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FUEL DEPOSITION AND MATERIAL MIXING IN A CASTELLATED TUNGSTEN LIMITER

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1. Introduction

All plasma-facing components (PFC) in ITER will be castellated, i.e. composed of small blocks separated by narrow grooves (~ 0.5 mm) in order to reduce thermally-induced stress [1]. The contact of plasma with several different elements (including carbon) on the first wall, as foreseen in ITER, will also lead to the co-deposition of eroded material together with fuel species in the castellation. This has been shown for castellated structures exposed to the plasma for long- [2,3] or short-term [4,5] in present-day tokamaks. There will be over one million of such grooves in ITER. As a consequence, fuel retention in the castellation may significantly contribute to the overall tritium inventory. This contribution provides an account on the detailed examination of surfaces inside the castellation of a tungsten (W) test limiter exposed in the TEXTOR tokamak during two experiments. The major aim of post-mortem analyses was to determine the morphology on the plasma-facing surfaces and in the castellated gaps: (i) fuel retention; (ii) material mixing and new compound formation.

2. Experimental

The mushroom-shaped W limiter was composed of twelve individual segments brazed to a copper (Cu) base. The detachable segments were separated by a 0.5 mm wide slits. Two separate experiments were performed. In the first one, the limiter was kept for 157 s of the plasma operation (including 135 s of heating by neutral beams) at r=48 cm, i.e. 2 cm deep in the scrape-off layer (SOL). In the second case the limiter was moved forward step-by-step in the SOL and eventually immersed 5 mm inside the plasma. The limiter was preheated to 450-520 ^{o}C, i.e. above the brittle-to-ductile transition temperature in order to avoid tungsten damage under heat shock. After each experiment, the limiter was dismounted thus enabling morphology studies in the poloidally oriented gaps (i.e. grooves of castellation). The amount of co-deposited elements (carbon, boron and deuterium) was quantified using accelerator-based ion beam analysis (IBA) methods. Surface topography was observed with scanning electron microscopy (SEM). The distribution of carbon and other elements in the gaps was analysed with energy and wavelengths dispersive X-ray spectroscopy (EDS and WDS, respectively). X-ray diffraction (XRD) was used to study the effects of material mixing inside the castellation.

3. Results and discussion

3.1 Visual inspection

Images in Fig. 1 a, b show the limiter after the first and second experiment, respectively. The most characteristic features detected on surfaces are indicated by arrows. Only in the first case the deposition was found on outer surfaces at the far end of the limiter on the side facing the ion drift direction. This is qualitatively similar to results of earlier experiments with solid W

limiters at TEXTOR [6,7]. In the second exposure the limiter surface temperature exceeded the melting point of W ($T_{m(W)}$=3410 ^{o}C). The surface remained shiny but material melting and melt layer motion occurred on some segments, as reported in [8,9].

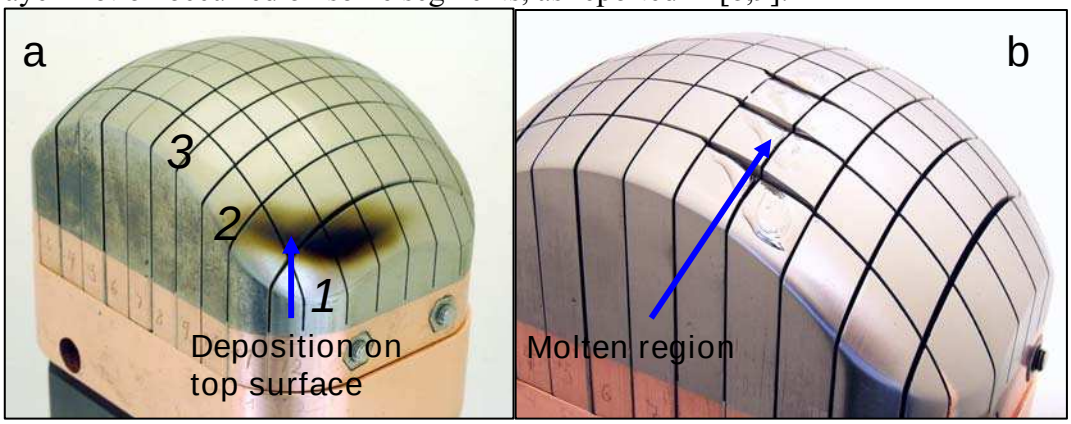

Figure 1. W macro-brush limiter after the first (a) and second (b) exposure at TEXTOR. Slices analised with IBA are numbered 1-3.

