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Scaling of L-mode heat flux for ITER and COMPASS-U divertors, based on five tokamaks

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Abstract

This contribution aims to improve existing scalings of the L-mode power decay length $\lambda_{omp}^q$, especially for plasma configurations with strike points at the ITER-relevant location—closed vertical divertor targets. We propose 13 new $\lambda_{omp}^q$ scalings based on data from the tokamaks JET, EAST, MAST, Alcator C-mod and COMPASS, and validate them against the output of the 2D turbulence code HESEL. The analysis covers 500 divertor heat flux profiles (obtained by

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\textsuperscript{a} See Labit \textit{et al} 2019 (https://doi.org/10.1088/1741-4326/ab2211) for EUROfusion MST1 Team.
\textsuperscript{b} See Joffrin \textit{et al} 2019 (https://doi.org/10.1088/1741-4326/ab2276) for the JET team.
\textsuperscript{c} See Harrison \textit{et al} 2019 (https://doi.org/10.1088/1741-4326/ab121c) for the MAST-U Team.
probes or IR cameras), measured in L-mode discharges with varying 12 global plasma parameters (all well predictable). We find that the two previously published scalings (Eich 2013 J. Nucl. Mat. 438 S72) and (Scarabosio 2013 J. Nucl. Mat. 438 S426), which were based on outer target data from AUG and JET, describe the JET, C-mod and COMPASS profiles well. This holds not only at the outer horizontal and vertical targets, but surprisingly also at the inner vertical targets. In contrast, EAST, HESEL and especially MAST data are poorly described by these two scalings. We therefore derive 13 new scalings, which account for 85–92 % of the measured $\lambda_{q,\text{omp}}$ variability across all five tokamaks. Although each of the scalings is based on a different parameter combination, their predictions for the ITER and COMPASS-Upgrade tokamaks are very similar. Just before the L-H transition in the ITER baseline scenario, the presented scalings predict values $\lambda_{q,\text{omp}} = 3.0 \pm 0.5$ mm. For the COMPASS-Upgrade tokamak, all the scalings predict $\lambda_{q,\text{omp}} = 2.1 \pm 0.5$ mm with a single exception of the scaling based on the stored plasma energy which predicts only 1.2 mm for both tokamaks. We encourage the reader to use as many of these scalings as possible, depending on available data. In attached plasma and using significant assumptions, our results imply steady-state surface-perpendicular heat flux around 10 MW/m$^2$ for ITER, and 20 MW/m$^2$ for COMPASS-Upgrade.

Keywords: tokamak, L-mode, divertor, scaling, heat flux, ITER, COMPASS-Upgrade

(Some figures may appear in colour only in the online journal)

1. Introduction to divertor target heat flux distribution

The most critical interaction of plasma with plasma-facing components (PFCs) in tokamaks takes place on the divertor targets at the strike points. The incident heat fluxes are usually studied under H-mode conditions, e.g. in [1, 2]. However, the first campaigns on ITER, as well as the start-up and landing phase of each ITER discharge, will be L-mode divertor plasmas. Even later, when the high performance H-modes with substantial fusion power are achieved, sudden H-L transitions may occur. It is therefore essential to predict the heat flux profile along the divertor target also for L-mode in order to assess the power handling of divertor PFCs. Such a prediction is also important for any L-mode fluid edge plasma simulation (using codes such as SOLPS, SOLEEDGE2D or TECXY) in order to set up the essential cross-field heat transport coefficient $\chi$, or for turbulent scrape-off layer (SOL) models (HESEL [3], TOKAM3X or GBS [4]) for experimental benchmarking. In this work we focus on attached plasma conditions, which yield the upper limit of the heat fluxes at the targets.

According to [1] and demonstrated in figure 1, the divertor parallel heat flux profile mapped to the outer midplane (OMP) can be described by the function

$$q_{\parallel}|(r) = \frac{q_{\parallel 0}}{2} \exp \left( \left[ \frac{S}{2 \lambda_{q,\text{omp}}} \right]^2 - \frac{\tau}{\lambda_{q,\text{omp}}} \right) \text{erfc} \left( \frac{S}{2 \lambda_{q,\text{omp}}} - \frac{\tau}{\frac{S}{2 \lambda_{q,\text{omp}}}} \right) \quad (1)$$

where $\tau = r - r_{\text{sep}}$ is the distance from the separatrix/strike point, $q_{\parallel 0}$ is the peak heat flux, $S$ parameter describes the heat flux spreading into the SOL and the private flux region, and $\lambda_{q,\text{omp}}$ is the power decay length. We also allow for a small offset $q_{\parallel 0} < q_{\parallel 0}$ corresponding to the background heat flux (result of radiation if measured by IR cameras). The $q_{\parallel 0}$ profile is mapped to the outer midplane in order to remove the effects of varying divertor geometries between the tokamaks and thus allow a universal comparison.

The value of $\lambda_{q,\text{omp}}$ is a result of competition between the essentially 2D upstream turbulent cross-field transport of blobs (see e.g. [5] for its complex velocity scalings in four SOL regimes) and the 1D downstream convection and conduction along the magnetic field lines. In effect, $\lambda_{q,\text{omp}}$ determines the plasma-wetted area and thus the engineering peak heat flux (see section 6). Thus it is crucial to predict its value in future tokamaks with sufficient accuracy. Therefore, based on new data from five tokamaks, we derive predictive scalings, compared also to previous scaling attempts [6–12].

