Power exhaust in ITER I: First Wall

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The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

A word on the contents of my talks at this school

- I will present two separate talks on ITER power exhaust
  - One mostly on first wall (this one)
  - The second on the divertor
- They will be largely inspired by two recent plenary talks I have given at fusion meetings:
  - ISFNT-13, Kyoto, Japan, September 2017
  - PSI-23, Princeton, USA, June 2018
- This first one will discuss some elements of building a heat flux specification for ITER First Wall components:
  - The information needed by engineers

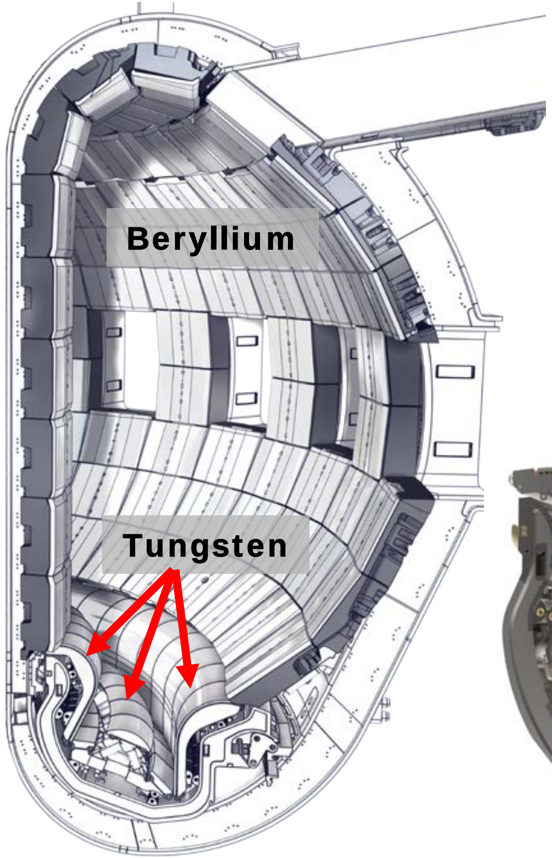
Content of Part I (first wall)

- The ITER plasma-facing components
  - Brief (1 slide) → much more in the talk from M. Merola
- Basics of the ITER PFC main wall heat load specification
  - The approach followed at ITER, definition of parallel heat fluxes are defined
- Some key examples of thermal load specs
  - Start-up loads on the central column → use this as case study and example of how international collaboration answered specific ITER questions during the design phase
  - Stationary heat loads during diverted operation (including ELM average)
  - Charge-exchange, radiative
  - Disruptions → important but not treated here (not "stationary exhaust")

A disclaimer

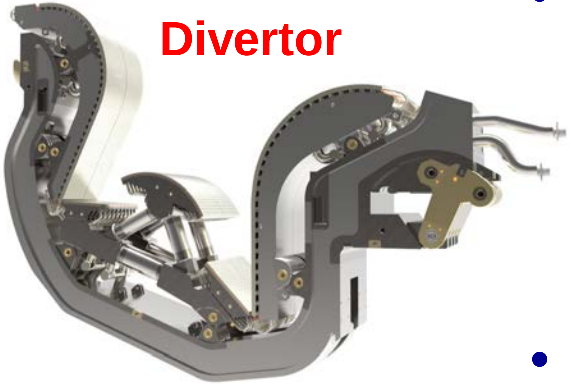
- I will discuss mostly thermal load specifications → the key physics inputs for PFC engineering design on ITER with regard to expected heat loads
- Only plasma related heat fluxes → neutronic loads not considered → not an issue for PFC integrity/lifetime on ITER → but will be on DEMO
- See the talk by F. Maviglia for some of the methods described here applied to the step after ITER

ITER Plasma-Facing Components



Blanket first wall

Divertor

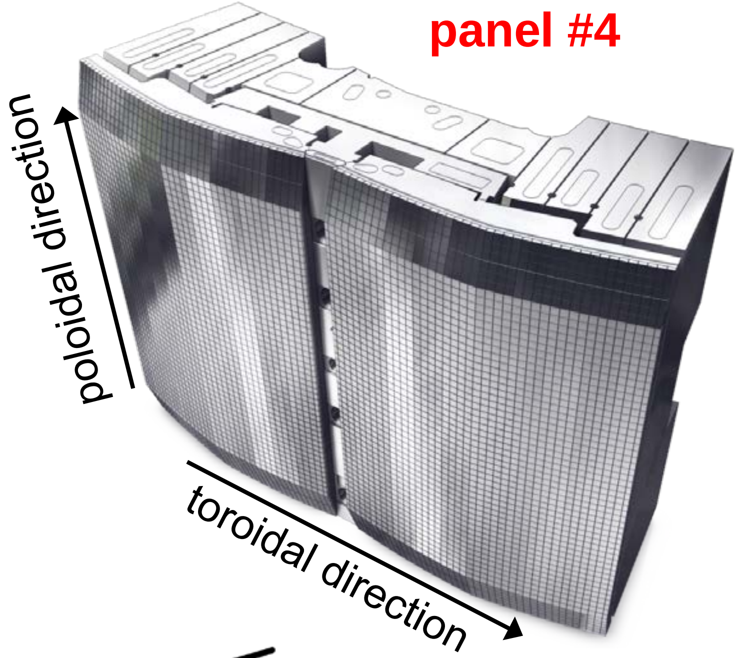


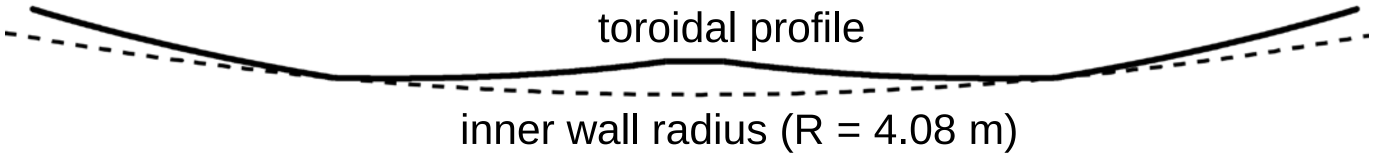
- ~700 m^2 beryllium
  - Low Z – good plasma compatibility
  - Good oxygen getter
  - Good thermal conductivity
- ~150 m^2 tungsten
  - Low sputtering yield, high threshold
  - Highest melting point
  - Low fuel retention
- All actively (water) cooled

First wall panel shaping

- All ITER first wall panels have toroidally shaped front surfaces (and sometimes poloidally)
  - Magnetic field line incidence angles are low due to much stronger toroidal field cf. poloidal field
  - Must avoid “leading edges” due to radial misalignments which are inevitable for components on the ITER scale
  - More on this later

e.g. Inner wall panel #4





So how to build a first wall plasma thermal load spec.?

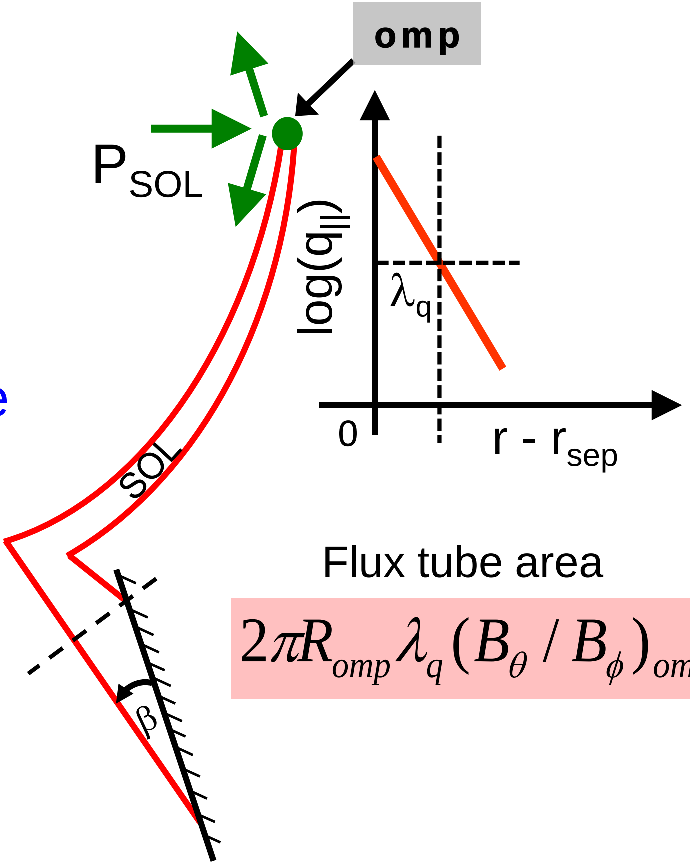
- The approach adopted at ITER has been to:
  - Be conservative, seek worst case first
  - Start simple, add complexity later where necessary
  - Be self-consistent (not always possible/easy)
  - Remain **"component independent"** as far as possible → prescribed heat fluxes should drive the design, not be modified to suit the design!
  - Separate "stationary" and "transient" loads:
    - PFCs cannot generally be designed for high energy transients (ELMs and disruptions) accessible in ITER and beyond
    - But consequences should be assessed and are a big part in setting mitigation strategies

Starting point: plasma heat flux definition

- For any given plasma equilibrium:
  - Simple model: specify P_{SOL}, λ_q
    P_{SOL} = power into SOL
    λ_q = characteristic width for SOL power flow parallel to B
  - Impose 0D power balance at outer midplane and construct radial profile of parallel power flux (this is an approximation):

$$q_{||omp} = P_{SOL} / (4\pi R_{omp} \lambda_q (B_\theta / B_\phi)_{omp})$$

$$q_{||}(r) = q_{||omp} \exp(-(r - r_{sep}) / \lambda_q)$$



Flux tube area

$$2\pi R_{omp} \lambda_q (B_\theta / B_\phi)_{omp}$$

Starting point: parallel heat flux definition

- Assume no volumetric power loss/gain along flux tube from omp (θ = 0) to any other poloidal location (ok for main chamber usually):

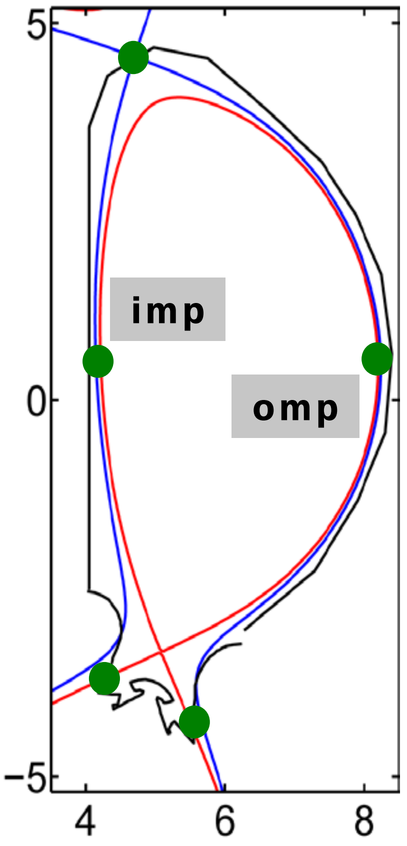
$$\frac{q_{||}(\theta)}{q_{||}(0)} = \frac{B(\theta)}{B(0)} \approx \frac{B_\varphi(\theta)}{B_\varphi(0)} = \frac{R(\theta)}{R(0)}$$

  - Distortion of flux tube parallel area ($A_{||}$)
  - Works for tokamak because $B_\varphi \propto 1/R$

$$q_{||PFC} = q_{||omp} R_{omp} / R_{PFC}$$

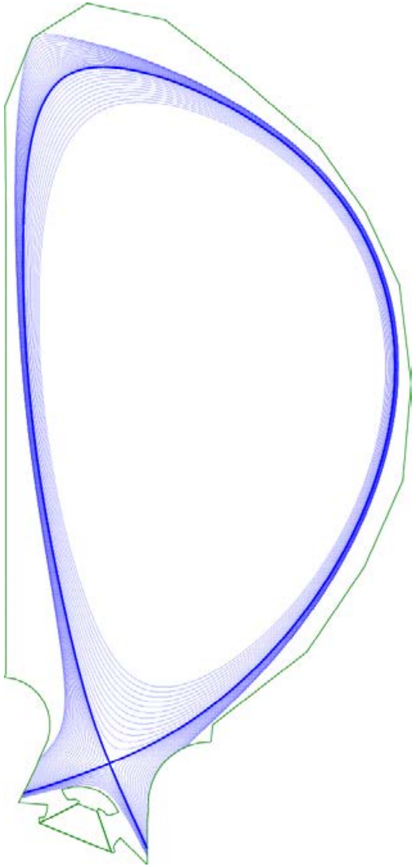
$$q_{\perp PFC} = q_{||PFC} \sin \alpha$$

- α total (3D) angle of incidence on the surface



Flux expansion

- To get complete picture of power flux density distribution on the component, full magnetic field line tracing is usually required:
  - Properly account for POLOIDAL flux expansion



Example new (shown for first time here!) ITER high triangularity expanded divertor flux equilibrium currently understudy for divertor power loading

