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Overview of long-term fuel inventory and co-deposition in castellated beryllium limiters at JET

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A B S T R A C T

Morphology of castellated Be tiles from the belt limiter exposed to the JET plasma for 56,000 s was examined on both sides of castellated grooves, on plasma-facing and side surfaces of the tiles. The essential results are (i) deuterium retention in the castellated grooves and in other locations is associated with co-deposition of carbon; (ii) the decay length of deposition in the castellation is around 1.5 mm; (iii) no deuterium is detected in bulk Be; (iv) bridging of gaps by molten beryllium occurred but gaps were not filled with Be; (v) on side surfaces of the tiles the formation of BeO layer was detected at a distance of 20 mm and more from the plasma-facing surface. The consequences for a long-term operation of a reactor-class device with several different plasma-facing materials are addressed.

1. Introduction

The assessment of fuel retention is of extreme importance for economy and safety in operation of next-step fusion devices. There are several pathways leading to the accumulation of hydrogen-containing species in the wall. In carbon surrounding the major mechanism is co-deposition. It is known that eroded material is transported and co-deposited together with fuel species in areas shadowed from the direct plasma line-of-sight [1–4]. Grooves of castellation and side surfaces of plasma-facing components (PFC), i.e. surfaces located in gaps separating tiles, are considered as a potential trap for vast amount of co-deposited fuel atoms. All PFC in ITER and all beryllium limiters in JET during the ITER-Like Wall (ILW) Project [5] will be castellated, i.e. composed of small blocks separated by narrow grooves (less than 1 mm) in order to reduce thermally-induced stress. The main question is whether fuel retention in such places may significantly contribute or even be decisive for the overall tritium inventory in ITER where the total number of castellated grooves will exceed 2 millions and the area of surfaces in the castellated grooves in the main chamber wall and divertor will be greater (by at least factor of 2) than the area of plasma-facing surfaces, as inferred from [6]. This calls for detailed studies of castellated structures and gaps between PFC tiles retrieved from present-day tokamaks.

Until now, JET has been the only tokamak operated in the past with a carbon wall and large-scale castellated metal structures: beryllium limiters and divertor tiles. Fig. 1(a) shows the in-vessel components in JET with belt limiters and Mk-0 divertor tiles made of beryllium. Other main chamber components (inner column cladding and upper plates) were made of carbon. Therefore, studies of Be limiters have given an unique chance to recognise the material transport to metal components during the long-term operation in the carbon-containing surrounding. Analysis of deposition in the narrow castellated grooves and in tile gaps can contribute to the assessment of material migration in environment containing both C and Be. It helps to distinguish the influence of chemical and physical processes in wall erosion and to draw conclusions regarding their impact on fuel inventory. The determination of migration into gaps and the resulting fuel retention is crucial for ITER. Therefore, in recent years systematic studies were done for components of the JET Mk-I divertor which was first operated with small carbon blocks and then with castellated beryllium tiles [7,8]. In the carbon divertor amount of fuel in the gaps (6–10 mm wide) between tiles was approximately twice greater than on plasma-facing surfaces [7]. In case of the castellated Be Mk-I divertor only small amount of deuterium was found in the narrow castellated grooves [8]. Steep decrease of the fuel in such grooves has been found: e-folding length $\lambda \sim 1.5$ mm. There have also been short-term experiments at TEXTOR [9] and ASDEX [10] showing the same steep decrease.

The intention of this work was to recognise the deposition and fuel retention in the JET beryllium belt limiter: the content and distribution of fuel and plasma impurity species on plasma-facing surfaces, side surfaces of the tiles and in the grooves of castellation.