2. Experimental database

The published L-mode $\lambda_{q,\text{omp}}$ scalings ([2] equation (7)) and [10] are based on measurements at the JET and ASDEX Upgrade tokamaks. However, a database comprising only two tokamaks may not form a reliable base for scaling studies, as it may not offer a substantial range of plasma parameters. This paper presents scalings based on a more variable dataset: the tokamaks COMPASS, EAST, Alcator C-mod (with high toroidal magnetic field, 2.5 $< B_\phi[T] < 8$), MAST and JET. The data was obtained in single-null L-mode plasmas (with the $\lambda_{q,\text{omp}}$, which in COMPASS limiter plasmas scales mainly as $\lambda_{q,\text{omp}} \propto I_p^{-1} \propto q_{95}$ in accordance with the HD-model [13, 14]) and the main SOL (power decay length $\lambda_{q,\text{omp}}$, scaled for limiter plasmas in [15]) is regularly observed [6, 16, 13]. In order to determine the peak heat flux, we focus on the heat flux footprint in the vicinity of the strike point and thus look for a scaling using a single $\lambda_{q,\text{omp}} \approx \lambda_{q,\text{omp}}$. Adding $\lambda_{q,\text{omp}}$ would unnecessarily complicate equation (1) and yield ambiguity to $\lambda_{q,\text{omp}}$.

\footnote{For $\tau > 4 \text{ mm}$, this fit clearly slightly underestimates the data, which is a consequence of using a single-exponential fit, even though double-exponential character is omnipresent: existence of the near SOL (power decay length $\lambda_{q,\text{omp}} \propto I_p^{-1} \propto q_{95}$) is a result of competition between two mechanisms: $\lambda_{q,\text{omp}}$ is a result of radiation if measured by IR cameras. The $q_{\parallel 0}$ profile is mapped to the outer midplane in order to remove the effects of varying divertor geometries between the tokamaks and thus allow a universal comparison.}
except for MAST - see later) with varying plasma parameters and divertor geometries, including both outer and inner targets.

The two published scalings were based on data from outer divertor targets and preferentially employed IR thermography, which can deliver data with high spatial resolution. However, for geometrical reasons, it is difficult to observe both the divertor targets if they are in the vertical configuration, such as on JET (figure 2(a)). As a substitution divertor probes can be used to obtain the necessary heat flux measurements as

$$q_\parallel = \gamma j_{sat} T_e$$

where $\gamma$ is the sheath heat transmission factor, which we assume to be constant along the target, $j_{sat}$ is the ion saturation current, measured in a standard way at highly negative voltage, and $T_e$ is the electron temperature, measured by various techniques described below. Due to greater data availability, probes measurements were used in all the tokamaks except for MAST, where only IR data was available and its consistency with probes was demonstrated in [12]. Figure 2 shows the divertor geometries in which heat flux profiles were obtained for the different experiments during the discharge flat-top phase.

On JET, Langmuir probes with standard slow voltage sweeping [18] were used to derive the $T_e$ and $j_{sat}$. In order to improve the spatial resolution, the strike points were slowly swept (5 cm / 0.5 s) as depicted by the arrows in figures 2(a) and (b). The inner target is analyzed only for the a) configuration since in b) it (tile 2) magnetically shields itself.

On COMPASS, only probes located at the outer target yield credible $q_\parallel$ profiles. $T_e$ is measured as the difference between the floating potentials of ball-pen and Langmuir probes [19]. The value of $\gamma = 11$ in equation (2) is selected to match the measurements of IR thermography [20] as demonstrated in figure 1; however, it should be noted that the specific value of $\gamma$ does not affect the resulting $\lambda_q^{omp}$. In the case of IR thermography [21], the parallel heat flux is obtained by the THEODOR code [22]. The error bars of the fit equation (1) are calculated using the bootstrap technique [23].

On EAST, only probe data from the graphite bottom inner target could be used. [7] Discharges heated by lower-hybrid waves were excluded from our analysis; however, [7] shows that they follow the same $\lambda_q^{omp}$ scalings, only with $\lambda_q^{omp}$ higher by 30%. The probes spatial resolution was 10–15 mm at the divertor, just sufficient to resolve the decay lengths.

On MAST, only IR thermography was used [12], employing a long wave (7.6–9 µm) camera observing the upper outer divertor. MAST discharges were performed in a double-null configuration which was closer to upper single-null, with $\lambda_q^{omp}$ marginally larger than the outer-midplane distance between the up and down separatrices. As experimentally verified in [24, figure 11], $\lambda_q^{omp}$ is not significantly influenced by this unusual configuration.

On Alcator C-mod, only outer target data were available, obtained by a combination of proud and flush Langmuir probes and surface thermocouples [6]. The data has high spatial resolution (0.1 mm mapped to the outer midplane) and wide dynamic range (4 orders of magnitude in heat flux).

3. Uncertainties in the database

Scans of the $\lambda_q^{omp}$ dependence on several global plasma parameters are shown in figure 3, demonstrating both the available parameter span within each tokamak and between them. One may observe its substantial dependency on $I_e$, $B_m$, $B_{pol}$, $q_{95}$, $\phi_p$, and $f_{GW}$. Table 1 contains description of all the used quantities.

