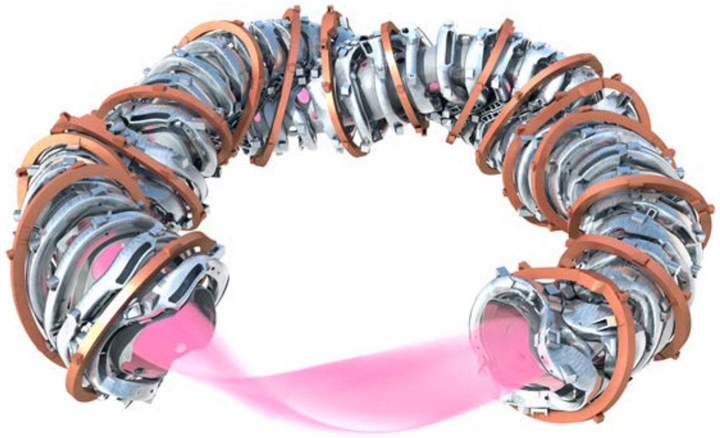

Plasma-Wall Interactions in Wendelstein 7-X Operating with Graphite Divertor

27.08.2020 I S. BREZINSEK AND THE W7-X TEAM

30th SPIG conference | Beograd | 25-28th August 2020

EUROfusion

HELMHOLTZ SPITZENFORSCHUNG FÜR GROSSE HERAUSFORDERUNGEN

NIFS NINS National Institutes of Natural Sciences WISCONSIN UNIVERSITY OF WISCONSIN–MADISON PPPL LPP ERM - KMS Wendelstein 7-X IPP JÜLICH Forschungszentrum

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* see T. Klinger et al., Nucl. Fusion 59 (2019) 112004

Wendelstein 7-X: Optimised Stellarator Design

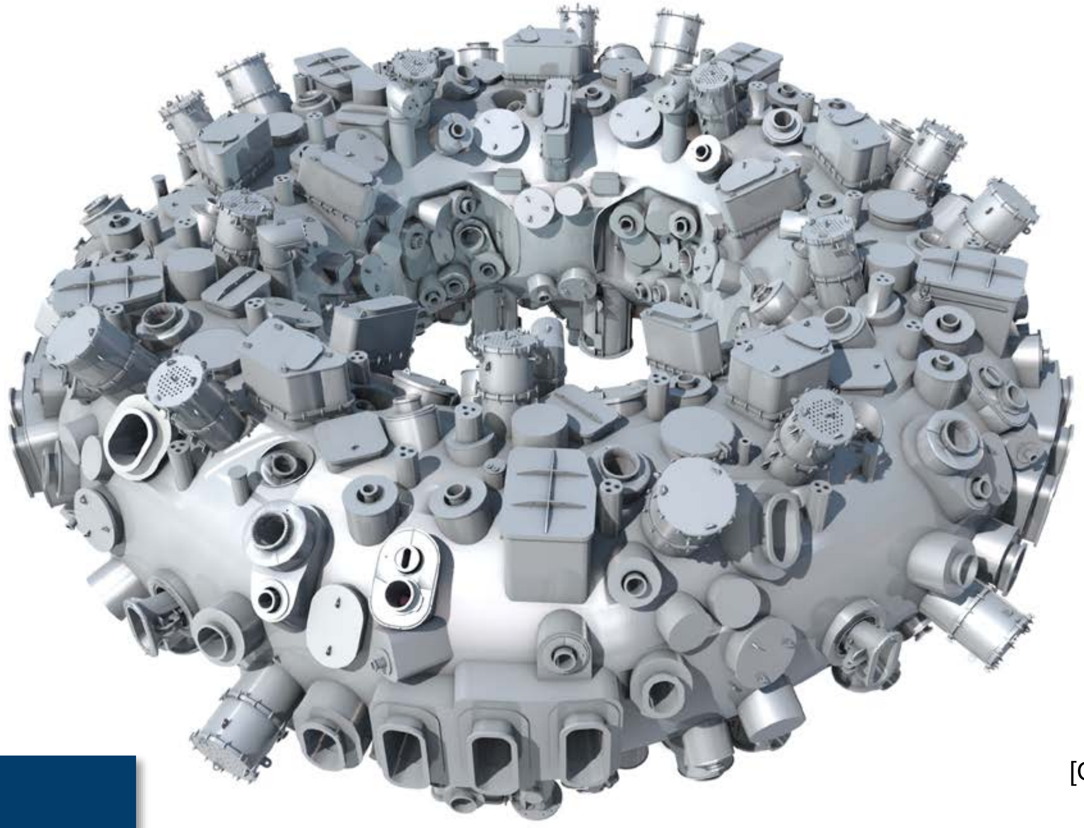
plasma volume: 30 m³

non-planar NbTi coils: 50

planar NbTi coils: 20

plasma vessel: 80 m³
in-vessel components: 265 m²

ports: 254
shapes: 120



machine height: 4.5 m
machine diameter: 16 m
device mass: 735 t
cold mass at 3.4 K: 435 t

[O. Grulke et al. EPS2019]

NbTi bus bars: 113
central support ring elements: 10

cryostat vessel insulation: 420 m³

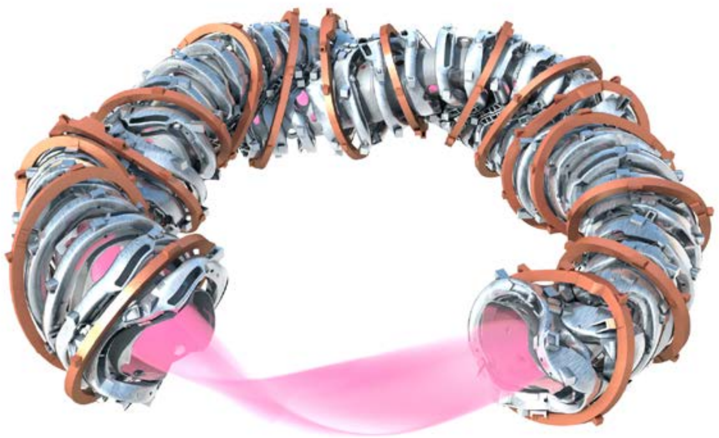
Wendelstein 7-X: Capabilities

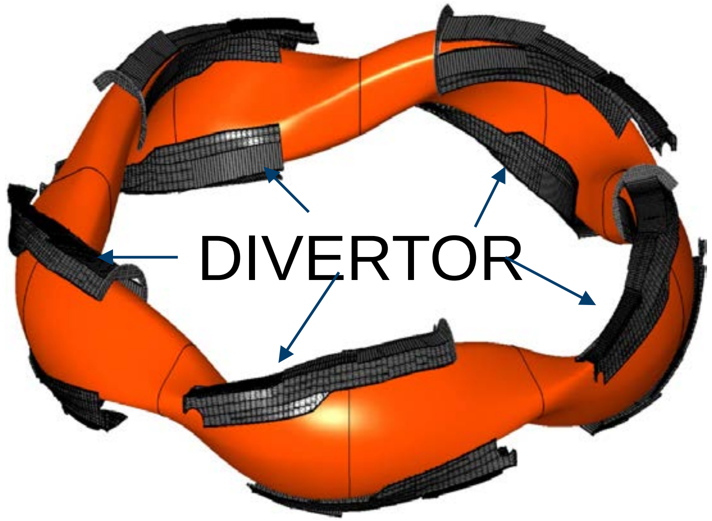
- Magnetic field of **2.5 T**
- Major Radius: **5.5 m** / Minor Radius: **0.53 m**
- Heating power: **7.5 MW ECRH / 3.4 MW NBI**
- Plasma volume: **30 m^3**

Island divertor to optimise power and particle exhaust as well as to screen impurities

Divertor operation

- Complete set of in-vessel PFCs
- No active cooling of graphite PFCs
- Divertor OP A: **~3776 s (He+H plasma)**
- Divertor OP B: **~9054 s (mainly H plasma)**
- Plasma duration: **≤100 s**
- Input energy: **≤0.2 GJ**





[T.S. Pedersen et al. PPCF 2019]

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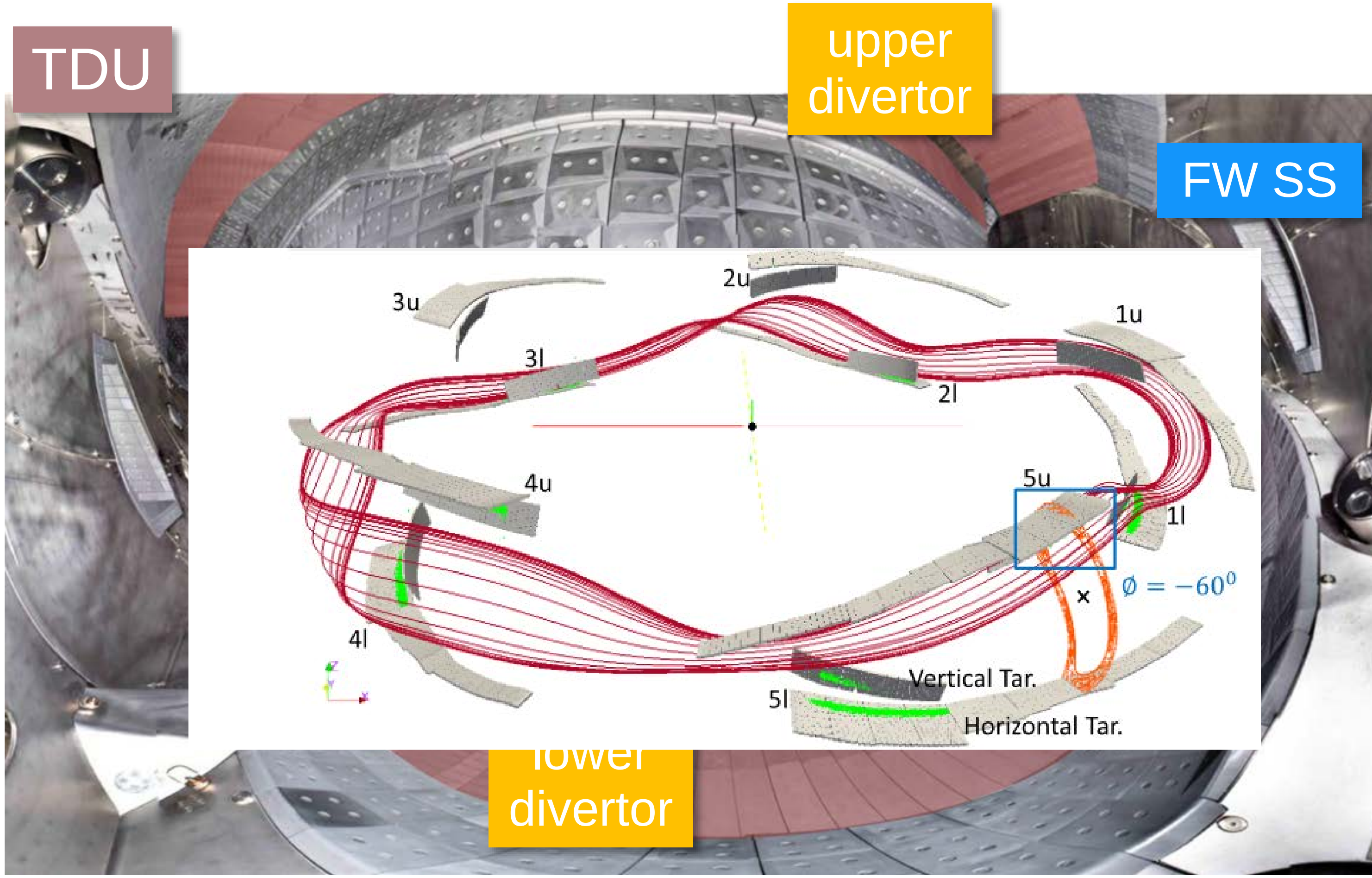
Wendelstein 7-X: Plasma-Facing Components

Island Divertor with Test Divertor Unit **(TDU)**

- 5 modules with 2 halfs
- Divertor material: fine grain **graphite**
- Divertor area: **19** (25) **m²**
- Max. divertor heat load: **10 MW/m²**

