Predictive Multi-Physics Integrated Modelling of Tokamak Scenarios using the ITER Integrated Modelling and Analysis Suite (IMAS) in support of ITER Exploitation

M. Romanelli[1,*], P. Strand[2], D. Coster[3], J. Ferreira[4], D. Yadikin[2], T. Jonnson[5], E. Fransson[2], F. Eriksson[2,1], M. Poradzinsky[12], M. S. Park[6], S. J. Park[6], Y. S. Na[6], M. Honda[7], O. Meneghini[8], S. P. Smith[8], T. Slendebroek[1,11], B. A. Grierson[13], H-T Kim[1], Ph. Huynh[10], M. Schneider[9], S.D. Pinches[9] and the EUROfusion-IM Team[a]

[1] *Culham Centre for Fusion Energy, Culham Science Centre, Abingdon OX14 3DB, UK*
[2] *SEE, Chalmers University of Technology Gothenburg, Sweden*
[3] *Max-Planck Institute for Plasma Physics, Garching, Germany*
[4] *Instituto de Plasmas e Fusão Nuclear, Instituto Superior Técnico, 1049-001 Lisboa, Portugal*
[5] *Royal Institute of Technology, Stockholm, Sweden*
[6] *Department of Nuclear Engineering, Seoul National University, 1 Gwanak-no, Gwanak-Gu, Seoul, 08826, Republic of Korea*
[7] *QST, Mukouyama, Naka, Ibaraki, 311-0193 Japan*
[8] *General Atomics, San Diego, USA*
[9] *ITER Organization, Route de Vinon sur Verdon, 13067 St Paul Lez Durance, France*
[10] *CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France*
[11] *Oak Ridge Institute for Science Education, Oak Ridge, Tennessee 37830, USA*
[12] *Institute of Plasma Physics and Laser Microfusion, Hery str. 23, 01-497 Warsaw, Poland*
[13] *Princeton Plasma Physics Laboratory, Princeton University, Princeton, NJ 08543,USA*
[a] *http://euro-fusionscipub.org/eu-im*
**Tokamak Energy Ltd, 173 Brook Drive, Milton Park, Oxfordshire, OX14 4SD, UK*

Main author's e-mail: Michele.Romanelli@tokamakenergy.co.uk

Introduction.

Preparation for the exploitation of future fusion experimental reactors requires the use of verified and validated models for the prediction of plasma response to actuators and of fusion performance. Several simulation codes have been developed by the fusion community for the modelling of plasma equilibrium, transport processes, MHD stability, heating and current drive and fusion processes. The integration of these codes into flexible, verified and validated workflows [1] has been the subject of the work carried out by the EUROfusion consortium which has culminated in the release and exploitation of the European Transport Simulator (ETS) [2]. EUROfusion is currently utilising five different Tokamaks JET, TCV, AUG, MAST-U and WEST to carry out its research plan. The analysis and modelling with ETS of data from the above set of different devices required a high degree of standardization and the development of a common data platform. The choice of EUROfusion is to fully embrace the ITER Integrated Modelling and Analysis Suite (IMAS) [3] based on the Interface Data Structure (IDS) for data standardization and code integration. A coordinated activity started in 2019 to develop tools, such as UDA plugins and the OMFIT [4] module IMASgo, for the mapping of experimental data in IMAS / IDS of all the EUROfusion Tokamaks along with a campaign for the validation of codes and models in ETS. The same tools have also been adopted at KSTAR and DIII-D. In this paper we will report on the results of the application of ETS to the exploration of different operational regimes in both EUROfusion Tokamaks and DIII-D, KSTAR, JT60-SA. First application to ITER modelling will also be shown.

Residual turbulent transport in ETBs.

One of the topics of the multi-machine study conducted with ETS on JET and Medium Size Tokamaks (MST) data has been the investigation of the residual turbulent transport in the ETB (external transport barrier) region of an H-mode plasma. Various tools have been developed for the mapping of experimental data into IDSs for the different EUROfusion tokamaks [5]. By using IMAS for the description of the heating systems of AUG, TCV and JET it was possible to run predictive simulations of multi-tokamaks on the same platform. The ETB model adopted in these simulations is one in which the transport in a region of prescribed width (from the ITPA scaling) is set up to be described by a constant conductivity, accounting for both anomalous and collisional transport. The level of conductivity in the ETB is such to reproduce the top of the pedestal temperature and gradient in the ETB region. For the core transport both BgB, EDWM and TGLF SAT1 have been used as transport models. The comparison between predicted profiles and measurements for an AUG and TCV pulses are shown in figures 1,2.

