Karlsruhe Institute of Technology

Tutorial

ICFRM-18
Aomori 2017

Overview of current reactor design and Fusion Structural Materials

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INSTITUTE FOR APPLIED MATERIALS (IAM)

https://www.nasa.gov

Outline

- DEMO reactors: current designs – blankets - divertors
- Reduced Activation Structural Materials (DEMO-oriented): recent progress
  - Reduced activation ferritic/martensitic steels
  - Oxide dispersion strengthened steels
  - Neutron irradiated steels – selected results
  - W alloys
- Database maturity & role of materials in fusion roadmaps

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- Reduced Activation Structural Materials (DEMO-oriented): recent progress
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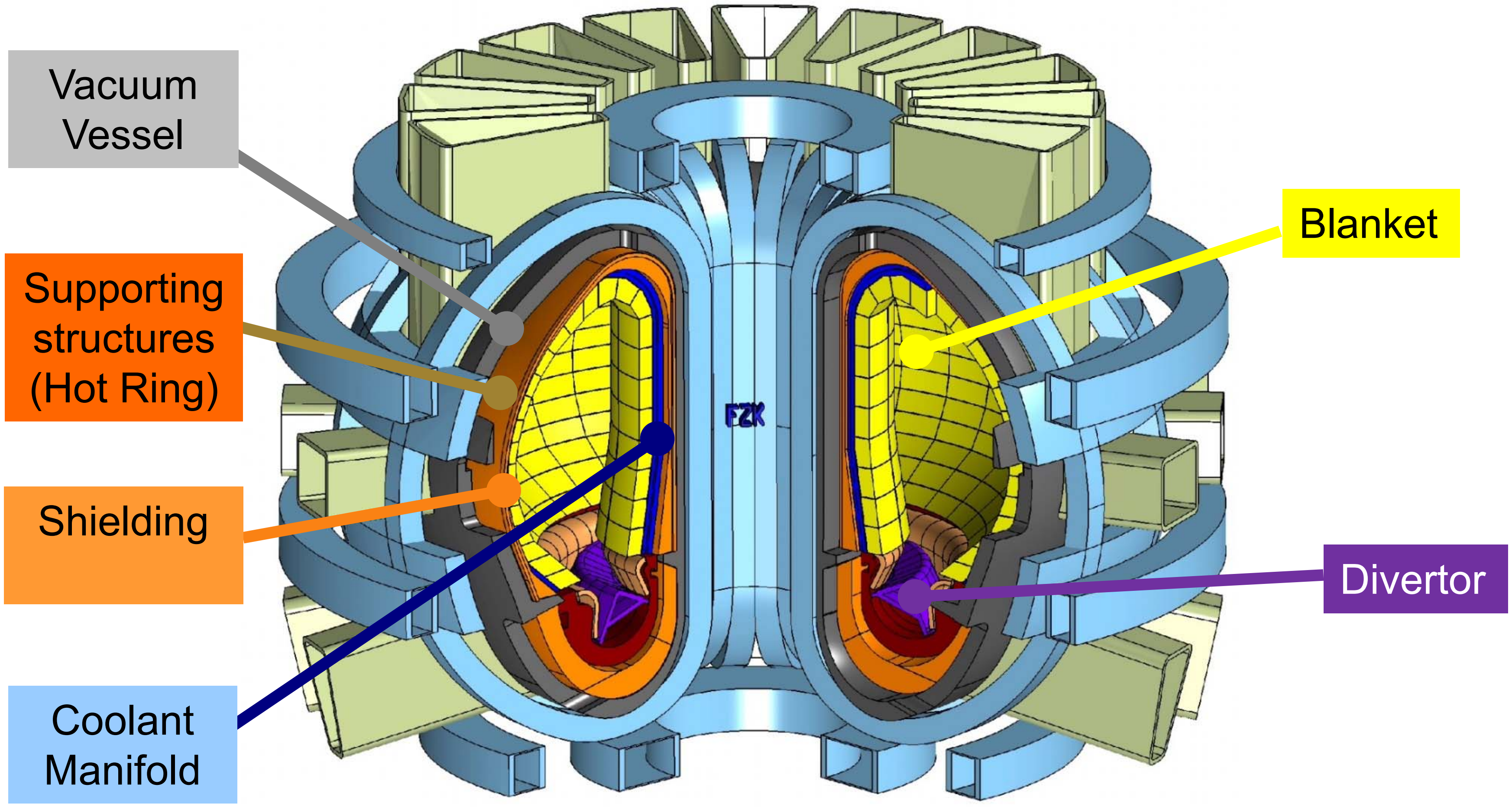
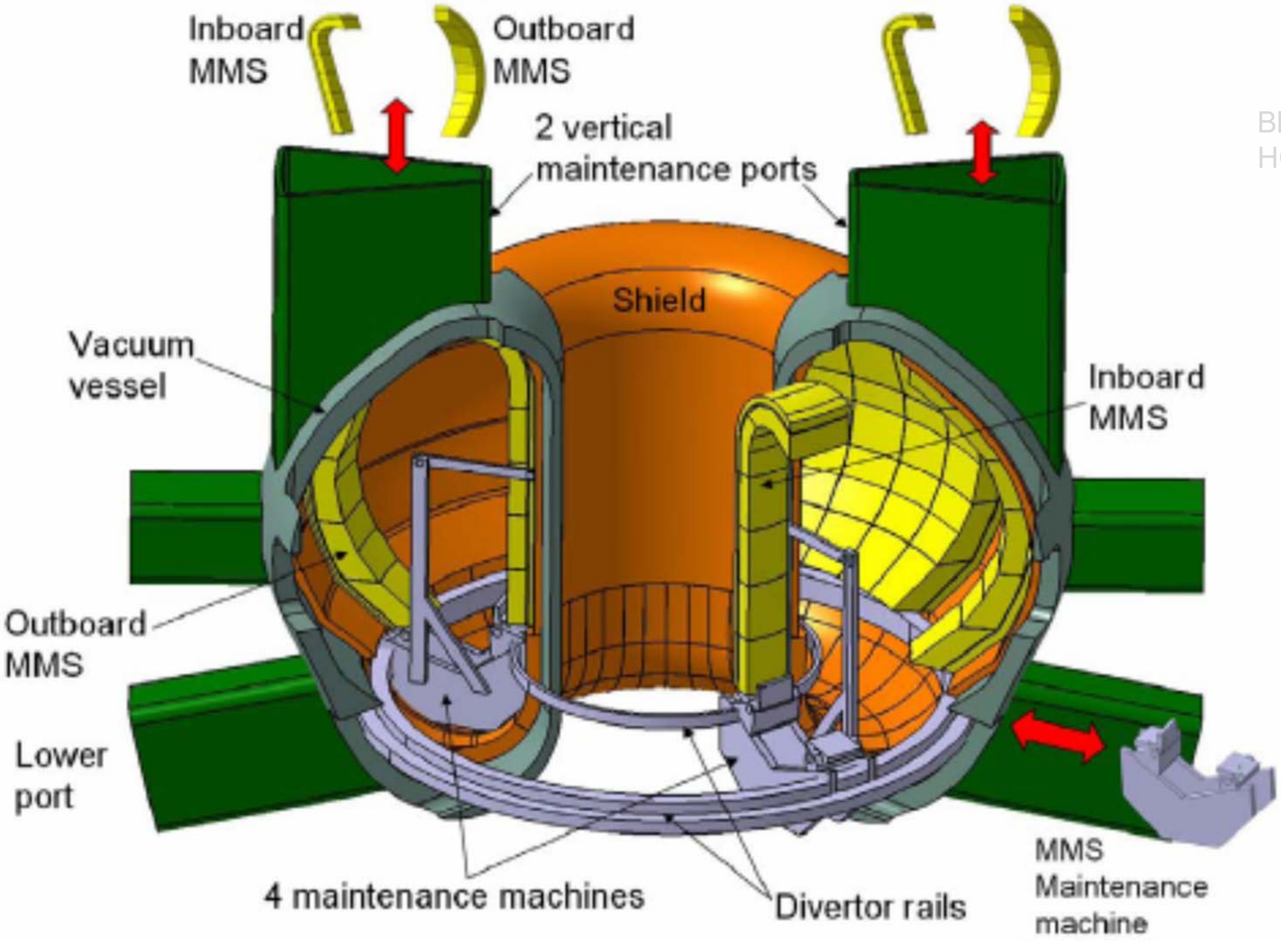
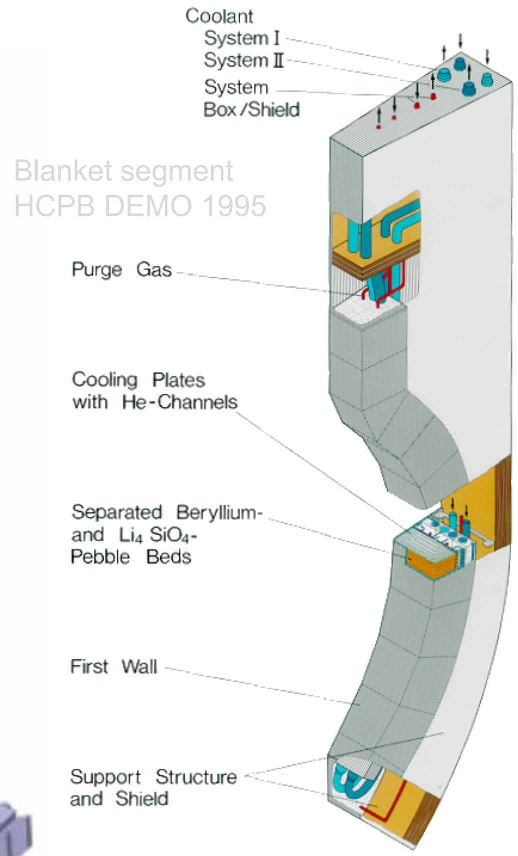
Fusion Reactor Thermonuclear Core



Design of a DEMO (Demonstration Fusion Reactor Plant)

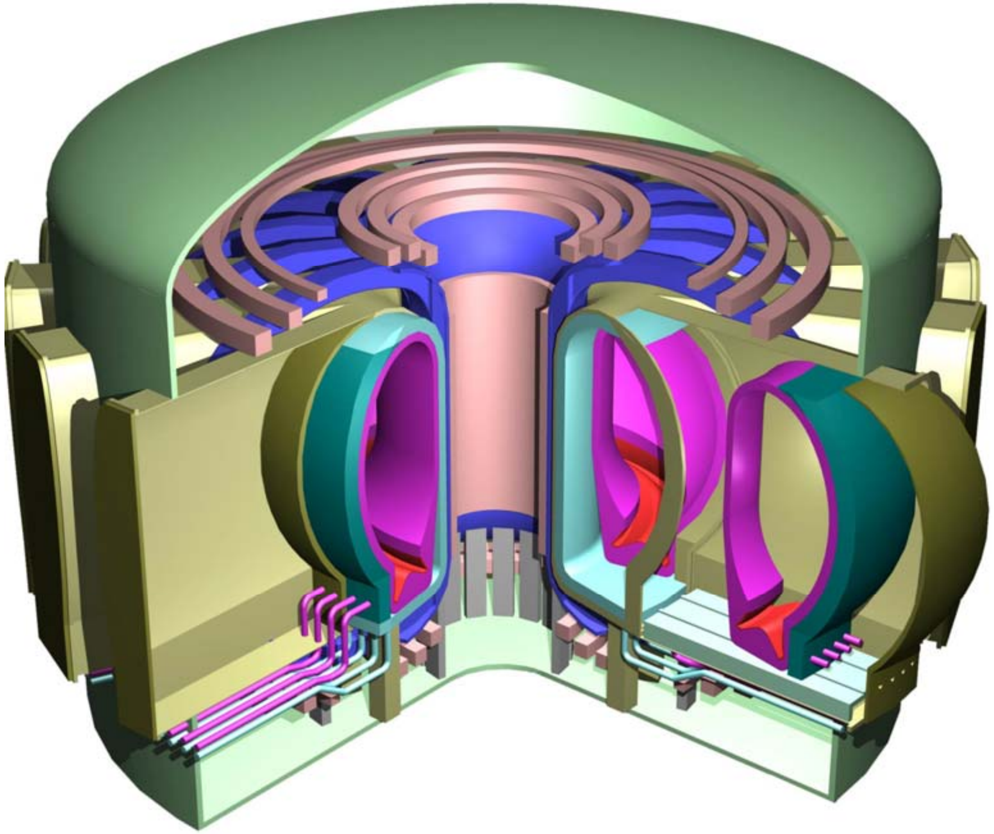
DEMO Design study:
Maintenance System - Vertical port



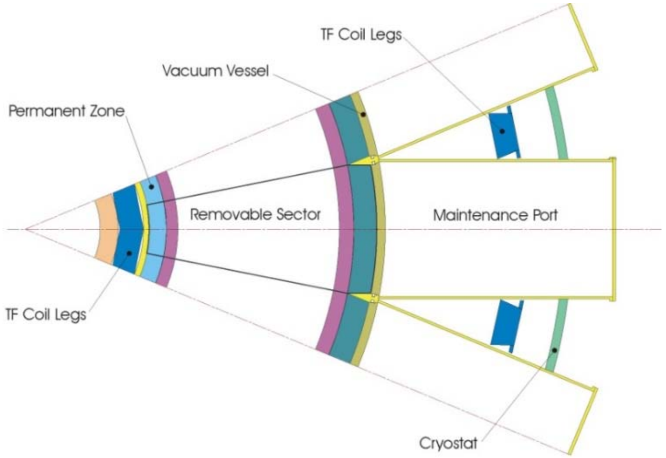


5 segments per sector
x 16 sectors
=> 80 segments

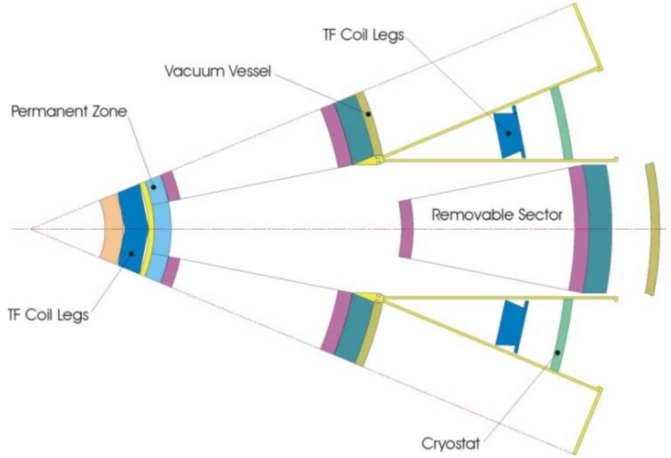
Design study:
Maintenance System - Large sector concept



Aries Studies, US



=> 16 sectors



DEMO design studies: Blanket architecture

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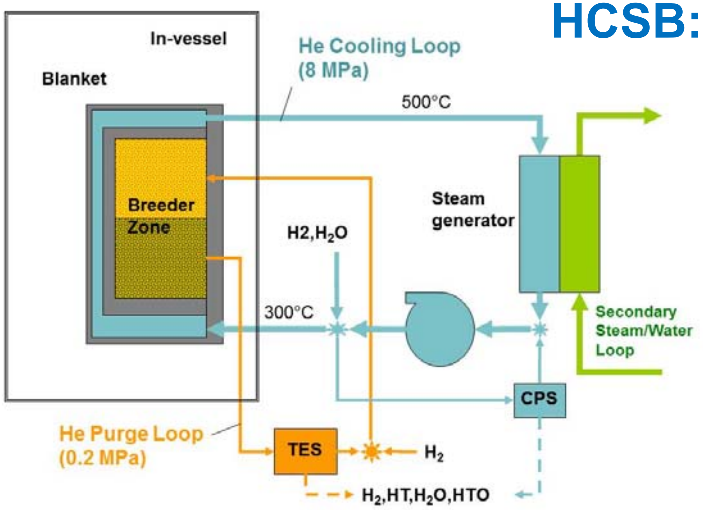
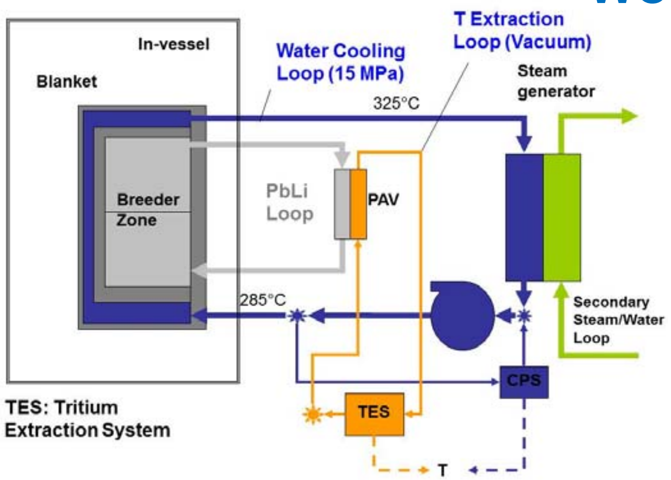
HCSB:



WCLL



- Very similar architecture, despite differences (coolant, breeder, etc.).
- Strict separation between coolant and Tritium breeder zones:
  - HCSB: Helium Cooled Solid Breeder
  - WCSB: Water Cooled Solid Breeder
  - HCLL: Helium Cooled Liquid Lead
  - WCLL: Water Cooled Liquid Lead
- The high pressure coolant (water or He) cools directly the steel structure flowing mainly in small channels
- A T carrier (a purge gas for the solid or the breeder PbLi itself in liquid breeder concepts) fills the breeder zone and flows in independent loops at low pressure transporting T outside the reactor.
- Also if PbLi is used as carrier, its recirculation rate (10-20 inventories pro day in WCLL and HCLL, respectively) is so slow that no significant heat is removed in these loops; the same is for the He purge in the HCPB.

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DEMO design studies: Suggested coolants

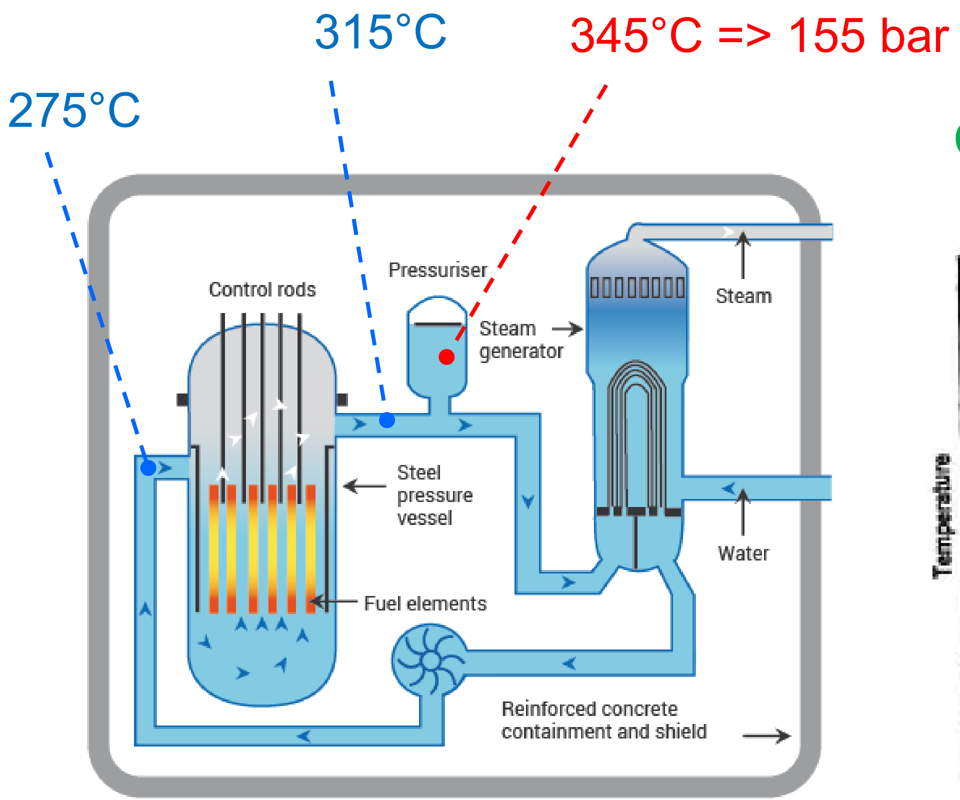
Water:
- ▲ Exceptional cooling capability. High density that allow small flow section.
- ▲ Low ΔT in Blanket. PWR range (275-315°C @15.5 MPa).
- ▼ Issues: T contamination, low temp. irradiation embrittlement, corrosion,…

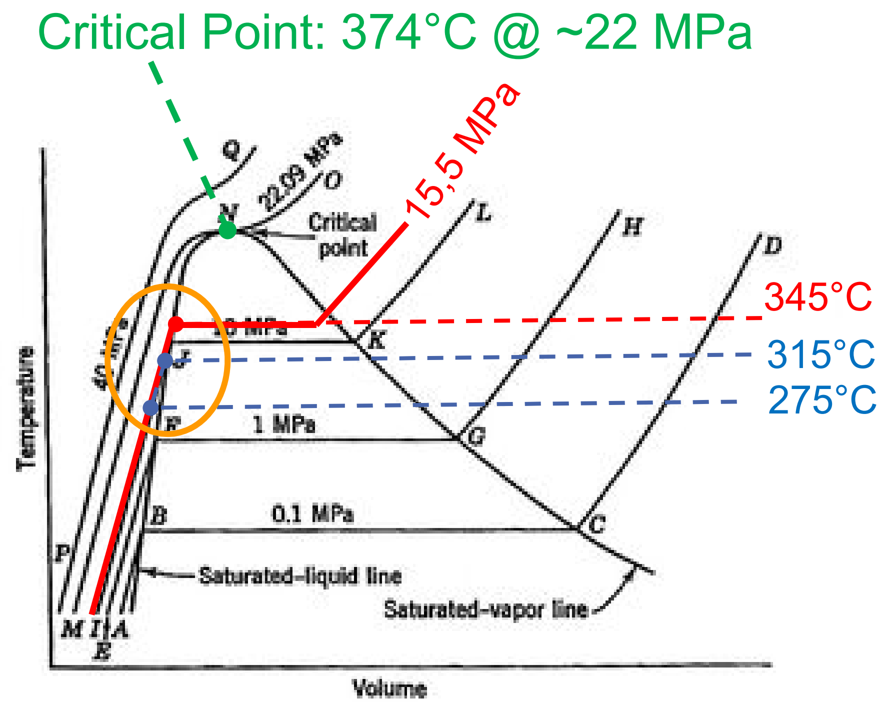
Helium:
- ▲ Exceptional compatibility with all materials
- ▲ Possibility to cope with all temperature windows
- ▼ Lower heat removal capability and higher pumping power.
  → Large tubes with low shielding features

Liquid Metal (PbLi and Li):
- ▲ Accomplish the double functions of heat removal and T transport
- ▲ High heat removal capability, low pressure
- ▼ MHD limitations (low velocity), corrosion.

