Wendelstein 7-X

IPP Max-Planck-Institut für Plasmaphysik

Physics in preparation of long pulse operation at W7-X

Marcin Jakubowski for the W7-X Team

MAX PLANCK GESELLSCHAFT

EUROfusion

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

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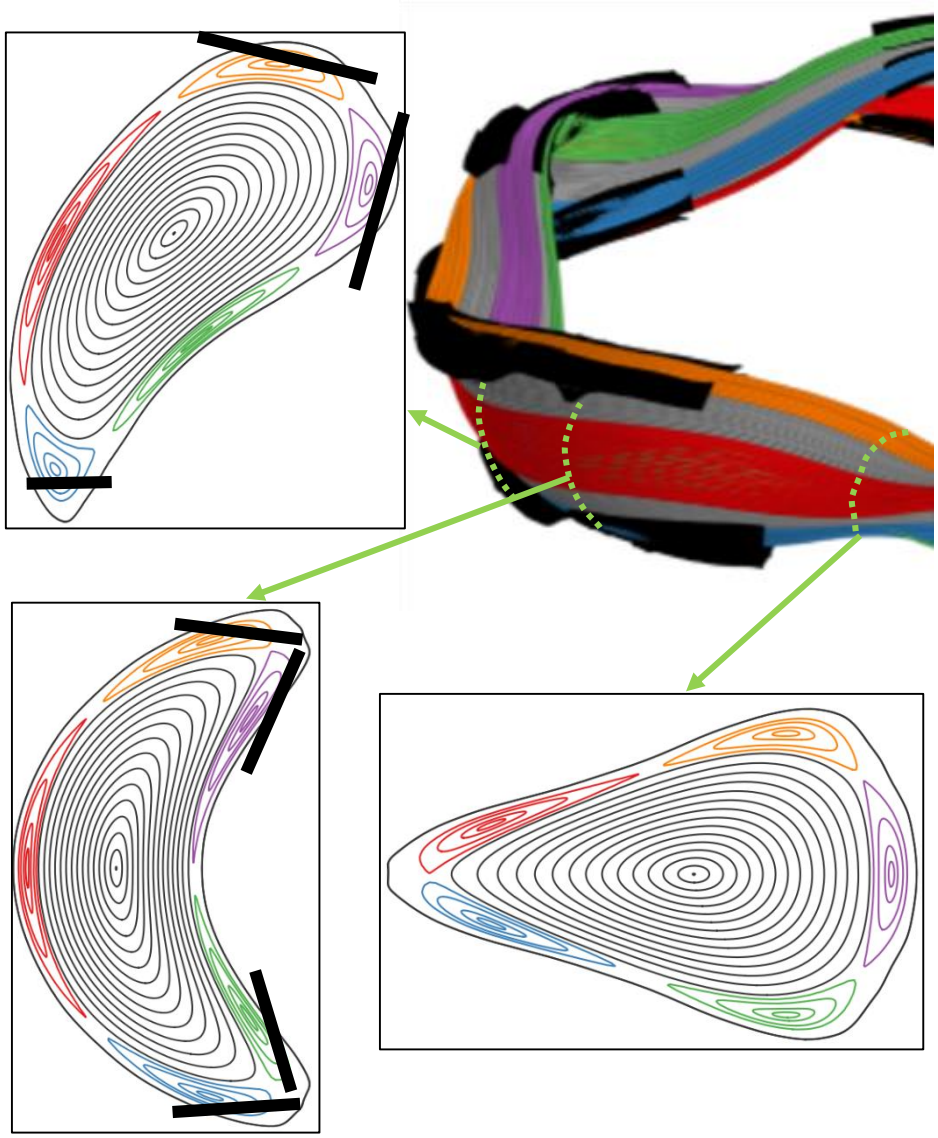
Island divertor: exhaust for helical plasma

W7-X wants to demonstrate **high power, high performance at steady-state**
10 separate divertor units, adapted to the shape of the flux surfaces

Large magnetic islands (standard: 5/5) form so-called **island divertor**.
Resonant islands require error field correction [1,2]



[1] S. A. Bozhenkov *et al.*, Nucl. Fus. **57**,126030 (2017)
[2] S. A. Lazerson *et al.*, Nucl. Fus. **57**, 046026 (2017)

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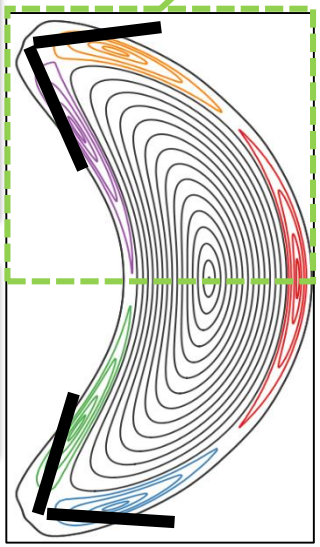
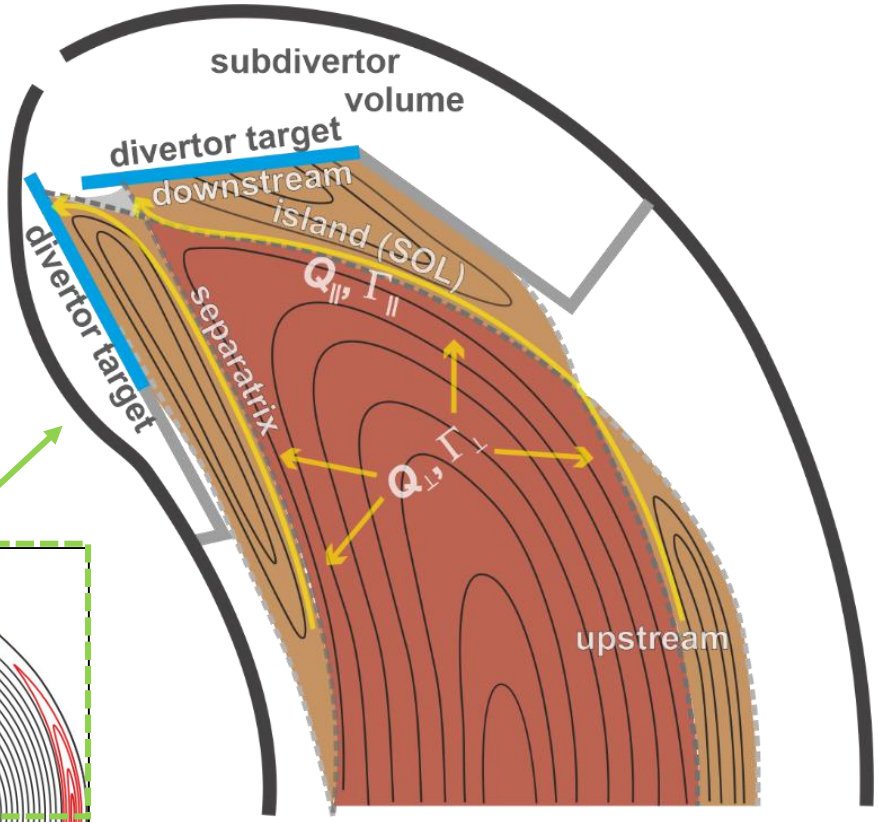
Resonant islands require error field correction

Open field lines guide ions lost from the confined plasma to the divertor target (**scrape-off layer**)
Island divertor features ExB drifts [1]

