EX/P4-24

Co-deposition and fuel inventory in castellated plasma-facing components at JET

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Keywords: D inventory, first wall, plasma facing components, plasma wall interactions

Abstract
This work is focused on the material migration into gaps between tiles and into castellation grooves on plasma-facing components from JET: water-cooled Mk-I divertors and belt limiter blocks. The essential results are summarised by the following: (i) co-deposition occurs up to a few cm deep in the gaps between the Mk-I tiles; (ii) fuel inventory in the the CFC tiles gaps exceeds that on plasma-facing surfaces by up to a factor of 2; (iii) in gaps between the beryllium tiles from the inner divertor corner and in belt limiter the fuel content reaches 30% of that on plasma-facing surfaces, whereas in the grooves of castellation in Be the fuel content is less than 3.0 % of that found on top surface; (iv) fuel inventory in the castellation of the Be divertor and limiter tiles is strongly associated with co-deposition of carbon. Implications of these results for a next-step device are addressed and the transport mechanism into the gaps is briefly discussed. The results presented here suggest that in a machine with non-carbon walls in the main chamber (as foreseen for ITER) the material transport and subsequent fuel inventory in the castellation would be reduced.

1. Introduction

An important mission of the JET tokamak is to optimise the wall and divertor structure. Over the years the geometry of PFCs in the main chamber and divertor has been modified several times in order to test a variety of operation scenarios under different wall conditions. This has also included the operation with bulk beryllium and Be-coated PFCs. The use of this material has been a distinct and unique feature of JET. In the period 1989-1992 castellated beryllium belt limiters (over 2000 blocks) were used. It was followed by the operation with a series of divertors including the Mk-I structure in which first carbon fibre composites (CFC), and then beryllium tiles were used. When fuel retention and material transport are considered, the simultaneous use of carbon and beryllium is very important for at least two reasons: (i) ITER plans to use both of these materials; (ii) it helps to distinguish the influence of chemical and physical processes in wall erosion and to draw conclusions regarding their impact on fuel inventory.

♦ See the Appendix to the paper of M. L.Watkins et al., Fusion Energy 2006 (Proc. 21st Int. Conf. Chengdu, 2006), IAEA, (2006).

The assessment of in-vessel fuel inventory is crucial for the safe and economic operation of a reactor-class device. The seriousness of this issue was realised only after the D-T campaigns performed in tokamaks with a carbon-based first wall: JET [1] and TFTR [2]. At JET operated with the Mk-IIA divertor structure, the most pronounced fuel accumulation was found in the inner divertor corner, in particular, in remote areas shadowed from direct plasma line-of-sight [3,4]. As a consequence of these measurements, detailed studies have been undertaken to assess the material fuel retention and transport to shadowed regions of other wall components of JET [4,5]. Emphasis has also been recently placed on studies of re-deposition and fuel retention in certain construction features similar to those foreseen for ITER, e.g. castellated tiles. The use of such structures for PFCs in ITER (divertor and the main chamber wall) is thought to be the best solution to ensure the thermo-mechanical durability and integrity of materials under high heat flux loads, especially when considering the use of metals (tungsten and beryllium) [6]. It is known, however, that eroded material may be transported and co-deposited together with fuel species in areas shadowed from the direct plasma impact. As a consequence, re-deposition occurs in grooves of castellation and in gaps separating PFCs. The enormous number of such grooves (over 1 million) that will be present in structures currently foreseen for ITER and the low level of tritium retention that can be tolerated (350 g), means that quantifying these processes in today's experiments becomes an urgent priority.

The deposition in gaps between PFC tiles has been studied in several machines [7-9]. However, still little is known about the fuel retention in narrow grooves. Until recently, beryllium belt limiters and Mk-I divertor at JET had been the only large-scale castellated structures used in fusion experiments. Analysis of deposition in the narrow castellated grooves and in tile gaps can therefore contribute to the assessment of material migration with the added, and important benefit, of an environment containing both C and Be. The general issue of material migration into gaps and the subsequent fuel retention are of crucial importance for ITER, in which there will be many such areas. All of them will be difficult to access by cleaning methods for fuel removal.

