Tungsten Migration between Main Chamber and Divertor of ASDEX Upgrade

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Introduction

Using graphite or CFC as plasma-facing material, strong erosion and intense carbon deposition occur and limit the performance of a fusion device. Especially, the formation of thick gas-containing carbon layers is a serious problem in fusion devices with long tritium pulse operation. As alternative tungsten is being considered for use as a plasma facing component in the main chamber. In the tokamak ASDEX Upgrade about 10 % of the graphite tiles at the bottom part of the inner heat shield were replaced by tungsten coated tiles ($d \approx 400$ nm) during the experimental campaign in 1999/2000 [1]. After this period two tungsten-coated tiles from the inner heat shield were removed and investigated by surface analysis techniques in order to measure the total amount of eroded tungsten. In addition, a graphite tile from the entrance plate of the inner divertor (facing the heat shield) as well as a CFC-tile from the inner divertor were removed and investigated with respect to tungsten deposition to get information on the tungsten migration.

Experimental

The temporal development of plasma equilibria of a typical discharge of ASDEX-Upgrade and the arrangement of the samples are shown in Fig. 1. The W coated tiles of the inner heat shield (samples $1, 2$) were exposed from November 1999 up to June 2000, whereas the graphite tile of the divertor entrance plate (sample 3) and the CFC-tile of the inner divertor (sample 4) were exposed from November 1997 up to June 2000. Between November 1997 and August 1999 11 boronizations and 3 siliconizations were carried out. In the experimental campaign from November 1999 until June 2000 2 siliconizations

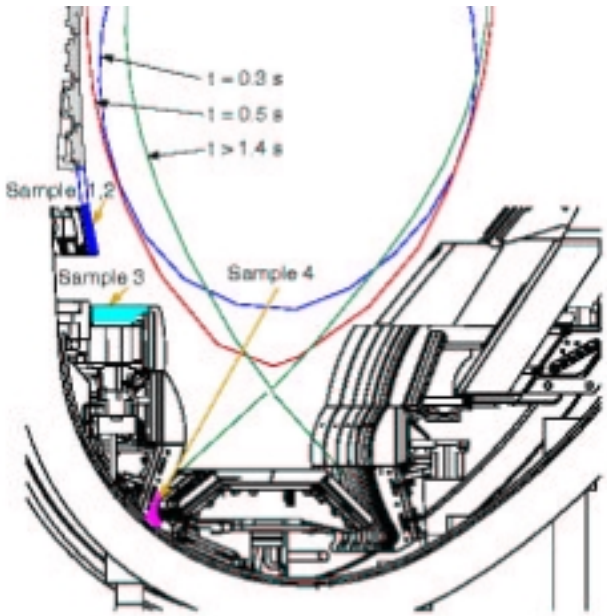

Figure 1: Typical plasma equilibria and arrangement of the samples exposed in ASDEX-Upgrade

were sufficient to guarantee an acceptable surface conditioning. In this last period with a total discharge time of about 4000 seconds the erosion and migration of tungsten in ASDEX-Upgrade were studied. This experimental campaign represents 792 ohmic and NB-heated discharges with heating powers up to 15 MW. The plasma densities near the last closed flux surface varied in the range between $5 \cdot 10^{18}$ and $7 \cdot 10^{19}$ m^{-3} [2] and the electron temperature in the range between 20 and 200 eV depending on the discharge conditions. After the exposure the tiles were investigated by Auger Electron Spectrometry (AES), Secondary Ion Mass Spectroscopy (SIMS) and Rutherford Back-Scattering (RBS).

Results

After plasma exposure the inner heat shield tiles exhibit different zones with prevailing erosion or deposition. Fig.2 shows the areal densities of the tungsten coating in toroidal direction of sample 1 measured by AES. A sharp transition from the deposition to the erosion zone was found at x=38 mm. The impurity contamination in the region with prevailing deposition of sample 1 measured by AES consists of carbon, oxygen and silicon as main components of about 85 % and iron, chromium, nickel and boron as minor components of about 15 %. The deposition layer was measured to be about 20 to 30 nm thick. Well-developed tungsten carbide layers were formed in the deposition areas indicating a temporary enhancement of the surface temperature up to values higher than 1000 K [3]. In the zone with prevailing erosion (x>38 mm) about $5 \cdot 10^{21}$ W-atoms m^{-2} were removed in comparison to the zone with prevailing deposition.

