EUROfusion

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

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The integration of modelling 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 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. [5]

Residual Turbulent Transport in the ETB of ELMy H-mode plasmas

One of the focus of the above study using 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 and the development of a simplified transport model applicable to the plasma edge.

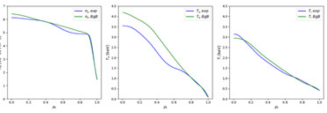

Figure 1: Comparison of AUG 36143 experimental vs predicted profiles at steady state using the BgB and the ETB transport models.

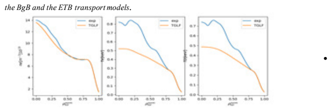

Figure 2: Predicted Te, Ti profiles for TCV 64770 (ECRH +NBI) at steady state, using TGLF SAT1 and the ETB transport model

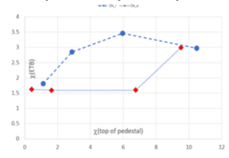

Figure 3: anomalous electron, ion conductivity in the ETB vs the conductivity at the top of the pedestal for a TCV, AUG and JET dataset.

- No correlation is found in the dataset between the conductivity in the two regions implying that the residual turbulent transport in the ETB is driven by the local pressure gradient. The initial data set used above will need to be expanded to include all the predictive simulations done with ETS for all the various tokamaks using the ETB transport model.

Confinement in KSTAR long pulse plasmas

Long pulse, steady-state plasmas in KSTAR have been simulated to validate the equilibrium and current diffusion modules 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.

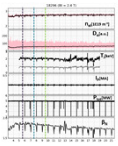

Figure 7: Summary time traces of KSTAR pulse 18296s

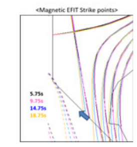

Figure 8: Drift of the outer strike point observed in the equilibrium reconstruction (#18296)

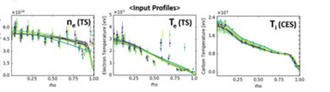

Figure 9: OMFIT radial fit of electron density and temperature from Thomson Scattering measurements and ion temperature from CES of pulse 18296

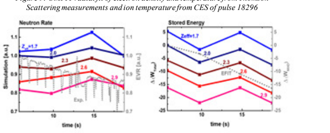

Figure 10: Neutron rate (left) and stored energy (right) calculated with ETS for different values of Zeff in pulse 18296 plotted against the measured values (dotted lines). n_c/n_e is assumed constant

Predictive simulations in preparation of the JET DT2 experiments and ITER 15 MA scenario

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.

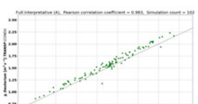

Fig. 13 Average heat diffusivity at r=0.5, ETS value against TRANSP

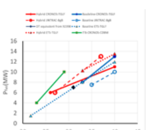

Figure 14: ETS projections for DT fusion power in JET baseline (blue dotted line) and hybrid (red dotted line) scenarios

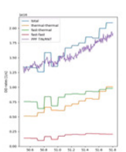

Figure 15: Neutron rate calculated with ETS/ASCOT for JET pulse 94442. The total rate (blue) is compared with the measurement from the neutron cameras

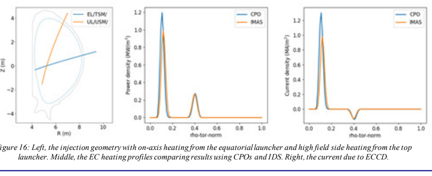

Figure 16: Left, the injection geometry with on-axis heating from the equatorial launcher and high field side heating from the top launcher. Middle, the EC heating profiles comparing results using CPOs and IDS. Right, the current due to ECCD.

ETS has been used to simulate the ITER 15 MA, 5.3 T baseline scenario

Impact of a variable NBI on DIIID plasma confinement and transport

Simulation of DIII-D scenarios with variable NBI injection angles have been used to validate the beam deposition codes in ETS against measurements of torque, ion temperature and neutron yield. 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.

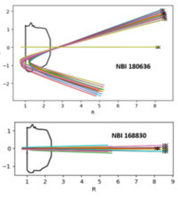

Figure 4: NBI IDS for two DIII-D pulses 168830 and 180636

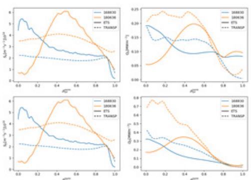

Figure 5: Top four charts: ASCOT / TRANSP power and particle deposition for the two different configurations, 168830 (off-axis, blue line) and 180636 (on axis, orange line).

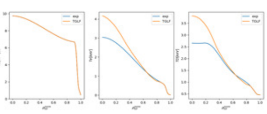

Figure 6: DIII-D 168830, ETS predicted profiles at steady state using TGLF vs profiles from TRANSP

High beta non inductive scenario of JT60SA

The high-beta fully non-inductive advanced scenario of JT-60SA has been simulated with the CDBM transport model including self-consistent calculation of the NBI and ECRH power deposition. 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 adapted to IDS. The adoption in ETS of the CDBM model developed in Japan demonstrates this concept.

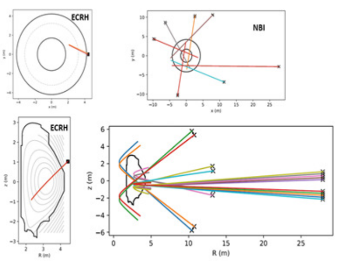

Fig. 11 Overview of the JT-60SA Auxiliary heating systems (ECRH, NBI) as implemented in ETS

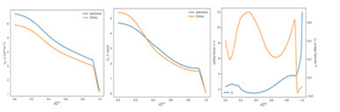

Fig. 12 Steady-state profiles for the JT-60SA high beta scenario. From left to right: electron and ion temperature, density, q, j profile. Transport model: CDBM and NCLASS.

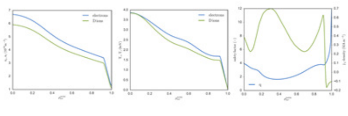

Fig. 12b Steady-state profiles for the JT-60SA high beta scenario. left to right: electron and ion density, electron and ion temperature, current density and q profile. Transport model: TGLF(SAT1) and NCLASS.

Conclusion

The results presented in this paper indicate 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 ITER exploitation.

[1] A Becoulet et al (2007), Computer physics communications 177 (1-2), 55-59

[2] P. Strand et al (2018), 27th IAEA FEC, TH/P6-14, IAEA CN-258

[2] F. Imbeaux et al (2015), Nuclear Fusion, Volume 55, Number 12

[4] O. Meneghini et al (2015), Nuclear Fusion, Volume 55, Number 8

[5] M.Romanelli et al (2020), Fusion Science and Technology, 76 894-900

[6] D. Farina, Fusion Sci. Technol. 52 (2007) 154

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GENERAL ATOMICS

JT-60SA Advanced Superconducting Tokamak

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