The contribution provides an overview and discussion of the results which have been obtained with particular emphasis on comparing the influence both of materials (carbon and beryllium) and PFC structure on fuel inventory. Such comparative studies allow the possible mechanism(s) governing transport to tile gaps and castellation grooves to be determined and the likely fuel retention to be expected in ITER as a consequence of the large scale use of castellation to be estimated.

2. Experimental

The study was carried out for: (i) beryllium belt limiters protecting the main chamber wall (castellation 14 mm deep and 1 mm wide) operated for 56000 s of plasma and (ii) for components of the Mk-I divertor. This water-cooled (base temperature 30-50 ^{o}C) structure was composed of small roof-shaped tiles: 72 mm long poloidally, 30-40 mm wide toroidally, 50 mm high and separated by gaps either 6 or 10 mm wide. As a result of roof-shaping and arrangement of the tiles with respect to the magnetic field lines, the lower part of a given tile was shadowed by the upper edge of the neighbouring tile. Views of the JET vessel with the Mk-I divertor and the arrangement of tiles can be found elsewhere [9,10]. The divertor was first operated for ~60000 s with CFC tiles and then for ~20000 s with castellated (6x6 mm with 6 mm deep and 0.6 mm wide groove) beryllium blocks. The divertor was first employed with carbon fibre composite tiles (CFC) for about 60000 s of plasma operation with 25600 s in X-point phases. Subsequently, the CFC was replaced by castellated (6x6 mm with 0.6 mm

wide and 10 mm deep groove) beryllium blocks and exposed for a further ~20000 s of plasma with 9150 s in X-point phases.

Tiles retrieved from the torus several years ago were recently examined *ex-situ* 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 performed with a 2.5 MeV $^3He^+$ beam, but to assess the thickness of deuterium-containing deposits, the ^3He beam energy was occasionally scanned in the range 0.7-3 MeV varying the information depth from 1.3 μm to approximately 10 μm, respectively. This highest attainable information depth was still too small in case of the thickest deposits (exceeding 25 μm). Enhanced proton scattering (EPS) with a 2.5 MeV H^+ beam was used to determine oxygen and carbon contents on the Be limiter tiles. To enable the analysis of co-deposits in the castellation two limiter tiles were cleaved in three places along the castellation. The study was performed on both sides of the grooves. In case of the Be divertor tile a few teeth were cut off to expose co-deposits in the castellation.

3. Results and Discussion

3.1. Deposition on the Beryllium Belt Limiter Tiles

The analyses of limiter blocks were performed on plasma-facing surfaces as well as on the side surfaces (i.e. surfaces in gaps between tiles) and in the grooves of castellation. In all cases beryllium, deuterium, carbon, oxygen and small quantities of metals (Ni+Cr+Fe originating from the Inconel vessel wall) were the main elements in the analysed surface layers but their amounts differed. Images in Fig. 1a,b show top surfaces of two fractured segments. The surface of some segments is rough because of Be melting. In some places, this process lead to bridging of the castellated grooves. From the EPS spectra in Fig 1c,d one infers that oxygen and carbon occur in fairly small quantities not exceeding $1.5x10^{18}$ cm^{-2}. This result is important because it shows that thick bulk beryllium oxide has not been formed even during the long-term storage of the tiles in air. The deuterium content on the plasma-facing surfaces ranges from $5x10^{16}cm^{-2}$ to $8x10^{17}$ cm^{-2}. Similar quantities of fuel have been found on the side surfaces of the tiles, i.e. on surfaces located in the gaps separating tiles.

