DIFFER

EUROfusion

Liquid metal conceptual divertor designs for the European DEMO

T.W. Morgan[1*], P.Rindt[2], N.J. Lopes Cardozo[2], R. De Luca[3], G.Dose[3], M.Iafrati[3], A. Mancini[3], G.Mazzitelli[3], S.Roccella[3], T. Barrett[4], F. Domptail[4], M. Fursdon[4], D. Horsley[4], D.Alegre[5], E. Oyarzabal[5], F.L. Tabares[5], V. Pericoli Ridolfini[6], Piotr Chmielewski[6], I. Ivanova-Stanik[6], M. Poradziński[6], R. Zagórski[6] R.Ambrosino[7], F. Crisanti[3] and B-E. Ghidersa[9]

[1]DIFFER, The Netherlands; [2]Eindhoven University of Technology, The Netherlands; [3]ENEA, Italy; [4]CCFE, United Kingdom, [5]CIEMAT, Spain; [6]Institute of Plasma Physics and Laser Microfusion, Poland; [7]Consorzio CREATE, Italy; [8]KIT, Germany

EUROfusion

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Going from ITER to DEMO involves large jumps in several parameters

Courtesy G. Matthews

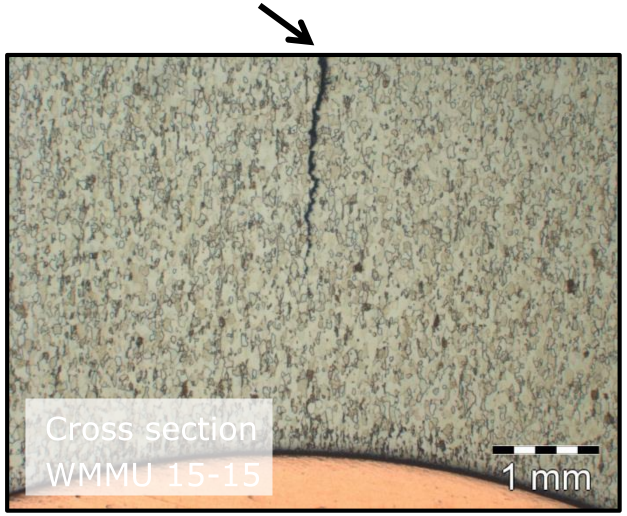
Property	ITER	DEMO[1]
Pulse length	~400 s	~7200 s
Duty cycle	**<2%**	**60-70%**
Neutron load	**0.05 dpa/yr**	**1-9 dpa/yr**
Exhaust power	150 MW	500 MW
Divertor area	~4 m^2	~6 m^2
Radiated power	**80%**	**97%**

Resilience to **neutrons** and **power excursions** on **long timescales** becomes more important

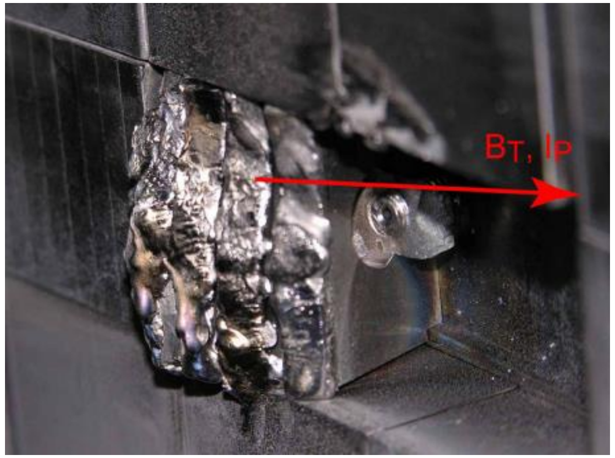
This is where **LM strengths** can play an important role compared to conventional solid divertor materials

[1] *Wenninger NF 2017*

Challenge for solid PFCs: avoiding component failure in DEMO



Hirai J. Nucl. Mater. (2015)



Lipschultz Nucl. Fusion (2012)

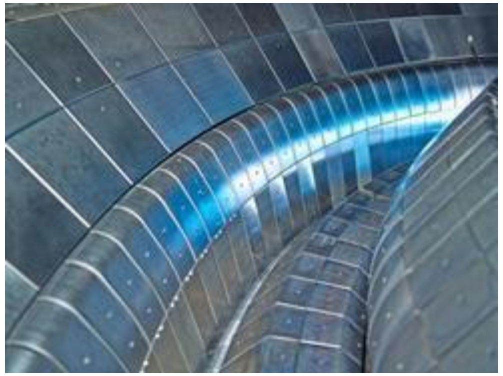
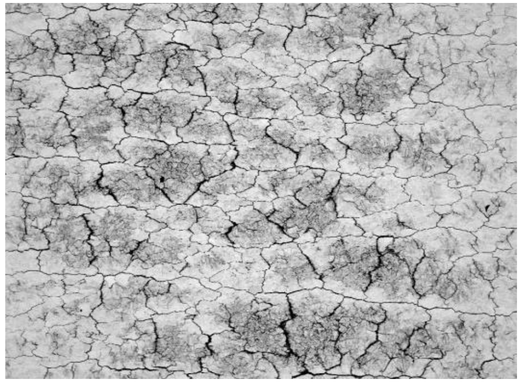
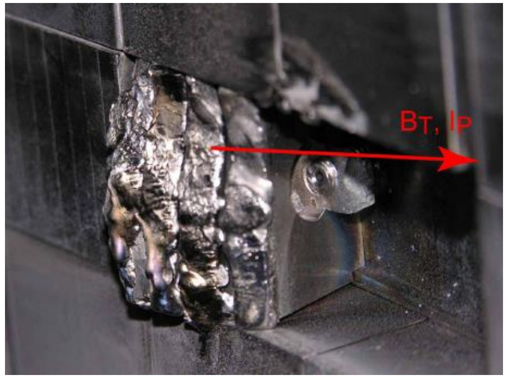
Need to avoid reaching component failure

- macrocracking leading to LOCA
- exceed CHF leading to melting
- fatigue failure
- Erosion limit

Any large unmitigated ELM or disruption could lead to failure (melting, LOCA)

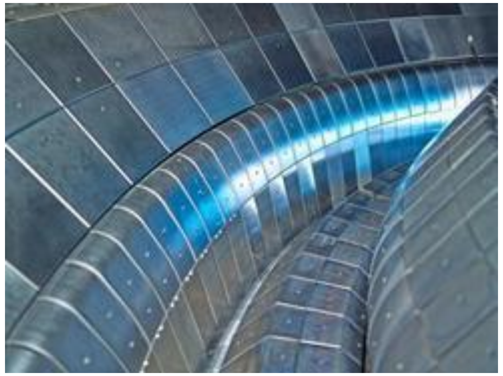
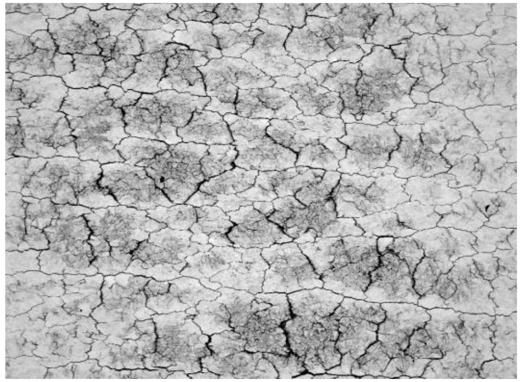
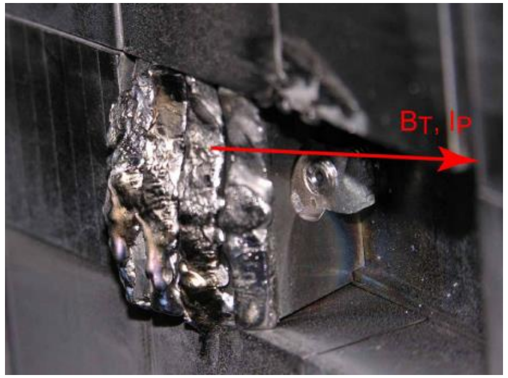
Planned replacement will require >6 months

Benefits of liquid metals for DEMO

	Sputtering	Thermal shock/fatigue	Big ELMs/VDEs/disruptions
Solid PFC:			
LM PFC:	Self replenishment Higher heat fluxes	No cracking Lowered stresses substrate ELMs possible(?)	Already molten Vapour protection