Neutral compression provided by neutrals **recycling** near the divertor target

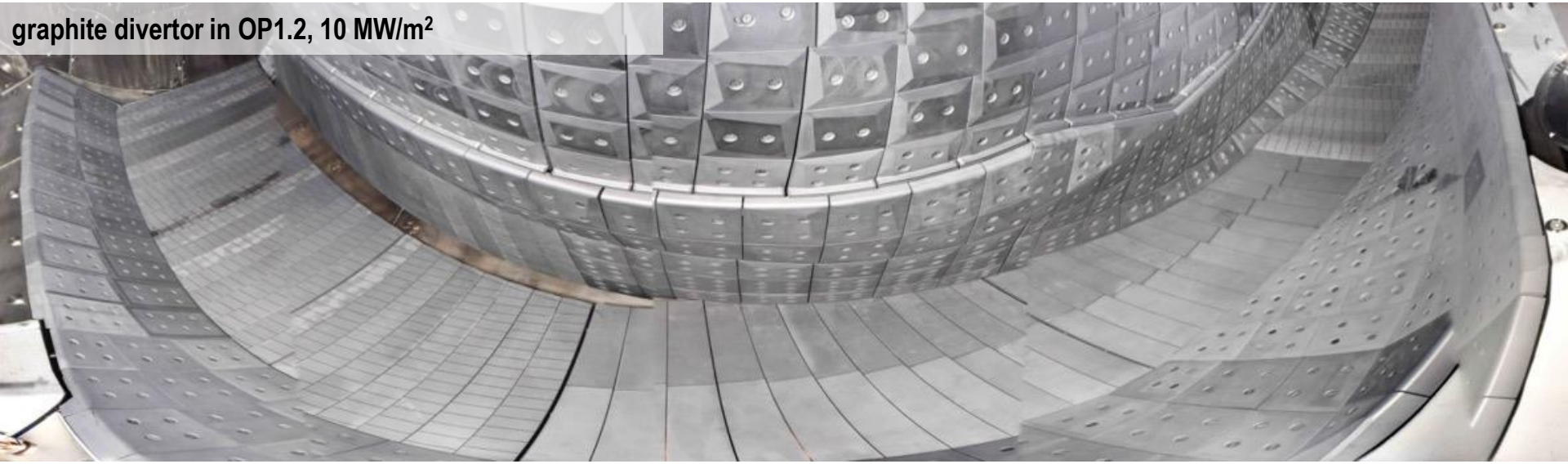




[1] K. C. Hammond *et al.*, PPCF **61**, 125001 (2019)

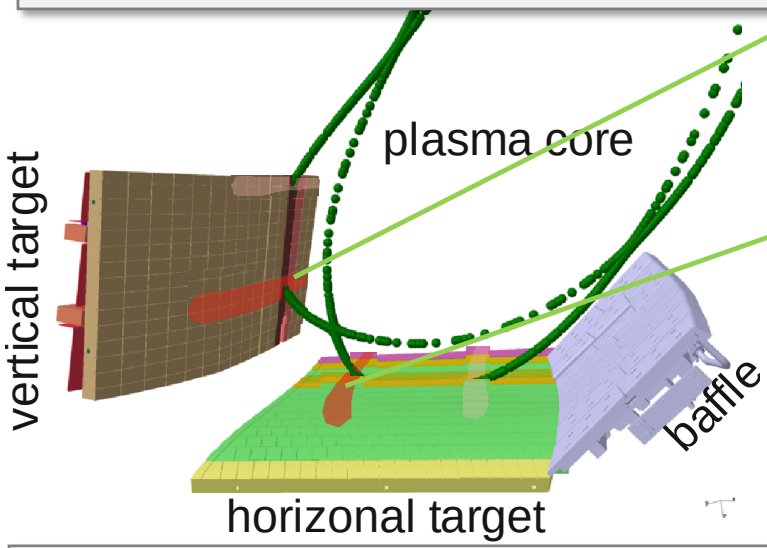
Towards steady-state compatible divertor

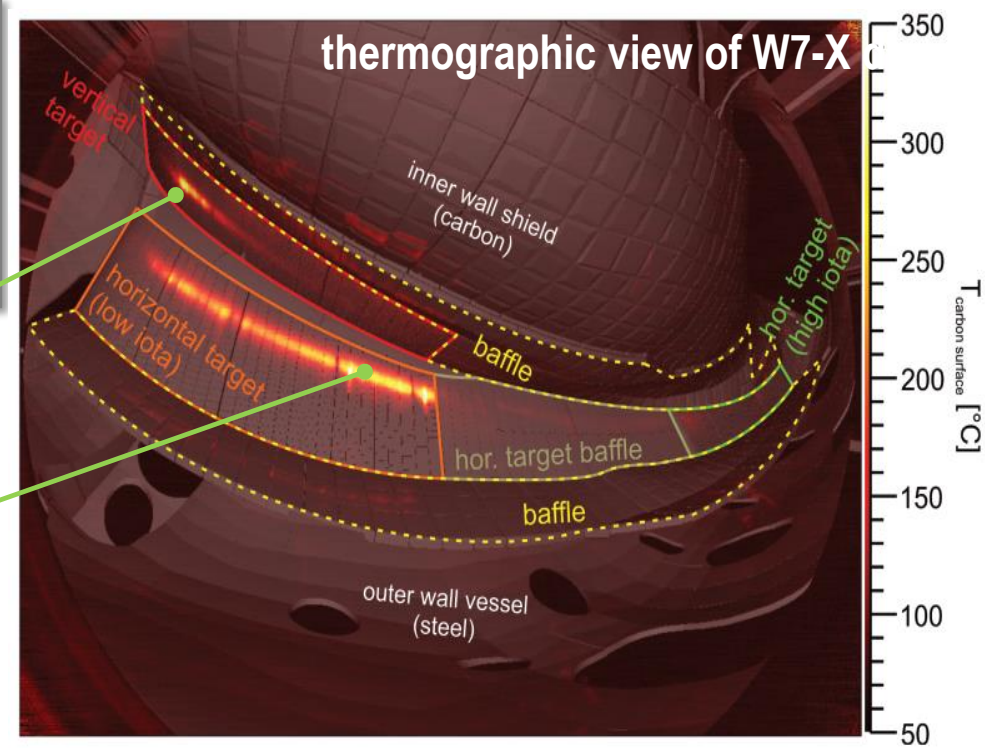
OP1.1	OP1.2	OP2	OP3
2015 / 2016 • 5 MW • 4 MJ • 6 sec	**2017 / 2018** • 8 MW • 200 MJ • 100 sec	**2022 - 2025** • 10 MW • 1GJ, →18 GJ • 30 minutes	**202x …** • Heating upgrades • 18 GJ • 30 minutes
Inertial cooling	Inertial cooling	Water-cooling of PFCs	carbon-free device



Three-dimensional strike line of the island divertor

- Island divertor at W7-X in **standard configuration** forms **two strike-lines** on horizontal and vertical target [2,3].
- Strike lines may be affected by toroidal currents [1-3]





