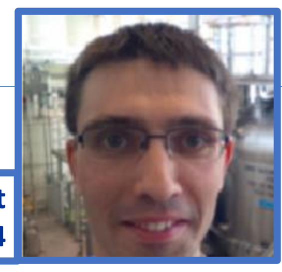
Analysis of the thermal-hydraulic effects of a plasma disruption on the DTT TF magnets



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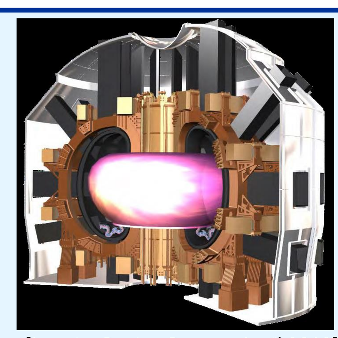
Introduction

The fully superconductive Divertor Tokamak Test (DTT) facility, currently under construction in Italy, will test several DEMO-relevant divertor solutions



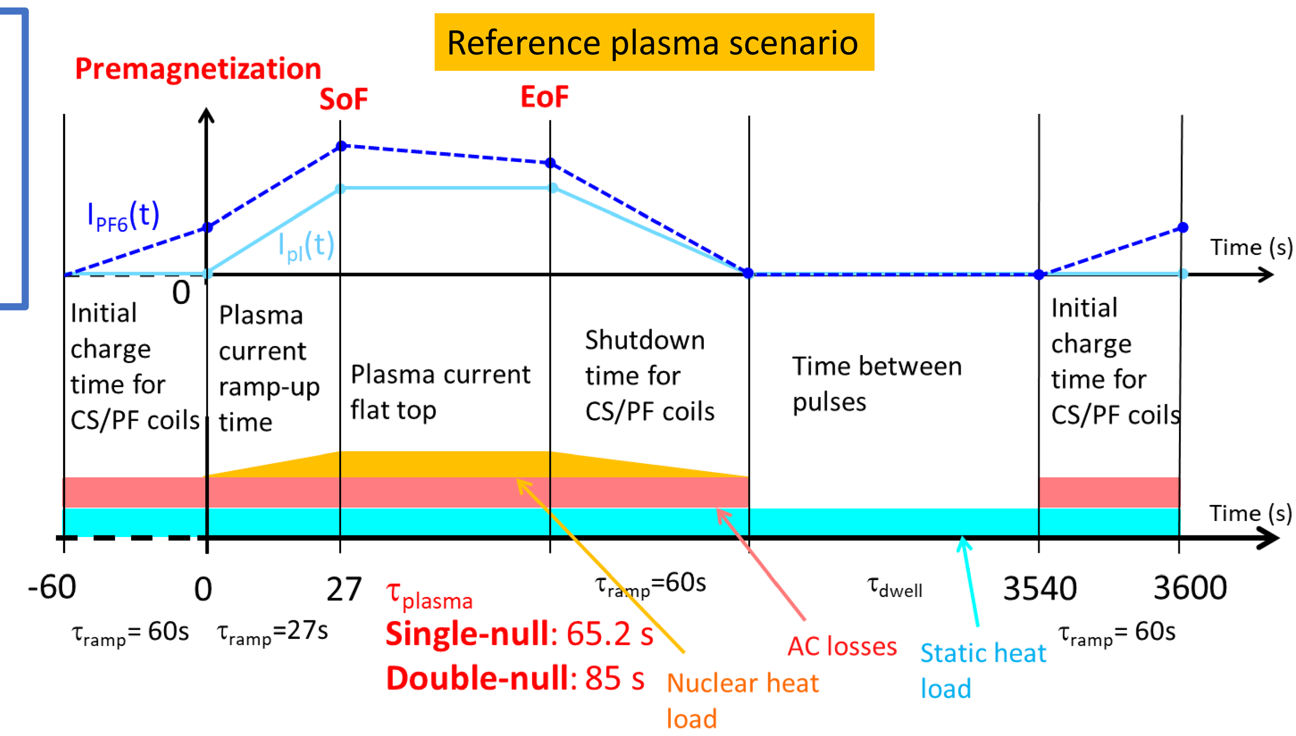
[Interim Design Report, April 2019]

- Major radius R_0 = 2.19 m
- Toroidal field on axis = 6 T
- Plasma current = 5.5 MA

[R. Albanese, et al., Design review for the Italian Divertor Tokamak Test facility, Fusion Eng. Des. 146 (2019) 194–197.]