Benefits of liquid metals for DEMO

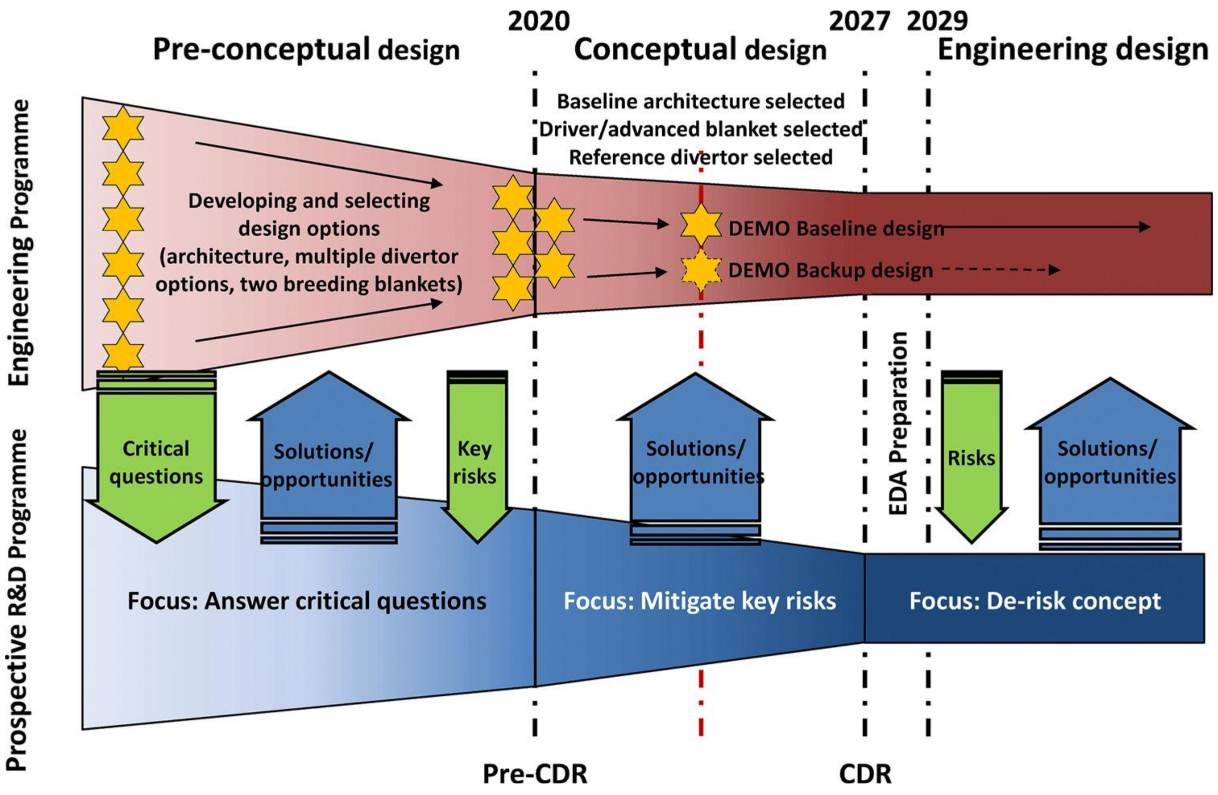
Neutrons

	Sputtering	Thermal shock/fatigue	Big ELMs/VDEs/disruptions
Solid PFC:			
LM PFC:	Self replenishment Higher heat fluxes	No cracking Lowered stresses substrate ELMs possible(?)	Already molten Vapour protection

Only influences substrate

Separation of PSI from neutron issue

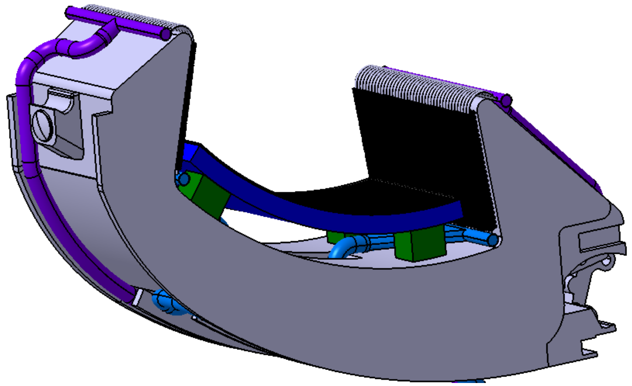
EU DEMO should have conceptual design completed by ~2027



PARAMETER	NEAR TERM DEMO[1]
P_{fusion}	2000MW$_{th}$/~500 Mw$_{e}$
t_{burn}	2 hours
R_0/a (m)	9.0/2.9
P_{sep}	153 MW
P_{sep}/R_0	17 MW m^{-1}
I_{plasma}	18 MA
$P_{neutron}$	1.04 MW m^{-2}
B_{axis}	5.9 T
W_{plasma}	1.18 GJ

[1] *Federici FED 2018*

EU LM strategy is risk mitigation: realistically should fit into DEMO design without huge changes to other components

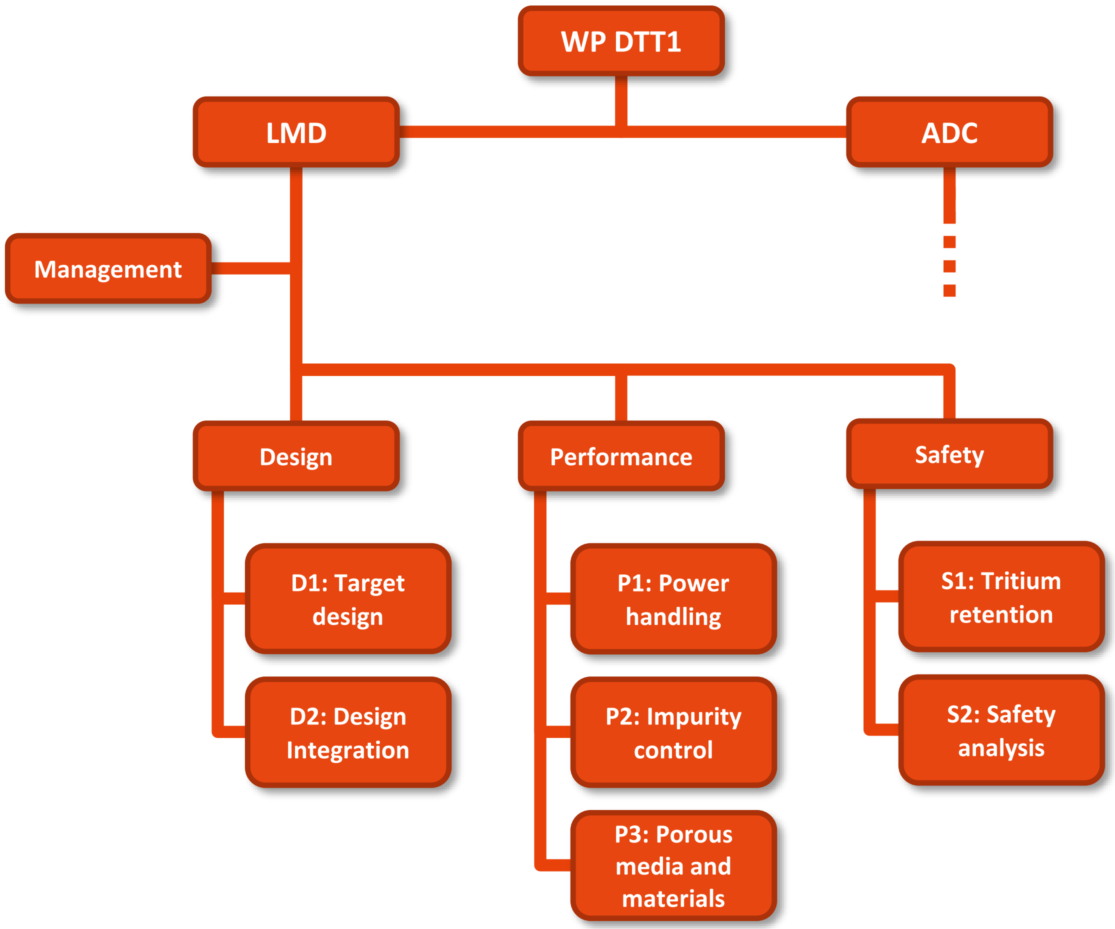


1. Can fulfil all divertor requirements (heat/particle handling, ash removal)
2. Compliant with plasma (impurity) and scenario
3. Mountable in divertor cassette (remote maintenance)
4. Compliant with in vessel components, pumps, diagnostics…
5. Something that can relatively easily be incorporated into Engineering design after decision point ~2027

Basic assumptions:

- want to use **most mature technology**, CPS
- **Cooling by conduction** to coolant

Description of WP (2018-2019)

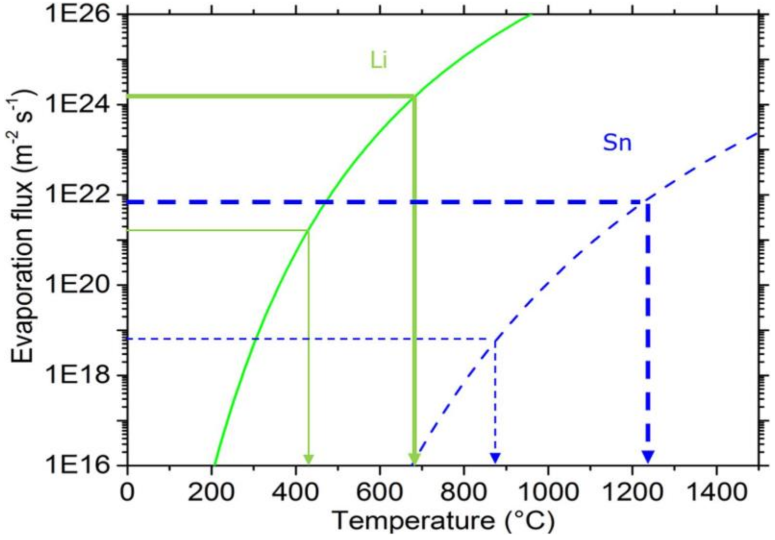


- Develop conceptual designs
- Integrate with rest of plant
- Assess power handling capability
- Influence of LM erosion on plasma performance
- Stabilization of LM surfaces by porous media
- Material compatibility in terms of wetting, corrosion, embrittlement
- LM fuel retention
- Safety precautions