Figure 3 shows the results of sample 2 by means of RBS-measurements. Sample 2 was closely located in mirrored symmetry (see Figs. 2 and 3). The areal density of tungsten before plasma exposure was determined by RBS to be nearly constant across the sample surface

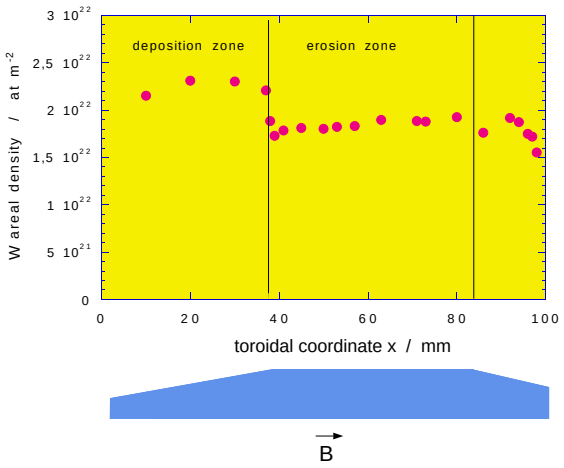

Figure 2: Toroidal dependence of the tungsten areal density of sample 1 measured by AES

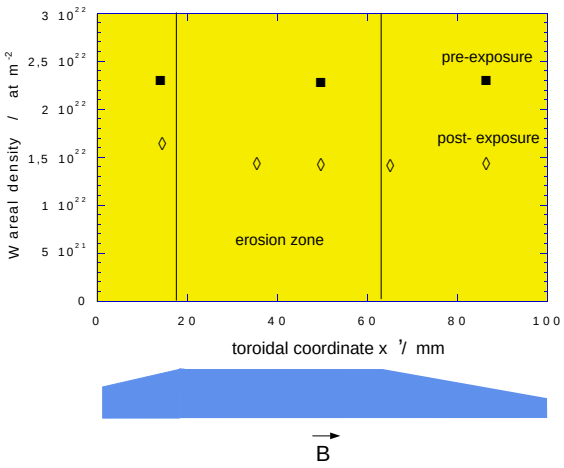

Figure 3: Toroidal dependence of the tungsten areal density of sample 2 measured by RBS

of about $2.3 \cdot 10^{22}$ W-atoms m^{-2}. This value agrees with the tungsten amount of the deposition dominated zone of sample 1 measured by AES. Sample 2 exhibits an eroded amount of the tungsten coating between to 6 and $9 \cdot 10^{21}$ W-atoms m^{-2}.

In addition, extremely non-uniform erosion of the inner heat shield tiles due to arcing has been observed. These arcing processes remove the total tungsten coating within the track widths of about 50 to 100 μm (see Fig.4). Regarding the erosion zone of the investigated tiles of the inner heat shield, the eroded amount due to arcing can be approximated to $< 1\%$ of the total eroded amount in this area.

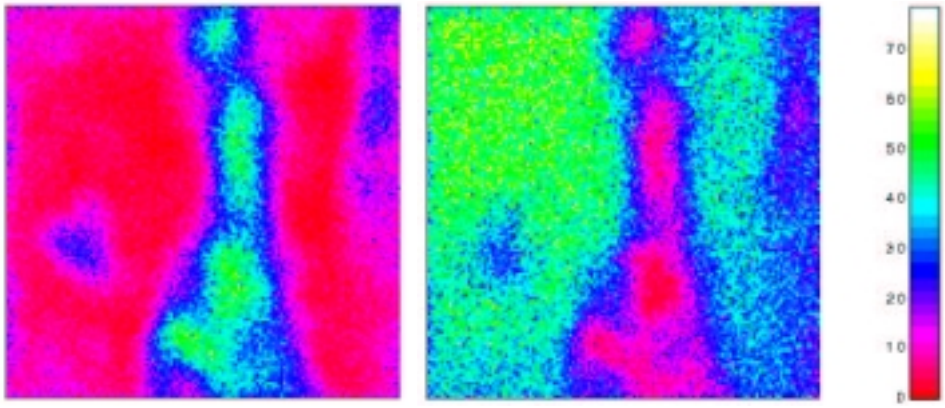

Figure 4: SIMS-images of the lateral distribution of C and W in the erosion area of the inner heat shield with an arc track. The images represent a part of the sample surface of 500 $\cdot$ 500 μm^2.

A fraction of the eroded tungsten was found in the deposition layer on the entrance plate of the inner divertor (sample 3). Increasing amounts from $8 \cdot 10^{19}$ up to $3 \cdot 10^{20}$ W-atoms m^{-2} were detected in the radial direction towards the plasma (Fig.5). The tungsten found on the entrance plate of the inner divertor indicates direct transport via the scrape-off layer plasma.