3.1 Co-deposition in castellated gaps

Images in Fig 2 a show the appearance of both sides of Slice 2 after the first experiment. The deposition inside the castellation has occurred in narrow belts (3-5 mm wide) just below the entrance to the gap. Detailed deposition profiles of D, B and C measured with IBA on both sides of the slice are in Fig. 2 b, c. One may notice some differences between the two sides of the slice. As can be noted on Side B, the deposition belt was formed only on a part of the slice. This is most probably related to the local temperature excursion on the limiter surface and consequential removal of the whole or part of the deposit formed during previous shots.

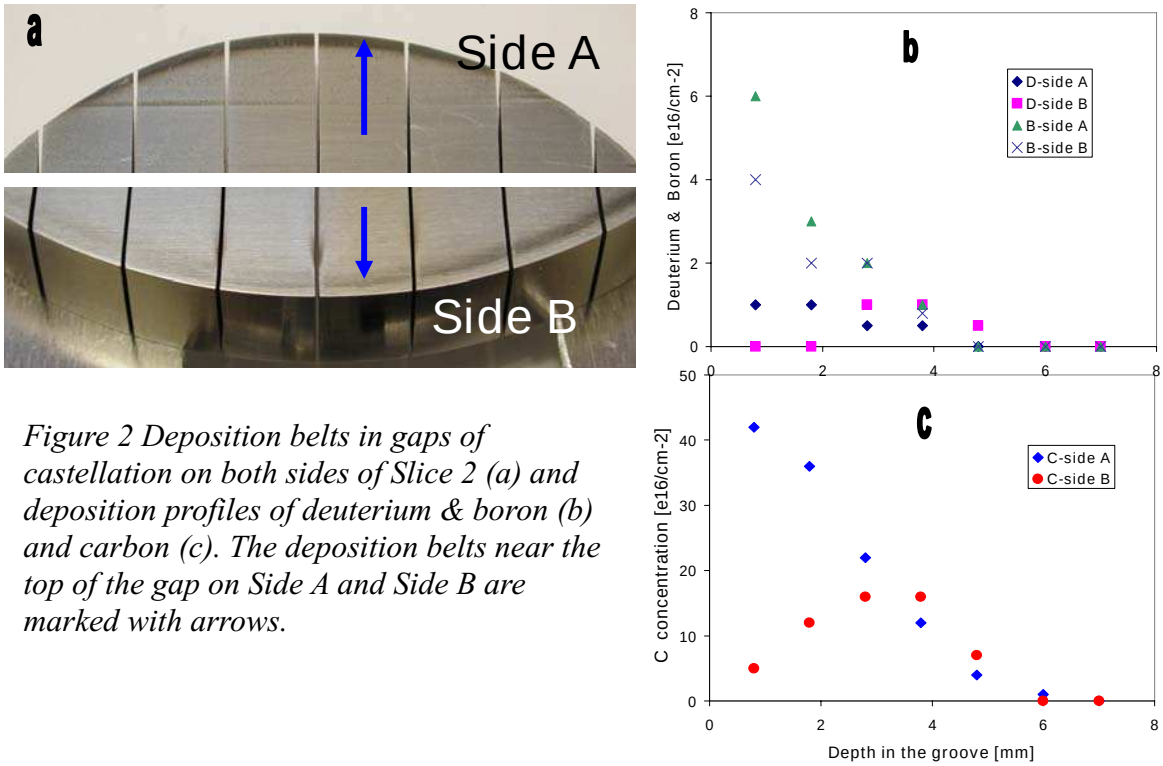

Figure 2 Deposition belts in gaps of castellation on both sides of Slice 2 (a) and deposition profiles of deuterium & boron (b) and carbon (c). The deposition belts near the top of the gap on Side A and Side B are marked with arrows.

