Fuel Inventory in Shadowed Areas of the JET Divertors

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Components of JET divertors have been examined to in order to assess the fuel retention in areas shadowed from the direct plasma impact: castellation groves in beryllium blocks, gaps separating tiles of Mk-I, and the interior of the septum module in the Mk-II Gas Box (GB) structure. The results show that: (i) co-deposition occurs up to a few cm deep in the gaps between the Mk-I tiles; (ii) in these gaps the fuel inventory exceeds that on plasma facing surfaces by up to a factor of 2; (iii) in the grooves of castellation the fuel content is less than 3.0 % of that found on top surfaces; (iv) fuel retention in the septum of Mk-IIGB is insignificant in comparison to that detected in the shadowed region in the inner divertor corner. Implications of these results for a next-step device are addressed.

Keywords: fuel inventory, material migration, carbon, beryllium, JET

I. Introduction

Safety and economy of a fusion reactor operation are the driving forces for the assessment of fuel accumulation in plasma facing components (PFCs). The issue has been known for long time, but its full seriousness has been realised following the D-T campaigns carried out in tokamaks with a carbon-based first wall: JET [1] and TFTR [2]. It has also been recognised that the divertor geometry and the related power deposition profiles have a strong impact on the transport of eroded material and its co-deposition together with fuel species. For that reason, an important mission of the JET tokamak is to optimise the wall and divertor geometry and to test a variety of plasma operation scenarios. Over the last decade the main chamber wall and divertor of JET have been restructured several times. Since 1994 it has been consecutively operated with the following divertors: Mk-I with carbon fibre composite (CFC) and then with beryllium tiles and, three Mk-II structures with CFC tiles: Mk-IIA, Mk-II Gas-Box (GB) and Mk-II Septum Replacement Plate (SRP). Tiles in the main chamber wall are made with carbon and the wall is regularly coated by an evaporated beryllium layer. When the fuel retention and material transport studies are considered, the simultaneous use of carbon and beryllium is very important for at least two reasons: (i) these two materials are planned to be used also in ITER; (ii) it helps to distinguish chemical and physical processes in wall erosion and to conclude on their impact on fuel inventory.

Following the full D-T campaign at JET with the Mk-IIA structure the most pronounced

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[*] See Appendix to the paper by J. Pamela et al., Fusion Energy 2004 (Proc. 20th Int. Conf., Vilamoura 2004) IAEA, Vienna 2004.

accumulation was found in the inner divertor corner, in particular, in remote areas shadowed from the direct plasma line of sight [3-5]. After this experience detailed studies have been undertaken to assess the material transport and fuel retention to shadowed regions of other JET divertors. The emphasis was given to studies of re-deposition and fuel retention in certain construction features similar to those foreseen for ITER: castellated tiles and dome. The issue of massive co-deposition in such areas is of crucial importance for ITER because they are difficult to access by cleaning methods for fuel removal.

The castellated structure of tiles in the ITER divertor 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). Until recently, the Mk-I divertor has been the only large-scale castellated structure used in fusion experiments. Therefore, analysis of deposition in castellated grooves and in gaps between tiles of Mk-I may contribute to the assessment and improved understanding of material transport and mixing in an environment containing both C and Be.

Strong emphasis given to the analysis of surfaces inside the gas box module of the Mk-IIGB divertor is justified by the fact that its structure has been, so far, the closest to the planned geometry of the dome in the ITER divertor. The inner part of septum has had a large surface area shadowed from the direct impact of confined plasma. Therefore, this region could be deemed as a potential trap for co-deposition associated with significant fuel inventory.

The main objective of this paper is to provide an overview of fuel inventory in structures resembling those to be implemented in ITER: (i) castellation which was used in Mk-I and (ii) the interior of the septum. The deposition pattern in the inner corner of Mk-IIGB and a brief comparison of deposition in two Mk-II divertors (Mk-IIA and Mk-GB) will also be given.

