SOLPS-ITER simulations of the COMPASS tokamak

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Tokamak transport codes are an invaluable tool in assessing the edge plasma physics. SOLPS-ITER [1] is a state-of-the-art transport code, developed by the ITER Organisation. Its complex treatment of neutrals by the Monte Carlo code EIRENE allows it to accurately capture plasma recycling and neutral transport. [2] This facilitates detailed investigations of momentum and power losses in the SOL, ranging from the simple SOL to detachment. In this contribution, we present the first interpretative SOLPS-ITER simulation of the COMPASS tokamak.

The COMPASS tokamak [3] is a compact machine operated at the Institute of Plasma Physics in Prague, Czech Republic. Its extensive edge diagnostics coverage [4] synergises well with interpretative SOLPS-ITER modelling. The principal diagnostics used in this contribution are depicted in Figure 1. Upstream measurements of T_e and n_e are facilitated by the Thomson scattering diagnostic (plasma top). Target measurements are carried out by an infrared camera (total parallel heat flux density $q_{\|}$) and a divertor probe array of ball-pen and Langmuir probes ($q_{\|}$, T_e and n_e at the outer target). Bolometric diagnostics are used to determine the power radiated in the plasma core, as their divertor coverage is not suitable for gauging the divertor radiation distribution. Lastly, these diagnostics are supplemented by a magnetic equilibrium reconstruction using the EFIT++ code. [5]

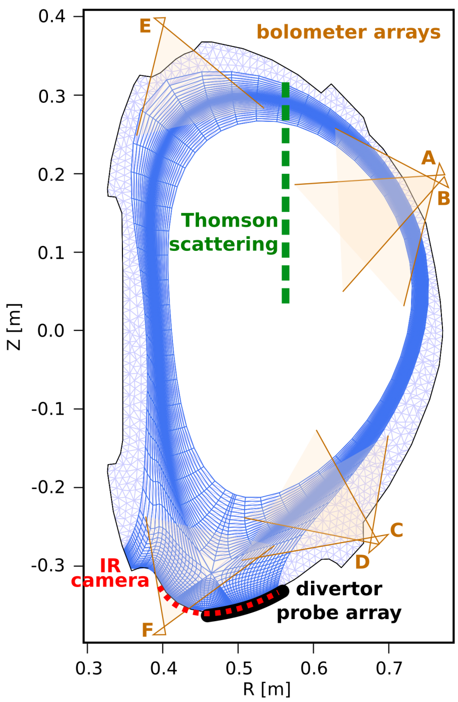

Figure 1: *Edge plasma diagnostics of the COMPASS tokamak, and B2.5 (blue) and EIRENE (lavender) grids.*

We present a simulation of the COMPASS tokamak discharge #17588 at the time $t = 1100$ ms. It is a deuterium Ohmic L-mode plasma in the divertor configuration, with the ion grad-B drift directed toward the divertor. The plasma current is $I_p = 180$ kA, the toroidal magnetic field is $B_t = 1.38$ T, the safety factor is $q_{95} = 4.2$ and the line-averaged density is $\bar{n}_e = 5 \times 10^{19}$ m^{-3}. The ohmic heating power is $P_{ohm} = 200$ kW, of which