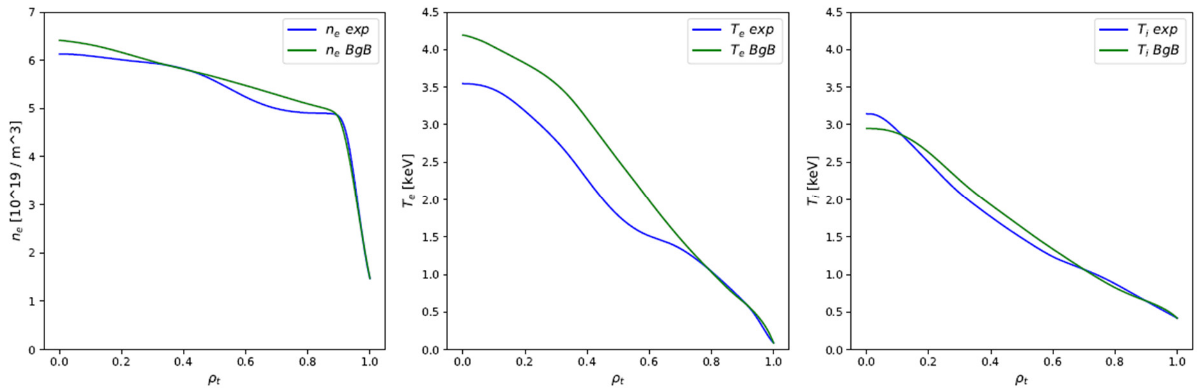

Figure 1: Comparison of AUG 36143 experimental vs predicted profiles at steady state using the BgB and the ETB transport models.

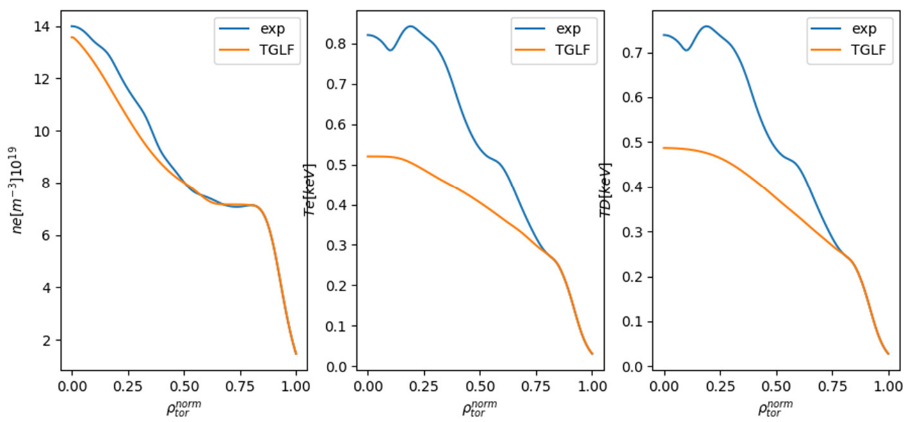

Figure 2: Predicted Te, Ti profiles for TCV 64770 (ECRH +NBI) at steady state, using TGLF SAT1 and the ETB transport model.

In figure 3 we report the value of the anomalous conductivity in the ETB (radially constant) versus the conductivity at the top of the pedestal (averaged over 10% of ρ_{tor} inside the pedestal top).

