Multi-tokamak scaling of main SOL heat flux width in limiter plasmas

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Introduction

Discharge on the ITER tokamak will preferably start and end up with plasma touching the inboard (HFS) limiter before shaping into the divertor configuration. In order to avoid exceeding the designed limit of the Beryllium enhanced heat flux limiters ($4.7MW/m^2$), its shape has to be optimized in order to receive the power everywhere equally. The optimum shape has been derived based on observation that the SOL radial profile of the power flux decays mostly exponentially, $q_{||} = q_{0||} exp(-r/\lambda_q^{\mathrm{omp}})$. Therefore, a logarithmic shape

$$y = -\lambda_q^{\mathrm{omp}} \cdot \ln(1 - \frac{C \cdot x}{\lambda_q^{\mathrm{omp}}}), \tag{1}$$

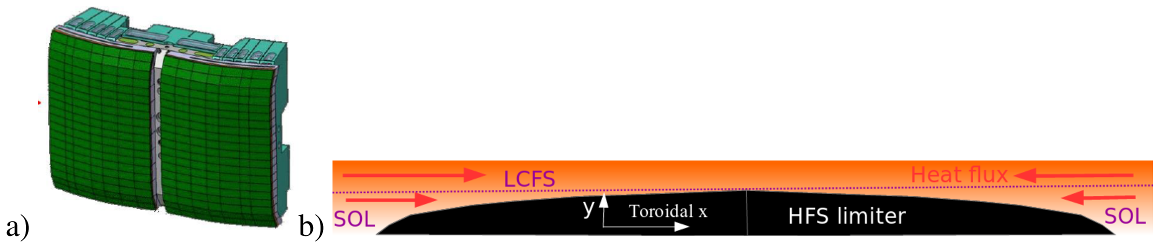

Figure 1: *a) 3D drawing of the ITER inner wall limiter [1]. b) Limiter optimized for constant power everywhere [1] for SOL plasma characterised by exponential decay length* λ_q^{omp}.

	R [m]	a [m]	I_p[kA]	B [T]	data	heating	isotope	Meas. of T_e	processed by
ITER	6.0	2.0	3500-7500	5.3	3	NBI	H,D,He	-	Shimada
JET	2.8	0.98	1500,2500	2.8	33	none	D	swept Langmuir	Silva, Horacek
Tore Supra	2.2	0.65	500-1200	2.6-4.1	121	none	D	tunnel + RFA	Gunn
DIII-D	1.7	0.6	600-1200	1.9	23	none	D	harmonic technique	Tsui, Rudakov
C-Mod	0.68	0.22	400-1100	4-7	19	none	D	scanning Mirror [6]	LaBombard
KSTAR	1.78	0.47	400	1.99	1	NBI	D	swept Langmuir	J.-G. Bak
TEXTOR	1.73	0.46	$\pm$ 200-400	$\pm$ 1.3-2.6	55	NBI	D	triple probe	Horacek
EAST	1.85	0.46	300	1.96	2	none	D	triple probe	G.S. Xu
HL-2A	1.67	0.36	100-220	1.36	39	ECH+NBI	D	triple probe	L. Nie, Wang
FTU	0.94	0.28	250-500	2.7-7.5	3x9	none	D	swept Langmuir	Maddaluno, Pericoli
COMPASS	0.55	0.2	80-180	1.15	91	none	H, D	swept Langmuir, BPP	Horacek, Seidl
CASTOR	0.4	0.08	9	1.3	3	none	H	swept Langmuir	ref. [2]

Table 1: *Overview of parameters of the used tokamak plasmas. TEXTOR includes both directions of B and* I_p*. The data being processed by different persons and temperature measured by different techniques yield probably to systematic errors in* λ_q^{omp}.

shown in Figure 1, yields theoretically an almost constant power flux across the entire surface of the limiter. Scaling of λ_q^{omp} is, however, available only for diverted plasmas, specified in the ITER Physics basis [4]. Experiments on Tore Supra demonstrated [5] that this scaling is not valid for circular limiter plasmas and that a single parameter determines λ_q^{omp}: the ohmic power.

