EUROfusion

Plasma facing components beyond ITER – solid materials

F. Maviglia

With contribution from:

G. Federici, C. Bachmann, L. Boccaccini, F. Cismondi, E. Diegele, R. Neu, G. Pintsuk, M. Siccinio, J. H. You, EUROfusion PPPT team and PLs

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Outline

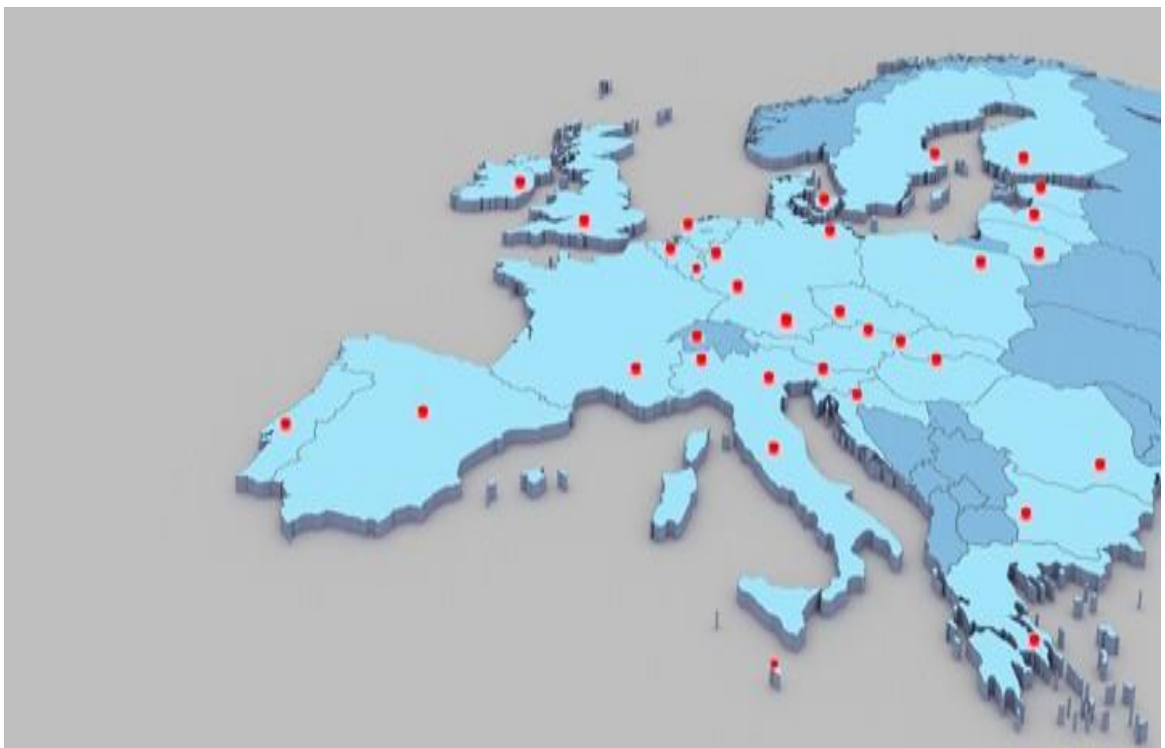
- Introduction: ITER and DEMO PFC requirements
- Materials R&D for DEMO – the baseline options
  - Plasma Facing Materials (Armor)
  - Copper-alloys (Heat sink materials)
  - Ferritic-martensitic steels (Structural Material)
- DEMO heat load requirements
  - First Wall (FW) and Limiters
  - Divertor
- Conclusions

EUROfusion consortium

Programme Management Office

ITER **Ph**ysics Department (IPH)

Power **P**lant **P**hysics and **T**echnology Department (PPPT)



EUROfusion consortium agreement signed in 2014 by:

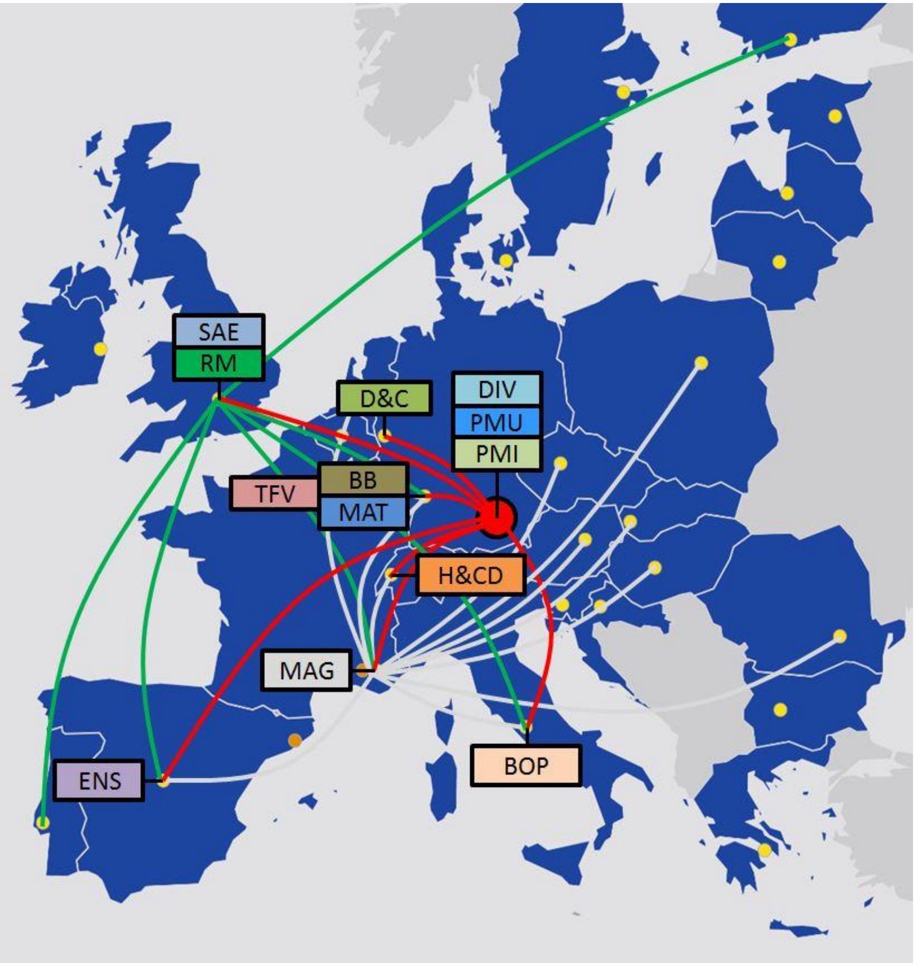
- 29 research organisations
- 26 European Union member states plus Switzerland signed and , as of 1 January 2017, Ukraine.
- In addition about 100 Third Parties contribute to the research activities through the Consortium members.
- EUROfusion collaborates with Fusion for Energy (Spain) and intensively supports the ITER International Organization (France).

Power Plant Physics and Technology Department (PPPT)

Pre-conceptual design of DEMOnstration (DEMO) Fusion Power Plant to follow ITER, capable of generating several 100MW of net electricity and operating with a closed fuel-cycle around the middle of the century.

Geographically distributed team:

- **WPPMI:** Plant Level System Engineering, Design Integration and Physics Integration
- **WPBB:** Breeding Blanket project;
- **WPBOP:** Heat transfer, Balance-of-Plant and Site project;
- **WPDC:** Diagnostic and Control project;
- **WPDIV:** Divertor project;
- **WPHCD:** Heating and Current Drive systems project **;**
- **WPMAG:** Magnets System project;
- **WPMAT:** Materials project;
- **WPRM:** Remote Maintenance System project**;**
- **WPSAE:** Safety and Environment project**;**
- **WPTFV:** Tritium, Fuelling & Vacuum systems project**;**
- **WPENS:** Early Neutron Source project**;**

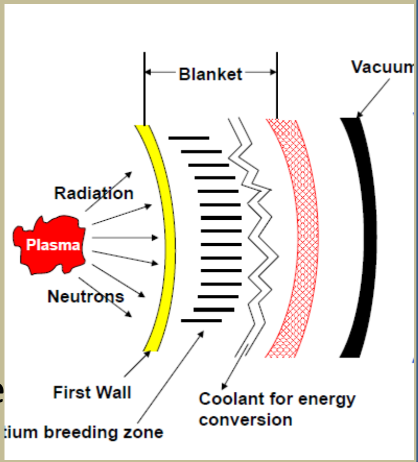


Recap of Major DEMO Design Issues

For any further fusion step, safety, T-breeding, power exhaust, RH, component lifetime and plant availability, are important design drivers and CANNOT be compromised

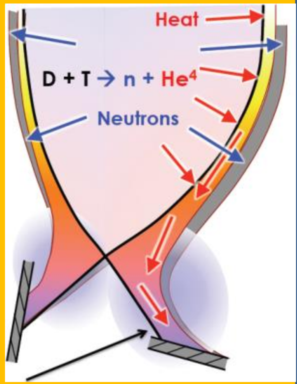
Tritium breeding blanket

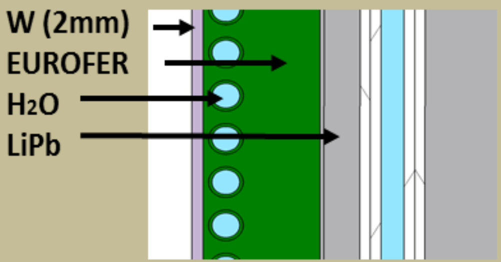
- ✓ most novel part of DEMO
- ✓ TBR >1 marginally achievable with thin PFCs/few penetrations
- ✓ Feasibility concerns/ performance uncertainties with all concepts -> R&D
- ✓ Selection now is premature
- ✓ ITER TBM is important



Divertor Power Exhaust

- ✓Peak heat fluxes near technology limits (>10 MW/m^2)
- ✓Use H_2O as coolant and Cu-alloy
- ✓ITER solution may be marginal for DEMO
- ✓Advanced solutions may be needed but integration is very challenging. A dedicated DTT is planned





Off normal transients are a major design driver. DEMO requires dedicated FW protections in some areas.

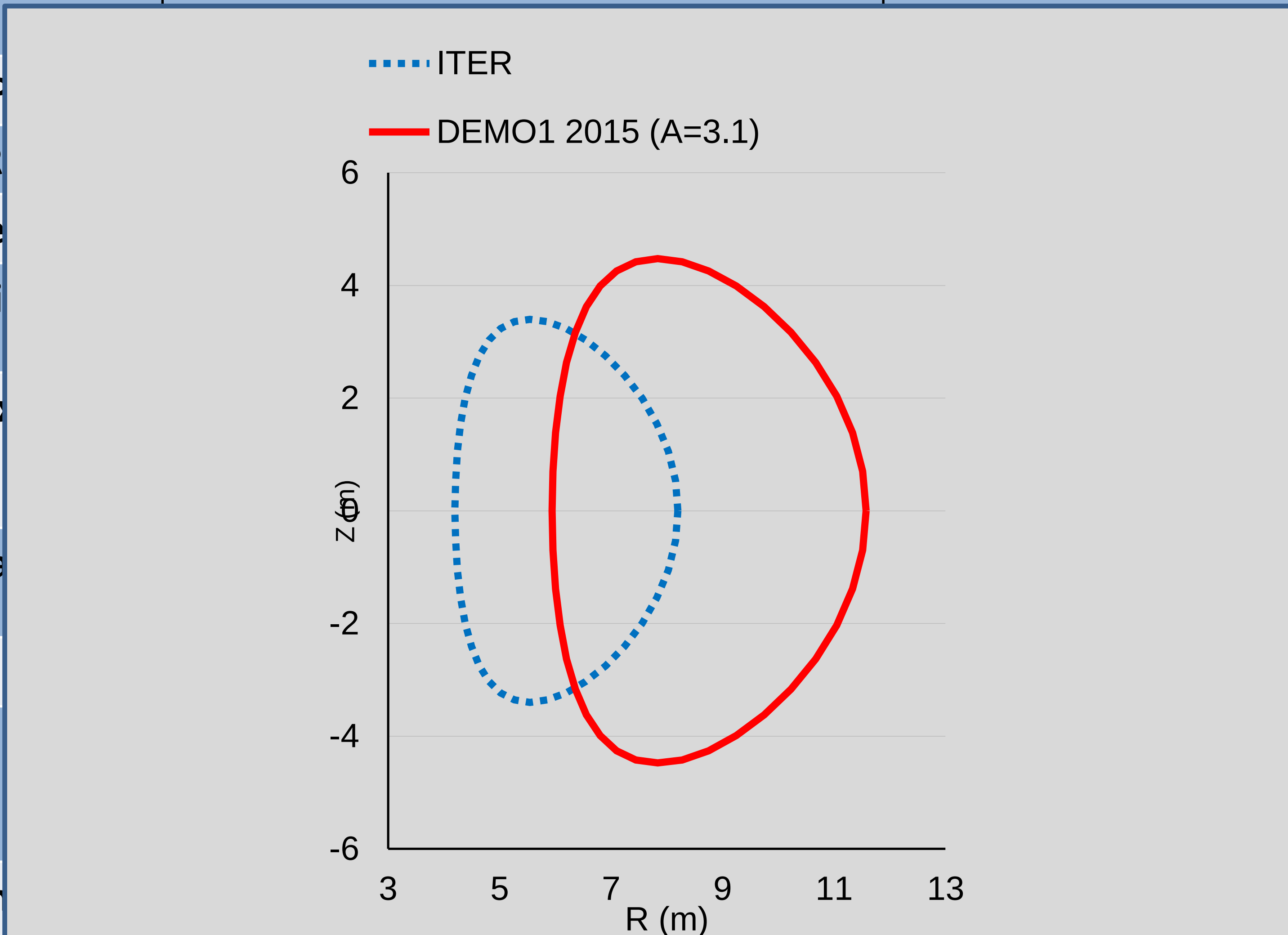
Plasma transients

Materials

- ✓ Embrittlement of RAFM steels at low temp. and loss of mech strength at ~ high temp.
- ✓ Progressive blanket operation strategy (1st blanket 20 dpa; 2nd blanket 50 dpa)
- ✓ Need irradiated matl property data and structural design criteria.
- ✓ Urgent need of a dedicated fusion irradiation facility (IFMIF-DONES)

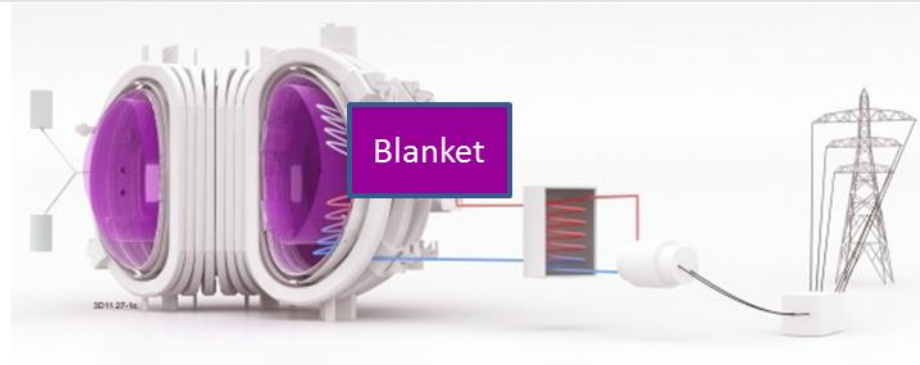
Main differences ITER and DEMO

	ITER	DEMO
Overall Mission	Experimental device with	Approaching a commercial power plant ... missions.
Fusion P		
Major R		
Pulse Le		on also studied
Availabi		r high
Complex		
Heat Tra		for electricity her temp.)
Tritium		f-sufficiency.
Materia		erials as ll and
Neutron Fluence		W steel (1st blank)



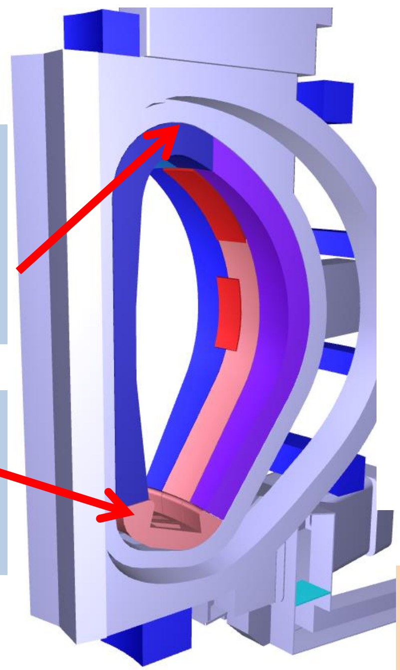
Fusion Materials Challenge

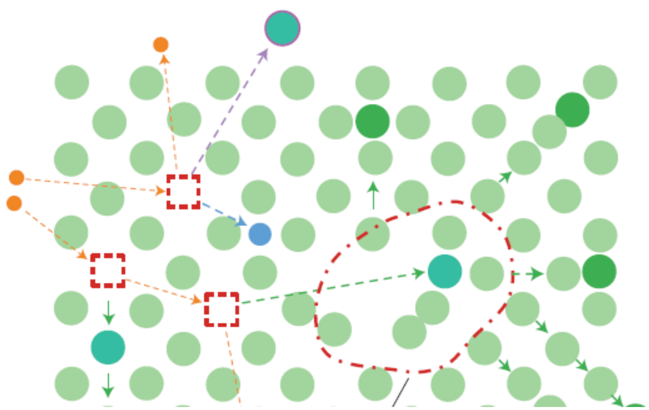
D&T react in the 'fusion furnace'
The energetic neutron stops in the "blanket", heating (to finally produce electricity), BUT



Ferritic-Martensitic *Steel*
Li-Ceramic or LiPb (*Breeder*)

Tungsten – *Armour*
Copper alloys – *Heat sink*





n- Damage in materials
- creates disorder
- vacancies
- transmutation
- helium

He/dpa = <1 (fission); > 10 (fusion)

H/dpa = 10 (fission); > 40 (fusion)

This damage is studied in irradiation experiments in Material testing reactors

Severity of damage measured by dpa (displacement per atom)
- **ITER ~1 dpa**
- **DEMO 20-50 dpa**

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Liquid metals to be presented in the next seminar by D. Andruczyk

Plasma Facing Materials (Armor)

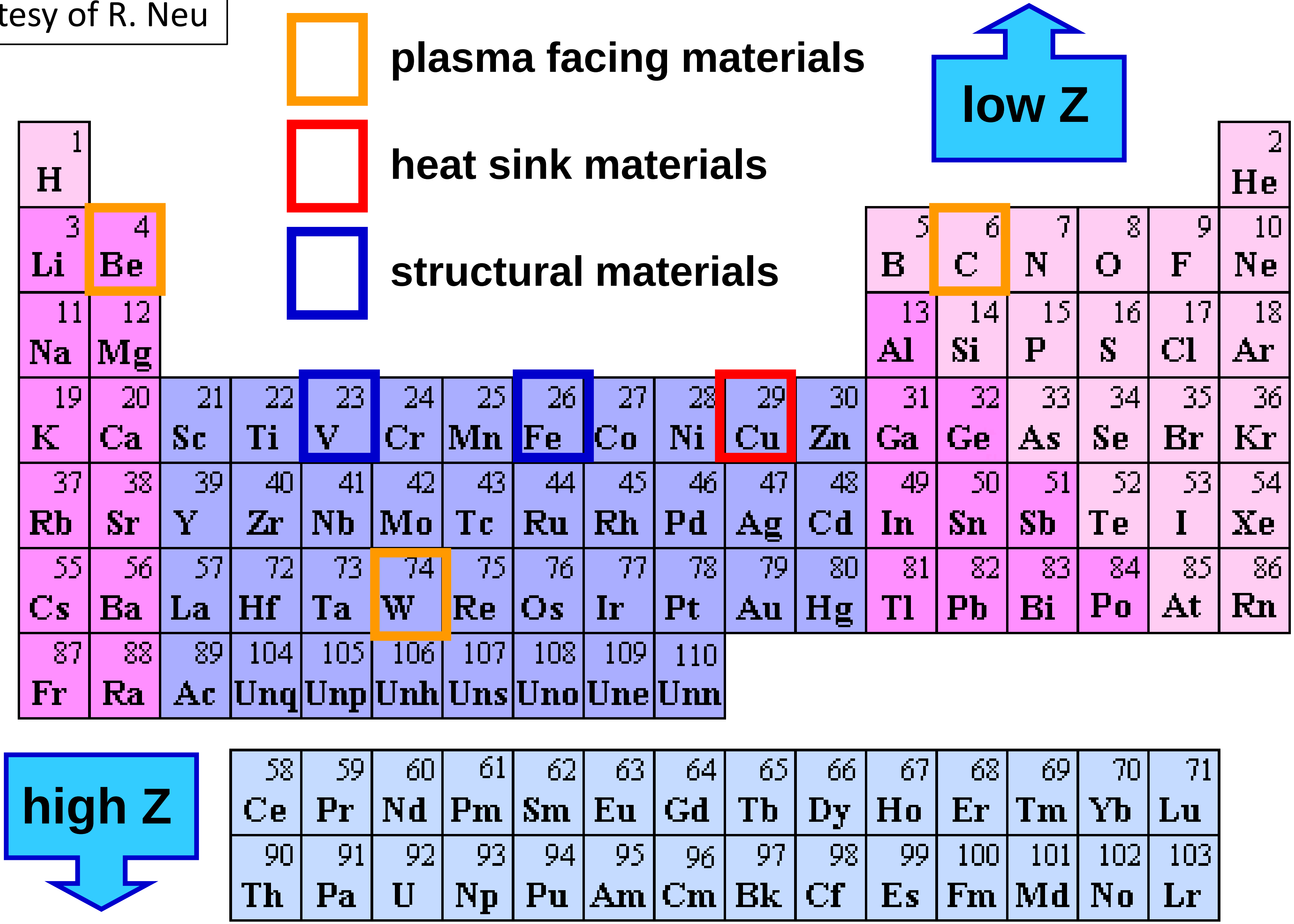
Materials for plasma facing components

Courtesy of R. Neu

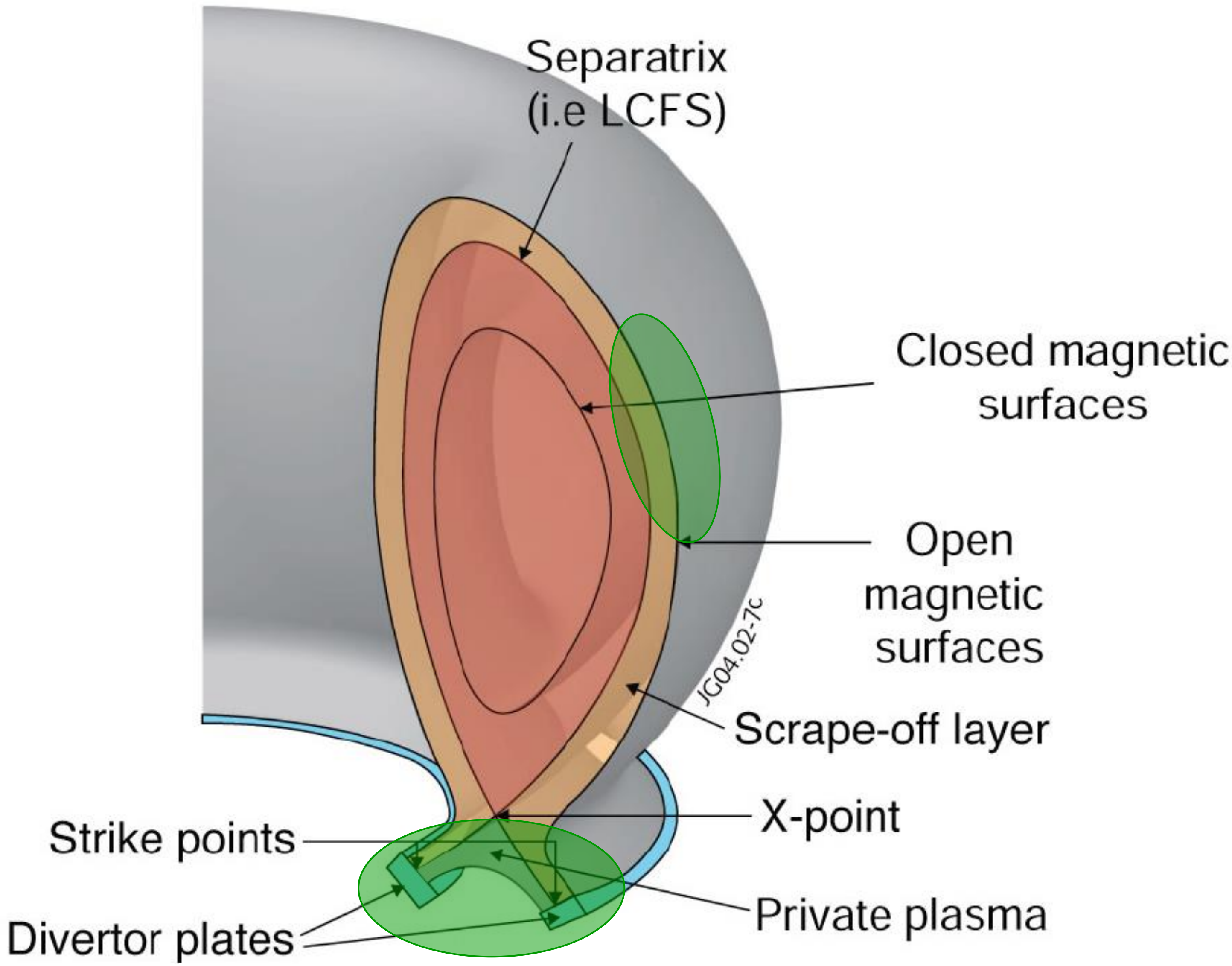
plasma facing materials

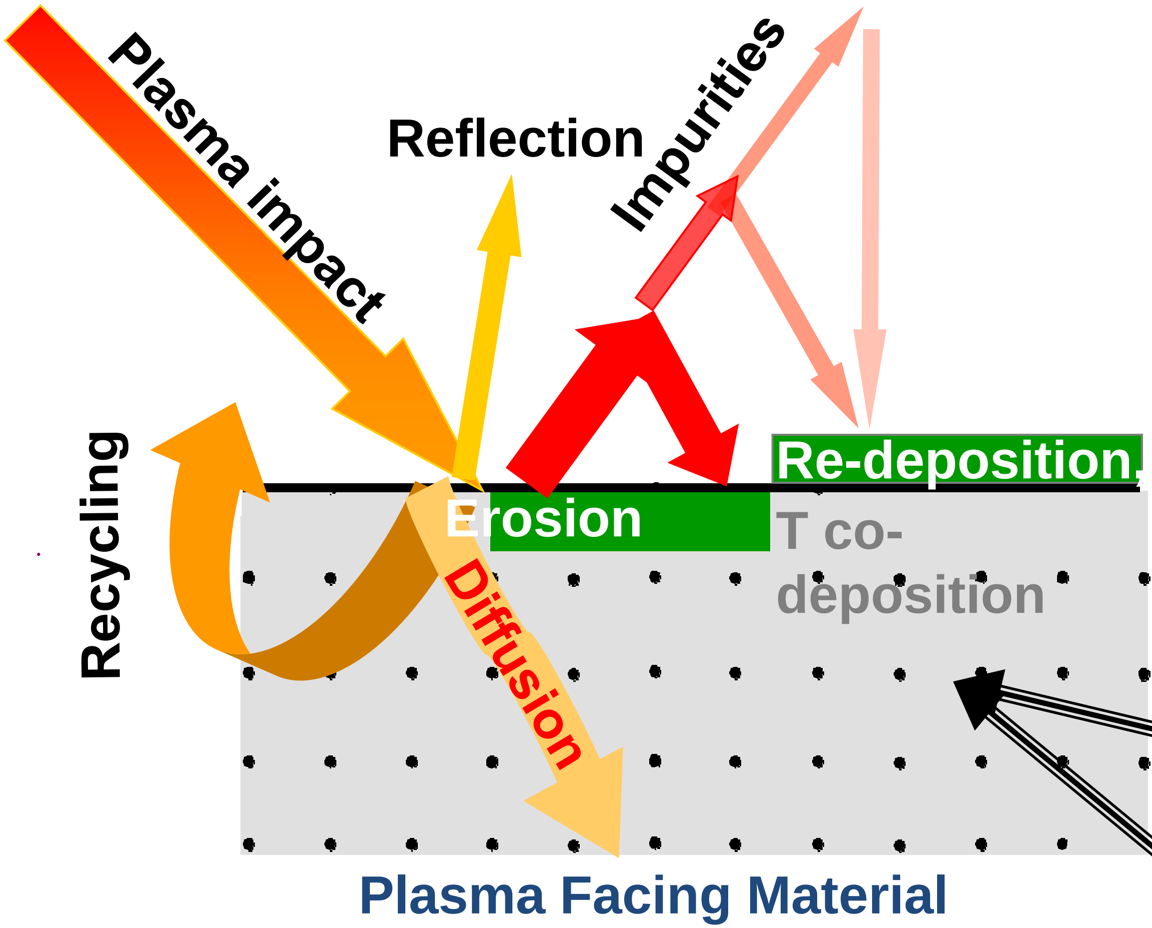
heat sink materials

structural materials



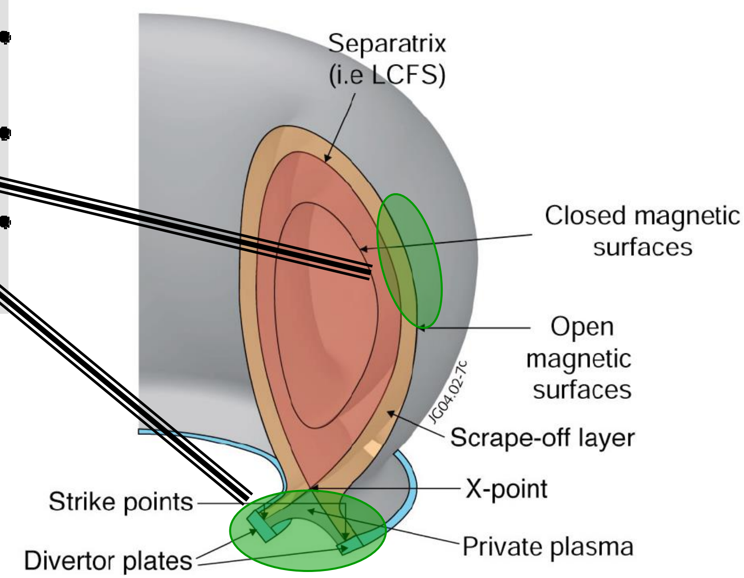
Plasma Wall Interaction in Fusion Devices