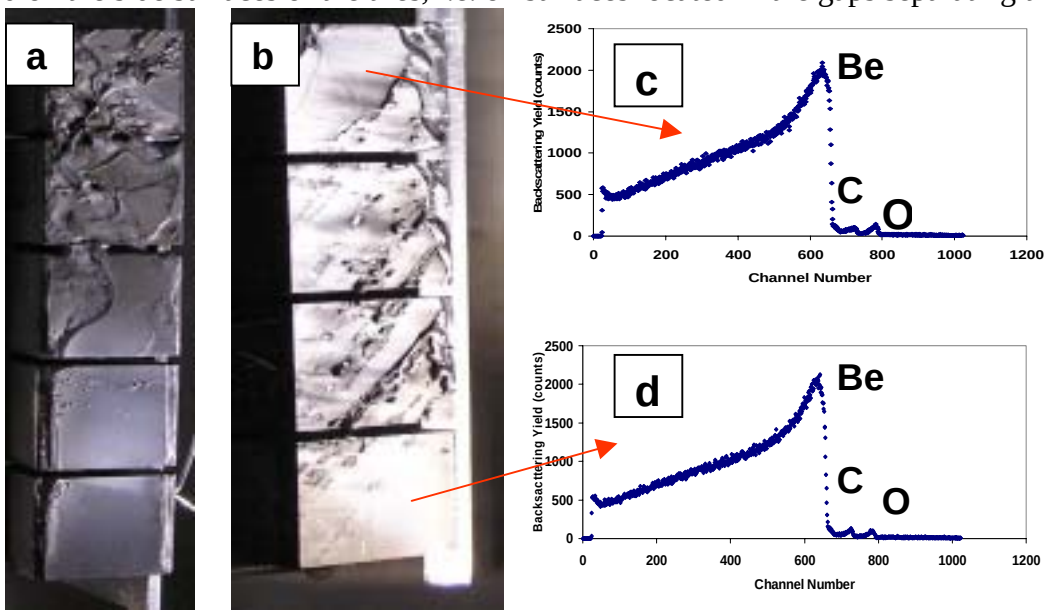

Fig.1. Plasma-facing surfaces of castellated beryllium tiles of the belt limiter (a) and (b); EPS spectra showing surface composition in two locations on the tiles (c) and (d).

Fig. 2a shows the deposition on two sides of the castellated gap, whereas in Fig. 2b deposition profiles of deuterium and carbon are plotted. Two essential conclusions can be inferred: (i) the deposition occurs in a narrow belt (up to 6 mm) close to entrance to the groove and (ii) the presence of fuel species is associated with the carbon co-deposition. The amount of deuterium in the deposition belt does not exceed $3x10^{17}$ cm^{-2} and it quickly decreases below the level $5x10^{16}$ cm^{-2}. The total amount of fuel detected in the grooves is less than 30% of that found on plasma-facing surfaces. Moreover, the analysis of cleaved surfaces did not result in detection of deuterium in bulk beryllium. Summary of these results leads to a conclusion that on the belt limiter tiles most of the stored fuel remains on plasma-facing surfaces and on the sides of the tiles, i.e. on surfaces located in the gaps between tiles.

