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Experimental investigations of castellated monoblock structures in TEXTOR

A. Litnovsky [a,*], V. Philipps [a], P. Wienhold [a], G. Sergienko [a], B. Emmoth [b], M. Rubel [c], U. Breuer [d], E. Wessel [e]

[a] *Institut für Plasmaphysik, Association EURATOM, TEC, FZ – Jülich, D-52425 Jülich, Germany*
[b] *Department of Microelectronics, KTH, Association EURATOM – VR, S-164 40 Kista, Sweden*
[c] *Alfvén Laboratory, KTH, Association EURATOM – VR, S-100 44 Stockholm, Sweden*
[d] *Zentralabteilung fur Chemische Analyse, FZ – Jülich, D-52425, Jülich, Germany*
[e] *Institut fur Werkstoffe und Verfahren der Energietechnik, FZ – Jülich, D-52425, Jülich, Germany*

Abstract

To insure the thermo-mechanical durability of ITER it is planned to manufacture the castellated armour of the divertor i.e. to split the armour into cells [W. Daener et al., Fusion Eng. Des. 61&62 (2002) 61]. This will cause an increase of the surface area and may lead to carbon deposition and tritium accumulation in the gaps in between cells. To investigate the processes of deposition and fuel accumulation in gaps, a castellated test-limiter was exposed to the SOL plasma of TEXTOR. The geometry of castellation used was the same as proposed for the vertical divertor target in ITER [W. Daener et al., Fusion Eng. Des. 61&62 (2002) 61]. After exposure the limiter was investigated with various surface diagnostic techniques. Deposited layers containing carbon, hydrogen, deuterium and boron were found both on top plasma-facing surfaces and in the gaps. The amount of deuterium in the gaps was at least 30% of that found on the top surfaces.

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1. Introduction

The plasma facing components (PFCs) in ITER will be subject to the intensive radiative, particle and heat loads. To ensure thermal stability and operational durability of the PFCs, it is proposed to manufacture the castellated armour of the divertor by splitting it's armour into cells [1]. However, the introduction of castellated structures imposes serious concerns on impurity accumulation and fuel retention in the gaps between the cells. The expected surface area in ITER will increase up to a factor of 4 due to the castellation [1]. Removal of deposits from the gaps seems to be a difficult task [2]. Data on deuterium retention in divertor volume available from JET, DIII-D and ASDEX-Upgrade already predicts unacceptable levels of fuel inventory for future fusion devices like ITER [3]. This highlights the necessity of detailed assessment of ITER-like castellated monoblock

* Corresponding author.
E-mail address: a.litnovsky@fz-juelich.de (A. Litnovsky).

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structures, mainly from the standpoint of impurity retention and fuel accumulation. An experimental program is underway on TEXTOR to characterize the behavior of castellated structures under the impact of SOL and edge plasmas. In a previous study, the thermo-mechanical properties of a tungsten castellated limiter had been tested [4]. It was demonstrated that tungsten limiter could be operated below the ductile brittle transitional temperature at power loads of up to 35–40 MW/m^2 without mechanical defects.

The aim of present experiments on TEXTOR is to study the erosion and deposition patterns formed during the long-term plasma exposure of a castellated metallic limiter with ITER-like geometry. Comparison of the amount of deposited carbon and deuterium on top surfaces and in the gaps is of particular interest.

2. Experiment description

The castellated limiter is shown in Fig. 1. The castellation consists of eight monoblocks with dimensions of 10×60 mm, which were slotted every 10 mm along their long sides. The width of the slits is 0.5 mm. These eight monoblocks were put into a holder and inserts of 0.5 mm were installed in between each pair of monoblocks, so that finally a castellated structure consisting of 6×8 cells with dimensions $10 \times 10 \times 10$ mm was formed. These dimensions are proposed for the vertical target of the divertor in ITER [1]. The castellated structure was made from TZM alloy consisting from 99% Mo, ~0.5% Ti and ~0.1% Zr. The limiter was installed into the TEXTOR limiter lock III and exposed at a radial distance $R = 48.5$ cm from plasma center, 2.5 cm behind the last close flux surface (LCFS) during 125 plasma discharges with a total exposure time to the plasma of 559 plasma seconds. The discharges were mixed NBI-heated with heating power between 0.3 and 1.3 MW, 24 ECR-heated discharges with 0.6 MW power and 52 ICR-heated discharges with power varying from 0.2 MW up to 1.4 MW. The line-averaged density was within $(1.5–5) \times 10^{13}$ cm^{-3} during the exposure. The particle fluence to the limiter was estimated using He-beam data and cross-checked with H_α spectroscopy, with an averaged value of 4.9×10^{19} D/cm^2. A mean radial decay lengths of the particle flux of 1.2 cm was estimated which gave a peak particle fluence of 2.7×10^{20} D/cm^2 on the plasma facing top ($R = 48.5$ cm) down to 3.3×10^{19} D/cm^2 at the far end of the limiter deep in the SOL plasma. The surface temperature of the limiter was varied from 200 °C to 400 °C which agrees well with expected values for the upper vertical targets of ITER divertor [5].

