CONFERENCE PRE-PRINT

PLASMA PREDICTION AND SIMULATION IN SUPPORT OF REACTOR DESIGN AND OPERATION AT TOKAMAK ENERGY

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Abstract

Plasma prediction and simulation activities at tokamak energy are centred around providing capabilities for the operation, modelling and design of both existing tokamaks and projected fusion pilot plants. This requires the development of validated plasma models based on tokamak data as well as targeted high-fidelity first-principles simulations. Once developed, plasma models are deployed in simulation frameworks such as SOPHIA for the support of plasma operation or in the staged approach of the design of a fusion pilot plant. Work is ongoing at Tokamak Energy in collaboration with our partners to bridge the knowledge gap between present day STs and large-scale spherical tokamak power plants. The development of a state-of-the-art plasma flight simulator (SOPHIA) has allowed TE to test ST40 discharges in advance and to train session leaders. SOPHIA has also been used to assess the controllability of plasma scenarios in future fusion pilot plants designs. A staged approach is adopted towards plasma modelling for fusion pilot plant design. The FUSE framework along with METIS-FreeGS provide higher fidelity description of plasma scenarios and allow the test of actuators and controllers for successful design of the plasma system / plant.

1. INTRODUCTION

Plasma modelling and simulation remains one of the most formidable challenges in the development of viable fusion power. Advances in artificial intelligence, high-performance computing, and experimental diagnostics are gradually improving predictive capabilities. However, achieving a truly reliable and scalable simulation framework that can guide the design of commercial fusion reactors is still an ongoing pursuit. Overcoming these challenges is crucial to making fusion energy a practical and sustainable power source for the future.

A fusion power plant is not just a plasma confinement device, it includes cryogenic cooling systems, breeding blankets for fuel generation, fuel cycle and heat extraction systems for power generation. Simulating the entire plant, from plasma dynamics to thermal and structural performance, requires coupling reliable plasma models with engineering simulations, further increasing complexity.

While fusion experiments provide valuable data, real-world measurements are limited by diagnostic constraints and other limitations. Many plasma properties, such as turbulence characteristics or magnetic reconnection events, are difficult to measure directly. This limits the ability to validate models, making it challenging to improve simulation accuracy. At Tokamak Energy we are carrying out a programme that aims at responding to the above challenges by designing dedicated experiments on ST40 and on other relevant tokamaks at our partner laboratories to develop reliable plasma models for spherical tokamak reactors in parallel to the development of simulation frameworks for the support of plasma operation and integrated plasma modelling for a fusion plant design.

2. ADVANCED PHYSICS MODELS FOR SPHERICAL TOKAMAK REACTOR DESIGN

The development of models for plasma operation in conventional tokamaks has benefitted from large amount of data and multiple years effort on validation against experiments. Conventional tokamaks have been operated with a broad range of collisionality, normalised ion Larmor radius and beta along with safety factor, plasma major radius and minor radius, toroidal magnetic field and plasma current. The challenge in modelling a spherical tokamak (ST) reactor is that present research on spherical tokamaks is limited to small size, short pulse devices operating at relatively low current and low toroidal magnetic field compared to those required for an ST reactor. There is very little available data to confidently validate models and project plasma operating scenarios from small STs to the reactor scale. A long pulse (with superconducting magnets) spherical tokamak of the size of JET, JT60SA or TFTR would be needed as stepping stone for the validation of models and codes. However, in the absence of such a device, ST40 [1] offers the best chance to date to extend the parameters space of ST operation. ST40 has operated with a toroidal magnetic field ranging from 0.7T to 2.1T and a plasma current up to 0.8MA. Moreover, in the last campaign, robust double null diverted scenarios have been developed allowing to access reliably H-mode confinement regime envisaged for ST reactor operation. Linear scaling of confinement with magnetic field has been observed in spherical tokamaks such as NSTX, START and GLOBUS-M, however the question remains on if such scaling can be used for projection to a fusion power plant operating at a field approximately ten times higher than that of small spherical tokamaks. The scaling of confinement with magnetic field at fixed plasma current in ST40 has been simulated with the TRIASSIC [2] framework which incorporates ASTRA [3] as a transport solver, CHEASE [4] for fixed boundary G-S equation solver, NUBEAM [5] for NBI injection modelling, NCLASS [6], and TGLF [7] for neoclassical and anomalous transport modelling respectively. The robustness of the models at different plasma aspect ratio has been successfully demonstrated by simulating similarity experiments (similar dimensionless plasma parameters) on ST40 and KSTAR (Fig. 1).

