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Modeling of tungsten transport in the SOL for sources at the central column of ASDEX Upgrade using DIVIMP

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Abstract

Tungsten erosion at the central column and tungsten transport in the scrape-off layer of ASDEX Upgrade have been modeled with the 2D Monte-Carlo impurity transport code DIVIMP. The main objectives were the reproduction of the erosion and migration patterns that are observed experimentally. This was accomplished by implementing a new erosion routine which allows to calculate the tungsten source term at the central column and subsequently the poloidal distribution of the W density. First results on W transport in the boundary plasma as well as erosion and migration patterns of W are discussed in relation to different models of the plasma in front of the central column. A comparison between measured and modeled deposition of W in the divertor already shows satisfactory qualitative agreement.

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1. Introduction

In recent years there has been a renewed interest in the use of tungsten as plasma facing material in fusion reactor experiments. This is mainly motivated by the large erosion rates of low-*Z* materials as Be or C and the problem of tritium codeposition with these materials.

Motivated by positive results during the W divertor experiment [1], increasing sections of the central column, the so-called heat shield (see Fig. 1), in the ASDEX Upgrade main chamber have been equipped with W coated graphite tiles starting with 1.2 m^2 at the lower heat shield during the first stage to an almost complete cover of 7.1 m^2 representing ≈1/6 of the whole plasma surface in the most recent campaign 2001/2002 [2]. The results look promising since up to the latest stage no adverse effect on the plasma performance could be observed as long as the heat shield is not used as a limiter for a long time. The W concentration mostly remained at least one order of magnitude below the acceptable maximum for ASDEX Upgrade.

The investigations on the use of W in the main chamber have been recently supplemented by modeling computations using the 2D impurity transport code DIVIMP [3], in order to allow a better interpretation of the experimental results. DIVIMP treats trace impurities on a background plasma, which at ASDEX Upgrade usually is provided by B2/EIRENE calculations. Since the area of direct plasma ion–wall interaction in DIVIMP is restricted to the divertor where the flux surfaces intersect the wall, DIVIMP had to be extended in order to accommodate for the tungsten source at the heat shield.

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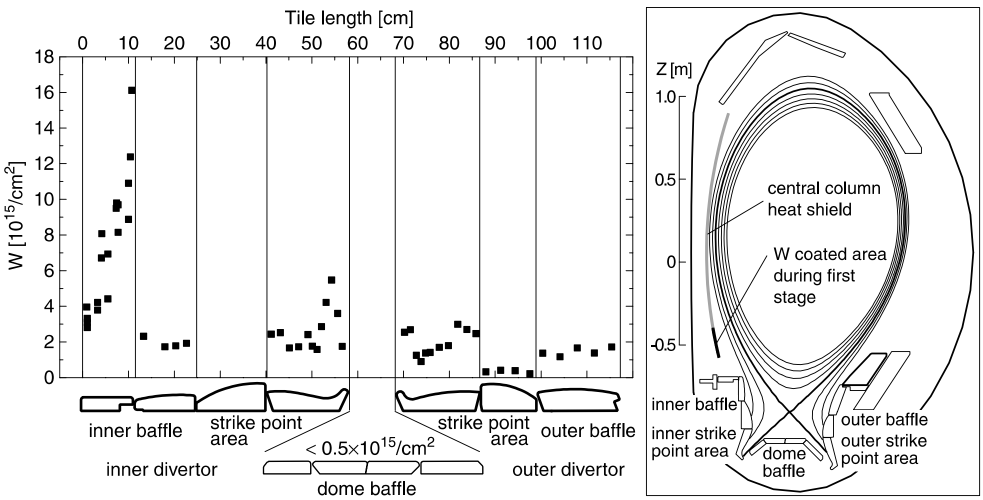

Fig. 1. Measurement of deposited W in the divertor of ASDEX Upgrade after the first stage with W tiles only at the lowest tile rings. One can see a strong deposition at the inner baffle and the inner strike point position. There is also some W deposited at the outer divertor.

2. Summary of experimental results

The influx of W from the heat shield was measured spectroscopically with an array of optical fibers, each monitoring a different poloidal position and observing the prominent WI line at 400.9 nm. However, the WI emission could only be detected in dedicated experiments where the plasma column was shifted against the central column. The influx in this case was $\approx 2.5 \times 10^{14}$ $m^{-2}\,s^{-1}$ from the tile that was closest to the separatrix. Therefore one would assume, that only a narrow ring contributes significantly to the influx.

