# The first ITER tungsten divertor: operating space and lifetime

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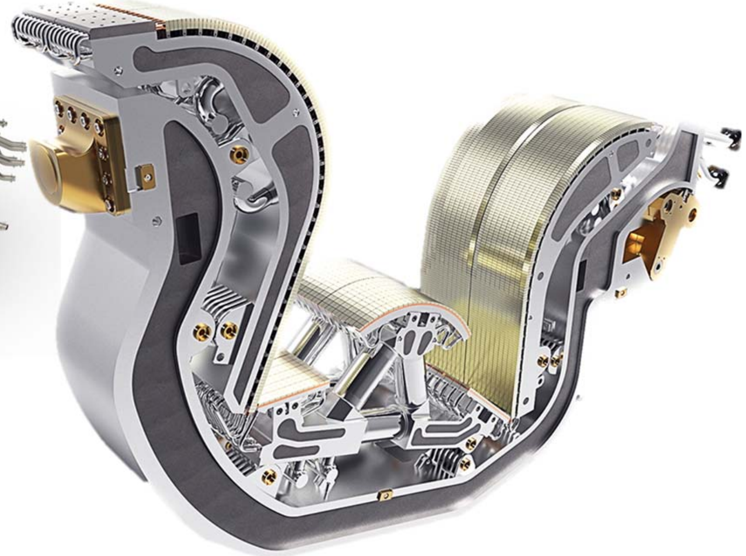
# Content

- 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 and definition of acceptable loads
  - Are scalings from simple models applicable?
- Summary

# Content

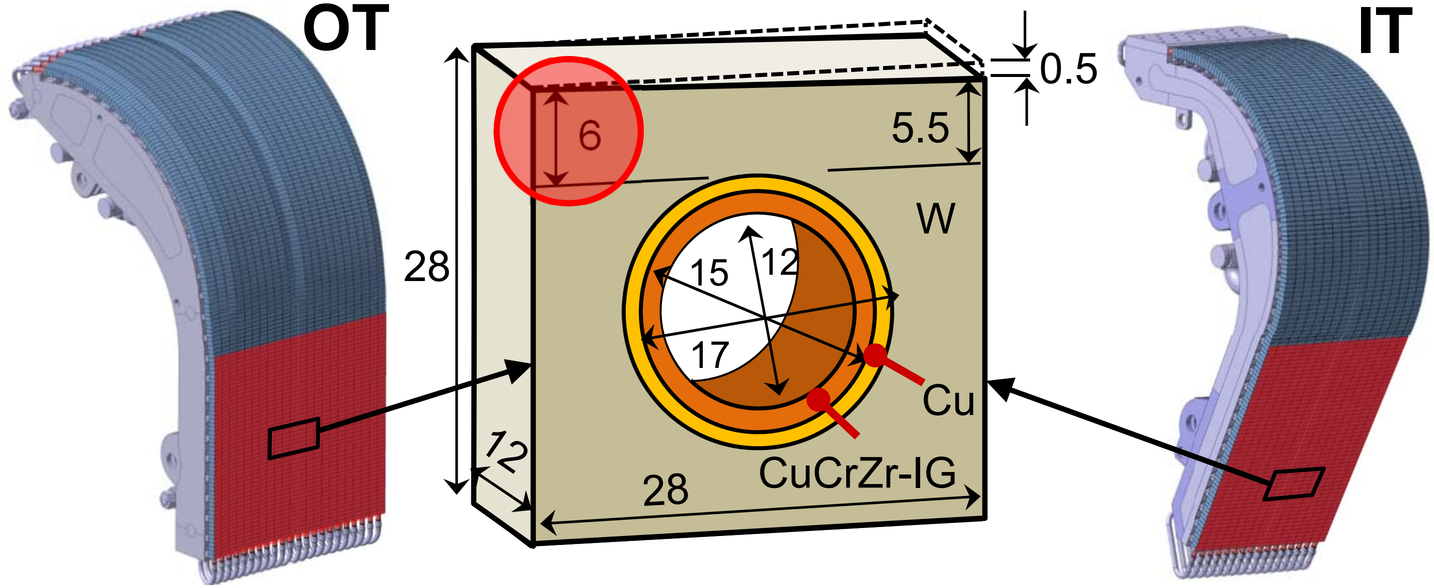
- Note 1:
  - 3-D fields aspects (ELM control) not considered here.
  - See talk (72) by H. Frerichs in 3-D fields session for ITER divertor modelling (Tuesday morning)
- Note 2:
  - Transients (ELMs) not discussed here (unless there is time)
- Note 2:
  - Much of the material in this talk can be found in the paper just published which accompanied the PSI-2018 review talk: R. A. Pitts et al., https://doi.org/10.1016/j.nme.2019.100696

# 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 complete

# Vertical targets and component shaping

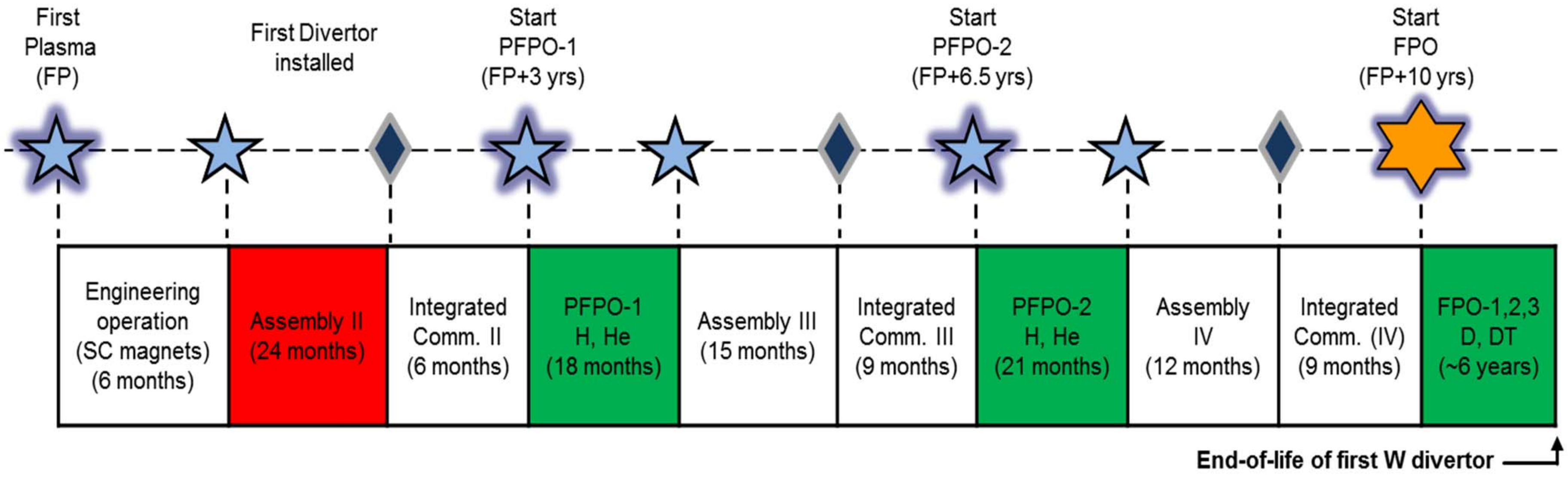



- Monoblocks in HHF areas will be toroidally bevelled to protect inter-PFU misalignments, vertical targets tilted to protect inter-cassette misalignments
  - Compromise between poloidal gap edge overheating and increased surface stationary loading

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

# Revised ITER schedule and divertor lifetime

- Divertor replacement currently foreseen in the ITER Research Plan at the end of the first D-T phases (3 campaigns, FPO-1,2,3) Plan
  - ~13 years after installation



# Updated ITER Research Plan

- Available publicly as ITER Technical Report (ITR-18-003)

- IRP informs the fusion community on details of experimental plans to achieve the Project goals and defines the required supporting R&D

- Expect ~900 days of D-T operation over ~5 years in FPO-1,2,3
  - ~12,000 pulses
  - **~8x10$^{6}$ s plasma time (~2200 hrs)**


ITER TECHNICAL REPORT

REPORT NO.
ITR-18-003

TITLE
ITER Research Plan within the Staged Approach (Level III – Provisional Version)

AUTHOR/AUTHORS
ITER Organisation

AUTHOR EMAIL(S)
ITR.support@iter.org

DATE
17th September 2018

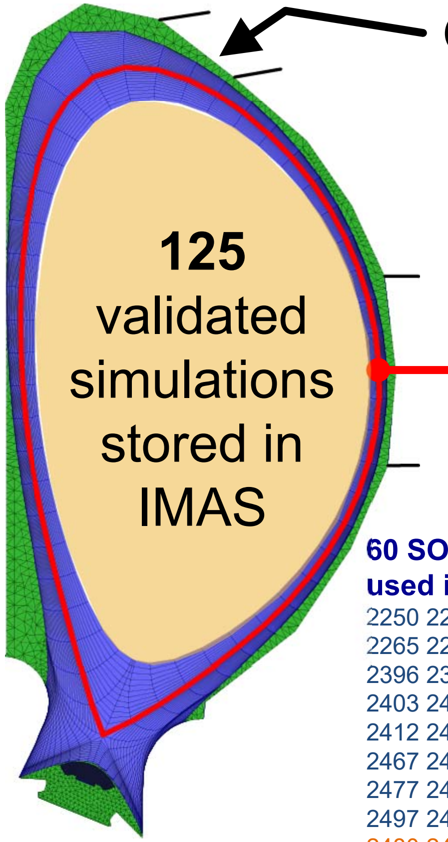
www.iter.org

# Burning plasma operating window

- Focus on "burning plasma" conditions → the most challenging for the ITER divertor
  - $Q_{DT}$ = 10, $P_{IN}$ ~ 100 MW
  - Ne and N seeding (emphasis on Ne where database currently largest)
  - No discussion of "integrated modelling" here
  - Divertor simulation database largely constructed with SOLPS-4.3, with more recent analysis using SOLPS-ITER
- 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 (i.e. not "DEMO-like")

# 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
  - $D_\perp = 0.3\ m^2s^{-1}$, $\chi_\perp = 1.0\ m^2s^{-1}$
- Scans in fueling, seed impurity, power into numerical grid ($P_{IN}$)
- All-metal walls
  - Assume Be everywhere, but no sputtering

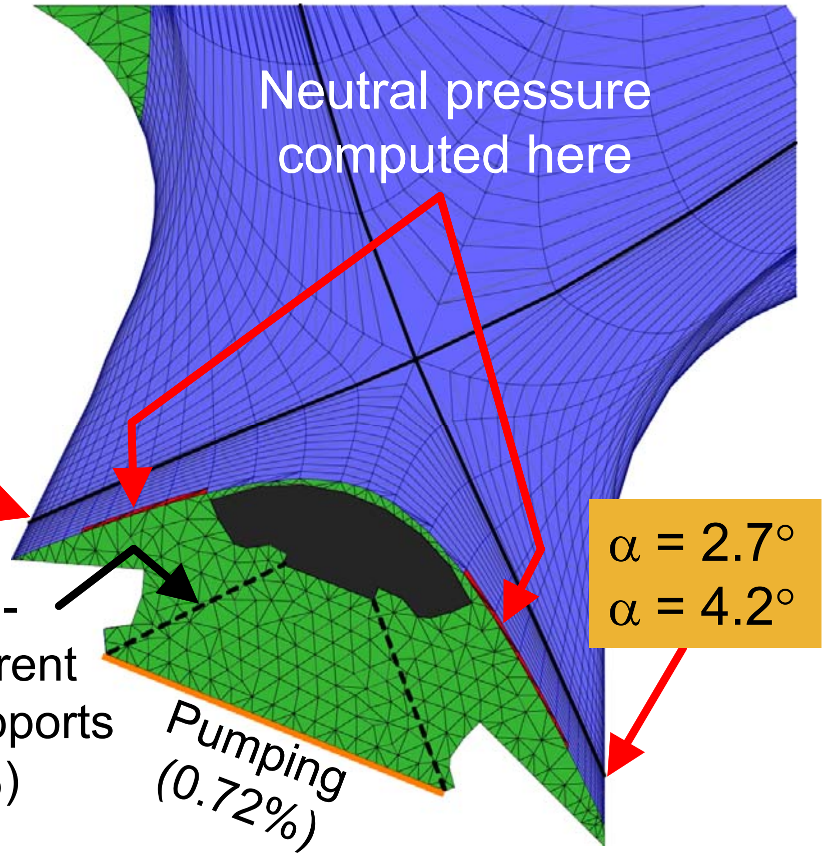



$\langle c_Z \rangle_{sep} = n_Z/n_e$
averaged around first ring outside separatrix

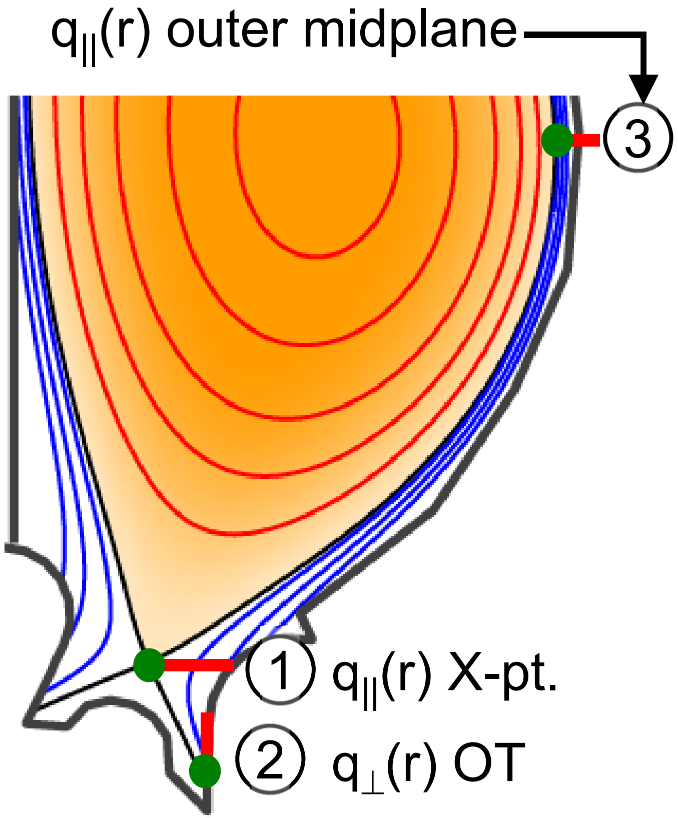
**60 SOLPS-4.3 case numbers used in this talk:**
2250 2251 2252 2253 2257 2258 2264
2265 2266 2269 2316 2317 2332 2333
2396 2397 2398 2399 2400 2401 2402
2403 2404 2407 2408 2409 2410 2411
2412 2413 2414 2415 2416 2439 2463
2467 2468 2469 2470 2471 2472 2476
2477 2478 2481 2483 2484 2485 2496
2497 2498 2508 2509 (Ne)
2480 2493 2494 2502 2503 2533 2534 (N)

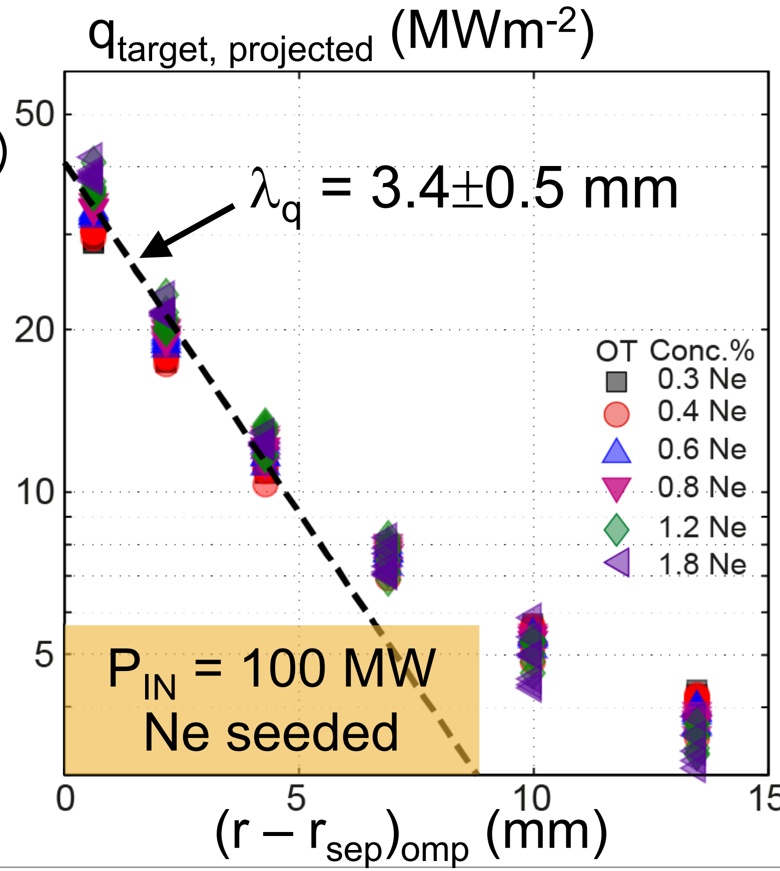
# Main simulation database parameters

- Seek to map out operational space in two key parameters:
  - Divertor neutral pressure
  - Peak divertor target power loading