Recently, a theory-based model capable of a credible prediction of λ_q^{omp} has been developed [3], concluding that λ_q^{omp} should scale only with q_{95}, ρ^*, ν_{gbs}. Those parameters rely, however, on probe measurements of T_e, n_e at LCFS which is subject of large error on current tokamaks (especially T_e and exact localization of LCFS) as well as n_e^{LCFS} in ITER is subject to factor of 2 uncertainty. Alternative (perhaps better) choice is thus a scaling based on engineering or those global plasma parameters well-known for ITER.

Scaling based on a single tokamak [5] is not sufficient for extrapolation to a larger tokamak, ITER. Therefore, we collected data from 450 probe strokes on 11 tokamaks worldwide, varying from the smallest plasmas (CASTOR with plasma volume 0.06 m^3, $I_p = 9$kA) up to JET with 70 m^3, $I_p = 2500$kA. Overview of the used tokamaks is compiled in Table 1.

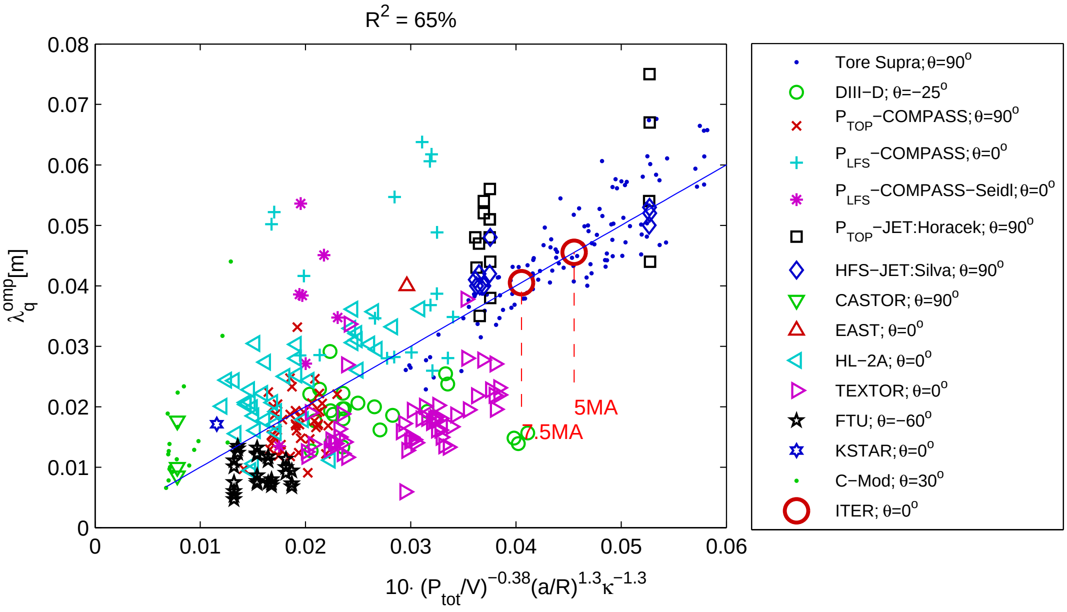

Figure 2: *An engineering scaling. The corresponding ITER predictions lie on the line.*

Experimental results

The experiments were performed with various reciprocating probes. Each measured radial profile of the ion saturation current density $j_{\rm sat}$ and electron temperature $T_{\rm e}$. The parallel heat flux is then $q \propto j_{\rm sat}T_{\rm e}$ and the desired $\lambda_q^{\rm omp}$ is then the decay length of the radial profile fit.

16 global plasma parameters were measured at the same time, thus constructing a valuable matrix of 16×450 data-points. We use the statistical package Gretl to perform multi-parameter least-squares fit with robust standard errors, weighting each value of $\lambda_q^{\rm omp}$ by the radial-profile fit quality (R^2 combined with $\lambda_q^{\rm omp}/\lambda_q^{error}$ ratio).

