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Testing tungsten Plasma Facing Components in WEST and AUG tokamaks : lessons for ITER

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* E. Joffrin et al 2024 Nucl. Fusion 64 112019 https://doi.org/10.1088/1741-4326/ad2be4
** http://west.cea.fr/WESTteam
*** H. Zohm et al 2024 Nucl. Fusion 64 112001 https://doi.org/10.1088/1741-4326/ad249d

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Need for tungsten PFC testing in tokamak condition

Next step fusion devices will face unprecedented heat loads and particle fluence:

→ **ITER divertor** has to survive > 10 years, 2000 hours of cumulated plasma time *[R. Pitts et al., NME 2019]*

→ **New ITER baseline:** Beryllium → **Tungsten** in first wall *[A. Loarte, TEC-2-3 Monday]*

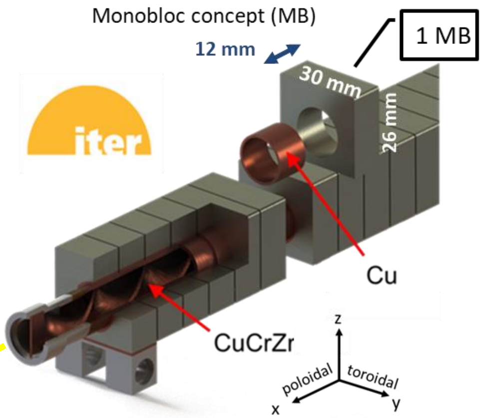
Two full-W tokamak devices in EU to address ITER urgent R&D:

Tungsten (W) Environment Steady-state Tokamak:

- Superconducting toroidal field coils, actively cooled
- Long pulse record with LHCD: 22 min. ← *[R. Dumont, EX-3-4 Tuesday]*
- Actively cooled tungsten ITER-grade lower divertor

15960 W blocks (5% of the ITER divertor)

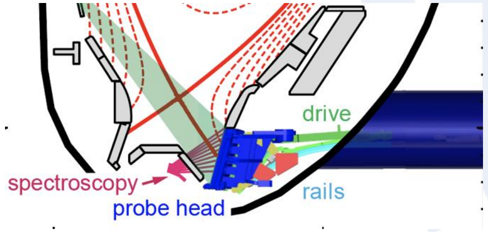
ITER technology:

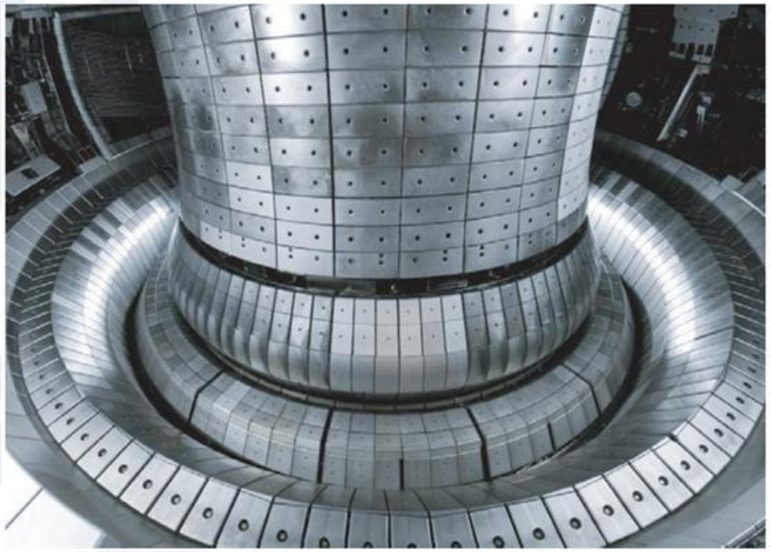


[J. Bucalossi, OV-2-4 Monday]

ASDEX Up-Grade:

- Full W device (upper and lower W-divertor)
- Divertor (DIM) and midplane manipulators (MEM)



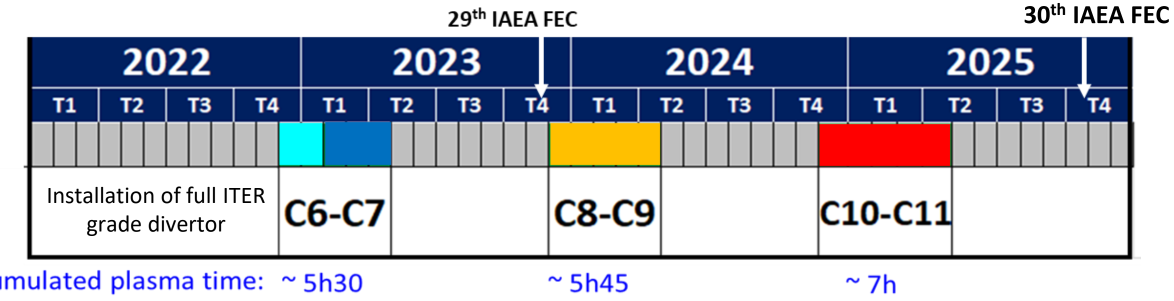


[T. Pütterich, OV- 3-2 Monday]