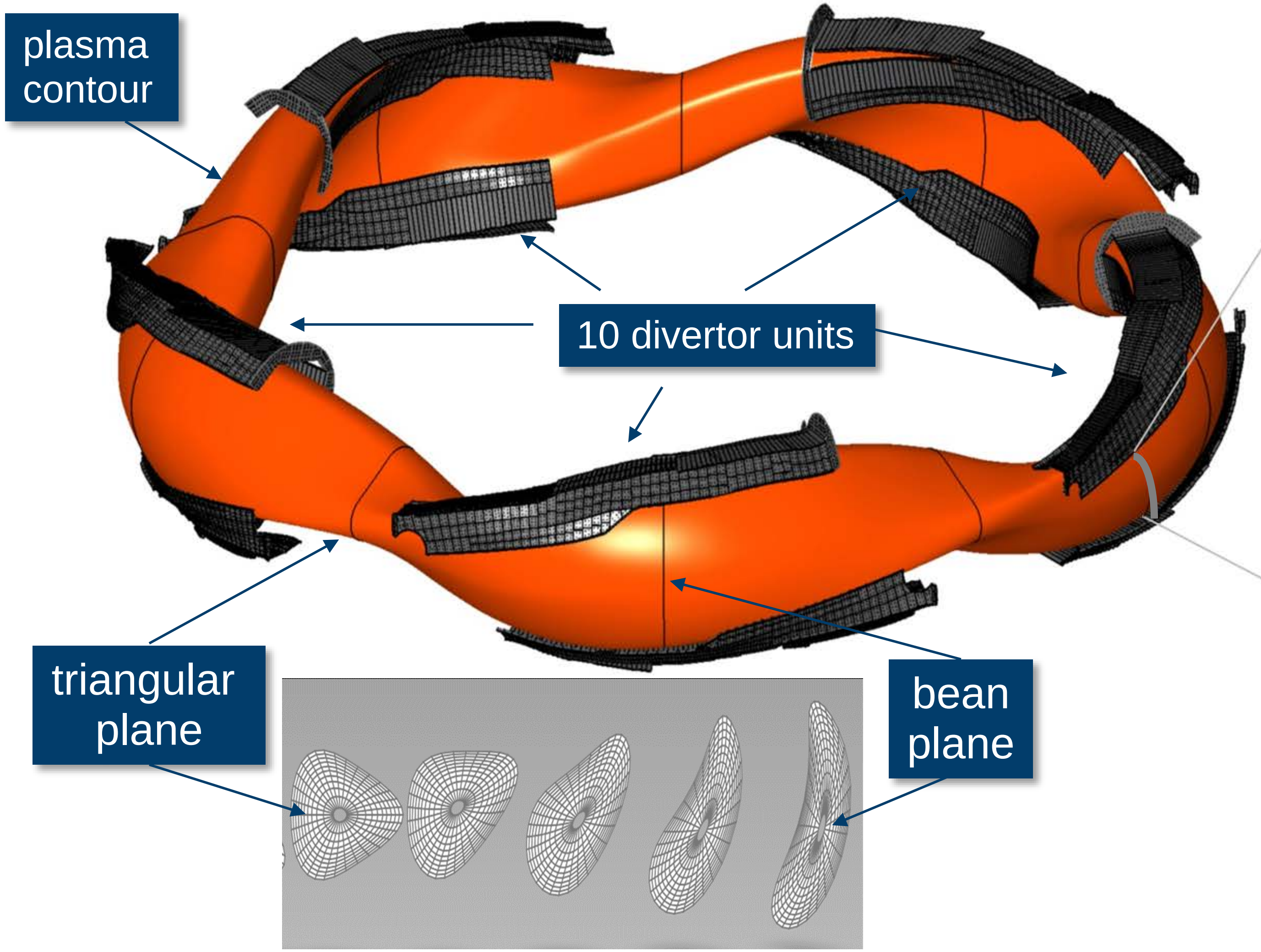
First wall coverage **(FW)**

- Main FW wall area: **45 m²** with C
  - 15 m² up to 0.5 MW/m²
  - 30 m² up to 0.25 MW/m²
- Recessed wall area: **70 m²** with SS
  - 70 m² up to 0.2 MW/m²
- Nominal wall temperature: **RT**

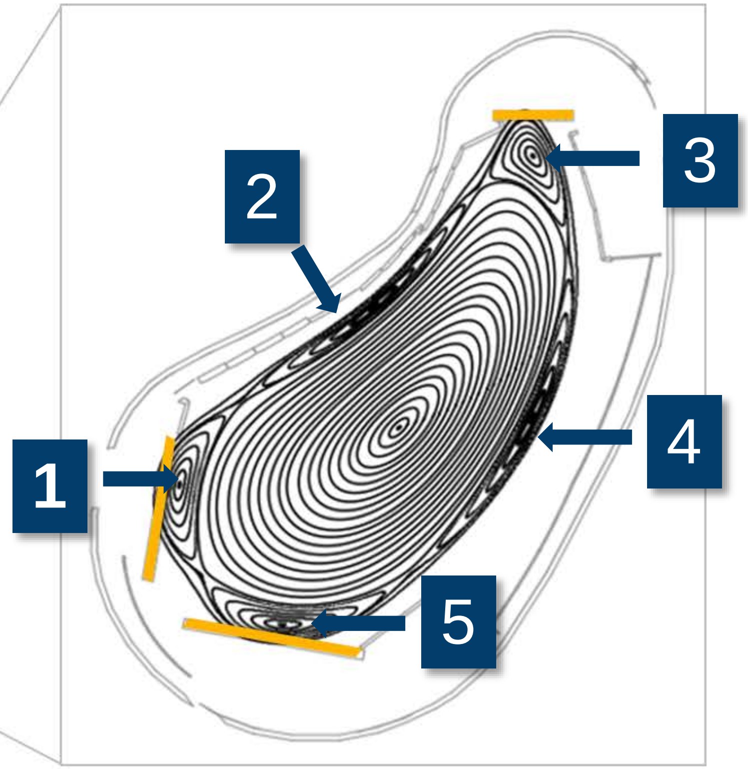


[CP Dhard et al.]

Wendelstein 7-X: Island Divertor Concept



2D cut: magnetic topology



standard divertor solution with 5 **magnetic islands**

[TS Pedersen et al. PPCF 2019]

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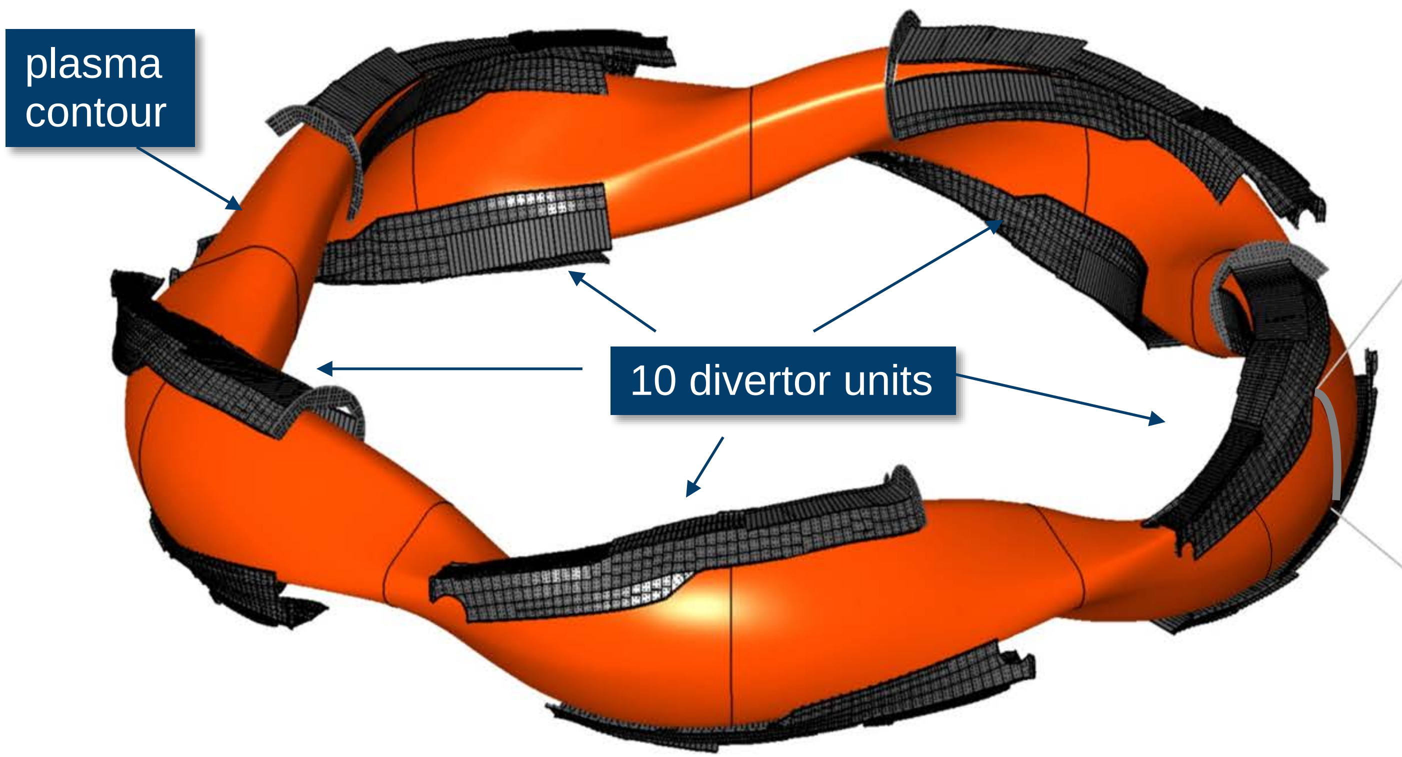
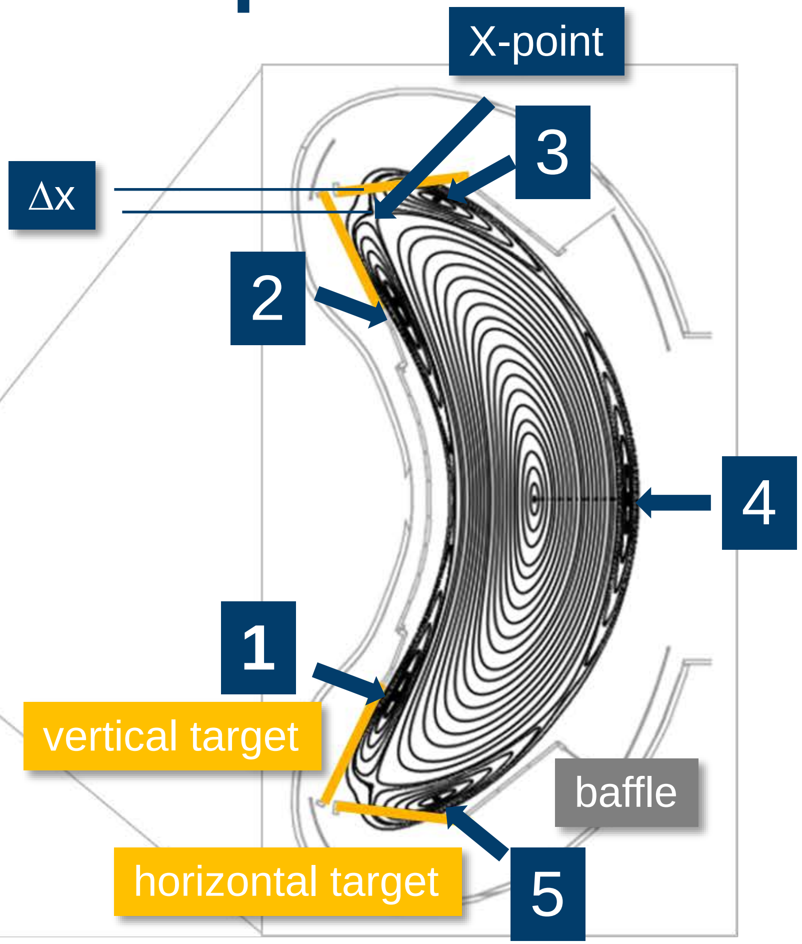
Wendelstein 7-X: Island Divertor Concept



