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Multi-machine Scaling of Main SOL Parallel Heat Flux Width in Tokamak Limiter Plasmas

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

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Introduction

Each discharge on the ITER tokamak will start and end up with plasma touching the inboard (HFS) limiter before shaping into the divertor configuration. In order to avoid exceeding the designed heat flux limit of the Beryllium limiter ($2.5 MW/m^2$), 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

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

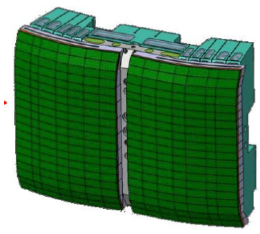
a)  b) 

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]	datapoints	heating	isotope	Meas. of T_e	processed by
ITER	6.0	2.0	3500-7500	5.3	3	NBI	H,D,He,T	-	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 [8]	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	[2, 3, 4, 5]

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.

Nowadays, there is no theory-based model capable of a credible prediction of λ_q^{omp} for ITER. The solution is to use a scaling based on well-known plasma parameters. Such scaling is written in the ITER Physics basis [6], however, based on experiments with diverted plasmas only. Experiments on Tore Supra demonstrated [7] that this scaling is not valid for circular limiter plasmas and that a single parameter determines λ_q^{omp}: the ohmic power.

Scaling based on a single tokamak is not credible 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) up to JET with 70 m^3. Overview of the used tokamaks is compiled in Table 1.

Experimental results

The experiments were performed with various reciprocating probes. Each measured radial profile of the ion saturation current density j_{sat} and electron temperature T_e. The parallel heat flux

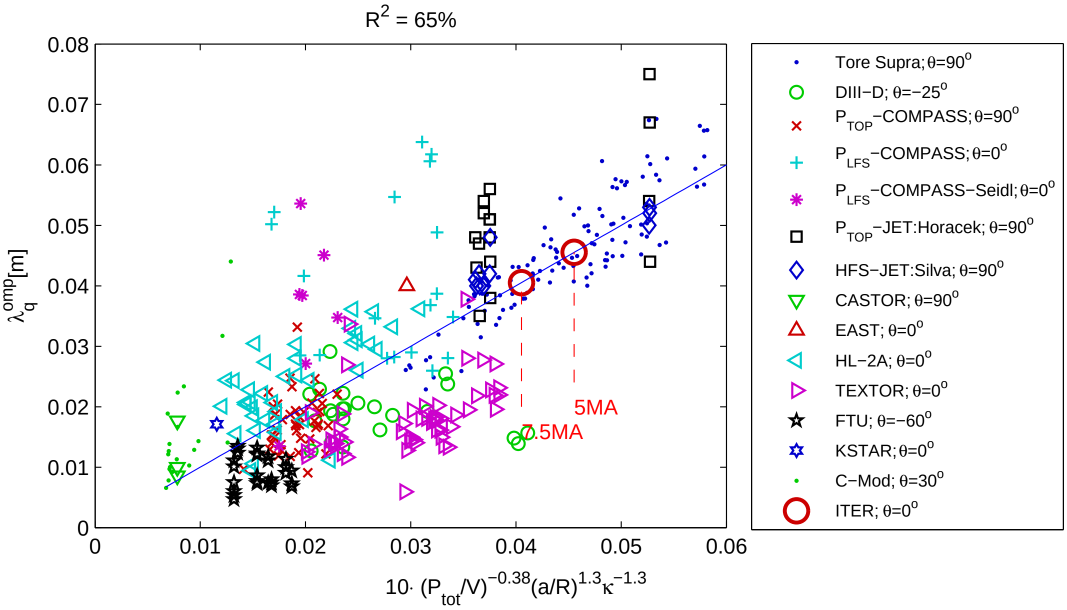


Figure 2: *The most credible scaling. The corresponding ITER predictions lie on the line.*

is then $q \propto j_{\mathrm{sat}} T_{\mathrm{e}}$ and the desired λ_q^{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 λ_q^{omp} by the radial-profile fit quality (R^2 combined with $\lambda_q^{\mathrm{omp}}/\lambda_q^{error}$ ratio).

We find that the dominant parameter of the scaling is either the total input power, P_{tot}, P_{SOL} or plasma current I_{p}, which vary from one discharge to another within each tokamak. Due to natural high mutual correlation of P_{tot}, P_{SOL}, I_{p}, only one of those can be used in a particular scaling. All the other parameters then only slightly refine λ_q^{omp} for a particular tokamak and, again, combinations with high mutual correlations must be excluded.

Prediction for ITER

We find a dozen of reasonable scalings using various combinations of global plasma parameters. The scaling with highest credibility (R^2 value) is $\lambda_q^{\mathrm{omp}}[m] = 10(P_{\mathrm{tot}}/V\,[W/m^3])^{-0.38}(a/R)^{1.3}\kappa^{-1.3}$, dependent on input power density, aspect ratio and elongation. It yields λ_q^{omp}= [7, 5, 4] cm for the three reference limiter plasma current scenarios I_p = [2.5, 5.0, 7.5] MA specified in the ITER Heat and Nuclear Load Specifications. Mapped to inboard midplane, the worst case (7.5 MA) corresponds to optimum toroidal shaping of the FW panel with $\lambda_q^{\mathrm{omp}} = 6 \pm 2$cm, thus consolidating the initial design choice of 5 cm.

Since the figure of merit R^2 of the dozen most credible fits is just a bit lower than for the best scaling, and since a priori we cannot claim which global parameter combination is the best to use, we can also ascribe the value of λ_q^{ITER} as the average across those many scalings using $P_{tot}/S_{LCFS}[W/m^2]$, $I_p/A[A/m^2]$, q_{95}, $B_0[T]$, a/R, $R[m]$, κ, $<n_e>[m^{-3}]$, $B_{pol}[T]$. **Fortunately, we observe that most of the reasonable scalings yield very similar prediction for ITER, even though using very different parameter combinations.** In [9], details of this work can be found, especially the experimental procedure, the full database and the alternative reasonable scalings.

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 [10] and scaled theoretically in [11]. Both predictions of the near and the here-scaled main SOL decay lengths are required for best ITER HFS limiter, designed finally in [12].

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