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Book of Abstracts

Table of Contents

Edge and Divertor Plasma Physics

Ammonia Molecular Assisted Recombination processes in nitrogen seeded deuterium plasmas

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Effects of the impurity seeding of nitrogen into the divertor plasma has been studied in Tokamak devices (e.g. ASDEX Upgrade, JET) to protect the tungsten divertor target from heat loads in ITER. Experiments have shown ammonia formation during H_2 discharges with N_2 seeding [1]. Here we investigate the chemical processes of N_2 or ND_3 seeded D_2 plasmas in the PISCES-E RF plasma device by using a rate equation model and a calibrated Electrostatic Quadrupole Plasma (EQP) analyzer, a combination of an ion energy analyzer and mass spectrometer. Because PISCES-E has relatively larger volume/surface ratio than previous laboratory experiments [2], the investigation is more focused on volumetric reaction processes. Typical plasma parameters used in this research were $T_e \sim$ 3-5 eV, $N_e \sim 10^{16}$-10^{18} m^{-3}, and the total pressure was P_{tot} = 10 mTorr while the fractions of the gas species were changed. A Residual Gas Analyzer showed formation of deuterated ammonia ND_3 which was up to 10% in the chamber during D_2/N_2 discharges. The model calculation suggested a possibility of a new Molecular Assisted Recombination (MAR) process enhanced by ammonia molecules which would occur in an electron temperature range 2-7 eV. This new MAR process, Hydronitrogen-MAR (HN-MAR), takes two steps like as other MAR processes via hydrogen [3] or hydrocarbon molecules [4]. In the first step of the HN-MAR, ND_y^+ (y=3, 4) molecular ions are formed due to the neutralization processes of D_x^+ (x=1-3) ions, i.e. charge/D^+ exchange reactions with ND_3. EQP measurements show formation of ND_y^+ during D_2-N_2 and D_2-ND_3 discharges. In the low-density D_2-N_2 plasmas ($N_e \sim 10^{16}$ m^{-3}), EQP measurements show ND_4^+ is dominant species when the partial pressure fraction of N_2 is more than 5%. In this partial pressure configuration, the model estimates the main neutralization processes of D^+ and D_3^+ are the first step of HN-MAR mentioned above instead of the surface loss. This neutralization channel of D_3^+ mainly contributes to form ND_4^+ in N_2 fraction range up to 5%. Even when the partial pressure fraction of ND_3 was only 0.3%, the ion density fraction of ND_4^+ was about 10%. This would be important observation for all PMI studies using nitrogen seeded plasmas. For our experimental configuration, when $N_e > 10^{18}$ m^{-3} the loss process of ND_4^+ ion will be dominated by electron ion recombination, which is the second step of HN-MAR. Therefore, high plasma density ($N_e \sim 10^{18}$ m^{-3}) is realized by adding Ar for investigating all the steps of the HN-MAR recombination process.

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abstract number 3

Initial Results from the Hotspot Detection Scheme for Protection of Plasma Facing Components in Wendelstein 7-X

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One of the main aims of Wendelstein 7-X, an advanced stellarator, is to investigate the quasi-steady state operation of magnetic fusion devices, for which power exhaust is an important issue. A predominant fraction of the energy lost from the confined plasma region will be removed by 10 so-called island divertors, which are designed to sustain a steady state heat flux of up to $10\,\mathrm{MW\,m^{-2}}$. They are subdivided into target elements which are covered with carbon fiber composites (CFC) layer on top connected via a copper interlayer to an underlying CuCrZr cooling block. A very important prerequisite for safe operation of a steady-state device requires an automatic detection of hot spots and other abnormal events. Simple temperature limits in IR thermographic images might get misguided by hotspots. In order to protect the divertor elements from overheating and to monitor power deposition onto the divertor elements, a near real-time hot spot detection algorithms for the analysis of carbon PFCs is implemented and tested at GLADIS [2].

During the recent operational campaign, inertially cooled test divertor units (TDU) are installed in order to prepare the steady state operation with water cooled divertor units. One of the difficulties in the hotspot detection in the carbon based machine is the co-deposition of the so called surface layers with poor thermal connection to underlying bulk material. We have developed and successfully tested a method [1, 2] to characterize surface layers. The surface layers can be detected in a steady plasma discharge during the initial rise and decay in temperature when a strike line touches the parts of divertor or wall and it can also be detected by manually modulating ECRH input power. This allows to detect overheated areas with reduced false positives. The algorithms, which have been prepared earlier and tested in GLADIS, were now successfully applied at W7-X. We were able to present automatic, near real time detection of hot spots (e.g. due to leading edges) and identification of surface layers.

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First FTU Tin liquid limiter results and their interpretation with the edge plasma code TECXY

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At the end of 2016 and for the first time in a tokamak device, a Capillary Porous Tin Liquid Limiter (TLL) was exposed to the plasma on FTU (Frascati Tokamak Upgrade).
To characterize TLL under FTU plasma discharges, the following diagnostics have been used: 1) a fast infrared camera observing the whole limiter surface 2) 4 Langmuir probes for measuring the electron temperature and density on both sides of the limiter and at two different poloidal positions 3) thermocouples for the measurement of the TLL temperature 4) a visible spectrometer looking at the limiter for the monitoring of Tin and D line emission. Thermal analysis has been carried out by applying the ANSYS code to the real geometry of the TLL. In the simulation we have taken into account: the shape and the position of the FTU plasma as reconstructed by the equilibrium code, the net power to the SOL (P_{input}-P_{rad}) and the decay length λ of the energy as obtained by the Langmuir probes at different TLL radial positions. In the first experiments on FTU, the tin limiter, not actively cooled, was inserted progressively in scrape-off-layer (SOL) up to a distance less than 0.5 cm from to the last closed magnetic surface (LCMS). It was exposed in ohmic plasma discharges with Ip=0.5MA, B_t=5.3 T and n_e=6.0x10^{19}m^{-3}.
A maximum heat load of 11 MW/m^2 was found by ANSYS code with the tin limiter at the distance < 0.5 cm from the LCMS in agreement with the indications coming from the Langmuir probes measurements. Within the range of the surface temperatures obtained in these experiments, the tin evaporation was negligible. Plasma performances did not change and no increase of Z_{eff} value was observed.
In this work, a comprehensive description of these first results with TLL will be given together with the prediction of the TECXY code, alreary satisfactorily used in the past on FTU to interpret many experimental data of the SOL region including those relative to the first application of the liquid lithium limiter (LLL) as first wall material.

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Effect of negative ions on the sheath formation by emissive probe and laser induced fluorescence methods

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To understand the effect of negative ions on the sheath formation, emissive probe (EP) and laser induced fluorescence (LIF) were used. EP have been used to directly measure the plasma potential profile and the sheath/presheath boundary is identified from the slop change of the emission current. LIF determines ion flow velocities and ion temperatures from the broadening of fluorescence lines. Basic plasma parameters such as plasma density and electron temperature are measured by single electric probe. Negative ion plasma is generated by the discharge of $Ar+O_2$ gas in a cubic chamber (24 x 24 x 24 cm^3) with DC filament source with the following conditions: $n_e \sim 10^8$ cm^{-3}, $T_e \sim 2$ eV, $T_i \sim 0.1$ eV). O_2 gas ratio have been changed 0 to 10 % to investigate the effect of negative ion to sheath formation. Change of sheath width, plasma potential, ion velocity due to ratio of negative ion concentration will be presented.

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Observations of efficient lower hybrid current drive at high densities in C-Mod by avoiding parasitic wave interactions with the scrape-off layer

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Efficient lower hybrid current drive (LHCD) is crucial for realizing an economical tokamak reactor, but a LHCD 'density limit' was found in Alcator C-Mod experiments [1], which appeared to prohibit effective non-inductive current drive at densities above $\bar{n}_e \approx 1.0\times10^{20}$ m^{-3}. However, in those experiments the total plasma current was typically set at a low value (~0.5 MA) with the goal of fully replacing the inductive component with non-inductive LHCD. The target plasmas therefore had high Greenwald fraction and under these conditions the scrape-off layer (SOL) plasma has broad density profiles [2]. Experiments have since identified LH wave interaction with the edge/SOL plasma as a potential contributor to the observed density limit phenomenon: (1) LH power was found to be deposited thermally near the last closed flux surface [3], (2) collisional loss for waves that enter the divertor region were found to be significant [4] and (3) parametric decay instabilities (PDIs) were observed at high density [5]. On the other hand, the density for onset of PDIs was found to increase with plasma current, suggesting that Greenwald fraction – and associated density shoulders in the SOL – may be a player.

Motivated by these results, plasma current was raised to 1.4 MA in C-Mod's 2016 campaign to see if PDI onset could be suppressed and efficient LHCD recovered at high density. Indeed, above 1.2 MA, the onset of ion cyclotron PDI was found to be delayed to $\bar{n}_e \approx 1.4\times10^{20}$ m^{-3}. In this case, hard X-ray production rate was increased by three orders of magnitude compared to low current (0.5 MA) plasmas at the same density. The drop in the loop voltage, ~0.2 V, is consistent with the injected LH power (600 kW) and an engineering current drive efficiency that scales as $1/\bar{n}_e$ as predicted by the theory. These observations, as well as a ray-tracing/Fokker-Planck analysis will be presented at this conference.

These results demonstrate the importance of controlling the edge/SOL plasma for efficient LHCD. They also support the idea of placing launchers on the high-field side (HFS) in double-null configurations [6, 7] to attain efficient CD at high Greenwald fraction. In this case, the HFS SOL is magnetically disconnected from the LFS. Because turbulence is suppressed on the HFS, density shoulders are absent. Similar to the experiments reported here, this situation is expected to help suppress first-pass parasitic wave-SOL interactions, including PDI and wave scattering, thus enabling efficient LHCD at high density.

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Fast-camera imaging of edge turbulence on Alcator C-Mod and W7-X

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Passive fast-imaging has proven to be fruitful on both tokamaks and stellarators for imaging filamentary fluctuations in the plasma boundary and for studying their dynamics. Fast cameras recording in the visible range have revealed the 3D structure of the SOL turbulence known as "blobs" to be emissive filaments aligned with the magnetic field. The filaments have been observed to propagate perpendicular to the field at speeds on the order of 1 km/s. More recently the dynamics of filaments in the divertor and private flux regions of the MAST [1], Alcator C-Mod [2], and NSTX-U [3] tokamaks have been studied. In this work we present additional analysis of the C-Mod fast-imaging, where intermittent filaments were recorded in an oblique view of the X-point region using a Phantom V710 camera imaging at ~400,000 frames per second. The filaments' emissivities in the R, Z plane of the divertor were reconstructed and tracked in time using EFIT field line projections.

During the initial run campaign of the W7-X stellarator, filaments rotating around the magnetic axis were observed by Kocsis, et al. using a fast camera imaging at up to 46.5 kframe/s [4]. For W7-X's 2017-18 run campaign, the first with a divertor, we added an additional fast-framing camera to the W7-X diagnostic suite. It has a tangential view of the machine cross section, including part of a divertor module. The view is coupled to the camera via a coherent fiber bundle. The spatial resolution is roughly 2 cm in the view and the typical plasma brightness limits the fastest frame rate to ~30 kframes/s. We observe ~10 kHz emission fluctuations along the divertor target and filaments in the plasma boundary that rotate around the boundary. Filaments in W7-X plasmas are compared to those obtained in the divertor region of C-Mod.

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Effects of the electrostatic shear on the edge plasma in a two-field $\kappa - \epsilon$ like model implemented in the transport code SOLEDGE2D-EIRENE

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Spontaneous generation of transport barriers at the edge of magnetically confined plasma characterizes the so-called high-confinement mode, or H-mode, in which it has been observed that the structure of the potential induces a strong $E \times B$ velocity shear that may explain a reduction of the turbulence [1]. This contribution aims to 1) introduce the description of such shearing phenomena in a 2D κ-ϵ-like reduced model for edge plasma turbulence, inspired by the RANS (Reynolds Averaged Navier-Stokes) approach, 2) define and investigate the model free parameters and 3) study the implications on plasma properties by implementing the model in a transport code. In this $\kappa - \epsilon$ approach, a dedicated equation describes the evolution of turbulence energy, akin to κ, and its non-linear self-regulation mechanism. A second equation on ϵ introduces a turbulence regulating field driven in particular by $E \times B$ shear flows. More precisely, the electrostatic potential being self-consistently calculated solving current equation, one finds that the radial electric field is proportional to the opposite of the ion pressure gradient in the closed field line region (force balance) and roughly proportional to the gradient of electron temperature in the scrape-off layer (boundary condition on the parallel current). This implies a strong $E \times B$ shear at the separatrix driving the growth of ϵ and potentially a suppression of the turbulence energy. Ultimately, the reduction of κ leads to a strong pressure gradient of ion pressure re-enforcing the $E \times B$ shear and stabilizing the onset of a transport barrier as can be found in [2].

The $\kappa - \epsilon$ model is implemented in the transport code SolEdge2D-Eirene and 2D diffusivities maps are computed hence fully constraining transport properties for given plasma scenario. The free parameters of the model are defined comparing SolEdge2D-Eirene simulations with experiments. Then power ramps are simulated and the possibility to recover flavors of the L-H transition with the reduced $\kappa - \epsilon$ turbulence model is discussed.

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A Spatially Hybrid Fluid-Kinetic Neutral Model for SOLPS-ITER Plasma Edge Simulations

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Design of tokamak divertors relies heavily on plasma edge codes such as SOLPS-ITER to extrapolate from current experiments to reactor-relevant operation conditions. These codes typically combine a finite-volume code (B2.5) to resolve fluid plasma transport equations, with a Monte-Carlo code (EIRENE) to solve the kinetic neutral transport. These codes suffer from significant increases in runtime for the (partially) detached plasma conditions that will be required for future fusion reactors. Especially the computational cost of the Monte-Carlo code for neutral transport calculations scales badly with increasing number of charge-exchange reactions. Since the fluid assumption becomes more accurate in these very same circumstances, fluid neutral models come to the fore as a valid alternative.

In this context, recent efforts of Horsten et al. [1] have focused on deriving transport coefficients and boundary conditions for fluid neutral models directly from the kinetic description. By starting from the same atomic physics databases for atom collision data (AMJUEL) and microscopic partial reflection of neutrals (TRIM), full consistency with the present EIRENE code was sought. These advanced fluid neutral models have recently been implemented in B2.5, and have shown excellent agreement with EIRENE for a detached benchmark case [2]. The fluid neutral model equations were hereby discretized on the plasma grid, which typically does not extend up to the vessel walls. The benchmark was therefore conducted on a geometrically simplified ‘Slab’ case instead of a real tokamak geometry, to circumvent this shortcoming.

In this contribution, we extend these advanced neutral models to accurately calculate the neutral transport in real tokamak geometries. To this end, a spatially hybrid fluid-kinetic neutral model is used for the neutral transport. The gap between fluid neutral simulation and vessel wall is closed by using the Monte Carlo code EIRENE in this region, while appropriate interface conditions are imposed at the grid interface. As such, the rarified neutral regions outside the plasma grid are treated with a kinetic neutral model. The approach is demonstrated on a plasma edge transport simulation of the ITER reactor with only deuterium atoms and benchmarked with a fully-kinetic neutral model.

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Increased Divertor Plasma Fluctuations with Detachment

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Divertor plasma turbulence levels increase when divertor detachment is approached in DIII-D. The root mean square (RMS) of fluctuation levels of saturation current, J_{sat}, floating potential, V_{fl}, electron temperature, T_e, and electron density, n_e, measured by a divertor scanning probe, possessing a bandwidth of DC-400 kHz, increase to 50-70% of the average levels as the divertor plasma starts detaching ($T_e < 10\ eV$) and becomes more homogenous. Data from instruments with lower sampling rate, such as divertor Thomson Scattering (DTS) and target probes, also indicate enhanced levels of scatter, near 50-70%, consistent with higher time resolution measurements.
Increased plasma fluctuation levels often lead to enhanced radial particle $\tilde{\Gamma}_r = \langle \tilde{n}\tilde{v}_r \rangle$ and heat $\tilde{Q}_r = 3k\langle \tilde{T}\tilde{v}_r \rangle - 3kT\tilde{\Gamma}_r/2$ transport and, in this case, lead to divertor plasma mixing and longer profile decay lengths. The increased mixing and transport lead to increased plasma-neutral interaction and therefore accelerate the detachment process. Fast filtered imaging of the divertor is used to identify the 2D structure of divertor fluctuations.
The upstream pedestal is often affected by divertor detachment, where upstream pressure drops by at least 10-20%, leading to reduced plasma performance [1-2], and the underlying physics mechanisms of such interaction is unknown although it has been fully ascribed to neutral fuelling. We have compared fluctuation levels at the divertor and at the midplane by using two scanning probes and see increased turbulent transport by 30-40% at the midplane during detachment, and a widening of the scrape off layer (SOL) density by factors of 5 and temperature by 30%. Modelling by SOLPS 5.1 with enhanced transport coefficients will be performed and compared to experimental data.

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SOLPS-ITER simulations of Ne-impurity experiments on EAST

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Optimal design of power and particle exhaust is key for future nuclear fusion reactors. Therefore, power dissipation by impurity radiation and the use of double null (DN) magnetic configurations are presently studied to improve power handling in the DEMO divertor [1]. With the possibility of long pulse operations and the feasibility to produce a DN configuration, the EAST tokamak is capable of pursuing advanced exhaust experiments including seeding studies in combination with alternative magnetic configurations. Neon seeding experiments in disconnected double null with main upper divertor configuration (or so-called DDN-up) with a divertor separation of around 1 cm have been carried out for the first time at EAST with plasma currents of 0.4 MA, a toroidal magnetic field of 2.5 T and auxiliary heating of P_{aux} = 4MW in total. In this discharge Neon seeding induces H-L-H-transitions and a maximum observed radiative fraction $f_{rad}=P_{rad,total}/P_{aux}$ of around 50%. In view of future EAST impurity experiments, numerical simulations of this discharge are set up to assess the most dominant physics processes induced by Neon seeding.

This numerical study will provide a SOLPS-ITER simulation to interpret this EAST discharge. As a result, the influence of geometry and neutral viscosity as well as volumetric recombination, plasma drifts and currents can be taken into account as required for detachment modelling [2]. Similar modeling of Argon seeding experiments in ASDEX-Upgrade has been successfully pursued previously with the SOLPS5.0 code in [3]. This work is taken as a reference during the SOLPS-ITER Neon seeding modelling.

In a first step, the presented simulations are important towards code validation for SOLPS-ITER on EAST. To this end anomalous transport parameters are determined manually in order to obtain agreement with experimental data as done in [3]. A comparison between the obtained power scrape-off width by SOLPS-ITER simulations and the predicted width by simplified analytical models (e.g. [4]) is part of this contribution. As the configuration is a DDN-up configuration, the power dissipation will especially take place in the upper divertor. However, based on the Langmuir probe data from the low field side, the influence of the lower divertor cannot be neglected. In order to study the influence of the lower divertor, the discharge is modelled with both an USN (upper single null) and a DDN-up configuration. Finally, the results are explored to study the relevancy of DN configurations with a metal wall for DEMO.

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Measurement and correction of the 1/1 error field for W7-X island divertor configurations

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In recent experiments on Wendelstein 7-X (W7-X) a small $m/n = 1/1$ error field with relative amplitude below 10^{-4} was measured by mapping the vacuum field with an electron beam. This error field could be compensated with a set of five trim coils installed for such corrections, which improved the symmetry of power distribution in the divertor by up to a factor of two. This result is essential for future steady state operation, as otherwise the performance could be limited by overloading the plasma facing components.

W7-X is a large optimized stellarator aiming to verify the fusion-reactor relevance of optimized stellarators. To eventually reach quasi-steady state discharges it is equipped with an island divertor [1], where an island chain, in most cases $m/n = 5/5$ at the resonance with rotational transform $\iota = 1$, forms the edge region. Due to its resonance nature the island divertor can be strongly perturbed by error fields caused e.g. by manufacturing and installation errors. During the assembly a special care was taken to minimize deviations and to correct accumulated errors [2]. Nevertheless, the remaining error fields are not necessarily negligible. In this contribution we present direct measurements of the most deleterious $m/n = 1/1$ error field and demonstrate that it can be successfully compensated.

The confining field of a stellarator is completely formed by external coils, therefore error fields can be precisely determined in vacuum by the flux surface mapping technique [3]. In the W7-X island divertor configurations, co-resonance of several modes at the same position complicates such measurements. To separate the 1/1 error field the magnetic configuration has to be modified either by cancelling the 5/5 islands with a set of control coils or by shifting the resonance position radially. In the limit of the latter approach the helical shift of the magnetic axis provides a measure of the 1/1. Both of these methods were conducted experimentally and result in the normalized 1/1 amplitude of about $(0.5 - 0.7) \cdot 10^{-4}$. Following the flux surface mapping measurements, the trim coils were applied in plasma discharges to symmetrize the power distribution between the divertor modules, as observed with a set of infra-red cameras and a set of thermocouples. It is confirmed that the divertor symmetry is improved by cancelling the 1/1 mode determined above.

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Simulation study on radiative divertor for HL-2M by impurity seeding with SOLPS-ITER

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HL-2M is a new medium-sized copper-conductor tokamak and is designed to operate with the ability to generate multiple types of advanced divertor configurations [1,2]. With the aims of high performance plasma and engineering toward ITER and even a fusion reactor, heat exhaust would be a serious problem for HL-2M. As an effective way to exhaust the heat power, radiative divertor operation has been demonstrated on many devices, using a variety of radiating impurities (e.g., N_2, Ne, Ar, etc). In this work, a series of impurity seeding is considered from the modeling point of view to solve the HL-2M heat exhaust problem by radiative divertor. SOLPS-ITER [3,4] simulations are performed for various kinds of impurities (N_2, Ne and Ar) from the lower dome with the standard lower single null configuration.

Simulation results demonstrate the similar high efficiency in reducing heat power of N_2, Ne and Ar, and impurity seeding is an efficient method to assure the power load to the target plates at an acceptable level even at high heating power (20 MW) during HL-2M discharges. The efficiency of the radiative power exhaust for different impurities are compared, and Ar can reach the highest radiative fraction and it radiates strongly in the divertor region as well as inside the separatrix. In addition, modeling shows that the total radiation fraction increases with increasing seeding rate of N_2 and Ne, with most of the radiation coming from the divertor region. Simultaneously, the impacts of impurity radiation on the ratio of radiated power to the power into the scrape-off layer and on the effective charge number are analyzed with different seeding rates. Moreover, the divertor plasma is found to have a transition from conduction-limited regime to complete or partial detachment with the increasing of seeding rate. In particular, the peak heat flux and peak electron temperature on the targets are reduced obviously and both move away from the strike point with increasing throughput of impurities. For controlling plasma-surface interactions, the results indicate that Ne is preferential to be used as the radiator during HL-2M operations. However, the level of Ne seeding rate should be modulated to maintain the effective charge number at acceptable values.

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Sensitivity analysis of plasma edge code parameters through Algorithmic Differentiation

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Anomalous radial transport coefficients, boundary conditions and reaction rates are among the main sources of uncertainty within plasma edge modeling. In principle, an analysis to determine the sensitivity of code results like maximum target heat load or temperature, or ion saturation current, on the uncertainty on model parameters can be easily implemented through finite differences. However, this incurs in error accumulations and allows to scan only one parameter at a time, requiring a huge computational effort. A possible solution to overcome this issue takes advantage of Algorithmic Differentiation (AD) [1].
AD assumes that a code is a series of sequential elementary operations that can be easily differentiated and as such, AD tools preprocess the source code and produce a model of the code which contains the additional derivative information. Usually, when dealing with sensitivity studies, the so-called forward differentiation approach is adopted, while in optimization processes or adjoint problems, the reverse differentiation approach is more efficient as its cost is independent of the number of design parameters. Moreover, multi-directional AD can be exploited during forward differentiation, meaning that the sensitivity of all objective functions on one model parameter can be evaluated with a single simulation, significantly reducing the computational effort required with respect to finite differences. Therefore, the user needs only to select the AD approach that efficiently matches the problem to be solved and then use AD tools to produce a differentiated model of the code.
In this work we demonstrate the feasibility of applying AD to plasma edge codes by using the TAPENADE tool [2] on the SOLPS-ITER [3] code in the forward mode. A simple slab case is adopted as reference scenario, with fluid description for neutrals, and the target peak heat load and strike point temperature are chosen as objective functions of which the sensitivities are evaluated. The model parameters considered in this first preliminary study are transport coefficients, both for the plasma and neutrals fluid description. The accuracy of the AD approach is proven by comparing the results with that obtained through finite differences.
Once the feasibility has been proven, the new technique is then applied to the more complex case of a real tokamak geometry. The sensitivity of transport coefficients and boundary conditions on the target peak heat load and strike point temperature is assessed, adopting again a fluid description for the neutrals, and proving how forward multi-directional AD can be exploited for an efficient and accurate sensitivity analysis in the framework of plasma edge simulations. In perspective, this builds a solid base for an efficient and reliable tool for optimization studies and parameter estimation, notwithstanding the possibility to make use of the reverse AD.

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Ion Heat and Particle Transport in the ASDEX Upgrade H-Mode Pedestal from Ultra-fast CXRS measurements

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Ultra-fast charge exchange measurements (CXRS) of the H-mode pedestal reveal new physics insights into the edge localized mode (ELM). For the first time, the dynamics of both the ion and electron pedestal were measured with a time resolution of 100 μs during an entire ELM cycle [1]. At the ELM crash, the ion temperature (T_i) at the separatrix is three times larger than the electron temperature (T_e) indicating that the parallel ion heat transport is not negligible and can be comparable to that of the electrons in the SOL. The measurements are consistent with the presence of high T_i filaments in the far SOL during the ELM crash. Furthermore, the characterization of the edge gradient recovery reveals a difference between the ion and the electron channels: the ion temperature gradient is re-established on similar timescales as ∇n_e, which is faster than the recovery of ∇T_e. After the clamping of the maximum temperature gradients, T_i and T_e at the pedestal top continue to rise while n_e stays constant which means that the temperature pedestal and the resulting pedestal pressure widen until the next ELM. The phases in the pedestal recovery are correlated with the onset of magnetic fluctuations and the effect of underlying instabilities on the electron and ion transport is characterized by means of predictive simulations.

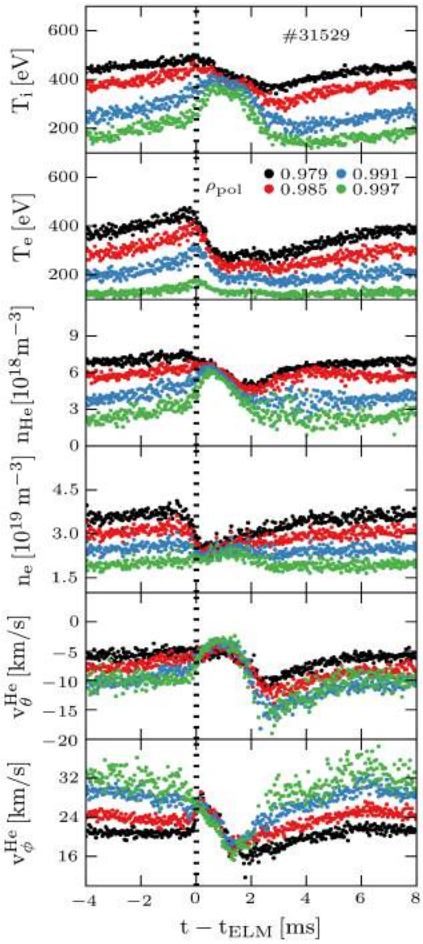

Fig. 1: Edge profiles evolution during an ELM cycle

The radial electric field (E_r) is determined via the radial force balance equation from the measurements of the poloidal and the toroidal impurity flows. At the ELM crash, E_r is found to reduce to typical L-mode values and its maximum recovers to its pre ELM conditions on a similar time scale as n_e and T_i. Within the uncertainties, the measurements of E_r align with their neoclassical predictions for most of the ELM cycle, thus indicating that E_r is dominated by collisional processes. However, between 2 and 4 ms after the ELM crash, other contributions to the ExB velocity, e.g. zonal flows or ion orbit effects, could not be excluded within the uncertainties. Similarly the ion heat transport in the pedestal is found to be close to the neoclassical level except between 0 and 4 ms after the ELM onset.

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The effects of particle recycling on the divertor plasma by a PIC-MCC modelling

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In order to ensure the lifetime of the divertor target, it is required to reduce the plasma temperature and heat-flux load to the target. During plasma surface interactions (PSI), the neutral particles are produced from the target surface by particle recycling. These neutral particles would collide with the incident plasma flux, thus radiate the energy and reduce the heat flux to the target, which could promote the achievement of the detachment [1]. The particle recycling includes two parts: (1) The reflection of the energetic deuterium (D) particles, which create the D atom; (2) The release of the absorbed D atom as the form of molecular D_2. The particle recycling depends strongly on the material property, i.e. for the carbon (C) wall, the reflection rate, which can produce D atom, is low; whereas, the tungsten (W) wall has high particle and energy reflection rates [2]. Therefore, it is important to understand the effects of particle recycling on the divertor plasma.

In this work, a PIC-MCC model, which can accurately describe the trajectories and collisions between charged particles and neutral particles, and the PSI processes [3][4][5], is developed to investigate the effects of particle recycling on the divertor plasma. The simulation domain is the whole scrape-off layer (SOL) in one-dimensional along the magnetic field line. At divertor plate, the reflected D and released D_2 [1] are considered. The collisions (elastic collision, excitation collision, ionization collision, charge-exchange collision, recombination collision etc.) are described by MCC method. It is found that the ionization collision of D atom can increase the plasma density and reduce the plasma temperature; the charge-exchange collision significantly contributes to momentum loss. The cases of divertor with C and W targets are simulated and compared. The results show that the energy and momentum losses of the C target are larger than those of the W target, which indicates that the detachment benefits from C wall, even without considering the C impurity radiation. The reason can be attributed to two aspects: (1) for W wall, the larger particle and energy reflection coefficients can produce the energetic atomic D, which can easily escape from the divertor region, resulting in the lower neutral concentration and power radiation; (2) for C wall, most of the neutral particles are recycled in the form of D_2 and dissociated into lower energy D atom, which can increase the neutral concentration in the divertor region.

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Pumping effect in tungsten divertor on plasma performance in EAST

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Neutral recycling behavior in the edge of magnetic fusion devices plays an important role in plasma performance. The status of boundary plasma, plasma shape, wall material, particle pumping efficiency, etc. will affect the edge neutral recycling behavior [1-3], which will further influence global plasma performance through altering edge density profile. The ITER-like upper tungsten (W) divertor, installed in EAST in 2014, has exhibited low recycling level and excellent heat exhaust capability compared to the lower graphite divertor, in which pumping effect could become more pronounced. Dedicated experiments have been carried out in EAST to investigate the effect of particle pumping in W divertor by turning on/off the top external cryopump that accounts for ~ 25% of the top cryopump capability and moving outer strike point (OSP) positions. Density ramping in upper single null configuration is employed.

The deuterium (D) recycling in W divertor increases when the external cryopump is shut down, which becomes more significant with higher OSP position, i.e., closer to the pumping slot. The H-mode is accessed and then transits back L-mode both at lower plasma density with the external cryopump shut down, compared to with external cryopump working. The plasma confinement is better with external cryopump on than off at higher OSP position, but no obvious difference is observed at lower OSP position. It is also observed that the onset of detachment in W divertor appears at lower main plasma density independent on OSP position. The radiative fluctuation with a central frequency at 5 kHz is observed prior to detachment and last longer with external cryopump shut down.

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Simulations of SOL-Divertor Plasmas in EAST with Tungsten Divertor

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Research at the experimental advanced superconducting tokamak (EAST) is strongly focused on preparations for the exploitation of ITER. To examine the feasibility of Tungsten as a material for plasma facing components (PFCs) in the divertor region, EAST was equipped with W-coated divertor plates and dome for the upper divertor, the remaining plasma facing components include the plasma facing components at lower divertor still consisted of Carbon materials, at the baffles and main chamber walls consisted of Molybdenum materials. The complex PFCs environment in EAST brings great challenges to the simulations of SOL-Divertor plasmas. Developing or using tokamak modelling tools and technology on EAST is one of key issues to identify the source of impurities and their density and radiation distributions by the simulations.

SOLPS-ITER[1,2] which represents a renewed coupling of the neutral Monte-Carlo code Eirene and B2.5 edge plasma fluid code is used for the simulation of SOL-Divertor plasmas in EAST with upper Tungsten divertor and the complex PFCs environment. D^0, D^{+1}, e^- and full Tungsten, Carbon and Molybdenum impurities are included in the multi-fluid species of SOLPS-ITER. Due to the plasma-wall interaction, the Carbon, Tungsten and Molybdenum impurities are sputtered from the first wall. The modified Roth-Bogdansky formula is chosen for physical sputtering yield calculation of C, W and Mo. The Carbon, Tungsten and Molybdenum impurities are not exhausted but redeposited. Ar impurity seeding is applied to the production of detached plasmas in the simulation with the complex PFCs environment. The heating power will be increased to 10MW in next several years in EAST, so, the power flux to the computational region is set to be P=10MW which is larger than the heating power used in the present EAST experimental campaigns. The Upper Single Null configuration from the experimental shot on EAST is used in the simulations. The edge density N_{edge} at the core-SOL interface is varied extensively for surveying the divertor operation over a wide range of plasma parameters. The density and radiation distributions of W, C and Mo impurities are obtained for $N_{edge}=1\text{-}5\times10^{19}m^{-3}$ and P=10MW with and without Ar seeding.

The simulation results show, at constant heating power increasing the plasma density can lead to lower divertor temperatures and correspondingly to a strong decrease of W concentrations. The W impurity produced by sputtering cannot produce enough radiation for the detached plasmas with $N_{edge}=1\text{-}5\times10^{19}m^{-3}$ and P=10MW. Ar impurity seeding is necessary for the detached plasmas and the detached plasmas can reduce the sputtering and density of the impurities, especially the sputtering and density of W impurity. The simulations play an important role in the W impurity control in EAST with the higher heating power.

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Pressure balance in a low collisionality tokamak scrape-off layer

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Understanding the physics governing the scrape-off layer (SOL) is necessary in order to reliably predict machine and operation critical quantities, such as the heat flux width at the divertor, plasma-wall interaction, material migration, effect of divertor condition on the pedestal profile, detachment of the divertor plasma, etc. Among the most basic quantities to predict is how the density and temperature in the SOL change from an upstream location to the divertor target.

Recent simulation results [1,2] using the axisymmetric gyrokinetic code XGCa showed several noteworthy features for a low-collisionality discharge of the DIII-D tokamak, including large variations of ion density and pressure along field lines in the SOL, experimentally relevant levels of SOL parallel ion flow (Mach number~0.5), skewed ion distributions near the sheath entrance leading to subsonic flow there, and elevated sheath potentials. Comparisons of the electron pressure variation in the divertor region between simulation and experiment showed good agreement [1] (measurements were made with the divertor Thomson system). However, the simplified fluid form for *total* parallel momentum was not conserved in the near-SOL [2], which implies kinetic effects are needed to properly predict the total pressure variation in the near-SOL. Taking care to include neutral friction and viscosity resulting from a Chew-Goldberger-Low (CGL) form of the pressure tensor (i.e. only the dominant diagonal terms) does not resolve the imbalance.

In this presentation, we consider detailed balancing of the momentum in the gyrokinetic code XGCa, to confirm the code conserves momentum globally and at the particle level. We then explore reduced models including additional pressure tensor terms, to determine their effect in the momentum balance in the scrape-off layer. This is similar to the "pressure tensor unfolding" by Chankin and Stangeby [3], but utilizing the full distribution function from XGCa to calculate the presumably higher order terms of the pressure tensor. We find that certain off-diagonal ion pressure tensor terms indeed have a non-negligible parallel variation, suggesting the need to include them in the full fluid parallel momentum balance equation. We will consider each of the pressure tensor terms in the full parallel momentum equation, and explore further simulations to determine the effect of collisionality on the importance of these terms.

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Kinetic and fluid modelling of non-local parallel heat transport: impact of a localized energy source

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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. Interestingly, in presence of a strong localized energy source mimicking for example antenna heating, the proposed non-local fluid approach is able to reproduce the modification induced on the electron temperature parallel profile, which is not captured by the usual flux limiter approach.

On the other hand, we consider a kinetic approach using a 1D-1V model where collisions are taken into account both via a standard Fokker Planck collisional operator and via a Multi-Particle-Collision (MPC) algorithm. The MPC method is based on a stochastic and *local* protocol that redistributes particle velocities, while preserving the global conserved quantities such as total energy and momentum. It is very efficient with respect to computation time reduction and has been recently applied to plasma physics [1, 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 results obtained with the non-local fluid approach previously mentioned. Moreover the comparison between the two approaches is also performed considering a localized energy source and its impact on heat exchange between the ion and electron channels.

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Exploring the use of Gaussian Process Regression for interpolating SOLPS simulations

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Versions of the SOLPS code have been used to design the ITER divertor, and are being used to further understand the ITER operational limits and refine the design of diagnostics. On the down-side, full fidelity runs can take months to complete, even with parallelization, making scoping studies infeasible with the full physics version of the code.

Recent work has used a reduced physics version of the code to perform such scoping studies related to future devices, reducing the time to produce a complete run to a week for each case (without parallelization).

In this work, the use of Gaussian Process Regression to interpolate within a database of these simulations is described, with the goal of providing a further speed-up.

The main data-set explored in this work consists of nearly 15 000 simulations for an ITER geometry, but with significantly more heating power than expected for ITER (250 MW crossing into the simulated edge plasma domain rather than 100 MW), where 3 parameters were varied: the gas puffs of D (equal to that of T), N and Xe. The geometry, input power and transport coefficient profiles were kept constant across these simulations. Four quantities of interest (QoI) were used to characterize the simulations: maximum peak power flux to the outer target, power crossing the separatrix, core Zeff and the outer mid-plane separatrix electron density, chosen as they form the criteria for finding "acceptable" solutions.

Gaussian Process Regression was then applied to fit each of the QoI as a function of the input parameters across the database, and these fits were then used to calculate the QoI (together with an estimate of their uncertainty) across a 3-D grid of points spanning the input parameters.

The ("quasi-continuous") interpolated data has been used to visualize the results as a function of the input parameters, and also to identify "interesting" parts of parameter space. Where the predicted uncertainty for these identified points was too high, additional SOLPS simulations have been triggered to produce refined solutions.

Finding the Gaussian Process Regressors, and then calculating the four QoIs for 1 000 000 points (100x100x100) takes about 1 day of elapsed time, compared to the approximately 20 000 years of compute time that would be needed for the reduced model and approximately 2 million CPU years for the full model.

Implementation of a 9-point stencil in SOLPS-ITER and implications for Alcator C-Mod divertor plasma simulations

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The SOLPS code suite has been used as the workhorse for the design of the ITER divertor [1]. The latest version of the code, SOLPS-ITER [2,3], is based on an up-to-date version of EIRENE and includes the advanced fluid drifts model from SOLPS5.2.

One of the ongoing developments of the code aims at enabling the use of grids extending up to the vessel wall. In this contribution, we report on a part of this effort that involves improving the discretization of the underlying plasma equations. Presently, the plasma solver in the code, B2.5, assumes perfectly field-aligned grids. However, such grids are not compatible with the strongly shaped divertor targets in most current machines, and as foreseen on ITER. Therefore, in practice the grids are only aligned with the magnetic surfaces, but radial lines may be far from orthogonal to the field. These distortion effects are currently neglected in B2.5.

Correctly accounting for grid distortion in the simulations is especially important for fluid neutral models, because neutral transport is isotropic. Neglecting grid distortion for fluid neutrals leads to both qualitatively and quantitatively incorrect divertor solutions [4]. The situation is somewhat less problematic for plasma transport, because the transport is dominated by fast parallel convection and conduction, and grids are still aligned in this direction. However, in the cold divertor conditions found in high-recycling and detached regimes, as expected in ITER, this argument no longer holds. Radial transport can now compete with parallel transport, and it is exactly in the divertor that grids are most strongly misaligned.

In this paper, we discuss the implementation of a 9-point stencil in B2.5 to account for grid distortion. With simple distorted slab cases, we show that radial plasma particle and heat flows can differ by up to 50% compared to cases where distortion is neglected. The flows may even change sign. Finally, we apply the new solver on complete B2.5-EIRENE simulations of the C-Mod divertor plasma including drifts, for discharge 990429019. The discharge, which has detached (cold) inner divertor and outer divertor in high-recycling conditions, has already been subject of intense numerical studies using the SOLPS-ITER code [5]. Moreover, the C-Mod divertor has strong, ITER-like, shaping, and provides an ideal testbed for the new code. In previous studies, we were able to reproduce upstream and outer target profiles with SOLPS-ITER, but inner target detachment remained elusive. We analyze the modified divertor flows predicted by the 9-point stencil and study their impact on inner-outer target asymmetry.

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Impact of dynamic desorption on the edge plasma modelling of JET H-mode discharge

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Particle recycling on Plasma Facing Components (PFCs) represents the main particle source for tokamak plasmas. In steady-state conditions, the ratio between the gas throughput and the recycling flux is estimated to be around 10% in present day tokamaks and is expected to be around 1% in ITER [1]. Hence, a complete understanding of the recycling phenomenon is important to ensure a reliable plasma density control.
The recycling behaviour strongly depends on the Plasma Facing Materials (PFMs). This phenomenon was highlighted at JET with the upgrade from a carbon wall to the beryllium (Be) – tungsten (W) ITER-like Wall (ILW). JET-ILW experiments have indicated that outgassing from remote areas could play a significant role in the recovery of the pedestal density after an ELM crash [2]. The strong and spread transient heat flux due to such events leads to an increase of the PFC surface temperature, inducing an extra-outgassing of the particles preloaded during the inter-ELM phase. This phenomenon appears to be strongly related to the PFCs fuel content (different for W-coated CFC and bulk-W) and surface temperature dynamic during this phase. However, it is difficult to distinguish the impact of the different extra-outgassing zones experimentally without a dedicated modelling effort.
In the present contribution such processes are investigated with the extension of the SolEdge2D-EIRENE code, which is now coupled with a dynamic thermal desorption module (Desorption from Wall ElemEnts, D-WEE). This module is based on a Reaction-Diffusion model for the desorption part [3], and on the quadrupole method for the thermal part [4]. At the present time, D-WEE is able to deal with multiple materials (e.g. Be-W), single plasma specie (e.g. D) and actively and/or inertialy cooled PFCs. This module is used in standalone mode using output from SolEdge2D-EIRENE for a JET H-mode discharge. The poloidal outgassing distribution is obtained during and after plasma discharges. Several PFMs poloidal distributions are considered, by adapting the different activation energies of traps and diffusivity. The resulting global desorption dynamic after a plasma pulse shows a $\sim t^{-0.8}$ trend, which is in good agreement with the one obtained experimentally from residual gas balance [5]. These results demonstrate the necessity to consider the entire set of PFCs, which experienced different particle flux and surface temperature time evolution. The poloidal outgassing distribution simulated in standalone mode is used as an initial condition for self-consistent SolEdge2D-EIRENE simulations. The D-WEE module introduces a space and time dependent recycling coefficient. The impact of this coefficient is significant during the simulations of transient events like ELMs or fast plasma strike-point displacement.

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Effect of edge plasma density on hot spot in LHCD plasma

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Hot spot [1] is a serious challenge limiting long pulse operation with LHCD (lower hybrid current drive) in tokamak. Edge plasma density in front of LH (lower hybrid) antenna plays an important role in LHCD, including LHW (lower hybrid wave)-plasma coupling, hot spot in the guard limiter of LH antenna, and current drive capability [1, 2]. A critical edge density above cut-off value must be required for good coupling. Effect of edge plasma density on hot spot and current drive has been further studied in EAST.

It is observed that the temperature in the guard limiter of the LH antenna induced by an IR camera increases with line averaged density, at which the hot spot is subject to appear. Also, the edge density increases with averaged density, hence leading the increase of electron temperature due to collisional absorption in SOL. This could account for the increasing temperature with density in the guard limiter and the observation of hot spot. Further simulation indicates that such spot corresponds to the peak position of edge density and LH electric field. Studies show that the hot spot is mainly ascribed to the heat flux in front of LH antenna, and that high edge plasma density and temperature are preferred for the hot spot in the guard limiter. In addition, the divertor footprints show that strike-point splitting behaviours appear with density increase, in agreement with current profiles in the edge region and the reduction of total driven current, suggesting that more power is deposited in edge region and LHCD capability is reduced at higher edge density. As indicated by the RF spectrum measurement, this is mainly explained by the stronger parametric instability (PI) behavior in the case of higher edge density.

Above studies suggest that on the condition that the LHW-plasma coupling is satisfied, low edge density, which is related to the interaction between plasma and wall, is preferred for LHCD. It offers one possible idea how to control the edge density so as to satisfy the coupling, mitigate the heat flux in the guard limiter, and improve current drive capability for fusion device. The comparison between 2.45GHz and 4.6GHz LH waves will be also presented.

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The effects of tungsten divertor on H-mode access and detachment in EAST!

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A notable reduction of the L-H transition power threshold is observed both in JET with ITER like wall (ILW) and ASDEX Upgrade (AUG) with full tungsten (W) wall compared with the previous C walls, indicating the important role of wall material on H-mode access. The reduced impurity concentration observed in JET-ILW plasma can reduce dilution of main ions, contributing to a higher edge gradient (higher E_r) [1, 2]. In AUG a much steeper edge electron density gradient with W wall, compared to a similar discharge with C-wall, is thought to be due to a higher wall-reflection coefficient for deuterium [3]. In the experiments of JET and Alcator C-Mod, the L-H transition power was observed to increase up to twice when moving OSP from a horizontal target to a vertical target, corresponding to decreasing the triangularity [2]. Simulation analysis shows that the different scrape-off layer (SOL) conditions and geometry can cause the variation of neutral recycling patterns, thus changing the radial electric field profiles in the SOL [2]. However, a consistent reason for these observations has not yet been identified.

The upper graphite divertor of EAST was converted into an ITER-like tungsten (W) divetor in 2014 [4], thus constituting a new wall configuration with top W divertor, bottom graphite divertor and Mo first wall in main chamber. Dedicated experiments have been implemented to compare plasma performance between with upper W divertor and with lower C divertor. It is found that the direction of toroidal magnetic field (B_T) has strong impacts on the H-mode access and detachment process in divertors. The lowest L-H power threshold is achieved in upper single null (USN) configuration with the reversed B_T direction, in contrast to previous observations with double C divertors. Three radiative fluctuation bands at ~ 1 kHz, 5 kHz and 10 kHz, respectively, are observed around the X-point region prior to plasma detachment in the W divertor. With the normal B_T direction, only a wide fluctuation band with a central frequency of 5 kHz appears both in USN and LSN configurations. Increasing triangularity by moving the outer strike point closer to pumping slot can reduce the H-mode power threshold, and could cause the onset of detachment at lower main plasma density. The edge neutral behaviour should play an important role, which can be related to the asymmetry of plasma wall interaction between upper W divertor and lower C divertor. Furthermore, the geometry of W divertor could be another non-negligible element, which can facilitate high shaping plasma.

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Edge plasma measurements on the Op. 1.2 divertor plasmas at W7-X using the upgraded combined probe

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During the second operational campaign (OP1.2a) W7-X will operate with the so-called island divertor. Because of the better protection of the in-vessel components higher power and energy levels are allowed and therefore higher densities (5e19-1e20 m-3) could be reached as compared to the previous campaign with the inboard limiters [1]. In order to assess the transport and the plasma boundary properties, such as the electron temperature and density, the ion temperature, the radial electric field and plasma flow, measurements have been performed with the combined probe mounted on the multipurpose manipulator. The probe contains a set of Langmuir pins, Mach probes, compensation coils and a set of 3D pick up coils for the measurement of the local magnetic field and fluctuations, and an ion sensitive probe to record the ion temperature profiles. A tungsten sample was placed at the front facing surface of the probe in order to provide a possibility to study material deposition and to relate it to local plasma parameters.

A comparison of the measured parameters is possible with the dedicated fluctuation probe, a Mach probe array, a retarding field analyzer (RFA) and a gas puffing probe head, for fueling and impurity seeding, which also features a set of Langmuir probes for obtaining the electron temperature and density. The data from the correlation reflectometer will be used to compare the measured radial electric field, this is especially useful to identify the position and the extend of the magnetic island and the location of the last closed flux surface. Field line tracing was used to map the measured temperatures and densities and the resulting heat fluxes onto the wall elements and to the results to measurements performed with the divertor Langmuir probes and the infrared cameras observing the divertor targets.

Measurements from the ongoing campaign report electron temperatures of up 100 eV and densities of up to 10^{19} m^{-3} in the edge. It has been observed, in addition to differences due to the magnetic configurations, that the magnetic topology is sensitive to the plasma beta. During the experiments the 5/5 island structure has been identified with the measured edge plasma profiles and compared with the predictions from the field line tracing code [2]. Those electron densities and temperatures have been used as an input for the EMC3-EIRENE modeling to improve the understanding of the plasma edge transport.

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Effect of turbulent fluctuations on neutral particles transport with the TOKAM3X-EIRENE turbulence code

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The scrape-off layer (SOL) of tokamaks shows turbulent fluctuations, with fluctuation levels reaching order unity. On time scales much larger than the turbulence correlation time (typically ~10μs), convection by the fluctuating velocity field results in cross-field transport, modelled by an anomalous diffusion term in transport codes such as SOLPS, Soledge2d-EIRENE. In fact, the latter codes solve mean field equations (i.e. time averaged fluid equations). A proper derivation of these equations shows that in addition to the gradient diffusion closure made on turbulent fluxes, a number of other non-linear terms are approximated in mean field codes. This is in particular the case for volumetric sources/sinks related to exchanges between neutrals and the plasma, as well as sputtered fluxes at the wall. Underlying fluctuations might thus affect the penetration depth of neutral particles into the plasma, as well as impurity production. The errors induced by these approximations in the calculation of averages have been investigated both for neutrals and sputtered fluxes, using a combination of stochastic models and 2D turbulence codes to model the statistical properties of fluctuations. Both ionization and sputtering are threshold processes, so that temperature fluctuations enhance theses processes at temperatures close or below the threshold. Moreover, density fluctuations also play a role in the transport of neutral particles, reducing the opacity of plasma to the neutrals.

In this work, we apply these results to the analysis of global TOKAM3X-EIRENE 3D simulations in X-point geometry. TOKAM3X is a 3D turbulence code, solving balance equation for particles, momentum and ion/electron energy density in the plasma edge. The code is coupled to the Monte Carlo solver EIRENE, using the same interface as in Soledge2d-EIRENE. The statistical properties of turbulent fields relevant to this work are discussed, both in the far SOL and for the first time in the divertor, including recycling. This includes the Probability Distribution Functions (PDF) or the various fields involved (n_e, T_e for ionization, n_e, ϕ, T_i for sputtering) as well as correlations between these fields. The neutral particle transport and sputtered fluxes are recalculated on the mean field plasma, and compared to the mean neutral particle density/flows obtained from the turbulent simulation, so as to assess the effects of the fluctuations, in particular on the ionization balance in the divertor. The latter effects become more and more pronounced as the high recycling regime is approached, in particular because the plasma temperature is low enough so that ionization and sputtering are strongly non-linear, but not only: the ratio of the neutral particle mean free path to the size of turbulent structures also plays a key role. Fluctuation dressed ionization rate coefficients and sputtering yields to be used in mean field codes are derived, and their parametrization in terms of the mean fields is discussed.

Width of scrape-off-layers in circular & diverted plasmas: a turbulent model confronted to experimental evidences

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Control of power loads on plasma facing components in current and future tokamak reactors largely dependents on perpendicular transport properties setting the scrape of layer width. This contribution aims at showing that interchange turbulence is a likely candidate, first in inner limited geometry, but also in diverted one. After confronting experimental evidences and simulation results, implications for future reactors are discussed.

For limited circular geometries, a 2D interchange transport model is validated against a broad set of fluctuation properties and mean density and heat load profiles collected in Tore Supra [1]. In particular (1) the ExB drift velocity of plasma filaments (blobs) are reproduced by an isolated blob model, and (2) values of the SOL width are matched by a regression scaling - on global control parameters - constructed from numerical simulations. Both agreements are within 30% error. The model scaling for λ_q depends mainly on the poloidal magnetic field strength, including weaker but finite sensitivity with machine size and total magnetic field strength. Predictions for ITER start-up phases reproduce recent extrapolations based on multi-machine regressions [2], let aside the possible existence of a narrow feature.

Extension of the model to diverted geometry has to suffer the evidence that λ_q is generally much smaller in this configuration than for circular inner limited plasmas, which cannot be explained by the 2D model in its state. On the other hand: (1) Experimental scaling laws constructed on JET and AUG L-mode lower single null data [3] return a parametric sensitivity of λ_q in good agreement with the model. (2) Recent TCV data [4] confirm this agreement. It points toward the existence of a positive sensitivity of λ_q with machine major radius. (3) Recent 3D flux driven turbulent simulations of the edge of both circular and diverted plasmas, made with TOKAM3X, show that interchange turbulence dominates transport in both cases and λ_q is much narrower (1:6) in diverted than inner limited configurations, ratio similar to experiments. Ongoing works focus on the role of magnetic shear & expansion effects on transport mitigation.

Implications are twofold. First, estimates of λ_q for ITER, either based on multi-machines extrapolations or drift-based heuristic approaches, could be too small by a factor of at least 2. Second, interchange turbulence is also likely to take place in the divertor volume. Simulations with TOKAM3X on equilibria with increasing length of the outer divertor leg - in TCV-like geometry - show that λ_q increases with the leg length whereas divertor spreading S does not [5]. These results are in good agreement with recent experimental evidences from TCV [4,5]. Besides questioning the physical interpretation of λ_q & S, it opens new perspectives in the optimization of turbulent transport in alternative divertor configurations.

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Recent progress in implementing *ExB* drift in EMC3-Eirene

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As an essential step on the way towards a self-consistent treatment of *ExB* drifts in EMC3-Eirene, a potential equation in the form of current conservation has been implemented. It is solved in general 3D SOL geometries where flux surfaces may not exist. Parallel currents along field lines obey the electron momentum balance and Bohm boundary conditions at targets. Perpendicular currents are driven by a finite anomalous cross-field conductivity, which is related to the particle diffusivity via Einstein's expression and can be modified by a user-defined constant. The potential equation is solved in two steps, following the basic idea of correction methods well-known in the literature. In the first step, the electron momentum equation is solved with zero perpendicular currents using the standard Euler method. In the second step, a potential correction due to finite cross-field currents is obtained by applying the same Monte-Carlo (MC) method as used for the other fluid equations in the EMC3 code. This hybrid Euler-MC technique is first benchmarked against a standard finite-difference method in a slab 2D geometry, and then applied to realistic 3D cases of W7-X. The paper will show details of this numerical concept and present the benchmark and first application results.

Flux-driven turbulence GDB simulations of the IWL Alcator C- Mod L-mode edge compared with experiment and stochastic model[1]

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Prior to predicting confinement regime transitions in tokamaks one may need an accurate description of L-mode profiles and turbulence properties. These features determine the heat-flux width upon which wall integrity depends, a topic of major interest for research aid to ITER. To this end our work uses the Global Drift Ballooning (GDB) model [1] to simulate the Alcator C-Mod edge and contributes support for its use in studying critical edge phenomena in current and future tokamaks. We carried out 3D electromagnetic flux-driven two-fluid turbulence simulations of inner wall limited (IWL) C-Mod shots spanning closed and open flux surfaces [2]. These simulations are compared with gas puff imaging (GPI) and mirror Langmuir probe (MLP) data, as well as the stochastic fluctuation model [3], examining global features and statistical properties of turbulent dynamics. GDB reproduces important qualitative aspects of the C-Mod edge regarding global density and temperature profiles, within reasonable margins, and though the turbulence statistics of the simulated turbulence follow similar quantitative trends questions remain about the code's difficulty in exactly predicting quantities like the autocorrelation time. A proposed breakpoint in the near SOL pressure and the posited separation between drift and ballooning dynamics it represents are examined.

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Advancements in 3D neutral gas and edge plasma modeling of resonant magnetic perturbations in ITER and their implications for fueling and exhaust

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The control of steady state and transient heat loads remains a key challenge for the next-step fusion device ITER in order for them to not exceed material limits for plasma-facing components. Transient events at the plasma edge - so called edge localized modes (ELMs) - are intrinsic to standard high confinement (H-mode) operation, but they may be mitigated or even suppressed following the application of resonant magnetic perturbations (RMPs) as demonstrated in present day machines. Steady state divertor operation, on the other hand, is affected by RMPs in terms of non-axisymmetric heat and particle loads, as well as reduction of core density (particle pump-out) which in turn can affect the onset threshold of detachment.

The development of the ITER divertor has been largely guided by two-dimensional (axisymmetric) modeling using the SOLPS code, yet in order to analyze the impact of symmetry breaking RMPs a three-dimensional model such as EMC3-EIRENE is required. We first present a benchmark of the two models for an axisymmetric case targeting the initial non-active phase in ITER with an L-mode, gas fueled $1.8\,\mathrm{T}/5\,\mathrm{MA}$ hydrogen plasma at $q_{95} = 3$ and using the SOLPS-ITER code version. In particular, we focus on cold divertor conditions and perform a density scan in which the divertor is pushed into roll-over, since numerical access to these conditions remains challenging for EMC3-EIRENE. To allow for a realistic particle balance in the new EMC3-EIRENE simulations we have implemented the volume under the divertor dome to match the machine pumping configuration and its representation in SOLPS. Furthermore, additional neutral gas processes such as elastic collisions with ions or other neutral particles have been switched on in EMC3-EIRENE. The 3D simulations with the improved neutral gas model are then performed for RMP application to a low input power ($\sim 30\,\mathrm{MW}$) H-mode discharge expected to be typical of what is currently planned for non-active operation on ITER. The clear distinction between confined and scrape-off layer (SOL) regions becomes diluted once RMPs are applied, resulting in field lines that connect the edge of the nominal confined region to the divertor targets. The degree of this field line loss depends on both the perturbation strength and plasma response effects. The latter are taken into account by coupling magnetic field calculations from the MARS-F linearized, resistive single fluid MHD model to EMC3-EIRENE. Results from a high and low plasma rotation case will be compared to the vacuum approach for the RMP field, and implications for the fueling efficiency and exhaust analyzed, resolving contributions from recycling and gas puffing.

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Abstract withdrawn

Divertor heat flux broadening induced by edge coherent mode in EAST

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Recent study on multi-machine database of inter-ELM divertor heat flux indicates that the midplane Scrape-off Layer power fall-off length for ITER is expected to be very narrow [1], ~1mm, which will induce small plasma-wetted area and high peak heat flux. In EAST, the edge coherent mode (ECM) near the electron diamagnetic frequency (20–90 kHz) is observed in the long pulse H-mode plasmas [2]. The integral heat flux width (λ_{int}) with ECM is comparable with the λ_{int} in L mode. This ECM is associated with a significant broadening of the integral heat flux width (λ_{int}) by up to 100%. The power decay length (λ_q) mapping results with ECM is much larger than the multi-machine experimental scaling results [1].
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A new high-order fluid solver for tokamak edge plasma transport simulations based on a magnetic-field independent discretization

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Power exhaust control in tokamaks, and its implications for steady-state and transient heat loads on divertor and limiter PFCs, is still an open issue. In transient situations in particular, such as during start-up or control operations, the evolution of particles and heat fluxes is little known, although being critical for the safety of the machine.
The heat load is largely determined by the physics of the Scrape-Off Layer (SOL), and therefore it depends on a large extent on the geometry of the magnetic surfaces as well as on the geometry of wall components. A better characterization of the heat exhaust mechanisms requires therefore improving the capabilities of the transport codes in terms of geometrical description of the wall components and in terms of the description of the magnetic geometry. For transient simulations, it also becomes crucial to be able to deal with non-stationary magnetic configurations. In particular, avoiding expensive re-meshing of the computational domain is mandatory.
As an attempt to achieve these goals we propose a new fluid solver based on a high-order hybrid discontinuous Galerkin (HDG) finite element method based on unstructured meshes. We propose to study the edge plasma transport during slowly varying magnetic equilibria, under the electrostatic assumption. We show how the particle fluxes at the wall vary in our model when evolving the magnetic equilibrium in time, for example, during the equilibrium construction skipping from a limiter configuration to a diverted one at the beginning of the operation.

Evolution of divertor plasmas in EAST with lithium aerosol injection[1]

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The Experimental Advanced Superconducting Tokamak (EAST) can achieve long-pulse, H-mode discharges[1] with ELMs eliminated by the use of real-time lithium aerosol injection[2]. Here we present results from upper single-null (USN) discharges, i.e. using the ITER-like W mono-block divertor, with low-field side lithium aerosol injection near the outer, upper strike point that eliminated ELMs. Divertor electron density, n_e and temperature, T_e profiles are measured by two Langmuir probe arrays installed in the upper divertor of EAST separated by 112.5 degrees toroidally[3]. Measurements from both probe arrays are combined and averaged in 100ms windows to create composite divertor profiles of n_e, T_e, and particle flux. A shot-to-shot conditioning effect[4,5] is observed where once lithium aerosol has been injected, discharges show a reduction in the peak divertor n_e at both the outer and inner strike points of 53% and 30% respectively as compared to an ELMy H-mode reference discharge. Upstream electron density is maintained at a constant 2.8 x 10^{19} m^{-3} during all discharges using active feedback control. T_e is unchanged both during the addition of lithium aerosol and shows no shot-to-shot variations beyond normal experimental variability. Ohmic cleaning discharges are shown to revert the discharges to ELMy H-mode behavior with increased divertor density.

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Impact of molecular deuterium on the particle and power balance in DIII-D detached divertor plasmas

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The atomic and molecular fluxes of deuterium off the target plates in DIII-D detached, low- (L-mode) and high-confinement (H-mode) plasmas are measured and compared to predictions from the edge fluid code EDGE2D-EIRENE. Previous measurements of these fluxes in DIII-D [1] and other devices, e.g., [2] in detached conditions showed that a significant number of deuterium molecules exist in the divertor scrape-off layer (SOL), dominating the particle, momentum and power balances at the target plates, and thus impacting the onset of detachment. Molecular processes also contribute to the total deuterium line emission and may contribute to the previously inferred radiation deficit in fluid edge code predictions [3].

Measurements of the line emission of the Lyman line series with the recently installed divertor Extreme Ultraviolet (EUV) [4] spectrometer (30-160 nm) will be used to validate the EDGE2D-EIRENE predicted line intensities, and to determine the particle and power balance across the low-field side (LFS) divertor leg, including recombination and Ly-α photon trapping, and deuterium and carbon radiation. The relevance of the Lyman-Werner molecular bands (120-170 nm) will be assessed with the same spectrometer. The atomic and molecular influxes were inferred spectroscopically via the Balmer line series and the Fulcher molecular bands (600-640 nm), and their spatial distribution imaged with cameras. Measurements of the electron temperature (T_e) and density (n_e) using the 2D divertor Thomson system show sub-eV T_e directly adjacent to the LFS plate, and the formation of a high n_e front within the LFS divertor plasma, facilitating strong plasma-molecule interaction in front of the target plate.

EDGE2D-EIRENE simulations predict the number density of molecules to exceed the number density of atoms by a factor of 5 in the vicinity of the divertor strike points for T_e of 1 eV, or below. However, while the simulations predict peak electron densities of $6x10^{19}$ m^{-3} and $5x10^{20}$ m^{-3} for L-mode and H-mode, respectively, consistent with the measurements, they do not indicate the formation of an electron density front away from the outer plate.

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The I-mode confinement regime on the ASDEX Upgrade tokamak: scrape-off layer properties and investigation of stationary and transient divertor heat loads

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The I-mode is an improved confinement regime of tokamak plasmas where an edge transport barrier is observed only in the heat transport but not in the particle transport. This is in contrast to H-mode confinement, which is characterized by transport barriers for both heat and particles. The I-mode does not exhibit any ELMs. Since the particle confinement is low, the I-mode does not suffer from high impurity content and no impurity accumulation is observed. It is characterized by an instability called the weakly coherent mode (WCM) which resides at the plasma edge. After substantial I-mode research by the fusion community in the last years, the mechanism which creates a transport barrier in only one of the transport channels is still not understood.

An overview of recent I-mode studies on ASDEX Upgrade is given, including L-I and I-H power thresholds, pedestal and confinement properties [1], extending previous studies to higher Greenwald fractions up to 0.7. The confinement improvement in I-mode is accompanied by a deepening of the edge radial electric field well and a reduction of turbulence with respect to L-mode [1, 2]. A striking feature of I-mode edge turbulence is a reduction of low-amplitude density fluctuations, concomitant with the appearance of strongly intermittent high-amplitude density bursts in the plasma edge inside the separatrix [3]. These density turbulence bursts are linked to the WCM. After their generation, they are expelled from the plasma and appear later in the divertor, observed by bolometry, infrared thermography and Langmuir probes. The importance of these bursts in terms of transient divertor heat loads is assessed.

Moreover, stationary I-modes have been obtained recently with neutral beam injection heating. The stationarity allows the characterization of scrape-off layer (SOL) fall-off lengths of density and temperature. While the former are similar to L-mode plasmas, the latter are comparable to H-mode plasmas, indicating that I-mode properties are also found in the SOL. Infrared thermography data yields information on the scrape-off layer power fall-off length and divertor loads, and implications for future devices are discussed.

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Impurity Ion Transport by Filamentary Plasma Structures

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In recent experiments of magnetic confinement devices, the intermittent filamentary coherent structures called “blob” or “hole” have been observed in the boundary layer plasmas. Such structures are thought to provide the radial non-diffusive transport in the boundary layer plasmas. Motivated by such experiments, many theoretical and numerical studies regarding the filament dynamics have been made on the basis of two-dimensional reduced fluid models [1, 2]. However, the size of such structures on the cross-section is thought to be in meso-scale in many situations. In such situations, the kinetic dynamics should be considered. Therefore, we have developed the three-dimensional (3D) electrostatic particle-in-cell (PIC) simulation code [3-5] and studied the kinetic dynamics on the filament phenomena with the 3D-PIC code. We have demonstrated the self-consistent current system and the temperature structure in a blob with 3D-PIC simulation [6]. Furthermore, the 3D-PIC code has shown that the dipolar profile of impurity ion density in a filament on the cross-section arises from the polarization drift and that a filament transports impurity ions [7]. The observed effective radial diffusivities of the impurity ion transported by a filament are comparable to the Bohm diffusion coefficient, $D_B = T_e / 16(eB)$. The impurity ion transport by a filament has been also studied by the two-dimensional interchange turbulence fluid simulation with the test impurity ion particle model [8, 9]. Since the direction of blob / hole propagation in helical devices is opposite to that in the low-field side of tokamak devices, the impurity ion transport by filaments might be able to explain the difference of impurity transport property between tokamak and helical devices. In this paper, we have investigated the dependence of the impurity ion transport by a filament on some parameters with the 3D-PIC simulation and found the correlation between the effective diffusivity and the filament size. Using such correlations, we will estimate the effect of filaments on the total transport of impurity ions.

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Fluctuation behaviour associated with the different phases of the ELM cycle in ASDEX Upgrade.

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High performance regimes (H-mode) are reached through the spontaneous formation of a transport barrier resulting from the shear-flow suppression of turbulence. The steep density and temperature gradients that form at the edge are subject to instabilities such as edge localized modes (ELMs), provoking repetitive collapses of the edge pressure profile. Between ELMs, temperature and density profiles build up again on different time scales [1], raising the question of which transport mechanisms determine the profile evolution until the next ELM. Intermittent density fluctuations persist in the inter-ELM phase, although the overall fluctuation level is strongly reduced as expected in the pedestal region. They appear in Doppler back-scattering signals as bursts at different levels, different time dynamics and statistical properties with respect to the L-mode, with significant non Gaussian Probability Density Functions. The intermittency and time dynamics of these density fluctuations evolve along the different phases of the ELM cycle that were identified in ASDEX Upgrade in the recovery of density and temperature profiles and in the divertor regime [2]. Inside the separatrix, the fluctuations change from irregular fast fluctuations to filament-like bursts that can be grouped in regular trains, with a repetition rate which evolves similarly to some of the mode frequencies observed in magnetics or ECE fluctuations along the ELM cycle [1,3]. In addition, strong bursts, filament-like or solitary structures, are observed in the late ELM cycle phase, often linked to radiation bursts later in the SOL and in the divertor Langmuir probes, indicating a potential role of these fluctuations in profile clamping between ELMs.

When the density is increased, typically when the H-mode degrades and the ELM regime changes from pure type I ELM to mixed type I and type II (or small ELMs) [4], fluctuation bursts at a high repetition rate become dominant, with an increased level. The PDF keeps a strongly intermittent character in the confined plasma while the PDF in the SOL becomes L-mode like. The role of these fluctuations in constraining the pressure gradient to avoid large ELMs will be discussed.

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Evaluating the impact of molecules on DIII-D divertor target heat flux densities using UEDGE*

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The impact of molecules on the divertor plasma conditions in DIII-D low-confinement (L-mode) plasmas is investigated in simulations with the multi-fluid code UEDGE, and compared to divertor measurements of target heat flux, radiated power and divertor plasma densities and temperatures. Previous predictive UEDGE simulations of plasmas in the Fusion Nuclear Science Facility (FNSF) showed that inclusion of molecules in the model reduces the peak plasma heat flux densities by about 20% in the main scrape-off layer (SOL) compared to simulations considering neutral atoms only [1]. The onset of divertor detachment, which significantly reduces peak plasma heat flux densities to the targets, is experimentally observed at lower separatrix densities than predicted by simulations, which is considered to be, in part, explained by molecular effects [2]. Furthermore, as elastic collisions with ions and molecules can transfer significant power to the molecules, a significant fraction of the heat flux to the divertor targets is carried by molecules in detached conditions.

The present molecular model in UEDGE considers neutral molecules as a diffusive fluid species. The continuity equation considers recycling of molecules at the vessel walls and targets as the sole molecular source, where the dissociation energy is extracted from the electron temperature equation. The dissociation of molecules into neutral atoms at a dissociation rate different from the ionization rate of the atoms is the sole molecular sink, where the Franck-Condon energy is returned through the ion temperature equation. The molecular motion is taken from the momentum equation where the molecular pressure is balanced by elastic scattering off atoms and ions, and the molecular temperature distribution is user-specified [1]. In this contribution, a temperature equation for the fluid molecules is introduced into the UEDGE molecular model to self-consistently evaluate molecular target heat flux densities.

The revised UEDGE model for molecules is evaluated against measurements of the target heat flux, the total radiation and line emission from deuterium, and the plasma temperature and density in unseeded DIII-D L-mode plasmas. The relative impact of molecules is contrasted against other physics models, such as cross-field drifts, and transport and radiation from carbon. The UEDGE predictions are compared to calculations with the multi-fluid edge code EDGE2D coupled to the kinetic Monte-Carlo neutral code EIRENE to elucidate differences due to fluid versus kinetic neutral models.

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Infra-red observations of ELM loading on toroidal gap edges of tungsten castellated blocks in the KSTAR divertor

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The issue of heat loading on the poloidal gap edges of castellated divertor structures has been intensively studied through coordinated, multi-device efforts within the International Tokamak Physics Activity [1]. There it was concluded that, at least for inter-ELM power loads, the deposition can be well described by an optical approximation. As a result, toroidal bevelling is being applied to the ITER divertor tungsten (W) monoblock top surfaces to hide these edges. However, toroidal gap (TG) loading remains problematic, particularly in the case of ELM loading, in which 3D ion orbit modelling [2] predicts that energetic ELM ions can access TG edges and can lead to edge melting even under mitigated ELM loads, far below those required to melt the top surface. To date, however, this predicted transient loading effect has not been observed experimentally in current devices. In this paper, we report indirect observations of ELM-induced TG loading on W castellated blocks installed in a special central divertor target tile [3] on the KSTAR tokamak. This region of the divertor is used as the outer strike point region for certain KSTAR discharges and, geometrically is equivalent to the ITER inner vertical target where electrons and ions will strike the upper and lower TG edges respectively.

The dedicated tile carries a matrix of 3×7 (toroidal, poloidal) castellated, brazed W blocks (W-Cu-CuCrZr) of 12 mm × 30 mm. The 7 rows of blocks are separated by TGs having widths of between 0.5 and 2.0 mm. All blocks were intentionally aligned within ±0.1 mm except for 3 blocks in the central poloidal column which were radially recessed -0.5 mm with respect to their neighbours. The choice of radial recess and TG width were guided by simulations performed using the tools developed in [2] to provide varying degrees of theoretical gap loading. Depending on the ELM parameters, this in turn is expected to yield various values of the ratio $\Delta T_{edge}/\Delta T_{center}$, with ΔT_{edge} and ΔT_{center} the ELM-induced temperature increases at the TG edge and block centre respectively. For typical KSTAR ELMing H-mode operation (B_T = 2.5 T, I_p = 700 kA, $T_{i,ped}$ = 400 eV), the ratio $\Delta T_{edge}/\Delta T_{center}$ can vary between factor 2-3. An IRTV system viewing the special tile finds values $\Delta T_{edge}/\Delta T_{center}$ = 1.5-3.5, but since the camera framerate was insufficient to resolve the ELM-induced temperature rise, the measurement is only indirect. Post-mortem SEM images of the TG edges indicate that the lower TG edges are more roughened than at the upper edge, suggestive of stronger loading at the location of ion bombardment. An additional outcome of the exposure of the special tile was the observation of severe melt damage on the poloidal edges of the first column of W blocks, caused by an error in alignment of the graphite frame surrounding the assembly with respect to the neighbouring tile. Despite the extreme damage, no effect on the KSTAR plasmas was observed throughout the campaign. The melt damage itself is largely in the direction expected as a result of j×B forces on the molten W.

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Hybrid fluid-kinetic neutral model for a 2D detached ITER case

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Most current plasma edge simulations are performed with a Monte Carlo (MC) simulation of the kinetic equation for the neutral particles with codes as, e.g., EIRENE [1]. This statistical approach poses computational problems: it hampers the convergence assessment of the coupled plasma-neutral equations and leads to exacerbated run times for high-collisional cases. For ITER and DEMO, these high-collisional (partially) detached scenarios are of crucial importance to reduce the particle and heat loads on the divertor targets. Simultaneously, in high-collisional regimes, one is closer to the fluid limit for the neutrals, which can be simulated much more efficiently. However, the fluid limit is only reached in some specific regions that are not necessarily known before starting the simulation. Therefore, we propose to use a hybrid model that combines the fluid and kinetic descriptions.

The hybrid method that we propose is based on a micro-macro decomposition of the kinetic equation, as explored, e.g., in Ref. [2] for the study of Landau damping in a plasma. The micro-macro decomposition implies that the total neutral velocity distribution is split up in a fluid and kinetic part. Then, the MC part is only used to compute a kinetic correction on the fluid solution, instead of solving the kinetic equation for the whole distribution. The resulting kinetic correction appears in the closure terms of the fluid moment equations. The choice of the fluid distribution leads to a particular fluid model. This method automatically determines the magnitude of the kinetic correction and avoids the complicated process for the localization of the kinetic and fluid regions. In addition, this micro-macro hybrid model is completely equivalent to the kinetic equation without any assumptions, at least in continuous form.

We have already explored this hybrid approach for a simplified 1D model that focuses on the poloidal particle motion [3]. There, we simulated a single flux tube in the divertor leg with a fixed background plasma, which is characteristic for an ITER partially detached case. Compared to an MC simulation of the full kinetic equation for the same computational time, the hybrid approach was shown to lead to a reduction of the statistical error on the particle and momentum source with about a factor five, and on the ion energy source with a factor three. In this contribution, we extend the hybrid model to the 2D plasma edge. Due to the additional toroidal particle motion, it is essential to include a parallel momentum equation in the fluid model. We add kinetic corrections on top of the fluid models from Ref. [4] and we assess the statistical error reduction for a given computational time, again for an ITER detached case.

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Extension of the I-mode confinement regime to 8 tesla on Alcator C-Mod*

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A crucial need for fusion devices is to achieve sufficiently high energy confinement, without the large transient heat loads typical of H-mode with Type I ELMs. The I-mode regime has a strong energy transport barrier, without a particle barrier, and is naturally free of ELMs. Particle confinement is at the L-mode level, which also avoids accumulation of impurities that may arise from plasma-surface interactions. Key questions remain for extrapolation of I-mode to reactors, notably the conditions needed to access the regime, and to maintain it without transitions to H-mode or L-mode. Experiments on the Alcator C-Mod tokamak have extended I-mode to the full range of field and current, B_T=2.8-8 T and I_p= 0.55-1.7 MA, and show that the regime becomes more robust against transitions to H-mode at higher fields. This is due to weak dependence of the L-I power threshold on B_T, while the upper range of power while staying in I-mode increases more strongly with field [1]. At 8 T, no discharges transitioned to H-mode even at maximum-available ICRH power (5 MW). I-mode discharges feature strong temperature pedestals, with T_e and T_i over 1 keV, and L-mode density pedestals. An E_r well, intermediate between L-mode and H-mode discharges, also develops. Several interrelated features in pedestal turbulence and flows are observed, including a low frequency GAM (10's of kHz), a high frequency 'weakly coherent mode' (few hundred kHz), and reduction in turbulence between these ranges. Some of these fluctuations may extend into the SOL.

The different characteristics of I-mode vs H-mode pose new questions and challenges for boundary solutions. While the elimination of transient heat loads due to ELMs is a major advantage, the stationary heat flux on the outer strike point can remain high, similar to L-modes and H-modes, with extremely narrow λ_q (<1 mm) at high poloidal field [2]. Because I-mode is usually accessed with ion $B\times\nabla B$ drift away from the active X-point, reversing ExB SOL flows, heat fluxes can also be high on the inner strike point [3]. Total power to the dominant divertor target has been significantly reduced by operating in near balanced configurations, with $|\delta r_{sep}|$ smaller than ~1.5 mm [1], but reducing the instantaneous *peak* heat-flux on any one target remains a challenge since the heat-flux widths are so small [3]. Combining I-mode with a detached divertor also remains an outstanding challenge [4]. Prospects for and needed research on the I-mode regime will be summarized.

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Determination of tungsten sources in JET-ILW divertor by spectroscopic imaging in the presence of strong plasma continuum

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ITER will feature a full tungsten (W) divertor and a beryllium wall cladding. The tungsten concentration in the core plasma has to be controlled and kept below $\approx 3 \times 10^{-5}$ to avoid large central radiation losses. Therefore, it is very important to get a complete understanding of the critical parameters for the erosion of tungsten components, particularly in the divertor. Especially, the knowledge of the W-erosion distribution in the divertor, which can be provided by video imaging spectroscopy, is important. However, the relative weak W I emission lines are often masked by the presence of the plasma continuum radiation (free-free, free-bound) and thermal radiation from the hot surfaces. For an attached divertor plasma, the bremsstrahlung emission starts competing with W emission at high densities and low electron temperatures as well as the high Z_{eff} levels. For n_e= 7×10^{19} m^{-3}, T_e= 5 eV and Z_{eff}= 1.6 the bremsstrahlung emission intensity at λ= 400 nm calculated using ADAS is about 9×10^{16} ph s^{-1} m^{-2} sr^{-1} nm^{-1} which is comparable to the measured W I emission at λ= 400.9 nm (~2×10^{17} ph s^{-1} m^{-2} sr^{-1} nm^{-1}). The recombination (free-bound) emission disturbs the W I line emission measurements in the completely detached divertor (n_e= 10^{20} m^{-3}, T_e= 1 eV). Under cold detached divertor with significant reduction of W sputtering, the tungsten is introduced into the plasma only during the ELMs.

The identification of the tungsten atom sources and the measurement of their radiation distribution in front of all plasma-facing components has been performed in JET with the help of an endoscope equipped with four digital CCD cameras (one megapixel, max. 33 fps at full resolution and 16 bits data output), each combined with filter wheels for narrow band interference and neutral density filters. This diagnostic system provides the same two-dimensional view simultaneously for different spectral lines. The scope of this contribution is to provide a clean image of the single spectral line of tungsten by using two camera images with interference filters of different bandwidths centred on the W I (400.9 nm) emission line. A new algorithm for the subtraction of the continuum radiation was successfully developed and is now used to evaluate the W erosion even in the inner divertor region where the strong recombination emission is dominating over the tungsten emission.

The detailed description of the algorithm will be demonstrated. Additionally, analysis of W sputtering and W redistribution in the divertor by video imaging spectroscopy with high spatial resolution will be presented.

[**] See author list in the paper, X. Litaudon et al., Nucl. Fusion, 57 (2017) 102001.

Impact of 3-D magnetic field topology on divertor heat flux under ITER-like RMP configurations

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The in-vessel, 3-row resonant magnetic perturbation (RMP) coils in KSTAR, just like in the planned ITER-RMP configurations, provides an ideal testbed to study the impact of 3-D magnetic field topology in ITER. Indeed, in support of ITER, KSTAR has been conducting a series of 3-D physics experiments whose RMP configurations might not be easily realized in other devices equipped with 2-row RMPs. Specifically, a variety of 'intentionally misaligned' RMP configurations have been systematically studied, while their compatibility with detached plasmas has been also explored. Taking full advantage of robustly reproducible full ELM suppression with either n=1 or n=2 RMPs, we have succeeded in measuring divertor heat flux striation patterns using static and rotating RMPs, whenever feasible. Unlike successfully diffused divertor heat flux during RMP-ELM mitigation [1], preliminary experimental analysis suggests that divertor heat fluxes were rarely dispersed by the previously effective misaligned configuration, once RMP-driven ELM suppression becomes dominant. This may be consistent with the classification that RMP-driven, ELM suppression would belong to a bifurcated state, while the ELM mitigation would be primarily governed by the linear plasma response to RMP [2]. However, another type of misaligned configuration, whose plasma response is dominantly enhanced by kink influence, appears quite promising in effectively lowering the peak of divertor heat flux, as well as in broadening the non-axisymmetric lobes. We will report the detailed analysis results of each RMP configuration impact, as well as discuss key physics parameters that might have contributed to ELM suppression in lieu of ELM mitigation.

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Dependence of the upstream separatrix temperature on impurity seeding and separatrix density in EDGE2D-EIRENE simulations for JET H-mode plasmas

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EDGE2D-EIRENE simulations of upstream density and radiative power scans in JET H-mode type plasmas in vertical target predict a *linear* relationship between the electron and ion temperatures, and power across the separatrix. This is in contradiction to the two-point model equations which states that the upstream temperature scales at the power crossing the separatrix to the power 2/7 - equation (4.87) in reference [1]. The upstream separatrix temperature – here defined as the low-field side mid-plane – is a critical boundary condition for pedestal stability analysis [2]. Using the two-point model 2/7 scaling as a physics basis, the JET separatrix temperature is routinely predicted at 100 eV. Semi-analytic models for the power decay width (λ_q) prediction also rely on the same two-point temperature equation [3]. Therefore, understanding the validity of temperature equation under varying plasma conditions is paramount and the focus of this work.

The multi-fluid code EDGE2D-EIRENE was used to simulate the upstream temperature for varying upstream density and radiative power from impurity seeding (nitrogen and neon). The simulated upstream temperature disagrees with two-point model temperature equation by approximately a factor of 1.6 – 2.0 in the ion temperature and by a factor of 1.1 – 2.0 in the electron temperature. The disagreement is strongly impacted when seeding impurities. The ratio of T_i/T_e remained mostly unchanged in our simulations unless the heat flux limiters were decreased from their current values.

To be consistent with experimental analysis we calculated λ_q from our simulated data using the method in [4]. The two-point model temperature equation is dependent λ_q, hence an upstream temperature can be calculated. Comparing this calculation to the upstream temperature from EDGE2D, the calculation overestimates by 10-50%. The range of disagreement is dependent on the upstream separatrix electron density, the seeding type and the radiated power of the seeded impurity.

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Abstract Withdrawn

Role of the divertor neutral pressure on power exhaust and operational limits in ASDEX Upgrade

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The divertor neutral pressure, p_0, has been found to be an important parameter for inter-ELM power exhaust as well as H-mode performance in ASDEX Upgrade. p_0 exhibits a close connection to the upstream separatrix density, $n_{e,sep}$, which in turn determines the H-L density limit [1]. For a closed divertor configuration and active pumping, the neutral pressure can be regarded as an engineering parameter, since it is largely determined by the gas puff rate. Experimentally found relationships are $n_{e,sep} \propto p_0^{0.31}$ [2], $n_{e,sep\ H\text{-}L} \approx 0.5\ n_{Greenwald}$ [1] and $P_{sep,detach} \propto p_0$ [3], where $P_{sep,detach}$ is the maximum separatrix power where detachment is still obtained for a given p_0. For impurity seeded cases, p_0 is a weighted neutral pressure sum of deuterium and seed impurities. These relationships allow the integration of H-L and power dissipation limits into an operational boundary for the neutral divertor pressure. Analytical models are used to extrapolate ASDEX Upgrade experimental results to future devices like ITER and DEMO. An important parameter is the exponent of the neutral pressure in relation to the separatrix density. The weaker experimental $n_{e,sep}$ dependence on the neutral pressure, namely $p_0^{0.31}$ versus the analytical high recycling dependence, $p_0^{0.5}$, allows thus for higher p_0 and hence results in favorable exhaust conditions. Possible reasons for the observed dependence and implications for divertor operation will be discussed.

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Investigation of supersonic plasma flow in DiPS-2 by a laser induced fluorescence system

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Flow measurements near X-points including E × B shear velocity and supersonic flow are still under debate [1, 2] in fusion devices, the generation and measurements of supersonic plasma flow at pulsed plasma system to be applied for the analysis of transient phenomena such as ELMs.

In this experiments, a concept of ion extraction system [3] has been adopted to generate supersonic plasma flows ($M_\infty > 1$) at weakly magnetized plasma in steady state condition. A cylindrical ion extraction electrode of stainless steel, which has a diameter = 5 cm and an axial length = 4 cm, was used. For the test of generation of supersonic plasma flow, the first result on the ion velocity distribution with supersonic plasma flow ($M_\infty = \sim 1.2$) was obtained in a capacitively coupled plasma with electron temperature (T_e) ~ 2 eV and plasma density (n_e) ~ 10 cm^{-3} by using a Mach probe. Ion extraction system was applied to a linear plasma device called DiPS-2 (Divertor Plasma Simulator - 2: length = 3560 cm, diameter = 20 cm, source = LaB_6 cathode, average density ~ 10^{11} - 10^{13} cm^{-3}, T_e ~ 1 - 20 eV for Ar plasmas) [4]. To analyze drift velocity in supersonic plasma flow in terms of discharge currents and biased voltages to ion extraction electrode, a laser induced fluorescence (LIF) system was adopted along with the measurement of a Mach probe. The LIF system composes of a tunable diode laser with a master oscillator power amplifier (MOPA), which has typical ouput power = 10 - 100 mW, line width = 1 MHz, coarse tuning range = 665 - 675 nm with a rotating grating, fine tuning range = 0.45 nm with piezo-electric actuator control from 0 to 100 Volt, and a mode-hop free tuning region > 16 GHz, with current coupling method, to pump Ar II transition $3d^4F_{7/2}$ metastable level to the $4p^4D_{5/2}$ level at 668.43 nm. The 442.60 nm fluorescence light emitted from $4p^4D_{5/2}$ level to $4s^4P_{3/2}$ level was collected to determine drift velocity in supersonic plasma flow. For validity of experimental results on supersonic plasma flow, LIF data was compared with those of Mach probe with various calibration factors.

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SOLPS 5.0 simulations of the inner divertor detachment of L-mode plasmas in ASDEX Upgrade with convection-dominated radial SOL transport

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SOLPS 5.0 simulations with a convection-dominated radial ion transport model have shown improved correspondence to measurements of inner target ion flux and inner divertor electron density (n_e) for unseeded, deuterium-only low-power L-mode plasmas in ASDEX Upgrade discharges [1] with different degrees of inner divertor detachment. Simultaneously, a match to the n_e and T_e measurements on the LFS midplane was maintained.

Cross-field drifts and decreased near-SOL ion diffusivity were found essential in simulating the experimentally observed inner divertor conditions in [2,3]. In this work, a convection-dominated radial ion transport model was applied poloidally across the scrape-off layer (SOL) with the radial diffusivity decreased to D~0.01-0.1 m^2s^{-1} and the steepening of the n_e profile compensated by radial convection with v~10-100 ms^{-1} away from the separatrix. The anomalous convection was used as a proxy for the non-diffusive transport processes. The drift terms were activated. This provided a significant improvement in the description of detached inner target conditions in L mode with respect to earlier work in, e.g., [2,4], without the need for poloidally varying transport assumptions as made in [5].

Detachment of the inner and outer divertors in the simulations was indicated by a significant decrease in the ion flux on both targets, consistent with the target Langmuir probe measurements. Consequently, the experimentally observed roll-over of the integrated target ion current at $n_{e,edge}>2.2\times10^{19}$ m^{-3} was reproduced. Concomitantly, the simulated inner sub-divertor neutral fluxes agreed within a factor of 2 with pressure gauge measurements.

The spatial evolution of the high-field side high-density front (HFSHD) [6] with increasing upstream density was simulated in agreement with spectroscopic measurements. As observed in [3], the decreased diffusivity was essential in confining the HFSHD in the SOL and avoiding its excessive diffusive transport into the core, allowing increasing the D_2 fuelling to experimental levels. An increase in the HFS SOL n_e with respect to the LFS n_e was observed also on the HFS midplane in agreement with reflectometry measurements.

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Divertor current measurements during type I ELMs in DIII-D

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In DIII-D, large currents flowing into the divertor floor during edge-localized modes (ELMs) have been measured by an array of shunt current resistors before an increase of heat flux is measured by IR thermography. These currents were predicted by a qualitative model for the non-linear ELM phase and are described as critical for ELMs, as they cause increased stochasticity in the Scrape-Off Layer (SOL) and thereby facilitate further particle and energy transport out of the plasma [1]. Typically, the current measured by a single tile during an ELM can reach 500 A. This amounts to 10-50 kA flowing in the divertor tiles. The temporal evolution of the ELM currents shows a first phase with large amplitude oscillations, occurring before the heat flux increase measured by infrared thermography at the same location, lasting approximately 0.05 ms. A second phase follows where the time evolution of the divertor current mimics the evolution of the divertor heat flux. Preliminary toroidal mode number analysis of the divertor currents appears consistent with a mix of low-n modes ($n<5$), within the limit of resolution. A consistent rotation pattern of the divertor tile current structures is not observed. The diagnostic consists of 40 tiles distributed in five concentric circles in the lower divertor and two circles in the upper divertor with sampling rates range between 50 - 500 kHz. While the present tile current array has been optimized for disruption analysis our studies encourage dedicated ELM current diagnostics and indicate the potential for influencing the ELM character by perturbations through non-axisymmetric divertor bias.

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Effects of stochastic magnetic field structure on edge impurity emission distribution in LHD

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In LHD, impurity emission distributions in the edge region have been measured with an imaging spectroscopy system for different charge states, by changing the stochastic layer thickness and plasma parameters. Emission from C^{1+} is located along the divertor legs, while those from C^{2+} and C^{3+} are mainly come from the stochastic layer. It is found that C^{2+} and C^{3+} emissions are strongly affected by the edge T_e (electron temperature) profiles, which is modulated by the stochastic magnetic field structure. Especially, a flattening of T_e caused by the stochastic layer leads to enhancement of the C^{2+} and C^{3+} emissions when the T_e at the flattening region enters ionization potentials of these charge states. This enhancement leads to a stable partial detachment in high density regime. It is also found that thicker stochastic layer effectively widens the emission distributions in radial and poloidal directions. The flow speed of C^{2+} estimated from Doppler shift analysis are in a range of 10 to 20 km/s and flow towards the divertor plates. It is found that the flow speed becomes faster for the thicker stochastic layer. These results suggest an important role of the magnetic field structure of the edge stochastic layer on the impurity emissions, transport, and the resulting divertor operations.

Edge magnetic field structure in the stochastic layer is considered to play an important role in terms of the impurity transport as well as of the resulting impurity emission distribution. Two dimensional (2D) distributions of the carbon impurity emissions are measured by using the imaging spectrometer system. An array of 130 fibres is installed in order to resolve the spatial structure in the edge stochastic layer, that is, the divertor legs, X-point, the stochastic layer, and the confinement region. From the reconstructed 2D distributions, clear correlation of the emission distributions with the magnetic field structure have been identified as described above. All the emissions shift toward the upstream region when the plasma density is increased, which is considered due to a systematic decrease of T_e for a fixed input power. It is found that the thin stochastic layer configuration provides a wide T_e flattening region at the inboard side, ~ 20 cm, which is caused by the stochastization around X-point. As the plasma density is increased, the T_e at the flattening region enters the ionization potentials of the C^{2+} and C^{3+}, 24.4 to 64.7 eV, at $6x10^{19}$ m^{-3} with NBI power of 15MW. Then the divertor particle flux shows sudden decreases and thus carbon source emission (CII 2s2p3s-2s2p3p, 514 nm) decreases as well. However, the emissions from C^{2+} (CIII $1s^22s3s$-$1s^22s3p$, 465 nm) and C^{3+} (CIV $1s^25f$-$1s^26g$, $1s^25g$-$1s^26h$, 466 nm) remain high despite the decrease in the carbon source, and the partially detached phase with enhanced carbon emission is maintained up to $10x10^{19}$ m^{-3}. At the conference, the detailed analysis of the relation between the impurity emissions, flow velocity, and the magnetic field structure are presented.

On the study of non-radiative collisional processes relevant to fusion edge plasmas under low energy ion and electron impact

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It is the era to search for the alternative to conventional source of energy. It was Albert Einstein who established the mass-energy equivalence that opened the window to think about the application of peaceful nuclear energy. Worldwide there are several fusion program going on in which DT reaction is confined either by magnetic or inertial confinement. A large variety of atomic and molecular species are present in the plasma edge, coupled with the relatively broad range of charge states of impurity ions which makes the collision physics of the edge plasma very complex. Non-radiative collisional processes play vital role in the understanding of the various mechanisms in edge plasma boundary. Tungsten is suited as the best candidate for the wall material in fusion devices and for radiadtive cooling inert gases are supposed to circulate in diverter region. In view of the above, we have developed an experimental set-up using time of flight mass spectrometry to investigate the non-radiative atomic and molecular processes relevant to fusion edge plasma using low energy ion as well as electron impact. Low energy N2+ ion beam produced from Coluron ion source is allowed to interact with tungsten surface and we found the formation of tungsten nitride layer on the surface of the tungsten surface [1, 2]. In addition, we investigate the collision of keV electron beam with neutral Argon gas which led the formation of multiply charged Argon ions [3]. The details of the experimental work with theoretical comparison will be presented and discussed.

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Inter-ELM pedestal fluctuations and their parametric (in-)dependencies

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In order to predict and optimize future fusion devices such as ITER, deep understanding of the plasma edge and its behavior is required since it strongly determines the overall plasma performance. In a regime of improved plasma confinement, the high confinement mode (H-mode), steep gradients in the plasma pressure build up forming the pedestal. These gradients are limited by an ideal magnetohydrodynamic limit, which if exceeded is leading to edge localized modes (ELMs) that relax the pedestal and expel large particle and heat fluxes towards the first wall and divertor, exceeding material limits. To avoid the stability limit and therefore, ELMs, fundamental knowledge on the processes, determining the pedestal structure, is required. Furthermore, transport across the pedestal determines the inter-ELM fluxes towards the wall, which will have a significant impact on the lifetime of future power plant.

Within the last years, inter-ELM fluctuations, located in the pedestal region, have been reported from several large tokamak experiments [1,2,3]. Mostly, they are also detected by local measurements of density fluctuations [4,5,6]. The onset of the inter-ELM fluctuations is clearly correlated with the evolution of the pedestal gradients [7], indicating the impact on the transport across the pedestal [8]. Furthermore, the associated density fluctuations exhibit characteristic, burst-like behavior [9].

The universality of the inter-ELM fluctuations and their connection to the evolution of the pedestal structure as well as the possible transport associated to them motivate studies across wide parameter ranges and cross experiment comparisons. This contribution characterizes the inter-ELM pedestal fluctuations across a range of plasma parameters, i.e. collsionality, triangularity or E×B rotation, and compares observations from different experiments. The most common features are present over the whole range of investigated parameters: The onset of the inter-ELM fluctuations is coupled to the evolution of the pedestal gradients and the detected fluctuations frequency in the lab-frame scales with the background rotation in the steep gradient region. These observations indicate a robust underlying generation mechanism that acts independently of all investigated parameters in the explored ranges.

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Parallel Energy Transport in Detached DIII-D Divertor Plasmas

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A comparison of experiment and modeling of detached divertor plasmas is examined in the context of parallel energy transport in order to validate and improve models used for divertor design. A previous modeling effort with SOLPS was able to reproduce experimentally observed divertor power dissipation and profiles of divertor T_e and n_e in detached helium L-mode plasmas for matched divertor entrance parameters [1]. However in this report, similar modeling of deuterium H-mode plasmas finds a radiation shortfall compared to experiment. To investigate the causes of this discrepancy, relative fractions of power carried by electron thermal conduction versus plasma convection are experimentally inferred from power balance measurements of radiated power and target plate heat flux combined with Thomson scattering measurements of T_e and n_e profiles. Relatively flat T_e profiles with $T_e \leq 20$ eV from the X-point to the target implies a large fraction of the power is carried by parallel convection, $\geq 50\%$, even before the onset of detachment. In contrast fluid modeling with SOLPS produces sharp gradients for $T_e \leq 20$ eV, indicating transport dominated by electron conduction. In addition SOLPS produces very little radiative dissipation of heat flux for $T_e \geq 15$ eV, while bolometer measurements indicate significant radiation in regions of similar temperature. To further compare modeling with experiment, inferred levels of parallel convective transport in the divertor will be examined more directly with insertable Mach probes and Coherence Imaging Spectroscopy (CIS), while contributions to radiative dissipation from carbon and deuterium will be measured with UV spectroscopy. Detailed analysis will aim to isolate and identify the physical processes responsible for the discrepancy between these measurements and SOLPS modeling.

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H-mode detachment and the asymmetry with ITER-like W divertor operation in EAST

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For ITER and future tokamak fusion reactors, the extremely high heat flux and erosion rate will limit the divertor target lifetime severely. The detached divertor operation, especially partial detachment, characterized by a clear reduction of both heat/particle fluxes and electron temperature on the target, is a very promising solution to these issues. In the EAST 2017 campaign, the H-mode detachment was obtained with ITER-like tungsten (W) divertor in upper single null (USN) discharges for the first time. The detached divertor operation in EAST is accompanied by a rollover of particle flux to the target with the increase of the plasma density. In the same time, the plasma electron temperature close to the target plates was reduced below 5 eV. The neutral density and pressure increase was clearly observed during the detachment, though the degrade of the plasma confinement was also observed, in agreement with the previous results on other tokamaks [1]. New results in EAST also show that the onset of detachments in the inner and outer divertors occur simultaneously, which is attributed to the geometry of the W divertor chamber. The denstiy threshold for the H-mode detachment ($n_e/n_G \sim 0.65$) is higher than previous L-mode plasmas in EAST. Moreover, the density threshold of detachment for the ion $B\times\nabla B$ direction towards the upper X-point is smaller than that with ion $B\times\nabla B$ direction away from the upper X-point.

In addition, the divertor asymmetry of peak particle flux measured by the divertor triple Langmuir probe arrays favors the outer divertor in USN for the $B\times\nabla B$ away from the upper X-point, while the asymmetry becomes more symmetric, or slightly reversed for $B\times\nabla B$ towards the upper X-point. The total particle flux [2] on the outer target is much higher than that on the inner target throughout the whole phase of plasma density ramping up, in both normal and reversed B_t discharges. It is also interesting to note that the total particle flux at the lower outer is often significantly enhanced as the plasma density increases in USN discharges. Future experiments with power scan, different divertor geometry, impurity seeding and 3D footprint effect are planned. New advances that may arise in the upcoming EAST campaign will also be presented.

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Modeling non-axisymmetry in the DIII-D small angle slot divertor using EMC3-EIRENE*

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The 3D edge transport code EMC3-EIRENE [1] is used to evaluate the effects of non-axisymmetric misalignment of the Small Angle Slot (SAS) divertor on detachment characteristics on DIII-D. The SAS is a slot divertor with a close-fitting baffle designed to control neutral recycling to yield a wide region of low electron temperature (T_e) across the target plate near the strike point and to achieve mitigated heat flux to the divertor at a lower upstream density as compared to a more open geometry. The current installation of the SAS divertor concept in DIII-D is misaligned to the magnetic field, with a deviation from ideal axisymmetry on the order of +/- 5 mm. As the SAS is expected to be sensitive to the position of the strike point relative to the small-angle target, this misalignment could cause toroidal asymmetry in the divertor conditions and the degree of detachment. The misalignment also complicates experimental measurements that are made at toroidally offset positions.

EMC3-EIRENE is used to verify SOLPS [2] predictions [3] of the key SAS features, with initial axisymmetric simulations for the optimized strike point position for the SAS showing that the transition to low T_e occurs at approximately the same upstream parameters as found in SOLPS. Further axisymmetric simulations using EMC3-EIRENE will be used to verify that the key features of the SAS, such as strong neutral compression in the slot is reproduced for the same input parameters used in previous calculations [3]. The axisymmetric SAS simulations will also be compared to an open divertor shape with the input parameters held constant. EMC3-EIRENE will be used to assess the impact of the misalignment, simulated as a sinusoidal offset with a toroidal mode number of 1 and a magnitude of 5.1 mm, which approximately describes measurements made in situ via a coordinate measuring machine. This is an extension of 3D simulations where a toroidal span of SAS geometry was modeled as an open divertor to assess the effect of local misalignment [4]. The effect of the offset on neutral pressure, divertor heat and particle flux, and downstream temperature and density will be evaluated. Finally, synthetic diagnostics will be used to evaluate the ability of the DIII-D diagnostic system to measure the effect of misalignment.

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Utilization of impurity granule induced ELM triggering in next step fusion devices

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Low-Z granule injection into high performance discharges on both DIII-D (Li, C and B_4C) and EAST(Li) has been shown to promptly trigger edge-localized modes(ELMs) without significant plasma degradation for all but the lowest power discharges. During the non-active phase of ITER, supplemental ELM triggering will be essential to provide impurity control. Thus, the utilization of impurity granules to decouple the ELM triggering mechanism from the fueling cycle supplements currently envisioned ITER ELM pacing techniques[1] and could extend the operational parameter space. In addition, we find that high frequency granule injection can result in partial ELM heat flux mitigation through pacing.

A series of lithium granule injection experiments on EAST [2] and DIII-D [3,4] utilized four distinct plasma parameters sets; we refer to these as EAST low-power, EAST high-power, DIII-D hybrid-shape, and DIII-D ITER-baseline (IBS) discharges. During the EAST high-power discharges ($W \approx 175$ kJ, $I_P = 400$ kA, $P_{AUX} \approx 6.5$ MW), injection of granules with diameter > 600 microns was sufficient to promptly trigger ELMs greater than 95% of the time. Injection frequencies up to 125 Hz were compatible with high performance ELMy H-modes, albeit with a 10% reduction in energy confinement. Due to a natural ELM frequency of >200 Hz in these discharges, however, the granule injection did not result in ELM pacing. In contrast, injection of the granules larger than 700 microns in diameter during EAST low power discharges ($W \approx 135$ kJ, $I_P = 450$ kA, $P_{AUX} \approx 4.3$ MW) resulted in H-L back transitions, indicating that the auxiliary heating was insufficient to replace the energy lost to granule ionization. In the DIII-D hybrid-shape scenario ($W \approx 800$ kJ, $I_P = 1.2$ MA, $T_{inj} = 2.9$ N m) [3] high frequency granule injection led to an ELM frequency multiplication of 3-5 over the natural frequency and a reduction of the ELM peak heat flux from ~ 700 +/- 50 W/m^2 to ~ 300 +/- 100 W/m^2. Complete heat flux mitigation was not observed during IBS discharges ($W \approx 700$ kJ, $I_P = 1.3$ MA, $T_{inj} < 0.1$ N m) with ELM pacing [4], where a bimodal structure appeared in the measured peak heat flux distribution function. Specifically, mitigated ELMs were interspersed with large ELMs with peak heat flux as large as those from the baseline discharges. Thus, while the ability of injected impurity granules to promptly trigger ELMs under a range of parameters is established and a scaling of granule size to ITER baseline conditions can be extrapolated, the occurrence of interspersed unmitigated ELMs in the IBS are problematic for granule ELM pacing as a mitigation technique unless the nascent cause for the bimodality can be determined and if possible addressed.

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The effect of lithium conditioning approaches for plasma-facing surfaces on the edge and core temperature and density profiles*

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Experiments on tokamaks at PPPL (NSTX [1], LTX [2], and CDX-U [3]), on the EAST and DIII-D tokamaks [4,5], and on FTU [6] have now investigated the effect of lithium surfaces on the scrape-off layer (SOL) and core plasmas using coatings on varying plasma-facing components (PFCs) including graphite or molybdenum tiles, porous tungsten surfaces, and stainless steel surfaces. These experiments have continued earlier work on lithium wall conditioning performed on T11-M [7], and TFTR [8]. Generally a reduction in the edge density gradient is observed with Li conditioning; however, both flat and peaked temperature profiles have been observed in different experiments [2,7], as well as widening of the H-mode temperature and density pedestal widths [1].

Technically, lithium has been introduced by pellet, granule, or aerosol injection, by direct evaporation, by evaporation into a backfill of helium gas, by between-shots fills of the bulk liquid metal into limiter structures, and by flowing liquid lithium over a limiting surface, during a discharge. A number of the tokamaks used in these experiments were limiter machines (TFTR, T11-M, CDX-U, FTU, and LTX), while others employed a divertor (NSTX, EAST). Background vacuum conditions and fueling have varied as well among the various experiments. While there are clear trends in the effects of lithium and lithium conditioned plasma-facing surfaces on the plasma performance, SOL effects, and (where analysis has been done) the surface science, there are also notable differences. Very recently a flat electron temperature profile with corresponding high edge temperatures was produced with a liquid lithium plasma facing surface in LTX [2], compared to more modest changes seen in NSTX and DIII-D [1,5], or even small decreases in the edge electron temperature seen in the EAST experiments with a flowing liquid lithium limiter plate [4]. The TFTR lithium conditioned discharges [8] had highly peaked temperatures profiles. Here we will show commonality between the various experiments, and attempt to reconcile a number of the disparate observations.

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Simulation study of the relation between P_{rad}, Z_{eff} and lifetime of W target for CFETR phase II with Ne and Ar seeding

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China Fusion Engineering Test Reactor (CFETR), which is designed to bridge gaps between ITER and DEMO, will be operated in two phases. The CFETR phase II is proposed to validate DEMO, therefore aimed to achieve over 1 GW fusion power, with major radius 6.6 m, minor radius 1.8 m and B_t 6~7 T [1]. In order to avoid the fuel retention issue, tungsten wall would be preferred for CFETR, which means there is no intrinsic radiator like C. Therefore, impurity seeding is indispensable to achieve a significant radiation power P_{rad} to satisfy the engineering limit of heat flux onto the divertor target. Besides, the divertor should effectively screen the impurities to reduce Z_{eff} in order to be compatible with the core plasma. Furthermore, the sputtering of the W target should be kept at low level to avoid frequent maintenance.

A series SOLPS simulation have been performed for the lower-single-null configuration with Ne and Ar seeding. It was found that, for similar P_{rad} (~80% of the 200 MW PSOL), the Z_{eff} is lower for Ar seeding due to the lower ionization energy (the ionization region of Ar is closer to the divertor target) compared with Ne [2]. However, W is easier to be sputtering by Ar for its larger atomic mass and possible higher charge state. To figure out the operational window for CFETR phase II, it is crucial to take the lifetime of W target into consideration.

In this work, the relation between P_{rad}, Z_{eff} and lifetime of W target are studied by SOLPS+DIVIMP simulation. By assuming 200 MW P_{SOL} and fixing cross-field transport coefficients which could give a SOL width according to Eich law, the simulations are performed for Ne and Ar with different impurity seeding rates and locations including inner and outer baffle and main chamber. Limitations on Z_{eff} and W concentration in core plasma, heat flux onto divertor and W target sputtering flux are then applied to find acceptable operational window for different impurity seeding scheme, which would be critical for the divertor design for CFETR and future fusion reactor.

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Abstract Withdrawn

Effects of Drift on the Divertor Plasma Transport in LHD

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The asymmetric plasma heat and particle loads on the divertor tiles, which are located at the positions of a symmetric magnetic field lines structure, have been observed in the Large Helical Device (LHD). The degree of the asymmetry depends on the toroidal field magnetic field direction and plasma operation, and drift is considered to be a cause of the asymmetry. In this presentation, the results of the evaluation of the degree of the asymmetry are shown, and the mechanisms are discussed.

In tokamaks with poloidal divertor configuration, in/out asymmetry is usually observed in divertor particle and heat flux, and that causes the asymmetry of deposition/erosion of plasma facing materials. The mechanisms of the in/out asymmetry are still under investigation, and plasma flow, drift, and the in/out asymmetry of magnetic field configuration can affect the asymmetry. Understanding of the mechanisms of divertor particle and heat load asymmetry is necessary to consider plasma-wall interaction in future fusion reactor.

In LHD, plasma experiments have been conducted under the helical divertor configuration, which is a built-in divertor magnetic field lines structure in the heliotron-type magnetic configuration. To investigate the divertor plasma properties, Langmuir probe arrays are embedded in the divertor tiles, which locate near the mid-plane at torus inboard-side divertor in seven of ten toroidal sections. At each toroidal section, a pair of the Langmuir probe arrays is located at two positions in symmetric relation with regard to the magnetic field lines structure. The asymmetric particle flux and heat load on the divertor tiles have been observed in LHD [1]. The asymmetry is inverted by changing the toroidal magnetic field direction. Therefore, $E\times B$ drift and/or $B\times\nabla B$ drift can be considered to be the cause of the asymmetry. Recently, the first deuterium plasma experiment in LHD was conducted in 2017. The isotope effect on the asymmetry was evaluated, and the result shows that the effect is not clear. That possibly suggest $E\times B$ drift is the dominant mechanism of the asymmetry.

In this presentation, the results of the further investigation of the asymmetry, such as plasma parameter dependences will be shown, and the mechanisms of the asymmetry in helical system will be discussed.

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The effect of the secondary x-point on the Scrape-Off Layer transport in the TCV Snowflake Minus divertor

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TCV experiments show evidence that the cross-field transport in the low-field side (LFS) Scrape-Off Layer (SOL) of the conventional Single-Null (SN) configuration is strongly enhanced when placing a secondary x-point in the common flux region of the primary separatrix. Such a magnetic configuration is known as the Snowflake Minus (SF-) [1].
The exhaust properties of the SF- configuration were investigated on TCV in Ohmically heated, L-mode attached plasmas with a range of x-point separations, magnetic field directions and locations of the secondary x-point (low-field side, LFS, or high-field side, HFS, SOL). The target heat flux profiles at all strike points are simultaneously measured with an Infrared Thermography (IR) system [2] and the main SOL plasma kinetic profiles with a fast reciprocating probe (RCP) [3]. The measured power repartition between the two branches of the SOL diverted by the secondary x-point yields an effective heat flux width $\lambda_{q,u}^{eff}$ for the SOL. It is found that the LFS SOL cross-field transport is strongly enhanced by the presence of the secondary x-point, for both magnetic field directions, with $\lambda_{q,u}^{eff}$ being a factor three larger than the value measured by the RCP at the outer mid-plane or by IR at the outer target of a comparable SN. The RCP inspection of the low poloidal field region reveals that the profiles of SOL density and ion saturation current develop steep gradients in proximity of the secondary separatrix, while the profiles flatten in between the two separatrixes, compared to the SN divertor. In contrast, the cross-field transport in the HFS SOL is not significantly affected by the presence of the secondary x-point. However, similar to comparable SN discharges, it is sensitive to the magnetic field direction, with a broader SOL width for reversed field.
Placing the secondary x-point on one side of the SOL also affects the power balance between inner and outer divertor. This is interpreted as the effect of changed parallel transport in the SOL due to the changes in the geometry of the flux tubes caused by the presence of the secondary x-point, and was already seen in fluid calculations assuming constant cross-field transport coefficients [4].

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The role of trapped neutrals in detachment onset and pedestal fueling studies on DIII-D

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A series of experiments on DIII-D makes use of the lower divertor floor and shelf and the upper divertor ceiling baffle configurations to isolate the effects of divertor closure and determine the role of trapped neutrals on detachment onset and pedestal fueling. We find that, in these experiments, increasing divertor closure can reduce the density at pedestal top, $n_{e,ped}$, at detachment onset by up to 30% compared to an open divertor. Matched discharges in each configuration are characterized with DIII-D's boundary diagnostic suite, including a high-density edge Thomson scattering array, in-target Langmuir probes, and divertor neutral pressure gauges. Experiments are interpreted with boundary modeling, primarily using SOLPS and OEDGE. Detachment at lower $n_{e,ped}$ appears to be a combination of multiple consequences of trapping recycling neutrals in the more-closed divertor: first, an increase in trapped neutrals leads to more-dissipative divertor conditions for similar upstream separatrix density, $n_{e,sep}$; and second, trapping of recycling neutrals leads to a lower $n_{e,ped}$ for similar $n_{e,sep}$.

A greater density of neutrals trapped in the divertor means lower divertor temperature[1] and more radiation losses—in the DIII-D divertor this is dominated by C radiation, which peaks at ~10 eV—decreasing temperature further and facilitating detachment. The enhanced trapping of neutrals in the closed divertor is supported by SOLPS5.1 modeling of matched H-mode discharges comparing the open and closed divertors. These simulations show 4 times higher neutral density in the closed divertor than the open for matched $n_{e,sep}$. This higher neutral density corresponds to a lower temperature in the closed divertor than the open, consistent with measurements of detachment at lower $n_{e,sep}$ in the closed divertor.

Trapping recycling neutrals also means reduced fueling of the pedestal; these experiments show an ≈15% decrease in $n_{e,ped}/n_{e,sep}$ in the closed divertor compared to the open. This reduction in pedestal fueling is demonstrated in ionization profile calculations: SOLPS5.0 shows increased core ionization[2] in the open divertor case, and OEDGE indicates a 50% increase in ionization inside the separatrix in the open case compared to closed[3]. These results are consistent with past experiments and UEDGE modeling[4,5] demonstrating heat flux reduction and reduced core ionization in the upper DIII-D divertor.

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Misalignment of particle and heat fluxes at the divertor plate: numerical modeling coupling turbulence and transport codes

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It is sometimes observed in both experiments [1,2,3] and simulations [4,5] a misalignment between the particle and heat fluxes (or more generally of plasma density and temperature) profiles at the divertor plates, with the peak values occurring ad different radial positions. Such misalignment can result in the increase of the surface erosion and can affect the main plasma performance through the reduction of the upstream electron pressure. Still, this issue is usually overlooked.
We investigate the influence of the divertor configuration on the alignment between the particle flux Γ and heat flux $q_{||}$ through a set of first principle and transport simulations.
The TOKAM3X code [6] is here used to model the plasma turbulence in the edge and scrape-off layer (SOL) in realistic diverted geometry, using experimental magnetic equilibria. Recent numerical improvements of the code allow to resolve electron and ion temperature in diverted configuration. The impact of plasma temperature on the edge and SOL average profiles and on the turbulence properties itself is discussed by comparison with a second simulation performed with the isothermal assumption.
The alignment of temperature, density, particle and heat fluxes profiles at the divertor plates is discussed.
The poloidal distribution of the effective diffusion coefficient given by turbulence is computed and used as an input for the 2D transport code SolEdge2D-EIRENE [7], allowing for an estimate of the impact of the neutral dynamics on the $q_{||}$ -Γ shift.

