ÉCOLE POLYTECHNIQUE
FÉDÉRALE DE LAUSANNE

CRPP

Materials development for fusion applications: current status and challenges

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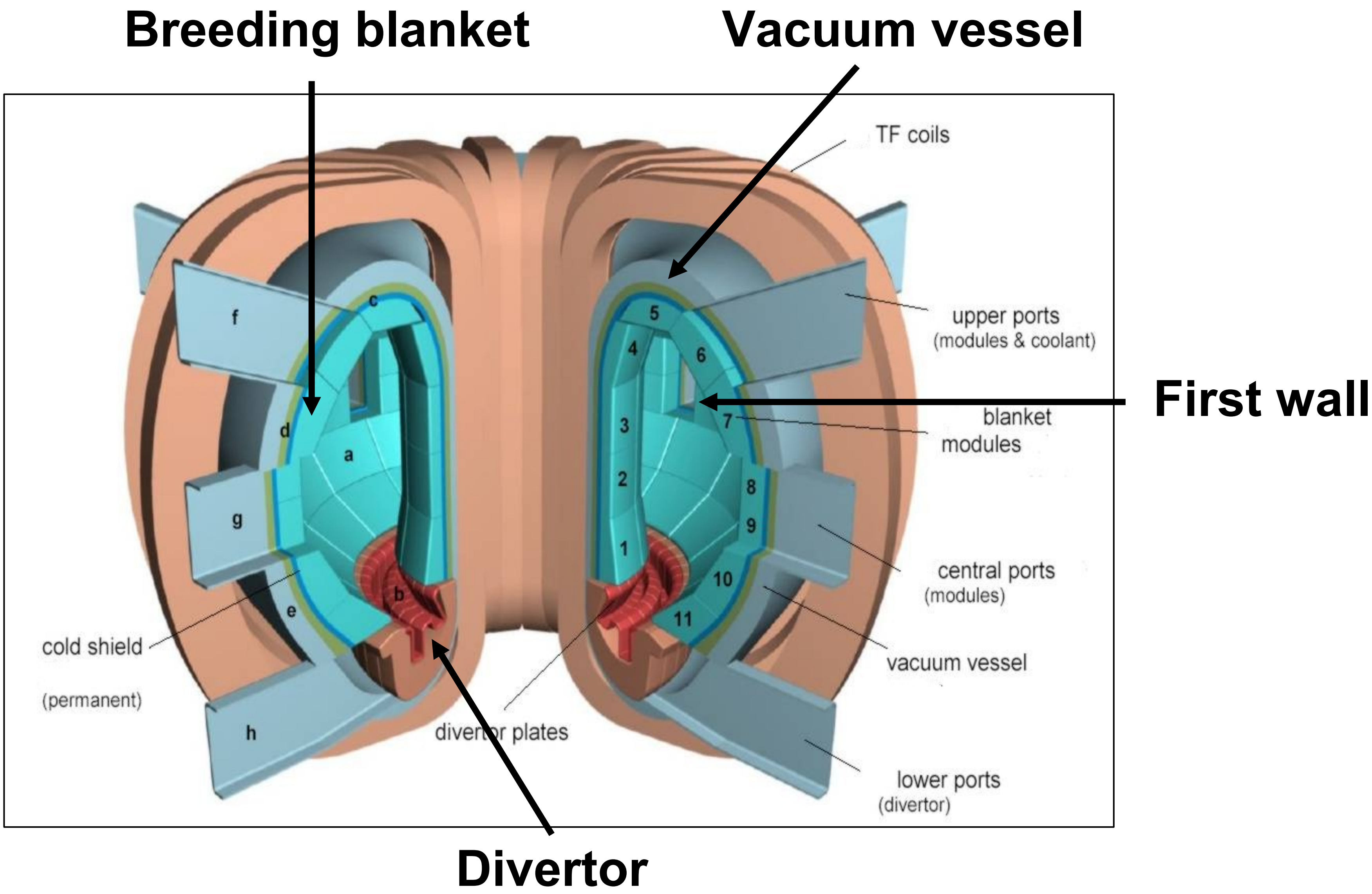
Outline

- Materials for ITER
- Materials for DEMO
- CRPP Fusion Materials group research activities


Breeding blanket
Vacuum vessel
TF coils
upper ports
(modules & coolant)
First wall
blanket
modules
central ports
(modules)
vacuum vessel
lower ports
(divertor)
cold shield
(permanent)
divertor plates
Divertor
a
b
c
d
e
f
g
h
1
2
3
4
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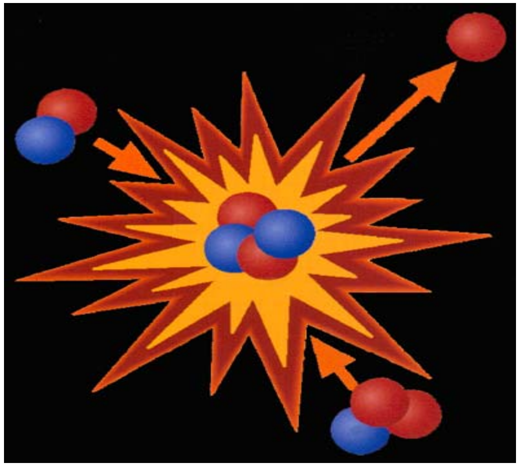
10
11

Wealth of materials for fusion reactor components

- Vessel and In-Vessel components
  - Plasma facing materials
  - Structural materials
  - Joining technology
  - Ceramics for electrical insulation
- Diagnostic components
  - Wires/cables, windows, optical fibres, mirrors/reflectors
- Magnets
  - Metallic and non-metallic materials, conductor materials

Products of D-T fusion reaction

deuterium ion



neutron 14.1 MeV

tritium ion

Exposition of the materials surrounding the burning plasma to radiation field of:

- high-energetic neutron
- plasma escaping particles

Specificity of fusion neutron irradiation (1)

For the fusion neutron spectrum, different nuclear reactions with the nuclei of the surrounding materials are possible:

neutron < 1 MeV: (n, n) elastic and inelastic scattering
(n, γ) capture

Primary knock-on atom energy up to about 40 keV.

****

neutron > 1 MeV: Transmutation reactions come into play
(n, p), (n, d) (n, t) (n, He)… H, and He production

Primary knock-on atom energy up to several hundreds keV.

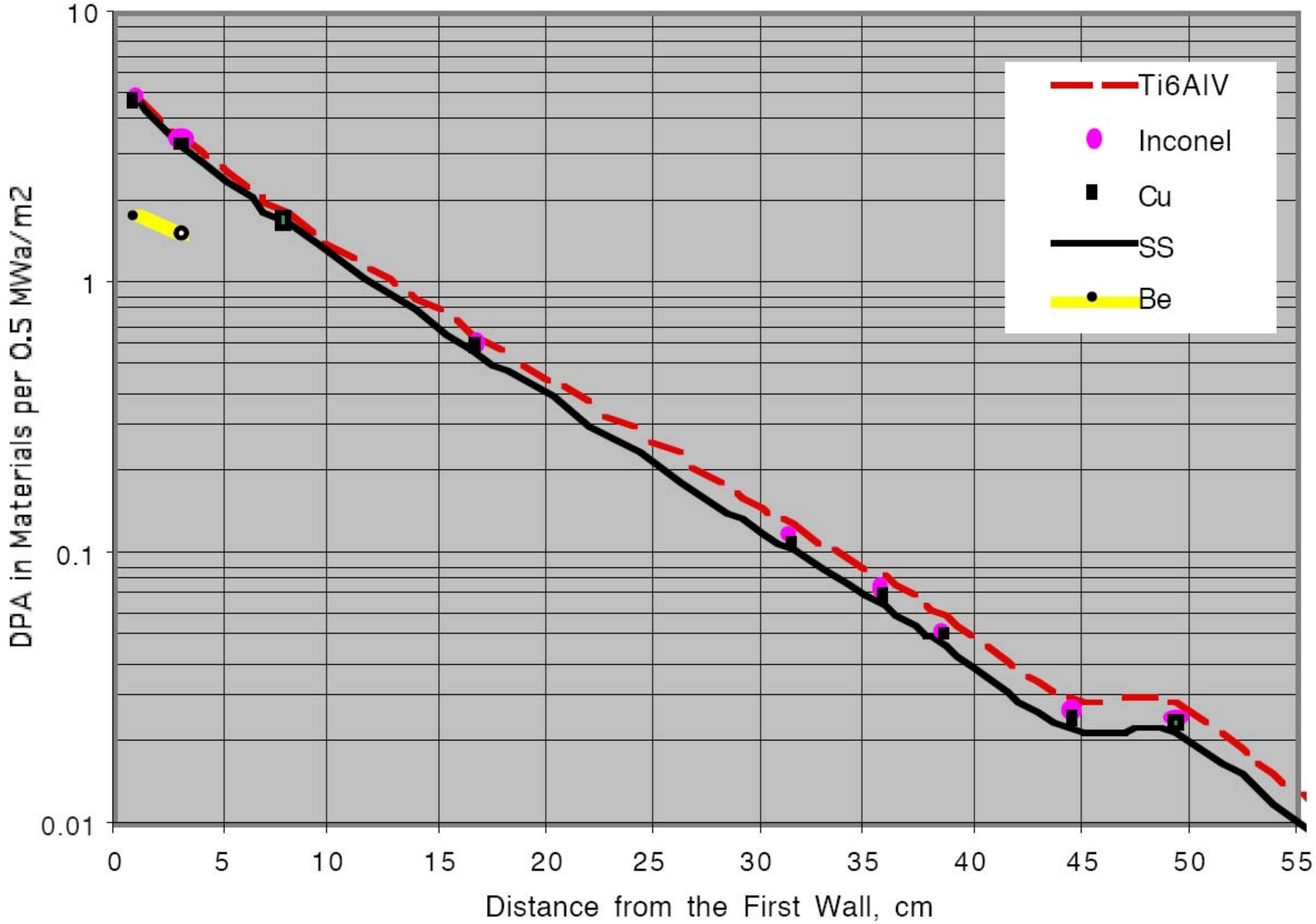
Specificity of fusion neutron irradiation (2)

