TECH/3-2Rb

Plasma Exhaust and Divertor Designs in Japan and Europe Broader Approach, DEMO Design Activity

Topics are selected from *Chapter 4: Divertor and Power Exhaust* in final report of Broader Approach (BA) DEMO Design Activity（DDA) Phase-I (2020 Feb.)

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Power exhaust concepts and challenges for JA and EU DEMOs

- EU and JA BA-DDA study covers common aspects of divertor physics and engineering design: water-cooled single-null divertor and appropriate geometry for plasma detachment.
- Both concepts handle similar thermal heating power (P_{heat}), and require large total radiation fraction ($f_{rad} = P_{rad}/P_{heat} \geq 80\%$) in order to reduce the peak heat load ($\leq$10 MWm^{-2}):

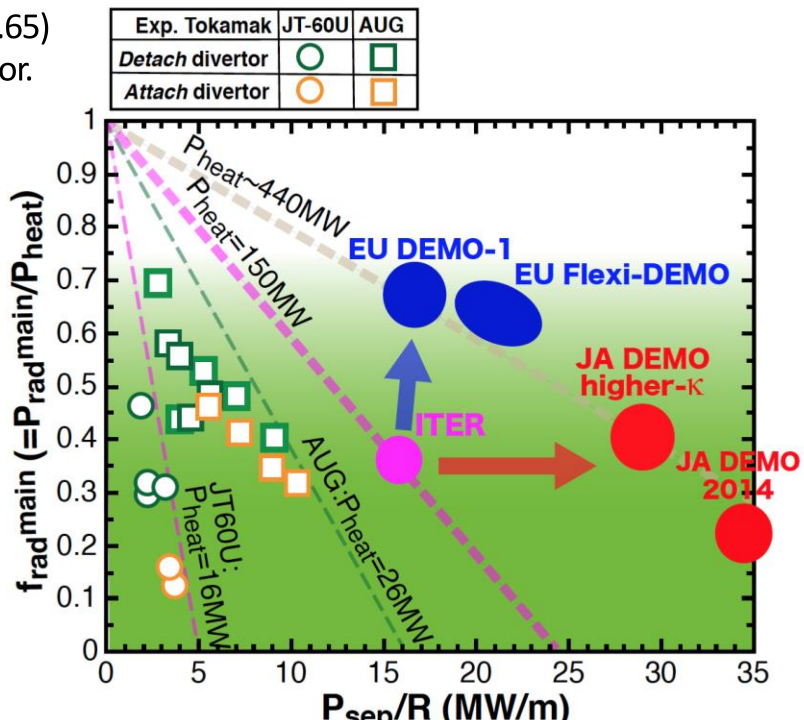
Divertor power handling is determined by requirements of f_{rad}^{main} and the plasma performance.

JA DEMO challenge (steady-state op.):
Lower I_p and higher HH with ITER-level f_{rad}^{main} ⇒ Large divertor power handling: P_{sep}/R ~30MWm^{-1}

EU DEMO challenge (pulsed op.):
Higher I_p and ITER-level HH with large f_{rad}^{main} by high-Z seeding ⇒ ITER-level P_{sep}/R =17MWm^{-1}

JA DEMO higher-κ proposal (κ_{95}=1.75) [1] rather than JA DEMO 2014 (κ_{95}=1.65) [2] is shown: having advantages on power exhaust in main plasma and divertor.

	Parameters	JA DEMO [1]	EU DEMO [3]
Power exhaust	**line-n_e^{main} ($10^{20}m^{-3}$)**	**0.86**	**0.87**
	n^{GW} ($10^{20}m^{-3}$)	0.73	0.72
	$n_{imp}^{main}/n_e^{main}$ (%)	0.6 (Ar)	0.039 (Xe)+Ar
	P_{heat} (P_α+P_{aux}, MW)	**435**	**457**
	P_{rad}^{main} (MW)	177	306
	f_{rad}^{main} (=P_{rad}^{main}/P_{heat})	**0.41**	**0.67**
	P_{sep} (MW)	258	154
	P_{sep}/R_p(MWm^{-1})	**30**	**17**



[1] Asakura, et al. Nucl. Fusion (2017), [2] Sakamoto, et al. IAEA FEC 2014&18, [3] Wenninger, et al. Nucl. Fusion 2017.

Divertor design and Power exhaust simulation

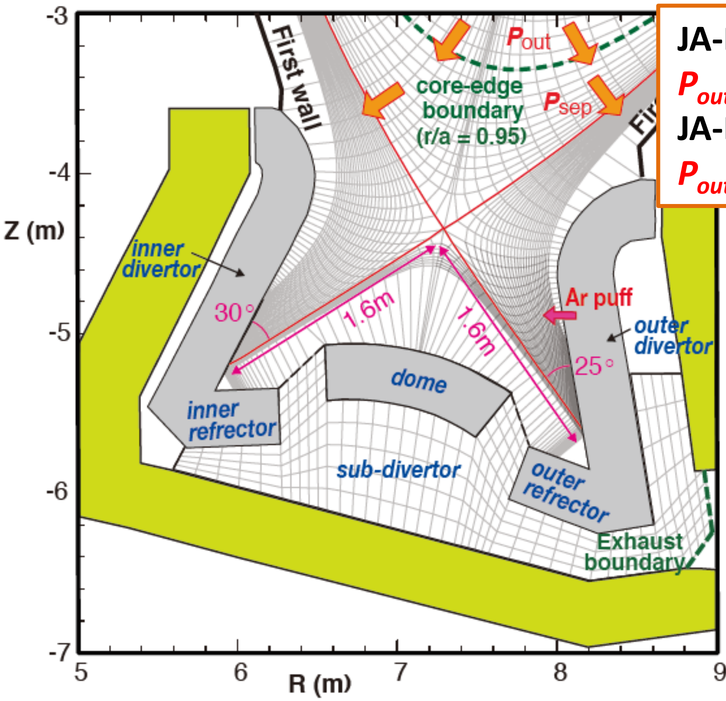
Conventional design concepts in JA & EU are based on the ITER divertor:

- Both DEMOs: Divertor leg is extended (outer L_{div}=1.6 m: 1.6 times longer than ITER).
- JA: Baffles cover divertor plasma for large P_{sep}/R handling ⇔ EU: Open and shallow geometry (ITER-level P_{sep}/R) to increase tritium-breeding area and reduce weight & process for remote maintenance.
- JA: Dome and reflectors are installed to enhance the neutral recycling near the strike-point. EU: Dome and reflectors are simplified ("liner") to reduce fast neutron flux to cassette and VV.

SONIC (JA-DEMO) and SOLPS-5.1 (EU-DEMO) simulations have been performed :

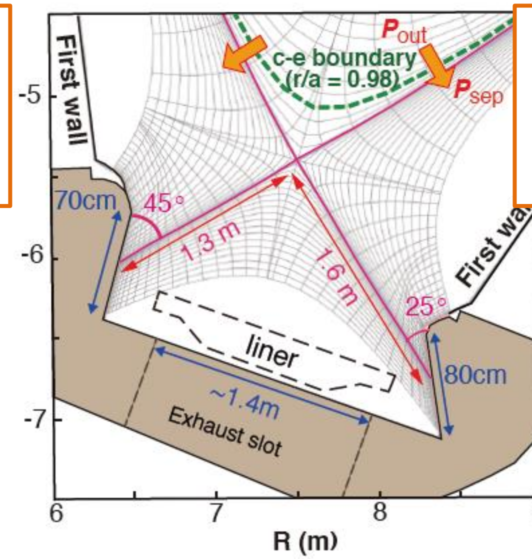
- Exhaust power (P_{out}= JA: 250-300 MW, EU:150-200MW) is given at core-edge boundary.

JA DEMO: SONIC simulation



JA-DEMO higher-κ:
P_{out} =250/P_{sep}~235MW
JA-DEMO 2014:
P_{out} =300/P_{sep}~283MW

EU DEMO: SOLPS simulation



Reference:
P_{out} = 150MW ~ P_{sep}
Power scan:
P_{out} = 180/200MW

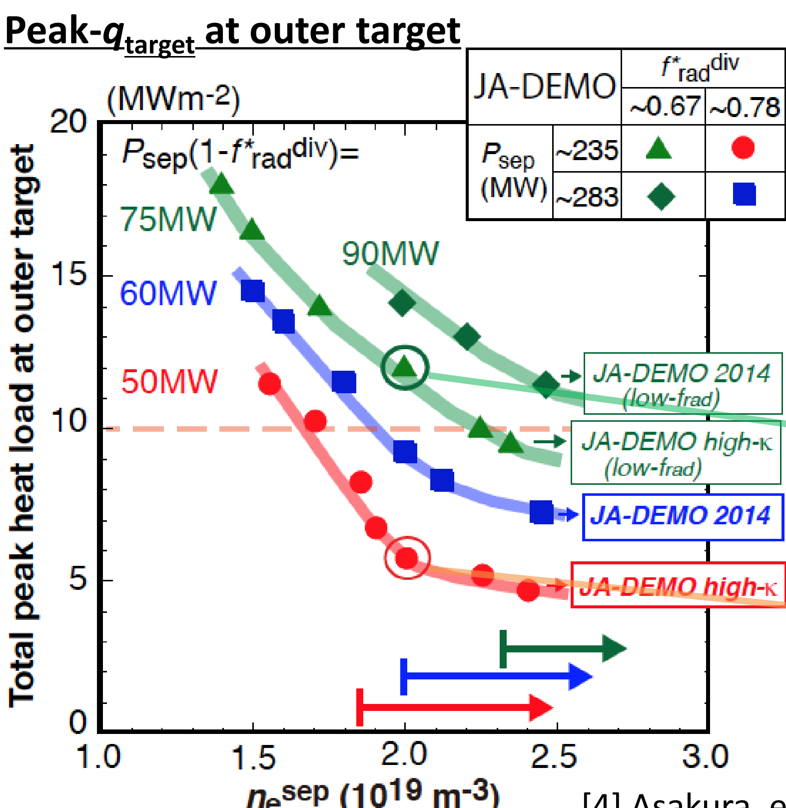
Note: $q_{//}$ profile width near separatrix is similar ($\lambda_{q//}$~3mm).

