Karlsruhe Institute of Technology

Aix-en-Provence
France
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Material Constraints due to neutron irradiation and degradation

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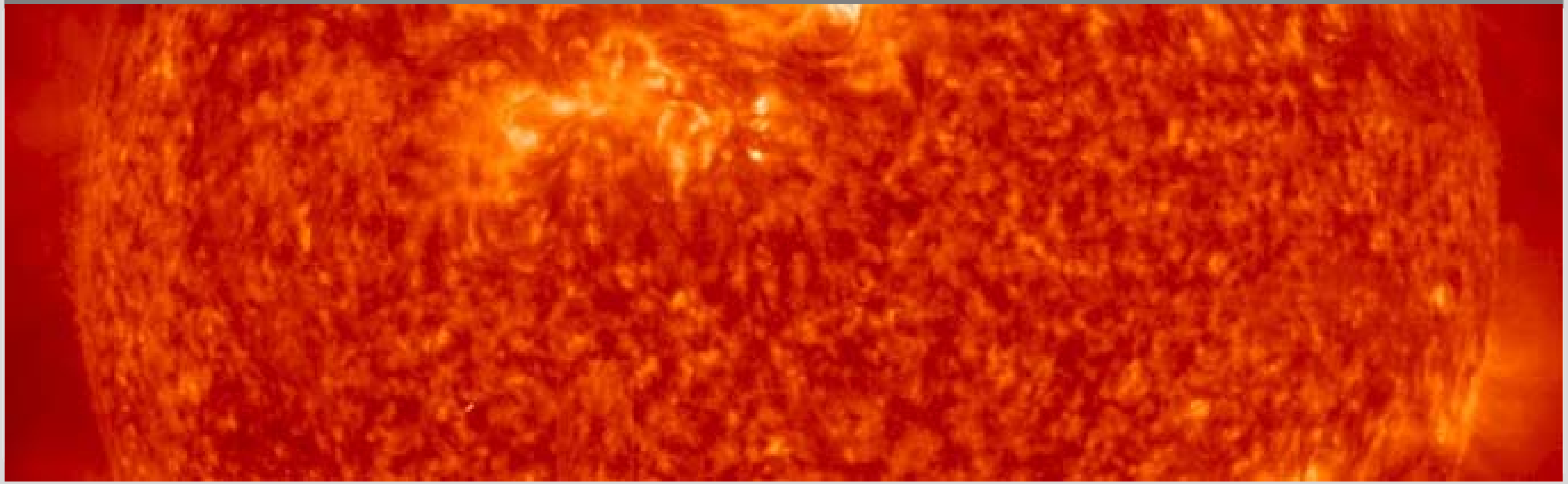
Institute for Applied Materials

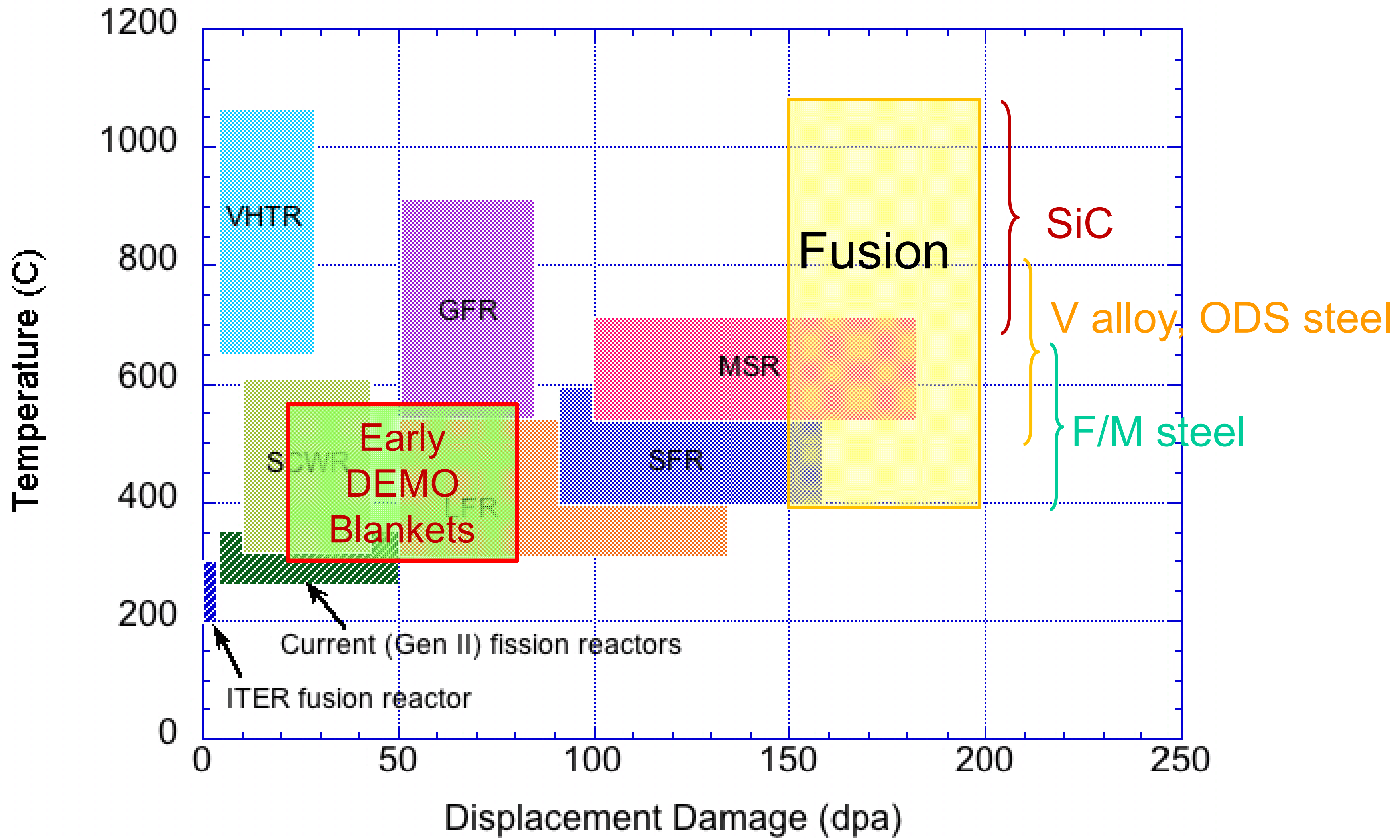


Outline

- Materials challenges: fusion – fission – spallation
- Neutron irradiation: Examples of progress and issues
  - Reduced activation ferritic/martensitic steels
  - Oxide dispersion strengthened steels
  - W alloys: I-1, I-3, O-1, O-2: Yesterday
    I-7, O-5, O-6, I-8: Today
- Technical Readiness and database maturity
- Role of materials in fusion roadmaps,
  Need for fusion specific irradiation source

Gen IV and Fusion reactors pose severe materials challenges

S.J. Zinkle & J.T. Busby, Mater. Today 12 (2009) 12



Requirements for “in vessel” structural materials

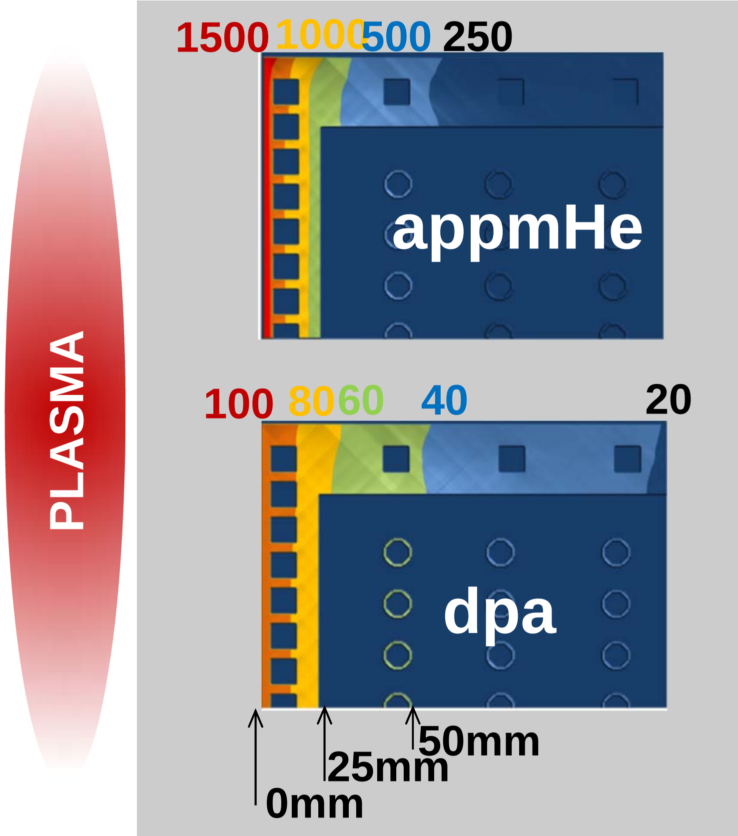
Fission – Fusion – Spallation: Three different irradiation loadings

	Fission (Gen. I)	Fission (Gen. IV)	Fusion (DEMO-PROTO)	Spallation (MYRRHA)
Structural alloy T_{max}	<300˚C	500-1000˚C	550-1000˚C	400-600˚C
Max dose for core internal structures	**~1 dpa**	**~30-150 dpa**	**~50-150 dpa**	**≤60 dpa/fpy**
Max transmutation helium concentration	**~0.1 appm**	**~3-10 appm**	**~500-1500 appm (~8 times more for SiC)**	**~2000 appm/fpy**
Particle Energy E_{max}	**<1-2 MeV**	**<1-3 MeV**	**<14 MeV**	**several hundred MeV**

Materials R&D towards:
- improved irradiation resistance
- enhanced temperature window
- convincing compatibility with coolants

Displacement damage and He production in Blankets

Helium production (appm) for 100 dpa at plasma facing side



H. Tanigawa, E.Wakai 2012

- "Only" the first few centimeters have a high He/dpa ratio
- In addition this part of the blanket carries the highest thermo-mechanical loads
- Therefore,
  - fission reactor irradiations are still meaningful for a significant fraction of in-vessel components
  - Nevertheless, a dedicated fusion neutron source is indispensable, but has to focus on plasma-near materials and loading conditions

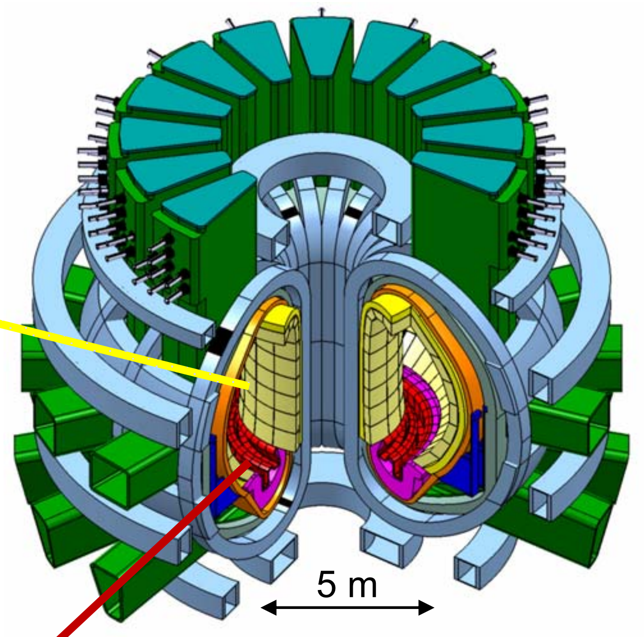
Fusion Power Plants: Structural Material Challenges

Blanket: ≤30 dpa/yr, ≤ 2.5MW/m²

- Plasma Facing Materials
- Reduced Activation Structural Materials:
  - RAFM Steels (EUROFER, F82H) 350-550 °C
  - RAFM ODS Steels 300-650 °C
- Functional Materials
  - Neutron Multipliers (Be), Li ceramics

Divertor: ≤10 dpa/yr, 10-15 MW/m²

- Refractory alloys (e.g. W-materials)
  850-1100 °C → ~600 - 1300 °C
- Nano-scaled RAF(M)-ODS Steels
  350-650 °C → ~300 - 800 °C



- Materials challenges: fusion – fission – spallation
- **Neutron irradiation: Examples of progress and issues**
  - **Reduced activation ferritic/martensitic steels**
  - Oxide dispersion strengthened steels
  - W alloys: I-1, I-3, O-1, O-2: Yesterday
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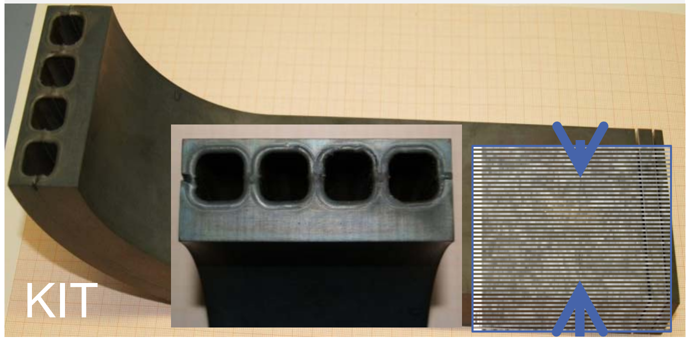
Recent past and present:

Qualification of RAFM steels EUROFER & F82H mod

Eurofer Blanket Module

ODS steel, diffusion welding



Main Achievements

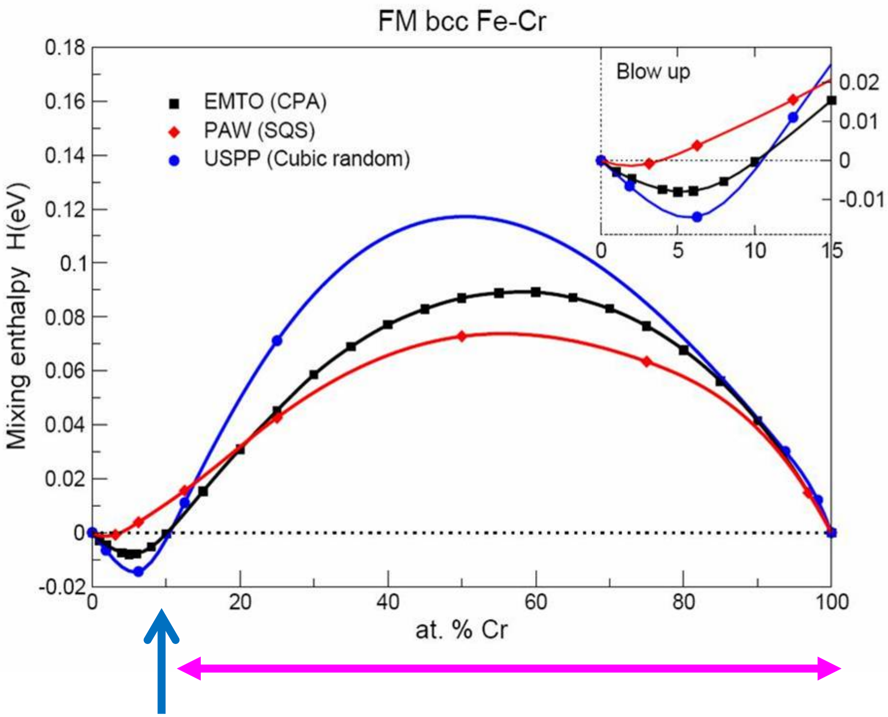
- Meanwhile ~35 tons of EUROFER & ~50 tons of F82H mod delivered
- Broad based Qualification Programme, including joining technologies and corrosion
- Fission neutron based materials data up to ~70 dpa
- Implementation of EUROFER database into the RCC-MRx code for ITER Test Blanket Modules ongoing

Future Mission

- Increase upper operation temperature and improve neutron/He/H embrittlement
- by fine tuning of alloying elements

Why RAFM steels are based on 9wt% Cr?