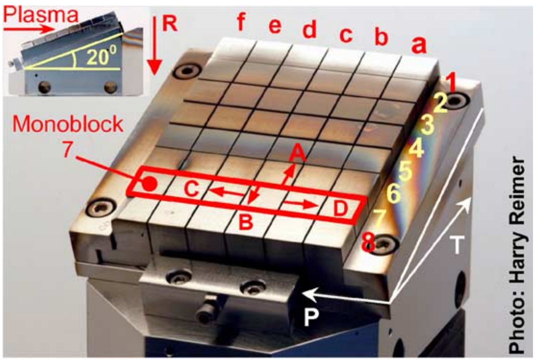

Fig. 1. View of castellated limiter after exposure, scheme of experiment and description of nomenclature used. P and T are poloidal and toroidal directions respectively.

After the exposure the limiter was dismantled and the monoblocks were analyzed using various surface analysis techniques. Time-of-flight secondary ion mass spectrometry (TOF SIMS) was applied to determine the depth profile of the elemental composition and to assess the thickness of the deposited layer. Nuclear reaction analysis (NRA) was performed both on top surfaces and in the gaps to determine the D-content in the deposited layers. Wavelength-dispersive X-ray spectroscopy (WDX) along with energy-dispersive X-ray spectroscopy (EDX) measurements were made to study the C and D pattern in the gaps.

We used the following nomenclature to uniquely identify the elements of the castellated structure. We number the monoblocks with numbers 1–8 as the radial distance from the plasma increases, so that the monoblock 1 is the one closest to the plasma (see Fig. 1). Gaps parallel to the poloidal direction and to the toroidal direction are named as poloidal and toroidal gaps respectively. Furthermore, four directions are used to uniquely identify each side of the poloidal and toroidal gaps: A, B, C and D as shown in Fig. 1.

3. Results and discussion

3.1. Visual observations

After the exposure a transition from net erosion on areas nearest to the plasma to net deposition deeper in the SOL was observed on the plasma-facing top surface of the castellated limiter. Such patterns usually occur on the surfaces of inclined targets exposed in the SOL of TEXTOR as described in details in [6,7]. At the radial position $R = 48.5$ cm monoblock 1 was exposed in the erosion-dominated zone while monoblocks 2–8 were subject to deposition. The analysis of interference colors was used to estimate the deposit thickness [8]. Colors were clearly visible and up to 3rd order of interference was observed on the deposit on the top surfaces. This corresponds to a film thickness of about 350 nm at the

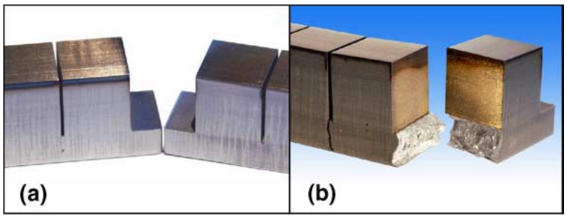

Fig. 2. Photograph of gaps after exposure: (a) poloidal gaps; (b) toroidal gaps.

maximum of deposition using a refraction index of $n \sim 1.7$ and an absorption index $k \sim 0.01$, as usually measured under similar conditions.

After dismantling the limiter deposited layers were found in the poloidal gaps located on a very thin stripe-like area with a width of 1–1.5 mm at the edges of monoblocks closest to the plasma, as shown in Fig. 2(a). Despite of the small width, the first order of interference color is visible on all the poloidal gaps of monoblocks 1–5 representing a film thickness of at least 100 nm.

Very recently, two monoblocks (2 and 3) were mechanically broken to allow studies in the toroidal gaps shown in Fig. 2(b). As can be seen, the deposits in toroidal gaps are significantly broader compared with the poloidal ones. The NRA measurements of deuterium content in the toroidal gaps are just started and this paper is mainly based on the analysis of the poloidal gaps.

3.2. SIMS measurements

SIMS measurements were performed on approximately 30 locations both on the top surfaces and in the gaps. A 2-beam SIMS TOF facility was used applying a 2 keV Cs-beam as a sputter beam and 25 keV Ga-beam as an analyzing beam. The device was calibrated with a reference sample, so that the sputter rate for a typical carbon deposit was known with an accuracy of 20%. Various elements have been scanned: H, D, B, C, O, Mo, MoC, MoO. SIMS analysis proved that the deposited film is a carbon film containing hydrogen, deuterium, oxygen and boron.

