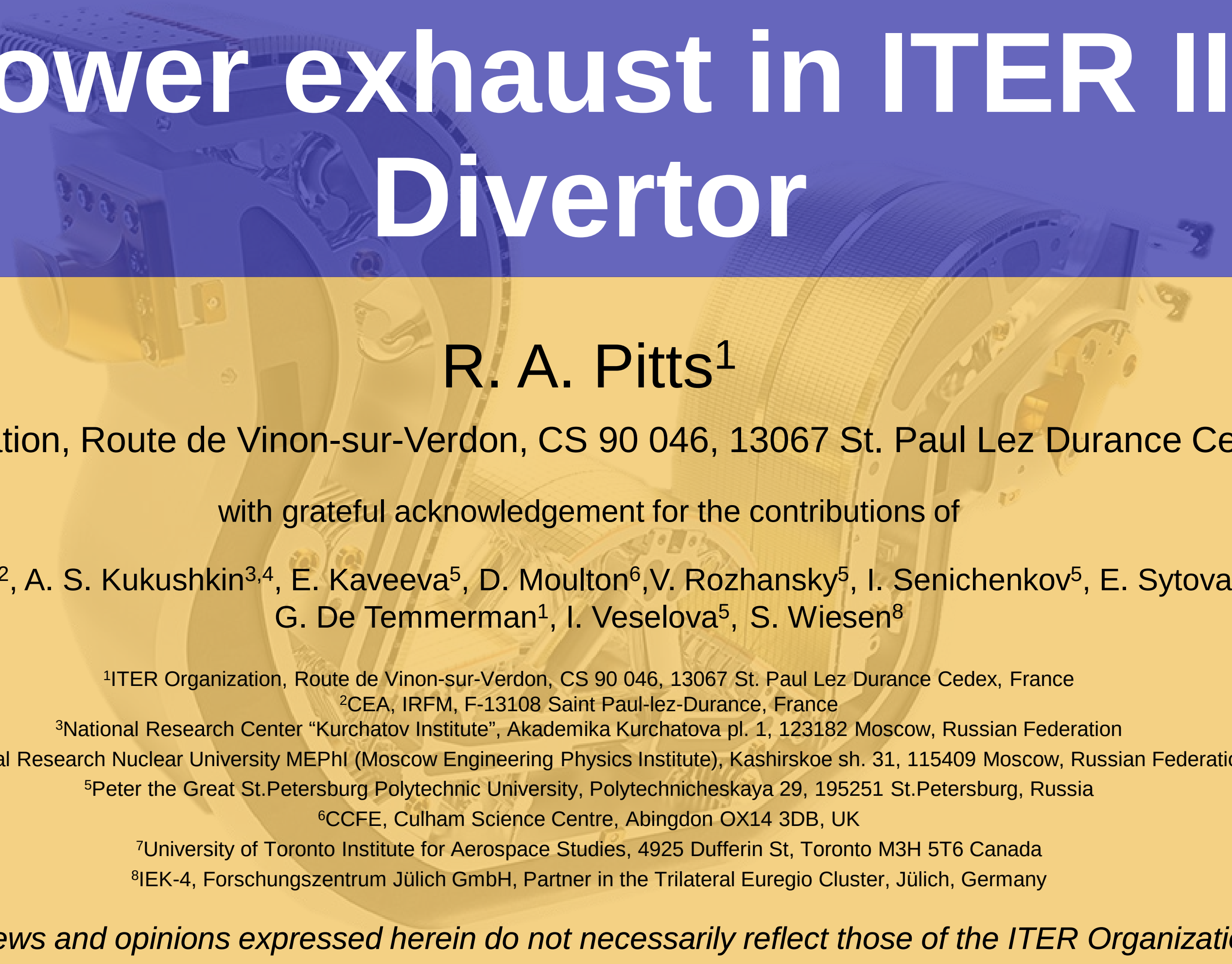
Power exhaust in ITER II: Divertor



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with grateful acknowledgement for the contributions of

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Content

- Introduction to the ITER W divertor
  - Basic physics/design features and expected lifetime
- Stationary power loading – the design simulation database
  - Overall characteristics
  - Focus on factors influencing the peak power loading
  - What really is the tolerable steady state power flux density?
  - What are the tolerable ELM loads?

- I will speak only about axisymmetric divertor heat loads → see lecture by O. Schmitz for the case of 3D fields for ELM control

But first, a brief return to lecture 1

- Apply 0D power balance to estimate peak $q_{||}$ at the divertor targets, e.g. for outer target:

$$q_{||out} = P_{div,out} / (2\pi R_{out} \lambda_q (B_\theta / B_\phi)_{omp}) \qquad P_{div,out} = P_{SOL}(1\text{-}f_{RAD}) A_{sym} / (1 + A_{sym}))$$

f_{RAD} = radiated fraction of power conducted to the divertor

R_{out} = major radius of outer strike point
$P_{div,out}$, $P_{div,in}$ = powers into outer/inner divertor, $A_{sym} = P_{div,out}/P_{div,in}$

- Example: λ_q = 5 mm, P_{SOL} ~ 100 MW, A_{sym} = 2
  "detached divertor", f_{RAD} ~ 60% → $q_{||out}$ ~ 300 MWm^{-2}
  "low recycling divertor", f_{RAD} ~ 20%, $q_{||out}$ ~ 900 MWm^{-2}
- Take α ~4° (component shaping → see later) → $q_{\perp,out} = q_{||,out} \sin\alpha$
  $q_{\perp,out}$ ~21 MWm^{-2} ("detached"), $q_{\perp,out}$ ~63 MWm^{-2} ("attached")!!
- Cannot be handled by technology on ITER

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NB: no losses associated with diffusion into PFR included → see lecture by H. Zohm

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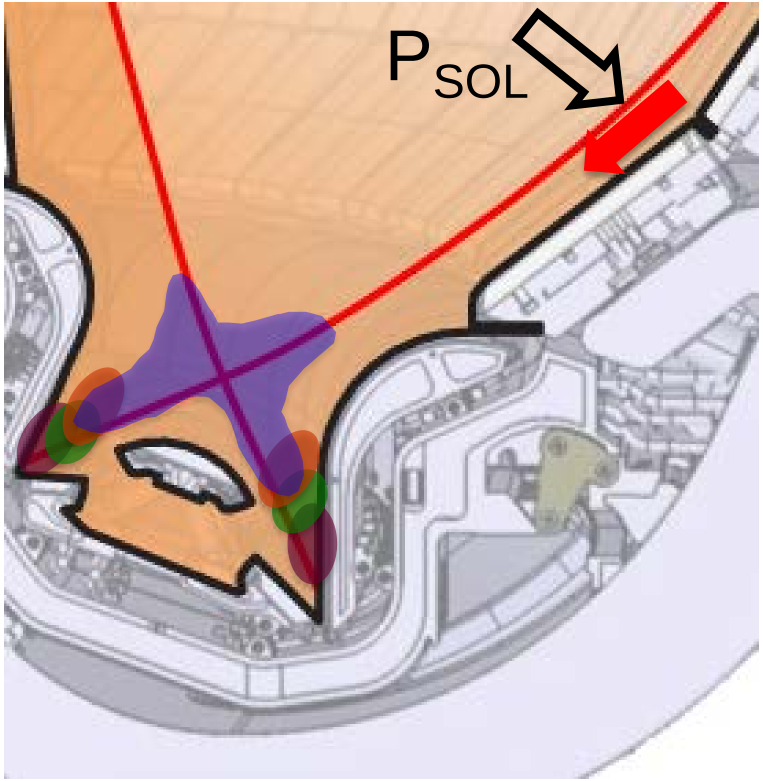
Now imagine that λ_q drops by factor 5 → $q_{\perp,out}$ increases by same factor …

- Cannot be handled by technology on ITER

Problem is that simple specification is too simple

- ITER Divertor is highly dissipative



Heat conduction zone

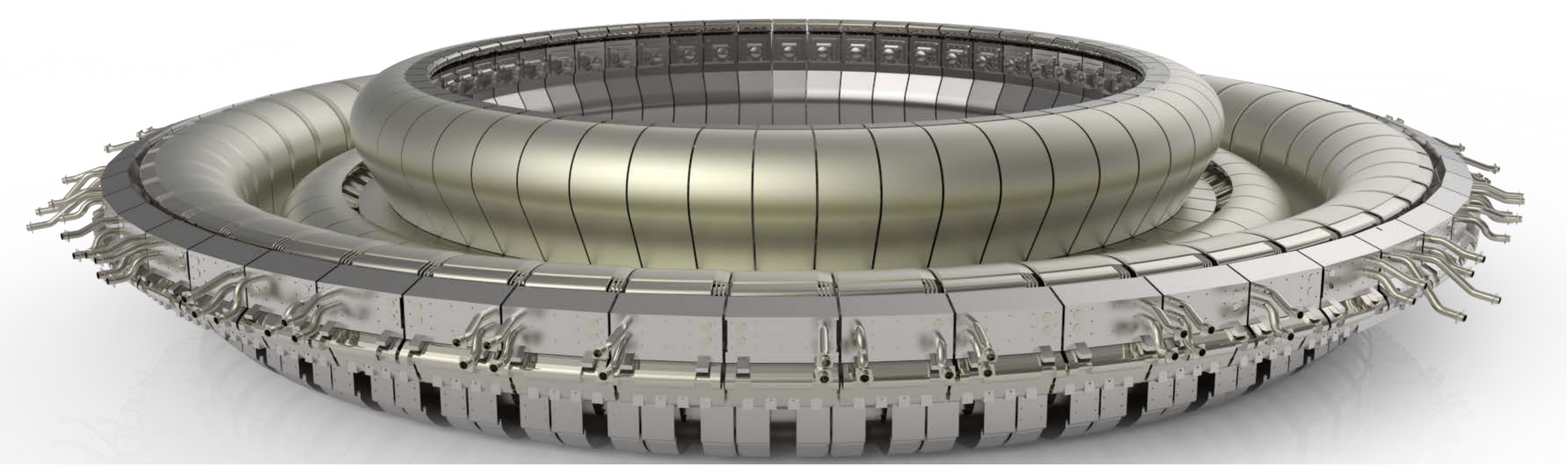
Impurity radiation zone

**$H^0/D^0/T^0$ ionization zone
(T_e > 5 eV)**

Neutral friction zone

**Recombination zone
(T_e < 1 eV)**

The ITER tungsten divertor

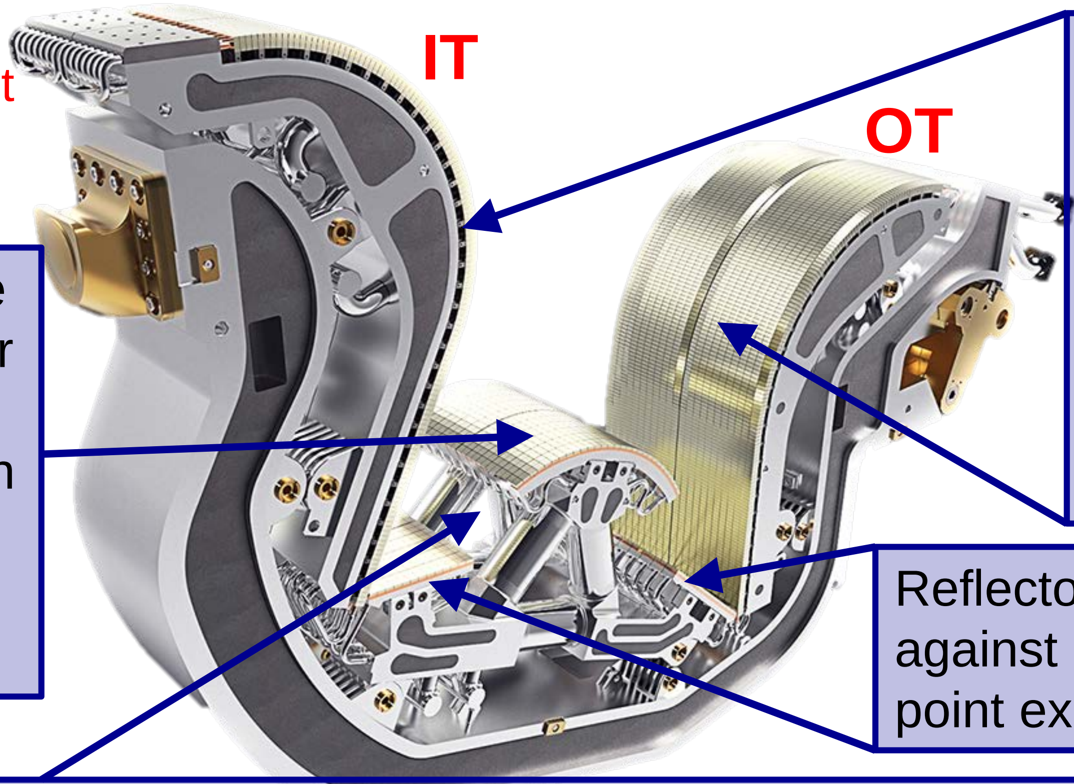


- The most sophisticated tokamak divertor ever built
  - 54 individual cassettes, fully water cooled, designed to handle up to ~100 MW in steady state
  - Now entering the procurement phase → design essentially complete

W divertor: key physics characteristics

IT = INNER target

OT = OUTER target



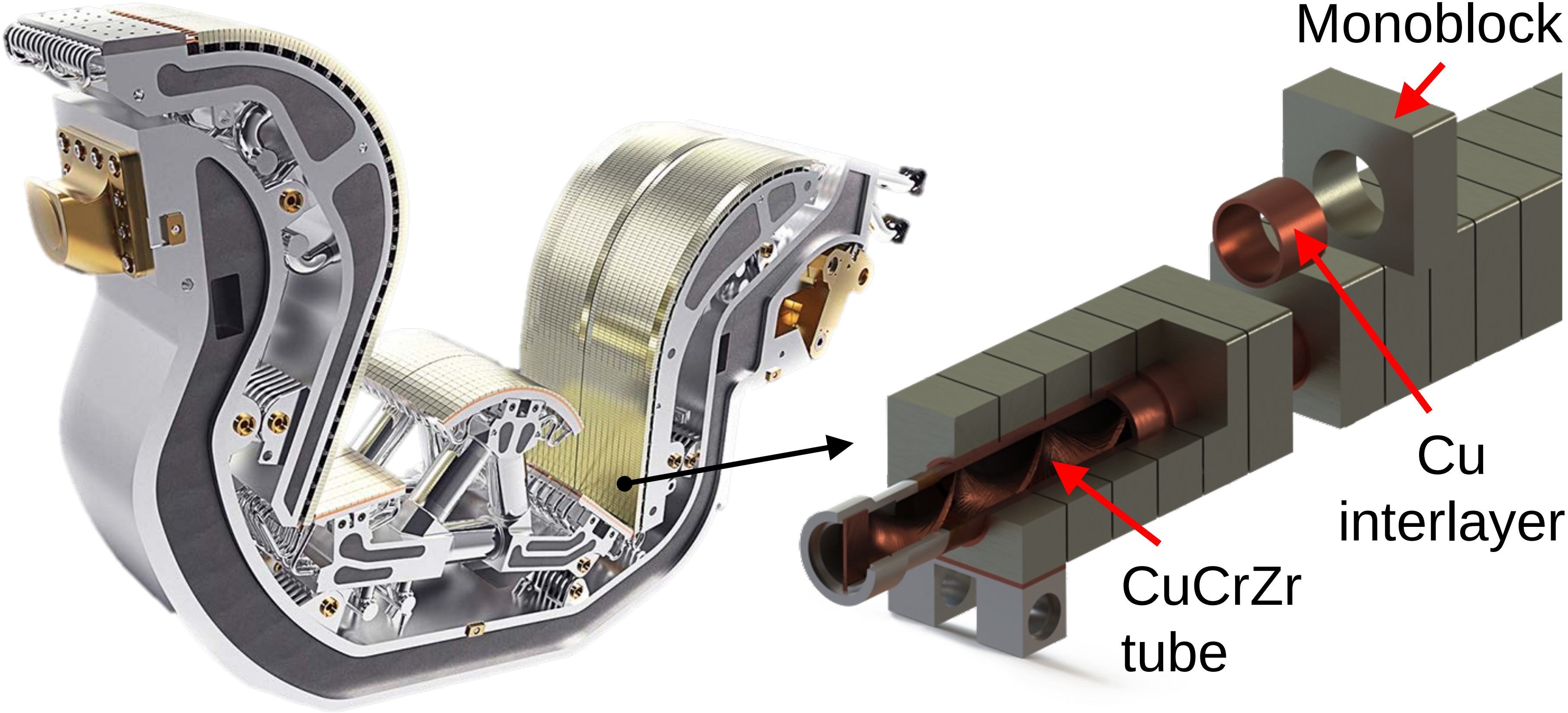
Dome to improve pumping → lower pumping speed required for given upstream He conc or fuel throughput

Deep vertical targets and baffle regions promoting detachment and reducing neutral escape to the core

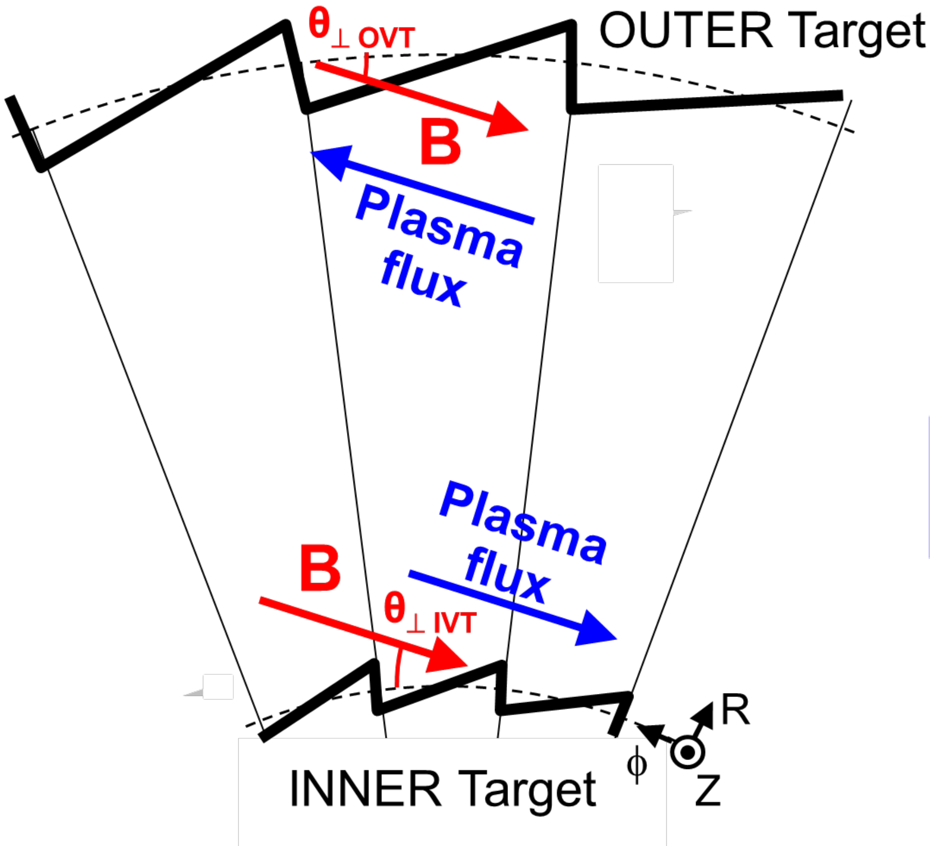
Reflector plates protect against downward strike point excursions

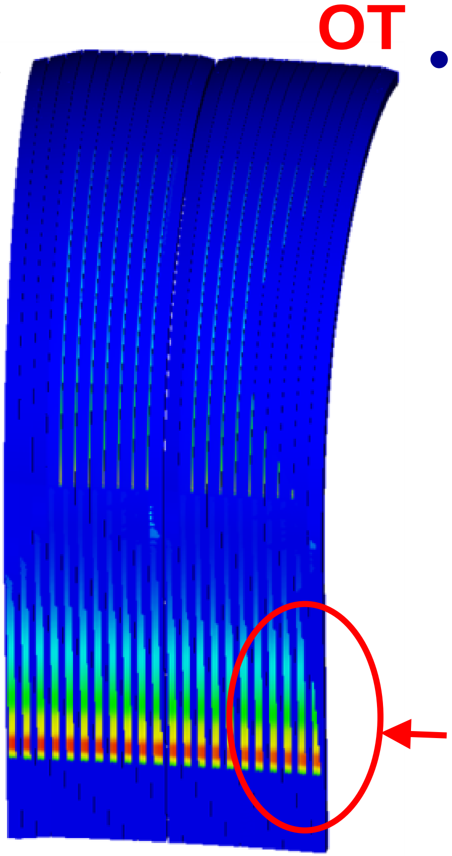
Transparency between targets for neutral recirculation – lower power asymmetries

