Materials under irradiation

PWR Structural Materials

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Short bio

Engineering degree Ecole Centrale Paris (materials major)

PhD in Appied Physics California Institute of Technology (USA)

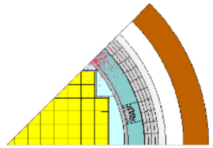
Research assistant at Los Alamos National Laboratory (USA)

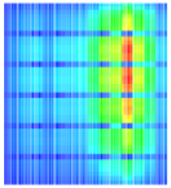
Post doc at the EU Joint Research Centre of Petten (The Netherlands)

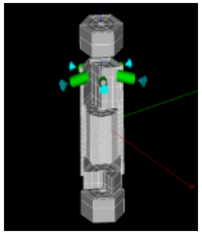
I joined CEA in 1996, held several management positions before becoming director of nuclear activities in Saclay in 2017

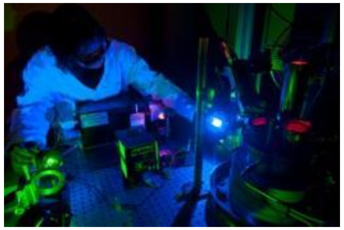
Adjunct professor at Centrale Supelec, ENSTA and Phelma Grenoble

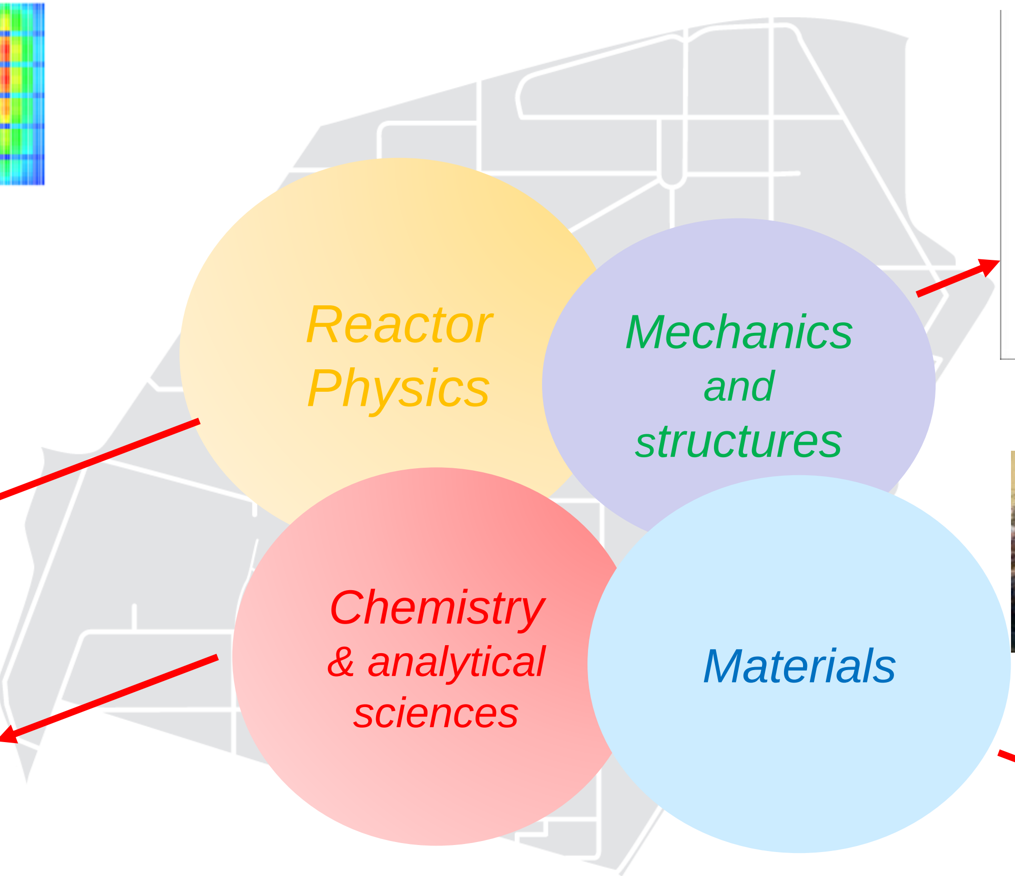
Direction of Nuclear Activities of Saclay









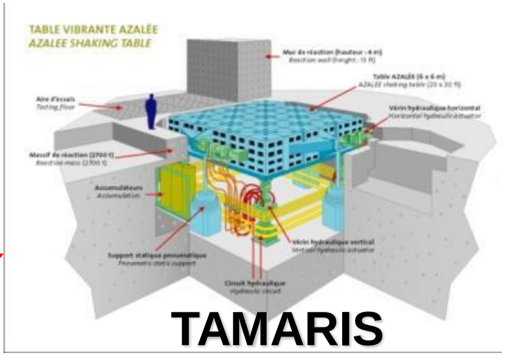


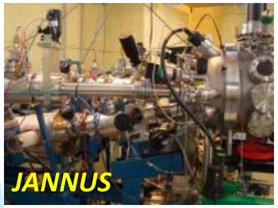
Reactor Physics

Mechanics and structures

Chemistry & analytical sciences

Materials





Basic Sciences for nuclear applications

125 PhD students

Outline

- **Effects of irradiation on materials**
- **PWR structural materials**
  - **The reactor pressure vessel**
  - **The internal structures**
- **PWR fuel assembly materials (excluding fuel)**
- **Gen IV structural materials**

Outline

Effects of irradiation on materials

- Macroscopic effects of irradiation
- Neutron – matter interaction
- Point defects, displacement cascades, irradiation damage
- Long term evolution of point defects : structure, mobility, clusters and sinks
- Microscopic evolution

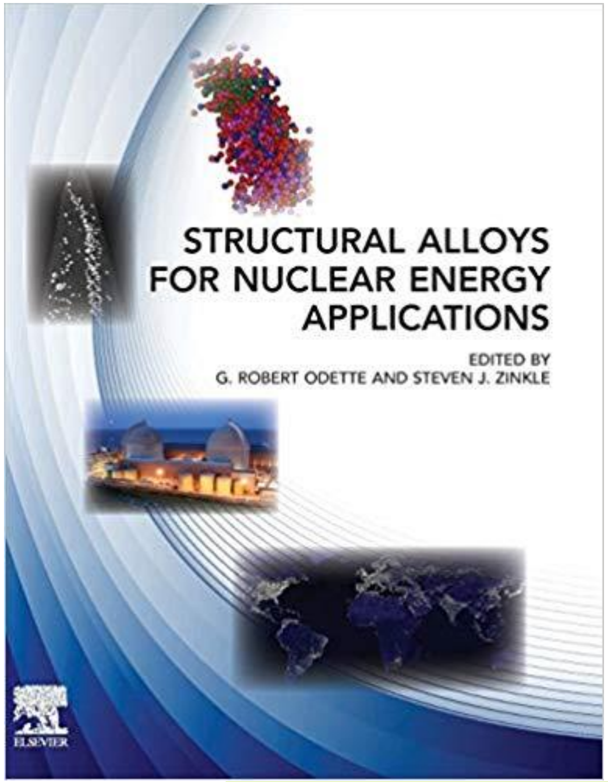
The reactor pressure vessel

- PWR design
- PWR fabrication
- Internal cladding
- Fracture of ferritic steels
- Irradiation embrittlement
- Pressure vessel integrity assessment
- Surveillance program

The internal structures

- Internals
- IASCC Intergranular corrosion (internals)
- Swelling

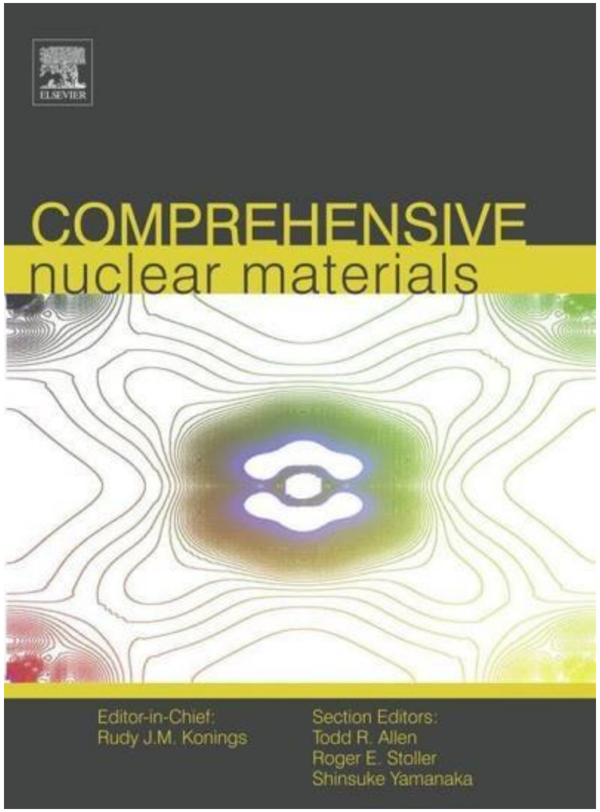
For further reading…



STRUCTURAL ALLOYS FOR NUCLEAR ENERGY APPLICATIONS

EDITED BY
G. ROBERT ODETTE AND STEVEN J. ZINKLE

ELSEVIER



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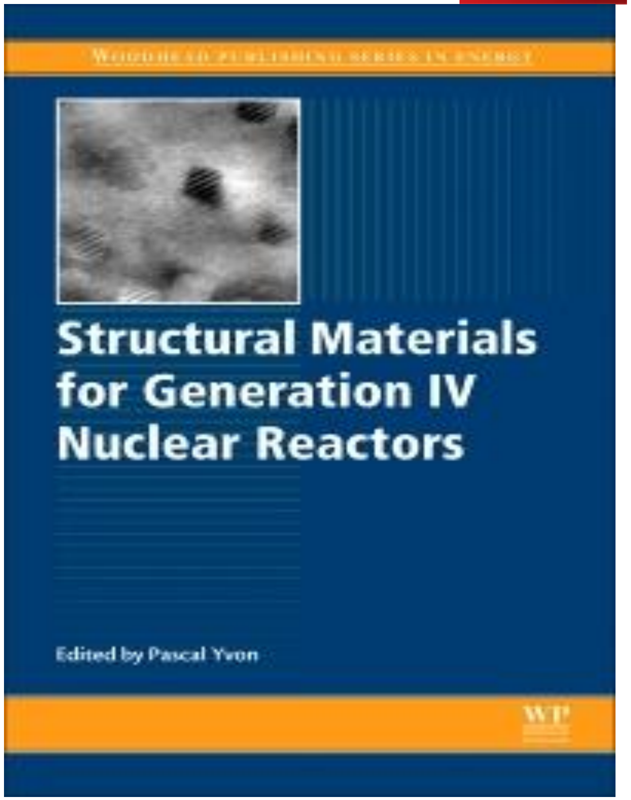


Structural Materials for Generation IV Nuclear Reactors

Edited by Pascal Yvon

WP

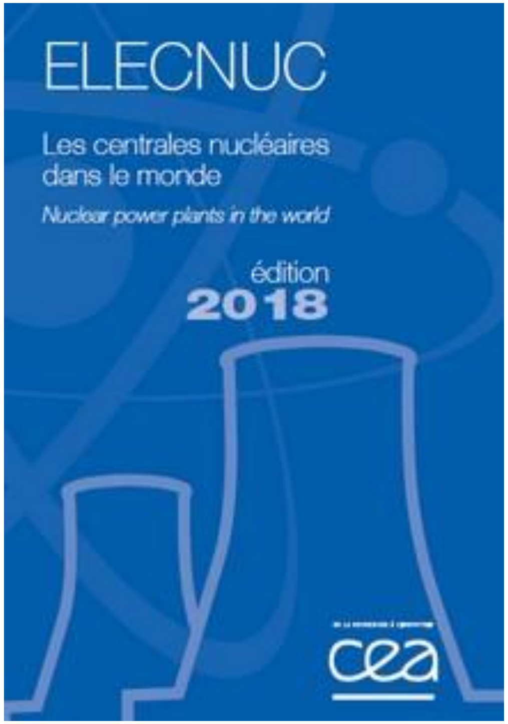
World nuclear fleet

450 reactors operating in the world

>99% have a thermal spectrum

2/3 are PWR (or VVER)

So the focus will be on PWR materials

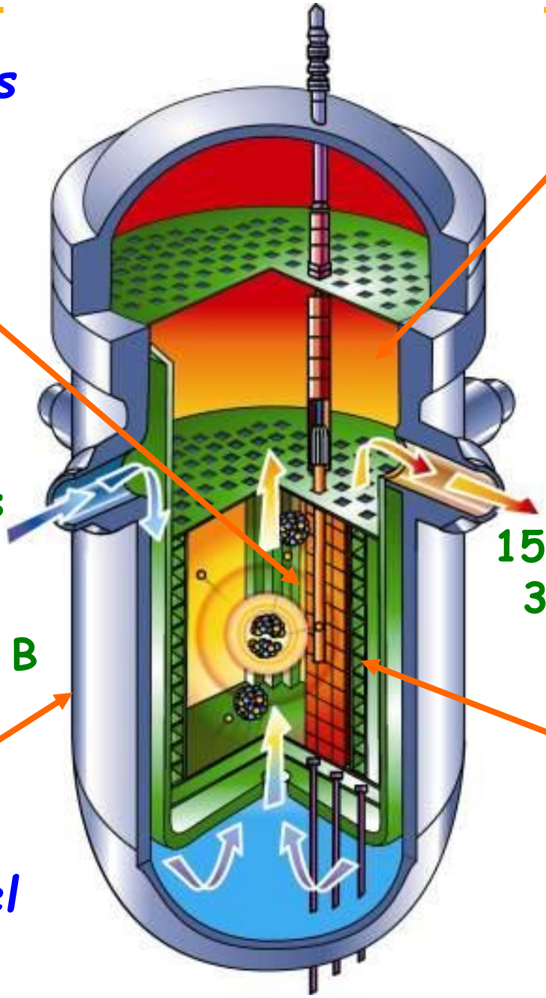

ELECNUC

Les centrales nucléaires dans le monde

Nuclear power plants in the world

édition 2018

PWR Irradiated Components



Fuel Assemblies
Zr alloys

300 – 400°C
10/15 dpa
5 – 6 years

Core Internals
Nickel alloys

~ 320°C
few 0.1 dpa
40 ➔ 60 years

Control rods
Austenitic steels

~ 320°C
~ 10 dpa
few years

155 bars
293°C
Water
H_2, LiOH, B

155 bars
328°C

~ 300°C
0.1 dpa
40 ➔ 60 years

Vessel
Bainitic steel
16MND5
A508 Cl 3

Core Internals
Austenitic steels

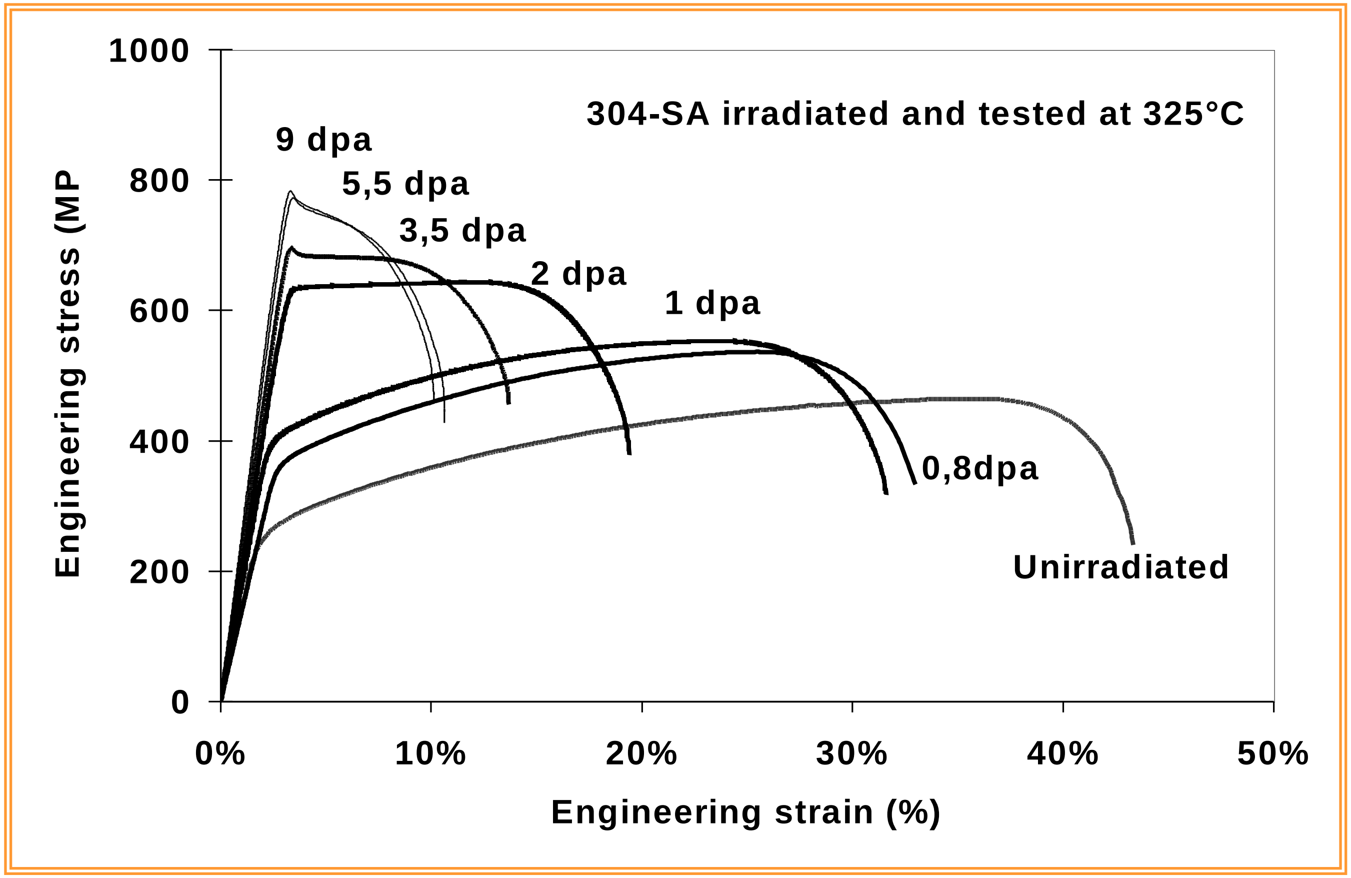
300 – 380°C
30 - 120 dpa
40 ➔ 60 years

- **neutrons**
- **temperature**
- **mechanical stresses**
- **environment**

Macroscopic effects of irradiation on materials

After irradiation an evolution of mechanical properties can be observed

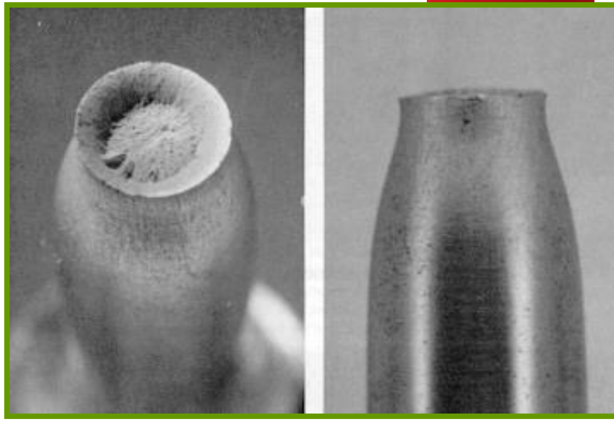
For instance the tensile testing properties of steel



Mechanical behavior : tensile test

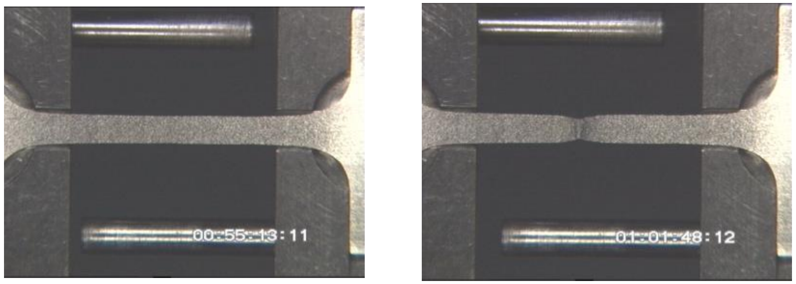
Low stress

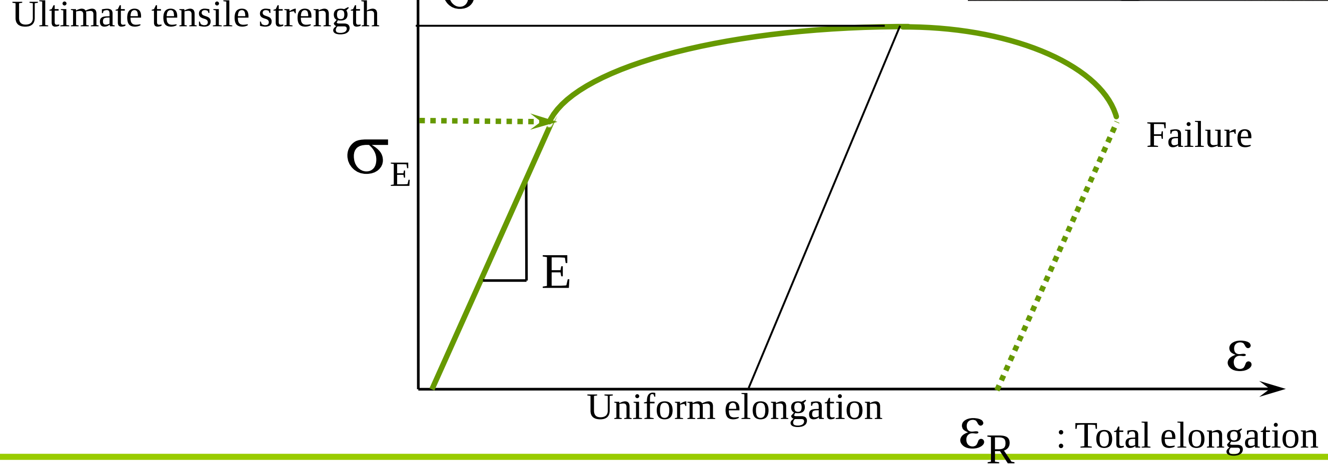
- Elastic deformation (fully reversible)
- $\varepsilon = \sigma/E$
  - σ stress
  - E modulus of Elasticity (Young modulus)