Molten salt (FLiBe):
- ▲ Accomplish the double functions of heat removal and T transport
- ▲ Low pressure, no MHD limitations.
- ▼ High corrosion issues, Low thermal conductivity, difficult chemistry.

Water usage in fusion reactor: PWR cycle





$P = Q_m \cdot C_p \cdot \Delta T$

P : power [W];

C_p: specific heat [J · kg^{-1} · K^{-1}]; ·

ΔT: temperature difference [K];

Q_m: massflow [kg/s];

For P=1GW with $C_p \approx 5.65$ kJ · kg^{-1} · K^{-1} => $Q_m = 4424 \frac{kg}{s}$

and with ρ = *714 kg/m*^{3} $Q_v = 6.19 \frac{m^3}{s}$

(water properties at 300°C)

Helium usage in fusion reactors:

Within the range of practical interest (power plant), the He behavior can be described with the "perfect gas law":

$$p \cdot V = n \cdot R \cdot T$$

p: pressure [Pa]
V: volume [m^3]
T: temperature [K]
R: universal gas constant (8.314 J · K^{-1} · mol^{-1})
n: mole number [mol]

Dividing with the mass m:

$$p \cdot \frac{V}{m} = \frac{n}{m} \cdot R \cdot T \quad \Rightarrow \quad p \cdot \rho^{-1} = \mathcal{R}_{He} \cdot T$$

ρ: density [kg/m^3]
$\mathcal{R}_{He}$: He individual gas constant
(2077 J · kg^{-1} · K^{-1}) as n/m = ~250 mol/kg

Helium density: $\rho = \frac{p}{\mathcal{R}_{He} \cdot T}$

@ 8 MPa and
T=300°C (573.15 K) => ~ 6.7 kg/m^3
T=500°C (773.15 K) => ~ 4.9 kg/m^3

Specific heat: c_p = 5.19 kJ · kg^{-1} · K^{-1}

To remove 1 GW with He at 8 MPa with T=300-500°C, the mass flow is

$Q = \frac{P}{c_p \cdot \Delta T}$ = ~963 kg/s

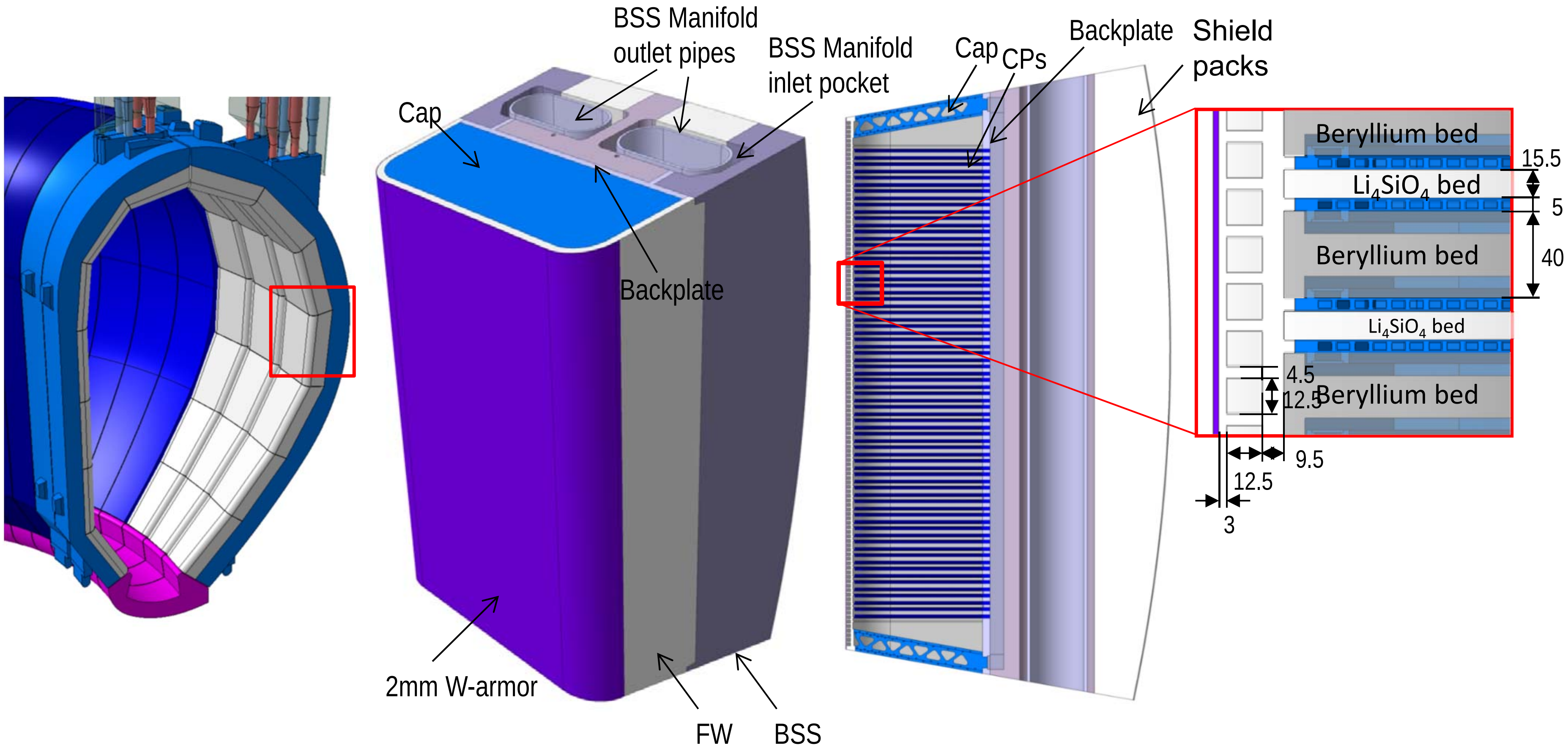
@500°C => 196 m^3/s

Liquid Breeders: Properties

	Metals		Salts	
	Li	**LiPb**	**FLiBe**	**FLiNaBe**
Reference composition	pure Li (metal)	Eutectic alloy ~15.8 at% Li	0.66-0.34 (LiF)-(BeF_2)	0.31-0.31-0.38 LiF-BeF_2-NaF
Melting Point MP [°C]	180	235	~731	~600
Density [kg·m^{-3}] at MP	533	9597	2056	2166
Electrical resistivity [Ωm] at MP	$2.5*10^{-7}$	$1.2*10^{-6}$	~$1.2*10^{-2}$	-
Thermal conductivity [W·m^{-1}·K^{-1}] at MP	42	12	~1	~0.70
Specific heat [J·kg^{-1}· K^{-1}] at MP	4394	190	~2380	~2200

DEMO design: HCPB architecture

Detail of a Helium Cooled Pebble Bed (HCPB) Blanket module



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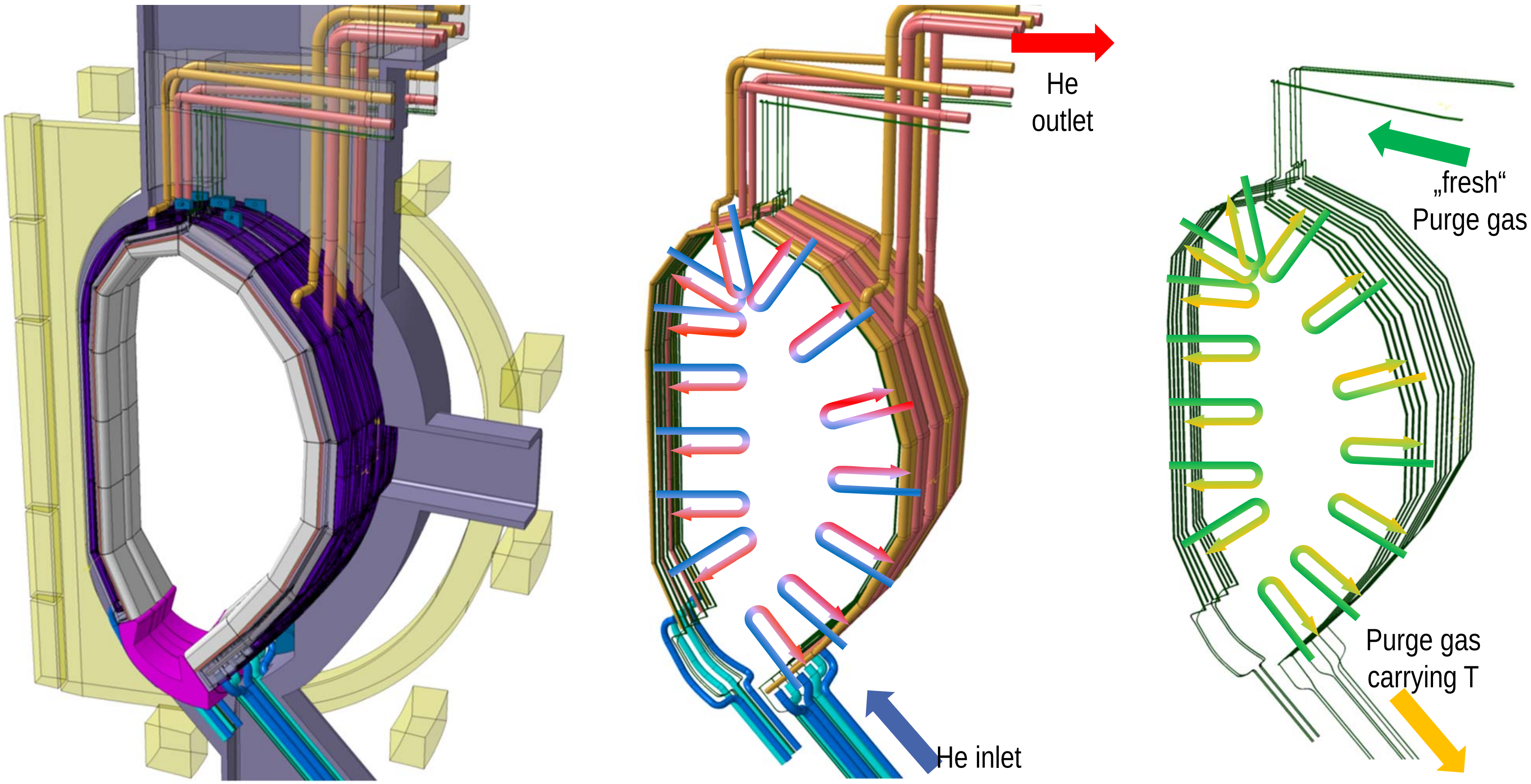
DEMO design: HCPB architecture

Coolant and purge gas feed pipes

DEMO with HCPB blanket sector

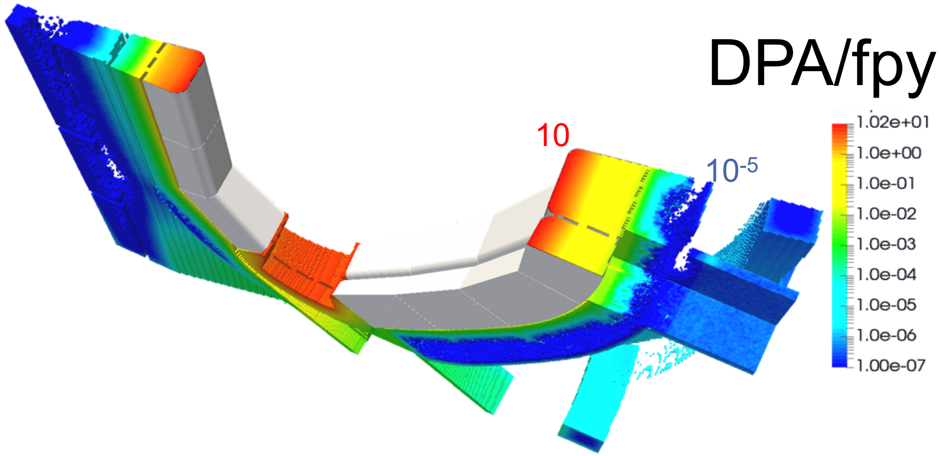
Coolant & purge gas feed pipes

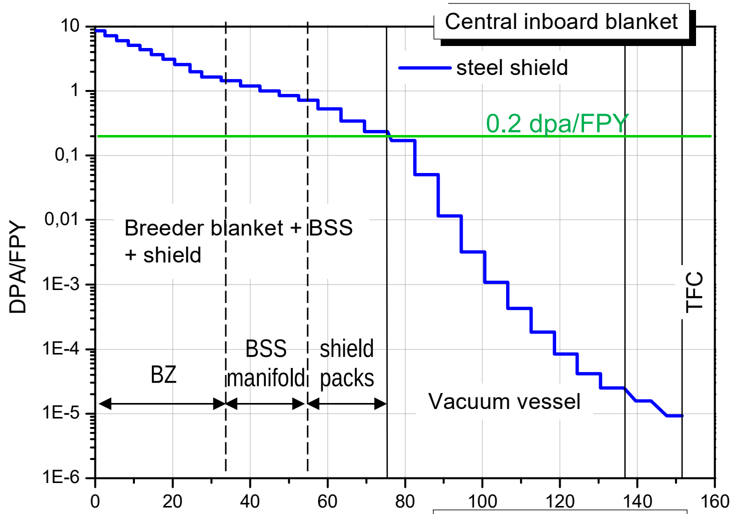
Purge gas feed pipes

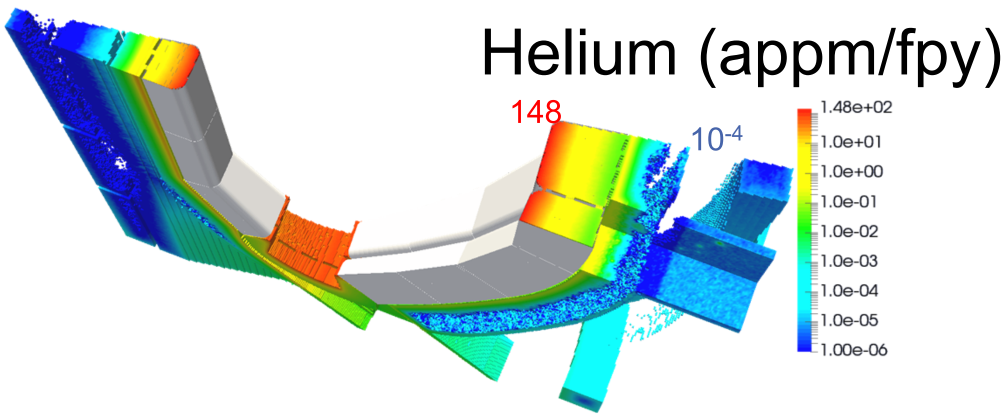


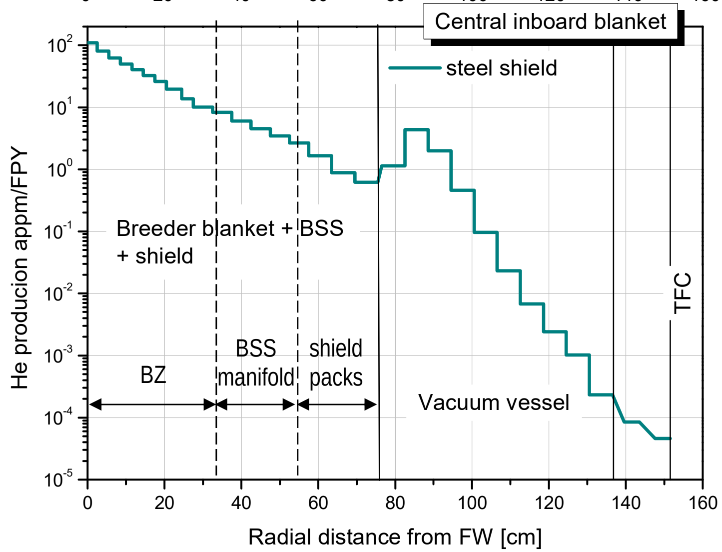
DEMO design: HCPB neutronics performance