After the experimental campaigns, the erosion of the W coated carbon tiles as well as the migration of tungsten onto the uncoated tiles, e.g. in the divertor, was investigated by ion-beam surface analysis. The detailed results are presented in [4]. A maximum erosion rate of 22 pm/s is found at the top of the heat shield and is much larger than the erosion rates due to sputtering by CX neutrals. This, as well as the observed shading effects, indicate a direct plasma–wall contact at the heat shield. The poloidal variation of the erosion can be attributed to the intersection pattern of the magnetic filed lines with the tile surfaces.

Measurements from Langmuir probes at the central column also show the presence of plasma directly in front of the heat shield. Estimated erosion rates are especially large during ramp-up and ramp-down when the heat shield is used as limiter. Depending on the poloidal distribution of the redeposited W during ramp-up and ramp-down, the measured erosion and deposition patterns of W might be dominated by these phases, yet one would expect the bulk of the eroded W to be redeposited on the heat shield again. Hence one has to keep in mind, that the model presented here only treats the flat-top phase of a discharge which, however, corresponds to the operation mode of a future steady state fusion reactor.

The areal density of W in the divertor measured during the first stage of the W coated heat shield is shown in Fig. 1. During this stage only the two lowest tile rings were coated with W, which means that there was no direct plasma–wall contact of this area during ramp-up or ramp-down. Thus the major part found in the divertor is eroded during the flat-top divertor configuration.

One can see that the deposition area is concentrated onto the inboard horizontal baffle and the strike point region, but also on the outboard side vertical baffle and strike point modules. The latter might be an indication of a certain W transport across the stagnation point at the top of the plasma. The observations suggest that W eroded from the heat shield is transported to the divertor rather quickly within the outer scrape-off layer (SOL).

However, when comparing these measurements with modeling results, one has to keep in mind, that all measurements are integrated over a whole experimental campaign and that significant deviations from the average might be found depending on the considered discharge type.

3. Modeling of W erosion and transport

Initially DIVIMP was designed to model erosion and transport of carbon in the divertor region and the SOL of tokamaks. Therefore in most versions of the code the plasma-ion–wall interaction is restricted to two areas, the inner and outer divertor. Erosion at the main chamber wall can only be calculated with neutral hydrogen fluxes taken from EIRENE simulations. For the calculation of the erosion source term at the central column, a simple plasma model was implemented where the plasma parameters are extrapolated from the outermost grid ring to corresponding points of the wall assuming an exponential decay of the electron density n_e and electron and ion temperatures, T_e and T_i, with a decay length λ. Alternative functional behaviors have been implemented as well. The decay length λ was taken from Thomson scattering measurements [6] at the low-field side plasma boundary of ASDEX Upgrade. The flux tube expansion between low- and high-field side plasma boundary suggests a $\sim$70% larger decay length which would lead to plasma parameters at the wall closer to the ones in the other case presented, where the plasma parameters in the outermost grid ring were directly mapped to the wall. The two cases should be considered as extreme limits until better experimental data become available, allowing a more realistic selection of the extrapolation function.

Ionization and recombination cross sections for W are taken from a modified ADPAK data base [7], the atomic data for line radiation is taken from measurements at the Berlin Plasma Simulator [8].

For the modeling of the W erosion source only physical sputtering needs to be taken into account. The sputtering data are taken from measurements during the W divertor experiment [5]. These effective sputtering yields are based on the findings, that the main erosion is caused by carbon impurities in the background plasma and differ significantly from the ones of pure deuterium. The sputtered W is then fed back into the appropriate outer grid cells analogous to the case of wall erosion by EIRENE generated neutrals. Due to the distance between grid and wall, prompt redeposition which might contribute significantly to the small SOL penetration probabilities is not included.

The discharge from which the background plasma data was taken is an L-mode hydrogen discharge # 11275, which was very well diagnosed at the edge and then analyzed thoroughly with B2/EIRENE. This discharge, of course, does not represent a typical discharge in ASDEX Upgrade but represents the best background plasma data presently available.

3.1. Results from erosion and transport modeling

Two computations have been performed for a poloidally fully W coated central column, comparing the case with the plasma parameters in the outermost grid ring directly mapped to the wall with the case assuming an exponential decay of temperature and density to the wall with a decay length of $\lambda = 80$ mm. For both cases the densities, temperatures and fluxes at the wall and in the outermost grid ring as well as the sputtering data are shown in Fig. 2. Thereby the minimum distance between the grid and the wall is 20 mm at the center, the distances at the top and bottom are about 50 mm each.