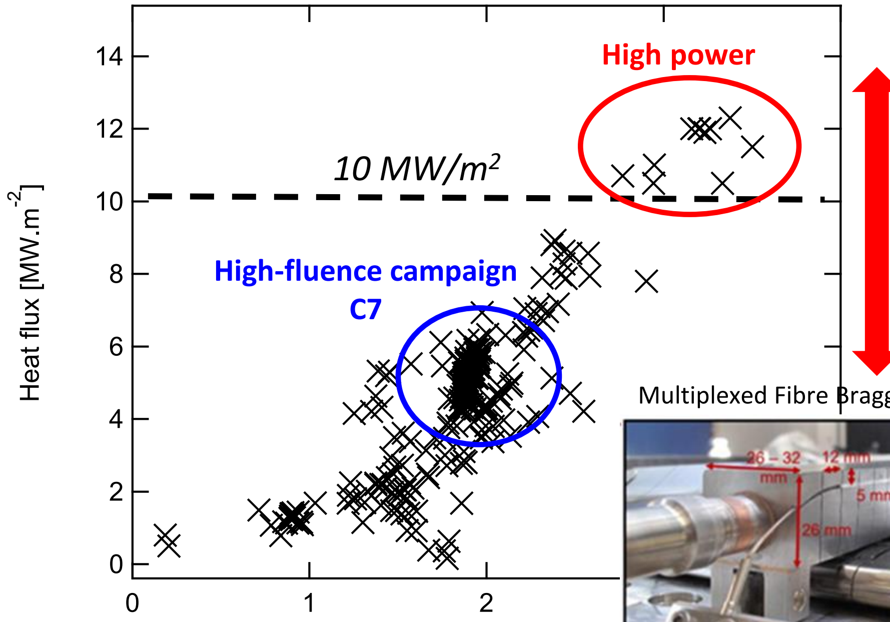
Outlines

- Introduction
- Assessing heat loads on castellated and shaped components
- Understanding damages : W-cracking and melting
- Summary and prospects

WEST phase 2 operation: ITER relevant ion-fluence and heat flux achieved

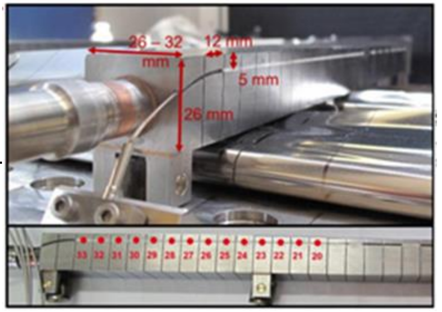


Steady-state heat flux computed with FBG sensing probes



[Y. Anquetin et al., NME 2024]

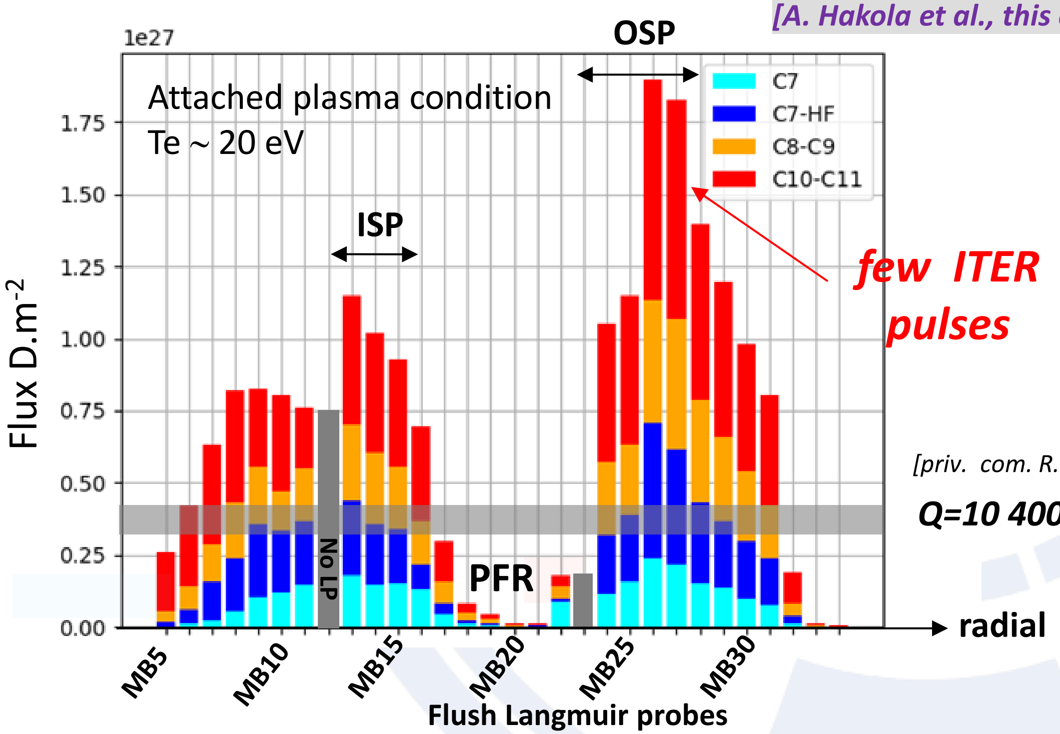
Multiplexed Fibre Bragg Gratings (FBG)



[N. Chanet et al. FED 2021]

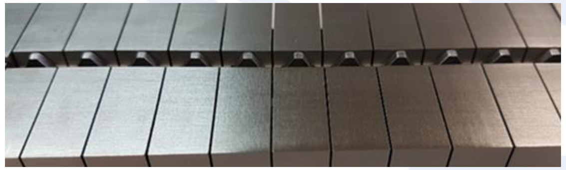
- > 4000 plasma performed / 2400 disruptions
- > 18 hours total plasma cumulated time
- Total cumulated deuterium fluence: 1.8×10^{27} $D.m^{-2}$

[A. Hakola et al., this conference]



[priv. com. R. Pitts and A. Pshenov]