II. Experimental

II.A. Mk-I Divertor

Mk-I was a water-cooled structure composed of small roof-shaped tiles: 72 mm long poloidally, 30-40 mm wide toroidally, 50 mm high and, as shown in Fig. 1, separated by gaps up to 10 mm wide. As a result of roof-shaping and arrangement of the tiles with respect to the magnetic field lines a lower part of a given tile was shadowed by the upper edge of the adjacent one. The divertor was operated first for a total of 59661 s with CFC and then for 20303 s with castellated (6x6 mm with 6 mm deep and 0.6 mm wide groove) beryllium blocks. Images in Fig. 2 feature the structure with CFC (a) and Be (b) tiles.

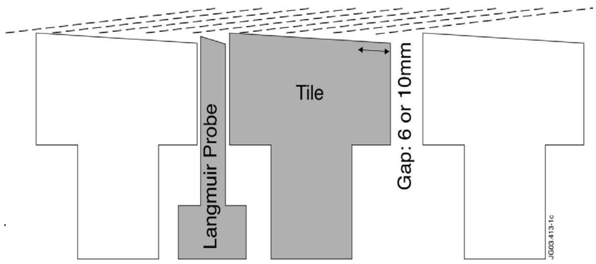

Fig. 1.Arrangement of tiles in the Mk-I divertor.

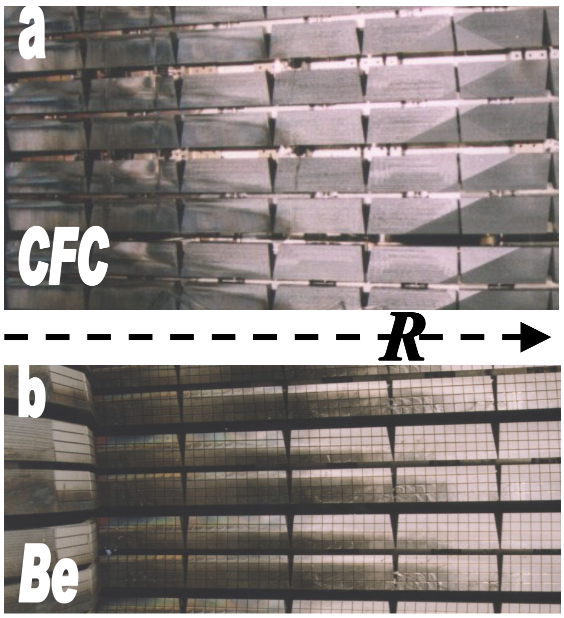

Fig. 2. Inner part of the Mk-I divertor with CFC (a) and castellated Be (b) tiles

II. B. Mk-IIGB Divertor

Fig. 3 a shows the cross-section of Mk-IIGB and the most typical position of strike points on the vertical divertor plates. In Fig. 3 b the scheme of the septum module is depicted. Tile 5 and divider plates were positioned toroidally, whilst the three support plates were oriented poloidally. Experimental campaigns with that divertor lasted for about 18 months in the period 1999-2001. The total operation time was 193291 s including 94299 s of X-point plasma.

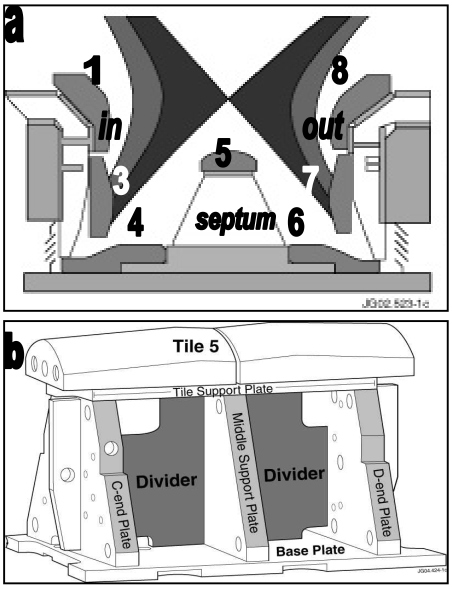

Fig. 3. Cross-section of the Mk-II Gas Box divertor (a) and the structure of septum (b).

