Plasma transient challenges (disruptions, detachment loss, equilibrium) and resulting requirements for the machine design of a DEMO tokamak reactor

8th IAEA DEMO Workshop, Vienna, 30.08.2022

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FUSION TECHNOLOGY DEPARTMENT

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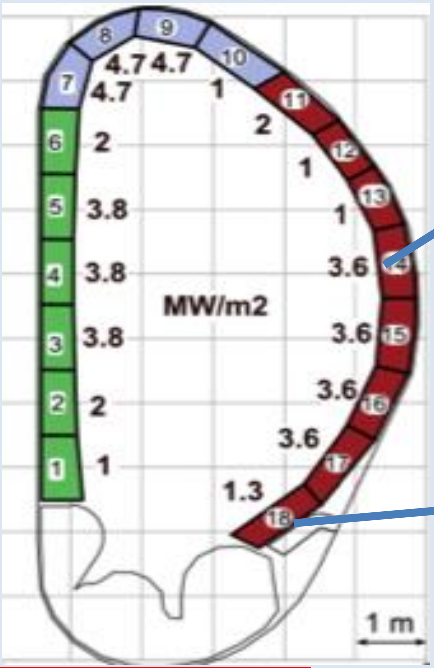
Outline

- Introduction: ITER and DEMO heat load requirements
- Plasma transient identification
- First Wall Transient Loads
- Divertor Transient Heat loads
- Conclusions

ITER and DEMO heat load requirements

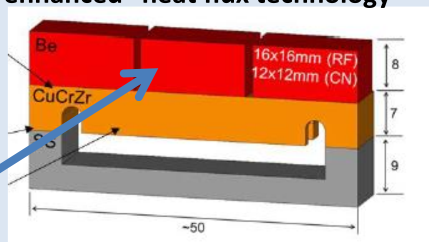
ITER:

- FW has no tritium breeding requirements.
- A large fraction of ITER's <u>Cu-alloy</u> first-wall can be designed for up to **~5 MW/m²**. (CuCrZr has extremely high K~300 W/mK but irradiation lifetime of only ~10 dpa)

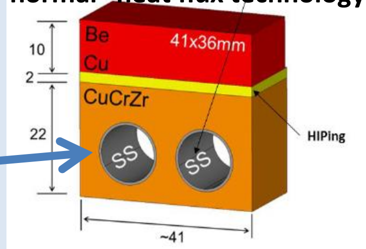


R. Mitteau, JNM 2011

"enhanced" heat flux technology



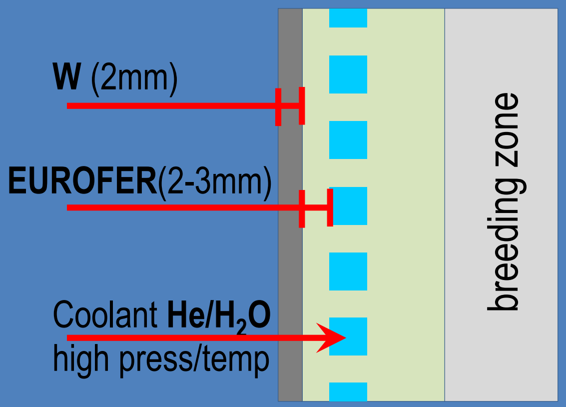
"normal" heat flux technology



DEMO:

- Tritium breeding: FW with thin layer of materials.
- DEMO FW structural material: <u>EUROFER</u> (much lower thermal conductivity K~30 W/mK, but high irradiation lifetime) → Steady state heat loads limited to **~1-2 MW/m².**
- 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.

First wall - breeding blanket



ITER conformal wall: precision required difficult to achieve with DEMO ≈9m tall BB segments

Present ITER SS limit up to 4.7MW/m²: DEMO (~1-2 MW/m²)load specification developed independently

EU-DEMO FW protection from plasma transients

EU-DEMO Gate review in 2020: Key Design Integration Issue#1: Design, performance and feasibility of wall protection limiters during plasma transients

Design process:

1) Transient identification:
- transport simulations
- ITER HNLS
- Experimental (JET, AUG, TCV, EAST)

2) Electromagnetic sim.:
- 2D/3D EM models
- Closed loop sim.
- Magnetic + kinetic (ongoing) see E. Fable

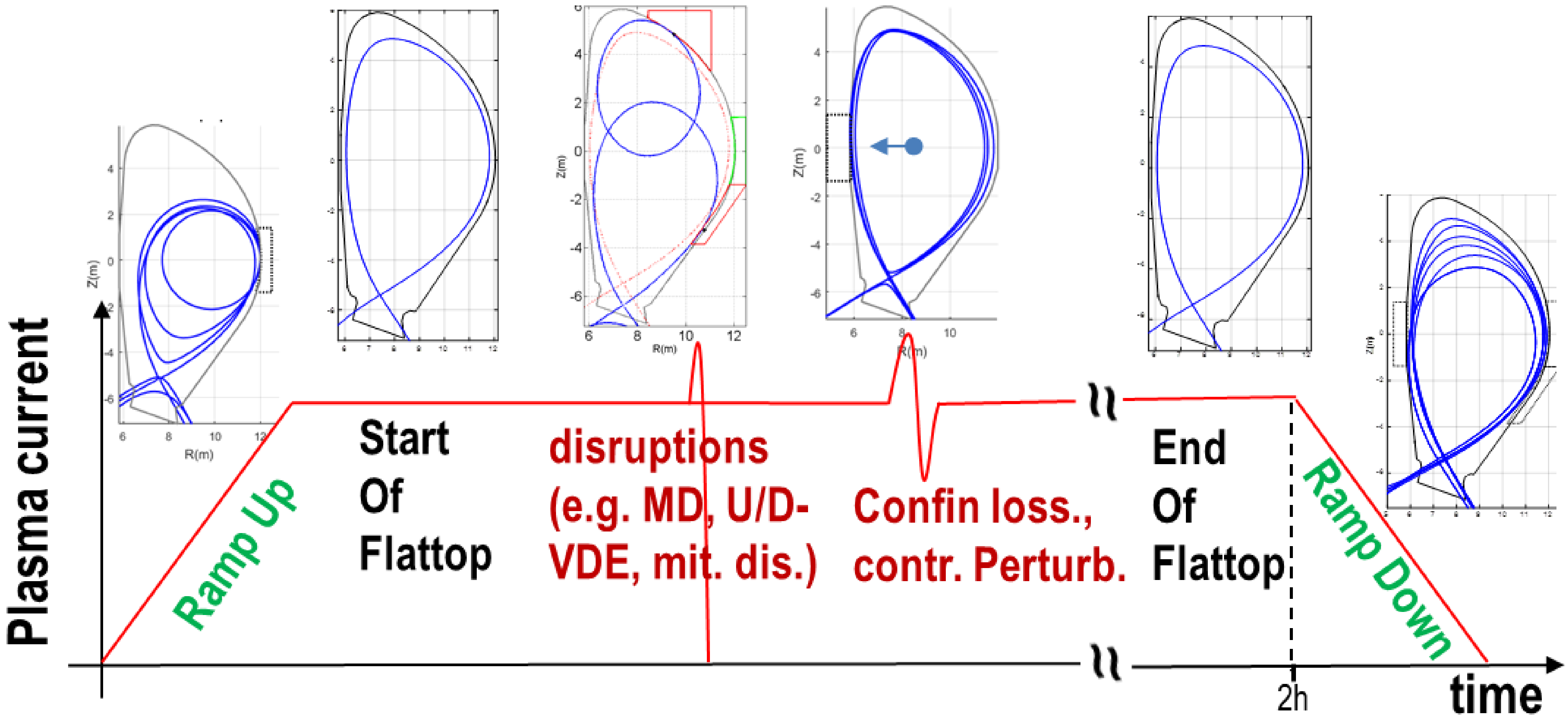
3) 3D Heat Flux calc.:
- Charged particles
- Radiation loads
- Misalignment, e-folding length sensitivity studies

4) Limiters shaping:
- shape + protrusion
- poloidal location
- number

5) Thermal calc./PFC des.:
- 1.5D/3D
- Vapour shielding
- Runaway electrons (REs)
- Advanced Materials

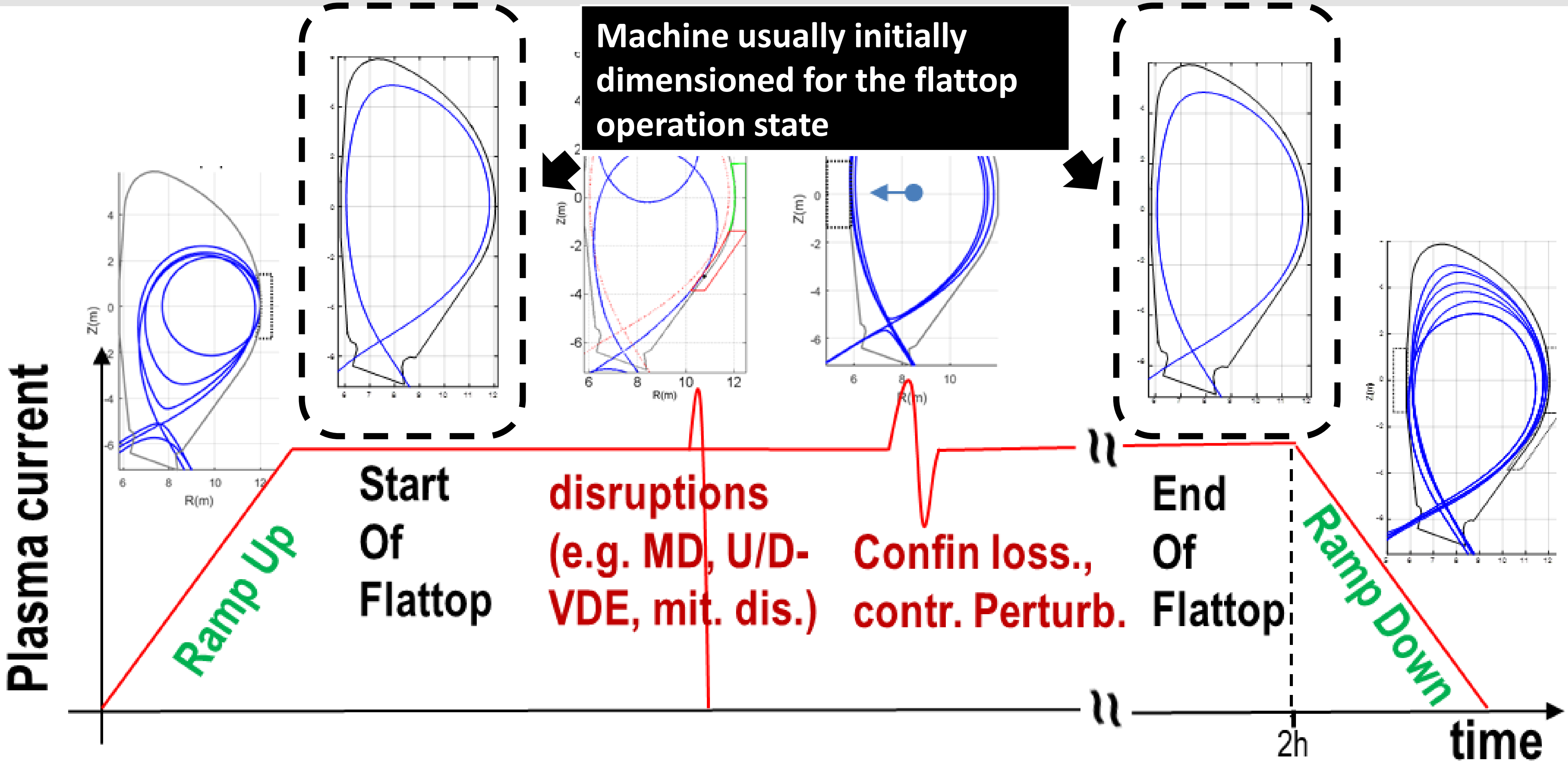
6) Limiter design:
- Integration
- Remote maintenance
- Inspection
- TBR

Transient list: Normal v.s. Off-Normal events



During RU/RD the plasma touches the wall when its current is smaller than surrounding currents, e.g. at the beginning/end of every pulse	Elongated plasmas are vertically unstable: If control is lost the plasma moves upward or downward	If plasma looses Energy, moves inward (if the movement is above the controller limits the plasma may touch the PFC)

Transient list: Normal v.s. Off-Normal events



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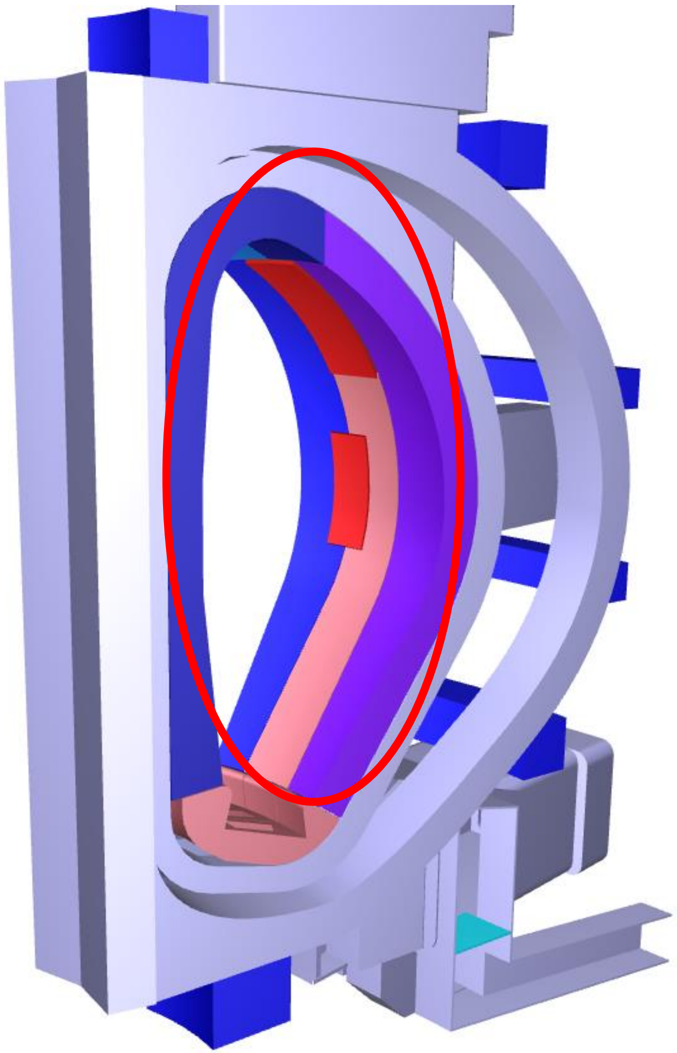
First Wall Transient Loads