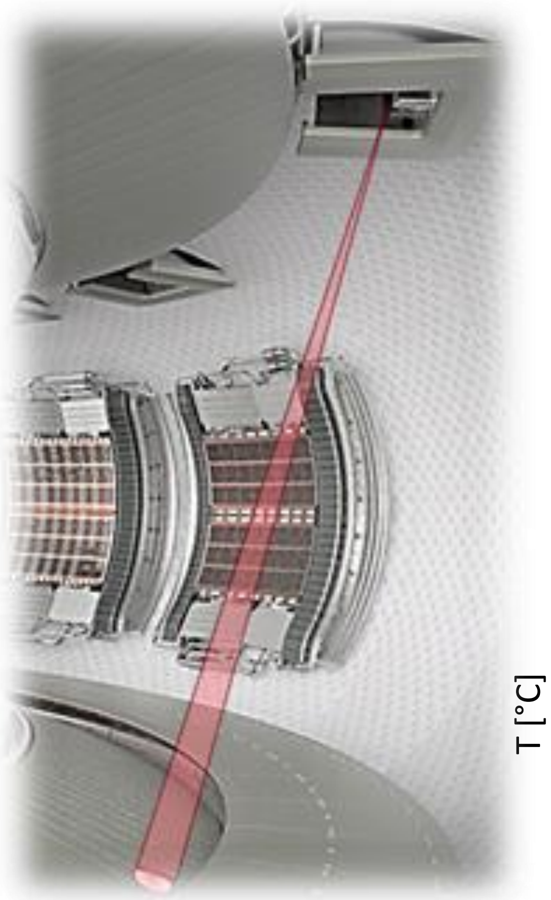
Q=10 400s ITER pulse



$$Fluence = \int \Gamma_i \, dt$$

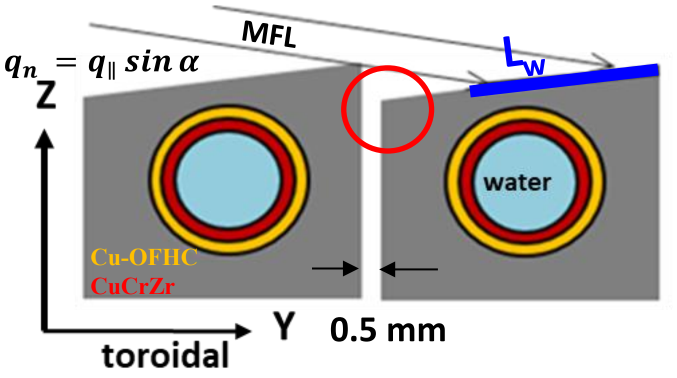
Very high spatial IR thermography to assess the toroidal bevel @ MB scale

Very High Spatial Resolution (VHR) IR camera 0.1mm/pixel



[M. Houry et al. NF 2024]

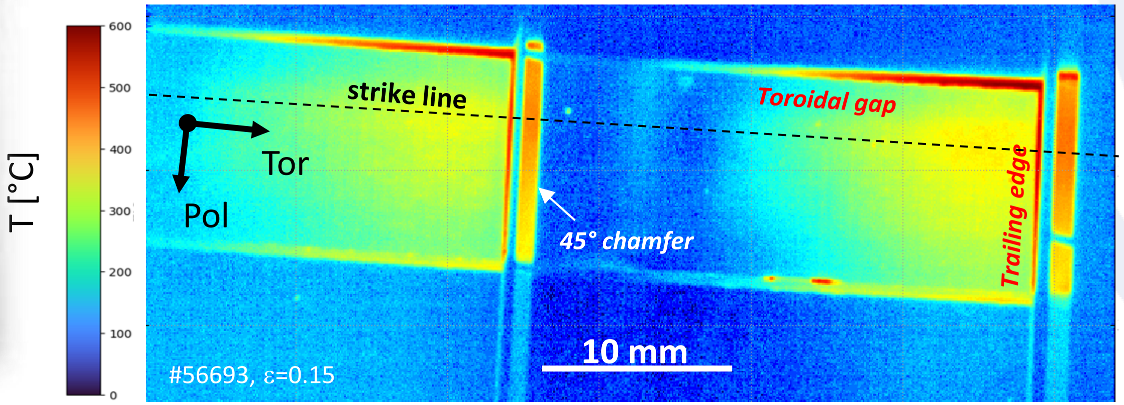
Shaping foreseen in ITER: 0.5 mm height toroidal bevel

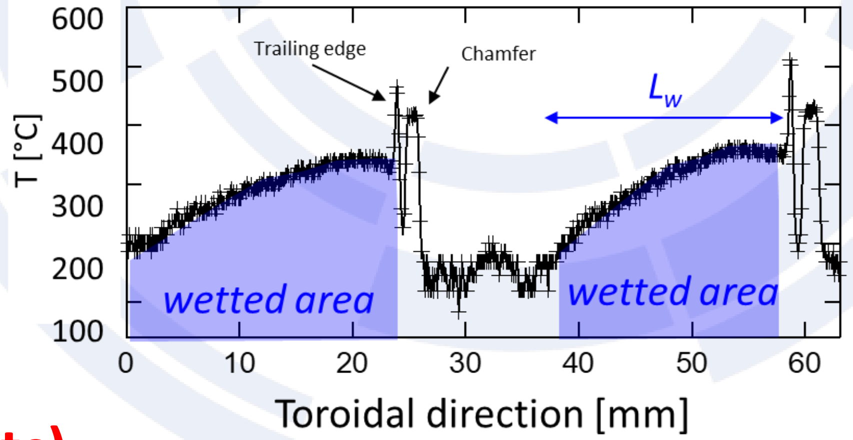


- Protection of the leading edges
- Reduction of the **wetted area** → +20% higher load compared to flat top geometry
- PFC assembly is critical due to grazing incidence of MFL
  - ➢ vertical misalignment (ITER tolerance = ±0.3mm)
  - ➢ MB tilt ±1° measured in WEST (ITER tolerance?)