Flux expansion

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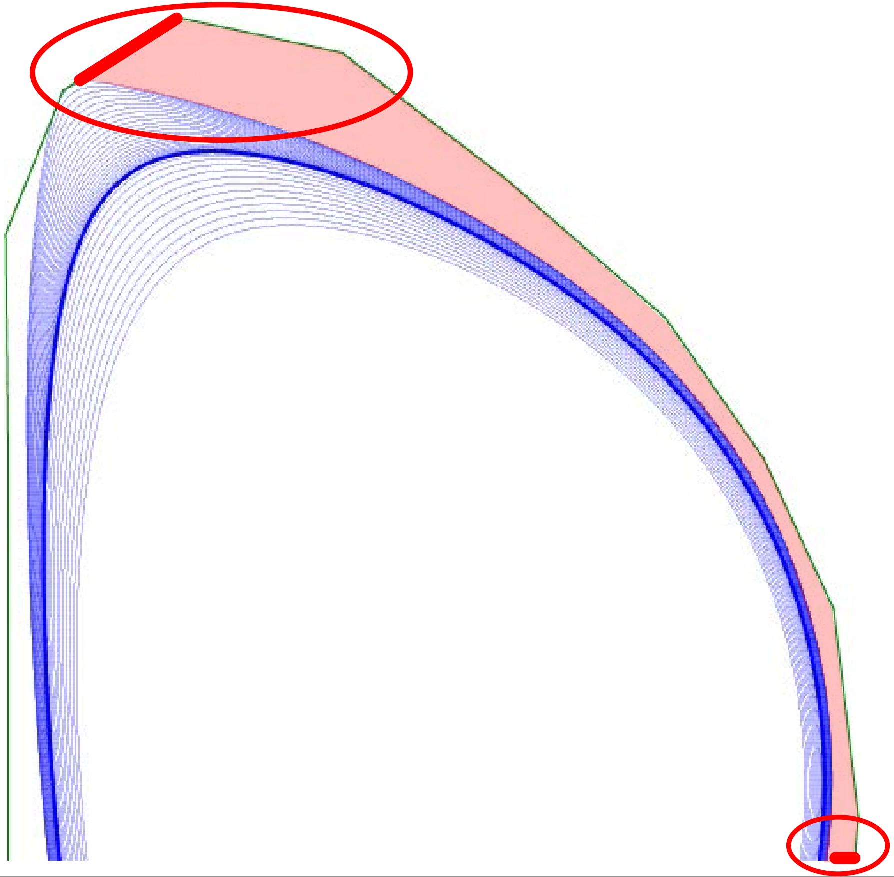
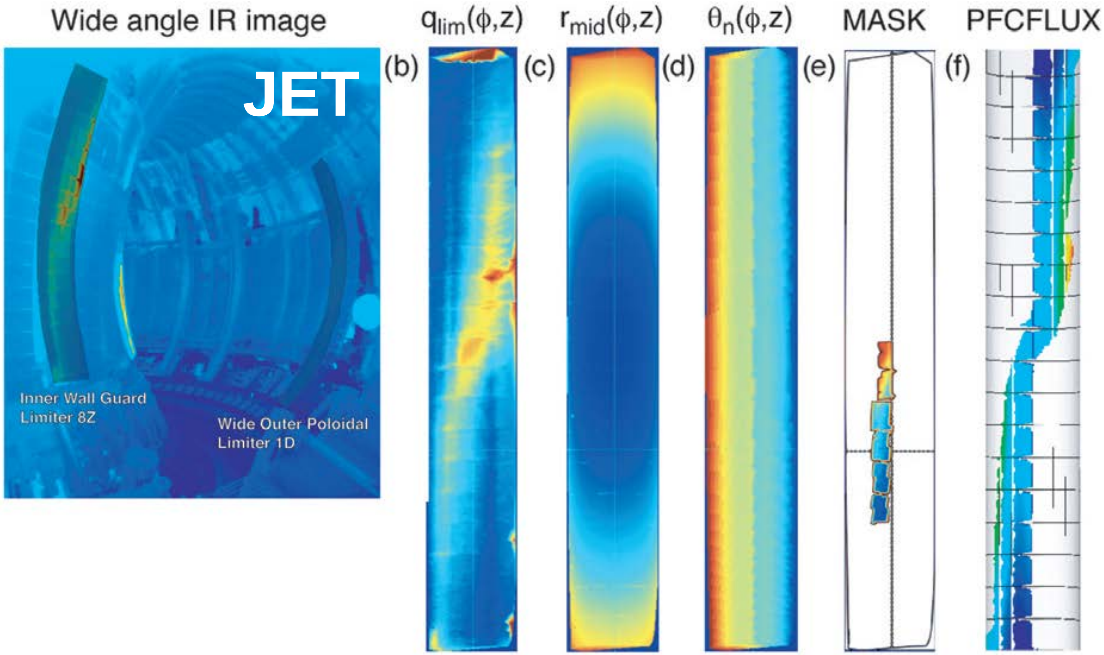


Field line tracing tools

- Most magnetic fusion institutes develop their own software
  - PFCFLUX* is an example of a more general tool developed at CEA
  - Used in characterization of limiter heat fluxes in the first JET ITER-Like Wall experiments (see later for more on this)
  - Also used in the past on ITER

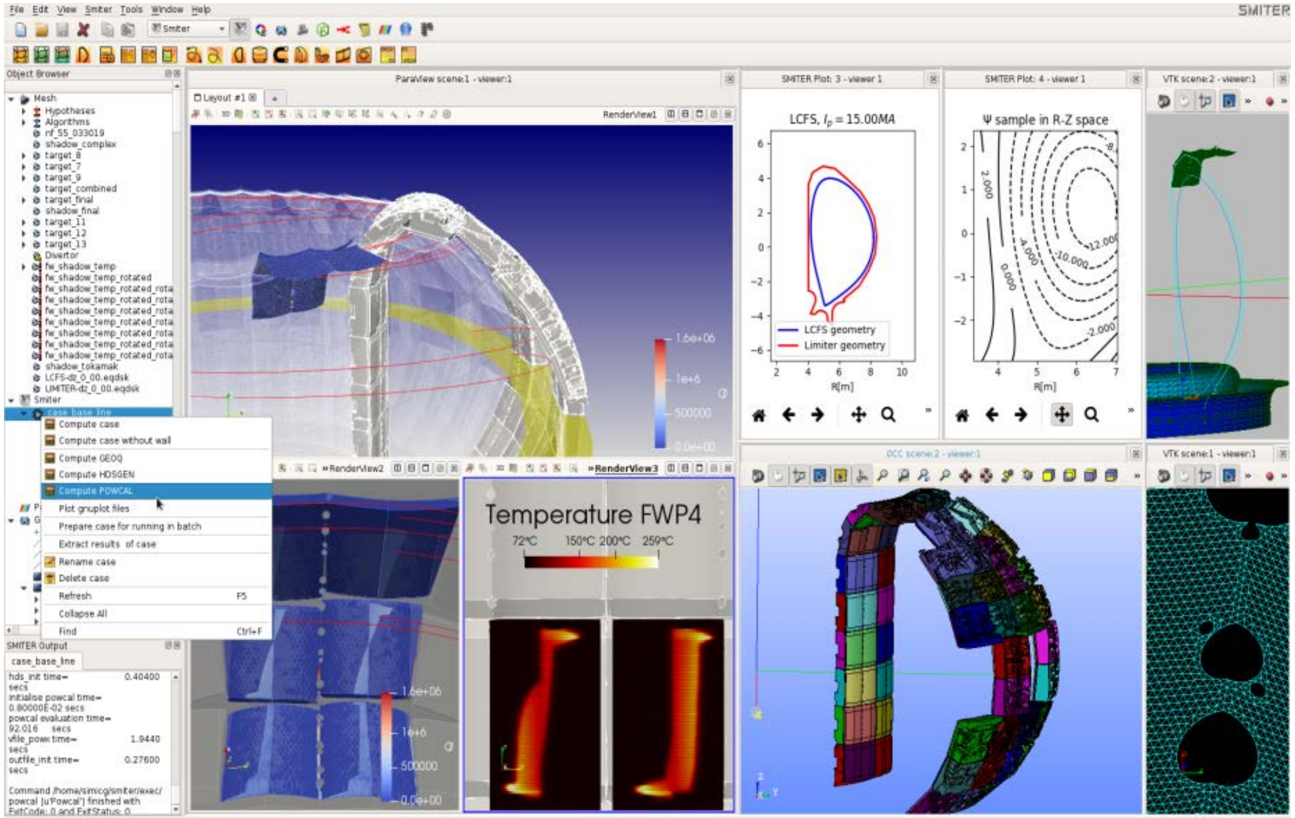
G. Arnoux et al., Nucl Fusion **53** (2013) 073016



*M. Firdaouss et al., J. Nucl. Mater. **438** (2013) S536

The SMITER code

- To provide a tool for ITER and available to all ITER Partners (unlike PFCFLUX), the IO has developed the SMITER code
  - Being incorporated into the ITER Integrated Modelling suite (IMAS)

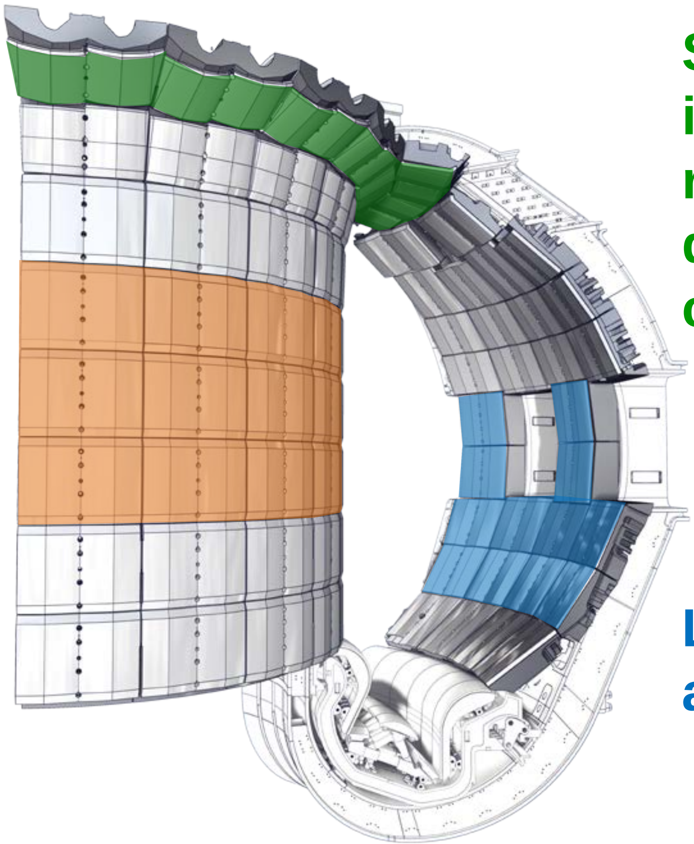


L. Kos, R. A. Pitts et al., to be published in Fus. Eng. Des.

See also poster by H. Anand at this school

Building up a thermal load spec: areas of concern

- Panels in these areas have highest heat handling capacity: $q_{\perp}$ ~4.5 MWm^{-2}



Building up a thermal load spec: areas of concern

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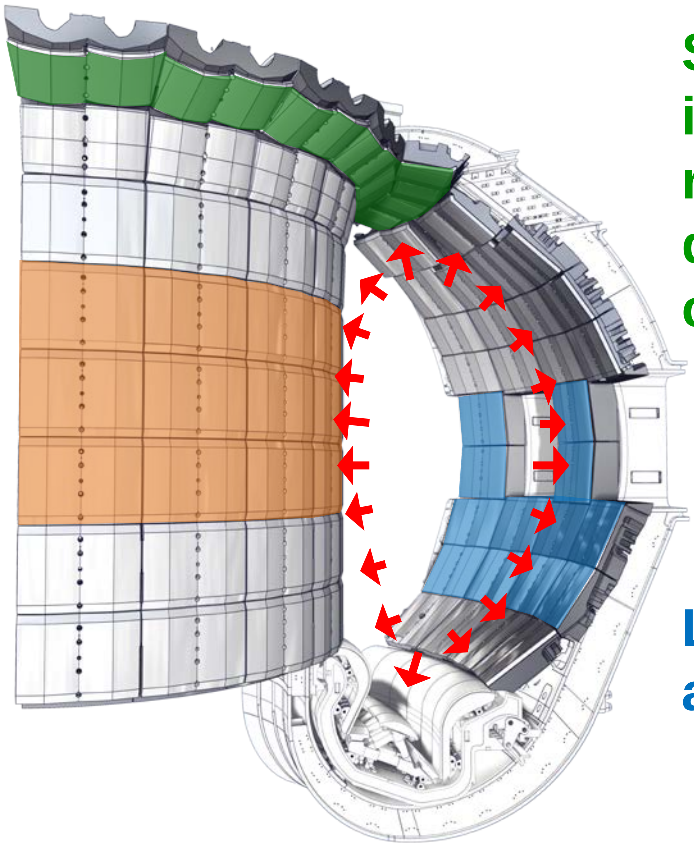
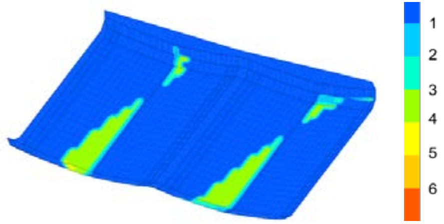


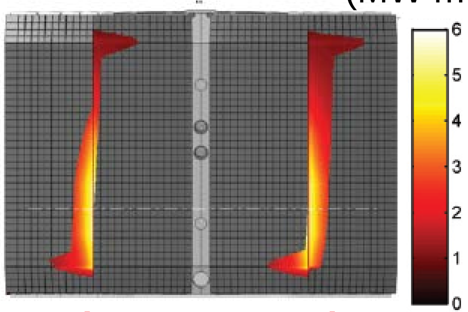
Price of shaping

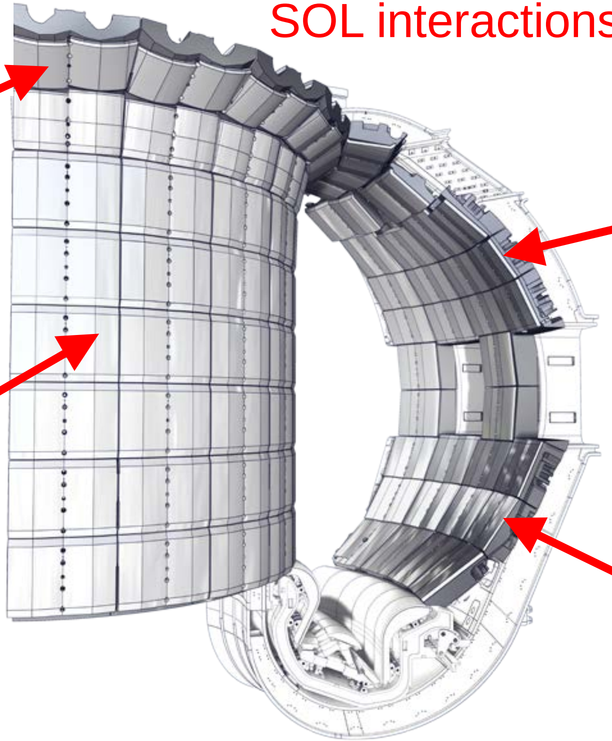
- Reduced projected area, steeper angles, higher power fluxes, peaked erosion→ possibility for local re-deposition in shadowed regions

Upper X-pt.