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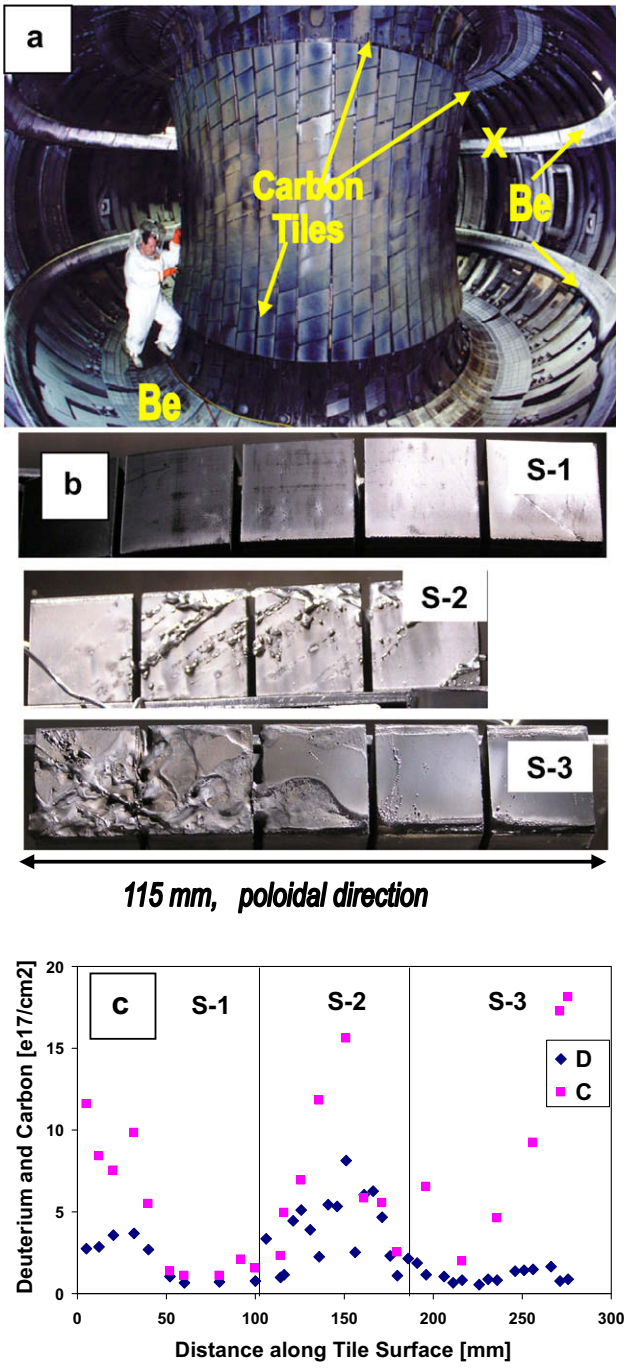

Fig. 1. View inside the JET vessel with beryllium components: divertor floor and belt limiters (a); surface topography of plasma-facing surfaces (b) and distribution of deuterium and carbon on consecutive segments of the cleaved limiter tile (c).

2. Experimental

The study was carried out for beryllium belt limiters protecting the main chamber wall (castellation 15–16 mm deep and 1 mm wide) operated for approximately 56,000 s during campaigns in 1989–1992. There were approximately 2000 tiles installed in the machine; each 379 mm long, 106 high and 22 mm wide. They were separated by 4–5 mm wide gaps with inconel plates as spacers. All castellated grooves on the limiter were toroidal, whereas the gaps between tiles were in poloidal direction. Two of the tiles were retrieved from storage for detailed studies. In the Beryllium Handling Facility at JET each tile was cleaved in three places along the castellation in order to expose surfaces in the groove thus enabling analysis of co-deposits formed on both sides of six grooves. The preparation of samples by cleaving minimised the risk of contamination of freshly open surfaces by deuterium from surfaces exposed to the plasma [11] and the production of Be dust which would inevitably occur in case of sawing the tiles.