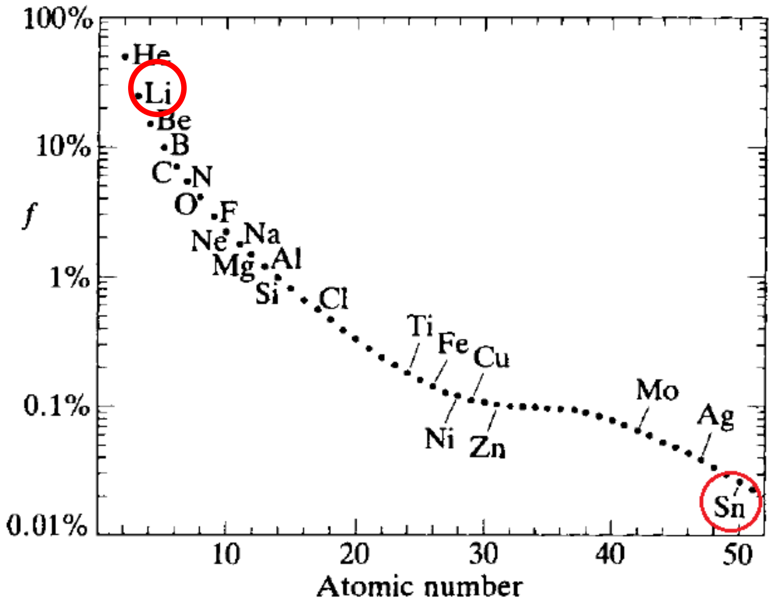
Material options of Li, Sn both have strengths and weaknesses

Choices once cost, availability, activation, material compatibility etc. taken into account

Lithium	Tin
Low Z	Higher Z
High vapour pressure	Lower vapour pressure
High T retention	Lower T retention



Morgan Plasma Phys. Control. Fusion (2018)



Wesson, Tokamaks (2004)

Need to develop design criteria to specify leading option

Design criteria: performance

Design requirement	Sn	Li
Must tolerate 10 MW m^{-2} in nominal operation	✓	✗ (✓)
17-21 MW m^{-2} during slow transients 3-10 s	✓	✗ (✓)
Heat load < 5 MW m^{-2} outside strike points	✓	✓
Withstand ≥1 disruption (80 GW m^{-2} 4 ms)	✓	✓
Coolant 40% safety factor CHF	✓	✓
Tritium inventory in-vessel <730g	✓	✗
Evaporation must not significantly reduce fusion output during normal operation	✓ 1250 °C	✗ (✓) 690 °C

Cannot simultaneously satisfy high heat loads and low evaporation rate for Li

Tritium inventory control with Li requires continual active removal

Design criteria: compatibility

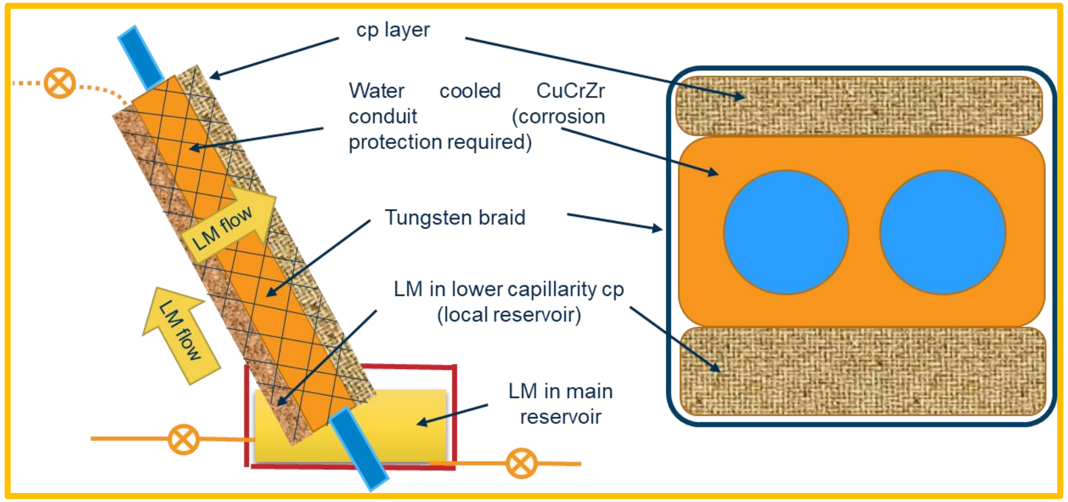
Design requirement	Sn	Li
High recycling divertor	✓	✗
Activation must be kept to limits for intermediate level waste	✓	✓
Lifetime 2 fpy	✓	✓
70 cm high vertical target	✓	✓
Need to be able to re-wet in-situ	✓	✓
Withstand atmosphere for 2 months during maintenance	✓	✗
Withstand 200 °C bake during startup	✓	✓
No major design changes to in-vessel components, diagnostics, first wall	✓	✗

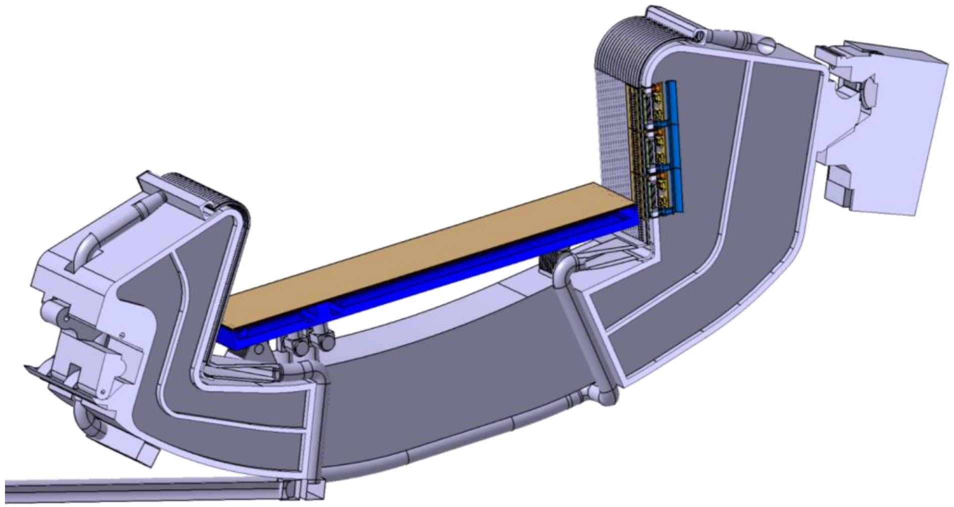
Li would act as low recycling surface and result in significant changes to the operational mode of DEMO

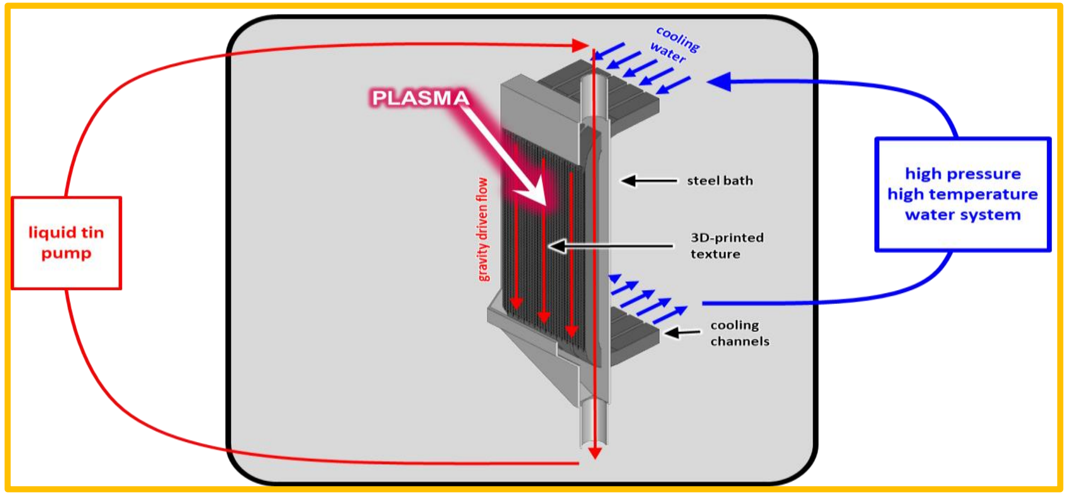
Long term Li may be better [see next talk] but technologically much more complex

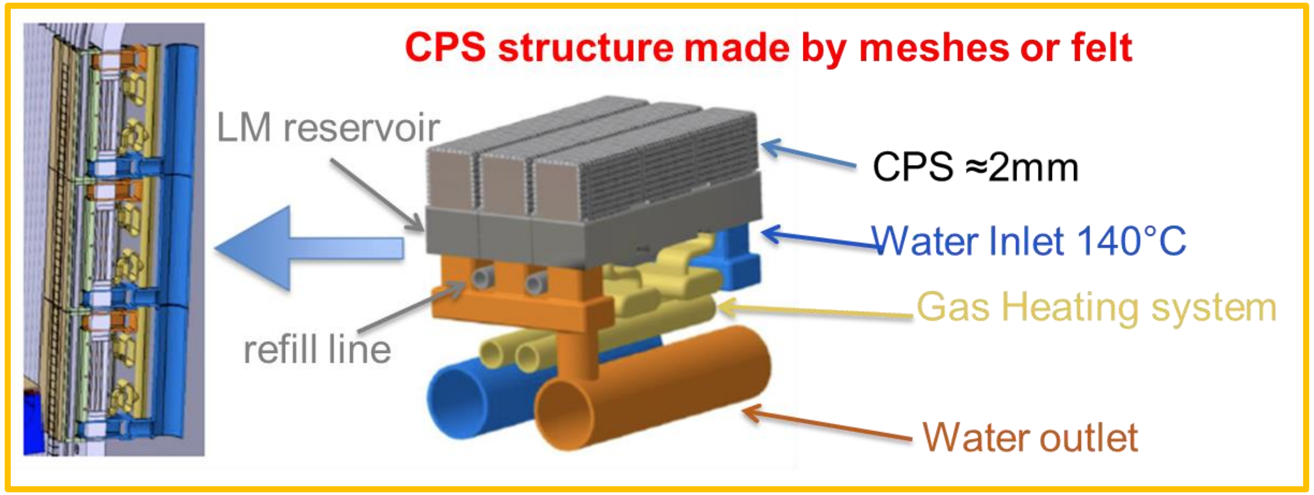
Sn chosen as candidate LM for this application

