# TTF2023

## Sept. 12-15, Nancy, France

**Transport Task-Force EU-US joint meeting**

**Interactive booklet of Abstracts**

Hosted by

INSTITUT JEAN LAMOUR

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*Injection of low-Z powders: a novel actuator to control pedestal, heat exhaust and plasma wall interaction*

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Herschel Manuel manuel.herschel@ipp.mpg.de
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Cano Megas Pilar pcano@ipp.mpg.de
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Austin Max max.austin@utexas.edu
*Initial Transport Results of the DIII-D Negative Triangularity Campaign*

Stewart Samuel sdstewart3@wisc.edu
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Vanovac Branka vanovac@mit.edu
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Kudashev Ivan ivan.Kudashev@univ-amu.fr
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Locker Franz Ferdinand franz.locker@uibk.ac.at
*Development of full-f gyrofluid simulations for edge turbulence and magnetic reconnection*

### *Shaping: Spherical Tokamaks*

Casson Francis francis.casson@ukaea.uk
*Mitigating uncertainty in integrated scenario design for the STEP prototype powerplant*

Giacomin Maurizio maurizio.giacomin@york.ac.uk
*Turbulent transport in the core of high-* $\beta$ spherical tokamaks and predictions for STEP

Imada Koki koki.imada@york.ac.uk
*Pedestal stability analysis of MAST-U H-mode plasmas and impact of plasma shaping parameters*

Patel Bhavin bhavin.s.patel@ukaea.uk
*Initial results of gyrokinetic analysis of the core plasma in MAST Upgrade*

***Impurities***

Fajardo Daniel daniel.fajardo@ipp.mpg.de

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Gleiter Tabea tabea.gleiter@ipp.mpg.de

*Experimental impurity transport studies for the plasma edge in different confinement regimes at ASDEX Upgrade*

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Dudding Harry harry.dudding@ukaea.uk

*An algorithmic framework for developing saturation rules in reduced core transport models*

Marin Michele michele.marin@epfl.ch

*Integrated modelling of ohmic ramp-up at TCV*

***Core transport/beyond GK***

Raeth mario mario.raeth@ipp.mpg.de

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***Core transport/modulation***

Zimmermann Carl Friedrich Benedikt benedikt.zimmermann@ipp.mpg.de

*Experimental validation of momentum transport theory in the core of a tokamak plasma*

Slief Jelle j.h.slief@differ.nl

*Applying self-consistent electron heat transport and ECH deposition profile estimation in DIII-D*

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Falessi Matteo Valerio matteo.falessi@enea.it

*Advanced energetic particle transport models*

Di Siena Alessandro alessandro.di.siena@ipp.mpg.de

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Plank Ulrike Ulrike.Plank@ipp.mpg.de

*Experimental assessment of the role of the main ion species composition on the access into H-mode*

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*Experiments and gyrokinetic simulations of TCV plasmas with negative triangularity in view of DTT operations*

1-03 Sales De Oliveira Diego diego.oliveira@epfl.ch
*Progress in understanding the impact of the magnetic geometry on divertor turbulence*

1-04 Di Giannatale Giovanni giovanni.digiannatale@epfl.ch
*Progress towards the understanding of negative triangularity improvements with gyrokinetic simulations*

1-05 Volcokas Arnas arnas.volcokas@epfl.ch
*Ultra long turbulent eddies, magnetic topology, and the triggering of internal transport barriers*

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*Quasilinear Gyrokinetic Modeling of Reduced Transport in the Presence of High Impurity Content, Large Gradients, and Large Geometric Alpha*

1-07 Wilks Theresa twilks@psfc.mit.edu
*Limiting factors for achieving peeling-limited pedestals in present devices*

1-08 Pankin Alexei pankin@pppl.gov
*Predictive modeling of Super-H mode in DIII-D using the TRANSP integrated modeling code*

1-09 Balbin-Arias Julio jjbalbinarias@wm.edu
*Role of neutral particles on pedestal structure for H-mode experiments in DIII-D*

1-10 Mckee George george.mckee@wisc.edu
*Turbulence and Transport Dependence on* $\rho^*$ *& Isotope Mass in H-Mode Plasmas on DIII-D*

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*Integrated modeling of WEST long pulse L-mode discharges*

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1-14 Panico Olivier olivier.panico@cea.fr
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1-15 Artaud Jean-Francois jean-francois.artaud@cea.fr
*Maximizing the ion temperature in an electron heated plasma: from WEST towards larger devices*

1-16 Fonghetti Theo theo.fonghetti@cea.fr
*Modeling of WEST plasmas with reduced Lower-Hybrid model: interplay with transport and sensitivity analysis*

1-17 Gillot Camille difpradalier@yahoo.fr
*Fidelity of Model Reduction: Implications of Near Marginality Lessons learnt from (i) quasilinear, nonlinear gyrokinetic (ii) gradient- & (iii) flux-driven simulations*

1-18 Ivanov Plamen plamen.ivanov@physics.ox.ac.uk
*Structure formation in plasma turbulence with imposed flow shear*

1-19 Acton Georgia georgia.acton@merton.ox.ac.uk
*Full Flux Surface* $\delta f$*-Gyrokinetic code*

1-20 Kit Adam adam.kit@helsinki.fi
*Enabling online pedestal stability analysis with machine learning*

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*On the validity of reduced quasi-linear transport models in the current ramp-up phase of TCV plasmas*

2-02 Najlaoui Anass anass-ahmad.najlaoui@univ-amu.fr
*Can TGLF model Kinetic Ballooning Modes turbulence in the center of high-performance tokamak plasmas?*

2-03 Fuhr Guillaume guillaume.fuhr@univ-amu.fr
*Gyro-Kinetic DataBase project*

2-04 Lanzarone Matisse matisse.Lanzarone@univ-amu.fr
*Neural Network Surrogate for Acceleration of Gyrokinetic Codes to Compute Instability Growth Rates and Frequencies*

2-05 Cavalier Jordan cavalier@ipp.cas.cz
*Statistical analysis of the COMPASS SOL turbulence by mean of a fast-visible camera*

2-06 Casolari Andrea casolari@ipp.cas.cz
*Study of heat transport properties in COMPASS Upgrade scenarios*

2-07 Jaulmes Fabien jaulmes@ipp.cas.cz
*Scenarios for operation of COMPASS Upgrade and ITER at larger plasma current*

2-08 Macha Petr macha@ipp.cas.cz
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2-09 Snoep Garud g.snoep@differ.nl
*Validation of reduced-order turbulence modelling in the L-mode near-edge of the JET-ILW tokamak*

2-10 Ho Aaron a.ho@differ.nl
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2-11 Enters Yorick ye525@york.ac.uk
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2-12 Schuett Tobias M tobias.schuett@york.ac.uk
*Nonlinear energy transfer between drift-wave turbulence and zonal flows in spherical tokamak plasmas*

2-13 Thomas Steven steven.thomas@york.ac.uk
*First characterisation of L-mode ion-scale turbulence on MAST-Upgrade with beam emission spectroscopy*

2-14 Cziegler Istvan istvan.cziegler@york.ac.uk
*Recent progress in the coordinated experimental and computational effort on flow-turbulence coupling*

2-15 Tamura Naoki tamura.naoki@nifs.ac.jp
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2-16 Jhang Hogun hgjhang@kfe.re.kr
*A generalized gyro-averaging operator with magnetic field inhomogeneity and its implication*

2-17 Bray Elisabetta s292538@studenti.polito.it
*Studies on the effect of impurities emitted from a liquid metal divertor on the plasma core for the design of EU-DEMO*

2-18 Marchetto Chiara chiara.marchetto@isc.cnr.it
*Progress on interaction between NTM Island and heavy impurities in AUG*

2-19 Aucone Lorenzo l.aucone@campus.unimib.it
*Experiments and numerical modelling of negative triangularity ASDEX Upgrade plasmas in view of DTT scenarios*

2-20 Kraemer-Flecken Andreas a.kraemer-flecken@fz-juelich.de
*Observation of quasi coherent modes in W7-X and relation to tokamaks*

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3-01 Luda Di Cortemiglia Teobaldo tluda@ipp.mpg.de
*Multi-machine validation of IMEP and fusion performance predictions for ITER and DEMO*

3-02 Proll Josefine j.h.e.proll@tue.nl
*Beyond ion-temperature-gradient turbulence in stellarators*

3-03 Hallatschek Klaus klaus.hallatschek@ipp.mpg.de
*Landau collisions for fluid and gyrokinetic simulations*

3-04 Tardini Giovanni giovanni.tardini@ipp.mpg.de
*ASTRA modularity and IMASification for integrated modelling workflows*

3-05 Romba Thilo thilo.romba@ipp.mpg.de
*Configuration dependence of regimes with suppressed turbulent impurity transport in Wendelstein 7-X*

3-06 Kappatou Athina athina.kappatou@ipp.mpg.de
*Studies of confined energetic helium ions in ASDEX Upgrade plasmas*

3-07 Tala Tuomas tuomas.tala@vtt.fi
*Isotope Mass Scaling and Transport Comparison between JET Deuterium and Tritium L-mode Plasmas*

3-08 Salmi Antti antti97b@gmail.com
*Electron density response to strike-point sweeping on JET*

3-09 Kirjasuo Anu anu.kirjasuo@vtt.fi
*Isotope dependence of intrinsic torque in JET ohmic plasm*as

3-10 Aaro Jarvinen aaro.jarvinen@vtt.fi
*Exploring state representation learning algorithms for multi-machine generative modelling of fusion plasmas*

3-11 Sama Juvert Njeck juvert-njeck.sama@univ-lorraine.fr
*Ion temperature gradient mode mitigation by energetic particles, mediated by forced-driven zonal flows*

3-12 Moritz Jerome jerome.moritz@univ-lorraine.fr
*Ion transport through Radio-Frequency sheaths studied by Particle-In-Cell simulations*

3-13 Lo-Cascio Guillaume guillaume.lo-cascio@univ-lorraine.fr
*Impurity transport through two types of transport barrier in 5D gyrokinetic simulations*

3-14 Chouchene Sarah sarah.chouchene@univ-lorraine.fr
*Mutual interactions of turbulent transport in COMPASS tokamak characterized by means of ultra-fast camera and machine learning*

3-15 Vermare Laure laure.vermare@lpp.polytechnique.fr
*Flow and phase velocity of turbulence in magnetized fusion plasmas*

3-16 Rienacker Sascha sascha.rienacker@lpp.polytechnique.fr
*Gaining insight into E √ó B flow control by plasma current in different magnetic configurations*

3-17 Mantica Paola paola.mantica@istp.cnr.it
*Detection of alpha heating in JET-ILW DT plasmas by a study of the electron temperature response to ICRH modulation*

3-18 Agostini Matteo matteo.agostini@igi.cnr.it
*Characterization of the edge turbulence and electron profiles in TCV tokamak with the Thermal Helium Beam diagnostic*

3-19 Balasubramanian Pradeep Somu pradeep.balasubramanian@uibk.ac.at
*Effect of approximations to the polarization equation in full-f gyrofluid turbulence simulations*

3-20 Grander Fabian fabian.grander@uibk.ac.at
*Hysteresis in gyrofluid resistive drift-wave zonal-flow turbulence*

3-21 Lin Wenhao linwenhao@swip.ac.cn
*Gyrokinetic simulations on the triggering and self-sustaining of internal transport barrier in HL-2A tokamak plasmas*

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# Understanding of near-edge physics in L-mode, H-mode and ELM-free regimes progress in validation, core/edge integration, role of particle source

C. Bourdelle
[1]CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France

Understanding the region between the separatrix and the pedestal top is of paramount importance. The performance of this region impacts strongly the overall discharge performance in 2 ways: 1/ towards the core, due to the stiff core turbulent transport, the energy content and fusion power is strongly coupled to the pedestal top; 2/ while the separatrix conditions are strongly coupled to the heat exhaust requirements at the divertor.

Between the separatrix and the pedestal top, across ~5% of the minor radius, takes place a complex interplay between the neutral particle source, turbulent transport, radial electric field and MHD stability.

Compared to tokamak in operation today, in larger devices such as ITER and DEMO, the interplay between fueling, transport, ExB shearing and MHD will be modified, preventing us from relying on linear extrapolation [1,2]. Moreover, in future larger devices, it will be mandatory to simultaneously achieve detachment at the divertor and optimized core performances. The control of the various phases of the discharges will be also mandatory to maximize the plasma duration as well as its performances. Hence, understanding of these phases in today's devices is of interest: the current ramps (up and down); L mode, H mode (with and without ELMs) and the transition from one to the other; attached and detached conditions at the divertor and transition from one to the other.

In this context, recent progress on turbulent transport characterization in L mode [3] and in the H mode pedestal will be presented [4] as well as the turbulence interplay with the ExB shear when transiting from L to H mode [5,6]. The role of fueling with gas or pellet in today's device on core performances in L and H mode will be discussed [7]. More largely the role of neutral recycling on core performances on the seperatrix parameters will be touched on [8,9]. Recent work modeling the complex interplay between MHD and transport compared to experiments [10], including as well the neutral penetration in H mode [11] and L mode [12] will be reviewed.

Some open issues will be proposed towards filling the gap from today's tokamak towards larger devices. For example, on how to anticipate the impacts of a decreased ratio of ionization length with respect to the divertor leg length and the wall clearance distance; of a reduced pumping vs the plasma volume; of detachment on ExB shear and H mode access; of dominant pellet fueling, etc.

This overview talk does not aim at being exhaustive. In the spirit of the Transport Task Force workshop, it will rather serve as an introduction to a session combining more detailed orals and stimulating discussions towards future work.

[1] A.R. Polevoi *et al* 2015 *Nucl. Fusion* **55** 063019
[2] F. Palermo et al 2019 Nucl. Fusion 59 096010
[3] N. Bonanomi et al Phys. Plasmas 28, 052504 (2021)
[4] D.R. Hatch et al Phys. Plasmas 29, 062501 (2022)
[5] C. Bourdelle Nucl. Fusion **60** (2020) 102002
[6] T. Eich and P. Manz Nucl. Fusion **61** (2021) 086017
[7] C. Perez von Thun et al, 48$^{th}$ EPS conference on plasma physics, 2022
[8] J.D. Strachan 1994 *Nucl. Fusion* **34** 1017
[9] C. Bourdelle et al 2023 Nucl. Fusion 63 056021
[10] T. Luda *et al* 2021 *Nucl. Fusion* **61** 126048
[11] S. Saarelma *et al* 2023 *Nucl. Fusion* **63** 052002
[12] C. Angioni *et al* 2022 *Nucl. Fusion* **62** 066015

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# Overview on modelling of edge-SOL transport

W. Zholobenko

Max-Planck-Institut für Plasmaphysik, D-85748 Garching, Germany

Fusion reactors require stationarity (no ELMs), high confinement, and tolerable heat and particle exhaust. The transport of plasma across the magnetic field is largely determined by turbulence. It is increasingly well understood in the plasma core: quasi-linear models are a great tool for reactor optimisation [1]. Perhaps the largest remaining uncertainties in core turbulence are the electromagnetic stabilisation by fast particles, as well as boundary conditions at the edge. But the edge is much more than a boundary condition to the core!

The study of edge-SOL turbulence is more complex than core turbulence in certain aspects. A key quantity of interest is the (exponential) radial decay length $\lambda_q$ for heat in the SOL, determined by the competition between parallel outflow through the divertor and perpendicular turbulent transport. Due to this exponential decay of the background, separating fluctuations from it does not make sense. Fluctuation amplitudes are generally large and characterized by ballistically propagating "blob" filaments [2], originating from the very edge of the confined region [3]. Therefore, a global model is required that does not separate turbulence evolution from the background. Additionally, the study of the particularly large $E \times B$ shear in the plasma edge, thought to be responsible for the L-H transition, is facilitated by a global model that includes both mean-field (neoclassical) as well as turbulence driven zonal flows. Such a model must also include the sustainment of the background by neutral gas ionization [4]. Lastly, the dynamics is susceptible to the magnetic geometry, including the topology change from closed to open field lines, large magnetic field shear and flux expansion.

Edge-SOL turbulence codes also necessarily run into higher computational costs due to the large domain volume, high resolution requirements, and the inability to use globally field-aligned coordinates across the separatrix. Because of these complexities, to date, the work horse for SOL modelling and divertor design are mean-field transport codes that approximate turbulence simply by an "anomalous" radial diffusion coefficient, such as SOLPS, SOLEDGE3X, UEDGE, ECM3... However, the significant interest in the subject and the advances in high-performance computing led to the development of many codes for direct edge-SOL turbulence simulations. Since collisionality is never high enough in the plasma edge to justify a collisional fluid closure, gyrokinetic approaches are often preferred, with codes like XGC, Gkeyll, COGENT or GENE-X. On the other hand, dealing with the high collisionality in the SOL, particularly in detached divertor conditions, is challenging for gyrokinetics. This motivates many groups to continue developing fluid models (GRILLIX, BOUT++, SOLEDGE3X, FELTOR, GBS...), which have to be extended for kinetic effects such as Landau damping and trapped particles.

These codes find numerous applications. Besides basic plasma physics, they contribute to the understanding and development of reactor relevant scenarios, in particular those that integrate exhaust, confinement and stationarity requirements [5]. However, due to the complexity of these tools, the results are often difficult to judge. Therefore, a growing effort is to carry out detailed validations of the codes against well diagnosed experiments [6].

Nonetheless, whether gyrokinetic or fluid, direct turbulence simulations will not be fast enough to allow large parameter scans for reactor design or especially real-time control. They can however aid in the development of reduced turbulence models, to improve on the transport codes with "anomalous" diffusion. While local quasi-linear turbulence models are available for this purpose for the plasma core, non-local turbulence propagation must be accounted for in the SOL. This is indeed being attempted [7].

[1] C. Angioni *et al.*, Nucl. Fusion **62**, 066015 (2022)

[2] V. Naulin, J. Nucl. Mater. **363-365**, 24–31 (2007)

[3] W. Zholobenko *et al.*, Nucl. Mater. Energy **34**, 101351 (2023)

[4] W. Zholobenko, Nucl. Fusion **61**, 116015 (2021)

[5] E. Viezzer *et al.*, Nucl. Mater. Energy **34**, 101308 (2023)

[6] D. S. Oliveira, T. Body *et al.*, Nucl. Fusion **62**, 096001 (2022)

[7] S. Baschetti, Nucl. Fusion **61**, 106020 (2021)

# Injection of low-Z powders: a novel actuator to control pedestal, heat exhaust and plasma wall interaction

A. Bortolon
[1]Princeton Plasma Physics Laboratory, Princeton, New Jersey, USA

Injection of radiative impurities is an important tool to address the core-edge integration challenge in confined plasma fusion. Impurities are injected in gas form, either ideal (Ne, Ar, Kr) or molecular gases ($N_2$). The recent development and implementation of powder injection systems for fusion applications has expanded the range of extrinsic impurities that can be used, including non-recycling, low-Z elements such as Li, B, C, Si, but also compounds such as BN, $B_4C$ [1]. Powder is typically injected gravitationally, at controlled rates up to $10^{20}$ particles/s, ablated in the plasma edge to provide a local particle source. Powder "droppers" have been implemented in tokamaks (ASDEX-Upgrade, DIII-D, EAST, KSTAR, WEST) and stellarators (LHD, W7-X) and utilized for a variety of applications, spanning from real-time wall conditioning, to pedestal control, heat exhaust, as well as a tool to study impurity transport, dilution and isotope effects. This work provides an overview of the experimental results in these domains. Injection of B in plasma discharges has been shown to improve the wall conditions in similar fashion than glow discharge boronization, leading to reduction of wall fueling and impurity sources. Real-time boronization has allowed achieving and/or maintaining wall conditions required for low density, low collisionality operation in AUG [1] and DIII-D [2]. Injection of BN powder in AUG with W metal wall resulted in changes of pedestal structure consistent with the $N_2$ induced confinement recovery, but significantly reduced ammonia production. Injection of Li, B and BN have been shown to allow achievement of ELM suppressed or enhanced pedestal regimes, typically associated with an enhancement of pedestal-localized instabilities of different nature, coherent [3], quasi-coherent [4], bursting-chirping [5]. Net improvement of energy confinement has been observed in stellarator plasmas, correlated with changes in neoclassical electric field [6] and/or core turbulent transport [7]. Injection of Li, B and BN at the outer strike-point have demonstrated detachment with near target low-Z radiators [8].
Powder/granule injection provided new tools to probe transport. For instance, controlled C powder injection has been successfully used in DIII-D to modulate $Z_{eff}$, to test the role of dilution on H-mode power threshold, and or provide validation datasets of carbon dominated plasmas.
Coupled with integrated modeling tools to address dust transport, fluid SOL transport, kinetic modeling of pedestal/core transport, powder injection has opened new research avenues and offered new tools for probing key physics in different areas of fusion plasma. The scalability of the technique to next-step fusion machines will be discussed.

Work supported by U.S. Dep. of Energy contract DE-AC02-09CH11466

[1] R. Lunsford *et al.*, Nucl. Fusion, 59 (2019) 126034
[2] A. Bortolon *et al*., Nucl. Fusion, 60 (2020) 26010
[3] E. P. Gilson *et al.*, Nuclear Materials and Energy, 28 (2021) 101043
[4] Z. Sun *et al.*, Physics of Plasmas, 28 (2021) 082512
[5] T. H. Osborne *et al.*, Nucl. Fusion, 55 (2015) 63018
[6] R. Lunsford *et al.*, Physics of Plasmas, 28 (2021) 082506
[7] F. Nespoli *et al.*, Nat. Phys. 18 (2022) 350
[8] F. Effenberg *et al*.,Nucl. Fusion 62 (2022)106015

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# From the core to the divertor: Status of the impurity transport investigations at Wendelstein 7-X

F. Reimold[1], T. Romba[1], Th. Wegner[1], V. Winters[1], B. Buttenschön[1], A. Langenberg[1], C.D. Beidler[1], R. Bussiahn[1], O. Ford[1], J.M. García-Regaña[3], T. Gonda[6], F. Henke[1], D. Medina Roque[3], C. Swee[5], N. Tamura[2], J.L. Velasco[3], D. Zhang[1] and W7-X Team[1,a]
[1]Max-Planck Institute for Plasma Physics, Greifswald, Germany, [2]National Institute for Fusion Science, Toki, Japan, [3]Laboratorio Nacional de Fusión CIEMAT, Madrid, Spain, [4]Princeton Plasma Physics Laboratory, Princeton, USA, [5]University of Wisconsin, Madison, USA, [6]Auburn University, Auburn, USA, [a]see T.S. Pedersen NF 62, 042022 (2022)

Impurity transport and dynamics - both in the confined and the boundary plasma - are crucial aspects to the operation and prospects of a fusion reactor. Particularities of neoclassical transport in non-axisymmetric magnetic configurations exacerbate this for stellarators [1,2,3]. In order to optimize power and particle exhaust as well as fusion performance consistently, a detailed understanding of the transport in the plasma boundary and the confined plasma is required. In the boundary plasma, significant radiative dissipation via impurity line radiation facilitates the power exhaust and sufficient impurity content in the neutral gas domain is required for impurity particle exhaust and control. Both favor large impurity concentrations in the boundary domain. On the contrary, impurity contamination of the confined region must be limited to low levels to avoid reduced fusion rates via energy degradation from radiative losses or fuel dilution. Impurity enrichment $\eta = {}^{c_{imp,core}}/_{c_{imp,divertor}}$ [4] is the crucial quality parameter that describes the ability to accommodate the conflicting requirements.

A new paradigm of impurity transport in the island divertor of Wendelstein 7-X (W7-X) has emerged from detailed analysis and 'code experiments' with the EMC3-Eirene code [5]. For experimentally relevant conditions, impurity leakage from the divertor via parallel transport is very small and perpendicular transport into and across the main flow stagnation region – centered around the island O-point – seem to limit the divertor retention of impurities in W7-X. Extrapolated to a reactor, this finding might allow significant optimization of the island and divertor targets to achieve higher impurity retention and tailored impurity distributions.

Core impurity transport in stellarators is widely seen as a potential showstopper for a stellarator reactor as the neoclassical impurity transport lacks the 'standard' temperature screening of tokamaks [1]. Experiments in W7-X have shown that in most conditions, in particular with strong electron cyclotron resonance heating (ECRH) (>2MW), turbulence-dominated regimes for the impurity transport persist. In these regimes, flat impurity profiles with no accumulation are observed [6,7]. However, in scenarios with improved energy confinement peaked impurity profiles with strong accumulation are observed [6,8,9,10]. In these scenarios, turbulent transport of impurities can be suppressed and the remaining transport is consistent with purely neoclassical predictions [3]. Fortunately, experiments show that accumulation controlled with application of additional ECRH power. In addition, magnetic configurations and reactor relevant conditions have been identified where neoclassical temperature screening – in particular for He – is predicted. This paves the way to further optimization of the neoclassical impurity transport [11,12,13] for reactor-relevant parameters. Experimental data is collected to provide experimental verification of these predictions.

In conclusion, this contribution summarizes the specific aspects and the current state of understanding and experimental verification of impurity transport in the W7-X stellarator for both the boundary and confined plasma domains and tries to combine the competing requirements with respect to a future stellarator fusion reactor.

[1] Helander PPCF 54 (2012), [2] Burhenn NF 49 (2009), [3] Sudo, PPCF 58 (2016), [4] Bosch, PPCF 39 (1997), [5] V. Winters, NF (submitted), [6] T. Romba, NF (submitted), [7] T. Wegner, JPP (2022), [8] D. Zhang, EPS (2022), [9] G. Wurden, EPS (2022), [10] R. Lunsford, PoP 28 (2021), [11] C.D. Beidler, NF (submitted),[12] J. L. Velasco, NF57 (2017), [13] C. D. Beidler (in preparation)

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# Overview of JET T and D-T experiments

J. Hobirk[1] on behalf of the JET Contributors*

[1]*Max-Planck-Institut für Plasmaphysik, 85748 Garching, Germany*

In the years 2021/2022 experiments were performed in the JET tokamak in order to gather information about isotope effects including Tritium and Deuterium-Tritium mixtures in a metallic wall environment similar to ITER (ILW). This campaign has led to plasma with high fusion powers, larger than 10MW [1,2], and fusion produced energies up to 59MJ in T rich (15% D-85% T) plasmas [2]. The increased isotope mass has consequences for the plasma wall interaction, e.g. the W sputter yield is higher for T compared to D for the same temperature, but also for the pedestal parameters. As part to this contribution an overview of the high fusion power scenarios and the impact of the higher isotope mass on the hybrid scenario [1] and the L-H transition [3] will be presented. The optimisation procedure in Deuterium and the adjustments necessary to achieve a radiation stable plasma in T and D-T will be described. In terms of global confinement the D-T optimised pulses reached similar values as the D reference plasmas performed before the D-T campaign (without scenario adjustments for higher isotope mass) but with different kinetic profiles. A density increase connected to the change in isotope mass, a change in heating deposition due to different NBI penetration, a change in distribution and energy of fast particles and also a different ion-electron heat exchange are possible causes. A key ingredient in achieving the high temperatures for high fusion rates is the ion temperature gradient screening in the pedestal region [4] which could also be reproduced in D-T.If on the other hand, when the D-T and T plasmas are repeated in D in an engineering similar manner (matching auxiliary+alpha heating power and gas injection) a clear isotope effect on the confinement is seen. The kinetic profiles are different but the differences are smaller compared to the pre-D-T references, they already deviate in the pedestal region and point towards a strong edge-core coupling which has been observed in other plasmas as well [5]. Also T rich [2] discharges were developed, which maximise the beam target fusion rate by using D NBI in a mainly T plasma. In addition fundamental D ICRH heating was applied to increase the population of fast D ions in the energy range of E=100-200keV optimal for D-T reactivity, and hence the fusion rate further. In the “after-glow” plasmas [6] clear alpha particle effects have been demonstrated. Toroidal Alfvén Eigenmodes, with α-particle drive and thermal damping modelling consistent with the observed unstable mode were found. This scenario was build utilising an ITB to maximise the fusion output in D. These transport reductions were qualitatively reproduced in T and D-T. In the hybrid scenario the transient ITB like structures in the D reference could not be reproduced in T/D-T. The results presented here augment our previous knowledge of isotope effects. They play a key role in achieving high performance fusion plasmas and are therefore of special interest for designing plasmas for future D-T experiments in the next generation devices like ITER.

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*See the author list of J Mailloux et al. 2022 Nucl. Fusion https://doi.org/10.1088/1741-4326/ac47b4

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# Gyrokinetic modelling of the radial electric field well in L-mode edge tokamak plasmas

Y. Sarazin[1], G. Dif-Pradalier[1], R. Varennes[2], L. Vermare[2], P. Donnel[1], X. Garbet[1,3], V. Grandgirard[1], P. Hennequin[2], Y. Munschy[1], K. Obrejan[1], S. Rienäcker[2]

[1] CEA, IRFM, F-13108 Saint-Paul-Lez-Durance, France
[2] LPP, Ecole Polytechnique, CNRS, 91128 Palaiseau, France
[3] School of Phys. and Math. Sciences, Nanyang Technological University, 637371 Singapore

The radial electric field $E_r$ and its shear are well admitted to play a key role in triggering and sustaining transport barriers in tokamak plasmas, most notably at the plasma edge in H-mode regimes. However, the precise dynamics of its radial profile at the transition, the weight of the various mechanisms at play and their causality in this build-up still remain domains of active research. This talk addresses new experimental observations on the WEST tokamak prior to the H-mode transition, as well as hints provided by flux-driven turbulence simulations with the gyrokinetic GYSELA code.

Both in Tore Supra and WEST L-mode discharges, featuring either an axi-symmetric limiter or an X-point divertor, the plasma current $I_p$ is found to have a dramatic impact on the $E_r$ well just inside the separatrix. Indeed, Doppler Back-Scattering measurements indicate that the $E_r$ well at the edge deepens at increasing current [1]. Consistently, the deepening of the edge $E_r$ well is observed in GYSELA simulations when decreasing the safety factor q – hence increasing $I_p$ – at constant magnetic shear [2]. The preliminary analysis points towards a subtle balance between collisional flow damping and turbulent flow drive through turbulent Reynolds stresses.

These simulations, performed in the flux-driven regime with adiabatic electrons (Ion Temperature Gradient driven turbulence), encompass core and edge plasmas as well as a simplified modelling of the scrape-off-layer in limiter configuration. They reveal how both the Reynolds stress and more critically its diamagnetic counterpart [3] play a key role in the buildup of an $E_r$ well in the L-mode edge of tokamak plasmas. In particular, the detailed analysis of the vorticity dynamics allows one to precisely unravel the causality chain [4]. The subsequent onset of a self-consistent transport barrier is reported. Although it is not that of an H-mode (indeed, the adiabatic assumption would need to be alleviated to enable particle transport and to address this issue in a relevant way), it sheds light on the potential dynamics at play in the L- to H-mode transition.

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# Pedestal turbulence in AUG and JET from a global gyrokinetic perspective

L. A. Leppin[1], T. Görler[1], M. Cavedon[2] , M. Dunne[1], E. Wolfrum[1], S. Saarelma[3], L. Frassinetti[4], J. Hobirk[1], F. Jenko[1], ASDEX Upgrade Team[5] and JET Contributors[6]

[1]*Max Planck Institute for Plasma Physics, Boltzmannstraße 2, 85748 Garching b. München, Germany*
[2]*Dept. of Physics "G. Occhialini", University of Milano-Bicocca, Milan, Italy*
[3]*CCFE, Culham Science Center, Abingdon OX14 3DB, United Kingdom of Great Britain and Northern Ireland*
[4]*Division of Fusion Plasma Physics, KTH Royal Institute of Technology, SE-10691 Stockholm, Sweden*
[5]*See author list of U. Stroth et al. 2022 Nucl. Fusion* ***62*** *042006*
[6]*See Mailloux et al 2022 (*https://doi.org/10.1088/1741-4326/ac47b4*) for the JET Contributors*

Turbulence is one of the key ingredients in shaping H-mode pedestals. Identifying the relevant turbulent transport mechanisms in a pedestal, however, is a great scientific and numerical challenge. Here, we address this challenge by global, nonlinear gyrokinetic simulations of two pedestals: One from ASDEX Upgrade (Type-I ELMy H-mode) and one from JET (hybrid scenario H-mode). The global simulations permit to calculate heat fluxes due to ion-scale turbulence in the steep gradient region encompassing the full pedestal from top to foot. They are supported by detailed characterizations of gyrokinetic instabilities via local, linear simulations at pedestal top, center and foot as well as dedicated nonlinear electron-scale heat flux calculations. Simulations are performed with the gyrokinetic, Eulerian, delta-f code GENE (genecode.org) and employ a new code upgrade of its global, electromagnetic model that enables stable simulations at experimental plasma beta values.

