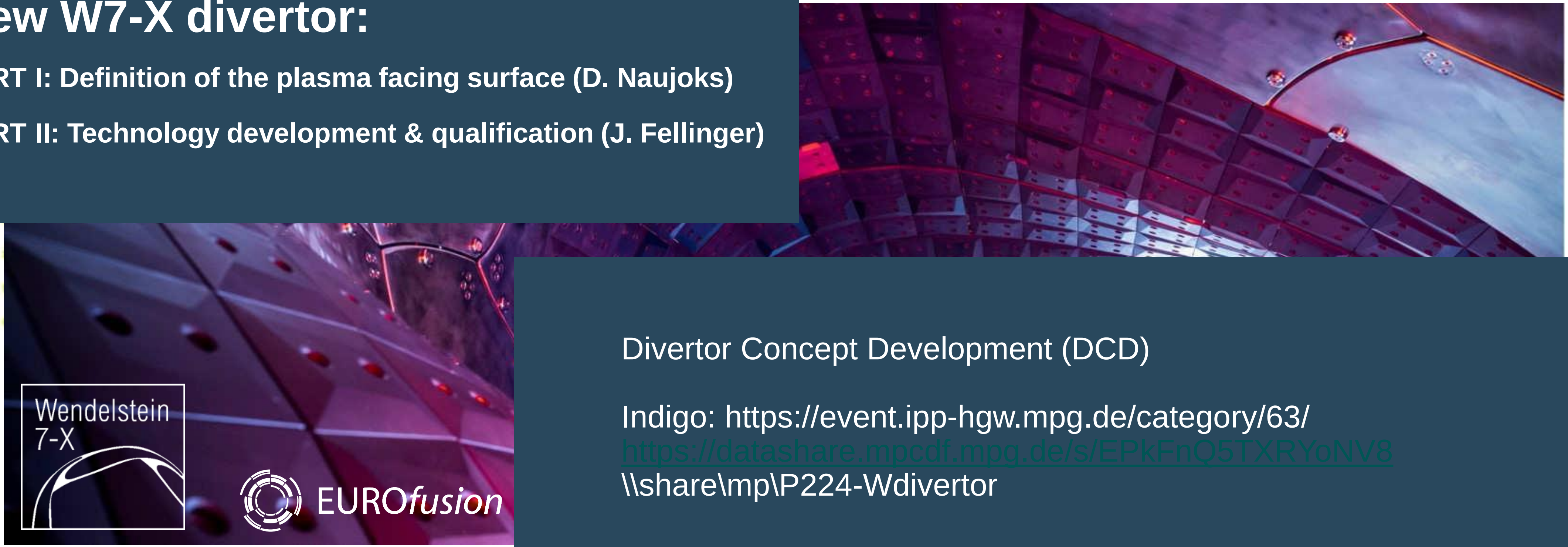
MAX-PLANCK-INSTITUT
FÜR PLASMAPHYSIK

New W7-X divertor:

PART I: Definition of the plasma facing surface (D. Naujoks)

PART II: Technology development & qualification (J. Fellinger)



Divertor Concept Development (DCD)

Indigo: https://event.ipp-hgw.mpg.de/category/63/
https://datashare.mpcdf.mpg.de/s/EPkFnQ5TXRYoNV8
\\share\mp\P224-Wdivertor

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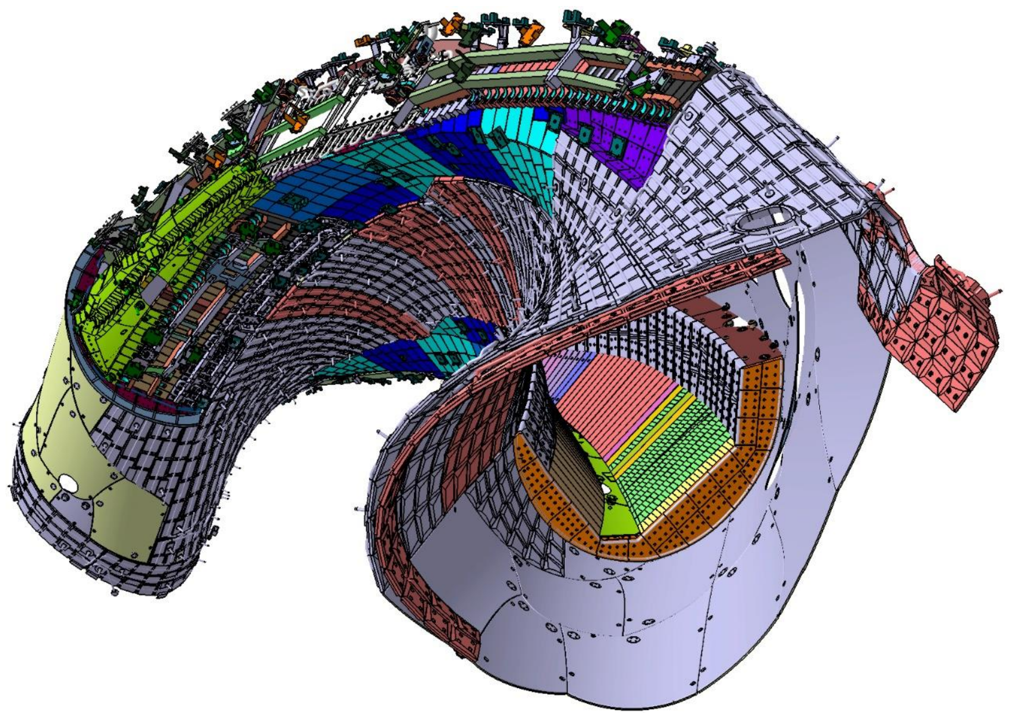
Content



Source: 1-ACE-Y0100.0

- Motivation – new tungsten divertor for W7-X
  - material replacement (C -> W)
  - local overloading
  - limited exhaust
- Divertor functions, objectives, two approaches
- Tools development for physics simulation & optimization:
  - 1 design tools (CATIA)
  - heat load calculations (SMoLID, EMC3-Lite)
  - neutral gas modeling (ANSYS, DIVGAS, EIRENE)
  - plasma/impurity modeling (EMC3/EIRENE, ERO2.0)
- Summary

W7-X Divertor Concept Development (DCD) for OP3

Temporary Working Group “W7-X Enhancement 2030” -> final report 1-YLF-T0000.0 [2B3SZA]
Resource and Schedule Review: OP3 divertor -> concept development 2022-2026 [2CM8YX]

The two major areas of required enhancements are the increase of the plasma heating power and **the transition to an all-metallic plasma facing component device**. The enhancement must not impede the W7-X long-pulse project goal. Operation at 10MW heating power for a discharge duration of 1800s must be ensured with the focus on three configurations (standard, high-mirror, high-iota).

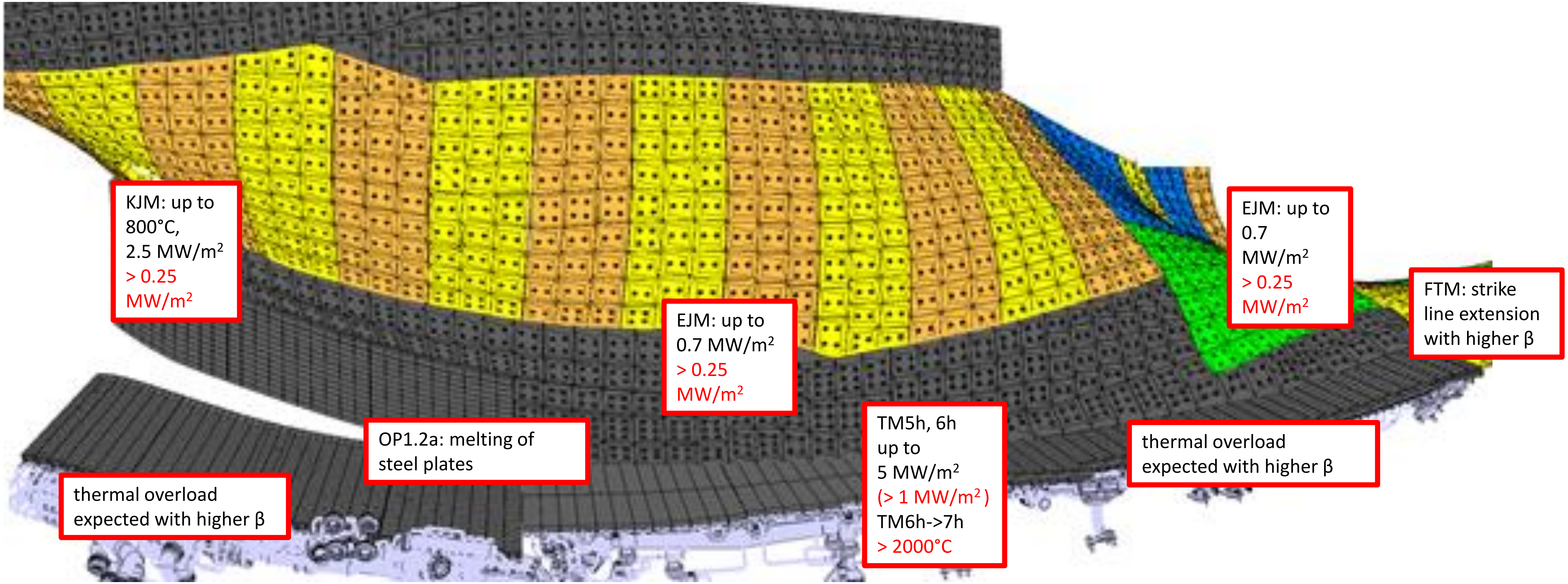
Transition of W7-X to reactor relevant plasma facing materials

- To prove that the stellarator concept can meet the requirements of a future carbon-free fusion reactor
- By demonstrating high-performance, steady-state HELIAS operation

Primary choice: W or W heavy alloy (WNiFe or WNiCu)

Concept phase 2022-2026 ➔ RSR topic → Go – No Go decision 2026 → Series production & installation phase 2027-2032

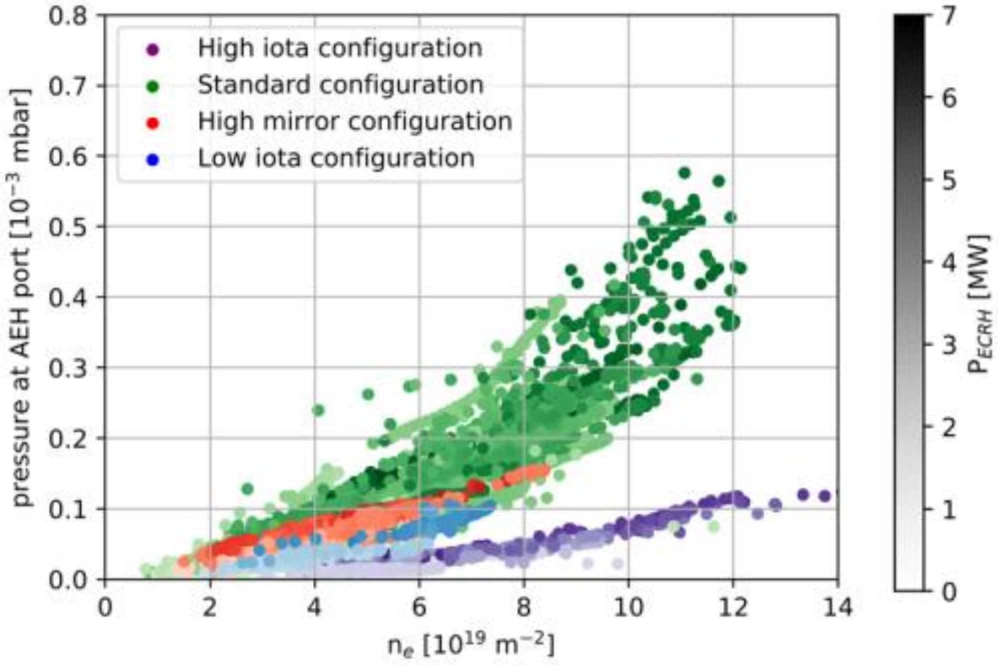
W7-X DCD: learned from OP1 – local overloading

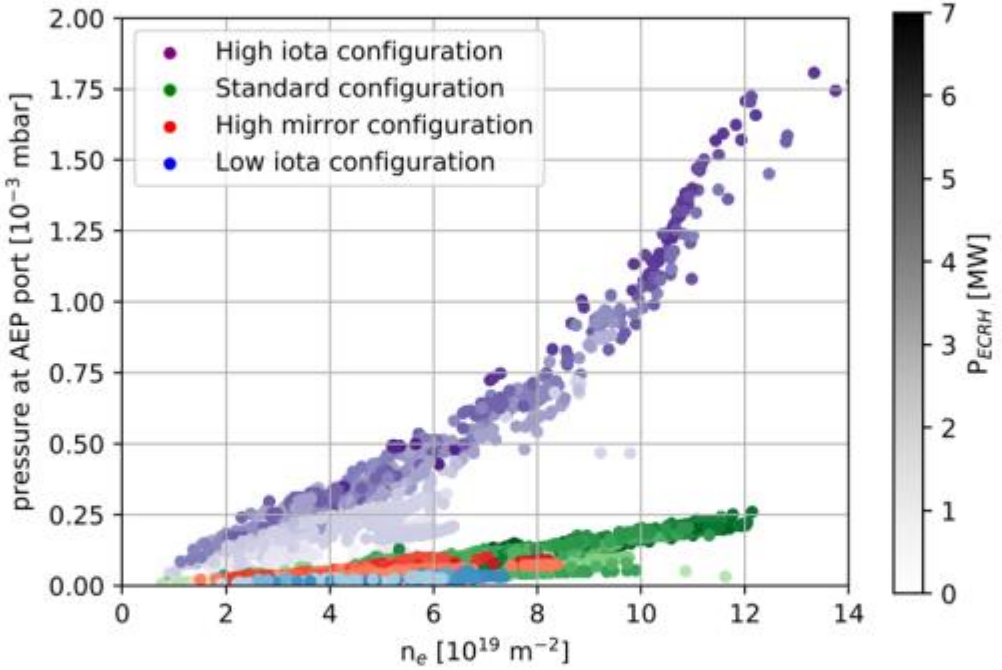


CAD source: M. Krause (1-QRH80-T0001)

W7-X DCD: learned from OP1 – limited exhaust

neutral gas exhaust was sufficient for plasma density control even during long discharges, but low neutral gas pressures in the W7-X sub-divertor region < 2e-03 mbar **(0.2 Pa)**



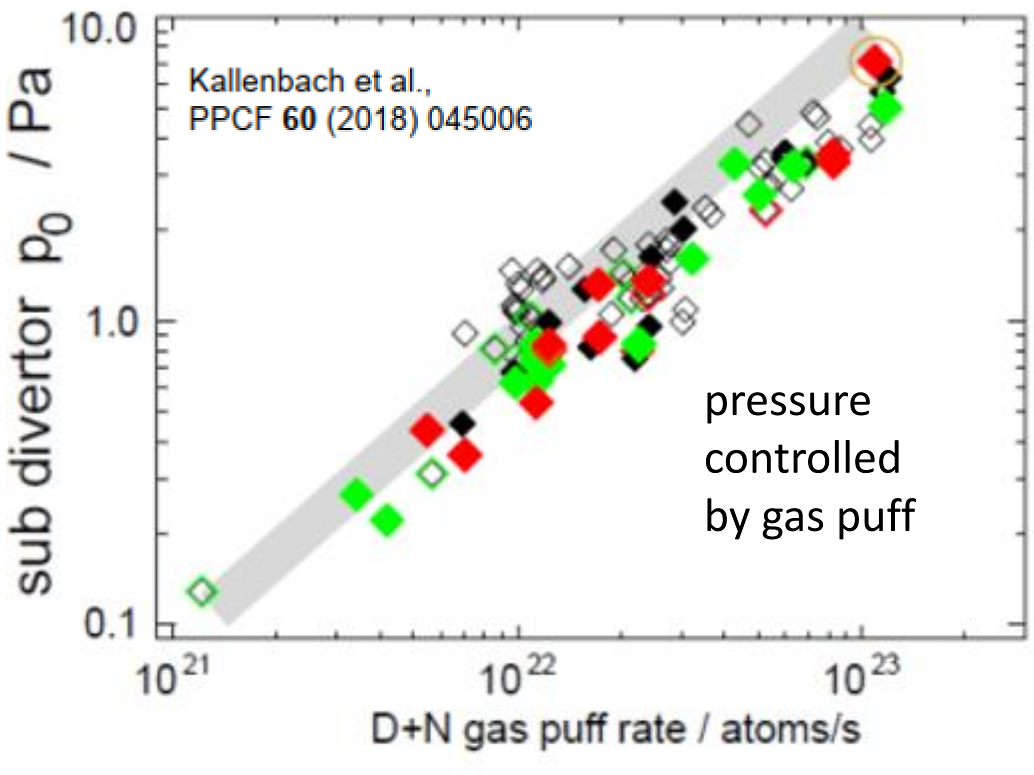


Source: V. Haak et al. Plasma Phys. Control. Fusion 65 (2023) 055024

W7-X DCD: learned from OP1 – limited exhaust

neutral gas exhaust was sufficient for plasma density control even during long discharges, but low neutral gas pressures in the W7-X sub-divertor region < 2e-03 mbar **(0.2 Pa)**