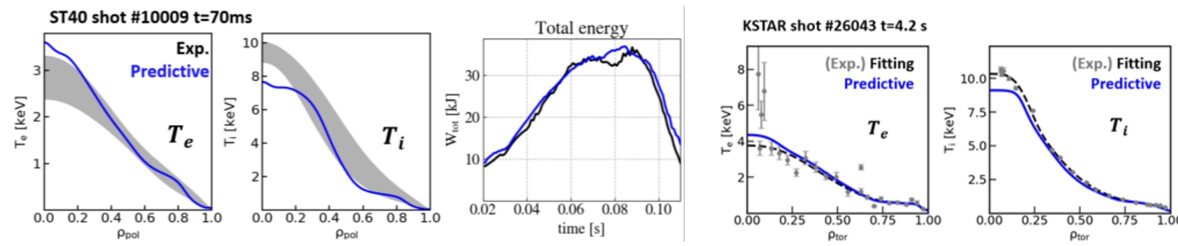

FIG. 1. Validation of the TRIASSIC model on similar plasmas in ST40 (left figures) and KSTAR (right figures)

The simulations showed good agreement between measured and TRIASSIC-TGLF predicted plasma performance in both ST40 hot ion mode [8] and KSTAR FIRE mode [9]. The direct comparison between ST40 and KSTAR plasmas aims at shedding light on the role of aspect ratio and edge safety factor on plasma performance.

A numerical experiment scanning the magnetic field from B=0.5T to B=2.2T has been performed with TRIASSIC based on the ST40 #13643 and #11515 plasma equilibrium and kinetic profiles. The simulations show a linear increase of confinement with magnetic field in the range 05T<B<1T and a saturation of confinement for B>1.6T where the slight increase of confinement with B follows the ITER scaling, Fig. 2. This suggests that the so-called ST confinement scaling (linear increase of confinement with B) is not universal and at reactor relevant field, confinement in STs may follow the conventional ITER scaling. Remarkably, the above result has been recently confirmed in ST40 dedicated experiments [1]. Analysis carried out also on Globus-M show that the saturation applies to a variety of mode of confinement (L-mode, H-mode and hot ion mode) [10]

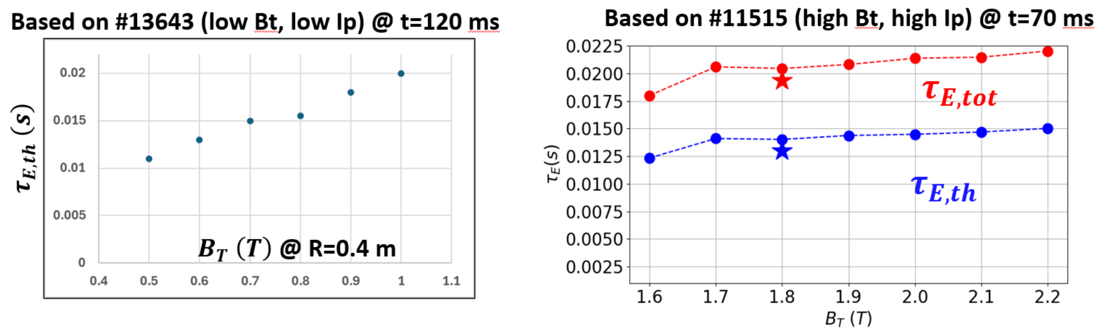

FIG. 2. Scaling of confinement time vs toroidal magnetic field with ST40 plasma parameters as simulated with TRIASSIC.