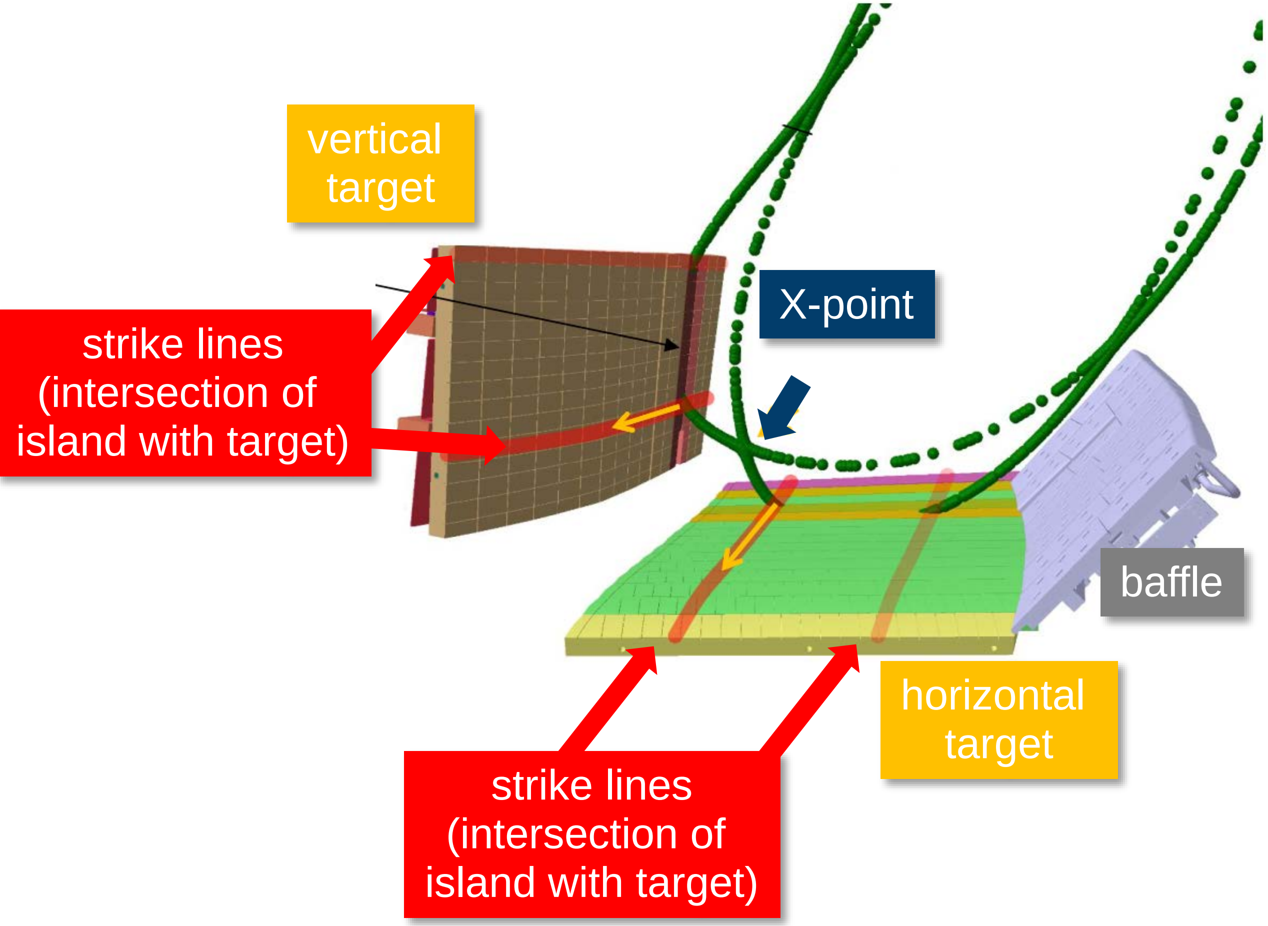


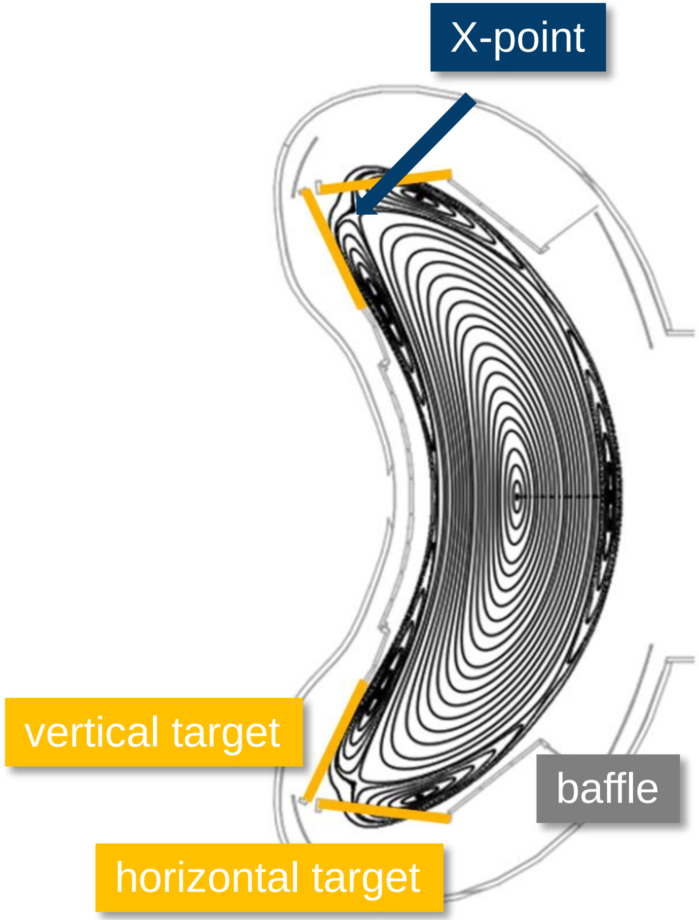
divertor target plates follow the magnetic island geometry in toroidal and poloidal direction

standard divertor solution with 5 magnetic islands (edge iota 5/5)

[EMC3-EIRENE predictions: Y Feng, et al. PPCF 2002]

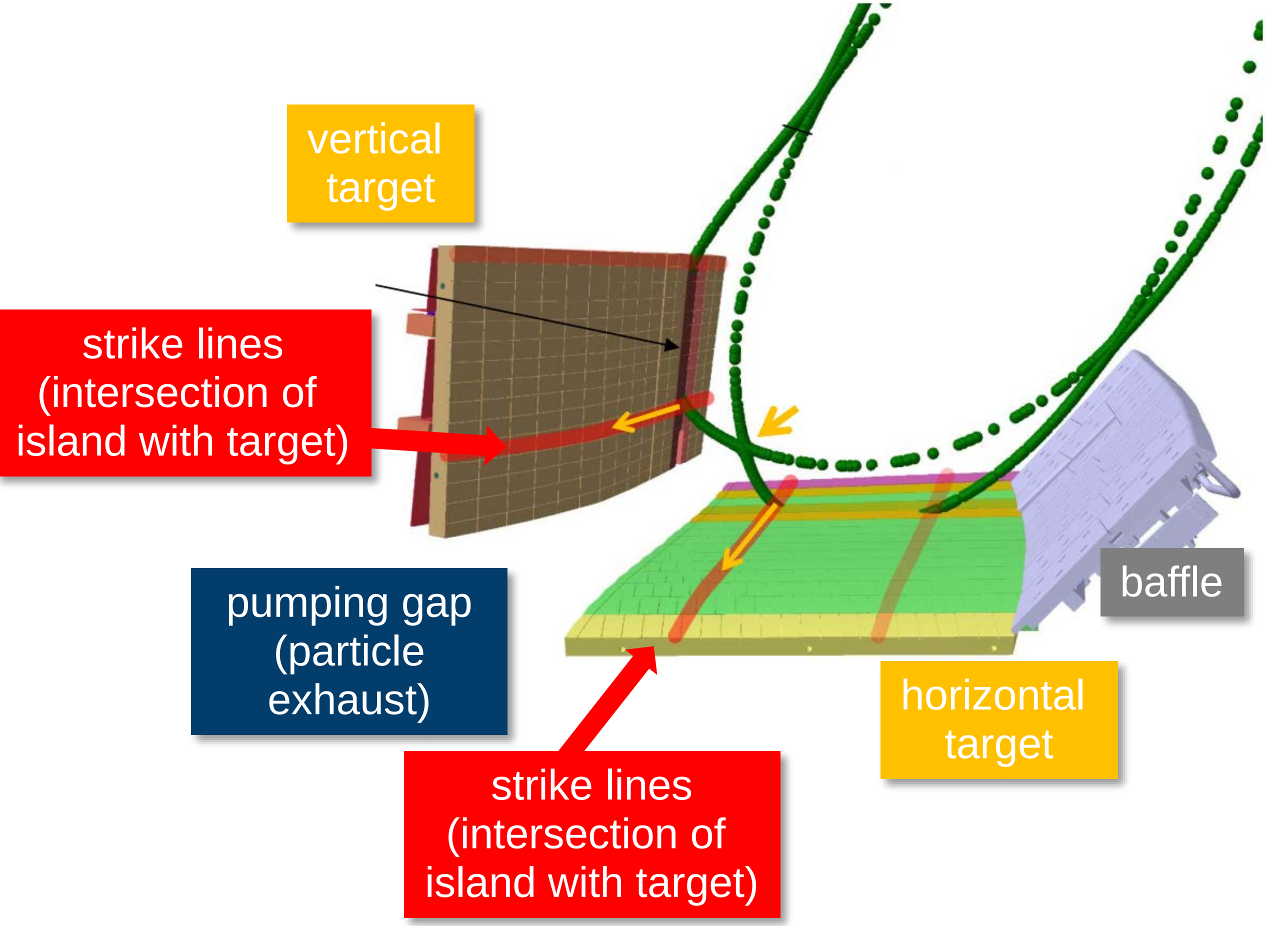
Wendelstein 7-X: Island Divertor Interaction Zones

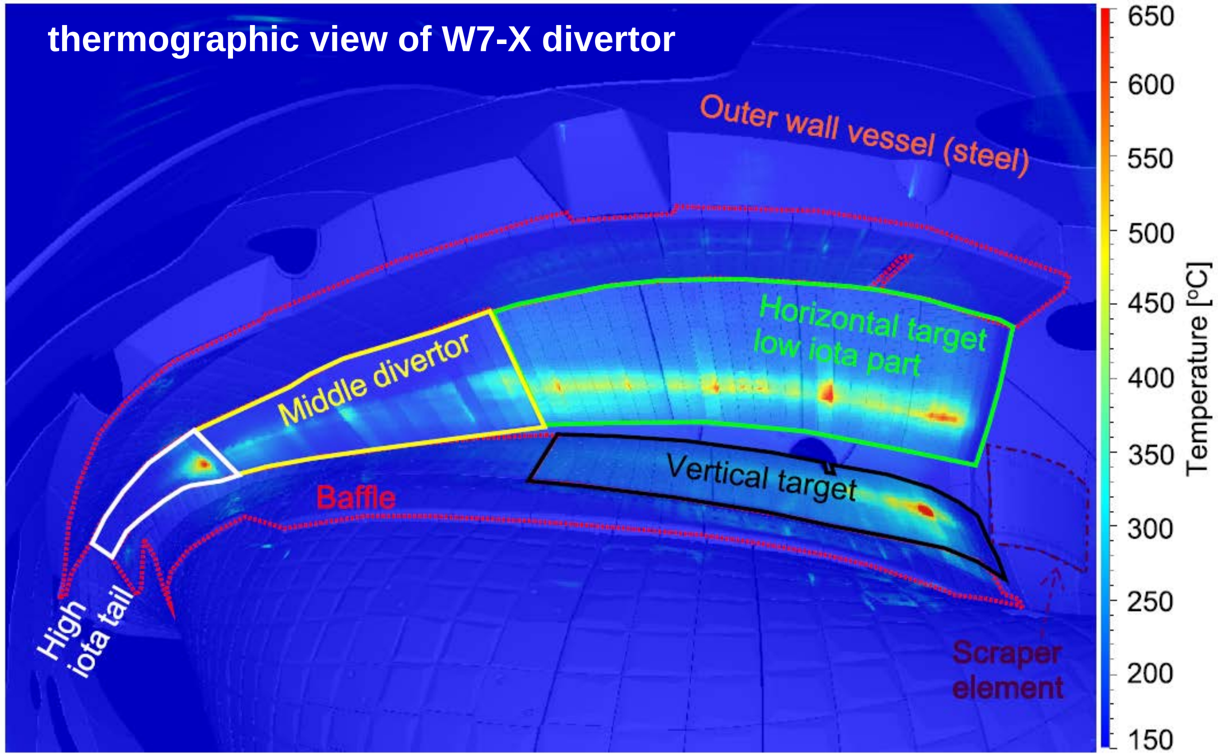




Wendelstein 7-X: Island Divertor Interaction Zones

- In **standard** magnetic **configuration two strike lines** interact with the target plates
- Visible in heat-flux pattern (power exhaust), imping particle-flux pattern (particle exhaust) and erosion pattern of target plates (impurity production) => main areas of **plasma-surface interactions**



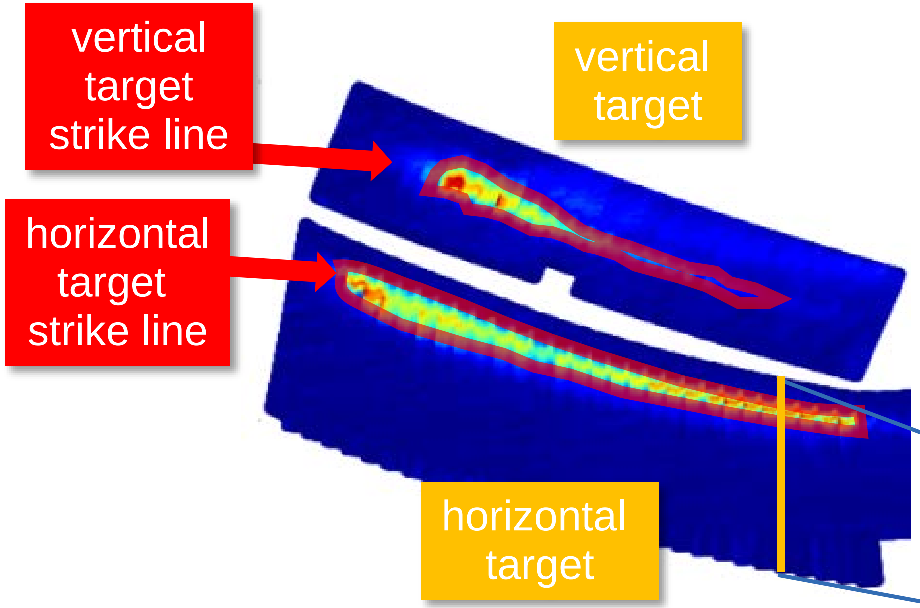


[Y. Gao et al. NF 2019]

Wendelstein 7-X: Heat-Flux Footprint on Target Plates

- Scrape-off layer width in present operational domain of W7-X in O(1cm) > wider than in tokamaks
- **Heat flux** is **spread** at the target plate due to **diffusive transport**
- Spreading and toroidal coverage define **wetted area** of the target plates

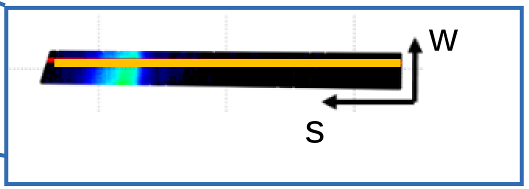
attached conditions



IR thermography
[standard magnetic configuration]

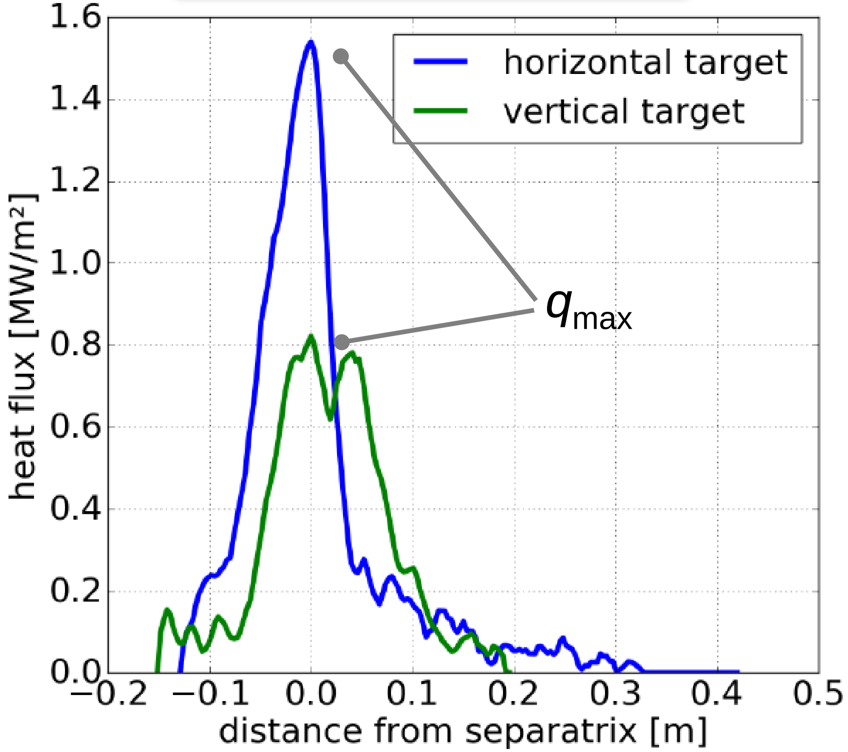
wetted area

$$A_{wet} = \frac{P_{div}}{q_{max}} = \sum_w \frac{\sum_s q_{div}(s,w)\,ds}{q_{max}}\,dw$$