In the cells corresponding to the central column the density at the outermost grid ring is relatively constant while the temperatures and accordingly the hydrogen

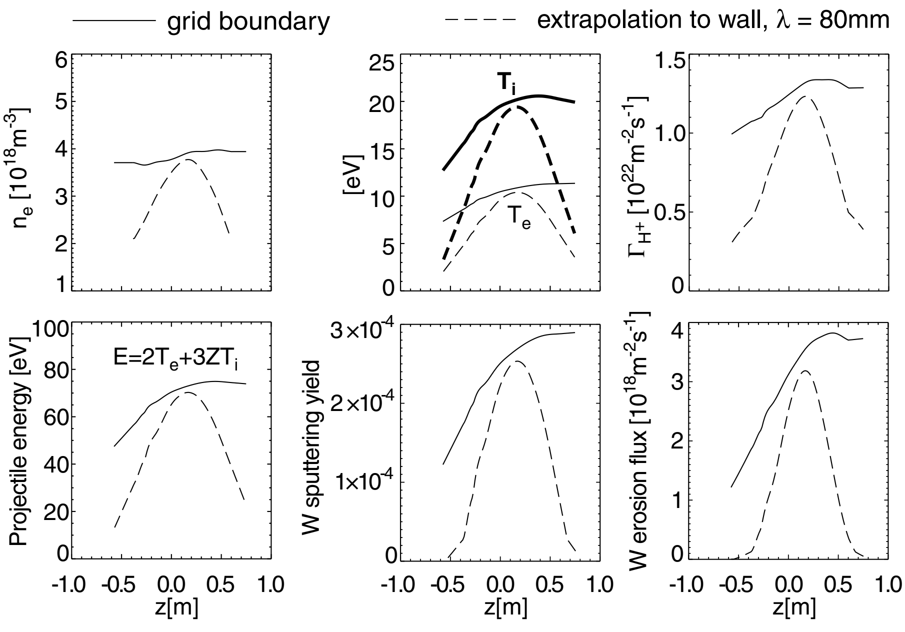

Fig. 2. Plasma parameters and sputtering data along the heat shield wall for the two cases of no decay and assuming an exponential decay of temperature and density to the wall with an decay length of $\lambda = 80$ mm. The definition of the abscissa z can be seen in Fig. 1.

flux are lower at the bottom. According to the decay length in the second case these parameters drop roughly by a factor two towards the top and the bottom of the heat shield.

In the first case the maximum erosion is found at the top of the central column while in the second case the maximum is located at the central heat shield. Accordingly the sputtered W flux for the first case agrees better with the experimental erosion pattern while the second case better fits the spectroscopic measurements. Thus for further simulations, the plasma parameters at the wall will be adjusted to Langmuir probe measurements expected to become available in the near future.

A poloidal cross section showing the distribution of the resulting density of some ionization states is shown in Fig. 3. While the lower ionization states are more concentrated around the source location and also in the divertor, the higher ionization stages are located more at the top of the plasma and within the separatrix.

3.2. Migration investigations

For the first stage of the W coated heat shield, calculations have been performed, to compare the modeled deposition pattern with the measured one described in Section 2. Since, as already mentioned above, there is no contact between the plasma and the tiles that were coated during this stage during ramp-up and ramp-down all the deposited W was eroded in the flat-top phase.

The poloidal distribution of the areal density of deposited W for this case is shown in Fig. 4. In this case, the WI line intensity, used for the determination of the W erosion flux, was below the spectroscopic detection limit. Thus a constant flux density from the tiles of $\Gamma_W = 10^{14}\ m^{-2}\,s^{-1}$, corresponding to the detection limit, was assumed.

The overall distribution of the modeled deposition agrees quite well with the measured deposition pattern, however with significant discrepancies in the small scale details. One has, however, to keep in mind the constrictions comparing the presented simulations with campaign integrated measurements. Nevertheless the strong deposition at the inner baffle and the inner strike point are well reproduced. This behaviour might be a consequence of the strong drag force on the W particles towards the divertor caused by friction with the plasma background, due to the plasma flow in front of the heat shield. This effect, in reality also present outside the computation grid, might be the reason that the

