Investigations on Tungsten Walls at ASDEX Upgrade

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In future fusion devices, tungsten (W) is a main candidate for being a first-wall material due to its low erosion rate and high melting point. However, when using W in a reactor, the central W-concentrations must be below 10^{-4} to avoid unduly large radiation losses. At ASDEX Upgrade, 85 % ($35.9\,m^2$) of the plasma-facing components (PFCs) in the 2005/2006 campaign are coated by W, while in the next campaign 100 % are envisaged [1]. In the actual campaign, special focus was put on the antenna limiters of the ion cyclotron resonance heating (ICRH) and guard limiters (both located at the low-field side (LFS)), as these have been coated with W recently and a considerable contribution to the W-contamination of the plasma has been expected. But also the decrease of carbon (C) densities following the reduction of graphite PFCs has been monitored closely, for better understanding the transport and migration of C.

Importance of Tungsten as Plasma Impurity

Fig. 1 shows the history of W-concentrations of ASDEX Upgrade along with the total area of W coated PFCs (top). The black squares show the full data set of an automatically produced database, which includes plasma phases with slow or no change in stored energy. The W-concentrations are derived from VUV emissions originating from a plasma region with $T_e \approx 1-1.5\,keV$ being typically at the edge of the confined plasma. The W-concentration at this region is not influenced by the central impurity transport, which may lead to impurity accumulation, an unfavourable state with an extremely peaked impurity profile. The issue of avoidance of accumulation will be discussed below. The full dataset exhibits a large scatter, as various plasma scenarios, different conditioning of the walls and diagnostic imperfections are naturally visible in such a large dataset. During impurity accumulation, which is identified by the peaking of the radiation profile, a hollow temperature profile may occur. This leads to a central plasma with $T_e \approx 1-1.5\,keV$ and the measurement position is shifted towards the plasma center, where a high W-concentration exists in a small fraction (typically 5 %) of the plasma volume. The red squares in fig 1. exclude impurity accumulation phases. For high density discharges without impurity accumulation and with sufficient heating power to reach $T_{e,0} \geq 2.5\,keV$ in the plasma center (blue squares), the W-concentration slowly increases with increasing surface of W-PFCs and reaches about $1 \cdot 10^{-5}$ in the recent campaign. The high density is considered ITER- and reactor-relevant, as for these devices high edge densities are envisaged, which will lead to a fast ionisation of inflowing W suppressing its penetration. When looking at the radiated power of the main plasma above the x-point and inside the separatrix (cf. fig. 2), W is only responsible for a minor part. Only in a few cases

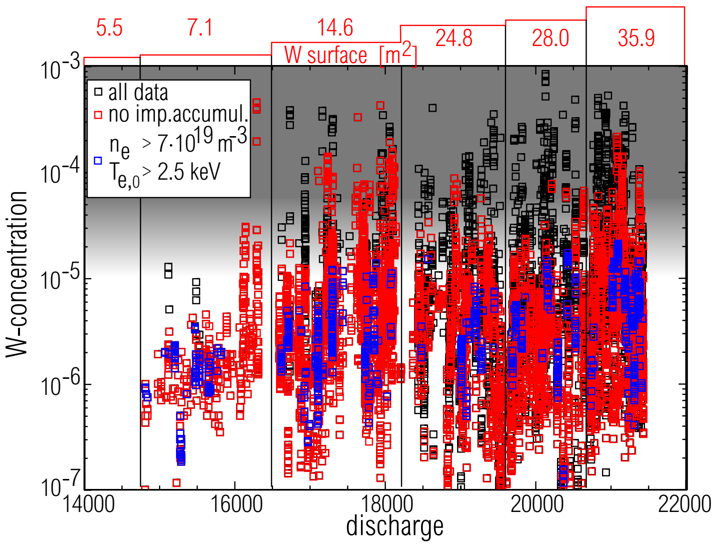

Fig. 1: *History of W-concentrations and areas of W-PFCs; shaded area marks the region of critical concentrations.*