[H. Niemann et al. NF 2020]

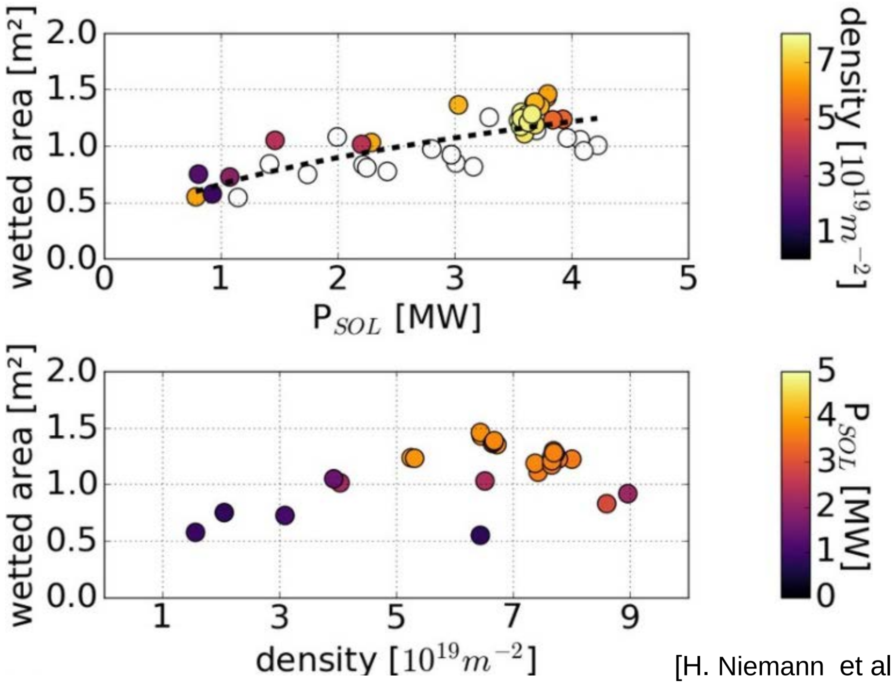
heat load profile



Wendelstein 7-X: Wetted Area in Attached Conditions

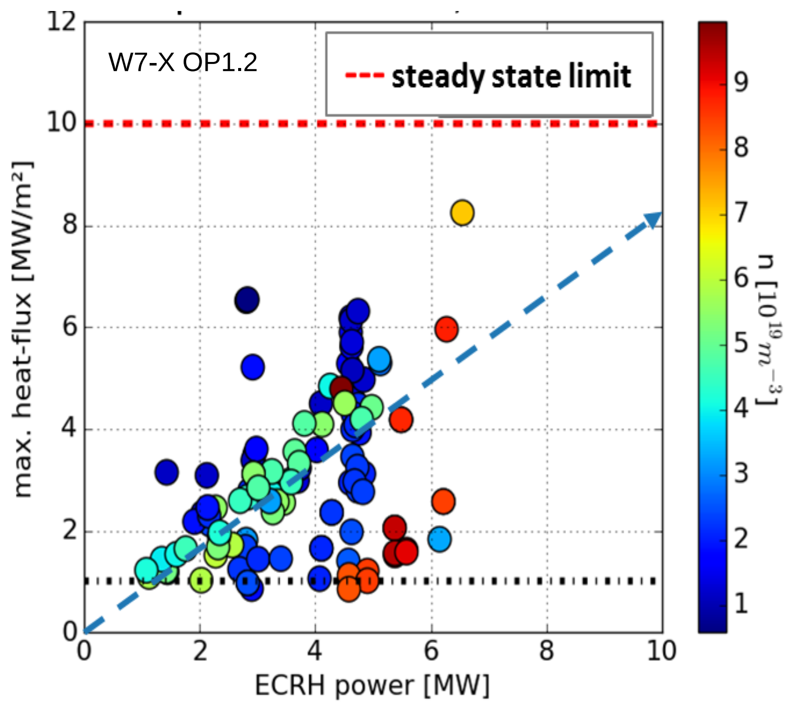
- Wetted area depends on magnetic configuration and rises with input power and radiation
- Total wetted area in standard configuration: 1.0m^2 -1.7m^2 (out of 19m^2)
- Peak heat load in all conditions below steady-state limit of PFC components of 10 MWm^{-2}

wetted area vs. density and power



[H. Niemann et al. NF 2020]

TDU peak heat loads



Wetted area represents main plasma-surface interactions zone for ionic species in W7-X plasmas

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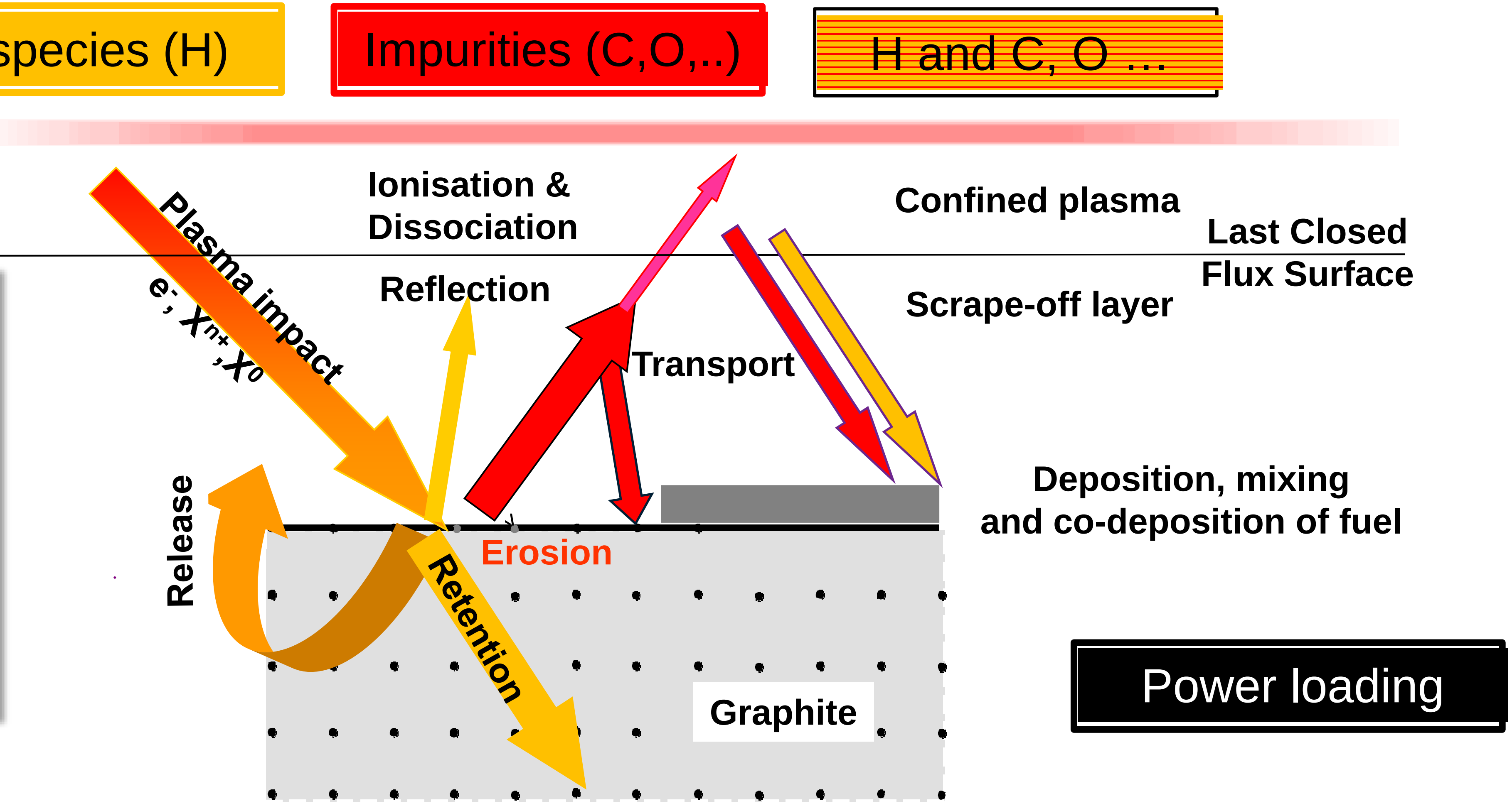
Plasma-Surface Interactions (PSI) Processes

Processes depend on plasma facing material, material / projectile mass, material mix, impact energy (E_{in}), impact angle (α), roughness and temperature (T_{surf}), plasma conditions



Objectives of PSI Studies in W7-X

- Exploration of operational window **wall conditioning, impurity radiation, fuel recycling**

- Investigate major PSI processes in the **3D environment** of Wendelstein 7-X with graphite PFCs: **material erosion, deposition, mixing, material transport, and fuel retention**

- Key access to PSI via spectroscopy (in-situ) and post-mortem analysis (ex-situ)
  - Experimental plasma information essential (T_e, n_e, Γ_{ion}, T_i, E_{in}, n_{neut}, T_{surf}, etc.)
- Post-mortem analysis: interpretation of single magnetic configurations or plasmas
- Interpretive modelling for code-experiment comparison
  - Plasma-edge code: 3D Monte-Carlo Code Package EMC3-EIRENE
  - PSI code: 3D Monte-Carlo Code ERO2.0

- Long-term goal: predictive modelling of PSI processes in W7-X towards
  - long-pulse operation with actively cooled graphite divertor: erosion (lifetime) & deposition (dust)
  - operation with metallic PFCs: W erosion (lifetime) and plasma compatibility (screening)

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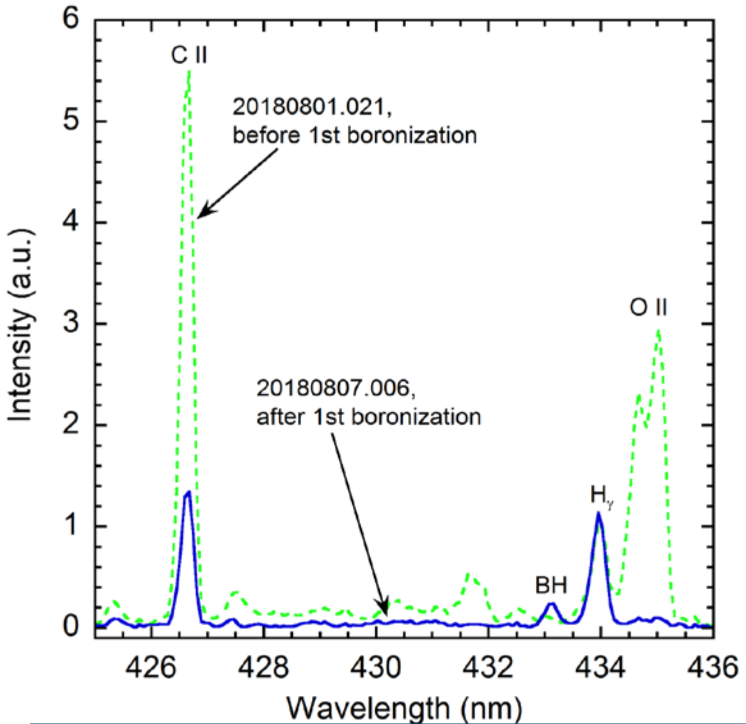
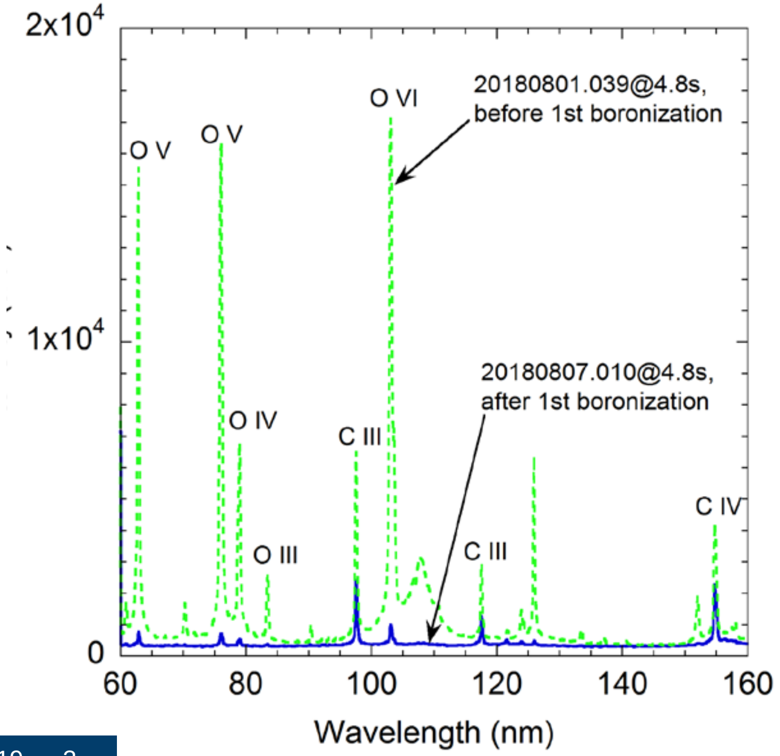
Intrinsic Impurities and Boronisation

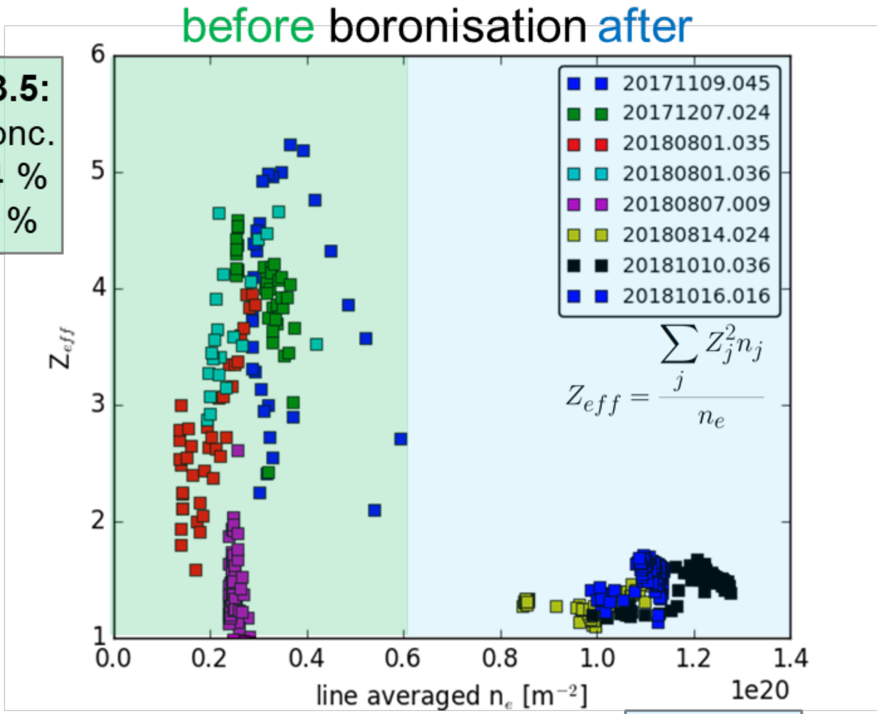
- Initial divertor operation compromised by high impurity content (O,C) and H outgassing [A. Goriaev et al. PS 2020]
  - residual water in graphite released during PFC heat-up by plasma impact => O in plasma
  - oxide layers on first wall sputtered by plasma and charge exchange neutrals => O in plasma
- Boronisation: Break of O cycle in plasma => lower O content and flux from PFCs [S. Sereda et al. NF 2020, E. Wang et al. PS 2020]
- Impurity radiation in confined plasma vastly reduced / high purity of H plasma [B. Butterschön et al., M. Krycjowiak et al.]