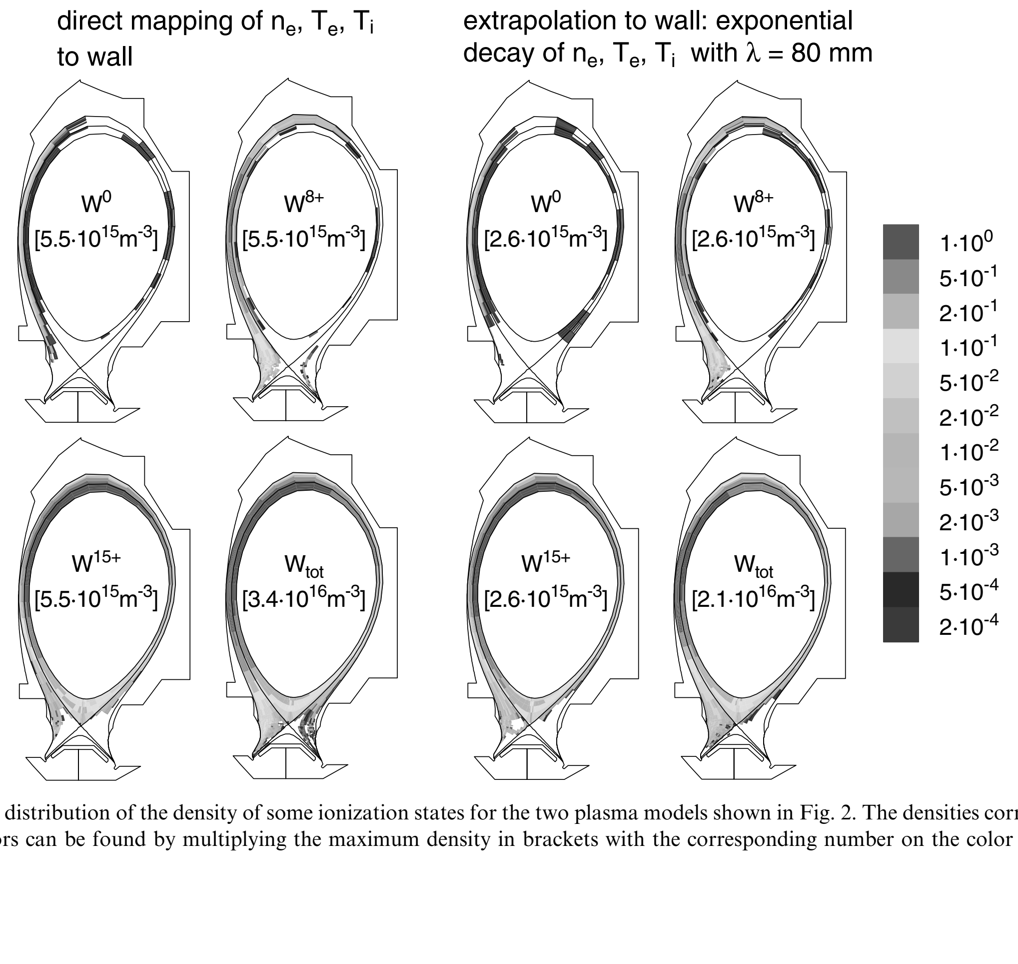

Fig. 3. 2D distribution of the density of some ionization states for the two plasma models shown in Fig. 2. The densities corresponding to the colors can be found by multiplying the maximum density in brackets with the corresponding number on the color bar.

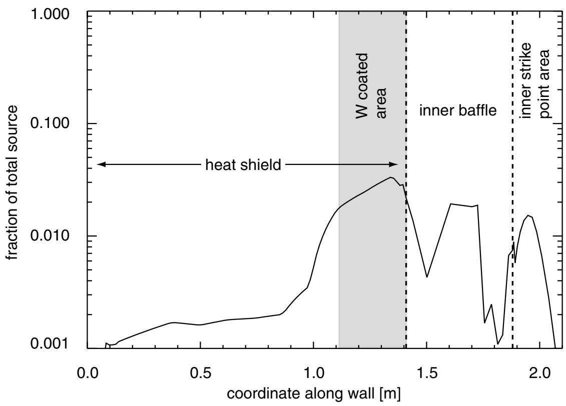

Fig. 4. Deposited W on the heat shield, the inner baffle and the inner strike point region as modeled with the DIVIMP code for the first phase of W coated tiles.

measured deposition on the horizontal part of the inner baffle is larger than the modeled one.

Another consequence of these drag forces is the fast decay of the W deposition towards the top of the heat shield, which is also seen in the experiment.

On the other hand the deposition at the outer divertor could not be reproduced with the code.

In cases where the W source is located more at the top of the heat shield, there is also some deposition of W in the upper divertor which is partly confirmed by the measurements at the inner upper divertor plate.

The code also predicts W deposition on the outboard wall, which could not yet be confirmed experimentally.

4. Conclusions and outlook

The usefulness of DIVIMP for the modeling of W erosion and transport in the SOL and the divertor of fusion plasmas has been demonstrated. Due to the early status of the work, however, the results should only be considered as a proof of concept.

The model of the plasma in front of the wall causing the erosion is flexible enough to describe the measured erosion but requires additional input data from the Langmuir probes at the central column.

The comparisons between code results and measurements already show some agreement taking into account the restrictions that apply for both. An important issue which needs to be addressed in the future is the migration of W during the ramp-up and ramp-down phases where the erosion of the W layer can become excessively strong.

Results from the W divertor experiment indicate an important contribution to the transport due to friction between carbon and W [9]. This effect needs also to be implemented.

New sputtering data for the simultaneous bombardment of W with deuterium and carbon have also become available [10] and will be implemented into the DIVIMP code.

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