Materials under irradiation

PWR Structural Materials

Pascal YVON
Director of Nuclear Activities in Saclay

pascal.yvon@cea.fr
Short bio

Engineering degree Ecole Centrale Paris (materials major)

PhD in Applied 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

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…
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
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

**Vessel**
- Bainitic steel
- 16MND5
- A508 Cl 3
- ~ 300°C
- 0.1 dpa
- 40 ➔ 60 years

**Core Internals**
- Austenitic steels
- 300 - 380°C
- 30 - 120 dpa
- 40 ➔ 60 years

- neutrons
- temperature
- mechanical stresses
- environment

- 155 bars
- 293°C
- Water
- H₂, LiOH, B

- 155 bars
- 328°C

- 300 - 400°C
- 10/15 dpa
- 5 - 6 years

- 10/15 dpa
- 5 - 6 years
Macroscopic effects of irradiation on materials

After irradiation an evolution of mechanical properties can be observed.

For instance the tensile testing properties of steel:

![Graph showing engineering stress vs. engineering strain for 304-SA steel irradiated and tested at 325°C. The graph compares unirradiated steel with steel irradiated to various doses of 0.8 dpa, 1 dpa, 2 dpa, 3.5 dpa, 5.5 dpa, and 9 dpa. The graph shows a decrease in engineering stress with increasing engineering strain, indicating a decrease in material strength with irradiation.]
Mechanical behavior: tensile test

Low stress
Elastic deformation (fully reversible)
\[ \varepsilon = \sigma / E \]
- \( \sigma \): stress
- \( E \): modulus of Elasticity (Young modulus)

Yield strength \( \sigma_Y \)
Irreversible strain induced by dislocation glide

Ultimate tensile strength

![Graph showing stress-strain relationship](image)

- \( \sigma_E \): Elastic limit
- \( E \): Young's modulus
- \( \varepsilon_R \): Total elongation
- \( \varepsilon \): Strain
Macroscopic effects of irradiation on materials

Dimensional changes

*With or without stress*

Corrosion
Creation of H$_2$ and He

15 mg/dm$^2$~1 $\mu$m

Activation

Oxidation time
Effect of neutrons

Depending on their energies, neutron can have

Nuclear effects (inelastic): - thermal neutrons

Fission

Capture (and subsequent nuclear reactions)

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.
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Neutron spectrum and units of irradiation

The neutron of different energies have different effects on the materials.

Unit of flux: \( \text{n.cm}^{-2}\cdot\text{s}^{-1} \)

Unit of fluence: \( \text{n.cm}^{-2} \)

Difficult to describe a spectrum by just one number.

Therefore use of the dpa (displacement per atom)
Displacement cascade ($E_t \gg E_d$)

- 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.
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

\[ \text{Cascade creation} \quad (\text{in agreement with KP}) \]
\[ \text{Cascade relaxation recombinations} \]
Formation energy of point defects

**Vacancy formation energy:**
low distortion of the crystal lattice $\rightarrow$ low formation energy

**Interstitial formation energy:**
High distortion of the crystal lattice $\rightarrow$ high formation energy

$\rightarrow$ Out of reactor it is much easier to create a vacancy than a self-interstitial

Point defect concentration at thermodynamic equilibrium

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

$\rightarrow$ 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

![Interstitial](image1)

Vacancy

![Vacancy](image2)

\[ D_{DP} = D_{DP0} \exp\left( -\frac{H_{DP}^m}{kT} \right) \]

<table>
<thead>
<tr>
<th></th>
<th>Vacancy</th>
<th>Interstitial</th>
</tr>
</thead>
<tbody>
<tr>
<td>Formation energy (eV)</td>
<td>1.4-2.1</td>
<td>2.8-3.5</td>
</tr>
<tr>
<td>Migration energy (eV)</td>
<td>0.5-0.9</td>
<td>Ea=0.01-0.06</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Ec=0.1-0.3</td>
</tr>
</tbody>
</table>