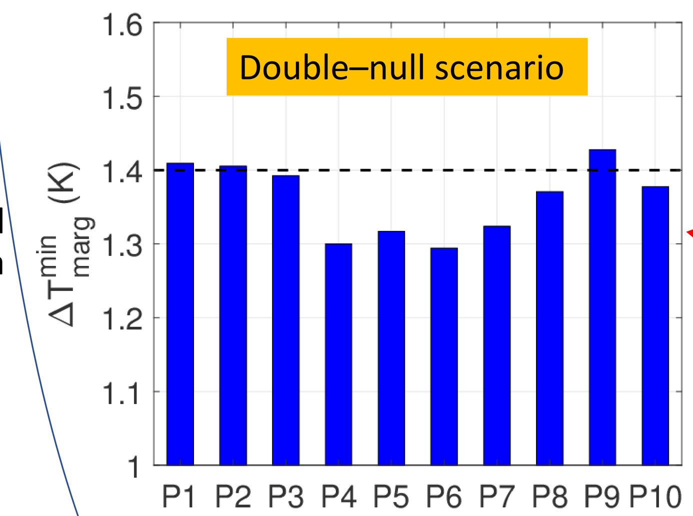
Aim of the work

Assess the thermal-hydraulic effects of a plasma disruption on the temperature margin of the DTT TF magnets during normal operation, with the 4C code

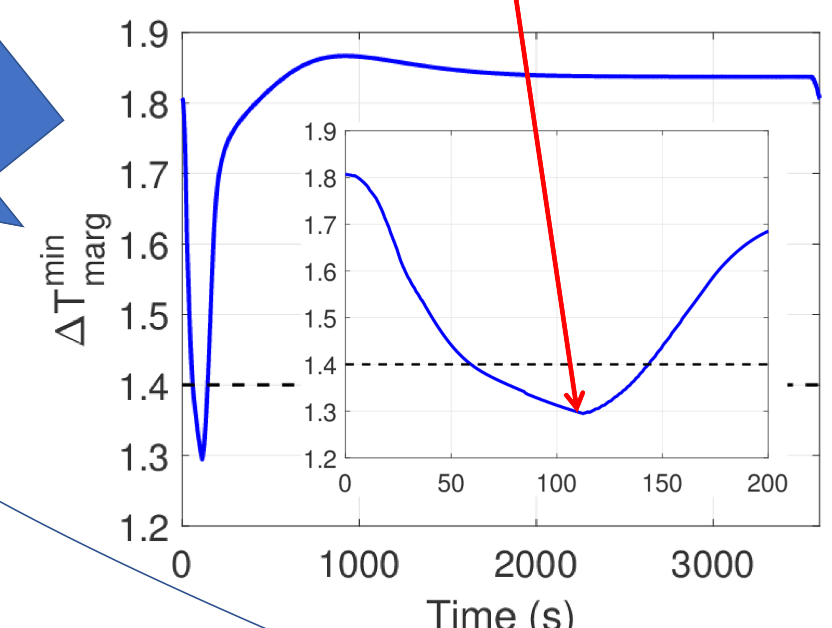


Only Double-Null scenario considered here (longer plasma duration → lowest ΔT_{marg})