Several Sn-CPS based pre-conceptual designs being developed

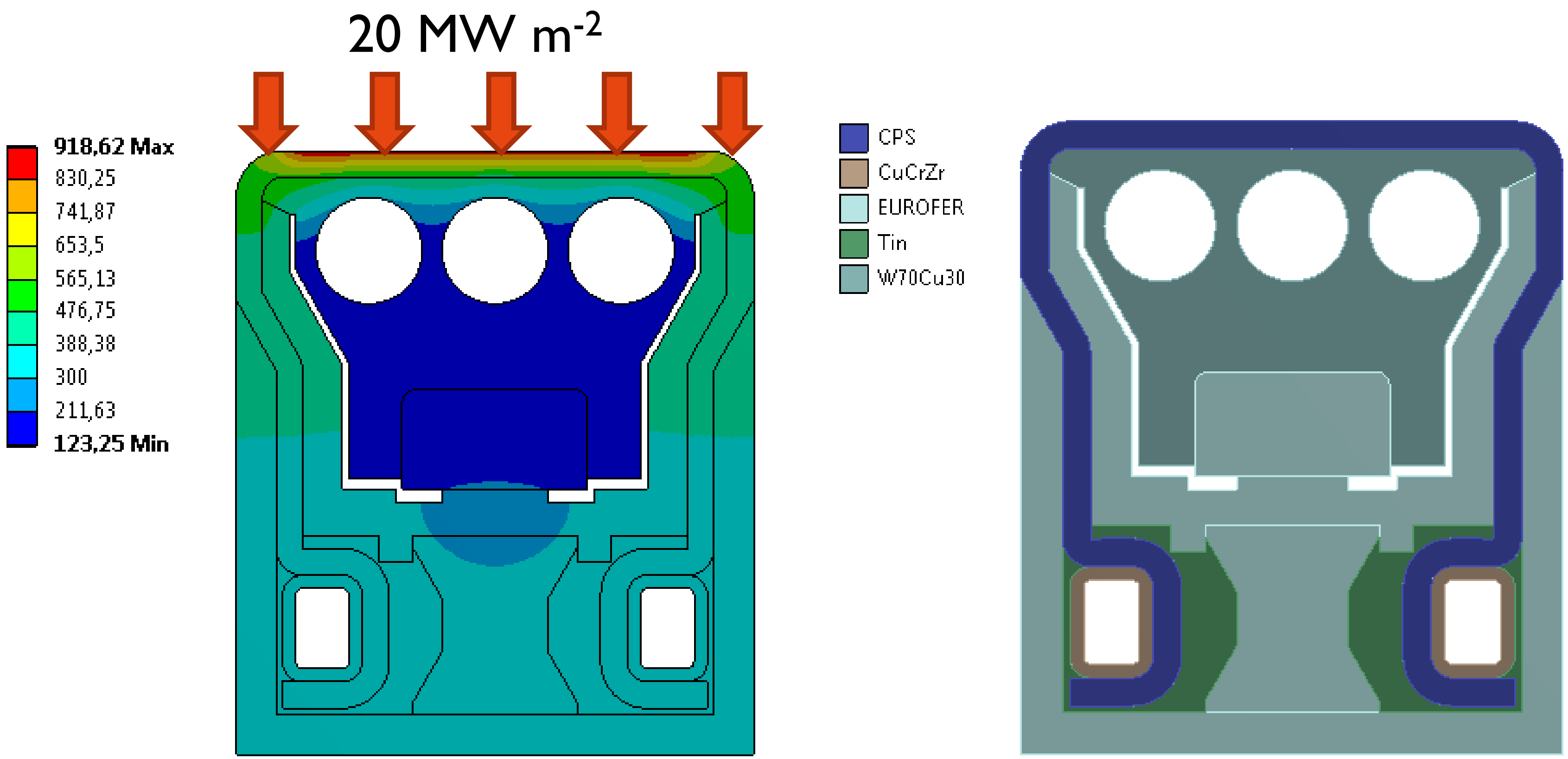








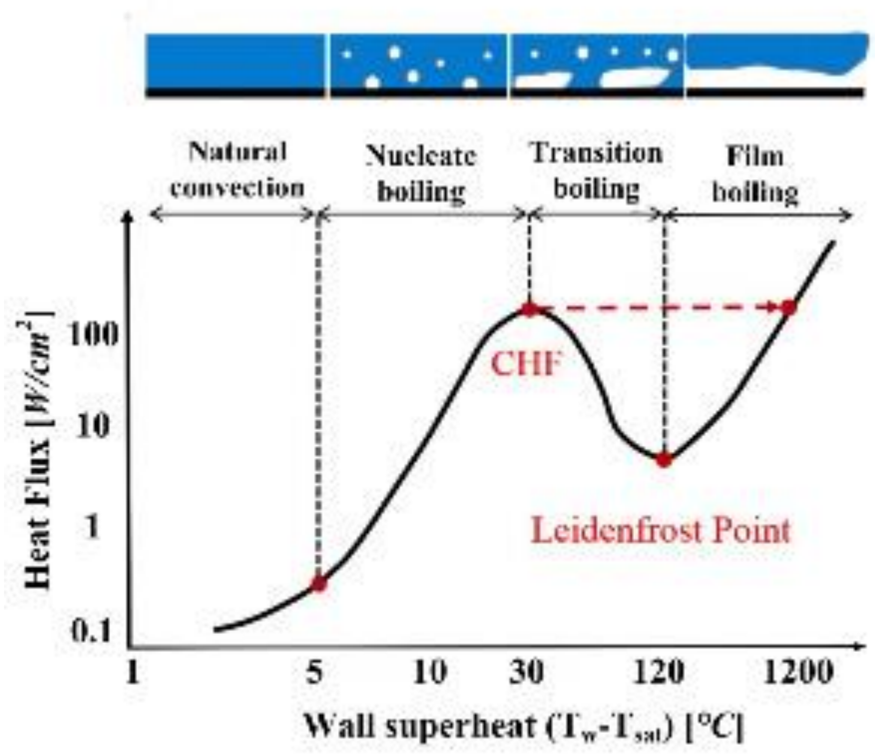
All designs capable of normal operation up to ~20 MW m^{-2}



Using CPS means component can be significantly thinned compared to W monoblock design

Possible to keep temperature below range where evaporation becomes significant

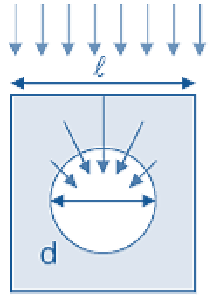
Critical heat flux is operational limit



Can keep temperature of Sn below point where evaporation becomes an issue

Main failure mode is therefore reaching CHF leading to severe reduction in heat transfer coefficient

$$q = h(T_w - T_{sat})$$



Some definitions:

Peaking factor: $f_p = q_w/q_0$

Incident CHF: $ICHF = CHF/f_p$

CHF Margin: $MCHF = ICHF/CHF$

Different design choices give different trade-offs: need to optimize

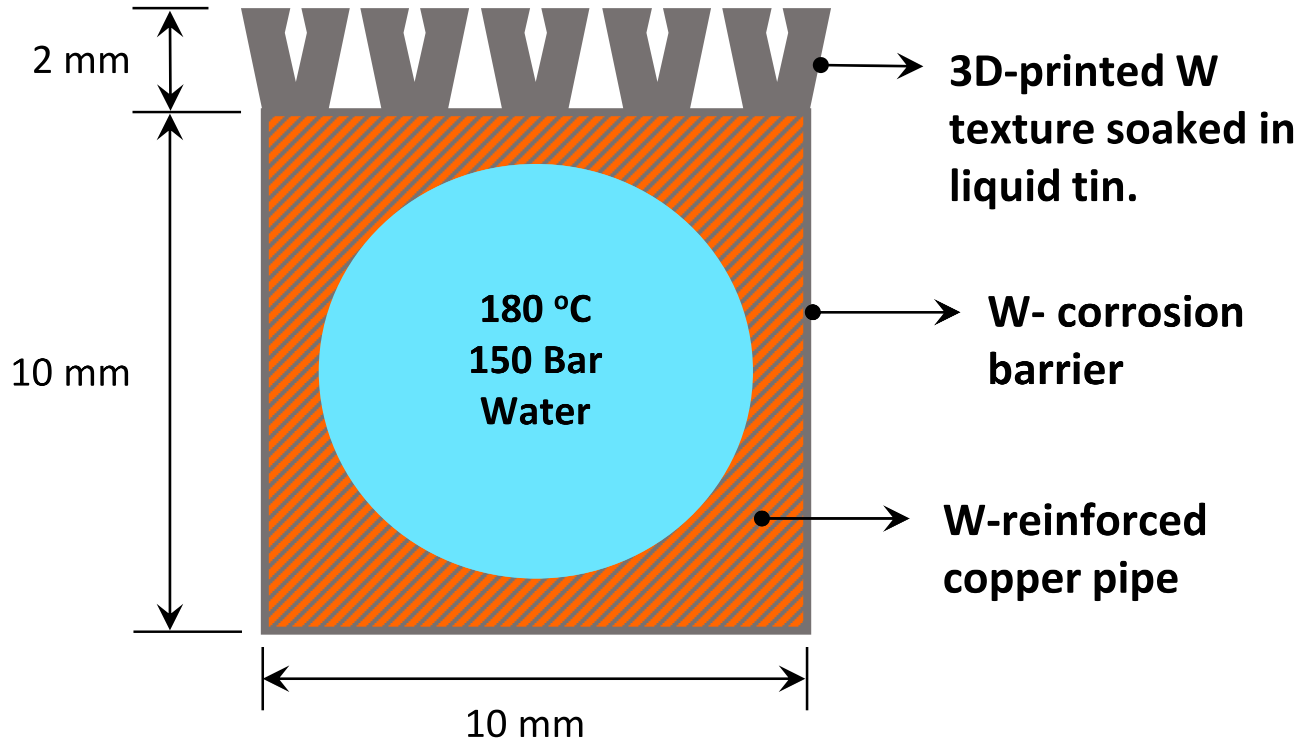
Design choice	ENEA	DIFFER	CCFE
Liquid metal	Sn	Sn	Sn
CPS type (2 mm thick)	Mesh/felt	3D printed	Braid
Sn resupply	Capillary reservoir	Capillary/flow	Capillary reservoir
CPS max pore size (µm)	50	50	20
Water temperature*	120 °C	180 °C	240 °C
Water pressure (bar)*	50	150	150
Water flow rate ($m\ s^{-1}$)*	12	14	16
Sn liquid?	>5 $MW\ m^{-2}$	During operation	Always
ICHF ($MW\ m^{-2}$)*	33	56	25
MCHF	1.65	2.8	1.25
Sn heating	Gas in PFC	Gas in cassette	Water