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Influence of recombination front region on plasma detachment in a linear divertor plasma simulator

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Various researches on plasma detachment are currently underway through complementary studies with magnetically confined fusion devices, linear diverter plasma simulators[1], and numerical simulations in order to control heat and particle loads on plasma-facing components. It has been investigated in the linear divertor plasma simulator NAGDIS-II that recombination front (RF) region, in which volume recombination strongly occurs in a detached plasma, has strong influence on the characteristics of plasma detachment. Detailed observation of plasma profiles and dynamical behaviors around the RF was conducted using newly developed two-dimensional driving Langmuir probe (2-D LP) as well as a laser Thomson scattering (LTS) diagnostics[2]. The LTS system, developed with international collaboration of DIFFER, is enable to measure electron temperature (T_e) less than 0.5 eV. LTS data can be utilized to calibrate 2-D LP data. Plasma fluctuations near RF were measured with a microwave interferometer (MI).

In a detached plasma, axial and radial profiles obtained with the 2-D LP show monotonically decreasing electron density (n_e) and T_e along the magnetic field line at the central region of plasma column. On the other hand, in the peripheral region of the plasma column, n_e peaks near the RF, which means a strong local cross-field transport from the central to peripheral region exists near the RF. Plasma instability accompanied with strong n_e fluctuation was observed by MI near RF[3]. The instability leads to the enhancement of the cross-field plasma transport near RF in a detached plasma. The local enhancement of the cross-field plasma transport also changes the plasma flow pattern, showing that inverse plasma flow along the magnetic field at the peripheral region of plasma column appeared[4].

We have also investigated the effects of magnetic field structure on RF by simulating magnetically expanding and contracting plasmas[5]. The total ion particle flux measured with a large-diameter target plate dramatically changed under the detached plasma condition compared to that in attached plasma. Under the detached plasma condition, the magnetically expanding plasma clearly exhibited a significant degradation of detached plasma formation.

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Integrated modeling of core, edge pedestal and SOL for high β_N steady-state scenarios on DIII-D

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A new integrated modeling of Core, Edge Pedestal, and Scrape-off-layer (CESOL) finds trade-offs between core performance and SOL/divertor control for high β_N steady-state scenarios on DIII-D. One of the key challenges for future reactors such as ITER, FNSF and DEMO is simultaneous optimization of core fusion performance and boundary techniques to maintain acceptable heat loads to the plasma facing components (PFCs), which requires improved understanding of interaction among the regions. CESOL consists of three independent, compound Integrated Plasma Simulator (IPS) workflows: FASTRAN (1-D core transport), EPED1 (edge pedestal), and C2/GTNEUT (2-D SOL plasma/neutral transport). In the core region FASTRAN computes all transport channels with TGLF and is self-consistent with an EPED1 edge pedestal. The total particle and energy fluxes are matched at the separatrix between the FASTRAN+EPED1 and C2 workflows in an iterative steady-state solution procedure. This specific coupling aims to determine the density and temperature at the separatrix, which are used to update the input to EPED1 and close the strong nonlinear dependency among the core, edge pedestal, and SOL regions.

Two self-consistent non-inductive scenarios are found at $\beta_N > 4$, both relying on the planned addition of a second off-axis neutral beamline and broad current profile at $q_{min} > 2$. Fully non-inductive operation requires relatively low density for efficient external current drive. If the divertor conditions are unrestrained (i.e., high temperature and high heat fluxes are accepted), a low density ($n_{e,ped} = 3.5\text{x}10^{19}$ m^{-3}) scenario has been found. In this case, CESOL predicts that the pedestal pressure increases with the separatrix density, which in turn improves the core confinement. The separatrix density depends strongly on the SOL/divertor conditioning such as wall recycling, gas puffing and pumping, as well as turbulent transport characteristics in the SOL as assumed with specified D and $\chi_{e,i}$ in the CESOL modeling. Requiring a dissipative divertor (low T_e and heat flux) leads to a high-density scenario, with $n_{e,ped} = 7$ x10^{19} m^{-3}. In this case the pedestal pressure increases rapidly with β_N, which tends to make simultaneous optimization of core performance and pedestal more promising. However, the pedestal pressure is not predicted to be sensitive to the seperatrix density, unlike the lower density case. CESOL modeling shows that higher density operation reduces the heat flux to divertor by more than factor 2, although the full detached divertor condition is not achieved. The increased bootstrap current fraction must compensate the reduced current drive efficiency at higher density, which constrains the fully non-inductive operation space at relatively higher $q_{95} > 6$ to satisfy exact current and power balance. This CESOL modeling elucidates the connection between core and SOL/divertor via edge pedestal in optimizing the tokamak fusion performance. Supported by US DOE under DE-AC05-00OR22725, DE-FG02-95ER54698, and DE-FG02- 95ER54309.

SOLPS-ITER analysis of L-mode KSTAR divertor detachment experiments

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Density scans in deuterium-only, lower single null in KSTAR low-confinement mode (L-mode) discharges show that the outer target detaches at lower upstream density than the inner target. In these plasmas the ion $B\times\nabla B$ direction was into the divertor. These experiments, carried out during the 2017 KSTAR campaign, were at a toroidal magnetic field $B_T = -1.8$ T, a plasma current -0.5 MA ($q_{95} = 4.7$-5.1), and 0.7 MW of neutral beam heating, resulting in a total conducted power of 0.5 MW across the separatrix. While detachment asymmetry is in contrast to the general finding in other tokamaks, it was similarly observed on TCV [1]. Like in TCV, the divertor plasma configuration in KSTAR is based on a short poloidal inner divertor leg on a vertical target and long poloidal outer divertor leg on a horizontal target. In addition, both TCV and KSTAR are full carbon devices.

The target particle flux profiles and their evolution from the attached to high recycling and detached state as predicted by the edge fluid code SOLPS-ITER [2] are in qualitative agreement with the fluxes measured by a set of embedded single Langmuir probes. The simulations were executed with and without carbon impurity using, for the latter case, the Bohdansky formula and 1% fixed yield for physical and chemical sputtering, respectively. Drifts were not yet activated in the simulations. As a consequence of the divertor geometry, the predicted neutral density due to recycling from both inner and outer targets peaks between the central and outboard vertical divertor target regions. This enhances volumetric losses in the outer divertor. When carbon impurities are included, radiation losses in the outer divertor region are 2-3 times higher than in the inner divertor, further promoting detachment at the outer target at lower upstream density.

Furthermore, the simulations indicate that the volumetric loss terms due to plasma-neutral reactions and perpendicular transport near the separatrix from the scrape-off layer to the private flux region are the main loss mechanisms of pressure and power. Electron impact ionization and charge exchange produces strong power and momentum losses in the separatrix vicinity, hence facilitating partial detachment there. Molecular assisted reactions cause significant power and momentum losses not only near the separatrix, but also in the far-SOL region, facilitating complete detachment.

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Liquid Li and Sn as DTT divertor targets, comparison on their effects on the heat loads and SOL properties

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The behaviour of the scrape-off plasma of the Italian projected tokamak DTT is analysed for the standard single null configuration by means of the 2D edge code TECXY1 when either Lithium or Tin are used as liquid target materials. The targets are modelled as a thin liquid metal layer superimposed on a tungsten substrate that faces the plasma and then is hit directly by the plasma particles, while the W bottom is kept at a fixed temperature by the cooling system. By solving the heat transport equation the temperature at the target top is calculated and the evaporation rate derived. The total impurity source strength is then estimated by including also sputtering. The impurity concentration and the involved radiative losses are calculated self-consistently by solving the multifluid plasma transport equations. A density scan is carried out for different target arrangements, in terms of the coolant temperature and thickness of the W substrate. Two different input power into the SOL P_{SOL}= 35 and 25 MW, are considered, following to the predictions [2] of the self-consistent edge-core coupled code COREDIV [3] for the maximum injected power P_{inj}=45 MW. 35 MW is approximately the value foreseen when no impurity is seeded in addition to the intrinsic one (W) over a quite wide operational density range. 25 MW are instead the reference value for the high density range, $\langle n_e \rangle \geq 1.5\times10^{20}$ m^{-3} when a small amount of impurity (N or Ar) is seeded. Lower P_{SOL} are not considered of interest here. Tin appears more promising than lithium in terms of radiative capacity, of wider ranges of applicability both of density and input power and of plasma purity. No clear detachment is observed for either Sn or Li except at very high density. For both solutions a regime where evaporation overcomes sputtering is more effective in dissipating the input power, provided that is kept low enough to ensure the stability of the code numerical solution. In this case a sort of vapour shielding seems to develop attached to the impurity source.

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High Performance Double-null Plasmas Under Radiating Mantle Scenarios on DIII-D

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We report on the reaction of high performance double null divertor (DND) plasmas to higher density radiating mantle conditions that lead to ≈50% reduction in divertor peak heat flux: $H_{98} \approx 1.4–1.7$, $\beta_N \approx 3$-4, $q_{95} \approx 6$, dRsep = 6 mm, and neutral beam plus ECH power input P_{IN} up to 15 MW, with ECH providing 3.5 MW. First, previous predictions from ELITE code analysis [1] indicate that τ_E should improve as density is raised by deuterium gas puffing under specific operating conditions, i.e., higher levels of P_{IN}, proximity to magnetic balance, and higher q95; our data largely support these predictions. We find that impurity profiles depend strongly on where ECH power is deposited inside the plasma. When the radial location of the ECH deposition was nearer to the magnetic axis (e.g., $\rho \approx 0.20$), the radial profiles of both the injected impurity (neon) and the intrinsic impurity (carbon) were flat or only slightly peaked. For deposition farther out (e.g., $\rho \approx 0.45$), the carbon and neon impurity profiles become much more peaked on axis due largely to a much stronger inwardly-directed pinch (STRAHL), suggesting that successful high performance DN plasma under these conditions using a higher-Z impurity may be more difficult due to greater impurity peaking. In fact, in comparing argon-based radiating mantles with neon-based mantles producing the same factor of two divertor heat flux reduction, we find that the plasmas using argon were more susceptible to triggering deleterious MHD modes. For the argon case, the changes in the current density profile may have triggered the formation of deleterious MHD modes (e.g., 5/2 and 2/1), which degraded τ_E and β_N; these modes were not observed in the corresponding neon case. Such deleterious MHD activity may complicate injection of higher-Z impurities into highly powered, high performance plasmas. However, the neon-based mantles still had significantly more core dilution than for argon-based mantles. We show that the choice of impurities, injection location, and details of scrape-off layer shaping all dictate where these injected impurities can be successfully pumped. Control over impurity movement is particularly important for DND and near-DND configurations where impurity pumping occur at multiple poloidal locations.

Taken together with previously reported studies, e.g., Ref. 1, these new results point to promising ways to access high power high performance regimes, while maintaining reduced divertor heat flux and adequate particle control. However, our data also show that there are still significant issues that must be overcome.

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Comparing theoretical and experimental scalings for power exhaust in seeded H-modes with SOLPS simulations of ASDEX Upgrade

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Material limits at the divertor target require future fusion devices to operate with a dissipative divertor, where the required dissipation is engendered by line radiation from seeded impurities. Extensive experience from experimental and theoretical investigations of seeded exhaust scenarios, see refs. [1,2] and references therein, guide the design of future devices. Existing modeling tools, such as SOLPS, are used to extrapolate to operational parameters outside the currently accessible parameter ranges.
Validated simulations of an ASDEX Upgrade H-mode [2] are used as a basis for parameter scans in heating power, neutral divertor pressure, plasma current and impurity concentration. The scaling of relevant detachment parameters within this database is analyzed and compared to experimental scalings of midplane and divertor properties [3,4] and analytical predictions for the required impurity concentrations for detachment [5,6]. It will be discussed what aspects of the more complete model in SOLPS are responsible for differences to the analytical models. The comparison with experimental scalings of divertor and midplane quantities like the neutral pressure or the separatrix density is used to asses the applicability of parameter scans derived from validated simulations for extrapolation from a given operating point to within the experimentally accessible parameter ranges.

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Atomic and molecular processes in plasma surface interactions and boundary plasma science

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It was recognized as early as 1968 by Bo Lehnert [1] that a zone of powerful gas - plasma interaction, formed near highly PMI exposed surface elements, can be the key to solving the PSI issue for sustainable nuclear fusion reactor operation. Today the "divertor detachment" regime in tokamaks is characterised by strong volumetric exchanges of particles, momentum and energy, provided by a gas plasma interaction zone near target surfaces. For any given plasma state, and set of plasma material interaction processes (boundary conditions), this gas - plasma interaction can be fully quantified. Firstly, by resorting to highly accurate atomic/molecular (A&M) collision data. These are made available from many national atomic data centers and numerous data coordination and evaluation activities initiated and driven by the IAEA nuclear data section [2]. Secondly the high standards in (kinetic) transport applications from nuclear (neutronics, radiation) applications. Their full mathematical analogy to neutral particle transport in plasmas has been carried over into now quite mature fusion plasma boundary Monte Carlo codes.

Today the underlying details of the particular A&M data set in edge plasma models are often not made very transparent. In the present review we will try to publicly expose the most dominant A&M processes individually, their uncertainty levels today, and to quantify their physical effects on boundary plasma dynamics (notably on divertor detachment) as well as the potential loss of information in reduced (approximate), computationally simpler models derived from them. The focus is on collision processes involving fuel (H), ash (He), wall material (e.g., Be, W) and seeding particles (e.g., N_2), including molecules and molecular ions formed from them. It is shown that the dominant gas plasma friction in typical detached divertor conditions is to be expected from molecule - ion collisions, rather than from the often quoted resonant charge exchange. Direct contributions from molecules and their ions to atomic lines also complicate visible light spectroscopy as a divertor plasma diagnostic. Furthermore they may lead to quite unexpected kinetic isotope effects in the D-T mixture of a future reactor divertor plasma.

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Abstract Withdrawn

Experimental Verification of Three-Dimensional Impurity Flows Due to Temperature-Driven Pressure Gradients

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Fluid modelling predictions of 3D flows due to parallel pressure gradients have been confirmed experimentally in the DIII-D tokamak using 2D Coherence Imaging Spectroscopy (CIS). 3D flows are of particular concern for large and next-generation stellarators where momentum loss from counter-streaming flows near magnetic islands can result in high-temperature detachment before onset of a high-recycling regime [1]. In tokamaks, the 3D flows in resonant magnetic perturbation (RMP) ELM-suppressed H-modes have been associated with the emergence of lobe-structures in the SOL and have been proposed as a possible mechanism for RMP-induced density pump-out [2].

On DIII-D, 3D flow effects have been observed during n=2 RMP ELM-suppressed H-mode discharges and in the presence of of large coherent n=1 magnetic islands. These islands were produced by external RMP coils during an L-mode inner-wall limited plasma. Using the CIS diagnostic, velocity images were obtained for a variety of island locations extending from the limiting surface to the far-SOL. A poloidally-alternating pattern of acceleration and deceleration, correlated to island positions, was observed with local velocity changes up to 10km/s and a scale length of 30-40cm at the mid-plane. Rotational screening inhibits the formation of 3D flows indicating that island formation, not simply 3D fields, is required.

These velocity perturbations were predicted with EMC3-EIRENE fluid modelling where 3D flows are generated by parallel pressure gradients resulting from the small temperature gradient across magnetic islands and an inboard/outboard density asymmetry. These parallel pressure gradients drive counter-streaming flows in island chains confined within the separatrix and flow velocity perturbation in the partially-stochasized SOL where the finite connection length of open field lines becomes an important contributor.

Quantitative comparison between EMC3-EIRENE and CIS measurements achieved using a newly-developed synthetic diagnostic reveals differences in the velocity perturbation's absolute magnitude of about a factor of 2. Other discrepancies reveal the importance of correctly simulating both poloidal and radial T_e profiles and suggest the need for inclusion of cross-field drifts for full-device modelling. Despite these differences, the pattern of alternating velocity perturbations throughout the SOL is captured, consistent with the role of parallel-pressure gradients in driving 3D flows.

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Effects of divertor target shape and baffling on plasma detachment: from current device to CFETR*

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The steady-state operation of next-step fusion devices requires both the deposited heat flux density on the divertor target below 10 MW/m^2 and plasma temperature at the target below 5 eV to ensure adequate lifetime. Therefore, it will be essential to achieve highly dissipative or detached divertor conditions for the control of heat flux and erosion in a fusion reactor. One of the most effective methods to promote the achievement of detachment is to improve neutral trapping and impurity screening in the divertor by changing the divertor structure [1]. Detailed analysis has been carried out by the SOLPS modeling to assess the relative importance of two key aspects of divertor-baffle geometry: (i) divertor closure, and (ii) field-target angle [2]; and a small-angle slot (SAS) divertor concept has been developed to validate the key features of a gas-tight slot divertor with optimized target shaping [3]. The initial tests of the SAS divertor in DIII-D experiment have shown that SAS can achieve a higher level of divertor performance than the open divertor, as predicted by SOLPS [4]. A correlation between T_{et} and n_{Dt2} for different geometries has been found, which indicates that the neutrals play a significant role in divertor energy dissipation processes [5].

All these studies highlight the importance of the divertor target shape and baffling on the plasma detachment. However, some critical questions still remain: (1) what is the range of input power that the current size tokamaks can operate leveraging the benefits from a closed divertor? (2) how much can we benefit from the divertor geometric effects with/without impurity-seeding for a fusion reactor, such as **C**hina **F**usion **E**ngineering **T**est **R**eactor (CFETR)? We have carried out a systematic analysis using SOLPS to address these points. For the current size tokamaks, direct comparison between the open and closed divertors is made to assess the effectiveness of divertor closure as a function of input power with and without impurity seeding. For CFETR, the SAS concept is evaluated under different input power conditions by comparing with a conventional horizontal target. Moreover, the potential advantages of employing SAS in CFETR with impurity-seeding and effective pumping are also discussed in this paper.

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First gas balance studies in Wendelstein 7-X operating with inertially cooled graphite divertor

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The advanced optimized stellarator Wendelstein 7-X (W7-X) is currently in its second operational phase, featuring an island divertor made from graphite. Plasma operation in this phase is done in either Hydrogen or Hydrogen/Helium after an intial phase of first wall conditioning [0]. With a divertor, plasma-wall interaction changes significantly compared to the first operation with a graphite limiter in the start-up phase of W7-X.
The first wall can act as a source (e.g. by recycling or outgassing) or sink (e.g. by co-deposition, or implantation) for fuel particles depending on the wall conditions, thus the actual competition of impinging and desorbing particles from the first wall. This has great impact on the global plasma particle balance. The global gas balance consists of all sources and sinks of particles.
We present a framework to assess all relevant terms of the fuel balance. Since the wall source is not directly accessible, only indirect determination by measurement or estimation of all other contributing terms is done. We will present the gas balance for different magnetic configurations and describe the influence of the wall source on density control for different plasma conditions, i.e. external fueling rates and heating power.
A major part of the gas balance is determined by the particle removal rate Q. Q is given by Q=p*S whereas p is the divertor pressure measured by a set of ASDEX pressure gauges at different locations in the sub-divertor region of W7-X [1], and S is the effective pumping speed.
These studies are of critical importance for density control already for the plasmas in OP1.2 but especially for the long-pulse discharges in the upcoming next operational phases with discharge lengths that eventually will exceed 1000 seconds.

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Development of Heating Scenario to Reduce the Impact of Bootstrap Currents in Wendelstein 7-X

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Wendelstein 7-X (W7-X) is a low-shear stellarator with 10 modular island divertor units designed for particle and heat exhaust. For the proper working of the island divertor concept, the edge magnetic field topology has to have a certain resonant structure, so-called magnetic islands, at the boundary of the plasma which lead the particle and power fluxes from the main plasma to the divertor targets designed for high power loads. The bootstrap current (BSC), which is driven by the plasma profiles, affects strongly the location of the edge islands thus changing their interaction with the divertor plates. Previous studies have predicted an overload problem due to the evolving toroidal net-current (consisting of the BSC and an exponentially decaying shielding current) near the pumping gap of the divertor where neutralized gas resulting from plasma divertor interaction should be able to escape the main vacuum vessel and should then be pumped out [1]. The overload occurs at the ends of the divertor targets where the maximum allowable heat loads are lower than in the centre of the targets. A new plasma-facing component referred to as scraper element (SE) has been proposed to protect the target ends intersecting the field lines which carry heat loads to the endangered locations. However, calculations predict a drop in pumping efficiency which thus may result in a degraded divertor performance [3].
Therefore, alternative experimental scenarios are investigated to avoid the necessity of the SEs. Since the overload situation occurs at full heating power at about half-way of the evolution of the net toroidal current, scenarios are explored where the critical range of toroidal current (around 22 kA) is reached and exceeded at reduced heating power levels.

For this study, the numerical tools used are the Variational Moments Equilibrium Code (VMEC) and the EXTENDER-code to calculate the magneto-hydrodynamic equilibrium field produced by the plasma and the external coils in the entire vacuum chamber. The plasma currents are calculated self-consistently by iterating between various codes until changes are negligible, as follows: the BSC is based on a VMEC-equilibrium calculation, from which transport coefficients are calculated with the Drift-Kinetic Equation Solver (DKES). These are used for transport simulations [2] that predict the plasma profiles, which then provide input for the next equilibrium calculation with VMEC. Based on the sequence of equilibrium fields from these self-consistent scenarios the compatibility of heat deposition with the capabilities of the divertor parts is examined.

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Current density features associated to Type-I and Type-III ELMs in COMPASS tokamak

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The presence of filamentary structures widely characterizes the edge region of fusion devices, independently from their magnetic configuration. These structures are generally revealed by pressure peak locally emerging on the cross-field plane from the plasma background and the name *filaments* is due to their extended size along the magnetic field line.

Filaments emerging from turbulence background share these general features with Edge Localized Modes (ELMs), which are responsible of a large fraction of transport towards the plasma wall and divertor plates. The filament electromagnetic features were experimentally studied in the recent years due to the relevance of this aspect on the transport, considering as an example the transition between closed and open magnetic field line topology, up to the possible magnetic field line bending effect in case of enough high current associated to ELM filaments, which could enhance their interaction with the first wall.

A recent preliminary analysis revealed the presence of parallel current in the COMPASS ELMs and evidenced their fragmented structure [1]. Aim of this contribution is to provide a direct experimental measurement of the current density associated to different type of filaments, such as type-I and type-III ELMs including also the inter-ELM filamentary structures in the Scrape-Off Layer (SOL) region of the COMPASS tokamak. Measurements were performed in D-shaped plasmas. Discharges were performed in both ohmic and NBI heated plasmas, achieving ELMy H-mode regimes characterized by different type of ELMs. The diagnostic equipment available allows a detailed evaluation of ELM following their development along the same magnetic field line. This study exploits in particular poloidal arrays of Langmuir probes and Ball Pen Probes [2, 3] in the divertor region, intercepting the same magnetic field line of a suitably inserted probe head, the Filamentary-probe [4]. This diagnostics provides the direct measurement of the current density associated to the filaments intercepting the probe, simultaneously to the other quantities, as their associated density, temperature and electric field fluctuations. Statistical evaluation, based on advanced statistical techniques, is applied to identify the most representative behavior of each filament type. A scan of the relative distance of the probe insertion is used to evaluate their radial evolution within the SOL region. Average properties of ELM and inter-ELM filaments will be investigated as comparing ohmic and NBI heated plasmas. The plasma density effect, which is expected to affect the filaments behavior [5], will be also investigated.

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Implications of recent and ongoing technological advances on the divertor performance requirements of fusion power devices

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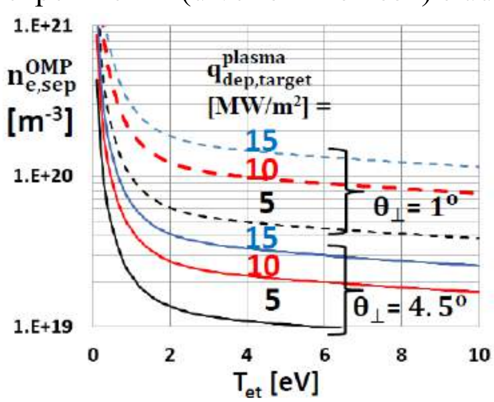
New analysis is presented of the divertor conditions required for fusion power reactors and recent technological advances are discussed that have potential importance for achieving these requirements. Reactors require values of $\bar{n}_e \approx n_{e,sep}^{OMP} \sim 10^{20}$ m^{-3} where $n_{e,sep}^{OMP}$ is the 'upstream' plasma density in the edge. A simple analytic expression is derived for $n_{e,sep}^{OMP} \propto \sim q_{dep,t}^{plasma} / \left(\sin\theta_\perp T_{et}^{1/2} [1 - f_{mom-loss}]_{T_{et}}\right)$, where $q_{dep,t}^{plasma}$ is the plasma power deposition density on the target, $\theta_\perp$ the angle between the target surface and **B**, and T_{et} the electron temperature at the target. $q_{dep,t}^{total}$ (total = plasma + rad'n load) will always be taken to the thermo-mechanical limit, presently ~10 MW/m^2 for W and CFC, to minimize the required edge radiation and the risk of degrading the confined plasma. It has been shown recently [1] that the momentum loss along the SOL is a function of T_{et} to a first approximation with $[1 - f_{mom-loss}]_{T_t} = 1.3\left(1 - e^{-T_{et}/1.8}\right)^{1.6}$ describing to ~ ± 2 the results for target/upstream pressure from 6 different code and an experimental (divertor Thomson) study. Thus $n_{e,sep}^{OMP}\left(q_{dep,t}^{plasma}, \theta_\perp, T_{et}\right)$, see figure.



The present sizes of the gaps and maximum misalignments in the modular ITER divertor require that $\theta_\perp$ be relatively large, ~ 4.5^o, compared to present tokamaks, $\theta_\perp$ ~ 1-3^o, to shadow-protect leading edges of the modules and monoblocks from damaging power loads. Thus $n_{e,sep}^{OMP} \sim 10^{20}$ m^{-3}, $\theta_\perp$ = 4.5^o and $q_{dep,t}^{plasma}$ = 5 MW/m^2 requires extremely strong detachment, T_{et}~ 0.5 eV. For smaller $\theta_\perp$ and/or larger $q_{dep,t}^{plasma}$ much less extreme values of T_{et} are required, e.g. ~ 4 eV for $\theta_\perp$= 1^o and $q_{dep,t}^{plasma}$= 10 MW/m^2. It is undesirable to resort to any stronger detachment than is necessary for target survival since this increases the risk of degrading the performance of the confined plasma, e.g. by X-pt MARFEs. The continuing rapid evolution of Advanced Manufacture (AM), robotics, and high temperature superconducting (HTS) magnets - each activity heavily funded by major non-fusion applications - is potentially transformative, requiring less extreme divertor detachment for fusion reactors: AM holds promise to increase $q_{dep,t}^{total}$ limits; advances in robotics could enable improved installation, maintenance and repair procedures for modular component devices like ITER, reducing $\theta_\perp$; HTS toroidal field coil magnets hold promise of being demountable (openable), making it possible to use *monolithic* rather than *modular* internal structure of the vessel with the pre-assembled, highly-aligned (i.e. small $\theta_\perp$) internal structure being lowered into place.

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Numerical study of turbulence in presence of an X-point with the flux-coordinate independent approach

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The complex geometry in the edge and scrape-off layer poses a challenge to simulation of turbulence in magnetically confined plasmas, since the usually employed field/flux-aligned coordinates become singular on the separatrix/X-point. In the flux-coordinate independent approach (FCI) [1,2] these coordinate singularities are avoided by using a cylindrical grid with discretization of parallel operators via field line tracing and interpolation (field line map). Within this framework an isothermal drift reduced Braginskii model [3] with Debye-Sheath boundary conditions at the target plates is implemented in the code GRILLIX [2,4,5]. The code was successfully verified with the method of manufactured solutions (MMS) and validated against TORPEX blob experiments [6]. The basic character of turbulence in presence of a separatrix is investigated with GRILLIX. It is found that fluctuations are driven mainly in the outboard midplane region where magnetic curvature is unfavourable, and due to magnetic shear the structures are distorted along magnetic field lines, especially strongly in the vicinity of the X-point. As such distorted structures are dissipated strongly the fluctuations die out towards the X-point, which tends to disconnect the low field side region from the stabilizing high field side region [7]. Furthermore, GRILLIX has been extended recently by a geometric multigrid solver, which allows relaxation of the routinely applied Boussinesq approximation. First results seem to indicate that the Boussinesq approximation has -at least up to moderate fluctuation levels- only minor effects [8]. An overview about ongoing effort concerning treatment of electromagnetic and thermal fluctuations within the FCI approach and application to 3D geometries [9] will be given.

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Integrated Edge-Core Plasma Modelling for the EU-DEMO

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The DEMO reactor aims at demonstrating the economically attractive feasibility of nuclear fusion by injecting for the first time fusion-produced electricity in the grid. A number of different projects are currently under consideration in different countries, among which EU is focusing on the so-called EU-DEMO, targeting at commercial grade electricity production by 2050. The experiment is now in the pre-conceptual design phase, where a number of critical decisions have to be made on how to solve critical issues like, among others, an optimal strategy to exhaust the ≥ 150 MW expected to enter the SOL crossing the separatrix. The power exhaust problem is worsened by the extremely large value of the $P_{sep}/R_0 \approx 17$ ratio. Such a value, larger than what characterizes any currently operating machine, or even ITER, essentially indicates that the target surface available for power exhaust will be strongly limited, which could lead to unbearably large peak heat flux values on the divertor target plates, if a proper mitigation strategy were not implemented.

Following the current baseline approach, most of the SOL entering power should be radiated before reaching the divertor plates through conduction/advection processes. This goal should be achieved by puffing a controlled amount of carefully selected impurity (or more likely a mixture of impurities) in the SOL, in order to spread as much as possible of the P_{sep} power onto a surface area as large as possible.

The optimal selection of the desired impurity mixture requires integrated consideration of divertor and core plasma physics. On the one hand, it should guarantee divertor operation to lie within acceptable operational limits (often placed at $T_e \leq 5$ eV and $q_{target} \leq 10$ MW/m^2 to avoid excessive W sputtering and allowing effective target cooling). On the other hand, we need to avoid excessive core contamination and maximize the energy gain to demonstrate economical attractivity of fusion. To date, some analysis has been performed focusing on divertor compatibility conditions with the SOLPS code [1], or more integrated estimates have been performed, they involved only preliminary edge plasma models [2].

We present an integrated modelling of the DEMO edge and core, obtained by coupling the SOLPS and ASTRA codes at the separatrix surface, where average fluxes of particles, energy, temperature and densities are matched iteratively. We consider several impurity mixes, comprising Xe, Ar, and Kr. We conclude that a suitable impurity mix is likely to be found, in which Ar should guarantee a sufficient radiation level from the SOL, still guaranteeing acceptable fusion performances

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Impact of self-consistent neutrals dynamics and particle sources on edge plasma transport and turbulence in 3D first principle simulations

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In tokamaks, heat and particle exhaust as well as confinement depend on the interplay of multi-physics phenomena occurring in the boundary of the plasma. A comprehensive modelling of the physics at play should involve a consistent description of plasma transport - including turbulence -, plasma-wall interaction, atomic and molecular physics, all treated in realistic magnetic and wall geometries. Due to the complexity of such modelling, the state of the art has for long been compartmentalized between mean-field codes, lacking a self-consistent description of transverse transport, and turbulence codes, ignoring neutrals physics and most often in simplified geometries. In the latter case, turbulence simulations have to be driven by an incoming flux of heat and particles arbitrarily imposed by the user, usually located at the inner boundary of the simulation domain.

In this contribution, we report the results of first principle simulations including self-consistently turbulence and neutrals physics. The TOKAM3X-EIRENE code package is used for that purpose, combining a fluid-drift description of the plasma and a kinetic description of neutrals. We analyse simulations run in an idealized circular limited geometry before turning to realistic diverted cases. In both cases, a comparison is made to simulations flux-driven from the inner boundary of the simulation domain. We show that the change of the location of the particle source from the core to the separatrix in the limiter case and to the target plates in the divertor case impacts both the equilibrium profiles and turbulence. In the closed field lines region, the need to evacuate power without an associated particle flux leads a flattening of density gradient and a change of turbulence nature from RBM-like to ITG-like. The ionization source due to recycling neutrals is found to fluctuate due to the interaction of the neutrals cloud with hot and dense turbulence filaments. The impact of these fluctuations on the non-linearities of atomic and molecular terms is evaluated. A density scan is also performed to show the dependence of these effects with scrape-off layer density regimes.

Analysis of indefinite multi-peak divertor footprint with proper orthogonal decomposition

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This study presents a characterization of the indefinite divertor footprint profile with multiple peaks in the Large Helical Device (LHD) by using the proper orthogonal decomposition (POD) technique. Moreover, we investigate the upstream parameter dependences in hydrogen and deuterium plasma discharges. Application of the POD is effective for not only heliotron but also tokamak devices with the resonant magnetic perturbation (RMP) field.

Clarification of mechanism determining the divertor footprint profile is an essential issue to predict accurate divertor heat and particle loads in future fusion machines. For this aim, in tokamak devices, characteristic parameters which represent the divertor footprint are investigated by using a fitting with an expression derived from the convolution of an exponential decay and a Gaussian function [1]. In heliotron/stellarator devices, however, similar approach is difficult, because there is complex three-dimensional (3D) magnetic structure in front of the divertor plate and the divertor footprint often has multiple peaks across the strike point. The same situation is also observed in tokamaks with the RMP; 3D robe structures contact the divertor plate, and thus a strike point splits. Therefore, establishment of characterizing procedure of such indefinite profiles is desired.

In the recent study, we employed the multivariable data analysis technique, POD, for characterizing the varying multi-peak footprint profile in the LHD hydrogen plasmas [2]. We analyzed a number of ion saturation current signals measured by the toroidal divertor probe arrays [3] in several hundred discharges; as a result, the particle flux profile can be characterized by the ratio between coefficients of two decomposed dominant bases. Furthermore, it was found that the ratio strongly correlates with the electron temperature and the density inside the last closed flux surface.

In this study, we apply the above-mentioned POD analysis to the divertor signals acquired in 19 cycle experimental campaign, which consists of hydrogen and deuterium plasma experiments in LHD. We will present the differences of orthogonal bases in between the hydrogen and deuterium plasmas, and discuss their dependences on upstream plasma parameters.

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Analysis of Detachment State using a Lagrange Scheme

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For future fusion reactors, it is necessary to maintain plasma detachment states in front of the divertor plates. Even though the basic physics of detached plasmas has been qualitatively understood, it remains an issue to reproduce the detachment state with a three-dimensional plasma fluid code.

One possible reason for the difficulty of simulating detachment scenarios is that the transport in detached plasmas tends to be dominated by convection, because of lower temperatures resulting in strongly reduced heat conduction. For this reason, we are developing a transport code using a Lagrange scheme which enables us to solve pure-convective problems by its semi-implicit treatment of the pressure gradient term. The Lagrange scheme uses pseudo fluid particles moving along the Lagrange coordinates. Therefore, this scheme is also well suited for complex geometry of the machines, e.g., non-orthogonal walls to the magnetic field.

As a first step, one-dimensional test-calculations have been done. To validate the scheme, benchmarks with a code using the Finite Volume scheme were done for a coupled system of continuity, momentum, and energy equations. Both codes agree quite well in most of the calculation domain though there are small differences close to the boundaries.

As a next step source terms are introduced, i.e., localized particle source by ionization, momentum loss by charge exchange, heat radiation, particle loss by recombination. This will enable us to approach scenarios with detached conditions. In the conference, we will present results for different operating conditions, representing both attached and detached states.

Reaction processes of molecular activated recombination leading to detachment of divertor simulation plasma in GAMMA 10/PDX

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Plasma detachment at the divertor is considered to be the most promising solution to reduce heat load on a divertor plate. The MAR (Molecular activated recombination) processes for the plasma detachment have been studied theoretically and experimentally in linear machines and tokamaks. In GAMMA 10/PDX, a divertor simulation experimental module (D-module) has been installed in the west-end region to study divertor plasma phenomena [1]. The plasma detached due to MAR when the additional gas was supplied to the divertor simulation plasma [2]. Interestingly, H_γ and H_δ emissions suddenly appeared when the hydrogen gas pressure in the D-module became more than 10 Pa. Reaction processes of the observed MAR phenomena will be discussed.

In this experiment, hydrogen gas was injected into the D-module to obtain dependence of plasma parameters such as electron density, electron temperature, Balmer line intensity and Fulcher α band spectrum on pressure in the D-module. In addition, the vibrational temperature of hydrogen molecules was obtained using coronal model [3] of Fulcher α spectrum intensity. Neutral gas pressure in the D-module increased up to ~20 Pa. The electron temperature decreased from ~25 eV to ~1 eV with increase in pressure and clear density rollover was observed at ~1.5 Pa and then electron density decreased down to ~2×10^{16} /m^3. After the density rollover, H_β intensity decreased as with the electron density but H_α intensity continued to increase with increase in pressure up to ~10 Pa. The vibrational temperature of hydrogen molecule increased from ~4000 K to ~9000 K with increase in pressure up to ~2 Pa and then decreased down to 3000 K. The observation of increasing of the vibrational temperature would be attributed to the production of vibrationally excited molecules through a reaction of $e + H_3^+ \rightarrow H_2(v) + H$. When the hydrogen gas pressure became more than 10 Pa, H_α intensity decreased and H_β intensity increased with increase in the neutral pressure and H_γ, H_δ emissions were suddenly observed and it increased as the electron temperature decreased. The sudden emissions could be attributed to that the reaction of $e + H_3^+ \rightarrow 3H$ become more dominant than that of $e + H_3^+ \rightarrow H_2(v) + H$ when electron temperature is below ~1eV. Quantitative discussion will be done by using a numerical simulation of CR-model including MAR process.

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Inter ELM coherent fluctuations in divertor Langmuir probe ion saturation current in KSTAR tokamak

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In the tokamak edge plasmas, coherent fluctuations regulating the transport has one important application, namely, significant reduction in the magnitudes of heat and particle flux loads on divertors, associated with edge localized modes (ELMs) [1]. In the present work, we report the experimental observation of coherent fluctuations in the ion saturation current on the outer divertor Langmuir probes, concurrently with density fluctuations and magnetic fluctuations in the inter ELM phase on KSTAR tokamak; density fluctuations in the mid-plane are observed by beam emission spectroscopy (BES) and the magnetic fluctuations through Mirnov coils. The observations were made in the high confinement (H-) mode deuterium plasmas, with lower single null and type I ELMs; other typical parameters: toroidal plasma current ~0.5 MA, magnetic field ~2.5 T, neutral beam heating power ~2.8 MW. These coherent fluctuations (frequency ~10 kHz) occur either continuously for tens of milliseconds or as short bursts for about one millisecond typically. The Langmuir probes near the strike point remain turbulent in spectrum whereas the probes slightly far in the outboard scrape off layer (SOL) show the coherent fluctuations. In the case of short coherent bursts, the mean ion saturation current on Langmuir probes near the strike point, on both the outer and inner divertors, decreases up to 30% and 50% respectively, whereas either a small or no significant change is found for the farther Langmuir probes on the outer divertor with coherent fluctuations. The coherent fluctuations on the outer divertor Langmuir probes are found for both upward and downward moving density modes in BES view, moreover with similar frequency as in the mid-plane; hence this rules out the possibility that observed coherent fluctuations at divertor Langmuir probes could be due to the structures themselves escaping into the SOL [2]. Preliminary interpretations indicate the substantial enlargement of the density structures in BES view in mid-plane may lead consequently to breaking of the structures; the broken structures possibly escape into SOL reaching divertor Langmuir probes. Additional factors such as the position and electromagnetic nature of the density structures in the plasma edge affecting the observed coherent fluctuations in divertor Langmuir probe current shall be investigated in future.[1]

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Effect of diverging magnetic fields on plasma profiles in super-X divertors considering the anisotropy of ion temperature

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A super-X divertor (SXD) is expected to drastically reduce the divertor heat load by a large flux-tube expansion [1] and its plasma characteristics have been widely studied by using edge plasma codes, e.g. SOLPS and EDGE2D. Plasma fluid equations in these codes are based on the Braginskii equations where a viscous-flux (VF) model is adopted [2]. Recently, we developed a plasma fluid model directly introducing anisotropic ion temperatures (AIT), $T_{i,//}$ and $T_{i,\perp}$, and found that the VF model in the Braginskii equations becomes remarkably invalid in diverging magnetic fields affecting parallel flow velocity V profiles [3, 4].

The VF model treats isotropic and anisotropic parts of ion pressure, $p_i \equiv (p_{i,//} + 2p_{i,\perp})/3$ and $\delta p_i \equiv 2(p_{i,//} - p_{i,\perp})/3$, instead of parallel and perpendicular ion pressures, $p_{i,//} = nT_{i,//}$ and $p_{i,\perp} = nT_{i,\perp}$. The δp_i is approximated by the viscous flux as $\delta p_i \approx \pi_i = -\eta_{//} \nabla_{//} V - \eta_{//}(V/2B) \nabla_{//} B$ where $\eta_{//}$ denotes the ion parallel viscosity. In diverging magnetic field regions ($V\nabla_{//}B < 0$), the ion viscous flux π_i can be positive. On the other hand, in an open-field system such as SOL/divertor plasmas, δp_i tends to be negative (i.e. $T_{i,//} < T_{i,\perp}$) [3, 5]. This is because the parallel energy component is lost faster than the perpendicular one by the parallel convection. Therefore, the VF model can be invalid in diverging magnetic fields towards the divertor plate, and direct introduction of AIT enables us to overcome this issue.

In this paper, therefore, we apply the above plasma fluid model introducing $T_{i,//}$ and $T_{i,\perp}$ to the SXD plasma simulation, and study the effect of diverging magnetic fields on plasma profiles. At first, we compare the above fluid code of AIT and the B2 code of VF model for simple-mirror configurations. In low-collisional regime, deviations by a factor of ~ 2 are observed in the V profiles from these two codes in the diverging magnetic field regions while good agreements are obtained in high-collisional one. For the diverging magnetic field configuration, the AIT fluid model gives a smooth profile of Mach number including subsonic-supersonic transition while the VF model shows an unnatural and unsmooth profile of Mach number influenced by the sheath boundary condition. In the presentation, we will compare and discuss in detail the results of SXD simulations using these two codes.

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AMMONX: a kinetic ammonia production scheme for EIRENE implementation

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The design of the ITER divertor and estimates of the required fuel throughput have relied for many years on simulations performed with the SOLPS plasma edge modeling suite, a new version of which, SOLPS-ITER, containing a more complete physics model, was launched in 2015 by the ITER Organization. The code uses the Monte Carlo neutral kinetic code EIRENE (www.eirene.de) as its main workhorse for solving the transport equations related to neutral atomic and molecular species, as well as for radicals and molecular ions.

High performance plasma operation in ITER will require the routine injection of an extrinsic low atomic number impurity to provide cooling of the divertor plasma via radiative dissipation. The two principal candidate gases are neon and the molecular gas nitrogen N_2. However, N_2 into hydrogenic plasmas is known to lead to the formation of ammonia, NH_3 [1], which, in the active operation phases would be tritiated. The formation of large quantities of tritiated ammonia has consequences for several aspects of the ITER plant operation in terms of tritium retention, gas reprocessing and duty cycle. It is therefore important to assess early on what fraction of ammonia may be found in the ITER gas exhaust when N_2 seeding is used.

Until recently the EIRENE suite only handled hydrogen chemistry (i.e. H atoms, H_2 molecules, and H_2 molecular ions and their isotopomers). Extensions also existed for hydrocarbon chemistry, including C_xH_y species (H is any hydrogen isotope), but no kinetic scheme including NH_x chemistry had been implemented. Such a scheme, presented here, has now been developed as a result of a recent collaboration between the LSPM-CNRS laboratory in Villetaneuse and the ITER Organization. It includes 50 reactive processes, implying a large set of N-bearing species (N, N_2, N_2^+, NH_x radicals and ions), such ionization, electron-molecular ion recombination, electron and ion impact dissociation etc. This mechanism is a part of a more complete scheme of 150 processes also including the metastable states of N_2 and N. The reduction from 150 to 50 processes has been obtained by comparing the reactive collision frequency for each species, and their individual diffusion times to the wall under the ITER divertor pressure conditions. The reaction rates for the different kinetic processes are obtained from the published literature, of which an extensive survey has been conducted during this work, and fitted with an Arrhenius type formalism (e.g. $k(T_e) = AT_e^n\exp(-E_a/T_e)$ with T_e the plasma electron temperature and E_a the activation energy).

The first modeling results obtained with this reduced kinetic scheme added to the EIRENE code database are presented here (the new database is called AMMONX). They show that the homogeneous chemistry processes in the plasma phase, even if they predict the formation of NH_3, cannot fully explain by themselves the levels observed in laboratory experiments. Hence a first reaction scheme for the formation of NH_3 on the walls is currently being validated on simplified 0D models, before a future implementation in the EIRENE code.

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The kinetic SOL: Understanding the sheath physics in tokamaks - progress in PIC modelling

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Kinetic modelling of the SOL plasma in fusion devices has more than three decades long history starting from simplified Particle-In Cell (PIC) models of the divertor plasma [1, 2] and SOL [3, 4] till our day massively parallel simulations of complex SOL geometry [5, 6]. PIC simulations, which can correctly describe the sheath in front of a divertor plate, require finest resolution in time and space and are computationally very expensive. Therefore, the PIC models of the SOL are continuously updated by inclusion of new sophisticated processes and optimizing of numerical schemes.

The aim of this work is to describe PIC technique for SOL simulations and present the recent results of the corresponding modelling. The atomic and molecular reactions play important roles in the SOL, e.g., recycling, radiative cooling, detachment etc., and the modelling for these reactions have been developed by introducing various Monte-Carlo techniques. These simulations including A&M processes have indicated that the parallel transport, as well as the cross-field drift transport, can significantly deviate from the classical one. Two main reasons for these deviations are: i) the presence of a super-thermal non-Maxwellian particle population carrying significant amount of the energy flux, and ii) strong cross-field gradients in the SOL, when the gyro-averaging approximation fails. These effects strongly influence divertor sheath physics too, e.g., the plasma and potential profiles are not necessarily monotonic in the pre-sheath and sheath, as it is usually assumed, and the normalized heat fluxes to the divertor plates (so called sheath heat transmission factors) can exceed the classical values by a factor of magnitude. PIC simulations have demonstrated that not only the stationary SOL but also the transient behaviors of the sheath and the SOL transport are affected strongly by the kinetic effects.

We quantify these effects and identify which of them have to be taken into account for studying of the plasma and impurity transport in the SOL. PIC results are compared with simplified analytical models and possible ways of their implementation into the analytical/numerical models of the SOL with lower computational costs are described too.

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Two-phases hybrid model for neutrals

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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 usually 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, Kn, which measures how "kinetic" the neutrals behave) is much larger than one. However, in the divertor region close to the neutralizer target plates 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, especially for large machines as ITER or DEMO. In these regions i) the kinetic description is too detailed (locally Kn << 1) because neutrals are quasi-maxwellian and ii) the Monte Carlo approach is very inefficient because neutrals undergo many collisions (charge exchange, elastic collisions) before being ionized or leaving the highly collisional region.
A hybrid kinetic/fluid model then becomes appealing for the neutral gas. Several types of hybrid models exist, and in this contribution we will focus on a two phases model [3], in which the two phases, fully fluid neutrals and fully kinetic neutrals, coexist in the whole domain. Additional processes connecting the two phases are introduced, mimicking evaporation and condensation reactions. The rate coefficients for these processes are calculated from the background plasma, in such a way that kinetic neutrals entering a highly collisional region condensate into the fluid phase after a few collisions. This entails running the kinetic code, Eirene, at a potentially much lower cost, together with a fluid code, here the one presented in [4].
In this contribution, both the computational efficiency and accuracy of this hybrid model are discussed on various types of plasmas, in realistic ITER geometry. The optimal values for the evaporation and condensation rate coefficients are also determined.

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Sensitivity of coupled plasma fluid-neutral kinetic edge simulations to the magnetized plasma sheath model

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Plasma edge modelling is a key to address challenges for next step devices (ITER), e.g. power exhaust. Transport codes, such as SOLPS or Soledge2D-EIRENE, 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 sheath is at the interface between the two models, and requires careful attention. For instance, the energy flux at the sheath entrance must be consistent in the fluid and the kinetic domains. If not, systematic errors are introduced in the energy transfers between electrons and ions in the sheath, hence on the energy of backscattered atoms. Ensuring consistency requires careful analysis of the assumptions made on the ion velocity distribution at the sheath entrance f_i, as well as of the consequences of these assumptions (e.g. value of the ion heat sheath transmission coefficient $\gamma_{i,}$ which is related to moments of f_i). Transport codes generally model f_i as a shifted Maxwellian with isotropic temperature T_i. Consistency conditions are discussed in this case, with particular emphasis on ion temperature anisotropy at the sheath entrance, related to the pre-sheath acceleration ($T_{i//}<T_{i\perp}$). Such an anisotropy is present in the fluid model when parallel viscosity is accounted for. Moreover, kinetic modelling shows that the parallel ion distribution is in fact far from a shifted Maxwellian (*e.g.* [1] and references therein), as shown for example in the improved model we previously developed to tabulate energy and angle distributions of ions on the wall [2]. The model, currently implemented in Soledge2D-EIRENE, is based on 1D-3V PIC code to treat sheath physics [3]. It uses as an input an ion distribution function f_i^K at the sheath entrance calculated with a SOL kinetic model. Consistency conditions for the energy flux are established for f_i^K. The two major differences between this model and the original Maxwellian one are i) the ion distribution at the sheath entrance and ii) the dynamics in the sheath, much more detailed in the 1D-3V approach. The latter model shows that the ions incidence angles are markedly shallower than what is expected from simplified treatments of sheath dynamics, the key player being the gyromotion. This has consequences both on energy recycling coefficients and on the subsequent neutral particle dynamics. We show that the two different sheath models can make a factor of 2 differences on the ion temperature in an extended region of the SOL, here on WEST cases. These effects both result from large differences in the incidence angles distributions, and from the heat sheath transmission factor γ_I, which are different in the two models. 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. This work highlights the remaining modelling uncertainties related to the description of the sheath, which have substantial effects on the modelling results

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Particle and heat spreading in the tokamak divertor via turbulent mixing

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A firm grasp of transport processes perpendicular to the magnetic field line is required in order to understand and predict heat and particle fluxes to divertor components in tokamaks. Perpendicular transport from the hot core into the scrape-off layer (SOL) upstream impacts the SOL profile at the divertor via parallel streaming along the magnetic field line. Downstream, below the X-point, perpendicular transport can spread heat and particles from the SOL into the private-flux region (PFR) thereby broadening the deposition profile on the divertor. A significant component of the upstream perpendicular transport is likely due to intermittent ejection of filamentary objects [1] which are under active study. Downstream perpendicular transport is less precisely understood, though recent camera measurements in MAST have demonstrated the presence of complex filamentary transport in the divertor also [2]. For many plasma processes in the divertor, localised turbulent transport is likely to be important yet it remains poorly understood. This paper focusses on the nature of downstream divertor localised turbulent transport through numerical simulation.

Simulations have been performed using the STORM [3] module of BOUT++ [4] in a slab geometry to mimic features of the tokamak divertor. Heat and particles are fed into the domain from an upstream source and sheath boundary conditions represent the interface with the divertor target. An axisymmetric laminar background is first evolved, within which a strong radial electric field arises on the separatrix where a rapid radial variation in the electron temperature leads to a differential biasing by the sheath potential, forming a transverse shear-layer. These conditions are shown to be strongly unstable to the Kelvin-Helmholtz instability [5] which drives a turbulent mixing layer in the vicinity of the separatrix position, drawing free energy from the gradient in the transverse ExB flow. An effective gravity is introduced as a proxy for magnetic curvature which narrows the mixing layer by forcing perturbations back towards the separatrix. Consequently, the orientation of the effective gravity, and by implication the orientation of the divertor leg with respect to the major radius, effects the turbulent mixing observed. The upstream density and heat flow into the domain are varied and the resultant change to the mixing layer will be analysed, particularly with respect to the target electron temperature; a key parameter identified in experimental scalings [6]. Finally the possible impact of a more realistic magnetic geometry will be discussed.

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Effects of divertor geometry on H-mode pedestal structure near divertor detachment in the DIII-D tokamak

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Dedicated experiments performed in DIII-D have found that divertor geometry significantly affects the H-mode pedestal profiles, especially near divertor detachment. Consistent with SOLPS modeling, the more closed divertor, i.e. both the upper-ceiling divertor with dome-baffle structure and the Small-Angle-Slot (SAS) divertor, tends to facilitate the achievement of divertor detachment at lower upstream density and higher pedestal temperature than an open (flat plate) divertor [1]. In both attached and detached plasmas, the more closed divertor results in the lower density and higher temperature at the top of the pedestal, due to lower pedestal fueling based on SOLPS and OEDGE analysis.

This different pedestal fueling and neutral distribution resulting from divertor closure could significantly displace the density pedestal from the temperature pedestal. For the closed divertor, increasing gas puffing and injected heating power gradually shifts the density pedestal profile radially outward away from the temperature pedestal profile. Such relative shift can reach up to 50% of the pedestal width in detached plasmas, which may suggest decoupling between the particle and thermal transport. However, the open divertor density and temperature pedestal profiles are aligned in both attached and detached plasmas.

Approaching detachment, the different divertor closure could deviate the pedestal width away from the empirical and theoretical $\beta_{p,ped}$ scaling [2]. During the attachment phase the pedestal width agrees with the scaling for all the divertor configurations. During the detachment phase, the pedestal width is strongly (30%) reduced for the open divertor. In contrast, for the closed divertors, the pedestal is significantly wider by up to 50% than the scaling. With such wider pedestal and lower pedestal fueling, the divertor detachment can be achieved while retaining high pedestal performance for the closed divertor.

In particular, it was found that high confinement is maintained with divertor detachment using SAS divertor in DIII-D, which features a gas-tight slot structure and an ITER-like slant target configuration. The SAS exhibits a significantly (~30%) higher confinement than the open divertor, and maintains good confinement while the divertor is detached throughout the density ramp during the discharge. In contrast, the open divertor exhibits further confinement degradation when the divertor starts to detach, with a pronounced drop right after the rollover of the ion saturation current measured by the target-embedded Langmuir probes.

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Particle and Power Exhaust for H-mode Operation over 100 Seconds with ITER-like Tungsten Divertor in EAST

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**See appendix of B. N. Wan et al., “Overview of EAST Experiments on the Development of High-Performance Steady-State Scenario”, Nuclear Fusion 57, 102019 (2017)

A world record long pulse H-mode operation of 101.2 seconds with H_98=1.1 and a total power injection of 0.3 GJ has been successfully achieved in the EAST tokamak with ITER-like top tungsten (W) divertor, which has steady-state power exhaust capability of 10 MWm^{-2}. The peak temperature of W target saturated at t = 12 s to the value T≈500 ºC and a heat flux ≈3 MWm^{-2} was maintained stably. Great efforts to reduce heat flux and accommodate particle/impurity exhaust simultaneously have been made towards long pulse of 10^2 s time scale. By exploiting the observation of Pfirsch–Schlüter flow direction in the SOL, the Bt direction with Bx∇B away from the W divertor (more particles favor outer target in USN) was adopted along with optimizing the strike point location near the pumping slot, to facilitate particle and impurity exhaust with the top cryo-pump. By tailoring the 3D divertor footprint through edge magnetic topology change, the heat load was dispersed widely and thus peak heat flux and W sputtering was controlled consequently. A fully non-inductive H-mode regime of small ELMs to minimize transient divertor heat load at low pedestal collisionality was developed. In addition, the control of target W sputtering and impurity screening in the divertor region was explored. Extensive lithium coating was employed to lower edge recycling, low-Z impurity content and W sputtering. ECH, high-frequency ELMs and RMP are effectively used for core high-Z impurity expelling.

The upgrade plan and status of EAST bottom divertor from graphite into W to accommodate more challenging particle and power exhaust for steady-state H-mode over 400 s and L-mode operation over 1000 s will also be presented.

Application of SOLPS-ITER edge plasma model to interpret seeded and unseeded JET H-mode discharges with metallic wall

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The prediction of particle and power exhaust for ITER relevant conditions (high-power H-mode, high-density with a partially detached metal-wall divertor) requires validated edge plasma models [1]. The SOLPS-ITER code [2,3] has been selected by the IO as the design tool to predict ITER divertor conditions. SOLPS-ITER includes a state-of-the-art transport model including plasma drifts and improved molecular kinetics, i.e. model features required to quantify a detached divertor operational regime. Over the past few years, 2D edge plasma codes have shown to be incrementally reliable especially for the case of metal devices [4]. With the only exception of [5] the new SOLPS-ITER code is still required to demonstrate that it can reproduce H-mode edge plasma conditions in increased size metal divertors.

A set of JET H-mode discharges with the ITER-like wall (ILW, tungsten PFCs in the divertor, beryllium for the main-chamber) has been selected for modelling the inter-ELM phase. In a step-wise approach a first discharge comprises of an unseeded H-mode diverted plasma in attached conditions in semi-horizontal configuration (Ip/Bt=2.0MA/2.4T, P_{NBI}=11MW, gas-injection into divertor, c.f. [6]) for which a post-mortem analysis of the PFCs exists. The well diagnosed edge plasma conditions are currently utilized for the analysis of Be/W transport in JET using the ERO2.0 and WallDYN codes requiring modeled background plasmas in 2D.

The second discharge is a nitrogen seeded H-mode discharge at higher power (Ip/Bt=2.5MA/2.7T, P_{NBI}=20MW) in vertical target configuration and in partially detached conditions at medium radiated fraction f_{rad}~45% (c.f. [8]). The results of the numerical analysis of this discharge are taken as a basis to validate further SOLPS-ITER for extrapolation towards a credible power exhaust scenario in ITER with a metal wall. It is part of an ongoing size-scaling experimental similarity study on power exhaust between JET and Asdex-Upgrade and in the context of a wider SOLPS-ITER modelling activity including the ITER scale within the ITER Scientist Fellow Network.

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Studies of impurity source location on edge impurity transport in EAST with EMC3-EIRENE modelling

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The transport property of impurities in the edge plasma is one of the most critical issues for the operation and performance of the fusion devices. The radiation of impurities during penetration into the confinement region of the plasma would lead to the degradation of energy confinement and even the termination of plasma discharge, which imposes constraints on routine operation of fusion facilities. However, on the positive side, a good control of the edge impurity radiation would lead to a strong reduction of power load on divertor target plates. Therefore, studies of edge impurity transport are important to obtain a better understanding of the underlying mechanisms of impurity screening and detached divertor plasma.

The studies of the transport of the edge impurity on EAST tokamak have been carried out with the EMC3-EIRENE code [1-5]. The impurities of the C^{1+} - C^{3+} are mainly distributed at the private region and X-point domain. However, for the higher charge states, the distributions of the C^{4+} - C^{6+} are mainly at the upstream domain due to the high ionization potential. A detailed analysis of the influence of the position of the impurity source on the distribution of the edge impurity for different charge states is performed by artificially switching on/off the divertor target erosion in EMC3- EIRENE code. It is found that the distributions of the C^{1+} - C^{3+} ions are strongly related to the location of the impurity source, while the distributions of the C^{4+} - C^{6+} ions are independent on the position of the impurity source.

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OEDGE-TRIM.SP simulations of inter- and intra-ELM tungsten erosion during DIII-D H-mode discharges

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To study the relationship between tungsten erosion, the D plasma background, and the inter-ELM impurity (C or W) distribution, sequential OEDGE and TRIM.SP simulations are performed for different DIII-D H-mode discharges. Shots with varying type I ELM sizes ($\Delta W/W$ changing from 2% to 12%) and ELM frequencies (changing from 10Hz to 200Hz) are chosen from both the DIII-D Metal Rings Campaign and dedicated experiments with W-coated samples using the Divertor Materials Evaluation System (DiMES). These results suggest that the inter-ELM tungsten erosion is dominated by C impurity flux to the W surface, but the tungsten self-sputtering can contribute up to 20 percent, consistent with previous studies [1]. The intra-ELM tungsten erosion may still be dominated by C impurities for typical DIII-D type I ELM, but the contribution of D becomes more important, as observed in other devices [2]. The inferred intra-ELM target C flux fraction varies from 1% to 3% and is generally similar to the inter-ELM C flux fraction.

The OEDGE code is employed to provide the inter-ELM impurity distribution and tungsten gross erosion rates. Then the TRIM.SP sputtering code is used to simulate the intra-ELM tungsten gross erosion profiles. For the intra-ELM simulation, the charge states of C are determined by a comparison of the C transport time from the pedestal to the target with the high charge states C recombination time during ELMs. The C fractions as well as surface C concentrations were treated as adjustable parameters to replicate the measured W erosion profiles for each case. These parameter values are compared to the inter-ELM C distribution to help understand where these C impurities originate, thus to further understand the tungsten erosion process during ELMs. Result shows that the erosion contribution of the high charge states C transported out from the pedestal is strongly related to the ELM size. Comparisons between local DiMES sample experiments and global tungsten ring experiments will also be performed to evaluate the role that tungsten that is transported out of the pedestal plays in the intra-ELM tungsten erosion process.

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Modeling tokamak boundary plasma turbulence and understanding its role in setting divertor heat flux widths*

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The BOUT++ code has been used to simulate edge plasma electromagnetic (EM) turbulence and transport, including quasi-coherent modes (QCMs), and to study the role of EM turbulence in setting the scrape-off layer (SOL) heat flux width (lambda_q) and its scaling with machine parameters. An important goal of this research is to develop a first-principles model that can reproduce the observed inverse current I_p scaling of lambda_q seen in the international tokamak database and that can project the SOL heat flux width for future machines. More than a dozen tokamak discharges from C-Mod, DIII-D, EAST, ITER and CFETR have been simulated with encouraging success. The plasma profiles inside the separatrix of these discharges used in simulations are taken from fits of a modified tanh function to real experimental data, mapped onto a radial coordinate of normalized poloidal flux for C-Mod, DIII-D, and EAST. The plasma profiles inside the separatrix of ITER and CFETR are taken from feasible burning plasma operation scenarios using CORSICA [1] and ONETWO codes.

For the C-Mod enhanced D_α (EDA) H-mode discharges, BOUT++ six-field two-fluid nonlinear simulations show a reasonable agreement of upstream turbulence characteristics and divertor target heat flux behaviour [2,3]: (a) The simulated quasi-coherent modes (QCMs) show consistent characteristics of the frequency vs poloidal wave number spectra of the EM fluctuations when compared with experimental measurements – frequencies are around 60-120 kHz and k_θ is around 2.0 cm^{-1} which are comparable to the Phase Contrast Imaging measurements; (b) The location of the QCMs is generally consistent with experiment; (c) the simulations yield similar lambda_q to experimental measurements within a factor of 2. The BOUT++ simulations have also been performed for inter-ELM periods of DIII-D and EAST discharges, similar quasi-coherent modes have been found in these discharges. The parallel electron heat fluxes onto the target from the BOUT++ simulations of C-Mod, DIII-D, and EAST follow the experimental heat flux width scaling of the inverse dependence on the poloidal magnetic field with an outlier [4,5]. Further turbulence statistics analysis shows that the blobs are generated near the pedestal pressure peak gradient region inside the separatrix and contribute to the transport of the particle and heat in the SOL region.

To generalize the Goldston heuristic drift-based (HD) model[6], BOUT++ transport model has been developed for transport simulations with the electric and magnetic drifts and with the sheath potential in the SOL. Transport coefficients are calculated from the experimental profiles inside separatrix, then extending to the SOL. Steady state solution of heat flux show: (1) a similar scaling to the Goldston's HD model; (2) The amplitude of the simulated heat fluxes is within a factor of 2 compared to the experiment data; (3) The ExB drift reduces the heat flux width by 30%. The results of the SOL heat flux width for ITER and CFETR will be presented from both BOUT++ turbulence and transport simulations.

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Rayleigh-Taylor instability of a liquid metal film with magnetohydrodynamic effects

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A flowing liquid metal film has been considered an economical material for tokamak divertors and protecting the walls from heat flux and radiation. However, a heavy liquid film imposed on the inner surface of a tokamak may be subject to the Rayleigh-Taylor instability and drip into the plasma core. With this motivation, we use linear stability analysis to investigate the Rayleigh-Taylor instability of an incompressible viscous liquid metal film in the presence of a transverse magnetic field, which is perpendicular to the gravity field. Previous studies [1,2] neglect both the viscosity and the finite thickness of the film. In this report, criteria on the instability are analyzed and the magnetic field is found to be mainly stabilizing the flow in a way similar to surface tension.

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Stationary ELM-free H mode and edge coherent mode performance in EAST

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The erosion of the inner wall materials of the fusion device due to impulsive heat flux from the repetitive ELMs is a critical issue for magnetic fusion research. The size of ELMs must therefore be tightly controlled in future large devices. This talk will more focus on the ELM-free H-mode characteristics which obtains in EAST. And three different stationary ELM-free H-mode regimes will be presented.

The first is an enhanced-recycling H-mode regime [1, 2], appearing at relatively high pedestal collisionality ($\nu_e^* > 1$) with RF-dominated heating (mainly LHCD), characterized by an enhanced divertor Dα emission and a high-n electrostatic Edge Coherent Mode (ECM) driving significant particle and heat transport in the pedestal steep-gradient region [3]. The ECM and the regime disappear at high heating power and low pedestal collisionality ($\nu_e^* < 1$).

The second is a low-recycling H-mode regime, achieved with RF-dominated heating (mainly LHCD) and extensive lithium wall coating (30 g for each time, usually twice) or during lithium power injection. Additional power (> 0.5 MW) from co-NBI will usually bring ELMs back. Counter-NBI can facilitate the access to this regime. Access to this ELM-free regime exhibits a clear density threshold and the density threshold increases with plasma current i.e., $n_{el} \geq 3\times10^{19}\ m^{-3}$ at I_P = 450 kA and $n_{el} > 3.5\times10^{19}\ m^{-3}$, at I_P = 500 kA, where n_{el} is the line-averaged density. The ECM usually still appears in this regime, but its frequency band becomes much broader, compared with that in the enhanced-recycling H-mode regime. The ECM tends to become much weaker during the lithium power injection but another mode in the pedestal region, a low-n (mostly n = 1) Magnetic Coherent Mode (MCM) [4], becomes stronger or more coherent. The MCM shows very weak density fluctuations but strong magnetic fluctuations as measured by fast Mirnov coils mounted on the wall.

The third ELM-free regime was obtained at high heating power and low pedestal collisionality ($\nu_e^* < 1$), with strong MCM but without ECM [4]. Dedicated experiments have been conducted in the last campaign to study the nature of MCM. Density ramp-up experiments exhibits a good linear scaling of the MCM frequency with the local Alfvén frequency and the frequency is near the local TAE frequency, suggesting the possibility of TAE modes. In addition, the MCM frequency was observed decreasing during Ip ramping down, i.e., q_{95} ramping up. This phenomenon can be interpreted as a continuous radially-inward shift of TAE gap up the pedestal density gradient. The role of MCM in ELM suppression is still under investigation.

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TECXY simulations of the ITER divertor

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The 2D boundary layer code TECXY [1] has been proven to be a useful tool for the investigation of specific physical questions in the scrape-off layer (SOL) of limiter tokamaks. The physical model is based on Braginski-like equations for the background plasma and rate equations for the impurity ions. An analytical description of two groups (cold and charge-exchange) of neutrals allows plasma recycling as well as sputtering and self-sputtering of impurity atoms at the target surface to be taken into account. Moreover the code incorporates drift motions and currents in a fully self-consistent way with plasma and impurity dynamics in the real curvilinear geometry of the tokamak SOL. Recently, an extended version of the code has been developed which takes into account multi-species impurity transport and can be run in diverted geometries [2].

In the present paper, we use TECXY to study the ITER plasma boundary in low power (P_{SOL} = 20-30 MW) non-active phase hydrogen discharges, benefitting from the much reduced computational demands of the code in comparison with the more complete coupled fluid-Monte Carlo neutral package SOLPS, which has been the workhorse of ITER simulations used to guide the divertor and fuel cycle design. TECXY is first benchmarked against the SOLPS-ITER code, the most recent SOLPS version now under the control and management of the ITER Organization. The benchmark focuses on a pure hydrogen density scan at low power and q_{95} = 3 (intended to represent either I_p, B_T = 5 MA, 1.8 T or 7.5 MA, 2.65 T non-active operation), covering the range of low to high recycling and then rollover to partial detachment with the aim of exploring which divertor physics TECXY can adequately represent despite the relatively simple neutral model.

The benchmarking exercise proved to be sufficiently convincing, at least for the non-strongly detached solutions, that the code could be deployed for first studies of the influence of drifts on the solution, in particular the in-out target power loading asymmetry which is a critical ITER issue with regard to power handling under burning plasma conditions. The less CPU intensive nature of TECXY allows a wider range of scoping studies with drifts activated to be performed than is possible with SOLPS-ITER, for which such simulations are extremely time consuming and have only just begun. First TECXY results indicate that for these low power cases, the influence of drifts on the target asymmetry is rather modest, but is strongly dependent on the plasma density.

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OEDGE simulation of W leakage from different divertor configurations in DIII-D H-mode discharges

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OEDGE simulation is carried out to understand the divertor W transport in the SOL for different divertor configurations of divertor metal rings campaign (MRC) on DIII-D [1]. It is found that the W leakage (the fraction of the sputtered W that enters the core plasma) from the strike point (SP) region only differs by 5% between partially closed and open divertor configuration discharges. Simulations of the partially closed divertor configuration discharge indicate that the leakage from a ring in the far SOL is about two times higher than from the strike point due to the high density near the strike point increase the W prompt re-deposition, even though the magnitude of the SOL tungsten source in these cases was lower.

During the MRC, two toroidally symmetric 5cm wide rings of tungsten coated tiles were installed in the outer divertor of DIII-D. In addition, different W isotopic compositions were used for each ring [2]. This allows the source of the W content measured in the far SOL plasma using collector probes to be identified. The OEDGE code is used to model the W source and SOL transport from both the partially closed divertor (SP on the floor) and open divertor (SP on the shelf) for selected inter-ELM H-mode plasma discharges. Experimental emission profiles from hydrogen, carbon and tungsten spectroscopy are used as constraints for modelling. Detailed modelling of the radial distribution of the W source and relative leakage for locations across both the floor and shelf is presented. The results show that W sputtering is dominated by C2+ and C3+ from local divertor sources. Moreover, all these cases demonstrated accumulation of tungsten near the crown of the plasma consistent with the collector probe measurements [3].

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Scrape-off layer density tailoring with local gas puffing to maximize ICRF power coupling in ITER

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The coupling of ion cyclotron range of frequencies (ICRF) power to tokamak plasmas depends critically on the width of the fast wave evanescence layer, the distance from the antenna strap to the cut-off density (typically of the order of 4×10^{18} m^{-3} in ITER standard heating scenarios). Previous experiments and simulations in ASDEX Upgrade and JET show that by shifting the fueling gas source from the lower divertor to the top or midplane of the machine, the scrape-off layer (SOL) plasma density in front of the antenna is increased and the evanescence distance is decreased, leading to an increase in the ICRF power coupling. In particular, it is observed that midplane gas puffing close to the ICRF antenna can increase the antenna coupling resistance by about 120% both in AUG and JET H-mode plasmas. Moreover, the local gas puffing can reduce the impurity sources from the walls in these experiments.

In the study reported here, the effects of local gas puffing on the SOL density in ITER are simulated for the first time with the 3D edge plasma fluid and neutral particle transport code EMC3-EIRENE. A toroidal 360° computation grid, extending to the wall and including all plasma-facing components, is used. The simulated plasma is a deuterium H-mode using the baseline Q = 10 magnetic equilibrium with $q_{95} = 3$. The following fuel gas puffing cases are examined: (1) toroidally distributed divertor gas valves injecting gas from below the divertor cassettes; (2) top gas valves magnetically connected to the ICRF antennas; (3) two midplane gas valves located toroidally between the two ICRF antennas; (4) two midplane gas valves located toroidally outside the two ICRF antennas. Based on the SOL density profiles obtained from the EMC3-EIRENE simulations, the antenna codes FELICE and ANTITER are used to calculate the coupling resistances. Parameter scans in EMC3-EIRENE have been made in order to understand the influence of transport coefficients on the coupling resistances. Power fluxes to the wall structures for the different gas puffing cases have also been extensively investigated.

The simulation results indicate that the SOL density in front of the antenna is most significantly increased with midplane gas puffing (Cases (3) and (4)) with similar rise for the two cases. Top gas puffing increases the local density to a much lower level despite magnetic connection to the antennas. The simulations thus confirm similar behaviour for ITER as seen experimentally in current devices and strongly suggest that ITER should attempt to modify the existing main chamber injection configuration to bring one of the four planned injection points closer to the antennas.

Experimental Characterization of the Lithium Tokamak eXperiment-β Scrape-Off Layer and a Theoretical Study of Electrostatic Potential in Collisionless Scrape-Off Layers*

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The Lithium Tokamak eXperiment (LTX) is a spherical tokamak device designed to study lithium plasma facing components (PFCs). The lithium coated wall of LTX has been demonstrated to produce a plasma edge with high electron temperature (200 eV or greater)[1, 2]. Plasma density in the outer scrape-off layer is also found to be very low, around 2×10^{17} m^{-3}, as a result of the low recycling lithium boundary [2]. The high temperature, low collisionality region of the plasma extends into the SOL. The recent upgrade to LTX-β [3] includes installation of a neutral beam, which will provide further heating and fueling of the plasma. Core and edge diagnostics will also be expanded. As part of this expansion, a Retarding Field Analyzer (RFEA) has been developed for the scrape-off layer (SOL) of LTX-β. Measurements of the ion temperature, ion energy distribution, and the local space potential will be performed in the (SOL) plasma using this RFEA. Upgraded high field side (HFS) and low field side (LFS) Langmuir probes will replace existing triple probes so that higher electron temperatures (over 100 eV at the wall) can be more reliably measured. The HFS probes are also positioned to give radial and vertical gradient measurements. The design of the upgraded edge particle diagnostics set will be presented, along with preliminary data.

Since a high temperature, low collisional edge is expected for LTX-β, with a high mirror ratio near the LCFS (around 4), the majority of particles in the SOL will be mirror-trapped. Trapped particle effects will therefore become significant in the physics of the SOL plasma, and warrants further theoretical investigations. Here we present a theoretical study of the ambipolar potential formed in the collisionless SOL via differential loss of the electrons and ions, known as the Pastukhov potential in the literature [4]. Preliminary numerical results will also be presented.

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Far SOL Transport and Plasma Wall Interaction in Main Chamber

Impact of ICRF on the scrape-off layer and on plasma wall interactions: from present experiments to DEMO

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During the last decade, studies of the impact of ICRF (Ion Cyclotron Range of Frequencies) power on interactions with the scape-off layer plasma have been actively pursued on many experiments with metallic walls: in ASDEX Upgrade (AUG), in Alcator C-Mod and in JET-ILW. The studies of the impurity production in particular are relevant for the use of ICRF systems in fusion reactors. This contribution gives an overview of the recent progress.

ASDEX Upgrade experiments show that the impurity generation associated with the ICRF power can be drastically reduced by minimizing the RF currents on the plasma facing antenna frame elements. This approach was at first demonstrated by using the modified broad-limiter 2-strap antenna, then further developed with the 3-strap antenna concept where the RF image currents on the antenna frame are cancelled by optimizing power balance between the straps. The concept of the image current cancellation was successfully extended to a 4-strap antenna by using the Field-Aligned (FA) antenna in Alcator C-Mod, where the RF-enhanced potential measured on the magnetic field lines connected to the active antenna was eliminated by the proper power balance. The impurity sources remote to the FA antenna, which govern the total impurity influx in Alcator C-Mod, were also reduced. The RF image current cancellation technique is thus fairly robust and is applicable to other experiments.

In AUG and JET-ILW, the DC biasing of the plasma on the magnetic field lines by the RF-current carrying structures manifests itself in multiple SOL modifications. In the vicinity of the antennas, increased DC currents and enhanced sputtering at the limiters are observed (AUG). Further away on the field line connections to the antennas, increased parallel ion energies (AUG) and increased sputtering rates (AUG and JET-ILW) are measured. At the same time, the density profiles close to the antenna (in AUG) and on the field lines connected to it (in JET-ILW) are modified due to the imposed 3D DC-field and the consequently forming E×B convective cells. Several numerical tools build a basis for the description of the interactions, with prospects of becoming quantitative. The electromagnetic codes TOPICA and RAPLICASOL for near-field calculations, the nonlinear code SSWICH for the modelling of the slow wave propagation and sheath rectification in the SOL, as well as EMC3-Eirene (a 3D fluid edge plasma transport code with kinetic neutrals for calculations of the density modifications) have been successfully applied to describe the experimental behavior. The latter was also used to study the effect of the local gas injection close to ICRF antennas in AUG and JET-ILW to increase the antenna loading and to further reduce the impurity sources.

Experience and techniques developed in the recent studies can be used to optimize antenna design and to recommend operational recipes, to ensure the high-Z and high power compatibility of the ICRF system in the next-step machines, such as ITER and DEMO.

Improved ERO modelling of beryllium erosion at ITER upper first wall panel using JET ILW and PISCES-B experience

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The lifetime of ITER beryllium (Be) first wall (FW) plasma-facing components (PFC) as well as other issues impacting the availability of the fusion device (e.g. T retention due to co-deposition with Be and Be-induced sputtering of tungsten divertor PFCs) are strongly dependent on Be erosion and transport in the boundary plasma. In recent years a number of related studies on JET equipped with the ITER-like wall (ILW) and on the PISCES-B linear plasma device have helped to improve the data for physical sputtering and chemically-assisted physical sputtering (CAPS). These data are used in the ERO 3D Monte-Carlo plasma-surface interaction impurity transport code, which has been widely deployed for Be erosion studies, including modelling of the plasma interaction on a single FW panel in the vicinity of the secondary X-point region in ITER [1]. The work reported here revisits these earlier ITER simulations in the light of the new physics incorporated into the ERO code.

The significance of the following effects has been demonstrated by ERO application to the JET ILW and the respective code extensions validated: angular and energy distributions of both sputtering ions (D^+, D_n^+, Be^{n+}) and released neutral Be (also Be_xD_y) species. In particular the angular distribution is very much determined, especially at grazing angles of the B-field to the PFC surface, by the electric field in the sheath treated now using a new semi-analytic approach. Also, the consideration of sputtering by charge exchange neutrals, the 3D distribution of the Be impurity concentration determining the self-sputtering and plasma shadowing effects is important. Applications to PISCES-B has confirmed the (~4 times) reduction of the Be sputtering yield at normal incidence due to the high D surface content in plasma-wetted areas compared to the commonly used SDTrimSP (binary collision approximation) simulations for pure Be. The effect of the quasi-metastable (^3P) state population after sputtering was implemented and shown to be significant for spectroscopy interpretation.

Taking into account the physics improvements, modelling of net Be sputtering at the lifetime-critical plasma-wetted regions of the shaped ITER FW panel results in the most optimistic number of ~4200 baseline Q=10 discharges possible, as predicted in [1], though, as always, such predictions are dependent to zero order on the assumptions made for the background plasma conditions, which are currently unknown for ITER. The new simulations suggest that this could be significantly reduced (by ~40%) as a consequence of the improved sheath model for oblique magnetic fields at the panel surface. The remaining uncertainties will be demonstrated, motivating further experiments. Related predictive simulations of Be impurity light emission can provide important input to assist (sensitivity and stray light issues) in designing of the ITER main chamber visual spectroscopy systems.

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On the effect of shoulder formation on thermal transport in the SOL of ASDEX Upgrade

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One of the main obstacles on the path towards a reactor-relevant tokamak is the unknown distribution of heat fluxes onto the different plasma facing components. While it is generally accepted that convective filamentary structures dominate particle transport in the far-SOL and substantial effort has been dedicated to their study in recent years [1,2], few works have attempted to provide a quantitative description of the perpendicular and parallel heat fluxes in the far SOL. Recently, data from different diagnostics were merged into a common database of equivalent L-mode experiments carried out on ASDEX Upgrade (AUG) for different values of divertor collisionality [3], thus providing all relevant SOL plasma parameters including density, electron and ion temperature, filament velocity, parallel Mach number, etc. This approach was validated by comparing the sheath theory calculation of power deposition on the midplane manipulator with its direct observation from independent IR data processed with the heat flux calculation THEODOR code. With this set of data, a radial profile of the perpendicular heat fluxes associated to filaments could be calculated in the outer midplane. It was concluded that the total perpendicular power carried by the filaments accounts for about 25% of power across the separatrix, P_{SOL}, around 20 mm in front of the separatrix, regardless of whether the shoulder had formed. As well, taking the temperature and density profiles measured at the midplane and the divertor, an estimation of the parallel heat transport was carried out. The main conclusion was that, when $T_i/T_e > 3$ is achieved at the separatrix (up to $T_i/T_e \sim 5$ was measured in the experiments), parallel transport is no longer dominated by electron conduction as ion conduction becomes at least equally important. In this work, we discuss the relevance of those results for next generation tokamaks. First, by comparing the parallel transport estimations with equivalent calculations carried out with SOLPS, we confirm that there are certain realistic regimes in which electron conduction does not dominate parallel transport. Second, we discuss under which scenarios it may be expected that filaments are able to advect significant amounts of energy and thus should be taken into consideration when assessing sputtering yields or heat loads onto plasma-facing components.

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Abstract Withdrawn

Estimation of Fuel Particle balance in steady state operation with hydrogen barrier model

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Simulations for a 3 GW thermal output fusion power plant show that $1.06\text{x}10^{21}$ D-T reactions occur per second under conditions where the ion out-flux from the core plasma is $5\text{x}10^{22}$ ions/s while the gas-puff rate is $1\text{x}10^{23}$ particles/s [1]. This implies a fuel burn-up fraction of only about 1%. Recent calculations that consider core fuel injection show that the fraction of the injected D-T that fuses only increases to 5% [2]. This means that the remaining 95% of the D-T fuel must be either pumped out or stored in plasma and plasma facing materials (PFMs). The mechanisms of adsorption, reflection, absorption, and desorption on PFMs all impact the fuel particle circulation and strongly depend on particle injection energy and the PFM surface conditions. Thus, fuel particles such as hydrogen (H), deuterium (D), and tritium (T) are continuously circulating during the plasma discharge as molecules, atoms, and several types of ions in the core plasma, scrape-off layer (SOL), vacuum regions, and the PFMs. Therefore particle circulation may play an important role in the performance of future fusion power plants. However, the impact of wall-stored fuel particles in PFMs on the particle balance during steady state operation is not well understood because of the lack of a proper wall model. Recently Q-shu University Experiment with Steady State Spherical Tokamak (QUEST) was able to achieve a long duration discharge lasting 1 hour 55 minutes with well-controlled wall temperature (hot wall) composed of atmospheric plasma sprayed W (APS-W) [3] and proper feed-back control of H_α level [4]. The surface of PFMs was microscopically investigated and it was found that the surface was covered with plasma-induced deposition layer of several tens nm in thickness [5]. The deposition layer has the capability to store fuel particles, and a hydrogen barrier is formed in the boundary of the substrate made of APS-W and the accumulation of fuel particles which results in higher surface recombination of fuel particles [6]. Consequently, the in-coming fuel particles into the PFMs are likely to balance the out-going fuel particle molecules that are under conditions of wall-saturation. Particle balance is calculated using a point model based on the QUEST wall condition with the presence of the hydrogen barrier. In the calculation, we assume that part of the in-coming fuel particles are reflected on the surface of the PFMs. High fuel particle reflection rate under constant gas fuelling results in an explosive increase of density in the plasma.

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RF Sheath Modeling of Spectroscopically Observed Plasma Surface Interactions with the JET ITER-Like Antenna

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Waves in the Ion Cyclotron Range of Frequencies (ICRF) enhance Plasma-Surface Interactions (PSI) near the wave launchers and magnetically-connected objects via Radio-Frequency sheath rectification. ITER will use 20MW ICRF power over long pulses, questioning the long-term impact of RF-enhanced localized erosion on the lifetime of its Beryllium (Be) wall. Recent dedicated ICRF-heated L-mode discharges documented this process on JET for different types of ICRF antennas.

Using visible spectroscopy, poloidally-localized regions of enhanced (by ~2-4x) Be I and Be II light emission were observed on JET outboard limiters during ICRF heating. Sequential toggling of the ICRF antennas combined with concurrent magnetic pitch-angle sweeping, consistently showed that these regions can be traced, *via* magnetic field-line connection, to regions of anticipated high $E_{//}$ RF near-field at an antenna [1,2]. The observed RF-PSI induced by the JET ITER-like Antenna (ILA) was qualitatively comparable to that induced by the JET standard, type-A2 antennas, for similar strap toroidal phasing and connection geometries [3,4]. Near-field modeling, using the antenna code TOPICA, was able to reproduce, qualitatively, the observed phenomena. However, a quantitative discrepancy persisted between the observed Be source amplification and the calculated, corresponding increases in $E_{//}$ at the magnetically mapped locations at the ILA. This discrepancy was specifically for current drive strap phasing, over a change from lower ILA to upper ILA feeding.

This paper revisits these ILA-specific current drive-phased cases using the Self-consistent Sheaths and Waves for Ion Cyclotron Heating Slow Wave (SSWICH-SW) code, coupling slow wave evanescence with DC SOL biasing [5]. The approach so far was limited to correlating the observed, enhanced emission regions at the remote limiters to the (fluctuating) near-electric fields, as calculated by TOPICA, and specifically for the antenna limiters' leading edges. The present approach includes a model for the rectification of these near-fields and of their transport to the remote PSI regions. As such it takes into account of the near fields throughout the recessed parts of the antenna. The aim is to determine whether more self-consistent modeling of the anticipated RF sheath induced effects, excited by the $E_{//}$ map from TOPICA at the ILA antenna mouth, could reduce the discrepancy.

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Statistical properties of the heat flux on the outboard mid-plane wall in Alcator C-Mod

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Plasma blobs are believed to transport a significant amount of particle and heat across the outboard mid-plane scrape-off layer onto limiters and other plasma facing components. Analytic theory, experimental observations, and numerical simulations suggest that their radial velocity depends on the line-averaged core plasma density $\overline{n}_\mathrm{e}$ [1]. Experiments further suggest that the plasma density fluctuations at the limiter radius are Gamma distributed [2] and that the average density increases with increasing $\overline{n}_\mathrm{e}$ while retaining an order unity relative fluctuation level. A stochastic model describes such fluctuation data time series as the super-position of uncorrelated exponential pulses, corresponding to individual blobs [3]. It predicts that the fluctuation amplitudes are Gamma distributed and there is a Lorentzian power spectral density (PSD). The parameters of these distributions are given by the degree of pulse overlap, the average pulse duration time, and the pulse asymmetry [3].
The Alcator C-Mod Mirror Langmuir Probe system [4] was used to investigate the statistical properties of heat fluxes on the plasma facing components in the framework of this model. We report on the statistical properties of one second long time series of the electron density n_e, temperature T_e, the electric drift velocity as well as particle and heat fluxes. The time series were sampled in a series ohmically heated lower single-null discharges where $\overline{n}_\mathrm{e}$ was between 10 and 60% of the Greenwald density. Both the n_e and T_e data time series are Gamma distributed and their PSDs are well described by a Lorentzian. The variation of the distribution parameters with $\overline{n}_\mathrm{e}$ will be discussed. Calculating the sputtering yield Y of the vessel wall with the here observed T_e fluctuations results in Y > 0 while using only the average T_e value yields Y=0. The convective heat flux is found to be larger than the conductive heat flux in low density discharges but switch for high density discharges. This flux roll-over will be discussed in terms of the phase relation of the n_e and T_e fluctuations and the radial electric drift velocity.[1]

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Scrape-off layer density shoulder formation mechanisms in JET ITER-like wall L-mode and H-mode plasmas†

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The characteristics of the SOL plasma (Scrape-Off-Layer) determine the level of power loading and erosion of, and DT implantation in, surrounding material surfaces, and thus help determine the viability of a tokamak-based fusion energy reactor. For example, the appearance of low-field side (LFS) SOL flattened or long e-folding length density profiles (so-called density ‘shoulders’ [1]) raise the density (neutral and ion) at main chamber surfaces, accompanied by plasma surface interactions. This study, which examines the potential mechanisms leading to SOL density shoulders in JET tokamak with the ITER-like wall (ILW), includes a factor of 2 range in plasma current, compares the effect of vertical (closed) vs horizontal (open) outer divertor target configurations, as well as the effect of divertor nitrogen (N_2) seeding vs deuterium (D_2) fuelling, primarily for L-mode plasmas.

There is evidence against local SOL ionization directly increasing SOL densities as a mechanism for shoulder formation. It is also found that a second, and commonly referred-to mechanism – the change of parallel resistivity in the SOL (characterized through $\Lambda_{div} =[L_{||}\nu_{ei}\Omega_i]/c_s\Omega_e$ [2]) leading to changes in SOL turbulence velocity and size – is anti-correlated with SOL density shoulder formation; while core density ramps through D_2 fuelling leads to increases in shoulder amplitude and Λ_{div}, N_2 seeding increases Λ_{div} but not SOL shoulder amplitude. The most consistent quantity that correlates with formation of SOL density shoulders is the amount of divertor recycling (quantified through Balmer D_α): a) shoulders form at the transition of the divertor plasma from sheath-limited to high-recycling condition; b) strike point sweeping in major radius changes divertor D_α and shoulder amplitude concurrently without changing Λ_{div}; c) N_2 seeding lowers both shoulder amplitude and divertor D_α while raising Λ_{div}; and d) switching the outer divertor leg from the horizontal to vertical target both lowers divertor D_α and shoulder amplitude. In attached divertor plasmas Balmer D_α light emissivity is a measure of the amount of charge exchange and ionization reactions which can change flows in the divertor, lowering or increasing the loss (sink) of ions from the upstream SOL and thus modify the density there. A comparative study of H-mode discharges indicates that similar conclusions about shoulder formation mechanisms are drawn for H-mode.

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† A paper corresponding to this work has been submitted to Nuclear Fusion

Modelling of Beryllium migration in JET-ILW with ERO2.0

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The JET ITER-like wall (ILW) can be used as a test bed for ITER [1] for studying erosion of main chamber beryllium (Be) and its migration into the tungsten (W) divertor [2]. While Be erosion affects the lifetime of main chamber wall components, its migration into the divertor can lead to W erosion there and also to tritium retention by co-deposition.

The newly developed Monte-Carlo code ERO2.0 code has been recently applied for modelling of the Be erosion and migration in JET limiter discharges [3]. It has been demonstrated that due to massive parallelization and other technical improvements, ERO2.0 is capable of simulating the entire 3D-shaped ILW (with high level of detail) and the edge plasma volume of JET. This provides several advantages in comparison to local modelling with the previous code version ERO1.0 [4]. Most notably, the increased volume allows the validation with a larger number of experimental diagnostics situated at different locations. These include 2D camera images (infra-red for heat flux and filtered Be I, Be II line and BeD band emission for particle fluxes) as well as multiple line-of-sight (LOS) integrated spectroscopy chords.

In the present contribution, we demonstrate that including the long-range migration of Be in the modelling of limiter discharges allows a self-consistent treatment of self-sputtering. This eliminates the need for Be plasma concentration assumptions which were unavoidable in the earlier simulations considering just the local volume. Furthermore, we present first ERO2.0 modelling of Be erosion and migration in JET diverted discharges, using plasma backgrounds simulated with SOLPS-ITER [5]. We study the influence of various main chamber erosion mechanisms, such as physical sputtering by ions, chemically assisted physical sputtering (CAPS) leading to release of BeD molecules, and physical sputtering by charge-exchange (CX) neutrals, which is dominant for the main chamber in discharges with divertor configuration. The simulation results are compared to LOS integrated spectroscopy [2] characterizing the Be effective sputtering yields to demonstrate improved understanding over ERO1.0 modelling. ERO2.0 validation is broadened by the simulation of 2D IR and spectroscopy measurements. We compare the erosion and subsequent migration and deposition of Be in diverted discharges to those in limiter configuration. This allows to provide a first preliminary estimate for the maximum possible tritium retention due to co-deposition with Be (re-erosion of Be deposits and outgassing are neglected).

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Scrape-off layer and near-separatrix divertor turbulence in NSTX and NSTX-U discharges

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The 3-D structure of divertor turbulence from the private flux region (PFR) to the far SOL is characterized in NSTX and NSTX-U by means of fast camera imaging. Edge and divertor turbulence contributes to particle and heat fluxes to the first wall and divertor plate in tokamaks. On the low field side (LFS) midplane, SOL turbulence is known to have parallel and perpendicular scale lengths of many meters and a few cm, respectively. However, it has not been clear until recently to what degree LFS turbulence extends to the divertor plate, or whether there is any additional turbulence generated in the divertor region. The characterization of the 3D structure of divertor turbulence is critical for the understanding of divertor heat and particle fluxes and the interpretation of divertor transport and radiation, usually based on average quantities. In this work, divertor fluctuations due to upstream blobs and divertor-localized fluctuations are characterized for NSTX and NSTX-U discharges.

Helical intermittent filaments on the divertor target were analyzed via surface-localized neutral lithium emission in diverted NSTX L-mode discharges. These helical structures represent the footprint of upstream blobs and correlate with n_e fluctuations measured by target Langmuir probes. Fluctuations in divertor light emission of up to 20-40% are correlated with outer midplane turbulence as measured by gas puff imaging, with correlation up to 0.7 for ψ_N~1.08-1.3. Fluctuation levels in the divertor and correlation with upstream turbulence progressively decrease approaching the separatrix. Blob disconnection due to X-point geometry and collisionality are being considered as explanation for the decreased correlation. Initial results from the linear eigenvalue solver ArbiTER [1] indicate that collisionality can limit the penetration of SOL turbulence to the NSTX divertor.

In the region disconnected from upstream turbulence, divertor-localized fluctuations are observed via imaging of C III and D-α emission in diverted L-mode discharges, similar to observations in Alcator C-Mod and MAST. Field-aligned filaments connected to the divertor target plate are radially localized at the separatrix on the outer divertor leg and in the PFR on the inner divertor leg and limited to the region below the X-point with fluctuations levels up to 10-20%. Filaments are characterized by comparable poloidal and radial correlation lengths (10-100 ρ_i) and parallel correlation lengths of several meters. Toroidal mode numbers are in the range of 10-20 and 2-8 for outer and inner leg filaments, respectively. Opposite toroidal rotation is observed for inner and outer leg filaments with apparent poloidal propagation of ~1 km/s. The poloidal motion of outer leg filaments is consistent with advection due to E×B drift as calculated by the multi-fluid edge transport code UEDGE with inclusion of cross field drifts and is opposite to the one observed for SOL blobs. Disconnection between inner and outer leg filaments as well as the absence of correlation with upstream turbulence support the hypothesis of X-point disconnection for divertor leg filaments. Work is ongoing with the ArbiTER code to understand mechanisms generating divertor-localized fluctuations.

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Carbon/deuterium particle transport measured by using cavity technique depending on the alignment to the toroidal magnetic field line at mid-plane region in KSTAR

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Plasma surface interaction (PSI) is the most important issue for future fusion device development and main concerns include the lifetime of plasma facing components (PFCs) due to steady state sputter erosion, plasma contamination by eroded material, tritium codeposition in redeposited material. In tokamak, toroidal and poloidal magnetic fields affect the motion of ion while that of charge exchange neutrals is relatively free but affected by plasma flow. Those are related to the parallel and perpendicular transport of carbon/deuterium particles at the mid-plain. In this paper, the technique of cavity sample is applied to study the carbon/deuterium particle transport depending on the alignment to the toroidal magnetic field line at mid-plane region in KSTAR measured by using cavity technique [1].
The cavity has a small volume inside stainless steel box, and a different two slit opening directions on the cover plate. Each cavity sample was installed on a different four direction (front/I_p/reversed I_p/top) with respect to the toroidal magnetic flied line (B_T). Samples were exposed to KSTAR Ohmic and H-mode plasma discharges by using multi-purpose mid-plane manipulator [2]. A standard reference sample of a:C-H thin layer is used to determine the net change of the thickness. The thickness profile of the a:C-H layer was measured by spectroscopic ellipsometer before and after the exposure. Plasma parameters of the shots were plasma current (I_p) 600 kA, B_T 2.0 T, and line integrated electron density (n_e) ~ 2.5 x 10^{19} m^{-3} in Ohmic plasmas. In the H-mode, it were I_p 600 kA and 640 kA, B_T 1.8 T, and n_e 2.9 x 5 x 10^{19} m^{-3}. Exposure time of the samples in Ohmic was 35.4 sec and 28 sec in H-mode. In Ohmic plasma, maximum deposition was observed at front cavity with 0.43 nm/s of deposition rate. No change was observed at the top cavity. I_p and rev.I_p cavities show similar pattern but with higher erosion rate at I_p cavity. In H-mode plasma, all cavities show net deposition except for the rev.I_p. Especially, cavity at the top shows maximum deposition rate of 1.02 nm/s due to ion bombardment. A detailed results and interpretation will be reported.

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Performance Estimation of Beryllium under ITER Relevant Transient Thermal Loads

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The employment of beryllium as armor material in ITER ensures a high level of plasma performance at the cost of a high erosion rate and the risk of melting during transient events. To assess the possible impairment of the beryllium armor by these events, a series of transient thermal load tests was carried out. The electron beam facilities JUDITH 1 and JUDITH 2 located at Forschungszentrum Jülich were used to exert transient thermal loads with ITER relevant edge localized mode and massive gas injection (MGI) like characteristics onto S-65 grade beryllium specimens. The test campaign covered a broad range of loading parameters, i.e. heat flux factors in the range of 3 – 32 $MWm^{-2}s^{0.5}$, pulse durations of 1 – 10 ms, base temperatures from room temperature up to 300 °C, and numbers of pulses of 1 – 10^7. This comprehensive dataset was used to map the damage behavior of beryllium. The results indicated that the loading conditions causing no damage on beryllium after 100 pulses are limited to a heat flux factor of ≤ 6 $MWm^{-2}s^{0.5}$ at base temperatures below 300 °C. For higher heat flux factors or base temperatures, the yield strength of the investigated beryllium becomes too low to compensate the thermally induced stresses fully elastically. Despite the fact that inflicted damage in the form of plastic deformation/roughening/cracking seems to be hardly avoidable under operational conditions, beryllium showed a promising long term performance under transient thermal loads. Tests with high numbers of pulses of up to 10^7 indicated that the induced damage saturates after 10^5 pulses as long as the applied heat flux factor does not exceed 9 $MWm^{-2}s^{0.5}$.

MGIs in ITER are intended to reduce the severe local damage caused by plasma disruptions in the divertor region by transforming the stored plasma energy to radiation, which is spread homogeneously across the reaction chamber wall. This radiation leads to transient thermal loads on the beryllium tiles capable of melting them [1]. The experimental simulation of these heat loads pointed out that, under the conservative assumption of 1000 full power disruptions mitigated with MGIs, the affected beryllium armor thickness is about 340 µm.

Overall, the post mortem analysis of the carried out transient thermal load tests revealed numerous surface morphology changes such as roughening, cracking, and melting but also the formation of pits, elongated filaments within cracks, and the detachment of the melt layer from the bulk material under repetitive melting. The pit formation and the detachment of the melt layer under repetitive melting were linked to the formation/segregation of beryllium oxide at the surface and at the grain boundaries, respectively. The oxygen partial pressure in the experiment was limited to 2×10^{-5} mbar, which is the lowest accessible value in JUDITH 1. Further investigations are planned to examine whether the beryllium oxide formation/segregation significantly affects the performance of beryllium under transient thermal loads at lower oxygen partial pressures, closer to the value of 10^{-9} mbar anticipated in ITER.

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Impurity Sources, Transport and Control

The DIII-D Metal Rings Campaign: Characterizing tungsten sources, SOL transport, and its impact on high-performance scenarios

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The DIII-D Metal Rings Campaign utilized a novel isotopic W tracer technique in the outer divertor to gain unique insights into W sourcing and scrape-off-layer (SOL) transport in the presence of a predominantly low-Z (i.e., C) material background. Notably, it is observed that the W atomic escape probability from the divertor region depends crucially on both the W source location and the ELM behavior. For large ELMs at low frequency, expected during any phases of poor ELM mitigation on ITER, W leakage into the main SOL is approximately equally efficient from both the strike point (SP) and far-target region (i.e., 3-5 heat flux widths from the SP). In contrast, for small ELMs, the divertor W source at the SP became the dominant driver of upstream SOL contamination. On-axis electron heating effectively suppressed the core neoclassical pinch to ion diffusivity ratio in such scenarios, nearly completely eliminating core W accumulation, as observed in other W devices. Only marginal reduction of the pinch/ diffusivity ratio occurs during off-axis electron heating, so strong W accumulation persists. In the far-SOL, asymmetries observed in the W collection pattern along two sides of a midplane collector probe are consistent with the formation of a 'potential well' driven primarily by the ion temperature gradient force along flux tubes near the separatrix. These results are supported by DIVIMP modeling, which suggests W impurities accumulate near the plasma crown in the near-SOL, then diffuse radially outward and flow back downstream towards the target.

ELM-resolved spectroscopic measurements of the divertor W source support ERO modeling predictions that inter-ELM phases are dominated by W sputtering via C impurities. ERO-OEDGE quantitatively reproduces the observed W erosion profile in L-mode discharges only when inward ExB drift effects are included because a substantial fraction of the C impurity flux to the W surface is carried by these drifts. Empirical SDTrimSP sputtering modeling predicts that both D and C contribute substantially to W sourcing during ELMs because the measured average ion impact energy increases from below to substantially above the threshold for D→W sputtering. The implications of the Fundamenski-Moulton 'free-streaming' model for parallel heat and particle flux during ELMs on W sputtering are explored. Consistency is only observed between measurements and predictions of how ELM deposited energy density scales with W source when broadening of the divertor heat flux footprint and enhanced target electron densities (e.g., via increased neutral recycling) are taken into account. Both effects are directly observed. Post-mortem analysis of the W-coated divertor tiles indicated unipolar arcing also contributed to high-Z sourcing during ELMs and arcing activity correlated with ELM size.[1]

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Analytical investigation of W sputtering in quasi-steady-state ELMy conditions at JET ITER-like Wall

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The ITER baseline scenario with 500 MW of DT fusion power and Q=10 implies an H-mode plasma with mitigated Edge-Localized Modes (ELMs). JET equipped with an ITER-like wall (ILW) material combination: tungsten (W) divertor and beryllium (Be) main chamber, provides the most relevant environment for ITER erosion studies including ELM effects. It was shown for JET ILW that inter-ELM W erosion in divertor happens mostly due to sputtering by Be plasma impurity as hydrogenic ion impact energies are below the respective W sputtering thresholds. However, an important W erosion mechanism in divertor is the intra-ELM sputtering by both, impurity and hydrogenic ions with energies determined by the pedestal temperature. It should be noted, that W impurity concentration in the core plasma of ITER just above $5 \cdot 10^{-5}$ can lead to unacceptable plasma cooling [1], therefore minimization and control of the intra-ELM W sputtering source is required.
The W intra-ELM sputtering flux from the outer divertor W target plate was assessed using the analytical interpretation of Langmuir probe (LP) flux measurements ("LP-Analytic") based on the "Free-Streaming" model (FSM) taking into account the angular dependence of sputtering yield [2]. These estimates provided an agreement within a factor of 2-3 with W I 400.9 nm line intensity measurements in JET-ILW hydrogen and deuterium plasmas.
In the present contribution the several improvements were implemented to refine this approach. We take into account the finite duration of the ELM source and follow time-resolved temperature and density drop during the pedestal crash. The improved approach is applied to the H-mode discharges during a two-week lasting JET ILW operation [3] with quasi steady-state wall conditions, providing a unique data set to statistically analyse the intra-ELM evolution of pedestal and target plasma profiles. The coherently ELM-averaged LP measured ion particle flux and heat flux profile from IR measurements presented in [4, 5] are used to assess the W gross erosion of the W JET outer and inner divertor plates under intra- and inter-ELM conditions. The modelling results are compared with passive absolute W I spectroscopy measurements mimicking the eroded W influx. The uncertainty due to Be concentration (<1%) in SOL plasma not exceeding 15% is demonstrated. The effect of the W self-sputtering including prompt effects is taken into account to assess the net erosion of W target plates.

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Chemically Assisted Physical Sputtering of Tungsten: identification via the $^6\Pi$-$^6\Sigma^+$ transition of WD in TEXTOR and ASDEX-Upgrade plasmas

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Tungsten (W) is selected as material for divertor plasma-facing components in ITER owing to its good power handling capabilities and the low fuel retention as demonstrated in the JET-ILW and ASDEX Upgrade (AUG) [1]. A critical issue for the plasma operation remains the source strength of W which can pollute substantially the plasma core. The W source strength is largely determined by **P**hysical **S**puttering (**PS)** processes caused by impinging impurities (e.g. O, C, N) and by high energetic fuel particle ions (H, D, T) as present during Edge Localised Modes (ELMs) [2,3]. A mixture of both species determines the W source, whereas the inter-ELM source can be mitigated by operation in the (semi-)detached regime with energies of impinging ions below the PS threshold; the intra-ELM source cannot be suppressed as long as energetic ions are expelled from the pedestal to the divertor.

Emission of the WD molecule was detected by spectroscopy near to the W surface during the exposition of a bulk W limiter to a series of D plasmas in TEXTOR with fluxes of a few $10^{23}D^+/m^2s$ and a variation of the local electron temperature (T_e) from 25eV to 85eV [4]. Detailed analysis of the spectral range between 673-678 nm revealed vibrational bands of the $^6\Pi$-$^6\Sigma^+$ transition of the WD molecule [5]. The band emission is thereby only present during W erosion but absent during WF_6 injections used for calibration which excludes the formation of WD in the plasma. W sputtering is TEXTOR is solely determined by C and O ions as the impact energy for deuterons is below the sputtering threshold [4].

The release of WD under combined high D and C ion flux can be explained by **C**hemical **A**ssisted **P**hysical **S**puttering (**CAPS**) either by (a) direct release of WD by D impact, (b) cascade-induced release by D impact, or (c) cascade-induced release by D impact. In all cases sufficient D content in the W matrix must be present and an impact energy threshold for the impinging ion for the release exists. The **CAPS** release is in competition to **PS** and the ratio of WI at 400.9nm to the WD band gives an insight on the strength of **CAPS**. Complementary to the ITER inter-ELM relevant case in TEXTOR (c), studies in AUG H-mode discharges with ELMs have been carried out to verify the release mechanism via (a) and (b).

Two comparable ELMy H-mode discharges (B_t=2.5T, I_p=0.8MA, P_{aux}=7.5MW, f_{ELM}=100Hz) in D were conducted with the outer-strike line located on a bulk W target plate in the outer divertor. T_e in the inter-ELM phase peaked at the strike line about 60eV and around 15eV deeper in the scrape-off layer whereas the intra-ELM impact energies are likely one order magnitude larger. The divertor spectroscopy system observes in the first discharge the standard WI at 400.9nm emission and in the second the WD band emission which was also detectable away from the strike-line where the impact energy for the main impurities in AUG (B, C, O) is lower. In the intra- and inter-ELM phase both WI and WD band emission can be observed, indicating that ELM-induced W sputtering takes place and also can cause the release of WD by CAPS.

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Investigation of impurity transport with turbulent simulations on WEST

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In the perspective of designing plasma facing components for future fusion nuclear devices, a still open issue relies in predicting the heat load on the divertor target and on the first wall. Nuclear reactors need to extract a large amount of the power in the plasma volume by radiation. To do so, light impurities such as Neon, Argon or Krypton can be injected to radiate near the separatrix. Simulations of impurity seeded ITER plasma rely mostly nowadays on 2D transport codes where the cross-field transport of impurities is taken into account by an ad-hoc diffusive transport, radial diffusivities being set to reproduce experimental results.
In order to go beyond this empirical approach, a dedicated effort has been made to provide first principle simulations of the impurity transport in the edge plasma. More precisely, the cross-field transport of impurities is at least partly driven by turbulent structures that require demanding numerical load to be dealt with. In this contribution, we present latest advanced of the code TOKAM3X which has been developed in the perspective to tackle the problematic of electrostatic edge plasma turbulence. Thanks to the cross-fertilization with the transport code SOLEDGE2D developed in the same team, TOKAM3X now solves Braginskii's equations for multi-species plasma taking into account the realistic geometry of the vessel using the penalization technique to treat Bohm boundary conditions. Neutrals transport and interaction with the plasma are taken into account by coupling with EIRENE.
The code TOKAM3X has been applied to simulate WEST tokamak scrape-off layer to simulate Nitrogen puffing. The impact of the light impurities on the turbulence is investigated. An estimation of the Nitrogen cross-field diffusivity is also given and compared with the main species (deuterium) diffusivity.

3D resolved effective charge state reconstruction with EMC3-EIRENE

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The three-dimensional Monte Carlo transport code EMC3-EIRENE [1][2] has proven to be a powerful edge plasma analysis tool for more than a decade now, especially, when three-dimensional magnetic fields strongly influence the plasma flows. We apply it as 3D interpretation tool to the stellerator Wendelstein 7-X (W7-X). Here, EMC3-EIRENE is used for two main purposes: Firstly, to provide full three-dimensional spatial resolution and reconstruction of further plasma properties which are not directly accessible experimentally. Secondly, to recalibrate measurement data sets, by using these numerically deduced plasma properties.

In this contribution, we present an approach to reconstruct the missing effective charge state Z_{eff} measurements in the first plasma operation phase of W7-X. Therefore, an iterative process between EMC3-EIRENE simulations and successive refinement of Langmuir probe measurement interpretations is used. Simulations within the iterative process were constrained by experimental bolometric power data, an array of up and down-stream profile measurements utilizing Langmuir probes installed on a mid-plane manipulator [3] and along the limiter. Matching refined measurement interpretations in each iterative step via the adjustment of free simulation parameters (mainly: anomalous radial transport coefficients) enabled a first consistent identification and even quantification of the effective charge state distribution Z_{eff}. The relative change for the last iterative step of Z_{eff} was found to be less than 1%, hence the effective charge state Z_{eff} is reconstructed consistently with the available information. Under the investigated experimental conditions, typically Z_{eff} differs between the up- and down-stream probe positions by a factor of up to two. Plasma material interaction parameters like sputtering coefficients can be included in the iterative process to further enhance the internal consistency.

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Deposition Profile Analysis of Enriched Isotopic Tungsten Tracer Particles from DIII-D Metal Rings Campaign Outer-Midplane Collector Probes

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First-of-its-kind experiments using isotopically-marked, W-coated divertor tiles coupled with midplane collector probes (CPs) have been performed on DIII-D to understand divertor impurity production and transport. Laser Ablation Mass Spectroscopy (LAMS) results are presented characterizing the isotopic ratios of deposited W on the mid-plane CPs and give quantitative information on the transport of W from specific poloidal locations within the lower outer divertor region. Two toroidal tile arrays (5 cm wide) of W-coated, TZM inserts in the lower outer divertor were used with the remaining plasma facing materials (PFMs) being carbon. The inner ring was coated in natural-W (with 26.5% W-182) and the outer ring was coated with 93% isotopically enriched W-182. The unique “isotopic fingerprints” for the W impurities released from each coating in a dominant C PFM environment enables their use as tracer particles to be collected and distinguished at other locations. A triplet set of replaceable graphite CPs each with collection surfaces on opposing faces oriented normal to the magnetic field and with distinct parallel collection lengths (determined by the CP diameters) were mounted at the outboard mid-plane and inserted at the distance of $R-R_{sep} <\sim 6$-8cm.

Rutherford backscatter spectrometry (RBS) analysis of the CPs has provided areal densities of elemental W content on each CP face and found peak deposited densities on the order of 5×10^{14} W atoms/cm^2; all CPs collected above the minimum RBS resolving threshold of 10^{12} W atoms/cm^2. These resultant W deposition profiles were compared with DIVIMP modelling of the SOL to better understand impurity transport in the edge plasma [1]. Higher peak W content was measured on the side of the probes connected along field lines to the *inner* divertor, potentially indicating a higher concentration of W in the upstream plasma. Radial decay lengths (RDL) between 5 and 50 mm were observed with consistently narrower RDL on the inner target face of the CPs. LAMS analysis has identified the increased presence of W-182 and yielded isotopic ratios of the deposited W. Using a two-source stable isotope mixing model (SIMM), the amount of W from each of the divertor rings that contributed to the total W deposition on the CP has been determined and shown to vary with the given plasma conditions, particularly ELM amplitude as examined through divertor spectroscopy and CP deposition profiles. Validation of the isotopic tracer technique is presented using a series of L-mode discharges with outer strikepoint on the enriched ring, which allowed for >95% accounting of the W deposited on the CP. Comparisons of deposited W profiles with strike point positioning, H-mode/L-mode, ELM frequency, and forward/reverse B_t are reviewed.

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Predictive ERO2.0 modelling of surface roughness effect on W physical sputtering and re-deposition in the linear plasma device PSI-2

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Surface morphology and its evolution during the plasma irradiation can have a large influence on the erosion and resulting lifetime of plasma-facing components. Effective sputtering yield Y_{eff}, re-deposition, as well as angular and energy distributions of the sputtered impurities can be significantly affected by the surface roughness prepared initially or arising during the irradiation process. Linear plasma devices such as PSI-2 have a number of advantages for plasma-surface interaction studies, including continuous operation. Numerical modelling can help to extrapolate the results of these experiments to large toroidal devices like ITER. The 3D Monte-Carlo PSI and local impurity transport code ERO is a well suitable tool for this purpose already applied for both ITER and PSI-2. This study is focused on tungsten (W) which is the main material for the ITER divertor area. The recently developed new ERO2.0 version of the code uses massive parallelization and is capable of simulating complex wall geometries including large toroidal devices (e.g. JET-IWL [1]).

In the present contribution we implement the effect of surface roughness into the ERO2.0 taking advantage of its capability of creating various surface geometries. The predictive modelling is aimed to assist in on-going preparation of experiments at PSI-2 with pre-defined roughness which are going to utilize several measurement technics like quartz-microbalance (QMB), mass loss, spectroscopy and SEM. To use the code on the micro-scale several modifications were introduced including direct incorporation of analytical tracing of incident plasma ions inside the magnetic sheath, various types of the regular structures representative for rough surfaces, angular distributions of the sputtered impurities dependent on the plasma ions incident angle, surface evolution during the plasma exposure due to the physical sputtering and re-deposition. The influence of the surface roughness on the Y_{eff}, re-deposition/prompt re-deposition, angular and energy distribution of the sputtered W in the conditions relevant to the PSI-2 facility ($T_e \approx$ 5-10 eV, $n_e \approx 1\text{x}10^{18}$ m^{-3}, normal B $\approx$ 0.1 T, E_i=30-250 eV) is investigated for several roughness types as a function of surface structures height and width. It is shown that the resulting angular distribution of W coming out of the surface peaks in normal direction to the surface with the roughness depth increasing (collimation). The preferential re-deposition in the valleys and erosion at the hills are leading to the flattening of the surface. It was shown that accounting for sputtered species angular distribution dependence on the sputtering ion incidence angle is responsible for the Y_{eff} decrease by at least 20%.

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Evidence of near-SOL tungsten accumulation using far-SOL collector probes and OEDGE Modeling in a DIII-D Metal Rings L-mode discharge*

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Toroidally symmetric rings of 5 cm wide tungsten-coated tiles were installed in the outer divertor of DIII-D. A collector probe array having multiple-diameter, dual-facing collector rods with axes in the radial direction, was inserted into the far scrape off layer (SOL) near the outer midplane to measure the plasma W content. Although the W source was in the outer divertor near the strike point, for most probes more W was found on the side connected along field lines to the inner divertor, with the largest diameter rod showing the largest divertor-facing asymmetries and the smallest rod showing no asymmetry. This is consistent with simulations showing accumulation of W impurities in the plasma upstream of the probe, as expected theoretically from the force due to the D^+ temperature parallel gradient, FIG, driving W upstream from an outer divertor source [1, 2]. The OEDGE code is used to model the W erosion, transport and deposition for a set of repeat LSN L-mode discharges. Simulations indicate that W accumulates near the apex of the plasma on field lines close to the separatrix primarily due to the balance between friction and FIG. The accumulation and subsequent transport to the outer far SOL where the probe array is located is dependent on the 2D D^+ temperature profile, the 2D D^+ flow velocity parallel to the field lines ($v_{\|}$), the radial transport, and the magnetic connection length geometry between the upstream W accumulation region and the probe array location. OEDGE has been enhanced with (i) extended grids including main chamber plasma wall contact and (ii) a collector probe model based on [3]. Radial transport plays a key role in the simulated collector probe deposition profiles because the probe is not located on the same field lines as the upstream accumulation. Through sensitivity scans, OEDGE approximately reproduces both the shape of the divertor-facing asymmetries and the radial decay of each collector rod profile for $D_\perp \sim 0.3 m^2/s$, a typical value used in SOL modelling. A boundary condition of $T_i = 2T_e$ at the outer target significantly increases the W divertor leakage and upstream accumulation while the modelled W asymmetry on the largest diameter collector probe increases modestly.