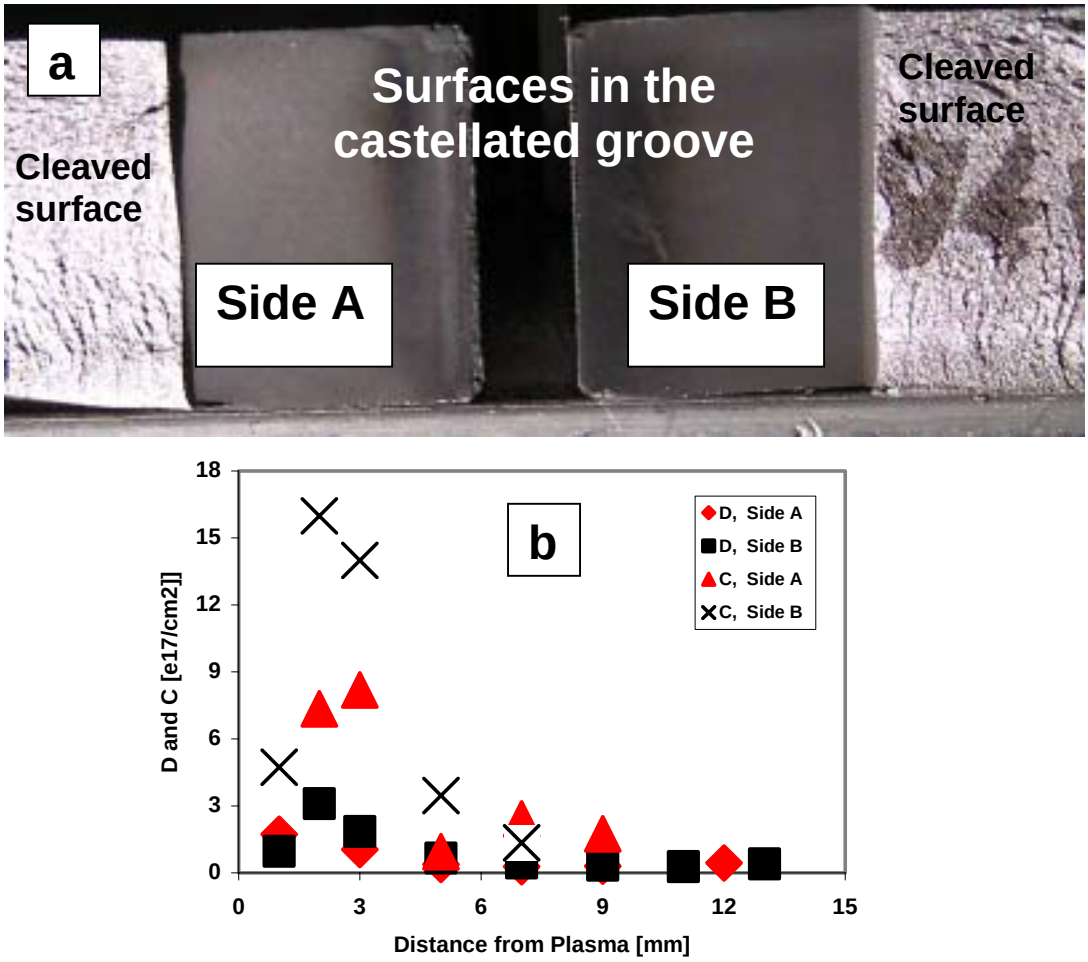

Fig. 2. Deposition pattern (a) and deposition profiles of deuterium and carbon (b) inside the castellated groove of the beryllium belt limiter tile.

3.2. Mk-I Divertor with CFC Tiles

Following the long-term operation with the CFC Mk-I divertor there was a significant deposition on both the plasma-facing surfaces and also the sides, i.e. on surfaces located in gaps between the tiles. The appearance of the plasma exposed CFC tiles with flaking co-deposits can be found in [9], whereas details regarding the elemental composition of plasma facing surfaces of the divertor floor have been reported in [10]. Only a brief summary of the most important facts is given here. On many tiles the maximum inventory, exceeding $5x10^{19}$ D at cm^{-2}, was detected in narrow (~5 mm wide) deposition belts in the shadowed part of the tile. There were no areas with D concentrations lower than $5x10^{17}$ cm^{-2}. Assuming toroidal

symmetry of the deposition pattern, the total inventory on the entire area of plasma facing surfaces on the divertor floor could be assessed on the level of 8.9×10^{23} D atoms. This value should be treated as a lower limit, because the information depth for deuterium with NRA using a 2.5 MeV $^3He^+$ beam was limited to approximately 8-8.5 μm whereas thicker co-deposits with very high deuterium content and the concentration ratio of deuterium-to-carbon, D/C ~ 0.5, were detected on some tiles.