Vertical target plasma-facing units



Global shaping: target tilting

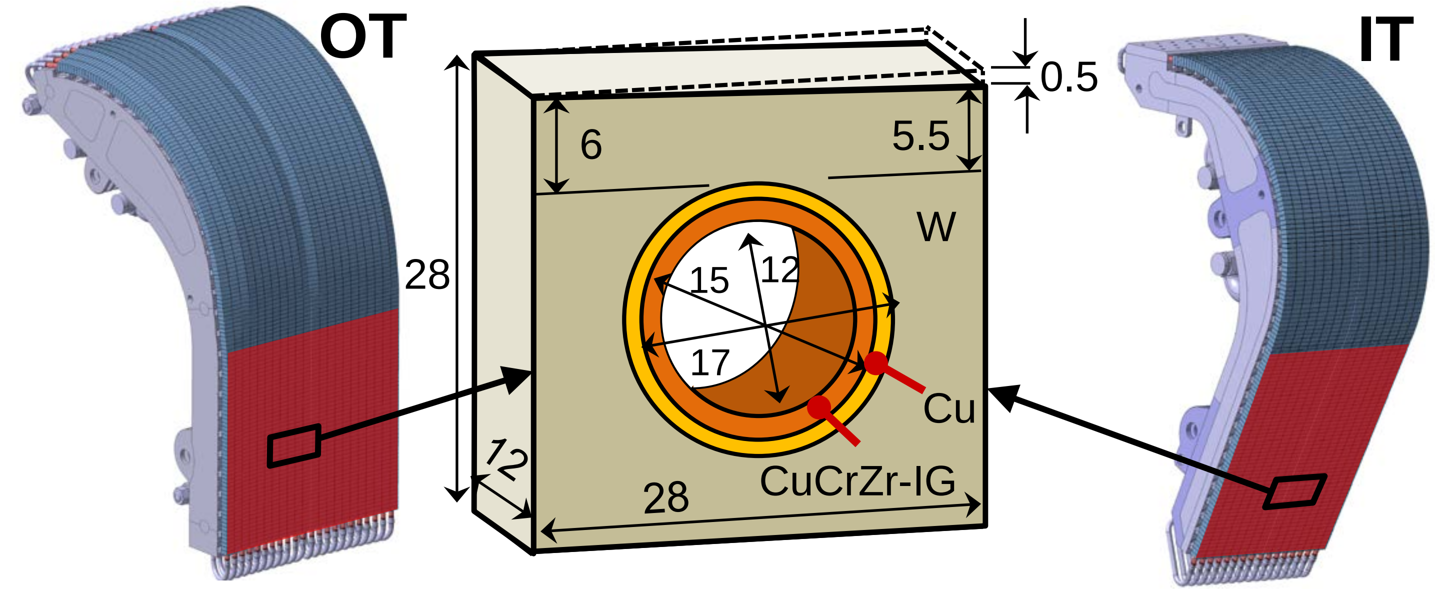




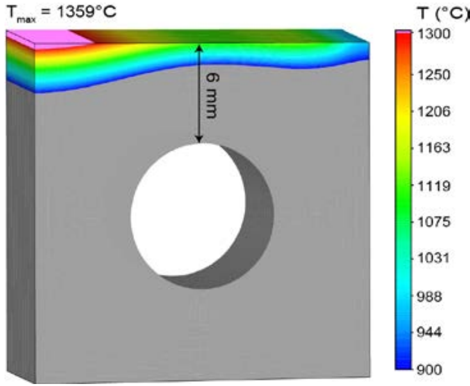
- Tilt of ~0.5° at both inner and outer targets to protect gross leading edges between adjacent cassettes

First plasma-facing unit fully magnetically shadowed

Local shaping: monoblock top surface

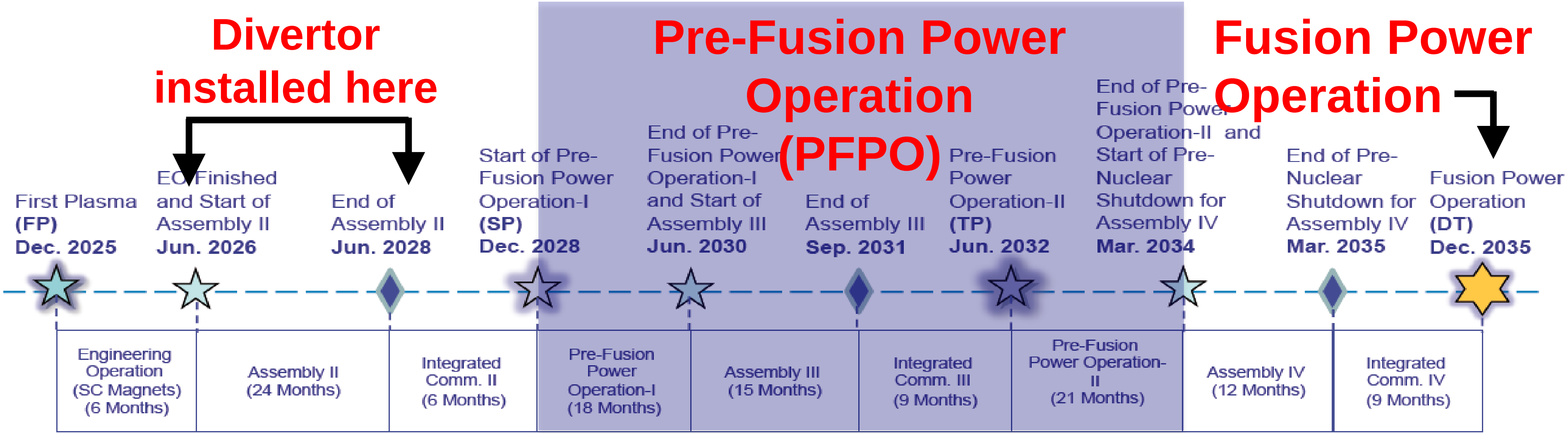


R. A. Pitts et al., NME **12** (2017) 60



- Monoblocks in high heat flux areas toroidally bevelled to hide radial misalignmensts between toroidal neighbours → compromise between poloidal gap edge overheating and increased surface stationary loading → See talk by J. P. Gunn

Revised ITER schedule and divertor lifetime



Divertor installed here

Pre-Fusion Power Operation (PFPO)

Fusion Power Operation

First Plasma (FP) **Dec. 2025**

EO Finished and Start of Assembly II **Jun. 2026**

End of Assembly II **Jun. 2028**

Start of Pre-Fusion Power Operation-I (SP) **Dec. 2028**

End of Pre-Fusion Power Operation-I and Start of Assembly III **Jun. 2030**

End of Assembly III **Sep. 2031**

Pre-Fusion Power Operation-II (TP) **Jun. 2032**

End of Pre-Fusion Power Operation-II and Start of Pre-Nuclear Shutdown for Assembly IV **Mar. 2034**

End of Pre-Nuclear Shutdown for Assembly IV **Mar. 2035**

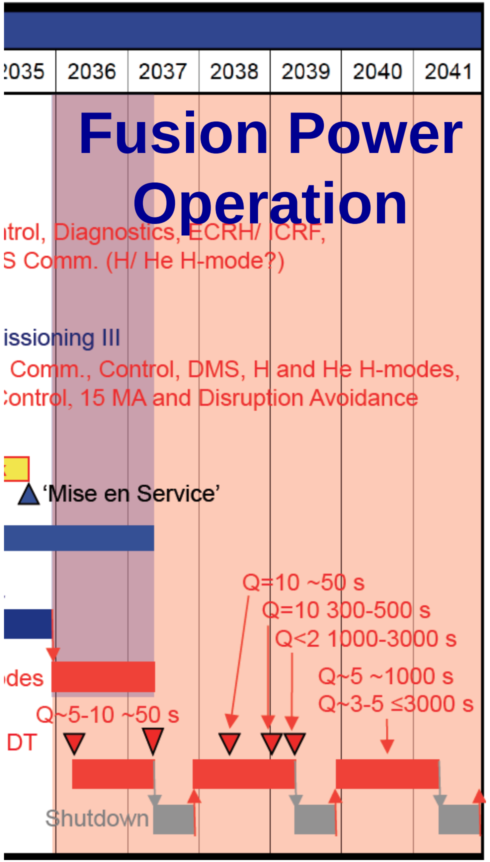
Fusion Power Operation (DT) **Dec. 2035**

Engineering Operation (SC Magnets) (6 Months)	Assembly II (24 Months)	Integrated Comm. II (6 Months)	Pre-Fusion Power Operation-I (18 Months)	Assembly III (15 Months)	Integrated Comm. III (9 Months)	Pre-Fusion Power Operation-II (21 Months)	Assembly IV (12 Months)	Integrated Comm. IV (9 Months)

- H and He operation in PFPO phases 1 and 2, L- and H-mode → detailed operational breakdown in new ITER Research Plan*:
  - Total days in PFPO-1: 470 → ~5700 pulses → **~3x10^5 s** up to P_{heat} = 30 MW
  - Total days in PFPO-2: 545 → ~5600 pulses → **~6x10^5 s** up to P_{heat} = 73 MW

* https://www.iter.org/technical-reports

Revised ITER schedule and divertor lifetime



- First DT campaigns roughly split into 3 phases (FPO-1,2,3):
  - Expect power into SOL to reach design value ~100 MW
  - Expect ~900 days operation over ~5 years
  - ~12,000 pulses
  - **~8x10^6 s plasma time**
    (~2200 hours or ~90 days)

First ITER W divertor required to survive until end of FPO-3

Divertor operation: design by simulation

- Physics operating mode for the ITER divertor is to a large extent based on plasma boundary simulations conducted over ~15 years with the SOLPS-4.3 code (B2-Eirene) → mostly C targets!
  - Well documented

A. S. Kukushkin et al. J. Nucl. Mat. **290-293** (2001) 887
A. S. Kukushkin et al. Nucl. Fusion **42** (2002) 187
A. S. Kukushkin and H. D. Pacher, PPCF **44** (2002) 931
H. D. Pacher et al. J. Nucl. Mat. **313-316** (2003) 657
A. S. Kukushkin et al. Nucl. Fusion **43** (2003) 716
A. S. Kukushkin et al. Fus. Eng. Design **65** (2003) 355
A. S. Kukushkin et al. Nucl. Fusion **45** (2005) 608
A. S. Kukushkin et al. J. Nucl. Mat. **337-339** (2005) 17
A. S. Kukushkin et al. Nucl. Fusion **47** (2007) 698
A. S. Kukushkin et al. J. Nucl. Mat. **363-365** (2007) 308
G. W. Pacher et al. Nucl. Fusion **48** (2008) 105003
H. D. Pacher et al. J. Nucl. Mat. **390-391** (2009) 259
A. S. Kukushkin et al. Nucl. Fusion **49** (2009) 075008
A. S. Kukushkin et al., J. Nucl. Mat. **415** (2011) 2011
H. D. Pacher et al. J. Nucl. Mat. **415** (2011) S492
G. W. Pacher et al. Nucl. Fusion **51** (2011) 083004
A. S. Kukushkin et al. Fus. Eng. Design **86** (2011) 2865

A. S. Kukushkin et al. Nucl. Fusion **53** (2013) 123024
H. D. Pacher et al. J. Nucl. Mat. **463** (2015) 591
A. S. Kukushkin et al., Nucl. Fusion **56** (2016) 126012

First real operating domain study for metal walls → will be a focus of this talk

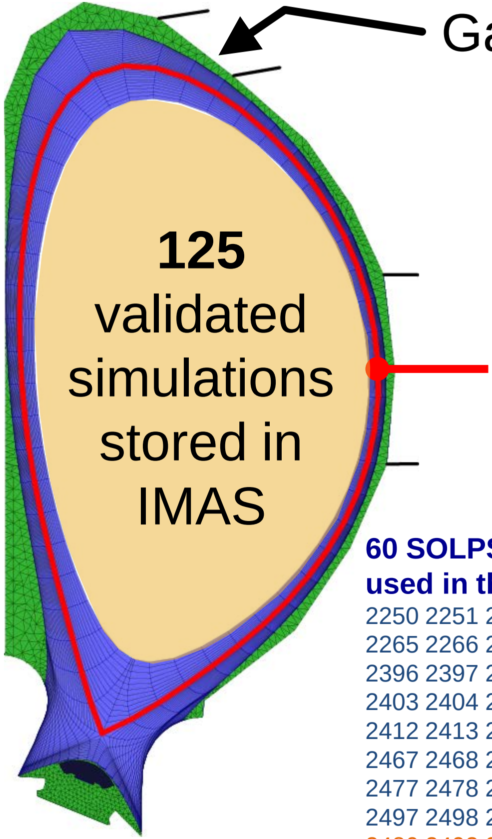
Since 2015, moved to new code version (incl. drifts)

SOLPS-ITER

S. Wiesen et al, . J. Nucl. Mat. **463** (2015) 480
X. Bonnin et al., Plasma & Fusion Research, **11** (2016) 1403102

Main simulation database parameters

- Steady state – no ELMs
- No fluid drifts, “L-mode” edge
  - Neutral-neutral collisions included
- Fixed equilibrium
  - q_{95} = 3, B_T/I_p =1.8/5, 2.65/7.5, 5.3/15
- Fixed cross-field transport
  - $D_\perp = 0.3\ m^2 s^{-1}$, $\chi_\perp = 1.0\ m^2 s^{-1}$
- Scans in fueling, seed impurity, power into numerical grid (P_{IN})
  - H, He, D, N_2, Ne, but only P_{IN} = 100 MW in this talk
- All-metal walls
  - Assume Be everywhere, but no sputtering



Gas puff

125 validated simulations stored in IMAS

n_Z/n_e averaged around separatrix

60 SOLPS-4.3 case numbers used in this talk:

2250 2251 2252 2253 2257 2258 2264 2265 2266 2269 2316 2317 2332 2333 2396 2397 2398 2399 2400 2401 2402 2403 2404 2407 2408 2409 2410 2411 2412 2413 2414 2415 2416 2439 2463 2467 2468 2469 2470 2471 2472 2476 2477 2478 2481 2483 2484 2485 2496 2497 2498 2508 2509 (Ne)

2480 2493 2494 2502 2503 2533 2534 (N)

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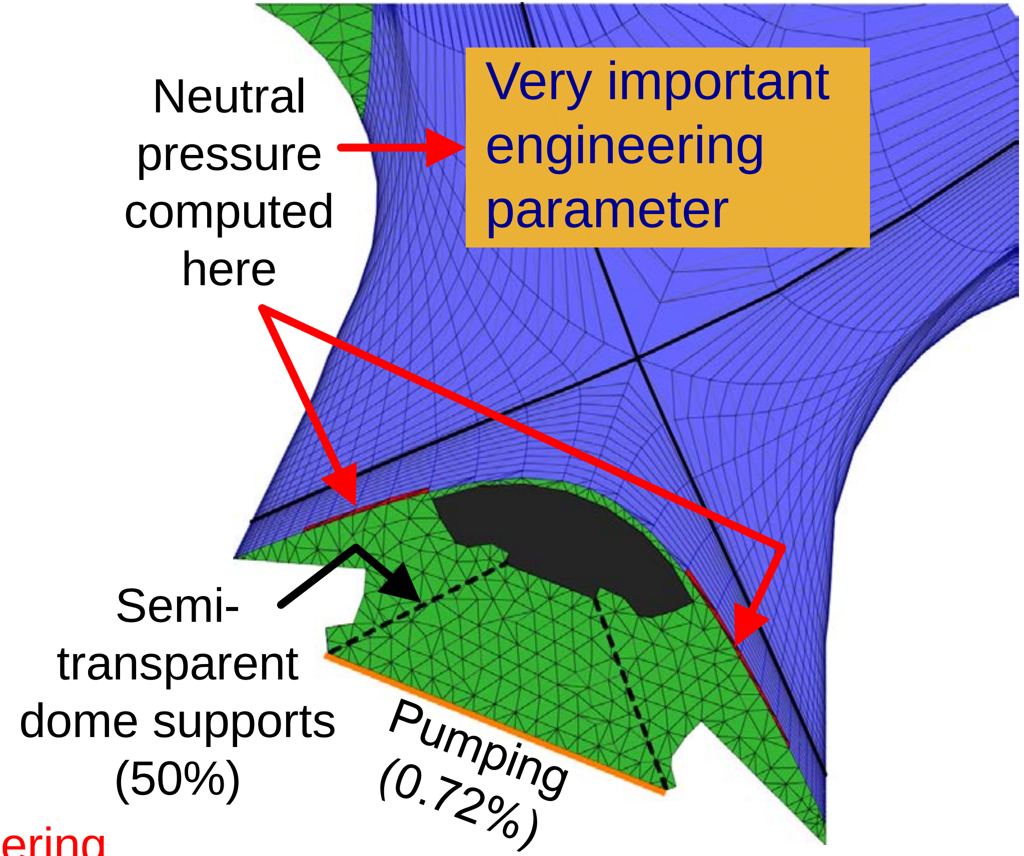

Neutral pressure computed here

Very important engineering parameter

Semi-transparent dome supports (50%)

Pumping (0.72%)

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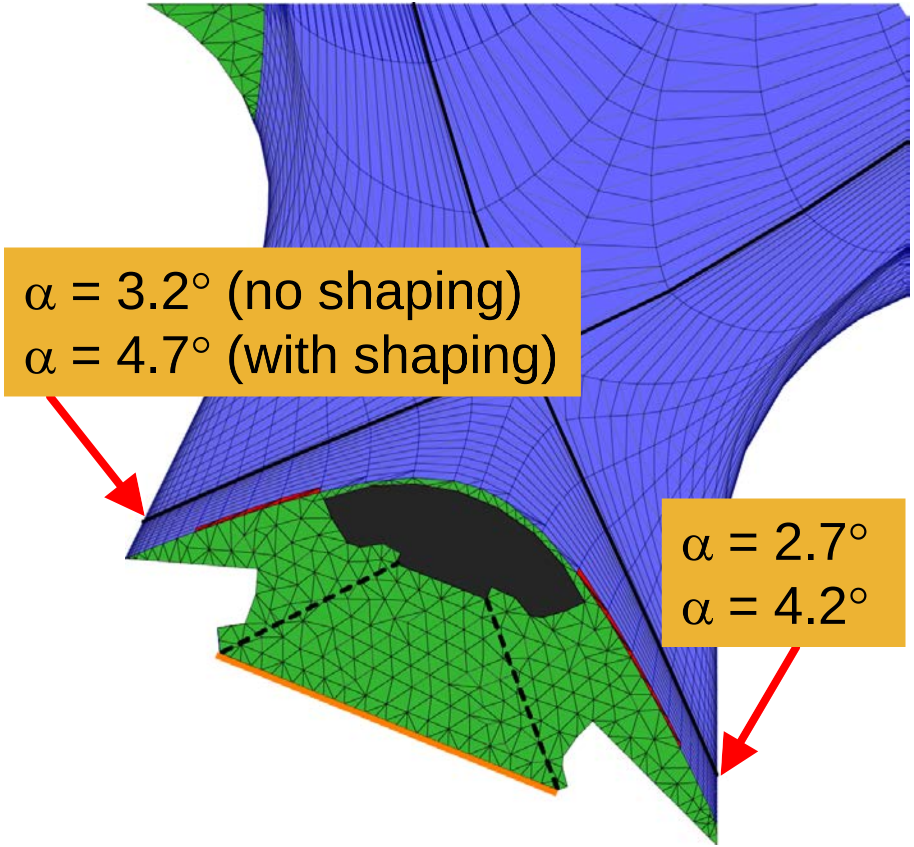


α = 3.2° (no shaping)
α = 4.7° (with shaping)

α = 2.7°
α = 4.2°

Burning plasma operating window

- Focus on "burning plasma" conditions → the most challenging for the ITER divertor
    - $Q_{DT} = 10$, $P_{IN} \sim 100$ MW
    - Ne and N_2 seeding (emphasis on Ne where database currently largest)
    - No discussion of "integrated modelling" here

- An important fact to bear in mind: ITER will operate always quite close to the H-mode power transition threshold
    - Cannot afford (too) much edge/core radiation

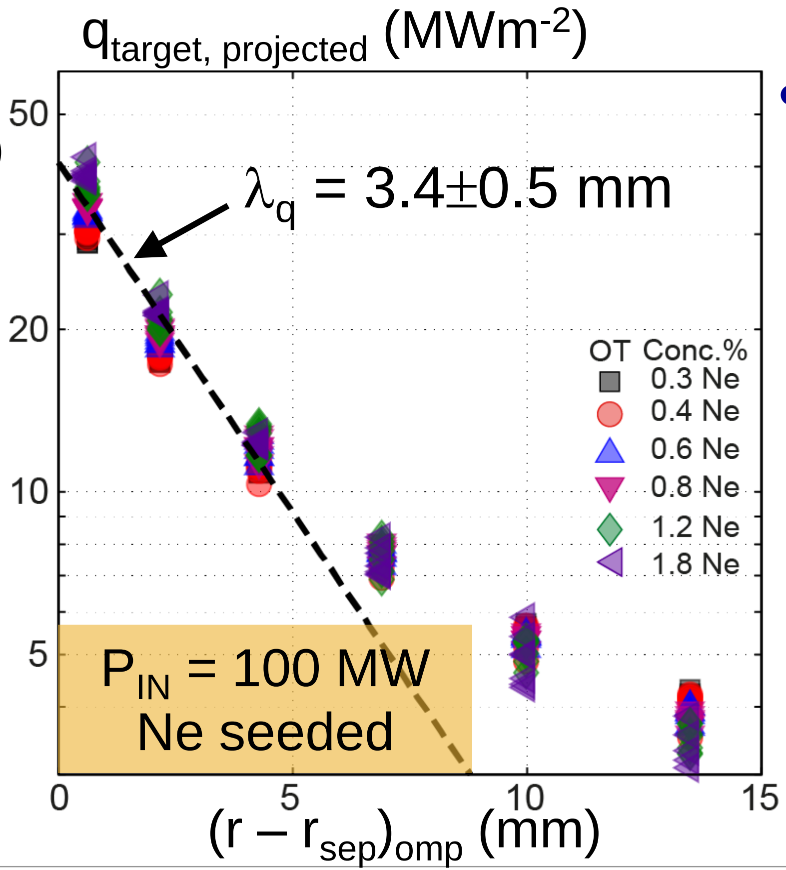
SOL heat flux width

$q_{||}(r)$ outer midplane

③

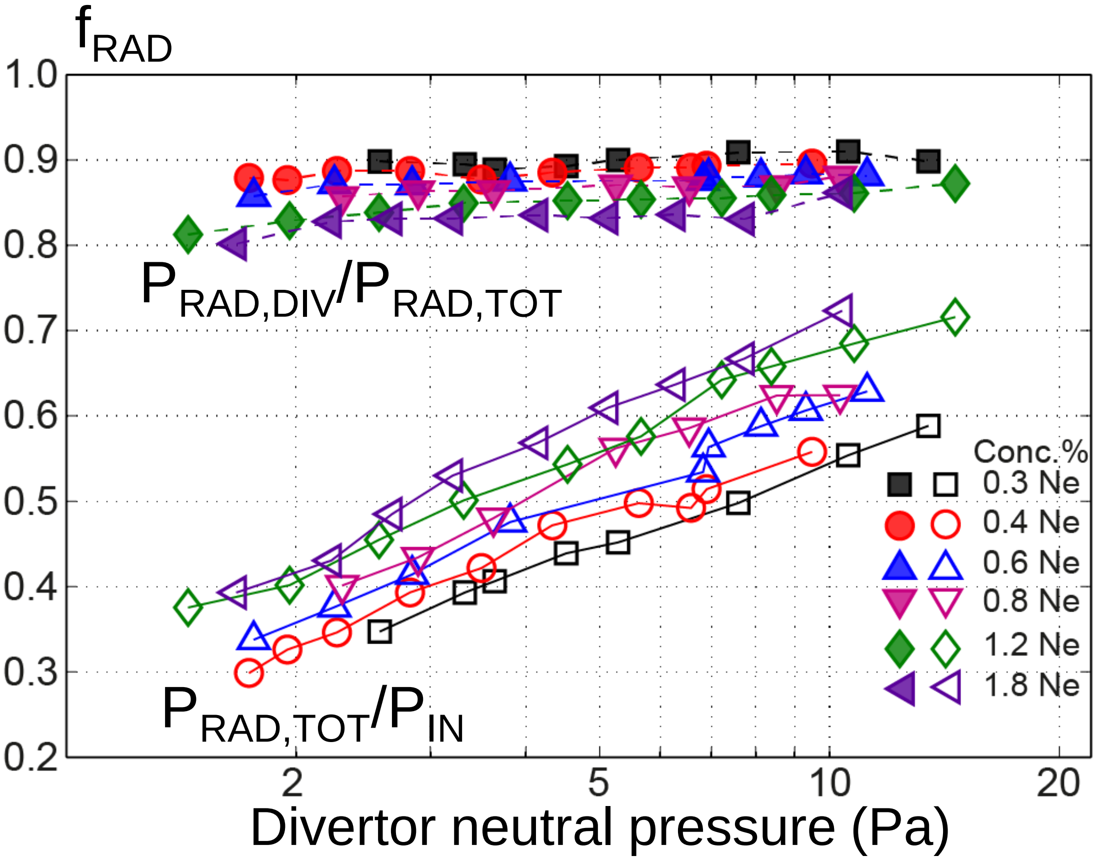
① $q_{||}(r)$ X-pt.