Expected dose at the end of life of a reactor.

Fusion reactor first wall: 100 to 200 dpa (displacement per atom) with a helium production of about 10 appm He/dpa and of hydrogen 40 appm H/dpa.

Fission reactor vessel: 0.1 dpa without He.

Neutron damage though blanket thickness



Radial dpa distributions in the bulk shielding blanket for different materials

Evolution of the microstructure (1)

- **Transmutation nuclear reactions create impurities:**

  He gas atoms, H gas atoms

- **Atomic displacement cascades yield point defects:**

  vacancies, intersitials, clusters of vacancies, clusters of interstitials

- **Segregation of alloying elements**

Evolution of the Microstructure (2)

- **The final microstructure results from a balance between radiation damage and thermal annealing**

- **<u>Complex secondary defects:</u>**

  Small defect clusters

  Interstitial dislocation loops

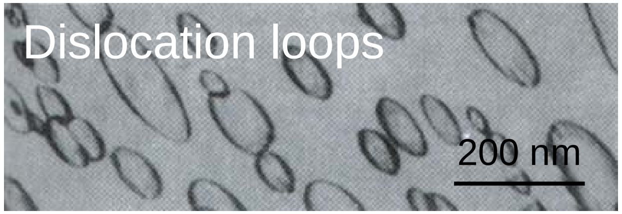
  Vacancy dislocation loops

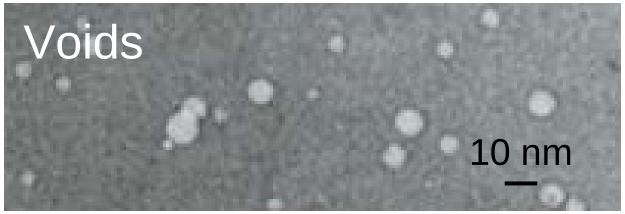
  Stacking fault tetrahedra

  Precipitates

  Voids, *void growth leads to swelling!!*

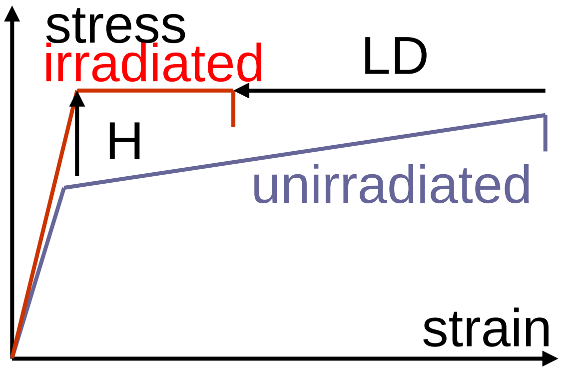
  He bubbles





Evolution of the properties

- **<u>Chemical composition:</u>**
  - **Change in the chemical composition**
- **<u>Physical properties:</u>**
  - **Decrease of electrical conductivity (low temperatures)**
  - **Decrease of thermal conductivity (ceramic materials)**
- **<u>Mechanical properties:</u>**
  - **Hardening (H)**
  - **Loss of ductility (LD)**
  - **Loss of fracture toughness**
  - **Loss of creep strength**



- **<u>Dimensions:</u>**
  - **Swelling, irradiation creep, irradiation growth**
- **<u>Environmental effects:</u>**
  - **Irradiation-assisted stress corrosion cracking**
- **<u>Radioactivity:</u>**
  - **Activation effects**

Plasma Facing Materials

- **The qualification of plasma facing materials is very demanding and limited by their capability for absorbing heat and minimizing plasma contamination.**

High heat flux of energetic particles: 0.1-20 MW/m^2

High temperatures: 500-3200°C

Sputtering erosion

High levels of neutron-irradiation: 3-30 dpa/year

Off-normal events: plasma disruptions

Hydrogen trapping

Plasma facing armor materials in ITER

Different plasma facing materials:

- ***Beryllium:*** **armor material for about 80% of the exposed surface to plasma, low effect on plasma contamination, low radiative power losses, low bulk tritium inventory.**

- ***Tungsten:*** **used on divertor baffles where erosion lifetime is the key issue. Lower erosion rate as compare to Be and C, low tritium retention. However, a smalll amount of W in the plasma can lead to very large radiation power loss.**

- ***Carbon fibre composite CFC:*** **used in the area exposed to large energy excursions (lower part of the divertor vertical target) owing to his absence of melting, high thermal shock and thermal fatigue and high thermal conductivity.**

Structural materials in ITER (1)

- ***Stainless steel 316L(N)-IGX:*** **austenitic steel for vacuum vessel and in-vessel components. L stands for low carbon content, N means that Nitrogen is content is controlled and IGX means ITER-grade with X the procurement specification (divertor cassette, thin-walled tubes etc…).**

  This steels is qualified in many design codes. Minor modifications in the chemical composition are required to improve the radiological properties.

  Reduction of Co content from 0.25% down to 0.05% decrease the decay heat by 20%.

  Reduce Nb to < 0.01%.

  Keep B content as low as possible to reduce He generation rate.

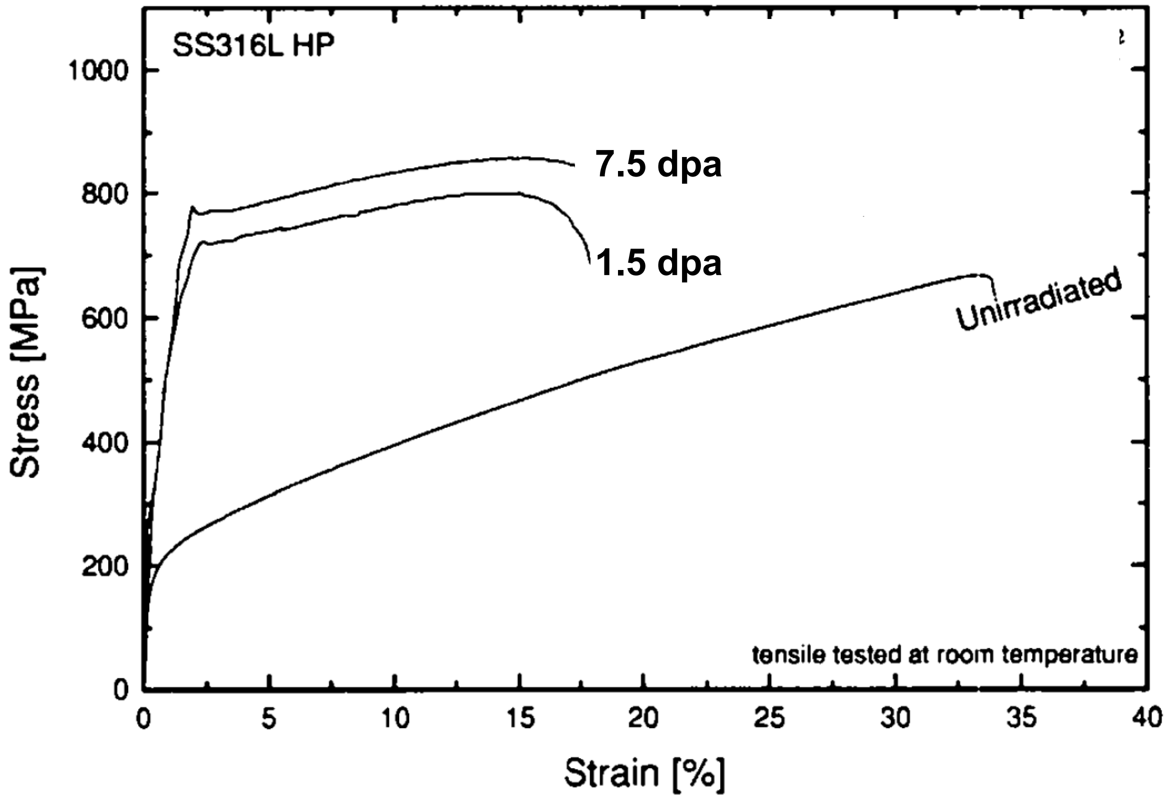
Structural materials in ITER (2)

- *Stainless steel 316L(N)-IGX:*

  At maximum fluence, about 2 dpa and 55 appm He, the irradiation temperature is between 100-300 °C. Hardening will occurs but uniform and total elongation (tensile test) will remain significant 10 and 20 % respectively.

