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The first wall of fusion reactors: a challenge for material research

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Overview

Introduction: the material challenge

Neutron damaged materials

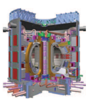
Research tailored towards fusion reactor materials

- Extrinsic toughening: composite materials
- Intrinsic safety: smart alloys

Summary and outlook

From ITER to DEMO

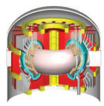
ITER



- operational flexibility (experimental device)
- transient heat flux events
- T-codeposition on „cold" surfaces
- no energy conversion (80°C water coolant)
- low duty cycle
- low neutron dose (wall: ~1 dpa)

Need to apply available materials and technology

DEMO

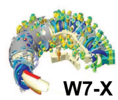


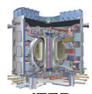
- lifetime (erosion, ageing)
- very limited transient heat flux events
- energy conversion (coolant: ≥300°C water, ≥400°C He)
- high duty cycle
- high neutron dose (wall: 80…150 dpa)
- low activation materials

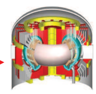
Need for innovation in non-activating materials and technology

PMI and material aspects in fusion research

current fusion experiments →







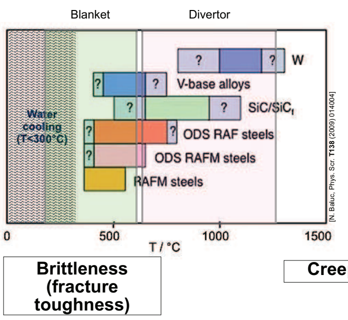
Wall materials
- tungsten as first wall
- C+Be+W material mix
- chemical erosion
- impurities and transport
- hydrogen inventory

Stationary operation
- actively cooled wall components
- bonding technologies (W, C / Cu, steel)

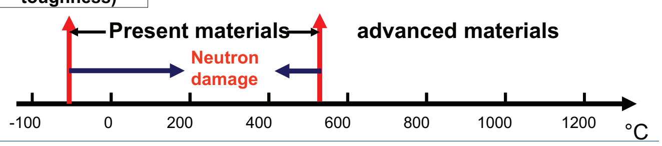
Tritium inventory
- permeation barrier layers
- functional coatings

Reactor operation
- high power heat sinks
- neutronen damage
- operational safety

New material approaches needed



- Conventional materials are limited mostly by their parameters in addition to engineering approaches
- New materials should consider limits and mechanisms to improve on limits (composites, FGM) - think component!



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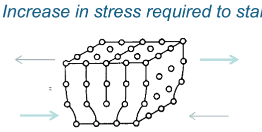
Effects of neutron irradiation

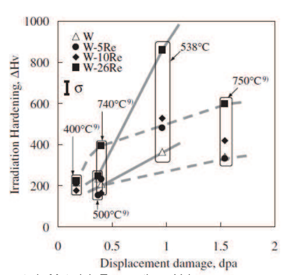
Defect production and migration Transmutation, He production	Hardening and deformations Fracture and embrittlement Creep, swelling surface modifications Precipitation and segregation He embrittlement	Crack formation/ enhanced erosion, melting Brittle destruction / Dust formation Fuel retention	Reduced life time of plasma facing components Issues for operational safety
Radiation damage event *Damage cascade* *Defect formation and diffusion* *Void and bubble formation*	*Physical and mechanical effects of radiation damage*	*Enhancement of PWI processes / material damage under heat and plasma loads*	*Relevance for nuclear fusion reactors*

Irradiation hardening

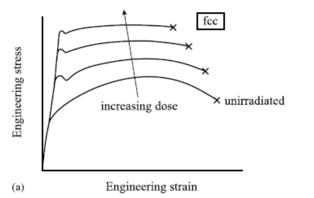
Increase in stress required to start a dislocation moving on its glide plane



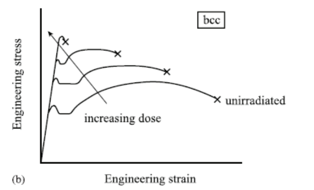
Gary S. Was, *Fundamentals of Radiation Material Science*, p. 268



Tanno et al., Materials Transactions, Vol. 52, No. 7 (2011) pp. 1447 to 1451



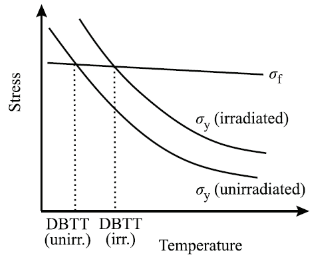
Temperature dependent!