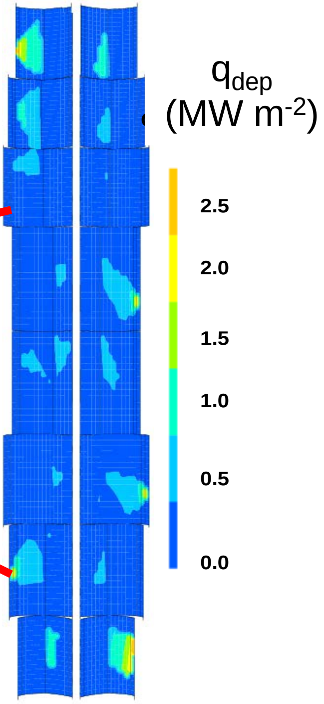


Inboard limiter start-up





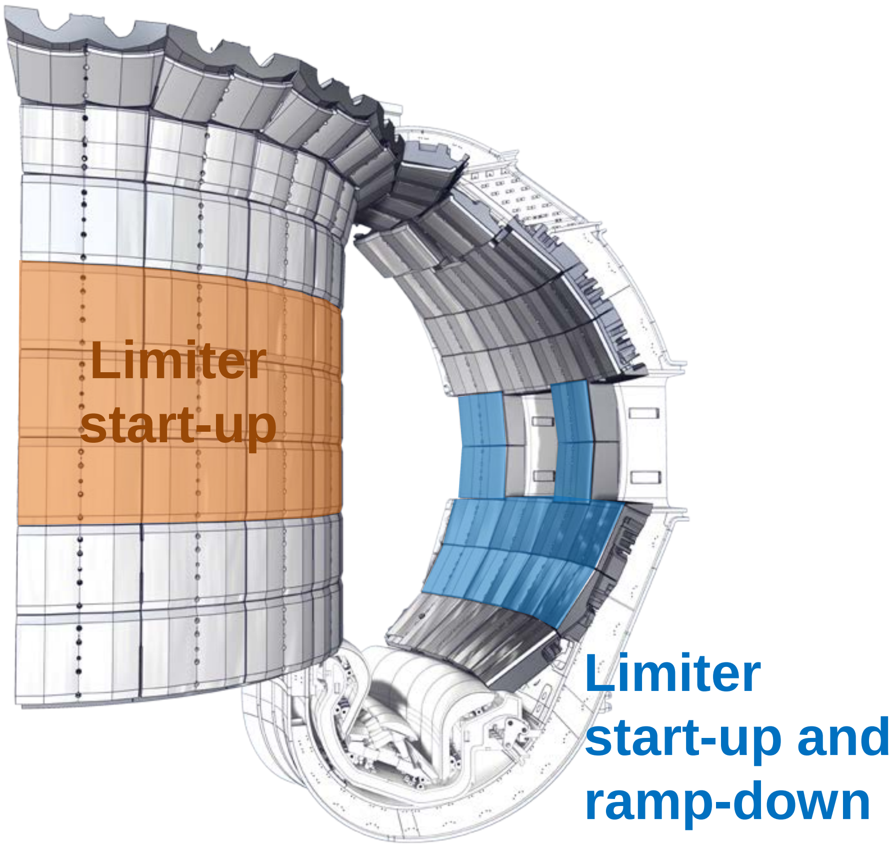
Outboard far-SOL interactions



- Very important to be sure about λ_q!

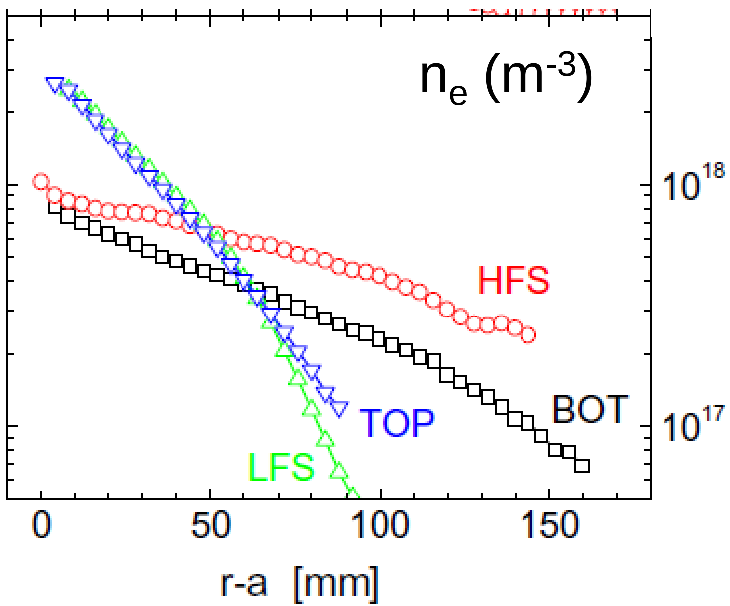
Case study: limiter start-up

- Original design spec. (2008) was:
  - $\lambda_{q,imp}$ = 50 mm (inboard limiter)
  - $\lambda_{q,omp}$ = 15 mm (outboard limiter)
- Based on tokamak data from diverted L-mode plasmas (1999 ITER Physics Basis)
  - Assumed also to hold for toroidally continuous limiters
  - Inboard-outboard λ_q difference due to ballooning transport and flux expansion
  - Design assumes P_{SOL} (MW) ~ I_p (MA)

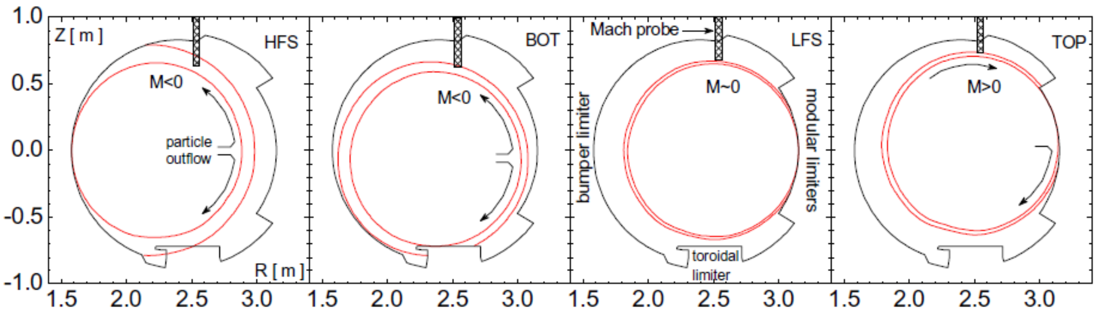


Example from Tore Supra

- Mach probe measurements

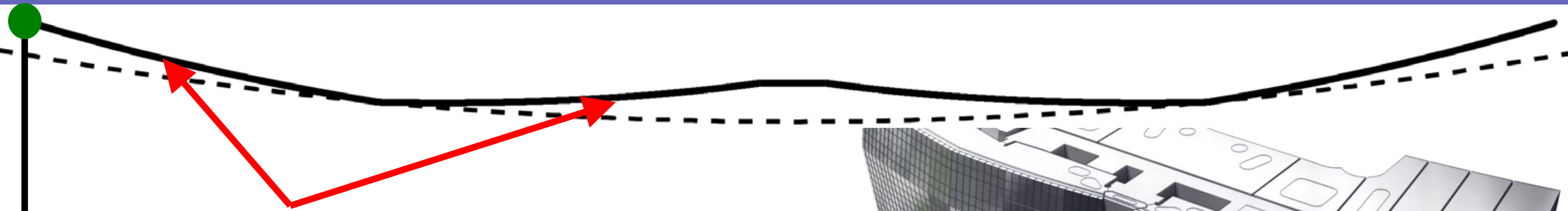


J. P. Gunn et al., J. Nucl. Mater. **363-365** (2007) 484

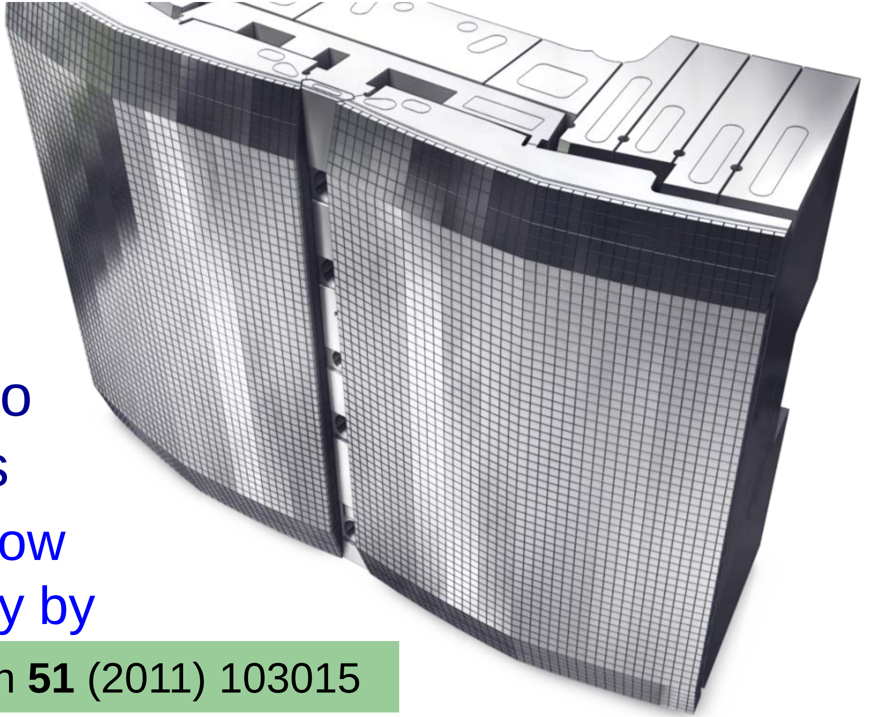


- ~factor 4 broader SOL density profile for HFS than LFS plasmas
  - Limiting on outer wall "cuts-off" LFS ballooning transport so λ_q shorter
  - Flux expansion factor $$f_x = \frac{R_{omp}}{R_{imp}} \frac{B_{\theta,omp}}{B_{\theta,imp}}$$ f_x ~1.6 (Tore Supra) → ballooning factor ~2.5
  - Assume same factor for ITER (and f_x ~1.3)

Shaping design

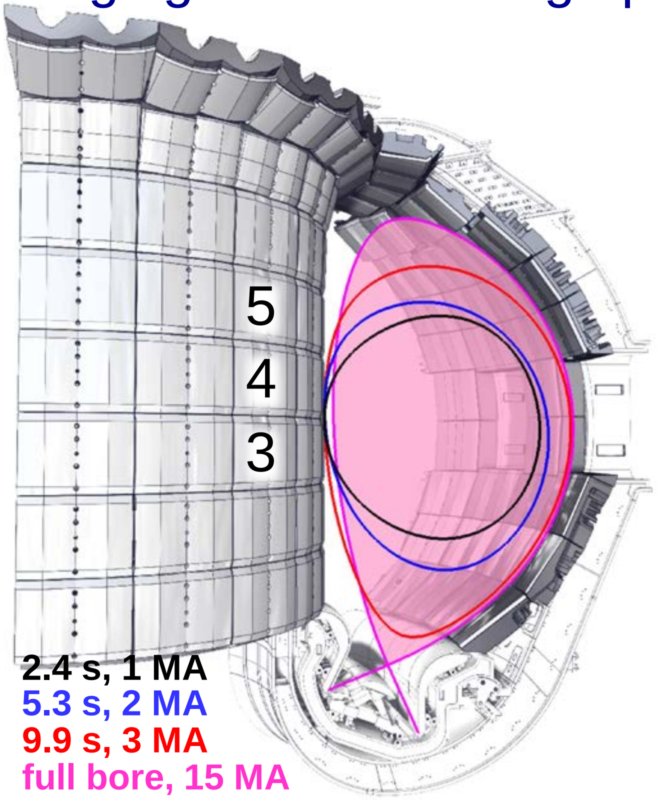


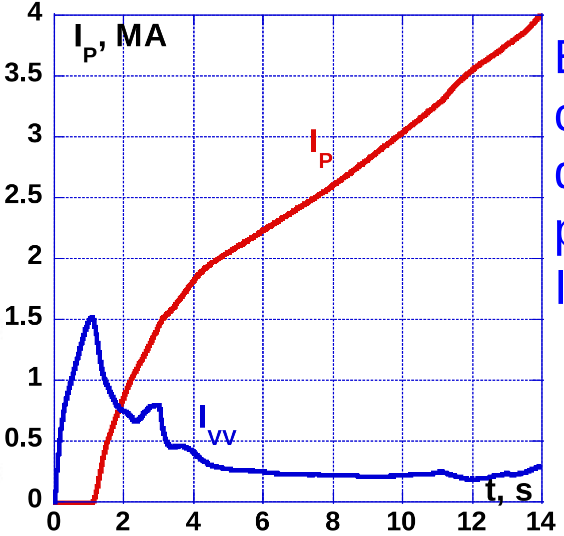
- The curvature of this profile is carefully designed to optimize power spreading for specified λ_q
- Radial "set-back" of panel wings to protect component misalignments
  - Courageous readers can find out how the shape is defined mathematically by consulting: P. C. Stangeby, Nucl. Fusion **51** (2011) 103015



Focus on inboard start-up

- Challenging on ITER → high potential heat loads, happens on every shot



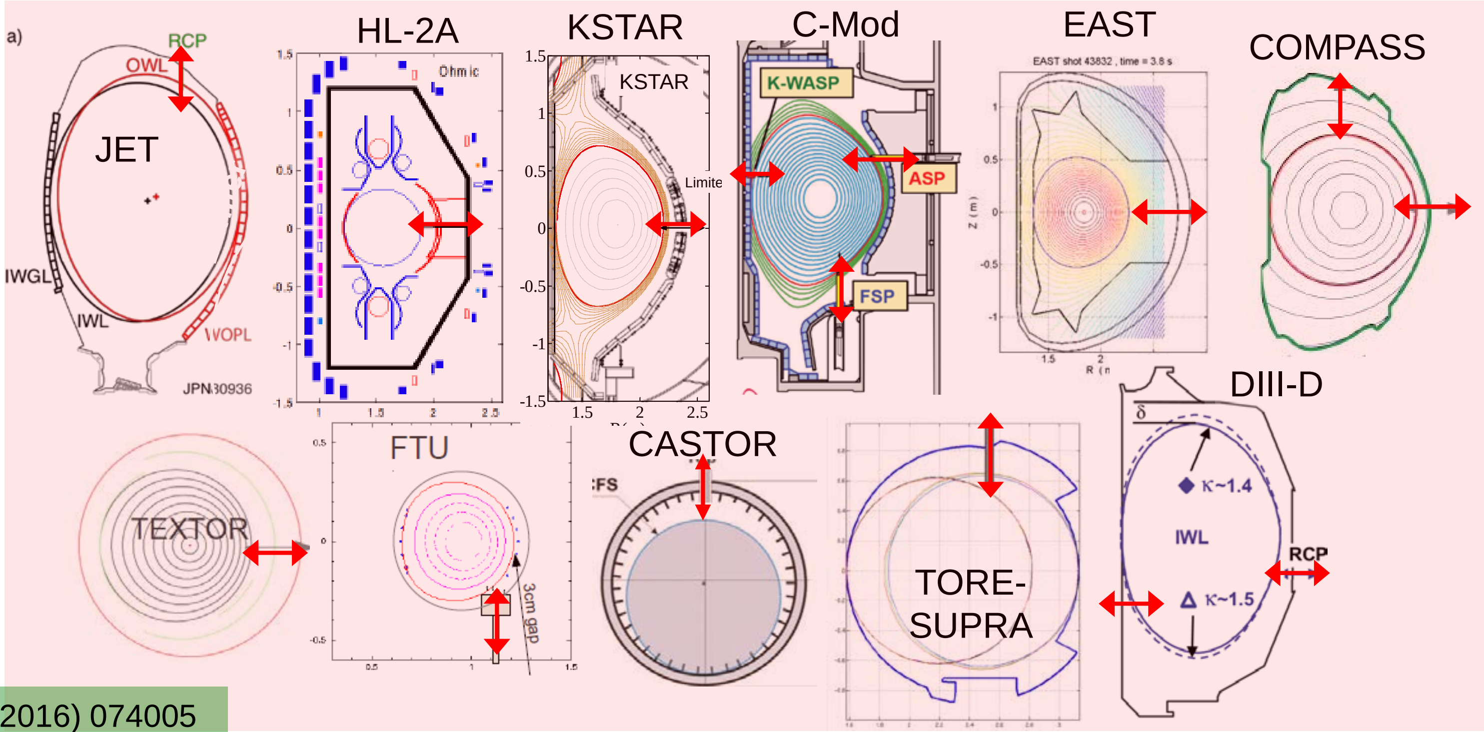


Example DINA code scenario design:
plasma diverts at I_p ~ 3.0 - 3.5 MA

Inner wall start-up preferred:
lower 3D stray fields (no ferritic inserts, no port openings, less impact of induced current in VV), better ECRH coupling, lower flux consumption

First question: is the assumed SOL λ_q correct?

