TRANSPORT IN THE DIVERTOR REGION OF TOKAMAKS AND ROLE FOR POWER EXHAUST IN CONVENTIONAL AND ALTERNATIVE DIVERTORS

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Developing a successful power exhaust solution for future fusion devices [1] depends crucially on our understanding of the transport processes in the Scrape-Off Layer (SOL) and the Private Flux Region (PFR). Increased transport in these regions broadens the target power flux profile, thereby reducing peak heat fluxes. Although transport towards the PFR is of similar importance as the SOL transport in determining the target heat flux width in present-day and potentially even more so in future devices, research has largely focused on the SOL transport, parametrized by the decay length λ_{SOL}. Thus, despite its importance, the PFR transport is comparatively poorly understood. In this contribution we present a novel fully analytical model for the minimum PFR decay length [2], which allows estimates for the integral heat flux width in future fusion devices. The model agrees quantitatively with measured PFR decay lengths obtained from target Langmuir probe profiles in ASDEX Upgrade (AUG). Additionally, our model provides an explanation for the approximately constant ratio between λ_{SOL} and the divertor spreading factor S, observed across several tokamaks [3]. Applying the model to the next-step tokamaks ITER and SPARC [4] results in a similar heat flux broadening as for present-day conventional divertors. However, for advanced divertor designs, with a large outer target radius, the model indicates increased target heat flux widths, which eases power exhaust constraints. Specifically, increasing the strike point major radius by about one third of the major radius, could halve the required impurity concentration for accessing detachment compared to conventional divertor designs.

The target heat flux width is a key parameter for developing power exhaust strategies for future tokamaks. Detachment scalings [5, 6] indicate that the impurity concentration required for accessing detachment decreases approximately inversely with the heat flux width. Using the SOL decay length λ_{SOL} and the divertor spreading factor S, the integral target heat flux width can be parametrized as $\lambda_{int} \approx \lambda_{SOL} + 1.64S$ [7]. λ_{SOL} has been extensively investigated in theoretical, numerical and experimental studies. According to an established scaling [3,8], λ_{SOL} scales with the poloidal *hybrid* ion Larmor radius, defined by $\rho_{i,p} = m_i c_s/(ZeB_p)$. Here m_i is the ion mass, c_s the sound speed, Z the charge number, and B_p the poloidal magnetic field. Due to the inverse B_p dependence of λ_{SOL}, narrow SOL widths $\lambda_{SOL} < 1$ mm are predicted for ITER and SPARC. Although recent studies [9, 10] have demonstrated that λ_{SOL} exceeds the scaling in certain regimes, low λ_{SOL} values remain a major concern for the operation of future tokamaks. Under such conditions, transport in the PFR could significantly broaden the target heat flux. However, compared to the SOL transport, the PFR transport is currently poorly understood. Although semi-empirical experimental scaling laws for the spreading factor S have been developed [11, 12], the underlying transport mechanisms in the PFR remain uncertain and no extrapolation to future tokamaks has been attempted. This contribution presents an analytical model [2] for the ion flux PFR width. The model assumes that the particle transport into the PFR is dominated by drifts, resulting in the following ion PFR decay length:

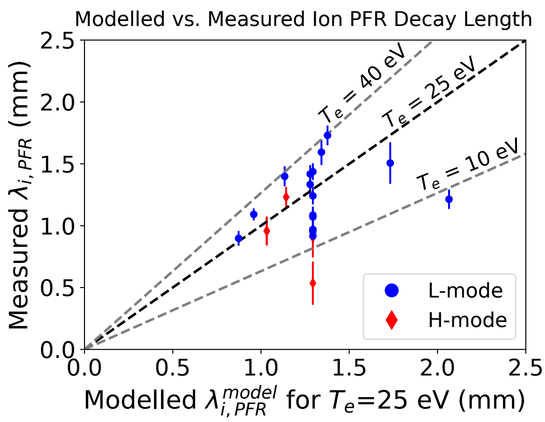

Figure 1: Modelled ion saturation current PFR decay length versus measured values in AUG in L- and H-mode datasets.

$$\lambda_{i,PFR,ot}^{model} = 4\frac{R_{ot}-R_{xp}}{R_{xp}}\rho_{i,p,ot} \qquad (1).$$

