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Comparison of tritium retention and carbon deposition in JET and JT-60U

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Abstract

Profiles of T retention in the Mark IIA divertor of JET and H/D and T retention in the W-shaped divertor of JT-60U are compared. Hydrogen (H, D and T) is retained in carbon deposited layers with nearly constant concentration throughout the layers, except high energy triton directly impinging into more than 1 μm in depth. However, carbon deposition profiles and hydrogen retention are strongly influenced by geometrical structure of the divertor and tile alignment as well as by magnetic field lines. Carbon deposition on the divertor base tile in JET shows stripes parallel to magnetic filed lines, suggesting a direct plasma deposition process. In JT-60U, the temperature of the co-deposited layers during operation plays critically important role on the hydrogen retention. It seems possible to reduce tritium inventory significantly by increasing the surface temperature of the plasma facing components.

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Keywords: Tritium and tritiated materials; Plasma–material interaction; Fusion reactor material

1. Introduction

Tritium co-deposition with carbon is a serious safety concern for ITER. Current predictions indicate that the in-vessel tritium inventory limit will be reached within only 60–150 ITER discharges [1,2]. However, past experiences also suggest that hydrogen retention behaviour is machine (and/or operation condition) dependent. In addition, retention data for H, D and T differ due to their different origins, such as residual gas, gas puffing, pellet injection, neutral beam injection or the D–D reaction [3]. The locations and temperatures of deposited layers must also play a certain role.

The recently developed imaging plate (IP) technique has given a detailed map of tritium deposition and made possible the direct comparison of tritium distribution and carbon deposition patterns [3–5]. This motivates

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us to compare the retention behaviour of H, D and T in different tokamaks and to investigate the origin of the differences. Taking the available data on carbon deposition and retention of H, D and T in the plasma facing carbon materials of JET and JT-60U into account, we discuss tritium retention in a carbon based D–T reactor.

2. Analyzed tiles

The CFC tiles measured here were a whole poloidal set of the JET Mark IIA divertor used in the first deuterium/tritium experimental (DTE1) campaign in which the tritium levels had already been measured by the combustion method as given in Fig. 1(a) [6,7].

A poloidal set of CFC divertor tiles were retrieved from the P5 section of the W-shaped diverter of JT60U. In the operation periods from June 1997 to October 1998, about 4300 discharge (3600 D–D discharge) experiments were made. More than 300 shots were with neutral beam injection (NBI) heating with power of 14–23 MW. During this period, a total amount of 18 GBq of tritium was produced and distributed in the vacuum vessel and/or exhausted [8]. Before ventilation, 700 discharges in hydrogen were carried out to remove tritium from plasma facing materials (PFM). All removed tiles were analyzed by imaging plate (IP), thermal desorption spectroscopy (TDS), secondary ion mass spectrometry (SIMS), X-ray photoelectron spectroscopy (XPS) and scanning electron microscopy (SEM) to measure tritium retention, H and D retention, depth profiling of H and D, surface characterization, and erosion/deposition measurements, respectively [9].

The imaging plates (IP) used here were BAS-TR2025, which were manufactured by Fuji Photo Film Co. Ltd. for detection of low energy β-rays such as those generated by tritium. The surface of the IP was in contact with the graphite tile surface in a dark shielded room. In order to avoid the contamination of the IP with tritium, a thin (2 μm) polyimide-film was inserted between the tile and the IP. The insertion reduces the sensitivity by a factor of about 10 and, in consequence, tritium within the depth of about a few microns can be analyzed. After the exposure, the IP was processed by an imaging plate reader, Fuji BAS-2000 or BAS-2500, to obtain the digitized photo-stimulated luminescence (PSL) intensity and its two dimensional mapping (“tritium image”) with areal resolution of about 100 μm. The PSL intensity, which is a measure of absorbed energy in the IP, cannot be directly converted to absolute tritium level (Bq/cm^2) but is nearly proportional to the tritium surface concentration. Details of tritium detection by IP technique were published elsewhere [4,5].

3. Results and discussion

3.1. Global tritium retention in the JET Mark-IIA divertor

Fig. 1 compares tritium images of a whole set of the JET Mark II-A diverter tiles (b), two poloidal line profiles of higher and lower tritium intensity along the red and blue lines respectively on the images (c), and tritium amount retained in the drilled cylinders cut from the tile, which was determined by the combustion method (d) [10]. As seen in (b) and (c), tritium retention is higher in the divertor target tiles both for the inner (BN4) and outer (BN7) horizontal divertor tiles, particularly at plasma shadowed region that was covered by thick carbon deposition. Heavy deposition was also seen on the bottom side of the inner divertor target tile (IN3). Although the IP technique can detect tritium within several microns depth, the IP line profiles agree well with that determined by the combustion method, suggesting that most of tritium was retained in near surface regions. Hence the PSL intensity and the tritium level determined by the combustion are compared in Fig. 2(a). One can see good agreement between the two, indicating that most of tritium was retained within a few μm detectable by IP. However, some deviations at lower concentrations are also appreciable. That is because the areas with low tritium content have less deposition and the contribution from tritium retained deeper into the surface or in the bulk becomes significant.