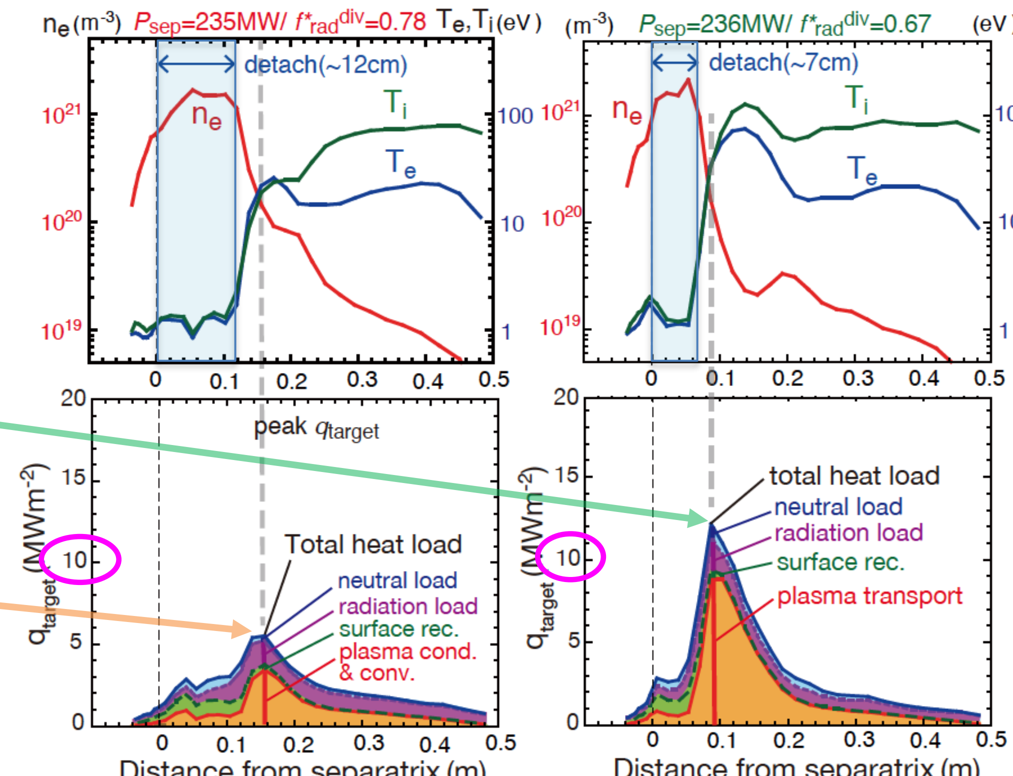
JA DEMO: Divertor operation in low density ($n_e^{sep} = 2\text{-}3\times10^{19}\,m^{-3}$)

Heat load can be reduced within the operation range ($q_{target} \leq 10$ MWm^{-2}) for $f^{*\,div}_{rad}$~0.8

In each density scan, Ar seeding rate was adjusted to obtain a given $f^{*\,div}_{rad} = (P^{div}_{rad} + P^{sol}_{rad})/P_{sep}$.

- **Higher-κ** (P_{sep}~235MW, $f^{*\,div}_{rad}$~0.8) reduces q_{target} ($\leq$ 6 MWm^{-2}), and allow enough operation margin.
- **JA DEMO 2014** (P_{sep}~283MW, $f^{*\,div}_{rad}$~0.8):
  Decreasing *detachment width*, and increasing T_i and T_e *of the attached plasma*.
  ⇒ peak-q_{target} is increased, and margin of the power handling ($\leq$ 10 MWm^{-2}) is reduced.
- ***Lower*** $f^{*\,div}_{rad}$ **~0.7** (P_{sep}~235 and 283 MW) cases:
  ⇒ **higher** n_e^{sep} (>2.3x10^{19}m^{-3} for *DEMO higher-κ*, >2.7x10^{19}m^{-3} for *DEMO 2014*) **is required**.





[4] Asakura, et al. Fus. Mat. Energy (2021).

EU DEMO: Divertor power handling by Ar seed for P_{sep}/R =16-22 MWm^{-1}

Heat reduction was achieved for all cases by increasing C_{Ar}^{SOL} (= $n_{Ar}/n_e)^{SOL}$ = 0.5-2.5%

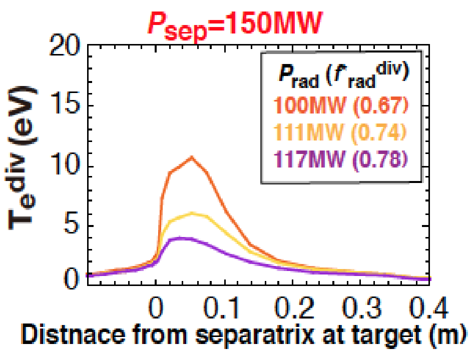
Geometry effect on plasma profile: *partial detachment* was not clearly seen in the open geometry.

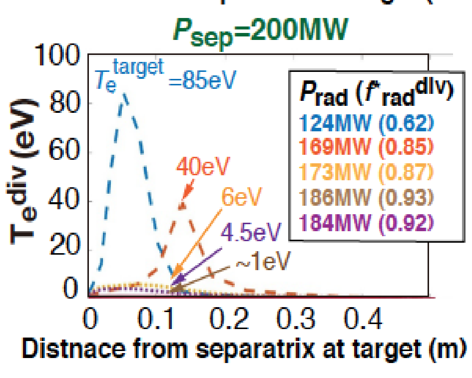
Baseline (P_{sep}=150MW): heat reduction ($q_{target} \leq$ 10MWm^{-2}) was achieved by increasing $f^*{}_{rad}^{div} \geq$0.7.
⇒ Low T_e^{div} ($\leq$ 5eV) was also produced *over wide outer target* for $f^*{}_{rad}^{div} \geq$ 0.8 ($C_{Ar}^{sol} \geq$ 0.8%).

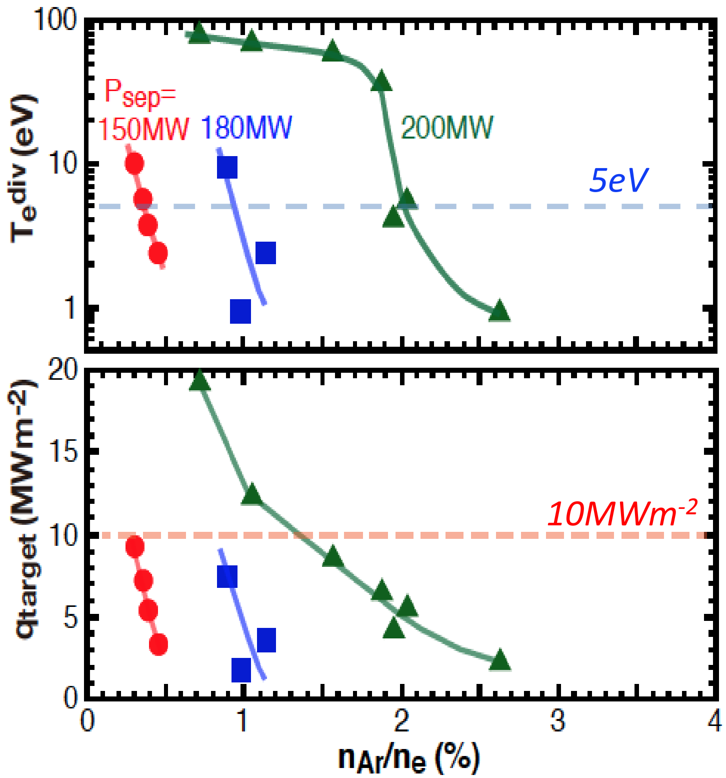
Larger P_{sep} case: q_{target} reduction was achieved ($f^*{}_{rad}^{div} \geq$ 0.75) ⇒ low T_e^{div} ($\leq$ 5eV) was required in higher $f^*{}_{rad}^{div} \geq$ 0.9 ($C_{Ar}^{sol} \geq$ 2%). Detachment (T_e^{div}~1eV) is seen in very high $f^*{}_{rad}^{div}$ ($\geq$ 0.93).