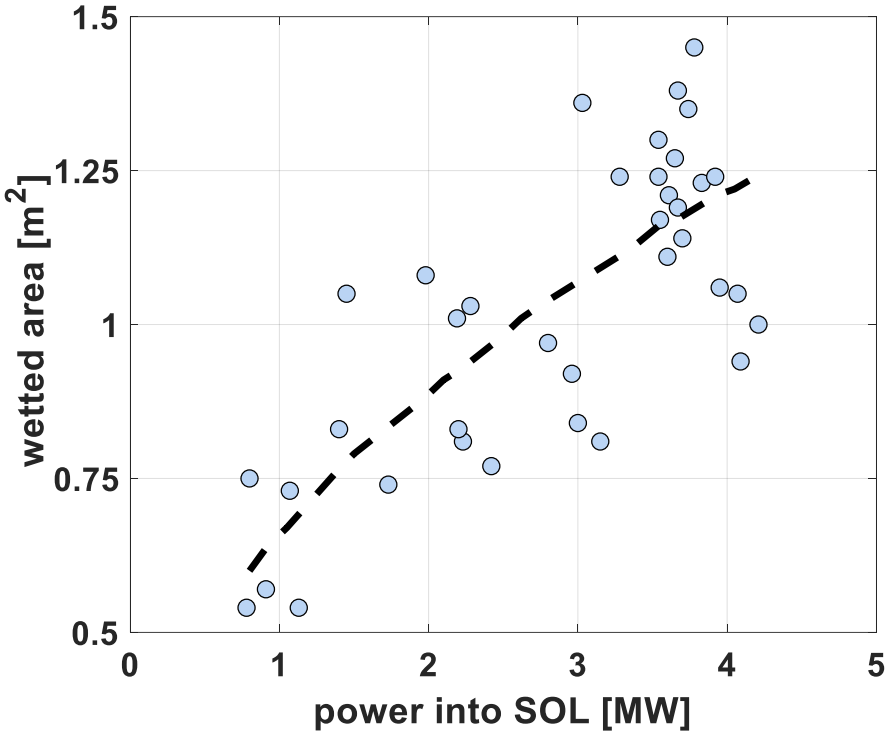
How well island divertor can **spread power** on the **divertor surface**? W7-X will operate with up to **30 minutes of plasma duration** with 10 MW of heating with technical limit of the divertor heat flux at 10 MW/m^2.

[1] Y. Gao *et al.*, Nuclear Fusion **59** (2019), [3] T. S. Pedersen *et al.*, Nuclear Fusion **59**, 096014 (2019)

Island divertor allows efficient heat flux spreading

- Wetted area A_{wet} is ratio of heat absorbed by the divertor (in eg. MW) to peak heat flux (in eg. MW/m^2)
- The allowed peak heat flux of the HHF divertor is 10 MW/m^2
  ➔ If P_{SOL} ~ 8 MW so A_{wet} > 0.8 m^2
- **Positive scaling with SOL power observed for attached plasmas** $\boldsymbol{A_{wet} \sim P_{SOL}^{0.44}}$

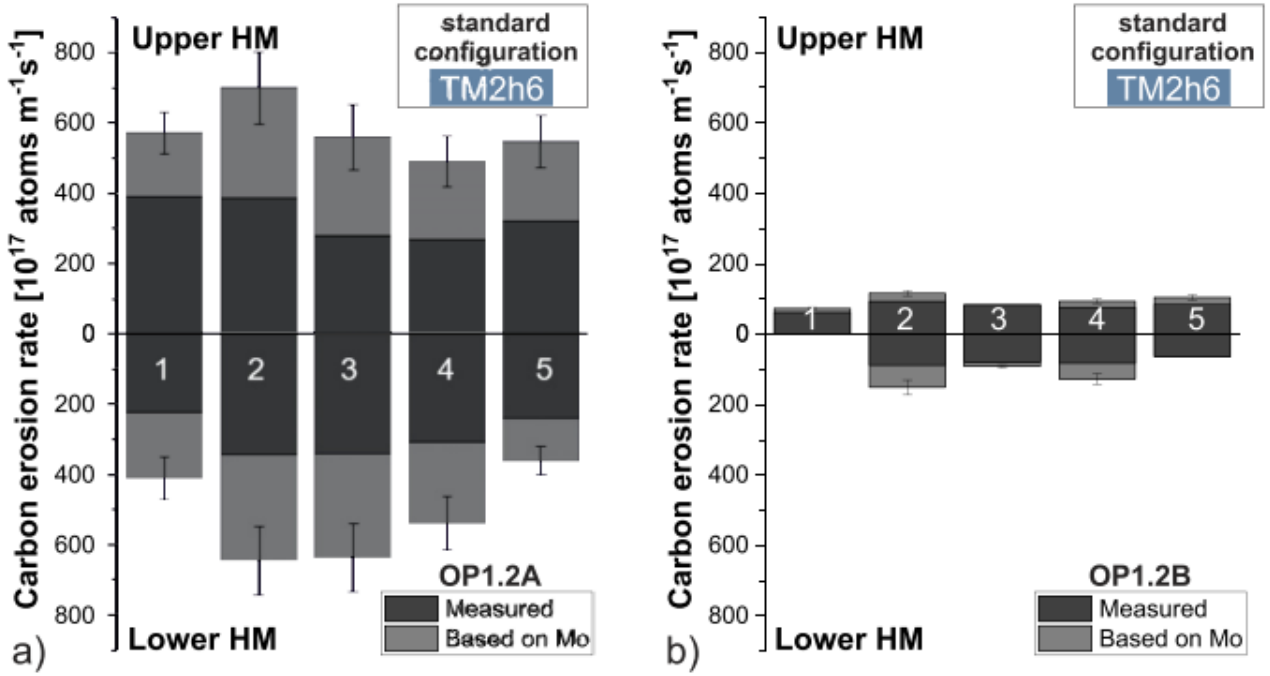
W7-X (standard): < 1.5 m^2 [1]
JET (L-mode): <1.6 m^2 [2]
ASDEX Upgrade (L-mode): < 0.8 m^2 [3]



[1] H. Niemann *et al.*, Nucl. Fusion 60 (2020) 084003
[2] T. Eich, et al., JNM 415 (2011) S856, [3] B. Sieglin, et al., PPCF 58

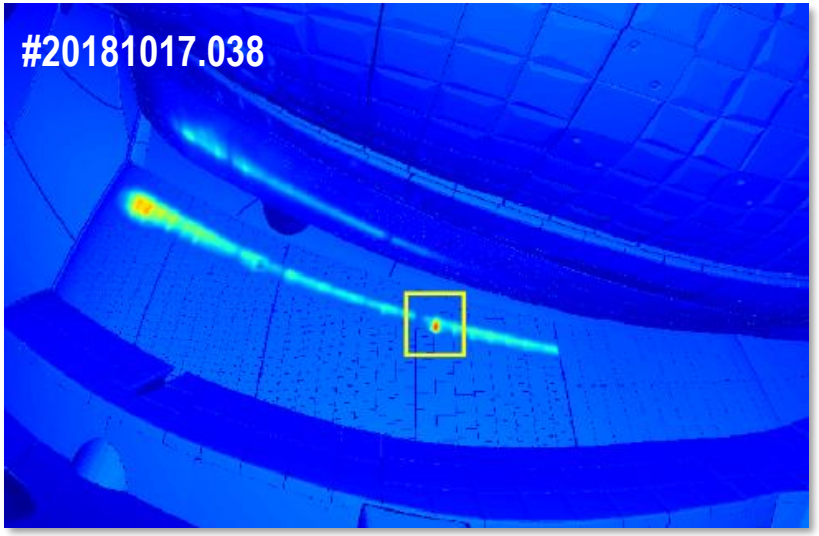
Sputtering of C reduced by boronization, but still may be an issue for long pulse

[S. Brezinsek *et al* 2022 *Nucl. Fusion* **62** 016006]



- Boronization reduces sputtering of C by a factor of 5.4 (due to strong suppression of O in plasma)
- Contrary to W – sputtering of C does not have a threshold in downstream temperature, i.e. chemical erosion is always present. Based on OP1.2b, the estimates for the attached plasmas are at the level of 7.6 g for a 30 minutes of plasma. This may be an issue for long pulse operations. More experiments planned for OP2