➢ **Surface metrology mandatory (→ wall protection)**

VHR IR image during WEST plasma experiment:



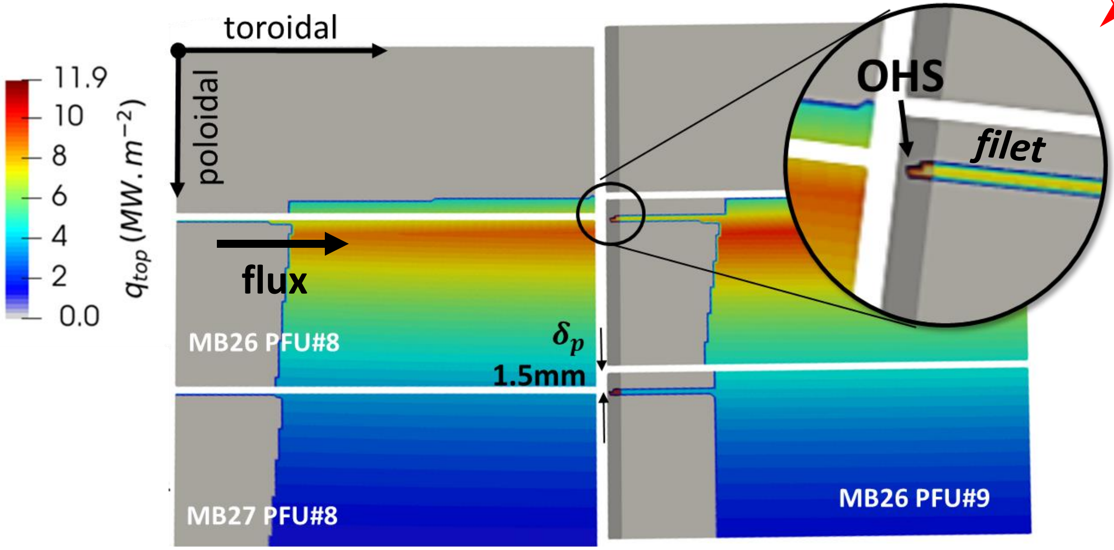


➢ **T° gradients on MB scale (# from HHF tests)**

Heat load on leading edges: prediction and observation

- 0.5 mm toroidal bevel offers **good protection on poloidal leading edges**
- MFL can still penetrate and strike the LE with high incidence angle*

*Optical Hot Spot (OHS)



Heat flux computed with the 3D PFCflux code

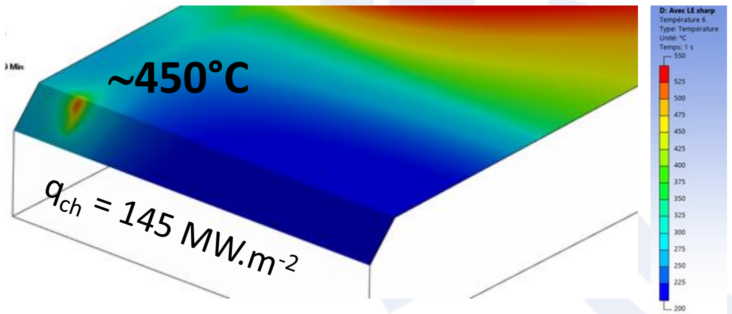
λ_q^t=3.3mm

➢ Very high heat flux expected on small surface

Thermal response estimation WEST L-mode plasma

q_n = 10 MW.m^{-2} top surface → q_{ch} = 145 MW.m^{-2}

Temperature maps: ANSYS simulation



- In WEST L-mode → acceptable temperature, with no vertical misalignments
- PIC modelling predicts higher heat load during ITER ELMs *[J. Gunn et al., NF 2019]*

→ High thermo-mechanical stresses, strong erosion and local melting

→ experimental validation is required: can we see the OHS with the VHR IR camera?

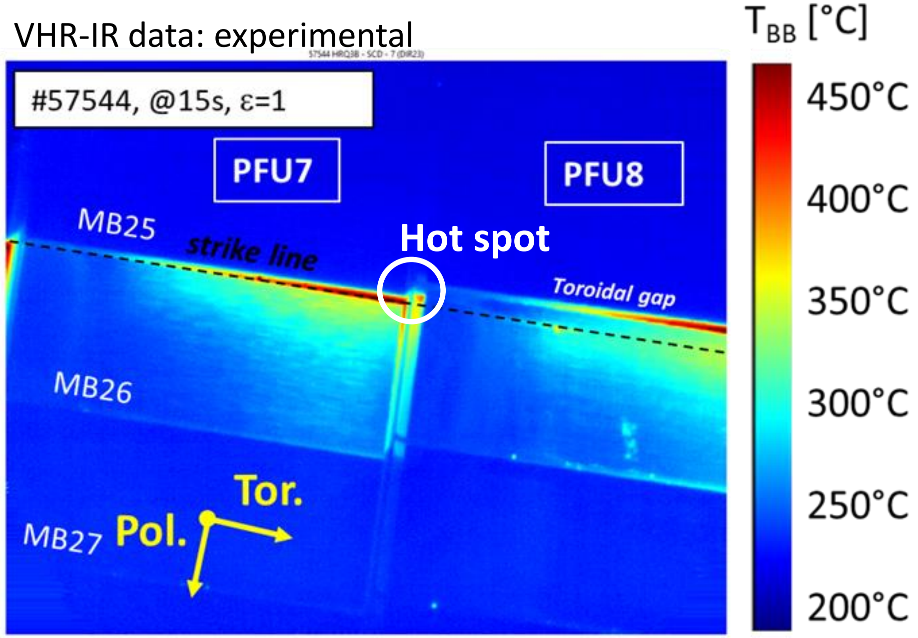
IR thermography in metallic environment: specular reflection disturbing

