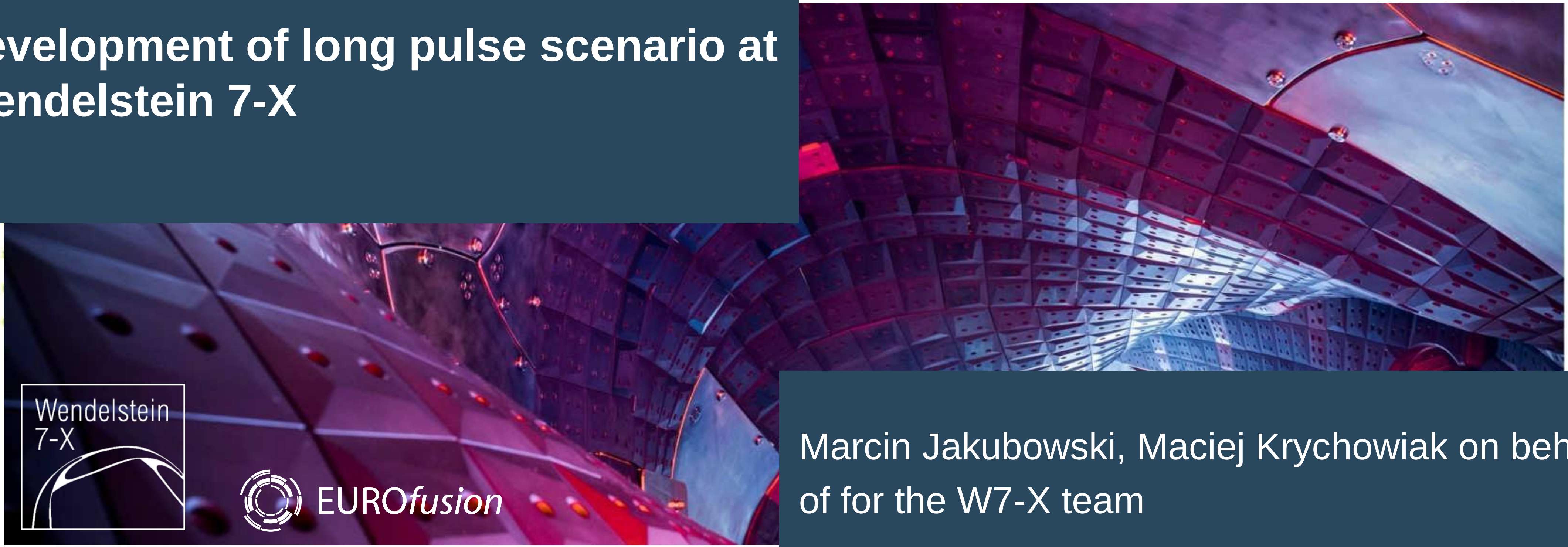
MAX-PLANCK-INSTITUT
FÜR PLASMAPHYSIK

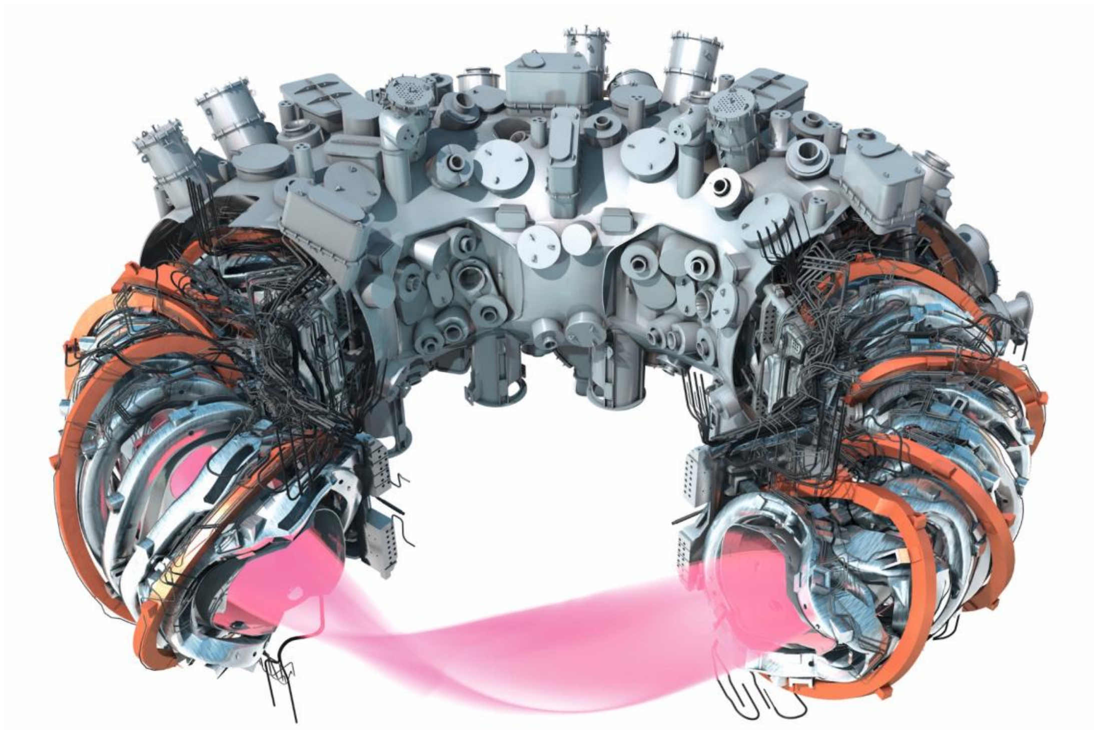
# Development of long pulse scenario at Wendelstein 7-X




Marcin Jakubowski, Maciej Krychowiak on behalf of for the W7-X team

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

# Optimised stellarator Wendelstein 7-X (W7-X)



*W7-X status/achievements (so far):*

- ✓ *4 experiment campaigns (2015-2023): OP1.1, OP1.2a/b, OP2.1*
- $P_{ECRH}$ = 8.5 MW (currently) (up to 30 min)
- $P_{NBI}$ = 6.6 MW (currently) (up to 10s)
- $P_{ICRH}$ = 1.0 MW (up to 10s)
- ✓ *Record plasma parameters close to NC[1] expectations:*
  - $T_i$ = 3.0 keV [2], $n_e$ = 1.4 x $10^{20}$ $m^{-3}$ [3]
  - $W_{dia}$ = 1.1 MJ [2], $\tau_E$ = 0.3 s [2]
- ✓ *Long pulse operation 480 s [4] (attached), stable detachment [5] for 120 s*
- ✓ *Plasma Facing Components developed for long pulse [6]*

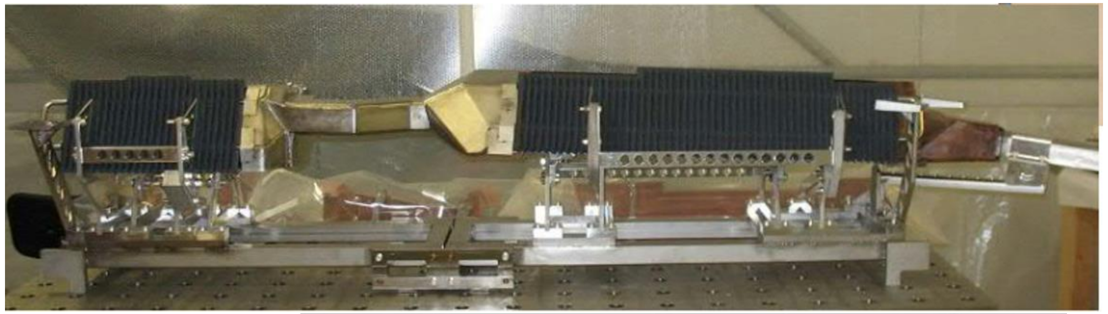

[1] C. Beidler, H. Smith, A. Alonso et al. *Nature* **596** (2021)
[2] S. Bozhenkov, Y. Kazakov, O. Ford et al. *Nuclear Fusion* **60** (2020)
[3] G. Fuchert, K. Brunner, K. Rahbarnia et al. *Nuclear Fusion* **60** (2020)
[4] H.-S. Bosch, P. van Eeten, O. Grulke et al. *Fusion Engineering and Design* **193** (2023)
[5] M.W. Jakubowski, et al. *Nucl. Fusion* ***61*** (2021)
[6] S. Brezinsek et al 2022 Nucl. Fusion 62 016006


*Demonstration of long pulse operation (30 min), extension of W7-X heating capabilities and good confinement of fast ions at high-beta envisaged for next W7-X campaigns*

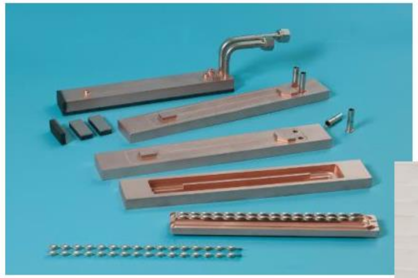
# Plasma facing components of Wendelstein 7-X

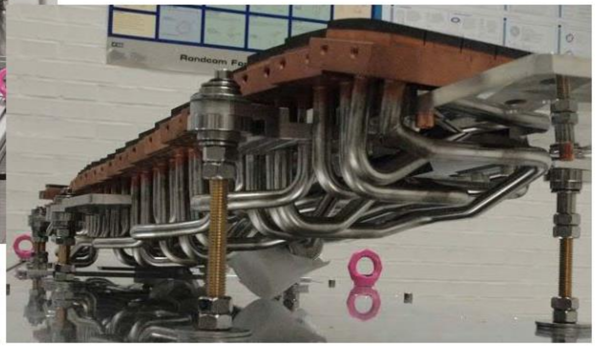
# Enhancements to Wendelstein 7-X before OP2

- Approximately 600 in vessel cooling circuits. All circuits proofed to be leak-tight and were hydraulically balanced.
- Steady-state pellet injector (comissioned in OP2.1)
- Ten identical cryopumps (CVPs) were installed in corresponding divertor volumes of Wendelstein 7-X
- All ten divertors were replaced by HHF divertors



- pressure water cooled divertor targets
- CFC blocks welded on CuCrZr heat sinks
- Steady state cooling with water

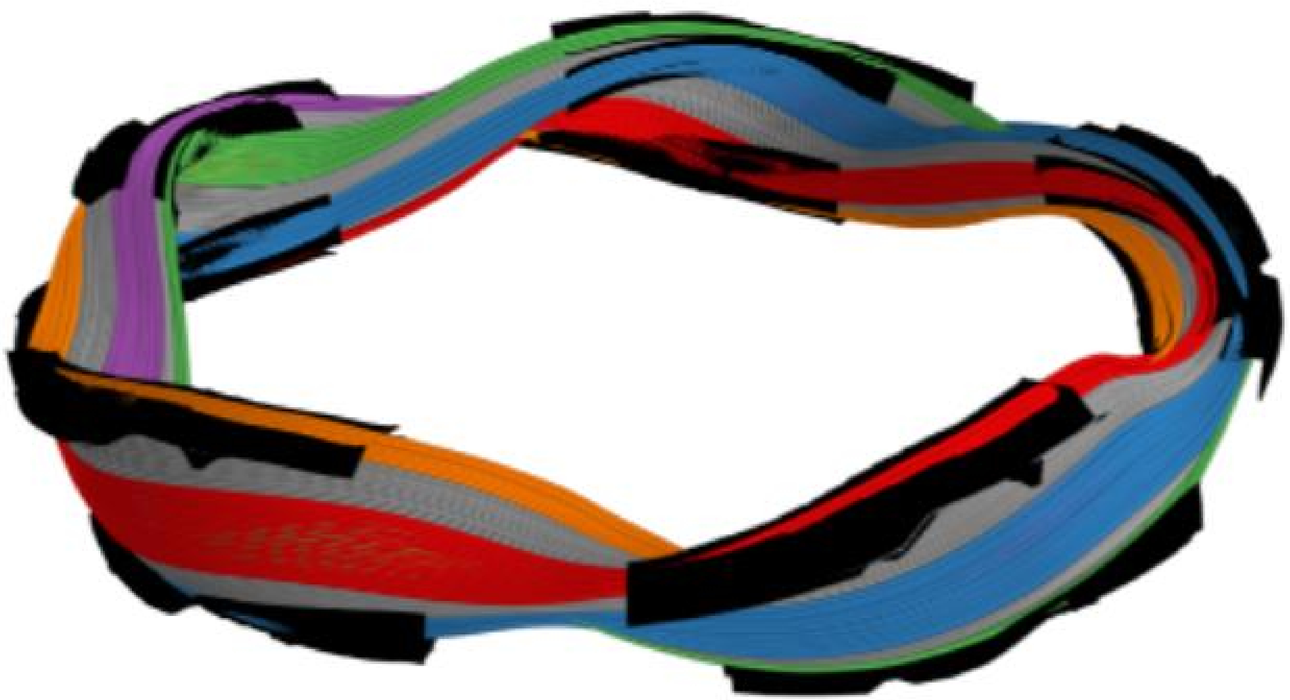




# The island divertor at W7-X

A low-order resonance placed in the plasma edge defines the edge island structure of W7-X:

Example of W7-X magn. *standard* configuration

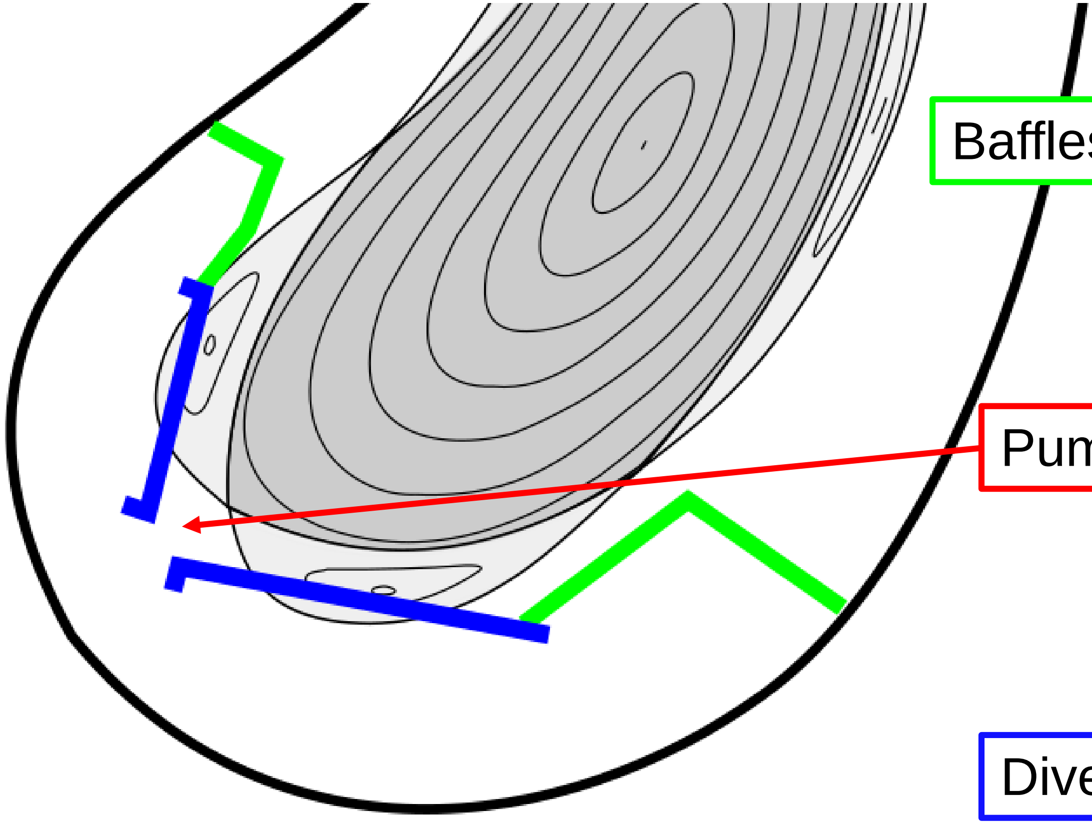


SOL with 5 independent islands (colorized)

**10 divertor units made of graphite (in black)**

Magnetic island twist around the core plasma and are intersected by divertor modules (made of carbon)

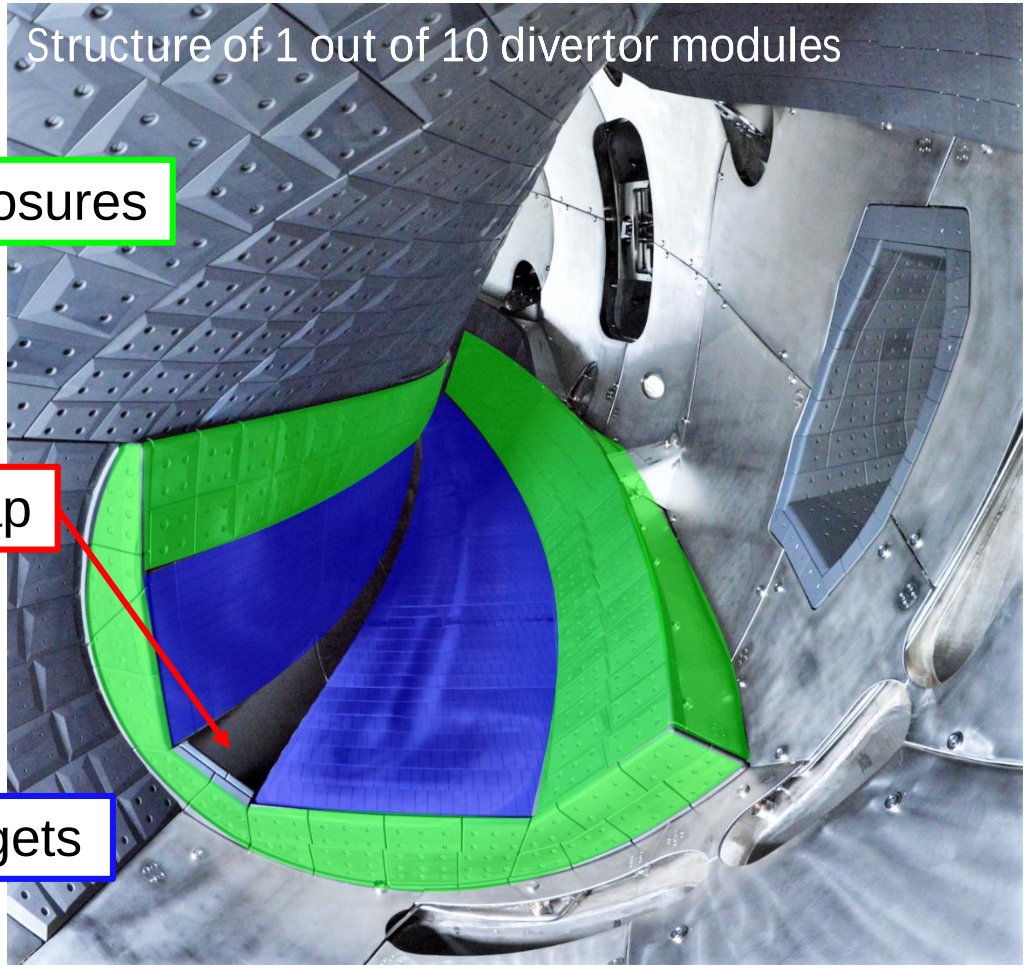
# The island divertor at W7-X – components



Baffles and closures

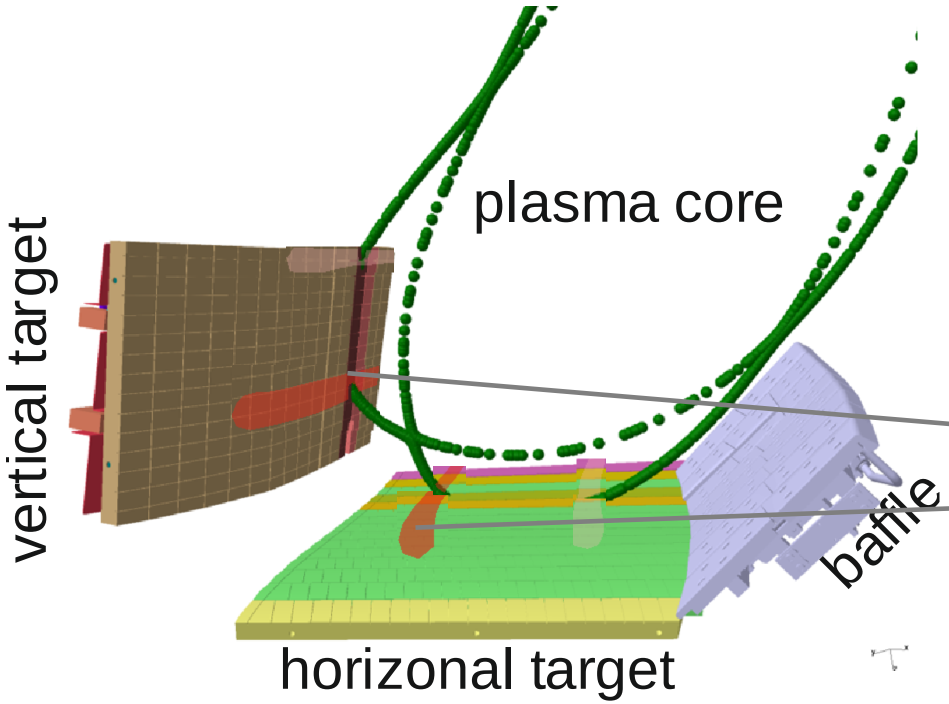
Pumping gap

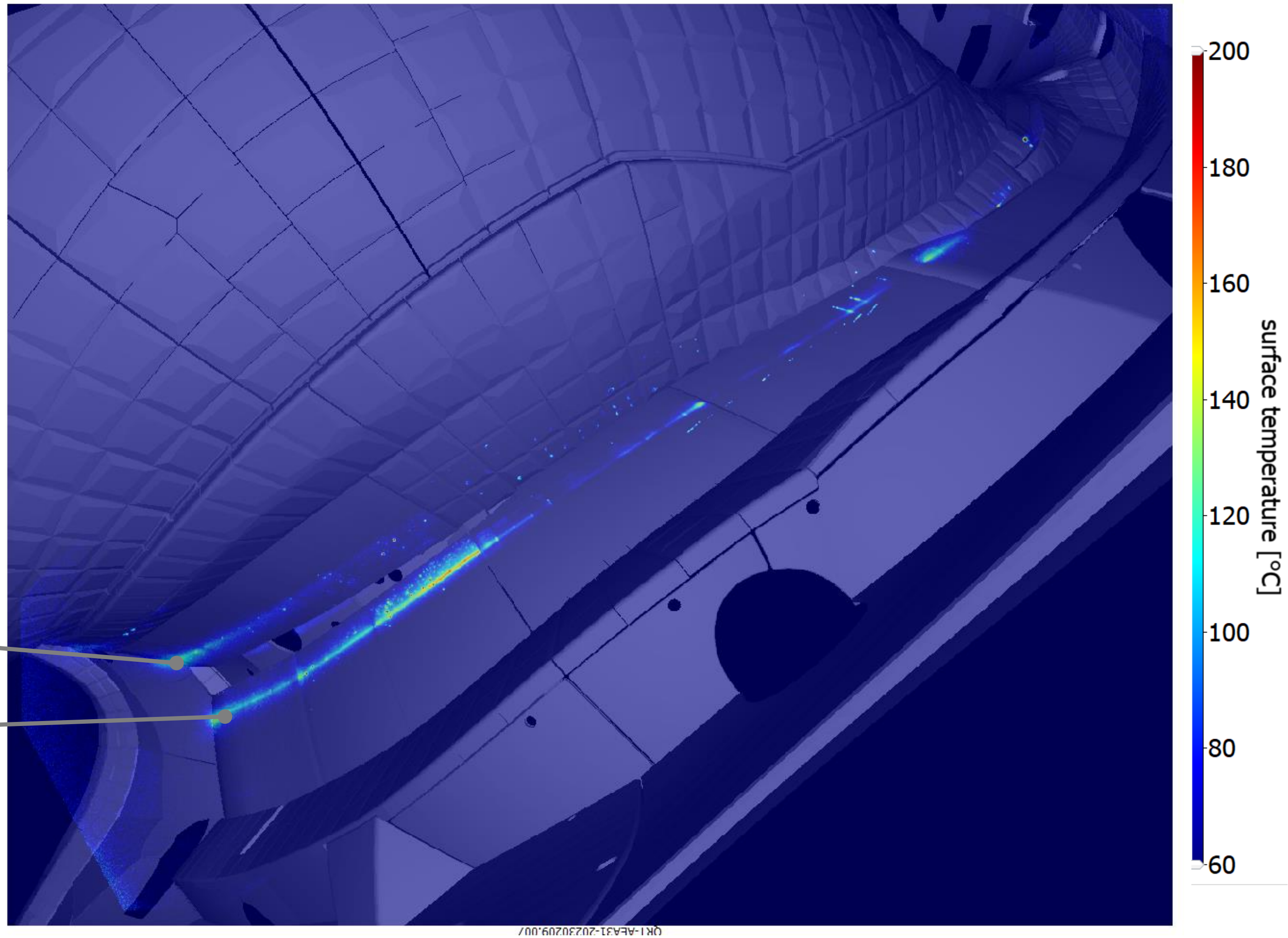
Divertor targets



# IR Thermography shows strike lines of island divertor (attached)

Island divertor at W7-X in **standard configuration** forms **two strike-lines** on horizontal and vertical target