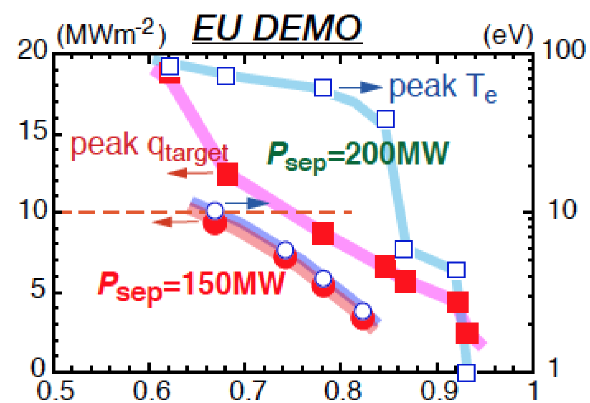
Ar seeding scan for P_{sep}=150/ 180/ 200MW

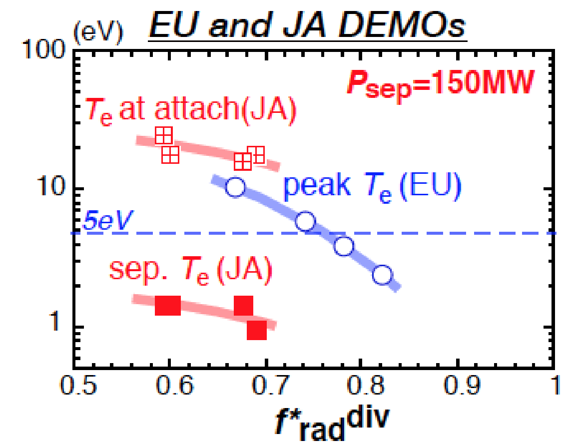
Note) n_e^{sep}~2.8x10^{19}m^{-3} is higher than JA-DEMO











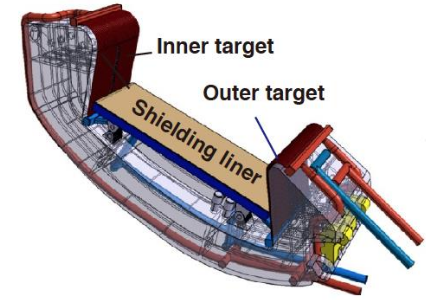
Design concepts for water-cooling DEMO divertor

W-PFC & CuCrZr-pipe is common baseline design based on the ITER divertor.

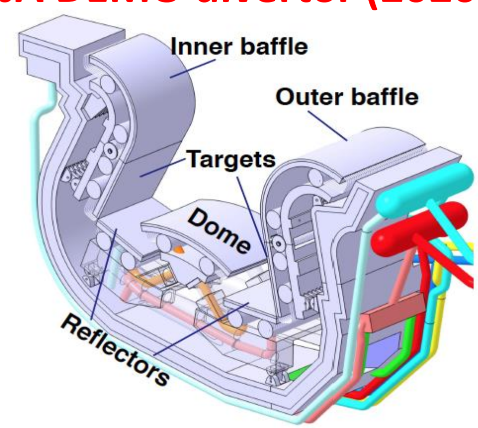
- ITER-like monoblock target is the first candidate for high heat load plasma facing component.
- Remote maintenance concept is also common issue: JA divertor is larger weight.
- Mechanical property of Cu-alloy and interlayer (1-2 dpa) may firstly determine PFC life time (maintenance) under DEMO n-irradiation condition, while coolant-temp. is increased (130-200°C).
  EU: R&D of ITER-like target to reduce stress and to strengthen pipe& interlayer.
  JA: applying ITER-like target *near the strike-points* (lower dose).

		EU DEMO [5, 6]	JA DEMO[1, 7]
Total number of cassettes		48	48
Number of divertor maintenance ports		16	16
Weight of one cassette (ton)		11	23
Target	PFC & Heat sink	**W&CuCrZr**	**W&CuCrZr**
	Water T(°C)/ Pressure(MPa)	**130/5**	**200/5**
	Dose on pipe/fpy (dpa)	<10	<1.5
Dome/Baffle	PFC & Heat sink	W&CuCrZr (***liner***)	W&F82H
	Water T(°C)/P(Mpa)	180/ 3.5	290/ 15
	Dose on pipe/fpy (dpa)	<10	<8.5
Cassette	Material	EUROFER97	F82H
	Water T(°C)/P(Mpa)	180/ 3.5	290/ 15
	Dose on struct. material/fpy (dpa)	<6	<3

EU DEMO divertor (2019)



JA DEMO divertor (2020)

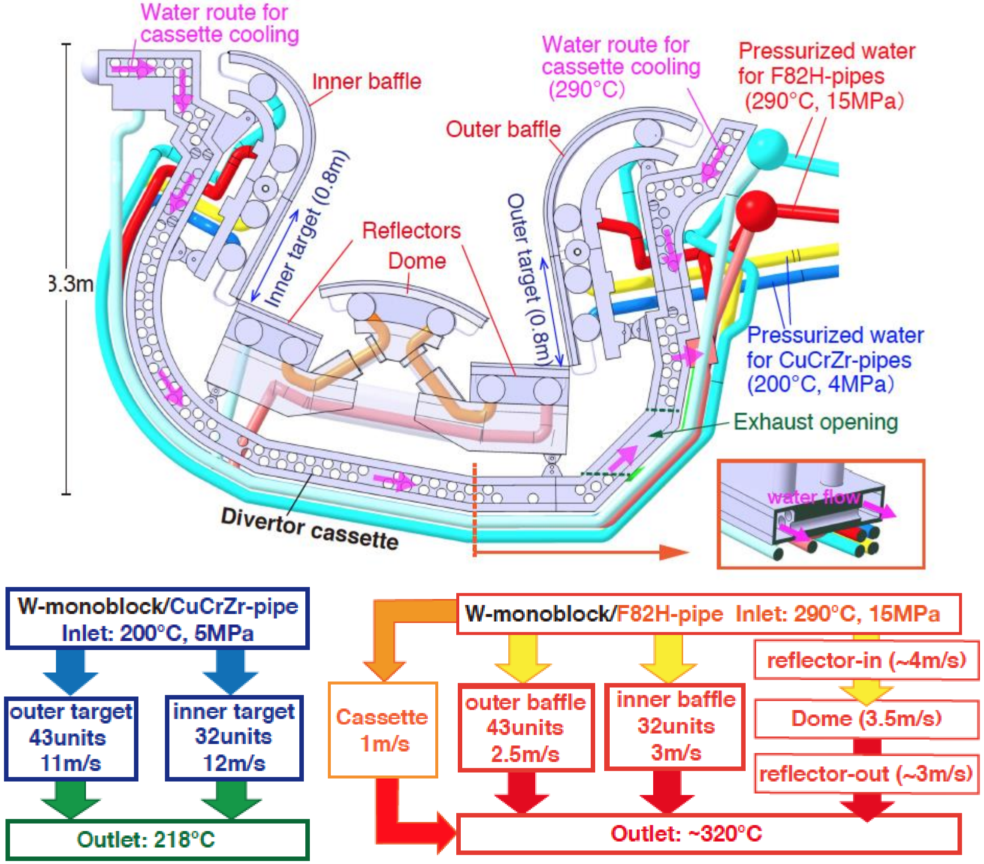


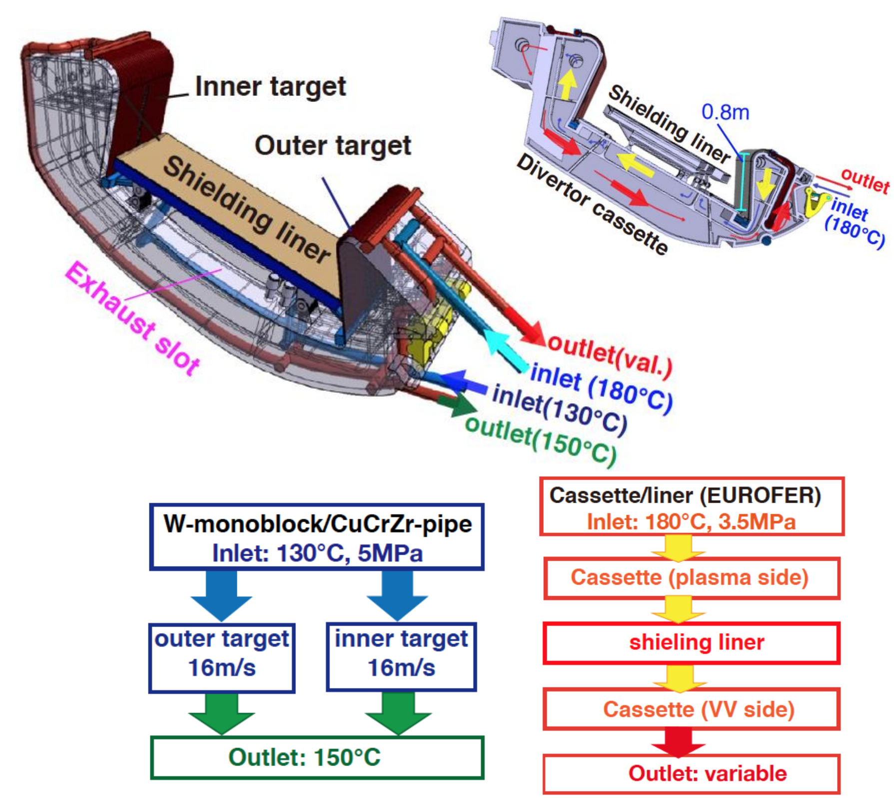
[5] J.H. You, et al., Fus. Eng. Des. (2017). [6] J.H. You, et al., Nucl. Mat. Energy (2018). [7] Asakura, et al. Fus. Eng. Design (2018).

Design concepts of divertor water-cooling for DEMOs:
Optimization of two water routes is required. Coolant-temperature is a design issue.

