ENGINEERING MODELS USED FOR PLIM

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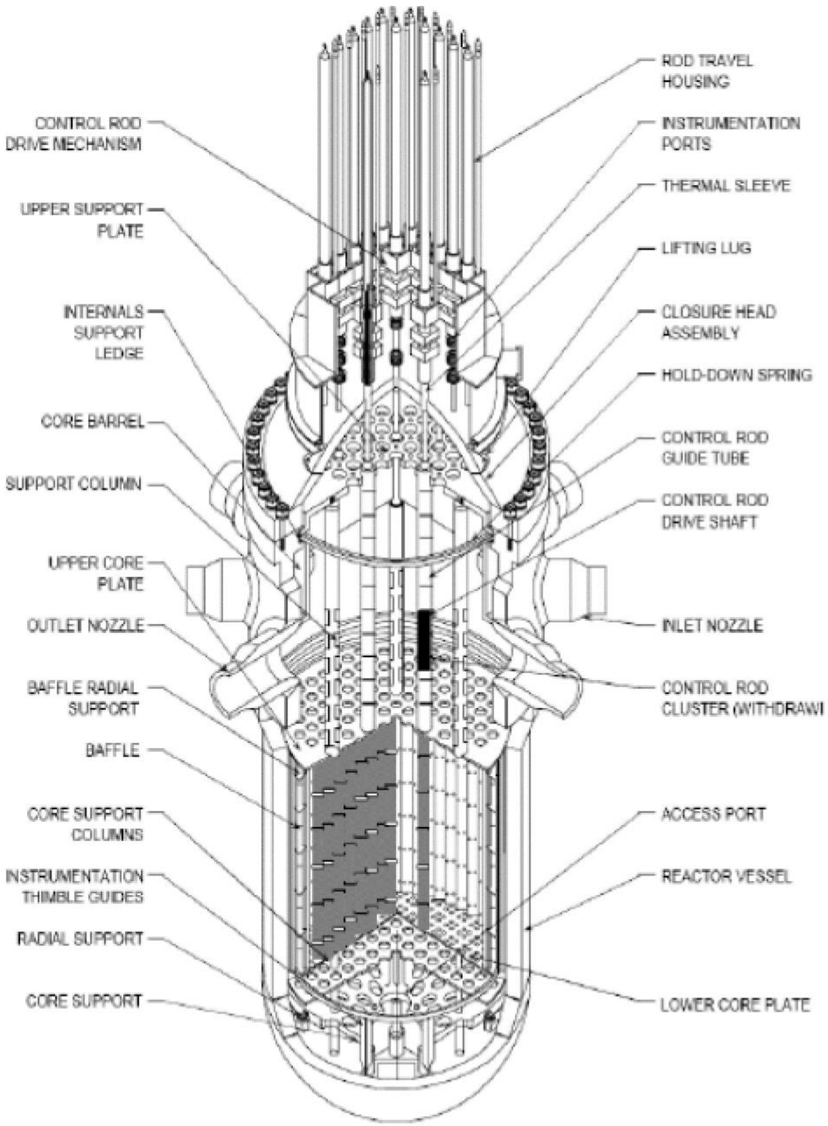
Outline

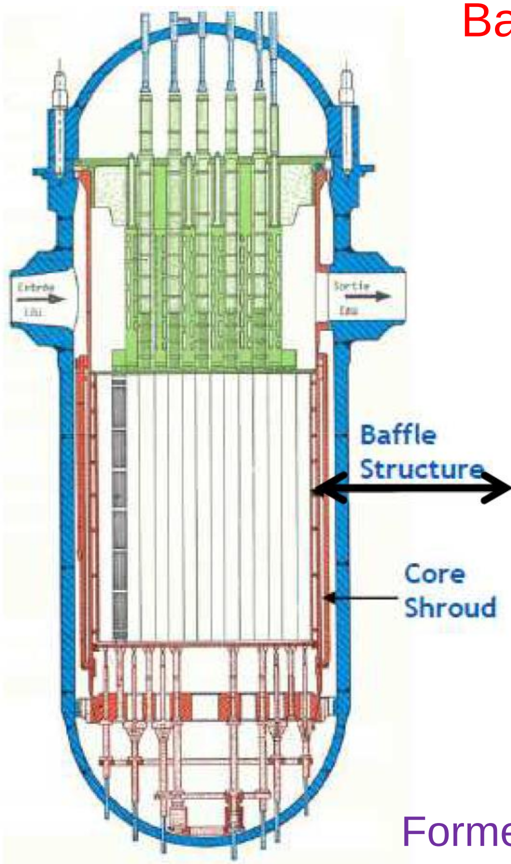
- Internals structure of PWR
  - ✓ Materials and Components
  - ✓ Operational Background
- Reactor Vessel Internals Management
  - ✓Methodology
  - ✓Determination of loadings
  - ✓Irradiated stainless steel constitutive model
- IASCC model
  - ✓ IASCC Engineering models
  - ✓Comparison between loading, IASCC criterion and field experience
- Conclusions and perspectives

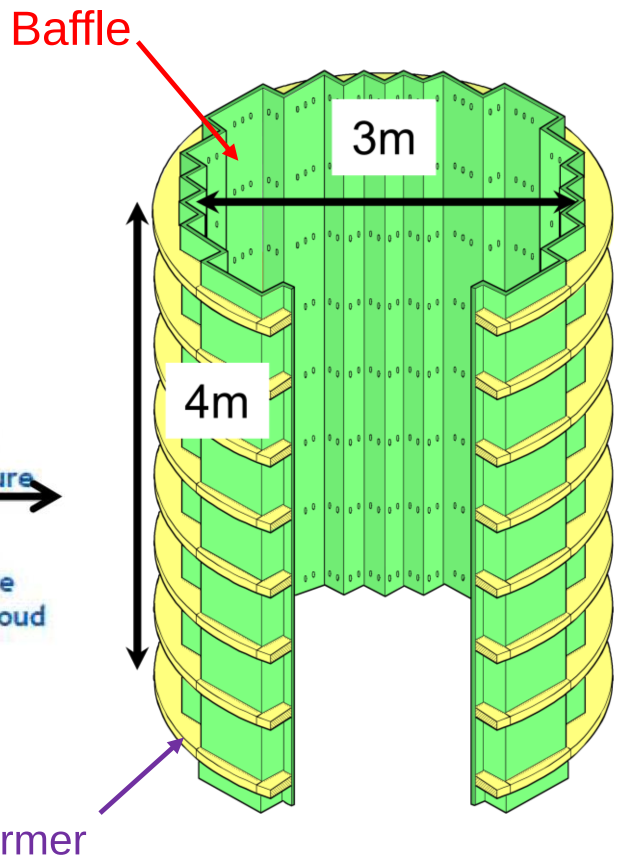
Internal structures of PWR

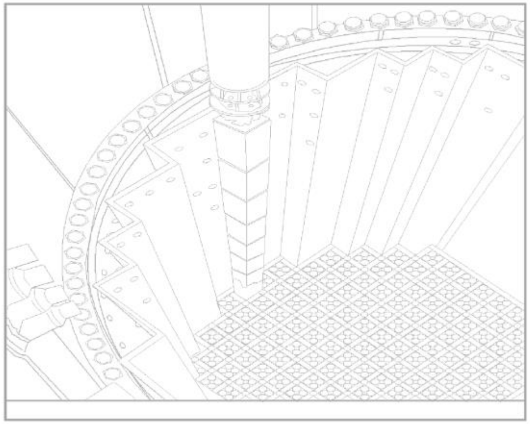


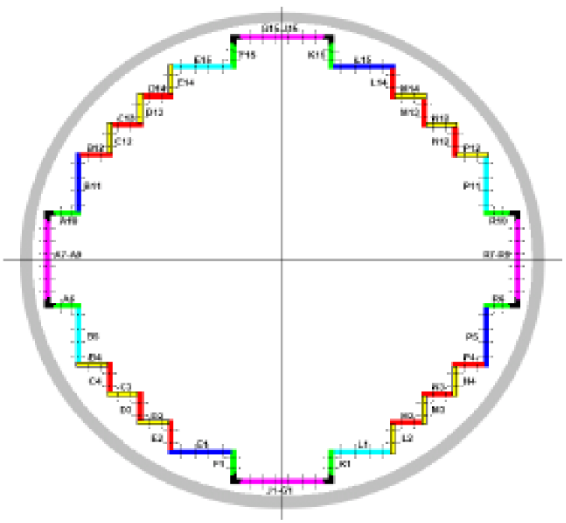
- Provide support, guidance and protection to the core
- Provide a path for reactor coolant flow to the core
- Provide a path for control elements and instrumentation
- Provide γ and neutron shielding for the vessel

