FAVOURABLE MODIFICATIONS OF SCRAPE-OFF LAYER (SOL) HEAT FLUX WIDTH THROUGH PULSED FUELLING IN ADITYA-U TOKAMAK

[1,2]SK INJAMUL HOQUE, [1,2]H. RAJ, [3]R. DEY, [1,2]S. BANERJEE, [1,2]K. YADAV, [1,2]K. SINGH, [1,2]S. DOLUI, [1,2]A. KUMAR, [1,2]A. KUMAWAT, [1,2]B. HEGDE, [1,2]A. KUNDU, [4]S. PATEL, [1,2]S. AICH, [1]R. KUMAR, [1]P. GAUTAM, [1]K. A. JADEJA, [1]K.M. PATEL, [1]P. VERMA, [1]R.L. TANNA, AND [1,2]J. GHOSH

[1]Institute for Plasma Research, Gandhinagar, India
[2]Homi Bhabha National Institute, Mumbai, India
[3]Department of Physics, Indian Institute of Technology Tirupati, Yerpedu, India
[4]University of Virginia, Department of Physics, Charlottesville, USA

Email: injamul.hoque@ipr.res.in

Enhancement of the scrape-off layer (SOL) heat flux width has been observed in the ADITYA-U Tokamak following the injection of short fuel gas pulses (~10^{17}– 10^{18} molecules). A notable reduction in parallel heat flux ($q_{\parallel}$) near the last closed flux surface (LCFS) is observed after each pulse. The measured global cross-field diffusion coefficient is increases and closely matching the Bohm diffusion coefficient during gas pulse injection. Comparative analysis indicates that pulsed fuelling is more effective in mitigating heat flux with improved core confinement than continuous gas feeding via real-time density control. Simulations with UEDGE code suggest that an increase in both cross-field diffusion coefficient and inward pinch velocity is necessary to replicate the experimentally observed broadening of the heat flux SOL width. These findings provide insights into efficient SOL heat flux control strategies for future fusion devices.

Effective heat flux management is crucial for protecting plasma-facing components (PFCs) and ensuring the safe long-term operation of tokamaks. A key parameter in this context is the scrape-off layer (SOL) heat flux width (λ_q) which determines how rapidly heat flux decreases from the last closed flux surface (LCFS) to the vessel wall. Studies indicate that λ_q decreases with increasing plasma current and density, posing a significant challenge for larger tokamaks during initial phase of plasma current ramp-up in limiter configurations. A broader λ_q is desirable as it reduces the localized heat load on limiters and divertors. Several heat flux mitigation strategies have been investigated, including impurity seeding [1], which increases SOL resistivity through radiative cooling. The effect of impurity seeding in different divertor configurations is also studied [2]. The real-time divertor heat flux control is achieved in COMPASS tokamak using impurity puffing with Langmuir probes and ball-pen probes [3]. Recent findings also suggest that high-frequency turbulence in quiescent H-mode plasmas can broaden λ_q [4], offering potential pathways for heat flux mitigation. Building upon these insights, in this article, our new findings report an optimized strategy for heat flux mitigation with improved core plasma confinement.

To study the effect of pulsed fuelling on edge heat flux, experiments are carried out with periodic short gas pulses and continuous gas feed by real time density control in plasma current flat-top of a single discharge (in different time window). A triple Langmuir probe (TLP) is used for measurement of heat flux with simultaneous measurement of ion saturation current and plasma temperature at normalised radius ρ = 0.992. It is observed that in the case of continuous gas fuelling there is steady heat flux of around 0.7 Mw/m^2 at edge in flattop of plasma discharge, whereas in case of pulsed fuelling there is periodic suppression of heat flux with average level of 0.4 Mw/m^2 (shown in figure 1). Our findings also indicates that pulsed fuelling is more efficient in controlling heat flux along with improved core confinement than the continuous gas fuelling. Further, SOL heat flux profile and heat flux SOL width are measured with an array of Langmuir probes. It is observed that immediately after each gas pulse SOL heat flux profile becomes flatter (i.e. λ_q increases). Assuming the simple SOL model, perpendicular particle flux into SOL can be equalled to parallel particle flux in SOL. This gives the relation between heat flux SOL width (λ_q) and cross-field diffusion ($D_{\perp}$),

$$\lambda_q = \left(D_{\perp}\frac{l_c}{c_s}\right)^{1/2} \quad \text{…….. (1)}$$

Where, l_c is connection length and c_s is ion sound speed in plasma. The effect of gas pulse on λ_q and $D_{\perp}$ are shown in figure 2.

