PET16

16th International Workshop

on Plasma Edge Theory in Fusion Devices

27-29 September 2017

Le Pharo

Marseille, France

Program and Abstracts

PET16

Program and abstracts

Workshop Schedule

	26-sept	27-sept	28-sept	29-sept
08:30				
08:45		Registration		
09:00			I3: M. Hoelzl	I5: M. Dorf
09:15				
09:30		Workshop opening		
09:45			O6: M. Hamed	O10: K. Hoshino
10:00		I1: S. Brezinsek		
10:15			07: M. Hosokawa	O11: G. Ciraolo
10:30				
10:45		O1: S. Carli	Coffee break	Coffee break
11:00				
11:15		O2: L. Casali	I4: P. Tamain	Poster Session P2
11:30				
11:45		O3: K. Galazka		
12:00			O8: A. Ross	
12:15		Lunch		
12:30			O9: N. Fedorczak	
12:45				
13:00			Lunch	Lunch
13:15	Registration			
13:30				
13:45				
14:00				
14:15	ITER visit	Poster session P1		
14:30			Tour	
14:45				I6: W. Dekeyser
15:00				
15:15				
15:30				O12: X. Bonnin
15:45				
16:00				O13: D. Coster
16:15		Coffee Break		
16:30		I2: K. Ibano		O14: G. Giorgiani
16:45				
17:00				Workshop closing
17:15		04: E. Marenkov		
17:30				
17:45		05: S. Togo		
18:00				
18:15				
18:30				
18:45				
19:00	Welcome Reception			
19:15				
19:30			Banquet	
19:45				
20:00				
20:15				
20:30				
20:45				

(I: Invited, O: Oral)

Committees and contact

International Scientific Committee

R. Zagórski (Chair)	IPPLM, Warsaw, Poland
Y. L. Igitkhanov	KIT, Karlsruhe, Germany
I. Joseph	LNLL, Livermore, USA
M. Kobayashi	NIFS, Japan
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D. Kh. Morozov	RCC Kurchatov Institute, Moscow, Russia
Y. Marandet	CNRS-Aix-Marseille University, France
V. Rozhansky	SPbPU, St.Petersburg, Russia
M. Yagi	QST, Rokkasho, Japan
D. F. Düchs (Hon.)	Münich, Germany

Local Organizing Committee (LOC)

Y. Marandet (Chair)	PIIM, CNRS-Aix-Marseille Univ.
P. Beyer	PIIM, CNRS-Aix-Marseille Univ.
G. Ciraolo	IRFM, CEA Cadarache
D.-M Fan	PIIM, CNRS-Aix-Marseille Univ.
P. Genesio	PIIM, CNRS-Aix-Marseille Univ.
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Workshop website

https://pet16.sciencesconf.org

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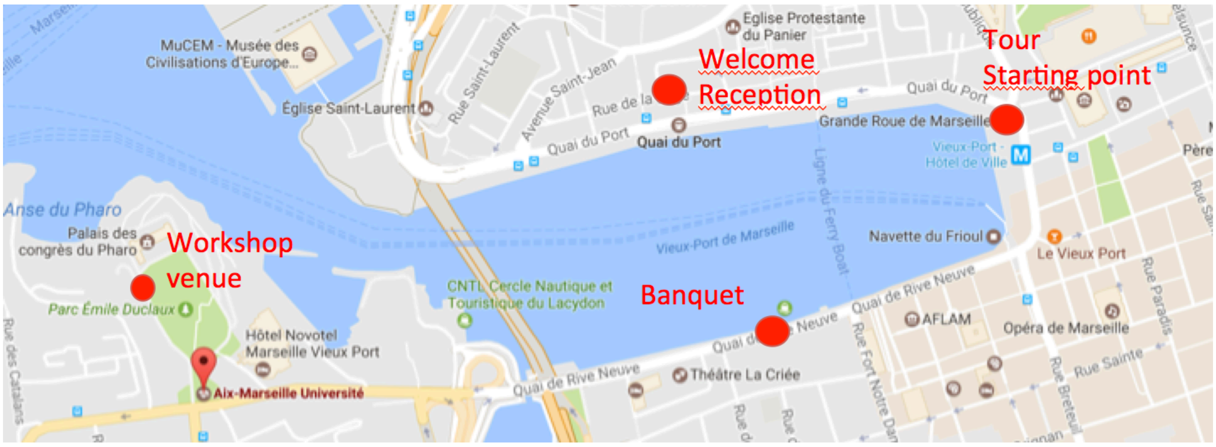
Maps

General map



Entrance of the Pharo park (58 Boulevard Charles Livon)

Workshop venue (Aix-Marseille Université)

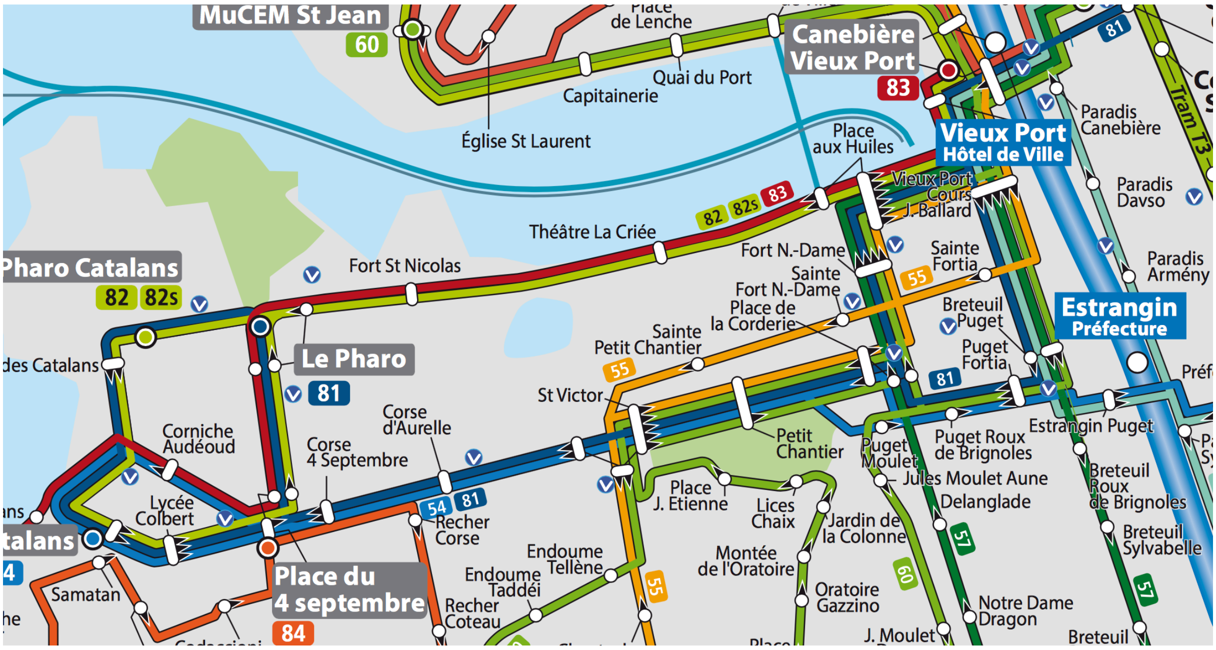
Transportation

From the metro station Vieux Port, the Pharo park can be reached by bus lines **82**, **82s** and **83** (bus stop close to the *ombrière,* see below).

From the Pharo Park, you can reach the Vieux port with lines **81**, **82**, **82s** and **83**. You may prefer taking a walk (20’), especially in case of traffic jam.

Tickets can be purchased from vending machines in metro stations or in the main bus stops, i.e. not at the Pharo bus stop (10 trips, 1€37/trip or a XL pass 72h for 10€80; tickets bought from bus drivers are more expensive 2€/trip). Once validated in the bus/metro/tram tickets are valid for 1 hour (back and forth,…), but do not forgot to validate every time you connect.

For those of you coming by car, there is a public parking at the Pharo, where you can book a parking spot online.



The ombrière du Vieux Port

PROGRAM

Tuesday September 26th

13:00 Registration

14:00 Departure for the visit of ITER (bus in front of the Pharo Park entrance)

19:00 Welcome reception (City Hall, Quai du Port)

Wednesday September 27th

8:30 Registration

9:15 Workshop opening

9:45 **I1** S. Brezinsek

Quasi-steady-state plasma operation in the Be/W material mix: from the JET tokamak to the ITER reactor

10:30 **O1** S. Carli

Effects of strike points displacement on the ITER tungsten divertor reflector plate heat loads

11:00 **O2** L. Casali

Modelling the effect of divertor closure on detachment onset in DIII-D with the SOLPS code

11:30 **O3** K. Galazka

Multiple impurity seeding for power exhaust management in JT-60SA tokamak with carbon divertor

12:00 Lunch

14:00 Poster Session P1

15:45 Coffee break

16:15 **I2** K. Ibano

Simulation study on the vapor shielding at solid walls under transients heat loads using weighted particle model

17 :00 **O4** E. Marenkov

On the radiation transport in inhomogeneous plasmas

17:30 **O5** S. Togo

SOL-divertor plasma simulation based on a generalized fluid model incorporating ion temperature anisotropy and mirror effect

18:00 Adjourn

Thursday September 28th

8:45 **I3** M. Hoelzl

What non-linear simulations can teach us about ELM physics

9:30 **O6** M. Hamed

Curvature effect on the micro-tearing mode stability

10:00 **O7** M. Hosokawa

Kinetic modelling of divertor fluxes between and during ELMs in a COMPASS-like tokamak plasma

10:30 Coffee break

11:00 **I4** P. Tamain

Impact of magnetic geometry and X-point configuration on edge plasma turbulence and transport in 3D first principle simulations

11:45 **O8** A. Ross

Non-Boussinesq turbulence studies in the SOL

12:15 **O9** N. Fedorczak

Width of turbulent SOL in tokamaks: from circular geometry to diverted ones

12:45 Lunch

13:45 Departure for excursion

19:30 Banquet

Friday September 29th

8:45 **I5** M. Dorf

Continuum kinetic modeling of axisymmetric plasma transport at the edge of a divertor tokamak

9:30 **O10** K. Hoshino

Multi-impurity divertor simulations using a Monte-Carlo kinetic impurity transport model

10:00 **O11** G. Ciraolo

Kinetic and fluid modelling of non-local parallel heat transport in magnetic fusion devices

10:30 Coffee Break

11:00 Poster Session P2

12:45 Lunch

14:30 **I6** W. Dekeyser

Divertor design through adjoint approaches and efficient code simulation strategies

15:15 **O12** X. Bonnin

Current SOLPS-ITER physics developments and activity

15:45 **O13** D. Coster

Characterization of oscillations observed in reduced physics SOLPS simulations

16:15 **O14** G. Giorgiani

A new high-order fluid solver for tokamak edge plasma transport simulations based on a magnetic-field independent discretization

16:45 Workshop closure

17:00 Adjourn

POSTER SESSION P1: Wednesday 27th, 14:00-16:00, Salle des Voutes

Board #	Paper #	First Author	Title
1	**I1**	S. Brezinsek	Quasi-steady-state plasma operation in the Be/W material mix: from the JET tokamak to the ITER reactor
2	**I2**	K. Ibano	Simulation study on the vapor shielding at solid walls under transients heat loads using weighted particle model
3	**O1**	S.Carli	Effects of strike points displacement on the ITER tungsten divertor reflector plate heat loads
4	**O2**	L. Casali	Modelling the effect of divertor closure on detachment onset in DIII-D with the SOLPS code
5	**O3**	K. Galazka	Multiple impurity seeding for power exhaust management in JT-60SA tokamak with carbon divertor
6	**O4**	E. Marenkov	On the radiation transport in inhomogeneous plasmas
7	**O5**	S. Togo	SOL-divertor plasma simulation based on a generalized fluid model incorporating ion temperature anisotropy and mirror effect
8	**P1-01**	H. Xie	Simulation of impurity behavior in EAST tokamak with the integrated COREDIV code
9	**P1-02**	S. Islam	Numerical simulation study towards plasma detachment in the end cell of GAMMA 10/PDX by a coupled fluid-neutral code
10	**P1-03**	R. Zagorski	Modelling of JET DT experiments in ILW configurations
11	**P1-04**	R. Chmielewski	TECXY simulations of multi-species impurity seeding in DEMO reactor
12	**P1-05**	E.T. Meier	Drifts effects and up-down asymmetry in balanced double-null DIII-D divertor configurations
13	**P1-06**	C. Norscini	First modelling of edge plasma density regimes in the COMPASS tokamak
14	**P1-07**	F. Subba	Analysis of highly radiative scenarios for the EU-DEMO divertor target protection
15	**P1-08**	S. Baschetti	Plasma turbulence reduction with a two field k-epsilon model for L-mode transport simulations with SOLEDGE2D-EIRENE
16	**P1-09**	H. Bufferand	Study of the impact of magnetic geometry on power exhaust with the transport code SOLEDGE2D-EIRENE
17	**P1-10**	A. Khan	WallDYN simulations of beryllium migration in ITER
18	**P1-11**	Y. Hayashi	Modeling of the linear plasma device NAGDIS-II with neutral gas puffing and pumping by using EMC3-EIRENE
19	**P1-12**	R. Mao	Plasma simulations of complex HL-2M divertor geometries using SOLEDGE2D-EIRENE edge plasma transport code
20	**P1-13**	K. Jesko	Soledge2d-EIRENE simulations of linear plasma devices Pilot-PSI and Magnum-PSI - a comparison with experimental data
21	**P1-14**	F. Subba	Advanced divertor configurations for DEMO
22	**P1-15**	M. Wigram	UEDGE Modeling of detached divertor operation for long-leg divertor geometries in ARC
23	**P1-16**	V. Rozhansky	Electric field and currents in the detached regime of a tokamak
24	**P1-17**	M. Shoji	Investigation of dust shielding effects by intrinsic ergodic magnetic field line structures in the peripheral plasma of the large helical device
25	**P1-18**	E. Sytova	Impact of a new general form of friction and thermal forces on SOLP-ITER modeling results
26	**P1-19**	K. Okamoto	Modeling of plasma and its wall interaction for long term tokamak operation
27	**P1-20**	S. Kajita.	Ignition and erosion of materials by arcing in fusion relevant conditions
28	**P1-21**	A. Fil	Study of how detachment characteristics are affected by the use of alternative divertors with SOLPS-ITER and SD1D
29	**P1-22**	J. Rosato	Spectroscopic models for tokamak edge and divertor plasma diagnostics
30	**P1-23**	R. Sheeba	Modeling of hydrogen line and continuum emission spectra of detached divertor plasmas
31	**P1-24**	M. Koubiti	A prospective spectroscopic study of hydrogen and impurity pellets in magnetic fusion devices
32	**P1-25**	J. Guterl	Modeling and analysis of tungsten sourcing in the outer divertor during the DIII-D metal tile campaign
33	**P1-26**	A. Holm	Assessing the ion-electron thermal equilibration in the SOL of tokamaks using UEDGE
34	**P1-27**	S. Pandya	Feasability study of possible runaway diagnostic methods in the edge plasma region region of ITER
35	**P1-28**	P.-S. Verma	Sensitivity of coupled plasma fluid/neutral kinetic edge simulations to the plasma wall interface description: effects of cyclotron orbits, sheath physics and surface roughness
36	**P1-29**	M. Meireni	Line shapes as a probe of turbulent plasmas
37	**P1-30**	R. Stamm	Possible spectroscopic signature of wave collapse in an edge plasma

POSTER SESSION P2: Friday 29th, 11:00- 12:45, Salle des Voutes

Board #	Paper #	First Author	Title
1	**I3**	M. Hoelzl	What non-linear simulations can teach us about ELM physics
2	**I4**	P. Tamain	Impact of magnetic geometry and X-point configuration on edge plasma turbulence and transport in 3D first principle simulations
3	**I5**	M. Dorf	Continuum kinetic modeling of axisymmetric plasma transport at the edge of a divertor tokamak
4	**I6**	W. Dekeyser	Divertor design through adjoint approaches and efficient code simulation strategies
5	**O6**	M. Hamed	Curvature effect on the micro-tearing mode stability
6	**O7**	M. Hosokawa	Kinetic modelling of divertor fluxes between and during ELMs in a COMPASS-like tokamak plasma
7	**O8**	A. Ross	Non-Boussinesq turbulence studies in the SOL
8	**O9**	N. Fedorczak	Width of turbulent SOL in tokamaks: from circular geometry to diverted ones
9	**O10**	K. Hoshino	Multi-impurity divertor simulations using a Monte-Carlo kinetic impurity transport model
10	**O11**	G. Ciraolo	Kinetic and fluid modelling of non-local parallel heat transport in magnetic fusion devices
11	**O12**	X. Bonnin	Current SOLPS-ITER physics developments and activity
12	**O13**	D. Coster	Characterization of oscillations observed in reduced physics SOLPS simulations
13	**O14**	G. Giorgiani	A new high-order fluid solver for tokamak edge plasma transport simulations based on a magnetic-field independent discretization
14	**P2-01**	M. Kobayashi	Temporal evolution of edge Te and ne profiles during detachment transition with and without RMP application in edge stochastic layer of LHD
15	**P2-02**	J. Artola	Non-linear MHD simulations of ELM trigerring via vertical kicks with JOREK-STARWALL
16	**P2-03**	M. Yagi	Nonlocal response of density and temperature fluctuations due to edge perturbation in tokamak plasmas
17	**P2-04**	D. Galassi	Spontaneous transport barrier buid-up in 3D global turbulence simulations of a diverted plasmas
18	**P2-05**	A. Fukuyama	Modelling of LH transition using the fluid-type transport code TASK/TX
19	**P2-06**	P. Paruta	Implementation of X-point configurations into the GBS code
20	**P2-07**	C. Baudoin	Drift driven vs turbulent heat transport in 3D edge plasma simulations
21	**P2-08**	W. Gracias	Analysis of key factors affecting filament dynamics in tokamak scrape-off layers using the TOKAM3X model
22	**P2-09**	N. Nace	Effect of safety factor and magnetic shear on edge turbulent transport and poloidal asymmetries
23	**P2-10**	D. M. Fan	Effect of particle fueling and recycling on the properties of SOL and Edge turbulent fluctuations in global TOKAM3X-EIRENE simulations
24	**P2-11**	A. Tanaka	A Coulom collision model for weighted particle simulations with energy and momentum conservation
25	**P2-12**	P. Migliano	An improved approximation for the analytical treatment of plasma kinetic linear instabilities in toroidal geometry
26	**P2-13**	Y. Homma	An extended kinetic model for the thermal force on impurity particles in relatively lower collisional plasmas
27	**P2-14**	W. Lee	Verification of 5D continuum gyrokinetic code COGENT: studies of kinetic drift wave instability
28	**P2-15**	Ph. Ghendrih	Electron burst driven by near electric field effects of lower hybrid launchers
29	**P2-16**	L. Chôné	Improved boundary condition for full-f PIC gyrokinetic simulations of circular limited tokamak plasmas in ELMFIRE
30	**P2-17**	Y. L. Li	Hot spot induced by LHCD in the shadow of antenna limiters in the EAST tokamak
31	**P2-18**	R. Tatsumi	Development of a Lagrange-Monte Carlo scheme for fluid modeling of SOL/Divertor plasmas
32	**P2-19**	M. Baeten	Identification of stochastic noise propagation in plasma edge simulations
33	**P2-20**	K. Ghoos	Accuracy and convergence of iteratively solved Monte Carlo codes for simulations in the plasma edge of nuclear fusion reactors
34	**P2-21**	N. Horsten	Hybrid neutral models for a detached ITER case
35	**P2-22**	B. Mortier	Enforcing conservation at Monte Carlo level in a coupled Finite Volume-Monte Carlo simulation
36	**P2-23**	S. Van den Kerkhof	Towards numerical optimization of novel magnetic magnetic topologies
37	**P2-24**	M. Blommaert	Implementation of a consistent fluid neutral model in SOLPS-ITER and benchmark with EIRENE for detached divertor conditions
38	**P2-25**	M. Valentinuzzi	Comparison between fluid, kinetic and hybrid descriptions for neutrals in the SOLEGDE2D edge plasma code
39	**P2-26**	T. Maeda	Analysis of the plasma blob formation and transport, and its effects on impurity transport in SOL Regions

Abstracts

I-01

Quasi-steady-state plasma operation in the Be/W material mix: from the JET tokamak to the ITERreactor

S. Brezinsek[1], S. Wiesen[1], D. Borodin[1], K. Schmid[2], and JET contributors

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ITER will operate with a metallic first wall and employ plasma-facing components made of Beryllium at the main chamber and Tungsten in the divertor. The positive aspects of the envisaged material mix for ITER, in particular low fuel retention, low erosion and material migration, low dust production, have been confirmed in JET equipped with the ITER-like wall JET-ILW [1], but plasma-surface interaction at these metallic components has shown vital impact on the operational space of the tokamak as well as on the plasma performance [2]. It should be noted that the unseeded plasma purity is very high in the metallic environment (Z_{eff}=1.2) and it is possible to conclude that the metallic wall revealed almost the pure plasma physics in the tokamak. In contrast, the previous performed plasma operation with carbon-based first wall, JET-C, effectively presented the plasma performance and plasma-surface interaction with effective intrinsic low-Z (C) seeding with beneficial impact on the plasma performance and divertor radiation, but inacceptable conditions on retention, migration, and dust production. However, comparing plasmas in the two machine configurations, JET-C and JET-ILW, helps to unmask the underlying plasma edge and plasma-surface interaction physics and helps to benchmark major edge and divertor codes used for ITER such as SOLPS-ITER [3], WallDYN [4] and ERO [5]. Moreover, impurity seeding with other low Z species as well as the variation of the divertor magnetic configuration, which both lead to recovery of the plasma performance help to decouple the role of radiation/dilution and fueling/recycling. In particular, a two weeks period of identical well-diagnosed plasma discharges in H- mode conditions accumulating more than 900s of plasma time and more than 30 000 ELMs can be used as perfect test bed to mimic ITER-like plasma discharge conditions which can be expected at half magnetic field operation as foreseen in the start sequence. In addition to the extended in-situ plasma information with statistical analysis of multiple quantities in the quasi-steady long pulse also post-mortem tile analysis information are available for the specific magnetic configuration. This unique set of information allows also the test of physics models related to specific questions like the fuel outgassing from Be and W surfaces, the prompt re-deposition of eroded W, local ELM-material modelling etc. as well as the compatibility and agreement with global modelling assumptions.
In addition to the benchmark of major codes also specific physics questions such as local ELM interaction, outgassing are addressed. Detailed analysis of discharges as well as an overview of associated modelling will be presented and critical revisited in view of ITER predictions.