Divertor conditions: $T_e \approx 30$eV and $n_e \approx 1\times10^{19}m^{-3}$

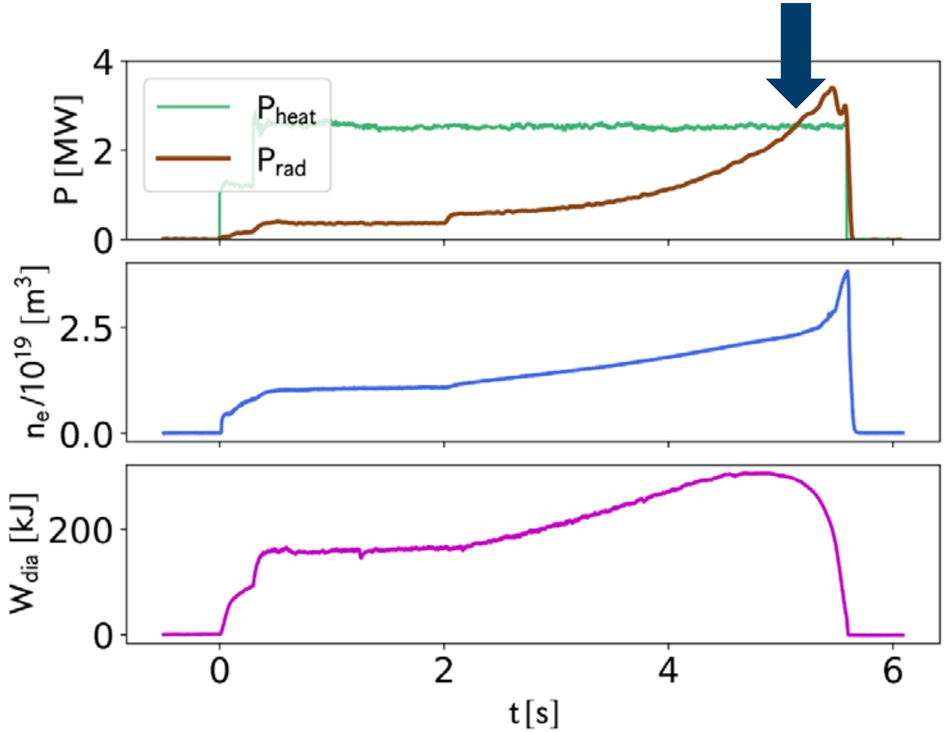
Z_{eff}=3.5:
ion conc.
C: 14 %
O: 7 %



Z_{eff}=1.5:
C, B, O

Radiation-Induced Density Limit

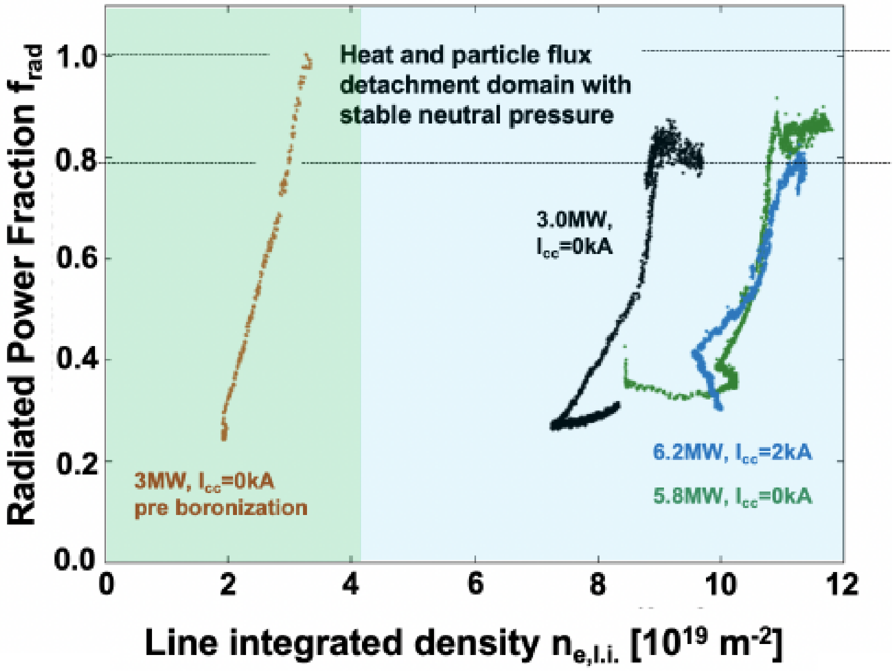
- No plasma current driven density limit in stellarators
- Plasma density limit determined by radiative collapse
- Radiation determined by H and intrinsic impurities (C, O)



$$\bar{n}_c = c \cdot \frac{P_{heat}^{0.6}}{f_{imp}^{0.4}}$$

[G. Fuchert et al. IAEA 2018]

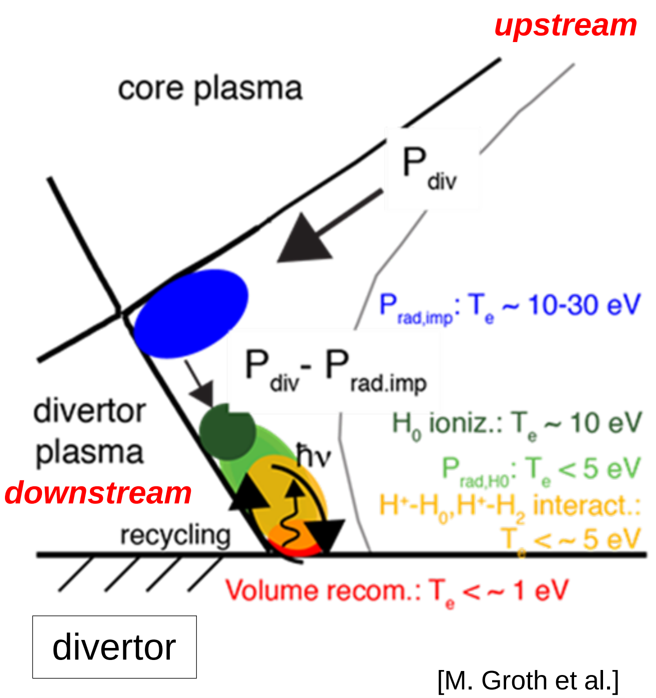
before - boronisation - after



- Divertor functionality with high divertor pressure and neutral compression after boronisation in W7-X [O. Schmitz et al. PPCF 2020]

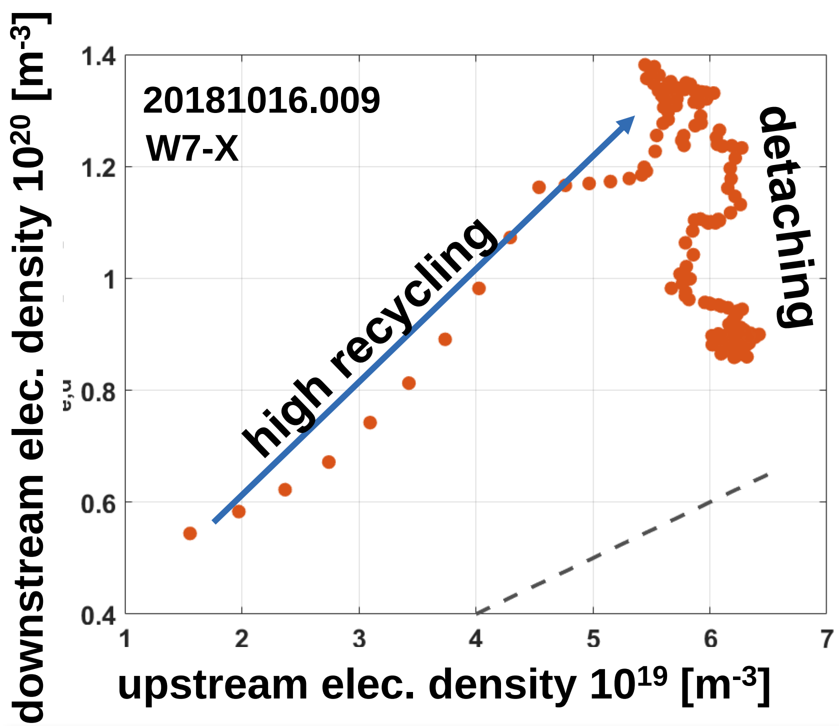
Fuel Exhaust and Recycling

- Access to stable high density operation in divertor and main plasma: power detached divertor regime in W7-X
- No significant volume recombination yet observed in W7-X



[M. Groth et al.]

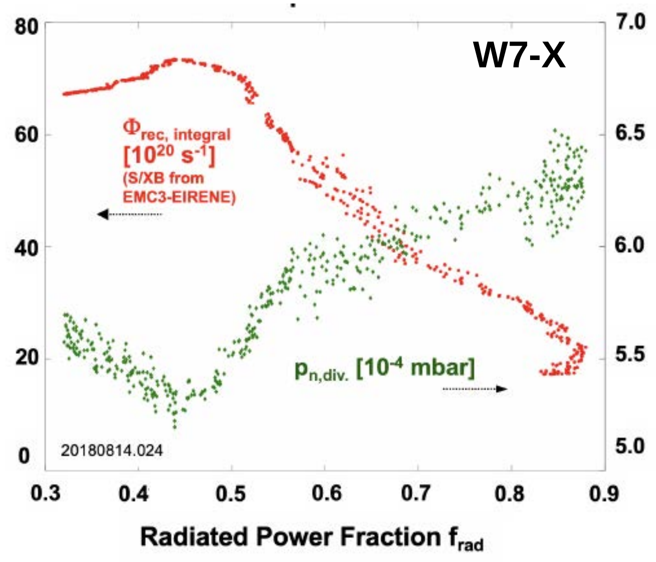
Detachment process in tokamaks (DIII-D, JET, AUG etc.)



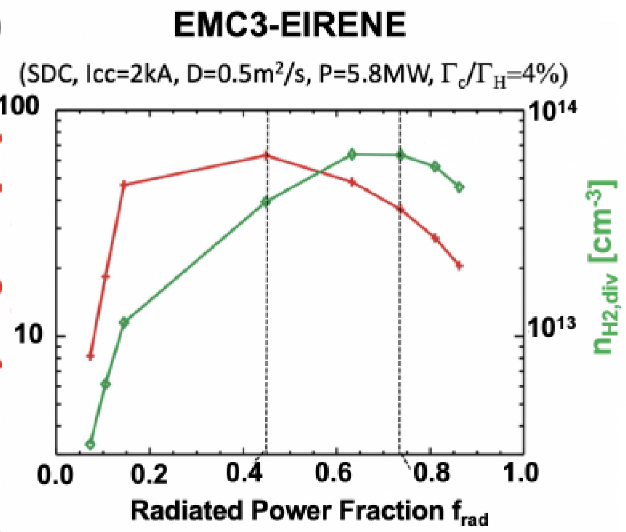
=> roll-over behaviour during fuelling ramp

[O. Schmitz et al. NF 2020]

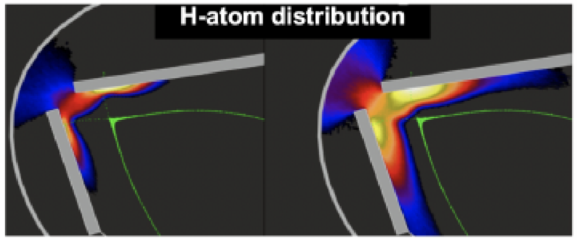
[D. Zhang et al. PRL 2019]