Gary S. Was, *Fundamentals of Radiation Material Science*, p. 582

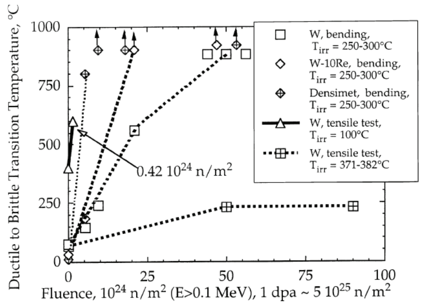
Embrittlement

- Ductile-to-brittle transition temperature (DBTT): yield stress σ_y equals fracture strength σ_f
- Irradiation causes DBTT to increase: different sensitivities of yield stress and fracture strength to neutron damage



Gary S. Was, *Fundamentals of Radiation Material Science*, p. 661

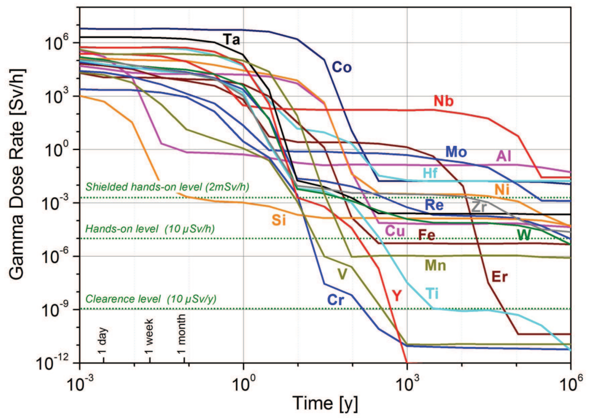
Embrittlement of tungsten



- Severe reduction of operational temperatures
- For PFMs: enhanced probability of crack formation

V. Barabash et al. / Journal of Nuclear Materials 283-287 (2000) 138-146

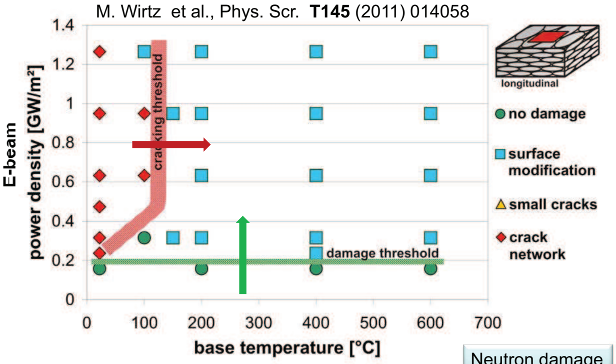
Transmutation: Decay times after n-irradiation



Source: R.A. Forrest et al., Handbook of Activation Data, 2009

Impact of synergistic loads on crack formation

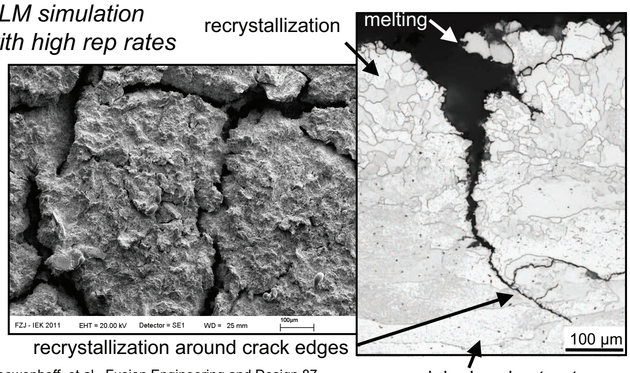
M. Wirtz et al., Phys. Scr. **T145** (2011) 014058



Neutron damage
Plasma load

Enhanced erosion of irradiated tungsten in synergistic loading conditions

ELM simulation with high rep rates



Th. Loewenhoff, et al., Fusion Engineering and Design 87 (2012) 1201-1205

Change of material composition

Radiation (plasma) induced:

- RIS: spatial redistribution of solute and impurity elements in a metal → enrichment or depletion of alloying elements near surfaces
- Reason: different coupling of solutes to defects
- Phase instabilities
- Issue for functional surface coatings (e.g. passivating layers or permeation barriers)
- Impact on surface composition as determined by preferential sputtering (e.g. EUROFER)
- *Cr enriches in F-M alloys, leading to grain boundary embrittlement [Gupta et al., J. Nucl. Mater. 351 (1-3) (2006), 162.]*