[1] S. Brezinsek et al., Journal of Nuclear Materials Vol. 463 (2015) 11
[2] M. Beurskens et al., Nuclear Fusion Vol. 54 (2014) 043001
[3] S. Wiesen et al.,Journal of Nuclear Materials Vol. 463 (2015) 480
[4] K. Schmid et al., Nuclear Fusion Vol. 55 (2015) 053015

Simulation study on the vapor shielding at solid walls under transient heat loads using weighted particle model

K. Ibano[1], A. Tanaka[1], S. Togo[2], H. Tae Lee[1], Y. Ueda[1], T. Takizuka[1]

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Erosion of plasma facing components (PFCs) is a major concern for the fusion reactor. Huge transient heat loads by ELM and disruption especially cause large erosions, and reduce the lifetime of PFCs. It is a challenge to predict the net erosion caused by these transient heat loads. There are strong interactions among incoming plasma flux, PFCs, and generated impurities. In order to study such strong nonlinear interactions, we have been developing a simulation code based on the particle-in-cell (PIC) method, and analysing the vapor shielding effects [1,2]. The code treats multi-species plasma by the PIC method with weighted particles. Density fraction of emitted impurities compared to the background plasma is varied from a very small amount << 1 to sometimes a very large amount >> 1. The weighted model is really indispensable to study the vapour shielding at PFCs. The wall temperature is followed by a heat transfer model. Energies of incoming plasma particles at the wall boundaries are counted as the heat flux and used for the boundary condition of the transient heat transfer calculation. With the increase of wall temperature, the material is melted and evaporated, and neutral impurities of the vapour are generated. In addition to the evaporation, plasma surface interactions, i.e., reflection, sputtering, thermo- and secondary-electron emission, are also included. Atomic interactions of the wall originated impurities with the incoming plasma are treated using the OPEN-ADAS library data. Transient behaviours of charged impurities in the plasma are simulated self-consistently by the PIC model, including the sheath and pre-sheath electric field. The present paper extends the previous works, and studies on the erosion behaviour including the vapor shielding on the wall plate during the transient loads. The comparison of simulation results with the basic experiments in linear devices will be shown, and the application of the code to the fusion plasmas as well. Summaries of physical phenomena during the wall evaporation and estimated erosion amounts per events with vapor shielding will be presented.

[1] K. Ibano et al., Contrib. Plasma Phys. 56 (2016) 705.
[2] K. Ibano et al., to be published in Nucl. Mater. Energy.

What non linear simulations can teach about ELM Physics

M. Hoelzl[1], F. Orain[1], G.T.A. Huijsmans[2,3], S. Pamela[4], J. Artola Such[5], M. Becoulet[2], A. Lessig[1], F. Liu[2], S. Futatani[6], S. Guenter[1], K. Lackner[1], I. Krebs[7], R. Wenninger[1], E. Wolfrum[1], F. Mink[1], T. Eich[1], E. Viezzer[1], M. Dunne[1], B. Vanovac[8], L. Frassinetti[9], E. Trier[1], the ASDEX Upgrade Team[1] and the EUROfusion MST1 Team*

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** See H. Meyer et al., Nuclear Fusion FEC 2016 Special Issue (2017)*

Edge localized modes (ELMs) are repetitive instabilities observed close to the boundary of H-mode plasmas in the region of large pressure gradients and current densities leading to a fast collapse of pedestal density and temperature profiles. During an ELM crash, parallel transport along open magnetic field lines leads to strong localized heat fluxes onto divertor targets expected to be beyond engineering limits for future machines like ITER. In the present contribution we emphasize the role of non-linear simulations for advancing the understanding of ELM physics as well as of methods for ELM mitigation, suppression, or avoidance.
The results are obtained with the fully implicit finite element code JOREK [1] which solves extended magneto-hydrodynamic equations in realistic tokamak X-point geometry including main plasma, scrape off layer, and private flux region. The model includes divertor boundary conditions [2], two-fluid and neoclassical effects [3] and a resistive wall model [4].
Simulations are approaching the stage where good quantitative agreement can be obtained for existing experiments in many respects. It is shown how modeling advances insights into ELM physics in a complementary manner to experiments by allowing to study aspects not accessible otherwise or enabling additional analysis. Reliable predictive simulations become more and more feasible. Nevertheless, remaining limitations are discussed as well – essentially the work plan for further modeling enhancements. Simulations of ELM crashes are compared to various experimental diagnostics to gain confidence in the modeling and complement experimental approaches. The role of plasma flows and of non-linear coupling of toroidal harmonics for the spatial and temporal structure of ELMs is emphasized in particular. The penetration of error-fields produced by external 3D coils is discussed, and the means by which these perturbation fields can lead to ELM mitigation or suppression. ELM triggering by "vertical magnetic kicks" and pellet injection as well as natural ELM-free states are addressed as well.

[1] GTA Huysmans, O Czarny, NF 47, 659 (2007)
[2] GTA Huijsmans, A Loarte, NF 53, 123023 (2013)
[3] F Orain et al, PoP 20, 102510 (2013)
[4] M Hoelzl et al, JPCS 401, 012010 (2012)

Impact of magnetic geometry and X-point configuration on edge plasma turbulence and transport in 3D first principle simulations

P. Tamain[1], D. Galassi[2], C. Baudoin[1], H. Bufferand[1], G. Ciraolo[1], N. Fedorczak[1], Ph. Ghendrih[1], Y. Marandet[3], N. Nace[1] and E. Serre[2]

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Transverse transport in the edge plasma of tokamaks is one of the key determinants of both power and particle exhaust – through its impact on the Scrape-Off Layer (SOL) width – and global plasma confinement –through the development of the H-mode transport barrier. Several transport mechanisms are likely to contribute, among which neoclassical transport, large scale steady drift flows and turbulent transport. The comprehensive modelling of these phenomena requires flux-driven turbulence codes in which no scale separation is made between the equilibrium and the fluctuations and including the key geometrical features of the edge plasma. Such modelling tools are currently being developed by several teams but the effort has until now focused almost exclusively on simple limited plasma in idealized circular geometry. In this contribution, we report on edge turbulence and transport in numerical simulations in realistic diverted geometry. Simulations are performed using the TOKAM3X 3D fluid turbulence code which can handle arbitrary axisymmetric magnetic geometries, e.g. limited, single null or double null plasmas. Simulations performed in JET-like and COMPASS-like magnetic equilibria are analyzed and compared to the more classical circular limited simulations. The key features of turbulent transport in circular geometry are recovered in diverted plasmas. In most of the domain, transverse transport is dominated by fluctuation-driven ExB fluxes. Fluctuations are strongly field-aligned but the parallel wave vector is non zero and large poloidal asymmetries exist both in the radial and poloidal directions. The fluctuation level and transverse transport are ballooned around the outboard mid-plane leading to quasi-sonic asymmetric parallel flows. The shaping of the magnetic surfaces, especially the flux-expansion, is found to play an important role in the level of turbulent transport and its spatial distribution. Nevertheless 2 important differences are highlighted between limited and diverted configurations. First, the X-point significantly perturbs turbulence and flows in its vicinity. It is in particular responsible for a local damping of turbulence, effectively decorrelating the near SOL in the divertor from the main near SOL. In the far SOL however, turbulent filaments from the mid- plane extend into the outer leg of the divertor with a pattern similar to that observed in experiments. The inner divertor remains quiescent, while the private flux region exhibits strong relative fluctuation levels but associated with little radial transport. Large amplitude steady convective cells also develop around the X-point leading to time averaged fluxes of the same order as turbulent fluxes in the outboard mid-plane. The second difference is the systematic observation of an edge transport barrier which is not present in similar conditions in circular limited simulations. Turbulent transport drops by more than 60% in a region a few Larmor radii wide located just inside the separatrix. As consequence a density pedestal builds up. The transport barrier is associated with a local ExB shear. Through careful analyses of the charge and momentum balances and dedicated simulations, we will discuss the origin of the transport barrier and the reason why it is observed only in diverted configuration.

Continuum kinetic modeling of axisymmetric plasma transport at the edge of a divertor tokamak*

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The first continuum kinetic calculations are presented for the axisymmetric cross-separatrix transport of plasma at the edge of a tokamak, including self-consistent variations of an electrostatic potential. The model solves the full-F gyrokinetic equations for ion species coupled to the long-wavelength limit of a vorticity equation for the electric field. Ion-ion Coulomb collisions are described by employing the full nonlinear Fokker-Plank operator. The simulations are performed with a high-order finite-volume code COGENT [1-2] developed by the Edge Simulation Laboratory collaboration. The code is unique in that it is based on a consistent high- order discretization and interpolation of the underlying equations, and hence the error (in particular near the X-point) can be bounded. It utilizes multiblock grid technology whereby the coordinate surfaces of each block are aligned to magnetic flux surfaces everywhere except near the X-point, and a high-order interpolation is used to provide data communication in the region where the grid blocks overlap.

The paper also presents recent progress towards extending simulation time for collisional divertor plasmas. To that end, the Implicit-Explicit (IMEX) infrastructure, which enables implicit treatment of selected terms, has been implemented within the COGENT code. As a practical example, the IMEX scheme is applied to the Fokker-Plank operator demonstrating significant speed-up of COGENT calculations in strongly-collisional regimes. Finally, the status of 5D COGENT and progress toward the full edge turbulence code is described. In particular, the slab- geometry 5D version has recently become available and is successfully verified in simulations of the collisionless drift-wave instability that involve gyrokinetic equations for both ion and electron species coupled to the long-wavelength limit of the 3D gyro-Poisson equation.

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Divertor design through adjoint approaches and efficient code simulation strategies

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Divertor design remains a key challenge in the development of fusion reactors. Power loads to plasma-facing components have to be limited to tolerable levels, while maintaining sufficient helium exhaust capabilities and core plasma conditions compatible with plasma burn criteria. Usually, one has to resort to plasma edge codes to extrapolate the current understanding of divertor exhaust physics towards the operational windows anticipated in reactors. It is generally accepted that partially detached divertor operation will be required to mitigate heat loads to the divertor targets. However, the speed and convergence of the edge codes become serious bottlenecks in these highly-collisional regimes. In combination with a large number of design variables and uncertain model parameters, this turns computational divertor design into a challenging, time-consuming task. Such complex design problems can be tackled effectively through adjoint-based optimization techniques. These have proven their virtue among others in the field of aerodynamics and structural mechanics. In the past years, we have successfully developed these techniques for the design of divertor target shape and magnetic field, as well as the estimation of unknown model parameters [1]. However, we focused on somewhat simplified models, in particular using fluid neutral approximations. To enable the use of these modern computational techniques for more realistic design problems using complete edge plasma models, we need to tackle both the critical code-speed issue, as well as extend the optimization algorithms towards the inclusion of kinetic Monte Carlo codes for neutral and radiation transport problems. In this contribution, we provide an overview of recent results and research tracks towards achieving these goals. Based on a critical analysis of numerical error components in coupled Finite Volume (FV) / Monte Carlo (MC) code systems [2], we show how numerical parameters of the code system can be optimized to achieve a minimum computational time for a given error on the simulation. By averaging the noisy simulation results over many iterations with a comparatively small number of MC trajectories, we obtain speed-ups of up to an order of magnitude compared to current computational practice. A further gain can be achieved by using approximate fluid neutral models, or exact hybrid fluid-kinetic neutral models, which treat the equilibrium part of the neutral solution with a deterministic fluid model, while only relying on a so-called δf MC simulation to provide kinetic corrections [3]. Next, we investigate how accurate sensitivities with respect to design variables can be computed in the presence of statistical MC noise. We compare so-called continuous and discrete adjoint approaches for sensitivity analysis. In the former, a continuous transport equation for the sensitivity computation is derived, and solved using MC techniques. However, in this approach keeping the correlation between MC particles in forward and adjoint systems – essential to reduce the variance on the computed sensitivities – is very challenging. In the discrete approach, on the other hand, the sensitivities are directly computed through linearization of the forward MC trajectories. While very effective in reducing the variance, these discrete adjoint trajectories have to be simulated in reverse order compared to the forward MC problem, thereby introducing an additional computational or storage cost. The relative merits of both approaches are compared in view of divertor design. Moreover, we assess the possibility of using approximate sensitivity information based on fluid or hybrid neutral models.

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Effects of strike point displacement on the ITER tungsten divertor reflector plate heat loads

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The baseline ITER burning plasma equilibrium is designed to place the divertor strike points deep into the "V-shaped" region formed by the high heat flux handling vertical targets (VT) equipped with tungsten (W) monoblock (MB) technology (10 MWm^{-2} steady state) and the reflector plates (RP), armoured with flat W tiles and using hypervapotron cooling (5 MWm^{-2} steady state) [1]. This provides adequate margin on the VT for the required power spreading in the scrape-off layer (SOL), including the possible effects of 3D magnetic perturbations for ELM control, and allows for good neutral pumping. At the same time, there is sufficient distance to the RPs (~16 cm for the outer, ~11 cm for the inner) to avoid intersection with the private flux region plasma. The RPs thus receive only photonic and charge-exchange (CX) neutral heat fluxes under normal operation, typically amounting to ~0.5 MWm^{-2}. The divertor plasma performance under these conditions has been extensively studied (see e.g. [2]) with the SOLPS4.3 plasma boundary code suite. However, one specific issue which has not been addressed is the case in which loss of vertical control, or a deliberate requirement (e.g. a few W MBs damaged in the nominal strike point areas), imposes a strike point position further down the VTs, or even directly on the RPs, particularly at the inboard side which is intersected first in case of a rigid vertically downwards displacement. This paper examines the consequences of such strike point displacements, both through SOLPS-ITER plasma boundary simulations and thermal analysis of the RP structure, aimed at assessing the RP response to increased radiative, CX and thermal plasma power deposition.

Starting from a baseline coupled fluid-kinetic neutral solution (without fluid drifts), corresponding to Q_{DT} = 10 with neon seeding, P_{SOL} = 100 MW and nominal strike point positions, the equilibrium is then progressively moved downwards in a series of rigid displacements and new stationary solutions are obtained, up to a maximum displacement of ~11 cm. At this point, the inner strike point is well onto the inner RP and the outer strike point is only a few cm from the extreme lower end of the outer VT. Beyond this point, the separatrix intersects the inner dome wing. As the RP-VT corner is approached and the strike point intersects the inner RP, the configuration is switched from vertical to a more horizontal target on the inner side. The high degree of neutral self-interactions and recombination helps in maintaining plasma detachment at the inboard divertor, mitigating the heat load deposited onto the inner RP, while at the outboard divertor the plasma condition is not significantly affected by the downward displacements, nor are the power fluxes to the outer RP.SOLPS results provide the heat load profiles for a finite element analysis, taking into account the full cooling geometry, to assess the thermal response of the VTs and RPs under the conditions exploited in the displaced scenarios.

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Modelling the effect of divertor closure on detachment onset in DIII-D with the SOLPS code*

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Modeling with the fluid code SOLPS has reproduced experimental observations of detachment of the more closed DIII-D upper divertor at 20% lower upstream density than the open lower divertor and thereby demonstrate divertor closure's utility in widening the range of acceptable densities for adequate heat handling. To achieve reduced heat flux and erosion at the plasma facing components, future devices will need to operate in at least partially detached divertor conditions. 2D fluid plasma models coupled to Monte Carlo neutral transport simulations, such as SOLPS, have been widely used to predict the onset of detachment. However, a predictive modeling capability corroborated with experimental data remains a great challenge. In modeling the DIII-D discharges the cross-field transport coefficients are constrained to reproduce the experimental upstream profiles. For the open divertor on the lower shelf, once the experimental upstream profiles are matched, the code simultaneously reproduced quantitatively the experimental density and power and particle flux at the target. The more closed upper divertor has been modelled using the same input parameters and mesh refinement as the lower divertor model. The model shows that the increased closure of the upper divertor improve the trapping of neutrals thereby reducing the power density deposited at the target and facilitating the detachment process, detaching at a lower upstream density than the open, lower divertor.

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O-03

Multiple impurity seeding for power exhaust management in JT-60SA tokamak with carbon divertor

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The main aim of the JT-60SA project is to analyze the near-fusion plasma conditions for support of the ITER experiment on its way towards realization of energy production in DEMO. In our previous studies for scenario #2 (according to the JT-60 Research Plan, high auxiliary heating power, medium electron density [1]) it was concluded that with the intrinsic impurity of C and seeded impurities N, Ne, Ar and Kr it was very difficult to reduce the power delivered to the divertor plate (P_{plate}) effectively [2]. Significant dilution of the main plasma and high impurity concentrations were unavoidable. On the other hand, the results for the same scenario but for tungsten divertor show that the high-Z intrinsic W impurity radiation in the core can be exploited as an additional energy loss channel allowing for increased scrape-off layer (SOL) radiation and strong suppression of P_{plate} [3]. In this work the carbon divertor is reconsidered with 2 external impurities: a medium Z (Ar) and a high Z (Xe) to restore this effect. The integrated core-edge COREDIV code [4] is used to analyze the influence of different seeding scenarios with Ar and Xe on plasma conditions, especially for radiative power exhaust and impurity accumulation.The COREDIV code describes self consistently the core and the SOL with the divertor regions of tokamak plasmas. The coupling between the core and the edge is imposed by continuity condition at separatrix of values and fluxes of temperatures and densities, respectively. In the core the 1D transport equations with semi-empirical transport coefficients for densities and temperatures are used. Transport coefficients include the anomalous and neoclassical transport and their profiles have been modified for describing the transport barrier. In the SOL the 2D boundary layer code EPIT is used based on Braginskii-like equations for the background plasma and on rate equations for each ionization state of each impurity. The sputtering processes of target divertor plates are included in the model.

The results of simulations confirm the idea that the use of selected gas mixture (Xe, Ar) might solve the power exhaust problem in the low density, high power steady state JT-60SA discharges. The power delivered to the SOL is reduced due to Xe radiation in the central plasma, but remains remarkably larger than H-L transition threshold. Under certain conditions (proper impurity puff position and recycling properties) it is possible to dissipate a significant part of the remaining heat flux by Ar radiation so that the P_{plate} reaches 10 MW, a limit tolerable for materials considered for a divertor in future reactors (assuming the total divertor surface in order of 1 m^2).

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On the radiation transport in inhomogeneous plasmas

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Radiation transport in edge plasmas contributes both to energy fluxes coming to the first wall materials and plasma composition through effects of photon excitation, photon ionization etc. It is known that the divertor plasma in typical tokamak conditions is opaque for hydrogen Layman-alpha line carrying the most of the radiance. Another important case where radiation transport has to be treated carefully is shielding phenomena of the first wall materials. Intense particle and energy fluxes to which the divertor elements may be subjected during transient events such as ELMs or disruptions induce evaporation of the material. The vapor quickly becomes ionized and the resulting plasma serves as a shield for the target. The shielding plasma has strongly nonuniform density and temperature profiles influencing radiation transport.

There is a large number of numerical codes including radiation transport in edge plasma simulation. The SOLPS package uses EIRINE code for Monte-Carlo simulation of the process. However, due to high computational complexity of the task a number of assumptions are employed. For example, it is supposed that change of excited atoms population caused by photon absorption is negligible compared to electron impact sources. Photons are sampled only for a limited amount of frequencies not too far from the line center. Finally, spatial resolution of radiation transport calculation is limited to only several cells in the edge plasma. The codes aimed at simulation of plasma shielding phenomena such as FOREV, TOKES, HEIGHTS suffer from the similar issues. Sometimes even stronger approximations are taken for example the radiation transport is supposed to be diffusive.

In the present work we demonstrate influence of mentioned effects on radiation transport. We use a full collision radiation model to simulate photon transport in inhomogeneous plasma with parameters taken as fixed background values. Instead of using Monte-Carlo or diffusion approaches differential equations for radiance are used. We show that variation of the line profile shape with plasma parameters can dramatically change the transport, even reverse the total radiation flux comparing with fixed profile width. We also demonstrate that for plasmas with parameters relative for shielding effects the source of atoms excitation due to photon absorption is comparable or in some cases larger than the corresponding sources due to electron impact excitation. Finally, we estimate the role of plasma shielding within our model for lithium which is considered as one of the alternative first wall materials in fusion devices.