Neutronics analyses with MCNP5-1.60



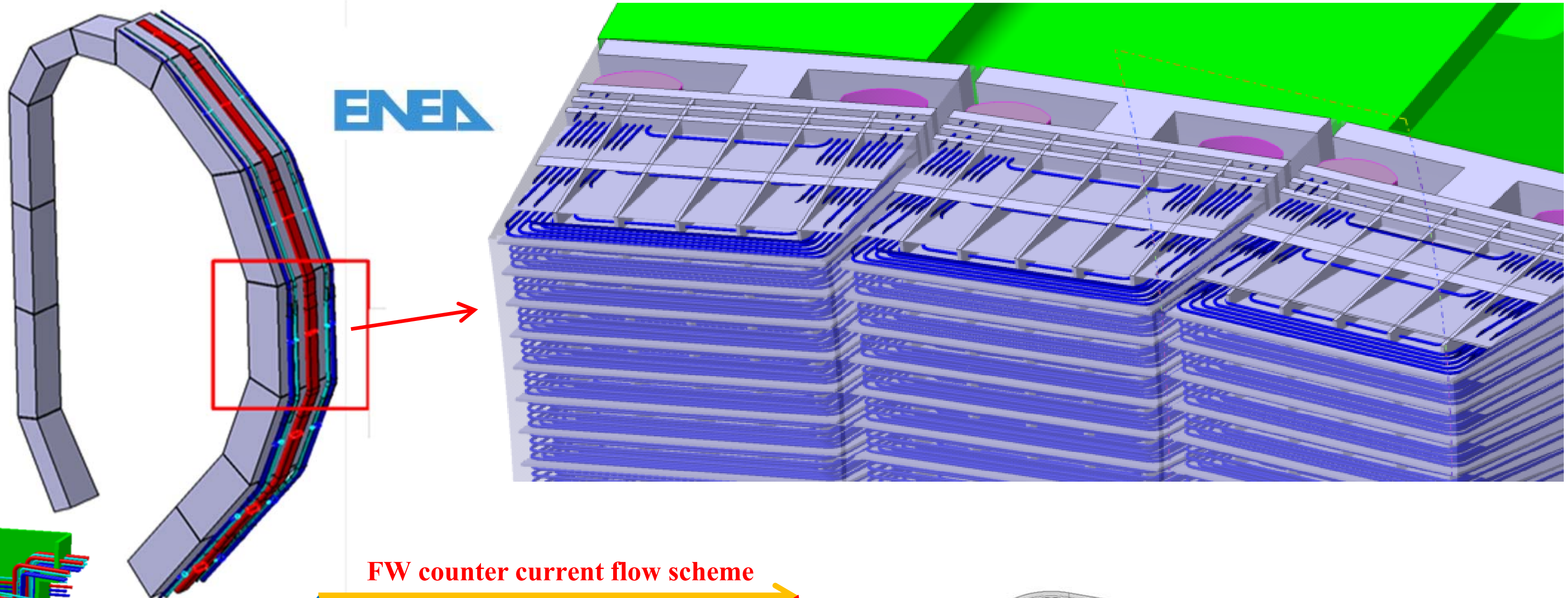


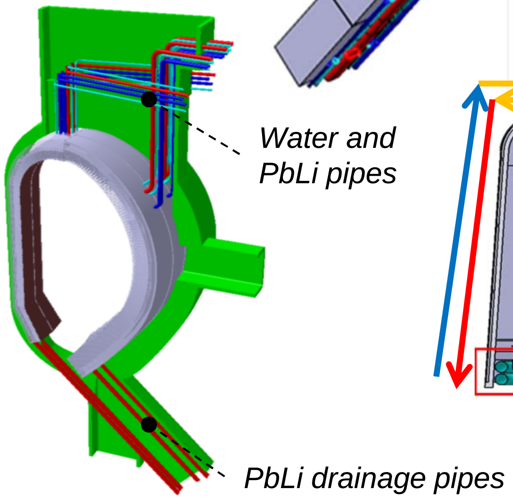


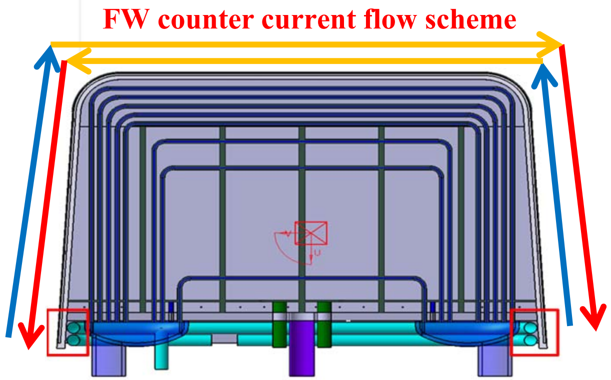


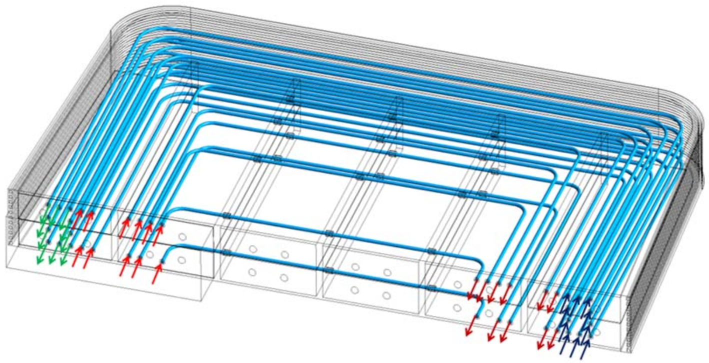
DEMO design: WCLL architecture

Detail of a Water Cooled Liquid Lead (HCLL) Blanket module







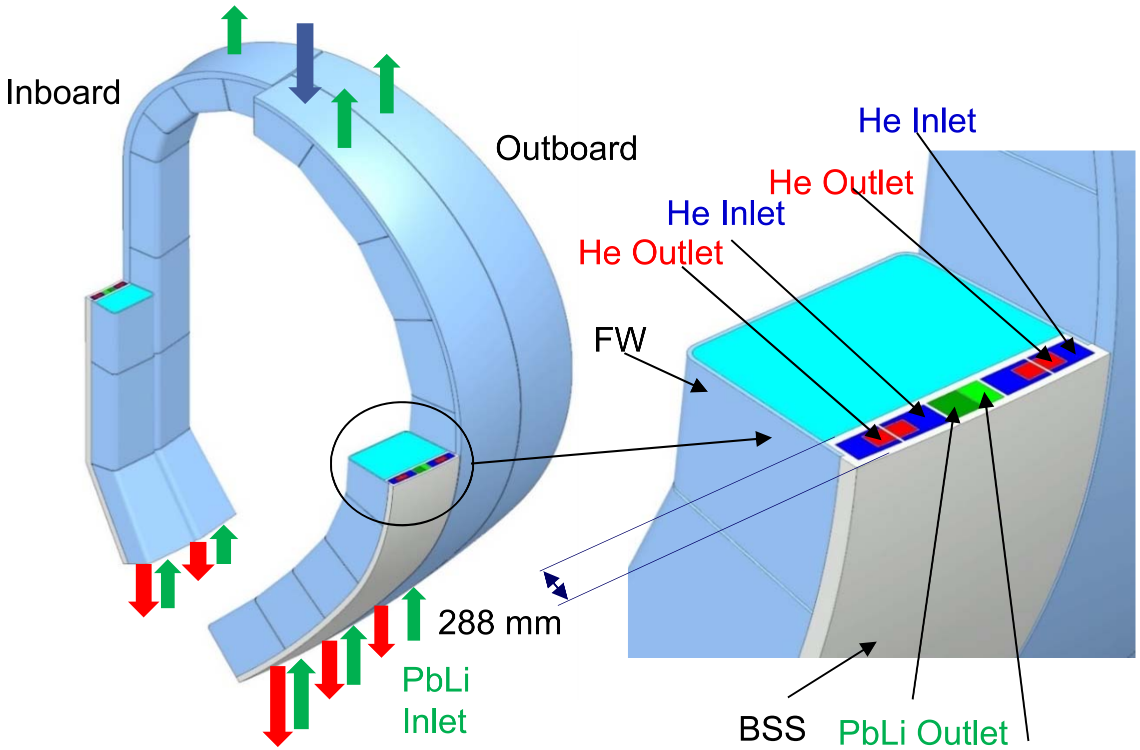


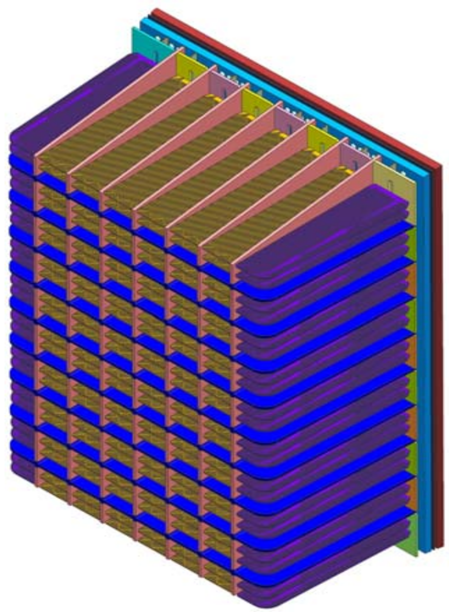
DEMO design: HCLL architecture

Detail of a Helium Cooled Liquid Lead (HCLL) Blanket module

Integration of the HCLL concept in the DEMO reactor. Organisation of the piping

Stiffening grids inside the box to cope with in-box LOCA event at 80 MPa

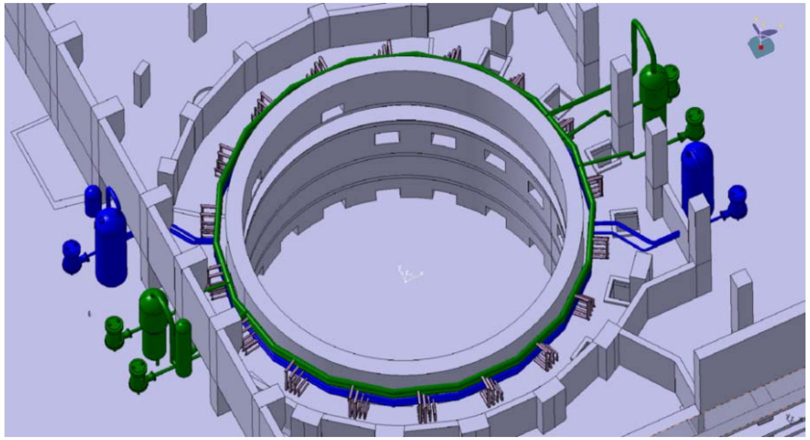


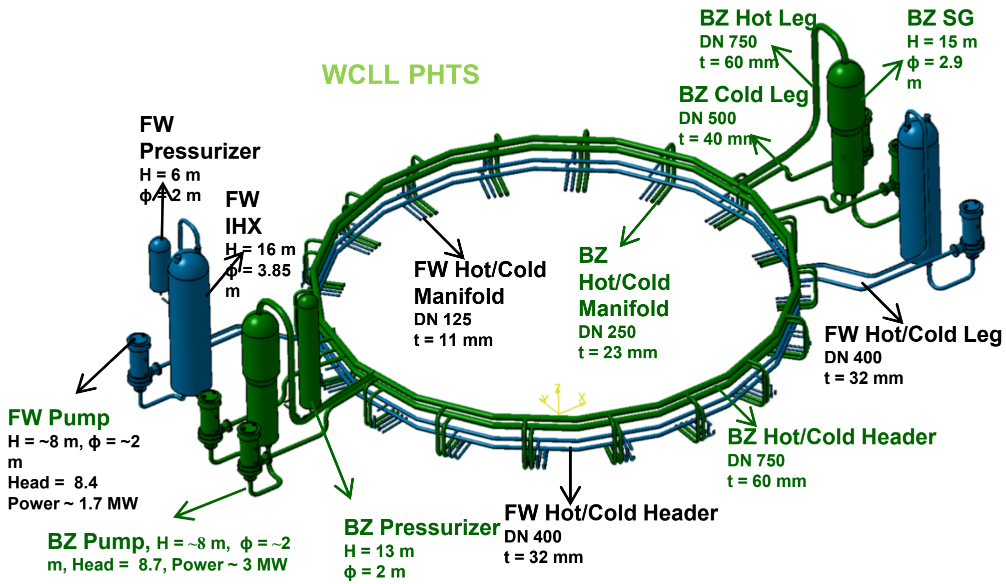


WCLL blanket: Primary heat transfer system (PHTS)

Overall WCLL PHTS Data	Breeding Zone	First Wall
Thermal Power [MW]	1577	467.6
Pumping Power [MW]	7	2
Total Volume [m^3]	308	137
Mass flow rate [kg/s]	8146	2415
Piping velocity range [m/s]	7 ÷ 19	7 ÷ 19
Number of loops [-]	**2**	**2**
Overall H2O Inventory [m3]	**445**	
Overall piping length [Km]	**~1.75** (whereof ~0.55 in rings)	
Total Pumping Power [MW]	**12.3**	
Main components per loop		
Hot/Cold Manifolds	9/9	9/9
Hot/Cold Legs	1/2	1/1
Hot/Cold Ring Header	1/1	1/1
Pumps	2	1
Heat Exchanger (Steam Generator)	1	1

WCLL PHTS Additional Data	
FW (H_2O/MS) HX Power	239 MW
FW HX Heat Transfer Area/volume	6704 m^2/21.8 m^3
BZ (H_2O/H_2O) SG Power	788.5 MW
BZ SG Heat Transfer Area/Volume	4369 m^2/15.4 m^3

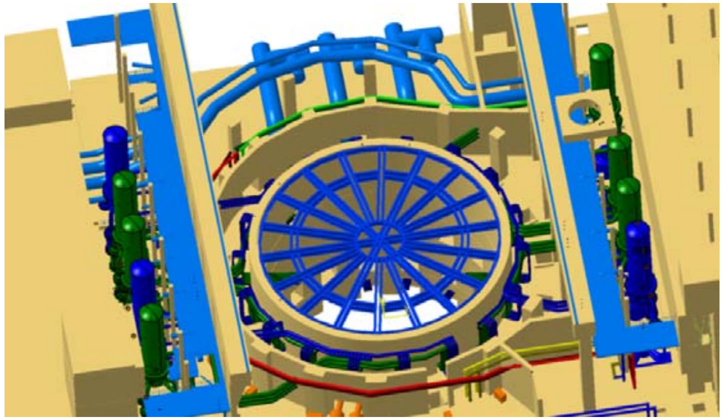


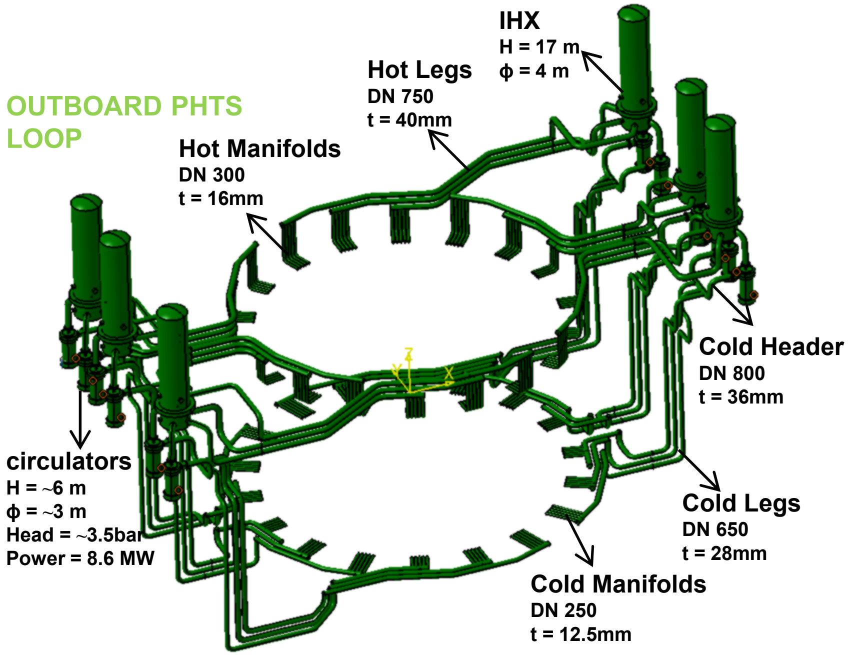


HCPB Blanket: Primary heat transfer system (PHTS)

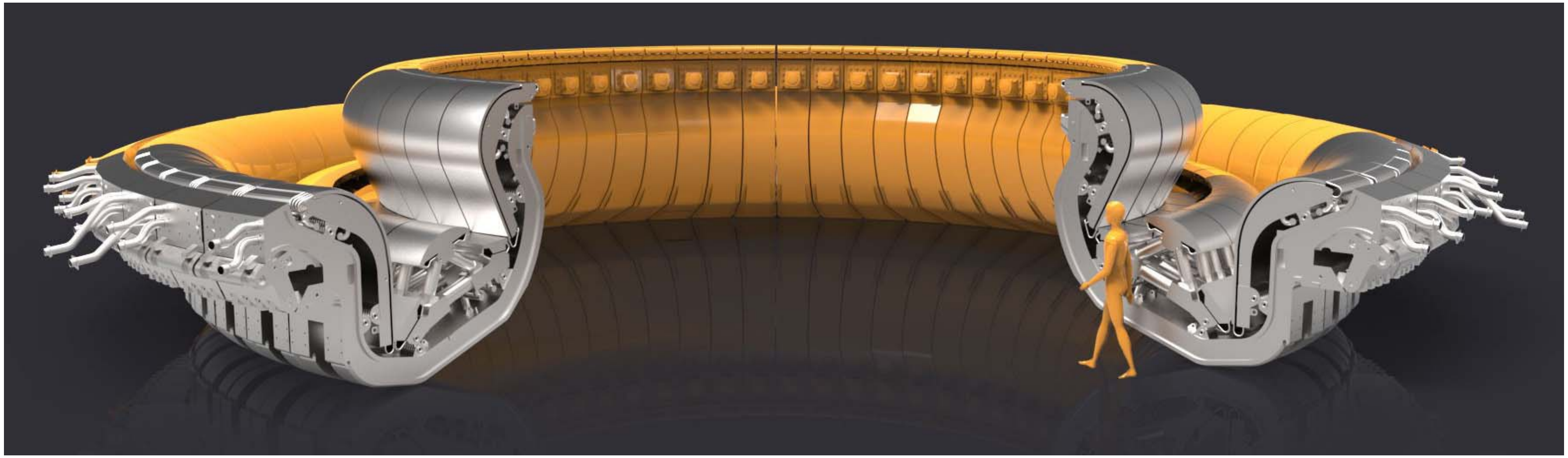
Overall HCPB PHTS Data	Outboard	Inboard
Thermal Power [MW]	1706	643
Pumping Power [MW]	103	46
Total Volume [m3]	1733	760
Mass flow rate [kg/s]	1642	657
Piping velocity range [m/s]	35 ÷ 45	35 ÷ 45
Number of loops [-]	**6**	**3**
Overall He Inventory [m3]	**2493**	
Overall piping length [Km]	**~9**	
Total Pumping Power [MW]	**~150**	
Main components per loop		
Hot/Cold Manifolds	18/18	24/24
Hot/Cold Legs	3/3	3/3
Cold Header	1	1
Circulators	2	2
Heat Exchanger	1	1

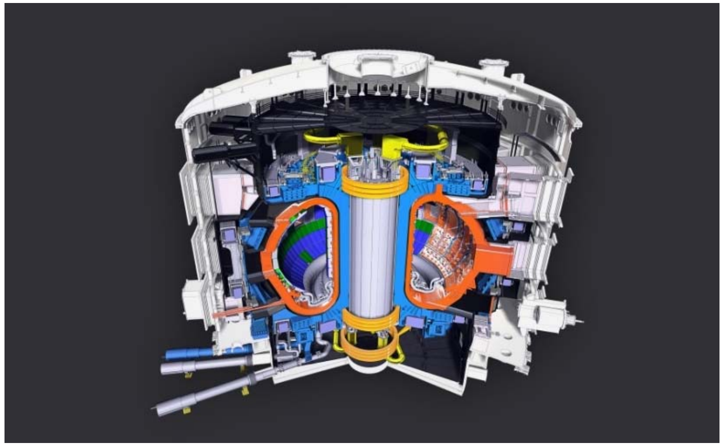
Outboard PHTS Loop Additional Data	
(He/MS) HX Power	301.6 MW
HX Heat Transfer Area	**~11500 m²**
HX Volume	~70m³





Divertor solutions: ITER





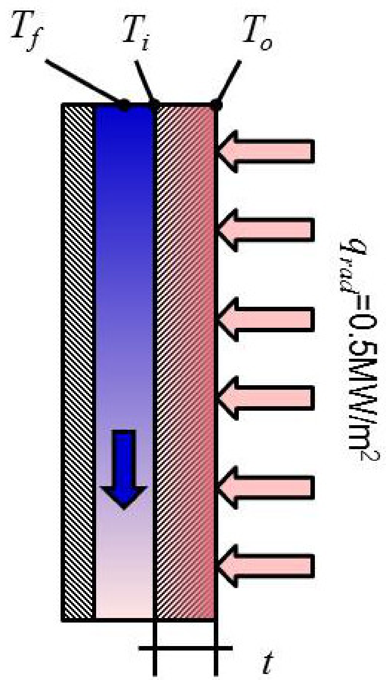
Divertor:

- 54 cassette (~10 t each)
- Able to withstand 10 MW/m^2 (transient to 20 MW/m^2) heat flux
- Installed/replaced by sophisticated remote handling

PFC - Fundamental function- „Cooling“

- **Wall surface heat flux caused by q_{rad}**
- **Requirement: $T_{mat} < T_{max,material}$**
- Where ? Outer side of first wall (FW) T_o
- Parameters
- T_f = bulk fluid temperature
- T_o = outer temperature FW
- T_i = inner temperature FW
- t = wall thickness
- λ = heat conductivity
- h = heat transfer coefficient

Example *
- T_f = 300°C
- t = 5mm
- λ = 20W/(mK)
- h = 8.000 W/(m²K)
  (w_{He}=80m/s)



$$T_i = T_f + \frac{q_{rad}}{h} \qquad T_o = T_i + \frac{q_{rad} \cdot t}{\lambda}$$

$$T_i = \left(300 + \frac{0.5 \cdot 10^6}{8 \cdot 10^3}\right)°C = 362.5°C$$

$$T_o = \left(362.5 + \frac{0.5 \cdot 10^6 \cdot 5 \cdot 10^{-3}}{20}\right)°C = (362.5 + 125)°C = 487.5°C$$

PFC - Operational functions- „Structures“

- **Thermal and other loads cause additional material loads**
- **Requirement:** $\sigma_{max} < \sigma_{Design}$ Where ? Everywhere, to be demonstrated

Several stress types:

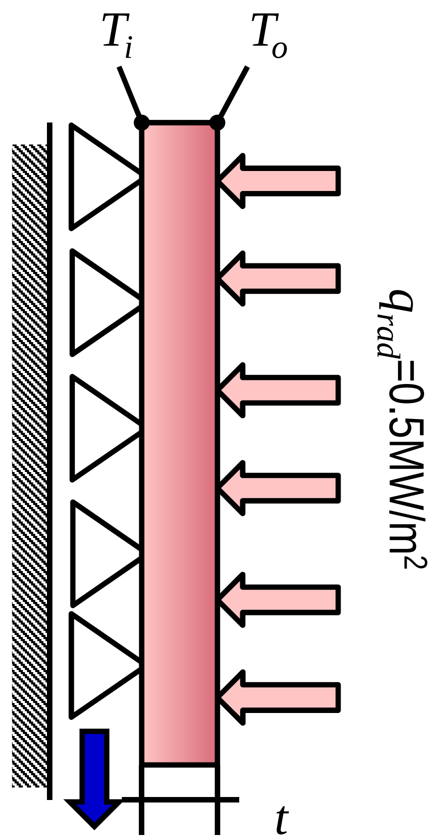
- *primary stresses* = pressure, mech.loads (bend, torque,….)
- *secondary stresses* = thermal loads
- *alternating stresses* = cyclic loads

Thermal loads on FW –plate:

- α = thermal exp. coefficient
- E = modulus of elasticity
- ν = Poisson ratio
- λ = Heat conductivity

Example:

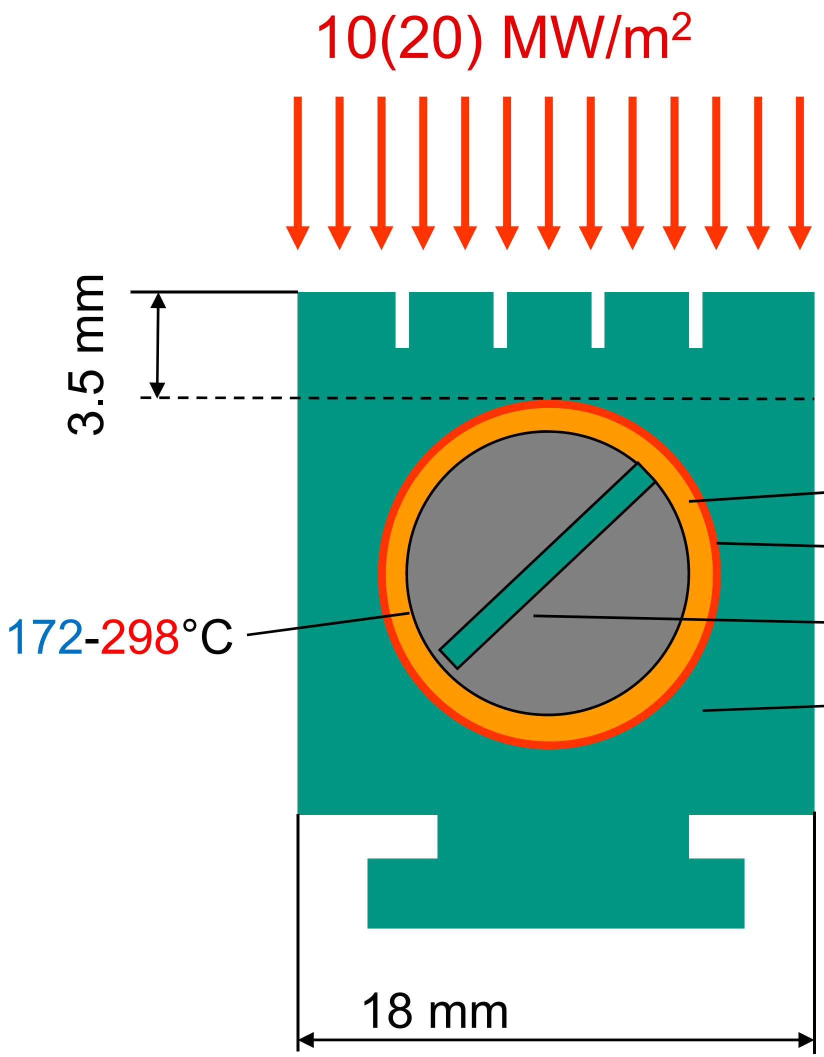
- α =1.8·10^{-5} 1/(K)
- E = 1.8·10^{11} Pa
- ν = 0.3, t = 5mm
- $(T_o - T_i)$ = 125°C
- q_{rad}=0.5MW/m^2

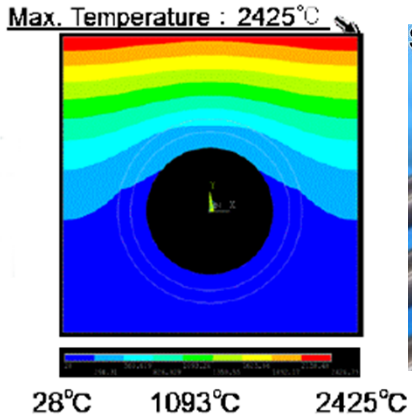


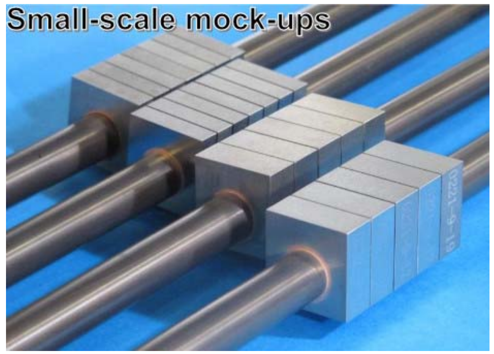
$$\sigma_{th,\max} = \frac{\alpha \cdot E \cdot (T_o - T_i)}{2(1-\nu)} = \frac{\alpha \cdot E \cdot q_{rad} \cdot t}{2(1-\nu) \cdot \lambda}$$

$$\sigma_{th,\max} \cong 290\ MPa$$

Water Cooled Divertor (Monoblock)