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abstract number 128

Bayesian Spectroscopic Analysis of N II Line Emission for the Characterisation of Volumetric Plasma Parameters and Atomic Behaviour in Diverted TCV L-Mode Plasmas

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In large scale tokamaks (ITER, DEMO) radiative losses from impurities will need to account for > 90 % of the power crossing the separatrix to keep the peak divertor target heat flux within tolerable limits, around 10 MWm^{-2}. [1, 2] This is key for steady state operation.[3] Impurities will be necessary for reaching this level of radiative loss. Understanding impurity transport is paramount for maximum impurity radiation in the divertor, with minimum core contamination and efficient pumping of impurities.

Bayesian analysis of atomic spectra can infer the electron density and temperature, and the ion/neutral densities. This produces a posterior distribution, N-dimensional probability distribution function (PDF), which gives the likelihood of a set of plasma parameters given experimental data for a range of points in N-dimensional parameter space. The posterior distribution gives complete information on the parameters and offers advantages over conventional analysis techniques, such as the ability to distinguish between global and local maxima, and being able to rigorously quantify uncertainties in the inferred parameters.

These techniques advance over existing methods of estimating the divertor impurity concentrations. Previous studies have used Collisional Radiative equilibrium (CRE) models, transport/diffusion corrected CRE models, and Metropolis Monte Carlo (MMC) searches of ion/neutral fractions.[4, 5, 6] This work uses a Markov Chain Monte Carlo (MCMC) search of the possible concentrations, which removes the need to make assumptions about the atomic equilibrium and can decouple the concentrations of different charge states of the same atom. Comparison of these concentrations with a CRE model provides an estimate of averaged ion confinement time and could be a source of transport model validation.

This analysis is applied to nitrogen line intensities and ratios from a detached, single null, L-mode TCV pulse to assess how nitrogen ions spread through the divertor leg during seeding and how this impacts the local electron density and temperature.

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Enhancement of helium exhaust during RMP-ELM suppression at DIII-D and analysis with 3-D edge fluid and kinetic neutral code EMC3-EIRENE*

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Resonant magnetic perturbation (RMP) fields used to suppress Edge Localized Modes (ELMs) in high confinement (H-mode) tokamak plasmas were found to lead to a strong enhancement of helium exhaust in recent experiments at DIII-D. The effective He particle confinement time $\tau^*_{p,\,He}$ in ELM-suppressed H-modes was reduced by 40% compared to unsuppressed discharges, and $\tau^*_{p,\,He}/\tau_E$, where τ_E is energy confinement time, was reduced by 10-20%. These first-time findings are important for ITER, where application of RMP fields is planned for ELM control, as they suggest RMP fields can replace the impurity exhaust produced by the ELM events.
Increased helium exhaust was measured for ITER-shaped plasmas at DIII-D using argon-frosted divertor cryo-panels for active pumping of He, which was injected in short test pulses into a deuterium plasma. A multiple-reservoir particle balance model was used for analysis of the experimental data. In both the plasma edge and core reservoirs, midplane He density measurements from charge-exchange spectroscopy show reduced n_{He} and a faster decay time during ELM suppression, suggesting faster outward transport and/or reduced He back-fueling after recycling. Increased He-I and He-II emission in the Scrape-off Layer (SOL) and increased neutral He pressure in the pumping plenum show that more He is retained in the SOL and neutral reservoirs, which is important for effective removal of He from the entire plasma. EMC3-EIRENE fluid plasma edge and kinetic neutral transport modeling of comparable scenarios, in addition to the experimental measurements, suggests two mechanisms behind the beneficial enhancement of helium exhaust. First, reduced parallel temperature gradients due to magnetic field stochastization in the vicinity of the separatrix can increase the friction force acting on impurities relative to the thermal force, which enhances outward transport in the region of the perturbed magnetic field. Second, the evolution of helical lobes, which connect the separatrix region to the divertor via a helical magnetic footprint, yields increased He neutral pressure due to increased divertor plasma plugging. Both effects are being analyzed with dedicated EMC3-EIRENE modeling including plasma response from M3D-C1 extended MHD code, which defines the level of magnetic field stochastization at the separatrix.

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Study on dynamic behavior of impurity transport along the magnetic field in detached plasma using linear plasma device.

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The reduction of heat and particle loads at the divertor in a magnetically confined fusion device is a crucial issue. It is necessary to reduce the divertor plate heat load to 10 MW/m^2. Plasma detachment is one method for reduction of heat and particle loads by using volume recombination processes. A stationary detached plasma is particularly important for efficient divertor performance. The divertor plasma is cooled due to the radiation loss from the impurity and appears to recombine. In the case of impurity seeding, the backflow of impurities is predicted[1]. The detached plasma has a steep electron and ion temperature gradient thus impurity ions are transported from the near-target region to the upstream region by the temperature gradient force[2]. It is important to understand the impurity transport phenomena around the divertor target for the impurity control. In large fusion experiment devices, impurity transport in the divertor region has been investigated by measuring the impurity density and its flow velocity spectroscopically, and studied by the code including many processes such as the friction between impurities and background particles and temperature gradients. However, it has not been clear how individual physical processes affect the impurity backflow. By observing the intensity distributions of impurity line emissions along the magnetic field, we simulated experimentally the impurity backflow using linear plasma devices.

The experiments are performed in the linear divertor simulator TPD-SheetIV[3], which can generate a steady state hydrogen plasma. The electron density and temperature of the plasma are the order of 10^{19} m^{-3} and 10 eV, respectively. The detached plasma is produced by injection of cooling gas in experimental region. The neutral pressure P in the experimental region is adjusted to be between 0.05 and 3.0 Pa with a cooling gas feed. The electron temperature and density, the ion saturation current along the magnetic field are measured by a reciprocating single probe. The profiles of impurity such as Ar along the magnetic field are measured by spectrometer. As a result, the steep gradient of the electron temperature and density are observed in detached plasma. Ar ions transported to upstream which is high temperature region with increasing temperature gradient. In the presentation, we will discuss the impurity transport and profiles of impurity along the magnetic field compared with 1D model. In addition, the experiments in the case of high temperature are conduct in large tandem mirror GAMMA 10/PDX.

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Analysis of the ramp-down phase of JET ILW discharges

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The percentage of disruptions in the JET discharges with the new ITER Like Wall (ILW) configuration at high current can exceed 30%. The understanding of the physics of the causes of disruptions remains an important subject in order to reduce disruptivity, particularly for the ITER-relevant high density and high current operation. The analysis of such scenario performed at JET is presented in this work. We have analysed ramp-down phase of the set of representative high current JET ILW discharges. As a first step we have analysed the experimental data for two discharges: #92437(disrupted) and #92442 (soft landing) characterized with high plasma current I_p = 3.5MA. Both shots are similar before the start of the termination of the discharge at 13.9s, but shot #92437 develops a hollow temperature profile which runs away to disruption. The main question is whether there is a difference in impurity content or critical plasma parameters before the start of termination or a difference in impurity source or transport during the termination phase appears and that causes the discharge to diverge. The analysis is performed for four different time slots at the ramp-down phase: t = 13.9s; 14.25s, 14.5s; and 14.75s, corresponding to different levels of the electron line density and auxiliary heating power.

Since the energy balance in tokamaks with tungsten divertor depends strongly on the coupling between bulk and the SOL plasma, proper modelling requires joint treatment of both regions. Therefore our approach is based on integrated numerical modelling of plasma parameters using the COREDIV code, which self-consistently solves the 1D radial transport equations of plasma and impurities in the core region and 2D multi-fluid transport in the SOL. In particular, we have studied the influence of the plasma heating and plasma density on the impurity production and transport during plasma termination phase. Since the deuterium gas flux are different, the influence of the separatrix density is also analysed.

The main conclusion from the preliminary simulations is the observation that for the same average electron density, a decrease of the separatrix density leads to an increase of the plasma temperature at the divertor plate leading to increased W production and consequently to larger W concentration and radiation in the core. When the central electron temperature approaches the 2keV level, corresponding to the maximum of the W cooling rate, enhanced radiation in the plasma center occurs which might be the reason for the experimentally observed hollow profiles of the electron temperature.

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Measurements of the impurity flow velocity and temperature in a wide plasma parameter range for Deuterium and Hydrogen plasmas in the divertor legs of the stochastic layer in LHD

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The impurity transport in the edge plasma is still not fully understood, yet its strong influence on the performance of the core plasma as well as on the tritium retention becomes more evident. This paper presents carbon impurity flow velocity and temperature measurements along the divertor legs for a wide density range $n_e = (1\text{-}14)\times10^{19}\ m^{-3}$, central electron temperature $T_e = 1.5\text{-}3.5\ keV$, and with total neutral beam injection (NBI) power of 9-12 *MW*. The isotope effect on the transport is studied in hydrogen (*H*) and deuterium (*D*) discharges.

For all selected discharges, the carbon ions flow towards the divertor in both *H* and *D* plasmas. Different velocities are obtained depending on the charge states. For C^{+} and C^{2+} they are in range 10-30 *km/s*, and 5-20 *km/s* for C^{3+} ions. It is also found that there is no change of flow direction even in the lowest density, where the impurity transport model predicts flow toward upstream in the thermal force dominant regime [1]. Flow velocities for *D* plasma are systematically slower, by the factor of 1.4-2, compared to *H* plasma. In *H* discharges velocities increase proportionally to the plasma density, while in *D* discharges this dependency is weaker. Temperatures of carbon ions in *H* and *D* plasmas remain same and are in the range 20-60 *eV*. Temperatures of C^{+} and C^{2+} increase slowly with n_e, while C^{3+} temperature decreases. Local plasma parameters, n_e and T_e, are calculated with helium and carbon collisional-radiative models [2,3]. The force balance and momentum transfer from fuel to impurity and isotope effect are discussed. The results are discussed in comparison with the impurity transport model in Scrape-Off Layer/Divertor simulations.

Flow velocities and temperatures of impurities are estimated by analysing the Doppler shift and broadening of the plasma emission lines, respectively. The following bands are used for analysis: C^{+} (CII), 515.1 *nm*, C^{2+}(CIII), 464.7 *nm*, C^{3+}(CIV), 465.7 *nm*, 580.1 *nm*, 772.6 *nm* [4].The Echelle spectroscopy is chosen for these measurements because of its wide range, ~ 409-801 *nm* in our case, and a good wavelength resolution, ~ 0.02 *nm*. This tool makes it possible to perform various analysis of the plasma light emission, useful for both impurity and fuel transport investigations. Two single channel Echelle spectrometers are used in this work, with the viewing area located at the divertor leg, almost tangential to the magnetic field lines.

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Abstract withdrawn

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Abstract Withdrawn

The calculation of radial electric field and its effect on divertor heat flux by using BOUT++ transport code

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Based on the C-mod discharges, the steady state radial electric field (Er) is self-consistently calculated from the plasma transport model with the quasi-neutral constraint using vorticity formulation under the BOUT++ framework. The effective particle and heat diffusivities can be determined from transport equations with sources only from inner radial boundary based on the experimentally measured plasma profiles inside the separatrix, and the profiles in the SOL which are getting by given a constant absolute value decay from separatrix. The results emphasize the effects of different physical models on the Er calculation. Outside the separatrix, the divetor sheath potential is dominant, which will form a larger positive Er. While inside the separatrix, Er is determined by the force balance where the ExB flow is balanced by the diamagnetic flow. The curvature drift does have a significant effect on the potential and Er though their contributions to the net flow in both the core and SOL region across the separatrix. The simulation results have been validated with the experiment.

Steady state solution of heat flux for both electrons and ions is calculated by this plasma fluid transport model with all drifts to mimic Goldston's Heuristic drift-based model for the power scrape-off width. The transport simulations show a similar trend to the Goldston's HD model. The electron heat fluxes dominant over ion heat flux, however ion heat fluxes become dominant when Bp increases. Magnetic drift has a significant influence on the heat flux width which can reduce it by 2~3 times, and ExB drift decreases the heat flux width by 10%~25% to improve Goldson's model.

Numerical study of Li species transport in edge plasma during lithium granule injection in the frame of BOUT++

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Application of Lithium (Li) in many fusion devices has manifested that it can facilitate plasma confinement, for example, it can reduce recycling and H content in the deuterium plasma, and suppress impurity accumulation in core plasma. In particular, from EAST and DIII-D experiments of Li pellet injection into the pedestal, it was found that a high frequency edge localized modes (ELMs) could be obtained; that even ELM-free regime could be realized. Pellet pacing ELMs is one of two baseline methods of controlling ELMs techniques for ITER. In comparison, the use of non-fuel pellets is desirable for ITER due to its decoupling ELM pacing from fuelling. However, there is little knowledge about how Li species released by Li pellet transport and evolve in the edge plasma. This information is important for further understanding how Li pellets trigger ELMs with MHD codes. In previous works we have developed an impurity transport module in the frame of BOUT++ [1] and also applied the neutral gas shielding model [2] to the pedestal region of a typical H-mode plasma [3]. In this work, we will study how Li species released from an injected Li pellet with a various size at a various velocity using a model by integrate these previous two parts of work [1, 3]. The results will be compared with those obtained by SOLPS at the same conditions. The work is now going under way. Detailed results and discuss will be presented in the coming submitted paper.

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High re-deposition ratio of high-Z metals under plasma exposure in Magnum-PSI

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Excessive core plasma contamination in ITER or DEMO by erosion and inward transport of impurities from plasma facing components (PFCs) will lead to an undesirable reduction in fusion power. This therefore provides an upper limit on the gross impurity flux from the wall materials entering the core plasma. A high local re-deposition rate of eroded material can minimize core impurities and increase the lifetime of a PFC by reducing the net erosion rate. Until now there was rather limited experimental data on the localized re-deposition rate measured under ITER/DEMO divertor relevant plasma fluxes, where very high incoming particle fluxes and low ion and electron temperatures are anticipated. In this work dedicated studies to investigate the re-deposition ratio of several different metals (copper (Cu), molybdenum (Mo) and tin (Sn)) under a high flux plasma beam were carried out in Magnum-PSI [1]. Sn is considered as a prospective material for use in a liquid metal divertor in DEMO and previously indicated a high re-deposition rate [2,3]. Cu was chosen as a metal with a relatively high sputtering yield and which can visually indicate the presence of impurities on its surface, while Mo was used as a reference where sputtering should be negligible.

Samples were exposed to high particle fluxes of 0.3−8.5x10^{23} $m^{-2}s^{-1}$ (B=0.2-1.0 T) in argon and helium plasmas with electron temperatures in the range 0.6–2.1 eV and with ion energies ranging from 38 to 59 eV. X-ray Photoelectron Spectroscopy identified that Mo from the clamping ring was the predominant impurity deposited, while Rutherford Backscattering Spectroscopy was used to determine the total mass gain from this deposition. Controlling for this impurity deposition, the re-deposition rate was determined via two methods: mass loss measured by micro-balance after exposures and mass gain on a quartz crystal microbalance measured during exposures. Both demonstrated a high level of consistency. For both Cu and Sn re-deposition rates greater than 99% were determined at high flux while Mo showed negligible erosion as expected. Considering the high plasma density and low electron temperatures plasma entrainment of neutral impurities is considered the most likely process which could cause such a high re-deposition rate, and mean free paths of a few mm are estimated. Such high rates imply that large gross erosion rates could be acceptable in DEMO. Particularly in the case where evaporation is the dominant erosion mechanism, which is the case for liquid metals such as lithium (Li) and tin (Sn), such a high re-deposition rate could also increase the temperature window for operation of such a PFC [4].

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Study of molybdenum impurity transport in EAST L-mode plasma with central electron cyclotron heating

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Effects of central electron cyclotron heating or ion cyclotron resonance heating on core impurity behavior and suppression of impurity accumulation have been observed in various tokamaks [1-3]. It has been proposed that the change in the temperature gradients during radio frequent heating may drive turbulent transport that is responsible for anomalous impurity transport [4].

In this paper, core metal impurity behavior in electron cyclotron resonance heated (ECRH) L-mode plasma of the EAST tokamak was investigated based on space-resolved ultraviolet spectroscopy and impurity transport analysis. Significant reduction of the core Mo^{30+} emissivity was observed during on-axis ECRH injection, accompanied with the decreasing of core ion temperature and the increasing of electron temperature. However, low-Z impurity ions such as C^{5+} and Li^{2+} increase when the ECRH is switched on which means a opposite trend as Mo^{30+}. Mo^{30+} is mainly located in the core area due to its high ionization energy (Ei= 1726 ev) and C^{5+} (Ei=392 ev) and Li^{2+} (Ei=122 ev) are mainly located at edge area, which means that suppressing of impurity accumulation by on-axis ECRH is very localized. The electron density reduction called 'particle pump-out' is well known as a typical behaviour during the ECRH. The impurity transport study before and during ECRH injection was performed by using line emissions of Mo XXX, Mo XXXI and Mo XXXII and the molybdenum transport coefficients were determined by reproducing the profiles of molybdenum densities to the experimentally measured results with a one-dimensional empirical impurity transport code, STRAHL.

The diffusion coefficients were found to increase substantially and inward convective velocity decreased and even change its direction in the core area (ρ<0.5) during on-axis ECRH injection, consistent with suppression of Mo impurity accumulation. Neoclassical impurity transport coefficients were also calculated and was found to be one order of magnitude lower than that of the experimental diffusion coefficient, which indicated that neoclassical transport is not responsible for the suppression of impurity accumulation, and the increase of diffusion coefficient and decrease of inward convective velocity might be possibly due to enhanced turbulent transport during the central electron heating.

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Impurity transport simulation in the peripheral plasma in the Large Helical Device with tungsten closed helical divertor

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Recent long pulse plasma discharges in the Large Helical Device (LHD) have been terminated by the radiation collapse induce by the incoming large amounts of carbon dusts released from closed helical divertor regions [1]. The exfoliation of carbon dominant mixed-material deposition layers accumulated in front of the carbon divertor plates (made of isotropic graphite) formed the large amounts of carbon dusts in the long pulse discharges. In order to suppress the carbon dust emission, the change of the divertor configuration to a full tungsten closed helical divertor one is now in the planning stage.

In the long pulse plasma discharges, sudden increase in the content of iron ions in the plasma has often been detected by impurity monitors, which is caused by plasma-wall interactions on the vacuum vessel (made of stainless steel) or by electric arcing between metallic components in ICRF antennas used for plasma heating and sustainment [2, 3]. The iron ions can induce the physical sputtering on the tungsten divertor plates, and the sputtered tungsten atoms are ionized in the peripheral plasma. It can bring about the catastrophic increase in the content of the tungsten ions by self-sputtering on the divertor plates, which leads to the interruption of long pulse discharges by radiation collapse.

For investigating the merits and demerits of the full tungsten divertor, a three-dimensional edge plasma transport simulation code (EMC3-EIRENE) has been applied [4, 5]. The code was modified to include the transport of iron and tungsten atoms/ions in the plasma, and the physical/self-sputtering of tungsten on the divertor plates. The modification enables to study the influence of the sputtered tungsten atoms on the sustainment of the plasma discharges in detail, which can contribute to finding operational windows for eliminating the catastrophic increase in the tungsten ions in the peripheral plasma.

In the conference, the authors will present the simulation results of the following two topics:

- On the operational windows (the plasma heating power and the plasma density) for minimizing the influence of the sputtered tungsten on the plasma,
- On the threshold density of iron ions in the peripheral plasma which induces the catastrophic increase in tungsten ions by the self-sputtering on the tungsten divertor plates.

The simulation can not only reveal the prospect of the full tungsten closed helical divertor for extending the duration time of long pulse discharges in the LHD but also indicate the optimized material configuration of the plasma facing components for sustaining steady-state plasma discharges in ITER and future nuclear fusion reactors.

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Spectral modeling of tungsten transport based on a compact advanced extreme ultraviolet spectrometer system for KSTAR

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In this study, we report a newly developed spectral model of tungsten transport for KSTAR plasma. Since tungsten (W) will be used as a plasma facing component on the ITER divertor and is presently employed in several devices including ASDEX-U, JET and WEST, study on the behavior of W impurity in fusion plasmas has become an essential issue. W has low sputtering yield and low erosion rate but it strongly radiates once it penetrates into the plasma, potentially resulting in severe radiation cooling. In order to prevent such case, the first step of the study should focus on monitoring W and the next should be a study of its transport to minimize the W accumulation inside the plasma. In KSTAR, a compact advanced extreme ultraviolet (EUV) spectrometer system (CAES) [1] was installed to measure spectra of W ions in EUV region where the charge states W^{27+} to W^{45+} lie. Spatially-resolved W spectra were successfully measured by the CAES which has 2.7 cm and 67 ms of spatial and time resolutions. Poloidal asymmetry of the W radiation was also observed by the X-ray pinhole camera based on the GEM detector and infrared video bolometer (IRVB) in the last KSTAR campaign. In order to interpret this set of measurements, a time- and space-resolved spectral model has been developed. The model calculates the intensity of many spectral lines and features of W in the given electron temperature and density profiles as well as their brightness, considering the geometry of the tokamak and the viewing lines of the spectrometer. Atomic Data and Analysis Structure (ADAS) was utilized to obtain ionization and recombination rates for the ionization equilibrium calculation. In order to find the global W density profile in the plasma, the continuity equation including diffusion and convection coefficients, radiation power loss relation [2] and the force balance equation with a centrifugal force effect [3] were individually considered. Flexible Atomic Code (FAC) was used to calculate the photon emission coefficient (PEC) of W^{10+} to W^{48+} allowing to determine the time- and space-dependent brightness. The modeling result of spatially-integrated quasi-continuum emission of W in 2 – 7 nm wavelength showed a good agreement with the measured data from the W injection experiment in KSTAR within a 0.2 nm error. The addition to the W model of light impurities such as carbon and argon is underway, which will help studying the charge dependence of impurity transport and comparing the previously reported results.

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Numerical study of Li species transport in edge plasma during lithium granule injection

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Experiments of lithium (Li) pellet pacing edge localize modes (ELMs) have been carried out on a few tokamak devices, and have obtained many positive results. To the end of grasping the physics behind Li pellet ELMs pacing, we should first understand how Li species distribute and transport in the edge plasma during a Lithium pellet injection. Based on our previous work[1] developed from neutral gas shielding model, the initial amount of Li species released from an injected Li pellet inside the separatrix can be evaluated over the space and time domain. In a short time of an order of 10^{-7} seconds, Li atoms released from a moving pellet hardly move and very few of them are ionized. With the source of these Li atoms as input to the code package SOLPS 5.1, the transport of Li species as well as the evolution of background plasma can be simulated for a specified magnetic field configuration. While a Li pellet of size ~1mm moves inward the separatrix of a typical H-mode plasma, the global temperature of electrons in background plasma feels a perturbation first, and then resumes its original state around in a time of around 5 μ s, but the global electron density responds much more slowly, rising first and falling to its unperturbed state in a time period of over 1ms. The temperature of electrons close to the separatrix at the outer mid-plane drops monotonically when the pellet passes by, while the electron density increases first and then decreases to a value that maintains the product of the electron temperature and the electron density temperature ($p=n\times T$) more or less the same as before, which indicates there is a sudden pressure rise and fall resulting from the injected pellet but the pellet does not change the pedestal features greatly. It is also found that an increase on the ion radial thermal transport coefficient during the ablation has happened in accompany with an electrostatic energy perturbation during the simulation, which indicates that rapid radial transport of energy and particles has taken place. The simulation results also show that most lithium species, mainly in forms of cations (Li+ and Li2+), deposit inside the separatrix; that almost all the neutral Li atoms (accounting for a tiny percentage of the total of Li species) diffuse outside the separatrix. More simulations are under way and detailed results and discussion will be presented in the submitted paper.

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Investigation of the impurity transport in edge plasma during the injection of the lithium powder in EAST by using EMC3-EIRENE

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The transport property of the edge impurity is one of the most significant topics for the long-pulse discharge operation of the magnetically confined fusion devices. A certain level of the impurity in the confined region would lead to strongly decreasing the available fusion power [1]. On the positive side, the radiation loss of impurity in edge region could contribute to the temperature reduction of the edge plasma and then drastically reduce the heat flux impacting onto the divertor target. Considering the lithium element as the low-Z materials, the lithium impurity would radiate mainly at the edge region characterized for the low plasma temperatures. In addition, the lithium impurity has the capability of improving the plasma performance and also availably removing high heat flux in the fusion devices [2]. Based on these reasons, the relevant experiments of the lithium powder injection have been designed and tested on the EAST tokamak. However, the transport mechanism of the lithium impurity in the edge plasma is still unclear.

In this work, on the basis of the EAST discharge experiment during the lithium powder injection, the transport characteristics of the lithium impurity in the scrape-off layer (SOL) of EAST is studied by the three-dimensional (3D) edge transport code, the EMC3-EIRENE [3,4]. The EMC3 code solves the fluid equations for particles, parallel momentum, and energies of electrons and ions, and the code is coupled with the neutral transport code EIRENE. In addition, The EMC3-EIRENE code has the tracing impurity model, which consists of equations of the continuity and the force balance. In this modelling, the lithium powder is injected from the top puffing port of the EAST. A comparative analysis of the impurity transport and the emission intensity, which is based on the simulation results and the experimental measurement, will be presented. Furthermore, the experimental results show that the sufficient lithium impurities could lead to the plasma disruptions [2]. Therefore, the influence of the injected amount of the lithium powder on plasma performance is also investigated in this work. The distributions of the heat flux impacting onto the divertor target with different amounts of the injected lithium impurity are studied in this modelling.

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Characterization of tungsten divertor sources and upstream edge plasma contamination using isotopic tracers in the DIII-D metal tile campaign*

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Experiments have recently been carried out with complete toroidal arrays of metal inserts coated with isotopically distinct tungsten (W) at two different poloidal locations in the lower outer divertor of DIII-D. The purpose of the experiments is to localize high-Z divertor erosion and migration near the outer strike point (OSP) and far-target in a mixed-material environment, i.e., carbon (C) and W, in ELM-y H-mode discharges. Particular to this work, the experimental setup employed measurements of both upstream collector probes (CPs) in the far-SOL and ELM-resolved spectroscopic measurements of gross W sourcing (W-I) with high spatial resolution providing new information on the transport link between different W divertor source locations and the SOL W content. It is found that the total amount of W deposited on the CPs shows a dependence on ELM frequency (f_{ELM}), strike point location, and position of the CP in the far SOL. Specifically, the far-target W source had minimal upstream deposition on the CP, i.e. within the uncertainty of the measurement, when f_{ELM} ~ 150HZ and delW/W ~ 0.5%. One the other hand with bigger ELMs (delW/W ~ 4%) with f_{ELM} ~ 50 HZ, the far-target W source deposited on the upstream CPs increased ~ 4x.
Ex-situ surface analysis techniques are applied to the CPs with inductively-coupled-plasma mass spectroscopy (ICP-MS) used to quantify the deposition of W isotopes and Rutherford backscattering (RBS) used to determine the total areal W. The CP radial W profiles are comparable in shape and magnitude to those obtained from other CP measurements, e.g. ASDEX-U divertor W ring experiments [1], and therefore gives confidence the probes were generally in the far-SOL. The isotopically marked W profiles are discriminated using a Stable Isotope Mixing Model [2] based on the ratio of isotopic pairs compared to the isotopic make-up of each W source. This technique allows determination of the fraction of W from each divertor source to within ~ 5%, and thereby these experiments separate the W sourcing between the OSP target (which had a natural W isotopic abundance) and far-target (which had an enrichment of the ^{182}W to ~ 93%). Under H-mode conditions the ^{182}W isotopic fraction (i.e. far-target source) on the CPs increased by up to 3x the natural isotopic levels, i.e. from 26.5% →~ 83% in some discharges, but typically the ^{182}W fraction was ~ 40%. It is also found that the fraction of W from each W source on the CP is directly proportional to the total amount of sputtering (ELM-averaged) during the CP exposure, and thus provides further evidence that ELMs with a larger wetted area contribute more to far-target SOL impurity contamination.

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Metallic impurity production during ICRH in EAST

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Waves in the Ion Cyclotron Range of Frequencies (ICRF~10→100MHz) are used on many tokamaks to heat the plasma. Impurity production is however often observed during ICRF wave injection, particularly p high-Z plasma facing components. Among the processes at play, RF sheath rectification is likely the main mechanism by which ICRF enhances impurities production. Although the wave-launching antennas and magnetically connected objects are likely contributors, the spatial origin of impurities during ICRH still remains unclear. This study takes profit from the diversity of materials in the Experimental Advanced Superconducting Tokamak (EAST) to shed some light on what happens where.

Using an Extreme UltraViolet (EUV) spectrometer, lines brightness of various high-Z impurities are investigated, each one characteristic of plasma-material interaction in a relatively precise location in the machine: iron (Fe) for antenna Faraday screens (underneath boron carbide coating), tungsten (W) for the upper divertor and molybdenum (Mo) for one half of the inner wall. In addition to these components of the EAST vessel, two passive plates in titanium (Ti) were installed on purpose at one toroidal location in the outer midplane to generate trace impurities. Ti was found appropriate since it can be used as a permanent plasma facing component with negligible impact on vacuum. Besides, Ti spectral lines can be measured by the EUV spectrometer and easily distinguished from other species. EAST is equipped with two ICRF phased strap arrays. Only one of them (B port) is magnetically connected to the Ti plates, while only the other one (I port) faces the Mo inner tiles. Spectroscopic signals are studied on L-mode plasmas in upper single-null configuration. Relative variations of line brightness are monitored over scans of lower hybrid (LH) and ICRF power by several combinations of both antennas, with several toroidal phasing between straps. They are here interpreted as a change in the sources of impurities rather than their transport.

The intensities of the different spectral lines, corrected for variations of the line-integrated density, exhibit different parametric variations over the scans. Materials closer to the antennas show much better correlation with ICRF parameters. In particular the Ti line reacts preferentially to the magnetically-connected B-port antenna. W line intensity (from upper divertor) increases with additional power from either ICRF or LH antennas. Mo line reacts preferentially to the I-port antenna facing high field side's Mo tiles. The analysis is complicated by the different designs of the two ICRF antennas: these can induce different levels of impurity production, independent of their spatial location in the machine.

Different potential mechanisms producing metallic impurities will be discussed. For shorter distances antenna-object, RF fields are stronger and plasma-material interactions seem dominated by RF sheaths. "Far field" sheaths, as well as power convected near the divertor strike points, could also play a role at other locations. Further experiments and RF-sheath modelling are engaged, aiming at disentangling roles of magnetic connections from antennas structural differences. Replacing titanium by an isotope of tungsten different from the one in the divertor region could also make the study more relevant for ITER objectives.

Investigation of the impurity transport in edge plasma during the injection of the lithium powder in EAST by using EMC3-EIRENE

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The transport property of the edge impurity is one of the most significant topics for the long-pulse discharge operation of the magnetically confined fusion devices. A certain level of the impurity in the confined region would lead to strongly decreasing the available fusion power [1]. On the positive side, the radiation loss of impurity in edge region could contribute to the temperature reduction of the edge plasma and then drastically reduce the heat flux impacting onto the divertor target. Considering the lithium element as the low-Z materials, the lithium impurity would radiate mainly at the edge region characterized for the low plasma temperatures. In addition, the lithium impurity has the capability of improving the plasma performance and also availably removing high heat flux in the fusion devices [2]. Based on these reasons, the relevant experiments of the lithium powder injection have been designed and tested on the EAST tokamak. However, the transport mechanism of the lithium impurity in the edge plasma is still unclear.

In this work, on the basis of the EAST discharge experiment during the lithium powder injection, the transport characteristics of the lithium impurity in the scrape-off layer (SOL) of EAST is studied by the three-dimensional (3D) edge transport code, the EMC3-EIRENE [3,4]. The EMC3 code solves the fluid equations for particles, parallel momentum, and energies of electrons and ions, and the code is coupled with the neutral transport code EIRENE. In addition, The EMC3-EIRENE code has the tracing impurity model, which consists of equations of the continuity and the force balance. In this modelling, the lithium powder is injected from the top puffing port of the EAST. A comparative analysis of the impurity transport and the emission intensity, which is based on the simulation results and the experimental measurement, will be presented. Furthermore, the experimental results show that the sufficient lithium impurities could lead to the plasma disruptions [2]. Therefore, the influence of the injected amount of the lithium powder on plasma performance is also investigated in this work. The distributions of the heat flux impacting onto the divertor target with different amounts of the injected lithium impurity are studied in this modelling.

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Preliminary study on Ne transport in EAST H-mode plasma

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Impurity accumulation in the core plasma can lead to serious ion dilution and energy loss. To find the effective methods of controlling impurity in the core plasma, the investigation and the understanding of impurity behavior, is therefore of utmost importance, which is also urgently desired for the Experimental Advanced Superconducting Tokamak (EAST) operations. Normally, the impurity transport could be studied by introducing transient perturbations of impurity into plasma by pellets injection (PI), laser blow-off (LBO), gas puffing (GP) or supersonic molecular beam injection (SMBI) system [1-2].

In EAST experiment neon has been externally introduced to H-mode plasma with GP and SMBI systems to study the impurity transport. For this purpose, two fast-time-response extreme ultraviolet (EUV) spectrometers working at 20-500Å and 10-130 Å respectively have been newly installed to monitor the impurity behavior in EAST plasma [3-4]. Based on the spectra measurement by the two spectrometers the line emissions in the wavelength range of 10 to 180 Å from NeIV – NeX have been identified. In this work, the behavior of Ne ions after injection with GP and SMBI are investigated and compared with each other and it is found that the SMBI is more adapted for impurity transport study. Therefore, the transport of neon which has been introduced by SMBI in EAST H-mode plasma is studied with one-dimensional impurity transport code combined with the experimental results, e.g. the transport coefficients of neon are determined by reconstructing the time evolution of identified neon line emissions according to the geometry of the diagnostic. In addition, the transport coefficients are calculated with neoclassic model NEO.

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Material Erosion, Migration, Mixing, and Dust Formation

Calculation of mechanical stresses and deformations near crack caused by pulsed heat load

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The pulsed heat loads cause the formation of cracks in tungsten. Several studies demonstrated formation of parallel to surface cracks resulting in the overheat due to the reduction of heat removal [1]. However, the calculation of residual mechanical stresses after pulsed heat load gives only tensile stresses along the surface leading to formation of perpendicular to exposed surface cracks [2]. The tensile stresses in perpendicular to surface direction may be result of the stress release around the perpendicular to surface crack. The problem of linear elasticity was solved for this situation. The two-dimensional differential equation was reduced to one-dimensional integral equation and solved numerically.

The first result of the calculations is the fact that the perpendicular to surface crack is reason of formation of stresses that tear off the surface layer. The development of the stresses may lead to delamination of the surface layer and formation of parallel to surface crack, decreasing of heat dissipation into the bulk. Moreover, the perpendicular to surface crack is not only the reason of the delamination stress formation, but also the end of crack may be a raiser of the stresses.

The second result is the calculation of the deformations around the crack on the surface. It was demonstrated that the edges of crack on the surface are elevated. The ratio of the elevation to the crack width in the absence of parallel to surface cracks was calculated to be π^{-1}. Potentially the significantly more ratio may be used as a sign of the parallel to surface crack formation. At the same time the width of the elevated area is about the crack depth. It makes this type of roughness more noticeable than the cracks.

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High heat loads producing large size dust particles in Alcator C-Mod

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Plasma facing units in the ITER divertor will be formed with chains of tungsten monoblocks (MB). One identified key issue is linked to the MB misalignment. Under cycles of heat loads and transient high heat loads, the MB leading edges could be melted and induce molten W splashes or droplet emission [1,2]. Resulting surface damage could compromise plasma operation by changing the W mechanical structure and by reducing the MB lifetime.

One effect of the tile misalignment was evidenced in the full-metal tokamak, Alcator C-Mod. During plasma operation, camera videos have shown an over-light emission of various leading edges of plasma facing components (PFC). These regions were destabilized during disruptions and led to an emission of molten material droplets across the vacuum chamber. Resulting typically rounded dust particles such as splashes and spheres were collected after 2007 and 2015 plasma campaigns. Dust under consideration was either in Mo or B, or in Mo/B alloy. The toroidal row of W tiles, which was inserted in 2007 in the strike point region of the lower outer divertor, also provided W and W/B dust.

Dust particles of composition mentioned above were selected with energy dispersive X-ray spectroscopy mapping in order to reject debris and dust of other composition. The main characteristic of the selected particles was their large size [3]. While the average size of W-dominated dust particles produced by arcing in ASDEX Upgrade is ~ 2 μm [4,5], the most probable size in Alcator C-Mod is ~ 50 μm and the largest can reach 450 μm.

To reduce the emission of molten material droplets during plasma operation, a slight rotation (~1° tilt) of all the modules of the low outer divertor was done in 2015 in order to shadow their leading edges. The weight of dust collected in the same sectors at the end of 2007 and 2015 were measured as obtained i.e. also with dust-debris of other composition. Within this limit, the dust weight in 2015 was 3 to 6 times lower than the dust weight coming from the same sectors in 2007. These results, added to the fact that the average energy injected in 2280 discharges in 2015 was 0.66 MJ/discharge against 2026 discharges in 2007 with 0.45MJ/discharge indicated that less dust was produced in 2015. The size distribution, which however remained similar to that of 2007, indicates the same origin mechanism.

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Angle-dependent sputter-yield measurements of keV D ions on Fe and W with a new high-current ion source

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SIESTA (Second Ion Experiment for Sputtering and TDS Analysis) is a newly built high current ion source used for research on wall materials for nuclear fusion devices. The system is composed of a DuoPIGatron [1] type ion source operating in steady-state, three grids for extraction and focusing, a differential pumping stage, a dipole bending magnet and the target chamber with a magnetic suspension balance.

The ion source can be set to an acceleration potential of up to 15 kV and can be operated with hydrogen, deuterium, helium and argon. Respectively, the grids serve for plasma limiting, ion beam focusing and acceleration to the desired potential. The operating pressure in the differential pumping stage is in the $10^{-4} – 10^{-5}$ mbar range, leading to a space-charge compensated beam. The water-cooled dipole magnet deflects and mass-filters the beam, and is capable of handling a 10 keV Ar^+ beam. The target chamber has a base pressure < 10^{-8} mbar, and an operating pressure < 10^{-6} mbar. The target can be rotated to study angle-dependent effects. It can be positively biased to facilitate exposure to ion energies as low as 200 eV. Implantation at temperatures up to 1300 K is made possible by electron impact heating. The magnetic suspension balance allows for in-situ sputter-yield measurements.

Ion currents at the target of 10 to 80 μA with D_3^+ ions were achieved for accelerating voltages of 2 kV and 10 kV respectively. In order to accurately determine the particle flux density to the surface, the area of the beam footprint on the sample was determined by exposing amorphous hydrocarbon thin films under identical experimental conditions and mapping the erosion crater by ellipsometry. The corresponding beam footprint area is 0.5 cm^2, signifying particle flux densities of up to 3×10^{19} $D/m^2/s$ for 10 keV D_3^+ ions.

As part of ongoing research on the influence of surface roughness on the sputter yield, nanometer and sub-nanometer smooth samples with slope ≤ 20° were produced by depositing thin Fe and W films on a Si substrate via magnetron sputtering. The surface morphology was determined by atomic force microscopy. In order to modify the surface structure as little as possible, the samples were exposed to low fluences of the order of 10^{22} D/m^2 by a 6 kV D_3^+ ion beam (2 keV/D) under varying angles of incidence ranging from 0° to 75° with respect to the surface normal. The layer thickness was measured by Rutherford backscattering with 2 MeV ^4He ions before and after erosion.

The resulting sputter-yields are compared to SDTrimSP simulations [2, 3], agreeing in all cases on the dependence of the sputter-yield on the incidence angle and, in the case of Fe, also on the absolute amounts. No dynamic evolution of the sputter-yield was observed in the simulations. There are deviations between the absolute values of the sputter-yield for W and the simulations, which is a known issue in SDTrimSP. The sputter-yields for W are compared to literature data, agreeing well with previous measurements at normal incidence.

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Impact and electrostatic remobilization of W dust on tokamak plasma-facing components

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Dust particles have been observed in fusion devices for decades. They can feed the plasma with impurities, thereby lowering its performances [1]. Correct depiction of dust/wall interactions is crucial to quantify dust sources and sinks in tokamaks. The Thornton and Ning model [2] for the normal impact of elastic-perfectly plastic spheres is implemented in the dust transport code MIGRAINe [3]. Another approach has been recently introduced in the CEA dust transport code DUMBO [4]. A 3D JKR-based discrete-element method [5] accounting for both viscoelastic effects and the variation of the adhesion velocity with the impact angle is used. Up to now, these effects were neglected in the models used for particles transport in fusion devices. In this paper, the tokamak-relevant consequences are discussed. We observe a significant increase in the adhesion velocity, the threshold which distinguishes between adhesion and rebound, along with an important reduction of this adhesion velocity for grazing impact angles.

As the counterpart of dust/wall collisions, remobilization can represent important dust sources. Extensive literature exists on both adhesion forces and the external forces that can lead to remobilization. We study the possibility of dust remobilization via electric forces generated by two different phenomena: (i) typical sheath potential drops and (ii) electron emission-induced electric fields in the absence of plasma. The experiments reported in [6-7], where W dust resuspension was performed with static electric fields, suggest that typical sheath potential drops encountered in tokamaks can remobilize W dust. In this paper, the Van der Waals-based Rumpf model [8] is shown to depict correctly the adhesion force when compared to data in [6].

The electric force generated by electron emission is estimated and applied in the cases of photoemission and thermionic emission. The expression is validated against experimental data from [9] where dust grains are removed from a W surface using UV pulsed lasers. Yet, when applied to tokamak conditions, it is shown that photoemission cannot lead to dust remobilization since photon fluences from typical plasma events (such as disruptions) remain much lower than that of UV lasers. However, when applied in the case of thermionic emission, we show that electron emission-induced electric fields can remobilize W dust of size $\lesssim 1$ µm.

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Abstract Withdrawn

Influence of glow discharge wall conditioning on the performance of ITER first mirrors

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Metallic first mirrors (FM) will be used as plasma-viewing elements for most of the optical diagnostic systems in ITER. The possible degradation of their surface reflectivity as a result of material deposition can severely impact diagnostic functionality. Simulations of FM deposition are associated with very large uncertainties and experiments are required to provide data for comparison. An example of such an effort is the extensive FM tests conducted at JET since 2006 [1]. Results from an additional experiment, in which a proxy for the ITER aperture/FM geometry has recently been exposed over a full JET plasma campaign [2], are currently being analysed. An issue with this experiment was the inability to include a shutter system, preventing the separation of glow discharge conditioning (GDC) and tokamak plasma exposure. Key physics issues are the degree to which energetic GDC ions impacting aperture surfaces and FMs drive erosion/deposition on the mirrors, and the related question of possible GDC plasma penetration into the apertures.

To support the interpretation of the JET mirror proxy exposure and to provide data for modelling [3] of the GDC plasma behaviour in the vicinity of diagnostic apertures, a series of hydrogen GDC experiments have been performed in a test chamber, for varying aperture geometries and for two combinations of aperture and chamber wall material (aluminium and stainless steel). Three different geometries were studied: a replica of the ITER proxy exposed in JET, a cylindrical and a conical aperture. In each case, a polished mirror was installed at a given distance behind the aperture. The aperture side walls are equipped with Langmuir probes to measure the impacting ion current, j_{ion} and a quartz crystal microbalance (QCM) at the mirror location was also used in some exposures to monitor the deposition rate. The gas pressure was varied in the range 0.1-5 Pa, the anode voltage was 500 V and the average wall ion current density was ~0.1 Am^{-2}, similar to typical tokamak GDC parameters.

For all geometries, j_{ion} decreases with distance from the aperture entrance. The higher the pressure the larger the relative ion current drop. In the range of 0.4-5 Pa, j_{ion} at any location increases with pressure. An increase of j_{ion} is, however observed at 0.1 Pa, and is explained by the transition of the GDC to a hollow cathode discharge inside the aperture. For the JET-like system, virtually no deposition was observed after a 100 hour GDC, as confirmed by SIMS and EDX analyses of the mirror surface. For the cylindrical and conical apertures, net erosion of the mirrors was measured by the QCM for all conditions. This suggests that although the flux of ions penetrating into the structure is only a small fraction of the GDC flux, this is sufficient to prevent the formation of deposits on the mirrors. The implications for ITER first mirrors will be discussed.

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Spatial distribution of dust events in ASDEX Upgrade studied by fast imaging

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Fast video data recorded from 2008 to 2013 in full-tungsten ASDEX Upgrade have been analyzed with the TRACE algorithm developed to automatically detect and track dust particles. A first study, based on the statistical analysis of dust rates under various discharge conditions, underlined the predominant role of off-normal plasma phases on dust creation and remobilization, while dust rates were found to be significantly lower than in tokamaks with carbon PFCs [1].
In this contribution, we present complementary investigations detailing spatial distributions of dust events and main characteristics of intrinsic dust trajectories. The recalibration of cameras lines of sight in consecutive campaigns shows evolutions in dust spatial distributions, which can be related to modifications in PFCs geometry and composition. Vertical Displacement Events inject dust originating from the PFCs which are heated by contact with the plasma. It is found that during type-I elmy discharges, dust is predominantly produced in the upper part of the vessel, before falling down with large vertical velocity component. We also show that different patterns of dust events and trajectories are observed depending on the heating scenario. Finally, it is found that while dust particle move in the SOL plasma, they remain visible on an average distance of 0.5 m at an average velocity of 80 m/s.

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Stochastic clustering of material surface under high-heat plasma load in fusion devices

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Materials of various chemical composition and initial crystalline virgin structure (tungsten, carbon materials and stainless steel) have been studied after the irradiation by high heat plasma fluxes in nuclear fusion facilities [1]. High-temperature plasma load on the plasma facing material in fusion devices during transients (disruption, ELMs, VDE etc.) produces several multiscale effects including surface erosion, redeposition of eroded materials, melting and melt motion over the surface, inhomogeneous solidification leading to specific surface clustering conditions which are strictly different from any other conditions of solidification and clustering of materials previously analysed. This study has demonstrated evidences of inhomogeneous stochastic clustering of the surface with properties of the self-similarity of granularity from nano- to macroscale. In particular, the hierarchical granularity and self-similarity with cauliflower-like shape of tungsten surface have been revealed for the first time. The clustering of materials irradiated by high-temperature plasma qualitatively differs from the ordinary Brownian surface roughness and from clustering under other conditions. This difference is shown by comparing the results with those for the molybdenum sample after exposure in the magnetron plasma discharge and for the industrial steel casting with the ordinary roughness formed typically at solidification after melting. The specific property of material solidification and clustering under plasma influence in fusion devices is due to a material's (ions, clusters, melt on the surface etc.) motion under the influence of stochastic electromagnetic field formed by the near-wall turbulent plasma. This field ensures memory effects, long-term correlation and conditions for the growth of agglomerates with a self-similar structure [2, 3]. In addition to such a process, effects of irregular motion and relaxation of the material (melt) on the surface contribute to the process of clustering at the extreme heat load on the material surface. These multiple effects are responsible for the fractal growth mechanism at scales from several tens of nanometers to hundreds of microns [2, 3]. Collective (synergistic) effects, rather than the specific physical and chemical properties of the virgin materials, dominate in such stochastic clustering. The reported experimental results possibly indicate universal mechanisms of stochastic clustering of materials under the high-heat plasma load in a fusion device. The quantitative characteristics of the statistical inhomogeneity of such surface structure, in particular, the broadening of the multifractal spectrum by 0.5–1.2, are in the range observed for typical multifractal objects in nature.

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Erosion Characterization of Innovative Plasma-Facing Materials on DIII-D using Focused Ion Beam Micro-Trenches

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Silicon samples possessing micrometer-scale trenches were exposed on the DIII-D tokamak using the Divertor Material Evaluation System (DiMES). The micro-trenches were prepared using a focused ion beam (FIB) to carve desired geometries into polished sample surfaces for erosion loss measurements. The exposures were performed in the lower divertor of DIII-D in ELMing H-mode discharges with the samples located outboard of the outer strike point between Ψ_N=1.02 and 1.04, far from the separatrix. Each DiMES sample was exposed to 13 total discharges, with T_e and n_e at the target surface ranging from 1 – 60 eV and 0.1 – 2.4 $*10^{19}$ m^{-3}, respectively. This test provided a good proof-of-concept, with samples demonstrating shadowing of the trench floor and edges as predicted by simulations, along with clear preferential sputtering on trench surfaces with respect to the trench orientation relative to the magnetic field. Trench surface fiducial markings indicated erosion depths approaching 0.2 μm were achieved in some cases. Shadowed erosion patterns also gave preliminary results for determining the average ion impact angle, estimated to be 53° with respect to the magnetic field and -55° with respect to the surface normal.

Plasma-facing materials (PFMs) in future tokamak fusion reactors must be designed to exhaust thermal loads while adequately protecting in-vessel components. The present work focuses on the erosive behavior of advanced, alternative materials under relevant high heat flux conditions. For bulk samples of these materials, the use of FIB micro-trenches for post-experiment erosion measurements is valuable. The micro-trenches have a well characterized depth measured via scanning electron microscopy and atomic force microscopy, as well as fiducial depth markings etched into the trench walls that are shadowed from plasma impact. The integrated, post-exposure changes in trench geometry combined with in-situ diagnostics provide a measure of the erosion rate for each exposed material.

The test exposure of FIB micro-trenched silicon included samples containing 10 x 4 μm rectangular trenches 4 μm deep with different orientations, along with one 10 x 10 x 10 μm square trench. Clusters of 10 x 4 μm trenches were oriented at 0°, 45°, and 90° with respect to the magnetic field. Observed sputtering patterns matched the expectations from MATLAB simulations, and the larger square trench allowed for ion impact angle calculations. Additional tests to study erosion properties of other innovative PFMs are planned on DIII-D and will be presented. The materials include: polycrystalline 3C silicon carbide, MAX phase ceramics Ti_3SiC_2 and Ti_2AlC, ultra-fine grain (UFG) tungsten, and silicon carbide coated graphite foam.

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Prediction modelling of erosion and deposition characteristics on rough surfaces under ITER-relevant plasma conditions

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The impurity erosion-deposition distribution on rough divertor of fusion device is inhomogeneous due to surface roughness, with a high erosion or small deposition on protruding parts of rough surface, and a lower erosion or larger deposition at the far side of ridges and at the bottom of recessions. This phenomenon is observed in different tokamak facilities, which may finally lead to a smoothing of initially rough surface. This effect of long pulse and steady state operation is difficult to study in present tokamak facilities, but predictive modelling can be performed to investigate it.

The three-dimensional (3D) Monte-Carlo code SURO [1-3] has been developed to study the impurity erosion and deposition on rough surfaces under ITER-relevant plasma conditions. The properties of background plasma and impurity near the divertor target are studied by the 1D Particle-In-Cell Monte-Carlo collision (PIC-MCC) code SDPIC, which are used as the input data for SURO code. The SURO code uses the test particle approach to describe the bombardment of background plasma and the deposition of impurity particles on the 3D surface topography. The dynamic change of surface topography as well as surface concentrations of different species due to erosion and deposition are taken into account in SURO, which has a very good flexibility for treating the process of material mixing. In this study, the beryllium impurity erosion-deposition on the tungsten substrate of divertor target relevant to ITER has been studied by SDPIC/SURO modelling. The detailed analysis of the impact of the background beryllium flux on the rough surface evolution is conducted. The areal densities of the background beryllium deposition on rough tungsten substrate are calculated with different beryllium fluxes. The effects of the rough surface topography on the beryllium erosion and deposition are discussed.

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Model validation on DIII-D experiments towards understanding of high-Z material erosion and migration in a mixed materials environment

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Modeling of recent DIII-D experiments reveals the critical role of the sheath and background low-Z impurities in determining high-Z material erosion and migration in a mixed materials environment. This understanding leads to promising methods for erosion control, which is critical for material lifetime, plasma impurity content, and tritium retention in fusion reactors. Thin Molybdenum coated samples with external biasing were exposed under well-diagnosed divertor plasma conditions in DIII-D using the divertor materials evaluation system (DiMES), and strong sheath effects on erosion rates have been observed. The 2D Particle-In-Cell (PIC) code SPICE2 and the 3D Monte Carlo code ERO have been used to simulate the particle flux and material erosion as a function of biasing voltage. Both the PIC simulation and the D_α emission measured by a fast camera reveal that with increasing biasing voltage the ion flux decreases at the biased area while increases at the adjacent downstream tile, although the biased sample potential is far below the plasma potential. Detailed modeling shows that the ion flux variation at different area is due to the strong gradient of the electric field in the sheath, which results in different magnitude of the polarization drift above the biased and non-biased surface. The reduced ion flux and incident energy are responsible for more than an order of magnitude reduction of erosion with slight positive voltage biasing (~40 V) in the experiments.

In addition, we have investigated tungsten (W) sourcing during the dedicated DIII-D experimental campaign with toroidally continuous W rings embedded in the divertor target. W migration across the surface was measured with several changeable inserts in a specially designed DiMES collector probe, while tungsten erosion was characterized by WI spectroscopy. The distribution of background carbon impurities in different charge states has been self-consistently calculated, taking into account mixed materials effects. ERO modeling shows that that the transport of carbon impurities not only dominates the tungsten sputtering source but also determines the overall erosion and deposition balance in the mixed materials surface. The ExB drift and lower electron temperature (~10 eV) at the radial outboard side of the DiMES collector probe lead to a net deposition zone where tungsten and carbon are accumulated. In the net erosion zone closer to the outer strike point, the tungsten coverage on the inserts is very low and saturated independent of exposure time. Good agreement with the measurements has been obtained.

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Measuring changes in the thermal and elastic properties of polycrystalline tungsten exposed to helium plasma using transient grating spectroscopy

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Tungsten (W) surfaces undergo surface morphology changes during exposure to helium (He) plasma at elevated temperatures. The implanted He precipitates into nanometer size bubbles. Eventually, after a certain incubation fluence of He has impacted the surface, the surface starts to grow nano-tendrils as He bombardment continues. This forms a highly porous layer referred to as W fuzz. Evidence has shown that this surface transition can occur in a tokamak environment [1], and the fuzz is expected to have a deleterious effect on tokamak operation. This motivates the need to understand how W fuzz develops, and how to mitigate, eliminate, and prevent W fuzz in future fusion energy devices.

It will be necessary to remotely monitor W plasma-facing surfaces for fuzz growth using a non-contact, non-destructive *in situ* technique. Transient grating spectroscopy (TGS) is being proposed as a viable method for this purpose [2]. TGS will allow changes in the W surface thermal and elastic properties to be sensed remotely as they occur, and these changes can be correlated to the presence and growth of fuzz.

This paper reports the first time this technique has been used to study W fuzz formation *ex situ*. Samples exposed to various fluences up to the incubation fluence at a temperature of 1020 K under a flux density of $3 – 6\times10^{21}$ He/m^2/s at 65 eV are analyzed with TGS after exposure. Changes in the observed standing acoustic wave (SAW) speed and rate at which the SAWs decay correlate to plasma-induced changes to the elastic and thermal properties of the W sample surfaces. By comparing TGS results with traditional analysis techniques (e.g. scanning electron microscopy), these property changes can be mapped to fuzz growth, thus verifying the applicability of TGS for *in situ* detection of W fuzz growth.

Under the irradiation conditions used for this study, a minimum in the SAW speed is found around 2×10^{24} He/m^2. This value is very similar to what others have reported as the incubation fluence at which W fuzz begins to develop. This means that by monitoring the SAW speed *in situ*, an observed minimum in SAW speed can be used to indicate the onset of W fuzz growth. Further analysis of the SAW can provide insight into the thermal property changes that are taking place as the plasma facing W surface becomes saturated with ~1nm He bubbles.

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First Principles Molecular Dynamics Study of the Liquid LiSn Surface as a Plasma-Facing Component

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One key challenge for fusion reactors is the materials used in the divertor region, where high-power loads impinge on the first wall [1]. For years, solid tungsten was the main candidate as a plasma-facing component (PFC); however, studies found that cracks will appear when the temperature at the divertor plates increases over the ductile-brittle transition temperature of tungsten (around 673 K) [2]. These mechanical issues (along with technical issues) therefore render tungsten unsuitable as a PFC. Liquid metals, with inherent disorder, have long been suggested as a solution because of their imperviousness to mechanical damage, allowing for a self-healing and self-replenishing surface. Promising candidates for liquid PFCs are lithium (Li), tin (Sn), and gallium due to their low melting points and evaporation rates [3]. The potential use of these materials in fusion reactors also depends on their affinity to retain hydrogenic isotopes. Liquid Li properties have been studied extensively both experimentally and computationally, as well as its retention of deuterium at different temperatures and concentrations [4]. Encouragingly, the evaporation rate of LiSn alloy for concentrations of 20-30 at.% Li is at least three orders of magnitude lower than that of pure Li. In addition, recent experimental data on $Li_{30}Sn_{70}$ suggests very low deuterium retention [5]. These factors thus make the liquid LiSn alloy an ideal candidate for PFC, although further research into its atomic behavior is desirable.

We have performed a first-principles molecular dynamics study of the surface of liquid LiSn at two specific concentrations: $Li_{20}Sn_{80}$ and $Li_{30}Sn_{70}$. The study was performed at several temperatures ranging from their melting points up to 973 K. Static and dynamic properties such as pair distribution functions, diffusion coefficients, and viscosities are evaluated at different depths from the surface. We also evaluate the temperature dependence of Li segregation at the surface. Overall, we provide insight into the atomic behavior of $Li_{20}Sn_{80}$ and $Li_{30}Sn_{70}$ alloys, which will be useful for further research into PFC for fusion reactors.

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Topics: Material Erosion, Migration, Mixing, and Dust Formation

Synthesis, densification and mechanical properties of nanometric tungsten for fusion applications

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Recent works have shown that low grain sizes are favorable to improve ductility and machinability of tungsten, as well as the resistance to ablation and spallation, which are key properties for the use of this material in thermonuclear fusion environment [1]. However, current production routes are not suitable for the fabrication of bulk nanostructured tungsten samples. We propose here a new methodology based on powder metallurgy. Powder synthesis is performed using the Self-propagating High-temperature Synthesis (SHS) process. SHS is based upon the reduction of tungsten trioxide with magnesium in excess, and using sodium chloride as a reaction moderator. The resulting powders show platelet-like grains as the main feature, below 1µm in their largest dimension and around 60nm in thickness. Densification is then performed using Spark Plasma Sintering (SPS) at temperatures of up to 2000°C, even though the densification seems to be complete below 1600°C. Relative densities of 99.99% have been obtained. While Scanning Electron Microscopy (SEM) reveals an apparent grain size in the micrometer range, Electron Back Scattered Diffraction (EBSD), which is sensitive to the grain crystallographic orientation, clearly indicates that these micrometric grains are in fact nanostructured (see figure). This nanostructure is being inherited from the initial powders, thus potentially preserving the ablation and sputtering resistance of the material. A global preferential crystallographic orientation is observed, although the surface analyzed is only 60x60 µm^2. Simultaneously, this nanostructure induces an increase in hardness to a value of 428 HV, much higher than microcrystalline tungsten (~327 HV). These first results confirms that the process presented in this paper go in the right direction e.g. bulk nanostructured tungsten material. Mechanical tests (such as compression test) are currently ongoing with different test samples. Plasma exposure experiments are also undertaken in order to evaluate how the new nanometric W material sustains plasma erosion. All these results will be presented in this contribution.

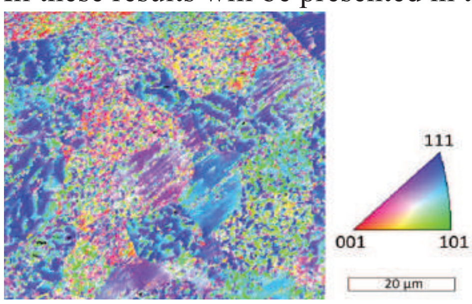

Figure: EBSD image of the surface perpendicular to the SPS compression axis, showing large grains (10 µm), with a finer, nanometric, substructure.

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ERO modeling of tungsten erosion and migration from a toroidally symmetric source in DIII-D divertor

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The W gross erosion profile from a toroidally symmetric W ring in the DIII-D lower divertor is reproduced with ERO simulations taking into account the effect of ExB drifts. ExB drifts are shown to have a significant impact on impurity migration in the divertor, i.e. by carrying a large amount of C impurities onto the W surface. The flux of C impurities eroding and redepositing on the W surface is shown to be larger than the flux of C impurities originating from nearby graphite tiles. In this C deposition regime dominated by erosion-redeposition, the flux of C impurities inducing W erosion strongly depends on the C implantation into and erosion from the W surface. Modeling of these processes with the analytic homogenous mixed material (HMM) model [1,2] is shown to be valid only for large electron plasma temperature at the divertor plate (Te>25eV).

Simulations were carried out for the dedicated L-mode discharges during the Metal Rings Campaign in DIII-D. Two toroidally symmetric W rings were inserted in the lower divertor of DIII-D and exposed to ~25 repeated attached L-mode shots in the reverse-Bt configuration with additional neutral beam heating (2.8 MW). The outer strike point was placed on the outboard W ring, where the radial profiles of the W gross erosion flux were measured spectroscopically. Modeling of those L-mode plasmas with OEDGE indicates that C impurities flowing onto the W ring are eroded and migrate within the outer divertor region only. Therefore, erosion, migration and redeposition of C and W particles are modeled within the divertor region exclusively using ERO with toroidal periodic boundary conditions. Plasma background information obtained with DIVIMP suggests that the horizontal projection of the total ExB drift points inward in the common flux region, as previously observed in the UEDGE simulations of H-mode plasmas with similar input power [3]. ERO modeling predicts that such inward drifts carry large amount of C impurities onto the W surface, which are necessary to quantitatively reproduce the measured W gross erosion flux. It is also shown that ExB drifts in the far SOL may explain the large W deposition pattern observed on graphite 5 cm radially outboard from the W ring.

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abstract number 164

Abstract Withdrawn

Thermal Stability of Tungsten Nanotendril Fuzz Grown Under Divertor-like Conditions

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Tungsten, the leading plasma facing materials of fusion reactors, is exposed to an extremely hostile service environment, characterized by high temperature, high heat flux, and intensive particle flux (e.g., deuterium, tritium, helium, and neutrons). The resultant thermo-mechanical degradation of tungsten will impose inevitable impact on plasma-materials interactions. A particular phenomenon for tungsten subjected to high fluence of helium plasma is the development of nanotendril fuzz on the surface. This might be one of the factors limiting the applicability of tungsten as the plasma facing materials. Significant efforts have been committed to understanding the fuzz formation through molecular dynamics simulation or other theory approaches, while limited experimental investigations have been conducted to characterize the microstructure of fuzz at various formation stages by employing advanced electron microscopy. Up to date, the underlying mechanisms controlling the development of nanotendril fuzz is not well understood and some critical experimental data of nanotendril fuzz are still missing.

In this presentation, we will discuss the thermal stability of nanotendril fuzz grown under divertor-like conditions through the employment of thermal desorption spectrometry (TDS) and transmission electron microscopy (TEM). Tungsten was exposed to 50 eV helium ions to a fluence of $4\times10^{26}/m^2$ at 900°C in PISCES-A device at UCSD, resulting in a thick mat of well-developed nanotendril fuzz on the surface. TDS measurements enabled the measurement of desorbed helium amount during the thermal annealing processes (0.5°C/s ramping rate to desired temperatures) and the identification of the dissociation energies of helium from trapping sites for each desorption peaks. A newly developed technique, Laser Ablation Mass Spectroscopy, was used to quantify the total amount of helium contained in the sample. The subsequent TEM observations revealed the microstructural evolution of the nanotendrils on the tungsten surface and the helium bubble evolution within the substrate tungsten (<500 nm) following TDS measurements at 900°C, 1400°C and 1600°C. The results indicated that nanotendril fuzz was reforming into the surface and no helium bubbles were observed within the substrate tungsten after 1400°C and 1600°C annealing except some remaining large open pores. For the 900°C annealing, the nanotendrils survived but shrunk, while a large number density of open pores were observed in the substrate tungsten. The dependence of the thermal stability of fuzz on temperature and helium pressure within bubbles will be discussed based on these experimental results.

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Growth process of nano-tendril bundles with sputtered tungsten

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Morphology changes on tungsten (W) surface during interactions between W and helium (He) plasma have attracted a lot of attention because those could lead to significant changes in thermal and mechanical properties for the divertor blocks in fusion devices. Therefore, an extensive investigation on possible morphology changes during the W-He interaction is of importance to minimize deleterious effects caused by unexpected morphology changes. Recently, a new type of morphology change was discovered by *Woller et al*, namely nano-tendril bundles (NTBs). Differently from typical nanostructure (W-fuzz), which is uniformly generated on all the surface, NTBs are isolated 'islands' of bundles of nanostructures (W-fuzz) and several tens of times taller than typical W-fuzz [1]. The W surface was not fully covered with fuzz, but NTBs formed in isolation. *Woller et al* used 13.56 MHz radio frequency modulated He plasma to fabricate NTBs.

More recently, it was revealed that NTBs formed under exposure to the He plasma produced by direct current (DC) discharge under sputtering regime with the addition of impurity gases such as argon (Ar), neon (Ne) and nitrogen (N_2) [2]. Even without the addition of the impurity gases, NTBs were also found by increasing the vacuum background pressure from ~10^{-5} to ~10^{-4} Pa. Most of NTBs were generated where net erosion yield for W sample against a He ion was in the range of 10^{-2} – 10^{-3}. NTBs formed upon fuzz-like morphology which was modified by He ion bombardment. Thus, it was indicated that erosion-deposition of W particles would be a key process for NTB growth, though the growth mechanism is still unclear. On the other hand, it has been known that the morphology of fuzz depends on surface temperature of W specimen [3]. This temperature dependence of fuzz formation can affect the erosion and deposition of the surface and the growth process of NTBs eventually.

To explore the growth process for NTBs, we investigated the morphology change of W with different plasma fluences. Two W samples were exposed to the DC He plasma with the addition of 2.4% of Ar gas. The sample was biased to −150 V and the surface temperature was 1480 K. Surface observations revealed that the formation of NTBs depended on the fluence. When the fluence was 2.0×10^{25} m^{-2}, a great number of NTBs with the height of ~20 μm was formed and the surrounding W surface was almost covered with fuzz, whereas NTBs did not appear but fuzz formed sparsely on the surface when the fluence was 1.5×10^{25} m^{-2} or lower. To reveal the influence of surface temperature on the growth of NTBs, two samples were exposed to He-Ne mixed plasma, with 4.4% of Ne gas ratio and −250 V of bias potential. The fluence was 2×10^{25} m^{-2} and the surface temperatures were 1500 and 1680 K. From the surface observations, it was revealed that the height and the number of NTBs decreased as the surface temperature increased. It was likely that the growth of NTBs slowed due to rapid surface diffusion by recrystallization and annihilation when the surface temperature exceeded the recrystallization temperature.

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Suppression of wall erosion by vapor shielding at low-Z and high-Z walls

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Wall erosion during transient heat loads is a large concern for ITER and future fusion devices [1] while vapor shielding is known as an inherent relief for the wall erosion during the high heat loads. It was pointed out that low-Z material such as Be easily causes the heat flux dissipation by the vapor shielding but high-Z material such as W hardly causes the shielding [2]. Meanwhile, low-Z materials are easily eroded. Here, a question is, how much erosion will happen at transient high heat load for low-Z and high-Z materials. In the present paper, we study wall erosion amounts and its suppression by the vapor shielding in a reactor condition.

In order to understand the difference between materials for vapor shielding, kinetic behavior of particles is a key factor. Thus, the authors are developing a particle-in-cell (PIC) simulation code, called PIXY, and have applied it to the vapor shielding simulation. Treating 1d3v movements of ion, electron, and neutral particles, PIXY simulates plasma behaviors in sheath and magnetic pre-sheath near a target plate. The plate temperature is simultaneously calculated to determine the vapor emission rate. Thus, the PIXY code treats evaporation, sputtering, self-sputtering, and prompt re-deposition self-consistently. A validation of the code has been taken with the experimental results of a plasma-gun injected to an Al coated W specimen [3]. The surface back temperature after the gun pulse was well reproduced by PIXY, which was measured as 2250 K for a W specimen and 1600 K for an Al/W specimen. The incoming heat flux to the plate of 2.6 GW/m^2 was dissipated to 0.5 GW/m^2 at 0.15 ms after the irradiation to the Al/W specimen started.

In the present paper using PIXY code, we study wall erosion during transient loads to Be and W walls at a reactor condition. In a comparison between low-Z and high-Z walls, a high-Z vapor particle tends to travel shorter than a low-Z particle. The larger radiation volume of low-Z vapor effectively dissipates the incoming electron energy. Plus, the low-Z vapor particles effectively dissipate the incoming hydrogen ion energy via the ion-neutral collision. Simulating this shielding effect, erosions via sputtering, melting, evaporation and ablation during the plasma load are estimated for different materials. Then, suppression of wall erosion by the vapor shielding is examined as a function of maximum heat flux, heat flux pulse duration, and remained first-wall armor thickness. Influences of ablation models, the limited dimensionality, and other model assumptions on the wall erosion estimation are also discussed. Findings from this parametric survey are summarized as reference data for the future device design and operation.

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Beryllium melting and erosion on the upper dump plates in JET facility during three ILW campaigns

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In 2016, the Joint European Torus (JET) completed a deuterium plasma campaign which ended the third period of operation between shutdowns for the ITER-Like-Wall (ILW). JET was upgraded to the ILW material configuration in 2011, the main elements are bulk beryllium (Be) tiles in the main chamber (inner and outer limiters and upper dump plate ribs), Be coated Inconel tiles on recessed areas of the inner wall and a mix of bulk W and W coated CFC tiles in the divertor. The first ILW campaign (2011-2012) focused on the impact of the new wall on plasma operation, material migration and fuel retention [1]; the second ILW (2013-2014) was expanded the operational space towards high power scenarios while optimising energy confinement and studied ITER relevant issues such as disruptions mitigation by massive gas injection. The third ILW (2015-2016) and most recent set of campaigns was characterised by a first push towards increased fusion performance along with integration of all necessary tools such as impurity control and disruption mitigation. In this work, data related to Be melting and erosion from the upper dump plate tiles during all three ILW campaigns is presented. High-resolution images of the upper wall of JET, taken after each plasma campaign operation shows clear signs of flash melting on the ridge of the Be dump plate tiles that form a set of 64 poloidal ribs or protection limiters. The melt layers move in the poloidal direction from the inboard to outboard tile ending on the last tile with an upward going waterfall-like melt structure [2]. The main reason for this melting is unmitigated plasma disruptions and it is the halo current which is believed to provide the J×B force that drives the melt layer along the surface. During all three ILW campaigns, 12618 plasma pulses were performed, with an average of ~4200 experimental pulses per campaign. Out of all these, around 15% of them were identified and catalogued as disruptions with most of these moving upwards and inwards. Using thermocouple data from the upper dump plates tiles we can see a reduction in energy delivered by disruptions once massive gas injection is used routinely with noticeably fewer extreme events in ILW3 compared to ILW1 and ILW2. This trend is consistent with trends seen in the high resolution in-vessel images. Each ILW campaign was followed by long-term sample removal from JET for *post-mortem* analysis which allows comparison of dump plate tile damage for each period. Be erosion was evaluated via tile surface profiling and precision weighing and initial results show increasing mass loss towards the outermost tiles with up to 0.5g mass loss from a single tile (one of 64 toroidally equivalent tiles) due to melting rather than plasma erosion. Although the poloidal flow of Be makes it hard to determine the net loss, the amount of Be mobilized is much greater than the quantity of Be dust as molten droplets estimated at ~0.5g following ILW2 [3]. Studies of the Be melt layers using microscopy are also planned with a view to determining the melt layer depth and impact on material structure and these results will also be reported.

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Measurements of Tungsten Erosion Using UV Emission from DIII-D and CTH Experiments

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A total of 27 W I emission lines in the UV region are identified, 19 of which have not previously been reported in fusion relevant plasmas. UV survey spectrometers (200-400 nm) are used to diagnose W erosion in the DIII-D divertor and from a W-tipped probe in the Compact Toroidal Hybrid (CTH) experiment. Initial W erosion measurements using the W I 361.8 nm line are compared to erosion source measurements using the standard 400.9 nm W I line and are found to be in good agreement. The relative intensities of W I lines throughout the UV spectrum in both DIII-D and CTH are consistent with atomic physics calculations. Material erosion can be diagnosed from spectral line intensities together with atomic coefficients representing the ionizations per photon (S/XB). Atomic physics calculations also reveal that the neutral W metastable level populations can significantly impact the S/XB ratio. The high density of W I emission in the UV region allows for simultaneous measurements of multiple W line intensities, which are needed to determine the effect of metastable level populations on the W I spectrum. An electron temperature diagnostic utilizing the W I 265.65 to 255.13 nm line ratio will also be presented. The broad neutral tungsten UV spectrum from 200-400 nm is also in agreement with W I theoretical calculations.

A new Collisional Radiative (CR) code, ColRadPy written purely in Python, has been developed and compared to the widely used collisional radiative code ADAS. ColRadPy solves the collisional radiative set of equations as well as the time-independent and time-dependent ionization balances. ColRadPy provides easy access to all parts of the collisional radiative matrix for extended post-processing analysis. ColRadPy allows the CR equation to be solved for an arbitrary number of metastable states. The new functionality has been used to model the six lowest metastable levels of W I and their impact on the photon emissivity coefficient (PEC), effective ionization rate (SCD) and S/XB coefficients. The capability to model more than three metastable states simultaneously has not been available previously. W I PECs are found to depend strongly on metastable level fractions while the SCD does not depend on the metastable fraction. The ColRadPy code will eventually be released as open source software.

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Angular dependence of Fe sputtering by Ar ions at polished and rough surfaces

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Erosion of the inner walls is one of the issues of a thermonuclear reactor. Material used for the inner walls will be irradiated by fast ions which will erode the material which will possibly contaminate the core plasma. Moreover re-deposition of eroded material contributes to additional loss of fusion fuel and retention and consequently reduce the operational life time of the inner walls. For perfectly smooth surfaces a distinct angular dependence of the sputtering yield is expected [1]. For rough surfaces this dependence is smeared out. Within the Eurofusion work package PFC a dedicated task was launched to quantitatively determine the influence of roughness on the sputtering yield.

To address this issue the erosion of magnetron sputtered iron (Fe) thin films on silicon by argon (Ar) ions was performed. To this end we constructed an experimental set up inside of an vacuum experimental chamber. Samples are exposed to Ar ions generated with Electron Cyclotron Resonance (ECR) ion gun. Ions are extracted with 1 keV, focused with an einzel lens and decelerated in front of the sample to 900 eV.

The sample with 260 nm thick Fe film deposited on silicon (Si) was mounted on a rotating table. Samples were exposed to fluence of $5x10^{20}$ ions/m^2 Ar ions at different impact angles to the surface. In the first experiment we exposed samples with mirror like polished surface and in the second one we exposed samples with high surface roughness. The thickness of the eroded layer was measured by Rutherford back scattering spectroscopy (RBS). Due to the small size of the erosion crater we perform the RBS measurements in our microbeam chamber with 1.5 MeV protons focused down to 3 μm. In the presented work results from both experiments will be discussed.

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WallDYN simulations of material migration and fuel retention in ITER neon-seeded DT plasmas

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ITER is a nuclear licensed facility and must therefore comply with tritium (T) and dust inventory limits. Tritium retention in ITER will mainly be driven by co-deposition with beryllium (Be). Predictions of retention rates during nuclear operations have been performed in recent years using the WallDYN code [1,2].

In ITER, neon (Ne) is a candidate seeding impurity for divertor power flux control [3]. To quantify the impact of Ne on Be deposition and T retention, a series of plasma backgrounds generated by the SOLPS-4.3 code are used as the basis for WallDYN simulations. The SOLPS-4.3 solutions (which do not include classical drifts) correspond to a fueling scan for high power $Q_{DT} = 10$ operation with $P_{SOL} = 100$ MW and with varying strengths of Ne gas injection, providing concentrations in the range 0.4-1.3% (defined with respect to the outer midplane separatrix electron density). The fuel throughput scan yields divertor states ranging from high recycling through to partial detachment, corresponding to a range of sub-divertor neutral pressures, p_n. OSM is used to extend the SOLPS solutions to the wall. To account for uncertainty in the nature of the ITER far-SOL, either an exponential extrapolation or increased radial velocities are assumed to account for the existence of a high density shoulder. The velocity of parallel plasma flow, $v_{\|}$ in the main SOL, is also varied, with $M = v_{\|}/c_s = 0.0$ and 0.5 towards the inner divertor as the limiting cases. The RACLETTE code is then deployed to estimate the surface temperature profiles at the inner and outer divertor targets. Finally, DIVIMP is used to calculate the redistribution matrix for input to WallDYN.

The Be erosion and re-distribution is strongly affected by the assumptions on the upstream far-SOL density. In case of an exponential profile, the Be deposition occurs mainly in the divertor, while assuming a flat density profile concentrates the co-deposition on the first wall itself. Because the wall is at a lower temperature than the divertor, this leads to an increase in T retention. Comparing a high recycling case ($p_n = 1.8$ Pa) to a partially detached case ($p_n = 9$ Pa), the retention is higher (by 50%) in the former because of the higher upstream electron and ion temperatures. Neon is found to play a marginal role on Be co-deposition and T retention, as expected because of the low Ne fraction and lower sputtering yield of Ne on Be compared to D on Be. Currently the effects of flow have been prescribed in OSM. SOLPS-ITER runs including classical drifts are in preparation and will be used to investigate the effect of more self-consistent SOL flows on the migration and retention.

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Modelling of tungsten erosion and deposition in the divertor of JET-ILW in comparison to experimental findings

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Tungsten (W) will be used as plasma-facing material in ITER and is under consideration for future devices like DEMO. However, the W concentration in the plasma should be minimised to limit plasma dilution and cooling. Thus, the prediction of the net W erosion is important and for this, experiments at existing fusion devices in combination with modelling are indispensable.

The present contribution focuses on the modelling of tungsten erosion and deposition in the divertor of JET. The 3D Monte-Carlo code ERO is used for this modelling and the results are compared with observations. Experimental studies at JET, mainly based on spectroscopy, can be found e.g. in [1, 2]. The tungsten gross erosion along outer divertor tiles has been measured by WI spectroscopy for intra- and inter-ELM conditions [1]. As in the experiment, the modelling shows that normally the gross erosion is dominated by the ELMs. For instance, in the example of [1], the intra-ELM sputtering is dominated by a factor of about 9 over inter-ELM sputtering. However, the modelling indicates that the inter-ELM phases can significantly contribute to the resulting net erosion as the re-deposition fraction during ELMs is larger than in-between ELMs. For the referred example, the modelled re-deposition fractions are about 84% for the inter- and 98% for the ELM-phase. This reduces the gross erosion by factors of about 6 for inter-ELM and about 50 for intra-ELM conditions leading to comparable net erosion due to inter- and intra-ELM phases. In general, the resulting contributions to the net erosion strongly depend on the inter-ELM electron temperatures – a parameter study will be presented. The modelling also shows that the W sputtering in-between ELMs is usually due to impurities (beryllium) unless the electron temperature is large enough such that sputtering due to deuterium also can contribute considerably. On the other side, the ion impact energy during ELMs is large enough (according to the free-streaming model) wherefore sputtering is dominated by deuterium ions. Finally, the effect of self-sputtering of returning W ions will be included in the modelling to study its influence on the resulting gross and net erosion.

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In-situ observation of reduced sputtering of nano-structured surfaces

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Sputtering of plasma-facing materials is a critical issue for the lifetime of the fusion reactor wall. Under the plasma bombardment, many materials form fine surface structures, such as nano-size fuzz, grass-like needles and cones. The surface roughness induced by these structures is able to significantly influence the effective sputtering rates of materials. Depending on the relation between the size of the structures and the distance between them, a considerable fraction of particles sputtered from the surface can be trapped on the protruding structures and, therefore, reduce the net sputtering yield.

The linear plasma device PSI-2 is equipped with imaging optical emission spectroscopy (OES) and quartz microbalance (QMB) systems for in-situ measurements of sputtering. The mass loss technique measures the amount of eroded material by the comparison of the sample mass before and after the exposure. The method, however, lacks time resolution and provides an integral value of sputtering. Both QMB and OES rely on a model for the interpretation of the measurements, i.e. for the calculation of the loss fraction of eroded particles on the way to QMB or calculations converting the number of photons measured by OES to the number of eroded particles. The ERO code used for this purpose allows us to numerically simulate the experimental conditions and directly compare data from various diagnostics with the model. An additional module is being implemented in ERO for simulating rough surfaces [1].

In the set of experiments described here, QMB, OES and the mass loss technique were simultaneously used to measure sputtering of tungsten and molybdenum by helium while fuzz structures were formed at elevated target temperatures, as well as sputtering of aluminium by deuterium while needle-like structures developed on the surface. The bulk target covered the entire diameter of the plasma column of 6 cm. A set of ten 5×5 mm^2 samples was incorporated in the target providing radially resolved mass loss data. In all cases, the incident ion energy of sputtering ions was selected to be well above the sputtering threshold. The surface morphology after the experiment was observed by scanning electron microscopy (SEM).

At an ion flux of $\sim1\times10^{22}$ m^{-2}s^{-1}, it typically took 30-60 min for the development of surface morphology resulting in a considerable (>25%) reduction in measured sputtering yield. While the analysis for W is ongoing, SEM confirmed the formation of fuzz for Mo and needle-like structures for Al. For Mo, both QMB and SEM observed a continuous reduction of sputtering up to a factor of two after a fluence of He of 4.3×10^{25} m^{-2}. For Al, sputtering reduced by 25% after a D fluence of 1.5×10^{25} m^{-2}, but did not change significantly thereafter, indicating a saturation of the effect. By a target tilt and, therefore, a change in the incident ion angle, it was possible to efficiently remove the needle-like structure. The Al sputtering yield recovered to a value close to the initial. Time resolved sputtering for the tested materials will be discussed in the contribution.

[1] A.A. Eksaeva, this conference

Carbon impurity transport study in an ECR Chamber by utilizing erosion and deposition of a-C:H thin layers

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Magnetic fields in a vacuum chamber which contains plasma affect the motion of ions, resulting in a different deposition pattern from that without magnetic field. The effect of magnetic field has to be carefully investigated to understand the characteristics of redeposited layers in a fusion device under strong magnetic field. In order to clarify the deposition and erosion characteristics as well as the properties of deposited layers of plasma facing components, we have exposed a graphite target to deuterium ion beam produced in an electron cyclotron resonance (ECR) chamber to simulate the effect in a simple magnetic field configuration.[1] We used a cavity structure with a standard a-C:H film on a Si substrate inside the coupon. Several coupons were installed vertically in the ECR chamber and the slit direction was parallel or perpendicular to B field. The optical constants were characterized by ellipsometric angles, Ψ and Δ, and chemical bonding structures were identified by Raman spectroscopy. Si substrates were positioned at the bottom and at the top of the coupons. According to ellipsometry, the standard a-C:H thin films at the bottom showed a significant erosion at the center due to the bombardment by neutral deuterium particles. In the middle of chamber, thickness variation showed equilibrium between deposition and erosion. The chamber heights where equilibrium between erosion and deposition occurred were different between slits parallel and perpendicular to B field. This phenomenon is attributed to non-negligible portion of ionic D^+ particles. The standard a-C:H thin films at the top showed a deposition of new a-C:H films at the center. A very thin polymer-like carbon layer has been deposited on the top coupons. The influence of the neutral particles causes the profile of eroded samples to have y-axis symmetry. Erosion depth profile was fitted using Gaussian lineshapes. Similarity of peak positions and lineshapes whether slit directions are parallel or perpendicular to B field indicates that neutral particles are dominant at bottom of chamber. Raman analysis showed that the intensity ratio, i.e. I(D)/I(G), was enhanced at the center, indicating that sp^2 contents have been increased. Estimating the erosion from the standard a-C:H films, we determined the erosion (sputtering) yield of incident D^0 particles on the standard a-C:H films as 7×10^{-3}. This value is consistent with literature values of 10^{-2} at 300K [2].

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[2] M. Schlüter *et al.*, J. Nucl. Mater. 33-37 (2008) 376.

Surface diffusion and growth of W self-interstitials during low-energy and large-flux H/He ion irradiations of polycrystalline W

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In this study, polycrystalline tungsten (W) has been irradiated with low-energy (20-200 eV) and high-flux (>10^{22} ions/$m^2 \cdot s$) H/He ions at the irradiation temperature of 130-1500 K. After H/He ion irradiation at a fluence of 10^{25}-10^{26}/m^2, their surface damages have been observed by scanning electron microscopy (SEM), SEM-electron backscatter diffraction, and transmission electron microscopy (TEM). The effect of O_2 addition into the irradiation system on the surface microstructures of irradiated W has been analysed, and tungsten oxide nano-rads have been observed by SEM and TEM. Our finding shows that surface diffusion and coalescence of W self-interstitials are formed during H/He ion irradiation at an elevated temperature, leading to the growth of W crystal and W oxide at the surface. During low-energy and large-flux H/He ion irradiations, W self-interstitials diffuse out of W surface layer in the [111] direction. The growth of W nano-fuzzes is well related to the surface diffusion and growth of W atoms at an elevated temperature. Our density functional theory calculation confirms the thermal instability of W self-interstitials, and the surface diffusion of W atoms due to H/He irradiation. Surface energy anisotropy provides selectivity in the driving force for the growth of W crystal, W nano-fuzzes, and W oxide.

Influence of surface tension on macroscopic erosion of castellated tungsten surfaces during repetitive transient plasma loads

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Understanding of plasma-surface interaction (PSI) effects during the transient events in future fusion reactors requires dedicated R&D activity in plasma simulators used in close connection with material characterization facilities as well as with numerical modeling. This report is focused on the analysis of surface tension contribution to the erosion features of tungsten resolidified surfaces and resulting material response to large number of repetitive plasma impacts. Experimental investigations of erosion processes on castellated tungsten surfaces in conditions relevant to ITER ELMs have been performed within powerful quasi-stationary plasma accelerator QSPA Kh-50. The surface energy load measured with a calorimeter was varied between the melting (0.6 MJ/m^2) and evaporation (1.1 MJ/m^2) thresholds, the plasma pulse duration was 0.25 ms. Observations of plasma interactions with exposed W surfaces, analysis of dust particle dynamics and the droplets monitoring have been performed with a high-speed digital camera. Development of surface pattern and material modification in results of plasma exposures have been studied with optical and SEM microscopy, profilometry and XRD.

Repetitive plasma loads above the melting threshold led to formation of melted and resolidified surface layers. Networks both macro and intergranular cracks appeared on exposed surfaces. Cracks propagate to the bulk mainly transversely to the irradiated surface. The splashing of dust/liquid particles has been analyzed in the course of repetitive plasma pulses. It was revealed that mountains of displaced material at the edges of castellated units are primary source of the splashed droplets. The solid dust ejection dominates by cracking processes after the end of pulse and surface resolidification.

Due to the continuously growing crack width (from fraction till tens μm) with increasing number of pulses the initially uniform melt pool on the castellated units became disintegrated into a set of melt structures separated by cracks. As result, a number of ejected particles essentially decreased after first hundred plasma pulses. Further increase of repetitive plasma impacts (above 200) led to considerable qualitative changes of surface morphology. Each cell of the crack network is strongly subjected to the surface tension that minimizes melt pool area. After large number of exposures the progressive corrugation of the surface occurred due to the capillary effects on exposed W surfaces.

Results of simulation experiments for castellated targets and developed surface structures are compared with repetitive plasma exposures of flat tungsten surfaces. Important contribution of surface tension to the erosion processes under the ELM relevant repetitive loads and it influence on suppression of droplets splashing is discussed.

Analysis of retained deuterium on Be-O films: ion implantation vs. in-situ loading

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ITER-relevant coatings containing beryllium (Be) are being developed in the scope of the EUROfusion Work Programme to supply standard samples for fuel retention investigations. Presently, deuterium (^{2}H) or helium (He) is being incorporated in relevant mixed samples to mimic the trapping of nuclear fuel in PFC re/co-deposits by two loading methods: in-situ during the deposition of the coatings by using the high-power impulse magnetron sputtering (HiPIMS) method or by ion implantation of ^{2}H or He ions on equivalent coatings without gas inclusions.

The document reports some of the preparative work performed with 400 nm Be-O-^{2}H films deposited on both Si and W plates. The efficiency of the two methods to incorporate low contents of ^{2}H in the films is compared. Also, the role of the interlayers between the films and substrates in the retained contents was evaluated.

Implanted films were prepared at room temperature by using 15 keV $^{2}H^{+}$ ion beams, being the fluence limited to 2 x 10^{17} ion/cm^{2} in order to avoid major morphological changes that may enhance gas release.

As-deposited and as-implanted coatings were analysed with ion beam techniques by elastic backscattering spectroscopy (EBS), Rutherford backscattering spectroscopy (RBS) and nuclear reaction analysis (NRA) using, respectively, 1600 keV $^{1}H^{+}$, 2000 keV $^{4}He^{+}$ and 1000 keV $^{3}He^{+}$ incident beams. Some months after implantation, aged coatings were analysed by time-of-flight elastic recoil detection (TOF-ERDA) involving 23 MeV $^{127}I^{6+}$ beams to evaluate ^{2}H release.

The present experiment pointed ion implantation under low fluence regimes as alternative to in-situ loading for ^{2}H incorporation in the range from 0.6 to about 1.2 x 10^{17} atoms/cm^{2}. As predicted and due to a lower diffusion of hydrogen on Si, Si substrates lead to a decrease in the retained contents down to 50 at.%. The release of ^{2}H in aged samples agrees with previous data. Also the Be, O and ^{2}H depth profiles evaluated by EBS/RBS/NRA or by TOF-ERDA are compatible.

Numerical studies from quantum to macroscopic scales of dusts in hydrogen plasma

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Dusts are frequently produced and dispersed during plasma operations. They are often generated by growth from molecular precursor and this growth leads to the formation of larger aggregates which induce a nucleation of solid germs. Particles formed are described by an aerosol dynamic including coagulation processes and accretion. Kinetic of these processes depends on the elementary charge processes and the charge fluctuation of the particle in plasma. So there is a strong coupling between particle characteristics and plasma discharge equilibrium.
This study is focused on the develop of a multiscale physic and numeric model of hydrogen plasmas and carbon particles around three essential coupled axes to describe the various physical phenomena: (i) macro/mesoscopic fluid modeling describing in an auto-coherent way, characteristics of the plasma, the population of formed molecular aggregates and properties of cloud of particles [1]; (ii) the classic molecular dynamics offering a description to the scale molecular of the chains of chemical reactions and the phenomena of aggregation; (iii) the chemistry quantum to establish the barriers of activation of the processes of distribution, reaction or of aggregation which drive to the formation of dusts [2].
An attention is focused on model carbon dusts formation in an argon DC discharge at 60 Pa where particle formation is initiated by graphite cathode sputtering. From chemistry quantum scale modelling, a systematic and accurate theoretical methodology is proposed to describe the elementary physical processes occurring relating to the growth of medium size clusters (C_n n<10). Based on Density Functional Theory (DFT) results, it is shown that the thermodynamic stability of clusters is independent of the exchange-correlation correction. For neutral carbon cluster size n <10, the most stable structure of the cluster is linear when the value of n is odd and cyclic if it is even. For anions, the most stable geometry remains linear irrespective of the size of the cluster. In both cases, the free energy per carbon atom increases in function of the cluster size. From classic molecular dynamics calculations, the thermodynamic stability of clusters until n = 60 is presented and sticking coefficient of little clusters on particles are calculated. At fluid scale, interplay between the different phenomena involved in the aerosol dynamics in dusty plasma is affected by the sputtering yield is detailed. The depletion of medium size clusters for the benefit of larger ones is thus numerically predicted and confronted to experimental data.

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Abstract Withdrawn

Global modeling of wall material migration following boronization in NSTX-U

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The National Spherical Torus Experiment Upgrade (NSTX-U) operated in 2016 with graphite plasma facing components, periodically conditioned with boron to improve plasma performance. Following each boronization, spectroscopic diagnostics generally observed a decrease in oxygen influx from the walls, and an in-vacuo material probe (MAPP) observed a corresponding decrease in surface oxygen concentration at the lower divertor. However, oxygen levels tended to return to a pre-boronization state following repeated plasma exposure. This deconditioning occurred on a faster time scale when conditioning with less boron. This behavior is interpretively modeled using the WallDYN mixed-material migration code, which couples local erosion and deposition processes with plasma impurity transport in a non-iterative, self-consistent manner that maintains overall material balance and has recently been applied to ITER [1]. A new model for spatially inhomogenous mixed material films has been developed for WallDYN, which allows for the differentiation between conditioning films of different thicknesses. This new capability improves model agreement with observed NSTX-U spectroscopic data. Plasma backgrounds representative of NSTX-U conditions are reconstructed from a combination of NSTX-U and NSTX datasets. Likely mechanisms driving the observed evolution of surface oxygen are examined, as well as remaining discrepancies between model and experiment and potential improvements to the model.

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Influence of heavier impurity deposition on Cr sputtering under He plasma exposure in multiple linear plasma devices

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As reviewed in Ref. [1], measured sputtering yields of various metals under high-flux light ion (D, He) plasma exposure are consistently much lower than theoretical values [2], while sputtering yields by heavy ion (Ne, Ar, Xe) plasmas are more or less consistent with theoretical values. To explore the cause(s) of the reduced sputtering yield, continuing experiments have been conducted with Cr targets exposed to He plasmas at incident ion energy, E_i, of ~80 eV in three linear plasma devices (PISCES-A, PSI-2, and NAGDIS-II). Spectroscopic measurements showed a significant drop of Cr I line intensities observed in front of the target during plasma exposures in PISCES-A and NAGDIS-II. Correspondingly, the sputtering yield, determined from mass loss, was found to be much lower than the theoretical value [2]. Surface observations revealed the development of cone structures. On the other hand, the sputtering yield in PSI-2 was nearly consistent with Ref. [2], and cone structures were formed only in tiny isolated spots.

This discrepancy between the devices may be explained by deposition of heavier impurities on the target in PISCES-A and NAGDIS-II. In these devices, the Cr target was clamped with a Ta cap and a Mo cover, respectively, which were also exposed to plasma, while the target holder was also made of Cr in PSI-2. While Ta is not sputtered by He at E_i ~ 80 eV, a trace amount of intrinsic impurity ions (e.g. C, O) as well as ionized Cr can sputter Ta. Sputtered Ta or Mo atoms can deposit and then agglomerate on the Cr target, leading to the formation of cone structures, since the sputtering yield of Ta and Mo is lower than that of Cr. It should be noted that Ta was actually detected with EDX (energy-dispersive X-ray spectroscopy) analysis on the Cr target surface after the plasma exposure in PISCES-A: ~98 at% of Cr and ~2 at% of Ta. The cone structures can reduce the sputtering yield due to direct line-of-sight deposition of sputtered Cr atoms onto neighboring cones, which is similar to observations on W fuzzy surfaces [3].

To further prove the hypothesis mentioned above, a stainless steel (SS) cap was used instead of Ta in PISCES-A. The mass of the main elements (Fe, Cr, Ni) in SS is similar to or the same as Cr. It was found that the similar-mass impurities did not lead to the formation of cone structures, and hence did not reduce the sputtering yield.

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Abstract Withdrawn

JET Be limiter tiles chemical bonding characterization by means of Raman microscopy and comparison with laboratory Be based samples: Be-O and BeO_xD_y identification

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Due to high flux plasma surface interactions, many processes, such as component melting, surface erosion, contamination (O, N, C…), dust production, element migration, hydrogen isotope retention or mixed material formation [1], can severely modify the tokamak plasma facing component (PFCs) properties.

Many studies using ion beam analysis techniques are devoted to determining the amount of given elements trapped, eroded or deposited in dedicated zones of existing tokamaks. Studies giving information on the material structural and chemical changes at the atomic scale are sparse although they can give clues to better understanding mechanisms at larger scales.

Raman microscopy is sensitive to chemical bonding and can give information on the way hydrogen isotopes are bonded to the other elements as for example in Be, BeD_x, Be_2C, BeO, BeO_xD_y [2-5]. The penetration depth of the probing electromagnetic field is a few nanometers which is well suited to investigate the supersaturated layer evidenced in many studies more precisely. Recently, micrometric BeO crystallites and BeO_xD_y species were identified on a tile extracted from the upper limiter of JET (2010-2012 campaign), with a combined approach using, confocal, and Raman and electron microscopies for experiment and Density Functional Theory calculations for modelling [2].

In this study, we report on a joint study of tokamak and laboratory synthesised Be based samples, using mainly Raman microscopy. Tiles coming from the inner and outer limiter tiles of JET (2010-2012 and 2012-2013 campaigns) are studied, together with their castellation sides. Besides, samples synthesised either by bombarding Be and BeO layers with D or by codepositing Be and O plus D, to mimic various mixed material possible formations, are also studied for comparison. A large amount of data collected on the JET limiter tiles are compared statistically to data collected on laboratory synthetized samples.

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Abstract Withdrawn

Impact of moisture on the disintegration of co-deposited layers and on dust generation: reactor safety case.

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In a system with kilometres of pipes there is a risk of cooling agent leakage, which is of particular concern for a nuclear device with a vacuum vessel. In the ITER safety assessment the impact of massive water ingress on the stability of co-deposited layers on plasma-facing components (PFC) is taken into account. Only the impact of ambient atmosphere on the layer detachment was discussed [1], but systematic study is missing. There are two major questions.

- Will the hot moisture/liquid water enhance the layer detachment and dust generation?
- If "yes", will it happen directly under exposure to water or upon drying the tiles?

Answers to these points are the aims of the recently initiated research programme which comprises: very detailed pre-characterisation (quantitative surface analyses) of actual tokamak tiles with flaking and with well-adhering deposits, treatment under moisture and then analyses. The project has two main phases. Complete feasibility studies (all steps listed above) are carried out on PFC from the de-commissioned TEXTOR, i.e. on real "tokamakium" deposited on tiles engineered to handle the load of plasma wall interactions. This ensures identification of critical situations to be studied carefully in a second stage with components retrieved from JET-ILW, when the availability of relevant materials for experiments is very limited and there are problems in handling large amounts of deposit/dust-containing beryllium. This contribution provides an overview of activities of the first phase and advances in studies of materials from JET. The major points are:

(a) Experiments have been carried out with 1.5x1.5 cm samples cut from the inner bumper limiter tiles (10 years operation) with a thickness of 9 μm.
(b) Samples were treated in three manners: (i) immersing in water and then heating to 250 ^{o}C; (ii) heating only; (iii) pre-heating to 250 ^{o}C and then applying water to simulate thermal shock in contact with moisture, i.e. both liquid and vapour.
(c) Most dust was generated in case (iii) with water on hot surface. This is not surprising as this is by far the most intense reaction including the thermal shock.
(d) About 1 mg of dust per 1 ml of water was trapped in the liquid. When the water was evaporated the dust tended to stick and not be mobile.

The (iii) type interactions will be the focus for the second stage using both the JET ITER like wall samples as well as deposited films with material with similar thermal properties as beryllium to avoid forming beryllium dust. Several different types of characterisation are planned such as imaging, measurements of atomic content including fuel and measurements of mechanical properties of the remaining surface before and after exposure.

Results of this work clearly indicate the need to understand details of the interactions, to identify scenarios which may appear and what strategies should be used to mitigate the risks.

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**See author list in the paper, X. Litaudon et al., Nucl. Fusion, 57 (2017) 102001.*

Performance of RAFM steels under deuterium plasma exposure with addition of seeded impurities

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Reduced activation ferritic martensitic (RAFM) steels such as EUROFER are current candidate materials for a blanket structure in DEMO design. Due to availability as well as technical and economic advantages over a tungsten-coated concept, a bare RAFM steel as a plasma facing material is also considered in future power plants.

To understand the erosion mechanism and morphology changes set of EUROFER, P92 and Fe samples were exposed in the linear plasma device PSI-2 at a sample temperature of 450 K and 950 K with an incident ion flux of about $3\text{-}5{*}10^{21}$ $m^{-2}s^{-1}$, an incident ion fluence of $1{*}10^{26}$ m^{-2} and an incident ion energy of 65 eV. Samples were exposed to deuterium plasma and with additional seeded impurities of He, Ar, Ne or Kr. Assessment of change of the surface morphology and erosion of exposed samples was performed using scanning and transmission electron microscopy (SEM and TEM). Investigation of tungsten surface enrichment was analysed by energy-dispersive x-ray spectroscopy (EDX) and compared with Rutherford backscattering spectrometry (RBS) results. Investigation of deuterium retention was investigated using nuclear reaction analysis (NRA).

Surface W enrichment increases for both types of steels when exposed to D and D+He plasma at the temperature of 450K. Exposure to D+Ne and D+Ar plasma reduces the tungsten enrichment on the surface, which disappear completely when exposed to D+Kr and D+He+Kr plasma. Addition of heavier species to the plasma resulted in smoothing the steels surface. Samples exposed to D+Kr as well as D+He+Kr plasma are very smooth, with almost no surface roughness. W enrichment was found locally, located in the spike structures. He addition to the plasma leads to increased D retention measured by NRA, for both types of steels. Increase of the D retention should be attributed with nano-bubble formation under the surface of both steels. Measured sputtering yield increased from about $1\text{-}3{*}10^{-4}$ for samples exposed to D and D+He plasma to about $1\text{-}3{*}10^{-3}$ for exposures with D+Ne and D+Ar plasmas and $1\text{-}2{*}10^{-2}$ when exposed to plasma with Kr seeding, for both types of steels. Moreover the sputtering yield is higher for pure Fe when compared to steels when exposed to D and D+He plasma, which could be attributed to surface W enrichment. The difference in sputtering yield between Fe and steels gets smaller when exposed to D+Ne and D+Ar plasma, where measured W enrichment is reduced. Pure Fe and steels exhibit comparable sputtering yields when exposed to plasma with Kr addition, which is explained by complete removal of surface W enrichment.
In the contribution the influence of exposure temperature is also investigated.

Interaction of adhered beryllium-proxy dust with transient plasma heat loads

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The interaction of adhered dust with transient plasma particle/heat fluxes determines - to a large extent - dust survivability on hot surfaces, which is connected to various ITER safety issues. For the configuration of μm W dust adhered to planar W substrates, comprehensive studies have been carried out in various tokamaks and linear plasma devices with ELMs & ELM-like pulses up to 550 MW/m^2 (normal to surface) under different B-field inclinations[1,2,3]. These studies have revealed that: **(i)** adhered W dust and especially dust-clusters can melt under much lower heat loads than bulk W owing to the small contact area and the imperfect thermal contact, **(ii)** wetting-induced coagulation as a novel mechanism of cluster transformation into single large spherical grains, **(iii)** despite the perfect liquid W - solid W wetting, the ELM duration is short enough for spreading dynamics not to fully evolve due to motion arrest by resolidification, as a consequence, molten dust remains nearly spherical, **(iv)** the remobilization activity is very low due to contact strengthening by liquid spreading and sintering. Overall, it has been concluded that adhered W dust can survive repetitive strong ELM impact but is then hardly mobilizable.

On the other hand, there have been few experimental studies with μm Be-proxy dust adhered to planar W substrates and concerned rather modest heat fluxes[4]. The results could possibly vary from the case of W dust because: **(i)** the involved melting points are much lower and thus the resolidification timescale should be higher, **(ii)** liquid Be proxy – solid W wetting should be weaker, **(iii)** contact strengthening should be limited due to the different materials involved.

Here we report the first systematic experiments on the interaction of adhered Be-proxy dust with ELM-like pulses. 1-10μm size Al (chemical proxy), Cr (mechanical proxy) and Cu dust (thermal proxy) as well as W dust (reference) were adhered to different well-defined regions – spots – of the same W substrates by gas dynamics methods[5]. The samples were then exposed in the Magnum-PSI hydrogen plasma[6] to repetitive ELM-like pulses (B=1.5T, α=20-90^o, triangular pulse, q_{max}=150-250MW/m^2, f_{ELM}=2-4Hz, $\Delta\tau$=1ms, N_{ELM}=1-50). Preliminary SEM analysis revealed a drastic remobilization activity in all the Be-proxy spots, whereas as always nearly no remobilization occurred in the W spot: *In the Al spots*, the surface morphology and chemical composition of the newly-exposed contact areas had drastically changed with thin Al-W fuzzy-like structures forming. *In the Cu spots*, nearly all Cu dust had remobilized without leaving any trace in the exposed contact areas. *In the Cr spots*, the remobilization activity was strong and thin molten layers were occasionally observed around the contact areas. The results are consistent with the phase diagrams of the respective binary Be (proxy) - W systems and can be explained by intermetallic compound formation and solid solubility arguments.

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The effect of thermionic electron currents on molten tungsten splashing from melt pools

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The splashing of tungsten (W) melt formed by the high transient heat fluxes on plasma facing components (PFCs) during plasma disruptions and edge-localized modes (ELMs) is of major concern for large fusion devices such as ITER. The W-melt splashing can lead to greatly enhanced PFC erosion and plasma contamination by heavy metal impurities. One of the primary mechanisms driving the W-melt motion is the Lorentz (JxB) force that results from the interaction of electric currents flowing across the W-melt layer with the strong magnetic field used to confine the plasma. The quantification of these currents and their effect on the stability of the W-melt layer is not well understood. The computational modeling of the dynamics of W-melt motion under the influence of the JxB force are of great importance for predicting and mitigating the conditions leading to the W-melt splashing.

In the present work, a computational model is developed to study the effect of thermionic electron currents on the stability of W-melt pools, such as those that could form on the ITER tungsten divertor during ELMs. It includes thermal, viscous, gravitational, surface tension, and magnetohydrodynamic (MHD) effects, and employs the volume of fluid (VoF) approach for treating the free surface of the W-melt pool [1,2]. This VoF-MHD model is then implemented in 3D geometries using the OpenFOAM C++ libraries of various numerical and physical models. The thermionic electron current is added to the current induced by the motion of liquid metal in the magnetic field. This VoF-MHD model can predict the motion and splashing of melt layers from PFCs due to volumetric forces, plasma or vapor pressure effects, and surface wave growth with the ejection of molten droplets.

The VoF-MHD modeling is performed to predict the conditions for W-melt spalshing from pools of different sizes (depth and length) and varying thermionic current densities. Preliminary results indicate a significant splashing of W-melt for current densities greater than 10 A/cm². The ejection of molten droplets from the W-melt surface is not revealed in the absence of plasma/vapor flow. The shape evolution of the melt surface during the splashing is also investigated as a function of the contact angle of a W-melt wetted solid W-substrate. The effect of wettability on the W-melt splashing is studied using both the static and dynamic models of contact angle. Depending on the wettability parameters used in the modeling, it is observed that the W-melt surface can take a variety of shapes. The wetting of W-melt to W-substrate is found to be very influential in determining its regime of splashing from a pool. The break-up flow of W-melt develops when the wettability of the W-substrate is decreased or the contact angle is increased. It is observed that the W-melt layer can partially bounce off or completely detach from the W-substrate with further increase of the contact angle. This underlines the importance of proper choice of wetting parameters in the modeling of W-melt splashing.

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Abstract withdrawn

Abstract withdrawn

Video analysis of intrinsic dust events in Experimental Advanced Superconducting Tokamak (EAST)

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Dust particles are commonly found in magnetic fusion devices [1]. Mechanism of dust production includes the erosion of the plasma facing components(PFCs) materials by plasma sputtering, flaking of deposited layers, arcing, melting and enhanced heat loads onto PFCs during disruptions, edge localized modes (ELMs), and other transient events. Dust production and accumulation can lead to tritium inventory rise, radiological and explosion hazards and also causes safety and operational issues in tokamaks such as ITER.

In this study, dust events recorded by fast framing cameras in between 5000 fps to 16000 fps during recent operational campaigns in EAST tokamak have been analyzed by the TRACE (**TRA**cking and **C**lassification of pinpoint **E**vents) code [2]. Fast Cameras combined with wide angle endoscope system for visible light observation make it possible to use different fields of view (FOV) and observe a wide area in EAST including the upper and lower divertor and the inner and outer walls [3]. The micron size dust particles with sufficient temperature can be detected as bright spots(local maximum of light intensity) in fast camera movies. TRACE code can efficiently detect those bright spots and track them in successive frames which enable us to calculate the dust production rates. Statistical analysis of dust production rates have been carried out for a large number of fast camera movies recorded in independent discharges during different campaigns. Dust rates have been found significantly larger than into ASDEX upgrade tokamaks with full tungsten PFCs. Dust particles are mainly produced during plasma current flattop and ramp down phase. Large dust explosion events caused by strong edge localized modes (ELM) filaments are often observed. During normal plasma operation dust particles originated from lower divertor move in toroidal direction while particles from upper divertor move towards last closed flux surface(LCFS).The dust particles trajectory and velocity following disruption are found inconsistent with gravity fall. The detailed investigation of the influence of the plasma current, plasma configuration, direction of vertical displacement event, plasma disruption energy, ELM size and frequency, power of different heating schemes, pulse length, cameras FOV and wall conditioning in different campaigns on dust observation rate will be presented.

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Modeling of erosion and deposition on ITER diagnostic first mirrors during glow discharge cleaning

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Most of the optical diagnostic systems in ITER will use metallic first mirrors (FM) as plasma-viewing elements. The possible degradation of their surface reflectivity as a result of material deposition can severely impact diagnostic functionality. Currently, modeling of FM deposition/erosion during plasma operations is associated with very large uncertainties. An additional unknown is the effect of glow discharge cleaning (GDC), which is the primary conditioning technique planned on ITER [1]. Since GDC operates in the absence of magnetic fields, GDC ions accelerated by the cathode sheath voltage drop (~500 V) will impact both the mirror surface and the sides of diagnostics apertures (stainless steel on ITER) built into the first wall, leading to both erosion and deposition on the mirror located behind the aperture. In addition, depending on the gas pressure and aperture dimension, penetration of the GDC plasma into the aperture might occur. To investigate these issues, a series of detailed hydrogen GDC experiments, reported in a companion paper [2], have been performed in a dedicated test chamber to study FM erosion/deposition and ion penetration in different geometries: a replica of the FM unit recently exposed in JET [3], a cylindrical and a conical aperture.

This paper presents modelling of plasma penetration into first mirror geometries and subsequent material deposition. A self-consistent 2D fluid sheath model [4] is first used to model the GDC plasma, providing estimates for the potential distribution on structures in the system and the incident ion fluxes and energies. The output from the plasma sheath code is then used as the particle flux input for 3DGAPS, a Monte Carlo impurity transport code [5], which simulates the material erosion and deposition. Assessments have been performed for two different first mirror geometries (conical and cylindrical), with a 500V anode potential and a range of pressures between 0.1 Pa to 5 Pa. The results are compared with data obtained from the experiments conducted at MEPhI [2].

Results from the sheath model show good agreement with the ion current density measurements in the MEPhI experiments. Preliminary findings from 3D GAPS simulations show very low net mirror deposition, also in agreement with experimental observation.

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3D global impurity transport modeling with WallDYN and EMC3-Eirene

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The global impurity migration code WallDYN [1] has been successfully applied to modeling Be migration in JET and making predictions for ITER [1]. WallDYN uses DIVIMP [2] for plasma impurity transport and is thus limited to toroidally symmetric geometries. While the plasma and SOL in these devices are essentially toroidally symmetric, the details of the first wall are not. They consist of 3D structures like poloidal limiters. Therefore, the calculations in [1] only gave toroidally averaged numbers of impurity deposition and layer growth.
Making more detailed predictions on deposition patterns or modelling Stellarator devices such as W7-X, which are not toroidally symmetric, requires taking the 3D structure of the first wall and the far SOL into account. For WallDYN this requires switching from the 2D code DIVIMP to the 3D SOL and impurity transport code EMC3-Eirene [3].
The coupling between WallDYN and EMC3-Eirene (≡WallDYN3D) follows the same scheme as with DIVIMP: Using an EMC3-Eirene calculated background plasma, the redistribution of impurities is recorded in a charge state resolved redistribution matrix $R_{i\rightarrow j@z}$ which denotes the fraction of material launched from wall tile i that ends up on wall tile j at charge state z. The information about impurity transport in $R_{i\rightarrow j@z}$ is then used in WallDYN to calculate the redistribution of eroded material and growth of layers.
In this contribution, we will show the first application of WallDYN3D by modeling the N migration in AUG from the puffing location at the divertor dome to the deposition location at the mid-plane manipulator (MEM). This N seeding experiment together with 2D WallDYN calculations was already presented in [4] where it was shown that this 3D problem could not be modeled with DIVIMP but requires a 3D SOL and impurity transport model. In revisiting this data, we model the radial N deposition profile on the MEM using WallDYN3D and compare the trace impurity transport models in DIVIMP and EMC3-Eirene focusing on the limitations of the latter.

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Elemental and topographical imaging of microscopic variations in deposition on NSTX-U and DIII-D samples

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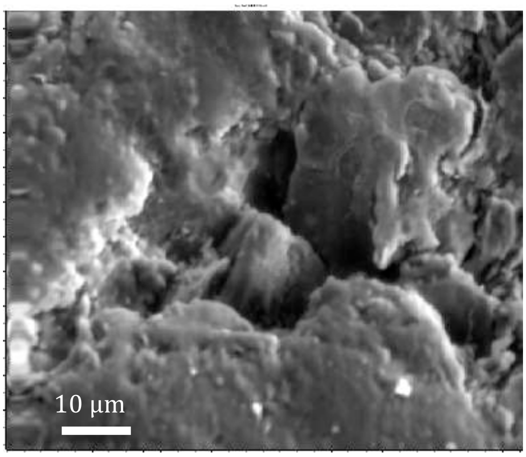
Tokamak plasma facing components have surface roughness that can cause microscopic spatial variations in erosion and deposition and hence influence material migration, erosion lifetime, dust and tritium accumulation, and plasma contamination. This has been confirmed in micro-ion beam analysis of JET tiles with a spatial resolution around 10 μm[1] and recent studies of aluminum migration using the DiMES probe[2], however high spatial resolution measurements on the scale of the surface roughness have been lacking. We will present elemental images of deposition on NSTX-U and DiMES graphite samples performed with a Scanning Auger Microprobe at sub-micron resolution that show strong microscopic variations and correlate this with 3D topographical maps of surface irregularities. The NSTX-U samples were boronized and exposed to deuterium plasmas; the DiMES samples have localized Al and W films and were exposed to dedicated He plasmas. Topographical maps of the samples are performed with a 3D confocal light microscope and compared to the elemental deposition pattern. The results reveal deposition concentrated in areas shadowed from the ion flux, incident in a direction calculated by taking account of the magnetic pre-sheath.



SEM image of pore on NSTX-U divertor tile

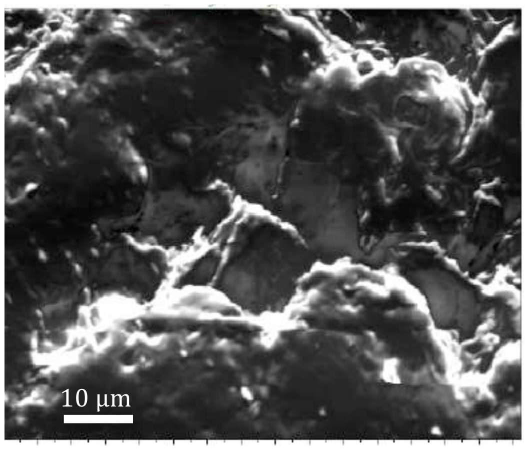


Image of same region from 171 eV boron KLL Auger electron emission.