Anisotropic diffusion of the self-interstitial

\[ \rightarrow \text{High mobility for interstitials, low mobility for vacancies} \]
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:**

\[ \square + \bigcirc = \text{nothing (if pure material)} \]
Point defect clustering

Vacancy disk

= dislocation loop

Point defects clustering: loop, cavity

\[ \square + \square = \square + \square = \square + \square = \square + \square \text{ or } \square \Rightarrow \text{Point defect cluster} \]
Annihilation on other sinks

Close to the sink: equilibrium concentration of point defects
Sur-saturation of point defects due to irradiation
-> point defect flux towards the sinks

• 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, $2 \times 10^{-8}$ dpa/s, $300^\circ$C, low sink density.

Hypothesis: Low sink density

- Hypothesis: Low sink density
- Increase with dose
- Thermodynamic equilibrium vacancies
- No interstitial
- Start of recombinations
- Interstitials reach sinks: Because of a higher mobility
- Vacancies reach sinks

Steady state

Vacancies concentration >> Interstitials concentration
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

Effect of elastic collisions on precipitates

• Amorphization

• Dissolution

Enhanced diffusion due to the super-saturation of point defects

• Precipitation

Nuclear reactions

• Changes in chemical composition
Amorphization of precipitates

Laves phases
(Fe,Cr)$_2$Zr

Amorphization of the precipitates
Dissolution of Fe into the matrix

Direction de l’Energie Nucléaire
Enhanced precipitation accelerated by irradiation

Sursaturation of point defects \[\rightarrow\] Accelerated diffusion of atoms \[\rightarrow\] Enhanced precipitation

Precipitation of $\beta$ Nb under irradiation in Zr alloys with Nb

Zr-Nb alloys, solide solution supersaturated 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 $\beta$-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

- CALOTTE
- BRIDE P
- BRIDE A
- VIROLE B
- TUBULURE
- VIROLE C1
- VIROLE C2
- ZONE E
- CALOTTE
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…
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

Direction de l’Energie Nucléaire
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

Forged cylindrical rings

Westinghouse

Le Creusot

Direction de l'Énergie Nucléaire
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)

<table>
<thead>
<tr>
<th>C_{max}</th>
<th>Mn</th>
<th>Ni</th>
<th>Mo</th>
<th>Cr_{max}</th>
<th>Si</th>
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<tbody>
<tr>
<td>0.2</td>
<td>1.15/1.55</td>
<td>0.5/.8</td>
<td>0.45/.55</td>
<td>0.25</td>
<td>0.1/.3</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>P_{max}</th>
<th>S_{max}</th>
<th>Cu_{max}</th>
<th>Co</th>
</tr>
</thead>
<tbody>
<tr>
<td>ppm</td>
<td>80</td>
<td>80</td>
<td>800</td>
</tr>
<tr>
<td></td>
<td></td>
<td>ppm</td>
<td>300</td>
</tr>
</tbody>
</table>

Low S, P et Cu to avoid irradiation induced embrittlement
Low Co to avoid $\gamma$ 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

Fragile
Transition F - D
Ductile
Fracture surfaces after Charpy impact test

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

**Ductile**: 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

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

$$K_I = \sigma \cdot \sqrt{\pi \cdot a}$$

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.
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.