Different technology options (CPS type, Sn resupply, CPS max pore size)

Different cooling assumptions (Water temperature, Water pressure, Water flow rate, Sn liquid?, ICHF, MCHF)

*c.f. DEMO ITER-like PFC: 140 °C, 50 bar 14 $m\ s^{-1}$ ICHF=45 $MW\ m^{-2}$ MCHF=1.4

Example design: DIFFER cross section single pipe row



CPS thin enough to get low surface temperature during nominal operation, but thick enough that VS occurs during transients (avoid reaching CHF)

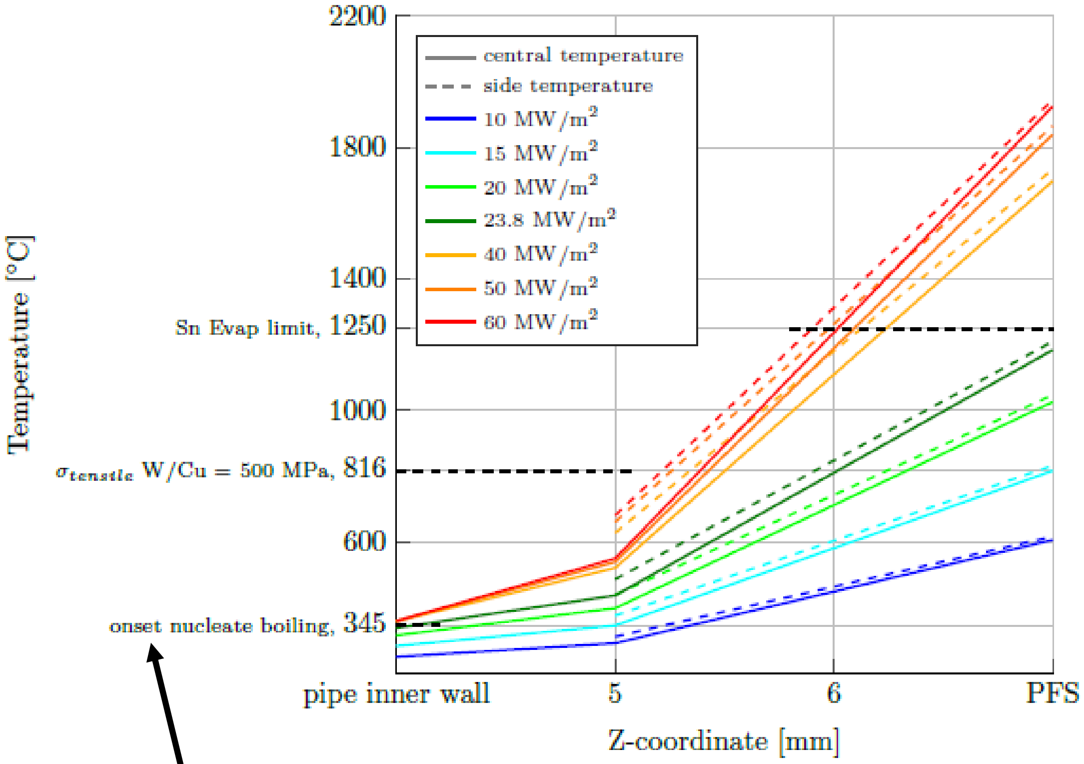
W/Cu composite has higher strength at high temperature (development WPMAT)

Only W faces Sn: no corrosion

Coolant is compromise between keeping surface cool and Sn liquid

No monoblocks (alignment issues), only rows of pipes

Performance analysis shows normal performance up to 24 MW m^{-2} possible with target damage only at 56 MW m^{-2}

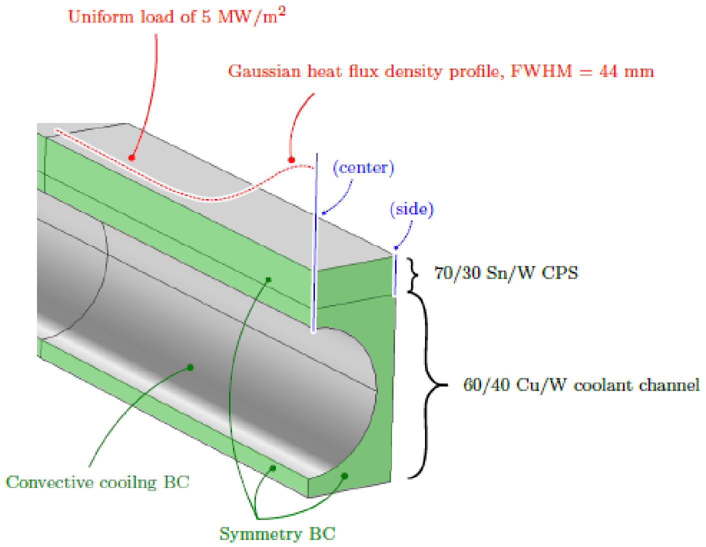


At 23.8 MW m^{-2} this becomes the limiting factor (only allowed during off normal events)

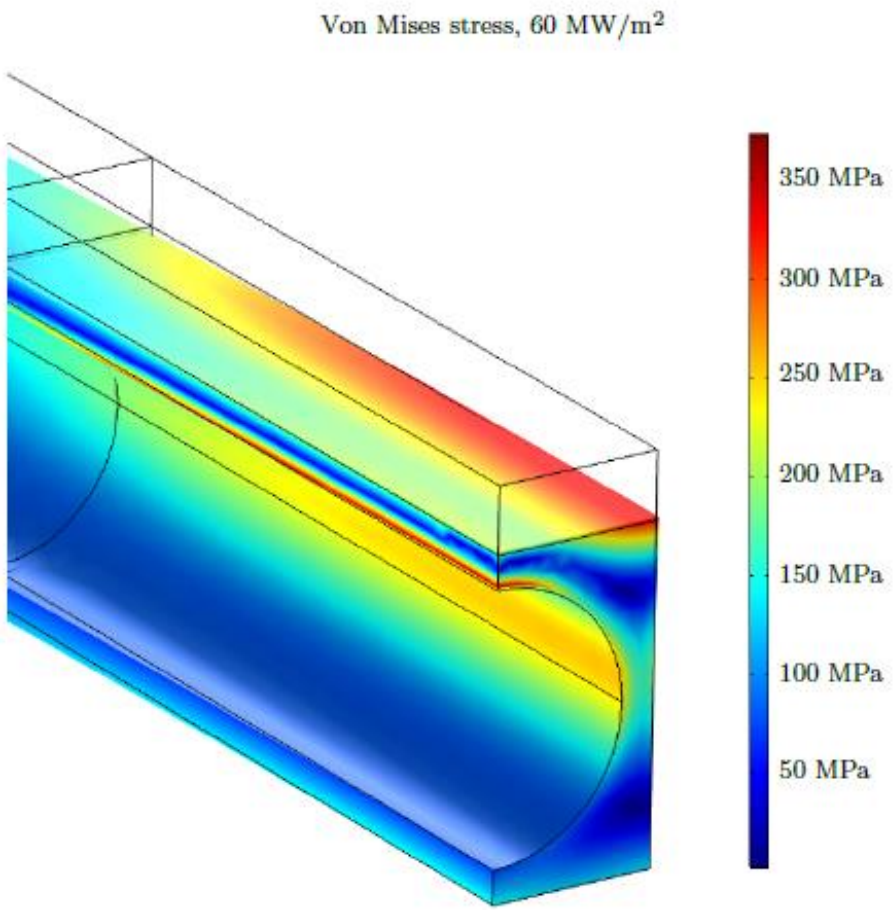
Normal operation up to 24 MW m^{-2} before onset nucleate boiling (slow transient only)

This is well below Sn evaporation limit defined

Target only damaged at 56 MW m^{-2} due to CHF reached



Von mises stresses are low in pipe even up to CHF so can operate in slow transients up to 56 MW m^{-2}