***Parallel* cooling route for inner and outer targets** is designed to avoid fast flow speed at inboard.

- **JA:** W-MB with CuCrZr/F82H-pipes was arranged for Plasma Facing Components with high/low heat load and low/high n-flux: $T_{coolant}$ =200˚C is used for CuCrZr-pipe to reduce embrittlement [8].
- **EU:** $T_{coolant}$ is reduced (130˚C) to increase the critical heat flux larger than 48 MWm^{-2} (for 150˚C) [9]. $T_{coolant}$(180˚C) for cassette (EUROFER97) to ensure sufficient fracture toughness at n-damage (<6 dpa).





Note: Total $P_{div}^{thermal}$:350MW +$P_{div}^{nuclear}$:120 MW is assumed.

[8] Li-Puma, et al, Fus. Eng. Des. (2013). [9] You, et al, Fus. Eng. Des. (2018)

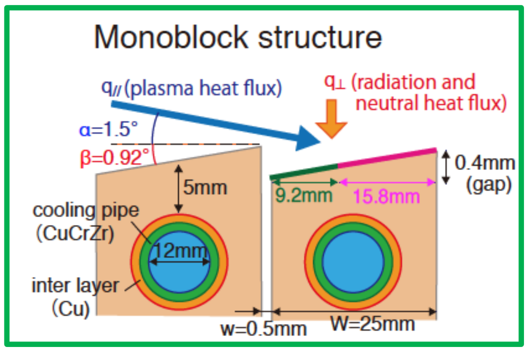
Heat analysis of W-monoblock and CrCrZr heat sink for JA DEMO

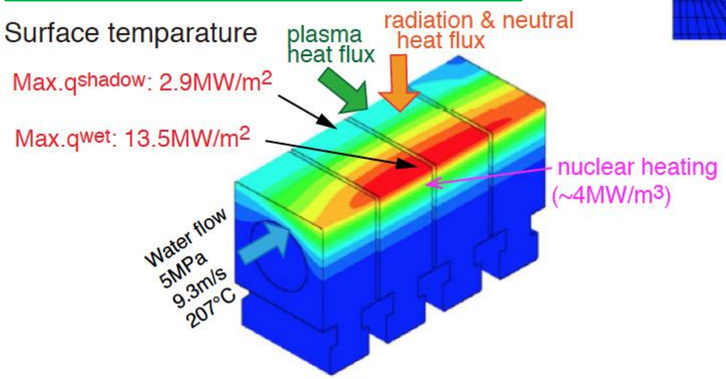
Acceptable power load depends on heat load components and target design

Heat load profile (plasma, radiation&neutral, nuclear heat) is applied to *ITER-like fish scale target*: peak heat load to flat tile (9.1 MWm^{-2}) corresponds to 13.5 MWm^{-2} to the wetted area.

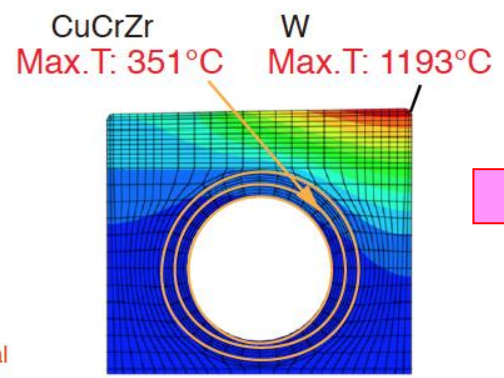
- The peak heat load is a critical, i.e. operation just below recrystallization temperature of W (1200°C). Irradiation-creep/softening of CuCrZr-pipe (351°C) is also anticipated.
- Max. heat flux from the pipe to coolant (18MWm^{-2}) is well below Critical Heat Flux (35MWm^{-2}).

Power exhaust by 200°C water is acceptable even for larger heat load on W (surface-T_W > $T_{recystalization}$).



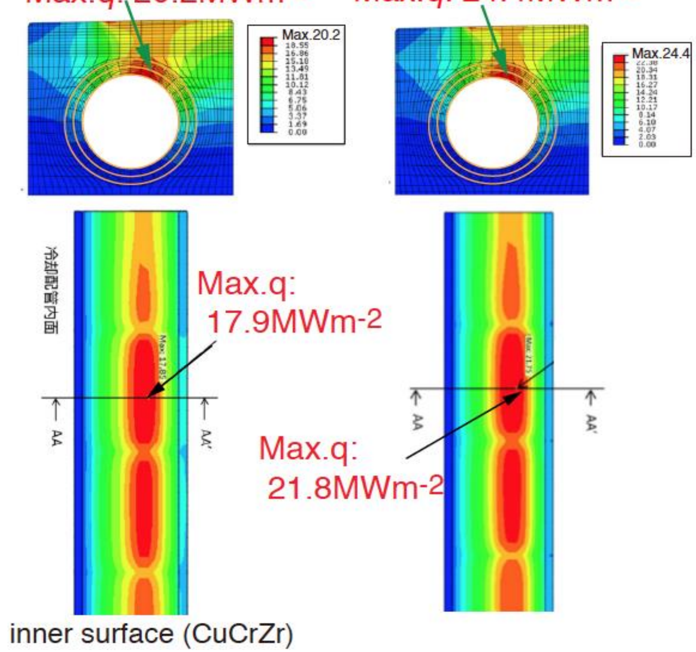


Temperature in MB target



Heat flux from CuCrZr-pipe to water

max.T_W~1200°C case **max.T_W~1400°C case**



In addition, **elasto-plastic stress analysis** was performed by repeating higher heat load (max. q: 15MWm^{-2}, W surface: 1400°C)⇒ mechanical strain on CuCrZr pipe (~0.25%) was not critical, while max. temp. became 365°C.

Summary: BA DDA for Power exhaust concept and Divertor design

Common design issues for Power exhaust and Divertor have been investigated in JA and EU.

- Requirements of f_{rad}^{main} and the plasma performance determined **divertor design concept:**
  Challenges of **JA** (steady-state): ITER-level f_{rad}^{main} (high *HH*) and larger P_{sep}/R =30-34MWm^{-1} and
  EU (pulsed): large f_{rad}^{main} (ITER-level *HH*) for ITER-level P_{sep}/R, contribute to optimize future reactor des.
  ⇒ **Same leg length (1.6 m: longer than ITER)** but **different geometry** (**JA:** ITER-like closer baffle, **EU:** rather open without dome and baffle) were proposed as baseline designs.

Power exhaust simulations of P_{sep}~JA: 250-300 MW, EU:150-200MW with Ar seeding have been performed, by using **JA: SONIC** and **EU: SOLPS5.1**, with similar $q_{//}$ profile width ($\lambda_{q//}$~3mm):

- **Large divertor radiation fraction ($f^*{}_{rad}^{div} = P_{rad}^{div}/P_{sep} \geq 0.8$)** was required to reduce peak-q_{target} ($\leq$ 10MWm^{-2}) and $T_{e,i}$ **in n_e^{sep} range** (JA: 2-3x10^{19}, EU:~2.8x10^{19}m^{-3}) **lower than ITER**.
- Divertor geometry affected *partial detachment profile*.

Integrated design of *divertor target*, *cassette* and *coolant pipe routing* has been developed:
water cooled ITER-like target (W-PFC and Cu-alloy heat sink) is a common baseline design.

- For a year long operation *under DEMO-level n-irradiation*, deterioration of mechanical properties of CuCrZr heat sink and Cu-interlayer is anticipated ⇒ restrictions of q_{target} & $T_{surface}$.
- **EU** has been developing W-MB target components to reduce stress and strengthen pipe & interlayer.
- **JA:** 2 cooling loops: *Cu-alloy heat sink* (target, 200°C) and *F82H heat sink* (baffle/cassette, 290°C).

⇒ Cu-alloy/interlayer concept and Operation-$T_{coolant}$ for DEMO divertor are common critical issues.

Joint studies on Plasma exhaust and Divertor design are extended to BA DDA Phase-II (-2024).

Development of water-cooled target components for EU DEMO
Mechanical property of heat sink and joint/interlayer is a key for Cu-alloy application

Development of candidate target concepts based on W-monoblock and Cu-alloy technologies:

- Divertor target concepts are developed for water-cooled targets:
  All are based on a Cu-alloy pipe with swirl tape to increase the heat transfer at the pipe wall:

Baseline:

- **ITER-like** MB & CuCrZr pipe with Cu-interlayer

→ Reducing thickness and width to reduce thermal stresses and prevent vertical cracking.

Reducing stress and strengthen pipe & interlayer:

- **Thermal break** interlayer /CCFC
- W wire-reinforced Cu **composite pipe** /IPP
- **Functionally graded** (W/Cu) **interlayer** /CEA
- W particle-reinforced Cu **composite** heat sink **block** /IPP

- Mock-ups of each concept have been fabricated and 100-level cyclic tested in a high-heat flux facility at 20-25 MW/m² with 130°C (20°C) water cooling

Note: He-cooling by multi-jet pipe (KIT) is an option.

Water cooled target concept

Target concepts	Interlayer	Heat sink
ITER-like (W monoblock)	Cu (1 mm)	CuCrZr pipe
Thermal break (W monoblock)	Cu (1.5 mm) bores	CuCrZr pipe
Composite pipe (W monoblock)	None	Wf/Cu pipe
FGM interlayer (W monoblock)	W/Cu (0.5 mm)	CuCrZr pipe
Composite block (W tiles)	None	W_p/Cu block

Divertor target concepts for EU DEMO

[10] J.H. You, et al., J. Nucl. Mater. (2021).