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Impact of surface roughness on the erosion of fusion relevant materials: comparison of experiment to morphology sensitive Monte-Carlo BCA codes

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In a future fusion reactor and in DEMO the lifetime of the plasma facing components (PFCs) will be mainly limited by plasma-induced erosion due to ion and neutral particle bombardment [1]. Sputtering of the wall material will in addition cause modifications of the surface topography (i.e. roughening or smoothening, ripple or blister formation) of the PFCs which in turn will influence the sputtering behaviour [2, 3].
We have investigated the influence of surface morphology modifications on the sputtering of fusion relevant materials by using a highly sensitive quartz crystal microbalance (QCM) technique. The morphology changes are induced by prolonged irradiation of 400 nm thick Fe and W films (25% roughness), by mono-energetic Ar ions at various angles of incidence. Atomic force microscopy measurements are performed to analyse the sample topography before and after irradiating up to a fluence of 10^{22} Ar/m^2 and to determine surface roughness parameters. Numerical modelling with dynamic and morphology including Monte-Carlo BCA codes, like SDTrimSP-(2D and 3D version) and TRI3DYN are performed for comparison [4-6].
Our investigations show that by including the local distribution of projectile impact angles, as derived from AFM measurements, as well as the elemental depth profile of the samples as an input to SDTrimSP-2D the agreement between experiment and simulation is substantially improved [3]. Further investigations with 3D Monte-Carlo BCA codes gave deeper insight into the dynamics of surface morphology changes and its influence to sputtering.

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Co-deposition of deuterium and impurity atoms on wall probes in the divertor of JET with ITER-like wall

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Plasma-wall interactions, material migration and the resulting surface modification of plasma facing components are identified as key elements in the preparation for future fusion devices [1]. To facilitate material migration studies in JET with the ITER-like wall, a number of probes have been installed in the divertor and on the main chamber wall [2]. They are retrieved for ex-situ analyses during major shut-downs. The aim of this contribution is to provide an account of the surface modification of covers for quartz crystal microbalance (QMB) erosion/deposition monitors and so-called spatial blocks (1.5x1.5x1.5 cm^3) attached to the ribs of the inner and outer divertor carriers. The components were retrieved from the divertor between 2012 and 2016, i.e. after three campaigns (ILW-1 to ILW-3) with the energy input 150, 201 and 245 GJ, respectively. Time-of-flight elastic recoil detection analysis (ToF-ERDA) and nuclear reaction analysis (NRA) were used to map the distribution of deposited species on the component surfaces and to obtain quantitative depth profiles of hydrogen, deuterium, nitrogen and tungsten in layers primarily made up of beryllium, carbon and oxygen. Improved resolution, achieved by separation of elements based on stopping power in a gas ionization chamber as described in ref. [3], also allowed the detection of boron despite the presence of pronounced carbon and beryllium signals. Compensation of the ToF-ERDA data for detection efficiency and ion-induced detrapping of deuterium during the measurement gave quantitative agreement with the NRA measurements, which strengthens the validity of the results. The points below summarize the present conclusions:

(a) The thicknesses of deposited layers on the studied components from the 2011-2012 campaign are limited to less than 4×10^{18} at/cm^2 or a few hundred nanometers.
(b) The amount of co-deposited deuterium is found to be proportional to the total amount of carbon in a layer, rather than the layer thickness.
(c) The presence of a few, in some cases up to 10 at. % of nitrogen-14 is detected on surfaces of all studied probes.
(d) High resolution microscopy revealed the layer structure and the presence of dust particles.

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*See the author list of X. Litaudon et al., Nucl. Fusion 57 (2017) 102001.

Demonstration of suppression of the dust generation and partly reduction of the hydrogen retention by tungsten coated graphite divertor tiles in LHD

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Three sets of tungsten coated graphite divertor tiles (VPS-W tiles) were installed in the closed helical divertor of the large helical device (LHD) instead of the graphite divertor tiles in 2012FY plasma campaign for examine the plasma wall interaction with the LHD plasma. VPS-W tiles well worked for suppression of the dust generation and partly reduction of the hydrogen isotope retention in the LHD.

The first wall panels and divertor plates of the LHD are consisted by stainless steels (SUS316L) and graphite, respectively. Long pulse discharges in LHD were greatly affected by two major issues that are closely related to plasma wall interaction (PWI) [1]. The first issue is the dynamic change of the wall pumping rate during the discharge. The second issue is the termination of the discharge with the exfoliation of the carbon based mixed-material deposition layers. Those issues are closely related with erosion and re-deposition of the graphite divertor tiles, because eroded carbon material deposits on the plasma facing components (PFCs), and change the physical properties of the PFCs. Therefore, we tried to exchange the three sets (6 pieces) of the graphite divertor tiles to VPS-W tiles with the 125 μm thickness of the W coating layer to confirm the effects of the reduction of the undesired mixed-material deposition (re-deposition) layer.

After the single plasma campaign of the 2012FY, we confirmed drastic suppression of the mixed-material deposition layer near the VPS-W tiles, and local reduction of the hydrogen retention on the VPS-W surface by means of ion beam analysis. Surface of the VPS-W tile can be locally categorized as the erosion dominant area and the deposition dominant area. In the former area, amount of the hydrogen retention was estimated to be ~1×10^{20} H/m^2. On the other hand, deposition dominant area, which covered with mixed-material deposition layer, amount of the hydrogen retention was over ~2×10^{21} H/m^2. This means that if we would want to totally suppression of formation of the mixed-material deposition layer and hydrogen isotope retention, carbon material should be completely eliminating in the PFCs. This paper also will be presented the detail of the surface morphologies of the VPS-W surface.

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The adhesion of tungsten dust on plasma-exposed tungsten surfaces

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The phenomenon of adhesion is central in many tokamak dust issues such as the development of removal techniques, the efficiency of post-mortem collection activities, resuspension in case of loss-of-vacuum accidents, remobilization under steady state or transient plasma conditions [1,2]. This motivated adhesive force experiments based on the electrostatic detachment method [3,4] as well as theoretical calculations based on the Lifshitz theory of Van der Waals forces [5].

Systematic measurements of the adhesion of spherical micron W dust on planar W surfaces have revealed the importance of surface morphology [3]. In particular, due to nano-roughness: (i) adhesion is not dominated by metallic bonding interactions but by London dispersion forces, (ii) adhesion is not deterministic but acquires a statistical nature and is better described by cumulative distribution functions. However, these experiments employed W samples that have not been exposed to plasmas. Plasma incidence can modify the surface morphology but also alter its chemical composition (removal of humidity and other contaminants that remained despite pre-cleaning, hydrogen retention) which could drastically affect the adhesive force.

Here we report the first pull-off force measurements of W dust adhered to plasma-exposed W surfaces. Planar W samples were first exposed to the deuterium plasma of the GyM linear device [6] with $n\approx5.2\times10^{10}\mathrm{cm}^{-3}$, $T_e\approx6\mathrm{eV}$, $V_{bias}\approx-400\mathrm{V}$ (-100V), $t_{exp}\approx90\mathrm{min}$, $T_{fin}\approx360^{o}\mathrm{C}$ ($270^{o}\mathrm{C}$). Nearly monodisperse spherical W dust (5, 9, 15μm) was then adhered to the samples through mechanical collisions below the sticking threshold with the aid of gas dynamics methods [1]. The samples were then adjusted into the hollow electrode of an electrical mobilization setup that allows the measurement of the adhesive force by lifting-up the dust grains after exerting an electrostatic force of well-known magnitude. Reference measurements were also carried out on the same samples prior to plasma exposure and after two months in air post plasma exposure.

The measurements revealed that the high-bias plasma exposure reduced the mean adhesive force by 50% for all monodisperse dust populations. This has been unambiguously attributed to the plasma cleaning (mostly sputtering but also desorption) of most surface contaminants as corroborated by the following evidence: **(i)** SEM analysis revealed no changes of micron-scale roughness and AFM analysis a small increase of nano-scale roughness after plasma exposure, which imply that surface modification did not play a role, **(ii)** ATR analysis revealed a complete disappearance of OH bands - present due to humidity - after plasma exposure, which implies that chemical modifications did occur, **(iii)** two-month exposure to air led to the retrieval of the high adhesive force values, **(iv)** samples biased at -100V were characterized by a quite smaller reduction of the adhesive force, being less subject to sputter-cleaning. The drastic effect of humidity on W-on-W adhesion and the decrease of adhesion after sputter-cleaning have also been confirmed by measurements on samples exposed to Ar sputtering discharges.

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Application of target surface illumination with laser radiation for in-situ examination of tungsten erosion under transient thermal loads

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Transient events with thermal load up to 1 MJ m^{-2} [1] on divertor plates are expected during operation of ITER-project. Even mitigation of type 1 ELMs does not totally exclude erosion of plasma-facing components. Researches on tungsten damaging during thermal shocks are wide spread but they mostly include post mortem analysis of the exposed samples. Experiments carried out on BETA (Beam of Electron for material Test Applications) are aimed for in-situ investigation of tungsten erosion during transient heating by means of optical diagnostics. Heat loads with heat flux factor (HFF) up to 300 MJ $m^{-2}s^{-0.5}$ at maximum, duration up to 300 μs and FWHM about 10 mm [2] can be generated on BETA-facility.

Two types of surface diagnostic sets are applied on this facility: recording of thermal radiation of the sample and imaging of the target illuminated with light of cw laser. Fast CCD-cameras and photodiodes followed by ADC are used for recording of spatial and time-resolved temperature distribution on the surface. Tungsten tiles were polished to mirror-like state for this research. Sample surface was illuminated with cw laser so that specularly reflected beam was absorbed. Such configuration reduces total intensity of the recorded laser radiation and makes it possible to detect of sub-micron surface modification due to deflection and scattering of the laser beam. A fraction of the scattered laser radiation was deflected by beam splitter and focused into optical fiber outfitted by photodiode and ADC. This diagnostic set allows to obtain data on material erosion during and after heating load even near room temperatures when thermal radiation of sample is too low.

Investigation with mirror-like samples of tungsten exposed at HFF of 30 MJ $m^{-2}s^{-0.5}$ revealed two types of material modification. At first, almost linear growth in the intensity of the scattered laser light was observed during entire heating pulse (250 μs). This signal slowly decreases during cooling (5 ms) to a constant level that is higher than the initial one. After this initial increase, there is one more abrupt rise that happens after relatively long delay (about 1 s). This surge occurs almost instantly (30 μs) and has no reduction after it. In the preliminary analysis, it was concluded that former rise is connected with surface roughening during transient heating and the latter one can be interpreted as formation of net of major cracks and surface elevation near the crack edges that is observed with visual inspection and laser profiler.

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Survival and in-vessel redistribution of disruption-induced beryllium droplets in ITER

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The potential for metallic dust and droplet formation in fusion reactors poses a variety of safety and operational issues. In ITER, the quantity of tungsten (W) and beryllium (Be) dust in the vessel is restricted by the nuclear licensing agreement, highlighting the need for predictive estimates of the dust inventory, originating from both normal and off-normal operation.

In terms of the potential quantities of material which can be evolved on ITER, surface melting of the Be first wall (FW) panels during the thermal and current quenches phases of major disruptions (MD) and vertical displacement events (VDE) is expected to be one of the principal contributors [1]. Such transient induced melting has already been observed on the JET Be wall at much lower plasma stored energies than those achievable on ITER [2]. If splashing of surface molten layers occurs (more likely for Be than W due to the lower mass), the ejected molten droplets may solidify into dust depending on the balance between heat fluxes from the quenching plasma and cooling processes such as vaporization and thermal radiation.

Here, we report on MIGRAINe dust transport code simulations [3] of the motion and temperature evolution of Be droplets during two realistic cases of late disruption mitigation in which the thermal quench (TQ) of MD and VDE events is unmitigated, resulting in Be melting and in which neon is injected only after the TQ to mitigate thermal loads and electromagnetic forces during the current quench. These simulations implement an updated droplet-plasma interaction model [4], which targets magnetized electron collection and thin-sheath ion collection, as well as electron emission processes induced by electron and high-Z ion impacts. Using plasma profiles obtained from DINA disruption simulations and exploring a wide range of initial conditions (droplet size, speed and injection angle) consistent with available results on droplet ejection from molten layers, each simulated particle is tracked until it is vaporized by the plasma or stops moving upon sticking/splashing on the FW/divertor boundaries.

The final amount of solid dust and its spatial deposition pattern are obtained by weighting the single-trajectory output with prescribed initial size and velocity distributions. The results show that at most 5% of the initial Be mass is converted into dust, which tends to accumulate near the outer divertor leg, and that MDs and VDEs mainly differ in which initial droplet sizes lead to maximum conversion. The characteristics of the deposited dust at selected locations are also estimated, providing input to guide both the design of the dust endoscope planned for ITER and other diagnostic systems potentially affected by the presence of dust.

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Relation between irregularities in the tungsten structure and hot spots observed in experimental simulations of ITER-relevant thermal shocks

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Transient thermal loads in the ITER divertor is one of the key problems for ITER operation. Recent calculations [1] show how difficult it is to avoid melting, seeking a compromise between the risks of melting the edge and the full wetted surface of the tungsten monoblock by uncontrolled ELMs. In this case, it is assumed in the calculations that tungsten is heated homogeneously by a uniform flux of plasma particles. However, experiments at the Budker Institute have shown that even with a uniform heating of the tungsten surface, on the image of the surface in its thermal radiation, hot spots are visible. Experiments are carried out on BETA (Beam of Electron for material Test Applications) facility and are aimed at *in-situ* investigation of tungsten erosion during transient heating by means of optical diagnostics. At the BETA facility, surface heating of samples is available with a duration of up to 300 μs and with a Gaussian heating profile having FWHM of approx. 10 mm, with a heat flux factor (HFF) of up to 300 MJ m^{-2} $s^{-0.5}$ [2], but a value of HFF about 30 MJ m^{-2} $s^{-0.5}$ was used in the experiments described below. The optical technique of fast imaging of tungsten surface in its own thermal radiation was described recently [3].

Two types of hot spots were observed by this method: small-scale (~ 0.1 mm) ones, visible during heating mainly on tungsten with the direction of grains perpendicular to the surface, and larger (0.2-1 mm) hot spots, more detectable later at the cooling stage. These latter hot spots are more typical for tungsten with a grain direction parallel to the surface. Precise mapping permits to correlate the thermal images with the SEM-images of the sample, and then with the images of cross-section of the sample. This comparison makes it possible to explain the appearance of hot spots of a larger size by the formation of cracks, parallel to the surface

The numerical simulations were carried out in order to calculate the geometry and size of the parallel to the surface cracks using the measured temporal and spatial dependences of the temperature in the overheated areas near the cracks on the exposed surface. Two-dimensional heat equation was solved in an area with a crack parallel to the surface, assuming the absence of the thermal conductivity between the boundaries of the crack

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First spectroscopic measurements of carbon erosion and transport in the divertor plasma of W7-X

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One of the key issues to achieve long stable discharges in fusion devices is the minimization of the erosion source of plasma-facing components which ensures a long lifetime of components as well as a low impurity concentration in the plasma. Key processes governing the erosion of graphite-based components is the physical and chemical sputtering of carbon [1] of which the latter is surface temperature-dependent and can occur also at low impact energies. The gross erosion rates owing to physical and chemical sputtering processes have been investigated extensively by optical emission spectroscopy observing the spectral emission of C I (909.5nm), CII (514.5nm), CH and C_2 in e.g. TEXTOR [2], JET [3], and DIII-D [4]. The emission bands of molecular species such as the CH Gerö band and the C_2 Swan band give useful information about the source location, its strength and dissociation processes of chemically released hydrocarbons from bare graphite-based PFCs or hydrogen-rich carbon layers due to co-deposition processes.

Wendelstein 7-X (W7-X) is a large advanced stellarator with the aim of demonstrating steady-state operation in high performance plasmas at high density and temperatures. In W7-X, ten test divertor units (TDU) made of graphite, two in each period of the fivefold symmetric device, have been installed. This so-called island divertor system should allow effective divertor operation with good particle and energy exhaust for a wide range of plasma conditions and magnetic configurations. The study of plasma-wall interaction (PWI) processes in the 3D divertor region is a critical subject in the initial operational phase before the divertor with actively cooled PFCs is installed.

In order to study the C erosion at the divertor target plates and the transport in divertor plasma, a set of spectroscopic diagnostics have been installed in W7-X. Here, we report on first results obtained from a tunable, fibre-coupled Czerny-Turner spectrometer with high spectral resolution. This spectrometer is attached to K port using 25 fibers, line of sights of which cover part of horizontal and vertical divertor plates. Initial studies with a subset of fibres focused on the study of the fraction of chemical and physical sputtering sources at the target plate as well as the dissociation chain of released hydrocarbons under typical edge plasma conditions in W7-X. Comparison of He/H and pure H plasmas provide a variation of the chemical contribution. Additional information concerning the break-up processes were obtained from the line shape analysis of the CI multiplet at 909.5nm.

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Confinement of helium precipitate growth within metal nano-layers and the implications to radiation damage

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Degradation of plasma-facing materials (PFMs) such as tungsten due to precipitation of implanted helium (He) from the fusion plasma is a key concern. Equiaxed He precipitates link up and form a foam-like structure, devastating mechanical cohesion and thermal conductivity [1]. Decades of research have mapped out the fate of He precipitates in metals, from nucleation and growth of equiaxed bubbles and voids to formation and bursting of surface blisters. By contrast, we show that He precipitates confined within a model nano-scale metal layer system (V-Cu-V) depart from their classical growth trajectories: they self-organize into elongated nanochannels as shown in figure below [2]. These channels form *via* templated nucleation of He precipitates along layer surfaces followed by their growth and spontaneous coalescence into stable precipitate lines. The total line length and connectivity increases with the amount of implanted He, indicating that these channels ultimately interconnect into percolating "vascular" networks. Vascularized metal composites promise a transformative solution to He-induced damage such as foam-like structures by enabling *in operando* outgassing of He and other impurities during the plasma exposure while maintaining material integrity.

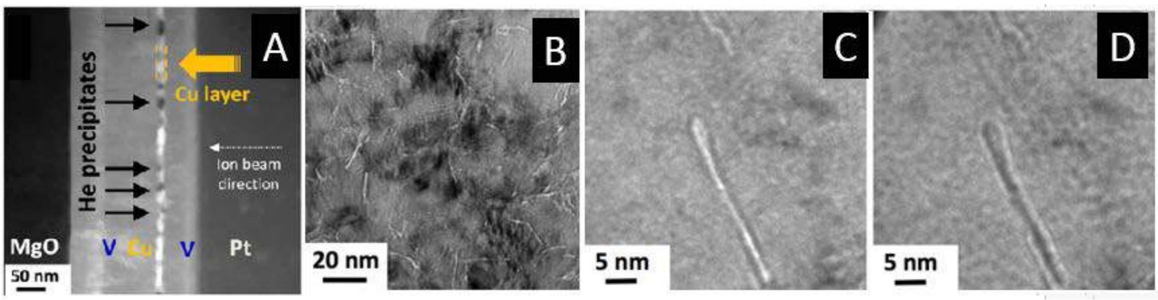

Figure: Cross-section (A) and in-plane (B) TEM imaging of helium precipitates in metal nano-layer interfaces. Direct observation of a helium nanochannel: (C) under-focus and (D) over-focus.

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TEXTOR whole-tokamak high-Z migration modelling and parameter studies with ASCOT code

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Global migration of high-Z impurities from eroded PFCs will influence the efficiency of energy production in future fusion power plants and the lifetime of PFCs. It is thus of highest economic interest to predict and control high-Z impurity migration. Transport codes can serve as a tool to answer this question and predict migration patterns. Their capabilities have been growing over the recent years to reproduce major plasma-wall interaction (PWI) phenomena, both on qualitative and quantitative level. Most of these simulations have been restricted to local scale, due to limited amount of information on global distribution and re-deposition of transported species.
The aim of this work is thus to combine results of comprehensive ex-situ analyses of high-Z metals on plasma-facing components (PFCs) from TEXTOR with ASCOT-based modelling. Tracer injection directly before carefully planned decommissioning and dismantling of the TEXTOR tokamak has offered a unique opportunity to investigate global migration of heavy impurities in great detail. Analysis of 140 tiles throughout the whole machine has made available global deposition maps on molybdenum, tungsten and medium-Z metals (Cr – Cu).

In this work, the global deposition map of molybdenum has been reproduced by ASCOT. ASCOT is a first-principle particle-following code that simulates the behaviour of minority particles in tokamak geometry using Monte Carlo methods to account for stochastic processes related to interactions with the plasma background. ASCOT can handle full 3D wall structures which is very important for global transport modelling. The boundary conditions were:

- simulation input consisting of the equilibrium magnetic field, full 3D wall structure of TEXTOR and parabolic fits to measured plasma density, temperature and rotation;
- injected Mo represented by 10.000 Mo^{3+} markers at the valve location with initial energy corresponding to the local temperature and isotropic velocity distribution;
- full neoclassical physics inherent in the simulations, with classical diffusion accounted for by simulating the full gyro orbits of the markers. The ADAS data base was used to calculate ionisation and recombination probabilities at each time step;
- marker tracing for 10ms, with most deposition taking place within 2ms.

Due to the lack of directly measured plasma profiles, both shape and values of the profiles were varied to find the best correspondence to measured global deposition. ASCOT allows separation of different mechanisms (drifts, rotation) which helped understanding the deposition pattern. Also anomalous diffusion and re-erosion were studied by including *ad hoc* models. Preliminary first-principle simulations have already shown qualitative agreement with measurements: (i) most markers were deposited on the main PFC, (ii) deposition on the inner bumper limiter was pronounced at limiter top and bottom. However, the marker migration directions differ: mainly toroidal in ASCOT vs. mainly poloidal in the experiment.

Deposition of impurity metals in JET ITER-like Wall campaigns

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There have been three operational periods of JET with the ITER-like Wall – 2011-2 (ILW1), 2013-4 (ILW2) and 2015-6 (ILW3). Post-mortem analysis of divertor and wall tiles removed after each period by Ion Beam Analysis show that the principle plasma impurity that was deposited in the divertor was beryllium (Be) that had been eroded from the main chamber tiles (which are mostly made of solid Be, or coated with Be); Be was also re-deposited in areas of the main chamber. However, within the deposits were also quantities of the metals Ni, Fe and Cr in approximately the same ratios as found in Inconel, and small amounts of tungsten (W).

Much greater levels of Ni, Fe and Cr were found on Tile 6 (outer divertor corner) after ILW2 than ILW1, with a similar level after ILW3 to that after ILW2. This is attributed to the failure of some tie-rods in Tile 7 which then slipped down so that their ends were resting on Tile 6, exposing a few centimetres of their length to power load. To determine if this contamination was merely local, tiles from the main area of plasma impurity deposition from each period (Tile 1 at the top of the inner divertor leg) have been analysed. Similar peak Ni, Fe and Cr concentrations were found on the apron of Tile 1 after both ILW1 and ILW2 (10×10^{17}, 4×10^{17} and 3×10^{17} atoms cm^{-2}, respectively), but the average over the whole Tile 1 was greater after ILW2 than ILW1 (4.92×10^{17} compared to 2.87×10^{17} atoms cm^{-2} for Ni, for example); the level was found to have returned to a lower level after ILW3. These results are in qualitative agreement with spectroscopic data. Ni, Fe and Cr have always been found at low levels within deposits in JET, probably due to erosion of the vessel wall and fittings (made of Inconel) by charge-exchange neutrals, but these ILW results indicate that impurities generated in the very corner of the outer divertor at Tile 6 are able to reach the main plasma and be re-deposited elsewhere in the vessel.

All divertor surfaces are of W – either as ~25 µm coatings on CFC or in the case of Tile 5 as solid W. There are also some W-coated CFC tiles in the main chamber, but well recessed from confined plasmas. Small concentrations of W are found within deposits at all parts of the vessel: Data are only reported for deposits on components that do not originally contain W. W levels on the stainless steel inner and outer deposition monitor covers after ILW1 were $72\text{-}104 \times 10^{15}$ and $29\text{-}50 \times 10^{15}$ atoms cm^{-2}, respectively: levels on inner louvres after ILW1 and ILW2 were $25\text{-}48 \times 10^{15}$ and $5\text{-}32 \times 10^{15}$, respectively, and on outer louvres $80\text{-}107 \times 10^{15}$ and $17\text{-}42 \times 10^{15}$ atoms cm^{-2}, respectively: levels on a Be Inner Wall Guard Limiter (2XR10) were also about a factor of two greater after ILW1 than after ILW2 (e.g. peak values of $500 \times 10^{15}$and 250×10^{15} atoms cm^{-2}, respectively. The decrease in W concentrations with time may be due to the decrease in the numbers of W particles dislodged by plasma interaction with coated artifacts, or by the coverage of W-coated surfaces with Be thus protecting against W erosion.

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Plasma Exposures of a High-Conductivity Graphitic Foam for Plasma Facing Components

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The plasma-surface interactions from samples of high-conductivity graphitic foam placed in 100 eV deuterium plasmas with densities as high as 10^{19} m^{-3} were investigated at the PSI-2 linear plasma device in Juelich and by exposure to W7-X plasmas using the Juelich manipulator in Greifswald. The purpose was to explore the possibility of using the material in a plasma facing component, and initial results were encouraging. Although some localized hot spots were noted, no cracking or ablation was observed. The PSI-2 samples received a fluence of $5x10^{25}$ m^{-2}.resulting in an average erosion of 39 microns or about 5 mg per sample. The exposures occurred at 200 ^{o}C and 500 ^{o}C, respectively. Spectroscopic data and weight loss measurements are reported. Residual Gas Analysis data were acquired to monitor sample outgassing. Laser-induced Breakdown Spectroscopy (LIBS) was used to measure deuterium retention in the porous foam and optical emission spectroscopy was used to identify outgassed species. After exposure, the surfaces were characterized with scanning electron microscopy and energy dispersive x-ray analysis. The graphitic foam has a thermal conductivity as high as 285 W/mK and is considered as a replacement to more exotic CFCs such as SepCarbNB31 or isotropic graphites like ATJ that are no longer produced but used in present-day tokamak experiments. Actively cooled monoblocks were made from the foam and underwent extensive materials characterization including infrared response studies at Oak Ridge National Laboratory. This material is under consideration for the proposed actively-cooled W7-X divertor scraper element.

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Physics Processes at the Plasma Material Interface

Cross-field electron transport inside an insulating cylinder of a baffled probe

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Short-circuiting of magnetized plasmas by conductive walls has been studied both as a basic phenomenon and for applications, including magnetic fusion devices. In this work, a different short-circuit effect is suggested that is due to bouncing of magnetized electrons off of the sheath near the walls. This effect is relevant to both non-conductive and conductive walls. Plasma-immersed wall experiments have been performed in a magnetized xenon plasma in a cross-field Penning configuration with density around $10^{12}cm^{-3}$ and an electron temperature around 2eV [1]. A cylinder with one open end and diameter of 1.4mm was placed across field lines so that electrons were blocked from reaching a wire recessed behind the shield while ions were unimpeded. This is accomplished by recessing the wire behind the shield further than an electron gyroradius but less than that of an ion. This configuration is relevant to the magnetically insulated baffled probe (MIB), a diagnostic for passively measuring plasma potential [2,3]. The reduction of electron current to the wire causes its floating potential to be closer to the plasma potential [3]. However, the measured electron current was much higher than expected even when the wire was recessed several electron gyroradii behind the baffle. The bouncing motion described above is suggested as a potential cause for the short-circuiting to the bulk plasma and has been studied with numerical approaches and with a separate experiment designed to isolate the effect. The proposed mechanism is a plasma-wall interaction that has a similar effect to the short-circuiting of plasma discharges caused by the Simon effect [4]. This work highlights an effect that may be important for cross-field transport near the walls in a variety of other configurations of magnetized plasmas.

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Concept of Effective Secondary Electron Emission Coefficient and V-I Characteristics of Glow Discharges

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Secondary electrons (SEs) emitted from the cathode surface determines all the discharge characteristics of direct current (DC) gas discharges. In dark discharges the discharge current ($\sim 10^{-6}$ A), plasma density, the photon and metastable fluxes is very low and hence the secondary electron emission (SEE) processes at cathode is dominated by ions [1]. These emitted electrons are characterized by the value of ion induced secondary electron emission coefficient (ISEEC, γ_i) and it is defined as the ratio between number of SEs emitted to the number of ion incident on the cathode surface. On the other hand, in glow discharge plasma (GDP) it is not the ions are the major species inducing secondary electrons (SEs) but the other non-ionic cathode directed species such as photons, metastables and energetic neutrals results from ions also induces SEs upon incident on the cathode surface. Adding, the contribution of non-ionic cathode directed species to the value of γ_i gives effective number of SEs emitted by single ion which is defined as effective secondary electron emission coefficient (ESEEC, γ_E) [1].
The value of γ_E is normally higher than γ_i and it depends on discharge conditions. Measurement of γ_E is difficult under GDP conditions. As a result, it is customary in plasma physics in most of the time to approximate the value of γ_E as γ_i. In contradiction to this approximation, we experimentally found that for Tungsten (W) and Copper (Cu) cathodes under identical discharge conditions, γ_i alone can't explain the measured discharge characteristics. On the other hand, the γ_E value obtained from the proposed self consistent model explains all the observed discharge characteristics. The proposed model is based on power balance at cathode [2, 3]. The model has been successfully applied for different operating gases such as Argon (Ar) and Nitrogen (N_2) and different operating pressures ranging from 0.15 mbar to 2.0 mbar. The results show that significant dependence of cathode material and discharge gas properties [4].

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Deuterium transport in a flowing liquid lithium loop under plasma bombardment

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It has been widely recognized that solid plasma-facing components (PFCs) would suffer from erosion and cracking when they are subjected to high power loads and particle fluxes in fusion devices. As a possible solution, the use of liquid metals for PFCs has been proposed because of their self-cooling and self-healing properties. Molten lithium is one of the candidates for this application because of its low atomic number and high absorptivity of impinging species. The understanding of hydrogen isotopes properties in liquid metals is essential for studying particle control in the edge plasma and tritium inventory issue.

Hydrogen transport parameters in liquid lithium have been investigated in plasma-driven permeation experiments by using a surface tension-mesh method in our previous work [1]. The precipitation of solid LiH under plasma bombardment has been observed. And natural convection effects and JxB forced convection effects on hydrogen transport in liquid metals have been investigated [2,3]. For any liquid lithium PFC concepts, the need for the liquid lithium recirculation is indispensable, because of the long-term tritium accumulation issue in a steady-state fusion system. A continuously flowing liquid lithium limiter based on the concept of a thin flowing film, has been successfully designed and tested in the EAST device [4]. The experiment has confirmed that the liquid lithium can be driven by built-in DC EM pumps to form a recirculating, closed loop system for fusion devices. However, there is no hydrogen extraction system for this closed loop, and the behavior of hydrogen transport in this system has not been considered. In the active radiative liquid lithium divertor concept, a liquid lithium loop with real time tritium removal techniques has been proposed. And it has been noted that the precipitation of solid LiT, the LiT collection and tritium recovery are important issues to be addressed [5].

In the present work, a proof-of-principle flowing liquid lithium loop with deuterium plasma implantation system and deuterium extraction system has been proposed. Deuterium transport behaviors, including deuterium diffusion, convection, desorption, LiD precipitation and dissolution along the loop are investigated by using a finite element analysis simulation. More detailed information will be presented in the coming conference.

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Effect of Bursting Process on Helium Bubble Evolution in the PISCES Experiments with Cluster Dynamics

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Plasma surface interactions in fusion tokamak reactors involve an inherently multiscale, highly non-equilibrium set of phenomena, for which current models are insufficient to predict the divertor response to and feedback on the plasma. In this presentation, we describe the latest code developments of Xolotl, a spatially-dependent reaction diffusion cluster dynamics code to simulate the divertor surface response to fusion-relevant plasma exposure. Xolotl is also part of a code integration effort to model both plasma and material simultaneously, as the gas accumulation in and release from plasma facing materials is important in determining fuel retention as well as wall recycling; the first benchmark for this effort is a series of dedicated PISCES linear device experiments.

We present the bubble bursting model created to take into account the helium release of over-pressured bubbles, which was added to the near-surface helium dynamics previously presented in [1] (including biased drift of mobile helium clusters and modified trap mutation). The tungsten adatom formation due to loop punching and trap mutation, the sputtering of tungsten atoms, and the tungsten ion re-deposition, are also taken into account to predict the surface morphology changes. These changes are studied under a set of different simulation conditions. Finally, 2D and 3D simulations are used to evaluate the evolution of surface roughness, which are compared to experimental results from PISCES as well as the sub-surface material composition.

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Atomistic Insight into D and He Near-Surface Implantation and Sputtering of Cubic-SiC Crystallographic Surfaces*

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We present results from atomistic simulations of near-surface (~10-50 nm) D/He and material interactions to better understand the processes of species implantation and sputtering of SiC. We observe stable He clusters within tetrahedral intersitices during implantation and further explore the synergistic effects between D and He within interstial and vacancy lattice sites. We focus on emphasizing these effects for (100), (110), and (111) particle-facing crystallographic surfaces of cubic-SiC. In the event of sputtering, we find preferential sputtering of either Si or C species depending on the surface and terminating species, for example, the (111) surface shows less sputtering of Si or C compared to (110) and (100) surfaces. We compare and contrast these findings to available experimental results for polycrystalline and composite SiC. Additionally, these results are compared to sputtering yields calculated from binary collision approximation (BCA) codes TRIM.SP and MARLOWE [1,2], where the surface binding energies (E_b) have been obtained from atomistic calculations [3] using different interatomic potentials [4–6] and ab-initio methods. We find better agreement between atomistic and BCA simulations with the calculated E_b compared to using the usual enthalpy of sublimation (i.e., cohesive energy) [1,2]. This work provides an improved insight into plasma-material interactions and the potential of low-Z plasma-facing materials by specifically tuning surface structure to control fuel recycling and reduce erosion. Furthermore, these results also have implications for the degree of structural disorder and bubble growth in SiC.

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Nonambipolar mechanism of plasma facing material heating under very high heat loads

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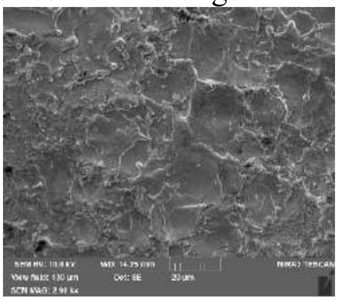
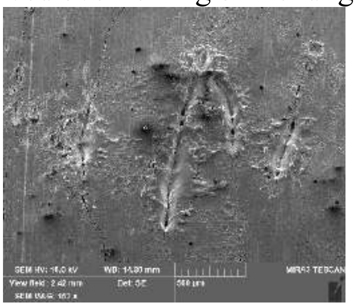
In reactor size fusion devices such as the ITER and DEMO are expected extreme high heat loads on the divertor plates both during steady state and transient events (disruption, VDE, ELMs). Plasma-surface interaction at a very high heat load on the material surface in a fusion device is influenced by several multiscale effects including surface erosion, redeposition of eroded materials, melting and melt motion over the surface, inhomogeneous solidification leading to the growth of the stochastic surface structure [1] and collective feedback effects. Conditions in plasma sheath over the roughen surface and melts are favourable for arcs and sparks ignition affecting the overheating of the material surface. Nonambipolar flow toward the surface is expected due to arcs and sparks leading to enhanced heating of PFMs under a very high heat load [2]. This ecton mechanism is caused by the phenomenon of continuous explosive electron emission during successive sparks ignition [3]. Unlike standard thermoemission, such mechanism can dramatically increase electron emission; as a result, sparks and arcs activity leads to a surface overheating and melting.
Experiments on the T-10 tokamak with ITER-grade tungsten PFMs under a powerful plasma load during ECR heating are presented. In such discharges, the tungsten plates installed at the inboard of the vessel were heated up to temperature exceeded 2000^0C, local thermal load were of more than 50 MW/m^2 on the plates edges leading to surface melting. Sparking (Fig.1a), arcing (Fig.1b), deep cracks and edge melting were observed on the ITER-grade tungsten tiles after the series of experiments. It is supposing that the nonambipolar flow due to the arcs and sparks effects on the tungsten tiles [2] leads to the observed overheating of the tungsten surface.


(a)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Sputtering of Li-C-O compounds under deuterium ion bombardment of plasma-facing component materials

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Lithium conditioning of plasma facing components (PFCs) in magnetic fusion devices has improved plasma performance and lowered hydrogen recycling due to the efficiency of lithium in binding hydrogen isotopes. In current fusion experiments [1], the presence of impurities inevitably creates Li-O and Li-C-O compounds. Therefore unravelling the fundamental mechanisms for hydrogen retention and sputtering for these lithium compounds is crucial for the application of lithium as a plasma-facing material. Additionally, Li-C-O may be of interest as new fusion-relevant materials. For example, lithium carbide has recently been proposed as a prospective breeder material for fusion reactors [2].

This work presents studies of deuterium ion sputtering and retention in Li-O and Li-C-O compounds using surface science techniques. The present experiments are performed on thin lithium films (10-20 monolayers) deposited on a Ni(110) substrate to avoid effects due to grain boundaries, intrinsic defects, and impurities diffusing to the surface. A D_2^+ beam was generated in a differentially pumped ion gun in the energy range of 400-1200 eV. Temperature programmed desorption (TPD) and Auger electron spectroscopy (AES) were used to measure deuterium retention and sputtering rates under different conditions. Li-C-O compounds were formed by dosing oxygen and hydrocarbons on pure lithium films at various surface temperatures.

Our experiments showed that upon oxidation, lithium thermal stability increased by 350 K. A drop of 60% in total hydrogen retention as the temperature varied from 90 K to 520 K was observed in both pure lithium and lithium oxide and confirmed by molecular dynamics (MD) simulations. Additionally, hydrogen retention measurements and MD simulations in lithium and lithium oxide films showed that they retain similar amounts of hydrogen [3]. These findings support the possibility that low hydrogen recycling can be achieved if lithium oxide is formed under fusion reactor conditions. Preliminary sputtering experiments showed that after irradiating 8 monolayers of lithium oxide with 500 eV/D^+ ions, the oxide layer remained chemically unaltered after 6 min (7×10^{19} D_2^+/m^2). The oxide layer, however, was fully removed after one hour of D_2^+ irradiation (7×10^{20} D_2^+/m^2). Dosing carbon monoxide on a pure lithium layer led to the formation of a Li-C-O surface that demonstrated increased lithium thermal stability by 150 K. MD sputtering data of the Li-C-O compounds showed chemical sputtering reduction of more than a factor two due to the presence of oxygen [4][1].

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Breakdown of the Conventional Sheath Models Under Strong Thermionic Emission: Application to Divertors, Probes, and Other Devices

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Electron-emitting sheaths are important to the plasma-surface interactions in many systems including tokamaks. Past theories dating back to Hobbs and Wesson assumed that the floating sheath potential must be negative [1]. Under strong emission, a small potential well (virtual cathode) was predicted to form to suppress some emitted electrons. This "space-charge limited" (SCL) sheath concept was used for sheath transmission factors in scrape-off-layer fluid models (see Stangeby's book). Recent analyses predict that the heating of tungsten divertor plates can lead to sufficient thermionic emission for a SCL sheath [2].

Our study shows that strongly emitting sheaths in experiments should not be SCL. Theoretical models [1] and simulation demonstrations [2] of SCL sheaths have always omitted collisions. We demonstrate [3] that when charge-exchange collisions are present, ions inevitably get trapped in the virtual cathode and their accumulation forces a transition to the inverse regime with a positive (ion-repelling) sheath potential. In the inverse regime, the force balance of electrons and ions throughout the plasma, and their fluxes to the surface, differ from the conventional regime. It may offer benefits to divertors. The lack of ion acceleration reduces sputtering. Also in the inverse regime, a high density of cold thermionic electrons ($T_{emit} < 1eV$) dominates the quasineutral plasma near the target. Under certain conduction-limited conditions this will significantly cool the target plasma and facilitate detachment, reducing the need to inject impurities. Ongoing investigations will determine whether intentional inducement of the inverse regime would be feasible and beneficial.

Inverse sheaths will change the interpretation of emissive probe potential measurements in divertor plasmas and other devices. The measured probe floating potential was long assumed be about 1Te below the plasma potential according to SCL sheath theory. Our inverse sheath model [3] suggests that a strongly emitting probe should float slightly above plasma potential, which was actually seen in past experiments [4]. We also demonstrate that SCL sheaths break down and transition to inverse sheaths even when the surface has a large negative bias and carries current [5]. This will alter the operation of plasma applications that rely on hot cathodes such as thermionic discharges, thermionic converters, and thermionic tethers.

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Review of Lithium Application to the Fusion Plasma

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Some scientists in fusion area have been puzzled by lithium splashing difficulty induced by the electromagnetic forces produced during the MHD activities in liquid lithium divertor R & D for almost 30 years since 1974. New theory to cope with this conundrum, the state-of–the-art new rationales, so called, ejection-free theoretical physics and relevant numerical simulation of an innovative compensating perpendicular magnetic field in divertor region are first time performed in this article. Our researches show splashing can be suppressed by adopting tiny discrete and electrically insulating capillary porous array system (CPAS) as the divertor target, because the conductivity among the capillary cells has been cut off by utilizing a kind of special ceramic composite material which is made of non-conducting ceramic, being able to withstand high temperature, and unbreakable composite, therefore, no any inductive current in liquid lithium surface can be produced except eddy current. Particularly, a compensating perpendicular field method is first time proposed by authors with synchronously solving Grad-Shafranov equation to meet the boundary requirement of perpendicular magnetic field on the lithium surface. The second option is to utilize multi-mesh layers of stainless steel immerged in liquid lithium to elevate the surface binding energy and a new original formula is first time derived. The facility for demonstrating this theory is under construction. Our highlights to solve this cutting-edge project are addressed in detail. An innovative fusion reactor concept with non-tritium breeding, non-shielding, non-auxiliary heating, non-tritium fueling might be potentially realized induced by internal lithium reactions with hot core plasma species, if tiny lithium grains are injected into reactor core plasma, or by way of backstream of evaporated lithium of first wall lithium surface, hence, the time schedule of fusion energy to reach the commercial milestone would be greatly shortened. However, a potential disadvantage has been discussed about utilizing solid lithium silicate grain pebble bed in blanket to breed tritium by considering tritium purge gas blockage resulted from the broken powder of lithium silicate grains.

Retarded recrystallization of helium-exposed tungsten

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Recrystallization is a general phenomenon occurring during high temperature annealing of metals and typically linked with a decrease in hardness and strength, and therefore to a deterioration of the thermal shock resistance. This may lead to the premature development of large scale cracks on the W monoblocks constituting the ITER divertor vertical targets in the case of excursions beyond the nominal peak heat flux handling capability of the component. Recently a possible retarding effect of plasma exposure was observed during joint high heat loading experiments [1]. In view of ITER operations, where recrystallization poses serious issues, it is important to understand the necessary conditions for retarded recrystallization to occur and identify mechanisms at play. A combined experimental and modeling approach is used here to study the effect of helium (He) plasma exposure on the recrystallization kinetics of tungsten. As trapping and clustering of He is strong in tungsten at high temperature, a prominent effect of He is expected and used in this study.

Polished tungsten samples (dominant by grain orientations of <001> and <111>) were exposed to He plasma with a flux of 1.5×10^{22} $m^{-2}s^{-1}$ and a fluence of 1.4×10^{26} m^{-2} in the linear plasma device STEP at Beihang University. The He incident ion energy was about 60 eV and bulk temperature was ~550 K. Subsequent isothermal annealing of exposed and reference samples was performed in a vacuum furnace for one hour at temperatures ranging from 1273 K to 1673 K. It is found that the Vickers hardness deceased as annealing temperature increased in all samples. While a higher Vickers hardness was found in He-exposed samples compared to non-exposed ones at temperatures from 1423 K to 1573 K. Recrystallization fraction was determined by electron backscattered diffraction. At temperatures from 1423 K to 1573 K and a depth up to 5 μm, a strong reduction in the recrystallization fraction was observed in He-exposed samples. The strongest reduction of about 50% in recrystallized fraction was found at annealing temperature of 1473 K, and the microstructure of the non-recrystallized region remained quite similar before and after annealing. Furthermore, the retarding effect of grain growth following recrystallization was even more prominent. At 1673 K, though a full recrystallization was found in both He- and non-exposed samples, the average grain size of He-exposed sample was only ~25 μm, much less than ~60 μm of the non-exposed one. Molecular dynamics simulations show that He clusters or bubbles play a critical role on the dislocation mobility and shear-coupled grain boundary migration, and thus on the recrystallization kinetics during annealing.

This study clearly demonstrates that He irradiation leads to modifications in the kinetics of recrystallization and grain growth in tungsten at temperatures from 1423 K to 1673 K, in the range of temperatures expected in the strike-point region of the ITER divertor. The implications of plasma-induced retarded recrystallization on the operational budget [2] of the ITER divertor will be discussed.

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Modeling of Aluminum Sputtering and Ionization in the DIII-D Divertor Including Magnetic Pre-Sheath Effects

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Recent analysis of Al (a proxy for Be) erosion in different attached L-mode divertor plasmas in the DIII-D tokamak has found that asymmetries in the Al I photo-emission plume and Al redeposition patterns are consistent with an asymmetry in the angular sputtering distributions expected with near grazing incidence angle ions. We have calculated the full incident ion angle and energy distributions for main and impurity ions using particle orbit tracking within the potential gradient of the magnetic pre-sheath (MPS), also known as the Chodura sheath [1]. Angular resolved sputtering distributions calculated using the SDTRIM.SP code were coupled with a Monte-Carlo model of electron-impact ionization and photo-emission of sputtered neutrals in the MPS electron density gradient to create synthetic diagnostic images of the eroding surface. Model predictions closely match the in-situ observed photo-emission plumes and are consistent with the microscopic and macroscopic distributions of material redeposition found on the samples.

The majority of ions are predicted to strike the horizontal DiMES surface at <15 degrees with respect to the surface plane, in a direction 30-60 degrees away from the toroidal field (when projected on the horizontal surface plane), after drifting 1-3cm in the E×B direction. Models of the sheath for magnetized plasmas predict a disappearance of the Debye sheath (DS) at surfaces where the angle between the magnetic field and surface plane is small (on the order of a few degrees) [2, 3]. Without the DS the entire ambipolar sheath drop, as well as the associated electron density drop, occurs across the MPS, which has a thickness on the order of 2-3x the main ion gyro radius. The weak near-surface electric field causes less deflection of the incident ions toward the surface, which leads to increased sputtering yield and energy in the forward-scattering direction, thus increasing the penetration of sputtered material into the plasma. Modeling also shows that the relatively thick region of the MPS increases the E×B drift distances which can govern the direction of material migration and create a significant near surface poloidal incident ion flow.

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Studies of impurity erosion and deposition on rough surfaces with 3D SURO modelling

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Experimental results in TEXTOR, JET and ASDEX Upgrade revealed a strongly inhomogeneous erosion-deposition distribution with large erosion on protruding parts of rough surface and smaller erosion or even deposition in recessions. A Monte-Carlo code called SURO [1-3] has been developed to study the influence of surface roughness on the impurity deposition characteristics in fusion devices. The properties of background plasma and impurity near the divertor target are studied by SDPIC modelling, which are used as the input data for SURO code. The SURO code uses the test particle approach to describe the bombardment of background plasma and the deposition of impurity particles on the three-dimensional (3D) surface topography. The dynamic change of surface topography as well as surface concentrations of different species due to erosion and deposition are taken into account in SURO, which has a very good flexibility for treating the process of material mixing.

The simulation of the deposition of carbon impurity on rough wall surfaces has been conducted with the rough surface code SURO. The temporal evolution of the net eroded substrate and deposited impurity areal densities has been studied under the low and high impurity fluxes. For the low impurity flux, the carbon and tungsten substrates maintain to be eroded during the exposure. While for the high impurity flux, the carbon and tungsten substrates can be protected by the heavily deposited impurity from erosion by the background plasma and impurity after the exposure of 1000 s. The areal densities of the deposited impurity and eroded substrate on rough wall surfaces are investigated with different carbon fluxes.

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Will tungsten fuzz form in ITER?

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It is now well-known that under exposure to a helium (He) plasma at elevated temperatures, a tungsten (W) surface is modified by the growth of a fibreform nano-structure referred to as "fuzz". Formation of W fuzz occurs when a specific set of conditions are met such as surface temperature (900 K<T_{surf}<2000 K), ion energy (E_{ion}>20 eV) and fluence (>10^{24}m^{-2}). Over the years, concerns have been raised that, should fuzz form on the W divertor targets in ITER, it might have adverse consequences for material lifetime and plasma contamination. These concerns include the risks of exfoliation of this fragile nanostructure and the possible increased probability of unipolar arcing, both of which could lead to enhanced dust formation and W release to the plasma. The key open question remains, however, of whether or not fuzz will actually form in ITER.

To identify areas where fuzz formation conditions could be satisfied, the ion flux and energy, and surface temperature profiles along the high heat flux regions of the divertor targets are calculated using a range of SOLPS simulations covering a range of plasma scenarios, from pure He discharges during non-active operations to high power D-T discharges,. Since fuzz formation is hindered by beryllium (Be) deposition, the output from WALLDYN simulations is used to identify Be-free regions. To include the effects of Edge Localized Modes (ELMs), the equilibrium growth model proposed in [1] is extended to take into account ELM-induced erosion and the effect of the transient temperature excursion. In addition, both the decrease of fuzz erosion rate and thermal conductivity with increasing thickness are treated consistently.

Fuzz formation appears possible at the outer target over a poloidal extent of 4-9 cm, while the inner divertor target appears to be deposition-dominated. Assuming that the peak temperature during ELMs does not exceed the value at which annealing is observed (~2200 K), it is found that the ELM-induced transient heating only leads to a 15% increase in the fuzz thickness. For a prompt re-deposition fraction of ~99.9%, as in [2], ELM-induced erosion appears too weak to play a role. The main finding is that as fuzz grows thicker, its thermal conductivity decreases and peak temperature attained during an ELM increases. Given the competition between growth and annealing rates, a critical thickness exists above which fuzz annealing dominates for a given ELM energy. For T_{surf}=1200 K, a maximum thickness of ~2 μm is determined for E_{ELM}=0.1MJ.m^{-2}, while no fuzz formation appears possible for E_{ELM}>0.5MJ.m^{-2}. The higher the base temperature, the lower the ELM energy density beyond which no fuzz formation is possible.

The existence of a critical fuzz thickness effectively decreases fears of macroscopic surface erosion and arcing, while the very fast annealing of the fuzz at elevated temperatures relaxes the concern of enhanced material losses during arcing.

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Physics of toroidal gap heat loading on castellated plasma-facing components

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For power handling reasons, the water cooled, plasma-facing components in the ITER tungsten divertor will be castellated, comprising tens of thousands of individual monoblocks (MB). Gaps in both the toroidal (TG) and poloidal (PG) directions introduce leading edges onto which plasma heat flux can be focused, even if misalignments between neighbouring MBs are eliminated. Depending on the magnetic configuration, regions which would be shadowed in a purely optical picture can be accessed by ions with finite Larmor radius, a problem which can be particularly acute during Edge Localized Modes (ELM) when energetic ions from the pedestal region impact the target during the transient heat pulse. Heat loads on the ITER MBs are being assessed using 3D ion orbit calculations [1], guiding efforts to find a shaping solution which avoids gap edge over-heating. Toroidal bevelling will hide leading edges on each side of PGs from inter-ELM heat loads (at the expense of higher perpendicular loads on the top surface), but a solution for hiding all the long TG edges on both inner and outer targets has not yet been found. The ion orbit simulations predict that even mitigated ELMs can melt TG edges in the strike point regions, posing a potential MB lifetime risk.

Although the consequences of TG loading may be significant for ITER, the phenomenon had never actually been seen in a tokamak until a series of dedicated experiments, reported here, was performed on the COMPASS tokamak. The low magnetic field in COMPASS makes the ratio of the ion Larmor radius to the gap width relevant to ITER ELMing conditions. Using special instrumented central column graphite tiles bearing poloidal and toroidal gaps, viewed with a high spatial resolution (0.5 mm/pixel) infra-red camera diagnostic, clear evidence for TG heating has been obtained for the first time. By changing the direction of the poloidal and toroidal magnetic fields, the four possible configurations of field line interactions with TGs have been examined. The resulting heat load distributions have been compared with heat flux profiles derived from both the optical approximation and simulations using the 2D particle-in-cell code SPICE, including self-consistent electric fields.

In two of the field configurations, plasma flux is observed to impinge on the unshadowed TG side, as expected optically, but in the other two, heat is deposited on both gap sides, and therefore in the magnetically shadowed regions, in excellent agreement with SPICE code results. This same code has been used to benchmark the 3D ion orbit calculations for ITER [2], providing confidence both in the existence of the TG loading phenomenon and in the validity of the simpler ion orbit approach. The experiment-code comparison has also emphasized the role played by local non-ambipolarity in determining the distribution of heat loading. In a separate experiment, time-varying bias voltage waveforms have been applied to a specially designed electrically insulated TG to select between electron-dominated and ion-dominated regimes. In the case of two-sided deposition and depending on bias voltage, the heat load is observed to switch sides, following theoretical expectations. These unique experiments demonstrate not only that TG loading does occur, but that the physics of this phenomenon is understood and must be accounted for in the ITER divertor shaping design.

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Synthesis of porous and nano W, W-O-N, WN_x and W-O coatings for plasma surface interaction studies

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In the present work the results of the synthesis of fusion relevant tungsten (W) coatings using pulsed laser deposition (PLD) and their exposure to deuterium plasma flux of GyM [1] are presented. PLD has the peculiar feature of tailoring the crystalline size, the stoichiometry, the gas inclusions and the morphology of the growing films. In this way, it is possible to produce W coatings that simulate different material modifications, which may occur at the first wall of W-based tokamaks.
Due to the interaction with the plasma, W plasma facing components (PFCs) are subjected to several modifications in terms of composition, crystalline structure and morphology. These changes can deeply affect the features of the first wall overall, e.g. increasing D retention. Moreover, the materials sputtered from the PFCs because of the plasma bombardment can migrate and deposit with properties very different from the ones of the pristine W tiles.
We deposited different kinds of nanostructured W, W-O, WNx, W-O-N coatings by PLD with the aim of mimicking redeposited W on tokamak first walls. The film morphology was changed from compact to porous, regardless the film composition. A proper choice of the process parameters and annealing procedure after deposition allow to control the crystalline grain size in the range 2 – 300 nm [2]. Considering the elemental composition of the films, O amount was varied between 12% and 75%, while N ranged between 9% and 35%. Depending on the deposition conditions, it was possible to deposit W with gas inclusions (O or N) or induce the formation of tungsten compounds (oxides, nitrides) whose crystallinity is related to the oxygen/nitrogen content.
To study the possible modifications of redeposited W coatings, different compact W films with different crystallinity and porous W films were exposed to GyM. Plasma flux, and fluence were 4×10^{20} ions/m s and 6×10^{24} ions/m respectively. The energy of the impinging ions was ranged between few eV to 300 eV. Post exposure morphology was assessed by scanning electron microscope. Nano lamellae [3] were formed after exposure on compact films depending on the energy of deuterium particles while no structuring was present on porous coatings. An evaluation of the sputtering yield of the different W films was also carried out by profilometry. As a preliminary result, the sputtering threshold of porous W seems to be lower than that of polycrystalline W.

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In-situ measurement of the spectral reflectance of mirror-like metallic surfaces during plasma exposition

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The capability to measure the optical emission of fast neutral atoms created due to plasma-surface interaction gives access to physical properties such as: energy and angular distribution of ions reflected at the metallic surface as atoms, optical properties of metallic surfaces such as spectral reflectance and polarization, etc….

The experiments presented are performed in a linear magnetized plasma at the PSI-2 in Jülich [1,2]. We used H-Ar mixed plasmas, with a composition of 1:1 [3]. The plasma parameters are the following: normal gas pressure p ~ 0.01 Pa, electron density $n_e \sim 10^{11}$ cm^{-3}, electron temperature $T_e \sim 6$ eV and the ion temperature $T_i \sim 2$ eV We measure the Balmer line H_α using optical emission spectroscopy (OES) and a high-resolution echelle spectrometer with a dispersion of approximately 1 pm. In a pure hydrogen plasma, no Doppler-shifted emission is detected above the signal-to-noise ratio [3,4]. But if a H-Ar mixed plasma is used, Doppler-shifted emission of the fast non-maxwellian H/D atoms, backscattered from the mirror surface, are observed and can be analyzed. Before the plasma exposition the target surfaces are polished thus they have mirror like properties. Furthermore, the mirror surface can be negatively biased up to -220 eV and thus the energy of the incoming ions and the backscattered atoms can be varied.

We present a method to measure in-situ the spectral reflectance of different metallic mirrors (Mo, W, Rh and Al) during the plasma operation. The spectral reflectance is achieved by measuring the Doppler-shifted emission of the fast atoms from the Balmer Line H_α at two different line-of-sights (LOS) namely 35° and 90° [5]. This gives also access to the temporal evolution due to surface modifications such as roughness variation by plasma impact. The measured spectral reflectance is also compared to theoretical values from the literature to prove the possibilities of the presented diagnostics [6]. Moreover, the polarization of the reflected light is measured for the different target. Therefore, the LOS is changed near to the Brewster angle of the used mirror materials of around 78°.

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The influence of nitrogen seeding on the beryllium erosion in JET

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ITER is expected to use impurity seeding to reduce the power loads on the beryllium (Be) first wall and the tungsten (W) divertor in order to protect these components and extend their lifetime. One possible seeding impurity is nitrogen (N_2) which is expected to be a good radiator at the predicted plasma temperature and density ranges in the divertor. Unfortunately, the chemical reactivity of N_2 can lead to the formation of ammonia [1] or chemical stable Be- and W nitrogenous compounds [2]. Furthermore, the introduction of N_2 can, like all seeding gases with atomic masses larger than the main plasma constituents, lead to increased physical sputtering of the plasma facing components. Therefore, in order to understand, model, and predict the interaction of nitrogen with the materials and conditions relevant for fusion devices, extensive experiments on laboratory (e.g. [3]) or tokamak scale (e.g. [1]) were performed. The study of the chemical and physical interaction of Be and N_2 is thereby of special importance as beryllium nitride does not decompose up to its melting temperature and Be as material with small atomic mass is strongly susceptible to physical sputtering. Interestingly, spectroscopic observation of N_2 seeded deuterium plasma in the linear plasma device PISCES-B showed that the introduction of nitrogen reversibly reduced the atomic BeI and BeD molecular emission in front of the sample. The reduction in optical emission has been attributed to a reduced erosion of the Be and was confirmed via sample mass-loss measurements [4]. To test if these findings are transferable to fusion devices, a series of similar N_2 seeded limiter discharges (B_t = 2.6 T, I_p = 1.94 MA, $n_{e,core} \sim$ 7E19 m^{-3}, $T_e \sim$ 1.8 keV) in JET, similar to studies in unseeded discharges [5] was performed. The quick repetition of the discharges allowed to study the influence of surface temperatures (400…570 K) on the interaction of the Be with the N_2 seeded plasma as each discharge raised the temperature of the limiter. Two additional neon seeded discharges at the beginning and end of the series allowed to directly compare the effects of edge plasma cooling and enhanced physical sputtering due to a chemical inert impurity with additional effects due to the chemical reactivity of the N_2. In this contribution we will present the results of the optical emission spectroscopy which show an reduction of BeI, BeII and molecular BeD measured intensities stronger than expected from the raise of the surface temperature alone [5] and discuss that this reduction is likely due to a suppression of chemical erosion of the beryllium.

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Effects of surface temperature on the chemical sputtering of boronized and oxidized carbon surface irradiated by deuterium

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The conditioning of plasma facing carbon components by lithium and boron is an important part of this research effort and it has led to the improved performance in fusion reactor experiments. In NSTX-U wall conditioning with boron was used to provide fuel density control and impurity reduction [1].

The preparation of an amorphous, boronized and oxidized carbon surface (BCO) is crucial in our molecular dynamics calculations of the effects of the variation of temperature in the 300K-1000K range on the retention of deuterium and sputtering by the BCOD surface. We define a BCO surface as a computer cell of 400 atoms with an initial atomic distribution of 20 % of O and B, and 60 % of C [2], with periodic surface boundary conditions. The target surface is energy optimized and thermalized to a required target temperature by using Langevin thermostat and is bombard by 15,000 D impacts at 10 for each target temperature. The resulted sputtering yields in Fig. 1 were obtained by applying the adequate statistics. Retention of D has decreased by more than 10% by increasing temperature for 300K to 1000K.

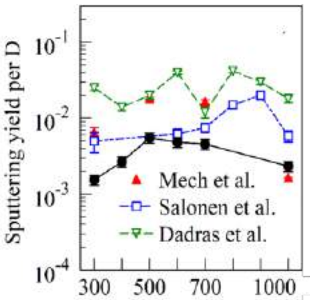

Figure 1. Chemical sputtering yield as function of the BCOD temperature for impact deuterium energy of 10 eV. The comparison is done with experimental results of a-C:D surfaces of Mech et al [3] and theoretical results of Salonen et al. [4] and Dadras et al [5].

As seen in Fig. 1, the suppression of sputtering by boron, seen previously at 300K [6], is also present at high temperatures. The maximum of the sputtering yield at about 500K is due to hydrocarbons and is explained by the complex chemistry in the BCOD surface. Measurements of the sputtering yield of the BCOD system is carried out using controlled irradiations with D^+ combined with Quartz Crystal Microbalance (QCMb) and Residual Gas Analysis (RGA) measurements. The comparison of the simulations and the measurements will be provided.

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Statistical modeling of surface morphology for multi-scale simulations of plasma-surface interactions

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Surface roughness greatly affects the erosion of plasma-facing materials. Material erosion of plasma-facing components (PFCs) in fusion devices is responsible for the release of impurities into the plasma and will have a significant effect on plasma performance. The complete effect of surface morphology on plasma-surface interactions is still yet to be fully understood. However, developing surface roughness models for multi-scale simulations of plasma-material interactions can capture both the effect of surface roughness on plasma-facing materials and, simultaneously, the effect of surface roughness on material emission into the plasma. Such models will be necessary to understand and describe how the changing surface morphology of PFCs will affect plasma device operation.

In this work we will present Fractal TRIDYN (F-TRIDYN), a Monte Carlo, Binary Collision Approximation (BCA) code based on the classic code TRIDYN and developed for integration into multi-scale models that includes two descriptions of surface roughness: an explicit, fractal surface model, and a new, implicit, statistical surface model.

Fractal surface models provide an explicit description of surface morphology by representing the surface as a polygonal fractal; however, a direct measurement of the fractal dimension that characterizes such a surface requires sophisticated molecular adsorption techniques that are not practically applicable to all material surfaces. On the other hand, statistical models of surface morphology are more computationally efficient than explicit models and are directly comparable to experiments via standard measurements of surface roughness (e.g., AFM, SEM, etc.). In the present study we show how complex surfaces can be modeled using a Kernel Density Estimation technique to represent an arbitrary distribution of surface heights. This strategy allows for fast simulations of complex surfaces in ion-solid interaction codes.

Sputtering yields of rough, fusion-relevant materials have been modeled using F-TRIDYN with both explicit, fractal surfaces and implicit, statistical surfaces. The roughness measured from AFM of irradiated tungsten surfaces has been used as an input to F-TRIDYN. We report a direct comparison of the two methods, showing agreement on both sputtering yields and sputtered particle distributions. The implicit method produces results comparable to the explicit approach, but at a fraction of the computational cost. F-TRIDYN has been coupled to both plasma and material codes such as Xolotl, GITR, and hPIC for the dynamic treatment of plasma-surface interaction processes.

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High-Order Harmonic Generation from Relativistic Laser-Plasma Interaction at Solid-Density Surfaces

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A high-intensity laser incident on vacuum-solid interface will create a dense expanding plasma whose behavior is governed by both the surface properties and the laser field strength. At relativistic laser intensities (more than 10^{18} W/cm^2 for 800 nm wavelength light), the interaction between the laser and plasma electric and magnetic fields is highly non-linear, accelerating relativistic electrons and producing coherent high-intensity extreme ultraviolet and x-ray radiation [1-5]. We consider in detail the role of the laser field sub-cycle structure in the harmonic generation process; the exact shape of the electric field substantially affects the interaction [6-8]. Sub-cycle control of the laser-plasma interaction is shown to allow a high-efficiency source of bright extreme ultraviolet and x-ray radiation.

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2D and 3D particle-in-cell simulations of electron nano-bunching in dense plasmas

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High harmonic generation from laser plasma interactions has been studied extensively in recent years as a possible route to attosecond and even sub attosecond pulse generation. It has been observed that when a laser interacts with an overdense plasma, a very thin, extremely compressed layer of electrons bunch together. These electron bunches can reach densities thousands of times greater than the original solid's density and with a width of just a few nanometers. These nanobunches of electrons have been shown to emit attosecond pulses many times greater in intensity than the incident pulse. Primarily this regime of high harmonic generation has been studied using 1D Particle in Cell (PIC) simulations. It is believed that these 1D simulations lead to an over prediction of the electron density. Therefore, in this work we consider the corresponding 2D and 3D PIC simulations of high harmonic generation in this nanobunching region. These simulations will provide a more realistic prediction of these electron nanobunches as well as the harmonics they generate.

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Abstract withdrawn

Uptake of Low-energy Neutral Deuterium Species in Sputter-deposited Tungsten Films due to Plasma Loading

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Hydrogen isotopes retention in tungsten (W) as plasma-facing materials in the divertor region of ITER and the foreseen first-wall material in next-generation fusion devices presents great concerns on cost and safety. In the divertor region, in addition to the ionic species, intensive fluxes of low-energy neutral species are expected to contribute to fuel retention, in particular for ITER divertor at detached conditions. At the first wall of the main chamber particle fluxes will be dominated by neutral species with a large fraction of low-energy species. However, no data can be found on the ratio between the contributions to the final hydrogen isotopes retention from low-energy and from high-energy particles during plasma loading.

The uptake of neutral species in W during deuterium (D) plasma exposure was studied. The special feature of the results presented here is that half of the samples were shielded from direct plasma exposure by a cover that prevents ions from a ECR plasma source impacting onto the sample surfaces. The used W materials were magnetron-sputtered W films with different thicknesses (60 and 400 nm) deposited on single crystalline silicon substrates, which can be used as model system for re-deposited W layers. Different plasma-exposure conditions, such as sample temperature, flux and energy of the ions reaching the sample holder, were performed and compared for identical samples. After D loading the retained D amounts were measured by ^{3}He (D, p) α nuclear reaction analysis. D retention and surface morphology for the plasma-shielded samples were compared with that from samples directly exposed to the same D plasma.

Results show that the retained D amount in all samples directly exposed to plasma are laterally homogeneous while the plasma-shielded W surfaces exhibit a clear decrease of the retained D amounts which increasing distance from the entrance slit for the neutral atomic D species. This decreasing profile is attributed to a decrease of the effective D atom flux reaching the respective surface area. The following preliminary results have been found:

1) In the investigated fluence range (up to 6×10^{24} D/m^2) D retention increases in both plasma-shielded and exposed samples and no indication of saturation is found. 2) The D amount depends strongly on temperature. D retention decreases with increasing temperature and the decrease is stronger for the plasma-shielded surfaces. 3) For the directly exposed samples the D amount increases proportional to the W film thickness which can be interpreted as a constant D concentration throughout the W film. On the contrary, the total D amount in the plasma-shielded samples is independent of the film thickness indicating that the D-containing layer is thinner than 60 nm. 4) Surprisingly for each investigated loading condition the averaged retained D amount in the shielded samples is only a factor of 3-5 lower than in the plasma-exposed samples and a relatively strong influence of ion energy is found for both shielded and exposed surfaces. 5) Most of the directly plasma-exposed samples show many regularly-round blisters with comparable size.

Plasma Mirror Reflectivity Performance

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Plasma mirrors can provide a high reflectivity surface for intense laser pulses. To achieve this a dense plasma is generated from the laser pulse's interaction with the surface of a solid target. Reflectivity can be minimized for low intensity pulses and at high intensity plasma mirrors can deliver reflectivities of above 70 percent [1]. Plasma mirrors are thus used to decrease the magnitude of pre-pulses and increase the laser contrast to the high levels desired for many laser-matter interaction applications [1]. Contrast improvements of up to 10^4 have been reported for single plasma mirrors [2]. Using multiple plasma mirrors in sequence can further better contrast [3]. Through varying the incident laser pulse parameters there can be a large variance in plasma mirror effectiveness. We will present our investigation into the performance of plasma mirror reflectivity. Particularly, we will discuss how reflectivity behaves at high intensities in comparison to previously published work.[1]

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Influence of Thermal Shocks on the He Induced Surface Structure on Tungsten

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W has been used in current devices and will be used in future reactors due to the good thermal and mechanical properties and moderate activation in neutron environment. At divertor in tokamak, W will be faces on constant He ion irradiation and transient thermal shocks likes ELMs (Edge Localized Modes: ~ 1 GW/m2, ~ 500 µs). It was reported that temperature have an impact on the He induced surface modifications even in relatively low temperature (less than1300 K)[1]. Transient temperature increase and decrease due to thermal shocks settles into the steady temperature in less than several tens ms. However, behavior of He induced structures such as bubble, hole and undulating structures in such short time have not been fully understood. In this study, the response of the He induced structures against thermal shocks were studied.

The samples were high-purity (99.995> %) W manufactured by Toho Kinzoku Co. Ltd. with a square shape of 20 × 20 × t5 mm. The samples were mechanically polished and then annealed at 1773 K under vacuum conditions for 2 h. He plasma exposure were performed using linear plasma device PSI-2 [2]. Irradiation temperature were from 500 to 1100 K. The typical He flux and fluence were 1 x 10^{22} /m^2/s and 3 x 10^{25} /m^2, respectively. The samples were biased -100V during exposures. The typical incident energy of He ion were 78 eV. Samples with or without He irradiation were applied thermal shocks in high heat flux test facility with electron gun named Active cooling teststand 2(ACT2) [3]. The electron beam had Gaussian-like profile with the FWHM of about 6.5 mm. The base temperature of the samples. peak heat flux , cycle numbers and the duration of the thermal shocks were about 300 K, 500 MW/m^2, 10 cysles and 500 µs, respectively.

From the SEM observations (before thermal shock experiments), undulating surface structures in samples irradiated at 500 K and dense holes in samples irradiated at 1100 K were observed. From TEM observation, both samples showed dense He bubbles with the size upto 20 nm. After the thermal shocks, dense holes in samples irradiated by He at 1100 K were modified into rugged surface with sparse holes. Undulating structures in samples irradiated by He in 500 K were modified into almost flat surface. The maximum temperature of the surface during thermal shock was estimated below the recrystallization temperature (less than 950 K). In addition, the surface temperature decreased below 500 K at 1.5 ms after of the thermal shock. He induced undulating structure and holes were flattened and closed in the short time.

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Activities at IAEA on data for plasma-material interaction in fusion devices

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The Atomic and Molecular (A+M) Data Unit of the International Atomic Energy Agency (IAEA) [1] reviews progress and achievements in the generation of Atomic, Molecular and Plasma-Surface Interaction (AM/PSI) data for fusion research programs, and coordinates international cooperation in measurement, compilation and evaluation of AM/PSI data for fusion. The Unit maintains a numerical database, ALADDIN [3], for atomic and molecular collisional data and plasma-surface interaction data; a bibliographical database, AMBDAS; and a wiki-style knowledge base, among other data collections.

The Unit also organizes various technical meetings and coordinated research projects (CRPs) [2], which bring together scientists worldwide to collaborate on a focused research topic. Currently two CRPs are in progress: a CRP on “Plasma-wall Interaction with Reduced-activation Steel Surfaces in Fusion Devices” and a CRP on “Data for Atomic Processes of Neutral Beams in Fusion Plasma”. A new CRP on “Atomic Data for Vapour Shielding in Fusion Devices” is in the early stages of planning.

Two further CRPs have recently concluded and will be described: “Data for Erosion and Tritium Retention in Beryllium Plasma-Facing Materials” was completed in 2016; “Plasma-Wall Interaction with Irradiated Tungsten and Tungsten Alloys in Fusion Devices” was a large project involving 19 research groups and included a Thermal Desorption Spectroscopy (TDS) round-robin exercise to test and validate experimental procedures for the measurement of hydrogen retention.

Continued support of quality assessment and uncertainty quantification (UQ) of plasma-material interaction data has been achieved through technical meetings over the last 2 years.

A progress report on the active and recently completed CRPs will be presented at the meeting, together with an overview of related database development and meetings.

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Optical Property of Nanostructured Tungsten for Plasma Emission Light

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Tungsten (W) has high durability for erosion by plasma irradiation and low tritium retention property. Thus, it becomes very popular as one candidate of plasma facing material (PFM), *i.e.*, first wall and divertor tiles. However a several problems to use W as the PFM remain unresolved. One of them is an influence of reflection of light on optical diagnostics, which is called stray light problem [1]. This problem is caused by the fact that W has much higher reflectivity than carbon based materials. Some of the authors, K.S, M.Y. and N.O., proposed the usage of nanostructured W to conquer the stray light problem. The nanostructured W is generated by the irradiation of helium particles under some condition [2]; it can be used for light absorber from ultra-violet and near infrared wavelength range [3, 4]. In Ref. [1], K.S. *et al.* also found that optical reflectance of the nanostructured W depends on the morphology of the surface. However the cause of the dependence is still unknown.

In this paper, we reveal the cause of the high absorptivity of the nanostructured W by finite-difference time-domain (FDTD) simulation with Drude-Lorentz model [5]. The FDTD simulation is one of the popular methods to analyse the electromagnetic-wave transmission, by which we have investigated surface plasmons on the surface of Au-nanorod [5].

In our FDTD simulation, the electromagnetic plane-wave is irradiated to two types of the nanostructured W targets; the first one is modelled as the fractal structure, *i.e.*, Menger sponge [6, 7]; another one is the three dimensional structure which is observed by transmission electron microscope (TEM). Waiting for a while after irradiation of the plane wave, it becomes the steady state. It is found that the electromagnetic field is localized in the target. This fact shows that the electromagnetic field is absorbed in the inside of the target. This is the reason why the nanostructured W has high absorptivity.

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Abstract Withdrawn

Development of tritium permeation barriers for future fusion devices

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In order to reduce fuel loss and due to safety issues, tritium accumulation into reactor walls and permeation through walls have to be prevented. Therefore, the development of tritium permeation barriers (TPB) is crucial for a safe reactor operation.

Deuterium permeation studies were performed on tungsten layers and steels, which are intended to be used as wall and structural materials in ITER and future fusion devices, namely 316L and Eurofer97. By comparing the deuterium permeation flux through these different samples, the requirement and important characteristics of TPB can be specified.

A thin coating of Al_2O_3 is a promising candidate for TPB, due to the high thermal stability and a reasonable hydrogen permeation reduction factor. The hydrogen permeation reduction factors are in the range of one to five orders of magnitude [1-3], as determined in deuterium permeation experiments by different groups. The reason for the large variation between the permeation reduction factors are mainly differences in purity, microstructure and texture of the oxide layers.

Different Y_2O_3 coatings are deposited on Eurofer97 substrates and studied in order to investigate the influence of the structure of an oxide layer on the deuterium permeation behavior [4]. Y_2O_3 is chosen due to the more favorable neutron activation behavior of Y compared to Al. The Y_2O_3 layers with thicknesses of 0.1µm to 1µm are deposited on both substrate sides by RF magnetron sputter deposition. Layers with three different magnetron process modes are prepared. The stoichiometry of these samples is the same, but the microstructure is different. After annealing the cubic crystal structure of all three process modes is verified by X-ray diffraction and the different microstructures are investigated by scanning electron microscopy and transmission electron microscopy. The permeation reduction factors of the three process modes are determined in gas-driven deuterium permeation experiments. Corresponding to each of the three process modes and microstructures, three different reduction factors are identified in the range of 30 to 4000. Thus, by improving the microstructure of the TPB, the permeation reduction behavior is strongly enhanced.

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Development of BCA-MD-KMC multi-hybrid simulation method for fuzzy nanostructure formation

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Helium plasma irradiation onto tungsten surfaces generates fuzzy nanostructures[1,2]. The fuzzy nanostructure causes the problems on divertor plates such as the arcing to generate micro crack on the material. Although it is considered that the self-agglomeration of helium bubbles is the trigger of the fuzzy nanostructure formation, its detail formation mechanisms are not well understood. Since the formation process of the fuzzy nanostructure is composed of complicated multi-scale multi-physics processes, the significance of researches on the fuzzy nanostructure formation is not only the scientific meaning of helium-tungsten interaction and but also the representative benchmark target for the simulation methods of plasma-material interaction in fusion science. Developed simulation methods for fuzzy nanostructure formation will be used to estimate wall condition and impurity generation in environments of future reactors.

We have been focusing on hybridization in simulation methods for plasma material interaction. In our previous works[3,4], the formation process of the fuzzy nanostructure at initial phase had been represented by the MD-MC hybrid simulation. However, the MD-MC hybrid simulation ignored the injection process of helium ions. In the injection process, the displacement of tungsten atoms kicked out by helium ions has some impacts on the growth of fuzzy nanostructure. Even though the kicked out tungsten atoms are not emitted to vacuum, some of them become surface ad-atoms. In addition, the injected hydrogen ion is easily stopped in a helium bubble and then its range becomes shorter than that in a pure bulk tungsten material[5]. This fact expects that the space distribution of helium atoms in a tungsten material depends on the injection process. Thus, the injection process should be considered in the hybrid simulation.

In the present work, we developed BCA-MD-KMC multi-hybrid simulation in which the injection process of helium ion is solved by using binary collision approximation (BCA), the diffusion process of helium atoms in tungsten material solved by using kinetic Monte-Carlo (KMC), and the deformation of the tungsten material due to the pressure of helium bubbles is solved by using molecular dynamics (MD). To combine with the MD which is parallelized with domain decomposition method (DDM), we made BCA code supporting the DDM. By this BCA-MD-KMC multi-hybrid simulation, we clarify the above effects of injection process in the fuzzy structure formation.

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Effects of Finite Saturation in Porous Surfaces During Particle Bombardment

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Liquid metal plasma-facing components (PFC) have potential application as divertor and first wall surfaces in fusion reactors. One particular embodiment of a liquid metal PFC is one constrained by a porous substrate as to stabilize the surface against electromagnetic body forces or surface shears [1]. Numerous approaches have been reported in the literature for different methods of optimizing the porous layer including wire meshes, flame-sprayed surfaces, laser-texturing, and chemical-vapor infusion lattices. Given the range of options for producing these porous layers, it is necessary to consider the pathways to optimization one may take. Recent theoretical treatments of the porous material indicate optimization is possibly through use of a permeability-enhanced Hartmann number [2]. These calculations, however, do not account for the impact of finite saturation on the wicking behavior if the liquid in the porous substrate.

The simplest expression of capillary pressure, P_c, is as follows: $P_c = 2 \sum cos(\gamma) / r_p$ where $\sum$ is the surface tension of the liquid, γ is the contact angle, and r_p is the pore radius. It is known in other porous systems that the capillary pressure varies as a function of the saturation of the material from values exceeding the simple expression at near-zero saturation to zero capillary pressure once fully saturated [3]. A completely saturated surface, then, has zero wicking potential, but generates finite capillary pressure once even a small amount of fluid is removed from the pore volumes. The actual capillary pressure relative to the nominal pressure in the equation above is often determined empirically and is referred to as the Leverett J-function. One example J-function for compacted sands indicates the capillary pressure can be less than 30% of the nominal value for a range of 50-100% saturation which reduces sorptivity [3].

If a liquid metal PFC is to operate with non-zero capillary pressure, as during passive replenishment, then a saturation of less than 100% is required. However, incomplete saturation implies the possibility of exposed substrate. This exposed substrate would be subjected to plasma bombardment and its destruction through plasma bombardment would defeat one of the main features of a liquid PFC. To explore the effect of finite saturation, we conduct particle bombardment on a novel, porosity-enhanced flame-spray material in the IGNIS experimental device at the University of Illinois. The composition of a liquid-metal infused, porous substrate surface is analysed with high-resolution XPS before and after bombardment of the surface with an energetic particle beam. The rate of passive recovery of a protective layer will be examined in both hydrogen and helium environments. Implications of the recovery rate and any implied minimum degree of saturation on the design of a porous liquid-metal PFC will be discussed.

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Directional growth of large scale nanostructures on metallic co-deposition layer

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Tungsten (W) is a candidate material for plasma facing components in fusion reactors. It has been revealed that interaction with helium (He) leads to various morphology changes including nano-tendrils and fuzzy nanostructures [1]. The mechanism of the morphology changes and its impact for fusion devices have been explored. In this study, we show that large scale nanostructures were grown by the He plasma irradiation under the condition where metallic co-deposition layer was formed. The thickness of the fur-like nanostructures became 1 mm, which was far greater than the thickness of the conventional W fuzz layer (<10 μm); the mechanism and the process of the morphology changes were significantly altered from those by pure He plasma irradiation. The enhanced growth process will give a new insight for the growth mechanism of the fuzz growth by He plasma irradiation.

Experiments were conducted using He plasmas in the linear plasma device NAGDIS-II (Nagoya divertor simulator). In addition to a W sample, a tungsten wire (sputtering wire) was installed adjacent to the sample. The sample was biased to -90 V typically, while the sputtering wire was biased to ~-500 V to induce sputtering of W atoms by He ion bombardment. The sample surface was in parallel with the magnetic field line. The distance between the sample and the sputtering wire was ~2 mm at the closest part. During the irradiation experiments, W particles, mainly ions because the mean free path is shorter than the distance to the sample, precipitated on the W samples together with He ions.

In an hour of irradiation, 1 mm-thick visible tungsten fur-like structures covered a tungsten metal substrate. Scanning electron microscope observations revealed that fine fiberform nanostructures and membrane structure comprised the fur-like structures. Further transmission electron microscope analysis showed that the thickness of the membrane was 20-30 nm and the structures contain fine He bubbles similar to fuzzy nanostructures. Systematic irradiation experiments suggested that the enhanced nanostructure growth occurred under almost the same temperature and incident ion energy conditions as the fuzz growth.

The fine structures were always grown from the edge of the sample and expanded to the surface. The growth direction of the fur-like structure can be identified visibly. In the cylindrical plasma, plasma rotates azimuthally due to $\boldsymbol{E} \times \boldsymbol{B}$ drift. In the field of view from the downstream, plasma flows in downward and upward direction in the left and right halves of the cylinder, respectively. In addition, parallel flow exists from the plasma source to the downstream. The growth direction was consistent with the plasma flow. Motion of the metallic ions influenced by the plasma flow might have caused the directional growth of large scale nanostructures under metallic ion precipitation conditions.

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Plasma interaction with liquid walls and electrodes

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Lifetime issues of plasma-facing components (PFCs) in fusion devices and in MHD generator channels motivate novel material exploration [1]. Heat loads from hot background gas, ion flux to the cathode, electron flux to the anode and interaction with chemically active gas species can cause severe damage to the plasma facing walls and electrodes. Liquid PFCs in tokamak and liquid electrodes MHD generator channels [2] are a promising approach to achieve long-lasting facilities.

A multi-functional code was developed on a base of a general purpose 3D CFD code ANSYS CFX. The code developed is capable of modeling: (i) MHD flow of liquid [3] including free surface flows in liquid plasma-facing components; (ii) fast plasma flow in MHD generator using extended equations accounting for flow compressibility, non-uniform chemical composition, electrical conductivity and the Hall effect; (iii) near-electrode plasma using non-equilibrium plasma model [4,5] taking into account ion and electron diffusion, thermal diffusion and their effects on electric field and heat fluxes as well as effects of space-charge limited sheaths.

(i) Flow of liquid metal having a free boundary was modeled. Effects of coplanar magnetic field on the flow and free surface shape were studied.

(ii) Supersonic 3D flow of coal combustion products in a linear MHD channel of a rectangular cross-section was modeled. Electrical conductivity of the gas mixture was enhanced by the addition of potassium carbonate. Heat load from plasma to the electrode surfaces were calculated, damage to the electrodes was estimated.

(iii) Electric arc was simulated in 2D using self-consistent model accounting for heat transfer in the electrodes, their radiation and electron emission. Heat load and particle fluxes to the electrodes were calculated and ablation rate of the electrode material was determined.

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Abstract Withdrawn

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A multi-technique analysis of helium plasma-induced surface modification of tungsten

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In this study, we apply spectroscopic ellipsometry, helium ion microscopy (HIM), and high temperature thermal desorption spectroscopy (TDS) to assess how tungsten surfaces are modified up to depths of 50 – 100 nm by He plasmas. Each of these techniques covers phenomena at different length scales, thereby enabling one to gain insight into how defects nucleate at the atomic scale and evolve into features ~100 nm in size and beyond. To characterize He-induced nanostructure at different stages of growth, we exposed a series of ITER-grade W samples to an RF plasma ($n_{pl} = 7\times10^{10}$ cm^{-3}; $T_e = 11.5$ eV; $E_{ion} = 80$ eV; $\Gamma_{ion} = 8.5\times10^{20}$ m^{-2} s^{-1}). The exposure temperature (T_{sample} = 550 - 930 °C) and total fluence ($F = 7.9\times10^{23}$ - 3.6×10^{25} m^{-2}) were systematically varied to explore in detail boundaries where the nanostructure is known to nucleate. HIM imaging revealed initial pitting of the surface at low fluence and temperature. At increasing temperature, these features evolve into ripples that are comparable to the diameter of tungsten nano-tendrils (~60 nm) that develop with further exposure. After the network of W tendrils formed (typically after the surface roughness exceeds 100 nm), we determined the depth of the nanostructure layer via focused ion beam profiling. Thermal desorption revealed He release peaks centered at 1004 °C and 1437 °C. These temperatures are roughly consistent with the de-trapping energies associated with vacancies and small bubbles presented in Ref. [1].

We correlated the structural information provided by HIM with the surface optical properties measured by spectroscopic ellipsometry over wavelengths of 280 – 1000 nm. A systematic decrease in both the extinction coefficient and index of refraction was observed over the visible wavelengths with increasing fluence and nanostructure thickness. These findings will be used for the basis of in-situ, real-time ellipsometry measurements using a M-2000 J. A. Woollam spectrometry system; this presentation will include initial results from this work.

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Floating potential of emitting surfaces in plasmas with respect to the space potential

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Strongly emitting surfaces affect the plasma-wall interaction in many applications, for example in divertors and limiters in fusion devices, emissive probes, and thermionic cathodes. The potential difference between a floating emitting surface and the surrounding plasma strongly impacts particle and energy transport to the surface. A variety of sheath models describe this potential difference, including the space-charge-limited sheath [1], the electron sheath with high emission current [2], and the inverse sheath with ion trapping [3]. Our measurements reveal that each of these methods has its own regime of validity. We determine the charge state of an emissive filament immersed in a variety of plasmas, emphasizing variations in emitted current density and neutral particle density. Depending on the regime chosen, emitting surfaces can float positively or negatively with respect to the plasma potential.

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Transient-induced tungsten melt motion studies on ASDEX Upgrade

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A series of experiments dedicated to the study of ELM transient-induced melting of tungsten have been performed on ASDEX Upgrade aimed at the provision of high quality data of key quantities for validation of the MEMOS-3D melt motion code, being used as the principal simulation tool for the prediction of melt damage and associated plasma-facing component lifetime on ITER. The experiments complement and extend similar studies on JET. Samples were installed in special tiles mounted on the divertor manipulator, allowing exposure to specific discharges without further modification due to subsequent plasma operation and rapid retrieval of samples for post-mortem analysis. Most critically, the probe head instrumentation provides measurement of current drawn and the opportunity to electrically insulate samples so that the melt motion with and without net current flow can be studied.

In the experiments, two principal sample geometries, a sharp protruding edge and an inclined surface, have been exposed to Type I ELMing H-mode plasmas with ELM energy density comparable to those obtained during the JET melt experiments with identical sample geometries. In conjunction with the MEMOS-3D modelling [E. Thoren, PSI 2018 submitted], net currents drawn by the samples quantitatively validate the picture of strong surface thermionic emission being the main driver for current flow through the samples and the resulting $\boldsymbol{j}\times\boldsymbol{B}$ force on the melt layer. In particular, the considerably lower current density through the molten surface area of the sloped sample in comparison to the leading edge case, and the resulting strongly reduced melt displacement in the poloidal direction are powerful evidence for the assertion that the lower probability for electron escape from the surface is the main driver for the decreased current density. This is an extremely important result supporting the assumptions being made in the melt modelling of shaped ITER divertor monoblocks. The highly complex topology of the final surface melt damage found on the inclined sample is a challenge for the modelling to reproduce and a good indicator of what ITER should expect to see in the case of full surface flash melting.

In new experiments focused on the sloped sample, the current measurements have been complemented by surface temperatures obtained with an IR system directly viewing the exposed sample top surface, providing unique, ELM-resolved observation of melt motion, the poloidal velocity of which can be extracted and is found to be in the range expected for a $\boldsymbol{j}\times\boldsymbol{B}$ driving force at the levels of net current detected during the exposure. The system is prepared and awaiting machine time for exposure of new, electrically floating leading edge samples, in which the flow of replacement current compensating the loss due to thermionic emission will be prevented. The resulting melt patterns are expected to be very different to the electrically connected case and should provide the definitive test of the mechanism driving melt motion.

Retention and chemical sputtering of the simultaneously lithiated, boronized and oxidized carbon surfaces irradiated by deuterium

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Coating of boron at the carbon surfaces brought a change of surface chemistry during the plasma irradiation which considerably influences D retention rate and suppresses wall-originating impurities in plasma by a significant suppression of chemical sputtering [1]. Plasma performance on lithiated carbon surfaces in NSTX has enhanced stored energy and plasma performance and suppressed ELMs [2], while a complex chemistry controls both retention of D and sputtering of Li-C-O-D surfaces [3].

Separate lithium and boron conditionings of carbon in PFCs have been studied in our previous work based on molecular dynamics, achieving qualitative agreement with XPS measurement in the National Spherical Torus Experiment (NSTX) and laboratory experiments [4]. Both lithiumization and boronization of PFC's based on carbon have been studied in the and in NSTX and related laboratory experiments and have demonstrated enhanced oxygen concentration in the surface. The retention of D is significantly influenced by presence of oxygen, being quantitative similar for B or Li coatings of carbon [4]. Presence of boron reduces chemical sputtering of carbon, while presence of oxygen additionally reduces the erosion [5]. Similar, though significantly weaker effect is found in erosion of LiCO surfaces [3]. The exciting specificities of effects on D retention and sputtering of the Li and B coated carbon surfaces raise questions on the possibility of control and enhancement of their beneficial effects by mixing Li and B.

In the present work we use molecular dynamics and *in-situ* experimental methods to study the effects of deuterium irradiation on D-uptake as well as sputtering of simultaneously boronized, lithiated, oxidized, and deuterated carbon surfaces. We present analysis of the bonding chemistry of D for various concentrations of boron, lithium and oxygen in carbon surfaces. The significant role in retention of D is played by B and O, while Li has mainly a role of catalysing O concentration in the surface layers. The presence of both Li and B in the mixture with O increases retention of D by 6% (in comparison to the BCO) and 9% (in comparison to LiCO). However, the total erosion as well as carbon erosion are significantly suppressed by presence of B and Li, being approximately between the LiCO and BCO results.

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Multi-physics modeling of the evolution of surfaces exposed to steady-state plasmas

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Plasma-surface interactions (PSI) involve physical processes that are diverse in nature, with characteristic time and length scales that span multiple orders of magnitude (ps–s, and Å–m). Therefore, comprehensive modeling of PSI requires the use of models that can accurately target each scale and subsystem.

In the present exercise, we integrate high fidelity plasma and material models using the Integrated Plasma Simulator (IPS) framework, to capture the evolution of tungsten (W) surfaces exposed to steady-state, mixed deuterium (D) - helium (He) plasmas. In our workflow, plasma characteristics are measured experimentally –when possible– or calculated by fluid codes that model the steady-state edge plasma, such as SOLPS. The impurity migration and re-deposition code GITR uses these plasma characteristics to model transport and re-deposition of particles eroded from the surface. Erosion rates and energy / angular distribution of sputtered particles are calculated by the binary collision approximation code Fractal (F)-TRIDYN, which accounts for key parameters in sputtering, such as ion impact energy, angle and surface roughness. F-TRIDYN also characterizes ion implantation (profile, damage to material, etc.) accounting for substrate composition. Sub-surface evolution of the implanted species is determined by the cluster dynamics - rate equation code Xolotl. Consequent changes in substrate matrix, surface height and thus morphology are also monitored by Xolotl. As surface composition evolves in Xolotl, F-TRIDYN is again called to update sputtering rates and implantation profiles. Plasma and impurity transport models may also update the input when sub-surface processes drive significant change, e.g., in outgassing or geometry.

Dedicated experiments in the linear plasma device PISCES allow to benchmark our integrated model and evaluate uncertainties. Sub-surface gas evolution, surface erosion, and impurity transport and re-deposition are measured in W surfaces exposed to pure He and mixed He-D plasmas. The W samples are biased (~250V) to ensure observable erosion yields (> $O(10^{-3})$). Variations in other exposure parameters, such as flux ($0.5 - 4\cdot10^{18}$ cm^{-2}s^{-2}), fluence ($0.5 - 1\cdot10^{22}$ cm^{-2}) and He:D ratio (10:90 – 30:70, and 100% He) allow for in detail validation of the sensitivity analysis. Using experimental values for plasma parameters, absolute W mass loss and re-deposition modeled in GITR are consistent with experiments (within ~30%). Changes in surface morphology and sub-surface composition (vacancy, D and He clusters) observed in the W targets are also compared to profiles calculated by F-TRIDYN and Xolotl.

New atomic data for use in W and Mo erosion measurements

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Erosion and re-deposition measurements of high-Z Plasma Facing Components (PFCs) can be inferred spectroscopically using atomic physics coefficients known as 'ionizations per photon' (S/XB) ratios and spectroscopic measurements. W and Mo are two of the most important PFCs for magnetically confined plasma experiments. The W I 4008.9 Å line is widely used for tungsten gross erosion measurements. We present here a summary of new atomic physics data for W I, W II, Mo I and Mo II. For both Mo and W, simultaneous observation of multiple emission lines in the UV provides the potential for carefully benchmarked S/XB diagnostics as well as allowing for spectroscopic checks on T_e, n_e, and metastable fraction. Metastable fraction cannot be determined theoretically, and must be measured using multiple lines observed simultaneously. The overall conclusion for W and Mo is that new atomic data can significantly change the S/XB ratios. In addition, one requires knowledge of the metastable populations, requiring simultaneous measurements of multiple spectral lines.

The focus of this work is to generate more accurate S/XB ratios for the low charge states of W and Mo. Current data for neutral W S/XB consists of data generated using approximate methods [1, 2]. We compare new W I Dirac *R*-matrix excitation data and new ionization calculations with the existing data. While there is reasonable agreement for some transitions, there is in general large differences with the data currently used in databases. Knowledge of the metastable fraction is critical in achieving agreement with DIII-D W I spectra, with each of the strong spectral lines being driven from a corresponding single metastable. The role of metastable populations on the S/XB ratio is investigated, with the effective ionization being relatively insensitive and the line intensity being strongly dependent. Preliminary R-matrix excitation and ionization results presented for W II, comparing with existing approximate data.

New R-matrix results are also presented for Mo I [3] and Mo II, along with associated S/XB ratios. For Mo II, spectral multiplets in the UV would provide an ideal S/XB diagnostic, while also providing a useful T_e measurement. The Mo II data compares well with spectra from Alcator C-Mod and CTH, using the appropriate metastable fraction. We also compare Mo I S/XB ratios with existing measurements from PISCES-B, showing good agreement.

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Blistering and hydrogen retention in poly- and single crystals of ITER relevant materials by a joint experimental-modeling approach

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Plasma-wall interactions present a serious concern in existing fusion reactors. Surface modification of PFC (Plasma Facing Components), and hydrogen retention are some of the problems that must be resolved before achieving sustainable nuclear fusion. Beryllium (Be) has been chosen as a first wall material due to its high thermal conductivity, low neutron activation, low Z and its affinity with oxygen. However it is a highly toxic material which has to be handled with caution. As proposed by [1,2] aluminum (Al) is a possible less toxic proxy material to Be, which presents a similar chemical behavior after plasma exposure. Its studies can therefore provide useful information that can be transposed to Be. In this paper, experimental and modeling results are first shown for hydrogen retention and blister formation in Al, in poly- and single crystals. Next a numerical comparison between Al and Be retention is presented.

Aluminum samples have been exposed to an hydrogen plasma generated by an ECR (Electron Cyclotron Resonance) microwave reactor [3]. The typical hydrogen ion flux is ~ $1.7\text{x}10^{20}$ ions/m^{2}s. The fluence is kept below typically ~$3.6\text{x}10^{24}$ ions/m^2, in order to study the first steps of nucleation and growth of surface and bulk defects, *i.e.* blisters and bubbles. By comparing surface and cross-section analysis (SEM, confocal microscopy, blister density measurements) between poly- and single crystals, we investigate and quantify the role played by grains boundaries in the hydrogen retention. Moreover, the comparison of results between three different single crystal orientations (<100>, <110> and <111>) allows to determine preferential orientations able to attenuate the formation of blisters.

A macroscopic rate equations code has been used to simulate hydrogen retention in materials and bubble formation [4,5]. This code simulates the depth profile of hydrogen isotopes, the hydrogen concentration in the material and the temperature distribution in the exposed material. The code was initially developed to simulate HI retention in tungsten (W), however it has been extended for Al and Be and used with the same plasma conditions used in our experiments. Temperature Programmed Desorption (TDP) [6,7] has been used on Al poly-crystals exposed to different fluences to quantify the amount of hydrogen desorbed from the samples. Three types of traps were chosen to simulate these experimental results on Al: vacancies, dislocations and bubbles. This numerical approach has been extended to Be and some differences on HI retention with Al are discussed.

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Investigation of Deuterium Retention in Thick Lithium Oxide Films on High-Z Plasma-Facing Components

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Lithium coatings on high-Z plasma-facing components (PFCs) at the Lithium Tokomak eXperiment (LTX) have led to improved plasma performance. The initial hypothesis was that lithium retains hydrogen by forming lithium hydride and thereby enabling low recycling in LTX [1]. However, recent in-vacuo measurements indicate the presence of lithium oxide in deposited lithium coatings. Improved plasma performance continued to be observed in the presence of lithium oxide [2]. These observations raise questions related to the nature of the lithium oxide surface, whether the PFC is an amorphous mixture of lithium and lithium oxide or something more ordered like a lithium oxide layer growing on top of lithium, and whether lithium oxide itself is responsible for any retention of hydrogen from the plasma. Another observation is that the hydrogen seems to be retained on time scales larger than the shot duration and smaller than the time between shots [3]. To investigate the mechanism by which the LTX PFC coatings might be responsible for low recycling, thick lithium samples were studied to represent the results of repeated lithium evaporation on LTX PFCs over the course of an experimental campaign. The samples were characterized by high-resolution X-ray photoelectron spectroscopy (HR-XPS) and Rutherford backscattering spectrometry (RBS). To provide baseline characterizations of thick lithium layers, samples stored with water vapor and oxygen content maintained under 10 ppm were initially used. HR-XPS measurements revealed a thick lithium oxide layer under a thin overlayer of lithium carbonate and adventitious carbon < 6 nm thick. RBS analysis showed the oxygen-containing region to be $< 22\ \mu$ thick. We will discuss HR-XPS, nuclear reaction analysis (NRA), and RBS analysis of these samples exposed to a deuterium ion fluence on the order of $10^{21}\ m^{-2}$ to simulate conditions in LTX.

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Tungsten surface enrichment in EUROFER and Fe-W model systems studied by high-resolution ToF-RBS

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Low-activation steels are potential candidates as first-wall materials for recessed areas of the plasma-facing wall in a fusion reactor. While most steels are not suitable as first-wall materials due to their relatively high erosion yields, low-activation steels contain small concentrations of heavy elements, especially tungsten. It has been predicted by computer simulations using the SDTrimSP code that during erosion by energetic hydrogen atoms or ions a W-enriched surface layer can be formed by preferential sputtering, which could lower the over-all erosion rate to acceptable levels. The existence of this W-enriched surface layer has been demonstrated experimentally [1,2]. However, due to the limited depth resolution of the used methods the comparison to the simulations proved to be difficult.
Time-of-flight Rutherford backscattering (ToF-RBS) with incident heavy ions (for example Si or Cu ions) offers a 10-20 times better depth resolution than conventional RBS with incident He ions. The new Garching ToF-RBS detector is located at a scattering angle of 150° with a free flight path of 1.313 m and a time resolution of 600 ps. The experimentally achieved depth resolution at the surface of W-containing Fe samples is 2–3 nm.
Model systems consisting of Fe layers with 0.7, 1.5 and 4.2 at% W and EUROFER steel (which contains 0.33 at% W) were eroded by 200 eV D ions to a fluence of 10^{23} D/m^2 at room temperature (RT) and 900 K. W depth profiles were measured using ToF-RBS with 2.0, 5.0, and 11.5 MeV Si ions, light impurities at the surface (especially C and O) were detected using time-of-flight elastic recoil detection at the Ruđer Bošković Institute with 20.0 MeV I ions at a recoil angle of 37.5°. The data sets from the two techniques were analyzed self-consistently using MultiSIMNRA [3]. Sample surface morphologies and -roughnesses were investigated using scanning electron and atomic force microscopy, information about the chemical states was obtained by X-ray photoelectron spectroscopy.
Annealing of the model systems with 0.7 and 1.5 at% W to 900 K for 10 hours in vacuum resulted in a W-enriched surface layer with a W concentration of 2-3 at% and a thickness of about 15 nm. The additional W at the very surface originated from a near-surface layer with a thickness of about 100 nm where the W was depleted. Erosion at RT or annealing at 900 K yielded a laterally homogeneously enriched W surface layer. The observed enrichment at RT is in good agreement with SDTrimSP simulations if the light impurities at the surface are taken into account. Erosion at 970 K resulted in segregation of W.

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Mirror cleaning of Be deposits with helium and deuterium

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Metallic First Mirrors (FMs) will play a crucial role in numerous ITER optical diagnostic systems. Being the first element of the optical path which allows radiation from the plasma to cross the neutron shielding, FMs will be placed close to the plasma and, therefore, will be subject to erosion and/or deposition. Especially net deposition can degrade the reflectivity of FMs severely, compromising the reliability of the optical measurements [1]. Regarding the mirrors in net deposition conditions, plasma cleaning using radio-frequency (RF) discharges is currently being considered as the most promising in situ cleaning technique [2].
This work presents the results of plasma cleaning experiments conducted on rhodium (Rh) mirrors contaminated with beryllium (Be) based deposits, either produced in laboratory or from exposure in JET-ILW tokamak. The cleaning was performed with sputter gases having a small impact on rhodium (mirror material), i.e. helium or deuterium (D). For the latter, two sputtering processes were studied, namely physical sputtering (200 eV ion energy) and chemically assisted physical sputtering (50 eV ion energy) [3].
Specific focus was given to the low energetic D cleaning as it prevents the sputtering of the FM while the removal of Be is enhanced through the formation of BeD molecules. Regarding the mirrors deposited with 30 nm of Be in laboratory, the use of He at 200 eV or D_2 at 200 or 50 eV led to a substantial decrease of the contaminants thickness (less than 2 nm Be remaining on the surface after cleaning) and accompanied with restoration of the reflectivity in the Vis-IR range. In the UV region, the recovery of optical properties was not complete (up to 24 % lower at $\lambda = 250$ nm) but no increase of the diffuse reflectivity was noted.
Two mirrors exposed in JET-ILW had co-deposits mainly constituted of Be with equivalent thicknesses equal to 59 and 591 nm. Plasma cleaning was performed with D_2 at 50 or 200 eV and the contaminants layer was decreased to 8 and 7 nm thickness respectively. Even with extended cleaning time, it was not possible to remove the last nanometers of Be, probably because of the formation of a tungsten enriched surface protective layer. No noticeable improvement of optical properties was observed for the JET-ILW samples.
The efficiency of low energetic D_2 cleaning on oxidized or deuterated Be deposits and Be/W mixed films as well as with He addition in the gas mixture will be presented as well.

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Melt tungsten layer surface and droplet splashing by pulse heat flux in the SPICA plasma gun

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The transient high-heat flux can generate melt-layer formation of tungsten (W), melt motion and droplet ejection, leading to surface erosion of plasma facing components (PFCs) and serious contamination in plasmas which become major concerns in large fusion devices such as ITER. Recently, modeling and computational studies of the physical mechanisms involved in splashing and formation of droplets have been performed [1], but detailed comparisons with experiments are required to understand physical process of macroscopic melt droplet formation under ELMs. So far, transient W melting experiments using plasma gun facilities do not include the imposed magnetic field on the target. This paper will present the experimental study of dynamics of W droplet splashing, 2D surface temperature and vapor shield effects with including the effect of the applied magnetic field which was carried out in the magnetized coaxial plasma gun SPICA facility at NIFS.

We demonstrated the droplet splashing from the melt W target plates (30x40 mm). These phenomena were observed under the condition of ITER-ELM relevant heat loads of 1.9 MJ/m^2 obtained with the peak gun current of 200-300 kA. The velocity of hydrogen plasma stream is 120-160 km/s. The electron density of plasma is $2\text{x}10^{21}$ m^{-3}. The $\boldsymbol{J}\text{x}\boldsymbol{B}$ pinch force produces the droplet ejection of the melt layer due to the plasma pressure. The droplet speed is about 26 m/s at t= 0.3 ms and then slows down to 13 m/s at t=1 ms. We have measured by using the fast camera (FASTCAM HX-3: Nac Image Tech.) combined with two colors optical system (TM2S) 2D surface temperature profiles of the W. The peak temperature has reached rapidly up to the melting temperature of 3695 K. This is consistent with 3D heat flux conduction analysis. The 2D surface temperature evolution shows a wave-like pattern which may be caused by Kelvin-Helmholtz instability [2].The number of the W droplet decreases as the externally applied $B_{ext} < 0.1$ T increases. The suppression efficiency depends on the direction of the parallel magnetic field. The propagation of surface waves may be damped by the imposed magnetic field parallel to the W-melt flow. Also, we will examine vapor shield effects by a spectroscopic measurement.

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Optical Property of Nanostructured Tungsten for Plasma Emission Light

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Tungsten (W) has high durability for erosion by plasma irradiation and low tritium retention property. Thus, it becomes very popular as one candidate of plasma facing material (PFM), *i.e.*, first wall and divertor tiles. However a several problems to use W as the PFM remain unresolved. One of them is an influence of reflection of light on optical diagnostics, which is called stray light problem [1]. This problem is caused by the fact that W has much higher reflectivity than carbon based materials. Some of the authors, K.S, M.Y. and N.O., proposed the usage of nanostructured W to conquer the stray light problem. The nanostructured W is generated by the irradiation of helium particles under some condition [2]; it can be used for light absorber from ultra-violet and near infrared wavelength range [3, 4]. In Ref. [1], K.S. *et al.* also found that optical reflectance of the nanostructured W depends on the morphology of the surface. However the cause of the dependence is still unknown.

In this paper, we reveal the cause of the high absorptivity of the nanostructured W by finite-difference time-domain (FDTD) simulation with Drude-Lorentz model [5]. The FDTD simulation is one of the popular methods to analyse the electromagnetic-wave transmission, by which we have investigated surface plasmons on the surface of Au-nanorod [5].

In our FDTD simulation, the electromagnetic plane-wave is irradiated to two types of the nanostructured W targets; the first one is modelled as the fractal structure, *i.e.*, Menger sponge [6, 7]; another one is the three dimensional structure which is observed by transmission electron microscope (TEM). Waiting for a while after irradiation of the plane wave, it becomes the steady state. It is found that the electromagnetic field is localized in the target. This fact shows that the electromagnetic field is absorbed in the inside of the target. This is the reason why the nanostructured W has high absorptivity.

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Deuterium sputtering and retention in lithium-coated plasma-facing components

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Coating plasma-facing components (PFCs) with lithium (Li) is becoming an increasingly popular technique to enhance the operational plasma performance of many magnetic confinement fusion devices. The National Spherical Torus Experiment (NSTX) and the Lithium Tokamak Experiment (LTX), among others, have leveraged the impurity gettering and deuterium (D) retention properties of Li PFCs to reduce the global wall recycling coefficient [1-3]. Reduced recycling of this manner leads to hotter edge plasmas and reduced instabilities, which in turn improves overall plasma performance. However, Li PFCs can also undergo significant erosion and sputtering by energetic ion bombardment, both limiting the material lifetime and contaminating the plasma. Though the effects of Li PFCs on tokamak plasmas have been demonstrated, the details of how Li PFCs retain incident D and how these surfaces sputter and evolve over time is still not well understood.

It is difficult to study the details of Li PFCs behavior *in-situ*, since tokamaks inherently create a complex environment for plasma and material characterization. In this work, as a representation of Li-coated PFCs in a tokamak [3], the D retention mechanisms and sputtering rates of thin Li films on a nickel (Ni) crystal were studied in a well-controlled environment. By working in a chamber with a base pressure of 10^{-10} Torr, Li films were studied without oxidation or contamination effects. A D_2^+ beam with a tuneable energy ranging from 400 – 1200 eV was produced with a differentially pumped ion gun. Auger electron spectroscopy (AES) was used to check surface cleanliness, and temperature programmed desorption (TPD) was used for retention and sputtering measurements.

Previous first-principles molecular dynamics (MD) simulations and plasma experiments at the Magnum-PSI linear plasma device have suggested that rock-salt like LiD precipitates in liquid Li under D_2^+ irradiation [4], and that surface erosion rates are reduced in the presence of LiD [5]. We compare our surface science measurements with these results to illuminate the underlying mechanisms involved in D retention and sputtering of Li and LiD. Preliminary data show that 2 monolayers of pure Li converts to LiD after 1 min (1×10^{19} D_2^+/m^2) of D_2^+ bombardment at 1000 eV and that 8 monolayers of LiD sputters within 1 hr (7×10^{20} D_2^+/m^2) under D_2^+ bombardment at 1000 eV. We provide sputtering rates of Li and LiD at different ion energies and characterize the D retention properties of these films.

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Simplified technique of secondary electron emission yield evaluation of a micro-architectured surface

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Secondary electron emission (SEE) in plasma-facing wall surfaces can lead to the overall reduction in performance of plasma devices. Materials with complex surface structures have low SEE properties due to the trapping of secondary electrons in their micro-cavities, and SEE suppression for such materials often depends directly on their surface structures [1]. We propose a new method using Scanning Electron Microscopy (SEM) developed at PPPL and at the Princeton Institute for the Science and Technologies of Materials which simultaneously evaluates the surface morphologies and SEE yield properties of a micro-architectured surface. This technique was applied to carbon velvets, and results show that the method is effective at estimating SEE yield by pixel intensity analysis of SEM-acquired images compared with direct laboratory measurements [2] and with recent theoretical models [3]. It is also shown that the material's surface conditions can be surveyed and quantitatively measured by SEM image processing. These properties, in the case of velvet surfaces, include the orientations of misaligned fibers, and the true fiber packing densities. The SEM technique is therefore demonstrated to be a quick method for SEE yield and corresponding local surface feature evaluation.

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The effect of grain size on the transport of deuterium in tungsten

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In this work we studied the effect of grain size on retention and transport of deuterium in tungsten. Tungsten consists of grains with distinct crystal structure that are separated by grain boundaries. These grain boundaries can act as weak trapping sites for hydrogen isotopes and also act as a faster way for deuterium diffusion into the bulk of the material [1]. From the results of this study we can extrapolate the influence of grain size on tritium retention and transport in the walls of future fusion reactors made of tungsten.

We carried out the experiment on three polycrystalline samples of tungsten that have different average grain sizes and a monocrystalline sample with surface orientation (100). In tungsten, native defects in the crystal lattice are present which act as strong trapping sites for hydrogen isotopes. Additional defects are created when the material is bombarded by particles from the hydrogen plasma and by neutron bombardment. In experiments samples were bombarded by high energy W ions as a substitute for neutron bombardment. We have structurally damaged the tungsten samples by irradiation with 20 MeV W^{6+} ions [2], creating defects to a depth of $2.3\,\mu$m. These samples with different grain sizes were then exposed to a flux of deuterium atoms at a temperature of 600 K for 70 hours. During the exposure we monitored the amount of retained deuterium in the material and its depth profile using Nuclear Reaction Analysis (NRA) by ^{3}He ion beam, explicitly using the reaction $D(^3He,p)\alpha$. The samples were also analysed with Thermal Desorption Spectroscopy (TDS), which, through simulation, provided us with the energies and densities of the hydrogen traps in the material.

The time dependence of retention of D during the loading with D atoms, which we acquired with the *in situ* NRA measurement, clearly shows a dependence on grain size in damaged tungsten in different rates of D uptake, while the TDS spectra show no difference between different tungsten samples. This gives our method of exposure to atomic hydrogen and *in situ* NRA analysis a significant place in the research field of hydrogen transport in materials. Our experimental set-up also allows the quantification of the acquired experimental data.

We have described the complex process of hydrogen transport in tungsten, where grain boundaries play a significant role, on the macroscopic level by changing the height of the potential barrier for entrance of deuterium atoms into the bulk. For this purpose, we simulated our experiment with the TESSIM code [3]. As it turns out, we could effectively describe the transport of deuterium in the bulk of the tungsten by reducing the potential barrier for samples with smaller grain sizes. Meanwhile the barrier for samples with large grain size approaches the value for the damaged single crystal sample.

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Impact of unipolar arcing on PFC surfaces in DIII-D divertor*

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Impact of unipolar arcing on plasma facing component (PFC) surfaces is studied in the lower divertor of DIII-D tokamak using fast filtered imaging and post-mortem analysis of exposed surfaces. The emphasis is made on arc erosion of high-Z materials (W and Mo), as it has been shown that arcing may be a locally dominant erosion process for high-Z PFCs [1], whereas previous studies of arcing on carbon PFCs in DIII-D showed arcing to be a relatively small erosion source compared to other mechanisms [2]. Arcing on metallic surfaces is also known to produce particulates that may be more efficient in contaminating core plasma with high-Z impurities than sputtered atoms.

Small material samples are exposed in the lower divertor of DIII-D under controlled plasma conditions using Divertor Material Evaluation System (DiMES). DiMES samples are viewed from above by filtered cameras that are able to register arc events. A DiMES head allowing external electric biasing of the samples has been designed and tested. The head consisted of a TZM (Mo alloy with 0.5% titanium and 0.08% zirconium) plate ~5 cm in diameter, half of which was coated with W, housing seven 6 mm diameter button samples including bulk W, W-fuzz, Ti-coated W, graphite and Si. External electrical biasing of the samples to -150 V did not result in any arcing in L-mode, whereas in H-mode arcs were observed only on W-coated part of TZM plate and W fuzz. From this and other DiMES experiments we conclude that: (i) W-coated surfaces are more prone to arcing than TZM; (ii) thin W coatings are more prone to arcing than bulk W; (iii) surface properties of the samples affect the arcing rates.

Arcing has also been observed on W-coated TZM tile inserts during the recent Metal Rings Campaign, which tends to occur mostly during Edge Localized Modes (ELMs) and disruptions [3]. About ten times more arc tracks were found on the outboard (shelf) metal ring compared to the inboard (floor) ring. This is attributed to the floor ring being shadowed from disruption loads and to different W coating surface properties and thickness on the two rings. Here we present results of further post-mortem analysis of these W-coated tiles, with the emphasis on the arc track properties and statistics.

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Impact of Ar-seeded and pure D plasmas on WCrY Smart Alloys

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Tungsten (W), owing to its various advantageous properties like low erosion yields and an exceptionally high melting point, is envisaged as first wall material for future fusion devices such as DEMO. W-based smart alloys address one of the drawbacks of W, its fast oxidation caused by a so-called Loss-of-Coolant-Accident (LOCA), a potential accidental reactor scenario with air ingress. The wall heats up to temperatures of up to 1200 °C for several months due to nuclear decay heat [1], which gives rise to sublimation of activated WO_3 and eventual release of radioactive material into the environment. By adding chromium (Cr) and yttrium (Y) as alloying elements, a protective oxide scale is formed on top of the smart alloy's surface when being exposed to oxygen. During regular plasma operation, alloying elements should be depleted at the surface, resulting in an enrichment of W.