Lower core internal structure of PWR

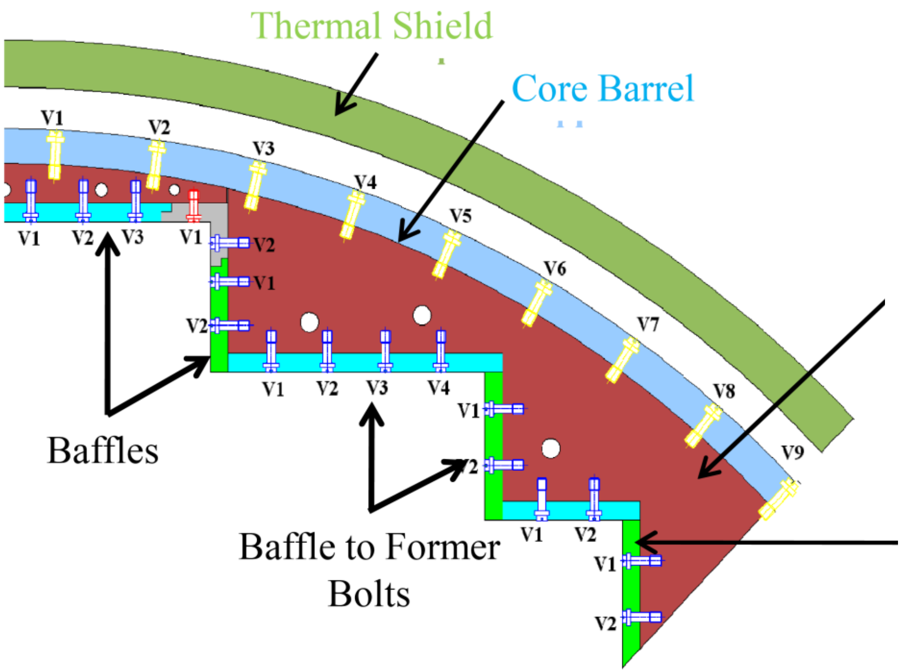


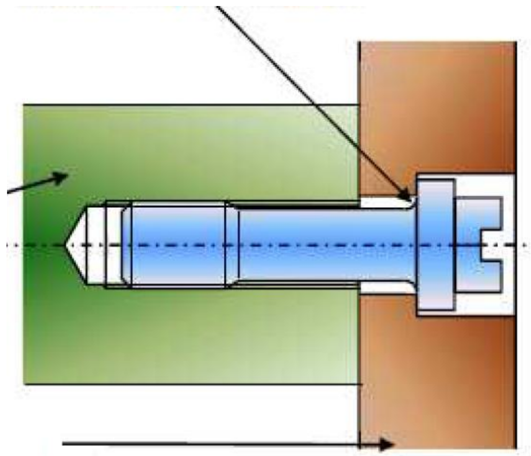






Components and Materials



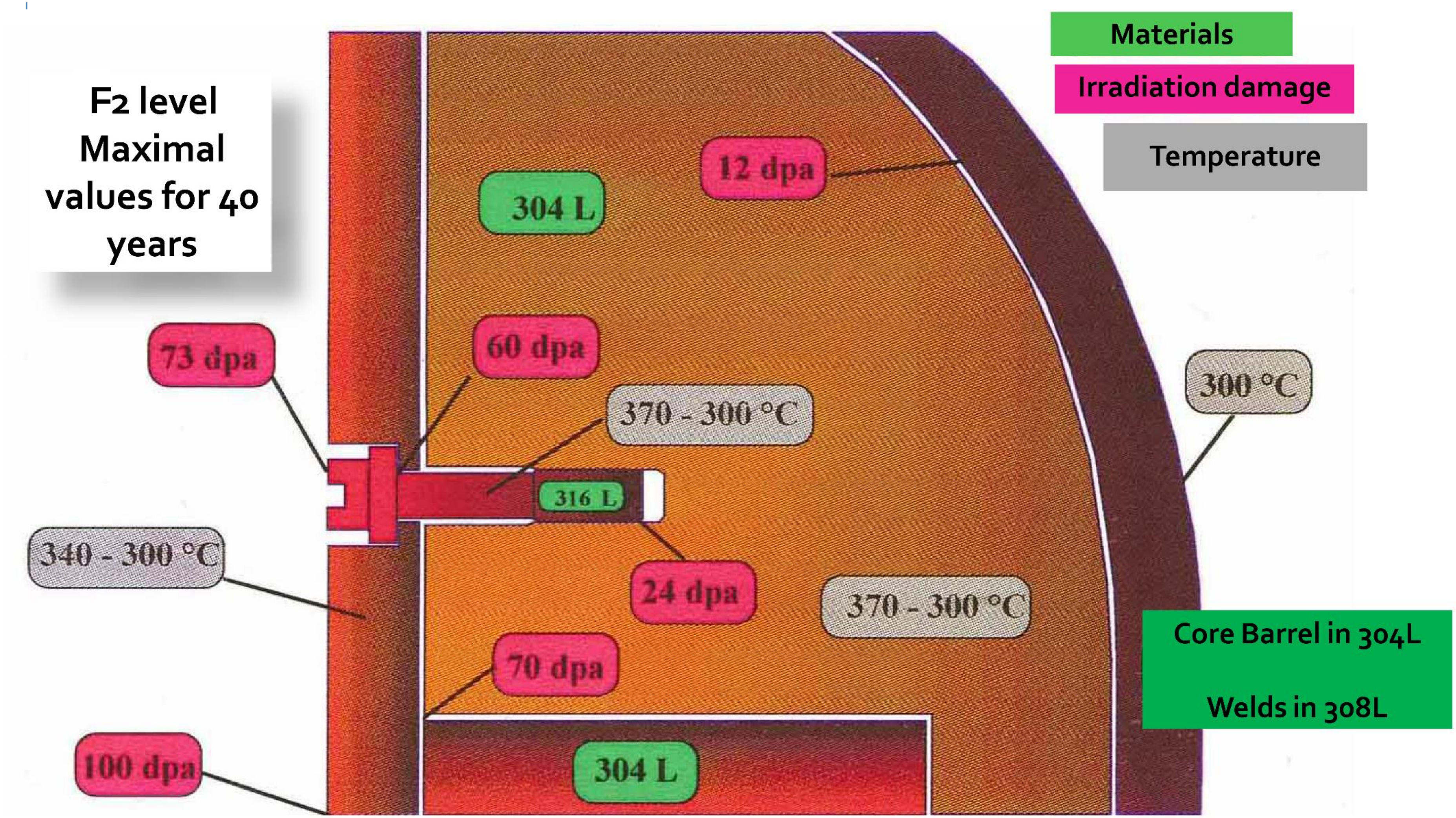


Baffle
Former
Core barrel
SA 304L
(18%Cr-10%Ni)

Baffle bolt
CW 316E or CW 316L
(17%Cr-11%Ni-2.5% Mo)

900-1000 bolts depending on design with only 1/3 necessary for component integrity

Characteristics

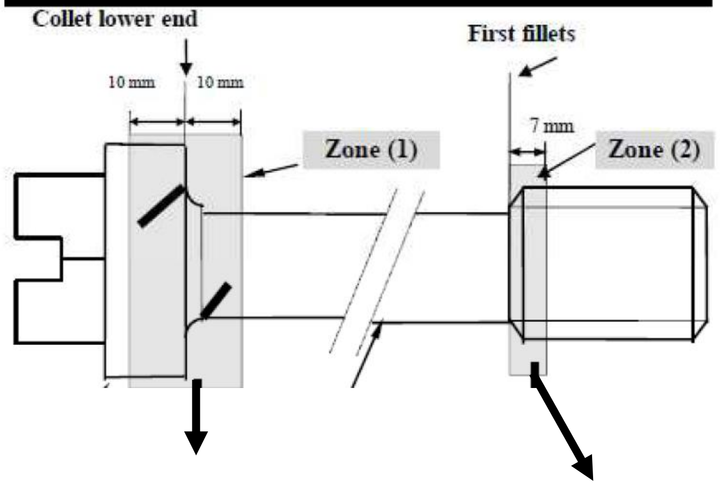


Operational background on first series of 900 MWe baffles

1980-1987	Baffle Jet (gaps in the baffle joint)	Consequences on fuel assembly
1981-93	Up Flow Conversion	
1988-	Metallurgical analysis of removed bolts	IASCC
1988-	**In Service Inspection (NDE)**	Acceptable degradation patterns
1998-	**Irradiation tests** to develop predictive model	IASCC, creep, hardening
2000	**Bolt replacements**	Bolts required for safety
2000-	**New bolt design**: cooling down of bolts, decrease tightening torque and prevent water	

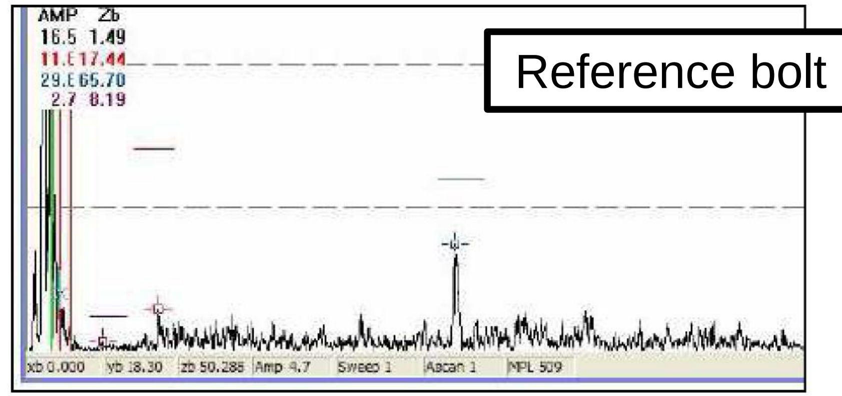
In Service inspection

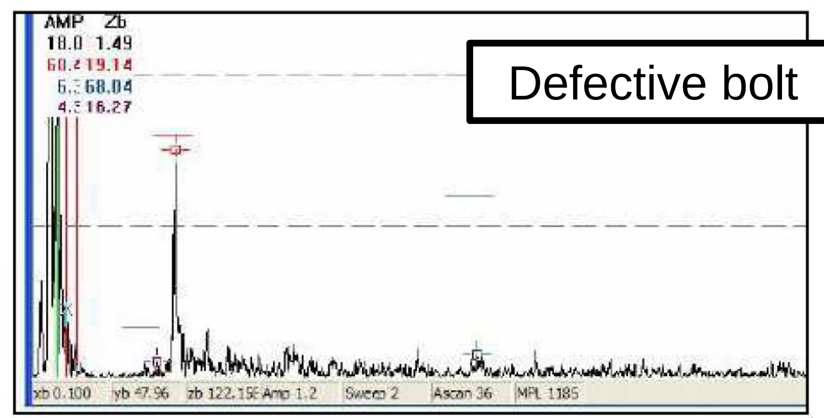
90% of the cracks heading towards the center of bold head



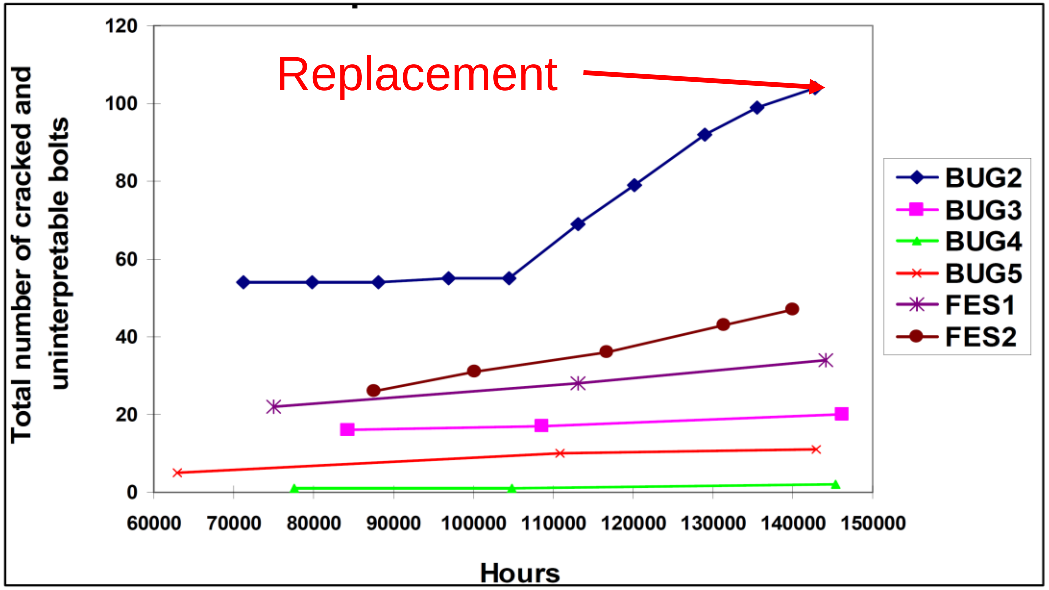
Fatigue (not observed)
IASSC observed

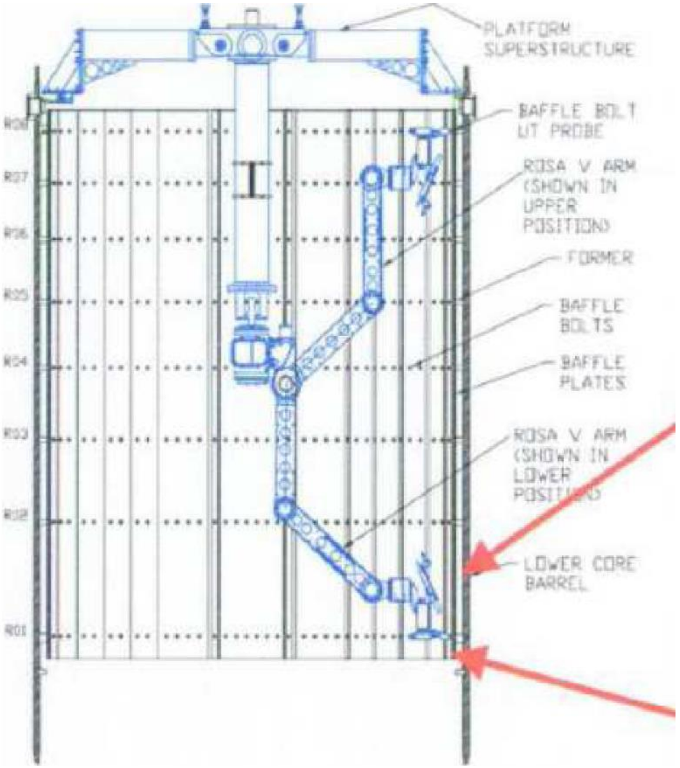
Fatigue (not observed)
Optional depending on the UT permeability of the bolt