There are several important features.

(i) The decay length (λ) of species in the castellation is in the range from 1.2 to 2.0 mm.

(ii) The concentration of all co-deposited species is small: below 4.2×10^{17} at cm^{-2} for carbon and 1×10^{16} at cm^{-2} for deuterium, thus showing - as expected - low fuel inventory on hot tungsten surfaces.

(iii) The presence of deuterium is always associated with the co-deposition of carbon.
The results form are highly consistent with other data for deposition in narrow gaps after the long- [2,3] and short-term [4,5] exposures of castellated structures to plasmas. They form a coherent picture indicating that no significant deposition and fuel inventory occurs in the grooves of narrow castellated metal PFC.

3.2. Material mixing

On the limiter overheated during the exposure (in the second experiment), no carbon and deuterium deposition could be traced. However, in the poloidal gaps there have been formed distinct belts with greyish are reddish deposits. An overview picture are detailed images of these regions are shown in Figure 3.

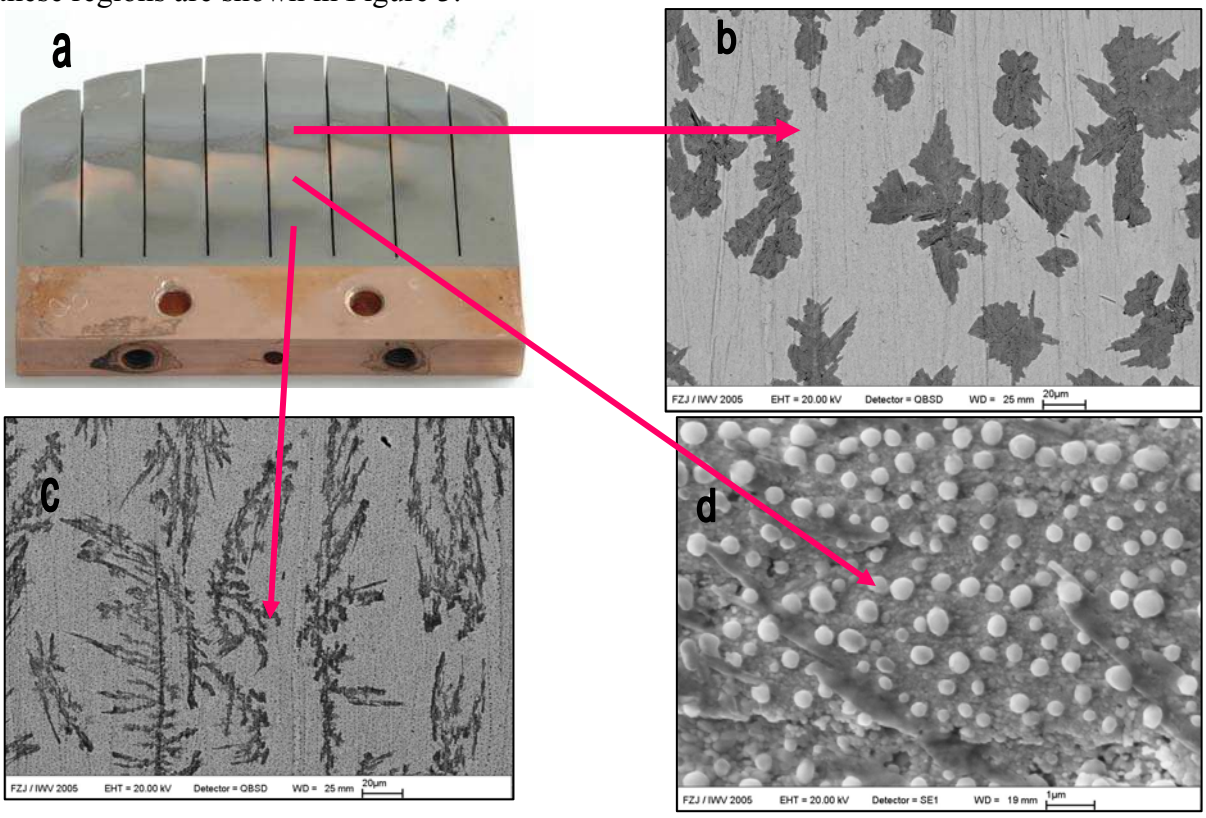

Figure 3. Overview (a) and SEM images of deposits structure in the poloidal gap (b) – (d).

EDS, WDS and XRD analyses identified tungsten dioxide (WO_2) in regions shown in Fig.3 b,c and copper droplets (Fig. 3d). The presence of these species may be attributed to the heating of the limiter to a very high temperature ($T_{surface}$ *over 3300* ^oC). The area covered with copper droplets is clearly separated from the limiter base thus indicating that the base was not the source of metal. The most probable explanation is that Cu originated from the toroidal gaps which were cut by spark erosion with a brass wire. The copper, remaining in the slots after machining, got molten ($T_{m(Cu)}$ *=1083* ^oC) and flew down to the colder region, where it spilled over and solidified. This hypothesis is consistent with the temperature profile in the limiter during its exposure. The calculation of the profile was based on the temperature measurements of the limiter surface (pyrometer) and in the base (thermocouples) [10].

To our knowledge, the presence of W oxide on PFC has been identified for the first time. Volatile tungsten di- and tri-oxides are known to be formed at elevated temperatures. It has also been reported that WO_2 may remain stable in a hydrogen-rich atmosphere [11]. One may assume that a volatile compound (oxide or H_2WO_4 [12]) was formed in the gap and then condensed on the surface cooler than 750 ^{o}C. The increase of the oxygen level at the limiter

