Modelling of the linear divertor plasma simulator NAGDIS-II by using EMC3-Eirene code

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Toward next generation fusion devices, such as ITER and DEMO, linear divertor plasma simulators (LDS) are widely used to obtain fundamental understanding of edge plasma physics as well as plasma-wall interactions because of their good properties: ability to produce steady state plasma at relatively low cost, good diagnostic access near the plasma-material interface and systematic control of plasma irradiation conditions. In order to make the scientific output obtained in each LDS universal, the numerical simulation to evaluate plasma characteristics in the LDS is quite essential.

The 2D fluid simulation codes, such as B2-Eirene code, have been utilized to simulate plasma characteristics in the linear divertor plasma simulators, PSI-2 and the MAGNUM-PSI [1, 2]. On the other hand, the Edge Monte Carlo 3D (EMC3)-Eirene [3] is able to simulate 3D effects in edge plasmas in fusion devices. Stochastic layer and divertor leg in helical divertor configuration of the LHD were analyzed using EMC3-Eirene code [4, 5]. The 3D effects are also important in the LDS, for example, in the experiment with a V-shaped divetor module being conducted in the GAMMA10/PDX [6].

In this presentation, the EMC3-Eirene code is adapted to analyse plasma profiles in the linear divertor plasma simulator NAGDIS-II. The schematic of the simulation model is shown in Fig. 1. The two regions are set up in the model: the ZONE1 is external particle and energy source region and the ZONE0 is the divertor plasma test region. The two regions are divided with the baffle plate. The ambient gas is deuterium or hydrogen. Optimization of the distribution of energy volume source is able to match calculated profiles of electron temperature T_e, and density n_e with experimental ones in the attached plasmas of the NAGDIS-II. Further, the detached plasma observed in the NAGDIS-II is going to be reproduced in this simulation.

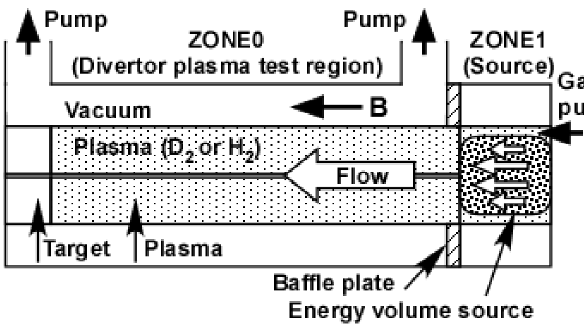

Fig.1 Numerical model of NAGDIS-II device

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Bohm criterion and virtual divertor model for SOL-divertor simulation

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The Bohm criterion is well known as a boundary condition for the plasma flow speed V_f at the sheath entrance in front of a plasma facing material. A simple criterion, $V_f \geq C_s$ (C_s is the sound speed), is given from a condition that the solution to Poisson equation describing the sheath potential is not an oscillating profile but a monotonic profile [1]. Applying this analytical method, the Bohm criterion for two-ion-species plasma was investigated by comparing with results of kinetic simulation using PARASOL code [2]. The analysis gives two curves like a shifted hyperbola in a ($V_f^{(1)}$, $V_f^{(2)}$) diagram to satisfy the monotonic profile formation, where $V_f^{(1)}$ and $V_f^{(2)}$ are the flow speeds at the sheath entrance of species-1 ion and species-2 ion, respectively. On an upper branch, both speeds are higher than those sound speeds, $V_f^{(1)} > C_s^{1)}$ and $V_f^{(2)} > C_s^{(2)}$, while on a lower branch $V_f^{(1)} < C_s^{1)}$ and $V_f^{(2)} < C_s^{(2)}$. On the other hand the PARASOL simulation showed that two flow speeds locate near the upper branch in case of rare collision. To explain the simulation results, we introduce a linear stability analysis to the sheath theory. We find that the upper branch is stable but the lower branch. This is because the upper branch is realized but the lower branch does never appear in the PARASOL simulation.

Recently we have been developing a new code for the SOL-divertor plasmas with a generalized fluid model taking account of the anisotropic ion temperature [3]. When we use two components of ion temperature, parallel to the magnetic field line and perpendicular to it, the parallel momentum transport equation is described by a spatially first-order differential equation without the second-order viscosity term. This equation does not especially require a boundary condition at the divertor plate. In order to simulate the existence of the divertor plate, we introduce a virtual divertor (VD) model [3]. The VD region is set beyond the divertor plate position. Plasma fluid equations are solved continuously from the SOL-divertor region to this VD region, where the artificial sinks for particle, momentum, and energy are added. By coupling the anisotropic-temperature formulation and the VD model, simulation results demonstrated that the flow speed at the divertor plate exceeds the sound speed automatically [3]. In this paper, we clarify the reason why the Bohm criterion can be established by the present modeling without treating sheath physics. The stability consideration described above is useful for this analysis. We also study the numerical restriction and error due to the finite grid size.

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Electrostatic characteristic of a spherical dust on PFW in sheath field

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Electrostatic characteristics of a spherical metal dust on plasma-facing wall (PFW) in plasma sheath field are studied analytically. The bipolar coordinate is applied to analyze this system, where one of the coordinates agrees with the boundaries, the spherical dust surface and the PFW plane surface. Plasma sheath field, where the decay length is Debye length λ_D, is applied to this system. In the bipolar coordinate the distance from the PFW is scaled by the dust radius. In this study the uniform potential and electric field, which are included in the sheath field, is considered as the 0-th order external field. The finite size of the dust is treated as perturbations to the 0-th order system.