Maintenance



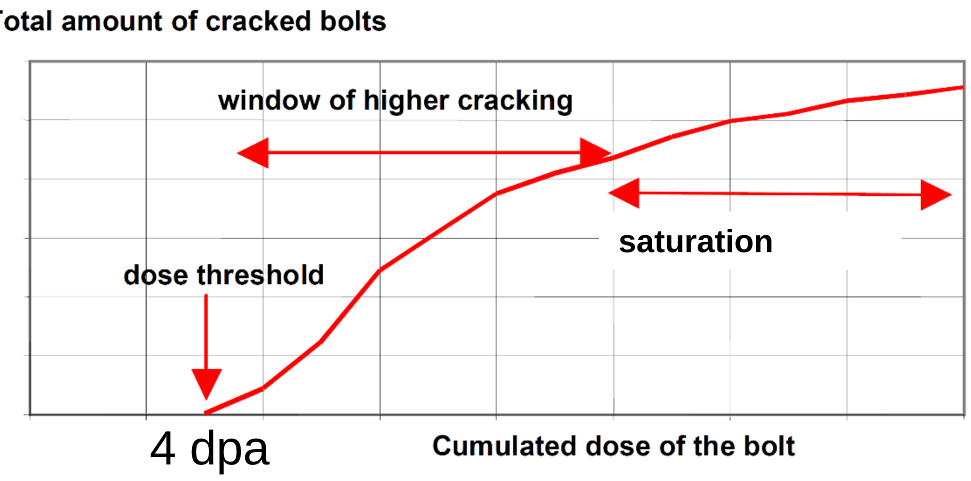


Large scatters in the total number of cracked or uninterpretable bolts as a function of the unit.

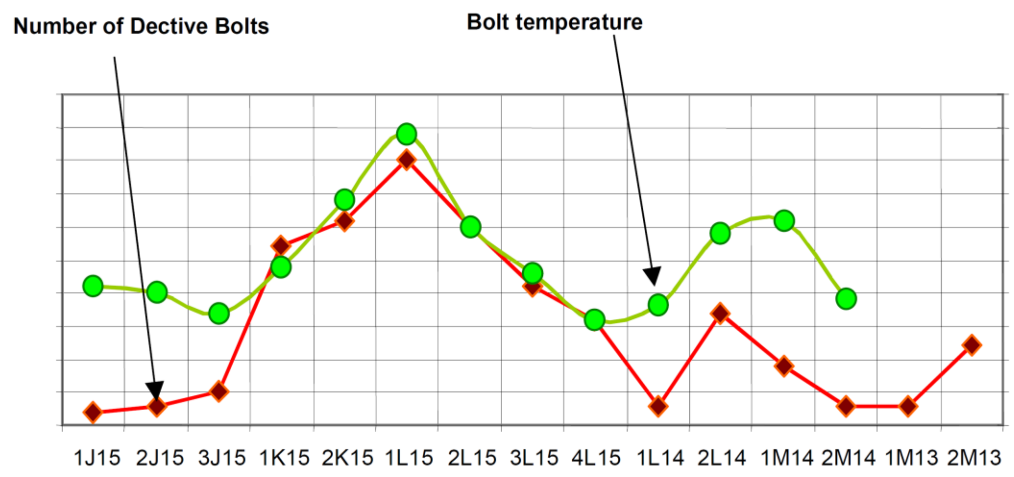
Need for predictive models to assess maintenance requirements

Influent factors on Cracking

- **Irradiation dose:** Compilation of data shows that there is a threshold for in-service baffle bolt cracking and that the rate of cracking decreases for higher dose



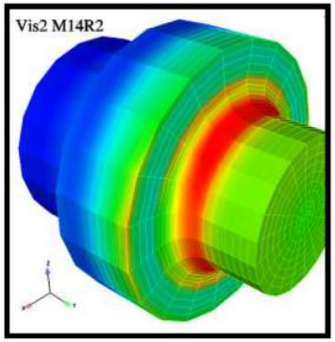
- **Temperature:** Correlation between the temperature calculated and the cracking level per equivalent colmn location in the reactor

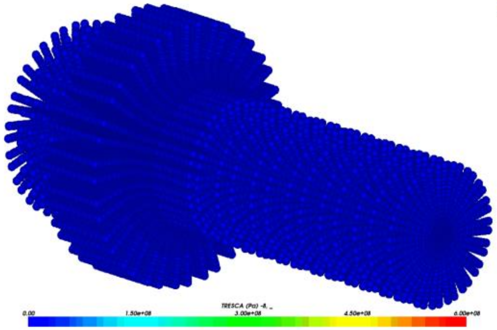


Influent factors on Cracking

- **Stress:** calculation of the loads applied to the collet-chunk region, position of the former.

Bolt location

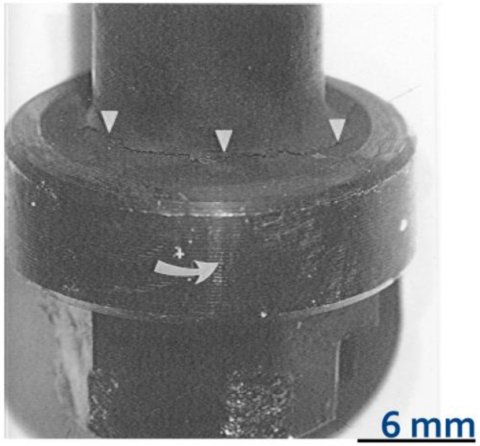




- **Material Chemical composition:** in BUG2 and FES2, bolts from same heat number, low Cr and high P steels with made in high frequency furnace

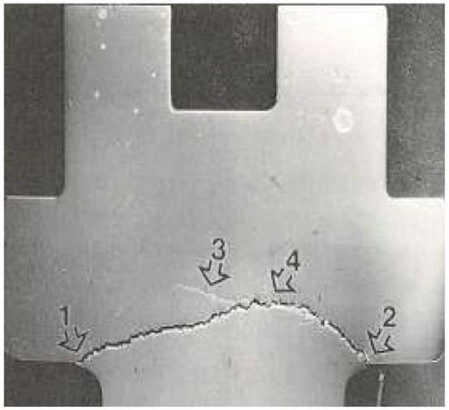
Need for more statistics to identify influent factors in cracking

Investigations on removed components

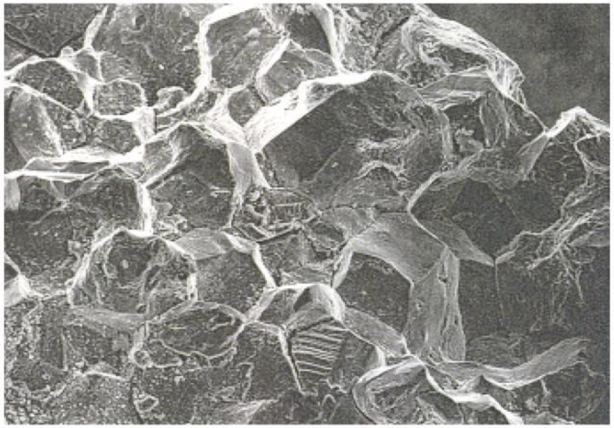


Visual examination of a cracked bolt

Intergranular aspect of a crack



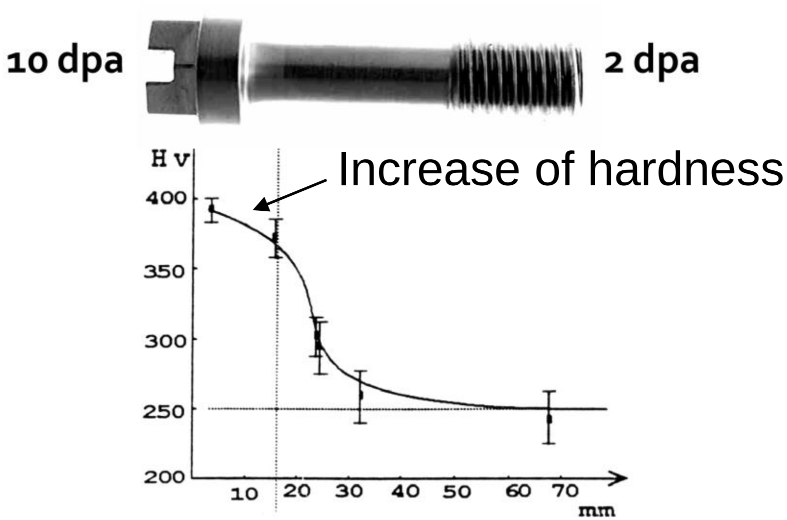
Metallographic examination of a cracked bolt



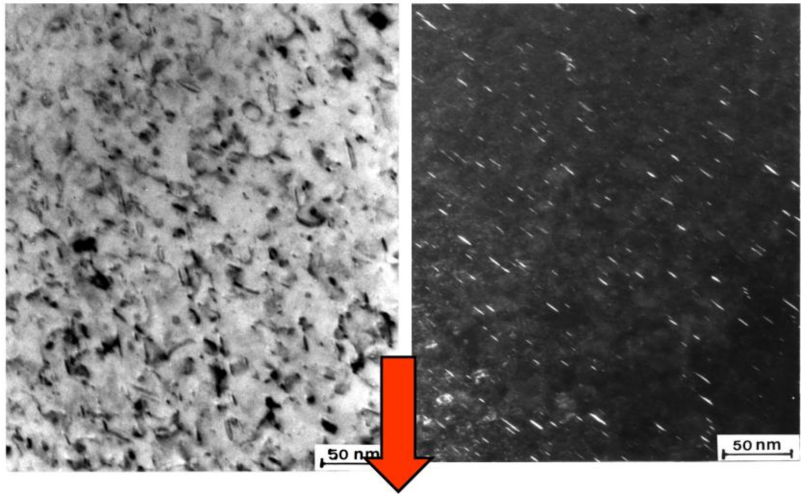
Intergranular fracture surface of a cracked bolt

Investigations on removed components

Hardness profile



Transmission Electron Microscopy



Most sensitive parameters controlling microstructure and hardening:

- ✓ Damage dose
- ✓ Irradiation temperature
- ✓ Initial state (dislocation density)
- ✓ Chemical composition (Radiation induced segregation

<u>Microstructure</u>

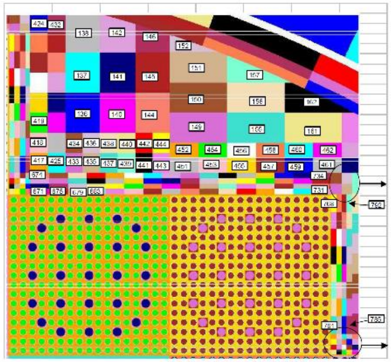
- ✓ Apparition of Frank loops
- ✓ Initial dislocation network (e.g. cold worked) disappeared
- ✓ Few cavities and bubbles for T< 350°C
- ✓ Precipitation ? Not in TEM but small precipitations in APT.