AUG: <0.1 mbar **(10 Pa)**

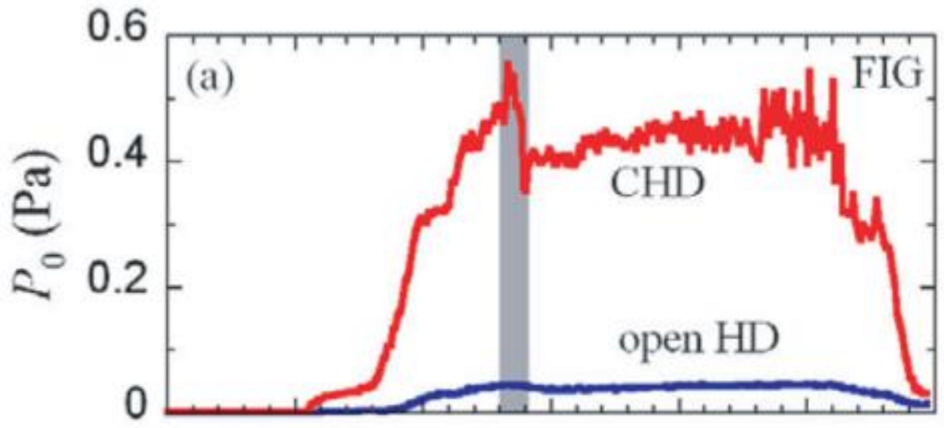


pressure controlled by gas puff

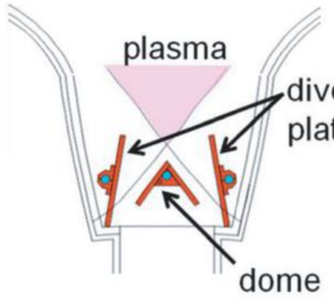
LHD: <0.02 mbar **(2 Pa)**

observed only in the inward shifted configuration with R_{ax} = 3.55 m, but not with R_{ax} = 3.6 m.

Source: U. Wenzel et al. 2024 Nucl. Fusion 64 034002



CHD – closed helical divertor.



Source: T. Morisaki et al. Nucl. Fusion 53 (2013) 063014

W7-X DCD: Divertor functions and objectives

- divert plasma particles
- neutralize plasma particles
- collect neutral particles
- remove neutral particles
- contain particles in sub-divertor
- plug neutral particles in divertor
- screen impurity particles from core
- survive – heat, erosion, forces

particle exhaust

particle retention

mandatory

high particle exhaust rate $\Gamma_{exhaust}$

reactor: $\Gamma_{exhaust\text{-}He} = \Gamma_{Fusion\text{-}He}$

operational: $\Gamma_{exhaust} = \Gamma_{sources, wall, NBI, pellet, gasinlet}$

Source: T. Kremeyer

W7-X DCD: Divertor functions and objectives

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retention of impurities in the divertor plasma

- W core concentration f_W < 2e-05

$$\frac{\partial n_z}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r} r \left(D \cdot \frac{\partial n_z}{\partial r} - v \cdot n_z \right) + S_z,$$

$$f_W = \frac{\lambda_{iz}\,\Gamma_{influx}}{n_e^{core}\, D_\perp S_{pl}}$$

heat loads on PFCs should not exceed their specific, defined limits

- p < 10 MW/m^2 steady state -> GLADIS tests with 15 MW/m^2
- T_{max} (WNiFe) < 1100 °C, (OFE-Cu) < 600°C, CuCrZr < 475°C, channel < 200°C
- wetted areas > 1 m^2, incidence angles < 5°

avoid localized excess heat loads (leading edges, fast particles).

acceptable net erosion

- net erosion rate < 0.2 nm/s ➔ @15 y * 40 days/y * 1800 s/day ➔ 0.22 mm

W7-X DCD: two approaches

Tungsten Focus

use of tungsten based PFCs with moderate modifications to the current W7-X setup

Reactor Focus

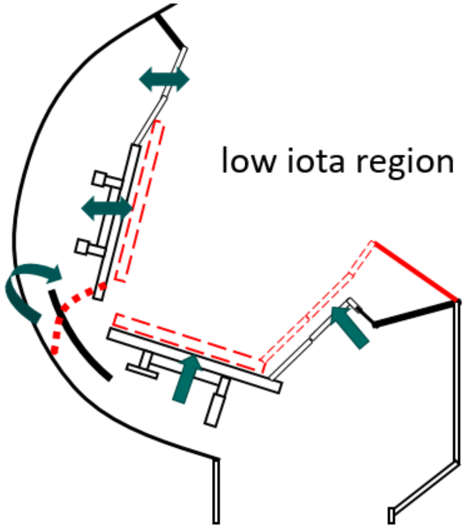
search for an optimized plasma facing surface that meets carbon-free reactor performance requirements on particle removal and impurity retention while guaranteeing target survival

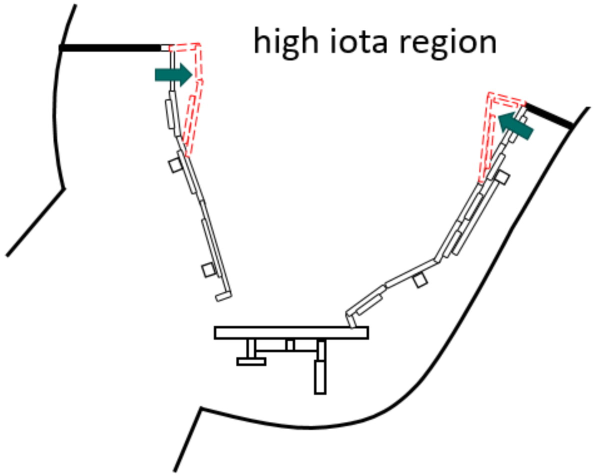
W7-X DCD: two approaches

Tungsten Focus

use of tungsten based PFCs with moderate modifications to the current W7-X setup

divertor with tight baffles for neutral screening

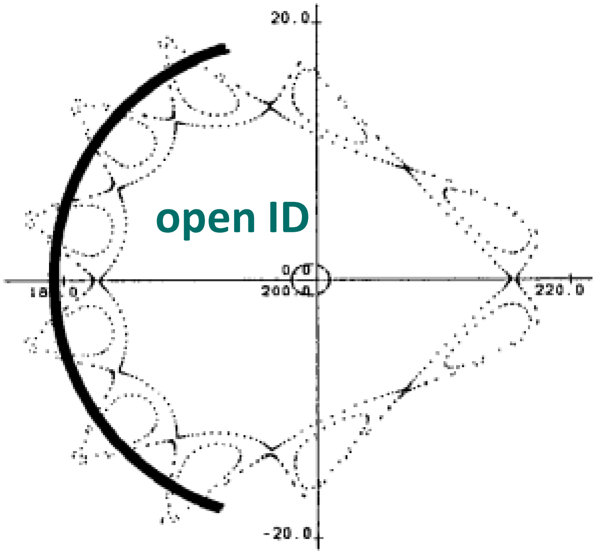


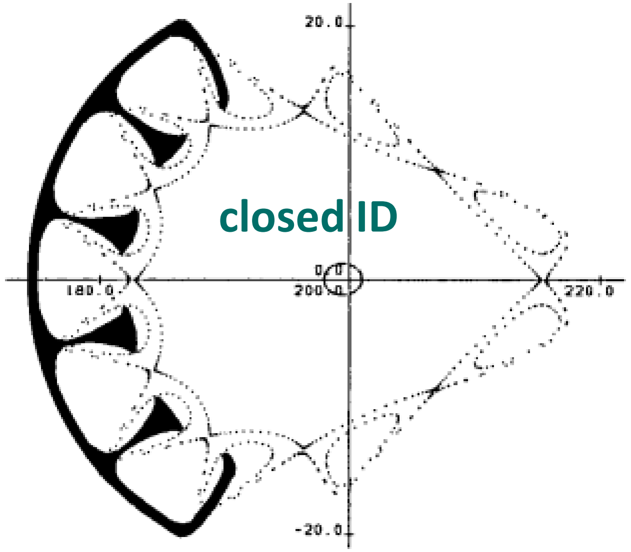


Reactor Focus

search for an optimized plasma facing surface that meets carbon-free reactor performance requirements on particle removal and impurity retention while guaranteeing target survival

“closed” Island Divertor (ID)





”closed” means that the recycling neutrals are mainly directed away from the plasma core

Source: F. Sardei et al. CPP 3-4 (2000) 238-250

W7-X DCD: two approaches

Tungsten Focus

use of tungsten based PFCs with moderate modifications to the current W7-X setup

pros

- \+ develop core/edge scenarios with reactor relevant Tungsten PFCs (low $T_{e,\,div}$ < 5 eV) and additional seeding (w/o carbon) earlier, as the available simulation tools are judged to be suitable for the design of a new W divertor
- \+ keep flexibility in the magn. field configuration space
- \+ resolve the heat load issues observed (incl. TM5h/6h)
- \+ moderate geometry changes to improve the screening of neutrals
- \+ gaining experience with optical diagnostics (incl. IR) with reflections from W surfaces

cons

- – restriction of the fueling systems (e.g. pellet system)
- – particle exhaust expected to improve only slightly – but not to values that would be relevant for extrapolation to a reactor
- – observed impurity retention may not be relevant for extrapolation to a reactor

risk of not being relevant?

Reactor Focus

search for an optimized plasma facing surface that meets carbon-free reactor performance requirements on particle removal and impurity retention while guaranteeing target survival

- \+ aims to reactor relevant particle exhaust and impurity retention
- \+ no exhaust restrictions to operate the fueling systems
- \+ possible reduction of the target area (smaller divertor)

- – extensive code development and validation is required to mitigate the risk of "unexpected" heat load problems (e.g. due to drifts (poloidal SOL flows), beta-, I_{tor}-effects)
- – reduced flexibility in the magnetic field configuration space
- – a new set of divertor diagnostics has to be installed in-vessel
- – modified divertor gas inlet system required
- – risk of having to modify the cryopumps

W7-X DCD: Upgrade strategy options

Case A – Tungsten Focus

- plasma facing surface (PFS) definition for an open W divertor until 2026
- fabrication/installation of an open tungsten divertor (Generation 2) + tungsten wall
- study physics & validate models
- development of a closed divertor for W7-X

Risk mitigation

- install uncooled closed test divertors (TMh5/6h) for one campaign (Generation 3), C or W

- fabrication/installation of a reactor relevant closed tungsten divertor (Generation 4)

Case B – Reactor Focus

- keep open CFC divertor and Carbon wall to study physics & validate models (10 years)
- development of a closed divertor for W7-X

Risk mitigation

- install uncooled closed test divertors (TMh5/6h) for one campaign (Generation 2), C or W

- fabrication/installation of a closed tungsten divertor (Generation 3)

W7-X DCD: current status of tool development

fast tools:
CATIA grid model (developed by DE)
+ EMC3-Lite
(developed by Y. Feng for fast heat load predictions)

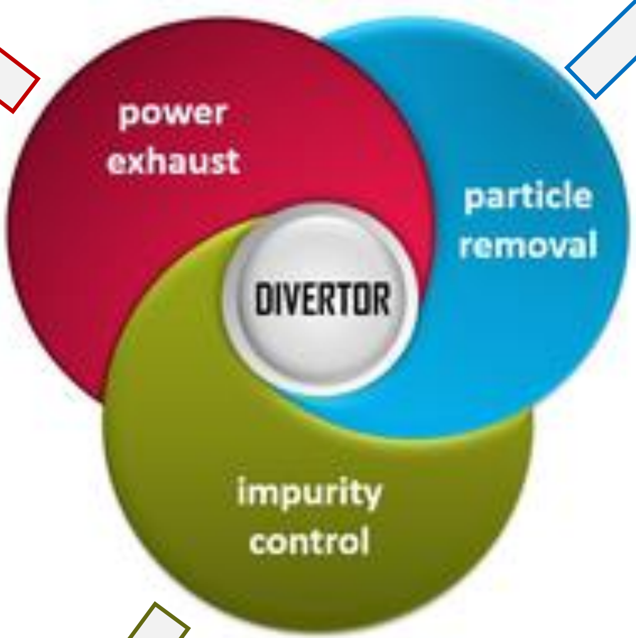
SMoLID (Simple Model for Loads in Island Divertor)
(developed by A. Kharwandikar)

Leading edge reduction optimization tools
(developed by A. Menzel-Barbara)

state-of-the-art: EMC3/Eirene

fast tools: in development (B. Davies), ANSYS (V. Haak)
state-of-the-art: EMC3/Eirene, COMSOL, DIVGAS (KIT)

Karlsruhe Institute of Technology



further activities

- Hermes-3 multifluid, 3D transport/turbulence, fluid neutrals model, extension to SOL planned (B. Shanahan)
- advanced fluid neutral (AFN) and hybrid fluid-kinetic approaches (KU Leuven T. Baelmans)

Source: https://conferences.iaea.org/event/286/contributions/25153/

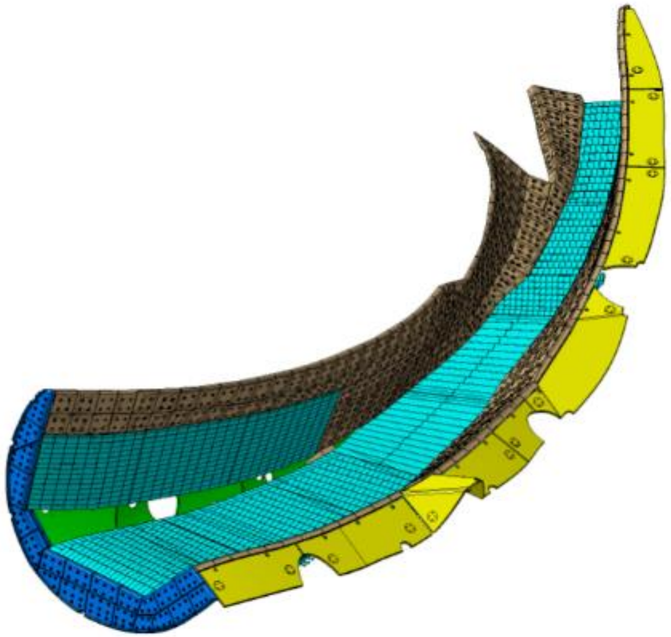
JÜLICH Forschungszentrum

state-of-the-art: EMC3/Eirene, ERO2.0 (FZ Jülich, IFN)