Max-Planck-Institut für Plasmaphysik

(New) Materials and Components for the DEMO Divertor

R. Neu[1,2], A.v. Müller[1], B. Curzadd[1,2], J. Riesch[1], J. W. Coenen[3,4], H. Greuner[1], T. Höschen[1], K. Hunger[1], G. Schlick[5], U. Siefken[6], E. Visca[7], J.H. You[1]

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Boundary conditions for plasma facing components (PFCs)

W as PFM:
- high melting point
- low sputtering yield
- low tritium retention
- low vapour pressure

no overlap of optimum operational conditions

Cu (alloys) as heat sink:
- high thermal conductivity

principle layout

W surface

T < 1500 K avoid recrystallization

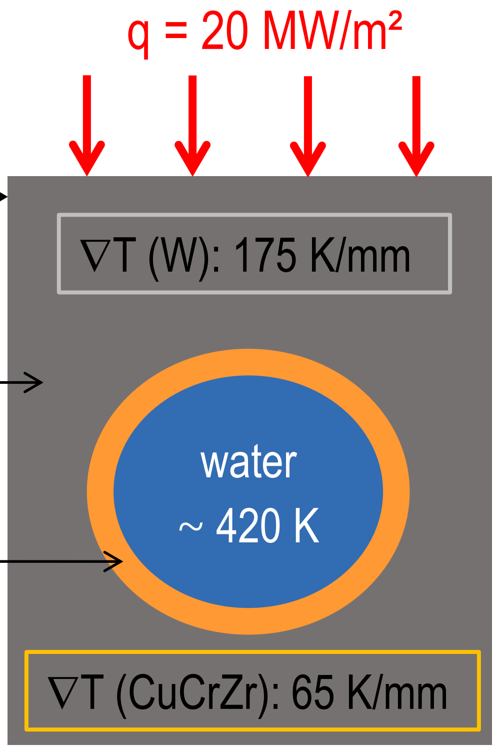
W bulk

T > 1100 K avoid brittleness

Cu bulk

T > 350 K avoid embrittlement

T < 620 K avoid loss of strength



deep cracking observed for ITER mock-ups during cycling at 20 MW/m² due to low cycle fatigue (crack initiation) and brittle behaviour during cool down

[M. Li FED 101 (2015) 1]

[G. Pintsuk, FED 88 (2013) 1858]

Advanced materials and designs for PFCs

Concepts for improving PFCs for the European DEMO

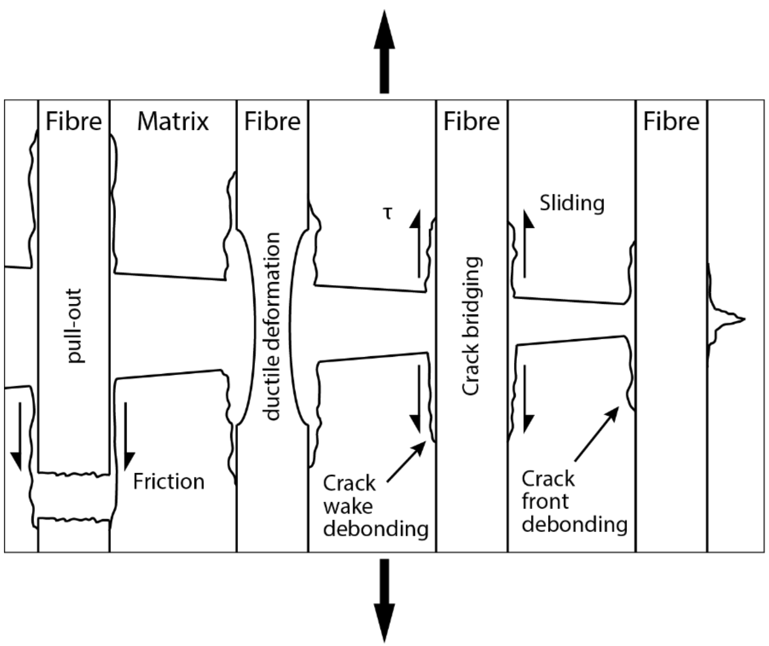
- optimisation of shape/size → lower temperature
- thermal break concept → homogenisation of temperature
- adaption of CTE by **functionally graded material**

} reduction of stresses

- improvement of material properties (strength, toughness) by **composites / fibre reinforcement**

⇒ **combination of concepts can further optimise function and lifetime**

[J.H. You, J. Nucl. Mater., 544 (2021) 152670]



Extrinsic toughening mechanisms in fibre reinforced materials:

→ stress redistribution by local energy dissipation

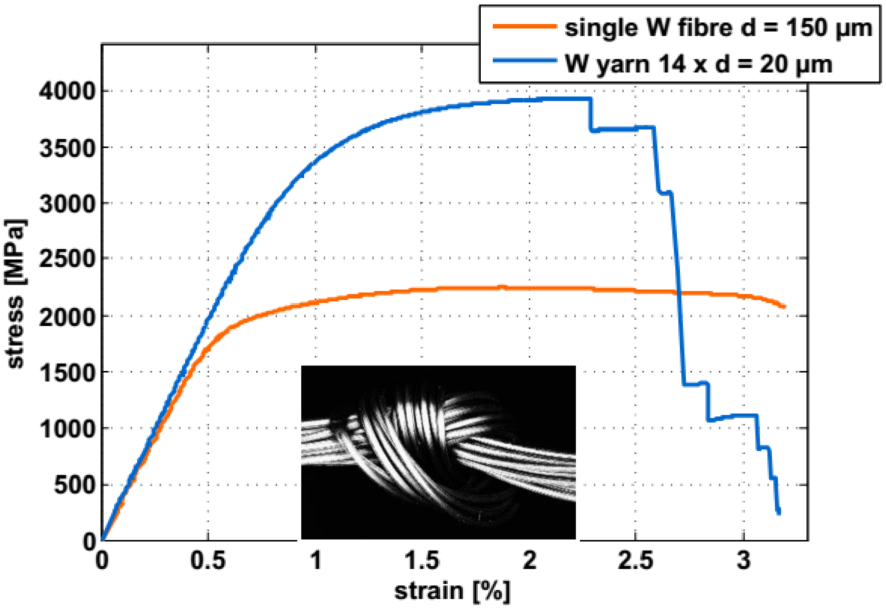
→ effective below DBTT & under embrittled state

[J. Riesch, Phys. Scr., T167 (2016) 014006]

W fibres and yarns

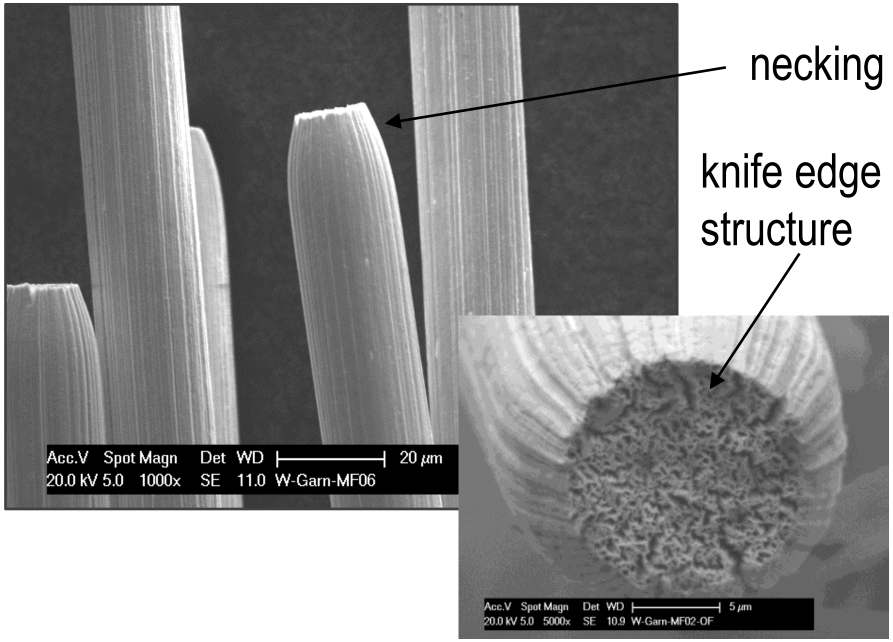
W fibres with small diameter (16 – 150 µm)

- highly deformed/fine grains:
  high strength, ductile already at room temperature
- potassium doping: **stable against recrystallization** up to ~ 2000 °C
- yarns show **increased flexibility and confirm higher strength** of thinner fibres



$$\sigma_{UTS} > 3900\ MPa$$

[J. Riesch, Phys. Scr. T170 (2017) 014032]



ductile failure behaviour confirmed

[H. Gietl, J. Comp. Mater. 52 (2018) 3875]

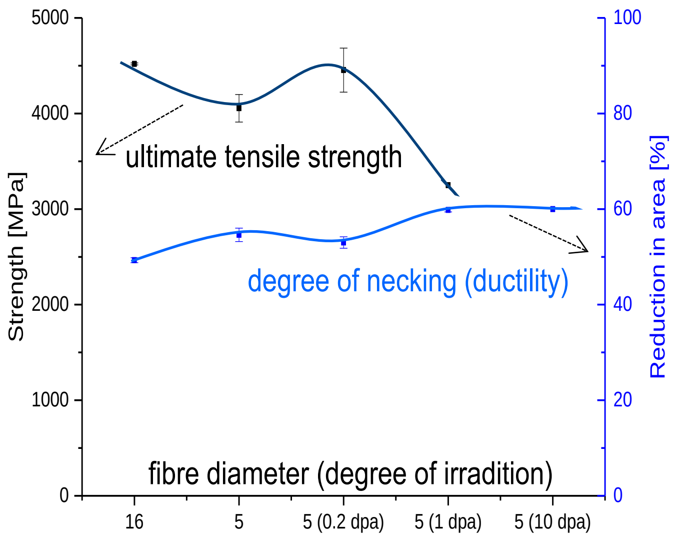
Ductile behaviour of irradiated W fibres