Reactor Vessel Internals Management

- In France, PWR are homogenous series
  - ✓900 MWe (3-loop plant): 2 kinds of NPP (CPo=:6 units, CPY : 28 units)
  - ✓1300 MWe (4-loop plant): 20 units
  - ✓1450 MWe (4-loop plant): 4 units
- No licensing process but a Safety Review for each 10 years of operation
- Safety Review include an Ageing Management Review
  - ✓Identify ageing mechanisms for components
  - ✓Define appropriate inspection
  - ✓Perform functionality evaluation
- Understanding and prediction of the behavior of bolt assembly
  - ✓Irradiated materials behavior
  - ✓Evaluation of loading (irradiation dose, temperature, stress and strain)
  - ✓Development of a predictive failure model
  - ✓Comparison with inspections

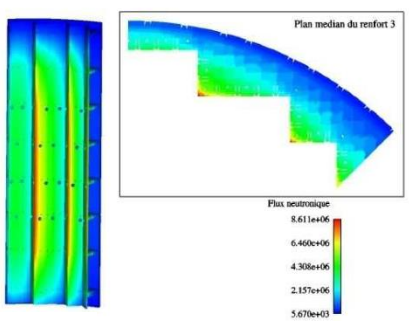
Methodology for loading calculations

Step 1

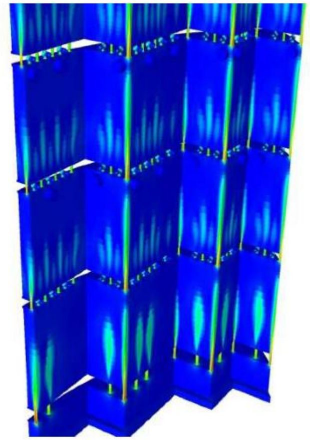


TRIPOLI computation

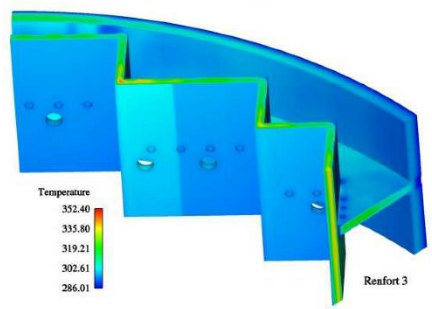
Interpolation of the heat deposit on the SYRTHES mesh



Step 2



Conjugate heat tranfert calculation SYRTHES/Code_Saturne



Temperature of the structures

Step 3

Interpolation of thermal fields on the mesh for Code_Aster

Stresses and strain in the whole structure

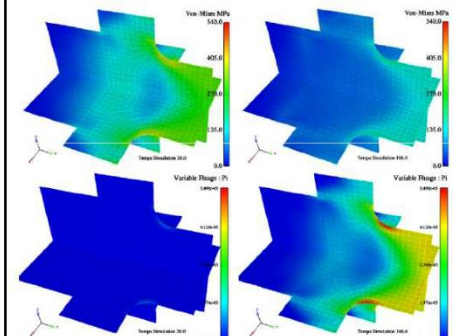
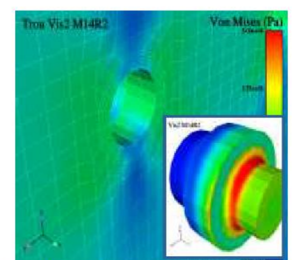


Figure 15 : Carte d'évolution de la contrainte et du fluage d'irradiation dans la vis.



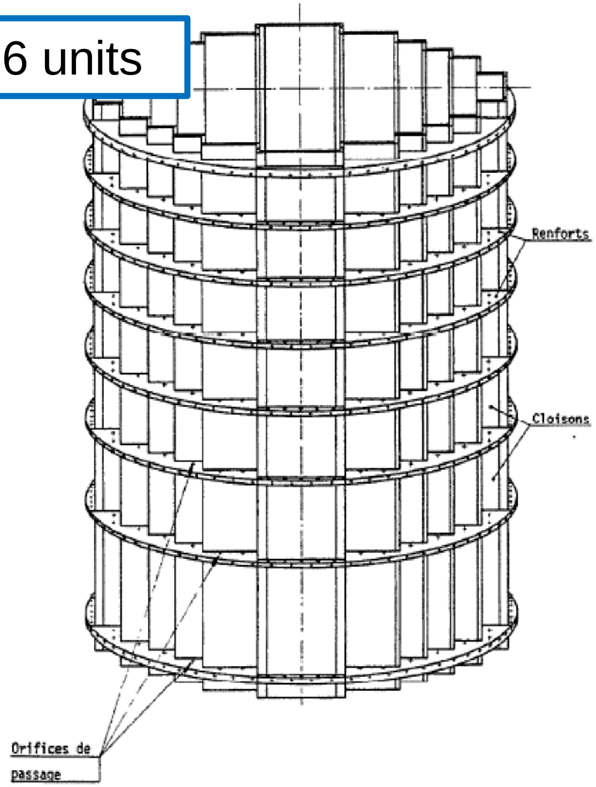
Code_Aster

Studies performed

- **Neutronics calculations**:
  - ✓ Different fuel configurations (standard, low fluence, …)
  - ✓ Different geometries of NPP: CPo, CPY (900MWe), PQY (1300 MWe), N4
- **Temperature calculations**:
  - ✓ Different hydraulic configurations for CPo (downflow, upflow)
  - ✓ Different geometries of NPP: CPo, CPY (900MWe), PQY (1300 MWe), N4
- **Mechanical calculations**:
  - ✓ Different loading conditions (downflow, upflow, fuel configurations)
  - ✓ Different material law coefficients (especially for void swelling)

Design of internals in French NPPs

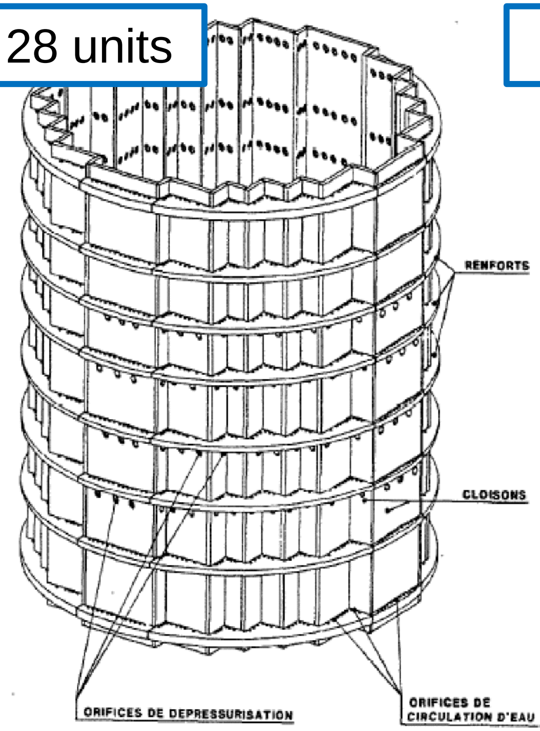
6 units



CPo
900 MWe

1st start up: 1977
Flow circulation: downflow up to 90's

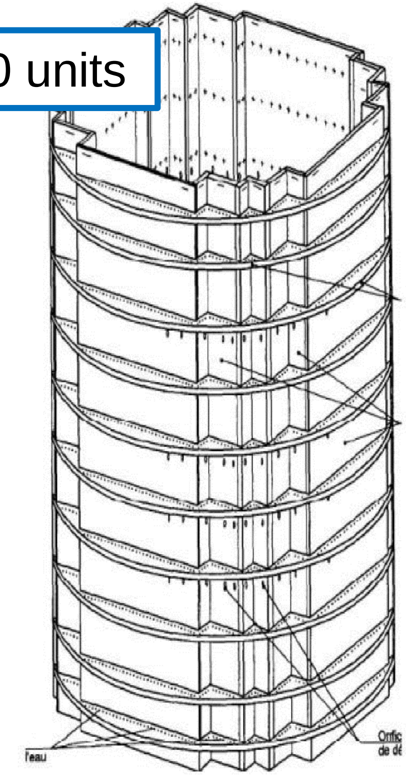
28 units



CPY
900 MWe

1st start up: 1978
Flow circulation: up flow

20 units

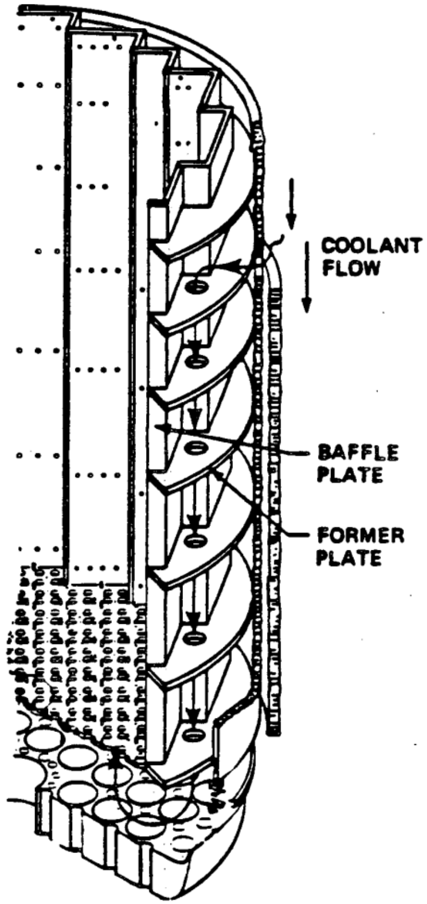


PQY
1300 MWe

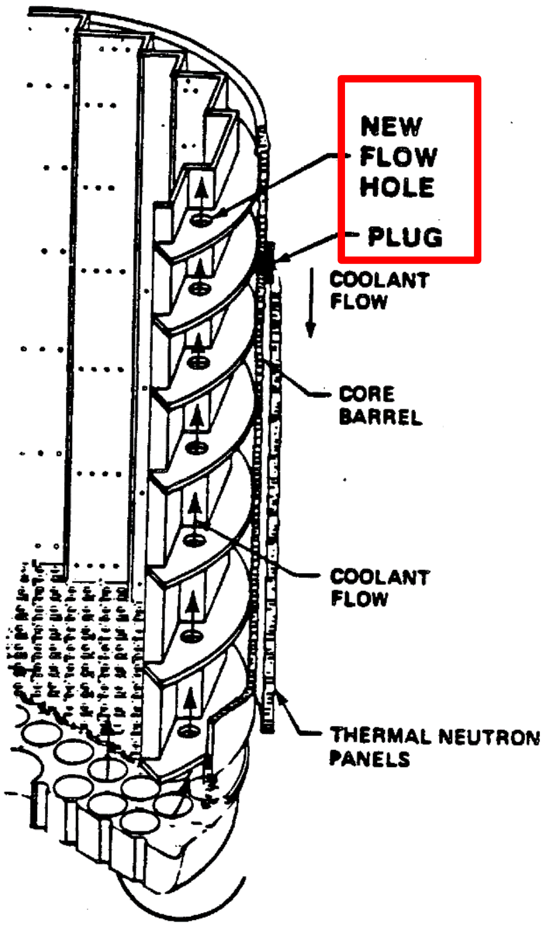
1st start up: 1984
Flow circulation: up flow

Downflow to upflow conversion

Reduce the differential pressure between the inner baffle region and the baffle to former region
→ avoid baffle jets and high stresses.