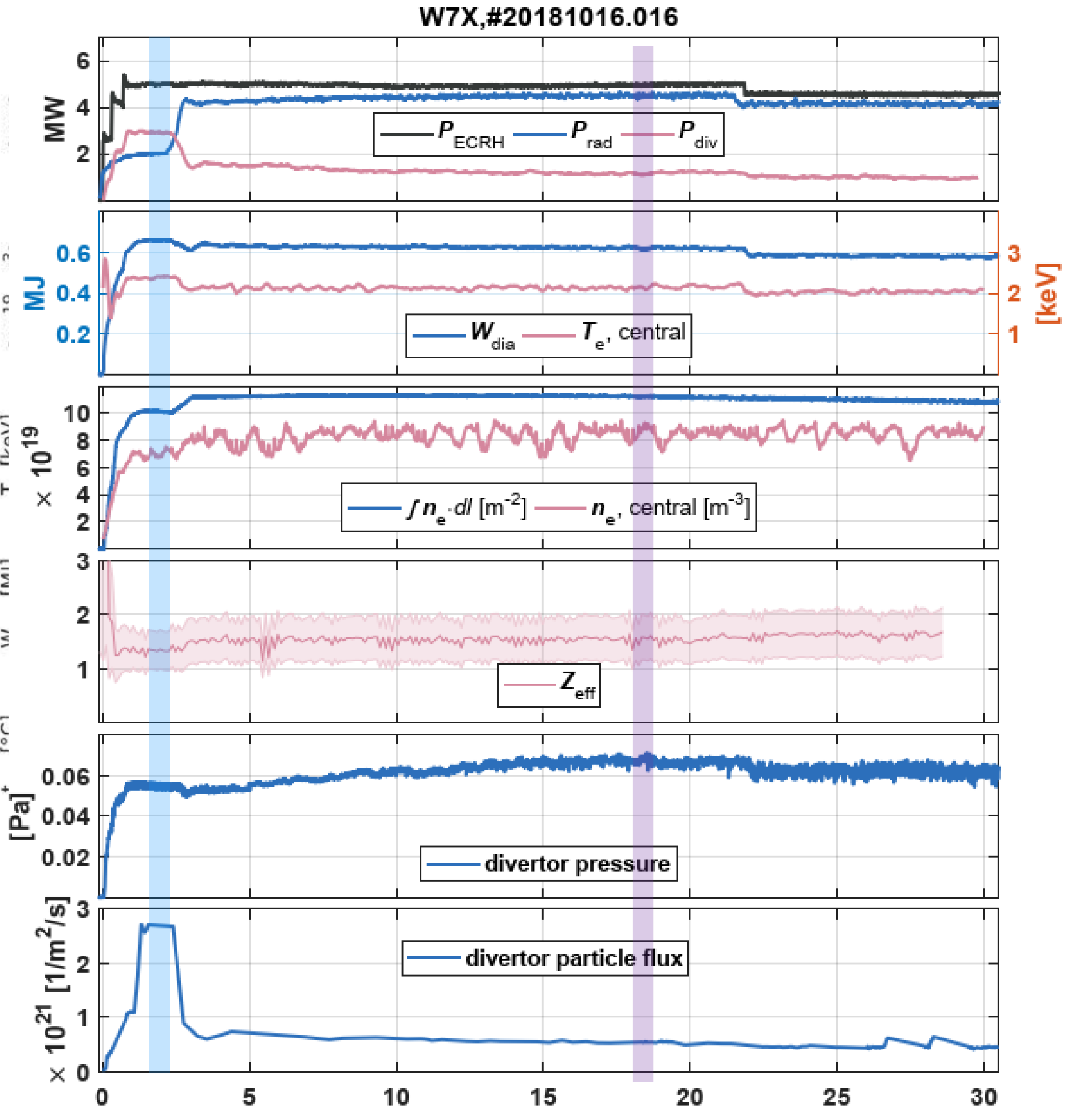
a.)



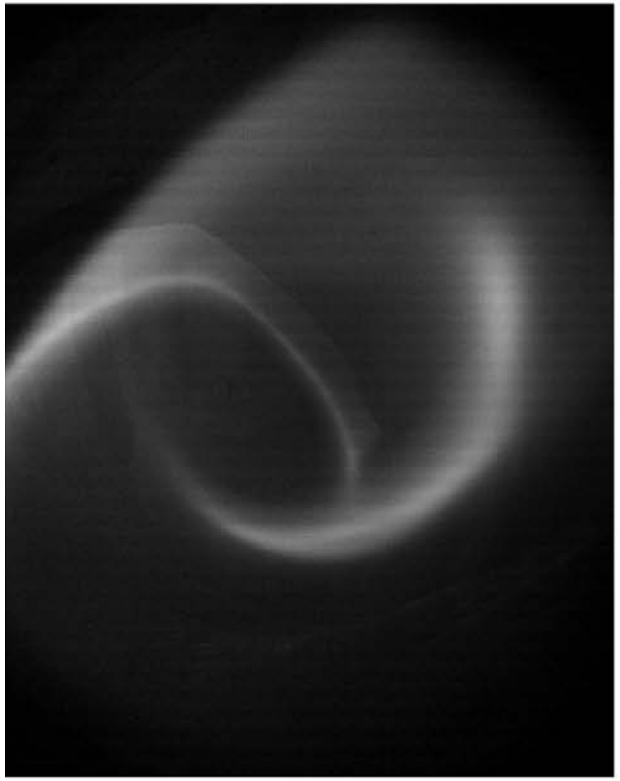
c.)

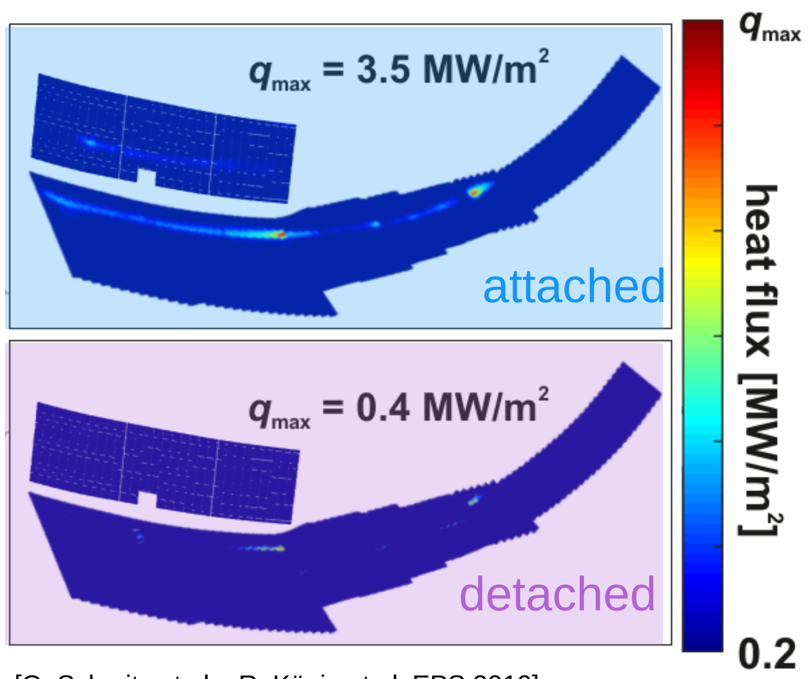


Long-pulse Steady-State Detachment in Wendelstein 7-X



[M. Jakubowski et al. ISHW 2019]





[O. Schmitz et al. , R. König et al. EPS 2019]

- Divertor detachment induced by C+H radiation and H recycling
- Symmetric in all five modules with the ten divertor units
- Steady-state operation for 26 s inertially cooled PFCs!

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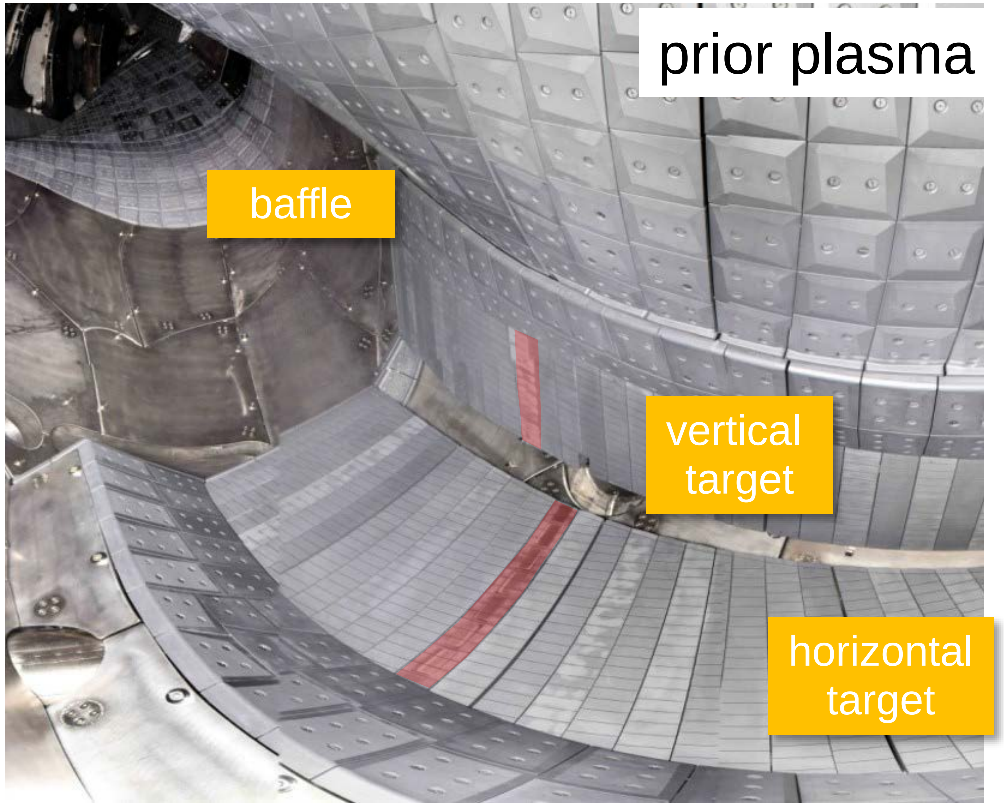
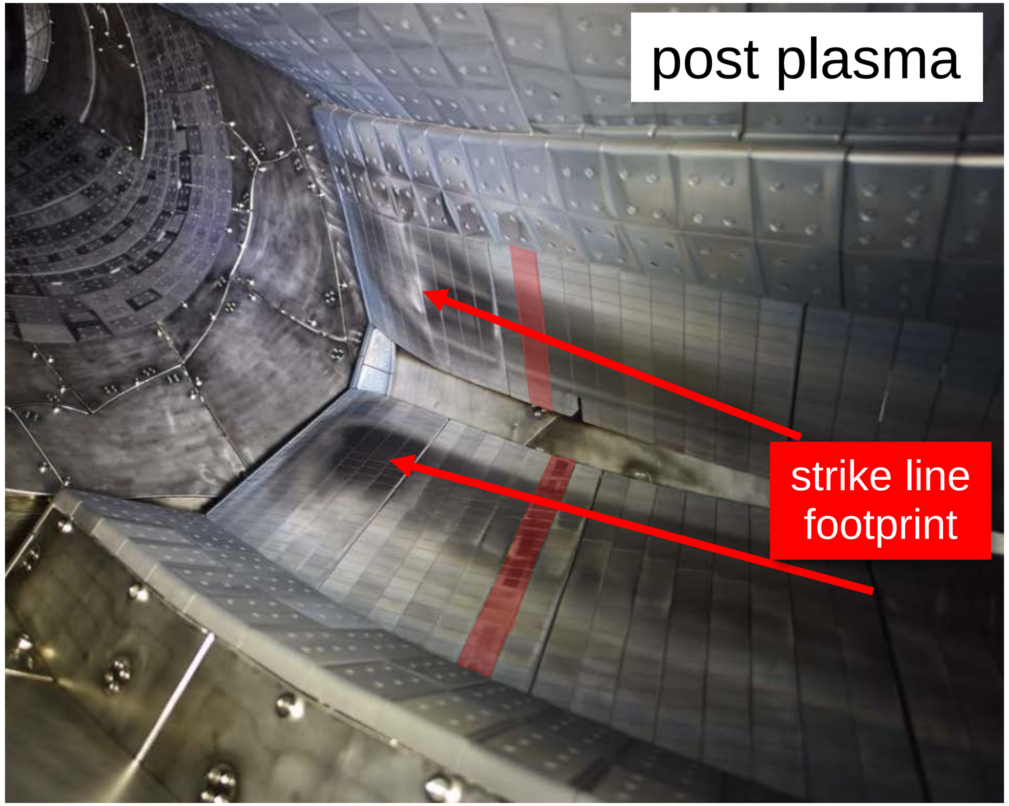
Visual Inspection of Divertor PFCs

- Campaign integrated pattern after first year of operation: predominantly attached divertor conditions
- Erosion and deposition pattern on horizontal target can be related to standard magnetic configuration



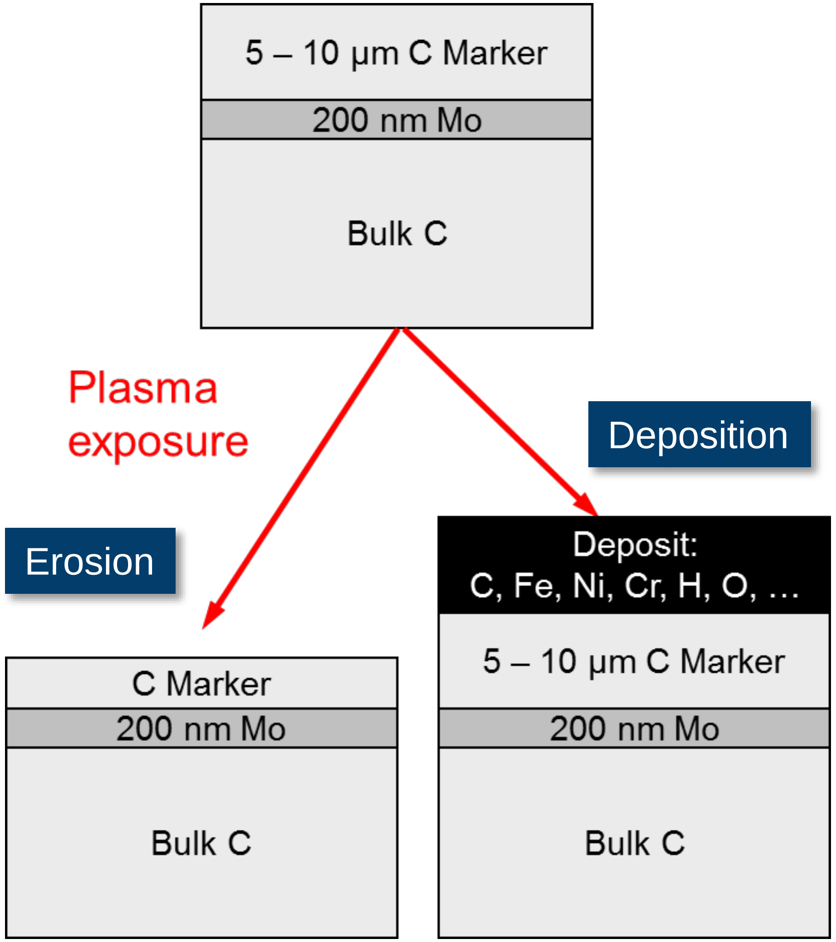


- Multiple erosion and deposition processes => net migration paths from tile analysis