There is evidence [25] that in attached conditions SOL properties, including $\lambda_q^{omp}$ and possibly its scaling, might depend on the SOL transport regime (sheath-limited (S-L) or conduction-limited (C-L)). Furthermore, according to [5] the radial blob velocity, which contributes to determining the cross-field flux, scales differently in these regimes. Therefore, to ensure the $\lambda_q^{omp}$ database coherence, we check that all the analyzed tokamaks operate in the same regime. Unfortunately, it was not possible to verify the transport regime for each individual data point, so we rely on analyses performed on discharges similar to those in our database. According to [26], Alcator C-mod SOL is in the conduction-limited regime for $\nabla_b > 10^{30}$ m$^{-3}$, which our data marginally satisfy. According to [27], the MAST SOL transport is also conduction-limited up to $\nabla_b = 4.5 \times 10^{19}$ m$^{-3}$, and detached beyond. According to [28], EAST SOL is conduction-limited for $\nabla_b < 4.5 \times 10^{19}$ m$^{-3}$.

Compared to these larger tokamaks, COMPASS has a relatively short connection length from the OMP to the outer target (typically < 5 m), which may facilitate transport in the sheath-limited regime. Indeed, a previous analysis performed during its operation in Culham, UK [29] has concluded that its SOL can operate in either the sheath- or the conduction-limited regime. To distinguish between the two regimes, we use the SOL collisionality parameter $\nu^* = 10^{-16} n_e L / T_e$ defined by equation 4.105 in [30], where $n_e$ is the upstream electron density, $L$ is the connection length and $T_e$ is the upstream electron temperature. If $\nu^* < 10$, the flux tube in question is in
the sheath-limited regime: if $\nu^* > 15$, it is in the conduction-limited regime. This criterion applies to each flux tube individually, and so a part of the SOL may be sheath-limited and a part conduction-limited. To account for this possible variation, collisionality was evaluated across the SOL using measurements of a reciprocating probe. $T_e$ was measured using the ball-pen probe and Langmuir probe technique [31]; $n_e = j_{sat}/0.5ec$, where $e$ is the elementary charge and $c_s = \sqrt{2eT_e/m_i}$ is the sound speed, was measured using a biased Langmuir probe and the aforementioned technique; finally $L_w$ was calculated from the equilibrium reconstruction as the connection length of the OMP to the outertarget. In the discharges where $\nu^* > 15$ over a majority of the SOL and >10 across its entire extent, the overall SOL transport regime was judged to be conduction-limited. Thus, the COMPASS data falls into two categories: either conduction-limited, or intermediate or sheath-limited. The choice between them is discussed in the following section.

Our results were cross-checked against the output of 2D slab OMP turbulence simulations by the code HESEL [3]. The simulations were set up for ASDEX Upgrade-like L-mode SOL parameters for a wide scan of relevant plasma conditions (see the golden symbols in figures 3 and 4). For consistency, we related the code-relevant parameters as $\kappa = 1.7$, assuming $B_\phi = B_{LCFS} \frac{a}{R_0}$ and $q_{95} = q_{cyl}$ because $q_{95}$ has no direct meaning in this 2D model. Similarly, since $(p)$ and $\beta$ are not input parameters in HESEL, they are excluded from relevant scalings.

4. Main result: power decay length scaling candidates

Figure 4(a),(b) shows the two previously published scalings [10] and [2] applied to our new data set. When all our data is included (that is, inner and outer targets, vertical and bottom horizontal targets and all tokamaks), only a small fraction ($R^2$) of its variation is described by the respective scalings. In contrast, the scaling [6] shown in figure 4(c) describes the available database surprisingly well considering that it is based on a single parameter $(p)$, the volume-averaged plasma pressure, and its source data come solely from Alcator C-mod. We included three more parameters alongside $(p)$ to create scaling E8 in table 2 which describes our database very well (excluding unfortunately MAST due to lack of data).

To find a new scaling based on all our data, we employed the methodology described in [15]. In contrary to many previous publications on tokamak heat flux scalings, e.g. [6, 2, 10–12], we look for all possible combinations of scaling parameters, not a single one. We assume $\lambda_{omp} = \alpha_0 \prod_{i=1}^N p_i^{\alpha_i}$. This translates to $\log(\lambda_{omp}) = \log(\alpha_0) + \sum_{i=1}^N \alpha_i \log(p_i)$ which allows the standard (least-square) regression analysis. To limit the resulting number of possible scalings, we apply strong restrictions on the scaling variables. The most prominent one is limiting the set of global plasma parameters to a set of 12: $f_0[A/m^2]$, $B_0[T]$, $B_{pol}[T]$, $\rho [m^{-3}]$, $P_{SOL}/S_{LCFS}[MW/m^2]$, $(p)[atm]$, $\beta$, $f_{GW}$, $q_{95}$, $q_{cyl}$, $\alpha/R_0$, $\kappa$ and $\delta_{up}$. This was done based on the following considerations:
Figure 3. Scan of the $\lambda_{omp}$ dependence on individual main plasma parameters as defined in table 1. The database includes five hundred COMPASS (197), EAST (19), JET (113), Alcator C-mod (91) and MAST (88) L-mode divertor profiles obtained by either probes or IR camera, as well as the output of 20 HESEL (2D slab turbulence) simulations. Dashed lines mark the parameters of ITER (yellow) and COMPASS-Upgrade (blue). Note that there is no single-parameter scan; these are just projections of the 12-dimensional parameter space. Note that the full database table of 550 entries of 47 parameters (only 12 used here) can be requested from horacek@ipp.cas.cz.
Table 1. Main plasma parameters of ITER [32] at t = 50 s and COMPASS-Upgrade [33] just before the L/H transition, calculated by METIS simulations. Parameter type: engineering = very precise, global plasma parameter, dimensionless = theoretically determining the plasma transport.