$\text{T}_{\text{irr.}} = 290^\circ \text{C}$  $\text{T}_{\text{irr.}} = 120^\circ \text{C}$: lower elimination of PD, less recombination, more hardening.
Saturation of irradiation effects

Ductile Brittle Temperature Transition increase vs fluence

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

Induced by hardening and also the formation of précipitates
Modelling of Cu precipitation by Kinetic Monte Carlo

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: $1.5 \times 10^{15}$ n.m$^{-2}$.s$^{-1}$
fluence: $9.7 \times 10^{23}$ n.m$^{-2}$
$T \approx 300$ °C

Diffuse solute clusters
$2$ nm - $5 \times 10^{23}$ m$^{-3}$

$\begin{align*}
Fe & \quad Cu & \quad P & \quad Mn & \quad Ni & \quad Si \\
bal & 20 & 3.5 & 4 & 3.5 & 3.5
\end{align*}$
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, $K_{IC}$
<table>
<thead>
<tr>
<th>Time of duration in vessel (years)</th>
<th>Capsule W (u)</th>
<th>Capsule V (v)</th>
<th>Capsule X (z)</th>
<th>Capsule Y</th>
<th>Equivalent time of irradiation of the Vessel (Years)</th>
</tr>
</thead>
<tbody>
<tr>
<td>4</td>
<td>7</td>
<td>9</td>
<td>14</td>
<td>11.2</td>
<td>19.5</td>
</tr>
<tr>
<td></td>
<td>28.0</td>
<td>39.1</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

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.
Embrittlement Formulas:

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

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

where $CF$ is the Chemical Factor, $\phi$ is the fluence ($10^{19} \text{ n/cm}^2$ with $E > 1 \text{ MeV}$) 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).

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

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 Action on Vessel Wall and Degradation mode under irradiation
- Cladding
- Defect postulated
- 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

$T(x,t)$ and $\sigma(x,t)$ distribution

Stress intensity factor $K_I$ ⇒ $K_{CP}$
- computation at crack tips

Fluence

initial properties of steel

initial transition temperature → $K_{IC}$ - computation at crack tips

Toughness

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\textsubscript{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

Internal barrier

- baffle

- baffle bolts

Fuel assembly

Core shroud (envelop) 304L
weld 308L

Baffle 304L

Baffle bolts 316 / 316L
960 / reactor

<table>
<thead>
<tr>
<th>Elément</th>
<th>Matériau</th>
<th>Dose (dpa) après 40 ans</th>
</tr>
</thead>
<tbody>
<tr>
<td>visserie</td>
<td>Cold worked</td>
<td>0 à 80</td>
</tr>
<tr>
<td>cloisons</td>
<td>Quench annealed</td>
<td>10 à 80</td>
</tr>
<tr>
<td>renfort</td>
<td>Hypertrempé</td>
<td>5 à 60</td>
</tr>
</tbody>
</table>
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

Internal structures of PWR

Internals (temperature, up to 380°C)
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)

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

For PWR : 0.5 to 1 ppm He/an
Irradiation assisted stress corrosion cracking

Radiation Hardening, Deformation and Creep Processes Impacting Crack-Tip Micromechanics

Crack Tip Processes During IASCC

Radiation-Induced Changes in Reactor Water Chemistry

Radiation Damage Vacancy & Interstitial Defect Production

Defect Clusters and Dislocating Loops

Radiation-Induced Changes in Grain Boundary Composition

Crack Tip Reactions at Crack Tip

Irradiation & Depletion of Grain Boundary

Surface Concentration (ppm)

Element

Chromium
Silicon

Analytical Electron Microscope Measurements

Irradiated 304 Stainless Steel

Bruegger PNNL 6/95

Direction de l’Energie Nucléaire
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.

Dose : $3.7 \times 10^{21} \text{ n.cm}^{-2}$, i.e. $\approx 0.1 \text{ dpa}$
Irradiation effects on Core Internals

Core barrel 304L
Welds 308L

Baffle plate
SA 304L

Baffle bolts
CW 316 / 316L
960 / reactor

Former
SA 304L

Cracked bolts
most irradiated
most stressed

Number of cracked bolts

UT inspections

CPO

hours

60000  80000  100000  120000  140000

Direction de l’Energie Nucléaire
Hardness of internal baffle bolts for PWR

- Saturation of the hardness around 5 dpa

High hardening
Cracking at the junction between: head - deformation area

Irradiation dose: 5 - 10 dpa

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)