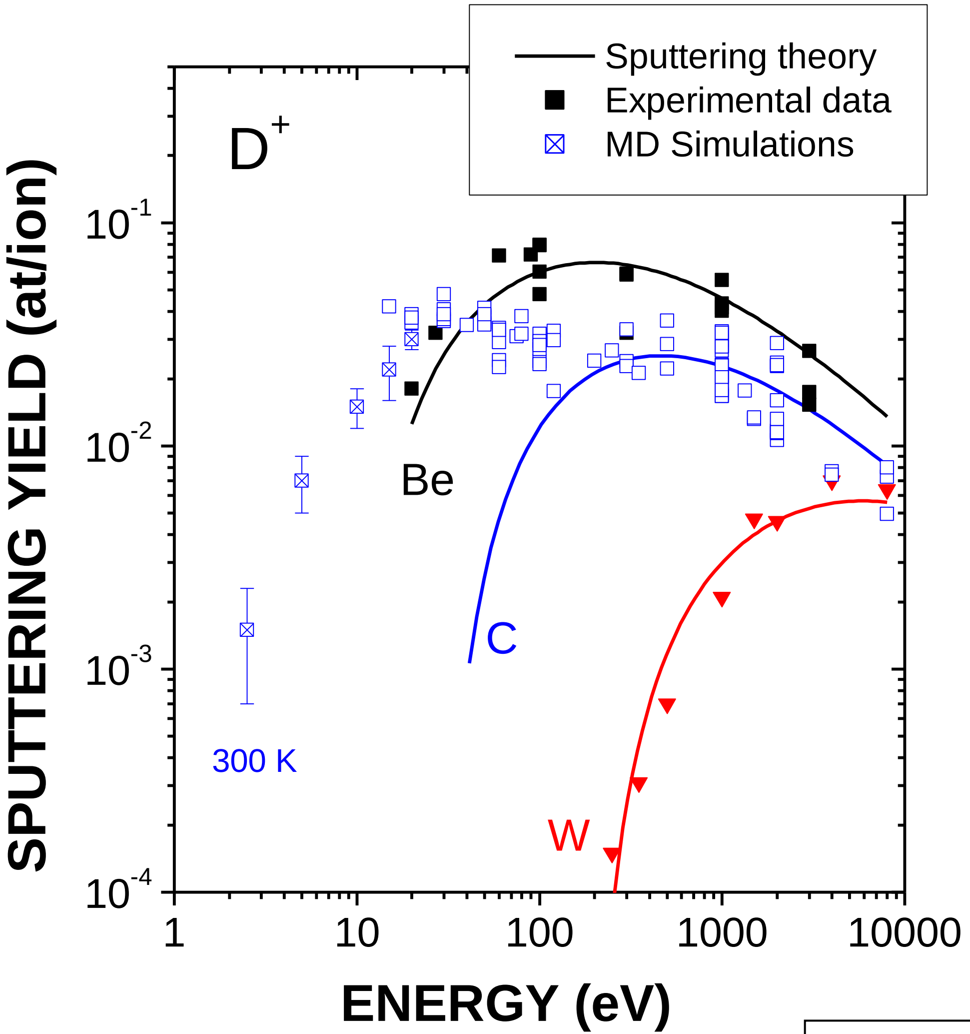
PWI & PFM determine

- component lifetime
- T retention
- dust production
- plasma compatibility



Sputtering yields of PFM

Erosion assessment from laboratory data:



Physical sputtering: understood and well predictable

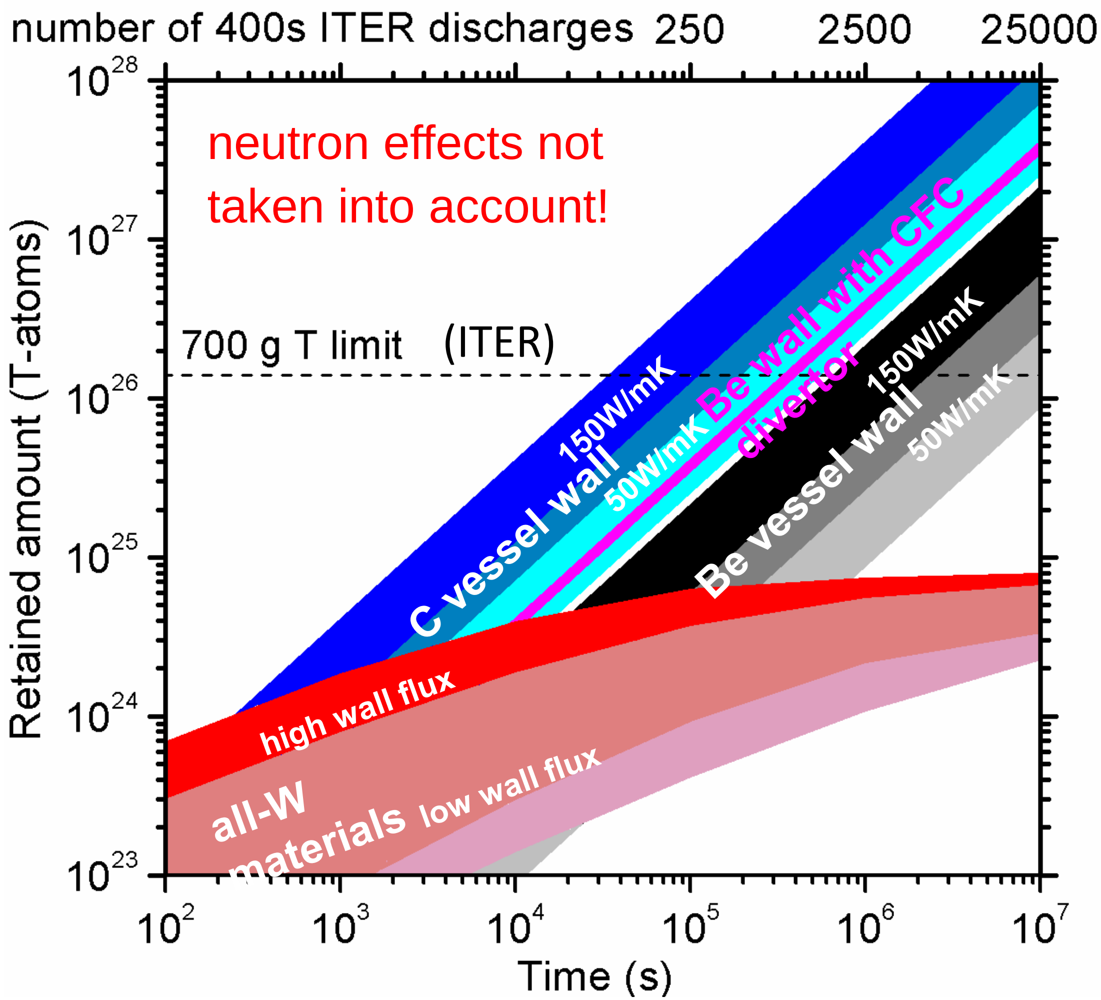
Chemical sputtering: complicated, multi-step process
can be strongly modified by material mixing
surface carbides inhibit chemical erosion

E. Salonen, Phys.Rev.B 2001, M. Balden, J.Nucl.Mat. 2000

T retention in PFM Projection to ITER

T retention given by
- Co-deposition
- Diffusion

Strongly dependent on
- background plasma flux
- erosion/deposition fluxes
- power fluxes / surf temperatures
- materials change under
  - He impact and
  - neutron irradiation



J. Roth et al., IAEA 2008

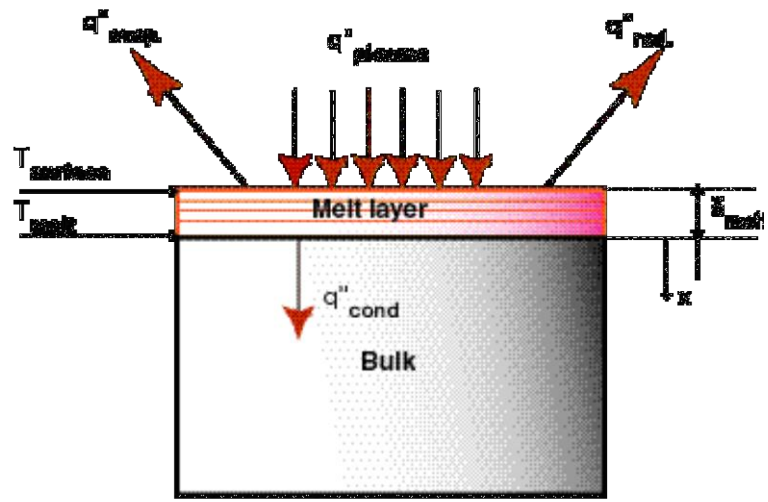
Properties of PFM candidates

	Be	CFC	W
atomic number Z	4	6	74
max. allowable concentration in the plasma	~3 %	~2 %	**~20 ppm**
thermal conductivity λ [W/mK]	190	200 ... 500	140
melting point [°C]	**1285*** **	>2200 (subl.thr.)	3410
coefficient of thermal expansion [10^{-6} K^{-1}]*	11.5	~ 0 **	4.5
n-irradiation behaviour	swelling	**decrease in λ**	activa-tion

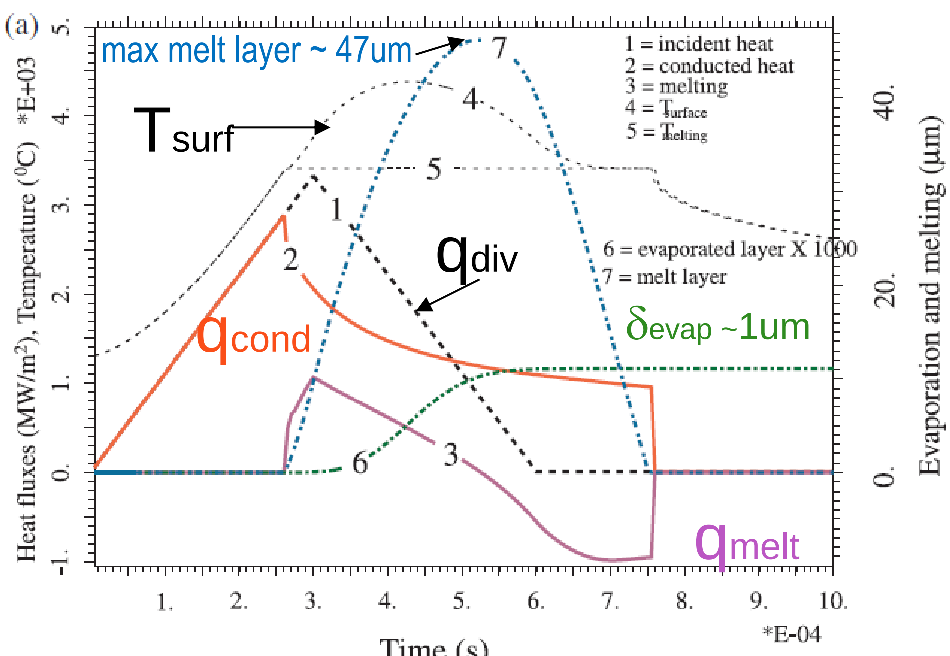
- CTE copper = $16 \cdot 10^{-6}$ K^{-1}
- ** NB31 in pitch fiber direction
- *** Be not suitable for divertor. Be/W mix less stable

Thermal loads during fast transients: ELMs

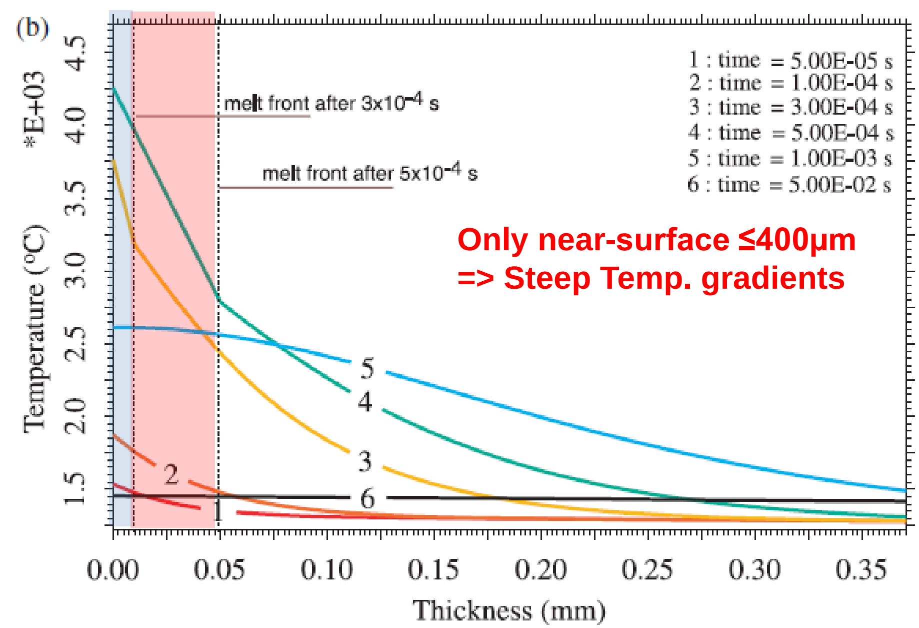
- Critical parameters are:
  - (1) energy loss from pedestal,
  - (2) fraction reaching the divertor,
  - (3) wetted area,
  - (4) duration/shape of ELM heat pulse.



e.g. E_{ELM}=1MJ/m^2 on 10mm W armour, inter ELM = 10MW/m^2

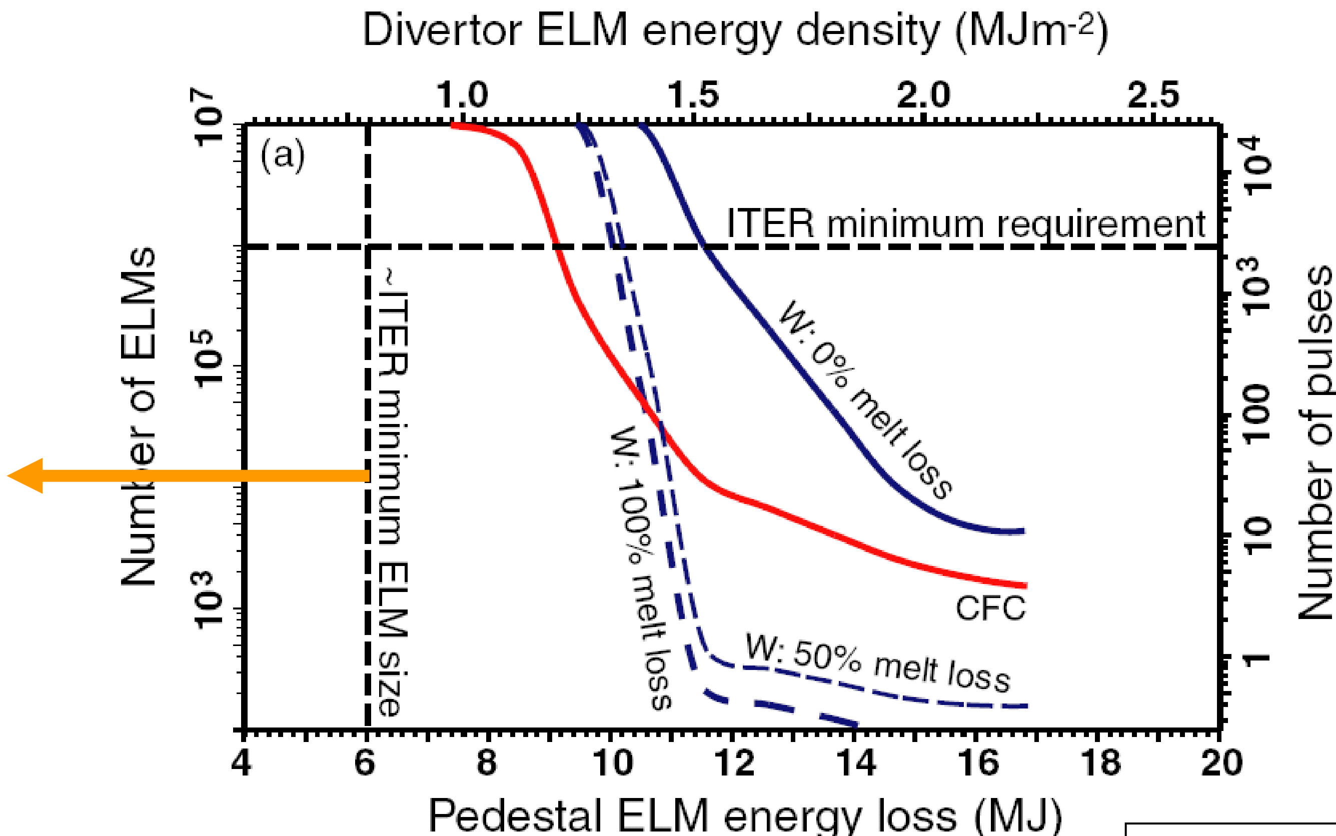


Vaporised depth ~1um
max melt layer ~ 47um



G. Federici et al., PPCF **45** (2003) 1523

ELM-size determines lifetime of ITER divertor



Federici et al., PPCF 2003

maximum ELM energy due to thermal fatigue (see below) →1 MJ in the divertor

Thermal loads during fast transients: ELMs

Cumulative material damage during ELMs

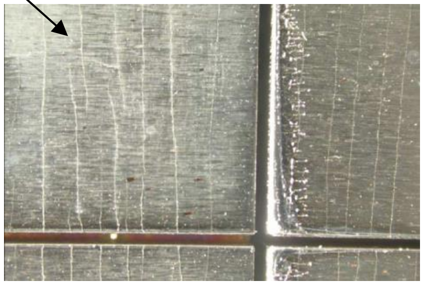
EU/RF collaboration, with experiments in Russian plasma guns and EU modelling

- ITER relevant thermal loads
- Tungsten target prior to exposure

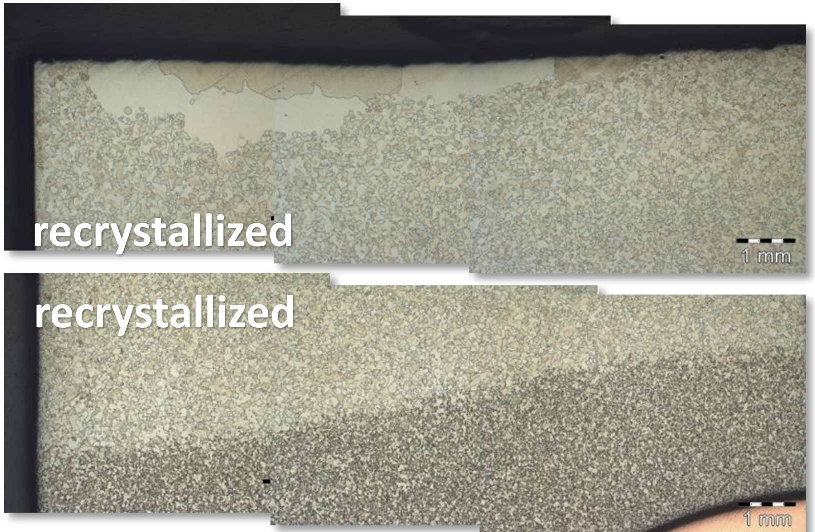
R. Giniyatulin (Efremov Inst.)

Zhitlukin, Safronov, Podkovyrov et al. (SRC RF TRINITI, Troitsk), Loarte, Merola

- Tungsten target after 5 shots in QSPA plasma gun test facility
  - ✓ 1 MJ/m^2
  - ✓ 0.5 ms
- Shots at 1.5-1.8 MJ/m^2
  - ✓ At 1.5 MJ/m^2 melt loss 15 mg/ELM, erosion dept **~0.3 μm/ELM**
  - ✓ (Model RACLETTE: **0.2 μm** @ these conditions).

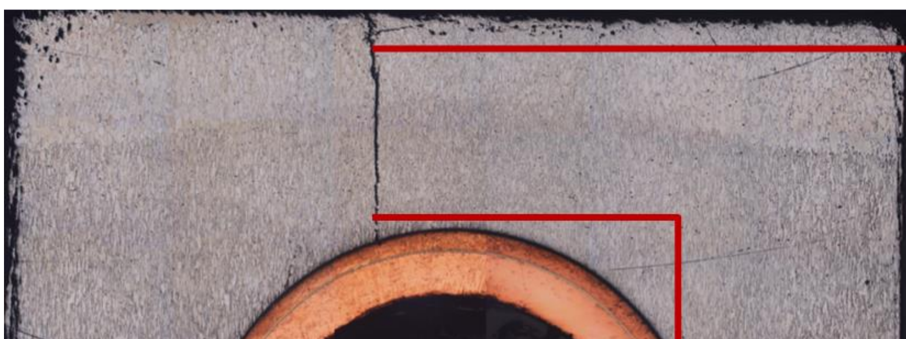


THERMAL steady state

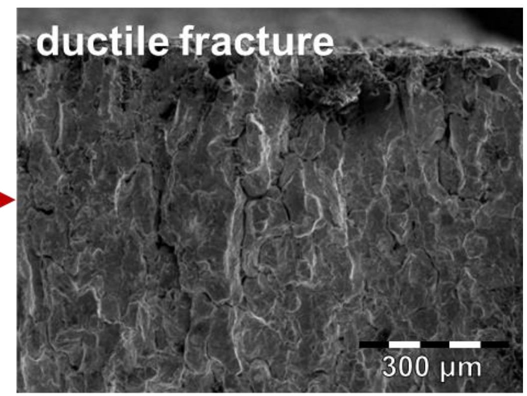


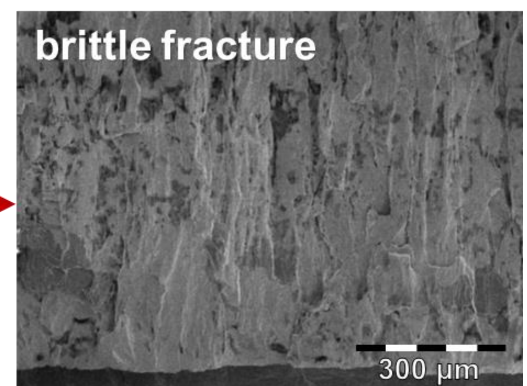
G. Pintsuk, FED 88 (2013) 1858– 1861

W-recrystallization/grain growth: (up to several mm): Temp limit 1300°C depending on **power density**, **joint quality**, **tungsten thickness** and **material properties**



ductile fracture region related to recrystallized zone?



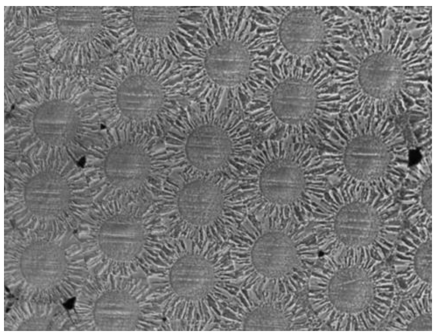


Improvement of tungsten properties: W_fW

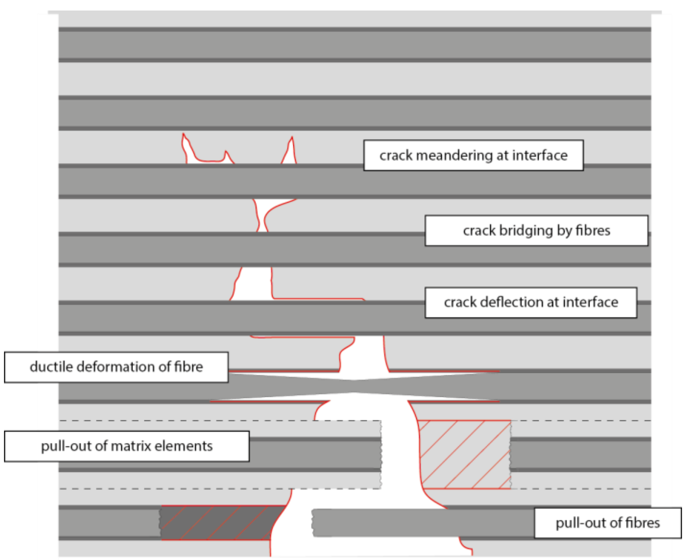
Production of **tungsten-fibre rein-forced tungsten** by chemical infiltration (CVI) of a W-wire arrangement

‚Proof of principle‘ successful:

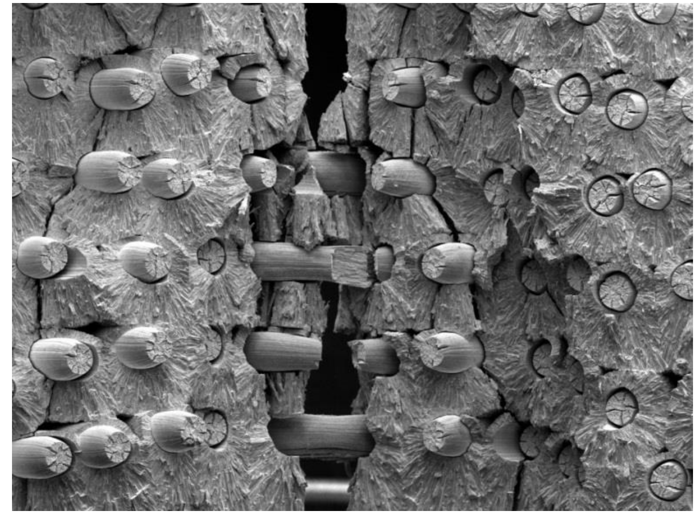
- high material density achieved
- strongly improved ductility