Stresses well below UTS of W/Cu composite (500 MPa at 816 °C)

Real failure point is yield stress however (unknown as yet)

Given safety margin operational limit during slow transients should be 43 MW m^{-2}

Disruptions can be survived without damage if significant vapour shielding occurs

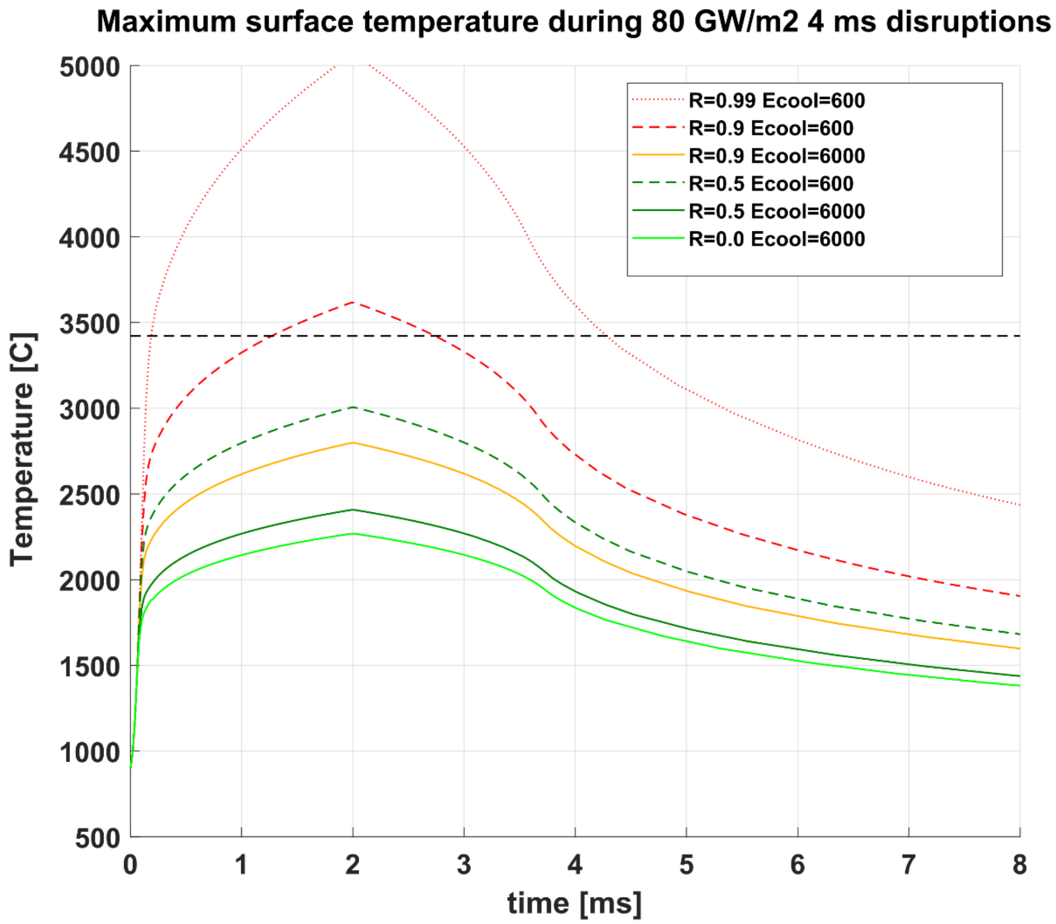
Apply 80 MW m^{-2} 4 ms disruption to FEM including VS model

Failure mode is W melting point

Given reasonable assumptions for ϵ_{cool} and R find that disruptions are survivable without damage

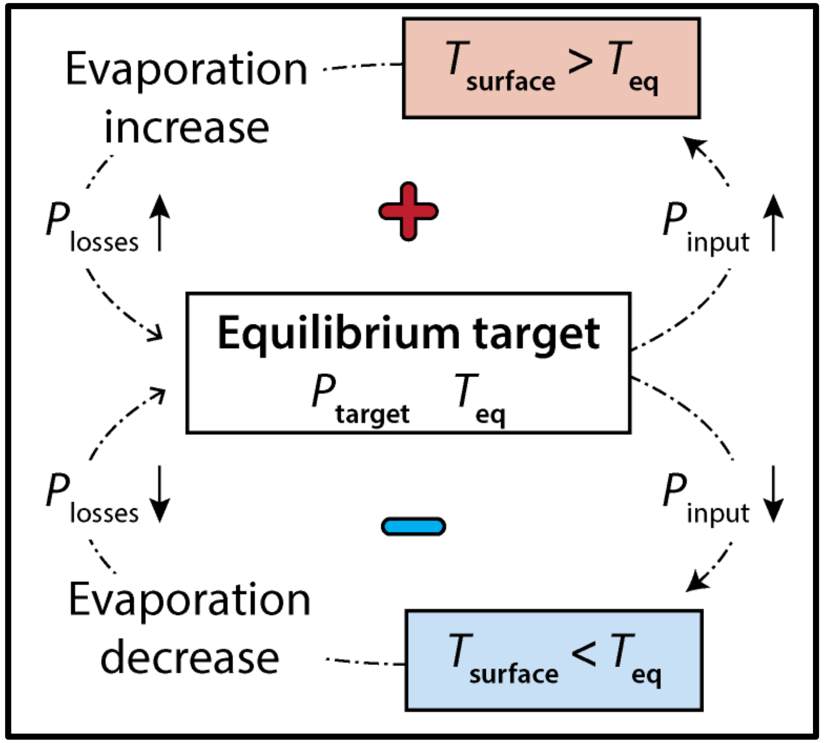
Similar for big ELMs

Don't need to worry about CHF, time too short



$$Q_{plasma} = \mathbf{Q}_{\text{cond}}(T_{surf}) + (1 - R)\Gamma_{\text{evap}}(T_{surf}) \cdot (\epsilon_{\text{cool}} + \epsilon_{col} + \epsilon_{evap})$$

Implications for operation



For loss of detachment get temperature increase to point where Sn evaporation will decrease fusion power in core

Lead to automatic protection of divertor component (no damage)?

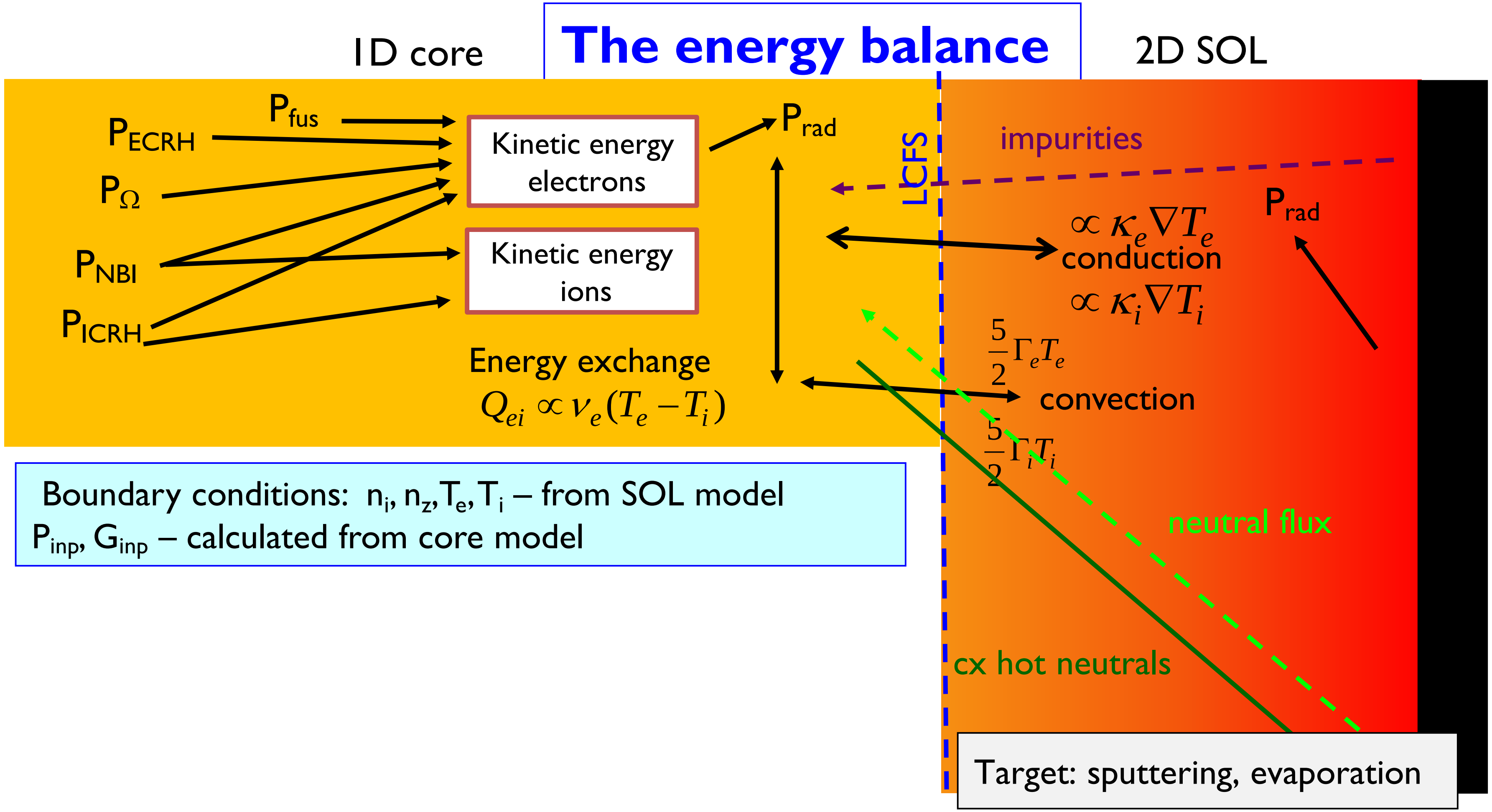
Divertor also may survive disruptions: less stringent limits for disruption mitigation? Same for ELMs?