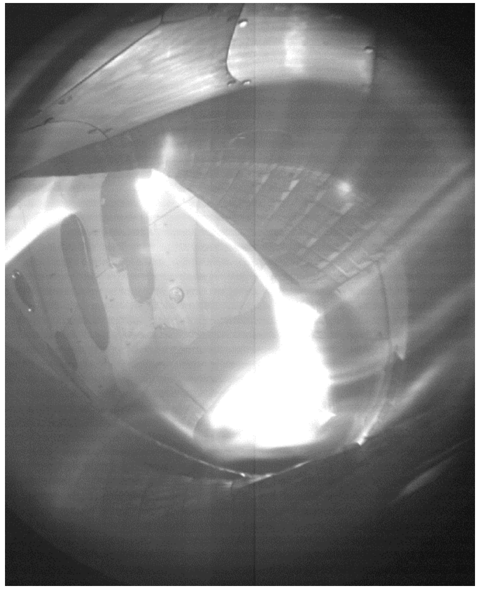



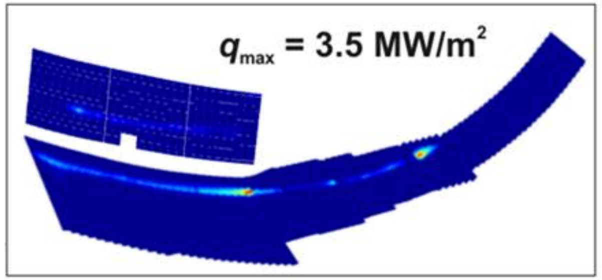
# Long pulse development at Wendelstein 7-X

- Future reactor will operate with power fluxes of up to 5 MW/m².
- In a stellarator-based 500 MW reactor > 80% of total heating power needs to be radiated away

→ W7-X with $P_{ECRH}$ = 8 MW cannot currently meet both requirements simultaneously:

## Attached plasma



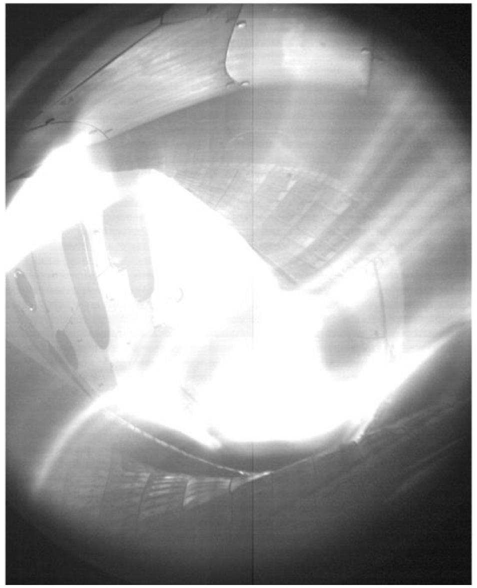



Low $P_{rad}$ $< 2\ MW$ →

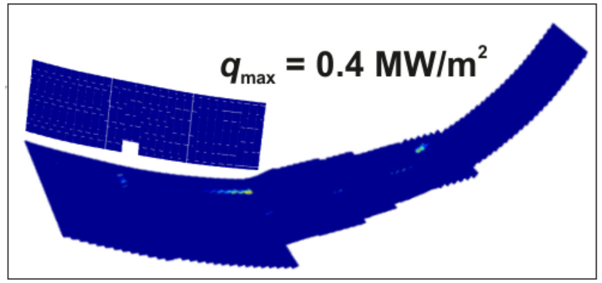
High $q_{peak} \sim 5\,\frac{MW}{m^2}$

Reactor-relevant power fluxes in attached plasmas

## Detached plasma



[Jakubowski (2021) NF 61 106003]




High $P_{rad}$ $=\geq 6\ MW$ →

Low $q_{peak} \sim 1\ MW/m^2$

Reactor-relevant divertor downstream parameters (e.g. $T_{e,downstream}$ ~ 5 eV and $f_{rad} \geq 0.8$)

# Outline

- (Introduction – The Stellarator W7-X)
- Detached long-pulse scenarios
  → *intrinsic detachment (w carbon)*
  → *seeded detachment (w neon)*
- Attached long-pulse scenarios
- Summary and Outlook

# Outline

- Introduction – The Stellarator W7-X
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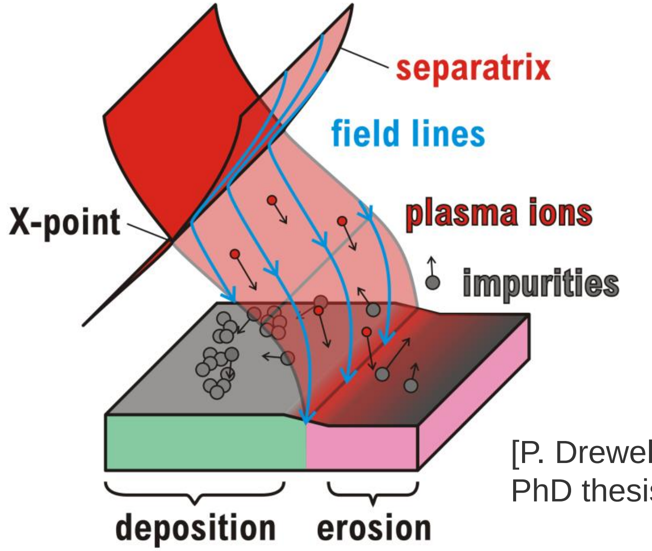
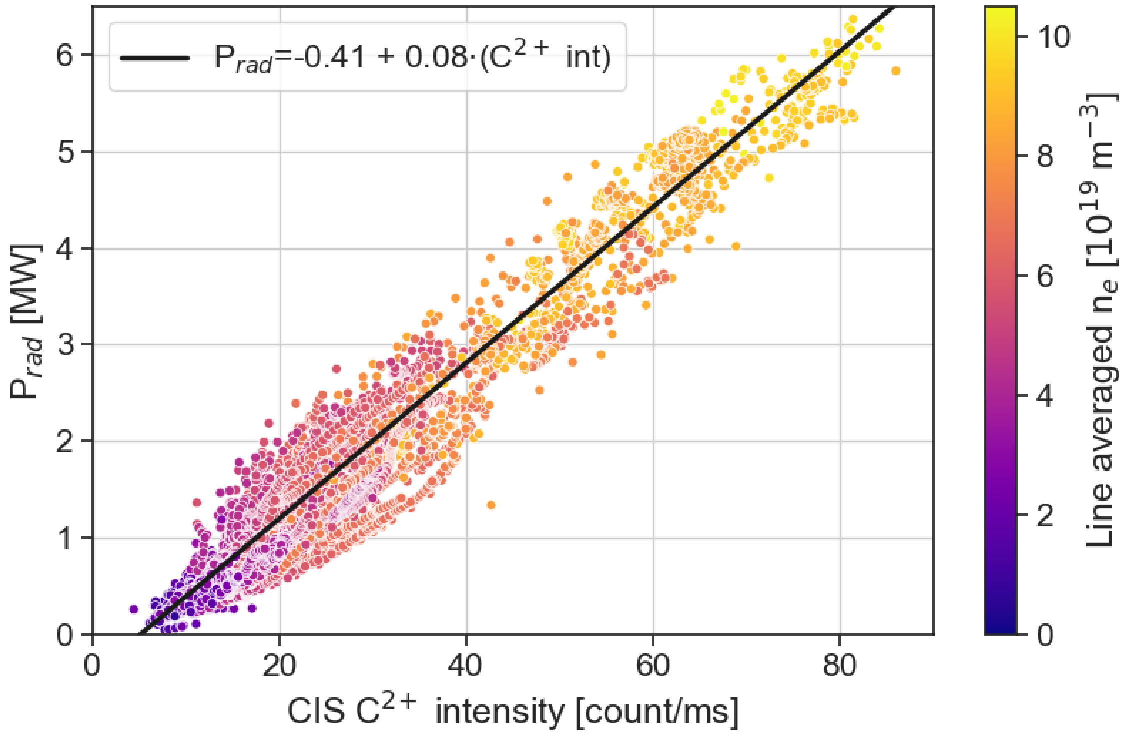
# Intrinsic detachment in the W7-X island divertor

- Obtaining detachment with help of intrinsic impurities (carbon) is rather straightforward at W7-X.
- Carbon is released from the plasma facing components




[P. Drewelow PhD thesis]