The local electrostatic field of the 0-th system φ_{ue} is obtained as [1]:

$$\varphi_{ue}(\alpha,\beta)=\varphi_s\left[1-\frac{R_d\beta}{\lambda_D(\alpha^2+\beta^2)}+\frac{R_d}{\lambda_D}\sqrt{\alpha^2+\beta^2}\int_0^\infty d\lambda\,\lambda\, e^{-\beta_0\lambda}\frac{\sinh(\beta\lambda)}{\sinh(\beta_0\lambda)}J_0(\alpha\lambda)\right],$$

where (α,β) is the bipolar coordinate, β_0 is one half, which is the value of β on the dust surface. The quantities φ_s (<0) and R_d are the dust radius and the sheath potential drop, respectively. The first term is the uniform electrostatic potential. The second term indicates the uniform electric field. The third term is the induced potential, which comes from the induced charges of the dust and PFW surface. The potential produced from the non-uniform sheath field φ_{ne} is calculated as:

$$\varphi_{ne}(\alpha,\beta)=\varphi_s\left[\sum_{n=2}^{\infty}\frac{1}{n!}\{-\frac{R_d\beta}{\lambda_D(\alpha^2+\beta^2)}\}^n\right.$$

$$\left.-\sqrt{\beta_0}\sqrt{\alpha^2+\beta^2}\sum_{n=2}^{\infty}\frac{1}{n!}(-\frac{R_d}{\lambda_D})^n\frac{1}{2^{n-1/2}\Gamma(n+\frac{1}{2})}\int_0^\infty d\lambda\,\lambda^{n+1/2}K_{n-1/2}(\beta_0\lambda)\}\frac{\sinh(\beta\lambda)}{\sinh(\beta_0\lambda)}J_0(\alpha\lambda)\right],$$

where Γ and K indicate the Gamma function and the second kind modified Bessel function, respectively. In the case of smaller dust radius than the Debye length $R_d / \lambda_D < 0.2$, the e-folding length of the electrostatic potential at $r = 0$ is the same as the Debye length. The effect of the finite size of the dust becomes remarkable at $R_d / \lambda_D > 0.2$.

By using the electrostatic potential on the dust and the PFW the surface charge density and the electrostatic force are able to be calculated. These characteristics of the dust on the PFW is applicable to investigate the dust dynamics of the dust wall interaction by using the MD simulation.

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Study of Hysteresis Properties in Tokamak Plasma Based on Bifurcation Concept

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This work aims to investigate the hysteresis properties at the *L-H* transition and *H-L* back transition based on bifurcation concept. The formation of an edge transport barrier (ETB) is modelled via thermal and particle transport equations. Both transport equations include neoclassical and anomalous effects with flow shear as suppression mechanism on the anomalous channel. The anomalous transport is modelled with critical gradient model. The flow shear, calculated from the force balance equation, couples the two transport equations. Analytical investigation reveals that the fluxes versus gradients space exhibits bifurcation behaviour with *s*-curve soft bifurcation type. Pressure and density fields show bifurcation nature separately. ETB can be formed when either the particle or thermal fluxes reach their critical threshold. Apparently, the backward *H-L* transition occurs at lower values than that of the forward *L-H* transition, illustrating hysteresis behaviour. The hysteresis properties, i.e. locations of threshold fluxes, gradients and their ratio are analyzed as a function of neoclassical and anomalous transport values, sources (particle and heat), and critical gradients.

Dynamical transport modelling of radial profiles in tokamak edge plasmas

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One of the fundamental issues in tokamak transport is the behaviour of radial profiles in the edge region. The profiles of density, poloidal and toroidal rotations, and temperature of charged particles (electrons, bulk and impurity ions), neutral density, and radial electric field are strongly coupled with each other. Therefore even the profiles in the ohmic and L-mode phases have not been well reproduced in transport simulations, needless to say about the formation of transport barriers in the H-mode. Most of core transport simulations use prescribed boundary conditions at the separatrix or at a magnetic surface slightly inside. The conditions have not been well validated though they strongly affect the performance of core confinement. A few core transport codes are coupled with two-dimensional peripheral transport codes, and self-consistent simulations have been carried out. The purpose of the present study is to reproduce the radial profiles by one-dimensional dynamical transport modeling and clarify the mechanism of determining the edge profiles. The multi-fluid transport code TASK/TX [1,2] is used for the analysis including the core and SOL regions. The dependences of the radial profiles on radial transport models in core and SOL region, parallel loss models in the SOL region, and boundary conditions on the wall are examined. The profiles are compared with experimental observations in ohmic, L-mode and H-mode phases. The rotation profiles of impurities and bulk ions are also compared. Preliminary analyses have shown that low and flat particle diffusion coefficients are required to reproduce density profiles, while thermal diffusion coefficients with parabolic profiles increasing in the edge region are required to reproduce the temperature profiles. The formation of edge transport barrier is examined using the CDBM transport model [3]. More systematic parameter survey is underway.

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Evolution of electron temperature in tokamak boundary plasma during a massive gas injection

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Abstract

It was found in tokamak experiments with disruption mitigation by noble gas injection in boundary plasmas that the energy in electron component remains lower than in ions. One reason for that could be a weak energy equipartition between electrons and ions compare with the electron radiation cooling within a magnetic tube and strong electron thermoconductivity along the magnetic field lines. To estimate the cooling time of electons we assume that the gas injection is localised somewhere downstream in poloidal and toroidal position. The time of temperature drop upstream the magnetic tube due to radiative energy losses in downstream can be calculated by solwing the 1D non-stationary thermoconductivity equation with nonlinear thermoconductivity coefficient. The exact analytical solution of the temperature evolution along the magnetic field lines is found assuming the given downstream temperature and the initial energy content within the magnetic tube. The result indicates that under some conditions the time of the upstream temperature drop can be comperable with the equipartion time.

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Nonlocal response of electron temperature fluctuation from edge to core in tokamak plasmas

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The nonlocal transport events are observed in toroidally magnetic confinement devices which can not be explained by the conventional local transport models [1]. For example, the cold pulse experiment shows that a rapid transient increase in the electron temperature in the plasma core in response to an abrupt cooling of the edge. The time scale of the transient transport is much faster than the diffusive time scale. Recent experiment indicates that a long-range fluctuation plays an important role for the fast pulse propagation [2]. Although theoretical studies on the nonlocal transport were carried out within a one dimensional system, the detailed physical mechanism was not identified [3]. In order to understand the physical mechanism of nonlocal transport, multi-dimension and multi-scale transport simulations are necessary. We have studied this problem using a 4-field reduced MHD model [4]. It was found a transient edge source generates the nonlocal transport in a tokamak plasma from edge to core through the large-scale vortex convection.

In this work, we extend simulation studies of nonlocal transport from edge to core after the transient edge source/sink in tokamak plasmas. By varying numerical models, we search for what physics is essential for the nonlocal transport. For the previous case of a 4-field reduced MHD model with Resistive-Ballooning-mode turbulence, we found that (1) a nonlocal response of pressure profile appears at the location far from the edge cylindrical source after it is switched off [4] and (2) a similar nonlocal transport appears for a spherical source but just after applying it in contrast to a cylindrical one [5]. Dynamics of the large-scale $\cos\theta$ component of $\tilde{P}_{\pm 1,0}$ play an important role in the nonlocal transport in these simulations. On the other hand, 3-field Landau-fluid simulation with ITG (Ion-Temperature-Gradient driven drift wave) turbulence with edge sink does not show the nonlocal plasma response clearly [6]. In this case, $\cos\theta$ component is stirred by the ITG turbulence in the core after the sink is terminated. As a consequence, its radial wavelength becomes shortened considerably in the core region. The existence of turbulence in the core can be an obstacle for the nonlocal transport via $\tilde{P}_{\pm 1,0}$. We are also developing a 5-field reduced MHD model that includes electron temperature equation. This model handles with the density source and electron temperature sink simultaneously so that more realistic pellet modeling in the edge is possible. We are investigating nonlocal response of electron temperature fluctuation using this model and simulation results will be presented.