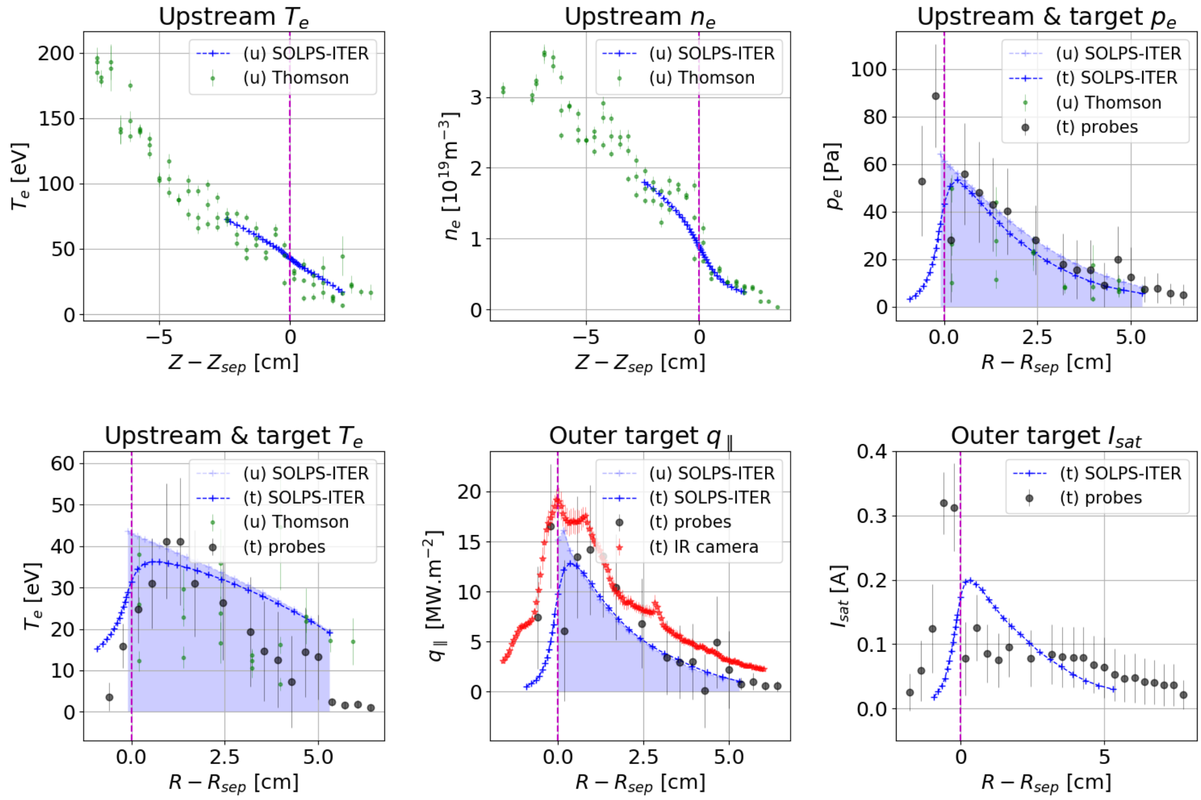

Figure 2: *Experiment-model comparison of the presented COMPASS simulation. Quantities labelled with (u) are taken from the upstream (plasma top, except for the $q_\parallel$ where the outer X-point is considered) and quantities labelled with (t) are taken from the outer target.*

$P_{rad} = 65$ kW is radiated in the core, yielding the power crossing the separatrix $P_{sep} = 135$ kW. The SOLPS-ITER simulation is coupled (B2.5+EIRENE) and drift-free. It does not account for any impurities, such as the carbon sputtered from the graphite divertor. Its resulting simplicity allows studying various simulation inputs; selected insights are discussed below.

Experiment-model comparison. Figure 2 shows that the simulation reproduces the experimental measurements with good accuracy and the profiles of T_e, n_e, $q_\parallel$ and I_{sat} match both upstream and at the outer target. The inner target heat fluxes (not shown here) are reproduced as well. This agreement may suggest that carbon impurities play a minor role in this discharge.

Magnetic equilibrium reconstruction. The equilibrium reconstruction forms the basis for building the B2.5 and EIRENE grids (figure 1). We compared two SOLPS-ITER simulations with identical inputs except for the equilibrium reconstruction used for grid construction. The two equilibrium reconstructions were constrained by magnetic measurements in the "standard" and "optimised" input configuration (more information in [6]), respectively, and their main difference was the separatrix outline ($\Delta Z = 2.1$ cm along the Thomson scattering chord). It was found that the "optimised reconstruction" (results presented herein) matched the upstream T_e

and n_e profiles automatically, while the "standard reconstruction" required an *ad hoc* upstream radial shift of 1.8 cm before it could reproduce the experimental results. This supports our previous suggestion that the "optimised" reconstructions are more accurate and illustrates the importance of high-quality equilibrium reconstructions in edge transport modelling.

Cross-field diffusion coefficient. In interpretative modelling, the diffusion coefficients associated with anomalous cross-field transport are usually determined by iteratively matching the upstream profiles. Here, this process yielded $D_n = 0.15\ \mathrm{m^2s^{-1}}$ and $\chi_e = \chi_i = 4\ \mathrm{m^2s^{-1}}$ (equal for lack of data). The D_n value was then compared to four D_n estimates:

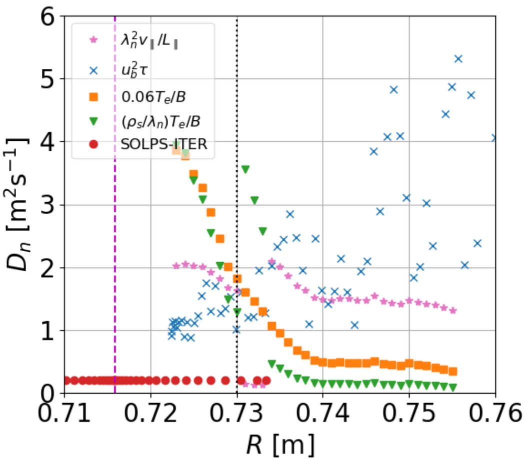

Figure 3: *Radial profiles of diffusion coefficient estimates (see text for details).*