S. Dudarev et al, UKEA, 2006

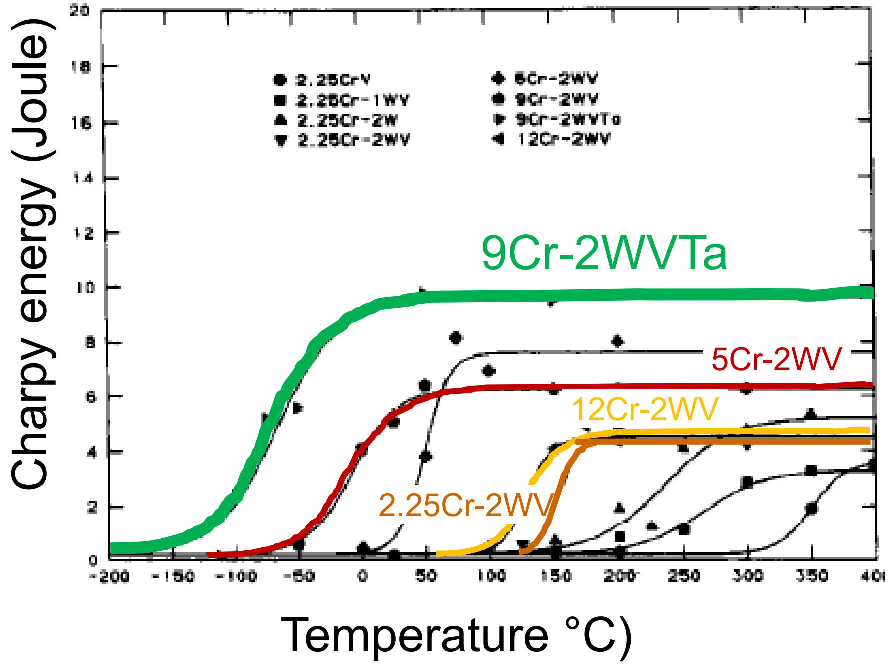


9-10%Cr is favorable

Above 10% Cr: formation of Cr precipitates (σ-phase, α')

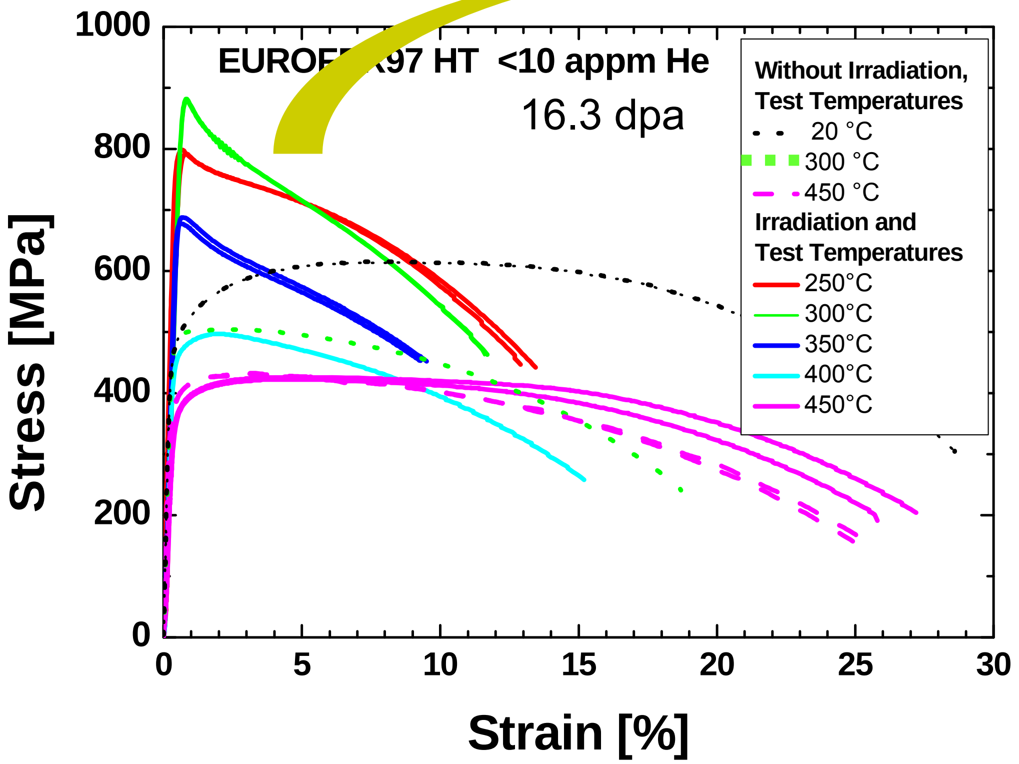
R. Klueh et al, J. Nucl. Mat. (1996) 336

Neutron irrad., 20-24 dpa, T_{irr} = 365 °C

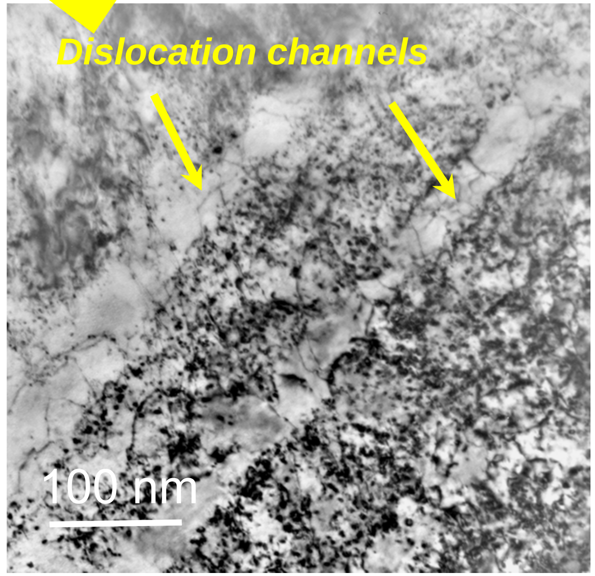


Experimentally verified:
9Cr-(1-2)WVTa steels have superior aging and irradiation properties

RAFM 8-10%CrWVTa-Steels without Helium: Tensile results after neutron irradiation



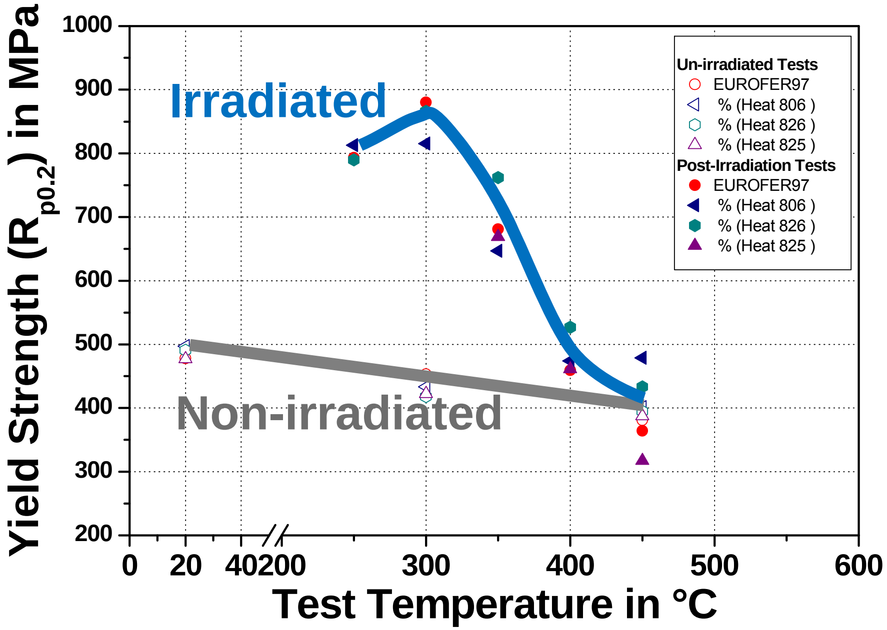
E. Materna-Morris, et al. JNM 386(2009) 422



N. Hashimoto et al., Fus.Sci.Tech. 44 (2003)

- Below ~400 °C: - Strain localization due to dislocation channeling
  - Despite the very small uniform elongation rupture stress still high

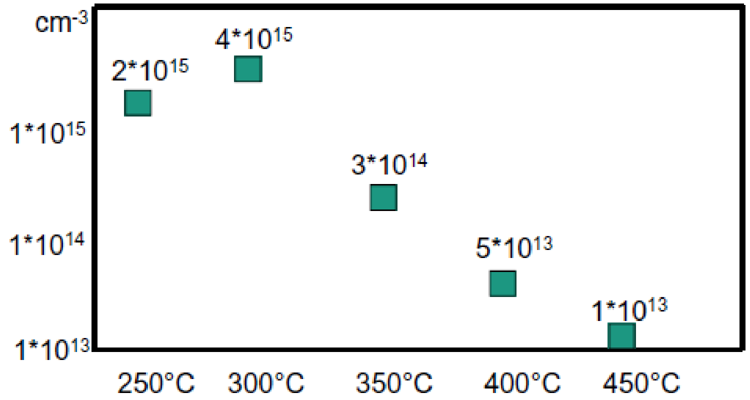
Substantial irradiation induced hardening below T_{irr} ~ 420°C by interstitial type defects

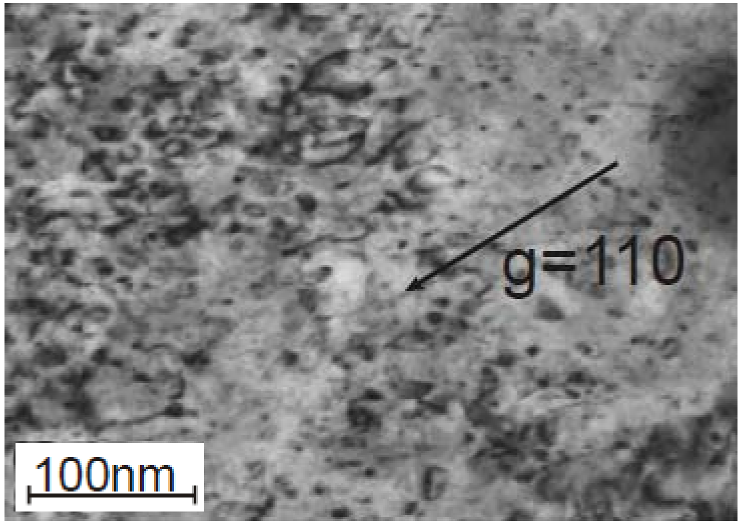


T_{irr} = 300°C:

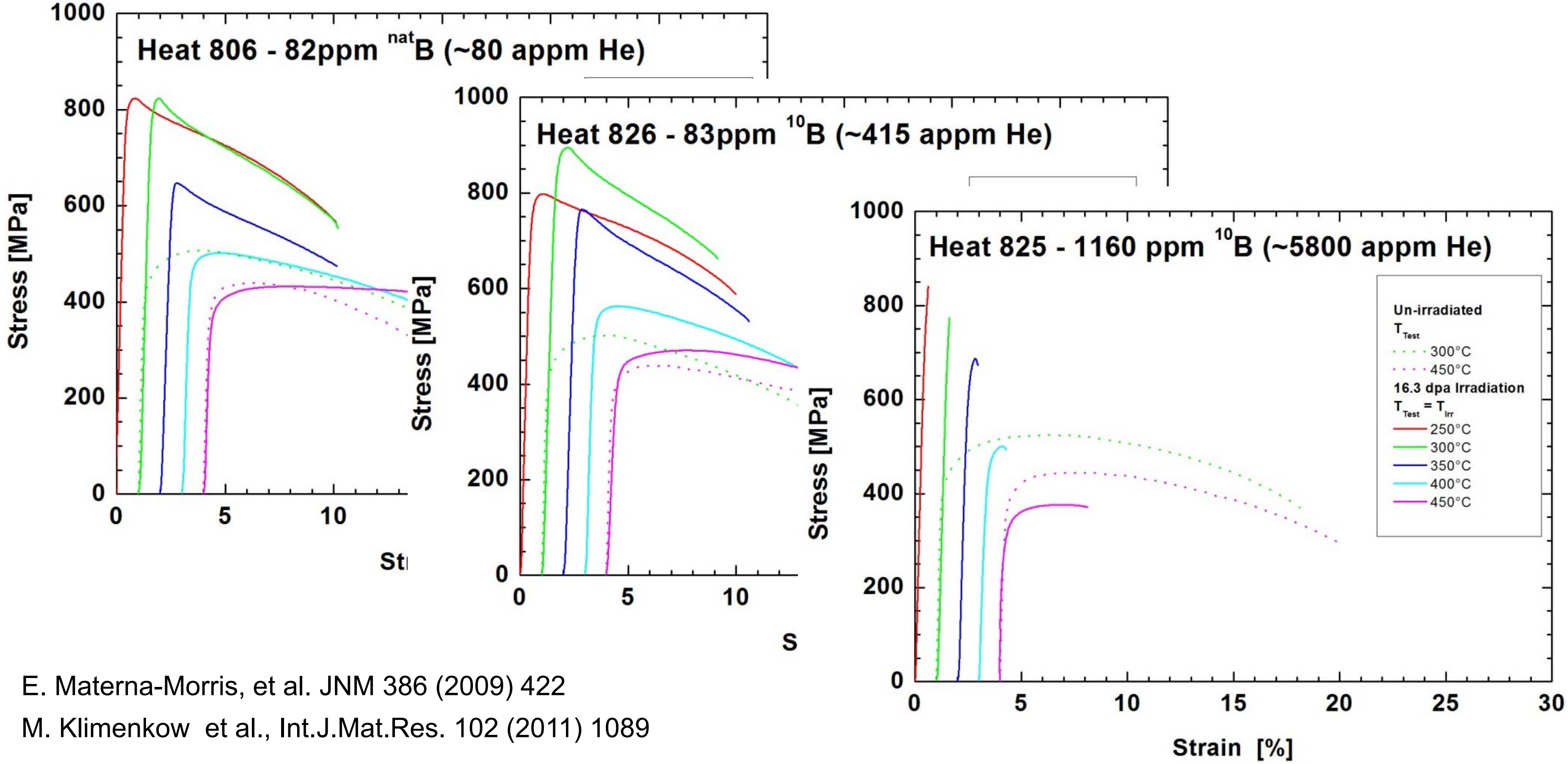
- Homogeneous distribution of point defects and dislocation loops
- Dislocation loops:
  - ½<111> Burgers vector
  - 5-25nm diameter

Concentration of irrad. induced defects





Ductile or brittle? The importance of strain rate $\dot{\varepsilon}$: Example: Eurofer, 16 dpa, B-doped, $\dot{\varepsilon} \approx 10^{-3}s^{-1}$

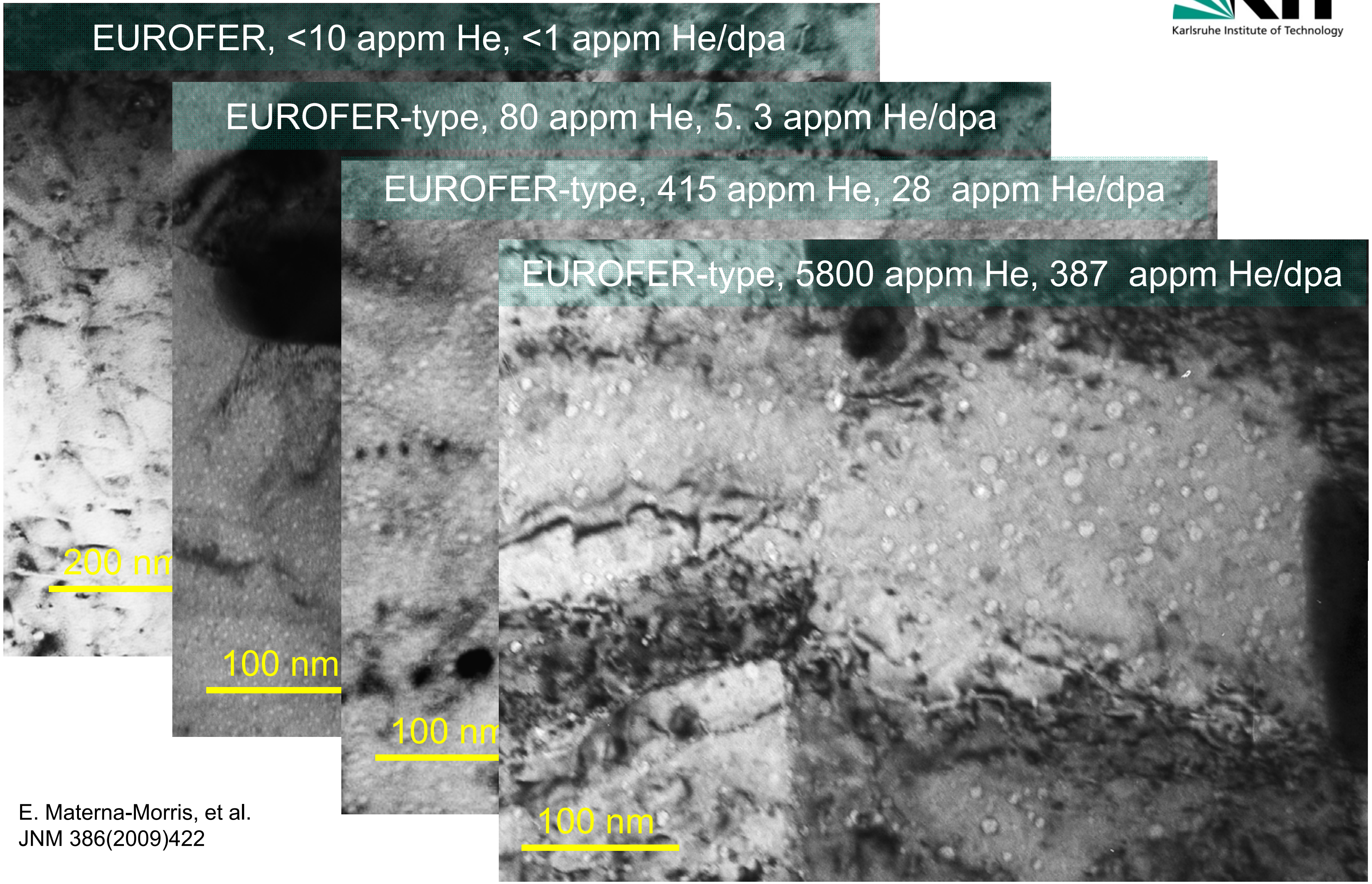


E. Materna-Morris, et al. JNM 386 (2009) 422

M. Klimenkow et al., Int.J.Mat.Res. 102 (2011) 1089

- ≤415 appm He: Almost no effect on tensile properties at small strain rates
- 5800 appm He: Entirely brittle fracture; total loss of plasticity

