Impact of Thermal Treatment and ICWC on Fuel Inventory in Co-deposits

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1. Introduction

Curtailing of in-vessel fuel inventory in fusion devices is essential in D-T plasma operation. The issue becomes particularly important with carbon plasma-facing components (PFCs) [1]. Currently considered basic schemes for reduction of fuel aim either at the desorption of hydrogen-containing species or at removal of the entire fuel-rich co-deposit. The first approach is based on surface heating by photonic means [2], long-term annealing of PFCs [3]. Ion cyclotron wall conditioning (ICWC) [4] is also proposed as a fuel removal technique but no detailed surface analysis of wall components has been carried out so far after high power radio-frequency (RF) pulses. Secondly, after PFCs cleaning by whichever technique, the fuel-depleted layers will remain in the vessel and they would be repeatedly exposed to plasma. The question is how do such layers respond to plasma during the repeated exposure? Systematic studies have been done to assess those two issues: (i) the efficiency of ICWC and (ii) fuel re-absorption by thermally treated co-deposits exposed to plasma after cleaning.

2. Experimental

The investigation was done with deposit-covered specimens obtained by sectioning graphite tiles from the TEXTOR tokamak: the main toroidal pump limiter at ALT-II (Advanced Limiter Test II) and inner bumper limiter being a shield of the dynamic ergodic divertor (DED). All specimens were analysed after tiles' retrieval from the tokamak and then at all stages of the experimental procedure by means of gas-phase and material research methods: thermal desorption spectrometry (TDS), ion beam techniques (e.g. NRA: nuclear reaction analysis using a $^3He^+$ beam) and scanning electron microscopy (SEM).

For thermal treatment a number of 1cm x 3cm plates were cut from the deposition zone of the ALT-II. Outgassing of each sample was performed at 1273 K with synchronous monitoring of masses M2 (H_2) M3 (HD), M4 (D_2) and masses M19 - M20 corresponding to D-containing water and C_1 hydrocarbons. For exposure in TEXTOR the outgassed plates were mounted on a holder attached vertically to the side of the test limiter (shaped as a single-roof block) to face the ion flux. The holder was made of a pure graphite plate which then served as a reference surface in the retention studies. The test limiter was positioned in the scrape-off layer (SOL) plasma at radial position r = 48.5 cm, i.e. 2.5 cm behind the last closed flux surface. The exposure was performed as a parasitic experiment during the commissioning of a charge exchange recombination spectrometer. The experimental program comprised 8 discharges (40 s in total), both ohmic and auxiliary heated by the two neutral beam injectors. The main plasma parameters were: the toroidal magnetic field strengths B_t = 2.2–2.6 T, line averaged electron density $n_e = 2.5–3\times10^{19}$ m^{-3}, plasma current I_p = 350–400 kA.

Another set of fuel-free samples was exposed to deuterium in a laboratory plasma device PADOS in which glow discharge is generated between two circular electrodes. The

outgassed and a graphite reference plate were placed on the lower electrode (cathode) at 450 K. The exposure was carried out for 3 h with a deuterium ion flux of about 1×10^{15} cm^2s^{-1}.

The exposure to ICWC discharges was done with a 5cm x 5 cm sample cut from the DED tile. As in the previous case, the plate was installed vertically on a complex test limiter holder and inserted from the bottom. ICRF pulses were generated using two antennae (50 kW each) in hydrogen (8×10^{-2} Pa) under B_t = 0.23 T. The entire programme comprised 550 pulses lasting 0.5 s, 40 pulses of 0.2 s and 45 of 2 s duration. In a given cycle, from 5 to 100 pulses, power was injected every 20 s. The exposure of the DED tile was performed during 490 pulses of 0.5 s duration.

3. Results and Discussion

The initial fuel content in thick co-deposited layers on the ALT-II tiles is in the range 7-11%, i.e. the deuterium-to-carbon concentration ratio (D/C) is about 0.1, as studied in detail after several long-term campaigns in TEXTOR [5]. As determined with NRA they contained around 4.7×10^{18} cm^{-2} D atoms layer of up to 7 μm. Thermal treatment enhances surface roughness and layer brittleness leading eventually to flaking and detachment of co-deposits, i.e. dust formation. These effects additionally complicate the subsequent analyses and handling of the deposit-containing samples. To find possible differences in the fuel content between the original co-deposits and layers exposed to deuterium after outgassing the following features were measured: (a) surface content of deuterium and depth distribution up to the accessible information depth with NRA; (b) total D amount and desorption characteristic with TDS.

