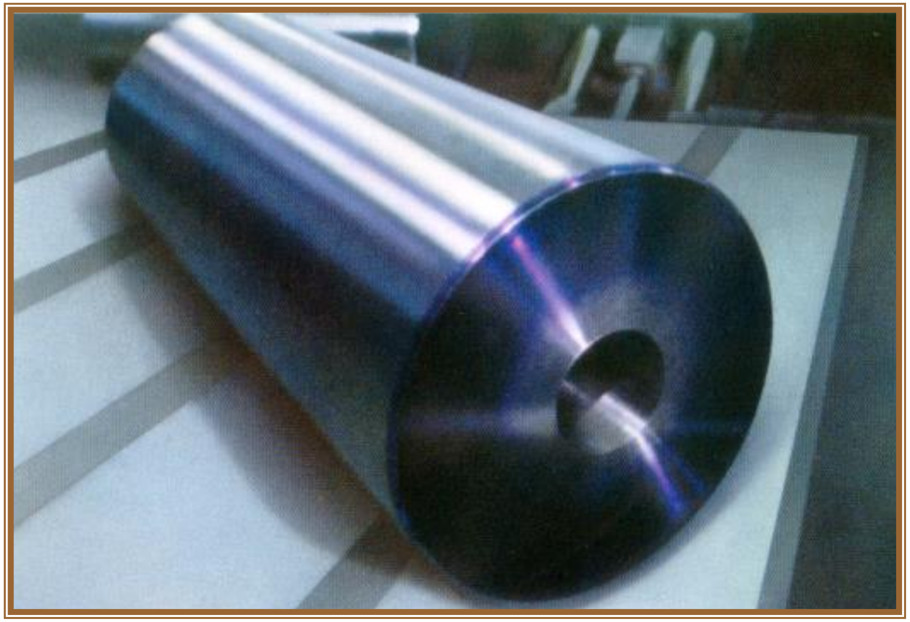
Output diam. 50-80 mm, thickness 15-20 mm

Extruded tube or shell

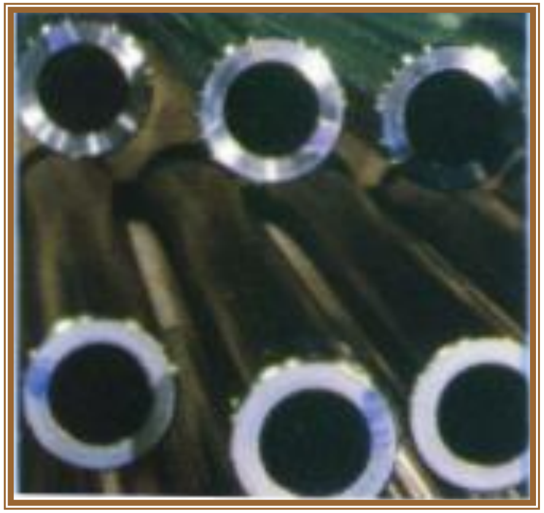
Annealing

The pre-heated billet is pushed between the die and the mandrel

Extrusion billet and extruded tube



Ready for pressing



After pressing
and cleaning

Cold pilgering

Reduction of the diameter and the thickness

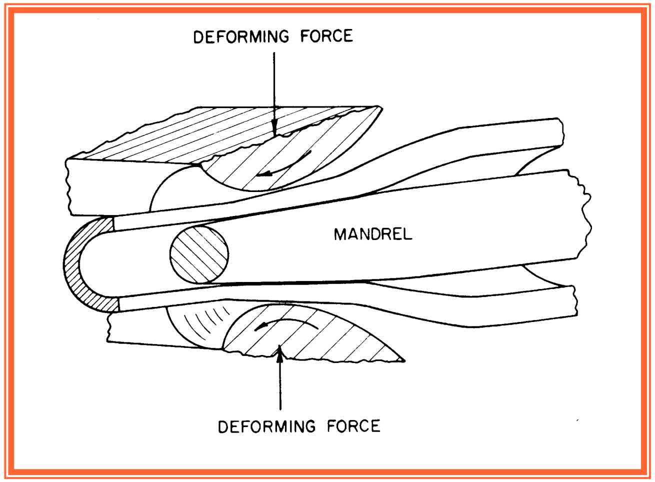
Two types of pilger mills : High Precision Tube Reduction (HPTR) and Vertical Mass Ring (VMR)

4 to 5 cycles cold pilgering and annealing

After extrusion + annealing

- Cold rolling: 63.5x10.9 SHELL
- TTH 850-1000K
- Cold rolling: 44.4x7.62 TREX
- Using VMR deformation up to 80% achieved (3 steps)

Rolling procedures



- Cold pilger - rolling process
  - grooved rolls
  - tapered mandrel
  - rocking
- Parameters
  - reduction in diameter
  - reduction in thickness

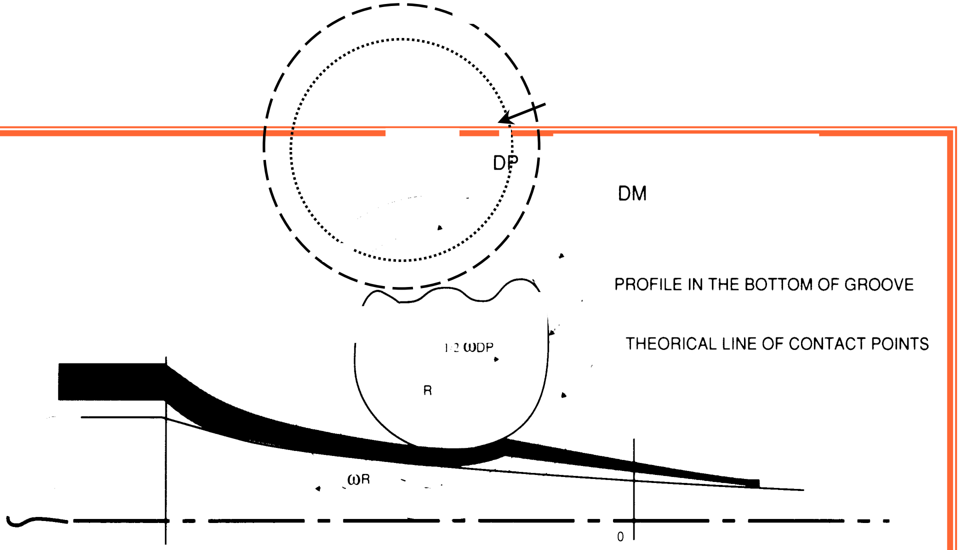
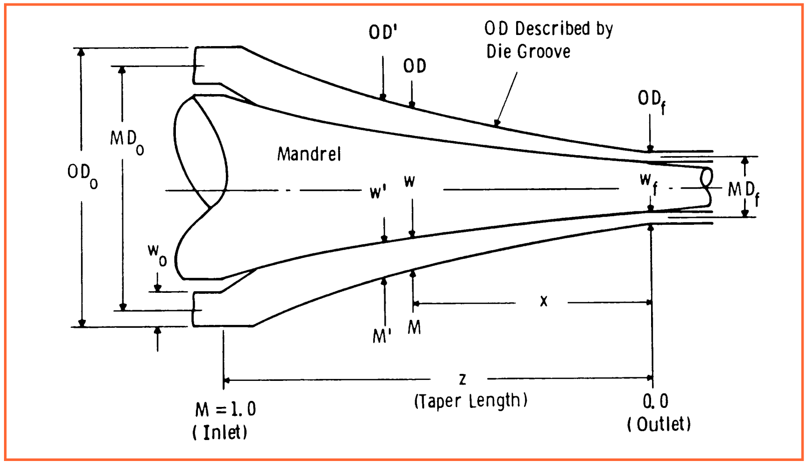
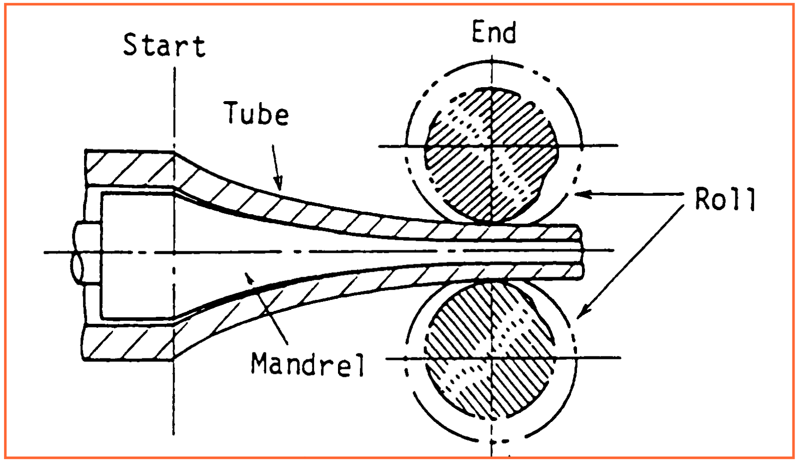


FIG. 2—*Slip factor influence.*

Pilger rolling scheme





Design parameters

Process parameters

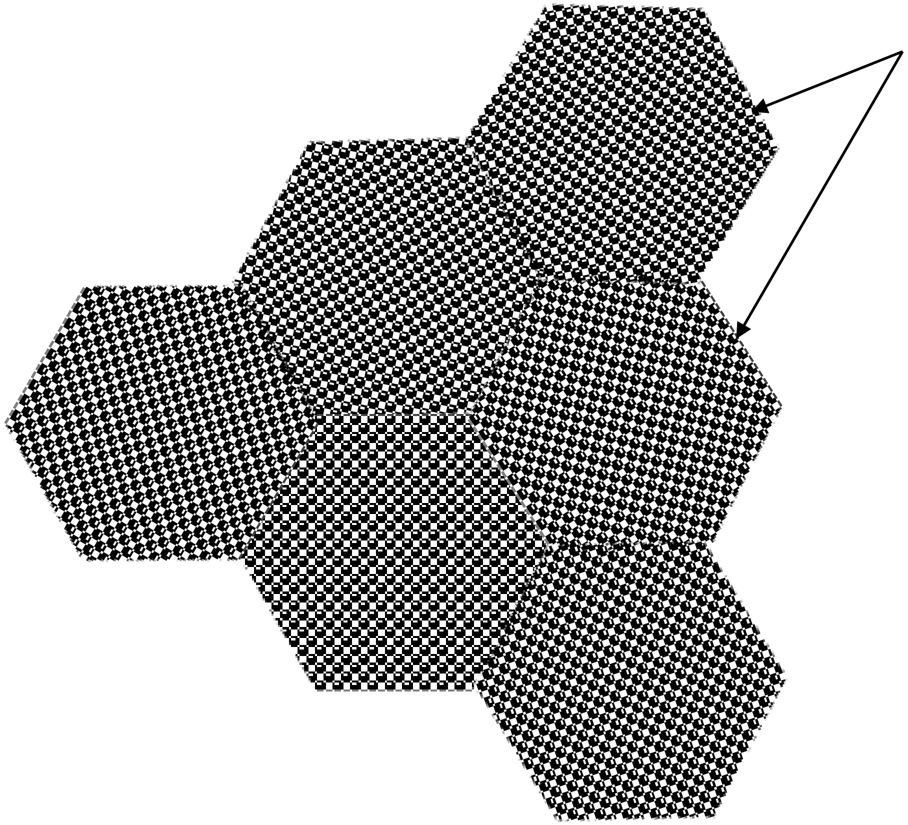
advance increments

rotation increments

EJC2019 école Joliot-Curie school

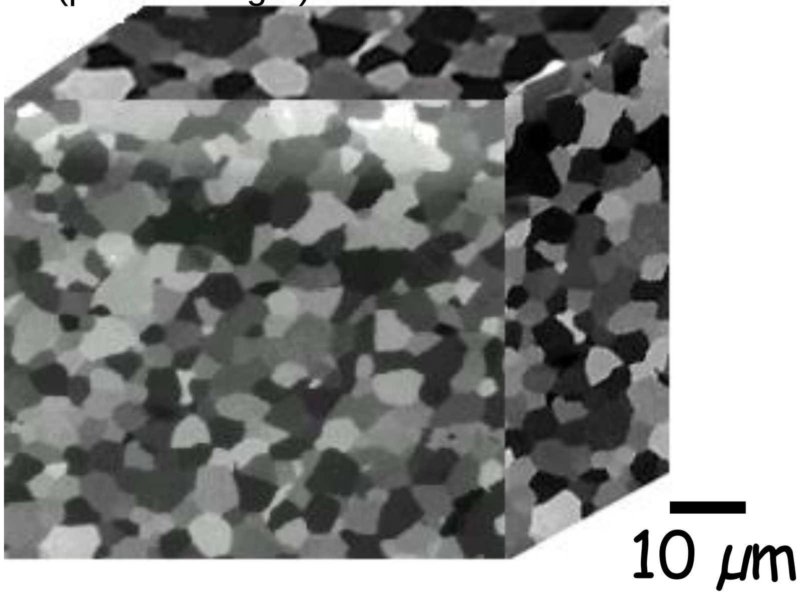
During the fabrication process => texture is created (preferred orientation of cristallographic planes)

Especially for Zr alloys with hcp structure



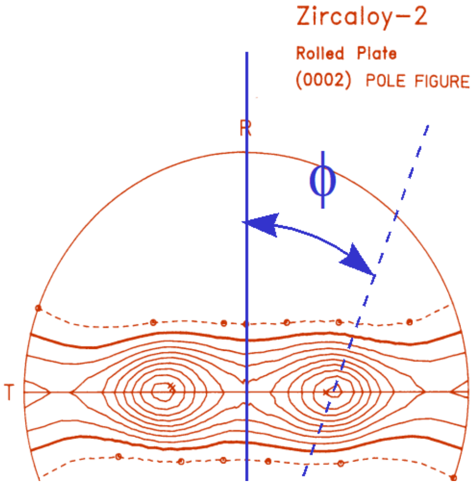
Grains with different orientation (atomic planes are not organized in a same manner)

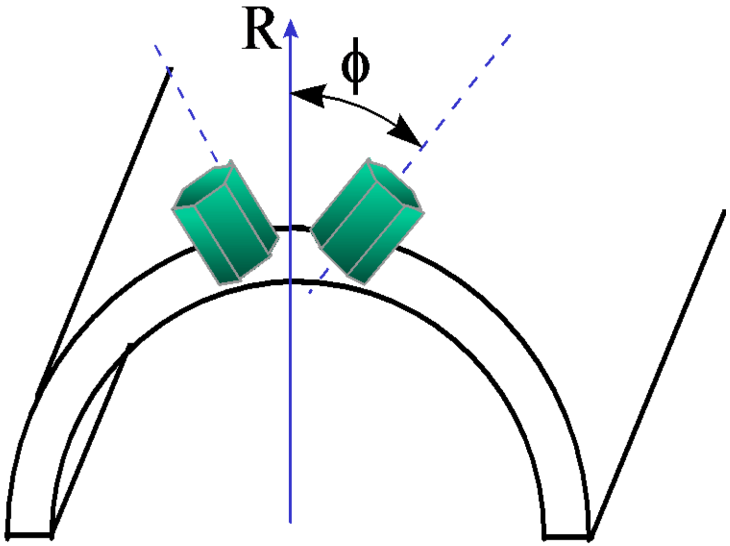
Contrast is linked to crystallographic orientation (polarized light)



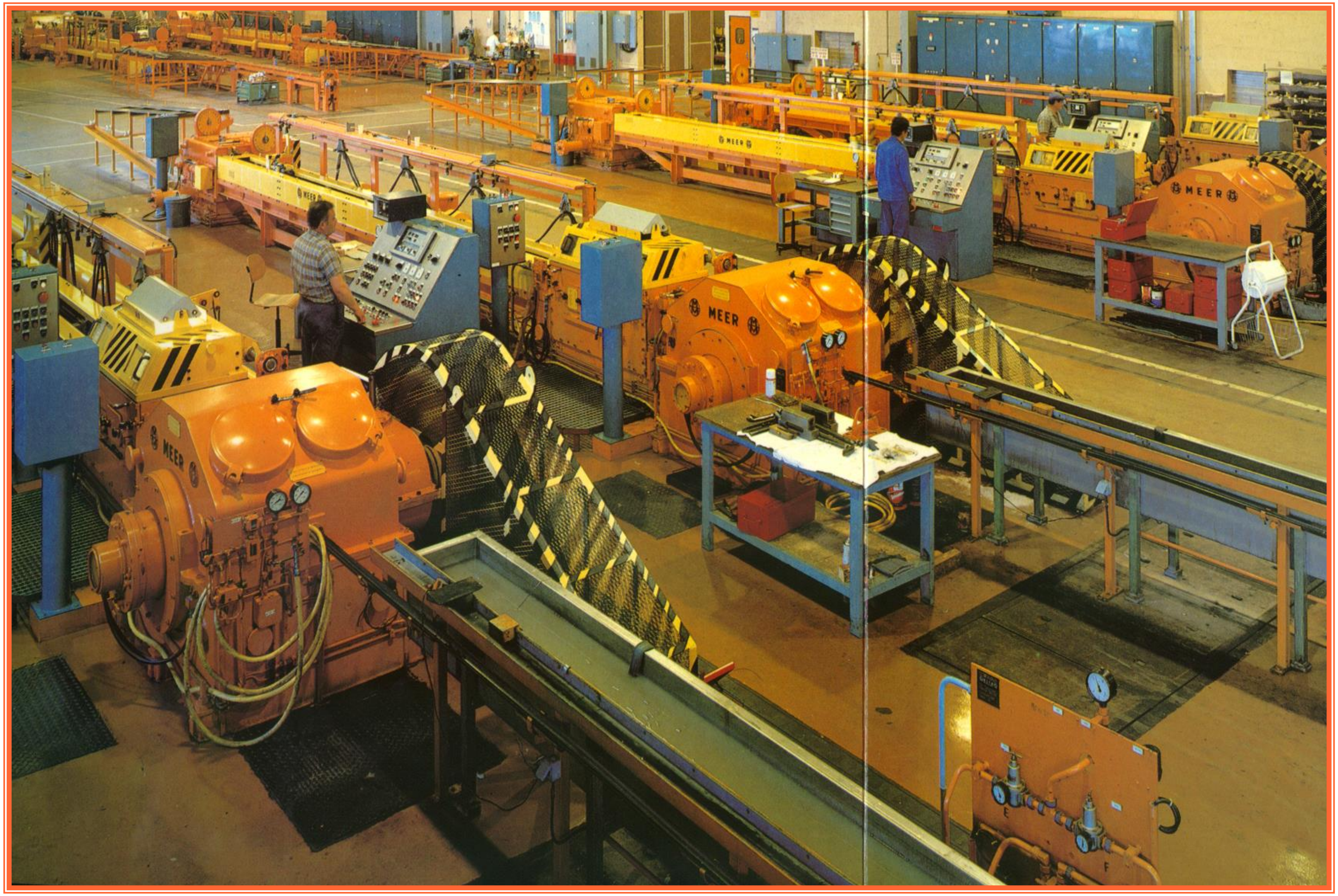
Polycristal (Zr cladding)