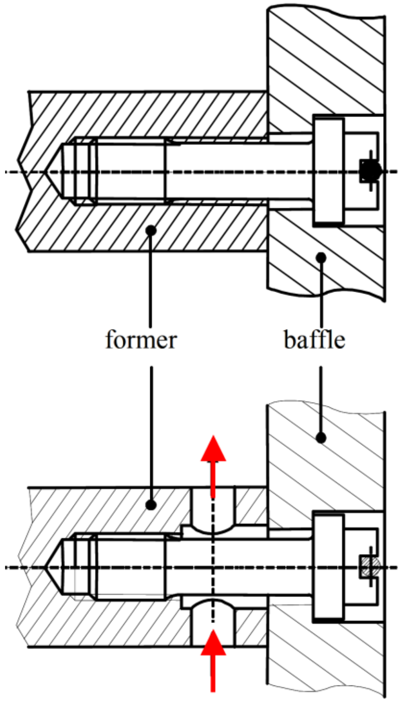
Down flow configuration



Up flow configuration

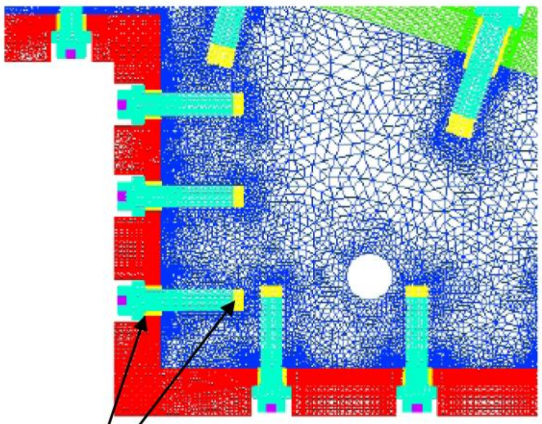
Baffle to Former Bolt Designs

Bolt cooling



CPY and PQY

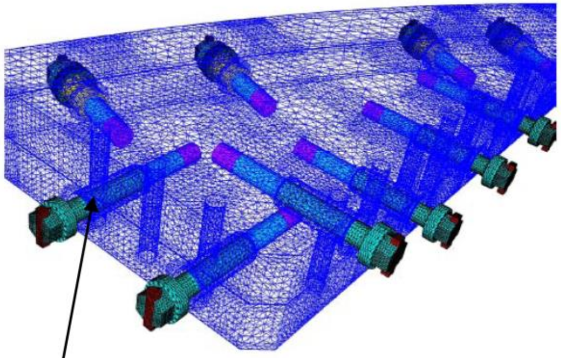
CPo



Confined Water

Fluid flow in the formers

CPY

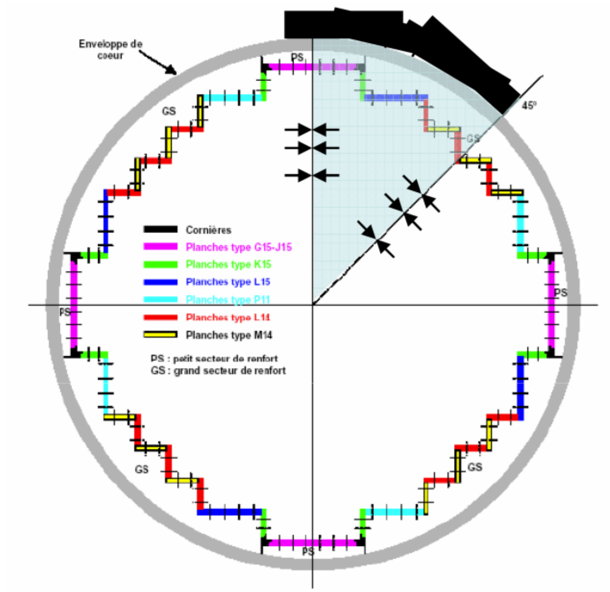


Water flow

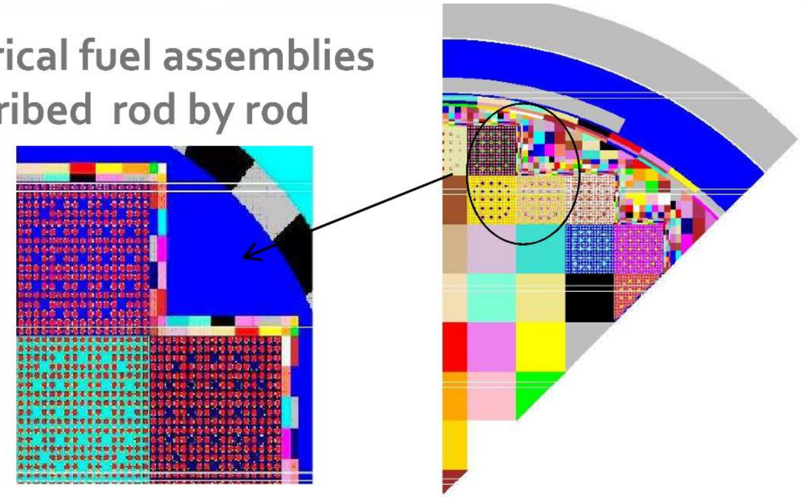
And also up flow conversion and low fluence operation

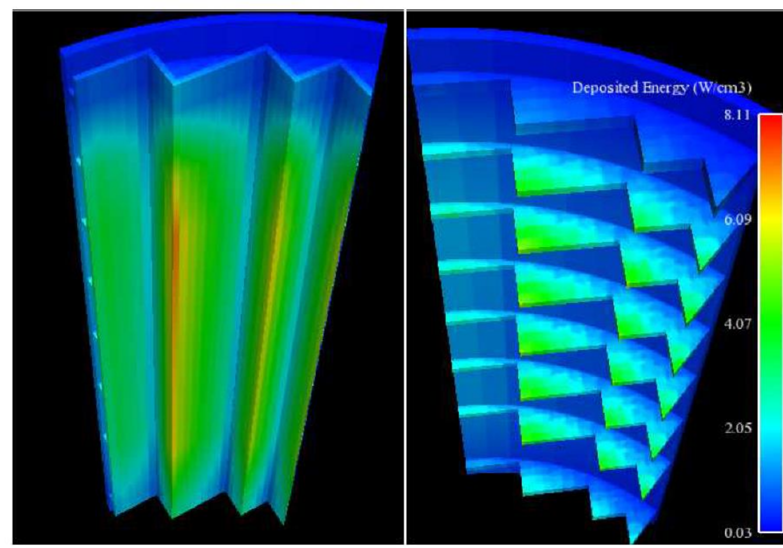
Neutron flux calculations

Calculation in 1/8 of core



Peripherical fuel assemblies described rod by rod

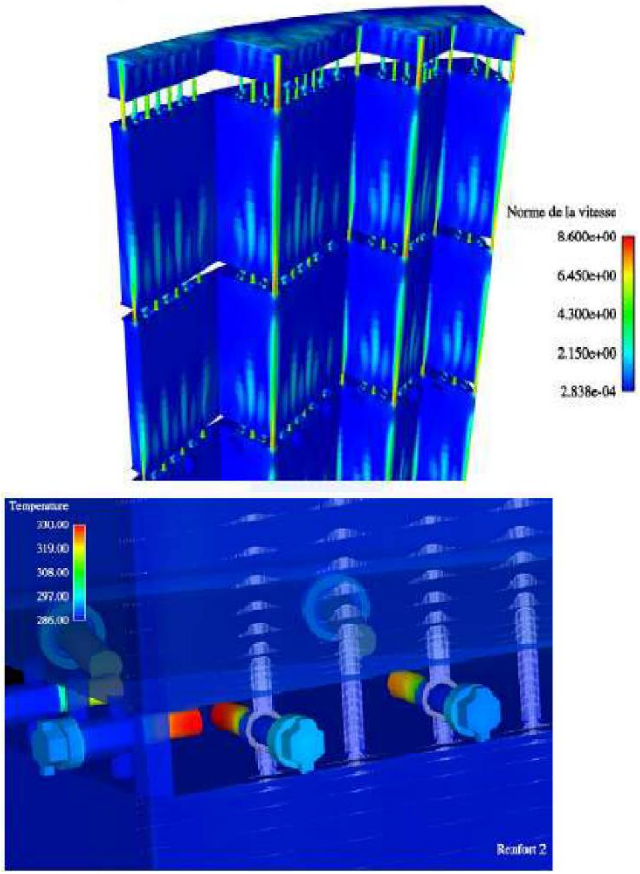




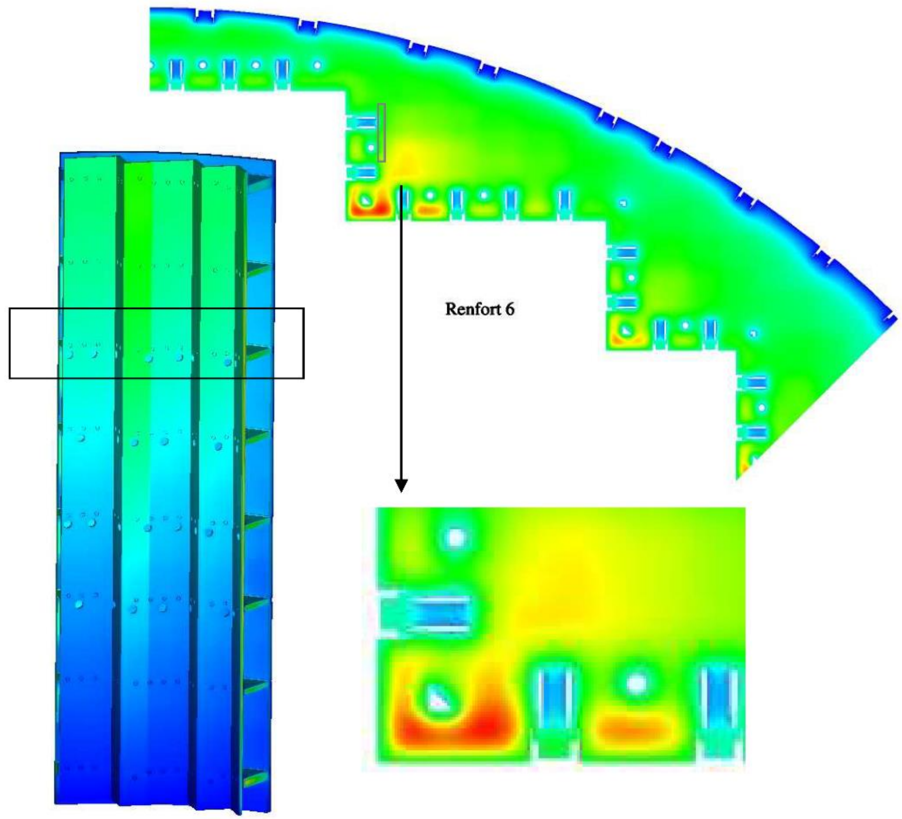
- ✓ TRIPOLI4 (CEA Monte Carlo code)
- ✓ CEA v5.1.1 library (jeff3.1.1 based-on library)
- ✓ 3 sources (γ fission, γ fission product, neutron)
- ✓ Dpa are coming from IRDF2002 evaluation