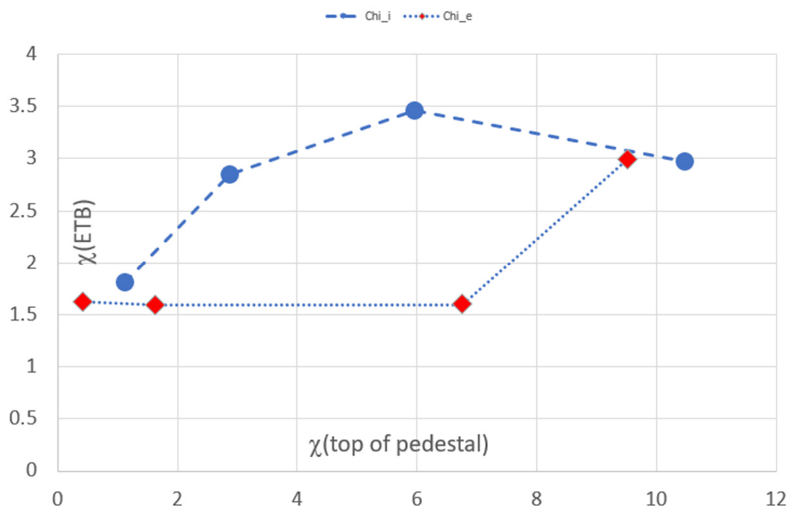

Figure 3: anomalous electron, ion conductivity in the ETB vs the conductivity at the top of the pedestal for a TCV, AUG and JET dataset.

No clear correlation is found in this small dataset between the conductivity in the two regions implying that the residual turbulent transport in the ETB is driven either by local pressure gradients or other mechanisms not related to core transport. The initial data set used above will be expanded to include all the predictive simulations done with ETS for the different tokamaks using the ETB transport model.

Simulation of DIII-D scenarios with variable NBI.

The DIII-D tokamak at General Atomics has a great flexibility in the angle and direction of NBI injection, making it very suited for the validation of beam deposition codes and for the study of the impact of different beam depositions on plasma performances. ETS simulations of DIII-D scenarios with variable NBI injection angles have been performed with the ASCOT, montecarlo code. The results of the power deposition and particle sources have been compared against TRANSP. TRANSP data, including the NBI configuration, have been mapped into IMAS using the IMASgo module in OMFIT, Figure 4. This allowed to configure the beam automatically without the need to modify the ASCOT configuration as it is usually the case when working outside IMAS. The impact of both rotation and heating on confinement has been studied with the model TGLF SAT1 in two different DIIID scenarios with on-axis and off-axis NBI injection. The results are reported in figure 5, 6.

