OPERATION OF ASDEX UPGRADE WITH HIGH-Z WALL COATINGS

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Abstract. The material for plasma facing components of a future fusion device is still not decided. At present most experiments use graphite, because of its good thermo mechanical properties and the low radiation potential of carbon. Due to the high erosion yield and ,especially, due to the codeposition with tritium, its use in a fusion reactor is still questionable. Based on the good experience using tungsten as divertor material in ASDEX Upgrade [1,2], which demonstrated that a divertor tokamak can be operated with a tungsten divertor without reduction of the performance, a step by step strategy was followed. Main sources of the carbon are predicted at the inner heat shield, which covers the central column. Tungsten test tiles confirm the erosion at this position due to charge exchange neutral, but also a non negligible ion sputtering component. A first step was done by siliconisation. In ASDEX Upgrade the maximal silicon concentration was 0.002. Consequently the performance of the experiment was not influenced by silicon radiation. A second step was done by tungsten coating of $1.2m^2$ of the inner heat shield. Experiments are done without subsequent wall coating, which would cover the tungsten. Spectroscopically measured central tungsten densities are always below $\approx 5 * 10^{-6}$ and mostly below the detection limit. Again no influence on the plasma performance parameters are found. Extrapolation to ITER conditions yields concentrations, which will not prohibit successful operation. The next step in ASDEX Upgrade will be a mostly tungsten covered inner heat shield at the next experimental campaign.
keywords: first wall, plasma wall interaction, plasma facing components, wall coatings

1. Introduction

The material for plasma facing components of a future fusion device is still under discussion. Although low-Z materials are preferred due to the high acceptable concentration in the plasma core, they suffer from high physical sputtering rates. In the case of carbon, which is presently used in the form of graphite because of its good thermo mechanical properties in most fusion devices, chemical erosion by low energy hydrogen isotopes has also to be considered. Additionally the codeposition of tritium in quite stable layers, makes its use in a reactor rather questionable [3]. An alternative may be high-Z materials having a high threshold for physical sputtering. Especially tungsten exhibits excellent thermo mechanical properties and low hydrogen inventories. However it has to be verified whether or not central concentrations are below the stringent limit imposed by the high radiation power of high-Z materials.

2. Pre investigations

The capability to use tungsten as divertor material under reactor relevant conditions was demonstrated in ASDEX Upgrade [2,4]. In contrast to early limiter experiments, where large central concentrations of tungsten were found, most of the discharges with the full tungsten divertor showed concentrations below $2 \cdot 10^{-5}$. This would be acceptable in a future fusion device. Since under reactor relevant conditions, the plasma temperatures in the divertor are rather low, the erosion of W is also very small and especially almost no sputtering of W by hydrogen occurred. The situation is quite different at the main chamber walls of a divertor tokamak. Here the plasma flux and the flux of impurity ions should be much lower. Recent model calculations for ASDEX Upgrade showed that sputtering by higher energetic charge exchange (CX) neutrals may be significant [5]. Results of this modelling are shown in Fig. 1. The abscissa presents the length along the scrape-off layer in ASDEX Upgrade leading from the inner lower divertor along the central column, the upper divertor to the outer lower divertor. A main source of impurities is predicted at the central column which is covered by the inner heat shield. Taking only CX erosion into account, a reduction of the impurity influx by a factor of 200 to 500 is expected, if the carbon is replaced by tungsten. Mid-Z materials as SiC will reduce the erosion only by a factor of 2 to 5, depending on the position. The CX erosion is dominant at the lower inner heat shield. Consequently the ITER-FEAT design uses tungsten at this position.