has been observed with spectroscopy during the exposure. It probably originated from the outgasing (water vapour) of bulk tungsten. The mechanism of oxide formation and transport under tokamak conditions is still to be studied during the next experiments planned with the tungsten limiter.

4. Concluding remarks

Detailed surface studies performed after two experiments with a tungsten limiter have provided a deep insight into the deposition, fuel inventory and material mixing occurring in the gaps of castellation. It has been shown that the deposition characterised by short decay length (~ 1.5 mm) occurs mainly in the region just below the entrance to the gap. The fuel inventory on hot metal PFC is small and it is strongly associated with the co-deposition of carbon. These results from TEXTOR and also earlier data from JET [2,3] operated in the past with beryllium limiters and divertor allow some optimism regarding the fuel inventory in a next-step device with all castellated PFC. However, the present results cannot be immediately translated into conclusions and quantitative predictions regarding the material migration and fuel inventory in ITER. The planned material configuration in the divertor (W and CFC) and on the main chamber wall (Be) will be different than in any present-day device. This will change both the scenario of material erosion and will influence fuel co-deposition. It will be one of the main issues studied at JET when it is operated in the future with an ITER-like wall [13]. One may suggest on the basis of results presented here that in a machine with non-carbon walls in the main chamber, the material transport and resulting fuel inventory would be reduced. It is unreasonable to expect, however, that the deposition in the castellation grooves will be completely eliminated. The development of efficient techniques of fuel removal from all parts of the in-vessel components remains, therefore, crucial for the operation of a reactor-class device.

It has been shown for the first time in studies of PFC that tungsten dioxide was formed inside the gaps. The mechanism underlying this process is still to be clarified in future experiments. The issue may be considered important because the formation of volatile W compounds, if massive, would be a meaningful pathway for tungsten migration in a tokamak, especially as consequence of a water-leak accident. However, based on the early results presented in this contribution, it is premature to attempt the assessment of the impact of this process for the operation of a next-step device with tungsten PFC.

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