<table>
<thead>
<tr>
<th>Type</th>
<th>Parameter</th>
<th>Unit definition</th>
<th>Description just before L/H transition</th>
<th>ITER</th>
<th>COMPASS Upgrade</th>
</tr>
</thead>
<tbody>
<tr>
<td>e</td>
<td>I_p</td>
<td>MA</td>
<td>Plasma current</td>
<td>12</td>
<td>2</td>
</tr>
<tr>
<td>e</td>
<td>A</td>
<td>m²</td>
<td>= πa²κ cross-section</td>
<td>14.2</td>
<td>0.41</td>
</tr>
<tr>
<td>e</td>
<td>j_p = I_p/A</td>
<td>MA/m²</td>
<td>Plasma current / cross-section</td>
<td>0.87</td>
<td>4.9</td>
</tr>
<tr>
<td>e</td>
<td>B_θ</td>
<td>T</td>
<td>Toroidal mag. field</td>
<td>5.3</td>
<td>5.0</td>
</tr>
<tr>
<td>e</td>
<td>B_pol</td>
<td>T</td>
<td>Poloidal mag. field</td>
<td>1.14</td>
<td>1.03</td>
</tr>
<tr>
<td>e</td>
<td>R_0</td>
<td>m</td>
<td>Major radius</td>
<td>6.5</td>
<td>0.89</td>
</tr>
<tr>
<td>e</td>
<td>V</td>
<td>m³</td>
<td>Line-averaged density</td>
<td>576</td>
<td>2.3</td>
</tr>
<tr>
<td>g</td>
<td>n_e</td>
<td>m⁻³</td>
<td>Power to SOL</td>
<td>18</td>
<td>3.7</td>
</tr>
<tr>
<td>g</td>
<td>P_SOL</td>
<td>MW</td>
<td>Power through LCFS surface</td>
<td>0.033</td>
<td>0.27</td>
</tr>
<tr>
<td>g</td>
<td>S_LCFS</td>
<td>m²</td>
<td>= 4π²R_0a√(1 + κ²)/2 surface</td>
<td>557</td>
<td>13.7</td>
</tr>
<tr>
<td>g</td>
<td>(p) = E_p/V</td>
<td>atm = 101 kPa</td>
<td>Power through LCFS surface</td>
<td>0.86</td>
<td>0.79</td>
</tr>
<tr>
<td>dg</td>
<td>β</td>
<td>≡ (p)2μa/B_θ</td>
<td>Greenwald density fraction</td>
<td>0.008</td>
<td>0.008</td>
</tr>
<tr>
<td>dg</td>
<td>f gio</td>
<td>≡ n_e/E_p/V</td>
<td>Safety factor at Ψ = 0.95</td>
<td>1.7</td>
<td>2.6</td>
</tr>
<tr>
<td>de</td>
<td>q_{SOL}</td>
<td>≡ π²B_θ/(1 + κ²)</td>
<td>Cylindrical safety factor</td>
<td>1.6</td>
<td>2.1</td>
</tr>
<tr>
<td>de</td>
<td>a/R_0</td>
<td>minor/major radii</td>
<td>Minor/major radii</td>
<td>0.28</td>
<td>0.3</td>
</tr>
<tr>
<td>de</td>
<td>κ</td>
<td>≡ b/a</td>
<td>Vertical plasma elongation</td>
<td>1.4</td>
<td>1.8</td>
</tr>
<tr>
<td>de</td>
<td>δ_w</td>
<td>Upper plasma triangularity</td>
<td>0.21</td>
<td>0.49</td>
<td></td>
</tr>
</tbody>
</table>

- We exclude plasma elongation since it varies too weakly, 1.6 < κ < 2.0.
- We normalise the parameters R_0, a, I_p, P_SOL, and E_p to j_p = I_p/A, a/R_0, (p) = E_p/V and P_SOL/S_LCFS. This naturally excludes any extrapolation towards ITER and COMPASS-U (in accordance with [15]), as shown by the vertical lines in figure 3 which all lie within the database span.
- As previously done in [3, 34], we exclude all local separatrix parameters which theoretically affect λ_{q}\text{amp}^{c}: the outer-midplane separatrix temperature T_{e,LCFS} and density n_e^{LCFS} or their peak divertor values T_{e,div} and n_e^{peak,div}. There were two reasons for this. One, their values for ITER cannot be reliably predicted without falling into self-consistent procedures, as ITER edge plasma modelling (e.g. in SOLPS-ITER) generally uses λ_{q}\text{amp} as input for the setup of its cross-field transport coefficients. Two, the separatrix parameters values cannot be reliably measured in experiment either, as the exact position of midplane separatrix is very questionable in current tokamaks. The typical systematic discrepancy of 1 cm yields error of both T_{e,LCFS} and n_e^{LCFS} values by a factor of \exp(1 cm/\lambda_{q}\text{amp}) > 10. It was attempted on COMPASS [35] and TCV [36] to identify the LCFS using various techniques, yielding to systematic shift outwards from the EFIT or Liiüe) magnetic outer midplane separatrix by ≈2 cm on COMPASS and 0.5 cm on TCV. However, even though this method seems more precise than the magnetic equilibrium reconstruction, it is not regularly used on tokamaks since it is quite complex and requires special diagnostics.