- Original spec based on L-mode divertor scaling and sparse limiter measurements → asked the R&D Community in 2012 for more (ITPA DivSOL Topical Group)
  - Pretty good response …!
  - All Langmuir probe data except JET which also had IR → see later

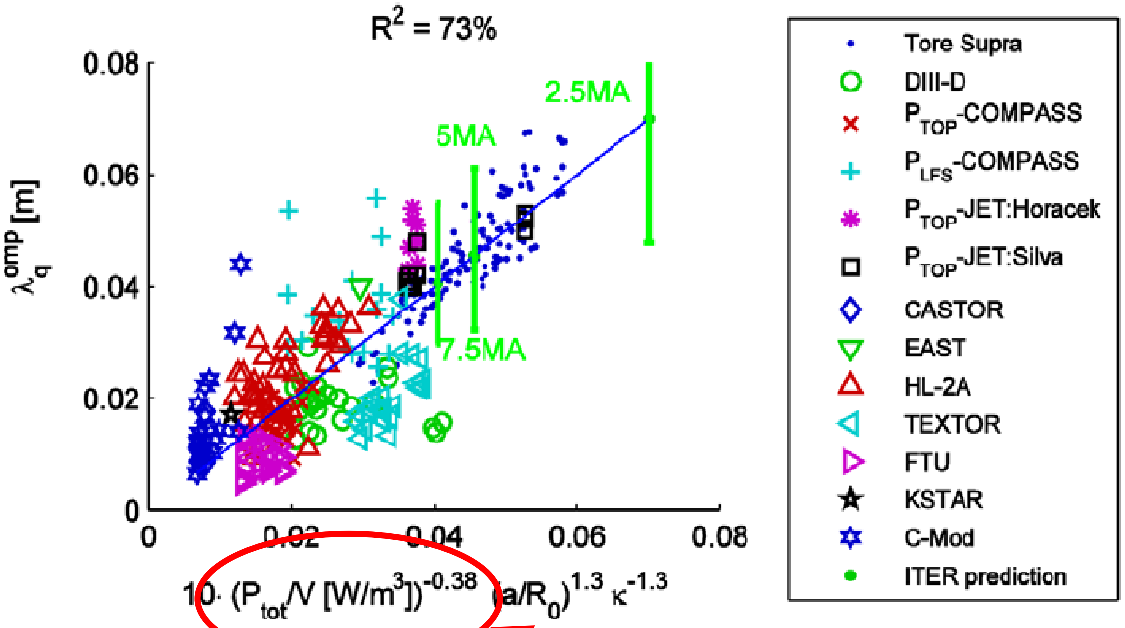


J. Horacek et al., PPCF **58** (2016) 074005

Answer is yes, more-or-less

J. Horacek et al., PPCF **58** (2016) 074005

- Lots of scatter (due in part to using Langmuir probes)
  - Many scalings tried
  - At highest allowed limiter phase I_p = 7.5 MA, best engineering scaling yields:
    $\lambda_{q,omp}$ = 44±11 mm →
    $\lambda_{q,imp}$ = 57 ± 11 mm
  - cf. original design
    $\lambda_{q,imp}$ = 50 mm
    → consistent so ~ok

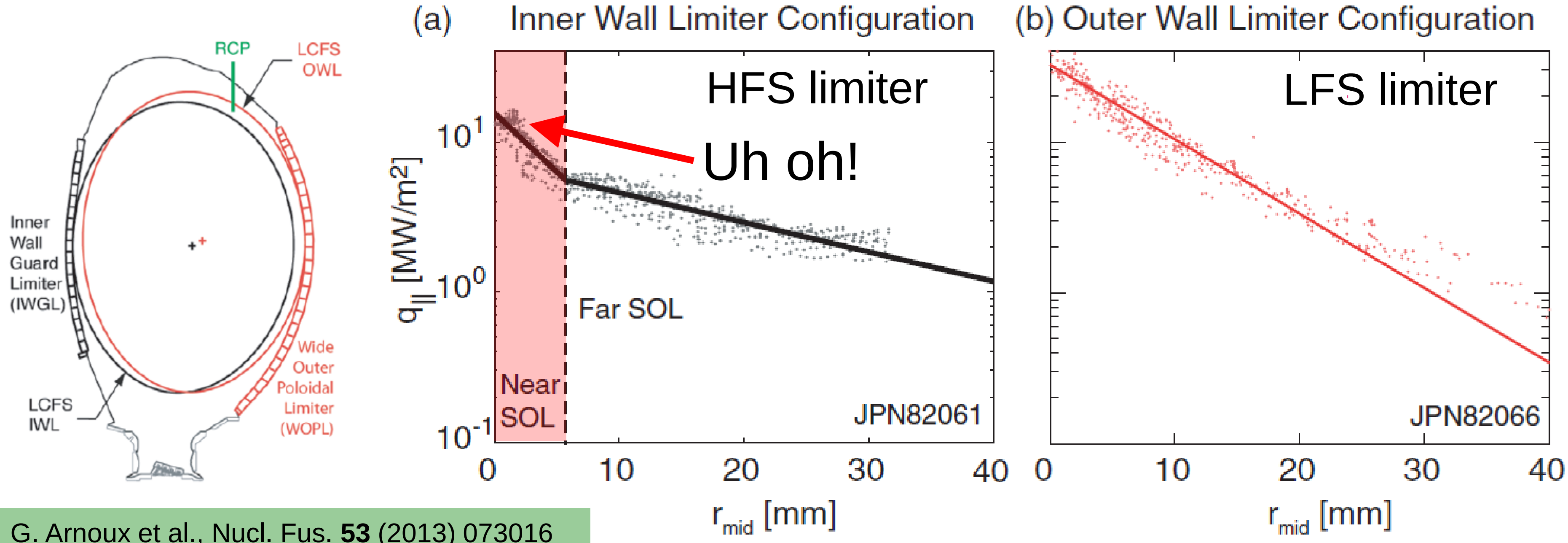


BUT…

$\propto I_p^{-0.4}$ for most limiter plasmas

A nasty little narrow feature seen at JET ....

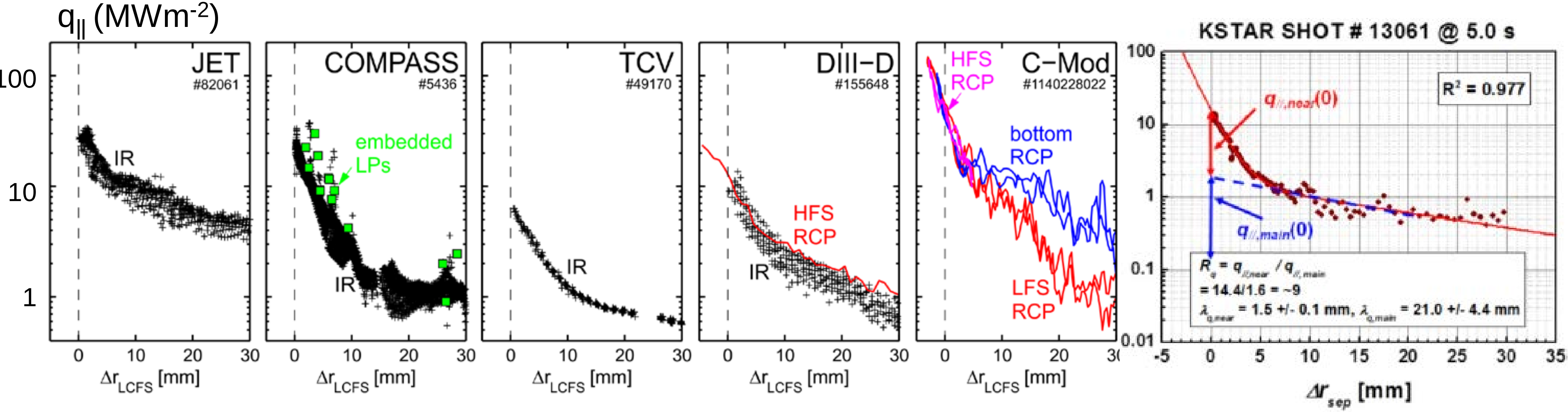
- Using inner limiter IR surface heat flux measurements, JET found a "double exponential parallel heat flux profile" for inner wall limiter plasma



G. Arnoux et al., Nucl. Fus. **53** (2013) 073016

Is the narrow feature unique to JET?

- Back to the R&D Community → big and enthusiastic response



- Narrow feature found in all 5 additional devices which looked for it
- Found both using Langmuir probes (SOL) and high resolution IR (limiter surface)

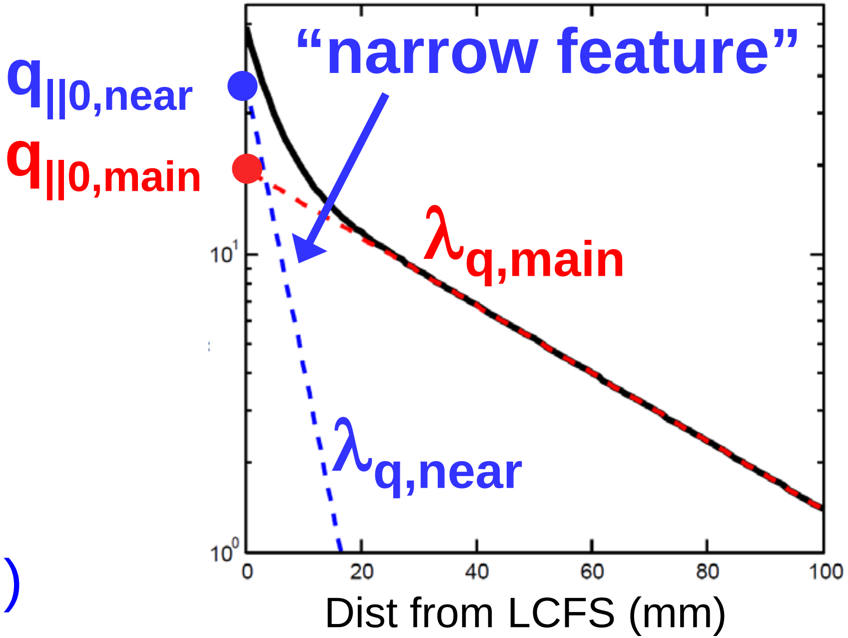
J. Horacek et al., JNM **463** (2015) 465
J.-G. Bak et al., NME **12** (2017) 1270
P. C. Stangeby et al., JNM **463** (2015) 369
F. Nespoli et al., JNM **463** (2015) 393
M. Kocan et al., Nucl. Fus. **55** (2015) 033019

Experimental HFS limiter SOL $q_{||}$ profile

- Profiles can all be fitted with a double exponential

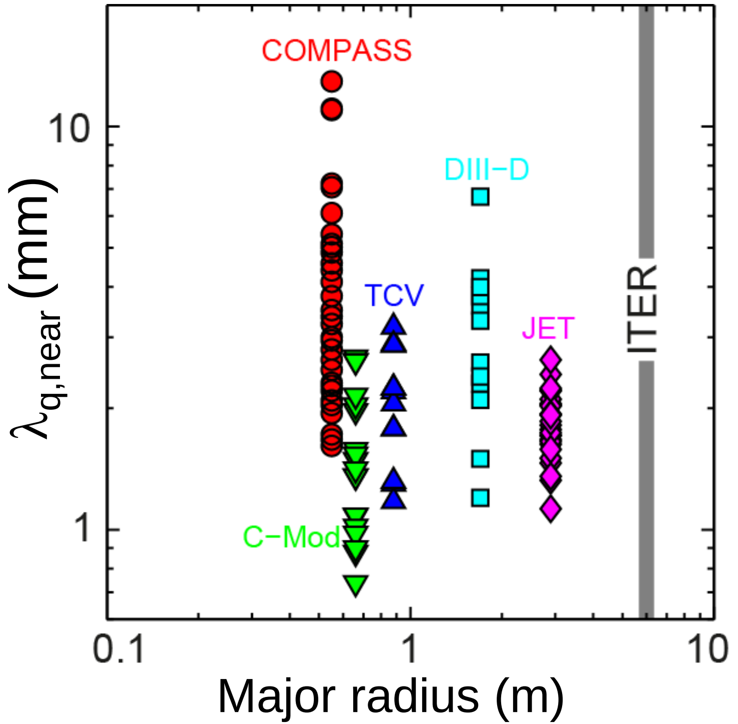
$$q_{||} = q_{||0\,main}\, e^{-\Delta r_{LCFS}/\lambda_{q,main}} + q_{||0\,near}\, e^{-\Delta r_{LCFS}/\lambda_{q,near}}$$

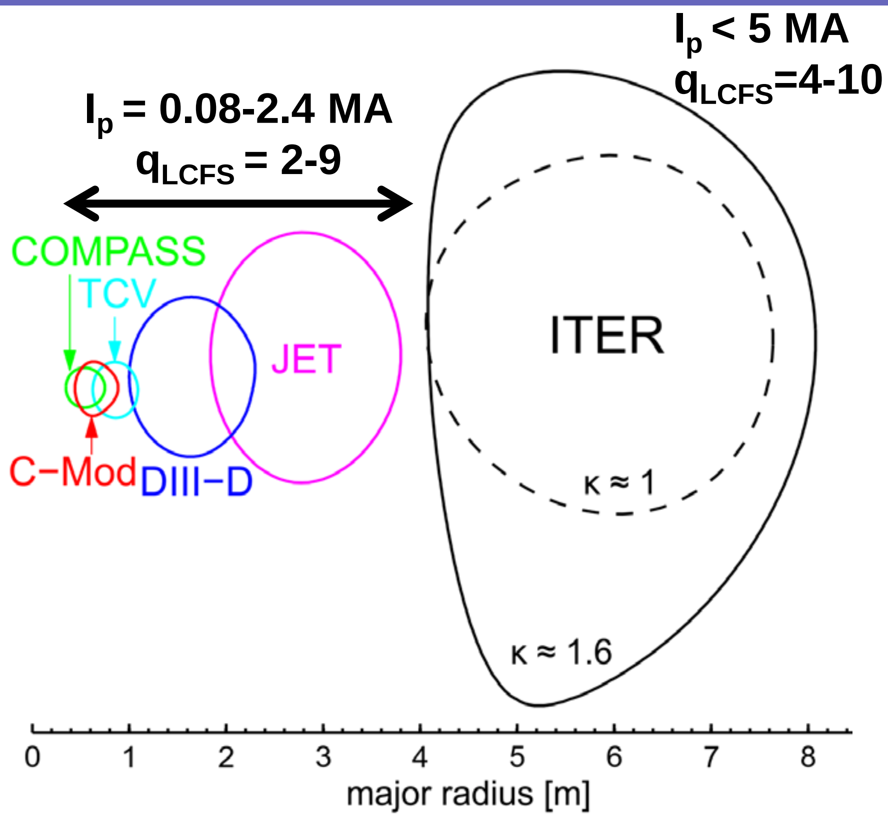
- $R_q = q_{||0,near}/q_{||0,main}$
- R_q, $\lambda_{q,near}$ and $\lambda_{q,wall}$ together determine the power carried in the narrow feature
- The "original" ITER inner wall panel toroidal profile assumed $R_q = 0$ (therefore single exponential)



How does the narrow feature scale?

