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Tests of ITER limiter L-mode SOL power width scaling in DIII-D

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Plasma start-up and ramp-down in ITER will use limiter configurations. The ITER first wall (FW) is being designed to allow startup on the actively cooled beryllium panels on both the high (HFS) and low (LFS) field sides, and plasma scenarios have been developed [1]. Here we report results of a dedicated experiment performed in the DIII-D tokamak that validate the key assumptions used to design the FW for power handling during limiter operation.

The power handling capacity is determined by the parallel heat flux density, $q_{||}$ and the FW panel shaping. The profile of $q_{||}$ is characterized by the scrape-off layer (SOL) power flux density e-folding length, λ_q. In the ITER Thermal Load Specifications [1,2] which form the design basis for the FW and divertor PFCs, λ_q in L-mode divertor phases is estimated assuming the scaling derived from measurements of divertor target power fluxes mostly from JT-60U and JET (with an uncertainty of a factor of ~2 around this value):

$$\lambda_q\,(\mathrm{m}) = (1\pm1/3)\ 3.6\ 10^{-4}\ \mathrm{R}\,(\mathrm{m})^2\ P_{\mathrm{div}}\,(\mathrm{MW})^{-0.8} \times q_{95}^{0.5} \times \overline{n}_{\mathrm{e}}\ (10^{19}\,\mathrm{m}^{-3})^{0.9} \times Z_{\mathrm{eff}}^{0.6}\ , \qquad (1)$$

where R is the major radius, P_{div} is the conducted power to the divertor, $\overline{n}_{\mathrm{e}}$ is the line averaged plasma density and Z_{eff} is the plasma effective charge. In the absence of a similar scaling for limiter plasmas, Eq. (1) has been applied to estimate λ_q for the limiter ramp-up/down phases in ITER by replacing P_{div} by the power to the limiters and taking into account the effect of a variable number of poloidal limiters following the model in Ref. [3].

Experimental measurements in tokamaks show considerably larger SOL width in HFS- compared to LFS-limited configurations ([4] and references therein). This is explained by the strong ballooning component of edge transport in tokamaks, which leads to larger SOL widths when plasmas are limited on the HFS. As a consequence, the value of λ_q mapped to the outboard midplane is usually expected to be ~2.5x larger in HFS limiter plasmas than in their LFS counterparts [3]. When flux expansion is taken into account, the local value of λ_q at HFS in ITER is expected to be ~4x larger than that on the LFS [2]. For given power into the SOL (P_{SOL}), this increase over-compensates the increased parallel power flux (due to the stronger toroidal field on the HFS) and makes HFS start-up advantageous compared with LFS configurations. There are in fact several other advantages to HFS start-up [1], so it is important to confirm that these ITER assumptions for limiter power loading are correct. Here we report results of the recent □$_q$ measurements in DIII-D performed in both HFS-limited (inner-wall-limited, IWL) plasmas of varying elongation, and lower single null (LSN) diverted discharges. A single discharge with the plasma limited at the top of the vessel was also executed as an approximation to LFS-limited conditions, for which the DIII-D FW is not optimized.

A poloidal cross-section of DIII-D together with the shapes of the last closed flux surface (LCFS) in configurations used in this study are shown in Fig. 1. Figure 1(a) includes two IWL configurations with slightly different elongation, κ~1.4 and κ~1.5. It is worth noting that δ, the distance between the top of the LCFS and the toroidally continuous "knee limiter" decreases with the increasing elongation. Figure 1(b) shows the separatrix in LSN and the LCFS of top-limited (TL) discharges along with the poloidal location of the midplane reciprocating probe array (RCP) and the field of view of the infrared camera (IRTV). The RCP is used to determine the e-folding lengths, λ_n and λ_T of n_e and T_e in the LFS SOL. Assuming $T_i = T_e$ (since T_i measurements are unavailable) and sheath-limited heat flux, $q \propto n_e T_e^{3/2}$ allows λ_q to be computed as $1/\lambda_q = 1/\lambda_n + 3/2\lambda_T$. The IRTV measures the heat flux profile across the lower divertor floor that is compared with the probe measurements of λ_q in the LSN configuration.

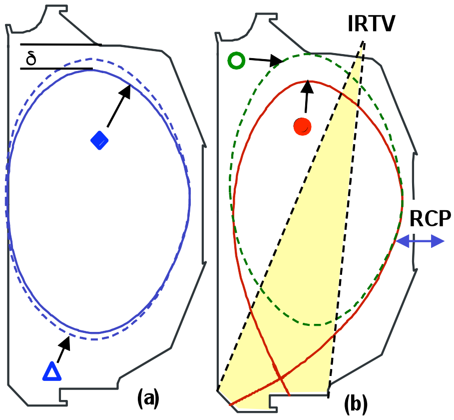

Fig. 1. Poloidal cross-sections of the LCFS in the magnetic configurations used in the study and diagnostic arrangement.