In both investigated pedestals from AUG and JET, we find turbulent transport to have a complex radial structure that is multi-scale and multi-channel. Electron transport in the AUG pedestal is found to transition in scale. At the pedestal top ion-scale TEM/MTM instabilities fuel electron transport whereas in the pedestal center electron-scale ETG transport takes over [1]. Turbulent ion heat flux is present at the pedestal top and strongly reduces towards the steep gradient region. Magnetic shear is found to locally contribute to the stabilization of microinstabilities and reduction of heat flux. In the JET pedestal, transport due to ITG is found to play a much more important role, particularly on the pedestal top/ outer core. In both pedestals, ExB shear is confirmed to strongly reduce heat fluxes in the global, nonlinear simulations. We discuss implications of our results for the applicability of quasi-linear transport models in the pedestal.

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# Drift wave nature of the Weakly Coherent Mode on ASDEX Upgrade

M. Herschel[1,2,3], T. Happel[2], D. Wendler[1,2], M. Griener[2], J. Kalis[1,2], P. Manz[4], U. Stroth[1,2], ASDEX Upgrade Team[5]

[1]Technical University of Munich, Physics Department, Chair for Plasma Edge and Divertor Physics, 85747 Garching, Germany
[2]Max-Planck-Institute for Plasma Physics, Garching, Germany
[3]Universität Ulm, Ulm, Germany
[4]Institut für Physik, Universität Greifswald, Greifswald, Germany
[5]See Author list of Stroth, U. *et al.*, Nucl. Fusion **62** (2022) 042006

Improved confinement regimes will play a key role in the operation of future fusion power plants. The I-mode [1,2], one of these regimes, combines good energy confinement with the absence of ELMs. It features a characteristic edge transport barrier in energy but not in density. The mechanism of this selective transport reduction is still under discussion. A feature in the edge density fluctuations called the Weakly Coherent Mode (WCM) [3] is often brought forward as responsible for the I-mode specific transport barrier.

Measurements obtained from Doppler reflectometry and thermal helium beam spectroscopy at ASDEX Upgrade (AUG) are combined to analyze the WCM in unprecedented detail. The analysis is based on well-diagnosed discharges, with a clear localization of the mode in the region of the strongest electron pressure gradient. A phase velocity of the WCM in the electron-diamagnetic direction consistent with the dispersion relation of a near ideal drift wave is found [4], as shown in figure 1. Additionally, the drift wave nature allows for a prediction of the central frequency of the WCM, matching the experimental values over twelve varying plasma discharges.

The found drift wave nature of the WCM is in agreement with a proposed mechanism responsible for the I-mode confinement regime [5].

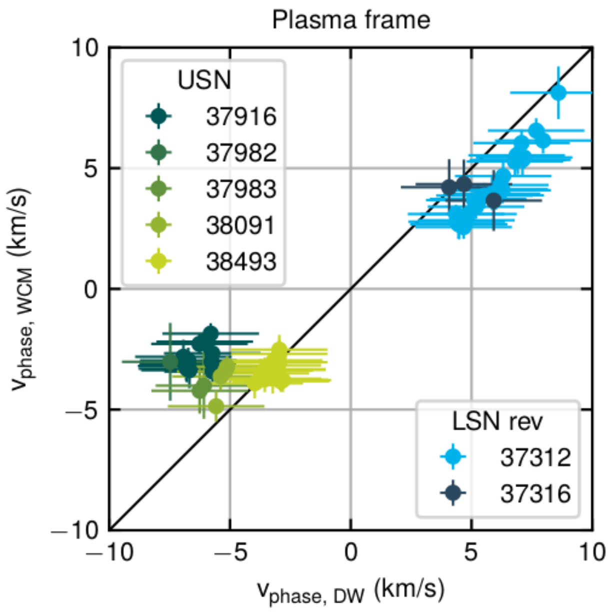


*Fig. 1. Measured WCM phase velocity versus ideal drift wave phase velocity. Shown are discharges in upper single null (USN) and lower single null with reversed magnetic field and plasma current (LSN rev). The deviations for #37916 and #37982 are likely due to a non-ideal drift wave at higher plasma beta.*

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# Edge deuterium temperature and toroidal rotation profiles at the ASDEX Upgrade tokamak and comparison to theory

P. Cano-Megias[1,2], E. Viezzer[3], R. M. McDermott[1], A. Jansen van Vuuren[3], E. Fable[1], C. Angioni[1], M. Cavedon[4], R. Chacartegui[2], D. J. Cruz-Zabala[3], R. Dux[1], T. Gleiter[1], F. Sciortino[1], U. Plank[1], C. F. B. Zimmermann[1] and the ASDEX Upgrade Team[1*]

[1] *Max-Planck-Institut für Plasmaphysik, Garching bei München, Germany*

[2] *Dept. of Energy Engineering, University of Seville, Spain*

[3] *Dept. of Atomic, Molecular and Nuclear Physics, University of Seville, Spain*

[4] *Dept. of Physics G. Occhialini, University of Milano-Bicocca, Milano, Italy*

The validation of transport models against experimental measurements is fundamental for understanding the nature of plasma transport and extending our predictive capabilities of the performance of future fusion devices. The complexities involved in diagnosing the main ion (in this work, deuterium) temperature ($T_D$) and toroidal rotation ($v_{tor,D}$) have prevented, for many years, the benchmark of theoretical transport models against main ion measurements. For this reason, $T_D$ is usually approximated by the impurity temperature ($T_z$), due to the rapid thermal equilibration expected between ion species. Additionally, $v_{tor,D}$ is typically estimated from $v_{tor,z}$ measurements in combination with neoclassical theory. In this study, direct measurements of main ions at the plasma edge using a Charge Exchange Recombination Spectroscopy system at the ASDEX Upgrade tokamak [1] are used to investigate main ion transport.

The edge main ion and impurity T profiles in inter-ELM H-mode phases have been measured in different collisionality and heating schemes. When the direct ion heating via NBI is high with respect to ECRH heating ($P_{NBI} > 2\ P_{ECRH}$), the deuterium and impurities are well equilibrated, $T_D = T_Z$. However, at low $Q_i/Q_e$, an unexpected difference between $T_D$ and $T_z$ at the pedestal top is observed [2]. Possible reasons for $T_D > T_z$ are examined in this work. Firstly, the possibility of a difference between the main ion and impurity core heat transport has been explored. To address this hypothesis, the main ion ($\chi_D$) and impurity ($\chi_z$) heat diffusivities have been evaluated by multi-species power-balance analysis with the transport code ASTRA [3-4]. The experimentally derived $\chi_D/\chi_z$ ratios are compared to a database of gyrokinetic simulations carried out with the GKW code [5]. The interpretative evaluation of the $\chi_D/\chi_z$ ratios is in good qualitative agreement with the GKW database. Secondly, it has been studied whether the discrepancy can be due to a different boundary condition for the ion species T at the separatrix. The temperature difference between the main ions and impurities at the separatrix is evaluated using the OEDGE package. OEDGE integrates OSM-EIRENE interpretations of the Scrape-Off Layer electron temperature and density with Monte Carlo simulations of minority impurity transport using the DIVIMP code. In addition, the difference between the main ion and impurity toroidal rotations ($v_{tor,D} - v_{tor,z}$) has been compared against neoclassical theory. The main ion toroidal rotation in the steep gradient region is in excellent agreement with NEOART calculations [7], while at the pedestal top neoclassical theory cannot quantitatively describe the experimental measurement. This result is consistent with previous works carried out at ASDEX Upgrade [8, 9].

* See full author list of U. Stroth et al. 2022 *Nucl. Fusion* **62** 042006

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# Initial Transport Results of the DIII-D Negative Triangularity Campaign

M.E. Austin[1], L. Casali[2], C. Chrystal[3], T.B. Cote[3], X. D. Du[3], D. Eldon[3], D.R. Ernst[4], F.O. Khabanov[5], P. Lunia[6], A. Marinoni[4], R. Mattes[2], G.R. McKee[5], S. Mordijck[7], A.O. Nelson[6], C. Paz-Soldan[6], L. Schmitz[8], F. Scotti[9], S. D. Stewart[5], K.E. Thome[3], A. Welsh[2]

[1]University of Texas–Austin, [2]University of Tennessee–Knoxville, [3]General Atomics, , [4]MIT Plasma Science and Fusion Center, [5]University of Wisconsin–Madison, [6]Columbia University, [7]William and Mary, [8]University of California, Los Angeles, [9]Lawrence Livermore National Laboratory

In the 2023 DIII-D negative triangularity (NT) campaign diverted discharges with $\delta \sim$ -0.5 were vertically stable, robustly ELM free, and attained high performance with $H_{98y2} \gtrsim 1$ and $\beta_N$>2.5. The plasmas exhibited elevated edge pressures and exhibit what is now referred to as an NT-edge, which is different than either typical L-mode or H-mode edges. An operational space with $I_p$=0.5-1.1 MA, $B_T$=1.0-2.1 T, $n_e$=2.0-14.5×10$^{19}$ m$^{-3}$ with a Greenwald fraction up to ~2, and $q_{95}$= 2.4 to 7 was covered and many different aspects of transport were studied in detail from the core to the scrape-off-layer. The dimensionless dependence of confinement was investigated by varying $\rho^*$ by a factor of 1.4 and $\nu^*$ by 3 and this analysis is ongoing. The effect of rotation on confinement was studied at $q_{95}$=2.8 and 4, where energy confinement at $q_{95}$=2.8 was reduced by ~30-40% and ~20-30% at $q_{95}$=4 when the torque was changed from strongly co-current torque to nearly balanced torque. These low-rotation NBI-heated plasmas will be compared with a dominant electron cyclotron heated plasma, where the electron temperature was much higher than the ion temperature. The impact of collisionality on particle transport was studied using gas puff modulation in a dimensionless scan. The electron thermal diffusion and convection was measured using modulated electron cyclotron heating to vary the electron temperature gradient. Alfvén eigenmodes are often observed in high-performance, high $q_{\min}$ NT plasmas, located at strongly-shaped plasma edge regions, deteriorating the energetic particle confinement and impacting the thermal performance. At this strong NT, plasmas retained weaker edge gradients due to a restricted second stability access (due to infinite-$n$ ballooning modes) in NT. At reduced NT of $\delta \gtrsim -0.3$, extended phases of limit cycle oscillations and H-mode are both observed. High radiation fraction plasmas with either Neon, Argon, or Krypton injection were created to assess the confinement impact and integration with mitigated divertor heat load. Scrape-off-layer heat flux measurements were taken throughout the campaign. Further turbulence and transport analysis of the DIII-D NT campaign will be presented.

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# Investigation of Turbulence Properties in Negative Triangularity Plasmas on DIII-D using Beam Emission Spectroscopy

S. Stewart[1], C. Chrystal[4], B. Geiger[1], F. Khabanov[1], A.O. Nelson[2], C. Paz Soldan[2], X. Qin[1], L. Schmitz[3], K. Thome[4], Z. Yan[1], G. McKee[1]
[1]University of Wisconsin – Madison, Madison, USA
[2]Columbia University, New York, USA
[3]University of California Los Angeles, USA
[4]General Atomics, San Diego, USA

Negative triangularity-shaped plasmas offer promising scenarios of high-performance plasmas with $H_{98}$~1, $\beta_N \geq 2.5$, and $f_{GW} > 1$while robustly avoiding ELMs. Experimental validation of the physical processes responsible for its improved confinement are yet to be demonstrated. In this work, a summary of turbulence characteristics in negative triangularity plasmas is performed using spatially localized ($k_y\rho_s < 1$), high speed (1 MS/s), multichannel (64 channels) density fluctuation measurements using Beam Emission Spectroscopy (BES). Turbulence amplitudes are shown to peak just inside the separatrix, similar to positive triangularity L-mode but with core turbulence amplitudes of $\tilde{I}/I<0.2\%$, similar to positive triangularity H-modes. Poloidal turbulent decorrelation lengths and times were found to peak just inside separatrix (5.5 cm, 10.3 μs) and fall off in the core. Scanning upper negative triangularity from -0.12 to -0.37 at the same input power induced a transition from H-mode to an ELM free NT-edge. Turbulence amplitudes are shown to decrease with the scan to stronger triangularity from a peak of 5% to 3% in the edge and from 0.3% to <0.1% at the top of the pedestal with nearly constant confinement time. Additionally, turbulence amplitudes are shown to increase nearly linearly with $q_{95}$, similar to positive triangularity. Increasing co-beam injected power tends to induce a radially broadened turbulence amplitude profile with lower peak values in both high and low torque shots. The turbulent scaling with an isolated scan of injected torque is also being investigated. Furthermore, multichannel velocimetry is computed using the full 2D BES array to investigate the poloidal flow spectra enabling investigation of Reynolds stress terms associated with the L-H transition in positive triangularity. Crossphase analysis of poloidally separated measurements supported by background poloidal velocity information may allow for the dominant turbulent mode (ITG, TEM etc.) to be identified via inference of the plasma advection velocity. These results are compared to *ExB* shearing rates as determined from CER data, to offer insight into the underlying turbulence saturation physics. Detailed BES analysis enables investigation of the turbulence properties responsible for H-mode like confinement in the robustly ELM-free negative triangularity regimes.

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# Pedestal properties of negative triangularity plasma in ASDEX Upgrade

B. Vanovac[1], M. G. Dunne[2], T. Pütterich[2], T. Happel[2],
R. Bielajew[1], D. Silvagni[2], G. Conway[2], J. Hobirk[2], D. Stieglitz[2], C. Yoo[1], A. E. White[1],
and the ASDEX Upgrade Team[3]

[1] *Plasma Science and Fusion Center, MIT, Cambridge, MA 02139, USA*

[2] *Max-Planck-Institut fur Plasmaphysik, 85748 Garching, Germany*

[3] *See author list of U. Stroth et al. 2022 Nucl. Fusion 62 042006*

Negative triangularity (NT) experiments in TCV [1], and DIII-D [2] have shown H-mode level confinement without Edge Localised Modes (ELMs), setting a path for NT as an operating scenario for burning plasmas. At ASDEX Upgrade (AUG), the first overview studies have shown that for the avoidance of H-modes at average triangularity $|\overline{\delta}| < 0.2$, AUG should operate with unfavorable ∇B drift direction [3]. For better physics understanding and the extrapolation of such regime, in this work, we compare MHD and turbulence between two NT discharges in favorable and unfavorable ∇B drift and two discharges at different $\overline{\delta}$ and the same ∇B drift. At the same $\overline{\delta}$, in both drift directions, plasma enters into an H-mode and is accompanied by ELMs with a high repetition frequency. A regular ELM repetition rate is observed in unfavorable while incoherent and non-constant in size ELMs are characteristic of the favorable ∇B drift. Despite increased ELM frequency with power, indicating their type-I ELM nature, the discharges are ideally peeling-ballooning (PB) stable. This suggests a potential role of resistive MHD or bursty turbulence during such ELM dynamics. Temperature fluctuations and correlation lengths, measured with the correlation ECE instrument, across the outer core and the pedestal do not differ between favorable and unfavorable ∇B drift direction at the highest matching heating power levels. The low $\overline{\delta}$ discharge stays in L-mode and has higher temperature fluctuations than its paired H-mode discharge at high $\overline{\delta}$. Correlation lengths measured with the correlation ECE system are the same, although the quality of confinement differs (L- vs. H-mode). This suggests that the measured fluctuation levels across the pedestal are insensitive to confinement transitions. In all discharges, we measure an increase in coherency levels when moving outwards towards the separatrix. Such strong coherency levels are characteristics of MHD modes or the weakly coherent mode as measured in I-mode.

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# Self-consistent simulations of plasma turbulence and neutral dynamics in detachment regime

D. Mancini[1], P. Ricci[1], N. Vianello[2,3]

[1] Ecole Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), CH-1015 Lausanne, Switzerland

[2] Consorzio RFX (CNR, ENEA, INFN, Università di Padova, Acciaierie Venete SpA), C.so Stati Uniti 4, 35127 Padova, Italy

[3] Istituto per la Scienza e la Tecnologia dei Plasmi, CNR, Padova, Italy

Simulations of high-density deuterium plasma turbulence in a lower single-null magnetic configuration based on a TCV discharge are presented. We reconstruct plasma dynamics of three charged species, electrons, atomic deuterium ions and molecular deuterium ions, interacting with two neutrals species, atomic and molecular deuterium, through ionization, charge-exchange, recombination and molecular dissociation processes. The plasma is modelled using drift-reduced fluid Braginskii equations, while neutral dynamics is described by a kinetic model solved without the use of Montecarlo methods. Increasing the D2 puffing we are able to reach a turbulent steady state where the inner divertor target is detached, presenting lower particle and heat flux for higher scrape-off layer density. The analysis of transport balance in the divertor volume shows that the decrease of particle flux is caused by a decrease of local neutrals ionization and parallel velocity. The effect of molecular reactions is to increase atomic neutral density, emitted by molecular activated recombination, increasing ionization and charge-exchange reactions above the X-point, see Fig. 1. Plasma energy losses are dominated by ionization reactions for temper-

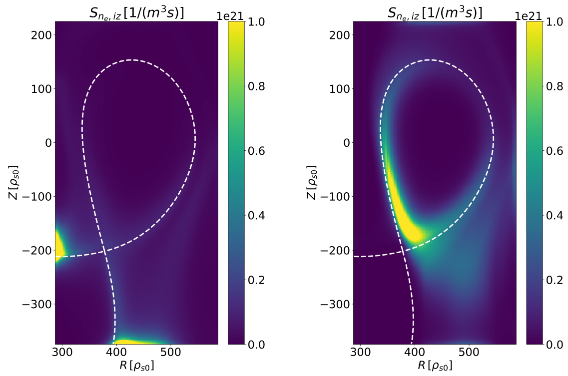


Figure 1: Time- and toroidally-averaged poloidal electron density source, $S_{n_e,\mathrm{iz}}$, for a low-density (left) and a high-density (right) simulations.

atures higher than 3 eV, by dissociation reactions for lower temperatures and by charge-exchange reactions for high neutrals density. The presence of strong electric fields in high-density plasma is also investigated through the analysis of Ohm's law, determining the importance of increased resistivity in the establishment of electrostatic potential gradients. A comparison between two simulations with different toroidal field direction leads to the assessment of the role of the ExB drift in the access of detachment conditions, in the asymmetries of the divertor targets and in the increase of turbulent transport with higher density at the mid-plane. The trends observed in the simulations are in agreement with experimental observations of increased density decay length, together with an increase of plasma blob size and radial velocity for increased plasma density.

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# Influence of self-consistently determined perpendicular transport coefficients on the numerical prediction of turbulent transport in a full WEST discharge

I. Kudashev[1], A. Medvedeva[1], M.S. d' Abusco[1], H. Bufferand[2], E. Serre[1]
[1]Aix-Marseille Université, CNRS, Centrale Marseille, M2P2 UMR 7340, Marseille, France
[2]IRFM, CEA Cadarache, 13108 Saint Paul-lez-Durance, France

Among all the issues to succeed ITER operation, the mitigation of heat and particles fluxes at the plasma facing component is certainly one of the most challenging. That requires a more complete understanding of the mechanisms at play, resulting from the complex interplay of transport processes in the plasma, losses at the wall and numerous atomic and molecular interactions.

Despite the constant increase of computational resources, 2D and 3D transport codes, in which plasma turbulence has been smoothed out by averaging, are nowadays the only ones able to tackle surface physics processes and atomic physics in realistic tokamak geometries. However, one of the main issues nowadays in such reduced models is the crude modelling of transverse transport fluxes resulting from the averaging of stresses due to fluctuations, assuming they are driven by local gradients, and characterized by ad-hoc diffusion coefficients (turbulent eddy viscosity) whose values are adjusted by hand in order to match numerical solutions with experimental measurements.

Our team recently proposed an advanced nonlinear model [1] that allows a self-consistent determination of the transport coefficients based on the turbulent kinetic energy and an ad-hoc characteristic time. Implemented in the high-order finite elements (Hybridized Discontinuous Galerkin) version of SOLEDGE3X [2, 3], we will show in this work how such modelling can improve the prediction of global transport properties in a full discharge of the WEST tokamak including the transient phases from start-up to ramp-down. Results from synthetic diagnostics [4] will also allow us to provide more reliable comparisons with available experimental measurements both in the plasma and at the divertor targets.

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# Development of full-f gyrofluid simulations for edge turbulence and magnetic reconnection

F. F. Locker, M. Held, A. Stürz, T. Stocker-Waldhuber, A. Kendl

Universität Innsbruck, Institut für Ionenphysik und Angewandte Physik, 6020 Innsbruck, Austria

Conventional gyrokinetic and gyrofluid models for nonlinear dynamics in magnetized plasmas are usually limited to either small fluctuation amplitudes (delta-f) and general perpendicular wave numbers (full-k), or to arbitrary amplitudes (full-f) but restricted to a long wavelength approximation (low-k). The latter approximation excludes some potentially important finite Larmor radius (FLR) effects on instabilities, turbulence and transport.

Recently, a more general "full-f full-k" gyrofluid model was developed (Held et al., Nucl. Fusion 2020) and first applied to test cases involving simulation of SOL blob filament propagation (Held and Wiesenberger, Nucl. Fusion 2023).

Here we report recent developments on "full-f full-k" gyrofluid codes for plasma edge turbulence and for magnetic reconnection. The models have so far been tested on simplified 2D (drift plane) geometries and are presently being extended to 3D edge/SOL geometry. First results are reported on what kind of differences and improvements in actual tokamak edge turbulence and transport modelling can be expected from these generalization to full-f with arbitrary wave numbers, in comparison to previously applied model approximations.

Furthermore, a 2D full-f gyrofluid model for collisionless magnetic reconnection, including finite Larmor radius (FLR) effects for a warm ions, is introduced and the differences to delta-f models are discussed. The linear tearing mode stability parameter is derived from the model to describe the initial evolution of a typical Harris-Sheet equilibrium situation. After introducing the numerical framework briefly, we discuss the status of our 2D simulations of magnetic reconnection under various conditions with a focus on tokamak parameters. The relevant (low) beta-regime, as well as the poloidal mode response is scanned to investigate the dynamics and morphologies of the evolving fields. The linear situation is compared to the turbulent one, to obtain insight on the respective evolution of the X-point geometry. We investigate FLR effects and the role of the electric potential in accelerating the magnetic reconnection. The results are compared to recent publications from delta-f gyrofluid, gyrokinetic and MHD simulations.

# Mitigating uncertainty in integrated scenario design for the STEP prototype powerplant

**F.J. Casson, H. Meyer, F. Eriksson, S. Freethy, D. Kennedy, S. Marsden, F. Palermo, B. Patel, E. Tholerus, T. Wilson, S. Bakes, C.D. Challis, T.C. Hender, M. Henderson, S.S. Henderson, M. Lennholm, K.G. McClements, D. Moulton, S.L. Newton, R.T. Osawa, C.M. Roach, S. Saarelma, and STEP team**

CCFE, UKAEA, Culham Science Centre, Abingdon OX14 3EB, UK

The STEP (Spherical Tokamak for Energy Production) programme has settled on a prototype reactor concept following a design space exploration phase integrating physics, technology, and engineering. The reactor is designed to generate net electricity, breed its own tritium, and run fully non-inductively. The spherical tokamak allows higher beta and elongation than in conventional devices but also presents additional challenges of limited space: The exhaust solution is likely to be double null with an exacting vertical control requirement; even a small distance between the two separatrices in double null can lead to significant loads on the inner divertor. A compact spherical design also limits the size of a central solenoid. A spherical tokamak reactor will rely on high bootstrap current fraction ($f_{bs}$) and non-inductive current drive, even during the current ramp up. The STEP concept has toroidal field $B_T$ = 3.2 T at geometric major radius R=3.6m, with plasma current Ip ~ 20 MA, aspect ratio A=1.8, elongation $\kappa$=2.8, fusion power $P_{fus}$~1.5 GW, normalised beta $\beta_N$ ~4.4, toroidal beta $\beta$ ~18%, bootstrap fraction $f_{bs}$ > 70%, and fully non-inductive steady state current drive provided by electron cyclotron and electron Bernstein systems. The small solenoid is used only for plasma initiation up to 2MA.

A spherical reactor will face the same reactor integration challenges as DEMO, with additional developments required for predictive modelling, due to the design space being less well explored: The anomalous transport is likely to be dominated by the electron channel. To predict the confinement, nonlinear gyrokinetic simulations of microtearing and kinetic-ballooning turbulence are being conducted, which then allow simple transport models to be calibrated and provides a basis to develop quasilinear models. We explore confinement predictions with different reduced models to produce a range of confinement predictions and to identify which confinement scalings are most relevant. Trends which persist across different transport models can be used to identify the scenario assumptions with the strongest sensitivities for performance. Several operating scenarios have been developed which define an operating window compatible with the machine design that mitigates the risks arising from the confinement uncertainty. Integrated modelling is embedded in a wider plant architecture integration programme which seeks to reconcile assumptions and constraints from different areas of plasma physics, technology, and engineering into a self-consistent burning plasma. These plasmas form the basis of more detailed modelling of heating and current drive, exhaust, MHD, fast particle losses, and technology selection.

Non-inductive current ramp up at reactor temperatures is quite novel; due to Faraday's law the ramp-up cannot be accelerated by adding additional current drive, and due to the required heating, the resistive timescale is very long. The non-inductive current ramp from 2MA to 20MA must be carefully managed to avoid the formation of current holes. To reach fusion burn conditions, the plasma must transition from a hot electron, low density plasma optimised for current drive efficiency to a high density plasma with hot ions and high bootstrap current.

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# Turbulent transport in the core of high-$\beta$ spherical tokamaks and predictions for STEP

M. Giacomin[1], D. Kennedy[2], D. Dickinson[1], B. S. Patel[2] and C. M. Roach[2]
[1]York Plasma Institute, University of York, York, United Kingdom
[2]Culham Centre for Fusion Energy, Abingdon, United Kingdom

The design and operation of future tokamaks, such as the Spherical Tokamak for Energy Production (STEP), which is an ambitious UK programme with the aim of demonstrating the ability to generate net electricity from fusion, require an accurate evaluation of the energy confinement time that depends on turbulent transport in the tokamak core. Particle and heat fluxes in the tokamak core are often evaluated by means of complex and computationally expensive nonlinear gyrokinetic simulations. These simulations are particularly challenging in compact ST devices, where the high β values achieved in the core provide a strong drive for electromagnetic instabilities, such as kinetic ballooning modes (KBMs) and microtearing modes (MTMs).

In this work, linear and nonlinear local gyrokinetic simulations are performed in the core of some of the most recent STEP pre-conceptual flat-top plasma reference scenarios.
The linear analysis shows the presence of a hybrid KBM instability at ion Larmor radius scale, which is found to couple to Trapped Electron Modes (TEMs). On the other hand, no instabilities are observed at electron Larmor radius scale. In addition to the hybrid KBM instability, subdominant collisional MTMs are also identified at ion scale. This hybrid KBM instability is carefully characterized by means of several parameter scans, which point out sensitivity to local gradients, magnetic shear, safety factor and $\beta$. In particular, the dominant mode is completely suppressed at $\beta'$ (the radial derivative of $\beta$) values higher than the nominal value.
The nonlinear analysis shows that, in the absence of equilibrium flow shear, the hybrid KBM instability drives heat and particle fluxes that are orders of magnitude larger than any acceptable value for STEP. The saturation at very large fluxes or the lack of saturation is consistently predicted by different gyrokinetic local codes and is shown to be consistent with a nonzonal transition. On the other hand, a strong effect of equilibrium flow shear on the heat and particle fluxes is observed. Importantly, fluxes are reduced to acceptable values when a level of equilibrium flow shear that is compatible with the diamagnetic flow is considered. The robustness of this result is improved by considering different gyrokinetic codes and different numerical implementations of the flow shear. Analogously to the linear analysis, the sensitivity to local gradients, magnetic shear, safety factor and $\beta$ is investigated in nonlinear simulations, once again revealing a strong role played by $\beta'$, with heat and particle fluxes that are reduced to acceptable values at high $\beta'$ even in the absence of equilibrium flow shear.
In contrast to the dominant mode, the subdominant MTM instability is found to drive negligible heat flux.

Finally, results of a multi-code linear and nonlinear benchmark are presented. An overall good agreement is found between the results of linear and nonlinear local gyrokinetic simulations carried out with different codes, improving the reliability of the simulation results presented here.

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# Pedestal stability analysis of MAST-U H-mode plasmas and impact of plasma shaping parameters

K. Imada[1,2], M. Knolker[1], T.H. Osborne[1], S. Saarelma[3], P.B. Snyder[4], H.R. Wilson[2] and the MAST Upgrade Team[3]

[1] General Atomics, San Diego, CA 92186-5608, USA
[2] York Plasma Institute, School of Physics, Engineering and Technology, University of York, Heslington, York, YO10 5DD, UK
[3] UKAEA, Abingdon, Oxon, OX14 3DB, UK
[4] Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA

Modern tokamaks can operate in high-confinement (H-)modes, in which a steep edge plasma pressure gradient is established, creating a "pedestal". H-modes are, however, subject to a class of explosive edge localised modes (ELMs), which could cause serious damages to the vessel walls of large tokamaks, such as ITER. Large ELMs, especially of Type-I kind, must therefore be mitigated or suppressed.
According to the "peeling-ballooning" theory[1], stability limit of the edge pedestal region depends primarily on two parameters: normalised pedestal pressure gradient, $\alpha$, and pedestal plasma current density, $J_{\text{ped}}$. Too steep a pedestal pressure gradient tend to push plasma towards high-$n$ ideal ballooning limit (where $n$ is the toroidal mode number), which typically result in Type-I ELMs. On the other hand, too high a pedestal current density at low $\alpha$ tends to make plasma unstable to low-$n$ peeling modes. Driving high pedestal current density at higher $\alpha$, however, can stabilise peeling modes and allow plasma to access the second stability region, with significantly elevated $J_{\text{ped}}$ and $\alpha$. Operating in such an improved confinement regime is not only an attractive option but likely necessary for future tokamak fusion reactors.
It has been observed in conventional tokamaks such as DIII-D that, amongst many other factors, plasma shaping parameters – elongation, triangularity and squareness – can have an impact on peeling and ballooning stability boundaries, as well as coupling between the two modes (which can close off access to the second stability region). Tailoring the plasma shape can therefore be one of the key factors to achieve high plasma confinement, required by future fusion reactors. However, comparatively little amount of study has been carried out in spherical tokamaks to date. Given that some compact fusion reactor concepts, such as STEP, are based on a spherical tokamak, it is important to determine if the plasma shaping parameters would affect pedestal stability in the same way for both conventional and spherical tokamaks and, if any, understand the differences.
This presentation focuses on the recent H-mode experiments on MAST-Upgrade spherical tokamak. It has been observed that, although the plasma was ultimately ballooning-limited, its $J_{\text{ped}}$ and $\alpha$ values were significantly higher compared to the predecessor, MAST. Furthermore, pedestal stability analysis using ELITE code has revealed that plasma was in a narrow region of stability between the peeling and ballooning stability boundaries, indicating weakening coupling of the two instability branches[2]. Dedicated experiments and theoretical modelling suggest that plasma shaping parameters do play a part in characterising pedestal stability. If they are found to play a significant role in weakening the coupling between peeling and ballooning modes, then our results will provide valuable information towards future spherical tokamak experiments, as well as furthering our understanding of pedestal stability physics for all tokamaks.