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Summary

Research tailored towards fusion reactors

Research focus:

- material development
  - definition
  - preparation and characterization
  - optimization
- PWI issues
  - erosion
  - retention
  - lifetime

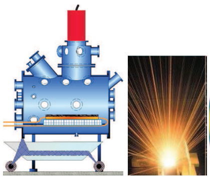
Materials tests:

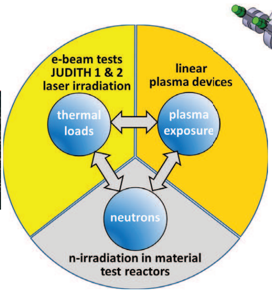
- neutron damage (simulation AND “real” neutrons)
- plasma exposure
- ELM (off-normal events) simulation

Material testing at FZJ

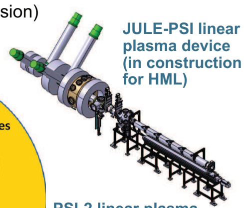
- Integrated characterization of thermo-mechanical and physical-chemical properties of neutron irradiated and toxic plasma-facing materials under high heat loads and plasma exposure
- Selection of plasma-facing materials tested under n-irradiation and optimized for PMI processes (tritium retention, embrittlement, erosion)

JUDITH 1 and JUDITH 2 e-beam facilities

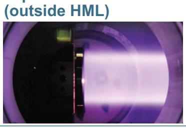




JULE-PSI linear plasma device (in construction for HML)



PSI-2 linear plasma experiment (outside HML)



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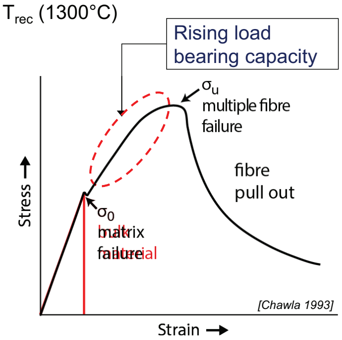
Tungsten: Brittleness problem

Limitations of operation temperatures for tungsten:

Lower limit: ductile-brittle-transition temp. T_{DBT} (260-650°C)
Upper limit: recrystallization temp. T_{rec} (1300°C)
plus: neutron embrittlement

- scattering in strength (small Weibull modulus)
- no damage tolerance
- uncertainty in lifetime prediction

Solution: extrinsic toughening (ductilization) mechanisms



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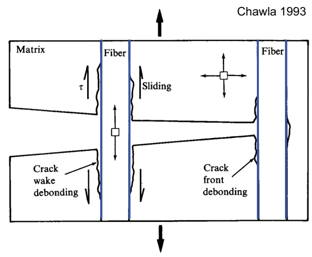
⇒ local energy dissipation
- crack bridging
- fiber oull-out
- crack deflection



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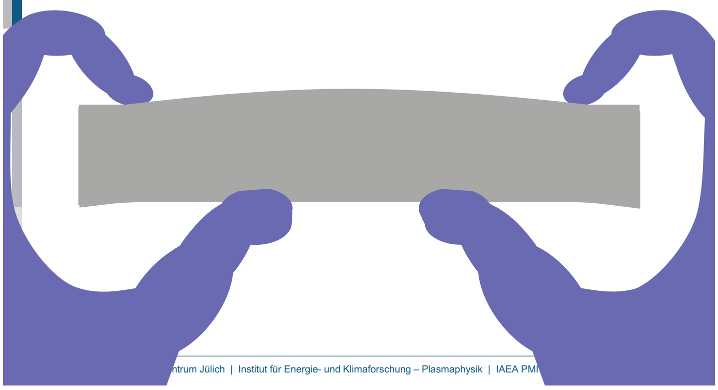
Main advantages for fusion