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Numerical Simulation Study of Plasma Flow in the GAMMA 10/PDX End-cell Using a Fluid Code

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The GAMMA 10/PDX ("GAMMA 10" and "Potential-control and Divertor-simulation eXperiments") tandem mirror in the Plasma Research Center, University of Tsukuba is the world's largest linear device which has high ion temperature than electron temperature (T_i = 100 ~ 400 eV, and T_e = 30 ~ 40 eV) [1]. By using the end-loss plasma flow from GAMMA 10/PDX, study of boundary plasma has been performed in the divertor simulation experimental module at the end region. It has been confirmed that the heat load on the target plate is almost dominated by the ion energy.

In order to investigate the energy loss processes and the plasma behaviors under the similar conditions of the divertor simulation experiments in the GAMMA 10/PDX, a numerical simulation code in the end-cell was modified based on the same fluid model as the B2-code, which has been originally developed by Braams for the numerical simulation of tokamak SOL and divertor plasmas [2].

In the previous study using a single-fluid code [3], it is reported that the importance of effective reduction of hydrogen ion energy by enhancing the charge-exchange reaction, since the dominant state of electron impact-ionization is kept by the energy transport from hydrogen ion to electron. In the multi-fluid code, Ar gas is applied as a impurity neutrals and the distributions of injected neutrals are determined as constant in end region. Under the condition of injecting neutral hydrogen and impurity particles, the charge-exchange reaction and the radiation loss for electron are strongly enhanced. Therefore, both hydrogen ion and electron temperatures decrease to nearly 3 eV and the heat load on the target plate is reduced by nearly 80% compared with the condition of low injection rate of neutral hydrogen and impurity. In addition, the plasma pressure in front of the target plate begins to decrease in the high density condition of neutral gas injection, which indicates the symptom of plasma detachment.

The divertor plasma depends on the atomic molecular interactions between charged and neutral particles. At present, the only atomic interactions are introduced in the multi-fluid code we developed. It seems that the importance of molecular particles interaction grows under the condition of high plasma density and low plasma temperature like detached plasma. In this paper, the detailed simulation results taking into account of the effects of molecular interactions and the evaluation of the gas injection effects on the plasma loss energy will be discussed by using multi-fluid code.

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Kinetic modelling of divertor fluxes during ELMs in ITER and effect of in/out divertor plasma asymmetries

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Particle and energy fluxes to the plasma facing components (PFCs) during uncontrolled edge localized modes (ELMs) are expected to unacceptably shorten the PFC lifetime for high Q scenarios in ITER on the basis of empirical extrapolations from existing experiments [1]. Non-linear MHD modelling of these particle and energy fluxes carried out for ITER has shown that some aspects of such empirical extrapolations, such as the scaling of the broadening of the ELM power footprint at the divertor with ELM energy loss, may not apply at the ITER scale [2]. However, the robustness of these findings is questionable because the particle and energy transport along the field lines in these MHD simulations are modelled in a fluid approximation. This is not applicable during an ELM in ITER because this transport is essentially collisionless given the high plasma temperatures in the pedestal plasma. In order to understand the consequences of kinetic effects on ELM energy and particle transport, modelling of typical edge plasma conditions during (and between) ELMs in ITER has been carried out with the PARASOL (PARticle Advanced simulation for SOL and divertor plasmas) particle-in-cell code in 1-D and 2-D approximations [3]. Initial simulations with PARASOL 1-D [4] had shown that both the in/out asymmetry of divertor parameters during ELMs as well as the ELM energy loss itself has an influence on the in/out asymmetry of the divertor fluxes, although the total energy deposited by the ELM tends to by biased towards the divertor with lower recycling between ELMs (outer divertor), which is contrary to experimental observations. This was identified to be due to the fact that large thermoelectric currents circulate between the two divertors during the ELM in the simulations. While the inner divertor ion flux during the ELM is largest (due to the higher recycling), the sheath transmission coefficient at the outer divertor is typically a factor of 2-8 times higher than at the inner divertor during the ELM, due to the thermoelectric currents, which leads to the ELM power flux at the outer divertor to be largest [4].

To understand the role of thermoelectric currents on ELM power deposition asymmetries further PARASOL 1-D simulations have been carried out where the two divertor targets are assumed to be floating so that no thermoelectric current can flow. In these cases, both the ELM power and particle fluxes to the inner divertor (higher recycling between ELMs) are highest indicating that the dynamics of the current flow in the SOL and the divertor structures can play an important role in the ELM deposition asymmetry. The degree of energy deposition asymmetry found in the simulations is typically much smaller than in experiment ($E_{in}/E_{out} \sim 1.2$ in the model with respect to $E_{in}/E_{out} \sim 2$ in experiment) although during some phases of the ELM the power deposition asymmetry can be very large reaching $q_{in}/q_{out} \sim 3.5$. The paper will describe the factors that are found to influence this in/out ELM asymmetry (between ELM in-out divertor asymmetries, ELM energy loss size, numbers of particles expelled by the ELM for a given ELM energy loss, etc.) and discuss implications for ITER. First simulations of ELMs with the PARASOL 2-D code will be presented.

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Mode structure analysis of detached plasma using a 2D image

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The intermittent cross-field convective plasma transport, so called plasma blob, has been observed in SOL in several magnetically confined plasma devices. However, the mechanism of enhancement of blobby plasma transport in detached plasmas has not been understood yet.

In this work, we have observed dynamical changes of plasma column accompanied with the generation of plasma blobs using a high-speed camera and performed the mode analysis of plasma instability.

The analysis was performed using 2D images taken by the high-speed camera in the linear divertor plasma simulator NAGDIS-II. Figure 1 (a) shows the contour plots of the dynamic behavior of the detached plasma. The quasi-axisymmetric plasma column deformed by plasma instability with non-axisymmetric strong emission. Then, a plasma blob is ejected in the radial direction from near the high-intensity region, and emission in the plasma column becomes original level.