II. C. Analysis methods

Tiles retrieved from the torus were examined *ex-situ* by means of ion beam analysis (IBA) techniques. Nuclear reaction analysis (NRA) was used to quantify and map surface contents of deuterium [$^3He(d,p)\alpha$], beryllium [$^3He(^9Be,p)^{11}B$] and carbon 12 and 13 isotopes: [$^3He(^{12}C,p)^{14}N$] and [$^3He(^{13}C,p)^{15}N$]. Most of the D, Be and C analyses were carried out 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 giving the information depth from 1.3 μm to approximately 10 μm, respectively.

III. Results and Discussion

III. A. Deposition in Gaps and in Castellation

III. A. 1. Mk-I Divertor with CFC Tiles

Fig. 4 shows the appearance of the CFC tiles after the long-term operation. There is a significant deposition on the plasma facing surfaces and also on the sides, i.e. on surfaces located in gaps between the tiles.

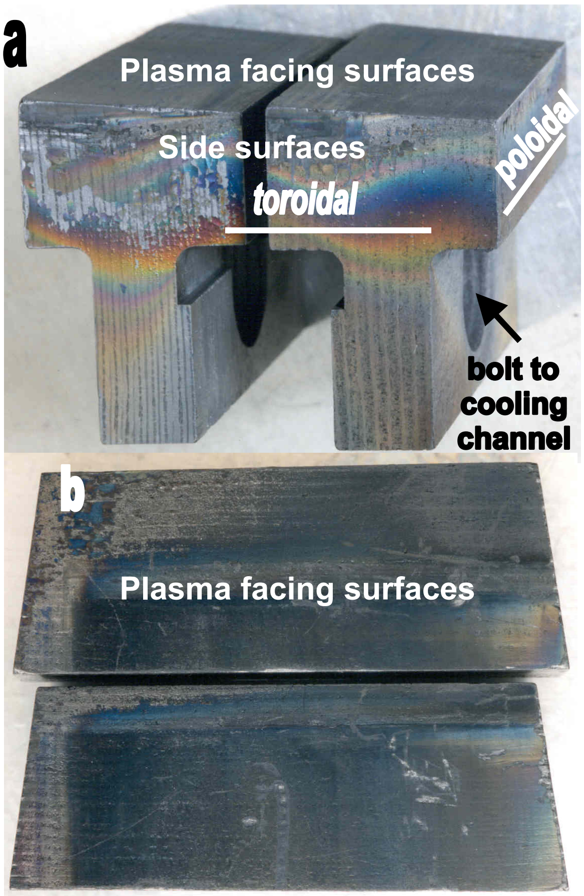

Fig. 4. CFC tiles of the Mk-I divertor: general view (a), plasma facing surfaces (b).

Details regarding the elemental composition of plasma facing surfaces of the divertor floor can be found in [6]. Only a brief summary of the most important facts is given here: (i) on many tiles the maximum inventory exceeding 5×10^{19} D at cm^{-2} was detected in narrow (approx. 5 mm wide) deposition belts in the shadowed part of the tile (see Fig. 1); (ii) there were no areas with the D concentration 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 (11.1 m^2, projected area of gaps subtracted) could be assessed on the level of $8.86x10^{23}$ 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. 5 shows deuterium profiles on poloidally and toroidally-oriented side surfaces. The most important result is that the deposition has occurred on the entire side surface. The deuterium content varies in the gap but on average it is not less than $3x10^{19}$ D cm^{-2}. On the same surfaces, the beryllium content is significantly smaller: up to $3x10^{18}$ Be cm^{-2} in the region 1-3 mm from the plasma facing edge and only $0.3–1.1x10^{18}$ Be cm^{-2} in the lower part of the tile. Therefore, one concludes that the fuel retention on the surfaces in gaps is associated with the transport of hydrocarbons and formation of carbon-rich co-deposits. The analyses have been carried out for a limited number of tiles, but the visual inspection of the whole poloidal set of divertor components has provided the evidence that such a deposition pattern was typical. Based on these results, the total integrated amount of fuel on side surfaces is assessed at the level of $15.9x10^{23}$ D atoms, i.e. nearly twice greater than on plasma facing surfaces.