Graphite Erosion and Deposition: Marker Layers

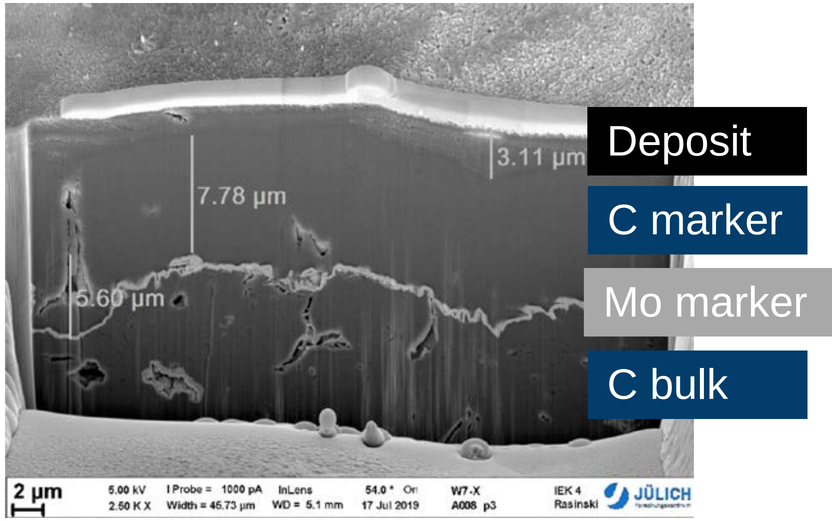
- OP A w/o boronisation
- C Marker locally fully eroded down to Mo

- Operational time in standard configuration: ~ 2100s in He and H
- Peak erosion rates: ~5.0nm/s at HT and 6.5nm/s on VT



[M. Mayer et al. PFMC2019]

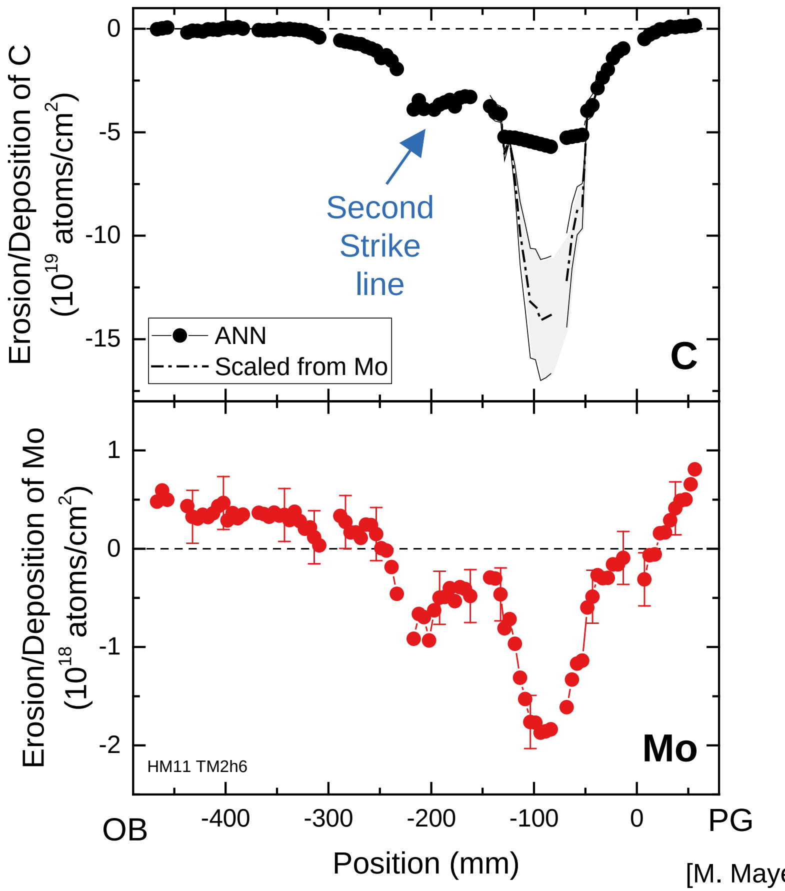
Post-mortem analysis
- EBS & NRA
- LIBS & LIA-QMS
- SEM, EDX, FIB



Cross-section in deposition zone in OP B (FIB - SEM)

Graphite Erosion and Deposition: Horizontal Target (OP A)

- Poloidal profile of C erosion in standard configuration on marker PFC
- Strong Mo erosion at strike line caused by impinging O and C ions
- Complete erosion of C marker at strike line => scaling from Mo erosion



[M. Mayer et al. PFMC2019]

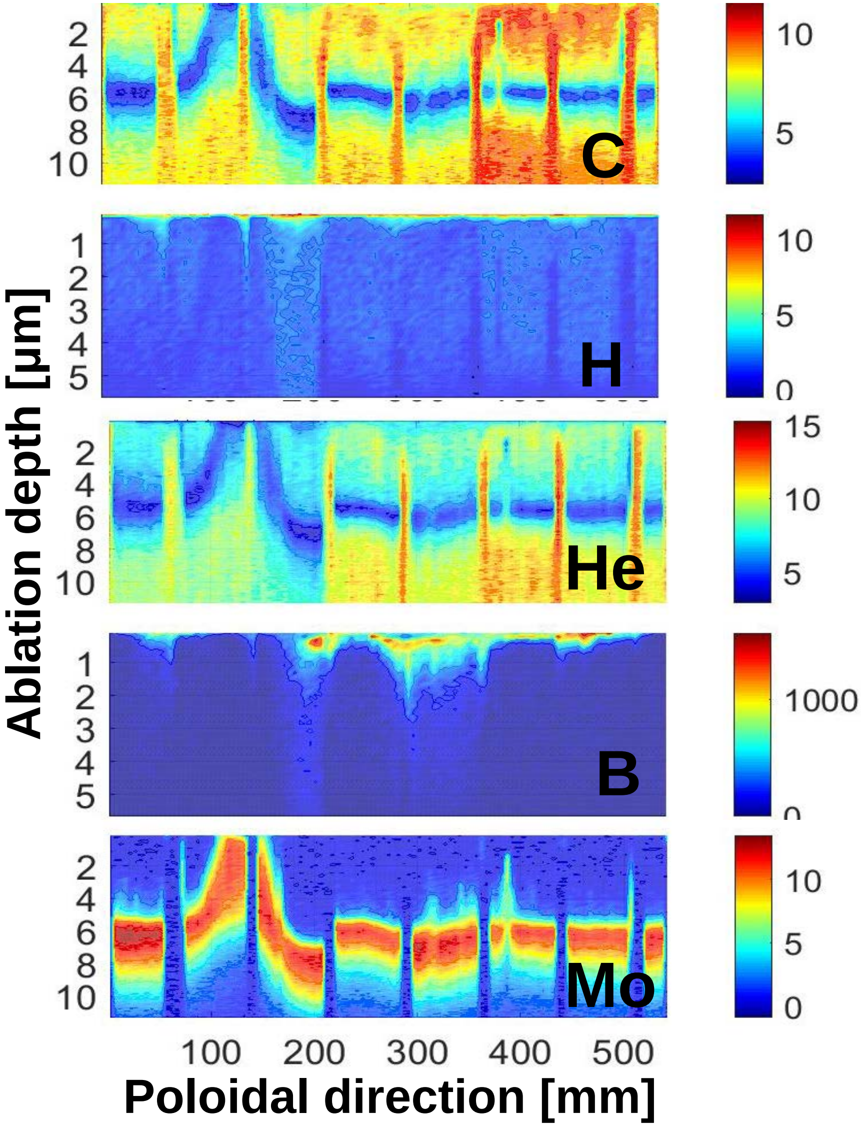
Initial C divertor net erosion estimate:

- Horizontal target: 34.5+/-8.4 g
- 13 marker PFCs measured
- Extrapolation to wetted area

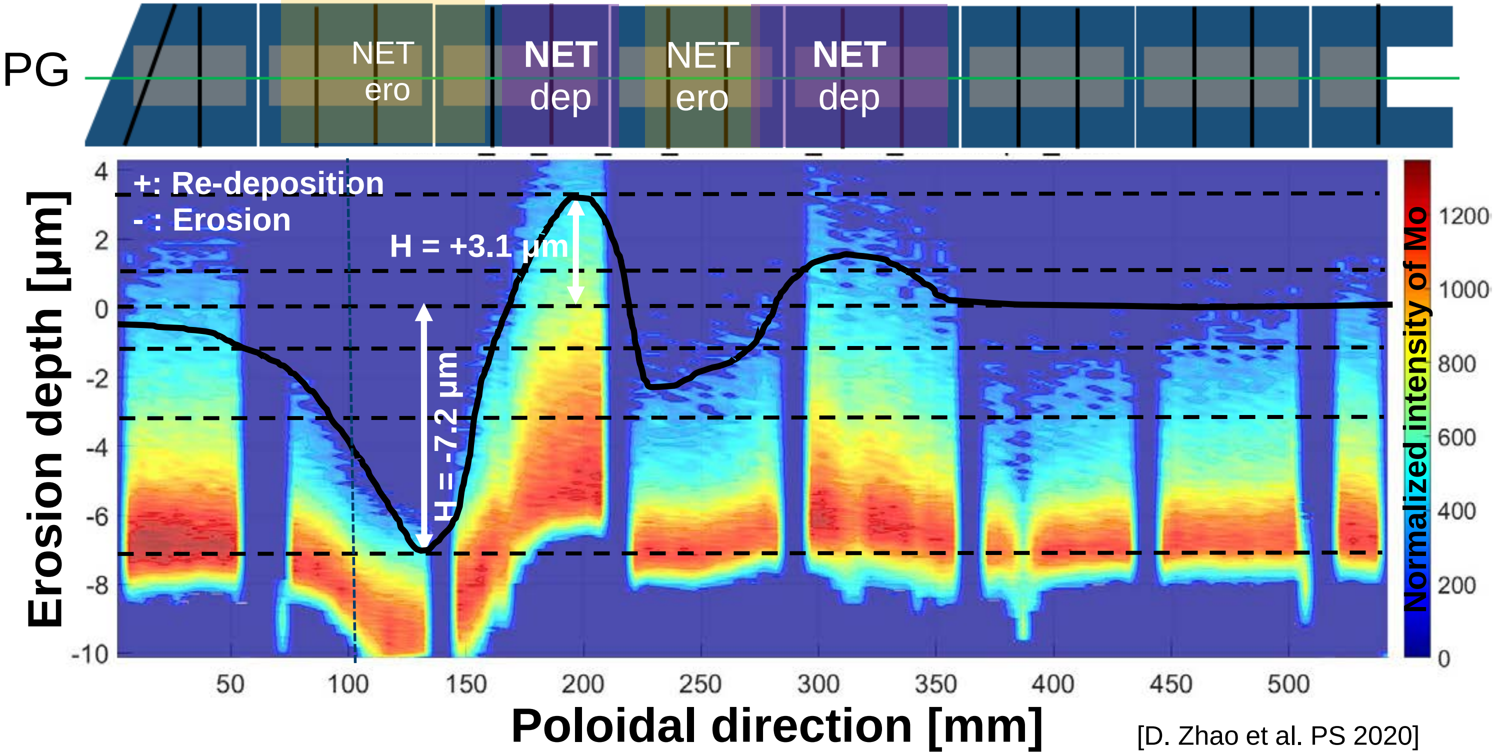
- Vertical target: 13.3+/-5.7 g
- 4 marker PFCs measured
- Extrapolation to wetted area

=> challenging for 30 min steady-state plasmas

Erosion and Deposition Pattern: Horizontal Target (OP B)



Poloidal distribution of erosion and deposition determined by LIBS

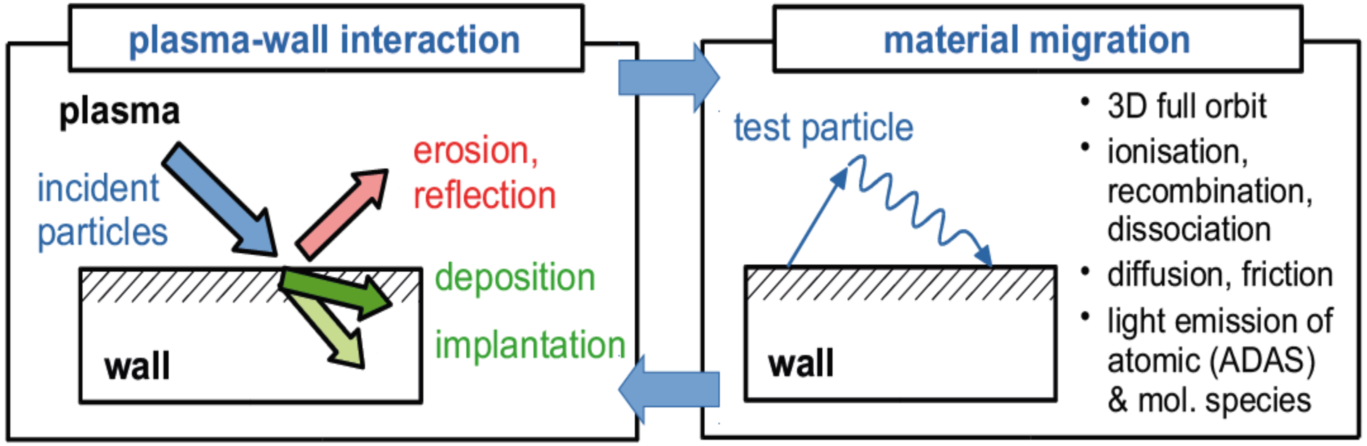


[D. Zhao et al. PS 2020]