Exceptional properties (strength & ductility)
⇒ ideal ingredient for composites for high-temperature applications

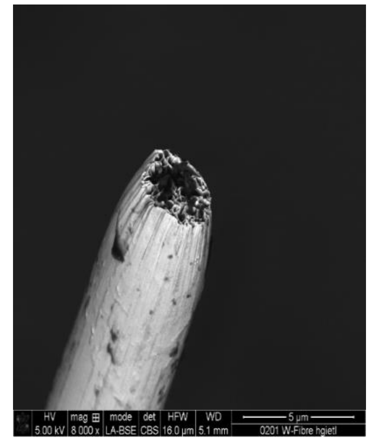
⇒ successful development **W_f / W and W_f / Cu** composites

Small fibre diameter (∅=5 µm) allows damage of complete volume by W ions (irradiation by 20.5 MeV W-ions simulating n-damage)

⇒ no strong degradation up to 10 dpa

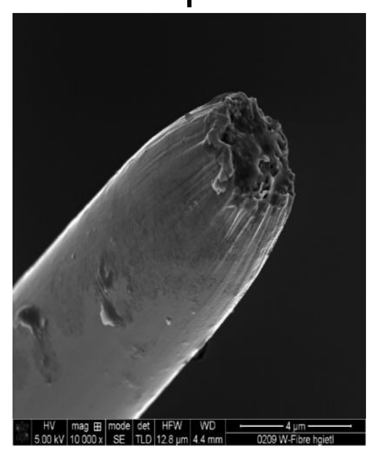


0 dpa

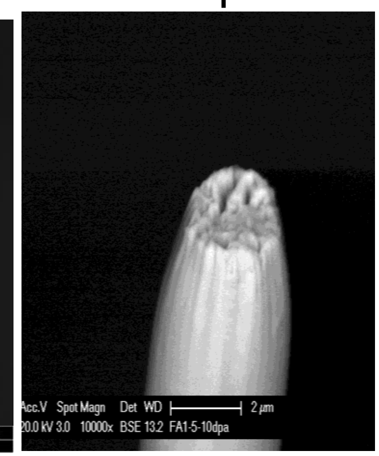


0.2 dpa

1 dpa

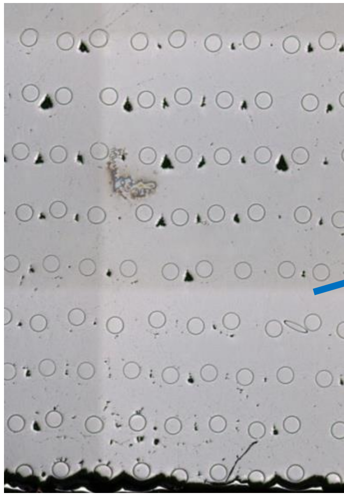


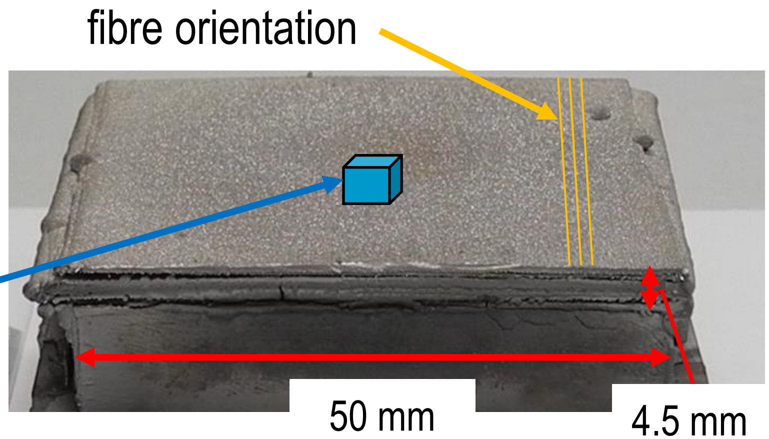
10 dpa



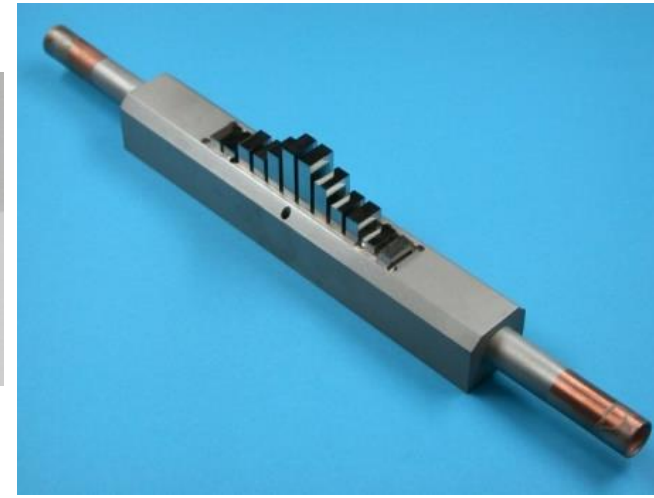
[J. Riesch, 17th Int. Conf. on PFMC, 2019, Eindhoven]

Production of bulk tungsten fibre reinforced tungsten (W_f / W)





[J. Riesch, Phys. Scr., T167 (2016) 014006]

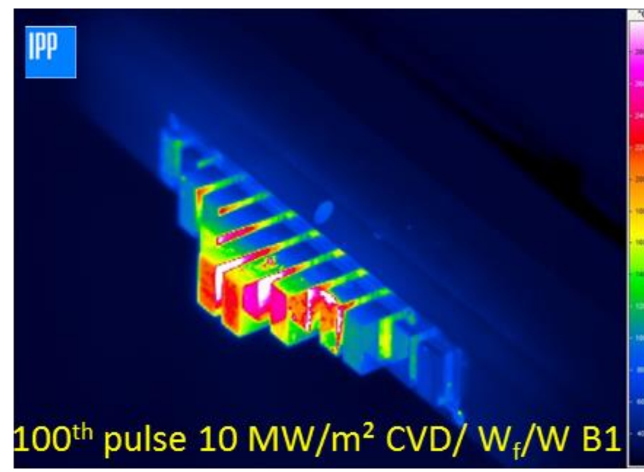


actively cooled W_f/W mock-up

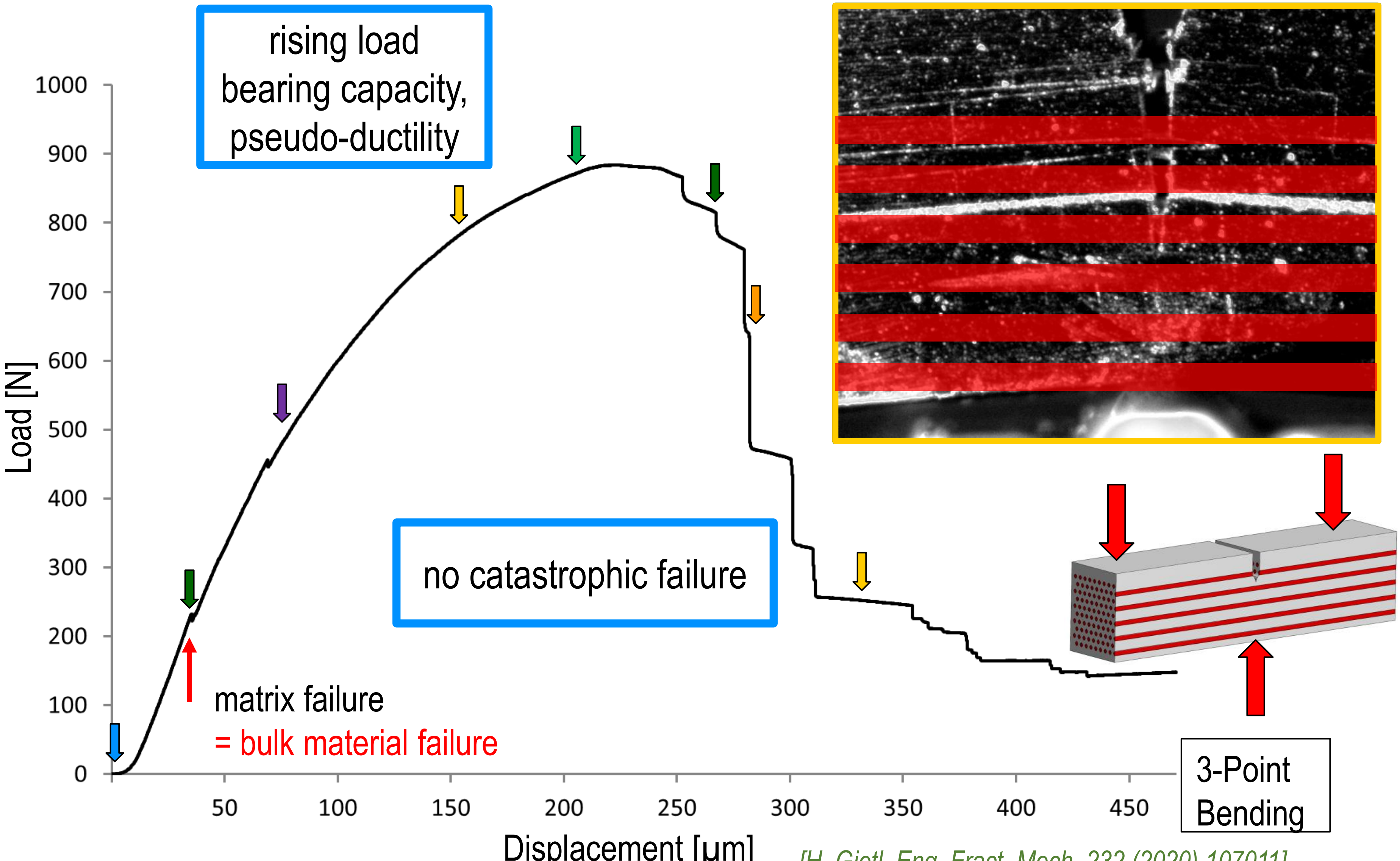
Production of W_f / W by Chemical Vapour Deposition
(CVD, decomposition of WF_6)