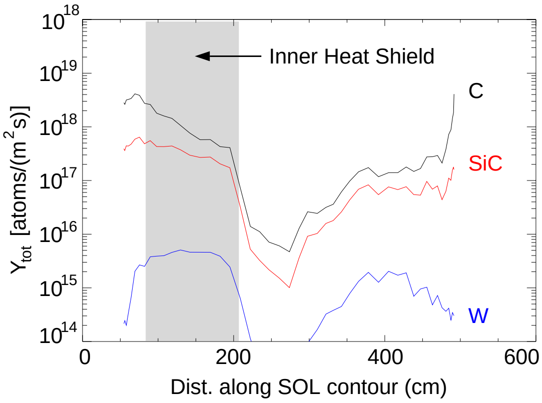

Figure 1: *Calculated erosion Yield Y_{tot} by CX neutrals at poloidal cross section.*

To verify these simulations four tungsten coated test tiles were mounted on the heat shield of ASDEX Upgrade (Fig. 2) at different poloidal positions for one experimental campaign. Analyses of these tiles using ion beam technique show very local erosion. At one side of the tile erosion of ≈ 2 *nm* was found, in good agreement with CX neutral calculations. At the other side of the tile the amount eroded was much higher, i.e. the applied layer of ≈ 7 *nm* was almost totally eroded at the edge of the tile. This signature could only be explained by ion sputtering [6]. One row of the inner heat shield consists of 64 planar tiles which form a polygon on the central column. Due to the inaccuracy of the mounting, one side of the tile could be hit by ions, whereas the other side was shadowed. This test shows that a coating of some 100 *nm* is sufficient to investigate high Z materials at the central column. These layers can be easily produced using in-situ coating like siliconisation or by physical vapour deposition. The erosion due to ion sputtering requires to cover the edges of the tiles. To investigate the maximal tolerable erosion yield, it has to be taken into account, that the material eroded in the main chamber may not benefit from divertor retention, but will penetrate into the main plasma more easily. To address the question whether materials with charge numbers Z beyond carbon may be used in the main chamber a stepped strategy was followed at ASDEX Upgrade.

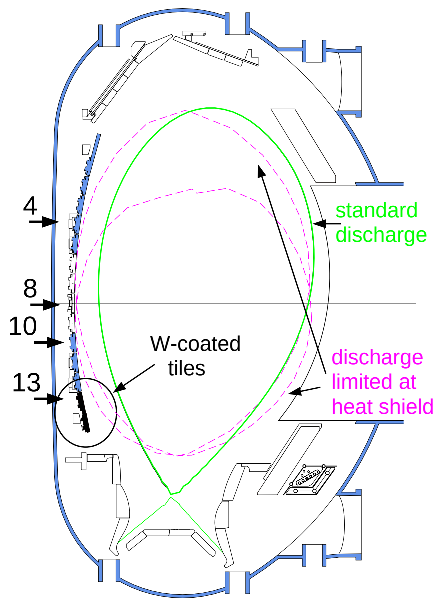

Figure 2: *Poloidal cross section of ASDEX Upgrade. The position of the W coated tiles is dark indicated. The test tiles are marked by arrows.*

3. First step: Siliconisation

In a first step the main chamber walls were coated with about 100 *nm* of silicon. This was done in-situ by depositing silicon by disintegration of silane in a glow discharge. Based on the experience in TEXTOR [7] the wall conditioning by boronisation was replaced by siliconisation. Like in the case of boronisation a similar reduction of the oxygen concentration was observed.

In contrast to a limiter tokamak, which was dominated by the silicon radiation [8], only small influence of siliconisation on the core plasma was found [9]. Beside maximum ohmic density limits, the same plasma performance was achieved as for a fresh boronised wall. The positive effects on the plasma are lasting much longer than for a boronisation. Consequently, siliconisation became a standard wall conditioning in ASDEX Upgrade. However within a few discharges after a fresh siliconisation, carbon remains the main plasma impurity. This hints to localised sources of carbon, where the silicon layer is eroded within this time. This high erosion can been explained by sputtering due to ion fluxes. Some of these sources could be identified after venting.