W7-X plasma is resilient to strong influx of impurities



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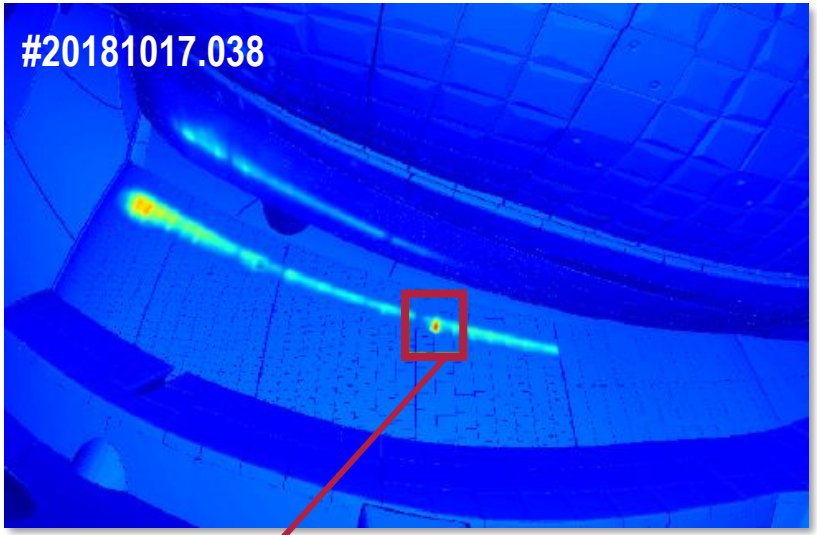
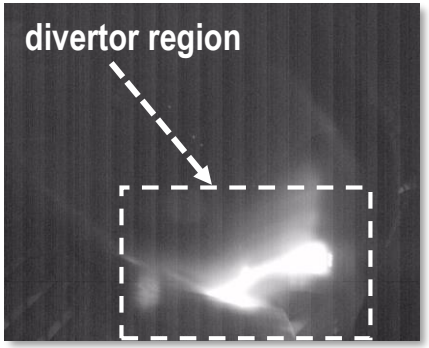
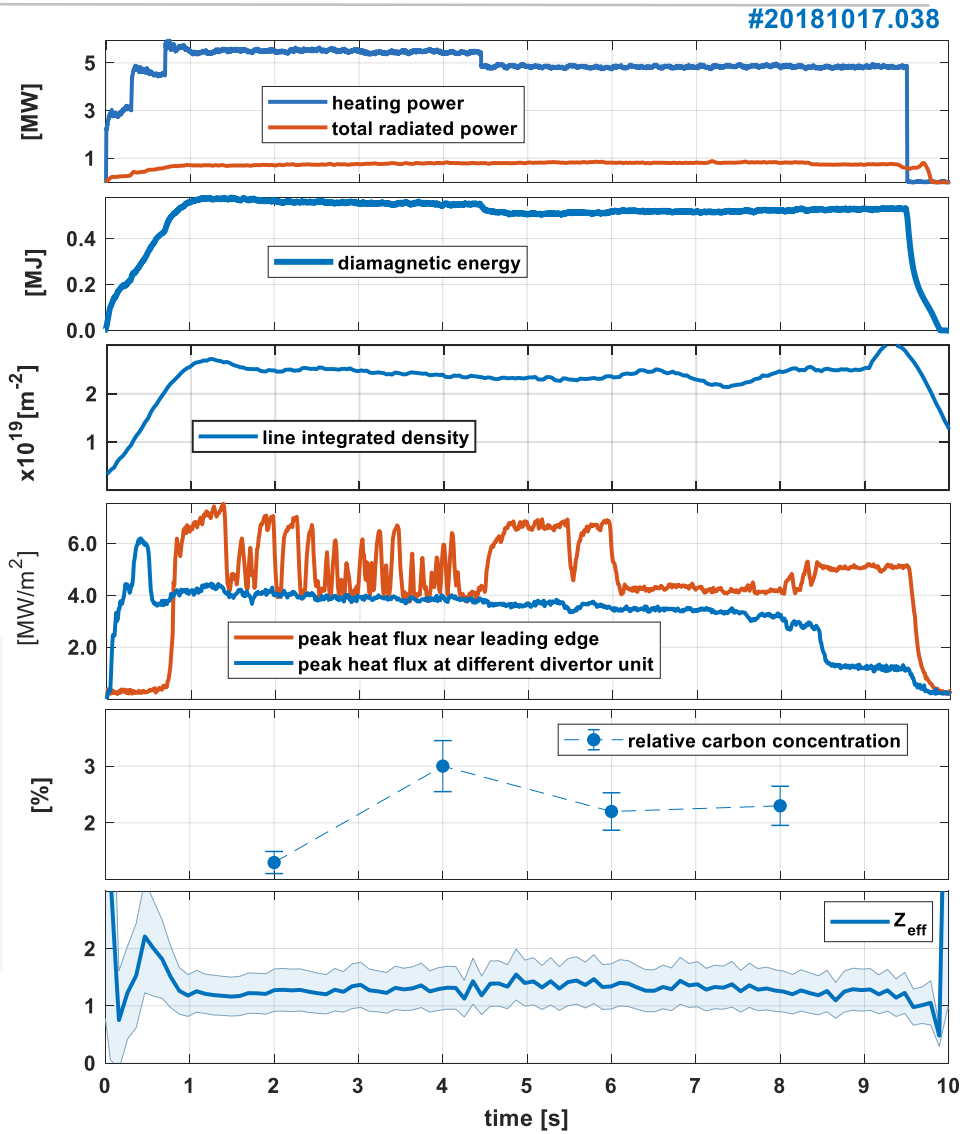






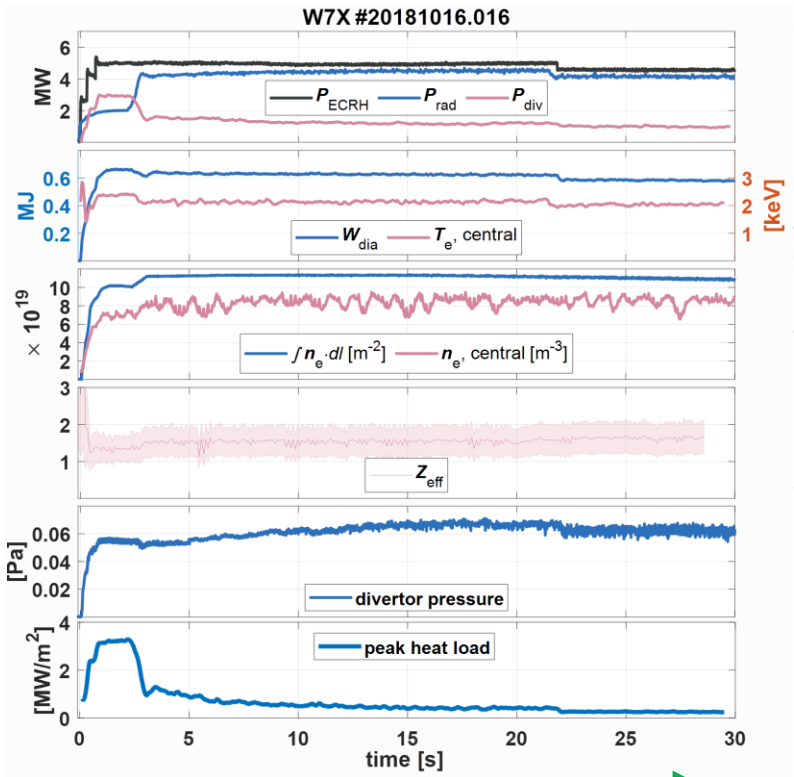
[M. Jakubowski et al 2021 Nucl. Fusion 61 106003]

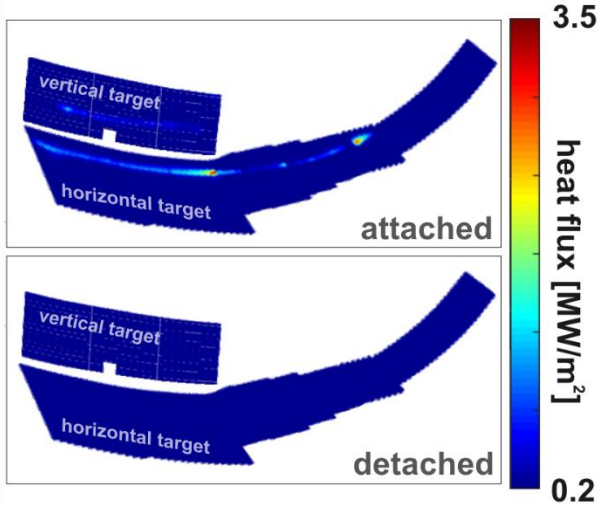
[B. Geiger, Nuclear Fusion, vol. 59, p. 046009, 2019]

Detached plasmas

Stable detachment regime

- Radiated power fraction of 80-90%
- Drop in diamagnetic energy ≲10 %
- Constant $Z_{eff} \approx 1.5$
- Sufficient sub-divertor pressure
- Virtually no convective loads to the target.





