UK STEP TOWARDS A FUSION POWER PLANT PLASMA

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The UK's Spherical Tokamak for Energy Production (STEP) programme has achieved its first major milestone of providing a conceptual design for a prototype magnetic confinement fusion power plant targeting 2040 [1]. Central to the design of STEP is a highly elongated, fully non-inductive (NI) plasma solution [2] as well as its control [3]. The current required in addition to the intrinsic bootstrap current ($f_{\mathrm{BS}} = I_{\mathrm{BS}}/I_{\mathrm{p}} \sim 0.8 - 0.9$) is driven by microwave-based heating and current drive (HCD) methods only. Options have been explored for (1) electromagnetic electron cyclotron wave current drive (ECCD) alone, and (2) a mix of on-axis ECCD and off-axis electrostatic electron Bernstein waves (EBCD) [4]. The normalised current drive efficiency for EBCD is predicted to be three times higher than ECCD, opening up the possibility to access a $Q_{\mathrm{fus}} = P_{\mathrm{fus}}/P_{\mathrm{aux}} \sim 30$ flat-top operation point (FTOP) compared to the ECCD only $Q_{\mathrm{fus}} \sim 11$ FTOP required for a net electricity output of $P_{\mathrm{net}} > 100$ MW [2,5]. The published design point (SPP-001) with $R_{\mathrm{geo,1}} = 3.6$ m, $A = 1.8$, $B_t\left(R_{\mathrm{geo}}\right) = 3.2$ T, which is predicted to generate $P_{\mathrm{fus}} \sim 1.5 - 1.8$ GW [2], has proven to be technically challenging due to the very limited inner build radius of $R_{\mathrm{ib,1}} = 1.5$ m. This has led to a design pivot to explore a larger design point (SPP-002) with $R_{\mathrm{geo,2}} = 4.3$ m with the same aspect ratio and fusion power, while continuing to pursue ways to reduce the size. In all cases, STEP plasma parameters are far from today's experimentally accessible regimes. Substantial work on both of these fully non-inductive design points has: extended the theoretical basis, reduced the uncertainty of the scenario and its control [3] and developed operating scenarios. This contribution gives an overview of the impact of the size change, and the more advanced understanding of the physics base for the STEP plasma scenario and its control.

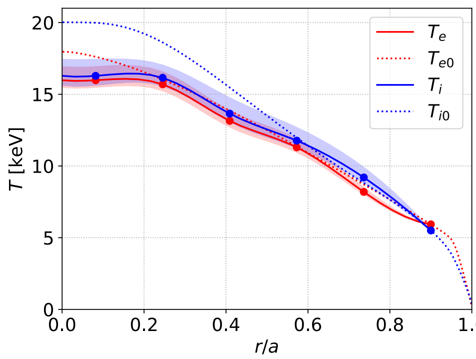

Figure 1: Predicted temperature profiles for SPP-001 (solid) from hKBM transport using a fixed equilibrium. The dashed line are the initial profiles from the integrated scenario simulation. The bands mark the range of solutions with different assumptions

In the absence of validated and sufficiently fast predictive transport models, the integrated scenario modelling is used as assumption integration with a Bohm-gyro Bohm (BgB) transport assumption scaled to achieve the fusion performance predicted by system code evaluations [4]. The possible scenario space is constrained by 7 conditions [5] including the divertor heat load $\frac{P_{sep}}{R_{geo}} < 40\,\frac{\mathrm{MW}}{\mathrm{m}}$, $Q_{fus} > 10$ and $q_{min} > 2.3$, and optimised with the aim of minimising the confinement assumption with respect to empirical scaling laws (e.g $H_{98(y,2)}$). Nonlinear gyrokinetic simulations for the derived profiles show that the turbulent transport in STEP is dominated by hybrid kinetic ballooning modes (hKBMs) with subdominant micro-tearing modes (MTM) [6] which can cause very large unsustainable transport fluxes. A new quasilinear reduced model for the hKBM transport will be described, including our first flux driven transport calculations which have increased the confidence in the existence of the FTOP solution with only a small decrease in fusion power by 10% [6] (see Figure 1). This transport solution is surprisingly insensitive to plasma rotation but relies heavily on the stabilisation due to the electron normalised pressure gradient β_e'. In addition, the impact of the high α-particle pressure and impurities in the burning plasma are neglected. A full integration into the JINTRAC framework or similar is planned. Due to the apparent stiffness the pedestal pressure is important, which is assumed to be ~15% below the stability limit for type-I edge localised modes (ELMs) accounting for performance losses due to operating in a no- or small-ELM regime or suppressing ELMs by 3D resonant magnetic field perturbations. The NI scenarios are very sensitive to the current drive efficiency and optimising the current drive power is key. CRAYON, a new fast relativistic ray tracing tool, modelling shows that relativistic EBCD rays can reach much deeper into the plasma than non-relativistic ray tracing suggests. Future coupling to JINTRAC allows for integrated EBCD scenario modelling.