Gyrokinetic analysis of the B-scan plasmas with the nonlinear gyrokinetic codes CGYRO-DMD [11] and STELLA [12] is currently in progress to identify which potential transition in turbulence regime could be responsible for the change in scaling of confinement with magnetic field at fixed plasma current.

Understanding the access to different regimes of confinement in a spherical tokamak is essential to be able to model plasma operation in a future reactor. The transition from L-mode to H-mode on ST40 is very often moderated by I-phase, or limit-cycle oscillations. The energy confinement time during the I-phase is found to correspond to an H98-factor of 0.5-0.9, which increases to 1 and above 1 after the transition to the ELM-free H-mode. Improved confinement regimes – ELM-free H-modes – were achieved in the latest campaign with the H98(y,2)-factor of ~1. Typical plasma scenarios had plasma currents of 400 – 700 kA, toroidal magnetic field on axis of 0.4 – 1.2 T, and total heating power provided by either one or two neutral beams, with the total heating power, including ohmic power, up to 2 MW. Energy confinement times in ELM-free H-modes plotted against the ITER 98y,2 predictions are shown in Fig. 3.

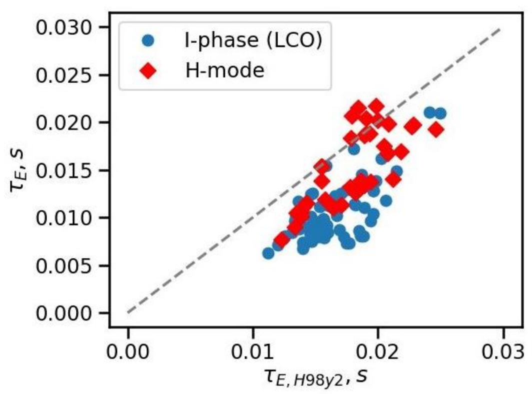

FIG. 3. Energy confinement time in I-phase (LCO) and H-mode plasmas versus ITER 98y2 predictions.

H-mode plasmas in ST40 are regularly ELM-free with wide pedestal as measured by Thomson Scattering. From pedestal stability analysis, the pressure gradient in the pedestal region is found to be below the ideal peeling-ballooning stability boundary. There is an ongoing effort at Tokamak Energy to develop a pedestal model based on the above observations (and those on other STs) to use for prediction for an ST fusion reactor.

Along with plasma confinement and performance other important variables for an ST reactor design are the heat loads on the first wall and the width of the plasma scrape off layer (SOL). ST40 exhibits narrow scrape off layer widths [1] that could be an issue when scaled to a reactor. Understanding the nature of the narrow SOL in ST40 is therefore important for modelling future devices. A fluid modelling of the edge plasma of ST40 #11890 pulse with “wide” heat flux was taken as a basic case. A numerical experiment of decreasing anomalous transport to achieve a narrow SOL has been performed. The SOLPS-ITER [13] code was used for the modelling. The appearance of the narrow peak is found to be linked to the observed increase in the absolute value of the floating potential near the strike point which can be explained by an increase in the radial currents flowing through the open magnetic flux surfaces. These currents are mostly induced by the diamagnetic currents, Fig. 4.