② $q_{\perp}(r)$ OT



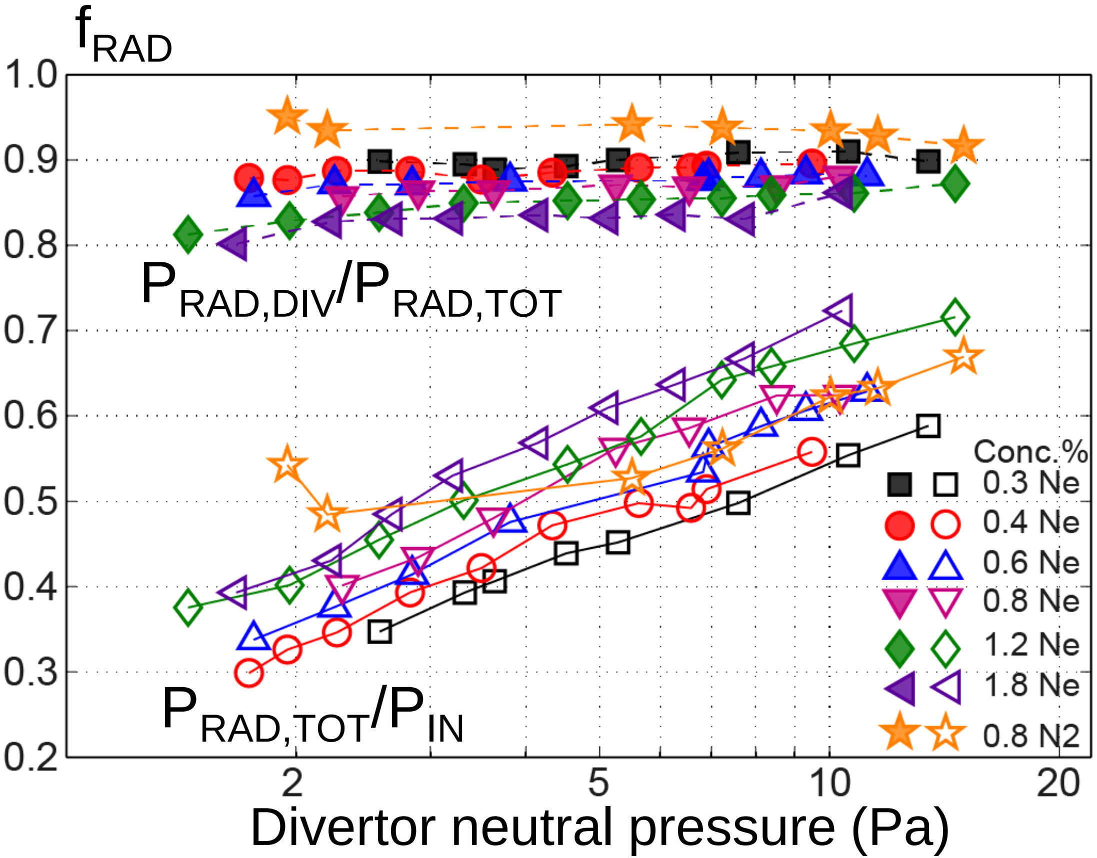
- Divertor conditions across database do not strongly influence upstream λ_q

Radiated fractions



- Radiation largely confined to the divertor region
  - $f_{RAD,DIV}$ ~ 0.8-0.9 across operating window for Ne
  - $f_{RAD,TOT}$ ~ 0.3 – 0.7

Radiated fractions



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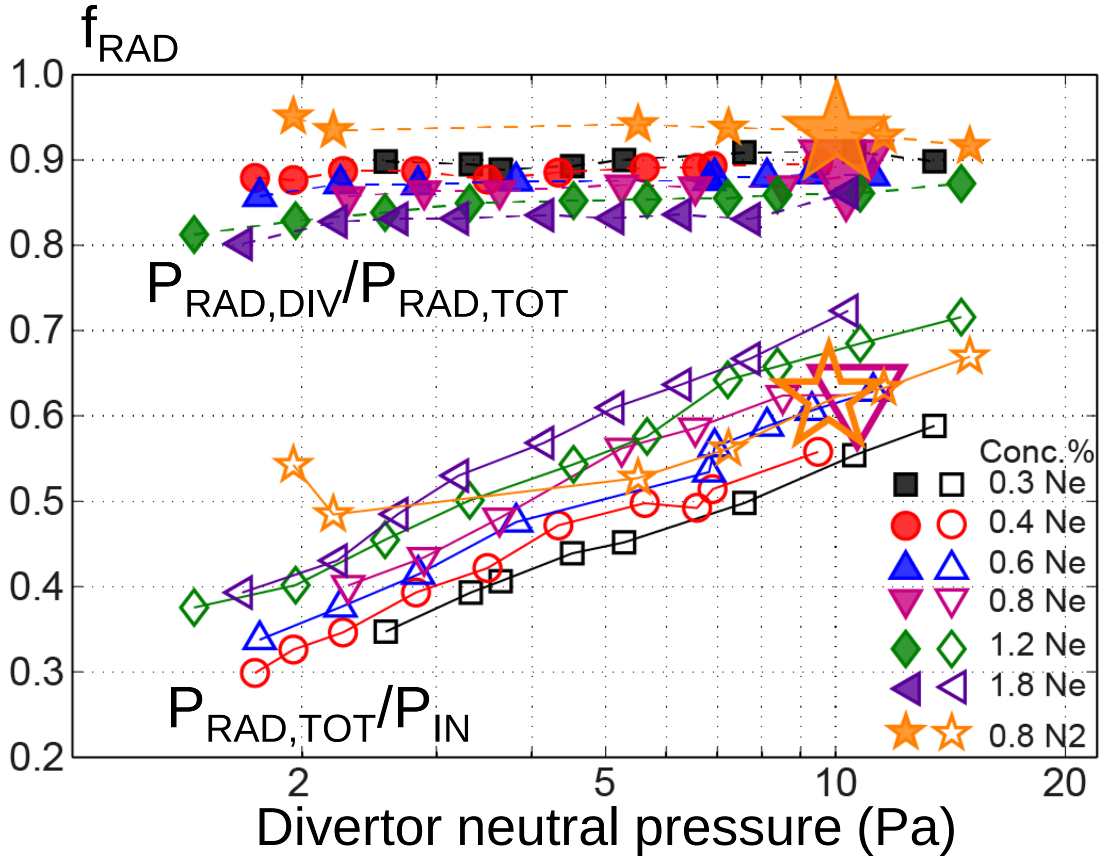
  - N more efficiently compressed than Ne
  - Lower core radiation with N

Radiated fractions



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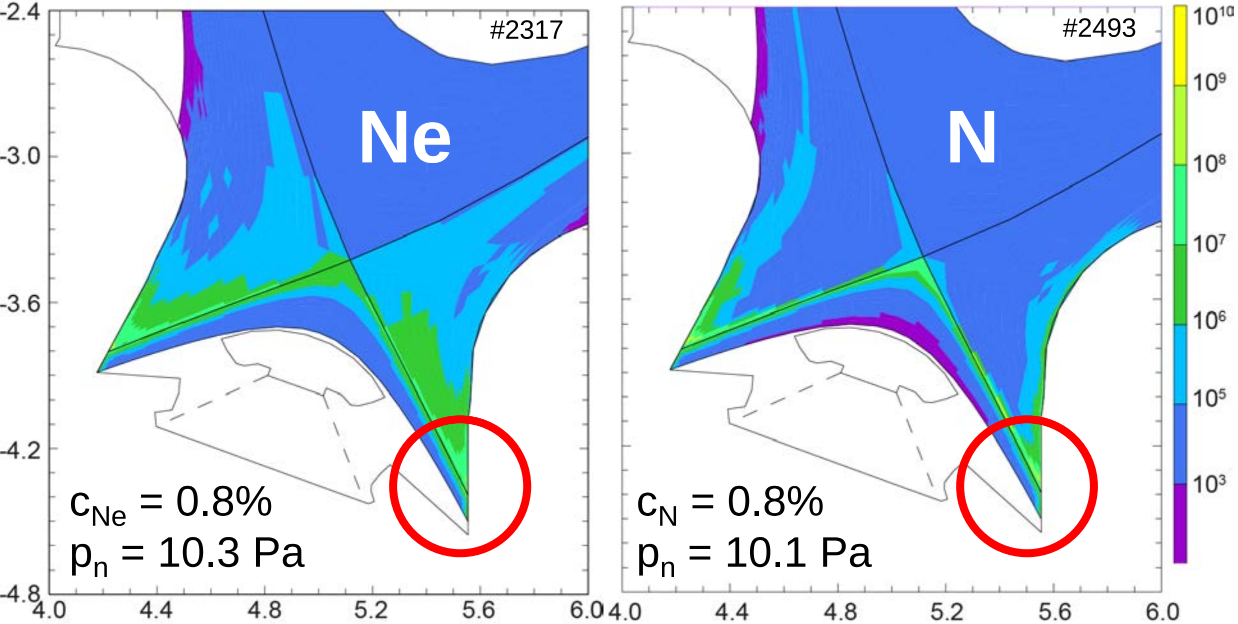
Divertor radiation distribution

Total radiated power (Wm^{-3})



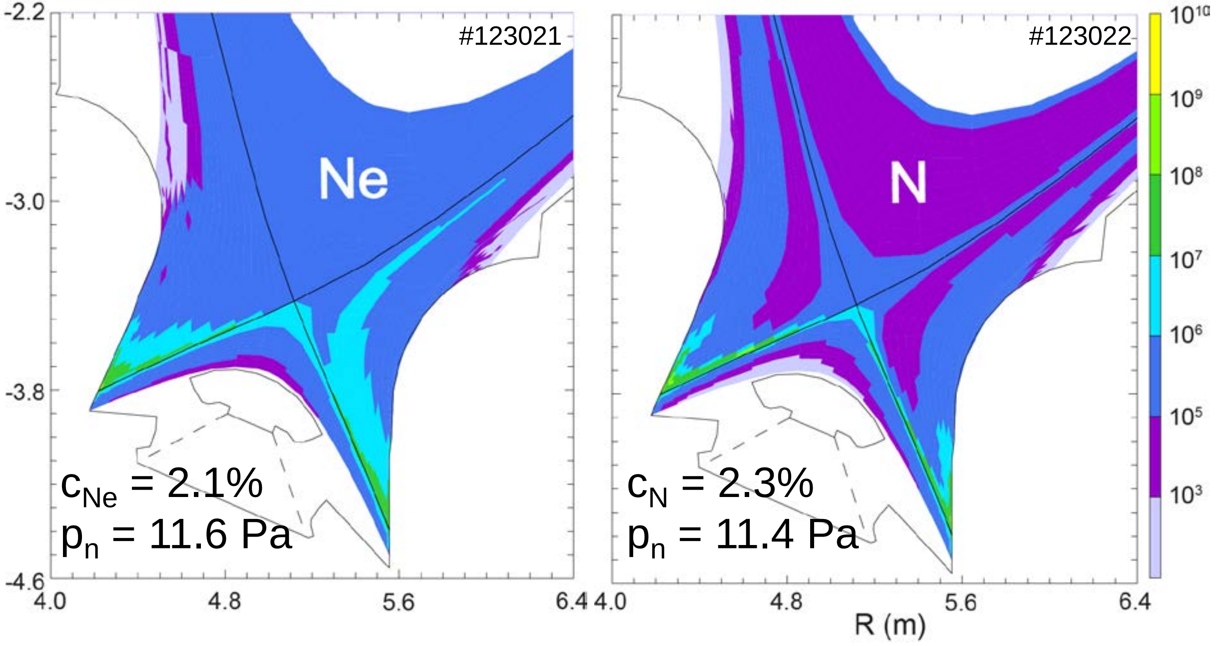
$P_{RAD,DIV}$ = 56.6 MW
41.3 (Ne) + 15.3 (D)

$P_{RAD,DIV}$ = 54.0 MW
38.6 (N) + 15.4 (D)

- Ne radiation more extended than N
  - Expected from differences in ionization potential
  - But still mostly confined to divertor volume
- Compression: $n_{Z,osp}/n_{Z,omp}$ ~100 (N), ~30 (Ne)

Divertor radiation distribution: with drifts

Total radiated power (Wm^{-3})



$P_{RAD,DIV}$ = 53.4 MW

38.7 (Ne) + 14.2 (D)

$P_{RAD,DIV}$ = 54.0 MW

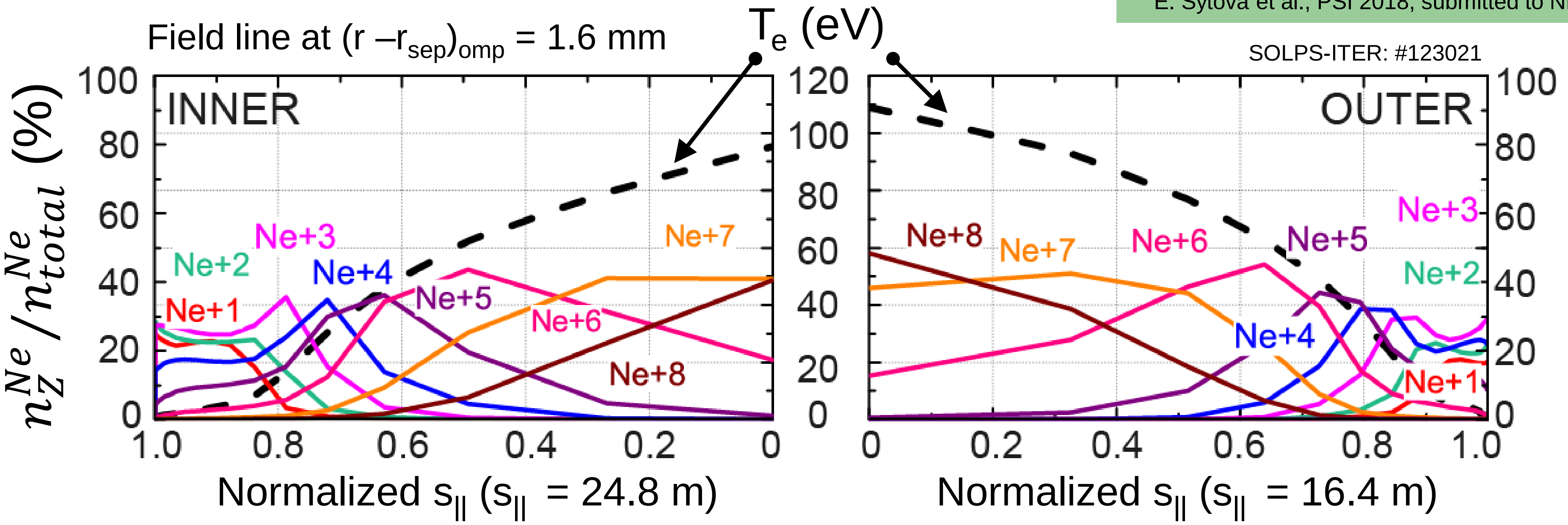
44.7 (Ne) + 9.2 (D)

- New results from SOLPS-ITER
  - P_{IN} = 100 MW
  - Matched Ne, N cases
  - H-mode pedestal
- Similar to SOLPS-4.3
  - Drift effects not important at high p_n
  - High magnetic field and better “divertor impurity screening”

E. Sytova et al., submitted to NME

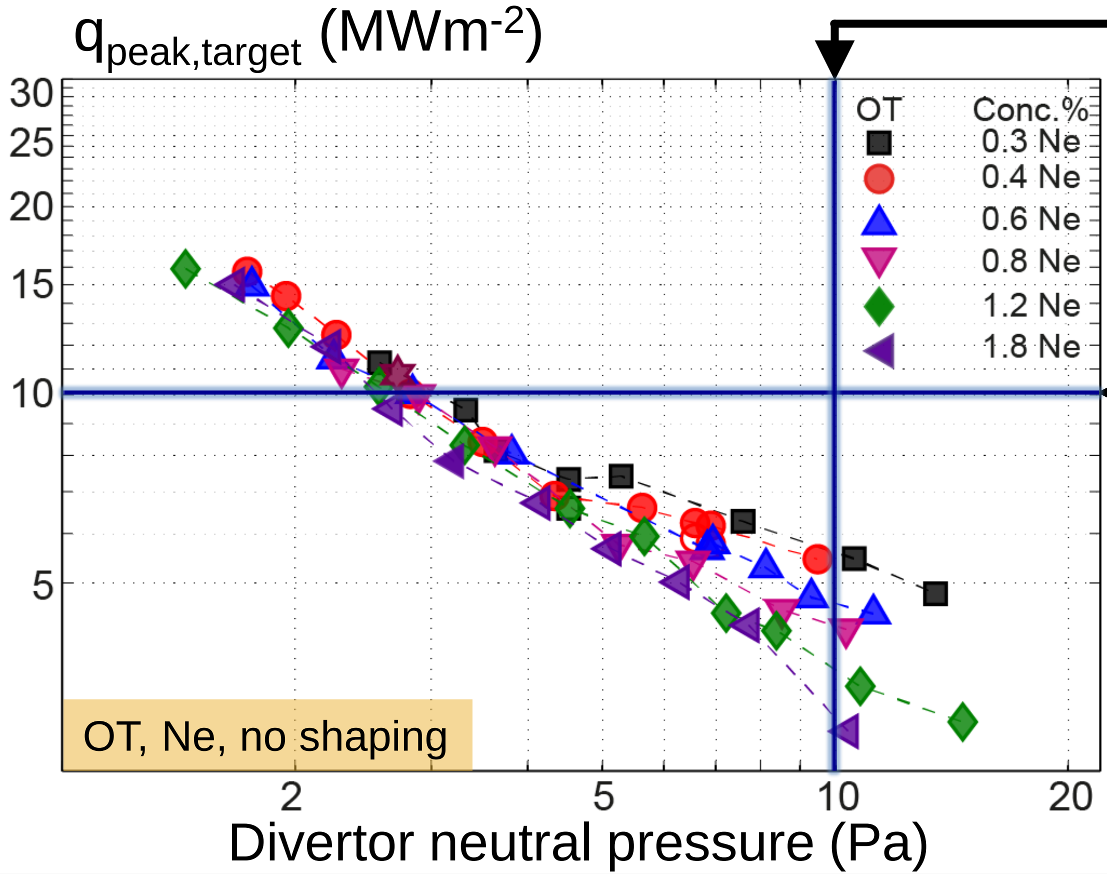
Impurity charge state distribution

E. Sytova et al., PSI 2018, submitted to NME



- ~87% of the divertor radiation comes from Ne^{+3} → Ne^{+6}
  - Well confined in the divertor region → T_e high enough, far enough
  - Ne fully stripped in pedestal region and cannot radiate

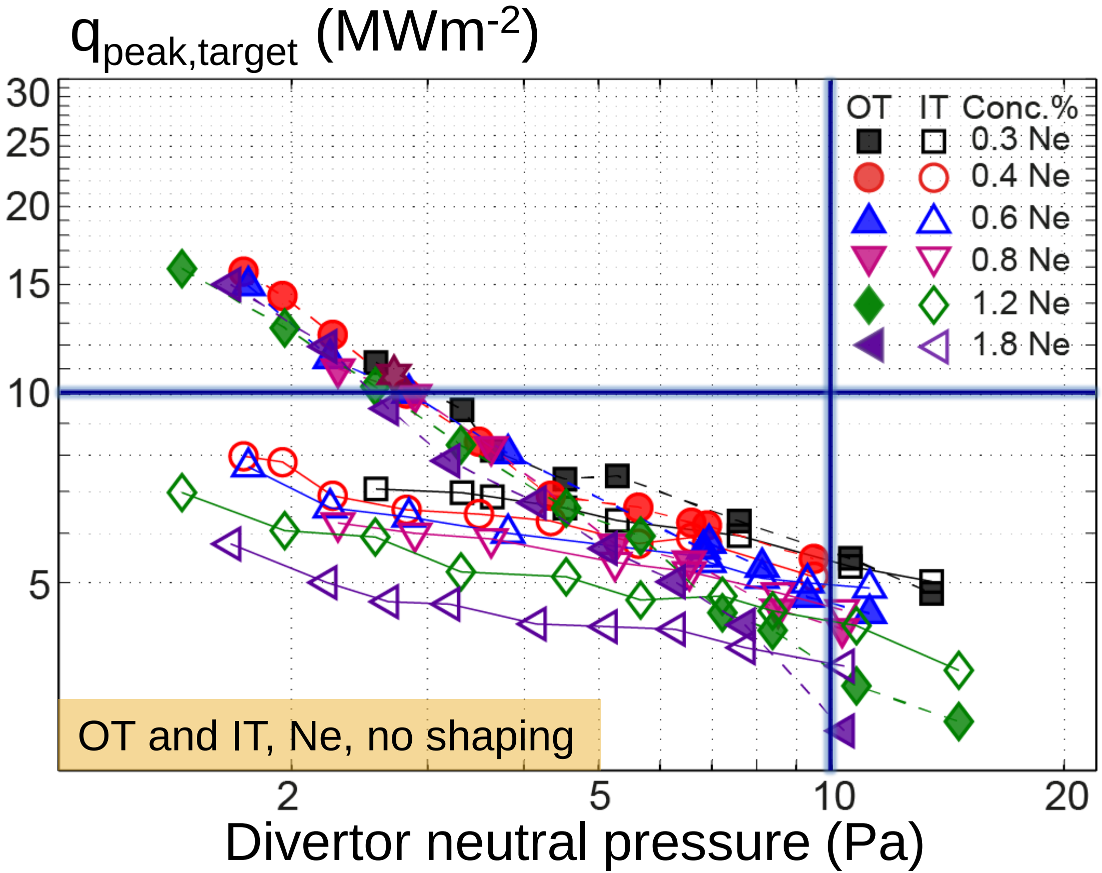
Operating window in peak power flux density