ΔT margin in normal operation



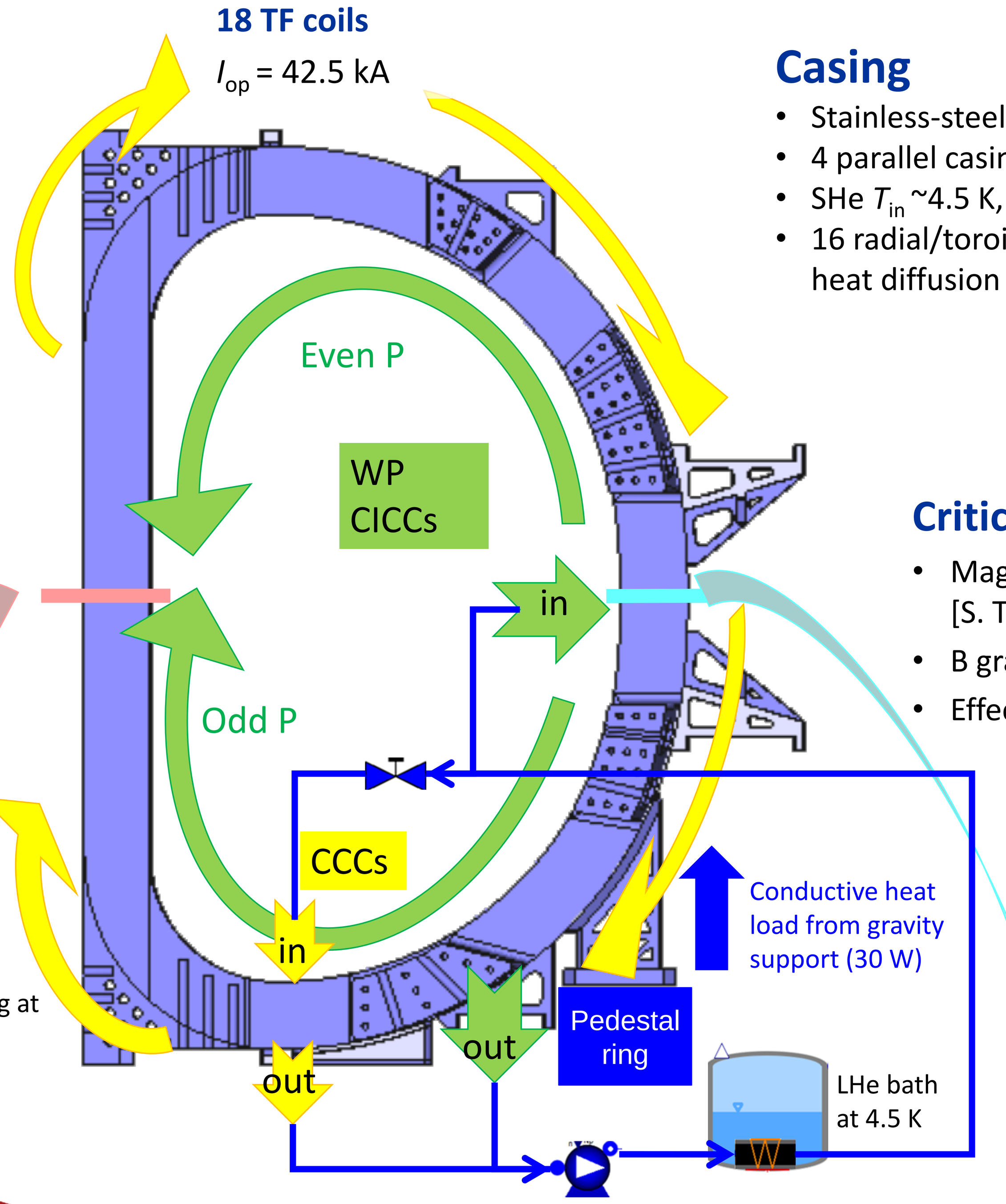
- The magnetic field peak is located at equatorial inboard section, at the WP center in toroidal direction → ΔT_{marg}^{min} is located in the central DPs (P6)
- The ΔT_{marg}^{min} is close to the target value (1.4 K) within ±0.1 K
- ΔT_{marg}^{min} is reached at End of plasma Flat-top (EoF)



4C code model

18 TF coils

I_{op} = 42.5 kA



Winding Pack (WP)

- Nb_3Sn CICC
- (Double) pancake-wound
- 10 hydraulic channels/coil (1D model)
- SHe T_{in} ~4.5 K, p_{in} ~6 bar, Δp ~2 bar