- No sign of a major radius scaling



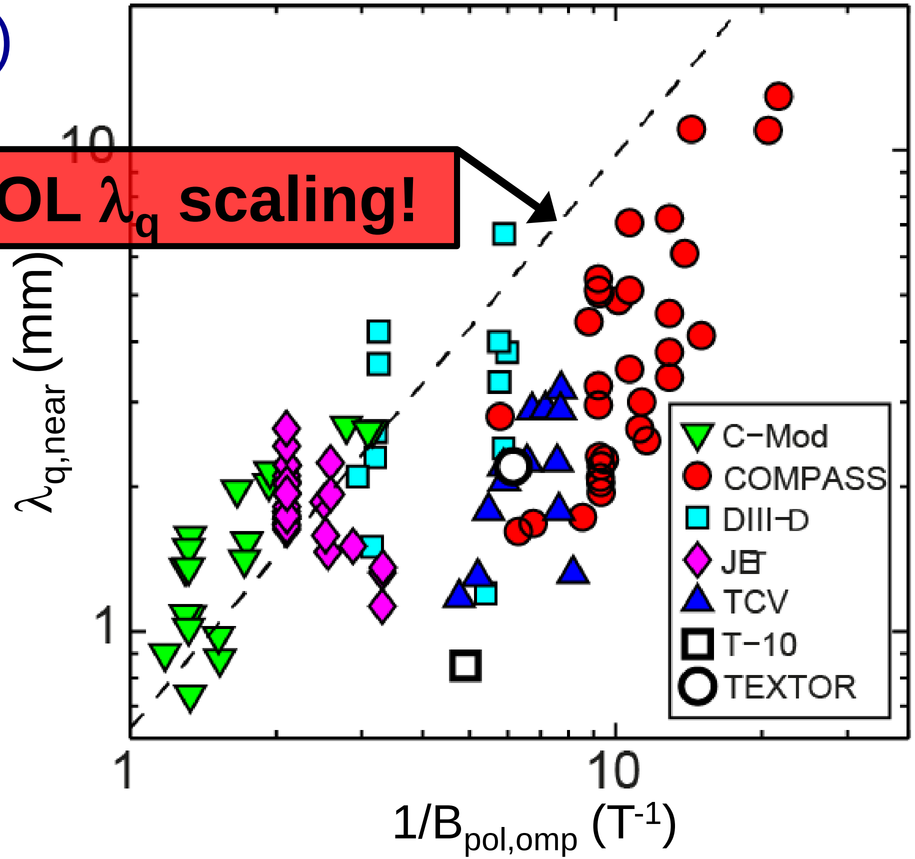


M. Kocan et al., Nucl. Fus. **55** (2015) 033019

How does the narrow feature scale?

- Trend towards $1/B_{pol,omp}$ $(1/I_p)$

H-mode divertor near SOL λ_q scaling!



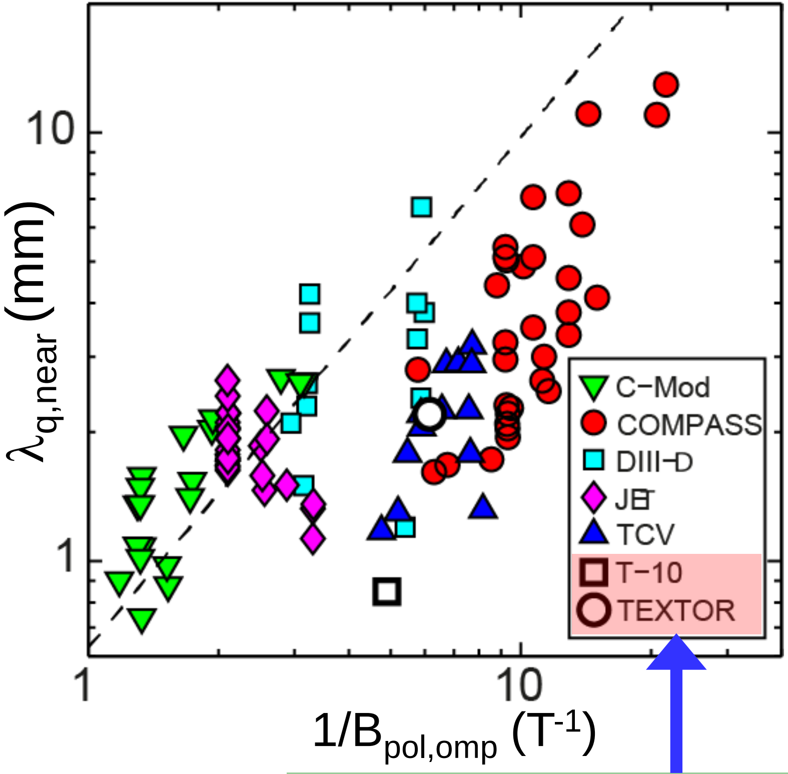
M. Kocan et al., Nucl. Fus **55** (2015) 033019

How does the narrow feature scale?

Good example of how things sometimes get “forgotten” in fusion research: we found these two points from much older papers → the “narrow feature” was seen nearly 3 decades earlier on T-10, and in 1999 in TEXTOR, both early limiter tokamaks.

We therefore have “known” about this from the start of ITER design, but the work had been forgotten …..

New work is of much higher fidelity → better and more modern diagnostics



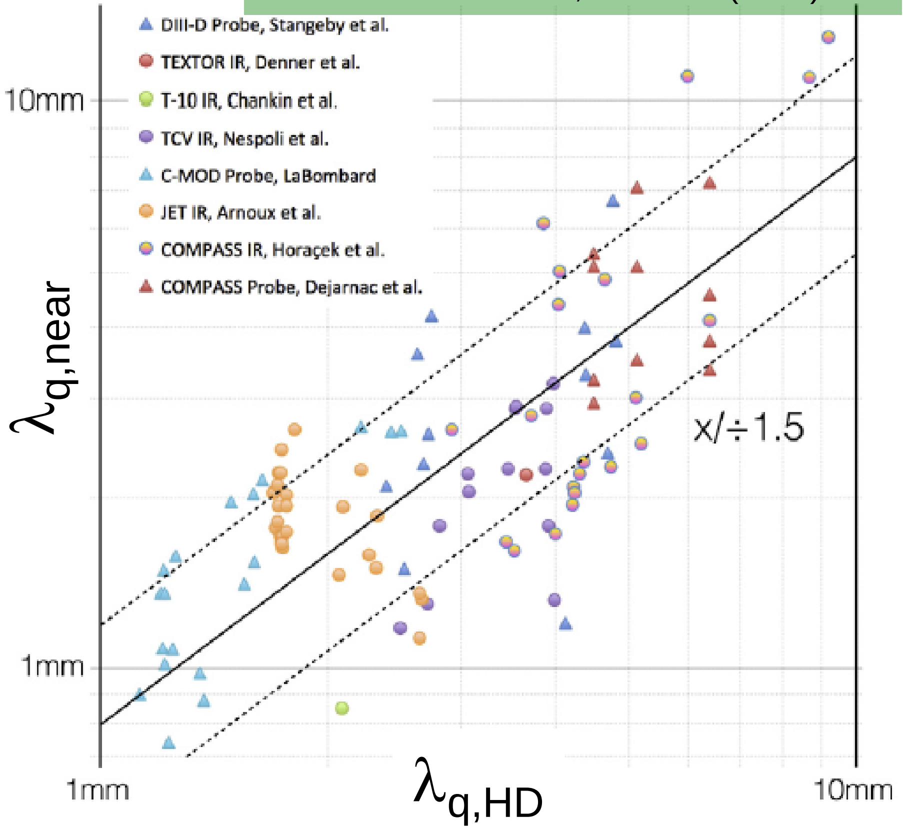
M. Kocan et al., Nucl. Fus. **55** (2015) 033019

T. Denner et al., Nucl. Fus. **39** (1999) 83
A. V. Chankin et al.,JNM **145-147** (1987) 789

How does the narrow feature scale?

- "Heuristic drift" model of Goldston (developed for H-mode, inter-ELM divertor λ_q scalings) does a good job for limiter plasma $\lambda_{q,near}$ database
  - Model balances $c_s/2$ parallel flows against ∇B and curvature B drifts → $\lambda_q \sim 2(a/R)\rho_L$
  - Should not strictly apply to limiter plasmas, but same basic physics is at work in the limiter SOL (neoclassical ion drifts and parallel flows)

R. J. Goldston, JNM **463** (2015) 397

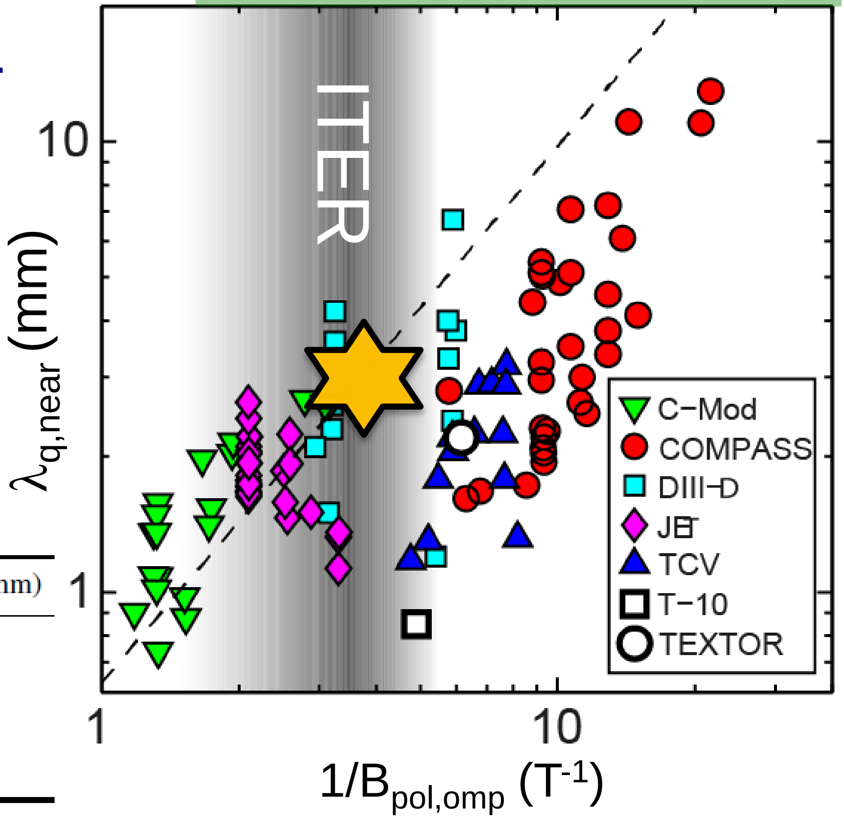


So what does all this mean for ITER?

- Physics recommended to in-vessel component engineers that the inner wall toroidal shaping be modified
  - Main SOL and narrow feature databases suggest: $R_q = q_{||0,near}/q_{||0,main} \sim 1\text{-}6$
  - $\lambda_{q,near}$ ~3 mm (omp) → ~4 mm (imp)
  - $\lambda_{q,main}$ = 50 mm (original specification)

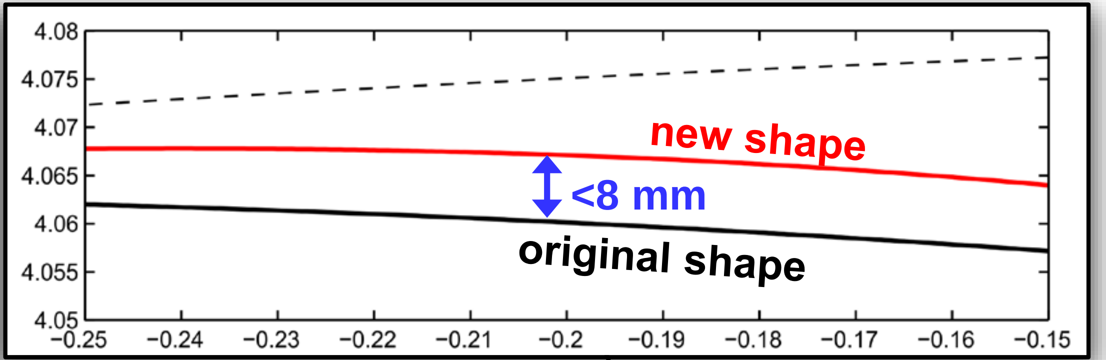
Tokamak	R_q	$\lambda_{q,near}^{OMP}$ (mm)
C-Mod	2–4	1–3
COMPASS	2–5	2–8
DIII-D	~1	2–5
JET	1–6	2–4
TCV	2–4	1–3

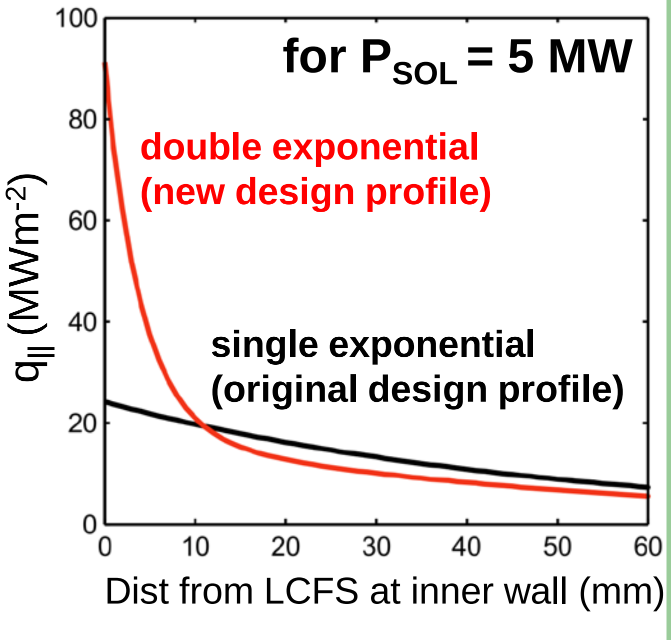
M. Kocan et al., Nucl. Fus. **55** (2015) 033019