Here R_{ot} and R_{xp} are the outer target and X-point major radii, respectively, and $\rho_{i,p,ot}$ is the poloidal hybrid ion Larmor radius at the outer target. Figure (1) compares the modelled PFR width given by equation (2) to experimentally obtained PFR widths of ion saturation current profiles measured by Langmuir probes in AUG L- and H-modes. In the model an electron temperature of $T_e = 25$ eV along the outer divertor leg is assumed for calculating $\rho_{i,p,ot}$. However, since the exact divertor leg temperature is uncertain, additional model predictions for $T_e = 40$ eV and 10 eV are included as upper and lower dashed lines, respectively. The measured PFR widths fall predominantly between these two model predictions, indicating that the model is able to give a quantitative estimate for the PFR width within a factor of two. Furthermore, since the PFR width model depends as λ_{SOL} on the hybrid Larmor radius, the model explains the approximately constant ratio between the divertor spreading factor S and λ_{SOL} previously measured across various tokamaks [3].
Combining the PFR width model with the theoretical expression for λ_{SOL} from Ref. [8] yields a model for the integral heat flux width λ_{int}, which is a critical parameter for power exhaust. Figure 2 (a) shows the predicted relative broadening $\lambda_{int}/\lambda_{SOL}$ as a function of the outer target radius R_{ot}/R_{xp}. For AUG, the model gives a $\lambda_{int}/\lambda_{SOL}$ of about 2, while for ITER and SPARC about 20% lower values are obtained. Figure 2 (a) also suggests that a divertor design with a large outer target radius, as for example implemented in the Super-X divertor [13], could increase λ_{int} significantly. Shifting the outer target radius outwards has the additional advantage of increasing the parallel flux tube area, due to the reduced magnetic field. The combined effects of a larger λ_{int} and the expansion of the flux tube area lead to a significant reduction of the impurity concentration $c_{z,det}$ required for accessing detachment: Applying the detachment model of Ref. [5], results in figure 2 (b), showing the dependence of $c_{z,det}$ on R_{ot}/R_{xp}. Here, the normalization constant $c_{z,det,0}$ is the detachment access impurity concentration for $\lambda_{int} = \lambda_{SOL}$ in the respective device. For AUG, ITER and SPARC, $c_{z,det}/c_{z,det,0}$ is about 50%, while increasing R_{ot} to $1.3R_{xp}$ reduces it to about 22%. Thus, a divertor design with a large outer target radius could reduce the required impurity concentration for accessing detachment, thereby reducing the risk of unacceptably high core impurity radiation and dilution.

In summary, the results on the PFR width presented in this contribution, together with the previous studies on the SOL decay length, complete our understanding of the processes determining the integral target width. These findings are therefore a crucial step towards extrapolating and controlling power exhaust in future divertor tokamaks.

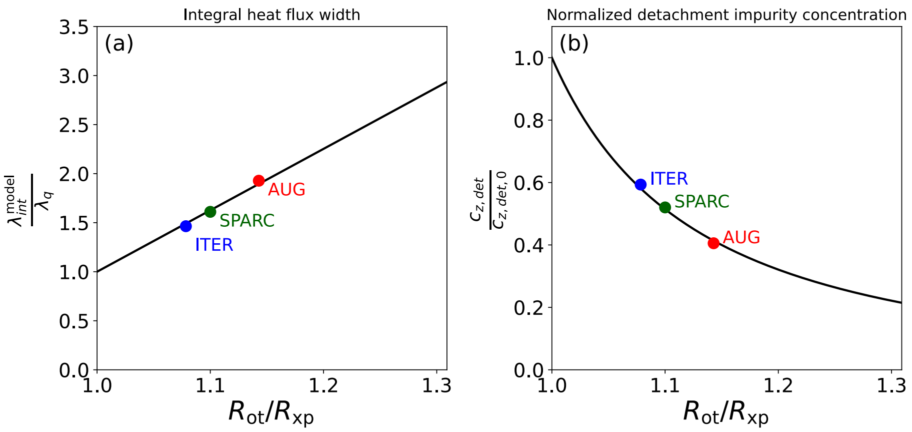

Figure 2: Modelled dependence of (a) normalized integral target profile width λ_{int}, normalized to the SOL heat flux width λ_q and (b) resulting impurity concentration $c_{z,det}$, normalized to $c_{z,det,0}$, required for accessing detachment, on outer target to X-point radii ratio R_{ot}/R_{xp}. Here, $c_{z,det,0}$ is the required impurity fraction for $R_{ot} = R_{xp}$.

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