Approximate "Detachment limit" (see later)

"Historical" power handling limit – will be higher in reality

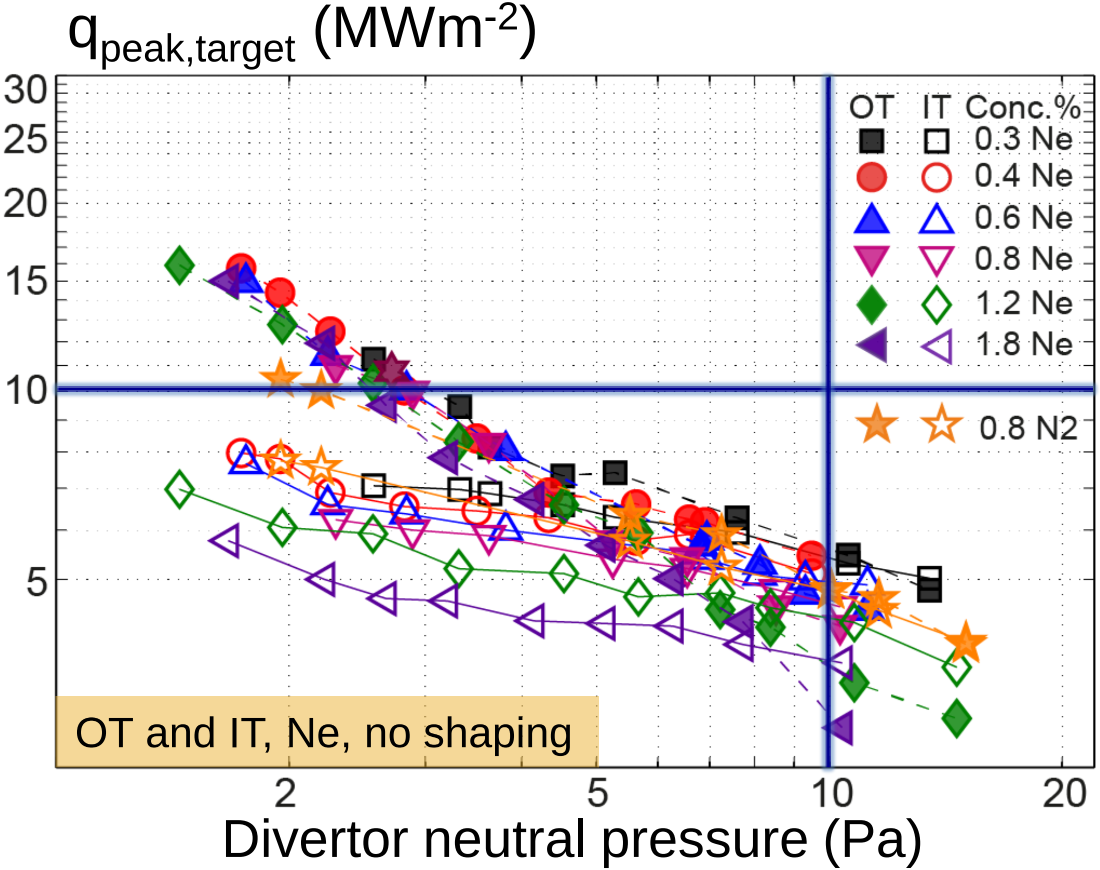
Operating window in peak power flux density



- Out-in asymmetry reduces at high p_{neut}
  - Strong neutral convection from inner to outer through private flux region balances ion flow from outer to inner target through the SOL → not a drift effect

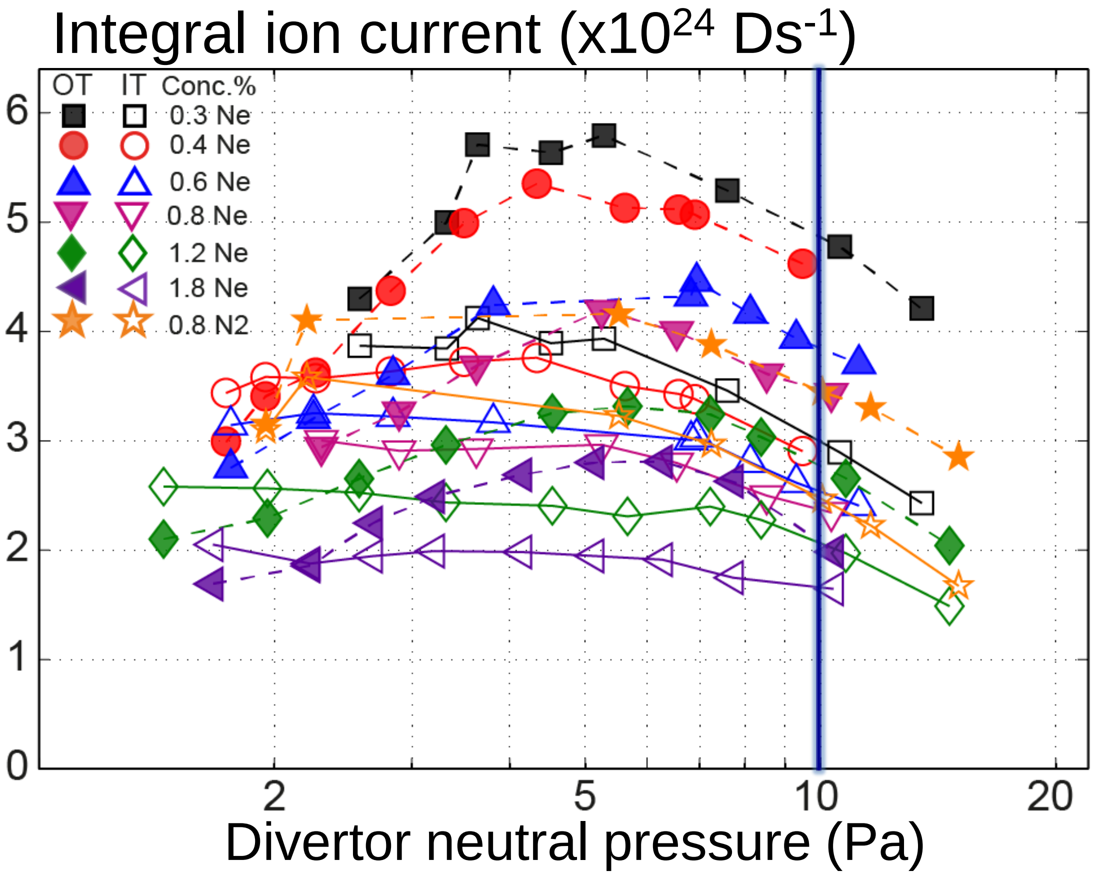
A. A. Pshenov et al, PoP 24 (2018) 072508

Operating window in peak power flux density



- Nitrogen points overlay well with Ne cases
  - Database more restricted for N but trends similar
  - Need 3-5x as much N than Ne in the code for given D fueling to obtain similar midplane impurity separatrix concentration

Integrated target ion fluxes

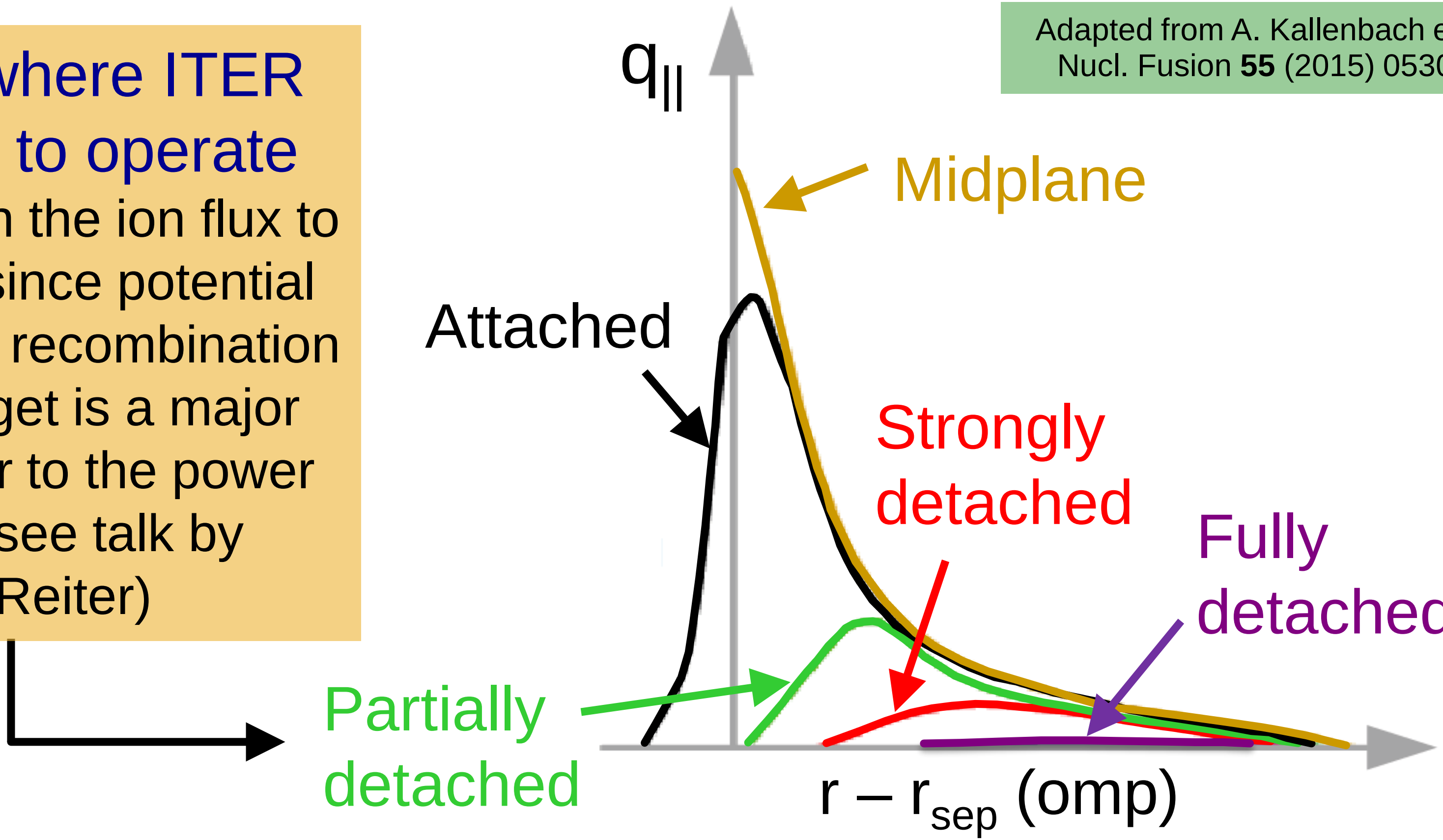


- Turnover in total plate current generally rather gentle
  - Loose criterion for "tolerable detachment" fixed as point at which integral flux reaches ~80% of peak value after rollover (based historically on discussions with JET) → happens typically near p_n ~10 Pa

Classic picture of detachment

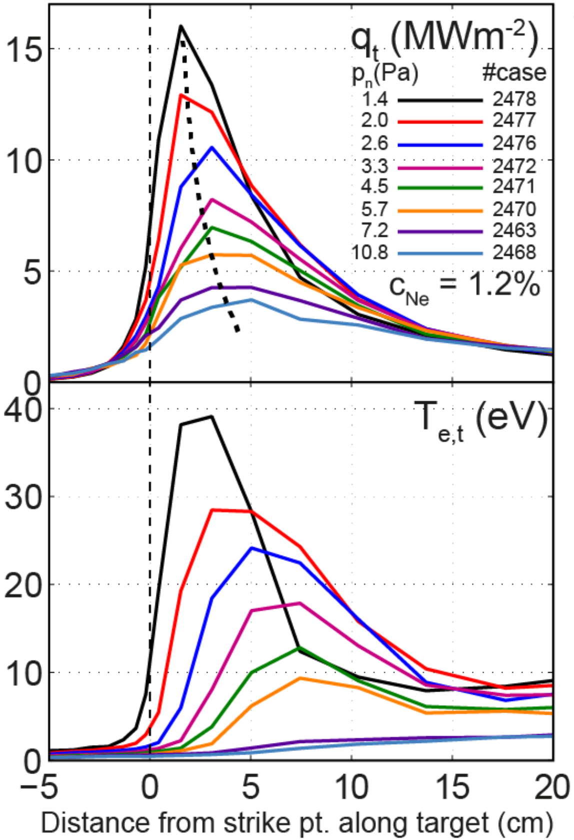
This is where ITER expects to operate

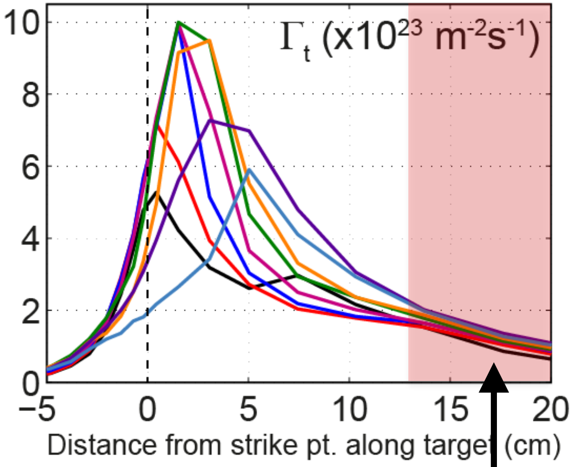
(bring down the ion flux to the plate since potential released in recombination at the target is a major contributor to the power load – see talk by D. Reiter)



Adapted from A. Kallenbach et al., Nucl. Fusion **55** (2015) 053026

Detachment evolution: outer target example



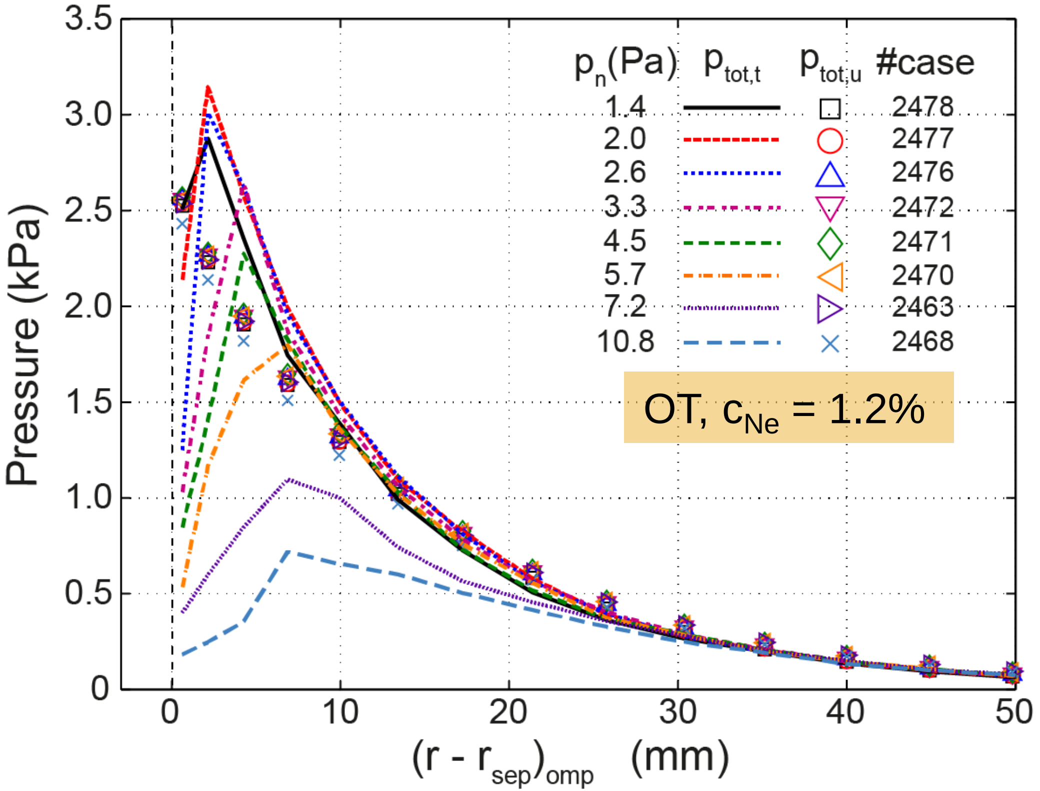


Avoid “complete” detachment → keep finite ion flux in outer part of the SOL to maintain sufficient neutral plugging

- “Classic” evolution from high recycling to partially detached state
  - He pumping improves with increased p_n

Total pressure-momentum losses

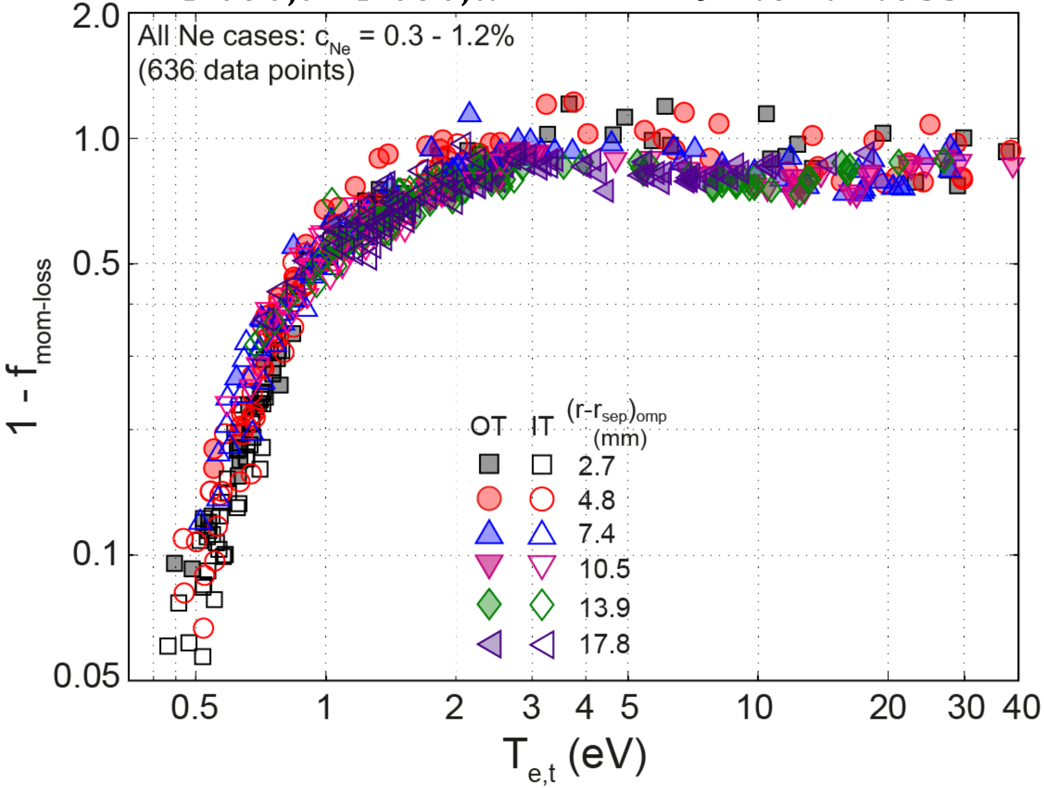
$$p_{tot} = k(n_e T_e + n_i T_i)(1 + M^2)$$



- Pressure loss downstream as p_n increases
  - Upstream p_{tot} unaffected by downstream conditions (as for λ_q)
  - Beyond region of pressure loss, upstream and downstream profiles overlap

