PLASMA TRANSPORT STUDY WITH 3D SHAPED FIRST WALL FOR LIMITER RAMP-UP PHASE OF ITER

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The early phase of each ITER plasma discharge takes place in a limiter configuration on the central column before transitioning to the divertor phase [1]. The ramp-up is an essential phase that has to be successfully negotiated such that power radiated upstream allows low power to enter in Scrape-off Layer (SOL) and enables the start-up plasma to reach to the desired divertor phase. In this study, the plasma boundary simulations are performed for the ITER inboard ramp-up phase using the 3D plasma transport code EMC3-Eirene [2]. One of the outstanding issue in ITER device is the incident power flux on the inboard first wall panels (FWPs). The FW is toroidally and poloidally to protect the leading edges between the adjacent panels. However, this shaping causes the power flux focusing onto the surfaces within the magnetically wetted areas.

The present 3D SOL plasma simulation work using EMC3-Eirene is the first effort in this direction where the coupled plasma-neutral transport is simulated for the shaped FW of ITER. The simulations are performed by employing a numerical grid based on the full 3D CAD geometry of the FWPs. A sufficiently large toroidal domain is included to capture all shadowing effects. The limiter ramp-up on ITER is rather slow (~11 secs from the burn-through to the transition to the diverted phase). The simulations are performed for the pure Hydrogen plasma of a fixed, slightly elongated magnetic equilibrium at $\mathbf{I_p \approx 2}$ **MA**, $\mathbf{B_T = 5.3}$ **T**, with systematically varied scrape-off layer (SOL) input power and separatrix density, based on predictions from DINA ramp-up simulations [3]. These conditions match those used in a related 2D SOLPS-ITER study, which considers impurity evolution, while this work focuses on pure hydrogen plasmas.

Along with the **3D-shaped first wall (FW)** study of plasma transport, a **2D axisymmetric FW simulation** has also been performed to benchmark EMC3-Eirene results against the 2D transport code SOLPS-ITER**.** A simplified, plain FW geometry is considered for this study, ensuring that identical input parameters are used for both codes. The comparison of plasma characteristics from the outputs of both simulations shows a similar range of values for key parameters, demonstrating consistency between the two approaches.

The present hydrogen plasma transport study is performed for the inboard limited ramp-up plasmas having power crossing the SOL is P_{SOL} = 2 MW and $n_{up} = 4 \times 10^{18}$ m^{-3} with uniform particle and energy diffusivities, $D_\perp = 0.5$ m^{2}s^{-1}, $\chi_\perp = 1$ m^{2}s^{-1}, respectively. The moderate power input of P_{SOL} = 2 MW into the SOL is considered as the high radiational loss expected from the W impurities which however are not treated as additional species in the present simulations covering transport in the SOL region based on their rather dominant radiation from the core region. Their major impact is therefore accommodated by considering lower input power entering into the SOL from the inner radial boundary of the simulation, or the LCFS which is the core-SOL interface covered by the present 3-dimensional computational grid. The 2D surface plots in the following discussions represent the plasma density and temperature on the (radial-poloidal) cross-section of the torus at a fixed φ .

The 2D radial-poloidal plane distribution of plasma density and electron temperature at the toroidal location near the apex region, i.e. $\varphi = 5^\circ$, is presented in Fig. 1 (a) and (b), respectively. The plasma density in the closed field line region is comparatively higher than in the SOL region and decreases on moving radially out. At the inboard mid-plane side, however, the plasma density grows higher with in the tangency region as the flux tubes are nearly parallel to the wall curvature in this recycling-rich zone, making it a higher-density region. The 2D electron temperature distribution indicates a low-temperature near-edge region of the orders of a few eVs and a high temperature near the core region.

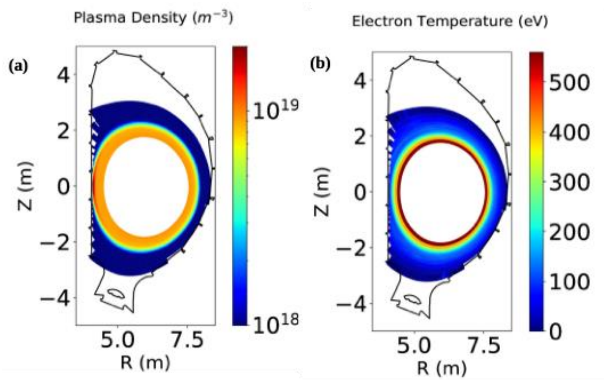

Fig.1 Plasma Density and temperature profile on the radial-poloidal cross-section at one toroidal location ($\phi = 5°$)

The 2D radial-poloidal distribution of the mach number and the particle flux at the toroidal location near the apex, i.e. φ = 5◦, is plotted in Fig. 2 (a) and (b) respectively. The strong downstream flow is observed on reaching the limiter surface due to the Bohm's boundary condition. The blue and red colour indicates the opposite directions of the toroidal flows. The ion flux observed in the SOL region is due to conventional boundary flows while in the near core region, the flows are opposite and are present due to poloidal density gradients present near FWP #3 as observed in Fig 1 (a) at the IMP of PFC.

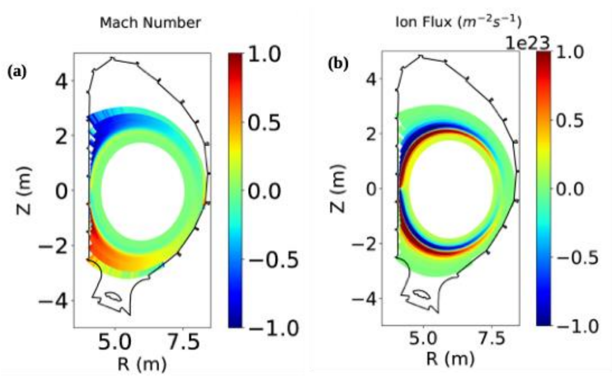

Fig.2 Mach Number and particle flux profile on the radial-poloidal cross-section at one toroidal location ($\phi = 5°$)

This profile serves as the first results and the advanced version of this study will include the W impurity species as actively radiating and undergoes transport. The input SOL power for such self-consistence study will accommodate the fraction radiated by the W species.

This work highlights the challenges associated with fully 3D grid-based transport simulations. These studies have important implications for the safe and efficient operation of ITER, particularly in terms of predicting and mitigating power loads on the FWPs. Moreover, this refined modeling approach contributes to the optimization of operational strategies for future fusion reactors and enhances the predictive capabilities of 3D plasma transport simulations.

References

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