Typical texture of cladding tubes





Pilger rolling equipment



Annealing

- ➢Under vacuum or inert gas
- ➢Static furnace (batch) or continuous furnace
- ➢In α phase
- ➢Intermediate annealing to recrystallize the product (enables further deformation)
- ➢Final annealing
  - •SRA: stress relieved anneal
  - •RXA : recrystallized anneal
- ➢Temperature is controlled to a few degrees
  - •475-750 °C
  - •1-10h

Metallurgical state and microstructure

Evolution for fuel assembly: increase the burnup => stay a longer time in PWR core => limit the oxide growth

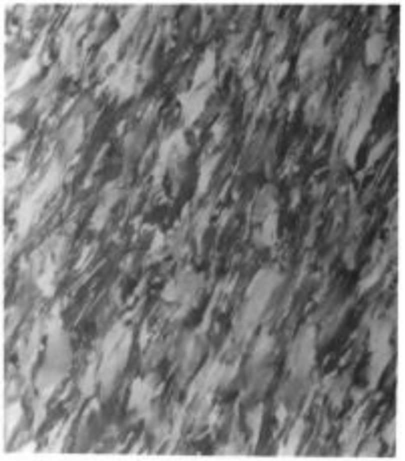
Zy-4 cladding **SRA** → M5 cladding **RXA**

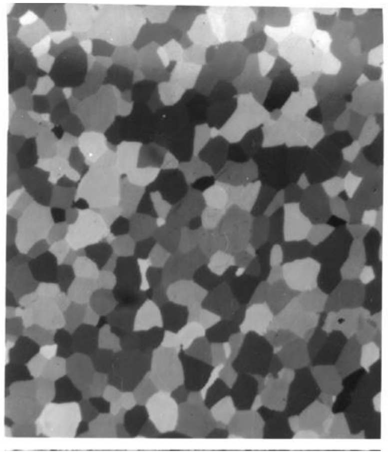
(high yield stress)

(slower oxide growth, good mechanical properties)

SRA

Elongated grains along the rolling direction : 2 x 15 µm²





RXA

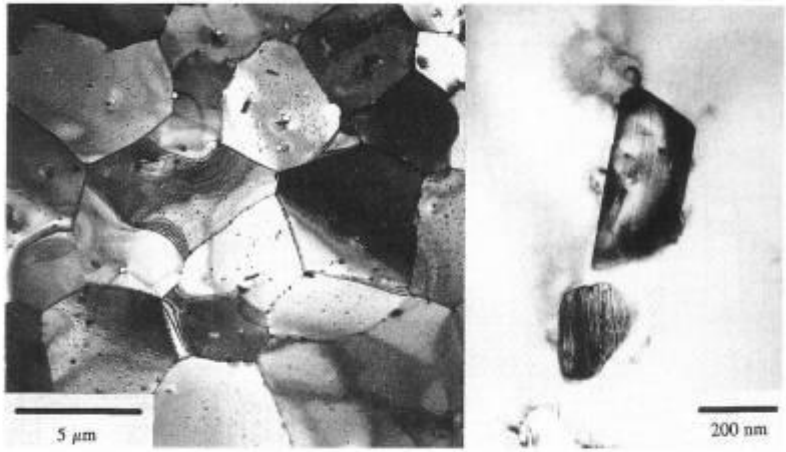
Equiaxed grains from 5 to 10 µm

guide tube et grids RXA
Zy-4 → M5 recrystallized

Precipitates

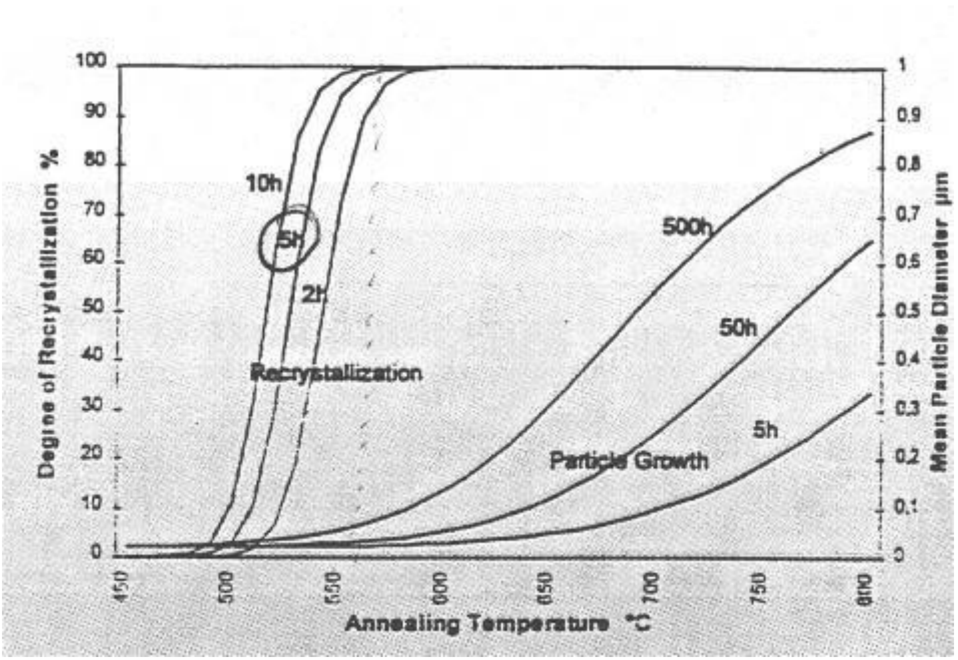
Need to control the precipitation (size, density, composition)
=> large influence on the behavior of Zr alloys (corrosion)

Typical distribution of intragranular precipitates $Zr(Fe,Cr)_2$



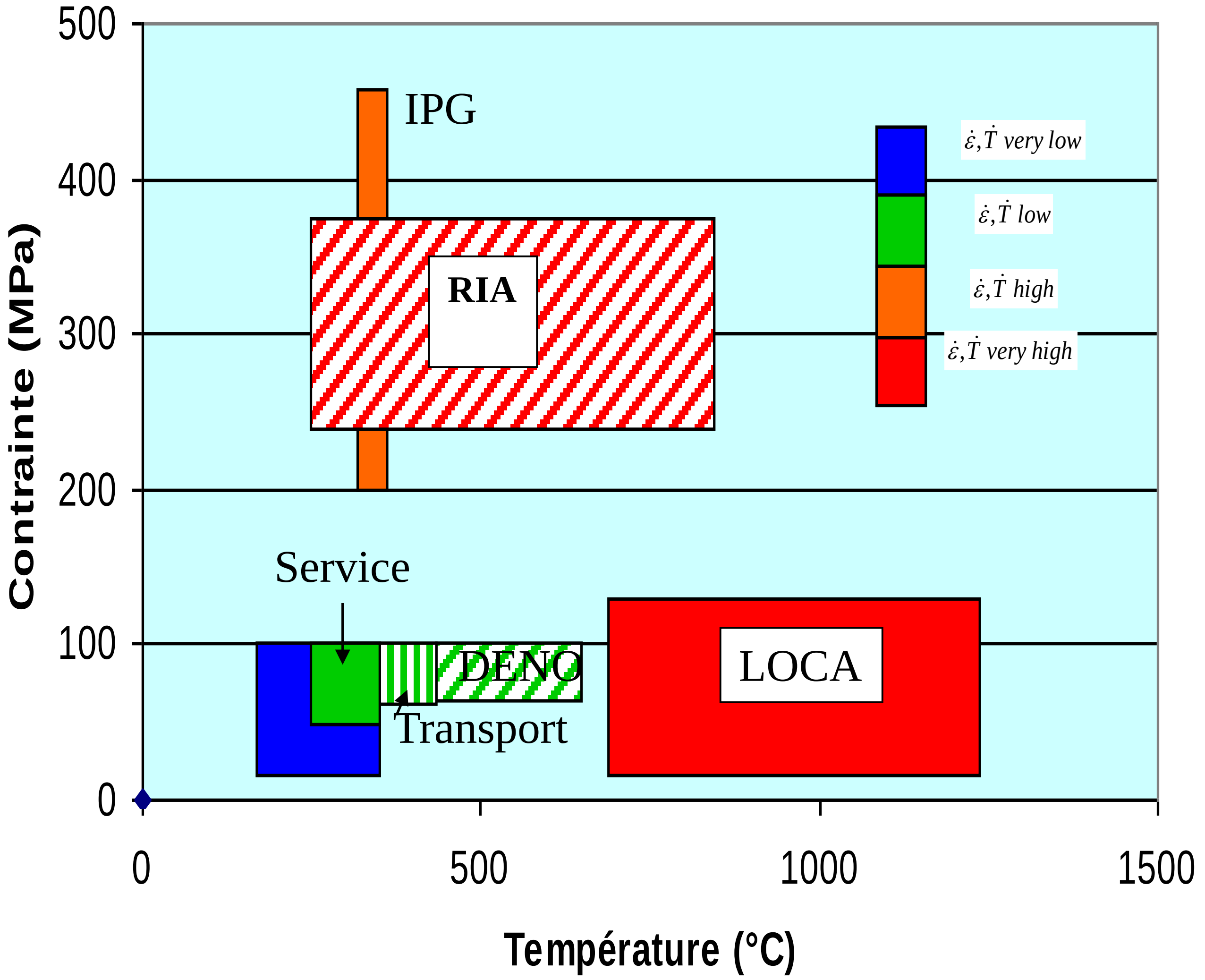
RXA Zy-4alloys

(slowdown of corrosion attributed to precipitates enriched with iron and chromium)



Influence of thermal treatment on the precipitate growth

PWR cladding – areas to explore



Cladding behavior

Under normal operating conditions the cladding is under external stress (155 bar), internal stress (30 to few hundreds MPa), irradiation, corrosive environment (inside and outside)

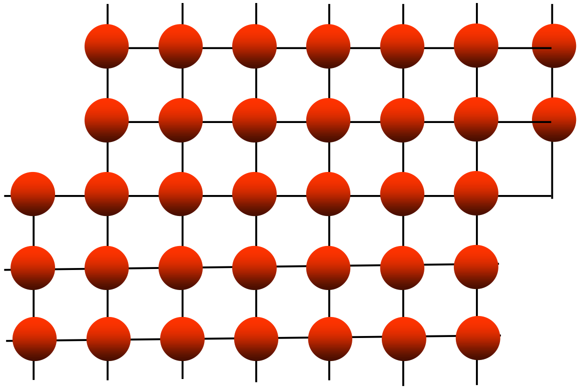
Hoop deformation of the cladding :

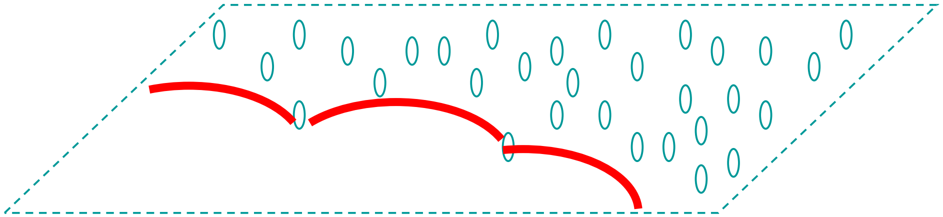
- Decrease in diameter (creep down) during 1st et 2nd cycle
  under water pressure
- Closing of the gap between 1 and 2 cycles

- Increase of diameter at high burn ups
  swelling of the fuel
- Modification of mechanical properties

Irradiation induced hardening

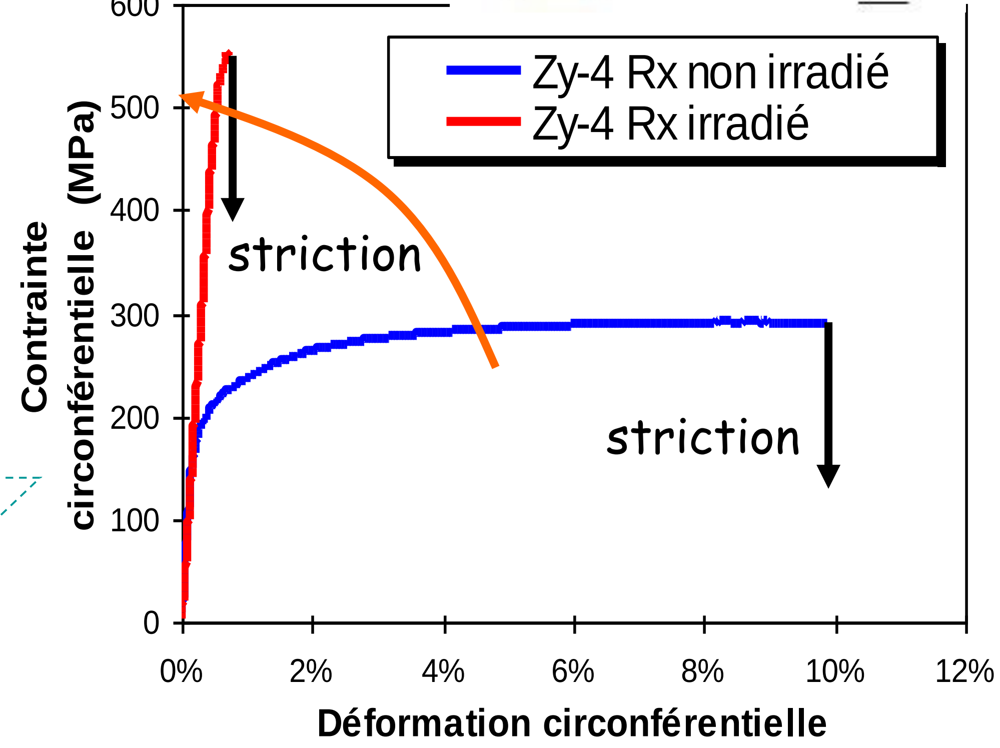
Dislocation sliding





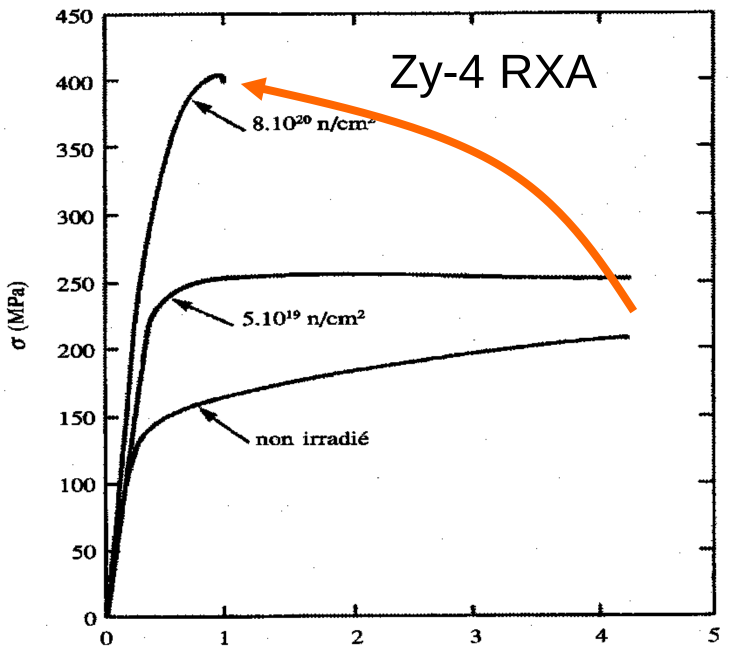
-> dislocation pinning on loops

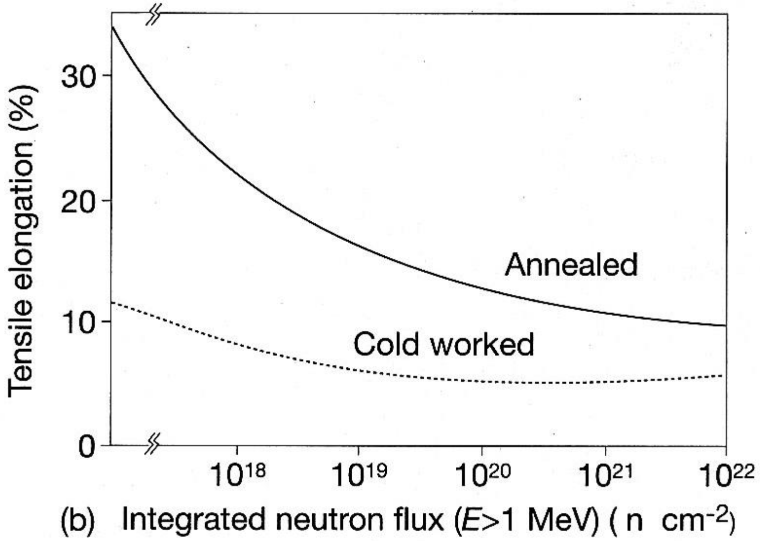
Internal pressure at 350°C

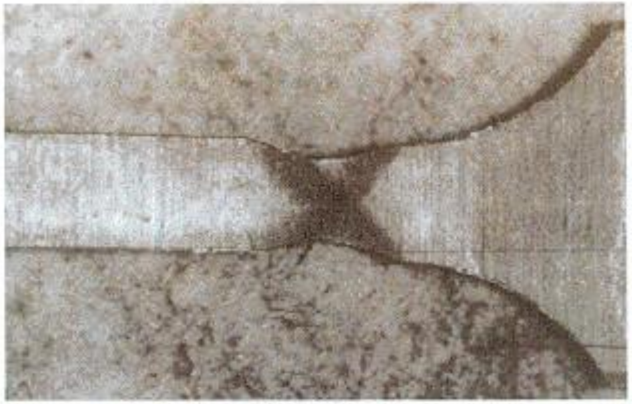


- Irradiation induced hardening
- Decrease of homogeneous elongation (macroscopic ductility)