Apart from constraining the overall global parameter set, further restrictions to the possible scalings are as follows:

- We require that all mutual cross-correlations M_{cc} between all the parameters used within each scaling are small enough:
  * M_{cc} < 2/3 across the entire database;
  * M_{cc} < 4/5 for each tokamak independently.

This is imposed in order to avoid scaling ambiguity, since future tokamaks may not follow the same parameter correlations. As a result, the number of scaling parameter combinations is drastically reduced from 3000 down to a few dozens. Furthermore, this requirement implicates that the scalings have at maximum N ≤ 4 parameters, otherwise some mutual correlation is too high.

- We require that each t−ratio = exponent α_i exponent α_i error > 3, meaning its statistical significance (the p-value) is above 99.9%.
- We further require that all |α_i| < 3, allowing for cubic dependency at maximum.
- Finally, let us limit the number of best scalings by requiring the minimum regression quality \text{R}^2 > 85%.

Satisfying all the above restrictions yields 13 credible scalings, listed in table 2 in black. The table also shows their predictions of λ_{q}\text{amp} for ITER and COMPASS-U for L-mode just before the L/H transition, using parameters from table 1.
Figure 4. Scalings of experimental values of L-mode divertor decay lengths \(\lambda_{omp}\) with main plasma parameters. Predictions are shown by the thick circle: yellow for ITER and blue for COMPASS-Upgrade. Inner targets and HESEL simulation are less credible (marked by light symbols). (a) P1 according to [2] (b) P2 according to [10], both derived from ASDEX Upgrade and JET only, (c) P3 according to [6] (derived from Alcator C-mod only). Examples of best NEW scalings based on (d) only outer targets D1, (e) all tokamaks F1.

5. Discussion of the scalings

To investigate the robustness of the derived scaling laws, we split the \(\lambda_{omp}\) database into six datasets:

- Dataset A keeps only the most conservative (credible) data obtained from the measurements at the outer targets. For COMPASS, only discharges with conduction-limited SOL were used. MAST data were excluded, as well as data obtained at inner targets (which may scale differently).
- Dataset B contains dataset A as well as the data from MAST, allowing to assess the role of near-double null plasmas and different aspect ratios. Note, however, that both influences are present in the MAST data simultaneously.
• Dataset C contains dataset A as well as all the COMPASS data to assess the role of SOL transport.
• Dataset D contains data from outer targets of all the studied tokamaks.
• Dataset E contains data from both the inner and the outer targets but excludes MAST, allowing to assess the possibly different scaling of $\lambda_{q,\text{omp}}$ at the inner target.
• Dataset F contains all the data.

Table 2 lists the scalings which match the criteria described in section 4 for each respective dataset. The second column shows the coefficient of determination $R^2$ meaning “what percentage of the data variation is explained/described by the scaling”. Each scaling was tested against each of the six datasets. The $R^2$ corresponding to its own dataset is marked in a large font and bold face.

For the most conservative dataset A, only one scaling A1 was found, with relatively red poor fit quality $R^2 = 77\%$. This is mostly due to the relatively small range of $\lambda_{q,\text{omp}}$ caused by excluding MAST data, which generally show large values of $\lambda_{q,\text{omp}}$ as seen in figure 3. The same problem befalls scaling C1. When the MAST data are added (dataset B), the scaling B1 with excellent fit quality $R^2 = 92\%$ is obtained. However, note that one of the scaling parameters is the aspect ratio $a/R_0$, which has limited variation on all tokamaks except for MAST (see figure 3). In principle, this parameter may not be itself responsible for the variation of $\lambda_{q,\text{omp}}$ but serves as a proxy for some other parameter, which is different on this spherical tokamak. Nevertheless, the aim of our work is not to uncover the underlying physics which determines the scaling of $\lambda_{q,\text{omp}}$ but to provide a reliable prediction based mostly on parameters which are straightforward to determine.

When all data obtained at the outer targets are used (dataset D), the dataset becomes sufficiently robust to yield a number of scalings with good fit quality. Some of them (D7, D0), however, perform poorly when applied to the more restricted datasets (having large negative $R^2$) and are therefore excluded from further considerations. Note that a majority of these scalings exhibit a similar dependence on $f_{GW}$ with the exponent value in the range of 0.5–0.9. Such dependence was not observed in the published scalings P1-4, although it may have been to some extend included in the dependence on pressure. All the credible scalings also depend approximately linearly on the aspect ratio, similarly to the dependence found in dataset B.

When data from outer and inner targets are combined (dataset E), it is possible to derive credible scalings even without the contribution of MAST data. We address three particular interesting aspects of these scalings. Firstly, the upper triangularity $\delta_{up}$ appears as a significant parameter. This was not the case in the previous datasets, even though they contained data with a significant range of $\delta_{up}$ (such as the C-mod data). This suggests that $\delta_{up}$ influences the transport to the inner target. Following the conventional picture of transport in the SOL, with the blob source in the vicinity of the outer midplane and subsequent parallel transport along the open field lines, it may imply that the location on the top of the tokamak (where the plasma shape is strongly influenced by $\delta_{up}$) plays a key role in this process. This conclusion is consistent with observations at AUG [37], where the $\lambda_{q,\text{omp}}$ measured at the inner target exhibited dependence on different parameters than measurements at the outer target. The second matter of interest is that, surprisingly, scalings derived from dataset E also contain a dependence on the aspect ratio $a/R_0$, although the variation of this parameter is not great in the absence of MAST data. This is demonstrated by many cross-group $R^2 < 0$, marked by the red color pointing to poor predictive capability of such scalings. This is because the $a/R_0$ exponent is always negative where MAST data is included. This is most probably caused by the presence of data from the inner target of EAST, which has a slightly smaller aspect ratio than other tokamaks in the dataset. This suggests two conclusions: (i) the scaling of $\lambda_{q,\text{omp}}$ at the inner target is indeed more complicated than the one at the outer target, and (ii) the aspect ratio is probably not a very reliable scaling parameter, which may be caused by the hidden dependencies on other parameters (as mentioned earlier).