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SOL-divertor plasma simulation based on a generalized fluid model incorporating ion temperature anisotropy and mirror effect

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The scrape-off-layer (SOL)-divertor plasma code packages, such as SOLPS, SONIC, UEDGE, etc., used for designing divertors of future fusion reactors mainly apply the plasma fluid model known as the Braginskii equations [1]. In this model, Coulomb collisions are assumed to be frequent enough to relax the temperature anisotropy of the plasma. It has been pointed out by the kinetic particle model, however, that the ion temperature tends to be remarkably anisotropic in the open-field SOL plasma even for the middle collisionality of ν^*10 [2]. In such plasmas with anisotropic ion temperature (or ion pressure), mirror effects can ~no longer be neglected. In a typical tokamak with inverse aspect ratio of $\varepsilon\sim 1/3$, the ratio of the toroidal field at the inner midplane to that at the X-point becomes $1/(1-\varepsilon)\sim 3/2$. Then, the mirror force term in the parallel momentum balance equation $(p_{//} - p\perp)(\nabla_{//}B)/B$ can be the same order as the pressure gradient term $\nabla_{//}p_{//}$ and affects the SOL plasma flow profile. In addition, some advanced divertor concepts with a spreading magnetic field toward the divertor plate, such as Super-X divertor [3], can be influenced by the mirror effect.We have been developing a one-dimensional plasma fluid model with anisotropic ion temperature in a homogeneous magnetic field, and simulating SOL-divertor plasmas of medium-size tokamaks [4]. The relation between ν^* and the ion temperature anisotropy agreed well with the result of particle simulation [2]. Moreover, thanks to the virtual divertor model which treats the divertor plate and accompanying sheath region as sinks for the plasma, the Bohm condition was automatically satisfied and the supersonic plasma flow in the radiative-cooling divertor region was self-consistently realized [4].
In this study, we relax the assumption of the homogeneous magnetic field in order to incorporate the mirror effect based on the generalized plasma fluid model proposed in Ref. [5]. The impact of mirror effect on the SOL-divertor plasma profiles including the generation of super- sonic plasma flows and the plasma detachment will be investigated. We also apply this model to the divertor simulation plasma of the mirror device GAMMA 10/PDX, which has a large mirror ratio of ~10 and high temperature ions [6], in order to conduct a benchmark study of our plasma fluid model with the experimental results.

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Curvature effect on the Micro-Tearing Mode stability

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In tokmaks, high confinement regimes (H-modes) can be accessed by exceeding a threshold in heating power. The H-mode is characterized by the formation of a narrow insulating region just inside the last closed flux surface, called the pedestal. In the pedestal, the radial transport is reduced and a steep pressure gradient develops. Many models have been proposed to explain the origin and dynamics of the pedestal, with various degree of success, but several open questions remain. In particular, the role played by the micro-tearing instability in the residual turbulence at the top of the pedestal is to be clarified. Micro-tearing modes (MTMs) are known to generate a large electron heat flux by modifying the magnetic field lines topology at the ion Larmor radius scale and have recently been predicted to contribute to a significant fraction of the pedestal electron heat transport [1]. However, the linear mechanisms at play in the MTM destabilization are not well understood. Recent gyro-kinetic simulations using spherical [2, 3] and standard tokamaks [4, 3] parameters show the existence of MTMs in the collisionless regime in contrast to analytical calculations which predict MTMs to be stable in this regime [5, 6]. In the present work, this discrepancy is investigated. The analytical theory is first extended to include the magnetic field curvature which was neglected in previous calculations. It is shown that the curvature drift can lead to the destabilization of MTMs in the collisionless regime. Dedicated comparisons between the linear dispersion relation and numerical simulations are on going to test whether the curvature is indeed responsible for the destabilization of MTMs observed in the simulations.

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Kinetic modelling of divertor fluxes between and during ELMs in a COMPASS-like tokamak plasma

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Particle and energy fluxes to the plasma facing components (PFCs) during edge localized modes (ELMs) are expected to unacceptably shorten the lifetime of PFCs in ITER [1]. Non-linear MHD simulation of ELMs for ITER have shown that some aspects of empirical extrapolations, such as the broadening of the ELM power footprint at the divertor plate, may not apply at the ITER scale [2]. However, these findings are questionable because the particle and energy transport along the field lines in these MHD simulations are modelled in a fluid approximation. The ELM transport in the ITER SOL-divertor plasma is essentially collisionless given the high pedestal plasma temperature. In order to understand the consequences of kinetic effects on the power and particle fluxes to PFCs by ELMs, particle simulations with PARASOL [3] have been carried out. Initial 1-D simulations for ITER showed that the in/out asymmetry of the ELM divertor power/particle fluxes are strongly affected by the magnitude of the ELM energy loss and by the thermoelectric current flow [4]. In order to understand the 2-D aspects of the ELM energy flow to the divertor, PARASOL-2D simulations for a small-size tokamak were carried out in the stationary condition and during an ELM including both the effects of drifts and divertor recycling [5,6]. It was shown that: (i) the direction of the ion drift has a strong effect on the steady-state in/out heat/particle flux divertor asymmetries and that this effect was larger during an ELM, (ii) the energy load to the inner divertor during an ELM was generally larger for “normal” ∇B and smaller for “reversed” ∇B and robust to modelling assumptions (recycling value, in/out recycling ratio, ELM energy loss magnitude and plasma collisionality). However, other PARASOL-2D results were in contradiction with typical experimental findings, e.g., the in/out heat flux asymmetry for steady state conditions for “reversed” ∇B was $E_{in}/E_{out} < 1$, where E denotes energy deposition to the divertor targets. This and other PARASOL-2D predictions depend on assumptions on the anomalous transport between/during ELMs. In order to test and refine these assumptions, while maintaining a reasonable computational time, PARASOL-2D modelling of edge and divertor plasma conditions between/during ELMs has been performed for a range of plasma conditions and of modelling assumptions for COMPASS-like plasmas. The comparison of the PARASOL-2D simulation results with COMPASS experimental findings [7] and the consequences for kinetic modelling of ELMs will be described in the paper.

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Non Boussinesq turbulence studies in the SOL

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The understanding of turbulence in the scrape-off layer (SOL) is of major importance for future fusion devices. Turbulent transport of particles and heat out of the core plasma mainly affects the plasma confinement characteristics of the entire device. Large amplitude effects like laments and blobs may have a strong impact on the plasma-wall interaction.

In the edge of a tokamak, where the plasma has a low temperature and high collisionality, a fluid model is appropriate. The Braginskii set of equations is suitable for the description of the effects in the edge region, however it demands too heavy computational resources. Using the fact that plasma confinement is mainly determined by low frequency dynamics a drift reduced full-f model can be used from the Braginskii system more suited for numerical treatment [1].

In addition the Boussinesq approximation is often employed, which reduces the numerical and computational complexity significantly. However, its use becomes questionable in the SOL where large fluctuations compared to the background may appear [2]. Moreover, we will discuss that the Boussinesq approximation has subtle effects on the energy theorem of the model, and a conserved energy-like quantity is only obtained if the Boussinesq approximation is applied in a consistent way. Finally, we will also present numerical and computational techniques in order to relax the Boussinesq approximation.

The drift-reduced full-f system is implemented in GRILLIX, a plasma turbulence code able to treat diverted geometries, by using the flux-coordinate independent approach (FCI) [3]. We solve a 4-field model evolving the density, the electron temperature, the vorticity and parallel ion momentum. In order to understand the impact of the Boussinesq approximation a flux tube geometry is chosen as the computational domain. As a check for the numerics the conservation of energy is derived and tested within the code, both in the full-f and the Boussinesq system.

The turbulence in the SOL is studied in the full-f and Boussinesq system on closed, open and partially open magnetic field lines. Probability distribution functions, correlation length and time are derived. The impact of the Boussinesq approximation is shown from a statistical perspective. Furthermore the effect of electron temperature is studied.

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Width of turbulent SOL in tokamaks: from circular geometry to diverted ones

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The relation between turbulent transport and scrape off layer width is investigated in Tokamaks.
First, circular plasmas toroidally limited on the inner wall are considered. A broad set of experimental observations collected in the Tore Supra scrape off layer is detailed and compared to turbulent interchange models. Blob ExB drift velocities measured in experiments agree reasonably well with an analytical model derived for isolated blobs. Based on a time averaged particle flux balance, it is also shown that the SOL width should depend on both the blob drift velocity and a blob intermittency parameter, which is so far not predicted by isolated blob models. To overcome this, a set of 2D isothermal turbulence simulations are performed with TOKAM2D code used to derive a power law regression of the density width function of global control parameters. Quantitative agreement is found between this regression and experimental density widths measured in Tore Supra, over a large set of plasma conditions. The sensitivity to control parameters (major radius, safety factor and normalized Larmor radius) is qualitatively explained by the sensitivity of the blob velocity model. The main control parameter is the poloidal magnetic field strength. The predictions are also extended to power decay length in limited plasma configurations, following assumptions on SOL temperature. For ITER start-up phases, the predicted power decay length fall in the range of extrapolations based on multi-machine regressions.
Second, we show that this model does not fully apply to diverted configurations: although the main parameter controlling the SOL width is again the poloidal magnetic field strength, SOL widths measured in L-mode diverted configurations are found to be much narrower than in inner limited ones. We explore mechanisms possibly involved in interchange turbulence reduction in diverted geometry: Flux expansion, magnetic shear, flow drive. An attempt is proposed to incorporate these mechanisms in a simplified transport model.

Multi-impurity divertor simulation using a Monte-Carlo kinetic impurity transport model

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Impurity species, such as intrinsic impurities generated from wall/divertor plate (W, Be, C) and seeded impurity (Ne, Ar, etc), has significantly important role for the divertor plasma transport. In an integrated divertor code SONIC [1,2], the impurity dynamics is described by a Monte-Carlo (MC) kinetic impurity model IMPMC, while most of other divertor codes, such as SOLPS, EDGE2D, etc., treat it as fluid. MC kinetic treatment has advantages in modelling of the impurity transport physics, such as plasma-wall interaction, kinetic effects and so on. Originally, the IMPMC has been developed for a single impurity species. Therefore, in previous SONIC works for multi-impurity species [3, 4], one impurity species is treated by IMPMC and the other species are treated by a modified coronal radiation (MCR) model[1]. In the MCR model, the impurity density is given as a fixed fraction of the background ion density.

In order to take into account transport of each impurity species and its interaction, the SONIC code is extended to multi-impurity species. For development of the multi-impurity model, the SONIC code has been restructured on the MPI-MPMD (Multiple-Program Multiple-Data) framework developed for the integrated modeling [5]. In the past version of SONIC, the IMPMC was strongly coupled with a plasma fluid model and a MC kinetic neutral model. Therefore, large effort, such as re-definition of global variable and subroutine, complicated data transfer, etc., is necessary. On the other hand, in the MPMD version, we can extend the SONIC to the multi-impurity species with small effort because several IMPMCs for each impurity species can be executed on the framework independently but concertedly. In addition, restructuring of the SONIC on the MPMD framework contributes to future developments of the SONIC, the integrated core-SOL-divertor modeling and so on.

The extended SONIC is applied to re-calculation of previous SONIC simulation for the JT-60SA divertor with the Ar impurity seeding [4], in which the carbon impurity was treated by the MCR model. The carbon radiation is widely distributed compared with the previous simulation due to the sputtering by the background plasma at the attached region and transport process different from the background plasma. Although the carbon sputtering by the Ar impurity has not been taken into account yet, the carbon radiation power is increased to ~8 MW from ~6 MW.As a result, the Ar gas seeding rate to decrease the target heat load to < 10 MW/m2 decreases from 0.16 to 0.1 Pa m3/s. Detailed comparison of results, impact of the carbon sputtering by Ar impurity and so on will be discussed.

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Kinetic and fluid modelling of non-local parallel heat transport in magnetic fusion devices

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Modelling parallel heat transport in edge tokamak plasma is a crucial issue for predictions of power loads on divertor targets. In the operational regimes of interest for a magnetic fusion device a significant temperature gradient will build-up along the field line between the upstream hot region that acts as a heat source, and the colder plasma region at the wall that acts as a sink. When collisionality drops, classical Fourier law fails in describing heat transport, leading to overestimated heat fluxes.

In order to improve the presently used ad hoc flux limiter treatment of parallel heat flux transport in edge plasma codes we propose here to combine two approaches. On the one hand, we consider a fluid description with a generalized version of the Fourier law implementing a non-local kernel for the heat flux computation. The parallel temperature profiles are computed for strongly, marginally and low collisional regimes. The Bohm boundary condition at the wall is recovered in the three regimes introducing a volumetric loss term representing the contribution of suprathermal particles to the energy out flux. As expected, this contribution is negligible in the strongly collisional regime while it becomes more and more dominant for marginally and low collisional regimes. This

In a kinetic approach, we consider a 1D-1V (PIC) model where collisions are taken into account via a Multi-Particle- Collision (MPC) algorithm. The MPC method, originally introduced by Malevanets and Kapral [1] in the context of mesoscopic dynamics of complex fluids, is based on a stochastic and local protocol that redistributes particle velocities, while preserving the global conserved quantities such as total energy, momentum, and angular momentum. It is very efficient with respect to computation time reduction and has been recently applied to plasma physics [2]. We simulate transport along an open field line connecting the hot upstream region to the cold target wall. A hot thermal bath mimics the upstream field line region where heat enters the domain from the core region. At the other end the heat loss at the wall is taken into account by a selection rule favoring the energy loss of suprathermal particles. The impact of collisionality on transport properties and on the energy and particle flux onto the wall is then compared to the fluid approach.

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Current SOLPS-ITER physics developments and activity

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The SOLPS-ITER code suite [1,2] was launched by the ITER Organization in early 2015 to give the edge plasma modelling community a new common tool for addressing a wide variety of scrape-off layer (SOL) and divertor physics phenomena. This tool reunited many of the then co-existing versions of SOLPS and provided a new streamlined work environment, source code repository, and maintenance infrastructure. Since its launch, SOLPS-ITER has been widely adopted by the edge modelling community across ITER member states, tackling a variety of research issues. At the same time, new features continue to be added to the code suite to improve its ease of use, range of capabilities, and understanding of its results.Benchmarking with other edge codes and validation against current experimental results is key to building confidence in SOLPS results, which have defined the ITER divertor design and fueling requirements and continue to play an important role in the development of ITER operational scenarios and the provision of synthetic signals for diagnostic design. A significant component of this effort is provided by expert collaboration through the ITER Scientist Fellow Network activity on divertor and SOL modelling, whereby results from a representative set of SOLPS- ITER cases are being compared to other plasma-edge modelling codes such as SOLPS4.3[1], SOLEDGE2D [3], TECXY [4], EDGE2D-Eirene [5] and EMC3-Eirene [6]. Other SOLPS-ITER specific work includes, for example, studies of plasma response to time- dependent impurity gas injection, enlargement of the ITER plasma boundary simulation database for low power (pre-nuclear) operation, detailed energy and momentum balance analysis, refinement of the physics models for high impurity content scenarios, assessment of the effects of drifts and currents on ITER operational window and tungsten transport simulations. In addition, a substantial number of technical improvements have been brought to the code suite, the most noticeable of which is a user interface (SOLPS-GUI) which allows for greatly eased run monitoring, code results visualization, and input file building. The paper will provide a review of these various activities, with emphasis on the new physics capabilities the code offers, and present some future development perspectives.

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O-13

Characterization of oscillations observed in reduced physics SOLPS simulations

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As part of a scoping study for an ITER sized tokamak, more than 38000 simulations have been performed of the edge plasma with SOLPS5.0-B2 using aggressive charge state bundling, fluid neutrals, and with constant-in-time boundary conditions. These simulations have 40, 80, 100, 125 and 250 MW crossing into the simulation domain from the core region, and a range of D/T and impurity gas puffs giving a variation of electron density and Z_{eff}. Most of the simulations are steady-state, but about 10% show oscillations where the range of peak power flux densities to the outer target exceed 1% of the average value, and about 3% where the normalized range exceeds 10%. These oscillating cases present a challenge in determining if a particular case has converged, or needs to be continued.

In a check that these oscillations are driven by physics rather than numerics, the time- step was increased and decreased by a factor of 10 compared to the base time-step for one of the cases with a clearly defined frequency. For this factor of 100 change in the time-step, the oscillation frequency changed by less than a factor of two, strongly suggesting that the oscillations are driven by physics.

This contribution will further analyze these cases to determine the mechanism(s) driving the oscillations, as well as attempting to localize them in parameter space.

A new high-order fluid solver for tokamak edge plasma transport simulations based on a magnetic-field independent discretization

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Our global understanding of power exhaust in tokamaks, and its implications for both steady-state and transient heat loads on divertor and limiter PFCs is still poor [1]. In steady state, the heat load results from the complex interplay of transport processes in the plasma, losses at the wall and complex atomic and molecular interactions. The heat exhaust properties are largely determined by the Scrape- Off Layer (SOL) and its width is thus a key parameter to determine the power load on the vessel. It depends to a large extent on a balance between transport of heat and particles across magnetic field lines and that along them. As a consequence, it narrowly depends on the geometry of the magnetic surfaces as well as on that of the wall components. Consequently, progressing towards predictive numerical simulation associated to a better characterization of the heat exhaust mechanisms requires improving the capabilities of transport codes[2, 3],with the aim to better discretize wall-components realistic geometry and to address versatile magnetic geometries possibly varying on time that corresponds to fusion operation in tokamaks. As an attempt to achieve this goal we propose a new fluid solver based on a high-order hybrid discontinuous Galerkin (HDG) finite element method [4]. The increasing interest of these methods in many areas of nonlinear dynamics comes from their suitability to construct robust stabilized high-order numerical schemes, on arbitrary unstructured and non-conforming grids, with high scalability on various types of computer architectures. However, only few studies have been carried out so far in magnetic confinement fusion both for fluid and kinetic simulations (see some examples in [5, 6]). Capitalizing on the experience acquired in the development of the SOLEDGE2D-EIRENE transport model [2], we propose to study the edge plasma transport in the frame of a reduced model (but containing most of the challenging issues regarding accurate numerical simulations) based on electron density and parallel momentum [7]. 2D simulations performed in realistic WEST geometry are benchmarked with well-referenced simulations of the literature [2], these latest being also compared with present simulations when the computation is extended up to the tokamak center removing edge core boundary conditions. In the frame of the heuristic model of Goldston [8], we analyse the dependence of the SOL width with respect to the intensity of the drift-based convective transport. Finally, we will show how the particle fluxes at the wall vary in our model when evolving the magnetic equilibrium in time, particularly during the equilibrium construction skiping from a limiter configuration to a diverted one at the beginning of the operation.

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P1-01

Simulation of impurity behavior in EAST tokamak with the integrated COREDIV code

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The upper divertor of EAST tokamak has been recently upgraded to ITER like tungsten plasma facing components (PFCs). The target materials for upper and lower divertors are tungsten (W) and carbon (C), respectively [1]. In EAST, lithium (Li) coating is an important wall conditioning method since it can effectively reduce particle recycling and improve plasma performance [2]. The wall composition and operation conditions lead to the situation where various impurities (such as W, C, Li, etc) may simultaneously exist with the fuel ions in the plasma during the discharges. It is well known that impurity penetration into plasma can contribute to fuel dilution and radiation enhancement. Therefore, it is essential to study impurity transport and assess power radiation losses in plasma. In the present work, the integrated COREDIV code [3], which self-consistently solves 1D radial energy and particle transport equations for plasma and impurities in the core region and 2D multi-fluid transport in the scrape-off layer (SOL), has been used to simulate the EAST discharges of upper single null (USN) and lower single null (LSN) configurations operated with W and C divertors, respectively. After implementing the main parameters of EAST typical shots into the code, the modelled radial profiles of plasma density and temperature are in good agreement with the experimental measurements. The different temperature profiles in different shots can be explained by different plasma heating regimes, which are modelled in the code by prescribing different power deposition profiles. For the simulation of C divertor with Li impurities, Li impurity is considered by an assumed puffing flux from the wall, whereas C impurity is generated by erosion processes from the target. With a proper assumption of Li influx, the simulated results such as the effective ion charge (Z_{eff}) and the total radiation losses (P_{rad}) reasonably agree with the experimental data. For the W divertor case, both C and Li impurities are considered in the plasma in the simulation as well. Since the W impurity is strongly dependent on the physical sputtering by the impurity ions present in the plasma, it is a very sensitive relationship between Z_{eff} and radiation loss. The high Z_{eff} and low radiation phenomenon observed from experimental measurements will be addressed in the simulations. High re-deposition rate for tungsten is taken into account. The fractions of core radiation and SOL radiation over the total radiation loss are compared with the corresponding experimental data, which can help to get a better understanding of impurity behavior in EAST.

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Numerical Simulation Study towards Plasma Detachment in the End-Cell of GAMMA 10/PDX by a Coupled Fluid-Neutral Code

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In GAMMA 10/PDX tandem mirror device [1], the divertor simulation experiments (D-module) have been progressed in order to reveal physical mechanism of plasma detachment, impurity transport, radiation cooling, etc. [2]. A numerical simulation study by using a fluid code [3] has also been progressed in GAMMA 10/PDX in order to clarify physical mechanism of plasma detachment in the end-cell of GAMMA 10/PDX. The fluid code has been developed based on the B2 code [4]. Meshes structure of the present simulation model have been designed according to the magnetic field configuration of GAMMA 10/PDX. A tungsten target has been assumed at the end of the mesh structure. For more detailed analyses of neutral particles, a neutral code has also been developed and coupled with the present fluid code. The present code consists of the fluid code and the neutral transport code. Plasmas parameters are calculated by solving the five fluid equations (Continuity, diffusion, momentum balance, ion and electron energy balance equation). On the other hand, source terms and neutrals distribution are calculated by the neutral code. The plasma parameters at the upstream region are defined as fixed boundary (Dirichlet boundary). Neumann boundary conditions are applied on the peripheral regions. We applied divertor boundary conditions on the target plate. Bohm condition through the sheath entrance is considered in the present study. The flux limits concept are used for parallel electron thermal conductivity and viscosity. The neutral particle has been modeled by solving Boltzmann equation. These equations have been solved simultaneously with numerical calculation methods. The atomic and molecular processes of hydrogen are included in the present neutral code. In the present model, recycle neutrals (particle reflection coefficient sets to unity) are considered as hydrogen atoms. Furthermore, hydrogen atoms and molecules have been added intentionally from the end of mesh structure.

Transport of hydrogen neutral towards the upstream region has been studied at fixed plasma parameters. A series of initial test calculations has also been performed by injecting hydrogen gas puffing atoms and molecules from the end of the mesh structure. It has been shown that neutral particles concentrate near the target plate and reduce towards the upstream region. The dependence of hydrogen gas puffing on the plasma parameters has also been investigated. Hydrogen gas puffing leads to strong reduction in the temperature of electrons and ions. In addition, reduction in the temperature has also been observed according to the increasing hydrogen gas puffing. For the strong gas puffing case (with 100 test particles/time step), T_e and T_i on the target plate reduces to less than 5 eV. Especially, a significant reduction in the ion temperature has been recognized due to hydrogen gas puffing. Hydrogen gas puffing in the plasmas edge re-gion strongly enhance the momentum losses via charge-exchange reaction. In addition, electrons lose their energy due to the ionization of neutral particles. Furthermore, power loss channels for both the electrons and ions increase significantly according to the increment of hydrogen gas puffing in the end-cell.