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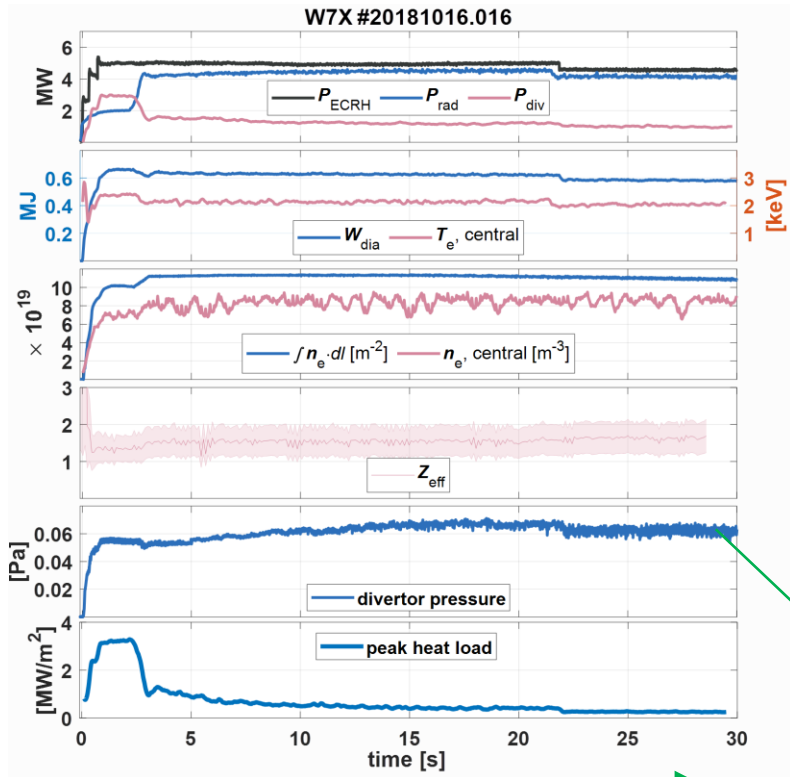
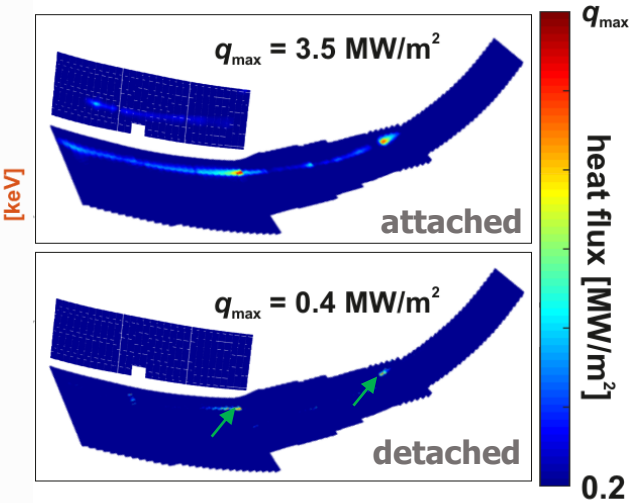
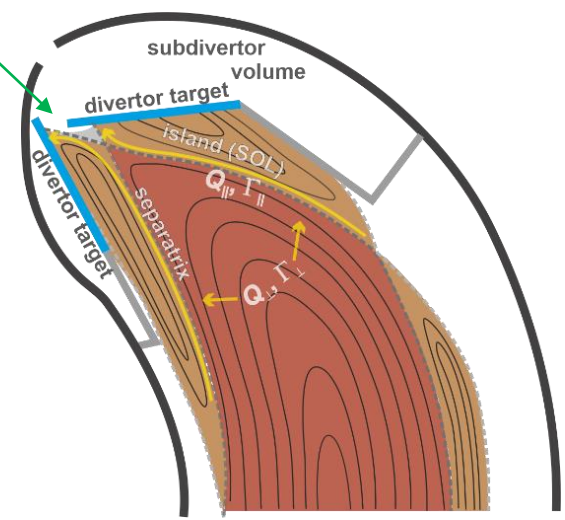