Re-absorption experiments in TEXTOR: The deuterium radial distribution and content determined by NRA in the re-exposed ALT-II deposit and in the graphite holder are plotted in Fig. 1(a). Images of the re-exposed sample and the pure graphite holder are shown in Fig. 1(b) and 1(c), respectively. Arrows indicate the direction of the analysis with a $^3He^+$ beam. The growth of a new deposited layer was observed on the reference graphite plate. Concentration of D atoms along the plate drops exponentially with a characteristic e-folding length λ = 0.9–1.0 cm. This value is in agreement with earlier measurements using surface collectors [6]. The retention profile on the re-exposed fragment of ALT-II has a plateau region which matches the surface structure and corresponds to the area with the remaining co-deposited layer. Fuel retention in the re-exposed deposit is 1.2×10^{17} $D\cdot cm^{-2}$. This quantity is significantly smaller than the retention in the reference graphite placed at the same radial distance from the plasma: $3.3\text{-}1.5\times10^{17}$ $D\cdot cm^{-2}$. The areas where deposits peeled off also show a lower level of fuel retention in comparison to pure graphite. In the area where the transition between the deposition and erosion zones of the original ALT-II tile starts, the concentrations of D atoms in the studied sample and the reference plate become almost the same. A similar flat profile and lower absolute values of fuel retention were observed in the second experiment in TEXTOR. Depth profiles of the D atoms show broader distribution of fuel species in the flaking porous deposits than in the pure graphite. Thermal desorption spectra for the original deposits on ALT-II are shown in Fig. 2(a) and 2(b); more details on such deposits is given in [7]. The release of selected species is plotted: HD (M3) D_2 (M4) and M19 and M20 which represent a mixture of C_1 hydrocarbons and deuterated water (HDO, D_2O). The most important feature is that the release occurs predominantly in the temperature range of 700 K –

900 K with a peak value around 750 K. Spectra obtained for the re-exposed plate have followed the same trend as measured for the original limiter tile, i.e. single broad desorption peak (650-1000 K) with the maximal desorption rate for M3 and M4 occurring at around 750 K. However, the desorbed amount of HD and D_2 is 30-40 times smaller than from the original sample, thus indicating that fuel-depleted layers are not immediately re-saturated during consecutive exposure to plasma.

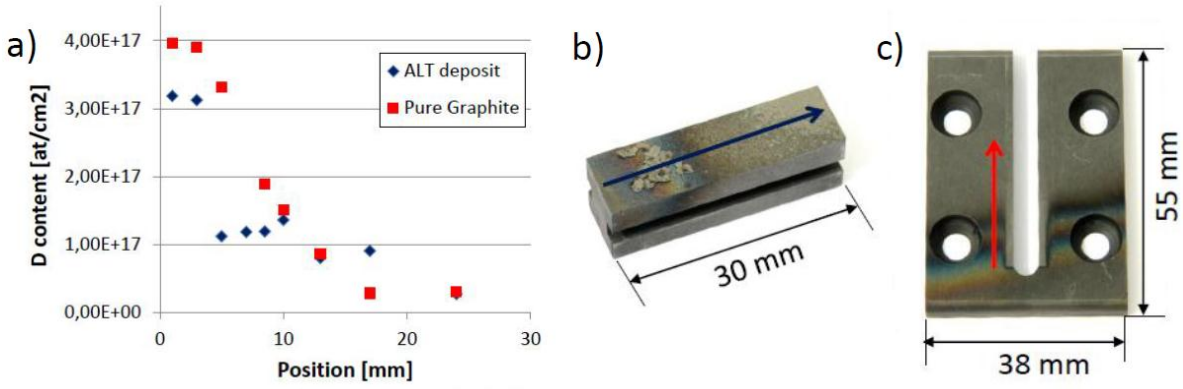

Figure 1: *(a) Deuterium content in the re-exposed deposit and reference plates after 40 seconds of exposure to SOL plasma in the TEXTOR tokamak; (b) Photo of the re-exposed specimen with the remaining deposit; (c) The pure graphite holder after the exposure. Arrows indicate the direction of the NRA scan.*

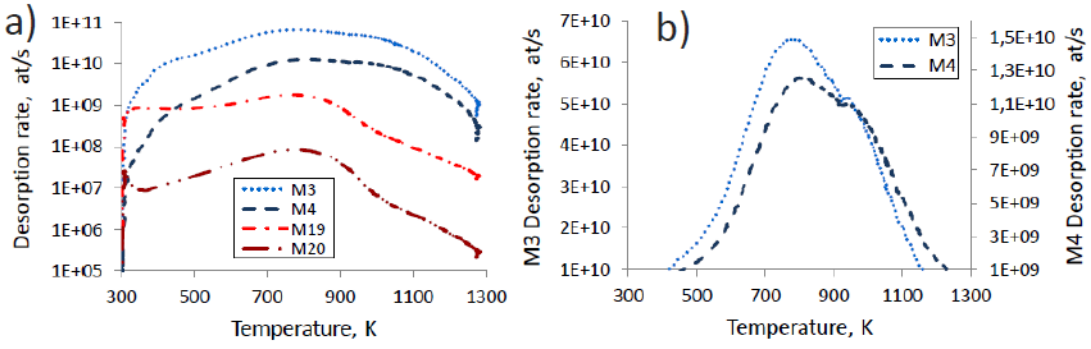

Figure 2: *TDS spectra for the initial and re-exposed deposits on ALT-II and a reference graphite plate. (a) Outgassing curves for M3, M4, M19, M20 from ALT-II plotted in logarithmic scale; (b) M3 and M4 spectra from ALT-II plotted in linear scale to emphasize the peak release around 750 K.*