Pressure-momentum losses vs. $T_{e,t}$

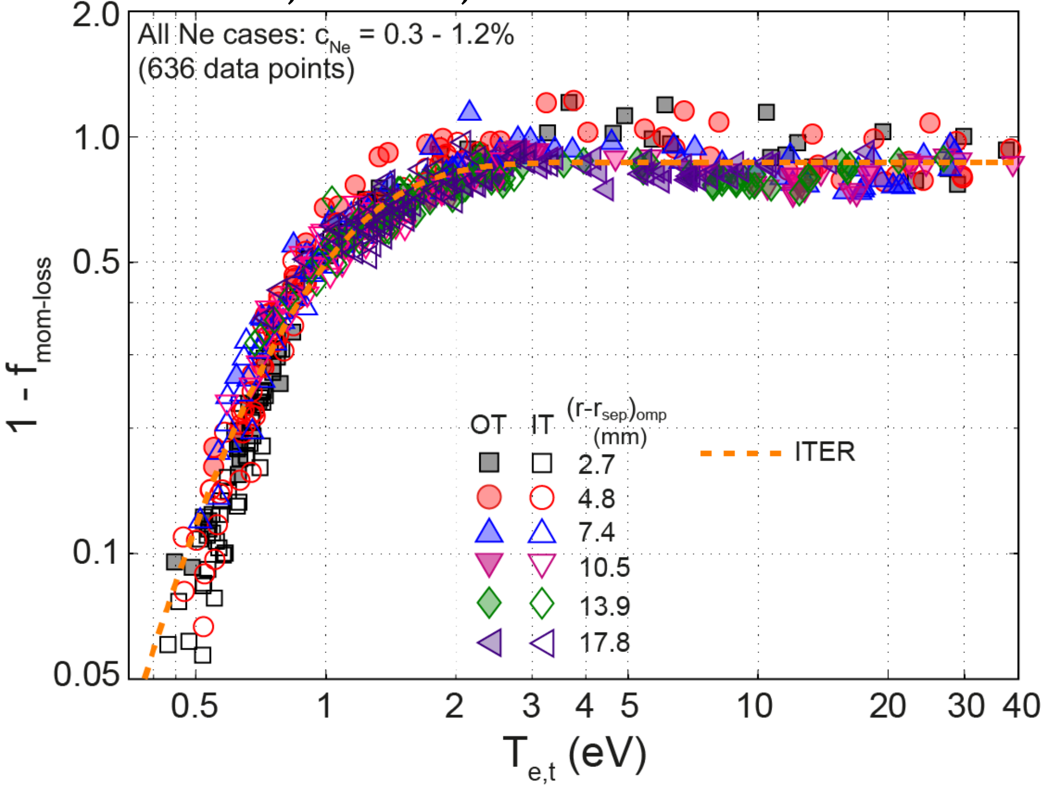
$$p_{tot,t}/p_{tot,u} = 1 - f_{mom-loss}$$



- Momentum loss strong function of $T_{e,t}$
  - Same at both IT and OT
  - Loss starts at $T_{e,t}$ ~ 3 eV
  - Strong below $T_{e,t}$ ~ 1 eV
  - Implies strong role for volume recombination → important for ITER (magnetic field incidence angles are high)

Pressure-momentum losses vs. $T_{e,t}$

$$p_{tot,t}/p_{tot,u} = 1 - f_{mom-loss}$$



- Momentum loss strong function of $T_{e,t}$
  - ITER SOLPS-4.3 simulation database well fitted by functional form proposed by Stangeby*:

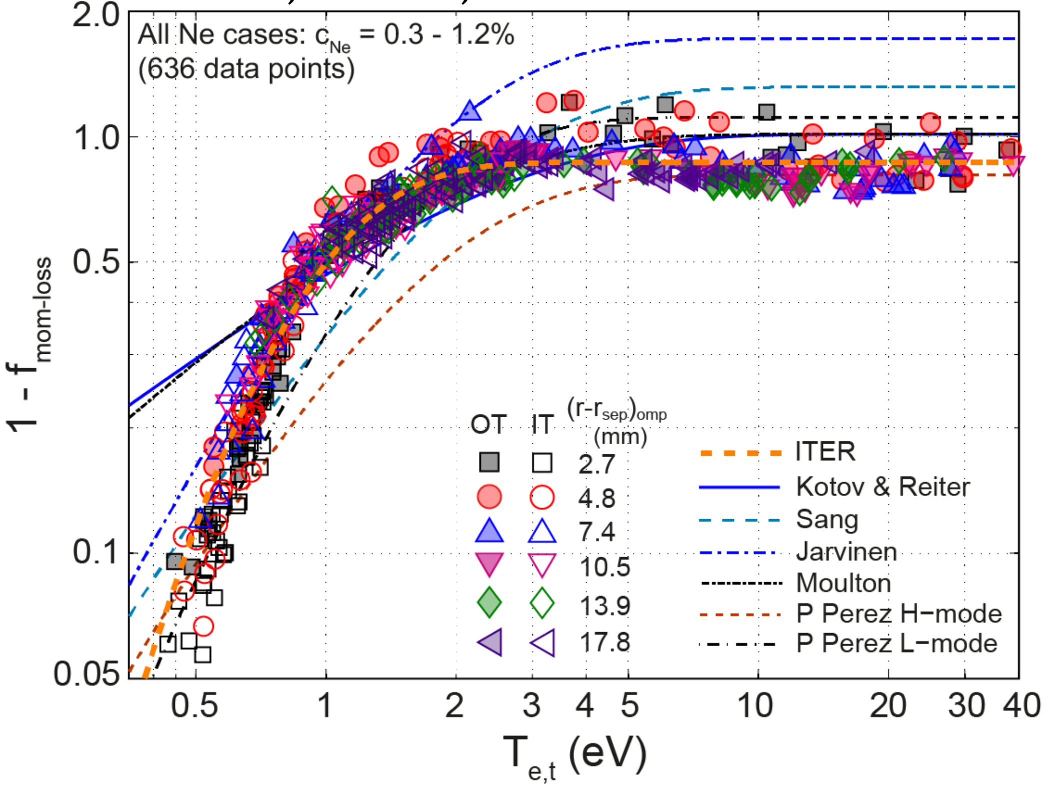
$$1 - f_{mom}^{tot} = A\left(1 - e^{-T_{et}/T^*}\right)^n$$

A = 0.87, T* = 0.43, n = 5.30
R^2 = 0.96

*P. C. Stangeby, PPCF **60** (2018) 044022

Pressure-momentum losses vs. $T_{e,t}$

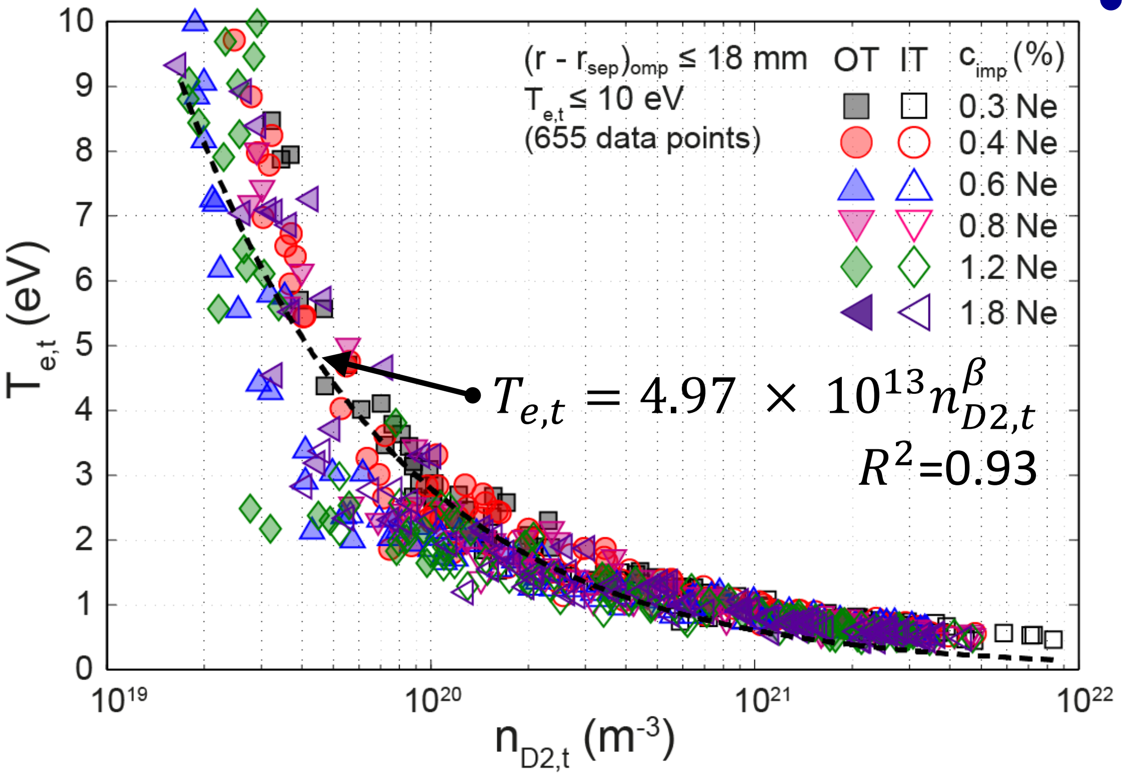
$$p_{tot,t}/p_{tot,u} = 1 - f_{mom-loss}$$



- Generally steeper trend than similar fits to other code studies*
  - BUT note: “balance analysis” now underway indicates that particle removal by VR not necessarily the dominant process
  - Neutral atoms and molecules created by VR responsible for up to half of $f_{mom-loss}$

*P. C. Stangeby, PPCF **60** (2018) 044022

Plate molecular density and $T_{e,t}$

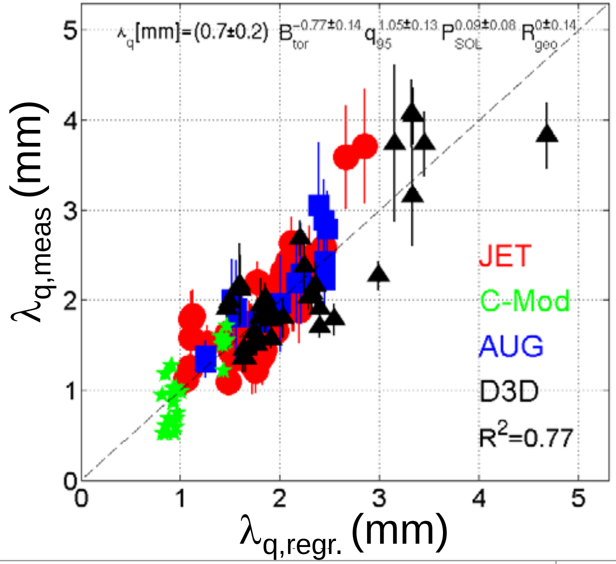


- $T_{e,t}$ and $n_{D2,t}$ tightly correlated ($T_{e,t} \lesssim 10$ eV)
  - Similar to findings from other code studies
  - $n_{D2,t}$ related to flux amplification at targets and hence to $f_{mom\text{-}loss}$ → expect link between $n_{D2,t}$ and $T_{e,t}$
  - Currently investigating link between $T_{e,t}$ and volumetric power losses

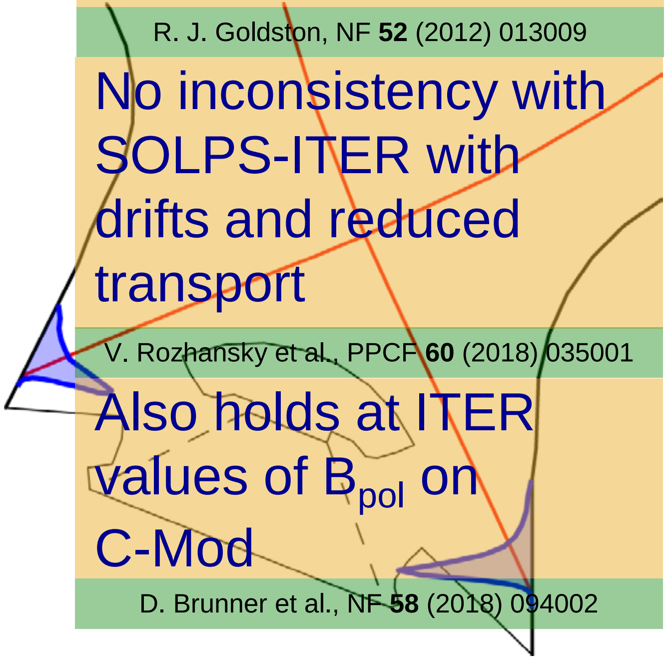
V. Kotov, D. Reiter, PPCF **51** (2009) 115002, P. C. Stangeby, PPCF **60** (2018) 044022

Factors influencing peak power density

λ_q narrower than we have assumed? → best current exptl. scaling gives λ_q ~1 mm for ITER at I_p = 15 MA



T. Eich, NF **53** (2013) 093031



Neoclassical ion drift model very good match to scaling

R. J. Goldston, NF **52** (2012) 013009

No inconsistency with SOLPS-ITER with drifts and reduced transport

V. Rozhansky et al., PPCF **60** (2018) 035001

Also holds at ITER values of B_{pol} on C-Mod

D. Brunner et al., NF **58** (2018) 094002

PFC shaping

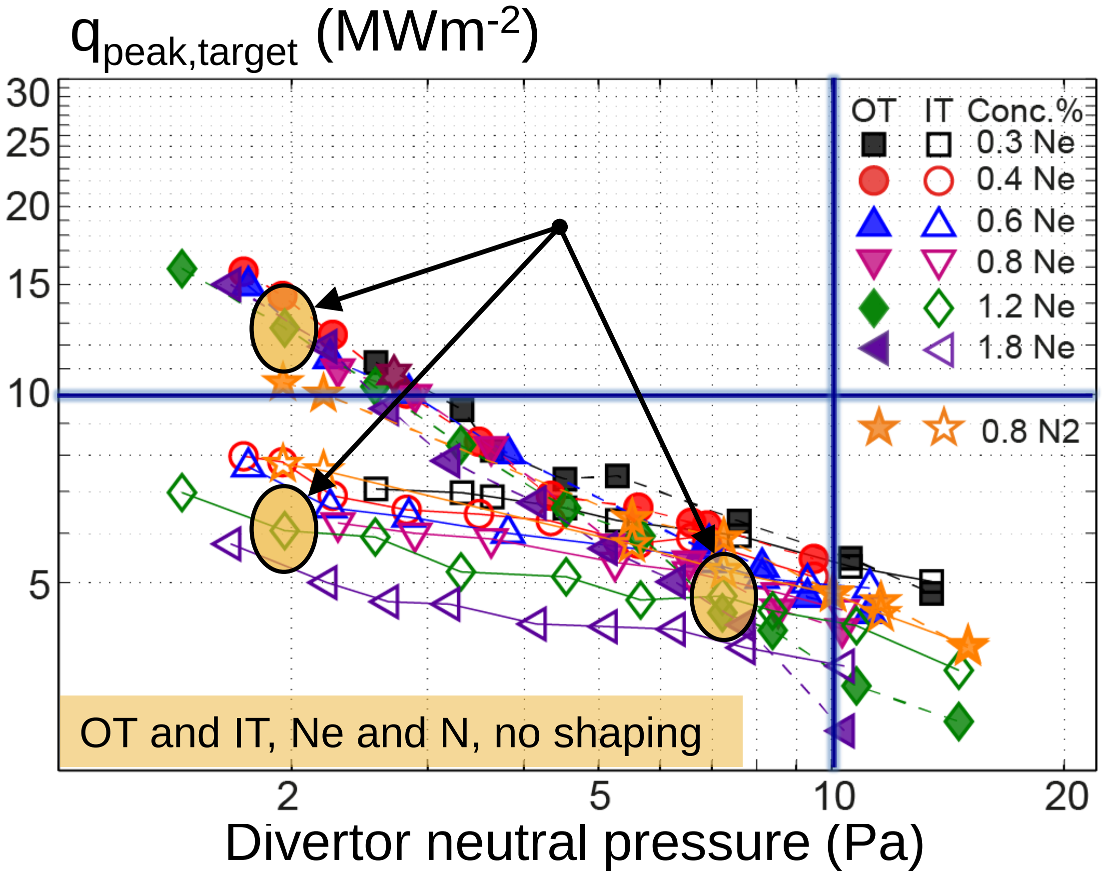
Fluid drifts? (SOLPS-4.3 has no drift capability)

Insufficient numerical simulation grid resolution?

Being actively pursued at Univ. Leuven

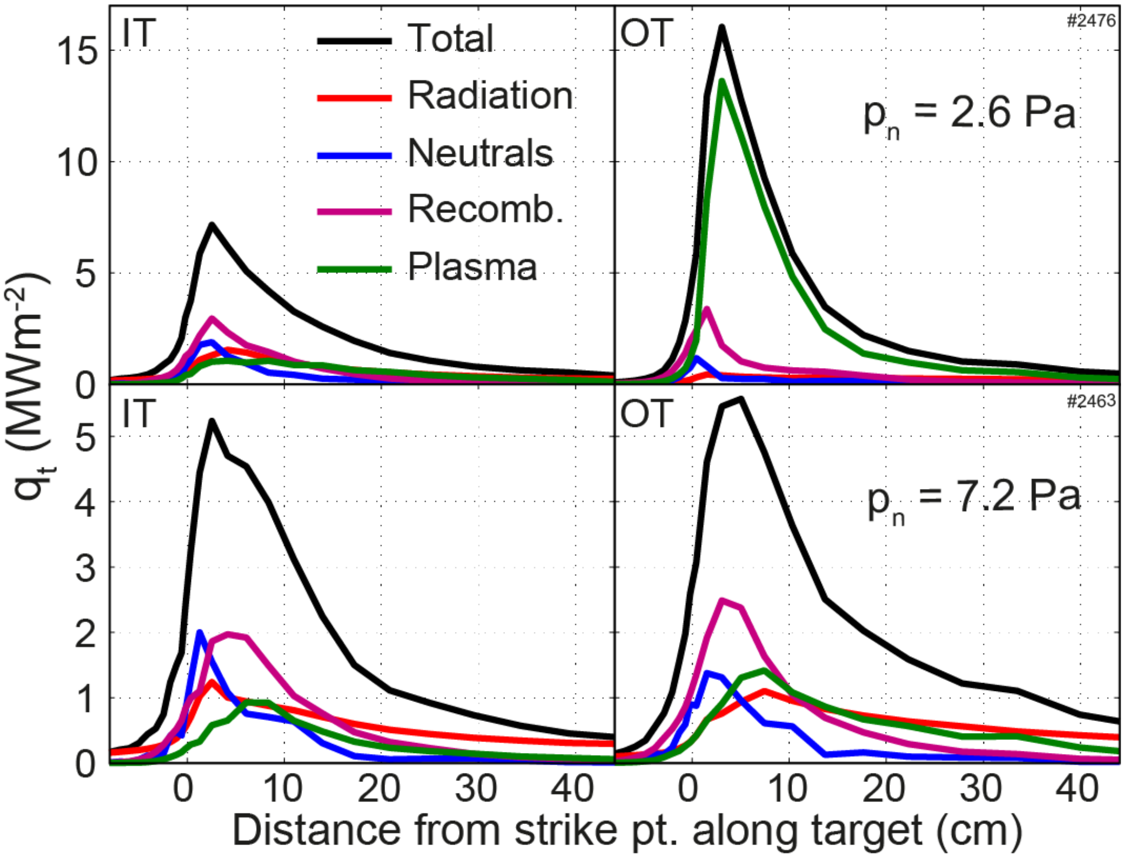
K. Ghoos et al, submitted to NF

Impact of shaping



- Need to apply angle corrections for global target tilting and monoblock toroidal shaping **only** to thermal plasma components
  - Kinetic plasma plus potential energy of recombination at the plate: $\gamma n_{et} c_{st} T_{et} + n_{et} c_{st} E_{pot}$

Impact of shaping



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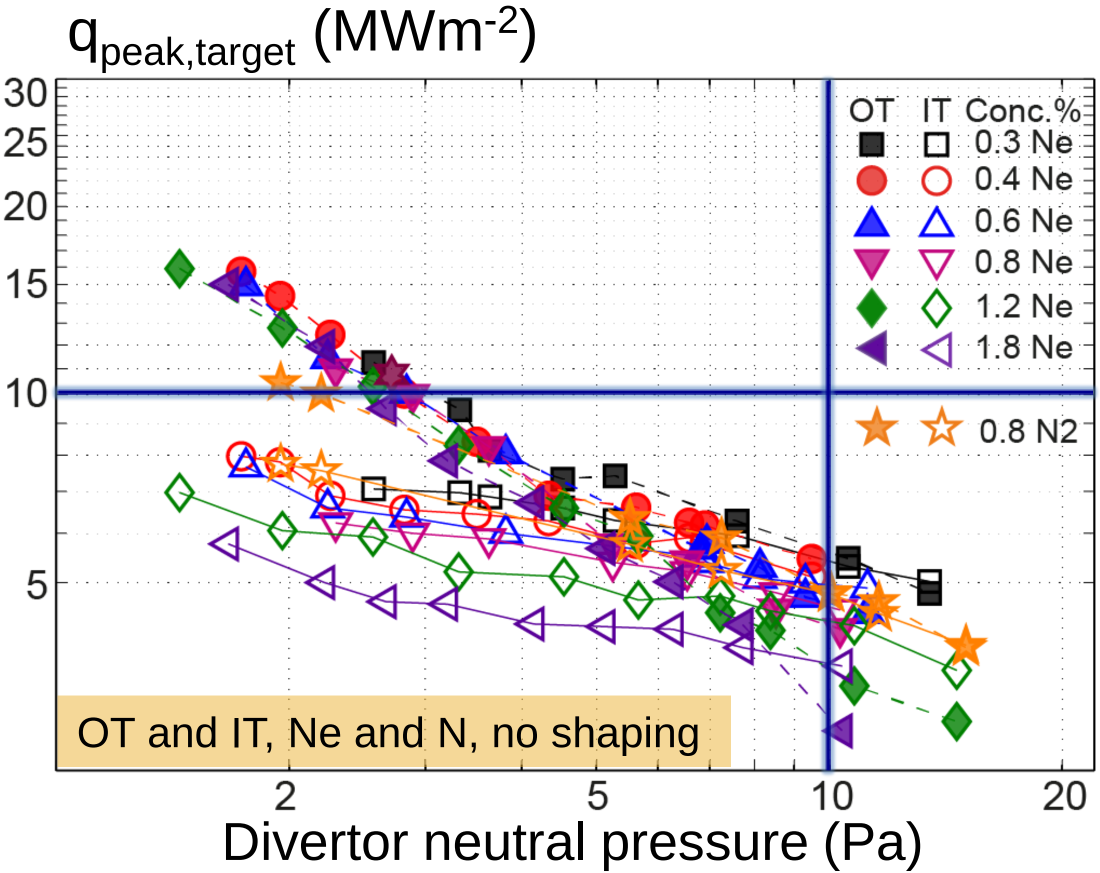
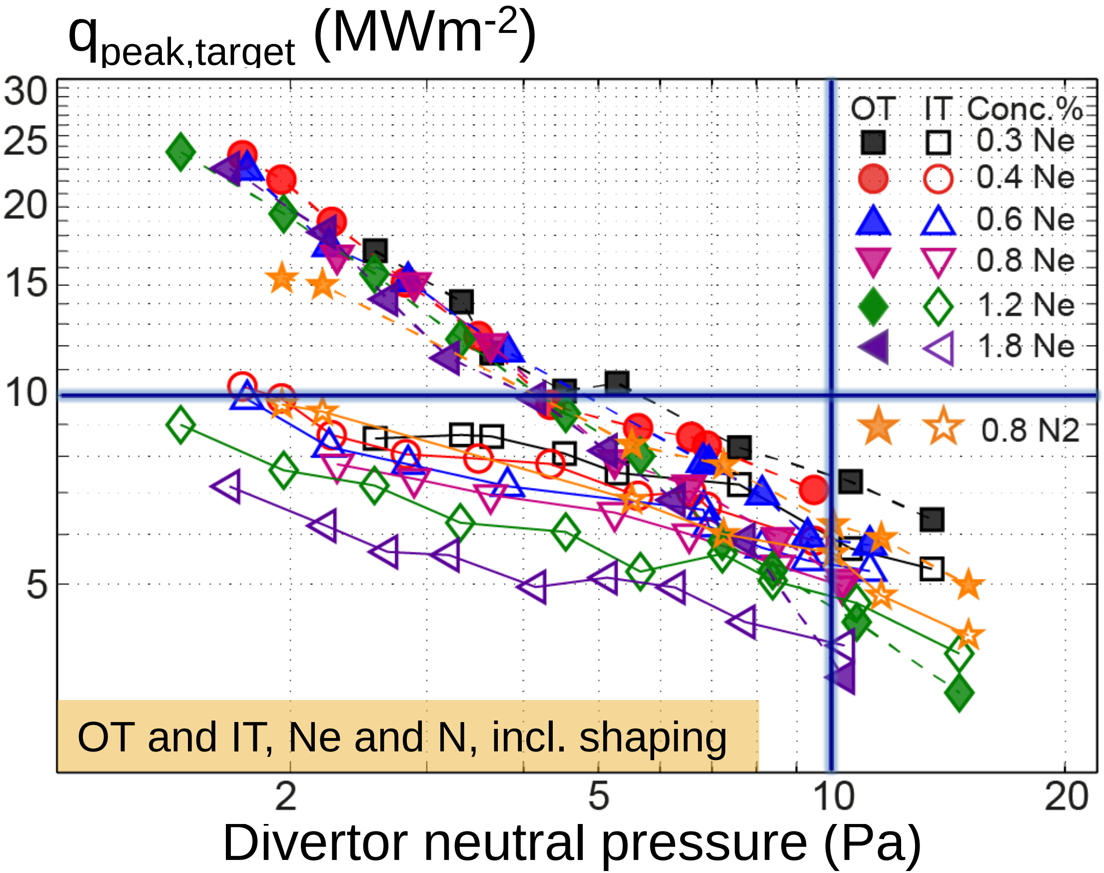
Impact of shaping