Thesis
J. Riesch,
TUM, 2012



Fracture surface of W_f/W after Charpy impact test



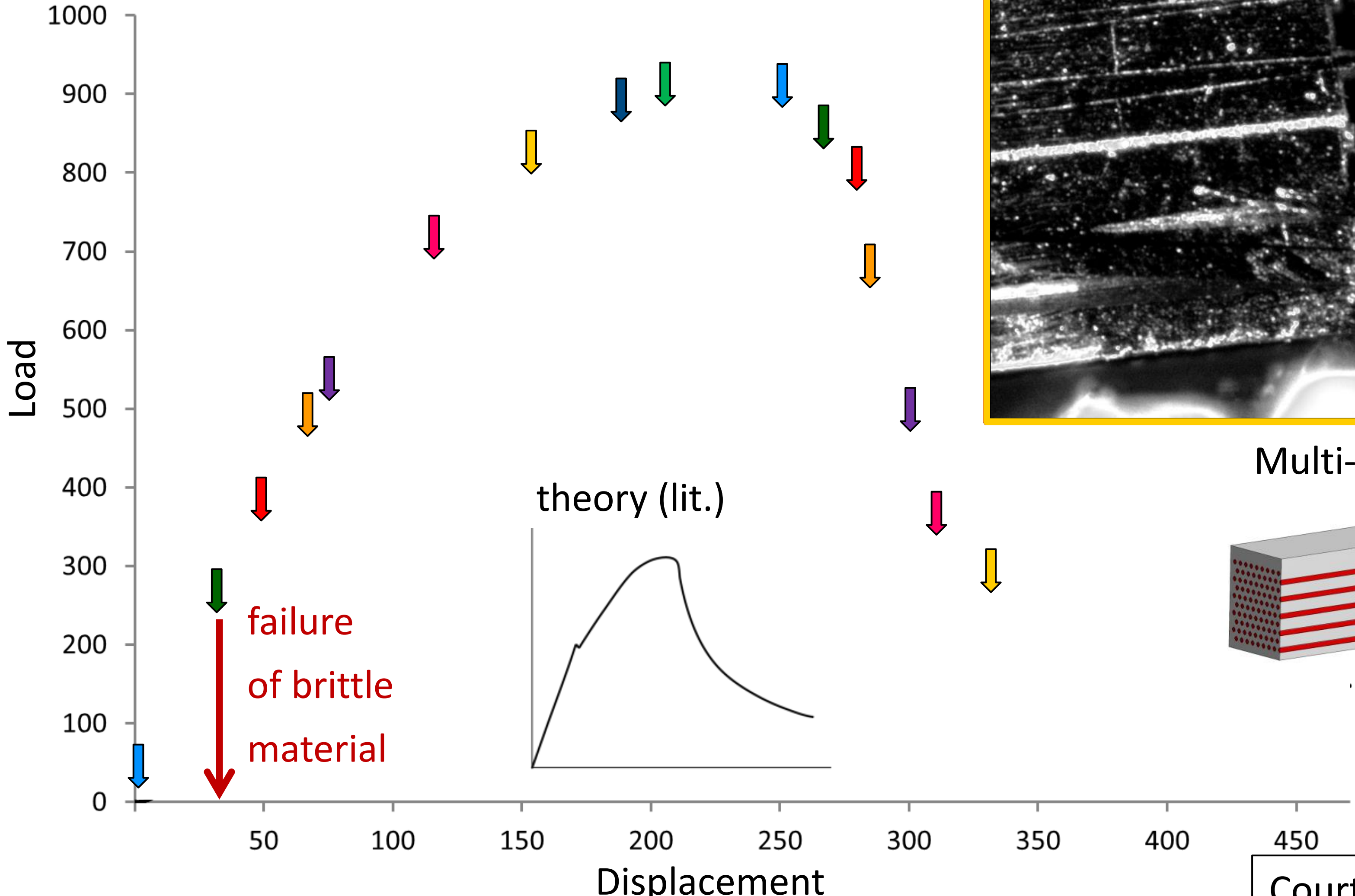
Aim

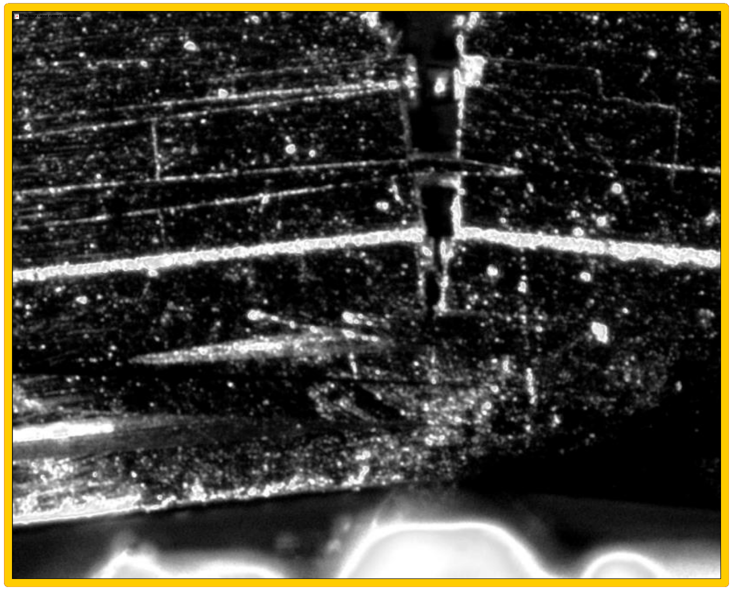
- Increase the toughness of W (resistance against cracking)

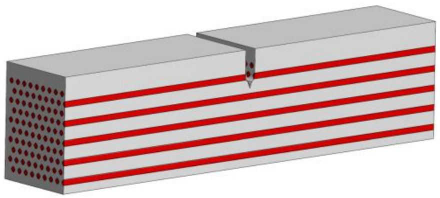
Characterization of toughness

- Charpy impact tests
  $\rightarrow$ ductile fibres
- Monotonic tensile test
  $\rightarrow$ no catastrophic failure
- Low cycle fatique testing
  $\rightarrow$ 10000 cycles at 60% of maximum stress reached even without optimized material

Improvement of tungsten properties: W_fW







Multi-fibre sample

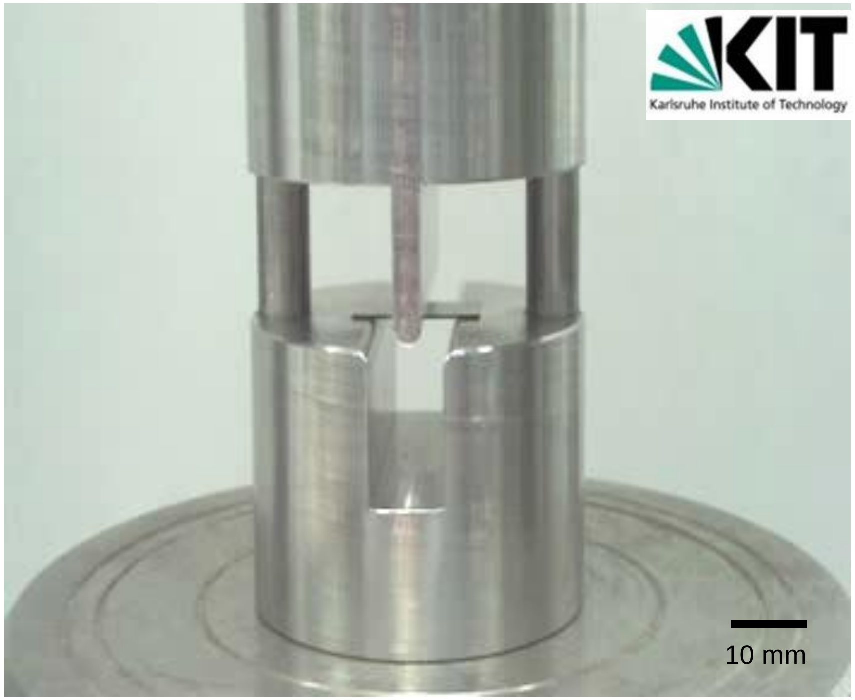
Courtesy of R. Neu

PFM: Developments for DEMO

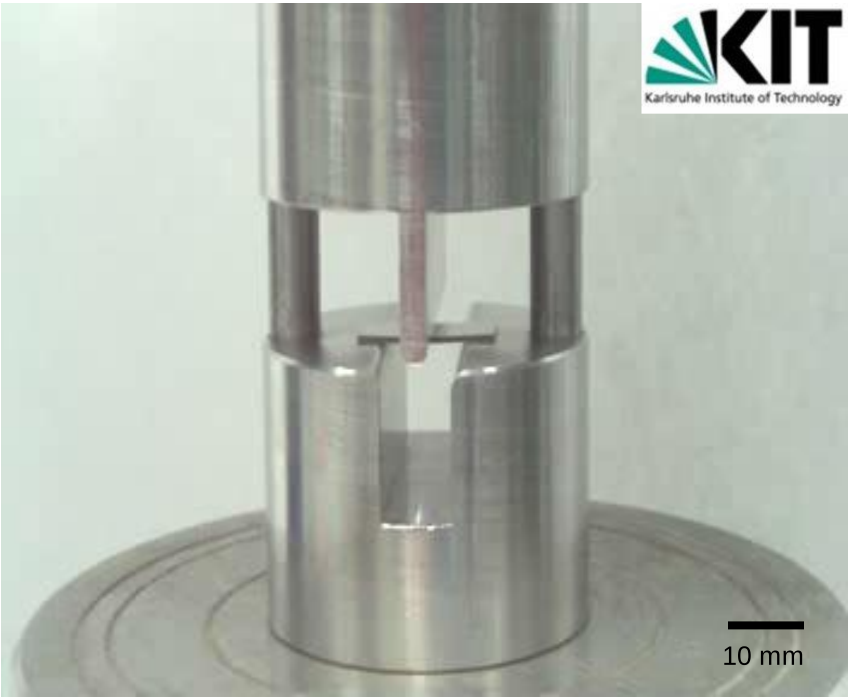
Refractory Materials for DEMO Divertors

In close cooperation with Plansee company

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



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



→ Severe cold-rolling makes W ductile

J. Reiser et al., Int. J. Refract. Met. Hard Mater. 64 (2017) 261–278

Material improvement

Copper-alloys (Heat sink materials)

Heat sink: material requirements

Solid elements with thermal conductivity κ > 50 W/mK (RT)

κ = listed in κ descend order

2		3/4	2	5	4	5*	5/6	3	5
Ag	Cu	Au	Al	Be	Ca	W	Mg	Rh	Si
429	401	317	237	201	201	174	156	150	149
3	1	2	2/5	3	1	1	2	1	5
Ir	Na	Mo	C	Ru	Zn	K	Co	Cd	Cr
147	141	138	129	117	116	102	100	97	94
2	3	1	1		2/3	3	1	5	1
Ni	Os	Li	In	Fe	Pd	Pt	Sn	Ge	Rb
91	88	85	82	80	72	72	67	60	58
	2	2	2	4	Failed criterion				
Ta	Th	Nb	Tc	As	element				
58	54	54	51	50	conductivity				

1. Melting point < 500°C
2. Activation/H solubility
3. Availability/Cost
4. Strength
5. Ductility
6. Water corrosion

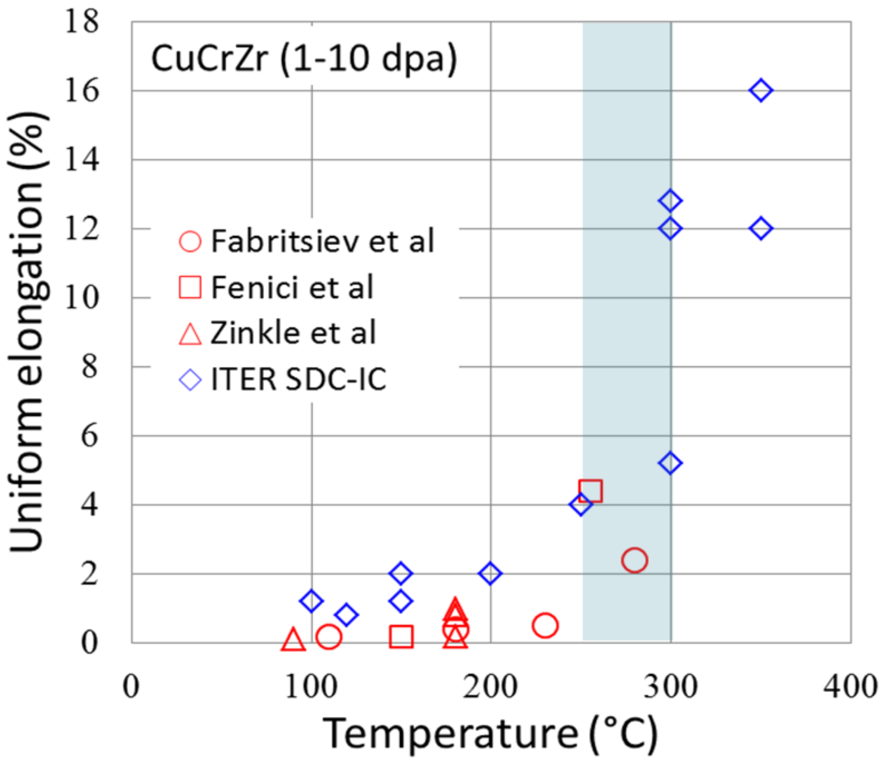
*water-cooled case (< 350°C)

J.H. You, Nucl. Fusion (2015)

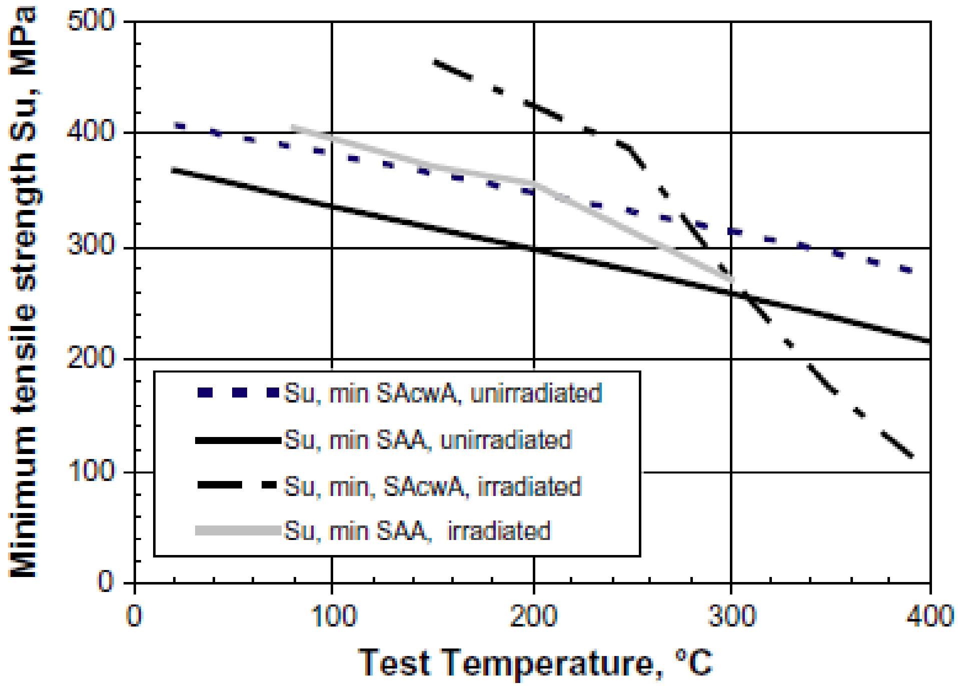
Irradiation effects in CuCrZr

Courtesy of J.H. You,

Embrittlement by irradiation



Softening by irradiation (ultimate stress-Temperature)



(SAcwA) solution annealing, cold working and ageing
(SAA) solution annealing and ageing
N.B. values for exposure above irradiation hardening saturation dose (>0.5 dpa).

Lower bound: 150°C

S.A. Fabrisiev et al. JNM (1996)

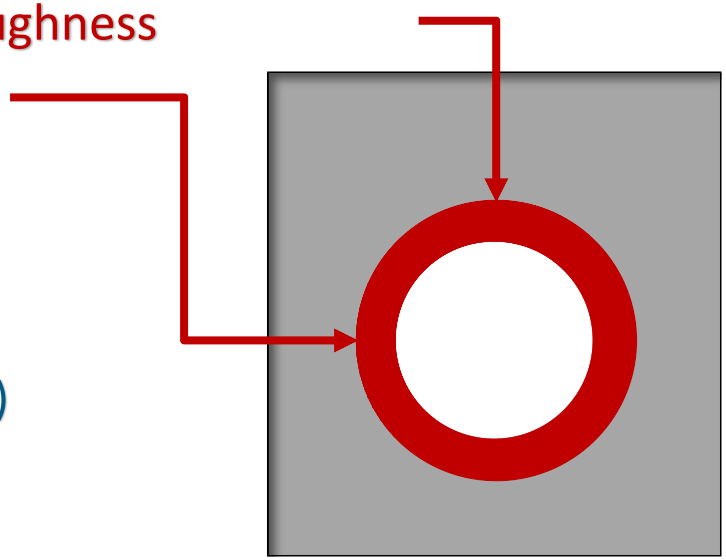
Upper bound: 300°C

V. Barabash et al. JNM (2011)

Heat sink: developments for DEMO

need to increase the mechanical strength and toughness

need to decrease CTE difference to PFM

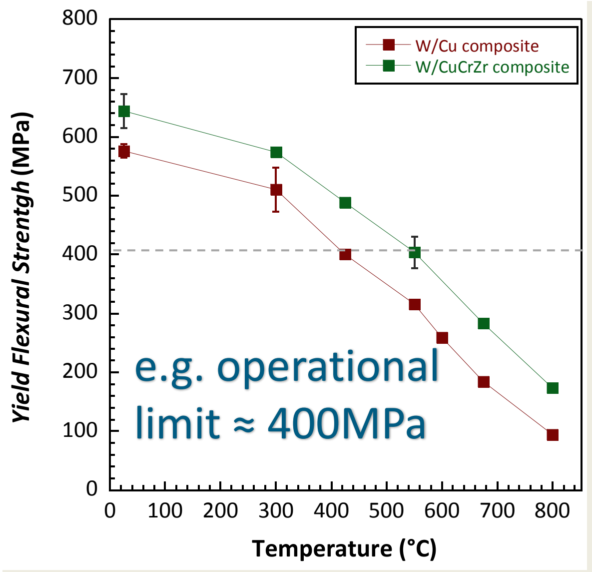


- particle reinforced materials, e.g. W_p/CuCrZr, …
- CuCrZr + X (X = Ta, V, …)

Issues

- increase of operational window (CuCrZr: 300-350°C)

W_p/CuCrZr (industrially available)



Heat sink: developments for DEMO

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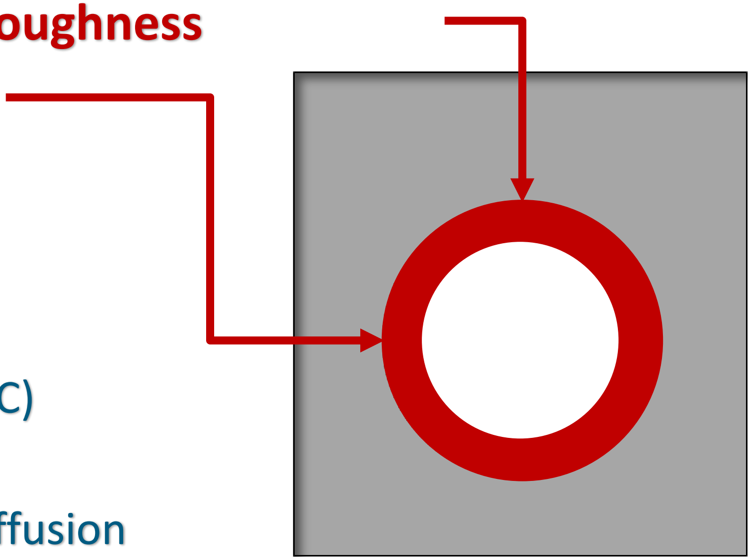
need to **decrease CTE difference to PFM**

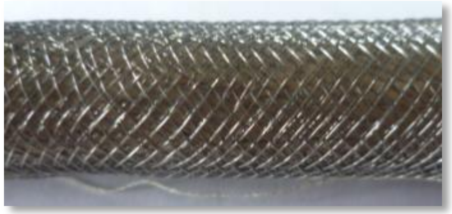


- fiber reinforced materials, e.g. W_f/Cu(CrZr), …
- laminates, e.g. W/Cu, W/V, W/Ti …

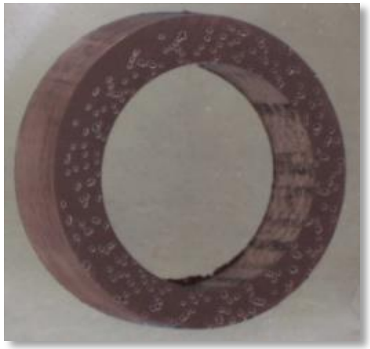
Issues

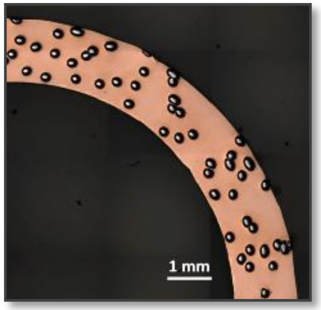
- increase of operational window (CuCrZr: 300-350°C)
- **fiber architecture – 2D/3D**
- microstructural stability multilayer systems incl. diffusion barriers) joints: laminate/PFM, laminate/steel (leak tightness)

W_f/Cu(CrZr)

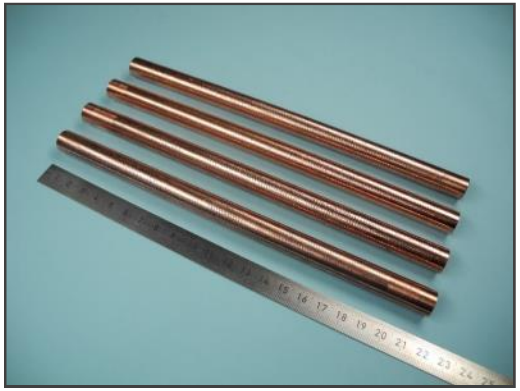


multilayer braiding





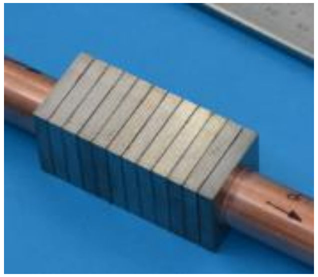
pipe cross section



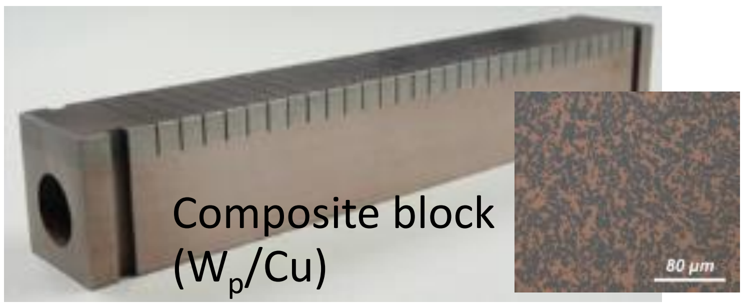
W fibre reinforced Cu pipes (length: ~200 mm)

Technology tested on small scale mock-ups

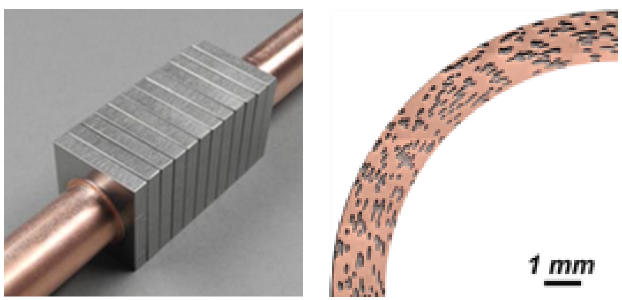
- Fabrication technology fully established.
- Mock-up production mostly completed.
- High-heat-flux testing reached 500 load cycles.



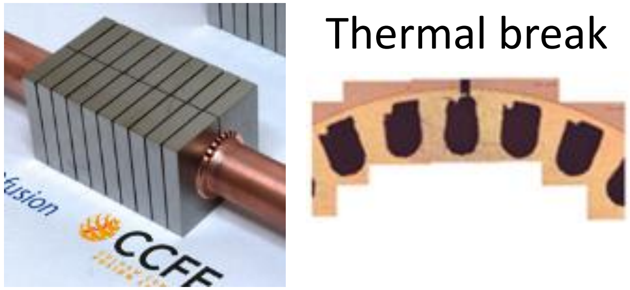
ITER-like



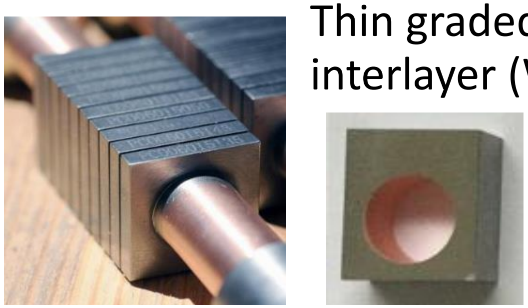
Composite block (W_p/Cu)



Composite pipe (W_f/Cu)



Thermal break

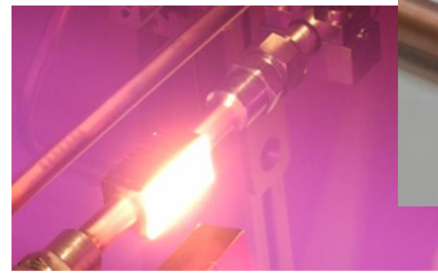


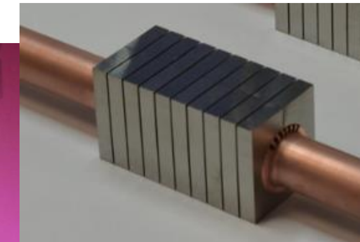
Thin graded interlayer (W/Cu)

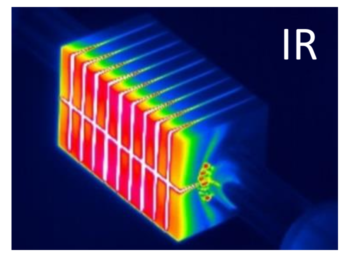
J.H. You et al., SOFT 2018

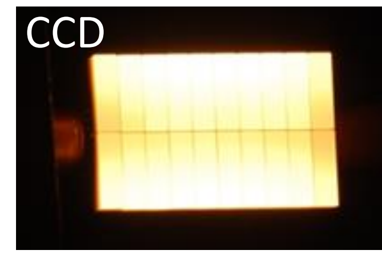
High-heat-flux (HHF) fatigue test: water-cooled

GLADIS: neutral beam (H/He) irradiation facility (IPP)









Technical data:

- Power: 2 x 1 MW ion sources
- Voltage: 15 - 50 kV
- Heat flux: 2 - 45 MW/m²
- beam size: ∅ 70 mm (80% central q')
- Pulse duration 1 ms - 45 s

Cooling

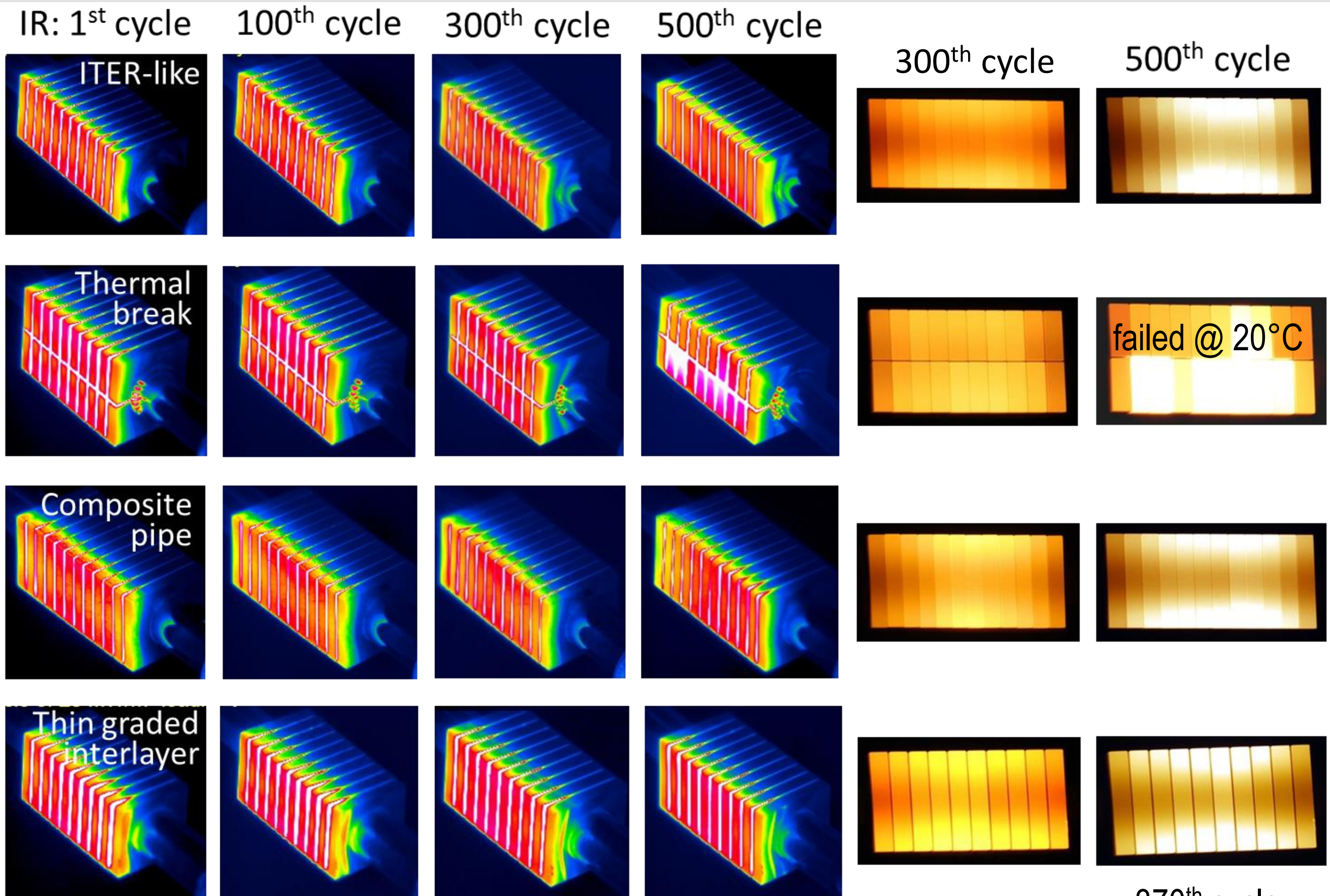
- T_{in}: 20 - 230 °C, T_{out}: < 250°C
- Flow rate: ≤ 2 (8.5) l/s, p ≤ 55 bar

Diagnostics

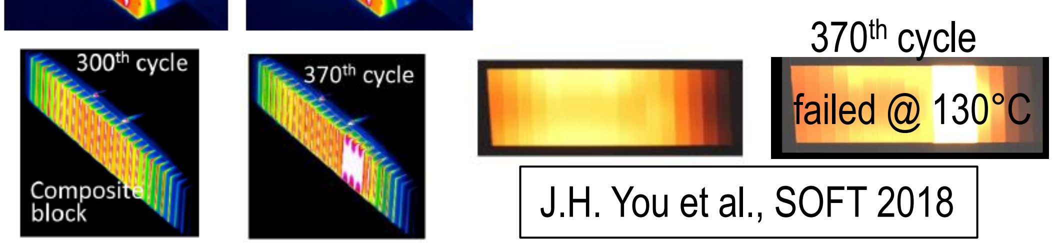
- Water calorimetry (thermocouples)
- Fast one and two-colour pyrometers
- High resolution CCD & IR cameras

HHF fatigue test: water-cooled (20MW/m², 130°C)

Tested in GLADIS with temp. coolant 20°C and 130°C



Change in surface temperature indicates defect evolution or local failure which affects the heat conduction leading to a temperature increase.