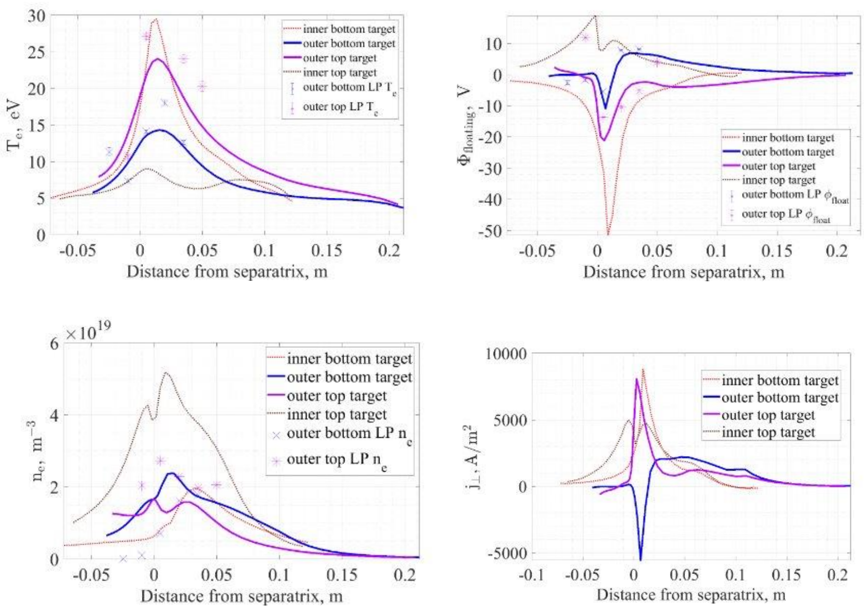

FIG. 4. Simulated electron Temperature, floating potential, electron density and perpendicular current at the target. Top/bottom Langmuir Probe measurements are plotted for model validation.

Full gyrokinetic simulations of the ST40 Scrape Off Layer / divertor performed with the code XGC1 [14] also show a narrow peak at the target, linked with electron dynamics, Fig. 5.

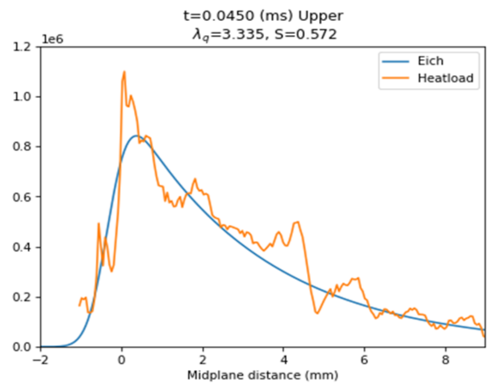

FIG. 5. XGC1 Simulation of the heat load at the outer upper divertor target.

3. PLASMA “FLIGHT” SIMULATOR IN SUPPORT OF TOKAMAK OPERATION

The main framework for simulating plasma operation at Tokamak Energy is SOPHIA [15], a versatile tokamak plasma simulator. It integrates models for the plasma, diagnostics, actuators and the plasma control system, providing a ‘control room’-grade experience to the operator. SOPHIA has been successfully deployed on the high-field spherical tokamak ST40 in the preparation of new scenarios (for both single and double-null diverted configurations), in the validation of experiments, the testing of new controllers and in the training of session leaders, Fig. 6. The validation of SOPHIA and its physics models on ST40 enables the development of plasma scenarios in the next generation of devices, such as a spherical tokamak Fusion Pilot Plant (FPP: Tokamak Energy, supported by the US DOE Milestone-Based Fusion Development Program), to be carried out with a higher level of confidence. SOPHIA can be used to test whether a proposed design point is feasible by assessing the controllability of the design under a set of assumed realistic models for the control system, device actuators, diagnostics and plasma dynamics. The integration of SOPHIA with existing pulse preparation and analysis tools makes it ideal for the training of new ST40 pilots and session leaders in charge of Tokamak operations.

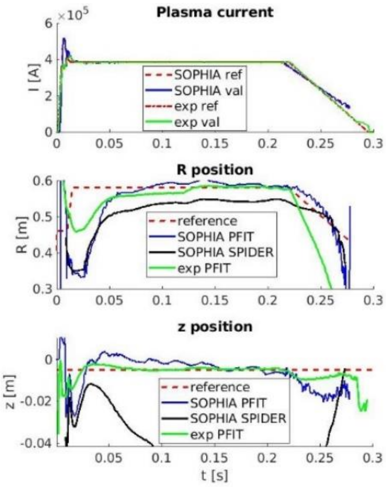

FIG. 6. Validation of a SOPHIA simulated pulse vs the corresponding ST40 pulse

Old pulses which had been executed with pulse preparation errors, as well as newly generated errors, can both be added to SOPHIA runs, enabling a pulse database to be constructed, from which AI based virtual pilots for future fusion plants could be trained.