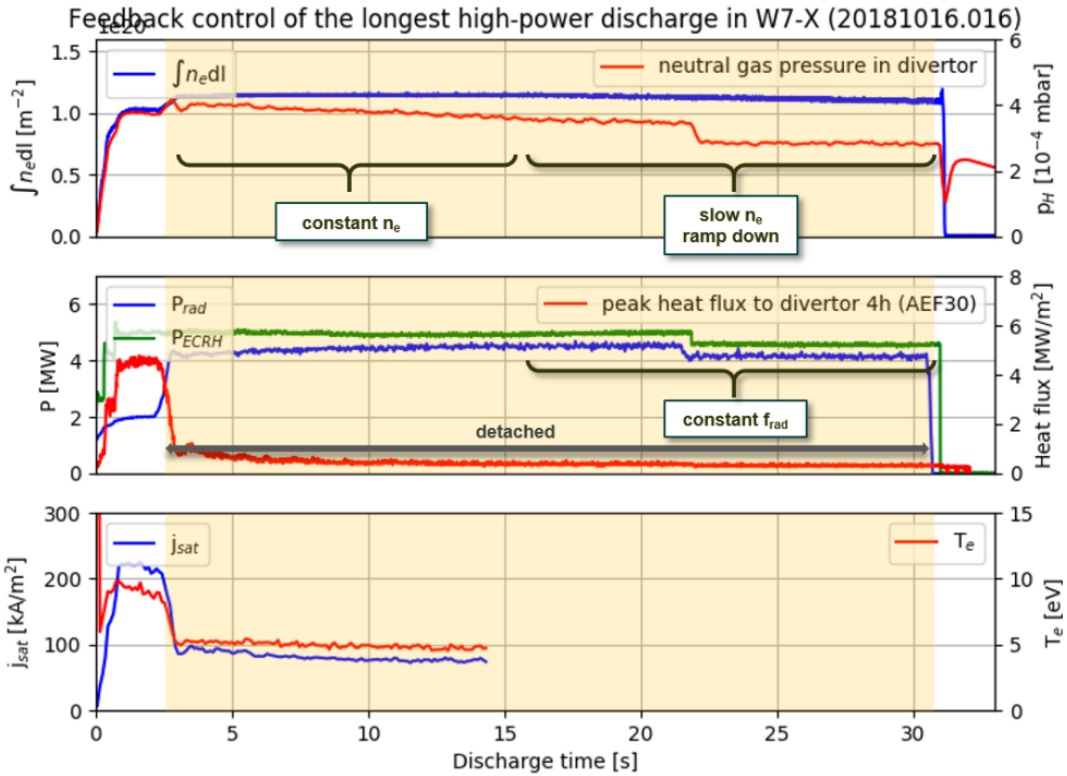


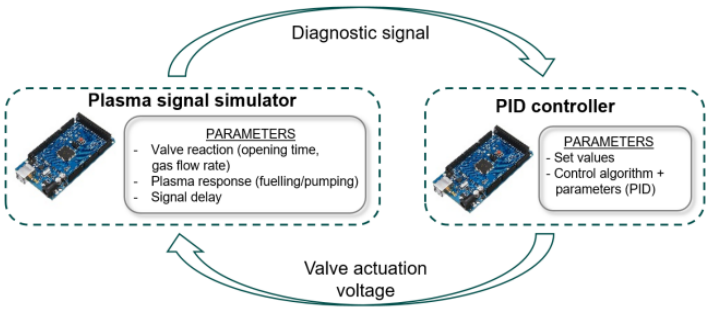


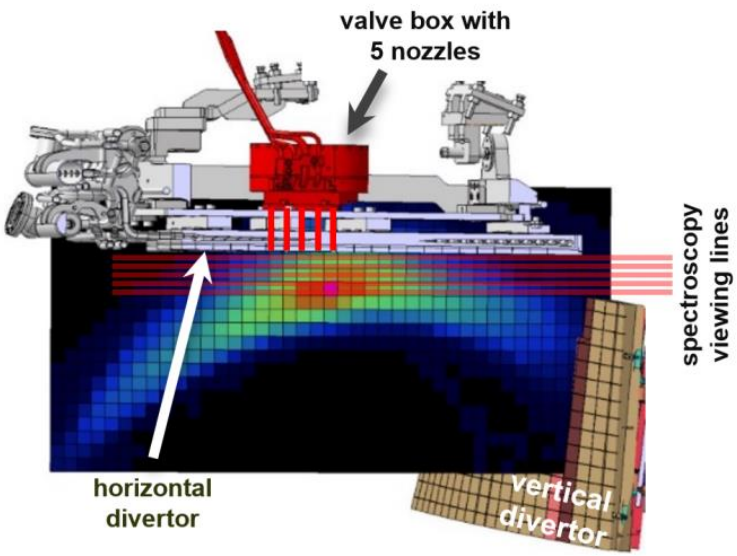
detachment

[M. Jakubowski *et al* 2021 *Nucl. Fusion* **61** 106003]

Stable detachment thanks to reliable feedback system



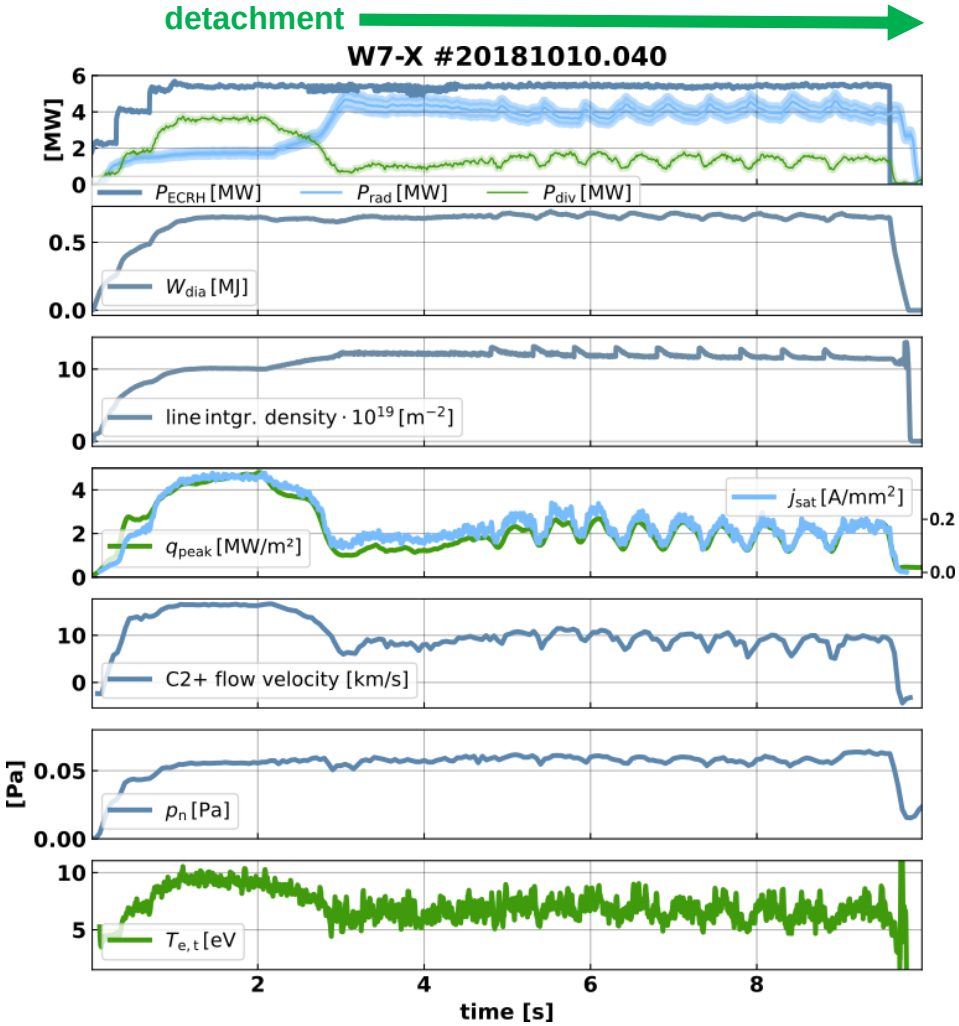




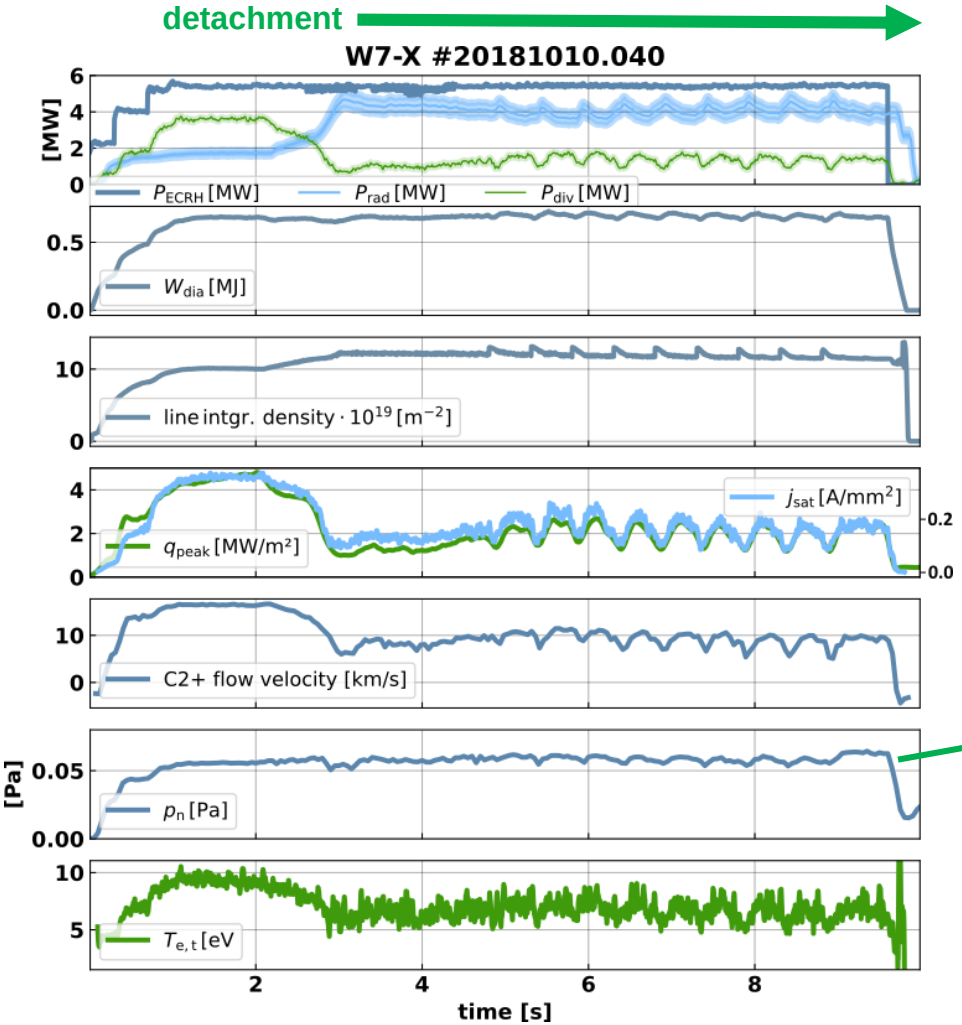
A feedback control system was established in W7-X to actively control the gas injection (actuator) for plasma fuelling and impurity seeding through the divertors