SIMS results are presented in Fig. 3, showing that the thickness of the deposited layer increases from monoblock 1 to monoblock 3 moving from the erosion to deposition-dominated area. On the top of the monoblock 3 the thickest deposit of 370 nm was measured. Moving away from plasma along the limiter surface the deposit thickness decreases from monoblock 3 to monoblock 7. This is due to the decreasing plasma and impurity fluxes lowering the deposition rate. The SIMS measurements agree well with the visual observations.

SIMS depth profiles of carbon in the gaps between the monoblocks 2, 3 and 4 were compared with those

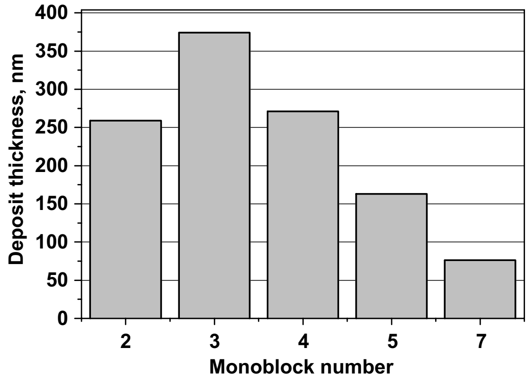

Fig. 3. Distribution of thickness of the deposit on the top surfaces of castellated limiter.

on the respective top surfaces. Examples of SIMS profiles are shown in Fig. 4 for the gap 3 from segment c, direction A and the top surfaces from monoblocks 3 and 4 from segment d. In the gap a deposit of 130 nm was found while on the top surfaces 3 and 4 the film thicknesses were 370 nm and 260 nm respectively.

3.3. NRA results

Measurements of deuterium content were done on a tandem accelerator with a 1.2 MeV $^{3}He^{+}$ beam in Uppsala, Sweden using the nuclear reaction [^{3}He (d,p) ^{4}He]. 4 locations in gaps and 5 locations on top surfaces of monoblocks 3 and 4 were studied. The average concentration on the top surfaces of monoblock 4 was found to be 1.7×10^{17} D/cm^2 and around 1.0×10^{17} D/cm^2 in the poloidal gaps of the monoblock 3, direction A. Combining these values with the SIMS results, D/C ratios of approximately 0.11 for the top surfaces and 0.13 for the poloidal gaps were found. However, TEXTOR was

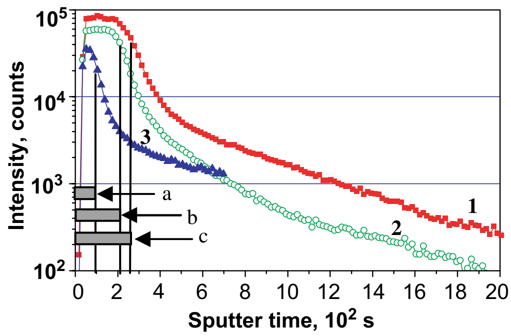

Fig. 4. SIMS depth profiles of carbon on top surfaces and in the gap: (1) top surface, monoblock 3 segment c; (2) top surface, monoblock 4 segment c; (3) gap in monoblock 3 segment d direction A. The half-width is given in nm: (a) 130 nm; (b) 260 nm; (c) 370 nm.

operated during the exposure mainly with H-NBI beams in D-gas fuelled plasmas, leading to significant hydrogen retention. The H/D ratio in the deposits was not measured directly, but estimated from the mean H/D ratio in the edge plasma of TEXTOR by spectroscopy and amounts to about 50%. Thus the overall (H + D)/C ratio is about 0.2.

First NRA measurements were made on toroidal gaps (after breaking) located on monoblock 3 segment b, direction *D* on three locations. Rather high areal *D* concentrations in the range of $4–7 \times 10^{17}$ D/cm^2 were measured. However, further investigations are needed to make firm conclusions on deposits in the toroidal gaps.

3.4. Results from electron beam diagnostics

A set of electron beam diagnostics (SEM, EDX and WDX) was used to analyze the elemental composition and depth distribution of elements in the gaps of monoblocks 2 and 7 using LEO 400 (Stereoscan) electron beam facility. Some EDX spectra are shown as an example in Fig. 5. A line scan was made in the gap of monoblock 2 in direction B. 10 keV electron beam was used, L_α photons of Molybdenum and K_α photons of carbon were detected in dependence on the distance from the edge closest to the plasma. Fig. 5(a) shows the decay of the K_α carbon line, representing the decrease of the C content on the surface while moving deeper in the gap. This is accompanied by the simultaneous increase in the Molybdenum L_α radiation, shown in Fig. 5(b).