  In the blanket manifolds and vacuum vessel the dose will remain quite low, < 0.06 dpa, resulting in no significant property changes.

Irradiation effect on the tensile properties of 316 steel after n-irradiation at 277 °C



Other structural materials in ITER

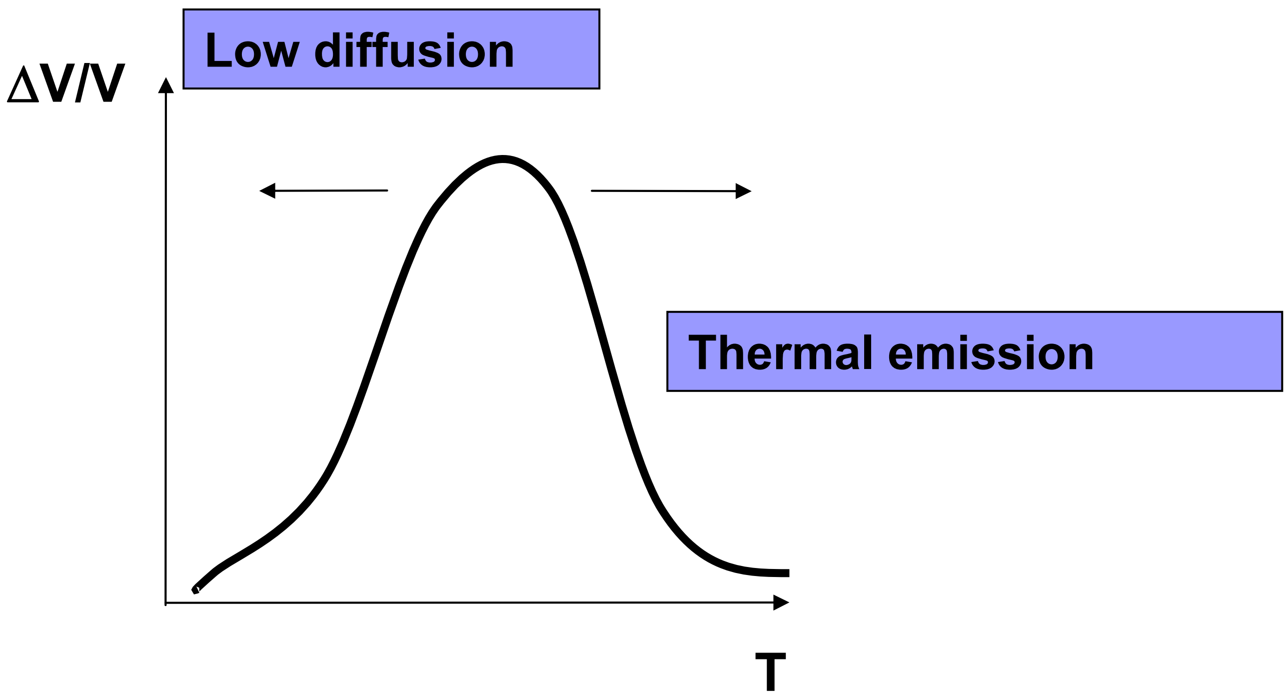
- Heat-sink Copper alloys: CuCrZr (first wall, divertor, limiter).
- High strength precipitation-hardened steel (bolting port plugs).
- Nickel-Aluminum bronze for nuts, bearing, connecting rods to connect plasma-facing components to the divertor.
- Ti alloys flexible cartridge to for blanket supports.
- Ni alloys for bolt attaching the shielding blanket modules through the flexible support.

Candidate Functional Materials

- **Selection of functional materials is very limited as it relies mainly upon the properties required by the envisaged function.**

Function	First wall	Breeding blanket	Divertor
Neutron multiplier material	-	Be, $Be_{12}Ti$, $Be_{12}V$, Pb	-
Tritium breeding material	-	Li, eutectic Pb-Li, Li-base ceramic materials	-
Coolant	-	Water, helium, eutectic Pb-Li, Li	Water, helium

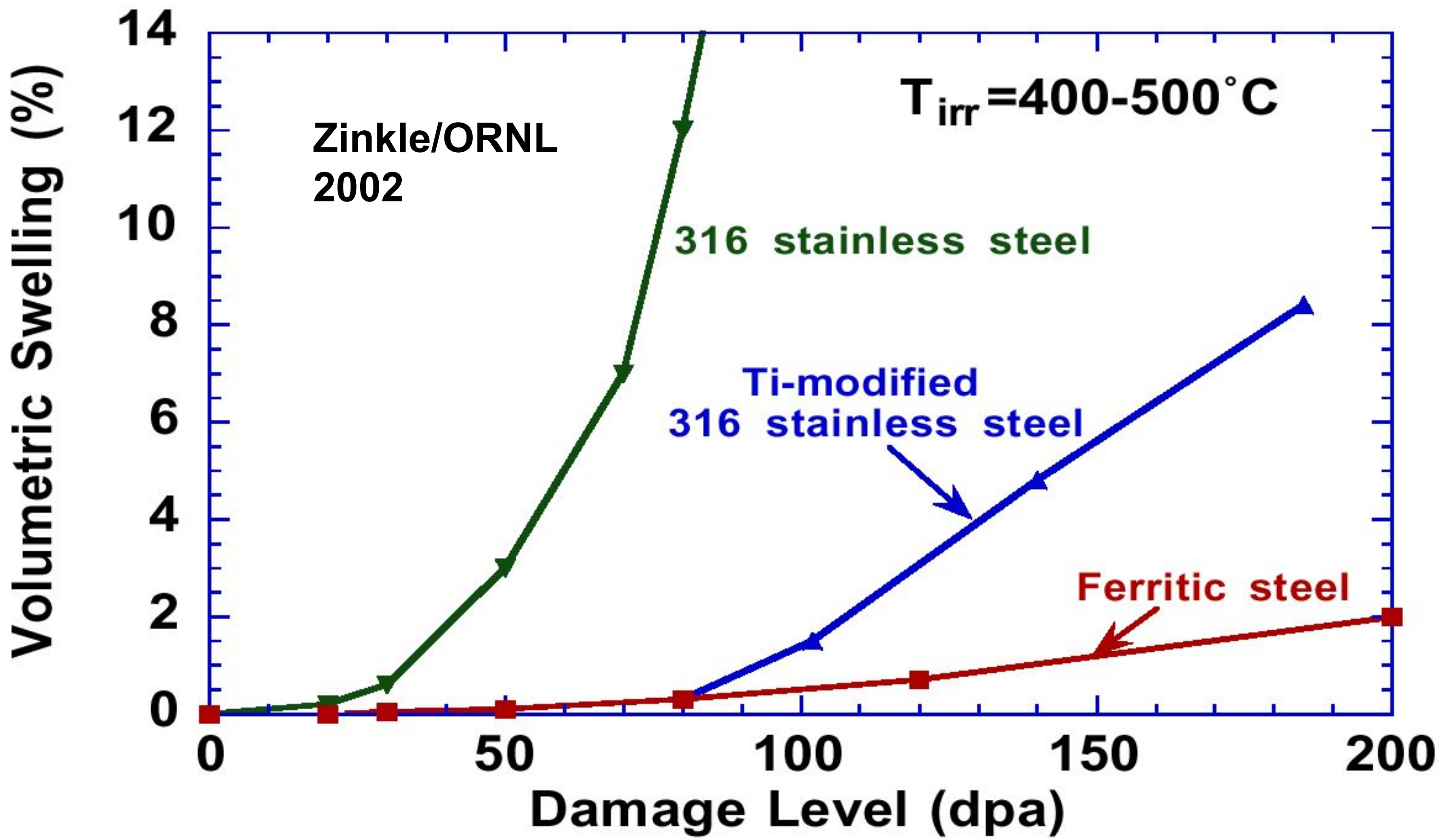
Structural materials beyond ITER (1)



Most metals swell in temperature range of 0.3 T_m < T < 0.55 T_m

Swelling of 316 steel will make it not applicable for a real fusion reactor.