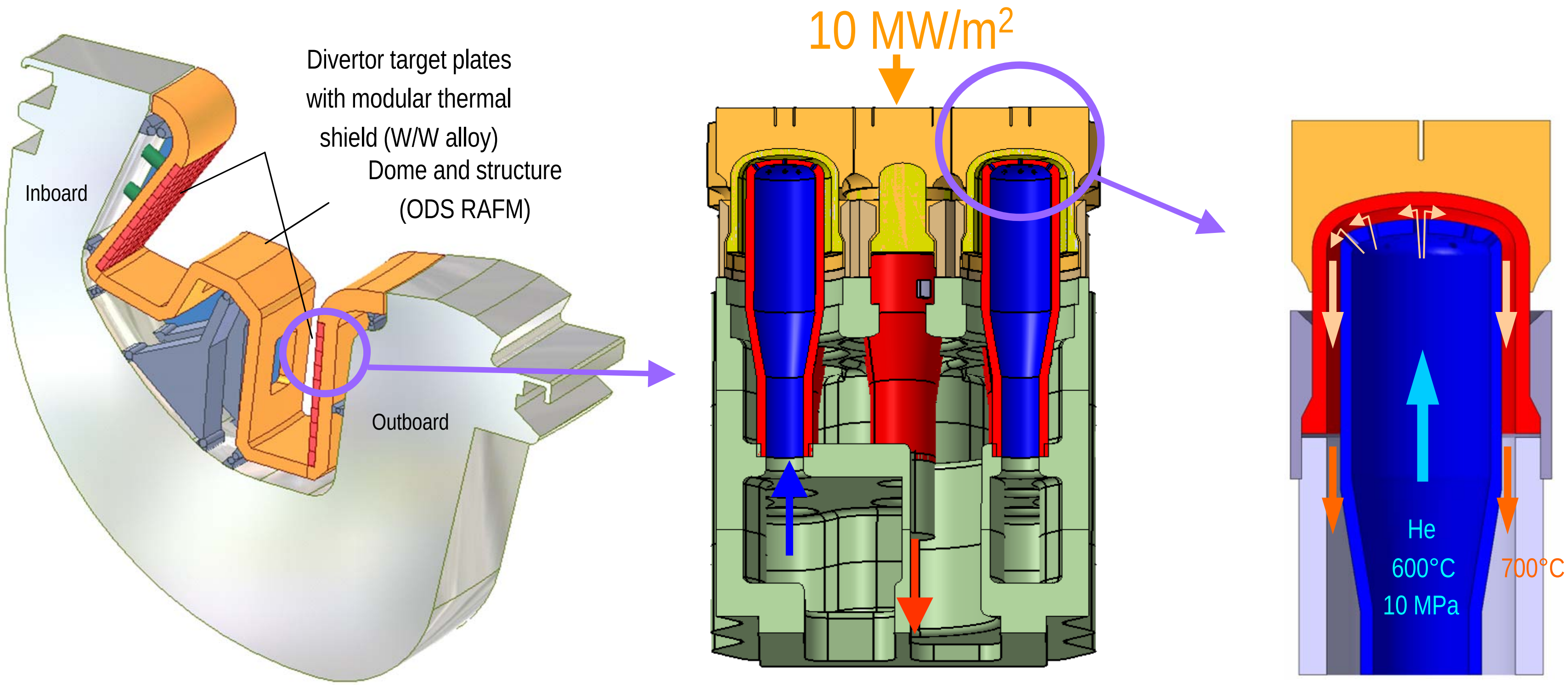






Coolant:
- inlet temperature 140°C
- pressure 4.2 MPa
- velocity 20 m s^{-1}
- Δt=27K (av.) (~167°C outlet)

Reference Design (RD): He-cooled modular divertor with jet cooling (HEMJ)



Divertor cassette

9-Finger module

1-Finger module

Outline

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  - Oxide dispersion strengthened steels
  - Neutron irradiated steels – selected results
  - W alloys
- Database maturity & role of materials in fusion roadmaps

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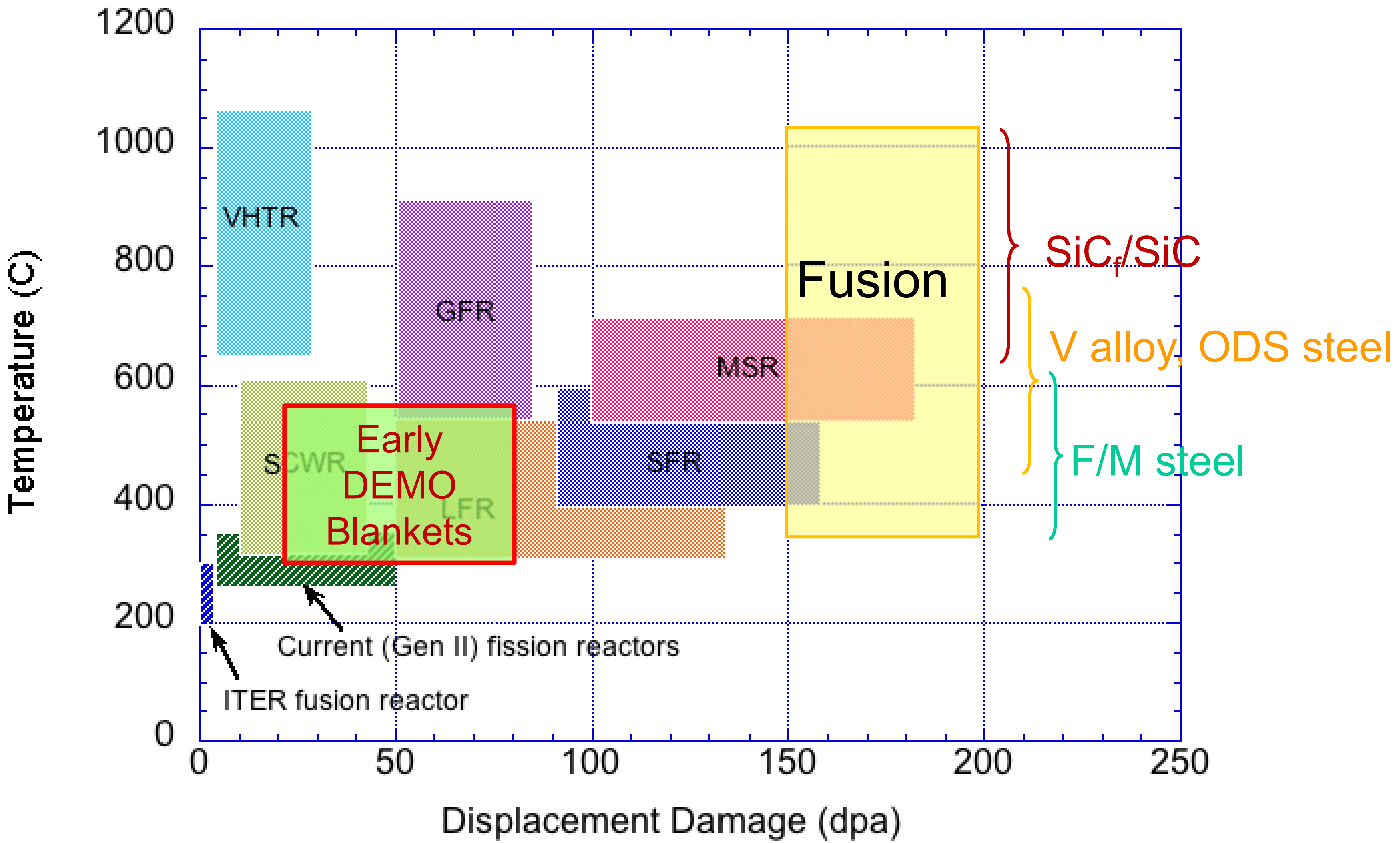
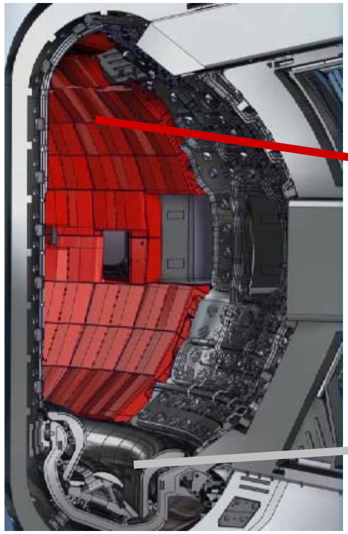
Gen IV and Fusion reactors pose severe materials challenges

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



Requirements for DEMO-Reactor Blankets

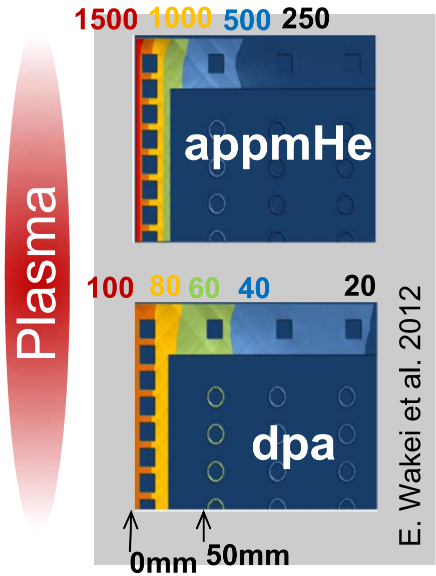


Blanket: ≤30 dpa/yr, 2.5 MW/m²

- Reduced Activation Materials:
  - RAFM Steels 350-550°C
  - ODS steels 300-750°C
- Functional Materials, Plasma facing materials

Divertor: ≤10 dpa/yr, 10-15 MW/m²

- Refractories (e.g. W-materials) 500-1300°C
- Low activation "high entropy" alloys?,……



He & dpa production for PFCs

- "Only" few centimeters have a high He/dpa ratio
- Plasma-near part carries also highest heat loads
- - Fission reactors:
  relevant for most of in-vessel structures
  - Dedicated fusion neutron source:
  indispensable for plasma-near materials

In Vessel Structural Materials
- Overview -

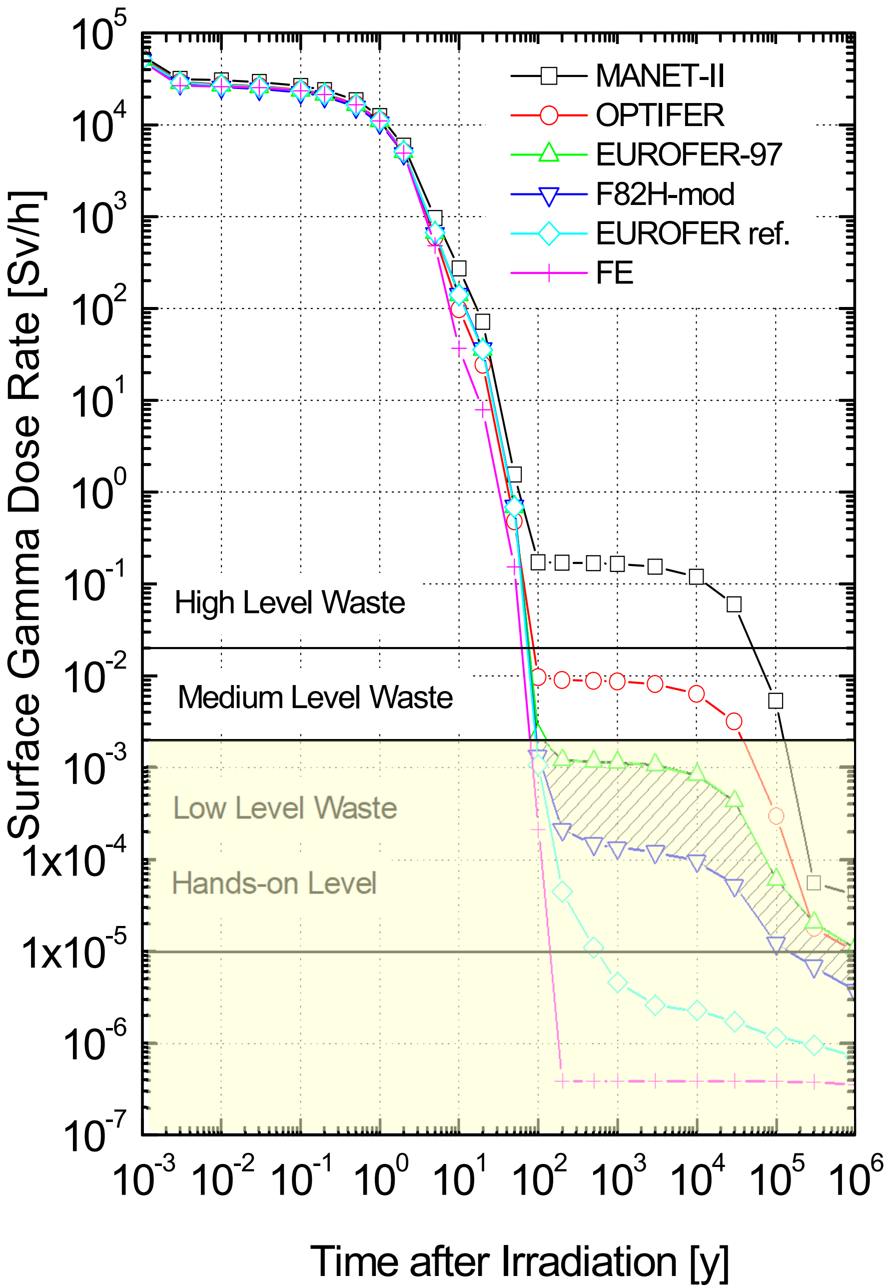
The requirements on Fusion Power Reactors have led in the past few decades to a worldwide concentration of the R&D activities on few material classes:

- Reduced Activation Ferritic-Martensitic (RAFM)
  9-10% CrWTa-Steels; present development: TMT optimisation
- Oxide Dispersion Strengthened ODS-RAF(M)-Steels with nanoscaled dispersoids (Y_2O_3 and Y-Ti-O type)

Vanadium alloys of type V-4Cr-4Ti

Fiber reinforced Silicon Carbides of type SiC_f/SiC

Refractory alloys and composites (W-based)



Priority:
Low activation capability

RAFM 8-10%CrWTaV steels

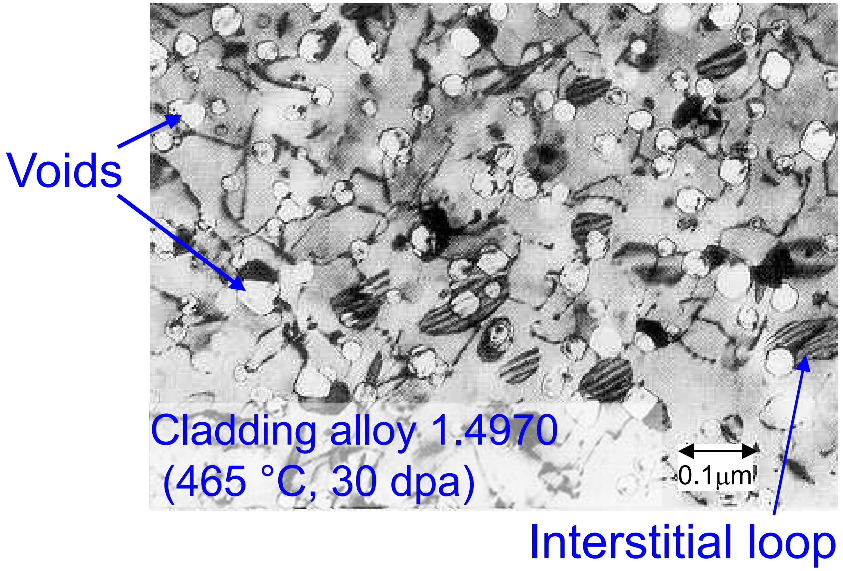
- „Low level waste“ already after 80-100 years
- No “high level” waste disposal
- The impurities Nb and Mo are dominating the hatched area
- >100 tons produced meanwhile worldwide
- Implementation of EUROFER database into RCC-MRx-code

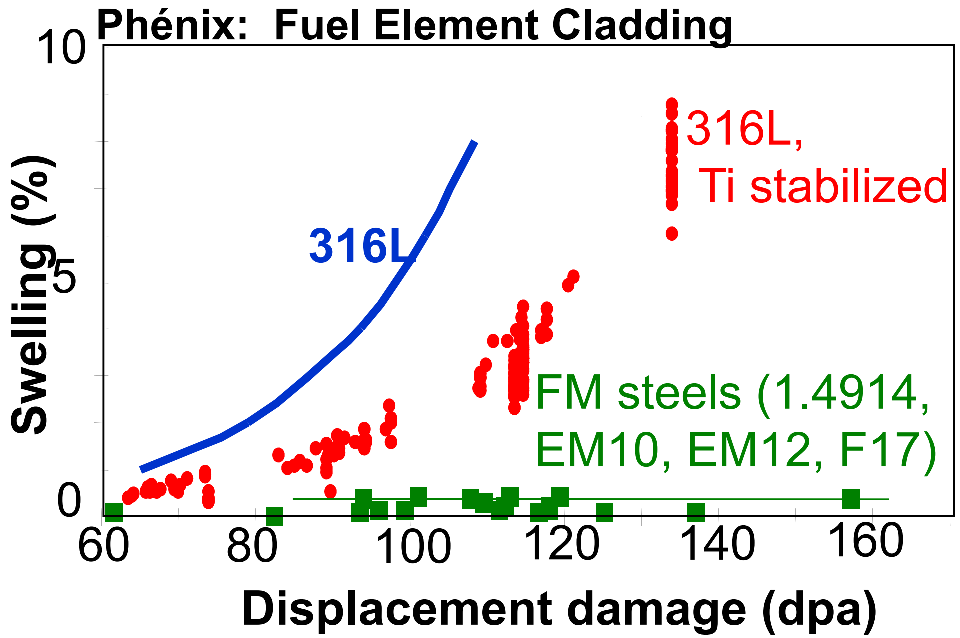
Long term irradiation ($12.5\ MWa/m^2$) of a DEMO reactor first wall

Selection criteria for in-vessel structural materials - Why not austenitic steels?

Historical development: In the conventional fission reactor technology (Generation 1 and 2) austenitic steels have been dominating the structural alloy application. They provide excellent welding properties, but have despite of enormous alloy improvements (e.g. DIN 1.4970, X10CrNiMoTiB 15 15) still substantial disadvantages.

- Strong swelling at high dpa doses

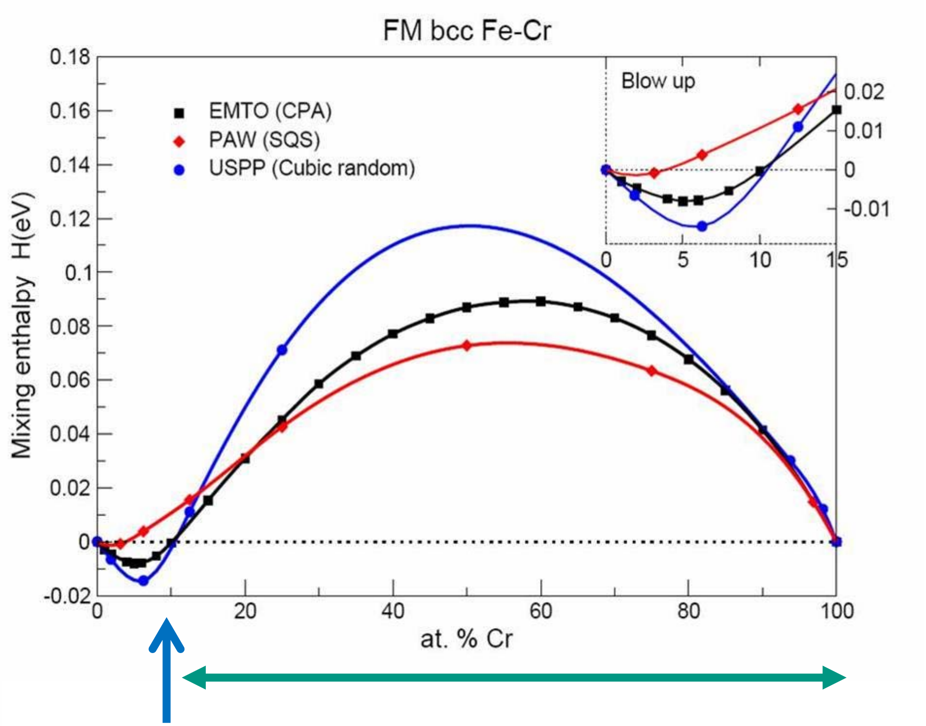




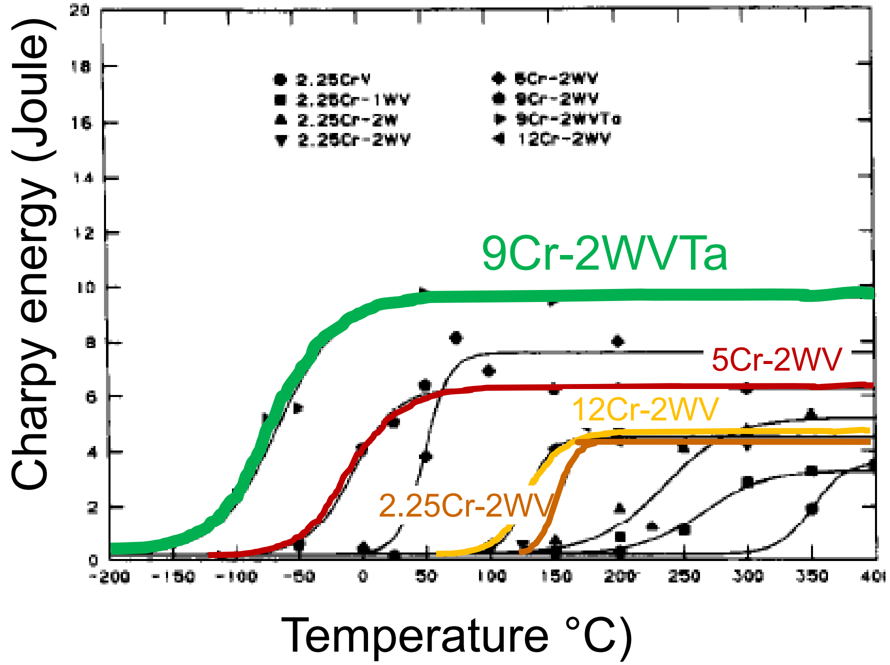
Why RAFM steels are based on 9wt% Cr?