WP detachment from casing at plasma side (→ no thermal contact) when the coil is charged

Casing

- Stainless-steel
- 4 parallel casing cooling channels (CCCs)
- SHe T_{in} ~4.5 K, p_{in} ~6 bar
- 16 radial/toroidal cross sections (2D heat diffusion model)

Critical current calculation

- Magnetic field due to PF/CS/plasma accounted for [S. Turtù, 2019], as in [Bonifetto et al., FED 2021]
- B gradient on the CICC cross section accounted for
- Effective strain –0.65%

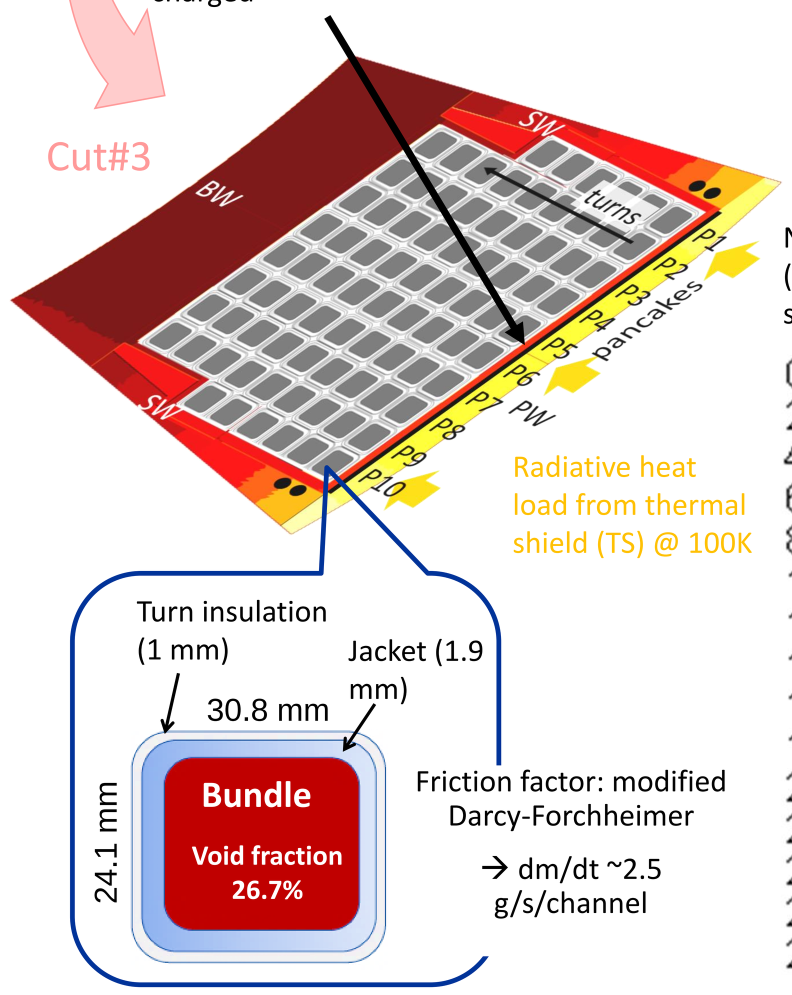
SHe circuit

Preliminary model of

- Cold circulator with realistic characteristic
- Heat transfer with LHe bath at prescribed T
- Piping and valves

in order to provide self-consistent boundary conditions to the magnet (circuit suitably scaled-down to account for a single TF)