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# Initial results of gyrokinetic analysis of the core plasma in MAST Upgrade

B.S. Patel[1*], I. Cziegler[2], A.R. Field[1], T. Rhodes[3], P. Shi[1], S. Thomas[1] & the MAST Upgrade team[1,**]
[1]UKAEA, Culham Science Centre, Abingdon, OX14 3DB, United Kingdom
[2]York Plasma Institute, Department of Physics, University of York, Heslington, YO105DD, UK
[3]Dept. of Physics and Astronomy, University of California, Los Angeles, California 90095, USA

*email: bhavin.s.patel@ukaea.uk
** See author list of J. R. Harrison et al. *Nucl. Fusion* 59, 112011 (2019)

First measurements of core transport in MAST-U, along with measured turbulence behaviour and comparisons to linear and nonlinear gyrokinetic simulations will be presented here. Understanding the core confinement of spherical tokamaks is a critical aspect of achieving high performance plasmas. MAST-U, having undergone a significant upgrade since MAST with its increased toroidal field and advanced divertor configuration, offers an opportunity to compare experimental results to first principle-based models.

In this work we examine a MAST-U 750kA L mode plasma heated by 1MW of SS NBI (#47107), which is found to have dominant electron heat transport compared to the ions, demonstrated by interpretive transport analysis using TRANSP (47107Q04) [1], shown in Figure 1. The turbulent electron heat diffusivity dominates over the neoclassical electron and the ion transport throughout the plasma. The turbulent ion diffusivity is below ion neoclassical level when $\psi_N$ < 0.55 but becomes comparable to the ion neoclassical transport up to $\psi_N$ =0.9.

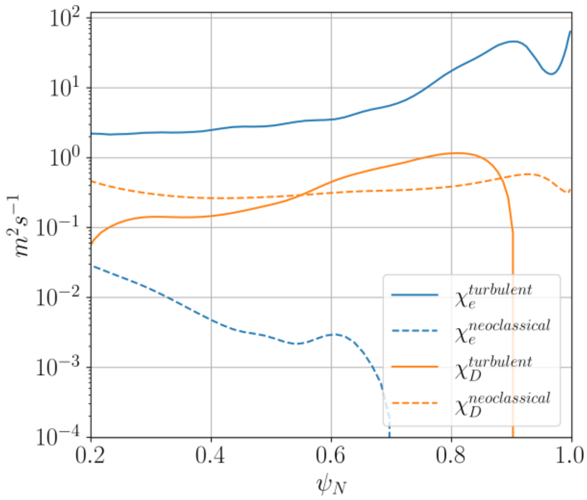


*Figure 1: Neoclassical (dashed lines) and turbulent (solid lines) diffusivities for the electrons (blue) and deuterium (orange) of MAST-U #47107 at t=0.6s*

This behaviour was typically seen in MAST [2], where it was shown that the ion scale turbulent transport is suppressed via equilibrium ExB shear [3], created by the rotation from the NBI.

This work presents initial linear and nonlinear gyrokinetic analysis found using the gyrokinetic codes CGYRO [4], GS2 [5] and GENE [6]. Preliminary linear gyrokinetic results find ITG modes peaking at $k_y\rho_s$=0.3 and ETG modes peaking at $k_y\rho_s$=20 for a mid-flux surface. The sensitivities of these modes will be examined in detail and nonlinear predictions will be compared to the experimental heat fluxes. BES and DBS diagnostic data, including intermediate $k_n$ spectra, fluctuation flow velocities and $\tilde{n}$ profiles will be compared to gyrokinetic predictions.

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# Increasing the predictive capability of impurity densities and their effects in tokamaks with integrated modeling based on theoretical transport models

D. Fajardo, C. Angioni, R. Dux, E. Fable, R.M. McDermott, G. Tardini and
the ASDEX Upgrade team[†]
Max-Planck-Institut für Plasmaphysik, Garching, Germany
[†]see author list of U. Stroth *et al* 2022 *Nucl. Fusion* **62** 042006

We present an integrated framework that demonstrates multi-species, multi-channel modeling capabilities for the prediction of impurity density profiles and their feedback on the main plasma through radiative cooling and fuel dilution. It combines all presently known theoretical elements in the local description of quasi-linear turbulent and collisional transport.
An ASDEX Upgrade (AUG) database of boron profile measurements [1] and GKW gyrokinetic results was used for the verification and validation of the transport models TGLF and QuaLiKiz. Neoclassical impurity transport is calculated with FACIT, which describes the strong effects of poloidal asymmetries at all collisionalities [2]. TGLF is complemented by the addition of centrifugal effects on the impurity transport coefficients in post-processing. The STRAHL-ASTRA coupling has been generalized to an arbitrary number of species. A scheme to correctly split the turbulent flux into diffusive and convective components was implemented for TGLF. This is essential for realistic ionization equilibrium calculations by STRAHL and allows for the consistent calculation of non-coronal impurity distributions and their radiated power profiles.

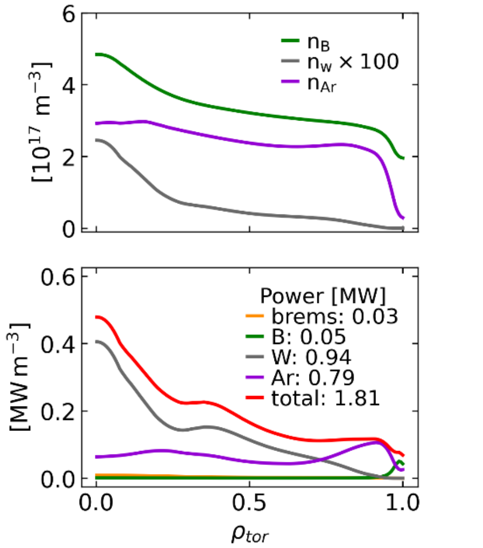


**Fig 1.** Full-radius simulations of B, W and Ar density and radiated power profiles in AUG #37041 with ASTRA-STRAHL-TGLF.

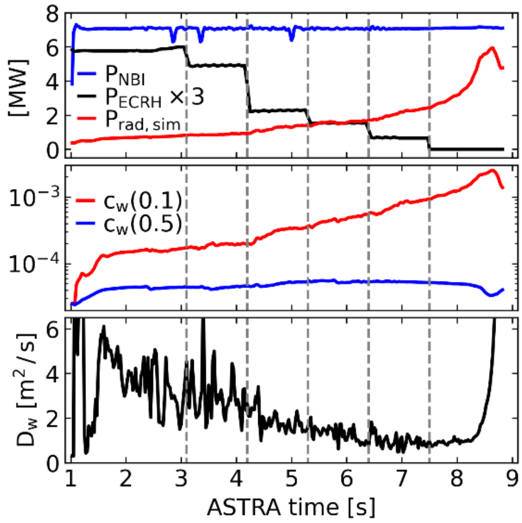


**Fig 2.** Time traces of NBI and ECRH power applied to AUG #32408, the predicted radiated power and the W on and off-axis concentrations and central diffusivity.

The workflow is shown to reproduce experimental results in full-radius L-mode and pedestal-inward H-mode modeling (Figs. 1 and 2). In particular, a highly radiative L-mode in AUG [3] with two intrinsic (B, W) and one seeded (Ar) species has been investigated and its measured radiated power and H-mode like confinement are obtained in the simulations. In addition, the control of W accumulation in NBI-heated plasmas with ECRH is studied in a dynamical simulation of an AUG H-mode featuring an ECRH ramp, obtaining good agreement with the experimental W behavior reported in [4] and highlighting the important role of enhanced central turbulent W diffusion with increasing ECRH. Comparisons to other modeling works [5], current limitations and future extension to full-radius H-mode simulations with IMEP [6] are discussed.

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# Experimental impurity transport studies for the plasma edge in different confinement regimes at ASDEX Upgrade

T. Gleiter[1,2], R. Dux[1], F. Sciortino[1], T. Hayward-Schneider[1], R. M. McDermott[1], T. Odstrčil[3], E. Wolfrum[1], U. Stroth[1,2], the ASDEX Upgrade Team[4]
[1]Max-Planck-Institut für Plasmaphysik, Garching, Germany
[2]Physik-Department E28, Technische Universität München, Garching, Germany
[3]General Atomics, San Diego, USA
[4]Authors of U. Stroth et al. 2022 Nucl. Fusion 62 042006

Understanding impurity transport in different tokamak regimes is necessary to devise reactor scenarios with suitable energy confinement and radiative power exhaust. For the most part, transport is dominated by turbulence, though, in regions with small gradients or suppressed turbulence, it can be close to neoclassical level, especially for impurity ions in high charge states. An important example is the edge transport barrier (ETB) that forms in the H-mode steep gradient region between occurrences of type-I edge localized modes (ELMs) [1]. At ASDEX Upgrade (AUG), the ETB produces an inward impurity transport, which is balanced by particle expulsion during the ELMs [2]. Whereas this mechanism was investigated for the type-I ELMy H-mode, thorough knowledge of impurity transport in the pedestal is lacking for other operation regimes. This includes promising high confinement modes for future reactor scenarios without type-I ELMs.

We present a comparative investigation of impurity transport at the plasma edge in AUG. High resolution charge exchange recombination spectroscopy (CXRS) data was acquired during a series of neon or argon seeded discharges in various confinement regimes, with a main focus on the quasi continuous exhaust (QCE) regime and the enhanced D-alpha (EDA) H-mode. Stepwise transitions into the type-I ELMy H-mode allow for direct comparisons to this reference regime. With a tailored diagnostic setup, we recorded absolutely calibrated spectra that comprise line emission from multiple impurity charge states. Density profiles are obtained from the radiances making use of ADAS atomic rates [3], the AUG neutral beam model, and the beam attenuation code COLRAD [4]. Reproducing the density data of multiple charge states with the 1.5D transport solver Aurora [5] allows to disentangle diffusive (D) and convective (v) contributions in our steady state discharges. Our analysis also takes into account charge exchange reactions between thermal neutral D and impurity ions. A Bayesian inference with the nested sampling algorithm MultiNest [6] yields the full probability distribution of the D and v profiles. In the course of this work, the ImpRad module in the One Modeling Framework for Integrated Tasks (OMFIT) [7] was extended, in particular for impurity transport inferences based on AUG CXRS data.

First results show that the pedestal gradient of the observed total neon density is less steep in the QCE regime than in the type-I ELMy H-mode and the related pedestal pinch of the v/D ratio is reduced by a factor of 2. The diffusion is raised above the neoclassical level expected in the ETB. We hypothesize that turbulence contributes to the edge impurity transport in the QCE regime and prevents impurity accumulation in this ELM-free operation mode at AUG.

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# An algorithmic framework for developing saturation rules in reduced core transport models

H.G. Dudding[1], F.J. Casson[1], D. Dickinson[2], B.S. Patel[1], C.M. Roach[1],
E.A. Belli[3] and G.M. Staebler[4]
[1]Culham Centre for Fusion Energy, Oxfordshire, UK
[2]York Plasma Institute, York, UK
[3]General Atomics, San Diego, US
[4]Oak Ridge National Laboratory, Oak Ridge, US

Quasilinear turbulence models such as TGLF [1] and QuaLiKiz [2] form an integral part of modern turbulence prediction and integrated modelling efforts, due to the rapidity with which they can estimate turbulent transport. This is achieved by combining the properties of the linearly unstable modes with a *saturation rule*, which prescribes the shape of the saturated potential spectrum of the turbulence against binormal wavenumber.
Saturation rules are typically tuned to databases of local nonlinear gyrokinetic simulations, which limits the quality of model predictions outside of the parameter space on which they have been trained. As a result, quasilinear models tend to be under continual development and verification as new parameter regimes are explored. Such a need is especially relevant today, with the approach of new experiments like ITER and conceptual power plants like STEP, which will operate in new regimes. This heralds the need for more generally applicable turbulence models to strengthen our confidence in future confinement predictions.
Here we present an overview of the framework we have adopted for developing such improvements, and as a case study illustrate how this approach was used to establish the new TGLF saturation rule SAT3 [3] to accommodate the dependence of confinement on main ion mass. A view as to the future of saturation rule development is also discussed, including the extension of the development algorithm to electromagnetic saturation; the role of community-based gyrokinetics projects such as Pyrokinetics and GyroKinetic DataBase; and the interplay between saturation rule development and machine learning applications in turbulent transport.

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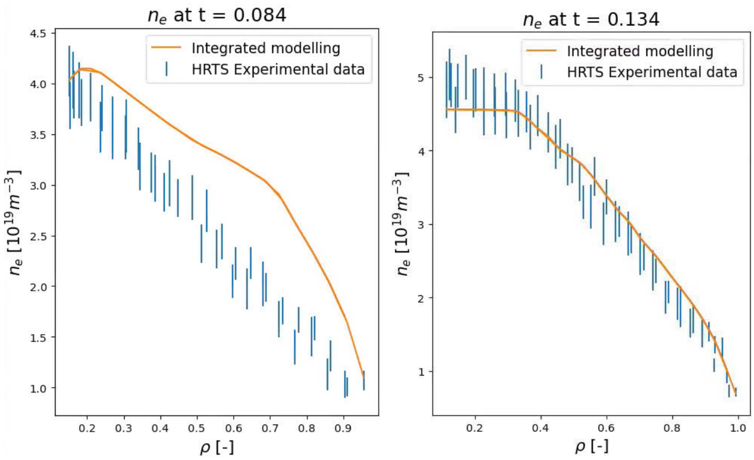
# Integrated modelling of ohmic ramp-up at TCV

M. Marin[1], Y. Camenen[2], P. Manas[2], O. Sauter[1] and TCV contributors
[1]EPLF, SPC, Swiss Plasma Center, Station 13, Lausanne, 1015, Switzerland
[2]Aix-Marseille Université, CNRS, PIIM UMR 7345, 13397 Marseille, France

The ramp up of a tokamak discharge is a critical phase of operation where the current is raised to its flattop value and that requires careful consideration of various engineering and physics aspects. In particular, it is necessary to simultaneously ensure MHD stability, minimize flux consumption and avoid disruptions. The high nonlinearity of the problem, the significant changes in the shape of the magnetic field and the uncertainty on the boundary and initial conditions are among the challenges in predicting the ramp-up. This work presents an integrated modeling approach for well-diagnosed TCV ramp-ups, developed in collaboration with experiments and modeling efforts conducted in multiple machines, including at JET [1], ASDEX [2], and WEST [3]. JINTRAC [4], a High Fidelity Pulse Simulator (HFPS) is utilized to self consistently predict the evolution of current, temperature, and density.

The self-consistent prediction of the density, including the evolution of impurities, has not been extensively explored and represents the main element of novelty in this work. To this end, the HFPS employs QuaLiKiz [5,6] to predict turbulent fluxes and FRANTIC [7] to predict neutral sources while using a closed loop feedback to match experimental line-averaged density. Our results exhibit good agreement with experimental data, with the simultaneous comparison of multiple channels increasing confidence in the prediction.




*Comparison of the measured and modelled electron density profile for the chosen TCV ohmic discharge. Current and density are still rising at the two selected times*

While no rigorous uncertainty quantification is performed here, we present sensitivities on the modeling choices and propose measures of agreement. The turbulence regime predicted by QuaLiKiz is analyzed, showing a transition from Trapped Electron Mode (TEM) to Ion Temperature Gradient (ITG) dominated turbulence, consistently with higher fidelity simulations [8]. Future work will include an increase in the number of shots for validation, more quantitative comparison of the QuaLiKiz predictions with higher fidelity simulations and additional experiments encompassing various ramp-up conditions.

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# Excitation of high frequency waves in non-linear 6D kinetic Vlasov simulation with steep gradients

**M. Raeth**[1], **K. Hallatschek**[1], **K. Kormann**[2], **N. Schild**[1]

1. *Max Planck Institute for Plasma Physics, Boltzmannstraße 2, 85748 Garching, Germany*
2. *Ruhr-University Bochum, Universitätsstraße 150, 44801 Bochum, Germany*

The Vlasov-Fokker-Planck-Maxwell system can describe physical phenomena in a tokamak plasma on all scales starting at large dynamics of the size of the device down to micro-scales of the order of the Larmor radius. However, up until recent years 6D simulations were beyond the available computational capabilities. As an approximation, gyrokinetic transport simulations are in good agreement with experiments in the tokamak core regime, where only small perturbation amplitudes and gradients are present. However, in regimes of high gradients and large turbulence fluctuation amplitudes, such as the plasma edge of a tokamak, the gyrokinetic approximation is debatable and at least those models based on a $\delta f$ approximation break down completely.
On the other hand, experimental results from the PLT (Princeton Large Torus) tokamak at the Princeton Plasma Physics Laboratory (PPPL) have shown the suppression of fluctuations through the injection of high intensity ion Bernstein waves (IBWs) [1]. These studies focused on the empirical effect of externally excited IBWs, but do not explain the mechanisms. IBWs break the gyrokinetic approximation and therefore neither their intrinsic stability nor their influence on energy and particle transport can be studied by current gyrokinetic turbulence and stability codes. The capability to simulate the excitation of IBWs, would be an important stepping stone toward a more comprehensive understanding of the high-frequency regime in the plasma edge.
We have developed an optimized and scalable semi-Lagrangian solver for the 6D kinetic Vlasov system based on a highly efficient scheme to treat the $\boldsymbol{v} \times \boldsymbol{B}$ acceleration from the strong background magnetic field. This allows us to simulate the excitation of plasma waves and turbulence with frequencies beyond the cyclotron frequency without a limitation by the gradient strength or fluctuation level. It is well tested in the low-frequency regime and produces correct results for the dispersion relation as well as energy fluxes in the linear and non-linear regime [2].
In this contribution, the first results going beyond the low-frequency regime will be presented, including a comprehensive understanding of the stability properties of the ion Bernstein waves. We have been able to show their destabilization for steep temperature and density gradients. The growth rates can thereby exceed those of the ITG instability, especially when a density gradient is present. The predicted instability is accurately reproduced by our simulation in a local gradient set-up as well as using a non-linear treatment of the gradients. The results will be presented along with the (rather non-trivial) description of the various turbulent fluxes in the kinetic system.

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# Experimental validation of momentum transport theory in the core of a tokamak plasma

C. F. B. Zimmermann[1 2], R. M. McDermott[1], C. Angioni[1], B. P. Duval[3], R. Dux[1], E. Fable[1], T. Luda[1], A. Salmi[4], U. Stroth[1 2], T. Tala[4], G. Tardini[1], T. Pütterich[1], and the ASDEX Upgrade Team[5]

[1] *Max-Planck-Institut für Plasmaphysik, 85748 Garching, Germany*
[2] *Physik-Department E28, Technische Universität München, 85747 Garching, Germany*
[3] *EPFL, Swiss Plasma Center, CH-1015 Lausanne, Switzerland*
[4] *VTT, P.O. Box 1000, FI-02044 VTT, Finland*
[5] *see the author list of U. Stroth et al. 2022 Nucl. Fusion 62 042006*

Understanding momentum transport is needed to reliably predict the plasma rotation profiles in future fusion device. This is relevant, as rotation impacts neoclassical and impurity transport, MHD instabilities, turbulences, and confinement. With the advanced momentum transport analysis framework presented herein, torque perturbation experiments can uniquely, separately, and concomitantly determine the contribution of diffusion, convection, and intrinsic torque to momentum transport within the core plasma. The analysis, self-consistently, includes the time dependencies of all transport mechanisms, which is essential to compensate for changes in the transport synchronous with the torque perturbation in order to separate the momentum fluxes and closely match the experiment. The experimentally inferred transport coefficients match gyrokinetic predictions for the Prandtl and pinch number providing an unprecedented validation. Scaling laws fitted to a database of gyrokinetic calculations are compared to main parameter dependencies from the experimental results. The intrinsic torque is particularly interesting, because it can spin-up the plasma from rest and is the largest uncertainty when predicting the rotation of future fusion devices. It is found to be co-current directed in ITG dominated discharges and originates at the edge of the plasma core. The size of this intrinsic torque correlates with the plasma pressure gradient in the pedestal, in agreement with theoretical predictions. This work opens new experimental possibilities for the validation of momentum transport theory, which should lead to the first, consistent, physics-based and validated predictions on momentum transport for reactor scenarios.

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# Applying self-consistent electron heat transport and ECH deposition profile estimation in DIII-D

J.H. Slief[1,2], M. van Berkel[1], E. Westerhof[1]
[1]DIFFER, Eindhoven, the Netherlands
[2]Eindhoven University of Technology, Eindhoven, the Netherlands

Perturbative studies of electron heat transport in fusion devices are typically based on the following transport model:

$$\frac{\partial}{\partial}\left(\frac{3}{2}n_\mathrm{e}T_\mathrm{e}\right) = \nabla\cdot\left(n_\mathrm{e}\chi\nabla T_\mathrm{e} + n_\mathrm{e}VT_\mathrm{e}\right) + KT_\mathrm{e} + S$$

where the evolution of density $n_\mathrm{e}$ and temperature $T_\mathrm{e}$ is determined by a set of transport parameters, namely diffusivity $\chi$, convectivity $V$ and reactivity $K$. By modulation of the heat source term $S$, the model can be linearized so that the transport parameters can be reconstructed based on time-resolved temperature and density measurements. Traditionally, it was assumed that spatial profile of the source term is well known, for instance through ray tracing calculations.

In the last decade, however, several experimental and numerical investigations have shown that there can be a mismatch between measurements of the electron cyclotron heating (ECH) power deposition profile and deposition profiles computed from beam and ray tracing codes [1,2,3,4]. This phenomenon is not yet well understood but potentially problematic for large machines like ITER, particularly for control tasks like profile control and the control of neoclassical tearing modes (NTMs) [5].

One of the difficulties with estimating EC power deposition profiles from measurements is transport broadening of the deposition profile on timescales shorter than the measurement time resolution [6]. To address this issue, we have developed tools that self-consistently include transport effects on the deposition profile [7,8,9], negating the transport broadening issue that plagued earlier approaches. These methods make it possible to directly compare beam or ray tracing deposition profiles against experimental measurements.

In this work, we apply our methods to measure the ECH deposition profile in a set of DIII-D discharges spanning a range of regimes and confinement modes. We validate the assumptions made in the methods and compare the measured profiles against results from the TORAY ray tracing code [10]. The results show that the different methods paint a consistent picture of deposition broadening, with measured deposition profiles between 1.5 to 3.5 times wider than the TORAY profiles. We show that this level of broadening, which falls within the expectation for ITER [5], could increase the power requirements for total NTM suppression to above the total power available in ITER.

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# Advanced energetic particle transport models

M.V. Falessi[1], P. Lauber[2], L. Chen[1,3,4], Z. Qiu[3,1], F. Zonca[1,3] and the ATEP Team

[1] Center for Nonlinear Plasma Science and ENEA C. R. Frascati, Frascati, Italy

[2] Max-Planck-Institut für Plasmaphysik, Boltzmannstr. 2, 85748 Garching, Germany

[3] Inst. for Fusion Theory and Simulation, School of Physics, Zhejiang Univ. Hangzhou,China

[4] Department of Physics and Astronomy, University of California Irvine, Irvine, CA, USA

The aim of ITER and magnetic confinement fusion is to achieve burning plasma operations. Although fusion particles would sustain the fusion process in ideal conditions, collective electromagnetic fluctuations may cause increased energetic particle (EP) losses. Therefore, understanding and accurately describing EP transport is crucial. EP transport involves different spatio-temporal scales and regimes compared to thermal plasma due to high energies and resonant interactions. Moreover, EP transport is a multi-scale process where EPs act as mediators of plasma cross-scale couplings [1]. Thus, a self-consistent, first-principle-based theoretical description is mandatory, based on a global e.m. gyrokinetic approach. However, accurately depicting meso-scales and the weak collisional nature of EPs require computationally demanding formulations, particularly on transport time scales [2]. This makes predictive analyses challenging, which calls for reduced descriptions.

To address these issues, the Advanced energetic particle transport models (ATEP) EUROfusion Enabling Research project has focused on constructing, implementing, verifying, and validating various reduced transport models for EPs. A recent general theoretical framework to describe EP transport has been established [2,3]. First, the need for developing a gyrokinetic theory for EP phase space transport will be presented [2]. The main differences with multi-scale gyrokinetics [4] will be discussed, emphasizing the assumption of scale separations between fluctuations and equilibrium meso-scale corrugations that does not apply to EPs. Furthermore, we will show how defining this theoretical framework lead to a renormalization of the usual plasma equilibrium in the presence of a finite level of electromagnetic fluctuations, called the zonal state. The governing phase space transport equations will be derived, resulting in a novel full f/ delta f mixed approach to be solved within a hierarchy of models of different complexity and need for numerical resources. The transport models will be integrated into the ITER Integrated Modelling & Analysis Suite (IMAS) based on stability analysis provided by local and global GK codes (DAEPS, LIGKA). We extended the well-known LIGKA/HAGIS suite to calculate EP phase space fluxes [5] and solve the EP transport equation in phase space for realistic EP populations such as NBI distributions at ITER. Finally, we introduce a dedicated diagnostic to monitor the nonlinear dynamics of the zonal state, highlighting its importance for verifying and validating EP transport models and reviewing its application to ORB5 [7] and HYMAGYK [8] codes.

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# Impact of supra-thermal particles on plasma performances at ASDEX Upgrade with GENE-Tango simulations

A. Di Siena[1], R. Bilato[1], M. Bergmann[1], A. Bañón Navarro[1], G. Merlo[2], T. Görler[1], E. Poli[1], M. Weiland[1], G. Tardini[1], V. Bobkov[1], R. Fischer[1], T. Dannert[1], K. Germaschewshi[3], B. Allen[4,5], F. Jenko[1], and the ASDEX Upgrade Team[*] and the EUROfusion MST1 Team[+]
[1]Max-Planck-Institut für Plasmaphysik 85748 Garching, Germany
[2]The University of Texas at Austin 78712 Austin, USA
[3]Department of Physics and Space Science Center, University of New Hampshire, Durham, USA
[4]Department of Computer Science, The University of Chicago, Chicago, USA
[5]Computational Science Division, Argonne National Laboratory, Lemont, USA
[*]See the author list of "U. Stroth et al. 2022 Nucl. Fusion 62 042006"
[+]See the author list of "B. Labit et al. 2019 Nucl. Fusion 59 086020"

Suprathermal particles generated by auxiliary heating systems have been shown to strongly suppress ion-scale plasma turbulence in flux-tube gyrokinetic simulations [1]. However, these studies are limited to micro-turbulence time scales, preventing changes in macroscopic quantities like plasma profiles and magnetic geometry. As a result, there is still no self-consistent physical picture describing the direct influence of fast particle turbulence stabilization on plasma profiles. Another issue is that quasilinear models often fail to incorporate supra-thermal particle effects on turbulence [2].

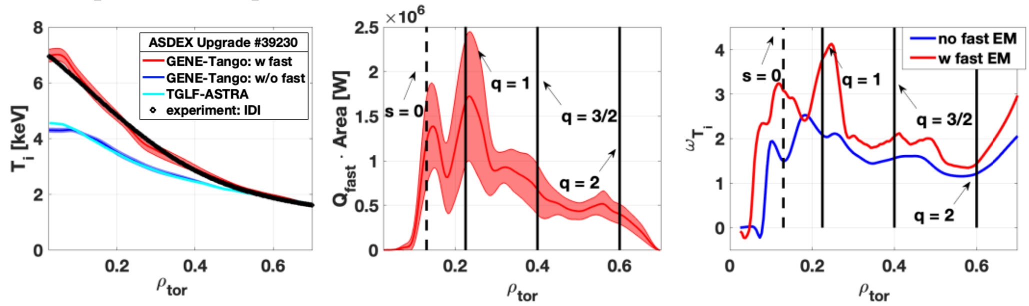


*Figure 1. Excellent agreement between the ion temperature profile computed by GENE-Tango with fast ions (red) and the experimental measurements for the ASDEX Upgrade discharge #39230 (left figure). TGLF-ASTRA (cyan) under-estimate the $T_i$ peaking, reproducing the GENE-Tango profile without fast ions (blue). The logarithmic thermal ion temperature profile obtained with GENE-Tango retaining fast ions peak at the location of energetic particle-driven modes (middle-right figure).*

We report the first-of-its-kind flux-driven-like GENE-Tango [3] simulations of an ASDEX Upgrade H-mode discharge, exhibiting a significant increase in plasma performance due to energetic particles. Only when the energetic particle species is consistently included in the radially global nonlinear electromagnetic GENE-Tango simulations we observe an excellent agreement for all plasma profiles (see Fig. 1). When fast ions are removed from the modelling, the ion temperature profile flattens significantly, matching the profiles computed with TGLF-ASTRA that significantly under-predicts the $T_i$ on-axis. Detailed analyses on the steady-state plasma profiles reveal that the thermal ion logarithmic temperature gradient strongly increases in correspondence with a RSAE (at s = 0) and an EP mode (at q = 1). The dynamics of this complex interplay between fast ions, Alfvén modes and the peaking of the on-axis ion temperature are discussed and explained.

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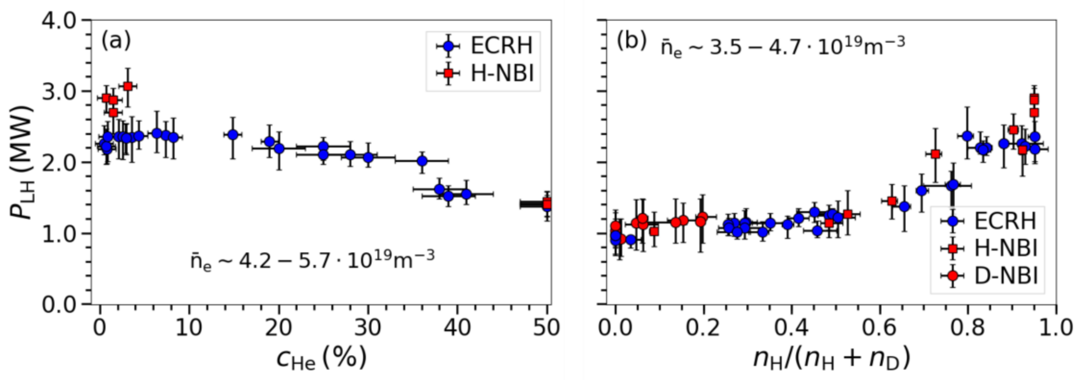
# Experimental assessment of the role of the main ion species composition on the access into H-mode

U. Plank[1], C. Angioni[1], G. D. Conway[1], T. Happel[1], A. Kappatou[1], T. Pütterich[1], P. Schneider[1], R. M. McDermott[1] and the ASDEX Upgrade Team[*]

[1]Max-Planck-Institute for Plasma Physics, Garching, Germany
[*]see author list of U. Stroth *et al.*, Nucl. Fusion **62** (2022) 042006 for the ASDEX Upgrade Team

Tokamak plasmas exhibit a strong dependence of the H-mode power threshold ($P_{LH}$) on the main ion plasma species. For hydrogenic species a general result is that $P_{LH}$ increases inversely with the ion mass [1]. However, future fusion reactors operate in mixed main ion species plasmas and for such conditions less conclusive data are presently available as they are sometimes even contradictory. To guarantee stable H-mode operation in future fusion power plants, it is therefore vital to improve our understanding of the impact of the plasma's ion composition on the transition from L- into H-mode (L-H transition).
At the ASDEX Upgrade tokamak dedicated L-H transition experiments were performed in mixed hydrogen-helium (H-He) and deuterium-hydrogen (D-H) plasmas over the past few years, now covering the entire range of H-He and D-H ion fractions. The non-linear dependencies of $P_{LH}$ in H-He and D-H plasmas are shown in Fig. 1(a) and 1(b), respectively. In both plasma mixes $P_{LH}$ changes most strongly when the turn-over from one dominant plasma species to the other one occurs.




Power balance analysis shows that the total edge ion heat flux follows the behavior of $P_{LH}$ [2,3], whereas the experimental edge ion temperature and the edge radial electric field gradients are found to be relatively constant at the L-H transition for similar edge densities. This implies that also the ion heat diffusivity changes non-linearly with plasma mixture and, thus, $P_{LH}$ seems to be directly linked to the edge transport properties of the different ion species, consistent with recent findings in JET deuterium-tritium plasmas [4] and theory [5].
This contribution also elucidates differences in L-H transition data of mixed ion species plasmas in the low and high density branches. In addition, the role of $Z_{eff}$ for the L-H transition in both D and H plasmas is investigated. Finally, comparisons to results on the L-H transition in mixed main ion species plasmas from other tokamaks are presented.