Fig. 3 compiles deuterium profiles on poloidally and toroidally-oriented side surfaces, illustrating the most important result that the deposition has occurred on the entire lateral surface. In the depth analysed by NRA no significant difference between the deposition in toroidal and poloidal gaps has been observed. The deuterium content varies in the gap but on average it is not less than 3×10^{19} D cm^{-2}. On the same surfaces, the beryllium content is significantly smaller: up to 3×10^{18} Be cm^{-2} in the region 1-3 mm from the plasma facing edge and only $0.3–1.1 \times 10^{18}$ Be cm^{-2} in the lower part of the tile. One may therefore conclude that the fuel retention on the surfaces in gaps is associated with the transport of hydrocarbons and formation of carbon-rich co-deposits.

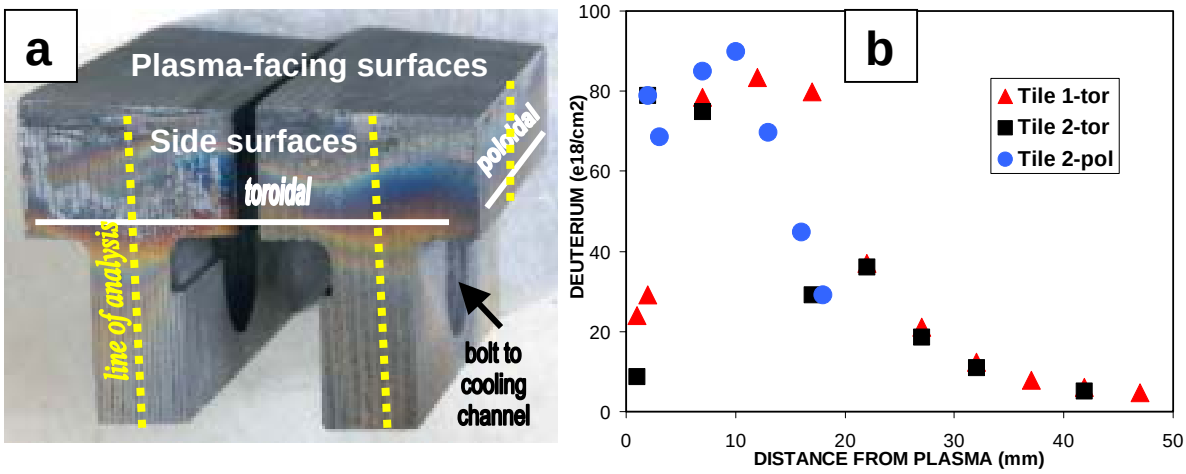

Fig. 3. Mk-I CFC tiles with flaking co-deposits (a) and the distribution of deuterium on surfaces in gaps between the tiles (b).

The detailed surface analyses have been performed only for a limited number of tiles, but visual inspection of the whole poloidal set of divertor components confirms that the deposition pattern found on these isolated units was fairly typical of the entire divertor. Based on these results, the total integrated amount of fuel on side surfaces is assessed at the level of 15.9×10^{23} D atoms, i.e. nearly twice as high as found on plasma-facing surfaces. This has profound implications for the use of tile castellation in the presence of carbon in the machine with high duty cycle.

3. 2. Mk-I Divertor with Beryllium Tiles

Fig. 4 shows a detailed image of the castellated Be tile. The inspection of these tiles has revealed three important features: (i) on the plasma-facing surfaces the deposition is particularly significant in the region shadowed by the adjacent tile; (ii) deposition on the side surfaces in gaps has occurred even at the bottom part of the divertor components, but no thick flaking co-deposits have been found; (iii) the castellation edges are still fairly sharp after long-term plasma exposure. Recent examination of the tiles from the inner divertor corner has permitted, for the first time, the assessment of material transport to the side surfaces and castellation grooves in Be (toroidally oriented groove has been studied). To make this study