# SOL heat flux width

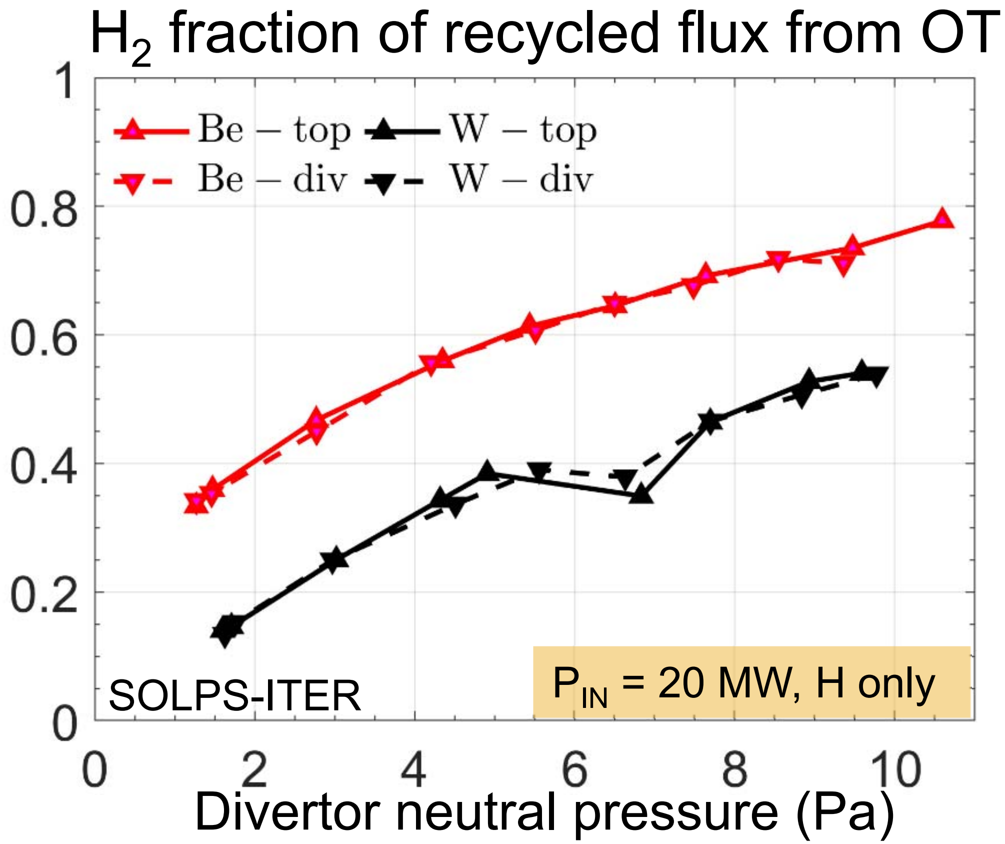



- Divertor conditions across database do not strongly influence upstream $\lambda_q$
- See talks by C.-S. Chang (66) and X. Xu (16) for more on $\lambda_q$ scaling

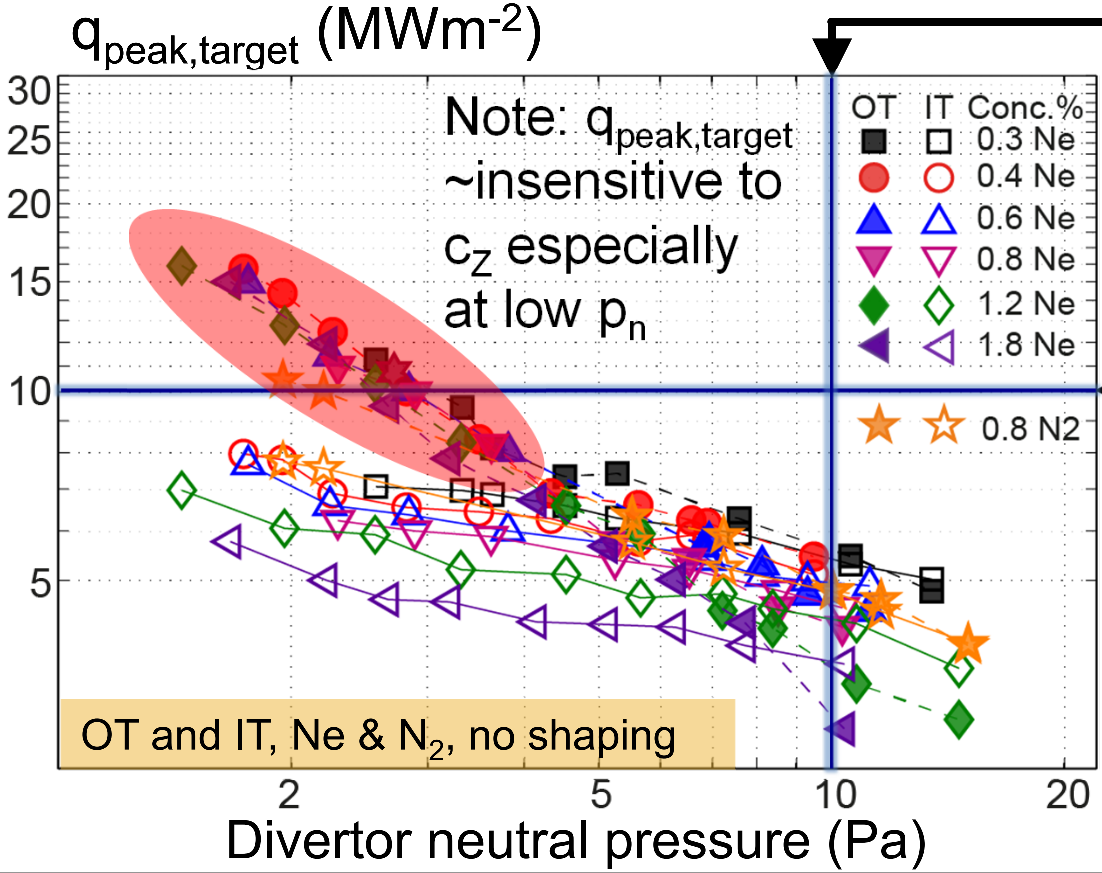
# Sensitivity to material?

- Target material decides ratio of reflected atoms/molecules
- More molecules from Be target, higher fraction of fast reflected atoms from W
- The overall effect of the two populations is to produce almost the same momentum and power losses




J. S. Park et al, APS (2019)

# Operating window in peak power flux density

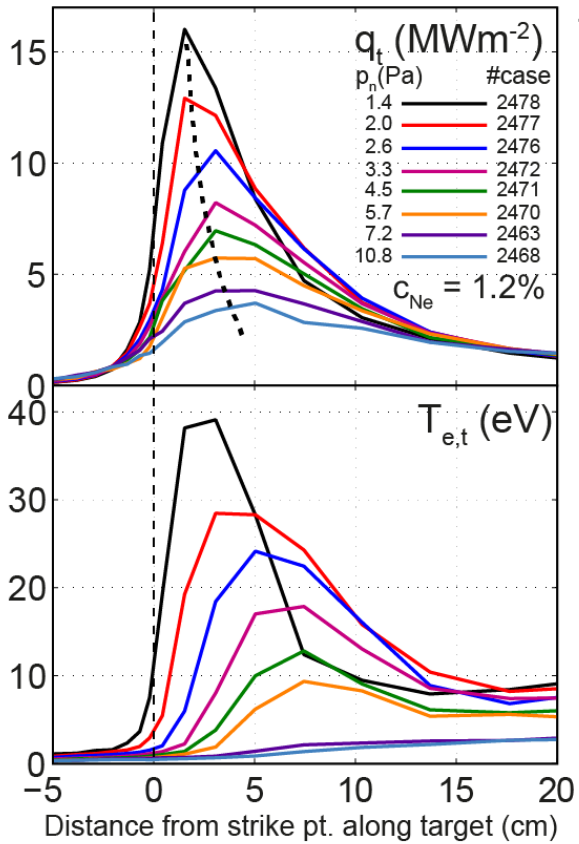


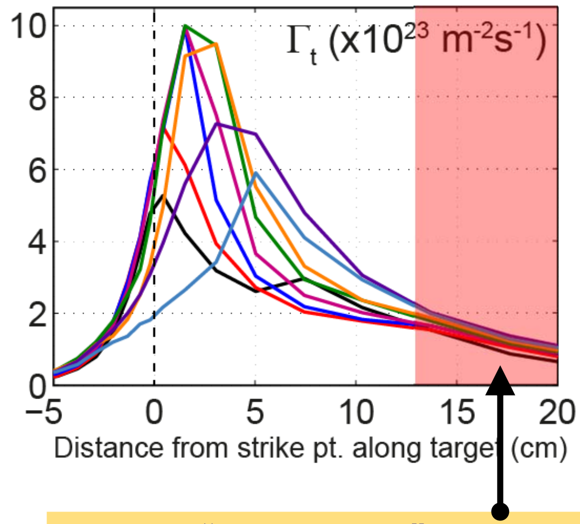

Approximate "Detachment limit"

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

- Out-in peak power asymmetry reduces in the code at high $p_n$
- $N_2$ and Ne behave similarly
- Need 3-5x N than Ne in the code for given fueling to get same $<c_Z>_{sep}$

# Detachment evolution

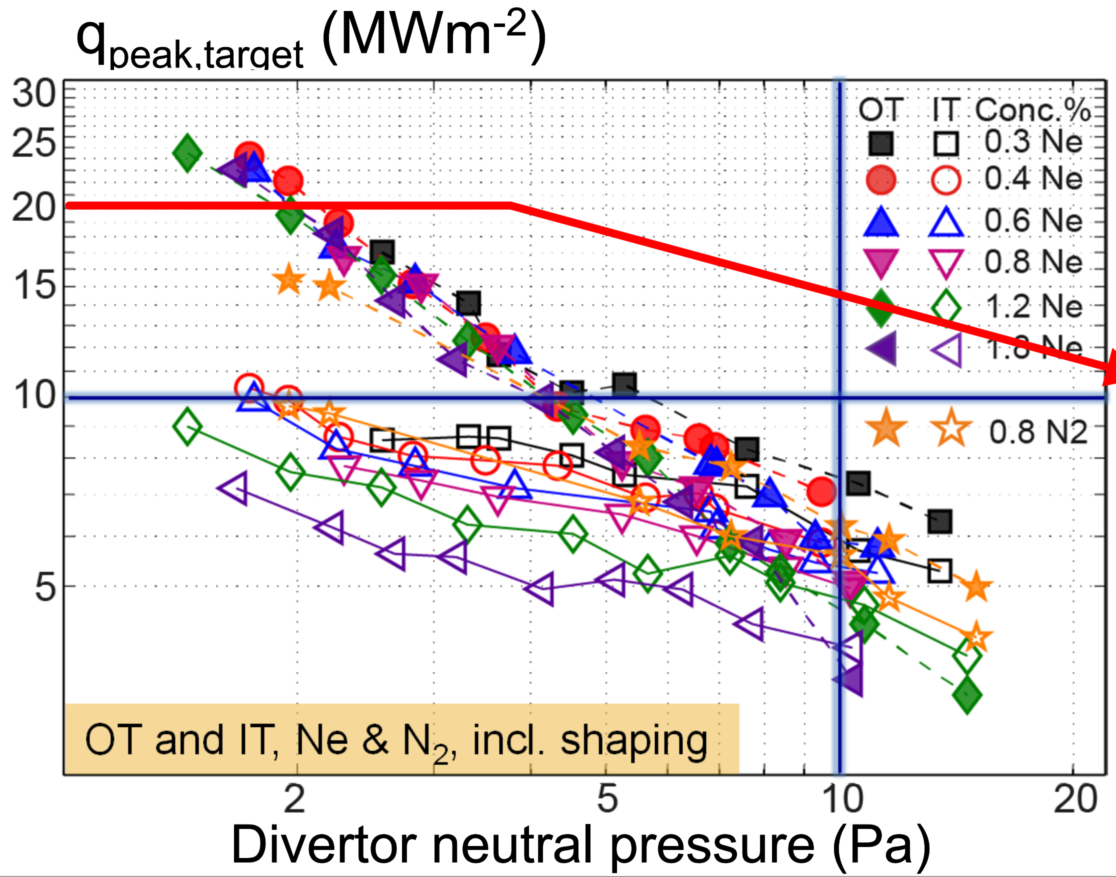



Avoid "complete" detachment → keep finite ion flux in outer part of the SOL to maintain sufficient neutral plugging

- "Classic" evolution from high recycling to partially detached state
  - He pumping improves with increased $p_n$ but not if far-SOL also detached

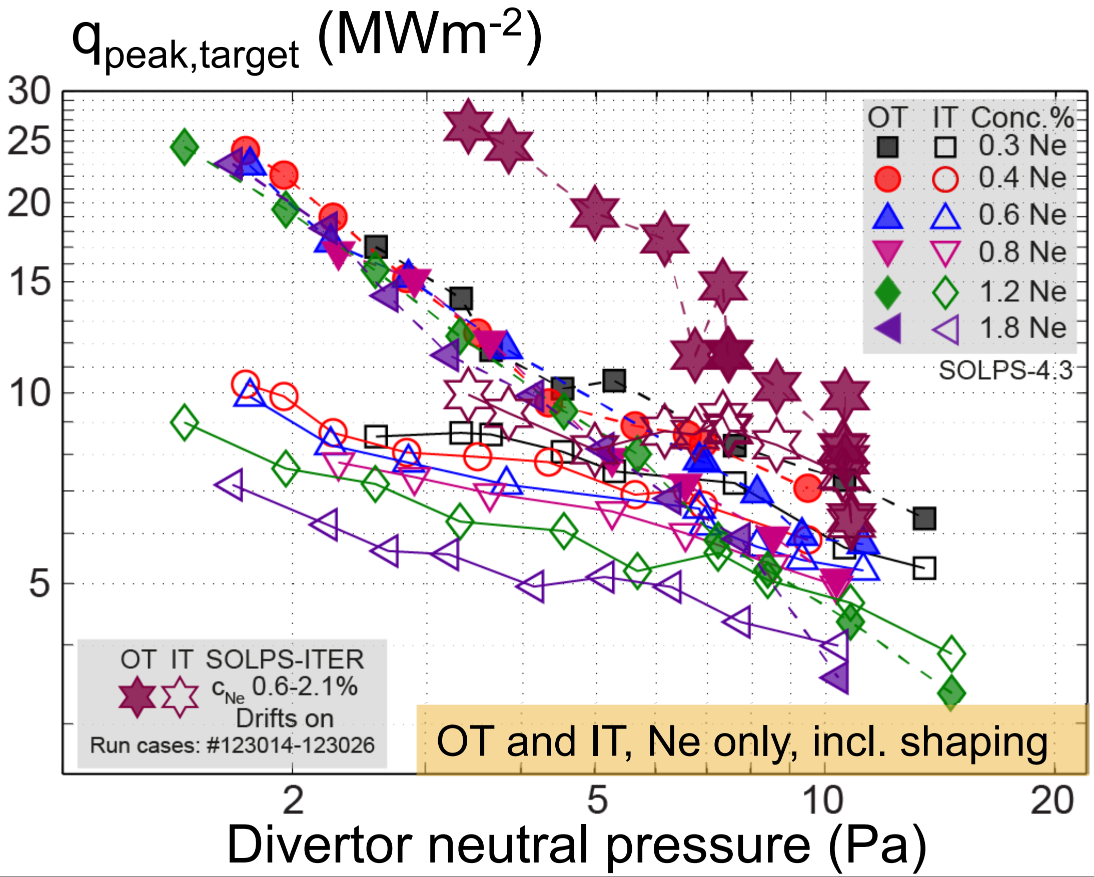
# Now add shaping




- Effects less marked at high $p_n$ where thermal plasma contributions lower
- 20 $MWm^{-2}$: CHF limit with factor 1.4 margin
  - Recently updated* after review of MB CHF test protocol and new dedicated tests now that final thickness decided