By design, SOPHIA is a machine-independent tool that can be deployed on other tokamaks. Using the MATLAB/Simulink environment, in the present implementation SOPHIA integrates the transport code ASTRA, which is tightly coupled to the equilibrium code SPIDER [16], as well as models for diagnostics, actuators and the control system. Other transport and equilibrium codes are being coupled inside the SOPHIA framework such as the CRATOS-GS equilibrium code developed in collaboration with the University of Cambridge [17]. During recent campaigns on the spherical tokamak ST40, it has become clear that internal reconnection events (IREs) have caused the plasma ion temperature and density to be significantly reduced, but without the plasma current collapse characteristic of a disruption. Sawtooth instabilities can have similarly deleterious effects. Databases have been collected including the instability time and type, and the physics and control current values at plasma times leading up to each type of instability. Back propagation neural networks have been used to train the Time To Reconnection TTR as a function of selected variables V_i for around 60 shots suitable for analysis. Suitable shots with no instability can be given a TTR comparable to the shot time. The stored trained network for any IRE type is able to reproduce the time to reconnection for any chosen values of the variables. In particular the differentials $dTTR/dV_i$, although often negative, are sometimes positive indicating that the onset of the IRE might be delayed by such a change. The process works both with salient physics variables, such as internal induction l_i and poloidal field B_P, and control currents, such as solenoid and poloidal field currents. It is anticipated that the SOPHIA code will be able to select configurations capable of giving significant improvements in plasma stability.

4. INTEGRATED PLASMA SIMULATORS IN SUPPORT OF REACTOR DESIGN

Tokamak Energy Inc, the US subsidiary of the UK based private fusion company Tokamak Energy Ltd, was selected as one of eight awardees of the U.S. Department of Energy's Milestone-Based Fusion Development Program, which is supporting designing a fusion pilot plant (FPP) based on the spherical tokamak concept and high temperature superconducting magnets. The design of FPP is based on a staged approach as described in [22]. To aid in the design of its FPP concept and provide a more refined plasma description than that in Tokamak Energy's system code PyTok, Tokamak Energy is making use of a new open-source JULIA-based software suite built around the FUSE [18] framework, developed by General Atomics. The FUSE framework is used to rapidly produce self-consistent candidate designs of an FPP, bringing together physics and engineering codes in an automated manner, minimising the need for time consuming manual intervention by domain experts. The plasma models used in the spherical tokamak version of FUSE are those developed by General Atomics and adapted to the case of a spherical tokamak and/or trained on spherical tokamak data and simulations. In particular, as described in section 2 of this paper, the pedestal model needed to be adapted to the case of large width and stable pressure gradient to peeling ballooning modes. A reduced scrape-off-layer model based on the Box model [19] has been implemented in the FUSE framework to enable core-edge integrated design studies. Also, work is underway in the development of a new divertor design capability. An optimisation tool, FORGE (FORGE Optimises Reactor Geometries to improve Exhaust), is being developed to enable a rapid and automated design of the magnetic geometry of the divertor in future reactors. Such a tool uses a simulated annealing approach to alter the magnetic geometry of the plasma in the divertor region, whilst leaving the core plasma unperturbed. In doing so, quantities related to ease of detachment, such as the connection length of the divertor leg, can be optimised for, without the need to re-run a sensitive non-linear Grad-Shafranov solver, Fig. 8.