Re-saturation in PADOS: As mentioned above, the original deposits on ALT-II contained 4.7×10^{18} cm^{-2} D atoms in a layer of up to 7 µm. Following the exposure to glow discharge in deuterium the outgassed and reference targets contained 1.44×10^{17} and 1.33×10^{17} $D\cdot cm^{-2}$, respectively. Though the absolute values are very similar, some difference is observed for the depth distribution of deuterium. In pure graphite about 70% of the D atoms is stored in a thin surface layer (< 1.5 µm) with a:C-D film while the rest is detected deeper (up to 4 µm). In the outgassed plate deuterium atoms are distributed evenly in the entire analysed depth. This can be related to the porous layer structure and it also indicates that re-absorption of deuterium atoms plays some role in the fuel retention in the outgassed and then re-exposed deposits, but the concentration is significantly smaller (at least 1 order of magnitude) when compared to the original amount. Distinct differences in fuel behaviour between targets exposed in TEXTOR and PADOS were revealed by TDS. While the maximum desorption rate of HD and D_2 for deposits formed at TEXTOR is around 750 K, outgassing of layers formed in PADOS is characterised by three peaks 500 K, 750 K and 1150 K indicating different binding states [8].

Fuel release by ICWC: Several DED tiles were retrieved from TEXTOR after several years of operation and analysed in detail with NRA for deuterium content. One such pre-characterised tile is shown in Fig. 3(a) and a piece of that tile after exposure to ICWC pulses is shown in Fig. 3(b), whereas plots in Fig. 3(c) show the deuterium content measured before and after that experiment. One perceives the decrease in fuel content following the ICWC in hydrogen. The drop is by a factor of more than 2. These are the first data of that kind obtained after cleaning a long-term wall component from a tokamak. These results are encouraging but still more detailed research is needed especially when it comes to the release of fuel from remote areas where the greatest deposition and fuel inventory has been observed. They are not accessible by ICWC.

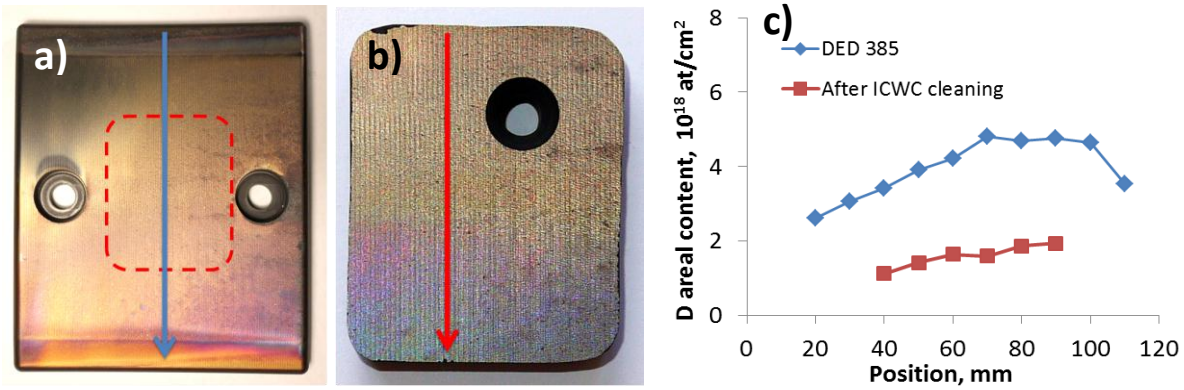

Figure 3: *(a) DED tile as retrieved from TEXTOR. The contour line corresponds to the area chosen for ICWC cleaning; (b) specimen sectioned from the tile and exposed to ICWC pulses; (c) deuterium content on the DED tile before after ICWC pulses. The direction of scan is marked in (a) and (b).*

4. Concluding remarks

The behaviour of fuel-depleted co-deposits was studied under repeated exposures to deuterium in a laboratory plasma device and in the TEXTOR tokamak. To our knowledge, this is the first report of systematic studies of fuel re-adsorption. The process did occur but the quantity of fuel species was much smaller than in the original layers. This may indicate that PFC surfaces that had been treated by thermal fuel removal methods are not immediately re-saturated by fuel during the repeated exposure to tokamak plasma. A positive result obtained by ICWC-induced cleaning should not overshadow the fact that this is an early results. At least a few other points should be carefully assessed: (a) toroidal uniformity of release; (b) depth of fuel release and (c) re-saturation of the layers during repeated exposure to fusion plasma.

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