F. Escourbiac et al, FED **146** (2019) 2036

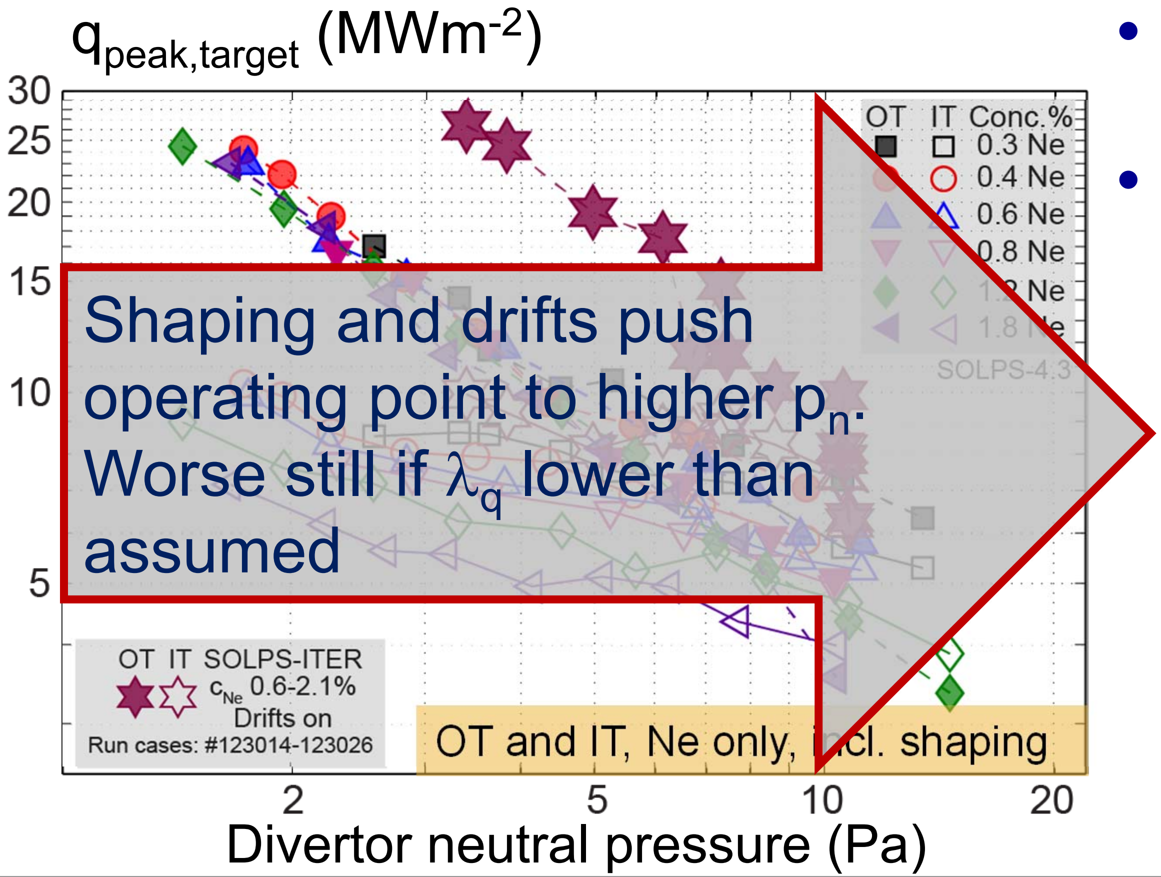
# Impact of drifts




- "H-mode" SOLPS-ITER drift modelling*
- Strong impact on OT loading at low $p_n$ but effect reduced as detachment deepens
  - Drifts increase characteristic pressure at which OT reattachment occurs due to increasing Ne leakage
  - Drifts increase need for good detachment control

*E. Kaveeva et al., submitted to NF

# Impact of drifts

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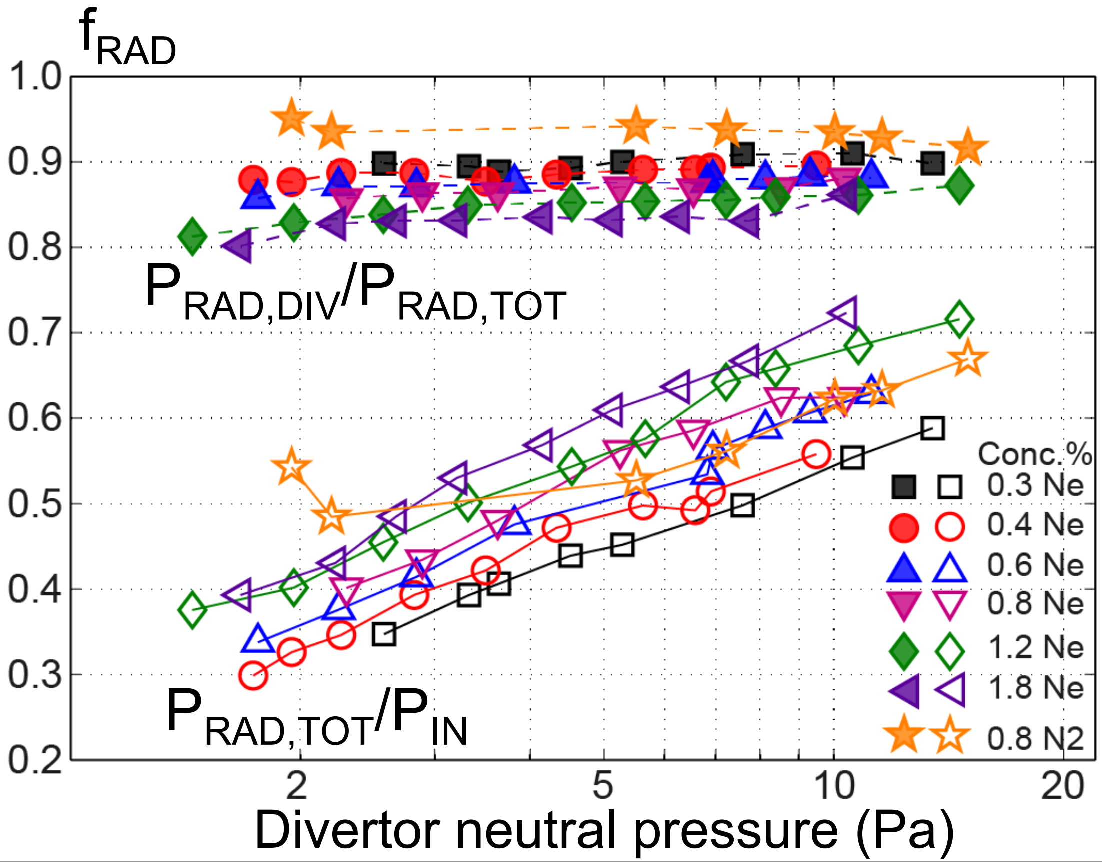
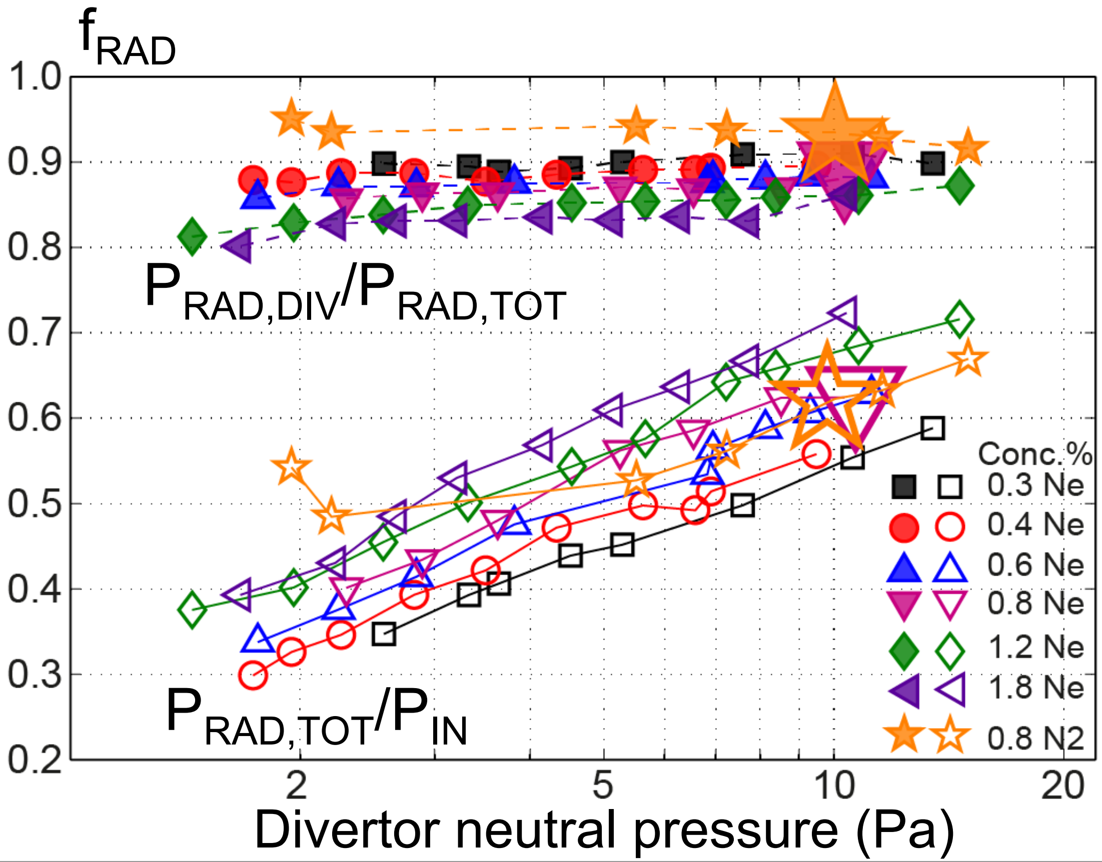
# Radiated fractions




- 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
  - N radiates more efficiently in the divertor than Ne for same $c_Z$
  - Lower core radiation with N

# Radiated fractions

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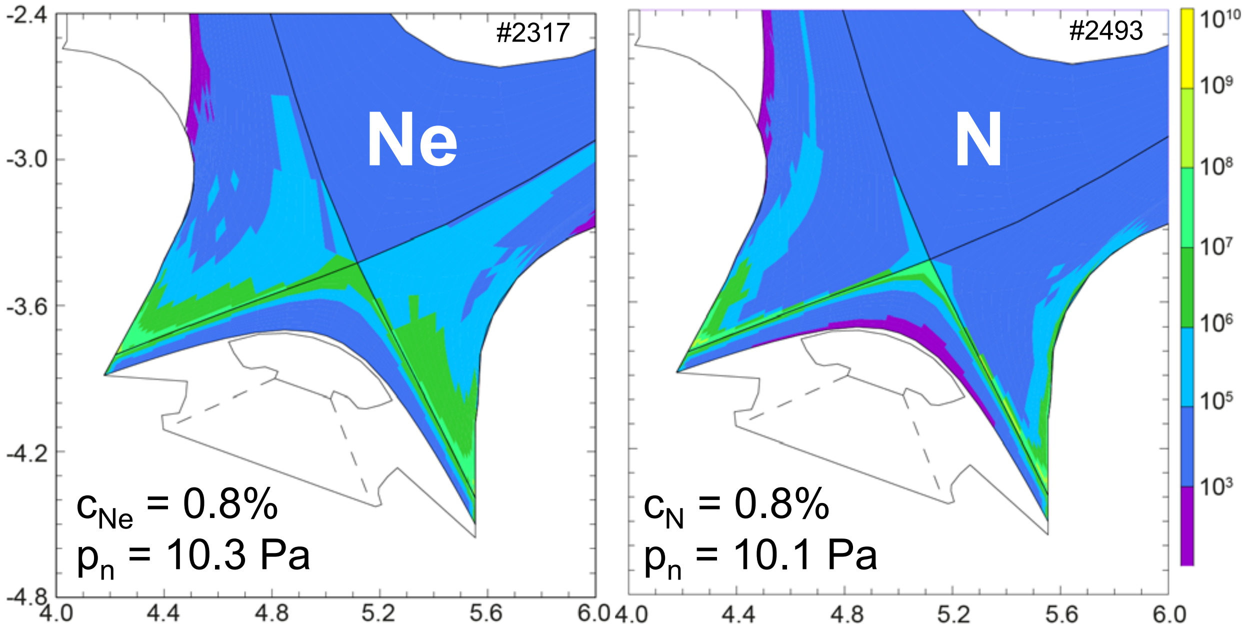
# Divertor radiation distribution

Total radiated power ($Wm^{-3}$)



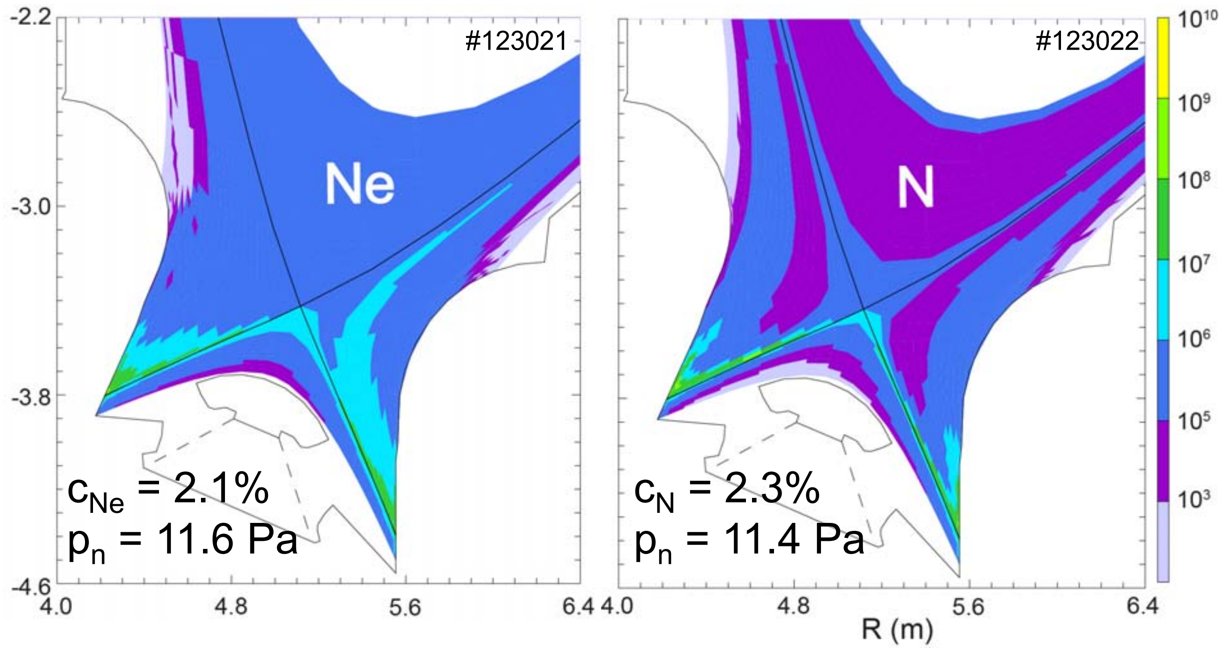

$P_{RAD,DIV}$ = 56.6 MW
41.3 (Ne) + 15.3 (D)