J.H. You et al., SOFT 2018

Material improvement

Ferritic-martensitic steels

(Structural Material)

Effects of n-irradiation on materials

Radiation damage mechanisms: embrittlement, thermal creep, swelling, etc. to be carefully considered in the design phase (eng. approach & safety margins).

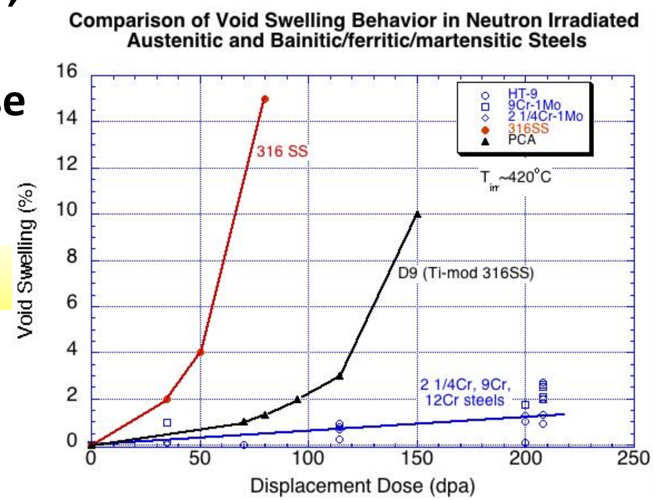
He/dpa = <1 (fission); > 10 (fusion)

H/dpa = 10 (fission); > 40 (fusion)

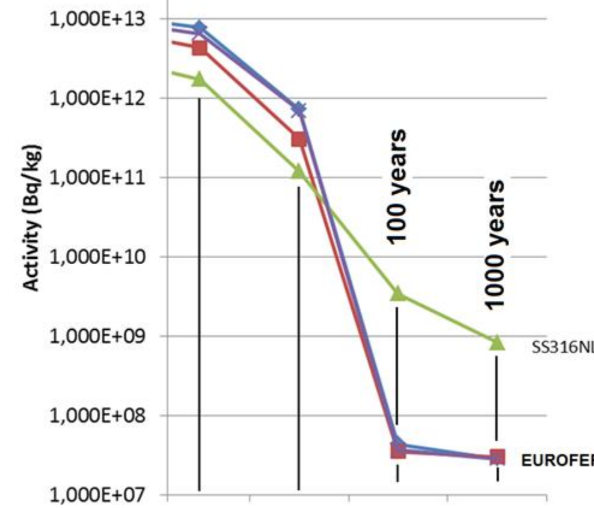
Lowest swelling occurs in BCC alloys (Ferritic steels)

J.L. Boutard

Conventional austenitic steels swell and get activated



After 6 FPY (DEMO lifetime)

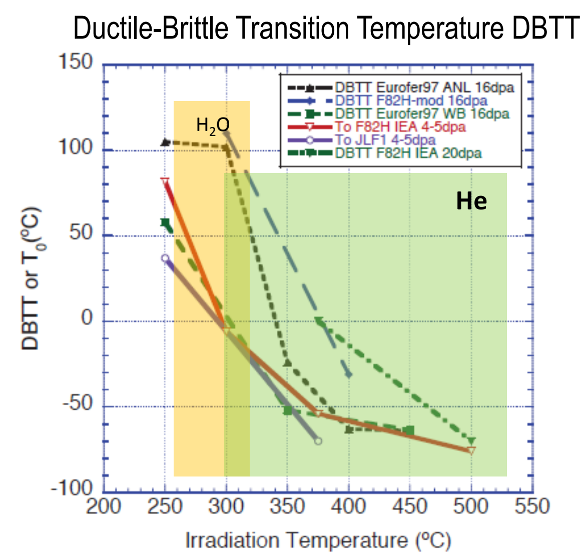


Reduced Activation FM Steels – elements that generate radioactive isotopes were replaced/reduced, e.g. Cr as major alloying element and Ta, W, V, repl. high activation elements (Ni, Al,...)

- FM steels are however subject to radiation embrittlement
- Lose mechanical strength at ~ 550°C (upper limit)
- Suffer from thermal creep (accelerated) by irradiation
- Unknown effect of helium embrittlement

Narrow design temperature operation window

❖ **High He conc. due to transmutation may further narrow design window (expected at dose > 20 dpa, i.e., 300-500 He appm)**



Baseline materials – DEMO – operation window

W (plasma facing material)

- lower limit determined by DBTT ≈ 300-400°C (non-irradiated state, strain rate dependent) and ~600-800°C (irradiated state)
- upper limit determined by recrystallization ≈ 1300°C (impurity dependent)

→ 800°C – 1300°C

CuCrZr (heat sink material)

- lower limit maybe determined by radiation hardening ≈ 250-275°C
- upper limit determined by material strength (softening)

→ 275°C (150-200°C?) – 350°C

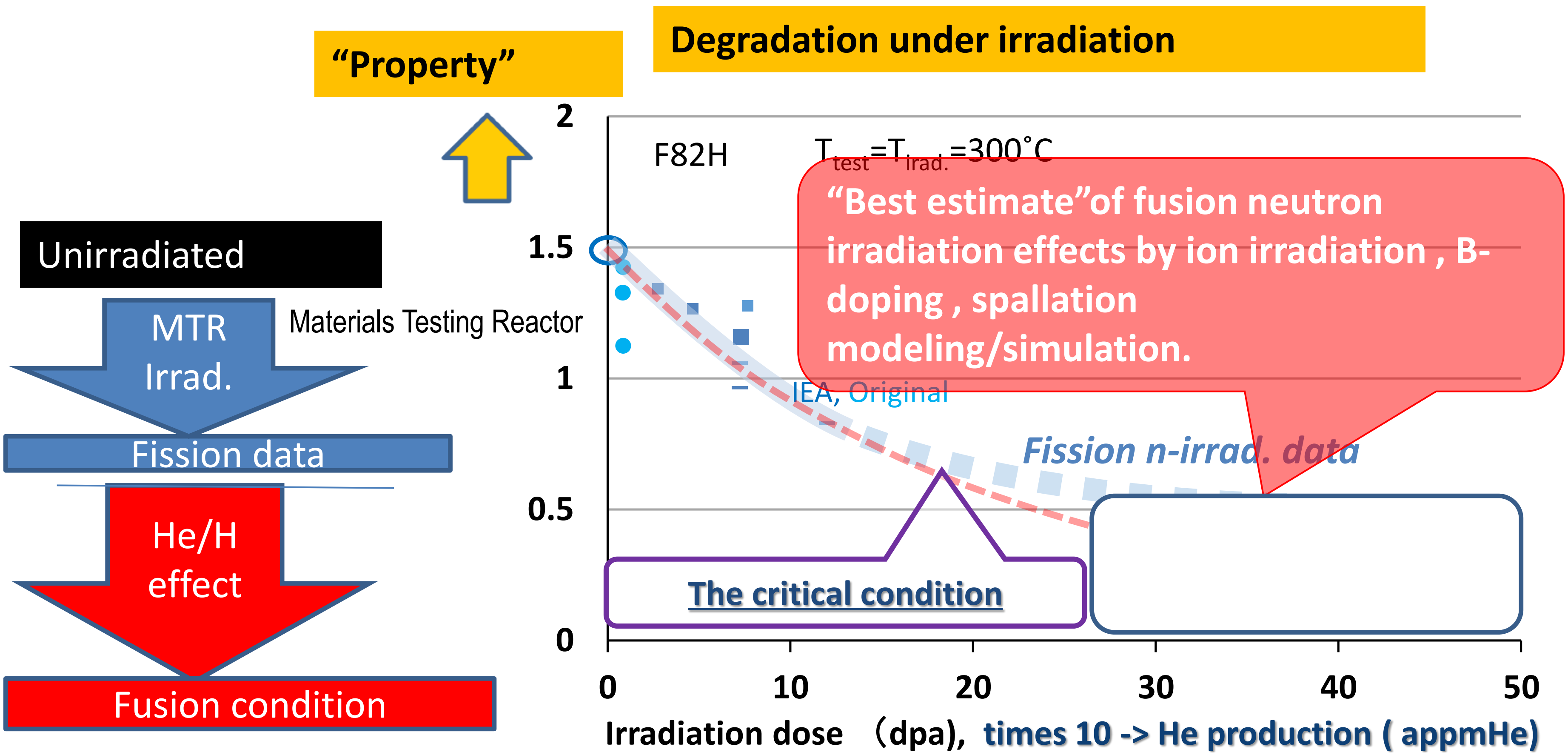
EUROFER97 (structural material)

- lower limit determined by DBTT ≈ -50°C (non-irradiated state) and ~200-300°C (irradiated state, ≤ 20 dpa)
- upper limit determined by material strength (softening)

→ ~300°C – 550°C

Irradiation effects – From MTR/Fission -> Fusion "Estimates"

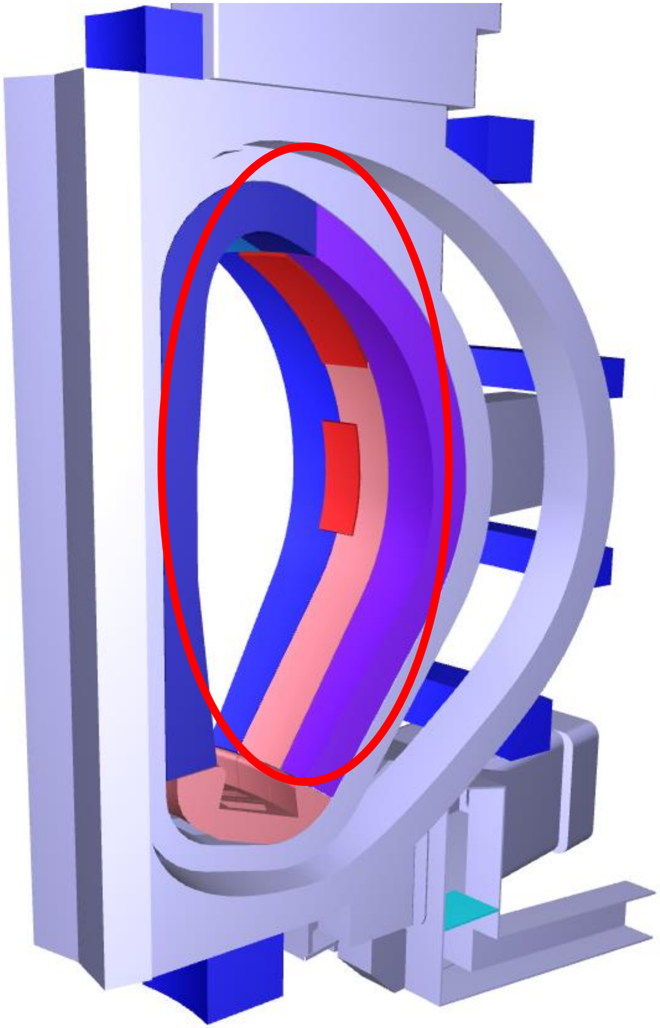
Modified from: Hiroyashi Tanigawa, QST



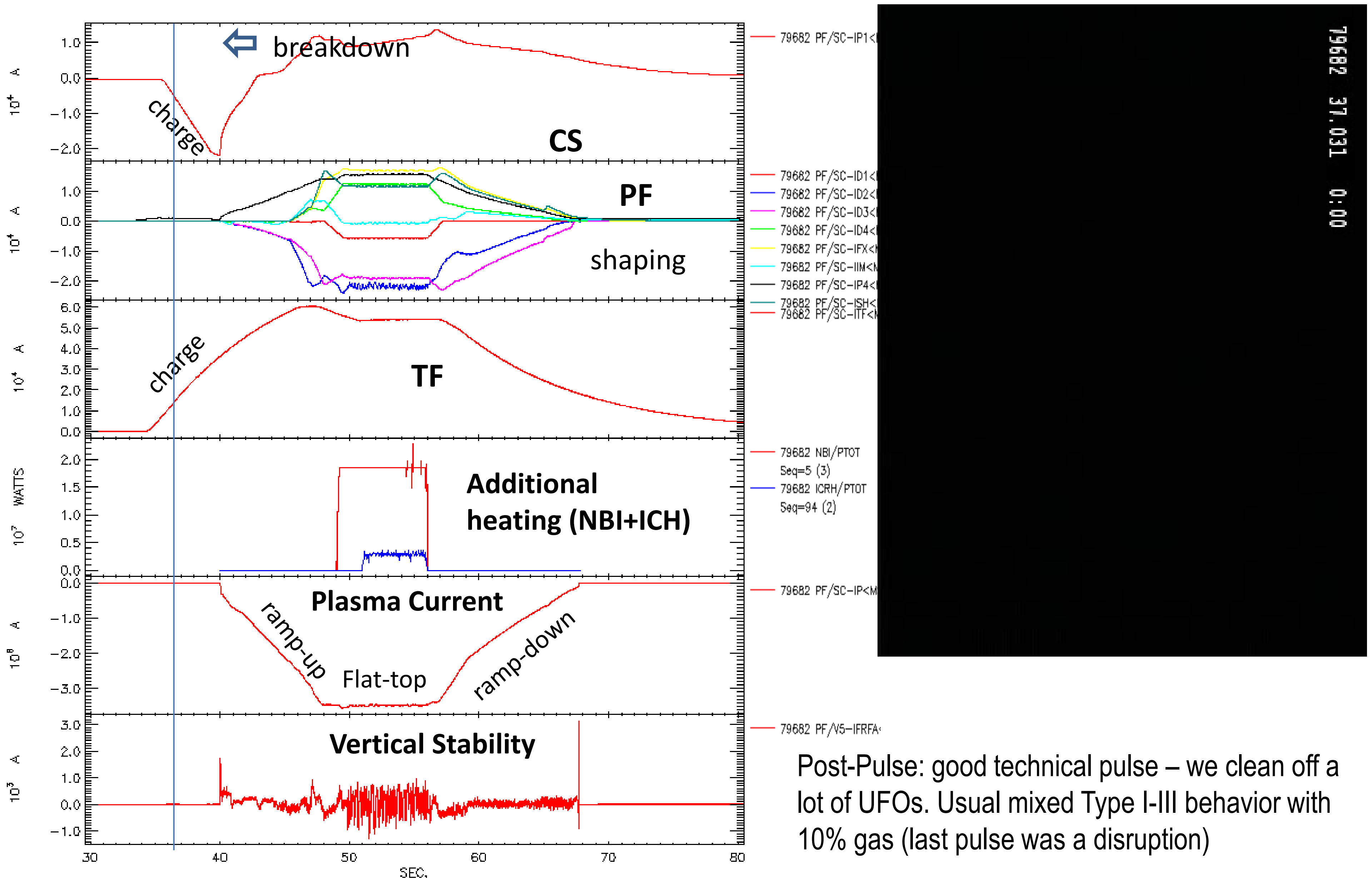
- ✓ **Need database from MTR & modeling/simulation and … as 1st estimate**
- ✓ **14MeV fusion neutron irradiation (like IFMIF-DONES) will be essential for both, validation/confirmation and reduction of unnecessary conservatism in "allowables"**

Outline

- Introduction: ITER and DEMO PFC requirements
- Materials R&D for DEMO – the baseline options
  - Plasma Facing Materials (Armor)
  - Copper-alloys (Heat sink materials)
  - Ferritic-martensitic steels (Structural Materi
- DEMO heat load requirements
  - First Wall (FW) and Limiters
  - Divertor
- Conclusions



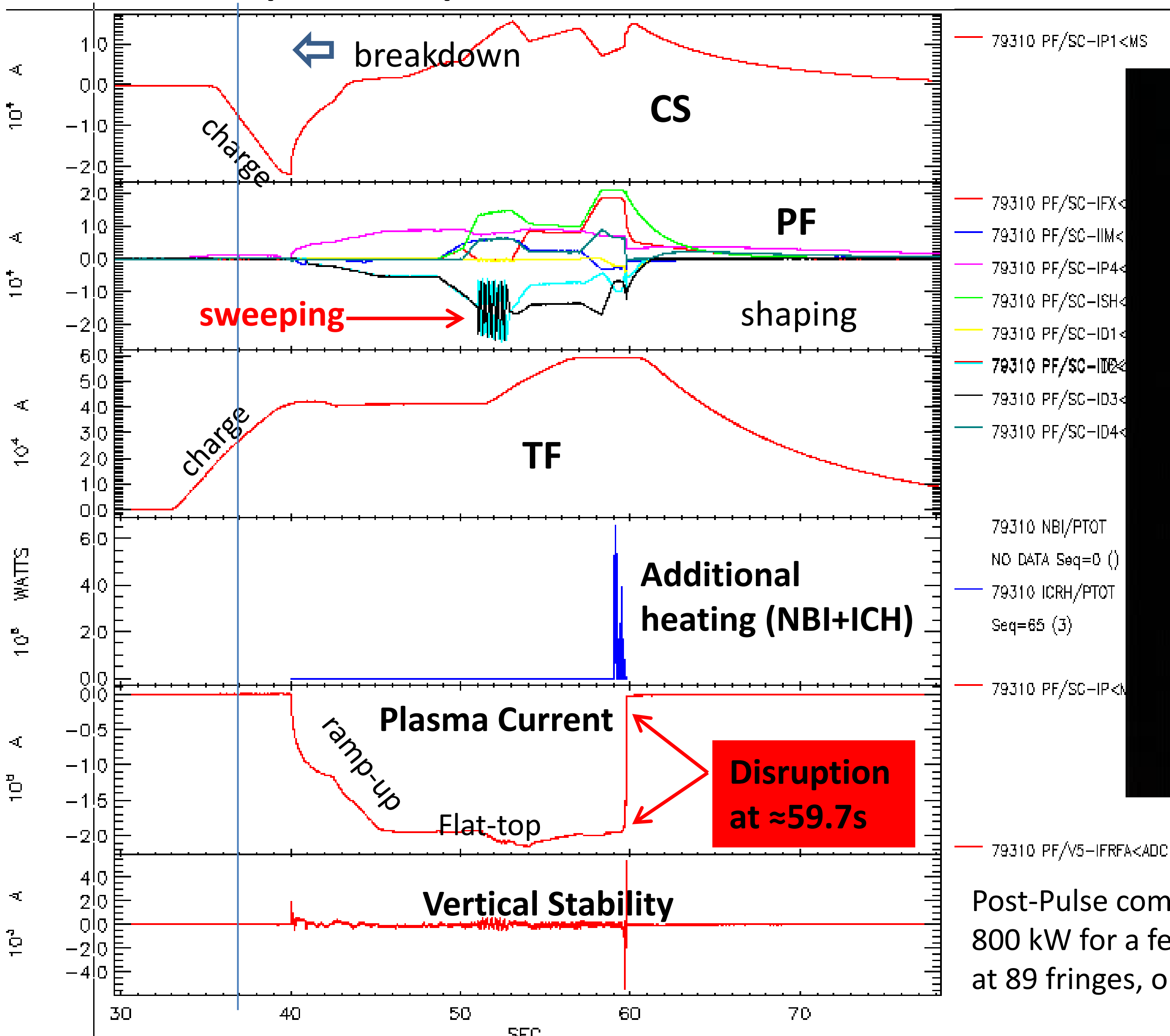
Plasma scenario: JET example

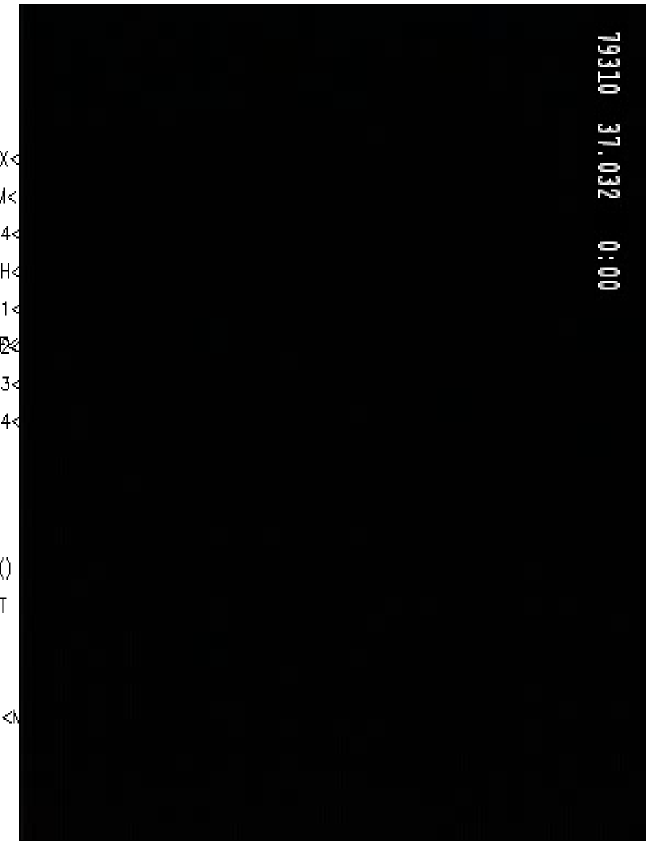


Post-Pulse: good technical pulse – we clean off a lot of UFOs. Usual mixed Type I-III behavior with 10% gas (last pulse was a disruption)

Plasma scenario: JET example

Plasma disruption: JET pulse 79310

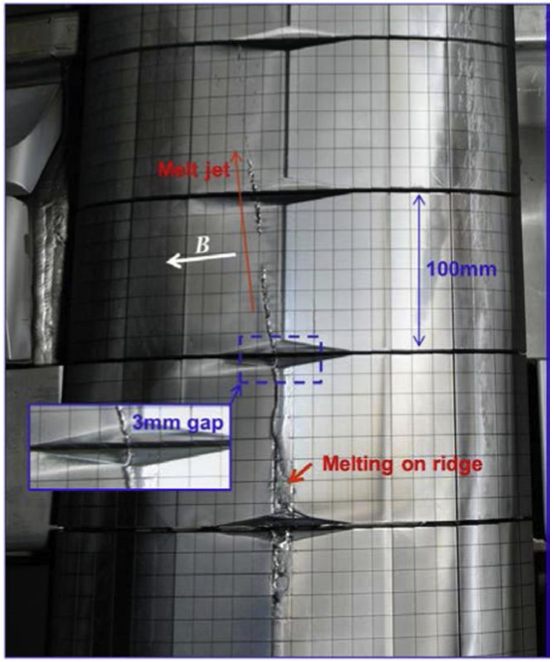


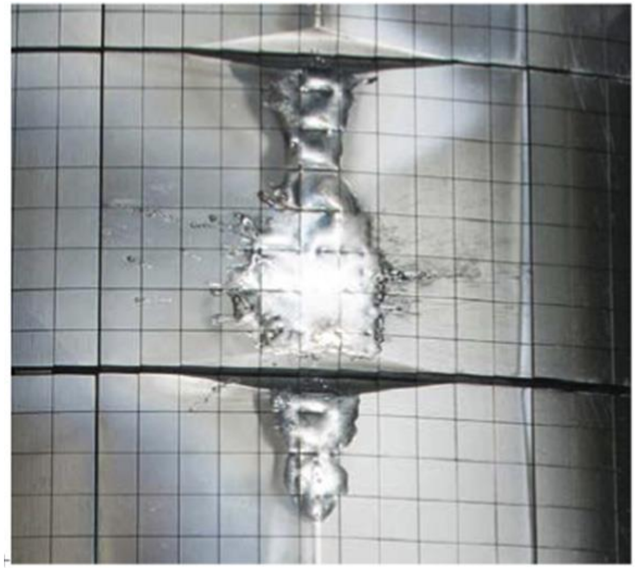


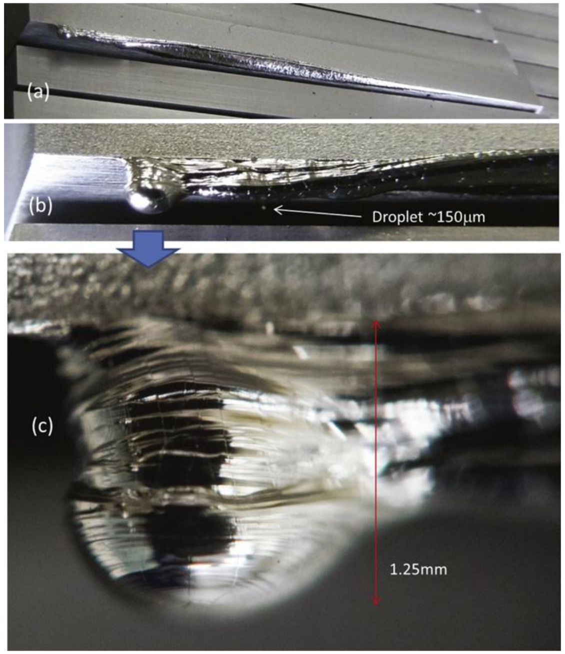
Post-Pulse comment: RF tripped completely only 800 kW for a few hundred ms. Disrupted at 59.7 at 89 fringes, ohmic density limit.

Plasma scenario: JET example

Possible damages due to plasma transients







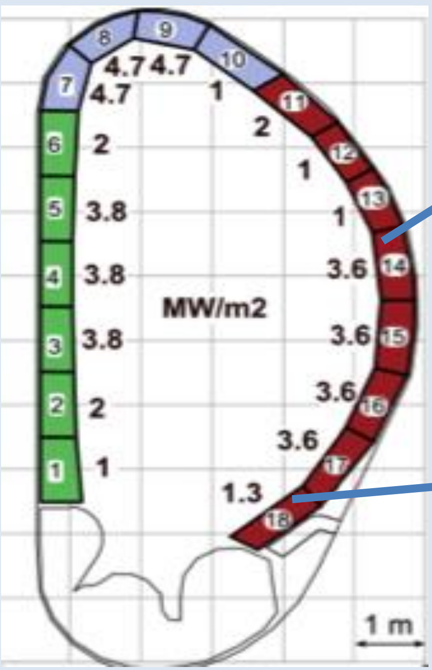
Slow melting of ILW Be limiters during plasma limited phase	Melting of Be limiter due to unsuccessfully mitigated runaway electrons (REs) experiment following a disruption	Image of the melted edge of the special divertor tungsten lamella during ELM-induced transient W melting

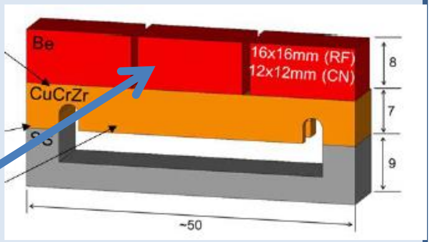
G F Matthews, et al., Phys. Scr. T167 (2016) 014070

Introduction: ITER and DEMO heat load requirements

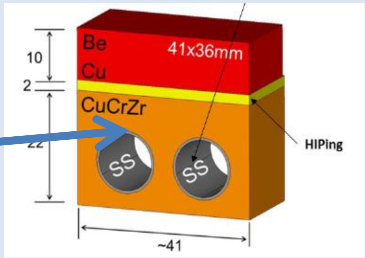
ITER:

- A large fraction of ITER's Cu-alloy first-wall can be designed for up to ~5 MW/m^2. (CuCrZr has extremely high K~300 W/mK but irradiation lifetime of only ~10 dpa)
- In case of heat load transients Be armour (low melting point) acts as a 'buffer' and off-sets temperature increase in structure by evaporation resulting in surface damage





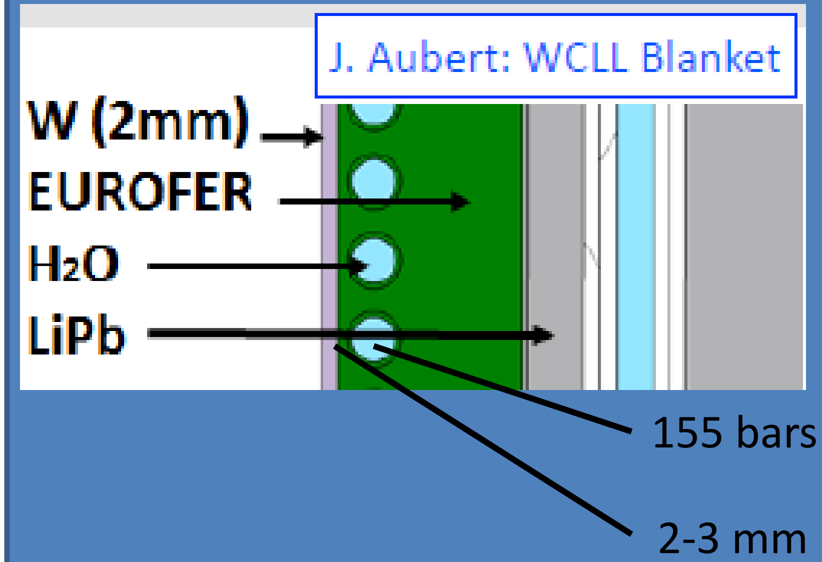
"enhanced" heat flux technology



"normal" heat flux technology

DEMO:

- Tritium self sufficiency
- W armour (high melting point) conducts heat to the heat sink overheating the cooling channels, evaporation only at very high T → poor resistance against heat load transients
- DEMO FW structural material: EUROFER (much lower thermal conductivity K~30 W/mK, but high irradiation lifetime) → Steady state heat loads limited to ~1 MW/m²

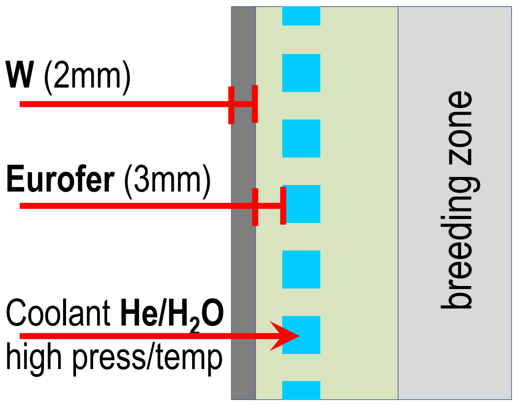


DEMO Breeding blanket wall load limits

DEMO breeding blanket requirements comparing to **ITER**:

- ❑ Tritium breeding self sufficiency
- ❑ Power conversion (High temperature → high efficiency)
- ❑ High neutron irradiation lifetime materials

First wall - breeding blanket



Difference in present design:

- ❑ Heat sink: **Eurofer** due to high neutron irradiation capability, (instead of **Cu**)
- ❑ Coolant: **H_2O** or **He** at high temperature for efficient power conversion.
- ❑ Armour material: **W** (instead of **Be**).