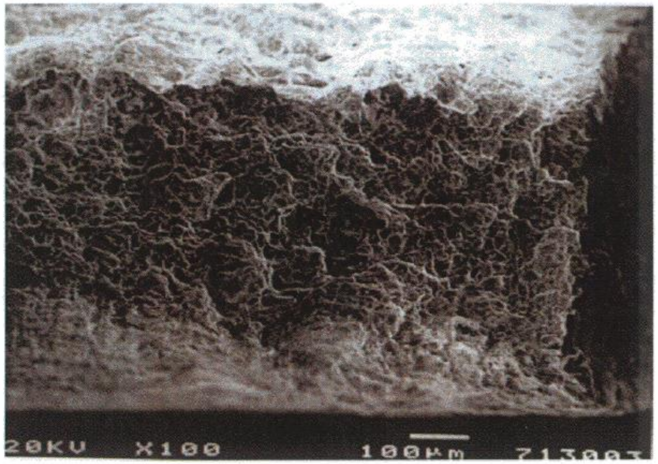
Reduction of the uniform elongation







-> Early necking of the specimen

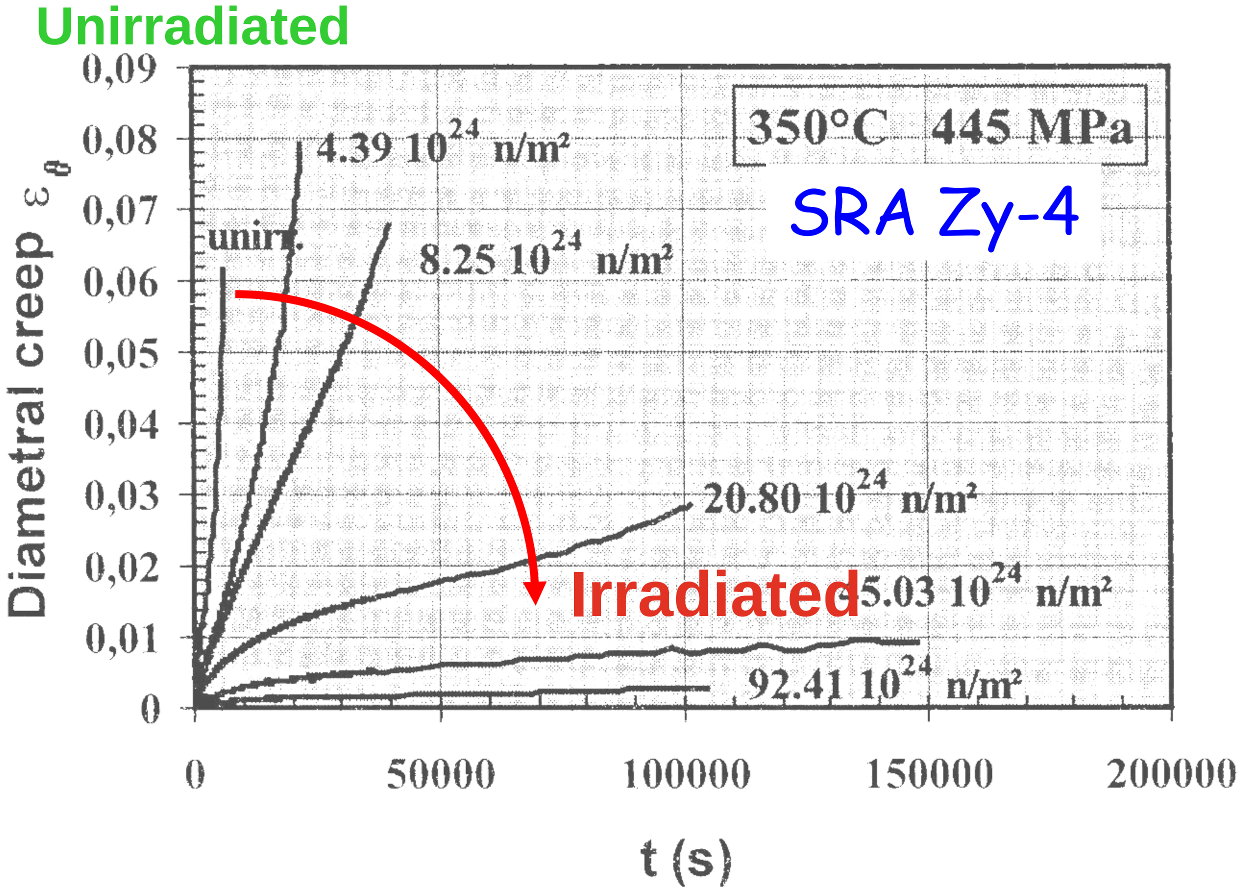


-> Ductile failure mode

-> Early localization of the deformation at the specimen scale

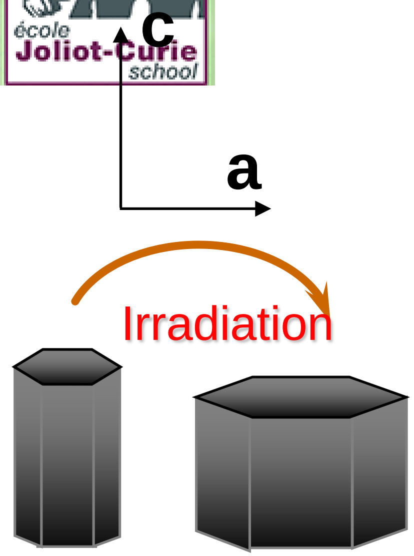
Effect of irradiation on creep behavior

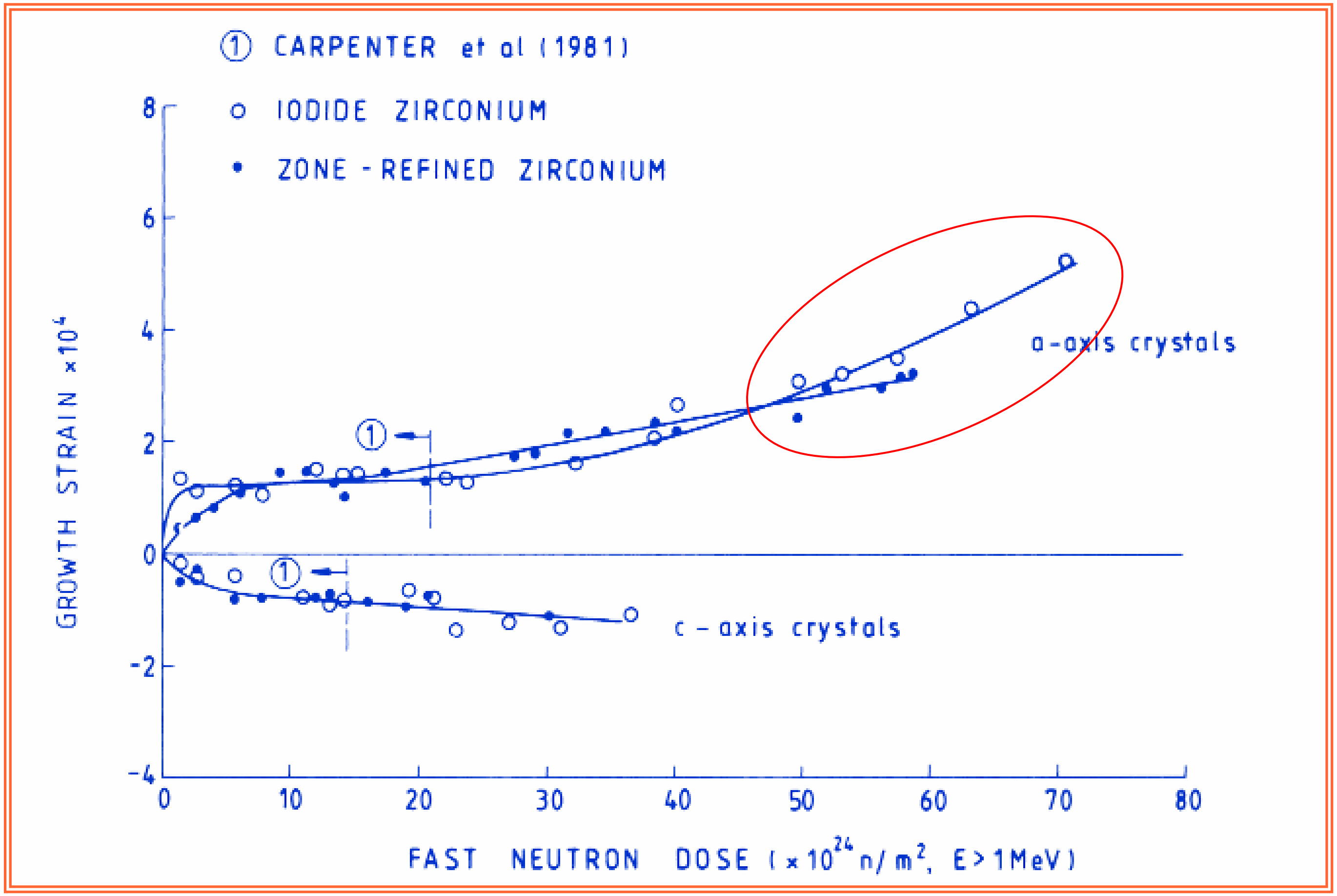
(post-irradiation creep)



Creep rate decreases with irradiation dose due to irradiation induced hardening

Growth of Zr single crystal

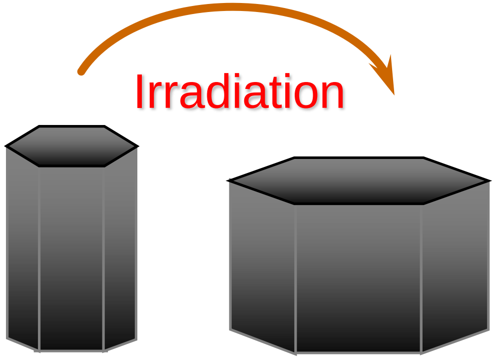


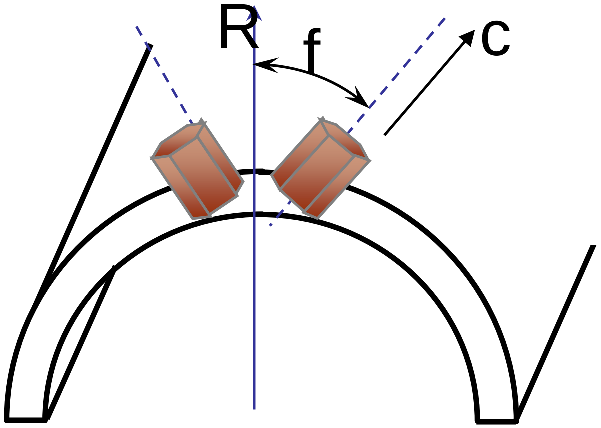


- Elongation in the <**a**> direction, shortening in the **<c>** direction (constant volume)
- acceleration of the growth for large doses (fluences)

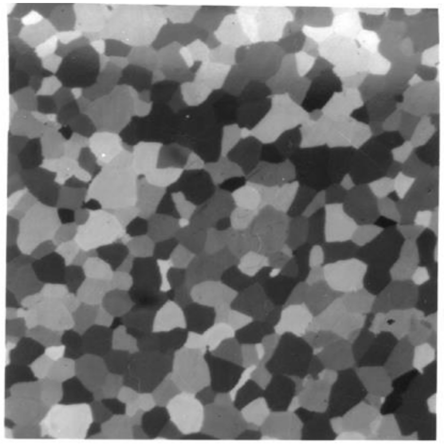
Growth under irradiation of the polycrystalline cladding

Specific to anisotropic materials with a strong crystallographic texture





RXA material

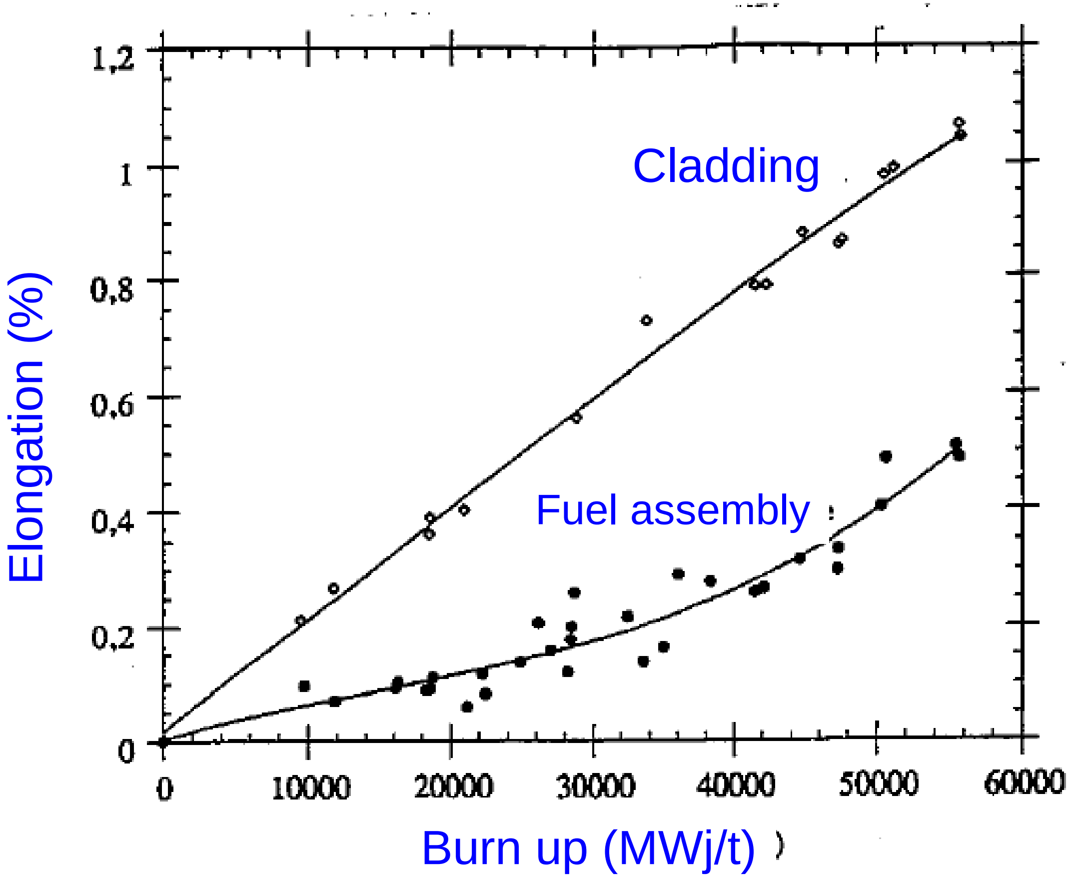


Strong texture :

- <c> axis close to the radial direction (r)
- The axial direction (z) is close to a <a> direction

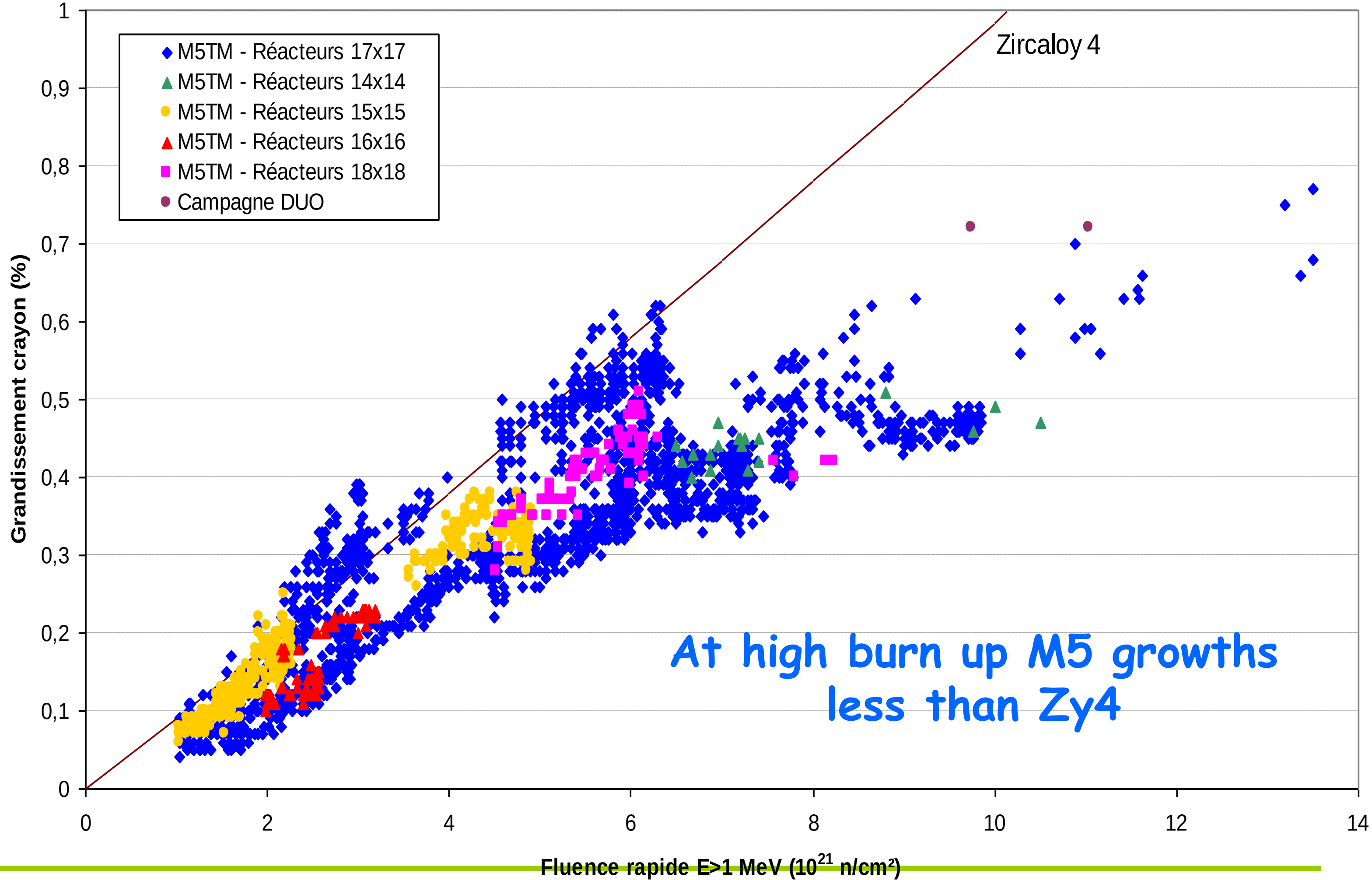
Irradiation -> Growth in the <a> direction -> Elongation of the tube

Growth of the polycrystalline cladding



In-reactor growth of the cladding (SRA Zy-4) and in-reactor growth of the fuel assembly.