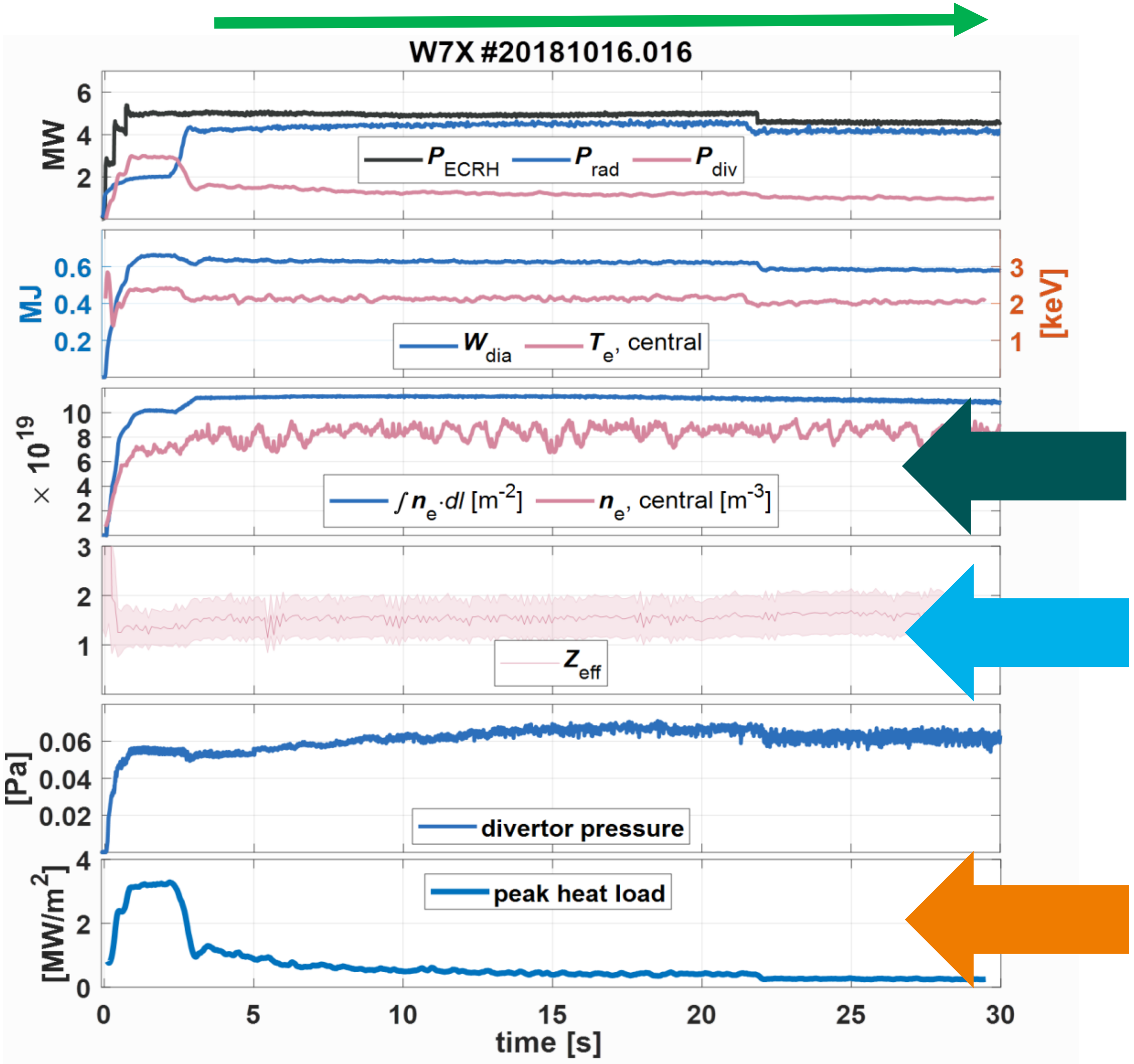
[V. Perseo *et al* 2021 *Nucl. Fusion* **61** 116039]




Plasma radiation proportional to C2+ intensity and plasma density → hydrogen fueling is our method to reach certain level of radiated power (and thus the SOL carbon concentration)

# Detachment can be obtained by raising the plasma density

- During OP1.2b (2018), about 30s of plasma duration were achieved with the divertor detached.
- In a typical W7-X detachment discharge, plasma line-integrated density is raised to a level above 1e20 m-2 which leads to frad > 0.8.
- Strong reduction of peak heat load and particle fluxes
- No significant increase in impurity concentration
- This state could be easily maintained with feedback for the whole plasma duration thanks to the PID controller. [M. Krychowiak, et al., NME, 34 (2023) 101363]



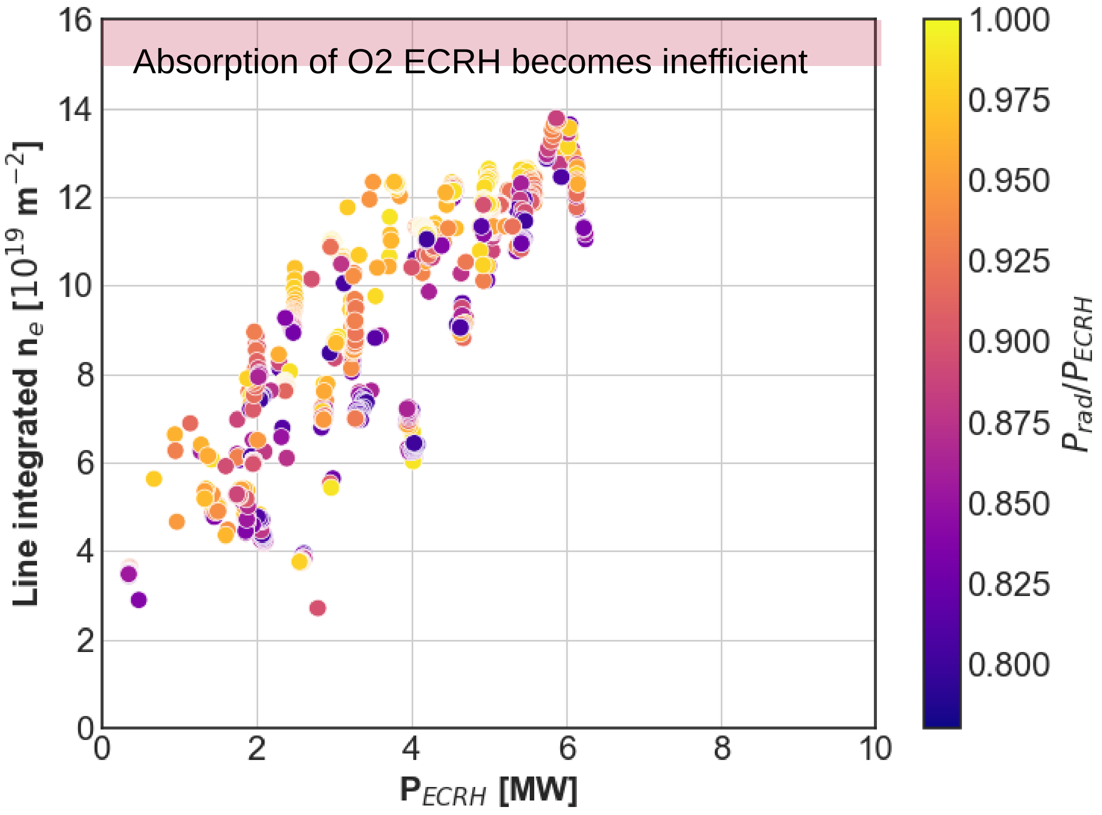

[O. Schmitz et al 2021 Nucl. Fusion 61 016026]
[M. Jakubowski et al 2021 Nucl. Fusion 61 106003]

# Intrinsic detachment in the W7-X island divertor

Right figure: Empirical density values required to reach detachment (frad > 80%)

Feedback system indispensable to “find” required (continuous) fueling level to reach required densities

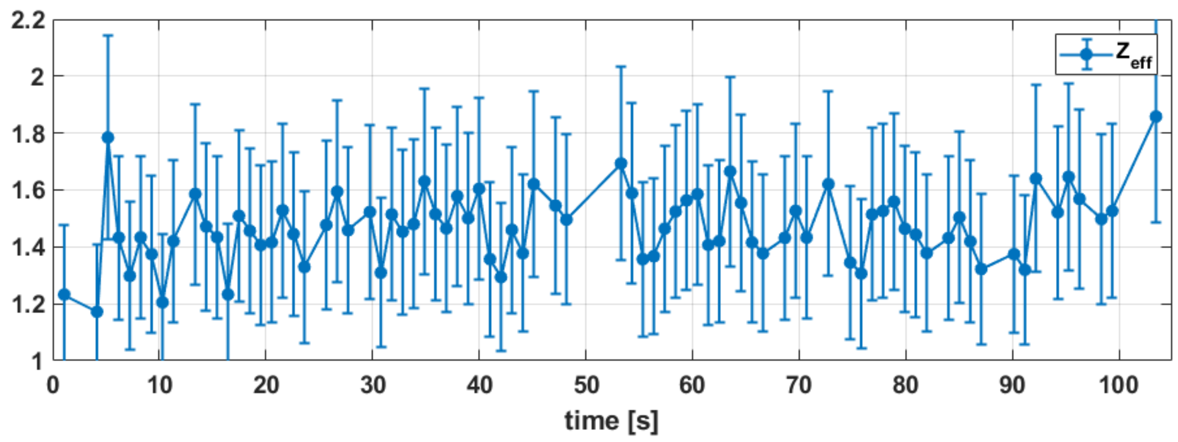
At too high ECRH power, the necessary density becomes too high to reach detachment
→ limit at ~16e19 m-2

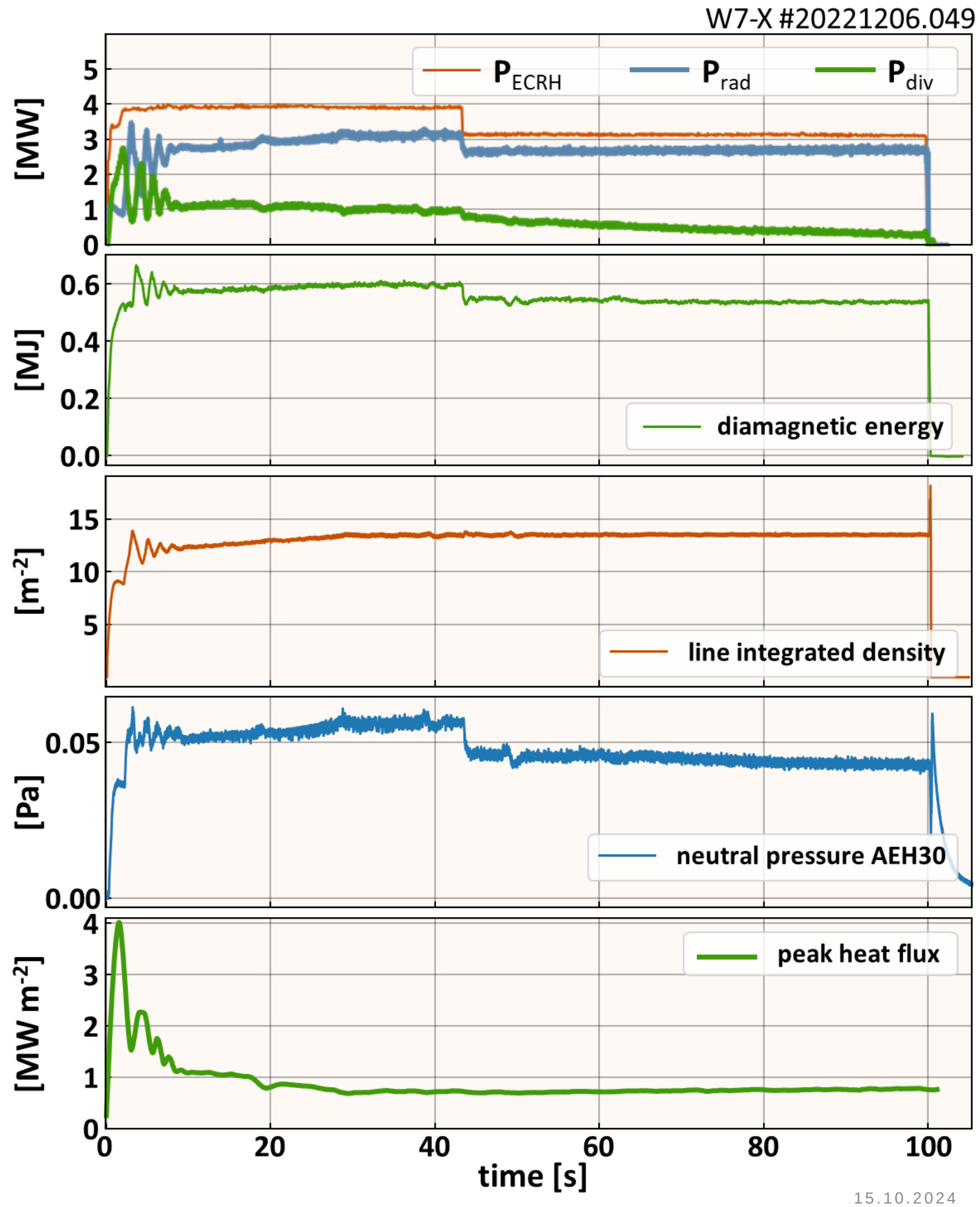



**So far, no problem in reaching (intrinsic) detachment in the explored power domain → at higher powers, intrinsic impurities not sufficient anymore, seeding required**