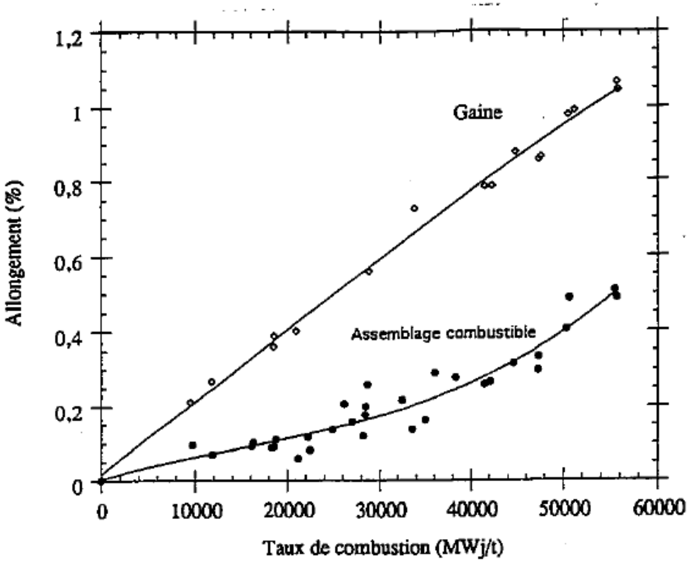
Yield strength σ_Y

- Irreversible strain induced by dislocation glide





Macroscopic effects of irradiation on materials



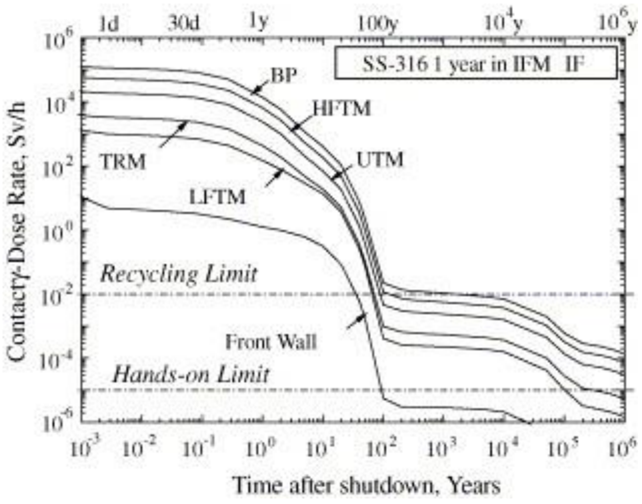
Dimensional changes

With or without stress

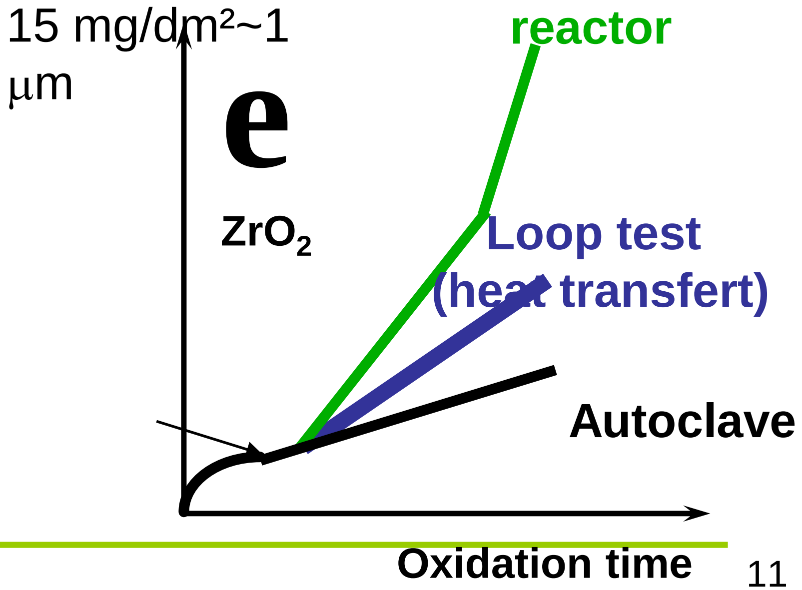
Corrosion

Creation of H_2 and He

en



Activation



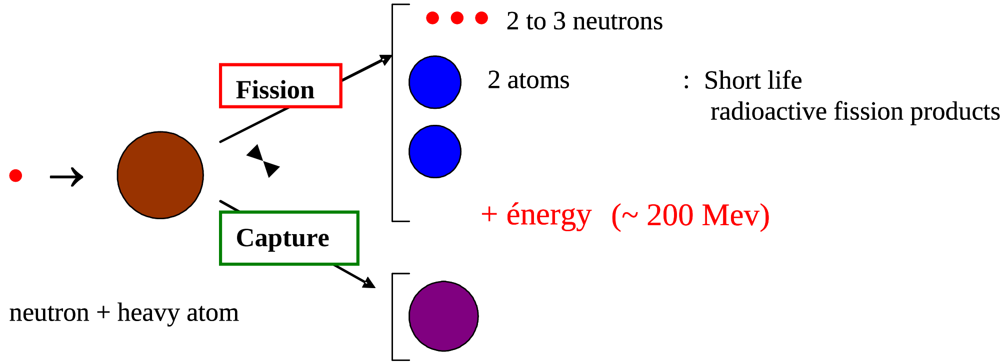
Effect of neutrons

Depending on their energies, neutron can have

Nuclear effects (inelastic): - thermal neutrons

Fission

Capture (and subsequent nuclear reactions)



2 to 3 neutrons

Fission

2 atoms : Short life
radioactive fission products

+ énergy (~ 200 Mev)

Capture

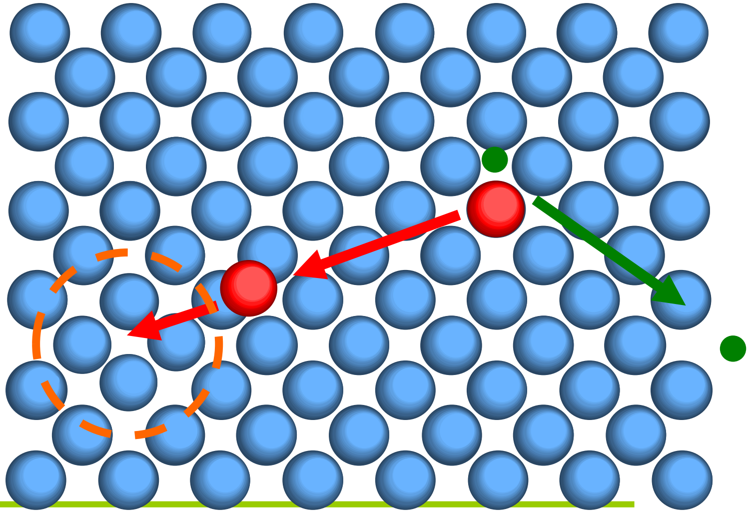
neutron + heavy atom

Ballistic effects (energy conservation) – fast neutrons

dpa – point defects

Effect of elastic collisions inside a crystal :

- For a transferred energy $E_t < E_d$ (threshold energy) -> vibration of the crystal lattice -> heating
- For a transferred energy $E_t > E_d$, the atom can be ejected from its atomic site and move through the crystal to other atomic sites (mean free path ~ several atomic sites)
- This creates a vacancy + a self-interstitial atom. This is a Frenkel pair.



PKA : Primary Knock on Atom

Effect of elastic collisions inside a crystal :

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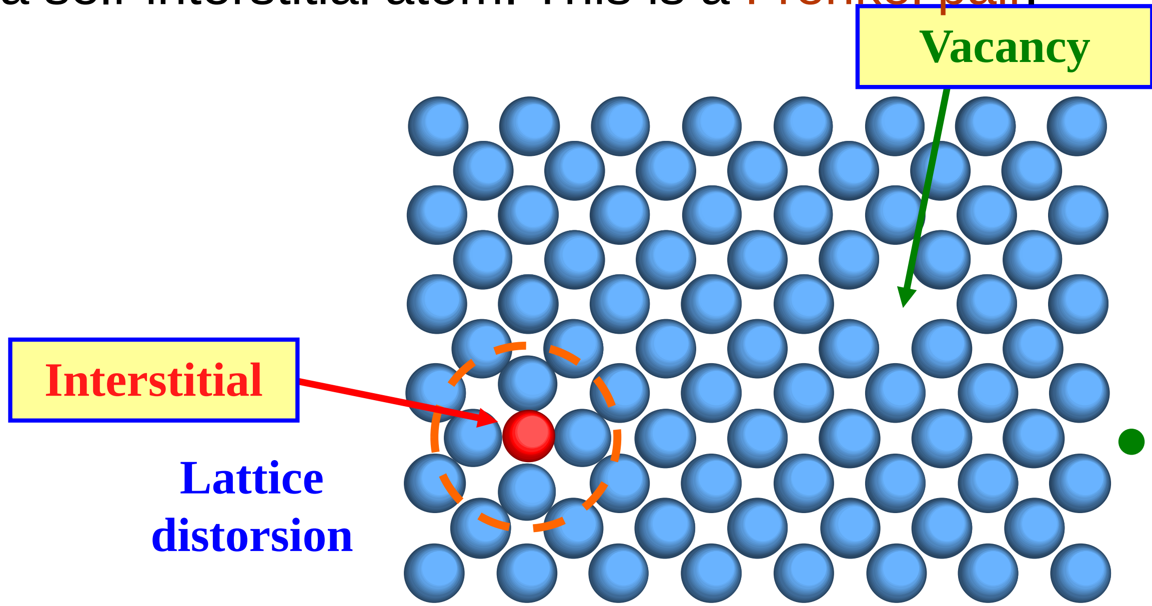


Neutron spectrum and units of irradiation

The neutron of different energies have different effects on the materials

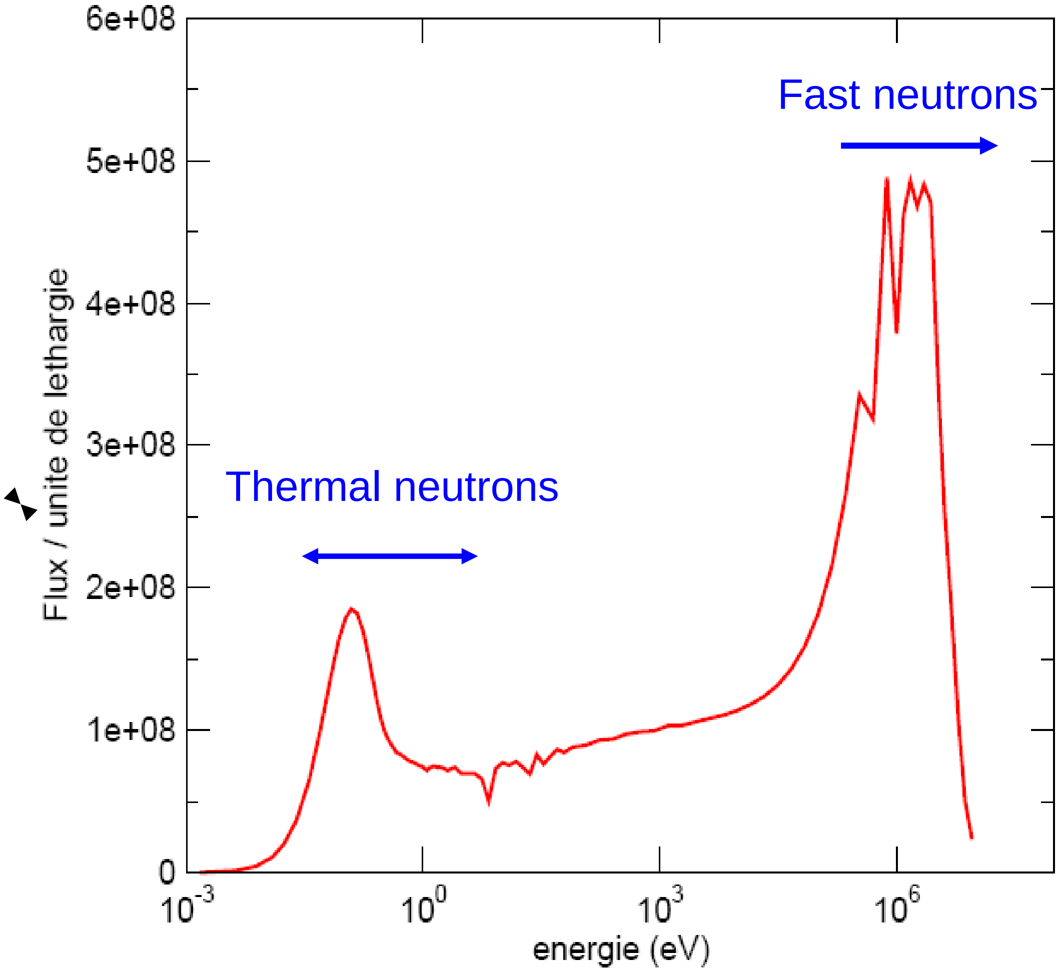
Unit of flux : n.cm^{-2}.s^{-1}

Unit of fluence: n.cm^{-2}

Difficult to describe a spectrum by just one number

Therefore use of the dpa (displacement per atom)

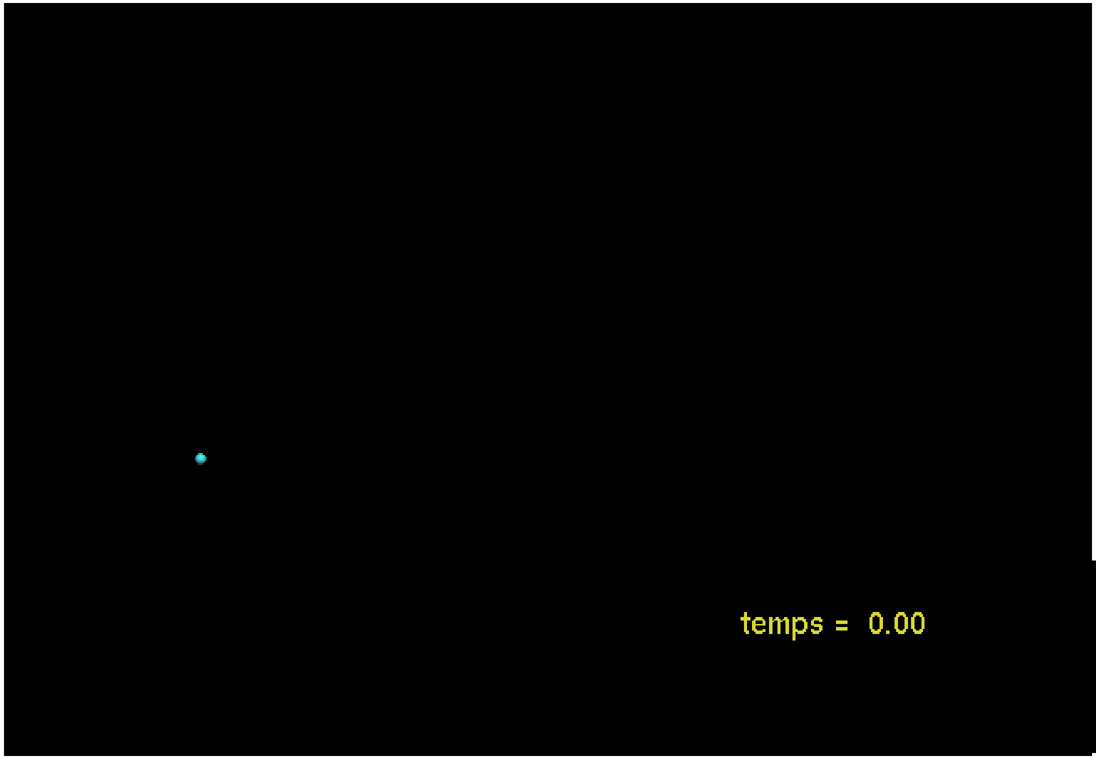
PWR neutron spectrum



Displacement cascade ($E_t >> E_d$)

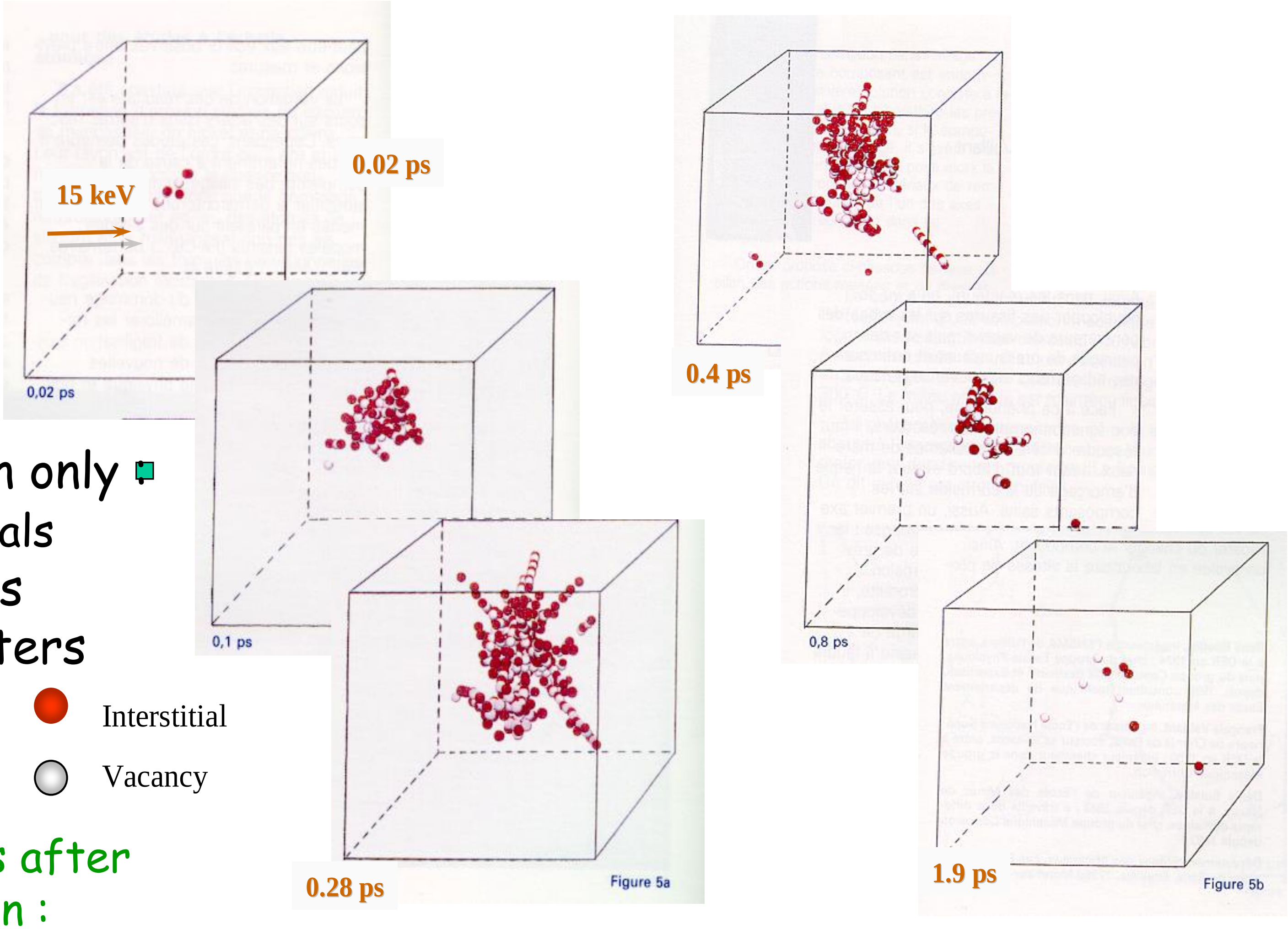
- For a transferred energy large compared to E_d, the ejected atom transfers part of its energy to other atoms of the crystal lattice...

  ... these other atoms can then displace other atoms.
- The primary knock on atom induces a displacement cascade



Vacancy : yellow
Interstitials : red
Displaced atoms : blue

Cascade evolution



After 10 ps, remain only

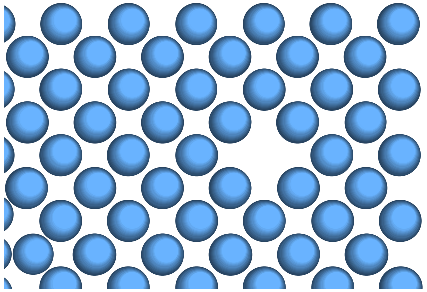
- Isolated interstitials
- Isolated vacancies
- Interstitials clusters
- Vacancy clusters

Few surviving defects after cascade relaxation : only 1/100 of displaced atoms remain

-> Cascade creation (in agreement with KP)

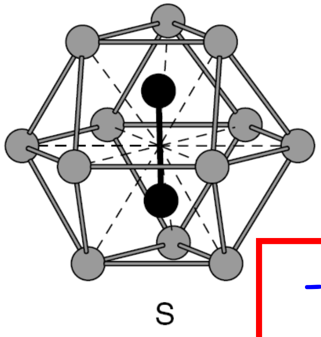
-> Cascade relaxation recombinations

Formation energy of point defects



Vacancy formation energy :

low distorsion of the crystal lattice -> low formation energy



interstitial formation energy :

High distorsion of the crystal lattice -> high formation energy

-> Out of reactor it is much easier to create a vacancy than a self-interstitial

Point defect concentration at thermodynamic equilibrium

$$c_{DPe} = \exp\left(-\frac{H_{DP}^{f}}{kT} \right)$$

Formation energy

-> No interstitial at thermodynamic equilibrium

2 important phénomena

Irradiation damage : creation of point defects and mixing of the atoms (dpa)

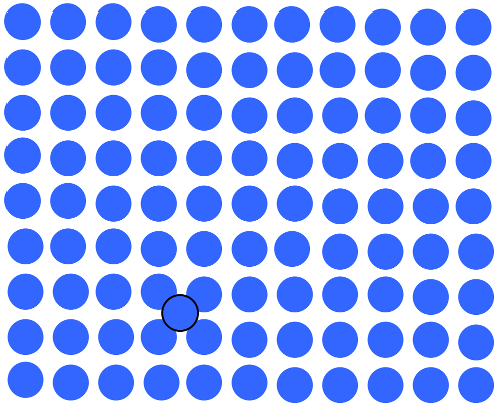
But also

Evolution of these point defects

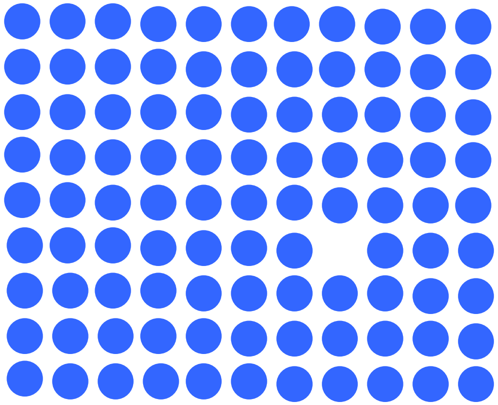
- Recombination
- Clustering
- Annihilation on sinks

Mobility of point defects

Interstitial



Vacancy



$$D_{DP} = D_{DP0} \exp\left(-\frac{H^{m}_{DP}}{kT}\right)$$

	Vacancy	Interstitial
Formation energy (eV)	1.4-2.1	2.8-3.5
Migration energy (eV)	0.5-0.9	Ea=0.01-0.06 Ec=0.1-0.3

