TOKAMAK ENERGY'S HIGH TEMPERATURE SUPERCONDUCTING MAGNET SPHERICAL TOKAMAK FUSION PILOT PLANT CONCEPT

Developed under the U.S. Department of Energy Milestone-Based Fusion Development Program

[1]S.A.M McNAMARA, J. ASTBURY, N. LOPEZ, X. ZHANG, AND THE TOKAMAK ENERGY FPP TEAM

[1]Tokamak Energy Ltd, Oxfordshire, United Kingdom

Email: steven.mcnamara@tokamakenergy.com

Tokamak Energy is one of the private companies selected for the U.S. Department of Energy's Milestone-Based Fusion Development Program, which is supporting the fusion industry in advancing fusion towards technical and commercial viability. The Program has an overall duration of five years and tasks awardees with producing a preliminary design for a fusion pilot plant (FPP) that can produce net electric power ≥50 MWe, be built with an overnight capital cost of ≤6 Bn USD (in 2022 $) and begin operations in 2034. During the first 18-month period of this five year program, working with collaborators from U.S National Laboratories, Universities and companies, Tokamak Energy will deliver a pre-conceptual design for a FPP based on the high-field spherical tokamak with high temperature superconducting magnets, and a set of associated technology development roadmaps. Spherical tokamaks offer an attractive route to commercial fusion due to their enhanced stability [1] and favourable transport and confinement properties [2]. Combined with HTS magnets, STs offer a route to more compact and potentially lower cost fusion power plants.

At the time of writing, an attractive baseline design space has been identified with the following high-level machine and plasma parameters: major radius R_{geo}=4.25 m, aspect ratio A=2, toroidal field B_T=4.5 T, plasma current I_P=13-16 MA, fusion power P_{fus}=800-950 MW, and auxiliary heating power up to P_{aux}=140 MW. Recognising uncertainties in the expected plasma energy confinement a series of design points have been developed with radiation corrected confinement enhancement factors in the range H_{ITER98}=1.2-1.6 with Ohmic plasma current drive fractions ranging from 20% to 0% and net electric power outputs between $P_{elec,net}$=70-110 MWe. The major plant technology choices aim to balance performance, plant integration and technology readiness. Design solutions currently being explored include: a full HTS magnet set (toroidal, poloidal and solenoid coils), monolithic toroidal field (TF) coils with a target TF full power life under irradiation damage of 5 years, and a significant solenoid capable of producing 45 Vs of inductive flux; a liquid, slow flowing natural lithium breeding blanket with He as the primary coolant, targeting a tritium breeding ratio of ≥1.1; a baseline plasma exhaust solution using tungsten plasma facing components, with an advanced liquid lithium concept also being developed; and a grade tungsten carbine and boron carbide centre column shield. The contribution will present the latest progress made towards the pre-conceptual design, with a focus on the plasma operating point and scenario.

A workflow for assessing and down-selecting design concepts has been developed and starts with the identification of promising design points using an in-house whole plant systems code, PyTok, that scans over a wide range of potential device parameters and allows the sensitivity of input assumptions and models to be evaluated. PyTok includes simplified models for all of the major plant systems, parametric CAD generation for cost modelling and neutronics assessments, large parameter space optimisation and sensitivity studies, and free-boundary plasma equilibrium generation. Promising design points are then taken forward for further assessment using a series of integrated physics and engineering workflows with increasing fidelity. For the plasma, the flat top operating point is developed using a 1.5D transport and equilibrium code to integrate various simplified models and produce plasma equilibrium and radial profiles that can be used for further assessment. The plasma kinetic profiles are either specified or estimated using a Bohm/gyro-Bohm analytic transport model [3], and are scaled to match the target fusion power and Greenwald

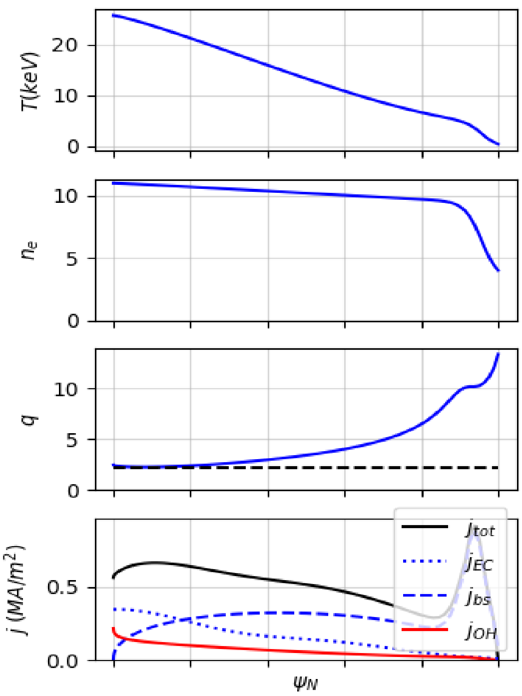

Figure 1 Profiles for the H_{ITER98}=1.4 operating point.