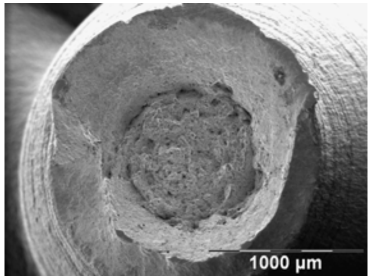
EUROFER, 15 dpa neutron irradiation at 250 °C

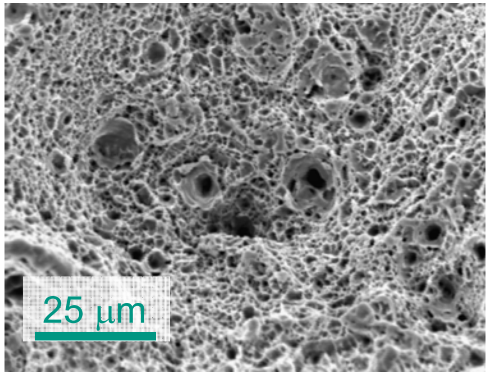


E. Materna-Morris, et al.
JNM 386(2009)422

Fracture behavior of tensile tested EUROFER 16 dpa, T_{irr} = T_{test} = 300 °C

EUROFER97, <10 appm He

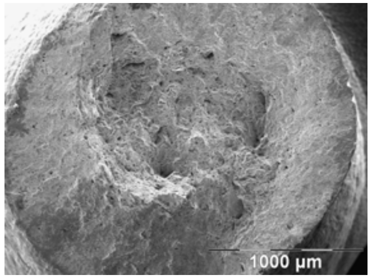




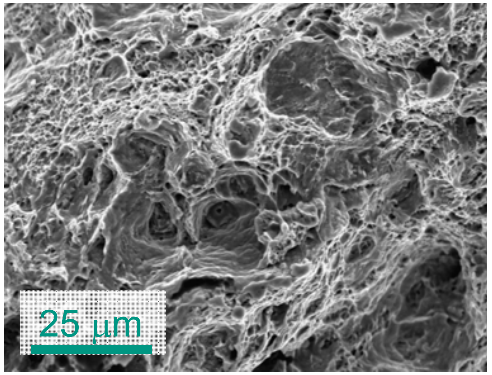
E. Materna-Morris et al., JNM 386-388 (2009) 422-425
M. Klimenkov et al., Micron 46 (2013) 51–56

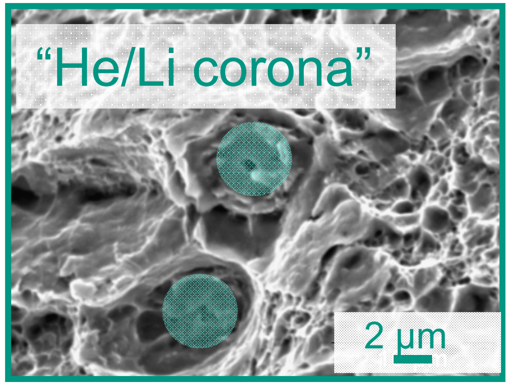
- ^{10}B-doping: He and Li effects cannot really be decoupled

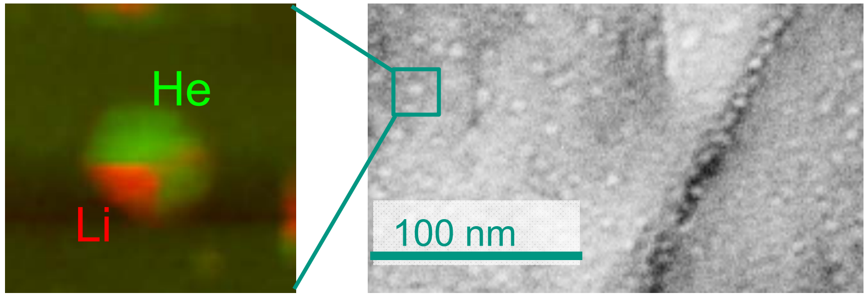
EUROFER-type, 415 appm He



n + ^{10}B → ^{4}He + ^{7}Li + 2.7895 MeV
range of He (1.0 MeV): 1.6 µm
range of Li (1.8 MeV): 2.0 µm

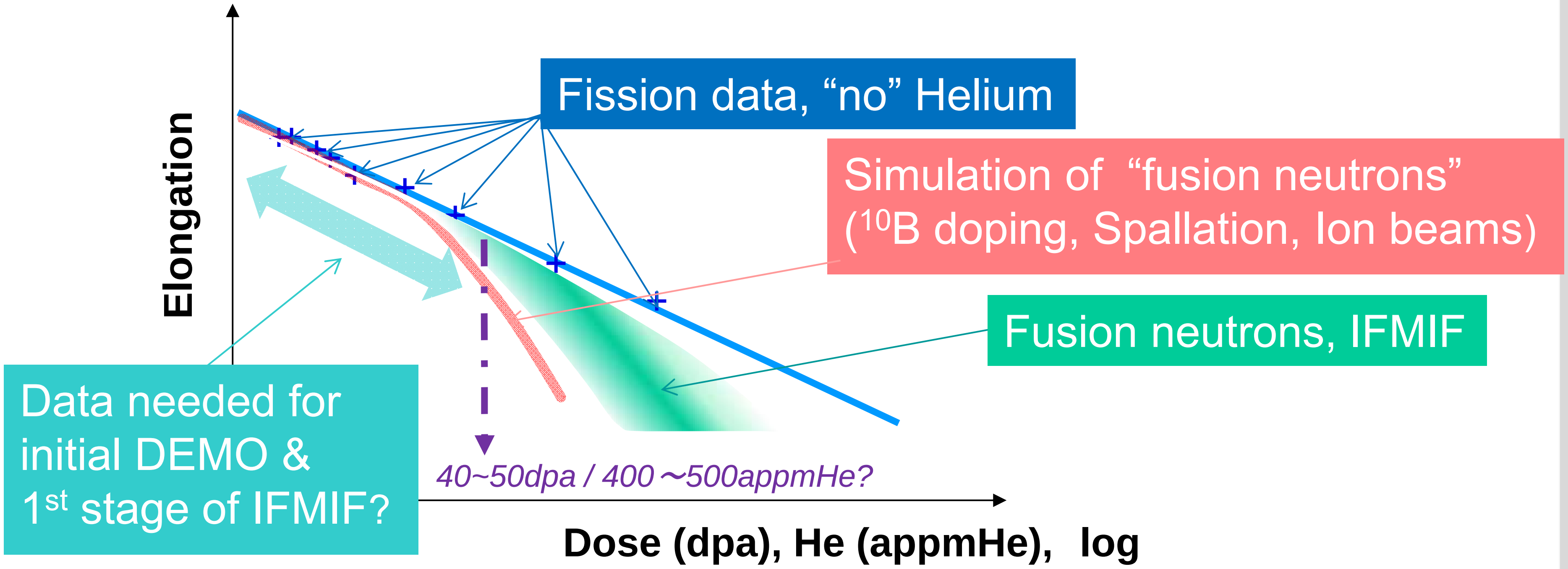






Possible steps towards a fusion design code: Tensile properties as example

In the style of H. Tanigawa, E.Wakai

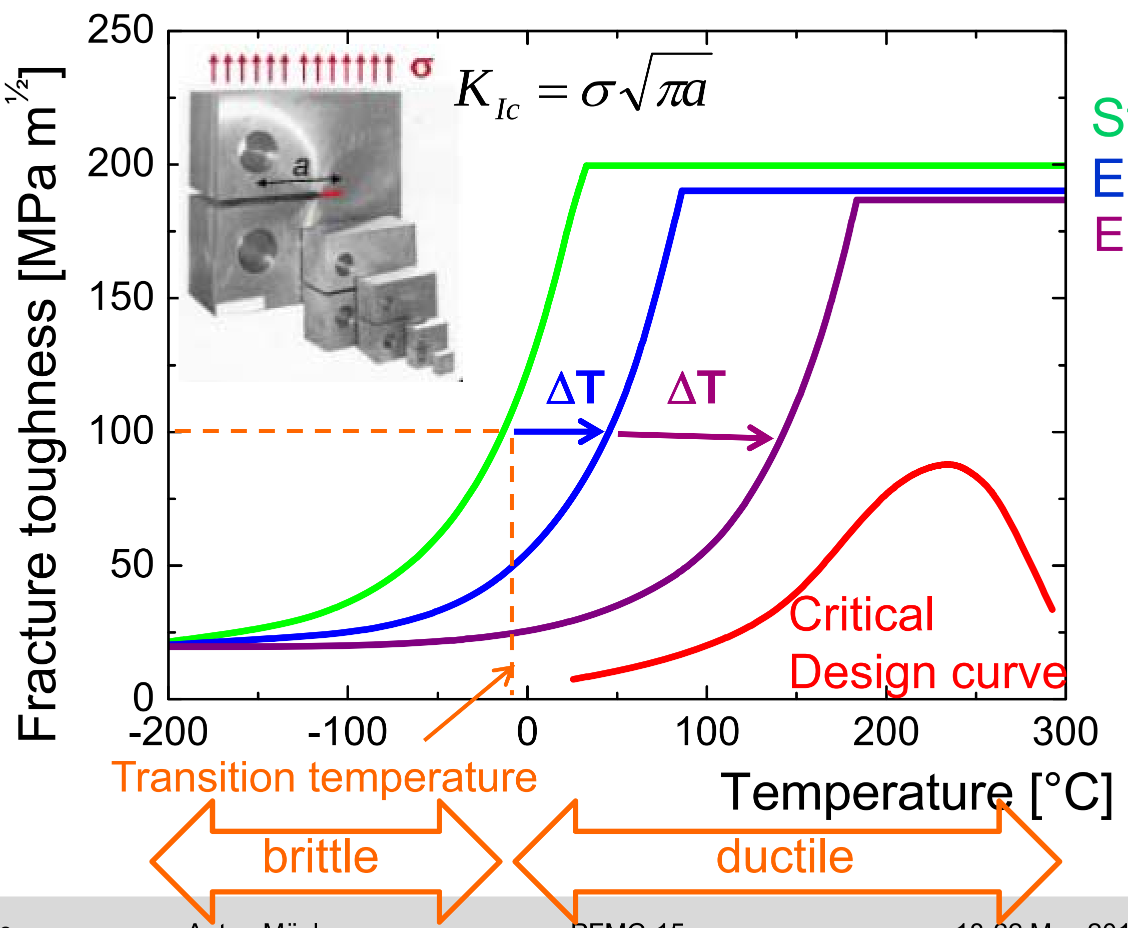


- ❑ <u>Small strain rate</u>: Critical condition 40~50 dpa / 400~500 appm He?
- ❑ This might be also the parameter window for initial DEMO and 1st stage of IFMIF and related design code development

Ductile or brittle?

Indispensable for safety, economy & life-time prediction

The accurate prediction of the ductile-to-brittle-transition temperature shift is fundamental for ensuring the structural integrity of reactor pressure vessels (Fission) and of blanket/divertor (Fusion)

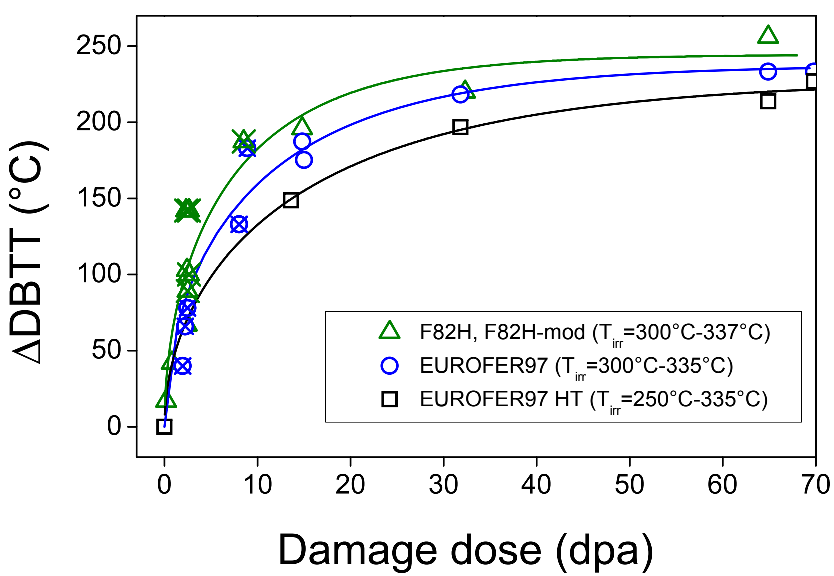


Fracture toughness specimens are indispensable in all test matrixes

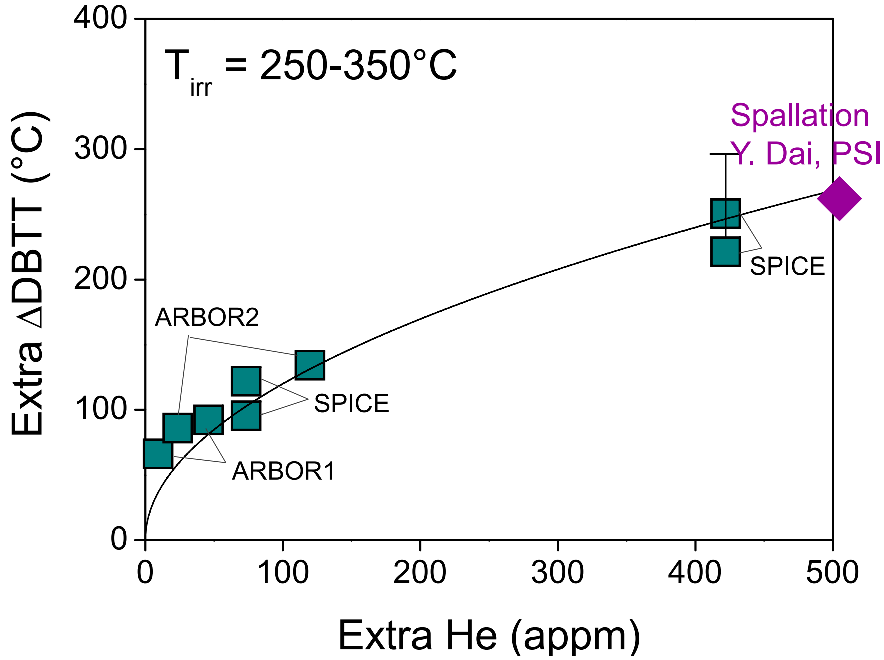
Ductile or brittle? The importance of strain rate ε: Example: Eurofer, 16 dpa, B-doped, $\dot{\varepsilon} \approx 10^2 s^{-1}$