Structural materials beyond ITER (2)



Lowest swelling is observed in body- centered cubic alloys (ferritic steel)

Structural materials beyond ITER (3)

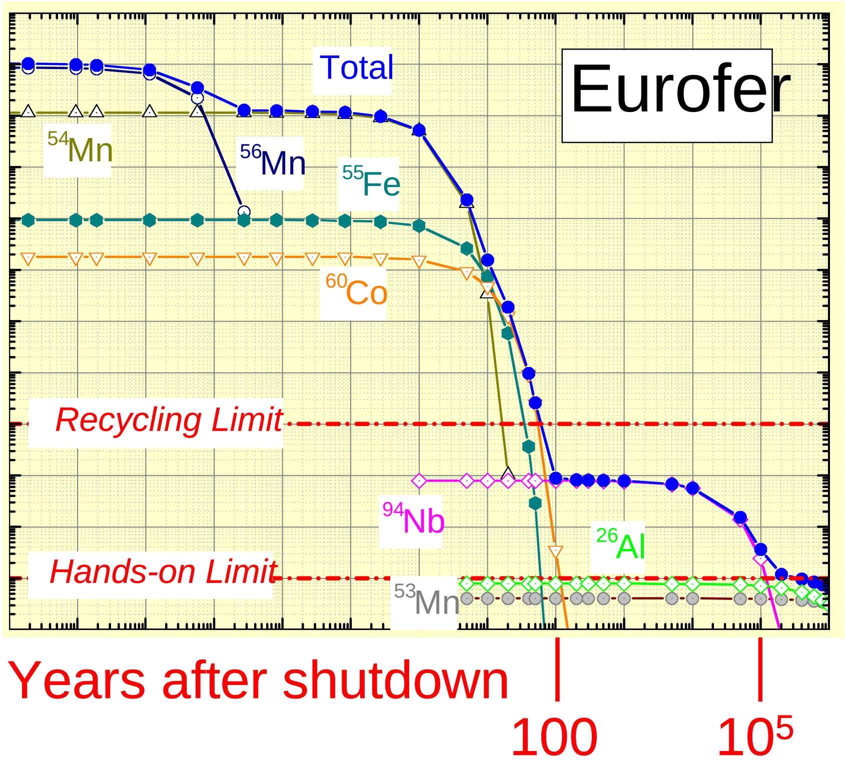
- **Reduced Activation Ferritic - Martensitic (RAFM) steels are considered as the most promising material for the first wall and blanket of future fusion reactor DEMO:**
  - **Good thermal conductivity**
  - **Reduced activation**
  - **Low swelling and radiation damage accumulation resistance**

- **Typical Aaloy composition: 9wt.%Cr, 1wt.%W, Mn, V, Ta, C<0.1 wt.%, (bal.: Fe)**

- **Elemental substitution for reduced activation:**
  - **Mo, Nb → W, V, Ta** ~~Ni~~

Activation of the reference Eurofer steel RAFM Reduced Actication Ferritic Martensitic

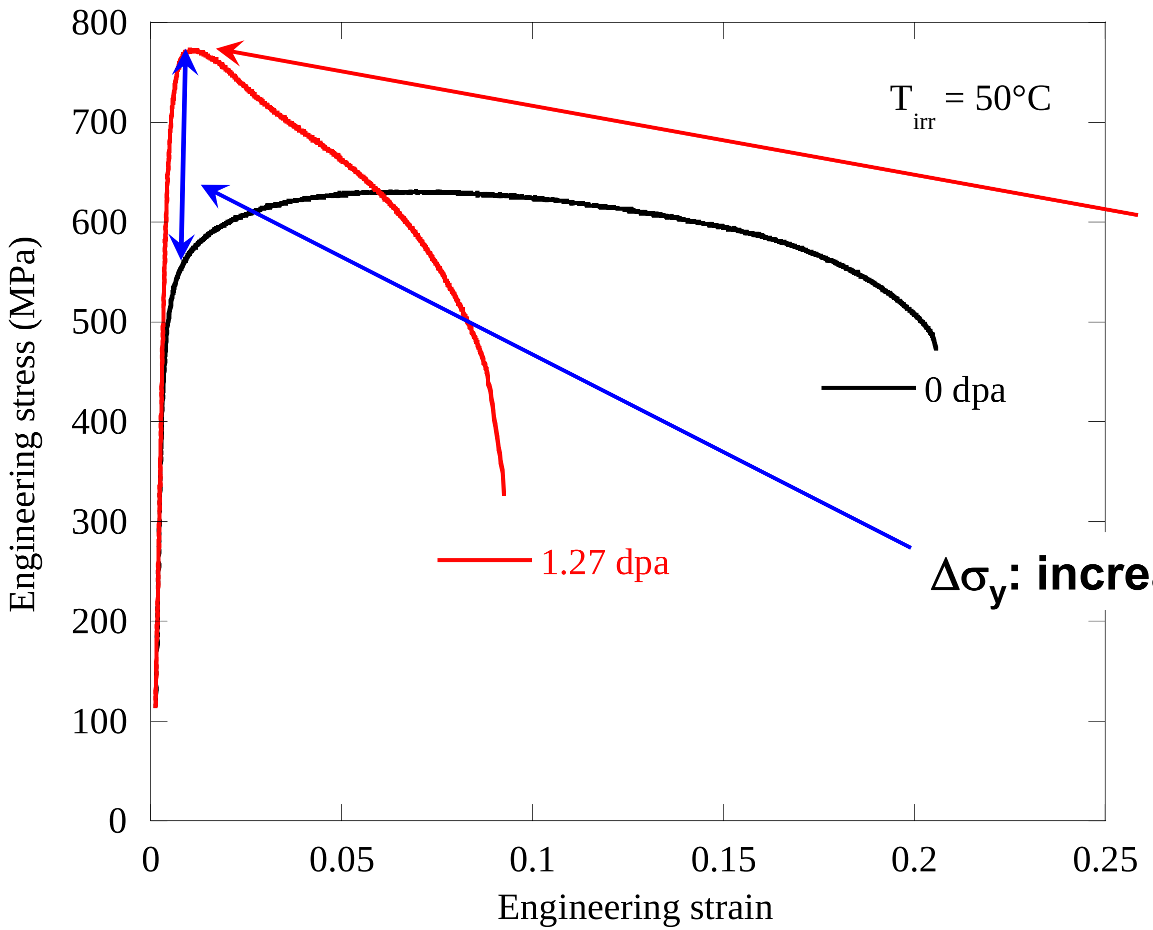
- Recycling dose rate level of 10 mSv/h is achieved after 50-100 years
- Hands-on dose rate level of 10 mSv/h is achieved after 10^5 years



- ***Assumptions:***

- **Fusion power: 3.3 GW**
- **Neutron flux: 1.53×10^{15} cm^{-2}.s^{-1}**
- **5 full power year irradiation**

Typical irradiation effect on tensile properties of high-Cr a tempered martensitic steel



Uniform elongation reduced to about zero plastic strain, even after low dose.