Temperature calculations

Fluid : velocity and temperature



Solid : temperature



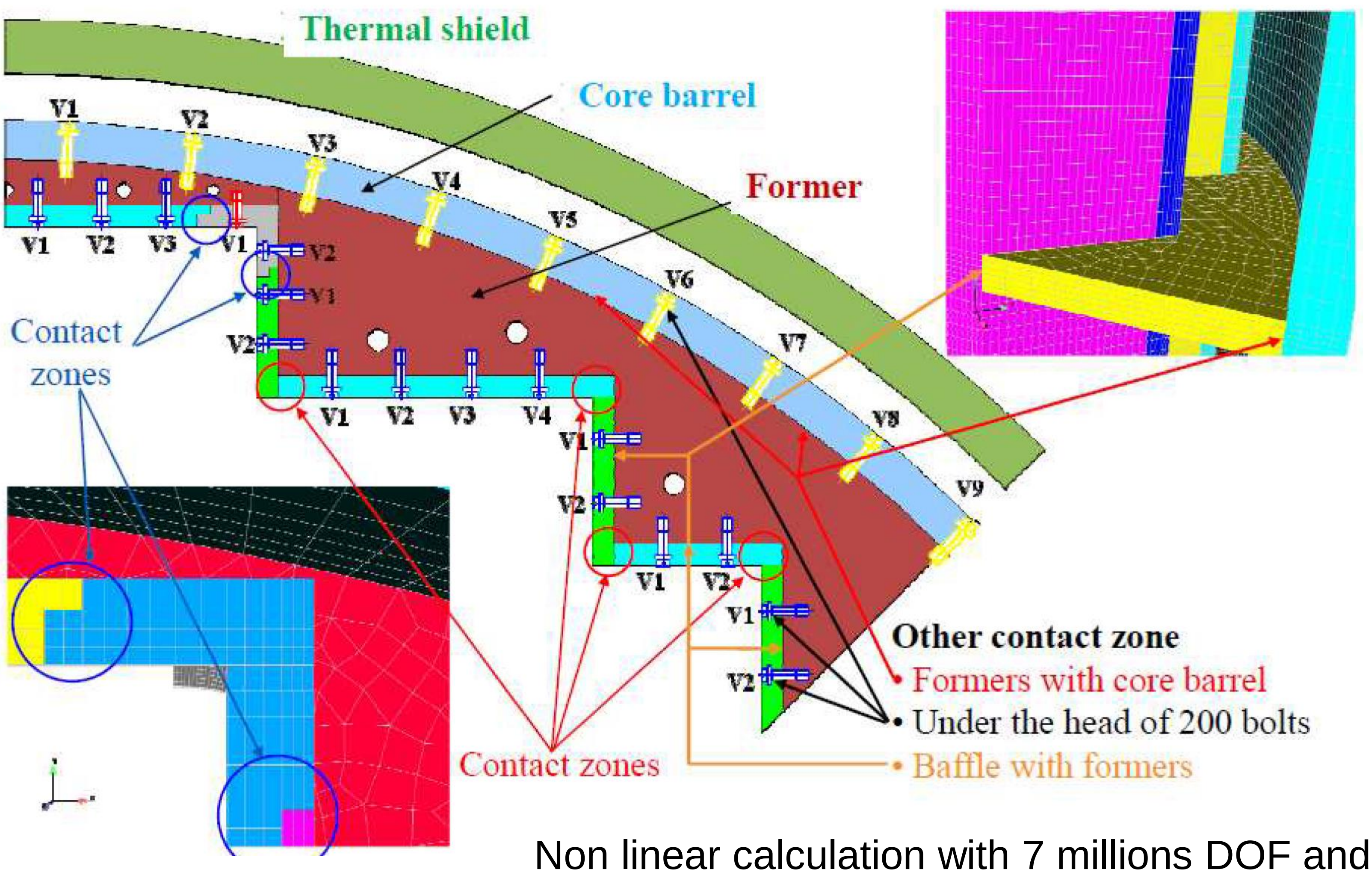
Code_Saturne

5 millions of cells

Syrthes

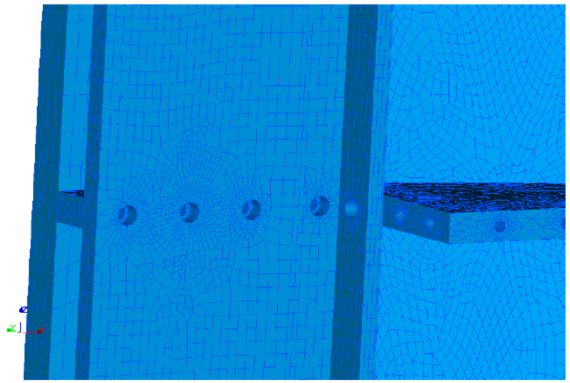
30 millions tetrahedra

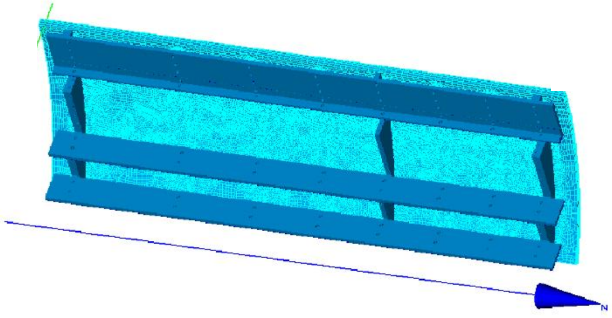
Mechanical calculations



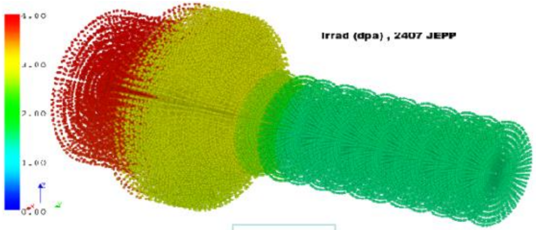
Non linear calculation with 7 millions DOF and 50000 contact nodes

Global model for the deformation of the structure

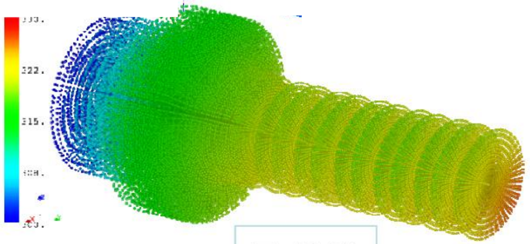




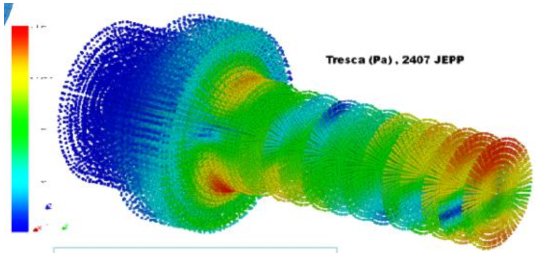
Local model for strain stress of the bolts



dpa

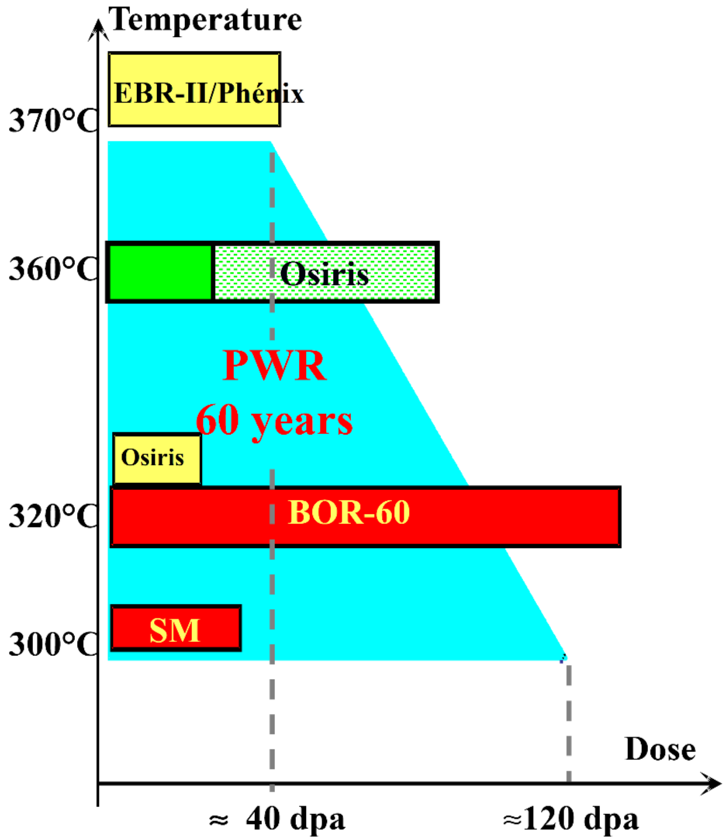


T (°C)



Stress (MPa)

Tests on Irradiated Austenitic Stainless steels



Determination of irradiated stainless steel constitutive law

- Irradiation in Material Testing Reactors
- Decommissioned Materials
- Materials representative of PWR Internals
  - ✓ SA304L
  - ✓ CW316 and CW316L
  - ✓ 308 Welds
- Specimens
  - ✓ Density specimen → Sweeling
  - ✓ Pressurized tubes → Creep
  - ✓ Tensile specimens → Tensile properties
  - ✓ CT specimens → Fracture toughness
  - ✓ Microstructural investigation

Irradiated stainless steel material constitutive law

$$\dot{\varepsilon} = \dot{\varepsilon}_e + \dot{\varepsilon}_{th} + \dot{\varepsilon}_p + \dot{\varepsilon}_{ic} + \dot{\varepsilon}_s$$

Elasticity

Thermal Strain

Plasticity

Irradiation creep

Swelling

- ❑ Thermo-elasticity
  - ✓ Young's modulus, Poisson's ratio, coefficient of thermal expansion are all based on non irradiated data base
- ❑ Plasticity
  - ✓ For non irradiated steels: handbook data
  - ✓ For irradiated steels: test after irradiation at RT and 330°C
- ❑ Irradiation creep
  - ✓ Experimentally based and cross validated on decommissioned materials
  - ✓ Thermal creep is extremely low at about 300°C.
- ❑ Irradiation swelling
  - ✓ No coupling between irradiation creep and swelling due to the lack of experimental data concerning PWR operating conditions.