IR thermography sensitive to surface emissivity and potential reflection from the environment

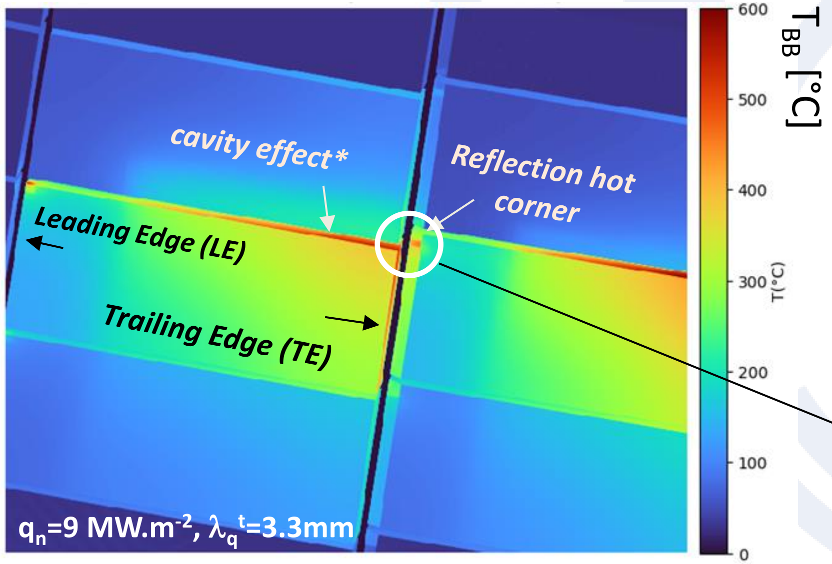
$$\Phi_{\text{mes}} = \varepsilon\,(\lambda, T\,)\mathcal{L}^0(\lambda, T\,) + \rho(\lambda\,)\textstyle\sum_i \varepsilon_i(\lambda, T_i)\mathcal{L}^0(\lambda, T_i) \qquad 0.1 < \varepsilon_W < 0.3$$

3D numerical workflow:PFCflux (heat load) / ANSYS (T°) / RAYMOND (reflection) ← *[A. Juven et. al., NME 38, 2024]*

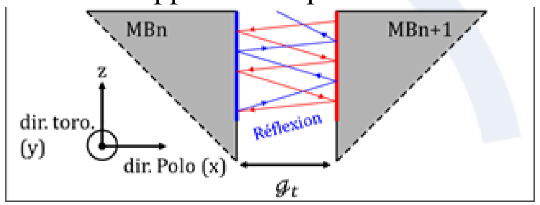
VHR-IR data: experimental

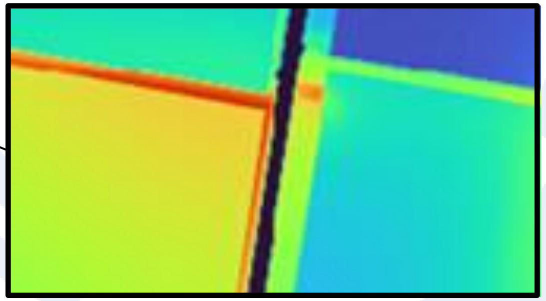


VHR-IR data: synthetic (thermal & photonic modelling)



*cavity effect = reflections on the W gap surfaces → enhanced local apparent temperature





[Q. Tichit et al., NME 2024]

→ Strong effect of the specular reflection in toroidal gap (TG) and trailing edge (TE)

→ in WEST the optical hot spots are diluted in the reflected signal (higher T° would be required)

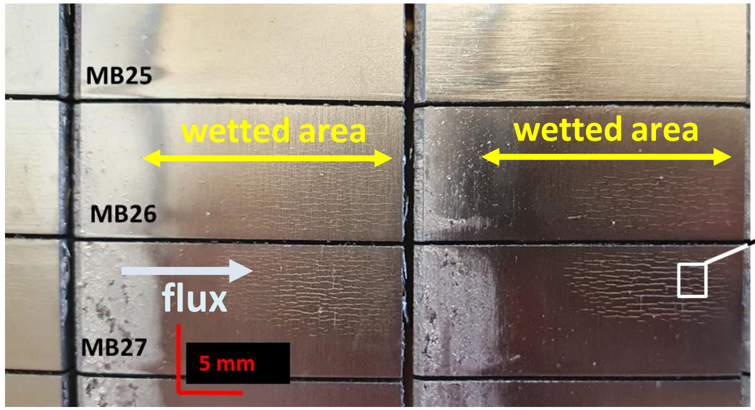
Outlines

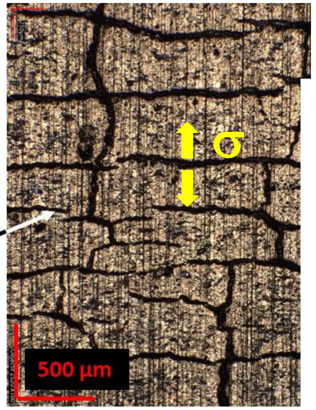
- **Introduction**
- **Assessing heat loads on castellated and shaped components**
- **Understanding damages : W-cracking and melting**
- **Summary and prospects**