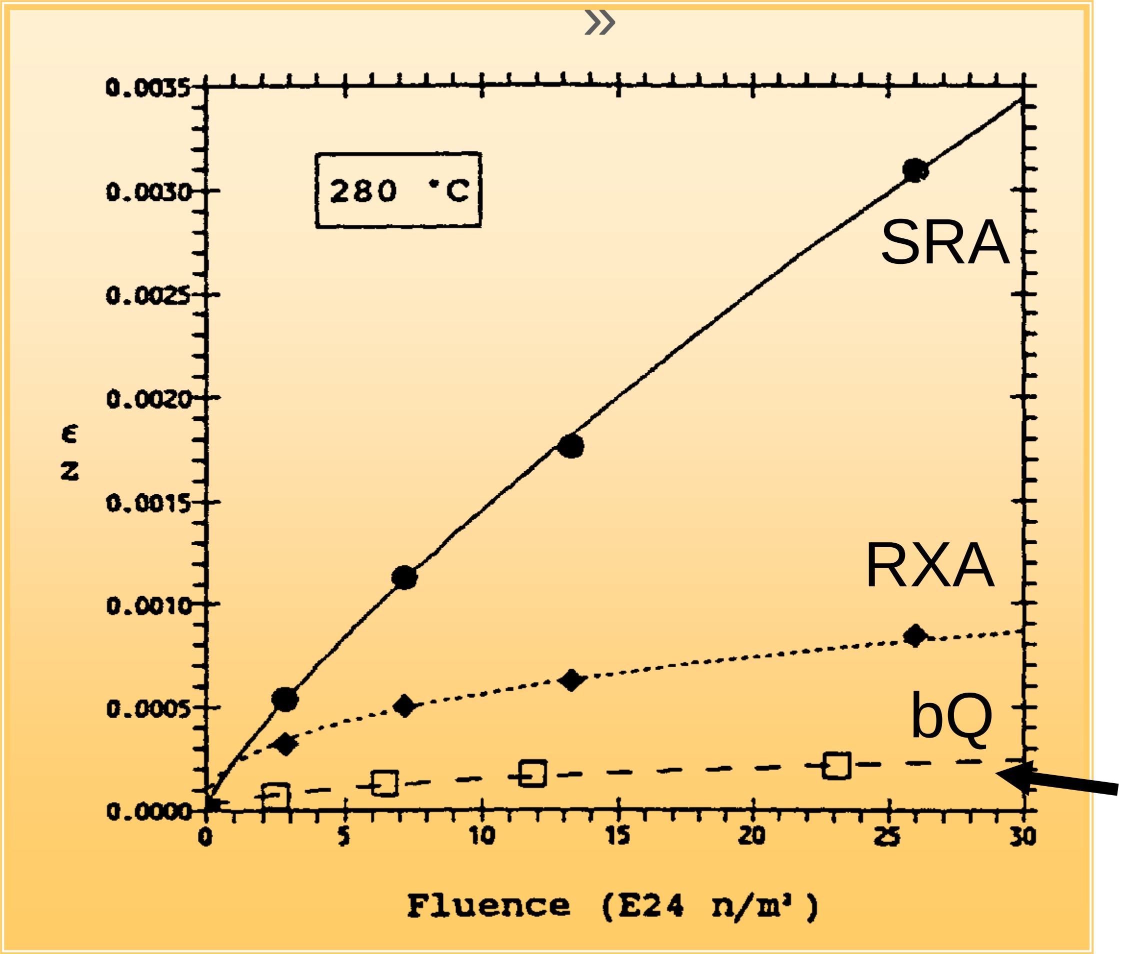
Growth



At high burn up M5 growths less than Zy4

Effect of metallurgical state on the growth rate of Zy-4

$$\varepsilon = \alpha\,(\phi t)^n$$



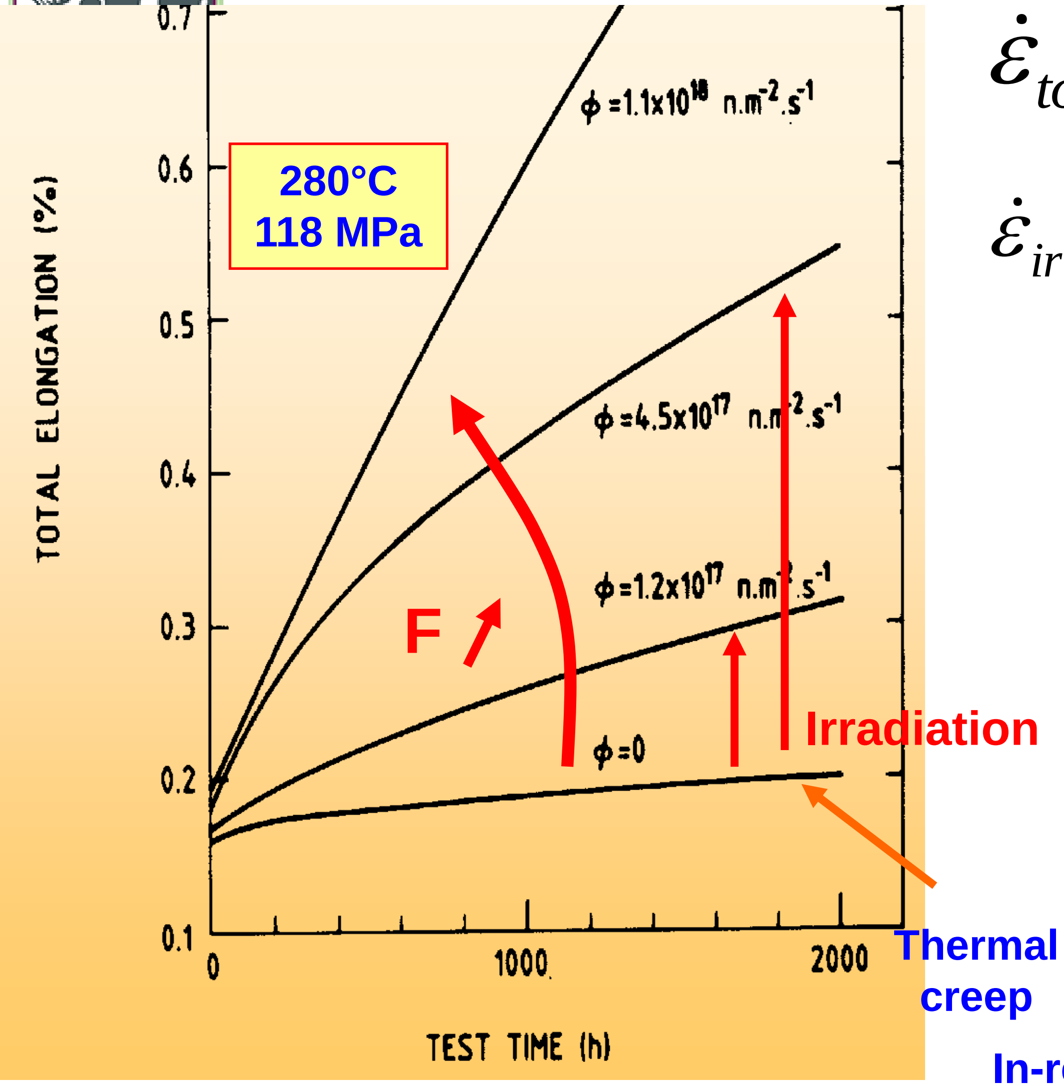
For PWR temperatures (300°C-350°C) :
SRA : n=0.7 to 1
RXA : n=0.5

Higher growth rate when higher temperature

b quenched,
Isotropic texture
-> low growth

Deformation under stress

Irradiation creep



$$\dot{\varepsilon}_{tot} = \dot{\varepsilon}_{th} + \dot{\varepsilon}_{irr} + \dot{\varepsilon}_{growth}$$

$$\dot{\varepsilon}_{irr} \propto \sigma^{n} \phi^{f} \exp\left(-\frac{Q}{RT}\right)$$

For pure irradiation creep :

n=1

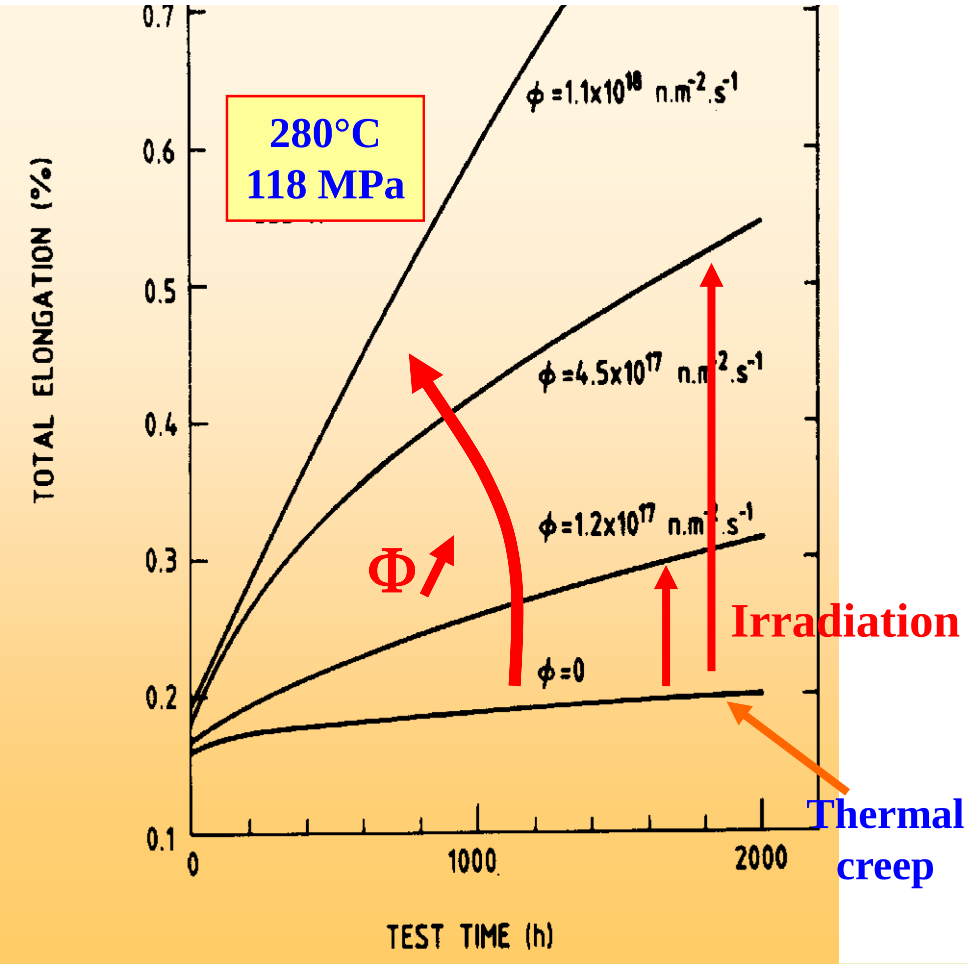
Q/R=2000 to 5000 K for T<300°C
-> low thermal activation
(low temperature effect)

For SRA : f=0.65 to 0.85
For RXA : f=1

In-reactor axial tensile test on SRA Zy-2

Mechanisms during irradiation creep

Without σ, dislocation: absorption is equal to emission =>dislocation do not move



280°C
118 MPa

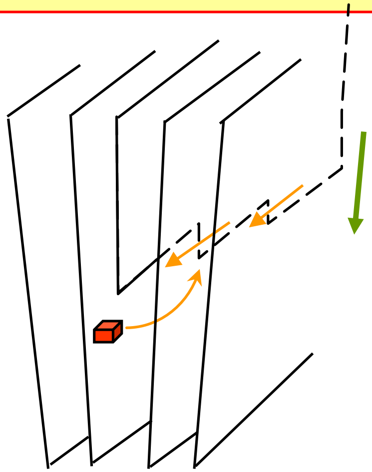
Φ↗

Irradiation

Thermal creep

Irradiation creep

Absorption of vacancy and interstitial help by stress => dislocation climbing



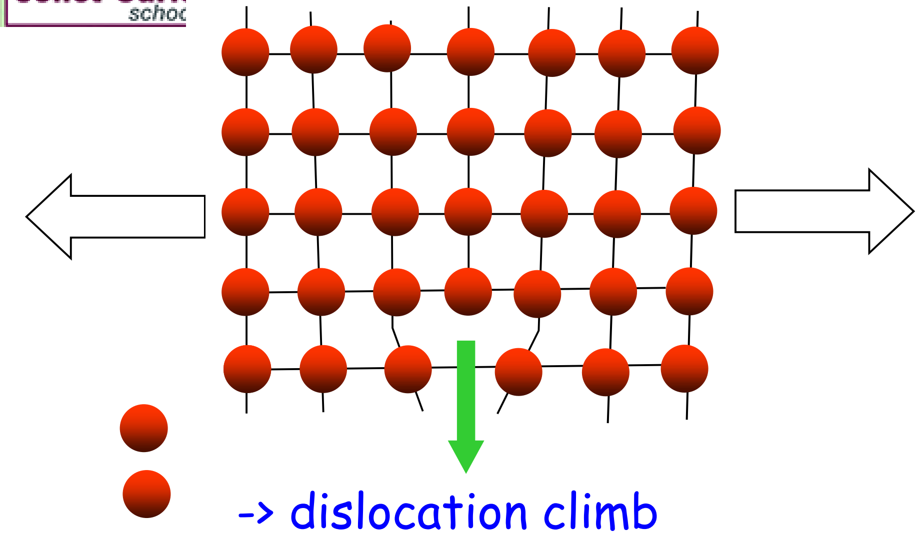
$$\dot{\varepsilon} = K_1.\sigma.\varphi$$

proportionnal to σ

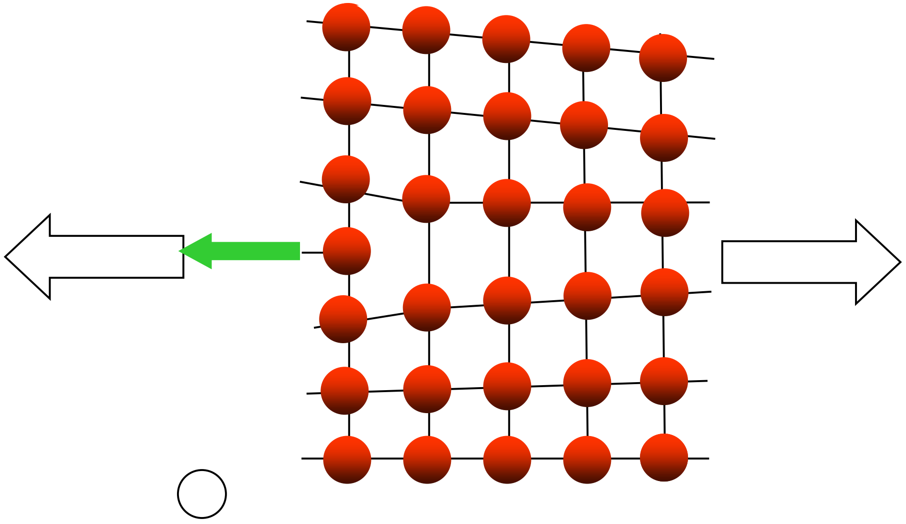
With σ, dislocation: absorption is favored compared to emission or the contrary =>depending on σ orientation

Irradiation Creep Mechanisms

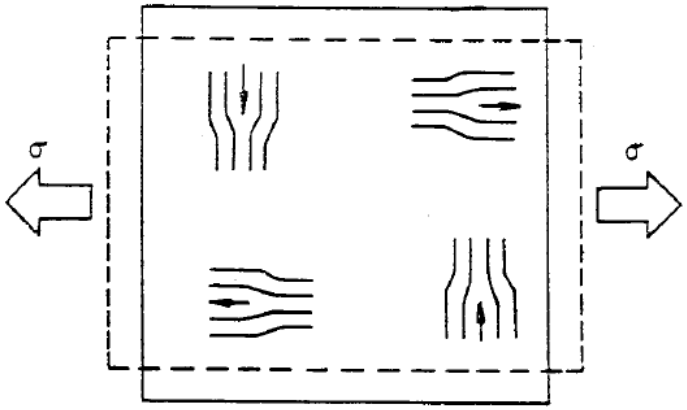
e.g. SIPA : Stress Induced Preferential Absorption



-> dislocation climb



-> dislocation climb

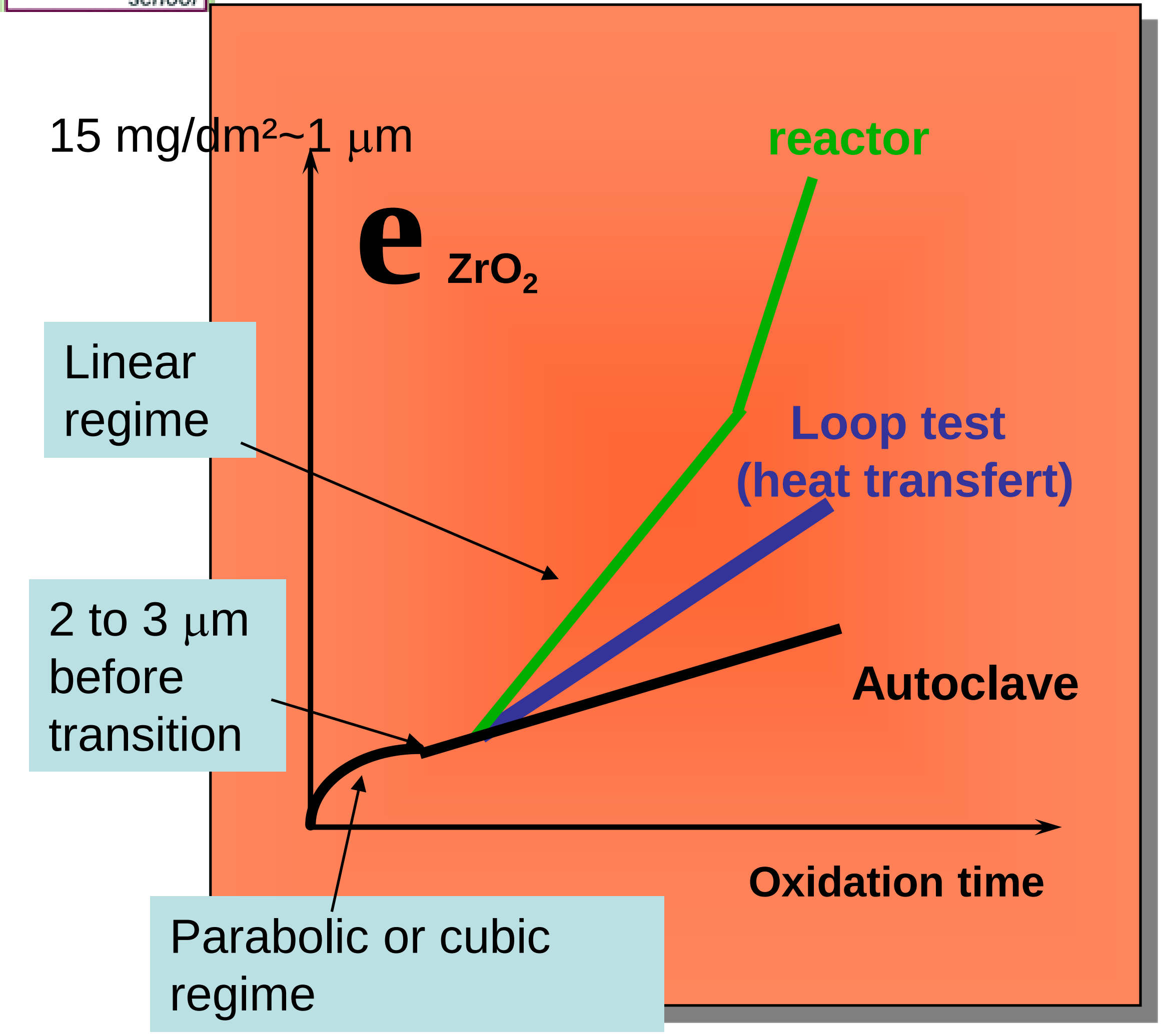


For dislocations showing different orientations with respect to the applied stress