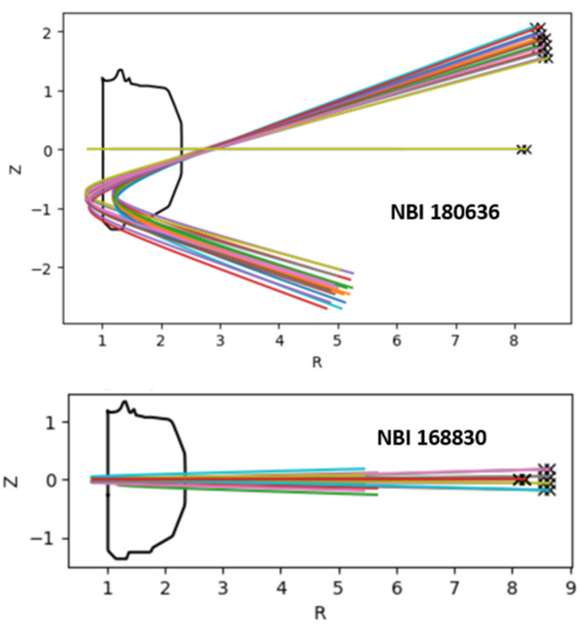

Figure 4: NBI IDS for two DIII-D pulses 168830 and 180636

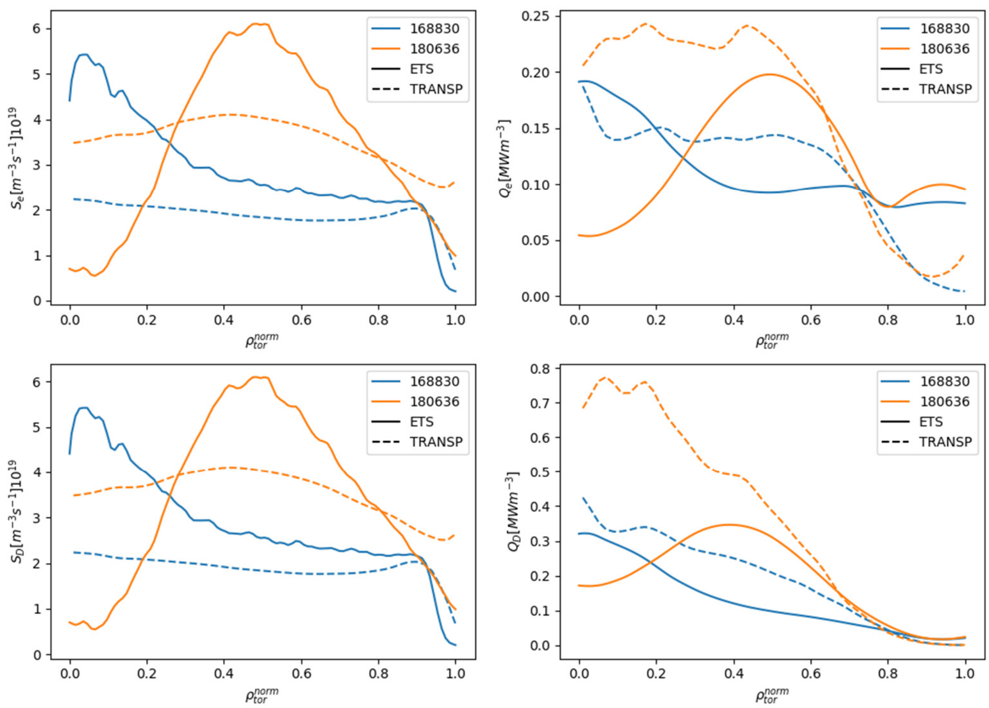

Figure 5: Top four charts: ASCOT / TRANSP power and particle deposition for the two different configurations, 168830 (off-axis, blue line) and 180636 (on axis, orange line).

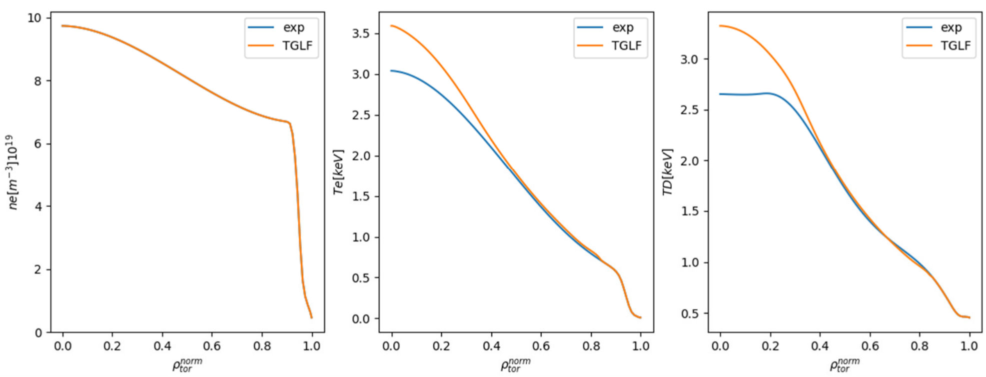

Figure 6: DIII-D 168830, ETS predicted profiles at steady state using TGLF vs profiles from TRANSP

Simulation of Long pulses in KSTAR.

A long pulse, steady-state plasma in KSTAR (pulse 18296) has been simulated to validate the equilibrium and current diffusion modules of ETS along with the impurity source and transport modules in non-transient conditions. The slow deterioration of confinement observed in KSTAR during long pulses has been addressed in the modelling by investigating the drift of the plasma equilibrium, by scanning ECRH power / resonance to control impurity accumulation and by controlling impurity influxes. Figure 7 shows the summary time traces for pulse 18296. Figure 9 shows the experimental profiles and their fit carried out with IMASgo in OMFIT. Figure 10 shows the result of ETS interpretative simulations with increasing values of Zeff. The result of the simulations show that the slow deterioration of confinement can be accounted for by an increased accumulation of carbon in the discharge. In turn the increased influx can be correlated to the slow drift of the outer strike point observed in the equilibrium reconstruction, Figure 8.