PFC ageing: spontaneous cracks forming in the most loaded areas

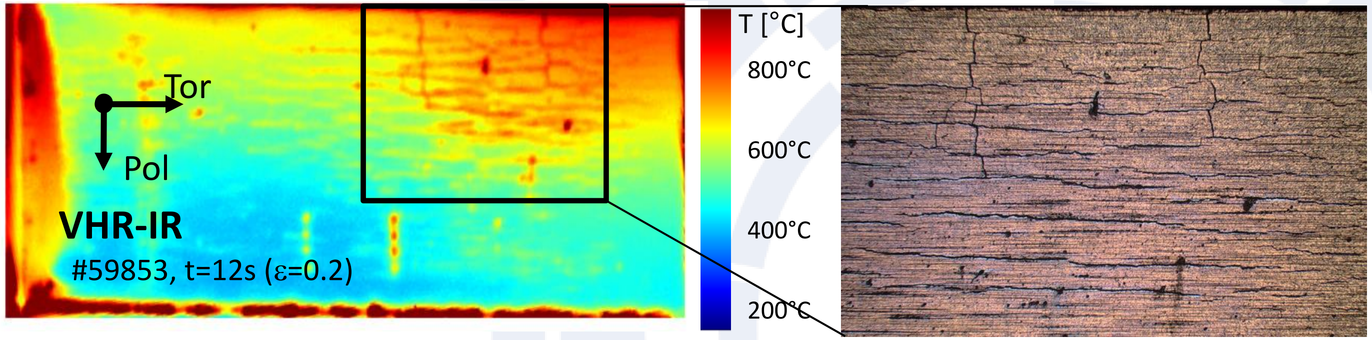
WEST plasma exposure leads to crack network on the top surface of MB in wetted area @ the outer strike point

crack width: 40-90µm





- Visible “in-operando”, heat flux ~10 MW/m^2

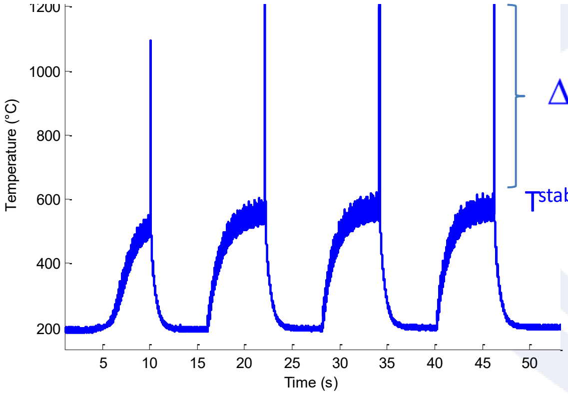


- Cracking not expected from standard HHF at those levels of steady-state → **transients initiate failure?**

Dedicated HHF testing (e$^-$ beam):

- 5 MW.m^{-2} steady state heat load
- With/without pre-heating, no λ_q
- 80-160 MW.m^{-2} transient (3ms, 6ms, 10ms) x100 cycles

HADES - High heAt loaD tESting - facility



→ No crack!

→ cracks related to thermal loading, but not only… plasma-induced effect?

Plasma operation with « macro-crack » failure

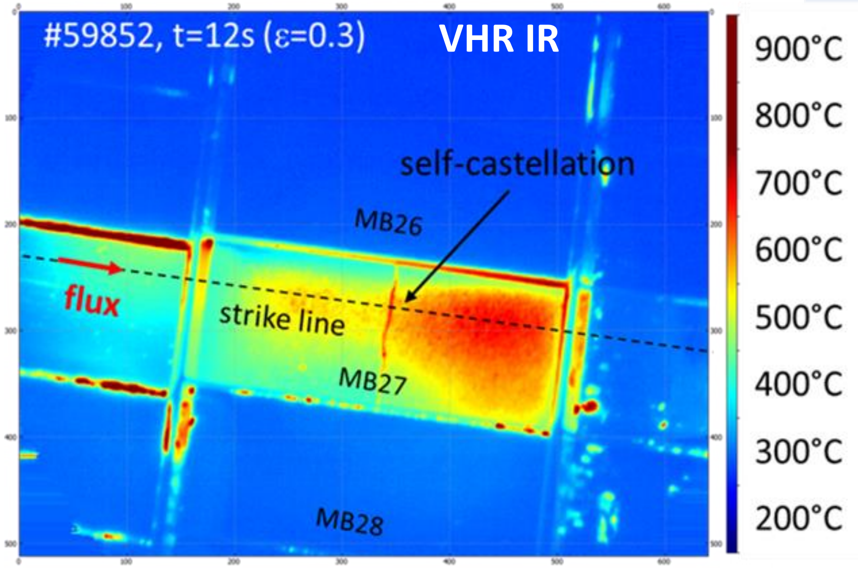
Strategy = use HHF test facility (e-beam) to generate well controlled and ≠ kind/level of damages
→ Mimic damage on ITER and expose the damaged component in tokamak device (in-situ)

JUDITH2

[M. Richou et al., NF 2022]

[M. Wirtz et al., NME 2017]

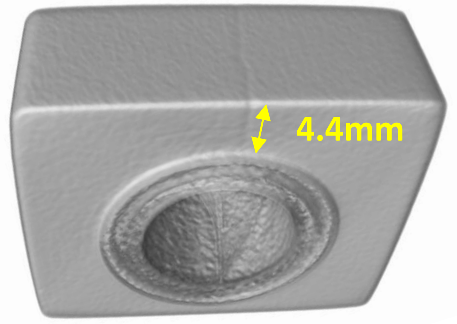
Deep crack accross MB



VHR IR data (0.1mm/pixel)

Damage type sets with the impact factor (F) and cycle number (representative of the ELM size/number) – JUDITH2