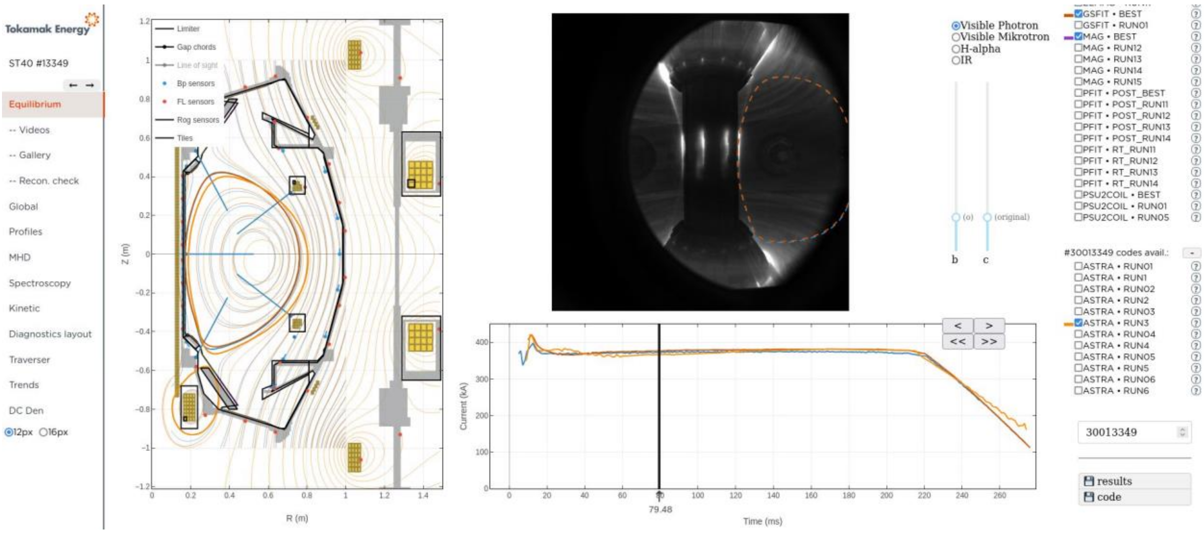

FIG. 7. SOPHIA simulation output plotted within the Physics Viewer software display along with experimental data. SOPHIA is fully embedded in the suite of codes used to operate ST40, including controllers offering to session leaders the same experience as running a real pulse.

Full plasma scenario modelling, such as current ramp up, requires the use of fast plasma simulators. One of the tools employed to model the current ramp-up of the FPP design point is an integrated workflow that couples the transport code METIS [20] with the Free-Boundary Equilibrium (FBE) Grad-Shafranov solver - FreeGS [21], within the context of magnet design. This integrated modelling approach is implemented through a Python interface that drives the FBE by prescribing from METIS the time-dependent plasma profiles of FF' and P', together with plasma shape constraints and the value of the poloidal flux ψ at the last closed flux surface (LCFS). In this way, the coupling ensures a self-consistent computation of plasma current density and magnetic field distributions, conserving the temporal evolution of ψ gradients and thus the flux consumption throughout the ramp-up phase. An example of the METIS simulation and coupled FreeGS result is shown in Fig. 9-12.

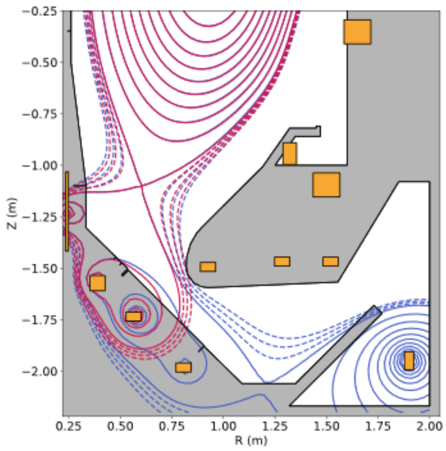

FIG. 8. Output of the FORGE optimization tool.