- layers of woven W wire fabric (distance 200 – 300 µm): K-doped, ø 150µm, 1µm Yttria interface layer
- fibre volume fraction ≈ 10 - 30%, unidirectional orientation
- density ≤ 99%

Samples for mechanical and high heat flux testing

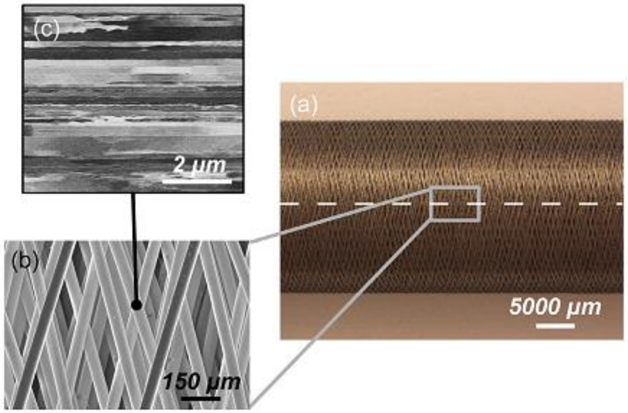


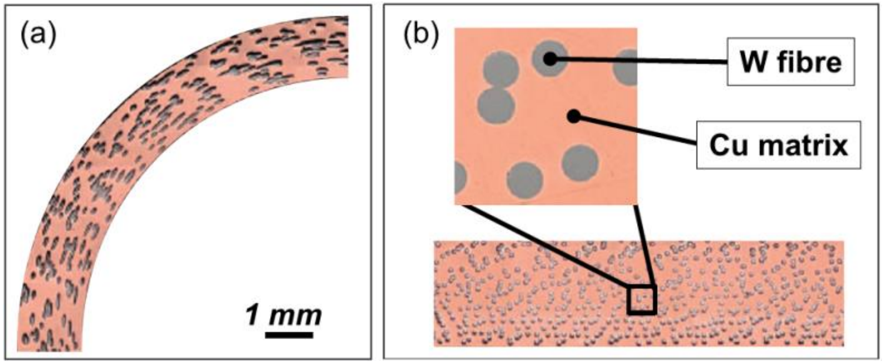
Testing of mechanical properties of W_f / W



[H. Gietl, Eng. Fract. Mech. 232 (2020) 107011]

W fibre-reinforced Cu heat sink





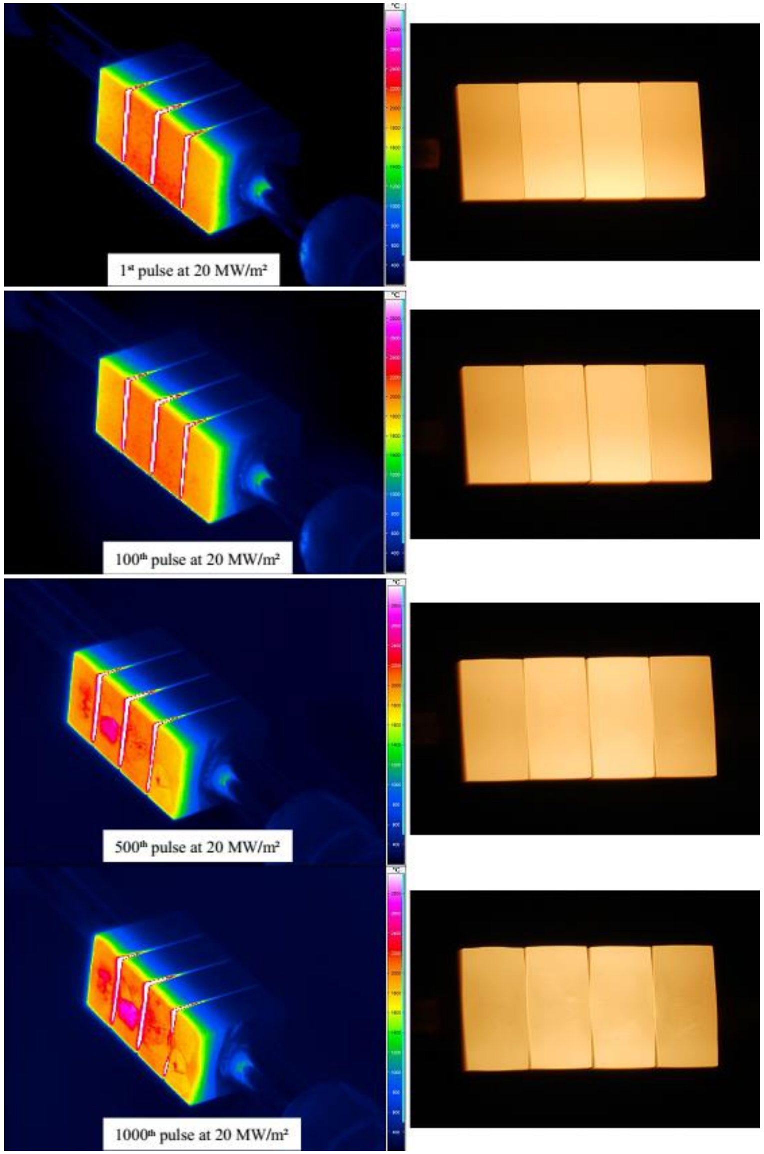
cylindrical multi-layered braiding made out of continuous W fibres or yarns with a nominal diameter of 50 μm

micro-sections of a W_f - Cu heat sink pipe produced by means of liquid Cu melt infiltration

[A.v. Müller, Phys. Scr. T171 ***2020*** *014003]*

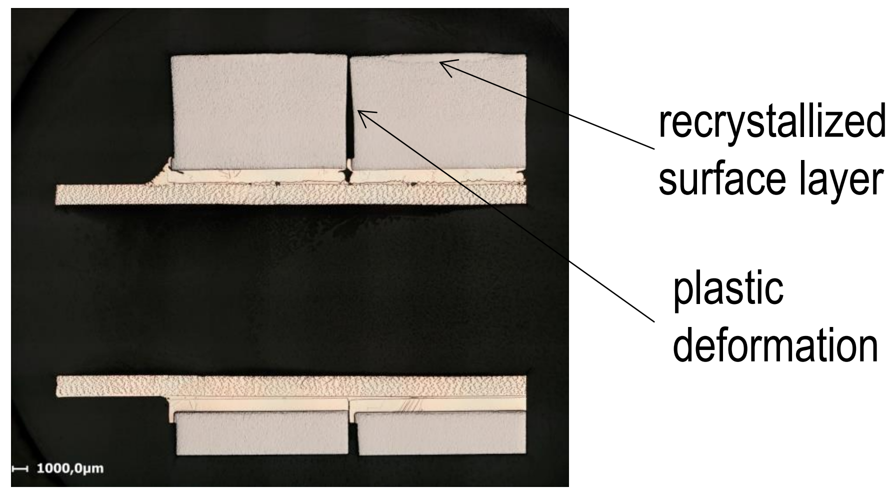
brazed joint between W mono-blocks and W_f - Cu heat sink pipe

High heat flux testing of W_f - Cu PFC mock-ups



- hot-water cooling conditions:
  130°C, 40 bar, 16 m/s → DEMO relevant
- 1000 load cycles at 20 MW/m² without indication of failure
- 100 load cycles at 25 MW/m²,
  screening up to 32 MW/m² (@ 20°C, 10 bar)

[A.v. Müller, Phys. Scr. T171 ***2020*** *014003]*



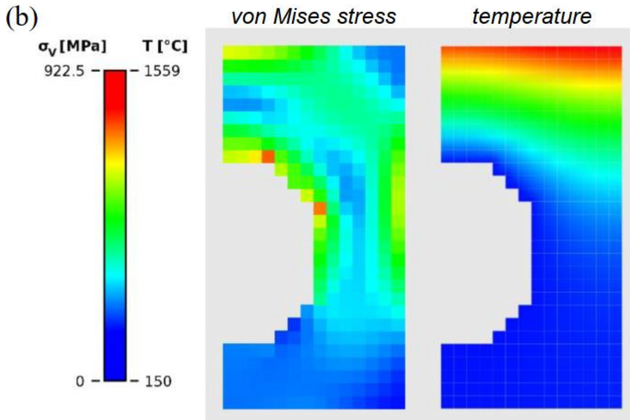
- no damage of bonding & W_f - Cu tube after 1000 cycles at 20 MW/m²

Use of additive manufacturing for topology optimisation

Stress/temperature distribution W monoblock @15 MW/m²

(a)

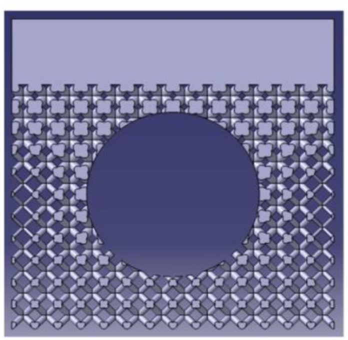
Q_N (MW m^{-2})	σ_{max} (MPa)
5	260.5
10	576.1
15	922.5
20	1264.2

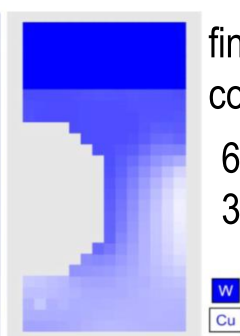


Reduction of thermal stresses by a factor of 6!