For testing the smart alloy's plasma compatibility and assessing the plasma's impact onto the self-passivation behaviour, WCrY and pure W samples have been exposed in the linear plasma device PSI-2 at FZJ [2]. For the two experiments conducted in steady-state pure deuterium (D) plasma a fluence of $1*10^{22}$ ions/cm^2 was reached. Plasma conditions are based on calculated DEMO first wall loads [3]. As a consequence of preferential sputtering, ion irradiation at energies of 220 eV resulted in a nearly doubled volumetric loss of WCrY in comparison to W. In contrast, at lower ion energies of 120 eV, Cr and Y were significantly depleted towards the surface while W was enriched. With Y and Cr depleted, the plasma faced a nearly pure W surface. Subsequently, measurements making use of the Focussed Ion Beam (FIB) Technique revealed similar sputtering yields for W and WCrY alloys.

Oxidation of WCrY samples exposed at ion energies of 220 eV did not show significant degradation in comparison to non-exposed samples [4]. Results of the oxidation performance of plasma-exposed samples indicating W enrichment will be presented at this conference. Furthermore, for more conservative lifetime estimates of W-based smart alloys, plasma exposure using D with a small amount of argon (Ar) ions serving as impurity species will be reported. Understanding the plasma impact onto the alloy's surface morphology and composition is supplemented by SDTrimSP modelling, using a one-dimensional and also two-dimensional target model.

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3D measurements and simulations of ion and neutral velocity distribution functions in the boundary of a magnetized plasma

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We present analysis of 3D laser induced fluorescence measurements and modeling of ion and neutral velocity distribution functions (I/NVDFs) in a 3D volume near a boundary of a magnetized plasma. These measurements are performed in the presheath region of an absorbing boundary immersed in a background magnetic field that is obliquely incident to the boundary surface (ψ =74°). Parallel and perpendicular flow measurements demonstrate that cross-field ion flows occur and that ions within several gyro-radii of the surface are accelerated in the **E** × **B** direction. We present electrostatic probe measurements of electron temperature, plasma density, and electric potential in the same region. Ion, neutral and electron measurements are compared to Boltzmann simulations, allowing direct comparison between measured and theoretical distribution functions in the boundary region.

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Parametric Investigation of Helium and Deuterium Concentrations in Tungsten Using Laser-based Techniques

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Fuel retention is one of the crucial areas to be investigated for future fusion reactors. Helium (He) and deuterium (D) ion fluence greatly influence fuel retention and permeation mechanisms. The purpose of this research is to use laser based techniques to investigate fuel retention as a function of tungsten microstructure and fluence. Single crystal, as well as polycrystalline tungsten specimen were exposed to voltage biased He, 10% He and 90% D, and D plasmas at PISCES. The single crystals included (100), (111), and (110) orientation variants, and the plasma exposure conditions consisted of an ~100 eV bias voltage and average surface temperature of ~500° C. The particle flux regimes investigated were 5e17 cm-2s-1 up to 1e18 cm-2s-1 and 1e18 cm-2s-1 up to 1e20 cm-2s-1. Laser Ablation Mass Spectroscopy (LAMS) with simultaneous Laser Induced Breakdown Spectroscopy (LIBS) has been used to measure the He and D atomic concentration as a function of depth in the specimen. LIBS and LAMS are similar ablation techniques using a focused laser beam to ablate a small volume from the surface forming a small high plasma. Within the plasma formation, LIBS investigate the elemental intensities within a few microseconds following ablation using optical emission spectroscopy. The plume is then quickly pumped out of the chamber and into the quadrupole mass spectrometer where the elemental mass is measured. Concentration is determined from the comparison of the emission and mass measurements. Results are shown as concentration as a function of depth for the various fluences and plasma configurations, as well as compared to other surface gas evaluation techniques.

Liquid Lithium Target for Neutron Generation

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At the University of Illinois, a compact, liquid lithium loop has been developed and tested. The compact, yet scalable, loop is comprised of a stainless steel trench system embedded with heaters and cooling lines, and was designed to handle large heat and particle fluxes in neutron generators as well as fusion devices. Lithium flow is driven through the sole use of thermoelectric magnetohydrodynamics [1]. The benefits of this device are two-fold: 1) produce high energy neutrons for materials testing and radiography, and 2) provide a self-healing, low-Z, low-recycling wall material. The flowing lithium will keep a fresh, clean surface, allowing Li-7(d,n) reactions to occur, as well as enhance the deuterium adsorption in the fluid. The enhanced deuterium absorption helps by increasing the total neutron output. Expected yields of this system would be 10^7 n/s for 13.5 MeV neutrons and 10^8 n/s for 2.45 MeV neutrons. Previous work has shown that using a tapered trench design allows for an increase in fluid velocity at the particle strike point. This shaping promotes less depression of the lithium surface, which helps to prevent dryout. Initial experiments, where a temperature gradient was imposed using the embedded heaters and helium cooling, peak velocities of 16 ± 4 cm/s were observed. For heat fluxes greater than 10 MW/m^2, COMSOL fluid models have shown that sufficient velocities (~70 cm/s) are attainable to prevent significant lithium evaporation. Future work will be aimed at addressing the wettability of lithium on stainless steel at lower temperatures to protect the permanent magnets in the system and experimentally determine the necessary velocities and geometry to prevent dryout. The system's neutron output will also be experimentally determined. The early results and discussion will be presented.

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MD simulation of atomic-scale processes on self-irradiation of tungsten

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Since tungsten is one of promising materials for plasma facing material in future nuclear fusion devices, related investigations are intensively performed. So far we have focused on atomic-scale electronic property of tungsten material by use of quantum chemistry simulation based on density functional theory (DFT) [1-3]. Larger scale physics and interactions with plasmas are to be considered in order to proceed material study with numerical simulation. In this context, a method of molecular dynamics (MD) simulation [4] is employed and organized calculations are carried out in this research, which can treat larger scale than DFT does and is expected to give a model of a material-side boundary condition for a plasma simulation in peripheral region.
Re-entering of sputtered tungsten atom is inferred to have large effect on tungsten material itself, since tungsten is a heavy atom compared to hydrogen or helium atoms. Thus it is important to study an influence of tungsten self-irradiation to tungsten material, especially in order for evaluation of a durability of tungsten materials against plasma exposure.
MD simulations of tungsten self-irradiation show that (1) in case of an incident energy less than 100eV, most of incident atoms slightly invade target tungsten material once, and then are trapped on the surface of target material; (2) increasing incident energy, incident atoms collide with an atom in target material and recoil atom preferably moves [111] direction. Sequence of the collision and recoil occurs while kinetic energy dissipates into circumjacent atoms, then propagates backward. Some recoil atoms are trapped as self-interstitial atom. Some are trapped on the surface or released from the surface resulting to sputtering; (3) in case of incident energy higher than 500eV, collisions of incident atom and subsequent recoil atoms occurs at the relatively shallow region near material surface, and collision chain in [111] direction, which is typically observed in a few hundred eV of incident energy, is blocked. As a result, sputtering yield is enhanced, and crystal defect like vacancy or interstitial atom is formed at shallower region.
In the presentation, comparison to results obtained by the binary collision approximation model (BCA) [5], which is simpler model and requires less computational resource, and DFT evaluation for typical atomic configurations observed in MD simulation are also addressed.

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MEMOS 3D modelling of ELM-induced transient melt damage on an inclined tungsten surface in the ASDEX Upgrade outer divertor

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Recent matched experiments in JET and ASDEX Upgrade [1,2] have exposed small tungsten (W) samples at the outer divertor strike point to high power H-mode plasmas, with Edge Localized Mode (ELM) energy densities and timescales similar to those expected during mitigated ELMs in ITER burning plasmas. Two different sample geometries were deployed in both devices, featuring an abrupt leading edge of ~1 mm height and an inclined surface with (slope angle 15°) such that, in each case, ELM-induced melting could be obtained on the inertially cooled surfaces. Good agreement with the ASDEX Upgrade leading edge experiment (post-mortem melt profiles, net electrical current to the W sample) has been obtained with the MEMOS-3D melt motion code under the conjecture that the current responsible for the **J**x**B** force on the melt layer is the electron flow through the bulk of the sample to replace the loss at the surface due to thermionic electron (TE) emission. The code has been recently updated with respect to the thermophysical W properties & the description of thermionic electron emission under space-charge limited conditions as well as a replacement current module has been added.

Application of MEMOS-3D has now begun for the sloped sample, the geometry of which provides a further important test for the replacement current hypothesis in that incident field line angles are lower in comparison with the perpendicular leading edge so that emitted electrons may escape the melt zone much less readily. The geometry also has the advantage that the loaded surface can be viewed with a fibre-based infra-red (IR) system, allowing direct observation of melt onset [3]. Using spatio-temporal incident heat flux profiles as input and assuming an optical approximation for the power loading, the maximum TE emission current is found following a scaling relation to the plasma flux obtained by dedicated PIC simulations [4]. As a consequence of the combined effects of the sheath virtual cathode and prompt electron re-deposition, the TE current density is already limited at values of surface temperature well below the W melting point and is dramatically reduced with respect to the nominal Richardson current [4]. Once ELM-induced melting is achieved in the simulations, the resulting **J**x**B** force on the melt layer is significantly lower than for the leading edge case, such that the predicted net poloidal melt displacement is on the order of a few mm, roughly consistent with experiment. Moreover, modelled melt velocities, on the order of a few tens of cms^{-1}, are close to those derived from IR images of the actual melt motion [3]. Much more challenging code runs are also underway seeking a qualitative match to the experimentally observed complex final melt profile on the sloped surface and thus understanding of the mechanism(s) leading to the onset and progression of the observed corrugated damage topology.

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Temperature dependence of the sputtering yield and the tungsten surface enrichment of Eurofer-97 steel exposed to the deuterium plasma of GyM

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In DEMO, the erosion by hydrogenic charge-exchange neutrals (CXNs), with the largest fluxes at energies of the order of 200 eV and below, will strictly affect the lifetime of the recessed elements of the first wall [1]. Among the possible materials candidate for these components, bare reduced activation ferritic martensitic (RAFM) steels, such as Eurofer-97, are a valuable economical and technological option. RAFM steels are iron-based alloys containing small amounts of high-Z elements like tungsten, W (~0.47 wt.% for Eurofer-97).
Due to the higher sputtering yield (Y) for hydrogenic particles of iron (Fe) compared to that of W, it was demonstrated [1,2] that iron of RAFM steels exposed to low energy deuterium (D) ions erodes fast, leading to a W-rich surface (with a depth of some nm) which is very difficult to be sputtered. Moreover, it turns out [1] that the erosion dynamics of RAFM steels strictly depends on their temperature (T). Thermal effects like inter-diffusion of Fe and W and also the W segregation toward the surface [3] could indeed respectively enhance and counteract the erosion. However, a comprehensive study of the impact of temperature on W enrichment of RAFM steels is still missing from literature.
For a controlled investigation of W enrichment as a function of RAFM steels temperature, mirror finished $10\times20\times1$ mm^3 Eurofer-97 samples were exposed to the D plasma (with electron density and temperature of $n_e = 5.3\times10^{16}$ m^{-3} and $T_e = 7.5$ eV) of the linear machine GyM [3], at three different temperatures: 600 K, 800 K and 1000 K. For each T, five ion fluences in the range of $4.5\times10^{24} \div 2.3\times10^{25}$ ions m^{-2}, were considered. In order to enhance Fe preferential sputtering, D ions energy was kept nearly constant to 200 eV applying a proper negative bias voltage to the samples holder, thus having a high Y of Fe still being below W sputtering threshold.
The erosion and the W enrichment of the surface of Eurofer-97 samples after GyM exposures were investigated and the results are here presented and discussed. The former was evaluated both from profilometry and mass loss measurements and a comparison between the local and global sputtering yield is given. The latter was estimated using Rutherford backscattering spectroscopy (RBS) plus the SIMNRA software. Due to the very limited depth of the W-rich surface layer, the SIMNRA reconstructed elemental profiles were corroborated by low-energy ion scattering (LEIS) spectroscopy (depth resolution of ~1 nm) measurements. Considering the highest fluence, preliminary results show that the erosion of Eurofer-97 at 1000 K is ~50% higher than that at 600 K probably due to inter-diffusion of Fe and W.

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Deuterium retention and erosion in liquid Sn samples exposed to D_2 and Ar plasmas in GyM device

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Divertor plates of tokamaks are known to be subjected to extremely high heat loads. Melting, cracking and other damages of Plasma Facing Components (PFCs) may occur [1]. Experiments on tokamak after severe melting of W tiles in the divertor have indicated that such damage events could have adverse effects on reliable machine operation when the plasma is in contact with those damaged tiles. Liquid Metals (LMs) present many potential advantages when compared to solid tungsten PFCs. LMs option has already shown to be a viable PFC in laboratory experiments simulating fusion environments, as well as in tokamak experiments [2]. Lithium [3] is the best-known liquid metal applied to fusion devices and it has shown low recycling operation in present day devices at low wall temperature, but there are serious concerns about tritium retention and high vapor pressure. Tin has emerged as a feasible alternative [4]. Preliminary studies have been done, but more detailed studies are needed.

This work will provide an overview of the recent activity carried out in GYM plasma device [5] on the topics of erosion and retention in liquid tin. As far as the erosion experiments, the S/XB spectroscopic parameter, converting the emission line intensity into an influx of Sn impurity atoms from limiting surfaces [6], has been evaluated for some prominent Sn I lines in the 300-450 nm range by Optical Emission Spectroscopy (OES) and mass loss measurements. Solid and liquid Sn targets have been negatively biased and exposed to an Ar plasma ($n_e \sim 1\text{-}2 \times 10^{11}$ cm^{-3}, $T_e \sim 5\text{-}10$ eV). Two magnetic configurations have been exploited, linear and cusp. The first, the most usual configuration in GyM experiments, has been employed for the exposure of Sn solid targets on the machine axis; the second has been employed for the exposure of Sn liquid targets without the use of the Capillary Porous System (CPS). This configuration, obtained by inversion of the current in two coils, curves the magnetic field lines and reduces the strength of the axial magnetic field at the target position (6 cm under the machine axis), allowing a horizontal exposure of the sample and limiting the self-sputtering by Sn ions which could influence the mass loss measurement. The sample was kept at a temperature of 250 °C so the evaporation flux was negligible. Mass loss measurements have allowed to determine the sputtering flux (3×10^{19} - 2×10^{20} m^{-2}s^{-1}) and the sputtering yield, which is in good agreement with the literature. The thermoelectric magnetohydrodynamic stirring of Sn liquid targets is here presented and will be the subject of further investigation.

Regarding the retention experiments, samples of liquid Sn have been exposed to D_2 plasmas ($n_e \sim 1.6 \times 10^{10}$ cm^{-3}, $T_e \sim 7$ eV) to a 10^{24} m^{-2} fluence (2h 15') in cusp magnetic configuration. Preliminary results show that deuterium retention, studied with Thermal Desorption Spectroscopy (TDS) as a function of Sn temperature (250, 300 and 340 °C), is negligible.

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Dynamic changes of sputtering and reflection yields upon energetic particle bombardment of self-consistently evolving 3-D surface morphologies

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The dynamic changes of surface morphologies under energetic particle bombardment can drastically affect quantities such as sputtering and reflection yields and can even change macroscopic properties, e.g. the classification of areas as net deposition or net erosion regions. Up to now, no self-consistent modelling of these processes was available. MD simulations are too expensive to model relevant system sizes of several micrometers in lateral direction and available binary-collision-approximation (BCA) based codes were restricted to two dimensions.

To provide a self-consistent modelling of the sputtering-induced dynamics of realistic geometries, the SDTrimSP code has been extended to enable for the first time the dynamic processing of arbitrary 3-D surface morphologies, as e.g. provided by AFM, STM, and confocal laser scanning microscopy or by MD simulations.
This allows to describe the fluence-dependent sputtering and reflection yields of the continuously modified surface including non-local effects like shadowing or flux-enhancements by forward scattering, overcoming limitations of previous approaches [1],[2],[3].

The simulation predictions are compared with experimental results (topography development as well as evolution of sputtering yield with fluence) from well-defined monoenergetic 5 keV argon ion beam exposures of 3-D (nano-)structured Si- and Ta-samples under various impinging angles in the high-current ion source SIESTA. The simulated and measured surface topographies, as well as integral weight-loss data are compared quantitatively and the observed differences are discussed.

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Radiation Damage and Deuterium Retention in Tungsten

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Tungsten is a promising candidate material for the wall of a future fusion reactor due to its low erosion yield and low hydrogen solubility. However, fusion neutron irradiation will induce radiation defects in the material which can strongly increase hydrogen retention. Therefore, it is important to study the mechanism of defect creation and its influence on hydrogen retention in tungsten. Neutron irradiation is often simulated by high energy ion irradiation, but it is not clear to what extent the displacement damage created by the ions resembles that of neutrons. In this study different ions with different energies are used to study the effect of the primary-knock-on energy spectrum on damage creation and deuterium uptake. Samples of hot rolled, polished and recrystallized tungsten were damaged with different ion species (p, D, He, Si, Fe, Cu, W) at energies between 0.3 and 20 MeV. The displacement of atoms was calculated using SRIM's quick calculation of damage option [1]. Deuterium (D) retention in self-damaged tungsten saturates at about 0.2 dpa [2]. In order to represent the linear regime and the saturation regime of the increase of D retention as a function of dpa two different damage levels of 0.04 dpa and 0.5 dpa (in the damage maximum) were used. For studying hydrogen retention in defects the samples were exposed to low-temperature D plasma to decorate the defects at 370 K. The D depth distribution was obtained by nuclear reaction analysis using the D(^{3}He, p)α reaction. The damage range calculated by SRIM is in good agreement with the region of increased D retention seen in the D depth profiles. Tungsten damaged by heavy ions (Si, Cu, Fe, W) to identical dpa values shows similar D depth profiles, i.e., D retention, is comparable. For tungsten damaged by light ions (p, D, He) the depth profiles show larger differences. In addition to NRA trapped D was measured by thermal desorption spectroscopy. Tungsten irradiated by heavy ions shows similar deuterium desorption. The desorption spectrum from tungsten damaged by helium shows a significantly different shape. Studies of the damage region by positron annihilation lifetime spectroscopy with low-energy positrons have shown no significant difference for the heavy ions. The damaged region was investigated by transmission electron microscopy. Differences in the dislocation structure in the damage zone damaged by different ions are visible even for those samples showing similar D depth profiles and similar D desorption. To avoid crystallographic effects which can be the reason for the observed differences, single crystals will be used in further studies. These studies may allow clarifying whether the microstructure is responsible for the observed differences in the desorption spectra.

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Interactions of the plasma and guard limiters during lower hybrid wave current drive on EAST tokamak

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Lower hybrid current drive (LHCD) has been successfully used to achieve long-pulse, high performance plasma on EAST tokamak. In recent LHCD experiments on EAST, hotspots near the guard limiters of lower hybrid antenna were often observed as the input power was over 2MW [1]. The hotspots not only caused serious damages to the guard limiters, but strongly degraded the plasma performance due to enhanced impurity productions. In published studies, the heat flux to the limiters was calculated by assuming that the energy absorbed by electron Landau damping was carried by the fast electrons to the limiters, and by ignoring the sheath structure formed in front of the limiter surface [2]. To understand the interaction of the plasma and limiters and to alleviate the hotspot issue, in this work, a particle-in-cell simulation code GCPIC is developed to investigate the interactions between plasmas containing fast electron component and the guard limiters, in which the secondary electron emission and the ion sputtering effects are included. In the one-dimensional simulations, the plasma in front of the launcher is bound by two guard limiters. The electrons are loaded with a distribution in the phase space determined by the resonant interactions between electrons and lower hybrid modes with high parallel refractive index of the launched spectrum. The preliminary results show that a sheath much thicker than a regular Debye sheath is formed near the limiter surfaces. The sheath potential is significantly increased due to the existence of the fast electron component. As a result, the ion's heat flux is enhanced and becomes comparable to the electron's. The total heat flux to the limiter surface thus increases, which gives rise to a significant increase of the limiter surface temperature and ion puttering. The secondary electron emission can reduce the sheath potential and the dependence of the heat flux on wall materials is also discussed.

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Abstract Withdrawn

Time Dependence of Deuterium Retention in Lithium and Lithium Compounds on Plasma Facing Components

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Lithium coating of plasma-facing components (PFCs) has led to improved plasma performance, such as longer discharge time and higher current density [1-2]. These effects are attributed to the effectiveness of lithium in retaining hydrogen isotopes and thus reducing recycling [3-4]. Since lithium readily reacts with background gases present in fusion devices, it is important to understand and parameterize the time dependence of deuterium retention in lithium and lithium compounds for applications of lithium under future long-pulse conditions.

Previous studies have examined the effect of oxygen impurities on deuterium retention in lithium-coated graphite and TZM [5-6]. The present work extends such studies by investigating deuterium retention in lithium and lithium compounds (Li_2O, LiOH and Li-C-O) as a function of time after exposure to deuterium ions (5 min to 24 hours) at various temperatures. This is a critical step in addressing what happens to deuterium trapped in lithium coatings between plasma shots, e.g., whether it diffuses out of the lithium, stays trapped or forms other compounds.

Experiments were conducted in an ultrahigh vacuum chamber with a base pressure of $2x10^{-10}$ torr to avoid contaminants. Thin lithium films of 10-20 monolayers were evaporated on a Ni substrate, known to be immiscible with lithium [7]. Li_2O, LiOH and Li-C-O films were formed by exposing the clean lithium film to O_2, H_2O and CO_2, respectively. These films were then irradiated with a well-characterized beam of 400 eV D_2^+. The samples were left undisturbed in the vacuum for different lengths of time (post-deuterium exposure time) before being analyzed via Temperature Programmed Desorption (TPD), X-ray Photoelectron Spectroscopy (XPS) and Auger Electron Spectroscopy (AES).

Preliminary TPD results indicate that the amount of D retained in clean lithium films is constant for post-deuterium exposure times up to 1 hr, but decreases to one-third of the original amount after 16 hr. Additionally, the amount of Li and LiD decreases, while the amount of LiOD increases, as a function of increasing post-deuterium exposure time. Further experiments are currently in progress to investigate deuterium retention as a function of post-deuterium exposure time for other lithium compounds.

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Quantification and Sensitivity Analysis on the Effects of Uncertainty On Impurity Migration In PISCES-A

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The extreme heat, charged particle, and neutron flux / fluence to plasma facing materials in magnetically confined fusion devices has motivated research to understand, predict, and mitigate the associated detrimental effects. Of relevance to the ITER divertor is the interaction of helium and deuterium with the tungsten divertor. Sub-surface helium bubble formation affects the trapping of deuterium and changes the tungsten surface morphology which results in changes in fuel recycling, ion implantation and sputtering dynamics, and impurity migration and deposition. The development of high fidelity coupled simulations of the plasma boundary and material surface interface are a necessary step for predicting the performance of plasma facing component lifetimes and performance. The global impurity transport code (GITR) has been developed as a high performance trace impurity module that is part of a larger integrated simulation to determine surface erosion, migration and deposition.

The linear device PISCES A [1] has performed dedicated experiments for high (4x10^22 m-2s-1) and low (4x10^21 m-2s-1) flux, 250eV He plasma exposed tungsten targets to assess the net and gross erosion of tungsten and volumetric transport of sputtered tungsten. The temperature of the target was held between 400 and 600 degrees C. While the PISCES A plasma is well characterized, the probe measurements of background plasma electron density, temperature, and ion fluxes have significant uncertainty associated with them. Further, these measurements are made 30 cm upstream of the tungsten target plate. Therefore the modeled plasma profiles also contain uncertainties. The uncertainty in GITR input affect atomic processes and the resulting forces on the impurities which modify the experimental observables of WI emission spectroscopy and mass loss/gain of targets.

We present results showing quantified uncertainty in the predicted erosion / migration / redeposition of W during the He exposure as well as predictions for He/D mixed plasma exposures using probabilistic forward uncertainty quantification methods. This allows for a bounding in the predictions of the experimental observables based on the uncertainty in the input data and models. Varying input parameters to match the experimental error bars shows moderate sensitivity resulting in estimates which bound the experimental measurements by approximately +/-30% error.

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Role of Grain Boundaries on Radiation Damage in Tungsten

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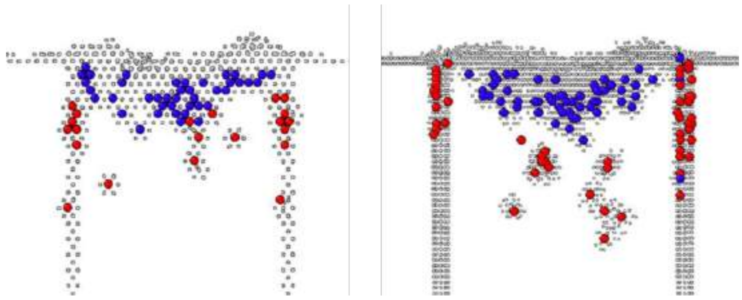
All plasma-facing components in the next-generation fusion reactors will be exposed to high fluxes of neutrons and plasma-particles, and will be expected to operate with walls at high temperatures to achieve thermodynamic efficiency in the fusion energy conversion. At this time tungsten is a favoured for the wall material due to its excellent thermo-mechanical properties, low hydrogen retention, low rate of erosion, and high melting temperature. In metallic materials, the ordered lattice, the surface, and the grain boundary behave drastically different. To make the prediction more accurate, all three components need be carefully studied.

We studied the evolution of the surface defects of nanograined tungsten surfaces due to the 1 keV cumulative self-atom impacts at 300K and 1000K. The simulations were performed by molecular dynamics with bond-order Tersoff-form potential [1], and were conducted on 12 structures, varying the surface planes ({100}, {110}), grain boundary configurations ($\Sigma3<110>\{112\}$, $\Sigma3<110>\{111\}$, $\Sigma5<100>\{130\}$, $\Sigma5<100>\{120\}$) and grain sizes(6nm, 8nm, 12nm). The mechanism of defect creation inside the lattice, interstitial bias absorption in grain boundary, surface roughening during radiation, and the stability of the boundaries at different temperatures were revealed. The basic roles of the grain boundary configurations could be explained by the results of the post analysis, with respect to the number of interstitials and vacancies, and to the deformation of the interfaces.



a) b)

Figure 1. a) $\Sigma3<110>\{112\}$ and b) $\Sigma5<100>\{130\}$6nm grain size interstitials accumulation in grain boundaries

When the collision cascade interacts with a grain boundary, the grain boundary will absorb some of the interstitials, and the number of which is strongly affected by the grain boundary configuration (Fig. 1).

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The effect of gyration on the deposition of beryllium and deuterium at rough surface on the divertor tiles with ITER-like-wall in JET

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Previous experimental results from micro beam analyses on divertor tiles, through 2011-2012 and 2013-2014 operations, showed that the deuterium retention and the beryllium impurity deposition were nonuniform on the rough surface. Frequently, the Be and D were accumulated within pits, cracks and valleys in the range of ~10 µm to ~100 µm and selectively at side slopes within larger pits [1]. Column growth in deposited layers was also observed in determined directions with respect to the magnetic field.

Modelling in [2] suggested the trajectory of carbon ions in the boundary sheath led to an inhomogeneous ion flux on rough surface (a few micro meters). Therefore the motion of ions and the impinging on surface could be the potential reasons for the nonuniform deposition. In this work, in order to figure out the effect of the ions gyrating motion on Be and D deposition on the real divertor topography, a calculation of ion trajectories in the plasma boundary electromagnetic field had been done with the isothermal plasma assumption. The surface topography had been measured by the focus stacking technique in optical microscopy, with the roughness ranging from 10 µm to 40 µm.

This calculation results suggest that:

- The gyration of the Be and D ions in grazing magnetic field ($\leq 5^\circ$) led to the deposition on the slope towards the magnetic field.
- The different impact angle (between ion velocity and surface normal) for Be and D could result in the accumulation in different slope.

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Comparison of surface blistering and deuterium retention between W and W – 5% Ta under 30 keV deuterium ion implantation

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Samples of tungsten (W) and tungsten with 5 mass percent of tantalum (W – 5% Ta) were exposed to 30 keV deuterium ion implantation at a fluence of 8.86×10^{21} ion/cm^{-2}s^{-1}. The surface morphology with blistering and deuterium retention were investigated and compared between W and W – 5% Ta. The blisters on W – 5% Ta surface were found less than that on W surface, and the deuterium retention was higher in the W system than that in W - 5% Ta. The trapping in the sub-surface cavities associated with blistering is considered as the predominant manner of deuterium retention. Therefore, the lower retention in W– 5% Ta was attributed to the less blistering.

Plasma Edge and First Wall Diagnostics

Surface characterization of PFC materials in the National Spherical Torus Experiment Upgrade with the Materials Analysis and Particle Probe

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Plasma Facing Components (PFC) conditioning has a crucial role in plasma performance in tokamaks. Traditionally, PFC materials are examined at the end of an experimental campaign, following hundreds of plasma shots. The Materials Analysis and Particle Probe (MAPP) is an *in vacuo* materials characterization facility installed in the National Spherical Torus Experiment Upgrade (NSTX-U) lower divertor [1-3]. MAPP is designed to examine mixed-material surfaces at ultra-shallow depth resolutions between a few monolayers to tens of nm where the interaction of hydrogen isotopes and complex materials is key in determining the effect of plasma-surface interactions on plasma behavior. A combination of MAPP data, *in-situ* ex-vessel data and computational modelling was used to elucidate the dynamic and complex surface chemistry of B-C-O interactions with energetic D particles [4]. Complementing this study, *post-mortem* analysis of samples extracted from ATJ graphite tiles located at three different radial locations along the inner and outer lower divertor of NSTX-U and from the center stack. Based on cumulative NSTX-U plasmas surface chemistry characterization between regions near the outer-strike point and private flux region was possible. Results show D-irradiation driven effects influencing B and O states.

MAPP was also used to conduct XPS studies on ATJ graphite and TZM alloy (>99% Mo, 0.5% Ti, 0.08% Zr) samples. These measurements showed that plasma induced oxidation of the boron coatings plays an important role in the chemical evolution of the surfaces, and as a consequence in plasma performance. Ex-vessel i*n-situ* laboratory experiments were performed to complement the observations made with MAPP, including D irradiation studies and XPS depth profiles. Results in ATJ graphite experiments show that D exposure increases oxygen concentration, highlighting the influence of these two species on the chemistry of the samples.

During 2018, MAPP will be transported to UIUC for hardware upgrades. The main changes made will be incorporating a new energy analyzer to improve the resolution of the XPS data and adding new diagnostic techniques such low-energy ion scattering spectroscopy, direct recoil spectroscopy and thermal desorption spectroscopy.

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Oxygen retention in boronized carbon surfaces and its dependence on plasma exposure in the National Spherical Torus Experiment-Upgrade (NSTX-U)

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Boronization is a Plasma Facing Component (PFC) conditioning technique widely used in tokamak machines [1]. The National Spherical Torus Experiment-Upgrade (NSTX-U) applied this conditioning method by using deuterated Trimethyl-boron (d-TMB) in a He DC glow [2]. The use of boronization during the campaign improved the plasma performance, allowing longer plasma discharges and H-mode access. An ATJ sample, used as a proxy for the graphite NSTX-U PFC, was inserted into the NSTX-U vacuum vessel using the MAPP diagnostic. The chemical state of the ATJ sample was monitored *in-situ* using X-ray Photoelectron Spectroscopy (XPS) without breaking vacuum. The XPS data showed a progressive increase (<5% to 23%) in the oxygen concentration of the boronized ATJ sample as the D^+ fluence increased [3]. Filterscopes were used to measure the light emitted by oxygen impurities in the plasma-surface interface. An increase in the registered magnitude of the OII line normalized to the D_γ intensity was observed as the concentration of oxygen on the ATJ surface increased. At the same time, plasma parameters including pulse duration, energy confinement time, confined energy and electron density suffered reductions close to 50% with increasing D^+ fluence. In this work, we present a quantitative analysis of the evolution of the chemistry of the ATJ surface, and correlate it with presence of oxygen in the plasma-surface interface. We use these results as evidence for the relationship between oxygen in the PFC and plasma parameters that are related to discharge performance.

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Flux Measurements and SiC Erosion Experiments in Proto-MPEX

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Particle flux is an important component in surface modification of fusion relevant first wall materials. The linear device, Prototype Material Plasma Exposure eXperiment (Proto-MPEX), has measured high particle fluxes of 5±2.5 x $10^{23} m^{-2}s^{-1}$ at the target using a new carbon target with an embedded ion flux collection probe and a double Langmuir probe (DLP) array. Ion flux measurements from an embedded target DLP match the derived fluxes from a DLP ~30 cm upstream of the target and Thomson scattering (Thomson) measurements ~4 cm in front of the target by using the measured electron density and temperature while assuming the Bohm criterion. The upstream DLP flux was calculated to be 4.36 x 10^{23} $m^{-2}s^{-1}$ and the Thomson flux to be 4.05 x 10^{23} $m^{-2}s^{-1}$. The carbon target leads to increased recycling at the target and a higher particle flux.

To explore the recycling in front of the target further a moveable target is being implemented on Proto-MPEX. Characterization of the plasma parameters in front of the target. The Thomson scattering measurement location is held at a fixed position in the device, while the target is moved along the z-axis of the machine, thereby varying the measurement location from the target face. In addition, D_alpha, D_beta, and D_gamma light emission will be measured with a spectrometer in front of the target. The ratios of these light emissions give insight to target recycling. Experiments on the erosion of a SiC/SiC composite targets will be presented for varying target temperatures.

Development of an in-situ diagnostic system for mapping the deposition distribution on PFCs of HL-2M

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Material erosion and deposition on the wall are curial for the operation and control of discharged plasmas in tokamaks since they may limit the lifetime of plasma facing components, dilute the core plasma with impurities, enhance the fuel retention and cause an amount of dust. The assessment of such effects due to material erosion and deposition is of primary importance for the steady-state operations of future fusion devices. However, many issues on the deposition and migration of impurities in a tokamak are unclear. In particular, a diagnostic system to investigate in-situ the deposition/codeposition distribution on the whole plasma facing components (PFCs) on tokamaks is required.

In this paper, an in-situ diagnostic for mapping the deposition distribution (IDMDD) on PFCs for HL-2M is developed. It has the advantages of evaluating the deposition behaviors over most of the vessel wall and eliminating the accumulation effects of tens or hundreds of discharges. This system is built with the laser-induced breakdown spectroscopy (LIBS) technique which is an in-situ diagnostic technique to analyze the constituents and the depth profile of deposition compositions on PFCs. The IDMDD system can scan poloidally from the lower inner wall via the lower divertor to the outer mid-plane and can also scan toroidally. Therefore, with this system, the wall properties over a broad area of the vessel can be measured. The system has been designed, integrated and tested in the lab. It will be applied to HL-2M later when this new tokamak is ready. All elements with the emission lines in the range of 180-850nm, including the possible deposited materials such as H, D, Si, C, Fe and so on can be analyzed and the isotope species H and D can also be distinguished with this diagnostic. The depth profiles of the deposited materials can be obtained and the fuel retention on PFCs can be evaluated in-situ. In order to investigate over a wide area of the wall, a Mo mirror in the vessel and its manipulator are applied. The mirror can move radially and rotate poloidally and toroidally in the vessel so that the LIBS can scan over the wall. The resolution of rotation is 0.1° and that of translation is 0.01mm. As a result, the spatial resolution of the IDMDD system on PFCs is a few millimeters in terms of its optical design for HL-2M. The system can be remotely controlled and its optical lenses, mirrors and fibers can be adjusted automatically when scanning over the PFCs. Then a clear picture of the deposition over the wall of HL-2M can be given. Characterizing the wall properties is important to understand the deposition behaviors and impurity migrations during operations.

Conceptual design of a first wall diagnostics manipulator with full in-vessel environment compatibility

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A remote control in-vessel operation manipulator system for China's Experimental Advanced Superconducting Tokamak (EAST) has been under development for many years. The previous manipulator prototype demonstrated practicable operation mode for full in-vessel workspace coverage. In this paper we mainly talk about adaptations of the current manipulator structure to improve its ultra-high vacuum environment compatibility inside the vacuum vessel. In the vacuum baking cask that stores the manipulator for pre-treatment before in-vessel operation, magnetic coupling structure is used to drive the long translating platform which sends the manipulator into vacuum vessel through the observation window on tokamak. All other manipulator drive motors are sealed from vacuum ambient for extreme environment protection and the movement is transmitted out through screw nut structure in metallic bellow. Most of the manipulator joints are revolute joints driven directly by push rod mechanism. For the end effector platform joint which requires a relative larger moving range and compact overall dimensions, a special pully structure is developed for movement transmission. Simple kinematics model of the manipulator is present at the end of the paper.

An extensive thermal diagnostics package for the determination of divertor heat flux footprints on HL-2A

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The maximum steady-state heat flux that can be safely handled by a material surface is typically $\sim 10MW/m^2$, the value which is presently set for ITER's peak divertor heat flux. Because of this constraint, the size of the decay length of divertor heat flux, λ_q, directly sets the acceptable fraction of power in the scrape-off layer (SOL) that may deposit onto the divertor targets. So the λ_q is a critical parameter for any tokamak with reactor-level power entering into its SOL. However, the scalings of λ_q in existing tokamaks are lacking for making extrapolations to ITER, and extrapolating present divertor results towards future machines also requires a detailed understanding of the heat flux pattern on the divetor targets.

For understanding this complex process, recently an extensive thermal diagnostics package that is comprised of infrared (IR) thermography, Langmuir probes (LPs) and thermocouples, has been developed in the HL-2A lower divertor chamber. The IR thermography system is equipped with a FLIR SC7300 camera to measure the temperature evolutions of the lower outer divertor plates through a vacuum window. The IR camera detects emission in the $3-5\mu m$ bandpass with 320×256 pixel resolution. The detector is cooled to 73K, and full-frames can be read out at a rate of 230Hz. However, the sampling rate can reaches up to 23KHz with a sub-window, which is useful for the analysis of heat flux footprints of edge localized mode (ELM). Moreover, the emissivity of the surfaces is non-uniform because the old plates have exposed to the plasma more than ten years, so a new carbon plate was used to cover the old plates to eliminate this effects. The IR system resolves ~1mm scale features on the carbon surface. Two LP arrays are installed on the inner and outer divertor plates respectively with a spatial resolution of 5mm near the separatrix region and 10mm far SOL region. Thus the heat flux 'footprints' can be cross-checked and then determined in detail with the two high resolution systems. In addition, 30 low-noise thermocouples are designed and embedded in the inner and outer divertor plates with a spatial resolution of 6mm near the separatrix region and 12mm far SOL region. This system is mainly used to calculate the heat flux during steady-state because of the low temporal resolution, and to supply a calibration source for the IR system since the temperature measured by thermocouple is independent on the surface emissivity.

Extensive cross-comparisons of heat flux are on-going among IR thermography, LPs, and embedded thermocouples. The preliminary results show that these data verify the full set of IR-inferred heat flux footprints during steady-state operation, and thus the λ_q is determined.

Development of a 30 – 165 GHz profile reflectometer and performance evaluation for ITER

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The Low-Field Side Reflectometer (LFSR) for ITER will provide real-time edge density profiles every 10 ms for feedback control and every 24 μs for physics evaluation. Laboratory evaluation of a prototype LFSR transmission line (TL) is underway to assess its performance compared to measurement requirements. The 40-meter TL includes circular corrugated waveguide, ten miter bends, reference-phase calibration mirror, waveguide switch, electron cyclotron heating (ECH) protection system components, Gaussian telescope, vacuum windows, and containment membranes. Frequency-modulated continuous wave (FM) transmission signals are generated by V-band (50 – 75 GHz) and D-band (110 – 165 GHz) transceiver modules. The transmission signals are quasi-optically multiplexed for combined propagation through the TL. Results from the laboratory tests are obtained in air with a reflector target to simulate the plasma cutoff for establishing the baseline performance as an ideal reflectometer. The integrated TL tests are performed in a staged approach that evaluates the specific purpose of the component as well as the effect of each component on the reflectometer antenna pattern, power loss, and FMCW performance. A method for calibrating the TL length with the use of a special miter mirror is investigated. Encouraging results were obtained that suggest the calibration technique is feasible for the ITER implementation. A signal-to-noise estimate of the reflectometer measurement on ITER is predicted by incorporating the laboratory test results with calculations of the expected noise levels and signal losses caused by the plasma. Current projections suggest that, with some further optimization of the transceivers, the LFSR will meet or exceed the measurement requirements for ITER.

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Topics: Plasma Edge and First Wall Diagnostics

ps-LIBS diagnostics for tritium measurements in W

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During its operation, a thermonuclear reactor produces a huge flux of particles which interact with the Plasma Facing materials (PFCs) leading to D/T implantation. For operational and safety reasons, the tritium wall inventory has to be measured. The Laser-Induced Breakdown Spectroscopy (LIBS) diagnostics is particularly relevant to perform this in situ measurement. In LIBS diagnostics, the laser-matter interaction leads to the local ablation of the surface and the formation of a radially expanding plasma. Its temperature is sufficiently high to cause radiative emission whose spectroscopic analysis leads to the composition of the plasma, therefore to that of the wall in case of negligible matrix effects. We have proved several years ago that ns LIBS is irrelevant to measure the D/T concentration in metallic samples [1,2]. Therefore we decided to start in 2011 the development of a ps LIBS system.

We are currently working on the ps LIBS technique to measure the concentration of H, D or T in different samples on our PLEIADES (Plasmas by LasEr IrradiationS AnD their Experimental Studies) setup. This development essentially follows four directions.

(1) **Ablation studies**. Equipped with a 30 ps laser source, this setup provides results corresponding to low ablation rates of the order of 100 nm/pulse in typical conditions. According to the pulse energy, this ablation rate can be reduced and the depth analysis resolution can be increased.

(2) **Spectroscopic analysis**. This analysis is performed on the [200, 800] nm spectral range for W (for fusion), Al (as Be substitute) and Si (as benchmark) implanted with mainly D^+ ions produced by plasma discharges or ion beams. The analysis is performed all along the plasma lifetime in order to identify the better time window compatible with high excitation and low thermochemical non equilibrium.

(3) **Plasma modeling**. The spectroscopic analysis is supported by the modeling of the plasma from the laser-matter interaction to its final extinction, ~ 1 µs later. The ECHREM numerical code (Euler code for CHemically REactive Multi-component laser-induced plasmas) has been developed and is used to analyze the thermochemical non equilibrium possibly observed experimentally.

(4) **Double pulse technique**. In the case where the laser-induced plasma temperature is not high enough, an additional nanosecond laser pulse is focused on the plasma longitudinally to the sample and is absorbed by inverse Bremsstrahlung. Then a significant signal-to-noise ratio enhancement is obtained in spectra.

In this communication, details will be given on the current status of the development of the previous four points. Moreover, comparisons performed between implantations measured by LIBS on W, Al and Si samples and those obtained by NRA will be presented and discussed.

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In situ study of hydrogen isotopes retention with co-deposited layer in EAST tokamak by Laser-induced Breakdown Spectroscopy

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Hydrogen isotopes (deuterium and tritium) retention is one of the crucial issues for plasma facing components (PFCs) in tokamak devices like ITER. Monitoring of hydrogen isotopes retention in PFCs is a mandatory task in consideration of the safety during ITER operation. Developing quick and easy-to-use material characterization techniques is of importance for the measurement of retention subsequently to plasma-wall interactions (erosion, deposition and migration) in nuclear devices.

Laser-induced breakdown spectroscopy (LIBS) is a promising method and suitable for *in situ* evaluating the retained hydrogen isotopes from the first wall material without disassembling of PFCs to assure continuous operation of nuclear devices. We reported previously an *in situ* and remote LIBS system has been established for the measurement of fuel retention as well as after wall cleaning in EAST tokamak [1,2].

In this work, the hydrogen isotopes retention in the co-deposited layers between tokamak discharges and after wall clean activates is investigated using *in situ* LIBS. Results from LIBS measurements on the first wall showed that water vapor still retained on the surface of the first wall even after several days baking and wall cleaning. During the initial phase of operation, deuterium was hard to detect on the first wall before the lithium wall conditioning. However, with continuous lithium conditioning, the plasma performance was improved, deuterium and hydrogen became easier to measure in the lithium deposited layers, which means the lithium layers may absorb the hydrogen isotopes to improve the plasma performance, and the hydrogen isotopes retention were driven by co-deposition layers. The depth of lithium layer and impurities were investigated from laser pulse by pulse after lithium conditioning and plasma discharge. The results indicate that the thickness of lithium layer almost unchanged after a whole day of discharges, but the lithium co-deposited layer was contaminated by impurities from the migrations. This could be the reason for plasma could not survive after a whole day discharges and requires the routine wall cleaning. After lithium conditioning, no hydrogen isotopes and impurities could be measured by *in situ* LIBS in the fresh lithium layers. H/(H+D) ratio was also investigated in EAST tokamak operation by LIBS, the wall appeared to be saturated during discharge after the high-performance discharge such as 100s H-mode discharge. Investigations of *in situ* LIBS in EAST will help to predict and optimize the wall conditioning for EAST operation and demonstrate the potential using LIBS in ITER.[1]

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[1] This work was supported by the National Science Foundation of China (Nos. 11575243, 11375010, 11605238 and 11605023).

Diagnostic Setup for the Divertor Manipulator at Wendelstein 7-X

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The investigation of plasma wall interactions in terms of erosion processes, fuel retention and new plasma facing components exhibits a scientific gap as discharges in tokamaks and stellarators last only seconds up to a minute. Currently, linear plasma machines are bridging this gap, while Wendelstein 7-X will provide steady-state discharges of up to 30 minutes in its second operation phase (OP2), in which a divertor manipulator is envisaged to expose samples to reactor-relevant plasma fluxes and fluences. With a design following the limiter lock system at TEXTOR [1] and the DIM-II [2] at AUG, the divertor manipulator at W7-X aims to study plasma surface interactions ex-situ by daily/weekly probe exchange and in-situ by embedded diagnostics and an observation system.
The design, layout and capabilities of the embedded and observing diagnostics determine the accessible parameters for the in-situ investigation of the exposed probe materials and are therefore detailed in this contribution. The adoption of the endoscope system in an adjacent port determines spatial and temporal resolution of the surface observation, while the integration and operation of electrical probes in the manipulator head is based on the sourrounding plasma properties. These range from plasma edge measurements in the manipulator shaft to intersecting the magnetic island structure on the manipulator head. The possibility of employing a coaxial laser and observation system adding active surface analysis techniques will be assessed. To prevent local overheating and excess thermal stress, essential diagnostics for temperature control are identified and combined with first considerations of the manipulator head shaping, based on the field line tracing tool of the W7-X web service. The effects of the magnetic configurations on the manipulator head shape and accessible plasma regions will be investigated.

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A Non-Invasive Method of Measuring the Height of Liquid-Metal Surface Waves

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Liquid-metal plasma facing components (LM-PFC's) could provide fusion reactors with improved tritium breeding capabilities, enhanced power removal, and 'self-healing' interior surfaces that are immune to both radiation damage and thermal stress. During reactor operation, fast-moving, smooth-flowing LM-PFC surfaces are preferred since surface waves may cause non-uniform heating of the LM-PFC and splashing of liquid metal could upset or extinguish the plasma. However, surface waves and instabilities on LM-PFC's can be caused by a number of different factors including interactions with tokamak surfaces (e.g. diagnostic ports), magnetic transients, and interactions with the 'plasma wind'. Identifying the location and measuring the amplitude of liquid-metal waves during reactor operation is an important step towards minimizing and controlling them. Therefore, a non-invasive electromagnetic diagnostic has been developed to quantify localized surface waves in LM-PFC's. This low-cost diagnostic is installed beneath the substrate that the liquid-metal flows so it is insulated from thermal transients. This paper provides details on the design, construction, and operation of the new diagnostic. Experimental data is compared to numerical results.

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Implementation of implanted depth marker technique to study high-Z surfaces in EAST

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While many ex situ measurements of tokamak surfaces exist, developing a deeper understanding of the dynamics of erosion, redeposition, and fuel retention (including material mixing) in these surfaces will require in situ observation of changes. An Accelerator-Based In-Situ Materials Surveillance (AIMS) diagnostic was developed for this purpose and first used in Alcator C-Mod to demonstrate the use of in situ Ion Beam Analysis (IBA) to study divertor surfaces with shot-by-shot resolution. The original IBA techniques utilized nuclear reactions of a 900 keV deuteron beam with low-Z isotopes in or on the surface. This method is not applicable to studying the bulk erosion of high-Z divertor material (e.g. molybdenum and tungsten) because the Coulomb barrier precludes nuclear reactions between high-Z elements and a low or medium energy ion beam (typically used for IBA) [1].

In order to measure this high-Z erosion, a new method of IBA using particle-induced gamma emission (PIGE) and implanted depth markers is being developed. Depth markers are a well-established method of measuring erosion and deposition; however, depth markers are traditionally co-deposited in layers with the surface material, which limits study to thin surface layers that could delaminate, and otherwise not exhibit bulk properties. Implanting the marker enables the study of bulk material and doesn't restrict the studies to specially-manufactured samples.

The following ex situ experiments have been carried out to assess the viability of this in situ technique. Samples with implanted depth markers have been studied in various environments, including the Material and Plasma Evaluation System (MAPES) and campaign integrated samples in the Experimental Advanced Superconducting Tokamak (EAST), located on the outer and inner midplanes, respectively. These samples experienced different conditions including plasma exposure and lithium conditioning, allowing for measurement of both erosion and redeposition. Different implantation depths were tested for optimal performance (shallow and deep implantations, approximately 200 and 1500 nm deep). Additionally, temperature stability was studied by varying implantation temperatures from 300 to 700°C and exposing samples to temperatures from 120 to 1000°C for times from 1 to 24 hours. A synthetic diagnostic was developed to allow interpretation of the experimental data and to test the sensitivity, with initial studies showing a match between predicted and experimental results. Selected results from the EAST study along with results from ex situ analysis and simulation will be presented.

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Fluctuations at the divertor surface measured by elongated, 'rail' Langmuir probes and their relationship to upstream turbulence

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Experiments indicate that plasma fluctuations in the low field side scrape-off layer (SOL) are dominated by the radial propagation of 'blobs' [1], *i.e.*, high density, high temperature plasma filaments that are aligned with the local magnetic field. In the *far* SOL, these structures are known to carry a significant particle flux to the main-chamber wall and are involved in the formation of a density 'shoulder', which is observed at high core plasma density normalized to the Greenwald density [2]. The statistical properties of the fluctuations, as measured by probe and imaging diagnostics, are very well characterized and accurately reproduced by a stochastic model consisting of the superposition of uncorrelated pulses [3]. According to theory [4], the radial propagation velocity of a filament depends on its electrical 'connectedness' to the divertor through the sheath, with collisionality playing an important role. Supporting evidence for this is seen in experiments [5]. However, it remains to be determined if these same dynamics influence turbulence in the *near* SOL region, where electrical connection may be additionally affected by other physics, such as magnetic shear. For example, when the divertor electrical 'connectedness' is modified by N_2 seeding, the near SOL profiles are not affected [6].

In principle, fluctuation statistics measured by divertor and upstream probes could directly indicate if filaments are dynamically connected to the target. An initial look at data from C-Mod's standard embedded divertor Langmuir probes with a 'proud' geometry was inconclusive in this regard [7]. However, we have since found that, elongated flush-mounted 'rail' divertor Langmuir probes [8] may provide more reliable measurements of ion saturation current and floating potential fluctuations; they are found to closely track fluctuation statistics measured upstream in cases when the 'proud' probes do not. This paper reports results from two ongoing investigations, made possible by this new diagnostic: (1) a comparison of fluctuations measured by 'rail' and 'proud' probes and the physical mechanisms that may be responsible for the differences; (2) a comparison of divertor and upstream fluctuation statistics, examining the potential role that divertor collisionality may have in affecting upstream filament dynamics.

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Abstract Withdrawn

Elemental Analysis on Wendelstein 7-X Limiter and Divertor Tiles by Laser-Induced Breakdown Spectroscopy (LIBS)

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Plasma-wall interaction (PWI) processes, such as erosion, deposition and fuel retention on plasma-facing materials (PFMs), are key issues for the safe and reliable operation of long pulse nuclear fusion devices. Analysis and understanding of elemental distributions in PFMs of are vital for the study of the 3D PWI physical process in Wendelstein 7-X (W7-X) stellarator. Laser-induced breakdown spectroscopy (LIBS) is a well-established elemental composition analysis method as well as one of the most promising candidates for *in situ* first wall diagnosis of fusion devices [1].

In the previous work [2], LIBS was successfully employed to *ex situ* analysis limiter graphite tile of W7-X which was exposed in the initial operational phase (OP 1.1). Depth profiles of each element and the ratio of H and C atoms on the surface along the toroidal were investigated. However, the poloidal elemental distributions on the limiter tiles and the divertor tiles of OP1.2a are still need to be studied.

In this work, an upgraded coaxial LIBS setup which has the similar configuration to the planned *in situ* laser diagnosis system of W7-X is employed in the lab. A picosecond pulsed laser system with pulse duration of 35 ps and wavelength of 355 nm is used to ablate samples in an ultrahigh vacuum chamber. W7-X limiter tiles and divertor tiles are analyzed by the LIBS system with higher spatial resolution (<2 mm) to obtain distributions of H retention and impurity elements, such as Fe, O and Na. Both toroidal and poloidal elemental profiles on surface of limiters and divertors tiles are studied and compared to heat and particle flux footprints. Simultaneously, mass spectra of retention gas are compared to LIBS spectra by laser-induced ablation quadrupole mass spectrometry. Depth elemental profiles in deposition and erosion zones will also be obtained by secondary ion mass spectroscopy to compare to the LIBS result. Quantitative LIBS information will be achieved by the thermal desorption spectroscopy measurements on the samples.

The studies on the elemental mapping of the PFMs surface by LIBS are crucial to understand the 3D plasma physics on the PWI and prepare for the *in situ* LIBS system in the stellarator geometry for further operation phases.

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An experimental assessment of methods used to compute secondary electron emission yield for tungsten and molybdenum electrodes based on exposure to Alcator C-Mod scrape-off layer plasmas

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Plasma potentials computed from Langmuir probe data rely on a method to account for secondary electron emission (SEE) from the electrodes. However, significant variations exist among published models for SEE and the associated experimental parameters used to evaluate them. As a means to critically assess SEE computation methods, a Langmuir-Mach probe head with two tungsten and two molybdenum electrodes was exposed to Alcator C-Mod boundary plasmas where electron temperatures exceed 50 eV and secondary electron emission is significant. Under these conditions, the computed SEE from molybdenum and tungsten can be substantially different, depending on the method used. Nevertheless, since the electrodes are exposed to identical plasma conditions, deduced plasma potentials should be the same regardless of electrode material. This self-consistency test is used to critically assess SEE evaluation methods.

Using a servomotor-driven fast scanning Mach probe [1] equipped with a Mirror Langmuir Probe bias system [2], the molybdenum and tungsten electrodes were scanned across the scrape-off layer of Alcator C-Mod, sampling electron temperatures up to 75 eV and plasma densities exceeding 10^{20} m^{-3}. Prior to the experiment, the electrodes were exposed to conditioning plasmas and elevated to high temperatures to ensure that oxide layers and other potential surface contaminants were removed. Plasma potential profiles were deduced for each electrode based on six different SEE evaluation methods. These six methods result from using two different model formulas (Sternglass, Young-Dekker) evaluated using three different experimental data sets (Kollath, Bronstein, and Walker).

Substantial differences in the deduced plasma potentials are seen, depending on evaluation method and electrode material exposed. Of the six methods used to compute SEE, only two are found to produce self-consistent results: the Sternglass model evaluated with Bronstein experimental parameters and the Young-Dekker model evaluated with Bronstein experimental parameters. In contrast, the method previously used for C-Mod data analysis (Sternglass model with Kollath parameters) was found to be inconsistent. We have since adopted Young-Dekker-Bronstein as the preferred method. An important consequence is that computed values of plasma potential drop across the SOL, electric field and E×B flow shear near the LCFS in Alcator C-Mod have changed substantially (~50%) compared to what had been reported previously.

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First Mirror Test in JET for ITER: complete overview after three ILW campaigns

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Metallic so-called *first mirrors* will be essential components of all optical spectroscopy and imaging systems in next-step devices. Therefore, their performance is crucial for reliable plasma diagnosis and operation. The First Mirror Test (FMT) for ITER has been carried out in JET [1-3] first in the presence of carbon wall and then during all three campaigns with the ITER-like wall (JET-ILW). The aim of this work is to provide an overview of results obtained for mirrors exposed during: (i) the third ILW campaign (ILW3, 2015-2016, 23.6 h plasma) and (ii) all three campaigns, i.e. ILW1 to ILW3: 2011-2016, 62 h in total. This is the first report on erosion-deposition in the shadowed zone of the divertor for the entire ILW operation.

Nine cassettes with 25 polycrystalline Mo mirrors were retrieved: one with five samples from the main chamber (ILW-3 only), two sets with 10 mirrors each from the divertor. One set was facing plasma for 24 h and the other one for 62 h. The examinations performed by means of optical methods for total and diffuse reflectivity determination in the range 400-1600 and 300-2400 nm (two systems used), microscopy (optical, atomic force and electron including EDS) and ion beam techniques (RBS, NRA, HIERDA) have brought a number of key results.

(a) Main chamber wall. The total reflectivity of all mirrors has decreased by 2-3% from the initial value. All of them are coated by a very thin co-deposit (5-15 nm) containing D, Be, C and O. This affected the optically active layer (15-20 nm on Mo) thus leading to the increase of diffuse reflectivity. Neither W nor N have been found on the surface. There are no differences between mirrors exposed in standard and baffled channels of the cassette.
(b) All mirrors from the divertor (inner, outer, base under the bulk W tile) lost reflectivity by 20-80%. This result confirms earlier findings and could be expected, but there are significant differences in the surface state dependent on the location and exposure time.
(c) The thickest layers composed mainly of Be are in the outer divertor: 850 nm after ILW1-3.
(d) The measured layers thickness is not directly proportional to the exposure time: 50-60% may be attributed to the last campaign when comparing results for ILW3 and ILW1-3.
(e) Only in a few cases, on mirrors located at the cassette mouse, flaking of deposits occurred.
(f) Nitrogen, tungsten and nickel are on all divertor mirrors. The highest N and W concentrations are in the inner divertor: N reaches 1×10^{17} cm^{-2}, W is up to 3.0×10^{16} cm^{-2}, while the content of Ni is the greatest in the outer: 2.5×10^{17} cm^{-2}.
(g) Oxygen-18 content is up to 7×10^{15} cm^{-2} (first report) and nitrogen-15 is over 1.8×10^{16} cm^{-2}.

The obtained results in this work will be compared with earlier data and the next steps in the FMT programme will be presented, i.e. impact of plasma or ion irradiation on pre-damaged mirrors. The implications for cleaning of mirrors will also be discussed.

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Estimation of three-dimensional structure on peripheral fluctuation using fast camera images and magnetic field calculation in Heliotron J

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In Heliotron J, measurement of peripheral plasma fluctuation/turbulence has been performed by using fast camera over a decade [1]. Many images are provided as two-dimensional information with time sequence by fast camera. They revealed the topological shape of fluctuation/turbulence and its complex behaviour. From experiences of the fast camera measurement in Heliotron J, almost peripheral fluctuation has a space structure elongated along the magnetic field lines. The filament structure along the magnetic field lines should be the basis shape of many kinds of fluctuation in Heliotron J (and perhaps in the other experimental devices). This is quite reasonable since the motions of electrons and/or ions along the magnetic field lines are very faster than those of vertical direction. Thus, normally peripheral plasma fluctuation is observed as bright or dark regions with a filamentary structure along the magnetic field lines.

The idea of this study is very simple. If the three-dimensional structure is obtained from an image with a two-dimensional structure, it will make a big jump to understand the edge plasma fluctuation/turbulence. For example, the speed of the apparent poloidal flow can be calculated using the accurate position of bright region of fluctuation if fluctuation moves with plasma. It will be very helpful for H-mode physics. Also, to identify the position of the plasma-wall interaction due to big fluctuation such as type-I ELM will be important for inner vessel damage in the future.

The curves of the magnetic field lines in the fast camera images were calculated from the magnetic field calculation code (Nakasuga code). As a result each magnetic field line is different shape in the camera image. Therefore, in principle the filament structure in the camera image can be identified by the magnetic field line.

Recent fast-camera measurements for pellet injection experiments show the existence of blob-like motion of the bright filamentary structures. By mapping the suitable magnetic field lines to the fast-camera images, three-dimensional information on the peripheral plasma fluctuation can be obtained.

This paper reports the challenge of extraction three-dimensional information on the peripheral plasma fluctuation from two-dimensional images with help of the geometry of the magnetic field lines in Heliotron J. Also the problems to be resolved in this scheme to increase accuracy are discussed.

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Depth resolved analysis of hydrogen in W7-X graphite components using Laser-Induced Ablation-Quadrupole Mass Spectrometry (LIA-QMS)

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Monitoring the fuel (hydrogen) and impurity content of first wall components in fusion devices like Wendelstein 7-X (W7-X [1]) is essential for a better understanding of plasma-wall interaction processes like erosion, deposition, retention and outgassing. Laser-based analysis methods like Laser-Induced Breakdown Spectroscopy (LIBS) and Laser-Induced Desorption (LID) offer preparation-free sample analysis [2] and are suitable as in-situ as well as ex-situ diagnostics, but are complex non-equilibrium processes, which impedes first principle quantitative measurements of unknown materials.
We present a novel method for depth-resolved analysis of volatile sample composites, using residual gas analysis after picosecond laser-induced ablation to release all co-deposited and implanted hydrogen and enhance the depth resolution. For this purpose a third harmonic (λ = 355 nm) of a Nd:YVO_4-laser with a pulse duration of τ_p = 35 ps is used. This results in a depth resolution in the order of 100 nm, as the ablation rate is in the same order of magnitude as the heat penetration depth. A spot size diameter on the sample of 1 mm with laser pulse energies of E_p = 40 mJ avoid significant matrix mixing effects at the edge of the laser-induced crater in the sample but enables the simultaneous and complementary LIBS measurement [3].
W7-X operated hydrogen plasmas in limiter and divertor configuration with graphite first wall components. The usage of graphite is leading to a complex erosion and deposition pattern with net erosion zones and implanted hydrogen as well as co-deposition zones with co-deposited hydrogen as identified post-mortem on limiters of the initial campaign [4]. LIA-QMS is one of a limited number of post-mortem analysis techniques which allows quantitative depth-resolved information of the hydrogen content in graphite components. We present the hydrogen depth distribution of a series of poloidal and toroidal locations on graphite limiter tiles of W7-X exposed in the initial campaign. The integrated hydrogen content for over 300 s of plasma exposure [5] is compared with corresponding thermal desorption spectra providing an independent measure. The LIA-QMS results are compared to simulations for the heat flux on the first wall during plasma operation [1], showing different fuel content in erosion- and deposition–dominated zones of the limiter.
Moreover, divertor graphite components exposed in the first divertor campaign will be studied in the same manner along the poloidal direction passing zones of high and low ion fluence as well as surface temperature.

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Comparison of LIBS results on ITER-relevant samples obtained by nanosecond and picosecond lasers

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The ITER strategy foresees applying laser induced breakdown spectroscopy (LIBS) for quantitative in situ diagnostics of fuel retention in the first walls during maintenance breaks. To this end, reliable detection of spectral lines of hydrogen isotopes, which in a laser-created plasma are partly overlapped due to the Stark effect, is required. Long delay times between the laser pulse and the beginning of data acquisition, t_d, will help in reducing the overlap due to the concentration of charge carriers in plasma decreasing with t_d. However, simultaneously the signal-to-noise ratio decreases. This contradiction sets the need to find alternative ways for reliable measurements of the line intensities.
Most fusion-related LIBS studies are carried out using lasers with nanosecond pulse durations. However, some studies [1] indicate that using lasers with a shorter pulse duration would be advantageous for determining the surface characteristics and retained fuel depth profiles of the analyzed samples. The aim of the present study is to compare the effect of different pulse durations on LIBS results of ITER-relevant samples.
Samples with D-doped W/Al coatings (Al is used as a proxy of Be) on Mo were tested. Two Nd:YAG lasers with pulses of different durations (first from Ekspla with 0.15 ns and the second from Quantel with 8 ns pulses) generating at 1064 nm, were used. In both cases, the beam was directed normally onto the sample surface. Argon at a pressure of 300 Pa was used as the background gas. To reach optimal conditions for the detection of hydrogen isotopes, t_d as well as the laser pulse energy were varied.
For both lasers the same setup was used for measuring the emission spectra. Spectrometers Mechelle 5000 for recording a broad range (250-850 nm) and MDR-23 for recording lines of hydrogen isotopes, positioned at 45° with respect to the laser beam. It was found that in case of picosecond laser

- Higher signal-to-noise ratio and slightly narrower spectral lines allowed a more reliable curve-fitting of lines of hydrogen isotopes that helped distinguish them.
- At a fixed fluence, single-shot spectra allowed to build more detailed deuterium depth profiles.

The LIBS results qualitatively matched with those obtained by Secondary Ion Mass Spectrometry.

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Abstract Withdrawn

Measurement of the ion species dependence of the intrinsic edge rotation in spherical tokamak QUEST

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A non-inductive method to initiate and sustain spherical tokamak (ST) plasmas using electron cyclotron heating (ECH) has been implemented in several devices currently under operation. The physics of the ECH ST-plasmas is, however, still not fully understood, and the diagnostics of basic plasma parameters and the construction of models describing plasma dynamics are important issues to realize a discharge with a better performance. Among several key parameters of interest, the intrinsic toroidal rotation has been intensively studied at the initial and steady-state phases of the discharges in a spherical tokamak QUEST.

At the initial phase of the ECH discharge, a simple toroidal plasma in a cylindrical geometry is produced at the resonance layer. The toroidal velocity of this plasma has been measured using multiple viewing chords spectroscopy. Complicated changes in the magnitude and direction of the velocity with the vertical magnetic field strength were found (Ref.[1] and succeeding studies), and the driving mechanism could be attributed to a balance among several factors such as the drifts, Pfirsh-Schlüter (PS) regime, pre-sheath electric field, and ion pressure gradient.

For plasmas in closed-field equilibriums, the toroidal velocity in the edge region measured by the spectroscopy significantly depends on the magnetic mirror ratio [1], suggesting a relation between the toroidal rotation and the selective loss of ions in a specific region of the velocity space. Also, a transient response of the ion flow velocity accompanying a strong gas puffing was measured using Mach probe at the outer scrape-off layer [2, 3], and the driving force of the toroidal rotation was attributed to the combination of the PS regime and pressure-gradient force.

Stimulated by these experimental findings, we have developed a new passive spectroscopic system that enables an accurate measurement of the radially resolved toroidal velocity on the midplane. An advantage of passive spectroscopy is its applicability to basically any ion species, and we can check the validity of physical models by confirming the ion species dependence. We have applied the diagnostic system to CIII and OII emission lines in the outer edge region and compared the measured velocities with predictions by plausible models.

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Shadowing Effects in Simulated Alcator C-Mod Gas Puff Imaging Data[1]

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The intense interest in predicting the divertor heat flux width for ITER is driving the search for an improved understanding of edge plasma turbulence. The gas puff imaging (GPI) technique [1] provides spatially and temporally resolved data that will likely be a critical ingredient in this search, particularly for validating turbulence simulation codes.
The neutral density field created by the GPI gas puff is, ideally, localized in the direction parallel to the magnetic field, yet varying slowly in the perpendicular directions and unaffected by plasma turbulence. In practice, the neutral gas cloud has a finite toroidal width and can indeed be altered by passing turbulent plasma structures. The principal concern with the latter is "shadowing" in which the plasma turbulence imprints its structure on the neutral density field, giving rise to spatial correlations not present in the underlying plasma turbulence. The shadowing associated with a prescribed, steady state, spatial perturbation in the electron density was examined in [2]. More recently, Wersal and Ricci [3] developed a synthetic GPI diagnostic for a fluid plasma turbulence code and demonstrated that shadowing can significantly impact the GPI light emission for penetration depths greater than one mean free path.
Shadowing is of particular concern for the GPI system on Alcator C-Mod given its high edge plasma densities and associated short neutral mean free paths, especially for helium gas puffs. We will quantify this effect using a DEGAS 2-based synthetic GPI diagnostic applied to an XGC1 gyrokinetic turbulence simulation of an Alcator C-Mod plasma; this was one of several runs performed to validate XGC1's ability to predict the divertor heat flux width [4]. A baseline result will be obtained by computing the GPI signal associated with the fixed, neutral density field obtained by puffing gas into an axisymmetric plasma. This will be compared with a more realistic, time dependent transport simulation of the puffed neutral atoms and molecules through the 3-D XGC1 plasma. In both cases, the synthetic GPI will be applied as a post-processor to XGC1 so that the impact of the gas puff on the turbulence will not accounted for. The neutral mean free path will be varied by repeating the exercise with both deuterium and helium gas puffs.

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Abstract Withdrawn

Abstract Withdrawn

Simultaneous vacuum UV and broadband UV-NIR spectroscopy for improvement of LIBS characterization of LiSn alloys

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In fusion reactors, the hot plasma is known to erode the reactor surface and subsequently the fuel (deuterium, tritium) is trapped within the walls. The concept of liquid walls offers advantages over the conventional Beryllium/Tungsten based materials of the wall/divertor, i.e. no nanocracks, release of impurities by heating. LiSn based alloys form good candidates due to favourable properties such as low melting point, higher thermal conductivity and heat capacity.

The ability of LiSn to retain D, N and O is investigated by means of Laser Induced Breakdown Spectroscopy (LIBS), a spark formed on the material surface. In addition to conventional UV-NIR range, the simultaneous detection in VUV range not only makes for observations of several Sn II and Sn III characteristic spectral lines, but also improves the precision on the data sets. These give rise to more reliable evaluation of the electron temperature using Saha-Boltzmann plots, thus improving the quantification of elements using a calibration-free method, which is already well established for quantitative analysis of metallic alloys [1,2].

Due to the peculiarity of surface heating by the laser, only the first shots were considered, as to prevent the influence of desorption of deuterium. Two spectrometers were used, broadband echelle type spectrometer (230–950 nm, ME5000, Andor) equipped with iCCD camera (iStar DH743, Andor) and the McPherson spectrometer (114 – 295 nm), coupled with iCCD camera (iStar DH740, Andor). Laser spark was generating by Nd:YAG laser (CFR, Quantel) operating at 266 nm. The electron density was determined from the Stark broadening of the hydrogen-alpha spectral line at 656.3 nm.

The measurements were done in Ar atmosphere at pressures 1 Torr, 10 Torr and 100 Torr. Gate delay and equal gate width were set to 100 ns, 300 ns and 500 ns resp. The electron temperature varies in the range from 1 eV to 1.5 eV. Chosen conditions permits us to detect and quantify retained N (*174.2-174.5* nm), O (*130.2* nm) and H/D (*121.5* nm) via interference free strong lines and Li/Sn elemental composition using numerous Sn II-III lines in VUV range.

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Abstract Withdrawn

Toroidal variation of the strike point in DIII-D*

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We report measurements of a 5mm toroidal variation of the outer strike point radial position using an array of three identical Langmuir probes distributed at 90 degree intervals around the torus (92, 180, 272 degrees). The strike point radial location is determined from the profiles of floating potential (Vf) and ion saturation current (Jsat) measured by the three 6 mm diameter domed Langmuir probes as the strike point is swept radially on a horizontal tile surface just outside of the upper small angle slot (SAS1) divertor. The steep radial gradient between the private flux region and outer scrape-off layer is used as a marker to determine the strike point location. Based on the three probe measurements, the strike point variation is consistent with the description of the strike point as a circle shifted 5 mm outwards from the torus centerline towards 225 degrees. This is consistent with direct in-vessel imaging of visual changes on the surface of graphite tiles resulting from plasma interactions during a recent vent of DIII-D. The toroidal variation of the strike point position and scrape-off layer plasma can affect the alignment with the SAS divertor and can therefore affect the ability of the slot geometry to achieve toroidally uniform divertor detachment [1] at a lower core plasma density than we can normally achieve. The bottom of the SAS divertor slot is 21 mm wide but the width is effectively about three times less because of misalignments between the slot hardware and the magnetic flux surfaces including error fields. The probe and slot locations are determined from detailed initial spatial measurements of the slot tiles using an in-vessel digital positioning instrument with less 0.1 mm error. The probe profiles are determined relative to the EFIT [2] equilibrium strike point. We find the EFIT strike point is close to the probe-determined strike point location near 180 degrees. These strike point measurements will be compared to previous error field measurements [3] and discussed with regard to the slot divertor alignment and its impact on detachment studies.

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Overview of first mirror cleaning using radio frequency plasma in EAST

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In-situ plasma cleaning has been regarded as the most promising method to recover the reflectivity of the contaminated first mirror (FM), which is very critical to ensure the accuracy and effectiveness of related diagnostic signal in ITER [1, 2]. First mirror cleaning experiments in EAST tokamak have been performed and radio frequency (RF) plasma cleaning has been proved to be an effective method. The ITER edge Thomson scattering (ETS) mock-up mirror using RF plasma has been successfully cleaned in-situ for the first time in a tokamak, which is promising for plasma cleaning baseline scenario of ITER.

Firstly, stainless steel (SS) mirror with carbon deposits due to tokamak exposure was effectively removed by argon (Ar) plasma in the laboratory. The reflectivity of the cleaned mirror was recovered by up to 90%. Then, the Ar plasma cleaning was applied for the non-plane large-size (80 × 300 mm^2) mirror used for EAST charge exchange recombination spectroscopy (CXRS) diagnostic. The inhomogeneous deposition with particles size of up to tens of micrometer was uniformly removed using RF plasma. More than 90% recovery of the reflectivity and negligible difference of the recovered reflectivity were obtained.

The FM in-situ cleaning system was developed using material and plasma evaluation system (MAPES) in EAST. The cleaning experiments for the ITER ETS mock-up (200 × 300 mm^2) with 5 small Mo mirror inserts on SS substrate have been carried out to address the application of the in-situ RF plasma cleaning in a tokamak. The neon and argon plasma can be successfully generated and maintained for several hours on the ETS mock-up with a 1.7 T magnetic field in the main chamber of EAST tokamak. For an inclined magnetic field angle of 20° and 5°, the achieved absolute self-bias was lower by factors of 2 and 10 respectively than that without the magnetic field. The 10 nm Al_2O_3-coating used as the substitute of Be deposits, was successfully cleaned for half an hour with a self-bias of -20 V and -80 V at 5 and 20 degrees, which was at least 40 times faster than laboratory experiments without magnetic field. Due to asymmetric sputtering caused by the local variation of ion flux, the central inset mirrors systematically exhibited higher diffuse reflectivity and surface roughness. Difference on the redeposition of the sputtered ETS mock-up materials was found for both cases as the iron (Fe) was only detected on the mirrors at 5°. Besides, the cleaning at 20° was slightly faster than that at 5° which is consistent with the higher absolute self-bias achieved at 20°. The total reflectivity was completely recovered for all the mirrors except the most edge one for the 5° case.

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Impurity radiation characterization for the first limiter and divertor plasmas of W7-X

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W7-X started operation with five inboard carbon-limiters at the end of 2015. Recently, the second operation phase has begun with ten test divertor units to explore the so-called island divertor concept. This paper compares radiation features of the first limiter and divertor hydrogen plasmas in W7-X.

In limiter plasmas, low-Z impurities like carbon and oxygen are identified to be the major radiators according to spectroscopic measurements. In quasi-stationary plasmas the total radiative power loss fractions measured by two bolometer cameras typically range from 20% to 60%, depending on the absorbed heating power and the plasma density. Radiation usually originates at the edge with $r \sim 0.8$. Strong poloidal asymmetries in radiation have been observed when plasmas approach a radiative density limit. In comparison to the limiter configuration, where the SOL is formed by smooth flux surfaces, divertor configurations are bounded by low-order magnetic island chains with fine structures, which are expected to have strong impacts on the radiation distribution of impurities. Intensive radiation has been observed near and outside the separatrix. This paper will demonstrate how existing bolometer channels, which have a spatial resolution of ~3 cm at the SOL, are used to estimate the radiation from the edge island chains of ~8 cm radial thickness under different plasma parameter and configuration conditions. Some experimental results will be compared with EMC3-Eirene simulations.

In-situ diagnosis of the first wall in nuclear fusion devices by Laser-Induced Breakdown Spectroscopy combined with Laser Speckle Interferometry

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Plasma–Wall Interaction (PWI) poses a critical issue in long pulse and steady-state operation fusion devices. The diagnosis of fuel retention and wall erosion as well as re-deposition is very important for understanding PWI and improving the performance of H-mode with long pulse operation. Currently, laser-based methods are only known possibility to obtain safety relevant nuclear inventory data at runtime. Laser-Induced Breakdown Spectroscopy (LIBS) has been developed for *in-situ* and online monitoring PWI such as fuel retention and impurity deposition in EAST [1, 2]. Recently, a remote Laser Speckle Interferometry (LSI) measurement at a distance of 3 m from the divertor tiles taken from EAST has been carried out in the authors' laboratory to simulate real detection condition on EAST [3].

In this work, a LIBS system, built in the same optical configuration as the *in-situ* LIBS system in EAST, has been used to investigate the fuel retention and impurity deposition on various materials of first wall from the high-field side of EAST after the whole 2017 campaign. A remote LSI measurement was developed for real time monitoring and reconstructing the crater produced by the LIBS measurements at a spacing of 3 m. Preliminary LIBS results showed that impurity elements (C, Si, Mo) were observed in the first four LIBS shots. However, deuterium was detected only in first LIBS shot. The morphology of the craters was characterized by LSI technique. The maximum depth, cross-section profile as well as surface topology for each crater were evaluated by analyzing the LSI data using a numerical model based on phase unwrapping algorithm, which was developed by the authors' group. The LSI results showed that the topological structure of laser produced craters was very complex due to the non-homogeneity of the laser beam spatial intensity profile and multi-physical coupling processes in the laser-matter interaction. In order to calibrate the measurements from LIBS and LSI, the craters were characterized by Scanning Electron Microscopy (SEM) combined with Energy Dispersive X-Ray Analysis (EDX) and Profilometer. The ablation depth per shot was determined to be ~ 60 nm by Profilometer. The maximum depth of craters was determined to be 100±30 nm by the LSI, which agreed well with the profilometer measurements. The LSI technique has demonstrated the ability to monitor the morphology evolution of the first wall at running time. These results indicate the potential application of LIBS combined with LSI methods to monitor the fuel retention and first wall erosion as well as re-deposition during PWI process in modern fusion devises.

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Plasma Exhaust and Plasma Material Interactions for Fusion reactors

Micro- and macro- elastic properties of tungsten fiber-reinforced tungsten composites probed by nano-indentation and laser ultrasonics

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Tungsten fiber tungsten composites (W_f/W) are presently being developed in the EU as next generation W-materials for plasma facing components. They possess pseudo-ductility and can overcome the limitations caused by the inherent brittleness of pure-W. If W_f/W composites are to be used as plasma facing materials, the effect of hydrogen plasma exposure on the mechanical properties need to be clarified. The macroscopic mechanical properties of composites depend critically on the microscopic interplay of the matrix, interface, and fiber. Hydrogen trapping at the microscopic scale will likely affect such interplay in W_f/W composites. Therefore, it is desirable to first clarify how micro level properties scale up to describe the overall bulk behavior of the bulk composite, such that the effects of hydrogen inclusion on mechanical properties can be elucidated.

In this research we characterized the elastic properties of W_f/W composites on the micro- and macro- scale by combined nano-indentation and laser ultrasonic method, respectively. Multi-fiber W_f/W samples, varying in fiber volume% (20-60%), manufactured by Spark Plasma Sintering (SPS) were studied. The fibers were distributed homogeneously in random orientations as seen from microscopy. Young's modulus was determined by nano-indentation hardness measurements of the W-matrix and W-fibers separately. The values were combined by simple rule of mixture, and compared to bulk elastic properties obtained from laser ultrasonics method [1]. We find that Young's modulus determined from the two methods agree within experimental error for fiber volume% of 30 or 40% However at fiber volume% ≥50%, the laser ultrasonic values are smaller than nano-indentation values. This difference may be due to wave scattering effects with increasing fiber%. Alternatively, it may indicate real loss in bulk strength. Microscopy shows that with increasing fiber%, there are increasing regions where no matrix is present between fibers (i.e. fiber to fiber contact only). We further compare such results against hydrogen-exposed samples to clarify the effects of hydrogen inclusion on the mechanical properties of W_f/W composites.

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First results of nano-structured substrates for a stable liquid metal plasma material interface under long pulse conditions.

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HIDRA is a long pulse toroidal plasma device [1] at the University of Illinois which is almost exclusively used to study the intimate relationship between the plasma interacting with surfaces of different materials and development of PFC technologies [2,3]. A Material Analysis Tool (HIDRA-MAT) has been built based on the successful Material Analysis and Particle Probe (MAPP) which is currently used on NSTX-U at PPPL [4,5]. This is an *in-situ* material diagnostic probe, meaning that all analysis can be done without breaking vacuum. This allows surface changes to be studied in real-time. HIDRA-MAT eventually will consist of several *in-situ* diagnostics including Langmuir probes (LP), Thermal Desorption Spectroscopy (TDS), X-Ray Photo Spectroscopy (XPS) and Ion Scattering Spectroscopy (ISS) [6], however currently only TDS is used. This paper will outline the HIDRA-MAT diagnostic which will include the design and implementation as well as initial results in characterizing and understanding adaptive nano-structured materials being developed at UIUC. A unique way of using a Residual gas analyser to distinguish between hydrogenic species and helium will also be discussed in the context of the diagnostic. The combination of HIDRA's unique long pulse capabilities and HIDRA-MAT's *in-situ* material analyse will facilitate a relevant, comprehensive study of the material evolution of various PFC systems in a prototypical confinement device over long exposure times (up to 60 minutes). A proposed material that will be tested in HIDRA-MAT is a low-Z/high-Z hybrid system consisting of a liquid metal (Li or SnLi) within a porous tungsten substrate developed by the RSSEL group at the University of Illinois. The progression of the hydrogen retention capabilities of such a system will be presented, and compared to more traditional PFC materials. Additionally, He entrainment within these systems are examined. One utility of these proposed systems is impurity gettering/release, and so the effects of glow discharges on cleaning these surfaces is also presented here. These experiments are supplemented by auxiliary experiments done in the IGNIS facility of the RSSEL group, where specific experimental parameters are replicated and their effects are observed with *in-operando* surface analysis techniques, such a XPS.

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Reduced D trapping by plasma-implanted He nanobubbles in radiation damaged tungsten

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Previous study results [1] show that a plasma-implanted helium nanobubble layer significantly reduces deuterium (D) retention by acting as D diffusion barrier in undamaged commercial ITER grade tungsten (W). In this paper we show evidence that this phenomena can survive displacement damage. Here, a He plasma exposure (sample temperature: 643 K, ion flux: 10^{22} m^{-2} s^{-1} at 100eV, fluence: 10^{25} m^{-2}) pre-treatment was performed to create a thin He nanobubble layer in the first ~30 nm of ITER grade tungsten samples. Samples were then irradiated by 5 MeV Cu ions at room temperature to create 0.001 to 0.1dpa with peak damage rates occurring about 860 nm below the sample surface. Samples without He plasma exposure pre-treatment were also irradiated by 5 MeV Cu ions to provide a controlled baseline. All samples were subsequently exposed to D plasma at 373K to a fluence of 10^{24} m^{-2}. Transmission electron microscope (TEM) was used to investigate the He bubbles' size and density in the samples. Nuclear reaction analysis (NRA) was performed to evaluate the D distribution profile and retention inventory to a depth of 2.5 μm. NRA results show that across a range of peak dpa ranging from 0.001 to 0.1 dpa, D retention inventory in the samples with He plasma exposure pre-treatment is reduced by a factor of 47.7% to 57.3% compared to samples without He plasma exposure pre-treatment. Thermal desorption spectroscopy (TDS) is used to measure the total D inventory retained in the samples and confirm these NRA results. The results suggest that the plasma-implanted He nanobubble layer can survive radiation damage and still function to reduce D diffusion and retention in tungsten-based plasma facing components.

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Abstract Withdrawn

A novel setup for the study of ammonia production from H_2/N_2 plasmas on tungsten surfaces

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Impurity gas seeding is required in ITER to reduce the power load onto the divertor target plates to values compatible with the divertor power handling capabilities. Specifically, nitrogen (N_2) is the preferred seeding species because of its favourable radiative properties as well as its apparent beneficial effect on plasma confinement [1,2]. However, once dissociated, nitrogen molecules chemically react with hydrogen (H) and its isotopes to form ammonia isotopologues. This formed ammonia could be a serious concern for ITER operation and maintenance. Consequently, it is important to understand the mechanism of ammonia formation and investigate the parameters influencing the production.
As shown in previous work [3], ammonia molecules highly stick on metal surfaces and specifically on stainless steel (SS) surface. As a consequence, measuring the production of ammonia in a standard stainless steel vacuum system would affect the accuracy of the results. Therefore, a new experimental setup has been developed to quantify the production of ammonia from the H_2/N_2 plasma. This setup consists of a cylindrical (metal free) quartz tube of 35(31) mm outer (inner) diameter and 1405 mm length connected to a waveguide surfatron plasma source (350 mm length). The plasma is created in the tube through a matching network by a 13.56 MHz radio frequency (RF) generator at 100 W. Inside the tube a 30×10 cm^2 tungsten foil is placed and heated with a three zones controlled temperature (250 mm) furnace up to a maximum temperature of 1300°C. N_2 and H_2 gases are introduced into the reactor tube at fixed partial pressures keeping their ratio constant to 1/9. On the other side of the quartz tube, a quadrupole mass spectrometer (SRS Residual Gas Analyser RGA200) is connected through a 2 mm diameter pinhole. The RGA is used to quantify the amount of produced ammonia. The total pressure during operation, measured by a Penning gauge, is kept around 10^{-5} mbar in the RGA chamber and 10^{-2} mbar in the quartz tube.
For the deconvolution of the measured spectra, a code that takes into account the different compound cracking patterns (CP) and calibration factors (CF) is employed. In fact, the CF for N_2, H_2 and NH_3 as well as for He or Ne are measured by introducing the pure gas and correlating the RGA peak intensity to the partial pressure measured by a baratron while the CPs are directly measured from the RGA mass spectra fragments of the pure gas.
Using this newly built setup and the deconvolution code, ammonia production from the plasma phase and on the surface will be presented. Furthermore, the effect of surface temperature and the impact of He or Ne addition onto the ammonia formation will be shown.

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Density Functional Theory Study of Hydrogen Interaction with Standoff Volume of Helium Bubbles Above Various Surfaces in Tungsten

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Plasma facing components in a fusion reactor are subject to high ion and heat fluxes which ultimately results in the formation of sub-surface gas bubbles containing hydrogen and helium in the tungsten divertor, which can produce substantial surface roughening as well as influence hydrogen retention and recycling behaviour. Consequently, it is critical to understand the underlying synergies between hydrogen and helium in this near-surface region. It has been shown in prior Molecular Dynamics (MD) studies [1] that hydrogen prefers to accumulate along the periphery of the helium bubbles at a region of low electron density. To this end, MD and Density Functional Theory (DFT) simulations have been used to study the interaction energies of hydrogen with the standoff distance created by the helium-tungsten repulsion. With this study, we hope to better understand the potential energy landscape around subsurface hydrogen in tungsten around the network of helium bubbles that inevitably form in reactor-like conditions. Such a study will provide an understanding of the synergistic effects between hydrogen and helium just below the surface of tungsten.

DFT calculations were performed using the Vienna ab initio Simulation Package (VASP), with exchange-correlation interactions modelled by GGA-PBE functional and plane wave cut off at 350 eV. The Brillouin-zone integration was performed within the gamma-point centered scheme using a (3x5x1) mesh. The periphery of the bubbles has been approximated by a planar geometry with a layer of helium in between two layers of tungsten. The effect of surface orientation was also considered. The initial DFT setup was taken from similar calculations performed with MD using the Juslin W-H potential with subsequent quenching to 0 K. Our results indicate that the standoff distance induced from the strong He-W repulsion creates a region of low electron density that results in high binding for the hydrogen at the periphery of these bubbles of about 2 eV.

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Nanosecond laser pulses for high heat fluxes tests on various tungsten materials under ITER-relevant conditions

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In this work, a numerical and experimental investigation of the thermal effects induced on various tungsten (W) materials (i.e. bulk W and nanostructured coatings) by nanosecond laser irradiation is proposed.
Due to plasma-wall interactions, the full W divertor of ITER will be subjected to extensive erosion, with consequent formation of thick co/re-deposits with complex and hardly predictable morphology, structure and composition. Because of the extreme thermal loads delivered during plasma instabilities (e.g. ELMs), both pristine W plasma facing components and co/re-deposited layers will suffer from severe thermal effects, which include recrystallization, surface melting, droplets formation and ejection, cracks formation and delamination. Since all these effects can be severely affected by the specific structure of the material, they must be consistently investigated.
In this framework, electron beams and millisecond lasers are routinely used at the laboratory-scale for testing plasma facing components under ITER-relevant high heat fluxes, exploiting pulses with duration of the same order of the ones of plasma transient events (i.e. milliseconds) [1, 2]. Thermal effects induced by these different sources can be consistently compared in terms of the heat flux factor (HFF), which is commonly exploited for photons, electrons and ions sources with constant temporal pulse shape in the millisecond regime.

Here, we propose to exploit nanosecond laser beams as an alternative, compact, easily accessible and cost-effective irradiation technique for testing plasma facing materials (both in the bulk and in the coating form) under ITER-relevant conditions. To this purpose, we firstly extend the definition of HFF to irradiation sources in the nanosecond regime and having a non-constant (i.e. gaussian) temporal profile. Then, we experimentally validate this approach through the irradiation of bulk W samples. In particular, we compare the obtained damage thresholds for melting and cracking in terms of HFF, with the ones present in literature coming from irradiation with electron beams and millisecond lasers. In addition, we irradiate various nanostructured tungsten coatings, deposited by Pulsed Laser Deposition, that mimic different coating scenarios in tokamaks (e.g. W coated tiles, redeposits of W, etc.). These coatings are deposited onto two different substrates, namely silicon and bulk W, which in particular is exploited to better mimic redeposits scenarios. Thermal effects thresholds are thus investigated, under both single and multi-shots irradiation, for metallic W coatings, with different structures and morphologies, and W coatings with high oxygen and nitrogen contents. The obtained thresholds turn out to be consistently lower than the ones found for bulk W, being severely affected by the porosity degree of the material and by the gas concentration. Moreover, delamination strictly depends on the type of substrate material, eventually occurring at very low HFFs (i.e. HFF = 1 MW m^{-2} $s^{0.5}$). We finally develop a 2D thermomechanical model, which can predict the damage thresholds for coatings with complex nanostructure and morphology under different thermal loads conditions by considering the effects of the porosity degree on various macroscopic thermo-physical properties of the material (e.g. thermal diffusivity, yield and ultimate stresses).

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Ammonia production and sticking on polycrystalline tungsten and 316L stainless steel

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During high-power operations in ITER, it will be necessary to inject extrinsic impurities into the edge plasma to dissipate part of the plasma exhaust power through radiation and maintain the power fluxes to the plasma-facing components within tolerable limits. Nitrogen (N) is one of the leading seeding candidates by virtue of its strong compression in the divertor and high radiation efficiency at the temperatures characteristic of high recycling and detached plasmas. Ammonia production has, however, been observed in the all-metal ASDEX-Upgrade and JET tokamaks during N-seeded discharges [1, 2]. The formation of large quantities of tritiated ammonia has consequences for several aspects of the ITER plant operation in terms of tritium retention, gas reprocessing and duty cycle. It is currently unclear how and where ammonia formation predominantly occurs in fusion devices, which makes it difficult to predict the ammonia formation rate in ITER.

In this contribution, we address the following questions:

- What is the dominant ammonia formation mechanism and how does the formation rate depend on the surface material?
- What is the sticking probability of ammonia molecules on ITER-relevant material (tungsten and 316L stainless steel)?

Our studies are performed in an ultra-high-vacuum environment [3] using ion beam and molecular beam exposure of polycrystalline tungsten and stainless steel (SS-316L) samples.

To understand the mechanism of ammonia formation, sequential implantation of N_2^+ and D_2^+ is performed at room temperature while Temperature Programmed Desorption (TPD) is used to quantify HD, D_2, N_2 and ND_3 production rates. On the one hand, we show that deuterated ammonia (ND_3) is produced on both metals when bulk deuterium (D) diffusion is activated. On the other, the absolute quantity of ND_3 produced is found to be strongly dependent on the sample material. This difference in ND_3 production rate is found to be related to dissimilarities in the formation process. On polycrystalline tungsten, we demonstrate that the formation mechanism involves N atoms present at the topmost layer of the surface and bulk nitrogen is unable to participate in the production of ND_3. In stark contrast, on SS-316L, N atoms naturally contained in the bulk alloy participate significantly in the formation of ND_3. Finally, measurements of the absolute sticking probabilities of ammonia molecules on tungsten and SS-316L samples will be presented.

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The research of interaction between helium plasma and lithated graphite substrate coated with silicon carbide

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Due to the properties of stable refractory surface, high toughness, thermal shock resistance, low neutron activation energy, good chemical compatibility and similar thermal expansion coefficient with the graphite substrate, The silicon carbide coating is served as protective coating on carbon-based materials in the ESAT fusion device. As lithium has the advantages of absorption impurity and good compatibility with inner plasma. The lithiation of plasma-facing materials has been confirmed to effectively improve the plasma characteristics in the EAST, but the lithium with strong corrosiveness is likely to aggravate harm to the graphite substrate coated with silicon carbide from escaped plasma including helium and hydrogen plasma. So the study of interaction between helium plasma and lithated graphite substrate coated with silicon carbide is very important on lithium applying to fusion.

The one-cathode linear plasma device built by LANE laboratory from Sichuan University is utilized to perform the experiments in this article. The OES, fast reciprocating probe and thermocouples are adopted to investigate the effect of the lithium on the plasma characteristics. The scanning electron microscope, energy dispersive X-ray detector, X-Ray diffraction, X-ray photoelectron spectroscopy and laser-induced breakdown spectroscopy are respectively used to analyze morphology, composition change, phase transformation and lithium penetration depth in the sample. This work would contribute to reveal the physical mechanism, it can provide scientific explanation for the experimental phenomenon in the ESAT, and provide scientific basis for protecting first wall.

Relationship between spherical nanoindentation stress-strain curve and microstructure of He-implanted tungsten

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As a plasma facing material, tungsten (W) is exposed to extreme environments. Understanding the effects of irradiation-induced damage defects on the mechanical properties of such materials used in nuclear fusion apparatus is vital for extending their service life. In this study, we utilized spherical nanoindentation to investigate the changes in the mechanical properties of W implanted with helium (He) at different doses. Both experimental observations and molecular dynamics (MD) simulation were carried out to reveal the intrinsic relationship between indentation stress-strain curve and microstructure. Multi-energy He implanted experiments were performed to create a flat damage profile with three damage levels (0.2 dpa, 0.5 dpa and 1dpa). Meaningful indentation stress-strain curves provided by spherical nanoindentation measurements indicated an irradiation hardening phenomenon that indentation yield stress increased with increasing damage dose. More specially, yield drop only occurred when damage level exceeded 0.2 dpa, whereas pristine tungsten and 0.2 dpa sample did not exhibit that characteristic. Microstructure evolutions were investigated by transmission electron microscope (TEM) and positron annihilation spectrometry (PAS) technique. Nano-scale He bubbles and small vacancy-type defects showed no significant changes with an increasing irradiation dose, indicating that these visible defects are not the determining factors for yield-related phenomena. Then MD simulation was applied to explore the intrinsic mechanism by further considering the pinning of screw dislocation lines by solute He atoms. The simulated stress-strain curves qualitatively explicitly reflected a tendency similar to that of the indentation stress-strain curves. This study proposes an efficient and promising method of measuring the damage to plasma facing materials and establishes an intrinsic correlation between the changes in the mechanical properties of He-implanted W and its microstructure.

Keywords: Tungsten, He implantation, Yield drop, Spherical nanoindentation, Molecular dynamics

Abstract withdrawn

Modeling of Fuzz Formation in Helium-Ion-Irradiated Tungsten

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Tungsten (W) and tungsten alloys are widely viewed as the most promising plasma facing material (PFM) candidates for divertor and first-wall systems in a nuclear fusion reactor. However, experiments aiming to investigate PFM performance under continuous operation and high first-wall temperature have shown that a nanostructure with a fuzz-like morphology develops on the W surface under the operating conditions of temperature, helium (He) impact energy, and He flux expected for ITER's divertor, which may adversely influence the reactor performance and operation. The aim of our work is to develop a fundamental understanding of the initial stage of fuzz formation and model the surface morphological evolution of helium-ionirradiated tungsten considered as a PFM.
We have developed an atomistically-informed (constitutive equations are parametrized using large-scale molecular dynamics simulations [1]), continuous-domain model to describe the surface morphological evolution of helium-ion-irradiated tungsten. Based on this model, we have conducted self-consistent numerical simulations of the dynamics of the irradiated W surface morphology and benchmarked the simulation results with experiments. Under the experimental conditions - an RF plasma source ($\sim 3 \times 10^{20}$ m^{-2}s^{-1}) exposure of 75 eV He on ITER-grade W at 840°C - the predicted surface morphology shows a good qualitative agreement with the experimental observations for the early stage of nanotendril formation on the W surface, a precursor to fuzz-like surface growth. Additionally, we have made quantitative comparisons between the experimental and simulation results; our model predicts the growth rate of the nanotendrils reasonably well, and provides reasonable approximate estimates for the nanotendril width and arrangement. Two plausible factors controlling the performance of our model are hypothesized: subsurface bubble dynamics and bubble bursting/pinhole formation; future model extensions to test these hypotheses, as well as the anticipated divertor performance are discussed.

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Ammonia formation in N_2-seeded H-mode discharges on JET and ASDEX-Upgrade

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Fusion devices with metallic plasma-facing components will require impurity seeding to reduce divertor heat loads to technologically feasible levels. In burning plasma discharges in ITER, ~70% of power entering the scrape-off layer will need to be dissipated in this way. Among the candidate impurities, experiments in present all-metal devices show that nitrogen (N) provides for the best plasma performance, notably as a result of efficient divertor compression. However, N-seeding leads to in-vessel ammonia formation. On ITER, where N is planned as a seeding gas option, the need to regenerate pumping cryopanels to high temperature to recover tritiated ammonia would have a significant impact on the machine duty cycle if the quantities were too high. In order to develop reliable estimates of ammonia production for ITER, data from present-day fusion devices are required. This paper presents the recent results of studies of its formation during N_2-seeded plasmas on ASDEX-Upgrade (AUG) and JET.

The experiments were performed as series of identically set-up N_2-seeded H-mode discharges. Constant seeding rates and heating powers were used to allow for a straightforward study of the evolution of the nitrogen inventory. Ammonia formation was detected with residual gas analysis (RGA) and divertor spectroscopy. The peak concentration of ammonia in the residual gas at AUG was found to be of the order of 1 % in the inner divertor. The amount of detected ammonia exhibited a pronounced build-up behaviour and legacy effects. Similar trends of radiated power, divertor temperature, confinement enhancement, as well as the core N density from charge-exchange spectroscopy confirm that the rate of ammonia formation is chiefly proportional to the density of N in the divertor plasma, and that the latter is strongly affected by the wall inventory. Spatially resolved residual gas analysis (at AUG) and divertor spectroscopy indicate higher concentrations of ammonia near recessed areas, which would suggest that a significant part of the net ammonia production occurs on areas not accessible to the plasma, via surface reactions between neutral N and H atoms.

Study of Surface Stability for Advanced Liquid Metal Divertors

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Fast flowing liquid metals are a proposed Plasma Facing Component (PFC) for fusion reactors, particularly for the divertor region. Liquid metal has the advantage of being able to be recycled and remove large amounts of heat at high flow speed, whereas solid materials may melt and/or accumulate radiation damage that requires a machine shutdown to replace. However, given the adverse conditions in the divertor region a free-surface flowing liquid metal may evaporate or splash without control. We present simulation results that utilize the existing magnetic field found in fusion reactors in conjunction with externally applied electric currents to generate a volume force (referred to as jxB force) to control the flow.

Surface behavior of liquid metal flows is of great interest, and how the behavior is affected by external magnetic fields and electric currents. Simulations can lend insight to these problems, specifically how different perturbations effect flow under various conditions. We simulate a range of flow conditions applicable to fusion reactors and determine the stability limits for phenomena that result in undesirable features such as splashing. The focus of investigating these limits is to determine the effect of jxB force in various configurations. Of particular interest are the effects of magnetic field gradients similar to fusion reactors (i.e 1/r), as well as externally applied, spatially varying, electric current densities.

The simulation results are benchmarked against known results and experimental data. Understanding the uses and challenges of jxB force on liquid metal flow stability can lead to development of control methods and algorithms that make a liquid metal flow perform well in PFC fusion applications.

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Simplified heat load modeling for design of DEMO discrete limiter

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The shaping of the Plasma Facing Component (PFC) is a fundamental challenge for future fusion reactors like DEMO. It has to be adapted to a large variety of plasma equilibria, ensuring that the PFC will not reach their thermal limits. The core heat load source is caused by charged particles circulating in the Scrape-Off Layer (SOL), following the magnetic field lines and impacting the PFCs. The EuroFusion program WPPMI is in charge of designing the First-Wall (FW) of DEMO.

Several codes like PFCFlux [1] or SMARDDA [2] use ray-tracing techniques to estimate the heat load deposited on PFCs on 3D CAD models. The heat load is assumed to come from the Outer Mid Plane (OMP), to be conducted parallel to the field lines and to decrease exponentially (decay length λ_q from 6mm to 50mm) in the SOL from the Last Closed Flux Surface (LCFS). This simplified model allows for fast estimation (couple of minutes) of the heat load, suitable with design studies and numerous go and back between CAD office and physic simulations.

However, the overall power balance should be satisfied. A way for validating this model is thus to calculate the ratio of the total power on the PFCs to the one circulating in the SOL (3MW to 69MW). This ratio is theoretically equal to 1. This is verified in all circumstances, except when discrete object exists close to the LCFS. This problematic is illustrated on an outer equatorial limiter design for DEMO with associated limiter equilibrium. A part of the energy coming from the OMP is not reported on the FW or the limiter, leading to a ratio as low as 30%. Several propositions to adapt the model for this case in order to recover the ratio are presented.

A first solution consists to follow the magnetic field lines on very long distances (up to 10km), to count how many locations of the OMP reach the same PFC and to cumulate the energy from those different locations. But this assumes that the radial deviation is still negligible at these long distances. A second solution is to assume that the missing power will reach the FW, which could be modeled by adding a second exponential decay length (λ_q^{far}) to the initial one (λ_q). In this study, it is shown that several λ_q^{far} are numerically possible for a given equilibrium. One last prospect is to force a radial deviation of the field line to be closer to the real movement of the particles.

Results with the different solutions are compared and assessed on their validity. The consequences on the design of a particular PFC are also analyzed.

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A review of natural and forced convection effects on power and particle removal by liquid plasma-facing components under plasma bombardment

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It is widely recognized that power and particle handling by PFCs (for plasma-facing components) is one of the critical issues, affecting the successful operation of a steady state magnetic fusion power reactor. Up to present, tungsten has been used for the PFCs in a number of the existing fusion devices and also is employed as the target material in the divertors for ITER and also for DEMO reactor designs. In these designs, tungsten is brazed on a heat sink made of copper alloys with good thermal conductivities. However, one predicts that the heat removal capability would be disabled if a reduced activation ferritic steel alloy, the thermal conductivity of which is typically one third of copper alloys, is used for the heat sink for DEMO reactors. Unfortunately, tungsten is also known to suffer from thermo-mechanical cracking due to its exceptionally high DBTT (the ductile-brittle transition temperature) around 400^{o}C.

In an attempt to resolve these technical issues, the use of liquid metals for PFCs has been proposed and implemented in several existing magnetic confinement experiments. In the case of LLD (for the liquid lithium divertor) in NSTX, despite the expectation, the liquid surface was found to be saturated with implanted deuterium, departing from the reduced recycling condition, just as rapidly as had been observed for solid lithium coatings.

It follows from these findings that either natural or forced convection could have helped de-saturate the liquid surface of LLD, a hypothesis which has recently been proved by the observations in a laboratory plasma facility: VEHICLE-1 [1, 2, 3], featuring a mechanically forced convection capability. In addition, electromagnetically forced (**J**x**B**) convection effects have also been found to enhance particle transport in selected liquid metals including Li, GaInSn, and Ga, which then results in the reduction of hydrogen recycling. Preliminary fluid dynamics modelling has been conducted to support these experiment observations [4, 5, 6, 7]. Based on these arguments, one expects the enhanced heat removal by liquid convection as well.

Reviewed systematically in the present work are these effects of liquid convection on hydrogen recycling and permeation. Also presented will be more recent results from the heat removal experiments by electromagnetically convected liquid metals and its associated fluid dynamics modeling.

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Experiments of continuously and stably flowing lithium limiter in EAST towards a solution for the power exhaust of future fusion devices

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Liquid lithium (Li) has effects to partly ameliorate lifetime and power-exhaust issues of plasma facing components (PFCs) by allowing for a self-healing, self-replenishing surface with no susceptibility to neutron damage in future fusion devices. Such a system is beneficial for the improvement of plasma performance, and therefore be attractive for future fusion devices. In this contribution, in order to allow for stable operation under tokamak conditions as well as its heat-exhaust capabilities, a flowing continuously flowing Li limiter on the concept of a thin flowing Li film on an actively cooling heat sink is designed and tested in high performance discharges in EAST [1,2]. In addition, experiments related to material compatibility, wetting, flowing as well as plasma impact were studied. A circulating Li layer with a thickness of <0.1 mm and a flow rate ~2 cm^3s^{-1} was achieved. Novel in-vessel electro-magnetic pumps (EMPs), working with the toroidal magnetic field of the EAST device, were reliable to control the Li flow speed. Some new ideals and techniques, i.e. increase the thickness of stain steel layer, using hot isotatic pressing (HIP) technology to braze stain steel layer and Cu heat sink, new design of distributor and a new set of high pressure He cooling system, were successfully explored to enhance Li coverage uniformity, erosion resistance of limiter surface and heat removal, and also plasma performances [3]. It was found the flowing liquid limiter is fully compatible with various plasma scenarios, including high confinement mode plasmas heated by lower hybrid waves or by neutral beam injection. The controllable Li emission from the limiter was beneficial for the reduction of recycling and impurities, for the reduction of divertor heat flux and the mitigation of ELMs (~150ms ELM-free phase), and in certain cases, for the improvement of plasma stored energy, which bodes well application for the use of the renewal of circulation of Li flows on targets of divertor in future fusion devices. The results shows that the flowing lithium PFC is an possible and alternative choice for the design divertor with high heat flux in future reactors by allowing for a self-healing, self-replenishing surface with no susceptibility to neutron damage to partly ameliorate lifetime and power-exhaust issues of PFCs. To further increase the flowing uniformity due to well wetting capacity between Li and Mo, new FLiLi upgraded using Mo instead of SS as support surface for Li is expected to be tested in HIDRA in Illinois and EAST in the end of 2017.

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Studies on formation of tungsten nano tendrils under irradiation of helium plasma in CIMPLE-PSI device

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Irradiation of tungsten surfaces by helium plasma may lead to the growth of a fibreform nanostructure knows as “fuzz”. The formation conditions of tungsten fuzz have been widely studied, although the formation mechanism is not yet fully understood. The formation of tungsten fuzz affects the erosion properties and fuel retention of tungsten. In addition, the fragile nature of the fuzz raises concern about possible enhanced formation of tungsten dust. Nanostructuring may also influence the surface mechanical properties of tungsten. We here report studies on the formation of helium nano fuzz in the CIMPLE-PSI device that has been designed, constructed and made operational at CPP-IPR, India, for controlled plasma surface interaction (PSI) research, under ITER–like divertor plasma conditions. Tungsten samples from Plansee (99.97% pure) were irradiated with pure helium plasma (19 kW) for a time duration of 1800 seconds under 0.3 Tesla magnetic field where the target was biased negatively (-45V). Fuzz formation was observed on the target together with isolated nanotendril bundles, as revealed by FESEM micrographs. The nanostructure was removed from the surface and analyzed using a high resolution Transmission Electron Microscope (HRTEM). The diameter of the individual filaments was found to be in the range of a few to 60 nanometers. Bubbles were seen embedded inside the filaments with relatively bigger sizes as compared to other reported values, growing maximum up to 20 nanometers, which is understood to be a consequence of higher surface temperature of the target. Grazing incidence x-ray scattering (GISAXS, performed at MiNaXS beamline P03 of PETRA III synchrotron radiation source, 13keV X-rays, Germany) was used to get information about the morphology of helium bubbles and their depth distribution, supplementing TEM observations. One interesting finding is that with increasing depth, the spatial distribution of helium bubbles exhibits some positional ordering. Results obtained for different plasma conditions (pure He and with 0, 20, 40% H admixtures) and different target annealing conditions. The plasma was diagnosed by optical emission spectroscopy, retractable Langmuir probe and water calorimeters that confirm PSI experiments were performed under ITER-like both extreme ion and heat flux regime.

Deciphering hydrogen isotope retention and sputtering in liquid metal/porous tungsten hybrid materials

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Tungsten is the material of choice for the plasma facing components (PFC) in the divertor region for its favorable thermomechanical properties such as its high melting point, thermal conductivity, low sputter yield, stability to neutron irradiation [1]. However, mechanical failure and ion induced surface morphology transformation such as fuzz [2], can expose the bulk W to energetic ion fluxes and heat fluxes increasing the probability for high-Z metal dust formation and un-tolerable radiative cooling effects. In addition, transient events can cause micro cracks, fractures that can propagate throughout the W armor resulting in microstructure failure, compromising fusion plasma confinement and overall reactor performance [3].

These challenges have motivated alternative concepts that may be able to address some of these technology gaps. One of these alternative concepts is the combination of a solid-phase matrix or hierarchical structure with metals in the liquid state [4]. Liquid metals at the surface of the plasma-material interface could perhaps address issues of surface replenishment and heat exhaust as well as other radiation-driven mechanisms. Aside from demonstrated impurity-gettering capabilities of some liquid metals, the intrinsic healing nature of a liquid metal wall mitigates radiation damage issues but are limited by drawbacks including but not limited to: macro ejection, liquid-metal corrosion and evaporation/redeposition. Exposed stainless-steel in current capillary porous systems in a dry-out scenario will have adverse effects on the edge and core plasma, so a PFC system that addresses LM/substrate material compatibility is a key area of investigation. A natural choice would a hybrid system of a viable liquid metal incorporated in a tungsten substrate. Issues related to hydrogen isotope retention and erosion are key PMI gaps.

Results presented here look to explore previously observed contrasting surface response to irradiation as well as differences in hydrogen isotope retention capabilities by exploring the role of the surface geometry [5]. Differences in the retention abilities of liquid metal/porous W systems (Li and SnLi) and the relevant Li, O, and C chemical complexes that are present during 500eV D_2^{+} at 230°C and 300°C are examined. The surface replenishment of the liquid metals on the porous substrate are observed by *in-operando* ion scattering spectroscopy of the surface. Additionally, preliminary thermomechanical properties, such as thermal conductivity, and plasma-material interactions (i.e. sputtering), are characterized of this hybrid system.

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Fast Flowing Liquid Metal Divertors for Fusion

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We present the results and developments from the fast flowing liquid metal divertor experiments, Liquid Metal eXperiment(LMX), and Flowing LIquid metal Torus (FLIT), at Princeton Plasma Physics Laboratory, PPPL.

Currently, there are no solid material that can handle both the neutrons and the extreme heat flux that will be present in a fusion reactor. A fast flowing liquid metal divertor is an alternative that aims to handle all the cooling requirements at the divertor allowing a steel alloy surface behind to be designed only for the neutron damage. Using lithium as the liquid metal may also have the added advantage that hydrogen isotope and possibly helium pumping in addition to core confinement enhancement. This concept has no limit on the heat that can be handled only an increase in liquid metal velocity is required. The main issue to overcome in this concept is the stabilizing the fast flow under MHD effects.

LMX at PPPL is used to study the fast liquid metal in channel configuration with magnetic field up to 0.3 Tesla and 2 m/s flow speeds. FLIT is an upcoming torus device at PPPL is designed to look at the annular flow at up to 1 Telsa and 10 m/s. We present the results from LMX and the design and construction of FLIT.

At LMX we studied the heat transfer in liquid metal, and found optimal channel surface shaping to obtain the maximum heat transfer from the surface that would minimize the evaporation in a reactor. The effect of shapes such as delta-wing shapes and various dimple configurations at the bottom of the channel and their effects in the experiment and comparison with the numerical simulations are shown. The effects of currents running through the liquid metal were explored. This effect is important for magnetic propulsion and flow stabilization. We developed an analytical and numerical model for the Lorentz forces. The relationship between the Lorentz force flow parameters and the hydraulic jump location is shown.

FLIT focuses on a liquid metal divertor system suitable for implementation and testing in present-day fusion systems, such as NSTX-U. It is designed as a proof-of-concept fast-flowing liquid metal divertor that can handle heat flux of 10 MW/m^2 without an additional cooling system. The 72 cm wide by 107 cm tall torus system consisting of 12 rectangular coils that give 1 Tesla magnetic field in the center and it can operate for greater than 10 seconds at this field. Initially, 30 gallons Galinstan (Ga-In-Sn) will be recirculated using 6 jxB pumps and flow velocities of up to 10 m/s will be achieved on the fully annular divertor plate. FLIT is designed as a flexible machine that will allow experimental testing of various liquid metal injection techniques, study of flow instabilities, and their control in order to prove the feasibility of liquid metal divertor concept for fusion reactors. Details of the design of FLIT will be presented.

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Impact of low-Z and high-Z ion-induced damage on the reflectivity molybdenum mirrors and sub-surface distribution of gas bubbles

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There is a broad range of research activities carried out worldwide aiming at the assessment of first mirrors performance in next-step devices. This work is focused on the determination of mirror performance following multiple ion irradiations to simulate both transmutation effects and damage which may be induced by neutrons under reactor conditions, e.g. in DEMO.

The research has been carried out for polycrystalline molybdenum mirrors. The irradiation was performed at the Ion Technology Centre (ITC) of the Uppsala University using the 350 kV Danfysik 1090 implanter with a beam current up to 1 mA: 30 keV $^{98}Mo^+$, $^{93}Zr^+$, $^{90}Nb^+$, 2 keV He^+ and 4 keV H_2^+ ions. The conditions were based on SRIM predictive modelling to implant in the optically active layer: 15-20 nm. Zr and Nb were to simulate transmutation, while H and He both transmutation and plasma impact. Irradiations at 300 K and 573 K were step-wise with regular reflectivity measurements in the 300-2400 nm range with a dual beam Lambda 950 spectrometer. This was followed by the quantification of implanted H and He using heavy ion ERDA and very detailed examination of the surface layer: sampling with a focused ion beam (FIB/SEM, Hitachi NB5000) and analysis with scanning transmission electron microscopy (STEM, Hitachi HD2700) operated at the accelerating voltage of 200 kV. Systematic studies were performed on over 20 mirrors. Key results obtained by this comprehensive approach are summarized below.

