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Physics drivers of the STEP divertor concept design

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The STEP programme aims to demonstrate viability of a GW-scale spherical tokamak reactor, with parameters around: geometric axis 3.6m, aspect ratio 1.8, elongation 2.9, plasma current 20MA, on-axis magnetic field 3.2T, core radiated power 350MW, with steady state power crossing the separatrix 150MW. Here we give a physics overview of the studies driving forward the STEP divertor concept design, focussing on operational phases during current ramp-up and flat-top.

The inner divertor in a spherical tokamak is challenging, with total flux compression along the leg and small strike point major radius suggesting unmitigated surface power loads of GW/m^2. The down-selected operational point thus focused around a double-null configuration, with surface area and poloidal variation of transport weighting the core power outflux towards the outboard. The outer divertor leg was extended to large major radius to increase the target wetted area, with unmitigated loads up to ~200MW/m^2, managed by seeding. Our recent studies indicate how synergy between connection length and flux expansion can be used to improve the detachment threshold and window, as well as front location sensitivity. Advantages of double-null geometry will only be realised with vertical position control precision to 1-2 millimetres, based on anticipated heat flux decay widths, so we have examined the impact of separatrix disconnection, dR_{sep}, on target fluxes. Up-down power sharing was found to be consistent with previous experimental modelling [Brunner et al (2018) Nucl. Fusion 58 076010], but the strong flux compression from outboard midplane to inboard target contributed to weak sensitivity of in-out power sharing, with the impact of drifts on this conclusion under investigation. This motivates testing of dynamic double-null (full power alternating from upper to lower divertor) for practical vertical stability control, with modelling indicating strong frequency and amplitude dependence of divertor component fatigue by temperature cycling. Partial detachment was recovered in a vertical inner target configuration, with high far-SOL temperatures driving unacceptable erosion. Full detachment required direct extrinsic impurity seeding or the introduction of an advanced divertor geometry, with the former giving unacceptably high core fuel dilution.

A toolkit has been developed which allows optimisation of divertor magnetic geometries in free boundary magnetic equilibria, in combination with compatible first wall geometries, while holding constraints within allowed margins. Constraints include core profiles, informed by stability and transport analysis, and field coil positions, defined by neutron shielding and current limitations, while divertor performance measures, such as connection length and poloidal flux expansion, are returned.

Performant geometries are passed to SOLPS-ITER, to assess trends in power handling ability.
The detached operating space is identified in terms of deuterium puffing and argon seeding, while minimising gas throughput, and maintaining low upstream density and impurity concentration. The inner divertor now focuses on a design approaching an X-divertor, finding substantial improvements in performance allowing reduced seeding levels, with ionisation profiles capturing the benefits of angled targets. The effect of pump location and inner target geometry on helium pumping efficiency is being optimised. The potential impact of gas puff location and pumping speed are also studied.

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