- damage tolerance
- mechanical effect

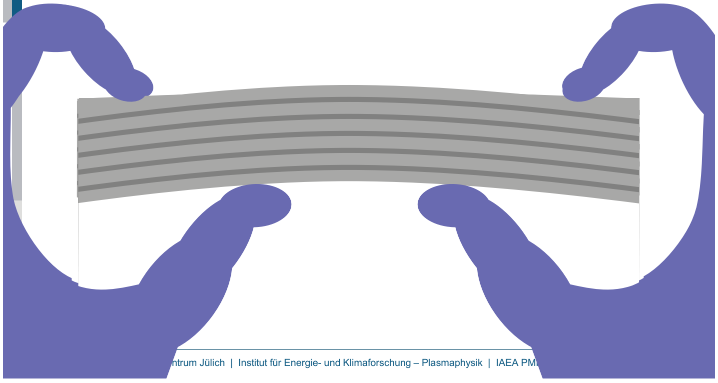
⇨ **less susceptible to operational embrittlement**

e.g. full tungsten tile under cyclic loading

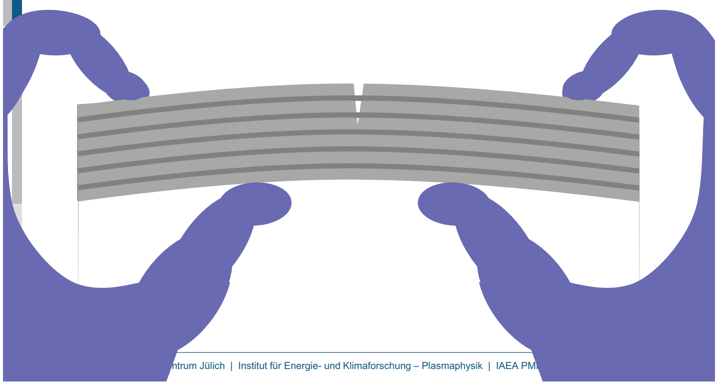


catastrophic failure by brittle fracture
after a random number of cycles
or caused by overload

W_f/W under cyclic loading

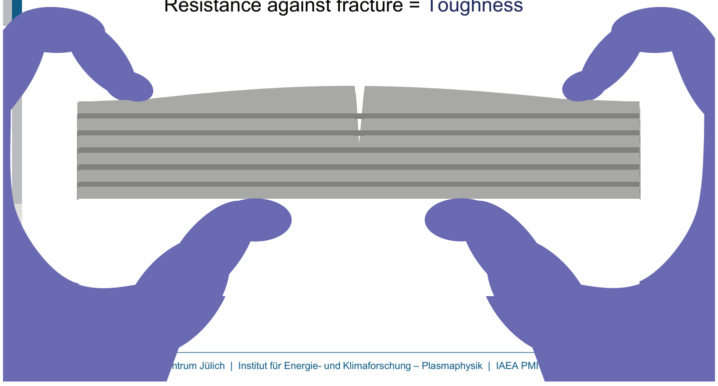


Crack is bridged by fibres



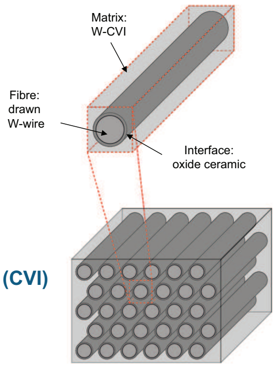
Further loading still possible

Resistance against fracture = Toughness

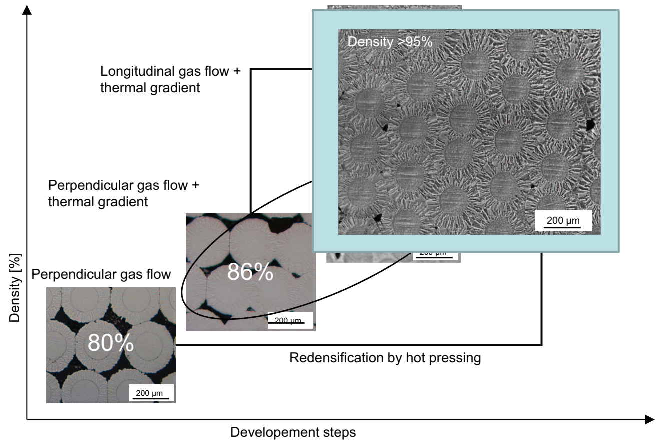


Architecture of W_f/W

- **Fibre**
  Drawn tungsten wire (d = 150 µm): high strength + some ductility
- **Interface**
  PVD coating: Optimised adhesion + stability
- **Matrix**
  Interface integrity + high density
  Develop **chemical vapour infiltration (CVI)** technique for W_f/W
  - No mechanical impact
  - Low process temperature

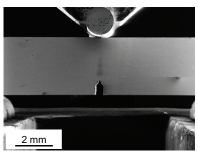


Development of CVI-tungsten



3-Point bending test (ESI Leoben)