S. Dudarev et al, UKEA, 2006

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



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



9-10%Cr is favorable

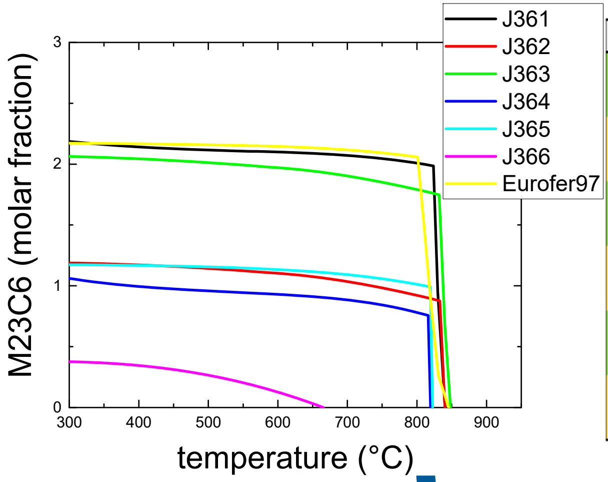
Above 10% Cr: formation of Cr precipitates (σ-phase, α')

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

Advanced ferritic-martensitic 9-10CrWVTa Steels:

Novel thermomechanical treatment

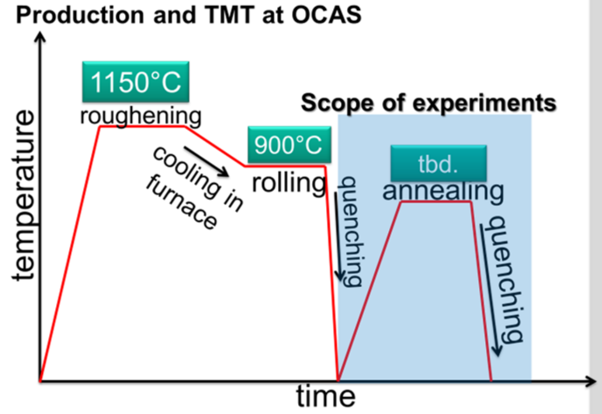
Thermodynamic Simulations



Heat variations

Nr	Name	Cr	W	V	Ta	N	C
1	EUROFER-s **J361**	9	1	0,2	0,1	0,04	0,1
2	EUROFER-LV **J362**	9	1	0,35	0,1	0,04	0,06
3	EUROFER-V **J363**	9	1	0,35	0,1	0,04	0,1
4	OPTIFER-LVwT **J364**	9	1	0,35	-	0,04	0,06
5	OPTIFER-LVwW **J365**	9	-	0,35	0,1	0,04	0,06
6	OPTIFER-vLvVTwW **J366**	9	-	0,62	0,14	0,04	0,06

Thermo-mechanical treatment



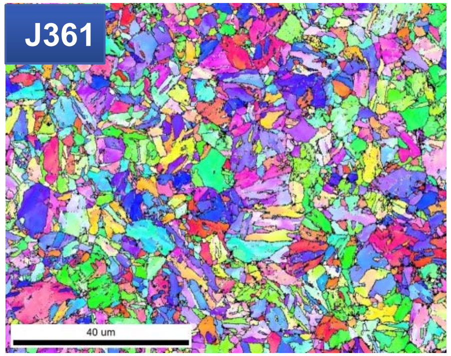
J. Hoffmann, M. Rieth, et al;
Nucl. Mater. Energy. 6 (2016) 12–17

>20 new alloys in Europe

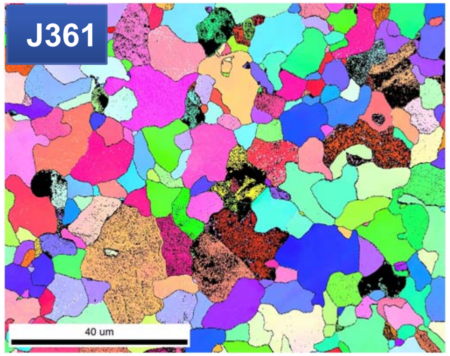
- Optimized toughness
  Broad based mechanical characterization

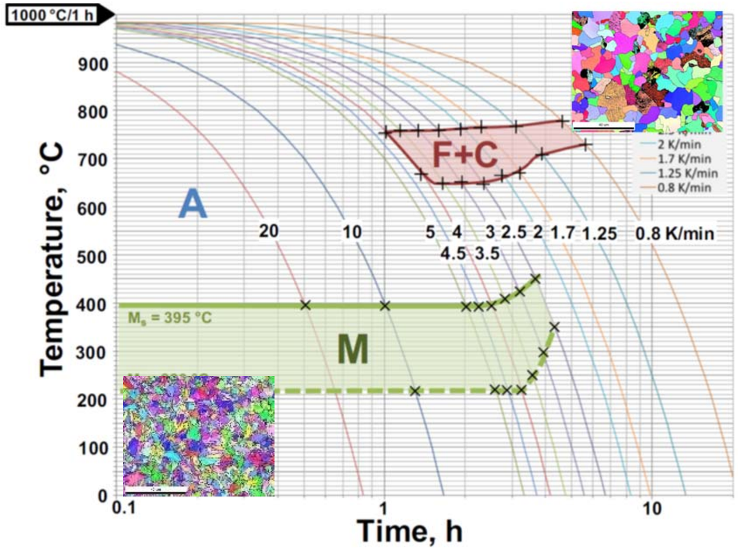
Advanced ferritic-martensitic 9-10CrWVTa Steels:
Novel thermomechanical treatment

Martensite



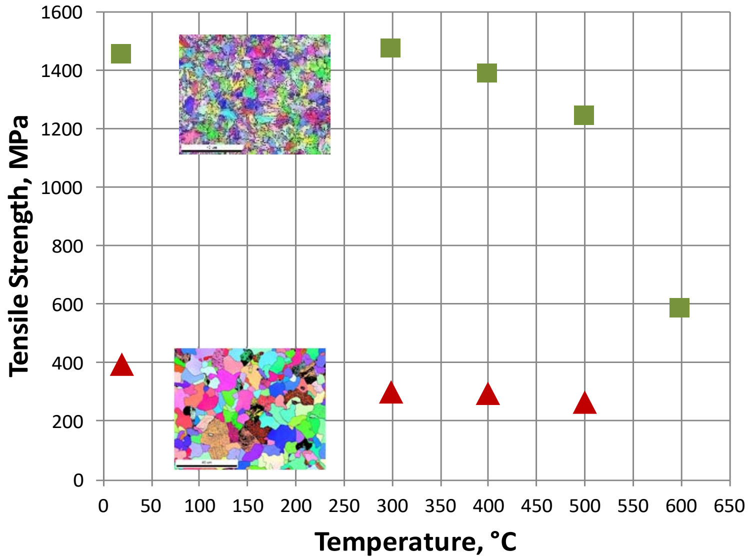
Ferrite + Carbides



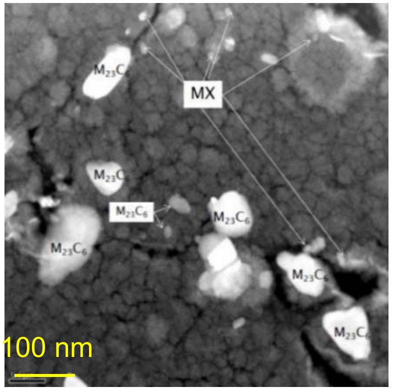


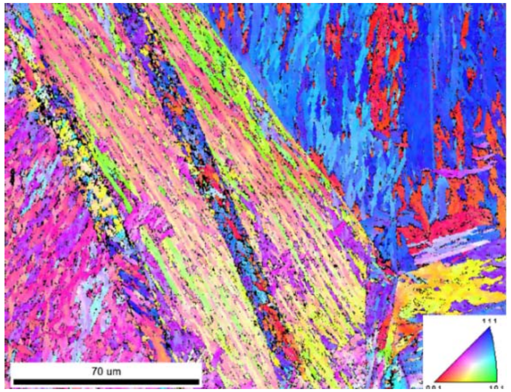
9-10CrWVTa steels can be efficiently optimized towards more low or more high temperature applications with

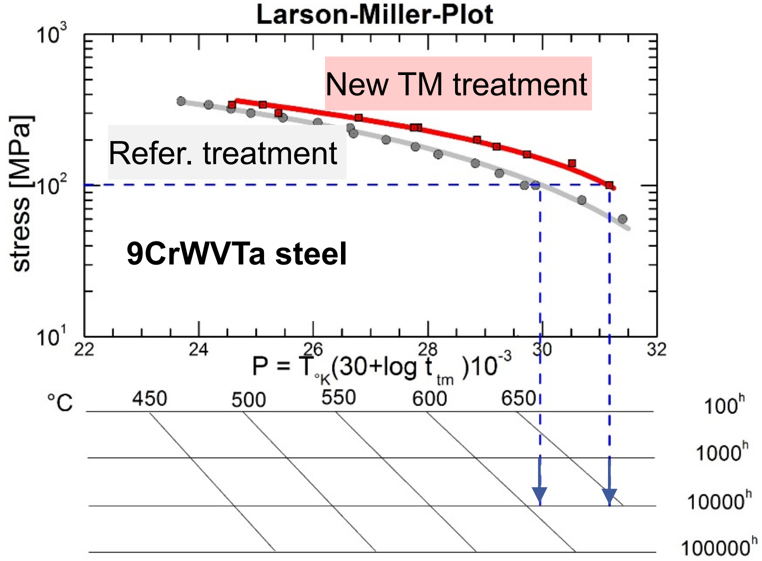
- Very minor alloy modifications
- Thermo-mechanical treatments after fabrication



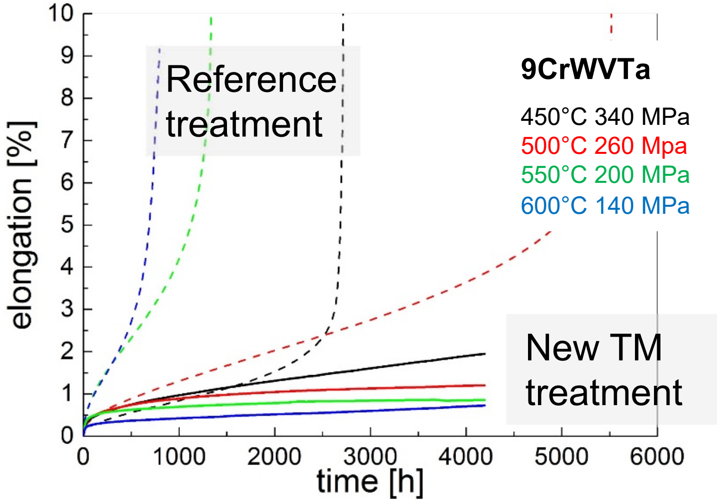
Advanced ferritic-martensitic 9-10CrWVTa Steels:
Novel thermomechanical treatment







Hoffmann et al., Nuclear Materials and Energy 2016
J. Hoffmann, M. Rieth, et al, Fus. Eng. Des. 98-99 (2015) 1986–1990



>20 new alloys produced 2016-2017

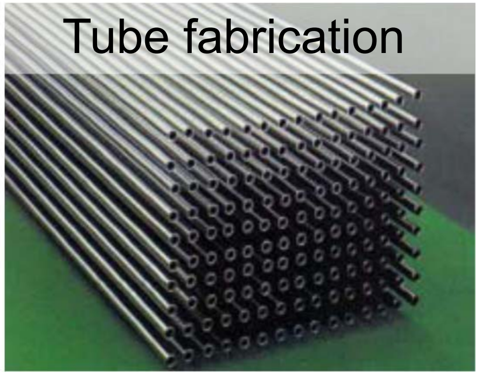
- Optimization of carbide and nitride phases
- Substantial improvement of high temperature creep strength

ODS steels for high temperature resistance and irradiation tolerance

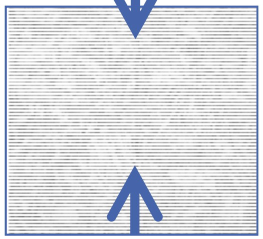
Oxide Dispersion Strengthened (ODS) Steels can be categorized as follows:

- ODS Ferritic-Martensitic steels, (9-10)Cr-(1-2)WVTa-0.35Y_2O_3
- ODS Ferritic steels, (13-18)Cr-(1-2)WVTa-0.35TiYO
- ODS Ferritic steels, (13-18)Cr-(1-2)WTa-(4-5)Al-0.6(Hf)ZrYO
- ODS Austenitic steels, 16Cr-15Ni-(1-2)WV-0.35TiYO

Fabrication processes meanwhile advanced

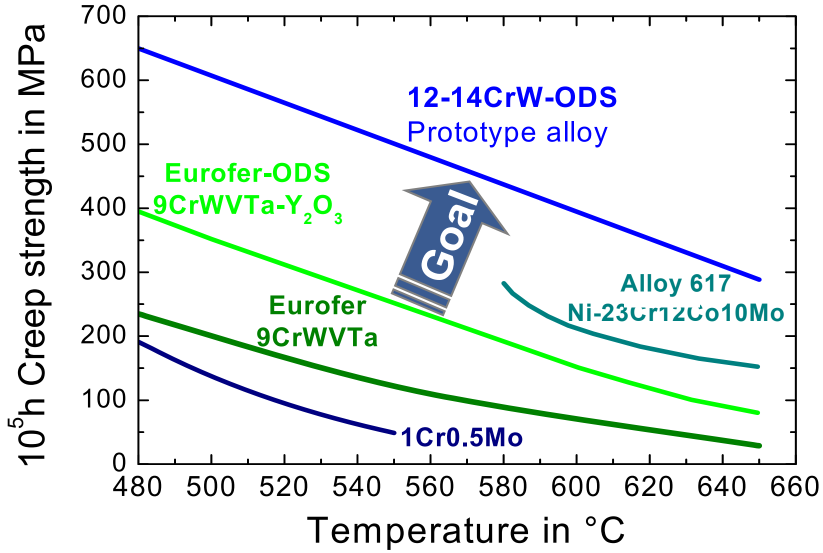


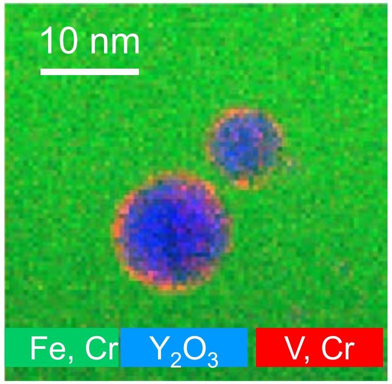
Diffusion weld hardly visible



International challenge: Development of nanoscaled iron based "super alloys" - (13-18)Cr-(1-2)WVTa-0.35TiYO

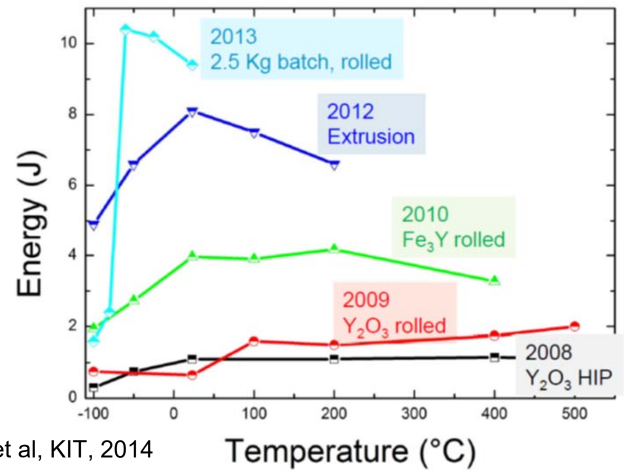
Superior Creep properties



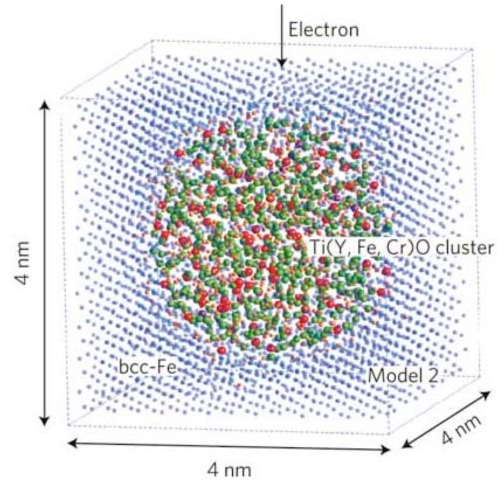


M. Klimenkov et al.,JNM 428 (2012)

DBTT substantially improved



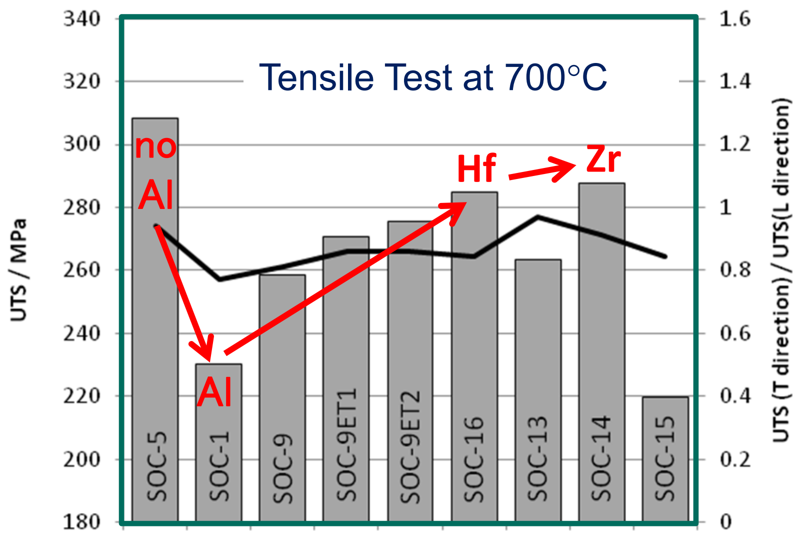
J. Hofmann et al, KIT, 2014



A.Hirata et al., Nature Materials 10 (2011)

Ferritic ODS steels with Al for high corrosion resistance - (13-18)Cr-(1-2)WTa-(4-5)Al-0.6Zr(Hf)YO

Japanese R&D program to improve the strength weakness of the highly corrosion resistant "PM2000-type" ODS-steels



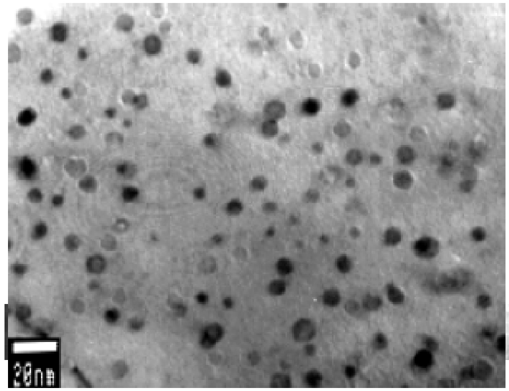
A. Kimura et al, 2015

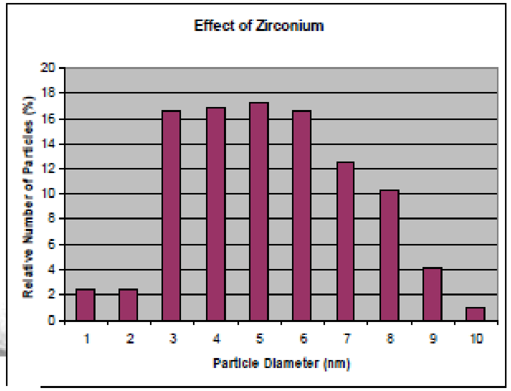
Additions of 0.6w% Zr(Hf) remarkably increases the tensile strength of Al-added ODS steels.

Ferritic ODS steels with Al for high corrosion resistance - (13-18)Cr-(1-2)WTa-(4-5)Al-0.6Zr(Hf)YO

Oxide particles in Al-ODSS
Ave. Diameter: 7 nm
Density: 1.6×10^{22} m^{-3}

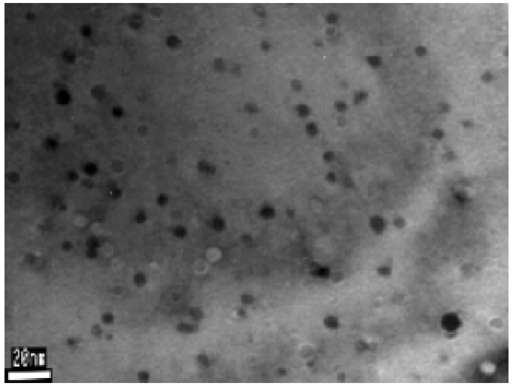
Zr or Hf addition resulted in fine oxide dispersion.

