Challenges of the Plasma Scenario for the Spherical Tokamak for Energy Production

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The Spherical Tokamak for Energy Production (STEP) is a fast-paced UK national programme that aims to construct the first prototype tritium self-sufficient magnetic confinement fusion power plant prototype (SPP) by 2040 delivering $P_{el} \geq 100$ MW to the grid. The spherical tokamak (ST) concept has great potential to provide a cost-effective steady state solution due to the combination of high natural elongation, $\kappa\sim3$, high normalised β, $\beta_N\sim4-5$, moderate toroidal field, $B_t(R_{geo}) = 3.2\,T$ and low aspect ratio $A = 1.8$ that maximises $P_{fus} \propto \frac{1}{A}(\kappa\beta_N B_t)^4$ in a compact design without inboard breeding. After a rigorous down selection of various concepts a major radius $R_{geo} \approx 3.6$ m, plasma current $I_p \approx 20-25\,MA$, auxiliary power $P_{aux} \approx 50-150$ MW delivering $P_{fus} \approx 1.5-\ 1.8$ GW was chosen as initial design basis. Central to the SPP design are the plasma scenarios. Four fully non-inductive (NI) burning plasma flat-top operating points (FTOPs) with $9 < Q = P_{fus}/P_{aux} < 30$ have been developed using the JETTO transport code as assumption integrator and possible NI pathways to and from the FTOP have been identified. A small solenoid is kept in the design to mitigate the risks of plasma formation. Simple semi-predictive modelling helps to understand the sensitivity on the different assumptions. The required β_N to achieve the required Q is slightly above the no-wall limit for $n = 1,2$ but well below the ideal wall limit if $q_{\min} > 2.3$. Therefore, both current profile and resistive wall mode control is needed. MARS-F calculations show that with 8 in-vessel mid-plane coils RWM control should be feasible. The assumptions on the required confinement for a NI scenario depend sensitively on the current drive efficiency. The parameter space in density $n_e \lesssim 2\cdot10^{20}\mathrm{m}^{-3}$ and B_t is chosen to enable the use of electron Bernstein wave heating and current drive (EBCD) whilst allowing on-axis current drive with electron cyclotron waves (ECCD). The launch geometry has been chosen to maximise current drive efficiency and scenario points with pure ECCD and ECCD + EBCD have been developed. Fast codes to assess EBCD including relativistic effects are being developed to aid the integrated scenario design. Predictive capability for the confinement in STEP is missing as the core transport is dominated by electromagnetic (EM) turbulence arising from hybrid kinetic ballooning (KBM) with subdominant micro tearing (MTM) modes. for which fast models are not existing. Non-linear local gyrokinetic (GK) simulations show that the hybrid-KBM transport, driven by β_e and needing a perturbation in $B_\parallel$, is stabilised by β_e' and flow shear. Whilst already diamagnetic flow shear levels $\gamma_{dia}\sim0.06\ c_s/a$ could be sufficient to strongly reduce the transport to the required values (Figure 1), STEP scenarios have, apart from the loss of α-particles due to 3D fields, no external momentum source. To assess the impact of the diamagnetic flow higher order GK modelling is needed. Heat and particle load during disruptions and edge localised modes (ELMs) are key concerns for the integrity of plasma facing components. The estimated ELM energy fluences of $0.2\,\mathrm{MJ/m^2} \leq \varepsilon_\perp \leq 0.3\,\mathrm{MJ/m^2}$ are lower than the melt limit of W monoblocs ($\varepsilon_\perp \approx 0.5\,\mathrm{MJ/m^2}$), but well above limits for cracking. Scenarios with no or small ELMs with fully detached divertor legs are required to reduce the damage and erosion of plasma facing components at the targets if conventional target materials are used. To mitigate the risk of ELMs active suppression using $n = 3$ 3D magnetic perturbation fields are considered in the design with both in-vessel resistive and ex-vessel super conducting coils. To aid the detachment at minimised core radiation fraction $f_{rad}\sim70\%$ a double null (DN) magnetic configuration with divertor legs that maximise connection length and flux expansion is chosen. SOLPS-ITER calculations with and without drifts have been used to scope the operational boundaries during flat-top and ramp-up. He pumping and the Ar concentration at the separatrix are key design drivers for the divertor geometry. To shield the inner divertor from excessive power loads and maintain H-mode operation an extreme accuracy of the vertical control is required and passive structures close to the plasma as well as active in-vessel coils are crucial. Unmitigated disruption loads are unsustainable in any rector grade tokamak. For STEP disruption mitigation using shattered pellet injection (SPI) will likely reduce the heat, particle, and EM loads during the thermal and current quench phases but DREAM modelling suggests that, as on ITER, the avalanche process will generate a large runaway electron (RE) beam current $I_\mathrm{RE} \gtrsim 12$ MA. Multiple SPI injections and likely other methods are needed to reach a benign termination regime where MHD aids in the dissipation of the RE beam. This contribution will give an overview of the STEP baseline scenario design and discuss the key challenges that remain. Significant progress has been made in reducing and assessing the plasma and control uncertainties for a STEP prototype power plant.

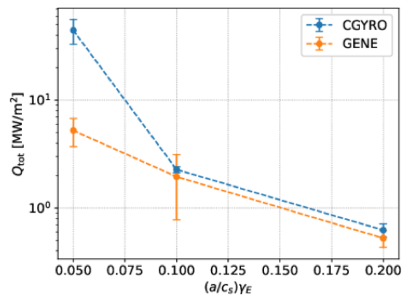

Figure 1 Flow shear dependence of hybrid KBM transport.