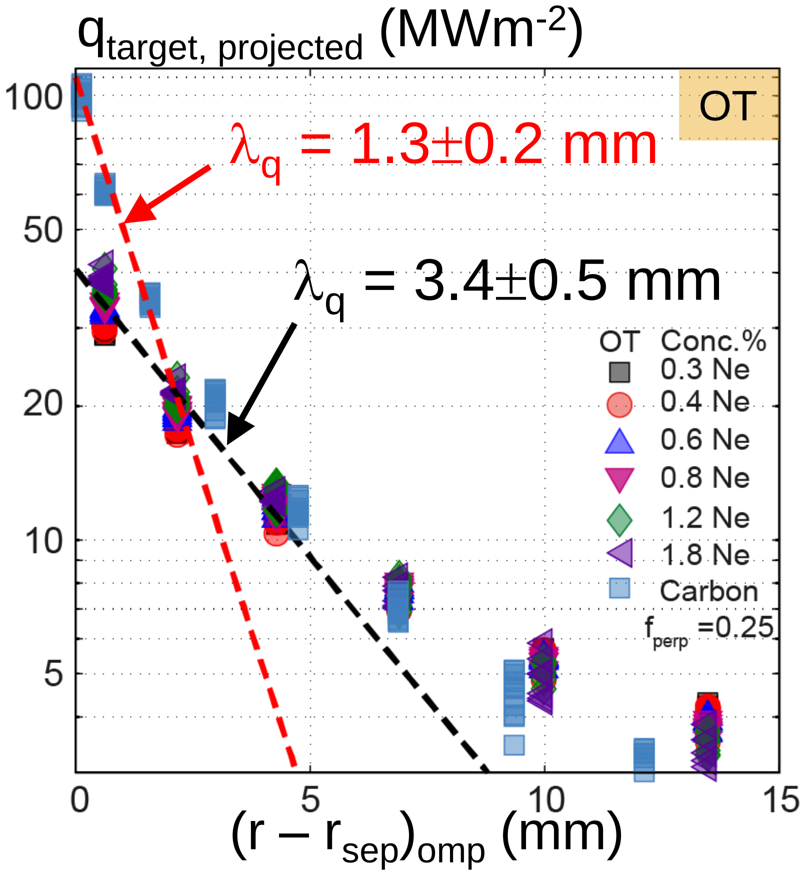
- Reminder, no shaping

Impact of shaping



- Now with shaping
  - Effects less marked at high p_n (where thermal plasma contributions lower)

Impact of reduced transport



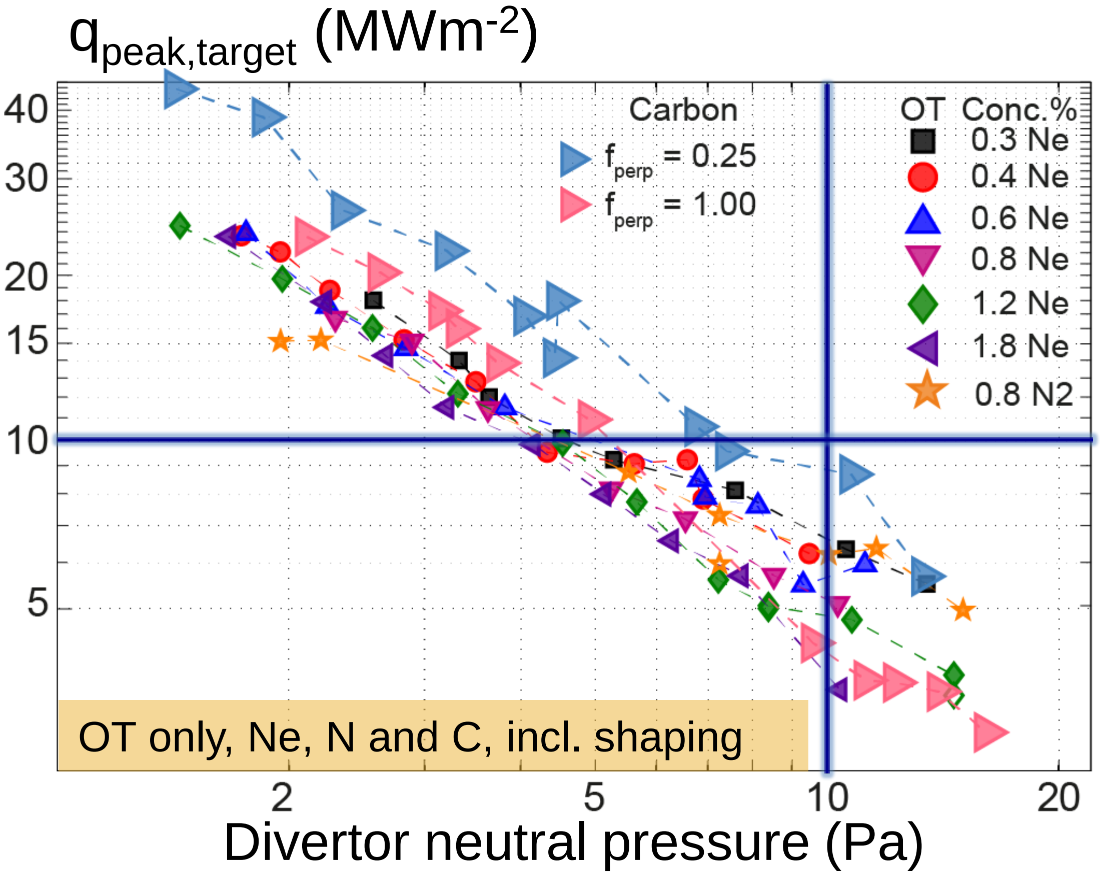
- Reduce $D_\perp$, $\chi_\perp$ by factor 4 compared to baseline:

  $D_\perp = 0.075\ m^2s^{-1}$

  $\chi_\perp = 0.25\ m^2s^{-1}$
  - Only old carbon divertor cases*
  - New SOLPS-ITER runs for Be/W underway
  - λ_q = 1.3 mm close to experimental scaling for ITER at 15 MA

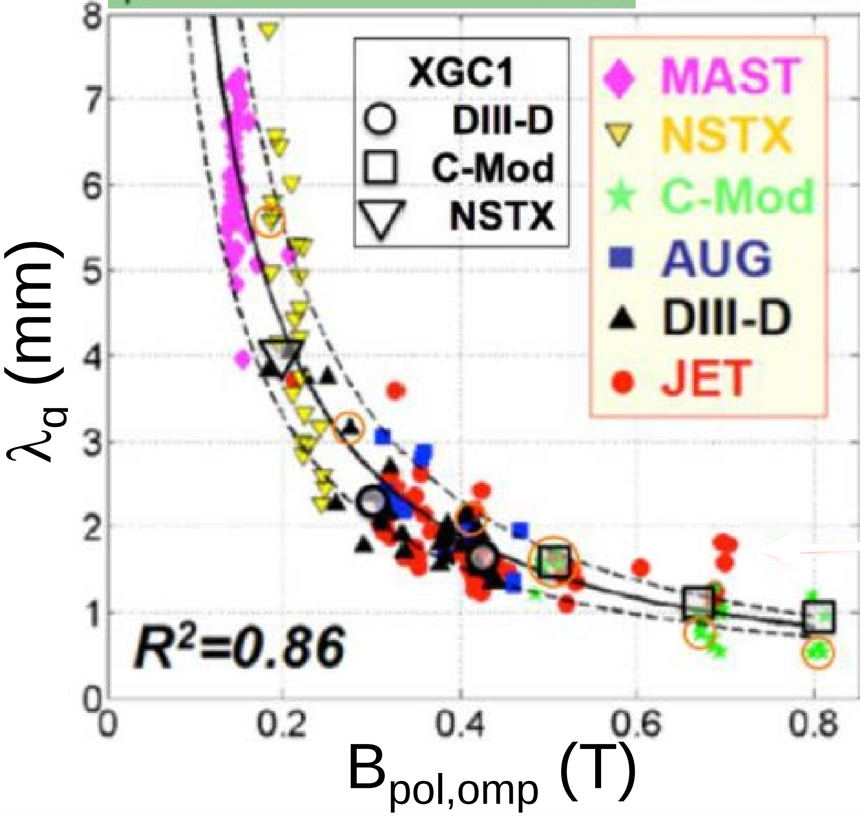
*A. S. Kukushkin et al. JNM **438** (2013) S203

Impact of reduced transport and shaping



- Window for target heat flux density narrows to higher neutral pressure
  - Proximity to complete detachment threshold?
  - Upstream density limits and detachment stability?
  - R&D priority

What will be the true λ_q on ITER?

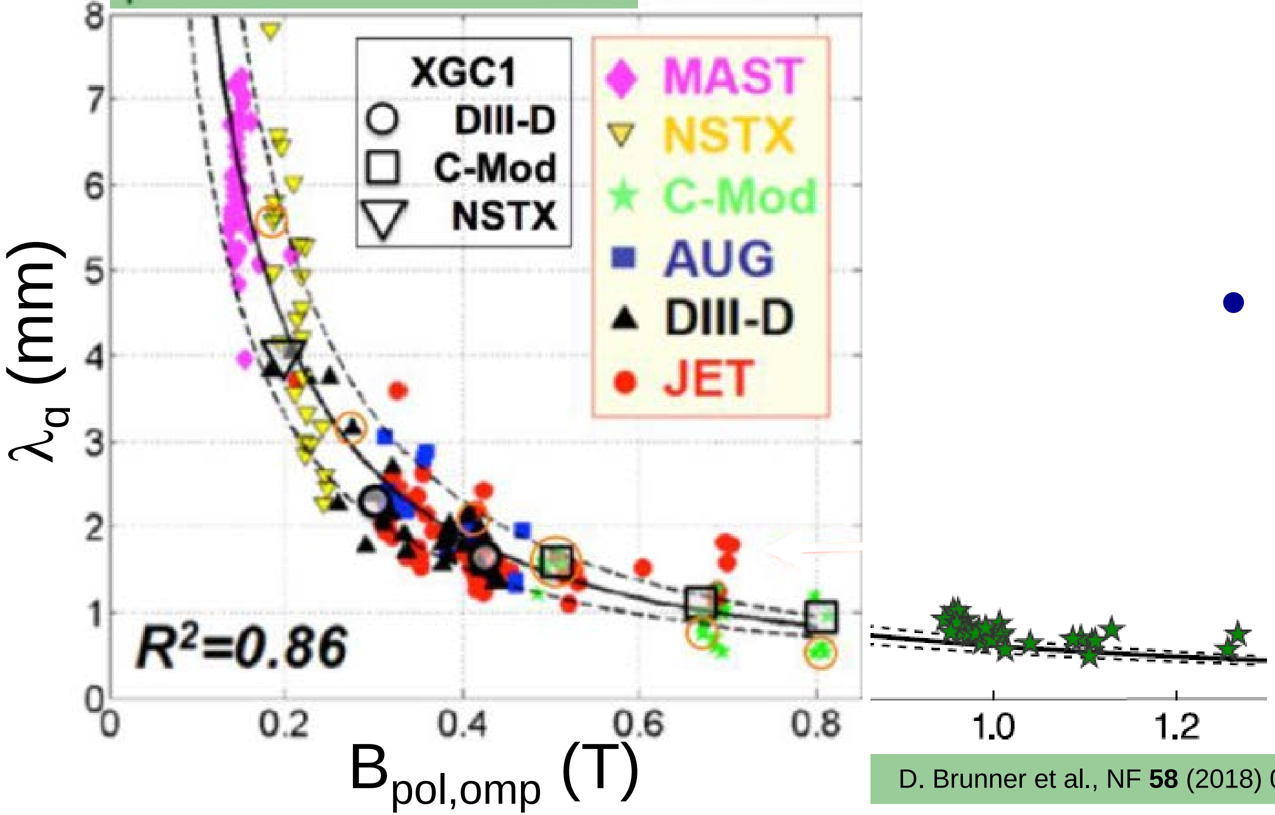


C. S. Chang et al. NF **57** (2017) 116023

- XGC1 electrostatic global gyrokinetic simulations match $\lambda_q \propto 1/B_{pol}$ scaling

What will be the true λ_q on ITER?

C. S. Chang et al. NF **57** (2017) 116023

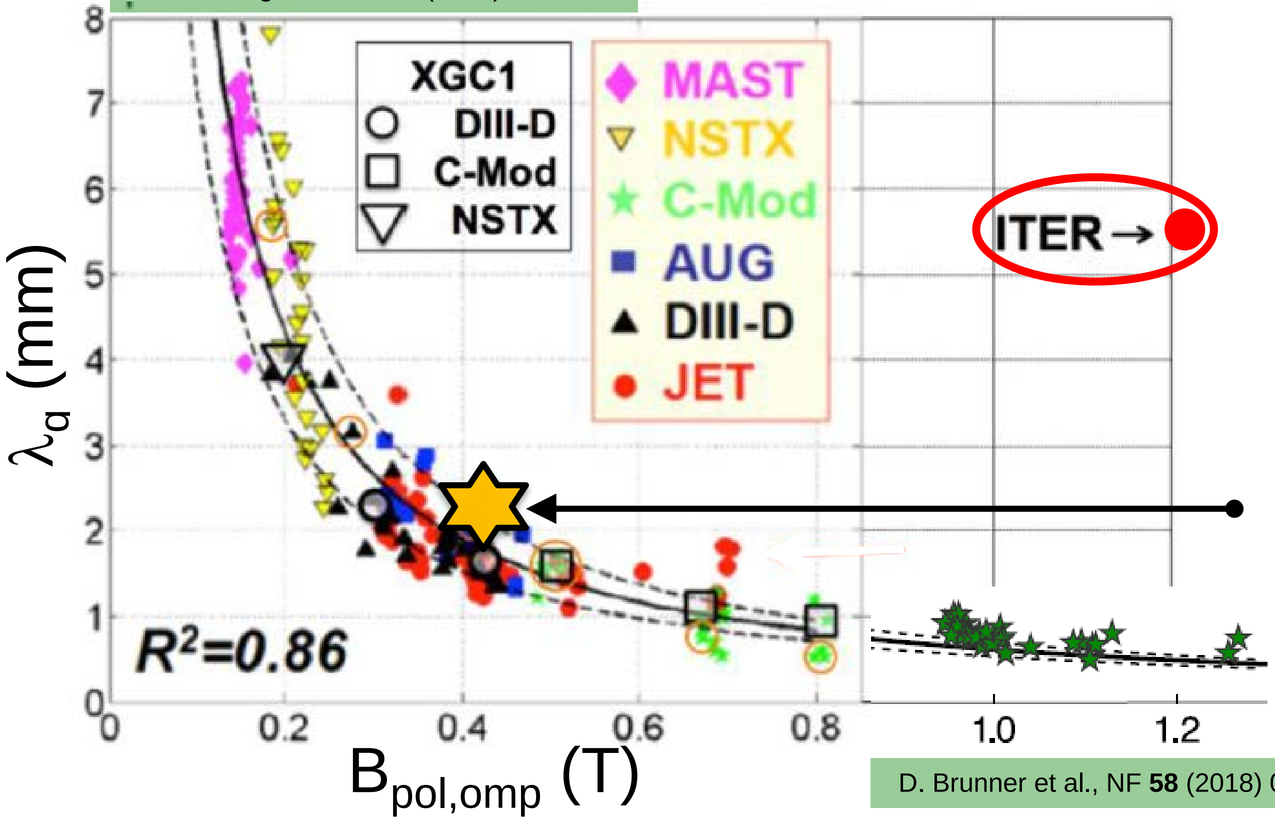


D. Brunner et al., NF **58** (2018) 094002

- XGC1 electrostatic global gyrokinetic simulations match $\lambda_q \propto 1/B_{pol}$ scaling
- Data out to ITER B_{pol} on C-Mod continue to follow scaling

What will be the true λ_q on ITER?

C. S. Chang et al. NF **57** (2017) 116023

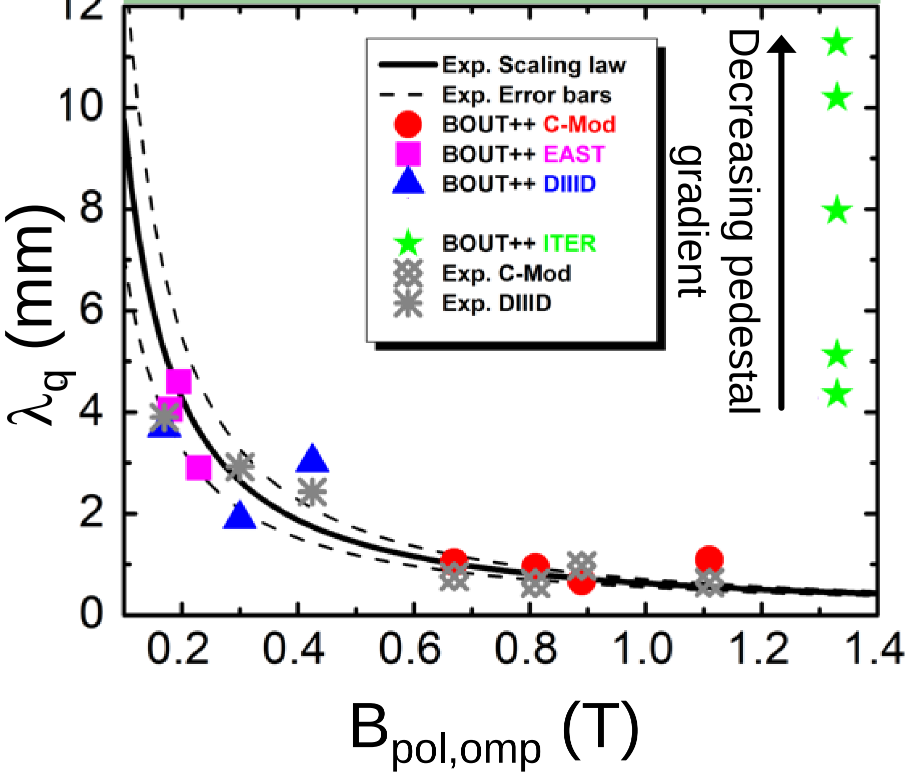


D. Brunner et al., NF **58** (2018) 094002

- But XGC1 dependence broken at ITER scale
  - Attributed to electron turbulence
  - Looks robust (for high I_p, B_T)
  - First XGC1 result for 5 MA ITER H-mode → follows empirical scaling

C. S. Chang et al., IAEA 2018

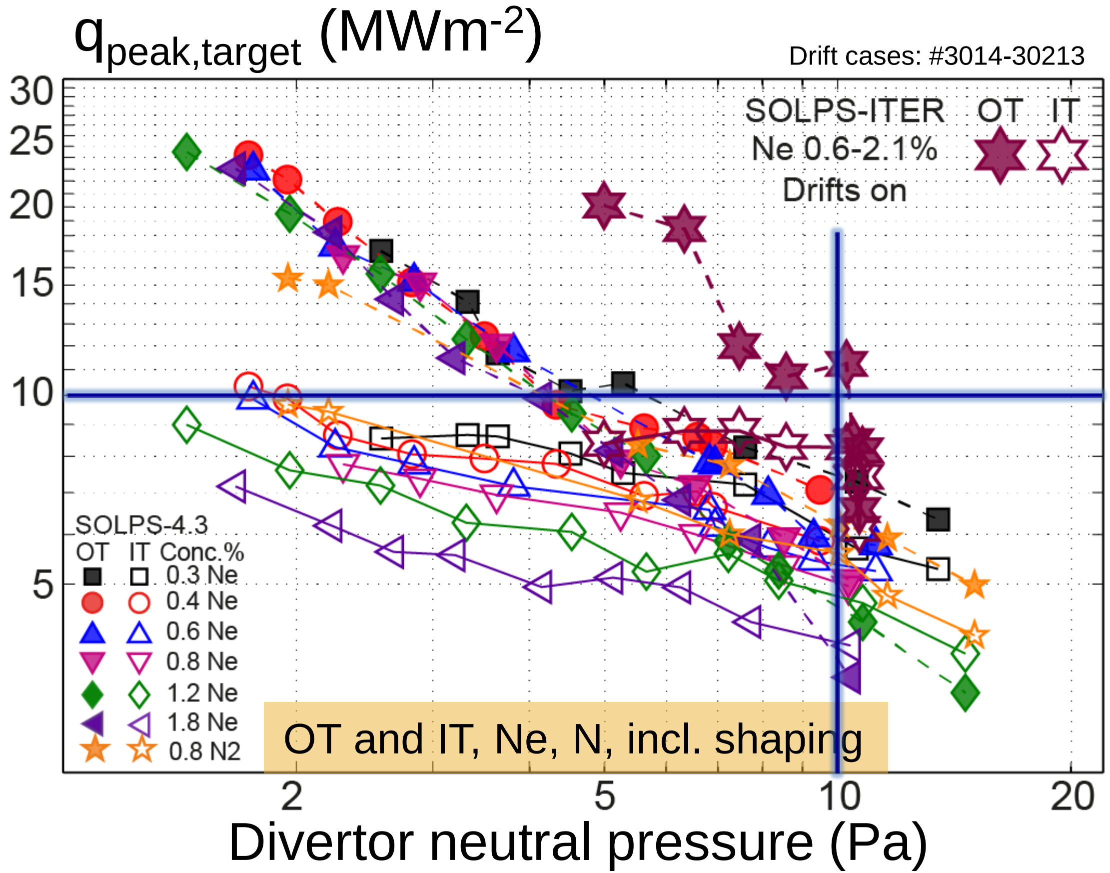
What will be the true λ_q on ITER?