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POSTER

# Tuesday Session

# Full-F turbulent simulations of a linear plasma device using a gyromoment approach

J. E. Mencke[1], B. J. Frei[1], P. Ricci[1]

[1]École polytechnique fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), CH-1015 Lausanne, Switzerland

We present the first full-F nonlinear turbulent simulations of a linear plasma device based on a gyromoment approach. Linear devices such as LAPD [1] or RAID [2] are ideal configurations to study basic properties of turbulence in a simplified setting, as a first step when developing new numerical codes for simulating plasmas. Furthermore, linear devices allow for a more direct comparison with experiments than other fusion devices. The model used for our simulations is based on the gyromoment approach presented in Ref. [3] with absence of magnetic shear and curvature. The gyro-centre equations of motion are derived directly from the gyro-centre Lagrangian retaining finite Larmor radius corrections. The full-F distribution functions are expanded in an Hermite-Laguerre basis which enables the simulation of the plasma dynamics in regimes far from thermal equilibrium, including advanced collision operators. The full gyrokinetic version of this model is able to describe both large- and small-scale electromagnetic fluctuations.

The model in Ref. [3] is implemented numerically by retaining up to second order Larmor radius corrections in the configuration of a linear device. The nonlinear and turbulent evolution of the ion full-F distribution function is obtained in terms of gyromoments, coupled with a fluid electron model. The simulation results show large amplitude and large size turbulent structures. (see Fig. 1.) The nature of the instabilities driving turbulence is investigated and compared with simplified fluid models.

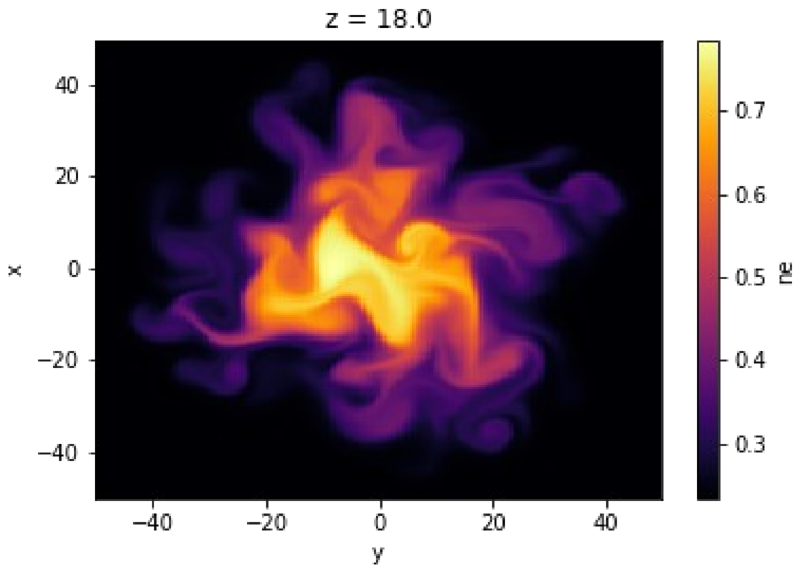


*Figure 1: Turbulent structures developing in the linear device. The electron density is shown in a cross section perpendicular to the magnetic field.*

This model is a first step towards a full-F gyrokinetic code for the simulation of the boundary of fusion devices which is expected to be particularly efficient with respect to current gyrokinetic models and with higher physical fidelity than fluid approaches.

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# Experiments and gyrokinetic simulations of TCV plasmas with negative triangularity in view of DTT operations

A. Balestri[1], P. Mantica[2], F. Bagnato[1], S. Coda[1], P. Innocente[3,4], P. Muscente[3,5], O. Sauter[1], M. Vallar[1] and the TCV team[1]

[1]École Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), 1015 Lausanne, Switzerland
[2]Istituto per la Scienza e la Tecnologia dei Plasmi, CNR, 20125 Milan, Italy
[3]Consorzio RFX (CNR, ENEA, INFN, UNIPD, Acciaierie Venete SpA), Padova, Italy
[4]Istituto per la Scienza e la Tecnologia dei Plasmi, CNR, Padova, Italy
[5]Centro di Ateneo "Centro Ricerca e Fusione", Padova, Italy
[6]See H. Reimerdes et al 2022 Nucl. Fusion 62 042018 (`https://doi.org/10.1088/1741-4326/ac369b`) for the TCV team

In the last few decades, experiments on several medium-size tokamaks (TCV [1], DIII-D [2] and AUG [3]) have shown that giving the plasma a Negative Triangularity (NT) poloidal crosssectional shape can cause a reduction of turbulent transport with respect to Positive Triangularity (PT) plasmas, while also inhibiting the transition to H-mode. This allows a NT plasma to have an L-mode-like edge pressure profile and H-mode-like core pressure level. This is a promising configuration for a future reactor, where detrimental edge localized modes have to be avoided and high confinement times are needed.

In order to explore the feasibility and the limits of NT, the Divertor Tokamak Test (DTT) facility, a novel superconducting tokamak under construction in Italy, is also considering a Negative Triangularity option for the full power scenario. Within an extensive framework of preliminary studies that involve both numerical modelling and experiments, during the 2022/2023 EUROfusion WPTE campaign two experimental sessions on TCV have been dedicated to testing the feasibility of such a scenario in DTT and tuning the parameters in order to optimize it.

The reference PT and NT magnetic equilibria envisioned for DTT have been reproduced in TCV and three different heating mixes, i.e. NBI, NBI/ECRH and ECRH, have been applied in order to access different turbulent regimes. In order to perform a thorough comparison, within a fixed heating mix three scenarios have been considered: a NT-PT L-mode pair with the same injected power and a high power PT H-mode scenario.

Independently of the heating mix, NT L-mode discharges proved to perform much better compared to PT L-mode ones with the same heating power and were also able to reach the central values of thermal pressure similar to those of PT H-mode shots. Looking at the logarithmic gradients of the temperature and density profiles, it can be seen clearly that all the beneficial influence of NT is limited to a radial interval $\rho tor = [0.8 - 1.0]$, where the gradients are indeed very large and where the absolute value of triangularity is still sufficiently high. This suggests that the reduction of turbulent transport in a NT plasma is not strongly affected by the nature of the turbulent regime, i.e. whether it is ITG or TEM dominated. Finally, preliminary predictive simulations performed with ASTRA-TGLF [4, 5] and local gyrokinetic ones performed with GENE [6] at a fixed radial location ($\rho tor = 0.85$) seem to be able to reproduce the beneficial effect of NT on confinement. More extensive gyrokinetic simulations are still ongoing and will be compared with previous predictive simulations performed for DTT by the authors.

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**Progress in understanding the impact of the magnetic geometry on divertor turbulence**

D.S. Oliveira[1], C. Theiler[1], O. Février [1], D. Galassi[1], S. Gorno[1], K. Lee[1], K. Lim[1], D. Mancini[1], G. Legoupil[1], P. Ricci[1], Y. Wang[1], C. Wuethrich[1], the TCV Team[a], and EUROfusion MST1 Team[b]
[1]École Polytechnique Fédérale de Lausanne, Swiss Plasma Center, Lausanne, Switzerland
[a] *See the author list of H. Reimerdes et all 2022 Nucl. Fusion, 62 042018*
[b] *See the author list of B. Labit et all 2019 Nucl. Fusion 59 086020*

An outstanding challenge in magnetic confinement fusion is to limit the heat flux reaching the vacuum vessel walls below material constraints while maintaining the necessary core plasma performance. For the power exhaust handling in DEMO, alternative divertor configurations (ADCs) are being considered as a possible substitute for the baseline ITER solution, since it is currently uncertain whether a detached conventional single-null plasma will meet divertor and core performance requirements in a fusion power plant [1]. ADCs are typically characterized by increased parallel connection length and plasma wetted area, which, among other benefits, facilitate access to detachment and actively reduce the peak heat flux at the outer target [2]. Recent works also highlighted the importance of divertor turbulence in exhaust solutions [3] and its capability in broadening divertor profiles [4,5]. These findings put into perspective the benefits of divertor turbulent transport for the power exhaust in ADCs, an open question whose first-principle basis has yet to be established [2].

In this contribution, we present the progress in understanding the impact of the magnetic geometry on divertor turbulence in an approach that combines experiments in TCV with simulations carried out with GBS, a 3D flux-driven, global, two-fluid turbulence code including a kinetic model for neutrals [6]. By varying the plasma current (and thus parallel connection length) and poloidal flux expansion, and benefiting from the X-point GPI diagnostic system [7] and the reciprocating divertor probe array installed in TCV [8], we assess the contribution of turbulence on divertor profiles and its main dependencies on these two magnetic geometry parameters. Detailed particle and power balances are performed in the simulations to unveil the contribution of steady-state drifts and cross-field turbulence to the transport in the X-point region. Preliminary results show that the ExB drifts are more important in the near SOL while turbulent transport is dominant in the far-SOL. The dynamics of turbulent filaments are also investigated in the simulations using a tracking algorithm and a synthetic GPI diagnostic. These analyzes show the presence of divertor-localized filaments, turbulence structures in the divertor that are disconnected from the upstream plasma, in the same region where a broadening of the divertor profiles occurs, similar to experimental findings [5]. We also perform comparisons of the SOL width in our experiments and simulations against scaling laws [9,10], discussing the level of agreement in terms of the underlying transport mechanisms.

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# Progress towards the understanding of negative triangularity improvements with gyrokinetic simulations

**Giovanni Di Giannatale[1], Stephan Brunner[1], Moahan Murugappan[1], Laurent Villard[1]**
[1] Ècole Polytechnique Féedérale de Lausanne (EPFL), Swiss Plasma Center (SPC), CH-1015 Lausanne, Switzerland

**Abstract.**
On the road to fusion energy production, many alternative scenarios have been investigated in order to address certain well-known problems of tokamak devices; among them, anomalous transport, ELMs, and disruptions. The studies on plasma shaping fall into this effort. Understanding and predicting this phenomenon is a key issue on the way to future fusion reactors. The understanding of turbulent transport reduction in negative triangularity (NT) with respect to positive triangularity (PT) has made many advances in the last decade thanks to dedicated experimental campaigns, analytical modeling, and huge efforts in numerical simulations. Nevertheless, a clear and definitive answer is still missing. Non-linear and non-local effects must be considered when attempting to study NT improvements. Linear analysis does not show significant differences in global codes, and beneficial effects of triangularity are deep inside, where triangularity is substantially very small.

Here, we focus on global effects that can only be observed when the full torus is simulated. We show that in global simulations, the Fick law is often a wrong approximation, especially in flux-driven simulations, where the source triggers some non-local effects. Using the global flux-driven ORB5 code and employing a model with collisions and kinetic electrons, we focus on two TCV shots with positive and negative triangularities.

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# Ultra long turbulent eddies, magnetic topology, and the triggering of internal transport barriers

Arnas Volčokas, Justin Ball and Stephan Brunner

École Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), CH-1015 Lausanne, Switzerland

E-mail: Arnas.Volcokas@epfl.ch

In this work, we use local nonlinear gyrokinetic simulations of tokamaks to demonstrate that turbulent eddies can extend along magnetic field lines for hundreds of poloidal turns when the magnetic shear $\hat{s}$ is very weak or zero [1]. Their length is limited only by critical balance — the distance that electrons can travel along the field line within the lifetime of a turbulent eddy. Such "ultra long" eddies can have significant consequences on turbulent transport due to parallel self-interaction as individual ultra long eddies will often "bite their own tail" [2]. Moreover, it makes correctly treating the field line topology, in particular whether a flux surface has a safety factor that is integer, rational, near rational, or irrational, all the more important, which is accomplished by carefully choosing the simulation domain length as well as the phase factor in the parallel boundary condition for $\hat{s} = 0$ simulations. To this end, we will show that field line topology can cause transitions between different turbulent modes and completely stabilize Ion Temperature Gradient (ITG) turbulence, both linearly and nonlinearly. Using Floquet-Bloch theory, we show how linear results from a domain that is one poloidal turn long can be used to calculate growth rates for any number of poloidal turns. Empirically, very weak or zero $\hat{s}$ has been identified as being one of the key conditions for facilitating Internal Transport Barriers (ITBs) [3]. We present low magnetic local gyrokinetic simulations that exhibit weak ITBs caused by the magnetic topology, which may inform a long-standing experimental observation that it is often easier to trigger ITBs where the safety factor has a low-order rational value [4]. Furthermore, we found that as the magnetic shear is lowered, the parallel eddy length scales like like $1/\hat{s}$ (up to a minimum value of magnetic shear), as a result of balance between diamagnetic and magnetic drift frequencies driving ITG modes. Lastly, we observe a novel physical effect termed "poloidal eddy squeezing" — when eddies become ultra long they can cover the full flux surface and, for specific values of the safety factor, strongly interact with themselves in the perpendicular direction. This can squeeze them, reducing their perpendicular size and ability to transport energy, thereby embodying an intriguing new strategy to improve confinement in tokamaks.

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# Quasilinear Gyrokinetic Modeling of Reduced Transport in the Presence of High Impurity Content, Large Gradients, and Large Geometric Alpha

C. D. Stephens[1], D. R. Hatch[1], M. Kotschenreuther[1], S. M. Mahajan[1], J. Citrin[2,3], C. Bourdelle[4],

[1]University of Texas at Austin, TX 78712-1192, United States of America
[2]DIFFER - Dutch Institute for Fundamental Energy Research, Eindhoven, The Netherlands
[3]Science and Technology of Nuclear Fusion Group, Eindhoven University of Technology, Eindhoven, The Netherlands
[4]CEA, IRFM, F-13108 Saint-Paul-lez-Durance, Franc

Transport barriers in tokamak discharges are often characterized by large gradients that can destabilize electrostatic microinstabilities, thereby driving anomalous turbulent transport [1]. However, large gradients can also lead to large geometric α_MHD, a stabilizing parameter in certain regimes [2]. The resulting transport is inherently constrained to be ambipolar; in effect, these large gradients can make this flux constraint impossible to satisfy, resulting in stabilization and the reduction of turbulent transport [3]. Due to the high computational cost of nonlinear gyrokinetic simulations, using a reduced turbulent transport model is ideal for predictive modeling. However, reduced models tailored for the tokamak core can become unreliable in transport barrier regimes, thus necessitating model development and improvement. We test the extent to which the gyrokinetic quasilinear code QuaLiKiz [4] can reliably predict anomalous transport in transport barrier discharge regimes to determine parameters that lead to turbulent transport reduction. We use the gyrokinetic code GENE [5], based on first principles, as a point of comparison for QuaLiKiz. Unlike GENE, QuaLiKiz uses many approximations to ensure computational tractability. In particular, QuaLiKiz assumes a Gaussian eigenfunction, uses s-α_MHD geometry, and only captures electrostatic fluctuations. To ensure accurate predictions in transport barrier discharge scenarios, we improve the approximations made for trapped particles, and thus the trapped electron mode (TEM), by incorporating the bounce-averaged electrostatic eigenfunction [6, 7]. The Gaussian ansatz allows us to analytically estimate this bounce-averaging effect with sufficient accuracy. We also improve the approximate methods used to solve for the mode structure in order to accurately calculate bounce-averaging effects.

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# Limiting factors for achieving peeling-limited pedestals in present devices

T.M. Wilks[1], T. Osborne[2], M. Knolker[2], P.B. Snyder[3], F.M. Laggner[4], M. Shafer[3], J.W. Hughes[1]

[1]Massachusetts Institute of Technology, Plasma Science and Fusion Center
[2]General Atomics
[3]Oak Ridge National Laboratory
[4]North Carolina State University

An optimized pedestal regime called the Super-H Mode (SH-mode) is leveraged to access peeling limited pedestals and study limitations on the integrated tokamak exhaust and performance (ITEP) gap. Analysis of DIII-D experiments demonstrates separatrix and mid-pedestal collisionality as key parameters in determining the state of both the pedestal (peeling vs ballooning dominant instabilities) and the divertor (attached/detached) regions. Simulations predict future devices such as SPARC and ITER will have pedestals limited by peeling instabilities [1-2], which are low toroidal mode number and driven by high pedestal bootstrap current and associated with low pedestal collisionality. Simultaneously, the requirement for a detached divertor necessitates high density and low temperature (and therefore high collisionality) in the divertor and scrape-off layer region to reduce the heat flux to the target plate, creating an ITEP trade-off. Experiments attempting to detach the divertor fuel and seed impurities in the divertor region, which increases the collisionality in both the pedestal and divertor regions. Bootstrap current decreases with collisionality, and therefore reduces the drive for peeling modes. Given a constant shape, collisionality in the mid-pedestal and separatrix regions emerge as a key parameters that set the boundary conditions impacting simultaneous optimization of pedestal stability and detachment in present devices, since pedestal and divertor collisionalities cannot be decoupled as expected in future devices.

Experiments were performed on DIII-D (Bt=2.1-2.2T, Ip=1.4-2.0MA, $P_{NBI}$=11-14MW) to determine the conditions where peeling and ballooning modes become strongly coupled. Primary actuators for these experiments included $D_2$ puffing to match pedestal density in semi-open and closed divertors as well as varied strike point location to impact the pumping efficiency in the divertor. Pedestal structure is modified by fueling and pumping such that the closed divertor requires higher gas puffing for a matched pedestal density due to the higher pumping efficiency. Therefore, the closed divertor configuration exhibits a steeper density gradient and lower scrape-off layer collisionality compared to the semi-open divertor configuration. Due to ~2x increased pumping efficiency in the closed divertor configuration, approximately 5x higher fueling rates were required to achieve similar pedestal conditions as the open divertor. The difference in pumping efficiency (~100 TorrL/s vs ~20 TorrL/s) and fueling for the same pedestal density indicates a change in pedestal transport, which leads to ~5x lower collisionalities at the separatrix and ~10x lower at mid-pedestal, $\psi_N = 0.96$, allowing access to peeling-limited pedestals.

Topical area: Core edge integration

Presenter email: twilks@psfc.mit.edu

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# Predictive modeling of Super-H mode in DIII-D using the TRANSP integrated modeling code

A.Y. Pankin[1], A. Bortolon[1], M. Gorelenkova[1],
F.M. Poli[1], T. Rafiq[2], M.W. Shafer[3], T.M. Wilks[4], Z.A. Xing[5]

[1]Princeton Plasma Physics Laboratory, Princeton, NJ, USA
[2]Lehigh University, Bethlehem, PA, USA
[3]Oak Ridge National Laboratory, Oak Ridge, TN, USA
[4]Massachusetts Institute of Technology, Cambidge, MA, USA
[5]General Atomics, La Jolla, CA, USA

The Super-H mode (SH-mode) is a good candidate for a stress test of theory-based models because the SH-mode discharges typically operate in unusual engineering parameters regimes that have not been intensively validated with theory-based models before. These regimes include high densities, high pedestal pressure, and strongly shaped plasmas. Accessing the SH mode requires dynamic control. Predictive modeling of this regime is a challenging task. The TRANSP code [1] includes a range of models for anomalous, neoclassical transport, sources, and H-mode pedestal with different fidelity levels. The DIII-D discharge 174783 is selected for this validation task. For the modeling of the H-mode pedestal in this research, we use the EPED-NN model [2], which is a neural network version of the EPED model. The updated version of EPED-NN that supports Super-H mode discharges is installed in TRANSP for this task. We initialize our simulation using the kinetic equilibrium EFIT reconstructions using the CAKE module [3] in OMFIT and plasma profiles generated by the Osborne module in OMFIT. The equilibrium is advanced in the TRANSP simulation in the TEQ module. The plasma profiles are advanced in the PT_SOLVER transport solver in TRANSP using different combinations of anomalous and neoclassical transport models. The TGLF and MMM models for anomalous transport, and the Chang-Hinton and NCLASS models for neoclassical transport are validated in this research. The neutral beam heating, NB particle source, and torque are simulated using the NUBEAM Monte-Carlo module [4]. The agreement of plasma profile predictions against experimental observation with different transport models is assessed. In particular, it is found that the simulation with TGLF and Chang-Hinton models for the electron and ion temperature profiles can reproduce the dynamics of the transition from the regular H-mode to the SH-mode reasonably well, but this combination of transport models results in somewhat underpredicted temperature profiles.

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# Role of neutral particles on pedestal structure for H-mode experiments in DIII-D

J. Balbin-Arias[1], S. Mordijck[1], T. Wilks[2] , L. Horvath[3], T. Odstrcil[4], R.Chaban[1],A. M. Rosenthal[2], J. W. Hughes[2], A. Bortolon[3], F. M. Laggner[5]

[1]William & Mary, Williamsburg, USA
[2]MIT Plasma Science and Fusion Center, Massachusetts, USA
[3]Princeton Plasma Physics Laboratory, Princeton, New Jersey 08543, USA
[4]General Atomics, San Diego, CA 92186-5608, United States of America
[5]Department of Nuclear Engineering, North Carolina State University, Raleigh, NC 27607, USA

We present a database study investigating the role of fueling versus plasma parameters in determining the edge electron density for DIII-D H-mode plasmas.

The database consists of multiple stable equilibrium (100-200ms stationary) lower single null ELMy H-mode discharges with engineering parameters ranging from toroidal magnetic field $B_T$ in [1.2, 2.2] (T), plasma current $I_P$ in [1.0, 1.8] (MA), Neutral Beam power $P_{NBI}$ in [1, 13] (MW) and safety factor $q_{95}$ in [3.0, 6.5]. Only data in 50-98% of the ELM cycle is included and the separatrix location is determined based on power balance techniques [1,2]. Previous research on AUG was able to link the divertor pressure directly to the upstream electron density using a 2-point model [3]. In our analysis, we will be able to directly include neutral density and ionization profiles from the LLAMA diagnostic [4]. The profiles of neutrals are fitted with an exponential function which allows us to link the neutral density at the separatrix directly to the pressure in the divertor, gas fueling rate as well as plasma and operational parameters. The neutral penetration depth will be compared to the width of the pedestal density [5]. These results will provide a systematical investigation of the various parameters that determine the edge electron density over a wide range of H-mode conditions in DIII-D.

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# Turbulence and Transport Dependence on $\rho^*$ and Isotope Mass in H-Mode Plasmas on DIII-D

G.R. McKee[1], C. Chrystal[2], B. Geiger[1], R. Hong[3], N. Howard[4], F. Khabanov[1], E. Perez[1], X. Qin[1], T. Rhodes[3], L. Schmitz[3], K.E. Thome[2], Z. Yan[1]

1 University of Wisconsin-Madison, Madison, WI, USA

2 General Atomics, San Diego, CA, USA

3 University of California-Los Angeles, Los Angeles, CA, USA

4 Massachusetts Institute of Technology, Cambridge, MA, USA

Normalized long-wavelength ($k_\infty \rho_i<1$) density fluctuation amplitudes are found to scale nearly linearly with the normalized local ion gyroradius, $\tilde{n}/n \sim \rho^*$, in both deuterium H-mode plasmas in which $\rho^*$ is varied while other dimensionless quantities remain nearly unchanged and in similar plasmas in which $\rho^*$ is varied by changing isotope mass in dimensionally matched hydrogen and deuterium plasmas. Interestingly, confinement is found to increase at smaller $\rho^*$ as $\rho^*$ is varied with a single ion species (deuterium), consistent with gyro-Bohm predictions, however, when $\rho^*$ is reduced by changing isotope mass (D->H), confinement degrades, counter to simple gyro-Bohm predictions, indicating that ion mass and/or the dimensionless electron-to-ion mass ratio are critically important to confinement scaling and the widely observed "isotope effect". The observed turbulence amplitude variation with $\rho^*$ is consistent with gyro-Bohm predictions of turbulent transport and to previous experimental observations in deuterium L-mode plasmas. The normalized ion gyroradius, $\rho^*=\rho_i/a$, was systematically varied in the matched deuterium plasmas while other dimensionless parameters, such as $q_{95}$, $\beta_N$, $T_e/T_i$, $\nu^*$, and $M$ (Mach #) are held nearly constant by appropriately varying plasma current, toroidal field, input power, torque and gas fueling to maintain similar normalized kinetic profiles; rotation and ExB shear was minimized using the counter-Ip NBI injection capability on DIII-D. This study is applied to ITER-like ELM'ing H-mode plasmas that exhibit $H_{98,y2}\sim1$, $\beta_N\sim2$, and $q_{95}\sim6$; $q_{95}$ and magnetic shear are higher than will be used in ITER in order to match previously obtained isotopic comparison plasmas and to minimize the impact of MHD. The $\rho^*$ deuterium discharges are similar to a set of dimensionally matched isotope-scaling plasmas that examined the impact of varying isotope mass (using hydrogen and deuterium) on turbulence and transport, allowing for comparison of plasma confinement in plasmas where the ion gyroradius is varied both by changing magnetic field as well as by varying ion mass. Comprehensive measurements of the spatiotemporal turbulence properties, such as spectra, amplitude profile, radial and poloidal correlation lengths, decorrelation times, and flow velocity, obtained with 2D Beam Emission Spectroscopy, are compared as the local ion gyroradius is altered via magnetic field and/or isotope variation. These turbulence measurements and analysis will seek to help resolve this conundrum. Transport properties and turbulence measurements in these plasmas will be compared with linear gyrokinetic simulations to help validate these models and enable more accurate projections of performance to ITER-scale plasmas that have similar dimensionless parameters except for $\rho^*$, thereby increasing confidence in extrapolation of fusion reactor performance predictions.

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# Extraction of the turbulence components and their dynamics using reflectometry data

L. Salazar[1,2], S. Heuraux[2], R. Sabot[1], J.-F. Artaud[1], C. Bourdelle[1], G. Dif-Pradalier[1], X. Garbet[1], A. Kramer-Flecken[3] and the WEST team

[1] CEA, IRFM, F-13108 Saint-Paul-Lez-Durance, France,
[2] IJL UMR 7198 CNRS, Université de Lorraine, F-54000 Nancy, France
[3] Institut für Energie- und Klimaforschung / Plasmaphysik, Forschungszentrum Jülich GmbH, D52425 Jülich, Germany

To achieve controlled nuclear fusion, it is crucial to have a deep understanding of turbulent transport as it plays a driving role in heat and particle transport at different regimes, times, and spatial scales. To reach the connection between transport and turbulence, it is necessary to discriminate the different types of turbulence, their energy exchanges and their associated dynamics.

Reflectometry diagnostic probes density fluctuations with high sensitivity and good spatial and temporal resolution. The frequency spectrum of conventional fluctuation reflectometer exhibits several components as identified separately by Vershkov [1] and Krämer-Flecken [2]: the Low frequency (LF) component, the Broadband (BB) component, the Quasi-coherent modes (QC modes) and the noise component.

The proposed approach to extract these components and their time evolution and interaction relies on information theory tools and machine-learning techniques to process the information obtained from reflectometry signal. The Continuous wavelet transform (CWT) exhibits properly the dynamics among different time scales. The first step is to eliminate irrelevant signals (high Doppler Effect and low Signal over Noise Ratio). Afterwards, various methods such as Complex Variational mode decomposition (mixed frequency tackling) and Mini-batch K-means (unsupervised learning) have been employed to perform the decomposition of the reflectometry spectrum into the QC, LF, BB and noise components.

This algorithm is fast enough for processing more than 100 000 signals taken during Ohmic phase from 3173 Tore Supra shots for statistical studies.

The QC modes are chosen for a first dynamics study. Indeed, H. Arnichand [3] showed QC modes were a signature of the Trapped Electron Mode (TEM) instabilities associated to the Linear Ohmic Confinement (LOC) regime. A statistical analysis of the QC modes probability apparition in the plasma current/ density diagram was performed for different radial positions. As expected, most of the QC modes are observed below the LOC-SOC transition threshold but two other clusters were identified, one at high plasma current- high density and the other one at very low plasma current.

For probing positions inside the $q = 1$ surface, an interaction between the QC-modes and a low frequency mode is shown, whereas for positions outside the $q = 1$ surface, no interaction is obtained. We observe that the low frequency mode shifts to higher frequencies up to QC modes frequency range during the ramp phase of the sawtooth. The energy of the QC modes increases at the sawtooth crash and then relaxes. Foreseen studies connecting local gradients, instability growth rates and dynamics are on the way.

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# Integrated modeling of WEST long pulse L-mode discharges

S.Y. Shi[1], C. Bourdelle[1], P. Manas[1], P. Maget[1], J.F. Artaud[1], R. Dumont[1], F.J. Casson[2], J. Citrin[3] and the WEST Team[4]

[1]CEA, IRFM, F-13108 Saint-Paul-Lez-Durance, France
[2]UKAEA, Abingdon, United Kingdom of Great Britain and Northern Ireland
[3]DIFFER-Dutch Institute for Fundamental Energy Research, Eindhoven, The Netherlands
[4]http://west.cea.fr/WESTteam

Combining high fusion performance and long pulse operation is one of the key integration challenges for fusion energy development in magnetic devices [1]. As one of the superconducting actively cooled devices, WEST, a large aspect ratio (A=5-6) full tungsten (W) environment tokamak is exploring long pulse operation using external current driven by Lower Hybrid (LH) waves [2]. Discharges up to about one minute were achieved using W coated actively cooled divertor in Upper Single Null configurations. Since 2022, the lower divertor has been a complete ITER grade W lower divertor, achieving more than 50 pulses lasting more than one minute with over 200MJ of injected energy, especially new records with two discharges of about 100s/300MJ.

In view of progressing both the technical and physical challenges of long pulse high performance operation, detailed physics understanding is needed to prepare real-time control aiming at maximizing the pulse length and the energy confinement time simultaneously. Towards this aim, integrated predicting modeling of current diffusion, heat and particle (including W and its radiation) of existing experiments is required.

In this work, we present WEST long pulse L-mode discharges predictive integrated modeling using the High Fidelity Pulse Simulator (HFPS) which is based on the JINTRAC suite codes using IMAS data format [3]. Five transport channels (current, electron and ion heat, main ion and impurities particles) are modeled simultaneously and predictively up to the separatrix together with magnetic equilibrium, plasma boundary shape being prescribed. Radiative losses due to W are self-consistently taken into account. Current Drive (CD) and heating sources are coherently modeled by a reduced LH model. Heat and particle turbulent transport are computed by state-of-the-art quasilinear model: TGLF or QuaLiKiz. The physics driving discrepancies between the 2 models, in particular for the particle transport, are discussed. Neoclassical transport is modeled by NCLASS.

Actuators to improve the confinement performances while maintaining a low loop voltage are investigated in order to propose future experiments in WEST on the ITER grade lower divertor. In particular, the impact of pellet fueling vs gas fueling is explored, thanks to the code HPI2 coupled to HFPS with self-consistent LHCD and heating, turbulent heat and particle transport, current diffusion and W radiations.

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# Nonlinear gyrokinetic simulations of boron density peaking: experimental comparisons and reduced transport model validation

P. Manas[1], C. Angioni[2], R. M. McDermott[2] and the ASDEX Upgrade Team[3]

[1]CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France
[2]Max-Planck Institut für Plasmaphysik, Garching, Germany
[3]see author list of U. Stroth et al 2022 Nucl. Fusion 62 042006

Accurately predicting turbulent light impurity transport in tokamaks has been a challenge for gyrokinetic codes, e.g. [1,2] and remains as such, in some regimes where the impurity profiles are hollow [2]. Furthermore, impurity transport modelling using reduced transport models is becoming an important ingredient in integrated modelling frameworks due to additional nonlinearities involved via light and heavy impurity profiles evolution and is well illustrated e.g. in current ramp-up modelling [3].

In this context, a set of nonlinear gyrokinetic simulations with GKW [4] have been performed to investigate possible discrepancies with the quasilinear approach applied in [2], in particular in regimes of mixed ion and electron heating and also in regimes where fast ions and electromagnetic effects have a non-negligible impact on the predicted heat fluxes. Starting from the experimental database presented in [2], ~20 nonlinear simulations have been performed (including neutral beam fast ions [5], electromagnetic effects and toroidal rotation), spanning a large range of parameters with, e.g., the normalized ion temperature gradient $R/L_{T_i}$ going from 3 to 9 and corresponding boron peaking $R/L_{n_B}$ from 2 (peaked profile) to -2 (hollow profile). Most of the simulations have been done at the normalized toroidal magnetic flux coordinate $\rho = 0.5$ and $\rho = 0.4$. Systematic comparisons with the electron and ion heat fluxes obtained from power balance analysis and with the experimental boron diffusion and convective coefficients are carried out.