Cut#3



Friction factor: modified Darcy-Forchheimer

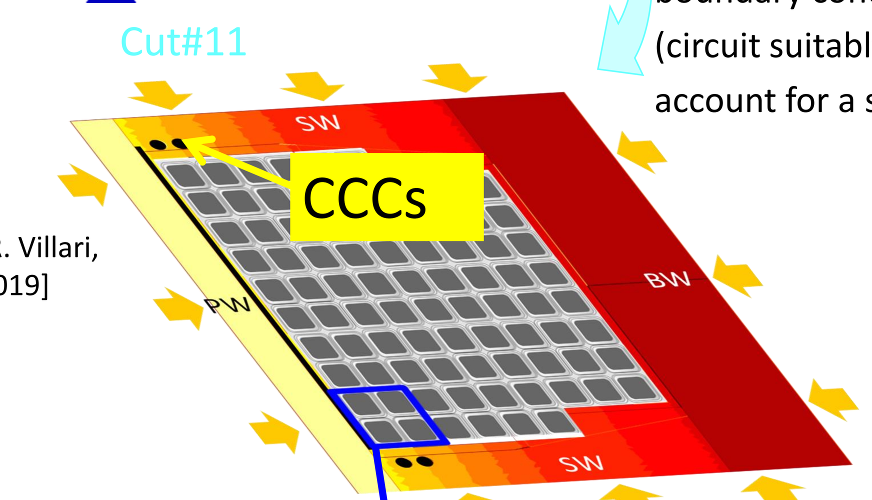
→ dm/dt ~2.5 g/s/channel

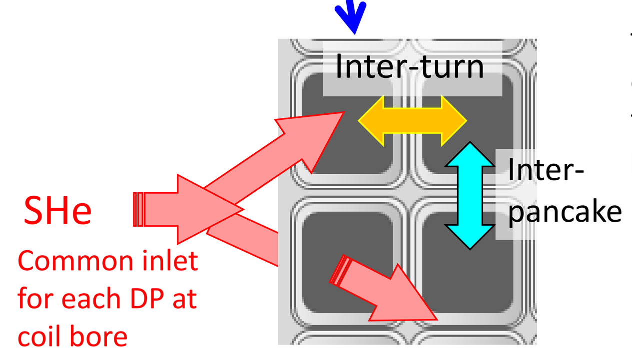
Nuclear heat load (without neutron shield) [W/m³]

[R. Villari, 2019]

0	0
200	100
400	200
600	300
800	400
1000	500
1200	600
1400	700
1600	800
1800	900
2000	1000
2200	1100
2400	1200
2600	1300
2800	1400