Ex situ examination of the tiles was done by means of nuclear reaction analysis (NRA) with a $^3He^+$ beam to quantify and map surface contents of deuterium, beryllium and carbon. Most of the D, Be and C analyses were carried out with a 2.5 MeV $^3He^+$ beam. Enhanced proton scattering (EPS) with a 2.5 MeV H^+ beam was used to determine oxygen and carbon contents on the tiles.

3. Results and discussion

3.1. Morphology of plasma-facing and side surfaces of limiter tiles

Images in Fig. 1(b) show the appearance of plasma-facing side for three consecutive segments (S-1 to S-3) of the cleaved tile. The corresponding deposition profiles of deuterium and carbon are plotted in Fig. 1(c). The central area of the tile (Segments 2 and 3) was molten during the plasma operation but the remaining parts at both ends of the tile have fairly smooth shiny appearance (Segments 1 and 4, the latter is not shown here). In the melt zone, the surface is very rough and some gap bridging by molten Be is noted. However, the melt layer remained on the surface and castellated grooves were not filled with liquid metal; see Fig. 2(a). Results indicate some variations in the fuel content along the tile. In absolute numbers ($1–7 \times 10^{17}$ cm^{-2}), the amount of deuterium is small in comparison to over 1×10^{19} D cm^{-2} as measured in shadowed areas of the JET divertors [1–4]. The integrated content on the plasma-facing side of one tile is around 2.1×10^{19} D atoms. The greatest content is found in the melt zone on Segment 2. Also the amount of carbon increases in that region but the quantity of carbon is quite small, maximum 1.8×10^{18} cm^{-2}. Both C and D have been detected only in the very surface layer. The increase of the D level in the melt zone agrees with previously published data for the melt region of the Mk-I beryllium divertor. This may be caused by fuel trapping in molten Be (see [12]) or in beryllium carbide formed on the surface at high temperature. However, if carbide is formed, its thickness would be limited to approximately 1 μm because small carbon quantity is found on the tile surface. The reason of slightly enhanced fuel trapping is still unclear but the issue certainly requires deeper insight and clarification in future studies.

Side surfaces located in gaps separating the tiles of several segments were analysed. Fig. 2(a) and (b) show, respectively, the appearance of a segment with the molten surface and the distribution of deuterium and carbon from the near-plasma region down to the bottom of the tile. The results clearly show that the amounts are small and the deposition profiles are steep with the e-folding length, λ, of around 2 mm. At the depth of approximately 10 mm into the gap the deuterium profile flattens and remains constant to the very bottom of the tile. A striking feature is the presence of a BeO layer, approximately 3–3.5 μm thick, found only in the region located deep in the gap (from 20 mm down) where the carbon content falls to the level below 1×10^{17} cm^{-2}. Fuel retention is still detected in that oxide film. It is not possible to conclude when the formation of the oxide took place, i.e. at the machine or during the long-term storage in air. However, the result may indicate that the carbon-containing film at the upper part of tile prevented beryllium oxidation. This hypothesis is partly supported by the fact that no BeO has been found on the plasma-facing surfaces containing some carbon.

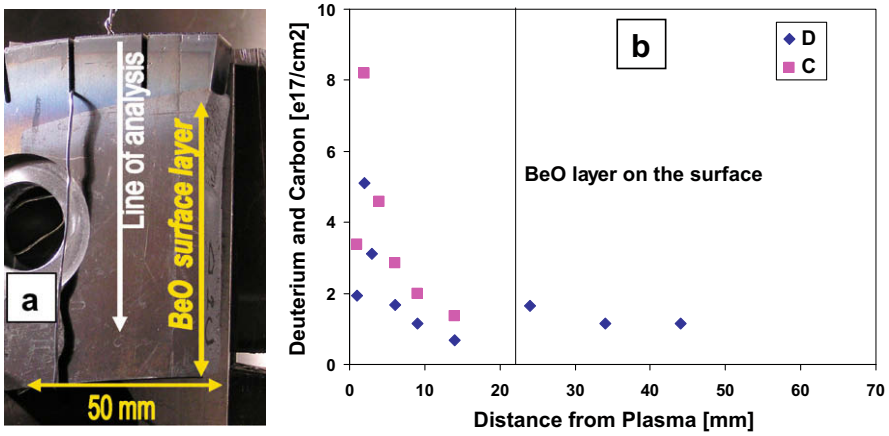

Fig. 2. Side surface (a) and deposition profiles of deuterium and carbon in the gap separating tiles.