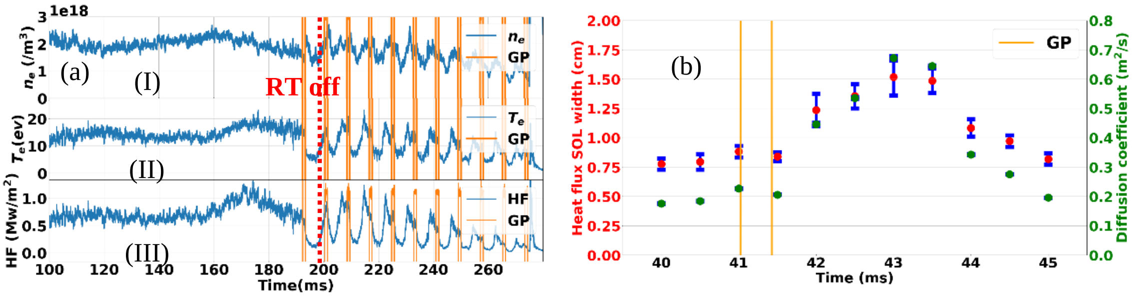

Figure 1: (a) Measurements of edge (I) density (II) temperature and (III) heat flux from TLP at ρ = 0.992 in shot 34647. (b) Variation of heat flux SOL width and cross-field plasma diffusion measured with Rake Langmuir probes in shot 36496. The orange vertical lines show time of gas pulses. The red dotted line in figure 1(a) shows the end time of continuous fuelling through real-time density control.

Thus, the increase in λ_q during pulsed fuelling can be explained in terms of increase in cross-filed diffusion from edge to SOL plasma. For validation of experimental results simulations have been done with UEDGE code taking the cross-field diffusion coefficient and inward pinch velocity as input. Taking $D_\perp$ of 0.2 m^2/s and inward ware-pinch velocity, V_p of 1.5 m/s simulated profile matches with edge density profile as experimentally measured before gas pulse. Simulation results show to match the experimental density profile after gas pulse increase in pinch velocity is needed along with the increased value of $D_\perp$. The most close match with experiments is found with $D_\perp$ = 0.6 m^2/s (same as measured experimentally) and V_p = -5 m/s.

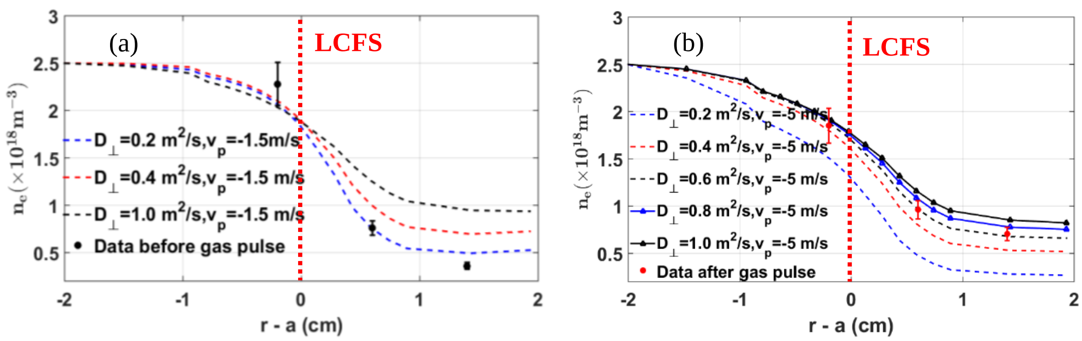

Figure 2: Density profiles from Simulation with UEDGE code (a) Before gas pulse results (dotted lines) with experimental data points (b) After gas pulse results (dotted lines) with experimental data points. Red dotted lines show the position of LCFS.

In conclusion, an application of short fuel gas puff results in increased cross-field diffusion to SOL along with increased inward pinch velocity. This leads to a broadening of SOL heat flux width subsequently resulting in an improved core confinement.

REFERENCES

[1] F. Nespoli et al. Impurity seeding for suppression of the near scrape-off layer heat flux feature in tokamak limited plasmas 2018 Phys. Plasmas 25, 052501

[2] Harshita Raj et al. Improved heat and particle flux mitigation in high core confinement, baffled, alternative divertor configuration in the TCV tokamak 2022 Nucl. Fusion 62 126035

[3] I Khodunov et al. Real-time feedback system for divertor heat flux control at COMPASS tokamak 2021 Plasma Phys. Control. Fusion 63 065012

[4] D. R. Ernst et al. Broadening of the divertor heat flux profile in high confinement tokamak pedestals limited by turbulence in DIII-D 2024 PHYSICAL REVIEW LETERS 132,235102