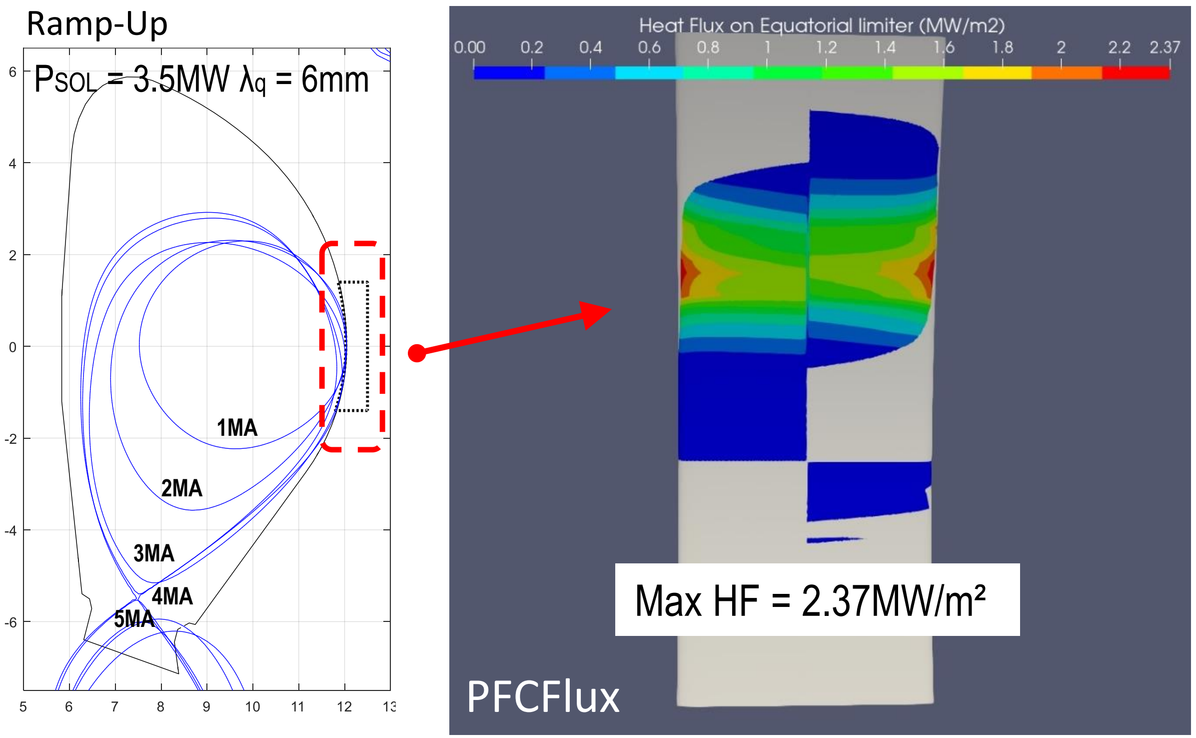
- Ramp Up/Down limited phases
- Upward/Downward Vertical Displacement Event
- Loss of Confinement
- Mitigated Disruptions

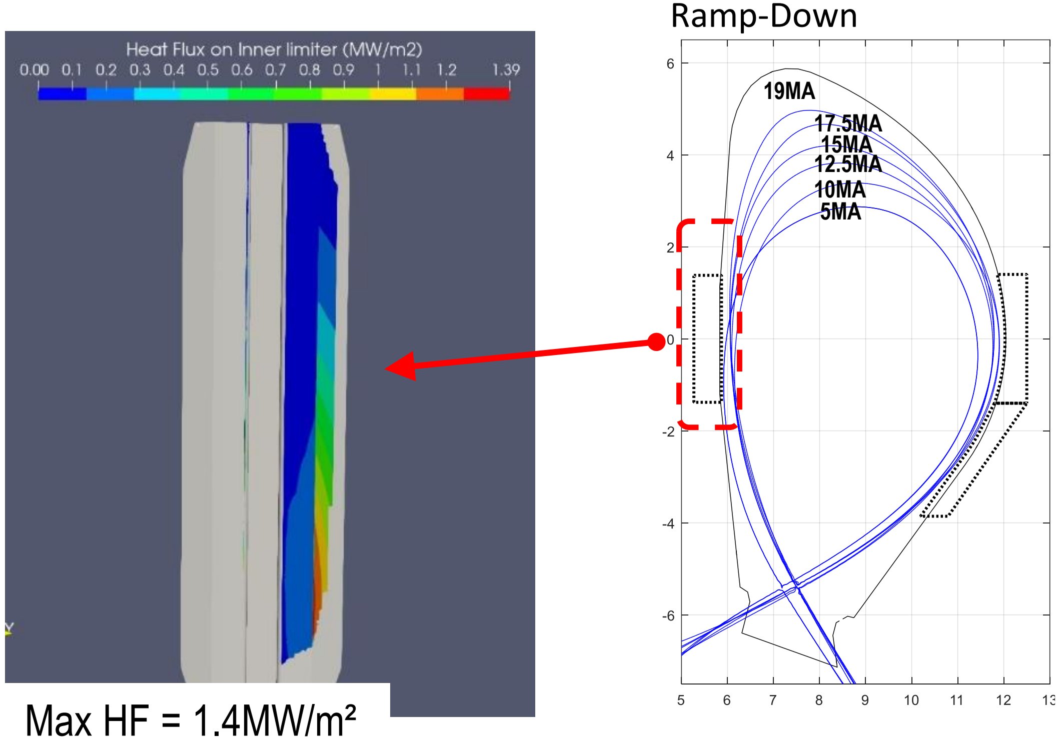
3D HF calculations and limiter surface design

Normal transients: **ramp-up/down** on Outer Midplane Limiter (OML)

- Plasma max Ramp-Up/Down assumed [0.1; 0.2]MA/s.
- **λ_q = 6mm, Psol[MW] = Ip[MA]** assumption (ITER like)
- RU: x-point formation in range at **[3.5; 6]MA**: **t_{RU}= 18 to 60s**

RU: Limited eq. 3.5MA, #4 OML

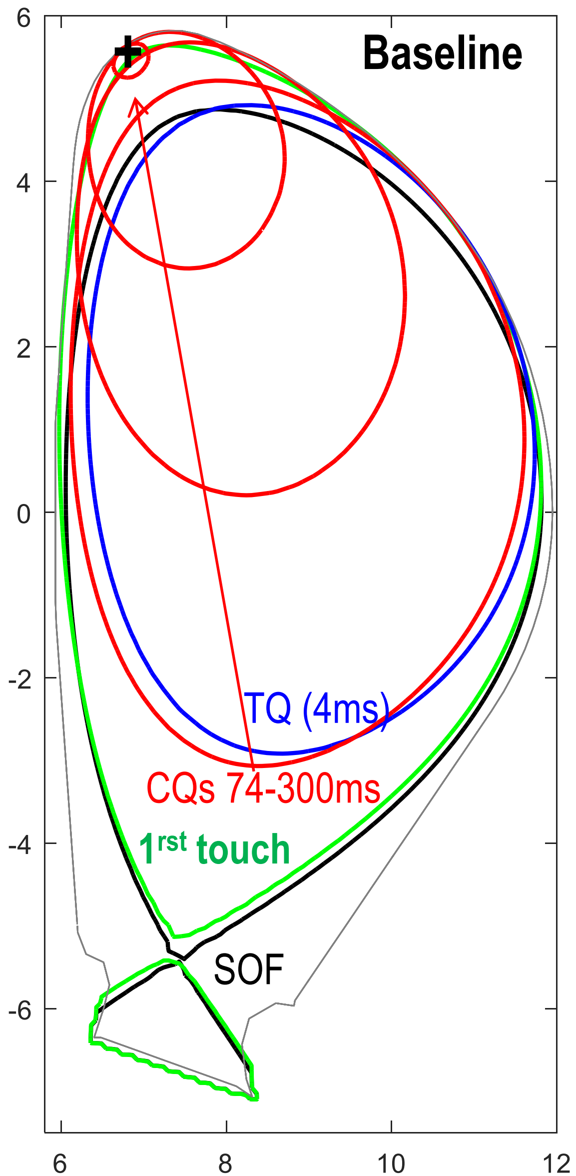




Misalignments studies performed. Max HF may be reduced if limiter adjustable at OMP port. Bare wall HF ≈3-4MW/m^2
No relevant HF found on other BB modules, nor on the limiter during flat-top phases
RD initial simulations: aim to remain diverted as long as possible (integrated simulations planned)

VDE simulations and heat loads calculations

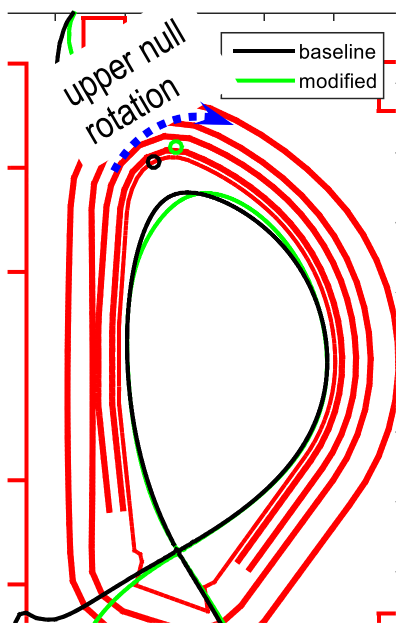
Off-normal transients: **Upward Vertical Displacement Event** and **scenario optimization**



Typical plasma VDE evolution:

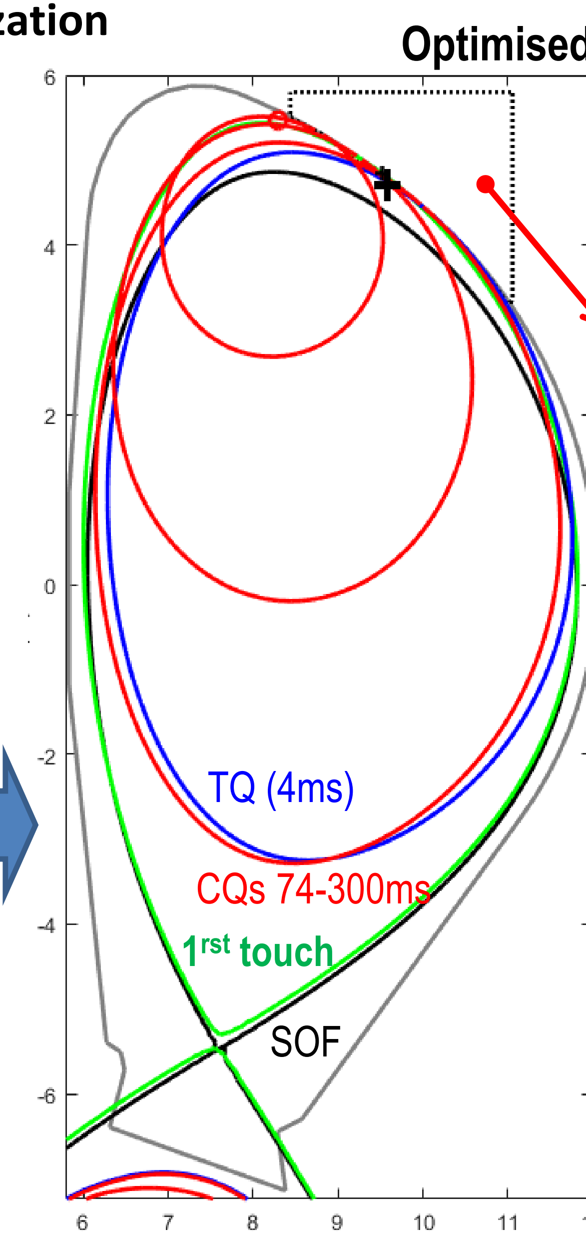
1) SOF (Start Of Flat-top)
2) **1rst touch** (**+** plasma moves vertically)
3) TQ (W_{th} from 1.3GJ to 0, in 4ms)
4) CQs (I_p from 19MA to 0, in 74-300ms)

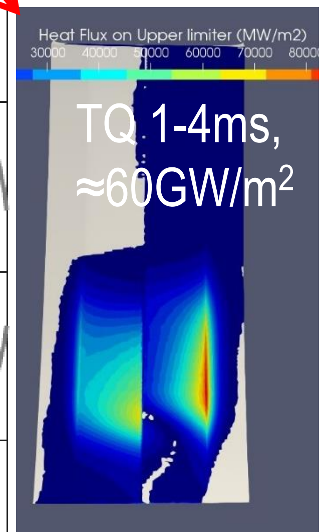
Baseline
- **1rst touch** close or at 11 'O clock
- **CQ** ends up at 11 'O clock



Optimised scenario
- **1rst touch**, TQ and CQ moved towards upper port area.

Obtained by moving upper x-point clockwise ≈60cm (upper-triangularity, $\delta_{95\%}$ from 0.33 to 0.25)





#8 UL used to prevent large charged particle HF reaching FW at TQ

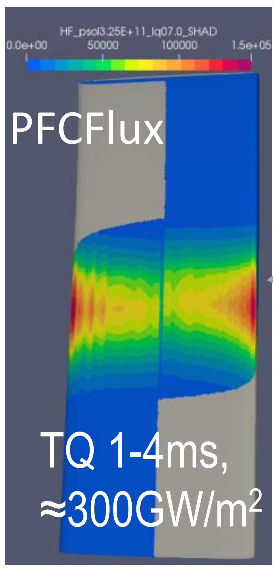
VDE simulations and heat loads calculations

Off-normal transients

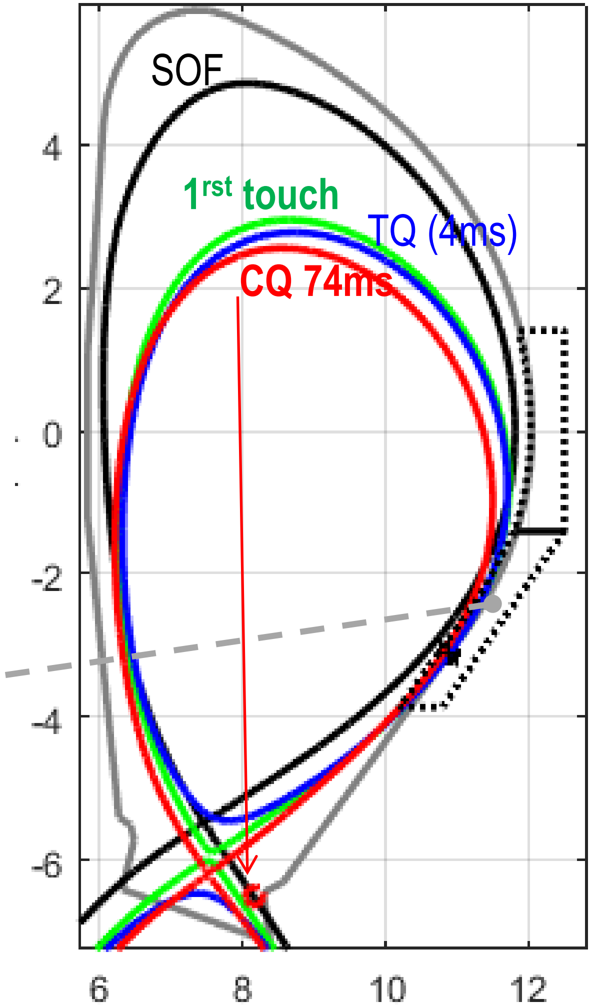
Typical plasma VDE evolution:

1) SOF (Start Of Flat-top)
2) **1rst touch** (**+** plasma moves vertically)
3) TQ (W_{th} from 1.3GJ to 0, in 4ms, λ_q=7mm)
4) CQs (I_p from 19MA to 0, in 74-300ms)

#4 OLL used to prevent FW damages during TQ

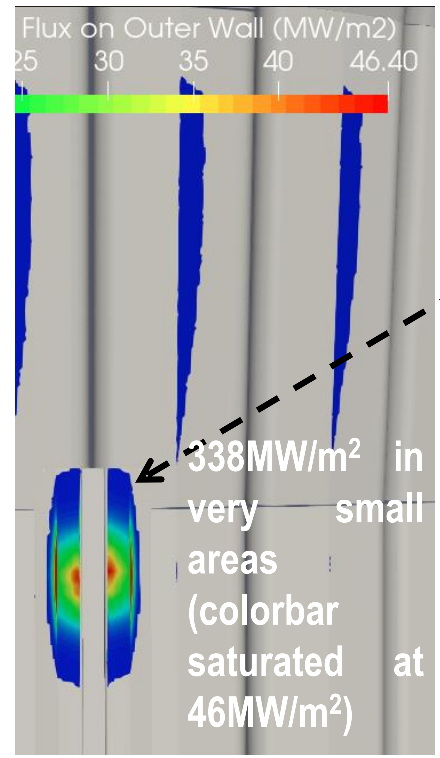


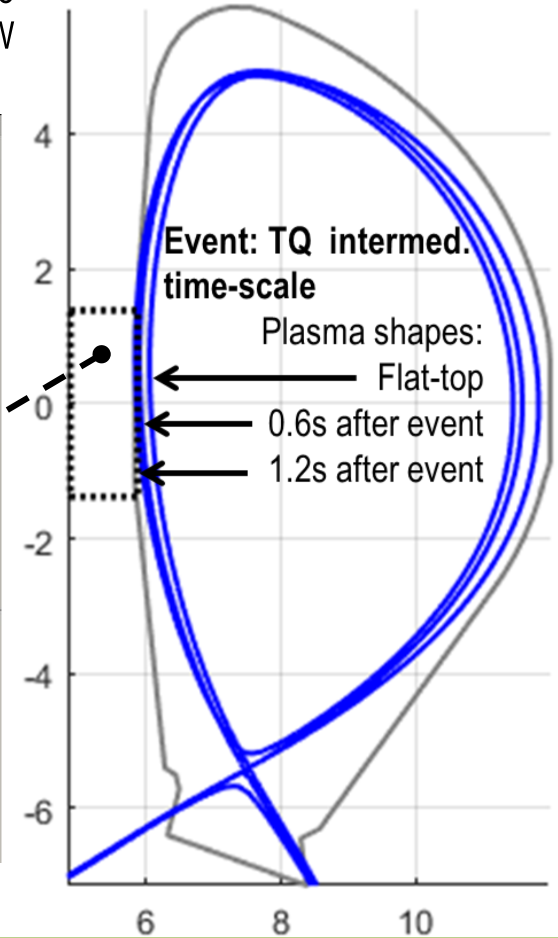
Downward-VDE on Outer Lower Limiter (OLL)



Loss of confinement: Conservative case on Inner Midplane Limiter (IML)

#4 IML used to prevent FW damages during TQ





Proposed limiters are able to prevent heat flux on the First Wall above the limits. Damages to the sacrificial limiters expected. Increase ***VDE controllability***

Inner Mid-plane Limiter far from maintenance ports: very challenging to maintain in case of damages. Strategies to ***enhance radial control*** being studied

Design criteria for the in-vessel equatorial coils: Vertical Stability

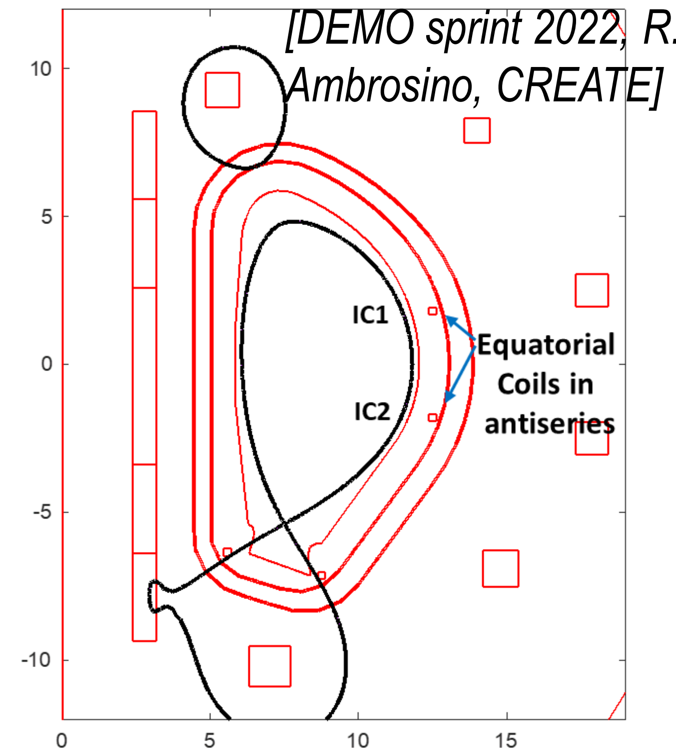
Vertical Stabilization: ITER requirement [*Y. Gribov NF 2015*] led to ITER-IVC introduction: V.S. system must be able to stabilize VDE events

- 'reliable' operation: max(Z0)/a ≈ 5% (15cm for DEMO)
- 'robust' operation: max(Z0)/a ≈ 10% (30cm)

Corresponding to [15-30]cm for EU-DEMO.

Simulations assuming 9 turns, (ITER has 4) with a range of poloidal beta $\beta_{pol} \in [0.1\ 1.04]$ and internal inductance $l_i \in [0.7\ 1.4]$:

With IVC	$Max\ Z_0$ $15cm$	$Max\ Z_0$ $30cm$
Voltage [kV]	2.07	4.14
Current [kA]	8.52	17.04
Power [MW]	17.6	70.54



Ex-Vessel coils VS performance

- Maximum 15cm VDE for $l_i \leq 0.8$
- Maximum 5cm VDE for $l_i = 1$
- Not stabilizable for $l_i > 1$

Assuming ITER technology (max current 60kA (peak), 15kA (DC) the ***control is achievable for present baseline only with IVC***.

Engineering integration and maintenance studies started for EU-DEMO during Pre and Conceptual Design Phase

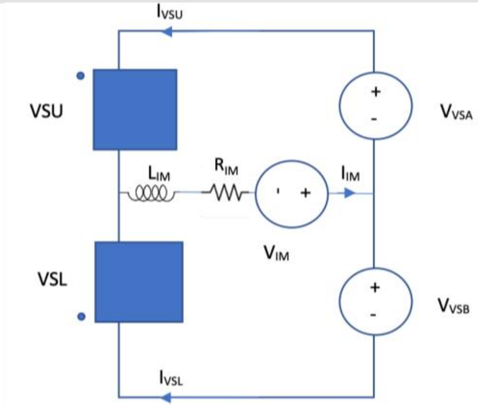
Design criteria for the in-vessel equatorial coils: Fast Radial Control

Fast Radial Control (FRC)

Able to provide a significant contribution to the vertical field during fast transients:

- Loss of confinement(e.g. H-L transitions, Additional Heating failures)
- in the plasma current raise during breakdown

Closed loop simulations including additional imbalance current circuit

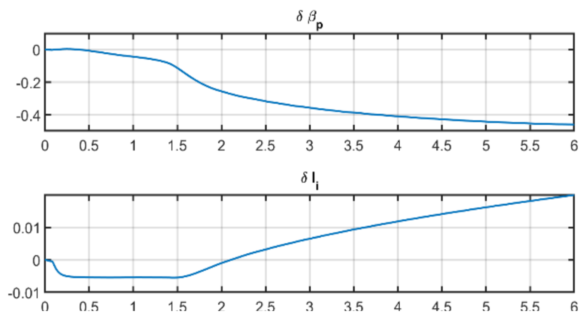


[R. Ambrosino, M. Ariola, CREATE]

Type of events

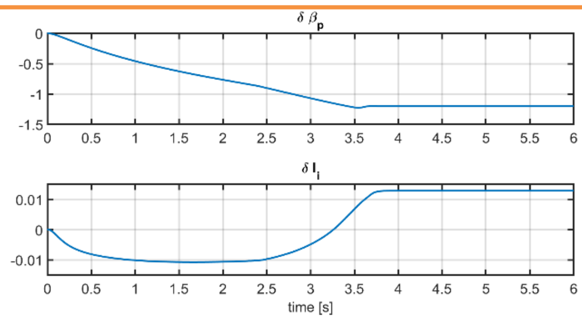
Performances

Loss of 40 MW NBI power



Controller	Minimum distance from wall	Max current	Voltage saturation	Maximum power
No FRC	5.7 cm	//	//	***700-900 MW***
FRC	***10.8 cm***	25 kA	1.65 kV	***26 MW***

Thermal Quench (TQ) intermediate timescale



Controller	Time to contact	Maximum current	Voltage saturation	Maximum power
No FRC	1.183 s	//	//	//
FRC	***1.974 s***	58 kA	2 kV	96 MW

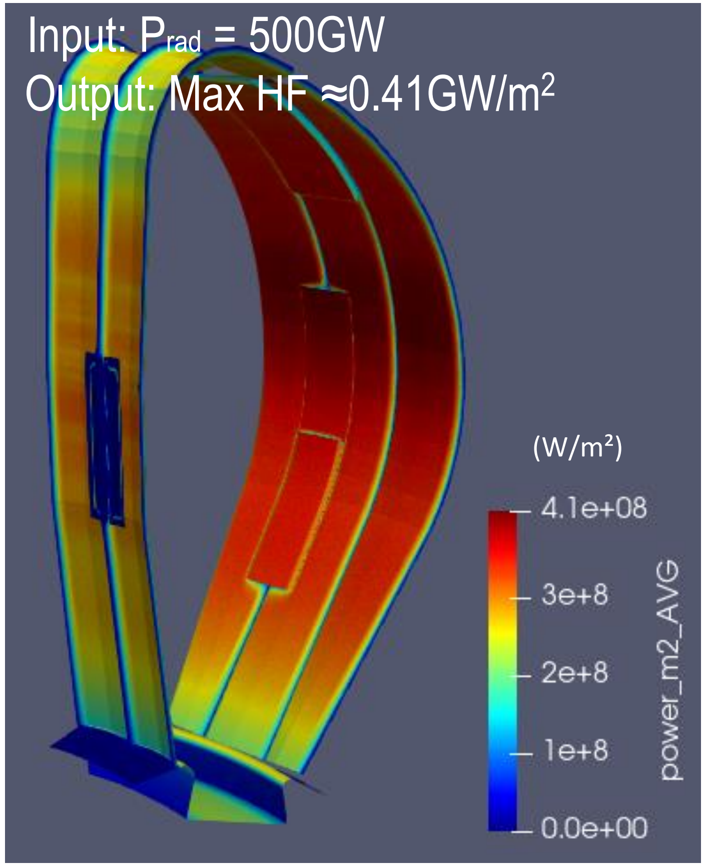
With FRC there is a significant improvement in the ***performance***, (e.g. ***plasma-wall distance***, ***control power***).

Only up to a certain class of events the plasma-wall contact can be avoided (e.g. not in the conservative TQ case)

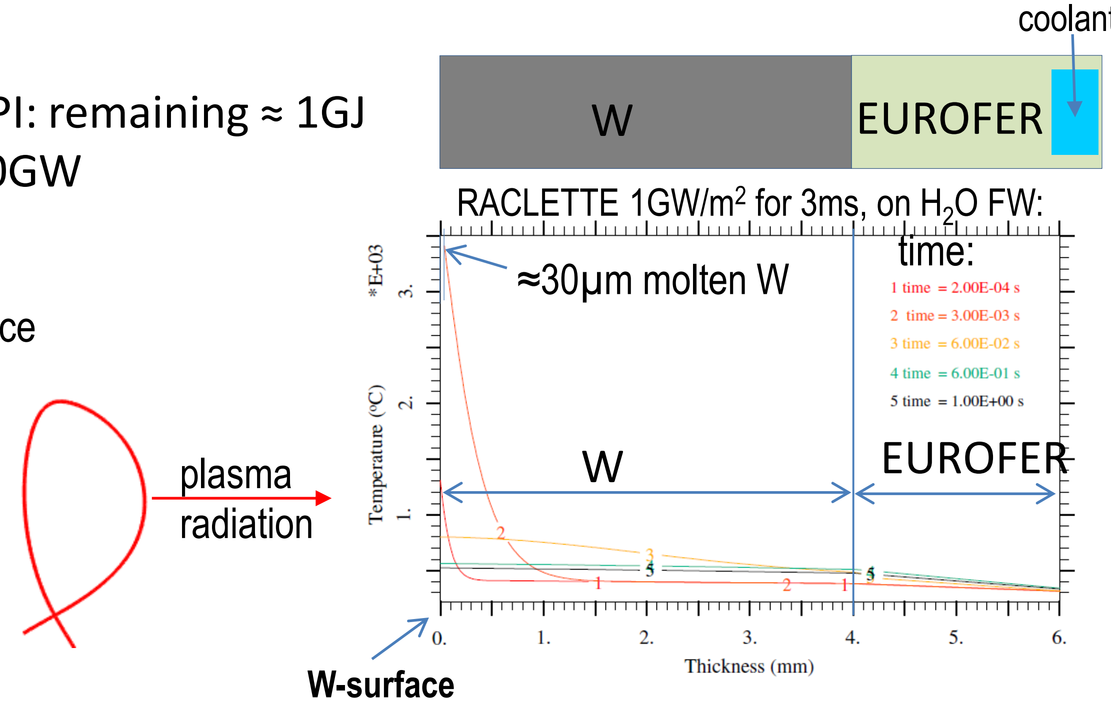
Radiation during mitigated disruptions

Preliminary results: Mitigated Major Disruption or U-VDE :

- Initial thermal energy W_{th}=1.3GJ: 20% radiated at pre-TQ at MGI/SPI: remaining ≈ 1GJ
- At TQ ITER aims at radiating 80% in 1-3ms (controllable) -> P_{rad}≈800GW



If Toroidal & Poloidal peaking factors(TPF)*=***2.8***, for P_{rad} = **800GW** (hence P_{rad} = 2.2TW) ⟶ max HF ≈ **2GW/m²**
*ITER uses TPF= 1.8 (tor.) and 1.5-4.5 (pol.).