Cut#11



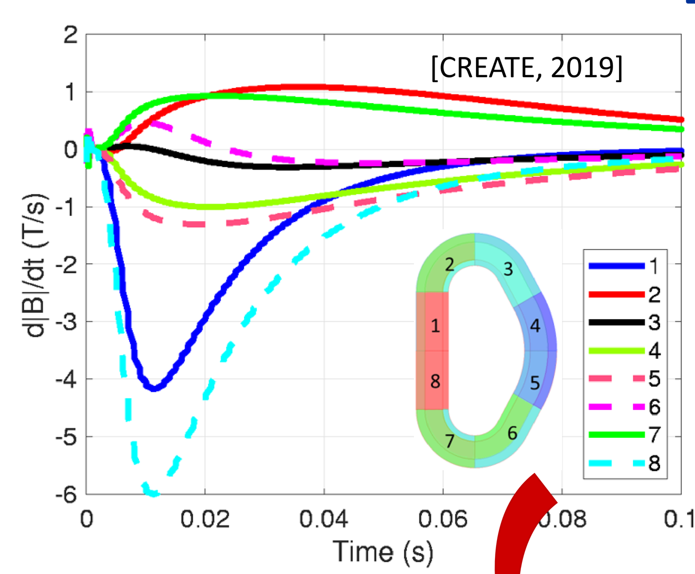


SHe
Common inlet for each DP at coil bore

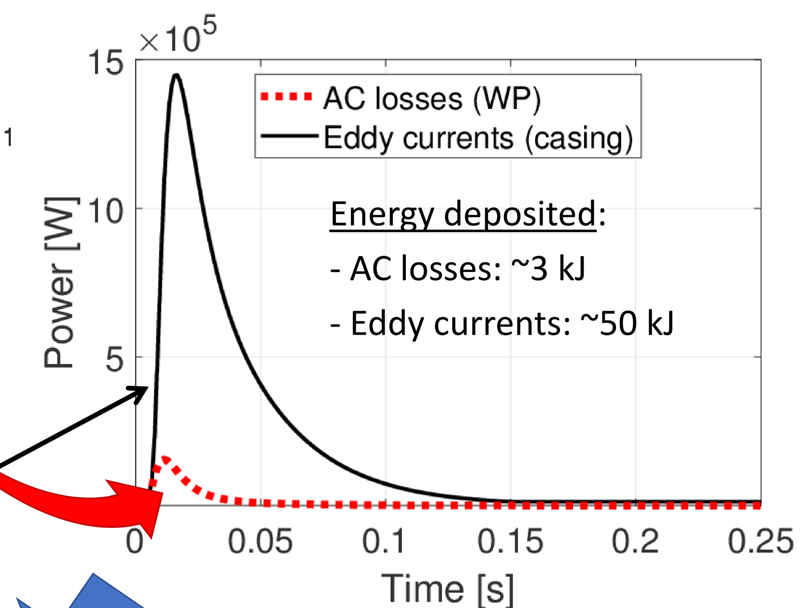
Turn-by-turn counter-current He flow in adjacent pancakes

Plasma disruption conservatively induced at EoF

Driver

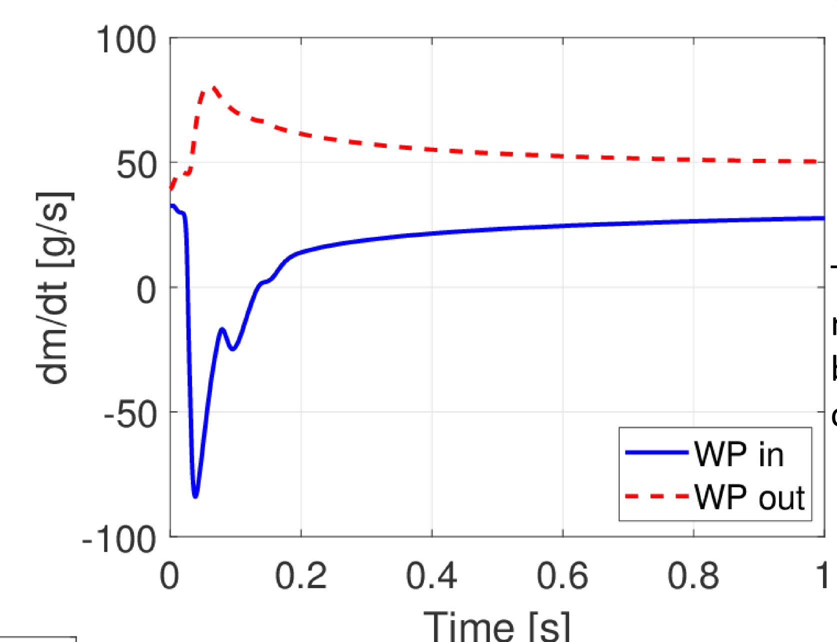


- Transport current of all coils (and the magnetic field) is assumed constant (as that at EoB) during the disruption
- Magnetic field variation during the disruption induces AC losses in the conductor ($n\tau$ = 250 ms)
- Eddy currents in the casing computed by EM model