X. Xu et al. IAEA FEC 2018, subm. to Nucl. Fusion

- Very recent BOUT++ electromagnetic turbulence simulations find the same trends:
  - $\lambda_q \propto 1/B_{pol}$ for small devices
  - Broken at high I_p (B_{pol}) on ITER
  - Comparison of “transport BOUT” with “turbulence BOUT” shows a transition from magnetic drift to turbulence at given machine size
- This is a highly controversial and important issue for ITER and reactors

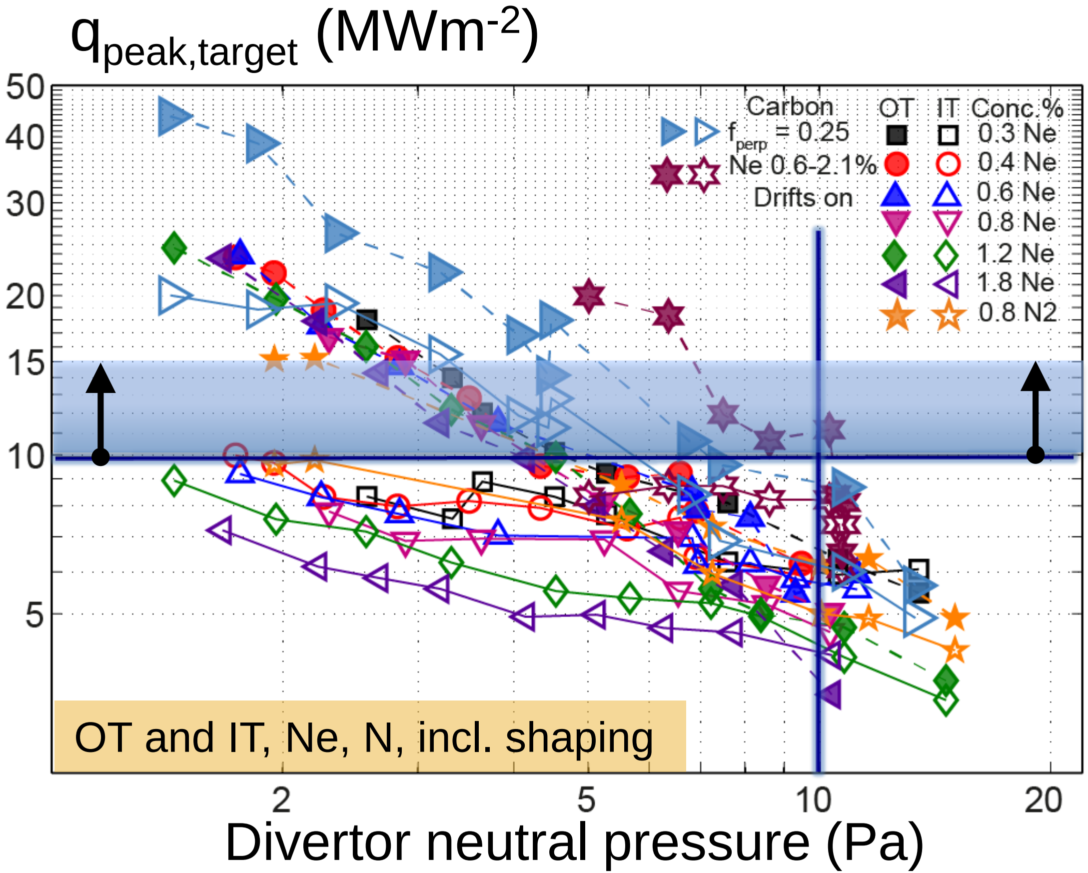
Impact of drifts



- Brand new results from “H-mode” SOLPS-ITER drift modelling*
- Strong impact on out-in asymmetry but effect reduced as detachment deepens
  - Neon very sensitive to drift-induced main ion flows
  - Only at the beginning of this study

*E. Kaveeva et al, in preparation for NF

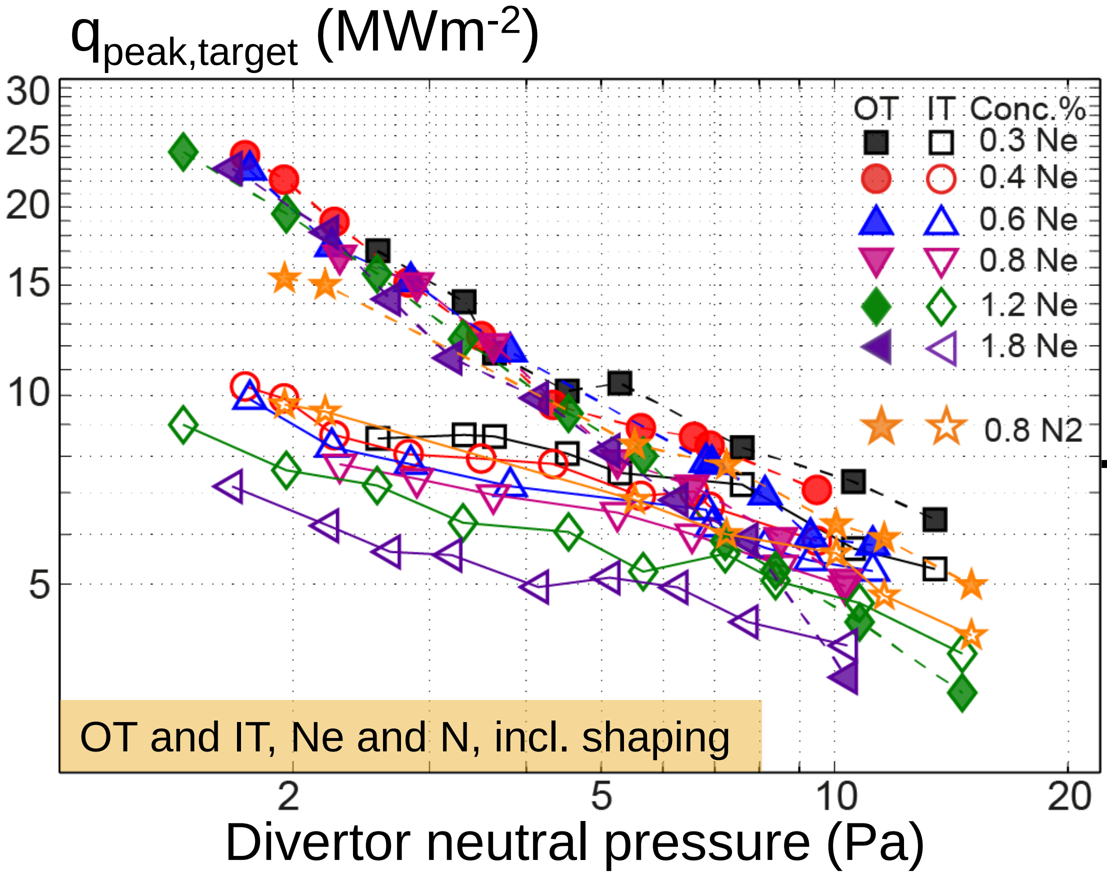
Being pushed into a corner?



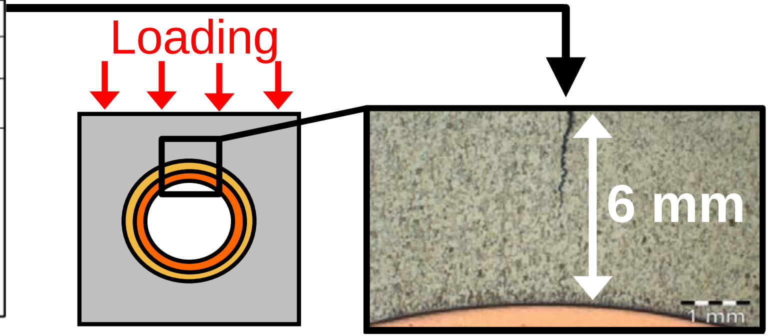
- Combined effect of all factors is push operation to higher divertor pressure
  - Good for He throughput
  - Potentially bad for detachment stability
  - New criterion for tolerable power handling helps a lot*
- Important to assess impact of operation at higher detachment degree

*G. De Temmerman et al, PPCF 60 (2018) 044018

What should be the true power load limit?

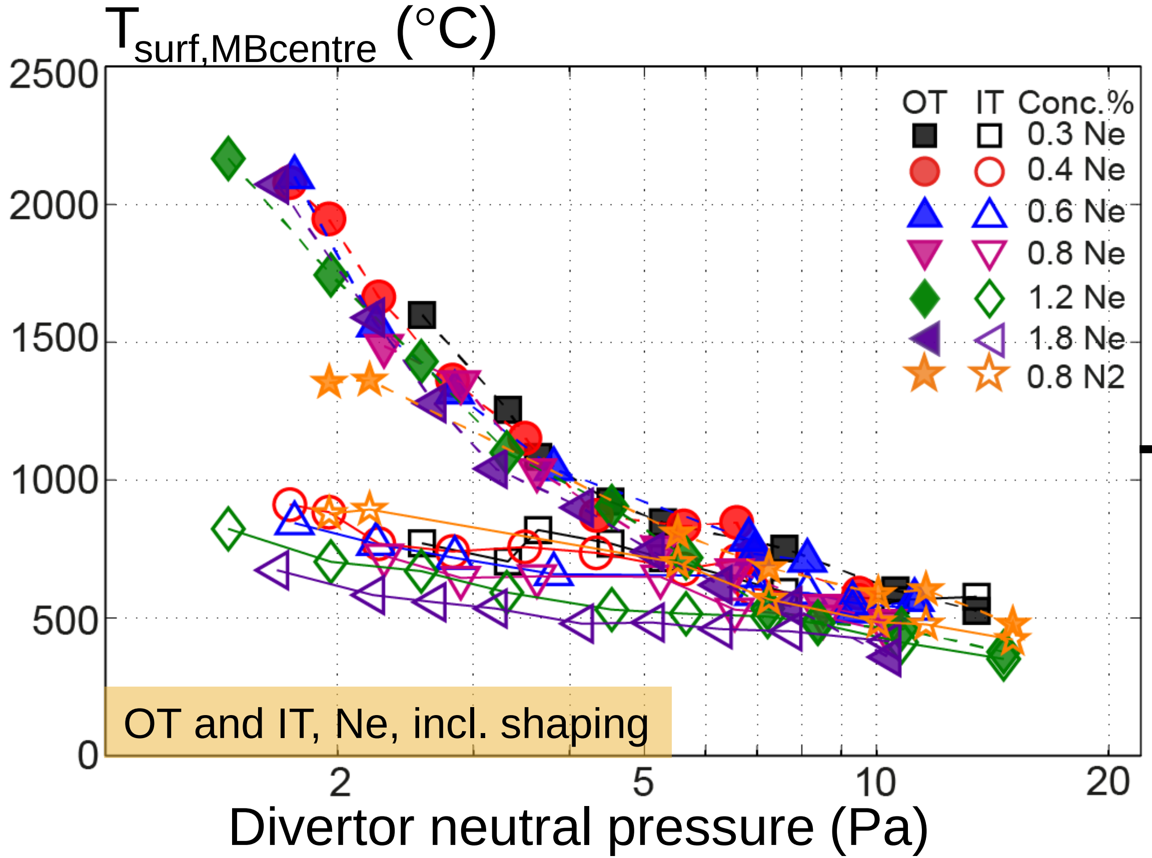


- Use FE simulations to transform q_{peak} to T_{surf}
  - Take value at monoblock centre → where cracking seen to start under high heat flux testing



S. Panayotis et al., NME **12** (2017) 200

What should be the true power load limit?



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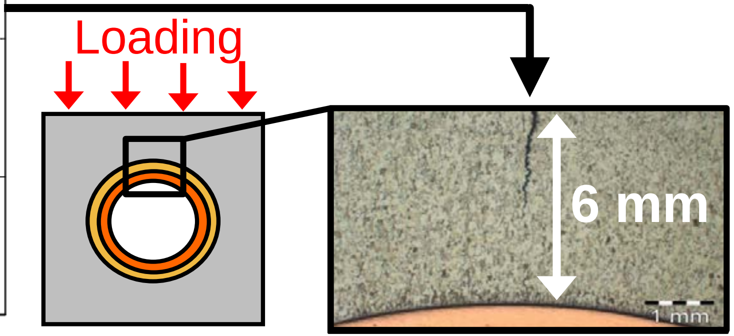
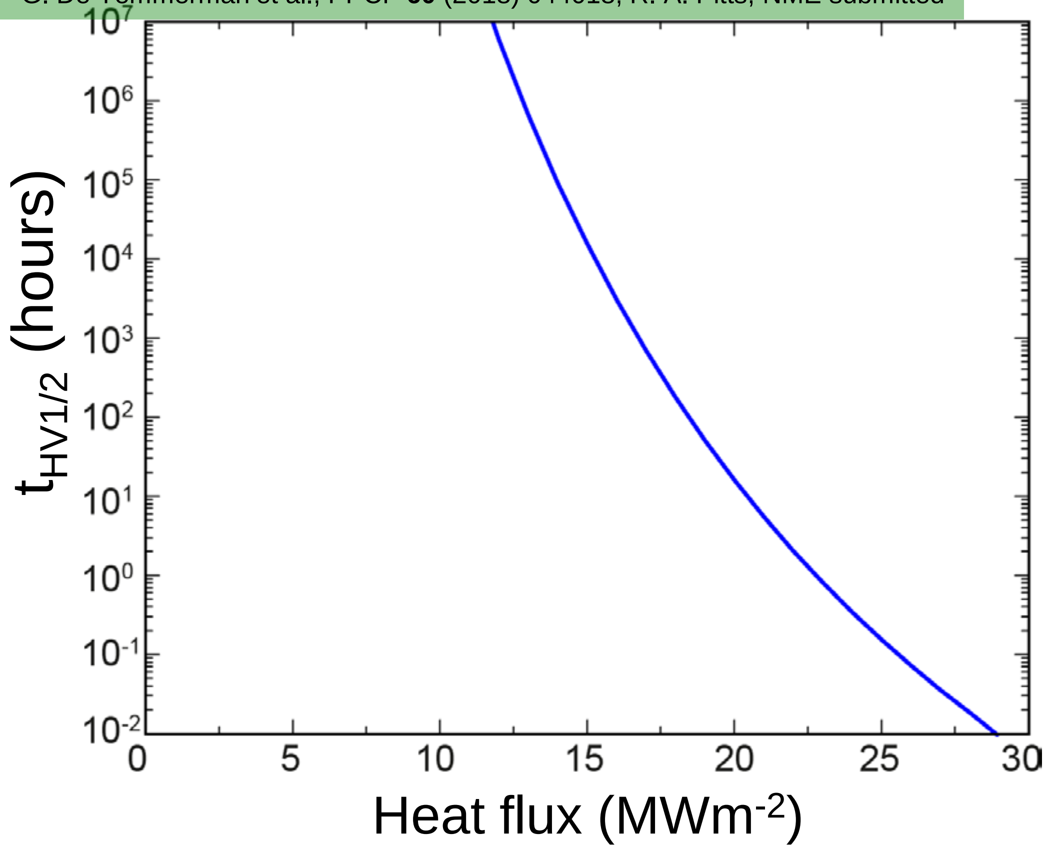


S. Panayotis et al., NME **12** (2017) 200

What should be the true power load limit?

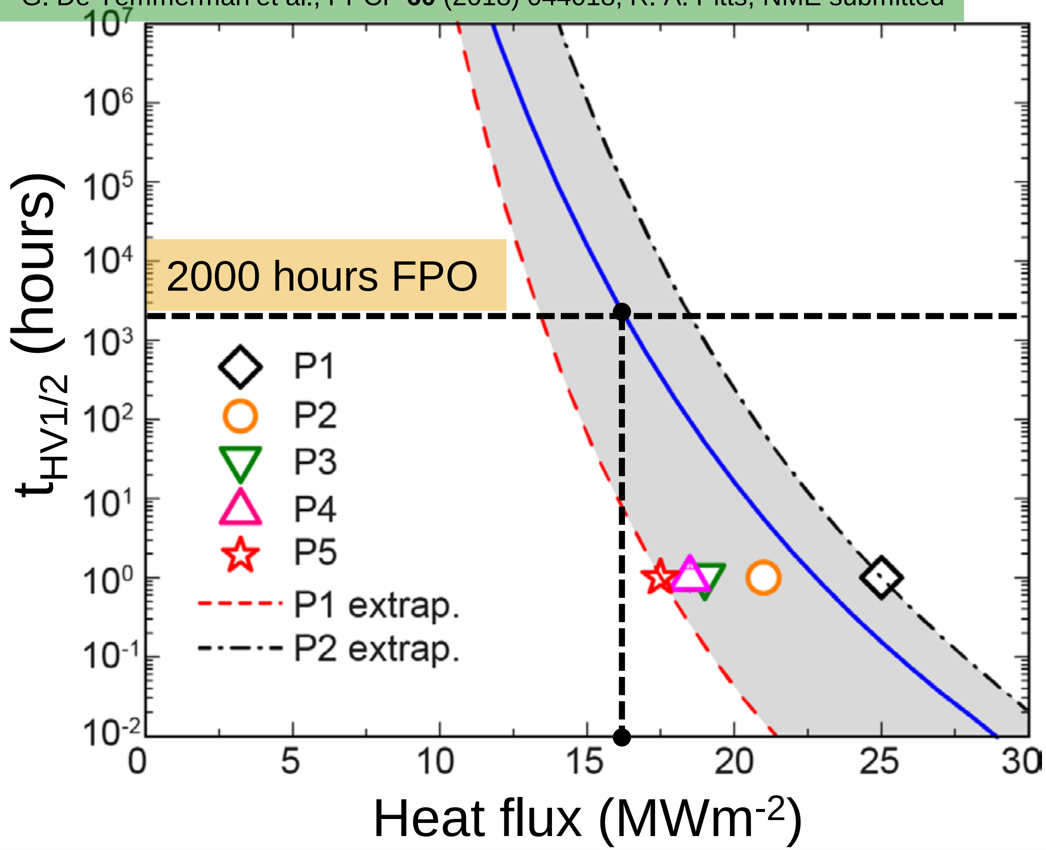
G. De Temmerman et al., PPCF **60** (2018) 044018, R. A. Pitts, NME submitted



- “Operational budget” in $q_{peak,target}$ defined by time required for hardness to drop by 50% at given depth below MB surface
  - ~2 mm recrystallization depth consistent with recent FEM modelling for crack onset due to low cycle fatigue[1]

[1] M. Li et al., Fus. Eng. Des. **101** (2018) 1

What should be the true power load limit?



G. De Temmerman et al., PPCF **60** (2018) 044018, R. A. Pitts, NME submitted

- Add a few points from specific measurements (1 hour annealing) on ITER grade W materials[1]
  - An idea of the range of uncertainty
  - Conclude that a reasonable max stationary heat flux could be **$q_{peak,target} \lesssim 15$ MWm^{-2}** for first ITER divertor to end of FPO

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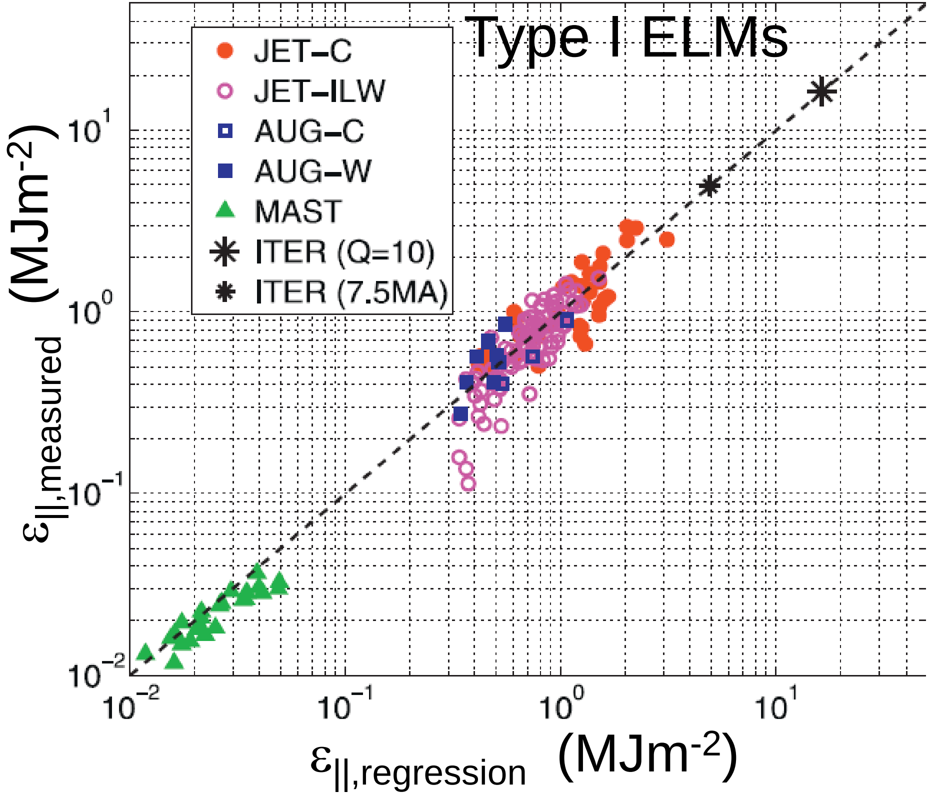
[1] S. Panayotis et al., NME **12** (2017) 200

ELMs – what if suppresion not possible?