We find that the dominant parameter of the scaling is either the total input power (consistent with [5]), $P_{\rm tot}$, $P_{\rm SOL}$, $I_{\rm p}$, or LCFS plasma pressure β, which vary from one discharge to another within each tokamak. Due to natural high mutual correlation of $P_{\rm tot}$, $P_{\rm SOL}$, $I_{\rm p}$, β only one of those can be used in a particular scaling. All the other parameters then only slightly refine $\lambda_q^{\rm omp}$ for a particular tokamak. Combinations with high mutual correlations must be excluded.

Prediction for ITER

A dozen of reasonable scalings are shown in Table 2 using various combinations of global plasma parameters. **Fortunately, we observe that those scalings yield very similar prediction for ITER, even though using very different parameter combinations. Mapped to inboard midplane (with flux expansion of 1.6), the worst case (7.5 MA) corresponds to optimum toroidal shaping of the FW panel with $\lambda_q^{\rm omp} = 7 \pm 2$cm, thus consolidating the initial design**

Scalings (dimensionless or in SI units)	λ_q^{omp}[mm] ITER prediction 2.5MA	5.0MA	7.5MA	R^2
$72.1 \cdot q_{95}^{-0.81}(a/R)^{3.64}R[m]^{0.54}\nu_{gbs}^{-0.18}\rho^{*-1.13}I_p[A]^{-1}$	70 ± 17	49 ± 12	43 ± 10	77%
$q_{95}^{-0.72}(a/R)^{2.96}\nu_{gbs}^{-0.21}\rho^{*-1.24}I_p[A]^{-0.81}$	59 ± 13	44 ± 10	40 ± 9	73%
$0.00158 \cdot \beta^{-0.33}(a/R)^{0.86}\kappa^{0.59}$	57 ± 13	53 ± 14	43 ± 11	73%
$4.07 \cdot (I_p/A[A/m^2])^{-0.4}(a/R)^{1.54}\kappa^{-1.34}\rho^{*-0.29}$	76 ± 21	49 ± 15	44 ± 12	70%
$10 \cdot (P_{tot}/V[W/m^3])^{-0.38}(a/R)^{1.3}\kappa^{-1.3}$	67 ± 19	47 ± 15	42 ± 13	65%
$501 \cdot (P_{tot}/V[W/m^3])^{-0.28}(a/R)^{1.46}\kappa^{-1.11} < n_e > [m^{-3}]^{-0.11}$	76 ± 22	50 ± 16	44 ± 13	64%
$20.6 \cdot (a/R)^{2.04}R[m]^{0.94}\kappa^{-0.82}P_{tot}[W]^{-0.34}$	67 ± 19	47 ± 15	41 ± 12	63%
$7.08 \times 10^6 \cdot (I_p/A[A/m^2])^{-0.77}(a/R)^{1.61}\kappa^{-1.8} < n_e > [m^{-3}]^{-0.14}B_{pol}[T]^{0.48}$	93 ± 30	53 ± 18	48 ± 15	62%
$6.76 \times 10^4 \cdot (I_p/A[A/m^2])^{-0.89}(a/R)^{1.5}\kappa^{-1.89}B_{pol}[T]^{0.52}$	91 ± 29	55 ± 18	50 ± 16	61%
$0.00108 \cdot q_{95}^{0.29}\nu_{gbs}^{-0.1}\rho^{*-0.29}$	86 ± 24	70 ± 19	57 ± 16	60%
$1.72 \cdot (P_{tot}/V[W/m^3])^{-0.28}q_{95}^{0.17}(a/R)^{0.96}$	88 ± 26	73 ± 21	56 ± 16	58%
Average	$\mathbf{76 \pm 21}$	$\mathbf{54 \pm 16}$	$\mathbf{46 \pm 13}$	

Table 2: *The final scalings and consequent ITER predictions (the 95%-confidence intervals) of the SOL width at outboard midplane for the three start-up scenarios.*

choice of 5 cm. Note that much steeper gradients have been found in the region near separatrix (λ_q^{omp} shorter by order of magnitude) at high-field side of tokamaks, described in [7]. Both predictions of the near and the here-scaled main SOL decay lengths are required for best ITER HFS limiter, designed finally in [8]. Details of this work can be found in [9].

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