- Irradiation with metal ions up to 30 dpa reduces the total reflectivity by 5-8%, while much stronger reduction (30%) is caused by helium and further by hydrogen ions. Diffuse reflectivity is always below 2%.
- Implanted He is retained in the layer of up to 40 nm. The concentration decreased to about 60% of the initial measurement when measured again 2 years later.
- He ions lead to the creation of bubbles within a thin layer below the mirror surface that differed in size. Bubbles of approximately 10 nm in diameter are created in Mo amorphous coating of 20 nm in thickness.
- Smaller bubbles of 1 nm in diameter and less are formed within the thin layer of approximately 20 nm below the amorphous coating. The bubbles tend to be formed on dislocations present in mirrors as a result of either grinding/polishing or heavy ions irradiation.
- The irradiation with Mo ions creates defects near the surface that probably trap/facilitate nucleation of smaller helium bubbles within the damaged zone leading to a more uniform bubble distribution than in the case of a low Mo dose or no Mo irradiation.
- A dense network of bubbles is created by H bombardment.

The results described above will be complemented by data on the temporal changes in the irradiated mirrors. Also, the impact of shallow irradiation on the deuterium uptake will be addressed and the implications for a reactor-class machine will be discussed.

Studying the Role of Second Phase Particles in Tungsten Alloys: Mechanical Properties and Response to Low Energy, High Flux D/He Plasmas

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Tungsten (W) is the current material of choice for plasma-facing components in future plasma-burning fusion reactors because of its high melting point, mechanical toughness, and high sputter threshold. However, the high melting temperature and low recrystallization temperature pose complications during fabrication [1]. Spark plasma sintering (SPS) is a powder compaction technique that provides high pressure and heating rates, allowing for a lower final temperature and hold time for compaction. In SPS, a strong electrical current is fed through the powder and die to heat the powder. An applied uniaxial force combines to compress the powder to a solid compact [2], leading to fully dense materials at lower temperatures as compared to conventional sintering [3].

One class of materials developed with SPS is dispersion-strengthened tungsten. These materials are fabricated with 500 nm W powder alloyed with zirconium carbide, titanium carbide, or tantalum carbide particles in weight percentages of 1.0-10 wt.%. Samples were fabricated via SPS at the University of Illinois by heating to 1800ºC under 60 MPa pressure with a three-minute hold time. In prior work, titanium carbide strengthened tungsten has shown better resistance to helium and neutron damage than pure tungsten [4]. It is conjectured that carbide-strengthened W may have enhanced ductility as the carbides capture impurity oxygen atoms while refining and stabilizing the W matrix [5]. To study this, the ductility of the above samples will be tested by 3-point bending tests at temperatures to 500ºC to determine the ductile-brittle transition. Recrystallization behavior of ZrC-strengthened samples up to 2100ºC will be studied to determine recrystallization resistance. Also, thermal conductivity of alloyed samples up to 500ºC will be presented.

The plasma-surface interaction of W samples alloyed with different types and concentrations of dispersed second phases will be studied. Samples will be exposed to <50 eV D and He ions at 10^{26} m^{-2} fluences and temperatures >800ºC with synergistic transient and quiescent ELM-like events simulated via laser pulses. The role of the type and amount of dispersed second phase on surface morphology, fuel retention, and surface chemistry will be examined via microscopy and spectroscopy techniques.

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Impact of plasma irradiation on selective laser melted tungsten and tungsten alloys

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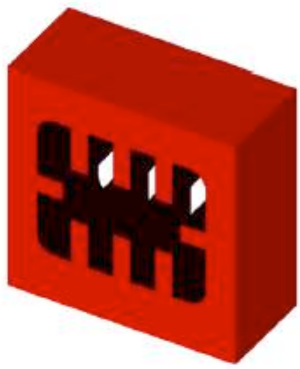
Advanced material and structure are needed to sustain the extreme environment in DEMO with over 20 MW m^{-2} heat load, high particle flux and neutron irradiation. A full-tungsten (W) divertor target shown on the figure is designed to achieve 20-30 MW m^{-2} heat load capability. In this design, the cooling tunnel is square shaped with fins structures on the side. The heat transfer path of the striking point in the edge is effectively shortened, and the effective convective heat transfer area is increased by the fin, thereby improving the heat transfer capability. However, it is very difficult to machine tungsten into complex shapes through conventional forming method. Additive manufacturing allows the whole engineering part to be built as a single component without further processing, thus brings the possibilities to overcome this shortcoming. However, densification and crack control remain great challenges in tungsten additive manufacturing. It is vital to produce dense, crack-free tungsten as well as assess its properties after plasma exposures.



In the present study, dense pure W and alloys of W (W-Ta, W-Re and W-ZrC) were fabricated by selective laser melting (SLM) and their behaviour after plasma exposure were investigated. The results showed that powder spheroidisation improved the densification process and alloying contributed to the crack control. Crack-free tungsten samples with relative density > 99% were obtained after alloying with ZrC. Plasma irradiation experiments in Magnum-PSI were carried out (H plasma, $\Phi \sim 4.1 \times 10^{25}$ m^{-2}; He plasma $\Phi \sim 5.5 \times 10^{25}$ m^{-2}) which revealed that tungsten alloys behaved very good performance under identical plasma exposure. Compared to conventional rolled W, hydrogen blistering is supressed significantly in W-Re and W-ZrC alloys. EBSD was applied to analyse the relationship between grain information and the blistering behaviour. The sub-surface morphology was further investigated by focused ion beam (FIB) / second electron microscope (SEM). Nanostructures were formed on W and W alloys under helium plasma irradiation and their morphologies were studied and compared. The nanostructure formation mechanisms were investigated via atomic force microscope (AFM) and transmission electron microscope (TEM). The first experiments on tungsten selective laser melting and following plasma irradiation tests showed very promising results. Crack-free, 99% relative density tungsten alloy samples were produced. The detailed study on tungsten selective laser melting process and plasma irradiation resistance properties were presented.

Avoiding deep cracking in tungsten divertor armor under high heat flux loads

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Currently, tungsten is the main candidate for the armor material of the divertor in future fusion reactors due to its high melting point, high thermal conductivity, modest thermal expansion, extremely low sputtering rate and negligible hydrogen solubility. However, the brittleness of tungsten below its ductile to brittle transition temperature raises a critical concern in maintenance of structural integrity under high-heat-flux (HHF) fatigue loads. Loss of structural integrity may lead to structural as well as functional failure of the component.

This critical concern in the divertor target has been already confirmed by variety of HHF tests with electrons and ion beams. ITER Organization chose a full tungsten divertor target, and plenty of HHF qualification tests have been conducted. The tested prototypes showed that the tungsten monoblock armor often suffered from deep cracking, when the applied HHF load approached 20 MW/m2. The deep cracks were initiated at the armor surface and grew toward the cooling tube in the vertical direction. Deep cracks have been also frequently observed in the adiabatically loaded solid tungsten divertor target in ASDEX Upgrade [1], in which a more complex and fusion reactor relevant loading condition is expected compared to the HHF test facilities. In both cases, the deep cracking seemed not to affect the heat removal capability of tungsten divertor, as most of them were perpendicular to the loading surface. However, the inherently unstable nature of brittle cracking may likely increase the risk of structural failure. Understanding the cracking mechanism is therefore of essential importance for the divertor design.

The mechanism of deep cracking in monoblock tungsten targets has been interpreted with the aid of numerical simulation [2]. The driving force of these deep cracks is the tensile stress induced by the plastic deformation generated by the HHF loads. To avoid appearance of deep cracking, a rigorous design study has been performed. The analysis shows that with a proper design the risk raised by the brittleness of tungsten might be significantly reduced, which has been confirmed by the HHF test in GLADIS.

Based on the lessons learned from monoblock targets, the thermal load might play a very important role for deep cracks in the tungsten divertor in ASDEX Upgrade [3]. Therefore, the impact of design options (smaller width or castellation) on the stress distribution induced by the thermal loads has been analyzed, which shows that the driving force of the deep cracks is largely reduced by applying the optimized design. These results are generally in line with the first results in ASDEX Upgrade campaign 2017 [4].

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Experimental assessment and simulation of stress distribution in tungsten exposed to ITER-like steady-state and transient plasma

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Understanding thermo-mechanical modifications of tungsten induced by ITER-like plasma is crucial for the lifetime predictions of the ITER divertor and selection and design of new materials for DEMO. Currently, different types of experiments by various devices have been carried out [1-3]. However, typically the thermal shock and fatigue behavior due to heat loading or the surface modifications due to plasma loading are considered alone, without combining both the synergistic plasma and mechanical aspects. Moreover, the underlying physics/mechanics are still poorly understood. More insights are needed, especially the assessment and simulation of stress distribution under combined steady state and transient plasma.

In this study, ITER-like high flux plasma (~10^{24} $m^{-2}s^{-1}$) superimposed with transient plasma was generated using the pulsed plasma source system of Magnum-PSI [4]. The spatial and temporal profile of the pulse plasma were well characterized by a time resolved (200 μs) Thomson scattering system while the 2D temperature distribution on the target was captured by a fast-framing (5 kHz) infra-red camera. Generally, the pulsed plasma has a triangular shape of 200 μs ascent and 1000 μs decay. The pulse energy was chosen to induce ΔT between 200 and 300 °C, which corresponded to peak heat flux in the range of 200 to 250 MWm^{-2}. This information can be used as boundary conditions in a coupled temperature-displacement finite element model, where stress distribution can be simulated. Secondly, 9 hot-rolled tungsten samples from Plansee® were exposed to different base temperatures (500 °C, 700 °C, 1100 °C respectively) and pulse energies. High resolution electron backscatter diffraction [5] is employed to determine the stress distribution experimentally. Then, an optimization algorithm is used to minimize the difference between simulations and a set of experiments by adjusting the material's constitutive equation. In this way, the constitutive equation of the plasma modified surface layer is determined. This can be compared to results from spherical nanoindentation, which is also capable of extracting part of the terms in the constitutive equation. The numerical model employing this constitutive equation can be used to predict the stress distribution under another set of plasma exposure conditions. This model will be further improved to account for damage evolution in the materials and finally predict their thermal shock/fatigue behaviors under long-term plasma exposures.

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Thermal and mechanical properties characterization of the surface damaged layer of tungsten

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As the potential divertor material of future fusion devices, the structure of tungsten (W) surface is modified by plasma irradiation. Thermal and mechanical properties of the damaged layer degrade and the melting and cracking thresholds decrease. The material can therefore fail more easily by transient events such as edge localized modes (ELMs), which produces extremely fast and powerful cyclic heat loads. Quantified criteria and parameters are needed to estimate the damaged level and to predict the potential failure rather than simply observing the damaged microstructure and analyzing the total fluence.
In the present study, the thermal and mechanical properties of the W surface layer irradiated by hydrogen (H) plasma with ion energy of 50 eV and fluence of 1.6×10^{26} $m^{-2}s^{-1}$ on Magnum-PSI, by helium (He) plasma with ion energy of 40 eV and fluence of 10^{25}-10^{26} $m^{-2}s^{-1}$ on PSI-II, and by He ions with energy of 30-190 keV and damage of 0.2-1 dpa were characterized. The transient thermoreflectance technique [1] was employed to test the thermal conductivity (TC) of W samples irradiated by He plasma/ions. Results show that the TC values of the W damaged layer dropped 1-2 orders of magnitude compared to the W bulk and decreased as the irradiation temperature and fluence increased. The serious degradation was due to the formation of He-bubbles and the damaged crystalline structure. Additionally, we utilized Berkovich/Spherical nanoindentation to test the mechanical properties changes induced by H plasma and He ion irradiation. Berkovich nanoindentation results indicate that irradiation hardening occurred and the hardness increased with increasing damage dose and exposure temperature. Moreover, the stress-strain curves extracted from Spherical nanoindentation reveal that yield stress increased with increasing dose and working hardening rate decreased after irradiation. Finally, the irradiated samples were loaded by ELM-like head fluxes with a pulse width of 1 ms and peak power density of 1.7 $GW\cdot m^{-2}$ produced by electron beam on EMS-60. Melting occurred and the grain size of the resolidified structure, as well as the depth of the molten bath increased as the TC of the damaged layer decreased. Obvious rounded cracks were observed around the molten baths of the He ions irradiated samples and became more severe as the TC decreased, which can be explained by the modified temperature field resulting from different TCs and the decreased ductility induced by the irradiation hardening [2]. The damage behaviour caused by the ELM-like heat load after previous irradiation is consistent with the results of the characterization of the thermal and mechanical properties.

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Plasma irradiation behaviors of an advanced W-Y_2O_3 alloy prepared by high energy rate forging

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A kind of W-Y_2O_3 alloy with Y_2O_3 content in the range of 0.5-2vol.% has been developed with liquid-liquid dopping technology for complex powder preparation and high energy rate forging for rapid deformation process. Its ductile-brittle transition temperature (DBTT) can be reduced to around 100°C and the recrystallization temperature is higher than 1600°C. Meanwhile, excellent mechanical properties were achieved [1,2].
Deuterium plasma irradiation of well developed W-Y_2O_3 alloy was performed in a linear plasma device (STEP, Beihang University) with the following parameters, D ion flux of about 1.2×10^{22} D/m^2.s at an ion energy of 50 eV. The applied fluences are 2.16×10^{25} D/m^2, 4.32×10^{25} D/m^2, 8.64×10^{25} D/m^2 and 1.296×10^{26} D/m^2, respectively corresponding to 0.5h, 1h, 2h and 3h irradiation time, and the sample temperature was stable in the range of 443-463K. Both surfaces, the parallel to and the perpendicular to the forging plane were irradiated. After D^+ irradiation, the deuterium retention was also measured by a thermal desorption spectrometer (TDS). Dense small-blisters (≤3μm) were found on the surface parallel to forging plane while hardly found on the surface of perpendicular direction. However, larger D retention was observed for the surface of perpendicular direction. The difference of surface morphology and D retention in two directions could be due to the elongation grains in the perpendicular direction, which induce deeper trapping of D atoms along grain boundaries.
Plasma facing materials have to receive the irradiation of hydrogen isotopes and helium mixed plasma in the future fusion reactors. To simulate the synergistic effect of heat loading and particle irradiation, irradiation experiments with a H neutral beam containing 6 at% He were performed in the high heat flux test facility GLADIS (IPP, Germany). The formation of blisters-pin holes-coral like structures was observed on the W-Y_2O_3 surface of perpendicular to the forging plane and the corresponding surface temperature is at approximately 600-1000-1500°C. After H/He neutral beam irradiation, repetitive electron beam pulses (1ms pulse duration) were applied on the irradiated surface. It was found that the critical cracking initiation did not changed compared with unirradiated samples.

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Deuterium retention in tungsten-based materials for fusion applications

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We investigate the deuterium retention behavior of tungsten based materials, which have so far not been investigated with respect to their properties as plasma-facing materials in fusion devices.
Tungsten is currently considered to be a promising material for application on plasma-facing surfaces in fusion devices. This is due to specific advantageous material properties, e.g. its low yield to physical sputtering with energetic hydrogen isotopes. However, tungsten also exhibits some disadvantages when used as a plasma-facing material. For this reason specific tungsten based alloys or composites are considered.
Self-passivating tungsten alloys are being developed to be resistant against oxidation at high temperatures. Otherwise, tungsten oxides would be released to the gas phase, when hot tungsten surfaces come in contact with oxygen in case of a loss-of-coolant accident with simultaneous air ingress. Bulk samples of such alloys are produced by mechanical alloying and subsequent hot isostatic pressing [1]. The material investigated in this contribution is W10Cr0.5Y.
HPM1850 is a commercially available material, which is currently explored as a divertor target material in ASDEX Upgrade [2]. This is a liquid phase sintered composite material consisting of tungsten and 3 wt% of a Fe/Ni mixture. It is employed to overcome problems with crack formation in bulk tungsten target plates [3].
The materials introduced above are investigated with respect to their deuterium retention. For this purpose samples are exposed to an ECR deuterium plasma for different durations to achieve fluences between $10^{23}m^{-2}$ and $10^{25}m^{-2}$. Subsequently the samples are investigated by nuclear reaction analysis (NRA) and thermal desorption spectroscopy (TDS), scanning electron microscopy, and confocal laser scanning microscopy.
The NRA analysis is performed with a ^{3}He beam with energies varying from 0.7 MeV to 6.0 MeV. This allows the determination of deuterium depth profiles in the near-surface regions of the samples. The integrated TDS spectrum yields the total amount of deuterium in the respective sample. It turns out that the total amount of deuterium retained in the HPM1850 material is comparable to that of pure tungsten while the self-passivating material retains a significantly higher amount. The temperature and fluence dependence of this effect is investigated.

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Review of the experiments performed with liquid lithium and tin limiters on FTU

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The use of liquid metals can improve the lifetime and reliability of plasma facing materials (PFM) because they are not subjected to erosion with dust formation and to a deterioration of their thermo mechanical properties due to intense hydrogen and helium influx and neutron irradiation as for solid materials. Possible liquid metal choices include Li and Sn. Lithium is a low Z material (Z=3), which allows for better plasma performance but its operational window is very narrow, 300<T<500 °C, to avoid strong evaporation. Tin is a high Z element (Z=50) having a large operation window due to the low vapour pressure, 300<T<1300 °C, low or negligible activation, low H retention and less safety problems in particular in combination with water cooling. Nevertheless, for tin, it is crucial to demonstrate that plasma operations are possible at a tolerable Z_{eff} value without plasma performance degradation.
Since 2006 on FTU tokamak, the possible use of liquid metals as PFM has been investigated. FTU is the first and only tokamak in the world operating with a liquid tin limiter. The compact high magnetic field FTU device can achieve a high power flux close to the last closed magnetic surface. Different limiter configurations, also actively cooled, were installed on FTU but all with the Capillary Pore System [1]. Four Langmuir probes are installed on the limiter itself to measure the density and the electron temperature close to the limiter. A fast IR-camera to monitor the surface temperature and a visible spectrometer to detect the line emissions for Li and tin are installed in a upper port looking directly to the limiter. Typical FTU plasma parameters close to the limiter are: density in the range of $0.4 \cdot 0.8\ 10^{19} m^{-3}$, electron temperature of about 10-20eV and e-folding lengths of heat loads in the range between 1-1.5cm. The liquid limiters have withstood heat loads less than 10 MW/m^2 for Li and 18 MW/m^2 for tin without any damages. Strong evaporation is present at these power levels but nevertheless no degradation in plasma performance has been observed. We have respectively estimated a maximum Li and tin concentration of about 10^{-2} and 5.10^{-4} of the electron density. These values are both compatible with reactor operations.
A review of the main results achieved and on the open points for a possible use of liquid metals as PFC in DEMO will be presented.

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Exposure of actively cooled ITER divertor mock-ups in ASDEX Upgrade

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In order to support the development of a qualification strategy for ITER divertor targets, a selection of mock-ups from several manufacturers have been subjected to high heat-flux tests in the GLADIS facility [1]. Specifically, one of the actively cooled W monoblock mock-ups by a European supplier has been successfully tested by applying 100 pulses at 10 MW/m and another 300 pulses at 20 MW/m . Each pulse lasted for 10 s, which allowed reaching steady state thermal conditions. Under the applied cooling conditions (inlet cooling water temperature 15 °C, 11 m/s water velocity) surface temperatures of 1100 °C and 2350 °C, respectively, were measured in accordance with finite element analysis. In order to investigate the further behaviour of the mock-up under ITER-relevant plasma and cooling conditions, it has been exposed in the divertor of ASDEX Upgrade (AUG) for several tens of discharges with a maximum heating power above 15 MW. For this purpose the latter mock-up has been mounted together with an identical, but unused muck-up on the divertor manipulator [2], using an active cooling loop. This allowed mimicking the ITER conditions as close as possible, namely using an inlet water temperature of 70 °C and a flow of 1 l/s. The mock-ups are equipped with thermocouples and the divertor manipulator is monitored during discharges by thermography and spectroscopy. A schematic few of the divertor manipulator head is presented in Fig.1.

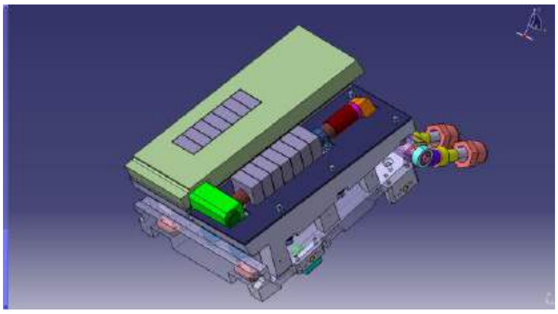

Fig.1: Schematic view of the divertor manipulator head with the two actively cooled ITER tungsten divertor mock-ups. In order to provide a better visibility of the actual technical arrangement, the right divertor target plate has been omitted in the figure.

Before the component was mounted in AUG, the status of both mock-ups has been documented and markers have been placed by the high resolution, heavy duty SEM/FIB available at IPP. As a consequence of the loading in GLADIS for 3000s at 20 MW/m , strong recrystallization and dimensional changes of the single monoblocks have been detected. By comparing the evolution of these surfaces with the ones of the unused mock-up after exposure in AUG, information on the initiation and evolution of damages under ITER relevant plasma condition has been extracted.

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Advanced Electron Microscopy Method Development to Improve Characterization of Plasma-Surface Interactions

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Plasma-materials interactions (PMIs) drive a large number of processes in the surface and sub-surface regions of the solid materials used as plasma-facing materials (PFMs). Degradation pathways of the materials, such as bubble and cavity growth, nanotendril "fuzz" formation, blistering and spallation, as well as neutron-induced damage modes that will interact with the plasma-induced damage, are all ultimately atomistic processes, and require atomic- or near-atomic resolution methods to properly characterize. We have been combining plasma exposure of materials, primarily polycrystalline tungsten, with advances in electron microscopy methods to improve the fidelity and resolution with which we can characterize plasma-induced damage in the PFMs, and analysis of damage and transmutation-induced precipitation in neutron-irradiated tungsten, which will allow understanding of PFM performance in-reactor.

In this presentation, we will provide examples of advanced electron microscopy analysis of defects evolving at the plasma-materials interface and in neutron-irradiated materials, using tungsten as the primary example. We will illustrate the use of transmission Kikuchi diffraction (tKD) to measure fine-scale (<5 nm) crystallography, such as the grain-boundary character distribution at the substrate/nanotendril "fuzz" interface, along with discussion of focused ion beam (FIB) protocols to produce nanotendril fuzz samples with minimized damage and preparation artifacts. Low-energy (50 eV) plasma exposure produced copious low-angle grain boundaries at the substrate/fuzz interface, along with small nanograins within the individual tendrils, implying that the plasma causes a continuous lattice rotation and grain boundary renucleation along the length of each tendril. Thermal desorption spectroscopy (TDS) applied to the fuzzy tungsten caused significant helium emission at high temperature, consistent with high pressure helium in the previously-observed cavities. Monochromated, aberration-corrected scanning transmission electron microscopy (MAC-STEM) was used to probe the pressure inside individual helium bubbles, and preliminary results are consistent with high helium pressure induced by the plasma exposure. Electron channeling contrast imaging (ECCI) applied to pristine and helium-plasma exposed tungsten is being used to study changes to the dislocation and defect structures caused by the plasma, as well.

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PG-QRO-1 a Plasma-Gun for PSI-Studies

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The study of the interaction of magnetized plasmas with candidate materials for fusion reactors, is a main topic in fusion research. While Plasma Simulators can produce intense beams their energy is usually limited to a few tens of eV. Plasma Foci, on the other side, produce a large spectrum in energy with energies ranging up to tens of keV. The plasma-gun PG-QRO-1, a coaxial electrode pulsed gun, has been tailored to produce plasmas with relevant densities but limiting the high energy spectrum in order to use it for plasma-wall-interaction studies. Deuterium low energy plasmas have been exposed to different materials like Tungsten, Titanium, Silicon and Pyrolytic Graphite in order to study PG-QRO-1 capabilities and limitations. The deuterium retention depth profiles in the materials are very shallow with penetration depths of the order of tens of nm, while showing good linearity of retention with the number of shots.

Capabilities of the prototype Material Plasma Exposure eXperiment enabling first reactor relevant SOL, detachment and PMI research

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The linear prototype-Material Plasma Exposure eXperiment (Proto-MPEX) device is a test bed for the high intensigy plasma source eventually to be used in the Material Plasma Exposure eXperiment (MPEX). MPEX is a planned device to address plasma-material interactions for fusion reactors including the capability to expose a-priori neutron exposed materials.

Proto-MPEX has increased its capabilities with respect to electron, ion heating, total injected power and its diagnostic suite over the last years. Plasma densities of about 8 x 10^{19} m^{-3} have been reached with a high-power helicon antenna. Electron heating via 28 GHz ECH/EBW up to temperatures of 20 eV have been obtained. Ion heating was observed at the plasma edge up to temperatures of 35 eV. With those conditions close to reactor relevant plasma fluxes have been measured on the target (~ 5 x 10^{23} $m^{-2}s^{-1}$). Power fluxes of ~14 MW/m^2 in low density discharges (~0.5 x 10^{19} m^{-3}) and ~2 MW/m^2 in high density discharges (~4 x 10^{19} m^{-3}) have been achieved.

With these capabilities, a variety of plasmas can be generated enabling the study of convection dominated scrape-off-layer transport as well as conduction dominated scrape-off-layer transport. Measurements were compared to B2-Eirene modeling for interpretation and will be presented. The results are favorable for the extrapolation to MPEX, which relies on the demonstration of a conduction limited transport regime. Depending on the heating mix and the neutral pressures in the different locations of the device (skimmers allow for differential pressures), attached or detached plasmas are obtained. Target ion fluxes, ion energies and sheath conditions approach those observed in the divertor of a tokamak. Albeit the limit in pulse duration, Proto-MPEX allows for first PMI experiments. An overview of potential PMI experiments utilizing the unique capabilities of Proto-MPEX will be given. In addition, a few examples of PMI experiments are shown.

Alternative divertor configurations for energy and particle exhaust

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Energy and particle exhaust is one of the key challenges of the tokamak approach towards fusion energy. It remains, in particular, not obvious that an exhaust solution based on a single-null, poloidal divertor, as envisaged for ITER, will extrapolate to DEMO and ultimately to an attractive fusion power plant [1]. Power exhaust generally scales unfavourably with size and, there is the concern that it also scales unfavourably with magnetic field, while a high magnetic field may be necessary for an economical power plant. Research on alternative, better performing, divertor configurations is, therefore, performed world-wide and has become a priority of the EUROfusion programme. This paper reviews a wide range of proposed alternatives to the conventional, single-null divertor that aim at reducing the heat and particle loads at the plasma-material interface and further improving the economical attractiveness of fusion energy.
Prominent proposed alternatives include concepts such as the “snowflake” [2], “X divertor” [3], “Super-X divertor” [4] and “X-point target divertor” [5] configurations but also further variants and up-down symmetrisations. All are based on a highly dissipative divertor relying on a partially detached divertor operating regime, similar to ITER, or even on full detachment. They employ geometric modifications of the magnetic configuration and the optimisation of the divertor target geometry and, in some cases, of gas baffles. Their differences lie in their power and particle exhaust handling. Improved performance can be manifested as a reduction of the peak heat to the targets, easier access to divertor detachment (e.g. with less impurity seeding), stable feedback mechanisms to keep the detachment or radiation fronts from moving into the main plasma, or an increase of the radiated power in the SOL whilst retaining sufficient core confinement. The main concepts are described and their underlying physics mechanisms exposed. While some mechanisms have been identified in experiments, a wider range of proof-of-principle experiments is still outstanding. Similarly, divertor transport codes can support some of the invoked mechanisms, but key issues such as the scaling of the cross-field transport and the stability of radiation patterns in the proximity of the X-point are not understood and limit our ability to predict and quantify the benefits in reactor conditions. These benefits will have to outweigh the increased complexity of a fusion power plant with an alternative divertor configuration. Among the discussed complications are the need for additional magnets, greater mechanical forces between magnets and larger toroidal field coils than in the conventional approach. Significant efforts are being undertaken to close the physics gaps, evaluate the engineering challenges and ultimately propose a compelling conceptual DEMO design.

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Development of tungsten fibre-reinforced tungsten as plasma facing material for DEMO

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Towards realizing fusion based energy systems the so-called DEMOnstration power plant is planned to be the first fusion reactor producing electricity. For the first wall material of DEMO unique challenges require complex features in areas ranging from mechanical strength and thermal properties to the typical plasma wall interaction issues retention and erosion.
Tungsten (W) is currently the main candidate material for the first wall due to its unique property combination e.g. the high melting point and the high temperature strength as well as excellent erosion resistance and low tritium retention. However, tungsten is brittle at room temperature and susceptible to embrittlement, i.e. by overheating or/and neutron irradiation. Tungsten fibre-reinforced tungsten composites (W_f/W) utilize extrinsic mechanisms to improve the fracture toughness and thus mitigate the brittleness problem of W. Thereby, an application as a plasma facing material under thermal transients and neutron bombardment becomes feasible [1].
A next step in the development process of W_f/W is the investigation of its behaviour within a fusion environment. In this contribution, we will discuss the development approach and present the first results of investigations of the hydrogen retention, erosion and irradiation effects. The interrelation of these properties as well as the complex microstructure of the composite material are key aspects. To study the retention, W_f/W model systems were specifically designed to investigate the respective influence of the microstructure and especially the composite constituents (fibre, interface layer and matrix). These samples are loaded in a deuterium plasma and afterwards examined by nuclear reaction analysis and thermal desorption spectroscopy. Bulk material has been exposed in the linear plasma device PSI2 to determine erosion under synergistic loading conditions. This initial test will clarify how fibres and matrix material are being eroded under these conditions. In order to study irradiation effects, ion irradiation was chosen as a substitute for the complex and time consuming neutron irradiation. 21 MeV W^{+6} were used to irradiate 5 µm thin tungsten wire allowing damaging of the complete volume. These fibres were then tension tested to investigate the influence of the irradiation damage on the mechanical properties. Different to conventional tungsten no sign of degradation was detected in first tests. It is planned to load irradiated fibres by deuterium to investigate the synergetic effects of irradiation, H retention and mechanical properties. The first steps of this study will be presented. As the behaviour of the 5µm wire is strongly related to the bulk behaviour of W_f/W the results will be used to predict the behaviour of bulk material.

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Two order of magnitude stress reduction in a 3D-printed tungsten/liquid lithium divertor target.

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3D-printing is a novel manufacturing technique that offers many potential benefits for fusion applications, such as the possibility to print functionally graded materials, or easily produce optimized geometries for individual divertor tiles. Regarding tungsten though, there are large concerns regarding the strength of the printed material. This work aims to demonstrate the usability of 3D-printed tungsten in a liquid lithium divertor (LLD) target, where strength requirements may be less restricting.

First 3D-printed tungsten is obtained from Philips Medical Systems [1] and tested for strength and thermal diffusivity. Most notably, the ultimate tensile strength is found to be around ~250MPa. The ductile phase is completely absent despite testing at 600℃. This may be due to low relative density of only 93%.

Next, an existing LLD concept is optimized and tested in linear plasma device Magnum-PSI. The “pre-filled” concept originally designed for NSTX-U [2] is chosen for simplicity. Here, lithium is supplied to the textured plasma facing surface via a wick that leads to a reservoir. This is a passive process driven by capillary forces. The optimized divertor target design uses a specific internal structure that can only be manufactured through printing. Two functions are fulfilled: First, the lithium flow to the surface is optimized and estimated at maximum 24 L/m^2s compared to 0.47 L/m^2s in the original design. Second, *thermal stresses in the component are reduced to levels < 8 MPa!* This is two orders of magnitude lower than expected for the ITER monoblocks [3].

The optimized target is realized by Philips Medical Systems and tested in linear plasma device Magnum-PSI [4] with power densities estimated up to 17.5 MW/m^2. SEM imaging of the target surface, and cutting the target to evaluate the internal geometry both show that the target was not damaged during plasma exposure.

In answer to the original research goal: 3D-printing can indeed be used in liquid lithium divertor targets. However, much more important: 3D printing has allowed a design where thermal stresses are reduced by more than two orders of magnitude! This enables many new possibilities in the material choice. E.g. a material could be chosen which is much weaker, but that is perhaps able to deal with neutron damage through continuous recrystallization.

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Initial testing of two liquid lithium based PFCs, LiMIT and FLiLi, inside a toroidal fusion device, HIDRA

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One of the great challenges in developing a viable nuclear fusion reactor and power station is the development of plasma facing components (PFC) that can survive the extreme environments they are exposed to. Currently, there are no viable solid materials that can take the next step from the current machines to future reactors, such as ITER and DEMO [1]. New solutions are being investigated, such as using liquid metals at the first wall and divertor [2]. Because of its low Z and ability to remove impurities, lithium shows great promise as a main candidate, and has shown improved stored energy in the plasma as well as better control of Edge Localized Modes (ELMs) [3]. Both the University of Illinois at Urbana-Champaign (UIUC) and Princeton Plasma Physics Laboratory (PPPL) have developed liquid lithium based PFCs, intended to be used as limiters and develop designs that will be eventually used at the wall and divertor of a fusion device. The PPPL design, the Flowing Liquid Lithium (FLiLi) consists of a flat plate on the surface of which liquid lithium would flow [4], while the UIUC design, the Liquid Lithium Metal Infused Trenches (LiMIT) [5], incorporates trenches on its surface and uses the thermo-electric magneto hydrodynamic effect (TEMHD) to flow the liquid lithium and maintain a fresh lithium surface.

These two PFCs have been manufactured, and tested inside the Hybrid Illinois Device for Research and Applications (HIDRA) [6-8], a stellarator/tokamak hybrid device at UIUC. For these tests, HIDRA was operated as a stellarator and used to develop and benchmark the LiMIT and FLiLi limiters. This paper presents the results coming from these experiments.

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Plasma impurity co-bombardment effects on sputtering of Beryllium and Tungsten

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In future fusion reactor, ITER, plasma facing materials (PFMs) will consist of tungsten (W) and beryllium (Be). Ions from the plasma as well as injected noble gas impurities (argon (Ar) and neon (Ne) here) as a coolant for the plasma, will lead to sputtering of PFMs. To study the effect of plasma impurities on the erosion and surface morphology of wall materials, molecular dynamics (MD) simulations were carried out. Therefore, we modeled irradiation of both W and Be surfaces with Ar-deuterium (D) and Ne-D mixtures, varying the fraction of Ar and Ne impurities from 0 to 20 percent with impact energies of 10-100 eV at 500 and 800 K surface temperatures for W and impact energies of 30-200 eV at 400, 600 and 800 K surface temperatures for Be.

In both materials, after a few hundred bombardments the sample surface was damaged and cell structures changed from crystalline to amorphous at lower ion energy and blistering-like effect was observed due to D_2 accumulation in the Be cells at higher energies. In W only the noble gas impurities were responsible for surface erosion in the energy range studied here and the sputtering mechanism was in physical region. For Be at impact energies higher than 100 eV, total Be sputtering yield, in the presence of Ar and Ne impurities is around three times higher than pure deuterium irradiations. The effect of surface temperature on the results is negligible here.

Abstract Withdrawn

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Investigation of TEMHD flow through large-pore metallic foams for use in plasma facing components

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Liquid metal plasma facing components (PFCs) have been shown to provide several benefits over traditional solid PFCs. These include providing a constantly refreshing, self-healing surface that can prolong device lifetime, decreasing edge recycling, reducing impurities in the core, and enhancing plasma performance. The Center for Plasma-Material Interactions (CPMI) at the University of Illinois at Urbana-Champaign (UIUC) has developed and extensively tested the Liquid Metal Infused Trenches (LiMIT) concept, showing passive thermoelectric magnetohydrodynamic (TEMHD) drive of liquid lithium through solid trenches at heat fluxes up to 3 MW/m^2 [1]. As the LiMIT system moves toward full-scale application, the technology must be able to cope with much higher heat fluxes, especially in the region of the divertor heat stripe. One difficulty that arises is the phenomenon of lithium dryout, which exposes solid trench material due to strong local TEMHD acceleration in the areas with the highest heat flux. A free surface model of this effect was developed at CPMI, and used to ascertain the propensity of trench shaping designs to mitigate the dryout problem [2]. It was found that inserting a step in the bottom of the trench just past the highest heat flux should effectively eliminate dryout for both slow flow (1 cm/s) and medium flow (10 cm/s) cases, and trench shaping solutions can be tailored for desired flow speed and heat flux. The LiMIT system was then modified and tested under electron beam impingement to physically test these modeled designs.

While these solutions can be tailored to a specific set of operating conditions, once built they may not be fully applicable to a wide range of heat fluxes and flow speeds. A better way to ensure lithium dryout does not occur, while still providing TEMHD flow, may be through the use of large pore metallic foams [3]. Similar to the passive capillary action seen in the capillary porous system (CPS) concept, the additional wicking of the lithium and its adherence to the structure can maintain the lithium level in the face of high heat fluxes that would cause dryout in an open surface trench-based system. Coupling this benefit with TEMHD pumping will allow for lithium recirculation and replenishment to avoid passivation and counter evaporation and ablation at the surface. The pore size of the foam can be controlled in order to maximize pumping while maintaining the benefits of capillary action. The results of these studies and the viability of this concept as a TEMHD-driven, liquid lithium PFC will be discussed.

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Generation and Transport of Metallic Impurities during the exposure of Liquid Metals to Hot Plasmas in TJ-II.

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Alternative to the use of refractory metals, mainly tungsten and its alloys, as target materials in a future fusion reactor, liquid metal (LM)-based concepts have been put forward in the international fusion community [1]. The motivation for that is the lack of permanent damage by plasma loads and neutrons as well as the possibility of *in situ* replacement inherent to these concepts. However, compared to the *standard tungsten solution*, to be implemented in ITER, these designs still require extensive R+D including the choice of the particular metal to be exposed to the divertor plasma. Due to the stringent requirements imposed by physical and engineering constrains, only three LM's are considered at present [2]: Lithium, tin and their alloys. One of the criteria used for the selection is the compatibility of the chosen element with the plasma as well as its tritium retention characteristics.
Several preliminary experiments have been already performed in the TJ-II stellarator aimed at these issues [3,4], and very promising results were obtained. Motivated by these results, a full campaign of comparative Li/ LiSn/Sn testing in TJ-II plasmas has been initiated. Solid and liquid samples of the three candidates have been exposed to the edge plasma in a Capillary Porous System (CPS) arrangement and the associated perturbation of the core plasma has been recorded. The surface temperature of the liquid metal/CPS samples (made of a Tungsten mesh impregnated in Li, LiSn, or Sn) has been measured during the plasma pulse with ms resolution by pyrometry and the radial profiles of Li, Li^+, Sn and Hα were recorded together with the electronic edge parameters. A simple 1D modelling was applied to the data, allowing for the evaluation of the kinetic energy (E_k) of the ejected atomic species as well as their screening at the edge by monitoring the ratio of first ion/neutral emission intensities. A clear evolution of E_k with sample temperature was deduced for Li samples, associated to the different relative contribution of sputtered/evaporated atoms, while a rather low (thermal) energy was always deduced for the LiSn case. Similar Li emission intensities from Li and LiSn samples were recorded even when their temperatures during plasma exposure were much higher for the latter. Also, different Li^+/Li ratios (screening) and normalized Hα emission intensities were observed in both cases. The recycling characteristics of tin samples at several temperatures together with the degree of plasma contamination by tin were also recorded and analysed.
In this presentation a full account of the results obtained and their implications for the use of LM/CPS concepts in a future Fusion Reactor will be addressed.

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Experimental studies on tungsten produced by powder injection molding as plasma-facing materials

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Tungsten is envisaged as plasma-facing material in fusion reactors because of its small tritium retention and low erosion rate as well as its high melting point and high thermal conductivity. However, it is very hard and brittle, which makes it difficult and expensive to fabricate and prone to crack formation under transient heat loads. The first disadvantage can be ameliorated using Powder Injection Molding (PIM) as fabrication route[1]. With its near-net-shape precision, the method offers particularly the advantage of cost saving. Furthermore PIM is an ideal tool for scientific investigations and efficient production of new oxide and carbide doped materials.

In this contribution, we report on plasma exposure of pure tungsten produced via PIM (sintered at 2400 °C, density 98.6 - 99%, with equiaxed grains) in the linear plasma device PSI-2[2] using deuterium and neon plasmas (to enhance physical sputtering) with a moderate plasma flux density of $4x10^{21}$ $m^{-2}s^{-1}$ to the targets. For the neon plasma exposure, the targets were biased to obtain an ion impact energy of 110 eV and the fluence was $1.6x10^{25}$ m^{-2}, for deuterium to 200 eV at a fluence of $5.2x10^{25}$ m^{-2}, respectively. The sample temperature has been kept to 150-200°C (neon exposure) and 420-450°C (deuterium exposure), respectively. In addition, the samples have been exposed to transient heat loads by a Nd: YAG- laser to simulate ELM-like heat pulses of 0.38 GWm^{-2} and a duration of 1 ms with a frequency of 0.5 Hz. 1000 pulses have been applied with and without plasma exposure. Reference samples (Plansee W, density > 99.97%, rolled, with a grain elongation perpendicular to the loaded surface and W with density > 99.95%, rolled, grain elongation parallel to the loaded surface) were exposed under the same conditions for comparison. Net erosion has been deduced from the measured mass loss, the surface roughness by laser profilometry and the resulting fuel inventory has been determined by nuclear reaction analysis and thermal desorption spectroscopy. The surface morphology has been analyzed prior and after the exposure by scanning electron microscopy.

We observe in all cases an enhanced erosion yield of the PIM material up to a factor of 3, the response of the material to the transient heat loads is similar in terms of roughness and surface morphology with a larger damage during neon exposure compared to deuterium exposure. Fuel retention in PIM-W shows strong variation not correlated with material density but with carbon content from remainders of binder. For PIM-W samples with lowest carbon content fuel retention is comparable to that in reference samples

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Additive Manufacturing of Tungsten for Plasma-Facing Component Application

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The refractory metal tungsten (W) is regarded as preferred plasma-facing material (PFM) for future magnetic confinement thermonuclear fusion devices. The reasons behind this choice are mainly that W exhibits a high threshold energy for sputtering by hydrogen isotopes as well as a low retention of tritium within the material. However, W is an intrinsically brittle metal with high hardness which means that the processing and machining of W is difficult and expensive. Furthermore, this also implies that very simple geometries, e.g. flat tiles or simple monoblocks, are typically used for W parts in plasma-facing components (PFCs).

Against these limitations, additive manufacturing (AM) technologies could be a highly versatile and innovative approach for the realisation of W elements for PFC applications. The characteristic feature of AM processes is that three-dimensional objects are created by sequential layerwise deposition of material under computer control which means that such a technology is capable of producing objects with more or less arbitrary shape.

Within previous work [1], it was found that bulk pure W can be consolidated by means of direct laser beam melting (LBM) with a comparably high relative mass density of more than 98%. However, it was also found during these investigations that pronounced crack formation can occur within the material as selective LBM processes typically induce high thermal gradients during bulk material manufacturing.

In this context, the present contribution will summarise topical results regarding the AM of pure W by means of powder bed based LBM for material manufactured by using preheated W substrates with temperatures of up to 1000°C. These elevated substrate temperatures are intended in order to maintain the ductility of the material during the LBM process to in turn minimise the formation of crack defects.

Furthermore, it will be discussed how the versatility of AM for producing tailored W structures can be exploited for PFC design in order to realise a high-integrity PFC with high heat removal capability and results regarding the manufacturing of a corresponding PFC mock-up will be presented.

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Abstract withdrawn

Crystallographic analysis of corrugations and nano-tendril bundle growth on tungsten exposed to helium plasma*

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Recently, we discovered nano-tendril bundle (NTB) growth on tungsten (W) surfaces exposed to RF helium (He) plasma [1]. These structures expand the evolutionary tree of structures that form on W surfaces saturated with He, which already included corrugation patterns and W fuzz. It has been shown that NTB growth can be more rapid than W fuzz growth and may represent another pathway for erosion of the surface [2]. Tungsten NTB growth, therefore, needs to be considered when evaluating plasma facing component operation in future fusion energy devices.

In this work, the propensity for NTB versus W fuzz growth on polycrystalline W under varying ion energy modulation conditions, from DC to peak-to-peak energy modulation of 42 eV at 13.56 MHz, is correlated to the crystal orientation of the underlying grains. Grains that are vicinal to crystal orientations with high surface diffusivity (e.g. {101} for bcc crystal structure) exhibit NTB growth at lower ion energy modulation amplitude than grains that are vicinal to low surface diffusivity orientations, such as {111} and {100} plane orientations.

A topographic instability model based on the balance of smoothing by surface tension and roughening due to surface diffusion barriers is applied to the corrugation patterns that develop in parallel to NTB growth and prior to W fuzz growth. A relationship between the corrugation wavelength versus surface temperature under otherwise identical ion irradiation conditions is derived from the model in which the sole fitting parameter is the surface diffusion barrier amplitude. The value for the surface diffusion barrier amplitude on surface orientations for which at least one of the major crystallographic directions in the plane is the close-packed direction <111> is found to be 0.145 eV, which is a reasonable value when compared to available data in the literature.

These results support that surface diffusion enhanced by ion bombardment plays a key role in the surface morphology evolution of W under He irradiation.

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Numerical analyses of CFETR scenarios with impurity seeding by the integrated COREDIV code

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The China fusion engineering test reactor (CFETR), as a next step fusion facility, has been proposed to bridge the physics and technology gaps between ITER and DEMO [1]. It is well known that for future tokamak reactors, reduction of the power delivered to divertor plates down to an acceptable level for existing materials is a critical issue. The radiative exhaust of energy in the pedestal and scrape-off layer (SOL) areas by intrinsic and in particular by externally seeded impurities is considered as a possible approach to spread energy over wider wall area. Since the energy balance depends strongly on the coupling between the core and SOL plasma, joint treatment for both regions is necessary.

In the present work, the integrated COREDIV code [2], which solves self-consistently radial 1D energy and particle transport of plasma and impurities in the core region and 2D multi-fluid transport in the SOL, is used to analyze the CFETR scenarios with different types of impurity seeding (Ne, Ar, Kr). The simulations are performed based on the CFETR steady-state scenarios: baseline scenario (phase I) and advanced scenario (phase II). Tungsten is assumed to be the divertor target material. It is found that for phase I, seeding by all considered impurities leads to an efficient mitigation of the power flux to divertor plates. The power flux to divertor plates and plasma temperature at the divertor target decrease with higher impurity seeding, due to the increasing of total radiation. On account of the simultaneous effects between reduction of plasma temperature and increment of impurity concentration near the divertor plates, higher impurity seeding can firstly increase and afterwards suppress the tungsten production from target and corresponding tungsten concentration in the core. For the cases of Ne and Ar seeding with sufficient flux, operation in semi-detachment condition is predicted. However, achievement of this condition will lead to significant reduction in fusion power and Q-factor (from $Q \approx 3.7$ to $Q \approx 2$) due to plasma dilution ($Z_{eff} \approx 5$). In the case of Kr seeding, low plasma temperature at divertor target can be obtained with low plasma dilution in the core. For phase II scenario (with higher plasma current I_p and averaged density $\langle n_e \rangle$), the fusion power is effectively enhanced ($Q \approx 16$). The total radiation power (about 180 MW) appears to be almost independent of the Ne puffing level and is dominated by W radiation ($\approx$ 70%). For both scenarios, when the divertor heat load is reduced to an acceptable level with impurity seeding, large radiation fractions ($f_{rad} > 80\%$) are achieved which are dominated by core radiation. Since core radiation is affected by the parameters such as impurity pinch velocity, radial transport coefficients in the SOL, and tungsten prompt re-deposition, sensitivity analyses for these parameters can help to find appropriate operational windows for the CFETR scenarios.

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Effects of double-layer tungsten coatings on hydrogen isotopes plasma- and gas-driven permeation through F82H

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Reduced activation ferritic steel alloys (RAFSs) such as F82H (Fe-8Cr-2W) are the candidate materials for the first wall of DEMO reactors [1]. For the blankets employing self-cooled breeder, the first wall is supposed to be subjected to bi-directional hydrogen isotopes permeation [2, 3]: in one direction by edge plasma-driven permeation (PDP) of deuterium as well as tritium into blankets, and in the other direction by bred tritium gas-driven permeation (GDP) into the edge plasma. It is important to note that deuterium PDP will hinder the recovery efficiency of tritium from the breeder and will probably necessitate isotopes separation as well. On the other hand, tritium GDP, essentially the same effect as gas-puff for fueling except that it is uncontrollable, will lead to an unwanted increase of the particle recycling in the first wall region, which could even affect core confinement performance as well as isotopes mixture imbalance. Because of their critical importance to steady-state reactor operation, these technical issues must be clearly resolved.
Tungsten (W) has been proposed as a candidate plasma-facing material for the ITER divertor because of its beneficial properties including high melting point, high thermal conductivity and low sputtering yield [4]. For a DEMO reactor, surface coatings made of W are necessary to protect the plasma-facing wall made of RAFSs such as F82H. In our previous work, hydrogen isotopes PDP and GDP through F82H coated with two different types of W coatings, i.e., sputter-deposited W (SP-W) and vacuum plasma-sprayed W (VPS-W) have been studied [5-8]. VPS-W coatings have been found to have a low density structure with connected pores, while SP-W coatings have a dense and pore-free structure. Experiments indicated that VPS-W coatings suppress hydrogen isotopes PDP from the plasma side to the blanket side, but does not suppress GDP in the opposite direction. In contrast, SP-W coatings suppress GDP, but, interestingly, tend to enhance PDP, which has been observed not only in the VEHICLE-1 linear plasma facility but also in the QUEST spherical tokamak. Therefore, double-layer VPS/SP-W coatings are proposed to reduce the bi-directional hydrogen isotopes permeation. Hydrogen isotopes PDP and GDP through VPS/SP-W coated F82H are investigated in this work and the effects of double-layer W coatings on hydrogen PDP and GDP are discussed, which is important to the plasma-wall interaction studies for a DEMO reactor.

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Effects of internal stresses on the irradiation resistance of tungsten-copper films

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This study investigates the effects of internal stresses on the irradiation resistance of tungsten-copper films. Tungsten films deposited on copper substrates were prepared by magnetron sputtering deposition at 400 °C. X-ray diffraction measurements shows the presence of internal stress in the as-prepared films due to the difference of thermal expansion coefficient. 38eV helium ions were introduced by plasma immersion at the a fluence of 1.0×10^{26} ions m^{-2}. After irradiation, the stresses became much smaller. Self-interstitial atoms tend to form clusters by attracting themselves. As the helium atoms increasingly accumulate, small He bubbles attract interstitial He atoms and the stress increase until saturation [1]. The stress leads to plastic deformation of the surface, which finally causes the release of stress and the swelling of W surface. This study might provide some ideas for the design of radiation-resistant nuclear fusion materials by properly control the stress of materials.

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Plasma exposure behavior of molybdenum and graphite in the EAST tokamak

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The effects of long-term plasma exposure on the microstructure of molybdenum (inboard/high field side surface) and two graphite (lower divertor and inboard) tiles in the EAST tokamak were investigated, including the surface damage, re-deposition of various metallic elements and hydrogen isotope retention. It is found that the high-flux transient heat induced by high flux deuterium plasma can result in the surface cracking with typical lengths on the order of one millimeter and melting at the edge of molybdenum tile, whereas no such cracking was found in both graphite tiles. Plasma exposed two graphite tiles exhibit ~8 μm and 60 μm thick damaged layers, respectively, in which the structure of graphite is transformed to amorphous carbon. Re-deposition of Fe, Mo, W, Ni, Cu, Ti and Cr, as well as hydrogen isotope retention are found in the surface layers of graphite tiles, whereas no significant re-deposition was found in molybdenum tile. This study suggests that sputtering is the dominated effect for molybdenum tile while the deposition effect is dominated for both graphite tiles. Furthermore both material properties and location in tokamak can influence the surface damage induced by the plasma exposure.

Effect of 800 keV argon ions pre-damage on the helium blister formation of tungsten-tantalum alloys exposed to 60 keV helium ions and its impact on helium retention

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We report the effect of Ar^{8+} ions pre-damage on the following He^{2+} irradiation behavior of tungsten-tantalum alloy (with 5 mass percent of Ta). In this paper, we compared the irradiation resistance performance against 60 keV He^{2+} ions of undamaged W-Ta alloy with that of pre-damage alloys with were preliminary exposed to 800 keV Ar^{8+} ions at a fluence of 4×10^{19} ions m^{-2}. The surface modification was studied with scanning electron microscopy (SEM), and helium retention was measured by thermal desorption spectroscopy (TDS). To better illustrate the densification characteristics, cross section imaging was performed by the focused ion beam (FIB). Our results demonstrate that the helium blistering of W-Ta could be effectively relieved by the Ar^{8+} ions pre-damage, while the helium retention was found to be systematically higher in the pre-damage W-Ta than that of the undamaged alloys. The Ar^{8+} ions irradiation-induced damage altered the morphology of helium bubbles in tungsten-tantalum exposed to the following He^{2+} irradiation significantly. The intensity of helium release peak at relatively high temperatures (>800 K) was enhanced due to Ar^{8+} ions pre-damage.

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Plasma Fuelling, Particle Exhaust and Control, Tritium Retention

Determination of retained tritium from ILW dust particles in JET

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Quantitative analyses of retained tritium on dust particles produced from plasma facing walls are important issues for safe maintenances in ITER and DEMO. In JET, the ITER-Like Wall (JET-ILW) experiment is a real simulation of plasma facing materials towards ITER; hence, the understandings of produced amounts, deposited areas, tritium amount and irradiation level of dust particles are required.

The first campaign of the JET operation (19.5 h of plasmas in 2011-2012) with ILW was followed by the retrieval of selected in-vessel materials for detailed ex-situ studies [1]. The first observations of total retained tritium from dust particles were done for ILW dust particles using the combustion method in QST, Rokkasho Fusion Institute. For ILW dust particles, specific activities of 750 MBq/g at inner divertor and 5.9 MBq/g at the outer divertor were observed. For carbon wall dust particles, specific activities of 5.5 MBq/g at inner divertor, 3.3 GBq/g at in/out louvres and 100 MBq/g at the outer divertor were observed.

The specific tritium activity of ILW dust is similar to dust from the carbon wall, however the total activity due to dust in the ILW is two orders of magnitude lower than that in carbon wall. This is due to a decrease in dust production greater than two orders of magnitude in the ILW (<2 g) compared with the carbon wall (>200 g). Carbon dust particles in the ILW dust samples have been shown to contain the highest amounts of fuel compared with beryllium and tungsten based particles. Such carbon particles are likely to be those remaining from the carbon wall era and probably account for the high specific activity of the ILW dust sample [2]. However there is evidence of impurity mixing with beryllium and tungsten dust particles [3-4] which will affect retention in dust. Further statistical analysis is required to evaluate the extent to which mixed material dust particles contribute to overall fuel retention.

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Evidence of D concentration driven trap formation in W

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The prevailing view on the retention of hydrogen isotopes (H, D, T) in tungsten (W) is that upon implantation the isotopes diffuse away from the surface encountering and filling pre-existing trap sites, thereby forming a diffusion front that progresses into the W. The pre-existing trap concentration is typically assumed to vary from 10^{-5}–10^{-3} intrinsic sites per W atom based on measured hydrogen depth profiles [1], and these values are typically required to model the hydrogen isotope uptake with a diffusion-reaction code like TMAP [2]. In a recent article, however, Gao *et al*. [3] show the existence of a 10 nm thick D-supersaturated surface layer with an unexpectedly high D concentration of ~10 at.% after irradiation with low energy ions, and suggest a synergistic interaction of implanted D ions and solute D atoms with the W lattice. In their proposition, solute D atoms prevent the recombination of vacancies with interstitial W atoms, produced by collisions of the D ions with W lattice atoms. The formed Frenkel pairs then become further D trapping sites, thus driving the D surface concentration. In the experiments to be reported, we show that a mechanism which drives hydrogen isotope trapping, but which is not understood, is also operative beyond the implantation zone, through a combination of experiment and thermal release modelling with the TMAP code.

In the experiments, two sets of three identically prepared W samples were produced and exposed to various plasma-material interaction regimes. Four were exposed at 560 K to 80 eV D ions to a fluence of 3×10^{21} m^{-2} in a D_2 plasma in the PISCES-A plasma device. Two of these are used as control samples for TDS analysis to examine the thermal release from the W only, while the other two samples are then coated with a layer (5–10 nm) of Be or W co-deposition using a dual sputter magnetron source [4] prior to the TDS analysis. The remaining two non-plasma exposed samples were also coated to obtain the TDS release from just the co-deposit layers. Modelling the TDS results with TMAP suggests that the pre-existing trap concentration of the coated, but not plasma exposed W substrate, is below 10^{-6} sites per W atom. However, for the same W material exposed only to D plasma, TMAP modelling can only reproduce the TDS release if the preexisting trap concentration is increased to above 10^{-5} sites per W atom. Release of D from the Be or W codeposit layer into the plasma exposed substrate during TDS results in an increase of the trap concentration to 10^{-3} sites per W atom to correctly model the release behavior. Taken as a whole, the results strongly suggest evidence of D concentration driven trap formation in W. Additional results, including NRA D depth profiles will be presented at the conference.

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Measurements of gap deposition profiles of different shapes of castellated tungsten blocks in KSTAR

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Deposition inside gap of castellated structures, such as toroidal pumped limiter (TPL) of Tore Supra (TS) and ITER divertor, can cause significant amount of fuel retention [1]. For instance, in TS DITS (deuterium inventory in Tore Supra) campaign it shown that more than 50 % of retention was caused by co-deposition inside the gaps of CFC (Carbon-Fiber-Composite) finger tiles [2]. Castellated CFC blocks in TPL have conventional square shape, while ITER divertor blocks will have beveled structure to protect leading edges of individual block. The aim of this paper is to study on the gap deposition profiles depending on the shaping of plasma facing components (PFCs): conventional square-shaped, trapezoidal, ITER chamfer, and rounded shape. Some of blocks were fabricated with intentional misalignments of various heights of leading edges (0.3, 0.6, 1.0, 2.0 mm). These tungsten block tiles are exposed to L- and H-mode discharges during 2015-2016 campaign [3], and carbon density inside the gaps were measured by EDX.

The carbon surface density inside of gaps is in a range from 7.0×10^{15} C atom/cm^2 at neat the top down to 1.0×10^{15} C atom/cm^2 at 5 mm from the top. It seems that 1.0×10^{15} C atom/cm^2 would be background level. Depending on the shape of the block, the deposition profiles inside the gap show different trends. Gaps behind leading edges show almost the same deposition profiles and distributed symmetrically on both open and shadowed side. Such profiles are obtained when the deposition was dominated by charge-exchange (CX) neutrals. Gaps of conventional square-shaped show asymmetric deposition profiles contributed by both ions and CX neutrals. The measured gap deposition profiles show qualitatively good agreement with modeling [4]. Although modeling of the deposition profiles is still on going and very difficult because of long exposure time (during a whole campaign), those profiles give an idea how the deposition profiles inside castellated tungsten blocks would be, i.e. those profiles are unique database to be modeled and compared with that of other machines.

Another interesting observation is that about 2 mm of very front top surface of most of blocks in the toroidal direction has no deposition. Two explanation can be given: 1) the area is the hottest spot on the top surface due to ion bombardment leading to thermal desorption of hydrogen and carbon atoms, 2) Ions with Larmor orbit effect can jump the poloidal gap and hit the surface, which enhances the erosion at the surface of the blocks.

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Fuel retention across the mid-plane of Outer and Inner Poloidal Limiters tiles in JET – summary of JET-ILW campaigns: ILW-1, ILW-2 and ILW-3

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Since installation of the ITER-Like Wall Plasma Facing Components (ILW PFC) at the Joint European Torus (JET), three plasma campaigns were run with this wall configuration: ILW-1 (2011-2012), ILW-2 (2013-2014) and ILW-3 (2015-2016). A global long term retention of deuterium (D) assessed based on results from the set of available samples from post-mortem analysis for ILW-1 and ILW-2 is ~0.2% of the D puffed into JET and is mainly held in deposits in the JET divertor. This paper presents new results of D retention across the mid-plane of the main chamber Be tiles (Outer Poloidal Limiter - OPL and Inner Wall Guard Limiters - IWGL) obtained from post-mortem analysis after ILW-2 and recent results available from ILW-3. The main points of the paper are:

- The results obtained by means of Ion Beam Analysis (IBA) and Thermal Desorption Spectrometry (TDS) analysis indicate the fuel retention patterns are similar for ILW-1, ILW-2 and ILW-3 across the mid-plane of the limiter tiles: namely low fuel retention in the central region of each tile – the points of plasma contact, where erosion and temperature are greatest – with higher concentrations of D at the ends of each tile, which is generally associated with re-deposition.
- The ILW-1 and ILW-2 plasma times were similar, the main difference between them however was that ILW-2 ended with an H-fuelled campaign, in which 10% of the total number of ILW-2 pulses were performed, whereas ILW-1 was predominantly a D-fuelled plasma campaign. Based on TDS of the Be samples it can be assessed that there is 3-10% more hydrogen in the ILW-2 samples than in the ILW-1 sample, however the overall D retention is comparable.
- The comparison between results obtained by TDS and IBA shows that the D inventory calculations for IBA and TDS agree within a factor of ~1.5 – 2.5. This error is considered acceptable as the main source of error in global inventory calculations is associated with the extrapolation from very few data points to a large surface area.
- Additionally, the D retention measured by IBA in deposits from the wing part of the OPL tile from ILW-2 is presented before and after annealing to 775°C in the TDS system – results not obtained before for post-mortem samples.

Displacement damage recovery in ultra-fine grain tungsten

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One of the main advantages of using tungsten (W) as a plasma facing material (PFM) is its low uptake and retention of tritium. However, in high purity (ITER grade) W, hydrogenic retention increases significantly with neutron induced displacement damage in the W lattice. This experiment examines an alternative W grade PFM, Ultra-Fine Grain (UFG) W, to compare its retention properties with ITER grade W after displacement damage up to 0.6 dpa (displacements per atom). Using deuterium (D) as a proxy for tritium, we find that displacement damage in UFG W creates trapping sites with lower binding energies, since this additional trapped inventory can be released by baking at low temperatures (< 500 ºC). Conversely, displacement damage in ITER grade W has an increase of higher binding energy trapping sites that require baking temperatures greater than 750 ºC to recover the trapped inventory, which may not be practical in a working reactor. Although the total trapped inventory in UFG W was 20% higher than ITER grade W in the undamaged case and 10% higher in the damaged case, this excess inventory is released at temperatures below 500 ºC. We conclude that UFG W has better retention properties after undergoing displacement damage and can keep tritium retention lower than with ITER grade W with low temperature baking.

In our experiment, we used 2 mm thick, 6 mm diameter, samples of ITER grade W and UFG W. Our UFG W samples have TiO_2 dispersoids at the grain boundaries that prevent the 1 μm grains from recrystallizing. Some samples were pre-damaged up to 2 μm from the surface with a 12 MeV silicon ion beam. All the samples were exposed together in the DIII-D divertor near an attached lower single null outer strike point to L-mode D plasma discharges using the Divertor Material Evaluation System (DiMES) manipulator. The peak plasma flux was 2×10^{22} $D^+/m^2/s$ for a fluence of 7×10^{23} D^+/m^2 in 10 discharges. Depth profiles of D were examined with Nuclear Reaction Analysis (NRA) up to 3.5 μm, and total D retention was measured with Thermal Desorption Spectrometry (TDS). NRA data show that D diffused deeper into the damaged UFG W samples, but TDS data show that most of the trapped inventory is easily recovered from damaged UFG W at baking temperatures below 500 ºC. For every sample, there are two major D_2 flux peaks in the TDS data around low (150 ºC) and high (750 ºC) temperatures. Displacement damage increased the high temperature peak in ITER W, which indicates an increase of in-grain vacancy traps (>1 eV), but the damage increased the low temperature peak in the UFG W samples. The increase of the low temperature peak may be a result of increased D trapping in low-energy grain boundary traps (<1 eV). This could be explained by displacement damage defects assimilating into a nearby grain boundary instead of remaining in the grain as a lattice vacancy. Diffusion modeling is underway to quantify these trapping energies.

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Hydrogen Isotope Exchange in Tungsten at Low Temperatures

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One possibility to reduce the tritium inventory of plasma-facing tungsten in future nuclear fusion reactors prior to opening the vacuum vessel is to perform hydrogen isotope exchange [1]. In this process a radioactive tritium (T) atom retained in a trap is replaced by a non-radioactive deuterium (D) or protium (H). Despite the extensive investigation of hydrogen isotope exchange in tungsten the underlying microscopic exchange mechanism remains unclear. Most of the available models consider the exchange empirically by the introduction of exchange cross sections or probabilities. A widely used microscopic model for hydrogen trapping in tungsten, termed classical model, assumes single occupancy of traps with hydrogen atoms and a fixed de-trapping energy depending on the trap type [2]. This model can explain the observed hydrogen isotope exchange only in very special cases and fails when the temperature is too low to allow thermal release of hydrogen atoms from the traps. In contrast to that, the recently developed fill-level model [3], which assumes multiple occupancy of traps and a de-trapping energy that decreases with increasing trap occupancy, is able to explain hydrogen isotope exchange on a microscopic scale also at low temperatures.
In order to test the fill-level model the exchange of D by H in tungsten is investigated in-situ in dedicated experiments with nuclear reaction analysis and mass spectrometry at low temperatures ranging from 150 to 290 K. In the first experiment the isotope exchange is investigated at 150 K for different D inventories prior to the exchange. The results show that a certain critical H fluence needs to be implanted before a decrease of the D inventory is observed. Furthermore, the lower the initial D inventory the larger the required critical H fluence. In addition, the decrease of the D inventory, once the critical H fluence is exceeded, is independent of the initial D inventory. In a second experiment the exchange of a pre-defined D inventory at different temperatures is studied. Also in this scenario, a critical H fluence is observed, which decreases as the exchange temperature increases. Furthermore, the decrease of the D inventory is stronger for higher exchange temperatures.
Finally, the experiments are simulated with the diffusion-trapping code TESSIM-X [3] in which the classical as well as the fill-level model is implemented. The observed dependences of the isotopic exchange on the initial D inventory and on the exchange temperature can be very well described with the fill-level model, whereas the classical model fails to reproduce the experimental results. However, since the exchange at low temperatures takes predominantly place within the implantation zone, kinetic de-trapping of D by energetic H cannot be neglected. When accounting for this effect in the simulations, also the classical model can describe the observed isotope exchange. This implies that an ensemble of classical traps with kinetic de-trapping behaves similar to a fill-level trap.

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3D numerical simulations by FEM of diffusion and transient hydrogen trapping processes in plasma facing components

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Understanding and controlling hydrogen isotope (HI) inventory is key for future fusion devices both to limit in-vessel retention and increase the tritium breeding ratio. HI trapping can also lead to embrittlement of plasma-facing materials. Simulations of diffusion and trapping of HI in metals is commonly made using a full diffusion-trapping process kinetics description. Specific macroscopic rate equations (MRE) codes [1,2,3], developed for plasma surface interactions, and based on the McNabb and Foster equation [4], allow to describe transient trapping kinetics. Most of theses MRE codes are dedicated to 1D problems, in which the influence of 3D defects (like gas bubbles) could be estimated [5]. Full 3D configurations, however, can not be simulated.
In this work, a full 3D MRE modelling is proposed based on a finite element method (FEM). The model, implemented in the 3DS Abaqus software [6], uses a generalized transport equation which accounts for mechanical fields, trapping, and their evolution with time [7,8,9]. To ensure the solution convergence and numerical stability, the transient trapping kinetic is introduced by using an approximation of the analytical solution the McNabb and Foster equation [10].
The implementation in a 3D FEM code of this original approach is first validated by the comparison with results obtained by the MRE 1D code called HIIPC [2] (validated by experimental confrontations) for specific 1D test cases. For theses test cases, the diffusion and trapping of plasma-implanted deuterium in tungsten and iron are simulated for selected durations where trapped and mobile concentrations are not at equilibrium (*i.e.* the Oriani's formulation [11] is not enough to describe the trapping phenomena [12]). The model is validated by comparison with experimental temperature programmed desorption [13]. Lastly, hydrogen inventory results are shown for relevant 3D structures: tungsten castellated structures and an ITER stainless steel diagnostic first wall.

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Tritium retention in W plasma-facing materials: impact of the material structure and of He and He-D irradiation

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Plasma-facing materials for next generation fusion devices, like ITER and DEMO, will be submitted to intense fluxes of light elements, notably He and H isotopes (HI). Our study focuses on tritium (T) retention on a wide range of W samples: first, different types of W materials were investigated to distinguish the impact of the pristine original structure on the retention, from W-coated samples to ITER-grade pure W samples submitted to various annealing and manufacturing procedures, along with monocrystalline W for reference. Then, He and He-D irradiated W samples were studied to investigate the impact on He-damages such as nano-bubbles (exposures in LHD or PSI-2) or W fuzz (PISCES-B exposure) on T retention.

We exposed all the samples to tritium gas-loading using a gentle technique which is not introducing new damage in the material. Tritium desorption is measured by Liquid Scintillation counting (LSC) at ambient and high temperatures (800°C). The remaining T inventory is then measured by sample full dissolution and LSC. Results on T inventory (activity/exposed sample surface) is observed to vary greatly depending on the material structure with a retention of 0.29 MBq/mm^2 for pure ITER-like W but only 0.09 MBq/mm^2 for monocrystalline W for instance. The presence of He bubbles in the close vicinity of the surface increases the T quantities trapped as gas (HT or T_2) in the material. The initial conditions of the sample surface also appear to have a huge impact on T desorption: for instance, a preliminary reduction of the native tungsten oxide at the sample surface allows a decrease in HTO desorption by a factor 3. The presence of W fuzz does not change T desorption dynamic, but while tritium desorbing as gas stays in the same order of magnitude compared to the original W material, HTO desorption at ambient temperature decreases by a factor 7, highlighting the presence of more permanent trapping sites for T.

To complement our T retention study and investigate potential isotopic impacts, D inventories were measured on the same set of samples by Temperature Programmed Desorption (TPD) at PIIM. D and T profiles were measured by Nuclear Reaction Analysis at JSI to allow an insight on the long term trapping sites localization. These results were coupled with modelling efforts to address the general tendencies of tritium trapping in W and extrapolate potential inventory for future full W machine.

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SOLPS-ITER analysis of nitrogen seeding interruption in JET H-modes

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High-power regimes of future fusion devices with water-cooled tungsten (W) divertor plasma-facing components will require (partially) detached divertor operation. On ITER, this will be achieved with extrinsic impurity (nitrogen (N), neon (Ne) and/or argon) injection. In order to thus control the divertor detachment, the plasma response to sudden changes in impurity injection rates must be known. A first step was taken for ITER using a series of time-independent SOLPS-4.3 plasma boundary simulations [1] with varying Ne injection, but a true time-dependent calculation is ultimately required. This is challenging at the ITER scale, but is more tractable for smaller devices.
To provide key benchmark data for such modelling, a dedicated experiment was performed on JET, where the divertor N injection (2e22 elec/s) was abruptly cut during a deuterium-fueled Type-I ELMing H-mode discharge (2,2 T, 2.0 MA, 8 MW NBI, $H_{98\text{-}y} = 0.8$, outer strike-point on the horizontal bulk W target). The injected N leads to partial divertor detachment, as seen in the rollover of the inter-ELM ion flux to the targets and decreasing ELM frequency, measured by embedded Langmuir probes at fixed negative bias. Some 300 msec after the rollover began, the nitrogen gas injection was interrupted. Within 100 msec of the seeding gas cut, the plasma starts to re-attach, the ion flux to the outer target and ELM frequency increase, and the pre-roll-over state is recovered in ~500 msec. The NII emission (500.40 nm) moves towards the X-point during the N seeding, then back towards the strike-point when the seeding is removed. These data provide valuable input as to the resilience of the detached state and the extent to which N remains in the system (partial recycling) following the gas input removal.
On the basis of these experimental observations, we use the SOLPS-ITER [2] code package to model the time-dependent JET divertor plasma response. We consider N_2 injection from the horizontal target, in the common flux region, as in the experiment. The simulations include all-metal walls, but neglect sputtering and the ELM transients. Fluid drifts are not activated in the interest of reducing the required CPU time. We first model the initial (detached) and final (re-attached) stages of the impurity injection interruption experiment, for which the upstream pedestal and SOL profiles show no significant differences, allowing the same profiles of cross-field heat and particle transport to be maintained throughout. The time-dependent simulation bridges the period of the gas injection interruption and the divertor plasma response. Sensitivity to the strength of N recycling is explored.

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Enhanced plasma and surface capabilities with beam fueling and heating in the Lithium Tokamak Experiment-Beta (LTX-β)[1]

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The Lithium Tokamak Experiment (LTX) has investigated the use of solid and liquid lithium coatings on metal surfaces surrounding a tokamak, leading to observation of the low-recycling, hot-edge, flat temperature profile regime after the termination of edge fueling [1,2]. An upgrade, LTX-β, will add a neutral beam to provide core fueling and achieve more stationary plasma density under low-recycling conditions without edge puffing of cold gas. LTX-β widens the LTX parameter space over many dimensions, primarily plasma performance, magnetic field, and wall conditioning. Additional plasma heating, density, temperature, and duration will increase heat and particle flux and fluence. Improved field coils, power supplies, magnetic sensors, and a plasma feedback control system will control the limiting surface location and eventually allow a separatrix. New between-shots midplane Li evaporation and Li granule injection, as well as enhanced pumping and bakeout, will better control the condition of the Li coatings.

The impact of these operational improvements on the low-recycling regime will be characterized by an enhanced diagnostic set. Development of the previously observed hot, low density edge will be studied with an enhanced Thomson scattering system with greater sensitivity, especially with new polychromators viewing the edge. Rotation, unimpeded by neutral drag, will be driven by the beam and measured with charge exchange spectroscopy, along with impurity temperature and density. The effects of the lithium coatings on core impurities, impurity sources, and recycled hydrogen will be investigated with doppler spectroscopy, AXUV diode arrays, resistive bolometery, filterscopes, and high-speed filtered cameras. A suite of Langmuir probes, a retarding field energy analyzer, and sample exposure probes will characterize the low collisionality scrape-off layer and unique lithium surfaces in LTX-β [3,4].

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Impact of divertor target material on recycling and discharge fueling during the full ELM cycle*

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Simultaneous ELM-resolved measurements of Balmer-α and Fulcher-band radiation from both D atoms and D_2 molecules, respectively, recycling at the outer strike point (OSP) were conducted in DIII-D during Metal Rings Campaign (MRC). The MRC involved operation with two toroidally continuous rings of W-covered TZM (Mo alloy with 0.5% titanium and 0.08% zirconium) inserts in the lower divertor. It was found that the relative fraction F of D atoms originating from D_2 molecules in the total recycling D flux changes during ELMs. Between ELMs, F on carbon and on tungsten is F_C~60% and F_W~85%, respectively, consistent with expectations if all atomic recycling is due to reflections. During ELMs, F dropped to F_C~40% and F_W~60%.
This effect has been studied with a variety of samples (C, Mo, uncoated and W-coated TZM, W, W fuzz, and Ti) exposed using Divertor Material Evaluation System (DiMES) manipulator in the lower divertor near an attached outer strike point in L-mode discharges. External voltage bias of square waveform between +10 V and -150 V with frequency 10 Hz was applied to the sample to investigate the dependence of atomic and molecular recycling on the D^+ ion impact energy (E_i). It was found that an increase of E_i by ~160 eV due to the bias leads to a transient increase of the recycling fraction above unity, similar to [1]. The flux of D_2 in contrast to D only showed a transient increase on C where ion induced D_2 desorption is an important channel of D_2 re emission [2]. Thus, the surface material and the ion impact energy as well as the surface temperature [3] are important factors in controlling the fraction of recycling molecules. This result has implications for both divertor detachment and pedestal fueling, as reflecting D atoms have longer ionization length and contribute to density pedestal recovery after an ELM. D_2 molecules, on the other hand, aid detachment and produce cold Frank-Condon atoms upon dissociation. During the MRC, with the total fraction of the W-covered area on each metal ring ~0.6% of the total wall area, the effect of W in the divertor could be seen in a ~10% increase of the line averaged density when OSP was placed on the W ring. This is qualitatively similar to what was seen on ASDEX upon a complete change from C to W PFCs [4]. We also present results of EDGE2D-EIRENE simulation assessing the effect of adding W in the divertor on divertor fueling and modification of the temperature and density profiles.

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Comparison of particle control potential using lithium injection and cryopumping in the EAST tokamak

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The use of both lithium wall coatings and divertor cryopumping systems are being studied in EAST, offering a unique comparison of the two particle control techniques. Here we analyze the particle removal efficiency of these two techniques in EAST.

The injection of lithium powder provides active wall conditioning in EAST, via reducing hydrogen recycling that extrapolates to long-pulse by continually replenishing the pumping lithium surface. The reduced recycling is evidenced by strongly reduced D_α emission in discharges with strong lithium injection, while the divertor ion flux measured shows small changes. The change in the divertor recycling coefficient has been analyzed using data-constrained SOLPS simulations [1]. Assuming Z_{eff}=1, a 20% reduction in the divertor recycling coefficient is inferred in the case with strong lithium injection, indicating efficient deuterium removal using lithium.

As part of an upgrade to the EAST lower divertor, an optimization of the cryopumping design has been performed, with a semi-analytic pumping model [2], using measured EAST divertor density, temperature, and particle flux profiles. Pumping under a 'dome' in the private flux zone is analyzed, including options for pumping either the inner or outer divertor leg, as well as an additional duct on the scrape-off layer side of the outer divertor. The optimized design includes a duct length on the outer leg of 5-10cm in order to provide sufficient reduction of the back-conductance to the plasma, to achieve high pressure in the pumping volume. The resulting design shows the ability to efficiently pump a wide range of outer strike point positions, with a plenum pressure approaching 1mTorr. 4% of the incident ion flux onto the divertor can be pumped, modestly below the pumping capability of the lithium injection approach.

These two techniques extrapolate quite differently to reactor conditions. Since lithium can be expected to bind at most one deuterium atom per lithium atom, achieving the needed hydrogen removing rate effectively sets the minimum lithium injection rate, whereas conventional pumping depends on the expected divertor neutral pressure and pumping speed. A comparison of these two approaches for achieving the same removal rate at reactor-like conditions will be presented.

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Fuel retention during ELM suppression by RMP in KSTAR

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For a tokamak fusion reactor, the control of edge localized modes (ELMs) in high-confinement mode (H-mode) operation is an essential requirement to avoid excessive heat and particle fluxes onto the plasma facing components (PFCs). So ITER will utilize RMP for ELM suppression. The ELM suppression is achieved by RMP in KSTAR since 2011 [1]. The control of long term fuel retention is one of the most critical issues for both ITER and other Tokamaks aiming at operating in steady-state conditions [2-4]. Furthermore, excessive tritium accumulation in the first wall will introduce a safety problem for the next fusion device, such as ITER, for instance, the limit of maximum tritium retained in ITER is 700g [3, 4]. Wall retention affects fuelling efficiency, plasma density control, and the neutral particle density in the plasma edge, which in turn affects plasma confinement [2]. Therefore, the investigation of fuel retention during ELMs suppression by RMP is of interest for ITER.

In this work, we have investigated the wall retention with and without RMP in KSTAR 2016 experiment campaign by particle balance. Particle balance is a simple but powerful method to trace the balance between injected particles and exhausted particles. We have compared the fuel retention with and without RMP under different plasma parameters (plasma current, electron density, pulse length and etc). Since H-mode discharges in KSTAR were driven and fueled mainly by NBI, the retention is strongly affected by NBI power input. In some cases, total retention level exceeds 5.4×10^{22} D atoms in a long pulse shot (longer than 25 sec) with a retention rate of 2.0×10^{20} D/s which is the saturation level of KSTAR inner graphite wall due to strong co-deposition. In ELM suppressed and mitigated shots, plasma density was pumped out while the background of D_α signal increased (peak levels by ELMs decreased), although the total radiated power was decreased. It is observed that there is no change of retention rate with or without RMP. Those results indicate that global characteristics of the plasma in terms of fuel retention calculation might be the same: Deposited power at inner and outer strike point, radiated power, plasma stored energy may be balanced by different manner with RMP which cannot be distinguished by simple particle balance retention model.

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3D modelling of the fuel retention on a fuzzy tungsten surface morphology

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Tungsten has been widely considered as a primary candidate material for the plasma facing components in next-step fusion reactors due to its high melting point, high thermal conductivity, low sputtering yield and tritium retention. However, experiments performed in linear helium-containing plasma devices indicate that a fiber form nanostructure, so-called “fuzz”, is generated on tungsten surface under certain conditions: the surface temperature is from 1000 to 2000K and the incident ion energy is higher than 30 eV. The diameter of each tendril is roughly 10-50 nm and up to micron in length [1]. The nanorods structure dramatically change the tungsten morphology which leads to the decrease of optical reflectivity [2] and thermal conductivity by several orders of magnitudes [3]. It also can cause the enhancement of tritium retention and tungsten release. In addition, the unipolar arc on the tungsten fuzz surface is easily ignited in response to the pulse irradiation [4]. Thus, the mechanism of the growth of the tungsten fuzz and its properties are needed to be studied.

In this study, a three-dimensional (3D) kinetic Mont Carlo (KMC) code, SURO-FUZZ (upgrade version of SURO code [5-8]) has been developed to get a tungsten fuzzy nanostructure in micro- and second-scales formed by the bombardment of helium ions. The growth rates of fuzz layer thickness in different temperatures are in good agreement with the experimental results. The retention rate of deuterium ions and the deuterium particles density distribution in a non-erosion tungsten fuzz morphology are calculated and also compared with the experimental results.

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Effects of Argon on Deuterium Retention in Polycrystalline Tungsten

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A burning plasma fusion device, such as ITER, is expected to experience localized melting in the tungsten divertor without impurity seeding [1]. However, introducing new gas species into the system will result in significant changes to the plasma-surface interactions in the divertor, including likely effects on the retention of deuterium and tritium in tungsten. Studies on argon, one of the candidates for impurity seeding, and its effects on deuterium retention in tungsten are few [2][3]. The study by Ishida, et al. [2] measured the permeation flux of deuterium under mixed D-Ar irradiation of tungsten at 500 to 1000 K, finding a decrease of permeation flux at higher temperatures compared to permeation flux for D-only. Another study by Zhu, et al. [3] observed an increase in deuterium retention in argon pre-irradiated tungsten upon exposure to deuterium plasma at 575 K, suggesting the deuterium occupation of both low-energy and high-energy trapping sites resulting from argon damage. However, there exists no comprehensive study of the effects of Ar on D retention in W under both sequential and simultaneous D-Ar ion irradiation in the literature. Performing a rigorous study on argon and deuterium with tungsten will provide a more complete picture of the mechanisms involved and the viability of argon as an impurity seeding candidate in ITER-like fusion devices.

In this study, we examine the effects of D-2.5% Ar ion irradiation on polycrystalline tungsten at 300-700 K under D-only, Ar-only, sequential, and simultaneous (SIM) irradiation. Ion energies of 500 eV/D^+ and 500 eV/Ar^+ are produced using a dual-beam ion accelerator, and the D retention measured by thermal desorption spectroscopy (TDS). We have found that the shapes and temperatures of the peaks in the Ar TDS spectra are consistent with a 1 keV Ar^+ implanted polycrystalline tungsten wire experiment by Kornelsen [4]. Comparing the SIM D-Ar results with previous studies with Ne and He, Ar appears to be more effective than He and comparable to Ne in reducing deuterium retention at 500 K [5]. At 300 K, deuterium retention is increased somewhat, as compared to the D-only case. However, at 500 K, deuterium retention was decreased by 40% or more. These results suggest that the dominant mechanism for deuterium retention changes significantly as the tungsten specimen temperature is increased above 300 K in the SIM D-Ar case. The retention of argon, however, is comparable to neon and less than helium at these temperatures. Two possible mechanisms are suggested for the observed effects on deuterium retention, with the relative dominance of each mechanism dependant on the tungsten temperature: sputtering from argon ions (which removes the implant layer), and the inhibition of deuterium diffusion due to D-Ar interactions at the near-surface region of the tungsten.

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Investigation of hydrogen isotope retention mechanisms in beryllium and beryllium tungsten mixed materials

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During operation of the nuclear fusion experiment ITER beryllium-tungsten mixed layers can be formed due to erosion and redeposition. Simultaneously, hydrogen isotopes will be implanted in plasma facing components (PFC) and mixed material layers or can be retained in codeposits. In this work, the retained amount as well as the temperature dependent release of hydrogen after implantation is investigated, which is of importance regarding the fuel balance in the plasma and possible changes of physical properties of the materials caused by hydrogen uptake. To predict the behavior of the PFC under the wide range of conditions that are to be expected in a fusion reactor, an understanding of the dominating mechanisms of hydrogen transport in the material and retention after implantation is required.

With this aim, the retention and temperature dependent release of hydrogen isotopes in Be-W mixed materials under well defined conditions is studied using the ultra high vacuum experiment ARTOSS. With this facility, the in situ preparation and analysis of atomically clean surfaces at a base preassure of 2 x 10^{-11} mbar is possible. The sample surface is cleaned by Ar^+ sputtering and subsequent 1000 K annealing cycles. The chemical surface state can be investigated by X-ray photoelectron spectroscopy. The ion bombardment caused by the hot plasma is simulated with a mass and energy separated ion source. Retained amounts of deuterium (D) are attained by nuclear reaction analysis (NRA). Additionally, information about the binding states of the retained D is accessed by thermal desorption spectroscopy (TDS). Different linear heating rates can be realized between 0.005-10 K/s up to 1000 K using an electron impact heater.

Experiments have been carried out utilizing the described experimental procedure for pure polycrystalline Be and W surfaces. Dynamic outgasing of deuterium at 300 K and 268 K after implantation with an energy of 1 keV/D is investigated to confirm low energy binding states of D in W. For W and Be samples, deuterium ions are implanted with different fluences starting from 10^{18} m^{-2} up to 10^{22} m^{-2}, revealing low energy binding states in Be for higher fluences. Additionally, a variation of the used temperature ramps is performed. Especially using very slow ramps of 0.01 K/s makes it possible to resolve at least three low energy binding states of 0.43 eV, 0.67 eV and 0.82 eV for the first time. Further investigation of the Be sample surfaces after implantation using SEM images shows that the low energy binding states seem to be not connected with the growth of Be-D dendrites on the surface. Instead, the surface of our sample shows a fluence dependent blistering effect, which is discussed to be the reason for the low energy binding states. Furthermore, a 600 nm layer of tungsten was deposited on a beryllium polycrystal substrate using a magnetron to create a 5 µm thick layer of $Be_{12}W$. During the annealing process, RBS measurements are performed to control the formation of the mixed phase. Retention mechanisms of D in the formed $Be_{12}W$ layer are investigated by means of TDS, NRA and SEM. The experimental results are finally compared to coupled reaction-diffusion modeling including parameters obtained from density functional theory calculations.

Effect of RMP application on neutral fueling and exhaust in MAST

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The application of resonant magnetic perturbations (RMPs) leads to reduced particle confinement times τ_p obtained from a global, single reservoir particle balance analysis. In L-mode plasma up to a 15% reduction in τ_p is measured, and in H-mode plasma a similar level of τ_p reduction is seen, however, the exact value depends on the RMP mode number and phasing. This establishes a clear link of the plasma response to the RMP fields. During all discharges with pump-out, the ionization source increased as the RMPs are turned on, but the confinement time decreased substantially enough to cause an overall density decrease.

RMPs are used to control edge localized modes (ELMs) and induce in most circumstances at different devices a reduction of the total number of confined particles N_{tot}, called density pump-out. The results presented in this paper relate this pump out for the first time for MAST to the neutral fueling and exhaust fluxes using the single reservoir, global particle balance. This particle balance was assembled using the plasma density and D_α emission measured by filter-scopes and a calibrated 1-D camera, as well as local values of S/XB coefficients determined by edge plasma parameter measurements, to infer the particle flux loss from the plasma and the incoming neutral recycling flux maintaining the plasma density.

In order to resolve the underlying effects in the neutral fueling and exhaust household inside the recycling and ionization layer, a multi-reservoir particle balance model [1] was revived, which includes both molecular and atomic species as well as the plasma and wall inventory. This model allows for inputs from experiments like fueling from gas puffing and neutral beam injection and the previously determined particle confinement time τ_p. With this model, the time evolution of the plasma density, neutral pressure in the vacuum vessel, and the D_α emission along a tangential chord in the plasma edge as a proxy for the recycling flux were accurately reproduced. This allows to use the underlying atomic and molecular physics constraints in the model to infer the impact of the RMP fields on the neutral fueling and exhaust dynamics. Because the 3-D plasma boundary induced with RMP fields changes the local plasma conditions inside of the ionization region, this interface is of seminal importance to understand the particle pump out. The model used requires an estimate of parameters like the probability of particles adsorbing on the wall and the fraction of ionized particles that remain in the confined plasma reservoir. These additional parameters enable to reconstruct changes in these terms and their relation to the particle pump out in a unique way. The results from this experimental analysis with both particle balance models are compared to results from numerical analysis with the EMC3-EIRENE code. Initial results from this comparison supports increased fueling efficiencies and reduced particle confinement times as a reason for the observed particle pump out.

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Abstract Withdrawn

Production of ITER-relevant Be-containing laboratory samples for fuel retention investigations

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Since 2014, a large project has been running under the EUROfusion Consortium to produce ITER-relevant test samples for fuel-retention studies. The strategy is to deposit mixed, Be-containing coatings at the National Institute for Laser, Plasma and Radiation Physics in Romania and distribute the samples for analyses and/or ion-implantation in partner laboratories. The composition, thickness, and surface structure of the deposits have been varied to study their influence on the efficiency of D retention and to make predictions for ITER. For benchmarking purposes, samples resembling the co-deposited layers on the inner divertor of JET during its ITER-Like Wall (ILW) campaigns [1] have also been produced. The properties of the samples have been determined using a variety of surface-analysis tools including Rutherford Backscattering Spectroscopy (RBS), Nuclear Reaction Analysis (NRA), Time-of-flight Elastic Recoil Detection Analysis (TOF-ERDA), Secondary Ion Mass Spectrometry (SIMS), Thermal Desoprtion Spectroscopy (TDS), and Laser-Induced Breakdown Spectroscopy (LIBS).

The focus has been put on D-doped Be-O, Be-W, Be-C-O (in the case of JET-ILW comparison) coatings with different surface morphologies in the nanoscale and thicknesses ranging from about 0.1 μm to some 15 μm. The relative D content of the samples can routinely be increased to 5-10 at.%, and in some coating types even 40 at.% has been reached. This is a good starting point to fabricate co-deposits with seeding gases and/or helium in the future. The Be-O-C-D coatings most closely resembling the JET-ILW co-deposits (O and C content 5-10 at.%, D content ~5 at.%, thickness ~15 μm) show very similar release behavior of D as real JET samples, indicating that the laboratory samples well mimic the structure of co-deposits in fusion reactors.

The surface analyses have revealed that increasing the O content from a few to 50 at.% in otherwise identical samples lowers the amount of D that can be retained, by up to a factor of 10. Implantation with D^+ ions results in similar retention behavior, though with more peaked D profiles, than direct doping during the deposition phase. The data also indicate that more D can accumulate in the sample if the thickness of the coating is increased, the surface becomes more modified and rough, or, in the case of mixed Be-W deposits, the relative Be fraction increases.

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Tritium distributions on W-coated divertor tiles and selected Be tiles used in the third JET ITER-like wall campaign

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Integrated tests of ITER reference materials (Be and W) have been performed in JET ITER-like wall (ILW) campaigns [1]. The authors have measured tritium (T) distributions on the W-coated CFC divertor tiles used in the first ILW campaign (2011–2012, ILW-1) and the second campaign (2013–2014, ILW-2) using an imaging plate (IP) technique. The IP images showed significant enrichment of T on the horizontal parts of the upper inner diver tiles and the shadowed region of the floor tiles due to co-deposition with Be and impurities [2,3]. In this study, T distributions on the divertor tiles and selected Be tiles used in the third ILW campaign (2015–2016, ILW-3) were examined using the IP technique.

To avoid contamination with T and Be, IPs were wrapped with thin polyphenylene sulphide (PPS) films, and then put on the tiles in the dark for 17–20 hours. After removing PPS films, the 2-dimensional distributions of intensity of the photo-stimulated luminescence (PSL) were analysed using a laser scanner. The W divertor tiles analysed were the high field gap closure tile (Tile 0), the inner vertical tiles (Tiles 1 and 3), the inner floor tile (Tile 4), the outer floor tile (Tile 6) and the outer vertical tiles (Tiles 7 and 8). The Be tiles examined were the outer limiter (4D14), the inner limiter (2XR10) and the upper dump plate. The locations of tiles can be found in Ref. [1]. Tritium detected in this study was produced by DD fusion reactions and therefore present as a minor isotope of hydrogen in the vacuum vessel.

As observed after ILW-1 and ILW-2, the T concentrations on the inner divertor tiles were higher than those of the outer divertor tiles. However, the distribution of T on the inner divertor tiles was more uniform than in the cases of ILW-1 and -2; T concentrations on the vertical tiles were relatively high, while those on the horizontal parts of upper divertor tiles and the shadowed region of floor tile were lower than the values after the previous campaigns. The T concentration on the outer limiter was higher than on other analysed Be tiles by orders of magnitude. The non-destructive depth profile analysis using β-ray induced X-ray spectrometry [4] is in progress and result will be also reported in the presentation.

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Modelling of the effect of ELMs on fuel recycling at the bulk W divertor target of JET

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JET tokamak has been operated successfully with its all-metallic ITER-Like Wall (JET-ILW) configuration (Be main chamber limiters, W divertor) showing a reduction in the global long-term fuel retention by factor of 10-20 and a decrease in the global fuel retention rate by a factor of >18 as compared to the all-carbon JET wall (JET-C) operations [1, 2]. Recently, assessments on the short-term retention in the JET-ILW divertor target have been done using spectroscopic methods as part of a dedicated experimental campaign with repeated plasma scenarios [3]. The results with steady-state plasmas showed the gradual increase of the W target base temperature during operation having a decreasing effect to the fuel retention. Further, the Edge Localized Modes (ELMs) were shown to increase the local fuel recycling at the W target. However, the underlying mechanism affecting the local fuel recycling and retention during steady-state and ELMy plasmas remains as a speculative issue. The present work focuses on the local fuel retention and fuel recycling by studying computationally the bulk W target properties affecting the recycling at the divertor target during typical JET steady-state L-mode plasmas and H-mode plasmas with type I ELMs.

The computational methodology comprises of a multi-scale method solving set of Rate Theory equations [4] describing all the plasma-material interaction events taking place during the inter-ELM and intra-ELM phases. Real experimental plasma parameters and reported target surface temperature evolution (Tile 5 stack C in Ref. [3]) have been used as input in the calculations. The methodology describes the sub-surface defect evolution influenced by an ELM, and the calculations have been extended covering JET pulses in full thus allowing a realistic assessment for the fuel recycling and release. For the steady-state L-mode pulse the low-energy fuel particle flux to the target was taken from the corresponding Langmuir probe at the strike point. For the calculations of the H-mode pulse with type I ELMs, the ELM frequency and energy, the inter-ELM energy, and the particle fluxes to the target were collected from BeII photon emission at the target and electron energy at the pedestal, and from the Langmuir probe data at the target, respectively. The results show the effect of surface temperature evolution to the fuel release dynamics during the plasma shot. Moreover, the effect of the ELM crash has a major role in the local recycling. The energetic intra-ELM particles get implanted and create sub-surface damage in the target, which act as additional fuel trapping sites. The inter-ELM low-energy fuel particles refill the near-surface region prior the next ELM crash. The H-mode local recycling is found to be highly complex interplay between the inter-ELM near-surface absorption, intra-ELM deep sub-surface implantation, defect creation and desorption.

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Determining fundamental transport parameters of hydrogen isotopes in tungsten

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Future fusion devices will use the hydrogen isotopes deuterium and tritium as fuel. The first-wall material will likely be tungsten for which retention and transport of hydrogen isotopes needs to be predicted.

The key quantity for transport is the diffusion coefficient. The generally accepted value for diffusion of protium in tungsten stems from Frauenfelder and was measured in the late 60ies [1]. Subsequently derived experimental values scatter by several orders of magnitude, trapping effects are presumably the reason. However, recent simulations and reviews also question Frauenfelder's value for the migration barrier of 0.39 eV [2, 3]. Furthermore his experiment was for protium only and is scaled by the inverse square root of the mass ratio to yield the value for deuterium. This is widely believed to be a valid scaling. However, a direct measurement would reduce the number of needed assumptions.

The objective of this study is to re-measure the solubility and derive the diffusion coefficient of protium in tungsten at temperatures between 1000 and 3000 K to resolve the discrepancy. In addition, the value is also derived for deuterium in tungsten. At temperatures above ~1400 K trapping effects are of vanishing importance and pure diffusion is the governing transport effect. Hence, a high-temperature ultra-high vacuum (UHV) experiment was designed and built. An induction furnace in combination with a water-cooled quartz glass tube is used for hydrogen gas loading at pressures of up to one atmosphere. Following the loading of single- and polycrystalline tungsten samples thermal desorption spectroscopy (TDS) is used in-situ to measure quantitatively the amount of the retained gas species. Finite-element modelling (FEM) of the loss of hydrogen reveals that outgassing during cool down cannot be neglected. For the parameters of Frauenfelder's experiment it predicts a loss of 14% and hence the solubility values derived need to be corrected for that.

The solubility of deuterium in poly-crystalline tungsten is currently measured for a set of pressures and with the theoretical values. The diffusion coefficient of deuterium in tungsten will be calculated afterwards. The setup was successfully commissioned and first measurements with deuterium are promising. First, with the given setup temperatures of up to 3000 K are reachable and second, the very low deuterium background signal allows to measure down to 1200 K. Both facts allow to extend Frauenfelder's parameter space extensively.
The same set of measurements will also be done for a single-crystal and compared.

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Comparison of the structure of the plasma-facing surface and tritium accumulation in beryllium tiles from JET ILW campaigns 2011-2012 and 2013-2014

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ITER-Like-Wall project has been carried out at Joint European Torus to test plasma facing materials relevant to International Thermonuclear Experimental Reactor – ITER. The first wall of the vacuum vessel is made of bulk beryllium tiles, whereas for the divertor bulk tungsten and tungsten coated carbon fibre composite tiles are used. Beryllium was selected due to low Z, good thermal conductivity, and high oxygen gettering characteristics. [1] Material erosion and tritium accumulation are the main issues related to performance of plasma facing materials in the vacuum vessel of fusion reactors. Thermal transient loads cause heating of beryllium surface and results in significant changes – material loss, melting, cracking, evaporation, formation of dust. Moreover, hydrogen isotopes retention in the wall materials and erosion product also takes place. [2]

During the shutdowns in 2012 and 2014, selected beryllium tiles were removed from the vacuum vessel. In this study, tiles from three positions were analysed and compared analysis results regarding both their position in the vacuum vessel and differences in the exploitation conditions during two campaigns. Tiles were from the inner wall (Inner Wall Guard Limiter), outer wall (Wide Poloidal Limiter) and upper region (Upper Dump Plate) of the vessel.

Two methods were used to determine tritium content in the samples – dissolution under controlled conditions and tritium thermodesorption. Prior to tritium determination, scanning electron microscopy and energy dispersive x-ray spectroscopy were used to study structure and chemical composition of the plasma-facing-surfaces of the beryllium samples.

Experimental results revealed that total tritium activity in the samples obtained is in range of several kilo Becquerel per square centimetre of the surface area. For example, in beryllium tiles from the shutdown in 2012 the largest tritium activity was found in the samples from outer wall 3.3 – 31.9 kBq/cm^2, activity decreases in inner wall 1.0 – 6.4 kBq/cm^2 and the least from the upper region tile middle part has tritium activity of 3.3 kBq/cm^2.

Results obtained from scanning electron microscopy has shown that samples surface morphology for beryllium tiles obtained during shutdown in 2014 are similar to samples surface morphology for beryllium tiles obtained in 2012.

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Hydrogen isotope exchange in ion-irradiated tungsten

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Tungsten, a candidate material for plasma-facing walls of fusion reactors, has suitable properties of high heat resistivity and low sputtering yield, while there is a concern of a large tritium inventory which leads to a significant amount of tritium loss due to radioactive decay and a more careful handling for radiological protection. Moreover, irradiation defects produced by D-T neutrons would trap tritium and increase the inventory [1]. When the plasma exposure is stopped and subsequently exposed to vacuum, tritium in the traps is detrapped. When the plasma particle is changed from tritium to hydrogen, tritium in the traps is exchanged to hydrogen. In both cases, the tritium inventory would be decreased. In the present work, effects of vacuum exposure and hydrogen plasma exposure is experimentally studied using deuterium as an analog of tritium.

In the experiment, a sample of tungsten disk was irradiated with 4.8-MeV tungsten ions to produce radiation defects at room temperature. An average damage was 2.6 dpa. Next, the sample was exposed to deuterium plasma and after the deuterium concentration reached the steady state, the sample was exposed to vacuum or hydrogen plasma. During the exposure, a beam of He-3 was injected to the plasma-exposed side of the sample and deuterium concentration was continuously observed by use of a nuclear reaction analysis.

The result showed that the deuterium concentration in the traps was drastically decreased by the hydrogen plasma exposure. This cannot be explained when detrapping of hydrogen isotope from the trap is independent. The most plausible explanation is that one trap site can trap more than one hydrogen/deuterium atoms and the detrapping energy decreases with increasing the number of trapped atoms [2]. In case of vacuum exposure, deuterium atom becomes difficult to be detrapped due to decrease in the number of trapped hydrogen and corresponding increase in the trapping energy. In case of hydrogen plasma exposure, hydrogen atom is trapped to a vacant trapping site. Since this exchange process keeps the number of trapped atoms, deuterium atom can be detrapped with low trapping energy.

Taking into account the above consideration, experimental results of another sample, which was preliminarily damaged to 5.9 dpa, were analyzed and it was found that the trapping energy varied from 1.8 to 2.4 eV, depending to the number of trapped atoms. The analysis also reproduced the thermal desorption spectra of deuterium from the 5.9 dpa-sample, which was not able to be explained by a single trapping energy.

From the above results, it was concluded that hydrogen/deuterium plasma exposure is more effective for removal of tritium from the radiation-damaged tungsten wall than vacuum exposure.

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Interaction between hydrogen clusters and point defects in W: an atomistic simulation

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Tungsten (W), as a plasma facing materials, will be exposed to hydrogen (H) plasma in future fusion devices. The supersaturated H will cluster and form a bubble in W. H bubbles will have effect on the microstructure and properties of W, such as reducing the thermal conductivity and sputtering threshold of W. The influence of H is directly related to the interaction between H clusters and point defects in W, which is still unclear so far.
We employ a first-principles method in combination with molecular dynamics simulation to investigate the dynamic behavior of H-defects as well as clusters at W surfaces. An EAM potential [1] of W-H system self-developed has been used, and several equilibrium conditions of H nucleation are involved into AKMC models [2].
H atoms are shown to prefer clustering in both bulk and the sub-surface of W. H and W atoms are in a ratio of 1:1 based on the binding energy of H clusters. The average binding energies of H clusters increase with the increasing of H number, in which dozens of H atoms clustered with a total binding energy up to 20 eV. H clusters show scattered distribution at 500 K and 1 at% of H, leading to the surface expansion by a vertical stress. The presence of vacancy consolidates the supersaturated H clusters with strengthening the anisotropic stress. Furthermore, it is found that there is strong attraction between H and self-interstitial atoms, and then W atom migration in the <111> direction will be restrained. The self-interstitial atoms are easy to be trapped by H clusters, which could block the recombination of Frenkel defects. Supersaturated H can induce defect growth and surface deformation.

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Abstract Withdrawn

Supersonic molecular beam injection in the tandem mirror GAMMA 10

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This study reports application of supersonic molecular beam injection (SMBI) to the tandem mirror device GAMMA 10, aiming at (1) understanding the SMBI fueling properties using different type of nozzles, (2) constructing a modeling method of the SMBI fueling and (3) producing high density plasmas for the sake of divertor simulation in GAMMA 10/PDX [1]. Since the total machine length between both ends of GAMMA 10 is 27m, trials of fueling at several locations are important for the optimization of particle fueling. In this study, we have carried out the SMBI fueling at two positions, the central-cell midplane [2,3] and the inner transition region of the anchor-cell.

In the central-cell midplane, three types of SMBI fueling with different nozzle (Laval, straight and without nozzle cases) have been tested to understand the fueling property [2,3]. Horizontal and vertical images of visible light emission have been captured with a fast-framing camera using a dual-branch fiber bundle. The directivity of injected gas has been calculated from the spatial profile of the light emission. In the Laval and straight nozzle cases, the directivity improves when the plenum pressure of SMBI up to 1MPa and it saturates more than 1MPa. To model the injected SMBI gas using a numerical calculation, the neutral particle transport simulation code has been modified [2,3]. The simulation results indicate that the SMBI fueling using Laval nozzle at the plenum pressure condition more than 1MPa has directivity of 1/5 as compared with the normal gas fueling with a conventional Piezo-electric valve. In the case of no nozzle at low plenum pressure around 0.3MPa, on the contrary, the directivity is almost the same as the conventional gas fueling. The line-averaged electron density of the central-cell increases more than twice after SMBI, while the decrease in the stored energy is significant due to the large charge-exchange loss of the ICRF heated fast ions confined in the central-cell.

Recently we have installed SMBI with a new Laval nozzle in the inner transition region of anchor-cell in order to reduce the charge-exchange loss of the fast ions. In this case, the reduction in the central-cell stored energy has been mitigated as compared with the SMBI at the central-cell midplane. We have observed a fast time response in the lost-ion flux to the mirror ends due to SMBI. The amount of lost-ion flux can be controlled by the injection condition of SMBI. Although the energy distribution of lost-ion flux should be investigated, this fast response of the lost-ion is applicable for the experimental simulation of edge-localized-mode.

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Abstract Withdrawn

Tritium penetration by isotope exchange in tungsten

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Tritium (T) transport in tungsten (W) will be governed by its interaction with other isotopes since T will always diffuse in the presence of deuterium (D) in a fusion device. The corresponding transport picture is a series of thermally activated diffusion, trapping, and de-trapping processes between the solute and trapped concentrations. A trap site that traps one hydrogen isotope is inconsistent with the experimental observation that D isotope exchanges with H at temperatures where thermal de-trapping is not allowed. This has led to the proposal of a fill level dependent multi-occupancy trapping model with recent attempts at experimental validation [1]. To provide insight into T transport by isotope exchange in W, we report on novel D-T and complementary D-H isotope exchange experiments.

Tritium penetration profiles in W samples with various microstructures were investigated at 300 and 573 K at University of Toyama, Japan. The samples were pre-implanted by 1 keV D ion irradiation then loaded with T from the gas phase. Samples were sectioned by etching and the radioactivity of the residual T activity (trapped T) was measured by imaging plate technique. Penetration distance was calculated from sample mass loss measurements following successive etching steps. The penetration profiles can be fitted well with an error function - indicating the inward transport of T is diffusion limited. Effective diffusion coefficients were derived by least squares fitting of the penetration profiles (10^{-20}~10^{-18} m^2/s). Very little dependence on W microstructure was observed. Comparing the values at 300 and 573 K, an activation energy of ~0.1 eV was estimated, which is much lower than the activation energy for diffusion of solute hydrogen in W (0.2~0.4 eV). This suggests that T transport by D-T exchange is governed by a characteristic rate constant that is very weakly dependent on temperature.

To test if such a characteristic rate constant and its temperature dependence holds for other isotopes, complementary D-H isotope exchange studies were performed using the dual beam experiment at IPP, Garching. Using nuclear reaction analysis, we followed the isothermal loss of pre-implanted D due to subsequent H implantation over a temperature range 250-450 K. The D loss as a function of implanted H fluence can be fitted well with a double exponential function. The initial decay constant of the first exponential has an activation energy of ~0.1 eV, while the second decay constant is nearly temperature independent. These results taken together support the notion that a single characteristic rate constant with an activation energy of ~0.1 eV can describe the isotope exchange process in W. The positive/negative implication on T-inventory and permeation will depend on the absolute magnitudes of such exchange processes, which is controlled by other parameters such as absolute hydrogen/trap concentrations or boundary conditions for diffusion.

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Investigation of deuterium trapping and release in the JET divertor during the third ILW campaign using TDS and TMAP

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JET is the largest tokamak with ITER-Like Wall design; bulk beryllium in the main wall and tungsten coated CFC-tiles and one row of bulk tungsten at the divertor. JET has operated with the ILW configuration since 2011 and the last campaign, ILW-3, was finished in 2016. Material lifetime and fuel retention are critical issues for the next step fusion devices such as ITER so the ILW experiments at JET provide unique information on these critical issues. Gas balance and post-mortem measurements at JET showed a factor 10-20 reduction in the long term fuel retention during the first ITER-Like Wall (ILW-1) operations in 2011-2012 when compared to JET with carbon wall (JET-C) [1,2]. During the ILW-2 campaign in 2013-2014 the inner strike point (ISP) was on Tiles 3 and 4 whereas the outer strike point (OSP) was mainly on Tile 6. During the ILW-3 campaign strike point distribution was somewhat different; ISP was on Tiles 1 and 4 whereas OSP was on Tiles 5 and 6. The present work continues the retention studies using TDS technique by analysing JET divertor tiles removed from the vessel after the ILW-3 experimental campaign in 2015-2016.

Selected samples from the divertor tiles exposed in 2015-2016 (except from bulk W Tile 5) were measured with TDS. TDS analyses were made in an ultra-high vacuum (UHV) system with a base pressure of ~10^{-9} mbar. Samples were annealed with linear ramp rates (10 K/min) from room temperature (RT) up to 1000 ^{o}C. After the ILW-2 campaign highest D amounts were observed in regions with thickest deposited layers, i.e. in the inner divertor on the horizontal part of Tile 1 and on High Field Gap Closure Tile. Increased deposition was found in the remote area outside the strike points on Tiles 4 and 6. Lowest D amounts were found on the outer divertor Tiles 7 and 8. Preliminary TDS results indicate that D retention on Tile 3 is at the same level as in ILW-2. TDS results will be compared with IBA and SIMS results. TMAP7 [3] simulations will be performed to try and reproduce the experimental TDS spectra by using realistic D depth profiles obtained from the SIMS measurements. A TMAP7 model for pure Be have been developed further for the co-deposited layers observed on JET tiles.

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**See the author list of “Overview of the JET results in support to ITER” by X. Litaudon et al 2017 Nucl. Fusion 57 102001.

Predictions for T retention in DT campaigns in JET ITER-Like Wall

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Particle control is a critical issue for the next step machines like ITER: particle injection and extraction systems must regulate the D-T fuel densities, exhaust helium ash, and minimize the tritium vessel inventory. Series of fuel retention experiments covering a wide range of plasma scenario have been performed in JET-ILW [1] showing a reduction of long term fuel retention of ~20 compared to JET carbon configuration. WALLDYN simulations have been performed for few of these experiments showing a consistent dependence as a function of the plasma scenario [2]. However the simulated long term retention is significantly lower than the one observed. The aim of this paper is to clarify and to quantify the main processes contributing to the retention. These include implantation and codeposition during plasma, long term outgasing in-between pulses and after the last pulse of the experimental sessions. Comparison of the modelling [2] with the experiments [1] are reported, whilst T retention predictions for JET-ILW are the main focus.

The analysis have been carried out for the series of ELMy H-mode pulses run in the frame of the “long term sample” (LTS) experiments performed in JET-ILW in July 2012. Particle fluxes and ion/neutral energy have been evaluated using the SOLEDGE2D simulations. The retention by implantation has been calculated based on these fluxes whilst the retention by co-deposition of D with Beryllium (eroded from the main wall) has been estimated assuming a fixed ratio of 0.3D/Be. The plasma scenario has been divided into three equally phases of 6 sec: the plasma ramp up (ohmic), the H-mode and the ramp down. Finally, the outgasing between pulses has been fitted through the experimental neutral vessel pressure following a law $\alpha\ t^{-0.7}$.

From these results, it is shown that, for JET-ILW, the implantation process leads to a nearly constant retention of ~$7\text{x}10^{22}$D. Indeed, it depends weakly on the plasma scenario and the pulse duration, since just few seconds are necessary for this amount of particles to be retained. Codeposition is a continuous process which dominates the long term retention and which also depends on the plasma scenario through the Be erosion. It is also shown that for the experiments dedicated to fuel retention, the gas collection resulting from the outgasing should be limited to about 5-6 hours after the last pulse of the sessions. This corresponds to the duration required for the neutral vessel pressure to recover the initial value of the beginning of the session. Over this duration, extra outgasing from previous sessions would be included and this would overestimate the fuel recovery. In these conditions, the retention rates have been estimated to be $2.5\text{x}10^{19}\text{Ds}^{-1}$ and $1.3\text{x}10^{20}\text{Ds}^{-1}$ respectively for the ohmic and H-mode phases. These results are very close to the experimental results [1] whilst the retention behaves as ~(wall flux)$^{0.7}$ well in line with WALLDYN [2]. For 10 LTS pure T pulses/day at JET-ILW (exp.) the long term retention $1.0\text{x}10^{22}$ T per day (0.05g of T) is predicted. Finally, the extrapolation of T retention for the JET DT campaign and main uncertainties are discussed.

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Detection by LIBS of the deuterium retained in the FTU toroidal limiter

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The quantitative detection of tritium retained in the ITER in vessel components is mandatory for deciding if the machine operation must be stopped and the exceeding tritium removed. Laser Induced Breakdown Spectroscopy (LIBS) is a suitable not invasive in situ diagnostic for detecting retained tritium: it is still under discussion whether the analysis from outside the vacuum chamber is sufficient to extrapolate the measured quantity to the entire ITER wall or a robotic arm equipped with LIBS system should be employed in the vented machine to analyze a more consistent area of the vessel. In this paper the LIBS measurement of the deuterium (used as a proxy for tritium) retained in and the surface elemental composition of the FTU Mo (TZM) toroidal limiter tiles, carried out from remote (~ 2.5 m) during machine maintenance, are reported. The experimental layout consists of a Quantel laser "Twin BSL" (λ = 1064 nm), a Andor "Istar DH320T-18F-63" ICCD camera with 1024x512 sensor (26 um pixels) and a Jobin Ivon "Triax 550" spectrometer (550 mm) with 2400 grooves/mm grating. The collinear transmission of the laser beam and detection of emitted visible lines is done through the 2 inches window of an equatorial port by using a dielectric mirror. Laser and optics for laser transmission and visible lines detection are mounted on a plate movable along three axes and able to be pivoted. Single pulse technique has been used so far with the FTU vessel under high vacuum or in Nitrogen or Argon atmosphere. Vacuum measurements resulted in a good resolution of D_α and H_α emission lines and in the detection, besides Mo, the main component of the TZM alloy, of Li coming from the lithium deposited during the experiments with lithium limiter, inserted in the vessel through a vertical port located 60° toroidally apart. Deuterium was also detected in shadowed zones in between tiles. Measurements carried out in Nitrogen atmosphere showed no evident D_α and H_α emission line: a possible explanation is the partial formation of NH and ND compounds, of which the emission lines in the LIBS plasma plume were not detectable in our experiments given the cut of emitted line wavelengths below 400 nm, caused by the dielectric mirror. The signals coming from D and H not chemically bound and/or from the brake-up of compounds could be below the detection limit. With Argon atmosphere (500 mbar) the deuterium and hydrogen lines were well visible although with worse resolution with respect to vacuum measurements, as it was to be expected because of the larger Stark broadening of emitted lines. After a fresh boronization (routinely performed in FTU) deuterium was also detected together with Boron lines. Both quantification of retained deuterium by using Calibration Free method and the choice of experimental parameters (laser energy density, gate delay, gate width etc.) for the best resolution of D_α and H_α emission lines, are discussed. These measurements were also carried out for supporting the proposed use of a robotic arm for an extended LIBS analysis of retained deuterium in and surface composition of the in vessel FTU components.