Note: extremely high flexibility for geometry, here deliberately classical geometry for comparison!

structure reproduced by bcc lattice

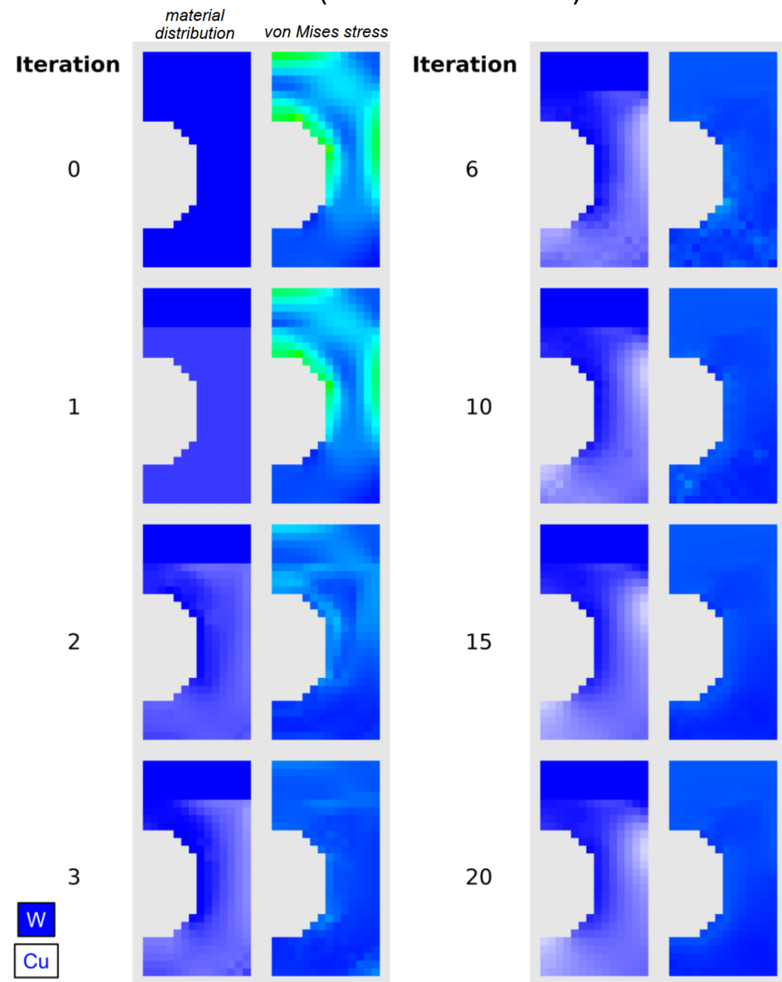




final avg. composition

61% W
39% Cu

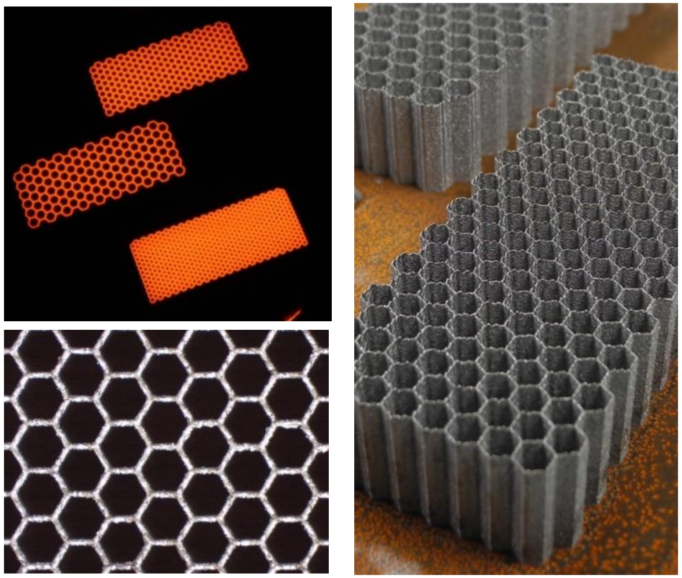
Optimisation of W/Cu distribution (10 MW/m² case)

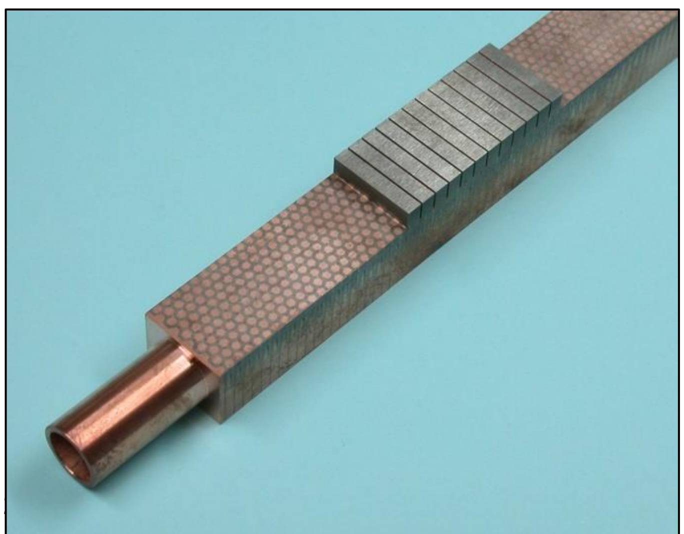


*[B. Curzadd et al., Nucl. Fusion 59 **2019** 086003]*

Additive manufacturing of actively cooled W components

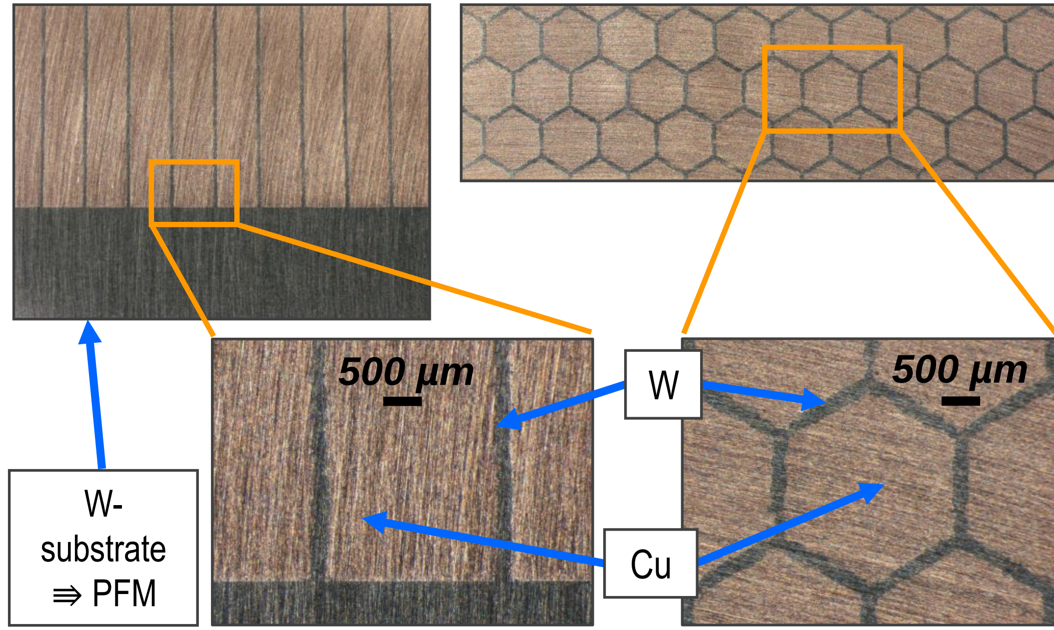
laser powder bed fusion of honeycombs





longitudinal cross section

perpendicular cross section



*[A.v. Müller et al., Nucl. Mat. Energy 19 **2019** 184]*

- **successful additive manufacturing** of honeycomb and optimized W preforms
- nearly **perfect Cu infiltration** of AM W skeletons
- HHF tests of optimized AM W mock-ups imminent

Summary and Conclusion

- **New composite materials can help to improve DEMO PFCs to allow for larger operational margin (higher cooling water temperatures / higher thermal loads) and lifetime**
  - increasing **high temperature strength of Cu** in the cooling structure, ameliorating consequences of Cu(-alloy) neutron damage
  - increasing **fracture toughness of W**
  - **adjusting the thermal missmatch** between armour and heat sink
  - tailored material distribution by **additive manufacturing to reduce thermal stresses**
- **Very promising behaviour of composites materials and composite PFCs in high heat flux tests**

Outlook:

First results for W fibres and W_f-Cu confirm superior behaviour under neutron irradiation!