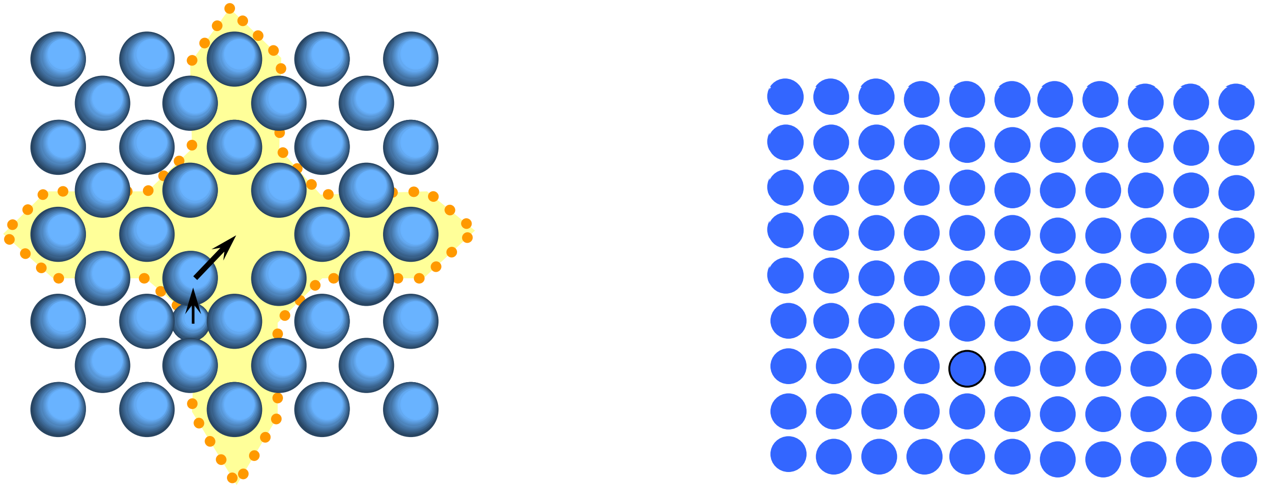
Anisotropic diffusion of the self-interstitial

-> High mobility for interstitials, low mobility for vacancies

Spontaneous mutual recombination

A self-interstitial close to a vacancy reacts spontaneously and the two defects are annihilated

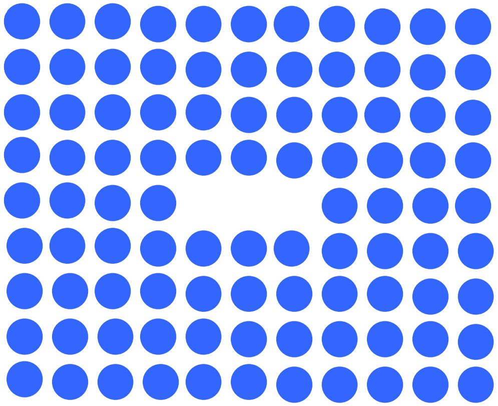
- The mutual recombination volume is about 100 atomes.
- Saturation of the point defects (if no other sink) due to mutual recombination

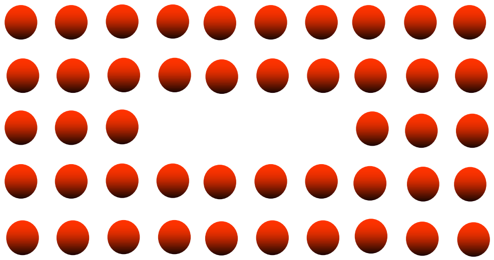


Mutual recombination :

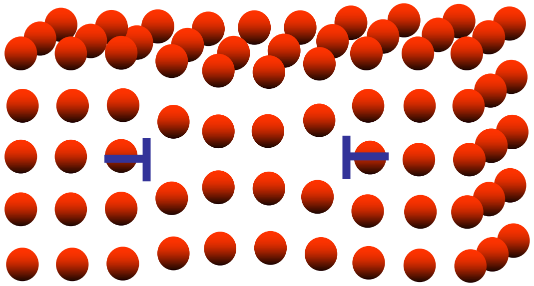
□ + ⚲ = nothing (if pure material)

Point defect clustering

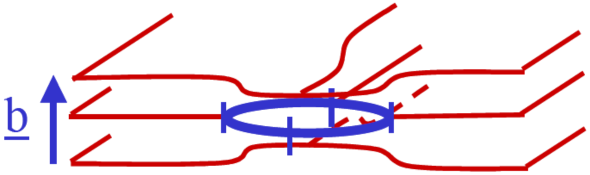




Vacancy disk

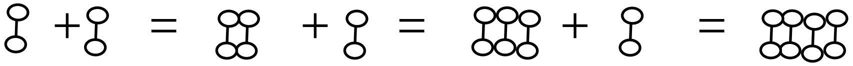


= dislocation loop



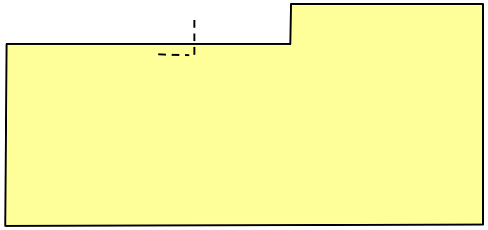
Point defects clustering : loop, cavity

□ + □ = □□ + □ = □□□ + □ = □□□□ or ⊞ -> Point defect cluster

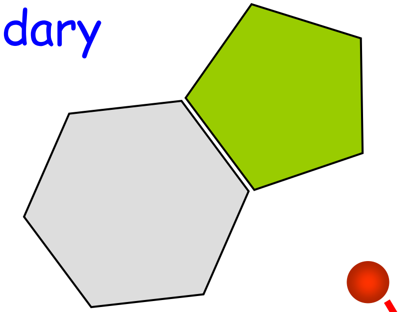


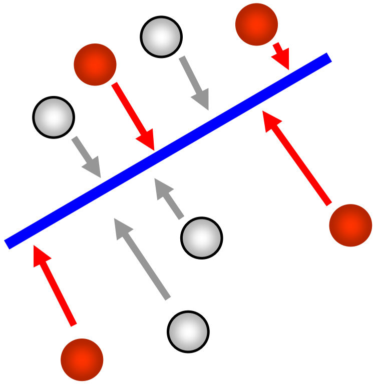
Annihilation on other sinks

Surface



Grain boundary



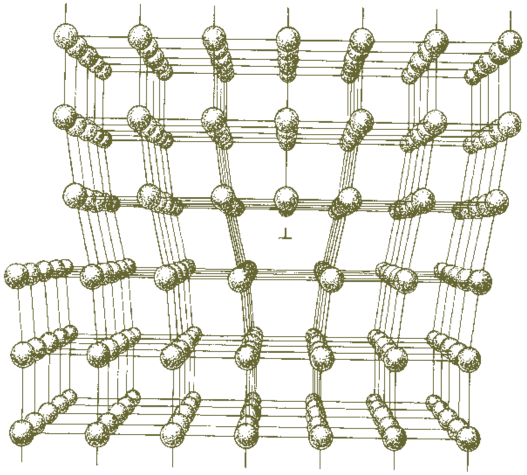


Close to the sink : equilibrium concentration of point defects

Sur-saturation of point defects due to irradiation

-> point defect flux towards the sinks

Dislocations

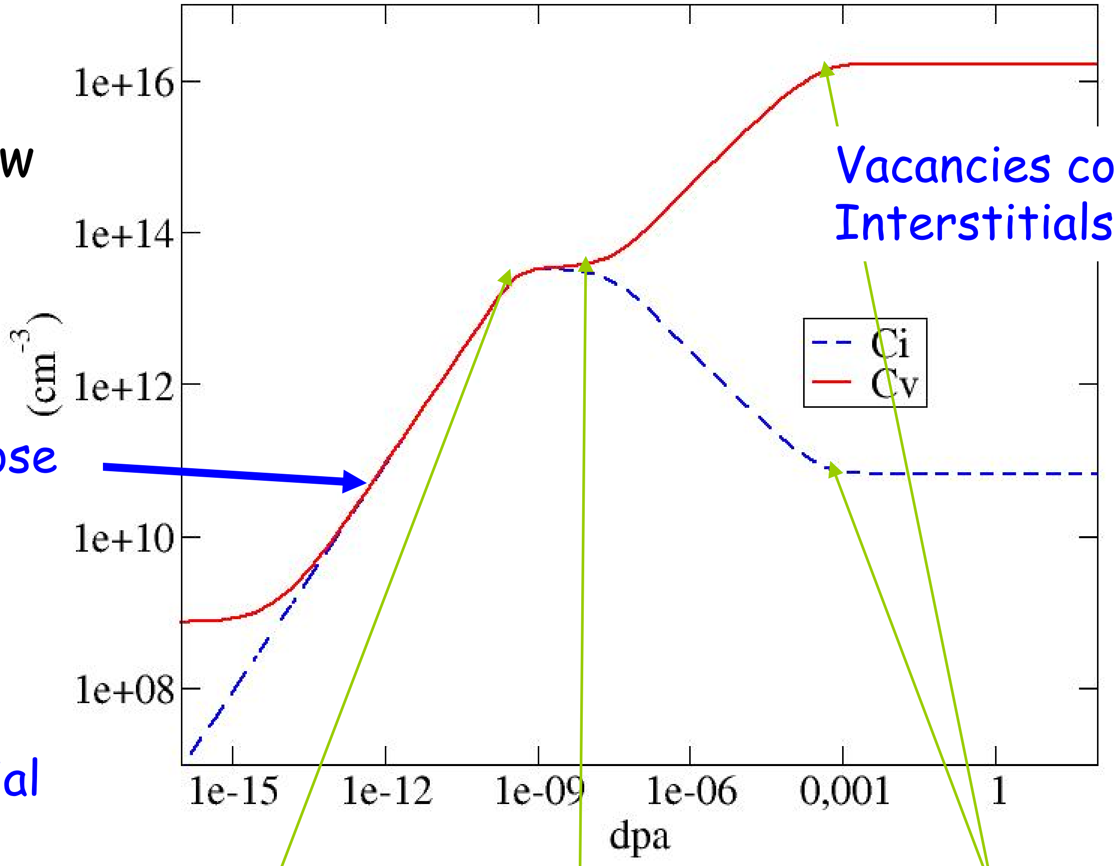


- High interaction between interstitial-dislocation
- Low interaction between vacancy-dislocation
- -> Higher chemical force on interstitials -> higher interstitial flux towards dislocations (if other types of sinks, such as grain boundaries)

Evolution of the point defects concentration

Iron, 2 10^{-8}dpa/s, 300°C, low sink density.

Hypothesis : Low sink density



Steady state

Vacancies concentration >> Interstitials concentration

Increase with dose

Thermodynamic equilibrium vacancies

No interstitial

Start of recombinations

Interstitials reach sinks

Because of a higher mobility

Vacancies reach sinks

Effect of the irradiation temperature

- There is competition between the creation of point defects and their elimination

•At low temperature, the point defect mobility is reduced. High density of point defects. The point defects are mainly annihilated by mutual recombination.

- At high temperature, the vacancies concentration at thermodynamic equilibrium is high, the increase due to irradiation in point defects is low.

When the temperature increases, so does the mobility of defects. Therefore, the system tends to go back to the equilibrium state

Beware of accumulation of energy

Microstructural evolution

<u>Effect of elastic collisions on precipitates</u>

- Amorphization
- Dissolution

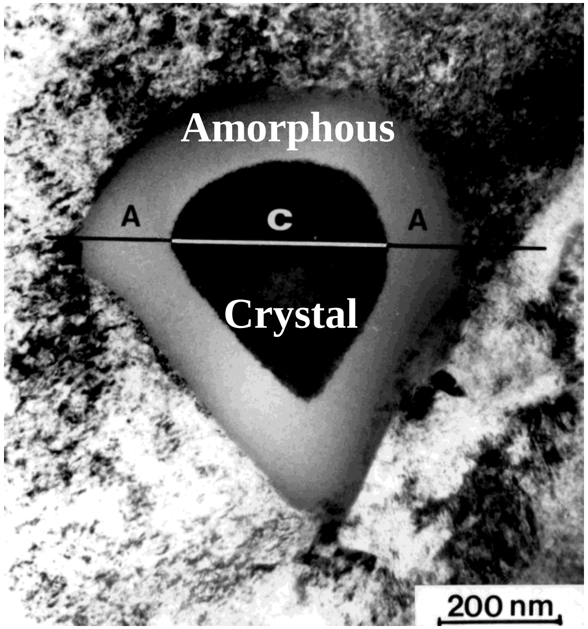
<u>Enhanced diffusion due to the super-saturation of point defects</u>

- Precipitation

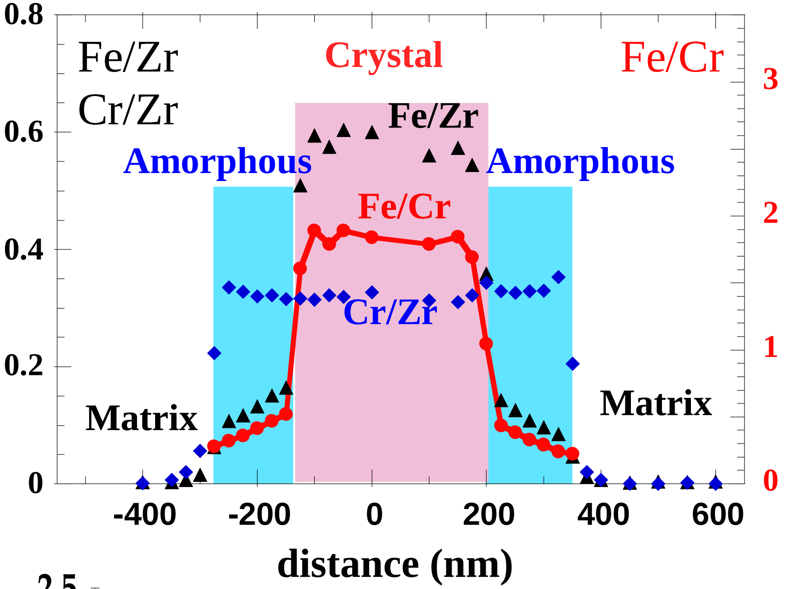
<u>Nuclear reactions</u>

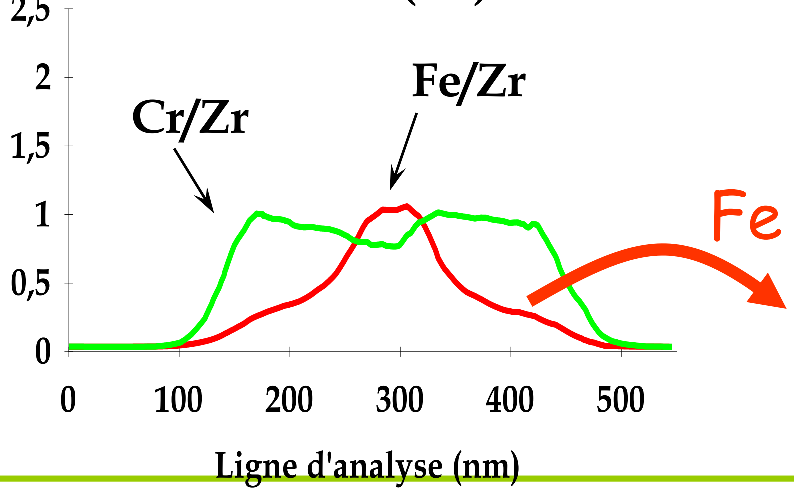
- Changes in chemical composition

Amorphization of precipitates



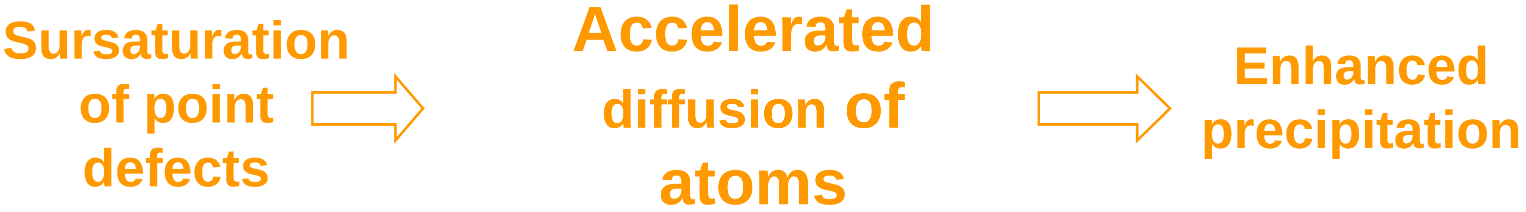
Laves phases
$(Fe,Cr)_2Zr$





Amorphization of the precipitates
Dissolution of Fe into the matrix

Enhanced precipitation accelerated by irradiation

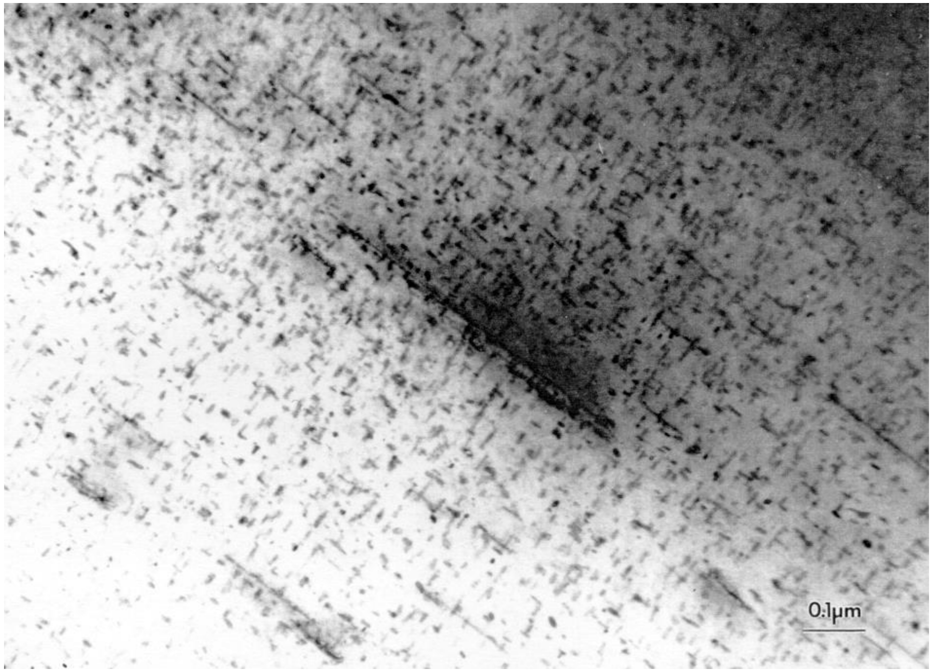


Sursaturation of point defects → Accelerated diffusion of atoms → Enhanced precipitation

Precipitation of β Nb under irradiation in Zr alloys with Nb

Zr-Nb alloys, solide solution super-saturated in Nb (out of thermodynamic equilibrium).

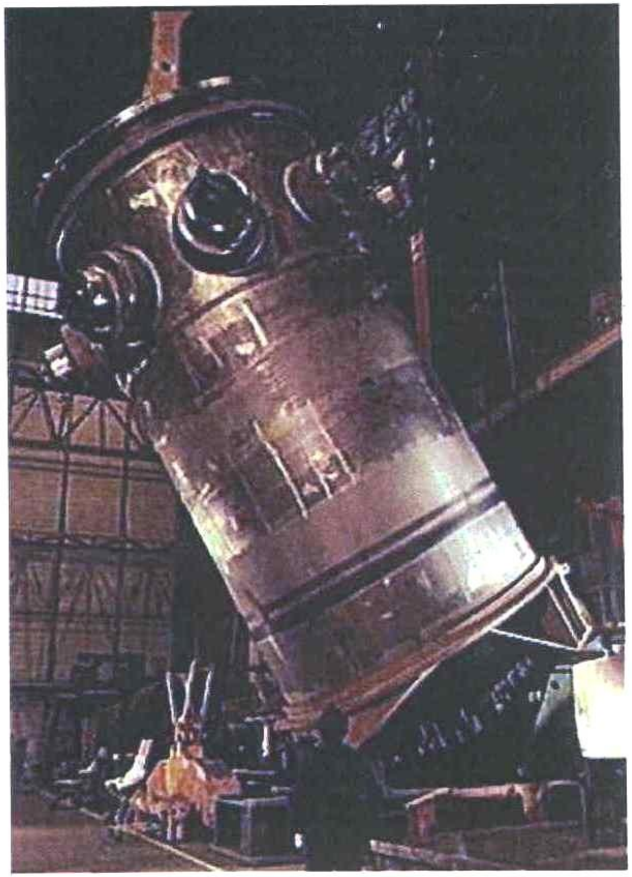
Under irradiation : the high vacancies concentration leads to a high vacancy flux that enable a fast return to thermodynamic equilibrium, and re-precipitation of β-Nb

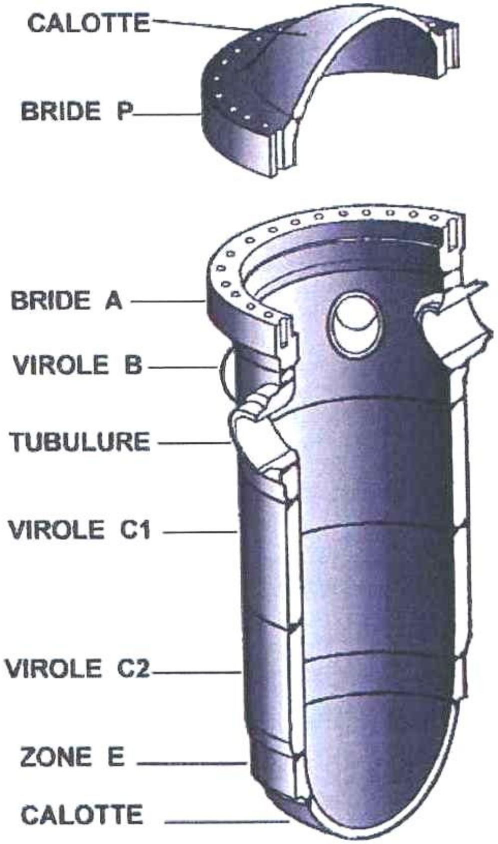


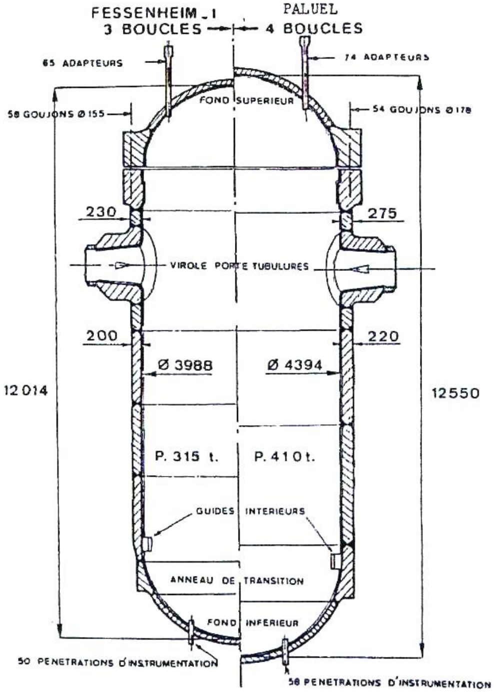
Long term evolution of point defects

- Point defects : vacancies and interstitials
- Under irradiation : creation of interstitial and vacancies (high V concentration)
- High mobility of interstitial low mobility of vacancy
- Anisotropic diffusion of the interstitials
- Vacancy and interstitial recombination
- Point defects clustering : loops
- Point defects elimination at sinks (dislocation, grain boundaries + loops)
- Complex evolution, depends on all the sinks present in the material, on temperature, on stress