80% radiation in 3ms may be above FW W-limit

MGI/SPI TQ radiation peak density should be minimised(ITER active research, see M. Lehnen 6th IAEA DEMO Workshop)

Mitigation techniques must consider FW damages (limiters are ineffective)

Cooling pipe below limits

3D HF calculations and limiter surface design

All considered perturbations and relative HF on limiters and FW:

Inputs					Outputs: max HF (MW/m^2) *(Italic):* with radiation, **Bold: GW/m^2**				
Scenario	Case	P_{SOL} (MW)	λ_q (mm)	Deposition time	OML	UL	OLL	IML	FW
SOF	Diverted	69	50	Steady state	0.53*(0.65)*	0.82*(1.10)*	0.09*(0.33)*	0*(0.19)*	0.40*(0.59)*
EOF	Diverted	69	50	Steady state	0.54*(0.74)*	1.01*(1.33)*	0.1*(0.36)*	1.84*(2.11)*	0.48*(0.67)*
Min disr	Diverted	69	50	15-50ms	<0.01	0.13	0.01	3.06	0.69
ELM	Diverted	69	50	15-50ms	1.40	0.56	0	0	1.48
Ramp-Up	Limited	3.5	6	17.5-35s	2.37	0	0	0	0.29
Ramp-Down	Limited	5	6	25-50s	<0.01	<0.01	<0.01	0.02	0.01
		5	50	25-50s	<0.01	<0.01	<0.01	1.39	0.60
U-VDE	First touch	69	1	20-35ms	<0.01	114[(2)]	<0.01	0	0
		69	5	20-35ms	<0.01	15.6	<0.01	0	0.02
	TQ	**325**	7	1-4ms	<0.01	**63**[(3)]	0	<0.01	138[(8)]
	Current Quench	10	10	74-200ms	<0.01	2.52	0	<0.01	0.01
		10	30	74-200ms	<0.01	1.53	0	<0.01	0.11
D-VDE	First touch	10 (*69)	10 (*1)	15-35ms	<0.01(*0.01)	0(*0)	<0.01(*24.8)	<0.01(*<0.01)	<0.01(*<0.01)
		10 (*69)	30 (*5)	15-35ms	<0.01(*0.01)	0(*0)	<0.01(*7.83)	<0.01(*<0.01)	0.08(*0.01)
	TQ	**325**	7	1-4ms	0.77(*182)[(1)]	0(*0)	**4.4(*306**[(4)]**)**	0.84(*11.3)	8.11(*292[(9)])
	Current quench	10	10	74-200ms	<0.01	<0.01	<0.01	<0.01	<0.01
		10	30	74-200ms	<0.01	<0.01	<0.01	<0.01	<0.01
H-L transition	Limited (inboard)	30	2	1-5s	<0.01	<0.01	<0.01	338[(5)]	0.23
		30	4	1-5s	<0.01	<0.01	<0.01	147[(6)]	2.2
Major Disruption (MD)	TQ	**325**	7	1-4ms	0.61	1.38	0.84	**8.5**[(7)]	336[(10)]
	CQ	10	10	74-200ms	<0.01	<0.01	<0.01	0.01	<0.01
		10	30	74-200ms	<0.01	<0.01	<0.01	0.21	0.05
Mitig.disr.	Mitig - TQ	**2.2**		1ms	**2**[(11)]	**1.8**[(11)]	**1.8**	**1.5**	**2**[(11)]

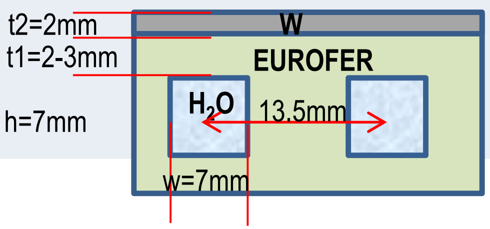
(n) critical cases in **red**

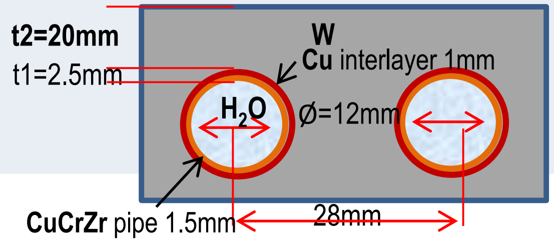
Preliminary misalignment studies for penalty factors.

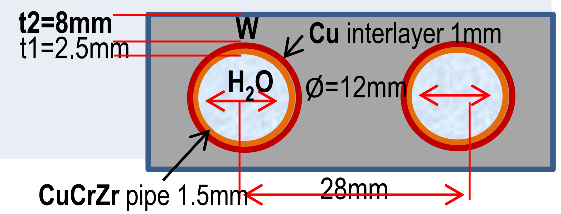
Thermal calculations with RACLETTE code

RACLETTE code used to quickly simulate thermal behaviour of PFC designs:

1 **BB FW** simplified model	2 **Sacrificial limiter** model	3 **Divertor like limiter** model
H_2O coolant, <u>EUROFER</u> heat-sink	**H_2O coolant, <u>CuCrZr</u> heat-sink**	**H_2O coolant, <u>CuCrZr</u> heat-sink**
Coolant parameters: Vel = 8m/s Pres = 15 MPa T_coolant = 300°C	Coolant parameters: Vel = 8m/s Pres = 5 MPa T_coolant = 160°C	Coolant parameters: Vel = 8m/s Pres = 5 MPa T_coolant = 160°C







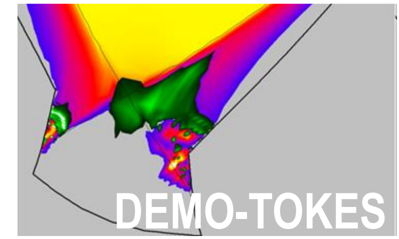
	Case	W-Evap. (µm)	W-Melt. (µm)	Surf. temp. (°C)	Heat sink temp. (°C)
3 →	**Divertor like limiter: (CuCrZr heat sink temp. lim. 350°C)**				
	D-VDE TQ [(1)]	0	0	958	171
	Mitig. Disr. [(11)]	0	58	4676	168
2 →	**Sacrificial limiter:**			**(CuCrZr heat sink temp. lim. 350°C)**	
	U-VDE FT [(2)]	0	0	1670	173
	U-VDE TQ [(3)]	2770	1084	7921	169
	D-VDE TQ [(4)]	Not converged			
	H-L [(5)]	15400	4246	5378	446
	H-L [(6)]	5300	4484	5075	313
	MD [(7)]	336	305	6695	168
	Mitig. Disr. [(11)]	0	49	4437	168
1 →	**First Wall**			**(EUROFER heat sink temp. limit 550°C)**	
	U-VDE TQ [(8)]	0	0	958	383
	D-VDE TQ [(9)]	0	0	1561	407
	MD [(10)]	0	0	1765	415
	Mitig. Disr. [(11)]	0	60	4676	429

- **All heat sink below limits**
- **FW armour protected, mitig. disr., to be tuned**
- **For VHHF sophisticated codes are being used**

RACLETTE is conservative when W vaporisation ≥tens µm: <u>possible</u> mitigation from vapour shielding

Vapor shielding model in Major Disruption

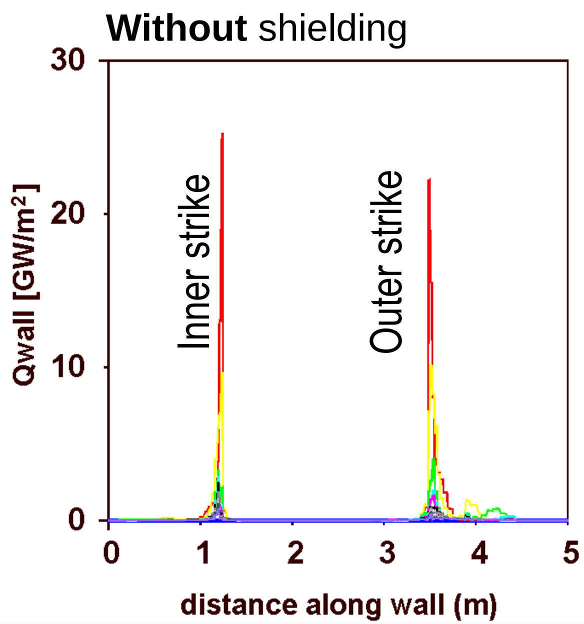
S. Pestchanyi, et al., *FST (2019)*
S. Pestchanyi, et al., NME (2020)



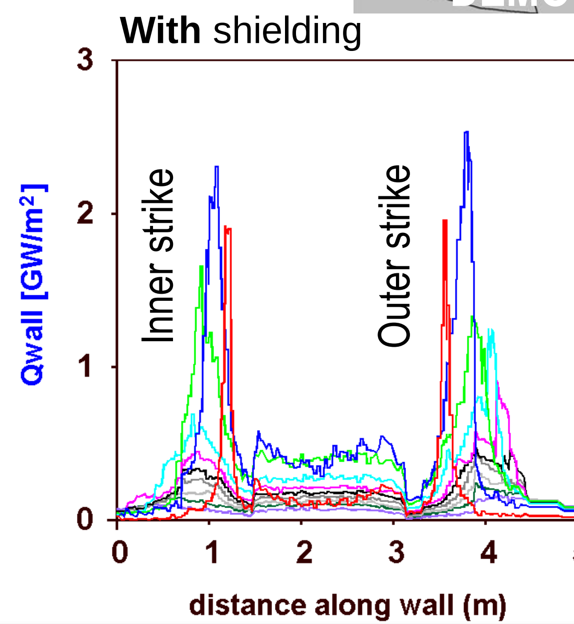
Preliminary simulations including vapor shielding have been performed on DEMO using TOKES code on:

Major (Central) Disruption (plasma in diverted configuration):

- Thermal quench duration **4ms**
- Charged particles energy = **0.65GJ** (1/2 E_{KIN_tot}) (to 1.3GJ)



Colors represent different instants from 1 to 10ms



With vapor shielding factor 10 reduction in Qwall (from 25 GW/m² to 2.5 GW/m²).

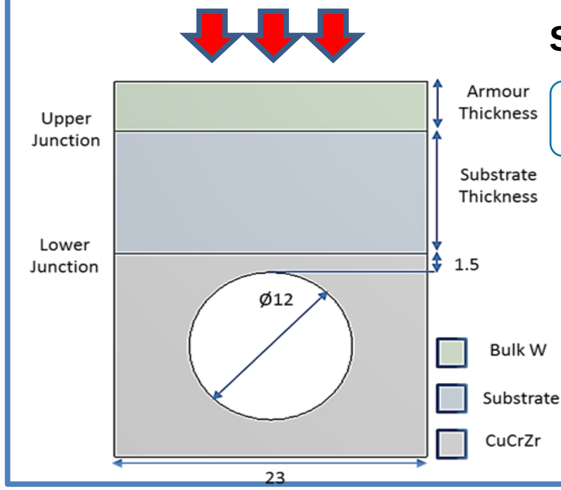
W-Vaporization is reduced from 700µm (in line with RACLETTE▲, ≈4e27 atoms) to 4µm (≈3e24 atoms). Melting from 400µm to 150µm

Preliminary results. In line with ITER modelling [1] and (old) exp. comparison [2]

[1] S.Pestchanyi, et al., FED, vol. 109, p. 141, 2016
[2] S.Pestchanyi, et al., FED, vol. 124, p. 401, 2017

Further DEMO experimental validation requested in QSPA

Armour R&D



Sacrificial limiter as the last protection resource of the reactor wall

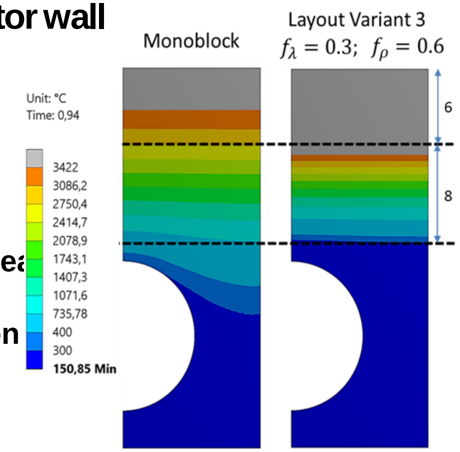
Heat Flux Reduction + **Longer lifetime**

Heat Flux Reduction:

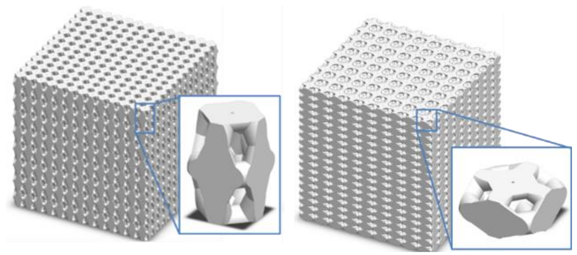
I. **Prompt vapour shielding**
II. Acceptable thermal conductivity
III. High S/V ratio

Longer lifetime:

I. **Less thermal stress and more ductile behaviour**
II. **Avoid overloading of the heat sink**
III. **Hindered crack propagation**



W lattice: tailored metamaterial designed to get desired characteristics and realized in additive manufacturing



e.g.: Anisotropy A=0.5
Constant profile ligament
ligament length L=0.33mm
ligament radius section R=0.15mm
Relative density 53.3%
Thermal Condutivity 48.8 W/mK

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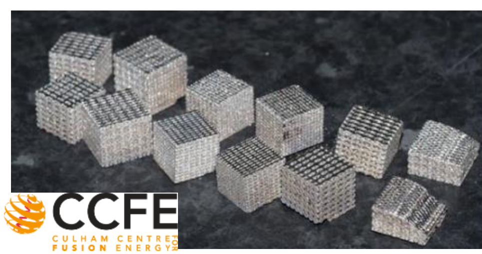
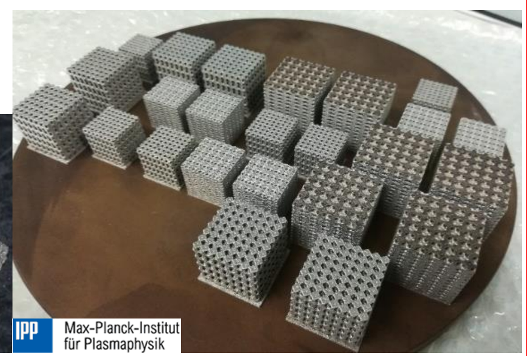