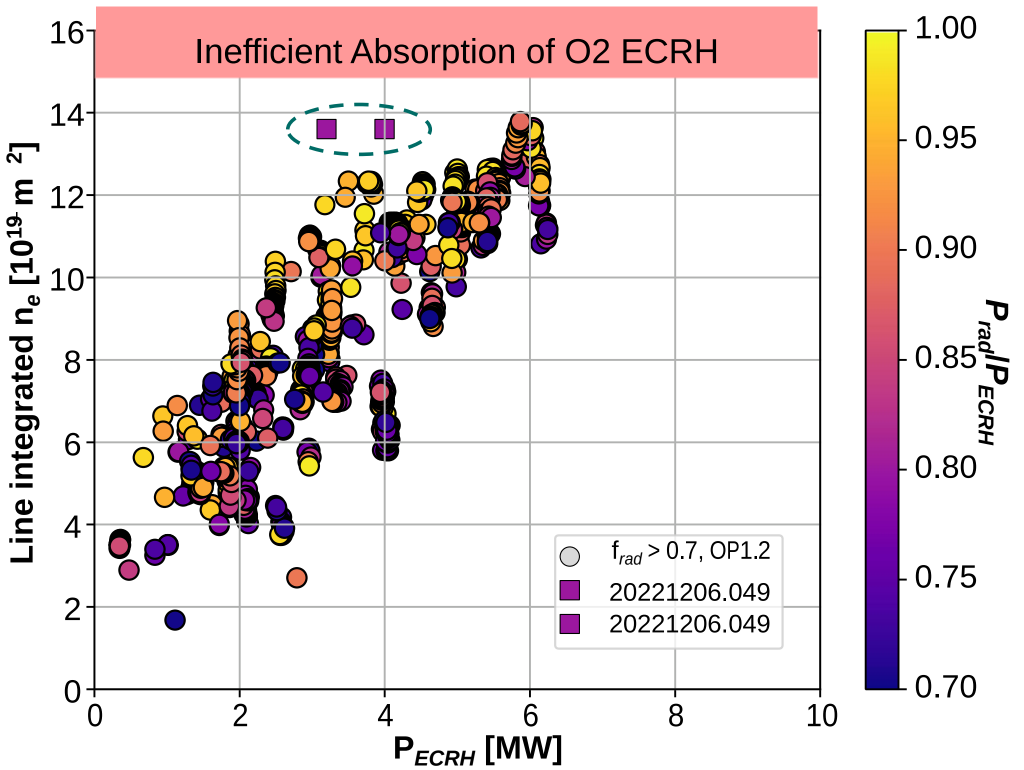
# Intrinsic detachment could be prolonged in OP2.1 to 100 s

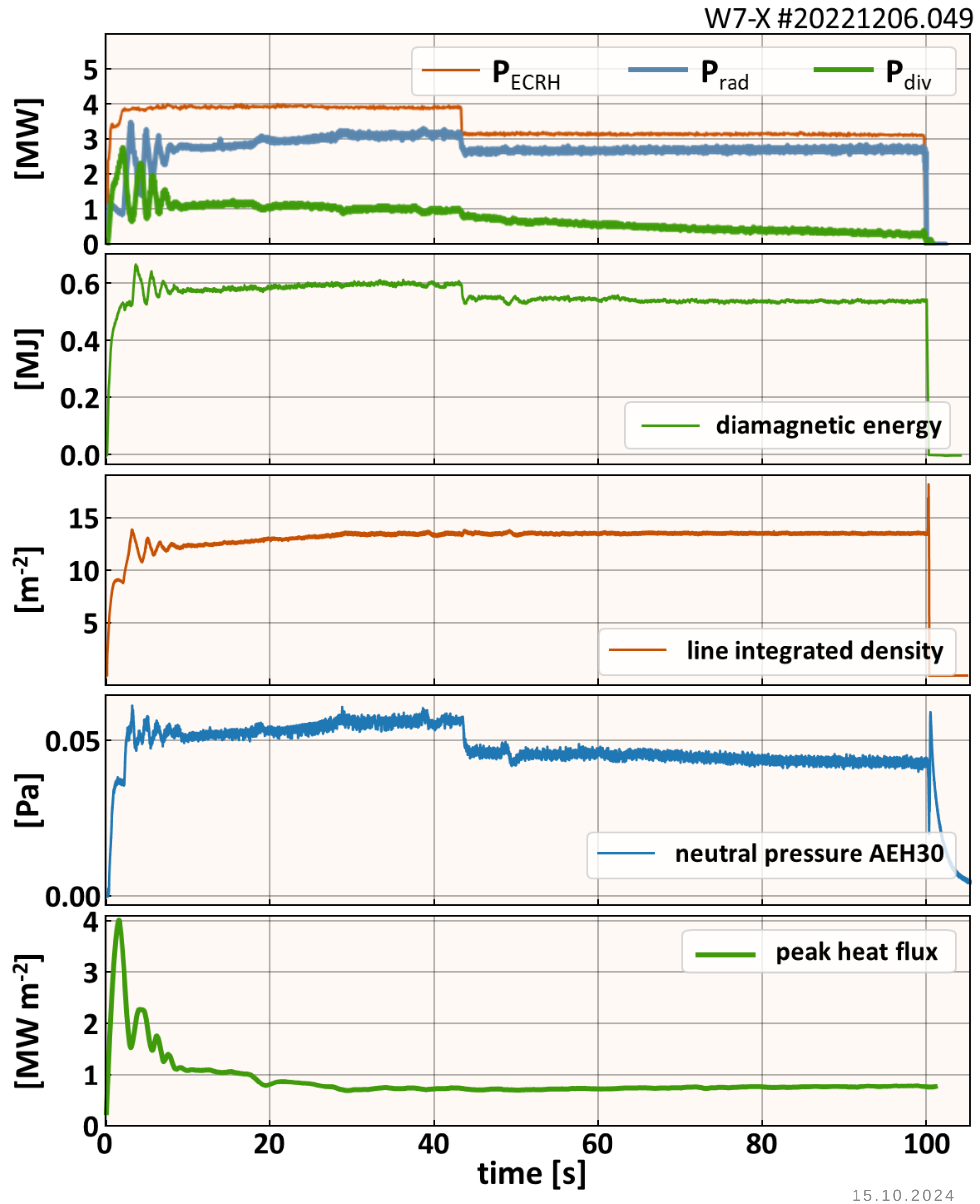
- This scenario could be succesfully prolonged in OP2, with 100 s long detached discharge
- **The feedback system was controlling plasma density and not plasma radiation.**
- Radiation fraction of ca. 0.8 required higher plasma density than in OP1.2
- No significant increase of core impurities either



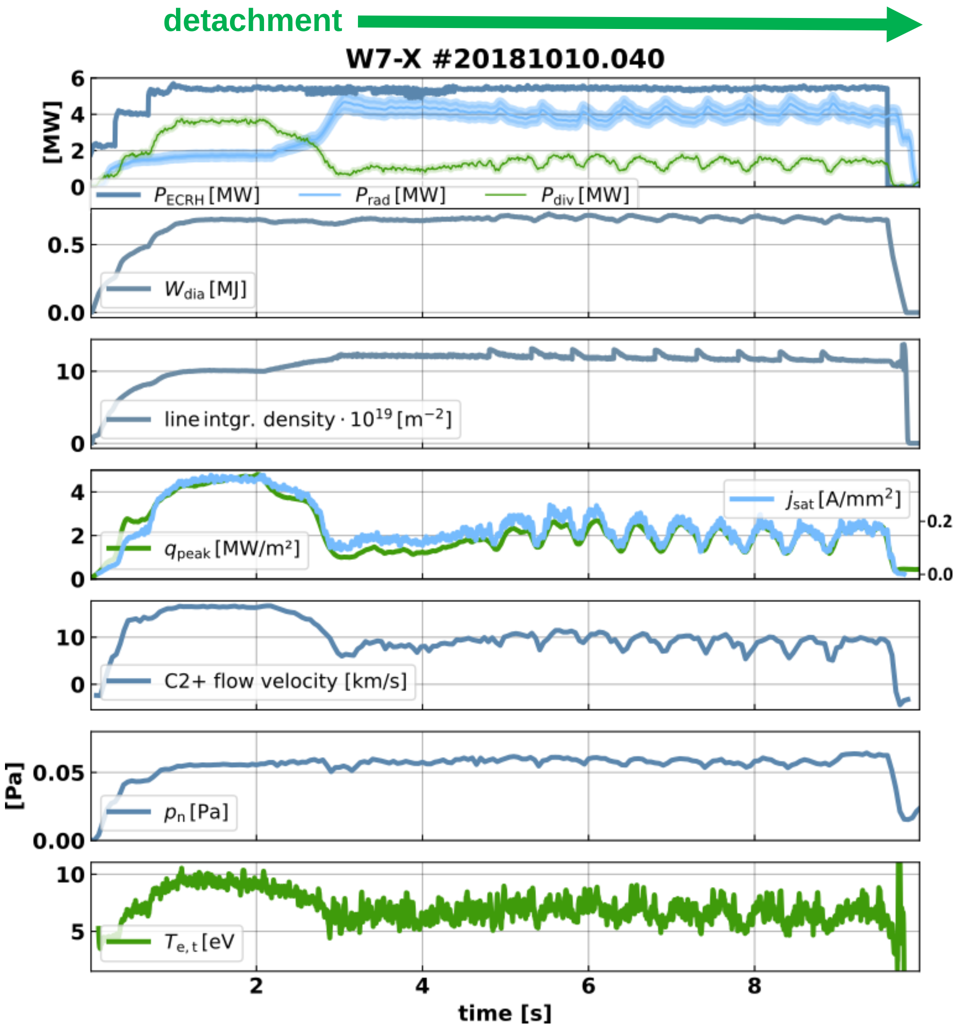
# Intrinsic detachment could be prolonged in OP2.1 to 100 s

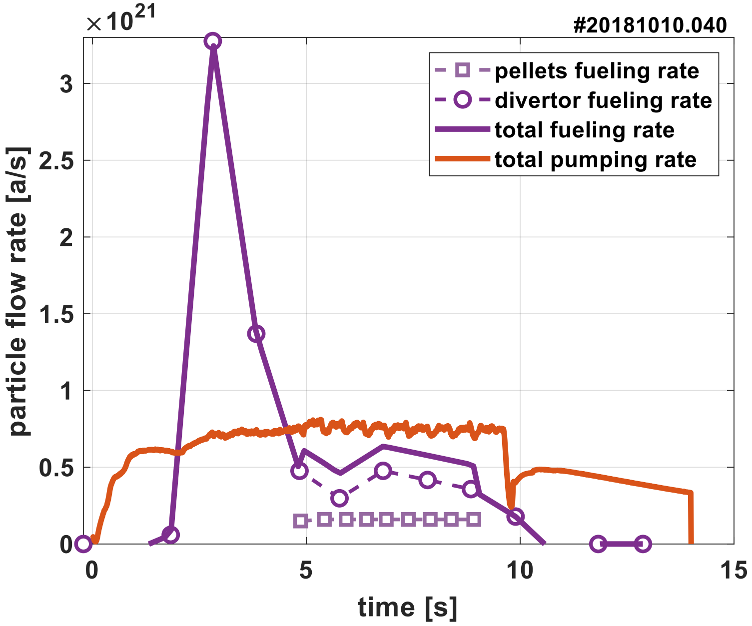



Scaling of the curve dependent on Zeff value → the smaller the impurity content (e.g. via boronisation or wall conditioning) the higher the necessary density