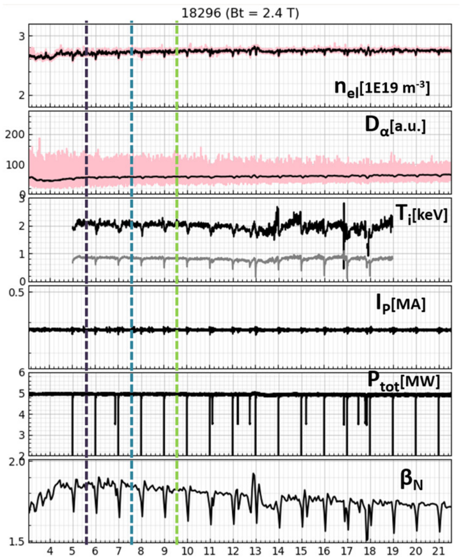

Figure 7: Summary time traces of KSTAR pulse 18296

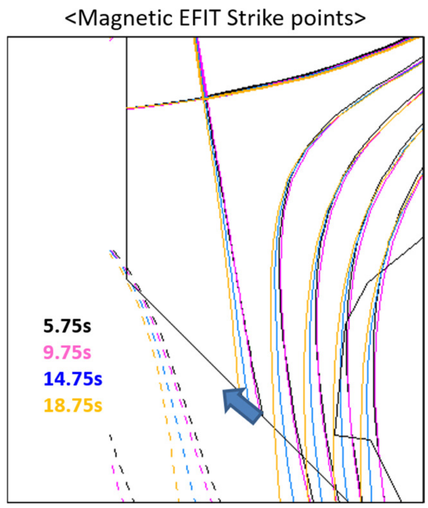

Figure 8: Drift of the outer strike point observed in the equilibrium reconstruction (#18296)

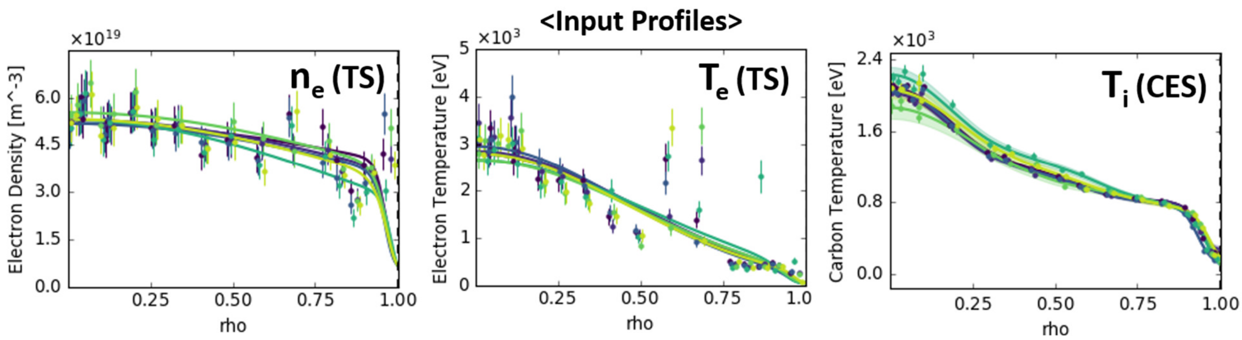

Figure 9: Radial profile fit of electron density and temperature (from Thomson Scattering measurements) and ion temperature (from CES) of pulse 18296 with IMASgo in OMFIT

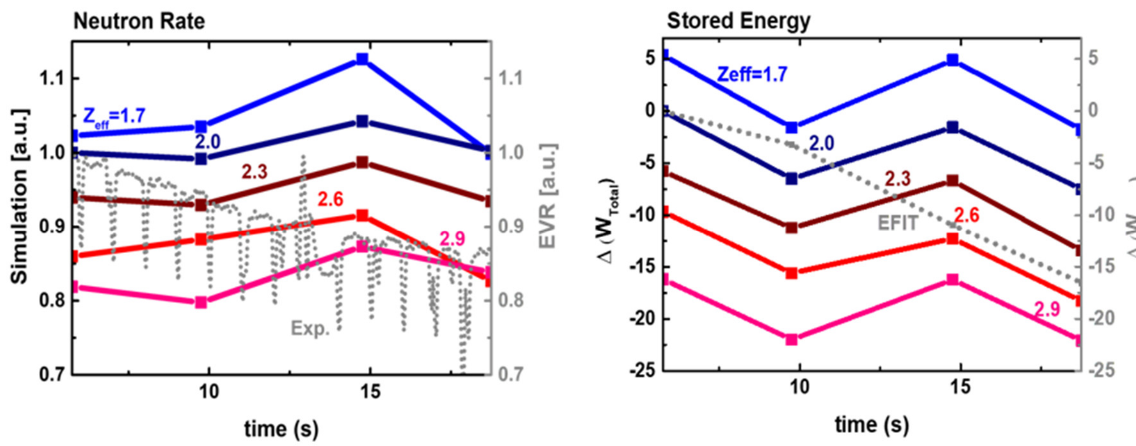

Figure 10: Neutron rate (left) and stored energy (right) calculated with ETS for different values of Zeff in pulse 18296 plotted against the measured values (dotted lines). n_C/n_e is assumed constant.