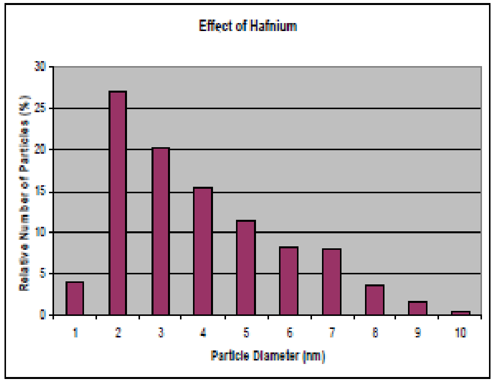




Zr addition

Ave. diameter: 4.7 nm
Density: 7.2×10^{22} m^{-3}





Hf addition

Ave. diameter: 3.5 nm
Density: 4.8×10^{22} m^{-3}

A. Kimura et al, 2015

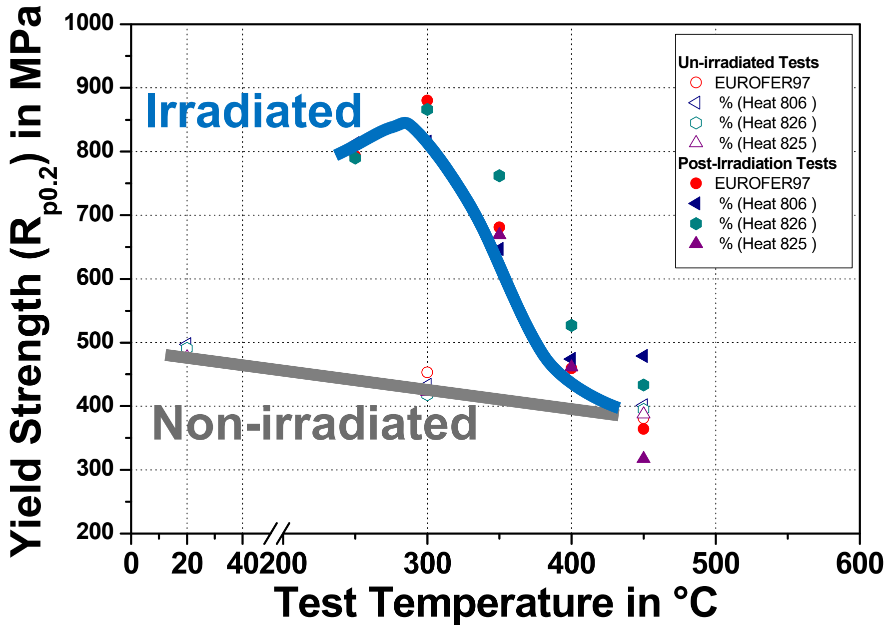
A. Moeslang Current reactor designs and structural materials ICFRM-18, 2017, Aomori, Japan

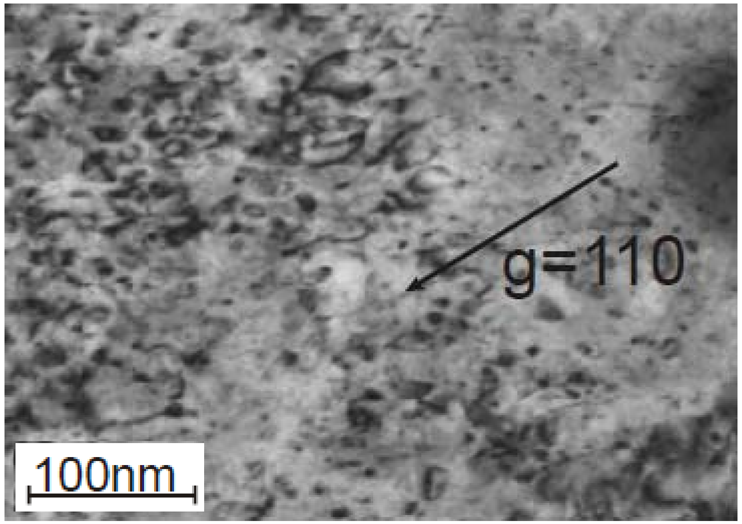
Outline

- DEMO reactors: current designs – blankets - divertors
- Fusion Reduced Activation Structural Materials (DEMO-oriented): recent progress
  - Reduced activation ferritic/martensitic steels
  - Oxide dispersion strengthened steels
  - Neutron irradiated steels – selected results
  - W alloys
- Database maturity & role of materials in fusion roadmaps

RAFM steels: Substantial irradiation induced hardening below T_{irr} ~ 400°C mostly by interstitial type defects

Karlsruhe Institute of Technology





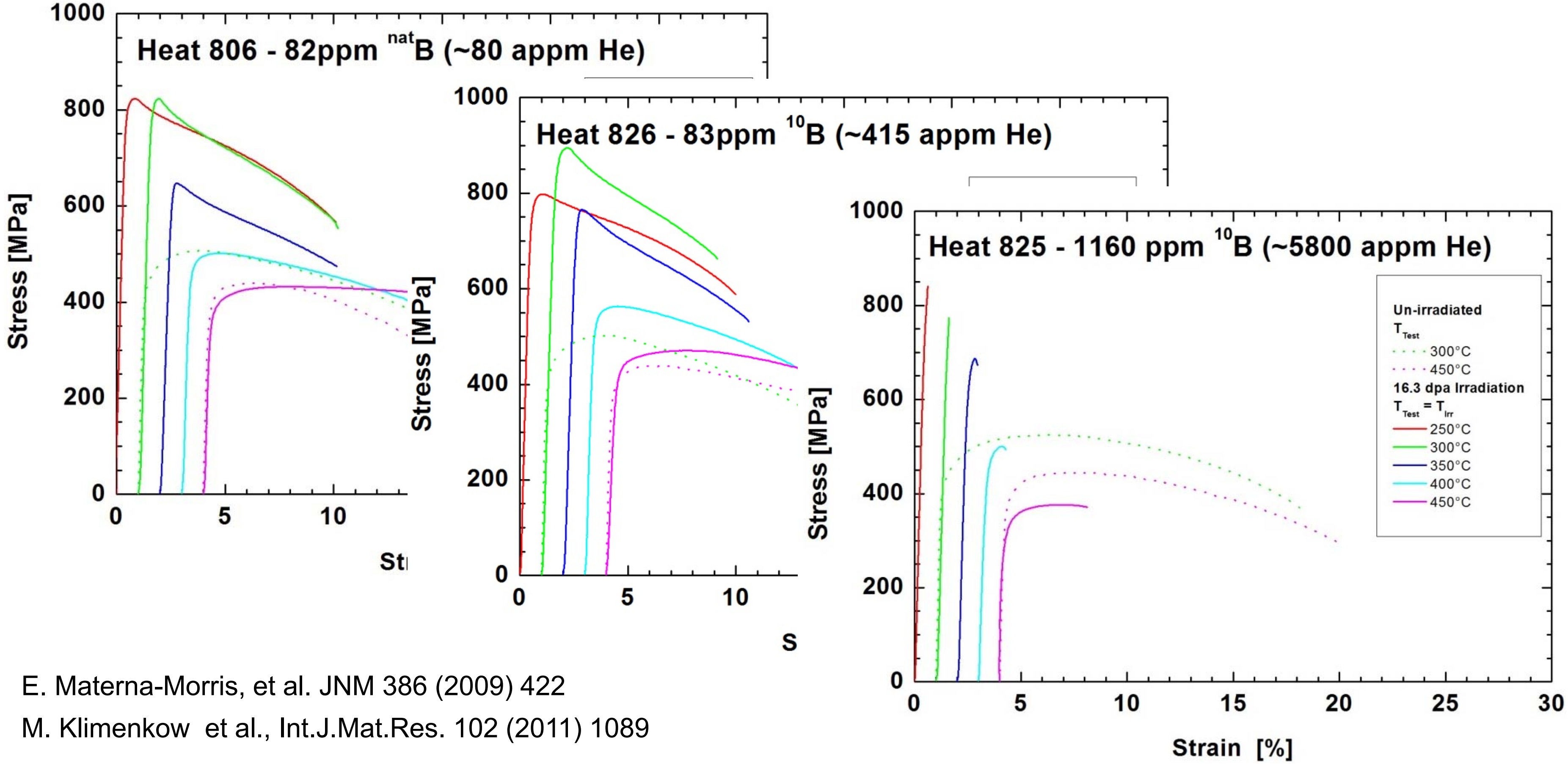
T_{irr} <~400°C:
- Homogeneous distribution of point defects and dislocation loops (½<111> Burgers vector, 5-25 nm diameter)
- Severe uniform elongation and fracture toughness reduction

T_{irr} >~400°C:
- No irradiation induced hardening,
- Only small ductility reduction and minor swelling
- Favorable irradiation tolerance even at high dpa doses

Ductile or brittle? The importance of strain rate $\dot{\varepsilon}$:

Example: Eurofer, 16 dpa, B-doped, $\dot{\varepsilon} \approx$ **$10^{-3}s^{-1}$**

Karlsruhe Institute of Technology



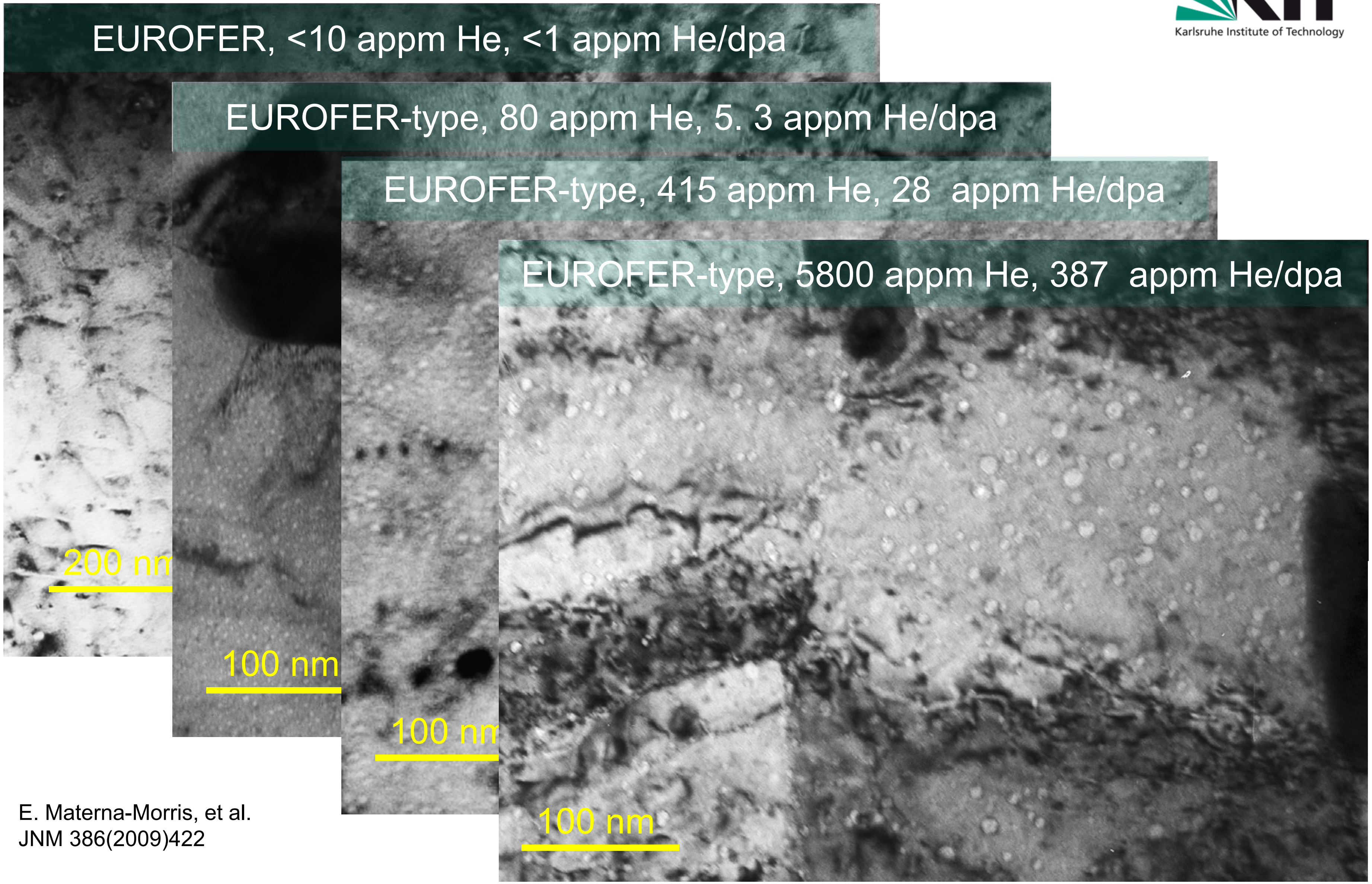
E. Materna-Morris, et al. JNM 386 (2009) 422

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

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

IAM Institut für Angewandte Materialien

RAFM Steels, 15 dpa neutron irradiation at 250 °C

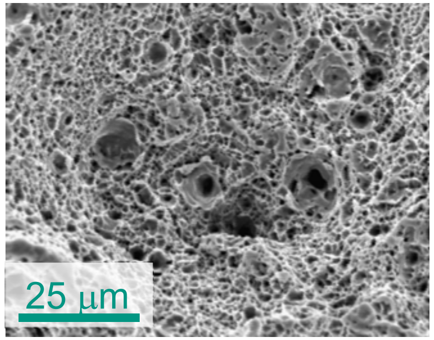


E. Materna-Morris, et al.
JNM 386(2009)422

EUROFER Steel: Fracture Behavior

Neutron irradiation: 16 dpa, $T_{irr} = T_{test} = 300\ °C$

EUROFER, <10 appm He

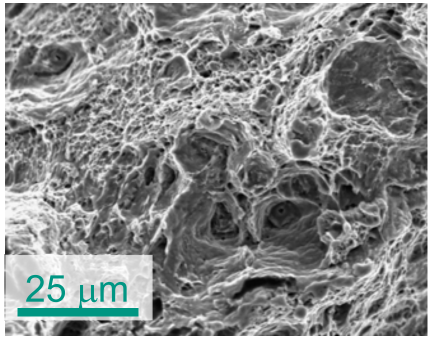


M. Klimenkov et al., Micron 46 (2013) 51–56

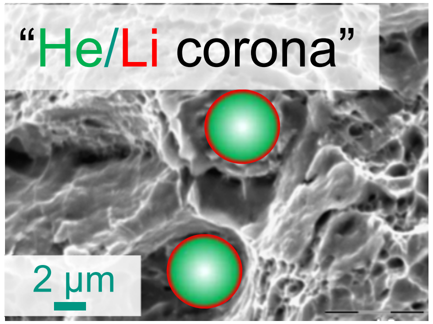
M. Klimenkov et al, J. Nucl. Mater. 462 (2015) 280-288

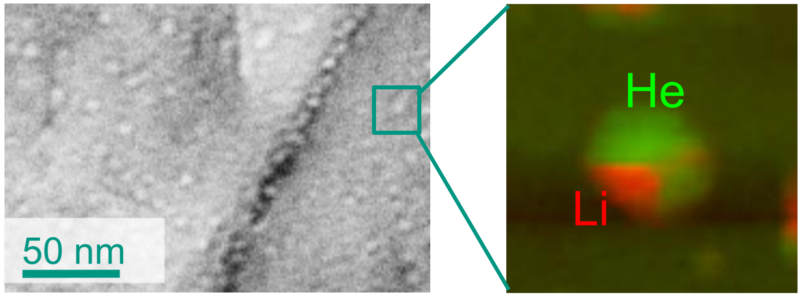
- KIT: Worldwide first direct observation of Li clusters
- Broad database on dpa/He effects
- ^{10}B-doping: He and Li effects cannot be decoupled. Intense neutron source needed

EUROFER-type, B-doped, 415 appm He



$n + {}^{10}B \rightarrow {}^{4}$**He** + ^{7}**Li** + 2.8 MeV
range of **He** (1.0 MeV): 1.6 µm
range of **Li** (1.8 MeV): 2.0 µm

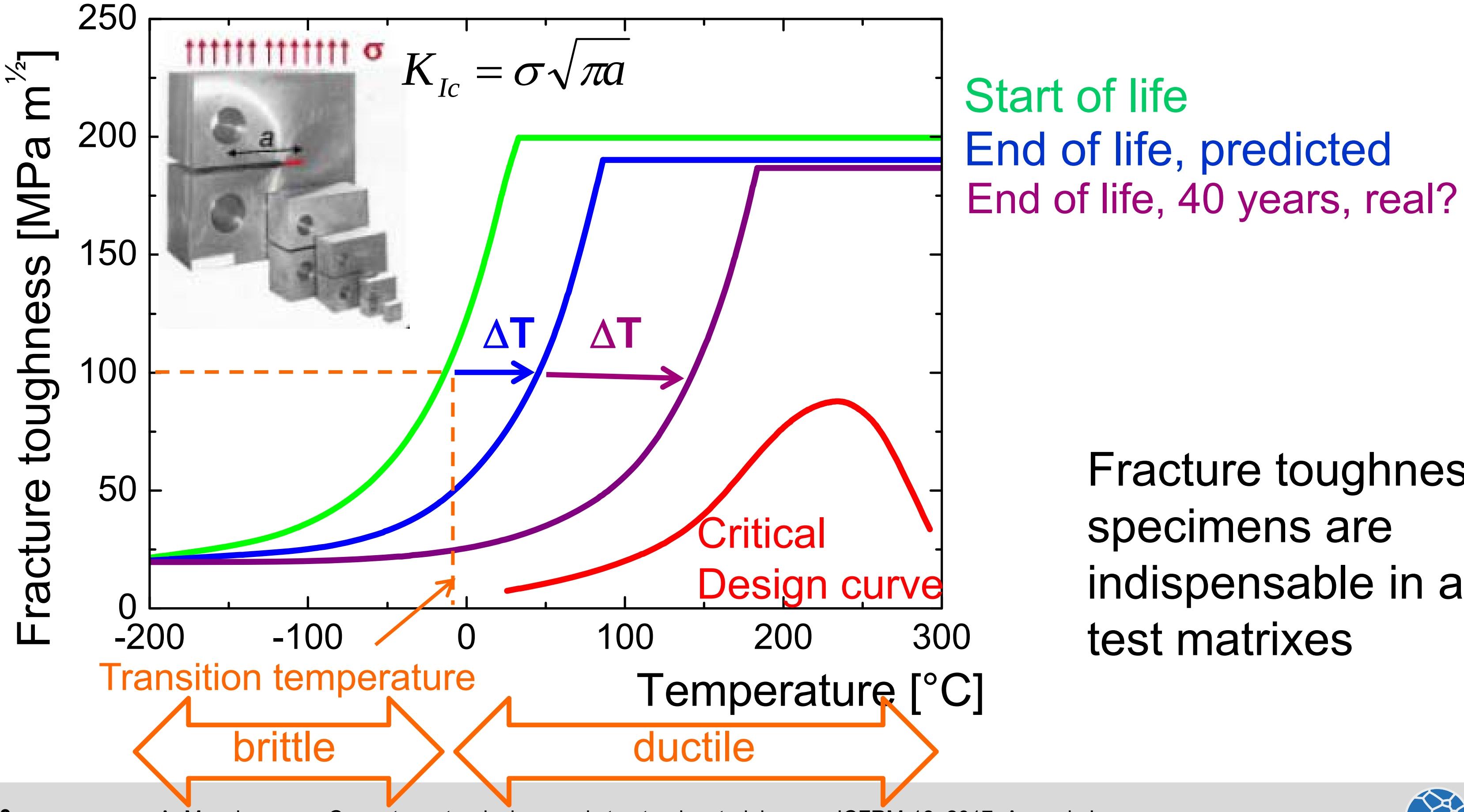




Ductile or brittle?

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)



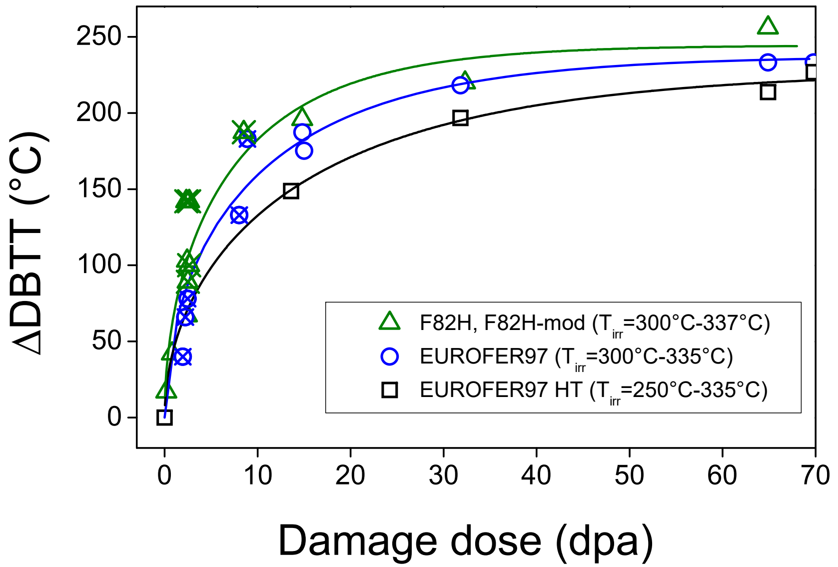
Fracture toughness specimens are indispensable in all test matrixes

Ductile or brittle? The importance of strain rate $\dot{\varepsilon}$:

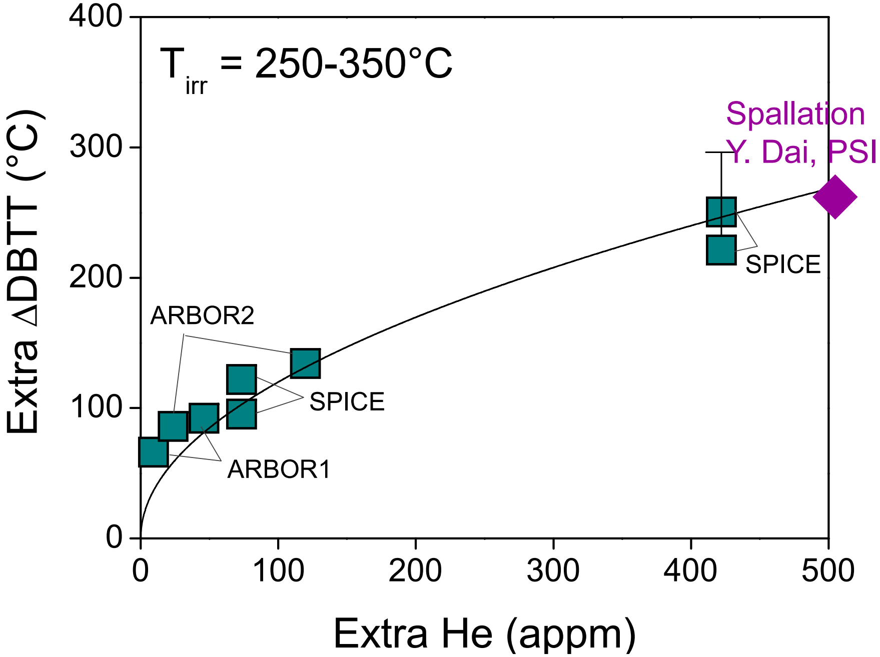
Example: Eurofer, 16 dpa, B-doped, $\dot{\varepsilon} \approx$ **$10^2 s^{-1}$**