At present, we are developing a 3D code which is more realistic to calculate vibrational hydrogen molecules. We will separately define vibrational molecules with each vibrational state. We have a plan to report contribution of Molecular Activated Recombination (MAR) on the plasma detachment.

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P1-03

Modelling of JET DT experiments in ILW configurations

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A series of high power discharges ($P_{aux}\approx$ 21-26 MW) using DT mixtures in ELM-free H mode was performed in 1997 during the JET DTE1 experimental campaign with carbon walls and divertor. A fusion power of 16.1 MW was achieved at 4.0 MA/3.6 T corresponding to the record fusion yield of Q=0.64 [1]. It is planned to perform again the DT experiments at JET with the ITER –like wall (ILW) configuration (beryllium walls and tungsten divertor) during the DTE2 experimental campaign in 2018/2019. This time however, it is not intended to repeat the transient scenario of DTE1, which in principle would be possible, but the focus is in steady state (for 5s) high performance plasmas. Experiments in DD show, that for the purpose of DTE2, high power operation compatible with divertor performance can be achieved with strike points sweeping. The high performance plasma scenarios which are foreseen for DTE2 operation are based on either a conventional ELMy H-mode at high plasma current and magnetic field or on the so-called improved H-mode or hybrid regime of operation with enhanced energy confinement. In order to assess the plasma parameters in the DTE2 experiments COREDIV code [2] has been used to perform self-consistent core-edge simulations of JET DT plasmas. The code has been already successfully benchmarked with a number of JET discharges for both carbon and ILW configurations [3,4]. The preliminary COREDIV extrapolation of the reference ELMy H-mode shot #87412 to high power (P_{aux}=41 MW) DT operation shows very good core plasma performance with fusion powers in the excess of 12 MW (only thermal contribution). Simulations show that such high performance discharge requires the heat load control by neon seeding, which in addition leads to rather beneficial effect on the plasma performance allowing for relatively wide operational window in terms of the amount of the allowed neon influx. In order, to further investigate the possible range of operating scenarios for the high power DT operation, numerical scans have been performed in terms of different input powers (P_{aux}=15-40 MW), H98 factors (=0.8-1.2), plasma currents (I_p=2.5-4.5 MA), magnetic fields (I_p=2.1-4.0 T) and plasma densities (n_e =0.6-1.1 10^{20} m^{-3}). Numerical scan at constant b shows that core and SOL radiations do not depend on the I_p and increase with seeding. When the seeding starts, degradation of the plasma confinement is observed and the resulting core plasma contamination by W ions is low $c_W << 10^{-4}$. However, the power scan at the constant I_p indicates that core radiation, power to the SOL and divertor plates (and even SOL radiation) saturates with seeding. In addition strong dilution with increasing seeding (Z_{eff} >> 3) and large W concentrations with increasing power are predicted.

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P1-04

TECXY simulations of multi-species impurity seeding in DEMO reactor

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Reduction of the heat load of divertors and plasma-facing components is a crucial problem of the future fusion DEMO reactor, where significantly higher fusion power and power to the Scrape-Off Layer (SOL) in comparison to present devices have been predicted [1]. As the threshold of the heat flux at the target has been postulated to be below 5 MW.m^{-2} and divertor plasma temperature below 5 eV to satisfy technological tungsten divertor limitations, the impurity seeding is required to reduce the energy flux flowing to the plate.Numerical studies have been performed to investigate the influence of the few impurity species on the heat load reduction and the SOL plasma parameters in European DEMO reactor [1]. Our studies were done with the use of the 2D multi - fluid TECXY code, which models edge plasma transport described by the classical set of transport equations of multispecies plasma derived by Braginskii [2,3]. Model treats the electrons and ions as a separate fluids and assume classical transport of plasma and impurities along the magnetic field lines with transport coefficients defined by twenty-one moment approximation of Grad [4] and anomalous transport across the magnetic field lines. The TECXY code solve transport equations for few impurity species and all associated ionization stages simultaneously.This paper describes numerical simulations of the edge plasma performed for DEMO reactor with various mixtures of impurities seeding (Ne, Ar, Kr). Investigations have been done in full edge geometry, i.e. the SOL region with private region. Our former findings show a significant decrease of the heat load and the temperature with different type of impurity seeding [5]. Thus, we aim to obtain the energy flux reduction to the target plate at the smallest possible value of the effective charge considering gas puff of various mixtures of impurities. It has been found that depending on the specific mixtures of impurity species application of reduction of the heat load to a to the target can be achieved at different levels of effective charge. Simulations have been performed for standard, snowflake and super X-divertor configurations and their influence on the decrease of the energy flux to the target plate have been discussed.

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Drift effects and up-down asymmetry in balanced double-null DIII-D divertor configurations

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The SOLPS-ITER 2D edge code is used to assess SOL flows and associated up-down asymmetries in a conceptual, but experiment-based, balanced double-null DIII-D AT plasma, using an up-down symmetric small angle slot (SAS) divertor [1]. In balanced double-null advanced tokamak (AT) configurations [2], up-down asymmetries in divertor target conditions may pose challenges and present opportunities. For example, if top or bottom is preferentially starved of particles, achieving detachment there may be problematic, and thus erosion and melting could be a concern. On the other hand, if the configuration can be designed to provide detachment top and bottom, but particle flow is preferential, a top- or bottom-only cryopump system may be possible, reducing overall device complexity and cost. Earlier modeling with SOLPS has shown that the SAS can facilitate detachment in single-null plasmas [3], but drift effects were not included and only single-null plasmas were considered. Previous research with SOLPS-ITER has shown the ability to capture strong Pfirsch-Schluter (P-S) SOL flows, and realistic heat flux widths [4]. These P-S flows, driven by vertical magnetic drifts, contribute to in-out asymmetries in single-null plasmas. In a similar way, an up-down SAS, with grad B drift down, for example, is expected to exhibit strong upward flow past the outer midplane, carrying the majority of particle exhaust — typically localized at the high-gradient outer midplane region — to the upper divertor. Simulations of the up-down symmetric SAS are now in progress. Analysis will focus on detachment behavior at the upper and lower outer divertor targets, and sensitivity of this behavior on pumping configuration. The results will help to inform, at a conceptual level, plans for up-down symmetric divertors for AT experiments and DEMO device designs.

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First modelling of edge plasma density regimes in the COMPASS tokamak

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In this work we present our first effort in (i) comparing SOLEDGE2D-EIRENE mean-field simulation with COMPASS experimental data and (ii) investigating density regimes and possible detachment scenarios, modeling a midplane density scan. For future fusion devices, such as ITER, operating in detached state is mandatory for reducing the plasma power flux incident on the divertor targets below engineering limits. If the basic methods to reach the detached state are known and have already been successfully tested on current devices, the exact physical mechanisms at play in the reduction of the fluxes to the targets are still not fully understood. This leads to large uncertainties in the extrapolation of the conditions necessary to reach the roll-over. Resolution of this issue requires clear understanding of the physics underlying detachment, hence validation and improvement of models and codes on comparison with experiments is needed. We here report on such effort based on experiments in the COMPASS tokamak which combines ITER relevant features (high triangularity divertor configuration, H-mode...) with a comprehensive edge plasma diagnostic coverage and the high flexibility of a relatively small machine [1]. The modelling is performed with the SOLEDGE2D-EIRENE mean-field code which treats the relevant physics [2]. Recent developments in the code include in particular the implementation of cross-field drifts and improved sheath models for the plasma-wall interaction. We first focus on a L-mode ohmic discharge, part of the isotopic campaign, which has been selected for modeling on grounds of good quality measured midplane density and temperature profiles and low helium level (monitored to be lower than 10%). Cross-field transport coefficients have been tuned automatically (using a numerical feedback procedure) to match mid-plane profiles measured experimentally via Thomson scattering and reciprocating probes. The resulting diffusion coefficient profiles are complex and illustrate the difficulty of describing the cross-field turbulent transport as diffusive. Divertor conditions obtained in the simulation are then compared to density and temperature measurements from target probes, showing already satisfactory agreement. An upstream density scan is then performed while keeping all other parameters fixed in order to identify density regimes and the expected level of transition from one regime to another up to total detachment. These levels, especially the threshold for roll-over, as well as plasma conditions in the divertor during the scans will be compared to experimental results. Finally, we discuss the impact of drifts on these results.

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Analysis of highly radiative scenarios for the EU-DEMO divertor target protection

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The development of an efficient strategy to operate the reactor under conditions compatible with an acceptable divertor lifetime (divertor protection) is critical for the successful construction and operation of DEMO. Current estimates suggest that the power decay length in DEMO should be within the range λq = 1mm and λ = 3 mm, with a power crossing the separatrix of at least P_{sep} = 150 MW, needed to safely operate the machine in H-mode. In addition to withstanding the resulting large heat flux on the targets, the divertor shape and material should be optimized to meet additional difficult constraints, such as guaranteeing an acceptable level of He compression and tolerating a continuous large fluency of high-energy neutrons. In order to allow a sufficiently long lifetime of the target surface, the peak power density should not exceed 5 – 10 MW/m^2 in steady state. This should count both the power conducted/advected by the plasma along the magnetic field lines and the contribution from volumetric radiation. SOLPS calculations place the latter at a maximum level of about 1 MW/m2, meaning that the power transported by the convection/advection channel should be kept below 4 MW/m2 (in the most conservative case). Furthermore, the need to avoid noticeable W sputtering and to operate in detached regime leads to estimate the maximum acceptable plasma temperature in front of the target at about 4 eV. It is now commonly accepted that such a combination of tight conditions cannot be met, unless at least 85% of the power crossing the separatrix is radiated along its way towards the target, and that this will require in turn the injection of a proper amount of impurities in the Scrape-Off Layer. However, the exact impurity (or mix of impurities) to be adopted is still to be determined, as well as the main plasma parameters which could allow obtaining the desired target conditions. Previous studies with the SOLPS code have focused on the analysis of Ar injection in the edge plasma, selected due to its relatively large radiative efficiency in the temperature window $T = 10 - 30$ eV, which candidates it as a very efficient SOL radiator. It was found that a possible operating window could require a large amount of Ar injection, and a relatively large outboard mid-plane density $n_{e,OMP}$, of the order of at least 4×10^{19}m^{-3}.

In this work we extend that study to a larger set of candidate impurities, now including Ar, N, Kr, and Xe, both individually and a few combinations thereof. We employ for the study the SOLPS code, mostly with the fluid neutral model and bundled charge-state approximations in order to speed-up the calculations. We consider a power level entering the pedestal of 150 – 300 MW, and an outboard density in the range $n_{e,OMP} = 2\times10^{19} - 5\times10^{19}$ m^{-3}. This leads to a multi-dimensional parameter space, in which a region of possible DEMO operating conditions should be identified. Our simulations show that it will be indeed possible to operate DEMO under divertor-acceptable conditions. However, the resulting operational space is narrow, and will require a large amount of impurity injection and an outboard density no smaller than $n_{e,OMP} = 3.5\times10^{19}$ m^{-3}. A critical constraint on the system is posed by penetration of the added impurities in the core. We show that, even keeping safely the reactor in H-mode, the amount of acceptable impurities will be limited by a significant fuel dilution effect, which lowers the final fusion power. We also present preliminary results of an ongoing more accurate analysis (full charge states model, kinetic neutrals), aiming at assessing quantitatively the uncertainty involved in the simplifications adopted.

Plasma turbulence reduction with a two-field $\kappa - \epsilon$ like model for L-mode transport simulations with SOLEDGE2D-EIRENE

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In order to achieve proper design and operation modes for the future ITER divertor, it is important to develop reliable numerical tools to simulate edge plasma properties, in particular turbulent dynamics, the latter being the main drive of perpendicular transport phenomena. First principle codes implement self-consistent equations for turbulent transport but are difficult to use with complete geometry and atomic processes and long durations as required for comparison to experiments, control and scaling to future experiments. Conversely, transport codes, which are dedicated to the latter issues, rely on reduced models with ad hoc turbulent transport properties constrained by experimental evidence. However, universal properties stemming from experiments are not sufficiently documented in fusion plasmas to determine the underlying transport parameters and narrow the range of free parameters. Therefore adding a proper model for self-consistent calculation of transport properties in transport codes appears as a crucial task in the investigation of plasma scenarios and in particular regarding the challenge of monitoring plasma- wall interaction.

We introduce a two-mean-field model for edge plasma turbulence, inspired by the RANS (Reynolds Averaged Navier-Stokes) approach, following the spirit of the $k{-}\epsilon$ description widely used in the neutral fluid community. In this approach, a dedicated equation describes the evolution of turbulence energy, akin to k, and its non-linear self-regulation mechanism. It is assumed that transport properties, modelled by diffusion, are proportional to this field. The model is completed by the second mean field akin to ϵ, which is generated by k and is introduced as a regulating field reminiscent of shear flow effects. In this picture k plays the role of a prey, whereas ϵ is its predator. Defining the appropriate regulating field and transport mechanisms depending on $k$is addressed for L-mode discharges, with proper magnetic equilibria and wall description, and where transport by blobs and large scale flows are strong contributors to transport. The model is implemented in the transport code SOLEDGE2D-EIRENE. Consistent 2D turbulent diffusivity maps are computed hence fully constraining transport properties for given plasma scenario of a given tokamak. The free parameters of the model are defined according to two procedures, matching of SOLEDGE2D-EIRENE simulations to experiments, and model reduction of the TOKAM2D turbulence model as a test bed to matching with turbulence codes such as TOKAM3X. For the latter turbulence model, one finds evidence that the regulator field encompasses the zonal flows as well as other modes driving large scale flows. Non-local kernels are also contemplated to investigate long range blob-like transport events.

P1-09

Study of the impact of magnetic geometry on power exhaust with the transport code SOLEDGE2D-EIRENE

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One of the key issues for preparing safe operation on future magnetic fusion reactors is the large uncertainty on the Scrape-Off Layer width: this sets the available volume for power dissipation and the deposition area of the exhaust power at the divertor targets. Experiments on a number of tokamaks suggest that the balance between parallel and radial transport in the main SOL results in an exponential spreading of the heat flux, whose width λq depends mainly on the poloidal magnetic field. In diverted configurations the heat flux is further spread by diffusion in the machine-specific divertor volume, described by a Gaussian of width S.

During a density ramp at fixed plasma current, it is found experimentally that the λq remains constant whereas the divertor spreading S grows linearly with the upstream density [1]. To better understand the dependence of the divertor broadening S with the density, simulations are performed with the transport code SOLEDGE2D-EIRENE. A density ramp is simulated in the JET configuration. The overall experimental behavior is well reproduced that is to say a rather constant λq and growing S while the density is ramped up.

The effect of the divertor geometry is also discussed comparing simulations and recent experiments performed on TCV [2]. A scan in the divertor leg length is performed at a fixed plasma current. The purpose here is to clarify how the variation of the parallel connection length between the X-point and the strike-point impacts the divertor spreading and if some information about the turbulent transport in the divertor region can be induced from simulation to experiment comparisons.

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P1-10

WallDYN simulations of beryllium migration in ITER

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ITER is a nuclear licensed facility and as such, must comply with limits for the in-vessel tritium and dust inventory. The ITER Research Plan (IRP)[1] foresees two non-active operation phases in which hydrogen and helium plasmas will be used aiming to demonstrate full technical performance of the device before nuclear operations with deuterium and tritium. Tritium retention in ITER will mainly be driven by co-deposition of tritium with beryllium. Predictions of tritium retention rates and material migration during nuclear operations have been performed in recent years using the DIVIMP and WALLDYN codes[2,3]. Many scenarios have been examined to assess the sensitivity of the results to the background plasma. Depending on the assumed scrape-off-layer plasma parameters, co-deposition of tritium with beryllium was found to occur either preferentially on the inner divertor baffle (as seen in JET operating with the ITER-Like Wall) or to concentrate on the first wall itself[2,3]. In addition to the issue of where the deposition will occur, uncertainty in the background plasma specification leads to a very large range in the prediction of fuel accumulation rates. The early phases of ITER operations shall be used to characterize hydrogenic retention and demonstrate techniques to be used later for control of T inventory. The ITER Research Plan therefore foresees dedicated fuel retention and material migration experiments making use of gas balance measurements and the dedicated tritium monitor (laser-induced desorption+ lock-in thermography system). To determine the feasibility of these experiments, further DIVIMP and WALLDYN simulations for low power hydrogen plasmas have been carried out. The plasmas under consideration are L-mode hydrogen plasmas at $q_{95} = 3$ (I_p/BT = 5 MA/1.8 T and 7.5 MA/2.65 T) with up to ~30 MW of input power (mixtures of ECRH and neutral beam heating). A sensitivity study has been performed to the background edge/divertor background plasma, using SOLPS-ITER simulations with separatrix densities in the range of $0.5\text{-}2\times10^{19}$ m^{-3} in combination with an exponential extrapolation in DIVIMP to extend the plasma solution right up to all plasma-facing surfaces. The RACLETTE code has also been used to calculate the surface temperature profile in the divertor from the heat fluxes calculated from SOLPS. Given the low input power, simulations show that measurable retention rate can only be achieved at relatively low plasma densities (and hence low divertor neutral pressures) when the divertor remains partially detached. In all cases, the codeposition is found mainly in the divertor. Retention ratios were found to lie between 3.5×10^{-5} and 1×10^{-2}.This work indicates that in order to get a measureable peak Be deposition of ~1 µm, plasma durations in the range 800-10000 s would be required, depending on plasma parameters, compatible with a short dedicated campaign of repeated discharges.

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Modeling of the linear plasma device NAGDIS-II with neutral gas puffing and pumping by using EMC3-EIRENE

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Neutral particle plays an important role in determining the edge plasma property especially under high neutral pressure condition associated with plasma detachment. In design concept of fusion devices, gas puffing and pumping units are crucial to determine the neutral particle behavior in divertor region as well as recycling. In order to understand gas puffing and pumping effects on the neutral particles, both studies of experiments and numerical simulations are essential. In the present study, we investigate the effects of gas puffing and pumping speeds in the detached plasma of the linear plasma device NAGDIS-II and perform the EMC3-EIRENE modeling, where the gas puffing ports and pumping unit are considered.

Linear plasma devices have been utilized to investigate edge plasma physics due to their experimental flexibilities, e.g. steady-state plasma and good accessibility to diagnostics. A simple geometrical configuration in the linear plasma devices enables a detailed comparison between the experimental results and numerical simulations. 2D fluid simulation codes have been utilized in the linear plasma devices [1–3]. The Edge Monte Carlo 3D (EMC3)-EIRENE code [4] was recently adapted to NAGDIS-II [5]. Fundamental calculations of plasma parameters were demonstrated, and neutral particle behavior of hydrogen atoms and molecules were discussed for different types of target plates. However, the neutral particle source and sink were the recycling at the target plate and ionization process, respectively. The effects of gas puffing and pumping were not included in the simulation.

For the elucidation of the energy balance among electrons, ions and neutrals in the detached plasma, neutral temperature is a key parameter. In recent study of NAGDIS-II, a laser absorption spectroscopy has been installed to measure the neutral temperature [6]. Under the high-pumping-speed condition, where the residence time of neutral particle was short, the neutral temperature increased compared to that when pumping speed was low. In order to investigate the enhancement mechanism of neutral particles with an increase in pumping speed, EMC3-EIRENE modeling including gas puffing and pumping will be performed, and the comparison with the experiments is reported in the presentation.

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Plasma simulations of complex HL-2M divertor geometries using SOLEDGE2D-EIRENE edge plasma transport code.

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In a fusion reactor, heat and particle exhausts are primarily handled by the divertor system. There, surface components need to sustain very high thermal loads from plasma interaction, which has to be mitigated by volume radiation/dissipation. The need of extrinsic impurity injection to radiate is however also impacting the burning performances due to divertor leaking toward the confined plasma. Envisaged solutions to tackle this problem aim at optimizing the divertor geometry to increase divertor dissipation, improve impurity screening between divertor and confined plasma region, and improve detachment access and stabilization. Experiments showed that changing from limiter to simple X-point diverted geometry could significantly improve tokamak performances. Current developments investigate more complex geometries with multiple X- points in the divertor: snow-flake, tripod, etc. An effective simulation code is required to properly estimate the power loads on the materials and understand diverted plasma specificities in terms of impurity screening and detachment access in these geometries.

SolEdge2D provide solutions for particle and energy transport in the edge plasma within complex and realistic 2D geometries. A penalization technique has been developed to model plasma-wall interaction in very flexible geometries. In addition, the plasma code has been coupled with the Monte- Carlo neutral code Eirene that implements the complex dynamic, atomic and molecular physics of neutrals within the plasma. This work makes SolEdge2D an efficient tool to investigate crucial problematics as main chamber recycling, impurity and divertor efficiency to control plasma wall interaction.

This work focuses on modelling of HL-2M scenarios, exploring different magnetic configurations: single null, tripod and snow flake, constructed with EFIT code. The input power and particle fluxes at the boundary of the simulation domains have been scanned to find operating points in term of thermal load and target plasma temperature for the different configurations. Thermal loads on the plasma facing components, as well as plasma parameter profiles, are compared in order to quantify the benefit of each configuration: power spreading along target, dissipation, access to detachment.

Furthermore, to establish the degree of uncertainty from artificial condition introduced into the SOLEDGE2D results, a sensitivity study of the variation of grid parameters, as well as the other control parameters, was performed. From this study, the efficiency grid generation mode has been obtained, balancing the convergence speed and computational accuracy.