Simulation of JT-60SA high beta, non-inductive scenario.

The high-beta fully non-inductive advanced scenario of JT-60SA (I=5MA, B=2.5T, Paux=24MW, 17 MW of NBI (5 MW from NNBI) + 7MW of EC using the 4 EC launchers at 110 GHz and 138 GHz) has been simulated with the CDBM and TGLF transport models including self-consistent calculation of the NBI and ECRH power deposition.

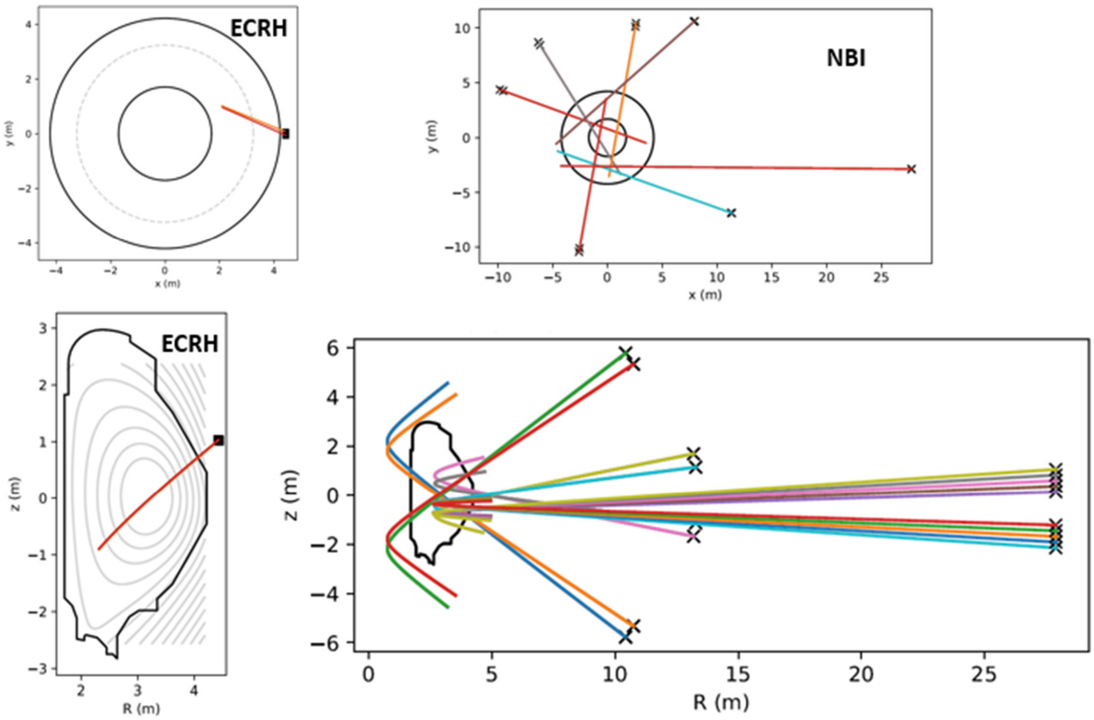

Fig. 11 Overview of the JT-60SA Auxiliary heating systems (ECRH, NBI) as implemented in ETS

The use of IMAS / IDS for the integration of modules (actors) in ETS makes it easy the testing within the framework of any physics code or module that has been adapted to IDS. The adoption in ETS of the CDBM model developed in Japan demonstrates this concept. Figure 12 shows the steady state profiles of the high beta scenario using the CDBM model. The simulation has been carried out with an internal boundary condition set up at rho toroidal corresponding to the top of the pedestal. Pedestal conditions are consistent with EPED.