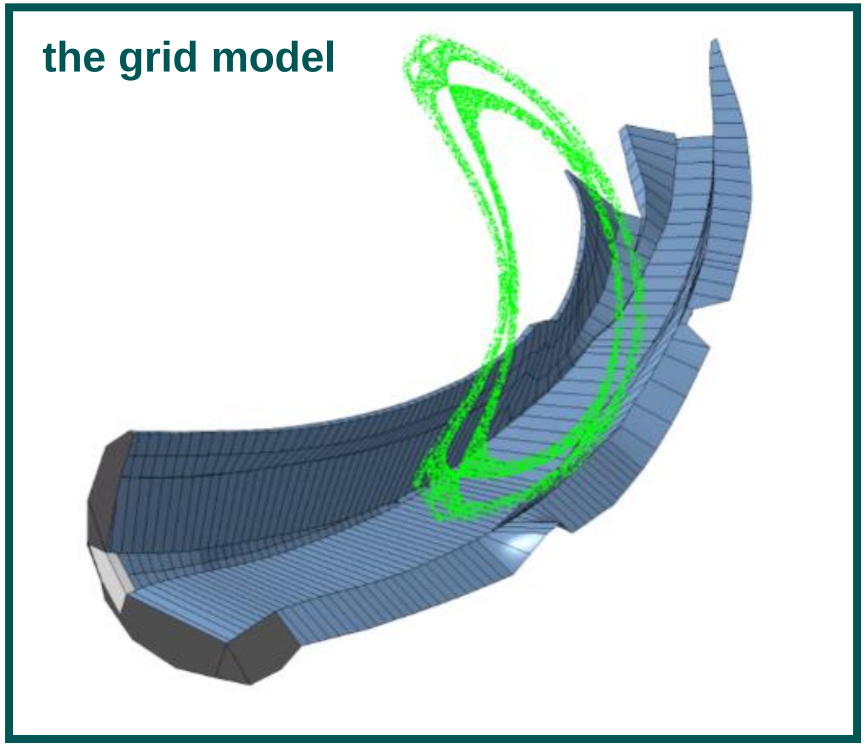
open issues:

- fast neutral particle modelling
- drift & turbulence effects
- impurity transport in SOL & core
- edge/core scenario development

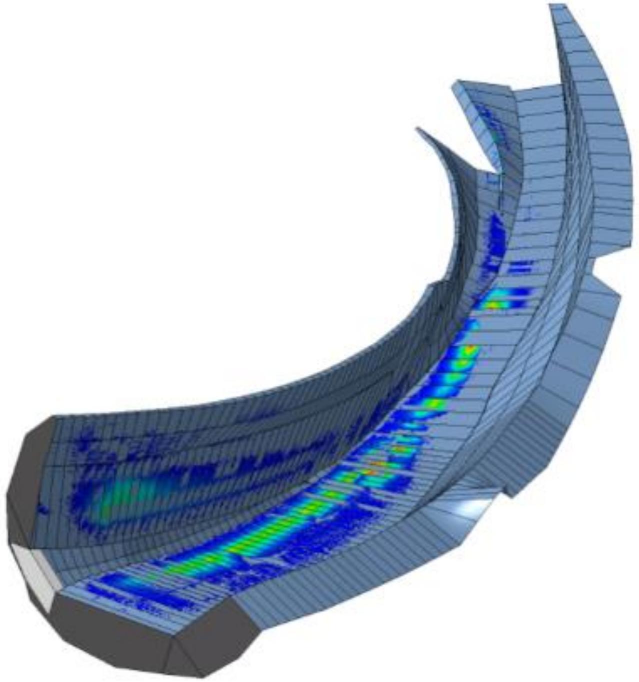
W7-X DCD: development of design tools CATIA+EMC3-Lite



detailed CAD geometry of
one divertor unit



reduced grid-based stream-line model
with limited number of grid points –
rdeveloped by DE IPP Greifswald



EMC3-Lite results mapped onto
the grid model

the modeling activities need be supported by the development of efficient engineering tools in a CATIA environment that process the complex 3D W7-X design data at different levels of sophistication to promote an efficient exchange with the physics-based codes

Source: T. Sieber

Source EMC3-Lite: Y. Feng Plasma Phys. Control. Fusion 64 (2022) 12501

W7-X DCD: development of design tools CATIA+EMC3-Lite

checking >200 configurations of standard, high-mirror, high iota for different beta (up to 4%) & I_{tor} (-20 kA to 20 kA) as provided by VMEC/EXTENDER (J. Geiger)

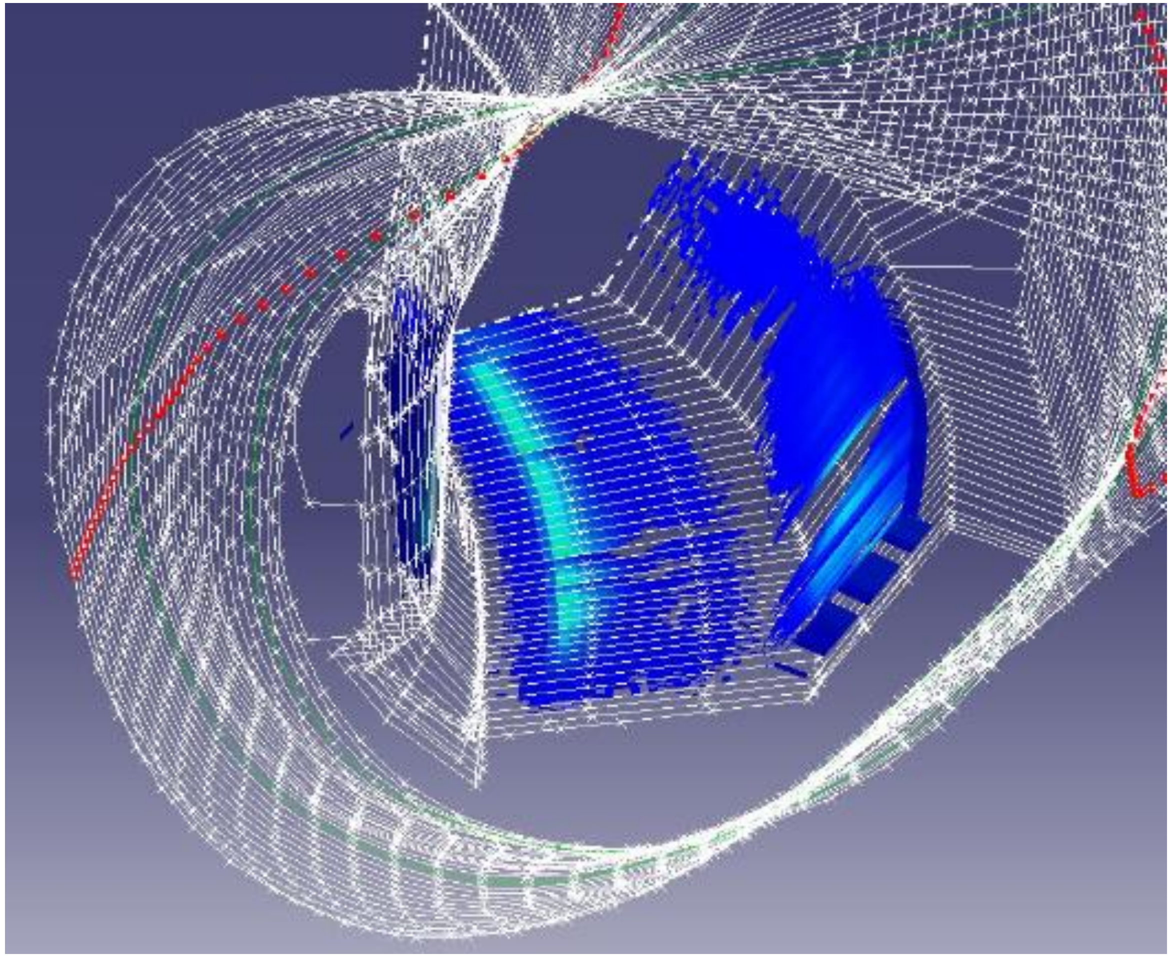
MHD-equilibrium => **VMEC**

- assumes existence of flux surfaces
- no separatrix structure, no islands
- based on energy minimization

EXTENDER to complete field outside of **VMEC**

combination VMEC-solution, EXTENDER-field and field of coils (Biot-Savart) enables fields everywhere

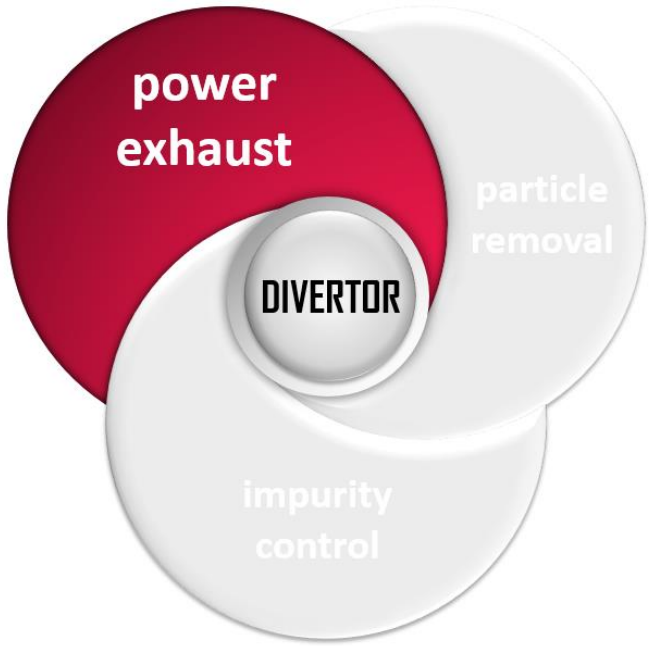
check with **HINT** is planned for selected cases



Example: heat load distribution for a modified geometry

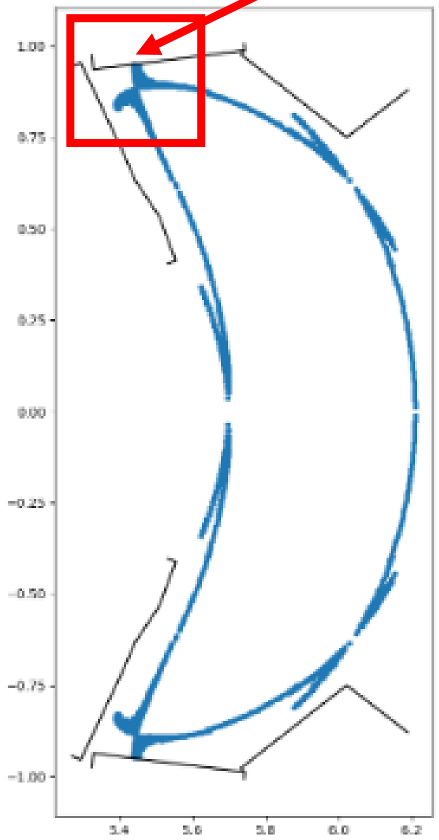
Source: J. Dettmar, T. Sieber, Ch. Voß

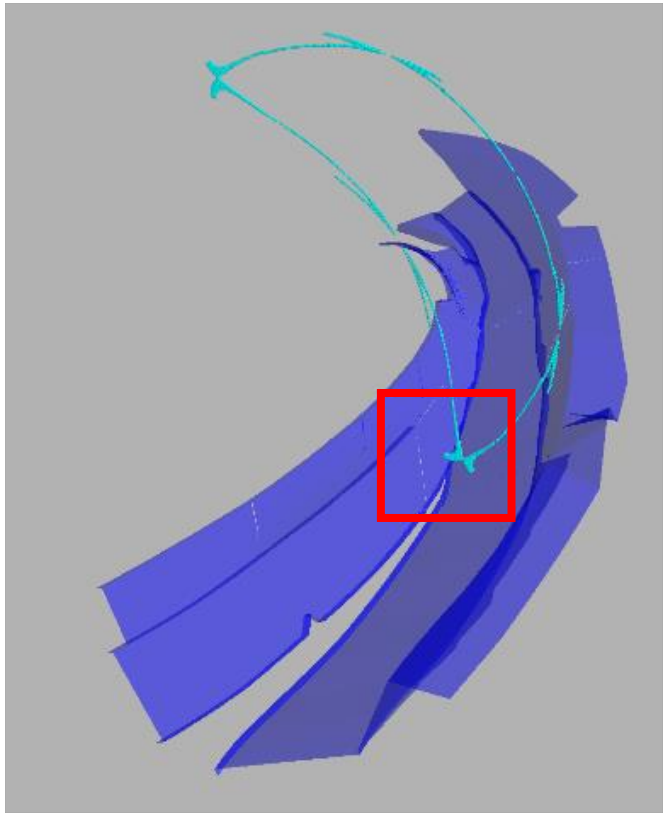
W7-X DCD – power exhaust



W7-X DCD: concept of “power carrying shells”

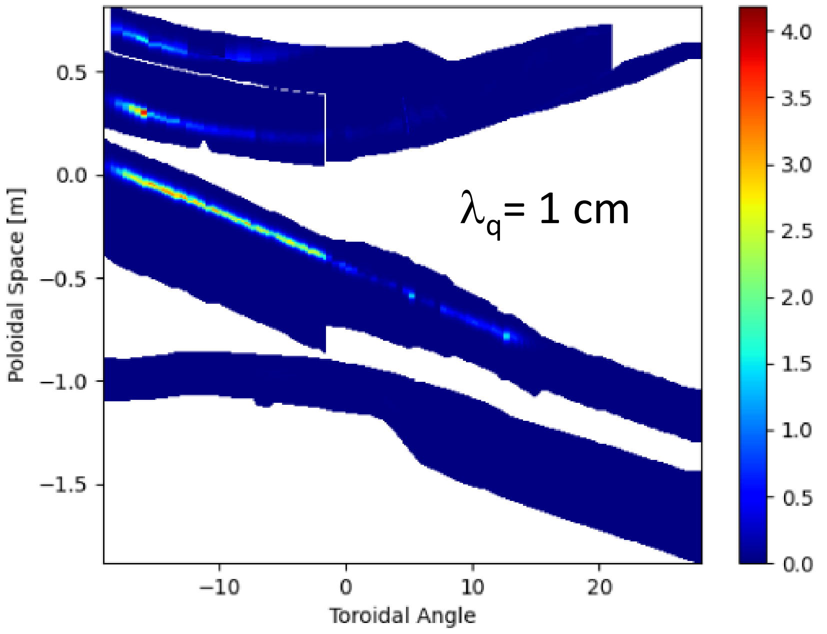
strike line positions





construction of a flux surface with
high parallel power fluxes = long connection length

SMoLID
(Simple Model for Loads in Island Divertor)