$P_{RAD,DIV}$ = 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

# Negligible drift impact at high $p_n$

Total radiated power ($Wm^{-3}$)



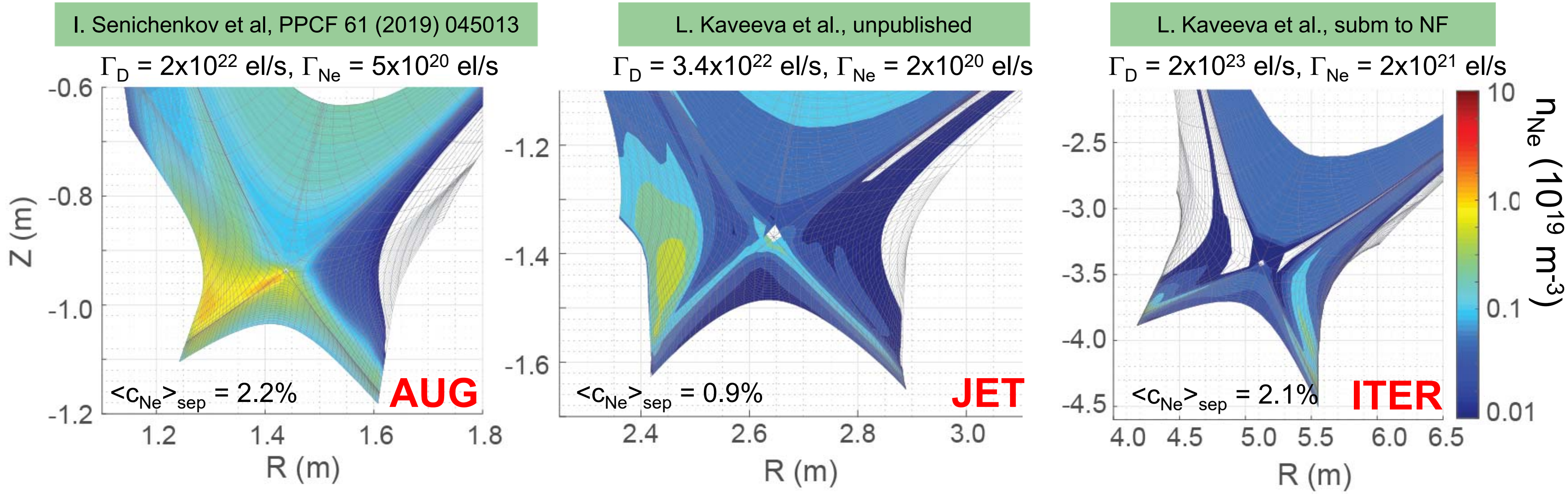

- SOLPS-ITER with drifts activated
  - $P_{IN}$ = 100 MW
  - Matched Ne, N cases
  - H-mode pedestal

$P_{RAD,DIV}$ = 53.4 MW
38.7 (Ne) + 14.2 (D)

$P_{RAD,DIV}$ = 54.0 MW
44.7 (Ne) + 9.2 (D)

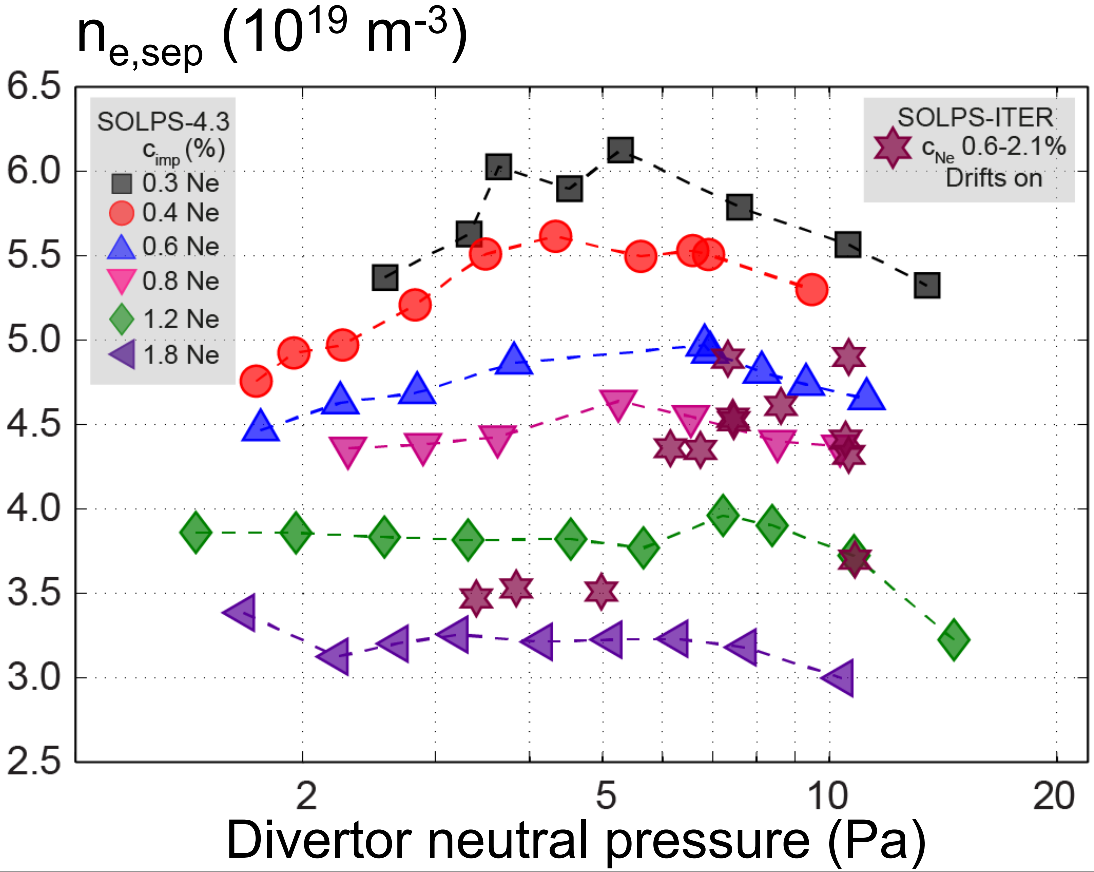
E. Sytova et al, NME 19 (2019) 72
E. Kaveeva et al., submitted to NF

# Impact of scale size?



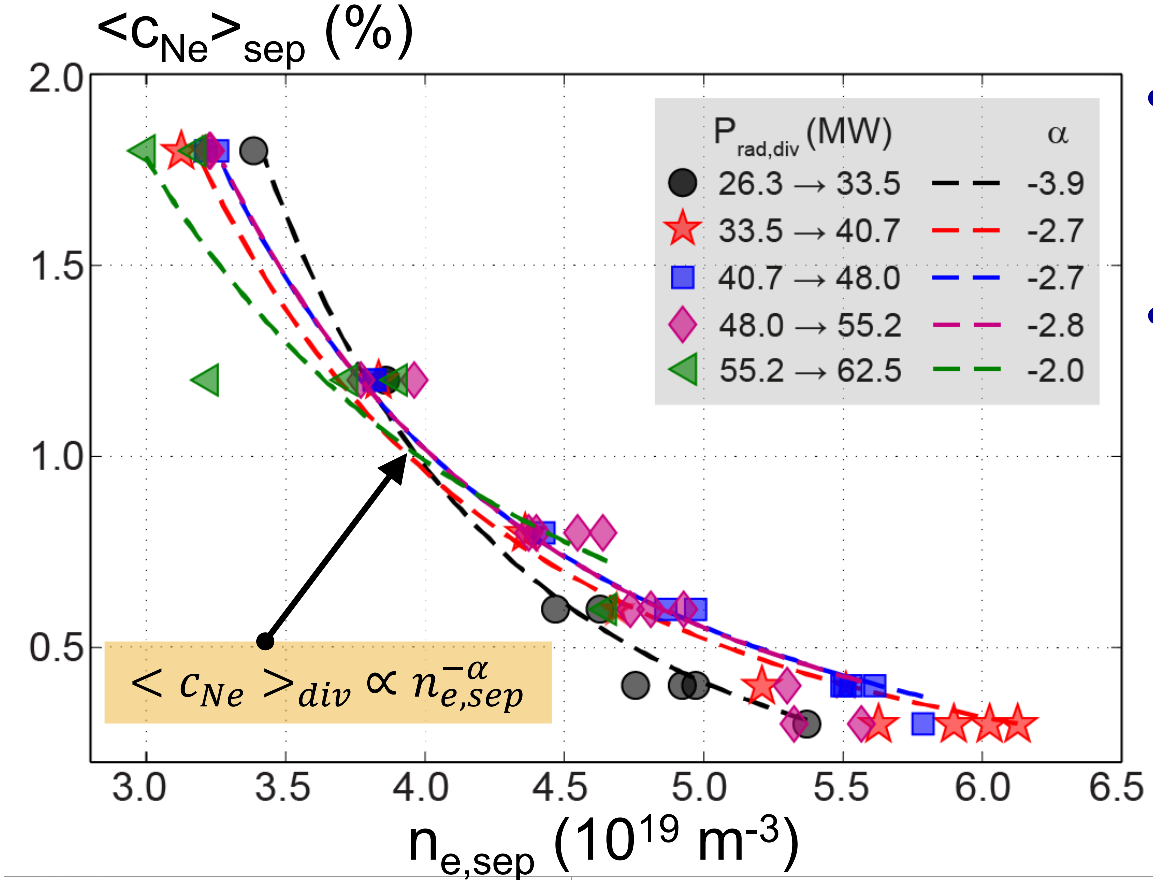

- Now have a set of SOLPS-ITER detached "H-mode" drift simulations with Ne-seeding across a factor 3 in machine size → we are in position to analyse the impact of scale
  - There appears to be a gradual evolution from stronger to weaker drift effect and weaker to stronger impurity retention with increasing size → Ne leakage still occurs in ITER, but cannot radiate in the edge/pedestal because too high $T_e$.

# Upstream density dependence



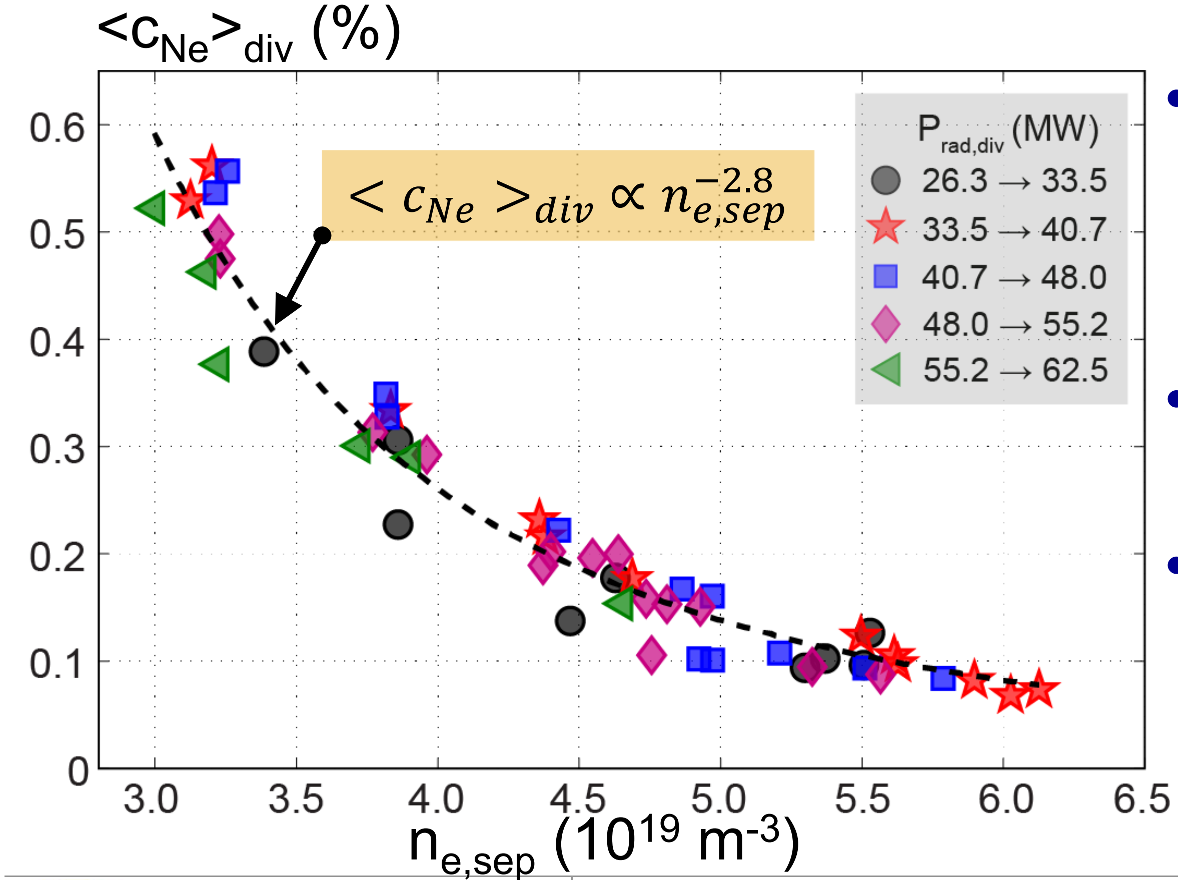

- Little or no dependence of $n_{e,sep}$ on $p_n$.
- Strong dependence on $c_{Ne}$ at fixed $p_n$.
  - $c_{Ne}$ ↑ → $P_{rad,div}$ ↑ → power available for dissociation/ ionization/excitation of fuel molecules/atoms decreases → $n_{e,sep}$ ↓
- Low $c_{Ne}$ favourable for $Q_{DT}$ = 10, but higher $c_{Ne}$ for higher $P_{rad,div}$, so compromise required (especially if $\lambda_q$ low)

# Density dependence of $c_Z$ (I)



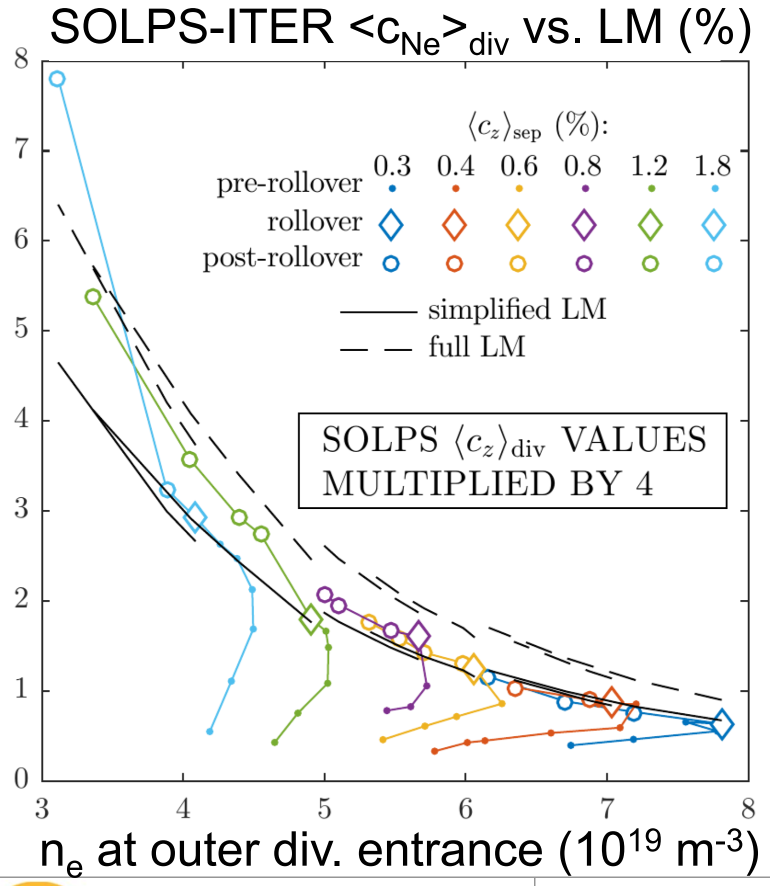

- Dependence ranges from $n_{e,sep}^{-4} \rightarrow n_{e,sep}^{-2}$ for points at low (high) $P_{rad,div}$
- Stronger than predicted by simple models (e.g. Lengyel) at lower $P_{rad,div}$