The reactor pressure vessel







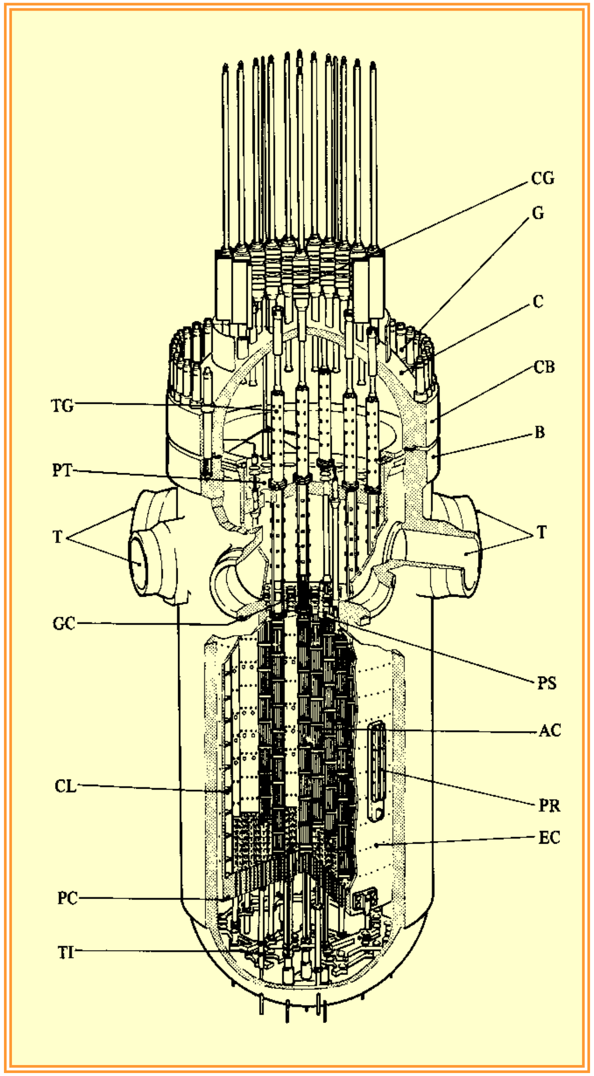
Functions of the reactor pressure vessel

The reactor pressure vessel is the second safety barrier

The pressure vessel is the only component which cannot be replaced

it has to keep its functions for the lifetime of the plant (environment, irradiation damage) in operating conditions, but also in accidental conditions…

PWR pressure vessel



Temperatures : 296 – 320°C

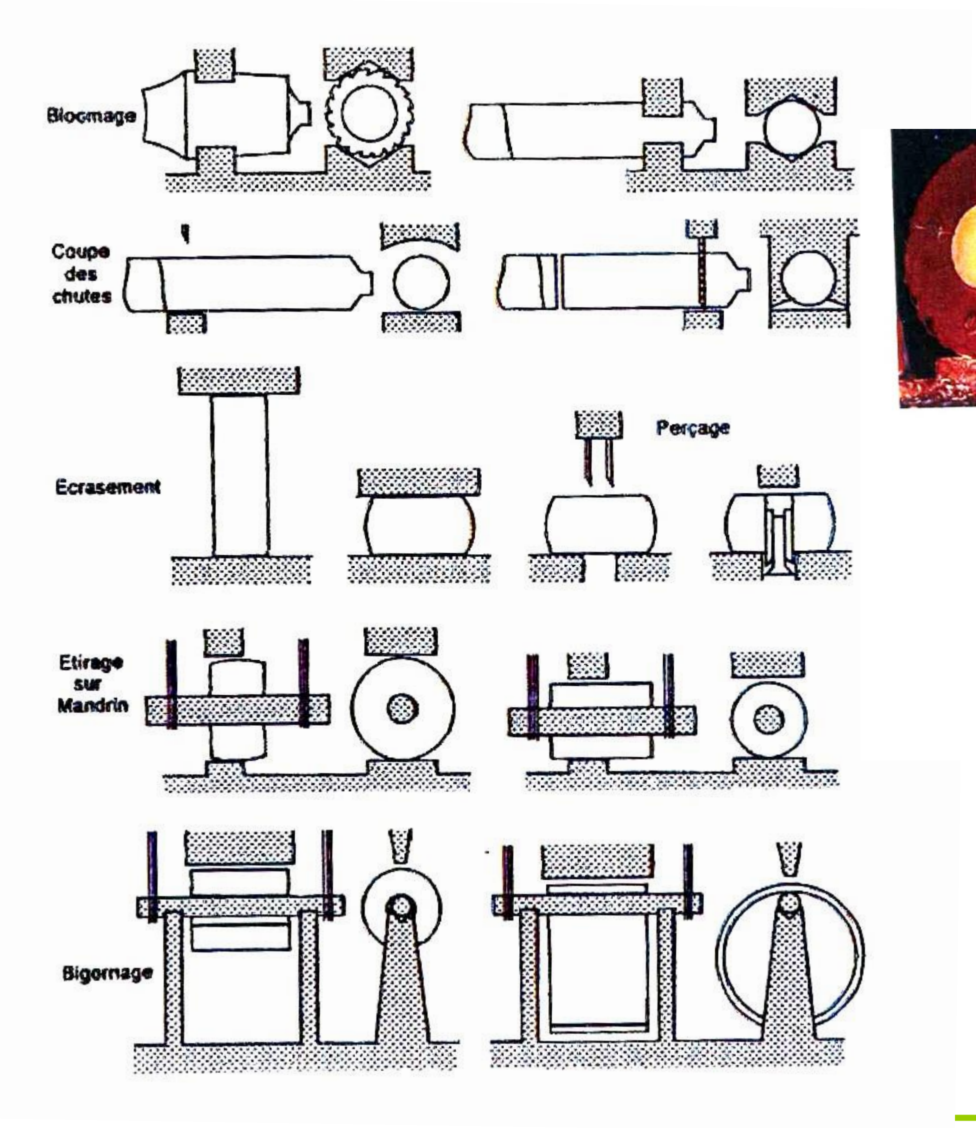
Coolant pressure: 155 bar

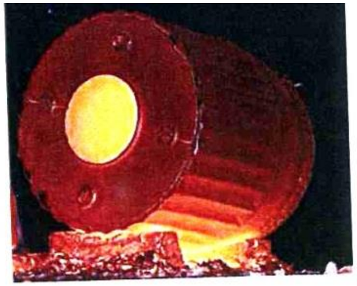
∅ = 4400 mm
e = 220 mm

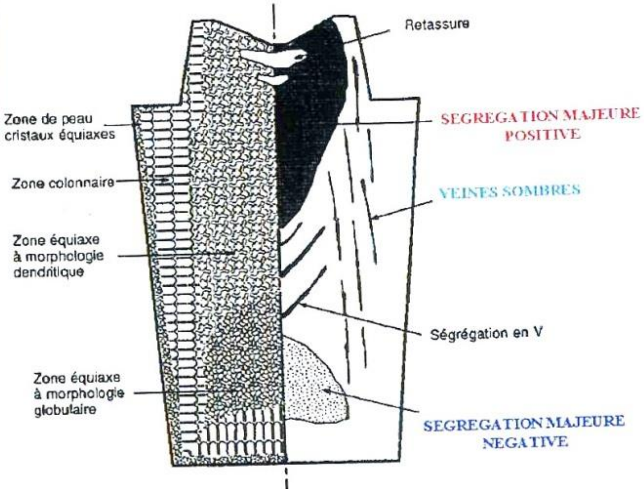
Gross weight 330 (900 MW) 440 t (1300 MW) EPR (520 t)

Steel A508 Cl. 3 = 16MND5 (bainitic steel)
Internal cladding: 304 L = Z2CN1810 (austenitic stainless steel)

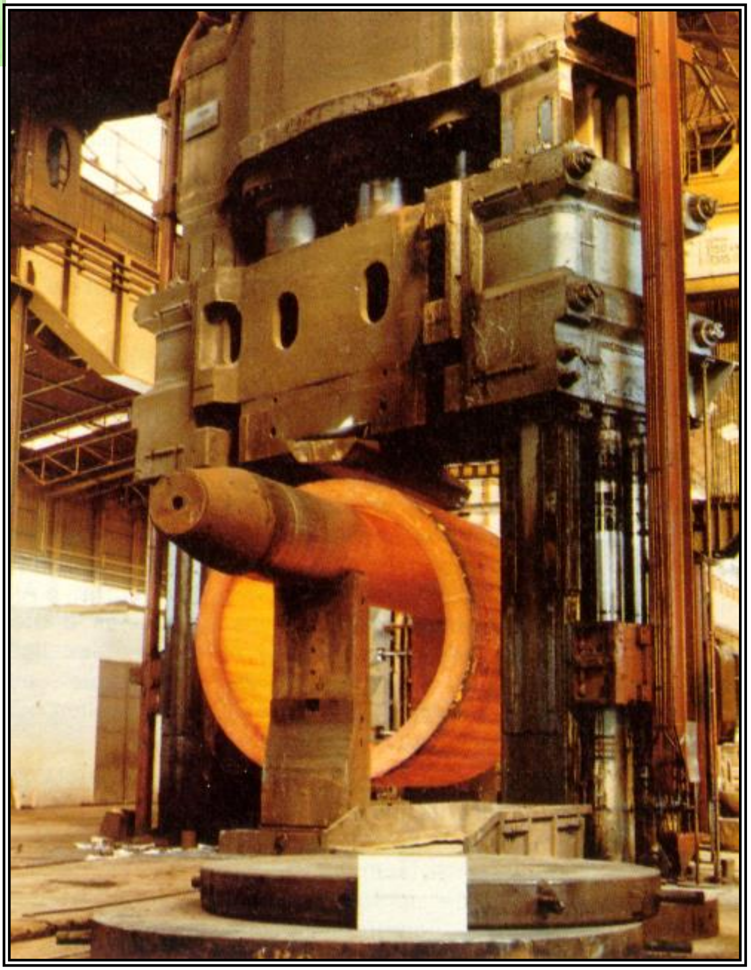
Fabrication process





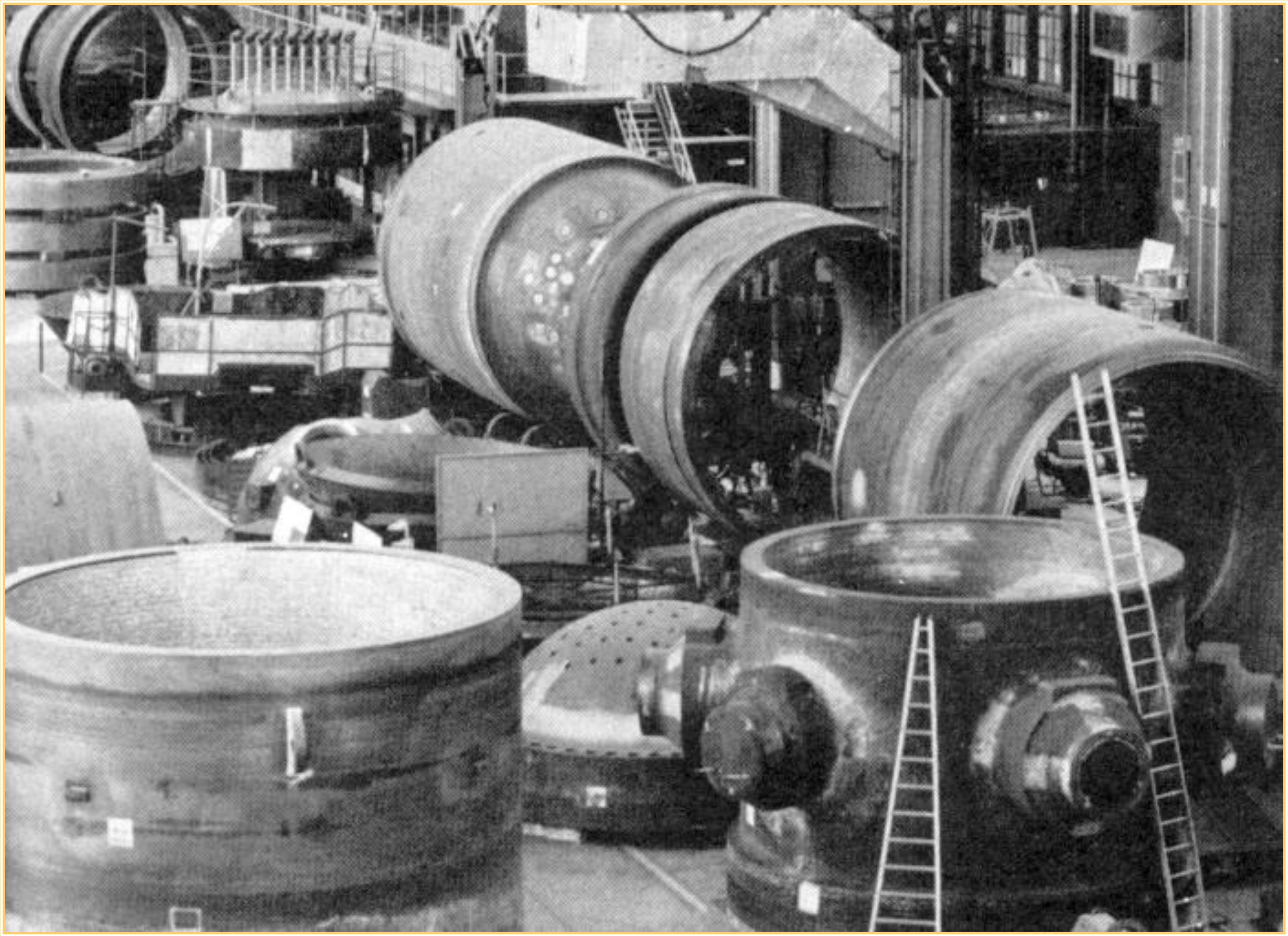


Forming of ingots



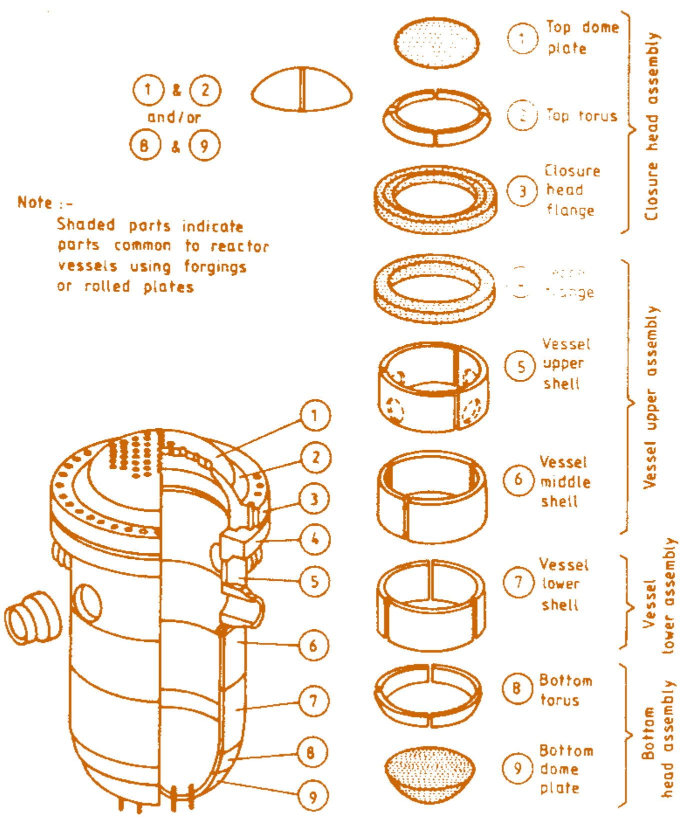
- Forging at 1100 - 1200°C
- Deformation ratio > 3
- Intermediate re-heating
- End of forming: hold at 600-650°C (H diffusion)
- Final heat treatments
  - γ -quench 850-920°C
  - tempered 635-665°C

Forged sections are welded



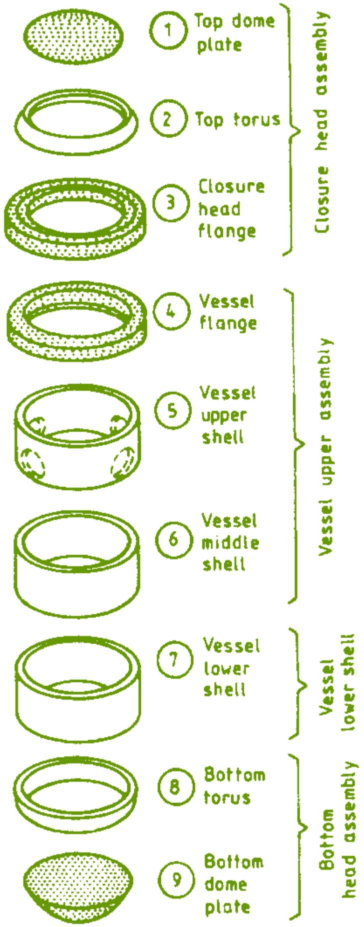
Assembly of the vessel

Rolled and welded flats



Westinghouse

Forged cylindrical rings



Le Creusot

PWR pressure vessel steels

Ferritic steel (bcc)
16 MnNiMo5 (A 508 Cl 3)

Low carbon (easy welding)
Strength by other elements, Mn….and structure (bainitic)

C_{max}	Mn	Ni	Mo	Cr_{max}	Si
0.2	1.15/1.55	0.5/.8	0.45/.55	0.25	0.1/.3

	P_{max}	S_{max}	Cu_{max}	Co
ppm	80	80	800	300

Low S, P et Cu to avoid irradiation induced embrittlement
Low Co to avoid γ irradiation

Internal cladding for vessel

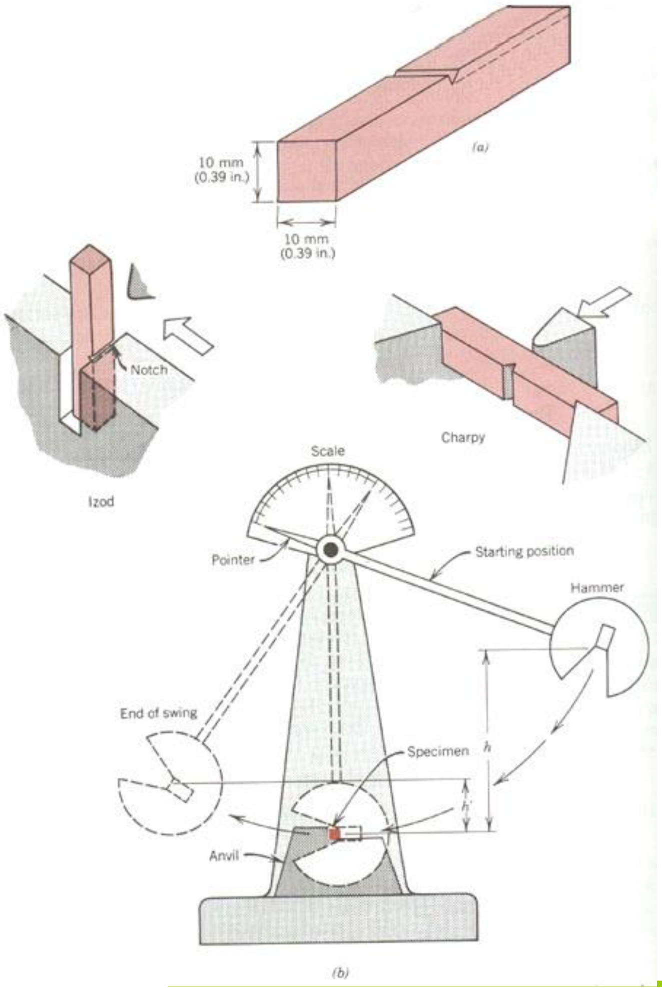
- To protect against the corrosion of the primary coolant: welding of a stainless steel layer (*beurrage in french*)
- Two layers are welded from planar sheets
  - 24 Cr, 12 Ni (Cr rich to compensate the loss due to dilution with base metal melted)
  - 20 Cr, 10 Ni (other layer similar to 304L-316L)
- Thickness: 8 mm (vessel steel: 220 mm)
- Formation of cracks under the layer
  - Cracking at low temperature (during the cooling, contamination due to H diffusion during the welding)
  - Pre- and post-heating

Introduction to fracture mechanics

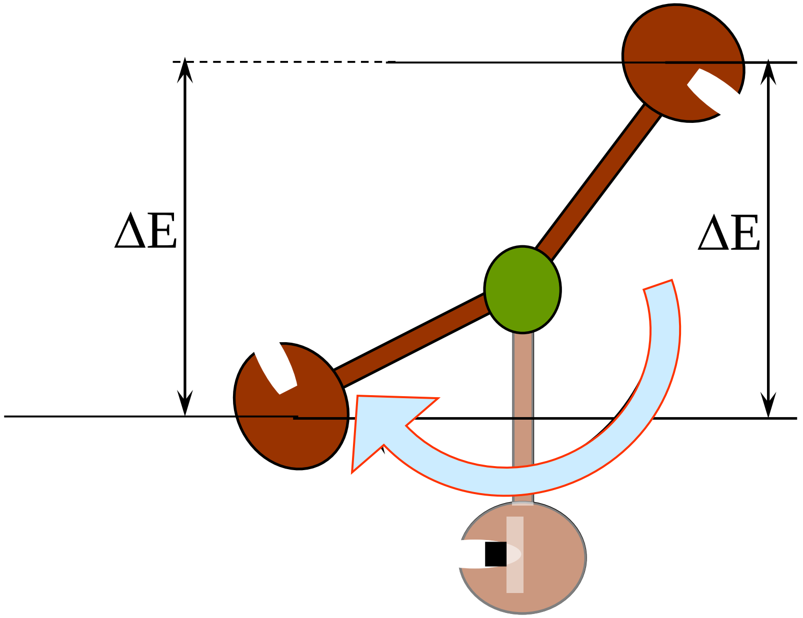
- **Charpy**
  - Indication of a tendency to brittle fracture (resistance to cracking)
  - Small samples
  - Easy irradiation
- **Fracture toughness** (K_{IC})
  - Mechanical value of a resistance to crack propagation (design)
  - Much larger testing samples

Brittle - ductile transition in ferritic steels (bcc)