It is found that $Q_e$ and $Q_i$ are well reproduced together with $R/L_{n_B}$ and in particular the trends when moving from dominant electron to ion heating. On the other hand, the boron convective coefficients are underestimated while the diffusion coefficients are overpredicted similarly to what was found in [2] using the quasilinear approach and power balance heat fluxes as a renormalization. These discrepancies on the diffusive coefficients are also less pronounced when moving towards higher NBI heating and higher $R/L_{T_i}$, while the convection direction becomes mispredicted. In these regimes it is also shown that $E \times B$ shearing has to be included to reproduce experimental levels of the ion and electron heat fluxes but has marginal impact on the boron peaking and corresponding transport coefficients.

Finally comparisons between nonlinear GKW simulations and quasilinear GKW/QuaLiKiz simulations have also been performed. While quasilinear and nonlinear GKW simulations are giving quantitatively similar results for the heat fluxes ratios and the boron peaking factor, the reduced gyrokinetic model QuaLiKiz [6], is found to overestimate the local $R/L_{n_B}$ and underpredict the electron heat flux. Further characterization of the quasilinear spectra, fluctuations amplitude and phase are made possible on a wide range of plasma parameters thanks to this nonlinear vs quasilinear comparisons for heat and particle transport.

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# Transport and zonal flows dynamics in flux-driven interchange and drift waves turbulence

O. Panico[1,2], Y. Sarazin[1], P. Hennequin[1], Ö. Gürcan[1],
R. Bigué[2], G. Dif-Pradalier[2], X. Garbet[2,3], Y. Munschy[2]
[1] LPP, Ecole Polytechnique, CNRS, 91128 Palaiseau, France
[2] CEA, IRFM, F-13108 Saint-Paul-Lez-Durance, France
[3] School of Phys. and Math. Sciences, Nanyang Technological University, 637371 Singapore

The saturation of heat and particle turbulent transport in tokamak plasmas is efficiently controlled by large scale sheared flows. While collisions govern the linear damping of these flows, nonlinear couplings of turbulent fluctuations provide source terms via Reynolds' forces [1]. Numerical simulations [2] later confirmed by experimental measurements [3] have shown that self-generated zonal flows (ZF) can structure in so-called staircases. The mechanisms of their generation, their impact on turbulent transport and their robustness with respect to the various types of turbulence remain active research topics.

In the present work, these issues are addressed by means of the reduced nonlinear model Tokam1D that features interchange and drift-wave turbulences, both suspected to be active at the edge of tokamak plasmas [4]. The model derives from the continuity and charge balance equations, where single poloidal and parallel wave numbers are retained and constant ion and electron temperatures are assumed. A generalized Ohm's law closes the system, linking the parallel current to the electric field and the electron pressure gradient. One of the strengths of this 1-dimensional model is to be flux driven: it evolves self-consistently the equilibrium and fluctuations of density and electric potential. It allows one to study the generation and structuration of large scale flows as well as their impact on turbulent transport.

The linear properties of both instabilities are controlled by two plasma parameters, the mean curvature g of the magnetic field and the adiabaticity parameter C that scales like the square of the parallel wave vector divided by the electron-ion collision frequency. They exhibit rich characteristics in the parameter space. Consistently with previous findings, all the three control plasma parameters – g, C and the ion to electron temperature ratio $\tau=T_i/T_e$ – are found to have a dual role, either stabilizing or destabilizing depending on the parameter regime. Also, they govern the phase shift between the density and electric potential fluctuations, hence the efficiency of the quasi-linear transport at prescribed fluctuation magnitude.

The generation and structuration of ZFs and their interplay with turbulence and transport are analyzed in nonlinear simulations on confinement timescales. Whatever the values of the scanned parameters g, C and $\tau$, ZFs are always active. They are driven by both components of the Reynolds stress, electric and diamagnetic [5], the contribution of the former being dominant when interchange dominates (large g). Two regimes are observed: ZFs are either structured in staircases or not. Staircases are found to emerge as a result of an anti-diffusive process. While maxima of the density gradient and of the shear of ZFs coincide, the ZF curvature governs the cross phase between density and electric potential fluctuations.

These results help to characterize the large scale flow dynamics and their efficiency in regulating turbulent transport, and to discriminate plasma regimes where staircases are likely to be observed experimentally.

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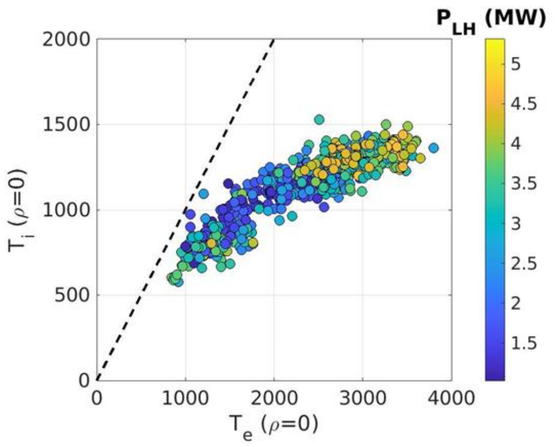
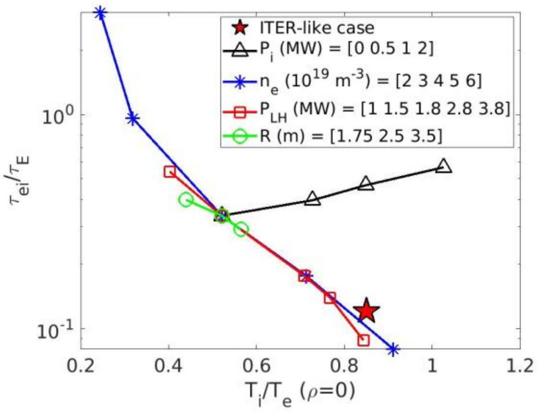
# Maximizing the ion temperature in an electron heated plasma: from WEST towards larger devices

P. Manas[1], J.-F. Artaud[1], C. Bourdelle[1], V. Ostuni[1], J. Morales[1], J. Citrin[2] and the WEST Team[3]

[1]CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France
[2]DIFFER, PO Box 6336, 5600 HH Eindhoven, The Netherlands
[3]see http://west.cea.fr/WESTteam

In predominantly electron heated plasmas, as the power increases, it is experimentally reported that the ion temperature ($T_i$) saturates while the electron temperature ($T_e$) increases with power [1]. As on AUG, W7X and elsewhere, saturates around 1.5 keV in WEST L-mode electron heated plasmas while reaches 4 keV. An entire WEST campaign of LHCD L-mode plasmas is successfully modelled using the integrated modelling METIS framework [2]. In METIS the collisional equipartition is modelled as well as the turbulent heat transport using the neural network regression of the gyrokinetic code QuaLiKiz [3]. The observed saturation is well reproduced by the modelling framework. As on AUG and W7X, the saturation correlates with a low ratio of the energy confinement to the volume averaged electron-ion collisional characteristic time. Similarly, to [4], we explore the role of the various players leading to enhanced coupling of ions and electrons. We therefore demonstrate that saturation in electron heated plasma is not expected in larger devices (as found in physics based integrated modelling) where the equipartition time is shorter than the energy confinement time.




Left figure: Central ion temperature inferred from D-D neutron rate against central electron temperature from ECE. The database corresponds to the C4 and C5 WEST campaigns altogether with injected power above one MW (LHCD heating only) and for a plasma current of 0.5 MA, a magnetic field of 3.7 T, D only plasmas. Right figure: All results of the different scans performed are reported in this figure showing the ratio of the collisional equipartition rate over the global energy confinement time versus the ratio of central ion to electron temperature. An ITER-like case is also highlighted.

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# Modeling of WEST plasmas with reduced Lower-Hybrid model: interplay with transport and sensitivity analysis

T. Fonghetti[1], P. Manas[1], R. Dumont[1], C. Bourdelle[1], J.-F. Artaud[1], Y. Peysson[1], A. Ho[2], A. E. Järvinen[3,4], E. Amnell[3], A. Kit[3], S. Smeets[5], V. Azizi[5] and the WEST Team[6]

[1]CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France
[2]DIFFER–Dutch Institute for Fundamental Energy Research, De Zaale 20, 5612 AJ Eindhoven, the Netherlands
[3]University of Helsinki, FI-00014 Helsinki, Finland
[4]VTT Technical Research Centre of Finland, FI-02044 VTT, Finland
[5]Netherlands eScience Center, Science Park 402, 1098 XG, Amsterdam, The Netherlands
[6]see http://west.cea.fr/WESTteam

**Abstract**.

Tokamak integrated modeling is crucial for the successful operation of ITER and future reactors. The integration of different physics modules into a single framework enables the self-consistent evolution of a simulated plasma, making computational time a key factor. If surrogate transport models like QuaLiKiz Neural Network (QLKNN) [1] accelerate the precise calculation of turbulent transport, the development of a fast and accurate Lower-Hybrid and Current Drive model remains a challenge. In the full W-environment WEST Tokamak, competition between radiated power and lower-hybrid heating in the core region is an issue [2]. Two different regimes are observed, depending on plasma conditions: a hot and well-confined plasma where the LH power dominates in the core region and a cold and badly confined plasma, with high central radiated power and prone to MHD activity [3]. The current work is aiming at validating a previously developed reduced LH model [4, 5] on a wide range of plasma parameters, and at exploring the sensitivity of the LH heat deposition in these two regimes. To do so, the reduced model is compared to a large number of ray-tracing/Fokker-Planck simulations using C3PO/LUKE [6]. Then, it is used in the integrated modeling framework METIS [5] coupled to QLKNN for the turbulent transport computations. This allows the investigation of the interplay between transport, radiated power and LH power deposition. While LH absorption and current driven in the reduced model are strongly dependent on safety factor and electron temperature profiles, they also strongly infer on the evolution of these quantities. At the same time, transport and confinement properties are determined from the plasma profiles. These cross-dependences leading to non-linear evolution of the electron temperature profiles are characterized and found to result in underpredictions of the central electron temperature. Finally, the LH model is coupled to the High Fidelity Pulse Simulator [7] to validate these cross-dependences using different transport codes such as the stand-alone version of QuaLiKiz [8] and TGLF [9]. Also, the sensitivity to edge profiles and LHCD input parameters is analyzed using duqtools [10] and the reduced model is optimized using Bayesian techniques [11].

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# Fidelity of Model Reduction: Implications of Near Marginality

Lessons learnt from (i) quasilinear, nonlinear gyrokinetic (ii) gradient- & (iii) flux-driven simulations

C. Gillot[1], G. Dif-Pradalier[1], Y. Sarazin[1], C. Bourdelle[1], Y. Camenen[2], J. Citrin[3], Ph. Ghendrih[1], X. Garbet[1,4], V. Grandgirard[1], P. Manas[1], F. Widmer[2]

[1] CEA, IRFM, F-13108 Saint-Paul-Lez-Durance, France
[2] Aix Marseille Université, CNRS, PIIM UMR 7345, Marseille, 13397, France
[3] DIFFER-Dutch Institute for Fundamental Energy Research, Eindhoven, Netherlands
[4] School of Phys. and Math. Sciences, Nanyang Technological University, 637371 Singapore

Quasilinear (QL) and nonlinear gradient-driven approaches are commonly used to predict turbulent fluxes in the open system made of tokamak plasmas. In this work, their validity is tested against flux-driven turbulent transport in magnetized plasma turbulence, characterized by less simplifying assumptions in the sense that fluctuating and equilibrium – flux-surface averaged – quantities are treated on an equal footing, without any scale separation. QL simulations are performed with QuaLiKiz – both in its standalone local version and in the flux-driven framework [1] – while local gradient-driven turbulent simulations use the gyrokinetic code GKW [2]. Local gradient-driven GENE simulations coupled to the transport framework TANGO are also added to the discussion [3]. These are compared with GYSELA flux-driven turbulent simulations [4].

Two distinct regimes are investigated in ITG (Ion Temperature Gradient) turbulence with Boltzmann electron response: the turbulence is either strongly driven or near-marginal. The distinction comes from the departure with respect to both linear and nonlinear thresholds for turbulence onset. The latter case is reported to feature complex turbulence self-organization at meso-scales, made of avalanches and zonal flow patterns [5].

It is found that linear predictions regarding phase shifts and amplitude ratio between fluctuating fields, central to the QL framework, retain reasonable validity in turbulent regimes for both cases, although several estimates of the Kubo number yield values around unity. This result is encouraging news for model reduction. Near-marginal regimes however pose another challenge: a significant heat flux is found to be carried below linear stability threshold in flux-driven computations, a property hampered in gradient-driven or QL approaches which postulate scale separation. It results in a significant flux mismatch between modelling frameworks, still present when running flux-driven transport simulations on the basis of QL transport coefficients [6].

Directions whereby to improve reduced models are discussed, possibly through the inclusion of inhomogeneous mixing, turbulence spreading and flow patterning.

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# Structure formation in plasma turbulence with imposed flow shear

P. G. Ivanov[1], M. Barnes[1], A. A. Schekochihin[1]

[1] *Rudolf Peierls Centre for Theoretical Physics, University of Oxford, Oxford, OX1 3PU, UK*

Gyrokinetic simulations of subcritical ion-scale turbulence have shown that, in the presence of background flow shear, the near-marginal saturated state is dominated by spatially localised, long-lived structures [1, 2]. Similar 'ferdinons' (see fig. 1) were later found in much simpler cold-ion fluid models of the Dimits transition in ion-temperature-gradient-driven turbulence [3, 4]. Additionally, beam-emission-spectroscopy measurements of MAST plasma have provided experimental evidence for the existence of radially travelling, long-lived, large-amplitude perturbations near the plasma edge that are consistent with the numerically observed ferdinons [5]. Here, we present a comprehensive study of the properties of these structures in the reduced fluid model [3, 4]. In the presence of steady mean flow shear, they have an infinite lifetime (in a periodic domain) and quasi-steady shape and amplitude. They are vortex dipoles in the perturbed $E \times B$ flow, and their instantaneous radial velocity is consistent with a balance of nonlinear advection and linear energy injection. We discuss their relevance for the Dimits transition between a zonal-flow-dominated, low-transport state and fully developed turbulence.

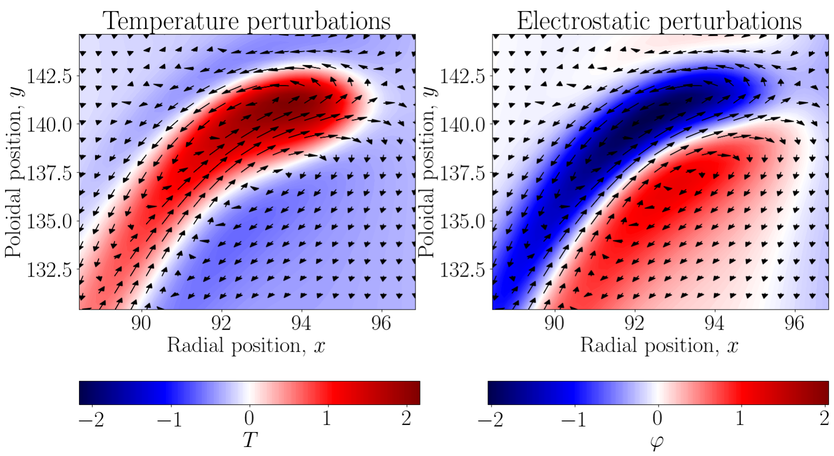


Figure 1: A 'hot' ferdinon moving radially outwards (to the right in these coordinates) in a background poloidal flow of positive flow shear. The arrows show the local perturbed $E \times B$ flow. Taken from [3].

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# Full Flux Surface $\delta f$-Gyrokinetic Code

G. Acton[1 2], S. Newton[2], and M. Barnes[3]

[1]Department of Theoretical Physics, Oxford University
[2]Culham Centre for Fusion Energy

May 2, 2023

Turbulent transport has long been understood to be the dominant transport mechanism in tokamaks. Stellarators, such as W7X, that have been optimised to reduce collisional (neoclassical) transport are also expected to be limited by turbulent transport [1] . Combined theoretical, computational, and experimental progress has advanced our understanding of turbulence properties and the resultant transport, specifically pertaining to tokamaks. In stellarators, however, the more complicated magnetic geometry gives rise to differences in turbulent behaviour [2]. In particular, the magnetic geometry is no longer replicated along each field line, but instead varies between field lines within a given flux surface in a non-trivial way. This has the consequence that the standard approach of simulating a single flux tube may be insufficient to capture the mechanisms that influence transport; zonal flows that allow for communication across multiple field lines require consideration of the turbulent evolution across an annulus encompassing the entire flux surface.

In order to address this issue computationally, it is necessary to develop an approach to treat the entire flux annulus. We have thus developed a new algorithm to confront this problem and have implemented this into the $\delta f$-gyrokinetic code stella that employs a semi-implicit treatment of electron dynamics and retains spectral accuracy in the plane perpendicular to the mean magnetic field. We will describe the new algorithm and show results from its implementation and application to a given stellarator equilibrium. The explicit Full Flux Surface version of stella with adiabatic electrons has been benchmarked against the existing global code GENE, and scans in $\rho^*$ have been performed yielding good agreement with expectation when comparing to flux tube simulations performed with stella. To illustrate the efficacy of the new approach we will then compare the explicit version of stella with kinetic electrons with the equivalent results obtained using the semi-implicit time advance. We present the numerical results obtained thus far, with the aim that such a code can aid future discussions as to the effects of zonal modes in 3D geometries.

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# Enabling online pedestal stability analysis with machine learning

A. Kit[1], A.E. Järvinen[1,2], I. Zaitsev[1], L. Chôné[1], J. Nurminen[1], V. Menkovski[3], S. Wiesen[4], L. Frassinetti[5], H. Nyström[5], M. Dunne[6] and ASDEX-Upgrade team

[1]University of Helsinki, Helsinki, Finland

[2]VTT Technical Research Centre of Finland, VTT, Finland

[3]Eindhoven University of Technology, Eindhoven, Netherlands

[4]Forschungszentrum Jülich, GmbH Institut für Energie- und Klimaforschung– Plasmaphysik, 52425 Jülich, Germany

[5]KTH Royal Institute of Technology, SE-1142B, Stockholm, Sweden

[6]Max-Planck-Institut für Plasmaphysik, Boltzmannstraße 2, D-85748 Garching, Germany

Current workflows for pedestal stability analysis require large computational times, as one must run equilibrium codes until convergence, e.g., HELENA [1], as well as MHD codes over many harmonics, e.g., MISHKA[2]. Additionally, it is difficult to conduct large-scale stability analysis across many disscharges given the large dimensionality of the data used for comparisons (kinetic profiles, equilibria, stability parameters, etc.,). Therefore, a (fast) reduced dimensional representation of the plasma state, parameterized by kinetic profiles and stability parameters, could facilitate large-scale comparison studies of the MHD tability in pedestal plasmas.

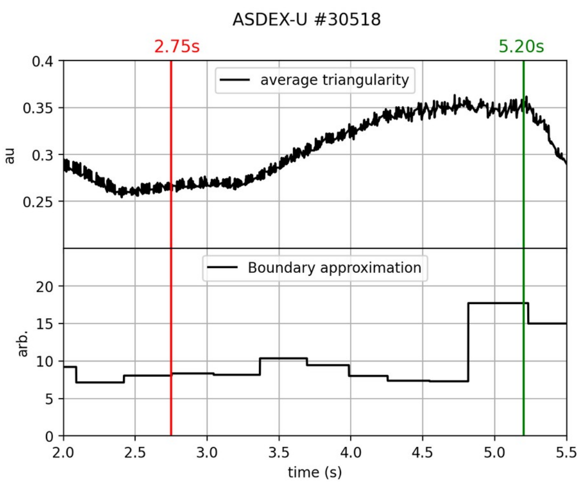


*Figure 1: Figure 1:As previously observed by M. Dunne et. al., ([8] Fig. 13), our analytical approximation also captures the outward shift of the stability boundary w.r.t to average triangularity.*

To obtain a reduced dimensional representation of the pedestal plasma state and its (approximate) MHD stability, we propose the use of Multimodal VAEs [3], which is conditioned on kinetic profiles and machine parameters (similar to our previous studies [4]), as well as first order analytical approximations of the stability boundary made using experimental data. The reduced model of the stability boundary consists of analytical expressions of the bootstrap current, $j_B$, as defined by Redl [5] and the normalized pressure gradients, $\alpha$, as defined in [6]. We then determine the stability boundary of a given time window to be the max value (in the pedestal) of the euclidean norm of $max(j_B)$ and $max(\alpha)$ within that time window, i.e., $\sqrt{max(j_b)^2 + max(\alpha)^2}$. Time windows of relatively constant machine parameters and euclidean norm are algorithmically identified using the PELT change-point algorithm [7]. This approach and reduced model shows qualitative agreement with previous stability analysis (Fig. 1).

We apply this method to over 4000 non-disruptive, deuterium H-mode pulses from ASDEX-Upgrade. It will then be investigated whether the representation model i) predicts the stability boundary given the electron profiles and machine parameters as an input and ii) encodes a lower dimensional representation for efficient stability comparison of different discharges. The use of vector databases will be explored for computationally efficient comparison studies for online analysis.

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# POSTER

## Wednesday Session

# On the validity of reduced quasi-linear transport models in the current ramp-up phase of TCV plasmas

Y. Camenen[1], M. Marin[2], P. Manas[3], C. Angioni[4], C. Bourdelle[3], J. Citrin[5], E. Fable[4], A. Ho[5], M. Lanzarone[1], P. Maget[3], A. Najlaoui[1], O. Sauter[2] and TCV team

[1] CNRS, Aix-Marseille Univ., PIIM UMR7345, Marseille, France
[2] EPFL, Swiss Plasma Center (SPC), Lausanne, Switzerland
[3] CEA, IRFM, F-13128 Saint-Paul-lez-Durance, France
[4] Max-Planck-Institut für Plasmaphysik, Boltzmannstrasse 2, D-85748 Garching, Germany
[5] DIFFER - Dutch Institute for Fundamental Energy Research, Eindhoven, Netherlands

In a tokamak plasma, the prediction of the plasma resistivity during the current ramp-up phase is essential to design scenarios that minimise the magnetic flux consumption and tailor the safety factor profile evolution to avoid deleterious MHD instabilities. The plasma resistivity directly depends on the electron temperature and the impurity content. Its prediction therefore largely relies on transport models for the electron heat flux and impurity flux (high Z impurities for the radiated power, which affects Te, and low Z impurities for the resistivity, proportional to Zeff).

The current ramp-up phase is characterised by plasmas with high Te/Ti, high q, high collisionality (low Te), low shaping and, often, a limiter configuration. For accurate predictions, transport models need to be validated in these very specific conditions.

This is the goal of the present study performed in the frame of TSVV11 activities on "Validated Frameworks for the Reliable Prediction of Plasma Performance and Operational Limits in Tokamaks". The study is focused on the ramp-up phase of TCV plasma #64965 at line averaged density, nel = $5 \times 10^{19}$ $m^{-3}$, plasma current, Ip = 320kA and safety factor q95 = 2.5 (flat-top values). Carbon impurity temperature, density and toroidal rotation were measured by charge exchange spectroscopy using TCV non-perturbative diagnostic neutral beam. Electron temperature and density were measured by Thomson Scattering.

Linear and non-linear gyrokinetic simulations have been performed with GKW for three selected time slices, t = 0.07, 0.12 and 0.17s, and two radial positions r/a = 0.5 and 0.7. In the early phase of the ramp-up, the plasma is deep in the TEM regime and progressively moves towards the ITG regime, starting first from the innermost radial locations. This is reflected in the non-linear electron to ion heat flux ratio which reaches more than 10 at r/a = 0.7 for the first time slice and drops to about 1 at r/a = 0.5 for the last time slice. Carbon ions and trace tungsten ions were included in the non-linear simulations to assess impurity transport.

The validity of quasi-linear predictions with TGLF and QuaLiKiz is assessed in this regime by comparisons of the linear response and quasi-linear estimates against the linear and non-linear GKW simulations, including impurity fluxes.

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# Can TGLF model Kinetic Ballooning Modes turbulence in the center of high-performance tokamak plasmas?

A. Najlaoui[1,3], Y. Camenen[1], C. Bourdelle[2], A. Loarte[3]
[1]Aix Marseille University, CNRS, PIIM UMR 7345, Marseille, France
[2]CEA, IRFM, Saint-Paul-lez-Durance, France
[3]ITER Organization, Route de Vinon-sur Verdon, 13067 St. Paul Lez Durance, France

A recent study [1], suggests that kinetic ballooning mode (KBM) turbulence may play a determining role in the central part of high beta plasmas. To confirm the role of KBM turbulence in experiments and explore its impact on transport and confinement in future devices, a validated reduced model suited for integrated modelling is desired.
In this study, we will present a KBM benchmark between the quasi-linear model TGLF [2] and the gyrokinetic code GKW [3], in conditions met in the core of hybrid H-modes (high plasma beta and low magnetic shear).
The benchmark was conducted using the IMAS unified standard for gyrokinetic simulations [4], involving a comparison of eigenvalues (growth rate and frequency), eigenmodes structures and cross phases for the linear response, as well as non-linear fluxes. The benchmark was first carried out for the GA standard case, and then extended to scenarios involving high plasma beta and low magnetic shear in JET plasmas.
TGLF managed to model the KBM threshold through a plasma beta scan, although the predicted growth rate value may deviate slightly from the actual value. Fig. 1.a shows an example corresponding to the GA standard case and Fig 1.b the spectrum obtained for the same case at $\beta = 1.4\%$
At low magnetic shear TGLF does not accurately reproduce the complex and extended eigenfunctions. which can result in significant discrepancies in fluxes. The growth rate is, however, reproduced with a relative error of about 15%.
Further analysis comparing the non-linear fluxes obtained with GKW to the TGLF quasi-linear prediction is presently on-going and will be presented at the meeting.

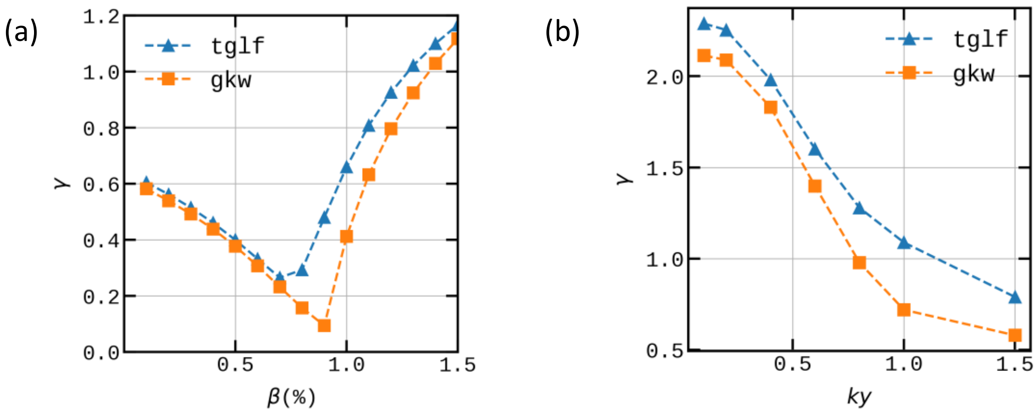


*Fig.1 Comparison of TGLF and GKW growth rate for the GA standard case. The left panel (a) shows the beta scan at ky = 0.4 and the right panel (b) the binormal wavevector scan at β = 1.4%*

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# Gyro-Kinetic DataBase project

G. Fuhr[1], Y. Camenen[1], K. L. van de Plassche[2,4], F. Almuhisen[3], C. Bourdelle[3], J. Citrin[4], M. Lanzarone[1], A. Najlaoui[1]

[1]Aix-Marseille Univ., CNRS, PIIM UMR7345, Marseille, France
[2]Ignition Computing, Keizersgracht 16D, 5611GD Eindhoven, Netherlands
[3]CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France
[4]DIFFER- Dutch Institute for Fundamental Energy Research, De Zaale 20, 5612 AJ Eindhoven, Netherlands

In recent years, a new type of neural network-based quasilinear turbulent transport model has been developed for the simulation of fusion plasmas, giving increasingly promising and fast results and allowing their use in integrated simulations [1][2]. These surrogate models are obtained by training NNs on large datasets of simulations generated with reduced quasi-linear codes like QuaLikiz [3] or TGLF[4]. While extremely powerful, this technique limits the accuracy of the surrogate model to that of the original one.

One way to further improve the capabilities of NNs based quasi-linear models is to train them on datasets generated with higher fidelity codes. For instance, the linear response of state-of-the-art gyrokinetic flux tube codes such as GKW [5] or GENE [6] could be used. Thanks to the growth of HPC resources, the generation of a dataset of a few million linear gyrokinetic simulations is now within the reach of a single research group. The size of the dataset can be further increased by mobilizing the community and collecting gyrokinetic simulations performed worldwide. To this end, we have extended the IMAS data model to include a unified standard for the inputs and outputs of gyrokinetic simulations. This standard is used to store gyrokinetic simulation results from different codes in a common database: the GyroKinetic DataBase (GKDB).

The GKDB is designed to be a repository of simulation data, a platform for code benchmarking, and a springboard for the development of fast and accurate turbulent transport models. The project is hosted and documented on GitLab (https://gitlab.com/gkdb/gkdb).

Thanks to the unified data model used for the database, quasilinear as well as linear and nonlinear simulations can be stored sharing compatible inputs and output. This offers the possibility to build fast quasi-linear models by training neural networks on the linear simulation data and to test their robustness against the non-linear simulation data.

Code comparison is always challenging due to the different normalizations and conventions used. The IMAS "gyrokinetics" standard greatly facilitates the benchmarking of codes (δf flux tube gyrokinetic simulations and/or quasilinear models) against each other. Proof of concepts of database usage including automated code benchmarks and data visualization will be presented.

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# Neural Network Surrogate for Acceleration of Gyrokinetic Codes to Compute Instability Growth Rates and Frequencies

M. Lanzarone[1], Y. Camenen[1], K. L. van de Plassche[3,5], F. Almuhisen[2], G. Fuhr[1], J. Citrin[3,4], C.Bourdelle[2]

[1]CNRS, Aix-Marseille Univ. , PIIM UMR7345, Marseille, France
[2]CEA, IRFM, F-13128 Saint-Paul-lez-Durance, France
[3]DIFFER: Dutch Institute for Fundamental Energy Research, Eindhoven, The Netherlands
[4]Science and Technology of Nuclear Fusion Group, Eindhoven University of Technology, Eindhoven, The Netherlands
[5]Ignition Computing, Keizersgracht 16D, 5611GD Eindhoven, Netherlands

Previous work [1,2] has successfully applied neural network (QLKNN) surrogates for the quasi-linear gyrokinetic simulation code QuaLiKiz [3] to predict core tokamak transport heat and particle fluxes, resulting in 3 orders of magnitude reduction in computation time for the full simulation with minimal (up to 15%, case dependent) loss of precision.

The current study aims to apply this concept using the gyrokinetic simulation code GKW, which has the advantage of being Electromagnetic and using realistically shaped equilibria whereas QuaLiKiz is electrostatic and uses s-α circular equilibria. In our case, we will develop a neural network trained to calculate instability growth rates, which allows flexibility in testing different saturation rules to ultimately calculate fluxes. Using GKW comes at the downside of heavily increased computation times which for linear simulations ranges from 1-100h as opposed to an average of 1s for QuaLiKiz. The goal of the neural network is therefore to produce results qualitatively similar to GKW simulations in a similar timeframe as QLKNN. This should effectively reduce the simulation time by up to 8 orders of magnitude while increasing the precision of predictions relative to experimental results. The first two major milestones of this study are presented.