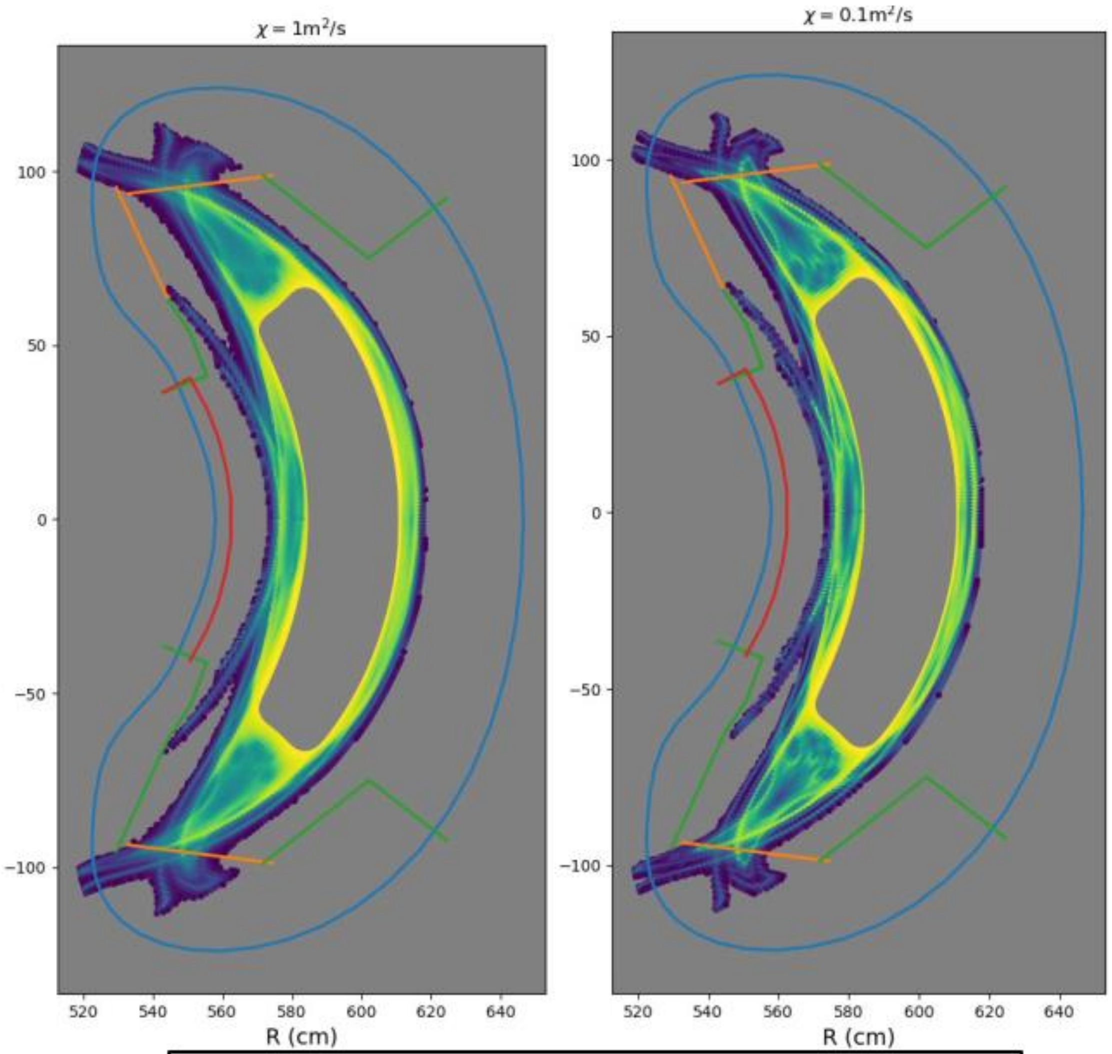
heat load distribution P_s

$$P_s\ [MW/m^2] = P_{parallel}\ ^{x}\ \sin\alpha\ ^{x}\ \exp(-\Delta/\lambda_q)$$

Δ distance from the power carrying shell to the components
λ_q power decay length in the SOL

Source: A. Kharwandikar, https://conferences.iaea.org/event/286/contributions/25129/

W7-X DCD: concept of "power carrying shells"



Exceptionally Chaotic W7-X configuration GYM+1750

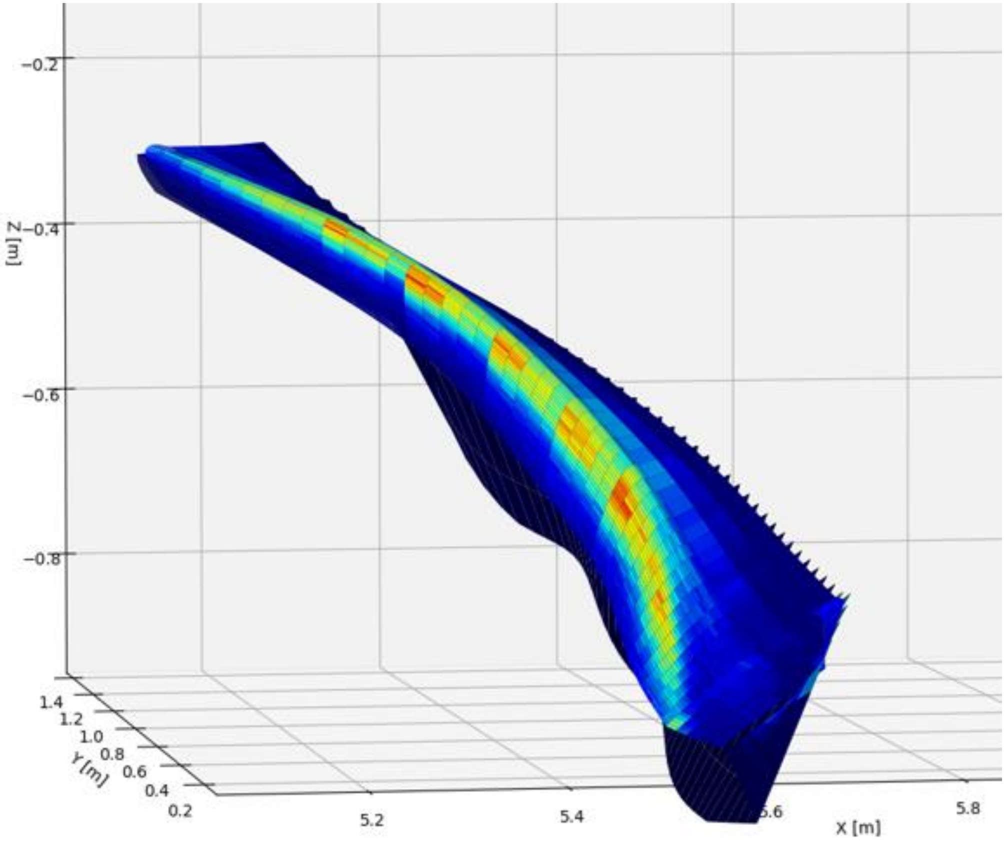
- **Aim**: Quantitatively describe edge heat transport, assuming diffusive transport.
- **Method**: Run EMC3-Lite simulation *without* PFCs and record trajectories of Monte Carlo particles. Metric describes accessibility of each EMC3-Lite cell
  - **Accessible: Many MC particles sample cell after a small number of timesteps**
- Benefits:
  - **Includes perpendicular transport**
  - **PFC-independent**
  - **Relatively fast (~1 CPU-minute)**
- Future work: benchmark against EMC3-Lite simulated PFC heat loads
- Possible applications for W7-X:
  - **Understanding magnetic chaos**
  - **Semi-automated PFC design**
  - **Divertor resiliency quantification**

target-independent heat transport metric

Source: R. Davies, J. Fellinger, S. Henneberg

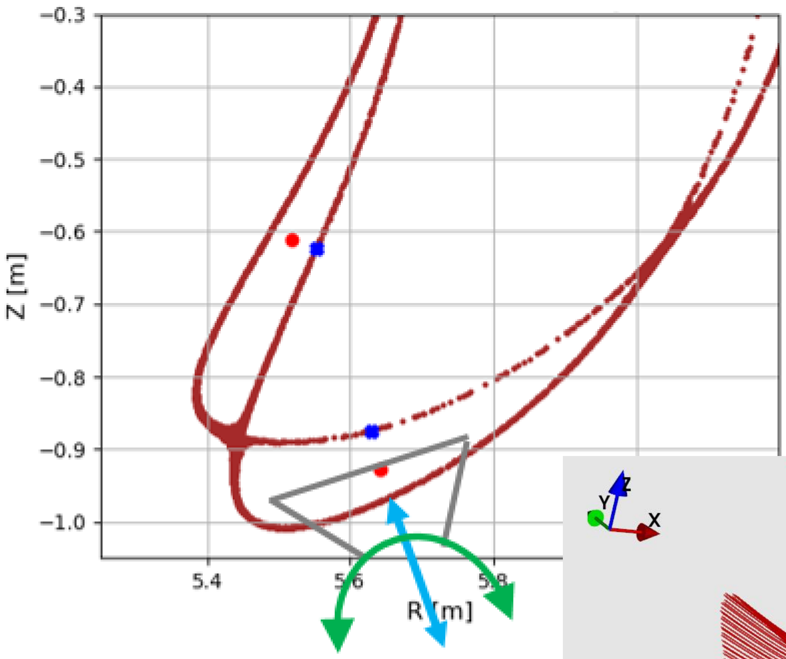
W7-X DCD: running studies

submerged divertor design tool

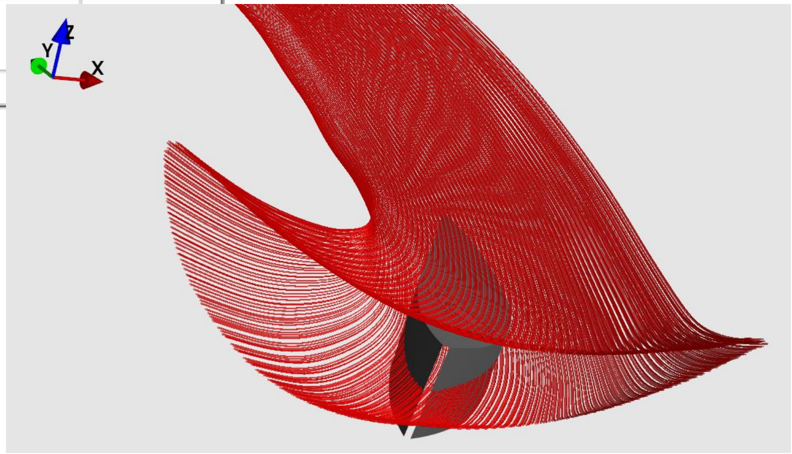


example: standard configuration, target phi=5° to 28°, loads below 10 MW/m^2

Source: A. Menzel-Barbara

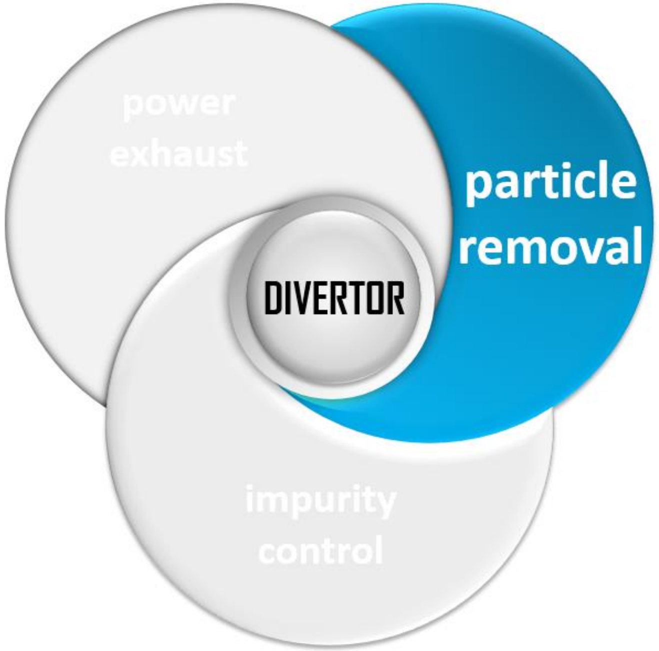


closed divertor concept: with targets deep in the island

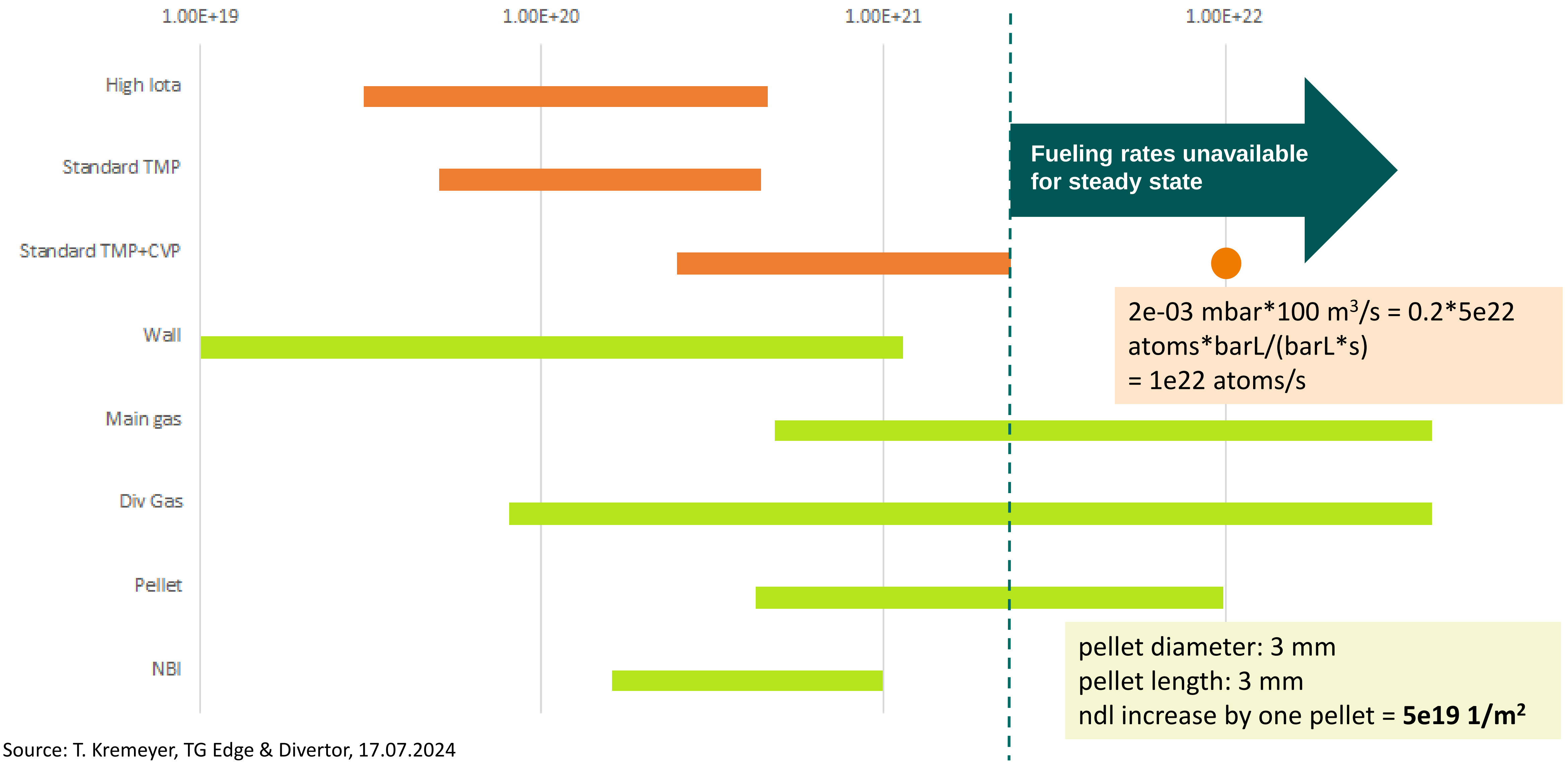


Source: A. Kharwandikar

W7-X DCD – particle removal



W7-X DCD: Exhaust limited density control



Source: T. Kremeyer, TG Edge & Divertor, 17.07.2024

W7-X DCD: Towards reactor relevant high-recycling regime

tokamaks show high-recycling regime

$$n_{tar} \propto n_{up}^3$$

beneficial for:

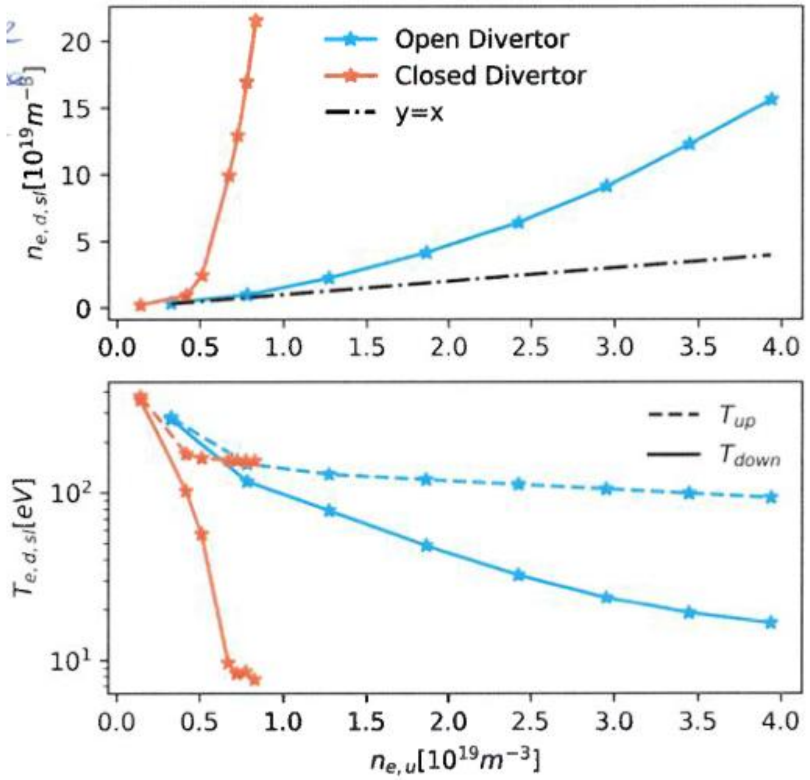
- plasma radiation
- particle exhaust
- impurity retention

absence of high-recycling regime in W7-X

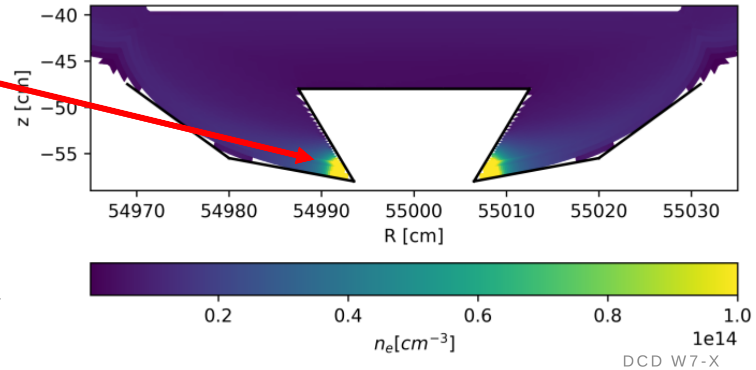
→ Can a closed divertor cure this?