E. Gaganidze et al., J. Nucl. Mater. 417 (2011)93-98

EUROFER, <10 appm He

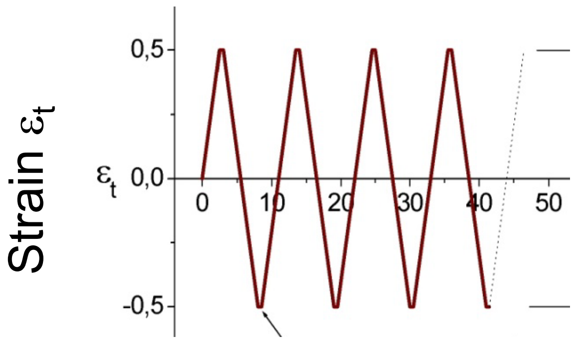


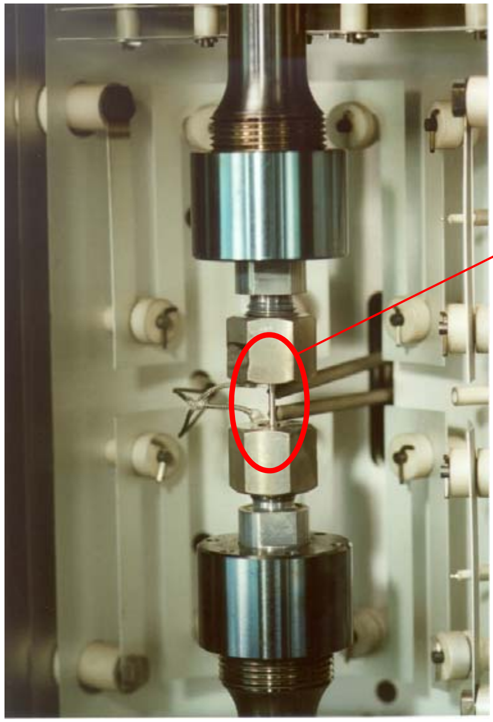
EUROFER, 10-500 appm He

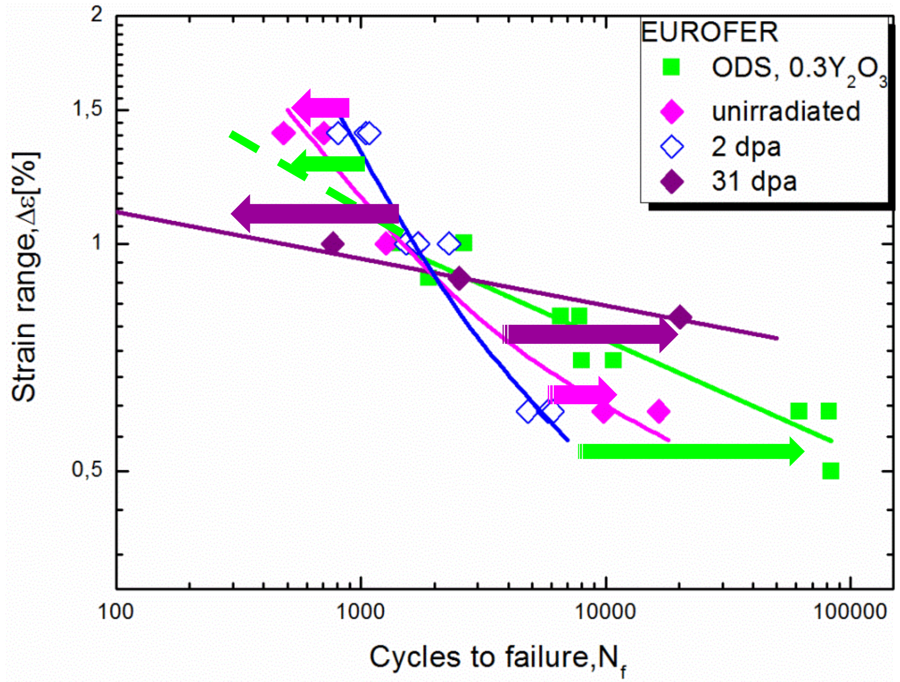


- ❑ <u>High strain rates</u>: Helium effects are an outstanding issue; Saturation??
  - ➡ He determines the lower operation temperature in DEMO blankets (~350°C).
- ❑ B-doping & Spallation neutrons are too aggressive
  - ➡ intense fusion n-source indispensable

Fatigue:
Irradiation can lead to shorter or longer lifetime







- high strain regime, shorter lifetime
 →accelerated crack initiation

- low strain regime, prolonged lifetime
 →Micro-crack growth impeded

$\Delta\varepsilon$ controlled LCF after 16.3 dpa irradiation

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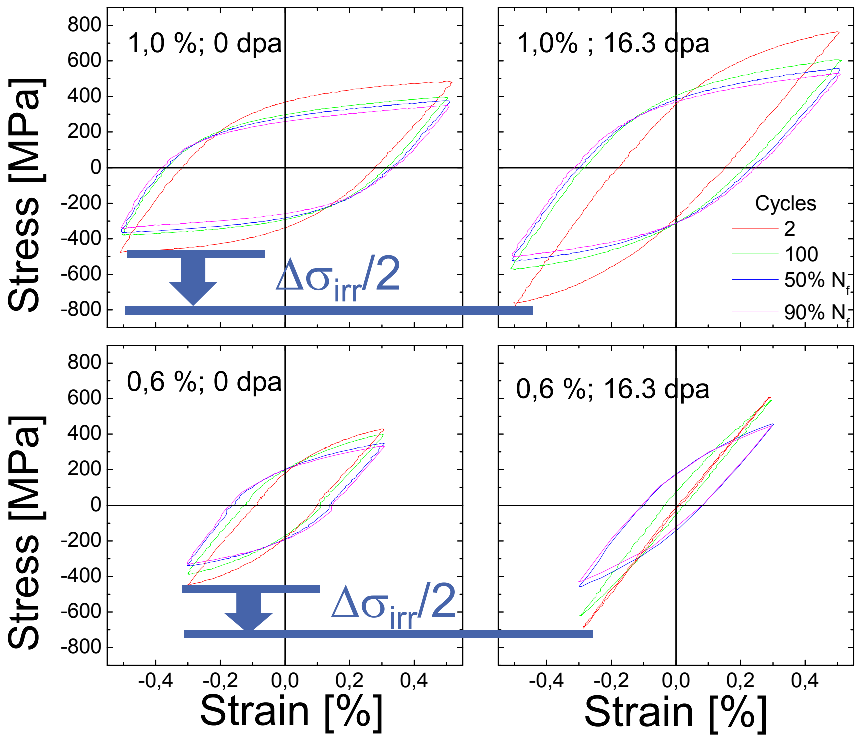
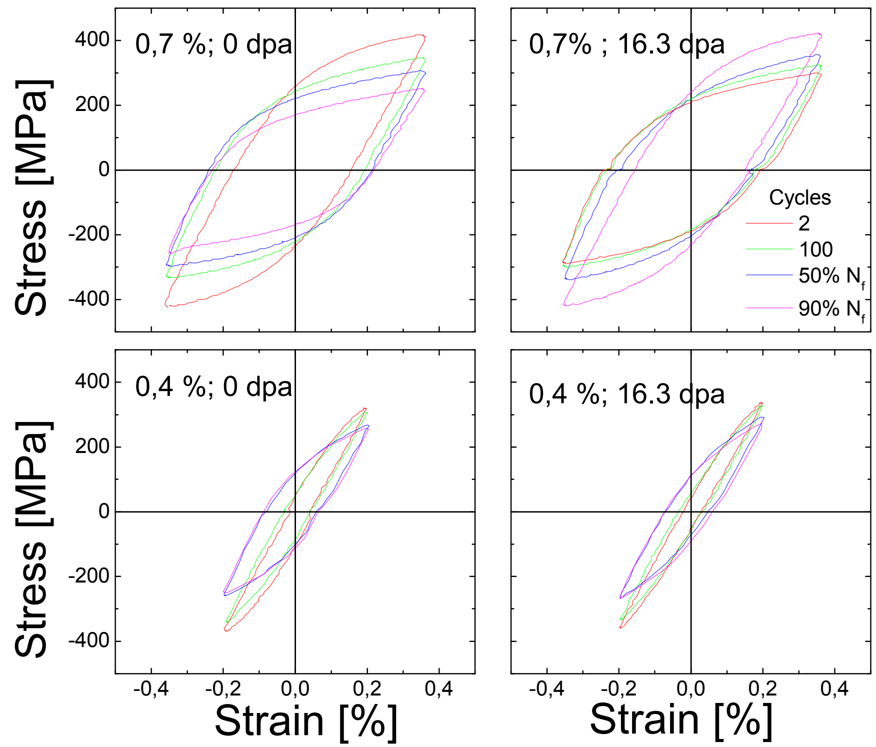
$T_{irrad.} = T_{test} = 250$ °C



$T_{irrad.} = T_{test} = 450$ °C



- Significant irradiation induced hardening $\Delta\sigma_{irr}$ at all $\Delta\varepsilon$
- Cyclic softening somewhat higher after irradiation

- No irradiation induced hardening $\Delta\sigma_{irr}$ at all
- Hysteresis loops very similar for irradiated and non-irradiates samples

Δε controlled LCF after 16.3 dpa irradiation

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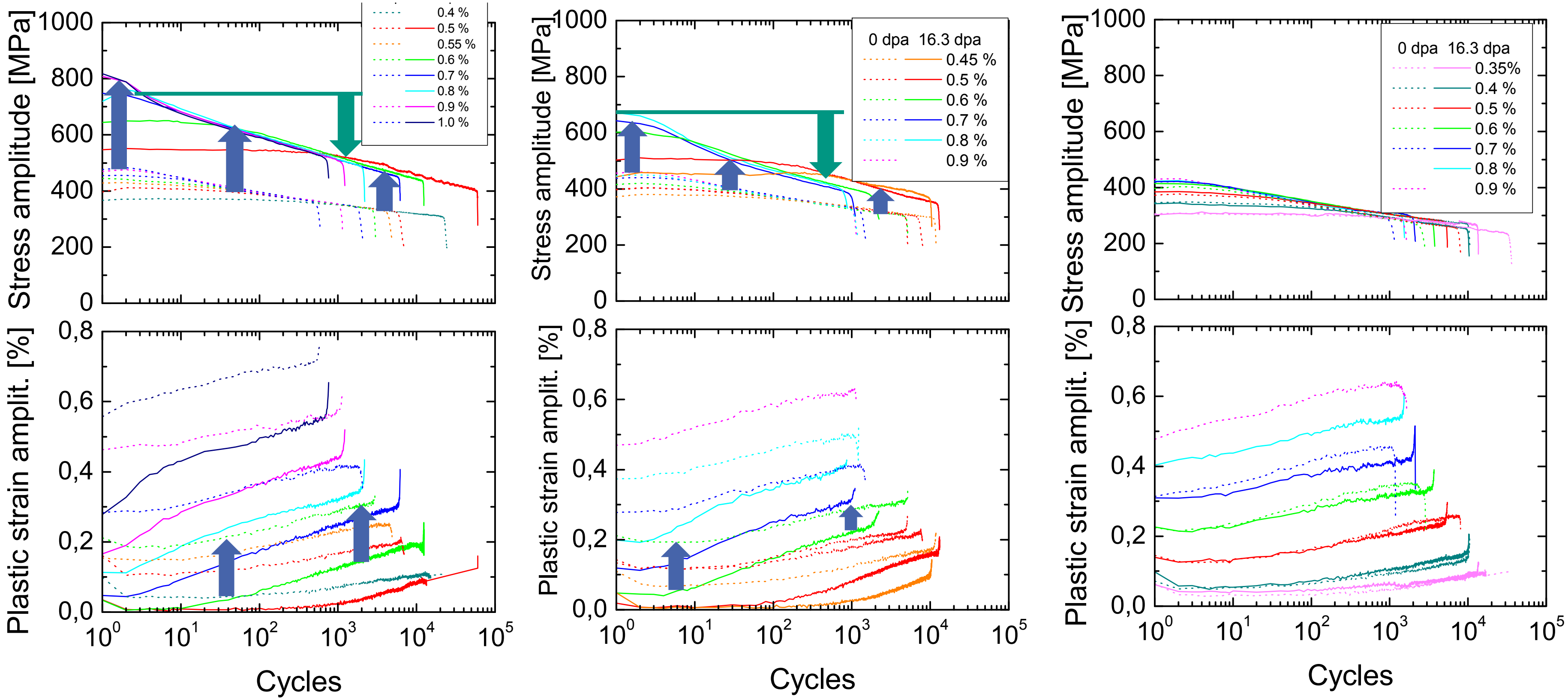
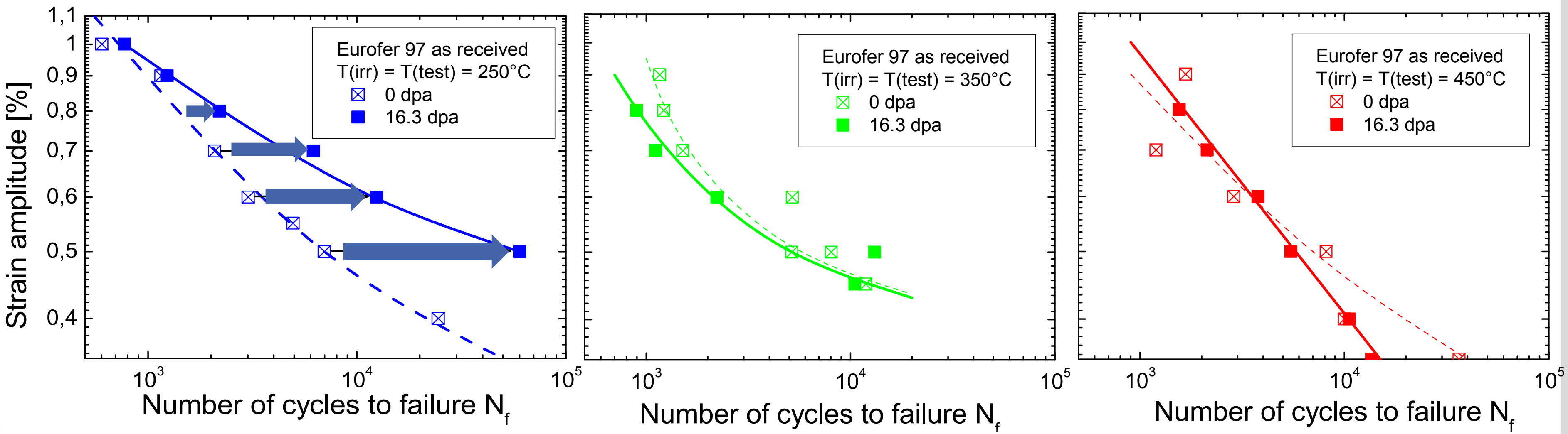
$T_{irrad.} = T_{test} = 250$ °C	$T_{irrad.} = T_{test} = 350$ °C	$T_{irrad.} = T_{test} = 450$ °C



- Below ~400°C: Significant effect of irradiation on Δσ and $\Delta\varepsilon_{pl}$ (⬆) as well as on **cyclic softening** (⬇)
- Above ~400°C: Interstitial type defects unstable → no irradiation effect

Effect of 16.3 dpa irradiation on lifetime

$T_{irrad.} = T_{test} = 250\ °C$ | $T_{irrad.} = T_{test} = 350\ °C$ | $T_{irrad.} = T_{test} = 450\ °C$