T. Eich et al, NME **12** (2017) 84



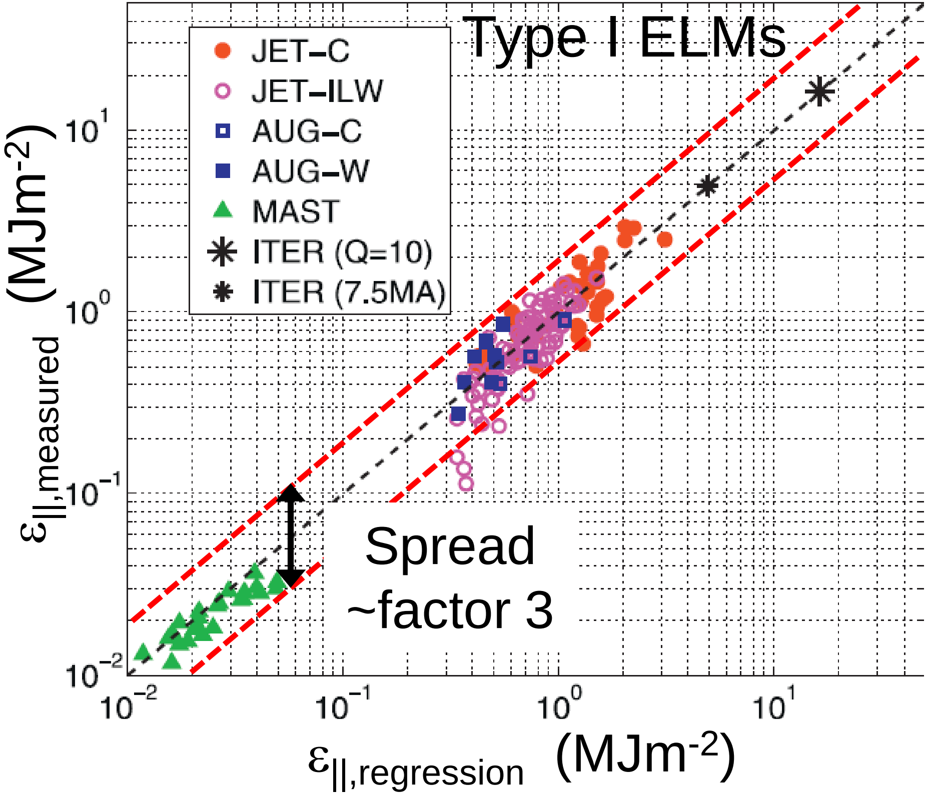
- Encouraging multi-device scaling for **outer target** peak parallel ELM energy density

$$\varepsilon_{||,scaling} = 0.28\frac{MJ}{m^2} n_{e,ped}^{0.75} T_{e,ped}^{1} \Delta W_{ELM}^{0.52} R^{1}$$

  - $\Delta W_{ELM} = W_{ELM}/W_{plasma}$
  - $n_{e,ped}$, $T_{e,ped}$ values of n_e and T_e at the top of the H-mode pedestal

ELMs – what if suppresion not possible?

T. Eich et al, NME **12** (2017) 84



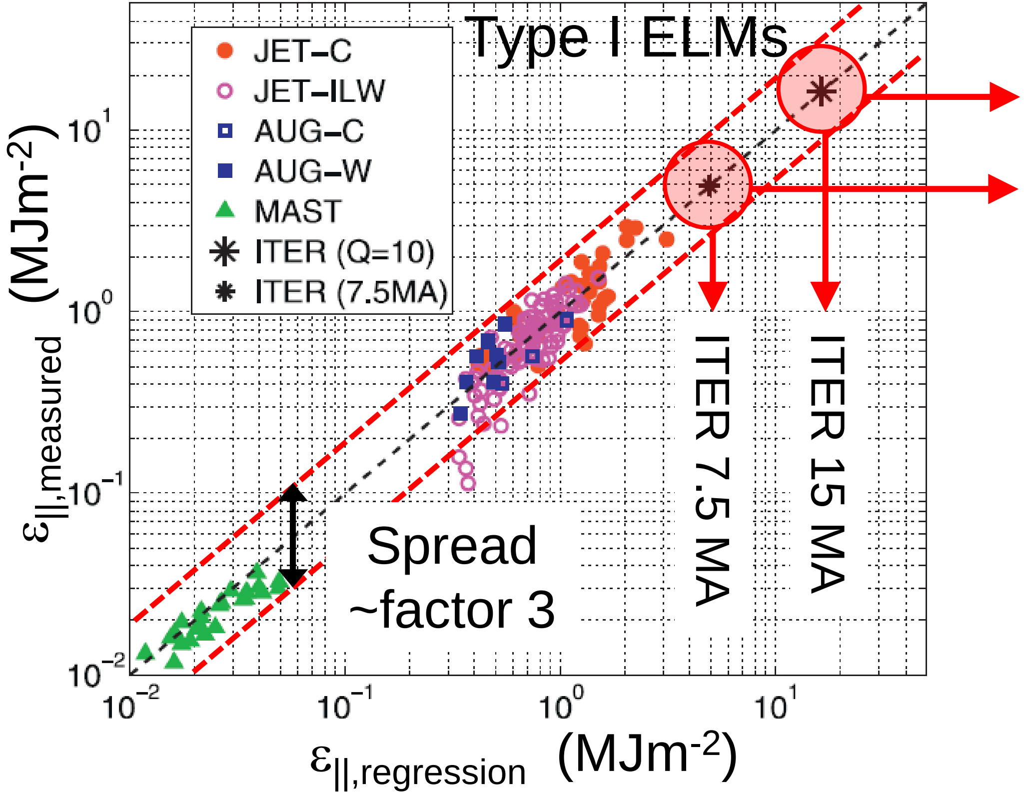
- Encouraging multi-device scaling for **outer target** peak parallel ELM energy density

$$\varepsilon_{||,scaling} = 0.28\frac{MJ}{m^2} n_{e,ped}^{0.75} T_{e,ped}^{1} \Delta W_{ELM}^{0.52} R^{1}$$

  - Parallel energy at targets dependent on pedestal top pressure and R
  - Favourable for ITER at Q_{DT} =10 compared to our previous scalings
  - Lower bound of data matched by simple model: $\varepsilon_{||} \approx 6\pi.p_e R q_{edge}$ (pedestal plasma connects to the targets during the ELM)

ELMs – what if suppresion not possible?

T. Eich et al, NME **12** (2017) 84

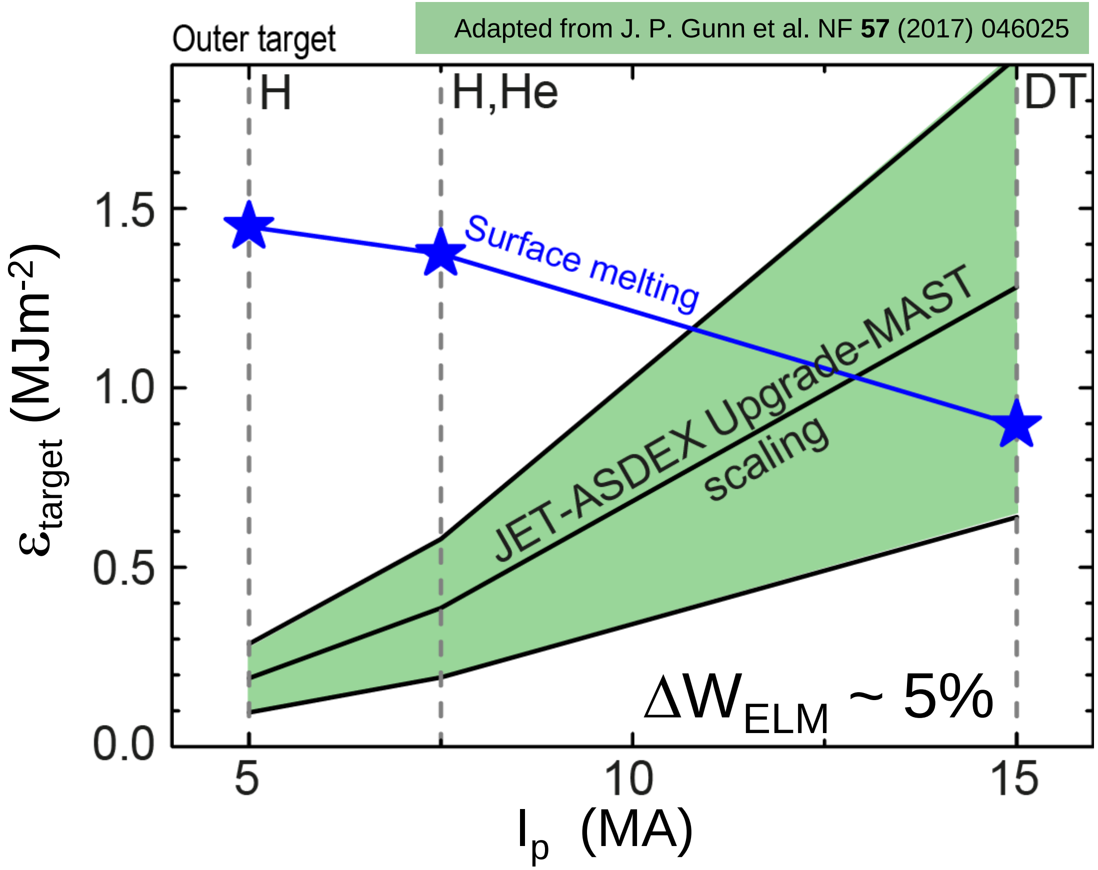


For $\Delta W_{ELM} \sim 5\%$

$\varepsilon_{target} \sim 0.60 – 1.80$ MJm^{-2}

$\varepsilon_{target} \sim 0.18 – 0.54$ MJm^{-2}

Natural Type I ELMs will still melt MB surface



Adapted from J. P. Gunn et al. NF **57** (2017) 046025

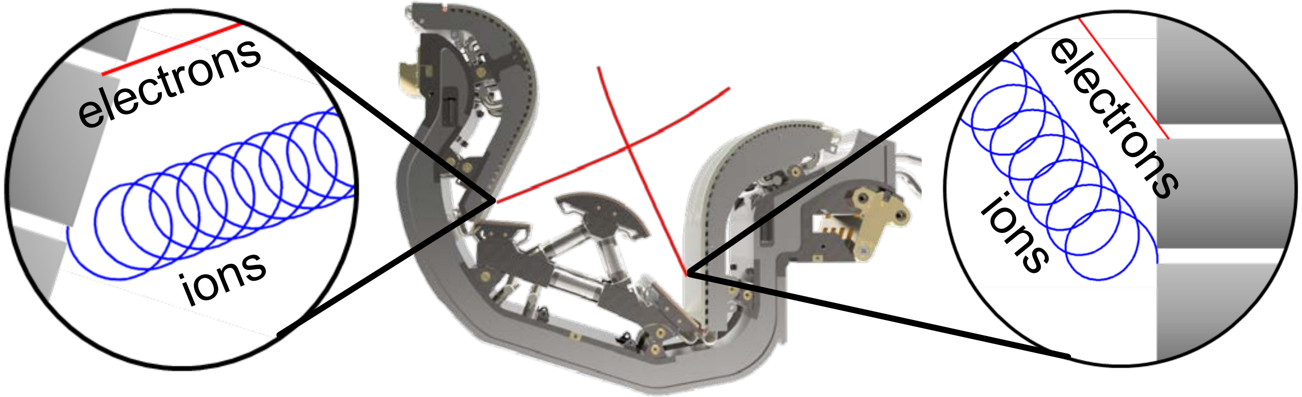
- Even with new scaling, monoblock surface melting will occur at $Q_{DT} = 10$
- Looks like reasonable margin to surface melting even for largest ΔW_{ELM}

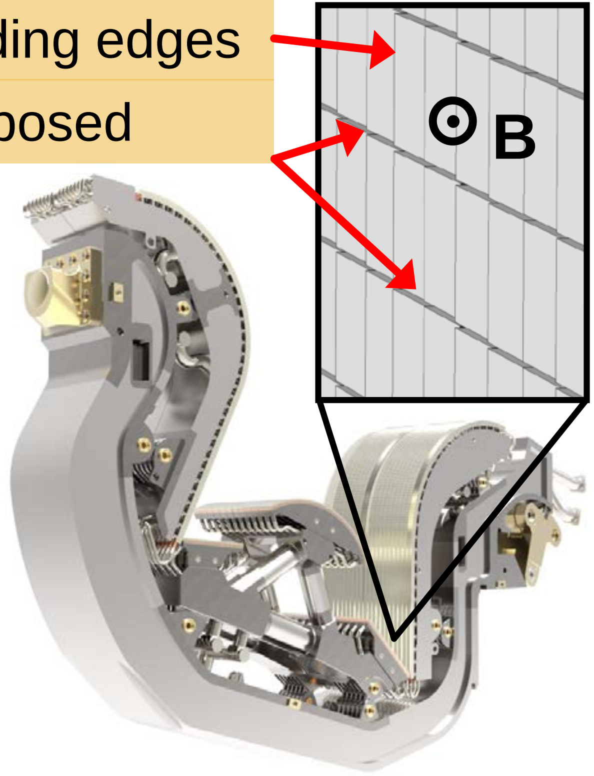
Problem of tile gaps

Toroidal bevel protects poloidal leading edges

BUT long toroidal edges are still exposed

- ELM ions problematic due to large Larmor radii of particles arriving from pedestal region

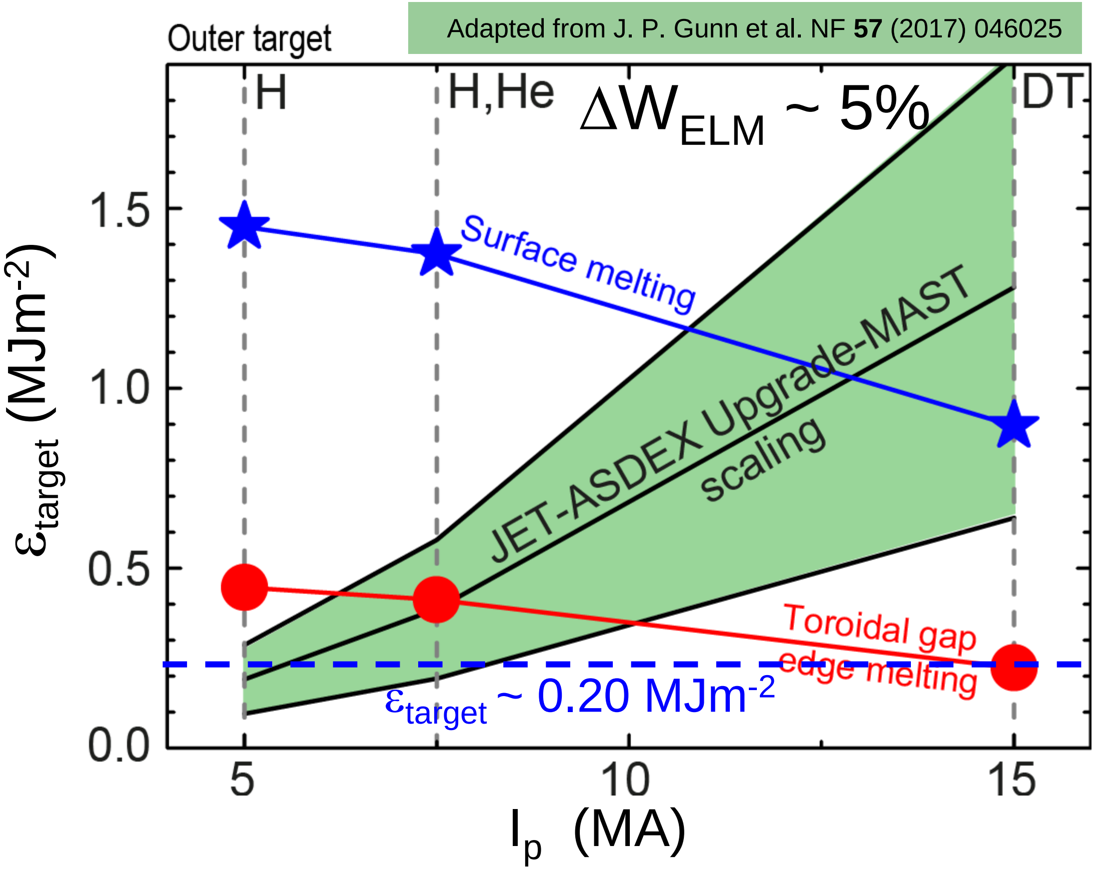




- Toroidal gap (TG) loading really does occur[1]
  - See talk by J. P. Gunn for more

[1]R. Dejarnac, Nucl. Fus. **58** (2018) 066003

Now add toroidal gap melting

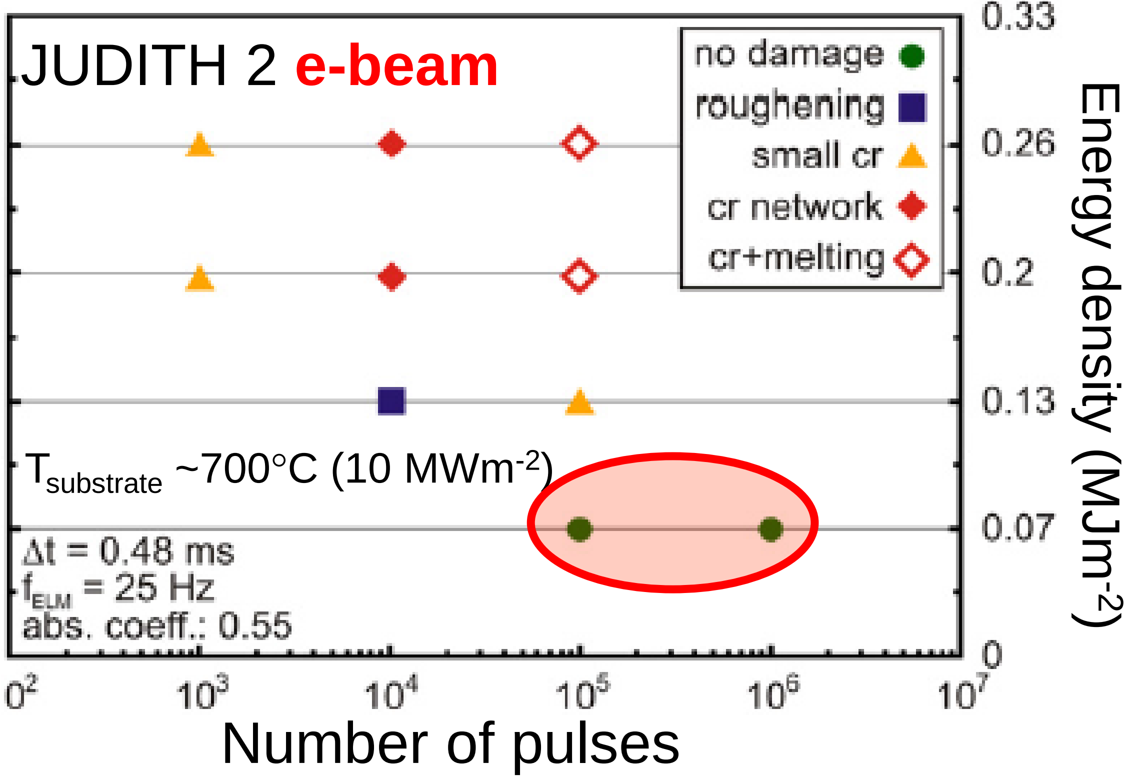


Adapted from J. P. Gunn et al. NF **57** (2017) 046025

- To avoid toroidal gap edge melting: $\varepsilon_{target} \lesssim 0.2$ MJm^{-2} → $\Delta W_{ELM}/W_{plasma} \lesssim 0.1\%$
  - Type I ELMs of this relative energy loss not found naturally …..

Surface cracking

T. Loewenhoff et al., Phys. Scr. **T145** (2011) 014057



- Frequent thermal cycling can lead to W surface micro-cracking
  - Threshold for zero damage formation $\lesssim$ 0.1 MJm^{-2} at high cycle number → similar to toroidal gap edge melting
  - Micro-cracks may be initiators for larger macro-cracks ….

ELMs: what to do for ITER?

We don’t know (yet) the consequences of repetitive ELM-induced monoblock toroidal gap melting

We don’t know (yet) the consequences for fatigue-induced surface cracking under simultaneous plasma exposure

We know that ΔW_{ELM} must be kept below ~1 MJ to avoid monoblock top surface melting for $Q_{DT} = 10$ (15 MA, 5.3 T) → ~0.3% of stored energy

Type I ELMs this small are not found naturally

So complete suppression is the only way to be sure.

But may also come with a price: see talks by Y. In, M. Fenstermacher, O. Schmitz

So that’s it, ITER wall and divertor (enjoy your lunch!)

but come back to listen to H. Zohm to how all this looks for the step beyond ITER. ..