CCFE Culham Centre Fusion Energy

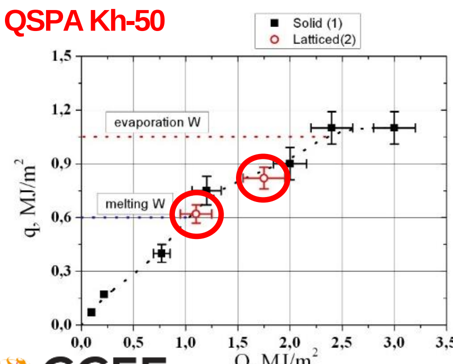


IPP Max-Planck-Institut für Plasmaphysik

Ongoing activities:

→ Different geometries samples production
→ Microscopic inspection
→ Material characterization (density, thermal diffusivity, mechanical testing)
→ Plasma compatibility and H-retention tests
→ **HHF experiments on linear plasma devices (QSPA Kh-50)**
→ **FEM-based tools for thermal simulation with melting / vaporization**

QSPA Kh-50



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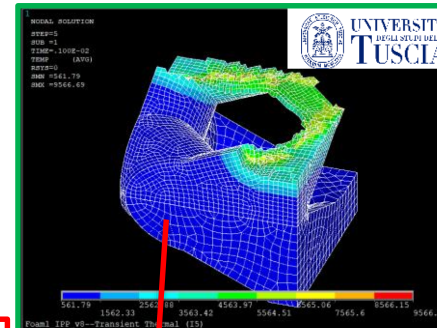
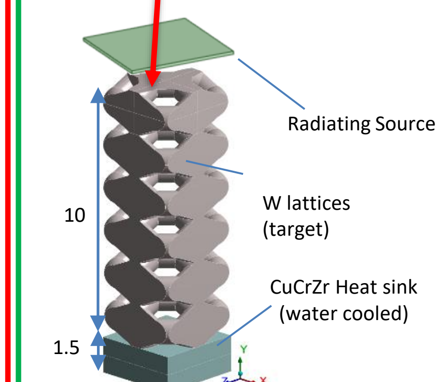
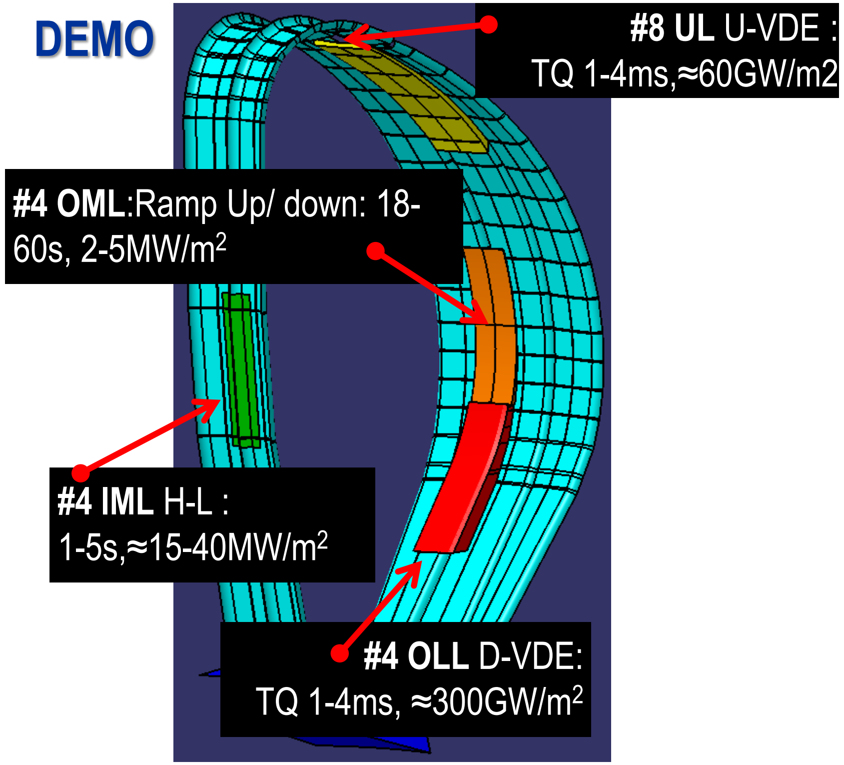
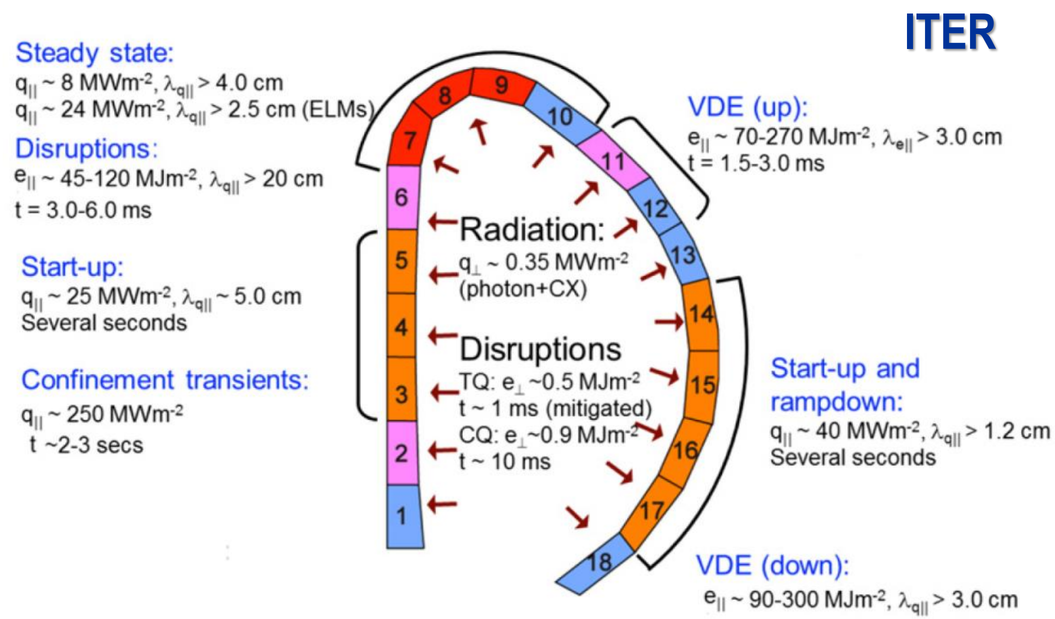




Limiters design

A 3D HF map is being created for DEMO, and will be kept up to date with new perturbation events, also experimentally based, similarly to ITER.

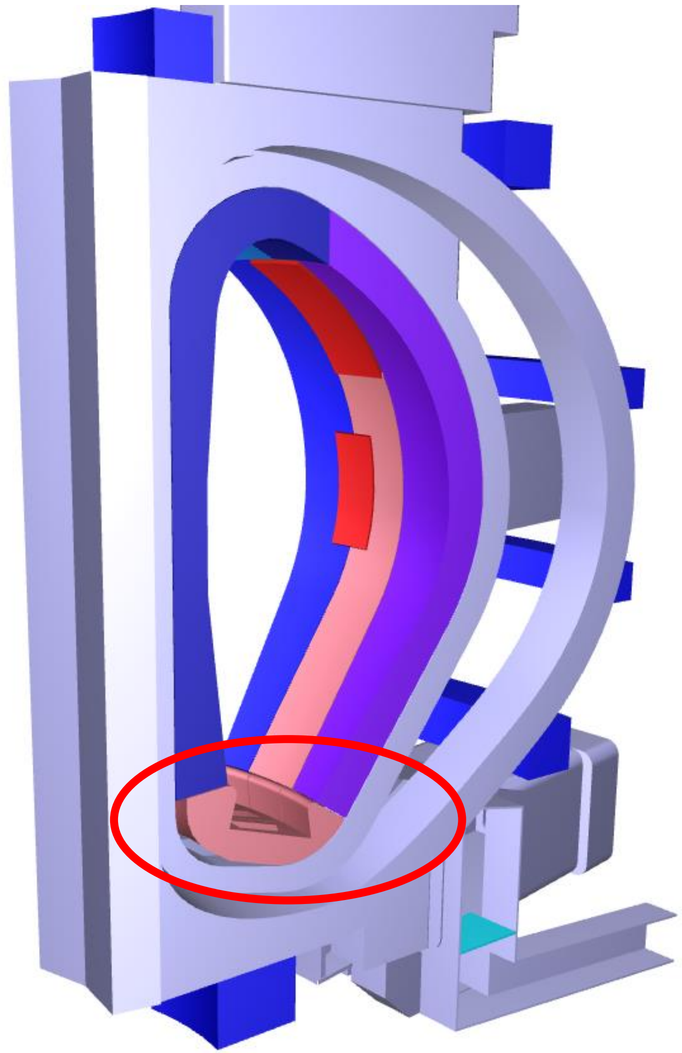




A.R. Raffray, NF 2014

The proposed limiters protects the BB FW in all the considered perturbations (evolving list).
Initial **vapor shielding** effects and REs simulation being performed.
Hardware **R&D** and testing proposed for PFC (*e.g.* lifetime duration, pipe protection).

Divertor Transient Heat loads

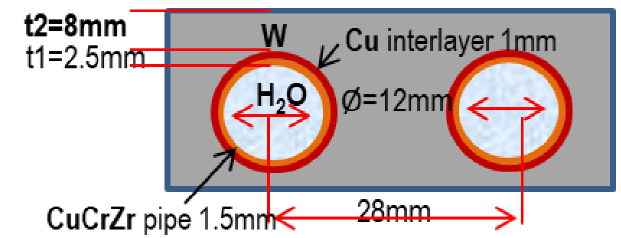


- ELMs
- Divertor reattachment

ELMs free regimes in DEMO

EU-DEMO estimated Type 1 ELMs *[Eich NME 2017]*

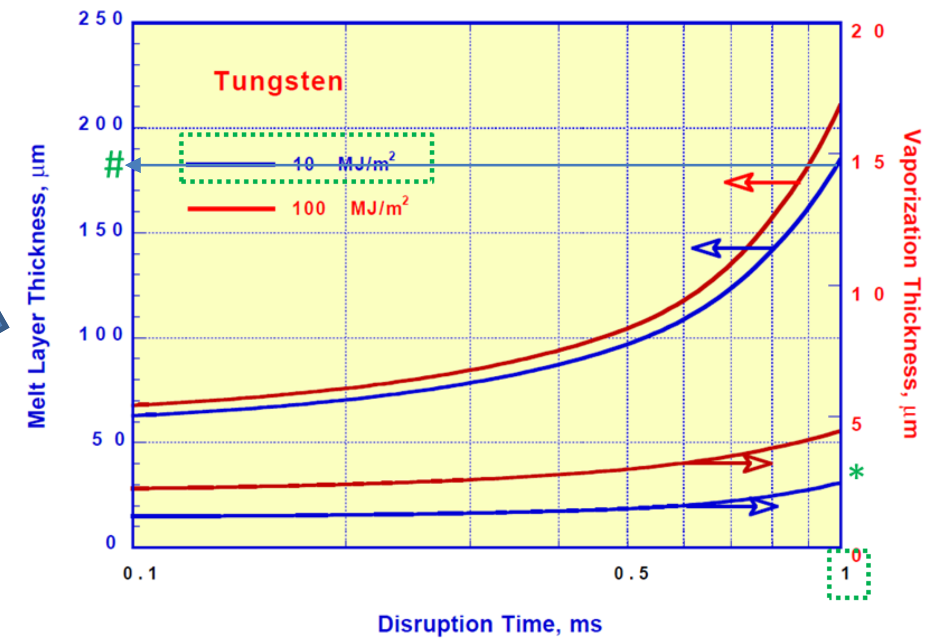
ELMs in DEMO - worst case	
Energy ELM - DeltaW/W = 0.2 [MJ]	**101.8**
tau_ELMs WORST CASE [ms]	1.0
Heat Load @Target [GW/m2]	9.2



Hassanein (2000) simulations including vapor shielding.

9.2GW/m², 1ms = 9.2MJ/m²

3µm(evap*)+182µm(melt#) = 185µm/event



Also, a single event is enough to reach melting temperature.

Once the complex mono-blocks armor features are lost, the problem worsen! *[J. P. Gunn NME 2021]*

The main strategy is to consider naturally ELM-free regimes as Priority EU-DEMO *[M. Siccinio, FED 2022]*

Divertor power exhaust in ITER and DEMO

Divertor Heat Loads is a machine design driver!

Heat flux design criteria (technologic limit):

- $10 MW/m^2$ steady state (order ~10^4 cycles).
- $20 MW/m^2$ transients for ~10s & ~1000 cycles.

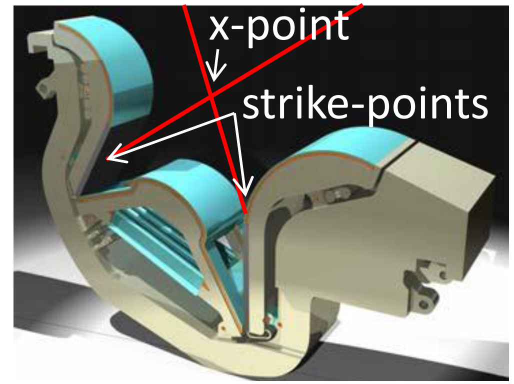
Successfully tested

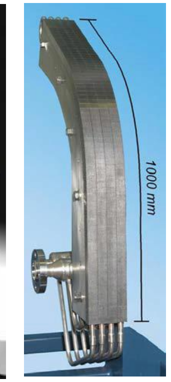
Technology R&D:

- Novel materials for heat sink & interlayer
- (e.g. W_f/Cu composite, W/Cu laminate)
- Mock-up fabrication, HHF tests & evaluation

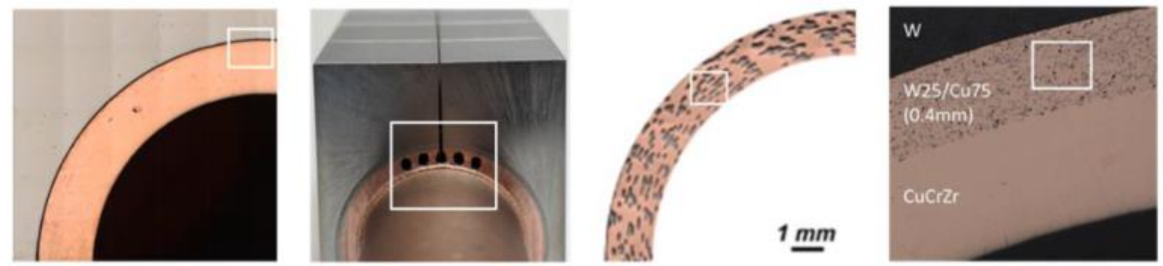
Divertor power exhaust strategies:

- Advanced Divertor Configurations
- **Detachment** (High radiation), **ELM mitigation**,…