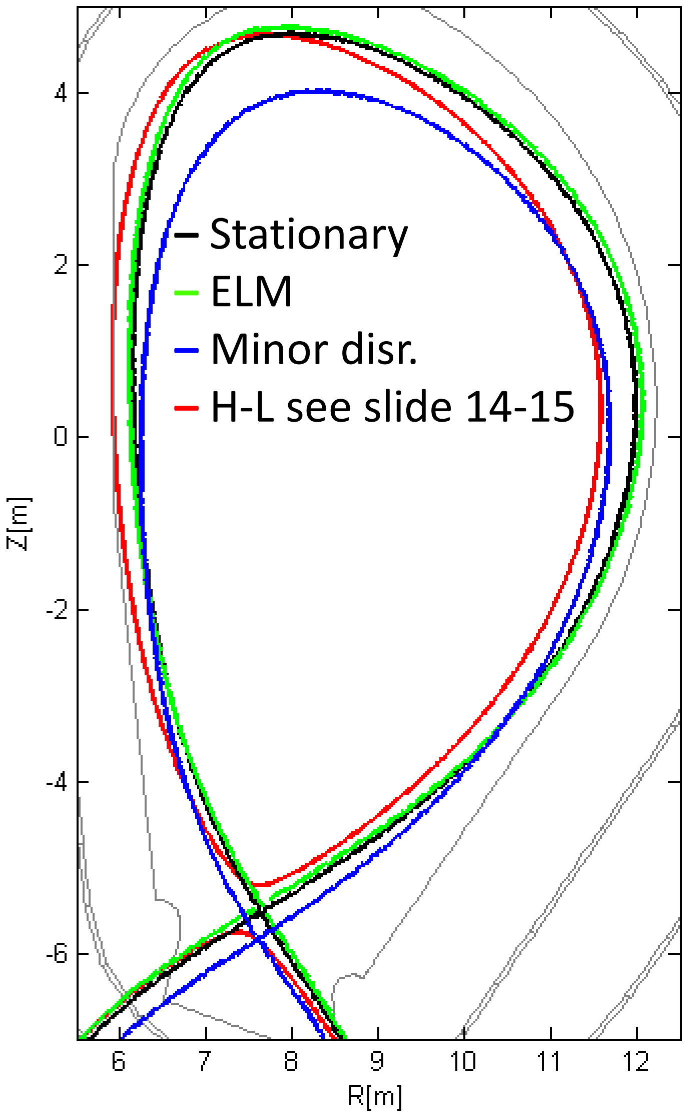
Static load limitations (from DEMO WPBB):

- ✴ **Water-cooled: ~1.5 MW/m^2**.
- ✴ **Helium-cooled: ~1.0 MW/m^2**.

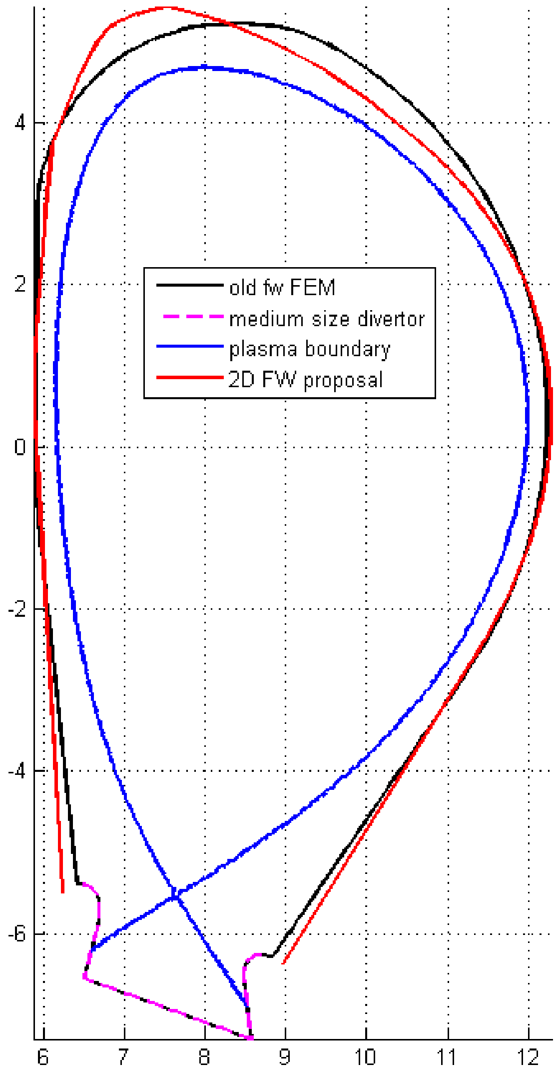
Present ITER limit up to 4.7MW/m^2: DEMO load spec. to be developed independently

Static loads: Conservative – P_{sep} slow transient

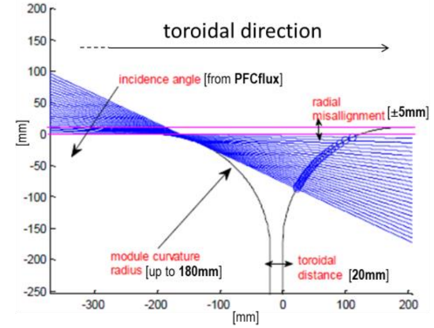
Plasma shape variations due to perturbations



CREATE NL dynamic sim.)



proposed FW as convex hull: verified in 3D (next slide)



E.g.: Automatic 2D FW proposal based on:

- -2D heat flux calculation
- -2D peaking factor
- -PSOL [**36-70**]MW/m^2
- -λ_q far SOL = [**5-15**]cm
- -target HF <= 0.5MW/m^2
- -3D approx. BB geometry
- -misalignments 0.5-2cm

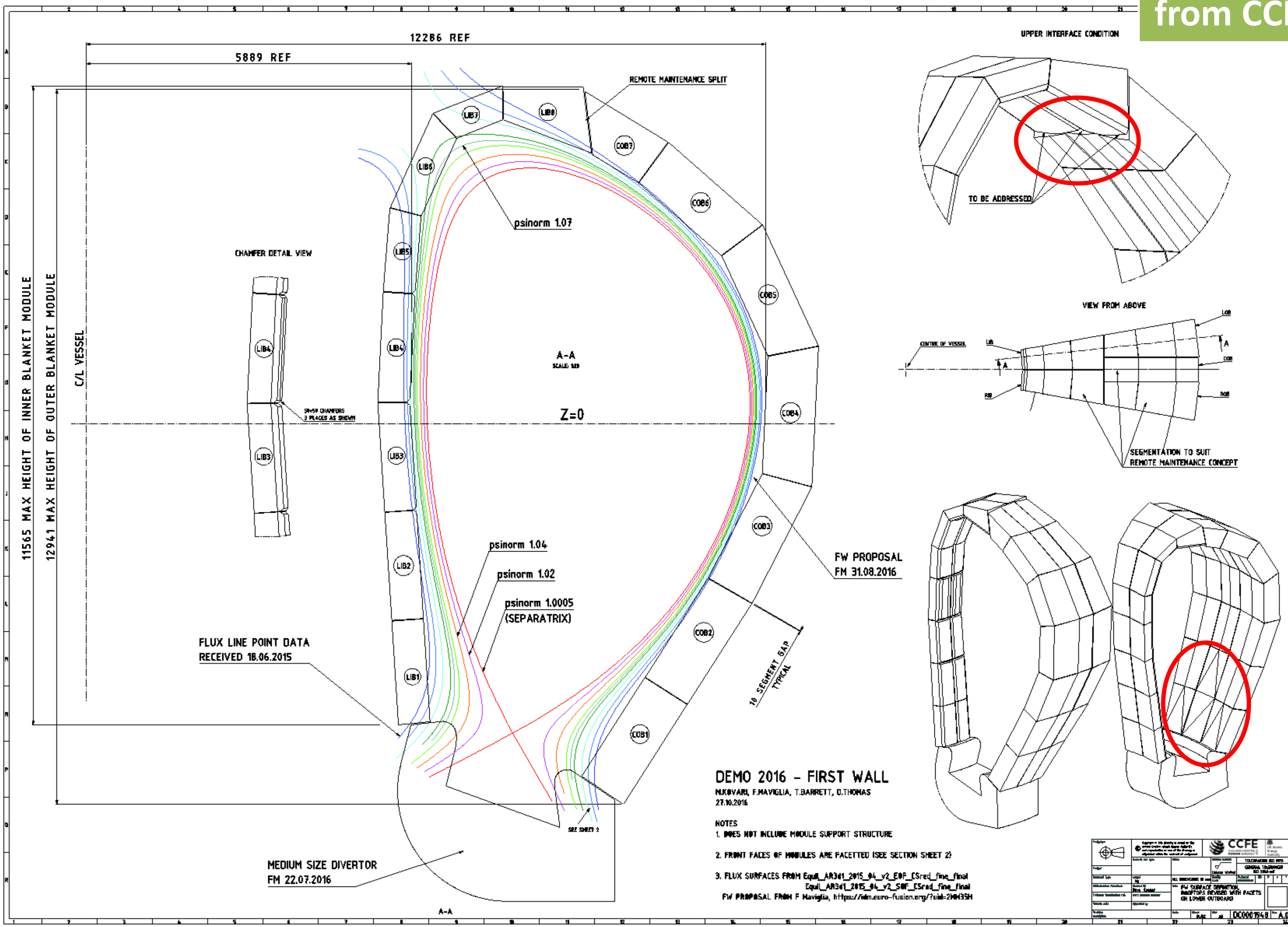
Perturbations used

- -ELM model 1 (control)
- -ELM model 2 (phys)
- -H-L controlled
- -Minor disruption

…

E.g.: 3D FW proposal

from CCFE



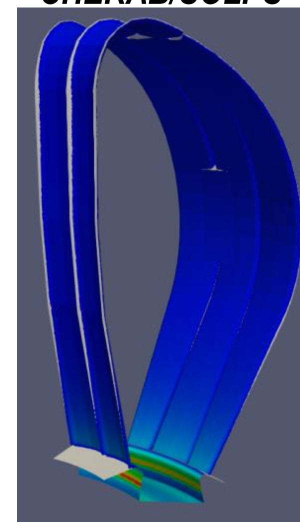
Static loads: Conservative – P_{sep} slow transient

Conservative, R. Wenninger, NF 2017

Radiation loads: **CHERAB** code using Core (**ASTRA**) + SOL (**SOLPS**) radiation

Charged particles: **PFCflux/SMARDDA** 3D field-line tracing

EOF: Carr/Subba
CHERAB/SOLPS



codes

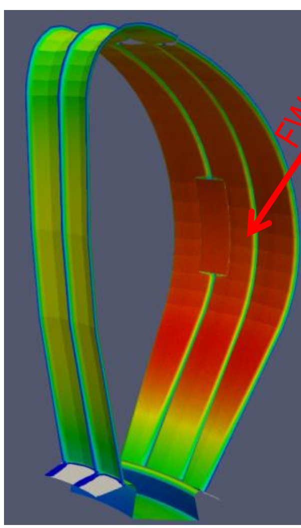
tot_rad_frac 70%		
P_alpha+P_aux 457 MW	P_sep,max 231 MW	P_rad,edge 40% 92 MW
		P_lambda_q=1mm 30% 69 MW
		P_lambda_q=50mm 30% 69 MW
	P_rad,core 226 MW	

FW & cluster in Div

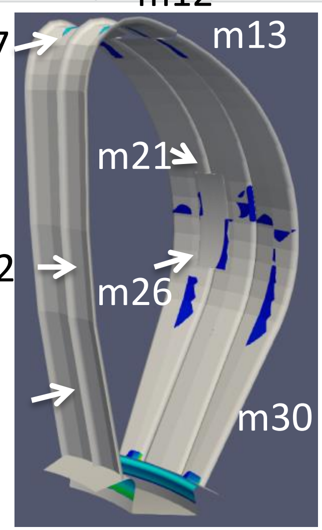
All Div (1mm OMP become ≈1cm at target)

95% Div, And 5% FW

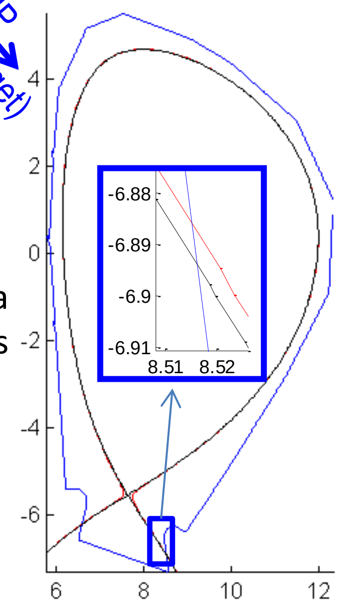
FW and Div



CHERAB/ASTRA



SOLPS: F. Subba
not shown in next slides



EOF: J.GERARDIN, M.FIRDAOUSS
PFCflux

3D fieldline tracing + radiation during steady state

EOF Inner FW	Charged particles MaxHF (MW/m^2)		Rad. Transfer Max HF (MW/m^2)		Tot HF Max HF (MW/m^2)	
	Left	Right	Left	Right	Left	Right
m1	0	0	0,23	0,23	0,23	0,23
m2	0	0	0,19	0,19	0,19	0,19
m3	0	0	0,19	0,19	0,19	0,19
m4	0	0	0,15	0,15	0,15	0,15
m5	0	0	0,15	0,15	0,15	0,15
m6	0	0	0,15	0,16	0,15	0,16
m7	0,26	0,27	0,17	0,17	0,43	0,43
m8	1,09	1,08	0,18	0,18	1,27	1,26
m9	0,96	0,96	0,20	0,19	1,14	1,14
m10	0,66	0,66	0,20	0,20	0,85	0,85
m11	0,48	0,48	0,20	0,20	0,67	0,67
m12	0,36	0,36	0,22	0,22	0,55	0,55

Limiter

	Charged particles MaxHF (MW/m2)	Rad. Transfer Max HF (MW/m^2)	Tot HF Max HF (MW/m^2)
Limiter	0,29	0,26	0,42

Divertor

	Charged particles MaxHF (MW/m2)	Rad. Transfer Max HF (MW/m^2)	Tot HF Max HF (MW/m^2)
Divertor	2,34	1,01	3,12

Outer FW	Charged particles MaxHF (MW/m2)			Rad. Transfer Max HF (MW/m^2)			Tot HF Max HF (MW/m^2)		
	Left	Center	Right	Left	Center	Right	Left	Center	Right
m13	0,11	0,01	0,43	0,23	0,23	0,23	0,33	0,23	0,63
m14	0,07	0,01	0,11	0,24	0,24	0,23	0,29	0,24	0,33
m15	0,02	0	0	0,24	0,24	0,24	0,25	0,24	0,24
m16	0	0	0	0,24	0,24	0,24	0,24	0,24	0,24
m17	0	0	0	0,25	0,25	0,24	0,25	0,25	0,24
m18	0	0	0	0,25	0,25	0,25	0,25	0,25	0,25
m19	0	0,00	0	0,25	0,25	0,25	0,25	0,25	0,25
m20	0	0,00	0	0,25	0,25	0,25	0,25	0,25	0,25
m21	0,01	0,01	0	0,25	0,25	0,25	0,25	0,25	0,25
m22	0,01	0,05	0	0,25	0,25	0,25	0,26	0,28	0,25
m23	0,05	0,06	0,05	0,26	0,26	0,25	0,28	0,28	0,28
m24	0,02	0,01	0,04	0,25	0,25	0,25	0,27	0,26	0,27
m25	0	0,01	0,02	0,25	0,26	0,26	0,25	0,26	0,27
m26	0	0,01	0,03	0,26	0,25	0,26	0,26	0,26	0,28
m27	0,04	0,02	0,04	0,26	0,26	0,26	0,29	0,27	0,29
m28	0,07	0,03	0,07	0,25	0,26	0,26	0,32	0,28	0,32
m29	0,13	0	0,12	0,25	0,25	0,25	0,38	0,25	0,38
m30	4,09	0,06	2,81	0,31	0,32	0,31	4,27	0,32	3,04

m30: Outer baffle area being corrected by more recessed BB

m8-m9:Upper-inner area 1/3-1/2 lower in nominal case ($P_{\lambda q5cm}$≈15-20MW): P_{sep} = 230MW not compatible with divertor limits for SS.

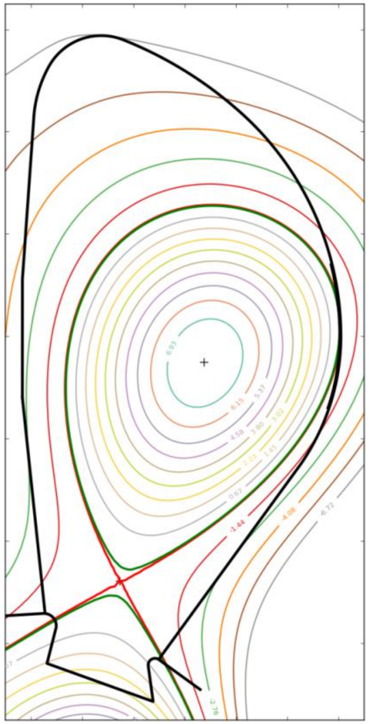
Misalignments penalty factor will increase the values, but shadowing may be possible in limiter are used

CEA: J. Gerardin. M. Firdaouss, CCFE: Carr

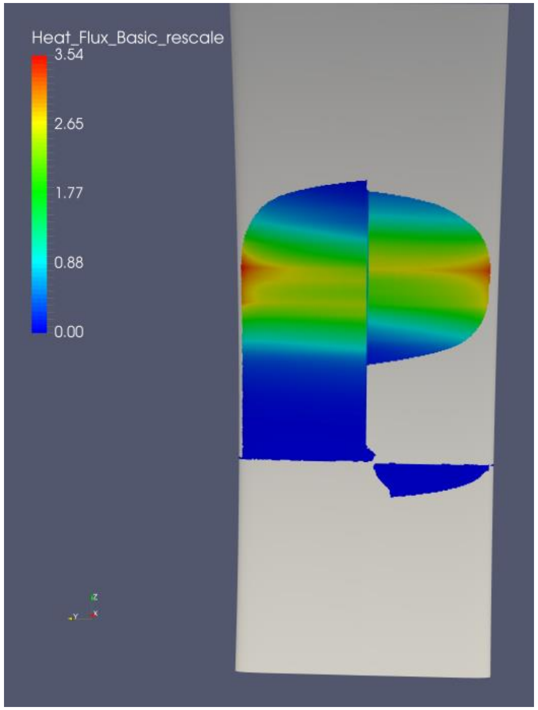
Transient analysis: Ramp-up limited phase

- Plasma ramp-up assumed from +0.1 MA/s up to + 0.2 MA/s.
- λ_q = 6mm, Psol[MW] = Ip[MA]
- X-point to be formed at 3.5MA to 6MA (based on ITER): t_{RU}= 20s to 60s

Limited eq. 6MA, 4 limiters



P_{SOL} = 6MW λ_q = 6mm



Max HF = 3.5MW/m² (ITER rescale)

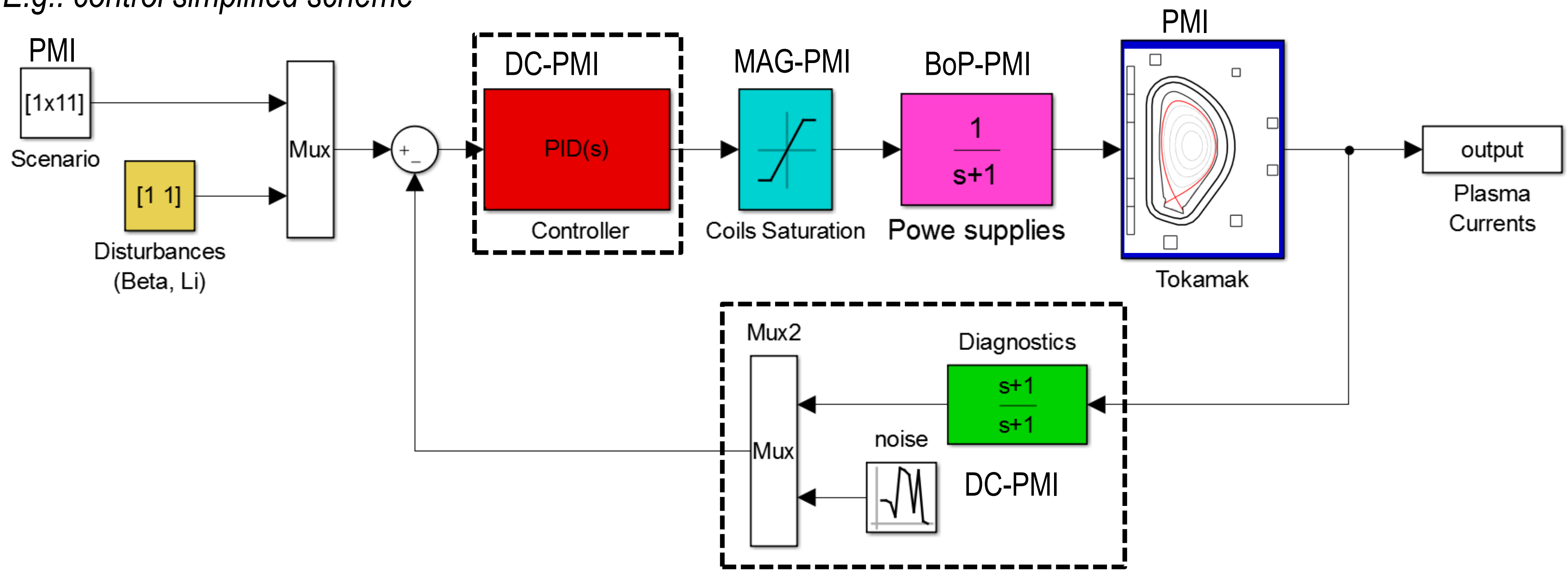
Misalignments may be reduced if limiter adjustable at OMP port. Bare wall HF ≈3-4MW/m²: variant 1 not compelling
No relevant HF found on other BB modules, nor on the limiter during flat-top phases

1: KDI1 disruption simulations: HF and REs

Definition of disruption cases, and relative inputs, *e.g.*:

- perturbation time evolution B_{pol}, L_i, I_{pla}
- TQ,CQ, times evolutions, Runaways Electrons (REs) energy fraction, Vapor shielding, etc.
- Control perturbations
- Electromagnetic simulations
- 2D heat flux (HF) calculation of radiated and charged particle
- Realistic controller-diagnostics from end 2018-2019

E.g.: control simplified scheme



Transient analysis for plasma-wall contact phases

Transient analysis for plasma-wall contact phases:

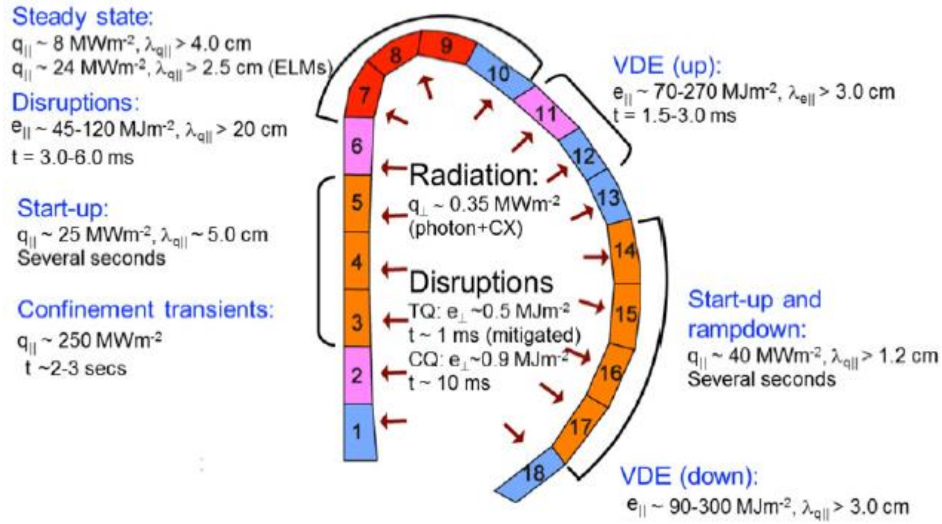
- Disruptive H-L transition
- VDE
- Ramp-up/down limiter phases

Preliminary disruptive events table develop.:

- Time duration estimated ranges
- Energy content
- Geometric position of Plasma-wall interaction

Will be used to evaluate the <u>technological solutions</u>, and to give the requirements for the HHF component designer: *e.g.*

- surface shape,
- components misalignment,
- number of toroidal modules,
- position (may be modified with plasma conf.)