$\Delta\sigma_y$: increase of the yield stress

Eurofer97 steel, proton irradiated

Effects of irradiation on the plastic flow properties

Key issues:

- $\Delta\sigma_y$, $\sigma(\varepsilon_p, \dot{\varepsilon}_p, T)$ and ΔT_o depend :

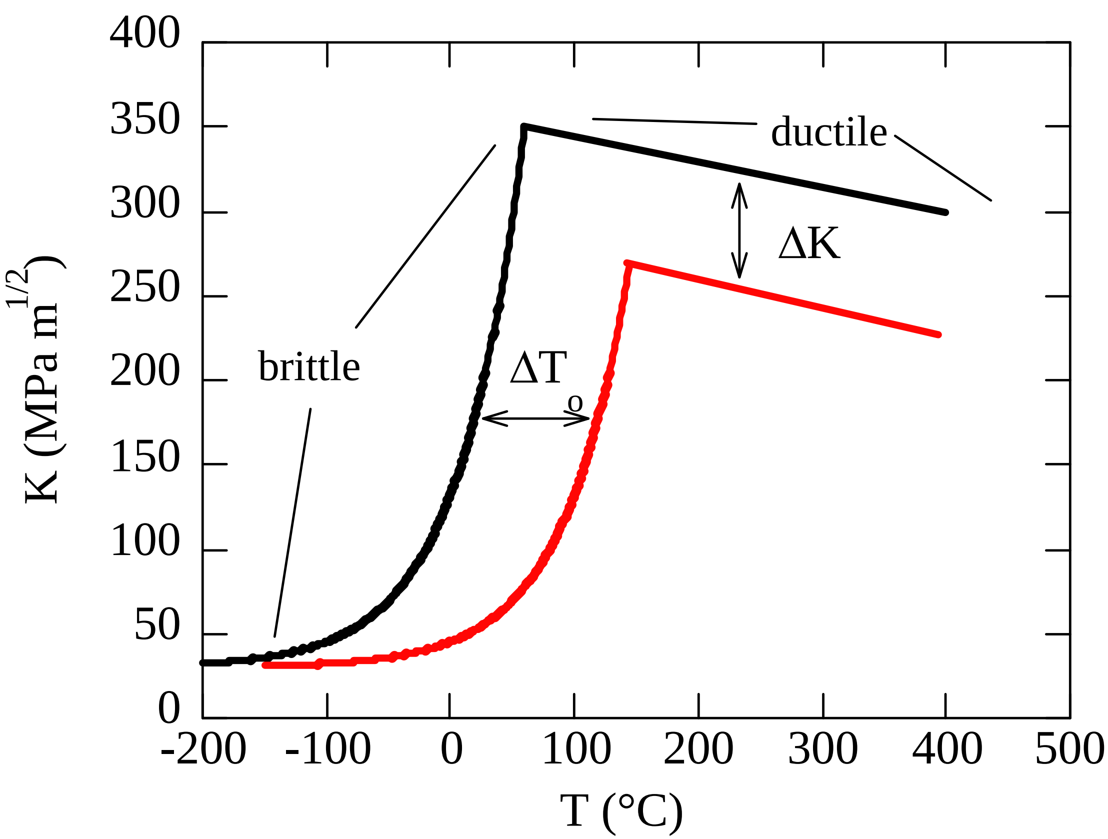
  neutron flux and spectrum,
  dose
  dose rate
  irradiation temperature
  chemical composition
  thermo-mechanical treatment
  final heat-treatment… ?

- How to quantify the separate and synergistic effects of all the variables?

- How to transfer high-flux data to low-flux data representative of service condition?

- How to extrapolate data into dpa, flux, T, regimes where no data are available?

Embrittlement of the "ferritic" steels following neutron irradiation at T < 450 °C



Like other bcc metals and alloys, "ferritic" steels exhibit a ductile-brittle transition of their fracture mode.

Two major effects of irradiation:

1) Shift of the brittle regime to higher T + ΔT_o.

2) Decrease of fracture toughness in the ductile regime ΔK.

Master-Curve approach to assess irradiation embrittlement

The K(T) curve has a constant shape. It is indexed by T_o on an absolute temperature scale at an reference toughness of 100 MPa $m^{1/2}$. T_o accounts for material variability.

$$K_{Jc\ mean}(T) = 30 + 70\exp(0.019(T - T_o))$$

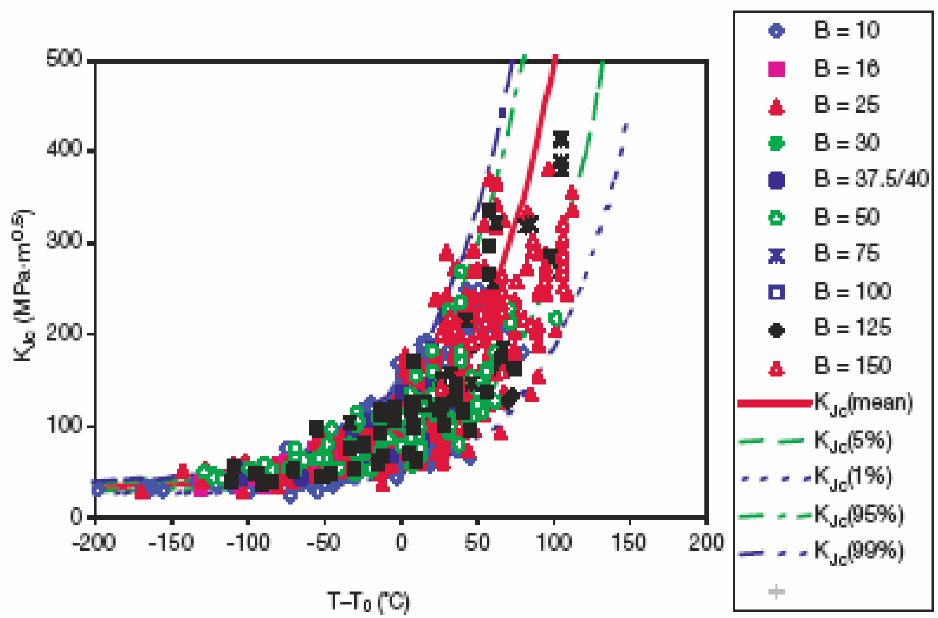
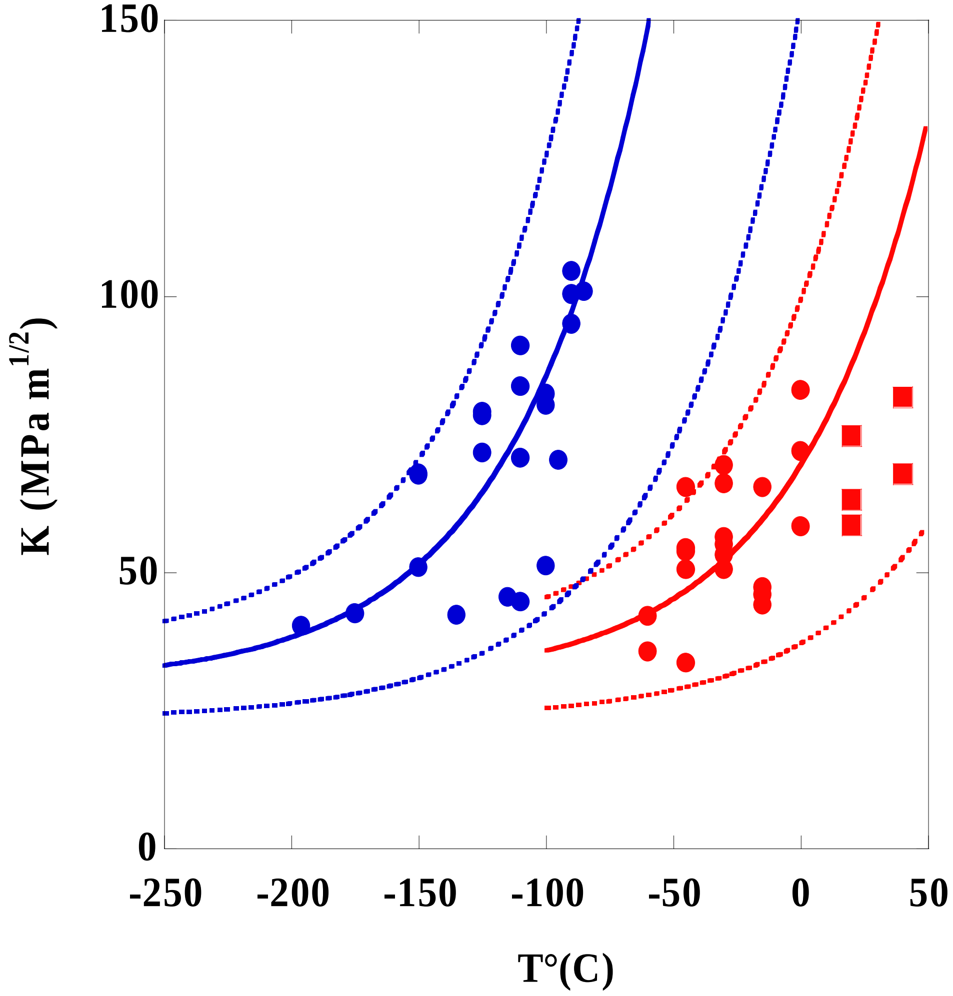


FIG. 17. Temperature dependence of static fracture toughness data for WWER-440 RPV materials.