aspects why this can be expected:

- better neutral & plasma baffling
- changed flow pattern (flow upstream reduced):
  - ➢ different momentum losses (f_{mom})
  - ➢ less convective parallel energy transport (f_{conv})

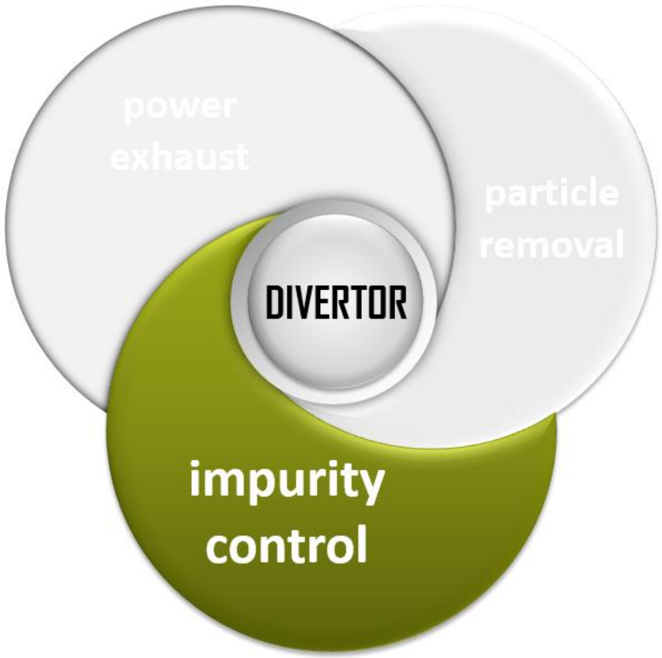


high n_e at the targets



Source: N. Maaziz, PSI 2024

W7-X DCD – impurity control

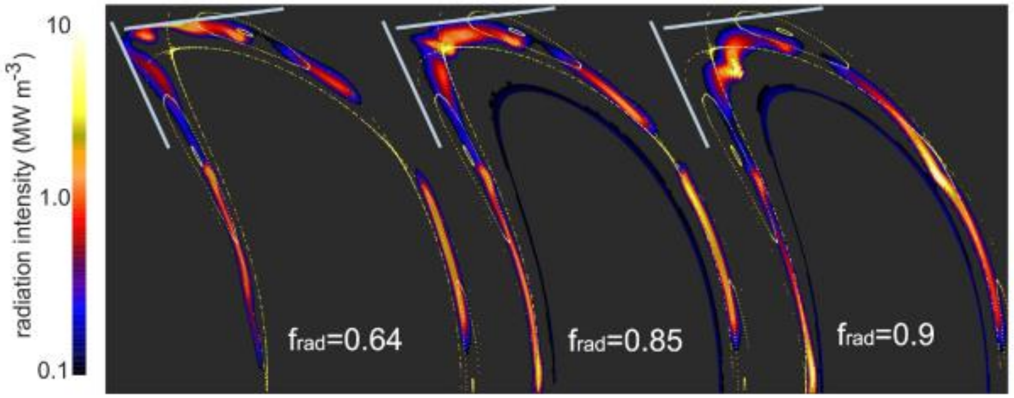


W7-X DCD: impurity transport in SOL and core

stable detachment (with impurity seeding) and sufficient impurity retention in SOL simultaneously maintaining power and particle exhaust as well as core performance

$$F_Z = -\frac{1}{n_Z}\frac{\mathrm{d}p_Z}{\mathrm{d}s} + m_Z\frac{(v_i - v_Z)}{\tau_s} + ZeE + \alpha_e\frac{\mathrm{d}(kT_e)}{\mathrm{d}s} + \beta_i\frac{\mathrm{d}(kT_i)}{\mathrm{d}s}$$

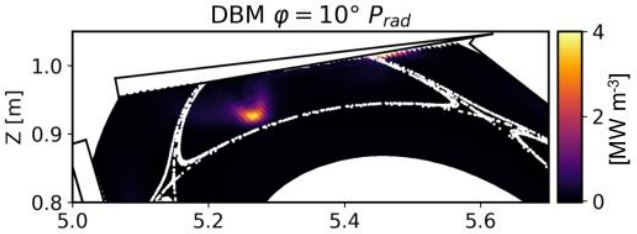
Source: Y. Feng et al. 2024 Nucl. Fusion 64 086027

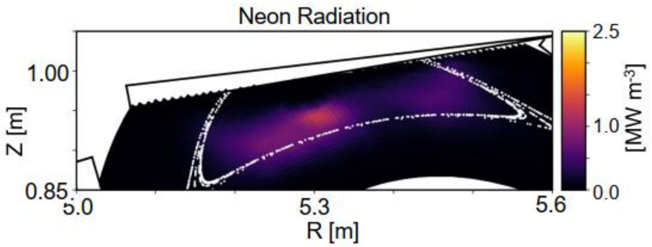


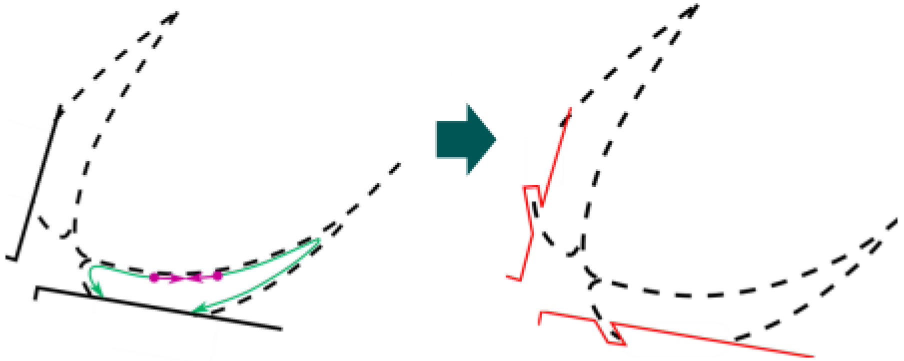
perpendicular transport is the limiting factor for impurity retention in the W7-X island SOL

- impurities leak to the LCFS via perpendicular transport across flow stagnation region
- ionization length still affects impurity retention by changing distance between ionization front and LCFS

closing the divertor would prevent impurity neutrals from escaping to island O-point region/keep them further from LCFS







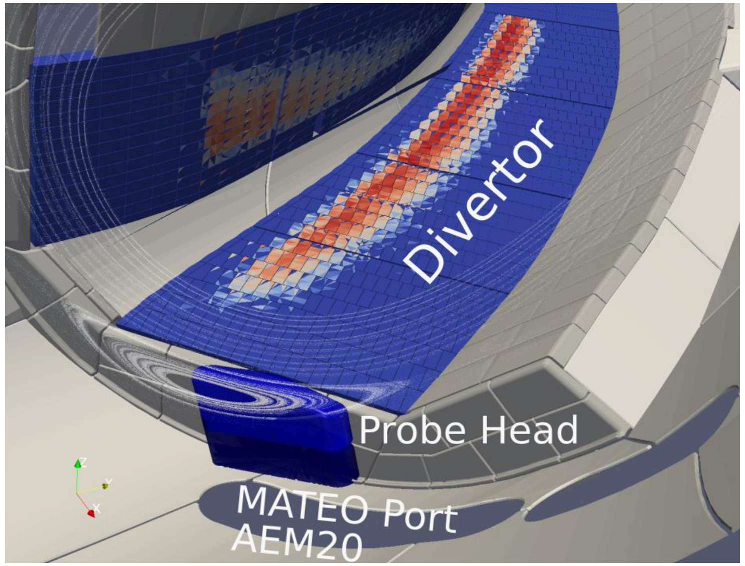
different radiating species may lead to more stable radiation scenarios: Neon radiates further upstream, and is more evenly distributed in simulations

Source: V. Winters, https://conferences.iaea.org/event/286/contributions/25140/, EPS 2024

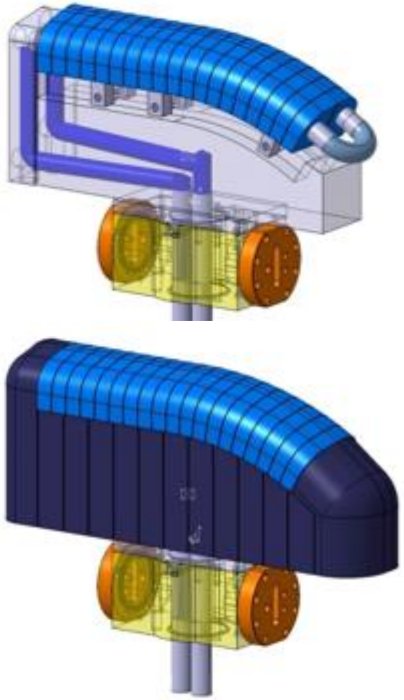
W7-X DCD: tungsten transport in SOL and core

- edge scenarios with tungsten (low Te and seeding required, low W core concentration) are different from scenarios with carbon PFCs (carbon as radiator, high Te allowed)
- permanent installation of tungsten PFCs (seeing all configurations and parameter variation during a campaign) has already shown critical issues (with ECRH, NBI operation)
- MPM, TESPEL, LBO, **MATEO**, powder injections will provide valuable data on erosion and transport

MATEO - Manipulator for Target Exposure and Observation



Source: S. Brezinsek, A. Knieps W7-X Physics Meeting, May 2024



first purpose:
controlled assessment of PWI effects (incl. erosion and deposition) on divertor materials

second purpose:
characterization of downstream plasma parameters near divertor (required for modeling of PWI processes)

PWI experiments require long-pulse scenarios with safe heat loads and known background plasma

W7-X DCD: current procedure

- **start with power exhaust analysis for attached conditions: definition of modified geometries meeting the criteria:**
    1. keep maximum heat load below 10 MW/m^2 with a heating power of at least 10 MW,
    2. keep the heat load only on the divertor targets (> 95%).
    3. minimize divertor target surface
- **evaluate modified geometries against particle removal requirements**
    1. ensure high neutral gas exhaust
- **identify potential impurity retention drawbacks**
    1. maximize distance to core plasma
    2. keep ionization front away from LCFS
    3. friction force should be dominant over thermal force

W7-X DCD: what about the baffles and heat-shields

conservative approach

Use graphite tiles with a 10 µm MedC tungsten layer (NILPRP Bucharest), replacing the carbon tiles step by step (a 2000 tiles).

pros

- proven technology, tested in various fusion experiments (incl. W7-X)
- a step-by-step approach to investigate potential problems by operating with tungsten surfaces at the baffle areas, which under certain conditions receive convective particle and heat loads
  -> low impurity retention? tungsten dust?
- manufacture and installation during the next years

cons

- regular cleaning is required (as in AUG with deionized water)
- carbon (edges, backside) remains in the device (otherwise full coating required)

innovative approach

Replacement of the carbon tiles with tungsten-coated copper plates, replacement of the complete heat sinks in the baffle areas. Replace all 8040 tiles in one step.

- the technology still needs to be developed and tested in HHF devices and in fusion experiments (first encouraging results have already been obtained)
- would make it possible to replace the carbon tiles in the device: a big step towards a "carbon-free" device

- a “big leap” could result in unexpected problems which could require correction activities (loss of operation time)
  -> OP2.1 with only about 200 tungsten tiles has already shown critical issues (with ECRH, NBI operation)

W7-X DCD: Summary

https://event.ipp-hgw.mpg.de/category/63/
https://datashare.mpcdf.mpg.de/s/EPkFnQ5TXRYoNV8
\\share\mp\P224-Wdivertor

For W7-X, a new design of the divertor components with a redefined plasma-facing surface (PFS) is planned, aiming at tolerable thermal loads, high particle exhaust and sufficient impurity control. In 2021, the development and qualification of target elements with tungsten-based armor material (W or W heavy alloys) with a load specification of 10 MW/m^2 in steady-state operation was started in the framework of EUROfusion (WPDIV).

parallel activity of the physics concept and technology development with continuous exchange of information:

-> development of technical solutions for the various individual components
-> definition of plasma facing surface will be an essential input for the integrated design of sub-divertor components

modelling activities within the plasma domain and within the sub-divertor domain with defined interface at the pumping gap:

-> several iterations as well as comparison with experimental results required (OP1.2, OP2)
-> outcome: definition of the plasma facing surface (PFS)

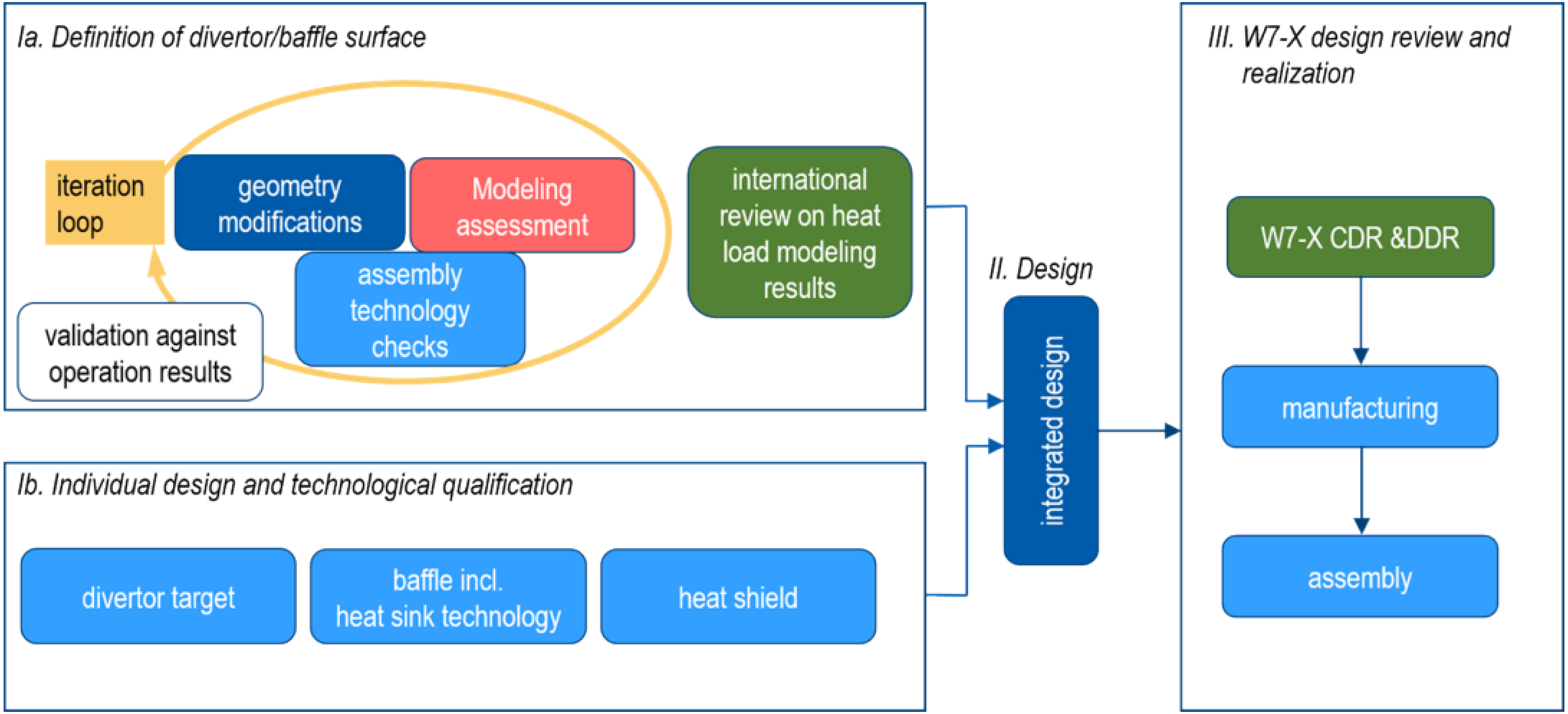
For case A (Tungsten Focus), the available tools are judged to be adequate to allow a proposal for the plasma facing surface (PFS) in the planned project time.