High energy X-Ray tomography (6 MeV)



fluence up to 10^{27} D/m^2, T_{surf} up to 700°C

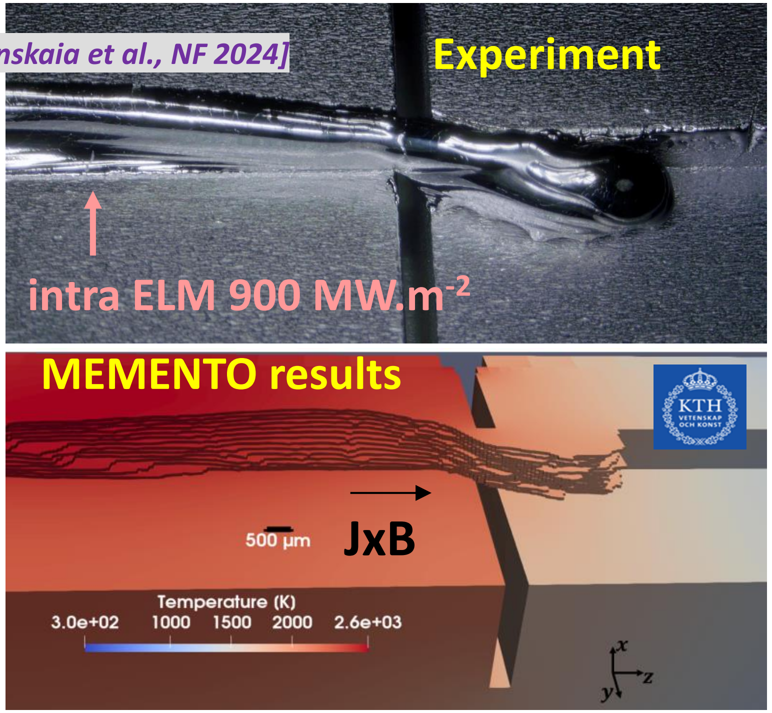
No observable evolution of the crack damaged surface (VHR IR data)

→ heat exhaust capabilities unchanged (rising and cooling time similar to healthy component)

→ post-mortem measurement scheduled next year

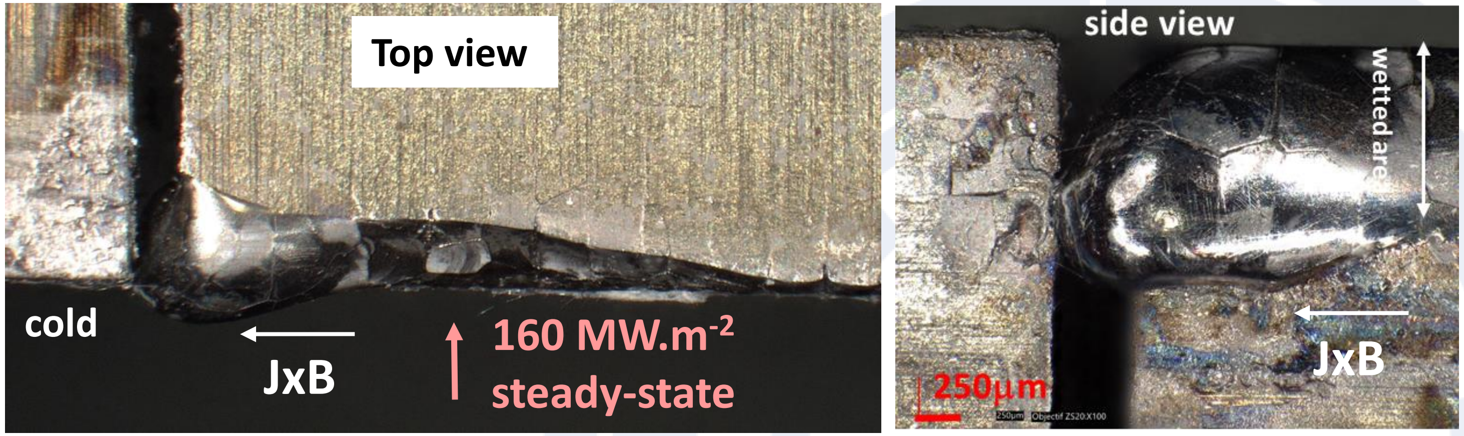
Tungsten melt transport across PFC gaps: sustained vs. transient

Transient melting in AUG: divertor manipultor



20 cm.s^{-1} melt velocities and pool depth of several tens of µm

Sustained melting in WEST: actively cooled PFC



few cm.s^{-1} melt velocities and ultra-thin ~µm layers

Prompt bridge freezing due to contact with the surface across the gap (cold)
- re-solidification prevents the melt from evolving around the corner

MEMENTO (macroscopic melt dynamics) extensively validated through dedicated EUROfusion & ITPA experiments *[S. Ratynskaia et al., NME 2022]*

Gap bridging without wetting of the inner gap (no infiltration) in both devices WEST/AUG