# Density dependence of $c_Z$ (II)




- $\langle c_{Ne} \rangle_{div}$ averaged over first SOL ring outside the separatrix below the X-point at the outer target
- Overall dependence
  $$< c_{Ne} >_{div} \sim n_{e,sep}^{-3}$$
- Similar to that found by S. Henderson on AUG with N-seeding close to detachment (talk 43)

# Comparison with Lengyel



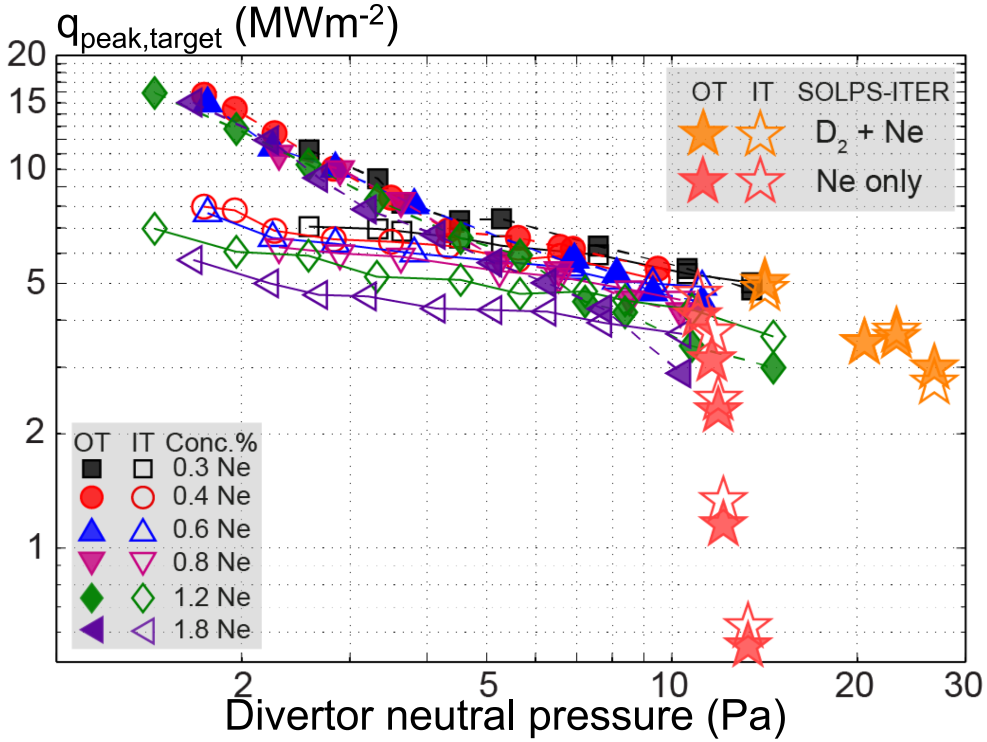

## Simplified LM ($T_{e,t} = q_{||,t} = 0$)

$$c_z = \frac{q_{\|u}^2}{2\kappa_{e\|0} n_{eu}^2 T_{eu}^2 \int_0^u L_z^{\mathrm{ADAS}} \sqrt{T_e} dT_e} \text{ with } T_{eu} = T_{eu}^{\mathrm{2PM}} = \frac{7}{2}\left(\frac{q_{\|u} L_{\|}}{\kappa_{e\|0}}\right)^{\frac{2}{7}}$$

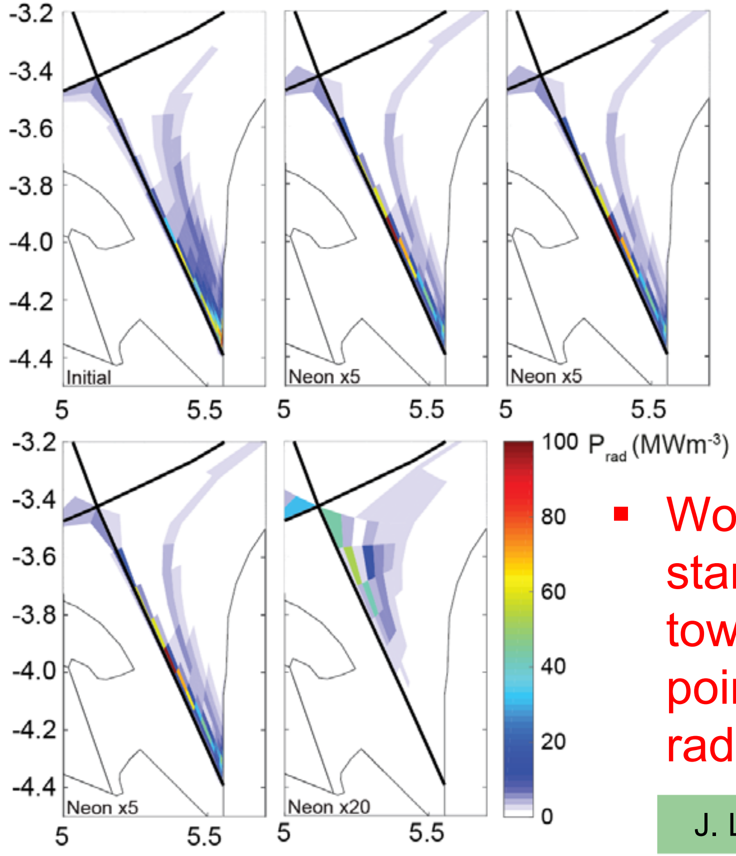
- Focus on region around rollover at each $c_Z$, and 3rd SOL ring outside separatrix $(r-r_{sep})_{OMP} \sim \lambda_q$
- Remarkably good agreement with trend
- x4 higher $c_Z$ predicted by LM due to:
  - Additional heat losses (radial transport and neutrals) and heat flux channels (convection, ion conduction) in SOLPS

D. Moulton et al., to be submitted to NF

# Avenues for reducing $q_{peak,target}$ (I)



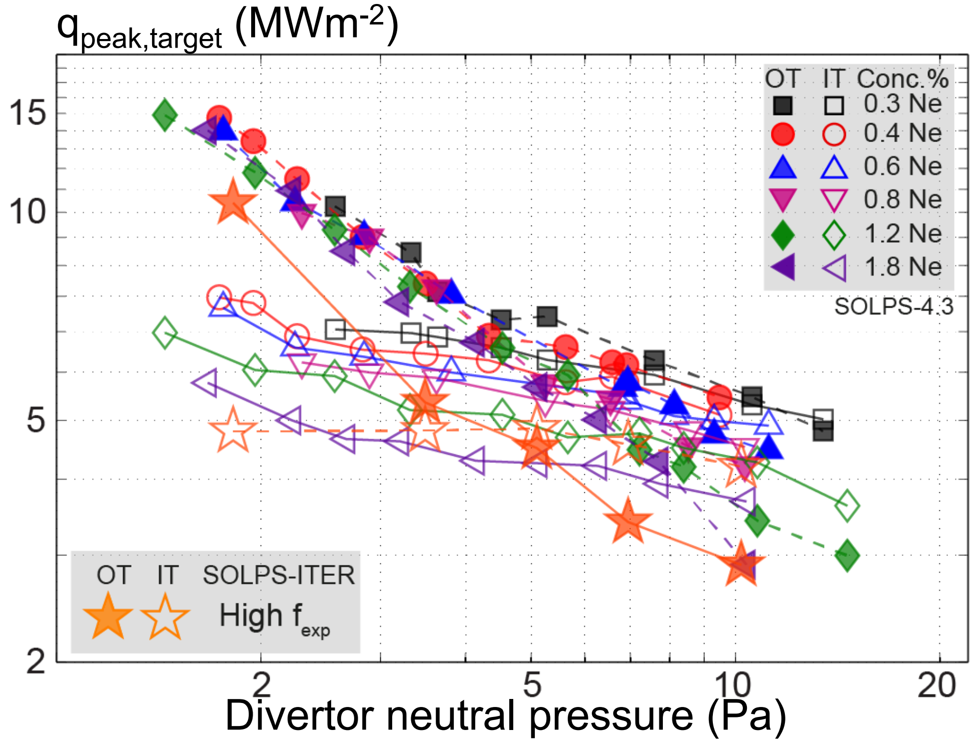

- SOLPS-ITER to push the detachment level beyond the SOLPS-4.3 database

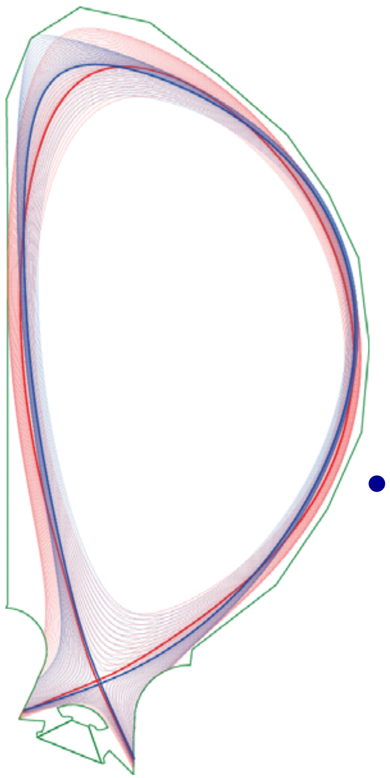


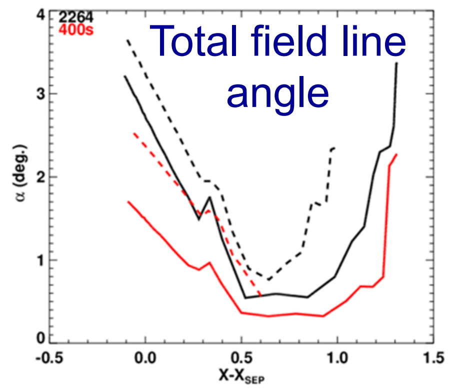

- Work just starting → towards X-point radiator

J. Lore, ORNL

# Avenues for reducing $q_{peak,target}$ (II)





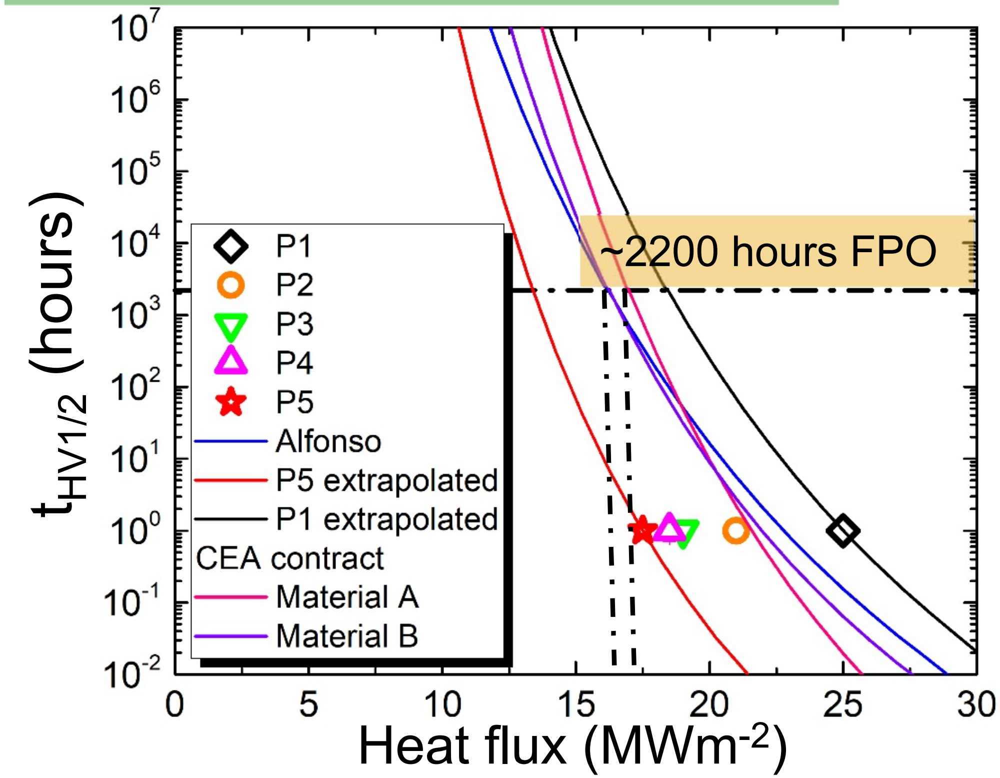

- SOLPS-ITER to study expanded (poloidal) configurations

- Magnitude of $q_{peak,target}$ reduction is roughly consistent with reduced angle of incidence

J. Canik et al., APS 2019

# “Lifetime” power density limit?

G. De Temmerman et al, PPCF, 60 (2018), ICFRM 2019
P1-P5, from S. Panayotis, NME 12 (2017)




- Define “Operational budget” for $q_{peak,target}$ in terms of time required for W hardness to drop by 50% at 2 mm depth below MB surface
  - ~2 mm recrystallization depth consistent with recent FEM modelling for crack onset due to low cycle fatigue
  - Two new curves from dedicated studies under ITER contract added since PSI-2018

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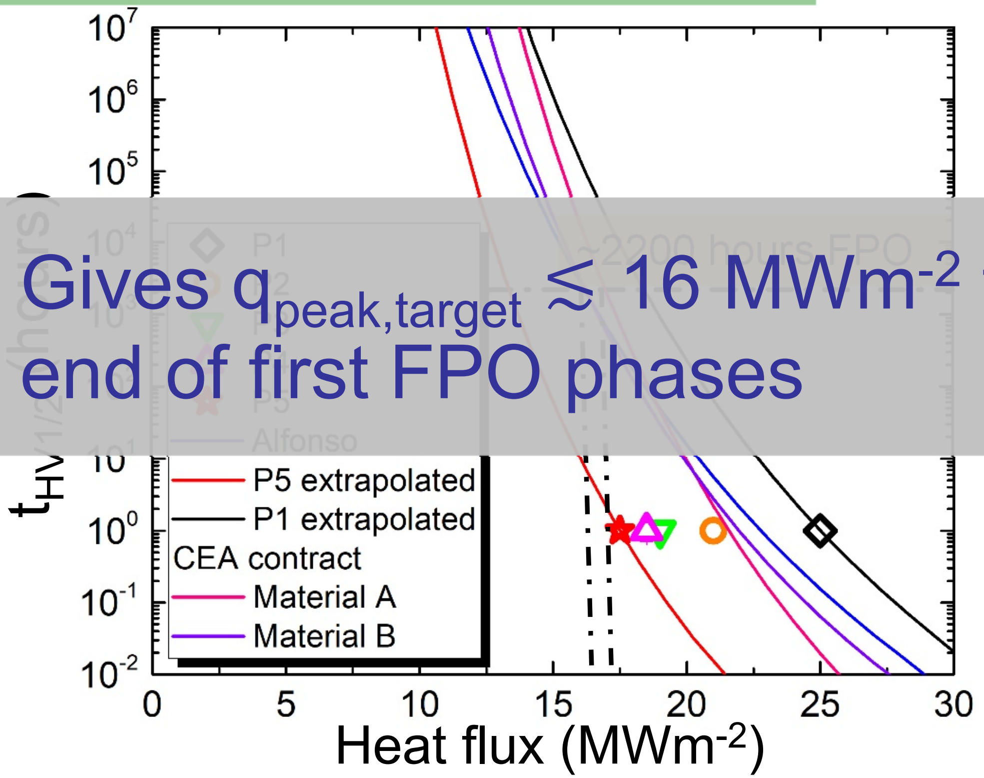



- Define “Operational budget” for $q_{peak,target}$ in terms of time required for W hardness to drop by 50% at 2 mm depth below MB surface
  - ~2 mm recrystallization depth consistent with recent FEM modelling for crack onset due to low cycle fatigue
  - Two new curves from dedicated studies under ITER contract added since PSI-2018

Gives $q_{peak,target} \lesssim 16$ $MWm^{-2}$ for first ITER divertor to end of first FPO phases