possible, a few “teeth” of castellation were cut from a single tile, allowing comparison of the fuel content on two kinds of surfaces: located in the groove and in the gap between the tiles. Fuel contents in various areas are indicated in Fig. 4. The D concentration in the toroidal gap between the Be tiles decreases sharply by a factor of 3 at a distance of 6-8 mm from the plasma-facing surface. When compared to the carbon divertor the total D content is over 10 times lower than on CFC. In the narrow castellated grooves the fuel content decreases by roughly an order of magnitude over a distance of 5 mm, from $6x10^{17}$ cm^{-2} near the plasma facing edge to $8x10^{16}$ cm^{-2} in the region deeper in the groove. The corresponding decay length, λ, is approximately 1.5 mm. These concentrations are 70-100 times lower than those on plasma-facing surfaces [10] and in the gaps separating tiles.

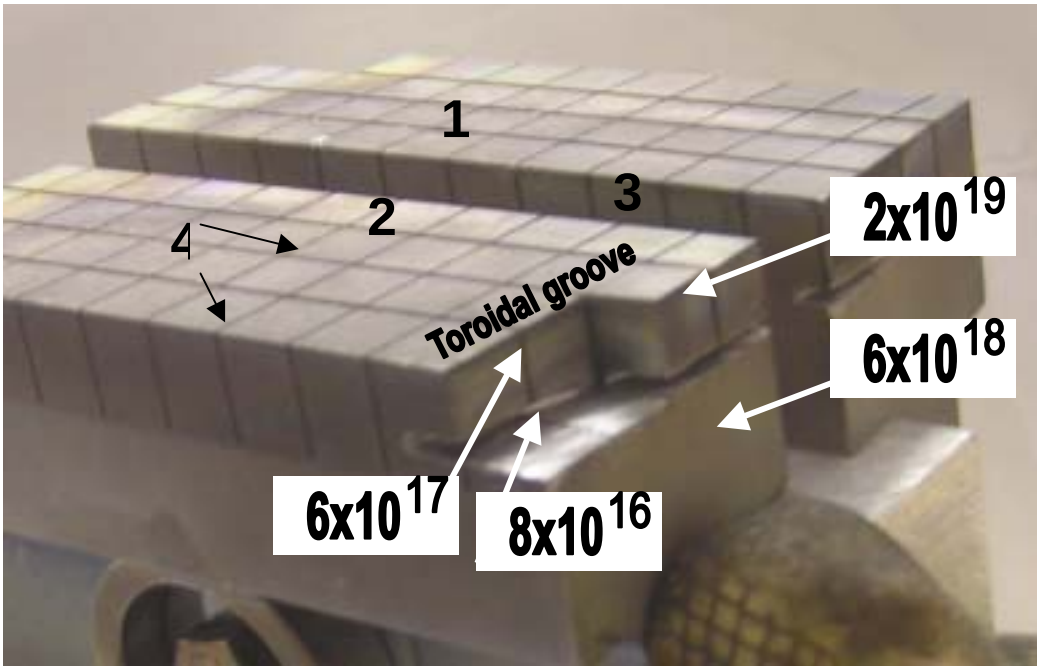

Fig. 4. Roof-shaped beryllium tiles of the Mk-I divertor: upper (1) and lower edge with an area shadowed by the adjacent higher tile (2);poloidal gap between tiles (3); castellation groves (4). Deuterium content on the side surfaces and in castellated grooves on the Mk-I beryllium tile is indicated.

From the analysis results compiled in [10], the integrated fuel content on the plasma-facing side of the inner corner Be tile could be estimated at the level of $9.32x10^{20}$ D atoms ($C_{surface}$). The recently determined integrated content in the castellation grooves of this tile is ~$0.22x10^{19}$ D atoms (C_{groove}), whereas $3.05x10^{20}$ D atoms (C_{gap}) are found on the surface located in the gap. One concludes that: (i) $C_{groove}/C_{surface}$ = 0.024 (2.4%) and (ii) $C_{gap}/C_{surface}$ = 0.33 (33%). The results clearly show that the fuel content in the shadowed regions of the Be divertor is distinctly lower than in the CFC structure.