P1-13

SolEdge2D-Eirene simulations of linear plasma devices Pilot-PSI and Magnum-PSI – a comparison with experimental data

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Predictions for the operation of tokamak divertors are reliant on edge plasma simulations typically utilizing a fluid plasma code in combination with a Monte Carlo code for neutral species. Magnum-PSI and its forerunner Pilot-PSI are linear devices operating with cascaded arc plasma sources that produce plasmas that are comparable to those expected in the ITER divertor ($T_e \sim 1$ eV, $n_e \sim 10^{21}$ m^{-3}). In this study, plasma discharges in Pilot-PSI and Magnum-PSI have been modelled using the Soledge2D fluid plasma code [1] coupled to the Eirene neutral Monte Carlo code. The plasma is generated as an external source of plasma density and power. These input parameters are tuned in order to match Thomson scattering (TS) measurements close to the cascaded arc source nozzle. In the simulations it is found that plasma transport from source to target is advection dominated. Supersonic flow regimes are found with M~ 2 close to the target plate. Simulation results have been compared with experimental findings using TS close to the target and in the case of Pilot-PSI, a Langmuir probe embedded in the target. Basic features of the plasma profiles observed experimentally can be recovered, with the exception of the strongly recombining regimes experimentally observed when the background pressure in the vessel is high (~ 10 Pa). It is speculated that this might be due to the simplified neutral model used in Eirene for this study, lacking elastic ion-neutral collisions and molecule assisted processes. Simulations with a neutral model containing these processes are currently under investigation.

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P1-14

Advanced divertor configurations for DEMO

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In order to demonstrate an attractive operational regime, DEMO will need to operate with high power crossing the separatrix ($P_{sep} \geq 150$ MW), low peak power flux at the targets (5-10 MW/m^2) and target temperature sufficiently low to limit W sputtering ($T < 5$ eV). The last two conditions will require most likely that plasma detachment should be achieved at all target plates. The baseline power exhaust strategy for DEMO extrapolates the ITER solution, employing a Single Null divertor (SN) and relying on extrinsic impurities injection in order to enhance SOL radiation and limit the power reaching the solid target via the basic plasma conduction/advection channels. However, it is expected that the DEMO SOL physics should be more challenging than in the case of ITER, because of the larger P_{sep}/R ratio (~ 19 MW/m for DEMO vs. ~ 17 MW/m for ITER) and the smaller power decay length (1 – 3 mm estimated for DEMO at the midplane, vs. ~ 5 mm for ITER), which makes uncertain the success of the baseline power exhaust strategy. Advanced configurations such as double null (DN), X-divertor (XD) and Super-X divertor (SX) are considered as a back- up solution for the power exhaust problem in DEMO, in case the single-null (SN) baseline strategy, already chosen for ITER, does not extrapolate favorably to the DEMO reactor conditions. These alternative solutions aim at lowering the target temperature and increasing the power deposition area by means of a longer connection length (XD, SX), proper flaring of the magnetic field lines in front of the targets (XD), placing the outer target at the outermost possible position to increase the toroidal revolution length (SX), or by increasing the number of targets available for power exhausts (DN).

A previous comparative study of advanced divertor configurations with Ar impurity seeding led with the SOLPS code showed the clear advantage of XD and SX over the SN in reaching low target temperature and target power densities; it also pointed out how, independently of the configuration considered, it was not possible to obtain acceptable target conditions unless the plasma density was sufficiently high and a considerable amount of Ar impurity was present in the divertor. This work extends the previous study in several aspects: we increase the number of considered impurities, by analyzing in details the effect of Xe, N and Kr in addition to Ar. The analysis of the SX configuration is now performed with reference to the newest equilibria, made available from 2017. First results confirm the significant advantage of advanced divertor configurations over the baseline SN, while it appears that a combination of more than one radiator is preferable over the single-impurity solution previously considered. In particular, a combination of lower-Z and higher-Z radiators (e.g. N and Xe) allows some flexibility to try setting independently the radiation level in the SOL and the confined plasma. Finally, the DN configuration has been added to the pool of geometries analyzed with the SOLPS code: we discuss the preliminary results currently available.

P1-15

UEDGE modeling of detached divertor operation for long-leg divertor geometries in ARC

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ARC (Affordable, Robust, Compact reactor) is a design concept for a high-field compact fusion reactor that makes use of high-temperature superconductors for a demountable toroidal field (TF) magnet and an immersion FLiBe blanket for tritium breeding and neutron shielding [1]. This concept allows the superconducting poloidal field coil set to be placed inside the TF while being fully shielded to neutron damage. Taking advantage of this feature, the ARC reactor concept has been recently updated [2] to include a tightly-baffled, long-leg divertor with an "X-point target" design [3] – with no reduction in available core plasma volume or reduction in tritium breeding ratio. Such tightly-baffled, long-leg divertor geometries are particularly attractive because they may provide access to passively stable, fully detached divertor conditions over a broad range of parameters, as has been recently found in modelling long-leg divertor geometries of ADX [4,5]. Moreover, they have the potential to greatly enhance the peak heat flux handling capability of the divertor, which is important for ARC – a device that is projected to have a scrape-off layer heat flux width of only 0.4 mm with total power exhaust (D-T alpha plus auxiliary heating) of 105 MW.

Using the recently updated ARC magnetic geometry, simulations of edge plasma and divertor are carried out with UEDGE, specifying varying levels of exhaust power from the core and the radial plasma profiles at the outer midplane anticipated for the device. Anomalous radial transport is modeled by radial diffusion and advection, to account for experimental levels of in-out transport asymmetry and plasma interaction with sidewalls. Neon impurity radiation is included in the "fixed-fraction" model, with the impurity fraction ~0.5%. Initial studies employing a super-X divertor configuration have shown that a stable detached divertor operational window exists for reactor exhaust power in the range of 72 to 92 MW. Under these conditions, the divertor heat flux is fully dissipated by radiation and radial losses to the sidewalls of the divertor channel. Simulations are presently being extended to study the performance of the X-point target geometry and to explore the sensitivity to input parameters, such as impurity concentration, upstream density, recycling coefficients and cross-field transport models.

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Electric fields and currents in the detached regime of a tokamak

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Structure of the currents and electric fields in the edge tokamak plasma is analyzed for the detached regimes of a tokamak. It is shown that strong electric fields in the divertor regions and near the X-point are typical for strongly detached regimes. The value of the poloidal electric field in the regions with the temperature 1-2eV is determined by the Spitzer conductivity and parallel currents. The latter are combination of the thermal and Pfirsch-Schlueter currents. The value and even direction of the poloidal electric fields in the detached regimes strongly differs from those in the high-recycling or semi-detached cases.

The ExB drifts have significant impact on the distribution of the main plasma and impurities in the divertor. Stability of the detached regimes with strong electric fields is also analyzed.

Results of simulations of several detached regimes for ASDEX-Upgrade performed with SOLP-ITER transport code are presented and the role of the ExB drifts associated with strong electric fields is discussed.

Investigation of dust shielding effects by intrinsic ergodic magnetic field line structures in the peripheral plasma in the Large Helical Device

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Sustainment of long pulse discharges in the Large Helical Device (LHD) have often been interrupted by emission of large amounts of dusts from closed divertor regions and the inner surface of the vacuum vessel [1]. In an ICRF heated long pulse discharge, it was clearly observed with a stereoscopic fast framing camera that dusts released from the surface of armor tiles (made of stainless steel) on a helical coil can in the inboard side of the torus interrupted the plasma discharge, which was synchronized with an abrupt increase of iron ion emission [2]. The analysis of the observations of the three-dimensional trajectories of the dusts at the plasma termination phase proves that the dusts penetrated into the main plasma confinement region through the intrinsic ergodic magnetic field line structure (ergodic layer) formed in the peripheral plasma. It is probable that the significant impurity radiation due to the iron ions caused by the evaporation of the penetrated iron dusts induced the plasma termination. In LHD, the geometry and the width of the ergodic layer can be flexibly controlled by changing the magnetic configuration (by shifting the radial position of the magnetic axis R_{ax}). The high temperature/density plasma formed along magnetic field lines with long connection lengths (L_c> 20km) near the Last Closed Flux Surface (LCFS) in the ergodic layer can evaporate the iron dusts. In addition to this, the plasma flow with high Mach numbers formed in the outer surface of the ergodic layer can sweep away the dusts, which can contribute to the protection of the main plasma from the penetration of dusts in the long pulse plasma discharges. Thus, a dust transport simulation code (DUSTT) [3] which is coupled with a fully three-dimensional edge plasma fluid code (EMC3-EIRENE) [4] have been applied to the detailed investigation of the effect of the dust emission on the sustainment of the plasma discharges. Impurity transport in the peripheral plasma caused by the iron dust emission from the surface on the helical coil can is investigated using the above two simulation codes. The simulation is performed in three-dimensional grid models for three different typical magnetic configurations which have a narrow, medium and wide ergodic layer (R_{ax}=3.60, 3.75 and 3.90m), respectively. A survey of the simulations of the impurity radiation power and iron ion content in the peripheral plasma is carried out in various plasma heating power and plasma density cases in the three magnetic configurations. It will find out an optimized operational window for shielding the main plasma from the dust emission for sustaining long pulse plasma discharges.

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Impact of a new general form of friction and thermal forces on SOLPS-ITER modeling results

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The SOLPS-ITER transport code suite [1] has been released in 2015 on the basis of older versions of SOLPS and the EIRENE Monte-Carlo code for neutrals. In the released code version, the original approach of B. Braams [2] is followed in which each of the ion species parallel momentum balance equations was combined with the parallel momentum balance for electrons so as to eliminate the electric field term (for convergence reasons). In doing so, the expressions for the thermal and friction forces were approximated for simplification. Such approximations break down for multicomponent plasma modeling. In [3] it was proposed to return to solving the original ion momentum balance equations, with the electric field term included, and with an accurate treatment of friction and thermal force terms, although the treatment of these terms still contains the assumption of trace density for impurities. In the present work, more general expressions for the friction and thermal forces, valid for high impurity density ($Z_{eff}-1>1$), are suggested. This is especially important when considering numerical studies of impurity-seeded divertor detachment regimes. Under these conditions, the accuracy of the impurity treatment achieved assuming trace impurities is insufficient. In their old form, coefficients of friction and thermal force depended only on main ion densities, which was valid for the Z_{eff} - 1<<1 case only. Thus, in the high impurity density limit, the thermal force acting on impurities from main ions does not go to zero, as it should do, and this leads to numerical instability in the SOLPS-ITER code runs. In their new form, these coefficients have the correct functional dependence on densities of all impurity ions (resolved by ionization states), which makes the new expressions valid for arbitrary impurity density and gives plausible results in numerical modeling. The derivation of these coefficients is based on the expressions from [4] and will be presented in this paper. The new form for the friction and thermal force terms has been implemented in SOLPS-ITER and has been tested by modeling ASDEX Upgrade Nitrogen-seeded discharges, with drifts and currents switched on. The paper will discuss the results of this test, emphasizing differences with the trace impurity approach. It will demonstrate that the trace impurity approach fails for simulations at high seeding rate, approximately at the Z_{eff}=2.

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Modeling of plasma and its wall interaction for long term tokamak operation

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Metal plasma facing materials (PFMs) is one of promising candidates for future fusion power plants, considering tritium retention. However, the interaction between plasma and metal PFMs in the long term operation has not yet been completely understood. The purpose of this study is to develop a plasma model which includes long term plasma-wall interaction especially metal PFMs and to validate the model by comparison with experimental results. In this study, a simple zero dimensional (0D) model for hydrogen (H) in the wall, and for neutral particles and ions has been develop in a system with a long term tokamak operation like QUEST[1] . In the 0D model, the wall-stored H density is obtained from the following equation,

$$\frac{dn_{Hwall}}{dt} = \frac{\Gamma_{in}}{d} - \frac{2k}{d} n_{Hwall}^{2} \ . \qquad (1)$$

Here, n_{Hwall}, Γ_{in}, k and d are the density of wall-stored H, in-coming particle flux Γ_{in}, parameter related to recombination in the wall material and the thickness of redeposition layer, respectively. The second term on the right hand side (RHS) is corresponding to molecule emission [2] toward the plasma. On the other hand, neutral particles and ions in the system are expressed by the following equation,

$$\frac{dn_j}{dt} = \sum R_{gain} n_k n_l - \sum R_{loss} n_m n_j - \left(\frac{n_j}{\tau_j}\right) + S_j \qquad (j = e^-, H^+, H^{2+}, H^{3+}, H, H_2). \qquad (2)$$

Here, n_j, τ_j and S_j are the density of species j, the confinement time and other sources (e.g. gas puffing and the particle source from the wall) or losses (e.g. pumping). Excited H atoms with n = 1,2, ⋯ ,5 are taken into account by the collisional radiative model to compare with spectroscopic measurements. The symbol Rgain and Rloss are rate coefficients. Fifteen important reactions (e.g. ionization, charge exchange (CX)) have been taken into account. The in-coming particle flux Γ_{in} in Eq. (1) is calculated from Eq. (2) and the Γ_{in} is including fast H atom produced by CX. The model above has been applied to the QUEST tokamak. Figure 1 shows time evolution of H atom wall-inventory (N_{Hwall}) with taking k as a parameter. As shown in Fig. 1, N_{Hwall} is not so much dependent on the parameter k for small t (t < 100s). On the other hand, N_{Hwall} with large k is saturated at shorter time than that with small k. By assuming that k depends on the wall temperature, these tendencies reasonably agree with those obtained QUEST experiments[3]. However, evaluation of fitting parameters such as τj is difficult and spatial distribution cannot be taken into account in the 0D model. In order to estimate τ_j in the steady state and to evaluate the effect of the special distribution, two dimensional model for the neutral and plasma transport is now undergoing. The result of the 2D calculations will be also discussed in the presentation.

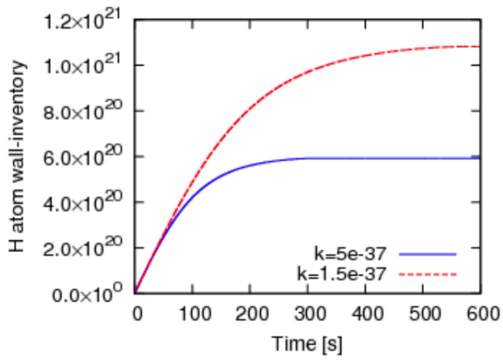

Fig. 1 Time evolution of H atom wall-inventory (N_{Hwall}) with taking k (see Eq. (1)) as a parameter.

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Ignition and Erosion of materials by arcing in fusion relevant conditions

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Arcing can be an important erosion process on plasma-facing materials for nuclear fusion [1]. In fusion devices, bombardment of nuclear reaction product He atoms can leads to significant morphology changes such as He bubbles, protrusions, and fuzzy nanostructures on tungsten (W) surfaces. When the surface morphology was altered, it was found that arcing was ignited frequently, especially in response to transients [2]. The arcing can leads to significant erosion and formation of dust, which can be significant influence on the material lifetime and core plasma performance. In this study, we provide theoretical understanding of the erosion and ignition property under the fusion relevant conditions based on experimental results obtained in a linear plasma device as changing various parameters.

Experiment was performed in the linear divertor simulator NAGDIS-II. A tungsten sample was exposed to the He plasma of which the typical electron density and temperature are $\sim 10^{19} m^{-3}$ and 5eV, respectively. The specimen was negatively biased so that nanostructures could be formed on the surface. A ruby laser, which has 694nm wavelength and 0.6ms pulse width, was used as transient heat load for igniting arcing on the surface of the samples. Mass erosion from the sample were measured with an electronic scale by measuring the mass before and after arcing.

The erosion rate was measured as 6 mg/s for 10.9 A and 4 mg/s for 3.9 A arcing. It is likely that higher current as well as longer duration contributed to the more erosion at 10.9 A than the case at 3.9 A arcing. The results are consistent with the scanning electron microscopy (SEM) observation, where we found that the trail width increased with increasing the current. On the other hand, the widths of single arc trail were the same regardless of the arc current, indicating that the eroding properties would not change in micrometer scale.

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Study of how detachment characteristics are affected by use of alternative divertors with SOLPS-ITER and SD1D

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Snowflake and Super-X topologies introduce new physics that have the potential to increase both the divertor power exhaust capability and control of detachment in tokamaks compared to conventional divertors, a problem which is crucial for the success of ITER and DEMO. A large effort is on-going on MAST-U [1] and TCV [2] to study these topologies, their properties and their control.

In this contribution, we study the detachment threshold, window and dynamics (movement) for two topologies (conventional and Super-X) with 1D models. We call ‘detachment window’ [3] the range of a given boundary condition, e.g. core density or power into the SOL, between detachment starting at the target and reaching the x-point.

SD1D (whose equations are similar to [4]) is a simple 1D fluid model (plasma and neutrals) developed on the BOUT++ platform. The ultimate goal is to have a fast, time-dependent code that can be used to understand the dynamics of detachment as well as the basic physics. The work and results consist of several parts:

1) The SD1D neutral model is further improved and compared to SOLPS-ITER 1D results (B2.5 standalone as well as B2.5-EIRENE), enabling a comparison of fluid and kinetic treatment accuracy. The goal is to make the fluid treatment, which is fast, as close an approximation to the kinetic treatment.

2) The impact of the toroidal flux expansion (movement of the strike point to large R and low |B|) and the upstream density on the detachment threshold and window is studied and compared with our analytic predictions [3] and to predict the characteristics of detachment in MAST-U and TCV [5]. Given that the physics included in SD1D is easy to change, we will also assess the impact of anomalous cross-field (poloidal) transport on the parallel plasma profiles, and compare to measurements from MAST.

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Spectroscopic models for tokamak edge and divertor plasma diagnostics

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Passive spectroscopy provides a non-intrusive diagnostic of tokamak edge plasmas: an analysis of the shape and the intensity of atomic spectral lines yields local values of the plasma parameters (N_e, T_e etc.), provided a suitable model is used. Recently, a Stark-Zeeman line shape database for the first Balmer lines has been designed for applications to current medium size tokamaks, in the framework of the MST1 European campaign [1]. After convolution with the atomic velocity distribution function (Doppler broadening), this database yields spectral profiles applicable in a fitting routine for an analysis of experimental spectra. The Stark-Zeeman model used in the database employs techniques reported in plasma spectroscopy textbooks [2]: a line shape is given by the Fourier transform of the atomic dipole autocorrelation function and the latter is evaluated using a model for the atomic evolution operator perturbed by external electric and magnetic fields. A total of 300 spectra for each Balmer line up to n = 7 have been calculated assuming plasma parameters in the ranges 10^{13} $cm^{-3} \leq N_e \leq 10^{16}$ cm^{-3}, 0.316 eV $\leq T_{e,i} \leq$ 31.6 eV, and $0T \leq B \leq 5T$[3,4]. In this work, we examine the limits of applicability of this database and present new spectra calculations. Two specific issues will be addressed: (i) the influence of strong magnetic fields, which can result in alteration of line shapes due to the thermal Lorentz electric field **v x B** [5]; (ii) the line broadening due to neutrals, which enters in competition with the Stark broadening in recombining regimes where the atomic density is high. After a presentation of the line broadening formalism, these points will be addressed successively and spectra calculations will be reported. An analysis of experimental spectra will also be performed.

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Modeling of hydrogen line and continuum emission spectra of detached divertor plasmas

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Ensuring the success of magnetic fusion requires tackling the major challenges facing the community. One of those issues is related to power exhaust, i.e., how to handle the huge heat and particle loads hitting the plasma facing components (PFCs) after their escape from the confined plasma. To avoid any damage, the power load should not exceed 20 MW/m^2 even for the most advanced materials. The most promising scenario to fulfil this requirement is the creation of a radiative mantle in the divertor region leading to the plasma detachment through volume recombination. Therefore, a big effort is being devoted to the study of plasma detachment in both L- and H-modes especially in the framework of EUROfusion MST1 [1] and JET1 workplans [2] as this scenario is foreseen for future large scale fusion devices like ITER. Impurities like nitrogen, neon or other noble gases are injected to reach plasma detachment. In support of such studies, we propose the modelling of line and continuum emission of hydrogen as well as line emission of impurities from detached plasma divertors in the aim of their characterization, i.e, providing their main plasma parameters (ne, Te and Ti). In fact, under detachment conditions hydrogen spectra consisting of isolated high-n lines of the Balmer series as well as a lowered continuum can be observed. Due to a density effect, the continuum is shifted towards higher wavelengths $\lambda_c > \lambda_B^{lim}$, where $\lambda_B^{lim} = 364\,nm$ is the theoretical Balmer series limit of hydrogen [3]. Under plasma detachment conditions (n_e~10^{20}-10^{21} m^{-3}, T_e~1 eV), high-n Balmer lines are mainly Stark broadened with a significant contribution from Zeeman effect for the lowest values of n depending on the B-field value, other contributions such as resonance (self-broadening) or Doppler broadenings being less significant. Such lines are useful to infer the electron density of divertor plasmas along each viewing chord of the spectroscopic system. The electron temperature can be inferred from the Boltzmann plot assuming a statistical equilibrium of the atomic populations over the excited levels but also from the slope of the continuum. For that purpose, the discrete to continuum radiation transition will be modeled using a dissolution factor approach which leads to a smooth merging of the lines into the continuum and resulting eventually in the lowering of the continuum below the theoretical limit λ_B^{lim}. Spectral modeling of lines emitted by injected impurities (N, Ne, ..) allow to obtain ion temperatures (T_i) from Doppler broadening. Modeling results will be confronted to experimental data.

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P1-24

A prospective spectroscopic study of hydrogen and impurity pellets in magnetic fusion devices

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There are a number of reasons pleading in favor of the adoption of the technique of pellet injection in future large-scale magnetic fusion devices like the ITER tokamak. Depending on their size, pellets made of hydrogen isotopes can be injected for plasma fueling or to control magneto-hydrodynamic MHD instabilities occurring at the edge of the plasma like edge localized modes (ELMs) or even to mitigate disruptions by attenuating the thermal and current quenchings which are the negative effects intimately linked to disruptions [1-3]. In the latter case, ice pellets of a relatively larger size are shattered just before entering the plasma [4]. Additionally, pellets made from other materials like carbon, aluminum, titanium or even tungsten are often injected in magnetic fusion devices like the Japanese LHD stellarator [5-6]. We investigate in this paper from the spectroscopic point of view all the main physics of ablation of hydrogen ice and impurity pellets in the aim to establish a link between the physical quantities characterizing of the ablation clouds of pellets and the host plasma. We will discuss the issue of how a possible collective motion of particles forming the ablation clouds can impact its radiative properties and whether or not it can be exploited for diagnostic purposes necessary for the improvement of the ablation models like the NGS model (Neutral Gas Shielding) in order to develop more appropriate models that can used for shattered pellets for instance.