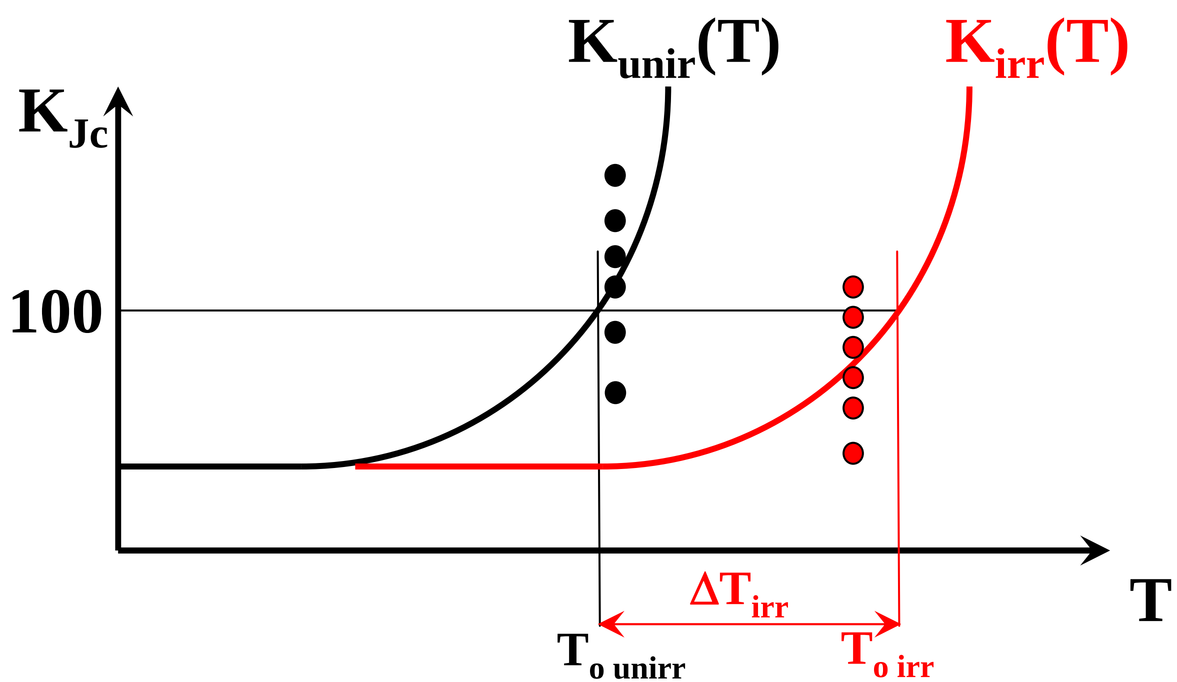
Shift of the K(T) curve after neutron irradiation at 60°C and 2.3 dpa



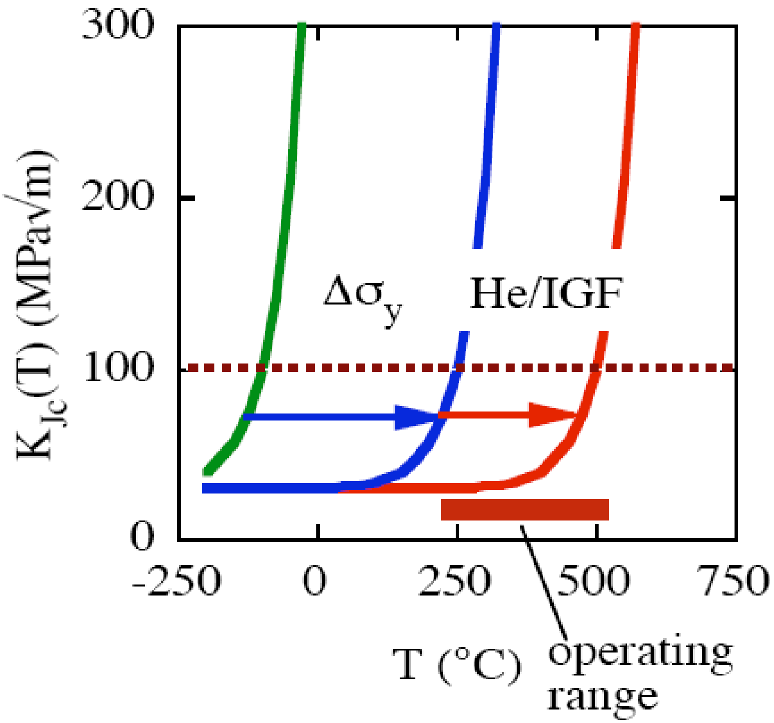
ΔT_o shift determination using the Master-Curve approach. ASTM-E1921 standard

The main advantage of Master Curve-Shift (MC-ΔT) method for fracture toughness is to determine T_o with few specimens (6).

K_{med}(T) has a constant shape before and after irradiation for all the "ferritic steels" and is indexed at an absolute temperature T_o for K = 100 MPa $m^{1/2}$.



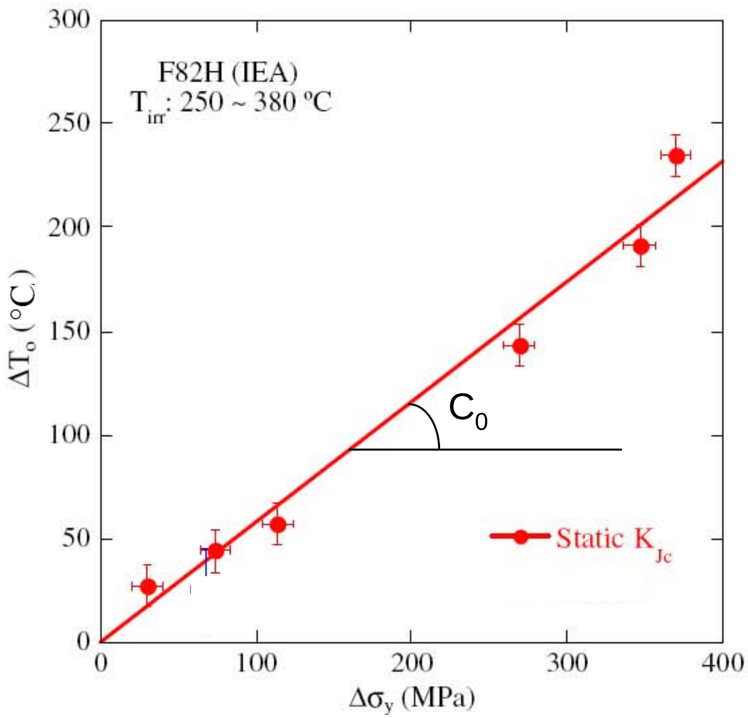
He effect on fracture at large concentrations
Non-hardening embrittlement

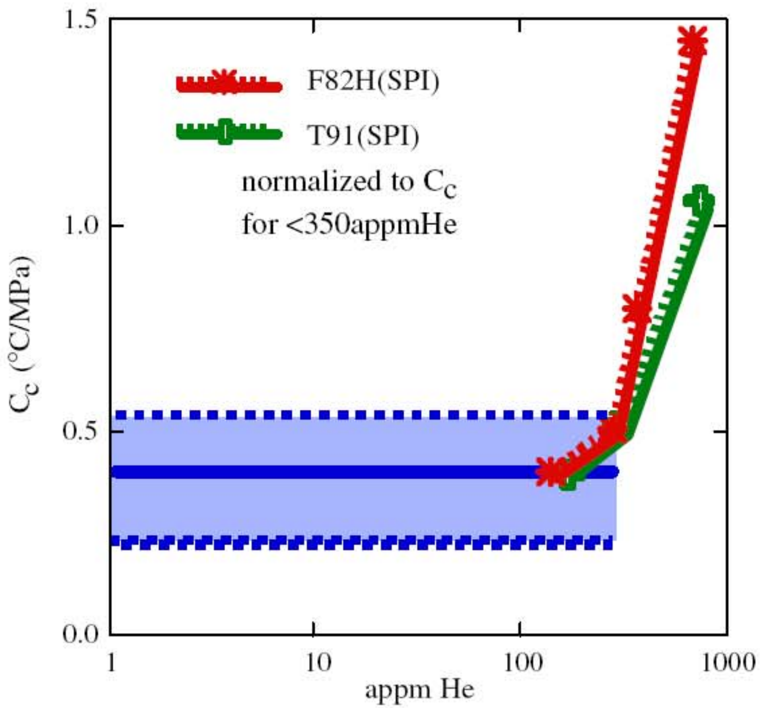


- **Irradiation experiments producing up to 400 appm He show no clear enhancement NHE.**

- **Spallation proton data suggests at > 600-800 appm He weakens grain boundaries producing very brittle intergranular facture that interacts synergistically with $\Delta\sigma_y$.**