Consequently, we have attempted to derive a scaling based on dataset E which would contain a dependence on the averaged plasma pressure, which is the principal parameter identified by Brunner [6]. The quality of such fit is decent and the $(p)$ exponent $–0.44$ is very similar to the one obtained by Brunner $(–0.48)$. However, two more parameters appear to be significant—the aspect ratio and $\beta$. The inclusion of $\beta$ is caused by the data from COMPASS, which were obtained at significantly higher values of $\beta$ than on other tokamaks.

Finally, when all the available data are used (dataset F), the fit quality decreases slightly compared to the more restricted datasets. The best quality scaling F1 depends on $B_{\phi}$, $q_{95}$ and $f_{GW}$. The safety factor includes a dependence on the poloidal magnetic field, but the exponent is different from the H-mode scaling derived by Eich [2]. The significant dependence on $f_{GW}$ does not appear in the H-mode scalings either. This may be caused by the generally larger variation of $f_{GW}$ in L-mode plasmas in our datasets. In H-mode, it is difficult to achieve measurements below the natural H-mode density or even to enter H-mode at such conditions, since the L-H power threshold is known to increase sharply in such a case. Moreover, high density discharges are usually not analysed when IR thermography is employed as the principal diagnostic, which was the case in [2], due to the parasitic influence of bremsstrahlung. Therefore, the dependence on $f_{GW}$ may have appeared as insignificant in such studies.

Note that scaling F1 (which has the highest $R^2$ for this dataset) does not employ the aspect ratio, however the $\lambda_{q,\text{omp}}$ predictions for ITER and COMPASS-U are quite similar to those obtained with scaling F2, where the aspect ratio is included. This suggests that although the role of aspect ratio is probably quite complex, it is not essential for determination of $\lambda_{q,\text{omp}}$ in the cases of our interest.

Two observations may be made across the datasets. Firstly, most of the credible scalings use one (and only one) of the parameters $B_{\phi}, B_{\text{pol}}, J_{95}, q_{95}$ and $q_{95}$. This is to be expected since they are all naturally correlated within the database. Secondly, the $R^2$ is not significantly modified when HESEL results are included in the fit, which leads us to the conclusion that HESEL is generally in good agreement.
Table 2. The most credible scalings for each dataset, ordered by the coefficient of determination $R^2$. The $\lambda^{\text{impl}}$ predictions are based on the parameters listed in table 1. Note that this METIS ITER scenario has very low $q_{\text{ps}} < 2$ which is probably MHD unstable. We verified that a more realistic ITER scenario with $q_{\text{ps}} = 2.2$ (due to faster shaping) yields predictions with $\lambda^{\text{impl}}$ longer (peak heat flux smaller) by 10% for half of the scalings. $R^2_{\text{ABCD}}$ for all the datasets are shown on the left. $R^2$ of the dataset which was used to derive the respective scaling is marked as $R^2_{\text{withoutHESEL}}/R^2_{\text{includingHESEL}}$. Red-colored scalings which have $R^2 < 85\%$ for all the datasets or have $R^2 < 0$ for any one dataset are not considered credible. Note that scaling E8, based on the average pressure, yields significantly shorter $\lambda^{\text{impl}}$ for ITER and COMPASS-U. $q_{\text{ps}}^{\text{peak}}$ is estimated according to equation (3). Further marked are the Eng.inerings scalings which use only highly credible inputs, independent from (uncertain) plasma conditions.