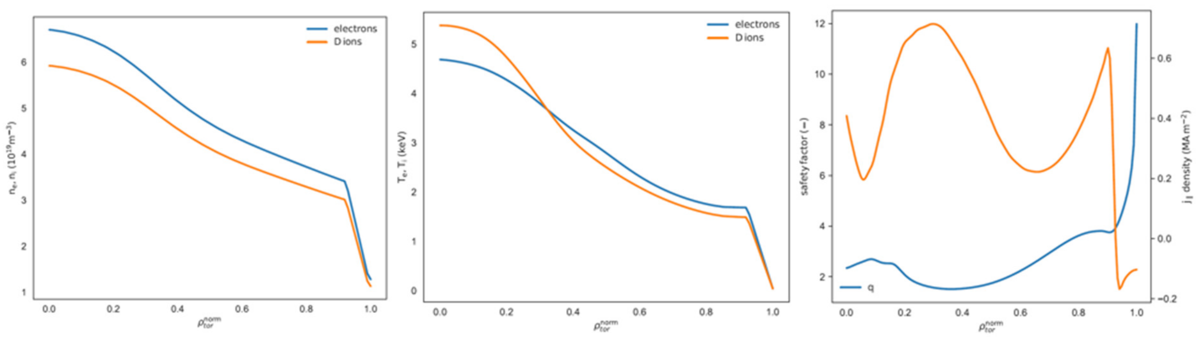

Fig. 12 Steady-state profiles for the JT-60SA high beta scenario. From left to right: electron and ion temperature, density, q, j profile. Transport model: CDBM and NCLASS.

Simulation of JET DT plasmas.

Preparation for an effective JET DT campaign requires extrapolation to DT of scenarios in deuterium in order to predict plasma confinement and fusion performance. An extensive validation of the heating and current drive / transport modules in ETS on JET deuterium discharges with both interpretative and predictive simulations has been carried out including the statistical benchmark with TRANSP on more than hundred JET discharges. Both baseline and hybrid scenario JET plasmas have been analysed with ETS using increasing NBI power. The power deposition profile has been evaluated with ASCOT which also provides the fast ion distribution function for the evaluation of the NBI contribution to the fusion power.

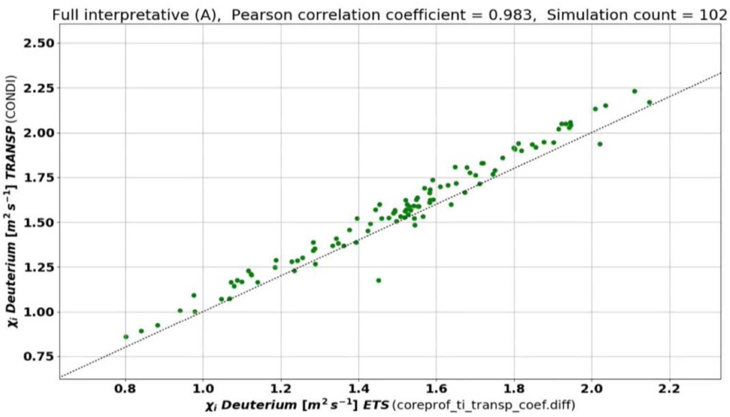

Fig. 13 Average heat diffusivity at r=0.5, ETS value against TRANSP

The neutron rate calculated with ETS/ASCOT compared to the experimental data for JET pulse 94442 is shown in Figure 3. The ETS results of the JET DT predictive modelling are summarised in Figure 4 and overlapped to the results of other codes (JINTRAC/CRONOS). The calculations were carried out with TGLF, with the saturation rule known as SAT1. These values for fusion power should, in principle, be taken as an upper bound. There are several reasons for this. First, we are using SAT1 in TGLF which gives higher temperature and we ignore charge-exchange losses on fast NBI particles due to neutral influx. A new saturation rule is available in TGLF SAT2 and the simulations will be repeated with the new transport model. Sensitivity scans on rotation should follow after the assessment of the saturation rules.