Proposed new inner wall toroidal profile

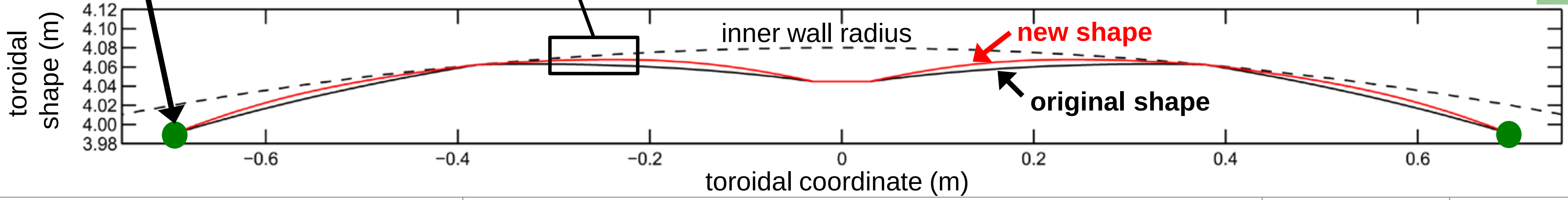
- New shape proposed based on:
  - $R_q = 4$, $\lambda_{q,near} = 4$ mm, $\lambda_{q,main} = 50$ mm





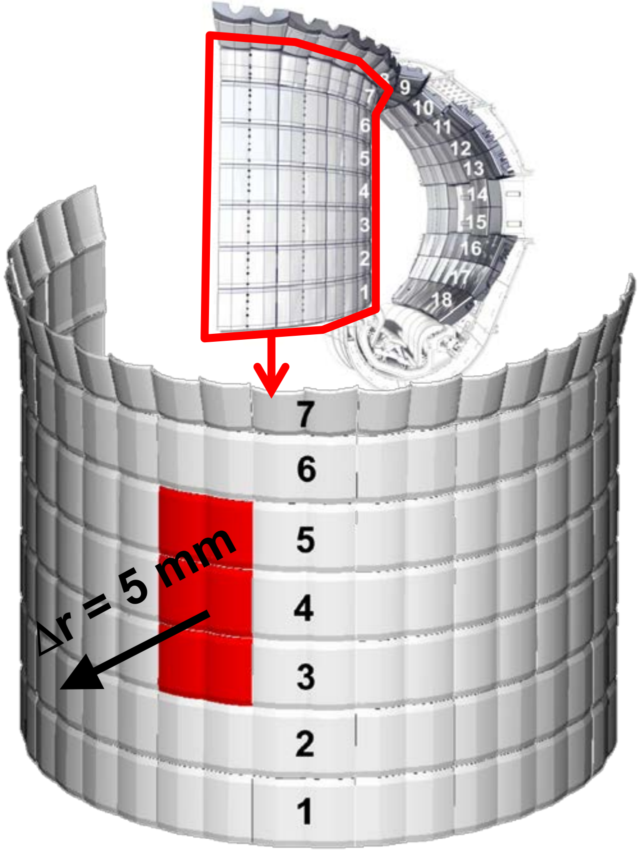
M. Kocan et al., Nucl. Fus. **55** (2015) 033019

Keep same radial set backs



Advantages and disadvantages

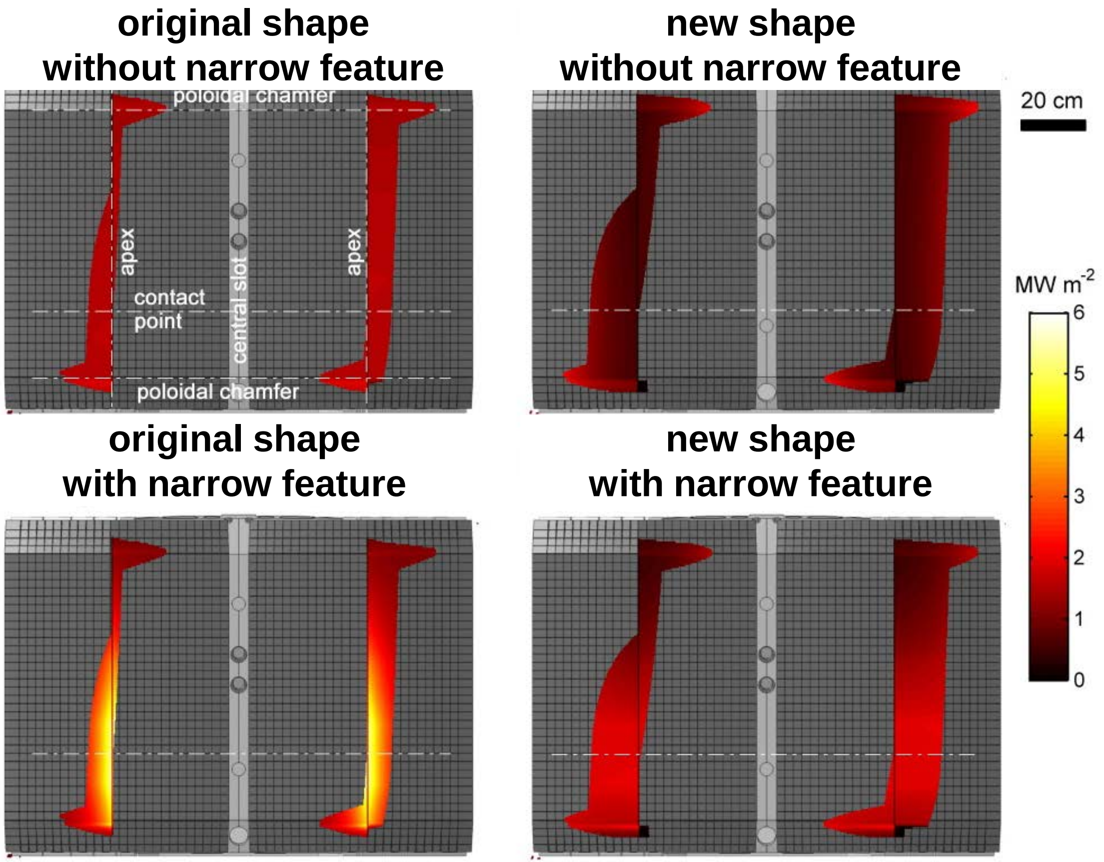
- Field line tracing to study what happens with and without the narrow SOL feature present and with and without the new toroidal shape proposal
  - Assume "worst case" allowed engineering radial misalignment between toroidally adjacent panels
  - Note that shape change only proposed for panels #3 - #5
  - Scan parameters R_q, $\lambda_{q,near}$



Surface heat loads on Panel #4

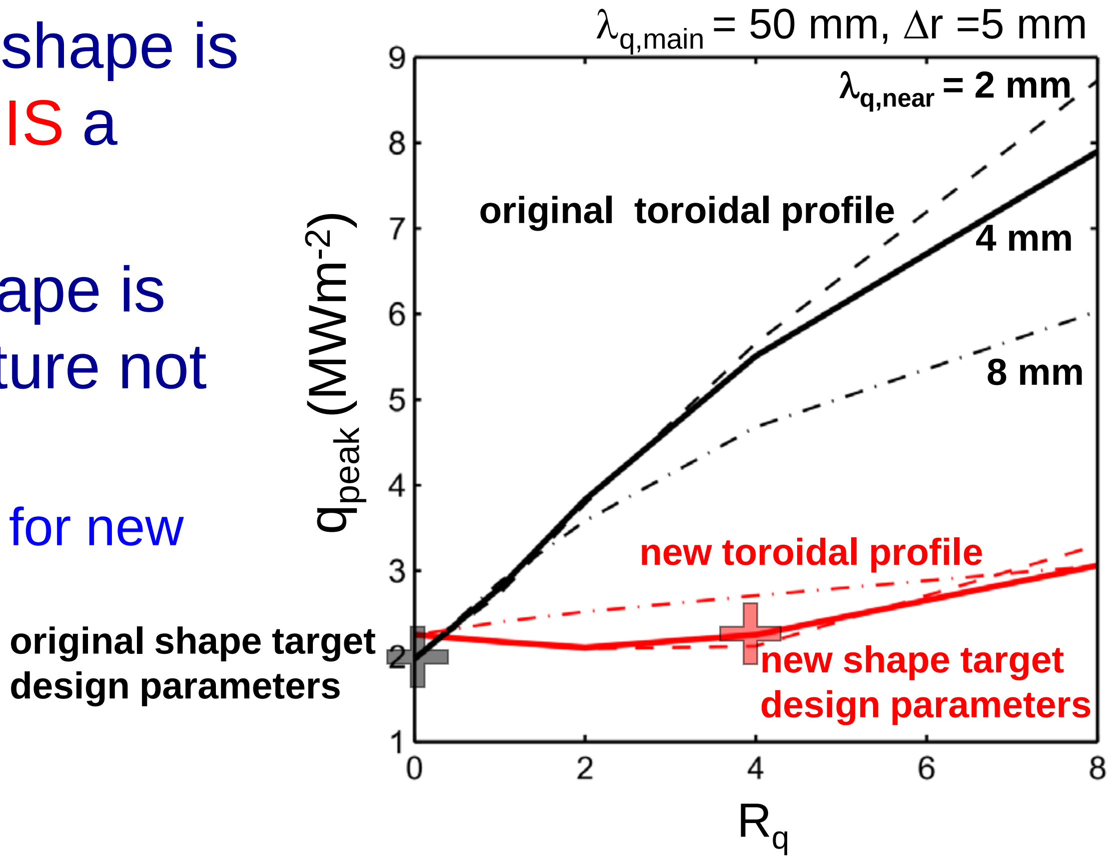
- Full bore ITER start-up plasma just before X-point formation
  - 5 MA is a worst case
  - Will be lower in reality

I_p = 5 MA
P_{SOL} = 5MW



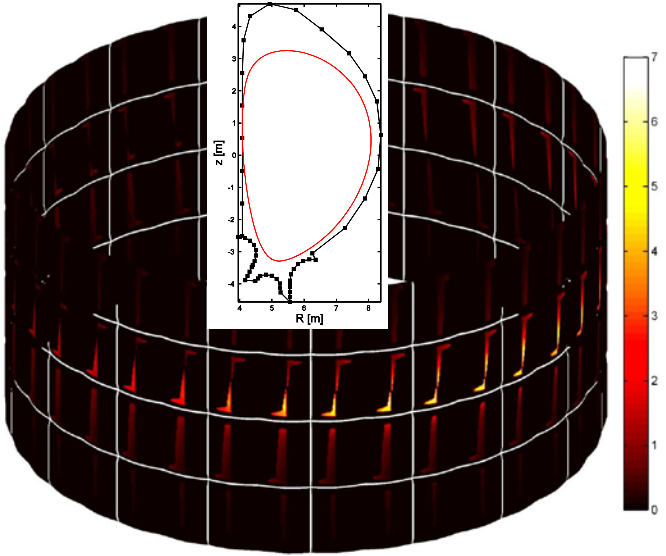
Sensitivity to R_q, $\lambda_{q,near}$

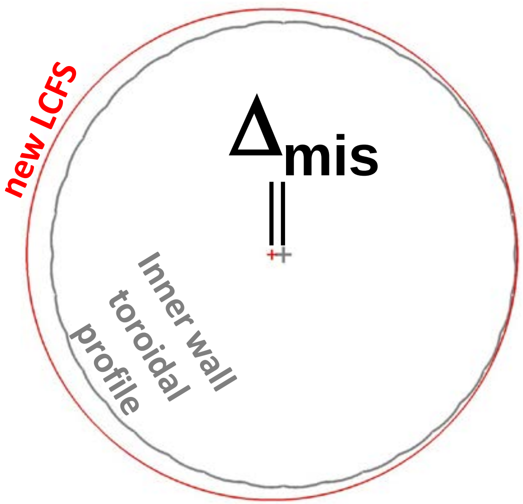
- Heavy penalty if toroidal shape is NOT changed and there IS a narrow feature
- Up to ~10% penalty if shape is changed and narrow feature not present
  - Low sensitivity to R_q, $\lambda_{q,near}$ for new shape

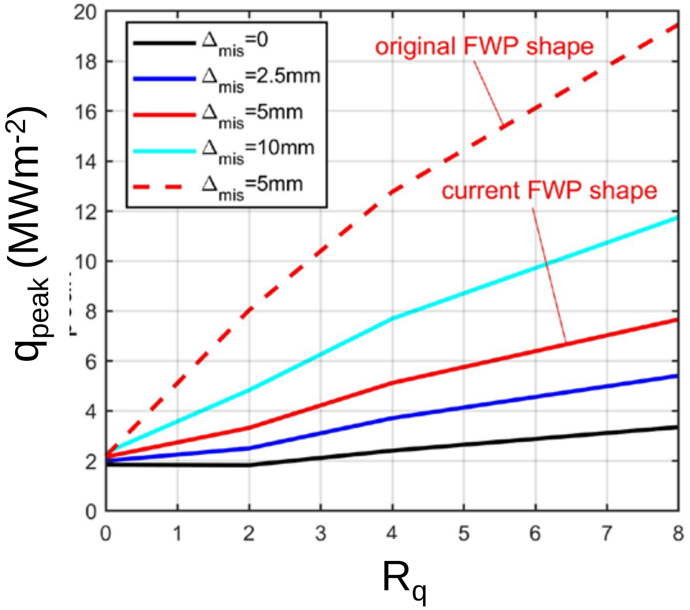


Global PFC misalignment

- These limiter studies highlight the importance of first wall panel alignment on ITER → limiter phase is long, stationary power handling of actively cooled Be panels not so high
  - Example case of n = 1 misalignment







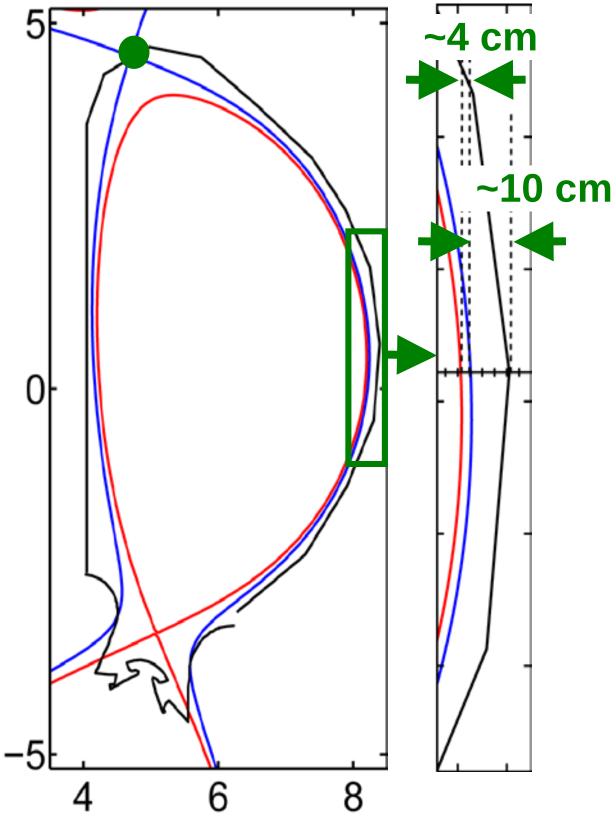
End result

- ITER Blanket Section and Domestic Agency partners agreed to change the inner wall toroidal profile shaping for panels #3 - #5
  - Good example of how physics R&D and engineering can work together during design activities
  - Hopefully some of you in the room will be around to find out if a narrow SOL heat flux feature eventually appears in ITER limiter plasmas!*