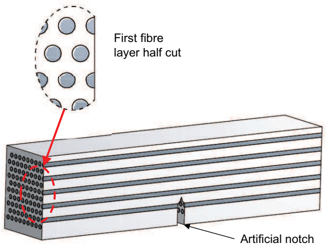
Stepwise 3-point bending



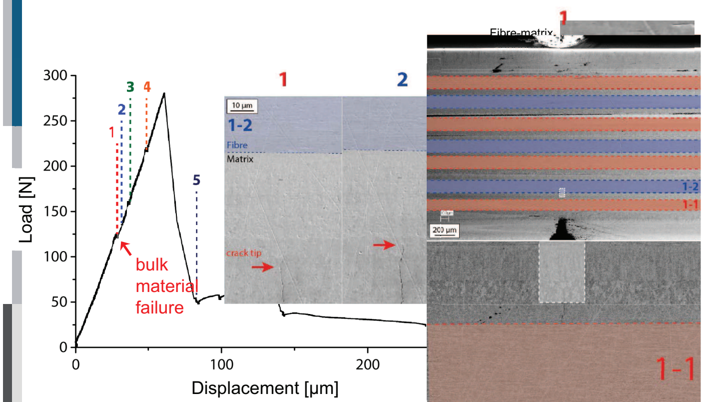
+

In-situ surface observation in electron microscope

Multi-fibre composite
→ W-CVI
→ 10 layers x 9 fibres
→ 2.2 mm x 3 mm



Stable crack propagation

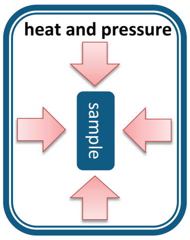


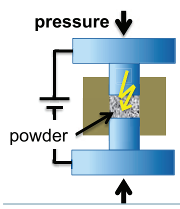
Controlled crack propagation + **risin**…

Alternative production routes

HIP – Hot isostatic pressing

- capsule filled with tungsten powder and fiber inside a pressure vessel
- powder compaction due to high pressure and temperature
- T_{max} 2000 °C
- p_{max} 350 MPa (via Ar)

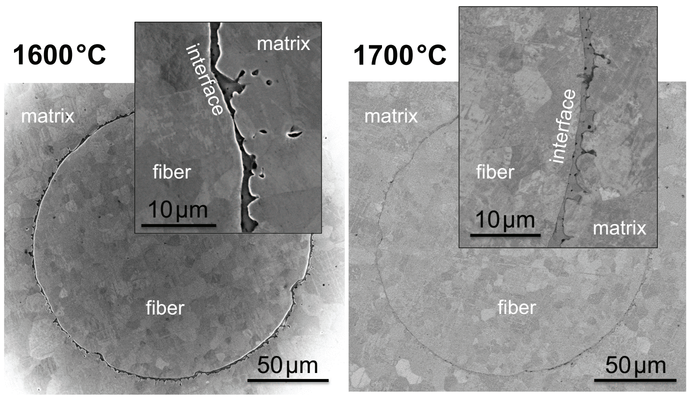




EDS – Electro discharge sintering

- tungsten powder and fiber are put inside an extrusion die
- powder compaction due to ohmic heating by a high current (500 kA) + uniaxial pressure (p_{max} 350 MPa)
- process time < 1 s, E_{max}=80 kJ

HIP W_f/W composites



- intact interface after HIPing
- dense matrix achievable

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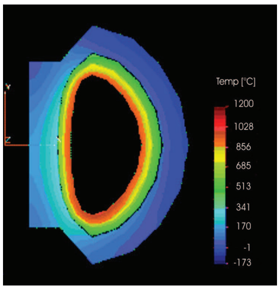
Accidental loss of coolant in reactor

Power plant conceptual study



Temperature profile in PPCS Model A, 10 days after accident with a total loss of all coolant.