A mode analysis of plasma instability has been performed using the following equations:

$$n_m(r,t) = \frac{1}{2}\int_0^{2\pi} n(r,\theta,t)\, e^{-\mathrm{i}m\theta}\,\mathrm{d}\theta\,, \qquad (1)$$

$$E_m(t) = \int_0^{\mathrm{R}} \left| n_m(r,t) \right| r\,\mathrm{d}r\,. \qquad (2)$$

The time trace of the mode structure of plasma blob generated has been investigated experimentally by this analysis (Figure 2 (b)). A magnitude relationship between mode components match with previous studies [1, 2]. However, this result is not periodically as predator-pray[2]. And it is an intermittent waveform.

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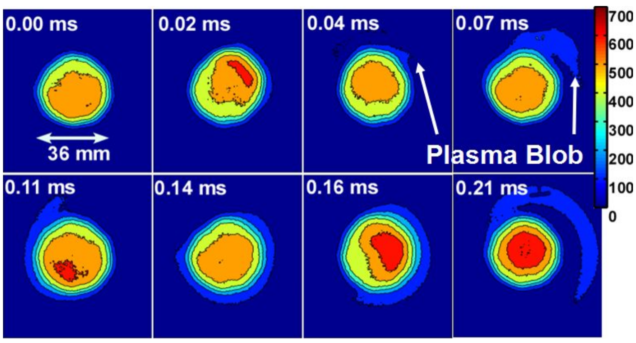

Fig.1 A contour plot of snapshots of the 2D image

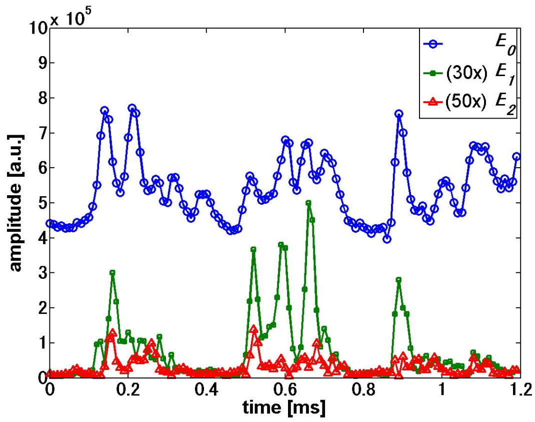

Fig 2. Time traces of the amplitudes E_0, E_1, and E_2.

SOLPS-ITER modeling of the Alcator C-Mod divertor plasma

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SOLPS-ITER is a new edge code package developed and hosted at the IO [1], as a backward-compatible successor to the SOLPS4.3 code that has been used as the workhorse for the design of the ITER divertor [2]. It is based on the most recent, MPI-parallelized version of the kinetic neutral particle transport code Eirene, and the B2.5 plasma code version developed in St. Petersburg, which presently includes the most advanced numerical treatment of drifts and currents [3].

In order to improve the predictive capabilities of the code and to identify and study remaining modeling issues, it is essential that the edge plasma model in SOLPS-ITER be continuously validated against other edge codes and experimental data from existing devices. As a part of this effort, we report here on the first results of an attempt to apply the new code package to modeling of the Alcator C-Mod divertor. With its magnetic field, divertor density and target shaping very close to ITER design values, the Alcator C-Mod divertor is one of the most ITER-relevant experiments in terms of divertor plasma and neutral particle parameters, and the inclusion of drift terms is expected to have a strong impact on the plasma solution due to the compact size.

As an initial benchmark for the new code, the deuterium, Ohmic, single null discharge #990429019 at 950 ms has been chosen. This particular plasma pulse, with I_p = 0.8 MA, B_T = 5.4 T, a line average density of 1.46×10^{20} m^{-3}, partially detached inner target and attached outer target, has a particularly comprehensive diagnostic data set for the divertor, and has previously been modeled extensively with the OEDGE code suite [4]. We discuss the setup of this case within the SOLPS-ITER environment, and an initial comparison to previous modeling efforts and experimental data. Runs with and without drifts and currents are compared, and the impact on in-out divertor asymmetries is assessed. Differences between model and experiment are analyzed and directions for further model improvement are suggested.

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Photon Absorption Effects in DEMO Divertor Plasma

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Handling of huge power exhausted to the SOL/divertor is crucial issue for design of DEMO fusion reactor. Study of power handling scenario and design window for a compact DEMO reactor has been progressed by using a suite of integrated divertor codes, SONIC [1, 2]. However, in the simulation, photon absorption effects were not taken into account. In the DEMO divertor operation, the complete detachment is required to decrease the target heat load and to suppress the target erosion. Under such divertor plasma condition, the neutral density is high and therefore the photon absorption can become considerably important.

In order to evaluate photon absorption effects on the DEMO divertor plasma, an iterative self-consistent collisional-radiative model [3] is applied to the DEMO divertor condition. The background plasma profile for a compact DEMO reactor (major radius: 5.5m, fusion power: 3GW) is calculated by the SONIC code. In this case, the atomic density near the inner target is an order of 10^{20} m^{-3}, and the decay length for the Lyman α line is less than 1 mm. The spatial profile of net ion particle source/sink without the photon absorption effects near the inner target is shown in Fig. 1(a). The ion particle sink due to the volume recombination can be seen in front of the target and in the private region. On the other hand, as shown in Fig. 1(b), the photon absorption significantly increases the ion particle source and therefore the inner divertor region becomes ionization dominant. The ion particle flux to the target and the target heat load due to the surface recombination of the ion flux can be increased by increase in the ionization source. At the same time, higher electron density and colder electron temperature due to the enhanced ionization can enhance the volume recombination and contribute to reduction of the heat load. In order to analyze this balance, the photon absorption effect is implemented to the SONIC code and self-consistent analysis among plasma and neutral transport and photon absorption is carried out.