*See recent work on TCV where the narrow feature was found to disappear at high SOL collisionality: F. Nespoli et al., Nucl. Fus. **57** (2017) 126029

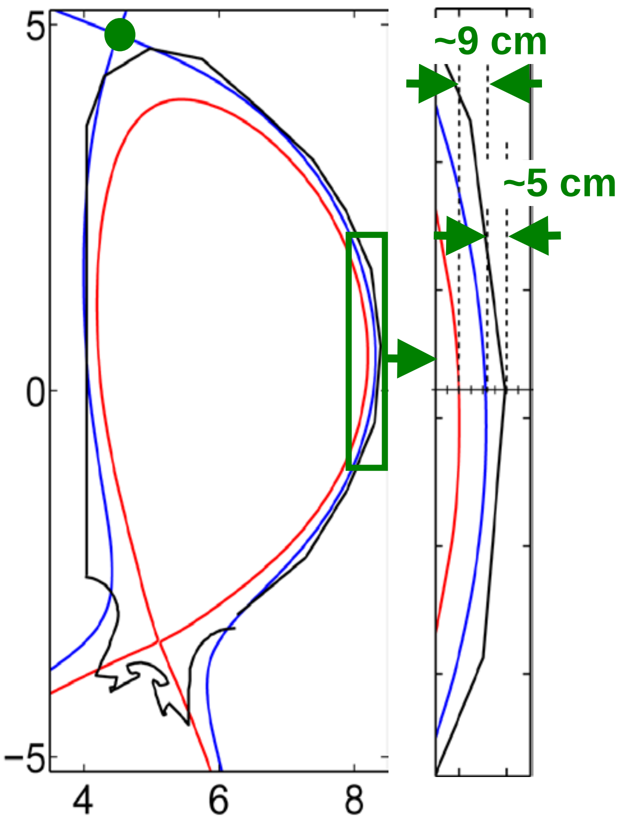
Now back to diverted equilibria …

- It is an ITER Project Requirement that the omp separation between 1st and 2nd separatrix be $\Delta r_{sep} \geq 4$ cm
  - Requirement on plasma control system
  - Quasi double-null with 2nd X-pt. just on top FW panels
  - Requirement is fixed by power handling of upper first wall panels



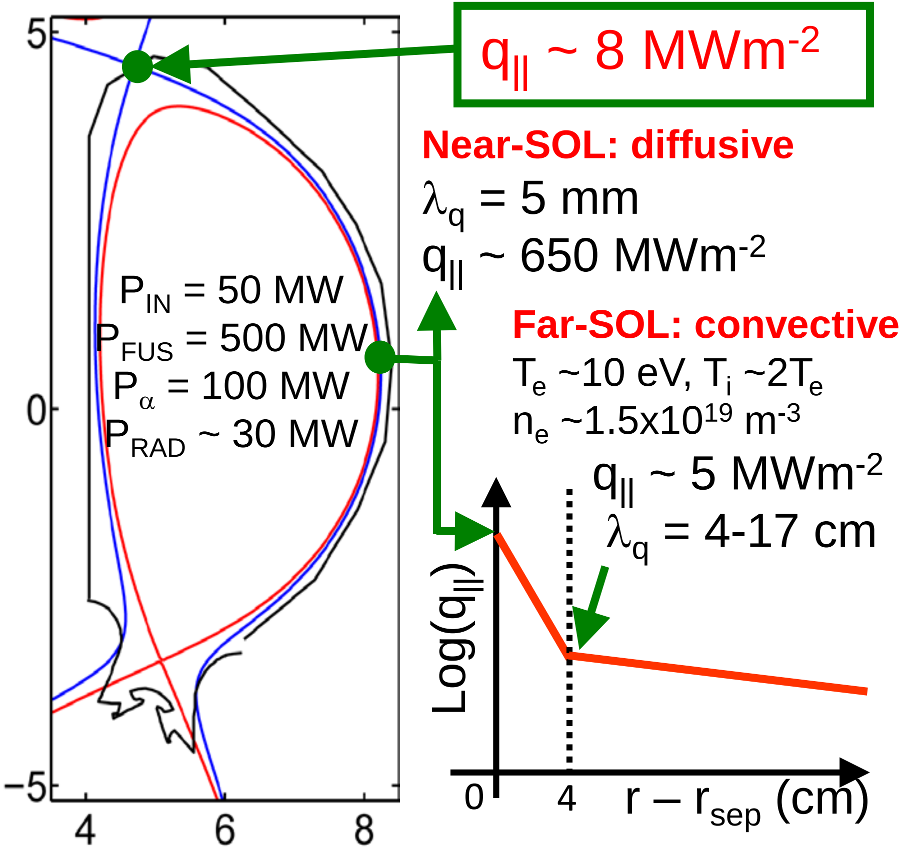
“Baseline” equilibrium

- Current reference equilibrium ($q_{95} = 3$) has $\Delta r_{sep} \sim 9$ cm
  - This is primarily to reduce heat fluxes to upper panels to lower values with some margin
- How to specify wall heat fluxes?



"Stationary" H-mode SOL heat flux

- Baseline spec for ITER is rather crude → physics basis for main chamber far SOL power fluxes still uncertain
- Take worst case:
  - Q_{DT} = 10, H-mode, high SOL density, smallest allowed Δr_{sep} = 4 cm
- Power balance:
  - $P_{SOL} \sim P_{IN}+P_{\alpha}-P_{RAD} \sim 120$ MW
- Assume SOL structure to be set by a mixture of diffusive and convective ("filamentary") components



“Stationary” H-mode SOL heat flux

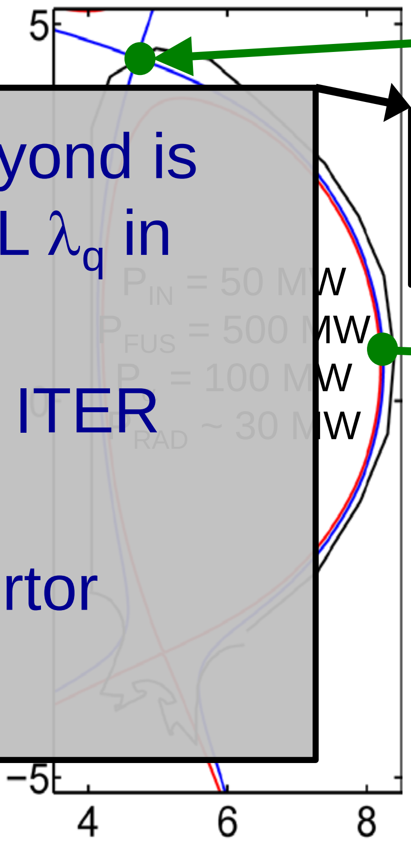
Important issue for ITER and beyond is what to assume for the near-SOL λ_q in high current plasmas

Baseline assumption for original ITER Heat Load Spec is λ_q = 5 mm

But this is a problem for the divertor

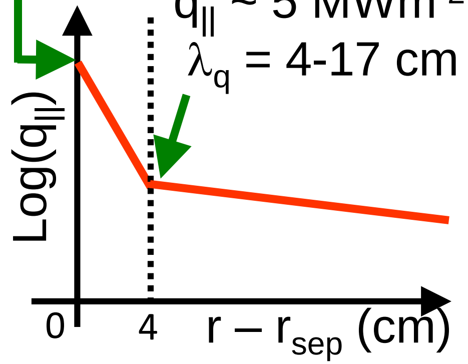
See my second lecture

$q_{||}$ ~ 8 MWm^{-2}

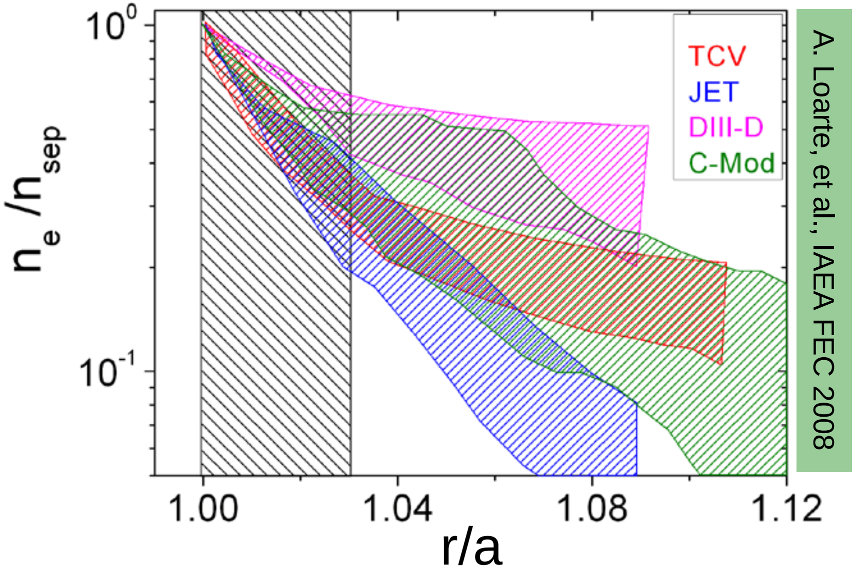


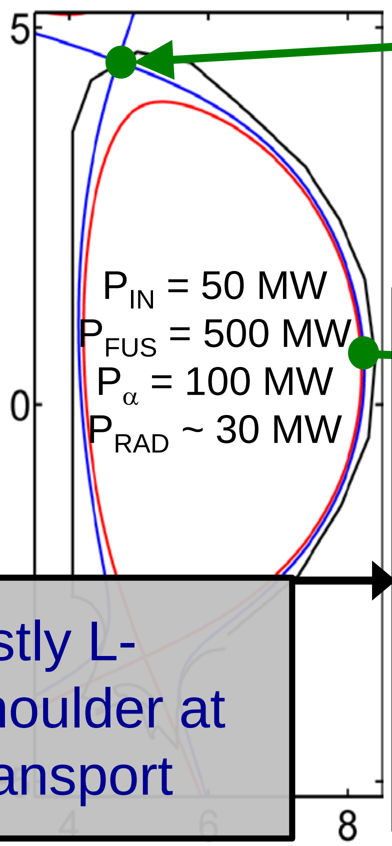
Near-SOL: diffusive
λ_q = 5 mm
$q_{||}$ ~ 650 MWm^{-2}

Far-SOL: convective
T_e ~10 eV, T_i ~2T_e
n_e ~1.5x10^{19} m^{-3}



"Stationary" H-mode inter-ELM SOL heat flux





Near-SOL: diffusive

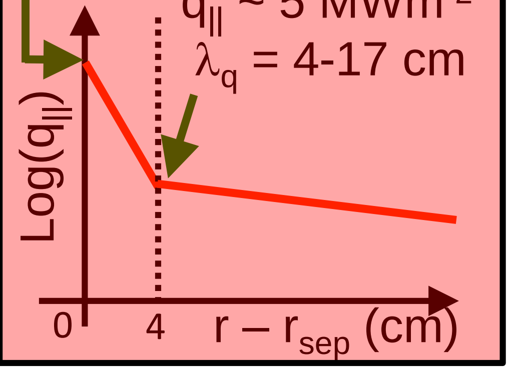
λ_q = 5 mm

$q_{||}$ ~ 650 MWm^{-2}

Far-SOL: convective

T_e ~10 eV, T_i ~2T_e

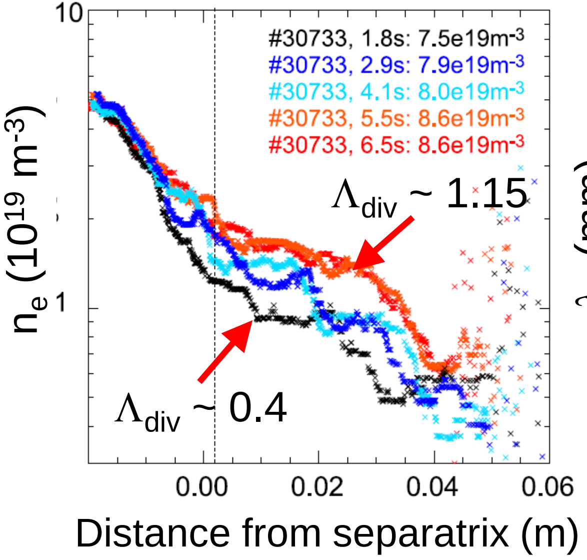
n_e ~1.5x10^{19} m^{-3}



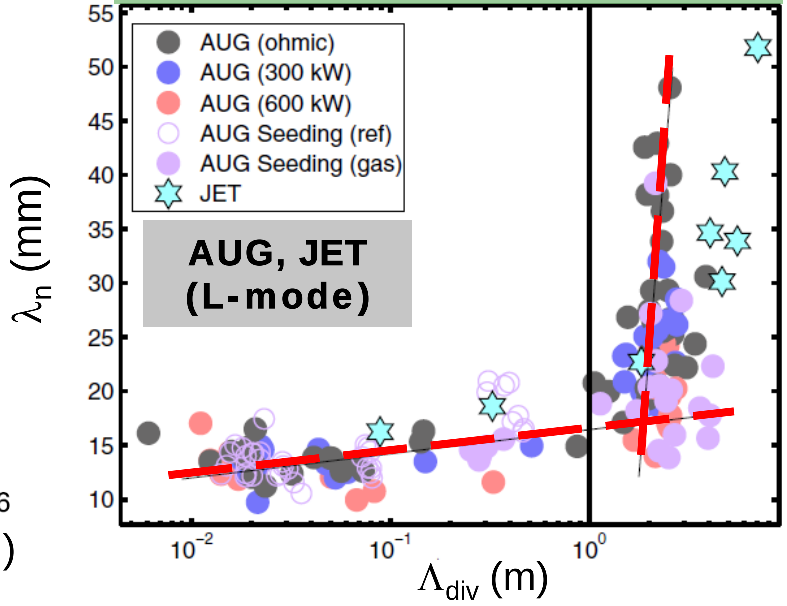
Based on experimental evidence (mostly L-mode) for formation of SOL density shoulder at high n_e caused by rapid filamentary transport

Origin of far SOL shoulders?