To arrive at tolerable stationary divertor heat and particle loads, high divertor pressures of $p_{\mathrm{DIV}} \approx 10-20$ Pa and substantial Ar seeding are required [7], even with divertor geometries having enhanced exhaust capabilities. For SPP-001 SOLPS-ITER simulations with drifts and kinetic neutrals have been performed to assess the divertor performance [8]. First coupled core edge (COCONUT) calculations suggest a good shielding of the Ar in the SOL from the core and adequate pumping capabilities. Achieving a sufficient He pumping capability requires optimisation of the divertor dome structure as pumping is only possible from the low field side. Turning to MHD and control, resistive wall mode control has been characterised, including $n = 1,2$ modes. A novel control scheme for vertical control maintaining the DN configuration has been developed. In addition, novel trajectory optimisation techniques for regulating the α-power have been developed.

The change in device size (SPP-002) as well as the characteristics of the hKBM transport have significant impact on the FTOP. The larger minor radius leads to a lower Greenwald density $n_{\mathrm{GW}} = \frac{I_p}{\pi a^2}$ at a similar current. With fixed fusion power range keeping $f_{\mathrm{GW}} \approx 1$ and the auxiliary power constant at $P_{\mathrm{aux}} \approx 150$ MW leads to lower β_N, lower bootstrap current but higher ECCD efficiency and therefore to a similar plasma current $I_p \approx 21$ MA. The lower density also means that $B_t(R_{\mathrm{geo}}) = 3.0$ T could be slightly reduced. This reduction was supported by a scan in B_t and f_{GW} to find the most optimal solution for current drive. The lower poloidal field at the edge leads to a reduction in pedestal pressure and the core transport changes adopted from the NLGK simulations in the BgB assumption led to a less peaked density but more peaked temperature profile. Both changes are unfavourable for the absolute off axis EBCD efficiency. Further optimising of the high field side pellet launch for deeper penetration as well as exploring $f_{\mathrm{GW}} > 1$ helps to partially recover the advantages of EBCD. The ramp-up and ramp-down trajectories have been adapted to the new size and have been further optimised using JETTO as well as a workflow using the control-oriented transport solver RAPTOR. The lower β_N of the larger device does not necessarily lead to a more favourable RWM stability. Work is ongoing to understand the sensitivity of the no-wall ideal MHD limit on the plasma profiles in STEP. The equilibrium provided by the new poloidal field coil set has been further optimised for core shaping and divertor performance.

There is an ongoing programme to improve the plasma solution with respect to controllability, reduction of the disruption risk and current drive power. Studies on the impact of plasma elongation and aspect ratio have been performed for SPP-002 showing a lower elongation and a larger aspect ratio require more challenging plasma assumptions. Questioning the lower limit of $P_{\mathrm{fus}} \geq 1.5$ GW arising from an initially too simplified balance of power calculation in the system code, FTOPs with lower radiation fraction and lower fusion power are being explored. Initial scans show that a solution with lower plasma current may be possible at constant $P_{\mathrm{net,el}} \approx 100$ MW but may require slightly more heat exhaust capability of the divertor or a higher confinement assumption. Reducing the plasma current has major benefits for the disruption risk and the resulting runaway electron (RE) beam. Shattered pellet injection alone may not be sufficient to mitigate the RE beam and 3D techniques are being explored to dissipate the $I_{\mathrm{RE}} \approx 10$ MA beam.

In conclusion, the plasma scenario and control work continue to reduce the uncertainty of the plasma solution for STEP. Tools and workflows have been developed to re-evaluate efficiently any design changes, and these are progressively being further optimised to increase modelling fidelity. For example, core transport solutions can now be based on predictive modelling though are still missing important features, such as fusion α-particles and impurities. Furthermore, the plasma design is strongly integrated with the engineering effort – a key aspect of power plant design.

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