the radiation due to W is dominating the radiative losses. Especially, for the high density cases the fraction of W-induced radiative losses is typically less then 30 % of the total radiative losses. For discharges, where W is causing 30 % of the losses or more, the total radiative losses are systematically larger, indicating bad conditioning of the walls. The green diamonds represent high density discharges with impurity accumulation. Their appearance at higher fractions of W-radiation documents a stronger tendency for impurity accumulation at higher W-concentrations. As impurity accumulation is degrading the stability of a discharge, its appearance is a more restrictive limitation on operation than just the power radiated by W. To avoid such situations central heating must be applied [2,3] and the total W-content needs to be controlled, e.g. by choosing operational parameters to avoid W-sources and/or wall conditioning. For conditioning, boronizations are applied regularly, which also lead to a temporarily coverage of all PFCs except the divertor as is revealed from surface analysis. In subsequent plasma discharges, the boron coverage is most rapidly eroded from the ICRH limiters, which is diagnosed by W-influx measurements at the ICRH limiters using the WI line at $400.8\,nm$ [4], as shown in fig. 3. Additionally, the impact on the W-density of the plasma are presented in fig. 3 along with the effective sputtering yield Y_{eff} [5]. The data originates from phases with ICRH during identical discharges ($I_p = 0.8\,MA$, $B_t = -2.0\,T$, $n_e = 3.5 \cdot 10^{19}\,m^{-3}$, $P_{ICRH} = 1.8\,MW$), while in between a variety of plasma discharges have been performed. Already after 20 discharges the boron layer is removed from the ICRH limiters, while at the guard limiters, which are more distant to the ICRH antennae (toroidally) and the plasma (radially), the boron layer is eroded within ≈ 50 discharges. The evolution of the W-density measured at $T_e \approx 1 - 1.5\,keV$ demonstrates that the ICRH limiters are the most important W-source for these discharges, while after about 100 discharges the W-density has reached a constant level (in this discharges) indicating that all important W-sources are active.

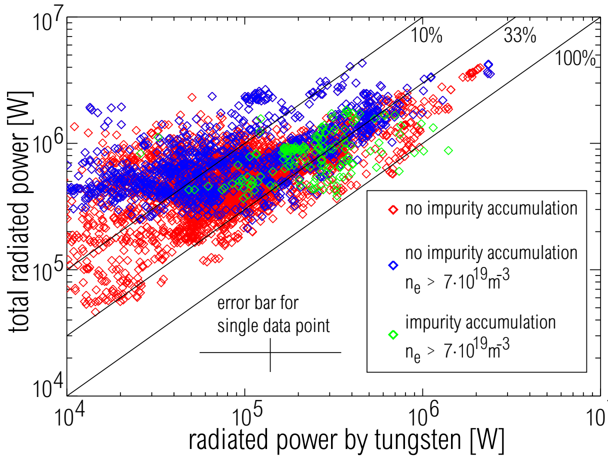

Fig. 2: *Radiated power vs. radiated power due to W-impurity radiation, derived from bolometer and spectroscopic measurements.*

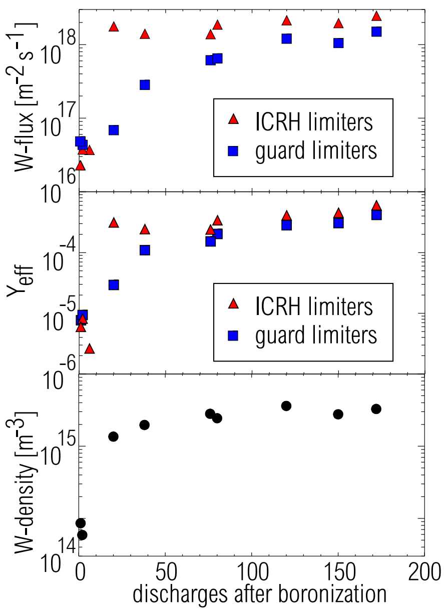

Fig. 3: *W-influx, effective sputtering yield and W-density for identical discharges following a boronization.*