-> macroscopic deformation -> creep

It can be shown that : $\dot{\varepsilon} = K_1.\sigma.\varphi$

Corrosion of Zr alloys



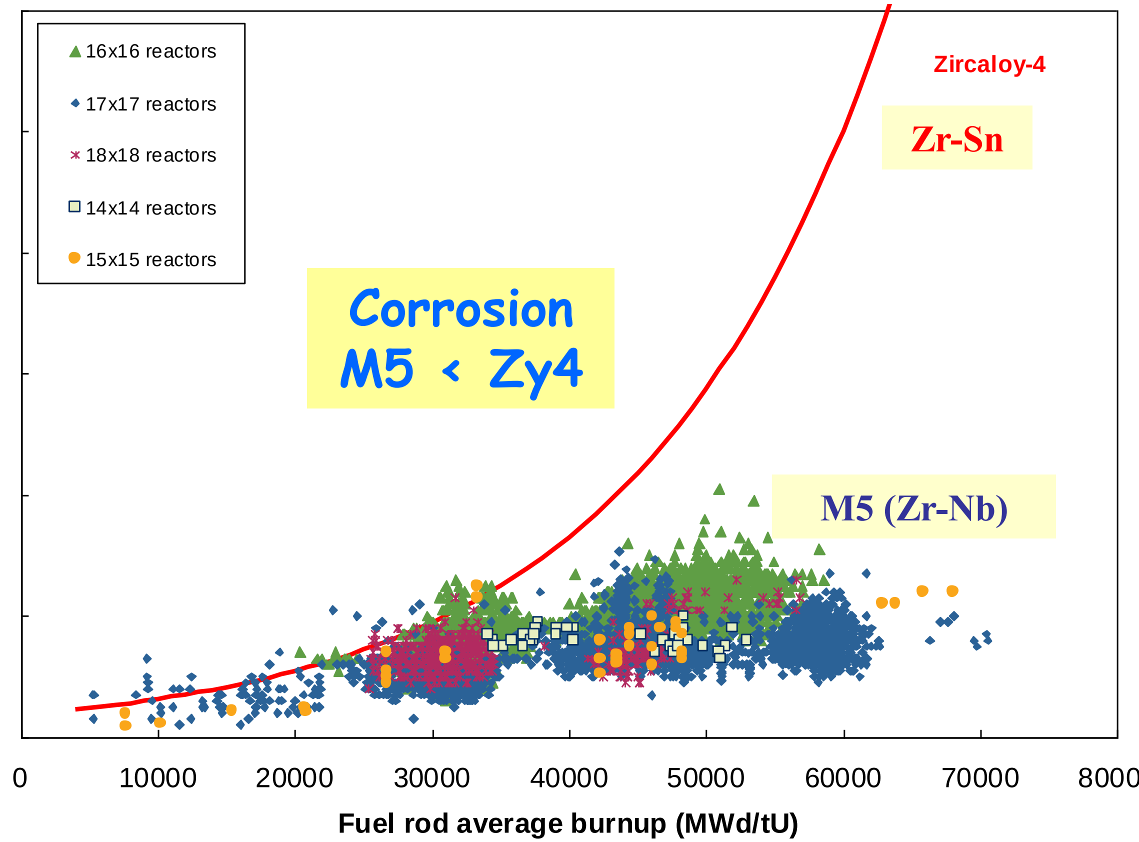
Linear regime

2 to 3 µm before transition

Parabolic or cubic regime

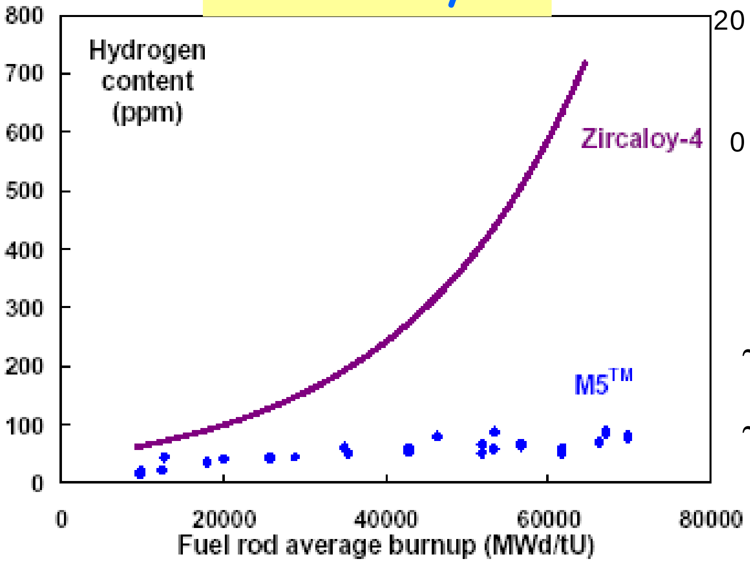
Zr oxide : ZrO_2 is very dense, stable, up to 2-3 microns

Corrosion of cladding



Corrosion
M5 < Zy4

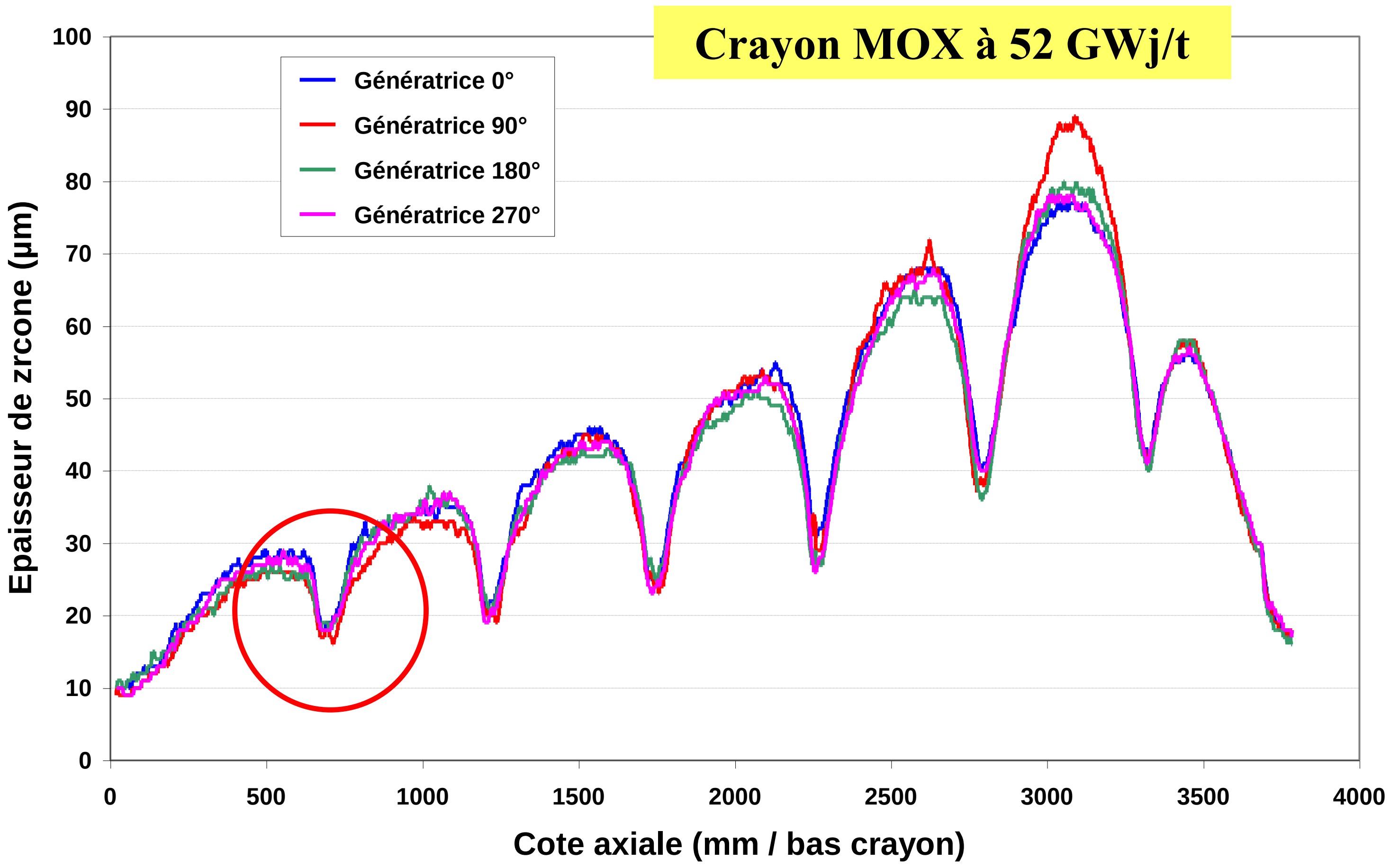
Hydruration
M5 << Zy4



~ 20 % H_2 ➔ Zircaloy 4

~ 10 % H_2 ➔ M5

Effect of temperature on oxidation of Zy4

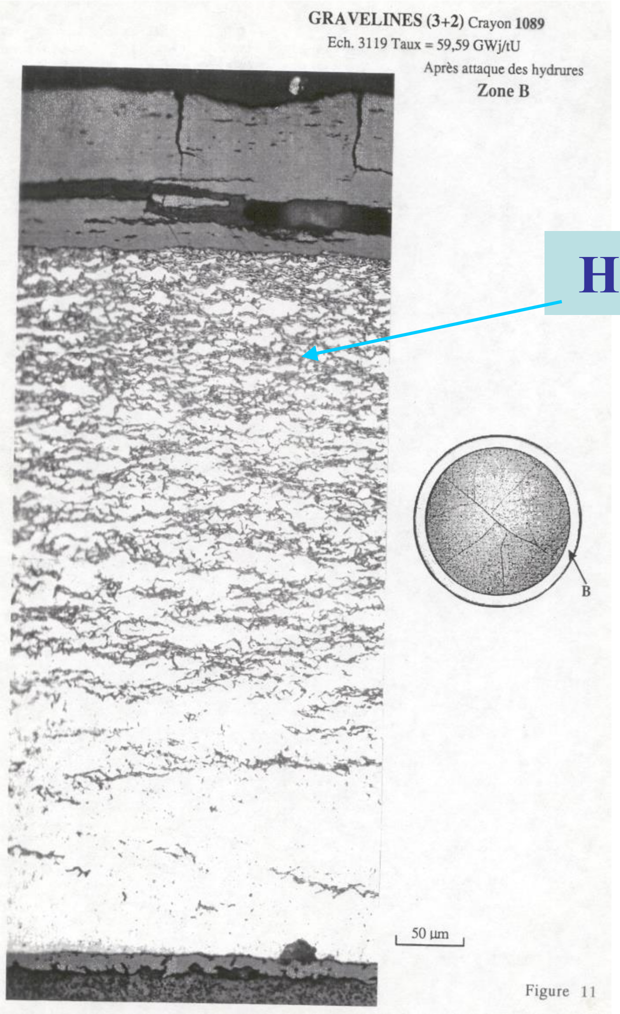


Bottom 290°C

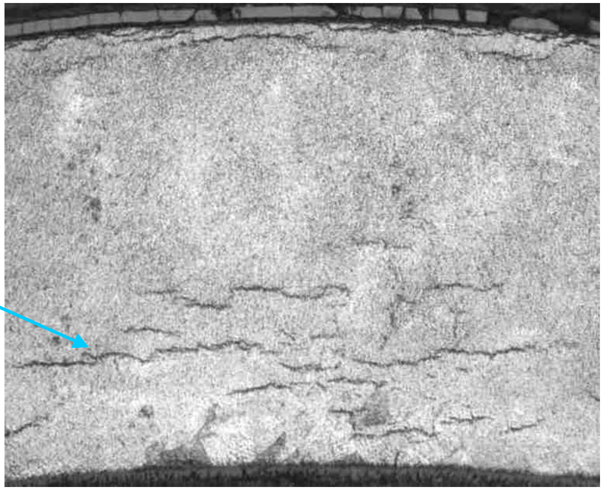
Top 325°C

Hydriding

Hydrides



Zy4, 5 cycles *60 GWj/t*



M5 6 cycles *72,6 GWj/t*

M5 is less hydrided than Zy4.

Hydryding

$\mathrm{Zr + H2O \Rightarrow ZrO2 + 4H}$

- A large part of H (reduction of water) get into Zr alloy (around 7 àto20 %)

(solubility at 350°C : ~100 ppm at 20°C, <1 ppm)

- Low solubility at room temperature, precipitation of hydrides $ZrH_{1.66}$
- Theses hydrides phases are brittle at low temperature (risky for materials handling and storage conditions)
- *(10 ppm H for 1 μm oxide)*

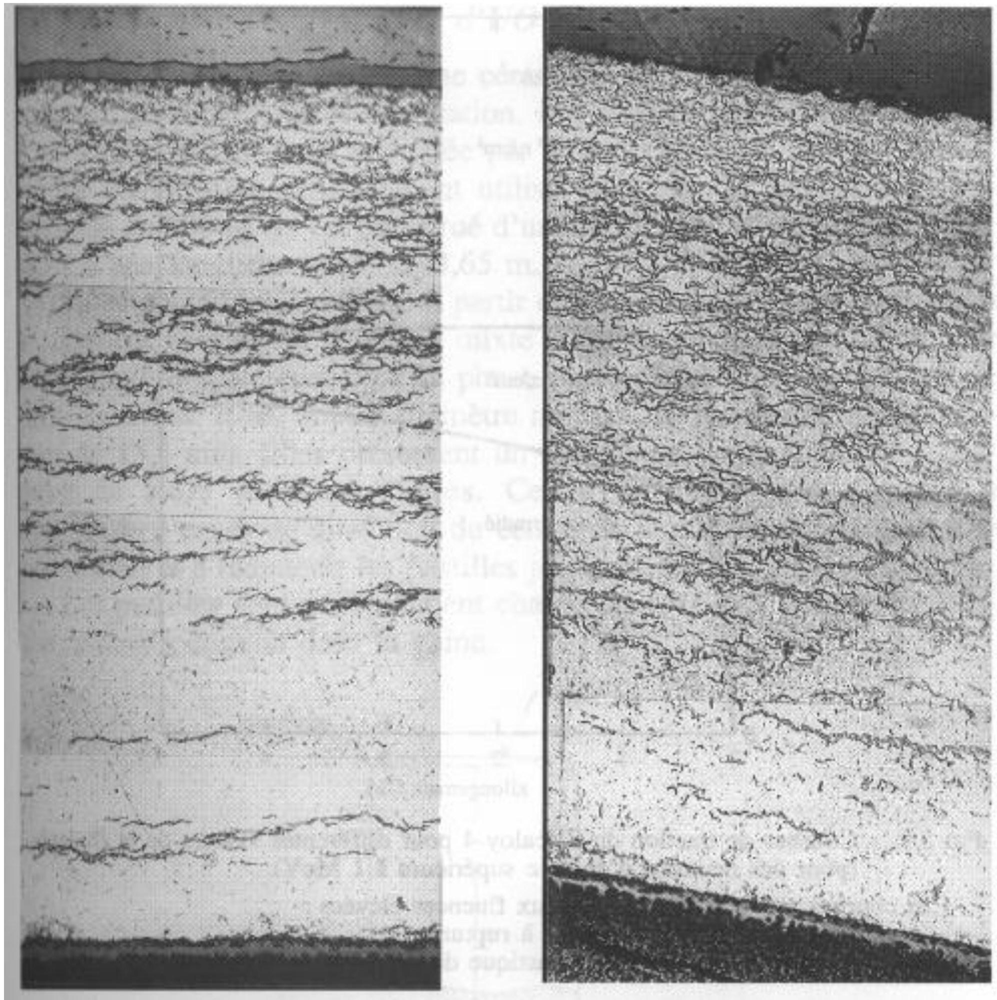
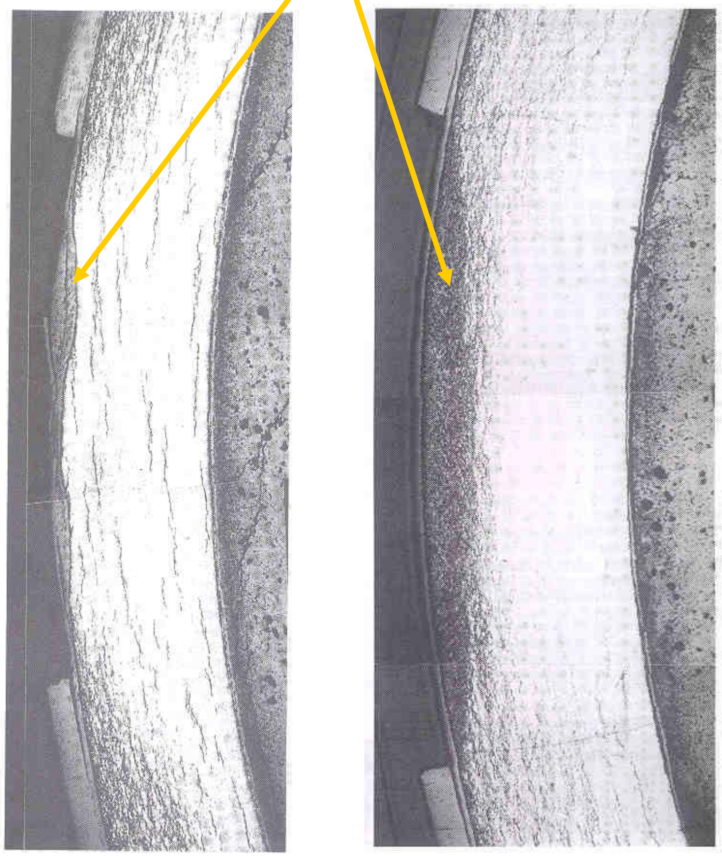


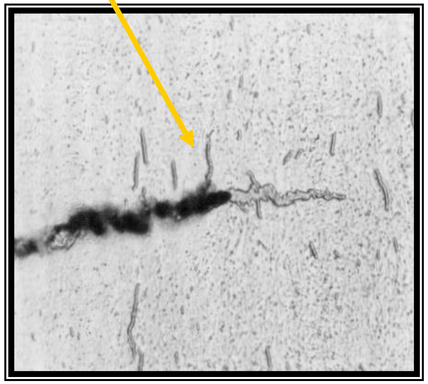
FIG. 5.8. – Micrographies de gaines hydrurées, irradiées en REP.