<table>
<thead>
<tr>
<th>#</th>
<th>R² [%]</th>
<th>fit quality</th>
<th>Scaling formulas</th>
<th>ITER $\lambda^{\text{impl}}$</th>
<th>Compass-U $\lambda^{\text{impl}}$</th>
<th>ITER $q_{\text{ps}}^{\text{peak}}$</th>
<th>Compass-U $q_{\text{ps}}^{\text{peak}}$</th>
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<tbody>
<tr>
<td>A1</td>
<td>77</td>
<td>77,77,33,20</td>
<td>ABCDE</td>
<td>$1.82 \cdot q_{ps}^{-0.43} &lt; p &gt; [\text{atm}]^{-0.48}$</td>
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<td>1.4</td>
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<td>B1</td>
<td>79</td>
<td>92,66,92,42</td>
<td>A + MAST</td>
<td>$4.03 \times 10^3 \cdot (a/R_0)^{1.06} \cdot q_{ps}^{-0.37}$</td>
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<td>1.5</td>
<td>9.4</td>
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<td>76</td>
<td>69,62,51,74</td>
<td>A + all COMPASS</td>
<td>$1830 \cdot q_{ps}^{-0.52} \cdot (P_{\text{SOL}}/S_{\text{CEF}})^{-0.4}$</td>
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<td>4.5</td>
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<tr>
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<td>79</td>
<td>92,67,92,91,42</td>
<td>All outer targets (with MAST)</td>
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<td>1.7</td>
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<td>2.2</td>
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<td>61</td>
<td>89,50,91,90,65</td>
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<tr>
<td>D5</td>
<td>22</td>
<td>88,15,89,88,16</td>
<td>$38.4 \cdot B_x^{0.22} \cdot (a/R_0)^{1.5} \cdot G_{\text{av}}$</td>
<td>3</td>
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<td>11</td>
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<tr>
<td>D6</td>
<td>55</td>
<td>88,40,89,86,17</td>
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<td>2.7</td>
<td>8.2</td>
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<tr>
<td>D8</td>
<td>32</td>
<td>86,26,88,85,75</td>
<td>$42 \cdot (a/R_0)^{0.41} \cdot G_{\text{av}}$</td>
<td>3.5</td>
<td>3.1</td>
<td>8.9</td>
<td>28</td>
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<tr>
<td>D9</td>
<td>80</td>
<td>85,67,87,82,22</td>
<td>Eng. $4350 \cdot (a/R_0)^{-0.43}$</td>
<td>3.1</td>
<td>1.6</td>
<td>10</td>
<td>28</td>
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<tr>
<td>D0</td>
<td>62</td>
<td>90,27,87,64,80</td>
<td>$13.9 \cdot B_x^{0.6} \cdot q_{ps}^{-0.23} \cdot G_{\text{av}}$</td>
<td>2.2</td>
<td>2.2</td>
<td>10</td>
<td>26</td>
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<tr>
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<td>85</td>
<td>34,72,11,92,92</td>
<td>$5600 \cdot q_{ps}^{0.25} \cdot (a/R_0)^{-0.9} \cdot q_{ps}^{0.27} \cdot G_{\text{av}}$</td>
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<td>4.1</td>
<td>7.6</td>
<td>27</td>
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<tr>
<td>E2</td>
<td>83</td>
<td>99,72,64,91,89</td>
<td>$961 \cdot (a/R_0)^{-1.41} \cdot \delta_{f_{\text{ave}}}$</td>
<td>4.5</td>
<td>4.5</td>
<td>7</td>
<td>26</td>
</tr>
<tr>
<td>E3</td>
<td>83</td>
<td>78,68,81,90,90</td>
<td>$44000 \cdot q_{ps}^{0.21} \cdot G_{\text{ave}}$</td>
<td>3.7</td>
<td>3.7</td>
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<td>84</td>
<td>56,69,26,89,89</td>
<td>$\text{Eng. 9070} \cdot q_{ps}^{-0.12}$</td>
<td>9</td>
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<tr>
<td>E5</td>
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<td>100,70,65,89,85,36</td>
<td>$\text{Eng. 2040} \cdot (a/R_0)^{-1.52}$</td>
<td>7</td>
<td>7</td>
<td>8.4</td>
<td>26</td>
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<tr>
<td>E6</td>
<td>81</td>
<td>77,64,81,85,85</td>
<td>$\text{Eng. 1.48} \times 10^3 \cdot q_{ps}^{0.46}$</td>
<td>3.5</td>
<td>3.5</td>
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<tr>
<td>E7</td>
<td>24</td>
<td>120,80,85,85,48</td>
<td>$2.8 \cdot (a/R_0)^{-1.31}$</td>
<td>4.5</td>
<td>4.5</td>
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<tr>
<td>E8</td>
<td>83</td>
<td>77,55,85,88,88</td>
<td>$0.024 \cdot &lt; p &gt; [\text{atm}]^{-0.44} \cdot (a/R_0)^{-1.96} \cdot q_{ps}^{-0.27}$</td>
<td>1.2</td>
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<td>F1</td>
<td>62</td>
<td>90,14,88,83,88</td>
<td>$8.39 \cdot B_x^{0.56} \cdot q_{ps}^{0.59}$</td>
<td>2.2</td>
<td>2.2</td>
<td>14</td>
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</tr>
<tr>
<td>F2</td>
<td>52</td>
<td>90,78,89,77,85</td>
<td>$12.1 \cdot B_x^{0.2} \cdot q_{ps}^{0.52} \cdot (a/R_0)^{0.42} \cdot G_{\text{ave}}$</td>
<td>2.5</td>
<td>2.3</td>
<td>13</td>
<td>20</td>
</tr>
<tr>
<td>F3</td>
<td>12</td>
<td>87,14,87,83,80</td>
<td>$15 \cdot q_{ps}^{0.56} \cdot (a/R_0)^{0.04} \cdot G_{\text{ave}}$</td>
<td>2.8</td>
<td>2.8</td>
<td>14</td>
<td>26</td>
</tr>
</tbody>
</table>

Published scalings

with all the (black-marked) scalings (see the yellow stars in figure 4).

Concerning the Published scalings, even though $R^2 \approx \frac{1}{2}$ for P1–P2, we consider them credible since they were derived from a fully different dataset (from 2013 Asdex-Upgrade and JET) than tested here. We consider P3 as marginally credible because it describes only 32% of the data variation, which is not surprising as it is based on a single parameter and derived from a single tokamak. Generalized version of P3 is the scaling E8 with $R^2 = 88\%$, however, they both predict (for unknown reasons) significantly smaller $\lambda^{\text{impl}}$ than all the other 15 scalings (not based on (p)).