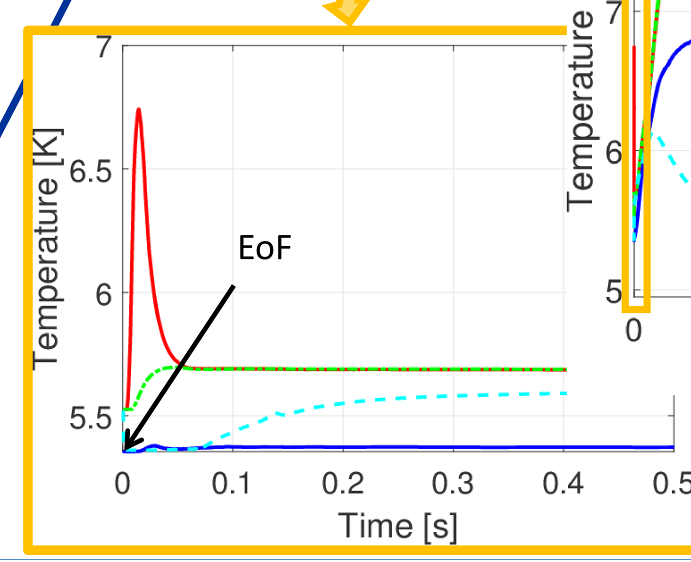


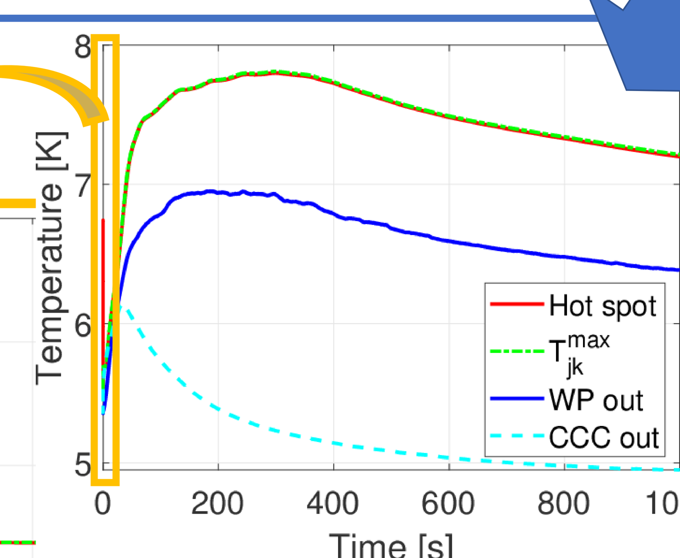
Energy deposited:

- AC losses: ~3 kJ
- Eddy currents: ~50 kJ



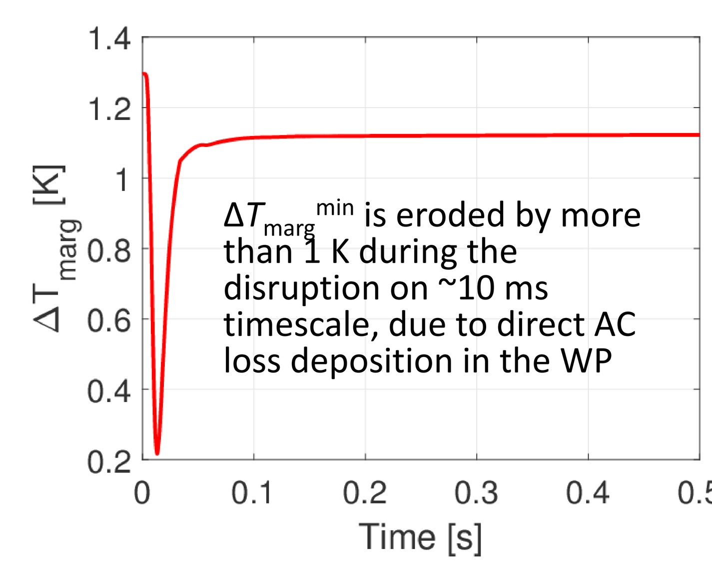
The mass flow rate reacts with a strong backflow at WP inlet on the 10 ms timescale





Power deposition heats-up:

- directly the WP on ~10 ms timescale
- the casing → heat is transferred to WP on ~10 s timescale (no effect on the ΔT_{marg}^{min} because involves outermost turns, where B is lower (and T_{cs} is higher)



ΔT_{marg}^{min} is eroded by more than 1 K during the disruption on ~10 ms timescale, due to direct AC loss deposition in the WP

Conclusions and perspective

Thermal-hydraulic effects of a plasma disruption at EoF in DTT TF coils analyzed with the 4C code:

- almost the entire temperature margin is eroded by the power deposited during the disruption at EoF → the coil is close to current sharing

In perspective:

- assess the impact of input uncertainties (*nt*, eddy currents, …) and of a fast current discharge following the plasma disruption