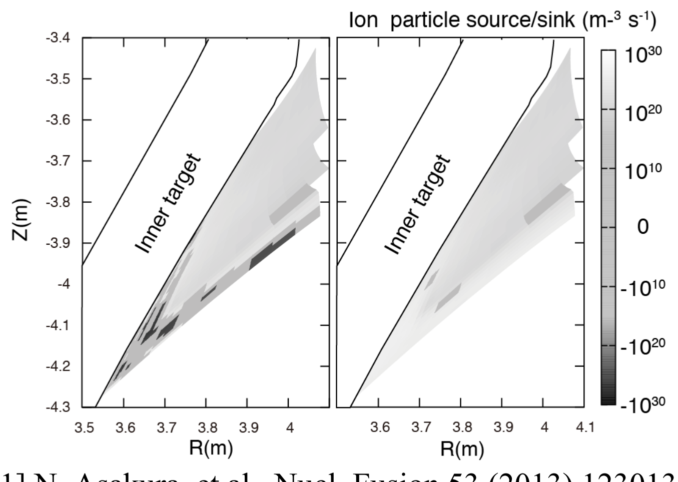

Fig.1 Spatial profile of ion particle source/sink near the inner target (a) without and (b) with the photon absorption effects. The net ion particle source/sink near the target and the private region is changed from the particle sink to the particle source due to the photon absorption.

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Hybrid formulation of radiation transport in optically thick divertor plasmas

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The preparation of large-scale devices (ITER, DEMO) requires accurate transport models for the plasma edge able to account for neutrals and charged particles in a self-consistent fashion. For certain, in particular ITER relevant, conditions the run time in edge codes is currently reported to be of several months, which rules out any practical analysis of results if progressing from ITER to the even far more demanding DEMO divertor scenarios. [1]. Modeling efforts are presently ongoing in order to speed up the codes or to develop more modern computational tools (see invited talk by H. Bufferand [2]). A specific point that needs to be addressed concerns radiation transport in atomic lines. A few years ago, it was shown using B2-EIRENE that the trapping of Lyman photons may result in an alteration of the ionization-recombination balance [3]. An accurate description of the machine operation requires this process be properly accounted for. In its current version, the radiation transport model used in B2-EIRENE involves a kinetic transport equation for photons of the Boltzmann type (referred to as ñequation of transferò in the astrophysics literature) and this equation is solved using a Monte Carlo method, in the same way as the Boltzmann equation for neutrals is solved. In the most optically thick conditions (e.g. with an atomic density approaching locally values of 10^{15} cm^{-3}), it can be shown that the divertor of ITER behaves in certain areas like a black body near the resonance line of hydrogen (Lyman α). This suggests the use of a fluid model can be as accurate as a kinetic approach, while being less time consuming. In the same spirit as [4], we report on the development of a hybrid formulation of radiation transport, designed in such a way to switch automatically between a kinetic and a fluid model according to the plasma conditions. Applications to a slab are performed as an illustration.

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COREDIV and SOLPS numerical simulations of the nitrogen seeded JET ILW L–mode discharges

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Next step fusion devices (ITER, DEMO) will operate with a tungsten (W) divertor and the reduction of power loads on the divertor target is mandatory in relevant heated scenarios to mitigate the tungsten source. Tungsten ions are sputtered in the edge region (SOL, divertor) and the main part of their radiation is in the core, which has important consequences on the device performance. From one side the core radiation should be limited in order to work in the H-mode which requires that power delivered to the SOL is greater than the L-H power threshold. From the other side, the strong radiation is beneficial for the reduction of the power to the target plates to technologically acceptable level (for ITER < $10MW/m^2$). Due to strong coupling between core and edge regions, reactor grade simulations require integrated modelling, where both regions are treated self-consistently.
Presently, different numerical codes are used for simulations of plasma performance in tokamaks (eg. SOLPS, COREDIV) They both solve the Braginskij fluid equations but are based on different numerical implementations, use different divertor geometries and neutrals models. The SOLPS code package is used to model only edge plasmas (until pedestal top) in the real vessel and magnetic geometry whereas COREDIV solves core and edge part self-consistently but works only in the slab geometry. A simple fluid/analytical model for neutrals is applied in COREDIV but in SOLPS, neutrals are simulated more precisely by Monte-Carlo package (EIRENE) which usually is more time consuming. Both codes have been benchmarked with existing devices (JET, AUG) [1-3] being able to explain some experimental findings.
Here the question appears, how much different model assumptions in the codes (full geometry or slab geometry, kinetic or fluid/analytical neutrals models) have influence on the simulation results. The main focus of this paper is to give answer to the above question by investigation how big and where the differences are between the two codes and what the numerical/physics reasons are for specific differences.
For this aim, JET ILW (ITER Like Wall) L-mode discharges have been chosen, with and without nitrogen seeding (#82291-9) and simulated by both codes. These shots are characterised by very low auxiliary heating (P_{NBI}=1.1 MW) and low electron density ($2.35x10^{19}m^{-3}$). Comparisons are made to the experimental measurements (e.g. radiation levels, Z_{eff} and W concentrations) and the differences between the two code results (e.g. temperature and density profiles at the outer divertor plate) will be shown and discussed.

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Reduced physics models in SOLPS for reactor scoping studies

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Heat exhaust is a challenge for ITER and becomes even more of an issue for devices beyond ITER. The main reason for this is that the power produced in the core scales as R^3 while relying on standard exhaust physics results in the heat exhaust scaling as R^1 (R is the major radius). ITER has used SOLPS (B2-EIRENE) to design the ITER divertor, as well as to provide a database that supports the calculations of the ITER operational parameter space. The typical run time for such SOLPS runs is of the order 3 months (for D+C+He using EIRENE to treat the neutrals kinetically with an extensive choice of atomic and molecular physics). Future devices will be expected to radiate much of the power before it crosses the separatrix, and this requires treating extrinsic impurities such as Ne, Ar, Kr and Xe — the large number of charge states puts additional pressure on SOLPS, further slowing down the code.

For design work of future machines, fast models have been implemented in system codes but these are usually unavoidably restricted in the included physics. As a bridge between system studies and detailed SOLPS runs, SOLPS offers a number of possibilities to speed up the code considerably at the cost of reducing the fidelity of the physics. By employing a fluid neutral model, aggressive bundling of the charge state of impurities, and reducing the size of the grids used, the run time for one second of physics time (which is often enough for the divertor to come to a steady state) can be reduced to less than one day. This work looks at the impact of these trade-offs in the physics by comparing key parameters for different simulation assumptions.

Presentation of the new SOLPS-ITER code package for tokamak plasma edge modelling

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Simulations of ITER divertor performance, geometry optimization and estimates of key related parameters such as fuelling throughput have historically relied on the SOLPS plasma edge modelling tool, more specifically its versions 4.0, 4.2 and 4.3, developed mostly in collaboration with FZ-Jülich [1]. However, the majority of the SOLPS code base development has been performed by research groups outside the ITER Organization (IO), most notably in IPP (-Garching and –Greifswald), Univ. Paris 13 and St. Petersburg, leading ultimately to versions 5.0 [2], 5.1 [3] and 5.2 [4], respectively.