Risk of radiative collapse or negative feedback mechanism to avoid it?

Core compatibility: COREDIV

COREDIV = 1D transport in the core self-consistently coupled to 2D model in the SOL

Aims at steady state description of plasmas with impurities



Boundary conditions: n_i, n_z, T_e, T_i – from SOL model
P_{inp}, G_{inp} – calculated from core model

Advantages:

- short running time (~one day)
- Impurity modeling, including radiation in the core and in the SOL and therefore possible assessment of the heat load to the target.

Drawbacks:

- SOL slab geometry
- Semi-analytical model of neutrals

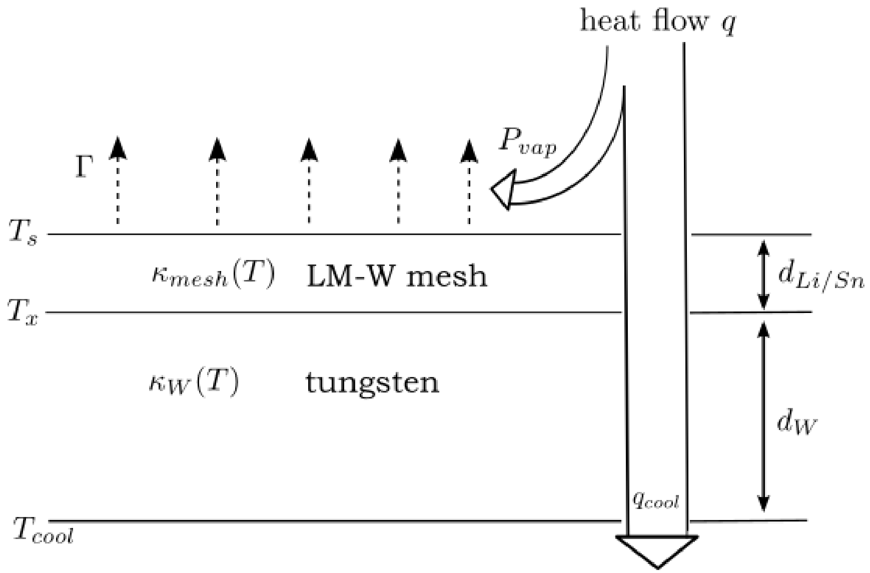
COREDIV handling of LM impurities

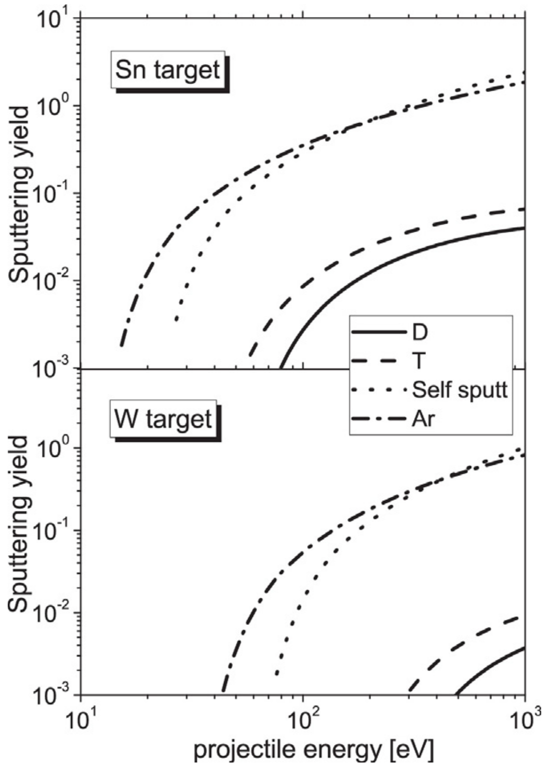
At divertor surface evaporation and sputtering sources coupled to incoming heat and particle flux

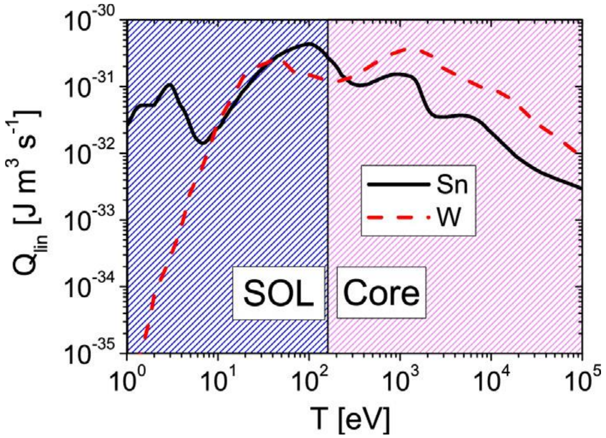
Radiation in core modelled using coronal curves

Source scanned by varying d_{Sn}, d_W or T_{cool}

Modelling DEMO-1 (2015) configuration

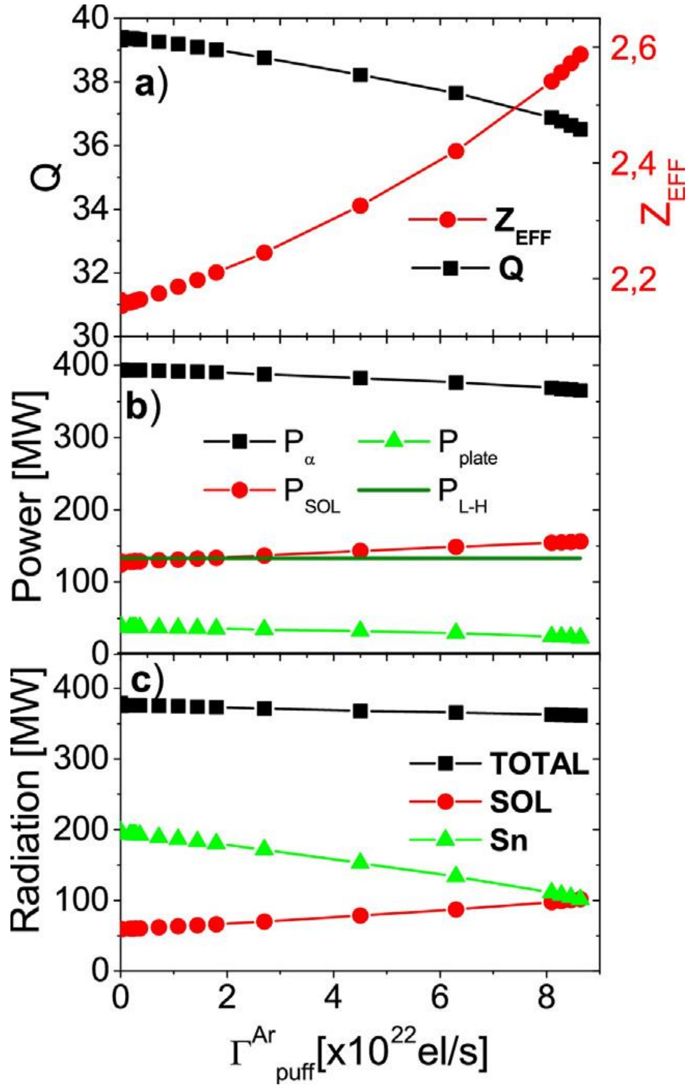






Poradzinski Fusion Eng. Des. (2017)

Use of Sn can be compatible with good core performance using Ar seeding



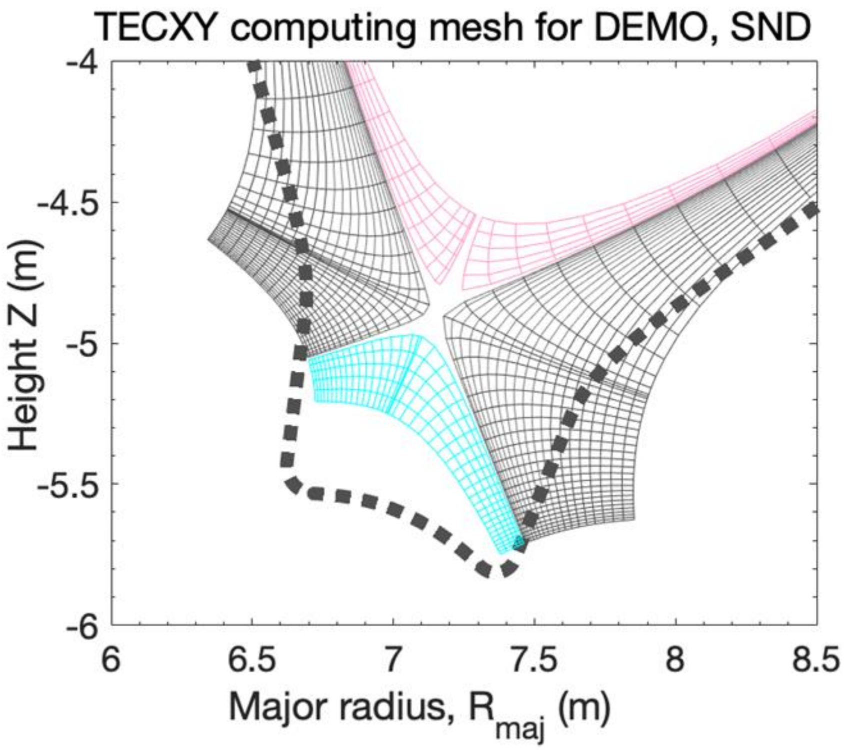
Seeding Ar increases SOL radiation and lowers core radiation by reducing Sn source