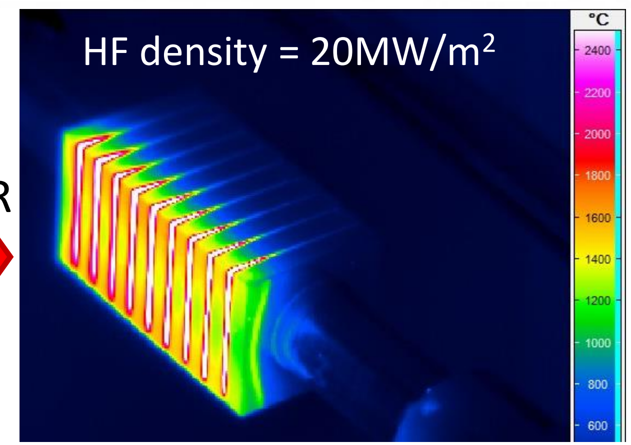


ITER-like (Cu interlayer) | Thermal break (bores, notch) | Composite pipe (W wire/Cu) | Graded interlayer (W particle/Cu)



J. H. You, FED 2022

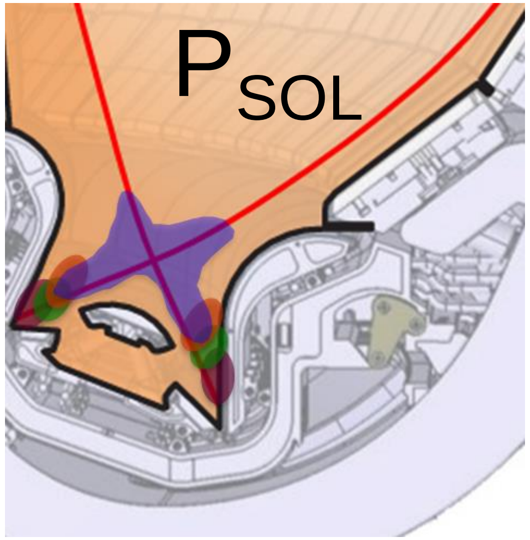
Tested in EU & inter. HHF facilities: e.g. IR termography



Full scale ITER div. casssette

SOFT2018

Divertor reattachment



[Figure: D. Whyte]

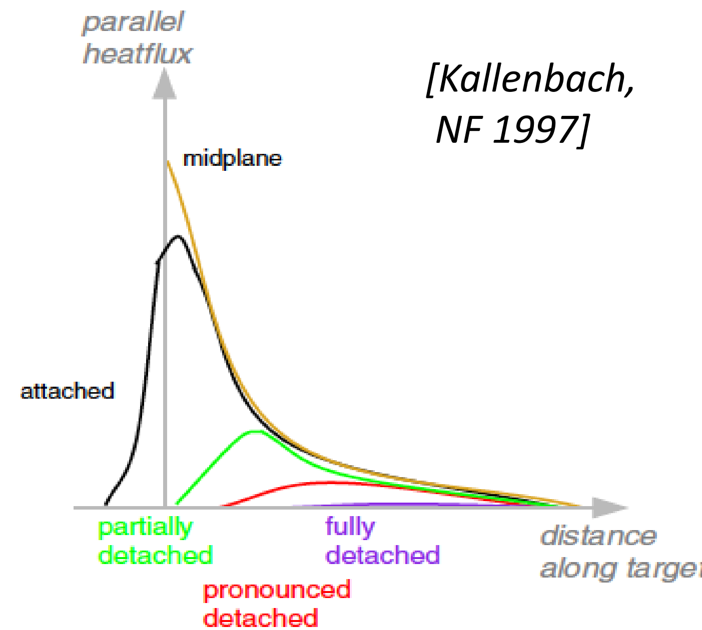
Heat conduction zone

Impurity radiation zone

$H^0/D^0/T^0$ ionization zone ($T_e > 5$ eV)

Neutral friction zone

Recombination zone ($T_e < 1$ eV) → detachment

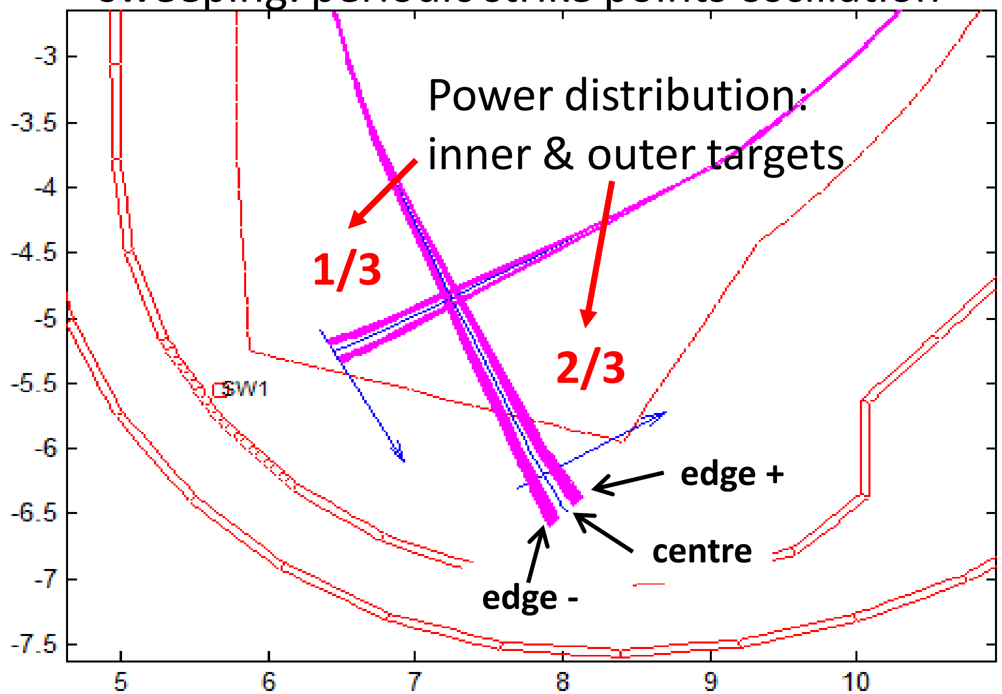


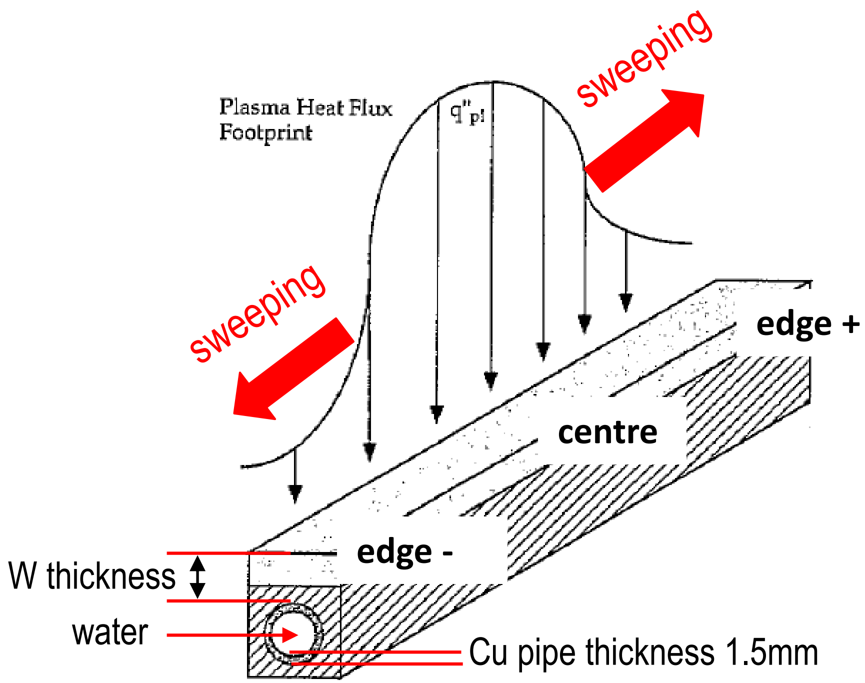
[Kallenbach, NF 1997]

- **The maximum tolerable steady-state HF** on the current DEMO SN divertor plate is **10-20 MW/m²** *[J. H. You FED 2022]*. This value can be achieved only in **detached** plasma conditions (with seeded impurities).
- **Detachment**: incoming plasma looses momentum and energy due to various mechanisms. A high neutral pressure in the divertor chamber is a necessary condition to achieve this state *[Kallenbach, NME 2019],[Pitts, NME 2019].*
- Simulations carried out with RAPTOR (with CREATE equilibria) show that the plasma current ***cannot be ramped down faster than ~0.1 MA/s*** without losing control: **In DEMO, there can be no fast plasma** termination (mitigated disruptions implications on the first wall are very severe).
- In case of **accidental divertor reattachment**, the divertor reaches burn-out (leading to an in-vessel LOCA) in **few seconds**.

DEMO strike point sweeping: loss of detachment mitigation technique

Strike point ***sweeping***: periodic strike points oscillation
Estimated heat flux re-attachment in DEMO up to >> divertor technological limits (10-20MW/m²)

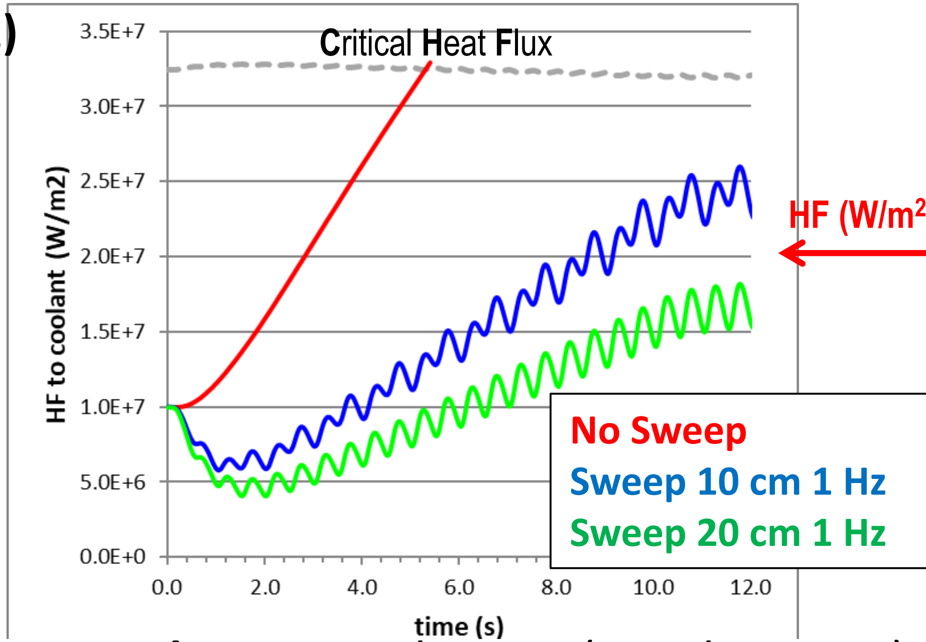


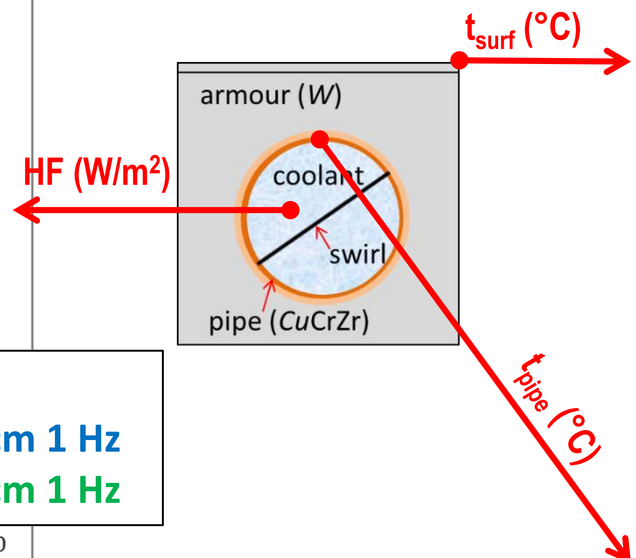


Thermal analysis with code RACLETTE

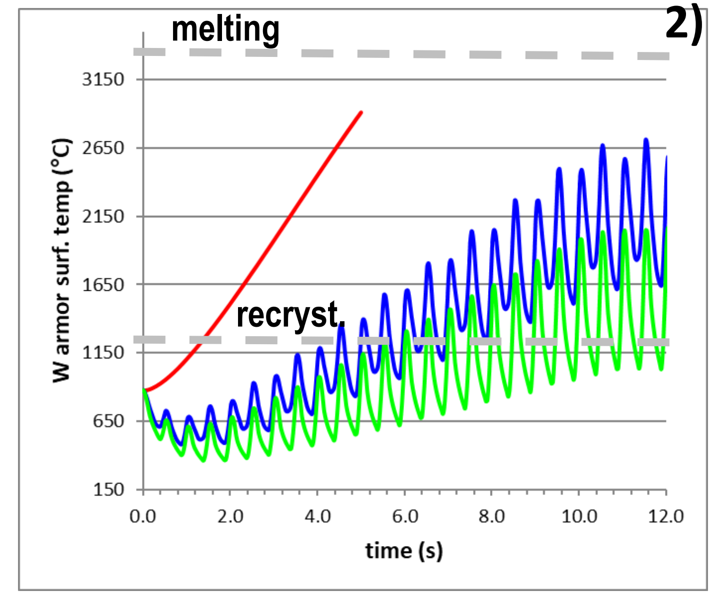
Results heat flux ramp from 10MW/m² to 70MW/m² in 10s:

1)

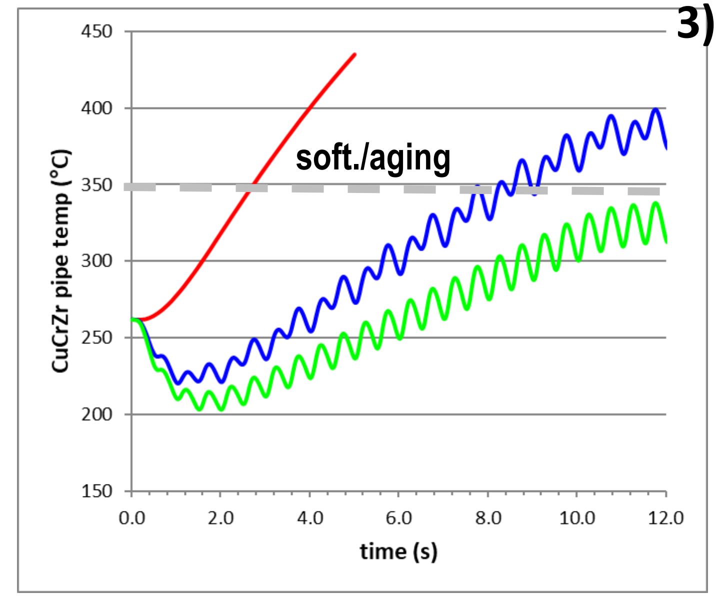




2)



3)



1) **HF to coolant**: In SS the CHF (pipe burn out) is reached in 5s, 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 ≈6s the CHF time, while in the sweeping cases it does not reach melting. In the 20cm-1Hz the temp. reaches 2000°C(> W-recrystallization: issue if kept too long).