While the work at the IO and FZ-Jülich focused mainly on neutral transport physics, the developments performed elsewhere have concerned on extending and refining the physics capabilities of the plasma model. For example, the St. Petersburg group has improved the numerical solution of the drift terms and the electric potential equation. These enhancements have reached a level of maturity at which it makes sense to merge them all. A decision to do this was taken at the IO in 2012 and work was commissioned under contract to couple the St. Petersburg version of the B2.5 plasma fluid solver with the newest version of the Eirene neutral transport code from FZ-Jülich [5] to yield a new code version, named SOLPS-ITER. The results of this activity were reported in [6]. Since then, further effort at IO has been devoted to including the refinements present in the 5.0 and 5.1 versions which were not all captured in the initial coupling exercise. The run environment for SOLPS-ITER has also been extended to the rest of the SOLPS code suite (mesh generation and I/O processing) and the code is now distributed by means of a dedicated repository maintained by the IO, including state-of-the-art version control and issue management.

This contribution will present the new code package and its implementation at IO, together with a discussion of further ongoing improvements (extended grids, new user interface design, additional physics switches). Following the initial, rather rigorous benchmarking part of the main coupling exercise, further comparisons have been made with existing SOLPS4.3 ITER baseline simulations and other experiments (see e.g. [7]), and will be presented here.

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Modelling of the JET DT experiments in Carbon and 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 campaignwith 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]. These high performance fusion shots were characterized by good confinement with very high central ion temperature up to 28 keV (hot ion mode), relatively low plasma density (~3-4 x 10^{19} m^{-3}) and by moderate values of Z_{eff} (~2-3).

It is planned to repeat the DT experiments at JET in 2017/2018 in the ITER –like wall (ILW) configuration, with beryllium walls and tungsten divertor. Direct extrapolation of the results from the old DTE1 experimental campaign to the new situation seems however difficult, since degradation of the plasma performance in the ILW configuration is usually observed and the compatibility of high power operation with divertor performance is questionable.

In order to assess the plasma parameters in the planned DT 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].

First, some selected discharges from the 1997 experiments are simulated to fix the code free parameters. In particular, the record shot #42746 is analyzed with the COREDIV code and good agreement with experimental data is found with the assumption of the ion transport reduction to obtain the hot ion mode features. The code is able to reproduce the density and the temperature profiles as well as the global plasma parameters (e.g. P_α, Z_{eff}, P_{fus}).

The preliminary COREDIV simulations for the ILW configuration (discharge parameters as for the shot #42746) show very good core plasma performance with even higher fusion power (dilution effect reduced with W) but the plasma parameters in the divertor might be not tolerable.

The plate temperature is high (> 40 eV) with significant heat load (> 15 MW), which results in very high tungsten production (Z_{eff}> 2 due to W ions).

Therefore the effect of seeding, which might be necessary, will be analyzed. Simulations results will be presented for neon and argon impurities for different values of plasma density and confinement level (H_{98}) and also for discharges with lower fusion performance but closer to the ITER modes of operation.

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Integrated modeling of impurity transport in core and SOL/divertor plasmas

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Impurities play a key role to reduce the heat load on divertor plates in tokamak plasmas. Simulations by an integrated divertor code SONIC for a full non-inductive current drive scenario in JT-60SA showed that the low divertor heat load (< 10 MW/m^2) with low SOL density (< 1.5 × 10^{19} m^{-3}), which is required for the scenario, was achieved by Ar gas puffing of 0.86 Pa m^3/s to the divertor region [1]. In the simulations, some amounts of Ar entered the core and thus the Ar accumulation in the core may reduce the core confinement by enhancing the radiation and dilution. On the other hand, the improvement of core energy confinement has been achieved with Ar seeding through the SOL region at high density in JT-60U H-mode plasmas [2]. Density peaking and dilution effects lowered the pedestal density at a given averaged density and led to high pedestal and core temperatures due to the almost constant pedestal pressure and the profile stiffness. The radiation was predominantly enhanced in the core edge region, while the radiation in the divertor region was less enhanced. This indicates the possibility to separately control the divertor heat load and the core energy confinement by impurity seeding at SOL and divertor regions. In order to study the transport of impurity in SOL/divertor and core regions and its effect on the divertor heat load and the core energy confinement, the integrated modeling code for core and SOL/divertor plasmas including impurity is one of the effective methods.

We have been developing an integrated modeling code TOPICS [3] (formerly called TOPICS-IB), in which a 1.5D core transport solver is coupled with SONIC [4]. One of the features of SONIC is the Monte-Carlo (MC) approach in an impurity-part code IMPMC, which has a lot of flexibility in impurity modeling such as interactions with walls and kinetic effects. The 1.5D core transport solver in TOPICS, however, treats only fully striped ions and can not treat many charge states of impurity and resultant radiations. To treat core impurity physics in TOPICS, we develop a core impurity transport code IMPACT, which calculates ionization/recombination processes, radiations, diffusive/convective transports of multi impurity species, and couple IMPACT with TOPICS. This integrated code is applied to the Ar core accumulation in the above JT-60SA scenario, where the Ar inflow to the core evaluated in the SONIC result is assumed to be injected as MC neutrals, the neoclassical transport is calculated by NCLASS, and anomalous diffusivities are set to the neoclassical level to consider the maximum accumulation. From integrated simulations, the Ar accumulation is found to be so mild that the plasma performance can be recovered by additional heating within the machine capability. Due to the strong dependence of accumulation on the pedestal density gradient, the high separatrix density is important for low accumulation as well as low divertor heat load. For more consistent modeling, the inflow of each charge state of impurity evaluated by IMPMC is used as inputs in IMPACT and the outflow of each charge state evaluated by IMPACT is used in IMPMC vice versa. We develop an interface model to couple the fluid-type code IMPACT and the MC-type code IMPMC, and study more consistently.