Quantification of the resistance to cracking

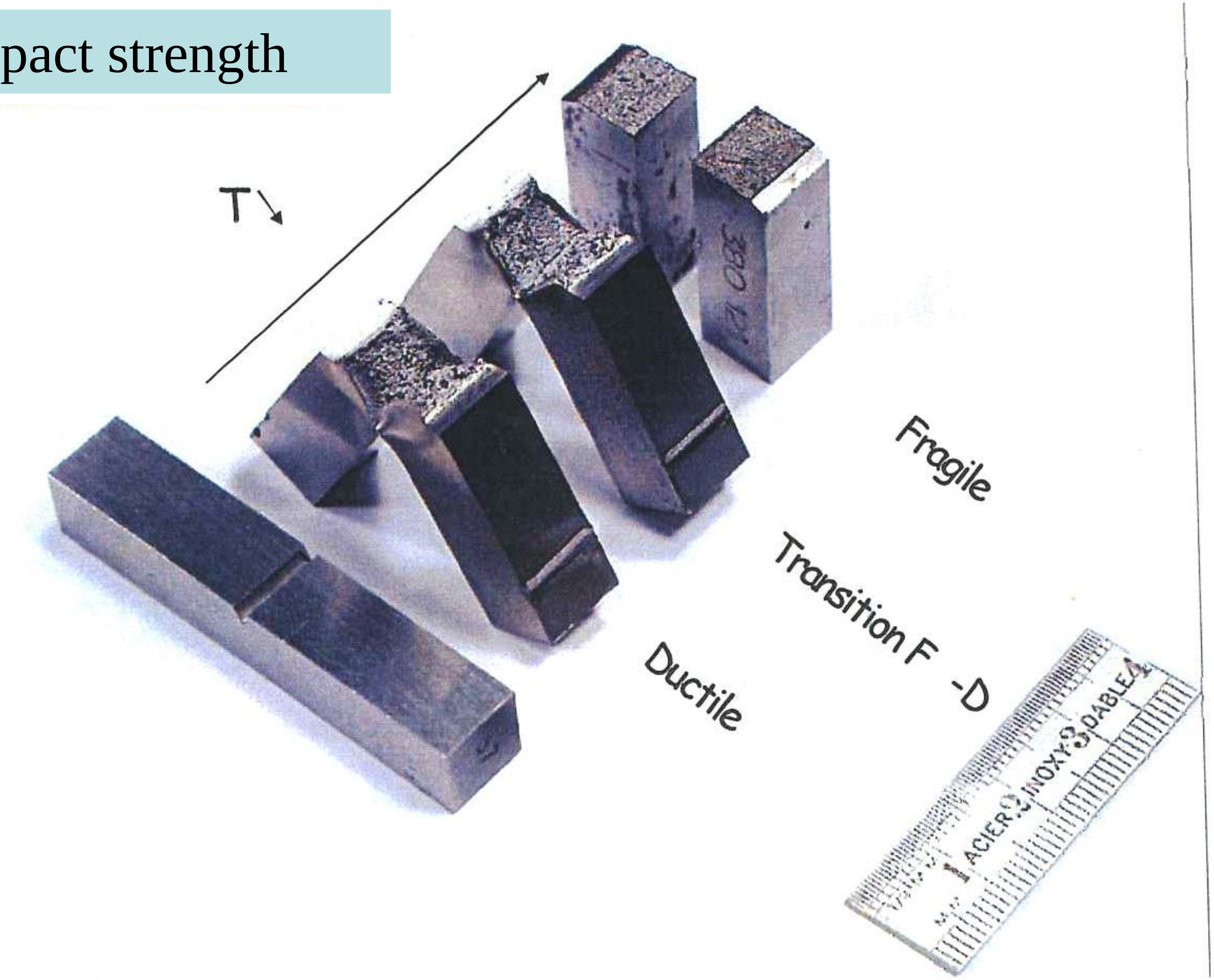


Measurement of the energy absorbed during the test



Charpy samples

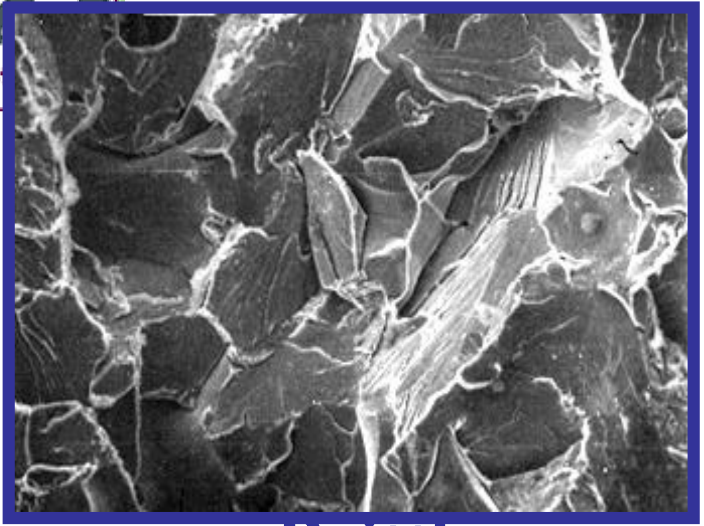
Impact strength



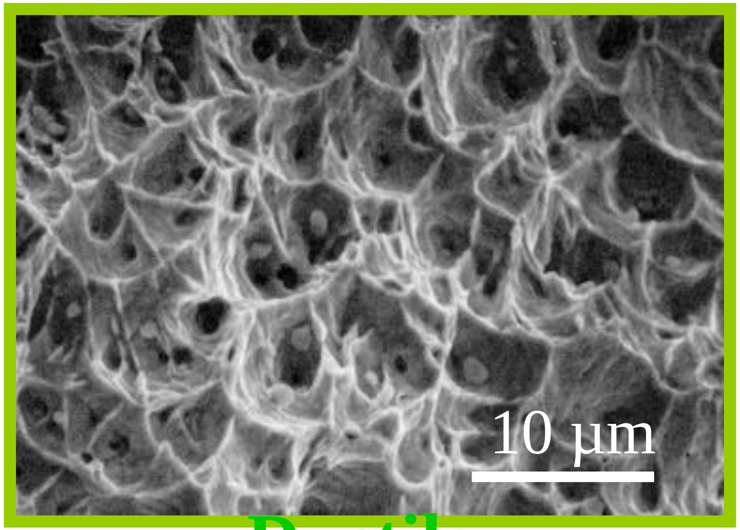
CHARPY

Test

Fracture surfaces after Charpy impact test



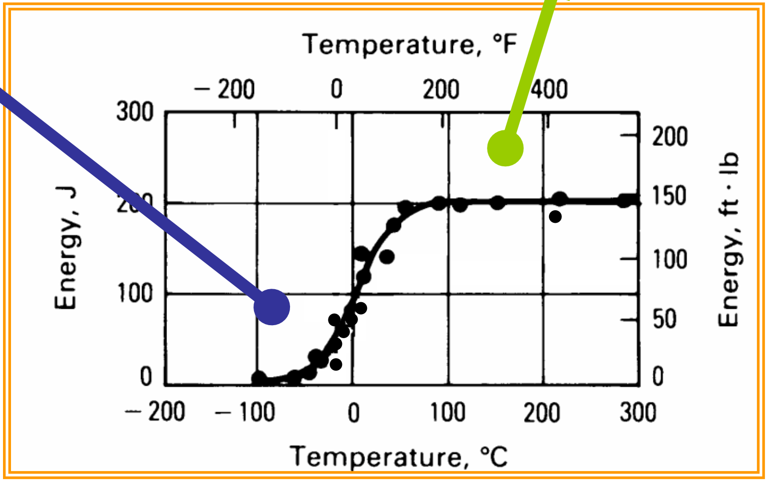
Brittle



Ductile

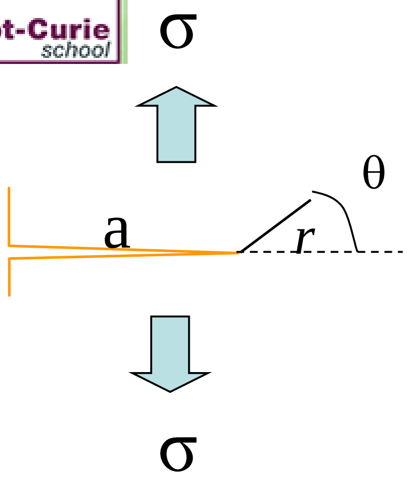
Ductile : plastic deformation and rupture after coalescence of cupules, high energy absorbed during the tests

Brittle : cleavage = de-cohesion of the crystal along specific crystallographic planes, low energy absorbed during the test before rupture



Evolution of the absorbed energy during the test as a function of the temperature

Fracture toughness (K_{IC})



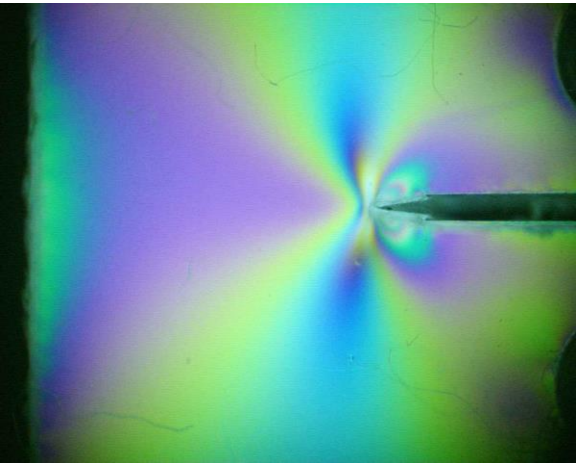
Resistance to crack propagation

At the root of a crack, the stress diverge => equal to infinity at the tip

$$\sigma_{loc} = \frac{K_I}{\sqrt{2\pi . r}} \cdot f(\theta) \qquad K_I = \sigma \cdot \sqrt{\pi \cdot a}$$

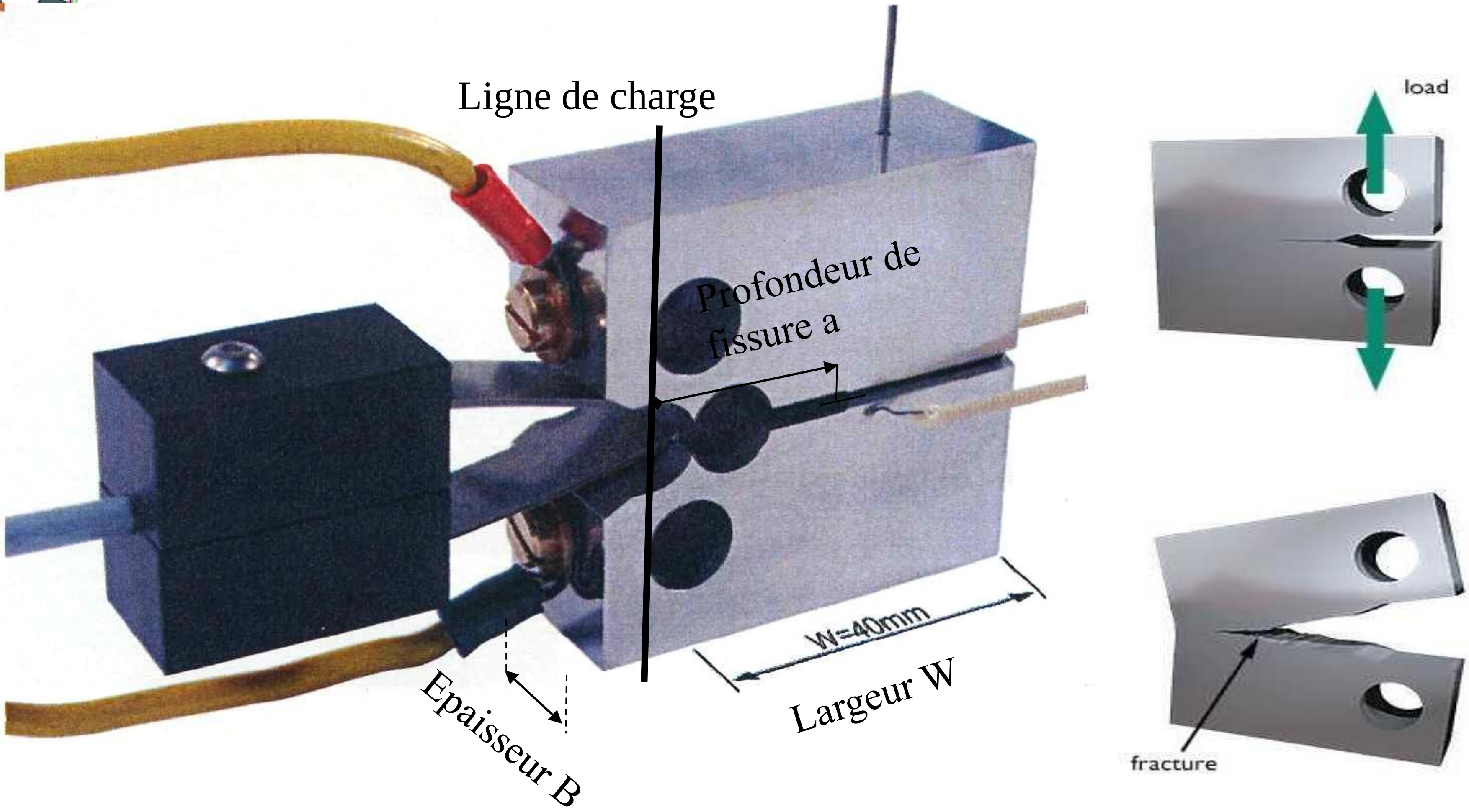
The stress intensity factor K_I allows to give a value to the singularity (in MPa.m$^{1/2}$)

The resistance of a material to crack propagation is measured by a toughness test. Fracture at $K_I = K_{IC}$ (ASTM E 399) (no propagation if $K_I < K_{IC}$, *if $K_I > K_{IC}$ than the crack propagates suddenly)*



Knowing the crack geometry and the stress state, allows to forecast crack propagation or not.

Fracture toughness: resistance to crack propagation

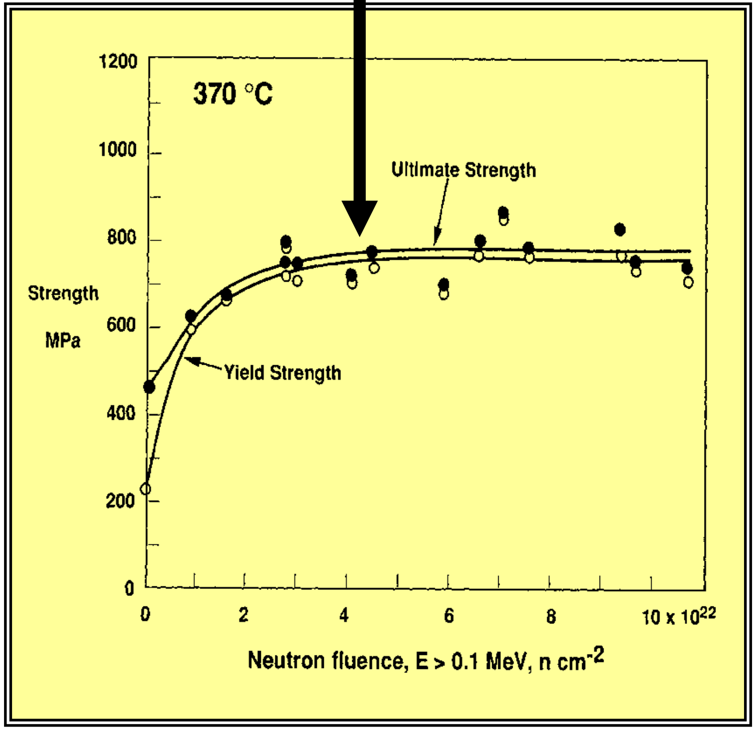


Increase in yield strength due to irradiation

- Pressure vessel irradiation: 10^{-10} dpa.s^{-1}

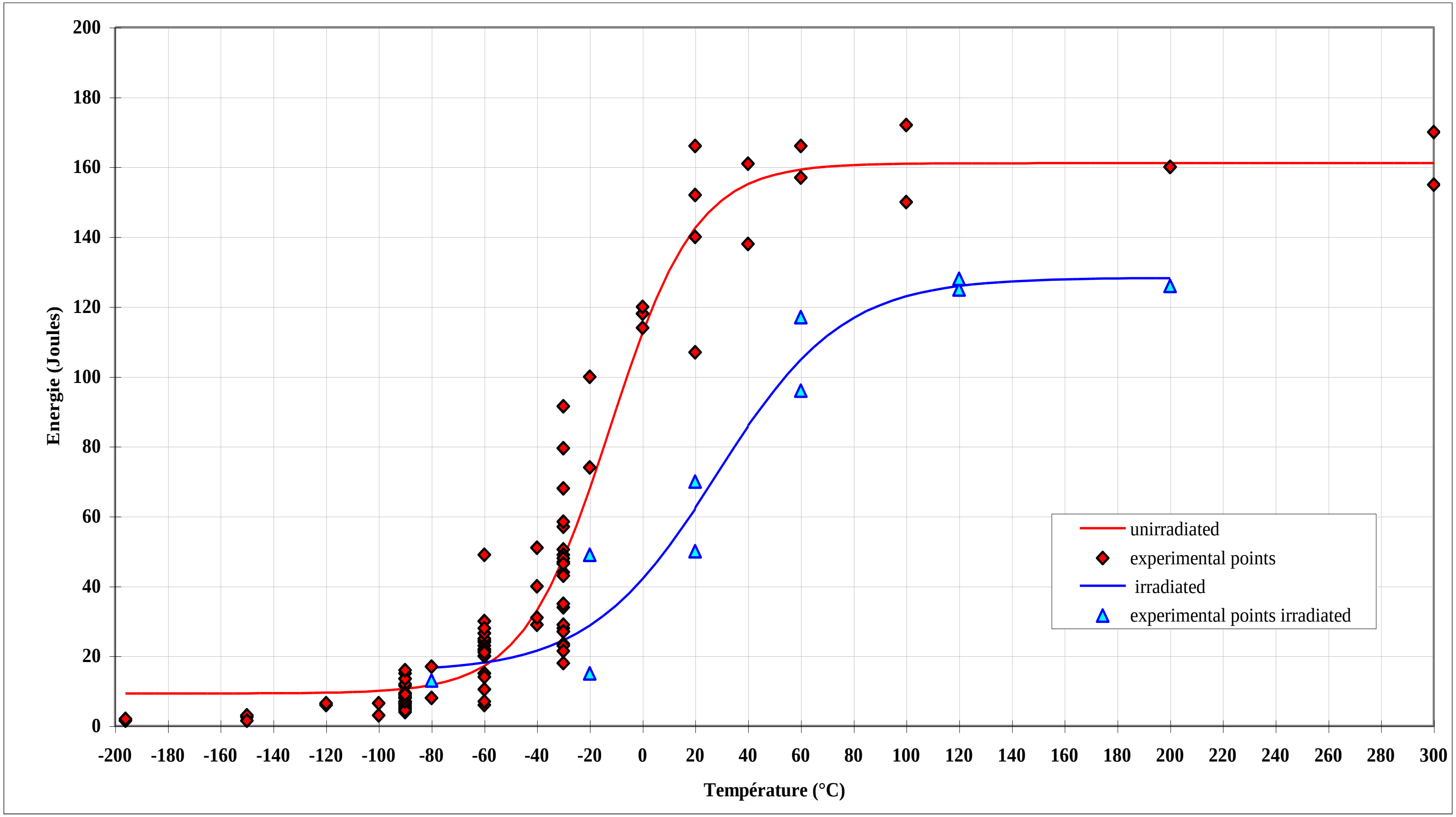
- Dislocation loop formation, and other defects like clusters of minor elements induce an increase in mechanical properties.

Saturation of the hardening due to the saturation of the microstructure (saturation of defects density)



Evolution of ductile to brittle transition temperature

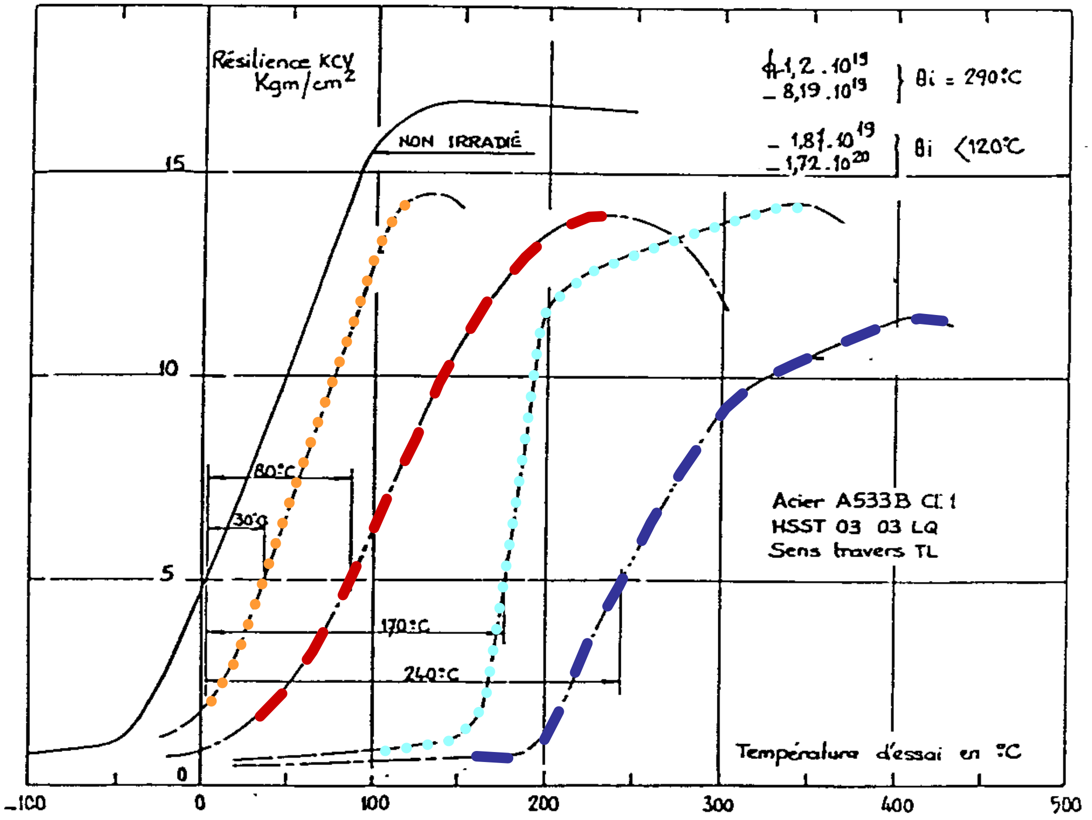
Higher strength increases the probability of failure by cleavage, leading to a higher transition temperature.



Impact of irradiation temperature

At higher temperature, continuous recovery occurs during irradiation and embrittlement is reduced.