# Summary

- The ITER W divertor design is now complete
  - Prototyping of all major components at an advanced stage
- Divertor burning plasma operating window well established
  - Impurity seeded, partially detached operation with radiation well confined to the divertor - N or Ne seem acceptable
  - But target shaping, drifts, possible narrow $\lambda_q$ all push window to higher divertor neutral pressure
  - Increased peak power handling capacity (based on W recrystallization threshold) adds some margin
- ELM suppression remains the objective
  - If ELMs are to be allowed, they have to be extremely small

# Still much to do for ITER

- The ITER divertor is being procured and the design cannot now change.
- But a lot of R&D still required in the years before the divertor is first used
  - See Page 23 of https://doi.org/10.1016/j.nme.2019.100696

# Thank you

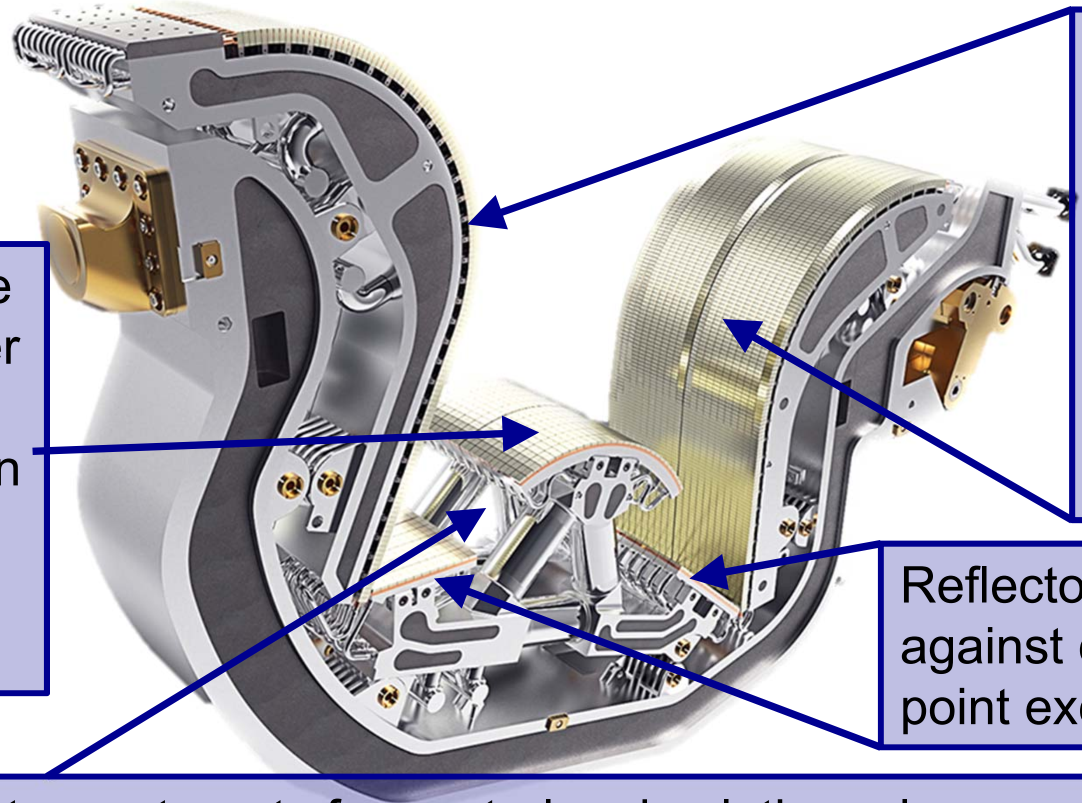
**ITER construction site 24/10/2019**

# Reserve slides

# W divertor: key physics characteristics

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

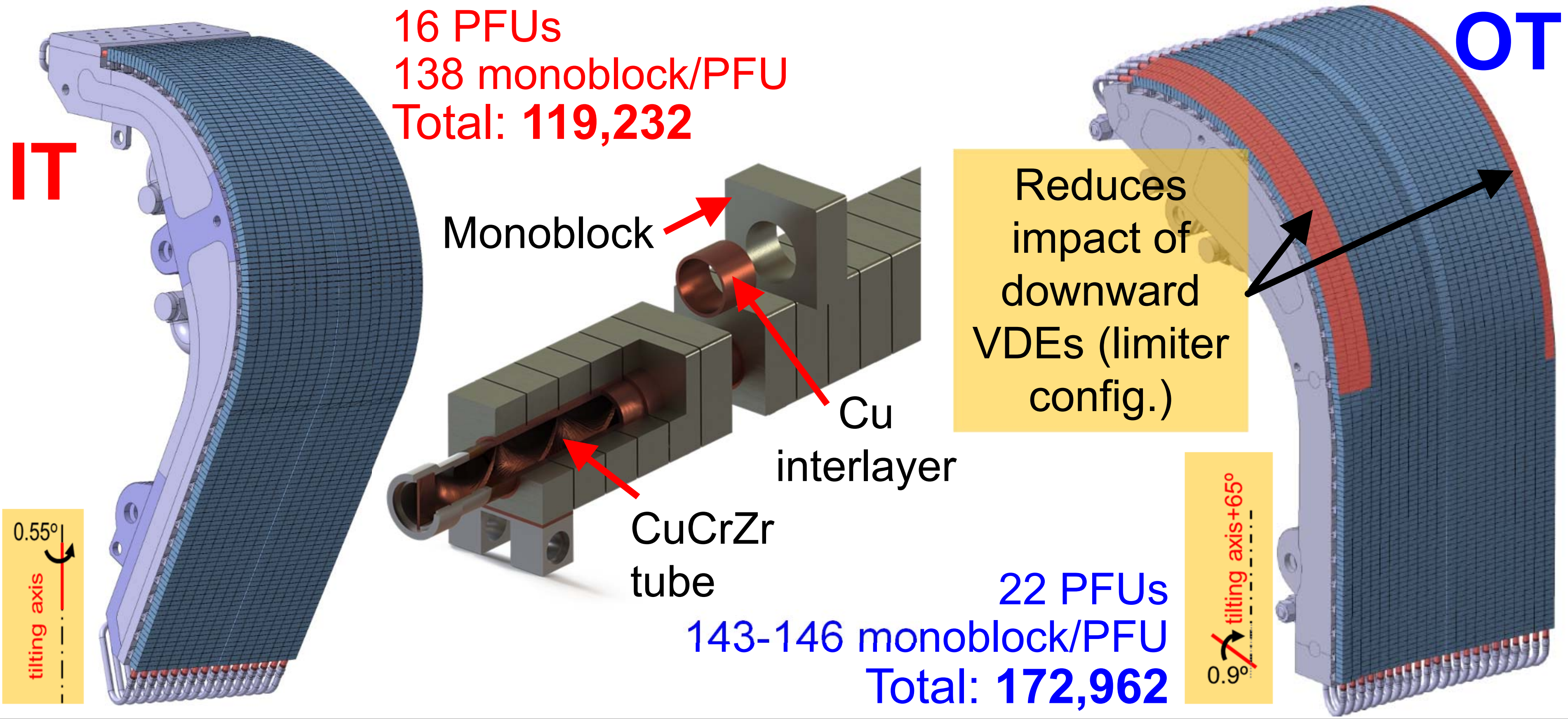
Dome to improve pumping → lower pumping speed required for given upstream He conc or fuel throughput



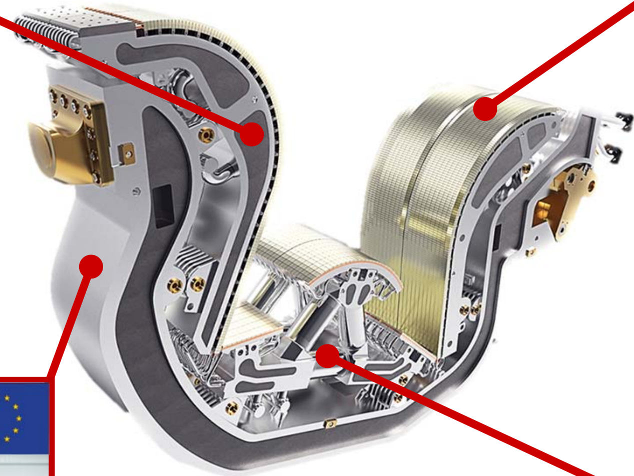
Reflector plates protect against downward strike point excursions

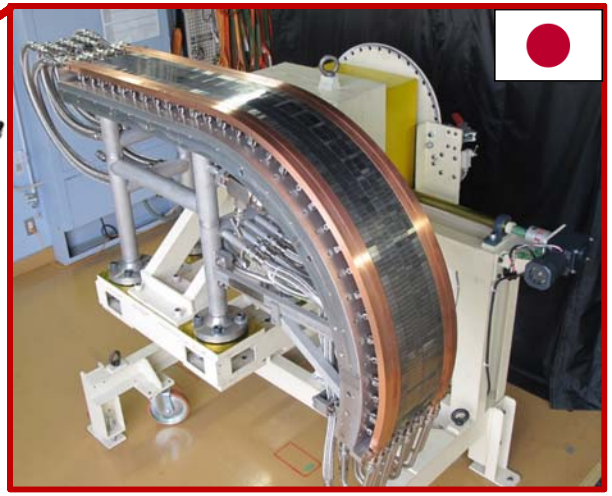
Transparency between targets for neutral recirculation – lower power asymmetries

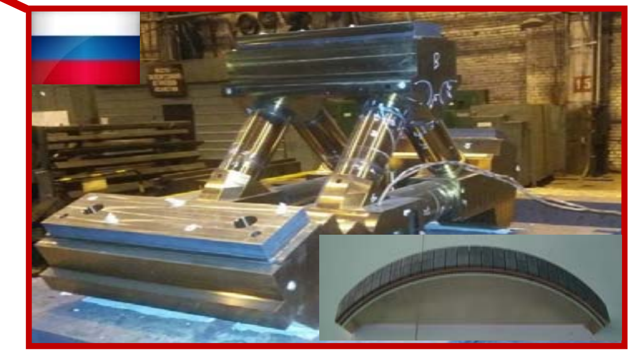
# Vertical targets and component shaping



# Progress in manufacture/prototyping



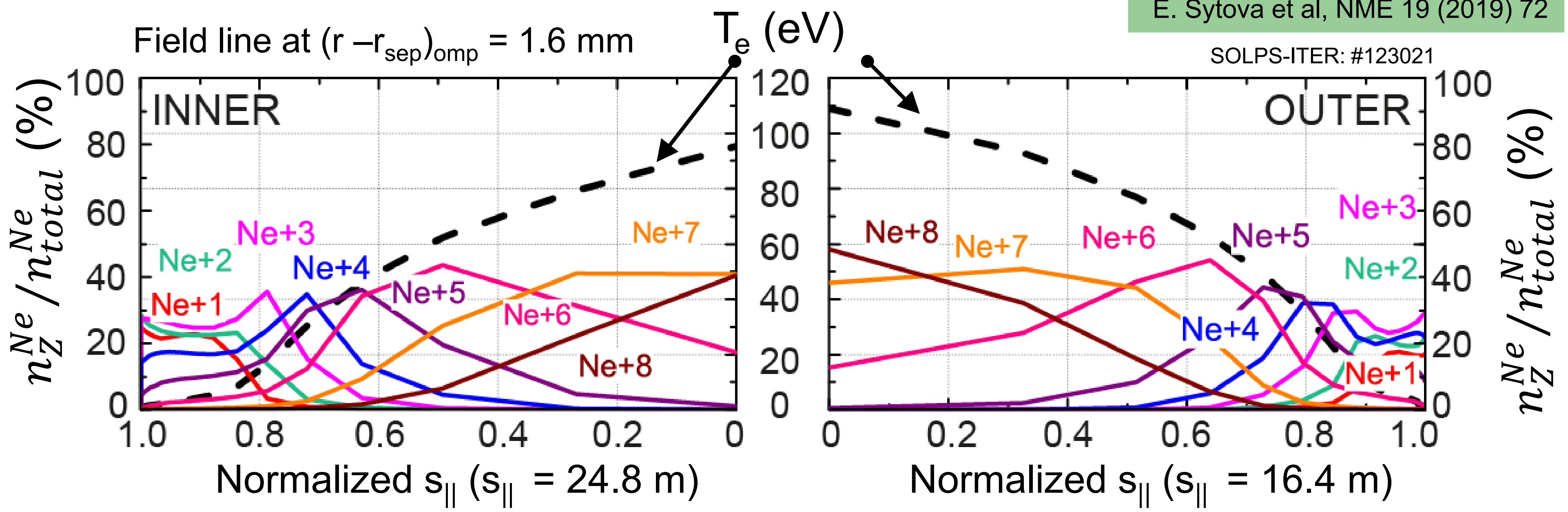




Both JA-DA and EU-DA have met tolerances on vertical target MB alignments

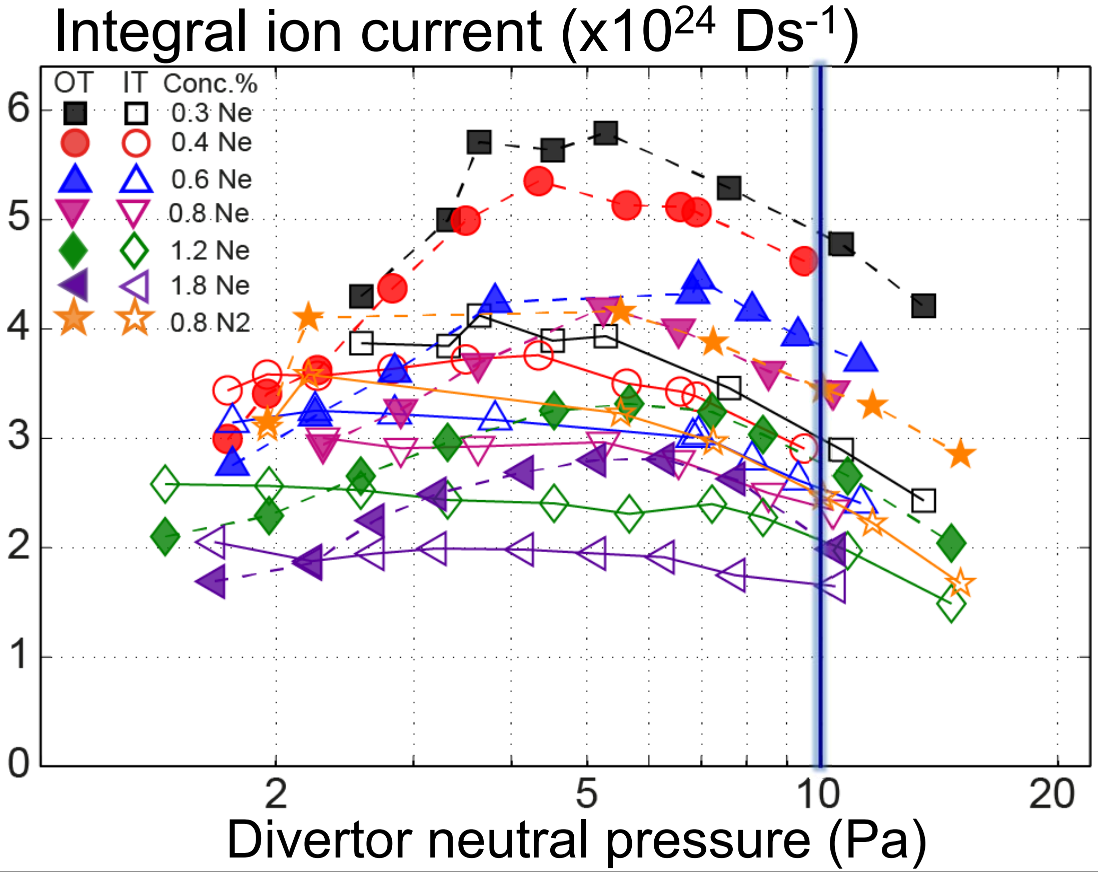
# Impurity charge state distribution

E. Sytova et al, NME 19 (2019) 72




- ~87% of the divertor radiation 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