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

EUROFER, <10 appm He

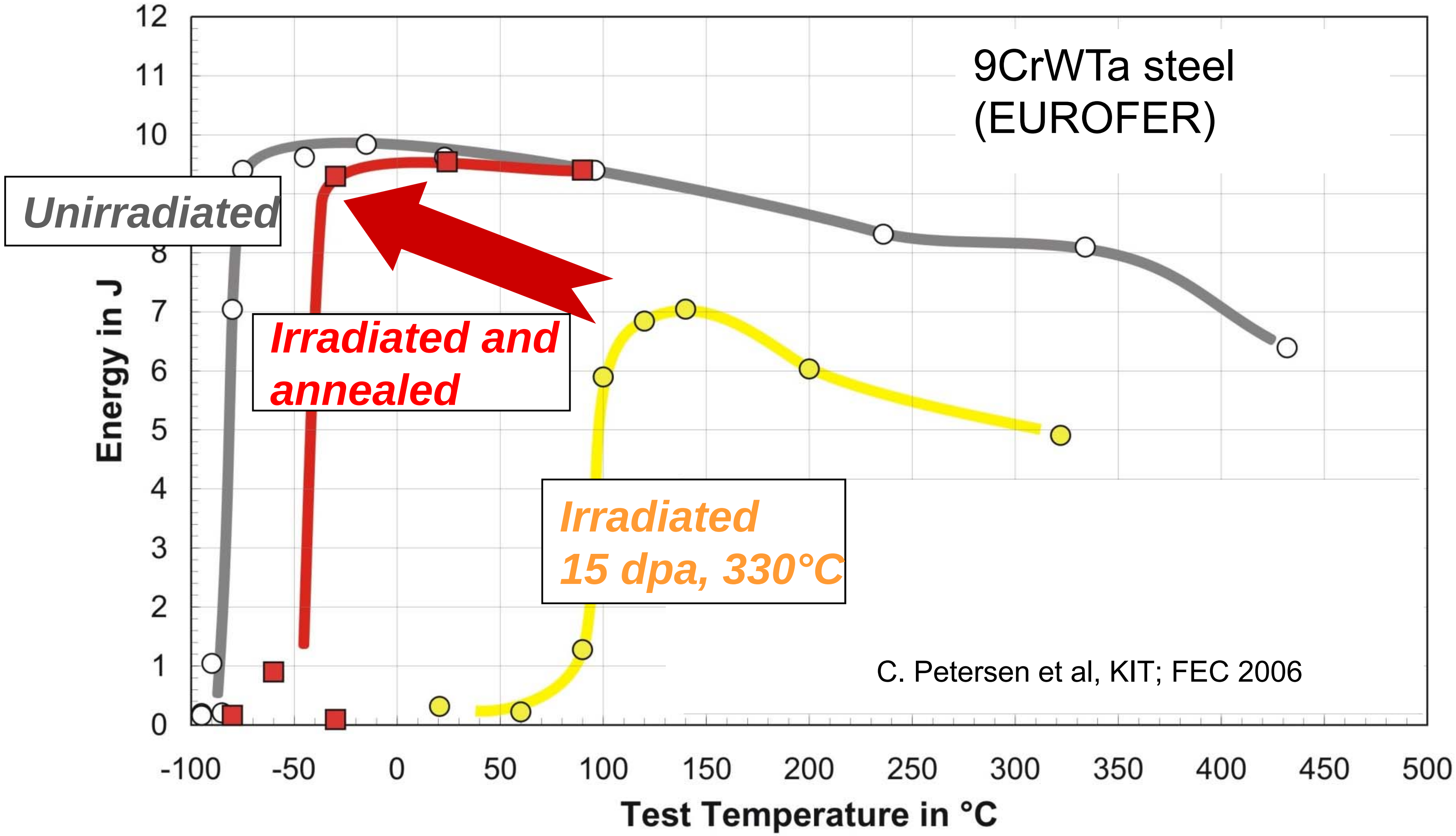


EUROFER, 10-500 appm He



- ❑ <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

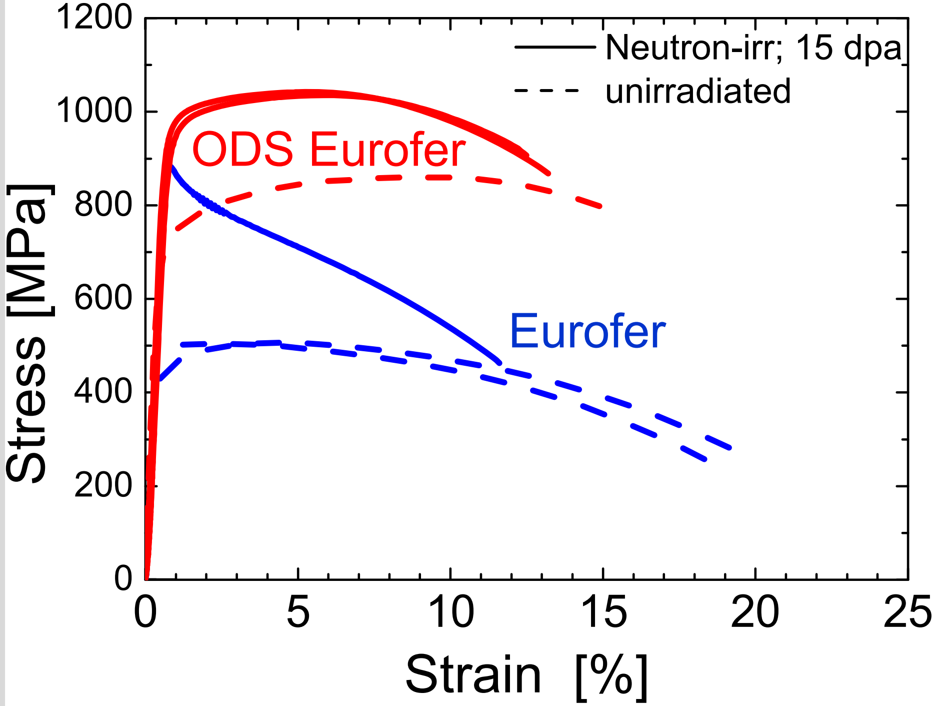
Is it possible to anneal irradiation embrittlement?



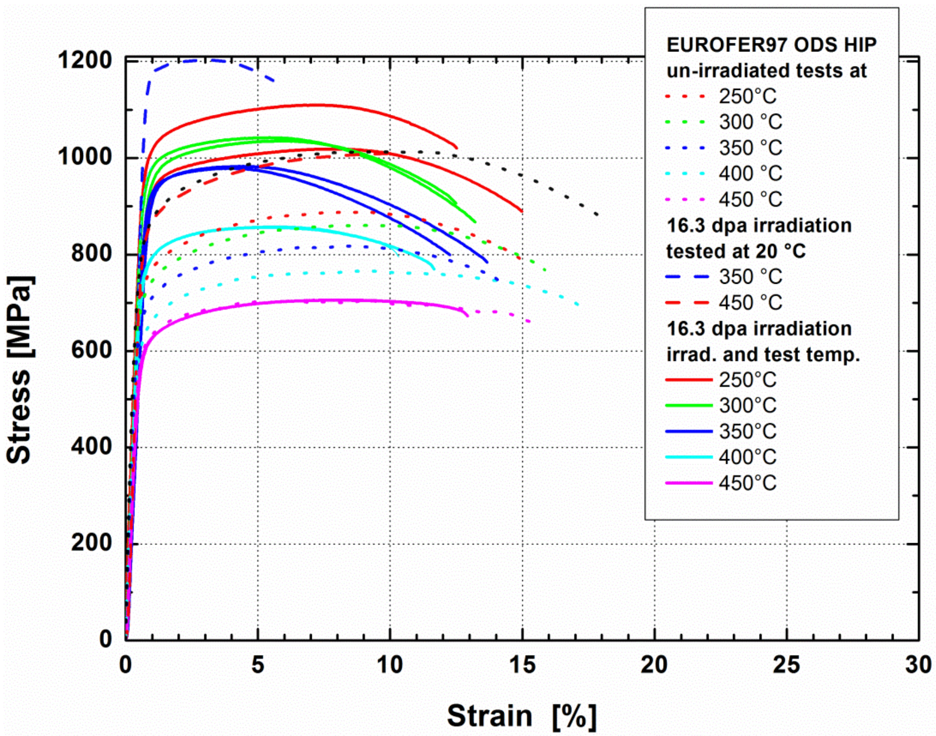
- How often can this recovery be repeated?
- What happens if large concentrations of He are present?

ODS EUROFER After Neutron Irradiation:
Substantial improvement of tensile properties

$T_{test} = T_{irr} = 300°C$



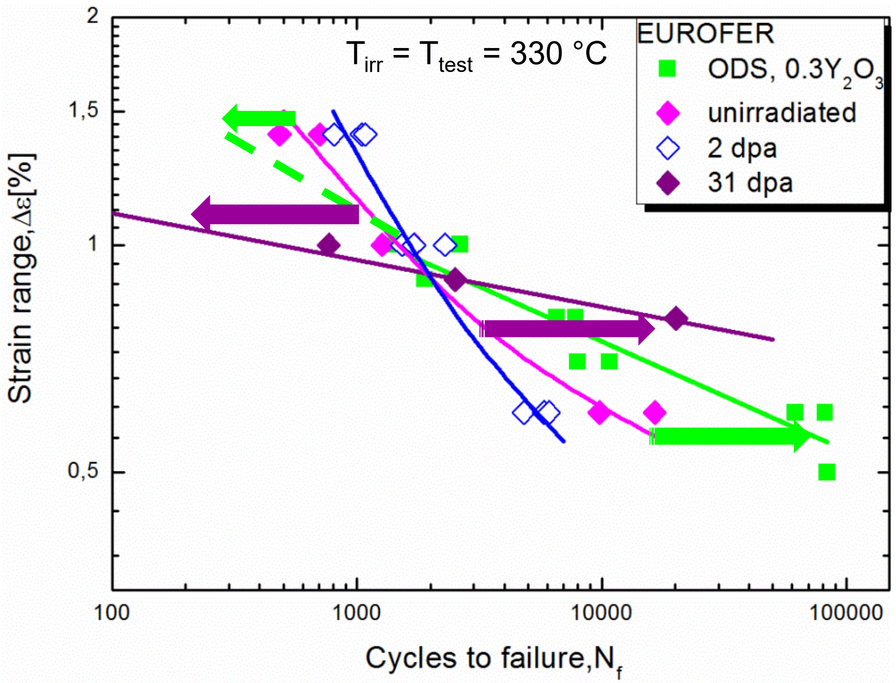
RAFM-ODS Steel, 250-450°C



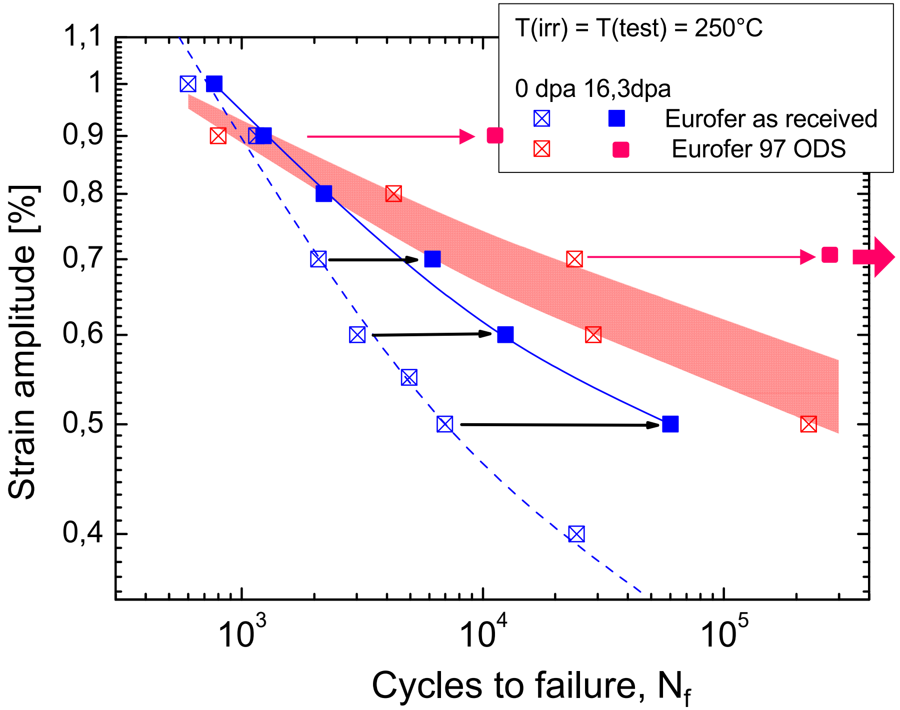
E. Materna-Morris, et al.; FusEngDes 98-99 (2015) 2038-2041

- ODS RAFM steel: superior uniform elongation and strength
- RAFM steels: Early strain localization due to dislocation channeling
- Still work hardening → almost no loss of uniform elongation (A_u ~7%) between 250 and 450°C

Fatigue Testing After Neutron Irradiation:
Substantial fatigue life improvement



E. Gaganidze et al., Nuclear Fusion 51 (2011), 083012



A. Möslang, H.Ch. Schneider et al, 2017, unpublished

- **High strain regime** → accelerated crack initiation → shorter lifetime
- **Low strain regime** → crack growth impeded → longer lifetime

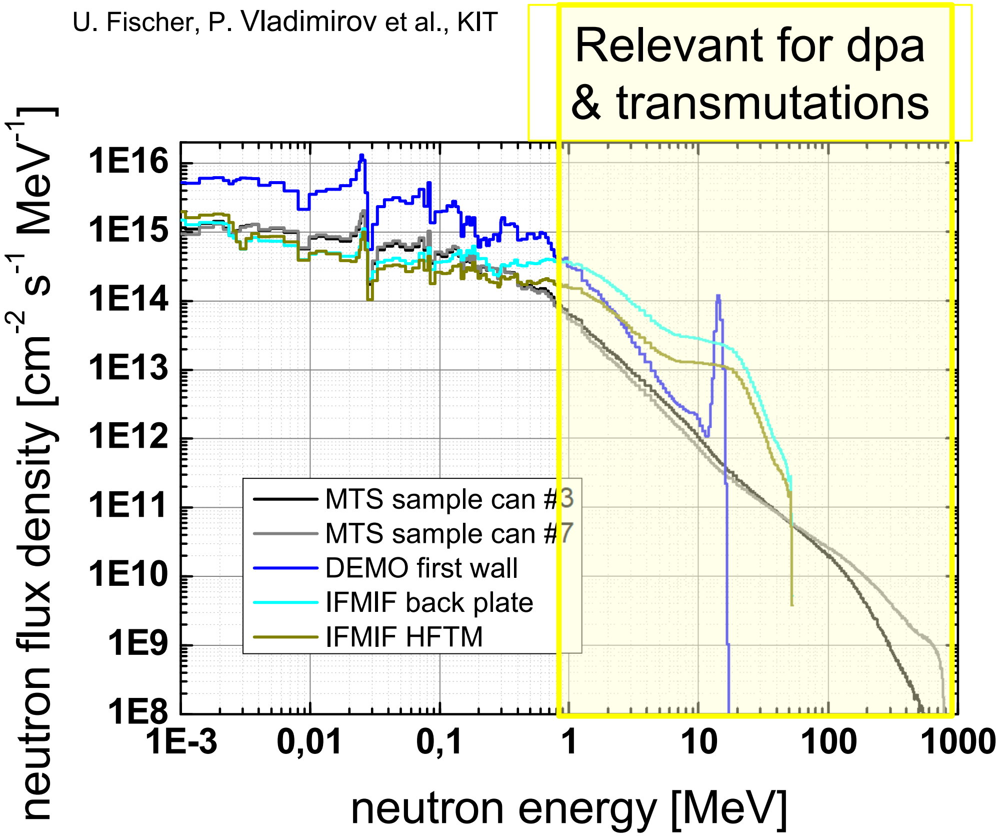
Nanoscaled ODS steels show:
- (Almost) no cyclic softening
- Unprecedented lifetime (fatigue testing interrupted at cycle 250000

Neutron spectra effects: Tensile properties

Karlsruhe Institute of Technology

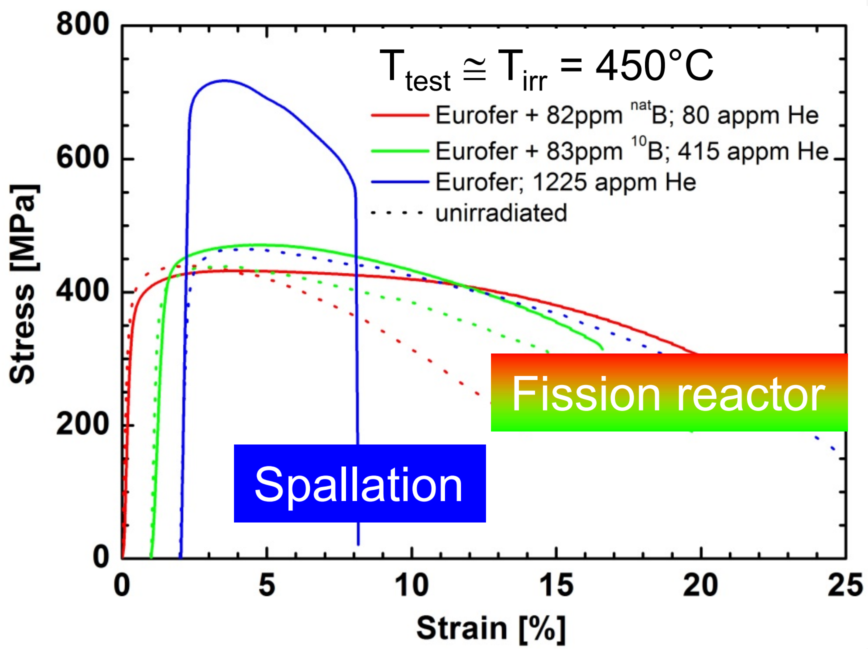
N-spectra: DEMO, IFMIF, Spallation

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

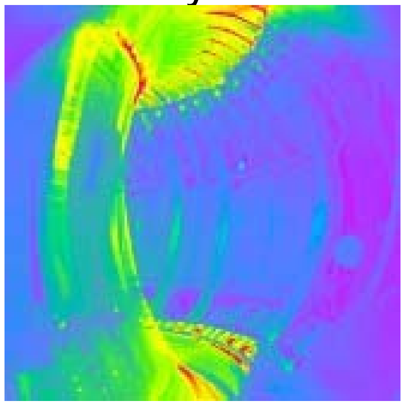
IAM
Institut für Angewandte Materialien

Outline

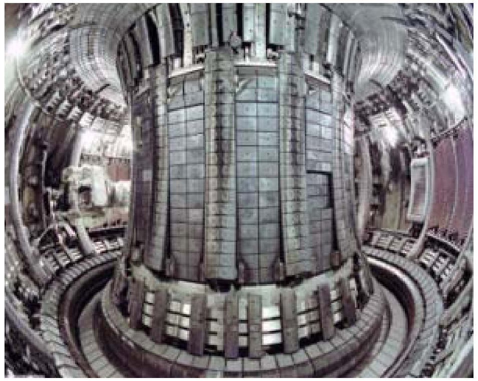
- DEMO reactors: current designs – blankets - divertors
- Fusion Reduced Activation Structural Materials (DEMO-oriented): recent progress
  - Reduced activation ferritic/martensitic steels
  - Oxide dispersion strengthened steels
  - Neutron irradiated steels – selected results
  - W alloys
- Database maturity & role of materials in fusion roadmaps

Tungsten materials: What is the problem?

- Divertor applications ask for a high temperature (1000°C) **structural** material
- W is the metal with the highest melting point of all metals (T_S = 3420°C)
- Disadvantages:
  - Low fracture toughness, K_{IC} [MPa $m^{1/2}$]
  - High brittle-to-ductile transition temperature (BDTT)
  - Recrystallization at high temperatures



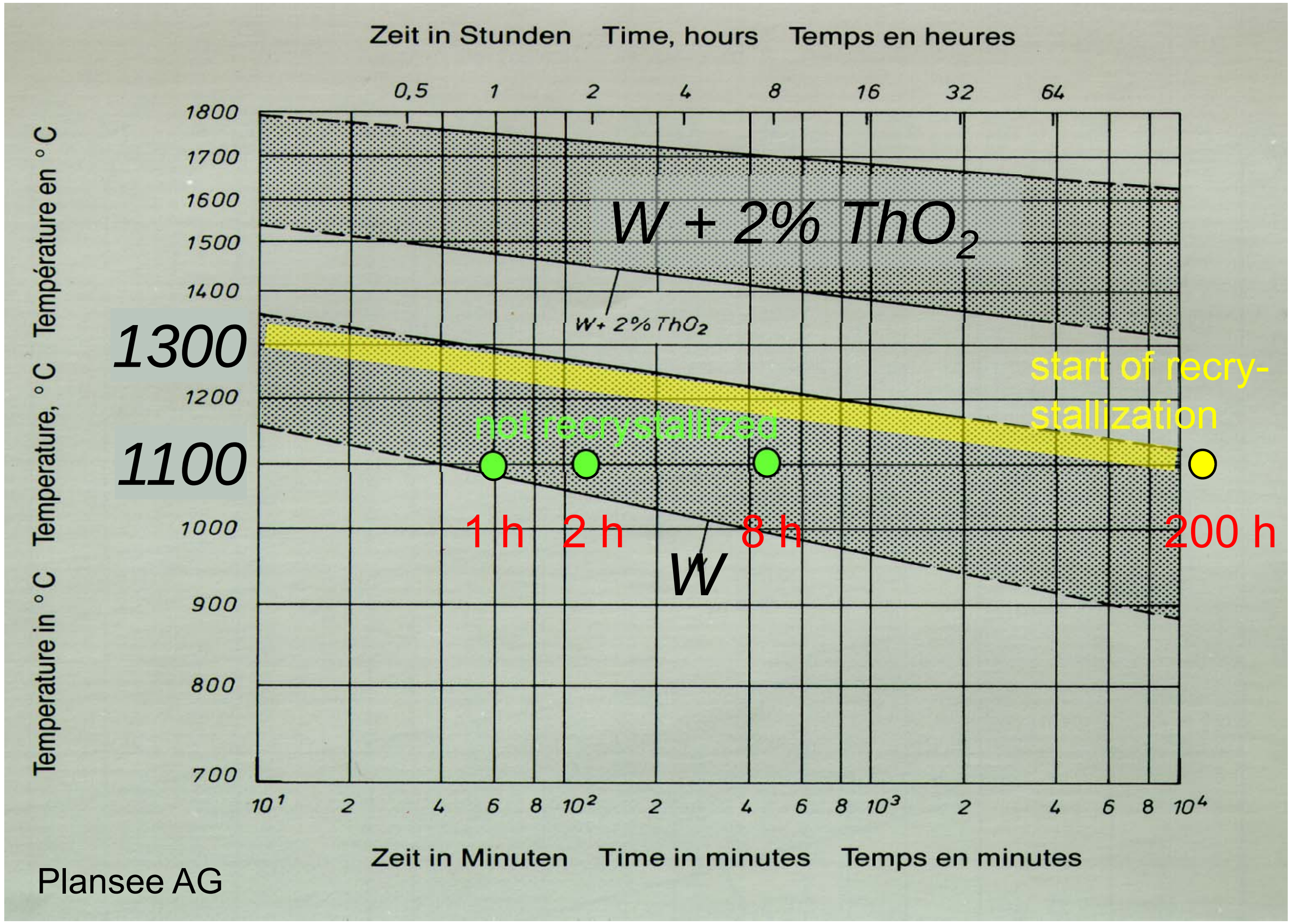
picture: ITER

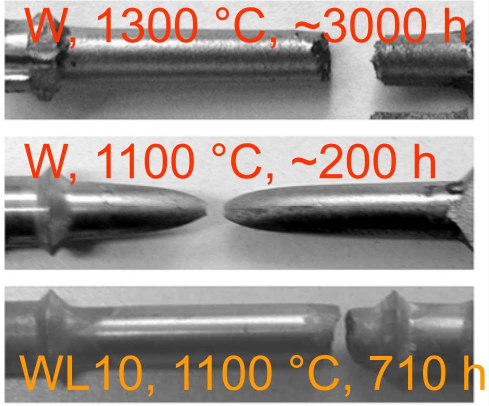


Tokamak fusion reactor

picture: PLANSEE SE

Tungsten alloys: Embrittlement issues
High temperature long-term creep limited by recrystallization