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Modeling and analysis of tungsten sourcing in the outer divertor during the DIII-D metal tile campaign*

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Plasma core contamination by high-Z material like W is a critical issue for future fusion reactors. Understanding physical processes governing the generation of tungsten particles in the divertor and the transport of these particles into the SOL is therefore crucial. A dedicated metal divertor tiles campaign has recently carried out in DIII-D with two toroidally symmetric tungsten rings inserted in the lower outer divertor to investigate W impurity source, transport and impact on core plasma performance. The contrast between the localized tungsten source in the outer divertor and carbon impurities from other plasma facing components in DIII-D greatly facilitates impurity source characterization. We present the modeling of attached L-mode plasma conditions of these tungsten ring experiments with impurity transport codes (DIVIMP [1] &ERO-D3D [2]), showing that the experimental tungsten erosion profile can be reproduced and that only a small fraction (~0.1%) of tungsten can migrate away from the divertor plate into the SOL.

Plasma conditions at the outer divertor plate are reconstructed with an onion-skin model (OEDGE [1]) using Langmuir probes and Thomson scattering data. Carbon erosion and deposition are modeled with the Monte-Carlo code DIVIMP. The calculated carbon influx is in reasonable agreement with spectroscopic measurements in the outer divertor. The simulations show that tungsten sputtering is mainly induced by the impact of carbon impurities produced in the vicinity of the tungsten ring, while deuterium sputtering is negligible in these cases. In addition, the erosion and redeposition of carbon and tungsten in the outer divertor region are elf-consistently modeled using ERO-D3D with the reconstructed background plasma from OEDGE, including the effects of carbon implantation into tungsten on overall erosion. Reproducing spectroscopically measured carbon and tungsten gross erosion profiles, the modeling shows that the fraction of tungsten particles, which does not promptly redeposit and migrate away from the target, is about ~0.1%. The calculated tungsten flux leaking from the divertor into the SOL is compared with the average tungsten flux deposited on the midplane collector probe. Finally, the outboard radial tungsten migration on the divertor surface away from the W rings, as measured by DiMES (Divertor Materials Evaluation System) will be examined.

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Assessing the ion-electron thermal equilibration in the SOL of tokamaks using UEDGE

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The ion-electron thermal equilibration in the divertor is assessed for pure deuterium plasmas and for deuterium plasmas with carbon minority impurities in a simplified *slab* geometry using the multi-fluid code *UEDGE* [1]. The study shows that the electron-ion temperature ratio (T_e/T_i) varies spatially and is a strong function of the assumed upstream plasma conditions and the applied physics models. Utilizing the full model in UEDGE, the T_e/T_i ratio is unity for a few conditions only, and varies by a factor up to 10, while excluding CX and recombination results in well-equilibrated plasmas. The UEDGE predictions allow for anticipation of the parameter space where analytic models of the SOL, such as the *two-point model* [2], are directly applicable, and scaling outside said parameter space. Parameter scans in upstream density and input power are carried out for pure deuterium plasmas on an orthogonal, two-dimensional slab geometry of tokamak-typical magnetic connection length L=30m using UEDGE [3]. To study the impact of CX and recombination on ion-electron equilibration, these processes are turned off for a subset of the simulations. Furthermore, the dependency of the ion-electron equilibration on toroidal magnetic field strength, the presence of deuterium molecules, and carbon as a minority impurity species are investigated. The simulations predict electron-ion temperature ratios that vary by a factor up to 10, depending on the assumed input parameters and spatial location in the slab. High core densities result in radial target T_e/T_i-ratios below unity, which for certain input parameters increase to 1.5 close to the separatrix. Low core densities result in radial target T_e/T_i-ratios between 1 and 2, which for certain input parameters increase to 3.7 close to the separatrix. The T_e/T_i-ratio is also dependent on poloidal position and can vary by a factor up to 6 between X-point and target. The temperature difference between electrons and ions result in over- and under-estimation of heat flux and particle fluxes, respectively, when assuming the electron temperature to be the common temperature in the two-point model if $T_e>T_i$, and vice versa. The UEDGE simulations predict that excluding CX and recombination at high core densities produces the most spatially constant T_e/T_i-ratios close to unity due to higher divertor electron densities. Including molecules show similar behaviour to the cases with CX and recombination effects excluded. The reasons for the T_e/T_i-ratio variation due to different input parameters are assessed and will be discussed.

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Feasibility study of possible runaway diagnostic methods in the edge plasma region of ITER

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Runaway Electrons (REs) produced inside the tokamak plasma possess a relativistic energy and they can damage the first wall components if localized high power deposition takes place. This requires detection of REs by all possible means for post-event analysis and also for possible investment-protection of first wall or divertor components. In ITER as well as in present tokamaks, RE diagnostics have to be in place in order to aid the commissioning of the disruption mitigation system. Several RE diagnostic methods are established that can cover the broad energy or wavelength band, nevertheless, each method has a limited sensitivity regime. The edge plasma region where the plasma parameters are relatively different from the core plasma region provides an opportunity to investigate new methods for RE detection. There are only a few techniques available for the detection of REs in the edge region such as Cherenkov detector probes or thermography using a movable limiter. These are mainly invasive techniques. In this paper, a few other non-invasive techniques are re-considered and investigated taking into account edge plasma parameters using relatively simple models and comparisons are made between the diagnostic signal levels with and without the presence of REs. These techniques are RE-induced changes in the radiative cooling rate of the impurity ions [1], effect of REs on visible spectroscopy signals [2] and applicability of the of the Inverse Compton Scattering (ICS) method [3-4] for RE detection in edge plasmas since the background signal level in the edge region could be relatively lower than the plasma core. In addition, the effect of the plasma edge parameters on bremsstrahlung [5] and synchrotron [6] radiations emitted by REs is also studied and reported herein.

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Sensitivity of coupled plasma fluid/neutral kinetic edge simulations to the plasma wall interface description: effects of cyclotron orbits, sheath physics and surface roughness

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SOL modelling is key to address challenges for next step devices (ITER), e.g. power exhaust and impurity contamination of the plasma core. Transport codes are currently the main workhorses to address these issues. They often consist of a fluid solver for the plasma and a kinetic Monte Carlo code describing the neutral gas dynamics. The magnetized sheath is the interface between the fluid domain (typically up to a few Larmor radii thick) and the wall, from which recycled atoms and molecules are emitted. It is well known that this narrow layer in front of the wall can play a critical role, in particular for tungsten influxes to the plasma, because of the prompt redeposition mechanisms. In this contribution, we elaborate on previous results obtained in our group by using a 1D PIC code to connect plasma conditions at the boundary to the fluid domain to the ion velocity distribution right at the wall, which determines recycling. Our results show that the ions incidence angles are markedly shallower than what is expected from simplified models implemented in most transport codes. This has consequences both on recycling coefficients and on the subsequent neutral particle dynamics. We show that in relevant cases in JET geometry and semi detached conditions, properly accounting for the ion dynamics in the sheath can make a factor of 2 differences on the ion temperature in an extended region of the SOL. Furthermore, cyclotron orbits and the sheath electric field also result in ions impacting the wall with a velocity vector in a plane tilted from that defined by the magnetic field and the surface normal, which also influences the dynamics of the recycled neutrals. However, these results assume a perfectly flat surface, and surface roughness should affect the physics of interest here. We thus investigate the sensitivity of our results to the latter using a simplified model in which the local deviations from the mean normal to the surface is properly sampled each time an ion impacts the wall. Roughness effects are found not to modify qualitatively our main findings in the perfect surface approximation, but do reduce e.g. energy recycling (both for simplified sheath treatments and our PIC-based one). Finally, an assessment of the sensitivity of the simulations results on the description of the plasma wall interface is provided.

P1-29

Line shapes as a probe of turbulent plasmas

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In plasmas, the thermal microscopic electric field yields a perturbation of the energy levels of atomic species (or multicharged ions) present in the medium that can be observable on spectra (Stark broadening). This line broadening mechanism is used for diagnostic purposes in magnetic fusion devices, particularly in divertor plasmas in detached regime, e.g. [1]. An estimate of the importance of the Stark broadening can be performed using the empirical formula DwS = n2ea0E0/ħ (hydrogen plasma case, static approximation) for the line width in the frequency domain, where n denotes the principal quantum number of the upper level and E0 is the characteristic amplitude of the microscopic electric field generated by the charged particles (the "perturbers") located at the vicinity of the atom. Other formulas and models are also available in cases where the electric field is dynamic during the characteristic atomic time of interest, which estimated as the inverse line width in the frequency domain (see [2] for a general presentation of the theory). We examine in this work the role of non-thermal electric fields associated with turbulent fluctuations of high-frequency. Following a previous work [3], we investigate the Stark broadening of hydrogen lines in a tokamak edge plasma subject to an energetic "runaway" electron beam. In this framework, the plasma "bump-on-tail" instability mechanism results in the generation of a turbulent electric field, which is comparable to the thermal (Holtsmark) microfield if the beam density is high enough. Using a quasilinear model, we address the role of the turbulent electric field on the line broadening and present new calculations of Lyman and Balmer lines. An application to ITER relevant conditions is performed.

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Possible spectroscopic signature of wave collapse in an edge plasma

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In plasmas coupled to an external energy source like a beam of energetic electrons, one can observe the nonlinear coupling of Langmuir waves with ion sound and electromagnetic waves. This nonlinear interaction of waves changes the structural and radiative properties of plasmas [1]. Coherent wave packets are trapped in regions of high intensity, and are subjected to a wave collapse cycle. During this cycle the electric field of the wave can reach values several hundred times larger than the Holtsmark field.

We have recently studied the main features of wave packets in a fully ionized and unmagnetized plasma affected by nonlinear wave collapse, and have proposed a simple model for evaluating the changes expected on a hydrogen line shape emitted under such conditions [2]. According to analytical studies (using e.g. Zakharov equations or a nonlinear Schrödinger equation [3]) or simulations calculations [1], the electric field experienced by an emitter is a sequence of solitons, each one corresponding to a cycle of wave collapse in a localized region of the plasma. We model such an electric field with a stochastic renewal process using an exponential waiting time distribution and a half-normal probability density function for the electric field magnitude of the turbulent wave packet. We can then calculate the dipole autocorrelation function and the shape of hydrogen lines by a numerical integration of the Schrödinger equation.

Using this model, we study edge plasmas in presence of relativistic (runaway) electrons generated during a plasma disruption. Runaway electrons have a long lifetime in a large tokamak like ITER and we conjecture that an interaction of the electron beam with the edge plasma locally creates wave collapse conditions, which can be observed by spectroscopy. Indeed, as the ratio of the wave energy to the plasma thermal energy reaches a critical value depending on the plasma conditions [1], nonlinear wave collapse may be observed and modify the spectra expected in a linear regime [4]. The aim of our work is an investigation of the possible changes in the spectra of the first hydrogen lines under such conditions. The previously developed model will be adapted for the typical edge plasma conditions expected for a large tokamak.

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Temporal evolution of edge T_e and n_e profiles during detachment transition with and without RMP application in edge stochastic layer of LHD

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An application of resonant magnetic perturbation (RMP) with n/m=1/1 mode leads to stabilization of detached divertor plasma if the edge island, which is created by the RMP, is located in the stochastic layer in LHD [1]. In such case, clear flattening of electron temperature (T_e) is observed in Thomson scattering data. In an attached phase, the T_e at the flattening is several hundred eV. With increasing density, the T_e decreases gradually, and after detachment transition, it is stabilized at ~ 10 eV and kept at this temperature range throughout the detached phase. The main radiator in LHD is carbon, which originates from divertor plates, and has maxima of radiation cooling rate at ~ 10eV. Using the T_e and n_e profiles obtained by the Thomson scattering system, radiation intensity is evaluated with assumed carbon density profile, and it indicates radiation is enhanced at the island region. Without RMP application, the T_e gradually decreases while the density is ramped up with fixed NBI heating. The carbon radiation is estimated using the same schema as above. The radiation profile peaks around T_e ~ 10 eV, but its radial width is much narrower than that in the case with the RMP application. Finally, the discharge collapses as soon as the peaked radiation reached the last closed flux surface (LCFS). A simple perturbation analysis is attempted by using the *Te* and ne data of the Thomson scattering. Keeping the shape of T_e and n_e profiles, small T_e perturbation (δT_e) is applied to the experimental data, where n_e is concomitantly changed assuming constant pressure (n_e* T_e). In the case with the T_e flattening (RMP case), the perturbation ($\delta T_e < 0$) leads to inward shift of the radiation peak, but with reduced radial width. As a consequence, it results in a reduction of total radiation. Thus the *Te* profiles recovers. In the case without RMP, on the other hand, the radiation monotonically increases and penetrates radially inward with application of $\delta T_e < 0$. Thus, there is no stabilizing effect. The analysis indicates possible role of edge plasma parameter profiles on the detachment stability.

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P2-02

Non-linear MHD simulations of ELM triggering via Vertical Kicks with JOREK-STARWALL

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Magnetic triggering of edge localized modes (ELMs) in ohmic plasmas was first reported for type-III ELMs in the TCV tokamak [1]. This method, showing reliable locking of the ELM frequency to an imposed axisymmetric vertical plasma oscillation (vertical kick), was also demonstrated in the ITER-relevant type-I ELM regime in ASDEX Upgrade [2] and in JET [3].

The authors of [3] and [4] concluded that the mechanism underlying the ELM triggering was a reduction of the MHD peeling-ballooning stability region due to an increase of the separatrix toroidal current during the plasma vertical motion. The induction of edge toroidal current was mainly attributed to a fast reduction of the plasma volume due to the motion through an inhomogeneous magnetic field.

In our research, we aim to provide a detailed description of the dynamics of this phenomenon by developing and using the reduced MHD free-boundary code JOREK-STARWALL [5].

First, we show a benchmark of the recently developed mutual interaction between plasma, wall, and PF coils in JOREK-STARWALL with a complex simulation of an ITER vertical kick performed by the code DINA [4]. Then we explore the range of the key parameters determining the edge current induction, such as the plasma resistivity, the plasma compression, the magnitude and velocity of the displacement, and the viscosity. Finally we investigate the effect of the vertical kicks on the non-linear ELM stability.

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Nonlocal response of density and temperature fluctuations due to edge perturbation in tokamak plasmas

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The nonlocal transport events are observed in toroidally magnetic confinement devices which could not be explained by the conventional local transport models [1]. For example, the cold pulse experiment showed a rapid transient increase in the electron temperature T_e in the plasma core in response to an abrupt cooling of the edge [2]. It was found that a sawtooth crash triggered quickly the L/H transition [3]. Recent experiment indicated that a long-range fluctuation plays an important role for the fast pulse propagation[4]. To understand the physical mechanism of nonlocal transport, we have studied the nonlocal response of T_e fluctuation using the reduced MHD model introducing edge source and sink. The model has been extended from a 4-field model [5] to a 5-field one [6]. Simulations with the 5-field model showed that (1) the central T_e increases according to the edge cooling, (2) the magnetic island located at q=2 rational surface plays an important role as well as non-resonant modes such as (m,n)=(0,0) and (1,0), (3) the collapse of T_e occurs after switching off source and sink where the streamer plays a major role. In this simulation, the energy is directly transferred from the edge source to meso-scale modes, which are main players for non-local transport.

In the present work, to investigate the role of turbulence, we further extend the 5-field to a 6-field model which includes ITG (Ion Temperature Gradient driven drift wave) in the fluid limit. It was reported that a 3-field Landau-fluid simulation with ITG turbulence with edge sink did not clearly show the nonlocal plasma response [7]. In this case, ITG is triggered near peripheral region due to the edge cooling (sink), and the non-local transport induced by the low mode-number vortex is suppressed by the ITG turbulence. The energy transfer from source and sink to meso-scale and micro-scale modes may depend on the modeling of source and sink as well as equipartition term. In this viewpoint, we are now investigating nonlocal response of density and temperature fluctuations using 6-field model, and simulation results will be presented.

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Spontaneous transport barrier build-up in 3D global turbulence simulations of a diverted plasma

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Turbulent transport in tokamak plasma edge is widely known to impact the global performances of the device, and particularly the confinement time. Electrostatic turbulence is responsible, through the action of the Reynolds force, for the driving of zonal flows [1] which contribute, in turn, to the shearing of propagating turbulent structures. The formation of these transport barriers could be directly linked to the transition from Low to High confinement mode, whose theoretical full explanation is still lacking. Until now, only 2D turbulence codes have been able to reproduce the L-H transition [2], studying the problem in a simplified geometry. In order to investigate the role of magnetic geometry on transport barriers formation, 3D flux-driven fluid turbulence simulations have been run with the code TOKAM3X [3], both in limited and in realistic diverted magnetic configurations, maintaining the same set of parameters. While in simulations of limited plasmas turbulent structures seem to stream freely from the closed field lines region to the Scrape-Off Layer, being subsequently drained by the parallel transport, a transport barrier spontaneously builds up in the closed flux surfaces of the diverted configuration, leading to a reduction up to ~70% in radial turbulent transport. When turbulence is fully developed, the transport barrier is narrow (few Larmor radii wide), strongly localized near the separatrix and steady in time. As a consequence, a local increase of the radial pressure gradient is immediately visible from simulations. In the vicinity of the separatrix, the level of fluctuations is drastically reduced, and only filaments with bigger size in the poloidal plane manage to cross the barrier.

In order to understand the build-up process of the transport barrier, a characterization of its efficiency is carried out by scanning different parameters of the model, as the ion temperature and the particle source driving the radial pressure gradient. The resulting picture confirms the central role of the ExB shear in the deformation and consequent depletion of turbulent structures. This shear is related to two superimposing mechanisms. On one hand, a background radial electric field builds up in the closed field lines in order to conserve the poloidal momentum. On the other hand, turbulence itself provides a driving mechanism for the poloidal velocity shear. Both contributions are evaluated, highlighting the differences between the diverted and the limited configurations. The strong Shafranov shift in the magnetic equilibrium and the elevated safety factor in the vicinity of the separatrix, characterizing the diverted magnetic configuration, are found as fundamental features for the triggering of the barrier.

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Modelling of LH transition using the fluid-type transport code TASK/TX

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One of the long-standing issues in improved confinement in tokamak plasmas is to understand the mechanism of L-H transition, onset of the edge transport barrier formation. The reduction of transport to the neoclassical level in the transport barrier has been attributed to the suppression of turbulent transport by various mechanisms. One of the basic mechanisms is the radial shear of the E×B rotation velocity due to the radial electric field which sustains the pressure gradient. Since the magnitude of the pressure gradient before the transition depends on various quantities, such as the total heat flux, turbulent transport, particle source, and transport in SOL, self-consistent transport modelling including radial electric field, plasma rotation, neutral transport, and SOL is required.

The one-dimensional fluid-type transport code TASK/TX [1,2] has been developed to solve the flux-surface-averaged multi-fluid equation and Maxwell's equation in a tokamak with circular cross section. Recently it was extended [3] to axisymmetric toroidal equilibrium with non-circular cross section and became compatible with conventional neoclassical transport modules, MI (Matrix Inversion) and NCLASS. The new version of TASK/TX correctly evaluate the neoclassical transport and the bootstrap current in the edge region, and self-consistently describe the time evolution of radial electric field, plasma rotation, pressure gradient, and neutral profile. A rather simple transport model, the CDBM turbulent trans- port model [4], was introduced with the reduction due to the radial shear of E × B rotation shear.

First the dependences of radial profiles of density, pressure, rotation, and radial electric field on heating power, plasma current and neutral gas-puff rate are studied. Then the sensitivity of L-H transition on the modeling parameters of the reduction factor in the CDBM model. Finally comparison with experimental observations is discussed.

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Implementation of X-point configurations into the GBS code

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Simulating the most external plasma region of a tokamak, the scrape-off layer (SOL), is of crucial importance on the path towards a fusion reactor as heat load on the vessel wall, impurity generation, and overall plasma confinement all depend on the plasma dynamics in this region. The GBS code has been developed to solve the drift-reduced Braginskii equations which de- scribe turbulent behaviour of the plasma in the SOL [1] [2] . Until recently, only limited scenarios were considered in GBS and it was not possible to investigate diverted ones. This was mainly due to the use of field aligned coordinates, which present a singularity at the X-point, where the gradient of the magnetic poloidal flux vanishes.

In the current work we will present the extension of the GBS code to the diverted case, performed thanks to the use of toroidal coordinates. Three main topics will be discussed: the analytical and numerical work behind the implementation of X-point configurations, the results of the code verification and convergence tests, and some first understanding of the plasma turbulence in the presence of a single-null configuration.