For case B (Reactor Focus), extensive code development and validation is required to mitigate the risk of "unexpected" heat load problems -> requires additional (not yet available) resources.

W7-X DCD:

APPENDIX

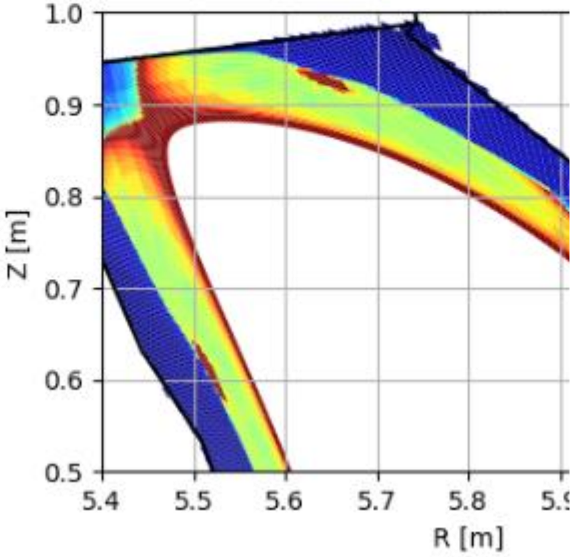
W7-X DCD: Steps of concept development (TWG report)

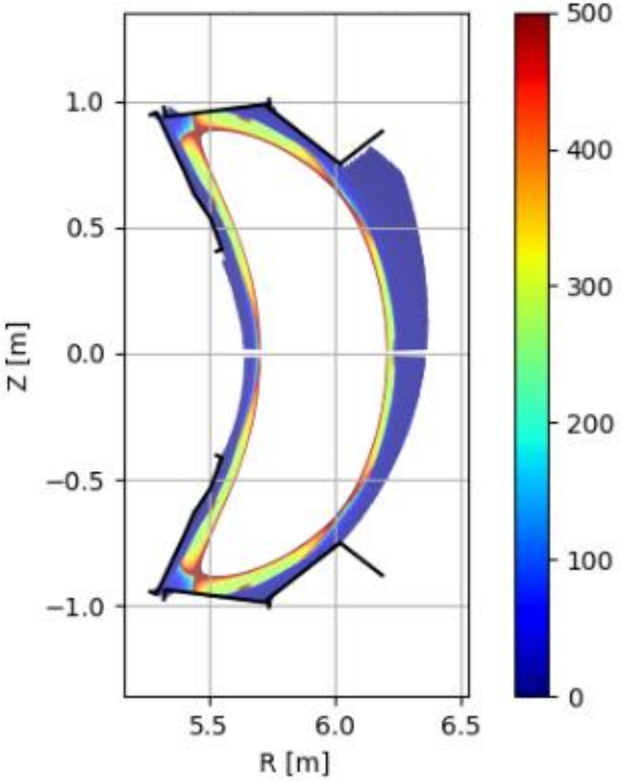


Source: TWG report 1-YLF-T0000.0 [2B3SZA]

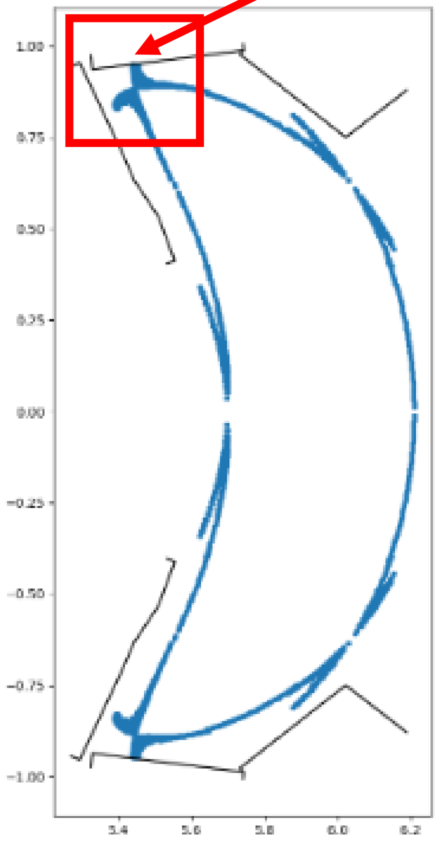
W7-X DCD: concept of “power carrying shells”

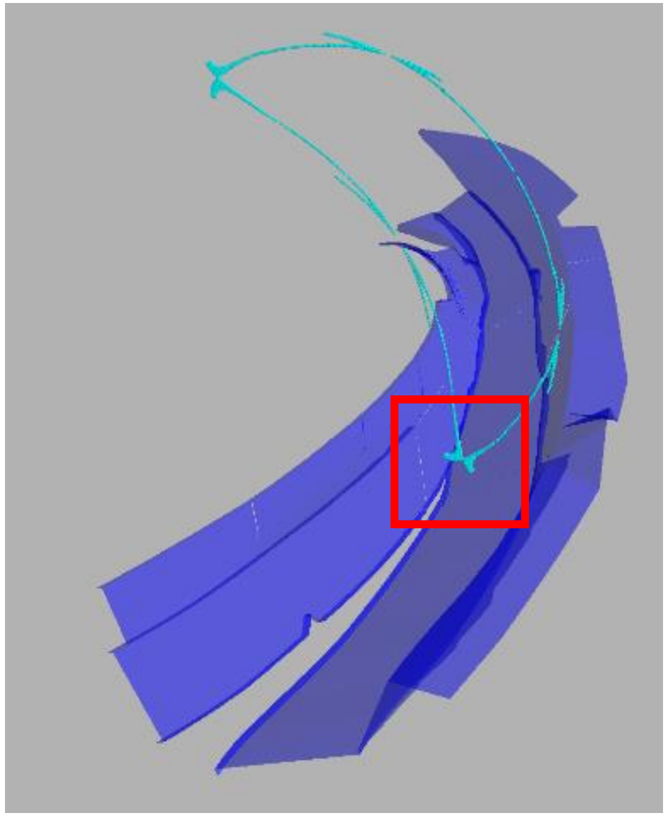
connection length
at phi=0





strike line positions



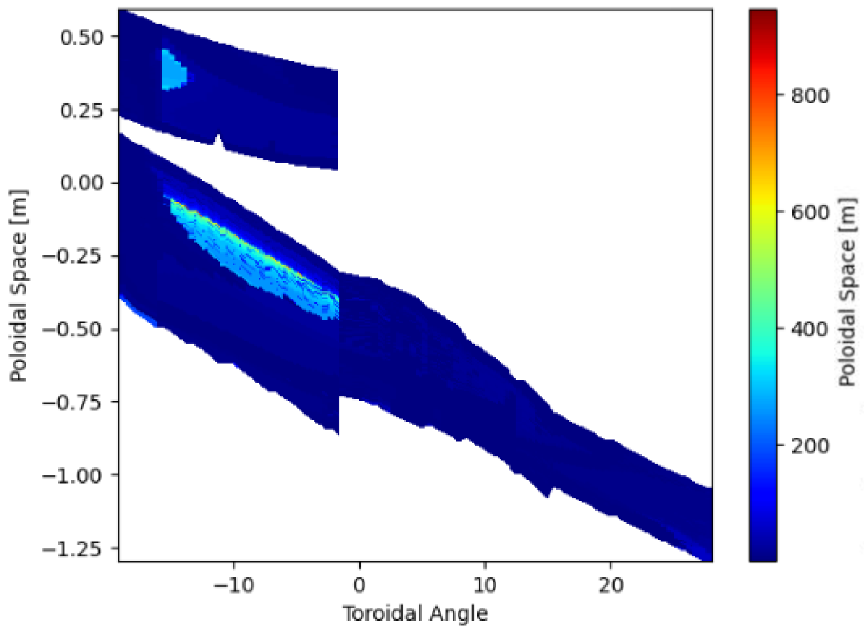


construction of a flux surface with
high parallel power fluxes = long connection length

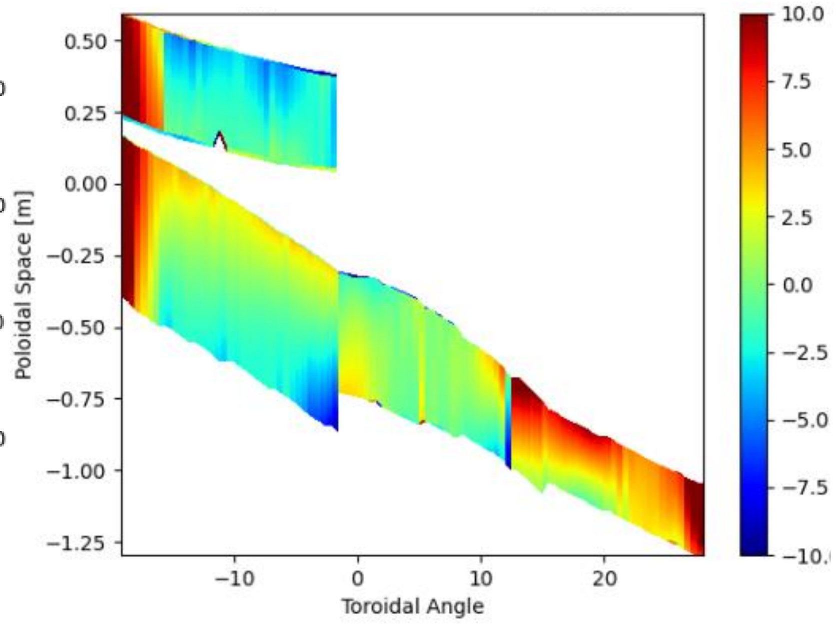
flux tubes with long connection lengths are supplied
with energy via cross-field transport from the core

Source: A. Kharwandikar, https://conferences.iaea.org/event/286/contributions/25129/

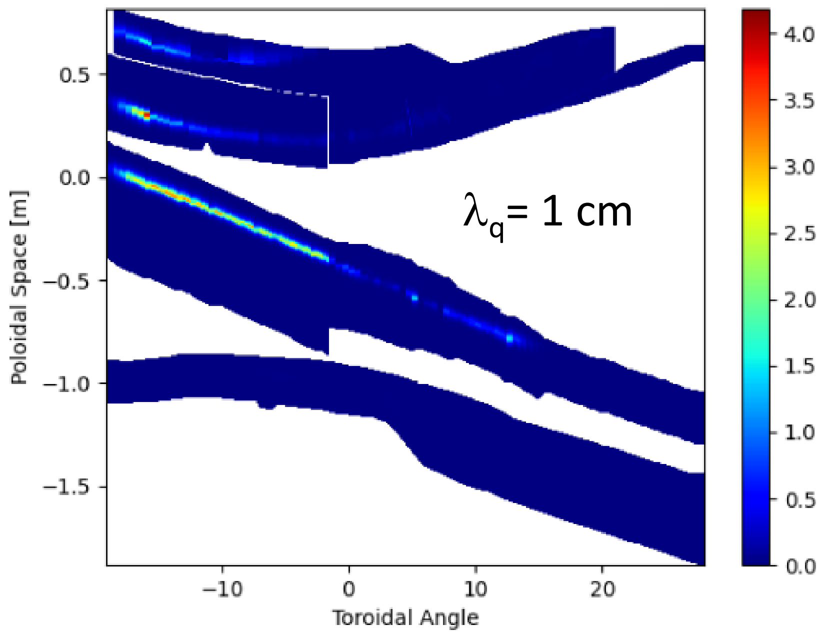
W7-X DCD: concept of “power carrying shells”



pattern of connection lengths [m]



pattern of angle of incidence



heat load distribution P_s

Δ distance from the power carrying shell to the components

λ_q power decay length in the SOL

3D heat load distribution is defined by parallel/perpendicular transport of energy and by the angle of incidence: P_s [MW/m^2] = $P_{parallel}$ ^{x} sinα ^{x} exp($-\Delta/\lambda_q$)

SMoLID

(Simple Model for Loads in Island Divertor)

Source: A. Kharwandikar, https://conferences.iaea.org/event/286/contributions/25129/

W7-X DCD: Power exhaust limit

regression analysis of radiation data

- consistent scaling with line-integrated density with intrinsic impurities and seeding

extrapolate detachment with radiation scaling

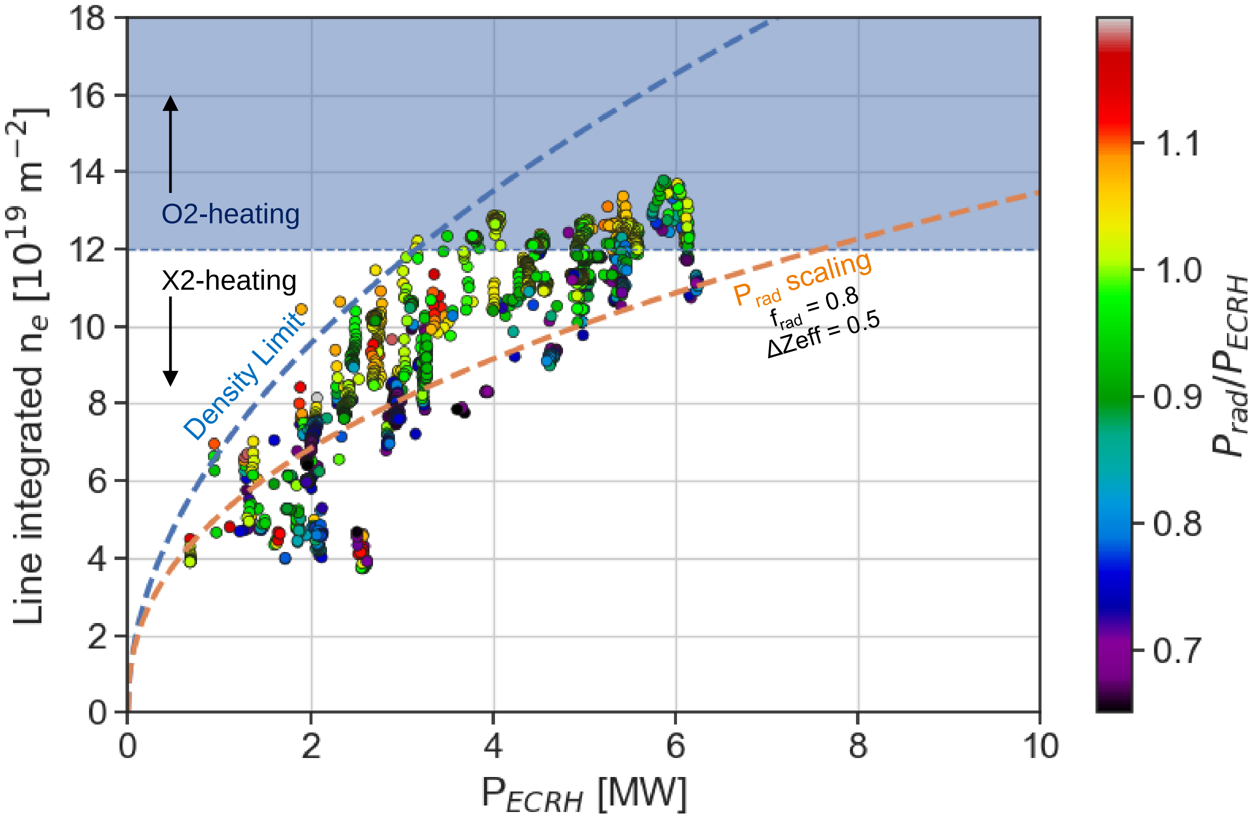
- detachment qualifier: $f_{rad} > 0.8$
- intrinsic impurities / low seeding: $\Delta Z_{eff} = 0.5$
- detachment limitations with ECRH:

X2-Heating $(n_c = 1.2 - 1.4x10^{20} m^{-3})$

$\rightarrow P_{lim,det} = 10\ MW$

O2-Heating $(n_c = 1.8x10^{20} m^{-3})$

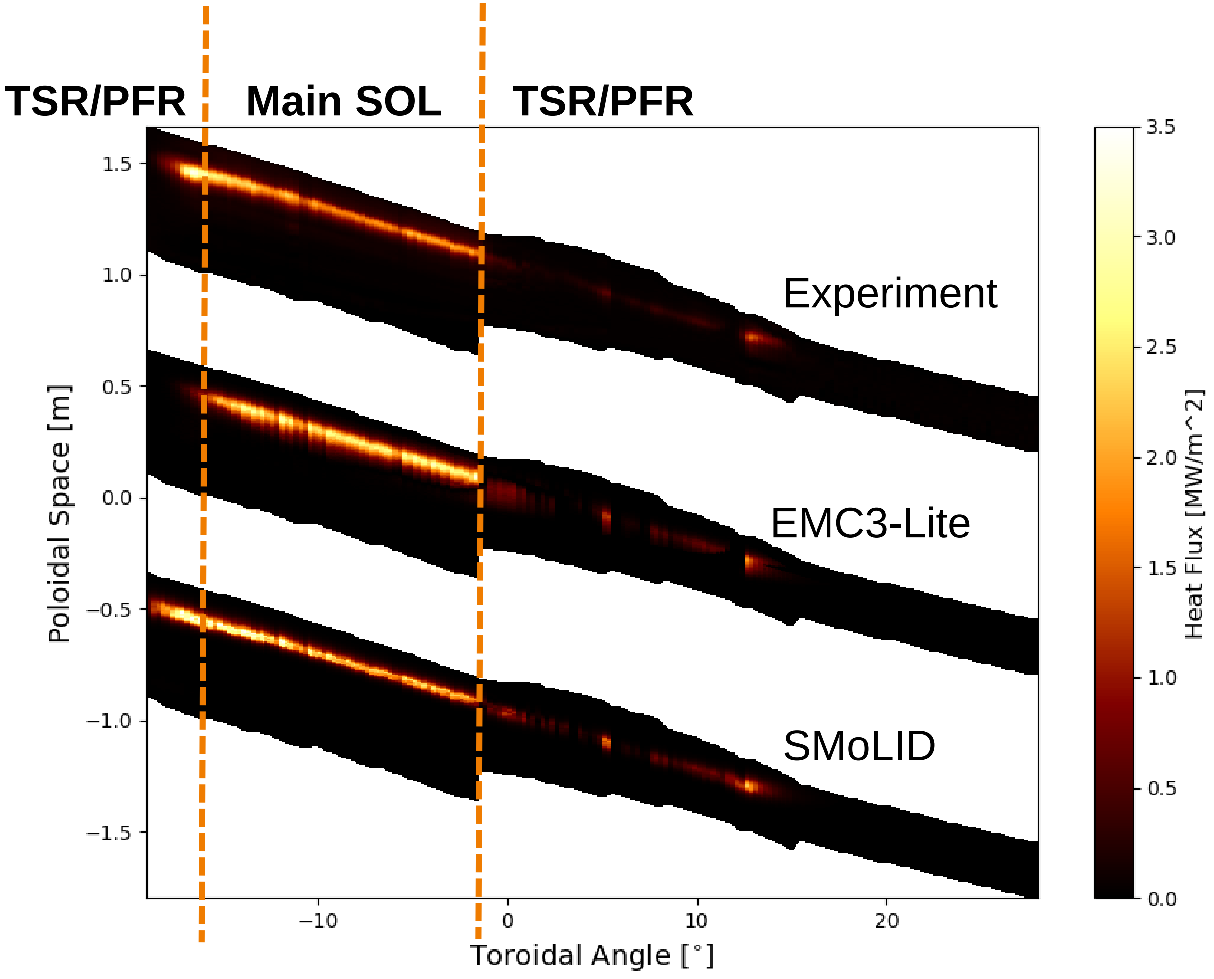
$\rightarrow P_{lim,det} = 20\ MW$



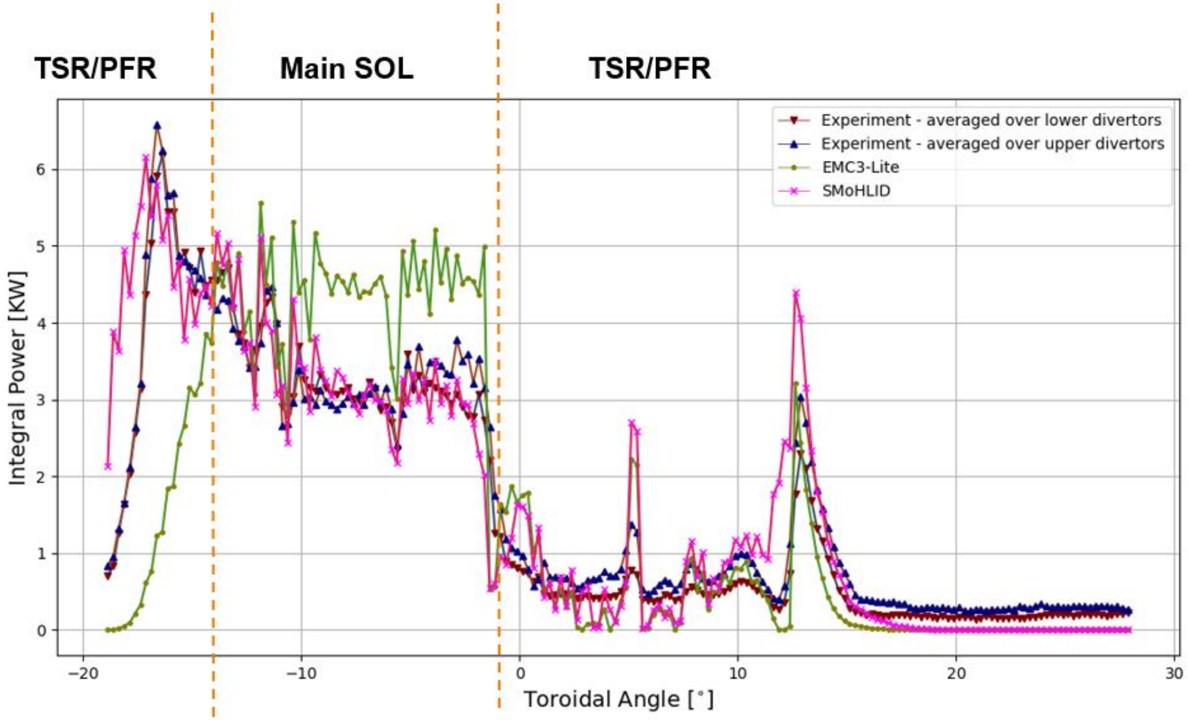
Source: F. Reimold, PSI 2024

Validation of tools – heat load distribution

unexpected difference in the toroidal distribution of the heat flux on the low iota target as predicted by EMC3/Eirene [D. Bold et al. 2022 Nucl. Fusion 62 10601]



Source: A. Kharwandikar



Source: A. Kharwandikar

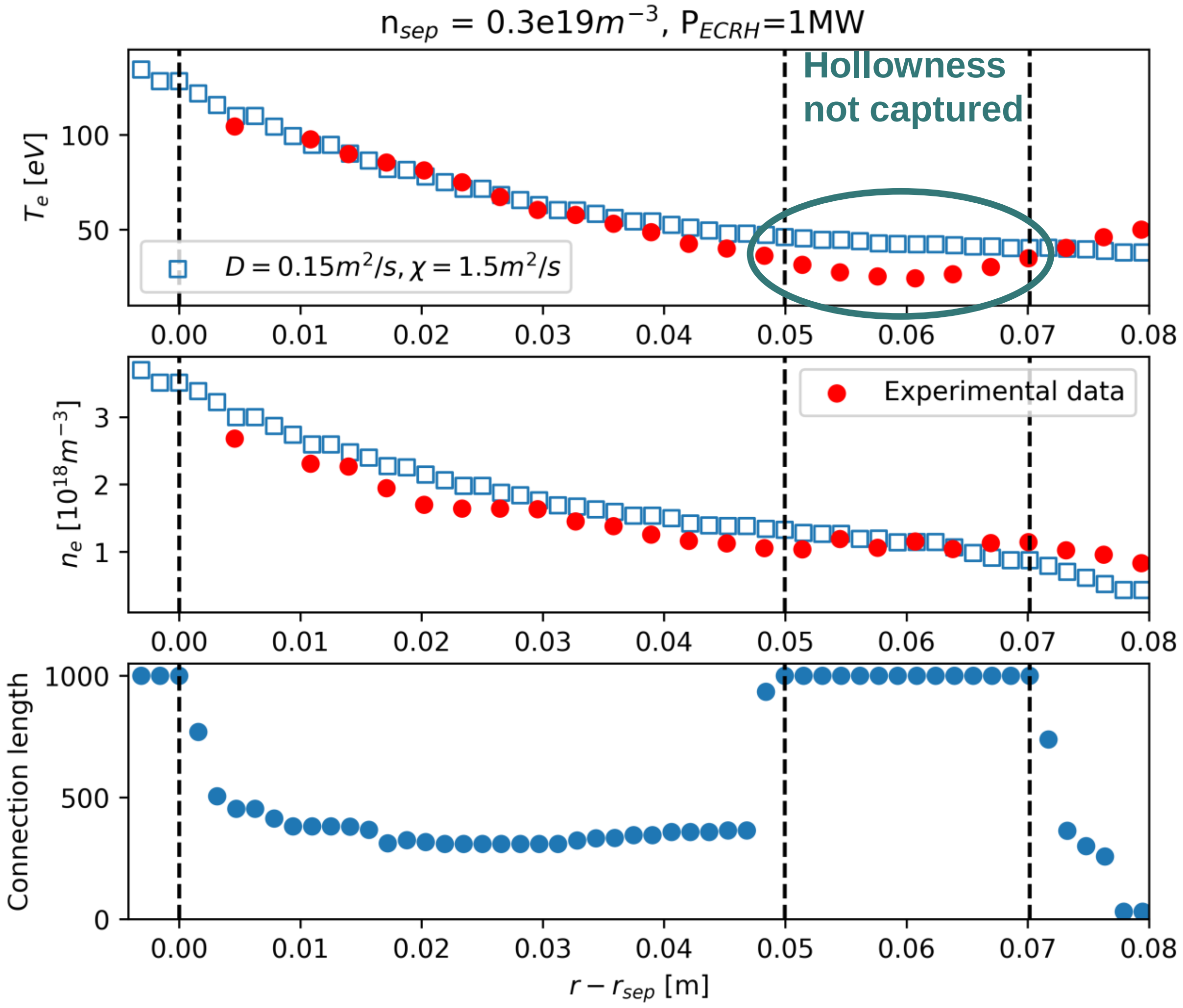
more experiments required to fill the data base for detailed comparison studies (P, n, beta, conf. scan)

Validation of tools – EMC3/Eirene choice of diffusion coefficients

- Diffusion coefficients are:
  - $D = 0.15\ m^2 s^{-1}$
  - $\chi = 1.5\ m^2 s^{-1}$
- Coefficients fitted for:
  - **Upstream** value
  - Fall-off lengths
- Low density and low power discharge

In [D. Bold et al, NF (2022) 62] ratio of heat diffusion coefficient χ to the particle diffusion coefficient D (0.2 $m^2 s^{-1}$), $\chi/D = 3$ leads to high T_e upstream.

no match for upstream and downstream plasma parameters with constant diffusion coefficients

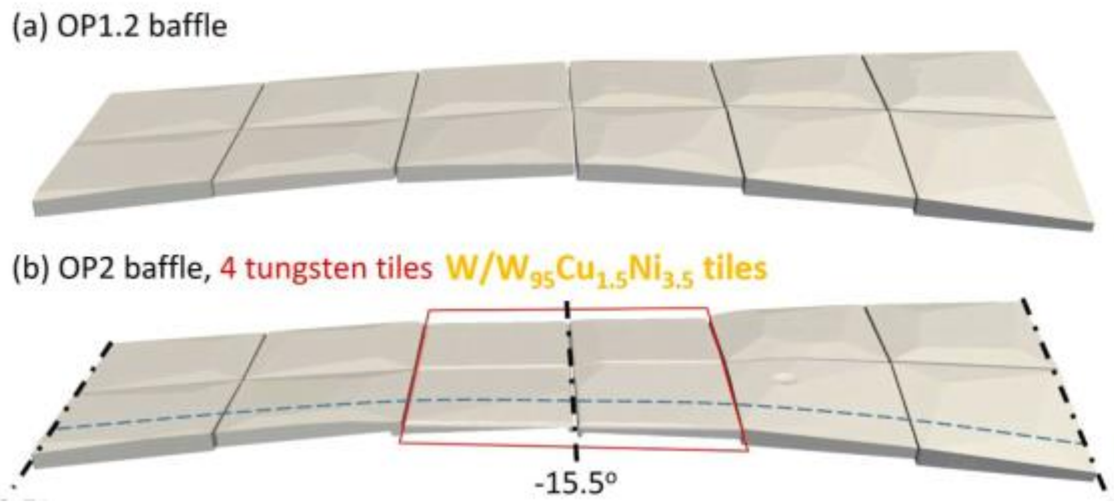


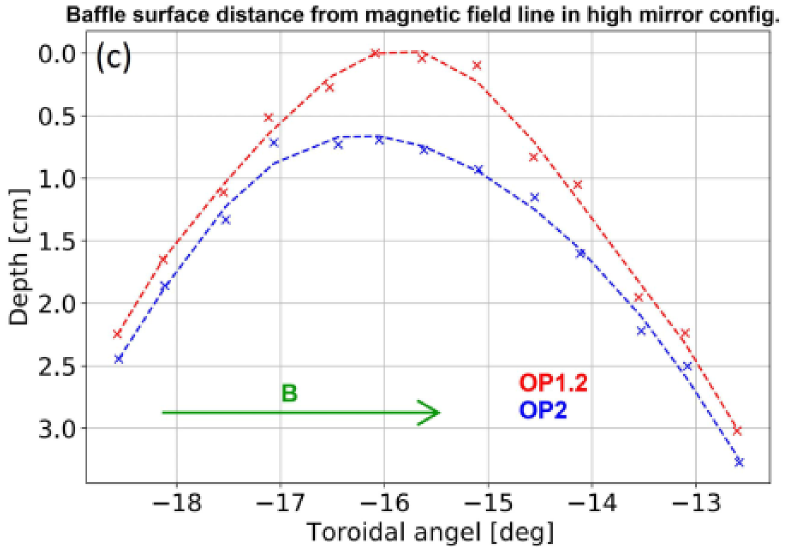
Source: N. Maazis, TG Edge&Divertor, 17.04.2024

more experiments required to fill the data base for detailed comparison studies (P, n, beta, conf. scan)

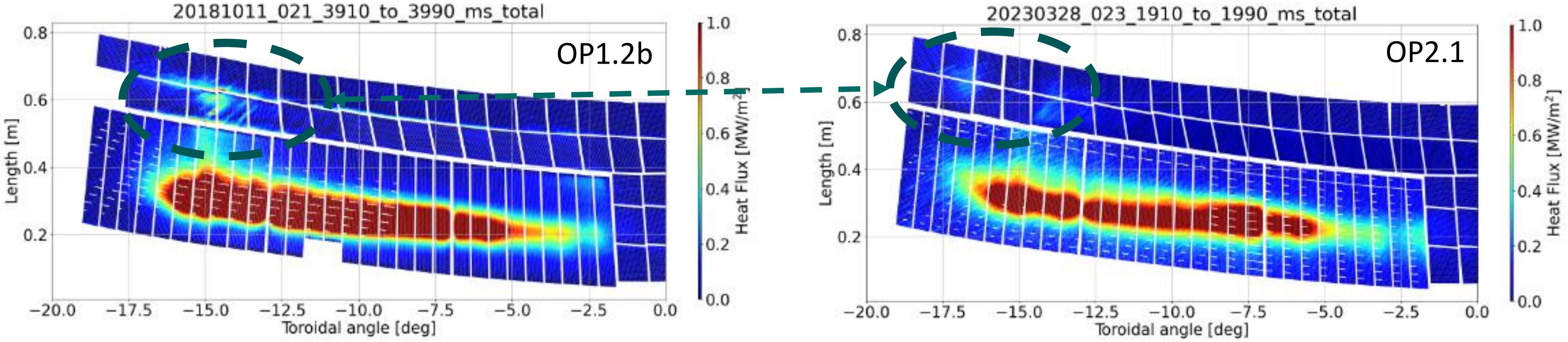
Validation of tools – reshaping of W baffle tiles for OP2.1

reduction of heat loads by a factor 2-3 as predicted by EMC3/Eirene





Source: Gao et al. Nucl. Fusion 63 (2023) 026031



Source: Naujoks et al. NME 37 (2023) 101514

Validation of tools – DIVGAS neutral gas modeling

direct comparison with corresponding experimental and numerical results has been performed using the experimental data as well as the ANSYS numerical data.