III. A. 2. Mk-I Divertor with Beryllium Tiles

Images in Fig. 6 a-c show details of Be tiles from the Mk-I divertor. Two features are important: (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 has occurred even at the bottom part of divertor components

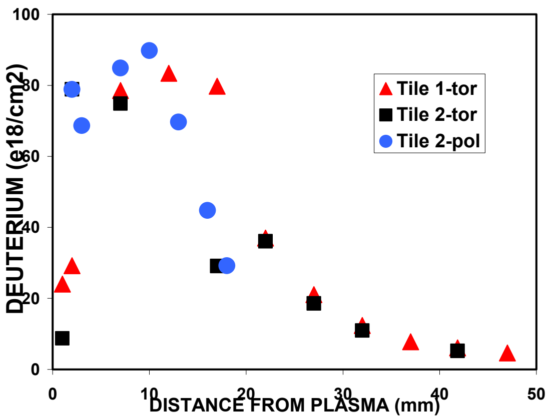

Fig.5. Distribution of deuterium on side surfaces in the gaps between the Mk-I CFC tiles

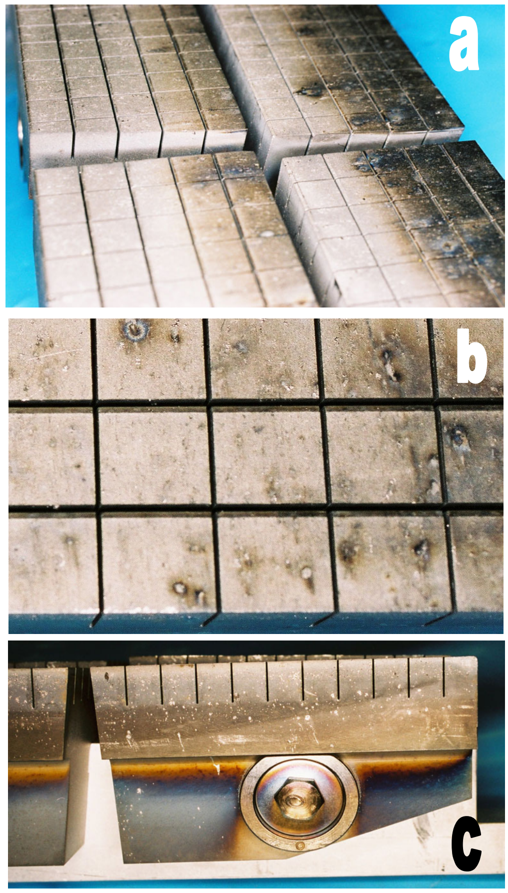

Fig. 6. Castellated Be tiles from Mk-I: plasma facing surfaces (a) and (b), side surfaces (c).

As previously reported [6], the greatest deuterium content on plasma facing surfaces was found on tiles from the inner divertor corner:

from $3x10^{18}$ cm^{-2} in the non-shaded part to over $3x10^{19}$ cm^{-2} in the region shaded by the adjacent tile. Recent examination of these inner corner tiles has permitted, for the first time ever, the assessment of material transport to the side surfaces and castellation grooves in beryllium. In order to enable the studies a few teeth of castellation were cut off. This allowed for comparison of the fuel content on two kinds of surfaces: located in the groove and in the gap between the tiles. Fig. 7 shows a detailed image of the examined tile.

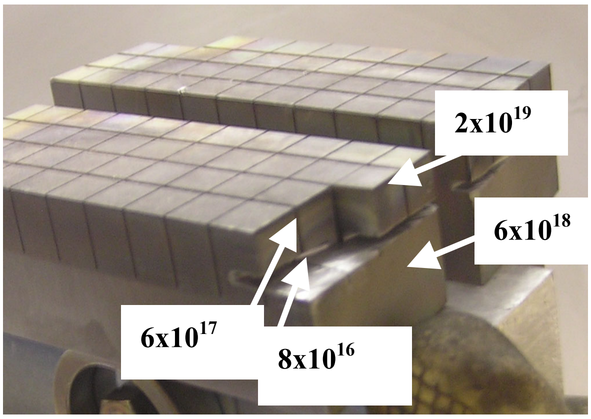

Fig. 7. Deuterium on side surfaces and in castellated grooves on the Mk-I beryllium tile.