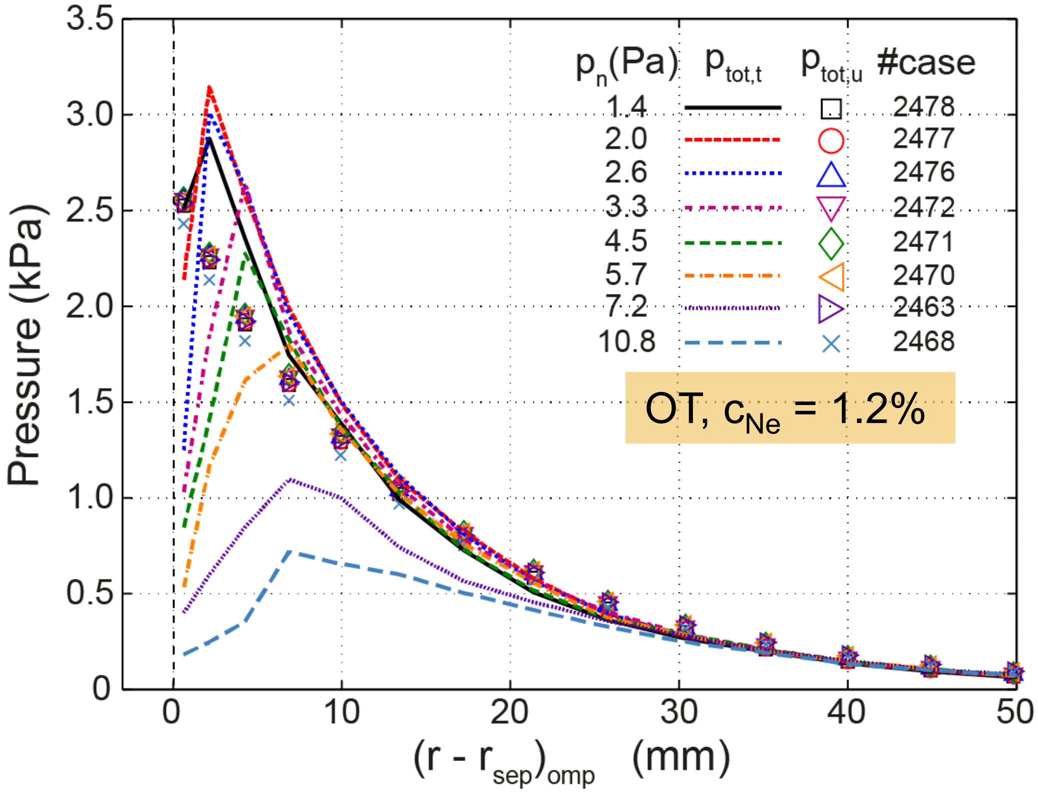
# Integrated target ion fluxes




- 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

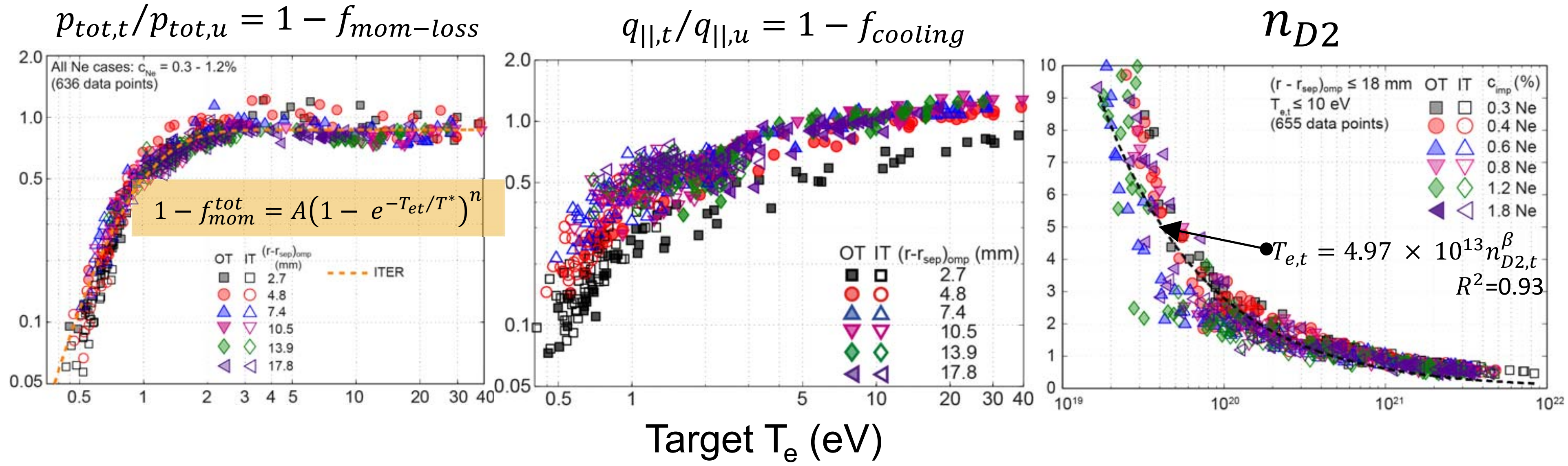
# Total pressure-momentum losses

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




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

# Importance of $T_{e,t}$

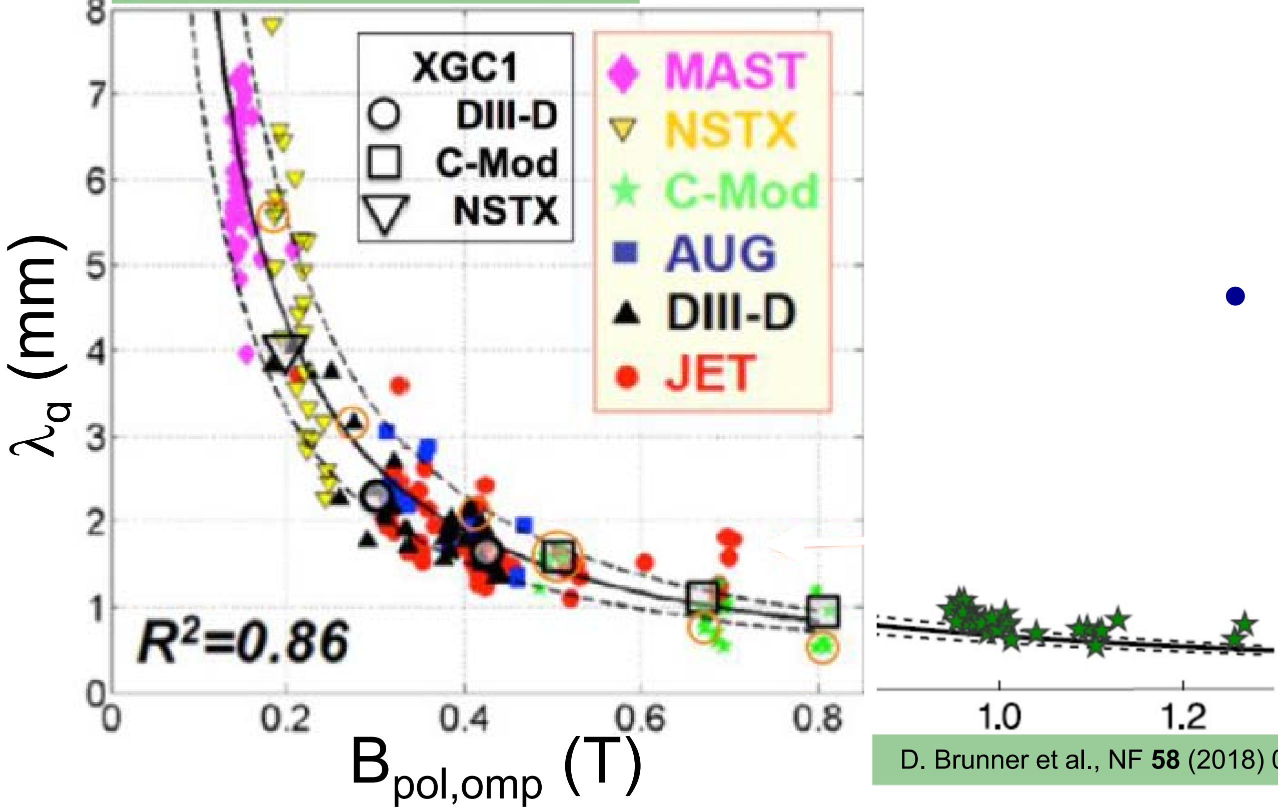



- Momentum and power losses in the ITER simulation database strongly correlated with $T_{e,t}$
  - Functions proposed by Stangeby* work well

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

# What will be the true $\lambda_q$ on ITER?

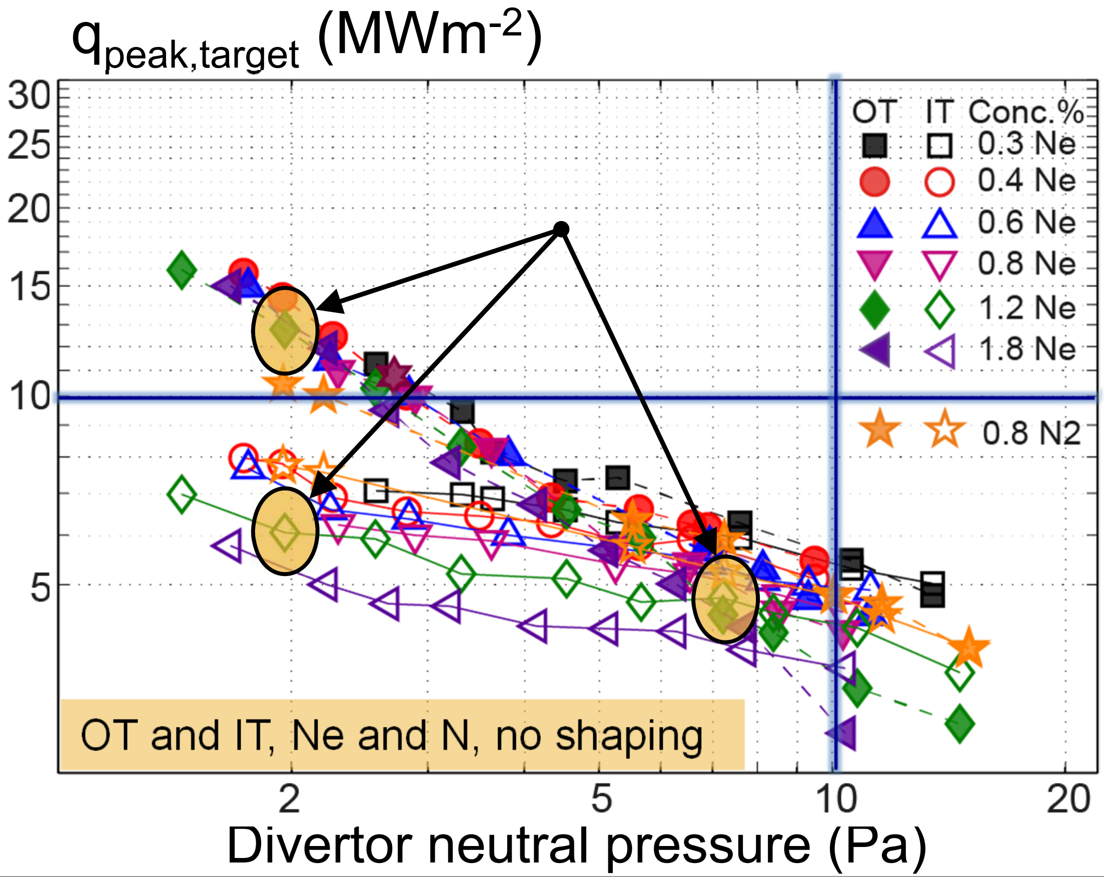
C. S. Chang et al. NF **57** (2017) 116023




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

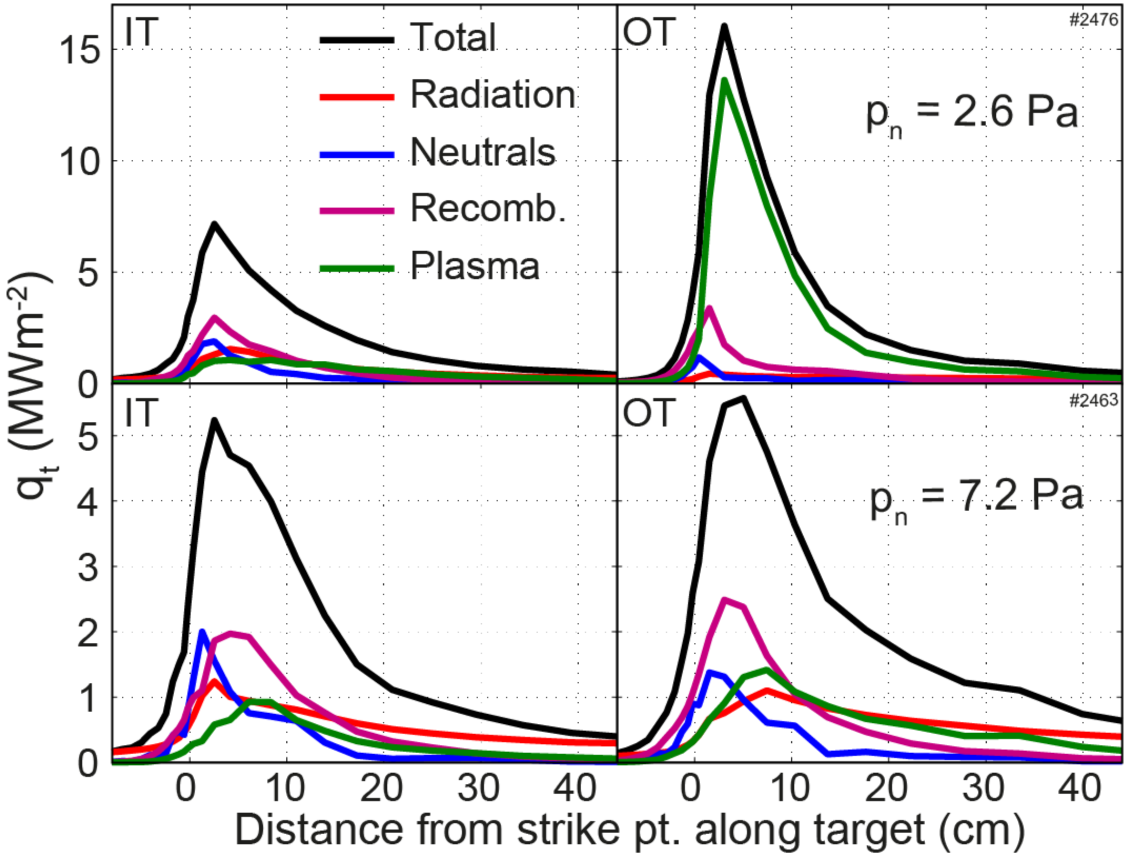
- XGC1 electrostatic global gyrokinetic simulations match $\lambda_q \propto 1/B_{pol}$ scaling
- Data out to ITER $B_{pol}$ on C-Mod continue to follow scaling