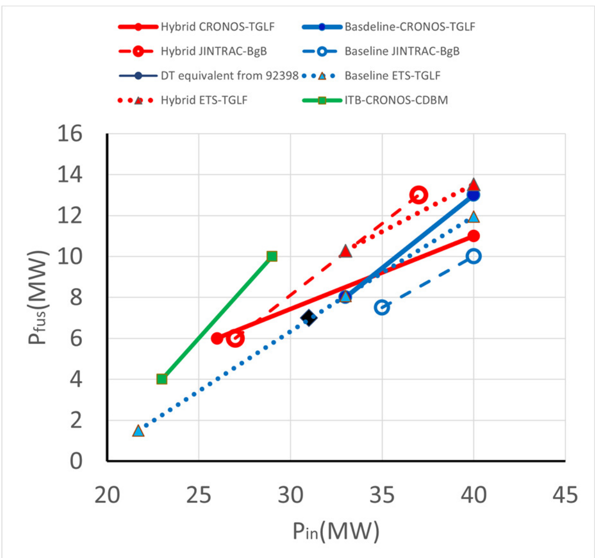

Figure 14: ETS projections for DT fusion power in JET baseline (blue dotted line) and hybrid (red dotted line) scenarios

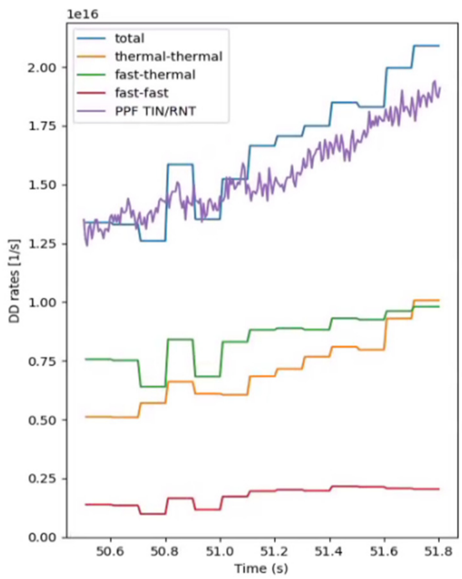

Figure 15: Neutron rate calculated with ETS/ASCOT for JET pulse 94442.

Simulation of ITER, 15MA, 5.3T scenario.

ETS has been used to simulate the ITER 15 MA, 5.3 T baseline scenario. This modelling was performed in the flat top with the aim to verify the IMAS implementation of the ETS by comparing with the former CPO based version. Within the study heating and current drive models, transport models and each of the transport equations were tested in detail. Below we show how a comparison of the electron cyclotron heating and current drive calculated with the GRAY code [6] with both the top and equatorial launchers active. Here the resonance position can be seen to be shifted by $\delta\rho_{tor} \sim 0.01$, due to slightly different treatment of the equilibrium near the magnetic axis.

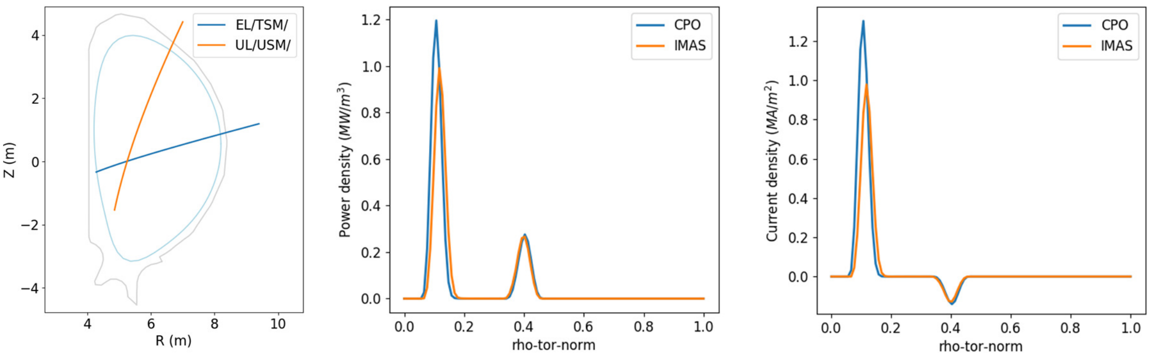

Figure 16: Left, the injection geometry with on-axis heating from the equatorial launcher and high field side heating from the top launcher. Middle, the EC heating profiles comparing results using CPOs and IDS. Right, the current due to ECCD.

Conclusion

The analysis presented in this paper show that IMAS is indeed an effective tool for facilitating the analysis of data across different Tokamaks and the exchange of physics modules. The use of IMAS allowed us to validate the models in ETS in various plasma conditions and operational regimes, building confidence in the predictions for ITER scenarios. The extensive use of IMAS in the fusion community will, in the longer term, provide a database of fusion data that can be exploited for theory studies, model validation, advanced Machine Learning and Artificial Intelligence applications in support of the exploitation of ITER and other fusion reactors.

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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