The effect of Beryllium Oxide on retention in JET ITER-like wall tiles

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Fuel retention in the JET-ITER like wall (JET-ILW) has decreased by a factor of 20 compared to operation with a carbon wall [1]. This reduction occurred due to a decrease in chemical erosion of the beryllium main chamber. A reduction in co-deposition of fuel with beryllium has been reported in the divertor [2]. However, 24% ($8.8x10^{22}$ D atoms) of fuel remains as long-term retention in beryllium dump plates and limiter tiles [1]. This contribution studies the Be limiter tiles exposed in JET, with the aim of understanding how the microstructure influences fuel retention. Dump plate, inner, and outer wall limiters tiles have been investigated, encompassing deposited, eroded and melted regions of the vessel. A wide range of techniques have been used to study these phenomena at different length and depth scales. Focused Ion Beam (FIB) 'serial milling' studies were used for compositional understanding of 20μm^3 beryllium cut outs. TEM studies were undertaken of 20μm^2 lift-outs. SEM and EDX studies were undertaken of the surface morphology and composition at different energies. Raman Spectroscopy was applied for the first time to JET tiles, to investigate the chemical bonding of surface layers up to 50nm depth. Raman investigations have uncovered the presence of BeO bonded to deuterium in BeO_xD_y, for the first time, on melted surface regions in upper dump plate tiles. It is proposed that this bonding is important in the retention mechanisms present for beryllium in JET. Preliminary Density Functional Theory (DFT) modelling was undertaken, which confirms the Raman band for the wurtzite BeO crystal structure. The literature supports the formation of BeO bonding even under Ultra High Vacuum (UHV) conditions above temperatures of ~630°C [3-4]. SEM-EDX studies of the samples support the presence of oxide island formation. Both oxide island size and number density decrease between co-deposited wing tiles and eroded central regions. A thorough investigation of Thermal Desorption Spectroscopy (TDS) was undertaken toroidally across the midplane of the outer limiter, inner limiter and dump plates. The initial trends in the TDS data, support the presence of a different desorption peak behaviour in the central eroded regions. This suggests a different retention trap behaviour. Some toroidal asymmetry is present on a first review of the TDS data. Peak behaviour differs between the left and right wing co-deposited positions. TMAP7 analysis will be applied to these TDS spectra in greater detail, utilizing the microstructural features found with the techniques above in the code.

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The effect of hydrogen presence on damage stabilization under simultaneous W ion damaging and D ion exposure

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In future thermonuclear devices such as DEMO tritium retention in neutron-damaged plasma-facing components will become a significant issue, due to safety of operation and fuel efficiency. The most suitable materials for plasma-facing components are tungsten (W) or advanced tungsten alloys. In order to study the influence of material irradiation by neutrons on fuel retention, high-energy ions produced by MV accelerators are used to simulate displacement damage [1]. Recently, we performed first experiments where W was simultaneously damaged by high energy W ions and exposed to deuterium (D) atoms [2]. These simultaneous exposures were also compared with sequential damaging and atom loading procedures. It was shown that the presence of deuterium has an influence on defect stabilization and consequently on fuel retention. This study has made a step closer to a real fusion reactor scenario where both implantation of energetic hydrogen ions and neutrals, as well as damage creation by neutron irradiation, will take place at the same time.

In this contribution we will present the next step of simultaneous exposure studies where polycrystalline tungsten samples (grain size 10-50 μm) are simultaneously irradiated by 10.8 MeV W ions and loaded by D ions with energy of 300 eV/D and an ion flux of $2x10^{18}$ D/m^2s at four different temperatures from 450 K to 1000 K. We keep the same damaging parameters as for the exposures with D atoms where four-hour exposure to energetic W ions resulted in a maximum of 0.5 displacement per atom [2]. In order to determine the density of the created traps in the material, the samples are after simultaneous damaging and loading additionally exposed to D ions for 39 h at 450 K (300eV/D, $2x10^{18}$ D/m^2s, fluence $2.8x10^{23}$ D/m^2). The deuterium depth profiles are measured by nuclear reaction analysis (NRA) using the $D(^3He,p)^4He$ reaction. When comparing with the atom exposures, the maximum D concentration in the damaged zone is more than a factor of three higher for the simultaneous W damaging and D ion exposure. These results are also compared with those obtained when samples are submitted to sequential damaging/annealing and D ion loading procedure and damaging at elevated temperatures. Even though there is a factor of three difference in the maximum D concentration as compared with the atom loading case the trend is similar. Namely, the D concentrations for the simultaneous experiment lay below the sequential damaging/annealing and ion loading experiment but above the damaging at elevated temperatures. Therefore, synergistic effects between damaging and presence of deuterium were identified also for the D-ion-loading case and are mostly pronounced at lowest (450 K) and highest temperature (1000 K). The results will be presented and discussed.

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Impact of Kr and Ar seeding on D retention in ferritic-martensitic steels after high fluence plasma exposure

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Reduced-activation ferritic-martensitic (RAFM) steels offer not only the best of steel bulk and acceptable activation behaviour, but also advanced surface stability in plasma environment and, therefore, are considered as candidate plasma-facing materials for the first wall in fusion reactors [1,2]. Retention of hydrogen isotopes as a fuel in reactor walls has a safety limit and could be influenced by cooling agents such as noble gases. In this experimental study we aim to find correlation between deuterium (D) retention in RAFM steel Eurofer'97 (EU'97) and seeding of argon (Ar) and krypton (Kr) in plasma exposures at fluences ~ 10^{26} D/m^2.
Range of exposed materials is extended by steel P92 due to its elemental as well as structural similarity with EU'97 and by pure polycrystalline iron Fe and tungsten W for comparison. EU'97 and P92 steels have in bulk <2 wt.% W that enriches the surface due to preferential sputtering of lighter constituents by plasma [1,2]. Outgassed polished samples were exposed in the PSI-2 linear plasma device to D+Ar and D+Kr plasmas (T_e=10 eV) at temperature 450-480 K, flux 4×10^{21} D^+/m^{2}s, impact ion energy 30-40 eV, fluences $\leq 1\times10^{26}$ D^+/m^2 and seeding 7-10% vol. Surface morphology, microstructural changes in the subsurface and elemental distribution were examined using scanning electron microscopy (SEM), focused ion beam (FIB) cross-sectioning and energy-dispersive X-ray spectroscopy (EDX). D content up to 8.6 µm depth and elemental composition were determined by nuclear reaction analysis and Rutherford backscattering spectrometry (NRA/RBS). D uptake in material bulk was measured by thermal desorption spectroscopy (TDS) till 1000°C.
According to SEM imaging, steels exposed to D+Ar / D+Kr plasmas exhibited rough eroded and stepped surfaces with sharp mountain-like profile 300 nm high, but preserved their original ferritic-martensitic microstructure in the bulk. No blisters or cavities were detected, grain borders could not be identified. Fe and W, in opposite, showed rather smooth surfaces with plenty of partially ruptured blisters and clearly visible grains. As shown by NRA, D retention in steels in the µm range was one order of magnitude less than after D or D+He plasma exposures in previous studies [3]: 0.2×10^{19} D^+/m^2 vs. 2.5×10^{19} D^+/m^2 respectively. However, as determined by TDS, total D inventory in steels exposed to Ar/Kr seeded D plasmas reached 5-8$\times10^{19}$ D^+/m^2 and is comparable with that after D or D+He plasma exposures [3]. The difference could be seen in thermal desorption spectra with broadened overlapped high temperature desorption peaks: surface sputtering by heavier Ar or Kr ions results in new additional trapping sites with higher energies.

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Modeling of H/D isotope-exchange in Be in UHV laboratory experiments

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Beryllium will be used as a plasma-facing material in ITER, as well as a neutron multiplier in the breeding blanket of DEMO, therefore a detailed understanding of hydrogen retention mechanisms in Be is of high importance. Earlier ion implantation experiments with deuterium indicated a saturation of retention at D/Be 0.3-0.4 in the implantation zone [1]. The emergence of a low-temperature thermal desorption stage has been observed for exposure fluences above 10^{21} D/m^2 [2]. These effects can be qualitatively explained by hydrogen trapping at radiation induced vacancies and by surface effects. However, good quantitative agreement between experiments and models is still missing. To gain a better insight into mechanisms of hydrogen retention, the so-called isotope-exchange experiments were performed, in which the material was subsequently irradiated with hydrogen and deuterium ions (and vice versa) and the results of thermal desorption at different stages were compared. It was shown that mainly the low temperature desorption stage was affected by the isotope exchange.

Reaction-diffusion modelling is used to interpret the experimental findings. The model describes the transport and desorption of deuterium implanted into beryllium based on DFT data for D diffusion and trapping at point defects, and accounts for hydrogen accumulation on the surface. The fluence dependence of hydrogen retention and release in beryllium can be qualitatively explained by the model. The model has been extended to include two hydrogen isotopes simultaneously, including multiple trapping in vacancies, and the results of simulations for isotope-exchange experiments will be presented and discussed in the contribution. Simulations indicate that there is indeed little isotope exchange for the high temperature desorption stage. The low temperature desorption, if governed by surface effects, can show strong isotope exchange when the surface is saturated or when sputtering of surface hydrogen/deuterium by ions of the second implanted isotope is significant.

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Peculiar release kinetics of deuterium in tungsten revealed by an *in situ* laser induced desorption technique

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Deuterium and tritium trapping at the divertor is one major concern in fusion devices such as ITER or DEMO because of tritium recycling issues as well as nuclear safety regulation related to tritium radioactivity.
In this contribution, the fundamental mechanisms behind deuterium detrapping in tungsten has been studied in a new ultra-high-vacuum apparatus combining the advantage of *in situ* Temperature Programmed Desorption (TPD) and laser induced desorption methods.

Single crystal W(111) and recrystallized polycrystalline tungsten samples were implanted at 300 K with 500 eV deuterium ions (D_2^+). The kinetics of deuterium desorption was studied using a Laser-induced TPD. The deuterium desorption rate is measured during a well-controlled increase of the sample temperature upon laser heating. We made varied the (Laser-induced)TPD heating rate from 0.1 to several hundreds of K/s by combining continuous wave infrared laser heating with power densities up to 10 MW/m^2 and standard radiative heating. Desorption products were quantified by using a differentially pumped mass spectrometer.

For heating rage below 10 K/s, a single desorption peak of deuterium is found for both tungsten single and poly-crystals. However, for higher heating rates in poly-crystals, the single desorption peak morphs in two desorption peaks, confirming the recent prediction of a dual trapping mechanism in recrystallized poly-crystalline tungsten [1]. One of the two desorption peaks presents a peculiar temperature shift with increasing heating rates. The very large shift of this deuterium release peak cannot be reproduced neither from a surface Polanyi-Wigner process, nor from 1D bulk MRE models (MHIMS code) using the usual attempt frequency of 10^{12}-10^{13} s^{-1}. A Falconer-Madix analysis of the Laser-induced TPD series with varying heating rates allows to retrieving an effective attempt frequency several orders of magnitude smaller than usual ones.

The complete kinetic parameters set (activation energies and attempt frequencies) being determined for both trapping mechanism in poly-crystalline tungsten, we used them in the latest version of MHIMS, which includes temperature gradients [2], to forecast the tritium retention in an actively cooled ITER-like divertor in its D-T start-up phase i.e. without neutron damage.

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Abstract Withdrawn

Tritium retention characteristics in dust particles in JET with ITER-like wall

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It is important to understand and determine tritium (T) retention and its behavior in dust with repect to the particle microstruture and composition. In the reported study, we have applied a tritium imaging plate technique (TIPT) in combination with an electron-probe microscopic (EPM) technique to examine a T retention characteristic in individual dust particle collected in the Joint European Torus with the ITER-like Wall (JET-ILW) after the first campaign in 2011-2012 [1].

Dust particles were collected from the inner divertor region in JET [2]. The total amount was on the level of 0.15 g. A tiny amount of dust was taken from that sample and placed on the surface of a disk made of indium (In). The content of T retained at a surface of or/and in the dust particles was evaluated by TIPT [3]: emission of β-rays from T to the imaging plate (IP) was measured over a whole surface of the indium disk. Afterwards, EPM analysis was conducted to determine morphology (composition and structure) of the dust particles in the same surface area where the IP measurement was conducted. The obtained maps (images) of the T distribution and elemental compositions were super-imposed to reveal characteristic features of T retention in the individual dust particle.

The results suggest that most of T was retained at the surface of and/or in carbon-dominated dust particles. The characteristics of T retention are in three categories dependent on the intensity of β radiation; moderate, high and very high. These different T retention characteristics are related to the size and microstructural roughness of the dust particle. The retention in tungsten, beryllium and other metal-dominated dust particles is relatively small: smaller by a factor of 10 ~ 100 in comparison with that in the carbon-dominated particles. This belongs to the category of moderate T retention, while the retention in carbon-containing and carbon-based particles is high/very high. The retention in each category of dust particles will be discussed with respect to size, details of microstructure and composition, and the distribution of tritium trapped in respective particles.

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Deuterium retention in tin exposed to fusion-relevant flux plasmas

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Tin is promising candidate for use in a liquid metal plasma-facing component (PFC) owing to its low evaporation rate and melting point. It is important to know how much deuterium is retained in liquid and solid tin and to understand its retention mechanisms under realistic fusion reactor conditions, in which ion fluxes and energy vary spatially by orders of magnitude [1]. However, few papers about deuterium retention in tin under steady-state plasma exposure are reported [2]. This work aims to fill the gap and systematically investigate deuterium retention in tin under low and high flux plasma exposures, respectively, using Nano-PSI [3] and Magnum-PSI [4].

In Nano-PSI, the typical ion fluxes of deuterium plasma are $10^{19} \sim 10^{20}$ $m^{-2}s^{-1}$. Free liquid tin surface targets were exposed with exposure time ranging from 10 minutes to 300 minutes. Deuterium retained in tin was measured using Thermal Desorption Spectroscopy (TDS) to be about $10^{-5} \sim 10^{-6}$ D/Sn or $10^{16} \sim 10^{17}$ D/cm^2 one week after the plasma exposure. This is one to two orders higher than the values reported in [2]. Furthermore, the retention was found to increase with deuterium fluence. During the exposure, droplet ejection of liquid tin was observed due to bubble bursting. This may lead to a higher retention of deuterium in liquid tin in the form of bubbles. A quick desorption of a small amount (0.02% ~ 0.3% of total deuterium retained in the samples) of deuterium gas happened at the temperature close to the tin melting point in the TDS desorption spectra.

Magnum-PSI can produce D^+ fluxes in the range plasma $10^{23} \sim 10^{25}$ $m^{-2}s^{-1}$ with an energy of 0.1-5 eV. Tin samples using a capillary porous system to stabilize against gravity and droplet production [5] are exposed to such deuterium plasma at different temperatures and fluences. Nuclear Reaction Analysis is carried out in the target analysis chamber in vacuum immediately after exposure. The total amount of deuterium retained in each sample is measured with TDS and compared with tin samples exposed in Nano-PSI.

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Deuterium retention in neutron irradiated tungsten-rhenium alloy and potassium-doped tungsten

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Tungsten (W) will be used as a plasma-facing material in ITER and possibly as an armour material in a fusion reactor. Hydrogen isotope retention of W materials under neutron (n) irradiation has to be investigated from the viewpoint of tritium inventory. Previous study [1] showed that defects induced by n-irradiation acted as trapping sites resulting in increased retention.

Advanced W materials, such as W-rhenium (Re) alloy and potassium (K)-doped W, have been developed in order to improve the mechanical properties of W [2]. Although their microstructures - Re precipitates in W-Re alloy or higher density grain boundaries in K-doped W - can act as sinks for irradiation induced defects, their hydrogen isotope retention after n-irradiation is not well understood. In this study, deuterium (D) retention in n-irradiated W-Re alloy and K-doped W was investigated using thermal desorption spectroscopy (TDS), to elucidate how the microstructures of W-Re alloy and K-doped W influence D retention and trap-site formation by n-irradiation.

W-5%Re alloy and K-doped W specimens (φ6 mm × t0.5 mm) were used. Neutron irradiation of the specimens was conducted in BR2 fission rector at Belgian Nuclear Research Centre at 563 K to 0.06 dpa. D plasma exposure was performed using the Compact Divertor Plasma Simulator [3] at Tohoku University (flux: $\sim 4.4 \times 10^{21}$ $m^{-2}s^{-1}$, fluence: $\sim 1.2 \times 10^{25}$ D m^{-2}, specimen temperature: 563 K). Following D plasma exposure, in-vacuo TDS was conducted to measure D retention. Compared to un-irradiated specimen, D retention in n-irradiated W-Re alloy increased by a factor of ~3.6 (2.8×10^{20} D m^{-2} in un-irradiated and 1.0×10^{21} D m^{-2} in n-irradiated specimens). The TDS spectrum of irradiated specimen showed one broad desorption peak between ~600-1200 K. The release at the high temperatures above 900 K was not observed for the un-irradiated specimen. Such difference indicates that new hydrogen trap-sites were formed in W-Re alloy by n-irradiation. In the case of K-doped W, n-irradiation increased D retention by a factor of ~1.3 (7.5×10^{20} D m^{-2} in un-irradiated and 1.0×10^{21} D m^{-2} in n-irradiated specimens). This smaller increase in D retention could mean that the higher density grain boundaries in K-doped W influenced on trap-site formation under n-irradiation.

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Momentum Injection and Precise Core Fuelling for Reactor Grade Fusion Plasmas

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High-performance ST and tokamak plasmas greatly benefit from plasma rotation and rotation shear to increase energy confinement and sustain high beta. This has been possible due to the injection of substantial momentum from tangentially injected neutral beam systems that also contribute to important core fuelling in such plasmas. In larger devices such as ITER or DEMO, higher beam injection energies are required to penetrate to the plasma core, and this reduces the momentum input per unit power. As a result, ITER is projected to have low toroidal rotation relative to present devices. Steady-state Advanced Tokamak (AT) scenarios rely on optimized density and pressure profiles to maximize the bootstrap current fraction. Under this mode of operation, the fuelling system must deposit small amounts of fuel where it is needed, and as often as needed, so as to compensate for fuel losses, but not to adversely alter the established density and pressure profiles. Conventional fuelling methods have not demonstrated successful fuelling of AT-type discharges and may be incapable of deep fuelling long pulse ELM-free discharges in ITER. The capability to deposit fuel at any desired radial location within the tokamak would provide burn control capability through alteration of the density profile.

Compact Toroid (CT) fuelling has the potential for toroidal momentum injection, and in addition has the potential to provide a source of deep controlled fuelling for density and pressure profile control in burning reactor grade plasmas. A CT [1,2,3] is a self-contained toroidal plasmoid with embedded magnetic fields. A reactor CT injector [4] would typically inject 2.2 mg toroids of DT plasma or pure tritium plasmas at a fuelling rate of up to 20 Hz. This 5 MW injector will impart the same momentum as a 69 MW, 500 keV neutral beam injector, while supplying 14 times more core fueling. These would be injected at a nominal velocity of 300 km/s, but have capability to vary the velocity (200-500 km/s) in order to vary the fuel and momentum deposition location. The injector would be positioned with some tangency with respect to the radial direction to be able to inject toroidal momentum. Two to three injectors positioned at different toroidal locations, and with different tangency could be used to control rotation shear. A CT injector does not require cryogenic tritium handling facilities, and has a simpler fuel cycle. In addition, the injected CT could have any combination of tritium and deuterium. The system is easily adaptable to a fusion reactor fuel cycle as the tritium extracted from the reactor exhaust can be directly cycled back into the injector gas control system. As a result of much higher tritium burn-up from deep core fueling, much less tritium will be present in the fuel cycle. These beneficial aspects of a CT injector, reactor implementation, and the required development effort will be discussed.

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Hydrogen isotope retention in W inserted a Cr thin layer

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Hydrogen isotope retention in plasma facing materials (PFMs) is one of the most critical issues for the fusion reactor. Tungsten (W) is a promising candidate for the PFM because it has good properties such as a high energetic threshold for physical sputtering, a high melting temperature and a low retention of hydrogen isotopes. However, such good properties can be lost due to neutron irradiation. Hydrogen isotope retention in neutron-irradiated W will significantly increase, since neutron-induced damage ranges over the whole region of the bulk W and hydrogen atoms diffuse deeper in the bulk W. In this study, W inserted a chromium (Cr) thin layer, which plays a role of a diffusion barrier, has been developed and possibility of reduction in retention in neutron-irradiated W has been explored.
The sample was made by a physical vapor deposition method. A Cr layer with the thickness of ~0.27 μm was formed on ITER-grade W (10 mm x 10 mm x 1 mm) and then a W layer with the thickness of ~1 μm was formed on the Cr layer. XPS measurement shows the surface of the Cr layer was naturally oxidized before the W deposition. The Cr oxide is expected to serve as a diffusion barrier for hydrogen isotopes [1].
The sample was exposed to deuterium (D) plasma in the compact PWI simulator APSEDAS [2] after annealing at 1173 K for 1 hour in vacuum. The ion energy and ion flux of the D plasma were ~30 eV and ~4 x 10^{21} D m^{-2} s^{-1}, respectively. The surface temperature of the sample during plasma exposure was 475 K ~ 495 K. The D plasma exposure and TDS measurement were repeated 5 times using the same sample to obtain the dependence of D retention on the fluence, which was changed from 2.2 x 10^{23} D m^{-2} to 2 x 10^{25} D m^{-2}. The D retention increased with increase in the fluence up to ~2 x 10^{25} D m^{-2} and then it saturated. After the series of the D plasma exposure and TDS measurement, the surface morphology was examined with SEM, showing there were many blistering with the size of a few μm to about 10 μm on the surface. The GD-OES measurement revealed that the D atoms existed in the W and Cr layers and did not exist in the bulk W, meaning the Cr layer acts as a diffusion barrier. The saturation of D retention is considered to be attributed to the diffusion barrier of the Cr layer.
Analysis of crystal structure of W and Cr layers is in progress by a combination of FIB and TEM. Besides, samples of which Cr layer has been exposed to oxygen plasma before W deposition will be exposed to D plasma and properties of D retention will be investigated to discuss the effect of degree of Cr oxidation on diffusion of Hydrogen.

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Influence of the presence of deuterium on displacement damage in tungsten

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In order to achieve tritium self-sufficiency in a future fusion reactor, the limitation in terms of tritium uptake needs to be very stringent [1]. Among many other favorable properties intrinsically low fuel retention makes tungsten, therefore, one material of choice as plasma-facing material. However, during operation defects in the tungsten lattice will evolve that will trap hydrogen isotopes. While for present day experiments this increased retention is only limited to the near surface it will take place throughout the whole bulk in future nuclear devices as a consequence of the neutron irradiation. Experiments with neutron-irradiated samples as well as MeV-energy ion-implanted samples showed that hydrogen isotope retention can reach values of the order of 1 at.% due to the displacement damage [2]. However, all these experiments were done sequentially: First damage was produced and only then transport and deuterium retention in this material was studied. First beam experiments where displacement damage was created while simultaneously dosing the samples with atomic deuterium revealed that even the presence of very small amounts of deuterium in the bulk can increase defect density and hence retention noticeably [3].
Here a different experimental approach will be presented that allows studying the influence of hydrogen isotopes on damage creation and stabilization. In this approach first 20 MeV W is implanted into recrystallized tungsten to a level where damage cascades overlap and defect density saturates within the 2 µm deep implantation zone. Second, the created defects are decorated with deuterium with a low temperature plasma achieving a homogenous deuterium concentration of 1.8 at.%. In a third stage, these samples are again implanted with MeV tungsten. In this step not only tungsten atoms are displaced and defects are generated but also the retained deuterium atoms are de-trapped. SDTrimSP calculations reveal that for the implantation parameters used, D atoms are recoiled ten times more effectively than tungsten atoms. Nuclear reaction analysis shows that during this second implantation no deuterium is lost from the sample, but it is only de-trapped and is effectively re-trapped again. Thermal desorption spectroscopy (TDS) shows that deuterium is redistributed from the low temperature de-trapping peak to the high temperature de-trapping peak. When samples are again decorated with deuterium after the second self-damaging step, NRA and TDS reveals that deuterium retention exceeds the initial saturation value by nearly a factor of two. These experiments clearly support the calculations by Kato et al. [4] and Middleburgh et al. [5] which predicted that hydrogen isotopes either stabilize vacancies or lower the vacancy formation energy which both leads to an increase in trap density.

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Improved Tritium Retention Modeling with Reaction-Diffusion Code Tritium Migration Analysis Program (TMAP)

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The in-vessel tritium inventory source term (i.e. tritium retention) is one of the critical safety evaluation terms used in reactor safety assessments for licensing fusion facilities. Improving the modeling capability of deuterium (D)/tritium(T) implantation, diffusion and trapping in plasma-facing components (PFC) is a key to accurately predicting tritium inventory. The Tritium Migration Analysis Program (TMAP) code, the reaction-diffusion code developed by the INL Fusion Safety Program, has been widely used for simulating D/T thermal desorption behavior from PFCs for D/T fuel retention research [1]. There were, however, two challenges in the previous version of TMAP; (1) Limitation on the maximum number of trap sites, (2) Inability to simulate near-surface phenomena affecting D/T behavior by high-flux plasma. For example, at high incident D/T flux and low temperature condition, the local concentration of D/T in solution within the implantation depth is high and the very high corresponding equilibrium gas pressure causes near-surface precipitation [2]. Interconnection of gas bubbles to the surface provides escaping paths for precipitated D_2/T_2, which reduces the diffusion length for release of D/T from solution, resulting in an increase in the release of incident D/T. To eliminate challenge (1), the TMAP trap modeling capability was recently enhanced to include as many traps as required by the user to simulate D/T behavior in neutron-irradiated tungsten [3,4]. To overcome challenge (2), efforts were made to simulate near-surface phenomena affecting D/T behavior. In our previous work, the increased D/T release by near-surface precipitation in neutron-irradiated tungsten was successfully simulated by adjusting a boundary condition of D/T ion at the surface [5].

In this work, we show the D thermal desorption spectra from polycrystalline tungsten specimens exposed to high-flux D plasma at low temperature with INL's Tritium Plasma Experiment, and report the adjusted boundary conditions used in the TMAP to simulate near-surface phenomena at different ion flux density (10^{21} and 10^{22} $m^{-2}s^{-1}$), different surface temperature (373, 473K, and 673K), and different ion energy (50, 100, and 500 V).

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Isolating the Detrapping of Deuterium in Heavy Ion Damaged Tungsten via Partial Thermal Desorption

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We investigated the use of partial Thermal Desorption Spectroscopy (TDS) of deuterium (D) from tungsten (W) trap sites to isolate both their spatial location and detrapping energies. W damaged with heavy ions and then implanted with D_2 plasma typically display multiple release peaks in the experimental TDS data. These release peaks are associated with various detrapping energies convoluted by an unknown spatial distribution of that type of trap. While Nuclear Reaction Analysis (NRA) gives some spatial information, it only measures the sum total of all D filled traps. Typically the NRA profile displays three distinct spatial zones: (I) the very near surface due to plasma implantation stress, (II) the region where heavy ions cause displacement damage, and (III) the intrinsic defects throughout the bulk.

We devised an experiment to successively depopulate each trap with increasing detrapping energy, thus allowing us to differentiate the spatial location associated with a particular detrapping energy. The experiment consisted of multiple W samples that were prepared identically with surface polishing and annealing prior to induced displacement damage with 5 MeV Cu ions and peak dpa of 0.12. All samples were subsequently exposed to the same D_2 plasma conditions and received a total fluence of 10^{24} D/m^2 over 1.5 hours at a temperature of 383 K. Next, one control sample was not thermally desorbed at this stage, while the other samples were subjected to a partial TDS (pTDS) up to various peak temperatures chosen to depopulate individual traps. Each sample reached a different peak, spanning 467 to 762 K, and was held for an hour at this temperature. NRA was then performed to determine the spatial profile of the D concentration remaining after pTDS as well as for the control. Lastly, TDS was performed up to 1300 K to remove all remaining D.

The simulation of the pTDS, NRA, and full TDS stages are well constrained by both the identical initial conditions and the controlled depopulation of each trap. Here we assume each trap concentration and spatial profile as well as the filling thereof during D implantation are the same for all samples. Comparing the NRA data from the control sample, without pTDS, and the lowest pTDS temperatures demonstrates that all of zone I was depopulated by holding the sample at 525 K. In addition, the highest detrapping energy is nearly depopulated completely by a pTDS temperature of 762 K. For pTDS peak temperatures between 525 and

762 K, the D profiles measured via NRA are similar to the displacement damage profile predicted by the Stopping and Range of Ions in Matter.

Model of hydrogen retention in tungsten with self-induced trap formation

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Tungsten has been selected as material of the divertor plasma-facing components (PFCs) in ITER tokamak and is a primary candidate material for PFCs in future fusion reactors. The choice of tungsten is driven in part by the relatively low tritium retention, as compared to carbon materials in previous tokamak designs [1]. Nevertheless, many hydrogen plasma exposure experiments, e.g. [2,3], show that high concentrations of retained hydrogen (up to a few at.%) are present in irradiated tungsten samples within a layer of thickness of order of several micrometers, which increases with increasing plasma exposure fluence approximately as $\Phi^{1/2}$. Such high concentrations cannot be explained by hydrogen retention in the intrinsic defects present in tungsten before the exposure, which typically have concentration of $\sim 10^{-3}$at.%. Therefore, it is reasonable to assume that hydrogen trapping sites are produced by a self-induced mechanism during plasma exposure. Such mechanism can be related to lattice stresses and dislocation jogs in tungsten produced by implanted hydrogen [3,4].

In this work we develop a model of hydrogen transport and retention in tungsten based on diffusion-reaction equations that takes into account the formation of self-induced hydrogen traps. Different analytical approximations of the trap generation, associated with hydrogen already trapped or dissolved in the material and corresponding to different possible physical mechanisms, are considered. The obtained analytical and numerical solutions demonstrate that in the case of low concentrations of intrinsic trap sites and of diffusing hydrogen atoms, the large concentrations of the trapped hydrogen can be realized for the high plasma fluences, consistent with the experimental measurements. The dependencies of the retained hydrogen quantities and of thickness of the hydrogen retaining layer on time are analyzed for this case and solutions $\sim t^{1/2}$, close to the experimentally observed results, are obtained.

The cases of neutron or ion damaged tungsten, having relatively high initial concentrations of hydrogen trapping sites with uniform and non-uniform spatial profiles, are also considered. The model analysis in these cases shows possibility of spatially localized run-away solutions, corresponding to formation of super-saturated trapped hydrogen layers, which can be associated with hydrogen precipitation and creation of bubbles. The conditions leading to the formation of hydrogen super-saturated layers are examined. It is also shown that in these regimes the time-dependence of the retained hydrogen quantities becomes non-trivial and can substantially differ from $\sim t^{1/2}$. The implications of the model for retention of hydrogen under very high fluence conditions in future fusion devices are discussed.

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Abstract Withdrawn

Direct detection of deep deuterium diffusion in different tungsten grades

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Tritium retention in plasma facing components (PFC) is a safety and operational concern for fusion devices due to restricted site inventories and potential for release. The two primary methods for investigating hydrogen isotope retention, include thermal desorption spectroscopy (TDS) and nuclear reaction analysis (NRA). These two techniques have provided invaluable data in this field, however both have limitations. For example, neither method is capable of providing the depth distribution of deuterium beyond a few micrometers. Measuring the depth profile into the bulk is important especially as experiments approach more ITER-like conditions where significant diffusion is expected, namely when PFCs are exposed to ultra-high fluences or neutron damage [1-2].

Recently, glow discharge optical emission spectroscopy (GD-OES) has been employed to show deep deuterium diffusion in tungsten in medium and low fluence sample implanted at a variety of temperatures [3-4]. In the present work, 3 keV deuterium is implanted at room temperature to various fluences for a variety of tungsten substrates, including laboratory grade mirror-polished tungsten, technical tungsten surfaces, and ITER grade tungsten. The deuterium diffusion profiles are assessed for each condition and grade of tungsten. As the grain and microstructure features for these samples vary, so does the deuterium depth profile, as has been already shown [5], however the depth profile beyond ~6 μm has not been previously investigated. Specifically, the depth distribution of deuterium is higher in ITER grade tungsten than rolled tungsten. In addition, we show and discuss the influence of surface conditioning on the depth distribution and diffusion.

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Implications of PMI and wall material choice on fusion-reactor tritium self-sufficiency

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Tritium self-sufficiency is a critical issue for the production of nuclear fusion energy. Earlier work [1] showed that for high recycling surfaces the acceptable tritium trapping probability must be below 10^{-6}-10^{-7} in order for the tritium breeding ratio (TBR) to exceed unity. That analysis, however, neglected the effect of co-deposition on the fuel balance. In the present work, we extend that earlier model to include this important effect, and use it to quantify the impact of co-deposition of fuel and eroded wall material on the tritium particle balance in a hypothetical reactor system. The expected ITER plasma parameters and geometry are used to estimate the amount eroded material from a device with either a full tungsten (W) or beryllium (Be) first wall. This eroded material eventually migrates into regions where is it permanently sequestered as a co-deposit. Measured D concentrations in W and Be co-deposits from laboratory experiments are then extrapolated to the wall temperature expected in future reactors and used along with these eroded flux estimates to determine the net loss probability of tritium from the device due to fuel co-deposition. The results show that co-deposition effects alone in a W first-wall device would lead to a tritium loss probability below 10^{-10} for a first-wall fluence corresponding to a few days of device operation, while the loss probability due to direct fuel ion implantation into un-eroded W would be about 10^{-8}. Thus a W first wall device should permit a TBR>1. Estimates for a Be based first wall device show tritium loss probability due to co-deposition in the range of 10^{-5}, too high for fuel self-sufficiency. We have also applied the same methodology to co-deposits formed by the erosion of a flowing liquid lithium or gallium divertor target. Results suggest Ga would have a tritium loss probability of ~ 10^{-9}, low enough for TBR>1. Use of low-recycling Li gives an maximum acceptable tritium trapping probability in the range of ~$3x10^{-4}$ (due to the larger fueling rate and throughput required with the low-recycling boundary), much higher than the value for high-recycling surfaces. However, our model indicates that such a Li-based system would have a fuel co-deposition loss probability of order 10^{-2}, too high for tritium self-sufficiency. More generally, the results show that low sputter yield plasma-facing materials without a chemical affinity for hydrogen at the desired operating temperature must be used in order to achieve tritium self-sufficiency.

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Hydrogen retention in MeV ion irradiated (H, Fe, W) and neutron irradiated tungsten

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MeV ion irradiation is used as a proxy for neutron damage to study how irradiation damage increases hydrogen retention in tungsten (W). Despite numerous contributions in this area of research, it remains an open question how well surrogate MeV ion irradiation approximates neutron damage in W. Neutron irradiated W samples will soon be available to better clarify the characteristics of neutron damage under Japan-US collaboration, PHENIX[1]. To enable comparison to neutron-irradiated samples, it desirable at this stage to survey the nature of MeV ion irradiation effects in W. However such comparative exercise is often hindered by the differences in irradiation or exposure conditions amongst the different groups (e.g. flux, temperature, etc).

Therefore in this contribution, we report on experiments performed in our lab and through collaborations that allows for more consistent comparisons. We summarize hydrogen retention behavior in W following MeV ion damage by H, Fe and W ions. Ion damage by H was made at the energy of 0.7 MeV by MTF at QST (formerly JAEA), followed by D ion irradiation with the energy of 1 keV. Fe and W irradiation were made at DuET at Kyoto University with the energy of 6.4 MeV, followed by exposure of D atoms by a glow discharge device at University of Toyama. D retention was observed by TDS and characteristics of trapping sites were estimated by using TMAP simulations.

By appropriate scaling over damage depth and fluence (dpa), we find remarkable similarities in the observed increase in total retention between the different damaging ion species used. Comparison to recent results on neutron-damaged samples at Tohoku University [2] is made to identify the similarities and the differences. Preliminary results showed that local D density trapped at damage sites and trapping energy (~1.8 eV) for the trapping sites are similar for ion damaged and neutron damaged W. From these results, damage made by high energy ions and neutrons would be similar as long as D retention properties are concerned. Detailed comparison is underway and will be presented.

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Assessment of the 3D geometrical effects on the DEMO divertor pumping efficiency

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One of the important aspects of DEMO divertor and the first wall design is the presence of gaps between the divertor cassettes, which cannot be sealed perfectly due to technical constraints. As a consequence, unintended particle fluxes will occur both in poloidal and toroidal direction and between the divertor cassettes. This issue has been already investigated for the case of Alcator C-Mod [1], in which the influence of the gaps in the poloidal and toroidal direction is significant and only if their existence is assumed, the corresponding numerical and experimental results may coincide. Furthermore, the same study has been performed for the case of ITER [2], where the simulations indicate that there is no major negative effect of the gaps on the divertor performance, although the parasitic flows caused by these gaps can be comparable to the pumping throughput. Additionally, another work, which describes in brief the influence of the gaps, is Ref. [3], which is focused on the ITER divertor. There, it has been found that a very strong back streaming of gas into the plasma occurs, mainly due to the flows through the inter-cassettes gaps. Furthermore, a recent study has been performed for the case of a 3D DEMO divertor [4], in which the existence of dome is not considered and the assumed divertor plasma conditions correspond to low divertor pressure levels (i.e low collisionality). The main reason for the latter assumption was the reduction of the needed computational cost. The outcome of this study shows that for an open divertor a strong outflux of neutrals towards the x-point is observed and for the reference case of 20 mm gap width a 10% reduction in the divertor pumping efficiency is obtained. Based on the above framework, in the present work a sensitivity analysis of the pumping efficiency of a generic 3D DEMO divertor configuration based on the size of inter-cassette gaps, is conducted. The numerical model includes three divertor cassettes, a pumping port as well as a dome structure (or alternatively a liner), while the assumed divertor plasma conditions are based on scaling laws proposed in [5], which result in high collisionality flow regimes in the divertor. The main reason of using the above boundary conditions is mainly due to momentarily lack of thorough edge plasma simulations. The numerical study will be performed using the DIVGAS code [6]. The analyses of the influence of the inter-cassette gaps and the dome structure on the divertor pumping efficiency are meant to contribute to the ongoing DEMO divertor design efforts.

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Topical category: Plasma Fuelling, Particle Exhaust and Control, Tritium Retention

TEM Analysis of Ion-Irradiation Damage in Cu-Implanted Tungsten

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Plasma-surface interactions in tungsten, or other materials, will be strongly influenced by neutron-induced radiation damage in the plasma-facing materials (PFMs). In this work, room-temperature Cu heavy-ion irradiation (room temperature, 0.5+2+5 MeV energy stacked) of single-crystalline tungsten has been used as a surrogate for neutron irradiation, and in this presentation, we use transmission electron microscopy (TEM) to characterize ion-irradiation induced damage. Companion work, using diffuse X-ray scattering, resulted in estimated sizes of ion-irradiation-induced dislocation loops of sub-nm to 1 nm in size, and with high number density. This poses a significant problem for TEM analysis, because non-destructive sample preparation methods (in order to allow further analysis of the specimens using other methods) require we use focused ion beam (FIB) technique to extract TEM specimens, and the 5-30 kV Ga ions used in FIB would induce defects substantially similar to the Cu-ion induced defects of interest, and abuts the TEM resolution limit of about 1 nm for dislocation loops.

In this work, we will discuss FIB-based protocols for sample preparation with minimized damage, such as post-FIB polishing by low-energy (900-1800 eV) Ar-ion, and statistical analysis of the defects measured within the known Cu-ion damage band compared to the below-ion-range, "pristine" area. In particular, the relatively larger loops induced by Cu-ion irradiation (as indicated by solid yellow arrows) can be differentiated from the small defects produced by Ga ions in FIB. We tentatively conclude that the radiation damage produced by the Cu-ion implantation can be qualitatively analyzed via TEM, and trends appear to match the diffuse X-ray scattering, but further explorations are underway.

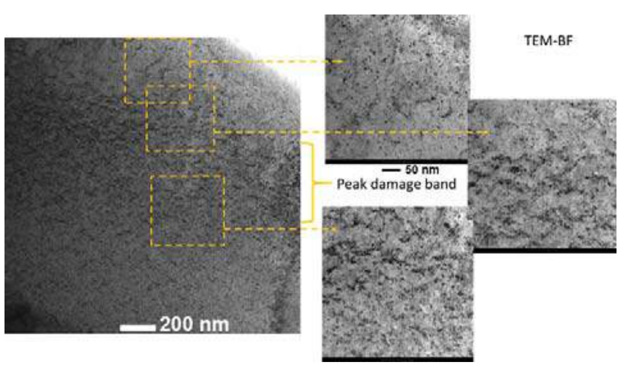

Figure: Bright-field TEM images of single-crystal tungsten Cu-ion irradiated to 2 dpa, and prepared using Ga^+ focused ion beam. Differentiation of Cu-ion damage and sample preparation damage is a challenge.

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Hydrogen and helium retention in materials

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Hydrogen isotopes and helium are common species in magnetic fusion environment [1]. Their interactions with plasma-facing surfaces and components span a wide range of time scales from sub-ns to months. Examples of fast interactions are physical sputtering and chemical sputtering. Examples of slow interactions include diffusion following their initial retention near material surfaces due to for instance physical and chemical adsorption and implantation. High-temperature plasma environment can modify the chemical reactivity of hydrogen isotopes and helium and therefore their interactions with plasma-facing components. Although the basic channels of surface interactions for bulk materials are understood quite well, quantitative understanding of hydrogen and helium retention remains elusive under the fusion environment and small particle sizes. The fact that the dust can also be generated on surfaces due to plasma erosion and modify the surface properties complicates the issues significantly [2]. We summarize initial results of hydrogen and helium retention studies using two LANL facilities: the hydrogen processing laboratory and the ion beam material laboratories [3]. Several materials including tungsten in both bulk and powder forms, are examined at different temperatures. The materials are also modified by the ion beams of hydrogen isotopes and helium at different energies. Further studies including more sensitive measurements and extension to neutron effects are possible. The experimental data will also constrain computer modeling and code development for plasma-surface interactions.

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Simulation of hydrogen isotope retention inside the tungsten material during low-energy particle flux irradiation

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Due to its favorable material properties, tungsten (W) has been chosen as the plasma facing components (PFC) in the future fusion devices such as ITER and DEMO. However, retention of tritium (T) will be enhanced when the material is heavily irradiated by energetic ions and neutrons. For safety and efficiency reasons, the total inventory of T in W should be controlled to below 700g for ITER [1]. In a fusion reactor, the divertor target may experience high plasma flux (10^{23}-10^{25} m^{-2} s^{-1}) with the electron temperature below 5 eV for long time. When a low energy particle (~eV) injected into W material, it may be adsorbed, and the chemisorbed lawyers of the W surface is formed, which prevents direct penetration [2]. Understanding the transport mechanism of fuel particles on material surface is a key issue for fuel retention in W.

In this work, hydrogen isotope (HI) retention in W material is investigated by a suite of codes HIIPC-SRIM, which has been preliminary applied to the fuel retention in the W materials [3][4][5]. However, the surface processes due to low-energy other than surface recombination have not been considered in the previous HIIPC model. To predict more accurately fuel retention in W, a surface model is developed to include surface adsorption, the Eley-Rideal (ER) and Langmuir-Hinshelwood (LH) recombination [6], adsorption and re-adsorption between surface and bulk. In the simulation, the W is divided into two zones, namely, surface and bulk. For the surface region, the new-developed surface model is applied; for the bulk region, the model based on the diffusion-trapping equation is used. The influence of surface parameters, such as desorption energies and wall temperature, on the low-energy particle absorption by the W surface are discussed. It is found that the energy barriers at the surface can reduce the HI flux from surface to bulk W. In addition, when W with vacancies and interstitials produced by energetic particles is annealed and interacted with HI, defect clusters with high de-trapping energies may be formed, which act as strong trap sites for HI. This process is also considered in the new developed model. The simulation reproduces well the experimental measurement [7].

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Deuterium retention in tin exposed to fusion-relevant flux plasmas

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Tin is promising candidate for use in a liquid metal plasma-facing component (PFC) owing to its low evaporation rate and melting point. It is important to know how much deuterium is retained in liquid and solid tin and to understand its retention mechanisms under realistic fusion reactor conditions, in which ion fluxes and energy vary spatially by orders of magnitude [1]. However, few papers about deuterium retention in tin under steady-state plasma exposure are reported [2]. This work aims to fill the gap and systematically investigate deuterium retention in tin under low and high flux plasma exposures, respectively, using Nano-PSI [3] and Magnum-PSI [4].

In Nano-PSI, the typical ion fluxes of deuterium plasma are $10^{19} \sim 10^{20}$ $m^{-2}s^{-1}$. Free liquid tin surface targets were exposed with exposure time ranging from 10 minutes to 300 minutes. Deuterium retained in tin was measured using Thermal Desorption Spectroscopy (TDS) to be about $10^{-5} \sim 10^{-6}$ D/Sn or $10^{16} \sim 10^{17}$ D/cm^2 one week after the plasma exposure. This is one to two orders higher than the values reported in [2]. Furthermore, the retention was found to increase with deuterium fluence. During the exposure, droplet ejection of liquid tin was observed due to bubble bursting. This may lead to a higher retention of deuterium in liquid tin in the form of bubbles. A quick desorption of a small amount (0.02% ~ 0.3% of total deuterium retained in the samples) of deuterium gas happened at the temperature close to the tin melting point in the TDS desorption spectra.

Magnum-PSI can produce D^+ fluxes in the range plasma $10^{23} \sim 10^{25}$ $m^{-2}s^{-1}$ with an energy of 0.1-5 eV. Tin samples using a capillary porous system to stabilize against gravity and droplet production [5] are exposed to such deuterium plasma at different temperatures and fluences. Nuclear Reaction Analysis is carried out in the target analysis chamber in vacuum immediately after exposure. The total amount of deuterium retained in each sample is measured with TDS and compared with tin samples exposed in Nano-PSI.

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Kinetics of Deuterium Penetration into Neutron-irradiated Tungsten

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Tritium (T) inventory in plasma-facing materials (PFMs) is one of the most important issue for assessing the safety of fusion reactors. Tungsten (W) is recognized as one of leading candidates for PFMs. A previous study has shown that the retention of hydrogen isotope in W significantly increased after neutron irradiation due to trapping effects of displacement damages [1]. The model of hydrogen isotope ingress into irradiated W has been proposed by previous researchers [1, 2] and it is expected that the hydrogen isotope retention increases in proportion to the square root of plasma exposure time. . However, the model has not been validated for n-irradiated specimen. In this study, n-irradiated W specimens were exposed to D plasma using a linear plasma device called Compact Divertor Plasma Simulator (CDPS) [3]. The variation of D retention with exposure time was examined to check the validity of existing model.

Tungsten specimens (A.L.M.T. Corp.) were irradiated with neutrons to 0.06 dpa in a fission reactor (BR2) at the Belgian Nuclear Research Centre and then exposed to a deuterium plasma in CDPS in the radiation controlled area of the International Research Center for Nuclear Materials Science, Institute for Materials Research, Tohoku University. The surface of the n-irradiated W specimens was electro-polished before plasma exposure in order to remove impurities and oxide layer. Deuterium ion flux and incident ion energy were 5.4×10^{21} $m^{-2}s^{-1}$ and 110 eV, respectively. The specimen temperature during plasma exposure was 563±5 K or 773±5 K. The plasma exposure time for each specimen was 1500 s, 6000 s and 24000 s. Deuterium retention was measured with the thermal desorption spectroscopy (TDS) with the rate of temperature ramp of 0.5 K/s.

The neutron irradiation resulted in significant increase in the D retention, as observed in [1]. The D retention in n-irradiated W increased clearly in proportion to the square root of the plasma exposure time. In addition, the TDS spectrum broadened toward high temperature region as exposure time increased. These observations suggest that the increase in D retention with increasing exposure time was due to deeper penetration of D. In the model proposed in [1, 2], it is assumed that there is an interface between the near-surface region where the traps are filled with D and the deeper region where majority of traps is empty, and the depth of this interface increases in proportion to the square root of time. The experimental results obtained in this study validate that the model proposed in [1, 2]. More quantitative information such as penetration depths and penetration rates will be reported in the presentation.

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Deuterium retention in re-solidified tungsten and mixed beryllium-tungsten material

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Leading edges of the ITER tungsten (W) divertor are expected to melt due to transient heat loads from edge localized modes (ELMs), and melting of the entire divertor surface will occur during vertical displacement events (VDEs) and disruptions. In addition, understanding tritium retention in plasma facing materials is critical for the successful operation of ITER due to safety reasons. Thus, the question of how melting affects hydrogenic retention is highly relevant for fusion devices, but limited data is available. Previous work [1] on re-solidified tungsten claimed that deuterium (D) retention in the melted layer was not significantly different from the unmelted surface; however, the surface was contaminated with carbon and the signal-to-noise of the nuclear reaction analysis measurement was low. Other work in JET revealed a ten-fold reduction of D surface concentration due to repeated transient melting from ELM heating on a modified sloped W lamella [2].

In the present study, we show that re-solidified tungsten subsequently exposed to D_2 plasma has a lower D retention than unmelted tungsten. Polycrystalline tungsten samples were melted using an Nd:YAG laser with power density up to 5 GWm^{-2} and 10 ms pulse duration. The re-solidified W was then exposed to deuterium plasma in the PISCES-A facility with fluence of 1.9×10^{25} Dm^{-2}, sample temperature ranging from 365 K to 573 K, and ion energy of ~75 eV. The depth of the laser-affected region was deeper than the plasma implantation and diffusion region. Control samples with no laser pre-melting were exposed to the same plasma, and thermal desorption spectroscopy (TDS) was used to measure D inventory in each sample after plasma exposure. Surface morphology was investigated with laser scanning microscopy and SEM imaging, revealing plasma-induced blisters on the unmelted samples and a smooth melt layer with grain growth on the laser irradiated samples. Plasma implantation and TDS were modeled using TMAP-7 with de-trapping energies of 0.9, 1.4, and 1.7 eV. The total D retention in the melted samples was reduced by 45% to 1.3×10^{20} m^{-2}, and the most significant effect due to melting was in the TDS release peak near 720 K (1.7 eV trap), which had a 77% reduction in the peak release rate compared with the unmelted sample. Modeling of the unmelted control sample indicated that the 1.7 eV trap had a significant near-surface D concentration. This, combined with SEM imaging, indicates that the lower D retention observed in the pre-melted samples was correlated with the suppression of blister formation.

The formation of mixed material due to eroded beryllium (Be) deposited on the divertor is expected in ITER. Transient heating of mixed Be-W material in PISCES-B demonstrated a lower cracking threshold of Be-W alloy compared to pure W [3]. In the present study, we show additional work on retention in laser-damaged and undamaged Be-W alloy.

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Fuel retention and recycling studies in long pulse H-mode discharges in EAST superconducting tokamak

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Fuel retention and recycling are critical issues for long pulse plasma operation in tokamaks, due to their strong effects on plasma density control and plasma confinement [1]. A comprehensive study on the fuel retention and recycling is carried out by using particle balance [2], in EAST tokamak with tungsten upper divertor and graphite lower divertor.

Major fuel retention during a discharge is always observed in the phase of Gas Puffing (GP) or Supersonic Molecular Beam Injection (SMBI) in EAST. Retention rate is ~ $(1.2\text{-}7.4)\times10^{20}$ D/s lower than GP or SMBI injection rate. Particle flux to divertor surface (measured by langmuir probes) strongly depends on plasma density and particle confinement time, but the relation between particle flux and fuel retention rate is very weak. These results indicate that fuel retention probably comes from the neutral energetic particles impacting the surface, which are produced by charge exchange between edge plasma particles and injected particles via GP or SMBI. These neutral particles would be deposited in the shadow region of first wall surface other than limiter or divertor surface.

Fuel recycling is strongly affected by plasma shape configuration, L/H-mode and heating power in EAST tokamak. Lower Single Null (LSN) divertor plasmas shows a lower recycling behavior than that in Upper Single Null (USN) divertor plasmas, probably due to a more closed structure of lower divertor in EAST, leading to higher particle exhausting rate in LSN. Recycling behavior is also different in L-mode from in H-mode. Particle flux to first wall surface in H-mode phase is lower than that in L-mod phase, inducing a lower particle exhausting rate, therefore H-mode plasmas has a lower absolute recycling flux but a higher global recycling coefficient (R_{global}) than L-mode plasmas. Experimental result also shows that increasing heating power could strongly enhance the fuel desorption from the first wall, both particle exhausting rate and fuel recycling are increased by higher heating power.

Low density discharges, with $(0.5 – 1.2)\times10^{19}$ m^{-3} and scanning strike points, is found to be an effective way to desaturate divertor surface ($<2.5\times10^{20}$ D/s) and to control fuel recycling. Lithium coating are also routinely used (15-30 g per day) to obtain a low recycling condition. 101.2 s long pulse H-mode plasma is achieved with plasma density 3.0×10^{19} m^{-3} and total injection power 3.0 MW. Fuel recycling is controlled to R_{global} ~ 0.87 – 0.95 during the whole discharge. An average wall pumping rate, i.e. the average retention rate during this long pulse H-mode discharge with lithium coating wall is ~ 1.0×10^{21} D/s. A continuous decrease of D_α emission and a gradual increase of Li-II emission during the whole discharge indicates that, fuel recycling is well controlled by the lithium film together with optimized plasma controls.

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Irradiation effect in tungsten resulting from ultra-high fluence deuterium plasma exposure in linear plasma devices STEP and Magnum-PSI

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To realize a commercially attractive fusion reactor, robust solutions to the issues of plasma facing materials (PFMs) are required. For future fusion devices, the effect of plasma irradiation on PFM performance is not well-defined under the ultra-high fluence and long duration plasma exposures expected. The cumulative plasma fluence to PFMs is expected to reach 10^{30}~10^{31} m^{-2} in a year in ITER. For DEMO and future reactors, the plasma fluence will be significantly higher than that of ITER due to the steady-state operation condition. Under high plasma fluence conditions, the safety issue related to the in-vessel tritium retention is critical considering the administrative limit of tritium. Additionally, degradation of PFMs under steady-state conditions may shorten its lifetime and should also be investigated. To fill the gap between present devices and future reactors, it is necessary to design and carry out dedicated experiment to study the PFM performance under ultra-high plasma fluence conditions.

A joint ultra-high fluence plasma exposure project is proposed under the collaboration between Beihang University (BUAA) and the Dutch Institute for Fundamental Energy Research (DIFFER). The goal of the project is to determine deuterium retention and surface morphology changes in tungsten under a single deuterium plasma exposure with an ultra-high fluence. Exposure experiments with a fluence up to 1×10^{28} m^{-2} are scheduled in STEP at BUAA and in Magnum-PSI at DIFFER in February, 2018. Experiment aiming at a fluence of 1×10^{29} m^{-2} will be carried out in Magnum-PSI if the success of the 1×10^{28} m^{-2} experiment is achieved.

Preliminary experiment with several tungsten grades (rolled tungsten and recrystallized tungsten) under deuterium fluences up to 2×10^{27} m^{-2} at surface temperatures of 450 K~500 K has been carried out in STEP. A strong dependence of blistering on plasma fluence is found. In rolled tungsten sample with surface direction parallel to the rolling direction, only very few small blisters with diameter less than 2 μm were found at 1×10^{27} m^{-2}, but both blister size and density were increased and many pores were present on the top surface at 2×10^{27} m^{-2}. Surface morphology changes in recrystallized tungsten at 2×10^{27} m^{-2} exhibited blisters with a high dome shape and flat top, while sizes ranging from 1 μm to 20 μm were widely observed. Besides, a high density of highly-oriented plate-like defects with an amorphous feature was observed by transmission electron microscope in the recrystallized tungsten, which is different from that observed at low plasma fluence.

The present result demonstrates that plasma exposures with high fluence up to 2×10^{27} m^{-2} have an effect on surface morphology which is significantly different from lower fluence exposures in both rolled and recrystallized tungsten. In the near future, the irradiation effect including deuterium retention and microstructure evolution in tungsten under higher fluence deuterium plasma exposure will be further demonstrated in the joint experiment in STEP and Magnum-PSI. These results would promote our understanding of PFM performance expected in future devices.

Continuously flowing liquid lithium as plasma facing divertor surface for tokamak and burning plasmas

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Liquid Lithium as the plasma pumping surface of divertor target plates in combination with Neutral Beam Injection (NBI) introduces the low recycling regime which can enhance plasma confinement by an order of magnitude. At realistic recycling of 50% the energy confinement is determined by plasma diffusion and by the best confined component (ions), while termal conduction plays only secondary role [1].

At the same time utilization of pumping by immobilized lithium is challenging. Wall conditioning last only for short time due to fast pacification of lithium surface by outgasing from the wall, while Li injecton to the wall enhance plasma edge cooling. Li flows, suggested for power extraction, are highly questionable due to magneto-hydrodynamic interactions with the strong magnetic field of tokamaks, limitation of 400^oC on Li surface temperature, and safety concerns.

The 24/7-FLiLi Flowing Liquid Lithium divertor, presented here, resolves the mentioned above fundamental issues with use of lithium in tokamaks. Aiming to implementation of the LiWall Fusion (LiWF) regime with recycling below 50%, it relies on the fact that the LiLi flow rate of 1 g/s is sufficient to absorb the now typical plasma particle flux of 10^{22}/s.

Accordingly, (a) the small flow rate makes the amount of free surface LiLi consistent with safety requirements; (b) the gravity driven flow with velocity of 0.5-1 cm/s and thickness 0.1 mm, determined by flow rate and LiLi viscosity, does not interact with tokamak magnetic field, small Reynolds and Harmann numbers make flow fully predictable and robust; (c) continuous 24/7 replenishment of free surface LiLi resolves the contamination problem: during plasma discharge 24/7-FLiLi pumps the plasma particles and creates the best possible confinement regime, while between shots ***highly chemically active 24/7-FLiLi serves as a "garbage collector" and cleans the vacuum chamber from outgasing from the wall***; (d) enhanced confinement and related special burning plasma LiWF regime [2] suggest the only practical way of solving the power extraction problem by enhanced confinement and reduction of heat flux to divertor; (e) contrary to existing misperception that high retention of tritium by Li prevents its use in tokamaks with burning plasma, ***24/7-FLiLi can be made uniquely consistent with needs in tritium recycling.***

The presentation describes the key elements of 24/7-FLiLi design, which relies exclusively on electromagnetic control of flow, and first steps of its development.

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The influence of helium on deuterium retention in beryllium co-deposits

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During the operation of a fusion device the wall material is eroded and re-deposited together with other species present inside the vessel. In the case of ITER, beryllium from the first wall will be re-deposited in the divertor region together with fusion fuel (D and T) and fusion ash (He). Formation of such co-deposits is an important mechanism of fuel loss and needs to be taken into account when estimating the total fuel retention in the wall. Many studies of D retention and release behavior from Be-D co-deposits have been previously conducted (e.g. [1, 2]), however the studies on the effect of He co-deposition on D retention are very limited.

The PISCES-B linear plasma device was used to produce Be-D-He co-deposits, similar to those expected to form in the ITER divertor area, in order to study the influence of He on D retention in such co-deposited layers. For this purpose a Be target was exposed to D_2-αHe plasma with a variable, spectroscopically determined He concentration ($0 \leq \alpha \leq 0.1$). Sputtered Be and back-scattered D and He particles were collected on a W plate directed towards the Be target and shielded from the direct interaction with the plasma. The average Be deposition rate was 0.07×10^{15} Be/cm^{2}s, resulting in the average layer thickness of 15 nm. The temperature of the W plates was varied between 300 K and 475 K.

Each W plate with a co-deposited layer consisted of two identical halves. The first half was used for TDS analysis, heating the sample at a rate of 18 K/min and using a high resolution residual gas analyzer to follow the partial pressures of H_2, HD, D_2 and He. The second half was used for NRA analysis with a ^{3}He ion beam to measure D and Be concentrations.

The results from the co-deposition with no He in the plasma compare favorably to the scaling law used to estimate the D retention in Be co-deposits [3]. Moreover, it is found that the inclusion of He in Be-D co-deposits has an observable effect at lower deposition temperatures. In the case of 10% He added to D_2 plasma a 23% and 37% decrease in D retention was observed for co-deposition at room temperature and 375 K, respectively. For higher deposition temperatures the influence of He on D retention is no longer evident. At reduced He concentrations ($\alpha = 0.01$ and 0.05 in the plasma) the effect of He is not obvious. In the case of ITER, He co-deposition is expected to play no role due to a high divertor temperature and low He concentration in the plasma.

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Observation of suppressed and aggravated D-induced blistering on pre-damaged W with different-flux D plasma exposure

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Tungsten (W), due to its high melting point, high thermal conductivity and high sputtering yield for energetic light particles, is considered as a promising candidate of plasma-facing materials (PFMs) in magnetic confinement fusion reactors. As a PFM it will inevitably face particle bombardment from intense hydrogen isotope (deuterium (D), tritium (T)) plasma and high-energy neutron. The possible formation of plasma-induced blisters raises concerns about long-term modifications of the material thermal and mechanical properties. So far, D-induced blistering formed on W with D plasma exposure has been widely investigated. However, the effect of neutron irradiation on D-induced blistering has so far been much less studied. In this work, the effect of pre-damage by heavy ion irradiation, as a proxy for neutron irradiation, on D-induced blistering of W is studied.

A 500 keV argon ion beam was used to produce dense displacement cascades in polycrystalline W. Subsequent D plasma exposures with high- (~10^{24} D/m^{2}s, in Pilot-PSI, DIFFER) or low- (10^{22} D/m^{2}s, in PSI-2, Forschungszentrum Jülich) flux were performed on pristine and pre-damaged W samples. During the two exposures, an incident energy of 40 eV was chosen; surface temperature was kept at ~550 K and a fluence of ~10^{26} D/m^2 was achieved. Surprisingly, different effects of pre-damage on D-induced blistering were observed in the two devices. After exposure to D plasma with flux of ~10^{24} D/m^{2}s, blistering was significantly suppressed on the pre-damaged W. On the contrary, after exposure with flux of ~10^{22} D/m^{2}s an aggravated blistering was found on the pre-damaged W. It suggests that under D plasma exposure with different fluxes pre-damage by heavy ion plays different roles in affecting D distribution and aggregation. Combining results from elastic recoil detection analysis measurements of near-surface D distribution and measurement of the total D retention by thermal desorption spectrometry, a hypothesis is proposed that high-concentration of D in the damaged layer promotes inward diffusion of the sequential incident D, thereby decreasing the likelihood of blister formation in the near surface. Low-flux D plasma exposure allows more D diffusing into the bulk and aggregate in a deeper depth in pre-damaged W, leading to the potential for forming large blisters with size of several tens to a hundred micrometer. While it is limited in the high-flux and short-term exposure case. The present results provide a possibility to mitigate blistering on W surfaces by predicting the critical condition of blistering under fuel plasma and high-energy neutron double irradiations.

Studies on hydrogen isotopes transport through ITER-like PFCs

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In ITER, the plasma-facing components (PFCs) will be subjected to intense plasma and neutral fluxes, the latter of which may directly reach the unshielded heat sink area (i.e. the bottom of the gaps between armor tiles, as shown in Fig. 1) and form a tritium leakage “short cut” [1]. The first-of-a-kind experiments on hydrogen isotope (D) transport through ITER-like W/Cu PFC mock-ups have been demonstrated using a newly-built steady-state plasma source at the Institute of Plasma Physics, Chinese Academy of Sciences.

For the permeation experiments, the gaps of the PFM tiles is ~0.5 mm and the angle between the gaps and the magnetic field line is set to be 5^o to ensure the heat sink is not exposed to charged particles from plasma. The plasma conditions are as follows: deuterium (D) is the working gas for plasma discharge, the electron temperature around 2 eV, the plasma density is of the order 10^{17} m^{-3}, the incoming D flux to the sample surface is estimated to be $\sim 1\times 10^{21}$ D m^{-2} s^{-1}.

The sample mock-ups have been found to be penetrated by D within several hours, which are much shorter than theoretical estimation for D diffusion through the 8 mm thick W bulk, suggesting that D can penetrate through heat sink without passing W PFM. Meanwhile, the measured permeation fluxes are orders of magnitude higher than calculated results for D through W, indicating D transport from the gaps of PFCs contributes the major fraction of hydrogen isotope leakage. For ITER divertor, where the neutral flux will be in the order of 10^{24} $m^{-2}s^{-1}$ [2], the T amount into the coolant may be underestimated without taking into account this permeation mechanism.

[1] Hai-Shan Zhou et al., J. Nucl. Mater. 493 (2017) 398.
[2] A.S. Kukushkin et al., Nucl. Fusion 45 (2005) 608.

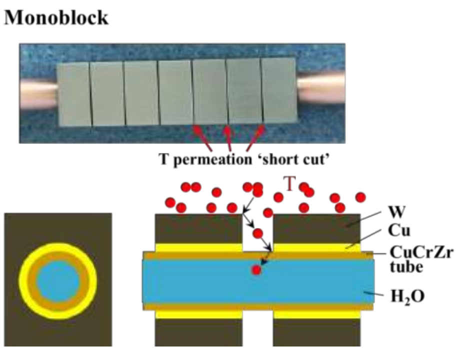

Fig.1. T permeation through ITER-like monoblcok

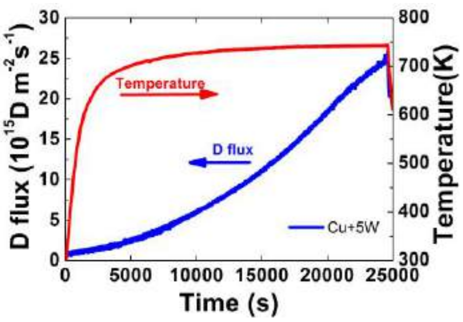

Fig.2. Measured D permeation flux through a W/Cu PFC mock-up.

Annealing and clustering of vacancies in tungsten and their influence on deuterium retention

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The hydrogen (H) inventory in tungsten (W) is governed by the presence of lattice defects acting as trapping sites for H. In addition to the intrinsic defects (dislocations, grain boundaries, etc.), in fusion reactors with a burning deuterium–tritium plasma, bombardment with 14 MeV neutrons will create a high concentration of radiation defects (vacancies, vacancy clusters, dislocation loops, etc.) through the entire thickness of the W armor, resulting in a significant increase in the trap density for H isotopes. Although these general trends have been elucidated, there is still a lack of understanding of how each defect type contributes to the total H retention and its temperature dependence. To unravel this, experiments studying H interaction using samples having one dominant and well–characterized defect type are required.

In order to introduce predominantly vacancies, high–purity single crystalline W specimens were damaged either by 4.5 MeV electrons or by 200 keV protons to low damage levels ($<10^{-2}$ dpa) at temperatures not exceeding 300 K. The samples were then annealed at temperatures in the range of 500–1800 K to investigate the stages of vacancy annealing and clustering. The resulting defects were characterized by positron annihilation lifetime spectroscopy (PALS) using a pulsed low–energy positron beam. In order to fill the defects with deuterium (D) without introducing additional damage, the samples were exposed to a low–flux (10^{20} D/m^{2}s), low–energy (10 eV/D) D plasma at a sample temperature of 450 K. The D concentration profiles in the samples were measured by nuclear reaction analysis (NRA) and the D binding states in the defects were determined by thermal desorption spectroscopy (TDS). The non–damaged samples exhibited a single bulk lifetime component of about 110 ps indicating a very low concentration of intrinsic defects. Damaging both by electrons and protons resulted in the appearance of a long–living positron lifetime component (about 200 ps), suggesting that in both cases the same type of defects was introduced. Annealing at temperatures above 600 K increased the value of the long–living component, indicating agglomeration of vacancies in clusters. The full recovery of the defects was observed after annealing at 1800 K, i.e. above the recrystallization temperature. Several temperature ranges corresponding to different sizes of vacancy clusters were identified by comparing the measured lifetimes with calculated ones using density functional theory [1]. The TDS spectra from the as–damaged samples exhibited a dominant peak near 650 K. After annealing at temperatures above 600 K this peak was replaced by a peak close to 930 K, indicating that the vacancy clusters exhibit a considerably higher D binding energy which is practically independent of the cluster size.

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Power Exhaust, Plasma Detachment and Heat Load Control

Investigating hydrogen plasma-chemical processes using Optical Emission Spectroscopy in detached Magnum-PSI scenarios

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In ITER and other next-generation fusion devices, divertor detachment will be critical to limit the heat flux to plasma-facing components, by means of plasma-neutral cooling channels and volume recombination. The linear device Magnum-PSI can produce plasmas in the same parameter range as the ITER divertor region, with electron temperatures $T_e \sim 1 - 5$ eV and densities $n_e \sim 10^{20} - 10^{21}$ m^{-3}. Experiments in Magnum-PSI are therefore vital to guide and validate predictive modelling code suites that aim to simulate detachment in the ITER divertor. A question of particular interest is which atomic and molecular processes involving the hydrogen species H, H_2, $H_2^{r,v}$, H_2^+, H_3^+, and H^- are essential to be included in models.

In Magnum-PSI, conditions of detachment with a high degree of plasma-neutral interactions near the target are mimicked by seeding hydrogen gas in the last of three differentially pumped vacuum chambers, raising the neutral background pressure from 0.4 Pa to up to $\sim$10 Pa in the final $\sim$80 cm in front of the target. Optical emission spectroscopy (OES) of the Balmer series using the recently installed multi-chord Czerny-Turner spectrometer enables the determination of radial profiles of excited H^* in the plasma beam from excited state $n = 4$ up to $n \approx 10$. Different atomic and molecular processes preferentially create H^* in different excited states, so OES can shed light on which processes are at hand. Further, Thomson Scattering (TS) measurements are used to monitor T_e and n_e at the same measurement location as OES.

TS measurements show that as pressure is increased, T_e falls monotonously and n_e first increases, then rolls over, and finally decreases. From the OES measurements, three pressure regimes can be distinguished. For low pressures, emission profiles are peaked in the centre of the beam, and the lower excited states are typically overpopulated with respect to the Boltzmann distribution. For medium pressures, emission profiles become hollow, and the overpopulation of lower excited levels becomes milder. Finally, for high pressures, emission profiles are peaked again, and underpopulation of the lower excited states is seen, particularly near the edge of the plasma.

These results suggest that in the regimes of low, medium, and high pressure, the atomic state distribution of H^* is dominated by ionization, Molecular Activated Recombination (MAR), and Electron-Ion Recombination (EIR), respectively. A detailed investigation of the relative contributions from the different species and processes is ongoing, which will rely on collisional-radiative modelling

Calibrated Helium and Carbon Ion Flow Measurements in the DIII-D Divertor Plasma

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UEDGE modelling with drifts included has highlighted the impact of poloidal and radial **ExB** drifts on divertor power exhaust in DIII-D[1]. To further explore the influence of drifts on ion flows in the divertor and Scrape-Off-Layer (SOL), two new calibrated Coherence Imaging Systems (CIS) have been employed, one viewing the whole plasma cross section[2], and a second focusing on the divertor; this paper focuses on the divertor. The divertor CIS system employs an interferometer that imprints interference fringes on the divertor plasma image. The phase of the fringes is proportional to wavelength, and an image of the line-integrated Doppler shift is obtained. A new tuneable calibration laser and precision wavemeter (0.01 pm) provides the "rest" wavelength for either CIII ~465 nm or He II ~468.6 nm plasma emissions resulting in < 1 km/s uncertainty in ion velocities that are in the range of ~30 km/s in the divertor plasma. The 2-D line-integrated velocity images have been tomographically inverted to obtain maps of ion velocity with time resolution ~ 0.3 ms.

These new calibrations have increased the confidence in ion velocity images and enabled new physics understanding. DIII-D was operated with He neutral beam injection into helium plasmas with both directions of the toroidal field B_T. UEDGE modelling indicates that the primary ion component near the divertor plate is He II (468 nm), becoming a minority species due to ionization moving away from the plate. The He II ion velocity is observed to change direction with B_T near the divertor plate as expected (flow towards the target in both cases). Near the x-point, where He II is more of a minority, the flow pattern is similar, suggesting entrainment. UEDGE modelling with drifts using the unique DIII-D Divertor Thomson Scattering (DTS) data agrees well at the plate, but *under*estimates the velocity upstream near the x-point [3]. CIII ion flow data indicates that the sign of the velocity is the same as the main ion; work is in progress to further quantify how well this ion is entrained. New preliminary data on how strongly the CIII ion flow is entrained in the He or D background plasma, and the influence of detachment on flows will be presented.

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[2] C.M. Samuell, et al., Journ. Instrum., **12** (2017) C08016
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Simulation Study of the Divertor Operation for a DEMO Fusion Reactor

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Handling of a large thermal power exhausted from the confined plasma is one of the most important issues for ITER and DEMO. Recently, the divertor design concepts were proposed for the Japan (JA) and Europe (EU) DEMOs with similar major radius of R_p = 8-9 m [1-3]. The radiative cooling scenario with impurity seeding is a common approach, and large radiation fraction totally in the main plasma, SOL and divertor ($f_{rad}^{tot} = P_{rad}^{tot}/P_{heat} > 0.8$, where P_{heat} is the total heating power of 430-460 MW) is required in order to reduce the peak heat load at the divertor target (q_{target}) to less than 10 MWm^{-2} level. In the JA DEMO, the divertor with a long leg of L_{div} = 1.6 m was proposed as a reference case to handle P_{sep}/R_p ~30 MW/m, which corresponds to two times larger than ITER and EU DEMO divertors. The divertor simulation result by SONIC code showed that q_{target} would be reduced to less than 10 MWm^{-2} in the detachment with Ar seeding and $f_{Ar} = n_{Ar}/n_e$ = 0.005-0.007 in the main, SOL and divertor [1].

Recently, key plasma parameters have been investigated to produce the plasma detachment and to determine the operation limits of the divertor design. Following result of case (1) was obtained, and calculations of other cases (2) (3) are in progress. In addition, comparison with those in the divertor of ITER size (L_{div} = 1.1 m) started.

(1) Radiation fraction in the SOL and divertor, i.e. $f_{rad}^{SOL+div} = (P_{rad}^{SOL}+P_{rad}^{SOL})/P_{heat}$, was reduced from 0.44 to 0.40-0.35 by decreasing the Ar puff rate. In the outer target, the partial detachment became narrow from r_{target} < 14 cm to 10-5 cm, and q_{target} was increased from 5 MWm^{-2} to 7-11 MWm^{-2}. At the same time, T_e^{div} at the attached region became higher from 20-30 to 50-100 eV, which would enhance net erosion of the tungsten (W) target by Ar ions by a few mm level during a year-long full power operation. Net sputtering rate is investigated by using the kinetic full-orbit impurity transport code (IMPGYRO) [4], under the partial detachment condition.

(2) Exhausted power from the core-edge boundary (r/a = 0.95), P_{out}, as an input parameter, is increased from 250 to 310-350 MW with increasing fusion power, P_{fusion}, providing an improvement of the impurity shielding at the edge, $f_{Ar}^{SOL}/f_{Ar}^{main}$, from 1 to 1.5-2, with similar $P_{rad}^{SOL}+P_{rad}^{div}$ as the case (1).

(3) Reduction in diffusion coefficients (χ and D): while χ = 1 m^{2}s^{-1} and D = 0.3 m^{2}s^{-1} in JA DEMO simulation were the same as those in the ITER simulation [5], e-folding length near the outer midplane separatrix ($\lambda_{q//}^{SOL-OM}$) of the parallel heat flux profile corresponded to 1.9 mm, which was narrow compared to 3.6 mm in ITER simulation due to higher T_e in the DEMO (350 eV). On the other hand, Eich's scaling [6] predicts smaller $\lambda_{q//}^{SOL-OM}$ of 1.3 mm. Effects of the smaller χ and D on the plasma detachment and heat load profile are studied to identify limits of the operation parameters in the long leg divertor.

These studies will provide a candidate of the conventional divertor design such as the divertor size in the DEMO design with large P_{sep}/R_p, and key operation parameters for the divertor design of the reactor.

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[5] A. Kukushkin, et al, J. Nucl. Mater. 438 (2013) S203. [6] T. Eich, et al. Nucl. Fusion 53 (2013) 093031.

High radiation scenarios and the X-point radiation regime at ASDEX Upgrade

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Future fusion reactors require a safe, steady state divertor operation. A possible solution for the power exhaust is the detached divertor operation, where the power leaving the core plasma is dissipated, primarily through radiation, being spread over large areas and lowering peak heat loads on surfaces. The radiation can be increased by seeding impurities. Depending on the used element and the plasma parameters, the radiation losses can be concentrated on specific plasma regions. In current devices, N is used for dominant divertor and scrape-off-layer (SOL) radiation and Ar for SOL and pedestal radiation. While ITER is planned to operate with a partially detached divertor, where only minor core and dominant SOL radiation is required, a DEMO-like device is predicted to require full detachment and a dissipated power fraction of about 95 % of the exhaust power, which is only achievable with additional radiation from inside the confined region. Here, a possible detrimental effect of the central radiator on confinement needs further investigation.

At the ASDEX Upgrade tokamak (AUG) with a full tungsten wall, stable full divertor detachment is achieved at the highest heating powers of up to 26 MW with radiated fractions of up to 90 %. This requires substantial seeding of either Ar or N. By the high enrichment of impurities in the divertor, this can be achieved with acceptable core concentrations (about 2 - 3 % in the case of N seeding).

With both, N or Ar, in the fully detached regime the main radiative losses are concentrated in a toroidally symmetric region of about 5 cm in diameter near the X-point. With increasing seeding, this radiation zone moves inside the confined region. However, even with such a strongly radiating region inside the separatrix, stable plasma operation is maintained with a reduction of the energy confinement time of only 0 - 20 %.

In this contribution, it is demonstrated that the position of X-point radiator can be controlled and an active feedback scheme using N seeding or heating is discussed. The active control will allow to study the impact on the core energy confinement in further detail and might help to reveal the physics mechanisms behind the X-point radiation. A possible extrapolation of the X-point radiation regime to ITER and DEMO operational parameters will be discussed.

Improved inference of atomic physics processes in detachment using Bayesian filtered camera tomography

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Filtered camera tomography is a crucially important tool for studying detachment in fusion devices. By providing a map of emissivity for a given spectral line across the (R, z) plane, it is the only diagnostic that offers 2D information on atomic physics processes, such as ionisation and volume recombination, which are central to detachment behaviour [1]. However, through testing with synthetic data we find that conventional approaches to divertor camera tomography can introduce significant artefacts and inaccuracies into the reconstruction. Additionally these methods cannot provide uncertainties associated with the reconstruction, leaving us unable to draw conclusions from the results with confidence.

We present several improvements to divertor tomography methodology, in addition to examples of how these have been successfully applied to TCV experiments, and synthetic MAST-U test cases. The linearisation of the tomography problem has been improved by including a higher-order approximation of the underlying emissivity, by switching from a rectangular grid to a triangular mesh and making use of barycentric interpolation. This recasts the solution as the emissivity at each vertex of the mesh, allowing it to properly conform to the often complex geometry of material surfaces in the divertor. A Bayesian solution to the linear system has been implemented as an extension of earlier work at JET [2], which properly accounts for uncertainties in the camera measurements and constrains the solution to physically-sensible length-scales.

The result of these improvements is a significant reduction both in the severity of artefacts introduced, and the average absolute error of the solution. In a synthetic TCV test case the Bayesian solution, combined with a refinement step which imposes additional physicality constraints, yielded roughly a factor of 4 reduction in the average absolute error when compared with conventional methodology. Additionally, the reconstructed emissivity is now well-behaved at material surfaces, and with the uncertainty estimates provided by the Bayesian approach, we are able to draw more informed conclusions from our experimental measurements.

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Power Exhaust and Detachment in Divertor Tokamaks with 3D Magnetic Perturbations in ASDEX Upgrade

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One of the major challenges for future fusion devices, such as ITER, is the control of heat flux onto the divertor targets. If the steady state peak heat flux is not mitigated it will exceed the material limit of about 10 MW/m^2 substantially. The ways to mitigate the target heat flux and access a detached divertor state have been investigated in numerous studies, which largely assume an axisymmetric magnetic field geometry. However, in recent years 3D Magnetic Perturbation (MP) fields, have been increasingly applied in several tokamaks, in order to control edge localized modes. If MP fields are applied in future fusion devices, their beneficial as well as harmful consequences for power exhaust have to be studied.
Under attached conditions, IR thermography shows the occurrence of toroidally localized heat flux maxima with MP fields. It has been speculated that these may 'burn-through' an otherwise detached divertor leading to intolerably high heat flux densities. In future fusion devices, this would require countermeasures, which would entail substantial engineering efforts. On the other hand, MP fields could lead to the beneficial effect of an increased toroidally averaged power decay length, and an increase of the volume in which impurities, such as nitrogen, can radiate efficiently.
In this contribution, experimental and numerical results of ASDEX Upgrade (AUG) discharges with MPs will be presented and their implications in view of ITER discussed. Based on measurements by divertor Langmuir probes in AUG with a rotating MP field, it will be argued that a burn-through event is unlikely in ITER, since toroidal asymmetries are smoothed out by perpendicular transport at low divertor plasma temperatures. In addition to that, the comparison of the experiments with simulations by the transport code EMC3-EIRENE revealed several strong arguments for the importance of screening currents induced in the plasma that reduce the amplitude of the MP fields in the confinement region. Due to this, the impact on both, the toroidally averaged fall-off length $< \lambda_q >$ and the radiated power is rather small and barely measurable with the present diagnostics in AUG. In ITER, however, where the power decay length is predicted to be of the same order as in AUG, while the radial perturbation of field line paths is much larger, a much stronger effect of the MP field is predicted.

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** *See the author list "H. Meyer et al 2017 Nucl. Fusion* ***57*** *102014"*

Extending the boundary heat flux width database to 1.3 Tesla poloidal magnetic field in the Alcator C-Mod tokamak

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The boundary heat flux width (λ_q), along with the power flowing into the boundary, sets the peak heat flux that must be exhausted in the boundary of magnetic confinement fusion reactors. Understanding what sets λ_q has largely been an empirical science [1], however physics understanding is progressing [2-4]. Results from a 6-machine international database of measurements of λ_q at the outer divertor in H-mode indicated that the poloidal magnetic field at the outer midplane (B_p) was the only significant parameter associated with the heat flux width: λ_q [mm] = $(0.63\pm0.08)\times(B_p\ [\mathrm{T}])^{-1.19\pm0.08}$ [1]. The maximum B_p in the database was ~0.8 T, whereas the 15 MA scenario in ITER will be 50% higher at ~1.2 T. Alcator C-Mod has been the only diverted tokamak in the world capable of operating at and above ITER-level B_p. Therefore, a major focus of the final experimental campaign on C-Mod was to extend λ_q measurements to reactor-relevant B_p, up to 1.3 T. Measurements clearly indicate a continuation of the inverse scaling of λ_q with B_p in H-mode up to and exceeding ITER-level B_p. While these results are broadly consistent with the HD model [2] they will, perhaps more importantly, provide a benchmark for first principles models [3,4], one of which presently projects [3] to ~10 times larger λ_q than the empirical scaling for ITER at the same poloidal magnetic fields.

In addition to the high-B_p measurements, we have assembled a database of λ_q measurements consisting of over 300 shots that span nearly the entire operating space of Alcator C-Mod (L-, H- and I-modes) under attached divertor conditions. As seen in earlier studies [5], λ_q at fixed values of poloidal magnetic field exhibit significant scatter that appears related to the core plasma confinement quality, i.e., discharges with the highest stored energy tend to have the smallest λ_q. Using the extended database, we are presently exploring correlations of λ_q with global plasma parameters and with conditions in the pedestal; we will report on the latest results at this meeting. In addition, the database now includes a composite of measurements made by surface thermocouples and Langmuir probes, both benchmarked against calorimetry. These sensors have much better spatial resolution and heat flux dynamic range than IR thermography, allowing for more accurate fits of λ_q to the measurements and resolving the role of heat flux spreading into the private flux region. We find that the standard analytic form [1] that assumes a symmetric, diffusive-like spreading of heat toward private and common flux regions is not appropriate under many conditions. Instead, we find that a 2-decay length model for both the private and common flux regions is a more appropriate empirical description.

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Comparison of DTT conventional and advanced divertor configurations

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The role of the DTT facility [1] is to bridge the gap between today's proof-of-principle experiments and DEMO [2]. It will help the development of a reliable solution for the power and particle exhaust in a reactor. To this aim DTT has been designed to study a large suite of alternative divertor magnetic configurations in order to ensure acceptable conditions at the walls while maintaining sufficient core performance. All of the present more promising alternative divertor configurations are realizable in DTT: the flux flaring towards the target (X divertor), the increasing of the outer target radius (Super-X) and the movement of a secondary x-point inside the vessel (X-point target) as well as the entire range of Snowflake (SF) configurations [3] and the presently reconsidered double null (DN). Most of previous configurations are produced using out-of-vessel coils but in DTT it is also possible a fine tuning of the magnetic field in the divertor region by small in-vessel coils. Here, we present a first comparative power exhaust study of conventional Single Null (SN) and alternative configurations by using the SOLEDGE2D-EIRENE [4] code which is one of the few codes able to deal with all presently envisaged divertor configurations. Closed divertors, with a full W wall, no impurity seeding and a level of power crossing the separatrix $P_{SOL}\approx 25MW$, have been considered in the simulations. In addition, the transport coefficient has been set up constant and an outer midplane decay length of 3 mm in SN attached condition has been assumed. A density scan for both the conventional and advanced configurations has been performed in order to investigate the behaviour of the different magnetic divertor solutions realized on the same vessel and divertor targets. In SND high power loads are foreseen by the code independently from the density, with a peak values higher than 20 MW/m^2. Lower peak power has been obtained in the alternative configurations. Furthermore, the codes predict detachment conditions for lower value of the upstream density. This behaviour is probably related to the benefit deriving from the geometrical feature of alternative configurations like the increase in the flux expansion and connection length.

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SOLPS modelling of detachment in the new Small Angle Slot divertor in the DIII-D tokamak

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SOLPS5.1 [1] modelling of divertor experiments carried out using the new Small-Angle Slot (SAS) divertor prototype [2] installed at DIII-D has indicated that detachment, strongly dependent on the orientation of the magnetic geometry within the slot, occurs at lower upstream separatrix density than a flat, open divertor, when the strike point is located near the small-angle slot target, in agreement with experimental observations.

The SAS divertor leverages the benefits of a gas-tight slot geometry with a critical small target angle to achieve simultaneous control of heat flux and erosion at relatively low plasma density required for non-inductive current drive in future steady-state tokamaks.
Monte-Carlo trajectories of neutrals from EIRENE simulations show that, for otherwise identical discharges, in contrast to the open divertor (horizontal target), SAS targets redirect recycling neutral towards the separatrix and into the private region (vertical + slot effect) [2] resulting in the build-up of the molecular density, which is further enhanced by the D_2-D+ elastic collisions. Early measurements from newly installed ASDEX gauges in SAS confirm a larger pressure in the near SOL compared to the far SOL when the strike point is located in SAS. Further detail comparisons will be made with the pressure gauges as well as local heat flux measurements from newly installed Surface Eroding Thermocouples.

SOLPS modeling of density scans shows that SAS achieves both lower T_e and heat flux at the strike point compared to the open divertor, consistent with the measurements from the target-embedded Langmuir probes. The modelling reveals a correlation between lower T_e, higher molecular densities and increased power losses in the slot, pointing out that it is the neutral dynamics which determines how effectively T_e and heat flux are reduced. Changes in the neutral ballistics, influenced by the small-angle target, promote enhanced volumetric losses facilitating divertor cooling.

In addition, SOLPS modelling shows a strong dependence of divertor cooling on strike point location [3]. In particular, the modelling highlights that there is an optimal strike point location in SAS, i.e., near the small-angle target plate, indicating an important relation between magnetic geometry and closure [4]. These findings have been confirmed by Langmuir probe measurements of T_e. This strongly suggests that rather than the closure alone, it is the synergy of a slot structure and a small target angle that is the reason for the higher neutral dissipation which characterizes the SAS divertor.

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Spontaneous plasma detachment studies in divertor relevant helicon plasmas

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We report observation and detailed studies of spontaneous axial plasma detachment in a radio frequency (RF) powered helicon plasma device, Controlled Shear Decorrelation eXperiment (CSDX). CSDX plasmas simulate the scrape off layer and divertor region of fusion plasma devices (plasma density ~ 10^{19} m^{-3} and electron temperature ~ 3 - 12 eV). The spontaneous (not due to any external perturbation such as a gas puff or biased electrodes) axial plasma detachment occurs simultaneously along with a self-organized global transition in the plasma dynamics, via a transport bifurcation with strong hysteresis, at a certain B_crit [1, 2]. For B < B_crit, the plasma is axially attached to the end of the 3 m long device and is dominated by density gradient driven resistive drift waves, rotating in the electron diamagnetic drift direction with low ***m*** (azimuthal mode number) modes. For B > B_crit, the plasma detaches axially from the end of the machine and a recombination region is observed with a very distinct recombination front. In this detached phase, the plasma exhibits steepened density and ion temperature gradients, strong shearing in the both the azimuthal and parallel velocities, and multiple, simultaneously present, radially separated plasma instabilities. The plasma is dominated by high ***m*** modes rotating in the ion direction at the center (***r*** < 2 cm), resistive drift waves at the location of the strongest density gradient (2 cm < ***r*** < 5 cm) and strong velocity shear driven instabilities at the edge (***r*** > 5 cm). Evidence from both Langmuir probes and fast framing camera imaging show the existence of a radial transport barrier in this detached plasma phase. The axial detachment also follows the same hysteresis curves associated with the transport bifurcation that led to the transition. Intermittent signatures of the ion mode occurring just prior to the plasma detachment seems to signify that changes in the plasma turbulence and transport characteristics lead to the axial plasma detachment. The phenomenon is universal, but the particular value of B_crit, and the width of the hysteresis loop, depends on the helicon source parameters (neutral pressure, RF power etc.) used. This study allows access to new regimes in a helicon device to study plasma turbulence and transport related to plasma detachment. Spontaneous axial detachment studies have serious implications on the relevance of similar RF heated helicon devices [3, 4] designed to study plasma material interactions (PMI). Two dimensional bifurcation diagrams show the regimes necessary to avoid plasma detachment for these devices to be relevant to PMI studies. In addition, this work also gives us the opportunity to study instabilities and fluctuations associated with plasma detachment, similar to those observed in fusion devices such as ASDEX-Upgrade [5] and are currently an important field of investigation for plasma detachment studies [6, 7].

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TECXY simulations of the onset of plasma detachment in the TCV tokamak

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An accurate description of the plasma transport in the tokamak edge and processes taking place during detachment are crucial as they strongly affect the power exhaust capabilities of the plasma. The TECXY edge plasma model has been used to investigate transport and energy dissipation processes preceding plasma detachment in the TCV tokamak experiments.

Calculations have been performed for a standard single null configuration of the TCV tokamak on the basis of recent TCV experiments of Ohmically heated plasmas [1]. A two-dimensional multifluid model of the edge plasma with Braginskii transport equations has been solved using the TECXY code [2]. The model assumes classical plasma and impurity parallel transport and anomalous transport across flux surfaces with ad hoc heat and particle transport coefficients as well as a self consistently calculated carbon sputtering from the target plate.

Experiments and modelling have been carried out to investigate the CIII emission evolution along the outer and the inner legs close before detachment stage when the ion current to the target drops [1,3,4]. Simulations performed with the TECXY code can reproduce the experimentally observed CIII line radiation pattern of carbon sputtered from the target plates. The numerical calculations show, in particular, the observed movement of the region of strong CIII radiation from both target plates to the X-point region along the separatrix in the early detachment stage.

The performed studies enabled applied edge plasma model validation and allows to describe the exhaust processes involved in the primary stage of detachment on the TCV tokamak. Thereafter, the TECXY model can be applied to comparison of advanced divertor configurations.

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Detachment studies in the Magnum-PSI linear device during both steady state and transient plasma operation

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Achieving divertor detachment is greatly facilitated by increasing the neutral background pressure, either by divertor closure, neutral trapping or gas seeding. Detachment studies on the Magnum-PSI linear device have been performed to investigate the relevant volume and surface processes responsible for detachment in tokamaks. The interaction of the plasma with a neutral background plays a crucial role in achieving a detached plasma regime. Detachment in Magnum-PSI is hence achieved by raising the H_2 background pressure by seeding H_2 in the target chamber. Detailed measurements of the various phases of the detachment process (with varying H_2 background pressure) give insight into the underlying plasma processes. Transient ELM-like pulses are superimposed onto the steady state detached scenario to probe the resilience of the detached scenario and to characterize the effect of high neutral background pressures on transient pulses.

In steady state operation, (attached) reference scenarios with electron temperatures of 3-5 eV and densities of around 10^{20} m^{-3} where used. Increasing the H_2 neutral background pressure first results in a decrease in T_e and an increase in n_e at approximately constant plasma pressure. At background pressures above 1 Pa, the heat and particle flux to the target and the plasma pressure start to drop, accompanied with a strong increase of Balmer line emission from the plasma. Background pressures up to 16 Pa where applied, leading to a fully extinguished plasma over a distance of 35 cm. A detailed analysis of the radially resolved n_e and T_e profiles, the heat and particle fluxes to the target, and the radiated power from the plasma will be presented.

The cascaded arc plasma source of Magnum-PSI is capable of producing short (about 1 ms) intense transient pulses with temperatures up to 20 eV and densities above 10^{21} m^{-3}. These pulses where applied to fully detached steady state plasmas to test the resilience of these scenarios. Detailed time resolved Thomson scattering profiles accurately characterize the size and shape of these pulses. Although all pulses where observed (by a fast framing optical camera observing the plasma beam and an IR camera monitoring the target) to break through the detachment, a strong effect of the high background H_2 pressure on the pulse height and shape was observed. A significant increase in density and a decrease in temperature point to strong ionization of the background neutrals. Although full detachment of the pulses was not achieved in the relatively short target chamber of magnum-PSI, both increased n_e and decreased T_e would greatly facilitate further volumetric losses in a much longer connection length tokamak scrape-off layer geometry.

Principles of Closure in the DIII-D SAS 2 Divertor for Optimal Heat Dissipation and Particle Control

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SOLPS 5.1 modeling of the next generation of Small-Angle Slot divertor [1] on DIII-D, termed SAS 2, reveals how recycling neutrals and impurities can be leveraged to dissipate the divertor parallel heat flux and reduce target erosion more efficiently than a similarly closed, flat divertor, by a factor of 2 or greater, by extending the benefits from closure concepts previously investigated on several devices [2,3,4]. SAS 2 models also indicate great potential for improved pumping efficiency; neutral trapping near the strike point increases local neutral density by a factor of 3 or greater, such that the chosen conductance path to the cryopump from the slot may be more flexible, due to higher collisionality. Greater neutral and impurity trapping in the SAS 2 divertor may facilitate detachment onset at the target while reducing fueling across the separatrix, enhancing core heating and current drive capability with suitable divertor heat flux control.

The prototype SAS 1 divertor utilizes features of both horizontal and vertical targets for the purpose of increasing dissipation at the strike point [1]. Modeling of the next-generation SAS 2 divertor, to be installed in 2021, aims to further improve SAS heat dissipation with a systematic approach to target shape and closure optimization. In this optimization procedure, the SAS 2 model target is parameterized into four contiguous segments of length L_n and orientation θ_n. Fifty-six permutations of L_n and θ_n are investigated in SOLPS as SAS 2 candidate targets, spanning a range of concave and convex features designed to control the distribution of recycled neutrals and impurities for efficient reionization in the divertor. Several candidate SAS 2 targets demonstrate superior strike point cooling over a flat target with similar closure. This optimization depends strongly on the anticipated heat footprint (midplane $\lambda_q \approx 2$ mm, $q_{\|,div} \approx 100$-150 MW/m^2) and magnetic geometries for high-power operation in DIII-D. Consequently, consideration is also given to SAS 2 models' sensitivity to input power ($P_{SOL} = 3$-12 MW) and cross-field transport parameters ($\chi_i = \chi_e = 0.2$-0.4 m^2/s), which are expected to strongly affect the state of detachment in the slot. Detailed analysis of the detachment onset density with $q_\|$ will be presented. Finally, analytical models of neutral conductance will be presented, proposing suitable pumping configurations for SAS 2. Sufficient neutral back-conductance from the pumping plenum may enable SAS 2 to operate in two distinct modes, emphasizing either greater detachment control or greater particle control.

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Initial Results of the First Wendelstein 7-X Island Divertor Experiments

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After a successful initial plasma operation phase with limiter configuration [1], the stellarator Wendelstein 7-X was upgraded with a pumped, inertially cooled island divertor. It consists of two divertor units in five toroidally periodic modules that are made of carbon and radially intersect the outer half of a large edge magnetic island chain. The divertor operation enabled plasma discharges for up to 20s with a plasma energy of up to 839kJ, 200ms energy confinement time and a line averaged density of $1.1x10^{20}$ m^{-3}. The edge plasma, typically T_e = 20 to 50 eV, is strongly influenced by gas recycling at the divertor, impurity generation (mainly carbon) and by the edge magnetic topology. The results of a first systematic study of these divertor properties in different magnetic configurations are presented in this contribution. Divertor heat loads were derived from infrared thermography on all ten divertor elements. It confirms that the designed divertor surface allows a symmetric distribution of power loads of up to 10 MW/m . Various magnetic configurations can be accessed by the modular magnetic field coil system of Wendelstein 7-X. A change of the edge rotational transform ι results in different edge islands intersecting the divertor with rather different strike line patterns. The compensation of a residual n/m = 1/1 magnetic perturbation mode with an external coil set is necessary to allow for safe operation of the future next-stage divertor, which will incorporate active water cooling. The dependency of different plasma fueling scenarios on the fuel injection location (gas puffing or pellet injection) and edge density and temperature conditions was investigated. Fast visible cameras allowed, for example, the tracing of fueling pellets paths and the characterization of their penetration depth. Since core fueling to higher densities proved to be challenging, injections of Ne or N_2 were used to lower the edge plasma temperature and allow better penetration of neutral gas. Impurity flows could be investigated with edge spectroscopy channels as well as a camera system with filters of hydrogen and carbon lines. The ratio between H_α and H_γ line radiation revealed high recycling divertor regimes. This, in combination with vanishing heat fluxes as seen by infrared thermography and divertor thermocouples, hints at a first successful partial detachment. Radiative collapses in high density operation were investigated in their relation to impurity generation and inward transport, based on the carbon line emission pattern.

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Detachment dynamics and sensitivity to control parameters in 1D simulations

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Detachment is an important process for high power fusion devices, which to operate must balance the often conflicting requirements for high performance, tolerable target heat fluxes, and efficient pumping of helium ash and impurities. The optimization of detachment can only be accomplished if we can control the detachment process including the amount of power and ions reaching the target and the location of the various sub-processes (e.g. radiation, ionization & recombination) along the distance from the target to the x-point.

A 1D *time-dependent* code (SD1D, based on BOUT++[1]) has been developed, and used to study the dynamics of detachment. The model includes a neutral fluid for atoms and coupling to ADAS for fixed carbon impurity concentration radiation. Magnetic topology effects such as variable field line length (poloidal flux expansion) and total flux expansion (moving the outer strike point to large major radius as in a Super-X divertor) are included as well. The model steady-state solutions have been successfully benchmarked against the extended 2-point model [2], and comparison to SOLPS with kinetic neutrals is ongoing.

Results of an area (total flux) expansion study confirm earlier SOLPS [3] and analytic model [4] predictions, showing that doubling the flux expansion at the target relative to the X-point halves the upstream density at which detachment occurs. In addition, we find that after detachment the motion of the detachment front is less sensitive to variation in upstream density at higher expansion factors, leading to an increase in the detachment window [4].

Variations in the loss rate of neutral atoms, and the redistribution of neutrals along the divertor leg, are used to study the effect of open and closed geometries on the detachment threshold and dynamics. In MAST-Upgrade like conditions neutral confinement times below 100 microseconds are found to significantly affect the high recycling regime, reducing or removing the nonlinear increase in target flux with upstream density, and moving the detachment threshold to higher upstream density.

Time-dependent simulations, using a PID feedback controller to set the upstream density by varying the upstream particle source, are used to study the speed of the detachment front motion and its response to external perturbations in power and density. Several timescales are identified: thermal energy propagation; sound waves due to pressure imbalances; and particle balance timescales. Close to detachment threshold regimes are found where the front moves at close to sound speed, and small windows of hysteresis are identified. In some cases the interaction between feedback control and detachment threshold can lead to limit cycle oscillations which do not reach a steady state solution.

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Exploration of Radiative Edge Cooling in the Island Divertor at Wendelstein 7-X

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Intrinsic and actively injected impurities are shown to directly impact on plasma edge conditions in correlation to the 3D magnetic structure of the scrape-off layer (SOL) at Wendelstein 7-X. The location of intrinsic impurities is directly correlated to the main strike line heat and particle fluxes. Therefore, active gas injection must be aligned with the island divertor geometry to be effective. The cooling capacity from dedicated injected gases is used to steer the heat flux to plasma facing components and also to manipulate the neutral fueling behavior. The results presented are an important first time exploration of the impact of radiative edge cooling by impurities on transport in the plasma SOL and at the core-edge interface.

The study encompasses a combination of results from a numerical analysis with the coupled 3D fluid plasma edge and kinetic neutral transport Monte-Carlo Code EMC3-EIRENE and experiments during the limiter startup campaign and the first island divertor campaign. In the startup configuration, the limiter positions and their poloidal extension resulted in a helical SOL consisting out of three types of separate helical flux bundles. These distinct magnetic flux tubes allowed a transport analysis based on simple SOL models and a heat flux characterization of power decay lengths $\lambda_{q\|}$ and peak heat flux values comparable to methods applied at tokamaks. Experiments with nitrogen (N_2) and neon (Ne) injection demonstrated an edge temperature reduction clearly correlated to edge radiation enhancement. The N_2 injection showed a fast response in which T_e recovered almost entirely after the injection was stopped. Ne featured higher recycling and terminated the discharge due to increased radiation causing radiative instabilities. These experimental results were analyzed with EMC3-EIRENE simulations. N_2 was modeled as a gas source neglecting recycling and, in contrast, Ne was sourced from the limiters to incorporate its properties as a fully recycling species.

Initial results from first island divertor experiments at W7-X show that core fueling and refuelling is probably prevented by island neutral screening effects as indicated by predictive modeling [1]. Here, radiative edge cooling is investigated for steering the neutral fueling by controlling the edge temperature. Initial explorations with EMC3-EIRENE indicate, that the location of the gas injection with respect to the island geometry matters. Experiments are commencing about changing the fueling and seeding locations from remote valves to specific SOL locations at the island O-point close to the mid-plane or directly in front of the divertor tiles near the main recycling and erosion zones. Results from fueling experiments aided by radiative edge cooling will be discussed. A focus of this work is the investigation of a correlation of local cooling features to the edge island geometry in the standard island divertor configurations. First, Ne was injected in discharges without using island control coils (Icc=0kA) and showed in response an enhancement of edge radition and correlated drop of divertor target heat fluxes. For Icc=2.5kA the absolute radition level reduces and a faster decay in response of the impurity radiation induced by Ne seeding was observed. These experiments were repeated with N_2. First results indicate a lower increase of edge radiation and fast drop of edge radiation in comparison to Ne seeded discharges. A systematic analysis of these experiments based on 3D numerical simulations with EMC3-EIRENE will be presented.

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Advances in radiated power control at DIII-D

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DIII-D's radiated power control system has been upgraded to convert detected power to locally emitted power in real time in the Plasma Control System (PCS) and to sum local power emissions into a sensor for total power radiated by the lower divertor $P_{rad,div,L}$ which agrees with standard offline analysis to within 20%. This is now possible due to more bolometer channels (now up to 12) being connected to the PCS for real time acquisition, and the new setup replaces previous use of a single proxy chord to represent power radiated by a region of the plasma. $P_{rad,div,L}$ has been increased to 70% above natural levels (with no N_2 seeding) and ramped up and down to demonstrate effective control. Spatial coverage is broad enough to enable radiated power control during the strike point sweeps which are commonly used to generate pseudo-2D divertor Thomson measurements in DIII-D divertor experiments. The new hardware and software should support similar sensors for P_{rad} from other plasma regions by using different channel groupings. Use of this control reveals challenges that may affect next step devices, which will require actively controlled extrinsic impurity seeding in order to manage heat loads.

Separate real time $P_{rad,div,L}$ measurements from two bolometer fans were averaged to provide the sensor used in a Proportional-Integral-Derivative (PID) system for controlling N_2 puffing during strike point sweeps. Radiated power control was tested using N_2 puffing through a port in the divertor and one near the crown of the plasma. Despite being located at the edge of the strike point sweep and raising concerns about the puff location changing from common flux to private flux region during the sweep, the divertor valve allowed good control.

When N_2 was instead puffed near the crown of the plasma, stable control was achieved for increases in $P_{rad,div,L}$ of 20% above natural levels in forward and reverse B_T, but requests for $\gtrsim$ 50% increases led to very unstable control with large oscillations in puffing and radiation. This may be because of longer delays between gas commands and changes in $P_{rad,div,L}$, even though the longer delays were accounted for in the PID tuning.

Finally, it was found that radiated power control can fail in detachment if enough of the divertor becomes too cold for nitrogen to radiate effectively. Then, additional puffing tends to lower T_e and reduce P_{rad} further. The control system responds by increasing its gas command, which can terminate in a radiative collapse if the PCS is not set up with appropriate checks and limits.

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Study of Lithium Vapor Flow In a Detached Divertor using DSMC code

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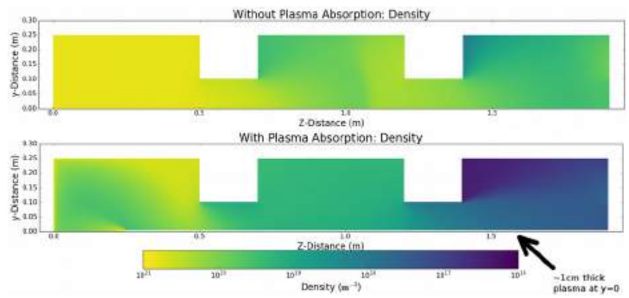
A detached divertor will be necessary to handle the plasma heat flux from a demonstration fusion power plant [1]. A key issue is control of the detachment location. The lithium vapor box divertor has poloidal baffles to form distinct chambers and so localizes the dense lithium vapor needed for detachment. The chambers are differentially pumped via condensation to reduce the lithium flux into the core plasma while still providing robust detachment [2]. We provide a simulation of the neutral lithium vapor flow in the divertor using the Stochastic PArallel Rarefied-gas Time-accurate Analyzer (SPARTA) Direct Simulation Monte Carlo (DSMC) code [3]. A Variable Hard Sphere (VHS) model with velocity-dependent effective diameter is employed for neutral-neutral collisions. The outer walls simulate evaporation and condensation by emitting particles and absorbing any impacting particles, while the baffle walls are diffusely reflecting. The code was modified with an axial weighting scheme to have an even distribution of simulation particles. The modified weighting achieved a factor of ten decrease in processing time and better statistics. In one study, vapor densities for a divertor without plasma computed with SPARTA are compared to densities predicted by solving enthalpy and mass flow equations in a choked flow model. The agreement between these models is acceptable. In another series of SPARTA studies, a toy model of a plasma modifies the lithium vapor transport. The plasma absorbs incident lithium and re-emits it in the center of the hottest box, with a temperature of 1 eV and directed speed corresponding to 1 eV for hydrogen. The density distributions for a Demo case with and without plasma are shown in the figure. The plasma absorption effect causes a significant reduction (greater than 90%) in the lithium exiting the vapor box. Lithium mass flow and density within each chamber are given for example configurations in Demo, FNSF, Magnum-PSI, and a vapor cylinder similarity experiment to be performed at PPPL. For the Demo case we find the lithium exit rate can be optimized to be less than 30 mg/s. This should be an acceptable in terms of effects on the plasma since NSTX has successfully operated with a 0.22 g/s injection with increased plasma performance [4].



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Energy balance during detached plasma operation in the divertor simulation experimental module of GAMMA 10/PDX

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Physics of plasma-gas interactions with high temperature ion and electron is one of the key issues for handling heat and/or particle load to the plasma facing components of magnetic fusion devices, ITER and a demo reactor. We have investigated energy and particle loss processes of high temperature plasma during detached plasma operations in the divertor simulation experimental module (D-module) located at the end-loss region of the tandem mirror plasma device GAMMA 10/PDX [1]. In this paper, we will discuss the energy balance between ion, electron and neutral based on recent results of systematic probe measurements using probe array on a V-shaped target plate in the D-module and several probes including ion sensitive probes (ISPs) [2] placed at upstream positions.

In the case of low neutral gas pressure range < 1 Pa with hydrogen gas puffing, for example, ion temperature (T_i) in the D-module measured by the ISP shows ~ 20 eV which is higher than electron temperature (T_e) ~10eV. These temperatures decrease with increasing hydrogen pressure. In high neutral pressure region > 5 Pa, T_i and T_e become ~ 10 eV and 2 eV, respectively. Though ion and electron have different energy loss channels each other, the tendencies of temperature drop are similar between them. Based on the experimental results, energy balance between ion, electron and neutral particles has been discussed. In our experimental conditions, ion energy loss channel is mainly charge exchange (CX) and ion-neutral (i-n) elastic collision since the influence of electron-ion energy relaxation is small due to low plasma density, while electron energy loss channel is ionization and excitation. Rate coefficients of CX and i-n collision are at least one order higher than ionization and excitation rates from the ground state hydrogen. It suggests that we should take into account the influence of exited hydrogen and H_2 molecule in order to explain the T_e decay. We will analyze energy balance equations for each specice considering these processes in detail. Energy balance along magnetic field will be also discussed based on parallel T_i measurements.

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Testing predictions of plasma detachment in TCV over a range in magnetic topologies through quantitative comparison to experiment

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The detachment process is crucial for ITER [1] and DEMO to strongly dissipate a large fraction of the Scrap-Off-Layer (SOL) power before it reaches the divertor targets. To achieve confidence in any predictions of future ITER operation or DEMO design we need a clearer understanding of both the physics of detachment and whether modification in magnetic topology will lead to significant improvements in divertor operation.

In this study we use the SOLPS-ITER code to model the results from several TCV discharges with different total flux expansion that is varied from a conventional divertor to the placement of the strike point R_{sp} at its maximum allowed in TCV [2]. The goal of the modelling is to characterize the detachment threshold and window [3], as well as understand the role of various processes (ionization, recombination, charge-exchange, etc) in detachment and in the loss of target ion current in TCV. The experimental density ramp is modelled by performing multiple steady-state simulations corresponding to different realistic gas puffing rates, wall pumping and physical and chemical sputtering of carbon from the wall. A carbon chemical sputtering rate of 3.5% has been chosen to match CIII experimental brightness measurements. Model results are also compared quantitatively to experimental measurements from Thomson Scattering, Langmuir probes, bolometry and filtered camera imaging.

We find that recombination is not a strong contributor to the rollover of the target ion current at detachment for all values of R_{sp}. In contrast, the divertor ionization source appears to play a significant role in magnitude and time, apparently dropping during the density ramps due to a drop of power available for ionization [4]. As the core plasma operating density is increased, the poloidal peak in the ionization shifts from the strike point to near the X-point while recombination remains peaked very close to the targets. The divertor ion source tracks the ion target current and both decrease at detachment for the entire range of densities observed experimentally, in qualitative agreement with experimental inferences using divertor spectroscopy [5]. The modeled target ion current drop occurs at lower upstream densities than in the experiment. In addition, the amount of target ion current loss is lower than observed in the experiment. While volume recombination strongly increases for upstream densities higher than can be obtained in the open divertor of TCV, it is still an order of magnitude lower than the divertor ion source.

The inclusion of drifts in the simulations will likely have a significant effect on the modeled loss of ion current and might be mandatory to obtain the asymmetry of detachment threshold between the inner and the outer divertors that is observed experimentally.

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Heat flux analysis of Type I ELM impact on a sloped, protruding surface in the JET bulk tungsten divertor

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Tungsten (W) melting due to transient power loads, for example those delivered by edge localised modes (ELMs), is a major concern for next step fusion devices. A series of experiments has been performed on JET to investigate the dynamics of Type I ELM-induced transient melting [1]. Following initial exposures in 2013 of a W-lamella with sharp leading edge in the bulk W outer divertor, new experiments have been performed in 2016-2017 on a protruding W-lamella with a 15° slope, allowing direct and spatially resolved (0.83mm/pixel) observation of the top surface using the IR thermography system viewing from the top of the poloidal cross-section. The new geometry results in a smaller temperature gradient compared to that on the top surface of the leading edge sample and correspondingly reduced sensitivity to the magnetic field incidence angle, permitting improved derivation of the incoming parallel heat flux in comparison to the sharp leading edge. One of the remaining issues of these most recent experiments is the understanding of the transient heat load as function of the geometry: protruding 15° slope and shaped standard lamella.

Analysis based on the forward approach using the full 3D heat load and thermal modelling of the lamellas has already been performed [2] assuming optical projection of the parallel heat flux on the lamella surface, together with a specific IR correction to account for effects related to spatial resolution. Using the same parallel heat flux, good agreement was obtained for three different geometries (sharp leading edge, protruding and standard lamellas) under L-mode conditions, validating the optical heat load projection assumption. The goal of the work described here is to extend the analysis based on the forward approach to the H-mode discharges, in particular during the Type-I ELMs used to achieve transient melting on the slope. Surface temperatures measured by the IR camera are compared with reconstructed synthetic data from 3D thermal modelling using heat loads derived from both the optical projection of the parallel heat flux and 2D particle-in-cell (PIC) simulations describing the influence of finite Larmor-radius effects on the deposited power flux [3]. Preliminary results show that the ELM power deposition behaves differently than the optical projection of the parallel heat flux, contrary to the L-mode observations, and may thus be due to the much larger gyro-orbits of the energetic ELM ions in comparison to L-mode or inter-ELM conditions.

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An assessment of nitrogen concentrations from spectroscopic measurements in the JET and ASDEX Upgrade divertor

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Reducing the plasma power exhaust impacting on plasma facing components during steady state operation is one of the major design issues in future tokamaks such as ITER. Impurity seeding, e.g. with N, is one method of achieving this and has been used for a long time in tokamak research [1]. In this work we exploit a recently developed spectroscopic N II line ratio technique for measuring the divertor nitrogen concentration [2] and extend the measurements over a wider range of discharges from ASDEX Upgrade (AUG) and JET.

The impurity concentration in the divertor is a necessary input for predictive scaling of divertor detachment [3,4]. Previously, divertor impurity concentrations have been estimated from indirect measurements based on either the decay rate of the impurity content after the injection phase or on the ratio of valve fluxes in constant conditions [3]. There is good availability of these indirect measurements for a wide range of discharges, however they typically come with large uncertainties. Direct measurements of the concentration using spectroscopy or a residual gas analyzer have been shown to provide better accuracy [5,6] but typically require dedicated diagnostic settings resulting in limited availability of the measurement.

The new spectroscopic technique for calculating the nitrogen concentration was purposefully designed such that the measurements were possible with the spectrometer settings used routinely to measure Stark broadening. Since the installation of the tungsten wall on AUG and JET, radiative cooling has become mandatory for high heating power scenarios and these discharges offer a wide range of parameter scans, including N_2 seeding and D_2 fuelling rates ranging from 10^{21}-10^{22} electrons/s and electron temperatures in front of the outer divertor target ranging from 5 – 50 eV. A nitrogen divertor concentration in the range of 5-25% was previously shown [2] for one partially detached H-mode discharge on AUG. In this analysis, we use a database of discharges from AUG and JET to expand these measurements and show that the divertor electron temperature decreases below 10 eV only with a strong increase of the nitrogen divertor concentration. A thorough assessment of the uncertainties will be presented along with comparisons of the flux ratio measurements of the nitrogen divertor concentration.