In contrary, all the scalings P4–P7 do not describe/predict well our database. The scalings P5-7 from year 1999 JET +
JT-60U contain $\frac{\pi}{2}/10^{19}$ and the unknown parameters we approximated as $Z_{\text{eff}} = 1.5$, $P_{\text{div}} = \frac{1}{4}P_{\text{SOL}}$, however, those assumptions do not influence the following judgements. The scalings PS–6 match reasonably both COMPASS and EAST, overestimate JET and HESEL by factor of two and fails completely for MAST and C-mod. The scaling P7 overestimates by factor of two MAST, EAST and JET, however, fails (overestimates by factor of 10–20) C-mod, COMPASS and HESEL.

In conclusion, we point out that despite the different scaling parameters and underlying datasets, all 15 credible scalings in table 2 (set in black) predict very similar values of $\lambda_{\text{omp}}$ for ITER, $3.0 \pm 0.5$ mm, and COMPASS-U, $2.1 \pm 0.5$ mm. This validates our approach to seek all possible scaling laws in order to assess the result robustness.

6. Implied estimate of peak heat flux

Following [38], $\lambda_{\text{omp}}$ determines the effective divertor wetted area as $A_{\text{omp}} = f_{\text{w}} 2\pi R \lambda_{\text{omp}}$, where $f_{\text{w}}$ is the toroidal wetted fraction [39]. $\lambda_{\text{omp}} \approx \lambda_{\text{omp}}^0 + 1.645$ is the integral power decay length and $f_{\text{w}}$ is the poloidal magnetic flux expansion between the OMP and the divertor. We further assume that:

- both targets receive an equal amount of power,
- negligible energy is radiated in the SOL, and thus the plasma is fully attached,
- no power is lost to the first wall since $\lambda_{\text{omp}}^0$ is much smaller than the separatrix-wall distance,
- the heat flux does not spread into the private flux region, $S = 0$ and $\lambda_{\text{int}} = \lambda_{\text{omp}}^0$,
- the OMP-to-divertor poloidal magnetic flux expansion is $f_{\text{w}} = 9$, and
- the toroidal wetted area fraction is $f_{\text{w}} = 0.8$.

Under these assumptions, $P_{\text{diverter}} \approx P_{\text{SOL}}/2$ and the resulting surface peak heat flux on two strike points is

$$q_{\text{peak}} \approx P_{\text{SOL}}/(2A_{\text{omp}})$$

(3)

The resulting predictions of $q_{\text{peak}}$ for ITER and COMPASS-U are shown in the right column of table 2. In the case of ITER, we consider the baseline scenario with plasma conditions just before the L-H transition (see table 1), using the respective predicted value of $\lambda_{\text{omp}}^0$ and the expected value of $P_{\text{SOL}}$. Note that from the point of view of the machine protection, the ITER divertor should keep the surface temperature of its water-cooled monoblocks below the cyclical damage limit (recrystallization), which corresponds to the steady-state heat load $q_{\text{peak}} \leq 10 - 15$ MW m$^{-2}$ [40]. This should prevent damage to the divertor monoblocks also in case when the L-H transition could not be achieved for some reason and the L-mode phase would extend on the timescale of the thermal response of the monoblocks. In case of COMPASS-U [33], we consider the scenario with the most demanding parameters ($I_p = 2$ MA, $B_{\phi} = 5.0$ T).

7. Conclusion

In order to improve the existing scalings of the SOL power width $\lambda_{\text{omp}}^0$ such as [2] and [10], we have measured, analyzed and combined five hundred COMPASS, Alcator C-mod [6], EAST [7], JET and MAST L-mode divertor heat flux profiles obtained by either probes or IR camera as well as output from HESEL (2D slab turbulence) simulations [3] using settings based on tokamak ASDEX Upgrade parameters. To assess the coherence of our heterogeneous database, we split it into six overlapping datasets and applied stringent criteria to the resulting scalings. This allowed us to reduce the possible ~3000 scalings to just 24 scalings, which are listed in table 2, each with very high credibility (describing 85–92% of the data variability). Further testing showed that only half of those new scalings had strong predictive capabilities outside their "native" datasets.

Our main result is that just before the L-H transition, all the 13 credible scalings (as well as [2] and [10]) yield consistent predictions of $\lambda_{\text{omp}}^0$ for both ITER and COMPASS-Up:grade:

- $\lambda_{q_{\text{omp}}}^0$ ITER = $3.0 \pm 0.5$ mm when using all reasonable parameter combinations. Note that this prediction is quite similar to the $Q = 10$ burning DT ITER H-mode [41, figure 4].
- $\lambda_{q_{\text{omp}}}^0$ COMPASS – U = $2.1 \pm 0.5$ mm.

It must be noted, however, that one of the 13 scalings (E8, based on the average plasma pressure $\langle \rho \rangle$, not available for MAST) yields a significantly shorter $\lambda_{q_{\text{omp}}}^0$ = $0.1 \pm 1.2$ mm for both tokaaks and that the reason is currently unknown.

We suggest that predictions of tokamak divertor conditions be done using as many black-colored scalings from table 2 as possible, excluding those in red color.

For an attached L-mode plasma all the scalings yield (using significant assumptions from section 6) divertor surface-perpendicular peak heat flux $q_{\text{peak}}^\perp \approx \langle P_{\text{omp}} \rangle \approx 10$ MW/m$^2$ and $q_{\text{peak}}^\perp \approx \langle P_{\text{omp}} \rangle \approx 20$ MW/m$^2$.

Acknowledgments

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