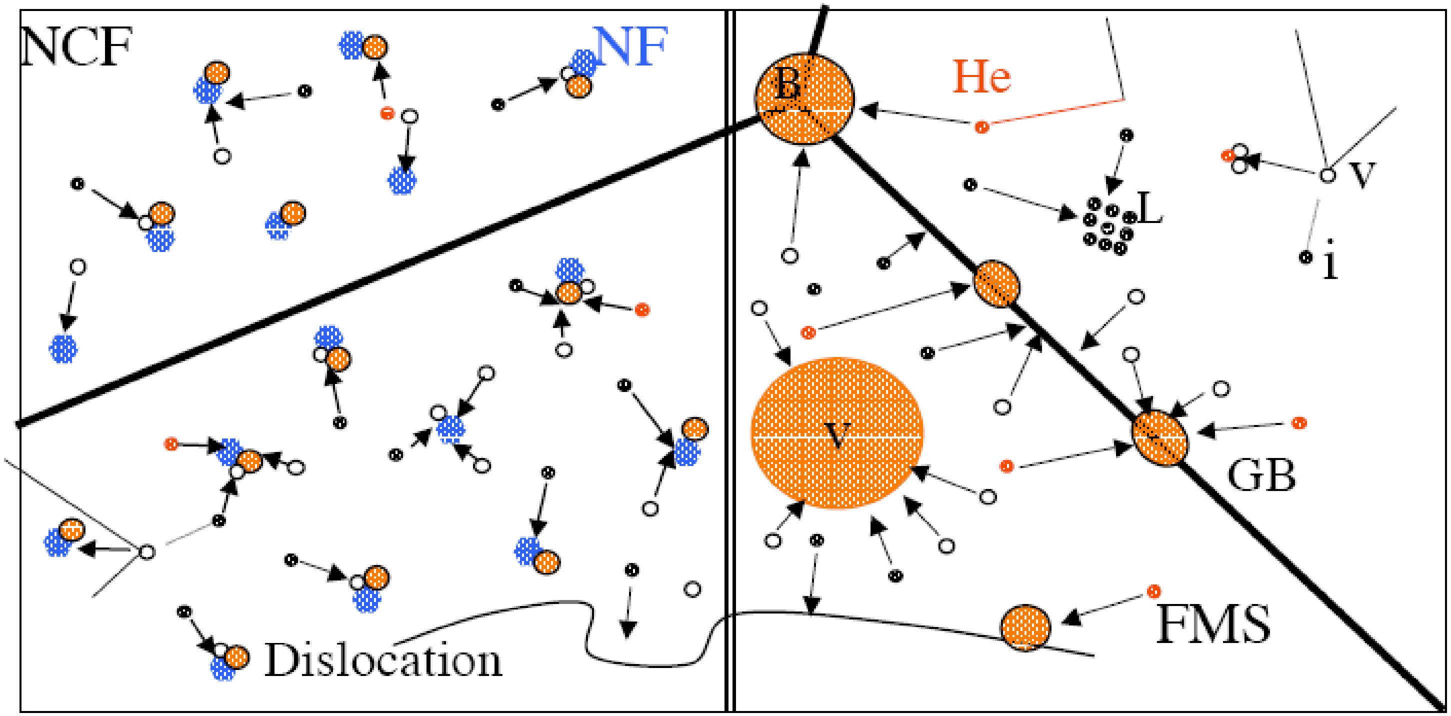
How serious and limiting the He effects can be?





"…If verified, such large ΔT_i would effectively eliminate low activation martensitic steels as candidate alloys for post-ITER applications to fusion power demonstration reactors, at least for a range of lower service temperatures…" G. Lucas et al. JNM (2007)

He management
Nano-composite ferritic vs ferritic steel



- Basic He & He/dpa mechanisms diffusion & clustering kinetics with dpa
- Trapping@dislocations/interfaces/boundaries
- Sink effects and partitioning, bubble sizes and stability, reaction to stress

Current Irradiation Facilities

- **There is no intensive 14 MeV neutron irradiation facility available.**

- **It is necessary to simulate irradiation by 14 MeV neutrons, by using either fission neutrons, or high energy protons, or heavy ions.**

Two Ways

- **How to account for actual irradiation conditions: fusion-relevant neutron spectrum, temperatures, accumulated damages (dpa), damage rates (dpa/s), production rates of impurities (e.g. appm He/dpa, appm H/dpa) ?**

Modeling **of radiation damage and radiation damage effects**	***Experiment-irradiation*** **with the International Fusion Materials Irradiation Facility (IFMIF)**

Numerical and Analytical Tools

- Radiation damage is an inherently multiscale phenomenon

- Ab initio / Density Functional Theory (DFT) calculations
- Molecular Dynamics (MD) simulations
- Monte Carlo (MC) simulations
- Kinetic Rate Theory
- 3D Discrete Dislocation Dynamics (DDD) simulations
- Transmission Electron Microscopy (TEM) image simulations
- Finite Element Modelling (FEM) calculations

IFMIF
International Fusion Materials Facility

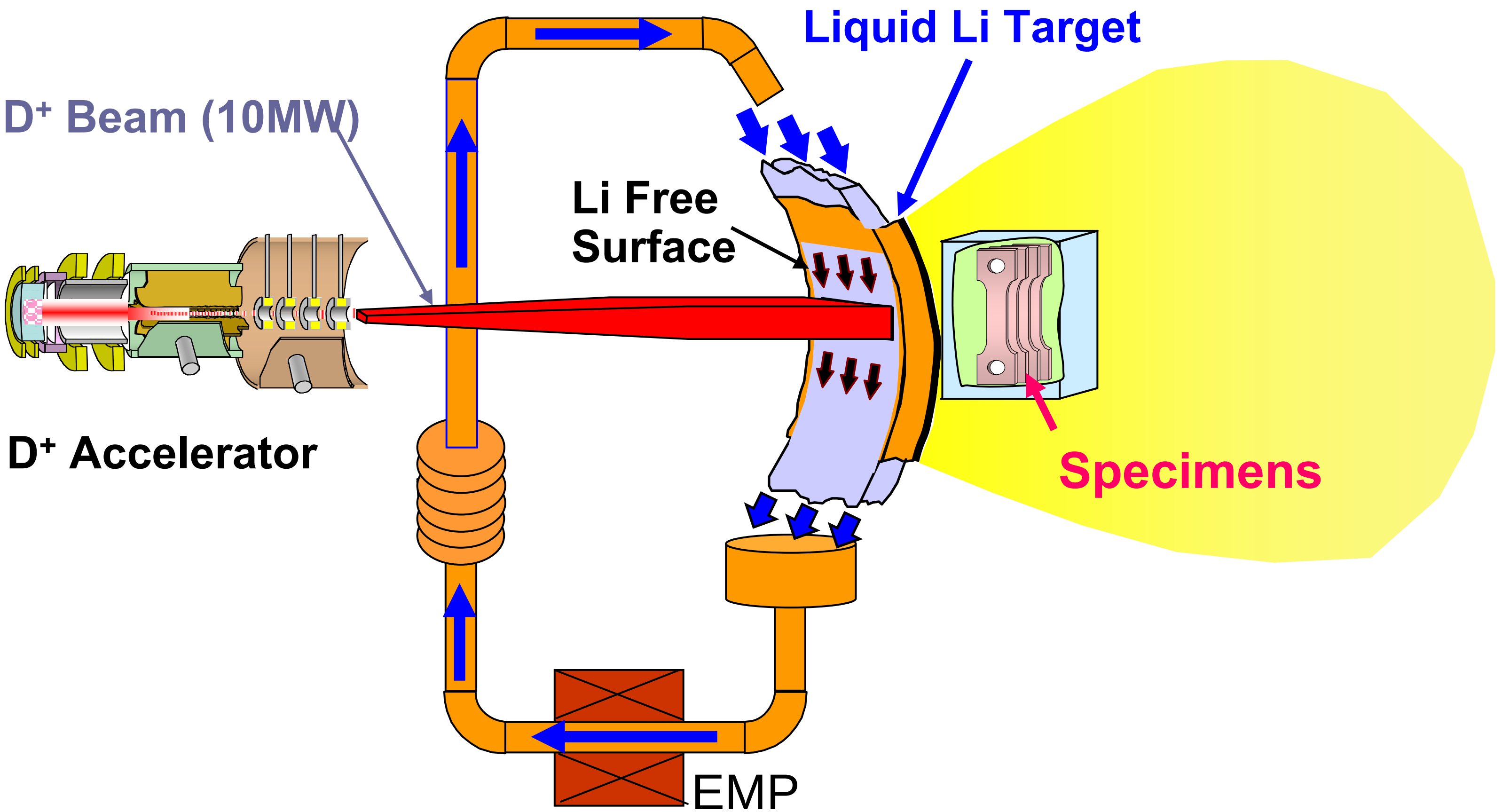
- **Intense source of high energetic neutrons (2x125 mA):** The corresponding neutron spectrum should create the same PKA spectrum and H/dpa, He/ dpa ratios as in the materials fusion power reactor.