For the same irradiation dose, the effect is higher at 120°C than at 290°C



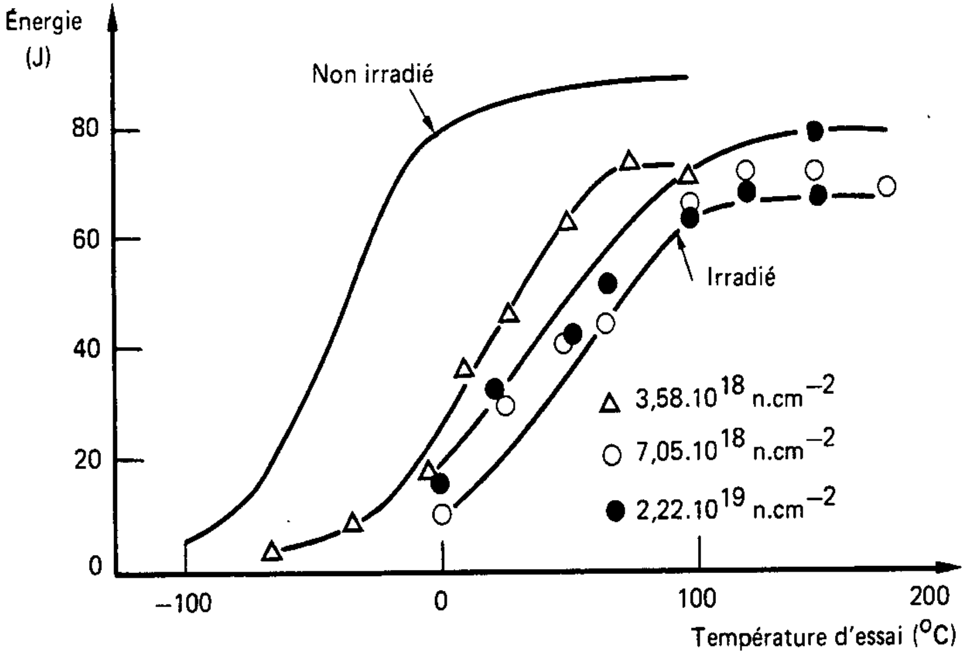
Tirr. =290°C

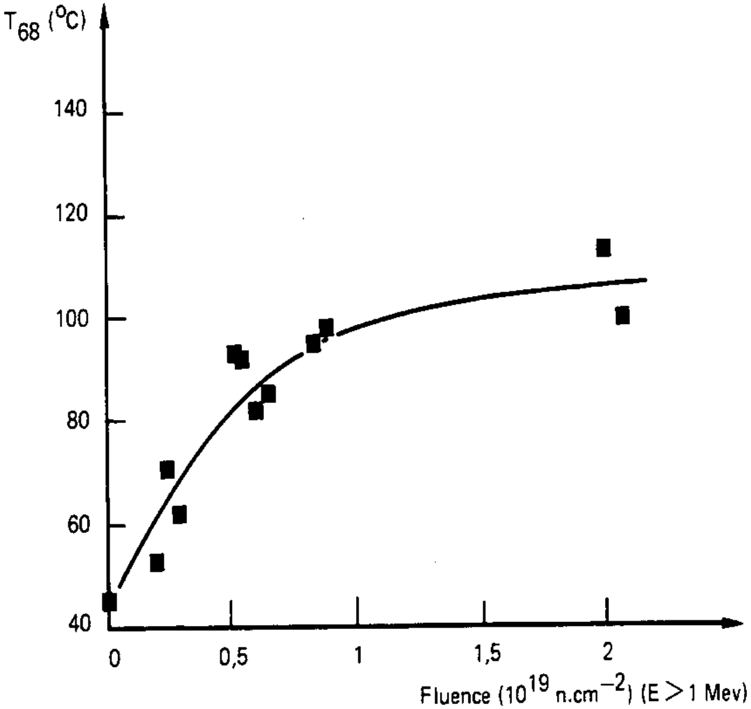
Tirr. =120°C : lower elimination of PD, less recombination, more hardening

Saturation of irradiation effects

Ductile Brittle Temperature Transition increase vs fluence

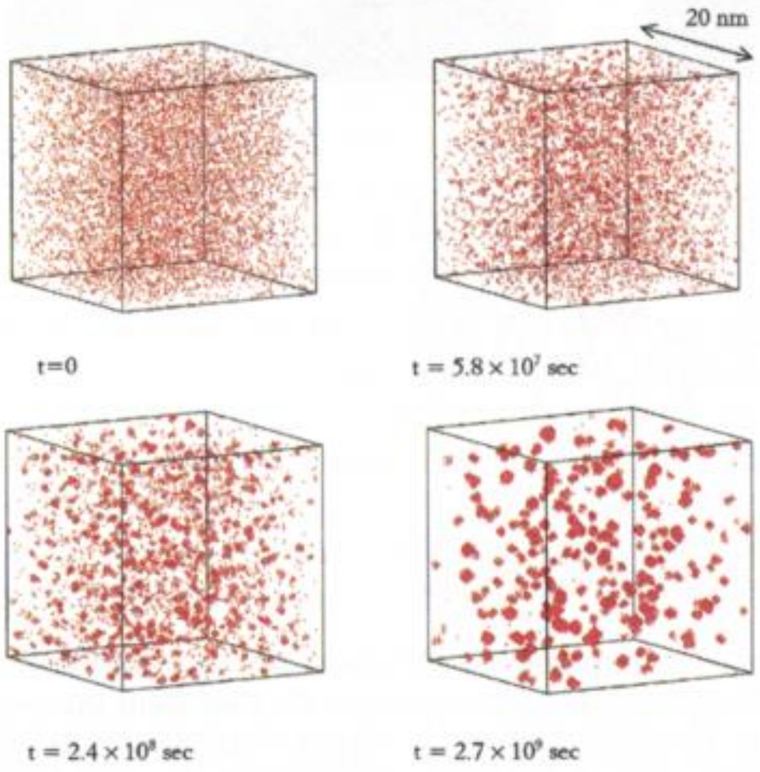
Induced by hardening and also the formation of précipitates





Non linear increase of the **DBTT** as a function of the fluence, rapid saturation

Modelling of Cu precipitation by Kinetic Monte Carlo



t=0

$t = 5.8 \times 10^7$ sec

$t = 2.4 \times 10^8$ sec

$t = 2.7 \times 10^9$ sec

Simulation Monte Carlo

Sonde Atomique 3D

- On each crystal site
  - Fe or Cu
- Thermodynamics for interactions between species
- Kinetics, according to probability of occurrence
- Simulated time: One century

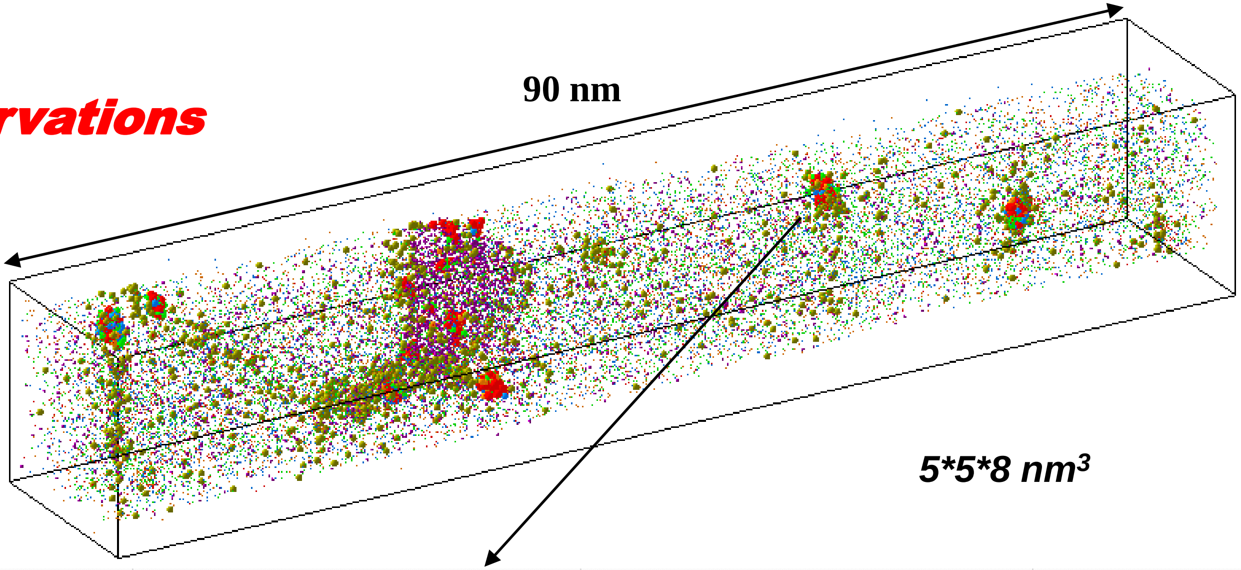
Behavior of real pressure vessel steels

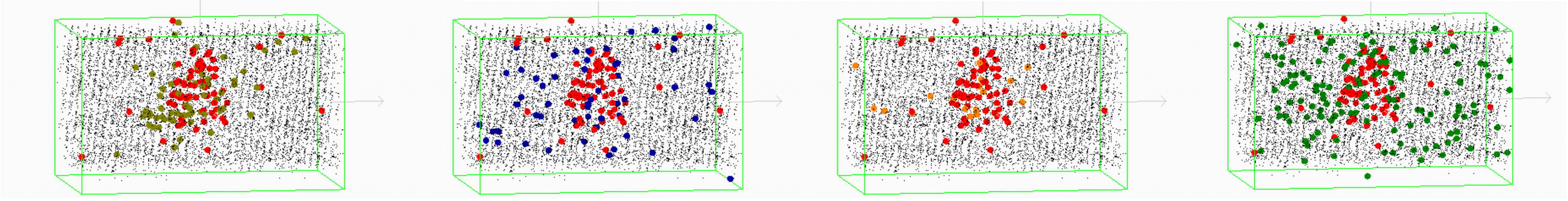
More complex chemistry

3D-Atom Probe observations

flux : $1{,}5.10^{15}$ $n.m^{-2}.s^{-1}$
fluence: $9{,}7.10^{23}$ $n.m^{-2}$
$T \approx 300$ °C

Diffuse solute clusters
2 nm - 5×10^{23} m^{-3}

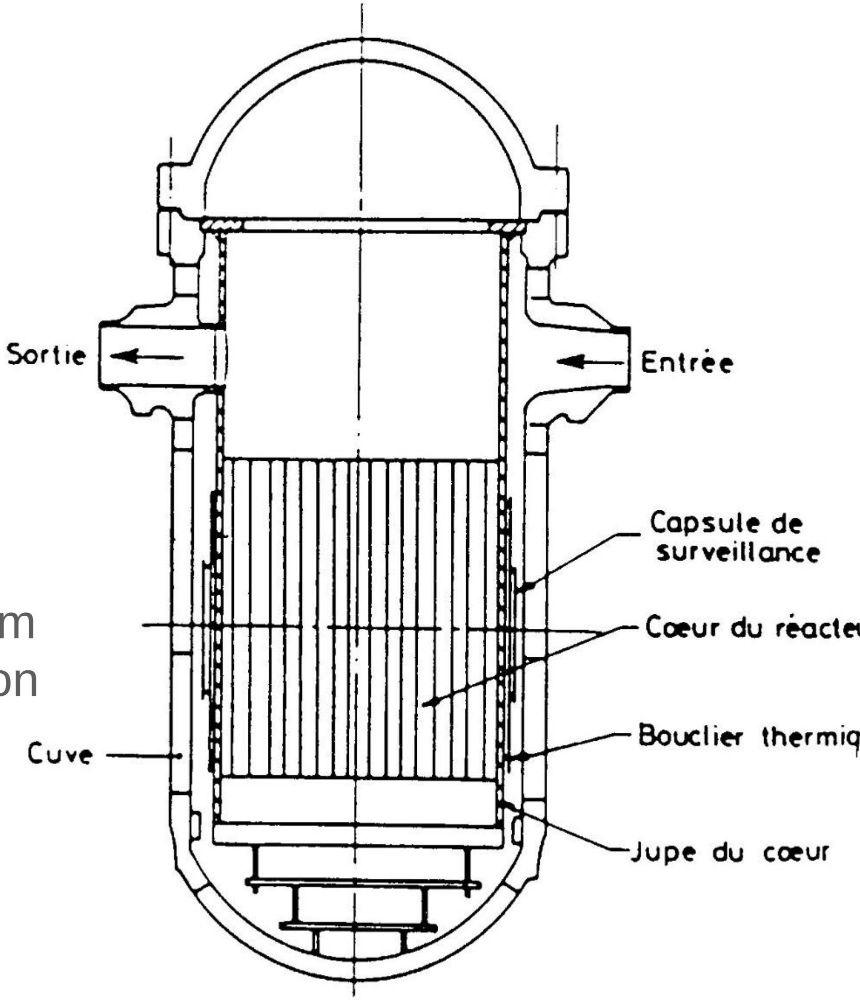




Fe	Cu	P	Mn	Ni	Si
bal	20	3.5	4	3.5	3.5

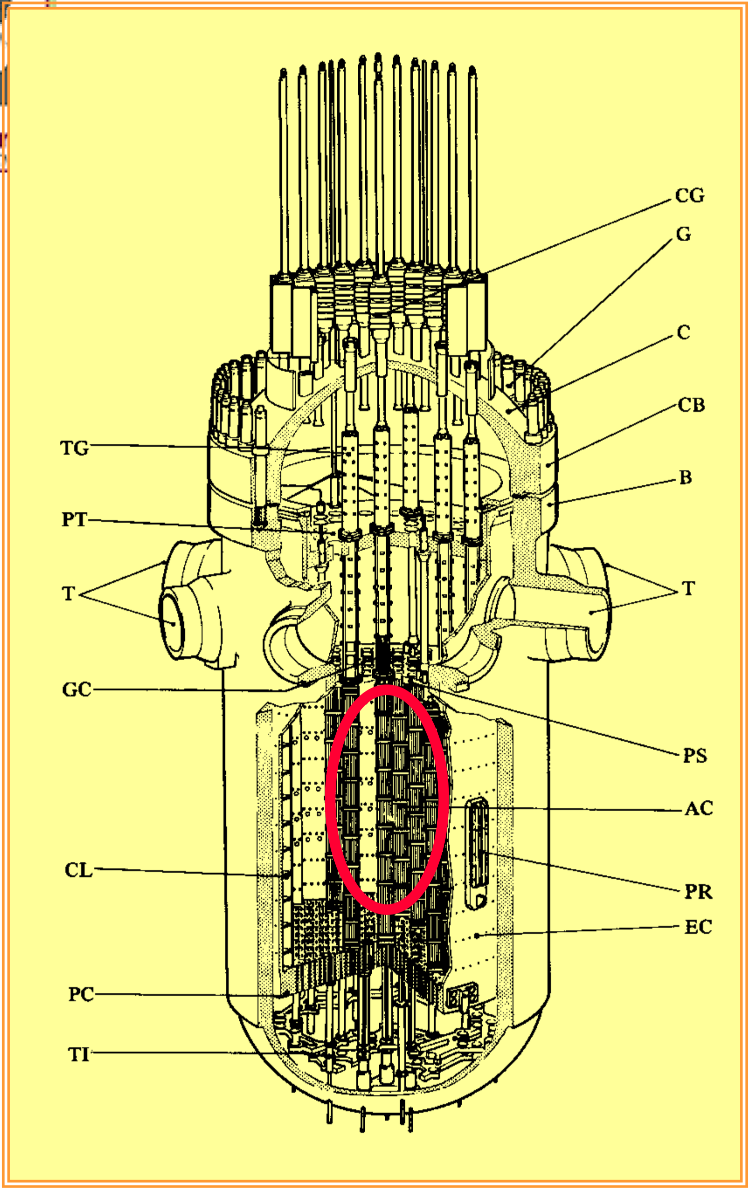
Surveillance Program

- Tensile and Charpy-V specimens are located at the periphery of the internal structures.
- The materials submitted to the surveillance program are the central part of the vessel, the C1-C2 weldment with its HAZ and a reference material common to all French nuclear power plants.



- Regulations stipulate that the surveillance program must be representative regarding the irradiation conditions (temperature, neutronic spectrum , …) and the materials
- Base metal taken from overlength of the shell
- Weld and heat affected zone elaborated under identical welding conditions as the core zone components

Irradiation devices



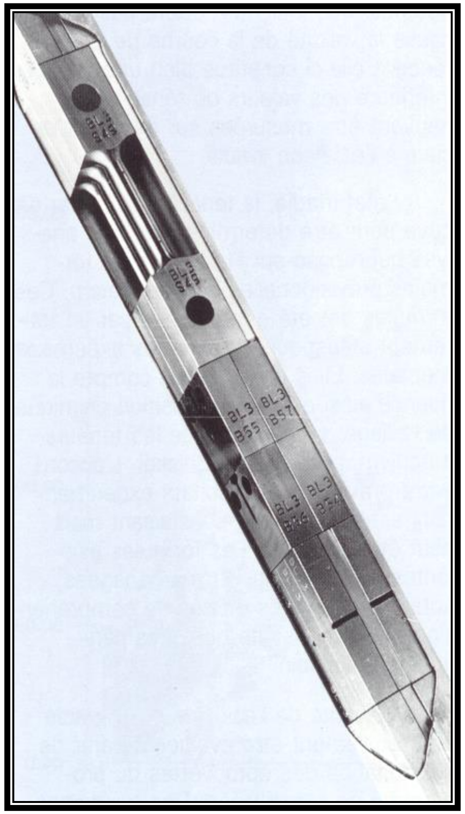
4 holders

Removed every 1/4 quarter design life

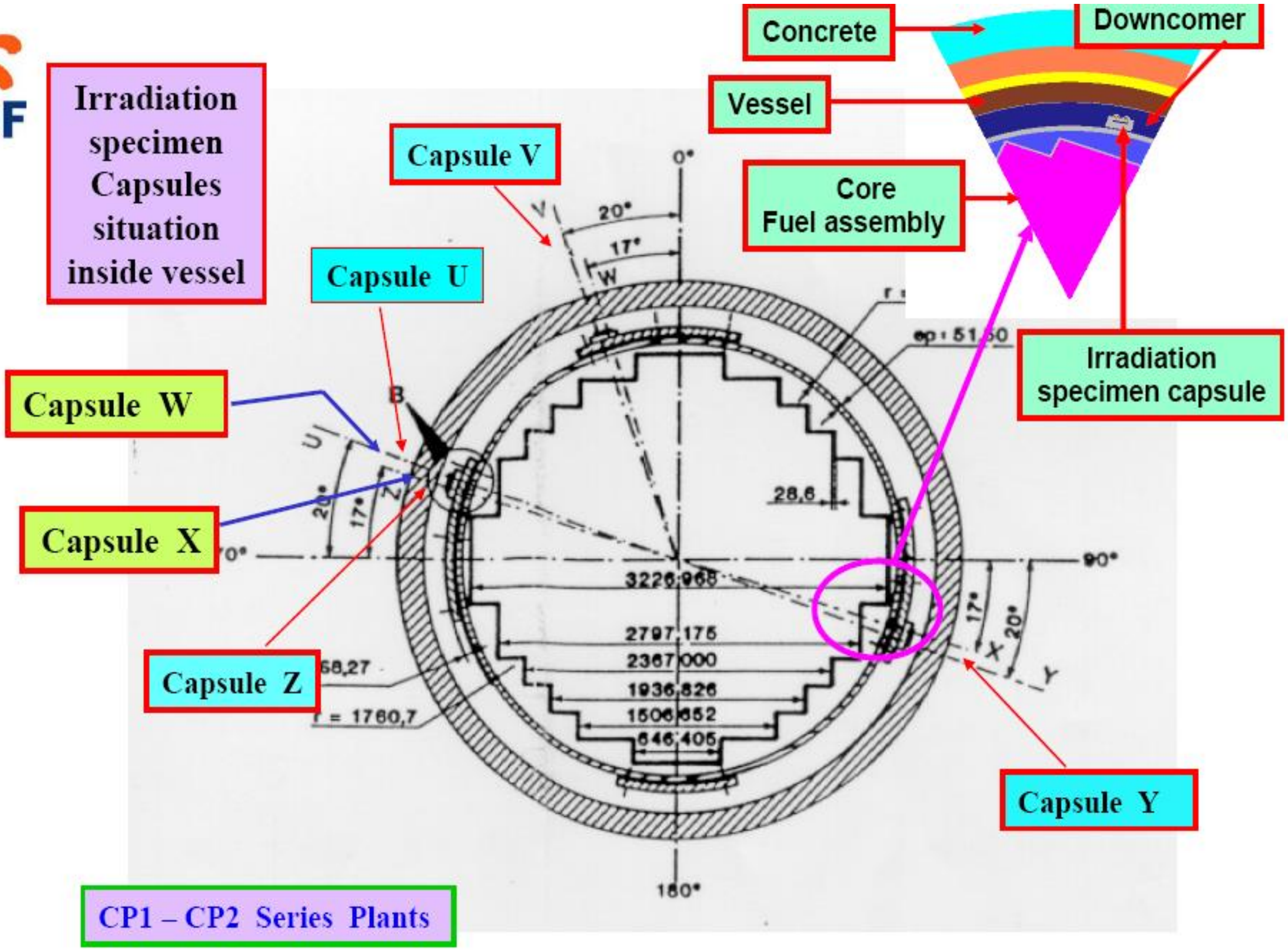
4, 7, 10 and 14 years

Fast neutron dose integrators

Samples: Charpy, tensile, K_{IC}



Surveillance programme



Irradiation specimen Capsules situation inside vessel

Capsule V

Capsule U

Capsule W

Capsule X

Capsule Z

Capsule Y

Concrete

Downcomer

Vessel

Core
Fuel assembly

Irradiation specimen capsule

CP1 – CP2 Series Plants

	Capsule W *(u)*	Capsule *(v)*	Capsule X *(z)*	Capsule *(Y)*
***Time of duration in vessel** (years)*	**4**	*7*	**9**	*14*
***Equivalent time of irradiation of the** Vessel (Years)*	***11.2***	*19.5*	***28.0***	*39.1*

Extension of the RPV irradiation surveillance programme based on the introduction of reserve irradiation capsules is engaged on the French plants since 1999 for all reactors.

Two reserve capsules W and X in place of capsules U and Z after removing from reactor these capsules U and Z.

Materials and fluxes

- Largely inspired from American regulations
- Fulfills the french safety Authorities requirements
- For all 900, 1300 and 1450 MWe reactors, the core zone is generally made up of two shells and the associated weld. As these materials experience a marked embrittlement under their design end of life (40 years), mechanical properties and particularly the rupture characteristics through impact strength are monitored for each reactor zone
- Materials positioned in capsules at locations well characterized for temperature and neutronic conditions
- Neutronic flux is higher (x3) than the one undergone on the vessel in order to anticipate the embrittlement. The flux is evaluated for each capsule through a lead factor, corresponding to the ratio of the neutron fluxes of more than 1 MeV energy undergone by the capsule and the vessel at the most irradiated point.

Material Ageing under Irradiation

- **Embrittlement Formulas :**

The general form is indexed on the temperature shift of Charpy-V :

$$\Delta RT_{NDT}\,(°C) = CF \cdot \phi^{G}$$

where *CF* is the Chemical Factor, ϕ is the fluence (10^{19} n/cm^2 with E > 1 MeV) and *G* an exponent.