Erosion of tungsten from the LFS limiters

After the removal of boron layers, dedicated experiments have been performed to investigate the W-erosion induced by different heating methods and paramters [4]. When ICRH is used, a strong increase of the sputtering yield and W-influx is observed at the ICRH limiters, while a less pronounced increase is measured at the more distant guard limiters ($\approx 0.8\,m$ toroidal distance) and an even smaller increase is observed at the inactive ICRH limiters ($\approx 5\,m$ toroidal distance). For an ICRH power fraction of $12 - 65\,\%$ the W-influx is enhanced by a factor of $2.5 - 10$. This enhancement appears within less than $1\,ms$ (i.e. the time resolution of the spectroscopic measurement) after switch-on of the ICRH. The fast timescale and the toroidal structure of erosion reveals that sheath rectification is most probably the source of this effect, while fast ions can be excluded. The change of the sputtering yield ΔY_{eff} allows for estimating the change in sheath potential $\Delta\Phi_{ICRH}$ in the range of $1 - 100\,V$ corresponding to ΔY_{eff} between 10^{-5} and 10^{-3}. For plasma discharges with ICRH and edge localized modes (ELMs), the spectroscopic measurements reveal a strong increase of Y_{eff} by a factor of 3 during an ELM. For a comparable set of purely neutral beam heated plasmas the erosion between ELMs lacks the enhancement due to ICRH, but during the ELM, Y_{eff} is comparable to that in ICRH phases (during ELMs). This suggests, that during an ELM more higher energetic ions hit the limiter, increasing Y_{eff} up to values, which cannot be enhanced by an additional sheath potential. Still, the W-erosion is enhanced in the ICRH case also during ELMs, because the impinging deuterium flux is larger by about a factor of 3 compared to pure neutral beam heated plasmas. The above findings demonstrate, that ICRH heating with ICRH limiters made from W may have both, beneficial and deleterious influence on plasma stability. On the one hand, it may prevent impurity accumulation by increasing central impurity transport, but on the other hand it increases the W-source, which leads more easily to impurity accumulation. Therefore, an improvement of the antennae is foreseen to minimize sheath rectification. Additionally, an extension of ECRH is on the way, which enables the avoidance of impurity accumulation, as has been shown earlier [2,3], and at the same time does not increase the W-source to the plasma.

Carbon transport and migration

In ASDEX Upgrade plasmas, C is still abundant at concentrations around 0.5 %. After coating 85 % of C-PFCs by W, the C-concentrations decreased only moderately to about 50 % of its initial value. Even though C-atoms are pumped during plasma discharges with a rate of about $0.6 \cdot 10^{19} s^{-1}$ the remaining C-sources are able to supply enough C. A numerical model for C-transport and migration was developed [6], which is able to describe a) the slow decrease of the C-concentration with decreasing area of C-PFCs, b) confirms spectroscopic observations and post mortem analyses of tiles and c) was tested by a dedicated experiment described below.

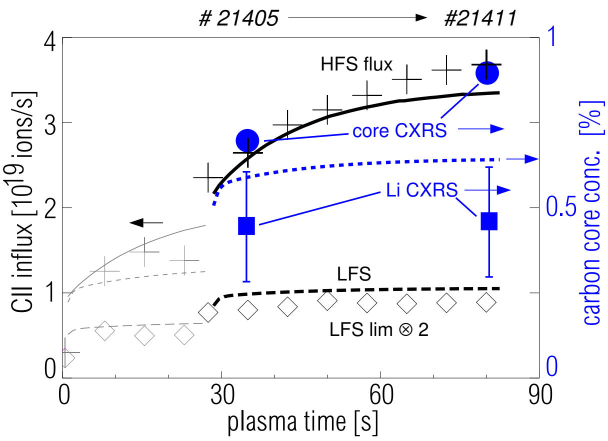

Fig. 4: *Comparison of C-transport and -migration model with measured fluxes (black) and concentrations (blue) vs. plasma discharge time. A net outer divertor source of* $1.5 \cdot 10^{19} s^{-1}$ *C-ions is used.*