The most important result is that deuterium has always been detected together with carbon indicating that its co-deposition occurred in the gaps and in castellated grooves. The shape of the deposition profile in the gap area is qualitatively similar to that on CFC tiles (see Fig. 6), but the total D content is distinctly smaller than on CFC tiles. This may be attributed to two factors: (i) a shorter operation time of Mk-I with Be tiles than with CFC components; (ii) a lack of the local carbon source in case of the Be divertor floor. The latter means that the carbon found in the divertor originated from the erosion of the main chamber wall. 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. These concentrations are 70-100 times lower than those on plasma facing surfaces [6] and in the gaps separating tiles.

From the analysis data in [6] the integrated fuel content on the plasma facing side of the inner corner Be tile is estimated to be $9.32x10^{20}$ D atoms ($C_{surface}$). As recently determined, the integrated content in the castellation grooves of this tile is around $0.224x10^{19}$ 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 indicate that the fuel content in the shadowed parts of the beryllium divertor is distinctly lower than in the CFC structure. Secondly, the retention in the castellated grooves is small in comparison to that on plasma facing surfaces and in gaps between the tiles.

III. B. Deposition in the Mk-IIGB Divertor

III. B. 1. Vertical and Horizontal Tiles

Plots in Fig. 8 a-c show deposition profiles of deuterium and beryllium in the inner leg of the Mk-IIGB divertor, i.e. on Tiles 1, 3 and 4. One observes significant variations in the distribution of species. On Tiles 1 and 3 it is related to the flux deposition profiles. The fuel content is in the range from $3x10^{18}$ to $1.8x10^{19}$ D atoms cm^{-2}. The situation changes drastically on Tile 4 where two distinct regions are clearly distinguished: (i) a very thick deposit in the area shadowed by Tile 3 and (ii) a moderate level on the part open to the plasma. As previously reported [7], the deposit thickness in the shadowed region reaches approximately 60 microns, whereas the results in Fig. 2 c reflect the concentration in a layer 8-8.5 μm thick, i.e. the depth accessible by IBA. Therefore, the actual total content of D in that region is at least 6 times greater and it amounts to about 7.2 x 10^{20} at cm^{-2}. Moreover, lack of Be in the shadow indicates that the deposit contains thick carbon film resulting from a long-range transport of hydrocarbons. Such a deposition pattern was identified and discussed following the studies of the Mk-IIA divertor [4,5].