Missions:

- Calibrate and validate the data generated using fission reactors and accelerator-based facilities (ion sources, spallation neutron sources).
- Qualify materials up to about full lifetime of anticipated use in a DEMO-type reactor.

Schematic View

H. Matsui, A. Möslang
SOFT Conference, 2004



Small Specimen Test Technology

H. Matsui, A. Möslang
SOFT Conference, 2004

- Using miniaturized specimens:
  - 80 dpa database for a few materials within five years for DEMO-predesign
  - 150 dpa database for a variety of materials within about 20 years

1 cm

Specimen type	Present geometry	Comments
Tensile		developed
Fatigue		developed
Charpy		Standard R&D ongoing
Creep		Verification
Crack growth		International R&D ongoing
Fracture toughness		International R&D ongoing

Missions of the FTM Group

- The FTM group has FIVE main missions, focused on metallic materials.
- Development of advanced metallic materials for fusion power reactors.
- Characterization of advanced metallic materials for fusion power reactors.
- Modelling of radiation damage and effects.
- Fracture mechanics and small specimen test technology.
- Qualification of metallic materials for ITER.

Mission 1

- Development of advanced metallic materials for fusion power reactors

- **Materials:**
  - ODS RAF steels
    - Fe-(12-14)Cr-2W-(0.1-0.3-0.5)Ti-0.3Y_2O_3
  - RA W-base materials
    - W-(0.3-1.0-2.0)Y, W-(0.3-1.0-2.0)Y_2O_3, W-(0.3-0.9-1.7)TiC

- **Manufacturing routes: powder metallurgy techniques**
  - Mechanical alloying
  - Hot isostatic pressing
  - Thermal treatments
  - Thermal-mechanical treatments

Mission 2

- Characterization of advanced metallic materials for fusion power reactors

- <u>Objective:</u> Characterization, before and after irradiation, of the relationships between the microstructure and the mechanical properties.

  - Advanced metallic materials developed by the FTM group
    - ODS RAF steels
    - RA W-base materials
  - Advanced metallic materials developed within the European Fusion Development Agreement (EFDA) programme
    - EUROFER RAFM steel
    - ODS EUROFER

Mission 3

- **Modelling of radiation damage and effects**

- **Calculations of the energetics of structural defects**
- **Creation of specimens containing a few million atoms and defects**
- **Simulations of TEM images of defects**
- **Simulations of the interaction of mobile dislocations with defects**
- **Simulations of the plasticity of materials (emphasis on He effects)**

- **Calculations/simulations are being focused on Fe-base materials (Fe, Fe-C, Fe-Cr, Fe-He, Fe-Cr-He, Fe-Cr-C), towards RAFM steels, and on W**

Mission 4

- **Fracture mechanics and small specimen test technology**

- **<u>Objective:</u> Model the temperature dependence of the fracture toughness of RAFM steels, in order to determine their lower temperature for use, in the unirradiated and irradiated conditions**

  - **Production of a large fracture toughness database and analysis of the results using the Master-Curve approach**
  - **Investigation of the effects of size and geometry on the mechanical properties, in order to extract standard parameters from testing of subsized specimens as irradiated in available facilities and in IFMIF**

Mission 5

- Qualification of metallic materials for ITER
- Objective: Qualify commercial metallic materials for the environmental conditions (temperature, irradiation and thermal-mechanical stress conditions) that will be encountered in ITER
  - Effects of irradiation and hydrogen content on the tensile and fracture behaviour of Ti-base alloys (e.g. Ti-6Al-4V, Ti-5Al-2.5V)
  - Creep, fatigue and creep-fatigue behaviours of the Cu-Cr-Zr alloy, in the unirradiated and irradiated states

Scientific Approach of the FTM Group

- **Investigating the production and accumulation of (radiation) damage as a function of irradiation conditions, thermal-mechanical stresses, temperatures, etc., at different length scales (micro-, meso-, and macroscopical) in relation to the (irradiation-induced) degradation of the mechanical properties**
- **A wide range of experimental and numerical tools is used to reach this objective**

Irradiation Facilities

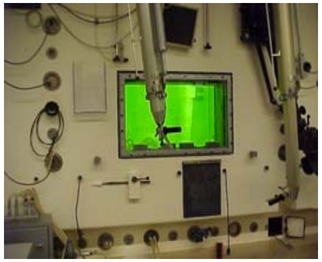
- The FTM group uses a mixed spectrum of high energy (≤ 570 MeV) protons and spallation neutrons, i.e., the target of the Swiss Spallation Neutron Source (SINQ), to simulate experimentally the effects of 14 MeV neutron irradiations
- The FTM group is also involved in specific fission neutron irradiations performed in reactors in the Netherlands, Hungary and Russia

Main Experimental Tools

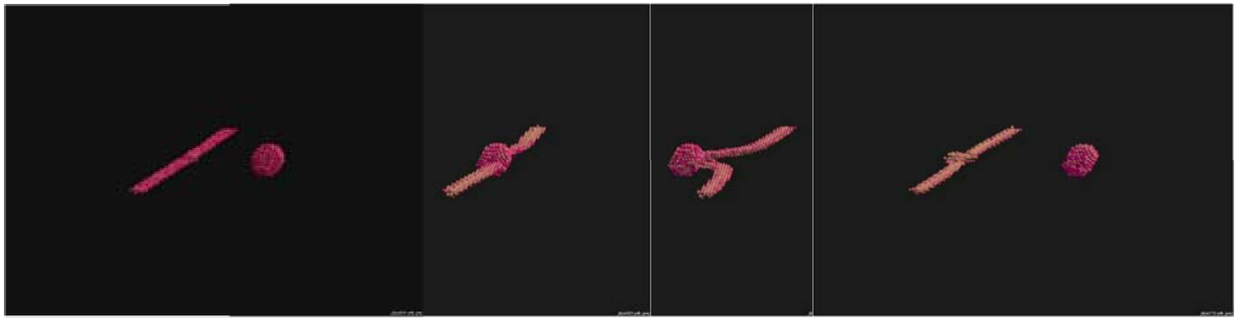



- **Manufacturing devices: ball mills, furnaces**
- **Deformation machines: tensile tests, Charpy impact tests, 3-point bend tests, creep tests, fatigue tests, small punch tests**
- **Nano-indenter**
- **Transmission electron microscope**
- **Focused ion beam**
- **Corrosion devices (flowing Pb-Li)**
- **Hot cells for mechanical testing of strongly radioactive specimens**

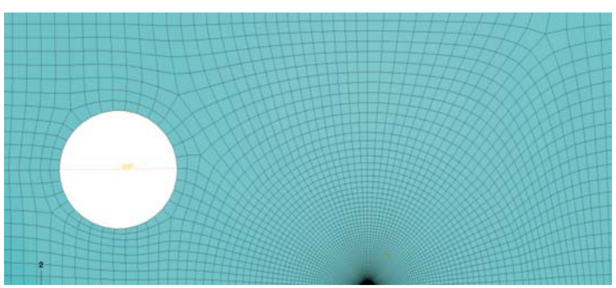


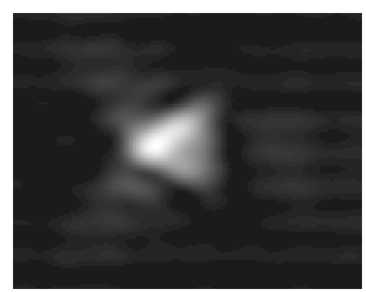


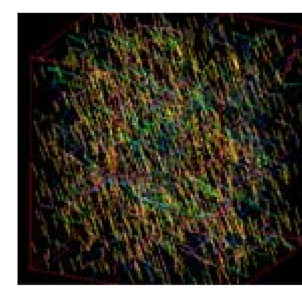

Main Numerical Tools



- Ab initio calculations
- Molecular dynamics (MD) simulations
- 3D discrete dislocation dynamics (DDD) simulations
- Transmission electron microscopy (TEM) image simulations
- Finite element modelling (FEM)
- <u>Equipment:</u> PCs Windows and Mac, and Unix workstations







Fusion will be or will not be…

it is a question of materials!