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P2-07

Drift driven vs turbulent heat transport in 3D edge plasma simulations

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Safe operation of future fusion devices relies on a sufficient power spreading on plasma facing component in order to stay within the material constraint, so the heat channel width is a key parameter for tokamak operation. However, the mechanisms driving the heat transverse transport in the edge plasma are not yet fully understood. In L-mode, it is accepted that turbulence dominates the cross-field transport. Nevertheless in H-mode turbulence is strongly reduced in the pedestal and it is thus not clear which is the main mechanism driving the heat transport in the vicinity of the separatrix. A heuristic model [1] based on the transport by the grad B drift predicts a scaling for the heat channel width in H-mode, giving $\lambda_q \propto B_{pol}^{-1}$ .This prediction is consistent with several experimental studies that retrieve this functional dependency in their scaling laws [2]. Yet there is still no common agreement on the main mechanism underlying the heat transverse transport. In this work, we investigate the relative importance of turbulence and drift convection in the heat transverse transport in different plasma regimes. We analyse turbulent and laminar simulations in circular ISTTOK geometry run with the fluid turbulent code TOKAM3X in its recently developed anisothermal version. In laminar simulations where the anomalous trans- port level is arbitrarily set via a diffusion coefficient and turbulence artificially suppressed, we observe that the grad B drift weights in the flux surface averaged transport only at very low anomalous transport level, below the neoclassical level. Moreover the drift-dominated regime is characterised by a complex plasma equilibrium coming along with supersonic transition, and inconsistent with one of the heuristic model assumption. The SOL width presents the correlations with temperature and B_{pol} predicted by this model, however functional dependencies are not found proving that more complexity have to be taken into account in the model. These results are also valid for both COMPASS and JET divertor configurations. For the turbulent simulations, that is to say with a self-consistent level of anomalous transport, we find that without the presence of a transport barrier the heat transport is largely dominated by turbulence with $\Gamma_{E,turb}/\Gamma_{E,tot} > 90\%$. In this regime, the large-scale convective contribution of grad B drift in the heat transport is negligible, and of opposite sign for ions and electrons. Finally a scan of the ion input energy, pointed out to trigger the creation of transport barrier in experiment [3] and in 2D fluid code [4], is realized to investigate whether a drift-dominated regime can be reached for turbulent simulations with a transport barrier.

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P2-08

Analysis of key factors affecting filament dynamics in tokamak scrape-off layers using the TOKAM3X model

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Turbulent filaments have long been observed in the scrape-off layer (SOL) region of a variety of magnetically confined fusion devices and they are suspected to be key, if not dominant, contributors to cross-field transport in the edge plasma and as such play an important role in determining the SOL width. Their dynamics are not yet completely understood and the critical conditions influencing their formation and longevity are yet to be systematically characterised. Recent three-dimensional (3D) seeded blob simulations performed with the TOKAM3X fluid model [1] have shown rather good quantitative agreement on radial blob dynamics with respect to both experimental measurements and simulations performed with other codes [2,3], all based on drift-reduced Braginskii equations. With the final aim to systematically expose some of the critical factors that affect the dynamics of filaments in the edge region, the TOKAM3X model's versatility to tackle different magnetic geometries is exploited in this study. We successively analyse the impact of a localized magnetic shear, the role of the discontinuity of boundary conditions across the separatrix and the influence of the spatial variation of the resistivity. We first focus on magnetic shear. A radially varying pitch angle of the magnetic field is shown to act as a filter for filaments. Above some threshold amplitude of the imposed shear, the filament effectively stops its radial motion in the shear layer and its density gets redistributed along flux surfaces on time scales much shorter than the parallel transport time. However, in the case of large amplitude filaments, the perturbation in the radial density profiles generates a secondary instability downstream of the shear layer, thus allowing the filament to partially cross it. We will also analyse how these dynamics change when the shear is poloidally localized, as it is the case with an X-point. Experiments have further shown that SOL filamentary transport is strongly dependent on the plasma density. This is usually interpreted as an effect of the increasing resistivity, especially in the divertor region [4]. Dedicated simulations with varying levels and spatial distributions of the plasma parallel resistivity have been performed. Results demonstrate a large impact of the resistivity on the radial velocity of filaments. This holds true whether the density perturbation is connected or not to the target. Finally, the impact of the separatrix itself is investigated. Experimentally, filaments have usually been observed to first appear around the separatrix [5] and the intermittency and skewness of the cross-field transport is seen to increase radially in the SOL [5,6] suggesting a role of the separatrix in the physics at play. In our simulations, the transition from closed to open field lines is responsible for local destabilization of turbulence likely to be due to velocity shear instabilities. Using filament tracking methods in turbulent simulations, we examine how crossing the separatrix affects the dynamics of filaments.

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Effect of safety factor and magnetic shear on edge turbulent transport and poloidal asymmetries

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H-mode power threshold is found to be reduced in divertor compared to limiter tokamaks. One of the differences between these two configurations is the presence of an X-point in the divertor case characterized by a zero magnetic field pitch angle which generates locally a strong magnetic shear. This magnetic shear is able to impact on turbulence by generating a spatial tilting of fluctuations which modifies the poloidal asymmetries of edge plasma [1]. Other key quantities related to the edge plasma are linked to the poloidal magnetic field amplitude such as the power SOL width scaling law or the Greenwald limit (through plasma current) but the role of the pitch angle and magnetic shear on these laws has not been established.

In this work, we focus on the impact of magnetic field pitch angle and its shear on basic edge turbulent properties. For this purpose, the fluid turbulence code TOKAM3X is used. It solves 3D fluid drift equations in edge plasma in a global and flexible magnetic geometry [2], thus providing the freedom to investigate the impact of the magnetic configuration. In order to focus on the impact of pitch angle, we consider cases with the minimum geometrical complexity (no poloidally varying flux expansion, no X-point) and opt consequently for limited plasmas in circular geometry. Our scanning parameter is the cylindrical safety factor $q_{cyl} = \langle r B_{tor} / (R B_{pol})\rangle_\theta$ where definition can be extended even in open field lines.

We first compare results obtained with realistic q_{cyl} profile for such circular limiter geometry – which has a parabolic shape – to a case without magnetic shear, i.e. a flat q_{cyl}-profile. Several simulations without magnetic shear are realized for different constant q_{cyl} values in order to discriminate magnetic shear effect from the effect of poloidal magnetic field amplitude. Magnetic shear is found to play an important role on the pattern of density and potential structures, especially in the Scrape-Off-Layer (SOL). Indeed, simulations with nearly zero magnetic shears lead to less elongated filaments and to a reduction of poloidal asymmetries in the SOL. Moreover, the magnetic shear impacts on mean radial electric field profile and thus E × B shear: the maximum of the radial electric field is radially shifted from LCFS to first open flux surfaces when magnetic shear is increasing. The effect of safety factor resonances on radial transport is also discussed. Resonant magnetic surfaces lead to a local flattening of density and potential profiles in closed field lines region, especially on the High-Field-Side midplane. Finally, we simulate a q_{cyl}-profile with a local maximum at the LCFS mimicking X-point q-profile. Note however that we do not simulate the strongly localized effect of this X-point. Such shape induces a strong magnetic shear at the LCFS and a negative magnetic shear in the first open flux surfaces which impacts on mean E × B shear and Reynolds Stress properties with an increase of poloidal asymmetries.

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Effect of particle fueling and recycling on the properties of SOL and Edge turbulent fluctuations in global TOKAM3X-EIRENE simulations

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Power exhaust is one of the key challenges of next step devices such as ITER and DEMO. Heat flux calculations presently rely on transport codes, which solve mean field equations in which the gradient diffusion hypothesis is applied to model turbulent fluxes. However, so far the resulting anomalous transport coefficients are not consistently calculated (see Baschetti et al., this conference), and are simply taken as input parameters of the simulations. As a result, these tools are more suitable for interpretation than for predictions. Going beyond these assumptions requires resolving turbulent fluctuations, in a global 3D geometry since strong interactions between mean/mesoscale flows and micro-turbulent fluctuations have been evidenced (e.g. Tamain et al., this conference). Going for global simulations (in the geometrical sense) also requires paying more attention to the forcing of the turbulence, especially for the charged particle source. In fact, the dominant particle source in the SOL/edge is generally recycling, which occurs close to the divertor plates in diverted plasmas operated in relevant regimes. Density profiles are thus expected to be mostly flat in the core (excluding pinches and/or NBI related sources in the plasma core). Compared to a situation where the particle source is imposed at the inner boundary condition of the simulation, this affects the pressure gradient in the edge region and thus the drive for interchange turbulence there. Recycling also strongly affects parallel flows in the SOL, especially in diverted configurations.

These issues are addressed here with the newly developed TOKAM3X-EIRENE code, coupling the 3D non-isothermal version of TOKAM3X to the EIRENE Monte Carlo solver for the neutral gas. The code package relies on the same interface as the Soledge2d- EIRENE code, which retains state of the art PWI and Atomic and Molecular physics. We first present results obtained in laminar cases (where turbulence is prevented from developing), with diffusive cross-field transport, with and without mean drifts. Limiter cases with different particle fueling strategies (core source, gas puff) are discussed, and density regimes are recovered in WEST-like diverted cases. The results are found to be consistent with those obtained with Soledge2d-EIRENE.

Then, first results of the fully turbulent cases including neutrals are presented, and the properties of turbulent fluctuations and of those of the resulting turbulent transport are compared to those observed without neutrals. The mean solutions (time averaged) are discussed in the light of the results obtained in laminar cases.

A Coulomb Collision Model for Weighted Particle Simulations with Energy and Momentum Conservation

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The Particle-In-Cell (PIC) model is powerful for kinetic simulations of edge plasma in magnetic fusion devices. In PIC simulations, so-called Takizuka-Abe (TA) model [1] is widely used as a basic Coulomb collision model. This model conserves both total momentum and energy in a binary collision between particles with the same weights. However, when low-density impurities are included in PIC simulations, it is necessary to introduce different weights between impurities and background plasma in order to maintain the statistical accuracy for these species [2]. Some Coulomb collision models with a weighted particle code have been developed by expanding TA model [3,4]. However, in these models, momentum and energy are not conserved in each individual collision, but can be nearly conserved on the average. Thus, a large number of particles (>> 100) are needed in each cell to maintain the accuracy in momentum and energy conservation, so that computational cost becomes higher. In this paper, we develop a new Coulomb collision model conserving both momentum and energy in each individual collision, and carry out test simulations. The new model is constructed also by expanding TA model. When a particle a with a weight w a and another particle b1 with a weight w_b ($w_a< w_b$) collide, the velocity of each particle changes in the 1st step as below, $v_a^{t+\Delta t} = v_a^t + (m_{ab}/m_a)\Delta u$,

1st $v_{b1}^{t+\Delta t} = v_{b1}^t - (m_{ab}/m_a)\Delta u \times (w_a/w_b)$

where $m_{ab}=m_a m_b/(m_a+m_b)$ is the reduced mass and Δu is the velocity change calculated by TA model. In this 1st step, the total momentum is conserved but the total energy is not conserved. The energy disagreement $\Delta E = 1/2\, w_a m_a\{(v_a^{t+\Delta t})^2 - (v_a^t)^2\} + 1/2\, w_b m_b \left\{\left(1^{st}\, v_{b1}^{t+\Delta t}\right)^2 - (v_b^t)^2\right\}$ is given by $\Delta E = -(1 - w_a/w_b)\, w_a m_b/2\, (m_{ab}/m_b)^2 \Delta u^2$.

For the present case of $w_a < w_b$, the total energy is decreased. Then, this energy disagreement is corrected in the 2nd step by using a third particle b2 of the same species as b1 chosen at random from the same cell. Correction velocities, Δv to the particle b1 and $-\Delta v$ to the particle b2, are added as below,

2nd $v_{b1}^{t+\Delta t} = 1^{st} v_{b1}^{t+\Delta t} + \Delta v \;\; v_{b2}^{t+\Delta t} = v_{b2}^t - \Delta v,$

where the total momentum is naturally conserved, and the corrected energy can be supplied as

$\Delta E^{corr} = w_b m_b\left\{\left(1^{st} v_{b1}^{t+\Delta t} - v_{b2}^t\right), \Delta v + \Delta v^2\right\}$. To determine the Δv vector, a ”collateral condition” is needed in addition to the ”energy correction condition”, $\Delta E^{corr} + \Delta E = 0$. Several kinds of collateral conditions are examined in test simulations using two species a (with mass m_a and weight w_a) and b (with m_b and w_b). The collisional relaxation of temperature between a and b species is a major indicator of the model validation. Comparison of these collateral conditions will be shown in detail, as well as the comparison with previous models.

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An improved approximation for the analytical treatment of plasma kinetic linear instabilities in toroidal geometry

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The analytical treatment of plasma kinetic linear instabilities in toroidal geometry is commonly tackled employing a power series expansion of the resonant part of the dispersion relation. This expansion is valid under the assumption that the modulus of the mode frequency is smaller than the magnitude of the frequencies characterizing the system (the drift, bounce and transit frequencies for example). We will refer to this approximation as High Frequency Approximation (HFA). We present a systematic analysis of the meaning and limitations of the HFA. As already known, the HFA is not valid for tokamak relevant parameters. A new way to approximate the resonant part of the dispersion relation, called here Improved High Frequency Approximation (IHFA), is therefore proposed. The IHFA is shown to be applicable to the treatment of plasma kinetic linear instabilities in tokamaks. A quantitative analysis of the Ion Temperature Gradient (ITG) instability is presented.

An extended kinetic model for the thermal force on impurity particles in relatively lower collisional plasmas

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Correct understanding of impurity behavior in plasmas is one of the most important research issues for development of fusion DEMO reactor. Carefully controlled impurity contents are required to ensure the core plasma performance for steady state operation, the heat exhaust scenario, and the detachment of divertor plasmas. Coulomb collisional force, consisting of the friction force $\mathbf{F}_0$ and the thermal force $\mathbf{F}_{\nabla T}$ (also called as temperature gradient force), plays a key role to determine impurity motion along the magnetic field line $\mathbf{B}$. Kinetic models for the thermal force [1, 2, 3] have been developed so far for the test impurity particle transport simulation where background plasma is given as fluid description. Those models are, however, valid under the high collisional plasma limit. The thermal force evaluated by those models does not take into account explicitly the collision frequency between background plasma ions ν_{ii}, whereas the friction force decays as the background plasma gets less collisional (i.e. $|\mathbf{F}_0| \propto \nu_{ii}$). Such independence of $\mathbf{F}_{\nabla T}$ from ν_{ii} is a consequence of the high collisional limit. Therefore, possible overestimation of the thermal force in relatively lower collisional plasmas is of actual concern to impurity transport modeling.

Purpose of this study is to develop an extended kinetic model of the thermal force applicable from higher to relatively lower collisional plasmas for test impurity particle transport simulation. The thermal force is due to non-Maxwellian component of background plasma ion velocity distribution (i.e. distortion by the 3rd moment of velocity). In our previous model [2, 3], this 3rd velocity moment was related to the conductive heat flux in the classical Spitzer-Harm form [4]. Applying the following heat flux limiting factors [5], enables us to include the plasma ion-ion collisionality information in our thermal force model: (1) standard constant flux limiter, (2) flux limiter by the Grad's 21-moment approach and (3) flux limiter by the generalized moment approach.

We investigate the effects and impacts of each heat flux limiter on the thermal force and on the impurity behavior in the SOL/divertor plasma, by using the full-orbit kinetic impurity transport code IMPGYRO [6]. For the comparison we use Japanese DEMO plasma parameters [7] as model profile, calculated by the integrated SOL/divertor code SONIC [8].

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Verification of 5D continuum gyrokinetic code COGENT: studies of kinetic drift wave instability

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COGENT (Continuum Gyrokinetic Edge New Technology) is a kinetic plasma simulation code which is being developed by the edge simulation laboratory (ESL) collaboration. The original version of the code has been developed in 4D phase space (2D configuration space and 2D velocity space) to address kinetic plasma phenomena in complex magnetic field geometry including core, magnetic separatrix and scrape off layer region[1]. This work is focused on extending the original 4D phase space to 5D phase space (3D2V) to address full kinetic turbulences in Tokamak edge region.

Here, we report the current status of 5D COGENT employed for shear-less simple slab geometry. As a verification study, we use the problems of collisionless drift wave instability (universal instability) and collisional drift wave instability. The electrostatic gyrokinetic equations for two kinetic species (ions and electrons) are solved self-consistently and coupled to the long-wavelength limit of the Poisson equation. Simulation results show that the growth rate and the real frequency of the drift wave correspond to the theoretical solution of the drift wave instability corresponding to the resonance wave-particle interaction in the magnetized inhomogeneous plasmas. Using the Krook collision operator in COGENT, we also found that the transition of the collisionless drift wave instability to the collisional drift wave instability shows good agreement with theoretical results for weakly collisional plasmas.

Extensive 5D runs have been performed to address the effects of the drift-wave instability on blob/filamentary structures characteristic of a tokamak edge. A helical-shape potential perturbation is observed to grow exponentially in time while spinning around the filament axis with electron drift frequency. The nonlinear stage of the drift-wave instability is observed and analyzed as well.

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Electron burst driven by near electric field effects of Lower Hybrid launchers

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Lower Hybrid launchers installed in the SOL plasma generate near electric fields that can accelerate SOL electrons. This mechanism is understood as the drive for hot spot generation on field lines connected to the launchers. Experimental evidence supports the latter fact and theoretical analysis provides a qualitative understanding. Recent experimental analysis has underlined the importance of turbulent in the hot spot generation. Open questions related to this problem are the optimum between density and distance in accelerating SOL electrons on the one hand and hot spot properties on the other hand. Regarding the latter, transport along the field line will determine the splitting between the ion and electron channel, and consequently the coupling to the sheath prior to deposition on a plasma facing component.

In the relevant range of parameters, a kinetic approach is found to be required to model the electron acceleration along the field line. In its simplest form, a 1D-1V kinetic model allows one to capture the key properties of the acceleration by a single wave, and for realistic cases with several overlapping waves. Several numerical schemes are implemented in the VOICE code to address multi-species 1D-1V Vlasov-Poisson problems. The version used in the present work is Eulerian and pseudo-spectral in both velocity and position directions. A BGK collision operator is included that will take into account some aspect of collisions.

Two issues of interest have been investigated and will be presented. First the electron response to a single wave is investigated. This is shown to govern a resonant amplification of the electric field amplitude. In the linear limit, an analytical expression provides a useful guideline. Very precise comparisons between analytical and numerical values of the amplitude and phase of the plasma response have been achieved with VOICE. Transients and beating with Langmuir waves are observed depending on BGK collisions and plasma density. The latter effect results from a trade-off effect of radial properties: the fast radial decay and wave-vector filtering of the near field perturbation due to the Lower Hybrid launcher and the plasma SOL profiles. These govern the resonance conditions and its radial location. Properties of the electron response to a single wave in the linear and non-linear regimes are completed by the analysis of the conditions for electron acceleration in the multi-wave case, hence generating a burst of accelerated electrons. A more complex problem is observed in the transport phase. The fast electron burst is found to expand but is partly confined due to the ion slow motion in the parallel direction. This leads to the development of a sheath at the forefront of the expanding burst of electrons associated to ion acceleration, hence energy transfer between electrons and ions. A first consequence, for large enough simulation domains, is density depletion in front of the launcher, and cooling of the remaining trapped electrons by this evaporation-like mechanism. The modified resonant conditions then tend to starve the electron acceleration mechanism. The possibility of a background return current, akin to secondary emission in the sheath problem, and/or the occurrence of a plasma source by turbulence, are then key aspects in determining the balance between the ion and electron energy channels, as well as the intensity of the hot spot with respect to the amount of power that can be channeled in the parallel direction and the width of the electron burst pattern. While these various non-linear mechanisms at work are well characterized when addressed independently, only qualitative results are discussed regarding the complete effects obtained by their coupling.

Improved boundary condition for full-f PIC gyrokinetic simulations of circular limited tokamak plasmas in ELMFIRE

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Predicting the dynamics of boundary plasma is of particular importance for magnetic fusion devices. The scrape-off layer (SOL) controls the power exhaust and plasma-wall interaction (PWI) in magnetic fusion devices. As such, it regulates the source of impurities to the core, hence the fusion performance of the plasma, and constrains the boundary of the confined plasma where edge transport barriers (ETB) form [1,2]. For these reasons, it is desirable to simulate the confined plasma and SOL as a coupled system for the goal of predicting impurity transport, improved confinement regimes such as H-mode, and the characteristics of the SOL itself (e.g. its width, or heat fluxes to first-wall elements).

In this work, we use the global full-f particle-in-cell (PIC) gyrokinetic code ELMFIRE [3] to simulate circular limited tokamak plasmas from the magnetic axis to the SOL, including a simple PWI model. Previous works have shown partial agreement of ELMFIRE simulations with theoretical predictions and experimental measurements of electric field, Mach number, density and temperature profiles [4], but also demonstrated a breakdown of the drift and gyrokinetic orderings close to the limiter and wall. To solve this latter issue, we make use of the so-called Logical boundary condition [5]. This sets the numerical boundary to the sheath entrance instead of the physical wall, bypassing the need for a sheath model and resolutions which are not compatible with full-torus turbulence simulations.

The implementation of this new boundary condition is verified against theory and compared to experimental results for the electric field, parallel flows and SOL width from FT-2 (poloidal limiters)–to which ELMFIRE has been successfully confronted in previous studies, e.g. [6]–and to the Mistral case from Tore Supra (toroidal limiter and quasi-axisymmetric contact points) [7]. The Mistral case contains four similar Tore Supra experiments with different contact points: to the top and bottom of the machine, and to the high-field side and low-field side. This design and extensive diagnostics in the SOL make it a reference case of choice for the validation of plasma edge simulations [8,9].

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Hot spot induced by LHCD in the shadow of antenna limiters in the EAST tokamak

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Hot spots induced by lower hybrid wave have caused serious damages to the guard limiters [1]. To study its physical mechanism experimental and theoretical analyses have been carried out. Experimental results have shown that the EAST plasma disrupts more easily when the 4.6 GHz lower hybrid current drive (LHCD) power exceeds 2.5 MW after lithium coating. A density scanning experiment indicates that there exists a safe operation regime within a certain plasma density range. Infrared camera images indicate that the wall temperature of the LHCD guard limiter located within hot spots is significantly enhanced by neutral beam injection (NBI), ion cyclotron resonance frequency (ICRF) and electron cyclotron resonance heating (ECRH) power, suggesting a synergistic effect on hot spots between the LHCD and other auxiliary heating. It has also shown that the wall temperature increases with the plasma density. Based on the experimental observations, a model has been developed to explain the mechanism of sputtering of graphite tiles due to hot spots [2]. The heat flux scaling results based on the model are also consistent with the experimental scaling in the Tore Supra tokamak [3]. The heat flux and fast electron flux driven by lower hybrid wave are the dominant factor to enhance the wall temperature and sheath potential, respectively. The results suggest that the disruption may be attributed to physical and chemical sputtering of LHCD graphite guard limiters in low and high density regimes, respectively.