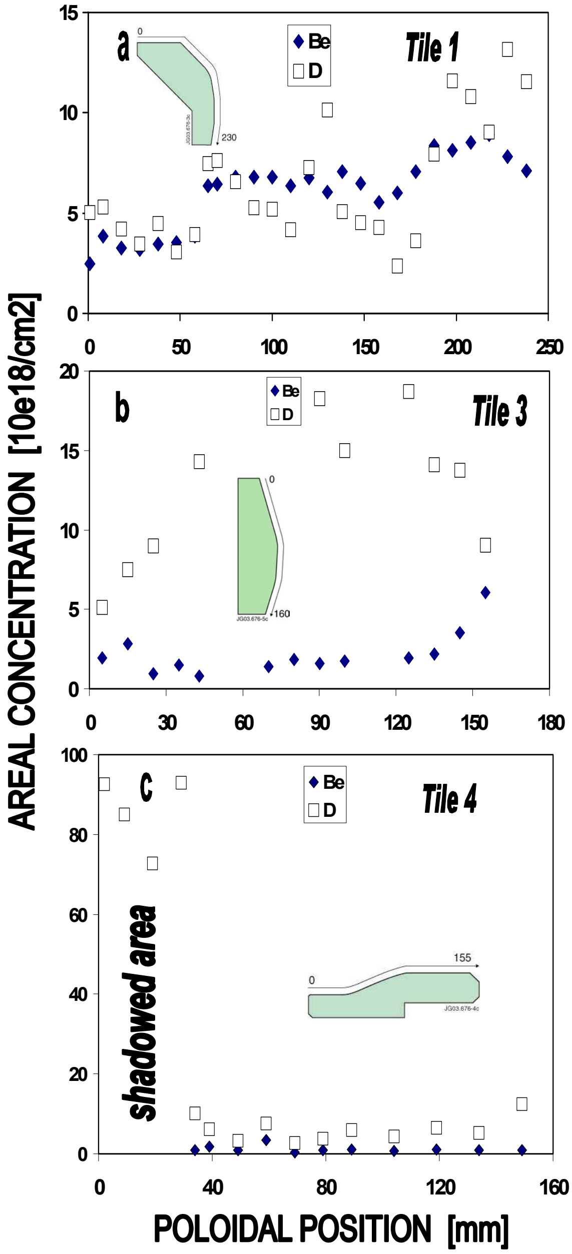

Fig. 8. Poloidal distribution of deuterium (open squares) and beryllium (filled diamonds) on Tiles 1, 3 and 4 in the inner leg of Mk-IIGB.

Deposition and fuel retention on plates from the outer divertor (6, 7 and 8) has been significantly lower than in the inner leg. The only exception was a narrow belt of strong deposition on Tile 6 in the region shadowed by tile 7. Assuming a toroidal symmetry in deposition, the integrated amount of retained fuel on all tiles would be around 4.68×10^{24} D atoms including as much as 3.63×10^{24} D atoms trapped in the shadowed region on Tile 4. These numbers, however, should be treated as a lower limit of retention, because for technical reasons, the material deposition and fuel retention in some areas of the shadowed region in the inner divertor could not be studied.

III.B.2. Deposition in the Septum

Surface morphology of all vertical support and divider plates of one septum module was examined. Fig. 9 a shows the appearance of the support plate (C-end, see Fig. 3 b). A dusty deposit with high fuel content (25×10^{18} D atoms cm^{-2}) is found only in a narrow belt, 10-15 mm wide, at the plate edge facing towards the inner divertor channel. Similar deposition pattern has been observed on other support plates. In some cases erosion by arcing has also been noted. No significant deposition zone is formed on the outer divertor side, where a constant level below 1×10^{18} D atoms cm^{-2} is measured. The Be content across the plate is in the range $0.5\text{-}1.5 \times 10^{18}$ atoms cm^{-2} and it is not increased at the edge facing towards the inner divertor. This indicates that the thick co-deposit is mainly composed of a hydrogenated carbon film. There are some small variations in the D and Be levels when all the analysed plates are compared, but the general distribution pattern remains unchanged.

Fig. 10 shows a divider plate and the amount of deposited species. These results are representative for all the plates under examination, both for the inner and the outer divertor side. Again, the amounts of D and Be are small and the coverage is fairly uniform over that surface: $0.5\text{-}1.5 \times 10^{18}$ atoms cm^{-2}. The analyses of the support and divider plates have given highly consistent results showing low level of co-deposition and, as a consequence, small fuel retention.