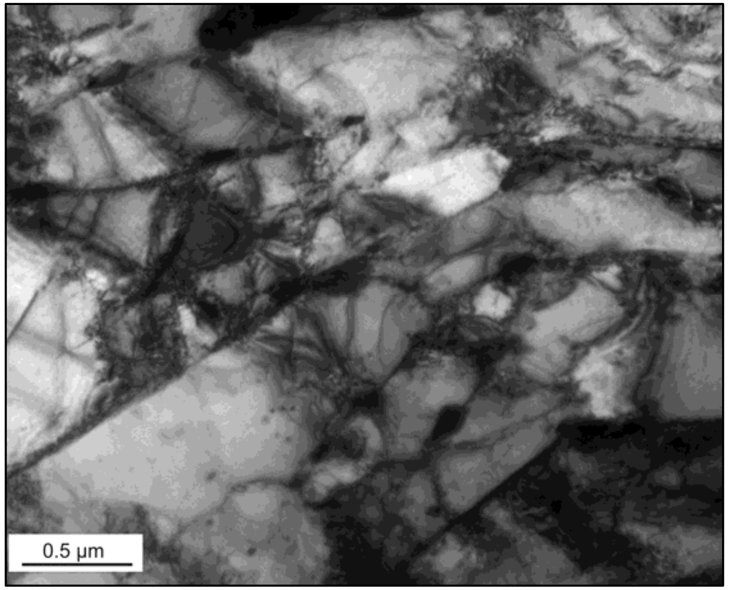


Pronounced increase of lifetime N_f after low temperature irradiation

Lifetime N_f in the range of non-irradiated specimens

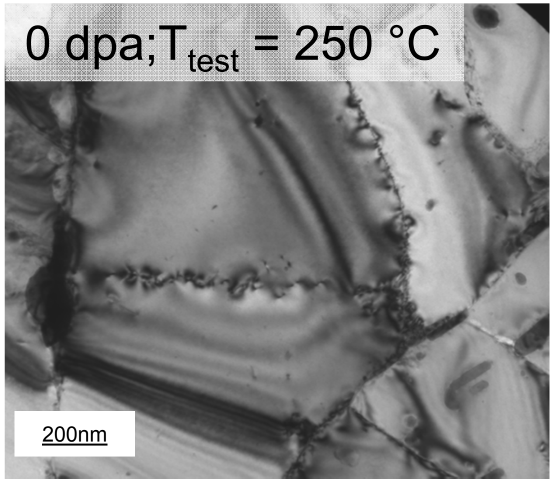
TEM analysis, EUROFER97 (as received)

0 dpa, Before fatigue testing

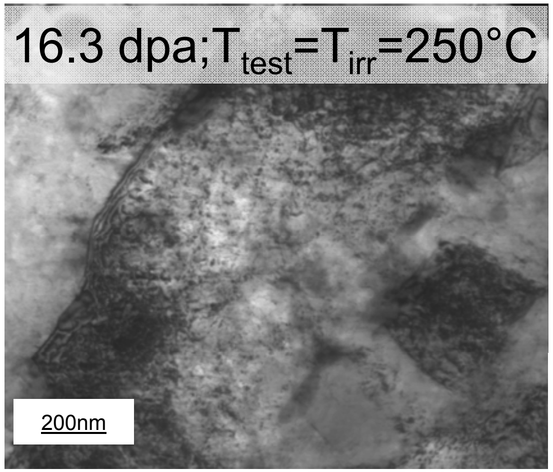


- Microstructure typical for tempered martensite
- High dislocation density

After fatigue testing, $\Delta\varepsilon$ = 0.5 %



- Pronounced sub-grain formation
- Low dislocation density within sub-grains
- Dislocation pile-up at sub-grain boundaries



- Also sub-grain formation, but less obvious
- High stability of irradiation induced defects, despite cyclic motion of dislocations

Is there a saturation of properties with increasing displacement damage ?

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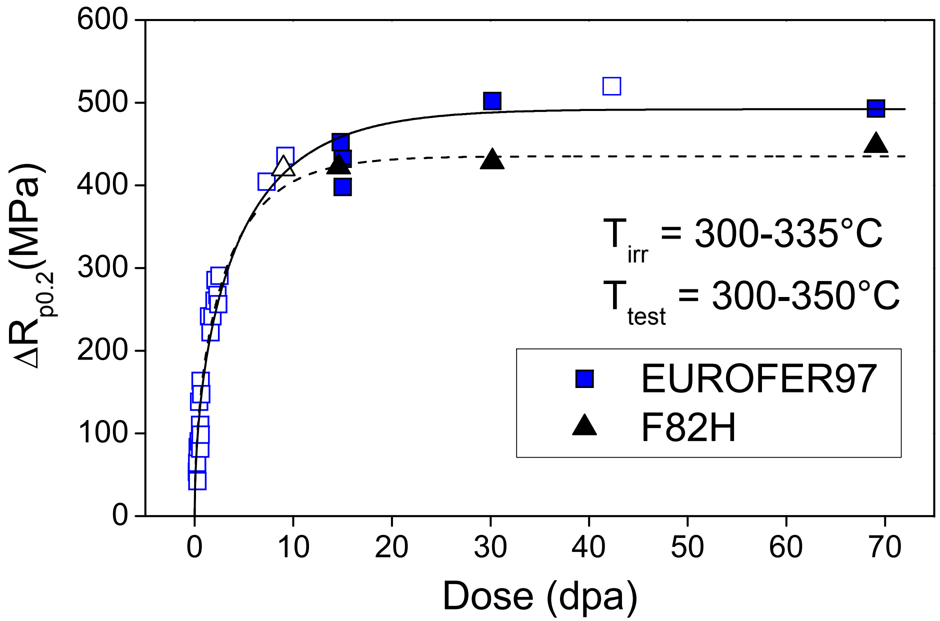
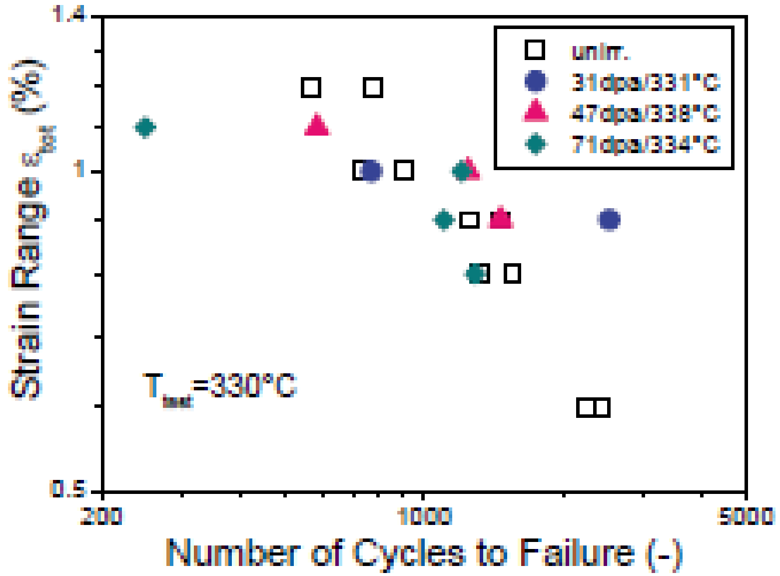
“no” He, Tensile testing



“no” He, fatigue testing



“Orowan-type fit:

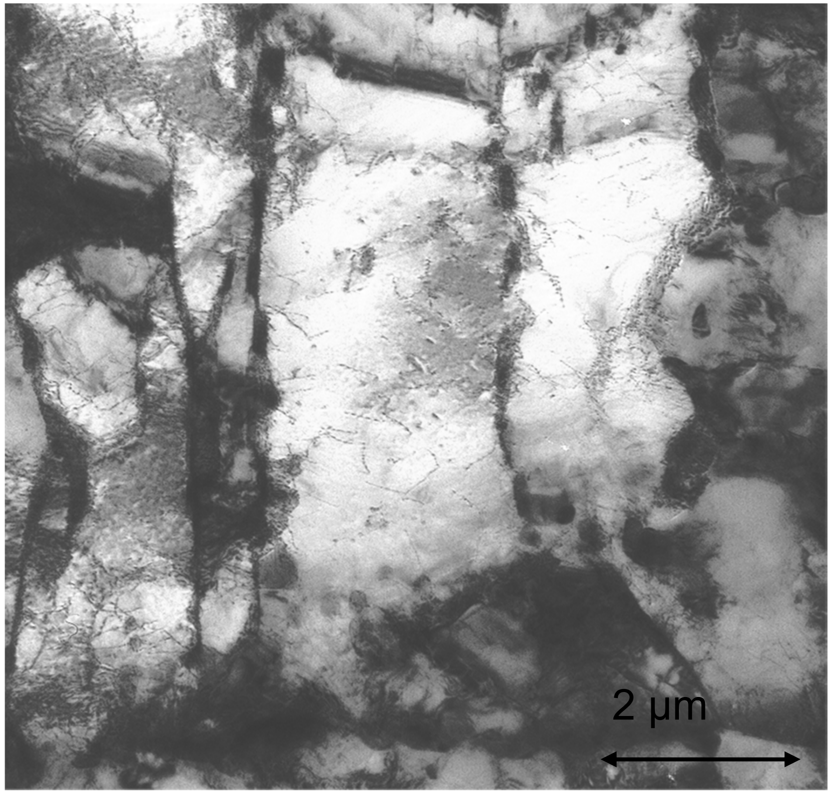
$$\Delta\sigma_{irr} = M\alpha\mu b\sqrt{Nd} \approx \Delta\sigma_{sat}\sqrt{\left[1-\exp(-\Theta/\Theta_0\right]}$$

E.Gaganidze, C. Petersen et al., KIT, 2011

➡ Irradiation induced tensile and fatigue properties seem to saturate beyond ~15 dpa

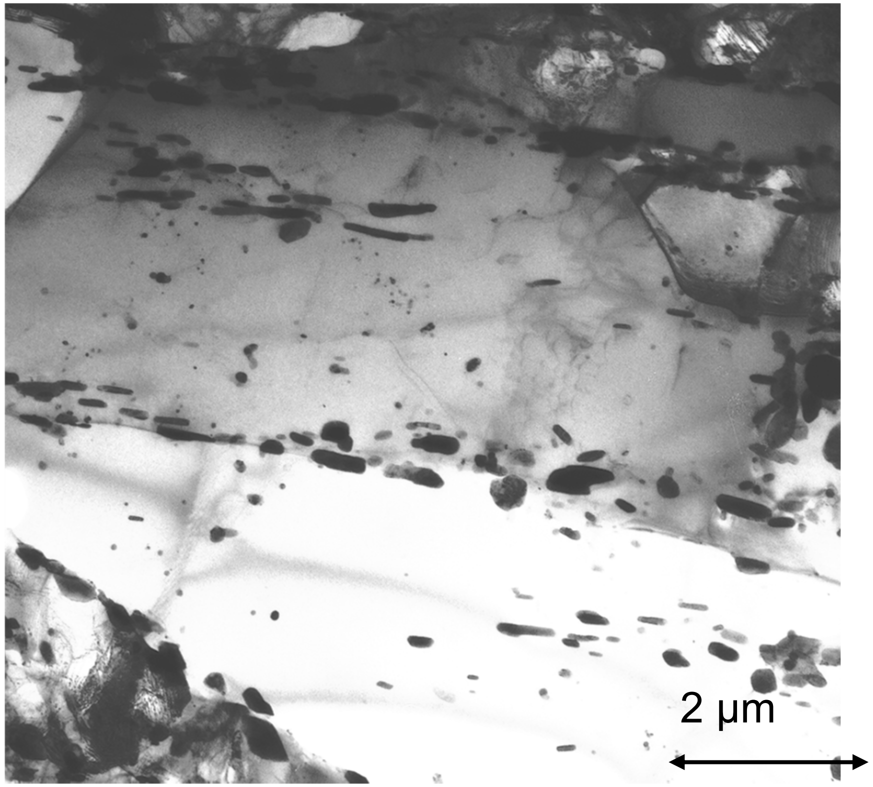
TEM Microstructure

Before testing



After Fatigue testing at 550 °C

$\Delta\varepsilon$ = 0.45 %, N_f = 60850

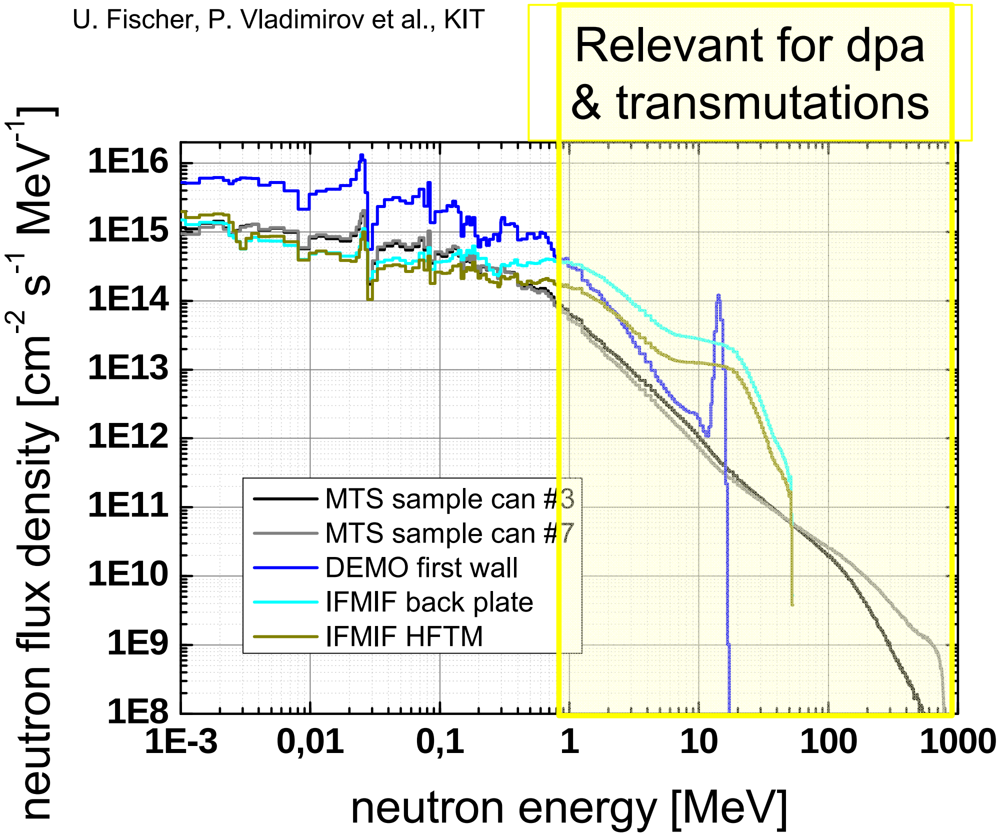


Fatigue testing: Besides sub-cell creation pronounced formation of precipitates (mostly $M_{23}C_6$) → substantial cyclic softening