- Work on AUG and JET found direct link between divertor collisionality ($\Lambda_{div} \propto L_{||} n_e T_e^{-3/2}$) and cross-field SOL filament transport in L-mode
  - Upstream SOL broadens when $\Lambda_{div} > 1$
  - But note big caveat: AUG on vertical divertor targets, outer strike on horizontal target in JET

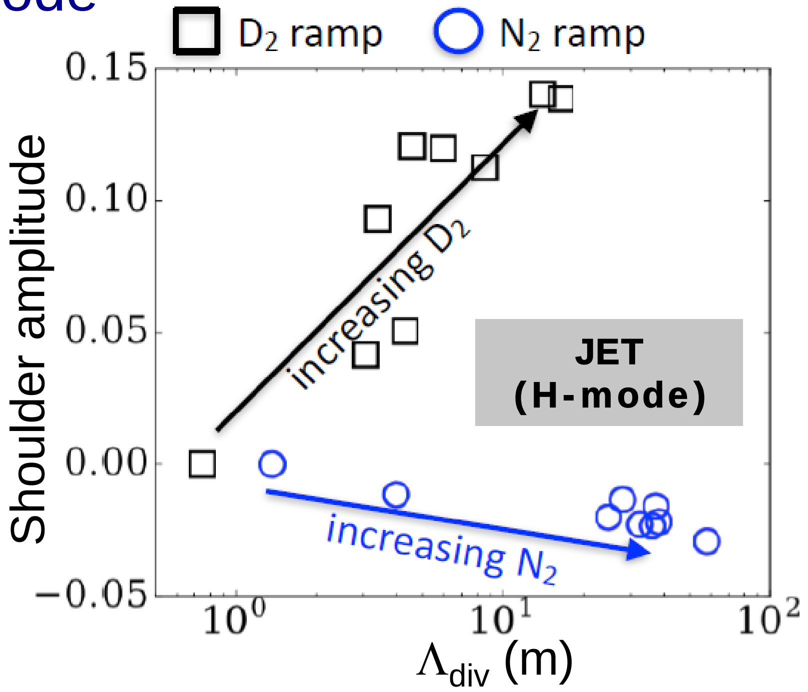


D. Carralero et al., PRL **115** (2015) 215002
D. Carralero et al., JNM **463** (2015) 123



Confusing situation still

- Recent measurements on JET conclude that Λ_{div} not a sufficient condition for shoulder formation in H-mode
  - Increasing collisionality with extrinsic seeding has little or no effect on upstream density profile
  - Broadening always seen with D_2 puffing → effect may be related to "neutral clogging" of the flux tube
  - Broadening not seen on JET in with outer strike on vertical target!



See A. Wynn et al., Nucl. Fus. **58** (2018) 056001

What to do for ITER?

In the 10 years since ITER Heat Load Specs were first established, precise extrapolation to ITER still not possible, and still not clear that the main SOL will broaden in H-mode at high density and divertor dissipation on ITER.

Current ITER specs:

Power to ITER FW <20 MW (20% P_{SOL})

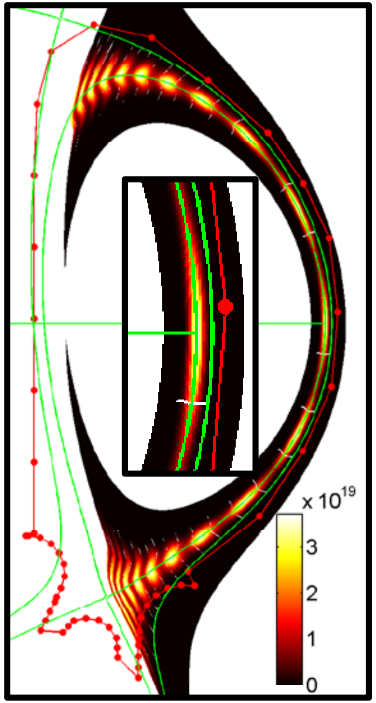
Particle flux to ITER first wall < $1\times10^{24}s^{-1}$ (10% Γ_{div})

Some evidence that main wall interactions could be reduced in a DEMO under some conditions by increasing wall gaps M. Beckers et al., NME **12** (2017) 1163

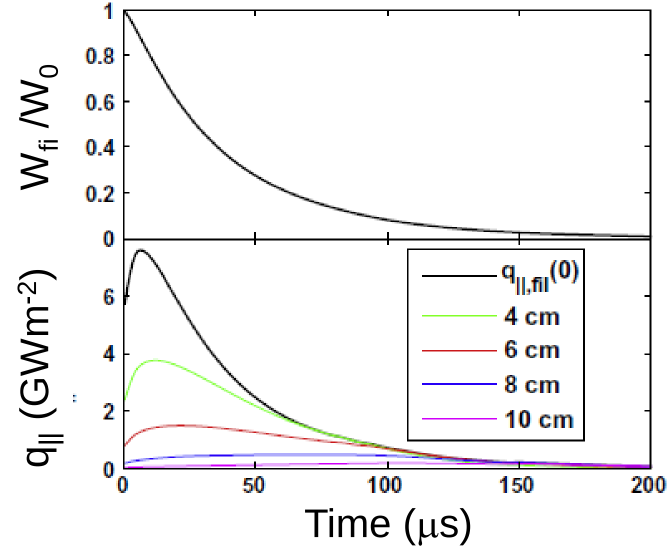
The same tentative conclusion reached by D. Carralero on the basis of AUG experiments D. Carralero et al., Nucl. Fus. **57** (2017) 056044

SOL ELM driven heat flux

- Total wall load is sum of stationary and ELM heat flux
  - ITER load spec is for **mitigated ELMs** only → uncontrolled ELMs at Q_{DT} = 10 not tolerable (FW melts)
  - Launch ELM filaments from omp separatrix with given radial and poloidal size, mode number and use fluid model for parallel losses
  - Compute avge. heat load for given f_{ELM} at intersection with FW



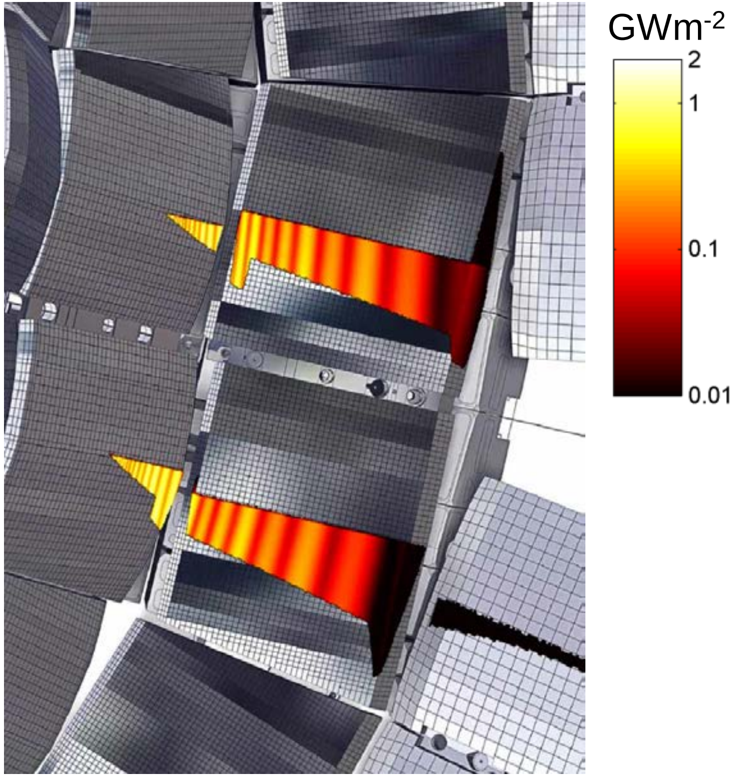
$$q_{\|,fil}\left(\Delta r_{sep1}, z, t\right) = -\frac{dW_{fil}}{dt}\frac{1}{4\pi\sigma_r\sigma_z} e^{-\frac{\left(\Delta r_{sep1} - tv_r\right)^2}{2\sigma_r^2}} e^{-\frac{\Delta z_{OMP}^2}{2\sigma_z^2}}$$



M. Kocan et al., J. Nucl. Mater. **463** (2015) 39

SOL ELM driven heat flux

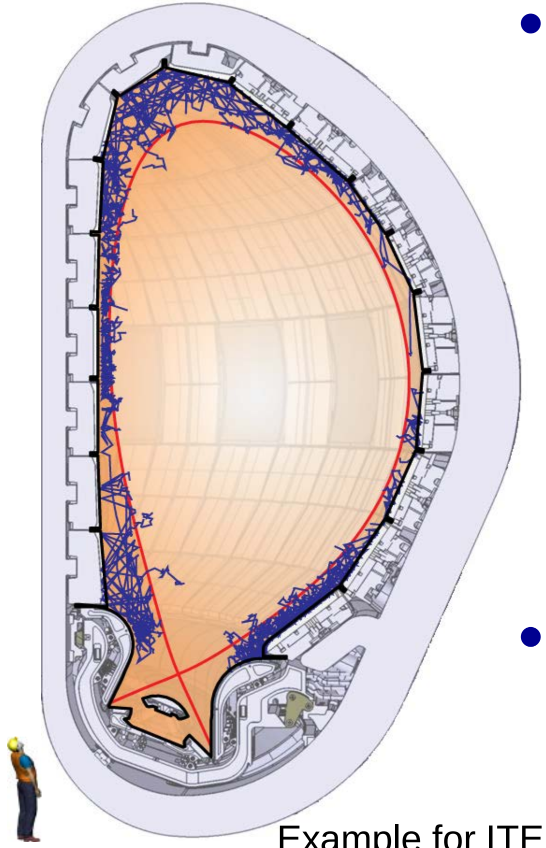
- E.g.: mitigated ELM, ΔW_{ELM} = 0.6 MJ, 10 filaments, σ_r= 3 cm, σ_z = 45 cm



- 150 ms interval during a single ELM, full field line trace on 3D first wall
  - For f_{ELM} = 70 Hz, ELM averaged heat loads at top panels:
    $\langle q_{||,ELM} \rangle$ ~15 MWm^{-2}, $\Delta r_{sep,omp}$ = 4 cm
    $\langle q_{||,ELM} \rangle$ ~ 9 MWm^{-2}, $\Delta r_{sep,omp}$ = 9 cm
  - ELM averaged heat flux to far SOL dominates the static (inter-ELM) loads

See also poster by H. Anand at this school for more details of how these ELM loads are computed

Stationary “perpendicular” fluxes



- Plasma contact with the walls generates CHARGE EXCHANGE (CX) neutrals

$$D^+ + D^o \longrightarrow D^o + D^+$$

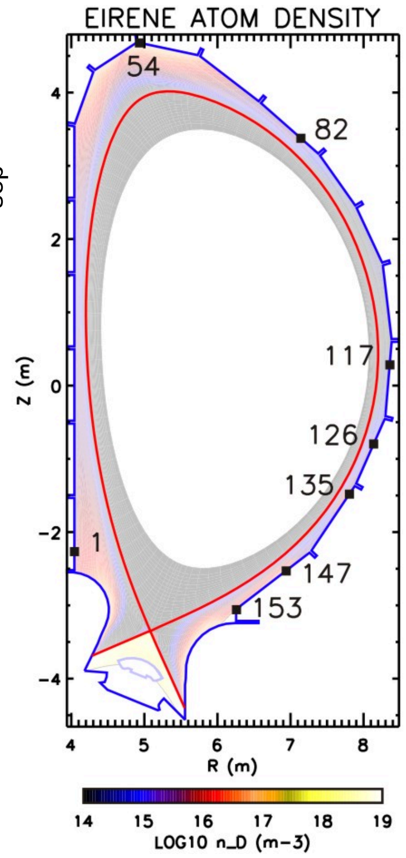
Hot ion — Cold neutral — Hot neutral — Cold ion

- Photons from core will load the walls quasi-uniformly

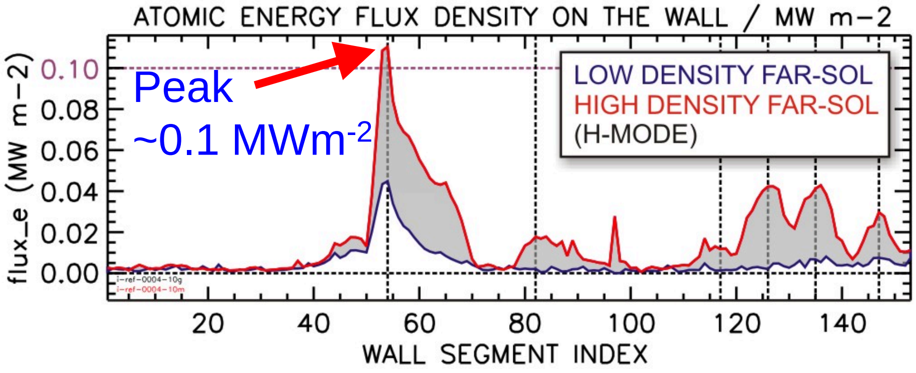
Example for ITER from the Eirene neutral transport code (divertor sources off)

Charge-exchange energy loads

$\Delta r_{sep} = 4$ cm



- Use SOLPS-OSM boundary codes to generate 2D plasma background right out to the walls (include H-mode pedestal)
  - Eirene code for CX fluxes, which are strongly dependent on far-SOL plasma values (e.g. shoulder formation)



Importance of CX erosion

- May well set the lifetime in a reactor for conventional walls

P. C. Stangeby et al., J. Nucl. Mater. **415** (2011) S278

Device	P_{HEAT} (MW)	t_{annual} (s/yr)	$E_{load/yr}$ ($P_{HEAT} \times t_{annual}$) (TJ/yr)	Net wall erosion (kg/yr)		
				Beryllium	Carbon	Tungsten
DIII-D	20	10^4	0.2	0.11	0.08	0.16
JT-60 SA	34	10^4	0.34	0.19	0.15	0.27
EAST	24	10^5	2.4	1.2	0.82	1.8
ITER	100	10^6	100	**64**	44	92
Reactor	400	2.5×10^7	10,000	5300	3700	**7900**

- 7900 kg (~ 2.5 mm) erosion per full burn year for a DEMO with R = 9 m assuming no local re-deposition, no thermal plasma interaction and poloidally uniform CX flux

This is what distinguishes the SOL from the divertor

Total perpendicular load for ITER

- CX: assume, conservatively **0.1 MWm^{-2}** on all FW panels regardless of poloidal position
- PHOTONS: assume at most: $P_{IN} + P_{\alpha} - P_{RAD,core} \geq P_{LH}$
  - P_{LH} ~ 70 MW for DT H-mode access at 5.3 T, $1x10^{20}$ m^{-3}
  - $P_{RAD,core} \leq$ 80 MW → ~0.12 MWm^{-2} on FW
  - Add factor 2 for maximum poloidal peaking → **~0.25 MWm^{-2b}**
- Total maximum recommended CX + photon FW load:

0.1 + 0.25 = 0.35 MWm^{-2} on any given FW panel in steady state at Q_{DT} = 10 → highly conservative since unlikely that so much core radiation can be supported for long and still remain in H-mode

Next lecture: the divertor ……
(enjoy coffee break!)