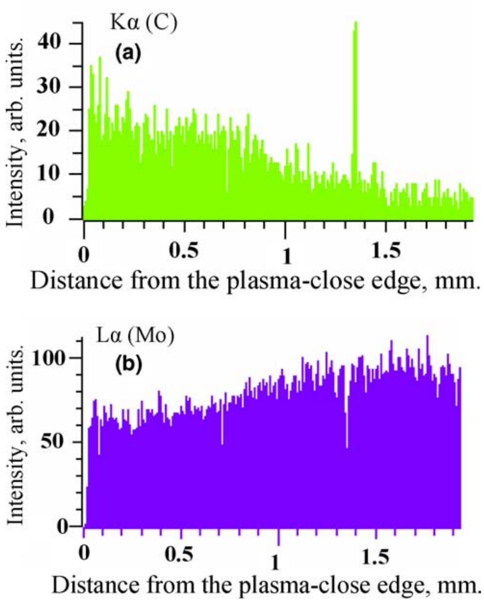

Fig. 5. Intensities of Mo L_α emission (a) and C K_α emission (b) as functions of the distance from the edge of a monoblock closest to the plasma.

The data show a decrease of the deposited layer thickness in the gaps, with an estimated decay length of about 1–2 mm.

4. Summary and discussion

Surface analysis and visual inspection showed the presence of deposited layers both on top surfaces and in the gaps of the castellated limiter. The deposit is an amorphous carbon film, enriched with hydrogen, deuterium, oxygen and boron. The deposited layers in the gaps are found within a narrow zone at edge of the monoblocks closest to the plasma. The thickness of the deposit is maximal at the plasma-closest position and reaches 200 nm there. Deposit thickness then decreases with the depth of the gap and levels-off at about 1–2 mm. An areal deuterium concentration in poloidal gaps ($D_{\mathrm{pol.gaps}} \sim 1.0 \times 10^{17}$ D/cm^2) is slightly lower than in the deposits on the top surfaces $D_{\mathrm{tops}} \sim 1.7 \times 10^{17}$ D/cm^2 representing the similar D/C ratio on both surfaces. The deuterium retention measured in the toroidal gaps is significantly higher ($D_{\mathrm{tor.gaps}} \sim 4–7 \times 10^{17}$ D/cm^2).

The ratio of the averaged total amount of deuterium retained in the gaps to that on the top surfaces is found to be $D_{\mathrm{total\ gaps}}/D_{\mathrm{total\ tops}} \geqslant 0.3$. This number is obtained from NRA data of the poloidal gaps in monoblock 3 and assuming the same decay length in the toroidal and poloidal gaps of 1.5 mm. Thus we treat this value as a lower limit since the data indicates more deuterium retention in the toroidal gaps.

However, the ratio of deuterium retention on top and in gaps changes also significantly along the surface of the castellated limiter. For the monoblocks closer to the erosion-dominated area (1 and 2), the deposit on the top surface is thinner (Fig. 3) whereas the deposits in the gaps do not tend to decrease, on the contrary they are becoming thicker. The data show a deposit of 200 nm for the gap of monoblock 2 segment c direction *A*, while the top surface has a deposit of 260 nm. Thus, the ratio $D_{\mathrm{total\ gaps}}/D_{\mathrm{total\ tops}}$ can increase significantly on the erosion dominated areas since the deposition is small on top surfaces while in gaps it remains high or increases.

5. Conclusions

The data show that the accumulation of fuel in the co-deposited hydrocarbon layers in the gaps of the castellated limiter is essential. A conservative estimate from the present experiment on TEXTOR gives a value of at least 30% of deuterium retained in deposits in the gaps of the castellated test-limiter with ITER-similar castellation. This data have to be compared with results from TFTR [9] and DIII-D [10] showing 15% and 40% of fuel accumulated in the gaps [11]. In comparison, the fuel content in the gaps of Mk-I JET divertor floor was found to be two times higher than on the plasma-facing surfaces [12]. Our data show also that the carbon layers are concentrated in the immediate vicinity of the gap entrance. This indicates that the transport is largely determined by the local geometrical structure of the gap with respect to the field lines. This is also supported by the preliminary analysis on the toroidal gaps, which are more in direction to the magnetic field line (not completely parallel). Recent analysis of gaps in MK I JET divertor has also demonstrated the importance of gap shaping and arrangement with respect to field lines [12]. Further investigations will be done in TEXTOR to study the effect of the geometry, gap size and surface temperature on the carbon transport and deuterium retention in the gaps. Modeling will be used to assess the carbon transport to the gaps. Development of techniques for possible mitigation of the deposition in the gaps and their removal is the targeted goal.

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