Silicon concentration was measured by spectroscopic methods. The line radiation of Si XIV enables to measure the central silicon density, whereas Soft X-ray radiation can be used to deduce the concentration of silicon at intermediate radii. For a low density H-mode discharge both methods are compared in Fig. 3. The measured silicon concentrations are in excellent agreement. In general maximum silicon concentrations are found during limiter phases, at plasma ramp-up and for low density H-mode discharges. Both types of discharges result in a high electron temperature, low density plasma at the SOL, which leads to a higher physical sputtering of silicon. No accumulation of silicon in the core plasma was observed. Even for internal transport barrier discharges without sawteeth only slightly peaked concentration profiles are found. The core silicon concentrations during plasma flat top are shown in Fig. 4 as a function of line averaged electron density. As expected the concentration is highest for H-mode discharges at low density, i.e. high temperature SOL. Typical silicon concentrations in range of $10^{-4} - 10^{-3}$ in H-Mode discharges with heating powers up to 15 MW were found. These concentrations are by far low enough to have no influence on the plasma performance. The silicon concentrations found were on the upper limit of the estimated values from the CX erosion source. Beside advantages for wall conditioning these experiments demonstrate that mid Z materials can by used at the main chamber of a divertor tokamak without significant increase of the radiation and adverse effects on plasma performance.

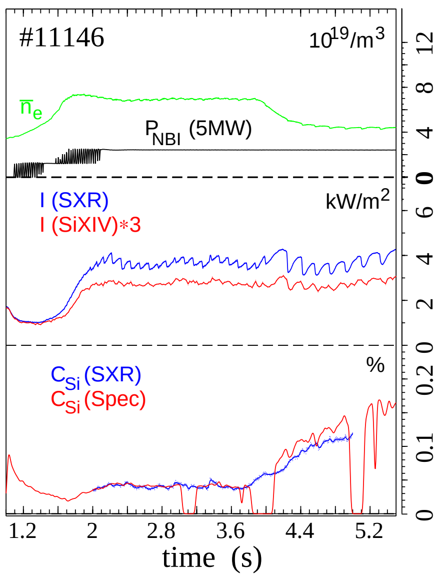

Figure 3: *Si concentration during low density H-mode discharge.*

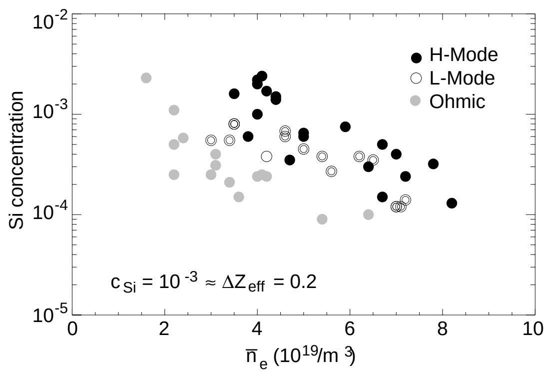

Figure 4: *Core silicon concentration with fresh coating for different plasma regimes.*

4. Second step: Tungsten coating

Motivated by these encouraging results, the use of tungsten coated tiles at the heat shield was decided. In the last experimental campaign the lowest two rows of graphite tiles at the central column were replaced by tungsten coated ones (Fig. 2). These represent a total area of 1.2 m^2

accounting for about 13% of the total area of the heat shield. The shape of the tiles was changed to reduce leading edge erosion due to ions (Fig. 5). Additionally the tiles were polished to get a surface roughness which is comparable to the thickness of the coating. The coating was applied by physical vapour deposition to a thickness of $200-500\ nm$. This thickness minimises the mechanical stress on the coating and is sufficiently large for the erosion expected at this position. Using the erosion rate of the marker experiment lifetime of about 50 h plasma operation in ASDEX Upgrade is expected. To investigate the operation with the pure W layer, operation was restarted without any wall coating. A new silicon layer was applied before shut down and activated again by He glowing after re-closing.