(i) The interplay of parallel and perpendicular transport may be approximated as $D_n = \lambda_n^2 v_\parallel / L_\parallel$ [7, Eq. (2)], where λ_n is the density fall-off length, $v_\parallel = Mc_s$ is the characteristic upstream parallel velocity and $L_\parallel$ is the connection length.

(ii) [7] newly suggests calculating D_n from the edge turbulence properties, $D_n = u_b^2 \tau$, where $u_b = \mathrm{Var}(v_r)$ is the characteristic radial blob velocity and τ is the v_r autocorrelation time.

(iii) The Bohm scaling posits that $D_n = 0.06T_e/B$.

(iv) The gyro-Bohm scaling posits $D_n = (\rho_s/\lambda_n)T_e/B$ where $\rho_s = \frac{\sqrt{m_i T_e}}{eB}$ is the Larmor radius.

The simulated discharge #17588 lacks horizontal reciprocating probe measurements needed to infer the listed plasma parameters; instead, we used the roughly similar discharge #6878. Figure 3 shows that the D_n estimates are in a bare order-of-magnitude agreement. This is a similar result to [7], but its uncertainty lends little clarification to the D_n value or physical meaning. While interpretative modelling can use the iterative matching procedure without rigorously justifying the anomalous diffusion coefficient, D_n choice in predictive modelling of future machines should be well informed. We conclude that this topic requires further inquiry.

The edge transport regime was gauged based on two criteria anchored in the two-point model [8]: the upstream-target T_e gradient and the momentum and power loss factors

$$1 - f_{mom} = \frac{p_t}{p_u} \qquad (1) \qquad \text{and} \qquad 1 - f_{pow} = \frac{q_{\parallel t} R_{\parallel t}}{q_{\parallel u} R_{\parallel u}}, \qquad (2)$$

where p is the total plasma pressure and the subscripts denote upstream (outer X-point) and target (outer target). As figure 2 shows, $T_{et} \approx T_{eu}$, and figure 4 indicates only small momentum and power losses. This corresponds to the sheath-limited regime at the outer target. At the

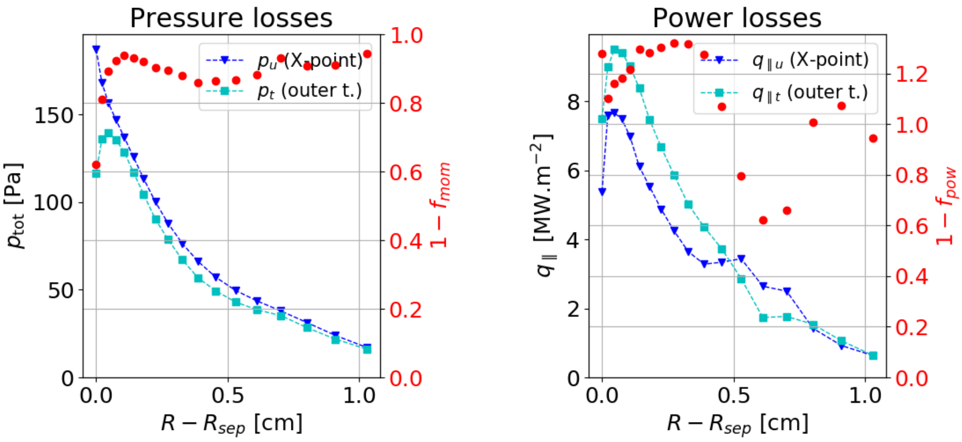

Figure 4: *Simulated profiles of the total plasma pressure p and the total parallel heat flux density* $q_\parallel$. *The corresponding loss factors* $1-f_{mom}$ *and* $1-f_{pow}$ *are plotted in red.*

inner strike point, $T_u/T_t \approx 2$ and the momentum and power losses remain small. Considering discharge #17588 is representative of a typical moderate-density COMPASS tokamak plasma, we conclude that the COMPASS tokamak typically operates in the sheath-limited regime. It is known that the transport regime can affect the spreading of turbulent structures in the SOL [9], and therefore we advise caution when using COMPASS in scaling studies together with machines operating typically in the conduction-limited regime.

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