The most important result is that deuterium on these tiles has always been detected together with carbon, indicating that co-deposition occurs in the gaps and in castellated grooves. Significant differences in deposition and fuel retention between the CFC and Be divertor cannot then be explained only by a three-times shorter operation time of Mk-I with Be tiles when compared to operation with the CFC components (see Section 2). All carbon found in the Be divertor must have originated from the erosion of the main chamber wall, so that the differences between the CFC and Be divertors may be attributed to the lack of the local carbon source in case of the Be divertor floor. The fact that the retention in castellated grooves is found to be small in comparison to that on plasma-facing surfaces and in gaps

between the tile demonstrates that the gap width plays an important role in determining the structure of the retention profile within the gaps.

Modelling of the transport to castellated grooves has been initiated using 2d-3v particle-in-cell (PIC) code self-consistently including full particle orbits and integration of Poisson's equation together with inclined target geometry. Details of the approach can be found in [11]. Fully ionised single-species (deuterium) plasma has been considered and following input parameters have been used in the simulation to model the typical conditions in the JET divertor near the strike point: electron density, $n_e = 7.5x10^{18}$ m^{-3}, electron temperature $T_e = 20$ eV, toroidal magnetic field B = 2.5 T. The groove width is 0.6 mm, the unperturbed ion flux onto the tile surfaces of $5.0x10^{22}$ $m^{-2}s^{-1}$ far away from the groove and the incidence angle $\alpha = 4^o$. The calculation indicates strong asymmetry in particle trajectories caused by *ExB* drift in the magnetized sheath [11]. For toroidally oriented grooves the decay length of deposited particles is 0.2 mm and a significant part of the flux may enter the groove. The missing flux is lost before entry due to strong focusing of the electric field on the tile corners.

The measurements with NRA and modeling prove short decay lengths of particles deposited in the groove. However, the experimentally determined e-folding length is 7 times greater, indicating that other processes than the direct flux deposition contribute to the transport phenomena during under the variety of the tokamak operation scenarios. This is most likely related to the deposition of neutral hydrocarbon penetration into the gap, because - as described above - deuterium in the groove is accompanied by the co-deposited carbon. Detailed kinetic simulations are under way to investigate the matter further.

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. Much more pronounced fuel accumulation is found in the presence of the CFC structure than in case of the Be tiles. This is measured both in various areas of the castellated limiter tiles and in the grooves of the tile from the inner divertor. The essential conclusion derived from the comparison of deposition in carbon and beryllium PFCs is that fuel inventory in gaps and castellation grooves is associated with the deposition of carbon. The difference in fuel content in the gaps between the tiles (less in Be than in CFC structure) also indicates that co-deposition in the divertor may be attributed to two factors: (i) transport of species eroded from the main chamber wall; (ii) transport from the local carbon source on PFC itself. The small inventory in the castellated grooves of beryllium tiles points to the probable influence of the gap width on the overall in-vessel fuel retention.

These results from JET operated with the carbon wall and relatively small quantities of beryllium should not 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 [12]. This will change both the scenario of material erosion and will influence fuel co-deposition and will be one of the main issues studied at JET when it is operated in the future with an ITER-like wall [13,14]. 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 in-vessel components remains, therefore, crucial for the operation of a reactor-class device.

Acknowledgements

This work has been conducted under the European Fusion Development Agreement and is partly funded by EURATOM, the UK Engineering and Physical Sciences Research Council, the Swedish Research Council (VR) under the Contract VR-621-2003-3170 and by the Grant Agency of the Academy of Sciences of the Czech Republic Grant No. GAAV B100430602".

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