Firstly we present the development of conversion software to convert QuaLiKiz input and output files to IMAS [5] standardised quantities. This allows the creation of a pipeline to train a neural network that accepts IMAS standardised inputs, important for later use with GKW. It also enables the use of QuaLiKiz inputs to build the pipeline which allows faster testing and validation before moving to the slower GKW simulations. This program is additionally part of a project aiming to facilitate the use of different simulations for cross validation and comparison of results by allowing for an easy conversion to and from the IMAS standard.

Secondly, the results of the neural network surrogate for QuaLiKiz are presented calculating the growth rates and frequencies of the most unstable modes. This network is trained using the IMAS standardised inputs and outputs of an existing QLK simulation dataset, converted by the aforementioned program and is an important stepping stone to the eventual goal of utilising a similar pipeline with GKW.

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# Statistical analysis of the COMPASS SOL turbulence by mean of a fast-visible camera

J. Cavalier[1,*], N. Lemoine[2], F. Brochard[2], A. Beltrami[3], S. Chouchene[2,4], A. Spir[2], L. Fevre[2], V. Weinzettl[1], J. Adamek[1], J. Svoboda[1], J. Horacek[1], M. Desecures[4] & COMPASS team[5]

[1]Institute of Plasma Physics of the CAS, Prague, Czech Republic
[2]Université de Lorraine, Institut Jean Lamour, Nancy, France
[3]Università degli Studi di Milano-Bicocca, Milan, Italy
[4]APREX Solutions, Pulligny, France
[5] See the author list in M. Hron et al. 2022 Nucl. Fusion 62 042021, https://doi.org/10.1088/1741-4326/ac301f
*cavalier@ipp.cas.cz

The interaction of turbulent plasma structures with neutrals naturally present in the COMPASS [1] (B = 1.15 T, Ip = 180 kA, diverted plasma) scrape-off-layer (SOL) has been observed by mean of a fast-visible camera (≥ 270 kfps). The light emitted from this interaction reveals a complex 3D movement of the structures, that can hardly be inferred from its projection onto the 2D camera chip plane. Assuming a constant light emissivity along field lines in the relatively narrow field-of-view of the camera, tomographic inversion [2] allowed to retrieve blob movements in a 2D poloidal (R,Z) plane of the torus. Using the TRACK software [3], individual blobs were tracked frame-by-frame and longitudinal and orthogonal velocities with respect to the magnetic flux surfaces were obtained, together with the blobs' heights, widths and orientations.

Mean (R,Z) maps and chosen histograms of these different quantities will be shown and commented. In particular, the mean longitudinal velocity map (see Fig. 1) clearly shows an inversion of the flow (shear layer) around 1 cm away from the position of the last closed flux surface, following the magnetic flux surfaces and in good agreement with reciprocating probe measurements. The structures are also found to be highly non-circular, especially in the shear region, where a ratio of about 6 is found between the longitudinal to perpendicular dimensions. Then, following the work of Tsui *et al.* [4], tentative scaling laws of the radial velocities with respect to the sizes of structures will be discussed.

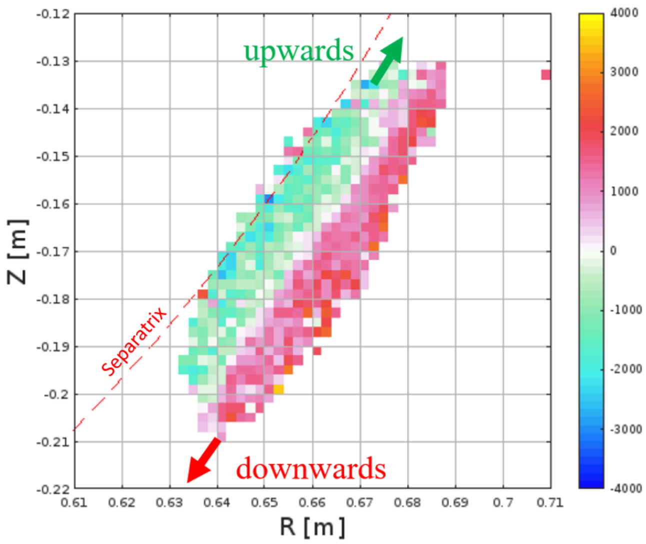


*Fig. 1: Mean longitudinal velocity map [m/s] obtained from tracking of individual structures after tomographic inversion of camera data. Inversion of flow (shear layer) is clearly visible and seems to follow the magnetic flux surface shape.*

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# Study of heat transport properties in COMPASS Upgrade scenarios

A. Casolari[1], F. Jaulmes[1], M. Farnik[1], M. Imrisek[1], E. Tomesova[1], O. Bogar[1], G. Tardini[2], E. Fable[2]

[1]*Institute of Plasma Physics of the CAS, Prague, Czech Republic*

[2]*Max-Planck-Institut für Plasmaphysik, Garching, Germany*

Additional heating in the form of NBI and ECRH is normally used in tokamaks to reach higher central temperatures, thus allowing to access H-mode and other advanced confinement modes. Central ECRH power deposition is also employed to mitigate or prevent impurity accumulation in the core, while NBI is also employed to deliver momentum to the plasma. Use of different combinations of ECRH and NBI heating was shown to affect the transport properties of the plasma [1][2]. COMPASS Upgrade (CU) will explore a variety of scenarios characterized by different magnetic field, plasma current and different combinations of NBI and ECRH heating [3]. For the design of CU scenarios we relied on the integrated transport code METIS [4] together with the free-boundary equilibrium solver FIESTA [5]: starting from these results, we carry out more advanced transport studies by employing the modular transport code ASTRA [6] coupled with different tools dedicated to dealing with the different aspects of plasma dynamics. The beam-tracing code TORBEAM [7] is used for the ECRH and the guiding-center particle code RABBIT [8] for the NBI, the turbulent transport coefficients are calculated by the gyro-Landau fluid code TGLF [9] while the neoclassical ones are calculated by the drift-kinetic code NEO [10]; the equilibrium is calculated by the fast 2D equilibrium solver SPIDER [11]. An integrated model for core plasma transport based on ASTRA and a similar set of tools has been developed in ASDEX Upgrade [12][13]. Here we employ these codes to analyze how different heating schemes affect electron and ion temperature profiles in different CU scenarios, with particular interest on the effect of central deposition of ECRH, and we try to characterize the dominant turbulent modes which contribute to heat transport. The role of plasma rotation and flow shear on turbulent mode stabilization is considered too. These studies will help us understand the physical processes that shape the equilibrium plasma quantities and design better heating strategies to achieve the planned results.

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# Scenarios for operation of COMPASS Upgrade and ITER at larger plasma current

F. Jaulmes[1], T. Markovic[1], A. Casolari[1], V. Yanovskiy[1], M. Imrisek[1], M. Komm[1], and J.-F. Artaud[2]

[1] *Institute of Plasma Physics of the CAS, Prague, Czech Republic*

[2] *CEA, IRFM, F-13108 Saint Paul Lez Durance, France*

This contribution focusses on the build-up of scenarios having lower values of the core safety factor $q$. We explore the engineering parameter space in larger values of plasma current $I_p$ and lower toroidal magnetic field $B_t$. In particular, we perform a systematic scan in $I_p$ up to values larger than what is usually applied in conventional experiments.

Prospective COMPASS Upgrade and ITER scenarios are modelled by the fast integrated code METIS [1]. METIS combines scaling laws, e.g. for global and pedestal energy with simplified transport and source models, whilst retaining fundamental nonlinear couplings. In terms of Edge Transport Barrier, we establish scenarios representative of the ELMy H-mode, the EDA H-mode [2] and the I-mode [3] using scaling laws [4].

The free-boundary equilibrium solver FIESTA yields detailed equilibrium and plasma shape. Using the Porcelli criteria [5], we describe the expected sawtooth cycle related to safety factor q=1 flux surface in the core of the resulting tokamak plasmas. In particular, we consider the operability of both ITER and COMPASS Upgrade at larger plasma current $I_p$, considering as upper boundary both possible limits $q_a$~3 and $q_a$~2.

We comment on the performance improvement, the density increase, as well as the expected modifications of the linear stability of MHD using the DCON [6] and MARS-F [7] codes. The increased severity of disruptions is considered. Possible synergies between the capabilities of the future COMPASS Upgrade and the investigation of ITER operation at larger plasma current are discussed.

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# Modeling of the COMPASS plasma SOL using GBS code

P. Macha[1,2], J. Seidl[1], D. Tskhakaya[1], D. Oliveira[3] D. Galassi[3], K. Lim[3], J. Horacek[1], J. Adamek[1], P. Bilkova[1], P. Bohm[1], P. Hacek[1], J. Krbec[1], M. Sos[1]

*1) Institute of Plasma Physics of the CAS, Prague, Czech Rep.*
*2) Faculty of Nuclear Sciences and Physical Engineering CTU in Prague, Prague, Czech Rep.*
*3) École Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SCP), CH-1015 Lausanne, Switzerland*

Understanding the dynamics of the plasma edge and the Scrape-Off Layer (SOL) is essential for optimizing the exhaustion performance in future fusion tokamak devices. These regions are responsible for maintaining a balance between plasma confinement and heat exhaust, directly impacting plasma-wall interactions and the overall energy and particle confinement. Edge and SOL dynamics are strongly influenced by turbulent transport, responsible for cross-field diffusion of heat, particles, and momentum. Consequently, the study and prediction of turbulent transport are of paramount importance. While scaling laws [1] have limited accuracy in predicting turbulent transport, and comprehensive theoretical models are lacking, numerical simulations provide valuable insights and predictions. Fluid turbulence codes, in particular, offer reduced computational demands compared to kinetic codes, while still delivering a realistic description of turbulence in the SOL. Accurate modeling of the SOL plasma is essential for designing and operating future fusion reactors like ITER and DEMO, where handling high heat fluxes and particle loads is critical.
In our work, the 3D, two-fluid, flux-driven, turbulence GBS code [2] is used to simulate the SOL of a typical discharge in the COMPASS tokamak [3]. GBS is one of the few codes in the world capable of performing self-consistent turbulence simulations in a realistic geometry and the full-tokamak size; electromagnetic effects and a kinetic model for neutral species are also implemented in the code. Recently validated against the TCVX21 scenario on TCV tokamak [4], our primary objective is to validate the GBS code against experimental data from the COMPASS tokamak.
In this contribution, we show the results of the first GBS simulation of the tokamak COMPASS. Experimentally measured radial profiles of the electron temperature, plasma potential, and ion saturation current were directly compared with the simulation. Very good agreement was observed. Based on the profiles, decay lengths were determined and compared with measured values. The electron temperature fluctuations, routinely measured on the COMPASS tokamak, were compared with GBS simulation results for the first time. Furthermore, the first comparison of turbulence properties, including the typical blob time trace provided by the conditional averaging method, probability density functions, and their moments of the electron temperature, plasma potential, and ion saturation current, was performed.

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# Validation of reduced-order turbulence modelling in the L-mode near-edge of the JET-ILW tokamak

G. Snoep[1,2], C. Bourdelle[3], J. Citrin[1,2], M.J. Pueschel[1], E. Delabie[4], A. Ho[1], M. Sertoli[5], E.R. Solano[6], P. Vincenzi[7] and JET contributors*

[1] DIFFER - Dutch Institute for Fundamental Energy Research, Eindhoven, The Netherlands
[2] Eindhoven University of Technology, Eindhoven, The Netherlands
[3] CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France
[4] Oak Ridge National Laboratory, Oak Ridge, TN37831-6169, United States of America
[5] CCFE, Culham Science Centre, Abingdon, OX14 3DB, United Kingdom of Great Britain and Northern Ireland
[6] Laboratorio Nacional de Fusion, CIEMAT, Madrid, Spain
[7] Consorzio RFX, Corso Stati Uniti 4, 35127 Padova, Italy
* See author list of J. Mailloux et al 2022 Nucl. Fusion 62 042026

The validity of reduced-order turbulence models, QuaLiKiz [1] and TGLF [2], in the L-mode near-edge was investigated using both higher fidelity gyrokinetic code GENE [3] and integrated modelling simulations JINTRAC suite [4].
The dataset is based on seven NBI-heated JET-ILW deuterium discharges with different upper triangularity and densities just prior on H mode transition [5].
For the lower densities, the dominant turbulence is in the ITG/TEM regime. In unshaped geometry QuaLiKiz and TGLF are in good agreement with GENE. But shaped geometry impact being large in these radial regions, TGLF using the Miller parameters performs best. Except for an electron direction mode that stabilises with increasing density-gradient, found in one low $\delta$ case only with GENE.
At higher densities, drift-resistive turbulence is found with GENE, consistent with previous work [6,7]. The dominant modes at low wavenumbers are driven by main ion parallel and curvature dynamics. In TGLF, these resistive modes are not as dominant as in GENE while they are not presently implemented in QuaLiKiz.
For the low density cases, heat flux driven modelling is performed with both QuaLiKiz and TGLF in the JINTRAC suite [4]. The predicted ion temperature up to the LCFS is in close agreement with measurement, while the electron temperature is underpredicted.

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# Large-scale JINTRAC validation with preliminary JET profile database

A. Ho[1], C. Bourdelle[2], Y. Camenen[3], F. J. Casson[4], J. Citrin[1], F. Koechl[5], JET Contributors*

[1] *DIFFER, De Zaale 20, 5612 AJ, Eindhoven, The Netherlands*
[2] *CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France*
[3] *CNRS, Aix-Marseille Univ., PIIM UMR7345, Marseille, France*
[4] *UKAEA, CCFE, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK*
[5] *ITER Organization, CS 90 046, 13067 St. Paul Lez Durance Cedex, France*
* *See the author list of 'Overview of JET results for optimising ITER operation' by J. Mailloux et al. 2022 Nucl. Fusion 62 042026*

To prepare ITER operation and contribute to DEMO design, a High Fidelity Pulse Simulator (HFPS) is required to evaluate the feasibility of specific plasma scenarios and trajectories within the known theoretical constraints of plasma physics. EUROfusion efforts towards this simulator are based around a Python workflow based on an IMAS-compatible version of JINTRAC (JETTO+EDGE2D) coupled to other IMAS-compatible modules, e.g. HCD. Due to the extrapolative nature of predicting ITER performance, it remains an open task to stress-test the generizability of this integrated model. As part of the TSVV-11 framework, an automated simulation setup tool was developed to reduce the undesired impact of human errors and subjectivity on the validation results. This was combined with a rudimentary classification procedure to configure the HFPS settings while accounting for best practices within the tokamak modelling community.

This work uses these tools to perform blind-execution of the JINTRAC-IMAS integrated model across a large number of experimentally-derived steady-state plasma scenarios, currently focusing exclusively on JET plasma discharges due to wide availability of raw and processed diagnostic data accessible to the EUROfusion community. The amalgamation of multiple pre-existing 0D outputs (e.g. $V_{\mathrm{loop}}$, $l_i$, $W_p$, $P_{\mathrm{rad}}$, neutron flux, etc.) and 1D outputs (e.g. $n_e$, $T_e$, $T_i$ at specified radial locations) allow the formulation of metrics to determine the 'goodness' of a given simulation. Further classification of these simulations via these metrics can potentially identify model discrepancies by their physical origins, providing a way to prioritize efforts to fill in missing physics within the simulation suite. Future work will incorporate a similar amount of data from the WEST, AUG, and TCV tokamaks to extend the validated regions and incorporate 2D line-of-sight information to increase the depth of validation possible.

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# The hunt for zonal flows and the ExB staircase through velocity field measurements in MAST-U

Y. W. Enters* [1,2], S. Thomas[2], T. Schuett[2], I. Cziegler[2]
[1]United Kingdom Atomic Energy Authority, Culham Center for Fusion Energy, Abingdon, Oxfordshire, UK
[2]York plasma institute, University of York, York, UK

Drift-wave turbulence and the associated zonal flows are ubiquitous phenomena in tokamak plasmas [1]. Zonal flows are driven primarily by the drift-wave turbulence and simultaneously act to strongly suppress the turbulent fluctuations from which they were driven. Considering that turbulence is the primary driver of anomalous radial transport in the confined region, the study of the drift-wave—zonal-flow system is crucial to the understanding, and ultimately control, of the anomalous transport [2]. Recent studies show that zonal flows self-consistently organise in a periodic pattern termed the ExB staircase [3], which is a promising step towards relating individual transport processes to global anomalous transport levels. Despite numerous indirect experimental measurements which hint at the presence of zonal flows, direct measurements remain scant [4]. As a result, zonal flow theories and related simulations currently remain unvalidated, driving the need for reliable zonal flow measurement techniques.

A recent uncertainty study of the two main image-velocimetry techniques, cross-correlation time-delay estimation (CCTDE) [5] and dynamic time-warping (DTW) [6], has broadly quantified operational limits and enabled confident velocimetry with improved inference frequencies [7]. These uncertainty tests, in combination with the advanced BES system on MAST-U, will be used to infer reliable velocity field inferences with frequencies up to ~100kHz. The velocity fields aim to resolve quasi-steady-state poloidal flows with radially periodic amplitudes. Bispectral analysis will be able to assess nonlinear energy transfer between the measured flow and the drift wave spectrum. Radial wavelengths, shearing rates, flow amplitudes, and lifetimes can be recovered from measured velocity fields, which are crucial metrics to compare with theory and simulation.

* Correspondence email: ye525@york.ac.uk

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# Nonlinear energy transfer between drift-wave turbulence and zonal flows in spherical tokamak plasmas

T.M. Schuett[1], I. Cziegler[1] and S.J. Freethy[2], D. Dickinson[1], S. Thomas[1], L. Howlett[1,2]

[1]York Plasma Institute, University of York, Heslington, YO10 5DD York, UK
[2]United Kingdom Atomic Energy Authority, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK

Turbulent transport is a key performance limiting factor of tokamak plasmas. The nonlinear interaction between drift-wave turbulence and zonal flows (DW-ZF interaction) can serve as an effective saturation mechanism with inherently reduced transport. Zonal flows do not contribute to radial transport and additionally break up radially elongated streamers, thus further reducing the radial transport. Hence, this interaction can serve as a physical explanation for the trigger of confinement transitions [1, 2].

In the fluid picture, this nonlinear interaction is usually analysed through the lens of the turbulent Reynolds-stress-mediated transfer [2, 3]. Nonlinear gyrokinetic simulations also exhibit significant DW-ZF interaction in the regime between the linear and nonlinear critical gradients also known as the Dimits regime. The nonlinear interaction in the gyrokinetic formulation captures the full phase space transfer between turbulent fluctuations and can thus serve as a kinetic extension to the Reynolds stress, offering additional insight into the underlying turbulent dynamics [4]. First preliminary results of three-wave-coupling analysis of fluid velocity transfer and free energy transfer in nonlinear simulations of ITG turbulence using the local flux tube code GS2 [5] will be shown. The analysis focuses on the impact of plasma shaping on the total magnitude and poloidal distribution of the transfers.

The analysis of the DW-ZF interaction in experiments requires $E \times B$ velocity measurements with high temporal resolution. Analysis of $E \times B$ velocities across confinement transitions inferred from density fluctuation measurements with the beam emission spectroscopy system in the spherical tokamak MAST-U will also be presented.

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# First characterisation of L-mode ion-scale turbulence on MAST-Upgrade with beam emission spectroscopy

S. Thomas[1,*], I. Cziegler[1], D. Dunai[2], A. R. Field[3], B. Patel[3], R. Scannell[3], and the MAST-Upgrade Team[3,†]

[1]York Plasma Institute, Department of Physics, University of York, YO10 5DD, UK
[2]Centre for Energy Research, Budapest, Hungary
[3]United Kingdom Atomic Energy Authority, Culham Centre for Fusion Energy, Culham Science Centre, Abingdon, Oxfordshire, OX14 3DB

[*]email: steven.thomas@york.ac.uk
[†]See author list of J. R. Harrison *et al.*, Nucl. Fusion **59**, 112011 (2019)

Turbulence plays a pivotal role in limiting confinement and performance in tokamak plasmas. It is thought to be responsible for the majority of transport. Its suppression through sheared flows can break apart larger turbulent eddies, which reduces transport, and may play a vital role in the mechanism behind the L-H transition [1]. As such, a good baseline understanding and measurement of turbulence in L-mode plasmas is required for reference in future turbulence measurements in MAST-U.

In MAST-Upgrade the beam emission spectroscopy (BES) diagnostic [2,3] is used to measure ion-scale ($k_{r,\theta}\rho_i < 1$) turbulent density fluctuations. The BES consists of an 8×8 array of avalanche photo-diodes (APDs) [4] covering an approximate 13×15cm area in the radial-poloidal plane, capable of sampling at up to 4MHz. In a series of three similar beam-heated L-mode discharges with the super-X divertor configuration [5], the BES was able to measure turbulence over the vast majority of the tokamak's minor radius ($0.15 < \Psi_N < 1.1$).

In these discharges the BES achieved signal-to-background ratios between 2 and 10. Preliminary results show measurements of relative plasma fluctuations of $\delta I/I = 2-10\%$ with correlation times, $\tau_c$, between 30-130μs. The frequency spectra increase in energy at low frequency as the measurement radii increases, as the fluctuation amplitudes increase and cover a wider frequency range. The frequency at which the slopes of the spectra changes increase towards the pedestal and plasma edge, as does the power in higher frequencies, before decreasing in the scrape-off layer. Additional analysis also characterises the turbulence through measurements of wavenumber-frequency spectra, correlation lengths, turbulence flow velocities, and non-linearity measurements.

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# Recent progress in the coordinated experimental and computational effort on flow-turbulence coupling

I. Cziegler[1], M. Barnes[3], B. Cook[1], Y. Enters[1,2], P.G. Ivanov[3], T. Schuett[1], S. Thomas[1]
[1]York Plasma Institute, University of York, Heslington, UK
[2]UK Atomic Energy Authority, Culham, UK
[3]Rudolf Peierls Centre for Theoretical Physics, University of Oxford, Oxford, UK

We report a coordinated effort of theory, improved metrics in computation and experimental diagnostics in support of understanding the complex interactions between turbulence and large-scale flows in tokamak plasmas. The turbulence-flow interaction thread goes through transport of momentum, intrinsic rotation, reduced model of zonal-flow–turbulence dynamics, new nonlinear metrics, and experimental advances on the diagnosis of equilibrium and zonal flows. The role of turbulence driven large-scale flows in the self-organisation of turbulence and thus in the quality of global confinement has extensive theoretical support, turbulence can be influenced by velocity shear and is thought to play a key role in the transport of momentum. Despite this, experimental identification of such features and the theoretical predictions of several of the above key features have been lacking in the field.
Previous experimental studies have suffered from a range of limitations from insufficient temporal resolution through sensitivity and specificity of flow velocimetry data to the restricted spatial domain in which to evaluate defining nonlinear characteristics. Several new methods have been developed for reducing the fail rate of velocimetry while optimising temporal resolution. Preliminary MAST-U results show general agreement with slower flow measurements and some key areas of much needed improvement. Taken together, tests highlight the areas in which ZF and flow-turbulence experiments have been unsatisfactory.
On the theoretical side, bistable regimes of turbulence are reported from gyrokinetic analysis, the impact of global physics on intrinsic rotation and the effect of profile curvature, a 3-field fluid ITG reduced model exhibiting emerging zonal structures and near marginal turbulence, along with recent advances in studying the poloidal structure of the nonlinear spectral transfer of energy and entropy in electrostatic an electromagnetic zonal drive.
This work was carried out in the Turbulence Dynamics of Tokamak Plasmas (TDoTP) programme, Engineering and Physical Sciences Research Council, award number EP/R034737/1.

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# Comparison of the impact of ECH and ICRH on impurity behaviour in NBI-heated LHD plasmas

N. Tamura[1, 2], M. Yoshinuma[1], K. Ida[1], C. Suzuki[1, 2], M. Goto[1, 2], T. Oishi[3], K. Mukai[1, 2], K. Tanaka[1, 4], T. Tokuzawa[1, 2], H. Funaba[1], I. Yamada[1]
[1]National Institute for Fusion Science, Toki, Japan
[2]The Graduate University for Advanced Studies (SOKENDAI), Toki, Japan
[3]Tohoku University, Sendai, Japan
[4]Kyushu University, Fukuoka, Japan

The impurity accumulation towards the plasma centre has become a severe concern in magnetic confinement fusion devices, especially in stellarator configurations. It can cause significant degradation of fusion reactor performance and lead to a radiative collapse of the discharges. Therefore, developing an efficient way of controlling the number of impurities in the core plasma is essential, especially in removing the impurities from the core plasma. In many tokamaks, it has been experimentally confirmed that RF heating, such as electron cyclotron heating (ECH) and ion cyclotron resonance heating (ICH), effectively eliminates impurity accumulation [e.g., 1, 2]. Also, in LHD, we have demonstrated the mitigation of the core impurity accumulation in high-density NBI-heated plasmas by applying an additional ECH ($P_{ECH}$ ~ 1.5 MW at $f$ = 154 GHz) [3, 4].

Here we present the impact of additional ICH on the behaviours of core impurities in NBI-heated LHD plasmas and the results of a comparison between the impact of ICH and that of ECH. In this study, to study the behaviours of low- and mid/high-Z impurities simultaneously, we have utilized the new type of TESPELs containing an inorganic compound [3] (in particular, lithium titanite ($Li_2TiO_3$: Z=3, 22 and 8), silicon hexaboride ($SiB_6$: Z=14, 5), sodium chloride (NaCl: Z = 11, 17), and calcium aluminate ($CaAl_2O_4$: Z = 20, 13 and 8)). Line emissions from the highly ionized impurities derived from the TESPEL were measured with EUV/VUV spectrometers. Further, the spatio-temporal behaviours of some of those impurities can be measured using a charge exchange spectroscopy (CXS) technique. The experiments have been performed in high-density (as a line-averaged electron density of ~ $5\times10^{19}$ m$^{-3}$) NBI-heated plasmas, where impurity accumulation has been observed. When the ICH ($P_{ICH}$ ~ 3 MW at $f$ = 38.47 MHz) was applied just after the TESPEL (tracer impurity) injection, the decay times of the intensities of line emissions from the tracer impurity ions became shorter, compared to the cases without applying the ICH. However, after the ICH was stopped, the intensities of line emissions from the tracer impurity ions were slightly recovered. The results indicate that ICH power of much more than 3 MW is required to completely suppress core impurity accumulation in high-density NBI-heated LHD plasmas. The results also suggest that much less total heating power is needed for ECH than for ICH to shorten the decay time of the intensities of line emissions from the tracer impurity ions. As you may know, the radial electric field significantly affects impurity transport in stellarator plasmas. Therefore, in this contribution, we will also discuss the impact of RF heating on the radial electric field in high-density NBI-heated plasmas.

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# A generalized gyro-averaging operator with magnetic field inhomogeneity and its implication

Hogun Jhang[1], S. S. Kim[1]
[1]Korea Institute of Fusion Energy, Daejeon, Rep. of Korea

In the description of magnetized plasma dynamics under the influence of electromagnetic fluctuations, it is convenient to study motion of the gyrating center (the gyrocenter) instead of full particle motion. The fast gyromotion around the magnetic field is averaged out in this approach. Then, a gyrocenter experiences averaged electromagnetic fluctuations during its gyration. This gyro-averaged field or potential is evaluated by a gyro-averaging operation of the physical quantity under consideration. In this work, We derive expressions for the gyro-averaging operator which are applicable to electrostatic fluctuations in the presence of a strong inhomogeneous magnetic field. A mathematical analysis has been carried out of the gyro-average operation given by the generalized expression

$$e^{-iy_c X}\left[\frac{1}{2\pi}\int_0^{2\pi} e^{-iX(\cos\alpha - y_c\cos 2\alpha + y_s \sin 2\alpha)}d\alpha\right]$$

where $X=k_\perp\rho_0$ with $\rho_0$ the local Larmor radius, $y_c=(\rho_0/4)|\nabla_\perp \ln B|\cos\gamma$, $y_s=(\rho_0/4)|\nabla_\perp B|\sin\gamma$ with $\gamma$ the angle between $\mathbf{k}_\perp$ and $\boldsymbol{\nabla}_\perp \ln B|$, and $\alpha$ is the gyro-angle. These expressions are expected to provide more accurate computation of _n, hence the potential being used in GK Poisson equation. In the low wavenumber limit, the gyro-averaging operator is shown to be represented by sums of Bessel functions with different orders. A simplified expression is provided as a Pade approximant in the low wavenumber limit. This form could be used in practical computations based on the gyrofluid formulation. In the high wavenumber limit, we make use of the method of stationary phase to compute the gyro-averaging operator. In this asymptotic calculation, we find that the operator reduces to the form,

$$\mathcal{G}=\sqrt{\frac{2}{\pi}}e^{i\pi/4}(\rho_0\nabla_\perp)^{-1/2}\left[\cos\left(X-\pi/4\right)+i2y_c\sin\left(X-\pi/4\right)\right].$$

which naturally involves fractional derivative. Discussions are made of a potential impact of this asymptotic expression in the high wavenumber limit, in particular in relation to the impact of energetic particles on micro-turbulence.

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# Studies on the effect of impurities emitted from a liquid metal divertor on the plasma core for the design of EU-DEMO

E. Bray[1], G.F. Nallo[1], F. Subba[1], C. Marchetto[2], C. Angioni[3], D. Fajardo[3], T. Luda[3]
[1]NEMO Group, Dipartimento Energia, Politecnico di Torino,
Corso Duca degli Abruzzi 24, Torino, 10129, Italy.
[2]ISC-CNR, Dipartimento Energia, Politecnico di Torino, Corso Duca degli Abruzzi 24,
Torino, 10129, Italy
[3]Max–Planck–Institut für Plasmaphysik, Boltzmannstrasse 2, D–85748 Garching, Germany

The power exhaust issue is one of the biggest challenges for tokamaks. Solutions to this problem are under study both from the engineering point of view, through the choice of plasma-facing materials and the design of highly effective cooling systems, and from the physics point of view, e.g. relying to atomic power loss processes in plasmas, which however may be harmful to the maintenance of the core plasma conditions.

One possible solution is the implementation of a Liquid Metal Divertor (LMD), which consists of a porous structure on the surface of the target where a liquid metal (typically lithium or tin) is slowly flowing, continuously replenishing the plasma-facing surface and self-shielding the target through evaporation and consequent plasma-vapour interactions.

Our group is presently performing, using the SOLPS-ITER code, studies on the behaviour of LMDs in terms of plasma-wall interaction and impact on the scrape of layer (SOL), with a view to applying the results to the design of EU-DEMO. Recently, a study started to assess the effects of this choice on the core plasma conditions.

A preliminary study on the requirements for EU-DEMO in terms of impurity density and radiated power in the core is reported in [1]. In our work, with the use of ASTRA code coupled with different, suitable modules for the calculation of main plasma and impurities transport, we study how the propagation of metallic ions produced by the LMD influences the core characteristics, focusing on the comparison between lithium and tin, and their different consequences in terms of plasma dilution and power losses.

One important part of the work is the description of the transition from SOL to core, from a geometrical before than numerical point of view. This in particular affects the coupling between ASTRA and SOLPS-ITER calculations, where we use the output of the SOL model [2] as input for the core one, employing the separatrix as the boundary and accounting for the scenario characteristics forecast for EU-DEMO [3].

We simulated the different behaviour in the core of lithium and tin, comparing it with the database of core transport evolution of metal impurities in existing experiments, and producing a comparison between the two impurities in terms of density distribution and radiated power.

This work has been carried out within the framework of the EUROfusion consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 – EUROfusion). Views and opinions expressed are however those of the authors and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

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# Progress on interaction between NTM Island and heavy impurities in AUG

C. Marchetto[1], E. Fable[2], C. Angioni[2], T. Odstrčil[3], T. Pütterich[2], S. Schmuck[4], G. Tardini[2],

and the ASDEX Upgrade Team[5]

[1] *ISC-CNR & Politecnico di Torino, Italy*

[2] *IPP Garching, Germany*

[3] *MIT, Boston, USA*

[4] *Consorzio RFX (Padova, Italy)*

[5] *See author list of H. Meyer et al. 2019 Nucl. Fusion 59 112014*

The use of Tungsten in plasma facing component in nowadays and future tokamaks is considered a solution for the problems of heat load on the walls and of Tritium retention. In some JET discharges Tungsten accumulates in the plasma core, with the possibility of leading to radiation collapse and possibly disruption. This accumulation seems exacerbated by the presence of a magnetic island.