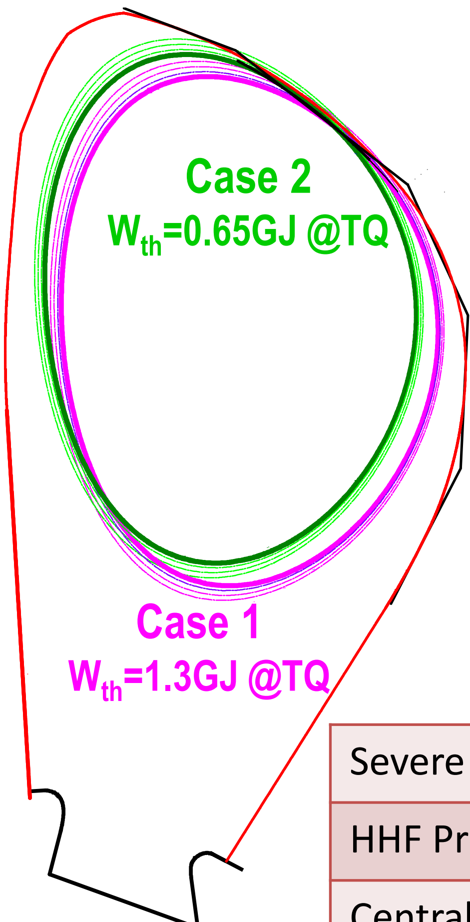
Aim to obtain ITER like load spec. and map

Transient analysis: thermal quench during a VDE

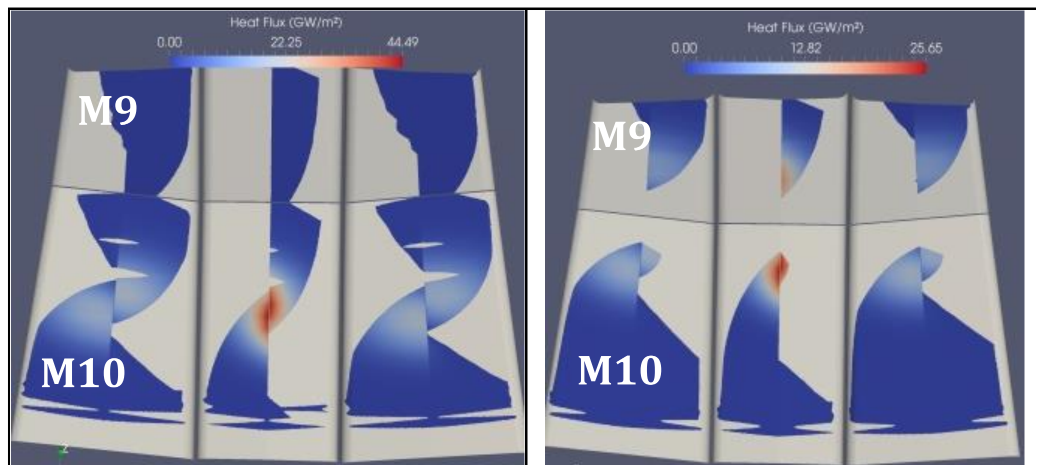
Upward Vertical Displacement Event (VDE) modelled as [R. Wenninger, EPS 2017]:

- Plasma moves upward, then becomes limited until q_a=2 when TQ is triggered
- Disruption SOL broadening: x7 from TQ onset (λ_q=7mm)

Plasma thermal energy content deposited in 4ms: 1) W_{th}=1.3GJ (Full), 2) W_{th}=0.65GJ (half)



Results	module	Max HF (GW/m²)
Case 1)	M10	44.5
Case 2)	M9	19.2
	M10	25.6

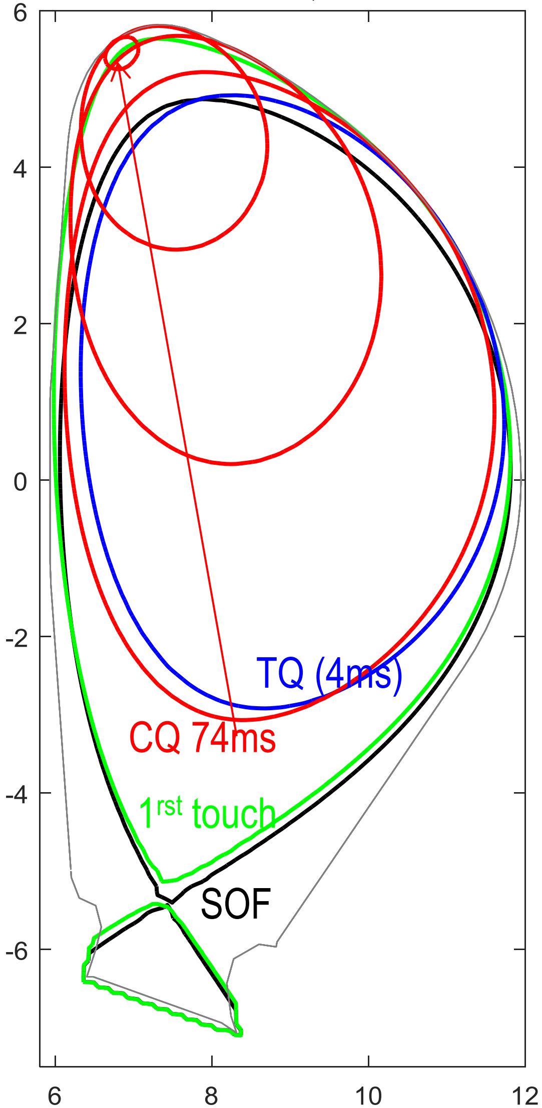


Results on standard FW, optimized for steady state λ_q=50mm.

Severe damages expected at tens of GW/m² on BB armour and cooling pipes.
HHF Protection panels (sacrificial?) concept being developed.
Central disruptions affects divertor. Downward VDE TQ, CQ, and RE being analyzed.

Unmitigated disruption simulations

E.g.

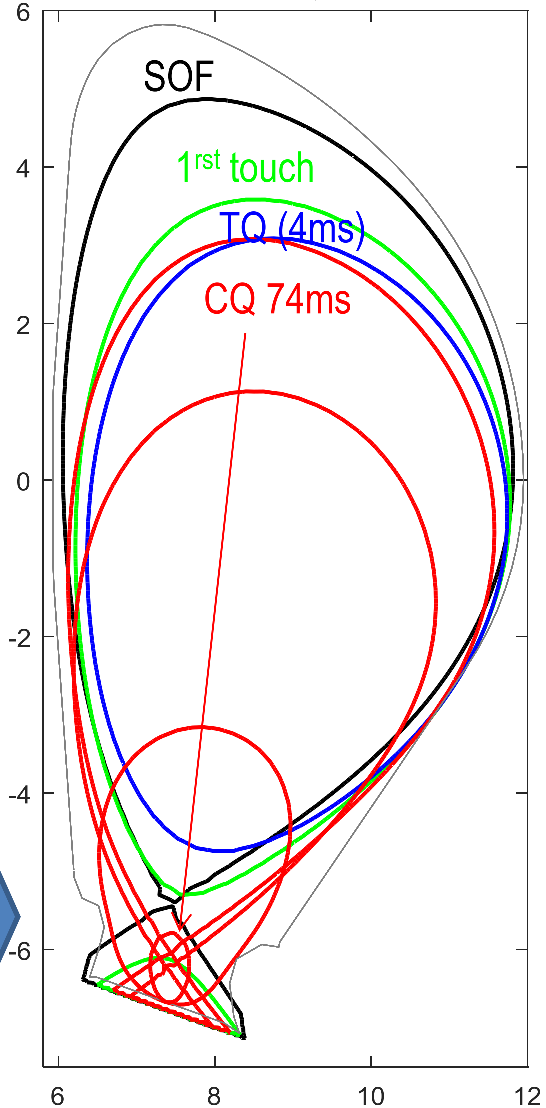


Typical plasma VDE evolution:

1) SOF (Start Of Flatop)
2) 1rst touch (plasma moves vertically)
3) TQ (W_{th} from 1.3GJ to 0, in 4ms)
4) CQ (I_p from 19MA to 0, in 74ms)

- 1rst touch close or at 11 'O clock
- CQ ends up at 11 'O clock

- 1rst touch right below OMP at 4 'O clock, Far from baffle
- TQ shrinks plasma which become diverted again (we cant rely on it!)

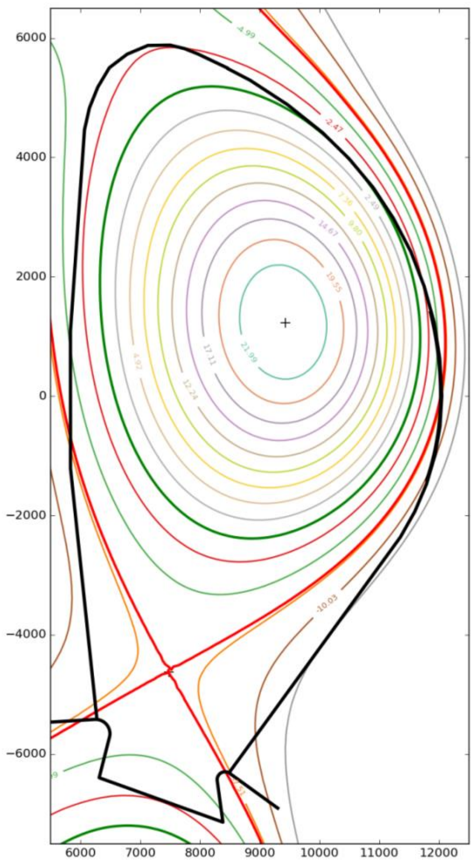


Unmitigated disruption simulations:TQ

Upward Vertical Displacement Event (VDE) modelled as [R. Wenninger, EPS 2017]:

- Plasma moves upward, then becomes limited until q_a=2 when TQ is triggered
- Disruption SOL broadening: x7 from TQ onset (λ_q=7mm)

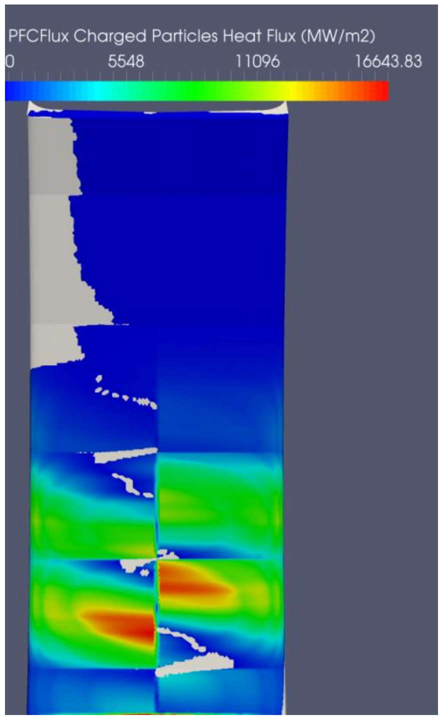
Plasma thermal energy content deposited in 4ms: 1) W_{th}=1.3GJ (Full)



P_{SOL} = 325GW

λ_q = 7mm

HF=**16GW**/m² for 4ms



	Max Heat Flux (MW/m²)			Int. Power (MW)		
	LOB	COB	ROB	LOB	COB	ROB
m13	0.000	0.002	0.001	0.000	0.000	0.000
m13-14	0.003	0.002	0.000	0.000	0.000	0.000
m14	0.453	0.289	0.000	0.092	0.075	0.000
m14-15	0.594	0.412	0.000	0.006	0.003	0.000
m15	10.279	6.990	0.000	3.395	0.783	0.000
m15-16	9.154	0.000	0.000	0.009	0.000	0.000
m16	27.108	21.953	17.431	1.792	2.087	3.434
m16-17	0.000	10.696	8.730	0.000	0.078	0.054
m17	10.288	14.385	10.288	4.046	7.132	6.025
m17-18	1.678	1.678	2.177	0.028	0.028	0.034
m18	1.171	1.178	1.178	0.265	0.614	0.614
m18-19	0.047	0.054	0.047	0.001	0.001	0.001
m19	0.022	0.022	0.033	0.008	0.008	0.009
Total				9.642	10.809	10.171

Eq. Lim	~0	~0
Upp. Lim	16643.0	112933

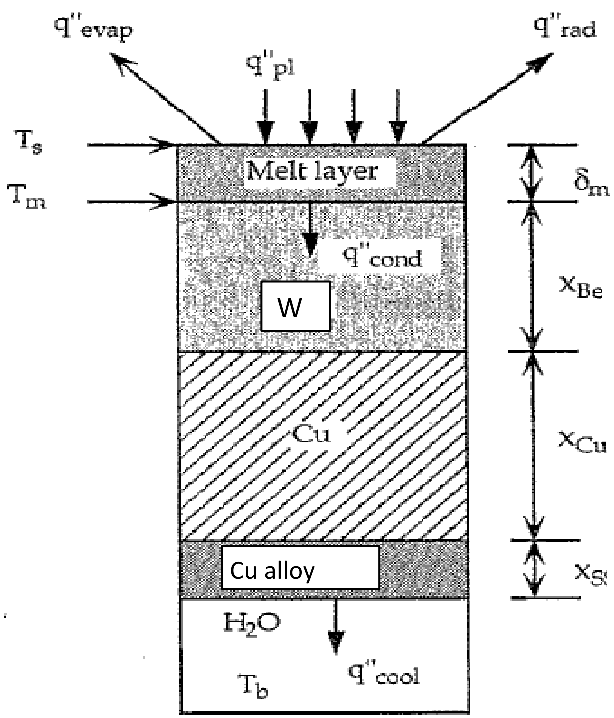
Total	112.964 GW
Ratio	0.348

Survival of the pipe may be possible, see RACLETTE and CB presentation

RACLETTE slow transient analysis

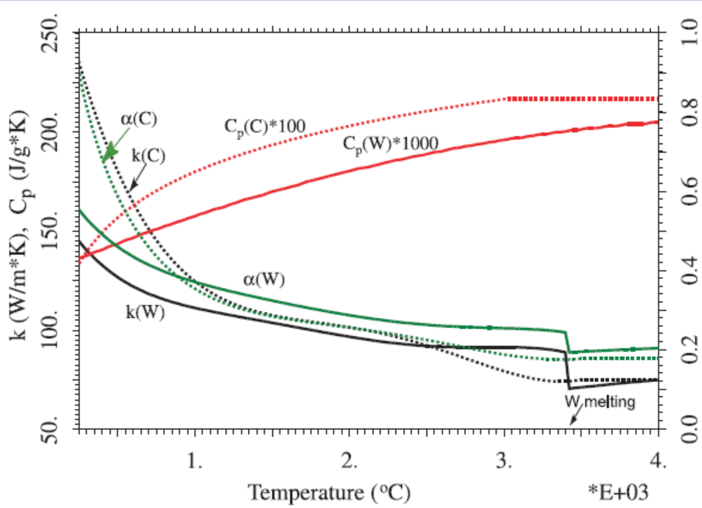
Analysis performed with code RACLETTE [1]. Fast thermo-hydraulic assessment, for broad parametric scans. It includes:

- 1D geometry with 2D corrections.
- All the key surface processes such as evaporation, melting and radiation.



- The surface interaction with bulk *PFC thermal response* and the *coolant*.

Model calibrated/validated with analytical, FEM multidimensional and experiments



Thermal properties (thermal conductivity, k, specific heat C_p, and thermal diffusivity, $k/\rho c_p$) of W and CFC as a function of the temperature used in the model.

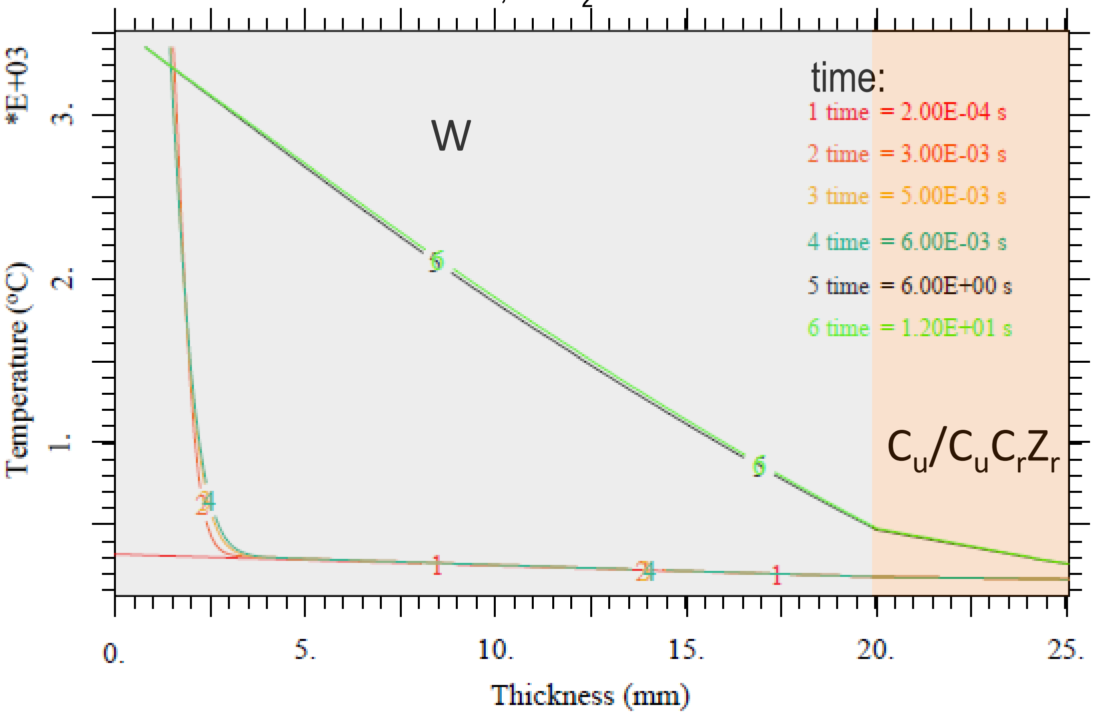
[1] A. Raffray, G. Federici, Journal of Nucl. Materials (1997).

RACLETTE: Thermo-hydraulic simulation

RACLETTE [1] simulation of Limiter (W-divertor like with 2cm armour) inputs:

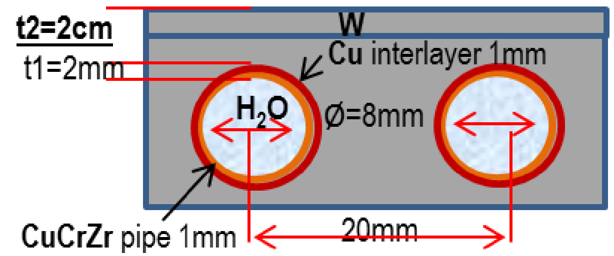
- Power density 0.2 to 20GW/m^2, deposition time 5 ms

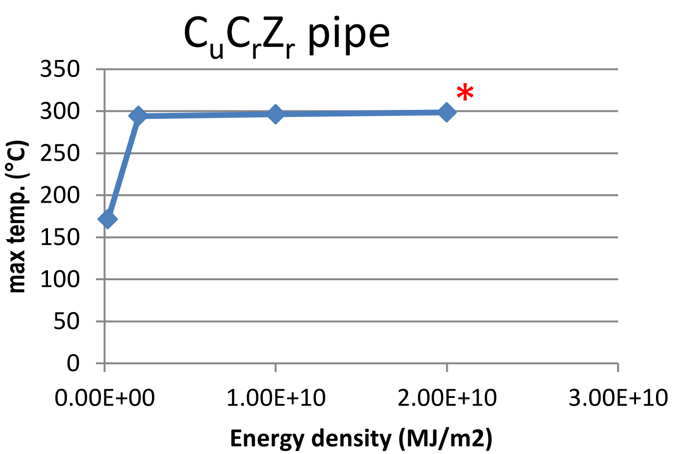
* RACLETTE 20GW/m^2 for 1ms, on H_2O FW:



H_2O coolant, CuCrZr heat-sink

Coolant parameters:
Vel = 8m/s
Pres = 5 Mpa, T_coolant = 150°C





- Temperature gradient between W-melting front and pipe ≈fixed if melting layer << W armour thickness
- **Slower transient:** CuCrZr below temp. limit (350°C) with **armor ≥ 20mm**. Mitigation expected by <u>vapor shielding</u>
- In steady state calculated HF ≈0.5 to 1 MW/m2 (mainly radiative): temperature at W-surf 800-1200°C

Wall protection concept

Wall protection concept is based on extruding limiters preventing the plasma contacting the BB FW

Rationale:

- BB FW will fail in case of heat loads causing melting of its armour (because it is made of Eurofer)
- Replacement of BB is time consuming, BB is also expensive

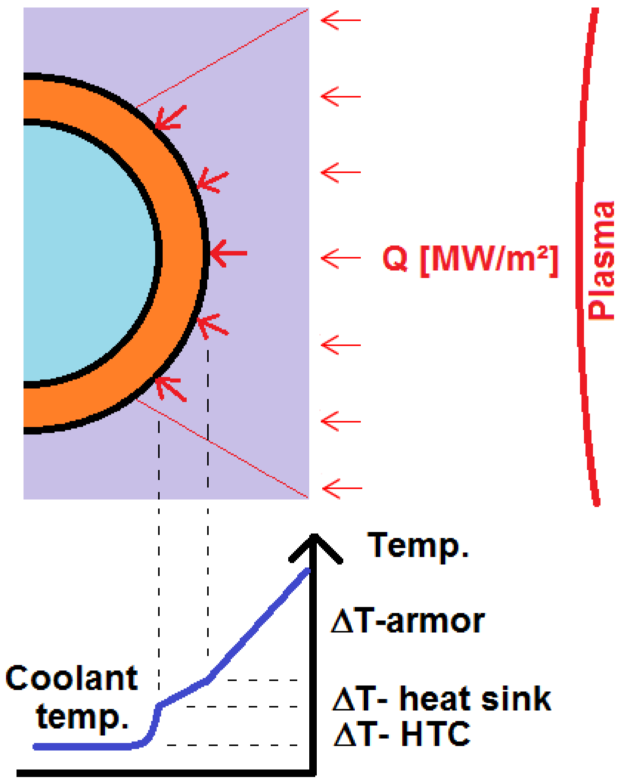
Discrete limiters:

Heat loads: ~0.5 MW/m² (steady state), ~0.1-10 GW/m² for 1.5-4 ms

- Better alignment options to toroidal field
- Separate, non-BB PHTS
- Leaks of limiters are less severe incidents than leaks of BB, and:
- We believe that divertor target-like PFCs could *prevent* the heat sink structure to fail during plasma-wall contact. This requires thermal insulation of heat sink structure → e.g. thick W armour:

$$t = \frac{\left(T_{W,melt} - T_{CuCrZr,limit}\right) \cdot \lambda}{Q} = \frac{(3422^{\circ}C - 350^{\circ}C) \cdot 140\ {}^{W}/_{mK}}{20\ {}^{MW}/_{m^2}} = 22mm$$

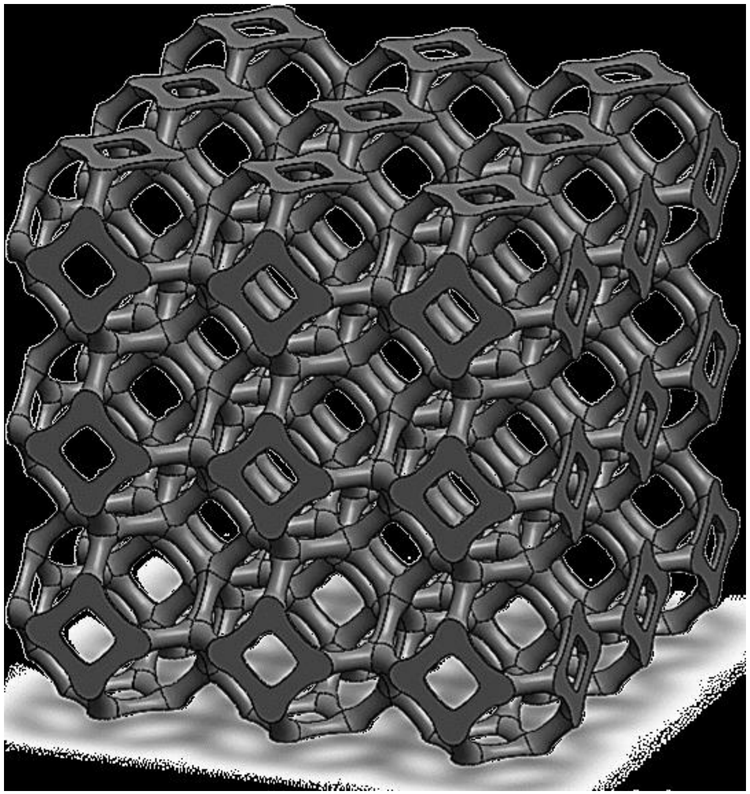
Damage of armour remains an issue of DEMO availability!

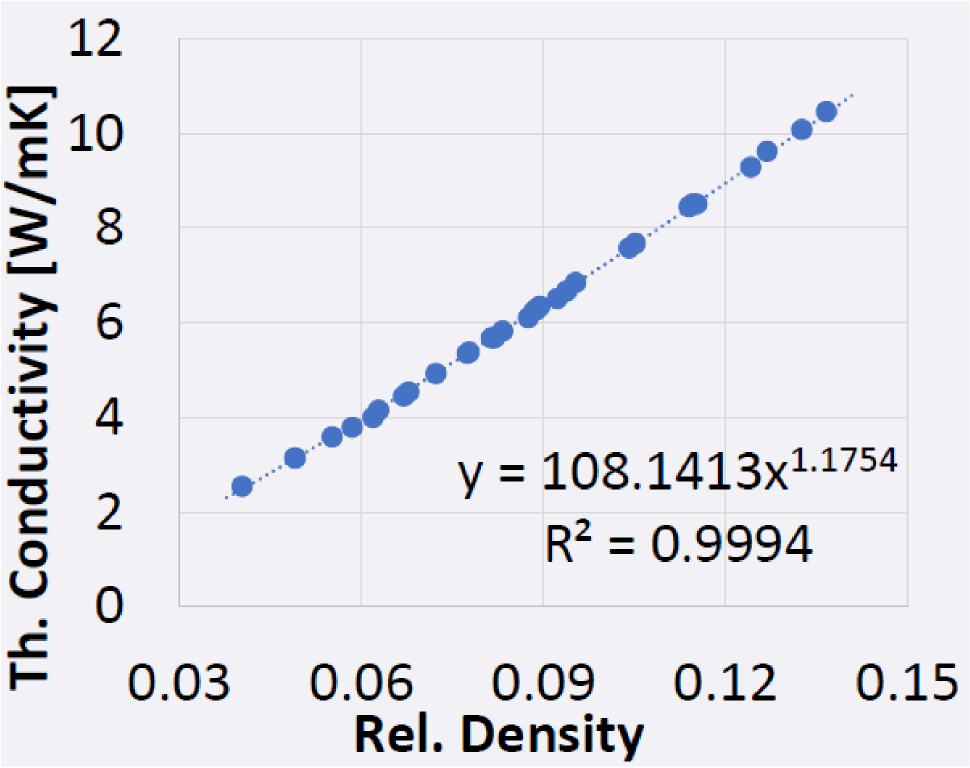


Limiter Armour R&D

R&D program required to develop:

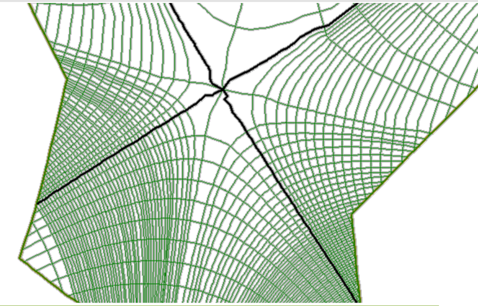
a) Armour providing thermal insulation, e.g. tungsten foam
b) Armour not requiring replacement after plasma-wall contact





R. De Luca et al, SOFT 2018
P. Fanelli, final meeting WPPMI 2018

Simulations including vapor shielding

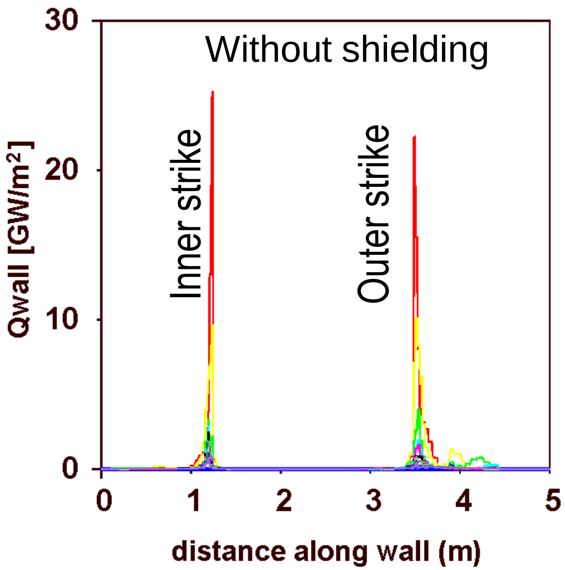


From S. Peschanyy, KIT

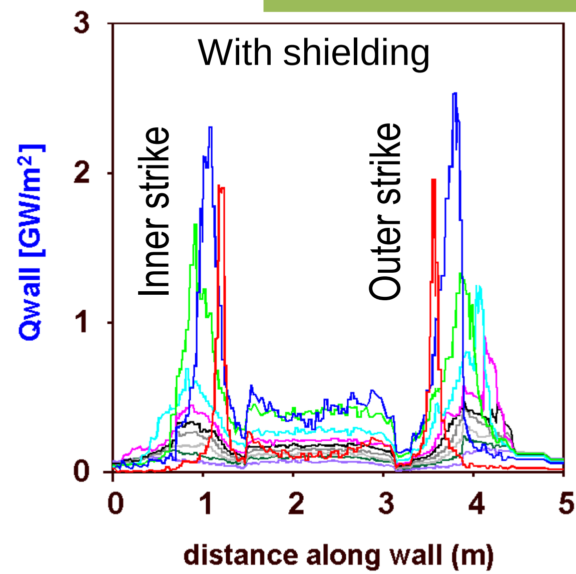
Preliminary simulations including vapor shielding have been performed on DEMO using TOKES code on:

Central Disruption:

- Thermal quench duration 4ms
- Charged particles energy = 0.65GJ (0.5 of total thermal energy)



Colors represent different instants from 0 to 10ms



With vap. sh. Factor 10 reduction in Qwall (from 25 GW/m^2 to 2.5 GW/m^2).

Max vaporization erosion is reduced from 700 μm to 1 μm.