3) **CuCrZr pipe temp.**: The pipe softening temperature of 350°C is reached in 2.7s in SS, and 8.6s in 10cm-1Hz seeping, while it is not reached for the 20cm-1Hz case.

Strategy for strike points sweeping control

Diagnostics

The use of sweeping would require the implementation of ***diagnostics*** able to detect reattachment promptly (<1s) (e.g. ***Spectroscopy+radiation, Thermography, thermo-currents***) *[Biel, FED 2022]*, to allow sweeping mitigation action.

Control

- ***Closed loop simulations show that In-Vessel coils (IVC)*** close to the strike points are needed to obtain the prescribed sweeping frequency (***1Hz***) and amplitude (***20cm p.p.***) with electrical power request $P_{el} < 5MW$.
- Standard PF coils could not guarantee the required performances due to vessel magnetic fields shielding effects *[R. Ambrosino, FED 2021]*.
- Integration of IVC in the divertor area is challenging!

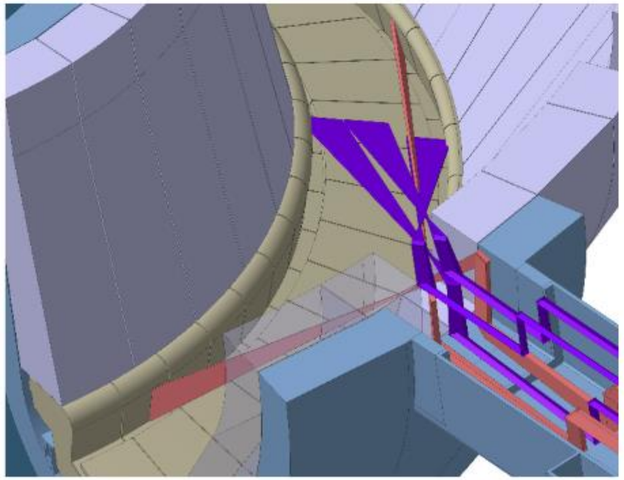
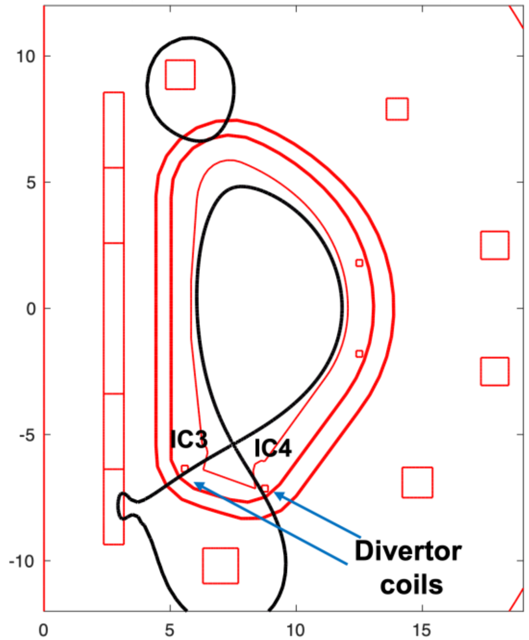


Fig. 6. Spectroscopic lines of sight for the divertor detachment control from an equatorial port plug.

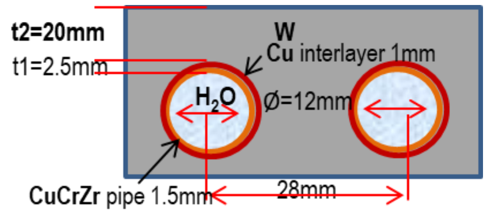


Conclusions

- DEMO power plants requirements are different from ITER (e.g. tritium self sufficiency, energy conversion, neutron resistant materials).
- Fusion machines are often designed for the flat top operation state. Plasma transients however can cause more severe load conditions and need to be considered too.
- A list as complete as possible of transient is being compiled, based on simulations, present machine extrapolations and ITER Specifications.
- Loads during plasma transients have a strong impact on key systems:
    - PFC fulfilling specific functions (e.g. sacrificial or normal operation limiters, divertor, breeding blanket – first wall),
    - Control systems, both in terms of Diagnostics (e.g. able to predict disruptions, loss of detachments), and actuators (e.g. the proposal to use of In-Vessel Coils for a faster response/lower control power, mitigation systems)
- The plasma operating scenario must be chosen to reduce the severity and the probability of the transient loads.

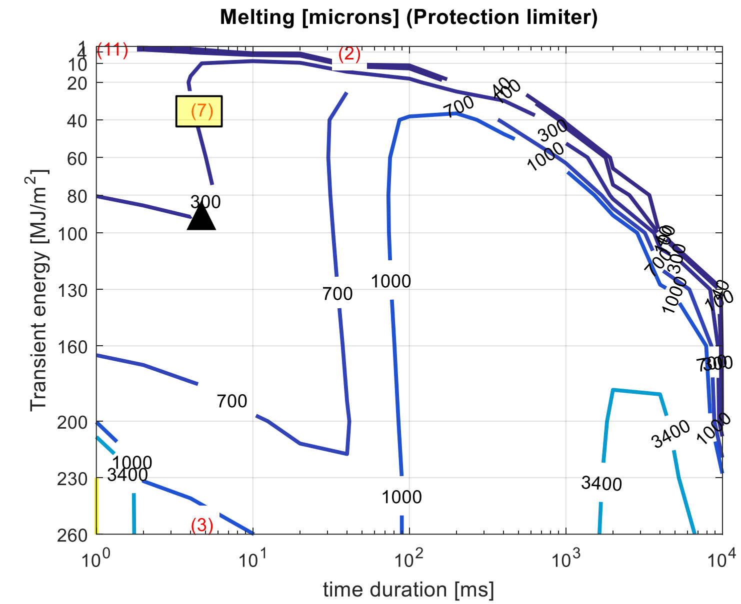
Thermal analysis scan to map different PFC technology limits and uncertainties

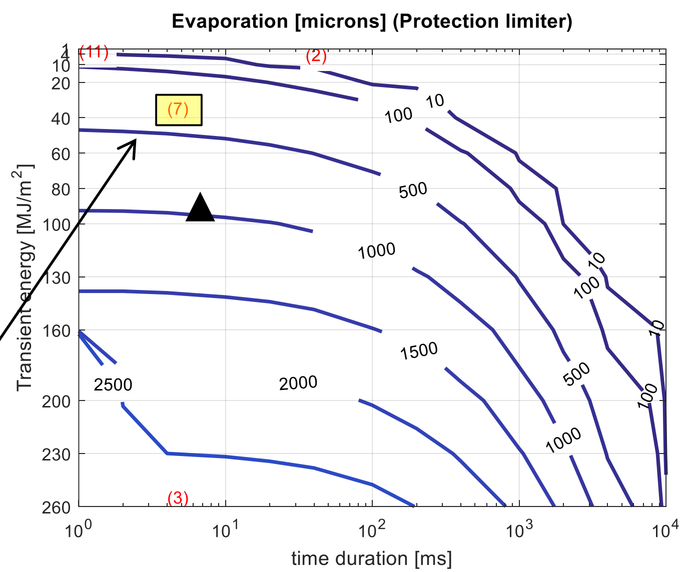
E([1-260MJ]MJ/m^2) **/ τ(**[1-10^4]ms) scan created for each PFC, to quickly assess vap./melt./temp./CHF



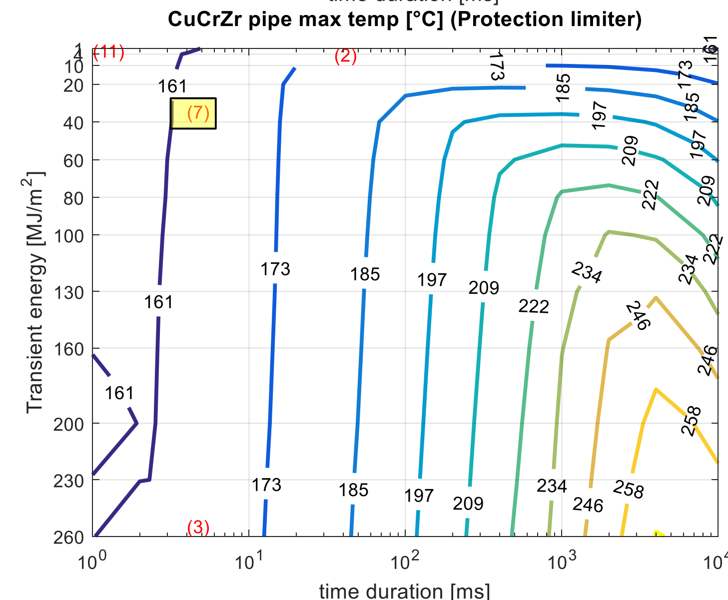
Example: Sacrificial limiter

Case	W-Evap. (µm)	W-Melt. (µm)	Surf. temp. (°C)	Heat sink temp. (°C)
Sacrificial limiter:				(CuCrZr heat sink temp. lim. 350°C)
U-VDE FT [2]	0	0	1670	173
U-VDE TQ [3]	2770	1084	7921	169
D-VDE TQ [4]	Not converged			
H-L [5]	15400	4246	5378	446
H-L [6]	5300	4484	5075	313
MD [7]	336	305	6695	168
Mitig. Disr. [11]	0	49	4437	168





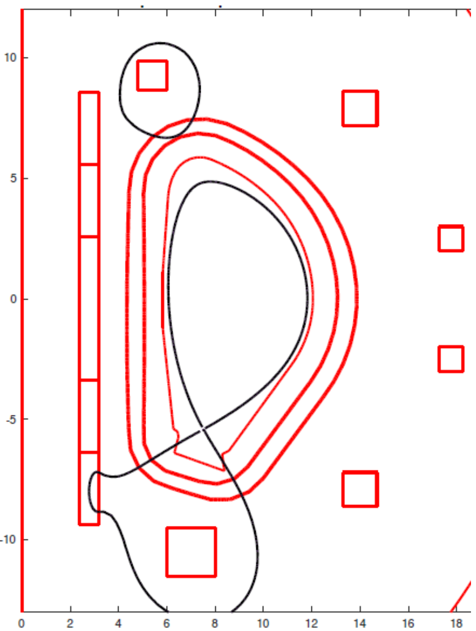
▲ -TOKES (no vapour shielding equivalent – 25GW/m^2 for 4ms = 100MJ/m^2), next page



RACLETTE is conservative when W vaporisation ≥tens µm: <u>possible</u> mitigation from vapour shielding

DEMO Static loads: Conservative – P_{sep} slow transient

E.g. baseline 2017
[*Siccinio FED 2022*]



I_{pl} = 19MA
β_p = 1.141
li = 0.8
R ≅8.94m
AR ≅ 3.1
k_{95} ≅ 1.65
δ_{95} ≅ 0.33
V_{pl} ≅ 2466m^3

Conservative, R. Wenninger, NF 2017

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

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

codes

tot_rad_frac	90%
P_alpha+P_aux	457 MW
P_sep,max	154 MW
P_rad,core	303 MW
P_rad,edge (70%)	108 MW
P_lambda_q=1mm (20%)	31 MW
P_lambda_q=50mm (10%)	15 MW

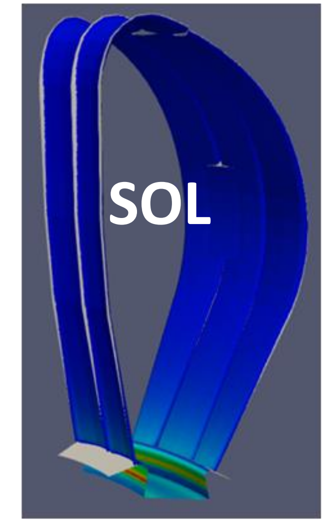
Cluster in Div → SOL

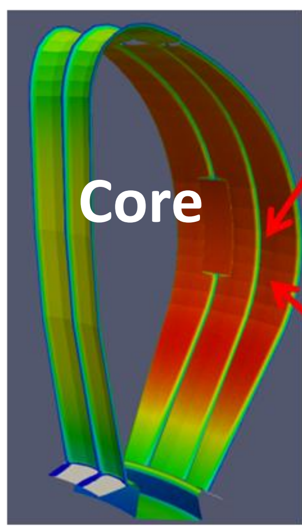
All Div (1mm OMP become ≈1cm at target)

95% Div, And 5% FW → Blobs

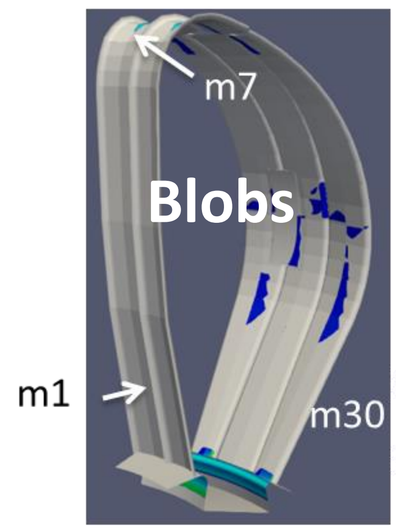
FW and Div → Core

EOF: ***CHERAB/SOLPS***

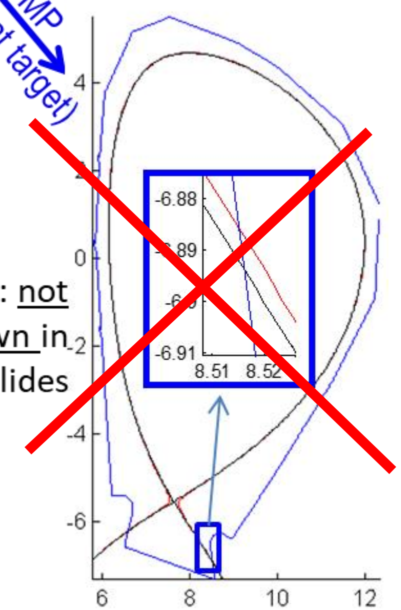




EOF: ***CHERAB/ASTRA***



SOLPS: not shown in next slides



Plasma scenario for EU-DEMO: status and plans

BACKUP SLIDE

M. Siccinio, et al., FED 2022

	EU-DEMO 2015	EU-DEMO 2017	EU-DEMO 2018	EU-DEMO (QH-mode)	EU-DEMO (I-mode)	**ITER**
R [m]	9.07	8.94	9.07	8.94	9.47	**6.2**
A	3.1	3.1	3.1	3.1	3.1	**3.1**
B_0 [T]	5.66	4.89	5.86	5.74	6.45	**5.3**
q_{95}	3.25	3	3.89	3.93	3.87	**3**
δ_{95}	0.33	0.33	0.33	0.33	0.33	**0.33**
κ_{95}	1.65	1.65	1.65	1.65	1.65	**1.7**
I_p [MA]	19.6	19.07	17.75	18.27	20.63	**15**
f_{NI}	0.44	0.5	0.39	0.52	0.219	**~0.2**
f_{CD}	0.10	0.11	> 0.05	0.16	>0.05	**> 0.1**
P_{fus} [MW]	2037	1998.3	2012	1871	1274	**500**
P_{sep} [MW]	154	156.4	170.4	178.5	240	**89**
P_{aux} [MW]	50	50	50	76	50	50
P_{CD}/P_{aux}	1	1	0	0	0	**0**
P_{LH} [MW]	121	107.5	120.8	N/A P_{LH} = 138 MW	N/A P_{LI} = 265 MW	**52**
H_{98}	1.1	1.1	0.98	0.89	0.8	**1**
$< n >/n_{GW}$	1.2	1.2	1.2	1.37	0.9	**~1**
$< T >$ [keV]	13.06	12.8	12.49	11.31	10.37	**8.9**
$n_{e,pt}$ [1e20m-3]	0.67	0.62	0.57	0.63	0.46	**~1**
$T_{e,pt}$ [keV]	5.5	5.5	3.7	4.6	2.7	**~3**
β_N [%mT/MA]	2.59	2.889	2.483	2.576	1.35	**1.8**
Z_{eff}	2.58	2.17	2.12	2.19	1.150	**1.78**
$P_{sep}B/q_{95}AR$ [MW T /m]	9.54	9.2	9.2	9.4	13.6	**8.2**
P_{sep}/R [MW/m]	17	17.5	18.9	19.8	25.34	**14.35**
Burn length [sec]	7200	7200	7200	7931	7200	**400**