To get an insight in this phenomenon through comparison with other machines, we model an AUG discharge where both Tungsten core accumulation and a (3,2) island are present. The underlying idea is that the onset of the mode suddenly changes the transport in its radial position by short-circuiting the region where W is present with a more central region where inward neoclassical transport dominates.

We use ASTRA (a 1.5-dimensional transport code calculating the evolution of plasma parameters in a time dependent axisymmetric MHD equilibrium configuration) with transport coefficients calculated from NCLASS and TGLF. We initialise the simulation with experimental profiles. We model the island as an enhanced diffusion zone; its radial position, onset time and width are determined from experimental data (Mirnov coils and ECE fluctuations). We compare the Tungsten SXR emissivity from the computed density (synthetic diagnostic) with the experimental SXR signals.

We will report on the simulation results and the status of the synthetic diagnostic.

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# Experiments and numerical modelling of negative triangularity ASDEX Upgrade plasmas in view of DTT scenarios

L. Aucone[1,2], P. Mantica[2], M. Bernert[3], T. Bolzonella[4], M. Dunne[3], R. Fischer[3], T. Happel[3], J. Hobirk[3], P. Innocente[4,5], A. Kappatou[3], A. Mariani[2], R. M. McDermott[3], P. Muscente[4,6], U. Plank[3], T. Pütterich[3], F. Sciortino[3], G. Tardini[3], C. F. B. Zimmermann[3] and the ASDEX Upgrade Team*

[1]Dipartimento di Fisica 'G. Occhialini', Università di Milano-Bicocca, Milano, Italy
[2]Istituto per la Scienza e Tecnologia dei Plasmi, CNR, Milano, Italy
[3]Max-Planck-Institut für Plasmaphysik, Garching, Germany
[4]Consorzio RFX (CNR, ENEA, INFN, UNIPD, Acciaierie Venete SpA), Padova, Italy
[5]Istituto per la Scienza e Tecnologia dei Plasmi, CNR, Padova, Italy
[6]Centro di Ateneo "Centro Ricerca e Fusione", Padova University, Italy
*See U. Stroth et al 2022 Nucl. Fusion 62 042006

This contribution presents experimental and modelling results of a comparison of negative (NT) and positive (PT) triangularity ASDEX Upgrade (AUG) discharges using the plasma shapes presently foreseen in the DTT tokamak, under construction in Frascati, Italy [1]. This work is part of a broader framework of investigation aiming at understanding whether the good properties observed in NT scenarios in devices like DIII-D and TCV will extrapolate to the DTT device and more in general to DEMO and fusion reactors.
Unlike in DIII-D and TCV, NT discharges in AUG tend to go into H-mode when operated in the common favourable ion ∇B drift configuration. Therefore, for the comparison both favourable and unfavourable configurations have been used [2]. The aim of the comparison with a PT H-mode has been to check whether the loss of the PT high pedestal is recovered in NT L-modes (or NT low pedestal H-modes) within a broader edge region, leading to similar core region kinetic parameters (hence fusion power in DT devices). This does not generally imply similar global confinement, given the weight of the edge volume in global parameters. Discharges with mixed NBI and ECR heating and with pure ECR heating have been compared, to study the effect of varying the ITG vs TEM relative weight, although ITG is the dominant mode in most cases.
For plasmas heated with NBI and ECR auxiliary power, the experimental results have shown that the NT geometry does not allow to recover the core performance of a PT H-mode. Instead, NT discharges with pure ECRH present logarithmic pressure gradients in the region $0.7<\rho_{tor}<1$ high enough to recover the PT thermal pressure. Predictive simulations of both PT and NT plasmas have been performed with the aim of investigating their transport properties. The simulations have been carried out using the transport code ASTRA [3] and turbulent transport model TGLF-SAT2 [4]. The transport of the main species is modelled up to the top of the pedestal in H-modes and up to the separatrix in L-modes. The impurities are predicted self-consistently up to the separatrix radius in all cases. Rotation is taken from experiment data, but not predicted. Results for both heating regimes are presented. In general, the integrated modelling reproduces well the discharges both in PT and NT configurations, including the reduced transport in NT ECH only cases. The physics understanding of the different behaviour of the NBI+ECR vs ECR plasmas is in progress, also with the help of gyrokinetic simulations.

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# Observation of quasi coherent modes in W7-X and relation to tokamaks

A. Krämer-Flecken[1], H.M. Xiang[1,2], K. Rahbarnia[3], S. Heuraux[4], R. Sabot[5], L. Salazar[4], G. Fuchert[3], J. Geiger[3], H. Thomsen[3], J. Proll[6], G. Weir[3], T. Windisch[3] and the W7-X-team

[1]Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung – Plasmaphysik,

52425 Jülich,Germany

[2]Advanced Energy Research Center, Shenzhen University, Shenzhen 518060, PCR

[3]Max Planck Institut für Plasmaphysik, Greifswald, Germany

[4]Institut Jean Lamour 7198 CNRS, Université de Lorraine, F-54000 Nancy, France

[5]CEA, IRFM, F-13108 Saint-Paul-Lez-Durance, France

[6]Technical University of Eindhoven, Eindhoven, The Netherlands

Turbulence in fusion devices is responsible to a large part for the radial transport of heat and particles. Turbulence spectra, as measured by different diagnostics, show a mixture of different turbulence components. To gain a deeper understanding of confinement dynamics in plasmas it is necessary to investigate these turbulence components. One of these components is the quasi coherent mode (QCM)[1,2]. Under certain plasma conditions/parameters QCMs are observed in the plasma. These modes are linked in general with the existence of trapped electron modes (TEM) in the tokamak plasma[3].

During the campaign OP2.1 at the stellarator W7-X, QCMs are firstly observed in the plasma core by poloidal correlation reflectometry[1]. The center mode frequency (140kHz-300kHz) and the width (50kHz-100kHz) of the QCM varies with the magnetic configuration and plasma parameters. These dependencies will be discussed in this presentation. The mode number of the QCM is obtained and together with the poloidal correlation length, which is calculated to be in the order of 20mm $k\rho_i <= 1$. This high $k\rho_i$ value supports the TEM nature of the QCM. Under certain circumstances the phase velocity of the QCMs is determined and is found in the electron diamagnetic drift direction, supporting this assumption.

As mentioned in the beginning the observation of TEMs is tokamaks [2,3,4] is known since long time. They appear in a quite broad frequency range and can be localized in the plasma core as well as in the gradient/edge region in case the electron temperature gradients are steep enough. Comparing the QCM in W7-X and Tore Supra in the plasma core, where the QCM are observed in both devices, shows that the mode frequency is a function of the poloidal rotation. The ratio of the poloidal rotation in the core of both devices is found to equal to the ratio of the center frequency of the QCM.

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# Multi-machine validation of IMEP and fusion performance predictions for ITER and DEMO

T Luda[1], C Angioni[1], M G Dunne[1], E Fable[1], A Kallenbach[1], N Bonanomi[1], P A Schneider[1], M Siccinio[1,9], G Tardini[1], P Rodriguez-Fernandez[2], J W Hughes[2], N Howard[2], L Frassinetti[3], S Saarelma[4], The ASDEX Upgrade Team[5], The EUROfusion MST1 Team[6], The Alcator C-Mod Team and JET contributors[7,8]

[1] Max-Planck-Institut für Plasmaphysik, Boltzmannstrasse 2, D-85748 Garching, Germany
[2] MIT Plasma Science and Fusion Center, Cambridge, MA 02139, United States of America
[3] Division of Fusion Plasma Physics, KTH, Stockholm, Sweden
[4] CCFE, Culham Science Centre, Abingdon, OX14 3DB, United Kingdom
[5] Stroth et al (https://doi.org/10.1088/1741-4326/ac207f) for the ASDEX Upgrade Team
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[8] EUROfusion Consortium, JET, Culham Science Centre, Abingdon, OX14 3DB, United Kingdom of Great Britain and Northern Ireland
[9] EUROfusion Consortium, Garching bei München, Germany

The Integrated Model based on Engineering Parameters (IMEP) [1-3] has been developed to predict the temperature, density, and rotation profiles over the whole radius of H-mode plasmas, from the magnetic axis to the separatrix. The workflow consists in the coupling of the ASTRA [4] transport code with the MISHKA [5] MHD stability code, and provides a self-consistent description of the core, pedestal, and scrape-off layer (SOL). Within the ASTRA package, a combination of theory-based and empirical elements are applied, such as TGLF[6]/QuaLiKiz[7] and NCLASS [8] turbulent and collisional transport models, as well as an empirical pedestal transport model based on multi-device experimental observations. A simple SOL model, obtained from an extension of the two-point model, provides the boundary conditions at the separatrix, without the need for profile measurements as input. The only inputs required are magnetic field, plasma current, heating power, fueling rate, and plasma geometry. The MISHKA MHD stability code is run on the kinetic profiles simulated by ASTRA to find the highest pedestal pressure stable to peeling-ballooning modes, corresponding to pre-ELM conditions. Extensive testing has been conducted by simulating over 100 stationary ELMy H-mode phases of Alcator C-Mod, ASDEX Upgrade, and JET-ILW discharges. The validation database includes wide ranges in operational parameters, such as heating power, current, magnetic field, triangularity, and fueling rate. The results indicate that IMEP reproduces the main dependencies captured by multi-device scaling laws, such as those of confinement time on the plasma current and on the heating power, while also predicting stored energies in significantly better agreement with experimental observations. In addition, IMEP offers an advantage over scaling laws by predicting the kinetic profiles and describing the change in confinement caused by triangularity, magnetic field, and fueling, the latter not being captured by scaling laws. The pedestal transport model accurately predicts pedestal height and width, allowing for the separate prediction of pedestal profiles for electron and ion temperatures and densities. Furthermore, IMEP can also predict conditions of the ITER baseline scenario and plasmas in the quasi-continuous exhaust (QCE or small ELM) regime, when the pedestal is limited by ideal peeling-ballooning modes (i.e. when MISHKA is applicable). Finally, the predictions for ITER and DEMO assess the fusion performance of their different scenarios and compare the estimated pedestal and global confinement with the predictions of scaling laws and previous integrated models. Overall, IMEP represents a significant step forward in our understanding and prediction of H-mode plasmas.

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# Beyond ion-temperature-gradient turbulence in stellarators

J. H. E. Proll[1], P. Costello[2], G. G. Plunk[2], M. C. L. Morren[1], R. J. J. Mackenbach[1], P. Helander[2], P. Mulholland[1], M. J. Pueschel[3]
[1]Eindhoven University of Technology, Eindhoven, The Netherlands
[2]Max Planck Institute for Plasma Physics, Greifswald, Germany
[3]Dutch Institute for Fundamental Energy Research, Eindhoven, The Netherlands

When it comes to optimising stellarators for turbulent transport, the ion-temperature-gradient mode (ITG) has so far taken front and centre – with advances in reducing the diffusivity in the strongly-driven regime or more recently in optimising for high ITG critical gradient. However, while ITGs do lead to large levels of turbulent transport, other instabilities must not be forgotten.

For trapped-electron modes (TEM), two routes for lower turbulence have emerged: On the one hand it has been found that so-called maximum-J geometry with the majority of electrons (Wendelstein 7-X is an approximation of that) experiencing bounce-averaged good curvature leads to reduced TEM growth rates and thus low TEM turbulence levels [1]. Even in the absence of the maximum-J property, low turbulence levels can be achieved: In configurations with very low shear, like the HSX stellarator, a plethora of subdominant and stable eigenmodes can lead to enhanced saturation efficiency and consequently low turbulence levels [2].

In the absence of the classical trapped electron mode another mode has recently been found to emerge: the so-called universal instability. This mode, with a much broader mode structure than the classical TEM, has been observed to lead to lower heat fluxes than comparable TEM [3,4]. For W7-X it is also found to become the dominant mode over a large range of wave numbers when trapped-particle modes are stabilised through collisions, highlighting its importance for transport in experimentally relevant conditions [5].

The story does not end with electrostatic modes, as effects of increasing plasma pressure must ultimately also be taken into account. In W7-X geometry, it has recently been found that an increase in plasma pressure gradient can lead to an early onset of a kinetic ballooning mode far below the typical threshold. This mode, while subdominant to the ITG, can lead to a strong increase in ITG heat flux [6].

One tool that might make classifying the instabilities unnecessary is the so-called available energy. The available energy can be computed at low computational costs and has been shown to correlate well with TEM heat fluxes in various geometries [7]. The available energy is now being used as a tool for direct geometry optimisation of turbulence in shaped Miller tokamaks.

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# Landau collisions for fluid and gyrokinetic simulations

**K. Hallatschek**
*Max-Planck-Institute for Plasma Physics, Boltzmannstr. 2, 85748 Garching, Germany*

A fast scheme for the machine accuracy calculation of gyro-kinetic (GK) and multi-fluid (MF) collision coefficients of the full (linearized) Landau collision operator has been developed. The code allows multiple species with finite mass ratios and differing temperatures and can be used to put the well-known Braginskii equations, which were obtained by analytic approximations with infinite ion/electron mass ratio, on a completely new basis. It will also be used to provide reliable tokamak edge turbulence simulations.

GK collision operators customarily are either based on a reduced number of moments for the (gold standard) Landau-Fokker-Planck operator, which have been transformed to GK gyro-center coordinates [1], or to the full Landau operator but in a purely drift-kinetic setting [2], owing to analytic limitations or overpowering computational cost. MF transport coefficients have been obtained in several complex analytical calculations and approximations [3] for infinite ion-electron mass ratio, but with rather intransparent ordering schemes and some missing coefficients. It should be noted that differing levels of collision representation in GK [4] and MF [5] turbulence simulations have precluded stringent comparisons between both frameworks for high collisionality, where both should be valid.

A code for the efficient calculation of the matrix elements of the complete linearized Landau operator using uniform orthogonal polynomials has been developed. On one hand, the computational cost is low – the coefficients for hundreds of polynomials can be obtained in seconds on a laptop. On the other, the matrix elements are computed up to machine precision, as confirmed by tests for various mass ratios ($1-10^{10}$), which makes them automatically fulfill all the required conservation laws and invariances. For GK simulations, the above matrix elements are then transformed to gyro-center coordinates. The collisional MF transport coefficients are calculated rapidly and completely by determining perturbed eigenspaces of the combined system of streaming and collision operator for arbitrary orders in perpendicular and parallel wavenumbers, which yields the complete transport matrix far more accurately than in literature [3, 6]. Due to the precision of the calculation, the (necessary) Onsager symmetries are automatically fulfilled. In addition, unlike the traditional analytic expansions, the code does not require infinite mass ratio and works for more than two species.

Due to the modular nature, the method can also guide the implementation of more accurate collision treatments, such as a combination of the small-angle Balescu-Lenard operator with the Boltzmann operator for large collision angles. This would only render the initial calculation of the matrix elements more costly but not the gyro-transformation or the transport coefficients. Moreover even the nonlinear Landau operator seems now to be in range, since it just requires a bilinear extension of the scheme.

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# ASTRA modularity and IMASification for integrated modelling workflows

G. Tardini, E. Fable, C. Angioni, D. Fajardo and the ASDEX Upgrade Team

Max-Planck-Institut für Plasmaphysik, Boltzmannstr. 2, 85748 Garching, Germany

Theory-based transport models have improved over the last decades, capturing trends and increasingly new features. Integrated transport modelling is now envisaged for a predict-first approach, especially core-edge integration and the self-consistent modelling of the source profiles. Modularity is highly desirable in order to benchmark new models, compare results and setup different modelling workflows with the due flexibility. Several paradigms for workflows based on IMAS objects are being developed nowadays, either in python, creating actors with the Iwrap package, or even language-agnostic, like those based on the MUSCLE3 software.

The ASTRA transport suite, however, is itself a container designed to enable users to setup their own workflows. It contains, in fact, a miscellany of pre-compiled equations and functions for the evolution of the plasma state. Additionally, however, there is a pre-processor which parses a list of user-defined instructions to generate a variable Fortran code, which gets eventually compiled and executed. Modularity turns out to be nevertheless important, both for the communication with the core of the code, as well as for separate testing and evaluation of single modules. Moreover, some of the fixed subparts of ASTRA, such as the transport solver, could be enveloped in external workflows, as it happens e.g. in the TRIASSIC suite of codes [1]. A significant step towards modularity was the recent refurbishing of ASTRA (v8.*), replacing all COMMON blocks with For-tran modules.

In this paper we present the modularisation of several codes coupled to ASTRA: the ECRH code TORBEAM, the NBI code RABBIT, and FEQIS, a newly developed Grad-Shafranov solver for ASTRA with prescribed boundary. A paradigm based on f2py wraps each module into a python driver, offering a comprehensive visualisation of the module's input and output. Moreover, the new coupling with the latest version of the STRAHL code for impurity transport is discussed. This was made possible by the new ASTRA python pre-processor. Finally, ASTRA is on the way towards IMAS-compatibility, taking IDS files as input and dumping IDS output files with the full time evolution of the simu-lation. As demonstration, some applications to experimental cases are also presented and compared to the High Fidelity Pulse Simulator as part of the TSVV11 activities.

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# Configuration dependence of regimes with suppressed turbulent impurity transport in Wendelstein 7-X

Thilo Romba[1], Felix Reimold[1], Oliver Ford[1], Peter Poloskei[1], and the W7-X team

[1] Max-Planck-Institut für Plasmaphysik, 17491 Greifswald, Germany

thilo.romba@ipp.mpg.de

Understanding of the impurity transport in fusion plasmas is crucial to predict the impurity content in a future reactor as this will limit the accessible parameter space for a burning plasma due to dilution (low Z) and radiative losses (high Z). In the stellarator Wendelstein 7-X (W7-X), impurity transport in plasmas heated by electron cyclotron resonance heating (ECRH) is found to be dominated by anomalous diffusion [1, 2] due to the neoclassical optimization of the W7-X magnetic field geometry [3]. In contrast, neutral beam heated W7-X plasmas in the high mirror magnetic configuration often exhibit a central accumulation of impurities, consistent with neoclassical transport calculations [4].

Previous work [4] showed that the suppression of the turbulent impurity transport in neutral beam heated plasmas is propagating radially inwards with transport characteristics suddenly changing. This work studies the radial location and conditions of the local change in transport, particularly its dependence on magnetic configuration. Various field configurations are assessed due to changes in the neoclassical transport which was identified as the dominant transport channel in the assessed type of discharges [4]. In addition to changes in the neoclassical transport, differences in the minor plasma radius between the configurations are expected to affect the anomalous impurity transport due to varying gradient lengths. Radial impurity density profiles of ionization states of low and medium $Z$ impurities are measured using charge exchange recombination spectroscopy (CXRS) [5].

Differences between the radii within which impurities accumulate are observed between the assessed magnetic field configurations in the radial impurity profiles from CXRS. In addition, the length of a phase of pure NBI heating prior to the impurity accumulation is found to be configuration dependent. To assess potential drivers of the suppression of the turbulence, correlations of local plasma parameters and regions of changed impurity transport are investigated. As an actuator to prevent the accumulation of impurities in such scenarios, the addition of low levels of ECRH is found to preserve low impurity concentrations in the plasma core while the main ion density increases due to beam fuelling. The impact of the added ECRH on the impurity transport is observed to behave threshold like with 1 MW of heating avoiding the impurity accumulation in the high mirror magnetic configuration.

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# Studies of confined energetic helium ions in ASDEX Upgrade plasmas

A. Kappatou[1], M. Weiland[1], R. Bilato[1], Ye.O. Kazakov[2], R. Dux[1], V. Bobkov[1], T. Pütterich[1], R.M. McDermott[1], the EUROfusion MST1 team[*] and the ASDEX Upgrade team[+]

[1] Max-Planck-Institut für Plasmaphysik, D-85748 Garching, Germany
[2] Laboratory for Plasma Physics, LPP-ERM/KMS, TEC Partner, BE-1000 Brussels, Belgium

Fast ion physics is an active field of research in the fusion community, but most studies focus on deuterium fast ions. The generation and investigation of energetic helium in present devices, however, provide significantly more insight into how the fast alpha particles produced from fusion reactions will behave in future reactor plasmas. Studies of confined fusion-produced helium studies were previously carried out at TFTR [1]. Such investigations have been conducted in non-nuclear devices by simulating the fast helium ion population either by helium neutral beam injection [2], or by accelerating either $^4$He-beam ions [3] or $^3$He ions [4] with ion cyclotron resonance heating (ICRF).

Confined fast helium ion populations can be measuredwith charge exchange recombination spectroscopy (CXRS) in the wings of the helium spectral line (He II $n$ = 4–3, 468.6 nm) providing information on their distribution function. We present CXRS measurements of energetic $^3$He ions, a first for ASDEX Upgrade [5]. In these plasmas, a three-ion ICRF heating scenario [6, 7] was used to heat a mixed hydrogen–deuterium plasma ($n_H/n_e$ ~ 70%–80%) with a small amount of $^3$He ions ($n_{3He}/n_e$<1%) accelerated to high energies (~1MeV). The CX spectral signature of the energetic helium ions is compared with the theoretical predictions, via forward-modelling of the spectrum utilising TORIC-SSFPQL [8] distribution functions. The expected energies of the ions agree well with the measurement, confirming that the spectral feature is due to ICRF-accelerated $^3$He ions. Such measurements provide a way to study confined fast ions without restricting the studies to deuterium. Furthermore, they can act as a test-bed for modelling assumptions.

Detailed comparisons between the experimental measurement and the modelling reveal discrepancies that illuminate details of the velocity distribution function of these ions. In particular, the differences indicate that other physics mechanisms might be at play leading to the radial re-distribution of the energetic $^3$He ions, for example orbit losses, radial fast ion transport, or MHD activity (sawtooth crashes, fishbones). We discuss the potential of the energetic $^3$He CXRS measurements to assess these mechanisms.

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[*] See author list of *B. Labit et al. 2019 Nucl. Fusion **59** 086020*
[+] See author list of *U. Stroth et al. 2022 Nucl. Fusion **62** 042006*

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# Isotope Mass Scaling and Transport Comparison between JET Deuterium and Tritium L-mode Plasmas

T. Tala[1], A.E. Järvinen[1], C.F. Maggi[2], P. Mantica[3], A. Mariani[3], A. Salmi[1], I.S. Carvalho[2], A. Chomiczewska[4], E. Delabie[2], A. Devasagayam[5], J. Ferreira[6], W. Gromelski[4], N. Hawkes[2], L. Horvath[2], J. Karhunen[1], D. King[2], A. Kirjasuo[1], E. Kowalska-Strzeciwilk[4], S. Leerink[5], M. Lennholm[2], B. Lomanowski[7], M. Maslov[2], S. Menmuir[2], R.B. Morales[2], R. Sharma[2], H. Sun[2], K. Tanaka[8], and JET contributors*

[1]VTT, PO Box 1000, FI-02044 VTT, Espoo, Finland
[2]UKAEA, Culham Science Centre, Abingdon, OX14 3DB, UK
[3]Institute for Plasma Science and Technology, CNR, Milano, Italy
[4]IPPLM, Warsaw, Poland
[5]Aalto University, Espoo, Finland
[6]Instituto de Plasmas e Fusao Nuclear, Universidade de Lisboa, Lisboa, Portugal
[7]Oak Ridge National Laboratory, Oak Ridge, Tennessee, US
[8]National Institute for Fusion Science, Toki, Gifu, Japan
*See the author list in J. Mailloux et al., Nucl. Fusion 2022, (https://doi.org/10.1088/1741-4326/ac47b) for the JET Contributors

The dimensionless isotope mass scaling experiment between pure Deuterium and pure Tritium plasmas with matched $\rho^*$, $\nu^*$, $\beta_n$, $q$ and $T_e/T_i$ has been achieved in JET L-mode with dominant electron heating (NBI+ohmic) conditions. 28% higher scaled energy confinement time $B_t\tau_{E,th}/A$ is found in favour of the Tritium plasma. This can be cast in the form of the dimensionless energy confinement scaling law as $\Omega_i\tau_{E,th} \sim A^{0.48\pm0.16}$. This significant isotope mass scaling is consequently seen in the scaled one-fluid heat diffusion coefficient $A\chi_{eff}/B_t$ which is around 50% lower in the Tritium plasma throughout the whole plasma radius. The isotope mass dependence in the particle transport channel is negligible, supported also by the perturbative particle transport analysis with gas puff modulation. The comparison of the edge particle fuelling or ionisation profiles from the EDGE2D-EIRENE simulations show that the absolute density differences that are necessary for the dimensionless match in the confined plasma dominate over any isotope mass dependencies of particle fuelling and ionization profiles at the plasma edge. Local GENE simulation results indicate a mild anti-gyroBohm effect at $\rho_{tor}$=0.6 and thereby a small isotope mass dependence in favour of Tritium on heat transport and a negligible effect on particle transport. A significant fraction of the isotope scaling and reduced heat transport observed in the Tritium plasma is not captured in the GENE simulations or in the

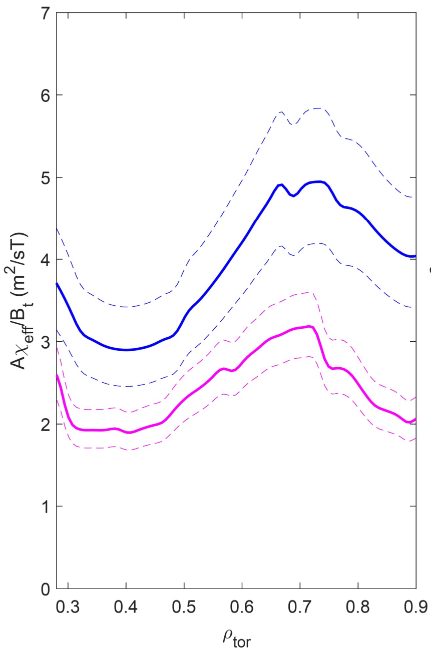


*Fig. 1. Scaled one-fluid effective diffusion coefficients $A\chi_{eff}/B_t$ for the matched (dimensionless) Deuterium (blue) and Tritium discharge (magenta) pair. The dashed lines indicate the error bars.*

ASTRA-TGLF-SAT2 simulations by simply changing the isotope mass for the same input profiles.

[1] T. Tala *et al.*, submitted to Nuclear Fusion Special Issue on JET T and D-T.

# Electron density response to strike-point sweeping on JET

A. Salmi[1], T. Tala[1], R.B. Morales[2], I. Carvalho[3, 4], P.J. Lomas[2] and JET contributors*
[1] VTT, Technical Research of Finland, Espoo, Finland
[2] UKAEA/CCFE, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK
[3] Instituto de Plasmas e Fusão Nuclear, Instituto Superior Técnico, Lisboa, Portugal
[4] ITER Organization, Route de Vinon-sur-Verdon, 13067 St. Paul Lez Durance Cedex, France
* See the author list in J. Mailloux et al., 2022 Nucl. Fusion **62** 042026

Strike-point sweeping, a technique often used to spread the heat loads on divertor targets, was employed in JET experiments for the first time to generate an edge-localized modulated particle source for investigating plasma fuelling and particle transport in the edge region [1]. This approach was motivated by the possibility to achieve higher modulation frequencies than those practically available from traditional gas puff modulation methods at JET (~6 Hz). Higher frequencies would enable the collection of more edge-localized information from the electron density response to the modulated particle source. Experiments utilized various sweeping frequencies commissioned up to 18.5 Hz. Both strong and weak electron density responses were observed in H-mode plasmas, depending on the strike-point configuration and the distance the strike-points moved during the sweep cycle. The electron density response exhibited a complex and unconventional behaviour (compared to gas puff modulation), which presented challenges for the interpretation.

In this study, we analyse one experiment in detail using an optimization framework to determine transport and particle source parameters by fitting our forward model parameters to experimental electron density measurements. Refer to Fig. 1 for an example of the best fitting simulation and the seemingly strange non-monotonic electron density modulation amplitude and phase observed in the experiment.

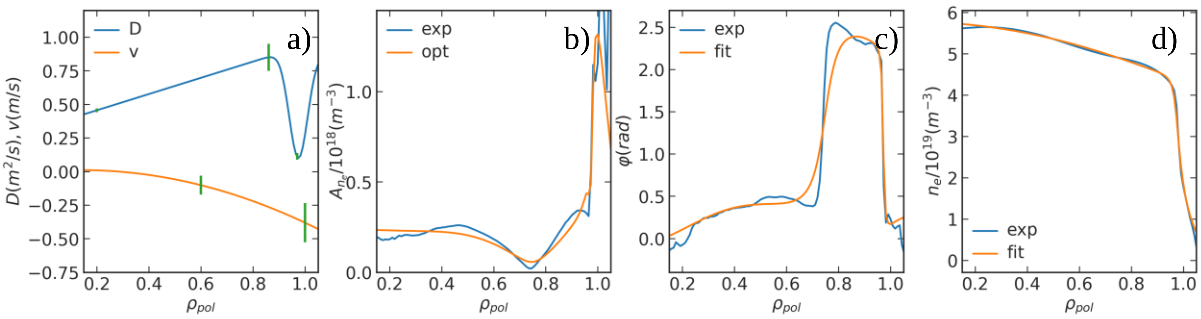


*Figure 1 (a) Inferred particle transport coefficients, and (b-d) Measured and simulated electron density profile and its response to the sweep at the 7.7Hz modulation frequency for discharge #92347 at t=52.85-56.4s.*

We demonstrate that a consistent picture emerges and that our approach can provide new insights into these data. However, we note that while strike-point sweeping creates the desired modulated edge-localized particle source, it also alters the properties of the edge transport barrier. Therefore, the strike-point sweeping methodology is a promising but challenging way to study edge particle transport and edge fuelling properties, requiring very precise measurements and complex modelling to disentangle the various effects.

[1] A Salmi et al 2023 Plasma Phys. Control. Fusion **65** 055025

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# Isotope dependence of intrinsic torque in JET ohmic plasmas

A. Kirjasuo[1], T. Tala[1], M.F.F. Nave[2], J. Ferreira[2], E. Delabie[3], and JET contributors[*]
[1] VTT, Espoo, Finland, [2] Instituto de Plasmas e Fusão Nuclear, Instituto Superior Técnico, Universidade de Lisboa, P1049-001 Lisboa, Portugal, [3] Oak Ridge National Laboratory, Oak Ridge, TN 37831-6169, USA, [*] author list in of J. Mailloux et al 2022 Nucl. Fu. 62 042026

A set of intrinsic rotation experiments[1] in H, D and T have been performed at JET. These ohmic L-mode discharges form a dimensional isotope mass scaling experiment with $B_t$ = 2.7 T, $I_p$ = 2.3 MA, $n_{e,ave}$ = 2.2x10$^{19}$ m$^{-3}$, and $T_{i,ave}$ = 550 eV. Isotope dependent intrinsic rotation is observed with peaked rotation profile in H, and hollow in D and T, with the most hollow profile in Tritium. In this work we have estimated the intrinsic torque profiles that are required for producing the experimental intrinsic rotation profiles.

In this work JETTO[2] is used to solve the momentum transport equation (1) and estimate the intrinsic torque required for producing the observed intrinsic rotation.

$$m_i \frac{\partial}{\partial t} n_i v_\phi = n_i m_i \left( \underbrace{\chi_\phi \frac{\partial v_\phi}{\partial r}}_{\text{Diffusive term}} - \underbrace{\cancelto{\approx 0}{V_\phi} v_\phi}_{\text{Convective term}} \right) + \underbrace{\Pi_{RS}}_{\text{Residual stress}} \qquad \mathbf{1)}$$

We have assumed that momentum diffusivity $\chi_\phi = Pr * \chi_{i,eff}$, where $\chi_{i,eff}$ is the effective ion heat diffusivity (from TRANSP) and $Pr$ is Prandtl number profile (based on earlier work in references [3,4]). In addition, we assume that there is no isotope scaling in the Prandtl number. Furthermore, since the rotation is very small, the convective term is assumed to be zero (as marked in equation (1)). Since we are analysing steady state intrinsic rotation ($\frac{\partial}{\partial t} = 0$ on the left-hand side in equation (1)), rotation is effectively determined by the balance between the diffusion and the residual stress components. In our analysis we are solving intrinsic torque density $\tau_{int} = \Pi_{RS}$ from the equation (1) with the assumptions listed above.