Figure 5: *Tungsten coated tile at the end of the campaign. The erosion dominated areas are marked by a broken line.*

The influx and the central concentration of W were measured spectroscopically. The detection limit was established using laser blow-off experiments. Tungsten concentrations during a standard H-mode discharge are shown in Fig. 6. During the shot the heating power and plasma density was enhanced. Maximum concentrations were found during the high power, low density phase, i.e. for high temperature SOL. The influence of the sawteeth on the tungsten concentration is evident. No tungsten concentrations above $C_W = 5 * 10^{-6}$ are found [10]. Even for limiter discharges and ITB scenarios the central tungsten concentration was below the detection limit. Therefore the concentrations, extrapolated for a full tungsten inner heath shield, are below the threshold tolerable in ITER and more than a factor of ten below concentrations, which might negatively affect the plasma operation of ASDEX Upgrade. Only small amounts of tungsten were measured using collection probes in the divertor region, were the plasma flux to the divertor is the highest. The plasma operation was not restricted due to the tungsten radiation in any way.

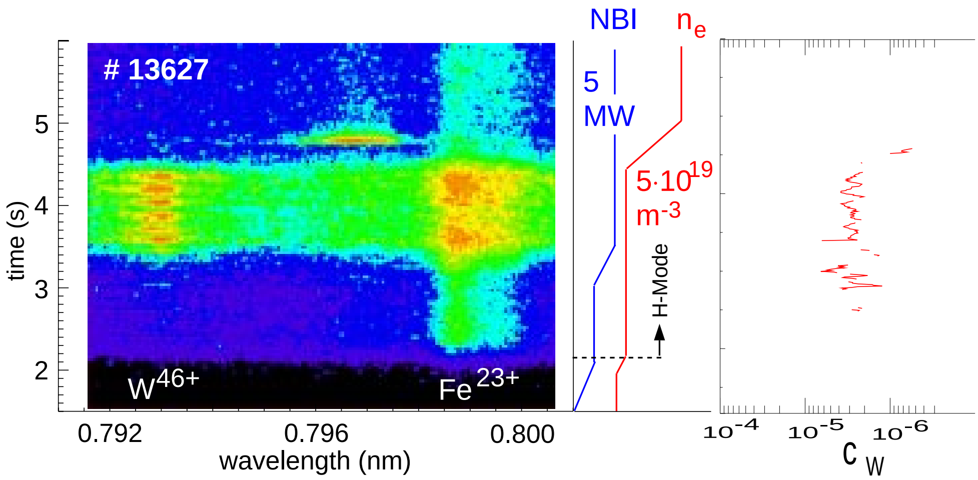

Figure 6: *Spectroscopic measurement of the central W density during a H-mode discharge. The deduced W concentrations are shown at the right picture.*

After the experimental campaign no macroscopic damage of the coating was found on the tiles [11]. Visual inspection shows regions, which are covered by Si and C and other areas with pure tungsten surfaces, as indicated by a broken line in Fig. 5. The deposition dominated regions are hidden to ions, which again demonstrates the importance of ion sputtering for the erosion at the inner heat shield of ASDEX Upgrade. A SEM analysis of the tungsten surface at the erosion dominated region showns no cracking, melting or delamination. The surface consists of pure tungsten, only the minor parts are silicon, remaining from siliconisation are found. Ion beam technique will be applied to investigate the composition of the layers.

5. Summary and Outlook

These results demonstrate the suitability of tungsten as a first wall material not only at the divertor but also in the main chamber of a fusion device. Furthermore analysis of the tiles show that, the erosion at the ASDEX Upgrade main chamber is strongly influenced by ions.

The next step of our experiments will be a coverage with tungsten of 2/3 of the total area of the heat shield of ASDEX Upgrade $(7m^2)$. The carbon will remain only at positions, which are used as limiter during special discharge scenarios. Different types of coating have been tested for thermal stability, adhesion and surface coverage [12]. As result of these tests, plasma arc coating with a thickness of 1 μm was selected. To minimise the costs, used tiles were machined in order to clean and polish the surface. The shape of the tiles was also modified to prevent edge erosion. In the spring 2001, ASDEX Upgrade will start with this tungsten coated inner heat sheath.

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