Preliminary results. In line with ITER modelling [1] and exp. Validation [2]

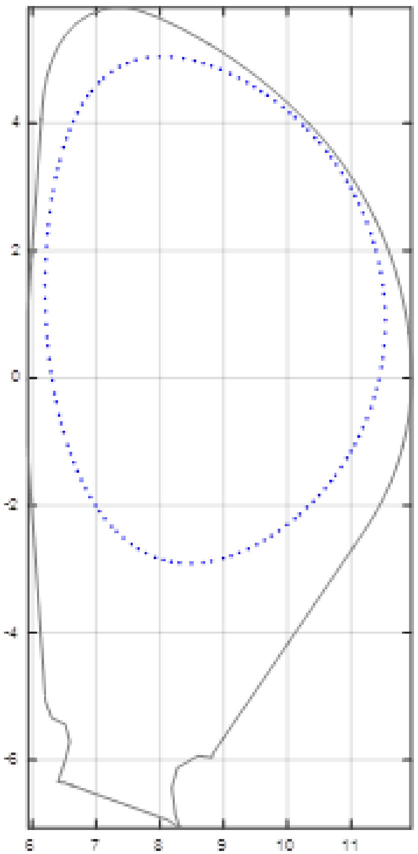
[1] S.Pestchanyi, et al., FED, vol. 109, p. 141, 2016

[2] S.Pestchanyi, et al., FED, vol. 124, p. 401, 2017

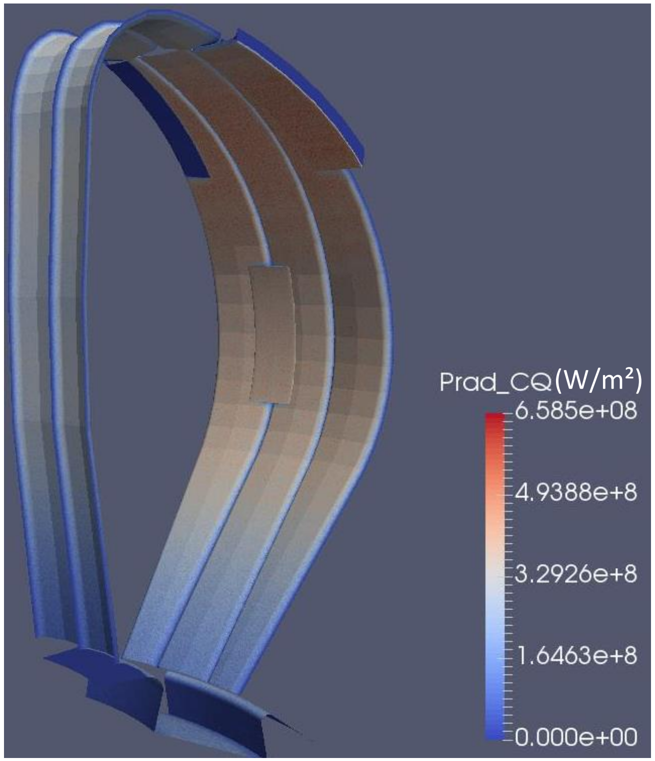
Mitigated disruption simulations:TQ

Preliminary results: Mitigated U-VDE as R. Wenninger, EPS 2017, help from T. Hender:

- Initial thermal energy W_{th}=1.3GJ: 20% radiated at pre-TQ at MGI/SPI: remaining ≈ 1GJ
- At TQ normally 80% is radiated in 1ms (controllable) -> P_{rad}≈800GW

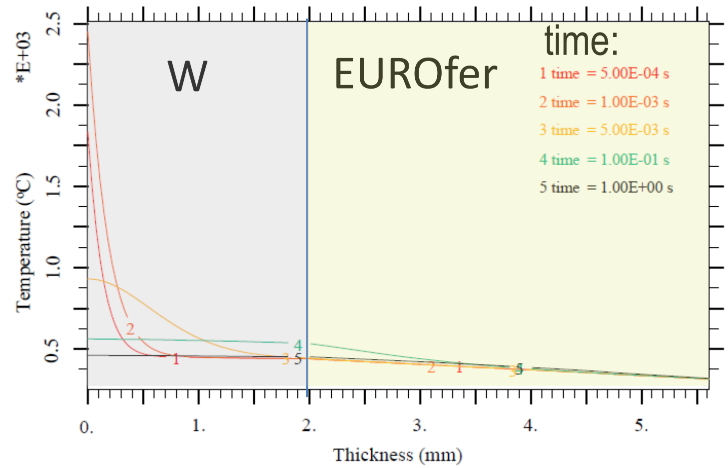


≈300 rad. sources used at boundary P_{rad} = 500GW



Max HF ≈660MW/m2, if TPF=2.8 is applyed -> ≈1.8GW/m^2

RACLETTE 1GW/m^2 for 1ms, on H_2O FW:



100% radiation in 1ms may be above FW W-limit

TQ radiation time may be slowed down with MGI/SPI

Mitigation techniques to consider FW damages (limiters ineffective)

Cooling pipe not damaged

Wall protection concept – inboard and upper null area

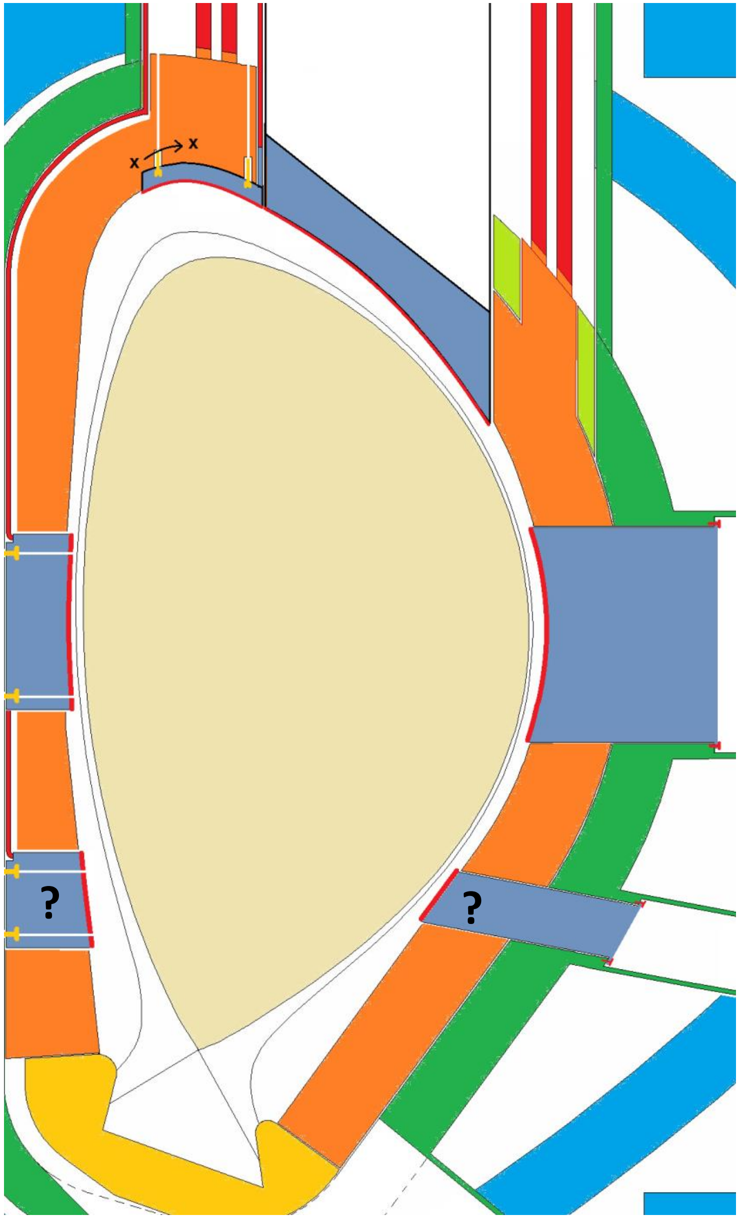
Protection concept for upper null area:

- In upward VDEs plasma moves towards 2nd null:
  - Move 2nd null clockwise, or
  - Reduce upper triangularity, i.e. shift 2nd null towards outboard
- 4 limiter components at new location of 2nd null
- Limiter interfaces can be accessed from the upper port
- Limiter is removed from the front

Protection concept for inboard:

- Use of e.g. 4 inboard segments as limiters abandoned because Cu-alloy assumed requiring scheduled maintenance
- 4 limiters at equatorial level + 4 limiters at lower level with front side access to mechanical supports and coolant pipes, directly attached to VV
- RH through 4 equatorial limiter ports
- Inboard BB remains installed and connected up to 50 dpa (unless BB failure occurs)

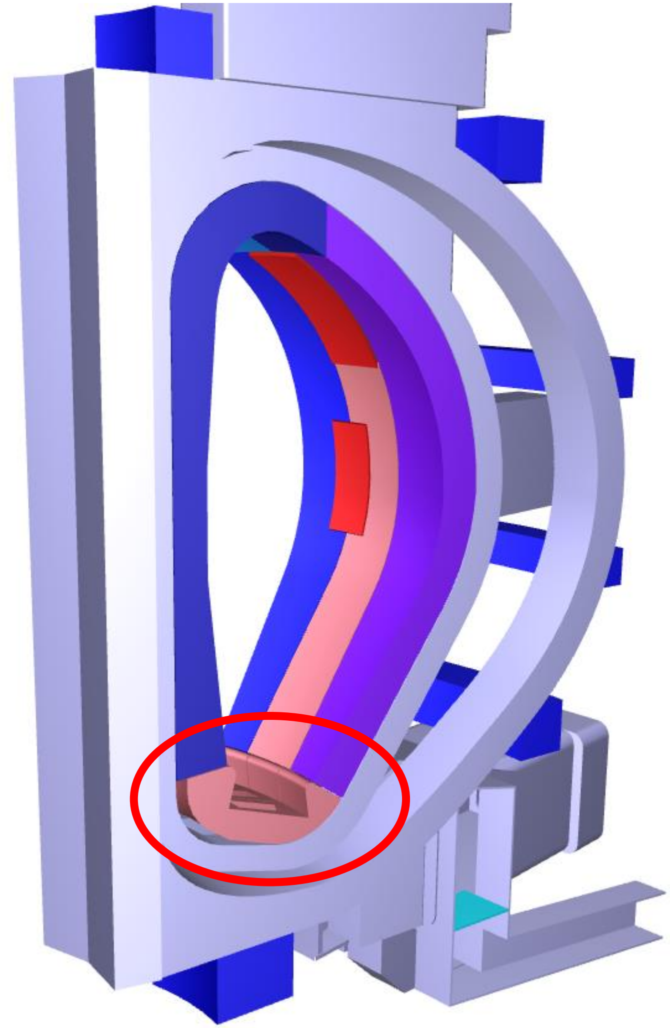
Alternative concept: Inboard segment with Eurofer-based PFCs with new thermally insulating armour with high lifetime.



❑ Plasma scenario studies: e.g. upper null moved outwards, magnetostatic final CQ point, plasma-FW distance, Inverted triangularity.

Outline

- Introduction: ITER and DEMO PFC requirements
- Materials R&D for DEMO – the baseline options
  - Plasma Facing Materials (Armor)
  - Copper-alloys (Heat sink materials)
  - Ferritic-martensitic steels (Structural Materi
- DEMO heat load requirements
  - First Wall (FW) and Limiters
  - Divertor
- Conclusions



Divertor power exhaust in ITER and DEMO

- Divertor power load is a key DEMO design constraint.

- **ITER targets heat flux design criteria:**
  - ✓ **10MW/m^2 steady state (order ~10^4 cycles).**
  - ✓ **20MW/m^2 transients for ~10s & ~100 cycles.**
  - ✓ **Coolant pipe burn out ~35MW/m^2 (factor 1.7 from transient).**
- DEMO heat flux removal capability margin reduced due higher coolant temperature to avoid Cu embrittlement at high irradiation[1-2] (TBV).

Presently studied regimes to lower divertor heat flux load:

- Techniques to radiate the majority of the loss power.
- Plasma detachment.

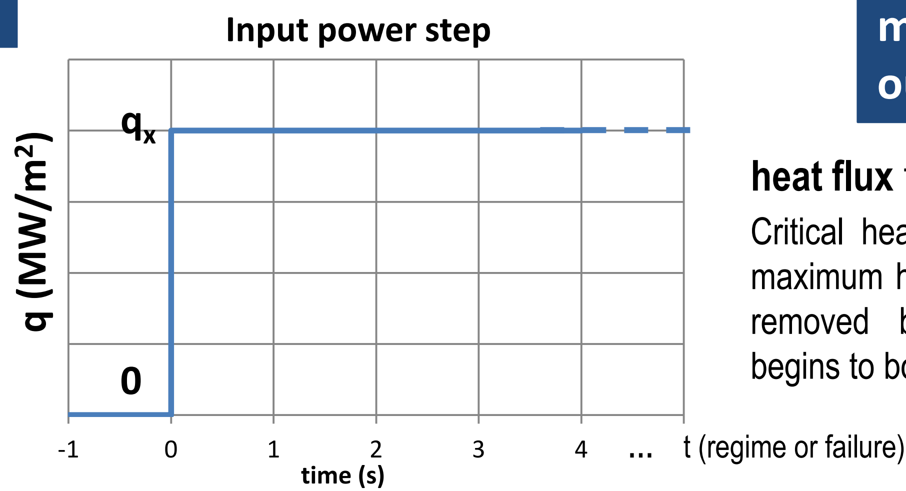
Failure of the above controls may lead to sudden increase of heat flux: Transient loads <u>critical</u> for DEMO due to reduced margin to pipe burn out.

[1] S.A. Fabritsiev, et al., Journal of Nuclear Materials (1996)
[2] S.A. Fabritsiev, et al., Plasma Devices and Operat., (1997)

Transient power load scan

- Sensitivity analysis to power steps

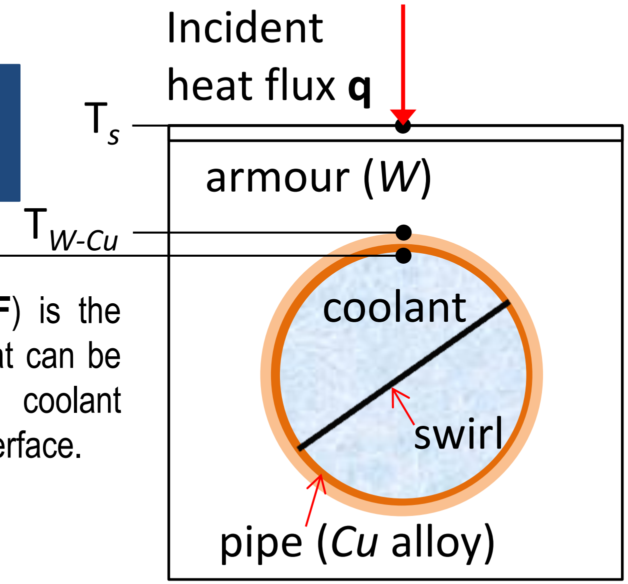
input:



model & outputs:

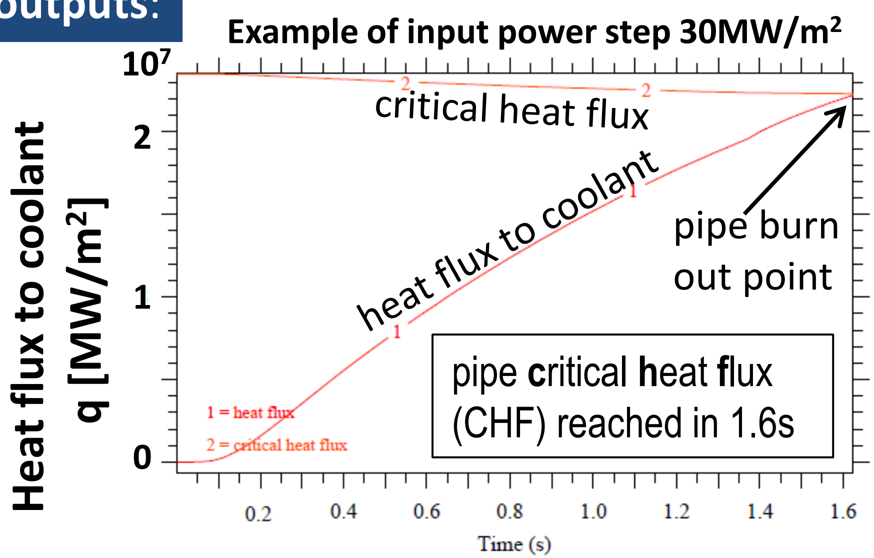
heat flux to coolant

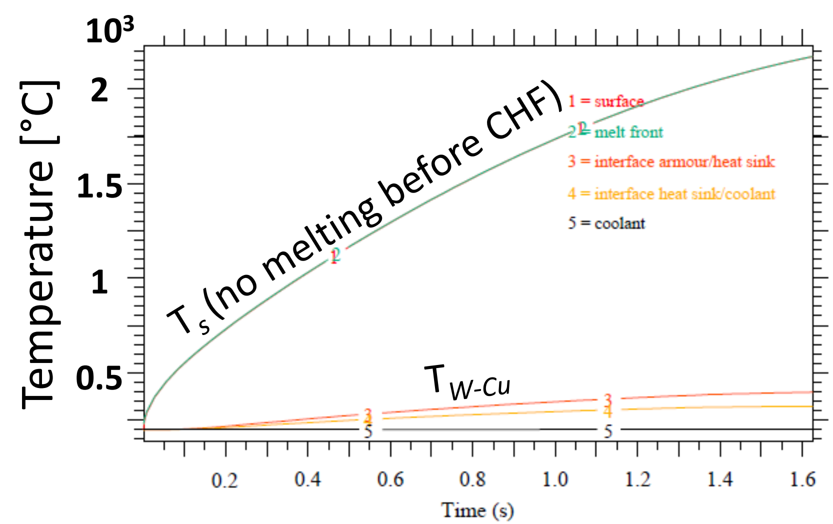
Critical heat flux (**CHF**) is the maximum heat flux that can be removed before the coolant begins to boil at Cu interface.



outputs:

Example of input power step 30MW/m²

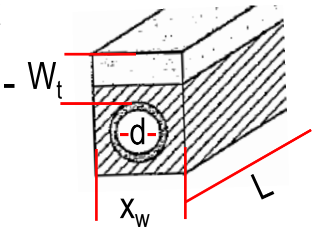




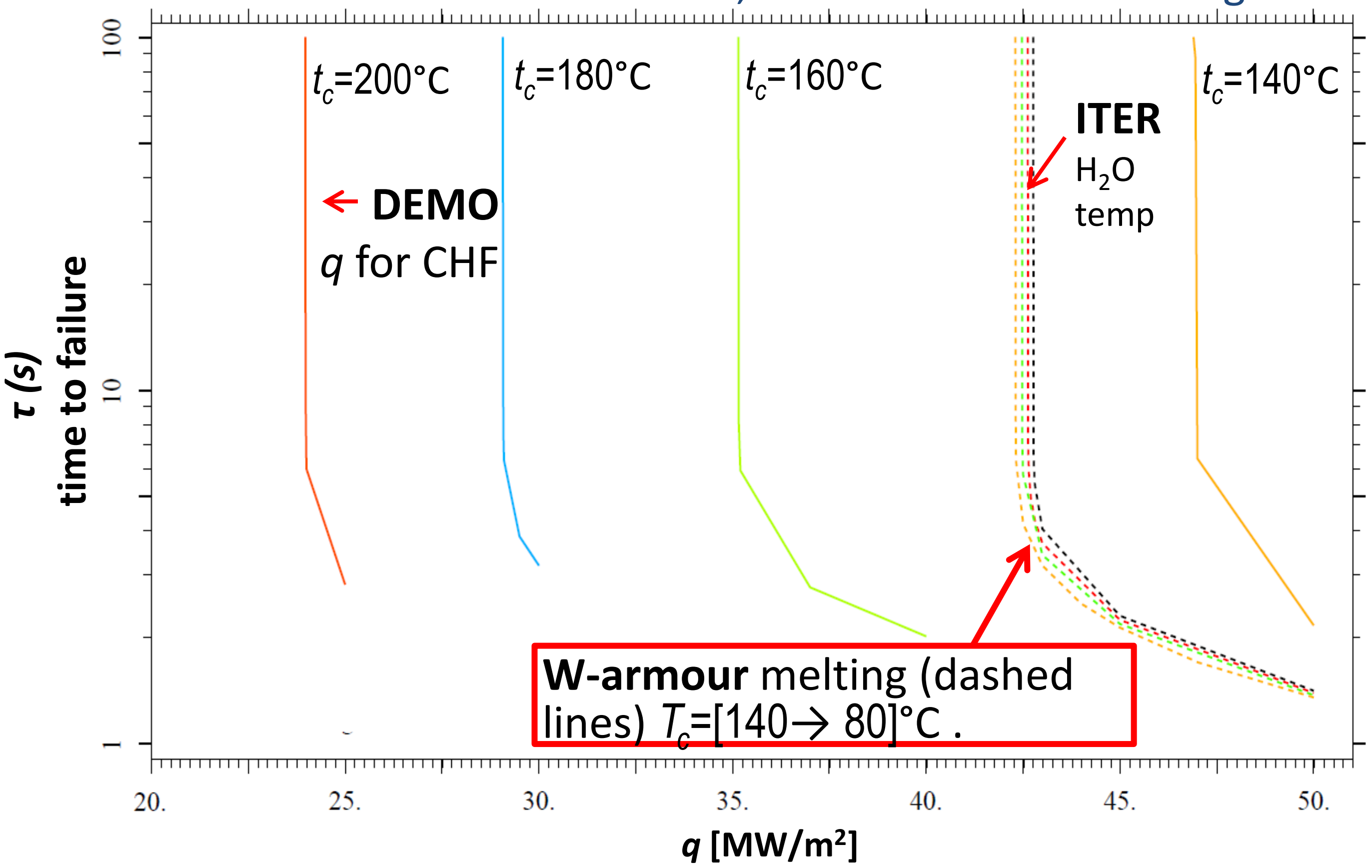
Transient power load scan

- Coolant temperature @monoblock range scan t_c= [80-200] °C

Main parameters: armour thickness 5mm (W_t), coolant **pressure 4MPa**, W mono-block width 28mm (X_w), water velocity 12m/s, pipe diameter (d)/length(L).



— Solid lines reach CHF, ···· dashed reach melting



- Melting is not the driving criterion for DEMO.
- CHF is in DEMO the limiting factor.

T H_2O [°C]	CHF [MW/m²]	Melting [MW/m²]
100	>50	43
120	>50	43
140	46	43
160	35	43
180	29	43
200	24	43

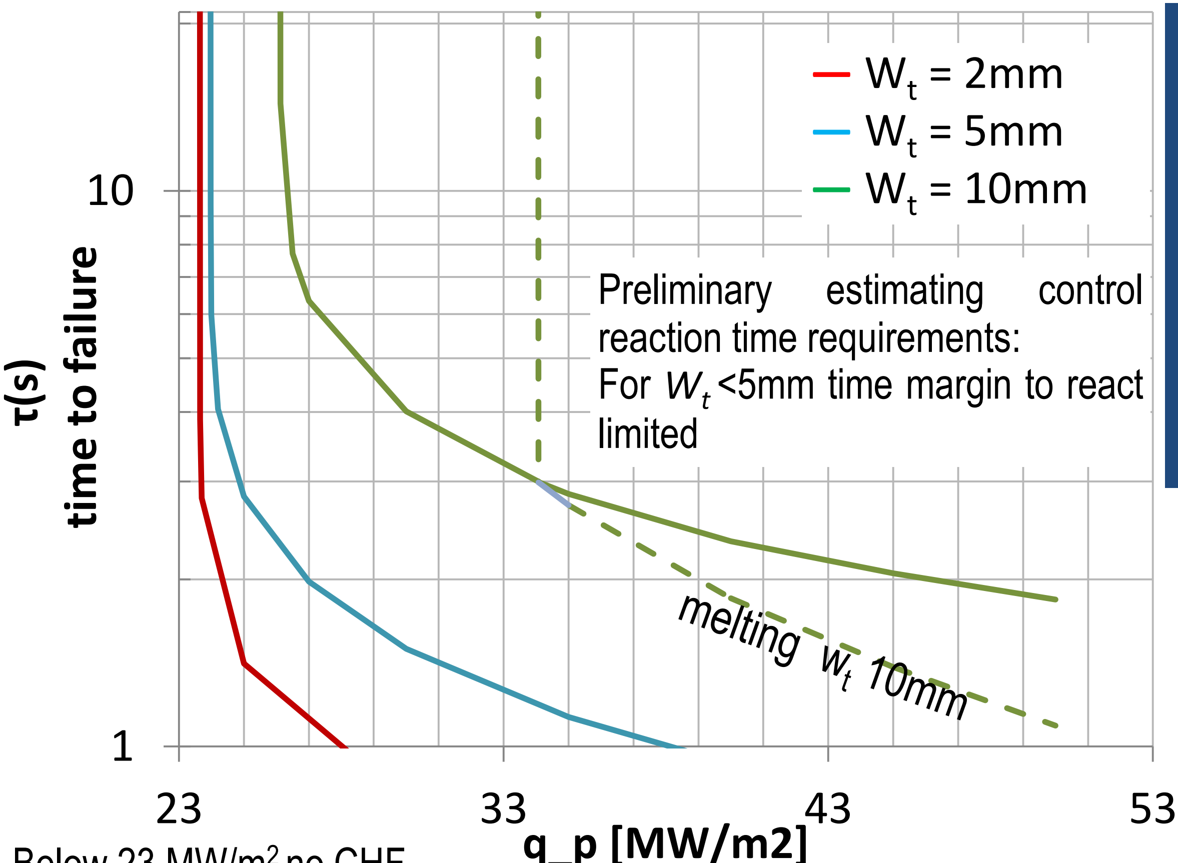
CHF margin substantially reduced in DEMO

Transient power load scan

- **DEMO** armour thickness range scan W_t = [2;5,10] mm

Main parameters: coolant temp. 200°C, pressure 4MPa, pitch 28mm, water vel.12m/s.