The experiment comprised a series of ohmic and neutral beam injection (NBI) heated L-mode discharges. Profiles of n_e and T_e were measured with the RCP twice per discharge, at $t = 2.5$ s and $t = 3.5$ s. Plasma current and density were scanned from shot to shot, while NBI heating power, P_{NBI}, was increased stepwise in some of the discharges from 0 to 1.25 MW at $t = 3.0$ s. The scaling parameters in Eq. (1) were varied in the following ranges: $q_{95} = 3.2$–7.4, $\bar{n}_e = 1.1$–4.5×10^{13} cm^{-3}, $P_{SOL} = 0.1$–1.4 MW. Here P_{SOL} is used in place of P_{div} in Eq. (1) and is calculated as the sum of ohmic and NBI heating power minus the power radiated from the plasma core. There was no systematic change in core impurity concentration throughout the scans with $Z_{eff} \sim 2$ in all discharges. We should note that it was not possible to change the scaling parameters independently. For example, an increase in the heating power typically resulted in an increase in the plasma density.

The full data set consists of 37 IWL, 10 LSN and 2 TL profiles. Figure 2 plots λ_T versus λ_n for all useable profiles in the dataset. A few profiles were discarded because the probe reciprocations did not allow close enough approach to the LCFS and/or due to excessive scatter in the raw data, resulting in poor fits. There is a good correlation, with $\lambda_T \sim 1.1\ \lambda_n$ on average (dashed line). The large open symbols show averages across the dataset for IWL (diamond) and LSN (circle) configurations, clearly demonstrating that in the IWL configuration, both λ_T and λ_n are ~2.5 times larger than in LSN and directly confirming one of the key ITER limiter load spec assumptions. The two available TL profiles have λ_T and λ_n comparable to LSN values (somewhat smaller than the LSN average), indicating that the ITER use of a modified divertor scaling law for limiter discharges has some validity.

In order to check the validity of the derivation of λ_q from the probe data, IRTV was used in LSN discharges to compare with the probe derived results. Out of 10 LSN profiles, 3 were obtained with the outer strike point (OSP) detached, and IRTV data could not be used. Six out of the remaining seven profiles show agreement to within a factor of 2 between λ_q values from IRTV (mapped to the LFS midplane) and the probe, which is reasonable within the measurement uncertainties.

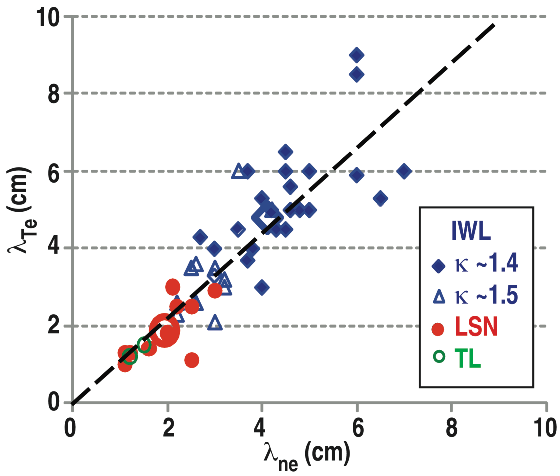

Fig. 2. Correlation between density and temperature e-folding lengths.

A comparison of the λ_q values derived from probe data of Fig. 2 with those calculated using the scaling in Eq. (1) is shown in Fig. 3(a) for the entire usable dataset, where, the IWL data have been scaled down by a factor of 2.5 to be comparable with LSN data and the scaling assumptions. It is evident from this comparison that our results do not confirm the assumed parametric dependence of the ITER λ_q scaling. However, the overall disagreement in absolute values is not very large. Moreover, not all experimental points may be suitable for comparison with the scaling. Equation (1) assumes attached conditions, while some of the higher density and lower I_p (higher q_{95}) discharges may have been detached. We do not have a good indication for detachment in IWL discharges, but those which are radiation-dominated (with low P_{SOL}) are likely to be detached. For LSN discharges IRTV data confirm that those with $P_{SOL} < 0.25$ MW are detached. In addition, a clear correlation was found between λ_q and δ, the distance between LCFS and the "knee limiter" (Fig. 1), with λ_q in higher κ, lower δ discharges being on the average ~30% lower than in lower κ, higher δ cases. Therefore, we conclude that proximity of the secondary limiter to the LCFS may affect the SOL width in higher κ discharges and that data from those discharges is likely to be suitable for comparison with the scaling of Eq. (1). Points with $P_{SOL} < 0.25$ MW and higher elongation have therefore been removed from Fig 3(b). All but one remaining IWL point and most LSN points (except