Cladding : Spalling of zirconia

$e_{ZrO2} \gtrsim 100$ µm ➔ spalling (oxide plates are torn)

➔ Cold spots ➔ précipitation of hydrides ➔ <u>embrittlement of the cladding</u>





Limit of Zy4

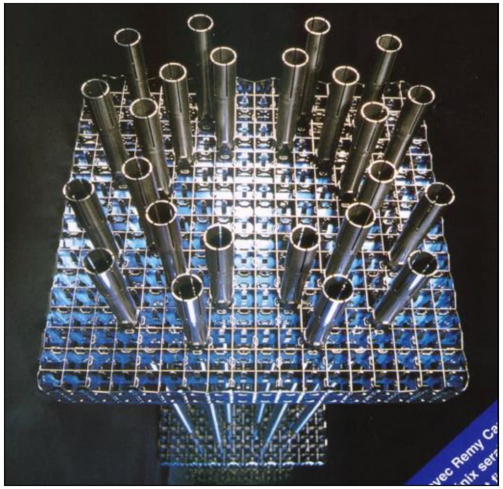
~ 60 GWj/t crayon

~ 52 GWj/t (ass)

➔Need for new cladding materials

➔ M5, Zirlo, …

Specific issues of structural materials

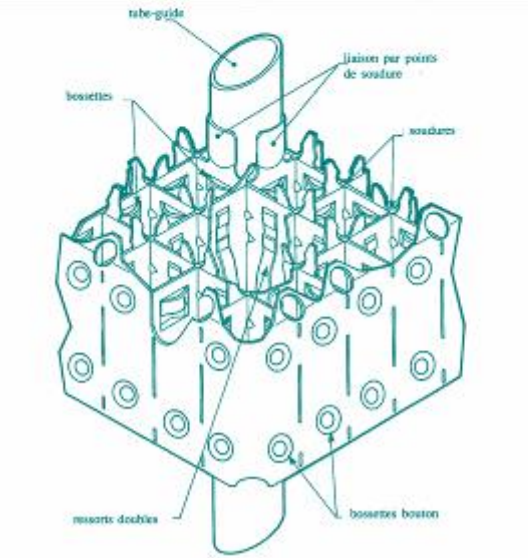


Beware of growth

Why?

Beware of growth

Beware of relaxation of dimples and springs



Need for flexibility

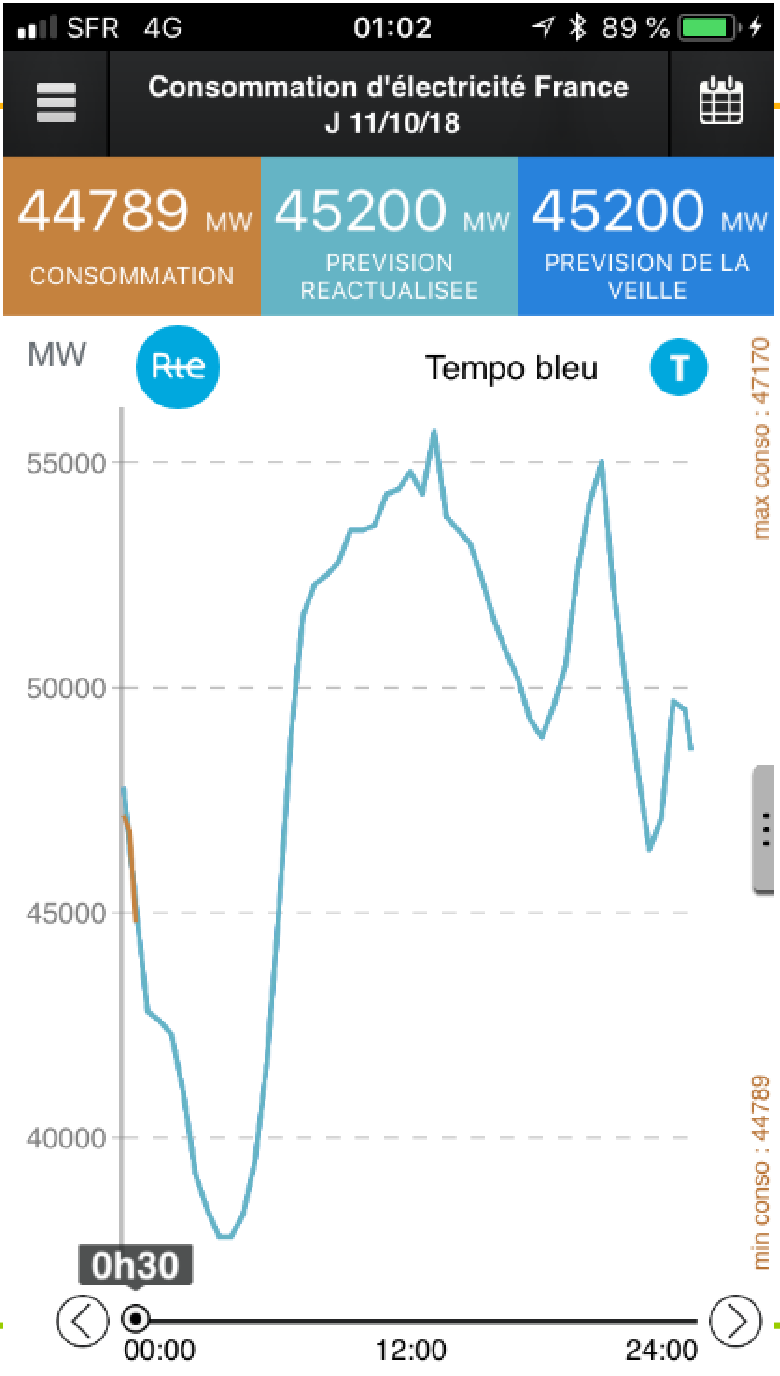
Why do we need to have variations of power in the French plants?

What are the impacts on the cladding?

Daily variation

RTE ECO$_2$mix app

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



Consommation d'électricité France
J 11/10/18

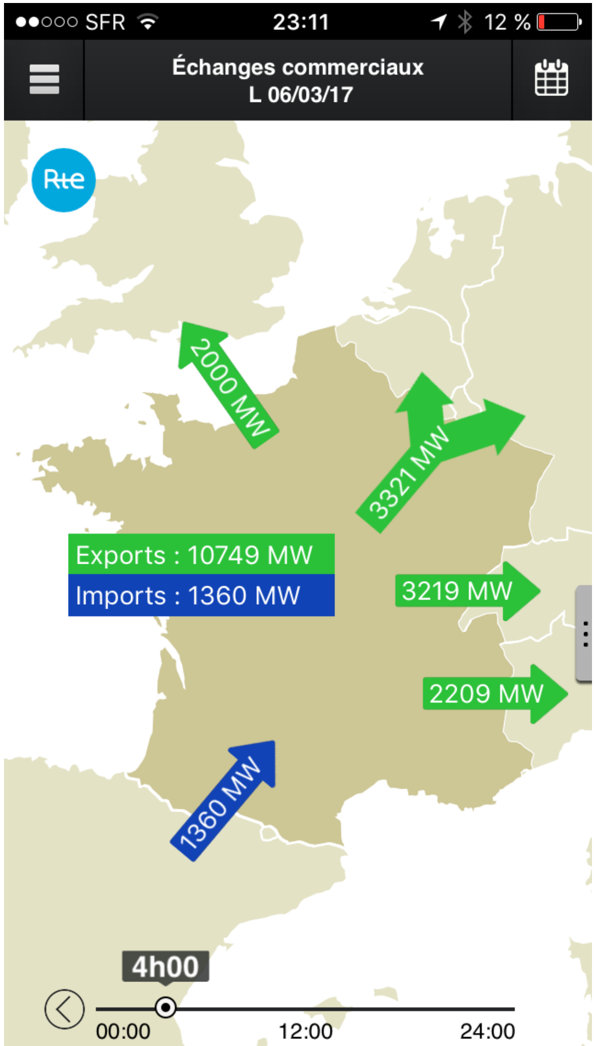
44789 MW CONSOMMATION

45200 MW PREVISION REACTUALISEE

45200 MW PREVISION DE LA VEILLE

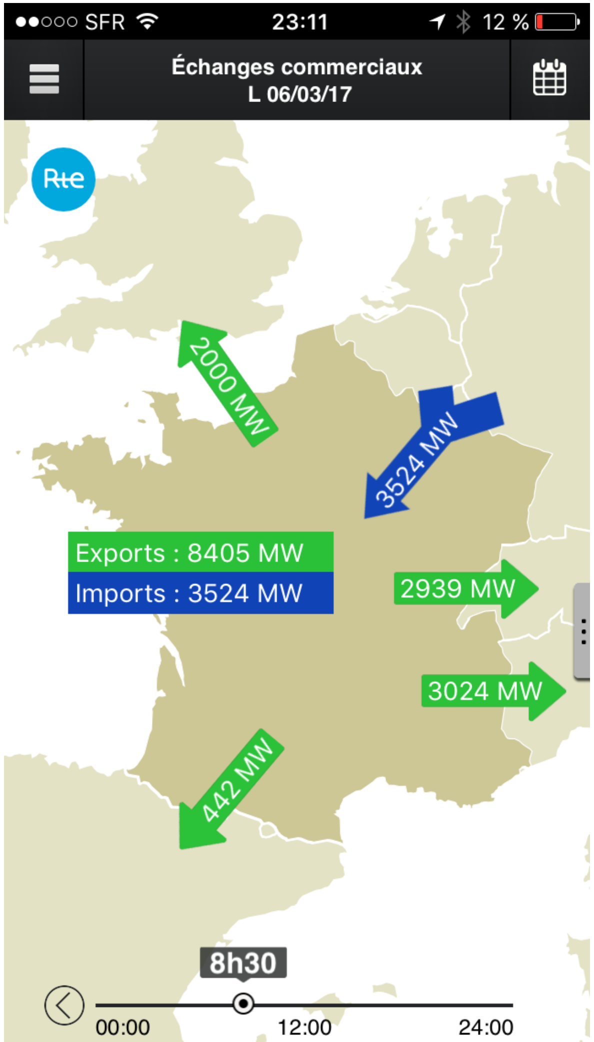
Tempo bleu

INTERCONNECTION



Échanges commerciaux
L 06/03/17

Exports : 10749 MW
Imports : 1360 MW

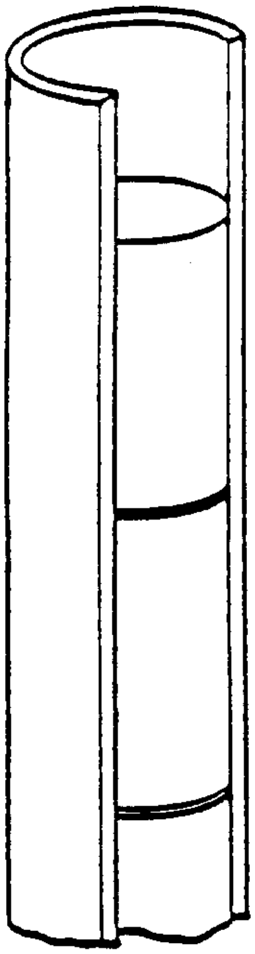


Échanges commerciaux
L 06/03/17

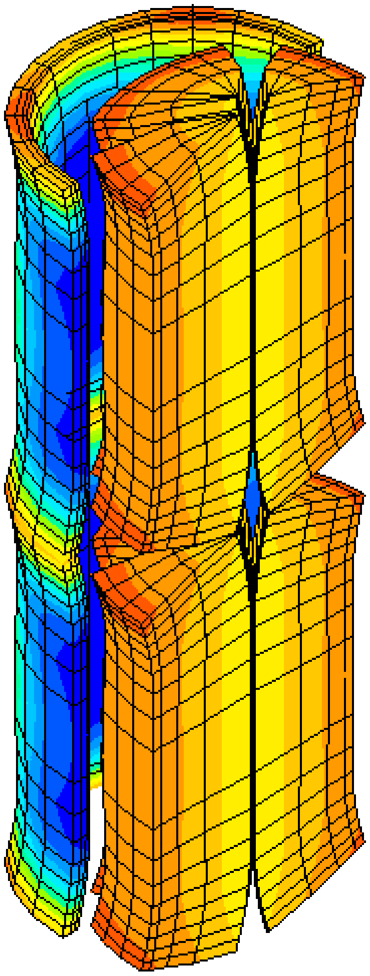
Exports : 8405 MW
Imports : 3524 MW

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**



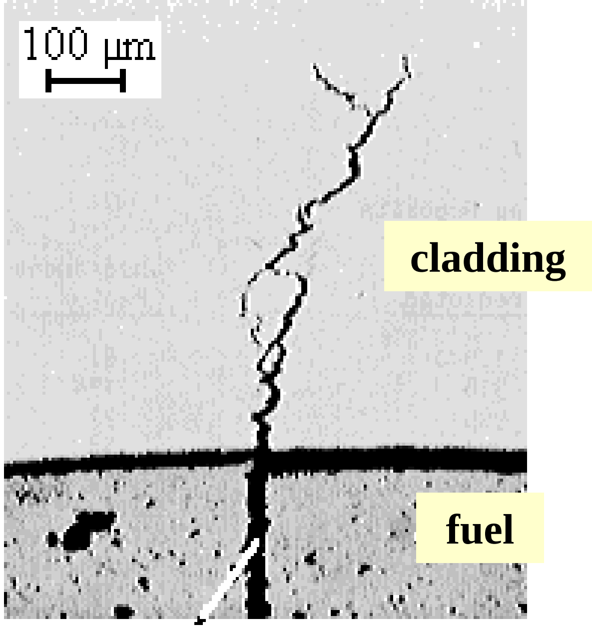
170 W.cm^{-1}



380 W.cm^{-1}

Pellet Cladding Interaction

I induced SCC
after power transient



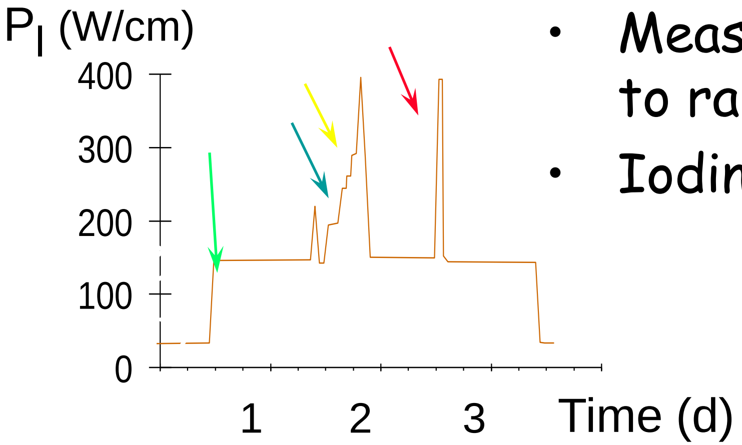
- Risk of cracks in cladding and rod failure
  - Transient> 420 W/cm
- Cracks appears after some mn at high power
  - Located at inter-pellet
  - In front of pellet cracks

    Where stresses are maximum

    And I escapes and condenses
- PCI risk seems to be maximum at the end of second cycle

Structure of a fuel rod after power transient

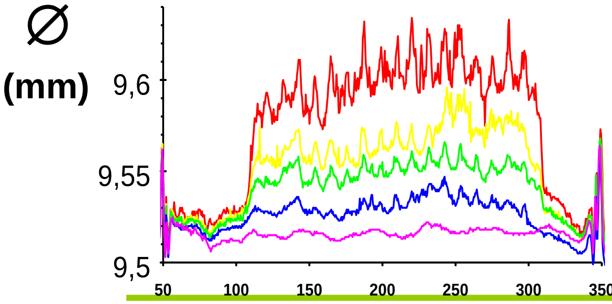
PCI on fuel rod during power transient

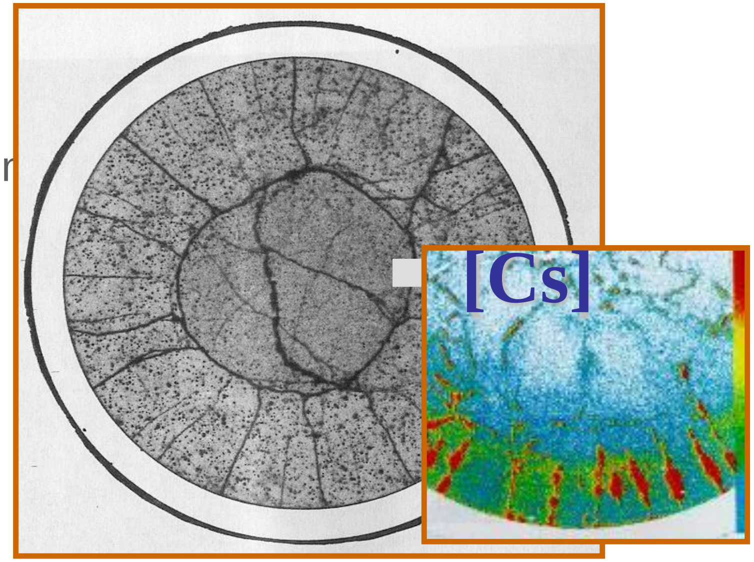


- Gas release fraction ⬀ 40%
- Measurements of Cs in the periphery (associated to radial cracks)
- Iodine is assumed to follow the same trend