density fraction. Alternatively, reduced transport models, such as TGLF, are used to estimate the radial transport and corresponding kinetic profiles without scaling to the target fusion power. For the flat top operating points, electron cyclotron heating and current drive is assumed and the deposition profile is prescribed and an empirical normalised current drive efficiency is used. The current drive profile is tailored to optimise the safety factor profile to be mostly monotonic and maintain q_{min}>2.2 to avoid neo-classical tearing modes and infernal modes. For the pedestal, simplified expressions for the height and width are used [4, 5, 6]. An example set of radial profiles for the H_{ITER98}=1.4 design point is shown in Figure 1. An initial definition of the plasma operating scenario is developed using METIS, a fast integrated tokamak modelling tool for scenario design [7]. This allows candidate scenarios to be explored and provides an initial estimation of the flux required to reach and sustain the target plasma current and the contributions from the solenoid and poloidal field coils. The flat top operating points and scenarios are then further assessed using higher fidelity modelling including, integrated core-edge modelling, MHD and energetic particle stability assessments, heating and current drive (H&CD) optimisation, turbulence and transport analysis, pedestal modelling, scrape-off layer and divertor modelling, free-boundary equilibrium and optimisation of coil placements. The general approach aims to increase the fidelity of modelling in a stepwise fashion with each step feeding information back to the preceding ones, whilst also looking to increase confidence in design points or exclude them from further consideration.

An initial design of the ECCD system for ramp-up and flat-top has been developed using ray-tracing and full wave modelling codes. For the flat top operating points, an acceptable normalised current drive efficiency of 0.3 can be achieved across the entire minor radius using O-mode polarised waves launched from the low field side with frequencies in the range 140-200 GHz. A computational efficient physics-based optimisation has also been developed as a means to quickly estimate the optimum launcher configuration and corresponding normalised current drive efficiency [8]. To manage the plasma exhaust in a relatively compact device a double null (DN) configuration is favourable to split the power load between the upper and lower divertors and reduce the high field side power loads. For the solid PFC baseline, a detached long-leg configuration with argon and possibly neon edge and divertor seeding is currently favoured and the possibility of using an x-point radiator is being explored. For the liquid lithium concept early work on making a concept down-selection is on-going. MHD stability analysis using the KINX ideal MHD code has identified a complicated interaction between the assumed pedestal height and width, no-wall and ideal-wall normalised beta, β_N, limits, vertical stability and proximity to a connected double null configuration. The no-wall and ideal-wall β_N limits are found to reduce with increasing pedestal height due to an increase in the edge current density. Increasing pedestal height also destabilises a n=0 mode localised near the x-points. This mode can be stabilised with conducting structures close to the x-point or by moving away from a connected DN configuration, however this reduces the no-wall and ideal-wall β_N limits. In summary, significant progress is being made to advance the pre-conceptual design of Tokamak Energy’s fusion pilot plant as part of the U.S. DOE Milestone-Based Fusion Development Program.

ACKNOWLEDGEMENTS

This work was supported in part by the U.S. Department of Energy under contract number DE-SC002488. This work was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, nor any of their contractors, subcontractors or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or any third party’s use or the results of such use of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favouring by the United States Government or any agency thereof or its contractors or subcontractors. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof, its contractors or subcontractors.

REFERENCES

[1] Y-K.M. Peng and D. J. Strickler 1986 *Nucl. Fusion* **26** 769
[2] S. M. Kaye *et al* 2021 *Plasma Phys. Control. Fusion* **63** 123001
[3] E. Tholerus *et al* 2024 *Nucl. Fusion* **64** 106030
[4] S. Medvedev *et al* 2016 *Phys. Rep.* **42** 472–485
[5] J.F. Parisi *et al* 2024 *Nucl. Fusion* **64** 054002
[6] J.F. Parisi *et al* 2024 *Nucl. Fusion* **64** 086034
[7] J.F. Artaud *et al* 2018 *Nucl. Fusion* **58** 105001
[8] N. Lopez *et al* 2025 *arXiv* 2501.04619