Neutron spectra effects: Tensile properties

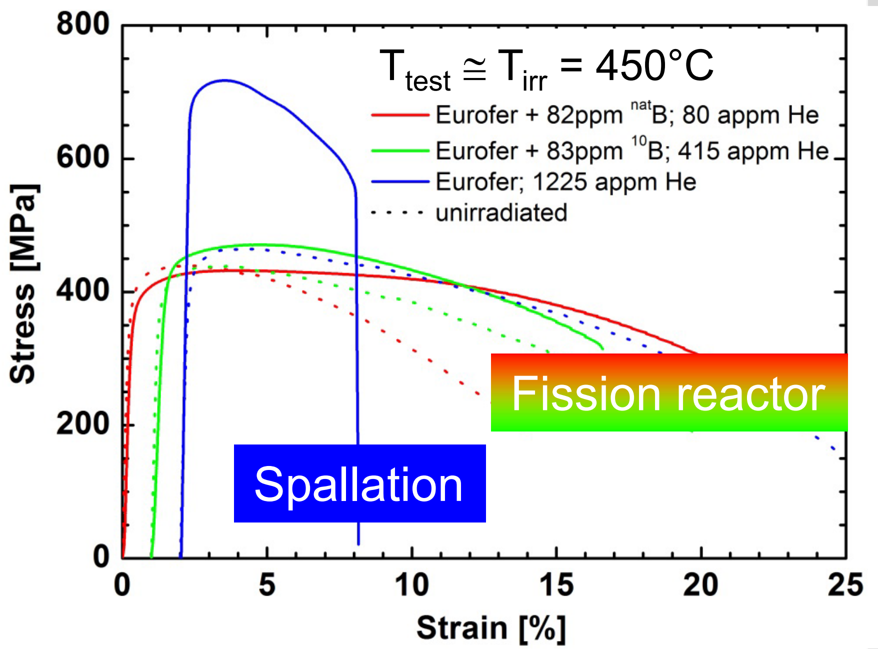
N-spectra: DEMO, IFMIF, Spallation

U. Fischer, P. Vladimirov et al., KIT



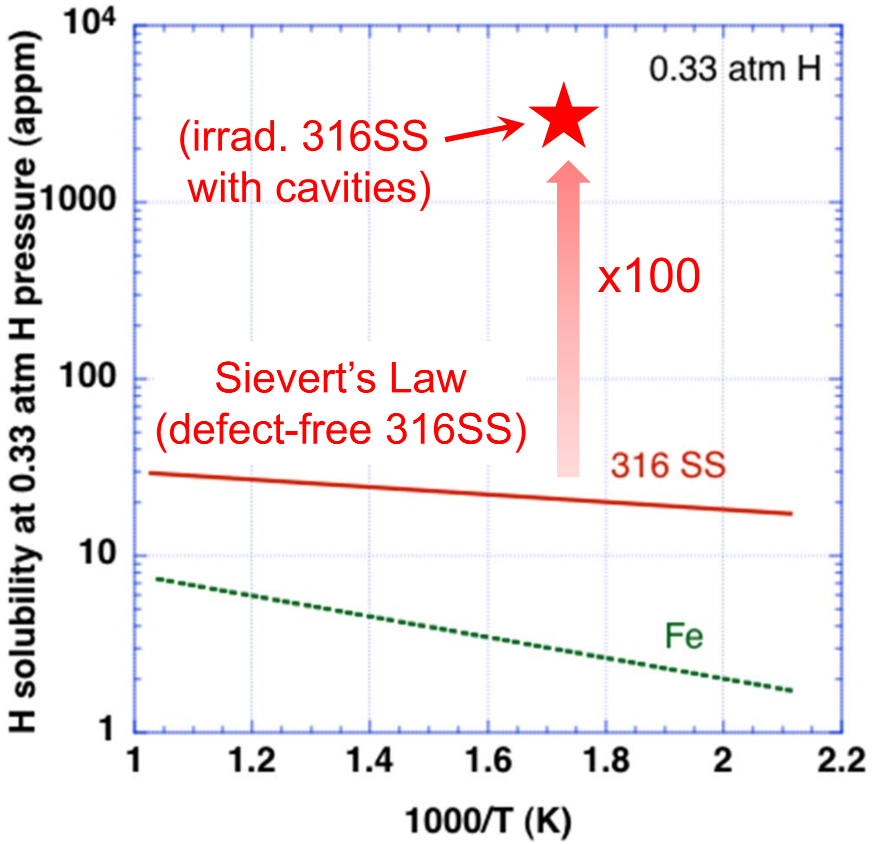
Fission vs. Spallation (PSI)

Y. Dai et al, He-dpa workshop, June 2009, PSI



- Spallation irradiation shows above T_{irr} ≅400°C much higher strength $\Delta\sigma_{irr}$
- What is the real fusion behavior? Intense N-source would answer this question

Hydrogen effects:
Retention of hydrogen in Fe and austenitic Steel 316



F.A. Garner; J. Nucl. Mater. **356** (2006) 122-135

S.J. Zinkle, A. Möslang, Fus. Engin. and Design 88 (2013) 472– 482

- Hydrogen effects may become a serious issue in fusion environments
- Typical for 14 MeV fusion neutrons is the simultaneous production of dpa, hydrogen and Helium → Intense fusion neutron source indispensable for realistic validation

- Materials challenges: fusion – fission – spallation
- **Neutron irradiation: Examples of progress and issues**
  - Reduced activation ferritic/martensitic steels
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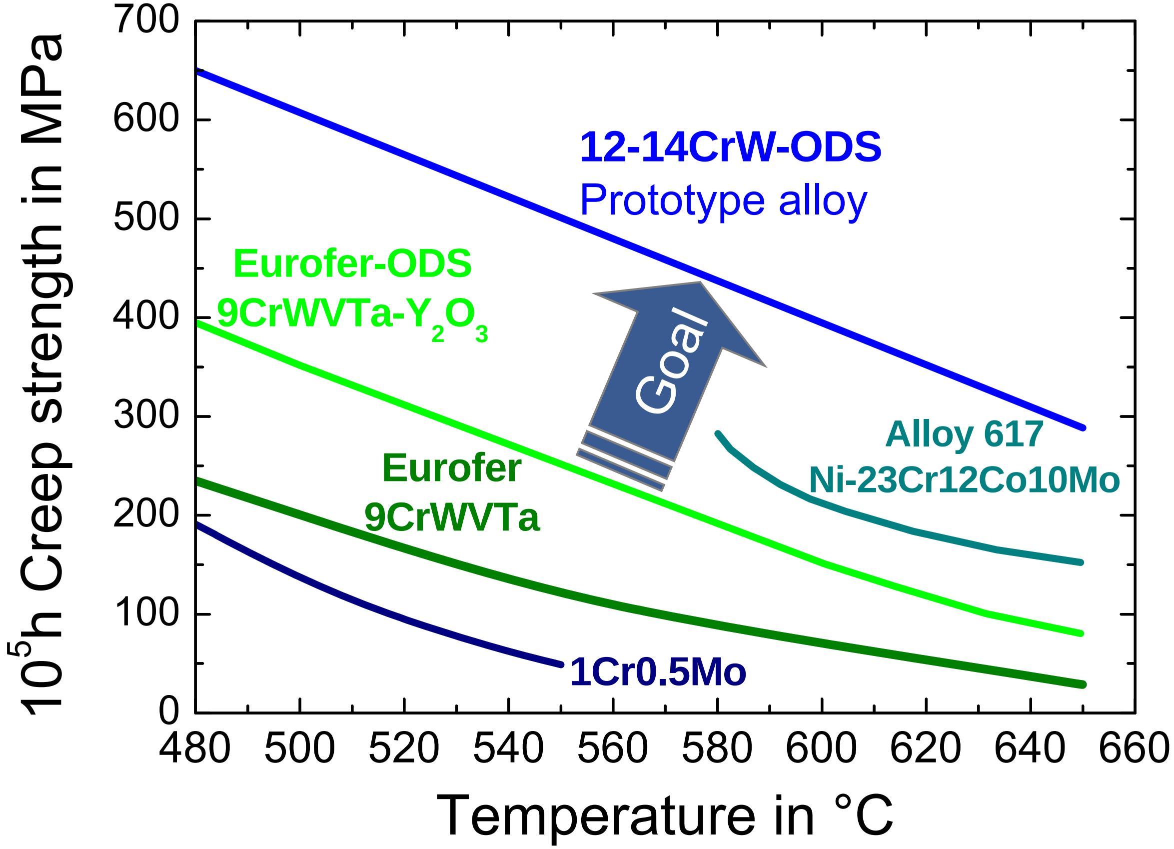
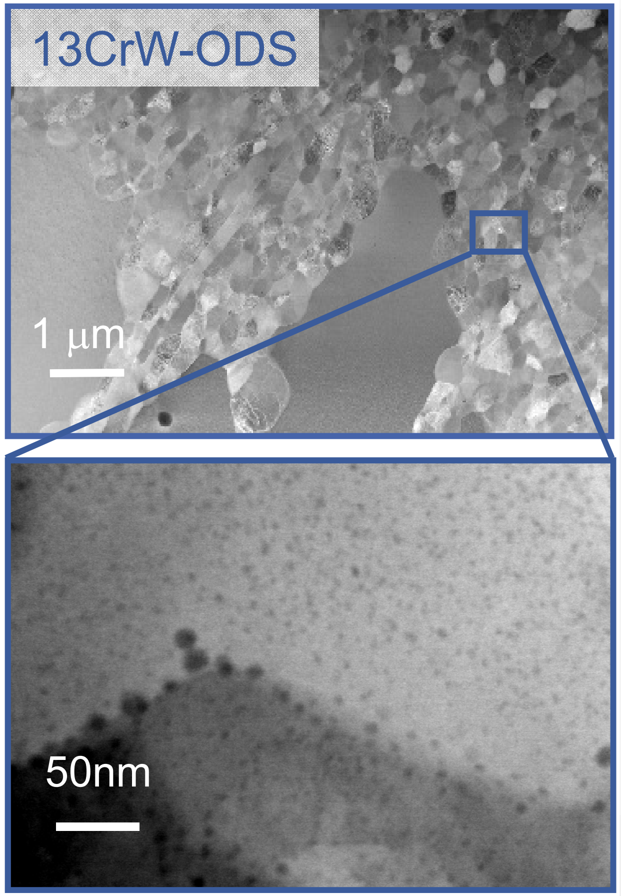
- Role of materials in fusion roadmaps,
  Need for fusion specific irradiation source

International challenge: Development of nanoscaled iron based “super alloys” (RAF-ODS)

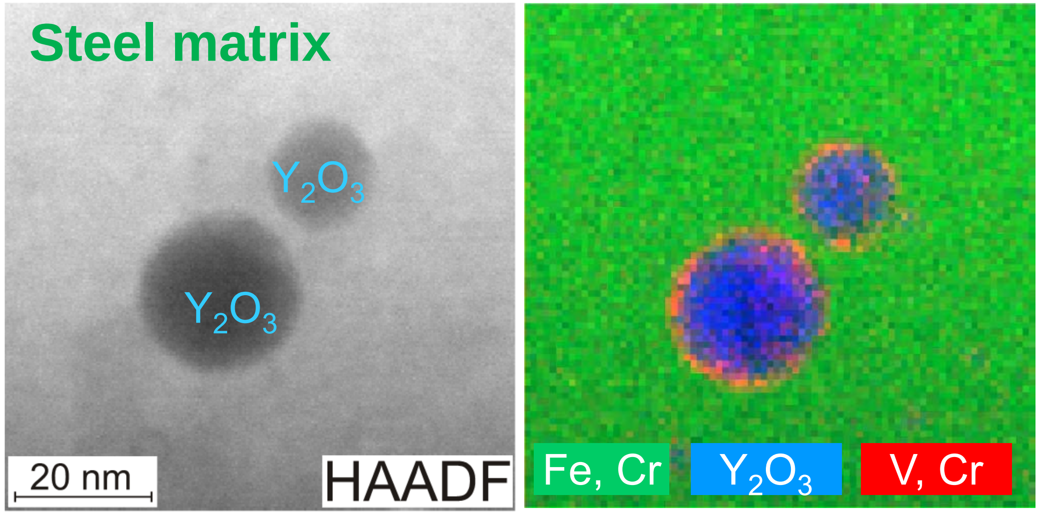




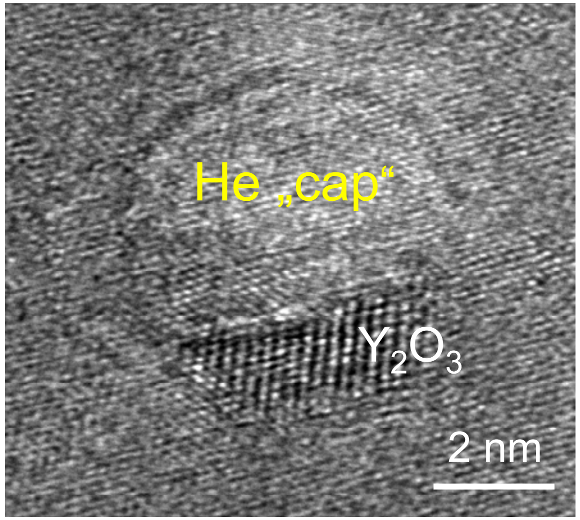
E. Eiselt, M. Klimenkov, R. Lindau, A. Möslang
J. Nucl. Mater 2010

Nanoscaled ODS-steels

1000 appm He impl. at 500°C



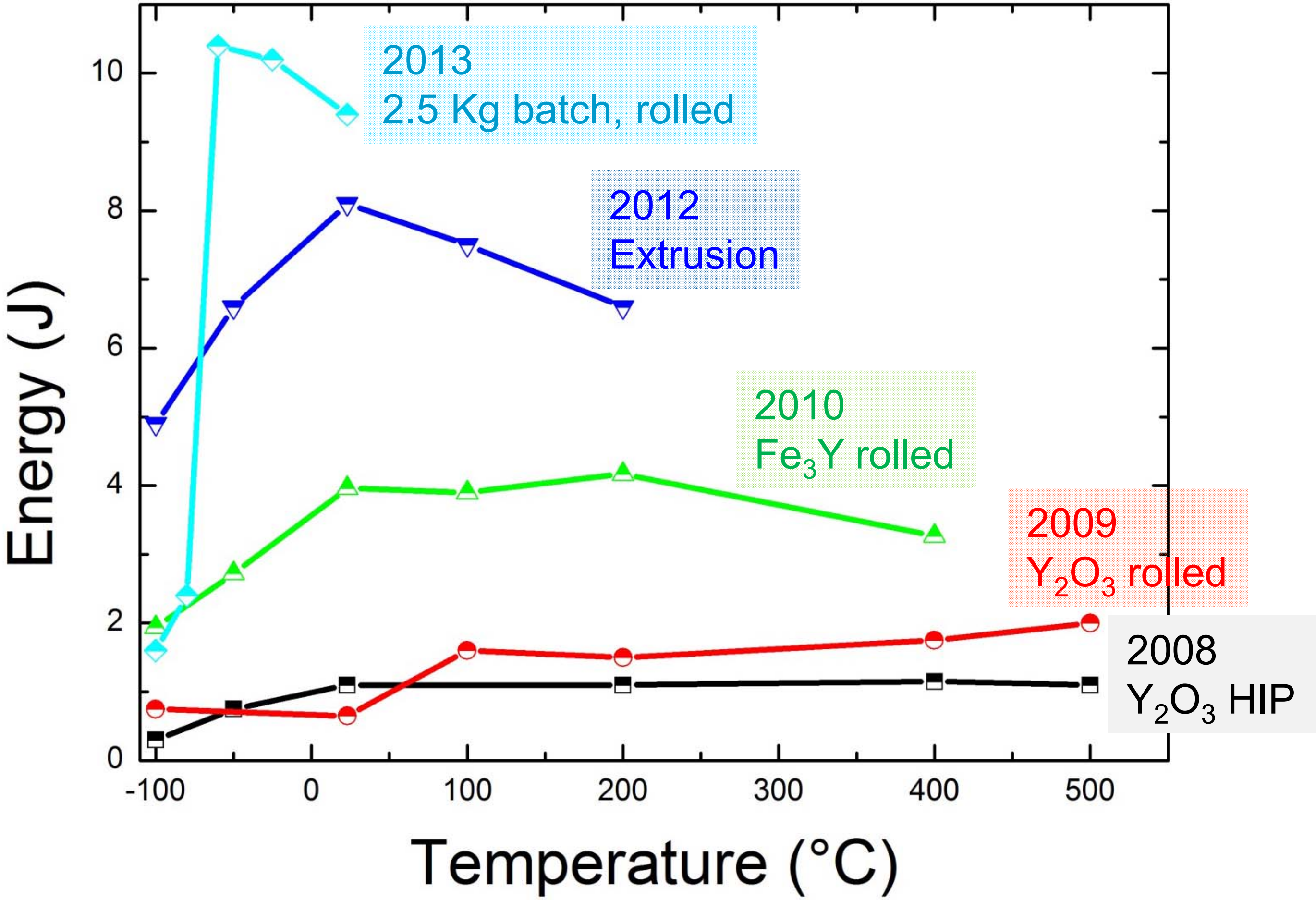
P. He, M. Klimenkov et al., J. Nucl. Mater. 428 (2012) 131-138



A. Ryazanov et al; JNM (2013) 153-157

- Nano-scaled ODS particles like Y_2O_3 or $Y_2Ti_2O_7$ are efficient trapping centers for diffusing alloying elements (Cr, V) and irradiation induced defects (vacancies, He)
- Therefore, nanoscaled ODS steels have potential for outstanding aging and irradiation resistance

Nanoscaled ODS-steels:
Substantial improvement of dynamic fracture toughness



J. Hoffmann et al, IAM, KIT, 2014

ODS EUROFER after Neutron irradiation:
Substantial improvement of tensile properties