Investigating the CFC-tile of the inner divertor (sample 4) by means of AES-depth profile measurements tungsten deposition was only found in the area of the separatrix position with amounts of 1 to $3 \cdot 10^{20}$ W-atoms m^{-2} (Fig. 6). A few centimetres away from the separatrix position no tungsten could be detected. The sensitivity limit was about $2 \cdot 10^{19}$ atoms m^{-2}. The occurrence of deposited tungsten near the strike zone of the separatrix at the inner targets may indicate that tungsten penetrated partly towards the separatrix region.

We estimated the total amount of tungsten deposited on the inner CFC-divertor tiles to be about 3 to $5 \cdot 10^{19}$ W-atoms and compared this value to the total amount of tungsten eroded from the surface of all 128 tungsten coated heat shield tiles using the values of Fig. 2 to be about $3 \cdot 10^{21}$ W-atoms. In that way, one gets, that only 1 to 2 % of the eroded tungsten was deposited on the inner divertor tiles. In similar way, the part of deposited tungsten on all graphite entrance plates of the inner divertor was evaluated to 5 to 10 %.

Tungsten erosion has been simulated with B2/EIRENE modelling, based on sputtering by

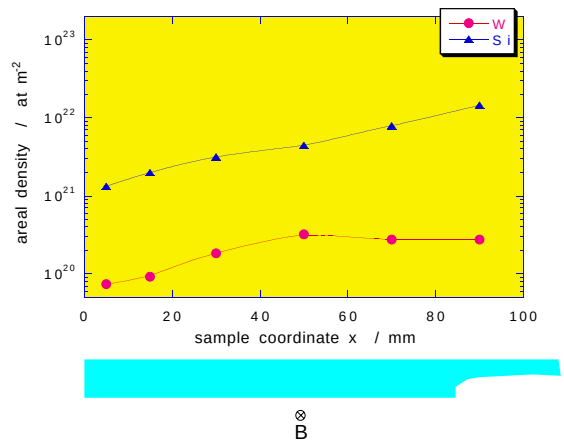

Figure 5: Deposition of W and Si onto the entrance plate of the inner divertor

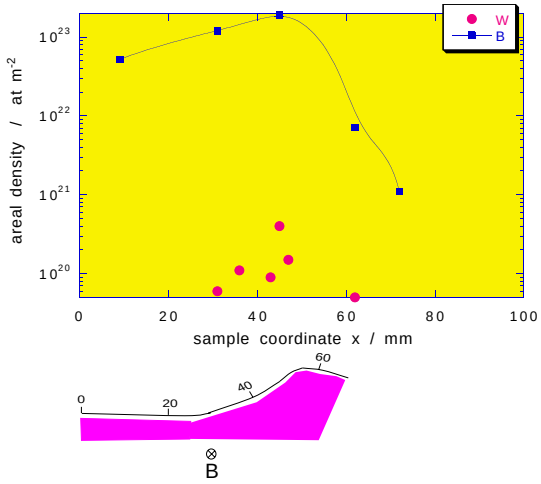

Figure 6: Deposition of W and B onto an inner CFC-divertor tile

charge exchange neutrals (CX). These calculations yield a tungsten erosion at the inner heat shield tiles of about 1 $\cdot 10^{20}$ atoms m^{-2} [4,5].

The measured erosion values are higher than the calculated ones by more than one order of magnitude. It seems that the process of CX-sputtering is not sufficient to explain the erosion processes at the central column of ASDEX-Upgrade. The different erosion in the toroidal direction in dependence on the angle between the surface and the magnetic field gives further indication that the dominant erosion process is sputtering by ion impact [5].

With 10 % W-coverage of the area of the central column heat shield (1 m^2) no significant impurity contamination of the central plasma has been found despite the relatively large erosion at the inner heat shield. The plasma performance was not affected [1], pointing to a very low penetration of tungsten into the confined plasma. This encourages the use of tungsten as a first wall material.

References

[1] R. Neu et al., J. Nucl. Mater. **290-293** (2001) 206-210

[2] H.S. Bosch et al., Plasma Phys. Contr. Fusion **41** (1999) A401

[3] D. Hildebrandt et al., J. Nucl. Mater. **290-293** (2001) 89-93

[4] H. Verbeek et al., Nuclear Fusion, **38** (1998) 1789-1803

[5] A. Tabasso et al., J. Nucl. Mater. **290-293** (2001) 326-330