Table 1. DEMO Physics Baseline 2017, 2018, 2019, QH-mode, I-mode relevant machine parameters and corresponding values for ITER. DEMO data have been produced with the systems code PROCESS. The parameter f_{NI} represents the sum of the driven current fraction f_{CD} and of the bootstrap current fraction..The subscript "pt" indicates quantities at the pedestal top. Cells containing values fixed by input in PROCESS are highlighted in blue (color online). Note that not all baselines have been built with the same input parameter set.

EU-DEMO – T. Eich scaling type 1 ELMs

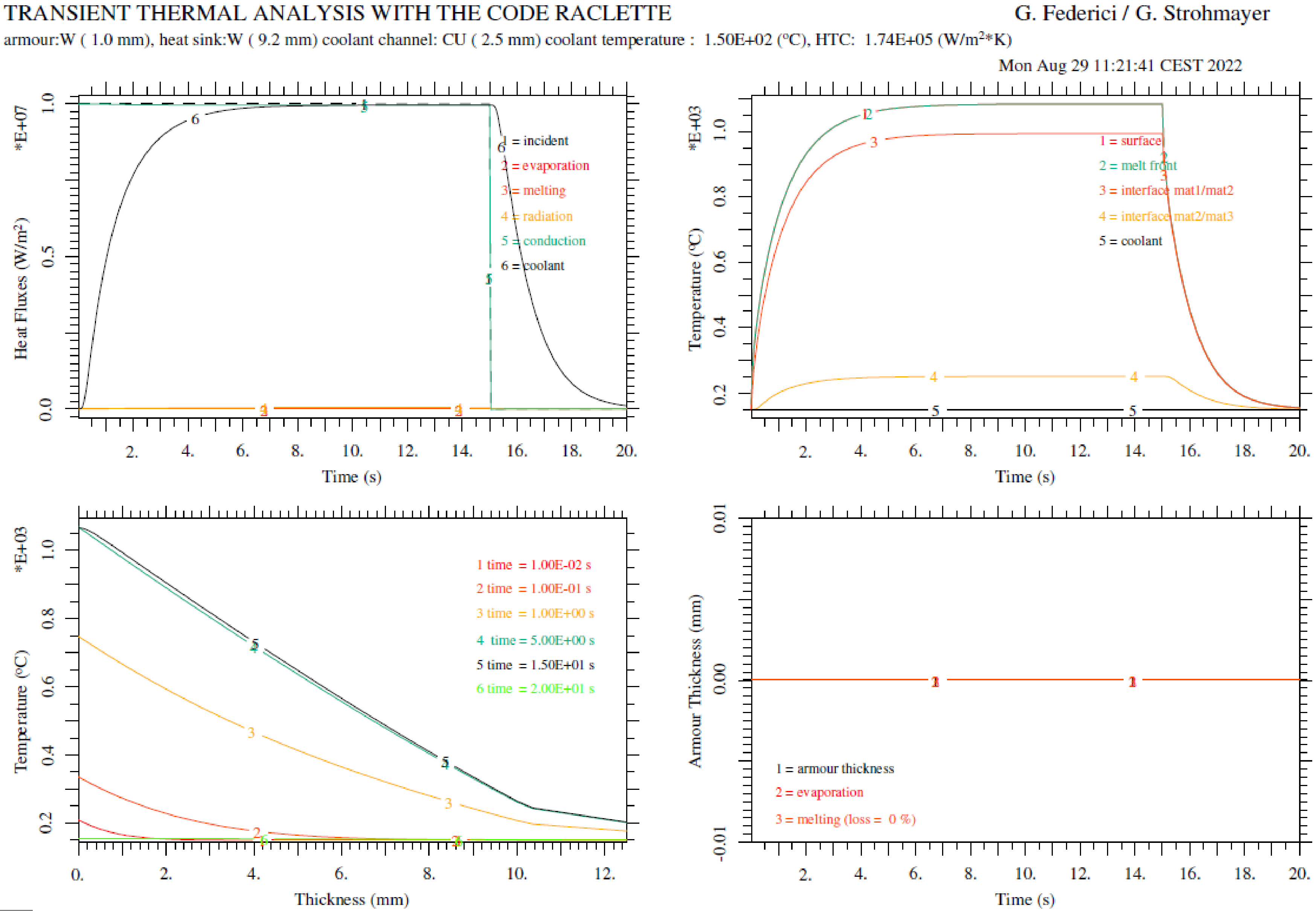
BACKUP SLIDE

$$\varepsilon_{||} = 0.28 \pm 0.14 \frac{MJ}{m^2} \times n_{e,ped}^{0.75\pm0.15} \times T_{e,ped}^{0.98\pm0.1} \times \Delta E_{ELM}^{0.52\pm0.16} \times R_{geo}^{1\pm0.4}$$

4		**DEMO SN**
5	**ELMs**	
6	nav [1e20m-3]	0.7
7	Tav [keV]	12.5
8	Volume [m3]	2924.0
9	Plasma Energy [MJ]	1272.2
10	**W Pedestal - 10% W plasma [MJ]**	**127.2**
11	**Energy ELM - DeltaW/W = 0.05**	**25.4**
12	**Energy ELM - DeltaW/W = 0.2**	**6.4**
13	**W Pedestal - 40% W plasma [MJ]**	**508.9**
14	**Energy ELM - DeltaW/W = 0.05**	**25.4**
15	**Energy ELM - DeltaW/W = 0.2**	**101.8**
16	**Eich Scaling**	https://doi.org/10.1016/j.nme.2017.04.014
17	R_Target [m]	8.5
18	nped [1e20m-3]	0.6
19	Tped [keV]	3.7
20	Target Inclination [deg]	30
21	E_ELMs/E_Ped BEST CASE [%]	1.0
22	E_ELMs/E_Ped WORST CASE [%]	10.0
23	tau_ELMs BEST CASE [ms]	3.0
24	tau_ELMs WORST CASE [ms]	1.0
25	WELMs BEST CASE - parallel [MJ/m2]	5.6
26	WELMs WORST CASE - parallel [MJ/m2]	18.4
27	Heat Load ELMs BEST CASE - parallel [MW/m2]	1854.5
28	Heat Load ELMs WORST CASE - parallel [MW/m2]	18422.8
29	**WELMs @Target - BEST CASE [MJ/m2]**	**2.8**
30	**WELMs @Target - WORST CASE [MJ/m2]**	**9.2**
31	**# of tolerable MIN ELMs (approx.)**	**170**
32	**# of tolerable MAX ELMs (approx.)**	**40**

EU-DEMO Div sim steady state

BACKUP SLIDE



2	mm	MW/m**2	s	microns	microns	grad C	grad C	grad C	grad C	grad C
3	1.0	10.0	15	7.27E-21	0.0	1084.4	1084.4	994.2	251.7	202.9

Mitigated disruption simulation: TQ

BACKUP SLIDE

$$\eta = W_{th}/A_{eff}/\sqrt{t}$$

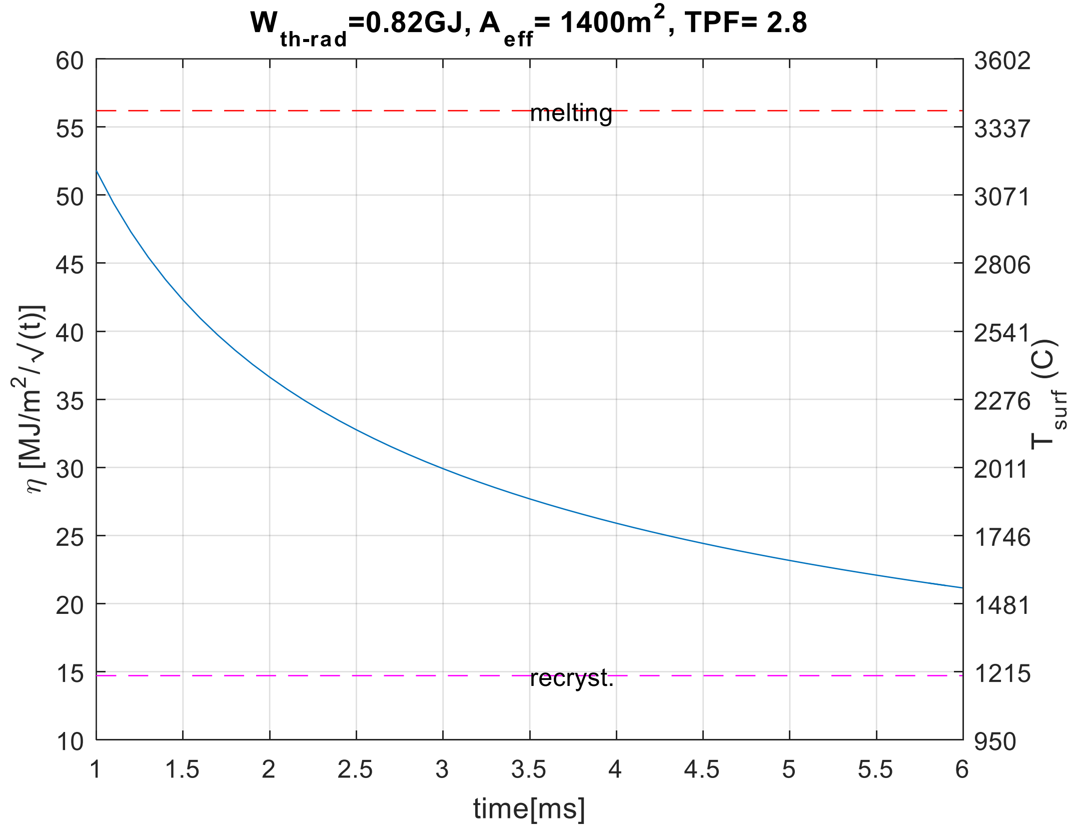
$$\eta_{crit} = (T_{melt} - T_{op})\sqrt{\pi\lambda\rho c/4}$$

$$T_{surf} = T_{op} + \eta_{crit}/\sqrt{\pi\lambda\rho c/4}$$

Data for tungsten:

λ	=	170	W/mK
ρ	=	19300	kg/m^3
c	=	138	J/kgK

From **W. Biel**



ITER Worst radiation case <u>conservative</u>: pre-TQ 30% W_{th} radiated, TQ 90% radiated -> $W_{th\text{-}rad}$=1.3GJ*0.7*0.9= 0.82GJ

With Toridal/Poloidal peaking factor (TPF) ≈ 3 the W-FW is close to melting temperature

This needs to be considered in the mitigation strategy choice (limiters not effective with radiation)

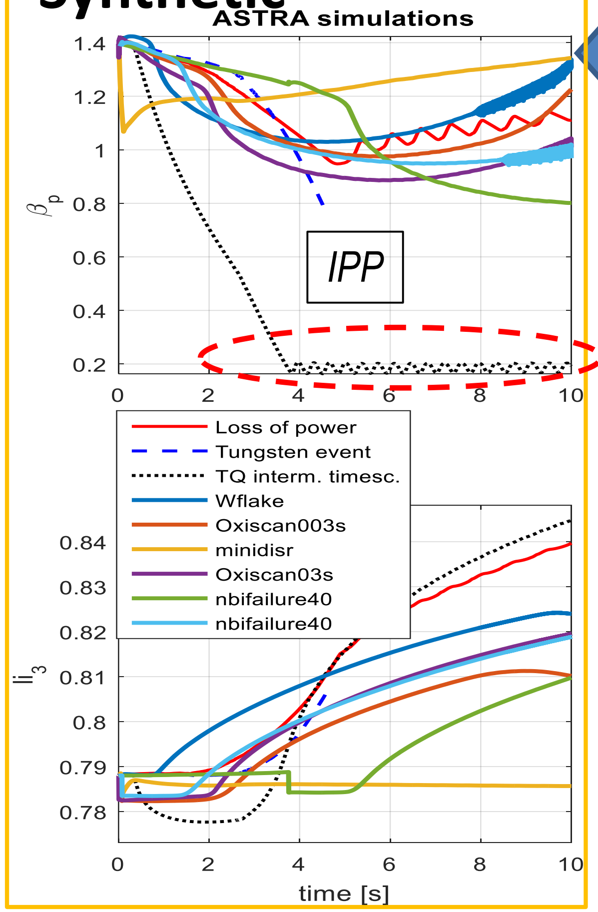
Plasma transient identification

BACKUP SLIDE

Several activities launched to predict possible contact points:

- Transport simulations to evaluate plasma perturbations (***$\Delta\beta_{pol}$, Δl_i, ΔI_p***). Integrated control, see E. Fable
- Inter-machine perturbation database: JET, ASDEX, EAST, TCV *[G. Sias, NF 2022]*
- ITER **H**eat and **N**uclear **L**oad **S**pecifications: *e.g.* U/D-VDE, unmitigated/mitigated disruptions

Synthetic



Synthetic (ASTRA) database, perturbations generated for:

- Loss of confinement
- ntm—like
- W influx
- H_2O influx
- ELM like
- Minor disruption
- TQ intermediate timescale (conservative)

Experimental database,

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

Experimental