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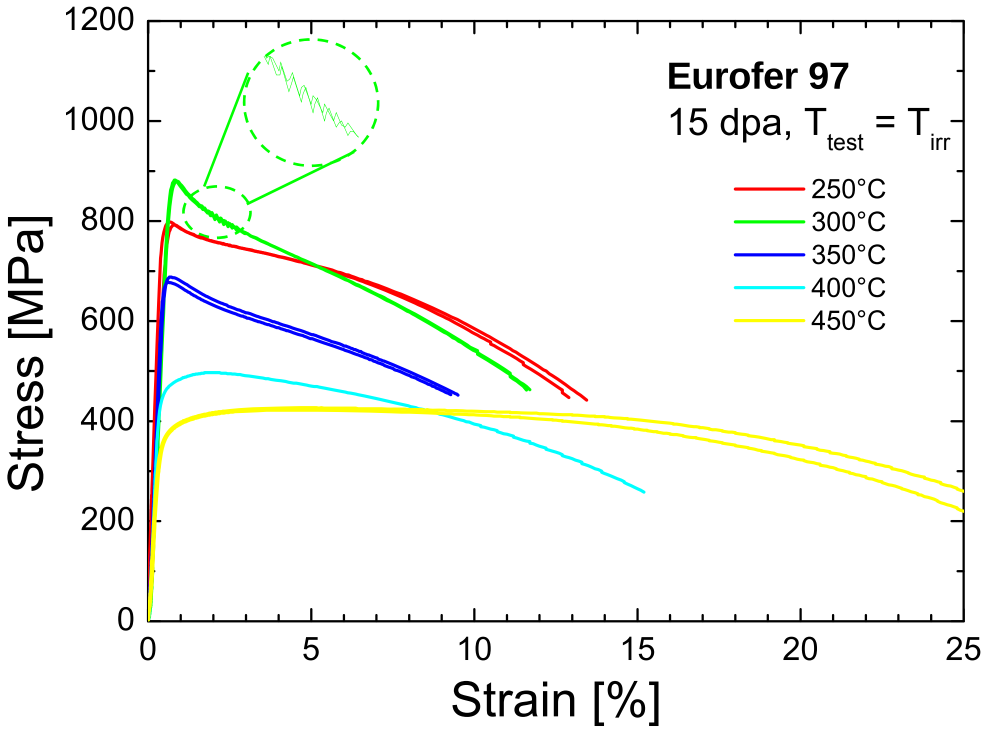
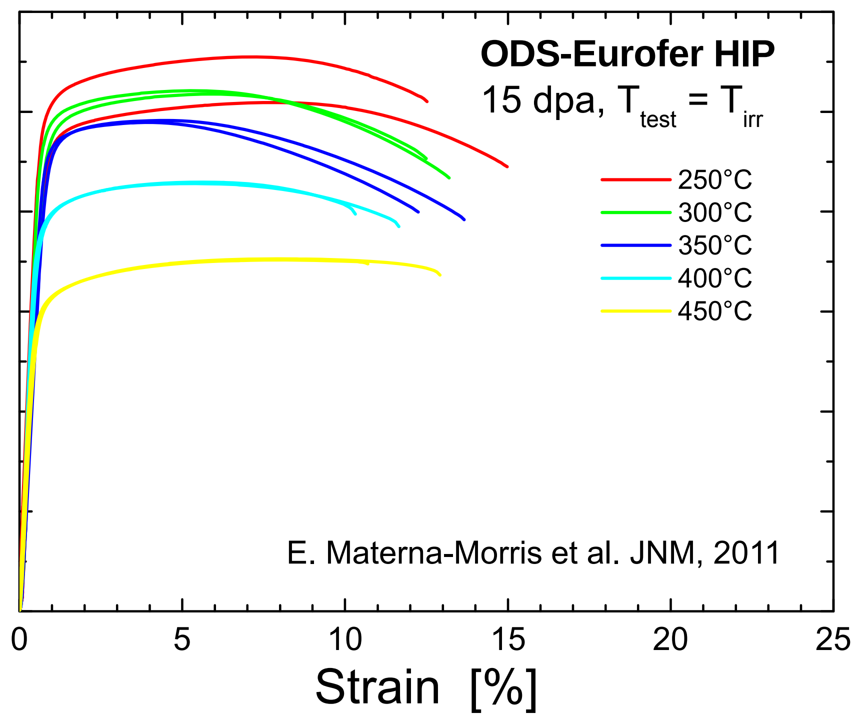
RAFM Steel



RAFM-ODS Steel

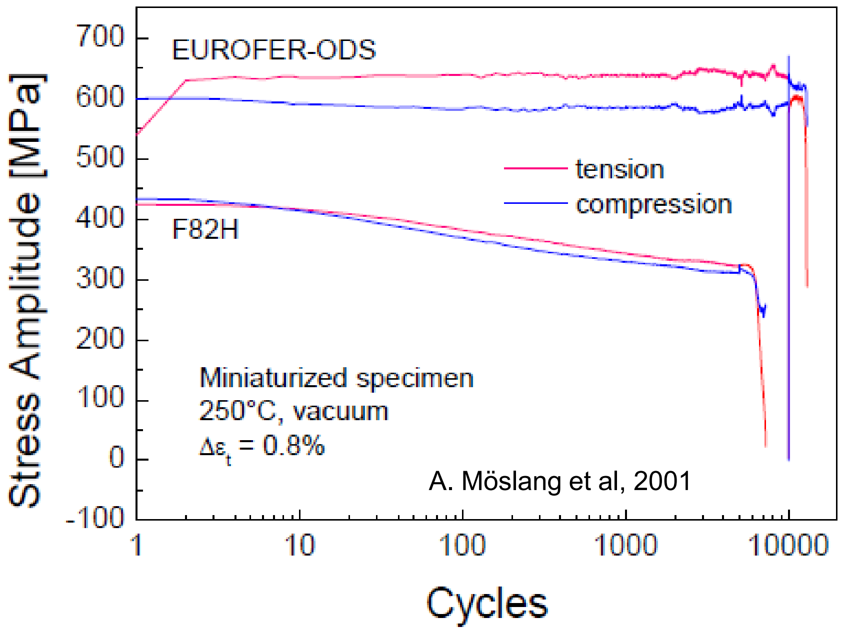


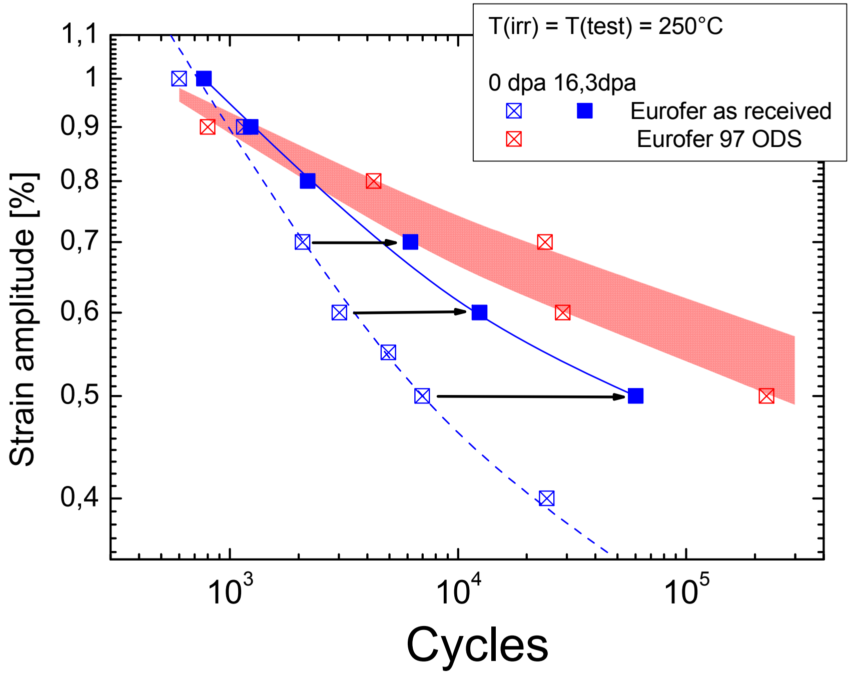
- Substantial irradiation hardening
- Early strain localization due to dislocation channeling → A_u ~0.3%

- Still work hardening → almost no loss of uniform elongation (A_u ~7%)

Outlook Fatigue:

Is it possible to overcome the cyclic softening phenomena of ferritic-martensitic steels?





Nano-scaled ODS particles are extremely stable, that is:

- (almost) no cyclic softening or dislocation channeling
- Suppression of alloy dissolution and aging due to their high sink strength

9Cr ODS steel fuel pins:
Production in Japan, irradiation in FBR BOR60, Russia

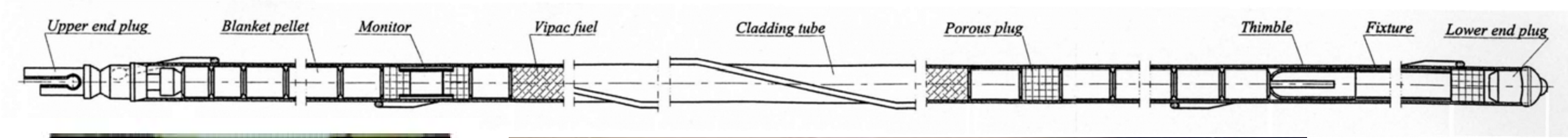
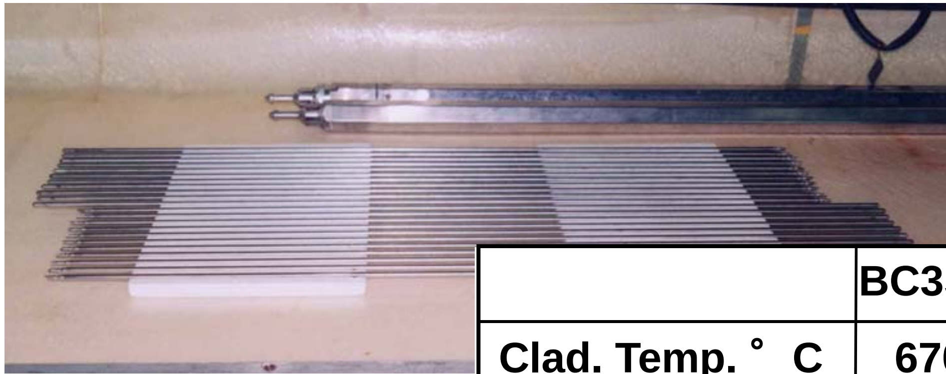


Fig. 12 Appearance of fabricated ODS fuel assemblies (BC358E and BC359E)

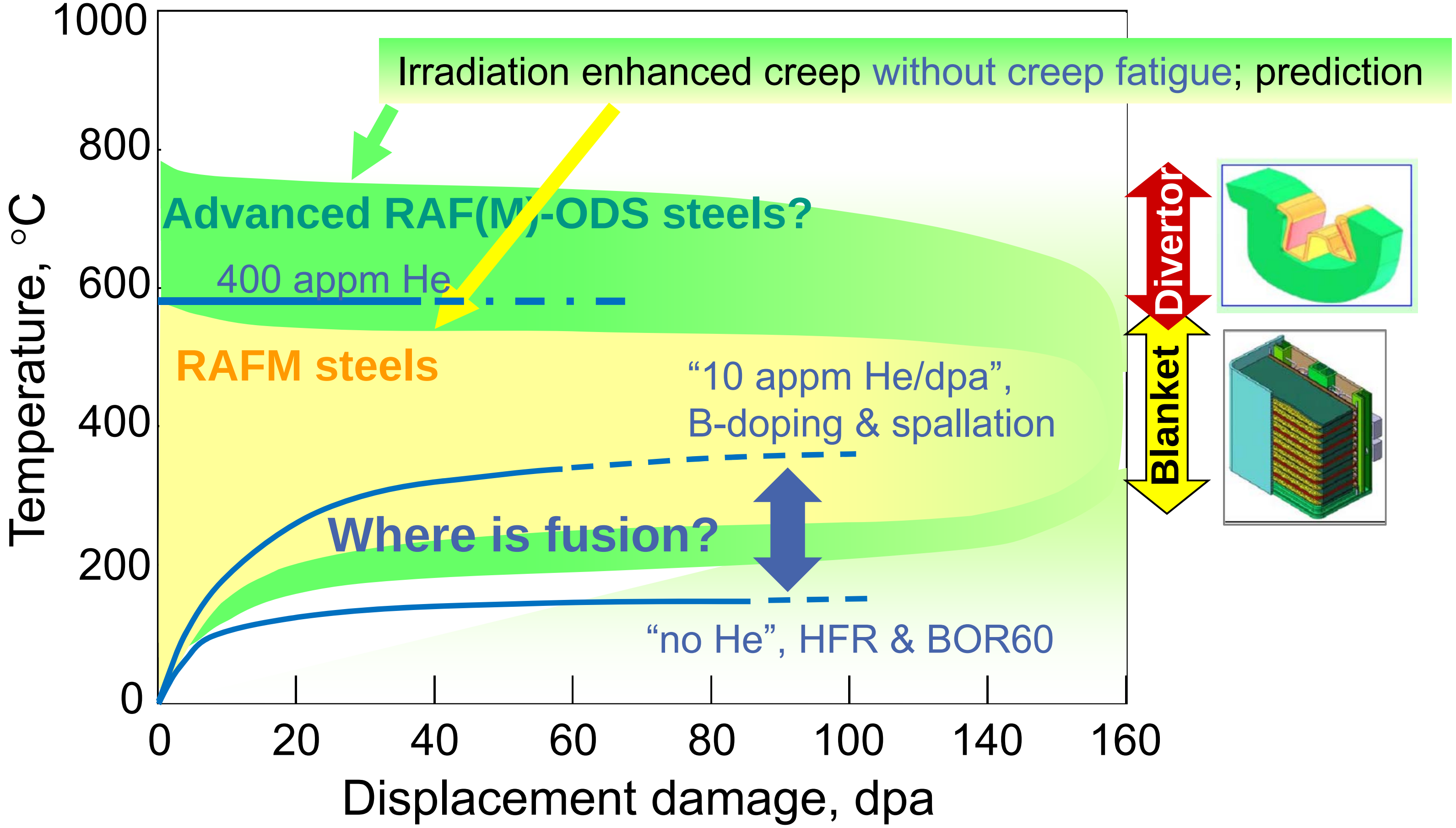


	BC358	BC359
Clad. Temp. ° C	670	720
Burnup at%	15	
Dose dpa	75	

S. Ukai et al., Hokkaido University, Japan

- Materials challenges: fusion – fission – spallation
- Neutron irradiation: Examples of progress and issues
  - Reduced activation ferritic/martensitic steels
  - Oxide dispersion strengthened steels
  - W alloys: I-1, I-3, O-1, O-2: Yesterday
    I-7, O-5, O-6, I-8: Today
- Technical Readiness and database maturity
- Role of materials in fusion roadmaps,
  Need for fusion specific irradiation source

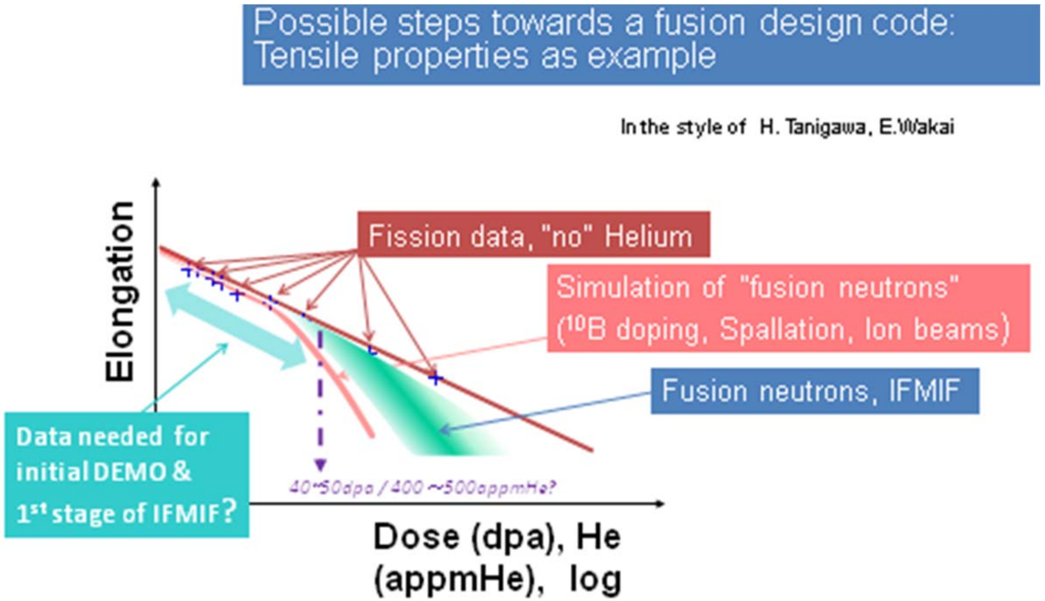
Application window of RAF(M)-ODS-Steels
- schematic -