Formulation based on steels and welds from the French nuclear program :

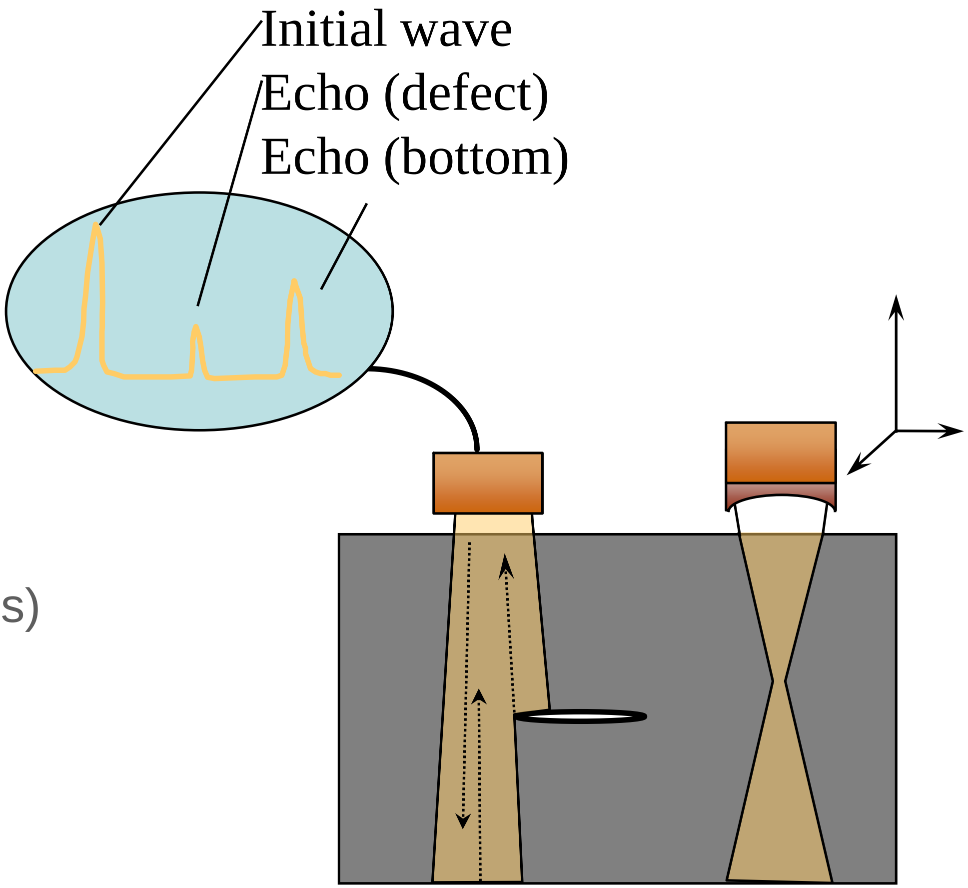
This particular formulation is for the highest effect of irradiation (FIS). Another formula is available in RCC-M for mean irradiation effect (FIM).

$$\Delta RT_{NDT}\,(°C) = 8 + \left[24 + 1537 \cdot (P - 0.008) + 238 \cdot (Cu - 0.08) + 191 \cdot Ni^{2} \cdot Cu\right] \cdot \phi^{0.35}$$

Non destructive examination (NDE)

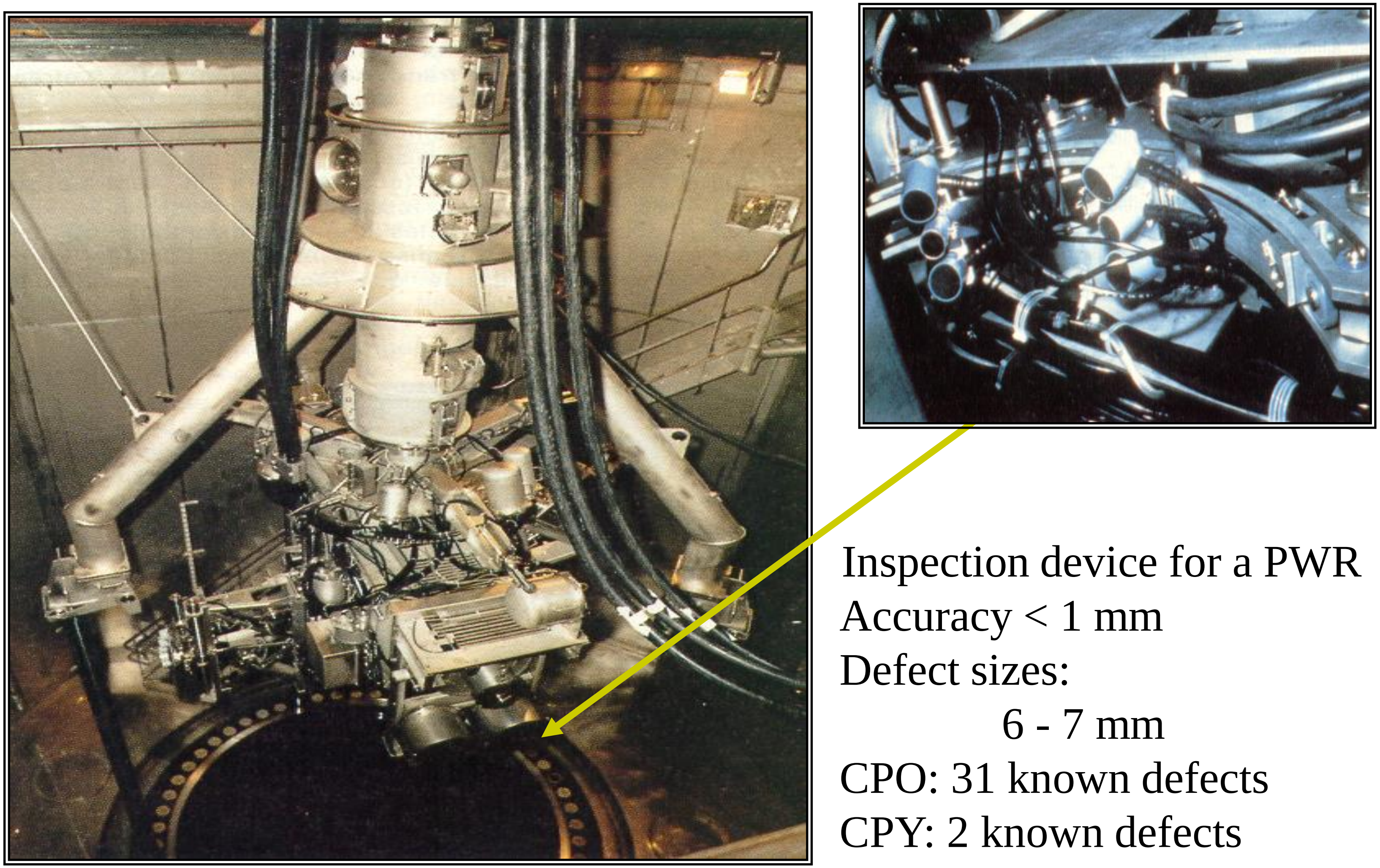
Ultra sonic testing

- Elastic strain wave
- Under water
- Focalized probes
    - x, y, z examination
    - 3D description of the cracks
    - A few mm accuracy
    - Inspection techniques (5 years)



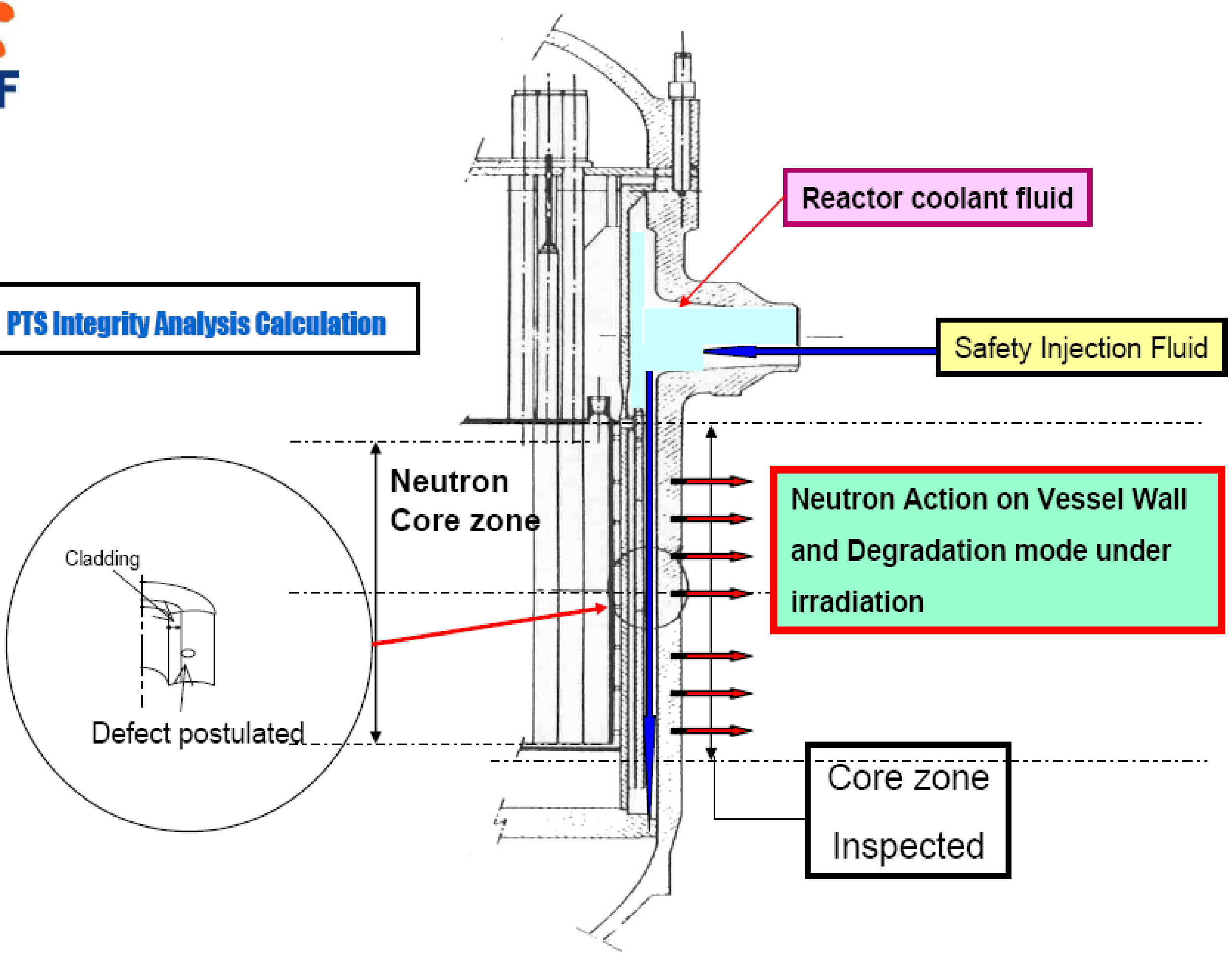
In service NDE

US probes



Inspection device for a PWR
Accuracy < 1 mm
Defect sizes:
6 - 7 mm
CPO: 31 known defects
CPY: 2 known defects

Assessment of vessel integrity



PTS Integrity Analysis Calculation

Reactor coolant fluid

Safety Injection Fluid

Neutron Core zone

Neutron Action on Vessel Wall and Degradation mode under irradiation

Core zone Inspected

Assessment of vessel integrity

Approach considered in relation with French codification

Demonstration of the integrity of the vessels in all conditions of loading, parameters :

- the RT_{NDT} ,
- all parameters fluence, defect distribution, transient, temperature.

The most severe conditions to be considered is the pressurized thermal shock (PTS) taking account for hypothesis shallow cracks beneath the cladding (subcladding area).

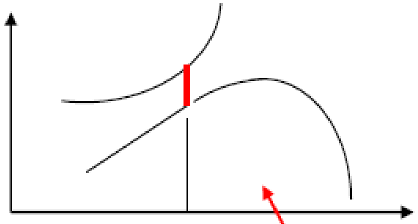
Justification of the Vessel integrity

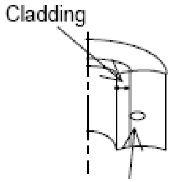
Demonstration of the margin on

- brittle fracture
- ductile fracture (ductile tearing).

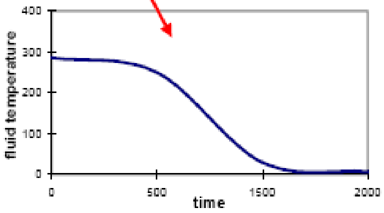
Assessment of vessel integrity

Assessment of RPV integrity :





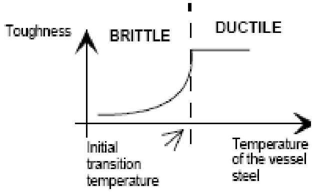
Defect



Transient

Fluence

initial properties of steel



T(x,t) and σ (x,t) distribution

Stress intensity factor K_I => K_{CP}
- computation at crack tips

Toughness K_{IC} -
computation at crack tips

Margin factor K_{CP} / K_{IC}

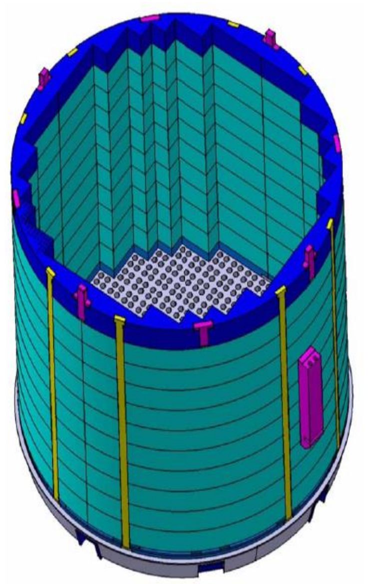
Extended life of PWR vessels

Old reactors

- Fuel management for reduced leakage
- Thermal recovery of pressure vessel
- Warm safety water tank

New reactors

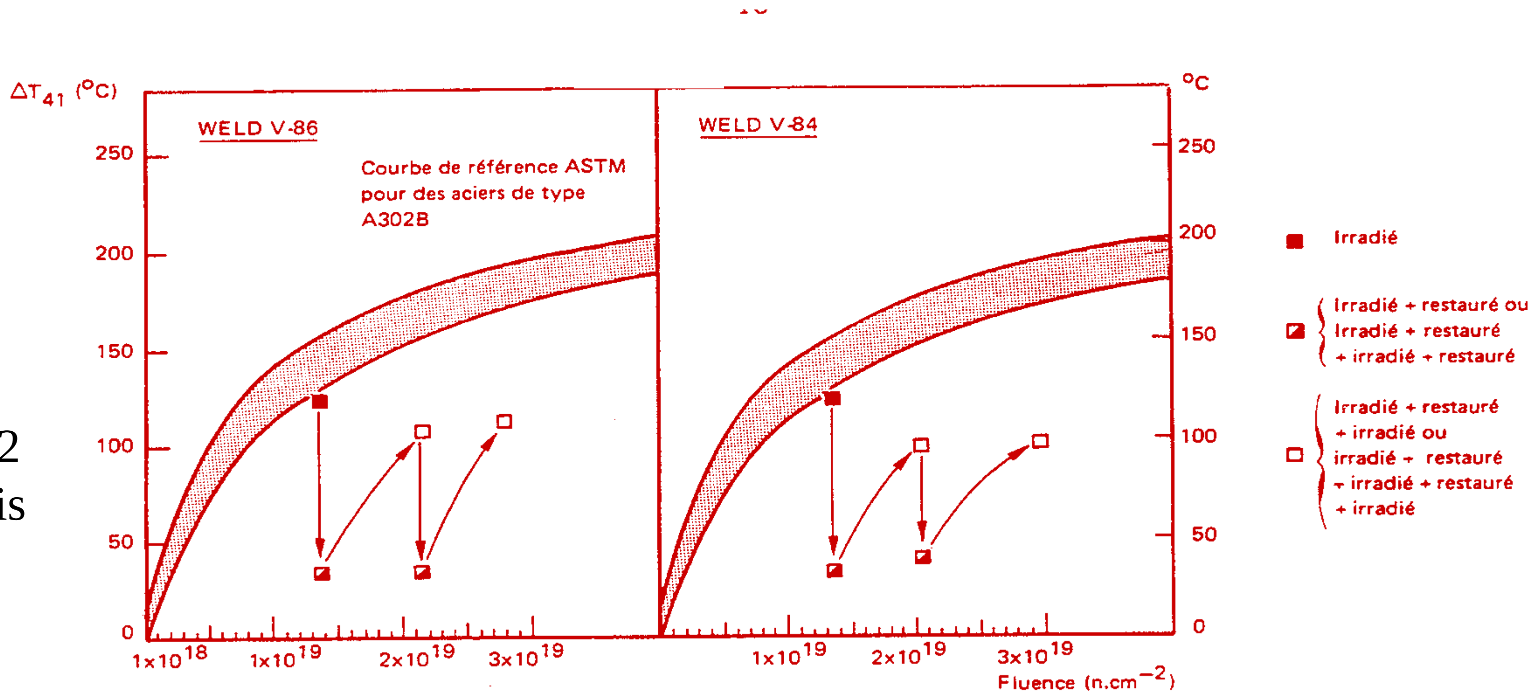
- Fine chemistry control during processing
- Reduced neutron flux by design (EPR heavy baffle)



Irradiation damage recovery

The clusters with different chemistry can be dissolved by intermediate temperature annealings. Yield strength increase is recovered as well as toughness.

This operation is performed in Russia where vessel steels are highly irradiated (2 x PWR, diameter is lower)



VVER: works at lower temperature ~ 270 °C and [Cu]<0.15% [P]<0.025% higher, RT_{NDT} increases faster than PWR for instance

Pressure vessel conclusions

Important metallurgical factors influence the properties, in particular the toughness, of French PWR steel (16MND5).

Careful control of elaboration, of microstructure and embrittlement under neutron irradiation are the key of good results.

The surveillance program warrants that no unpredicted deviation occurs.

Comprehensive work on the basic mechanisms at the origin of Copper embrittlement is also under investigation with cascade dynamic simulations and to be extended to get closer to real compositions

Enormous stakes to increase the lifetime of the vessel

Internals

Pressure vessel

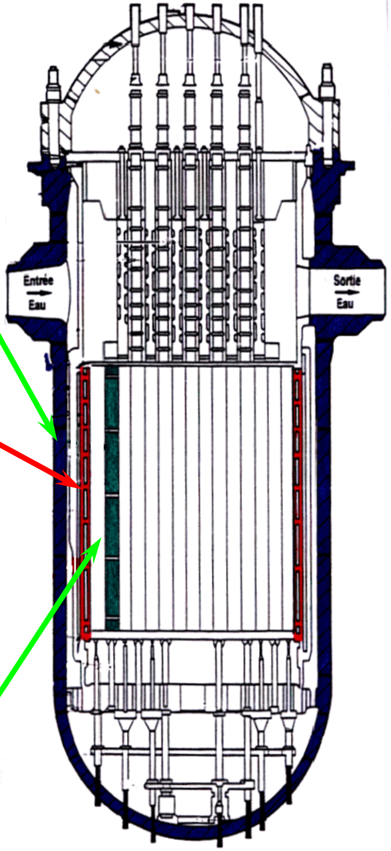
Internals

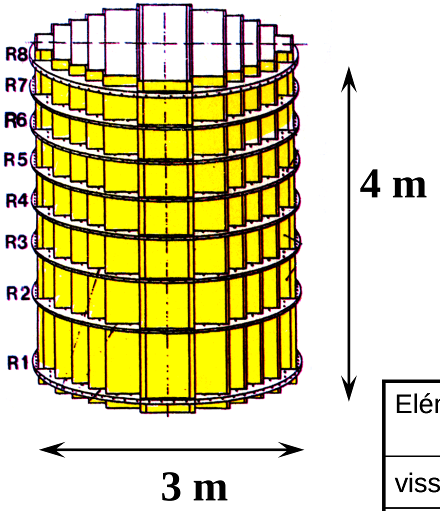
-barrier

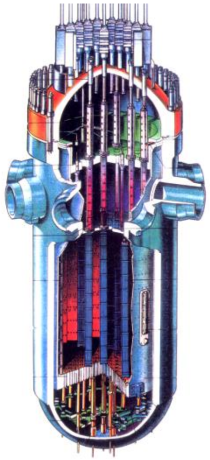
-baffle

-baffle bolts

Fuel assembly

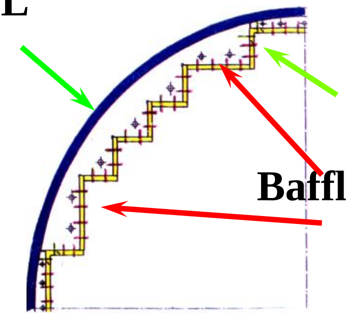






Elément	Matériau	Dose (dpa) après 40 ans
visserie	Cold worked	.0 à 80
cloisons	Quench annealed	10 à 80
renfort	Quench annealed	5 à 60

Core shroud (envelop) 304L weld 308L

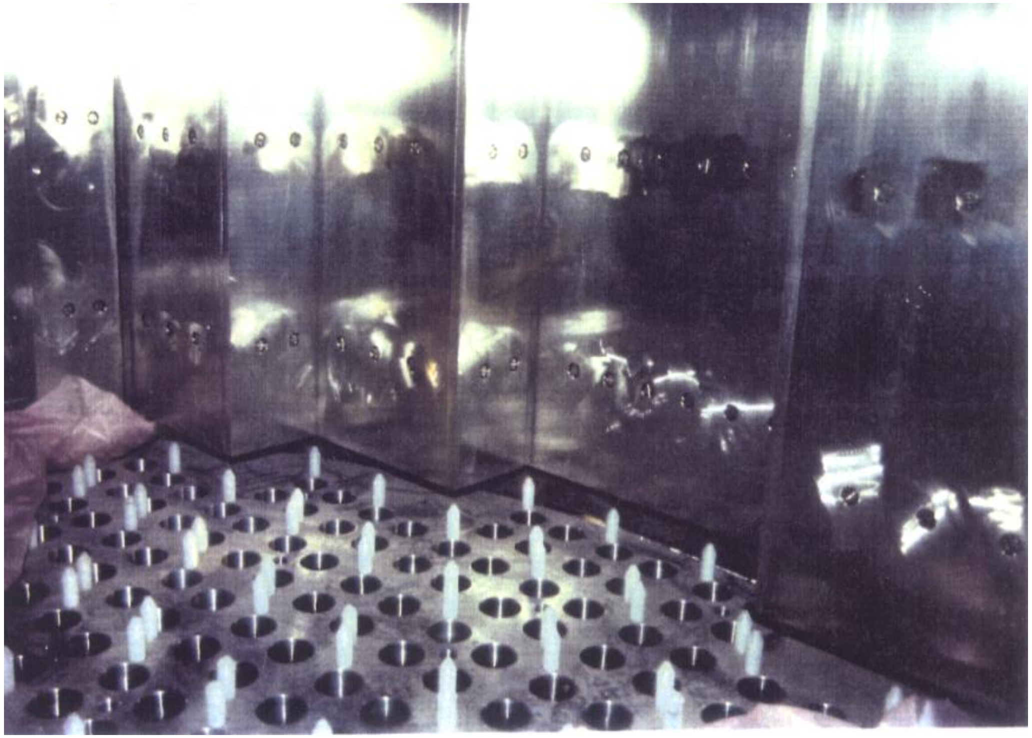


Baffle 304L

Baffle bolts 316 / 316

960 /reactor

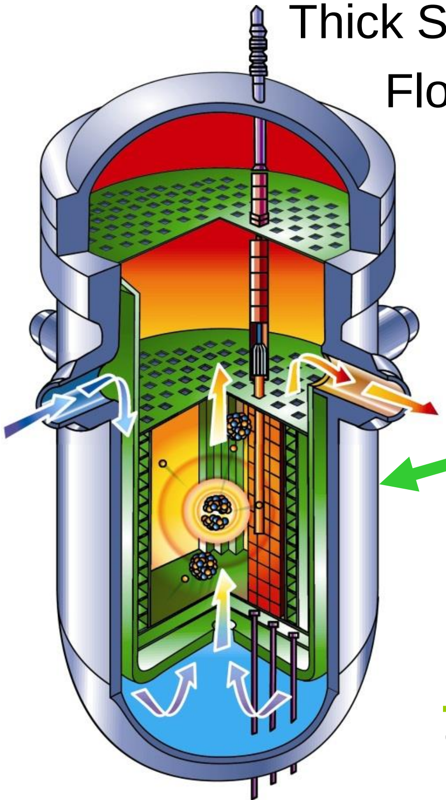
Internals N4 (PWR 1350MW)