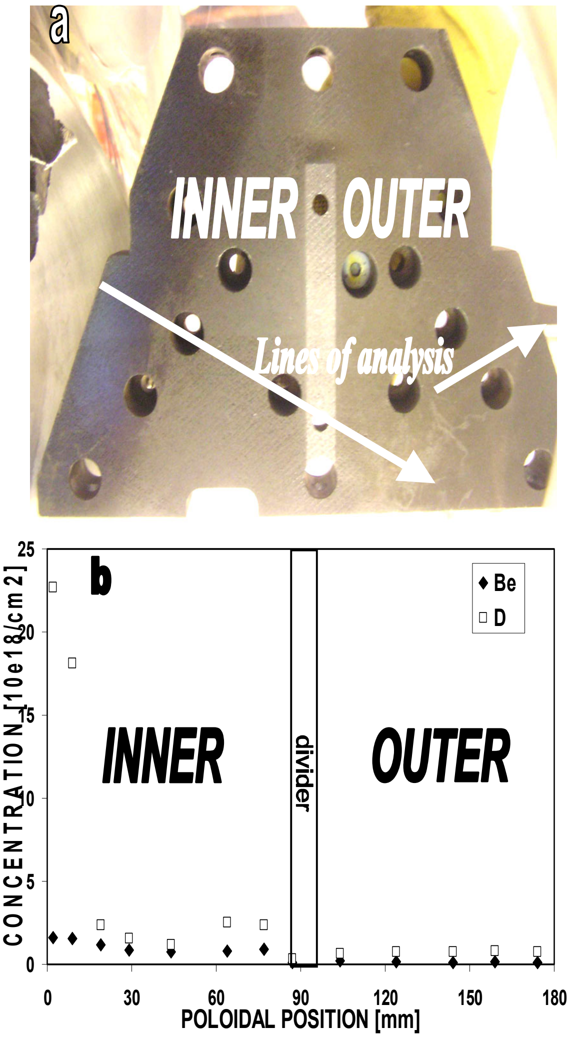

Fig. 9. A poloidal support plate of the gas box module (a) and distributions of D and Be on the plate (b).

The total surface area of the gas box components is 14.85 m^2. Taking into account the average D content of $1.5x10^{22}$ at cm^{-2} outside the deposition belts and $12x10^{18}$ at cm^{-2} in the belts and assuming the toroidal symmetry of deposition in the septum, the extrapolated upper limit of fuel retention would be $3.28x10^{23}$ atoms. When absolute numbers are compared, this amount is over one order of magnitude smaller than that assessed in the shadowed part of Tile 4, i.e. $36.3x10^{23}$ D atoms retained on an area of around 0.5 m^2.

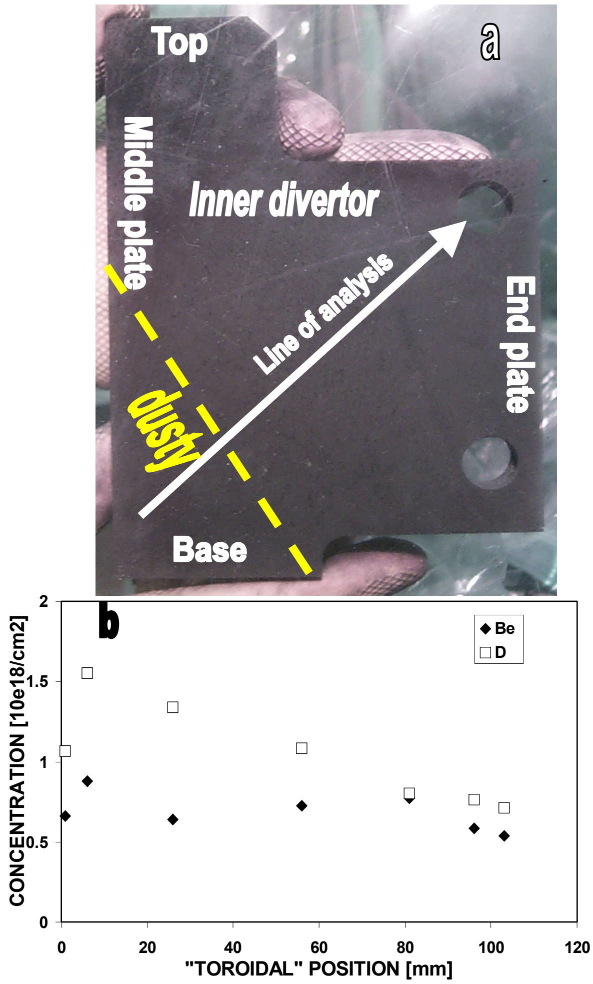

Fig. 10. A divider plate of the gas box (a) and distributions of D and Be on the plate (b).

III.C. Comparison of Deposition in Mk-IIA and Mk-IIGB Divertors

Mk-IIA and Mk-IIGB divertors were in operation for a similar period of time. The majority of retained fuel was found in the inner divertors, especially in the zone shadowed from the direct plasma impact. Fig. 11 shows the deposition histogram on the plasma facing surfaces. A brief comparison of the deposition allows for the statement that in both cases greater deposition has occurred in the inner than in the outer divertor. The exception is a high deposition region on Tile 6 being in the shadow of Tile 7 in Mk-IIGB. However, there is a significant difference in the deuterium-to-beryllium concentration ratio when the composition of the surface layer is compared: $C_D/C_{Be} > 1$ in Mk-IIGB and $C_D/C_{Be} < 1$ in Mk-IIA. The difference can be primarily attributed to the end phases in operation with the divertors. Immediately after the D-T experiment in JET with Mk-IIA, the vessel was intensively cleaned-up by various means in order to reduce the tritium retention [1]. This certainly reduced also the deuterium content in the surface layer. On the contrary, no special cleaning procedure was done after the operation with Mk-IIGB because discharges were fuelled only with deuterium and helium [8].