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



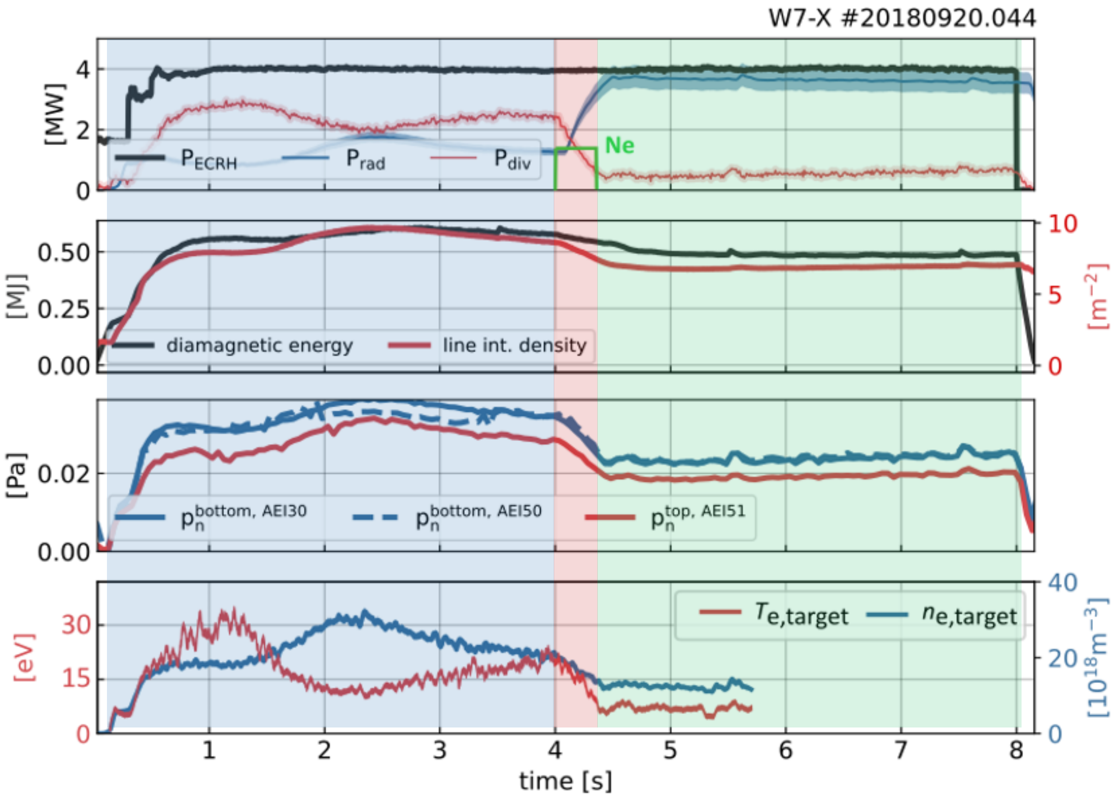

**New steady state pellet injector from 2024**

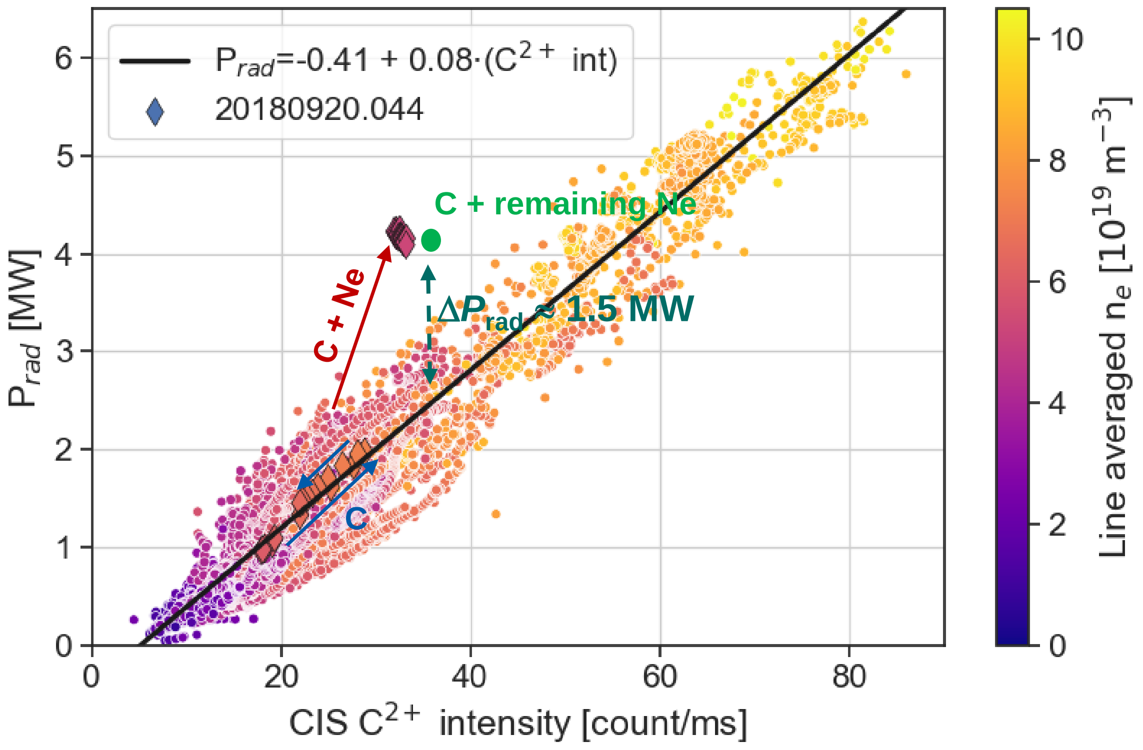
- pellet material: $H_2$ or $D_2$
- pellet size: 2mm – 3mm (adjustable)
- pellet speed: 250 – 1000 m/s
- repetition frequency:
  single on demand, continuous up to 10 Hz
- injection duration: up to 30 minutes

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

# Detached long pulse scenarios (seeded)

# Detachment via Neon seeding



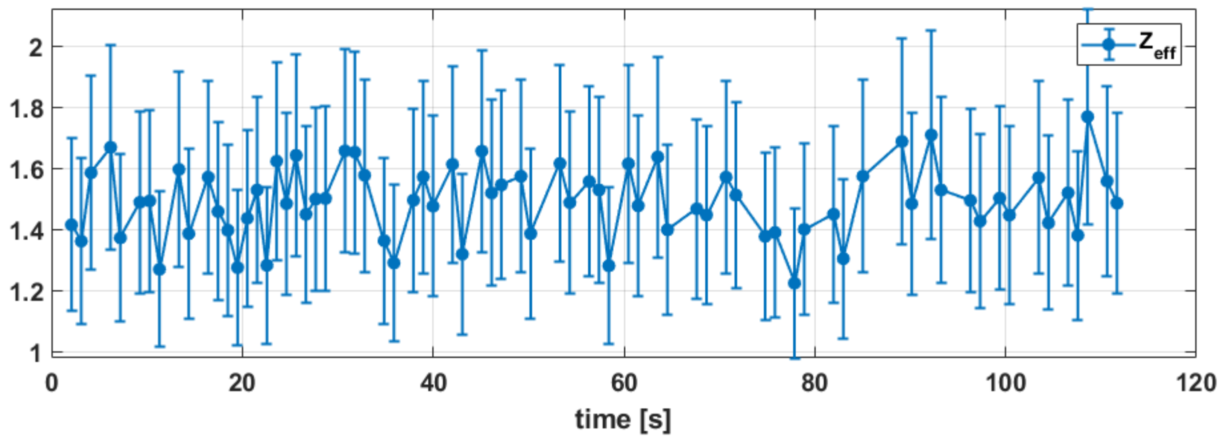



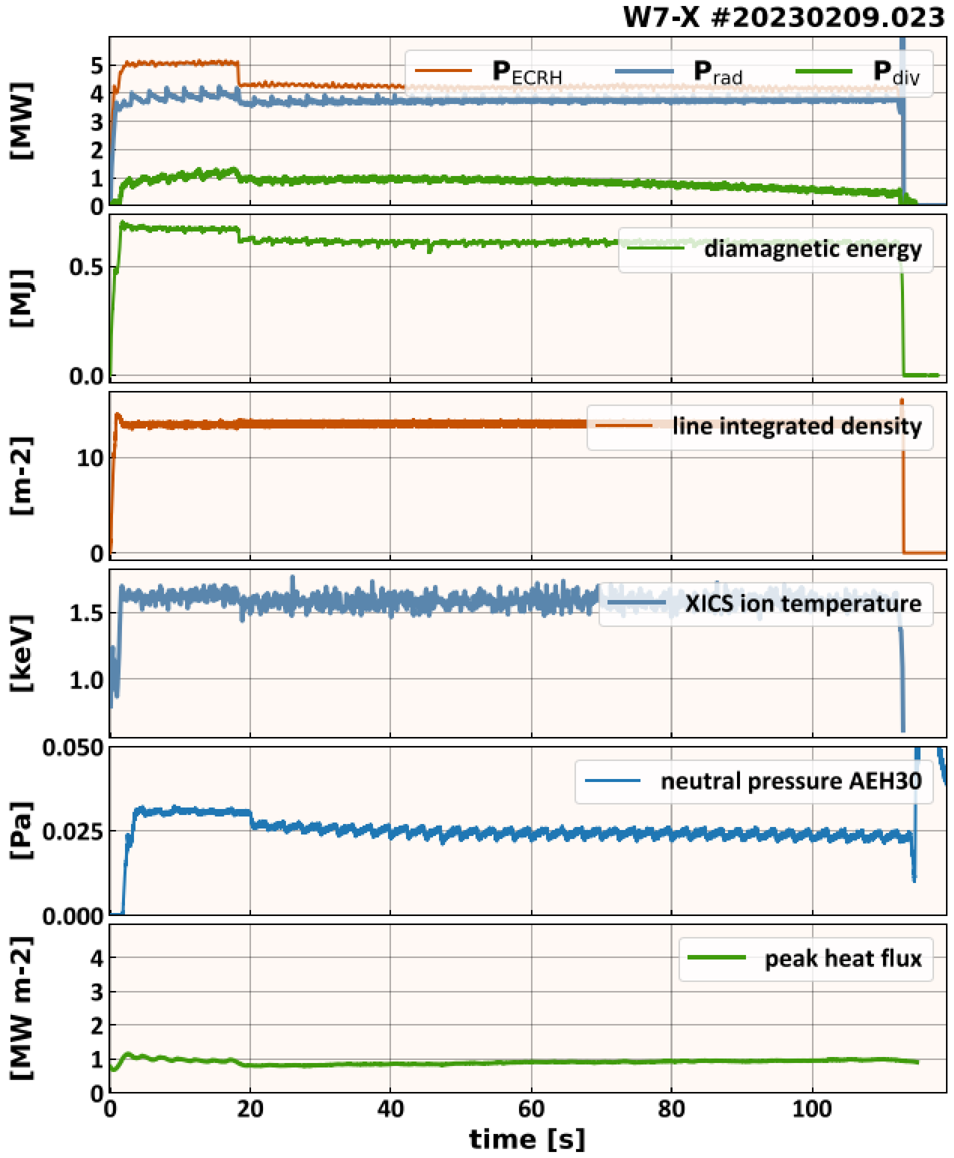

(2) Via Neon seeding, the $P_{rad} \propto I_{C2+} \propto n_e$ relation is broken, higher $P_{rad}$ achieved at lower density

(3) A global decrease of power loads achieved even though seeding occured in only one island; long-living due to high recycling of Ne

# Longest detached scenario with Ne seeding (OP2.1)

- Feedback on $f_{rad}$ not yet implemented, resulting in several attempts to find optimal Ne seeding settings for proper $P_{SOL}/P_{rad}$ to keep divertor and baffle in the safe temperature range
- Puff of ca. $8 \cdot 10^{17}$ Ne atoms every 2.5 seconds kept $P_{rad}$ baseline constant at $f_{rad}$ = 0.8. Gyrotrons dropped after t = 18 s, which further increased $f_{rad}$
- No impurity accumulation observed in core:



# Attached long pulse scenarios

# Record Heating Energy of 1.3 GJ (in attached state) in OP2.1

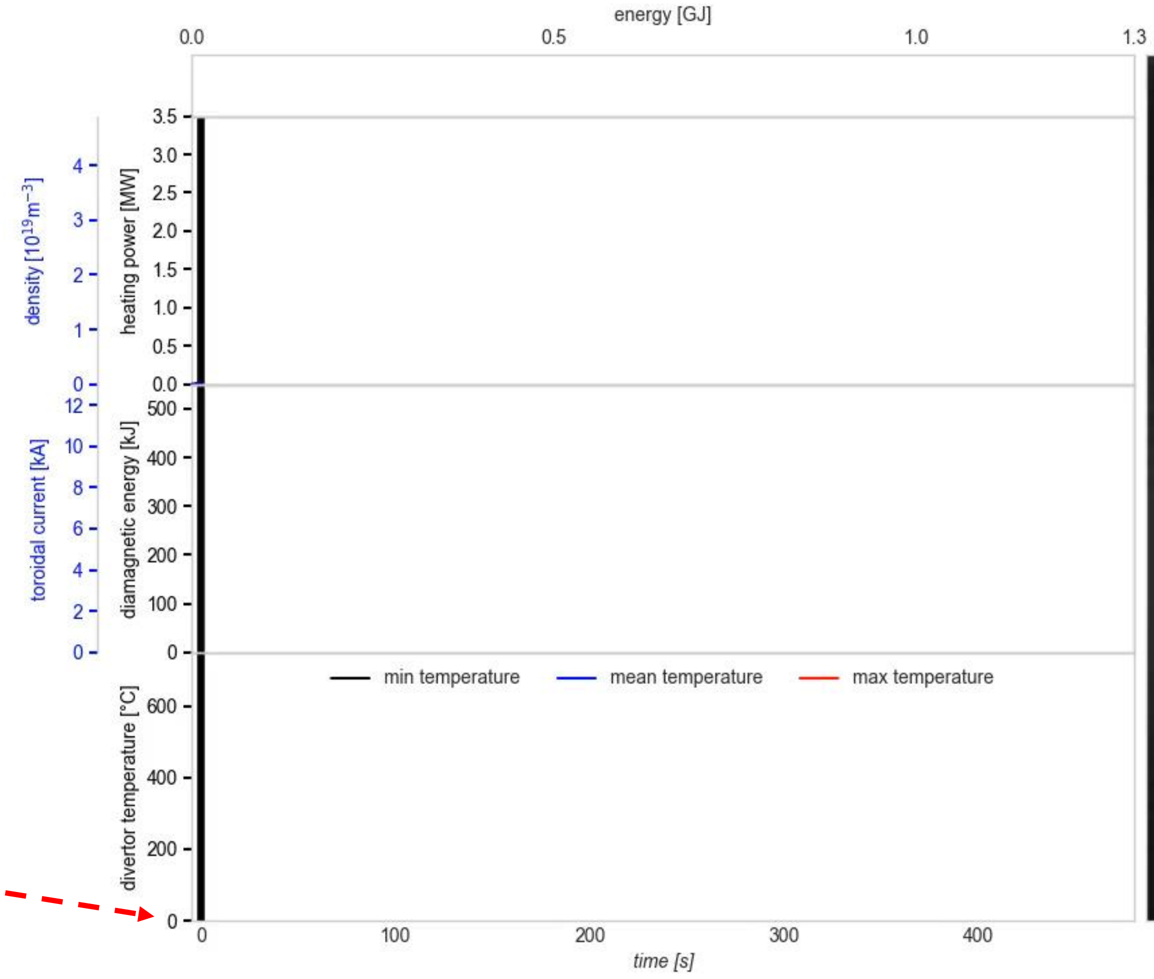
$W_{dia}$ ~ 430 kJ

$T_i$ ~ 1.5 keV

$T_e$ ~ 2 keV

$n_e$ ~ $5 \cdot 10^{19}$ 1/m²

**480 s plasma duration**




20230215.32


Line-integrated density pre-programmed to slowly decrease with the feedback system.

No overloading of the surface components facing the plasma observed.

# First Wall behavior in LPO

**Comparison of two long pulses with different divertor scenario** [G. Schlisio, et al., to be published]

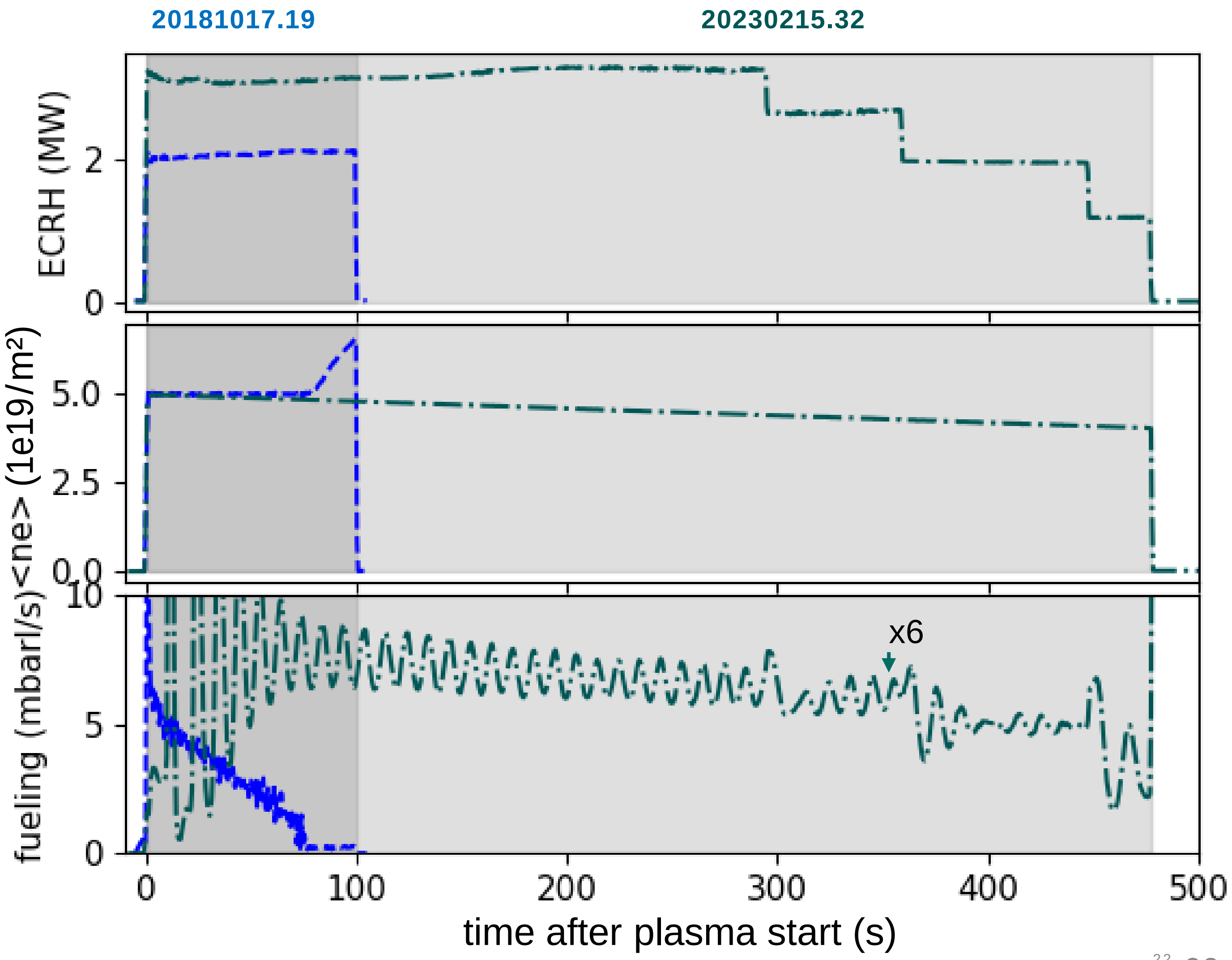
- Both are:
  - ECRH only
  - Run with feedback-controlled $H_2$ fueling

Scenario 1: (20181017.19) OP1.2b

- 100s plasma duration
- **Inertially cooled divertor**
- Diminishing fueling
- Density runaway after 80s

Scenario 2: (20230215.32) OP2.1

- 480s plasma duration
- **Water-cooled divertor**
- Near-constant fueling-density-ratio
- Stable density



# First Wall behavior in LPO

[G. Schlisio, et al., to be published]

**Inertially cooled divertor:**

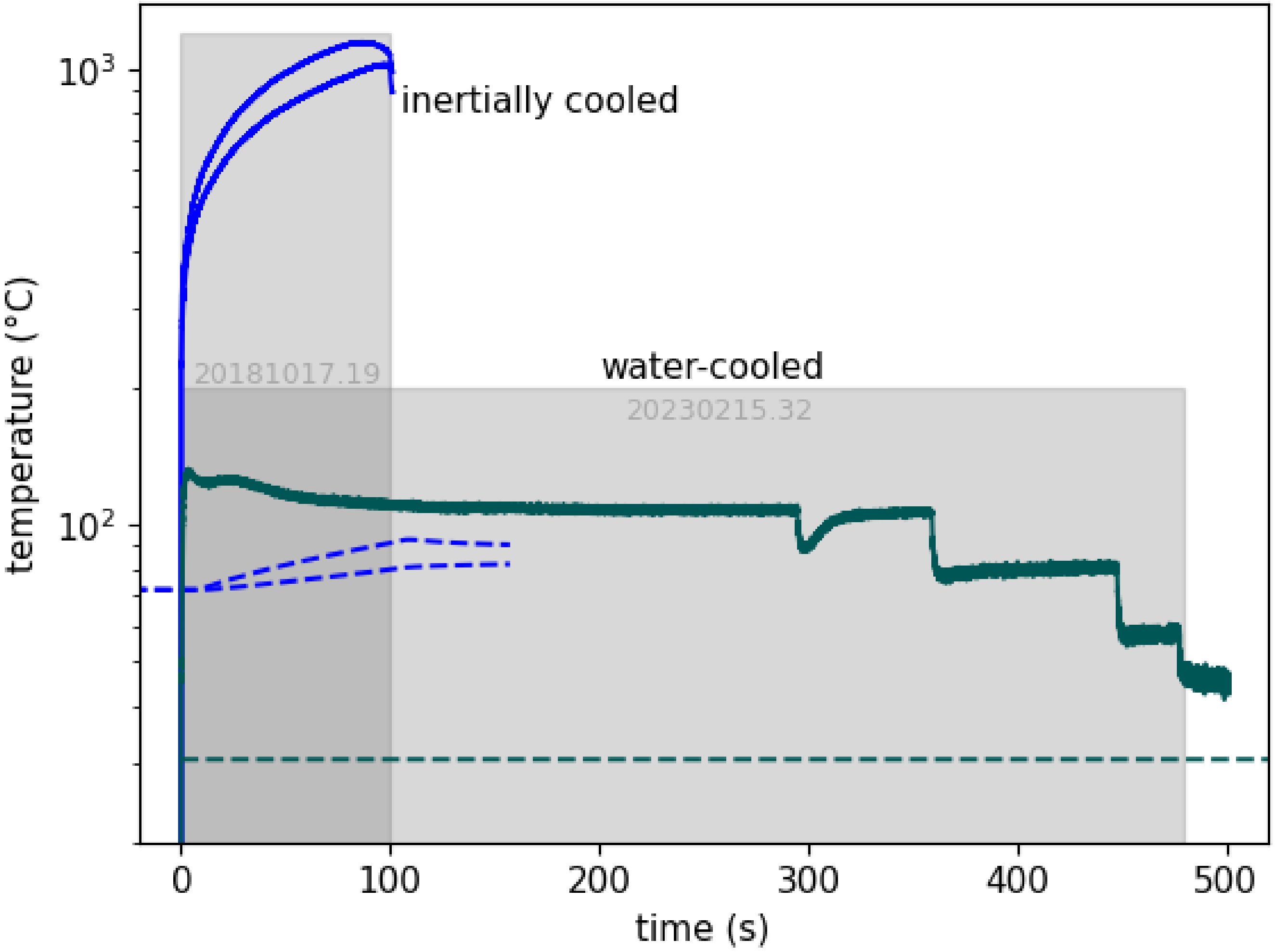
- Surface temperature (solid) raises above 1000 °C
- Bulk temperature starts to follow (dashed)

→ tile is heating up – outgassing

**Water-cooled divertor:**

- Surface temperature (solid) stabilizes very quickly
- Cooling water temperatures virtually unchanged (dashed)