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Power exhaust management by impurity seeding in ASDEX Upgrade tokamak modeled by COREDIV code

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Due to material issues in a future thermonuclear tokamak reactor the maximum power delivered to the divertor plate needs to be limited. A way to reduce the energy dissipated in the divertor region is to increase the radiation in the discharge area, which would result in shifting the power exhaust to the chamber wall. To realize this scenario a controllable amount of carefully chosen impurities needs to be supplied to the plasma discharge.
This paper uses numerical modeling with the self-consistent COREDIV code [1] to analyse the influence of the impurity seeding with N_2, Ar and Kr on the key plasma parameters. In COREDIV the core plasma is modelled by 1-dimensional transport equations with semi-empirical transport coefficients for densities and temperatures, modified to describe the transport barrier, and in the scrape-off layer (SOL) by 2-dimensional fluid equations developed by Braginskii and by rate equations for all ionization states for each impurity. The coupling between the core (defined as the region limited by the separatrix) and the SOL regions is imposed by appropriate continuity equations at the separatrix.
The calculations for each impurity are directly compared to the experimental results obtained on ASDEX Upgrade (AUG) tokamak with the same impurities [2]. Specifically, the simulations are performed for shots: #29254 (with N_2 seeding) and #29257 (with Ar seeding), both with 14 MW auxiliary heating power. A good agreement between simulation and experiment for the concentrations of N_2 (1.3 %) and Ar (0.2 %), the respective radiation fractions, total radiations and Z_{eff} is obtained. The calculated values of the radiation fraction in the core region for N_2 is smaller (0.15 MW) than for Ar (0.6 MW) and the main radiation fraction in the core is due to W (7.3 MW in the case of N_2 and 7.6 MW in the case of Ar seeding). The calculated electron temperature (T_e^{plate}) on the divertor plate can be reduced from the unseeded value (30 eV) down to 10 eV for N_2 seeding and down to 13 eV for Ar seeding, respectively. This values correspond well to the experimental results of T_e^{plate} measured by Langmuir probes before the injection of impurities and during N_2 and Ar gas seeding. There is a remarkable agreement for the heat flux values as well. However, there is a discrepancy between the measured and simulated W concentration (factor 6). In order to access the reason for these differences, simulations will be also performed with the promt re-deposition model switched on (above presented results do not account on the prompt re-deposition phenomena) and for different assumptions regarding the radial transport in the SOL. Additionally, for the same plasma parameters (density, temperature, auxiliary heating) and configuration (electric current, magnetic field) a scan with Kr seeding is performed. We simulate the shot #29986 at t = 4.3 s with higher auxiliary heating power (18 MW), where the simultaneous influence of N_2 and Kr impurities on the plasma and divertor parameters is studied.

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Magnetic Field Models and their Application in Optimal Magnetic Divertor Design

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At present, several plasma boundary codes exist that attempt to describe the complex interactions in the divertor SOL (Scrape-Off Layer). The predictive capability of these edge codes is still very limited. Yet, in parallel to major efforts to mature edge codes, we face the design challenges for next step fusion devices. One of them is the design of the helium and heat exhaust system. In past automated design studies, we introduced optimal design methods to reduce the often excessive heat loads that threaten the divertor target surface. To this end, coils were controlled to improve the magnetic configuration. Results of these studies indicated large potential reductions in peak heat load by an increased magnetic flux divergence towards the target structures [1,2].

To obtain these results, a perturbative magnetic field model was used in combination with a plasma edge model. This perturbative model assumes that magnetic field changes due to small coil perturbations can be sufficiently described by analyzing the effect of external coil changes in vacuum. The influence of altering plasma currents and their position is hereby neglected for the small changes concerned. However, because of the rather large magnetic field changes that were found in past automated design studies, the legitimacy of the perturbation approach has to be questioned. Therefore, we have recently included a free boundary magnetic equilibrium (FBE) solver into the simulation framework to improve the magnetic field model [3].

In this paper, we analyze under which conditions the simplified perturbation approach is legitimate by comparing both models. First results will be shown for an optimized heat exhaust of the new WEST (Tungsten Environment in Steady-state Tokamak) divertor currently under construction in the Tore Supra tokamak at CEA (Commissariat à l'Energie Atomique, France). It is found that the sensitivities and the related optimization paths are strongly affected.

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A model of self-similar radiative transfer in resonance lines for testing the edge plasma codes

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The representation of kinetic equations in the self-similar variables allows one to obtain analytic solutions, which may be very helpful for testing the respective blocks of numerical codes for transport phenomena. The examples include the steady-state collisional kinetic equations for electrons [1] and neutral atoms [2] in a strongly inhomogeneous plasma. The self-similarity appears to be applicable to the cases of nonlocal (non-diffusive) correlations of the distribution function like the cases of superthermal electrons [1] and fast neutrons produced by the charge exchange [2]. Another type of self-similarity was found [3] for the non-steady-state Biberman-Holstein (B-H) equation for radiative transfer in resonance atomic/ionic lines in a homogeneous media. Here again the self-similarity is expressed in terms of characteristics of nonlocality of the B-H radiative transfer.

In this paper we extend the approach [1,2] to the case of the steady-state Biberman-Holstein equation in an inhomogeneous plasma slab. It is shown that for some types of similarity of spatial profiles of three characteristics, namely, background plasma density, line shape width and non-radiation source of atomic excitation, the profile of excited atoms density may be described analytically in terms of the similarity of the above-mentioned profiles. The revealed cases of analytical solutions are suggested for testing the radiative transfer codes in edge plasmas, including the codes for radiative transfer in a background plasma and for radiation losses by an impurity in plasmas.

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Effect of PSI on momentum input to plasma-facing material surfaces

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Power transmission factor through sheath has been intensively studied so far [1, 2]. But, momentum input to plasma-facing material surfaces would be also important, especially melt layer on tungsten walls or lithium wall in fusion devices. There have been few reports on this subject although plasma momentum related to propulsion of thrusters has been investigated [3-5].

Recently, momentum input to tungsten target from high-heat-flux deuterium plasma was observed experimentally as shown in Fig.1 [6], and was estimated numerically with a simple and phenomenological argument. Motivated by the above observations, a comprehensive formulation for momentum input will be tried to include a momentum increase due to ion reflection on the wall by defining a momentum reflection coefficient in addition to ion particle and ion energy reflection coefficients, a momentum due to evaporated atoms and clusters with account of cos θ emission law, an increase due to sputtering, effects of surface morphology, and plasma flow.

Plasma fluctuation or turbulence with spatial variation would affect some inhomogeneities of momentum influx to the wall which could couple with fluid-mechanical motion such as Kelvin-Helmholtz instability of melted tungsten layer.

New experimental results will be introduced in relation to the above arguments.