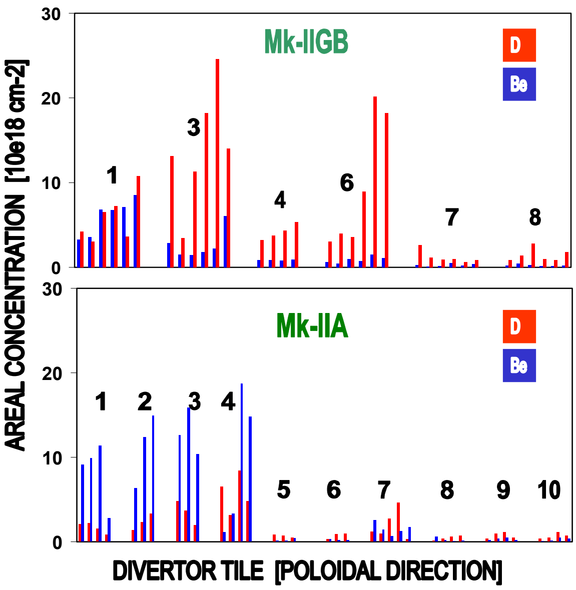

Fig. 11. Comparison of deposition on vertical and horizontal tiles of the Mk-IIA and Mk-IIGB divertors. Results for shadowed regions on Tile 4 are not included.

IV. Concluding Remarks

An important contribution of this work to tritium retention studies is the assessment of fuel accumulation in structures similar to those foreseen in ITER: castellation of PFC and dome. They have large areas shadowed from the direct plasma impact and, potentially, can trap a lot of co-deposited fuel. Until recently, the only experience with large-scale structures of this kind has been at JET operated with castellated tiles in the Mk-I divertor and with the gas box module in Mk-IIGB.

The essential conclusion derived from the comparison of deposition in the CFC and Be Mk-I divertors is that fuel inventory in gaps and castellation grooves is associated with the deposition of carbon. Much more pronounced accumulation has occurred in the presence of the CFC structure than in case of the Be tiles. It is not intended here to give predictions for ITER, but one may suggest that in a machine with non-carbon walls in the main chamber the material transport and consequential fuel inventory in the castellation would be reduced. The difference in fuel content in the gaps between the tiles (less in Be than in CFC structure) also indicate that the co-deposition in the divertor is attributed to two factors: (i) transport of species eroded from the main chamber wall; (ii) transport from the local carbon source in the divertor itself. Small inventory in the castellated grooves of beryllium tiles points to the probable role of the gap width on the in-vessel fuel retention. This issue will be the subject of future studies.

The assessment of the deposition and retention inside the gas box was particularly important because many diagnostics in ITER plan to gain access through the septum structure and excessive deposition would obscure viewing capability of optical systems. The measurements clearly indicate that the septum region should not be considered as a major trap for transported and co-deposited material. Fuel inventory in the septum is insignificant in comparison to that in the shadowed areas in the inner divertor corner. From the very small deposition found on the divider plates, one also concludes that the direct cross-divertor transport of material (neutral atoms and molecules) through the dome, from the outer to the inner leg, can be considered negligible.

These results from JET operated with the carbon wall and evaporated beryllium coating in the main chamber should not be immediately translated into conclusions and predictions regarding the material migration and fuel inventory in the ITER divertor. However, the results allow for some optimism showing that no dramatic deposition inside the dome structure would occur, especially, since the use of carbon in the next-step machine is planned to be strongly limited. According to present materials choice for in-vessel components of ITER [9], the source of carbon will be significantly reduced (bulk Be tiles on the main chamber wall). This will change the scenario of material erosion and will also influence the co-deposition of fuel.

Acknowledgements

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