3.2. Morphology of castellated grooves

Images in Fig. 3(a) show two sides of the castellated groove created by cleaving the tile. Bright areas at far ends are shiny surfaces of metallic beryllium in the freshly fractured part. Deposition profiles of D and C inside the groove are plotted in Fig. 3(b). The results obtained for both sides of six castellated grooves allow a statement that, despite some differences in the appearance of surfaces, all profiles are very similar. Therefore, the most important features are summarised by the following. In all cases the presence of deposit is observed just below the entrance to the gap. Smaller amounts of species found very close to the entrance may be attributed to thermal desorption of gases (D_2 and C_xD_y molecules) due to high temperature of the limiter surface. The profiles reach their maximum at the depth of about 2–3 mm and then decay to a level below 1×10^{17} cm^2 already at 4–5 mm. In neither case the D content exceeds 7×10^{17} cm^{-2}, which is considered to be a very small amount in comparison to that found on side surfaces of carbon tiles [7]. The e-folding length, λ, for deposition is 1.2–1.6 mm. This value agrees very well with that determined for the castellated Be Mk-I divertor [8] and also found in some short-term experiments [9,10]. The presence of D and C is detected down to the bottom of the groove. In some cases there is an increase in concentration at the bottom. This may indicate that the groove bottom is a final trap for C–D neutrals entering the gap at right angle to the tile surface. As inferred from EPS spectra, some oxygen (up to 6×10^{17} cm^{-2}) is present in co-deposits but there is no evidence of bulk BeO formation. The presence of oxygen is probably related to penetration of some quantities of oxygen-containing gases (water vapour, O_2, CO_2) into the film during the tile storage for 14 years. However, the amount of gases was limited because the tiles were stored in tightly sealed bags.

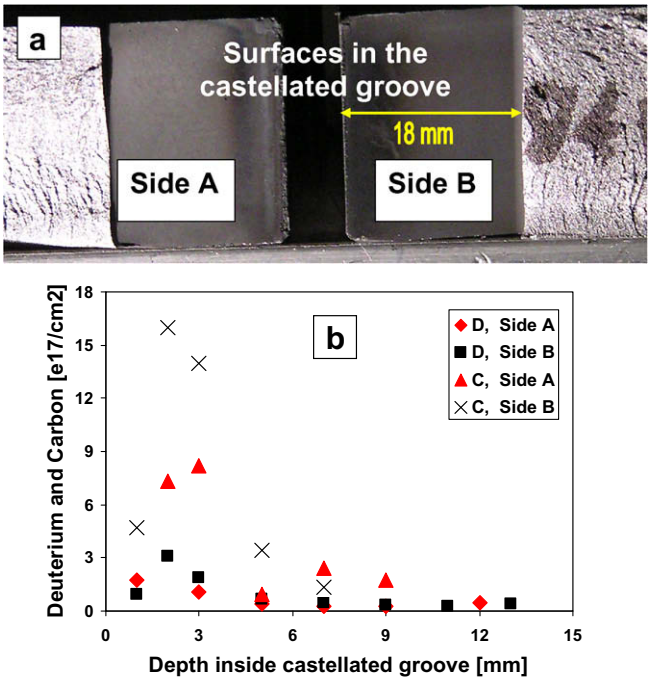

Fig. 3. Deposition pattern (a) and deposition profiles of deuterium and carbon (b) inside the groove.