Determination of plasticity law

- Yield Strength ($Rp_{0.2}$), ultimate stress (R_m) and uniform elongation (e_U)

$$R_{0.2} = R^0_{0.2}(T)\eta_1(c)\xi_1(d)G_1(d, F_1(T))$$

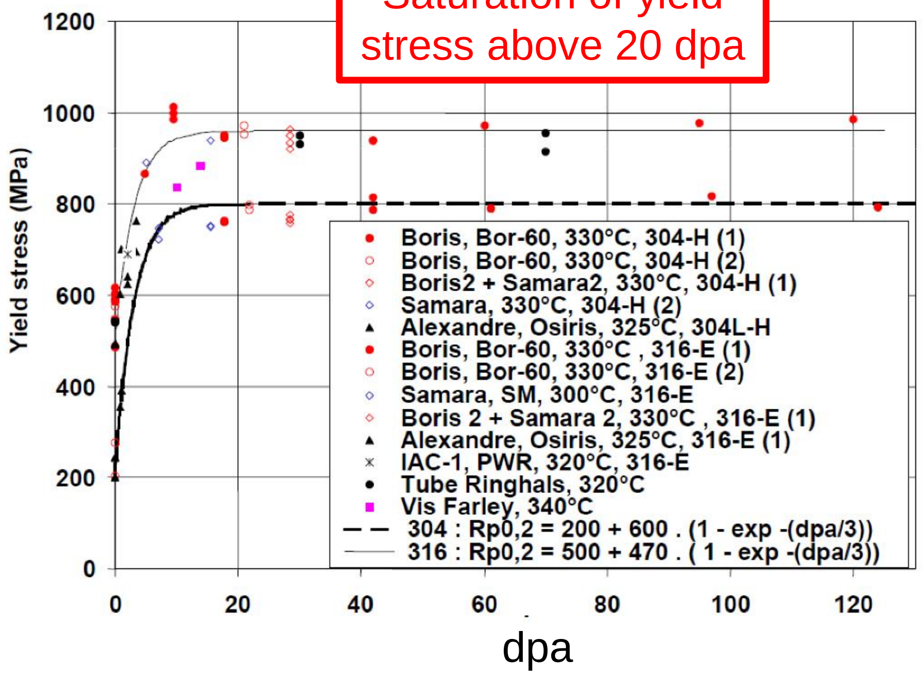
Temperature dependence fitted to data from unirradiated materials

Work hardening dependence fitted to data from irradiated materials at 330°C

Dose dependence fitted to data from irradiated materials

Cross dependence T and dose fitted on irradiated data d>20 dpa and T=20°C

Saturation of yield stress above 20 dpa

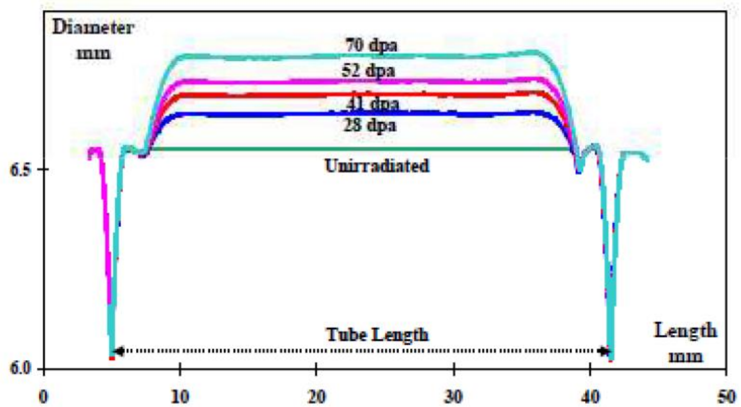


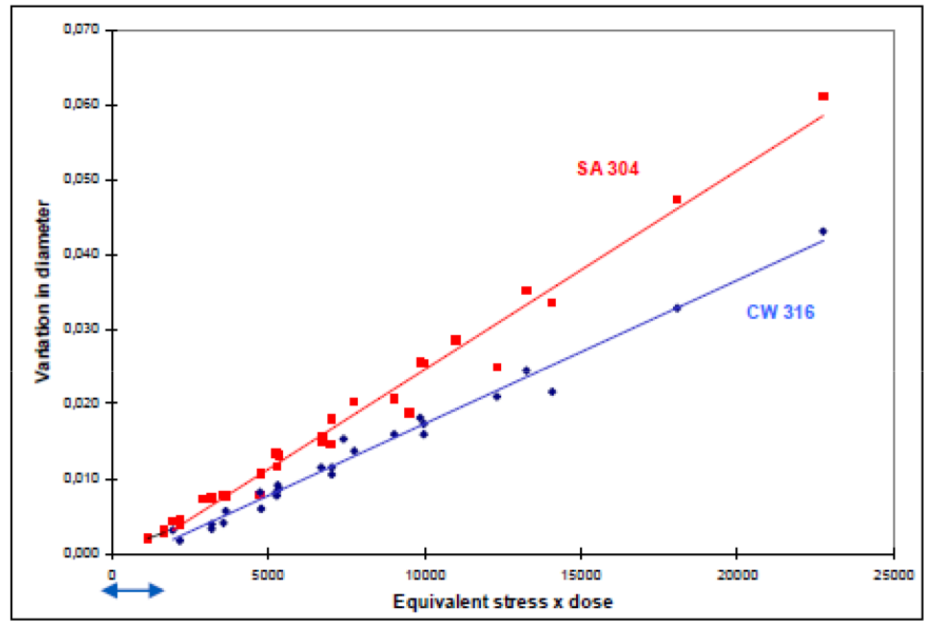
Data obtained from post irradiation tensilte tests on MTR irradiated stainless steels

Saturation observed for $Rp_{0.2}$, R_m and e_U due to complex microstructural interplays

Determination of irradiation creep law

Creep assessment by means of gas pressurized tubes deformation measurements different dose levels





Threshold for creep occurrence

- ❑ ε_i: deformation due to irradiation creep
  - ✓ For σ.Φ> C, ε = (A. σ.Φ)-B
  - ✓ A the compliance of creep (%/Mpa-dpa), σ von Mises stress, Φ fluence in dpa, B is the threshold for the initiation creep

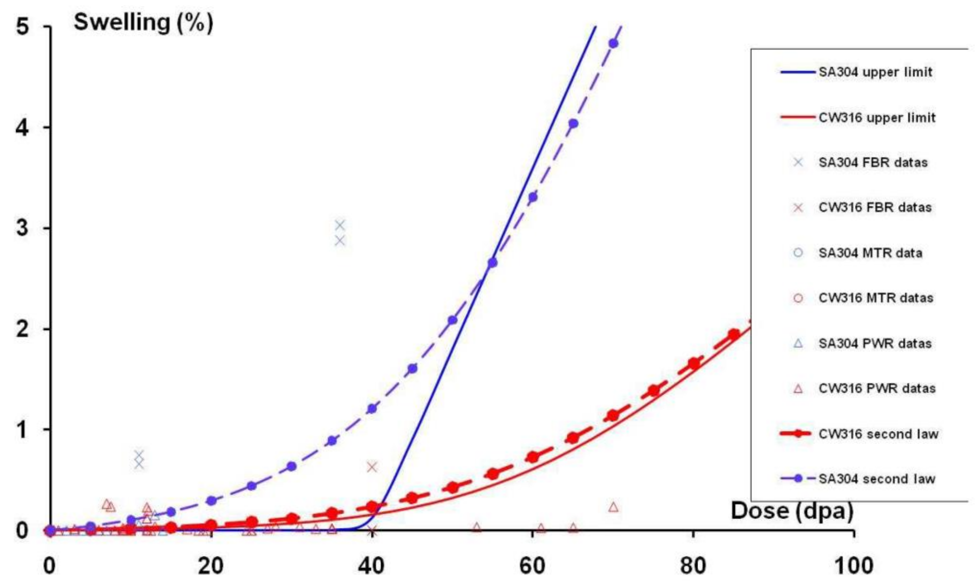
Determination of swelling law

- **No macroscopic swelling data for PWR condition**
  - ✓Use of a general Foster Flinn equation

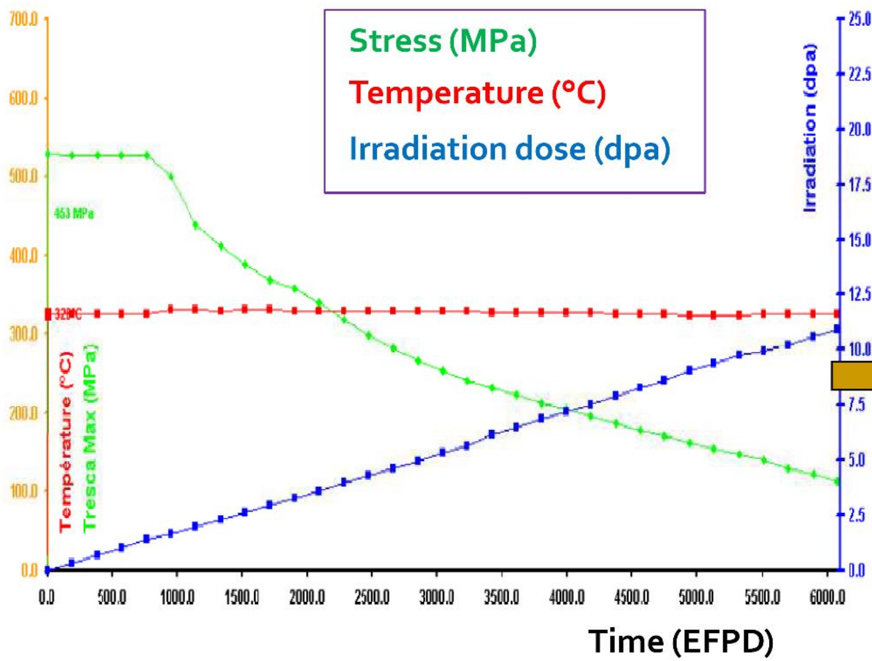
$$\frac{\Delta V}{V} = R\left(\Phi + \frac{1}{\alpha}\log\left(\frac{1+\exp(\alpha(\Phi_0 - \Phi))}{1+\exp(\alpha\Phi_0)}\right)\right) \qquad \zeta_g(T) = \frac{1}{2}\left(1+\tanh(\mu_g(T - T_c^g))\right)$$

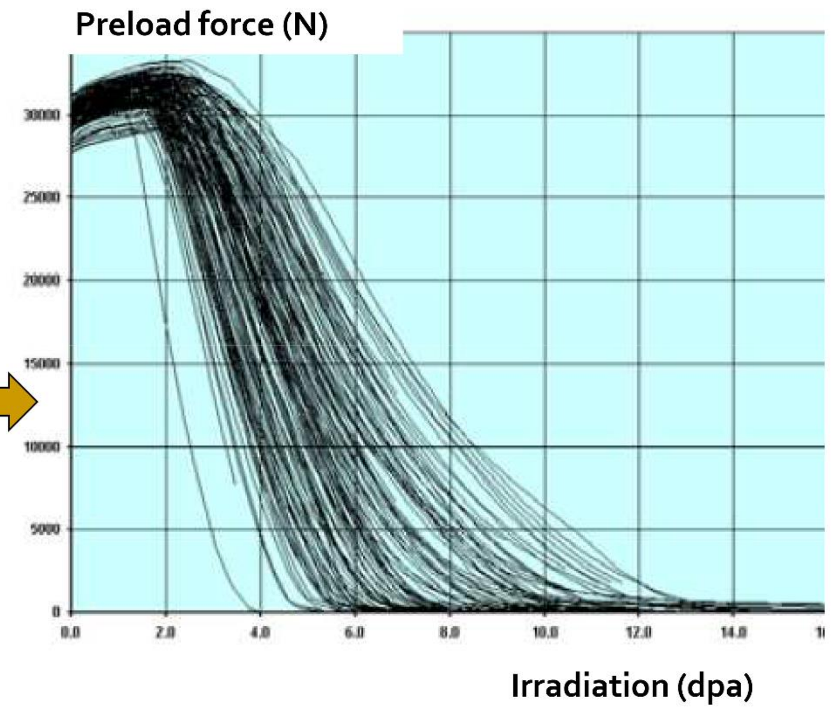
- **Two material parametrization especially for SA304**
  - ✓No swelling before 40 dpa and then linear evolution
  - ✓Swelling since beginning

