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SOLPS-ITER modelling of reference plasmas in the ITER Research Plan

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The design of the ITER divertor and assessment of its performance have relied on an extensive boundary plasma simulation effort, using the SOLPS suite of codes as the main tool [1]. The main burning plasma divertor simulation database [1], extensively used in the divertor design activity, in the specification of the ITER fuel cycle, and deployed extensively to provide synthetic diagnostic signals, assumes that erosion of the beryllium (Be) main chamber walls will lead, through material migration, to the tungsten (W) divertor targets being coated with Be. In addition, gas fueling for detachment control, including extrinsic impurities, is assumed to be injected through gas feed lines in the main chamber.

A recent substantial revision of the ITER Research Plan (IRP) [2] follows the staged approach to ITER operation and specifies in more detail the early, non-active phase operational campaign, in which hydrogen discharges with shorter pulse duration and lower scrape-off layer (SOL) power, P_{SOL} will be the first to be studied. Since the original high power (P_{SOL} = 100 MW) database was constituted, there have been many developments in the simulation code itself, and in 2015, SOLPS-ITER [3], the most advanced SOLPS package, was launched. Results are now routinely stored as ITER Integrated Modelling and Analysis Suite (IMAS) Interface Data Structures (IDS) in a public database [4].

To characterise divertor performance in the early campaigns and examine the sensitivity to the assumptions of Be coated targets and gas puff location made for the main burning plasma database, a new simulation programme with SOLPS-ITER is underway at the ITER Organization. Studies have also been conducted of the feasibility of using "raised strike point equilibria" as a means to outgas fuel from Be co-deposits accumulating on the inner divertor baffle region and, of the trend, found in the high P_{SOL} database and observed also at lower power, for the upstream separatrix density to saturate with increasing divertor neutral pressure (and hence detachment degree).

The key findings, to be discussed in detail during the presentation, may be summarized as follows:

- Switching from a fully coated Be surface to a pure W target mainly affects the ratio of atoms and molecules among the recycled neutrals, modifying the electron temperature in the target vicinity. The contributions to momentum transfer and power losses from molecule-plasma and atom-plasma interactions tend to compensate one another, reducing the impact of divertor surface material.
- At low P_{SOL}, gas puffing location (main chamber vs divertor) has only marginal effect since in the near SOL regions, most important for dissipation, neutral particles rapidly accumulate near the divertor.
- The observed saturation of upstream separatrix density with increasing fuel throughput (hence higher divertor neutral pressure or increased detachment degree), can be ascribed to the ITER vertical target geometry, which promotes strong local cooling in flux tubes close to the separatrix.
- The initial studies of raised strike points reveal that for the maximum plasma current and input power at which such equilibria are likely to be possible, the inner target heat flux density will be insufficient to raise the surface temperature to values at which significant fuel desorption will occur.

References

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[3] Bonnin X. et al., 2016 Plasma Fusion Research 11 1403102
[4] Imbeaux F. et al., 2015 Nucl. Fusion 55 123006