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Development of a Lagrange-Monte-Carlo Scheme for Fluid Modeling of SOL/Divertor Plasmas

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Understanding of plasma transport in the SOL/divertor region is one of the most important issues in order to reduce successfully heat and particle loads onto the divertor plates in future fusion reactors. For this purpose, several numerical models [1-4] were developed including three- dimensional (3D) models.In this study, a Lagrange-Monte-Carlo scheme was developed. It integrates two schemes, namely a Lagrange scheme for the convective part and a Monte-Carlo scheme for the diffusive part. The advantage of the Lagrange scheme in the convective part is the semi-implicit treatment of the pressure gradient term. As seen in Fig. 1, a 1D pure-convection coupled problem for the continuity and the momentum equation is solved correctly by the Lagrange scheme, while the Monte-Carlo scheme has not obtained the right solution. This is mainly due to the strong coupling by the pressure gradient term which poses severe numerical problems in the standard Monte-Carlo approach for purely convective regimes.

The work aims as a first step at the solution of 1D results for the three coupled fluid equations of continuity, momentum, and energy eq. Generalization to higher dimensions is then straightforward.

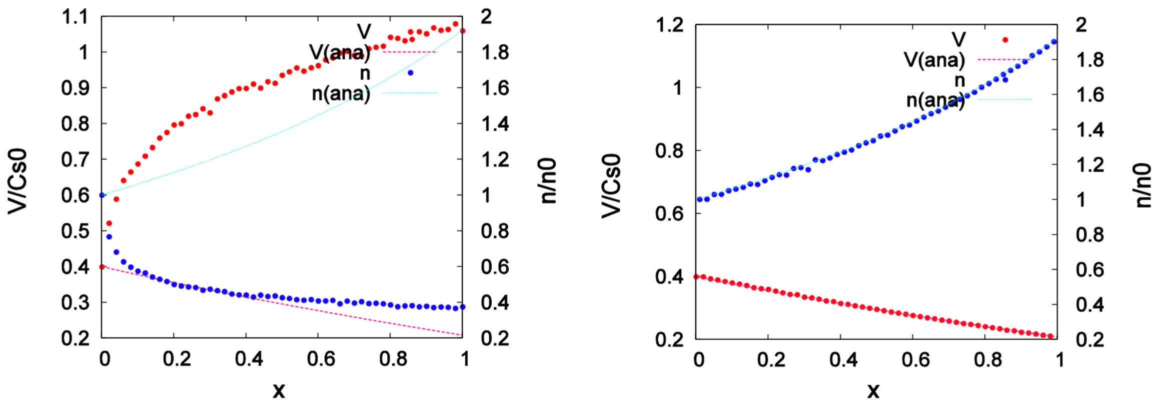

Fig. 1 Results of the 1D pure-convection problem
(Left: Monte Carlo scheme, Right: Lagrange scheme).

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Identification of stochastic noise propagation in plasma edge simulations

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At present, the plasma and neutral transport in the plasma edge region of magnetic confinement fusion devices is numerically simulated using plasma edge codes such as B2-EIRENE [1]. The plasma is modeled with PDEs (Braginskii equations) that express conservation of mass, momentum and energy, while the neutral particles are described by kinetic equations (Boltzmann equations), that model the neutral distributions in a position-velocity phase space. These two sets of equations are strongly coupled because of mutual interactions between plasma and neutral particles. On the one hand, these interactions give rise to source terms in the fluid equations. On the other hand, the physical parameters in the kinetic equations strongly depend on plasma quantities. A Finite Volume (FV) method for the plasma equations and a Monte Carlo (MC) simulation of the neutral particles are iteratively coupled and solved alternatingly until convergence is attained. The numerical error contributions originating from non-convergence, FV discretization and the finite amount of MC particles are examined numerically in [1]. Also, practical procedures to estimate these errors are proposed in [1]. However, the underlying mechanism of how the stochastic noise of the MC source term evaluations in every iteration, results in a deterministic bias and a statistical error (with zero mean) on the plasma solution, is not yet understood completely. In this paper, we study the propagation of the stochastic MC noise throughout the iterative scheme and the resulting noise on the plasma solution in a general mathematical setting. We denote the iteration scheme as follows:

$$X^{n+1} = G\big(X^n, S_n^{MC}(X^n)\big),$$

where X_n is the plasma state at iteration n, $S_n^{MC}(X^n)$ is the MC estimate of the source terms at iteration n based on the plasma state X and G represents the iterative scheme. The track-length estimator used in the MC method produces an unbiased source term estimate. Hence

$$S_n^{MC}(X) = S(X) + \eta_n,$$

where $S(X)$ is the exact source term corresponding to the plasma state X and η_n is a random variable with mean zero representing the MC noise. The plasma state X_n can be decomposed as: $X^n = X^* + E_n$, where X^* is the exact solution satisfying $X^* = G(X^*, S(X^*))$ and where E_n is the error on the plasma state at iteration n. We derived formulas for the deterministic bias $\mathrm{E}[E_n]$ and for the variance $\mathrm{VAR}[E_n]$, quantifying the statistical error as a function of the variance $\mathrm{VAR}[\eta_n]$ of the MC estimator. The goal is to get insight in the way that the simulation parameters (like the relaxation factor, time step, number of FV cells, ...) effect the magnitude of the error E_n on the plasma solution. We will present the main results of this analysis and illustrate the results via simulation experiments that confirm the correctness of the derived formulas for the bias and the variance of E_n.

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Accuracy and convergence of iteratively solved Monte Carlo codes for simulations in the plasma edge of nuclear fusion reactors

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Iteratively solved Monte Carlo (MC) codes, such as EMC3-EIRENE [1], are frequently used for plasma edge simulations. However, accuracy and convergence assessment are still an unresolved issue. Typically, convergence is monitored via the relative change of the simulation result over subsequent iterations. Often, this relative change is considered to be a measure for the statistical error of the MC scheme [2]. This straightforward approach does, however, not necessarily yield a reliable stopping criterion.

In an intermediate step towards error calculations in coupled MC/MC simulations, we examine the accuracy and convergence behavior of an iteratively solved MC code for two simplified one-dimensional cases. The first case simulates one-dimensional neutral transport in the vicinity of a pump with a non-linear Boltzmann equation. The second case assesses a plasma transport model governed by fluid equations with fixed source terms.

In this work, we describe and estimate the different components of the numerical errors in an iteratively solved MC code, based on the classification of numerical errors in coupled finite volume / MC codes in [3]. The numerical error consists of several contributions, originating from statistical noise, discretization, correlations and incomplete convergence. Based on this classification, combined with appropriate error estimates [3] and taking into account a preset affordable computational time, we are able to determine suitable numerical parameters for the above mentioned simulations.

We discuss the accuracy and convergence of three simulation approaches. With Correlated Sampling, strict convergence can be reached if the particle trajectories remain correlated between the iterations. Because this correlation is sometimes hard to achieve, we discuss techniques to increase the fraction of simulations that converges. With Random Noise, the statistical noise prevents strict convergence of the code (in terms of the relative change tending to zero with an increasing number of iterations). However, by averaging over iterations, accurate results can be obtained efficiently. With Robbins Monro, averaging is used during the simulation. Although the relative change between the iterations seems to indicate a large increase in accuracy, the actual error decreases slowly.

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Hybrid neutral models for a detached ITER case

P2-21

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Most current plasma edge codes use a Monte Carlo (MC) simulation to solve the kinetic equation for the neutral particles. The EIRENE code [1] is an example of such an MC code. However, the statistical noise hampers the convergence assessment when solving the coupled plasma-neutral equations. Moreover, the calculation time strongly increases for high-collisional detached cases. The high number of charge-exchange collisions in the (partially) detached regime justifies the use of deterministically solved fluid neutral models in large parts of the simulation domain [2]. Here, we present a hybrid method that combines such fluid models with the kinetic description, based on hybrid models used in rarefied gas dynamics [3]. In Ref. [2], the results from three different fluid neutral models with increasing degrees of complexity are compared to an MC simulation of the kinetic equation for a detached ITER case. The first neutral model, which consists of a single pressure-diffusion equation, gives only accurate results for the particle source. When adding a parallel momentum equation, also the parallel momentum and ion energy sources are predicted within 30% of accuracy. This error is further reduced to 15% by solving a separate neutral energy equation. This leads to the conclusion that discrepancies remain between the fluid and kinetic solutions. To eliminate these discrepancies, we add a kinetic correction on the fluid solutions by exclusively solving a kinetic equation for the closure terms from the moment equations of the fluid models (e.g., stress tensor, heat flux vector,...). The application of this hybrid method for the underlying fluid neutral models from Ref. [2] leads to three hybrid neutral models, which we assess on the basis of their reduction of calculation time for a certain statistical error for a detached ITER case. The advantage of this type of hybrid neutral model is the enormous reduction of computational time for a single particle. E.g., the average time for simulating a single particle is reduced with a factor 100 compared to an MC simulation of the full kinetic equation for this detached ITER case. This is due to the absence of time-consuming charge-exchange collisions in the kinetic part of the hybrid model. However, this computational time reduction per particle does not necessarily lead to an equivalent speed-up of the neutral transport simulation due to the different statistical nature of the hybrid approach in comparison to the traditional – full kinetic – MC approach. On the one hand, there is an increased statistical error of the kinetic correction since it consists of canceling positive and negative parts. On the other hand, the kinetic correction has only a minor influence on the total neutral solution, because a large part follows from the fluid solution. Thus, both effects counteract each other and the effects are different for the three fluid models. In this work, we compare the performance of the hybrid models with the full MC simulation and we propose measures to further reduce the computational time of plasma edge simulations for detached divertor conditions.

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Enforcing conservation at Monte Carlo level in a coupled Finite Volume – Monte Carlo simulation

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A finite volume (FV) plasma simulation and a Monte Carlo (MC) neutral simulation form the backbone of SOLPS-ITER simulations [1]. These two simulations are iteratively coupled until convergence. The coupling leads to the introduction of statistical noise from the MC simulation into the FV simulation. Under the influence of this noise, the FV code can not converge in a classical sense, i.e., by requiring the residuals to reach machine precision upon convergence, except when using a very large number of particles in the MC simulation. A possible alternative measure of convergence is provided by the Kukushkin metrics, which monitor the stagnation of the energy level and can be regarded as a global convergence criterion [1]. Of course, such a metric can only be used in a setting where the energy equations are not included in the FV model.

The coupling between the MC and FV parts is bi-directional: the simulated MC particle paths depend on the plasma state, and the plasma state is determined by neutral source terms (such as mass and momentum) that are computed from the MC particles. Selecting MC source term estimation procedures with reduced variance often entails the loss of strong conservation of mass, momentum and energy: in an individual simulation, the source terms in the FV part do not correspond exactly to the amount of mass, momentum and energy that disappeared from the MC part. Conservation is only maintained in a weak sense, i.e., in expectation over an infinite number of MC particles. The lack of strong convergence leads to statistical noise on the balances of these three quantities and precludes the use of Kukushkin-like metrics without a very large number of particles [2]. On top of this, the physically-relevant conservation properties of FV simulation are also removed.

We propose a rescaling of the to-be-conserved properties at the end of the MC simulation as a means of enforcing conservation while continuing to benefit from the reduced variance of non-conserving estimation procedures. An analytical approach based on a Taylor expansion and Edgeworth series expansion demonstrates the bias resulting from the proposed approach is of order $O(N-1)$, with N the number of MC particles, and hence asymptotically negligible compared to the statistical error, which is of order $O(N-1/2)$. This negligible bias will be illustrated for a 1D simulation with the same features as SOLPS.

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Towards Numerical Optimization of Novel Magnetic Topologies

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In the search for new methods to tackle the heat and particle exhaust challenge in tokamaks, the potential of novel magnetic topologies is currently being investigated. Some of these topologies exploit multiple and/or higher order magnetic nulls or x-points, in contrast to the well-known single-null divertor. One of the promising topologies is the snowflake topology, named after the typical hexagonal structure of the separatrix near the second order x-point. In the past decade, a considerable amount of research on the plasma edge physics of these topologies has already been performed. However, no work on automated numerical optimization of these topologies has yet been reported, although target shape [1] and magnetic divertor[2] optimization have already been developed for the single-null divertor and leads to promising results. The optimization involves the minimization of a cost functional that indicates the performance of the design. In our case, this is the integrated heat load over the targets. In this paper, we make the first step towards numerical optimization of snowflake configurations by verifying whether cost functional and gradient can be calculated in an accurate way using an in-parts adjoint approach [3]. One of the obstacles along the road to optimization of novel topologies, is the inflexibility of current grid generators to automatically construct high quality aligned grids throughout the entire plasma edge region for these novel topologies. We developed a grid generator that can automatically identify the magnetic topology and construct an adequate aligned grid throughout the plasma edge. A general approach not limited to snowflake configurations is elaborated. In order to assess the discretization error on the cost functional, a GCI discretization error analysis is performed [4]. We find that the error on the cost functional is below 0.5%. However, it is found that large errors on plasma state variables occur near the x-points due to the large flux expansion. In some cases they can locally amount to over 100%. To improve this, a local grid refinement method is added to the grid generator. As a first step towards automated optimization of novel topologies, we perform a sensitivity analysis of the two most prominent snowflake configurations: the snowflake minus and snowflake plus. Starting from an initial magnetic equilibrium, magnetic field changes are modelled using the vacuum approach. An in-parts adjoint approach is then used to calculate the sensitivity of cost functional to the coil currents that shape the magnetic field. A discretization error analysis is also performed on the gradient.

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Implementation of a consistent fluid neutral model in SOLPS-ITER and benchmark with EIRENE for detached divertor conditions

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At present, (partially) detached divertor operation is considered as a necessity to obtain acceptable divertor heat loads in future fusion reactors, while maintaining an adequate Helium exhaust. The simulation of the edge transport under such conditions, however, poses severe challenges for current plasma edge codes. Especially the accurate simulation of neutral transport with a kinetic Monte Carlo (MC) code rapidly becomes inaccessible for such high-collisional cases because of the tremendous computational cost. It is exactly in these conditions that simple fluid models perform best and might provide a feasible alternative.

Although prior attempts to benchmark fluid neutral models with kinetic codes in plasma transport simulations still gave large discrepancies [1], these could often be attributed to the use of inconsistent fluid neutral models or lacking physics in the fluid models compared to the much more mature MC codes like EIRENE. Recent efforts of Horsten et al. [2] therefore focused on elaborating fluid neutral models with transport coefficients derived consistently from the underlying collisional data from atomic physics databases, and improved boundary conditions that include the effects of microscopic (partial) reflection of neutrals at solid surfaces. In a benchmark of this new generation of fluid models with an in- house kinetic code on a fixed background plasma, these models have proven to offer an accurate alternative for MC codes in high-collisional plasma edge regions.

In this contribution, we report the first benchmark of these novel fluid neutral models with EIRENE on a fixed background plasma. To this end, we implement the boundary conditions for partial reflection of neutrals in the SOLPS-ITER code. This new SOLPS-ITER fluid neutral model incorporates the microscopic rate and reflection coefficients from respectively the AMJUEL-HYDHEL and TRIM reflection databases. As such, a fluid neutral model is envisaged that is fully consistent with EIRENE in the limit of highly- collisional plasmas. The model includes parallel momentum conservation for neutrals and assumes ions and neutrals are in thermal equilibrium. A benchmark is then conducted on a slab case in the absence of molecules.

Finally, we examine the convergence of the neutral model in coupled plasma edge transport simulations with a full-fledged plasma model from SOLPS-ITER. The performance of this fully-fluid edge transport model for detached divertor conditions is then extensively compared in terms of accuracy and convergence speed to that of the hybrid fluid-kinetic B2.5-EIRENE code (SOLPS-ITER).

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Comparison between fluid, kinetic and hybrid descriptions for neutrals in the SOLEDGE2D edge plasma code

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Power exhaust is one of the major challenges of future devices such as ITER and DEMO. Because of the lack of identified scaling parameters, predictions for divertor plasma conditions in these devices have to rely on edge transport codes, which often consist of a fluid code for the plasma (like Soledge2D [1]) coupled to a kinetic Monte Carlo code (such as Eirene [2]) for the neutral particles. The latter incorporates the complex atomic, molecular and surface processes characteristic of edge plasmas. The use of a kinetic description for the neutral gas stems from the fact that in most of the device the ratio of the neutrals' mean free path to a representative physical length scale (the Knudsen number, K_n, which measures how "kinetic" the neutrals behave) is much larger than one. However, in the divertor region close to the neutralizer target plates, especially for large machines as ITER or DEMO, the situation can be very different owing to high density of the order of 10^{20}–10^{21}m^{-3} and low temperatures, below 5eV. In this regime, the kinetic description is too detailed (locally $K_n << 1$) and the codes tend to be very inefficient due to the Monte Carlo approach.

A hybrid model then becomes appealing for the neutral gas, treating the latter as a collisional fluid in low-temperature and high-density areas and kinetically elsewhere. In order to achieve this, a new fluid neutrals code has been developed: it solves a multifluid (one fluid for the atoms and one for the molecules) 2-D Navier-Stokes system of equations for the neutrals, with a set of boundary conditions consistent with the kinetic description [3]. From the numerical point of view, the code solves the system of equations using a Hybridizable Discontinuous Galerkin (HDG) method [4,5], which has various advantages, among the others: reduction of the number of degrees of freedom of the numerical problem; the possibility of using arbitrary meshes (in particular, the same mesh used by Eirene); stable high order discretization of convective and diffusive operators.

In this contribution, we thus describe the new fluid model and present some simulations performed in WEST geometry with the neutral code fully coupled to Soledge2D. The impact of the neutral code on the computational time is investigated comparing results with the fully-fluid description, the fully-kinetic one (i.e. Eirene) and a preliminary hybrid fluid-kinetic version implemented in Soledge2D, in high recycling or detached divertor regimes.

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Analysis of the Plasma Blob Formation and Transport, and Its effect on Impurity Transport in the SOL Regions

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Understanding the impurity transport is one of the most important issues for realization of future fusion reactors. As for the SOL background plasma transport, the recent studies have shown that a non-diffusive transport like plasma blob is important [1]. The formation mechanism of the blob itself and also the effect of the blob on the impurity transport have not yet been well understood. Final goal of this study is to clarify the effect of the blob formation/transport on the impurity transport. As a first step, we focus on the plasma blob formation by applying the 2D interchange turbulence model to the edge and the SOL regions. This model consists of the density and vorticity equations with **E**×**B** drift and diamagnetic drift [2]. The computational domain is a simple 2D slab space. Figure 1 shows the preliminary result of the blob structure for the plasma density at $\tilde{t} = 500$ with starting from the initial potential fluctuation at $\tilde{t} = 0$ ($\tilde{t} = \omega_{\mathrm{ci}} t$: ω_{ci} is the ion cyclotron frequency). At first, the linear growth of the Rayleigh-Taylor instability has been observed, then the plasma blobs have been formed from the end of the linear growth phase to the nonlinear saturation phase. The linear growth rate in the simulation reasonably agrees with the theoretical result from the linear dispersion relation. With these results of the density and potential structure, the effect of blobs on the impurity transport has been studied by using a simple test particle model of impurities with their equations of motion. The equations of impurity motion have been solved and test impurity trajectories are followed by taking account of the background plasma density and potential fluctuations. Figure 2 shows a snapshot of the 2D distribution of the impurity test particle ($\tilde{t} = 500$). The results show that the plasma blob significantly affects the impurity distribution. The details of the results will be discussed in the presentation.

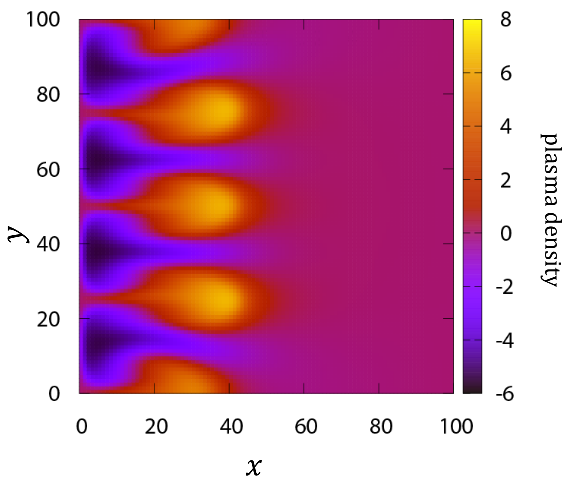

Fig. 1 Blob structure at $\tilde{t} = 500$. (The x and y coordinates are corresponding to the radial and poloidal directions normalized by ion Larmor radius respectively.)

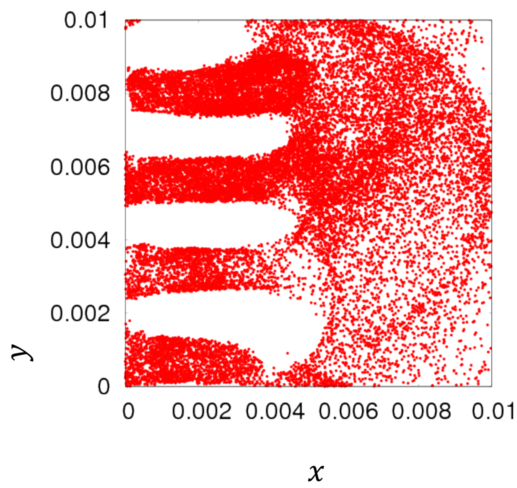

Fig. 2 Snapshot of the 2D distribution of the impurity test particles at $\tilde{t} = 500$with the background density and potential profiles obtained by the 2D interchange turbulence model (Fig. 1). Background temperatures are assumed to be $T_{\mathrm{e}} = T_{\mathrm{i}} = 50$ eV.

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