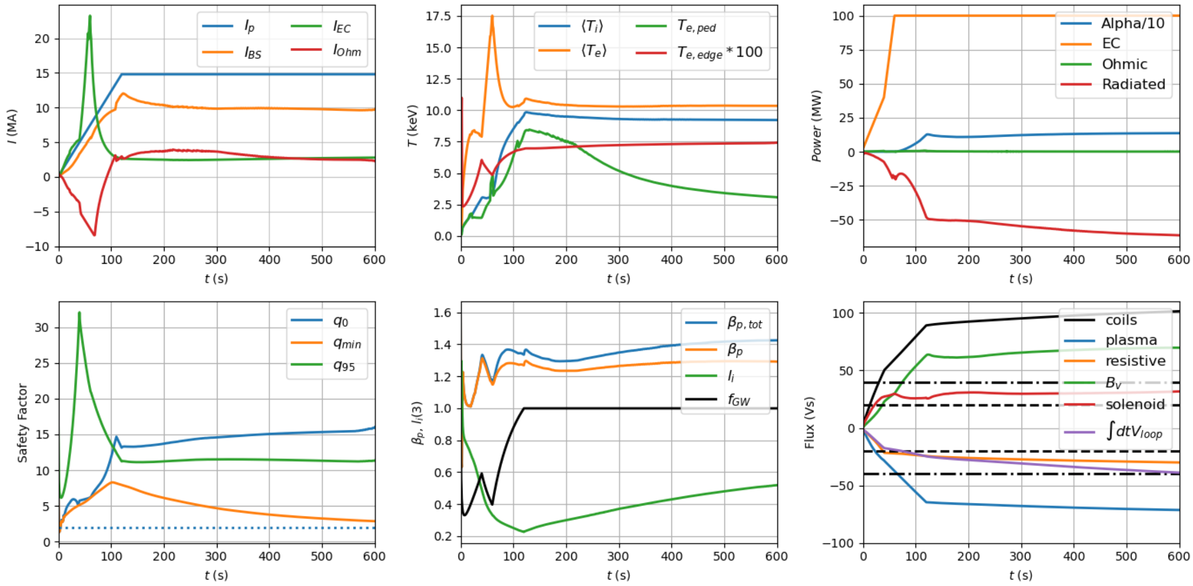

FIG. 9. Time traces of simulated main plasma parameters for a ST500 FPP operation scenario.

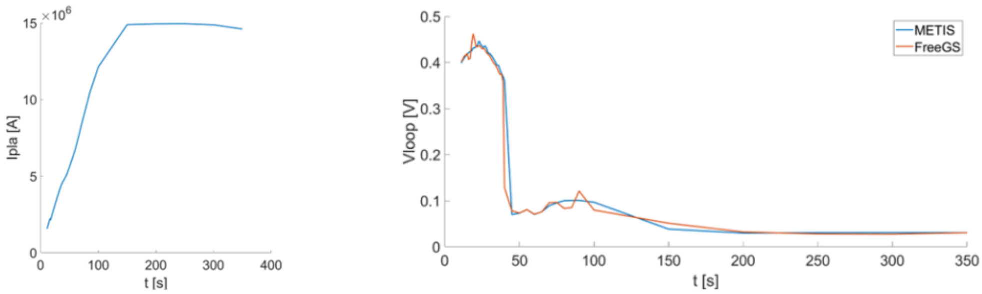

FIG. 10. FBE Plasma current and METIS-FreeGS Vloop comparison.

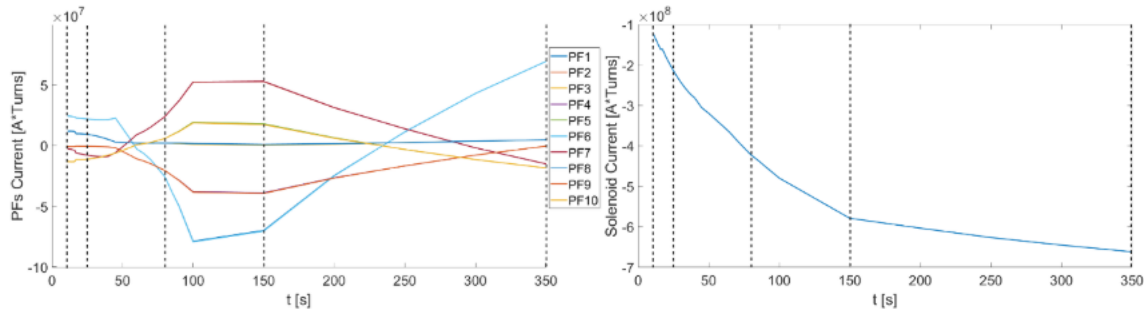

FIG. 11. FreeGS PFs and Solenoid currents waveforms.