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Modeling of power exhaust in DEMO alternative divertor configurations with SOLEDGE2D-EIRENE

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Modelling of power exhaust of the DEMO device is one of the activities required to evaluate the possibility to build a reliable fusion reactor. Indeed recognizing the difficulty of the standard Single Null (SN) divertor configuration to provide a solution compatible with present and foreseen solid divertor target technological solutions, many alternative divertor magnetic configurations have been proposed for DEMO. The aim of any alternative magnetic configurations is the reduction of the heat load on the divertor targets lowering the detachment thresholds and increasing the radiation power in the SOL. Among all proposed alternative magnetic configurations in this study the following ones have been compared to the SND: the flux flaring towards the target of a X divertor (XD), the increasing of the outer target radius of a Super-X divertor (SXD), the connection length increasing and strike points doubling of a Snowflake divertor in plus and minus variants (SFD-plus, SFD-minus) and finally the strike points doubling of the double-null divertor (DND). By using the 2D SOLEDGE2D-EIRENE [1] code here we present the first comparative study of these configurations performed with a fluid plasma code coupled to a neutral kinetic code. All configurations with nearly equal global parameters (major radius, plasma section and plasma current) have been modeled with a full W wall with unitary recycling, reasonable pumps, no impurity seeding and no drifts in two conditions: at various density but at the level of power crossing the separatrix P_{SOL}=150MW, required in DEMO to operate in H-mode, at a fixed middle density $n_{sep}=2.5\cdot10^{19}$ m^{-3} but different P_{SOL}=50÷150 MW to simulate the effect of impurities. As transport parameters the couple $\chi_\perp = 0.18$ and $D_\perp = 0.42$ m^2/s which provides an upstream e-folding decay length of the power flowing channel on the outboard equator of $\lambda_{q,u}\approx3$ mm for the SND in attached condition.
Given the uncertainties in the transport parameters knowledge and on some physical phenomena the aim of the modeling was not to provide the "exact" evaluations of main exhaust parameters but to provide the best comparison between various configurations with a "state of the art" edge code. In these contest the simulations have shown different advantages of each alternative configuration against the others in terms of power load on targets and volume losses but in general alternative configurations performs equal or better than the SND one.

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Effects of the Gas Puffing Neutral on the Plasma Parameters in the End-Cell of GAMMA 10/PDX by using the Multi-Fluid Code "LINDA"

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GAMMA 10/PDX is the world's largest linear plasma confinement device [1]. The divertor simulation research has been effectively progressed in the end-cell of GAMMA 10/PDX by using the end-loss plasmas ($n_{\mathrm{e}} \sim 2 \times 10^{18}\mathrm{m}^{-3}, T_{i\parallel} \sim 200\ eV, T_e \sim 30\ eV$) in order to predict and to explore the physical mechanism of plasma detachment [1-3]. It has been found that radiator gases injection into the divertor simulation module of GAMMA 10/PDX is effective on the radiation cooling and generating detached plasma [1-3].

A numerical simulation study has been started in the end-cell of GAMMA 10/PDX for understanding the detailed physical mechanism of energy loss processes (such as radiation cooling, CX loss, Ionization loss, etc.) [4-5]. The LINDA (Linear Divertor Analysis with fluid model) code is a 2D multi-fluid code which consists of continuity equation, momentum equation, energy equation for ion and electron [4]. The mesh structure has been designed according to the magnetic field configuration of GAMMA 10/PDX. A tungsten target plate has been designed at the end of the mesh. The divertor boundary conditions are applied on the target plate. The neutral models for both the impurity and hydrogen are given by solving 1D fluid equations which are solved iteratively together with the fluid equations. The atomic processes of hydrogen and impurity gases (Xe, Ar, Kr, Ne and N) have been included in the present code. In the study, comparison among various gas species has been performed numerically to analyze the detailed energy loss processes during impurity injection. The electron and the ion temperature reduce near the target plate due to the impurity gas injection. Especially, a remarkable reduction in the electron and ion temperature has been observed for Xe and H injection, respectively. On the other hand, Ne injection is the least effective on the electron cooling. Xe injection increases significantly the radiation power loss. For the strongest Xe injection case, T_{e} on the target plate reduces to about 3 eV. The charge-exchange loss between proton and hydrogen neutral has been detected as major energy loss channels for ion. On the other hand, the radiative power loss of the impurity gases has been detected as major energy loss channels for electron.

In the paper, we are also aiming to report effects of the Molecular Activated Recombination (MAR) on the plasma detachment in GAMMA 10/PDX.

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Impact of Drifts on Divertor Power Exhaust in DIII-D

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Radiative divertor experiments and 2D fluid simulations show strong impact of cross-field drifts on the low field side (LFS) divertor target heat flux and volumetric radiation profiles through the transition to detachment in DIII-D high confinement mode (H-mode) plasmas. This code model validation by direct comparison to data increases physics understanding and confidence in predictions of detachment in ITER and future reactor scenarios.

Operating with the **B×∇**B-drift towards the X-point (fwd. B_T) with deuterium and nitrogen injection, the LFS divertor radiation front is observed to rapidly shift from the target to near to the X-point at the onset of detachment. In contrast, operating with the **B×∇**B-drift away from the X-point (rev. B_T), the radiation front is observed to remain closer to the target plate and to be radially shifted towards the far SOL. Consistently, flat target heat flux profiles, q_{LFS}, with the peak reduced by ~5x relative to the attached conditions are measured in fwd. B_T. In rev. B_T the q_{LFS} profile remains peaked and the peak is reduced only by ~3x.

UEDGE simulations optimized to reproduce experimental measurements from multiple diagnostics, indicate that these observations can be largely explained by the poloidal and radial **E×B**-drifts in the divertor. In fwd. B_T, the poloidal **E×B**-drift in the private flux region (PFR) drives the main ions from the LFS divertor to the high field side (HFS) divertor, thereby reducing n_e and increasing T_e in the LFS divertor. At the detachment onset, the reduction of the radial T_e and potential gradients diminish this **E×B**-drift sink in the LFS divertor, such that the LFS divertor can evolve into a high n_e, low T_e, strongly detached state with the same upstream separatrix n_e. In rev. B_T, on the other hand, the poloidal **E×B**-drift in the PFR drives the main ions from the HFS divertor to the LFS divertor, thereby increasing the LFS divertor n_e and facilitating radiative divertor conditions. However, the radial **E×B**-drift in the LFS common SOL in rev. B_T competes with this mechanism, driving ions from the strike line region towards the far SOL. As a result, n_e at the strike-line becomes depleted, limiting the degree of detachment next to the separatrix, while leading to a formation of a high n_e, radiation front in the far SOL. A similar comprehensive physics picture is also emerging from SOLPS-ITER simulations including drift effects. These observations are consistent with 2D divertor Thomson scattering volumetric n_e and T_e data, 2D radiation distributions from bolometer and visible emission reconstructions, and target Langmuir probe measurements. New divertor EUV emission data, capable of directly identifying the contributions of the primary species contributing to the total radiated power, will also be presented. Collectively, these sets of multiple code simulations validated by data from multiple diagnostics significantly increases physics understanding of detachment and confidence in predictions of heat flux and radiation profiles for future devices.

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Simulating Magnum-PSI target gas puff experiments with the SolEdge2D-Eirene transport code

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The Magnum-PSI linear plasma device [1] uses a cascaded arc plasma source to generate a high density (~10^{20} m^{-3}), low temperature (~1 eV) plasma beam, similar to the one expected in the ITER divertor. The plasma source operates with a steady gas flow rate around ~5 standard litres per minute (slm). The ionisation fraction of the source is <10% and the residual gas flows into the vessel. A differential pumping system exhausts most of the gas from two successive chambers. In a third chamber, the plasma beam impinges upon a solid target. The neutral pressure in this chamber is determined primarily by recycled neutrals, rather than residual gas from the source, making the experimental conditions similar to those in a tokamak divertor. The possibility to introduce an external puff of cold neutral gas into the target chamber provides control over the neutral background pressure P_n. In this work we employ the Soledge2D-Eirene code [2] to simulate the three chambers of the Magnum-PSI device including the differential pumping. The cascaded arc plasma source is not modelled self-consistently, the plasma is generated using external volumetric sources of energy and plasma particles in order to match Thomson scattering (TS) profiles measured close to the source nozzle. Initial results of target gas puff simulations show that reduction of T_e with P_n can be reproduced by the code. The rollover of n_e observed experimentally by TS with increasing P_n is reproduced, but not at the axial position of the TS measurement (2 cm in front of target). Instead, in the simulations it occurs approximately 10 cm further upstream. In the simulations, the rollover is caused by the competition of deceleration due to ion-molecule friction, which tends to increase the density, and radial particle losses due to perpendicular transport, which act in the opposite way.

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Kinetic simulation of heat pulse propagation through the tokamak scrape-off layer*

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Tokamak fusion reactors are currently envisaged to rely on high performance H-mode operation which suppresses edge turbulence, but comes at the cost of repeated edge localized mode instabilities (ELMs) that deliver heat fluxes that can be large enough to erode target plate and plasma-facing component surfaces. For an H-mode tokamak, the pedestal plasma delivered to the scrape-off layer is typically hot enough to reside in the collisionless regime, where the mean free path is longer than the connection length. In this case, parallel transport along field lines becomes nonlocal and cannot be treated quantitatively with a fluid model [1]. The nonlinearities involved in the plasma dynamics, the materials dynamics, and their mutual interactions imply that kinetic effects can be important for making quantitative predictions that can be validated against experimental results.

In this work, the transient behavior of a plasma heat pulse that travels along a flux tube is studied using the 4D drift-kinetic COGENT code [2] in order to predict the heat flux impinging on the target plate of tokamak divertor. Simulations use both kinetic electrons and kinetic ions in order to determine the detailed dynamics and detailed particle distribution functions. These simulations report on the first use of sheath boundary conditions [3] within COGENT, required to keep the boundary quasi-neutral and to remove short temporal and spatial scales, respectively on the order of the inverse plasma frequency and Debye length, from the solution. Results are compared to heat pulse propagation test problems [1,4-6] that have been used to compare physics models and numerical algorithms. Comparison to fluid models will be made in order to determine under which conditions kinetic effects become important.

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Change of confinement mode during detachment transition with RMP application in LHD

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Compatibility of detached divertor plasma with good core plasma performance is important for future reactor operations. During detachment transition with RMP application in LHD, recovery of energy confinement is observed up to the level of ISS04 (international stellarator scaling in 2004), while in the attached phase the energy confinement is usually degraded far below the scaling due to the edge magnetic island formed by the RMP and to the enhanced impurity radiation.

At the detachment transition with RMP, the impurity radiation increases by a factor of 2, and divertor power load estimated by Langmuir probe decreases significantly. At the same time, the plasma stored energy increases discontinuously. Then the confinement enhancement factor, which is τ_E (energy confinement time) normalized to the ISS04, recovers from 0.6 to 1 during detachment transition, and it is kept in the whole detached phase of several seconds until the end of discharge. On the other hand, in the attached phase, the confinement factor decreases down to 0.6 with increasing density and impurity radiation. The radial profiles of electron pressure, i.e. product of electron temperature and density obtained by Thomson scattering, becomes peaked at the center of plasma after the detachment transition. The peaking is due to the density increases in the detached phase while the temperature slightly decreases. Without the RMP application, such confinement mode transition does not occur, and τ_E monotonically decreases with increasing density.

Assessment of stored energy profile and the energy transport analysis show that the confinement improvement occurs at the outer region of plasma (ρ ~1), and then slowly propagates towards inner radii. The transition, i.e. the propagation lasts for about 1 second. The estimated cross-field energy transport coefficient, χ_e, which varies in a range of 10 ~ 2 m^2/s depending on the radii in the attached phase, decreases down to ~ 1 m^2/s for the entire radius in the detached phase.

Magnetic fluctuation analysis shows gradual decrease of high frequency component (several tens kHz) with increasing density in the attached phase, and finally those components vanish after completion of the confinement mode transition in the detached phase. Instead, broad lower frequency component less than 10kHz, peakd at 5kHz, is enhanced in the detached phase. Fluctuations of ion saturation current at the divertor plates also shows enhancement of low frequency component around 100Hz after the detachment transition. The profile of ion saturation current exhibits broadening toward the private region.

Observations of Strong Reduction of the Power Load onto the Island Divertor Targets of Wendelstein 7-X in the OP1.2a Experimental Campaign: Complete Stable detachment?

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The island divertor concept, which has for the first time been successfully demonstrated in Wendelstein 7-AS (W-7AS), was also realized in the Wendelstein-7X (W-7X) device. In W7-AS stable detachment was found to be always partial, with an about 20x reduced divertor power load over most of the divertor and a small only partially detached region with a power load reduction of about a factor 2-3. A similar partial-detachment feature was reproduced by EMC3-Eirene modeling and attributed to the in/out asymmetry in surface compression of the W7-AS configurations in which impurity radiation prefers the inboard side. If this is true, the pololoidally more symmetric flux surface geometry of W7-X should allow for a more symmetric impurity radiation distribution at the edge and thereby a more homogeneous reduction of heat fluxes on targets under detached conditions. This has been indeed shown by EMC3-Eirene simulations [e.g. Feng et al NF2016].
In the presently running W7-X experimental campaign OP1.2a we are trying to find operational routes to stable detachment. Initially, reaching core plasma densities of more than a few times 10^{19} m^{-3} proved rather difficult in hydrogen with just gas fuelling. This, however, was fully in line with the EMC3/EIRENE modeling prediction of efficient neutral screening by the island structures enclosing the confined plasma region. Since this situation was expected from modeling, a pellet injector previously used on ASDEX-U had been refurbished and installed on W7-X for the OP1.2 experimental campaign. In the recent development of hydrogen pellet fuelling discharges, we observed at the end of the pellet fuelling phase in a 3MW ECRH heated hydrogen discharge ($\langle n_e \rangle_{max} \approx 5\text{x}10^{19}$ m^{-3}), a complete drop of the peak heat flux densities across the entire divertor targets to values below 0.2 – 0.4MW/m^2 on all 10 discrete island divertor units made of graphite and monitored by IR cameras. At the same time, two bolometers viewing an entire triangular plane measured a radiation strength that is comparable with the total heating power with radiation peaked at the very edge. Edge radiation in H plasmas is believed to come primarily from H, C, O composition. This plasma state was stably maintained over several energy confinement times, until the pre-programmed end of the discharge. In comparison to the pre-pellet phase, no remarkable change in the linear-average density and the energy content could be identified. The presently still ongoing investigations focus on understanding and optimizing this plasma state. In particular, the divertor spectroscopic systems and Langmuir Probes will be used to get further insight into why the high density pellet fuelled phase is required to gain access to this operation regime. Interesting results including possible drift effects will be re-examined with reversed field, which will be performed in the last week of the experimental campaign.

Divertor plasma detachment: past and future

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Large heat flux to and fast erosion of plasma facing components as well as tritium retention are critical issues of plasma material interactions (PMI) for future magnetic fusion reactors. Detached divertor regime, which is characterized by large reduction of both heat and plasma particle fluxes to divertor targets, suggests a plausible solution at least for some of the PMI issues. Even though divertor detachment was studied extensively for almost three decades, still there are some controversies with the roles of different processes including impurity radiation loss, cross-field plasma transport, plasma-neutral interactions, and plasma recombination in the reduction of plasma flux to the target. In addition, detached divertor regimes can be subject to different instabilities, resulting in significant fluctuation of plasma parameters, and the bifurcation-like phenomena causing the jumps of detachment front and MARFE formation, which, potentially, can cause disruption of the discharge. The physics of such instabilities and bifurcations is not very clear yet, although different models were suggested in the literature. Finally, it is plausible that some synergistic effects of plasma and wall physics play an important role in detachment related issues. We note that all of these can be complicated by ELM effects.

In the talk we outline the physical picture of divertor detachment and discuss the role of each physical process resulting in the redaction of plasma flux to the target. We also emphasize how these roles could change in advanced divertors in comparison with current tokamak divertors. We review existing models predicting instabilities and bifurcation phenomena in detached and semi-detached regimes, compare them with existing experimental data and give the projection for advanced divertor designs. Finally, we discuss the most crucial gaps in our understanding of edge plasma physics and how they may affect our assessment of divertor performance in future tokamak reactors.

The effect of feedback-controlled divertor nitrogen seeding on the boundary plasma and power exhaust channel width in Alcator C-Mod

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The scrape-off layer (SOL) power channel width, λ_q, is projected to be ~0.5 mm in power reactors, based on multi-machine measurements of divertor target heat fluxes in H-mode [1,2]. However, these measurements were performed at low levels of divertor recycling and radiant heat dissipation, allowing the heat flux 'footprint' at the divertor target to be interpreted as a measure of the 'upstream' λ_q. Important questions therefore arise: Does upstream λ_q change with the level of divertor dissipation and/or the degree to which the divertor target plate is electrically connected to the upstream SOL? Theory indicates that electrical connection to the divertor target can play a role in SOL turbulence, e.g., parallel currents can reduce blob polarization and transport [3]; sheath potentials can impose $E\times B$ shear, regulating turbulence [4]. We report results from Alcator C-Mod in which feedback controlled nitrogen seeding in the divertor was used to systematically vary divertor dissipation in a series of otherwise identical L-mode plasmas at three plasma currents: 0.55, 0.8, 1.1 MA, with 5.3 Tesla toroidal field. Core plasma density was chosen so as to place the divertor in a sheath-limited heat flux regime prior to nitrogen injection. Outer midplane profiles were recorded with a scanning Mirror Langmuir Probe; divertor plasma conditions were monitored with 'rail' Langmuir probe and surface thermocouple arrays. Key observations are:

(1) In all cases, N_2 seeding resulted in a factor of ~10 reduction in divertor target plate heat fluxes while maintaining core plasma conditions relatively unchanged;
(2) In all cases, divertor target conditions near the strike point changed from sheath-limited to high-recycling, attaining partial detachment;
(3) In all cases, net parallel current densities to the target plate – driven by Pfirsch-Schülter and thermoelectric currents – were reduced by over an order of magnitude;
(4) Despite these dramatic changes in divertor conditions, upstream λ_q remained largely unchanged, within statistical uncertainties.

We interpret observation (3) to indicate that an order-of-magnitude variation in the 'electrical connection' between target plate and upstream SOL was obtained. The results have important implications for our understanding of boundary plasma transport and for advanced divertors:

- Divertor dissipation does not reduce peak heat flux densities entering from 'upstream'.
- Divertor plasma conditions, including sheath boundary conditions, do not play a defining role in the physics of cross field transport in the near SOL region.
- Upstream λ_q will be largely unaffected by advanced divertor details – leg length, flux expansion, x-points.
- Divertor fan width can be designed based on empirical λ_q scaling.

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Understanding of the outer divertor heat flux splitting during RMP-ELM suppressed regimes in KSTAR

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One of the most important concerns for burning-plasma devices such as ITER is the issue of divertor target power loading, both in steady state and during ELMs. The latter will be particularly challenging with regard to tungsten target lifetime and must preferably completely suppressed or at the least strongly mitigated. It has been demonstrated that resonant magnetic perturbations (RMP) can be an effective method to achieve this on several tokamaks. Along with the role of RMPs in suppressing ELMs, the broadening of divertor target power loading in the presence of the 3D fields may be an additional benefit of the application of RMPs, although the situation is unclear experimentally regarding the distribution of heat flux density in the more complex loading pattern compared with the non-perturbed situation.

In KSTAR, striation of the outer target heat load has been clearly observed during RMP-ELM suppression regimes using an infra-red (IR) thermography system. The data has been obtained mainly in attached divertor heat load conditions in KSTAR discharges with I_p = 500 ~ 600 kA, q_{95} = 4.5 ~ 5.5, B_T = 1.8 ~ 2.0 T and P_{NBI} = 2.8 ~ 3.5 MW and a full carbon wall. For the ELM suppression, n=1 and 2 (n is the toroidal mode number) RMPs are applied. Experiments in which the RMPs phase is rotated during a single discharge have enabled us to argue that the non-axisymmetric outer divertor target heat flux pattern is mainly determined by the configuration of RMPs such as toroidal mode number and phase. Several issues remain unresolved, however, such as the plasma response to the external magnetic perturbations which results in the fine detail of the heat flux striation pattern, in particular the spacing between peaks and sharpness of each peak. KSTAR experiments have also demonstrated that the plasma response can be very different according to plasma conditions such as pedestal collisionality.

In this paper, several plasma response models including vacuum and ideal plasma response models will be tested by comparing field line tracing calculations to the outer divertor heat flux profile measured by divertor IRTV on KSTAR. For a more accurate description of the plasma pedestal region, the kinetic-EFIT result coupled with the measured plasma density and temperature profiles will be taken into account. In addition, the 3-D EMC3-EIRENE plasma boundary code will be applied to investigate the possible role of edge plasma transport in the formation of the striation pattern.

SOLPS modeling of radiative divertor plasma with impurity seeding during ELMy phase in EAST

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Impurity seeding is a conventional method to increase the extrinsic radiating fraction of tokamak with metallic PFMs in order to reduce the power loading at the divertor target. Numerical simulations of radiative divertor plasma during inter-ELM phase have been wildly conducted [1], and the power exhaust physics of impurity with steady state plasma is gradually revealed. However, there also remains a doubt whether the controlling physics are still suitable for the H-mode plasma exhibited ELM, because repetitive bursts of particles and heat arising during ELM phase may significantly impact the power exhaust efficiency of impurity particles. In this work, SOLPS5.0 code package has been applied to simulate the ELM in EAST, and the impact of ELM on the distribution of impurity density and power radiation is investigated. The fraction of impurity ions with different charge states, and the power exhaust ability during ELM phase are also compared with that during inter-ELM phase. According to [2], ELM is simulated by enhancing transport coefficients (radial particle diffusivity or the radial convective velocity) for a short time, and the content of extrinsic impurity ions across the separatrix is compared between these two different simulation methods.

As known that impurity puffing mode plays an important role in the performance of radiative divertor, so in this work, the time dependent neutral source of SOLPS is applied in order to simulate several puffing modes: constant puffing speed, pulse puffing, and “feedback control”. Radiative feedback control has been applied in AUG [3] and JT-60U [4] as a conventional method, which is an effective technique to sustain the high radiation fraction. Here using EAST magnetic configuration, we manually increase the extrinsic impurity puff rate at the approaching of ELM burst in order to dissipate much more heating power, then the “feedback control” is roughly simulated.

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Spectroscopic investigation of N_2 and Ne seeded induced detachment in JET ITER-like wall

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The degree and operating space for nitrogen and neon impurity seeded induced detachment in JET with ITER-like wall (JET-ILW) L-mode discharges in the high-recycling divertor regime is shown to be regulated mainly by the combined effect of i) the local radiative cooling efficiency in the outer divertor; and ii) the degree of ionization front incursion towards the X-point with increased impurity seeding. Quantitative spectroscopic analyses show that at the onset of detachment the net particle balance at the outer target is dominated by a marked decrease in the ionization source between the X-point and the target with increased nitrogen and neon seeding, and only a marginal increase in the volume recombination rate. Local radiated power dissipation (nitrogen only) and upstream P_{SOL} reduction (both nitrogen and neon) with increased seeding leads to a reduction in the outer target T_e and an inward shift of the ionization front away from an enhanced ionization region at the strike point caused by observed high Lyman series opacity. For the same power flow into the divertor, an additional 30% reduction in the outer target ion current was observed for nitrogen seeding, and is attributed to local radiative dissipation not observed in the neon seeding cases.

Spectroscopic measurements of the nitrogen radiation pattern and plasma parameter estimates from deuterium radiation were used to constrain EDGE2D-EIRENE simulations. The large drop in ionization source with increased impurity seeding is only recovered if the simulations include an *ad-hoc* opacity treatment informed through measurement of the Ly_β/D_α ratio. A novel N II spectral line ratio technique was also used to constrain the prescribed radial impurity transport in the SOL. Inferred T_e and n_e estimates from N II 4f→3d and 3p→3s transitions are consistent with low transport close to ionization balance (residence time $\tau\sim\infty$), while the EDGE2D-EIRENE simulations with prescribed cross field transport ($D_\perp$=1.0 m^2s^{-1}) yield a shorter N^{1+} residence time ($\tau\sim$0.1 ms). EDGE2D-EIRENE simulations show that a decrease in the prescribed radial nitrogen transport coefficient from $D_\perp$=1.0 to 0.1 m^2s^{-1} results in a redistribution of the radiated power in the divertor towards lower ionization stages (from $N^{3+,4+}$ to $N^{1+,2+}$), which leads to a factor of two increase in the nitrogen concentration needed to reach the same total radiated power. The impact of including the full photon transport model in EIRENE, as well as the effects of convective transport on the impurity radiation distribution in the divertor and X-point regions will be presented.

* See the author list of "Overview of the JET results in support to ITER" by X. Litaudon *et al*. 2017 *Nucl. Fusion* 57 102001.

Snowflake- and X-Divertor configurations in TCV and the future upper divertor of ASDEX Upgrade

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Power exhaust is regarded as one of the greatest challenges for the next generation of fusion experiments and in particular for a future fusion reactor. For this reason alternative divertor concepts are currently discussed in the community to mitigate the peak heat flux densities occurring near the divertor strike points. TCV has pioneered the experimental realization of a Snowflake (SF) [1] configuration in 2009 [2] and is playing a leading role investigating alternative configurations in Europe, while NSTX and DIII-D investigate these configurations in the US. In order to study such configurations in a tokamak with large heating power compared to its size, the installation of a pair of in-vessel coils with currents $|I_{IV}|<= 50$ kAt in the upper divertor of ASDEX Upgrade (AUG) [3] was recently decided. While the standard single- (SN) and double null (DN) configurations can still be performed, a series of new configurations ranging from an X- divertor (XD), to a low- and finally a high field side snowflake minus (LFS SF- and HFS SF-) will be accessible with these coils. The design of these coils was supported by simulations with EMC3-EIRENE, which can rather easily handle topologies with two X-points and which identified a series effects that might mitigate the heat flux to the divertor targets [4]. One of the most important questions about the suitability of alternative divertor configurations for a reactor is whether they facilitate the access to the detached divertor regime, while simultaneously maintaining a high confinement. Recent SOLPS simulations required volumetric recombination as well as drifts to describe the details of the detached SOL in AUG [5], processes that are not taken into account in EMC3-EIRENE. SOLPS was not applied to a SF topology, so far, due to the technical requirement of a block-shaped computational grid. This requirement was now met by sub-dividing the physical grid for the SF into several regions and adjusting their resolution adequately. A previous study with rather low input power P_{in}=1.6 MW [6] is now extended to more relevant P_{in}=10 MW at radiative fractions >= 80%, where a substantial degree of detachment is achieved in the SOL. While the numerical details of the simulation as well as further extrapolations are presented in [7], here we focus on the physical effects expected to occur in the new configurations as well as the technical realization of the divertor and its diagnostics.

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High-confinement steady-state operation with quasi-snowflake divertor configuration and active radiation feedback control in EAST

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Advanced divertor configuration and active feedback control of radiated power are two promising solutions to the power exhaust issue in tokamaks, especially for steady-state operations of future fusion reactors. Effective heat load reduction in long-pulse operations with ITER-like tungsten divertor was successfully achieved on EAST in quasi-snowflake (QSF) configuration and active radiation feedback control phase, respectively.

Fully non-inductive steady-state QSF plasmas have been obtained with all the main plasma parameters being very stable, with the longest steady-state shots demonstrated up to 21 s without any sign of instabilities [1], only limited by the technical imposed scenario parameters. A mix of different auxiliary heating power, i.e., ECRH, LHW and ICRH, has been injected up to total 6.2 MW in QSF configuration with top tungsten divertor, without coupling problems. All the discharges were in H-mode with $H_{98} \geq 1$, neither edge D_α nor core impurity accumulation was observed. EAST QSF diverted plasmas have reached an increase of the connection length by ~30% and the flux expansion by a factor of ~3 in the outer strike point region. This QSF configuration is suitable for the superconducting tokamaks with integrated poloidal field system like EAST, which wasn't originally designed for the snowflake divertor configuration. A heat flux reduction, up to a factor of ~ 2.5, was achieved with bottom QSF on the graphite divertor. In all the steady-state QSF discharges, the ELM activity was quiescent, indicating a possible non-linear interaction between the downstream magnetic topology and the upstream kinetic gradients.

By utilizing impurity seeding, assisted with normal divertor gas puff as feedforward and supersonic molecular beam injection (SMBI) as feedback control, the radiation can be effectively controlled with slight degrade of core plasma performance, i.e., the loss of plasma stored energy within 7 - 11%. More importantly, both divertor heat and particle particles were significantly reduced during the active feedback control phase. The radiated power was successfully maintained with the maximum relative control error decreasing from 24% to 13% when the controlled radiated power raised from 0.6 MW to 1.2MW in long-pulse H-mode experiments. In addition, active feedback control of radiation in L-mode plasmas was also performed. In the next step, this radiation feedback control, showing promising potential for heat flux control with good core plasma performance, is planned to be integrated in longer pulse with QSF divertor configuration on EAST.

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Power accounting using divertor extreme ultraviolet emission in the transition to detachment in DIII-D

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Extreme ultraviolet (EUV) emissions in conjunction with 2-D imaging of divertor plasma parameters are measured in order to quantitatively account for radiated power from atomic and ionized plasma constituents and compare to bolometry in the transition to detachment in DIII-D. New measurements in the 150-1700Å vacuum ultraviolet region are made using a vertically-viewing SPRED (Survey, Poor Resolution, Extended Domain) spectrometer[1], Divertor SPRED (DivSPRED) which complements the horizontally-viewing dual-range core SPRED. DivSPRED shows that EUV emissions in the divertor are dominated by C IV, 1549Å and D Lyman-α, 1215Å, along with significant emissions for other carbon charge states (CII and CIII), in agreement with previous observations[2]. Observation of the D Lyman series, and the Lyman-Werner D_2 molecular band are expected as divertor density is increased. EUV C emissions are found to plateau while Ly-α increases by >5 fold during the detachment 'cliff' transition at the outer strike point ($T_{e,OSP}$ from ~10-15 eV to 2-3 eV). Detached divertor operation at the targets is critical for high power, long pulse operation in ITER and future fusion reactors. This condition has proven challenging to model with state-of-the-art boundary simulation codes where multiple fluid codes modelling large devices have shown an increasing deficit of predicted radiation compared to experimental measurements with increasing Greenwald fraction; a radiation shortfall.

Measurement of electron parameter profiles in the divertor is made possible with a combination of coincident divertor Thomson scattering (DTS), tangential filtered 2-D and infrared imaging, visible (VIS) and near-infrared (NIR) spectroscopy, bolometry, and fixed and reciprocating Langmuir probes.1-D DTS, DivSPRED, VIS and NIR observations are extended to 2-D by sweeping of the X-point and targets past the observed field of view at a common major radius, including both outer and inner strike points. Fast-framing capability of the DivSPRED allow ELM-resolved measurements of D and C species emission and radiated power. Modeling with UEDGE and SOLPS-ITER with synthetic diagnostic capabilities for EUV emissions are compared with experimental data to help resolve the shortfall in code-simulated radiation.

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Dynamics of detachment movement in MAST-Upgrade Super-X divertor

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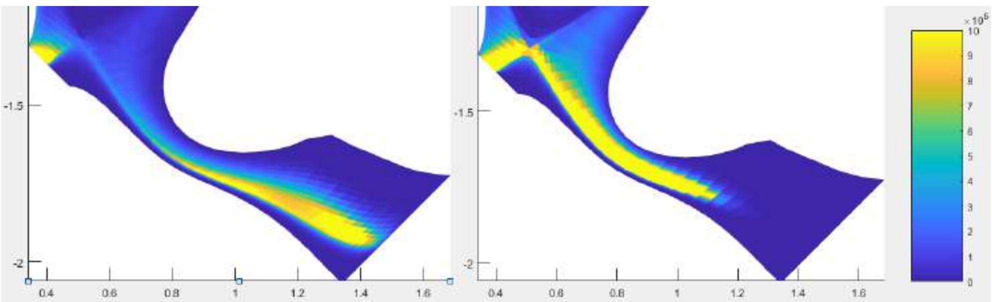
For future high power devices, detached divertor operation will likely be required to avoid divertor damage by excessive heat flux and to limit target plate erosion to acceptable values [1]. In this work, the SOLPS-ITER code has been used to study gas puff (non-seeded) and impurity (nitrogen-seeded) driven detachment evolution in MAST-U Super-X geometry. The main ion puff rate and impurity seeding rate scans were performed for a fixed P_{SOL}=2.5MW to obtain well equilibrated detached solutions. We also compared the effect of nitrogen and D_2-injection locations – at the entrance to the divertor and at the divertor target.

In the non-seeded case, ionisation and radiation peaks detach from the target at the same puff rate at which the target flux rolls over for either D_2-injection location. The steady-state ionisation and radiation peak locations range from 4% and 2% respectively to 58% and 54% of the poloidal distance from the target over the injection scan. The ionisation and the radiation fronts have a narrow peak when detachment begins, which broadens significantly as they move upstream. The ratio of the change in location of the steady-state peaks with the increase in D_2-injection rate decreases as the front moves along the leg towards the x-point; the peak location is less 'sensitive' [2] to the D_2-injection actuator. The recombination and density profile peaks never leave the target but the recombination peak magnitude increases with increasing D_2-injection. Higher injection rate cases are being modelled.

Similar trends with regards to the movement of the ionisation and radiation peaks are observed in the nitrogen seeded cases. The ionisation/radiation front movement slows down with increasing seeding rate and the fronts are located ~ 50% of poloidal distance from the target to the x-point at the highest seeding rate in this scan. In contrast to the non-seeded case, there is no sign of a recombination front forming at the target even for the highest seeding rate in this scan. Higher seeding rate cases are currently being explored.

We will also present a comparison of the evolution of the 2D energy and momentum loss profiles for the seeded and non-seeded cases to understand what drives the roll-over of the target flux in each case. Finally, the sensitivity of the thermal front location is compared to the predictions of the thermal front model [2].



2D impurity radiation (Wm^{-3}) profiles of a partially detached (left) and strongly detached case (right).

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Spatial distribution of highly charged Ne ion in detached divertor plasma of JT-60U

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Impurity seeding is one of promising techniques to mitigate significant heat load onto the divertor target. One dominant energy loss channel is line-radiation from highly-charged impurity ions, which spatial distribution is, however, difficult to obtain because of too short wavelength for visible spectroscopy. Over this difficulty, we made use of a Ne^{7+} visible line from a highly-excited level to study transport of a highly-charged Ne ion for the first time, by spatial-resolved measurement together with intrinsic C impurity transport in JT-60U divertor plasma.

In the Ne seeded detached divertor plasma of an H-mode discharge, it was found that a peak of Ne VIII n=8-9 line (Ne^{7+}) was located at the high-field-side (HFS) of the X-point. In contrast, C IV 3s-3p line emission had no peak and was widespread over the divertor region with weak intensity. Given that Ne VIII was the biggest radiator as already studied (but without spatial-resolved information) [1], these observations indicate the divertor radiation came predominantly from the HFS of the X-point due to Ne ion emissions. However, this Ne VIII emission peak was not maintained for long even with a feed-back control of a Ne puffing rate. The Ne VIII emission peak diminished and moved toward the inner strike point. In contrast, C IV 3s-3p emission peak was formed at the HFS of the X-point together with C IV n=6-7 emission peak. Once the Ne VIII peak disappeared, it was not possible to recover it with the feed-back control of the Ne puffing rate. In other words, the C IV peak was very stable against the Ne puffing rate change.

A similarity between the Ne VIII and the C IV emission peaks was found; at both of the peaks, recombination of higher adjacent charge state ions, that is, Ne^{8+} and C^{4+}, was indicated by, respectively, Ne VIII n=8-9 line and C IV n=6-7 line. This situation is similar to an X-point MARFE of a low power heated L-mode discharge [2]. Further in order to obtain a strong radiation peak by other impurities such as Ar, it is essential to control plasma parameters so as to occur recombination of Ar ions, probably Ar^{8+} due to a closed shell structure.

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Impact of additional plasma heating on detached plasma formation in divertor simulation experiments using the GAMMA 10/PDX tandem mirror

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This paper presents the detailed results of the effect of additional plasma heating under the condition of plasma detachment produced in the large tandem mirror GAMMA 10/PDX. In Plasma Research Center of the University of Tsukuba, a research project on divertor simulation research has been conducted using a large tandem mirror device and the experiments have been extensively performed towards the development of the divertor for future fusion devices [1-3]. GAMMA 10/PDX is a large linear device with 27 m in length and utilizes many plasma production/heating devices, such as radio-frequency (RF) wave, microwave and neutral beam injection systems. We have investigated characteristics of high heat and particle fluxes produced by the above plasma heating systems. So far, the particle flux Γ_{ion} of 3.3×10^{23} particles/s•m^2 was achieved by using ICRF wave and superimposing ECH pulse of ~400 kW attained the maximum heat-flux value of ~30 MW/m^2 at the exit of west end-cell.

Divertor simulation experiments for the plasma detachment have been carried out using the divertor simulation experimental module (D-module) at the east end-cell [3, 4]. In detached plasma formation from high temperature plasma (T_{i} ~150 eV, T_{e} ~30 eV, n_{e} 10^{16}~10^{17} m^{-3} at the inlet of D-module), effects of gas injection on the reduction of T_{e}, Γ_{ion} and P_{heat} on the tungsten target installed in D-module were examined using a number of radiator gases (N, Ne, Ar, Kr and Xe). Detailed comparison among these gases probed that Xe gas showed the stronger effect on electron cooling and achieving detached plasma than the other gases. Kr gas showed the intermediate performance between Ar and Xe.

Recently additional plasma heating was applied to detached plasmas and its influence on the detachment state of plasmas was evaluated. In this experiment, the ECH pulse of 150 kW was applied to the upstream region of detached plasmas established by injecting heavy noble gases (Ar, Kr and Xe) and the significant phenomena on detached to attached transition were measured with Langmuir probes, calorimeters, spectrometers and a high-speed camera. In the presentation, detailed behavior of the plasma parameters is described and the effect of ECH is discussed in terms of radiation loss power and ionization and recombination processes.

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Power loads in the divertor phase of Wendelstein 7-X

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Wendelstein 7-X, the world largest superconducting advanced stellarator has five-fold symmetry. It aims to demonstrate high performance steady-state discharges with plasma duration up to 30 minutes. The magnetic field is set up in a way to realize the heat exhaust in the scrape-off layer with magnetic islands. The second operation phase (OP1.2a) started in August 2017. For this campaign the majority of the wall was covered with high-density graphite armor. The plasma surface interaction is realized with 10 uncooled divertors. They are symmetrically placed as an upper and lower divertor in each module following the stellarator symmetry. Later they will be replaced with cooled, high heat flux (HHF) divertors capable of handling steady heat fluxes of up to 10 MW/m . The surface temperatures of the divertors are observed with infrared cameras. In total 9 microbolometric cameras (8-10 µm) and one high resolution IR camera (3-5µm) were installed in a way, that each camera observes one divertor. From the measured evolution of the surface temperature, the heat flux is evaluated using the THEODOR (Thermal Energy Onto DivertOR) code[1]. We have seen heat fluxes of up to 10 MW/m onto the divertor. Power and density scans were performed in three different magnetic field configurations, where each configuration means different magnetic equilibrium and heat flux pattern on the divertors. As expected, the heat flux pattern to the divertor varies strongly with the magnetic configuration, i.e. heat flux location and pattern. Depending on the configuration we have measured strike line width in the range from ca. 1 cm to ca. 3 cm. However, one needs to stress that in stellarators with 3D strike line, the simple parameter of strike line width does not describe properly the wetted area.

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Effect of gas puff and pump on plasma detachment associated with molecular activated recombination in GAMMA 10/PDX

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Molecular activated recombination (MAR) is one of the important recombination processes which can lead to plasma detachment. The MAR reaction depends not only on electron temperature (T_e) and density (n_e) but also on ion temperature and molecular states. In order to clarify effects of the molecular states on plasma detachment, we have performed experiments of hydrogen gas puff and pump in divertor simulation experimental module (D-module) which is installed in the end region of GAMMA 10/PDX tandem mirror. It is shown that the contribution of MAR processes to detachment is changed due to puff and pump even though T_e is the same.

The D-module consists of a stainless-steel cuboid chamber with an inlet hole and a V-shaped target. The end loss plasma is exposed to the target. Additional hydrogen gas can be supplied to divertor simulation plasma (i.e. plasma in the D-module) from the vicinity of the inlet. The D-module has an exhaust door at the back side. When the door is open, neutral particles flow out (i.e. pump).

The additional hydrogen gas was supplied to the divertor simulation plasma in the cases where the exhaust door was closed and open. In the both cases, T_e near the corner of the V-shaped target decreased from ~20 eV to ~2 eV and rollover of n_e was observed. More hydrogen gas was necessary for T_e to decrease to ~2 eV in the case where the door was open. In the both cases, the density rollover occurred at T_e ~6 eV and n_e ~0.7×10^{17} m^{-3}. The dependence of n_e on T_e was almost the same for the both cases until T_e decreased to a few eV. After that, n_e with the door open was higher than that with the door closed at the same T_e. As T_e decreased to ~2 eV, n_e with the door closed decreased to ~0.2×10^{17} m^{-3}. On the other hand, n_e with the door open decreased to ~0.6×10^{17} m^{-3}. It looks like the dependence of n_e on T_e bifurcated when T_e decreased to a few eV. Ion particle flux showed the similar tendency to that of n_e. The flux rollover occurred at ~13 eV and the dependence of the flux on T_e bifurcated at T_e ~10 eV. When the door was closed and T_e was ~3 eV, hydrogen molecular rotational and vibrational temperatures (T_{rot} and T_{vib}) were estimated to be ~2000 K and ~9000 K from Fulcher-α spectrum, respectively. On the other hand, T_{rot} and T_{vib} with the door open were ~1000 K and ~7000 K at T_e ~3 eV. It is suggested that the change in the contribution of MAR processes to detachment occurred due to changes in hydrogen molecular rotational and vibrational levels caused by puff and pump.

ELM Power Deposition on a Tungsten Leading Edge in a DIII-D He Plasma

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Power exhaust is among fusion's major issues. High heat loads on surfaces that protrude into the plasma (leading edges), power bursts onto plasma facing components (PFCs) during Edge Localized Modes (ELMs) and requirements for aligning and shaping PFC elements are some specific concerns. A goal in the 2016 DIII-D DIMES tungsten (W) block experiment (WBX) in support of ITER was to study heat loads from H-mode ELMs. The Divertor Materials Experiment Station (DiMES) held 10-mm-wide W blocks with 0.3 and with 1 mm leading edges during exposure to helium (He) plasmas heated with Electron Cyclotron Heating (ECH). We reported the slight melting of the 1-mm W leading edge and our best-fit estimates of ~50 and 10 MW/m^2 for the parallel and top surface heat loads from analysis of infrared (IR) data, probes and thermal modeling. [1] This heat load ratio is consistent with geometry when the ~2.9° pitch of the block's top surface is added to the angle of incidence. While H-mode ELMs were present, events that we describe here as extended ELMs associated with H-L back-transition dominated the transient particle exhaust. We estimated their duration of 1-2 ms from alpha-light data from a filter scope. This paper presents new information on the role of these unusual extended ELMs in melting a W leading edge. Their pathology may have relevance for ITER He plasmas in its start-up phase. The extended ELMs in shot 166843 occurred with a frequency of ~10-20 Hz through most of the shot. The particle exhaust during these events significantly reduced the plasma density, enough in some cases to trigger gas puffing. The peak heat loads measured by an infra-red (IR) camera viewing a graphite tile at another toroidal location during and between these extended ELMs respectively were 3.1 and 1.2 MW/m^2. We assume these values, extracted automatically using THEODOR, give a good value of the integrated heat load under the ELM peak, although the individual peak values of a series of multiple overlapping ELMs may be higher. Another IR camera viewed DiMES from overhead. The rise in surface temperature of the W block from heat absorbed during multiple extended ELMs was discernable in comparisons of IR data frames over intervals of 100 ms from this camera. In our evaluation, we select periods (200-1000) ms during shot 166843 with several repeated and separated extended ELMs of similar size. We extract heat loads by matching surface temperatures from a detailed thermal model to the IR data where the camera is not saturated ($T_{saturation}$ of ~1200° C for WBX). We calculate temperatures beyond this range, e.g., at the leading edges, with the thermal model. The analysis at temperatures near melting of a series of extended 2-ms ELMs with a frequency of 15 Hz shows the temperature of the leading edge rising by ~265° C. The ratcheting per cycle of ~40-50° C declines as the temperature increases. We will report more detailed analysis that include melting and radiation losses.

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Improved understanding of detachment on JET through improved camera tomography

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The next generation of fusion devices, such as ITER and DEMO will require strong mitigation of the steady-state heat and particle fluxes to divertor plasma-facing surfaces (PFCs), which could be achieved through the detachment process. Understanding detachment and being sure we have the proper representation of the underlying physics would then be essential to designing and operation of a Demo reactor. However, testing and exploring detachment physics is limited by both our diagnostic capabilities and analysis; The research presented herein is part of a plan to address those limitations.

A first step is the application of Bayesian inference techniques to JET H-mode plasma divertor camera images (development of inversion method covered elsewhere [1]), filtered for particular Balmer and carbon emission lines. This has provided more accurate renderings of the underlying emissivity in our current studies by factors of 2-3 (from comparison of the inversion of a generated brightness image by current and Bayesian technique). From the emissivity profiles of D_γ, NII and D_γ/D_α we have extracted the movement of the peak in ionization, N radiation and recombination from target to x-point, eg ΔZ_{NII} (m), as detachment develops in response to the influx of N_2 ,$\Delta\Gamma_{N2}$ ($1x10^{22}$/s). We find that the sensitivity of the inter-ELM , ΔZ_{NII} (m)/$\Delta\Gamma_{N2}$ ($1x10^{22}$/s) [2] to be ~ 0.3. Similar sensitivities are found for the position of the recombination front to the N_2 injection rate. We are currently examining how such sensitivities vary as the divertor conditions are changed along with benchmarking the new camera image inversions against the chordal spectroscopy measurements of the same lines. Incorporating of the chordal spectroscopy data into the inversions should allow quantitative studies of ion sinks and sources and their roles in the target ion current rollover at detachment.

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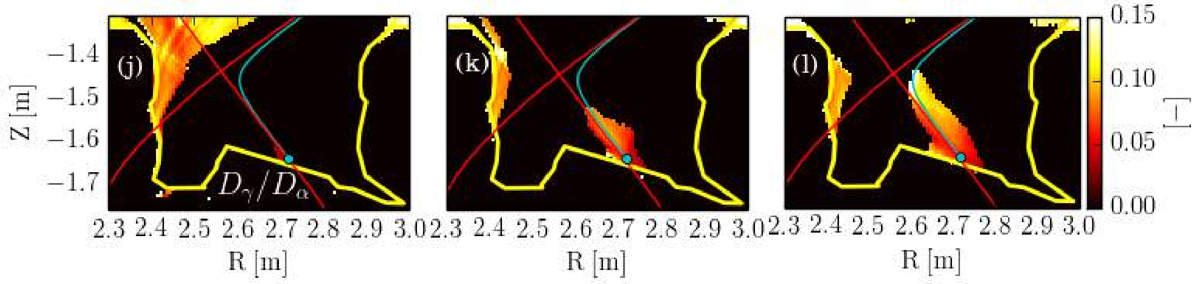

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SOLPS simulation for alternative upper divertor geometries in ASDEX Upgrade

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Alternative divertor configurations, such as the X-Divertor (XD) [1] or the low-field side snowflake-minus (LFS SF-) [2] are currently discussed as a possible solution for the power exhaust problem in a future fusion reactor. For this reason ASDEX Upgrade (AUG) recently decided the installation of a pair of in-vessel coils in the upper divertor [3] to study these alternative configurations experimentally in a machine with a high heating power compared to its size. EMC3-EIRENE was the first code to study configurations with two X-points numerically [4], while the applicability of SOLPS was also shown recently [5]. In contrast to EMC3-EIRENE, SOLPS includes volumetric recombination and drifts. These terms were required to describe the AUG scrape-off layer plasma including the LFS/HFS asymmetries [6] with a detached divertor [7]. The access to detachment as well as the radiative capabilities will be important criteria to evaluate the suitability of the different configurations for a reactor that will need to operate with a significant degree of detachment. In fact an enhanced radiative fraction as well as a substantial up- to downstream pressure drop indicating a significant degree of detachment were found in SOLPS L-mode simulations (without drifts) for the LFS SF-, while a reference standard single null case with the same nitrogen gas puff rate, power influx and upstream pressure was still attached. In contrast to [8] we will focus here on the numerical details of the simulation analyzing the importance of the different particle- and momentum loss processes. In addition to this, the effect of a neutral particle baffle, foreseen to be installed at a later stage of the divertor upgrade, on the detachment threshold is studied. We will also show first preliminary simulation results with partially activated drifts for the LFS SF-. These drifts may play a role for the transport across the primary- and secondary separatrices [9] and therefore for the activation of the secondary strike points.

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Assessment of particle and heat loads to the upper open divertor in ASDEX Upgrade and comparison with SOLPS simulations

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Drifts and divertor geometry play a fundamental role in power, momentum and particle transport. An experiment has been carried out in the upper open divertor of ASDEX Upgrade to characterize their impact.

Infra-red thermography measurements of the heat flux profiles in upper single-null, low confinement mode discharges in ASDEX Upgrade show that the heat loads onto the upper low-field side target increase by factors up to ~ 8 when changing from unfavourable ($B_T < 0$, ∇B drift downwards) to favourable ($B_T > 0$, ∇B drift upwards) toroidal field directions. The evolution of the heat loads with increasing plasma current and core density is different for both field directions. Increasing the plasma current has a greater impact on heat flux profiles, q, in favourable direction than in unfavourable direction: the ratio of $q_{peak,B_T>0}/q_{peak,B_T<0}$, in attached conditions, increases from ~ 2 at $I_p = 0.6\,$MA up to ~ 8 at $I_p = 1.0\,$MA; however, heating power increases more with I_p for $B_T < 0$ (up to factor of ~ 2) than for $B_T > 0$ (up to $\sim 40\%$). The effect of increasing core density on the heat flux profiles, while in attached conditions, is more pronounced in unfavourable direction than in favourable direction: for $B_T < 0$, q_{peak} increases up to a factor of ~ 2, while almost no change is observed for $B_T > 0$; the heating power changes with core density slightly for $B_T < 0$ (up to $\sim 20\%$). At the highest densities obtained, power detachment is observed for both toroidal field directions. The onset of power detachment for $B_T < 0$ is apparently triggered at a lower core density.

This dependence on the B_T direction of the evolution with increasing plasma current and core density of the target profiles has also been observed in the collected ion saturation current, I_{sat}, measured by the Langmuir probes. The main difference is that the roll-over of I_{sat} is only observed for $B_T < 0$.

The open divertor configuration in the upper divertor will be compared with the existing lower closed divertor to asses the effect of the divertor geometry. SOLPS simulations for the upper single null configuration will be used to interpret the plasmas and to improve the understanding of the large impact of drifts and divertor geometry observed in these AUG experiments.

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The influence of N_2 seeding in a detached-like H_2 plasma by means of linear machine Magnum-PSI and numerical simulations.

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Experiments have shown that impurity seeding in the tokamak's divertor region leads to a net reduction of power loads onto the targets. Nitrogen is currently the leading candidate for impurity seeding in ITER. Volume processes such as molecular-activated-recombination (MAR) and electron-ion recombination (EIR), together with impurity radiation losses and elastic collisions, may all contribute to achieve a detached plasma regime, in which the heat and particle fluxes are greatly reduced before reaching the surface. This implies tolerable power loads on the divertor's plates. Little is known on the detailed plasma-chemical processes occurring in such scenario in the presence of nitrogen.

To study this complex system, an extensive global plasma model of H_2+N_2 chemistry has been set up on using PLASIMO code.[1] The model has generated qualitative results highlighting new molecular-assisted reactions paths, suggesting NH as main electron donor in charge exchange with H^+ and N_2H^+ as principal ion mediator. The resulting primary mechanisms have been implemented in Eunomia[2], a spatially-resolved Monte-Carlo code based on the test particle approximation method. Such code is coupled with B2.5, a fluid code for the simulation of charged particles. These codes provided further insights into the most relevant physical-chemical mechanisms occurring in such scenario.

Dedicated experiments on plasma-surface interactions and detachment with and without nitrogen seeding have been carried out in Magnum-PSI, a unique linear plasma generator[3] located at DIFFER, capable of reproducing ITER-relevant divertor plasmas. The influence of different nitrogen contents in a detached-like hydrogen plasma has been studied and the results are reported. Together with numerical simulations, a large set of diagnostics have been used i.e. Thomson scattering, calorimetry, optical emission spectroscopy, mass spectrometry and bolometry. Notably, a reduction of plasma pressure of 30% has been observed in the case with a N_2 fraction of 20%, suggesting an enhancement in both the recombination of ions and molecular-induced processes.

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Radiative Heat Exhaust in Alcator C-Mod I-Mode Plasmas

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Integrating tokamak operational regimes at high normalized energy confinement with reactor relevant boundary solutions is a key challenge to realizing net energy production from magnetic confinement fusion. The I-mode regime [1] has a strong thermal edge transport barrier, L-mode like particle confinement times and is naturally free of ELMs, but solutions for power exhaust and erosion control for long-pulse have not yet been demonstrated. Using impurity seeding, experiments conducted on Alcator C-Mod were not able to maintain the I-mode pedestal while achieving divertor detachment. Only modest reductions in heat flux to the outer divertor were achieved, from 23-27 MW/m^2 to 10-13 MW/m^2 while maintaining the I-mode. Impurity levels and mixtures were widely varied, scanning levels of pre I-mode recycling gas seeding, trying separately neon and argon, and adding private flux nitrogen fueling into established I-modes. This was done for a fixed I-mode target at I_p=1.1 MA, B_T=5.7 T, line-averaged n_e=1.5e20 m^{-3} and approximately 4 MW of heating power. The L/I transition was suppressed at elevated early seeding, while in nearly all cases, an I/L back-transition occurred even for modest nitrogen seeding levels. In contrast to the H/L back-transition, the I/L back-transition is weakly perturbing to the boundary plasma, with little to no change in the edge density profile or the loss power. In the L-mode following the I/L transition, strong reductions in parallel heat flux were achieved, dropping from 1.5 GW/m^2 in I-mode to below 0.2 GW/m^2 in L-mode, indicating that the separatrix density, and impurity fraction were sufficient for radiative exhaust, but that impurity seeding directly or indirectly impacted the I-mode pedestal sustainment physics. Experiments to try and detach at lower pedestal collisionality were also unsuccessful. This approach focused on modifying damping of zonal flows which are thought to play a role in the I-mode pedestal [2]. Detailed descriptions of the results of the I-mode experiments are presented, contrasting them with results of recent C-Mod H-mode detachment experiments where appropriate.

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Physics basis for the ITER tungsten divertor

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Building on about 20 years of physics simulation, engineering design and component testing, the ITER tokamak divertor is the largest and most complex ever to be constructed. At the time of the last report to the PSI Conference Series on the ITER divertor status in 2012, the strategy to begin operations with full-tungsten (W) armour had been proposed by the ITER Organization (IO) and was under study. The decision was taken formally in 2013, since when the physics basis in support of the final design has been further developed, with invaluable and numerous contributions from the research community within the ITER Parties. On the eve of component procurement, this paper will discuss the present basis, beginning with a reminder of the key elements defining the overall design, and outlining relevant aspects of the Research Plan accompanying the new “4-staged approach” to ITER nuclear operations which fix the overall lifetime constraint of the first divertor.

The main focus will be on steady state and transient power fluxes in both non-active and DT phases, the main drivers for design and future divertor operation. Stationary loads are obtained from simulations using the 2-D SOLPS-4.3 and SOLPS-ITER plasma boundary codes, assuming the use of the low Z seeding impurities N, Ne and now, for the first time, including fluid drifts, allowing more realistic accounting for in-out target power asymmetries. Imposed by the power handling requirement, the use of W monoblock technology on the divertor targets introduces gap edge heat loading, which must be prevented by surface shaping. This in turn increases the surface heat flux density for given thermal plasma power $q_{||}$ arriving parallel to field lines, with further potential increases due to drifts and narrower than assumed SOL heat flux channel widths. Avoidance of W recrystallization sets an upper limit on the allowed stationary power flux density and since increased fuel gas puffing will increase upstream densities beyond those which may be compatible with disruptive stability, stronger seeding is the principal mitigation route to decrease $q_{||}$ in the event that margins are too eroded. There are, however, limits on allowable impurity concentrations before confinement is compromised and/or full detachment occurs, a process often experimentally observed to be rapid and therefore undesirable from the control point of view. Moreover, whilst N seeding is found to be preferable on today’s all-metal tokamaks regarding performance, the divertor compression of both N and Ne on ITER is predicted to be similar, an important physics issue occupying current R&D. Work is also progressing on the assessment of power loading in the presence of magnetic perturbations for ELM control using the EMC3-Eirene 3-D code suite.

The issue of tolerable limits for transient heat pulses is still an open question and will be addressed here. Although a new scaling for ELM power deposition has shown that there may be more latitude for operation at higher current without ELM control, the ultimate limit is likely to be set more by material fatigue under large numbers of sub-threshold melting events. In the case of disruptions, recent simulations have shown that W vapour shielding should provide significant surface power flux mitigation.

Simulations of a high-density, highly-radiating lithium divertor

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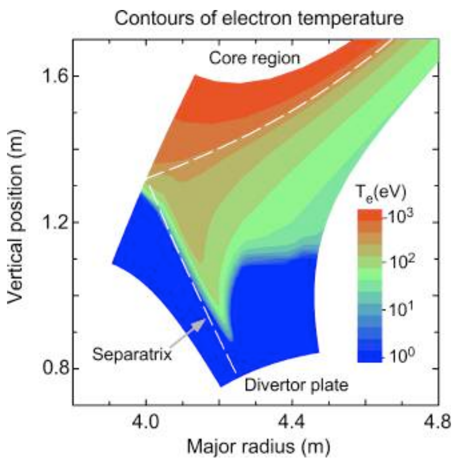
Lithium has been proposed as a plasma-facing material to possibly improve core performance [1] and simultaneously manage the high heat-flux problem on surfaces by radiating most of the exhaust plasma power in the divertor region, where a vapor-box scheme [2] is one possibility. Here results of edge-plasma simulations are reported with the geometry and parameters of the recent FNSF study [3]. A set of calculations are performed with the 2D UEDGE plasma model and a simple diffusive neutral model [4]. Equations are solved for the density and momentum of a DT species and all three charge states of Li, in addition to separate ion and electron energy equations. To mimic a crude vapor-box, Li gas is injected near the divertor plate from the private-flux and outer divertor leg regions and is removed assuming a wall albedo of 0.5 on both PF and outer walls, which allows steady state solutions. The hydrogenic (DT) fuel ions and accompanying electrons transport the core exhaust power into the scrape-off layer. Because of the assumption of Li on surrounding surfaces, recycling of hydrogen ions into neutral gas is negligible. For a range of Li gas input, steady-state, detached-plasma solutions are shown where well over 90% of the exhaust power is radiated by Li, resulting in peak surface heat fluxes ≤ 2 MW/m^2 on the divertor plate, outer wall, and private-flux wall. The figure shows the electron temperature contours in the outer leg for a detached divertor-plasma solution where a high-density Li neutral and ion populations surrounds the plasma along the boundary of the blue, low T_e region. While Li ions dominate in the divertor leg, their density is much less than the DT density at the midplane. The collisional parallel thermal force plays a key role in determining the midplane ion Li density, and sensitivity of results to different model assumptions are discussed. Here the key issue is possible dilution of the core DT fuel. A more complete model of lithium neutral gas is also underway using the Monte-Carlo Direct Numerical Simulation SPARTA code both to simulate the gas behavior for the conditions studied here and also to analyze a more detail vapor-box geometry [5].



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A lithium vapor box divertor similarity experiment for a linear plasma device

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The lithium vapor box divertor [1] is a potential solution for power exhaust in toroidal confinement devices. The divertor plasma interacts with a localized, dense cloud of lithium vapor, leading to volumetric radiation, cooling, recombination, and detachment. To minimize contamination of the core plasma, lithium vapor is condensed on cool (~400 °C) baffles upstream of the detachment point. Before implementing this in a toroidal plasma device with a baffled-slot divertor geometry, we consider an experiment with a scaled baffled-pipe geometry in a high-power linear plasma device such as Magnum-PSI [2].

As shown in Figure 1, the baffled-pipe system is composed of three cylindrical steel boxes with length and diameter of ~15cm, joined end to end by 'nozzles.' The system is oriented horizontally so that the plasma beam may enter the open nozzle. A few tens of grams of lithium will be placed in the farthest downstream of the boxes, which is coated internally by a porous material to wick liquid lithium. When that box is heated to 660 °C, it is predicted to be filled with lithium vapor with a density sufficient to cause detachment in a 4kW, 9MW/m^2 plasma beam.

Without plasma present, the upstream boxes at 400°C will condense the 1.5g per minute that evaporates from the downstream box. To validate a model of vapor flow requires measurement of the flow of lithium and of the box wall temperatures. Mass flow can be measured to within 10% by weighing the ~1kg boxes to 0.15g accuracy before and after 15 minutes of operation at temperature, or by measuring the rate of heating of the upstream boxes due to the latent heat of condensation to 10%. To reach a 10% uncertainty in predicted flow rate, *absolute* temperatures of the box walls must be measured to 3K, which may be achievable by careful application of very stable thermocouples. Such an experiment, without plasma, is planned at PPPL.

In experiments with a plasma beam we aim to validate the model of detachment and power spreading of the plasma beam. In this case, changes in temperature at the strike point and on the box surface will be measured by thermocouples during plasma exposures of a few seconds. When detachment occurs, the 4kW plasma beam is expected to transition from heating the strike point at a rate of hundreds of K/s to heating the whole box at 9K/s. In addition, power to the strike point and box wall can be inferred by calorimetry of two separate cooling systems, measuring coolant-in and coolant-out temperatures. If the box can be sufficiently cooled, minutes-long experiments could test the 'plasma plugging' effect, expected to decrease lithium efflux from the first box by 90% or more, by measuring changes in the mass of the second box.

Design of the experiment including heating, cooling, and temperature measurement systems will be presented.[1]

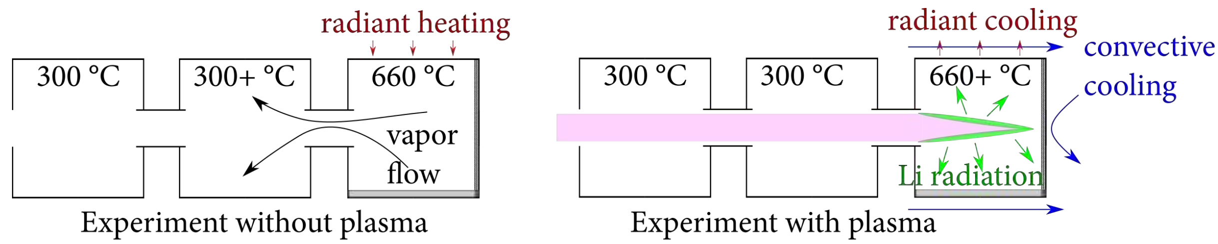

Figure 1: Experiments without and with plasma present. The first experiment tests vapor flow and the second tests detachment and power spreading. Example temperatures and modes of heat transport are shown.

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1 This work supported by DOE Contract No. DE-AC02-09CH11466.

Dependence of Neutral Pressure in Variable Divertor Geometry on Detachment on DIII-D

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Neutral pressure measurements in closed divertor configurations on DIII-D show a large increase when the divertor plasma shifts into high recycling and detachment, consistent with present understanding. Multi-point pressure measurements presented here are from the partially closed upper divertor where more diagnostics are able to quantify detachment onset and the more closed small angle slot (SAS) divertor [1] where measurement access is restricted. These measurements are used to help understand detachment and validate divertor design metrics.

During experiments ramping density to detachment in the upper divertor with the ion B×∇B drift toward the divertor target, a rapid increase in neutral pressure in the plenum near the outer strike point is observed at the onset of detachment [2], which is understood to be the transition from a high recycling regime to fully detached [3]. This is further verified by the occurrence of the rollover of the ion saturation current to the target as density increases and the CIII radiation front lifting away from the target. This pressure rise reaches up to 30 mTorr from a baseline pressure of 1 mTorr, which indicates a strong neutral buildup at the detached leg and is consistent with a high recycling regime and detachment. If the outer common flux area is not actively pumped, the pressure can come to near equalization on either side of the leg, while pumping on the outer leg requires larger gas puffing to achieve detachment, albeit at similar density and less plenum pressure (<10m Torr).

With the strike point in the optimal position in the SAS divertor, in-tile pressure gauges were designed to measure neutral pressure in the "near" SOL (inside Ψ_N~1.005) and the "far" SOL (inside Ψ_N~1.015). Initial experiments performed with the ion B×∇B drift away from the divertor target show a larger pressure in the near SOL (4 mTorr) compared to the far SOL (1.2 mTorr) by a factor of up to 3 indicating neutral pressure buildup in the narrow region. A rapid increase in pressure is also observed in this configuration as density increases, suggesting a transition to a high recycling regime. Assuming neutral temperatures in the SAS are ~4eV, a simple model balancing the particle flux at the gauge inlet can relate the atomic deuterium pressure in the divertor to approximately 10 times higher to the wall-thermalized molecular deuterium pressure in the gauge. Additional planned experiments will further exploit these pressure measurements with the ion B×∇B drift toward the divertor target where detachment onset happens more rapidly with increasing density. Work supported by US DOE under DE-AC05-00OR22725, DE-FC02-04ER54698.

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Divertor detachment optimization in an unfavorable open geometry horizontal target divertor configuration in NSTX and NSTX-U.

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Tokamak divertor detachment optimization is usually achieved by strong neutral baffling with divertor hardware structures and vertical divertor targets, both aimed at increasing neutral compression and ionization source in the divertor, as is implemented in, e.g., the ITER divertor design. We show that divertor geometry advantages similar to the baffled vertical plate divertor can be obtained by optimizing a standard divertor *magnetic* configuration albeit using an unfavourable open geometry horizontal target divertor. Such divertor geometries have been realized in spherical tokamak (ST) NSTX and are planned for NSTX Upgrade, where projected inter-ELM peak divertor heat fluxes may exceed 10 MW/m^2 in 2 MA, 10 MW, 5 s long discharges thus challenging plasma facing component thermal limits. High-performance H-mode discharge scenarios in ST-based devices are based on projections of increased neutral beam current drive efficiency and improved confinement at low electron collisionality. At low collisionality and lower density, conventional divertor heat and particle load mitigation techniques based on induced volumetric power and momentum losses (the radiative divertor), face challenges, further aggravated by the inherent compactness of the ST divertor.

In 1 MA, 6 MW NBI-heated NSTX highly shaped H-mode discharges with P_{SOL}~4.5 MW and ion $B\times\nabla B$ drift toward the main X-point, controlled variations of the X-point height and the outer strike point major radius R_{OSP} were used to produce a divertor database with varying geometries (connection length, flux expansion, divertor leg length cf. ionization m.f.p., divertor leg angle) and plasma properties (density, heat flux profiles, D_a and carbon emission profiles, and neutral pressures). The poloidal angle between the separatrix and the horizontal divertor target, as well as the X-point height were found to play a key role in controlling neutrals. Significantly increased ionization source and neutral compression were found in the lower X-point height configurations with $R_{OSP} < R_{Xpt}$, where recycling neutrals were directed toward the separatrix and baffled by the plasma separatrix and horizontal target. Modelling of these configurations using the multi-fluid code UEDGE with charge-state resolved carbon and inertial neutrals, as well as the database analysis of parallel divertor heat fluxes and recombination rates at higher upstream densities, suggest that such configurations are more favourable for detachment. The modelling shows that the configuration reaches partial OSP detachment at lower density and the divertor temperature lowering is monotonic with upstream density, unlike all other configurations that show an abrupt temperature drop. The findings are also consistent with previous NSTX detachment experiments conducted in similar standard divertor and snowflake-minus configurations with and without additional deuterium puffing.

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Abstract Withdrawn

Spectroscopic studies on the enhanced radiation with high Z rare gas seeding for divertor detachment in LHD plasmas[1]

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In the forthcoming ITER (International Thermonuclear Experimental Reactor) tokamak, the achievement of the divertor detachment is indispensable for mitigating heat loads onto the divertor plates. One of the solutions is radiative cooling with inert gases seeded into the edge plasmas so as not to deteriorate the confinement properties of core plasmas. Therefore, injections of nitrogen (N_2), neon (Ne), argon (Ar) and krypton (Kr) have been extensively carried out so far in existing tokamaks and helical devices including the Large Helical Device (LHD) at the National Institute for Fusion Science [1]. Low Z impurities (N_2 and Ne) radiate mainly outside the confinement region, while high Z impurities (Ar and Kr) may radiate in the region closer to the core plasmas. Though the difference in radiative region between Ne and Kr has been qualitatively demonstrated in LHD using an imaging bolometer [2], each contribution of each ion stage to the radiation enhancement has not yet been clarified in detail, especially for higher Z impurities.

In order to answer this question, spectroscopic studies have been performed in the LHD plasmas with inert gas seeding for detached plasmas. In particular, extreme ultraviolet (EUV) and soft X-ray emission spectra of Kr ions have been measured in a variety of wavelength ranges using a grazing incidence spectrometer. We have identified isolated lines or quasi-continuum bands from ion stages in the range of Kr^{5+}–Kr^{25+}, the time trends of which were compared with that of the total radiation power measured by a bolometer. The trends of Kr^{18+} and Kr^{17+} line intensities roughly follow the trend of the enhanced radiation, which implies major contributions of ions with n=3 outermost electrons. Also, difference between Ne and Kr is discussed based on a trial of Kr and Ne injections in the same discharge for more effective radiation enhancement in wider areas in the edge plasma. The line intensities of Ne and Kr ions clearly differ in time trend, and differ also from the enhanced total radiation including contributions of both the impurities.

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Comparing N versus Ne as divertor radiators: SOLPS-ITER simulations of impurity seeding in ASDEX Upgrade-like geometry

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To remain below technologically feasible stationary power loads on its tungsten targets, burning plasmas on ITER will require partially detached divertor operation actuated by the use of impurity seeding. As a consequence of operation close to the H-mode power transition threshold (P_{sep}/P_{LH} ~2), it is currently expected that ITER will use the low Z extrinsic radiators, nitrogen (N) and neon (Ne), since higher Z impurity will lead to too much core radiation loss. Experiments on current all-metal wall devices (e.g. ASDEX Upgrade and JET), however, indicate that there are important differences between the two species in terms of impurity transport and divertor compression ($\eta_{imp}=c_{imp,div}/c_{imp,core}$). Nitrogen is generally observed to be more efficiently compressed, providing more localized divertor radiation. In contrast, simulations for ITER at high performance indicate that compression will be similar. The fully recycling nature of Ne makes it more attractive as a divertor radiator on ITER so it is extremely important to understand the physics determining the differences in impurity behavior.

In an attempt to address some aspects of this problem, the SOLPS-ITER code package has been used to compare scrape-off layer (SOL) distributions of N and Ne in test simulations under inter-ELM, deuterium H-mode conditions in ASDEX Upgrade geometry with all drifts and currents switched on. Similar studies have been conducted with older versions of SOLPS code [1]. In the present work recent, more accurate description of thermal and friction forces acting on the impurity ions is used [2]. The ratio of seed gas to fuel throughput in the simulations (5.6% N, 3% Ne in terms of electrons) is such that the new treatment of parallel momentum balance strongly influences the SOL impurity distribution.

The cases feature completely detached inner target and are partially detached at the outer. They correspond to P_{SEP}/P_{LH}~3 and more than 60% of the 5 MW of input power to the simulation domain is radiated. Under these conditions, the code finds $\eta_N \approx 2\eta_{Ne}$ with a resulting factor of ~2 higher value of the ratio $P_{RAD,DIV}/P_{RAD,TOT}$ for N. The improved thermal and friction force description [2] leads to a greater accumulation of impurity in the upper SOL regions, an effect which is stronger for Ne than N. The code findings regarding divertor impurity compression are consistent with experimental observations on ASDEX Upgrade, for which preliminary benchmarking of SOLPS-ITER to a closely matched pair of H-mode N/Ne seeded discharges is underway.

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*See the author lists: A. Kallenbach et al., Nucl. Fusion 57 (2017) 102015; H. Meyer et al 2017 Nucl. Fusion 57 102014

Effects of curved divergent magnetic field on heat load in the linear divertor simulator TPD-Sheet IV

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Steady-state fusion reactors and DEMO reactors will have much higher heat flux from the core than that from ITER, which itself exhibits heat flux that is several times larger than that available in the current fusion reactors. Therefore, for the development of such advanced reactors, it is necessary to enable additional heat removal. Recently, advanced divertor aiming at additional heat removal by improving the magnetic field structure has been studied. The use of the advanced divertor such as a Super-X divertor (SXD) and Snowflake divertor (SFD) is one possible solution for the issue of heat load. A number of studies have simulated the advanced divertor, and experiments had been conducted on several large devices. However, the experiment is still not enough and it has not reached the practical stage. Also, fundamental experiments to clarify characteristics of advanced divertor have not yet been performed. Therefore, we conducted fundamental experiments to investigate the variation of characteristics about the backflow, detached plasma and heat load by the magnetic field. We had reported the characteristics of the neutral-particles backflow under the curved divergent magnetic field [1]. In a recent work, we experimentally examined the generation of detached plasma in a curved divergent magnetic field and the relationship between the plasma-facing area and the magnetic field. The experiments had been performed by a linear divertor simulator, known as the Test plasma Produced by Directed current discharge for the Sheet plasma IV (TPD-Sheet IV) [1-3]. The plasma parameters such as the space potential, the electron temperature and electron density were measured by a Langmuir probe. In addition, a visible spectroscopy was performed for the same position of the probe measurement. The heat load on the target was evaluated by an IR camera. As a result, it was confirmed that the expansion of the plasma by the magnetic field is limited. In this experiment, it was also confirmed that the space potential became deeper with the divergence of the magnetic field. Therefore, it is considered that one cause of this phenomenon is the well-type potential of the plasma. However, in the convex-type potential observed in large devices, the possibility of efficient heat load reduction due to the converse effect is expected. In this contribution, we will report details on these results.

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Heat-flux Footprints at the Inner Divertor Target During I-mode on the Alcator C-Mod Tokamak

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Most of the attention around the boundary heat flux width (λ_q) has been focused on the *outer* target. For example, a multi-machine database for λ_q at the outer target for attached plasmas with H-mode confinement has been assembled [1] and has yielded the so-called "Eich-scaling", λ_q [mm] = $(0.63\pm0.08)\times(B_p\ [\mathrm{T}])^{-1.19\pm0.08}$, where B_p is poloidal magnetic field at the outboard midplane. This result, in particular the absence of a favorable size-scaling, has accentuated the exhaust-power-handling challenge for a reactor. A number of advanced divertor concepts addressing this challenge have been invented, e.g. the "Super-X"[2], the "X-point Target" [3], and the "Li Vapor Box" [4], but all are directed toward handling the power on the outboard divertor leg. There are fewer ideas for handling power on the inner leg, where there is less space to implement them. This has motivated our measurements of heat-flux footprints on the inner target of C-Mod's I-mode plasmas in single-null (SN) and near-double null (DN) configurations. We find that for LSN I-mode plasmas the minimum values of λ_q are ~1-1.5 mm and are consistent with the "Eich-scaling" over the B_p range from 0.65 to 0.95 T. The I-mode plasmas are of particular interest because to date we have been unable to detach either the inner or the outer target in that confinement mode with impurity seeding [5]. While the minimum detected widths follow the scaling, we also measure widths that are significantly larger in many instances, both within a single discharge and between discharges with nominally similar core plasma conditions. We are presently searching for "hidden variables" that might lead to this scatter above the observed minima.

Because heat transport to the boundary is dominant on the low-field side, DN is one way to mitigate the inner target heat-fluxes. We have investigated the influence of magnetic flux balance on the inner target profiles in the lower divertor. By modulating the separation between the lower and upper separatrices ("δR_{sep}"), the experiments showed that we could easily reduce the total power on the lower target to half the value seen in LSN. But they implied that δR_{sep} must be controlled to significantly less than λ_q in order to share the *peak* heat-flux between lower and upper targets.

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Attainment of passively stable, fully detached regimes in long-legged, tightly baffled divertors

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Passively stable, fully detached divertor regimes have been recently found in numerical modeling of divertor configurations with radially or vertically extended, tightly baffled, outer divertor legs, with or without a secondary X-point in the leg volume [1]. The model parameters are based on those of the ADX tokamak design [2], and the simulations are carried out with the tokamak edge transport code UEDGE. A fully detached regime is found to persist for a wide range of input power from the core; as input power is varied, the location of a stable detachment front in the leg shifts away from, or closer to, the divertor target. At the lowest input power, the detachment front location approaches the primary X-point, which defines the lower limit on the detachment power window. When a secondary X-point is embedded in the divertor leg, the detachment power window is largest, accommodating a factor of ~10 variation in core input power. Consequently, the maximum power handling capability of the long-legged divertor appears to be an order of magnitude higher than a conventional geometry operating with otherwise identical conditions. These features are highly desirable for a reactor: fast power exhaust transients might be accommodated passively while the location of the detachment front may be detected and actively controlled over long time scales [3]. Two important questions are: (i) Does this behavior persist as modeling assumptions are varied? and (ii) Does this detachment regime scale to power-plant-size devices operating at their full design parameters? To help answer the first question, we have explored the sensitivity of the detached divertor response to various assumptions and parameters used in the model. These include anomalous plasma radial transport, neutral transport, impurity species, and the geometry of plasma-facing material surfaces. With several essential model assumptions varied, the overall picture of a stable fully detached regime in a tightly baffled long-legged divertor still holds, lending confidence in these modeling results. The key physics for the detached divertor regime in these simulations involves the interplay of strong convective plasma transport to the outer wall of the divertor leg, confinement of neutral gas in the divertor volume, geometric effects including a possible secondary X-point, and atomic radiation. The second question is addressed in a companion paper [4] that explores the performance of these long-leg divertors exposed to the power exhaust of the ARC pilot-plant design. These results motivate experimental measurements and modeling of plasma turbulence and transport in long-legged divertors and the consideration of how these processes might scale to the plasma conditions that are anticipated for the divertor of a fusion reactor.

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Divertor Heat Flux and Particle Control on NSTX-U via Optimization of Plasma Boundary and Divertor Shape

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This work presents strategies for reduction of divertor heat flux and enhancement of particle pumping on NSTX–U through magnetic optimization and control of the plasma boundary and divertor shape. One of the primary goals of the NSTX–U research program is to assess spherical tokamak energy confinement scaling as a function of an expanded range of plasma parameters. To achieve this goal, significant upgrades to the device were recently completed, including the installation of a larger-diameter centerstack and a second neutral beam injection system, that increase the achievable plasma current (2 MA), toroidal field (1T), input heating power (10 MW), and plasma discharge duration (5s) [1]. Further improvements will likely include a cryopumping system for enabling enhanced density control. Long-pulse plasma operations on NSTX–U at maximum current, field, and power will require robust control of both the heat flux profile on the divertor as well as the location of scrape-off layer plasma relative to cryopump surfaces. A procedure is developed for tailoring these parameters through optimization of the magnetic geometry in the divertor region. Simple models for the scrape-off layer heat flux and particle profiles are used to map these quantities onto divertor surfaces in a variety of magnetic configurations such as the standard and snowflake (SFD) [2] divertors. An algorithm is developed for determining magnetic geometries in NSTX–U that are optimal for achieving some user-defined criterion, such as maintaining the peak divertor heat flux below some maximum value or obtaining a desired level of density pump-up. Recently-developed feedback control schemes [3] are augmented to enable real-time implementation of the optimization approach.

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New insights into the physics and dynamics of divertor ion current loss during divertor detachment in TCV

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Detachment is predicted to be of paramount importance in handling the power exhaust for future fusion devices, such as ITER. However, a direct experimental quantification of the role and spatial profile of the various atomic processes controlling the loss of divertor ion current during detachment has, until now, not been available due to lack of ionisation measurements.

The physics of the target ion current loss, a defining and important feature of detachment, is studied in TCV using density ramp, or N_2 seeded, L-mode discharges with various plasma currents. Novel spectroscopic analysis techniques utilising the hydrogen Balmer series have been developed to infer the ionisation source magnitude and distribution. This provides, together with the ion sink in the plasma (recombination), electron density and divertor power balance measurements, a detailed picture of particle and power balance along the outer divertor leg. The results for a conventional single-null topology show the divertor ion source tracks the ion target flux both in magnitude and in time: both the ion target current and ion source decrease together at detachment. Surprisingly, the volumetric recombination ion sink – commonly thought to be the primary detachment ion loss mechanism – is only a small (sometimes negligible) portion of the ion current losses. New evidence from TCV shows the driver of the divertor ion source decrease is a decrease in power flowing into the ionisation region combined with an increase in the energy required per ionisation – essentially 'starving' the ionisation region of power. SOLPS modelling of a TCV density ramp discharge with a conventional divertor configuration has reproduced the general characteristics of the ion source reduction. 'Power starvation' of the ionization *source*, therefore, appears to be central to loss of divertor target ion current.

The divertor plasma *sink* for ions, recombination, is maximised at the highest divertor densities, which are achieved at the highest core densities. Nitrogen seeding enables detachment by 'power starving' the ionisation region at lower core densities and hence lower recombination sink. The ratio between the recombination ion sink and the ion target flux increases when poloidal flux expansion is increased (x-divertor) under constant core conditions. Understanding these changes may be key to understanding the role of magnetic geometry on detachment and possibly the detachment process itself.

Estimate of 3D wall heat loads due to Neutral Beam Injection in EU DEMO ramp-up phase

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High energy Neutral Beam Injection (NBI) is one of the methods being considered in EU DEMO pre-conceptual design phase to provide auxiliary power to the plasma. From recent studies [1], it appears clear that auxiliary heating power is needed during the ramp-up (and ramp-down) phase to guarantee a robust access to H-mode (and to compensate for high radiation power losses in ramp-down). The use of NBI during ramp-up has to be carefully considered due to possible shine-through losses which can exceed the maximum heat load tolerated by the first wall (for DEMO the steady state peak heat flux limit is 1 MW m^{-2} [2]). In ITER, shine-through losses pose a lower limit on density for NBI operation at $n \sim 3 \times 10^{19}$ m^{-3} [3]. This limits for ITER the operational window of the NBI system and can prevent its use during the ramp-up phase due to low plasma density.

In this work the heat wall loads due to NBI shine through and orbit losses are calculated for the diverted plasma ramp-up phase of EU DEMO pulsed scenario by numerical simulations performed by BBNBI [4] and ASCOT [5] Monte Carlo codes. The simulations have been done in a complete 3D geometry considering the latest DEMO NBI design [6], which foresees NBI at 800 keV energy with respect to 1MeV beam energy for ITER. Location and power density of NBI-related heat loads at different time-steps of DEMO ramp-up are evaluated and compared with the maximum heat flux limit. Since NBI shine-through losses depends mainly on the beam energy, plasma density and volume, DEMO has a more favorable situation than ITER, enlarging NBI operational window. This increases the appeal of neutral beam injectors as auxiliary power systems for DEMO.

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Performance assessment of tightly-baffled long-leg divertor geometries for application in the ARC reactor concept

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Divertor power loading remains an unsolved problem for fusion energy. High-power, reactor-level devices need significant heat flux mitigation in the scrape-off-layer (SOL) to avoid surface damage/erosion to divertor targets and plasma facing components. A number of advanced divertor configurations have been proposed as potential solutions, including double-nulls, long-legs and magnetic field flaring with secondary X-points. Modelling of tightly-baffled, long-leg divertor geometries in the divertor test tokamak concept ADX have shown the potential to access passively stable, fully detached divertor conditions over a broad range of parameters [1].

The question remains as to how these divertor configurations may perform in a reactor. To test this, we study these configurations in the context of the ARC reactor concept [2]. ARC employs high-temperature superconductor toroidal field (TF) coils that are demountable, allowing the poloidal field coil set to be placed inside the TF coils while still being sufficiently shielded to neutron damage by the blanket. The ARC design has been updated [3] to include a tightly-baffled, long-leg divertor with an X-point target [4]. ARC provides an appropriate reactor testing scenario for advanced divertor configurations, with a projected SOL heat flux width of 0.4 mm and total plasma power exhaust requirement 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 modelled by radial diffusion and advection, consistent with experimental levels of in-out transport asymmetry and plasma interaction with sidewalls. Initial studies employing a super-X divertor configuration and neon impurity radiation included in the "fixed-fraction" model (with fraction ~0.5%) have shown that a stable detached divertor operational window exists for reactor exhaust power in the range of 72 to 92 MW [5]. Under these conditions, the divertor heat flux is fully dissipated by radiation and radial losses to the sidewalls of the divertor channel. Simulations are 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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Modulated heating scenario development for detection of surface layers and hotspots in W7-X

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Multiple infrared thermography views are used to measure temperatures of graphite surface components inside of the Wendelstein 7-X stellarator. But to determine heat flux loads (up to ~10 MW/m^2) on the divertor, it is necessary to know the thermal properties of the materials, and in particular, to be able to distinguish between poorly coupled surface coatings or hot spots, and undisturbed wall/divertor surfaces. Furthermore, these thin (H, C, O) [1] surface layers can change during the day, or even within long discharges. Typically, in tokamaks transient power loads (e.g. ELM's) are used in order to study the evolution of the surface layers. Since ELMy regimes are rather exotic in stellarators, we have developed specific scenarios with modulated input power to allow surface layer detection. This is in addition and in contrast to numerical algorithms designed for use in the absence of external modulations (reported elsewhere in this conference A. Ali, et al). During the first operating phase (OP1.1) on W7-X, we had a FLIR camera with 424 Hz framing rate [3]. It could easily track 17 Hz ECRH modulations (50% power depth) effects on the graphite limiters. But during the present second operating phase (OP1.2a) with un-cooled graphite divertors, the majority of the IR diagnostics (with a wide field of view) have only a 100 Hz framing rate, which restricts us to using slower input heat (ECRH) modulations. Target plasmas typically have > 100 millisecond energy and particle confinement times. As a result, we developed a plasma scenario with a 5 Hz (6 MW, 200 millisecond on/off) ECRH pulse train, performed once or more per week. During the "off" phase, fast and slow decays of surface temperature can be seen, depending on the presence of coatings, or of a clean surface. Additionally we have swept the strike line over the divertor with movement of ca. 8 cm and frequency of 5 Hz, which also heats and cools different regions during the plasma pulse. We will report on the evolution of the divert via these observations over the course of the first divertor campaign.

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Turbulent Transport Integrated Numerical Studies of Power Exhaust in H-mode Plasma Boundary in Tokamaks

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In tokamaks, the power transported across the separatrix spreads over a narrow radial width of about 1 mm at the midplane, resulting in excessive heat flux on the divertor targets [1]. Therefore, one of the most essential issues that need to be tackled in tokamaks is the power exhaust.

Impurity seeding has been successful as a tool to reduce/mitigate heat flux on the targets. Modeling tools such as SOLPS, SOLEDGE, EDGE2D, UEDGE etc have been used to help understanding related physics and pursuing after steady operation with impurity seeding. One drawback of such modeling studies, though, is the *ad hoc* anomalous radial transport coefficients used. Experimentally the transport in the scrape off layer is found to be of turbulent nature [2]. Moreover, seeded impurity will change the background plasma. Thus the turbulence transport may also be changed. This dynamic interaction between plasma and impurities is not taken into account in the aforementioned modeling practices.

Coupling the codes dedicate for boundary plasma modeling with a turbulence transport code in an iterative way offers a possible solution to the problems discussed above. This contribution presents the work of coupling SOLPS with HESEL, a well developed turbulence transport code [3]. In the coupling process, SOLPS provides HESEL with the background plasma profiles of ne, Te, Ti and other parameters required by HESEL. Then HESEL outputs radial fluxes, which will be passed to SOLPS as source terms. Both one-time exchange as well as iterative crosstalk between the two codes are possible and the respective results will be compared.

In this contribution, we first compare the diffusive coefficients derived from HESEL simulation of L-mode plasma to the *ad hoc* ones used in SOLPS modeling for the same discharge. The parametrized parallel loss terms in HESEL are compared to SOLPS calculations to give a reference of the credibility of parametrization in the parallel direction in 2D turbulence transport codes. Then the results obtained by coupling the two codes in iterative manner are compared to the previous modeling results of Ar seeding obtained with SOLPS alone, as well as to the experiments on ASDEX Upgrade. We investigate how well SOLPS modeling, now with turbulent transport integrated, reproduces the experiment. We are especially interested in seeing how including turbulence transport deviates the impurity/radiation distribution and divertor plasma solution from previous results to assess the importance of properly described radial transport in boundary plasma simulations.

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2D self-consistent modelling of a box-type liquid metal divertor for the DTT facility

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The DTT facility [1], which is currently undergoing preliminary design in Italy, will assess the effectiveness of alternative solutions for the steady-state power exhaust problem in future fusion reactors (e.g. DEMO). The present study concerns one among the strategies which are foreseen to be tested, namely the use of liquid metals (LM) as plasma-facing materials, where the possibility of testing a closed "box-type" divertor based on [2] is being considered among others [3]. That divertor, which could employ e.g. Li or Sn as the LM, foresees a LM evaporation chamber (EC), which is open towards a second divertor box, i.e., the differential chamber (DC), in turn connected to the main plasma chamber (MC).
A simple, 0D model for this type of divertor has been recently developed by our group [4], with simplifying assumptions concerning the LM recirculation and a basic treatment of the metal vapor-plasma interactions.
In this work, a new 2D CFD model of the LM vapor is developed in OpenFOAM and benchmarked against the abovementioned 0D treatment. Then, the model is extended in order to describe more accurately the effect of eroded LM (evaporated + sputtered) on the SOL plasma, based on [5], allowing to estimate 2D heat load profiles on the target and on the EC and DC walls. The resulting temperature distributions in the divertor walls are evaluated by means of a 2D FEM model and employed to evaluate evaporation and condensation rates, which in turn affect the SOL plasma behavior. In this way, through an iterative procedure, self-consistency is achieved. Thanks to this feature and to the relatively low computational effort required for a simulation, this model can be employed to support the pre-conceptual design of a closed LM divertor for DTT by providing information on the hotspot temperatures and on the LM recirculation requirements.

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Heat load and ELM control with impurity mixture SMBI seeding for ELMy H-mode plasmas in the HL-2A tokamak

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In standard high confinement regime (H-mode) of tokamak plasmas, the edge-localized mode (ELM) usually produces high transient heat loads on plasma facing components. Over the years, intensive effort has been dedicated to find an optimal technique for heat load and ELM control. Specifically, supersonic molecular beam injection (SMBI) [1], an effective fuelling tool has been first demonstrated for ELM mitigation in HL-2A [2], and then in EAST and KSTAR. In addition, impurity seeding has been used to mitigate ELMs in several devices. It can reduce the heat load by converting the heat into impurity seed radiation. Moreover, it can control the ELM activities by affecting the pedestal dynamics and instabilities.

On HL-2A, the impact of fuelling and impurity on pedestal dynamics and instabilities has been studied, recently [3]. It has been observed that a broadband electromagnetic (EM) turbulence can be excited by peaked impurity density profile at the edge plasma region, and governed by double critical gradients of the impurity density [4]. Recently, experiments have been performed in HL-2A with impurity SMBI, which can inject a mixture of gas including D_2 and light impurity gas (Ne or Ar, etc). The SMBI impurity seeding is beneficial for forming an edge radiation layer and preventing impurity core accumulation. It has been observed that with pure impurity SMBI injection, the H-mode plasma confinement has been improved. For mixture impurity seeding, the ELM behavior or plasma confinement varies with the ratio of the impurity gas to D_2. It has been found that large ELMs are mitigated to very small bursts with 30% Ne-SMBI seeding. Experiments in HL-2A seem to indicate that there should be an optimal impurity ratio for ELM and heat load control, and suggest that pedestal dynamics and heat loads can be actively controlled by exciting pedestal instabilities and forming a steady edge radiation layer.

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Divertor heat flux reduction by active and passive Li injection in EAST

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Reduction of divertor heat flux could decrease wall erosion and impurity production. Li has served as a low Z plasma facing material in several devices, and has been shown to reduce the peak heat flux on a liquid Li divertor[1].In EAST, Li evaporative coating by oven, Li powder injection, and flowing liquid Li limiter (FLiLi) have all been successively applied[2].In certain experiments, a bright Li radiative mantle appeared at the plasma edge during Li powder injection and FLiLi operation[3-4], which was accompanied by heat flux reduction. Here we compare active Li powder injection with passive Li evolution from FLiLi.

Li powder was actively injected into plasmas from the top of EAST at ~5×10^{21}atom/s (~70mg/s); in comparison, Li particle passive efflux from FLiLi into the plasma was estimated at >5×10^{20} atom s^{-1}, due to surface evaporation and sputtering. A few small Li droplets ~1 mm diameter were ejected from FLiLi due to the strong interaction between the liquid Li surface and plasma. These Li particles were ionized in the scrape-off layer(SOL) plasma, and transported toroidally to gradually form a bright Li radiative mantle with an obvious poloidally asymmetric non-uniform space distribution. Significantly, this strong Li radiation effectively reduced the divertor peak heat flux, which causes a ~15% temperature reduction of divertor strike point by Li passive injection from FLiLi surface compared to that ~45% by Li powder active injection. Overall the characteristics resemble those from heat flux reduction and detachment onset via low-Z impurity injection in many devices. Moreover, active and passive Li injections have successfully mitigated transient heat fluxes during ELMy H-mode in EAST.

In future fusion device, by applying various Li injection methods, the amount of Li passive injection into plasma from divertor liquid Li component could automatically self-regulate Li efflux into plasma, due to enhanced (reduced) emission at increased (reduced) surface temperature. By combining with active Li injection, both steady state and transient high heat fluxes could be mitigated.

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Wall Conditioning and Tritium Removal Techniques

Real-time wall conditioning by controlled injection of boron and boron-nitride powder in full tungsten wall ASDEX-Upgrade

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We report results from ASDEX-Upgrade (AUG) in which injection of boron nitride and pure boron powders in plasma discharges improved wall conditions in subsequent H-mode plasmas, using a newly developed powder injector designed to handle a variety of materials. Specifically, pure boron injection appeared to improve wall conditions qualitatively similar to boronization, while boron nitride injection was dominated by the effects of nitrogen, including confinement improvement as observed with nitrogen gas puffing.

In the full tungsten wall AUG, boronization, typically performed by injection of B_2D_6 gas in He glow discharges [1], has proven effective for control of W influx and access to low density and collisionality. The beneficial effect of boronization lasts for a few days up to two weeks of operation, after which a new boronization cycle is required for access to low collisionality. Here we attempted to improve wall conditions with injection of boron-rich compounds during plasma discharges. The experiments were enabled by a newly deployed powder injection system [2], specifically designed to handle a variety of materials, including viscous powders that naturally form conglomerates and obstruct flow. Powders were delivered gravitationally into lower single null, H-mode plasmas (I_p=800 kA, P_{NBI}=10 MW, n_e=6x10^{19} m^{-3}) through a vertical guide tube installed on the top of AUG, at approximately constant rates for intervals of 1-4 s. The plasma shape was optimized with small but poloidally constant outer gap to favor redeposition of ablated powder onto the W limiters.

Ablation of powder particles occurred mostly at the injection location, as indicated in wide-angle discharge imaging. A fraction of particles was observed to migrate on approximately field-aligned trajectories. Core impurity line emission from charge exchange recombination spectroscopy increased with impurity injection rate, as expected. During B powder injection, radiative losses increased by up to 50%, with only a minor effect on plasma characteristics. Visible spectroscopy indicated an increase of B emission at the limiters and in the divertor, during B (and also BN) injection, suggesting deposition at these locations. Increased W emission was also observed in certain cases. An increase of energy confinement of 10-20% was also observed with BN injection, which is attributed to the improved pedestal stability associated with N injection [3]. These preliminary results suggest that injection of B and/or BN powder in dedicated wall-conditioning plasmas can be used to partially restore and possibly extend the beneficial effects of a boronization in AUG.
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Designing high efficiency glow discharge cleaning systems

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Glow Discharge Cleaning (GDC) is one of the best established techniques used in present fusion devices in order to remove deuterium and impurities from the plasma facing components. At least in its simplest form, with Direct Current (DC) power supply, GDC operation is not compatible with the presence of a magnetic field in the vessel volume, that inhibits the discharge, hence in superconducting coils devices DC-GDC cannot be used for inter-shot conditioning; nevertheless DC-GDC is implemented also in ITER, as fundamental tool for wall conditioning after maintenance with vessel venting and for tritium recovering after D-T campaigns. The effectiveness of the wall conditioning is closely connected to the ion current impinging on its surface, given that in the process the wall plays the role of discharge cathode and undergoes ion sputtering. It was already found [1,2] that in toroidal fusion devices, where the cathode area is typically 2-4 order of magnitudes larger than the anode area, the current does not spread uniformly over the wall, hence cleaning efficiency is not everywhere the same. The mitigation of such non-uniformity pattern is a key issue in order to get good conditioning of the plasma facing wall.

In the present work we report about dedicated experiments conducted at RFX-mod toroidal fusion device, devoted to enlighten the effect of different anodes configuration on the profile of ion current density at the wall, j_{wall}. RFX-mod (R=2m and a=0.459m) is very well suited for the purpose since its wall, the cathode, is circularly shaped and uniformly covered by tiles, without discrete limiters or any other protruding structure: the regular shape of the plasma boundary gives high sensitivity in detecting the effect of any parameter on the j_{wall} profiles and allows a direct comparison of the measures to models, based necessarily on simplified geometry. Moreover, RFX-mod is equipped with an extensive set of electrostatic probes, organized in a uniformly distributed toroidal array of 72 probes and a poloidal one of 7 probes, for the full reconstruction of the ion current profile. The j_{wall} pattern has been measured with either one or two working anodes, with anodes of different size and material and with anodes placed at different insertions with respect to the wall, from the center of the poloidal section down to the wall edge. A working pressure scan was also performed. The role of the anode area has been quantified and j_{wall} non-uniformity has been characterized in terms of ratio of anode and cathode areas. Anode discretization and size were also found to impact on the V-I characteristic curve of the discharge. Experimental results shown here were found to be in good agreement with models of glow plasmas found in literature and constitute the basis of the project for a new GDC system for the RFX-mod machine upgrade.

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Development and optimization of He Electron Cyclotron Resonance Heating and He Glow Discharge wall conditioning scenarios for W7-X.

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The operation of the superconducting stellarator W7-X [1] demands to control the surface state of the plasma-facing-components (PFC) made of carbon-based material by means of wall conditioning methods [2]. The results of the first operational campaign in limiter configuration showed that temporary depletion of the wall from H and impurities was achieved by He glow discharge cleaning (He-GDC) between experimental days. The recycling coefficient could be reduced by ~ 25 % owing to dedicated Electron Cyclotron Resonance Heating (ECRH) conditioning plasmas in He during the experimental day, both leading to a significant extension of the discharge duration (up to 20 s) [3]. The conditioning schemes have been continuously optimized throughout the research program in divertor configuration of W7-X. In particular, it was found that maximizing the removal rate of H in He-GDC, through the impact energy increase, causes an increased sputtering yield of first wall claddings. He-ECRH conditioning procedures have been developed to sustain a high H discharge repetition rate at best plasma performance (i.e. high and controlled density) and recovering from frequent radiative collapses. Namely, both reasons induced to improve the He-ECRH cleaning discharges or to investigate alternatives. One of the most promising strategies which has been successfully tested on W7-X is sequences of 10 short He-ECRH discharges with a high duty cycle of 1.5s/30s that is able to increase H depletion of PFC significantly [4].

The results of the He-GDC and He-ECRH wall conditioning optimization on W7-X are presented. The suitable GDC parameters, i.e. voltage, anode current and gas pressure, are defined to keep the balance between maximum possible hydrogen removal rate by He-GDC and minimum PFC erosion. The efficiency of single He-ECRH cleaning pulses and sequences of He-ECRH discharges are compared. The effect of varying the main parameters of He-ECRH cleaning pulses such as gas prefill, input power, pulse length are described. Additionally, it is shown that slight changes of the magnetic field configuration and modulation of injected ECRH power during the He-ECRH discharge, also allow to change the balance between retention and removal of H from the PFC. The results of developed cleaning schemes implementation show significant grow of the wall cleaning efficiency. Further improvement of wall conditioning methods can also contribute to better plasma performance for long-pulse operation during the actively cooled divertor phase of W7-X.

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In-operando observation of helium ion effects on deuterium retention in lithium films on tungsten substrates

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The plasma surface interaction within a fusion device is one of the limiting factors to long pulse, power generating operation. A plasma facing component material will require effective heat tolerance, minimal erosion yield, and minimal fuel retention properties. Tungsten (W) has been selected as the divertor material for the International Thermonuclear Experimental Reactor (ITER) because it has a high thermal conductivity and high sputter threshold. However, when W is exposed to high particle flux (>10^{26} $cm^{-2}s^{-1}$) at high surface temperatures (>600ºC), the surface will develop defects such as pits, blisters, and nano-structured tendrils. These morphology changes reduce the beneficial properties of W. To overcome this limitation, a more radiation tolerant thin film material could be used, such as lithium (Li). In multiple tokamak devices, Li, as a wall coating, has improved the plasma performance by reducing fuel recycling from the walls, which stabilized the edge plasma and decreased the number of edge localized modes (ELMs). Because ELMs help eject impurities from the core plasma, the complete suppression of ELMs is detrimental. Methods to regulate the frequency of ELMs have been investigated using gas puffs [1]. Another method to control the ELM frequency is to control the fuel recycling via helium (He) ions. In Li films, the deuterium (D) is retained through a chemical relationship between Li, O (oxygen), and D [2]. Previous work has shown that when He ions are introduced with D ions, in a dual beam irradiation of Li films on W, a reduction in the D retention is observed [3]. To further investigate this phenomenon, 500 nm films of Li on W were exposed to both simultaneous and sequential irradiations of D and He. The simultaneous experiments used He concentrations of 1, 5, and 10% up to a D fluence of $1x10^{17}$ $cm^{-2}s^{-1}$. For sequential irradiations, the He fluence was ≈5% of the D ($1x10^{17}$ $cm^{-2}s^{-1}$). The energies for the He and D ions were 1000 eV and 250 eV/amu, respectively and samples were exposed at room temperature. The surface chemistry was characterized with x-ray photoelectron spectroscopy (XPS) to determine changes in retention. The XPS scans occurred *in-situ* and *in-operando* for the simultaneous and sequential irradiations, respectively. The simultaneous irradiations show a decrease in retention close to a factor of 6 for the 1 and 5% He cases, while for the 10% case, the retention drops by only 2x. The sequential experiments showed a decrease in the retention when He follows D ions and little change in the retention when D follows He. This may indicate that He breaks the D retention mechanism in Li.

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Surface conditioning of ASDEX Upgrade with tungsten plasma facing components

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For reliable plasma operation, conditioning of the plasma facing components (PFCs) is essential. Transition from carbon to tungsten PFCs changes the requirements on wall conditioning significantly. For instance, glow discharge cleaning (GDC) between plasma discharges was necessary with carbon PFCs and, therefore, routinely applied in ASDEX Upgrade (AUG) [1], whereas GDC is intentionally omitted for tungsten PFCs. The actual procedure for wall conditioning starts during a vessel opening with wet wiping to remove layers from previous experiments. The next step is to bake at 150 ^{o}C during pump down to reduce the water content of the installed tungsten coated carbon tiles. Finally, He GDC is applied to enable the start of tokamak plasma operation. Using high deuterium gas fluxes and a typical heating power of 10 MW, the W-PFCs are conditioned for normal plasma operation within 2 experimental days allowing routine operation of AUG [2].
Scenarios which require low collisionality plasmas may be affected by large W influxes resulting in tungsten accumulation. To reduce the W source, a boronization using 90 % He 10 % B_2D_6 mixture is applied. Earlier investigations have shown that a major W source is sputtering during ELMs at the ICRF antenna limiters [3]. Boronization covers these limiters for, typically, 30 discharges. Further Boron layers on top of deposits reduce the low-Z impurity content, which plays an important role for the W sputtering. Typically a new boronization is applied after 200 discharges. As the boronization allows more flexible operation, it is now routinely applied again.
GDC is mostly used after warm-up of the cryo pump or massive gas injection for disruption mitigation. He GD implants gas in W PFCs, which reduces the plasma performance [4]. To reduce the wall implantation, pulsed GDC either in D_2 or He is used. This technique reduces re-ionization of released ions, i.e. enhances the pumping effectivity, and reduces the implantation of He. The flux onto PFCs is reduced by a factor of 5, but the cleaning effectivity is almost preserved in comparison to continuous GD. A newly installed roughing pump allows continuous operation on an experimental day without regenerating the LHe panel of the in-vessel cryopump, which prevents reloading of the PFCs during regeneration. The sum of these techniques together with the upgrade of the ECRH power enables operation at low densities, which is, for instance, required to develop ELM control scenarios and current drive experiments. An attempt to develop a robust criterion on the status of the boronization using standard signals is presented.

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Can tritium be extracted from a flowing lithium divertor system fast enough?

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As lithium has grown in popularity as a plasma-facing material, efforts have been placed on examining its viability as a first wall candidate. Lithium has proven over numerous studies to improve core confinement, while allowing access to operational regimes previously unattainable while using solid, high-Z divertor and limiter modules [1, 2]. These benefits are due to the fuel retention capabilities of lithium, which allow it to be an almost ideally absorbing boundary, which is both beneficial and problematic. As such, extraction technologies must be constructed and verified within the scope of a larger scale lithium loop system that separates lithium impurities, recovers deuterium and tritium, and recycles clean liquid lithium back to the plasma-material interface.

Recent studies conducted at the University of Illinois Center for Plasma-Material Interactions indicate the rates for isotope recovery will be able to balance the wall loss rates as described by Krasheninnikov, et al. [1]. Maximum recovery rates, however, are greatly dependent on the atomic fraction of hydrogen isotope present within a given volume of lithium. Looking at the extreme case of pure lithium hydride (LiH), having the highest available hydrogen atomic fraction, has shown that LiH will decompose and evolve molecular hydrogen at a peak rate of $(9.6 \pm 1.2) \times 10^{17}$ s^{-1} at 696 °C. From a thermodynamics standpoint, the activation energy for this process was found to be 95.7 ± 2.4 kJ mol^{-1}, which differs from the magnitude of the enthalpy of formation for LiH at 90.7 kJ mol^{-1} by 6% [3]. A differential lithium element in a LiMIT-style plasma-facing component [4] will be subjected to a dose of slightly greater than 1×10^{17} deuterium and tritium ions per trench per pass, meaning the isotope atomic fraction per pass will be well below saturation. Related studies, investigating absorption and recovery as a function of exposure conditions, have shown that solid lithium samples with masses of approximately 0.11 ± 0.05 g display evidence of saturation at plasma fluences of nearly $(1.4 \pm 1.0) \times 10^{21}$ m^{-2}. Similar experiments performed on liquid samples of the same mass show that the energetics of the ions and radicals in a plasma influence absorption above and beyond the simple thermodynamics associated with neutral exposure. There also exists a non-monotonic functionality in amount absorbed versus the sample temperature. The sum of these findings culminated in the development, construction, and testing of a prototype distillation column along with a design for a supplementary recovery technique. These results and the path forward for future hydrogen recovery systems will be discussed.

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Active wall conditioning for long pulse plasma by using lithium powder injection in EAST with tungsten divertor

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Tungsten is a promising plasma-facing material for use in future fusion devices. However, tungsten has previously been observed to accumulate in the plasma core of fusion devices, which impedes achievement of high-power, long-pulse H-modes[1]. Wall coatings for metal wall with different material has been demonstrated to improve overall plasma performance[2]. However, this coating technique may not extrapolate to future long-pulse devices, since the coatings will be eroded by the continual plasma flux onto the surface.

In order to provide active wall conditioning and control high Z impurities, a new technology has been developed that is capable of injecting Li powder into the scrape-off layer(SOL) plasma during plasma discharges, where it quickly liquefies and turns into an aerosol. The real-time injection of Li powder into 40 s long H-mode discharges is effective to suppress impurity influx and control recycling on EAST with an ITER-like tungsten divertor. With Li powder injection, stable profiles of tungsten are sustained both in the core and edge, while the concentrations are halved compared to the normal ELMy H-mode discharge, suggesting a reduced tungsten source. During Li injection, several effects play a role in the suppression of tungsten influx. These effects include a reduced SOL temperature and a reduction of the heat flux on the tungsten divertor and hence reduced erosion source, mitigation of ELM size[3], and impurity segregation via deposition of a Li film on plasma-facing surfaces.

Real-time Li injection has also been applied to enhance the particle recycling control on EAST. Quantitative analysis of the recycling changes using the SOLPS edge plasma and neutral transport code indicated a ~20% reduction in recycling coefficient with Li injection. This effect is strongest in the active tungsten divertor, confirming the Li is transported to strongly plasma-wetted areas.

This research indicates that active Li, or possibly boron or another low-Z material, powder injection could be an effective method to enhance the particle recycling control and suppress tungsten impurity influx in future steady-state fusion devices.

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Abstract Withdrawn

Wall conditioning throughout the first carbon divertor campaign on Wendelstein 7-X.

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Wall conditioning is commonly applied in magnetic controlled fusion devices to improve plasma performance and discharge reproducibility [1]. The super-conducting stellarator W7-X keeps its magnetic field charged during the experimental days. Therefore, conventional wall conditioning by glow discharges (GDC) cannot be routinely used. Overall, the usage of He-GDC operation was limited throughout the first W7-X campaign in divertor configuration (OP1.2a) to avoid sputtering of plasma facing components as evidenced by passive spectroscopy. Alternatively, Radio Frequency based wall conditioning scenarios are foreseen. The multi-megawatt Electron Cyclotron Resonance Heating (ECRH) system of W7-X [2] is well suited for this purpose, since in a stellarator the confining magnetic field already exists in the vacuum; the confined ECRH plasma is sufficiently dense and hot to provide good absorption of ECRH power minimizing stray radiation. Conditioning by boronisation and RF discharges in the ion cyclotron range of frequencies will become available in the next operation campaign with actively cooled divertor.

This contribution analyses the initial conditioning cycle by baking and GDC before plasma operation, and the contributions of GDC and ECRH discharges to the performance improvement throughout the OP1.2a campaign. One week of baking at 150°C after machine venting removed about 10^{25} H_2O molecules and suppressed the footprint of higher hydrocarbons in mass spectrometry measurements. The remaining impurities, mainly CH_4 and CO could be reduced by x10 within 3 cumulated hours of H_2-GDC. Mass spectrometry time traces of subsequent H_2-GDC's operated before the first ECRH plasma as well as throughout the campaign follow one continuous decaying trend in time ($t^{-0.7}$) suggesting a modest removal contribution by He-GDC and He/H_2-ECRH operation. He-GDC effectively but temporally desaturates the walls from hydrogen, hydrocarbons and carbon oxides. Whereas CH_4 and CO dominated the outgassing in He and H_2 ECRH discharges at the start of the campaign; typical H recycling for C-devices was obtained towards the end of the campaign. The limited efficiency of plasma core fueling makes controlling the fuel recycling at the plasma facing components particularly important and challenging. Recycling conditions below unity are retrieved by He-ECRH discharges or wall fueled (H_2) discharges. Sequences of repetitive He-ECRH discharges with pre-defined duty cycle are found to be the most (time-) efficient tool presently available for providing reproducible recycling conditions in high density (>$4 \times 10^{19} m^{-3}$) and long pulse H-operation (>10s) with density control.

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Laser Induced Desorption of co-deposited Deuterium in Beryllium Layers on Tungsten

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In fusion devices with beryllium (Be) and tungsten (W) as plasma-facing materials like the JET-ILW, the major contribution to the long-term hydrogen retention is found in co-deposited Be layers on W divertor components [1]. In ITER, a comparable behavior with the formation of hydrogen isotope containing Be layers is expected where the tritium (T) retention is limited by the nuclear license. To study and monitor the spatial distribution of the H, D and T content of the wall without tile removal, Laser Induced Desorption (LID) is being designed as fuel retention diagnostic detecting released hydrogen isotopes either spectroscopically during plasma operation [2] or between plasma phases with the aid of mass spectrometry [3].
During the LID process a wall area of 3 mm in diameter is heated up by laser pulses to high surface temperatures, even close to melting. The retained gases diffuse through the material and will be desorbed. However, up to now only a fraction of the retained hydrogenic fuel could be released by ms laser pulses, e.g. utilizing a 10 ms laser pulse [4]. Complete fuel desorption has only been observed at the Be melting temperature utilizing 30 ns laser pulses [5]. According to numerical desorption calculations for Be [3] optimised laser pulse parameters (laser intensity, pulse duration, frequency and number of pulses) should increase the desorption fraction significantly, but an experimental verification is required. Therefore, the influence of different laser pulse parameters on the desorption efficiency of LID is studied here on homogeneous Be layers with different deuterium content and layer thickness.
Layers of 1 and 10 μm thickness have been deposited on polished W substrates (Plansee, IGP grade) by HiPIMS (High Power Impulse Magnetron Sputtering) in INFLPR. During the sputtering process deuterium (D) was co-deposited with the beryllium in two different concentrations of either 0.01 or 0.1 D/Be. Hence, the optimal LID parameters for different Be layer thickness and deuterium concentration can be investigated in a multi-parameter space of sample parameters and laser parameters with up to 500 W time-averaged power and pulse durations between 0.1 and 20 ms. The rapid laser heating is performed in the Fuel REtention DIagnostic System (FREDIS), where for the first time LID is compared to slow Thermal Desorption Spectrometry (TDS) in a combined device using the same quadrupole mass spectrometers. This comparison permits to determine the desorption efficiency of LID for different parameters.

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Micro-structured tungsten; an advanced plasma-facing component

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Tungsten (W) is considered as the most promising plasma-facing material (PFM) in the area of high flux exposition like in the divertor of ITER and in upcoming fusion devices. Recently, several investigations have been done on extruded or laminated tungsten such as thin foils and fibers. These highly strained tungstens, address a major flaw of bulk tungsten, which is its brittleness (more specifically, the laminating process lowers the ductile to brittle transition temperature (DBTT) of W based materials) and provides some plasticity to W (owing to the resulting grain structure [1]. The achieved benefits are exploited in tungsten fiber composites (WfW) [2].

This experimental work addresses a slightly different approach, using the same base material (laminate/extruded tungsten), but not to improve tungsten structural properties as such, but its direct performance as plasma facing components (PFC). For this purpose, the base W material is made out of 2000 to 5000 fibers with 240 and 150 µm diameter, respectively, or 56 µm thick and 2 mm wide foil roll. Fibers have been assembled into compact cylindrical piles with hexagonal lattice while the foil has been rolled into a 1 m length spiral, both as $10\times10 mm^2$ samples, having no interlayer matrix between laminate W components. Several benefits are expected relative to plain/bulk tungsten as PFM: i) mitigation the thermal fatigue effects, by allowing much easier thermal expansion and consequentially reducing the thermal stress accumulated after thermal cycling, ii) a similar or reduced fuel retention due to its high surface-to-volume ratio of the structure easing desorption less limited by diffusion, while having a minimal tradeoff relative to sputtering thanks to its 90% density (compared to bulk W), iii) a better power handling by increasing emissivity [3]. These PFC have been exposed in the linear plasma facility PSI-2 to neon plasma (95 eV impact energy, 500K, 5.4×10^{25} Ne^+ m^{-2} fluence, 100 min) and afterwards to deuterium plasma (51 eV impact energy, 450K, 5.1×10^{25} D^+ m^{-2} fluence, 150 min) to evaluate the influence of the microstructure on sputtering yield and on fuel retention. The thermal fatigue effect was investigated by having a combined steady state plasma loading on the PFC and transient laser pulses (1000 pulses, 3 mm spot diameter, 23 J, 0.5 Hz) to simulate ELM cycling fatigue with an intensity of 0.38 GW/m^2. Solid tungsten samples were also exposed in all measurements providing a clear reference.

Post-characterization of exposed novel and reference materials included several methods. Material losses between 3.83 and 5.21 mg were measured (about 78% of expected theoretical sputtering of Neon at such energies) and related to scanning electron microscopy (SEM) sputtering evaluations and surface morphology changes. D depth NRA profiling up to 3.6 µm showed similar surface retention between 4.7 and 5.4×10^{20} D/m^2 while the total amount released by thermal desorption spectroscopy (TDS) showed also very similar retention values between 1.51 and 3.66×10^{21} D/m^2.

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