When the composition of deposits in all grooves of the analysed tile is considered, the integrated amount of fuel in castellation is estimated at 6–10 $\times$ 10^{18} D atoms, which corresponds to about 30–50% of deuterium detected on the plasma-facing surfaces (compare to numbers given in the previous section), thus showing that the amount of retention and carbon deposition in castellated Be grooves is small. The most important fact coming from tile analysis is that the fuel presence is associated with co-deposited carbon. These results are highly consistent with data obtained for the castellated Be tile of the Mk-I divertor [8]. Secondly, on beryllium surfaces freshly opened by cleaving no oxygen was detected with the methods applied in the study, detection limit 1×10^{17} cm^2. It means that during six weeks after fracturing the tile, the oxide formation was limited to the very surface layer.

4. Concluding remarks

An important contribution of this work to tritium retention studies is the assessment of fuel inventory in castellated PFCs, i.e. in structures similar to those foreseen in ITER. As shown previously, much more pronounced fuel accumulation is measured in the presence of the CFC [7] structure than in the case of Be tiles [8]. Only small amount of fuel has been found in all studied regions of the castellated tiles: maximum 7×10^{17} cm^{-2}. One may suggest that during the long-term tile storage at least some fraction of the originally retained deuterium desorbed due to isotope exchange with hydrogen from water vapour. However, the small amount co-deposits and steep profiles of the co-deposited carbon indicate that the original deposition was not pronounced. The results from studies of co-deposits on carbon substrates from TEXTOR have shown that in 5 years the D content decreased by approximately 25–30% [13]. With this scenario for the Be tiles with C–D deposits,

the deuterium content would have decreased by 50–55% after 14 years.

There is no difference in the amount and profiles of deuterium and carbon between narrow toroidal castellated grooves (1 mm) and wider poloidal gaps (4–5 mm) separating tiles. This might suggest that operation with beryllium limiters and carbon on the central column and upper plates (as it was in this particular case) does not result in the formation of thick deuterated carbon films. It does not exclude, however, the formation of such films in other locations, as discussed previously [1–4].

These results from JET operated with the carbon wall and relatively small quantities of beryllium should not be immediately translated into quantitative predictions and far reaching conclusions regarding the material migration and fuel inventory in ITER because the planned material configuration in the divertor (W and CFC) and on the main chamber wall (Be) will be different than in any device operated to date. This will change both the scenario of material erosion and will influence fuel co-deposition. The clarification of this issue will be one of the main targets in the science program of JET when it is operated in the future with an ITER-like wall comprised of castellated beryllium limiters and tungsten lamellae-structured load bearing tiles in the divertor [5]. 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 in-vessel components remains, therefore, crucial for the operation of a reactor-class device.

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References

[1] P. Coad et al., J. Nucl. Mater. 290–293 (2001) 224.
[2] R.D. Penzhorn et al., J. Nucl. Mater. 288 (2001) 170.
[3] M. Rubel et al., J. Nucl. Mater. 313–316 (2003) 321.
[4] J.P. Coad et al., Nucl. Fusion 46 (2006) 419.
[5] G.F. Matthews et al., Phys. Scr. T128 (2007) 137.
[6] W. Daenner et al., Fusion Eng. Des. 61&62 (2002) 61.
[7] M. Rubel et al., Phys. Scr. T111 (2004) 112.
[8] M. Rubel, J.P. Coad, R.A. Pitts, J. Nucl. Mater. 367–370 (2007) 1432.
[9] A. Litnovsky et al., J. Nucl. Mater. 367–370 (2007) 1481.
[10] Krieger et al., J. Nucl. Mater. 363–365 (2007) 870.
[11] B. Emmoth, M. Rubel, E. Franconi, Nucl. Fusion 30 (1990) 1140.
[12] J.P. Coad, M. Rubel, C.H. Wu, J. Nucl. Mater. 241–243 (1997) 408.
[13] M. Rubel et al., J. Nucl. Mater. 363–365 (2007) 877.