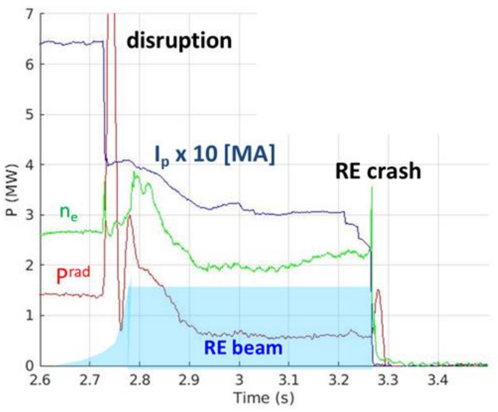
→ Castellation preserved underneath

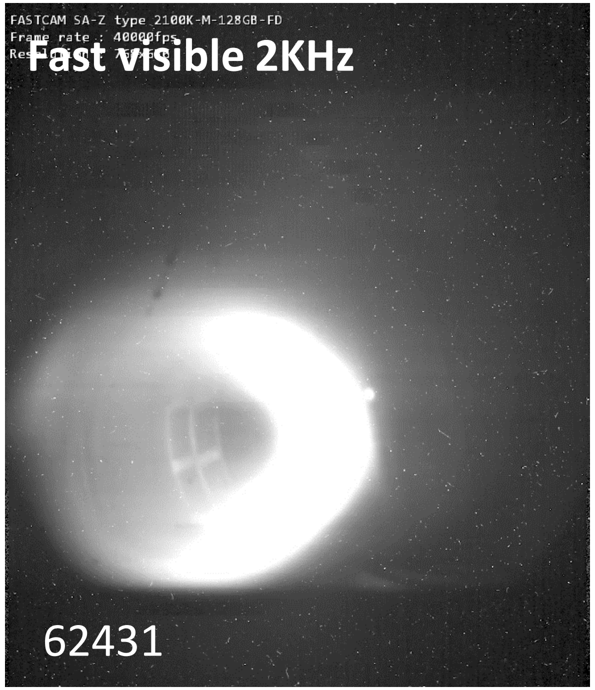
→ subsequent consequences EM load during disruption to be assessed ...

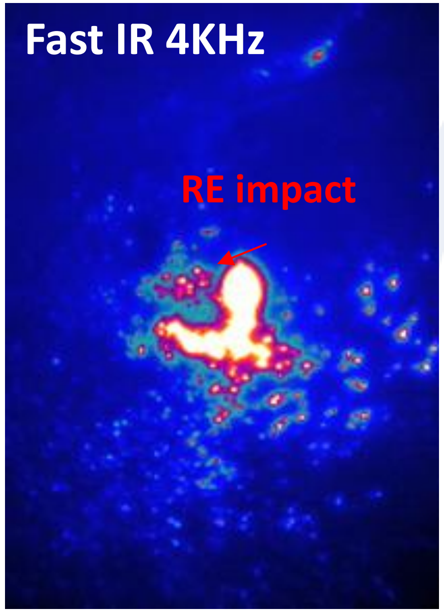
W PFC damage induced by runaway electron incidence

- 1st controlled RE impact experiment on W-material (instrumented tile with TC)
- dedicated set of diagnostics to characterize the RE beam

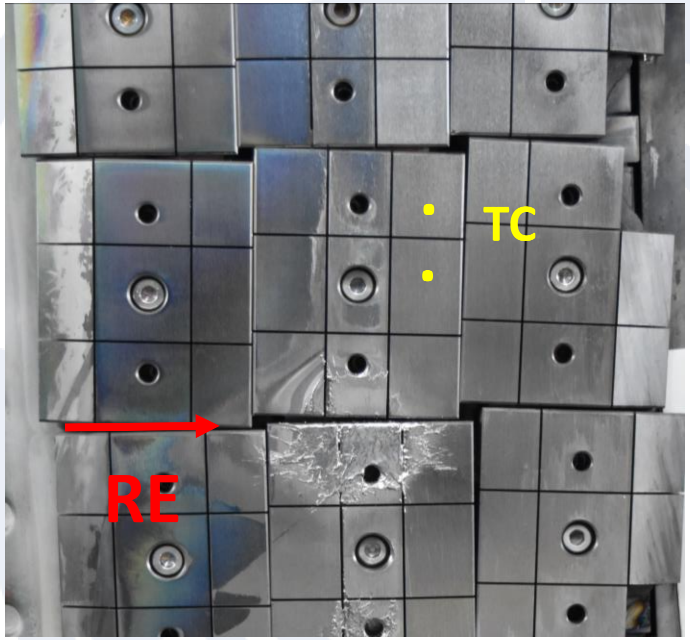
I_{RE}=230kA







Bulk W (inner bumpers)



- **Major damage despite high melting point of W**

RE energy (Runaway Electron Imaging Spectroscopy):
- 17 MeV monoenergetic beam

PFC energy load (TC):
- ~50 kJ (preliminary estimation with TC)

Duration of the RE impact (fast visible and IR data):
- ~3 ms

[G. Ghillardi et al,. PPCF 2025]

[S. Ratynskaia, Plenary talk, 51th EPS Plasma Physics]

PFC damage: melting
- circular impact ∅~8cm
- ~ 1 mm depth
- bridging of the castellation

Work-flow (GEANT4 - MEMENTO/LS-DYNA) to test predictive capabilities for ITER

Summary

- **PFC testing in tokamaks and HHF test facilities are essential for the validation of the numerical tools used for ITER or other next step fusion devices**
  - Damage mechanisms, failure modes → PFC lifetime and safe operation
- **No major failure observed in WEST, no evolution of the heat exhaust capabilities**
  - Unexpected cracking observed: combined thermomechanical “transient” & plasma induced-effect suspected ?

→ post-mortem on-going, further exposure scheduled in 2025-2026 with highly radiative scenario (X-point radiator) *[N. Rivals et al., EX-D 3067]*

- **Various W damages tested:**
  - Exposure of pre-damaged PFUs performed at various levels:

    No significant degradation/evolution of the macro crack observed
  - Tungsten melt transport across PFC gaps:

    Gap bridging in both devices WEST (sust.)/AUG (transient)
  - First controlled RE impact on W material performed in WEST

    Major damage despite high melting point of W

    Data available to test predictive capabilities for ITER

→ experimental data used to test the predictive capabilities in ITER (PFC lifetime + operation)