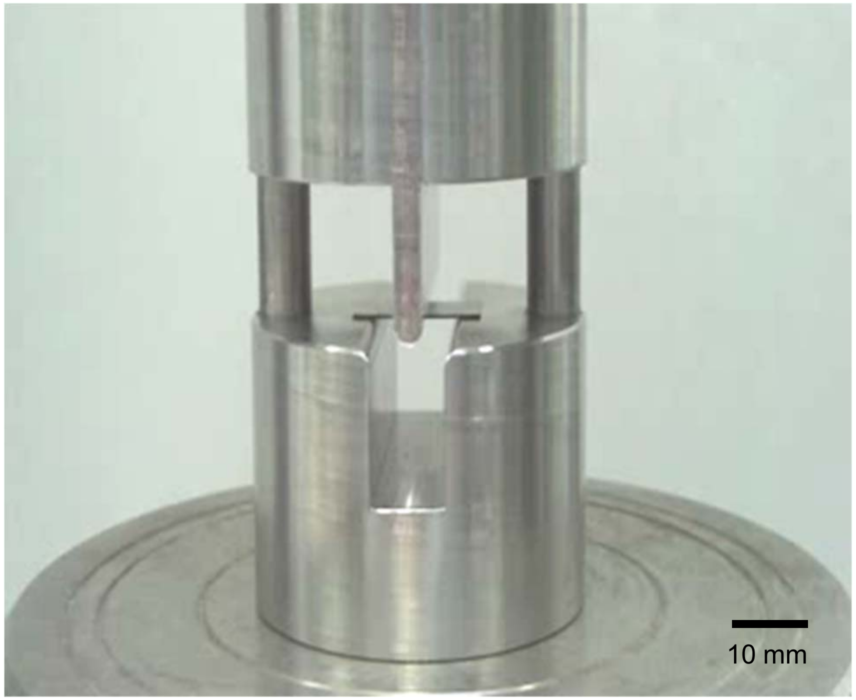




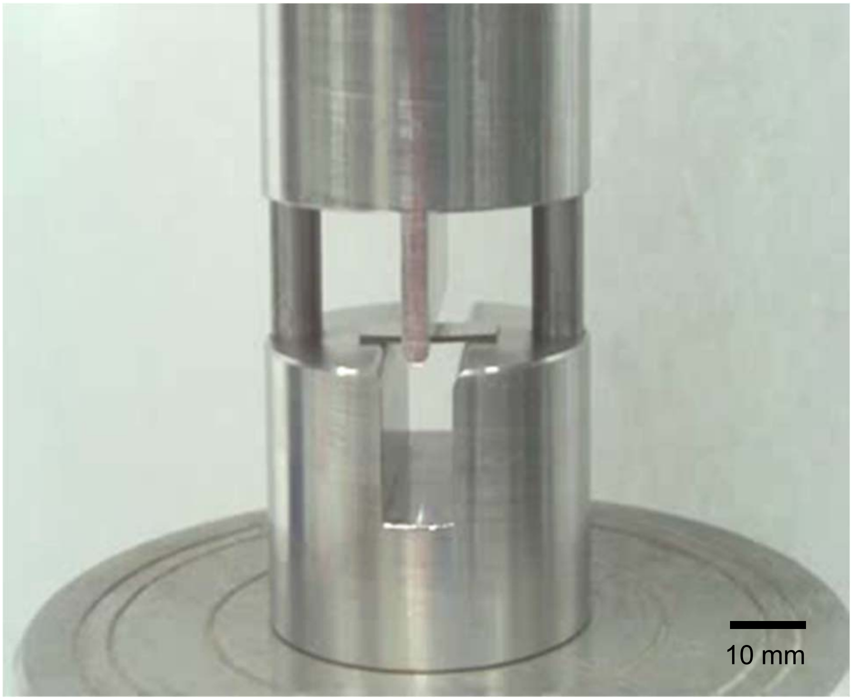
Refractory Materials for DEMO Divertors

Tungsten: Improvement of ductility and fracture toughness

Hot-rolled, coarse-grained W
Test temperature: RT



Severely cold-rolled, ultrafine-grained W; Test temperature: RT



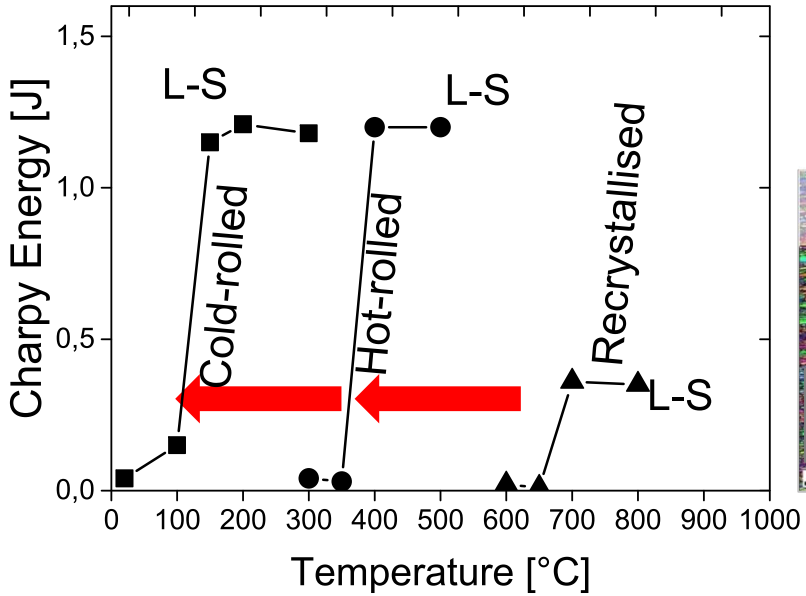
→ What happens during cold-rolling that makes W ductile?

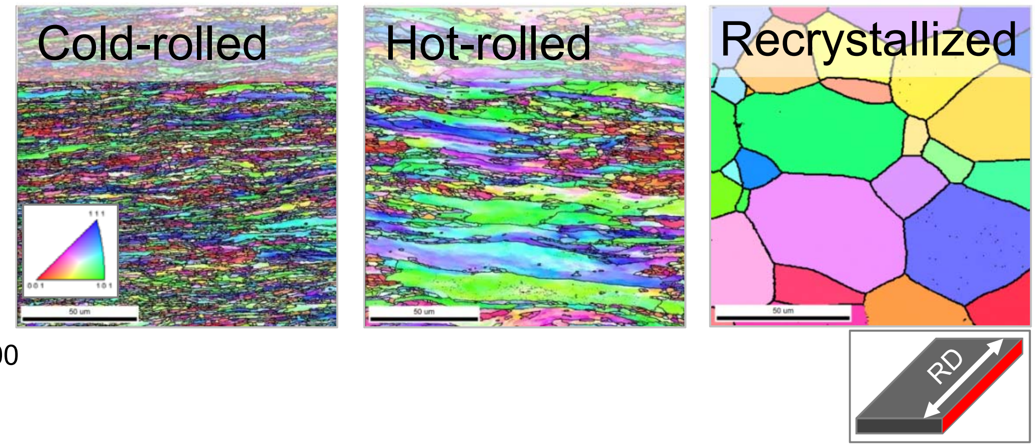
Refractory Materials for DEMO Divertors

Tungsten: Improvement of ductility and fracture toughness

Ductilisation of W through cold-rolling

- **Brittle-to-ductile transition**





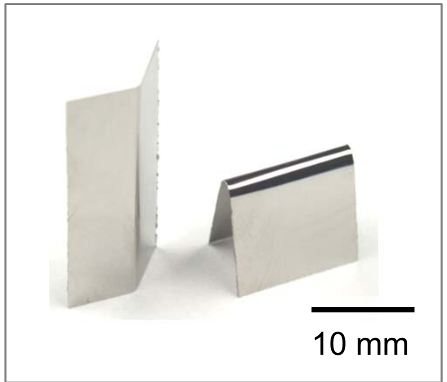
J. Reiser et al., Int. J. Refract. Met. Hard Mater. 54 (2016) 351–369.

Refractory Materials for DEMO Divertors

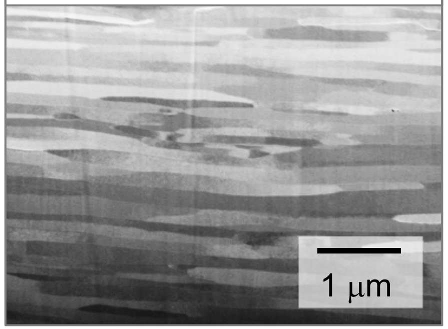
Improvement of ductility and fracture toughness:
→ Tungsten laminated composites

J. Reiser et al. Internat. Journ. of Refractory Metals & Hard Materials 69 (2017) 66-109

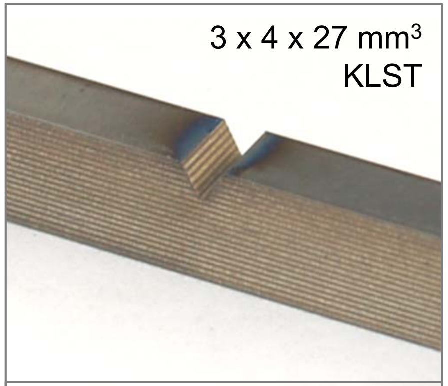
W-foil



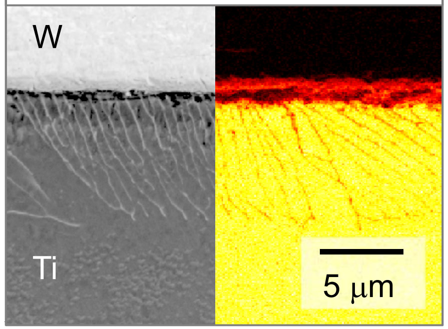
- Metal physics



W laminate plate

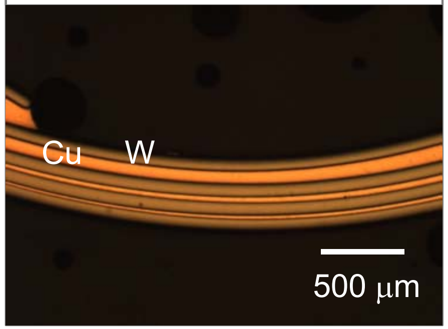


- Bonding and ageing

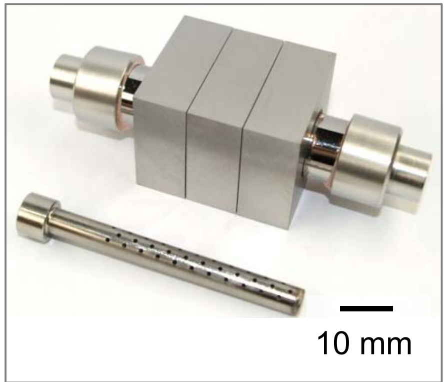


W laminate pipe

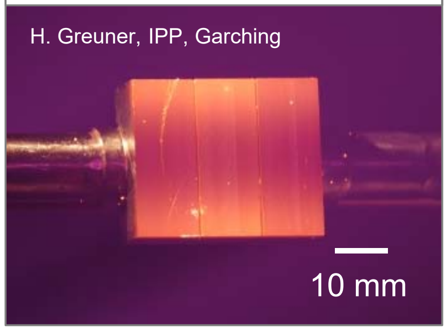
- Joining technology



Applications



- Fabrication and testing



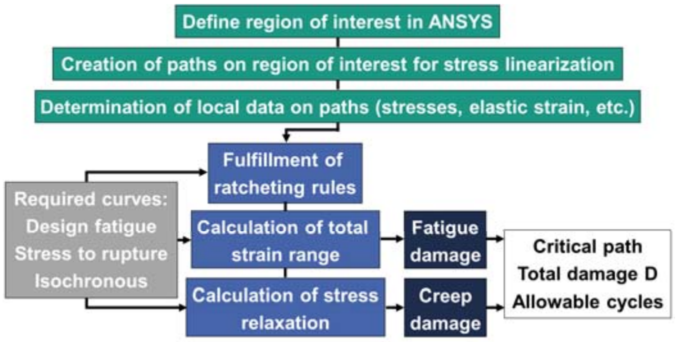
Outline

- DEMO reactors: current designs – blankets - divertors
- Fusion Reduced Activation Structural Materials (DEMO-oriented): recent progress
  - Reduced activation ferritic/martensitic steels
  - Oxide dispersion strengthened steels
  - Neutron irradiated steels – selected results
  - W alloys
- Database maturity & role of materials in fusion roadmaps

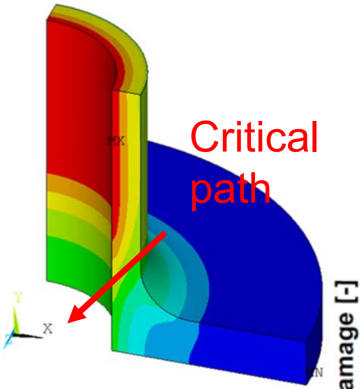
Creep-Fatigue Assessment (CFA) Tool

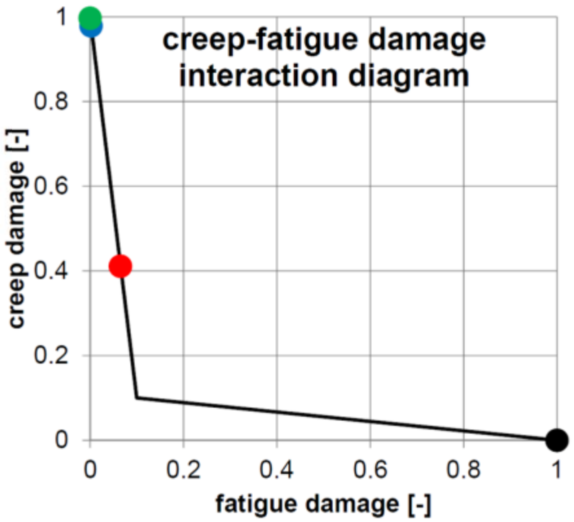
M. Mahler et al., NME 9 (2016) 535-538

- Objectives
  - Fast design evaluation of in-vessel components
  - Creep-Fatigue Assessment tool covering the complexity of design codes and new C-F rules for EUROFER97

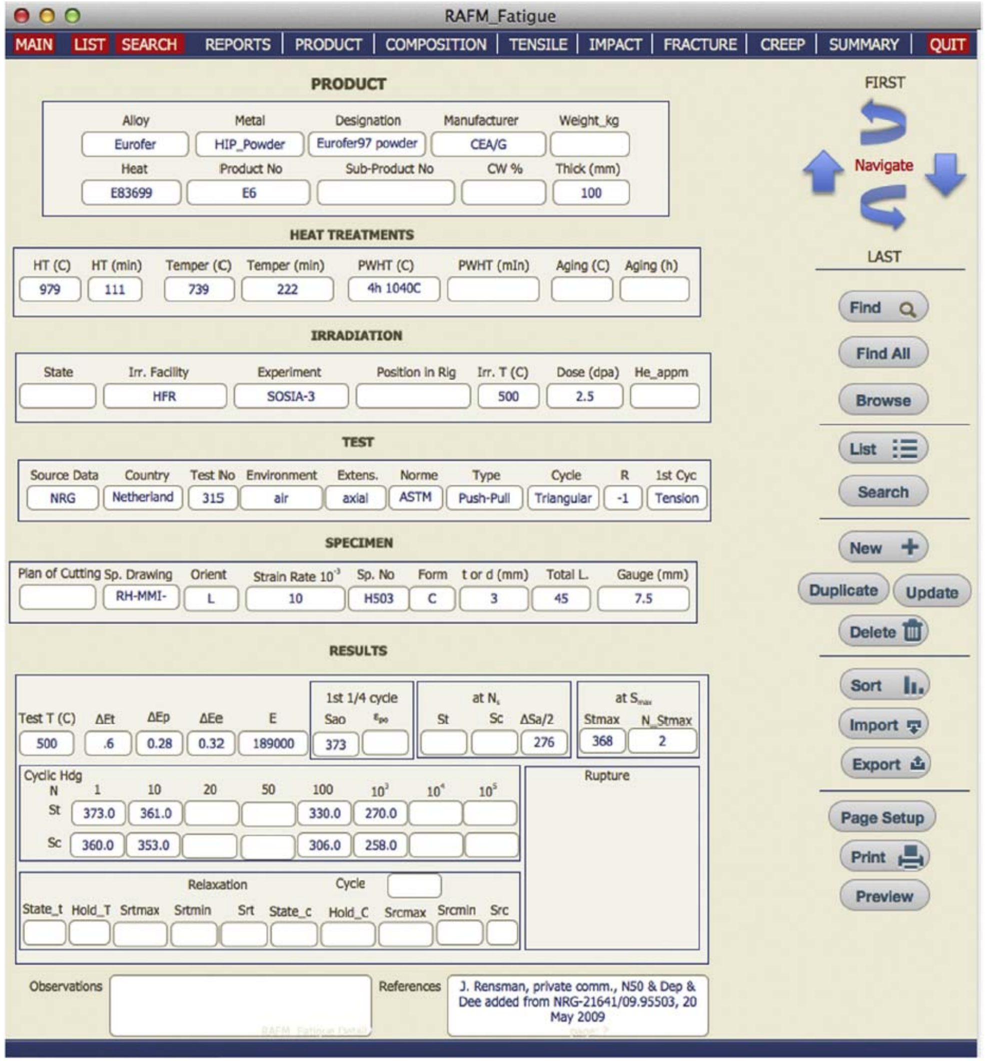


- CFA tool result output
  - Creep damage
  - Fatigue damage including cyclic softening
  - Allowable number of cycles
  - CFA tools are being developed as post-processing for ANSYS
  - Allow automated identification of critical region in 3D structure





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

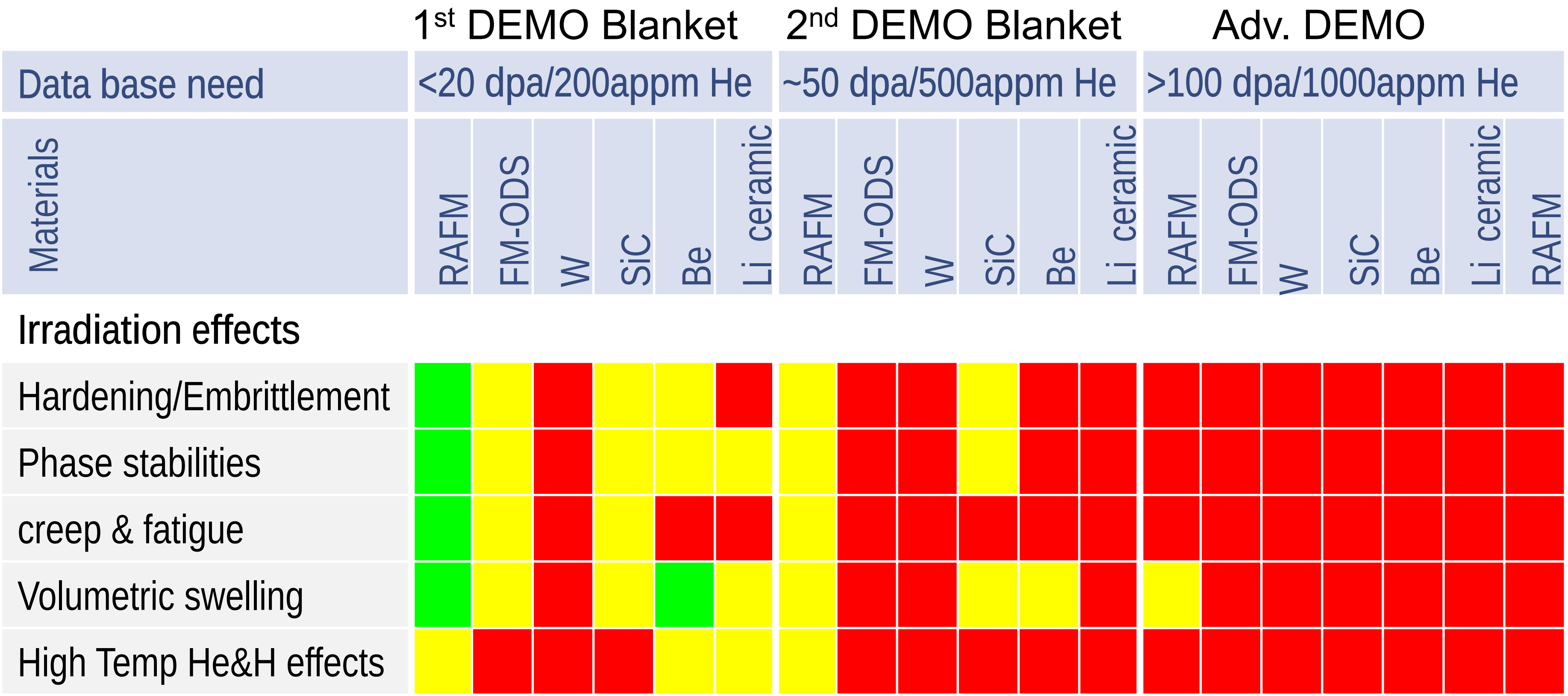
↓

RCC-MRx Design code implementation of the new material class

Irradiation effects: Materials Database Maturity

Karlsruhe Institute of Technology

Complimentary: R. Kurtz, PNNL



Adequate knowledge base exists

Partial knowledge base exists

No knowledge base

Note: He levels are only for FM steels

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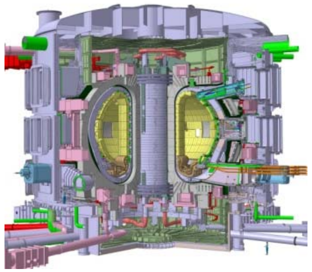
Institut für Angewandte Materialien

Role of Materials in Fusion Road Maps - simplified -

Karlsruhe Institute of Technology

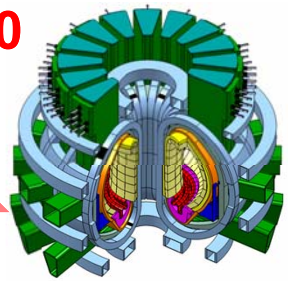
ITER,DT-phase beyond 2027



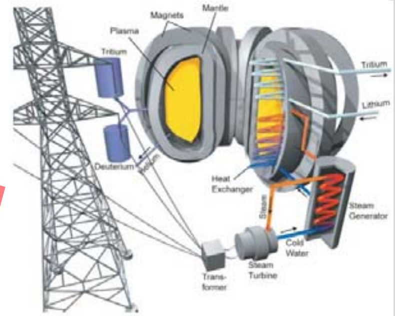
Early DEMOs

EDA end ♦ ♦ Start operat.

~2030



FPP beyond 2060



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?

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