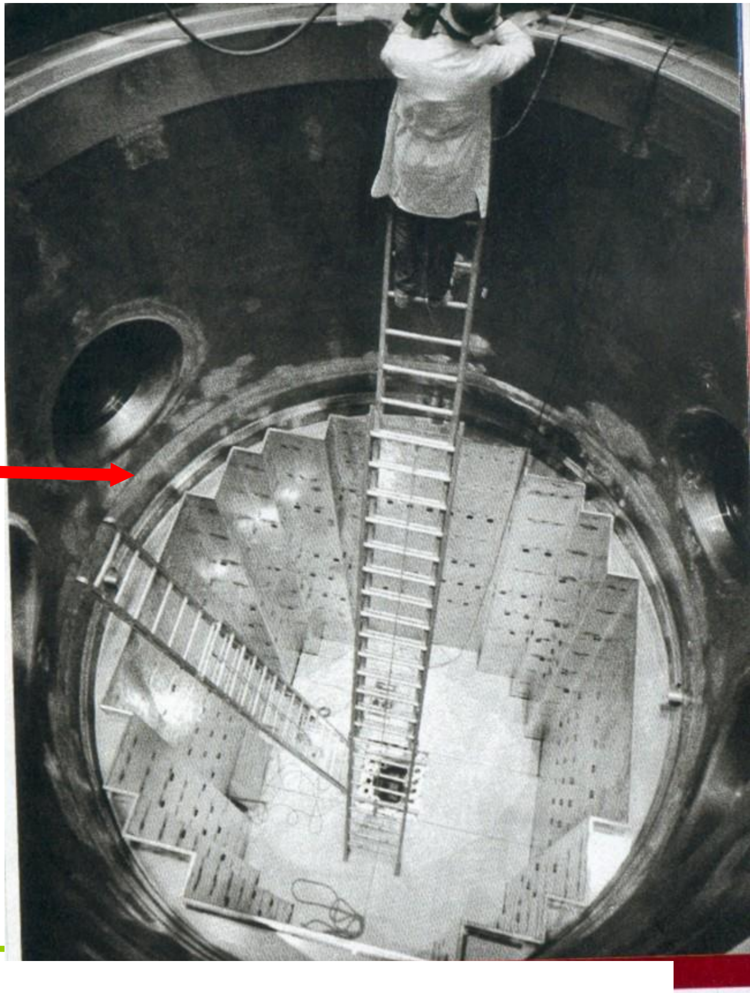
Role of internals for PWR reactors

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%Ni (18-08) : 304 et 316

2 to 100 dpa : end of life dose

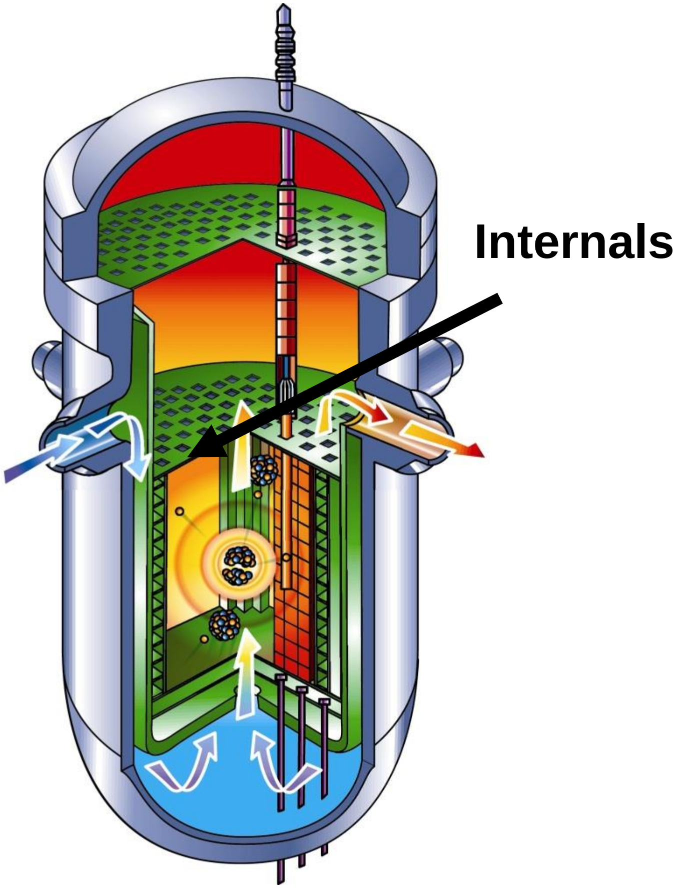
Irradiation effects: Aggregation of points defaults: loops, voids : **hardening**

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

- Swelling is possible at high doses

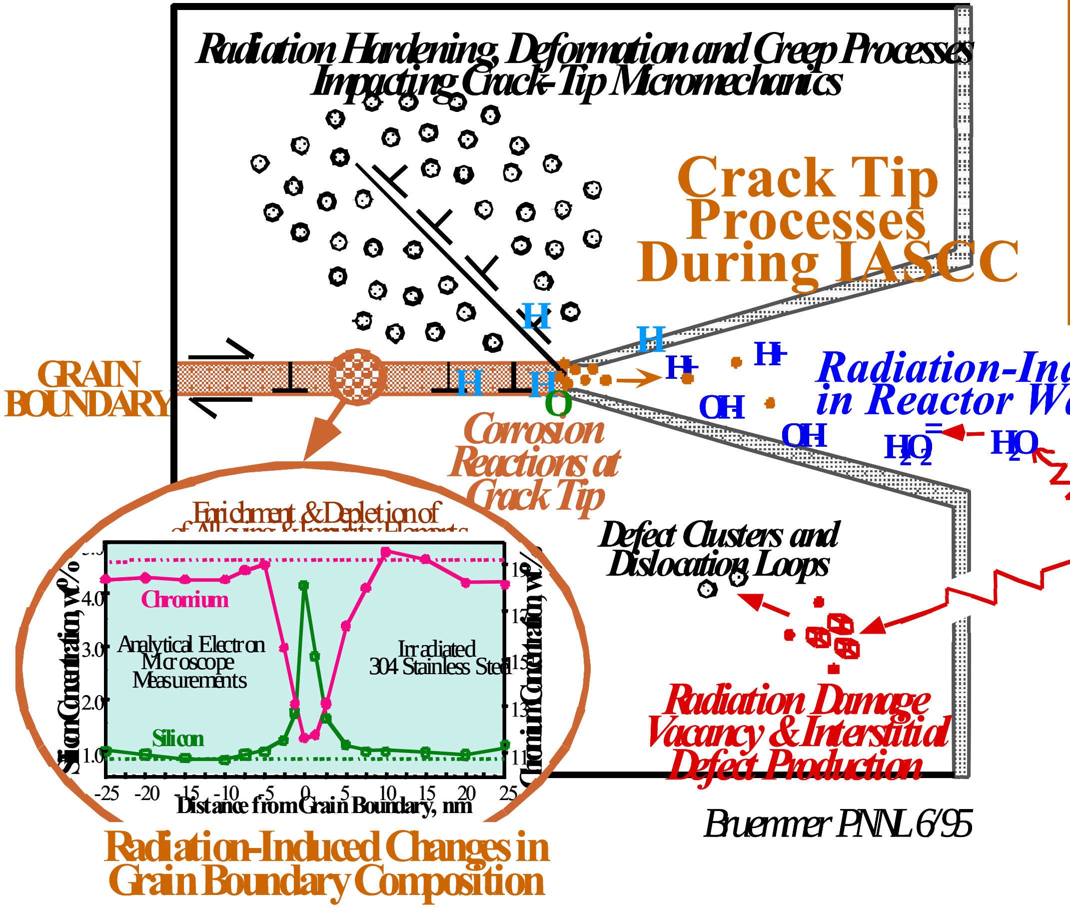
- He formation *in situ* (hardening)

$^{58}\mathrm{Ni}\,(n,\gamma)\,^{59}\mathrm{Ni}$ then $^{59}\mathrm{Ni}\,(n,\alpha)\,^{56}\mathrm{Fe}$

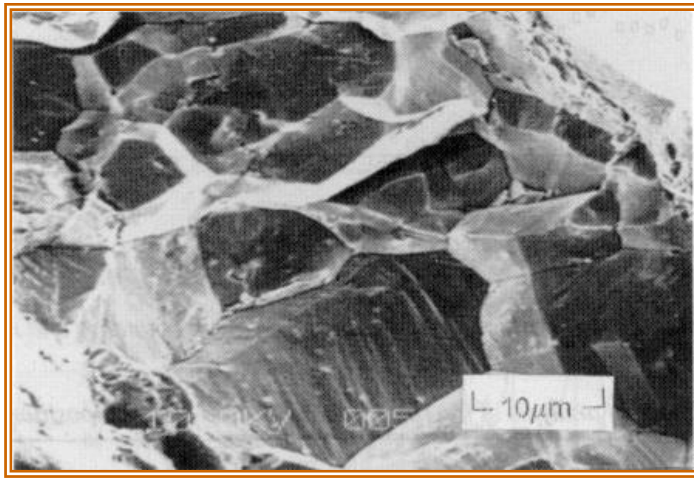


For PWR : 0.5 to 1 ppm He/an

Irradiation assisted stress corrosion cracking

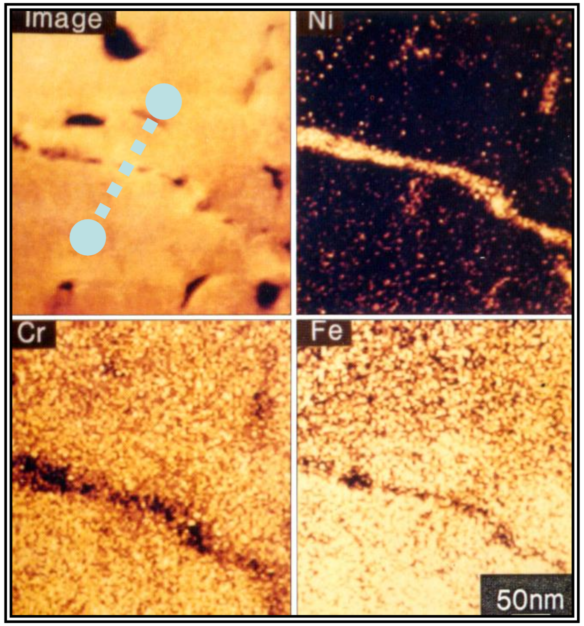


Bruemmer PNNL 6/95

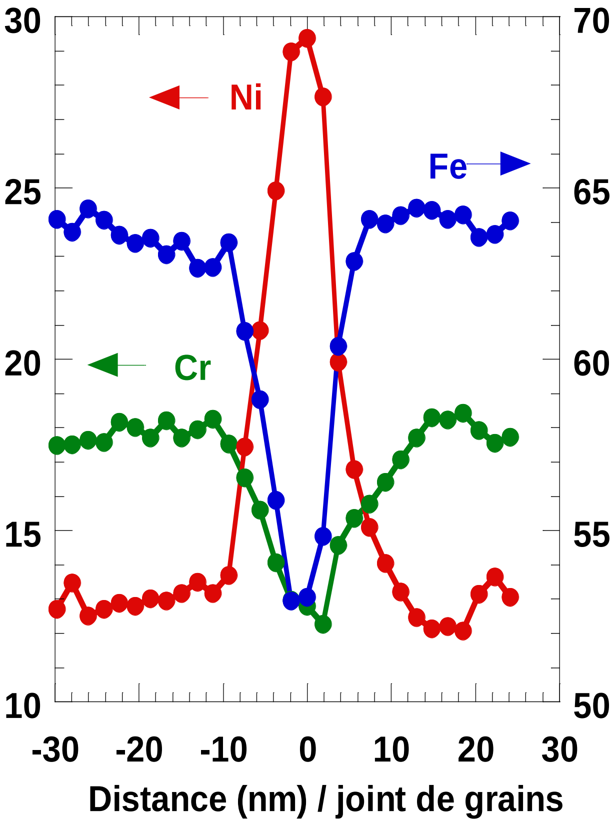


Grain Boundary Segregation

Typical GB chemistry

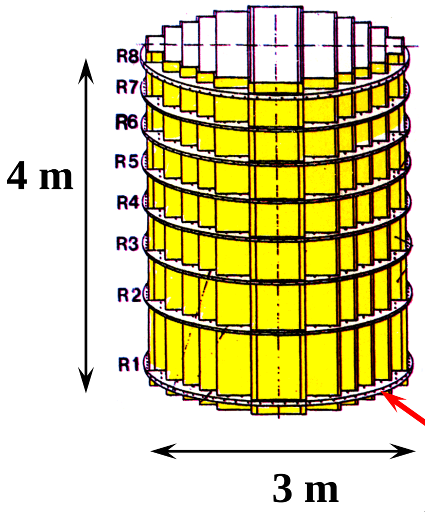


Dose : 3.7 10^{21} n.cm^{-2} , i.e. ≈ 0.1 dpa



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



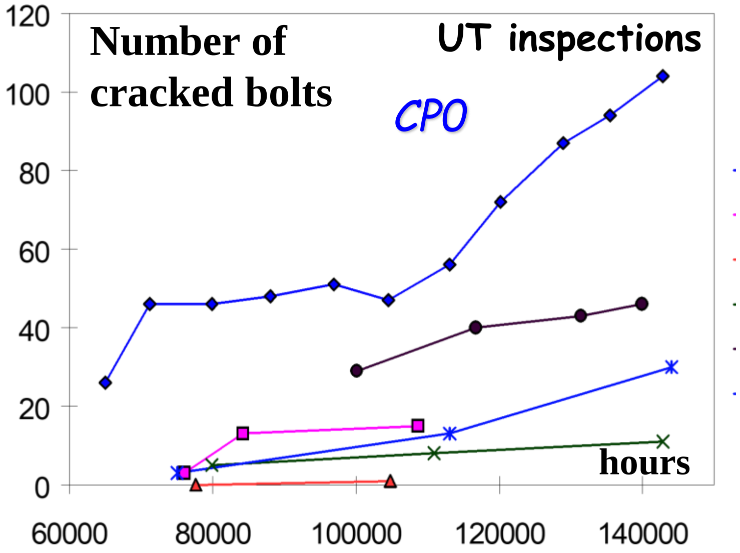
Cracked bolts

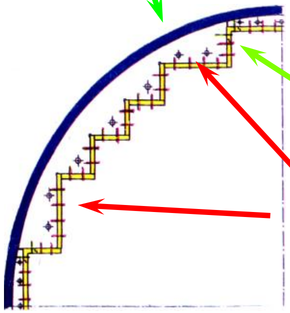
most irradiated

most stressed

Former
SA 304L

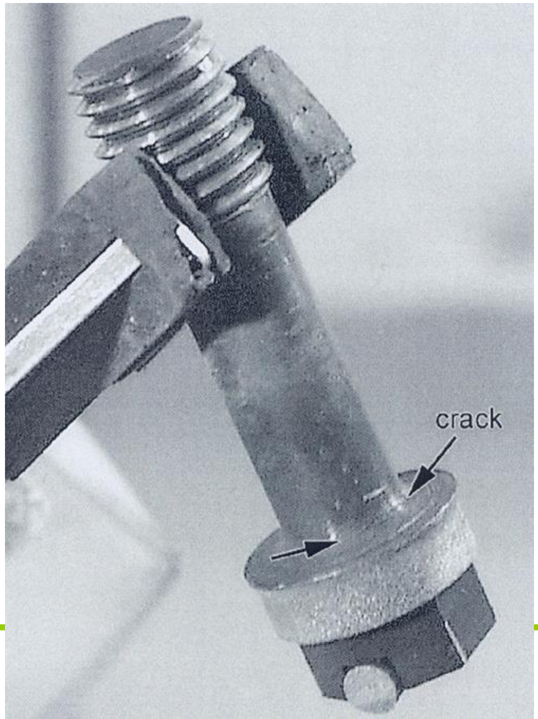
Core barrel 304L
Welds 308L



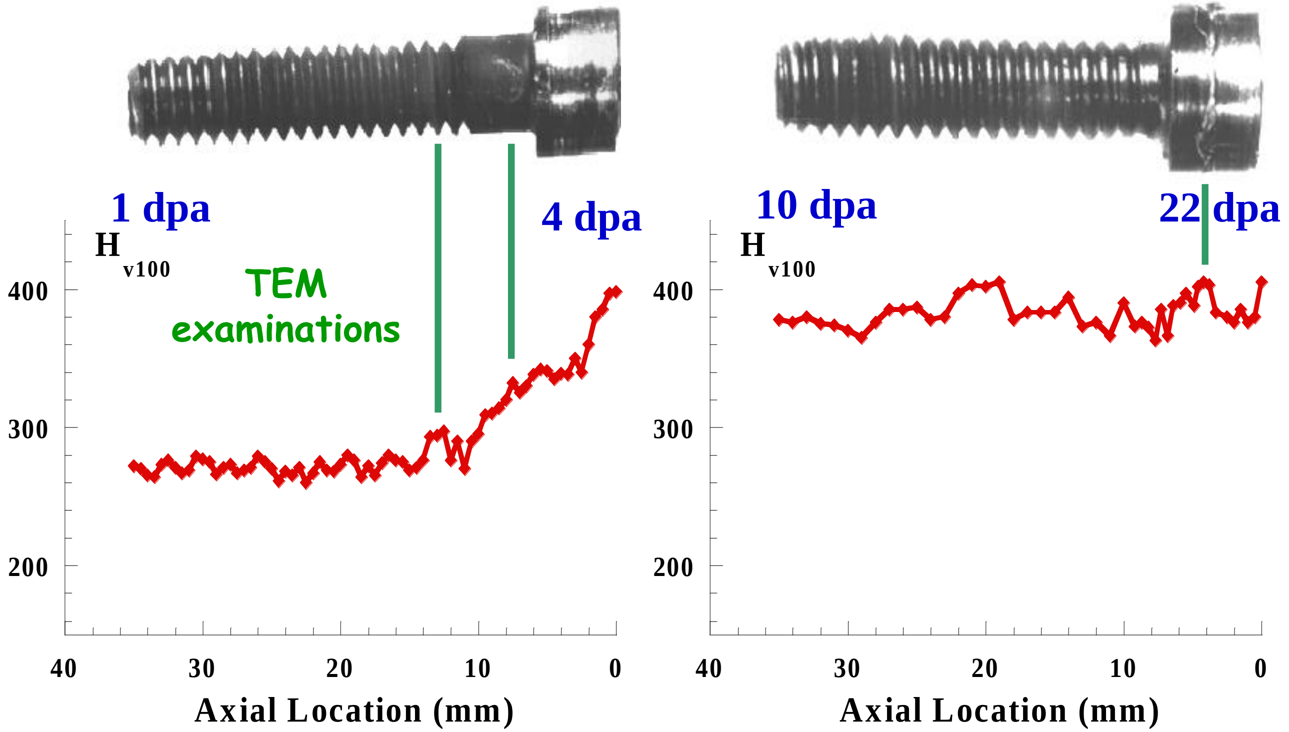


Baffle plate
SA 304L

Baffle bolts
CW 316 / 316L
960 /reactor



Hardness of internal baffle bolts for PWR



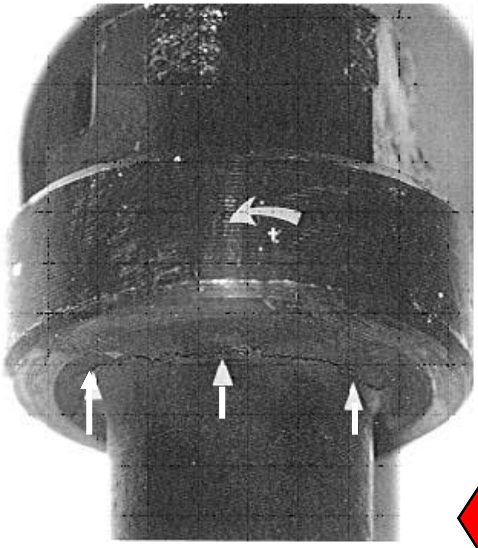
-> Saturation of the hardness around 5 dpa

High hardening

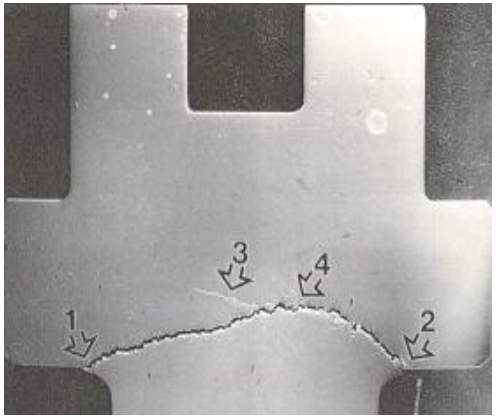
Cracking of baffle bolts for PWR internals IASCC

(Irradiation Assisted Stress Corrosion Cracking)

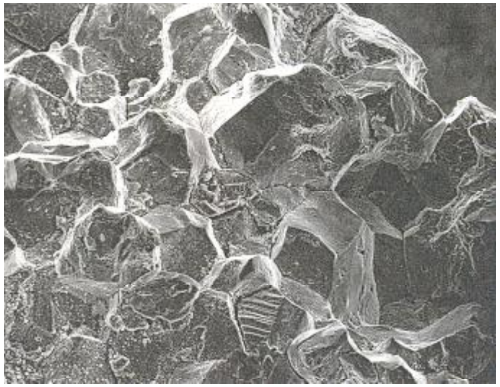
Irradiation dose : 5 - 10 dpa



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)