— solid CHF, - - dashed melting



Below 23 MW/m² no CHF

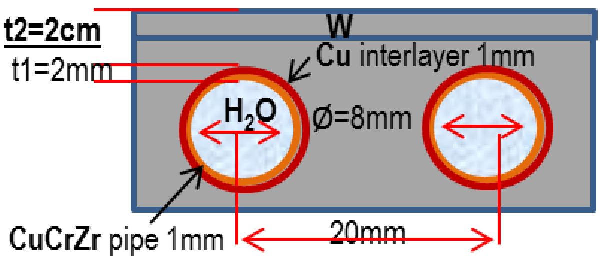
- The larger ***W_t*** the slightly higher the CHF.
- Larger ***W_t*** could allow longer time to CHF up to certain **q** values (plasma shut down strategies, mitigation?)
- Larger W_t *for erosion.*
- Drawbacks: higher W-$t_{surface}$

W_t [mm]	CHF [MW/m²]	5s to CHF [MW/m²]	4s to CHF [MW/m²]	3s to CHF [MW/m²]
2	22.8	22.8	22.8	22.9
5	24	24	24.2	25
10	26.2	28.4	30	**34**

HF transient map: Energy - Deposition time

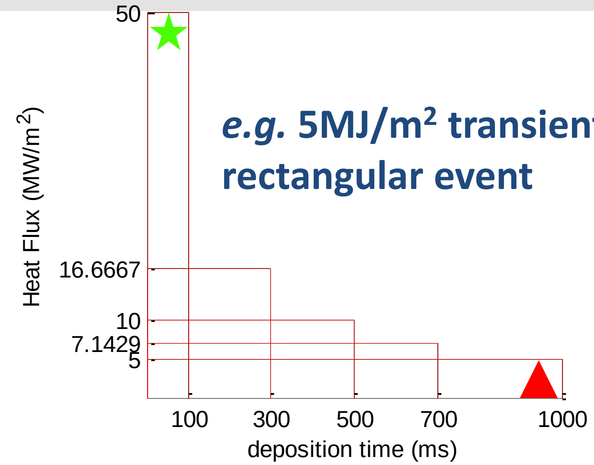
RACLETTE inputs:

- Energy (MJ/m^2) : 1 to 100
- Deposition time (ms): 0.1 to 10^4

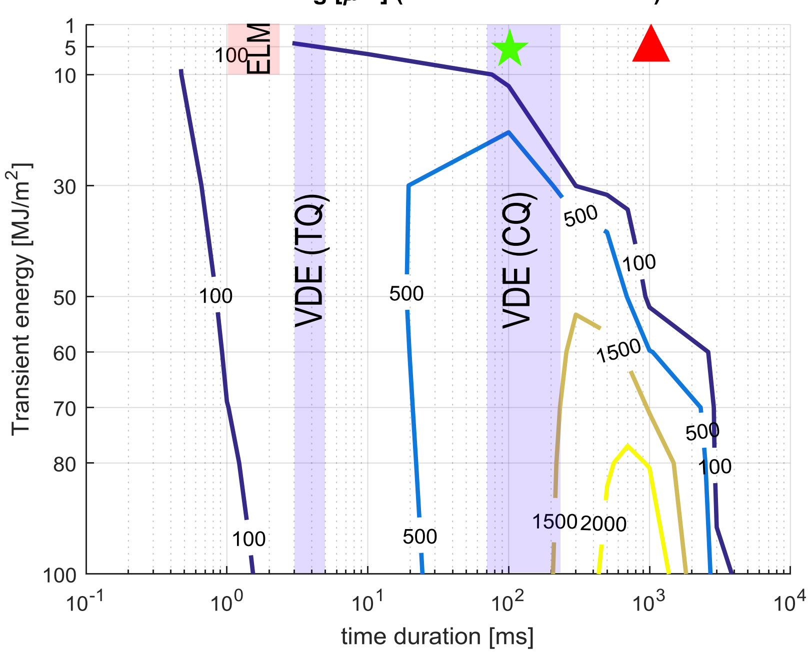


H_2O coolant, CuCrZr heat-sink

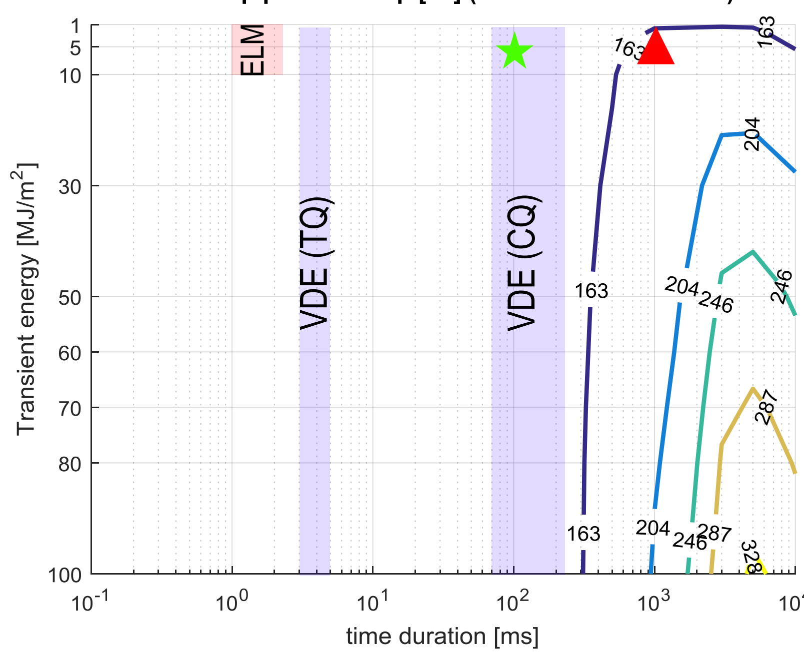
Coolant parameters:
Vel = 8m/s
Pres = 5 Mpa, T_coolant = 150°C



W Melting [μm] (W20mm-Cu-H2O150C)



CuCrZr pipe max temp [°C] (W20mm-Cu-H2O150C)



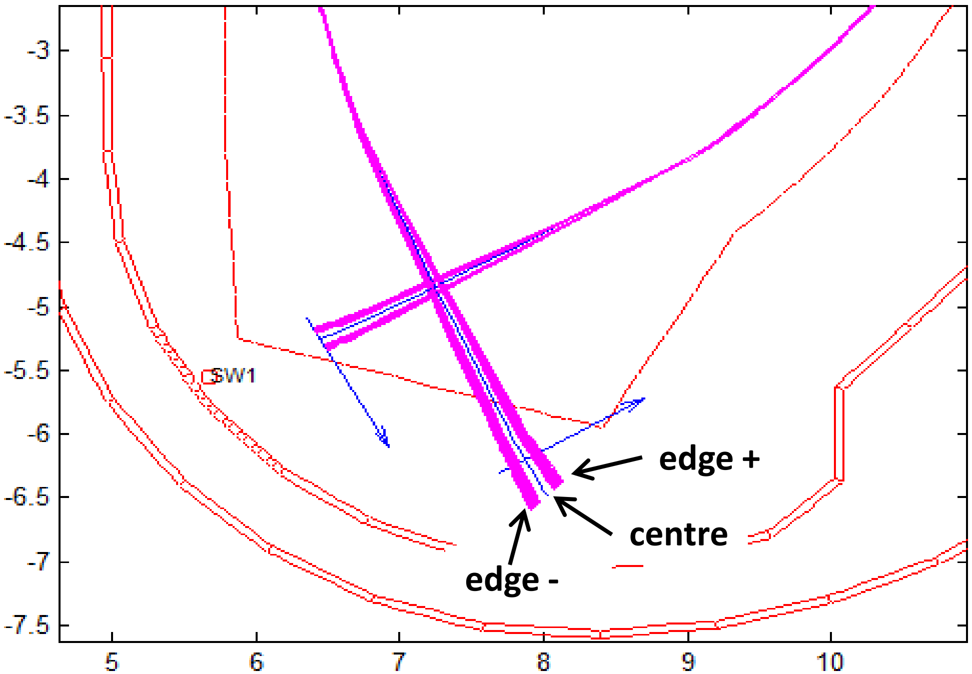
- **Fast transients** (≤ 2-3ms): only the armour surface is affected. W melt limit is quickly exceeded
- **Slower transient:** CuCrZr below temp. limit (350°C) with armor ≥ 20mm. Mitigation expected by vapor shielding

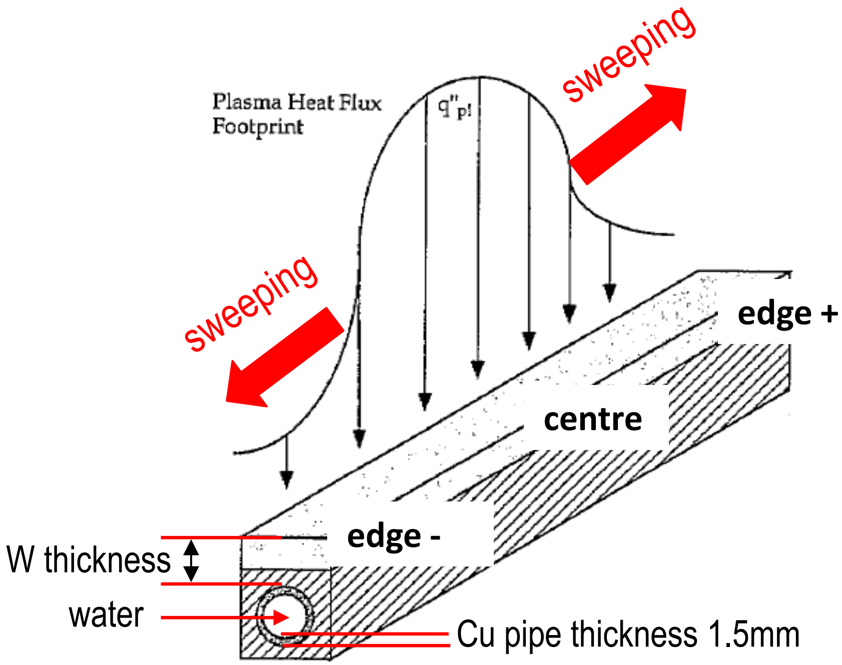
Strike point sweeping parametric scan

Parametric scan: Heat flux chosen levels Q : [20, 30, 40] MW/m²

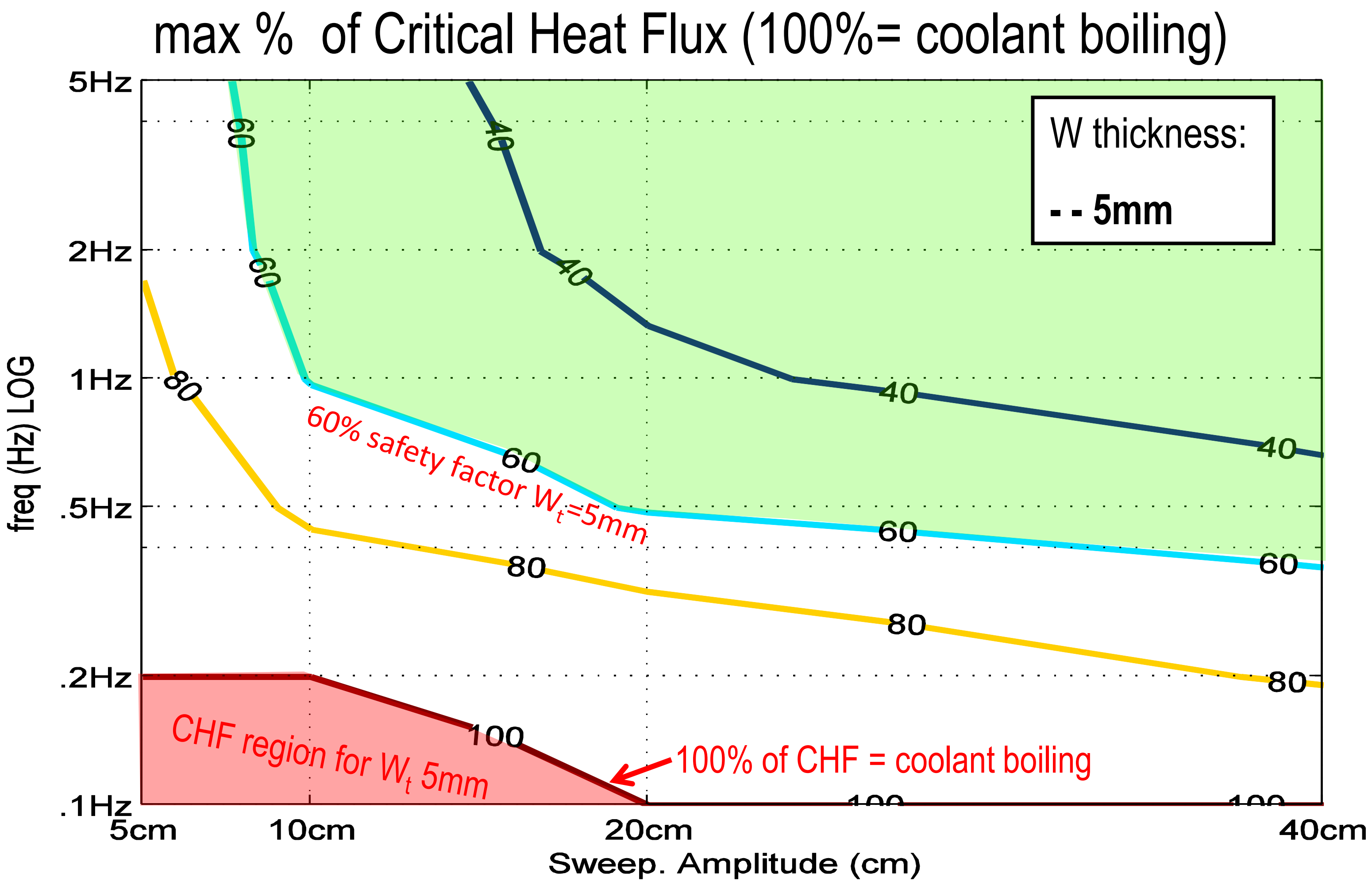
Fixed model parameters DEMO case:	**Scan parameters**
❑ Coolant inlet temperature 200°C;	❑ Armour W Thickness : {5, 10} mm
❑ Coolant pressure 4MPa;	❑ Sweep. Amplitude : {5 ,10, 20, 40} cm
❑ Water velocity 11m/s.	❑ Freq. : {0.1, 0.2, 0.5, 1.0, 2.0, 5.0} Hz

sweeping: periodic strike points oscillation



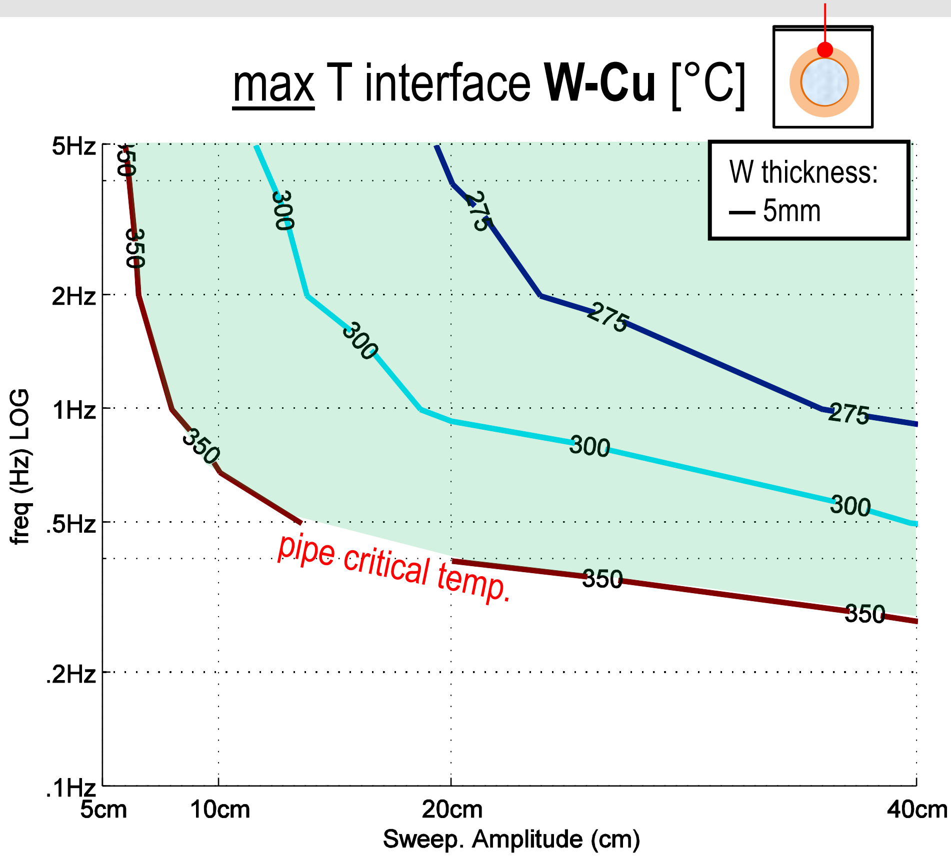


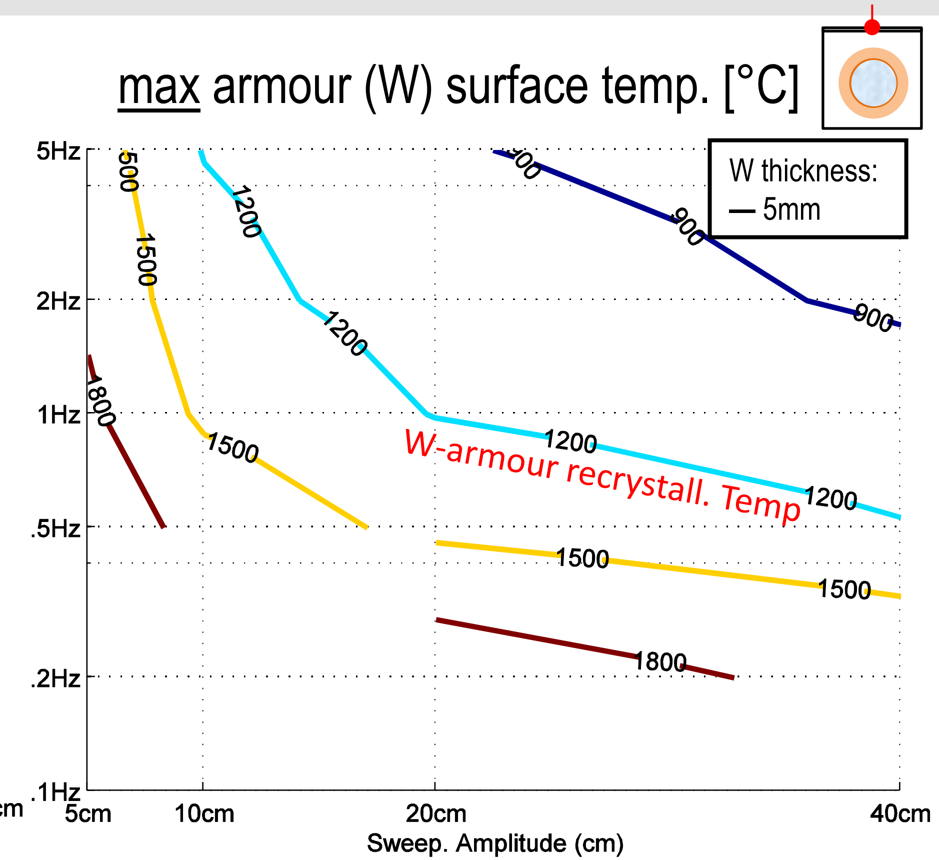
Sweeping frequency-amplitude operational range



The region where the HF to coolant **< 60%** (**safety factor= 1.7**) of CHF is a)0.5Hz & >20cm, and b)10cm&>1Hz, (for Q=30MW/m^2)

Sweeping frequency-amplitude operational range





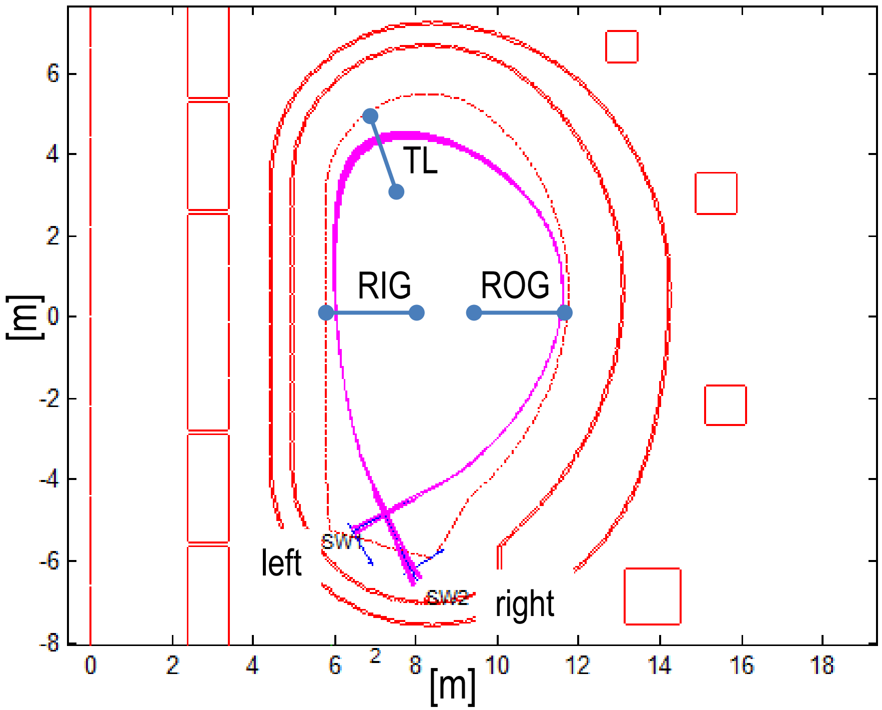
- **Cu pipe critical temperature = 350°C.**
- **For Temp. ≥ 300 °C Cu-alloy start softening/aging**

- **No melting in this operational space.**
- **If recrystallization to be considered the op. space is limited at 30MW/m^2**

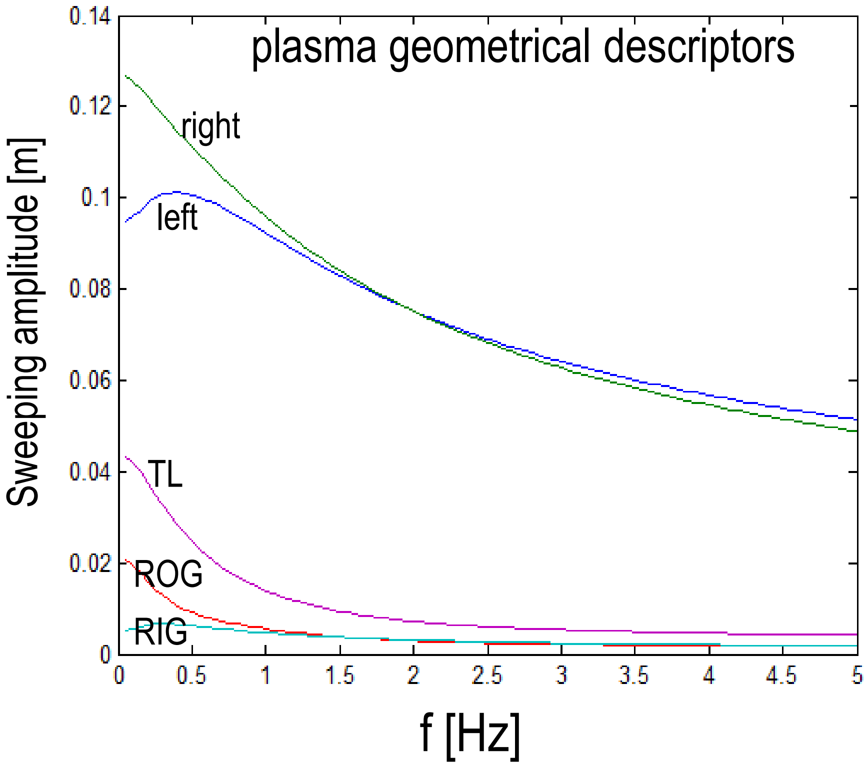
Union of **CHF**, **W-surf temp**. and **pipe temp**. ranges determines operational space of interest

*Missing point if CHF reached.

Sweeping effect on overall plasma boundary variation

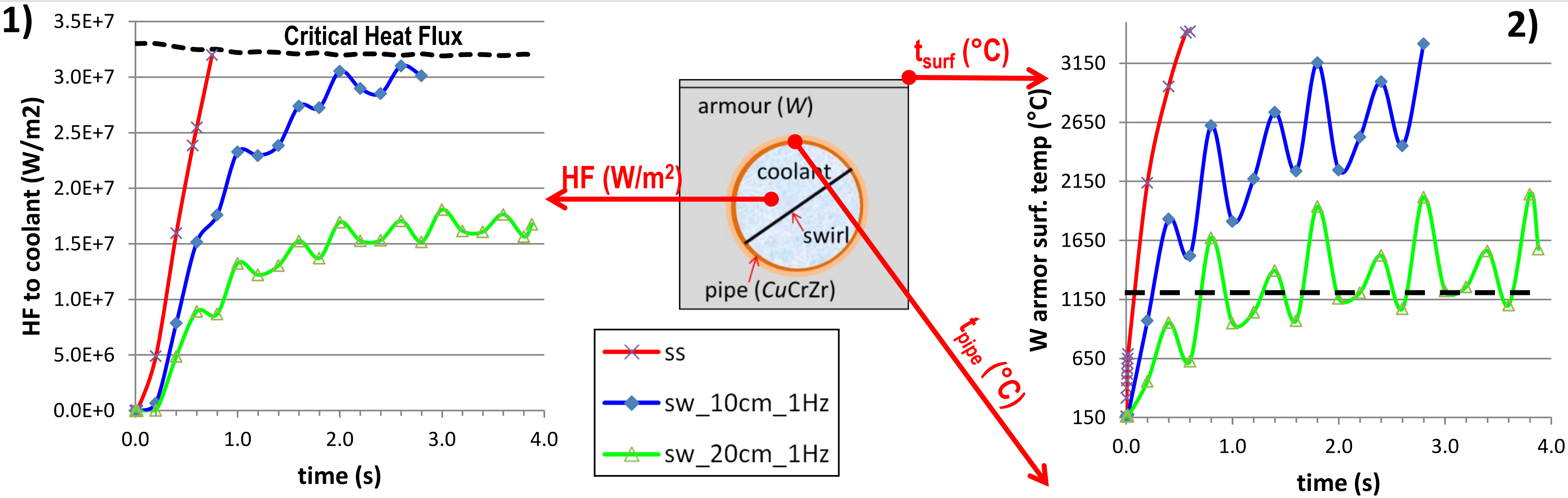


Plasma shape at sweeping edges



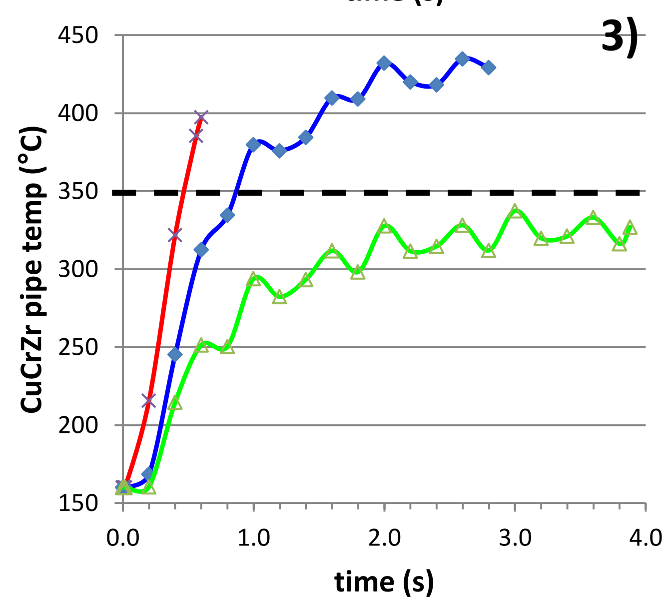
- Motion of the plasma core, including Radial Inner/Outer Gap (**RIG/ROG**) limited to <u>less than 15%</u> of the strike-point motion.
- Top Left (**TL**) Gap moves 30% of the strike-point motion at 0.2 Hz (slightly less than 20% at 1 Hz), due to the vicinity of a null point.

Thermal analysis with RACLETTE: 70 MW/m²



Results with incident Heat Flux = 70 MW/m²

1) **HF to coolant**: In SS the CHF (pipe burn out) is reached in 0.7s, while the 10cm-1Hz sweeping is marginal, and the 20cm-1Hz allows 50% margin.
2) **W armor temp.**: In SS the W surface melt at the CHF time, while in the 10cm-1Hz it reaches melting in ≈3s, and in the 20cm-1Hz the temp. reaches 2000°C(> recr.).
3) **CuCrZr pipe temp.**: The pipe softening temperature of 350°C is reached in 0.5s in SS, and 1s in 10cm-1Hz seeping, while it is not reached for the 20cm-1Hz case.



Conclusions

- DEMO requirements are different from ITER: wall load specification needs to be developed independently.
- Present first wall heat load limits of $1MW/m^2$ can be achieved for steady state and controllable perturbation. Critical areas: baffles and upper FW
- Control margins, and tolerances detrimental effects will require further technology, geometry, and plasma optimization.
- Transient events as RU/RD plasma limited phases, and <u>disruptive events exceed the standard BB limit</u>: specific designs required to protect the wall.
- Discrete (sacrificial) limiters requirements to avoid FW-BB severe damages, *e.g.* Loss Of Coolant Accident events.
- Prediction and design of sacrificial limiters for plasma-wall contact to be carefully assessed, possibly for any foreseeable and unforeseeable event, via geometry and plasma optimization:

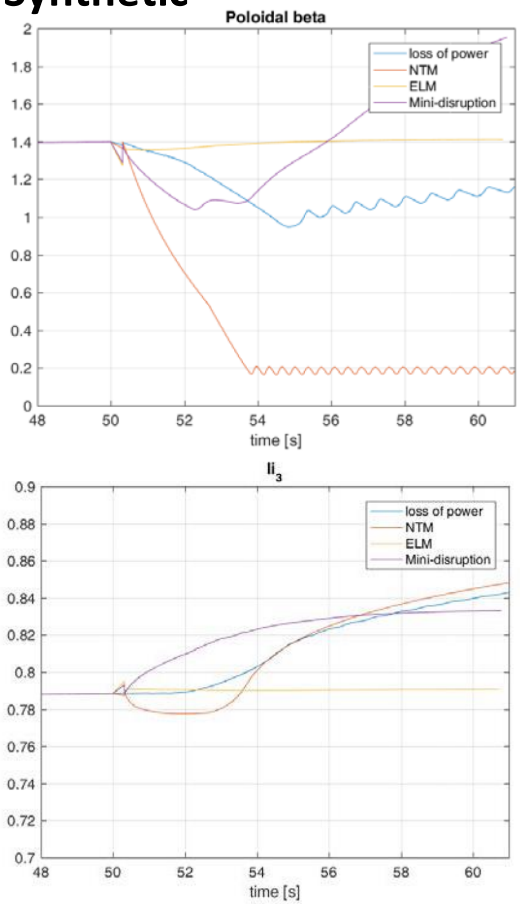
Disruption simulations: HF and REs

(backup slide)

Several activities launched to predict possible contact points:

- ❑ Inter-machine perturbation database (JET, EAST, ASDEX, TCV)
- ❑ Modelling of perturbation effect on plasma shape-movement
- ❑ Simulations with CARMA0NL/CREATE & MAXFEA

Synthetic



Synthetic (ASTRA) database, perturbations generated for:

- ❑ ntm—like
- ❑ W influx
- ❑ ELM like
- ❑ Minor disruption,...

IPP-CREATE simulations

Experimental database,

- ❑ JET, EAST, ASDEX, TCV:
- ❑ H-L, L-H
- ❑ ELMs
- ❑ Minor disruption
- ❑ SN/DN

Universities of ***Tuscia/Cagliari***

Experimental