→ tile in thermal equilibrium – no significant outgassing

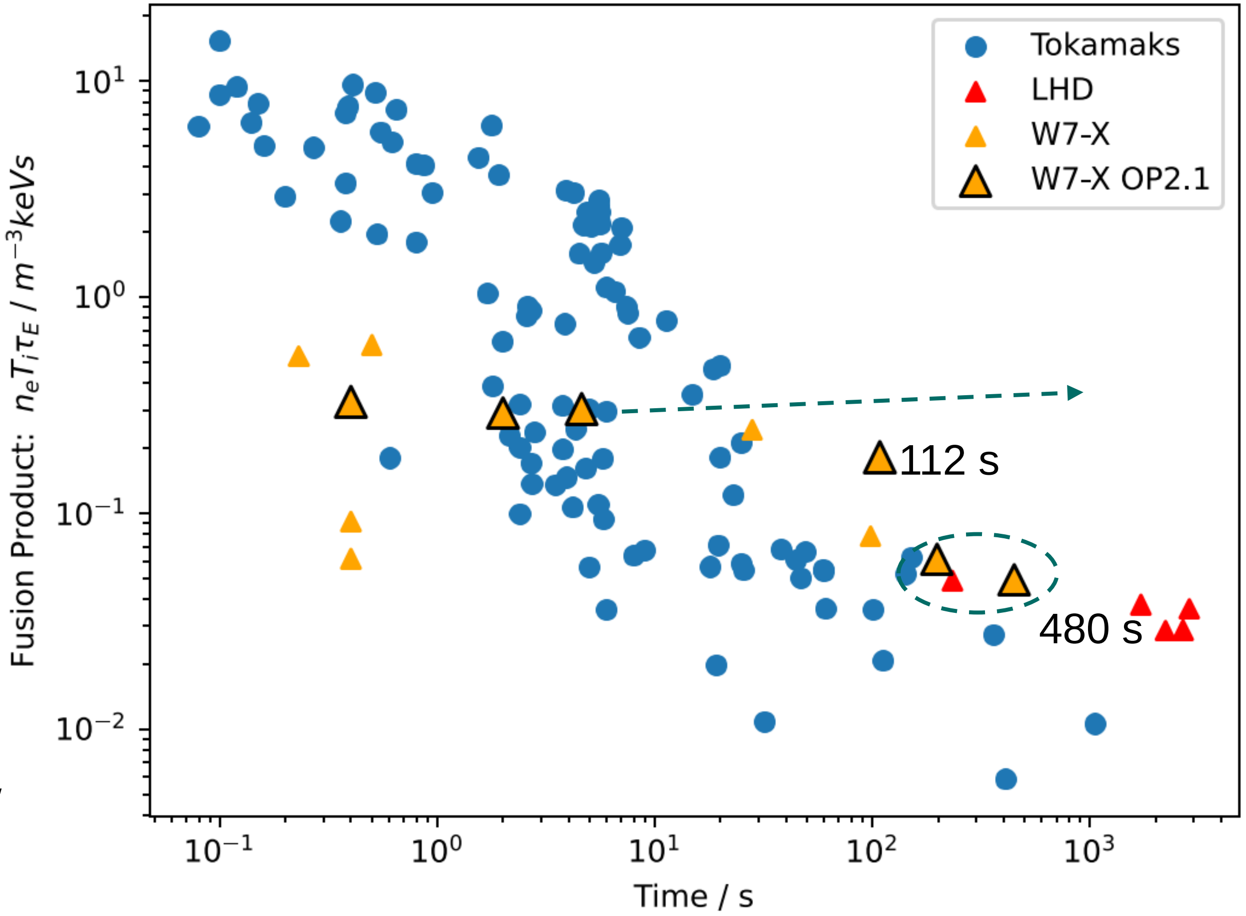



**Divertor cooling and Feedback system crucial for steady LPO with significant particle fluxes**

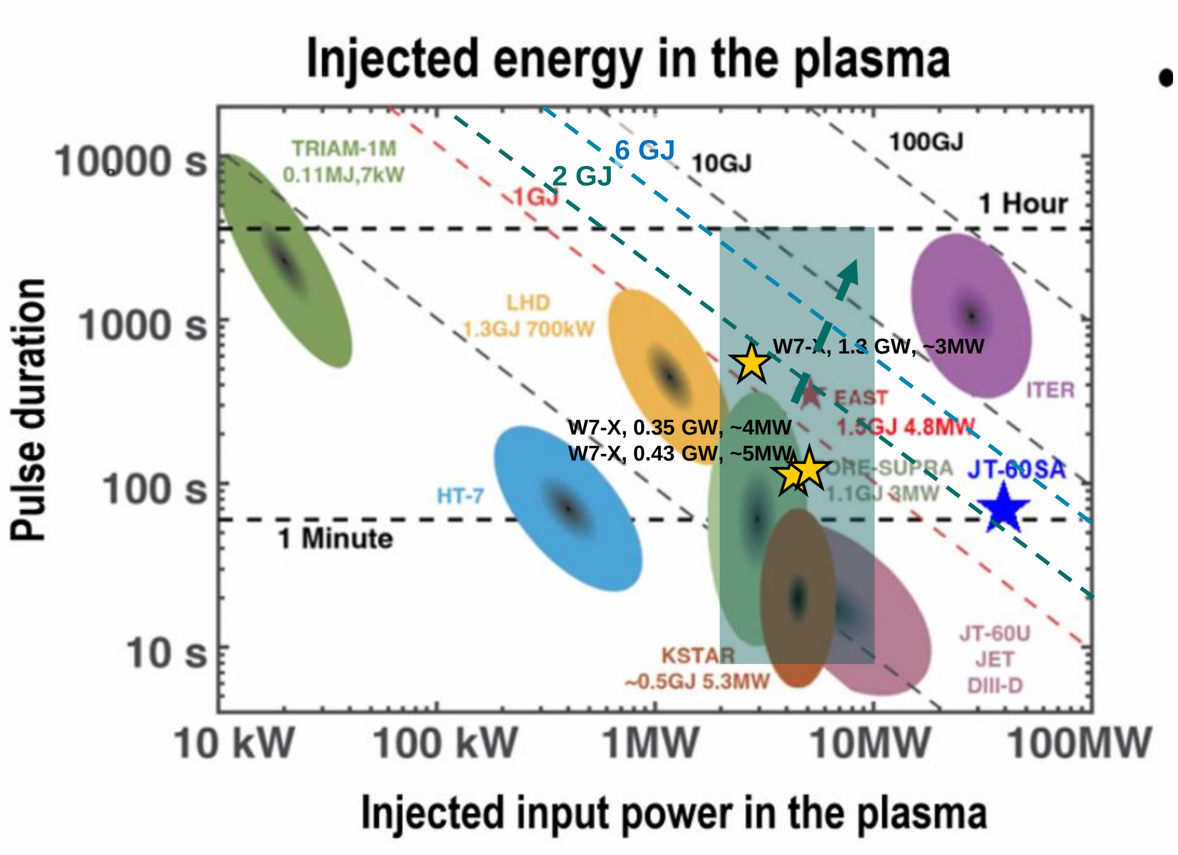
# Summary

- First campaign with water-cooled plasma facing components (incl. HHF divertor) allowed to gain first experience with long pulse scenarios:
  - 8 minute attached plasma
  - >100 s detached plasma
  - Future developments need to enhance the plasma core performance in LPO (w e.g. steady-state pellet injector)
  - Extension of heating system and feedback control ongoing
  - Develop (detached) scenarios with high plasma radiation (impurity seeding) and low core impurity contents

[X. Litaudon, et al., Nucl. Fusion 64 (2024) 015001]



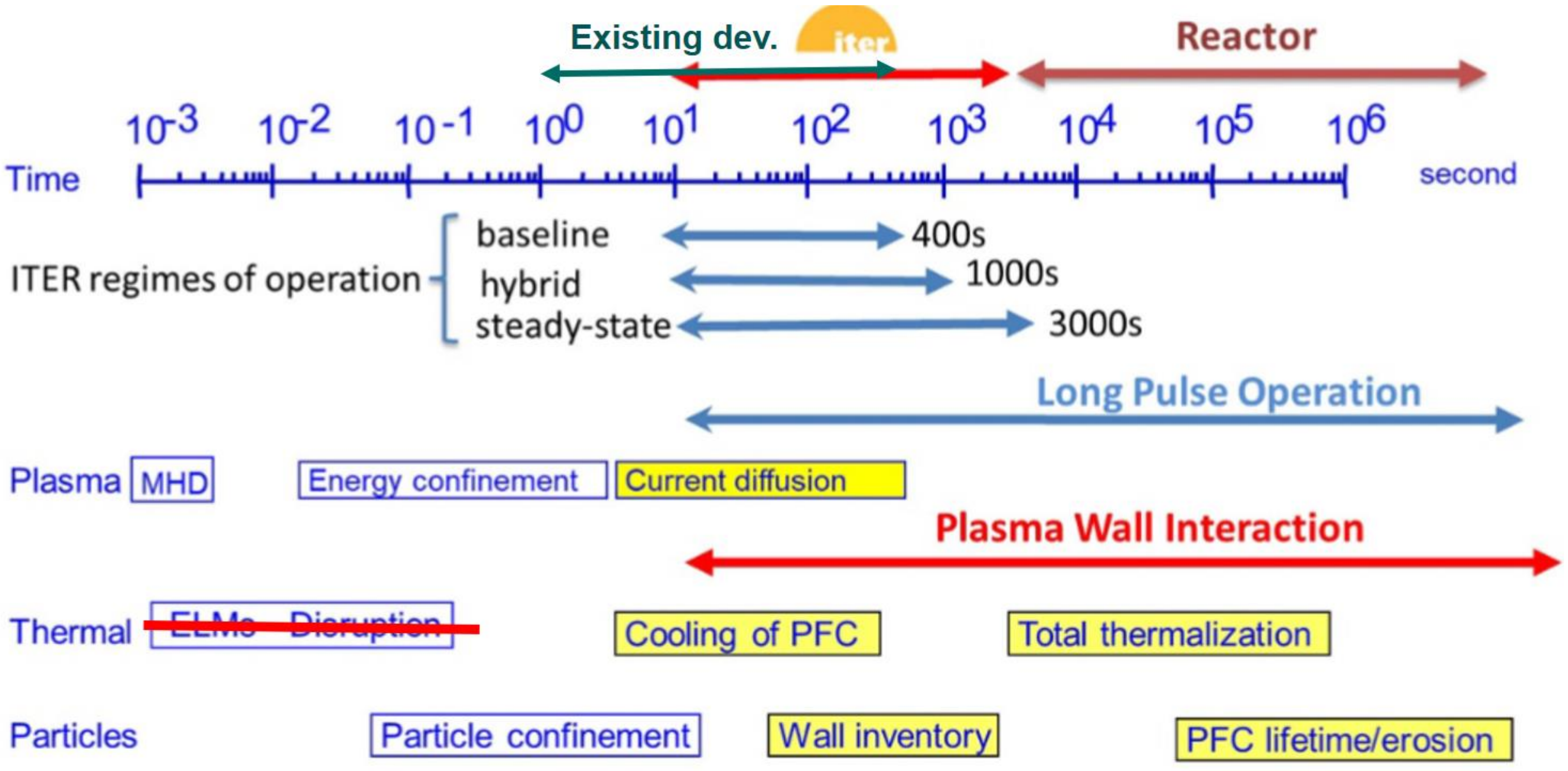
# Outlook




**W7-X in prime position to further explore LPO with relatively high injected input power in future campaigns (2GJ in OP2.2/2.3, at least 6 GJ planned for OP2.4)**

# Time scales in magnetic confinement fusion

Extending plasma pulse length is essential for future reactors. However, Long Pulse Operation (LPO) is often called the 'Grand Challenge' for fusion science, requiring pushing the limits of physics and technology integration in a nuclear environment for fusion reactor applications.




[X. Litaudon, et al., Nucl. Fusion 64 (2024) 015001]