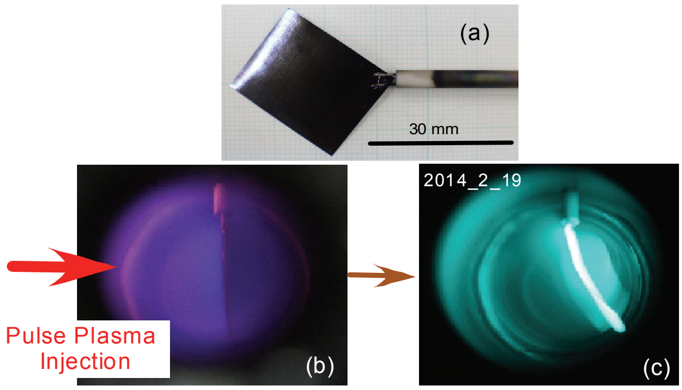

Fig. 1 Movement of tungsten foil due to pulse plasma injection. (a) tungsten foil and its support, (b) total light photo of tungsten foil looked through vacuum window at mid point before pulse plasma injection and (c) photo through the interference filter of tungsten atomic line just after the pulse plasma injection .

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Particle simulations on effects of plasma-tungsten interaction to the prompt redeposition and the self-sputtering

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Net Erosion of plasma facing walls is one of the most serious concerns in future fusion reactors. Significant heat and particle loads can cause erosion by sputtering, vaporizing, grain ejection, or melting. Tungsten plasma facing walls are planned for use in the divertor of future devices, due to its high threshold energy for sputtering by light particles like hydrogen and helium. However, physical sputtering increases considerably by other heavier impurities. One positive aspect of sputtered tungsten particles once ionized is its prompt re-deposition behaviour in a magnetic confinement device. It is believed that its short ionization length compared with its large gyro radius works as a trap so that the ejected tungsten particles can not escape but return to the surface close to its original location. Besides with experimental studies [1], several simulation works have been carried out by Monte-Carlo approaches [2] and Particle-In-Cell (PIC) approaches [3]. Both studies examined situations in which ejected tungsten particles do not strongly disturb the condition of divertor and SOL plasma. However, as the ejected tungsten particle flux increases, as in the case of disruptions or ELMs, the interaction between the ejected tungsten particles and the plasma becomes important, e.g., cooling of the plasma by ejected tungsten and resultant reduction of the melting, sputtering etc. Thus, a code based on the PIC technique was developed to study self-consistently the plasma-tungsten interaction. In the simulation, tungsten ionization, recombination, line radiation, electron from tungsten, recycling neutrals and many other kinds of reactions and particles were treated simultaneously. The code treats one dimensional in space and three velocity components. Thus, simulations were repeated for different magnetic pre-sheath conditions by changing the angle between the magnetic field line and the wall surface and rate of the total and prompt re-deposition were examined. As well as this prompt re-deposition phenomenon, behaviours of tungsten self-sputtering were studied. It was concerned that the re-depositioning tungsten accelerated by the sheath potential may reach an energy which sputtering yield exceeds unity. If a particle incident generates more than one sputtering particles, an impurity burst may happen and cause a significant erosion of tungsten. Performing PIC simulations for various conditions on the magnetic field and the incident plasma, possibility of the sputtering burst was investigated.

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Simulation analysis of carbon deposition profile in the closed helical divertor configuration in the Large Helical Device

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A closed helical divertor has been installed in the inboard side of the torus for effective particle control in the peripheral plasma in the large helical device (LHD). The divertor plates made of isotropic graphite (carbon) have been installed along the strike points on the vacuum vessel. The front surface of the divertor plates is faced to the inboard side of the torus to enhance the density of neutral particles behind a dome structure which is installed along the space between the two divertor plates in the inboard side [1].

Recent long pulse discharges in the LHD have often been collapsed by emission of large amounts dusts released from the closed helical divertor region near upper and lower ports [2]. It was found that the exfoliated carbon-rich mixed material layers were deposited on the surface of the closed divertor components on the site [3]. It is possible that the carbon sputtered from the divertor plates formed the deposited layers on the surfaces to which the divertor plates directly face in the divertor region, and the exfoliated deposition layers caused the emission of large amounts of dust by interaction with the peripheral plasma, which could lead to the radiation collapse in the long pulse discharges. For this reason, the configuration of the closed helical divertor near the upper and lower ports was modified in order to suppress the carbon deposition in the last experimental campaign in FY2014.

Carbon deposition profile in the divertor region is calculated by a three-dimensional peripheral plasma simulation code coupled with a neutral particle transport code (EMC3-EIRENE) [4-7]. The detailed three-dimensional configuration of the modified structure of the closed divertor near the upper and lower ports is included in the simulation. It is found that the calculations of the heat load profile on the surface of the modified divertor plates is consistent with the temperature profiles on the divertor plates observed with an infrared camera, which supports the validity of the simulations of the carbon deposition profiles using the code.

The simulation analysis will propose an improved closed divertor configuration to suppress the carbon deposition in the closed divertor region. It can contribute to extension of the plasma duration time in long pulse plasma discharges in the LHD.

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TECXY study of a liquid lithium divertor for DEMO

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Very high heat load densities on the target plates are envisaged for devices of DEMO size. For economic reasons, the machine should work for long periods of time before its walls need replacing. Those factors mean that most resistant materials are needed for the divertor plates. One of the proposed divertor designs for the prospective thermonuclear reactor DEMO is a liquid metal divertor. A preliminary assessment using the TECXY code is reported in this contribution.
The vast majority of planning and testing focused so far on solid materials, such as carbon composites (CFC) and tungsten. Among other disadvantages, like tritium retention by carbon and heavy element plasma pollution in case of tungsten, solids can withstand strictly limited power fluxes before their surface is gradually but irreparably destroyed. Liquids, although they release building material to the chamber more easily than the solids mentioned, can continuously be replenished, so that their surface may remain undamaged. Molten Li flowing through a capillary porous system seems to be a promising design since any droplet release driven by Lorentz forces is strongly suppressed.
Based on current, encouraging experience with liquid lithium limiters studied in the FTU tokamak [2], a similar solution is considered for the DEMO divertor. A new, low H recycling mode (due to strong pumping properties of Li) is expected. Neutral H radiation is in some sense substituted for by intensive radiation of lithium. This comes at the price of plasma contamination by released Li atoms.
The effectiveness of lithium radiation in cooling plasma streaming towards the plate, temperature and divertor heat load profiles, as well as distributions of Li^{n+} impurities in the DEMO divertor are studied with the help of 2D, fluid code, TECXY. The simulations are carried out for several values of plasma flux and power crossing the separatrix.

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