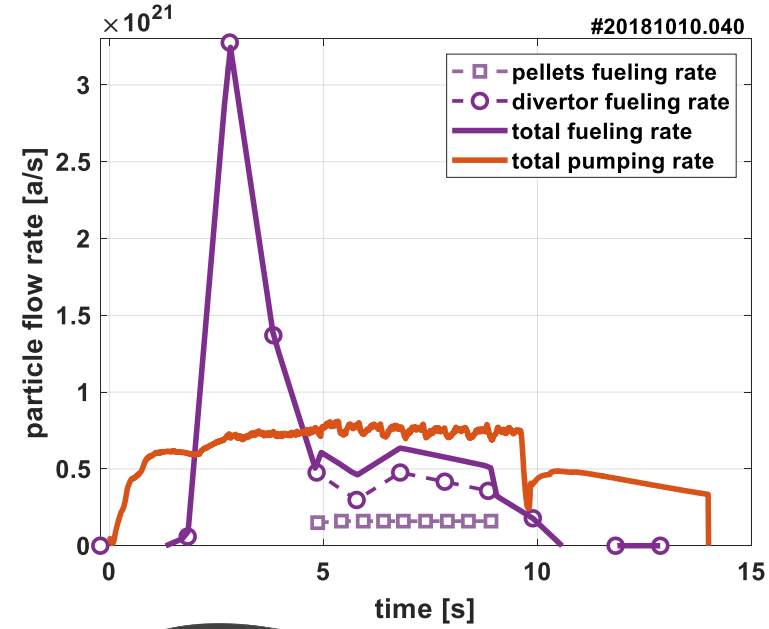
[M. Krychowiak, et al., PSI 2022]

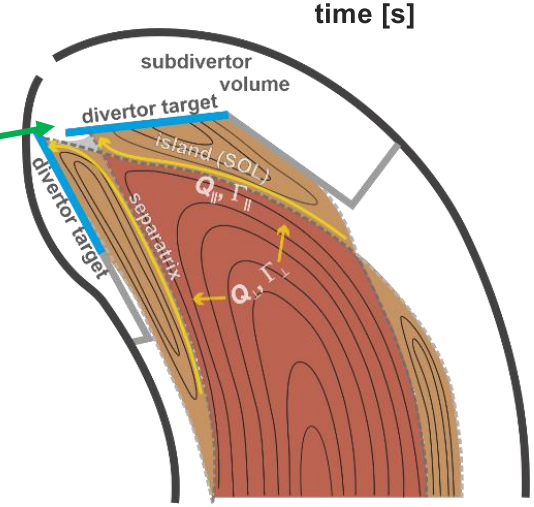
Upstream pressure defines downstream parameters



Neutral pressure sufficient for steady-state density control at W7-X

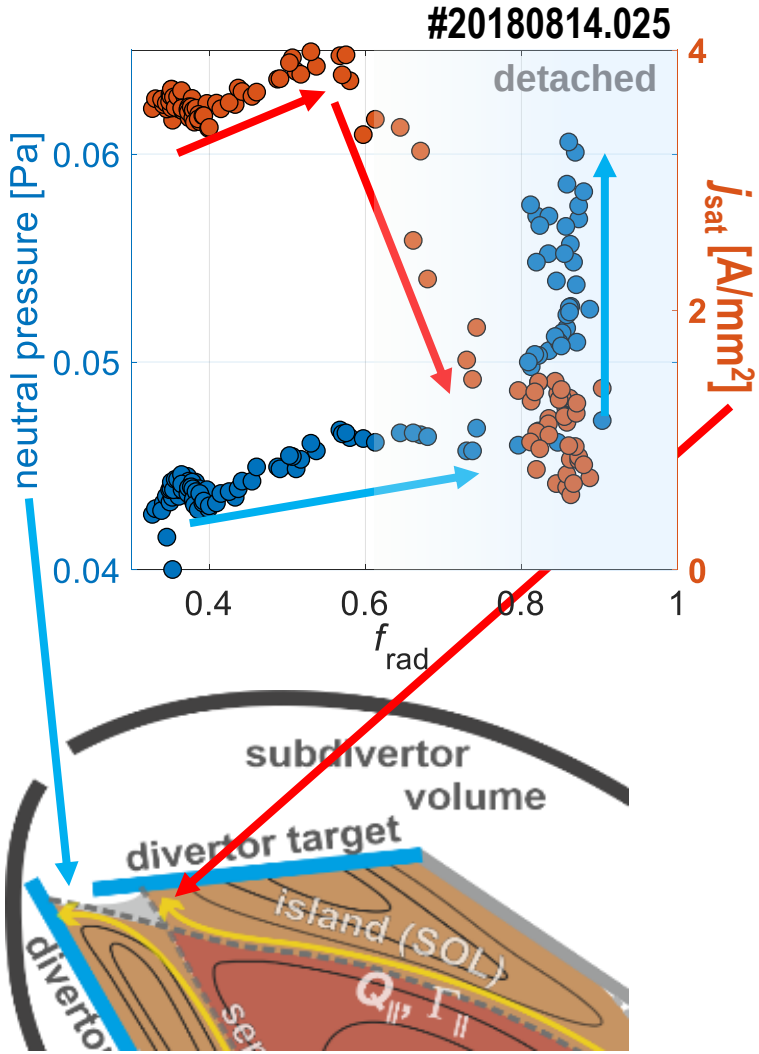






See also [G. Schlisio *et al.*, Nuclear Fusion **61**, 036031 (2021)]

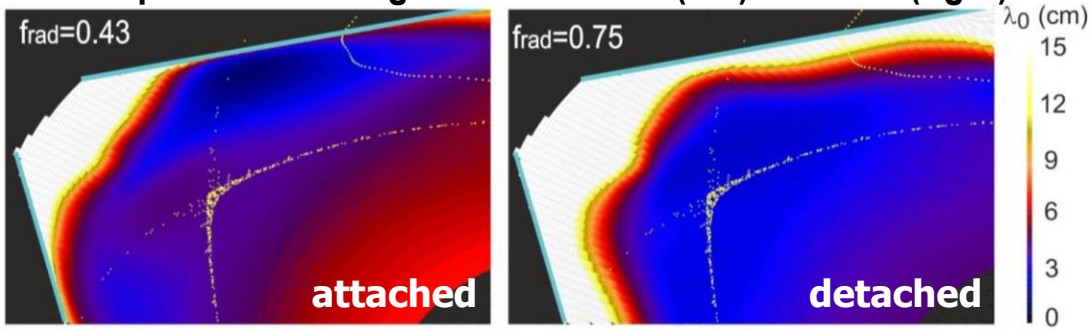
Distribution of neutrals defines neutral influx into pumping domain