- Erosion rate lower that in OP A (min. 4x lower in OP B)
- First erosion zone in-line with strike-line location
- Material composition in depth for C, H, O, B, Mo
- B eroded at strike line / B deposit away from strike line

3D simulation of PSI in W7-X

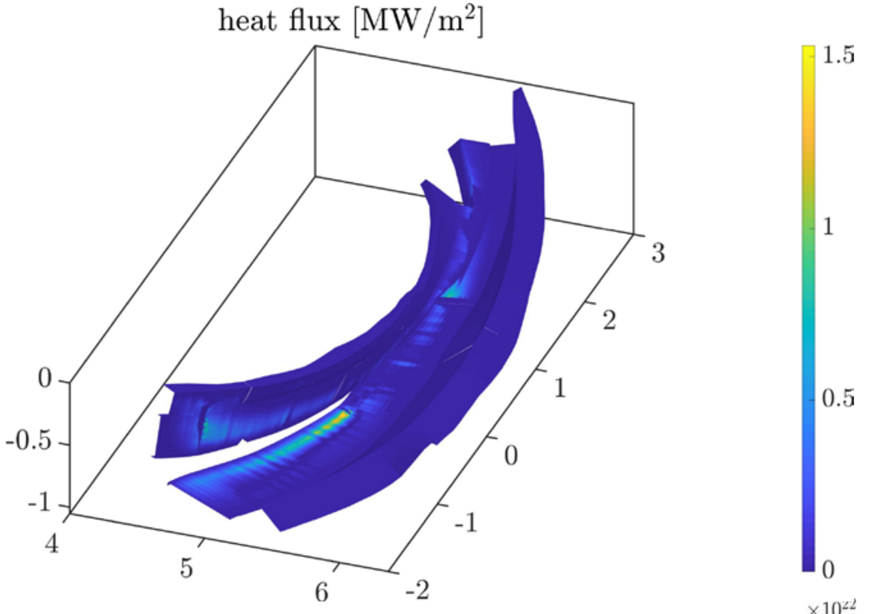
- Significant operation in standard configuration (>50%)
- Erosion / deposition pattern on divertor modules measured: spectroscopy (gross) and post mortem (net)
- Main interaction: strike lines / wetted area (IR)

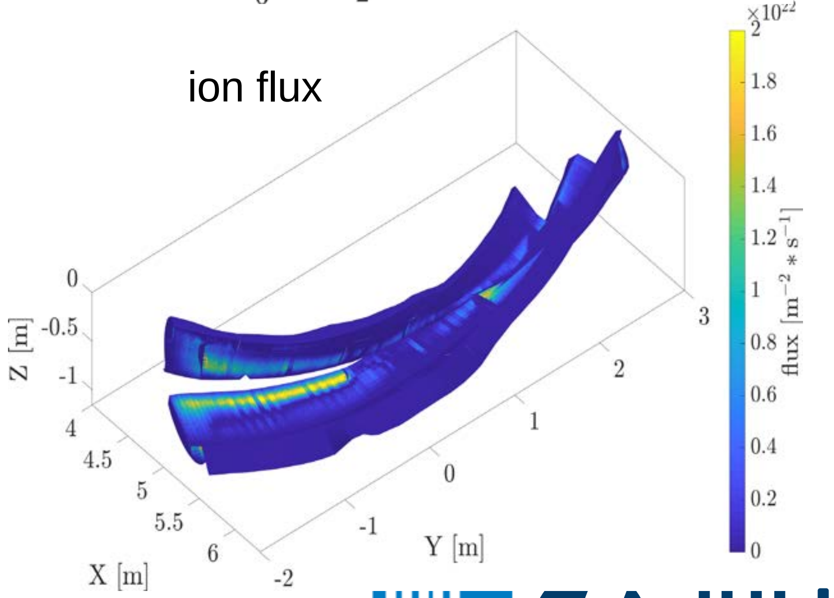


ERO2.0 calculations on HPSC (JURECA)

- 3D simulation of PSI with ERO2.0 of one W7-X module: periodic boundary conditions of fivefold symmetry
- 3D plasma background from EMC3-EIRENE: reference plasma in H with C impurity fractions

Plasma information from EMC3-EIRENE





[F. Effenberg et al. NF 2019]

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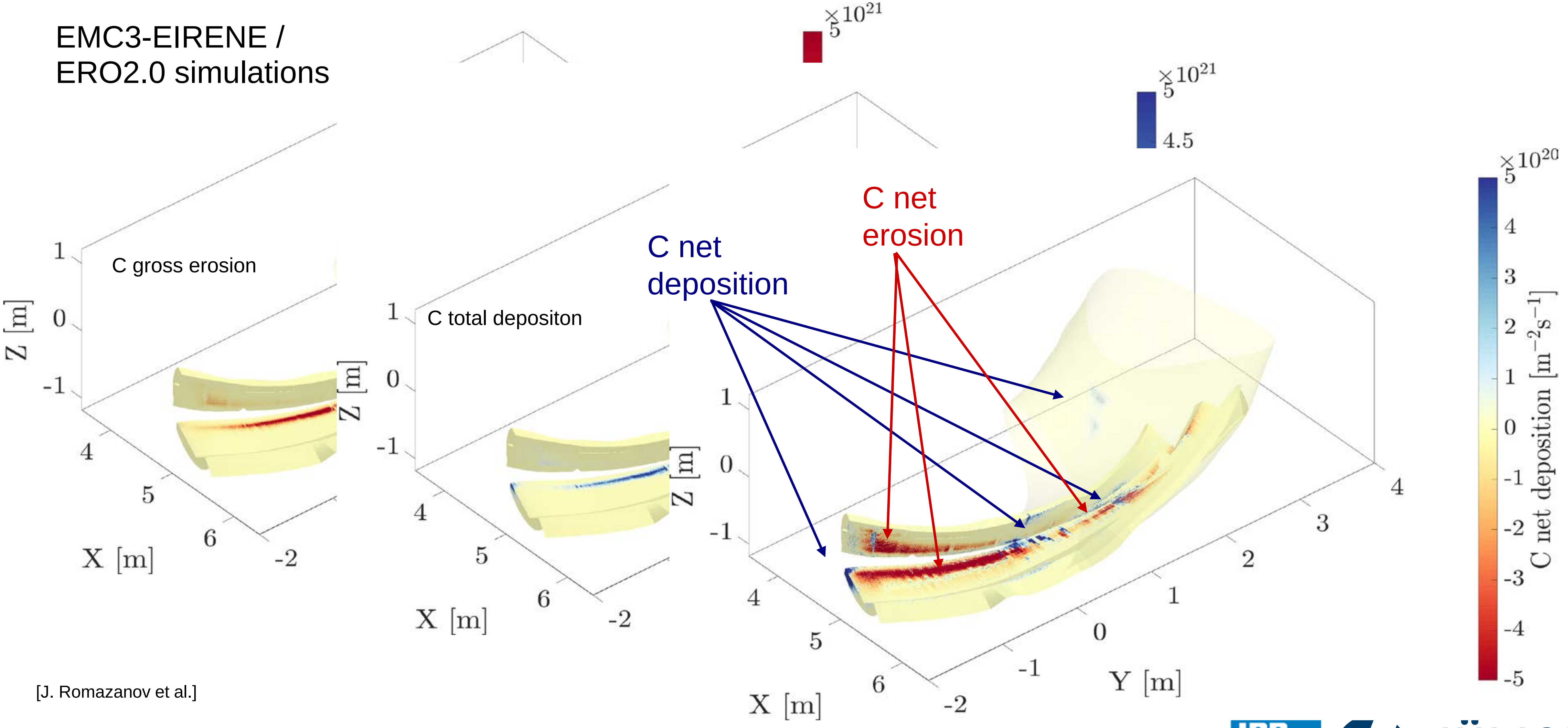
C Erosion and Depostion Balance in Standard Configuration

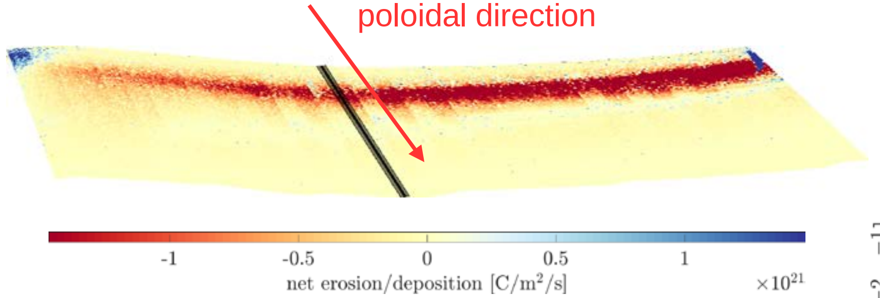
EMC3-EIRENE /
ERO2.0 simulations

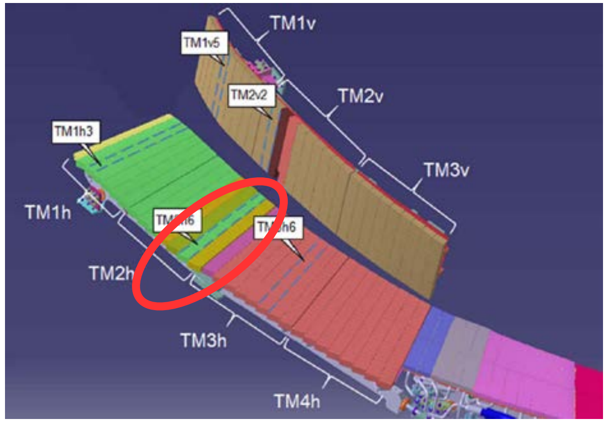


[J. Romazanov et al.]

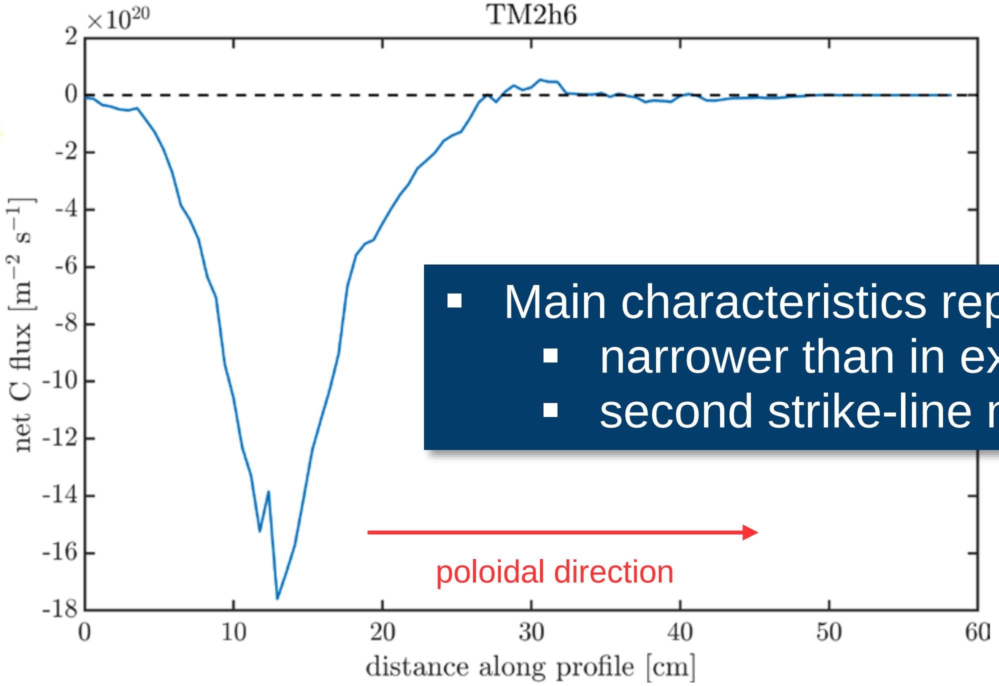
Initial ERO2.0 Modelling of C Erosion and Deposition in W7-X

- Modelling of net C erosion and deposition on the horizontal target plate
- C sputtering by physical and chemical sputtering considered





poloidal C net flux profile at marker location



- Main characteristics reproduced, but
  - narrower than in experiment
  - second strike-line much weaker

- Benchmark: $^{13}CH_4$ injection through gas inlet in horizontal divertor (330 plasma seconds)
- Last TDU experiment: clean ^{13}C deposition pattern in divertor – analysis ongoing [S. Brezinsek PFMC2019]

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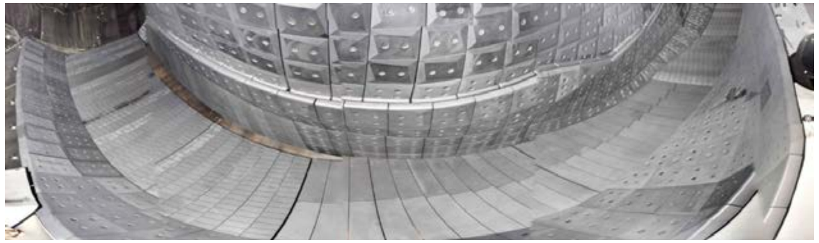
Summary and Outlook

- W7-X island divertor: characterisation with power loads, recycling and detachment
- PSI processes with graphite PFCs: erosion / deposition and role of boronisation
- Modelling of plasma edge (EMC3-EIRENE) and PSI (ERO2.0)

Uncooled graphite divertor (TDU)



Actively cooled CFC divertor

Actively cooled divertor with W PFCs

2017 / 2018
- C divertor
- 7.5 MW
- 200 MJ
- 100 s
- H/He

2021/2022
- CFC divertor
- >10 MW
- 18 GJ
- 1800 s
- H, He, D

202X with X > 7
- W divertor
- “optimised” design
- Metallic first wall
- More input power
- H, He, D

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