# 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




- Need to apply angle corrections for global target tilting and monoblock toroidal shaping **only** to thermal plasma components
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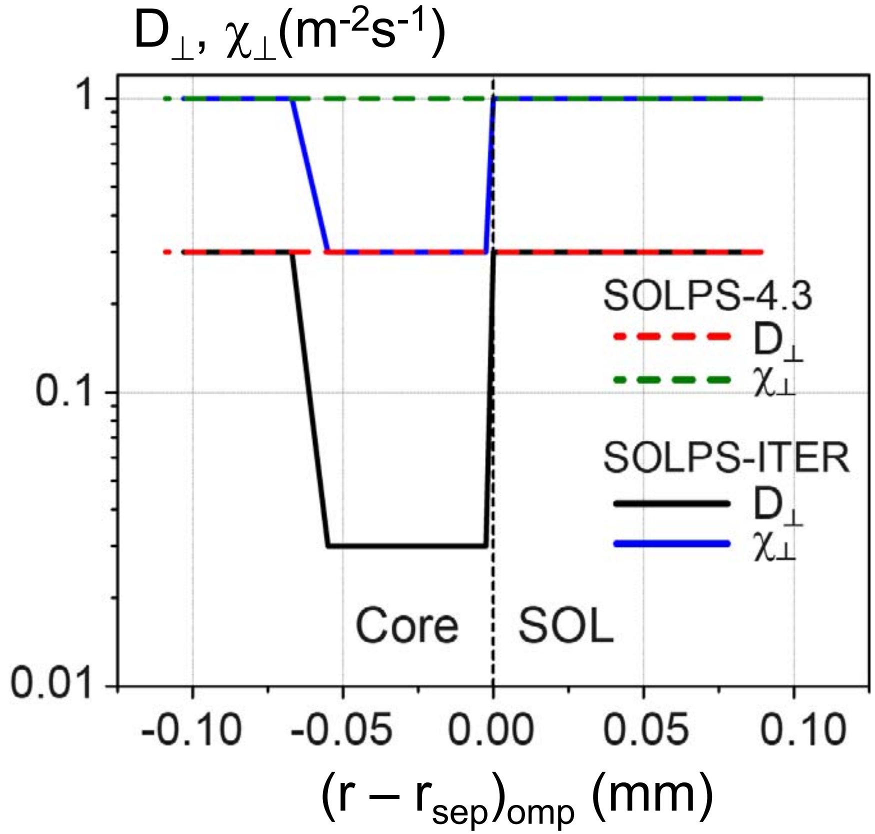
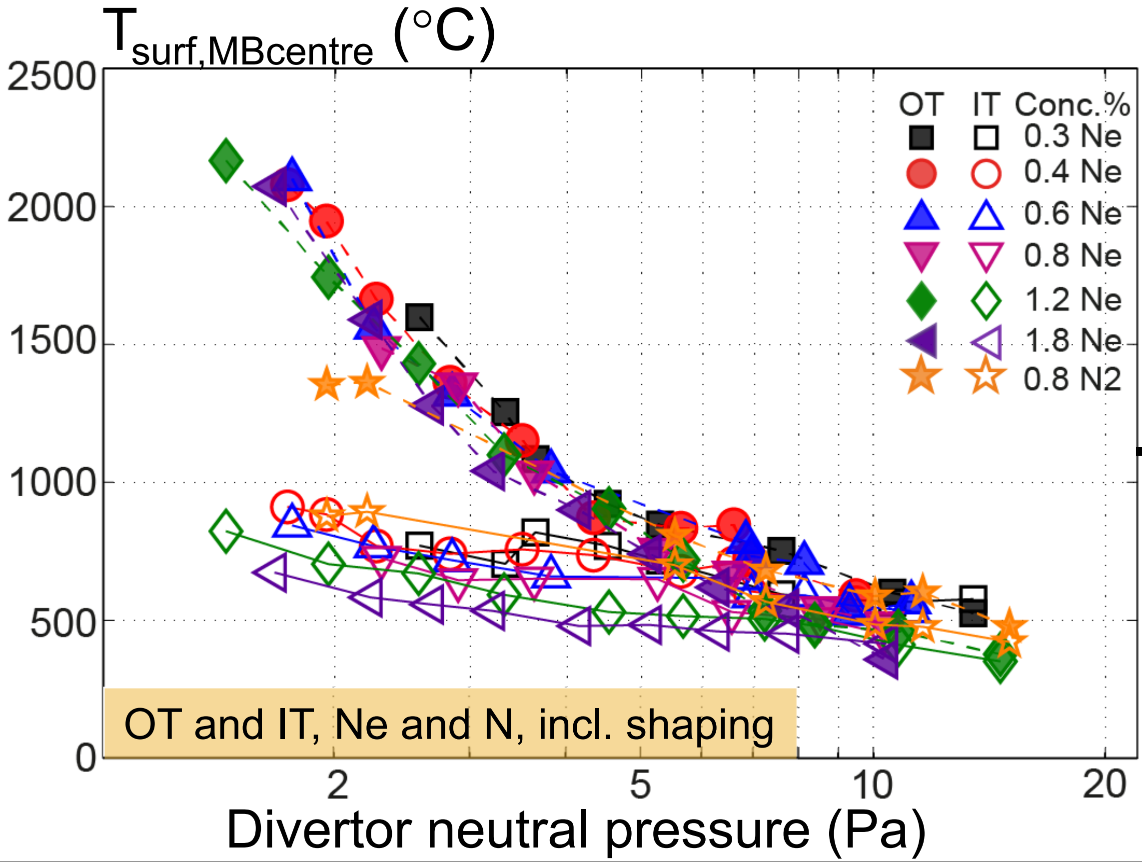
# SOLPS-ITER drift simulation transport

- Brand new results from SOLPS-ITER
  - Be/W walls, same SOL transport as SOLPS-4.3 database
  - Ne seeding
  - $P_{IN}$ = 100 MW
  - H-mode pedestal now included
  - Sophisticated code speed-up schemes required just to make drift runs possible[1]

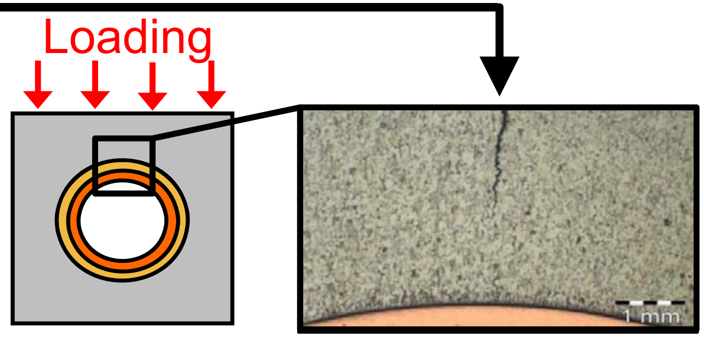



[1]E. Kaveeva et al, NF 58 (2018) 126018

# “Lifetime” power density limit?

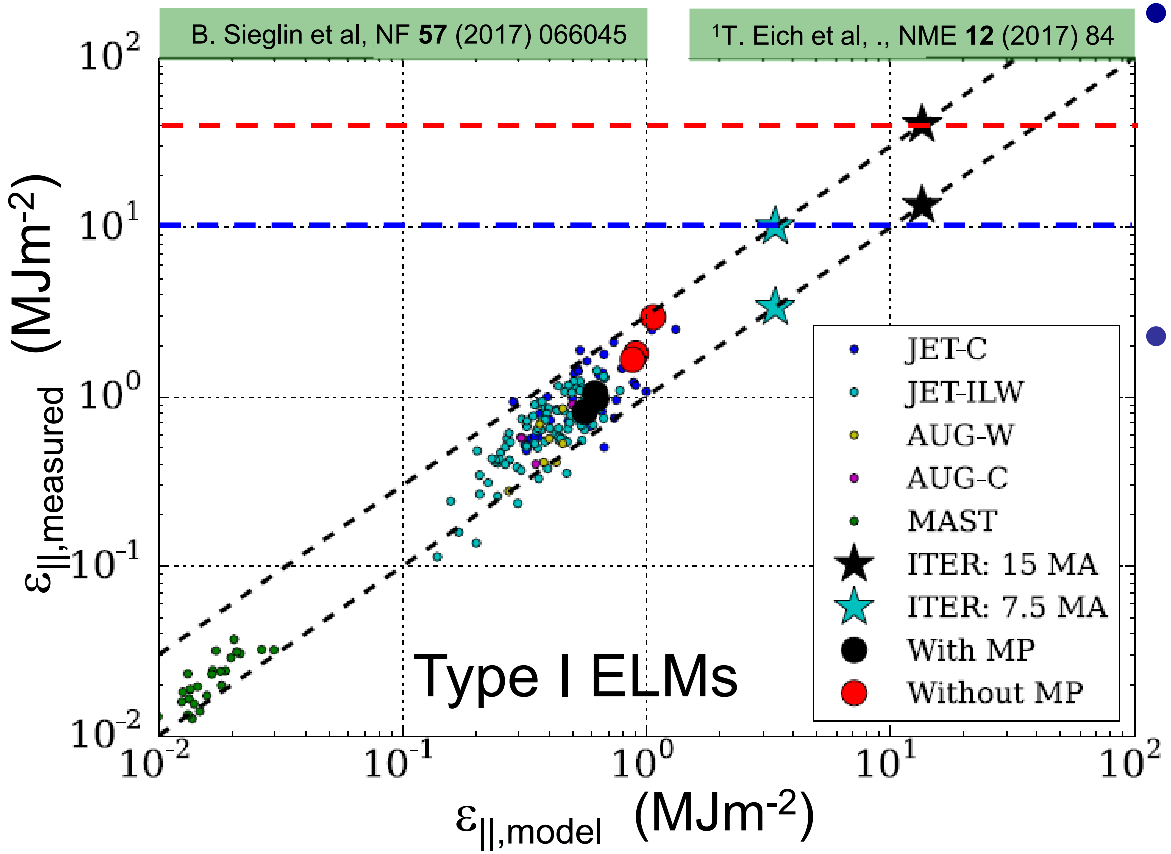



- Use FE simulations to transform $q_{peak}$ to $T_{surf}$
  - Take value at monoblock centre → where cracking seen to start under high heat flux testing




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

# ELMs – what if ELM suppresion not possible?




- Encouraging new scaling for target parallel ELM energy $\varepsilon_{||,targ} \approx$ $\mathbf{6\pi . p_e R q_{edge}}$
- For ITER targets with shaping, scaling gives:

  7.5 MA:

  $\varepsilon_{\perp,targ} \cong 0.36 \pm 0.18$ MJ m$^{-2}$

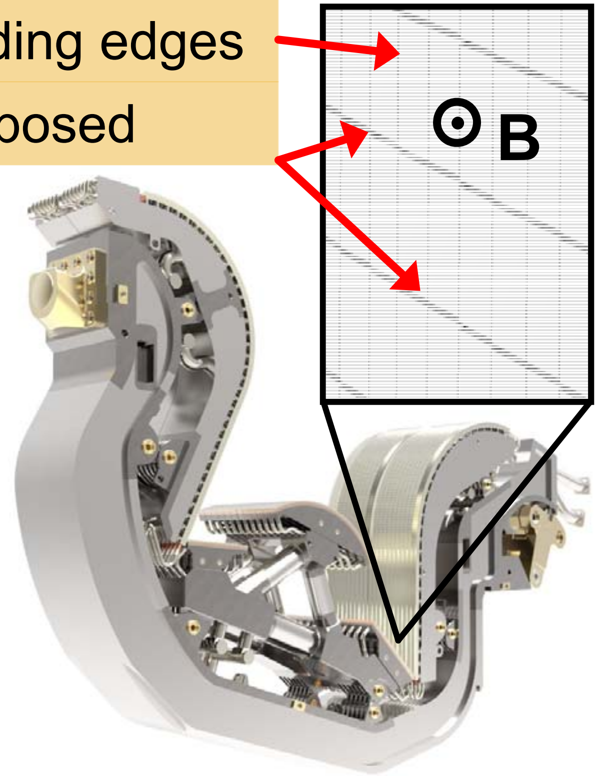
  15 MA:

  $\varepsilon_{\perp,targ} \cong 1.2 \pm 0.6$ MJ m$^{-2}$

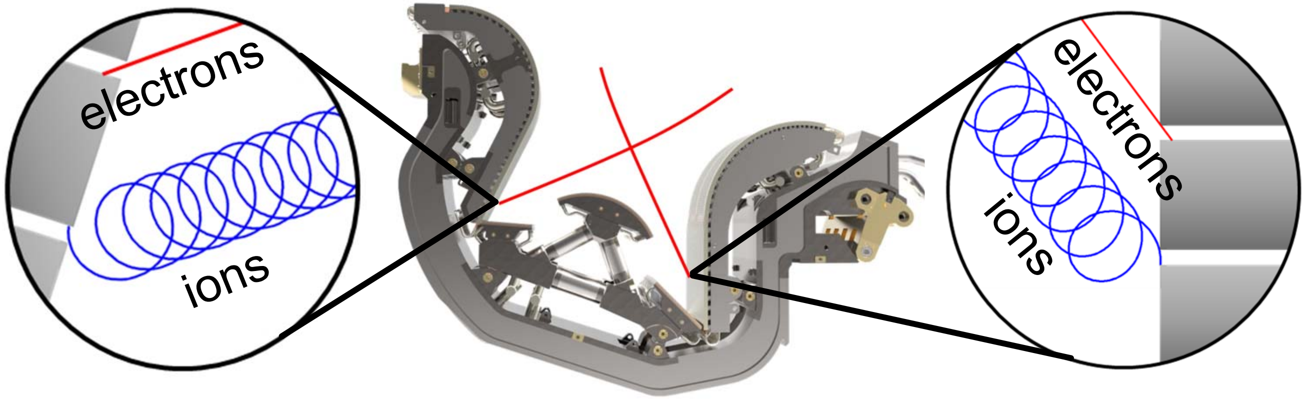
# Problem of toroidal gaps

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

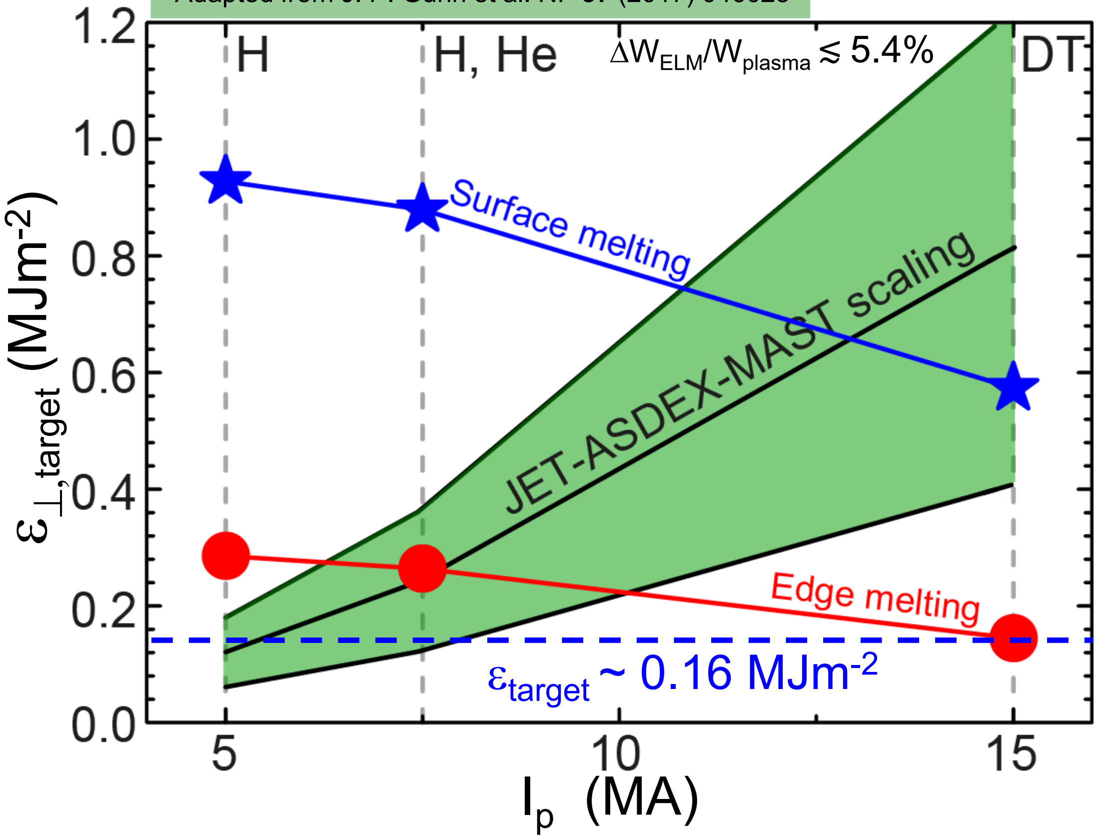



- Toroidal gap (TG) loading really does occur[1]

[1]R. Dejarnac et al, NF 58 (2018) 066003

# Constraints on ELM energy loss

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




- To avoid toroidal gap edge melting, $\Delta W_{ELM}/W_{plasma} \lesssim 0.2\%$
  - Very similar to limits found for no surface damage from large cycle number, ELM-like e-beam testing of ITER-grade W*
  - Such ELMs not found naturally on tokamaks
  - ELM buffering in detached divertor?

*M. Wirtz et al., NME **12** (2017) 148