The analysis shows an isotope effect in the intrinsic torque. While there is a difference in the $\chi_{i,eff}$ and $\chi_\phi$ profiles between the different isotopes, this difference is not sufficient for explaining the difference between peaked and hollow experimental rotation profiles. The experimental hollow intrinsic rotation profiles in D and T cannot be reproduced without intrinsic torque direction change in the gradient region. The effect is most pronounced in T, where the rotation profile is the most hollow. In T a steep change of the intrinsic torque direction at mid-radius is required to produce the deeply hollow experimental intrinsic rotation profile. Intrinsic torque direction change is also required to reproduce the hollow D intrinsic rotation profile. In H the intrinsic torque direction remains co-current, enabling the peaked intrinsic rotation profile.

We have carried out several sensitivity tests, including changing the Prandtl number profile both in magnitude and profile shape (constant and increasing from core to edge), and changing $\chi_{i,eff}$ and $\tau_{int}$ profiles between the studied pulses. The observed results of the intrinsic torque profiles and their isotope dependence survived all performed sensitivity tests.

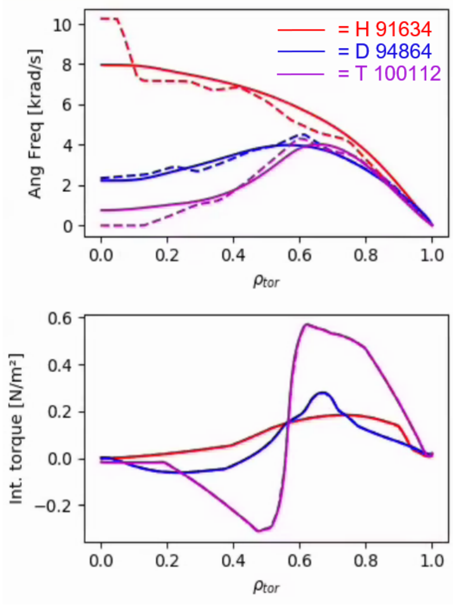


*Figure 1. Upper: Experimental (dotted line) and simulated (solid line) intrinsic rotation profiles. Lower: The solved intrinsic torque profiles*

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# Exploring state representation learning algorithms for multi-machine generative modelling of fusion plasmas

A.E. Järvinen[1, 2], A. Kit[2], Y. Poels[3], S. Wiesen[4], V. Menkovski[3], L. Frassinetti[5], M. Dunne[6], the ASDEX Upgrade Team* and JET Contributors†
[1]VTT Technical Research Centre of Finland, FI-02044 VTT, Finland
[2]University of Helsinki, FI-00014, Helsinki, Finland
[3]Eindhoven University of Technology, NL-5600 MB, Eindhoven, Netherlands
[4]Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung – Plasmaphysik, DE-52425 Jülich, Germany
[5]Division of Fusion Plasma Physics, KTH Royal Institute of Technology, SE-100 44 Stockholm, Sweden
[6]Max-Planck-Institut für Plasmaphysik, D-85748 Garching, Germany

Latent variable based deep learning algorithms are investigated for establishing machine size aware state representations for plasmas in JET and ASDEX-Upgrade (AUG) (Fig. 1) [1]. Dynamical systems with high dimensional observations, such as magnetic confinement fusion devices, can significantly benefit from a lower dimensional representation that conforms to the actual degrees of freedom of the system. This is basically the aim of the traditional scientific workflow that produces analytical and numerical models to explain observations and to provide generalized physics understanding that extrapolates beyond observations. Recent developments in deep generative, latent variable models opens a pathway to algorithms that learn these reduced dimensionality state representations automatically from databases of observations. Such state representations could be useful for many applications ranging from predictive modelling to integrated data-analysis and large-scale model validation [2, 3]. This work explores algorithms to establish these representations based on multi-machine databases of observations. Using Domain Invariant Variational Autoencoder (DIVA) type algorithm [4] with auxiliary regression modules for the device control parameters and size, the algorithm encodes size dependence in the latent representation (Fig. 1). Previous studies demonstrate inverse size dependence of the pedestal electron density, $n_{e,PED}$, encoded in the latent representation consistent with the empirical Greenwald density limit scaling (Fig. 1) [5, 6]. In this work, the size dependencies will be further elaborated and compared to other expected device size dependent features.

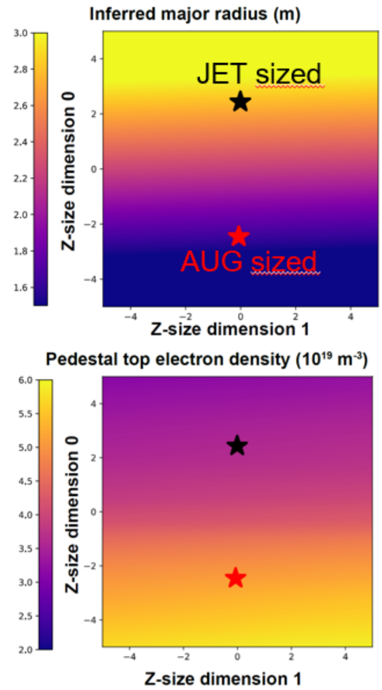


*Figure 1. Contour of the inferred device major radius (upper) and pedestal top electron density (lower) as a function of two of the latent space dimensions.*

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* *See author list of U. Stroth et al. 2022 Nucl. Fusion* ***62*** *042006*
† *See author list of" 'Overview of JET results for optimising ITER operation' by J. Mailloux* et al *2022* Nucl. Fusion ***62*** *042026*

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# Ion temperature gradient mode mitigation by energetic particles, mediated by forced-driven zonal flows

J.N Sama[1], A. Biancalani[2], A. Bottino[3], D. Del Sarto[1], G. Di Giannatale[5], A. Ghizzo[1], T. Hayward-Schneider[3], Ph. Lauber[3], A. Mishchenko[4], M. Muruggapan[5], B. Rettino[3], B. Rofman[5], F. Vannini[3], L. Villard[5] and X. Wang[3]
[1]Université de Lorraine, CNRS, IJL, Nancy, France
[2]Léonard de Vinci Pôle Universitaire, Research Center, Paris La Défense, France
[3]Max Planck Institute for Plasma Physics, Garching, Germany
[4]Max Planck Institute for Plasma Physics, Greifswald, Germany
[5]Swiss Plasma Center, EPFL, Lausanne, Switzerland

Heat and particle transport due to drift wave turbulence is of concern to fusion plasma reactors due to its ability to deteriorate plasma confinement. Temperature gradient driven modes like the ion temperature gradient modes (ITG) [1] are the most common drift waves present in tokamaks. The major saturation mechanism of ITG modes is believed to be their spontaneous non-linear excitation of zonal flows [2]. Recently, global electromagnetic simulations with the gyro-kinetic particle-in-cell code ORB5 [3] have been performed, where the self-consistent interaction of ITG turbulence, zonal flows, and Alfven modes has been studied. As a result, it was conjectured that zonal flows forced-driven by Alfven instabilities might be used to reduce the level of ITG turbulence [4]. This would represent an indirect way of interaction of energetic particles (driving Alfven modes unstable) and turbulence.

In this work, we investigate the effects of force driven zonal flows on ITG modes by isolating this mechanism in the self consistent nonlinear simulations and prove that it is indeed effective. To this aim, the amplitude and radial structure of the forced-driven zonal flows are measured and imposed on electrostatic ITG simulations. Imposing this self-consistent zonal flow leads to a significant mitigation of the ITG instability, which induces a significant drop of the ITG growth rates. A scan of the zonal flow amplitude shows an inverse relation between the ITG growth rate and the zonal flow amplitude.

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# Ion transport through Radio-Frequency sheaths studied by Particle-In-Cell simulations

J. Moritz[1], S. Heuraux[1], M. Lesur[1], E. Gravier[1] and F. Brochard[1]

[1] *Université de Lorraine, CNRS, Institut Jean Lamour, UMR 7198, F-54000 Nancy, France*

This study, based on PIC simulations, concerns the sheath properties and the transport of ions in the edge of a 1d plasma bounded by two parallel conductive walls when a RF voltage is applied between them. A constant magnetic field B is assumed in this numerical model with an inclination with respect to the surfaces ranging from a few degrees up to 90°. The simulations were run for different ion-to-electron mass ratios corresponding to hydrogen and deuterium, as well as for different RF angular frequency $\omega_{rf}$. The latter is monitored such that the ion plasma angular frequency $\omega_{pi}$ to $\omega_{rf}$ ratio is in the range 10 to 1, conditions met in the neighborhood of ICRH antenna in fusion plasma devices. The time evolution of the particle flux at the walls and of the electron sheath position is analyzed for a broad range of RF frequencies investigated. We show that for normal incidence of B, ions transit time $\tau_i$ in the sheath is smaller than the RF period so that ions experience instantaneous local electric fields as long as $\omega_{pi}/\omega_{rf} > 1$. However, the ion flux at the collecting surface exhibits a time modulation due to the fact that $\tau_i$ depends on the phase at which the particles enter the sheath with respect to the RF modulation [1]. The ion energy distribution exhibits two peaks whose relative amplitude depends on $\omega_{pi}/\omega_{rf}$ and can be related to the ion transit time [2].
Finally, for decreasing incidence of B at the same $\omega_{rf}$, the ion density at the sheath entrance decreases as well as the effective $\omega_{pi}$, which modifies $\tau_i$ and changes the ion energy distribution as well as the average energy collected by the walls as seen in [3].

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# Impurity transport through two types of transport barrier in 5D gyrokinetic simulations

G. Lo-Cascio[1], E. Gravier[1], T. Réveillé[1], M. Lesur[1], X. Garbet[2,3],Y. Sarazin[2], K. Lim[4], G. Dif-Pradalier[2], P. Donnel[2], A. Guillevic[1], T. Rouyer[1], V. Grandgirard[2]

[1] *IJL, UMR 7198 CNRS, Université de Lorraine, 54000 Nancy, France*

[2] *CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France*

[3] *School of Physical and Mathematical Sciences, Nanyang Technological University, 637371 Singapore*

[4] *Ecole Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center, EPFL SB SPC, Station 13, CH-1015 Lausanne, Switzerland*

Both collisional and turbulent transport govern the impurity transport in tokamaks. Inward neoclassical convection is known to dominate transport for heavy impurities[1]. Using 5D gyrokinetic GYSELA[2] simulations, the impact of two types of transport barrier is studied. The first type of transport barrier is triggered by an external poloidal momentum source (i.e. vorticity)[3] that locally polarizes the plasma. This results in a locally reduced heat diffusivity coefficient and a slight core pressure increase compared to a case without transport barrier[4]. The second type of barrier is triggered through an imposed strong radial density gradient for the main species (D) at a localized radial position. This creates through the force equilibrium $E_r = -\frac{1}{e_i n_i}\frac{\partial P_\perp}{\partial r} + v_\theta B_\varphi - v_\varphi B_\theta$ a sheared radial electric field and thus a strongly sheared poloidal flow. Also, the strong density gradient stabilizes linearly ITG modes, thus creating an effective transport barrier with low heat diffusivity.

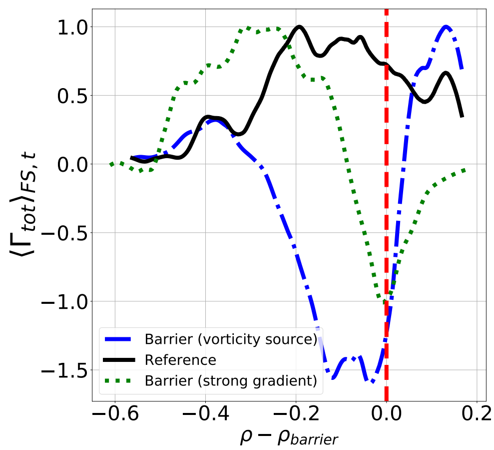


Figure 1: *Normalized time-averaged total radial particle flux of helium impurities with (blue dash-dotted and green dotted lines) and without transport barrier (black solid line). The red dashed vertical line indicates the transport barrier position.*

A range of impurities in the trace regime with different collisionality regimes is explored in the presence of each of the previously described transport barriers, like helium ashes ($Z = 2$, low collisionality "Banana-Plateau"), argon ($Z = 18$, medium-high collisionality) and tungsten ($Z = 40$, high collisionality "Pfirsch-Schlüter"). As shown in Fig.1, the radial flux of helium changes sign from positive (outward) to negative (inward) in the vicinity of the transport barrier. This is due to the strong reduction of turbulent transport, which is the main transport channel for low-Z impurities like helium. A strong impact of poloidal asymmetries and anisotropies on neoclassical transport is observed in the vorticity source case.

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# Mutual interactions of turbulent transport in COMPASS tokamak characterized by means of ultra-fast camera and machine learning

S. Chouchene[1,3], F. Brochard[1], J. Cavalier[2], M. Desécures[3], N. Lemoine[1], V. Weinzettl[2]

[1] *Université de Lorraine, Institut Jean Lamour, Nancy, France*

[2] *Institute of Plasma Physics of the CAS, Prague, Czech Republic*

[3] *APREX Solutions, Pulligny, France*

The dynamics of turbulent transport in a tokamak plasma is of significant interest due to its impact on plasma confinement and stability. Turbulent filaments, also called blobs, can be investigated experimentally with a variety of diagnostics like beam emission spectroscopy, electrostatic probes, reflectometry and fast visible imaging [1]. In our investigation, high-speed videos of the plasma edge have been captured in COMPASS tokamak and analysed after applying tomographic inversion in a poloidal plane across the separatrix with a spatial resolution of 2.5 mm [2]. Both classical statistical methods and machine learning algorithms have been set up to gain new insights into the behavior of blobs, including their propagation speed, size, and shape through the analysis of data acquired at unprecedented rates in passive visible imaging up to 1 million frames per second, revealing rich and complex dynamic behaviors such as splitting, coalescence and equivocal transient inversions of the poloidal propagation direction of filaments (Fig. 1). To date, such kind of mutual interactions between filaments has never been reported experimentally, even if they have been investigated theoretically [3].

In order to analyse large sets of data, we have developed a machine learning model capable to distinguish and quantify these interactions, achieving a mean average precision of 89 % during the prediction step. This model is helpful to better understand results obtained with classical tracking methods applied to data recorded at lower frame rates.

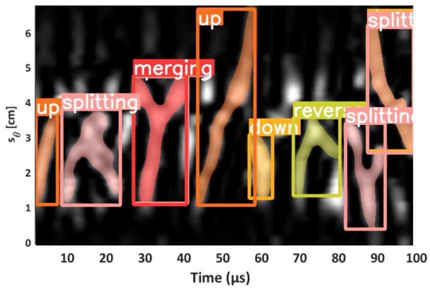


Figure 1: *Light kymograph along the separatrix showing filament interactions, where $S_\theta$ is a poloidal coordinate.*

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# Flow and phase velocity of turbulence in magnetized fusion plasmas

L. Vermare[1], H. Corvoysier[1], O. Gürcan[1], G. Dif-Pradalier[2], X. Garbet[2], V. Grandgirard[2],
P. Hennequin[1], C. Honoré[1], S. Rienäcker[1], Y. Sarazin[2], R. Varennes[1] and the WEST team[2]
[1]LPP, Ecole Polytechnique, CNRS, 91128 Palaiseau, France
[2]CEA, IRFM, F-13108 Saint Paul Lez Durance, France

Turbulence and flows play a key role in tokamak plasmas performance. While the energy confinement time is limited by the presence of small-scale turbulence, plasma flows can induce a strong velocity shear, which tends to reduce the spatial scale of turbulent structures and thus the associated energy transport. Indeed, a significant increase of performance is made possible by the onset of a transport barrier, i.e. a localised radial layer in the plasma where turbulent transport is reduced thanks to strong sheared flows. The formation of transport barriers is a topic of utmost importance in the community of fusion plasma physicists. In particular, understanding of underlying physics of turbulence, its characteristics and its interaction with flows when approaching the transition appears to be mandatory.

The aim of the present contribution is to gain insight into the nature of the instability at the origin of the turbulence and its impact on the flow [1] by studying the dynamics of the density fluctuations and by identifying the contribution coming from the "phase velocity" of the density fluctuations. This study is far from straightforward and requires a combination of detailed experimental characterisation, via Doppler backscattering measurements, and a numerical study based on turbulence simulations.

Doppler backscattering (DBS) gives access to the intensity and velocity of density fluctuations at a selected spatial scale [2]. DBS is a microwave diagnostic that is commonly used in the community to measure the radial profile of the perpendicular plasma velocity. The velocity detected is the sum of the mean flow plasma velocity plus the phase velocity of the density fluctuations. In the most common case, density fluctuations are considered as tracers that are advected by the plasma that give access to the velocity of the plasma flow. In this sense, the intrinsic velocity of the fluctuations, which corresponds to the phase velocity in a linear stability analysis, is considered negligible. This approximation seems appropriate at the edge of the plasma where the mean flow velocity can be large. However, depending on the plasma conditions, this phase velocity can contribute significantly to the measured velocity [3]. An evaluation of its amplitude can be obtained by measuring the evolution of the density fluctuations velocity as a function of the wavenumber, since the mean flow velocity does not depend on the wavenumber. New experiments have been carried out in the WEST tokamak to evaluate the wavenumber dependence of the density fluctuations velocity by changing the probing angle during the stationary phase of a single discharge.

Furthermore, in order to go beyond this amplitude assessment, combined studies using results from first-principles numerical simulations in different plasma conditions (close or far above the instability threshold) are performed to characterise the density fluctuation velocity as a function of the wavenumber. The results in different extreme cases (strong / quiet zonal flow activity) are compared to experimental observations.

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# Gaining insight into $E \times B$ flow control by plasma current in different magnetic configurations

S. Rienäcker[1], L. Vermare[1], P. Hennequin[1], R. Varennes[1], J.P. Gunn[2] and the WEST team[2]
[1]Laboratoire de Physique des Plasmas (LPP), 91120 Palaiseau, France
[2]CEA-IRFM, F-13108 Saint-Paul-Lez-Durance, France. WEST team list.

The role of sheared $E \times B$ flows is long recognized as decisive for turbulence control and access to improved confinement regimes in tokamaks and other fusion devices. To date, the principal mechanisms controlling these flows are not clearly understood despite continuous theoretical and experimental progress. This contribution investigates the influence of the magnetic configuration on the radial electric field behavior, particularly with respect to its sensitivity on plasma current. Understanding the mechanisms involved could also point more generally to the dominant drivers of $E \times B$ flows.

Typical L- and H-mode tokamak plasmas exhibit a significant negative radial electric field $(E_r)$ in a thin radial layer close to the separatrix. This $E_r$ "well" presumably acts as a major regulator for turbulent transport via the associated sheared $E \times B$ drift, hence the need to elucidate its origin. Experimental observations point to an important – yet unexplained – sensitivity of the $E_r$ well to certain parameters, including plasma current $I_p$ and magnetic topology. For instance, a deepening of $E_r$ with $I_p$ has been observed on several tokamaks [1–3]. Recently, this trend has also been recovered numerically from first-principle simulations [4]. While the interpretation of the $I_p$-dependence is the subject of ongoing work presented elsewhere [5], a related intriguing observation draws particular attention: in "unfavorable" configuration on WEST, i.e., with the ion $\nabla B$ drift pointing away from the active X-point, the sensitivity of $E_r$ to $I_p$ is significantly enhanced compared to the opposite "favorable" situation [1].

While the asymmetry between unfavorable and favorable $B \times \nabla B$ directions in terms of L-H power threshold is observed in most tokamaks, a robust explanation for this phenomenon is yet missing. A key player in this context is presumably the $E_r$ well. In fact, in L-mode plasmas, the $E_r$ well is more pronounced in favorable configuration, as observed on WEST [1] and on other devices [6, 7]. However, on WEST this trend fades or even reverses when increasing the plasma current, which strongly amplifies the $E_r$ well in unfavourable, and only mildly in favorable configuration. Thus, at high enough current and with additional heating, unfavorable discharges surprisingly exhibit stronger $E \times B$ shear.

This puzzling observation raises the question examined here: how does the $B \times \nabla B$ direction (relative to the X-point) control the edge $E \times B$ drift velocity and particularly its dependence on $I_p$? To tackle this issue, edge to near-SOL perpendicular flow profiles obtained from Doppler backscattering [8] are investigated during $I_p$ scans in L-mode. Further information is provided by parallel SOL flow measurements via Mach probes. Improved data processing methods are used for a systematic investigation, revisiting experiments from past campaigns on Tore Supra/WEST and presenting new results from the recent campaign.

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# Detection of alpha heating in JET-ILW DT plasmas by a study of the electron temperature response to ICRH modulation

P.Mantica[1], F.Auriemma[2], I.Casiraghi[1], D.Gallart[3], K.Kirov[4], E.Lerche[5], M.Maslov[4], A.Salmi[6], Z.Stancar[4], A.Dal Molin[1], E.Delabie[4], J.Eriksson[7], J.Garcia[8], P.Jacquet[4], T.Jonsson[9], V.Kiptily[4], E.Litherland-Smith[4], C.F.Maggi[4], M.Mantsinen[3], S.Menmuir[4], M.Nocente[10], E.Peluso[11], G.Pucella[12], D.Rigamonti[1], G.Szepesi[4], M.Tardocchi[1], D.Van Eester[5] and JET Contributors*

[1]Institute of Plasma Science and Technology, CNR, Milano, Italy
[2] Consorzio RFX ISTP-CNR, Padova, Italy
[3] Barcelona Supercomputing Center, Barcelona, Spain
[4] UKAEA, CCFE, Culham Science Centre, Abingdon, Oxon, United Kingdom
[5] Laboratory for Plasma Physics, ERM/KMS, Brussels, Belgium
[6] VTT, PO Box 1000, FI-02044 VTT, Espoo, Finland
[7] Department of Physics and Astronomy, Uppsala University, Uppsala, Sweden
[8] CEA, IRFM, Saint-Paul-lez-Durance, France
[9] Fusion Plasma Physics, EES, KTH, Stockholm, Sweden
[10] Università di Milano Bicocca, Milano, Italy
[11] University of Rome "Tor Vergata", Roma, Italy
[12] ENEA C.R.Frascati, Frascati, Italy
* See the author list of ' J. Mailloux et al 2022 Nucl. Fusion 62 042026'

Detection of α-particle heating in DT plasmas with fusion gain <1 is difficult by steady-state power balance analysis, since the α power is small compared to the external heating and falls within experimental uncertainties. A new detection method has been attempted in the JET DTE2 campaign, based on the dynamic response of the electron temperature $T_e$ to modulated Ion Cyclotron Resonance Heating (ICRH). ICRH modulation induces both a modulation in $T_i$ (i.e. in α-particles from thermal-thermal fusion reactions) and a modulation in fast D or T ions from Neutral Beam Injection (NBI), depending on the chosen ICRH scheme (i.e. in αs from beam-thermal/beam-beam fusion reactions). This modulated electron heating from αs has a longer phase delay than most other electron heating sources, due to the α high energy and long slowing down time, resulting in a delayed $T_e$ response in comparison to the $T_i$ response.

Best results have been obtained in JET DTE2 in the T-rich Hybrid scenario [1] at 3.86 T, 2.5 MA with 15%-85% D-T mix, ~29 MW D NBI power and 4 MW ICRH power in n=1 D scheme, square-wave modulated at 1 Hz with 50% duty-cycle. Fig.1 shows profiles of amplitudes and phases of $T_e$ and $T_i$ modulations. Amplitudes are ~ 10% with centrally peaked profiles, as expected. The key result is the large phase delay of central $T_e$ with respect to $T_i$: whilst ions show an expected central phase delay ~ 50 deg, increasing towards the edge, electrons show maximum phase delay in the centre, ~105 deg, decreasing towards values similar to the ion phases when approaching the edge. This is a clear signature of a central $T_e$ modulation at least in part due to fast ions with long slowing down times.

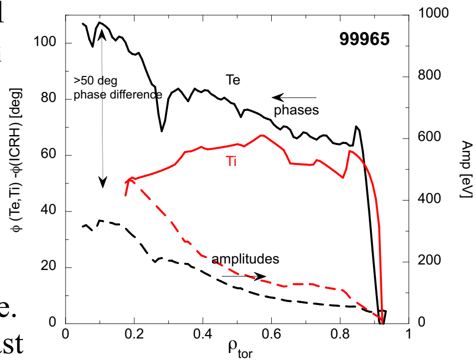


*Fig.2: Amplitudes (dashed) and phases with respect to ICRH power (solid) of $T_e$ (black) and $T_i$ (red) for shot 99965.*

Modelling of the heating deposition shows that $D_{bulk}$ and $D_{NBI}$ accelerated by 4 MW n=1 D ICRH form tails up to 250 keV, which boost the fusion reactions and induce the large neutron and α modulation, but do not have long slowing down times to justify the observed $T_e$ delay. This indicates that the large phase delay observed in central $T_e$ can only be due to α heating. Integrated modelling of the discharge indeed reproduces the experimental results only when α heating is included in the source terms. The estimated time-averaged amount of electron α heating is ~2 MW in the first phase of the pulse and its modulation amplitude ~ 0.5 MW.

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# Characterization of the edge turbulence and electron profiles in TCV tokamak with the Thermal Helium Beam diagnostic

M.Agostini[1], M.Ugoletti[1], M.La Matina[1], Y.Andrebe[2], B.P.Duval[2], N.Offeddu[2], C.Theiler[2], Y.Wang[2] C.Wüthrich[2], S.Coda[2], C.Piron[3] and the TCV team[3]

[1] *Consorzio C.so Stati Uniti 4, 35127 Padova, Italy*
[2] *EPFL, Swiss Plasma Center (SPC), CH-1015 Lausanne, Switzerland*
[3]*Unità Tecnica Fusione, C.R. ENEA Frascati, CP65, 00044 Frascati, Italy*

A Thermal Helium Beam diagnostic, recently installed on TCV, allows measurement of the radial profiles of electron density and temperature at the outboard midplane edge and Scrape Off Layer, together with their high frequency turbulent fluctuations up to 200 kHz. The edge region is studied for a range of L-mode discharges, with different heating schemes: purely ohmic, with Neutral Beam injection and/or Electron Cyclotron heating. Edge profiles are strongly modified by auxiliary heating, and the changes in the gradients of the electron density and pressure are linked to those in the turbulence behaviour: fluctuation levels and blob dimensions are found to be strongly correlated with the characteristic radial lengths of the profiles.

A comparison with H-mode discharges is also discussed, in terms of inter-ELM turbulence and local gradients.

*[3] See author list of "H. Reimerdes et al 2022 Nucl. Fusion* ***62*** *042018*

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# Effect of approximations to the polarization equation in full-f gyrofluid turbulence simulations

B.P Somu, A. Kendl

Institute for Ion Physics and applied Physics, University of Innsbruck, Innsbruck, Austria 6020

In full-f gyrokinetic and gyrofluid models, which treat the evolution of the full distribution function instead of small fluctuations, the gyrocenter densities are coupled to the electric potential through the polarization equation. This takes the form of a generalized Poisson problem, $\varphi\nabla N\cdot\nabla = Q$, whose solution for $\varphi(x,y)$ in the 2D drift plane can be computationally expensive to obtain.

Turbulence codes often employ “linearized” approximations to this problem, such as assuming a constant and/or static polarization density ‘N’ instead of a dynamically fully evolving ‘N(x,y,t)’.

Here we investigate the effects of such approximations in a full-f gyrofluid model. Results are first obtained for 2D gyrofluid Hasegawa-Wakatani resistive drift wave turbulence simulations, and the impact of the approximations on transport for various tokamak edge parameters is studied.

The status of the extension of the underlying full-f full-k gyrofluid code TIFF to 3D electromagnetic field-aligned simulations in tokamak edge/SOL geometry is reported.

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# Hysteresis in gyrofluid resistive drift-wave zonal-flow turbulence

F. Grander, F.F. Locker, A. Kendl

Institut für Ionenphysik und Angewandte Physik, Universität Innsbruck, 6020 Innsbruck, Austria

Finite Larmor radius (FLR) and hysteresis effects on drift wave turbulence, transport and zonal-flow interactions are investigated within a paradigmatic modified Hasegawa-Wakatani model by means of a δf isothermal gyrofluid code.

In the limit of cold ions (without FLR effects) previous results (Numata et al, Phys. Plasmas 14, 102312, 2007) on the transition between turbulence dominated and flow dominated states are reproduced. The impact of gyrofluid FLR effects (warm ions) on the transition is studied.

The observed characteristics similar to a state bifurcation with the resistive coupling coefficient as a control parameter leads to the question if a hysteresis behaviour can be observed, with differences in the forward and backwards transitions between turbulence and flow states. This is related to the problem of possible dependence of the turbulence and flow system on initial conditions.

Here we show by dynamical variation of the resistive coupling parameter throughout extended simulations, that different zonal flow modes, and in relation different transport levels, can be achieved, depending on the history or initial conditions of the numerical experiment. The changes in the zonal flow states are found to be more pronounced in the presence of FLR effects.

The status of generalization to full-f full-k gyrofluid simulations with the code TIFF, and extension to 3D thermal field-aligned simulations in tokamak edge/SOL geometry is reported.

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# Gyrokinetic simulations on the triggering and self-sustaining of internal transport barrier in HL-2A tokamak plasmas

W.H. Lin[1], J. Garcia[2], J.Q. Li[1], S. Mazzi[2], Z.J. Li[1], X.X. He[1]

[1]Southwestern Institute of Physics, Chengdu 610041, China

[2]CEA, IRFM, 13108, Saint-Paul-lez-Durance, France

**Abstract**

Internal transport barriers (ITB)[1], characterized by steep pressure gradients and high temperatures and densities near the core region of tokamak, provide a promising way to the reduction of energy loses and improvement of plasmas confinement. The formation of ITB have been exposed to extensive studies and various mechanisms were proposed such as the E × B shearing stabilization. But nevertheless, there are aspects that remain unclear, e.g. whether a single mechanism or multiple ones are responsible for the ITB triggering and whether such mechanisms continue to play their role on the ITB sustainment once it is fully formed. To clarify these aspects, the state-of-the-art gyro-kinetic code GENE [2] is utilized to perform non-linear simulations for the ITB discharge #22453 [3] in HL-2A tokamak, with a particular emphasis placed on the individual effects of finite-β, fast ions and E × B shear. A new paradigm for the ITB formation is proposed in which different physics mechanisms play a different role depending on the ITB formation stage. It is confirmed that the fast ion dilution effect, arising from increase of fast ion fraction due to Neutral Beam Injection (NBI), is capable of stabilizing the Ion Temperature Gradient (ITG) modes, reducing the ion fluxes to a low value and finally triggering ITB in HL-2A. However, such a mechanism plays a minor role once ITB is fully formed. We define the concept of ITB self-sustainment, as the ITB is supported by an interplay of zonal flows and electromagnetic modes with low toroidal number in the presence of finite-β and steep pressure gradients once the ITB is triggered. Unlike turbulent transport reduction by E × B shearing, which is not found to play a role on the ITB formation in our work, the mechanism found in this study could be important in future tokamak devices as long as finite-β effects are strong enough.

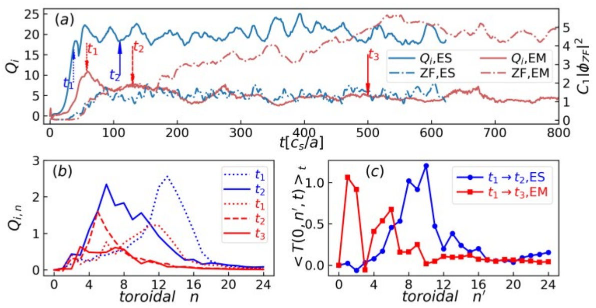


Figure. 1 (a) Time trace of the total ion fluxes and zonal field energy, labelled by $Q_i$ and ZF respectively, for the cases at 650 ms with with and without finite-β, labelled by EM and ES respectively. (b) Flux spectrum at the normalized time marked in (a). (c) Time-averaged nonlinear term contribution from each toroidal mode to the growth of ZF energy for the EM and ES

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