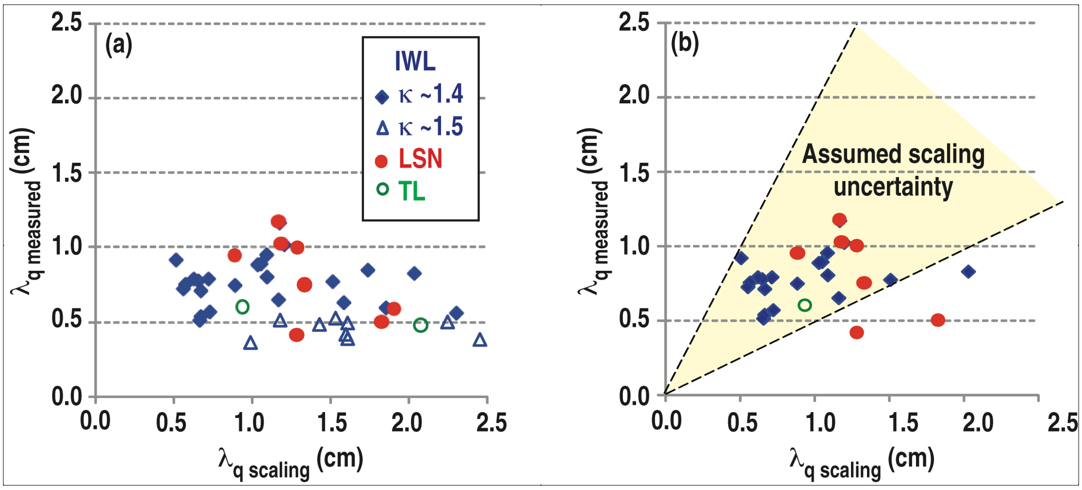

Fig. 3. Comparison of the measured heat flux e-folding length with assumed ITER scaling of Eq. (1) over full dataset (a) and with questionable points removed (b). Note that measured IWL values are scaled down by a factor of 2.5.

for two with P_{SOL} ~ 0.3 MW that are close to detachment) agree with the scaling within the assumed uncertainty factor of 2.

The primary goals of our experiments were to benchmark the ITER SOL power width scaling of Eq. (1) in both limited and diverted configurations and demonstrate the larger λ_q for HFS versus LFS limiter configurations. Three of five scaling parameters (q_{95}, $\overline{n}_e$ and P_{SOL}) were varied in a rather wide range, although they do not vary independently and it is thus impossible with this dataset to check the individual scaling dependencies of Eq. (1). Moreover, the measured λ_q values show no correlation with the scaling trends as the plasma parameters change. On the other hand, with the exception of detached discharges and those affected by a proximity of the secondary limiter, the absolute measured values of λ_q agree with the scaling within the assumed uncertainty of a factor of 2. This result provides some confidence that the scaling relationship may be a reasonable assumption provided that the FW design accounts adequately for the uncertainty.

We have shown that the SOL width measured at the outboard midplane in IWL configuration is on average ~2.5 times larger than in LSN, confirming the assumptions used by ITER. The strongest dependence of the scaling in Eq. (1) — the one on the major radius — could not be directly tested in our experiments. However, the fact that our results are in reasonable agreement with a scaling based on data from JT-60U and JET, machines with a considerably larger R, constitutes an approximate confirmation of the validity of the R^2 dependence in Eq. (1). This is an important result, greatly increasing the confidence in the application of Eq. (1) to ITER.

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[1] R. A. Pitts *et al*., Proc. 19th PSI Conference, San Diego, USA (2010), paper submitted to J. Nucl. Mater.
[2] A. Loarte, *et al.,* Proc. 22nd IAEA Fusion Energy Conf., Geneva, Switzerland (2008), Paper IT/P6-13, http://www-pub.iaea.org/MTCD/Meetings/fec2008pp.asp
[3] P.C. Stangeby, Nucl. Fusion **50** (2010) 035013
[4] M. Kocan and J.P. Gunn, Plasma Phys. Control. Fusion **52** (2010) 045010.