[Final Report of the European Fusion Power Plant Conceptual Study, 2004]

- Accidental loss of coolant: peak temperatures of first wall up to 1200 °C due to nuclear afterheat
- Additional air ingress: formation of highly volatile WO_3 (Re, Os)
- Evaporation rate: order of 10 -100 kg/h at >1000°C in a reactor (1000 m^2 surface)
  → large fraction of radioactive WO_3 may leave hot vessel

⇨ Development of self-passivating tungsten alloys

Requirements for "smart" W alloys

Normal operation conditions

- W-dominated plasma-wall interactions
- Limited and controlled H isotope retention

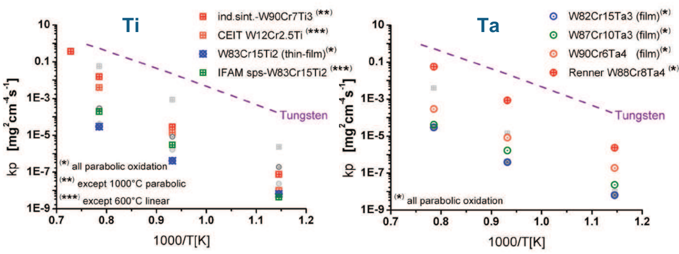
After LOCA event (loss of coolant accident)

- Strong reduction of oxidation rate
- Stable protective layer

General

- Large-scale bulk material production routes
- No formation of brittle phases

Si-free alloys: W-Cr- (Ti / Ta / Y)



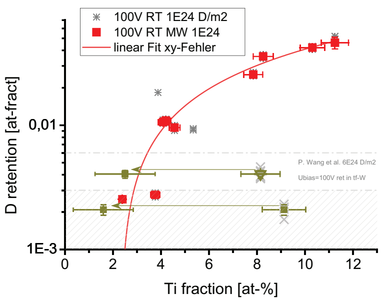
Reduction of oxidation rates

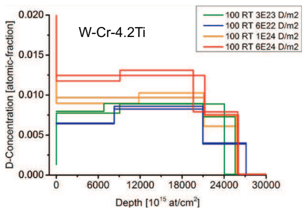
- Model thin films: several orders of magnitude
- Bulk materials: less reduction, different mechanisms?

Composition

- Both Ti and Ta alloys successful
- Maximise W fraction: **W-Cr6-Y0.04**: 82 at% W Oxdiation rate <5x10^{-6} $mg^2cm^{-4}s^{-1}$

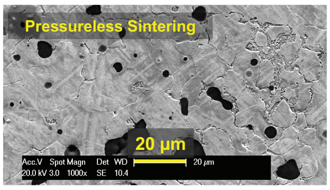
D retention after plasma exposure



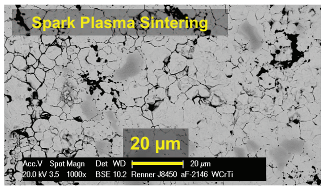


- correlation between Ti concentration and D retention
- comparable to PVD tungsten for low Ti fraction
- bulk material: similar after Ti correction for oxide fraction

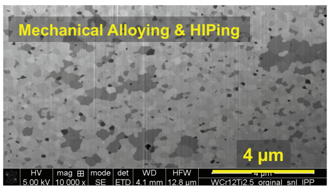
W alloy bulk production methods



T = 2000 °C, t = 30 min



T = 1474 °C, P = 64 MPa, t = 10 min



T = 1300 °C, P = 150 MPa, t = 60 min

MA + HIP:	submicron grains, Ti inclusions
PS:	grain size ~ 20 µm, TiO_x inclusions
SPS:	grain size ~ 10 µm, „cracked" Ti inclusions
all:	Cr-rich/Cr-poor grains

Summary

PMI and materials for DEMO / a fusion reactor

- Combination of neutron / thermo-mechanical / particle loading
- No operational window for available materials
- Development and testing for new material (composites) required

W fiber / W matrix composites

- Development of W-CVI and powder metallurgical routes
- Verification of toughening effect: Stable crack propagation + rising load bearing capacity: damage tolerance
- Active toughening mechanism for fully brittle samples: resistance against embrittlement

Self-passivating W alloys

- Up to 1/1000 reduction of oxidation rates for ternary alloys
- Transfer from thin films to bulk material successful
- PWI processes: D retention quantified

Outlook

Multi-component materials

- Combination of materials solutions: brittle alloys with composites
- Hydrogen isotope inventory (PWI processes):
  - Dynamic evolution of composition during operation
  - Composites and alloys: new transport/trapping channels for T
  - n-induced damage: increased T retention?

DEMO: open issues, new ideas

- Steel first wall / breeding blanket: T permeation barriers required
- Thermomechanical properties after 14 MeV neutron irradiation?
- Neutron damage: large T inventory, erosion behavior?
- Transmutations: intrinsic formation of alloys (ref. to all above!)