Cladding hoop strain due to fuel thermal expansion

ΔØ µ 17 to 25 µm for each 100 $W.cm^{-1}$





Pellet Cladding Interaction

Iodine induced Stress Corrosion Cracking

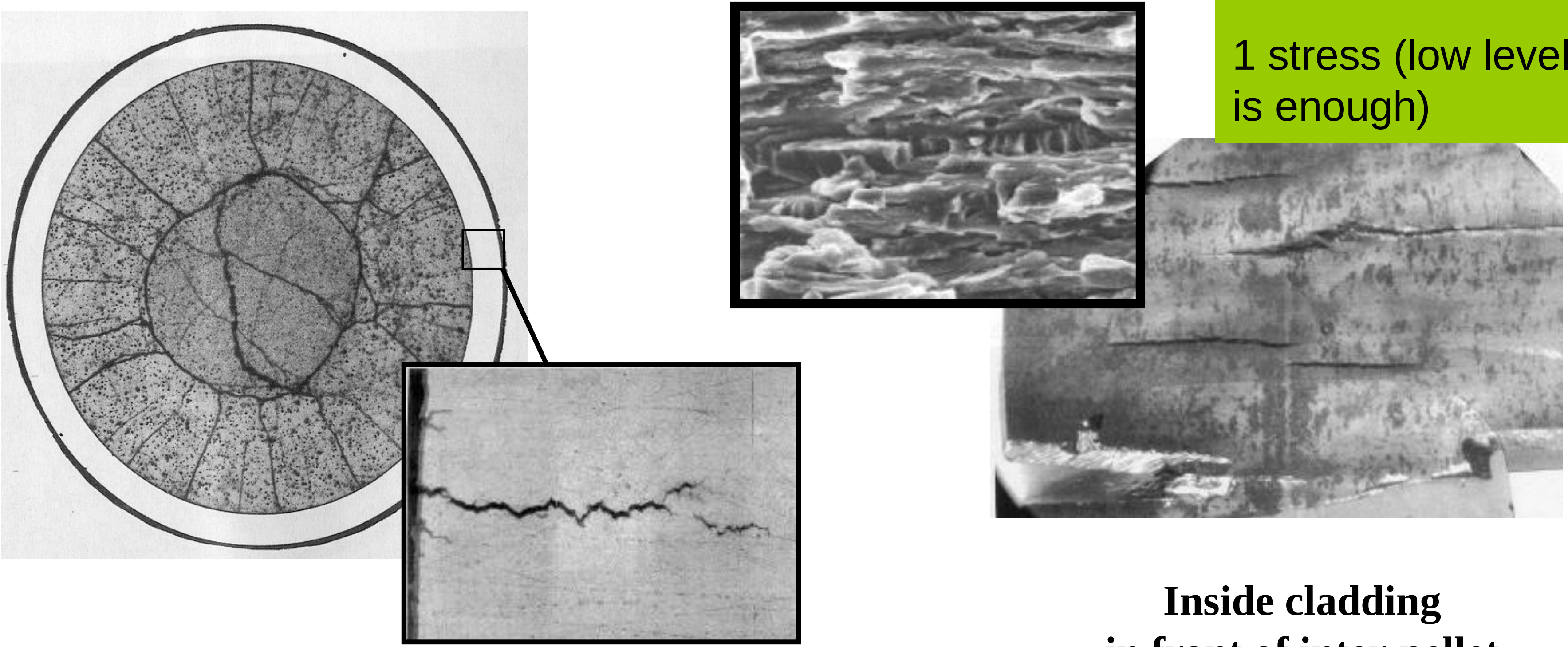
- Intergranular cracking

3 points are needed:

1sensitive material

1 environment

1 stress (low level is enough)

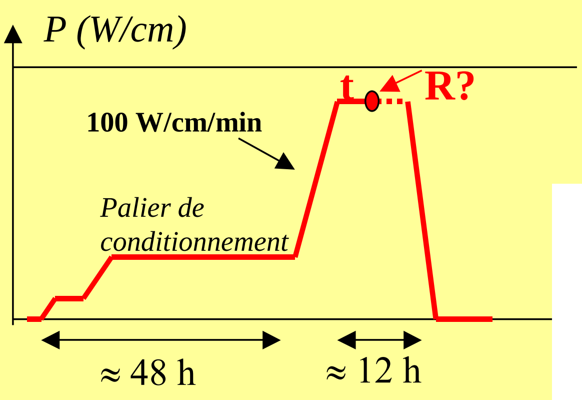


Inside cladding in front of inter-pellet

Technological limit deduced from experiments

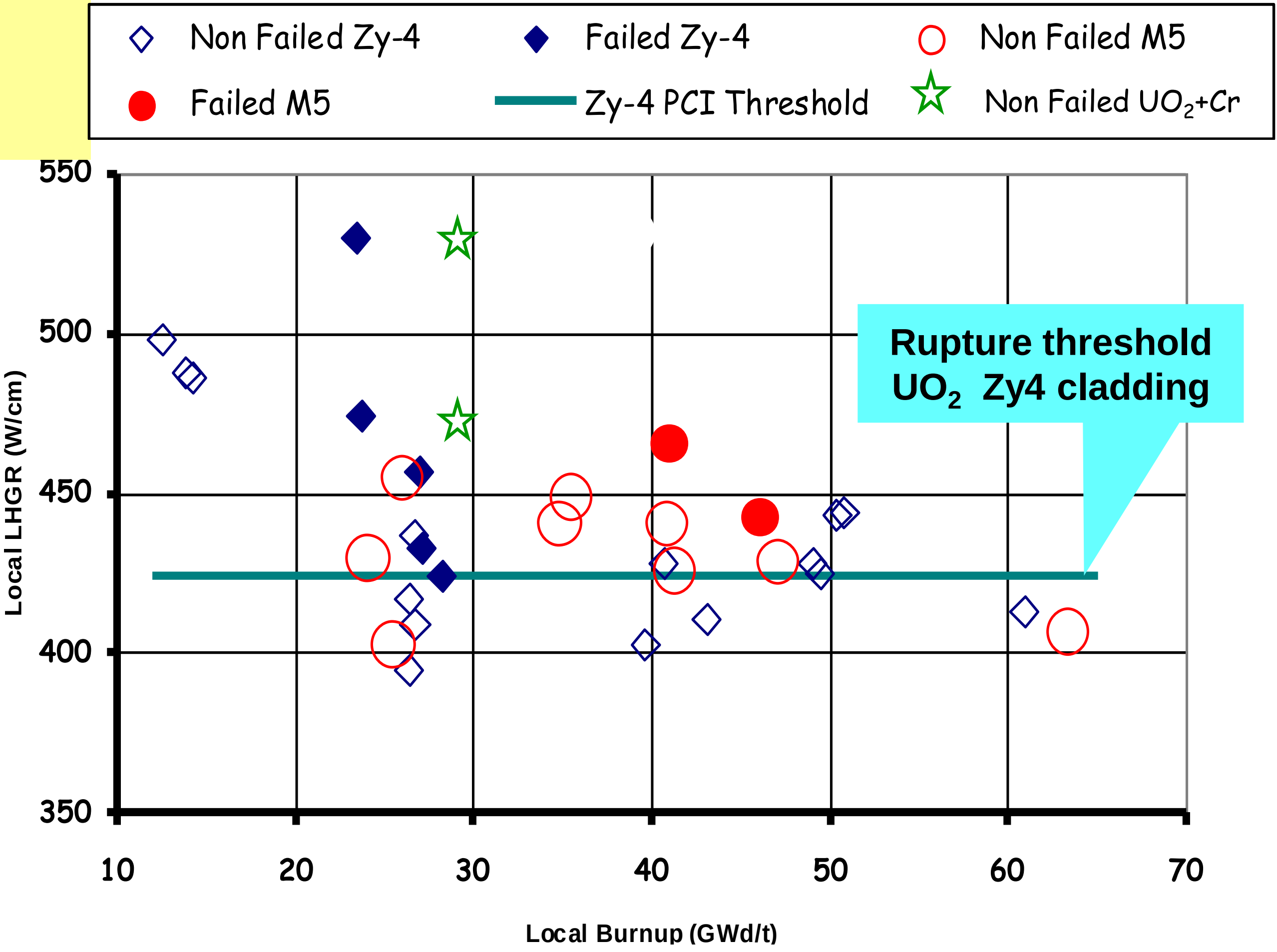
Power ramps

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



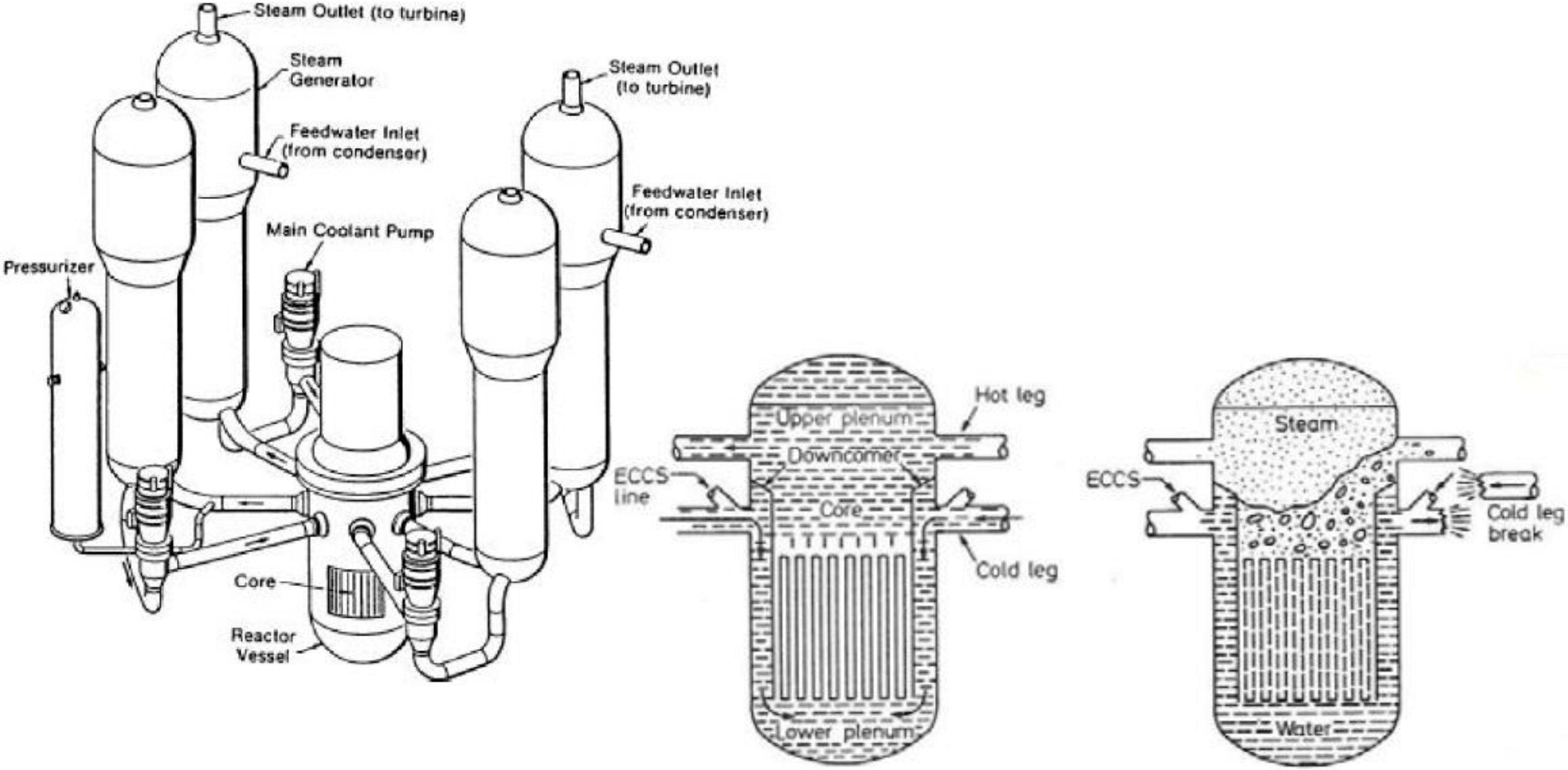
Paramètres de fonctionnement
Plin max, DPlin, vitesse

Paramètres du crayon
Bu, gaine, combustible



Rupture threshold
UO_2 Zy4 cladding

Accidental conditions LOCA



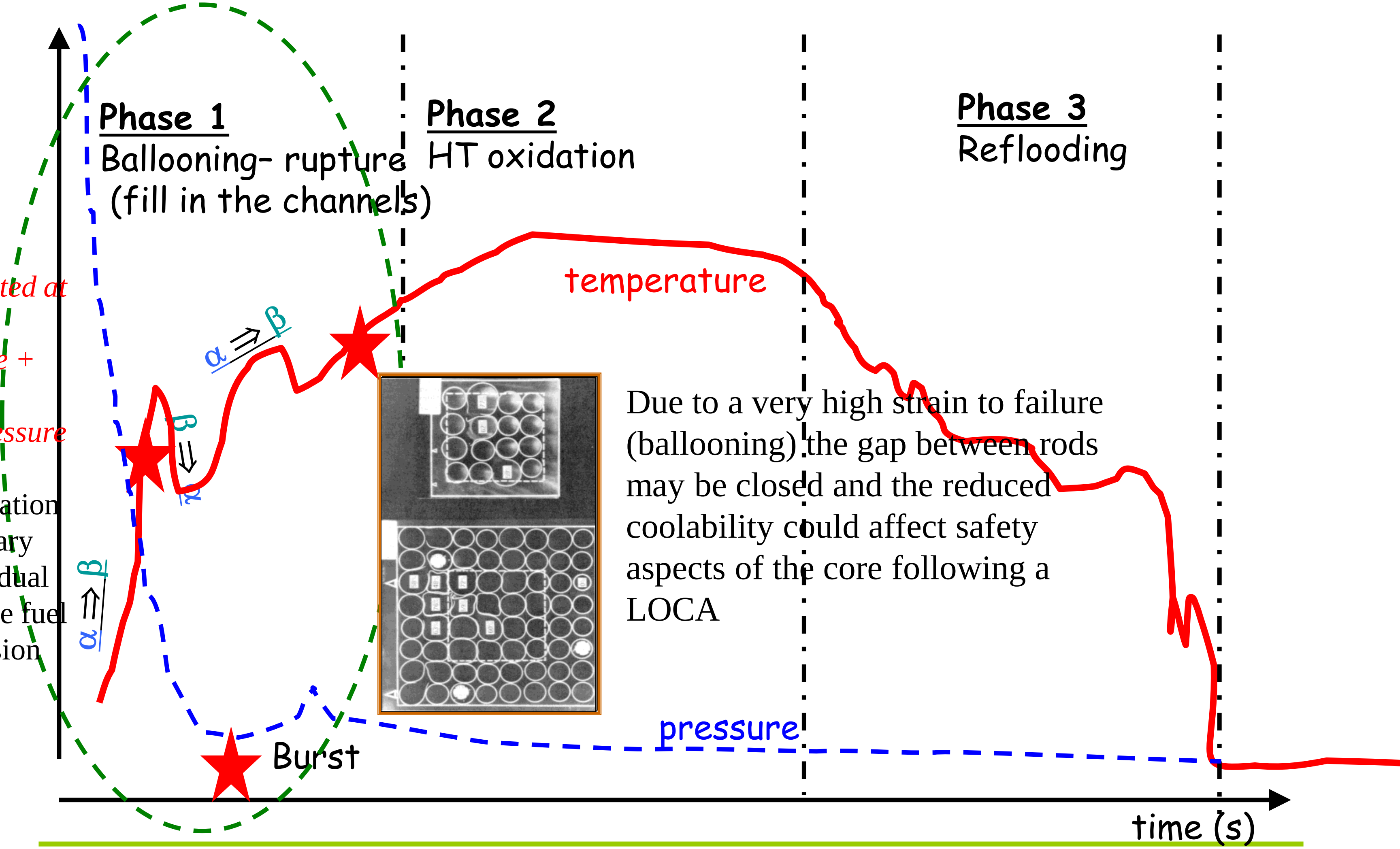
Normal operation

Blowdown

Behaviour for accidental conditions (high temperature)

First phase of LOCA - Cladding ballooning and recoolability

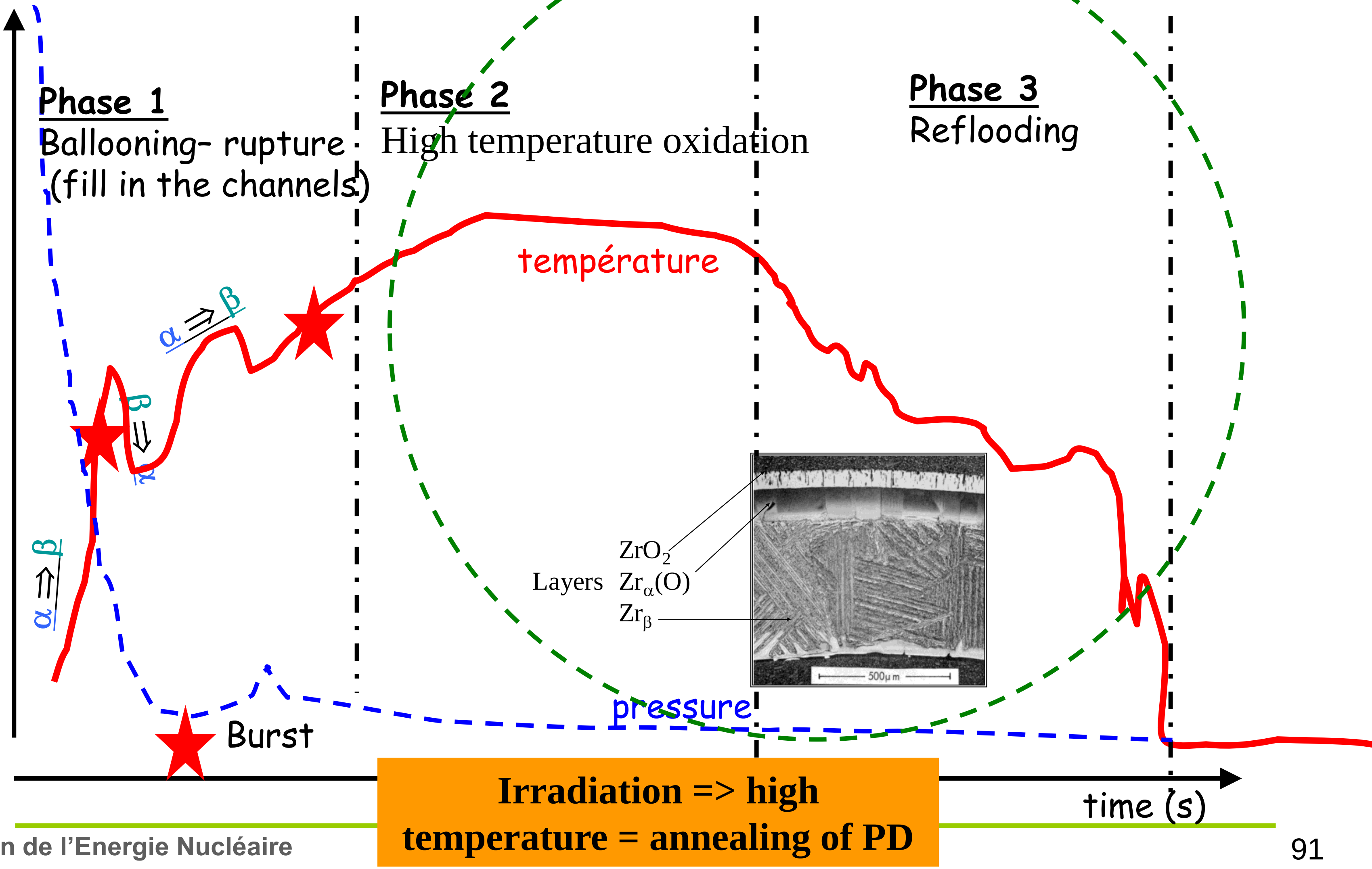
During accidental conditions LOCA, the cladding is *quickly heated at high temperature + increase of internal pressure* due to the depressurization of the primary circuit, residual power of the fuel and gas fission release