The model assumes a sputter efficiency of C which is dependent of the wall coverage by C until a thickness of 2 monolayers is reached. The walls of the main chamber are building up a certain C-coverage until the sputtering and deposition reach an equilibrium. Transport from the sources in the lower divertor to the surfaces in the main chamber and into the plasma is taken into account, as well as transport out of the plasma and into the inner divertor, which acts as net sink. The free parameters have been calibrated by experimental findings, such as spectroscopic measured C-influx from W-coated surfaces in the main chamber and drift velocities in the SOL from the outer divertor to the main chamber and further to the inner divertor. From the equilibrium C-concentrations the model allows for deducing a net C-source from the outer divertor of about $1.5 \cdot 10^{19} s^{-1}$ ($\approx 10\,\%$ of turnover in outer divertor) penetrating at the separatrix upstream. This C is redistributed on all surfaces of the main chamber, giving rise to a C-concentration of about $0.5\,\%$ in the plasma. After about 5 cycles of deposition and re-erosion in the main chamber the C-atoms are deposited or pumped in the inner divertor. From the inner divertor also some back-flow of C is taken into account as spectroscopic measurements clearly support this feature. To test this model, 8 identical discharges have been performed following a boronization, in which the use of ICRH was minimized to leave the boron layers intact. Accordingly, the lower divertor is the only major C-source, as it is not affected by the boronization. In fig. 4 the model results are presented for these 8 discharges along with the corresponding measurements. (In gray, the results of the model and the measurements are shown for preceeding discharges, which exhibit slightly different plasma parameters.) For the actual 8 discharges of the experiment, fig. 4 demonstrates that at the LFS the timescale of C-coverage is only a few seconds, while the walls at the high-field side (HFS) are covered with C on a timescale of about $30\,s$. Both is documented by the C-influx (black symbols) from corresponding surfaces and good agreement between model and measurement is found. The measurements of C-concentration (blue) are in the right range, however, a detailed comparison to the model gives no significant information as the uncertainties are too large. The timescale for the increase of C-concentration in the model is closely connected to the influx on the LFS and the outer divertor. This experiment confirms the divertor net source to be about $1.5 \cdot 10^{19} s^{-1}$. In terms of C, today's PFCs at ASDEX Upgrade resemble an ITER-like wall, because besides the lower strikepoints nearly all PFCs are C-free. In ITER the C-migration must be minimal to suppress the co-deposition of tritium along with C. However, the above results suggest, that for an ITER-like wall the C in the divertor is very mobile and will show up in the main plasma at concentrations comparable to a full-C machine. It may be noted, that the introduced model incorporates many free parameters, not all of which are determined by the experiment. Therefore, some uncertainties about the exact mechanism remain, and especially the prediction for ITER lacks the influence of the beryllium impurity.

Conclusion

At ASDEX Upgrade, 85 % of the PFCs have been coated with W leading to W-concentrations up to $\approx 1 \cdot 10^{-5}$ for reactor relevant densities. The W-coated LFS-limiters act as a major W-source, when ICRH is switched-on, probably due to sheath rectification. An elevated W-concentration may lead to impurity accumulation even before the radiated power of W is critical for the discharge. As central RF-heating is frequently necessary to prevent impurity accumulation an improvement of the ICRH antennae and an extension of ECRH is foreseen. The latter is increasing central impurity transport, while no additional W-source is produced. Enhanced erosion from the limiters is also observed during ELMs, for which a similar effective sputtering yield in discharges with ICRH and with pure neutral beam heating is observed. The C-concentration is decreasing slowly with the reduction of graphite surfaces in ASDEX Upgrade, reaching now about $50\,\%$ of the original values. A numerical transport and migration model is able to describe this consistently with the available measurements. The model also allows for estimating the C-source from the outer divertor to be about $1.5 \cdot 10^{19} s^{-1}$. Additionally, the transport timescales from LFS to HFS support the picture of the model, which implies considerable leakage of C out of the divertor. For the oncoming campaigns the aim of a C-free machine is feasible, as the C-sink by pumping is of the same order as the now dominant C-source, which will be replaced by W this year.

References

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