- ❑ Operation window 350-550 °C: Are RAFM steels limited to ~ 50dpa at the FW?
- ❑ Fusion relevant data well beyond 50 dpa urgently needed → fusion n-source

Design code extension by RAFM steels

In order to get the licensing (and to protect the investment) all these issues must be taken into account in the design codes used during the design phase



Fusion-like irradiation data are needed

- To provide timely a fusion relevant materials database
- To monitor a few properties at high doses (to see the potential of the material for advanced DEMOs and FPP)

Existing database on neutron irradiated steels
- schematic -

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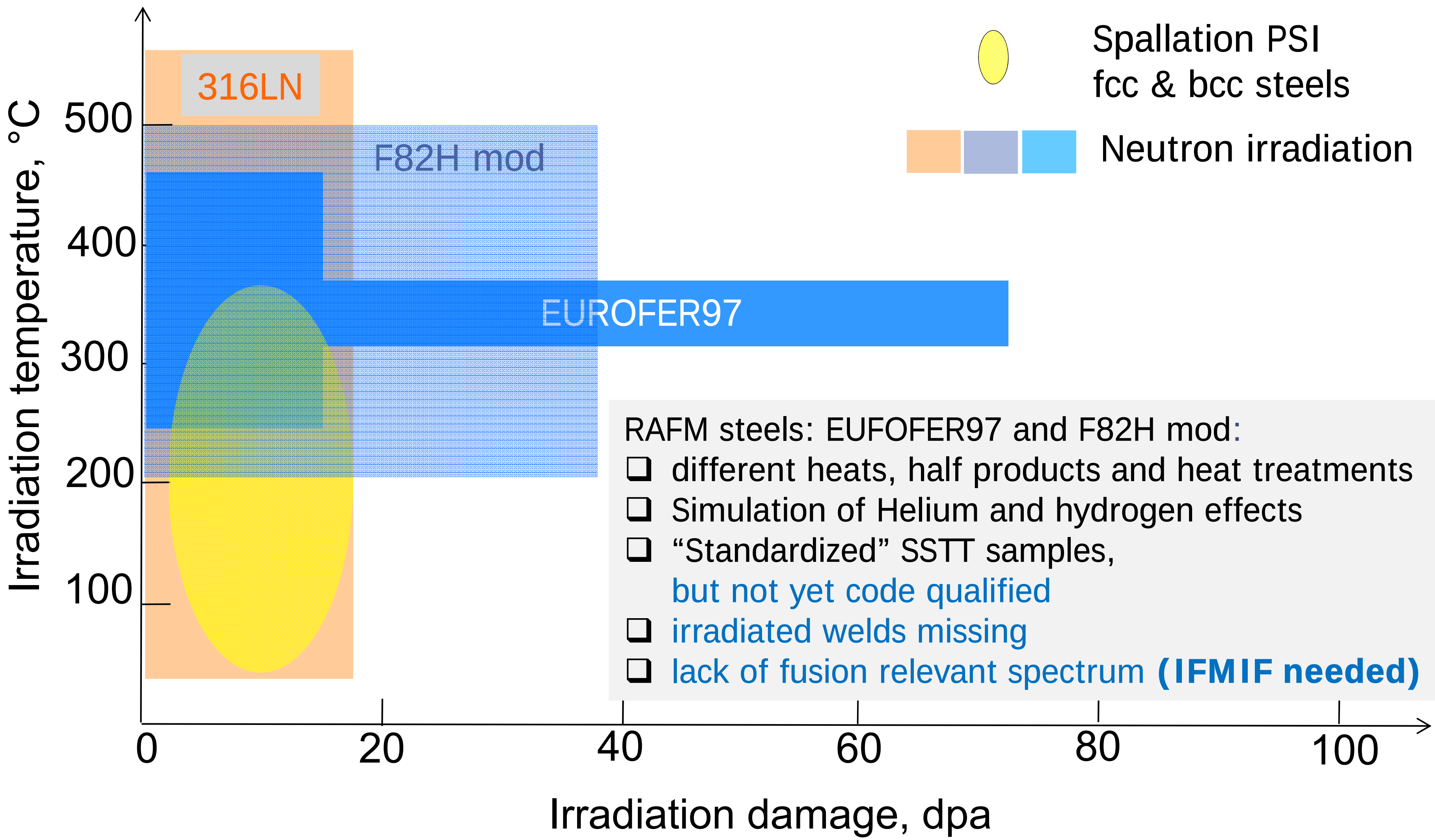
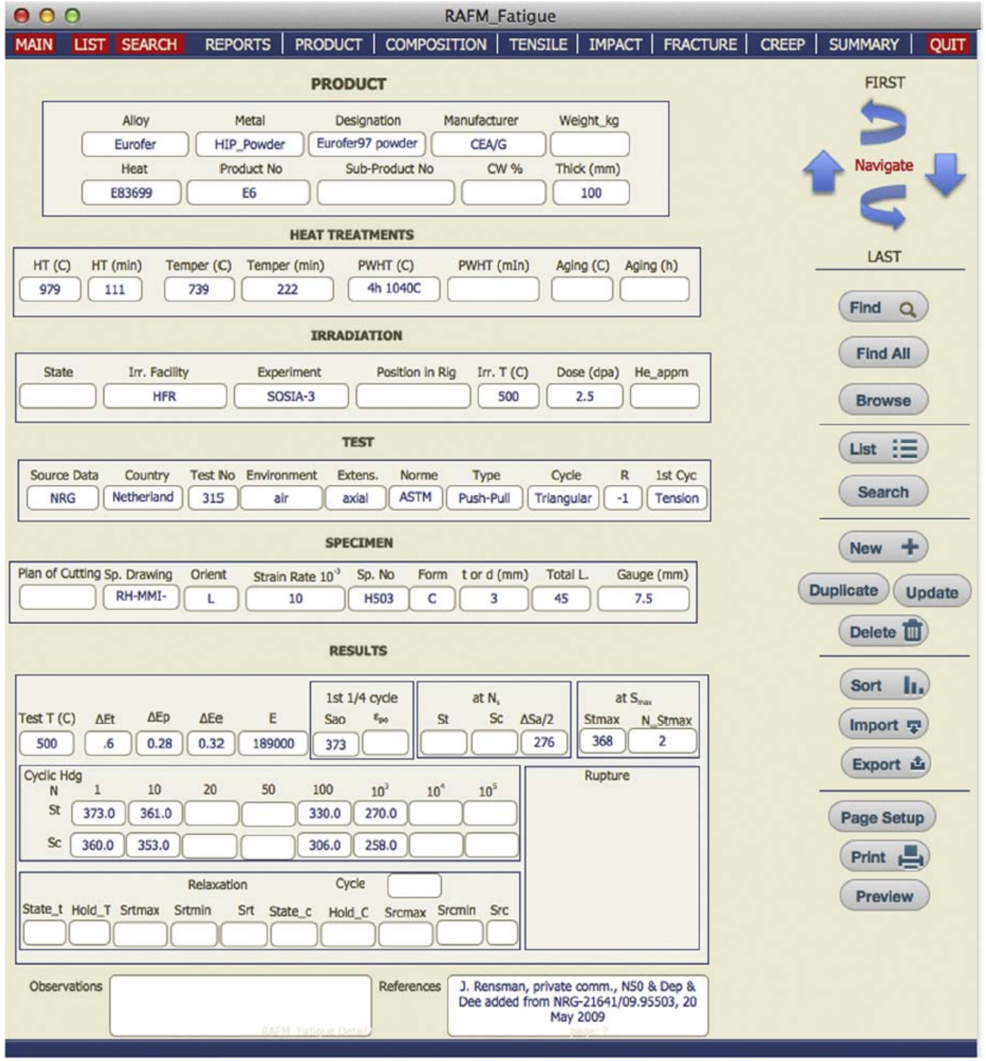


RAFM steels: EUFOFER97 and F82H mod:

- ❑ different heats, half products and heat treatments
- ❑ Simulation of Helium and hydrogen effects
- ❑ "Standardized" SSTT samples, but not yet code qualified
- ❑ irradiated welds missing
- ❑ lack of fusion relevant spectrum **(IFMIF needed)**

Design code qualification, e.g. for RCC-MRx



Collection of broad based materials data

↓

Validation by expert groups

↓

code qualified “Materials Properties Handbook”

↓

Distribution to manufacturers and designers for comments

↓

RCC-MRx Design code implementation of the new material class

Irradiation effects: Materials Database Maturity

Complimentary: R. Kurtz, PNNL

	1st DEMO Blanket						2nd DEMO Blanket						Adv. DEMO						
Data base need	<20 dpa/200appm He						~50 dpa/500appm He						>100 dpa/1000appm He						
Materials	RAFM	FM-ODS	W	SiC	Be	Li ceramic	RAFM	FM-ODS	W	SiC	Be	Li ceramic	RAFM	FM-ODS	W	SiC	Be	Li ceramic	RAFM
Irradiation effects																			
Hardening/Embrittlement	Green	Yellow	Red	Yellow	Yellow	Red	Yellow	Red	Red	Yellow	Red	Red	Red	Red	Red	Red	Red	Red	Red
Phase stabilities	Green	Yellow	Red	Yellow	Yellow	Yellow	Yellow	Red	Red	Yellow	Red	Red	Red	Red	Red	Red	Red	Red	Red
creep & fatigue	Green	Yellow	Red	Yellow	Red	Red	Yellow	Red	Red	Red	Red	Red	Red	Red	Red	Red	Red	Red	Red
Volumetric swelling	Green	Yellow	Red	Yellow	Green	Yellow	Yellow	Red	Red	Yellow	Yellow	Red	Yellow	Red	Red	Red	Red	Red	Red
High Temp He&H effects	Yellow	Red	Red	Red	Yellow	Yellow	Yellow	Red	Red	Red	Red	Red	Red	Red	Red	Red	Red	Red	Red

- Green: Adequate knowledge base exists
- Yellow: Partial knowledge base exists
- Red: No knowledge base

Note: He levels are only for FM steels

Suggested TRLs for in-vessel materials

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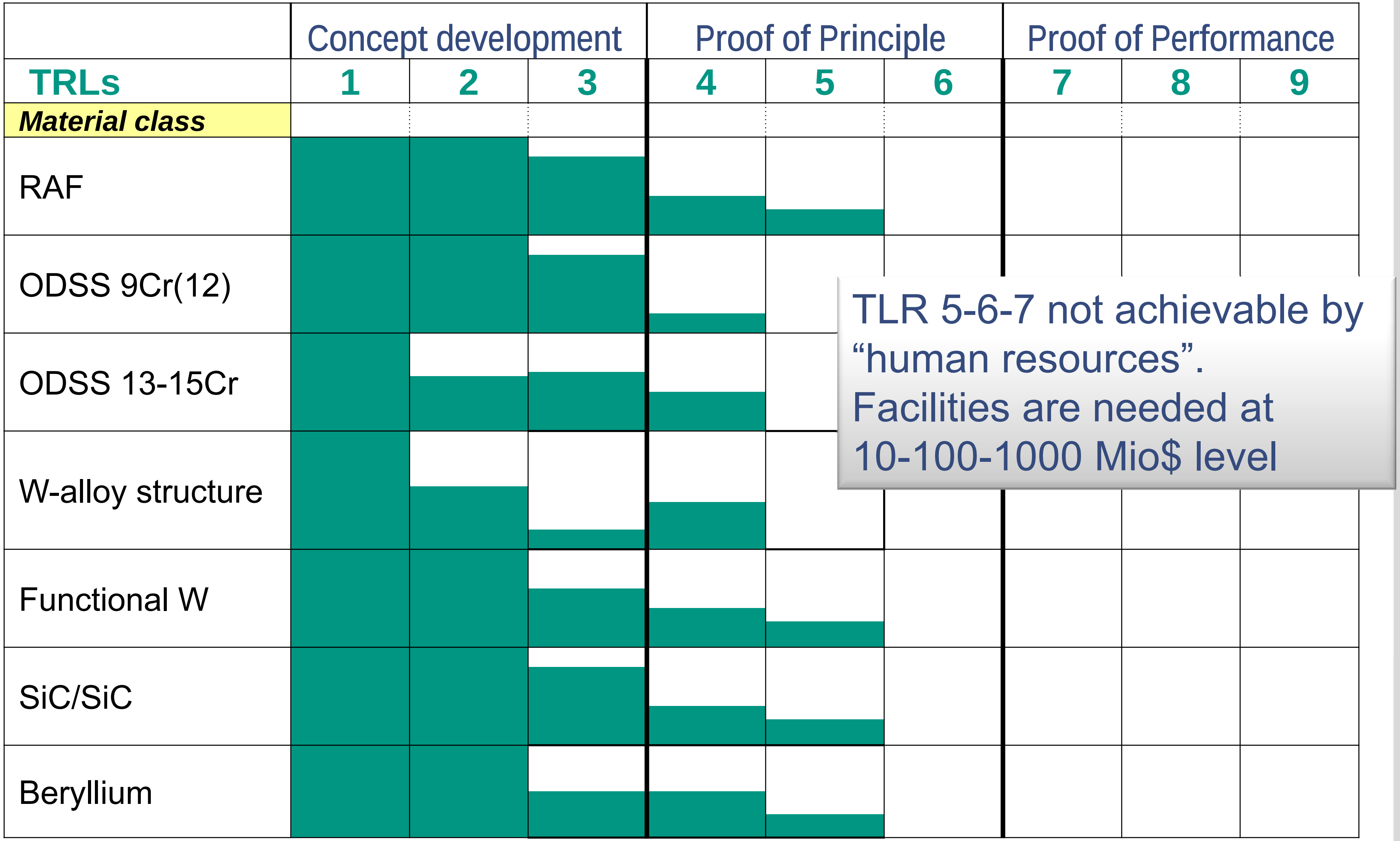
According to M.S. Tillak et al, ICFRM-15, 2012



Role of Materials in Fusion Road Maps - simplified -

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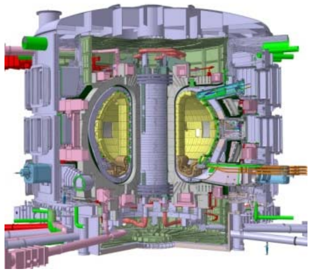
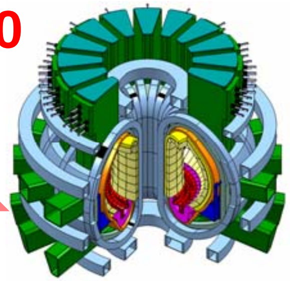
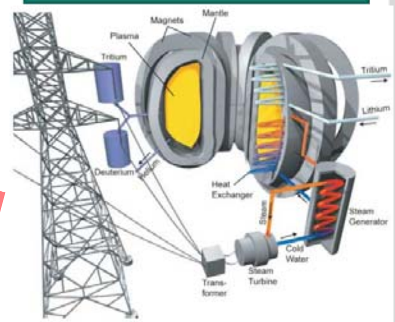
ITER,DT-phase beyond 2027	Early DEMOs	FPP beyond 2060
	EDA end ♦ ~2030 ♦ Start operat.	







Materials Database

D-Li type fusion n-source
- ♦ 20 dpa 1st blanket
- ♦ 50 dpa licensing?
- ♦ 100 dpa, small volume

preferable option

"Simulation": Fission reactors, ion implantation, spallation

ongoing

Plasma based n-sources (e.g. FNSF):
≤10 dpa/fpy; 0.1-0.3 duty cycle, start~2030
→ 40 dpa (≥ 15 yrs) not before 2045

For DEMOs too late

Pillars of the EU Fusion Materials Programme

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FP7				Horizon 2020						
10	11	12	13	14	15	16	17	18	19	20

Materials Technology for ITER TBMs and DEMO
F4E coordinated
EUROFER, Joining technology, Be, Li ceramics,….

Fusion Materials for DEMO; Advanced Materials
EUROFUSION coordinated
Improved RAFM steels, ODS alloys, W components

Broader Approach, Materials
funded by JA and EU countries
Welds, BeTi, Li ceramics

Bottleneck:
DEMO specific irradiation and PIE programmes

International Road Map

High-temperature materials are at a critical path

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