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*



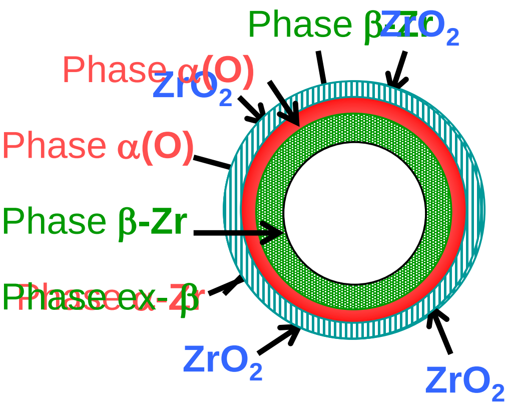
Phase 1
Ballooning- rupture
(fill in the channels)

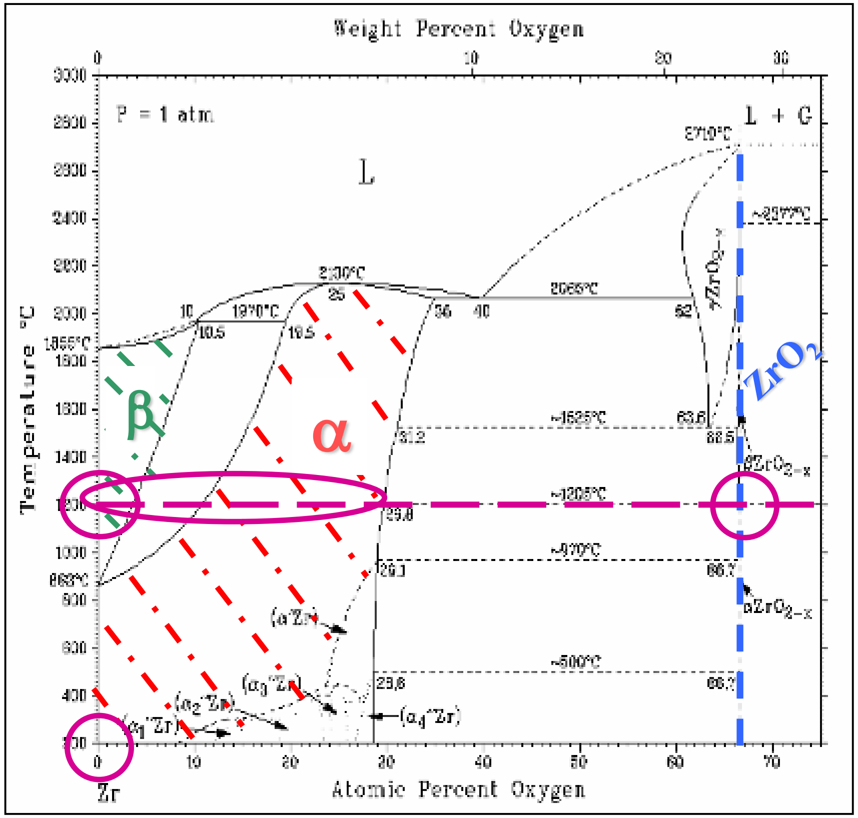
Phase 2
High temperature oxidation

Phase 3
Reflooding

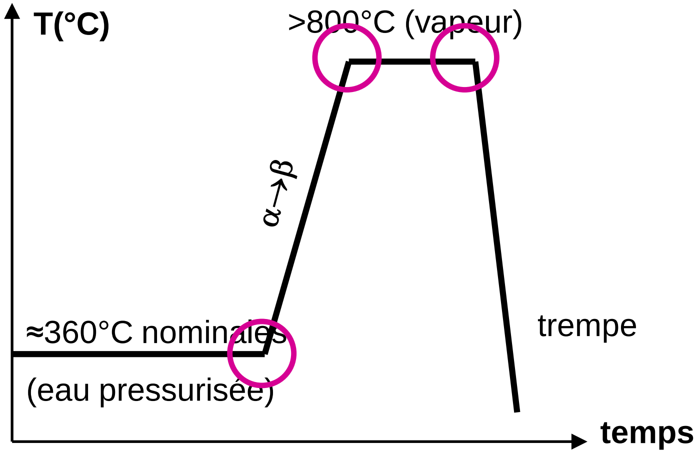
Irradiation => high temperature = annealing of PD

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





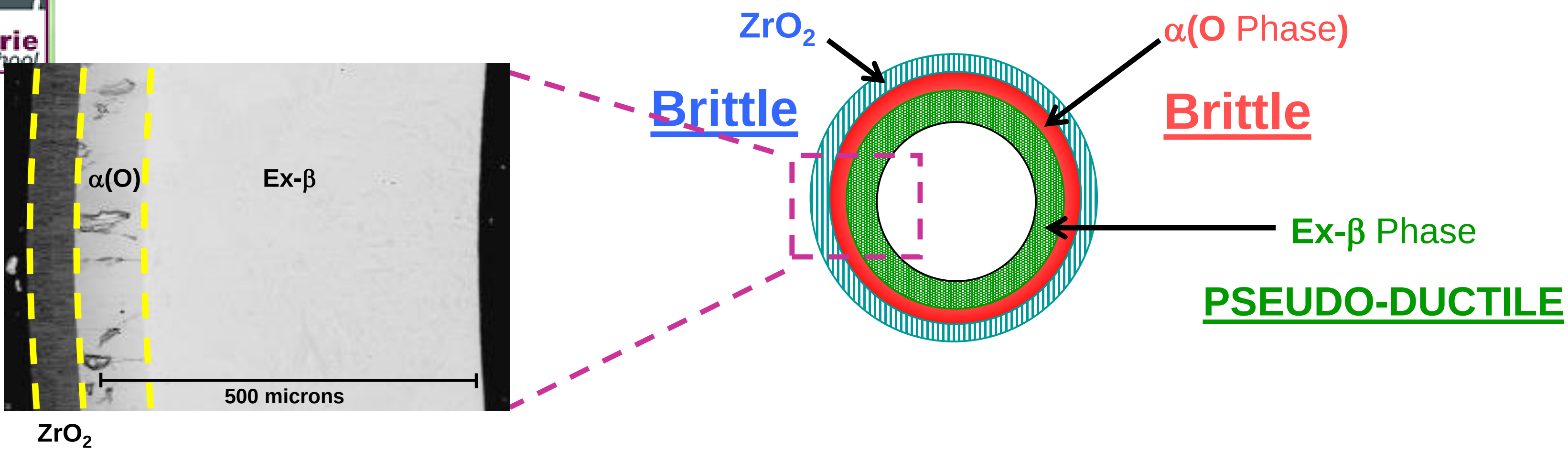
Binary diagram Zr-O



Transitoire de trempe simulant les conditions hypothétiques accidentelles (APRP)

microstructurals and mechanical consequences :

Accelerated oxidation ⇒ oxygen diffusion inside the metal ⇒ brittleness



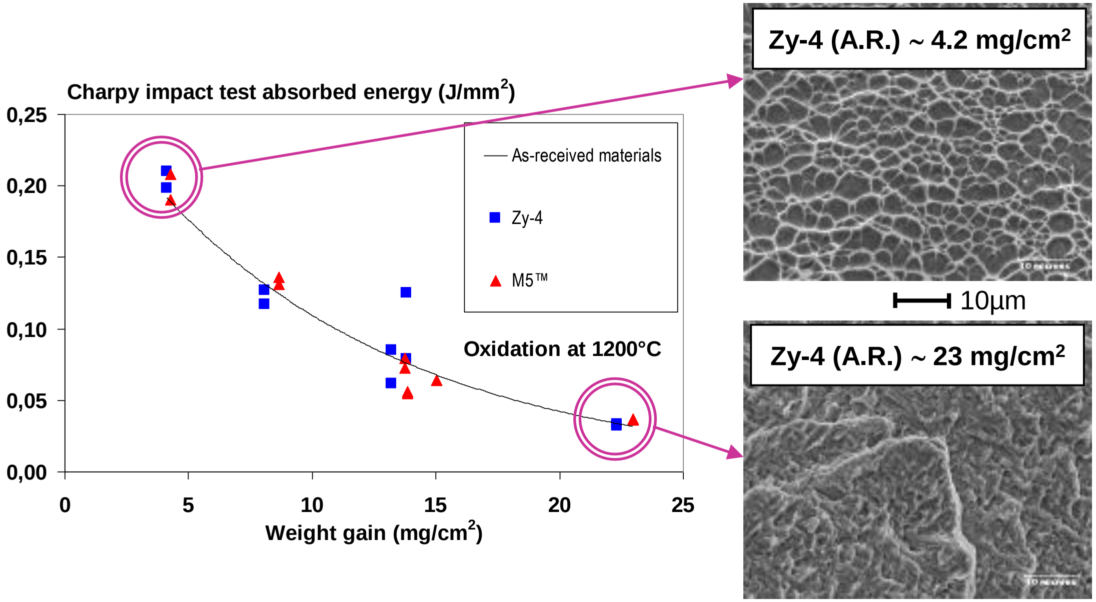
Residual ductility of the cladding depend on :

(1) **Ex-β** phase thickness

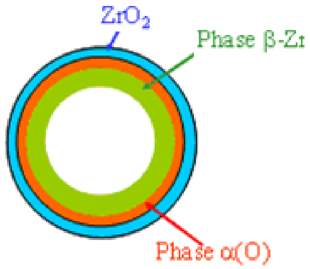
(2) C(O) in the Ex-β phase after **quenching**

ductile ⇔ brittle transition

à $[O]_{\text{critical}} \sim 0.4$wt.%



post oxidation /quench mechanical tests



← Low oxidation level: ductile material

High oxidation level: fragile material →

LOCA accident

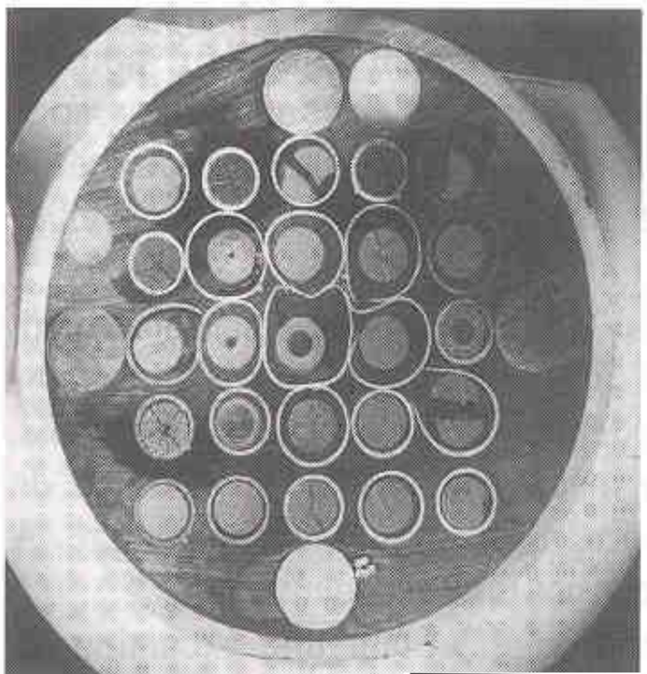
Test in PHEBUS reactor

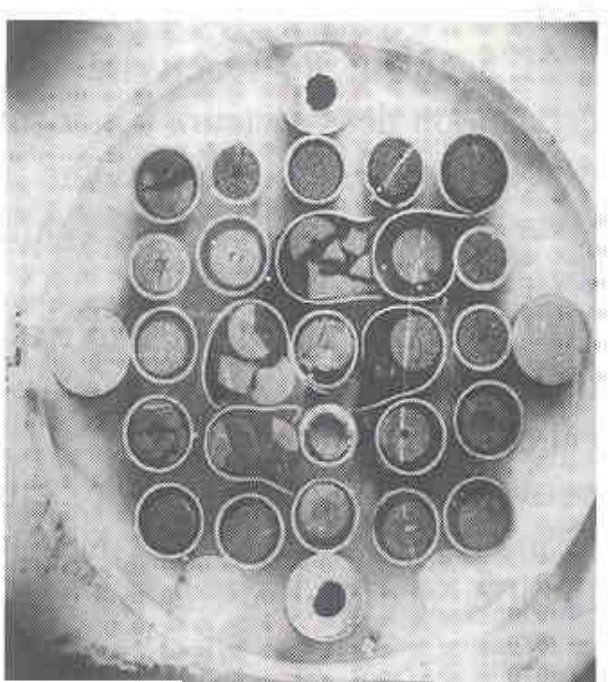
Current criteria :

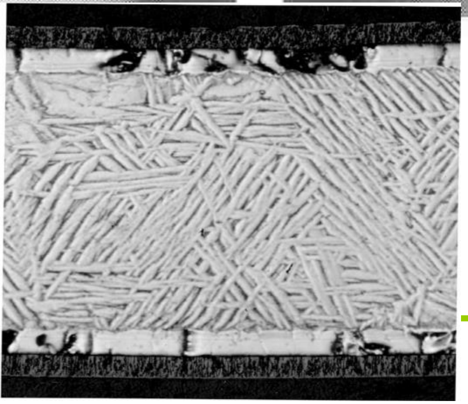
- T cladding
PCT < 1204°C

- ECR < 17 %
(Equivalent Cladding Reaction)

New criteria being discussed







Reactivity Insertion Accident (RIA)

- Ejection of a control rod
    - ➔ very fast increase of reactivity
    - ➔ Fast power transient (few tens of milliseconds)
    - ➔ Quasi adiabatic heating of the fuel
      Tmax in periphery
    - ➔ Strong mechanical fuel -cladding interaction

Essai Cabri
REP Na 8

Hydride blister

cladding

Fuel with grain boundary decohesion

Reactivity Insertion Accident (RIA)

- Reactivity accident (RIA)
  - At high Burnup, in hydrided (hence fragile) cladding PCI ➔ failure of the pin
    - Instantaneous release of intragranular fission gas can in some cases contribute to the mechanical loading
    - Ejection of fuel fragments
      ➔ generation of mechanical energy by thermal interaction between fragments and water.

Criterion : this mechanical energy must stay low

Avoiding cladding rupture is a way to guarantee it

➔ Limits on injected energy (# 80 cal/g at high burnup)

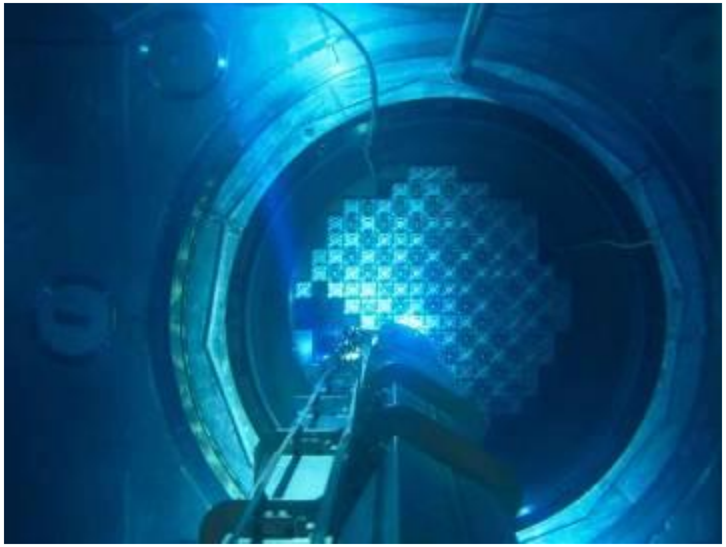
➔ Residual ductility of the cladding (no spalling of the oxide)

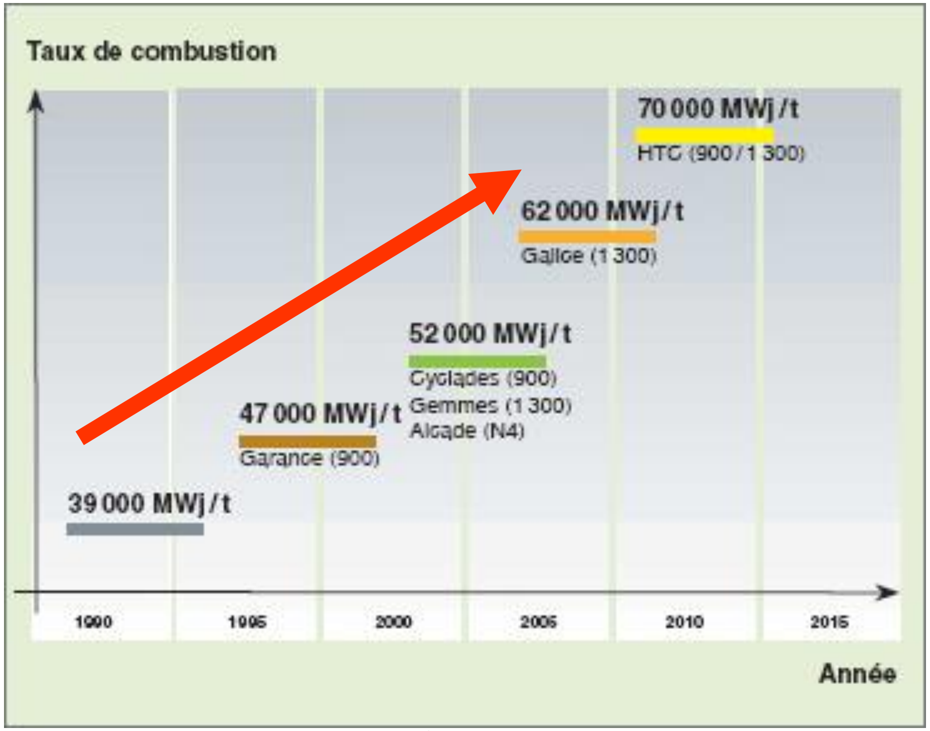
Conclusions

Huge gains in improving the fuel performances

Old fuel assemblies: 3 PWR cycles ~30 GWj/t

New fuel assemblies: 5 PWR cycles ~50-60 GWj/t





(1 PWR cycle ~ 2.10^{21} n/cm²)