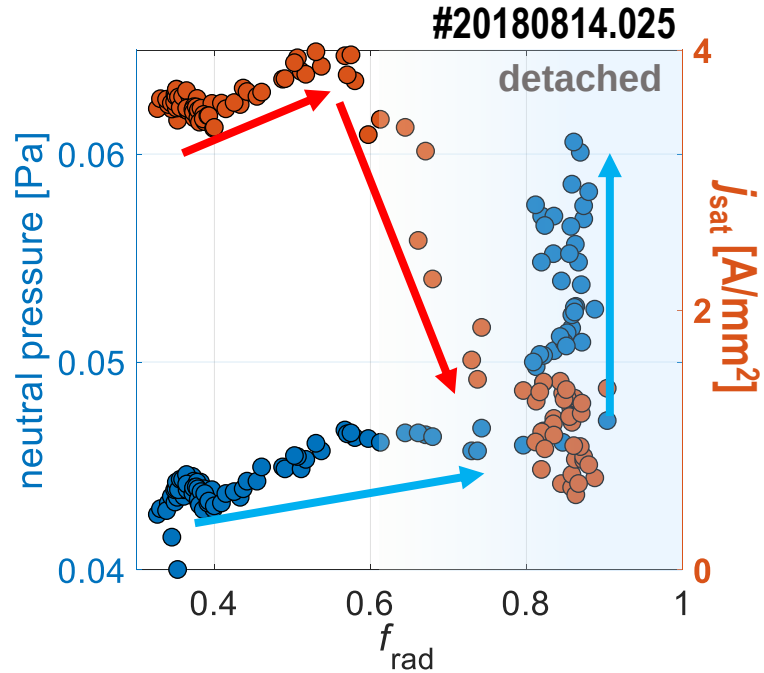


Distribution of neutrals defines neutral influx $\Gamma_{in,\,H^\circ}$ into pumping domain

EMC3-Eirene, Y. Feng, NF (submitted)

neutral penetration length at frad = 0.43 (left) and 0.75 (right)





$$\lambda_0 = v_{ith}/n\sqrt{<\sigma v>_{CX}<\sigma v>_i}$$

Two effects increase neutral pressure:

1. Longer penetration length mainly due to lower plasma temperature (5 eV) in the islands

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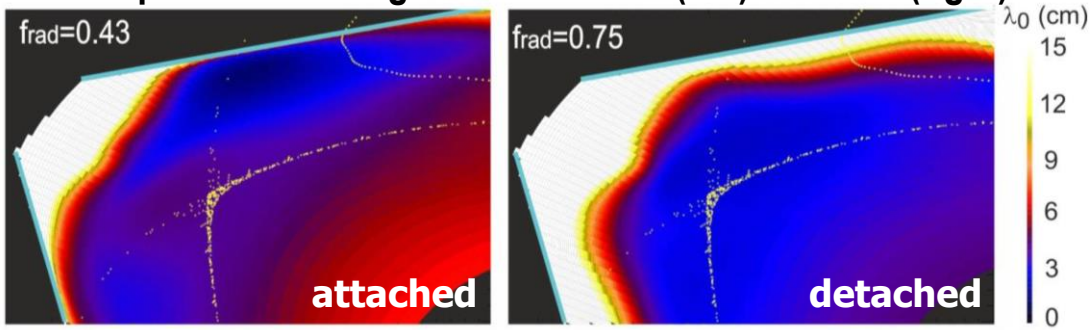
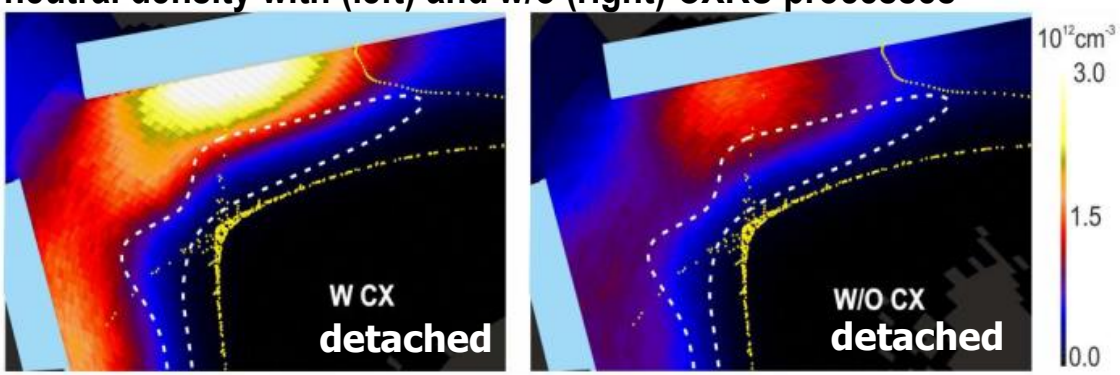
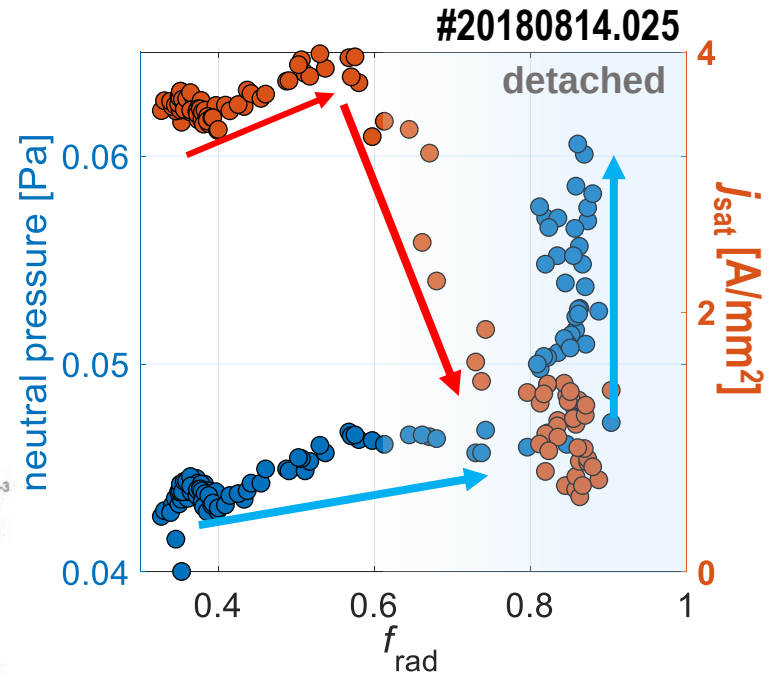


neutral density with (left) and w/o (right) CXRS processes





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Two effects increase neutral pressure:

1. Longer penetration length mainly due to lower plasma temperature (5 eV) in the islands
2. CXRS processes between fast ions coming from upstream and slow neutrals downstream.

Summary

- **Wendelstein 7-X** demonstrated that **island divertor** is an attractive exhaust concept.
- In attached state
  - We can efficiently spread heat with **wetted area of up to 1.5 m^2** Moreover it **increases with P_{SOL}**.
  - Although carbon erosion is expected to play a significant role in the long pulse scenarios…
  - **overloading divertor** leading edge showed that **plasma core is robust against impurity** accumulation.

- **En-route to detachment** we observed **higher recycling regime**. This has been enabled by **good separation of counter-streaming flows** in the island divertor.
- **Stable (≤ 26 s), complete detachment** in many scenarios was achieved with the feedback systems.
- Longer pulses (1 GJ, ~100 s, 2023) → (18 GJ, ~ 30 minutes, 2025) are planned for the following years.

Thank you for your attention