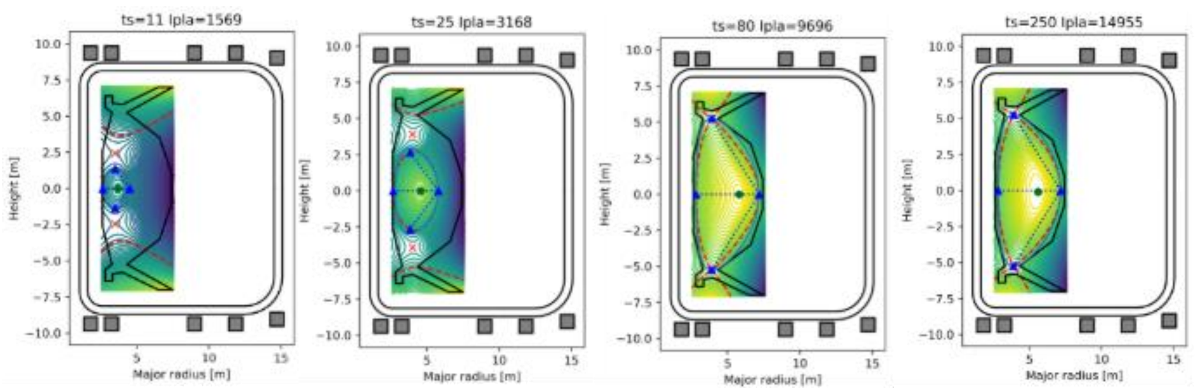

FIG. 12. Snapshot equilibrium during the current ramp up.

The METIS-FreeGS tool does not include modelling of the SOL / Divertor, however post-processing analysis with SOLPS-ITER allows to assess the heat loads on the first wall and to prescribe mitigating solutions such as impurity seeding. Seeded impurities can be included in the METIS-FreeGS scenario simulation and can also be prescribed in the FUSE analysis.

Plasma stability is calculated in the post-processing phase using MHD stability codes such as KINX. The controllability of the scenario is then tested with SOPHIA to provide a simulation of the full plasma operation.

Plasma disruptions pose significant risks to high-performance plasma operations, threatening both machine integrity and operational availability. Consequently, the design of future fusion devices aims to minimise the causes of such events as much as possible. Therefore, evaluating and characterising the effects of plasma disruptions is crucial in the design of a future fusion energy reactor. To support the conceptual design phase of a Spherical Tokamak Fusion Pilot Plant (ST-FPP) project, developed by Tokamak Energy, we have developed high fidelity MHD codes in collaboration with the University of Cambridge and we have been carrying out disruption simulations following the electromagnetic modelling approach using a 2D axisymmetric evolutionary equilibrium code applied to different conceptual design stages of a reactor scale machine, Fig. 13.

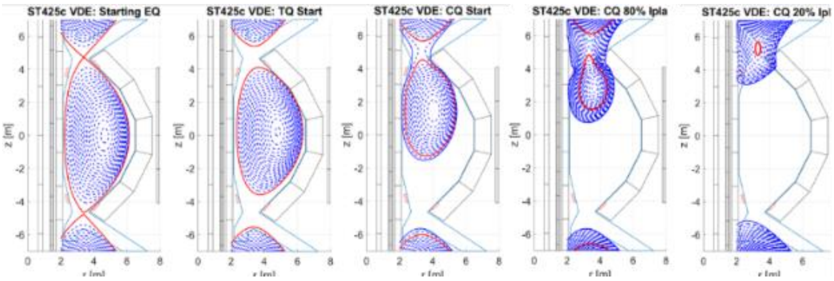

FIG. 13. UVDE ST425 Plasma Evolution and plasma-wall collision areas.

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