Can operate above L-H threshold while also detached at target plate

Similar scenario to W target with Ar or Ne seeding (Q=34-36) [Ivanova-Stanik J. Nucl. Mater. (2015)]

n.b. deliberate scenario with v. high Sn source (d_{Sn}=2 mm, d_W=50 mm, T_{cool}=500 °C, T_{surf}~2000 °C)

TECXY has also been used to study the effect of Sn on performance



2D multifluid description– Braginskij like equations simultaneous treatment of few impurity species

Classical parallel transport

Radial Transport: several options – constant diffusion coefficients, Alcator-like, turbulent transport

2 Temperatures Model (Te and Ti) all ions same Ti

Atomic processes: ionization, recombination, excitation, charge exchange (Li, Be, B, C, O, Ne, Ar, Si, Ni, Sn, Mo)

Neutrals (analytical & fluid model for neutrals): hydrogen recycling, impurity sputtering and EVAPORATION

Curvilinear tokamak geometry

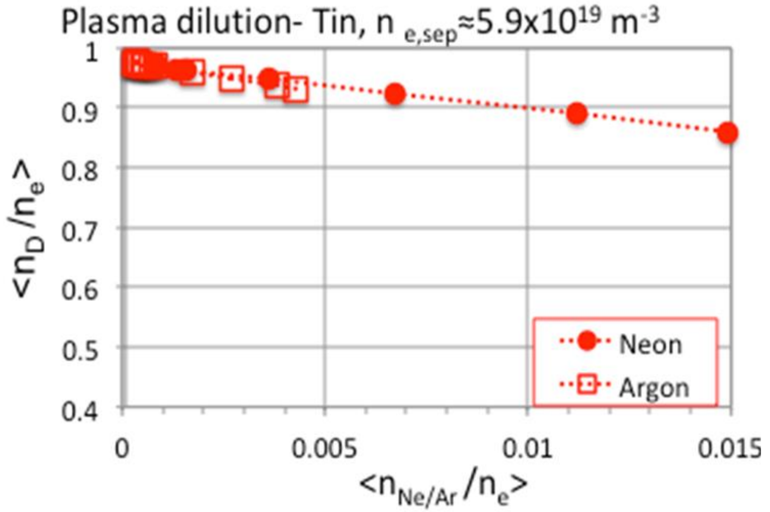
Similar conclusions reached that with seeding impurities good performance achieved (modelling for I-DTT but similar for DEMO)

Again high Sn source conditions chosen (d_{Sn}=1 mm, d_W=7 mm, T_{cool}=700 °C)

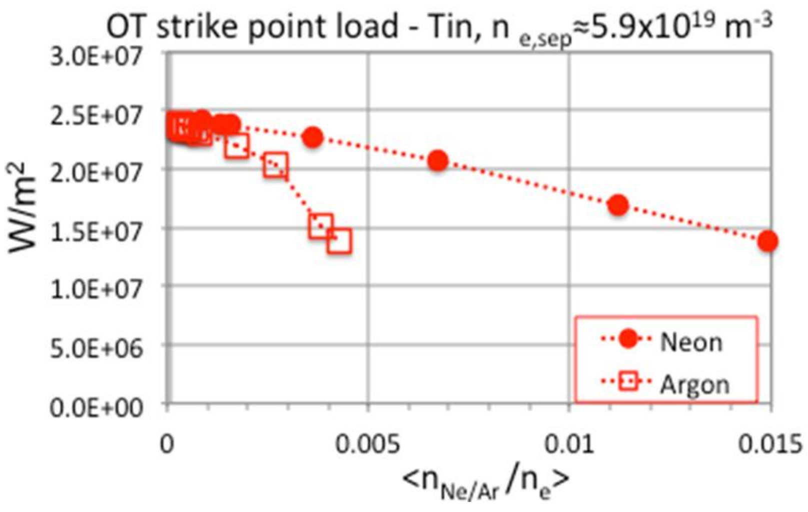
High density scenarios result in detachment and low core impurity even without seeding

Even at low density addition of seeding impurity leads to reduction of power at plate while keeping plasma dilution low

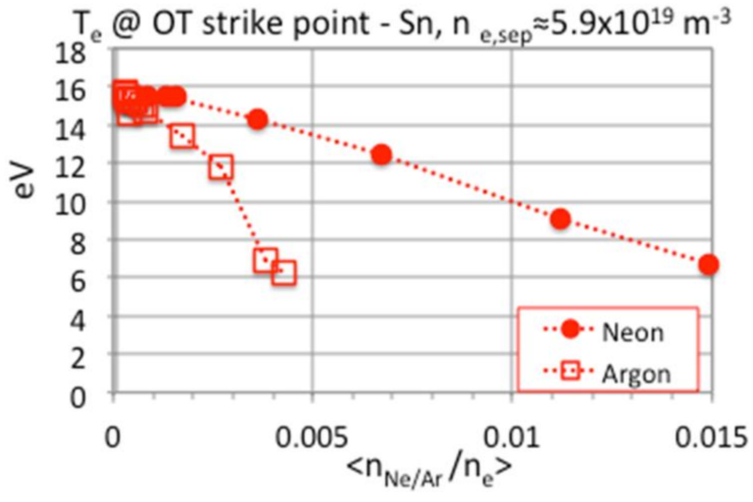
Plasma dilution



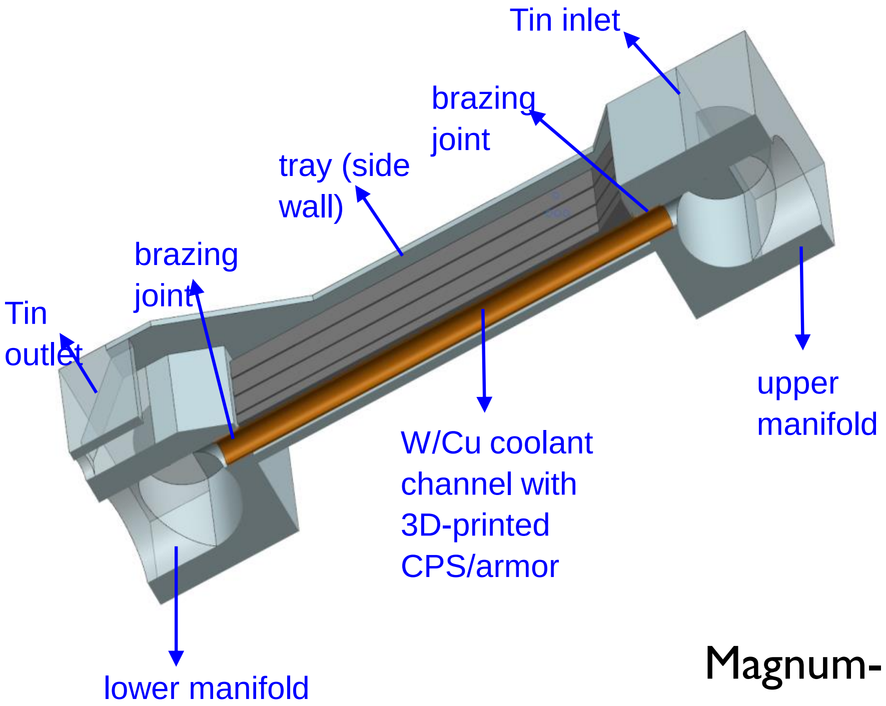
Heat load



Electron temperature



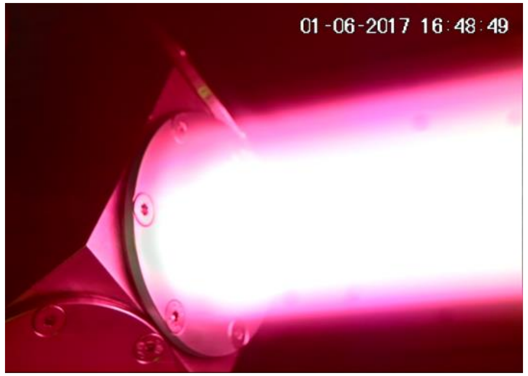
Next steps



Development of prototypes based on pre-conceptual designs

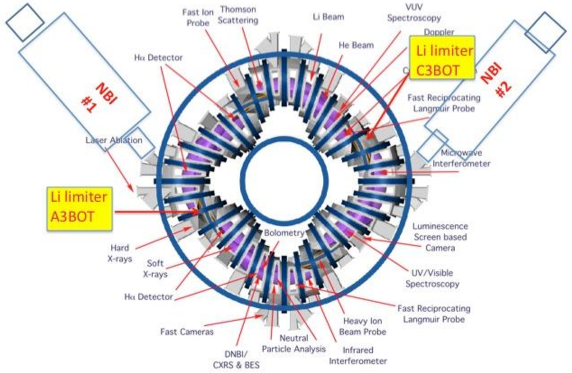
Testing in plasma and heat load devices

Magnum-PSI



Plasma/ELMs

OLMAT



Heat loads

QSPA-Kh50

ELMs/disruptions

Conclusions

DEMO heat exhaust challenges can be addressed using a Sn-CPS based divertor solution

A 20 MW m^{-2} steady-state heat load can be sustained while maintaining good core performance

Development and testing of prototypes is ongoing as a step towards deployment in confinement devices to demonstrate these benefits