	Discharge 20181010.08		Discharge 20180904.31	
	AEH Section	AEP Section	AEH Section	AEP Section
Flow regime	Transition Kn ~O(1)	Free molecular Kn~O(10)	Free molecular Kn~O(10)	Transition Kn~O(1)
Incoming neutral part. flux through the pumping gap Γ_{in} (s^{-1}) [EMC3/Eirene (Y. Feng)]	3.23×10^{20}	2.66×10^{19}	1.15×10^{20}	2.78×10^{20}
Measured pressure P_{exp} (Pa)	0.014	0.0082	0.0024	0.041
Estimated ANSYS pressure P_{ANSYS} (Pa)	Not valid in this flow regime	0.0041	0.0018	Not valid in this flow regime
Estimated DIVGAS pressure P_{DIVGAS} (Pa)	0.0093	0.0029	0.0033	0.031

Source: S. Varoutis et al 2024 Nucl. Fusion 64 076011

this work will be extended to cases with lower Knudsen numbers (i.e higher incoming neutral fluxes), in which the neutral-neutral collisions dominate the flow behavior and aim to a quantitative comparison with corresponding experimental measurements.

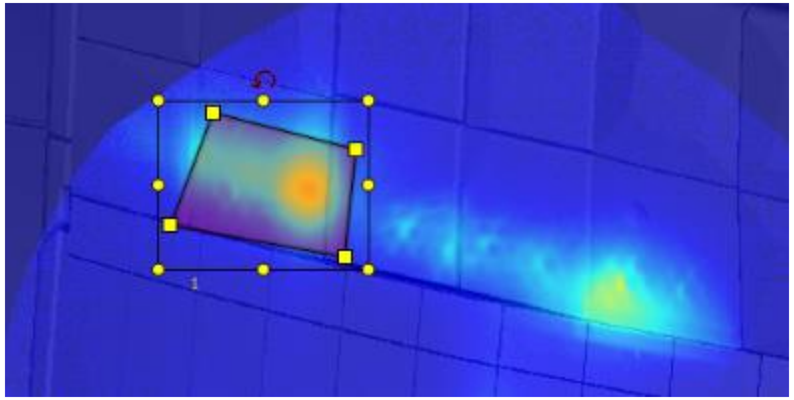
Tungsten PFCs in W7-X: Experiments in OP2.2/2.3

source: IMG_9008

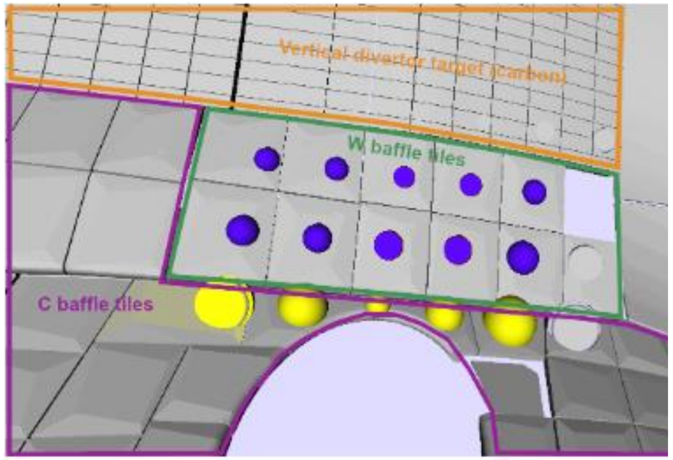
IPP_D_2KLQUC: PV after MP2.2

- definition of the operation limits (erosion/accumulation)
- characterize the scrape-off layer retention for tungsten impurities (eroded from baffle and heat shield)

MMG000+2520 (low-shear): BM1v (AEK31) up to 700 °C



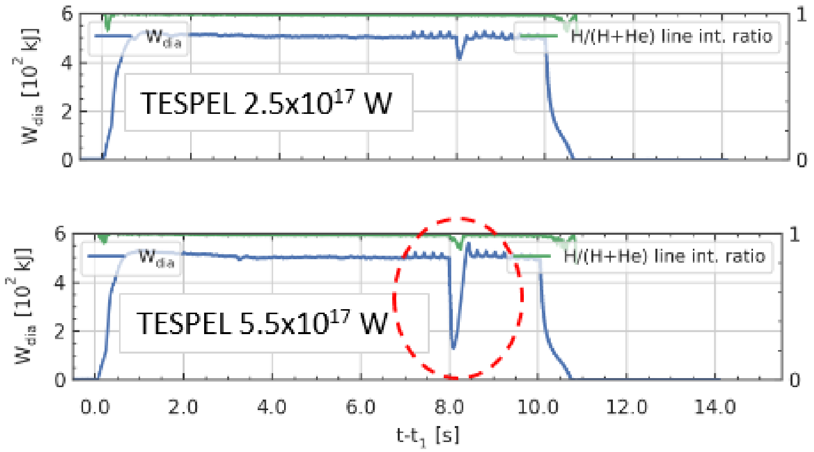
observation by divertor spectroscopy (15 fibers viewing BM1v from AEK41, 2 from AEF30)



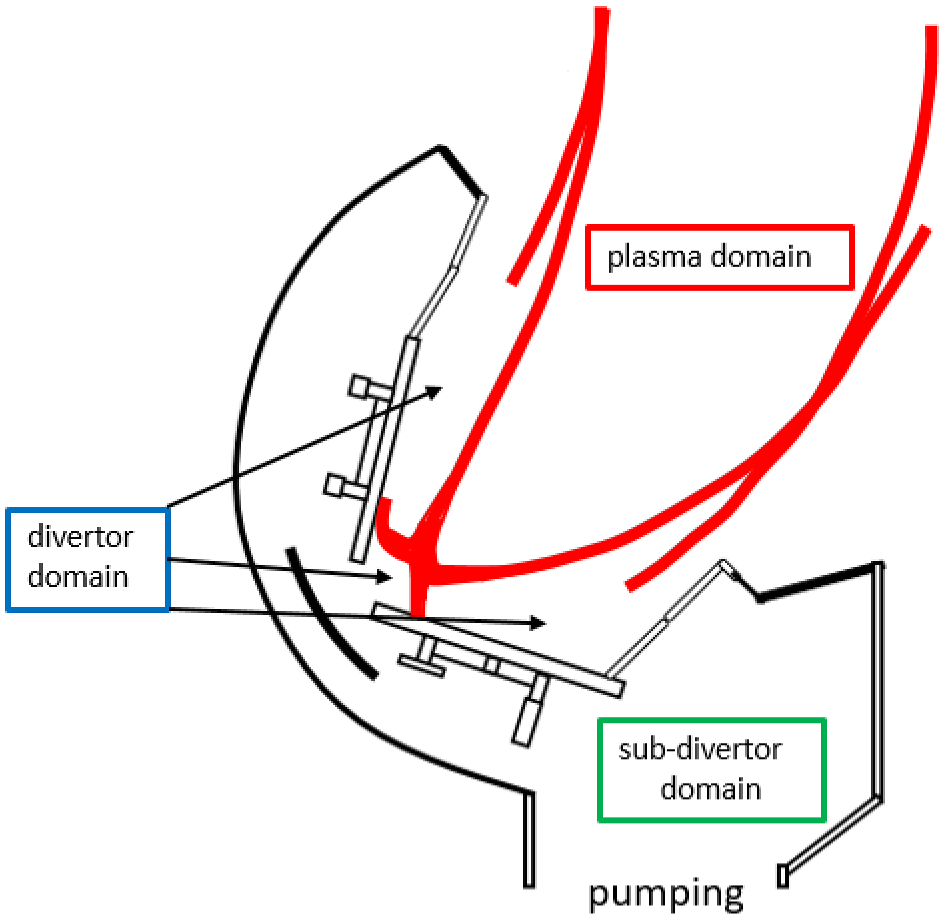
Source: M. Kriete (2024)

and

by core spectroscopy (HEXOS calibration with TESPELs)



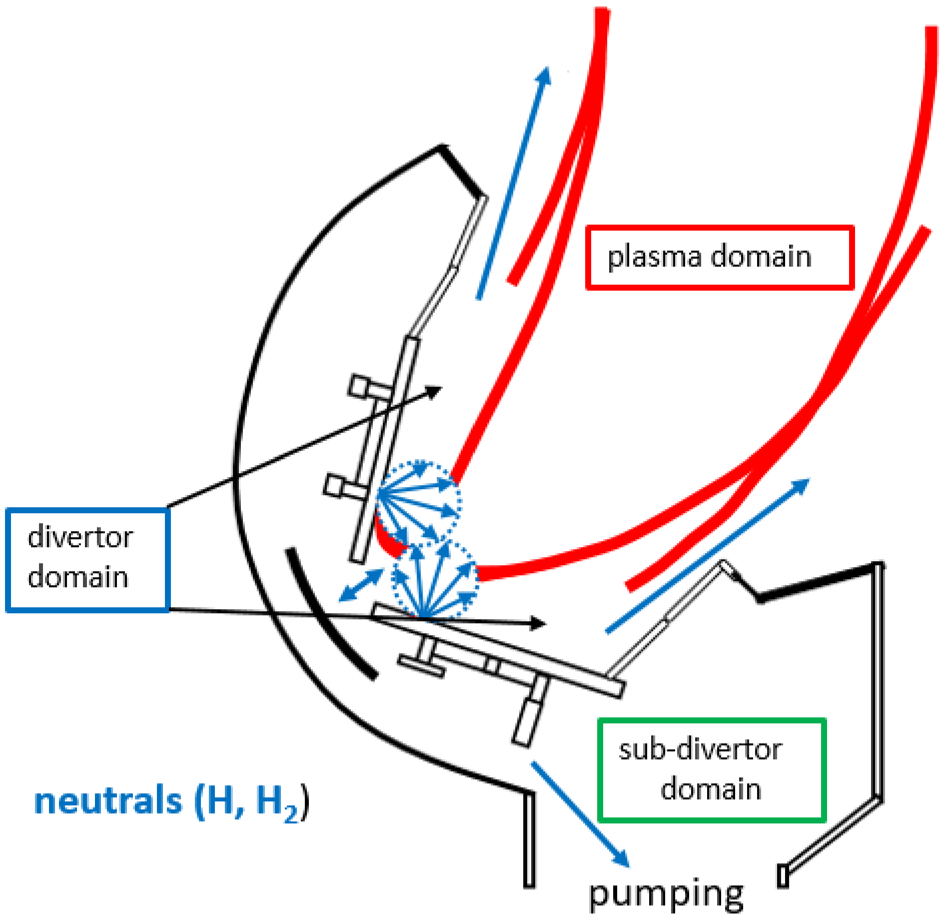
W7-X DCD: sub-divertor pressure & pumping



$$p_{div} = \frac{\Gamma_{pl}\sqrt{\pi\, m_{H_2} k_B T_{H_2}/2}}{A_{leak}^{div} + A_{PG}\, S_{eff}/\left(A_{PG}\sqrt{k_B T_{H_2}/(2\,\pi\, m_{H_2})} + S_{eff}\right)}$$

$$p_{sub} = p_{div}\,\frac{C_{con}}{C_{con} + S_{eff} + A_{leak}^{sub}\sqrt{k_B T_{H_2}/(2\,\pi\, m_{H_2})}}$$

W7-X DCD: sub-divertor pressure & pumping



neutrals (H, H_2)

$$p_{div} = \frac{\Gamma_{pl}\sqrt{\pi\, m_{H_2} k_B T_{H_2}/2}}{A_{leak}^{div} + A_{PG}\, S_{eff}/\left(A_{PG}\sqrt{k_B T_{H_2}/(2\,\pi\, m_{H_2})} + S_{eff}\right)}$$

$$p_{sub} = p_{div}\,\frac{C_{con}}{C_{con} + S_{eff} + A_{leak}^{sub}\sqrt{k_B T_{H_2}/(2\,\pi\, m_{H_2})}}$$

C_{con} >> S_{eff}

$$C_{con} = \tfrac{1}{4} A_{PG}\,\overline{v} = \tfrac{1}{4} A_{PG}\sqrt{8 k_B T_{H_2}/(\pi m_{H_2})}$$

if p_{sub} should be at least $p_{div}/2$, then

$$\frac{S_{eff} + A_{leak}^{sub}\sqrt{k_B T_{H_2}/(2\,\pi\, m_{H_2})}}{C_{con}} = 1$$

$$A_{PG}^{min} = S_{eff}/\sqrt{k_B T_{H_2}/(2\pi m_{H_2})} + A_{leak}^{sub}$$

the exhaust rate $\Gamma_{exhaust}$ is usally two orders of magnitude smaller than the incoming plasma ion flux

pumping gap area A_{PG} requirement

W7-X DCD: Summary

https://event.ipp-hgw.mpg.de/category/63/
https://datashare.mpcdf.mpg.de/s/EPkFnQ5TXRYoNV8
\\share\mp\P224-Wdivertor

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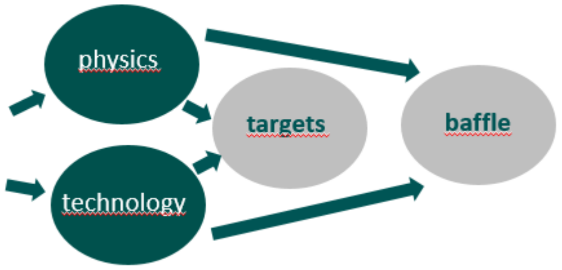
- optimization objectives

1

- optimization criteria
- optimization tools



Physics Concept Development
D. Naujoks et al. Divertor concept development for the W7-X stellarator experiment, 4th Technical IAEA Meeting on Divertor Concepts, 7-11 Nov 2022, IAEA Headquarters, Vienna
-> https://conferences.iaea.org/event/286/contributions/25129/

Target Element Development:
J. Fellinger et al. Tungsten based divertor development for Wendelstein 7-X, contribution to the PFMC-19 (2023), paper is under submission to the Journal of Nuclear Materials and Energy (2023)
-> https://users.euro-fusion.org/repository/pinboard/EFDA-JET/conference/106333_paperfellinger.v3.docx

Experiments with Tungsten in W7-X
D. Naujoks et al. Performance of tungsten plasma facing components in the stellarator experi-ment W7-X: recent results from the first OP2 campaign, contribution to the PFMC-19 (2023), paper is under submission to the Journal of Nuclear Materials and Energy (2023)
-> https://users.euro-fusion.org/repository/pinboard/EFDA-JET/conference/106863_naujoks_pfmc_2023_tungsten_pfcs_in_w7-x.pdf