Fitted on experimental data obtained from experimental reactor and attemptedly adapted to PW: **considerable overestimation of swelling**



Mechanical calculation using the constitutive law

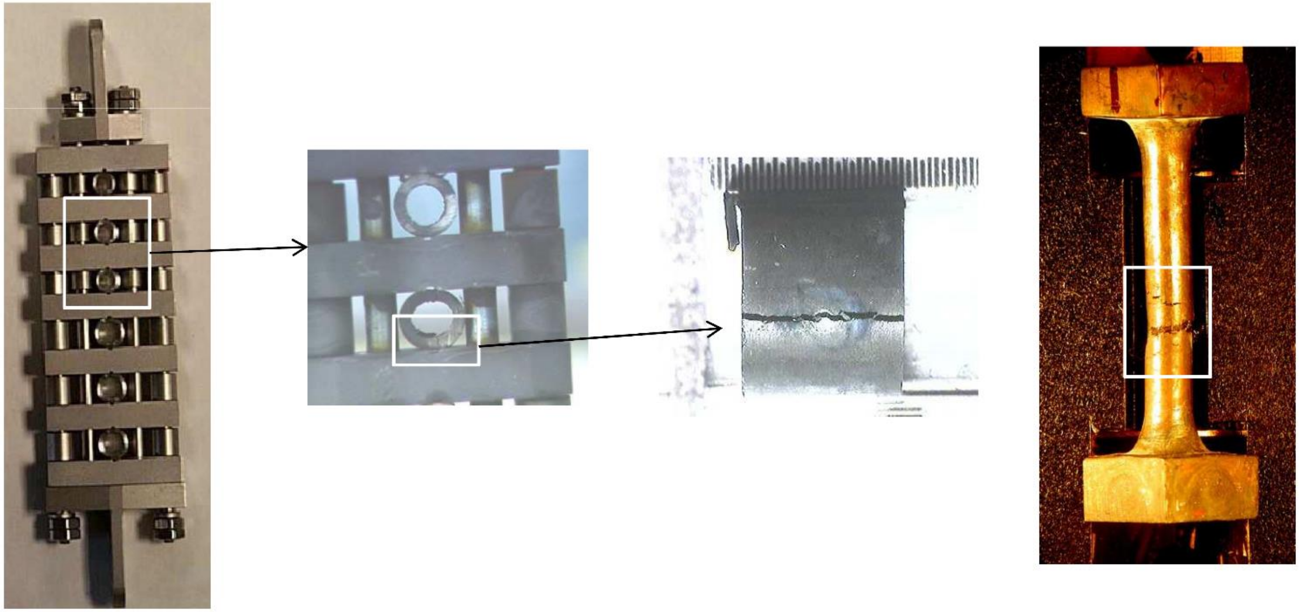




Decrease of stress due to irradiation creep is not compensated by the swelling law

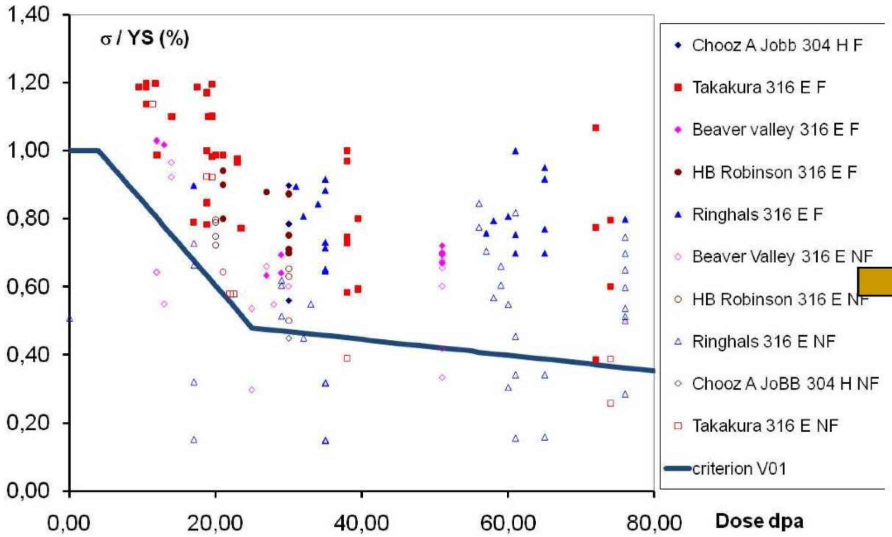
IASCC Criterion

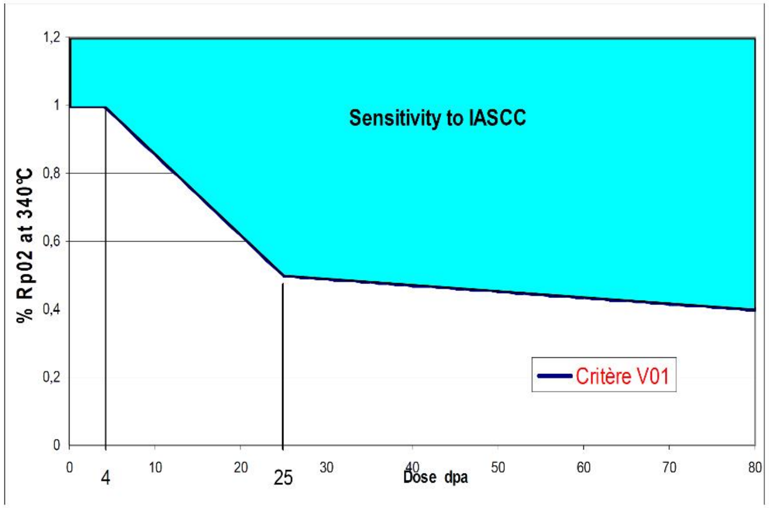
- ❑ Definition of a sensitivity area for IASCC initiation depending on dose
  - ✓ Experimentally based during international programs
  - ✓ Post irradiations O-ring and CL tests in environment on decommissioned materials between 290 and 340°C.
  - ✓ Different applied stress <YS and different dose
  - ✓ Few in-pile tests



IASCC Criterion

- Definition of a sensitivity area for IASCC initiation depending on dose
  - ✓ Function of the applied stress divided by YS at dose and temperature
  - ✓ Database internationally shared



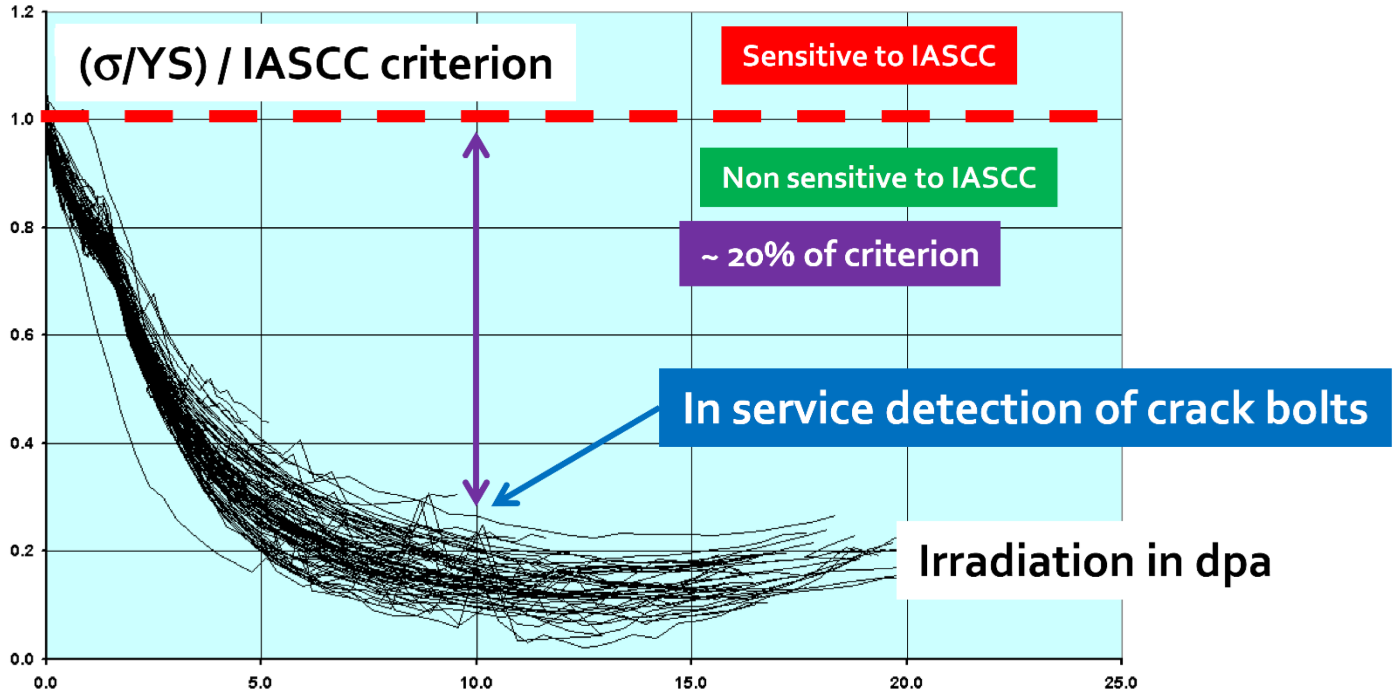


Three sensitivity domains:

- ✓ dose < 4dpa → σ_{TRESCA}/YS>1
- ✓ 4dpa <dose< 25 dpa → σ_{TRESCA}/YS>(577/525)-(13/525)*dose
- ✓ 25 dpa< dose → σ_{TRESCA}/YS> (699/1300)-(3/1300)*dose

Calculations vs IASCC Criterion

Stress calculations vs IASCC criterion



- ✓ Stress in the head-shank junction of the bolts decrease rapidly and the applied stress remains below the IASCC criterion
- ✓ Field experience: IASSC occurs starting at 10 dpa

Validity of loading calculation and/or IASCC criterion ?

Conclusion and work in progress

- ❑ Validity of calculation methodology
  - ✓International benchmark on methodologies in progress (4 companies on 3 continents)
  - ✓Uncertainties propagation on load or dose calculation
- ❑ Constitutive law for irradiated stainless steels in PWR
  - ✓Creep at high stress (~YS)?
  - ✓Swelling law for PWR conditions ?
  - ✓Physically based constitutive model ?
- ❑ Better definition of realistic loading
  - ✓Role of dynamic straining during shut-down, start-up or reactor trip
  - ✓Role of chemistry in confined area of bolts ?
- ❑ Improvement of IASCC criterion
  - ✓Definition of a criterion based on deformation rate or other parameters ?
  - ✓Complexity of the stress and strain on the bolt in service compared to lab specimen
  - ✓ Data for low irradiation dose (< 10 dpa)
  - ✓In pile-test versus post irradiation test
  - ✓Role of surface state (residual stress due to machining) on IASCC criterion