Since the photo stimulated luminescence (PSL) intensity obtained by the IP is relative, we can convert the PSL intensity into absolute tritium level using the linear relationship in Fig. 2(a). Then we have integrated the tritium level on the whole surface areas of tiles, and the integrated levels for all tiles are compared with the extrapolated tritium retention from the combustion methods, assuming homogeneous tritium distribution in Fig. 2(b). The agreement between the two is very good, though there is some difference. This clearly indicates that most of tritium is retained in the deposited layers. The difference is most significant on tile BN4 on which tritium distribution was quite inhomogeneous, due to localized thick deposition, particularly on the plasma shadowed area.

3.2. Detailed tritium distribution on the inner divertor

A detailed tritium image including the plasma shadowed area of tile BN4 is given in Fig. 3, where deposited layers on a limited area were removed by an adhesive tape. The IP image for the rear side of the removed layers is also shown. One can clearly see that the tritium image of the rear side of removed deposits is quite similar to that of the original surface. The removed layers retained 2.67 GBq of tritium which was determined by

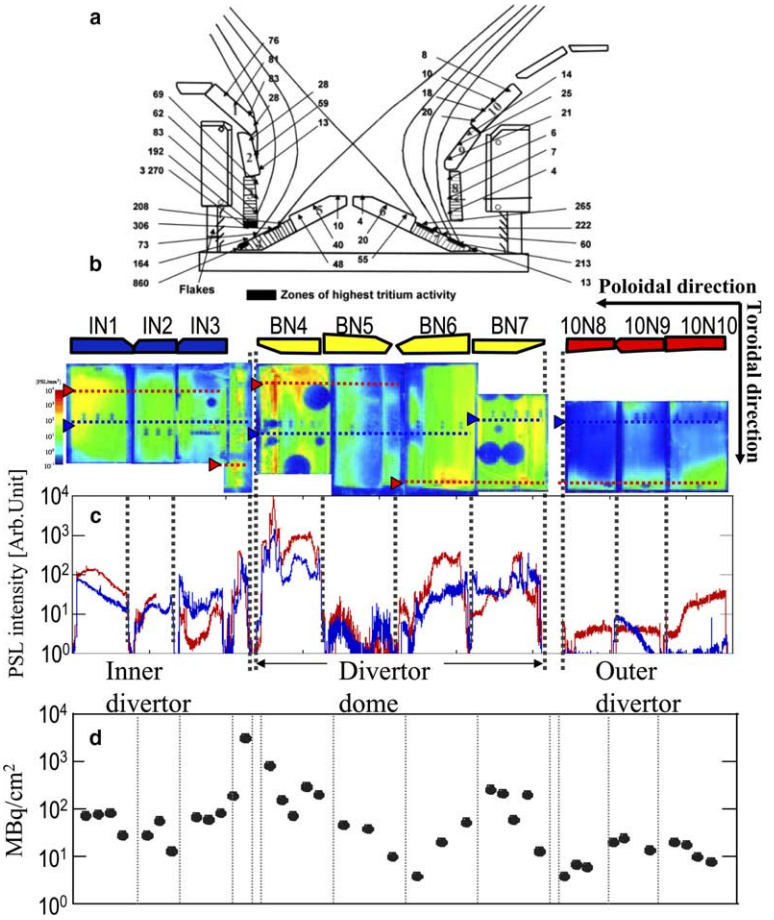

Fig. 1. Tritium distribution on JET Mark II-A divertor tiles. (a) Schematic drawing of JET Mark-II divertor. Numbers in the figure indicate tritium activity in MBq measured by combustion method (After *R.-D. Penzohorn et al. [10]). (b) Tritium areal profiles given by imaging plate technique. Red and blue areas respectively correspond to high and low tritium retention areas. (c) Poloidal line profiles of tritium given by PSL (Photo Stimulated Luminescence) intensity at higher (red) and lower (blue) tritium retained area. (d) Tritium amount determined by combustion method for 3–5 drilled holes in each tile in toroidal direction. (After *R.-D. Penzohorn et al. [10]).

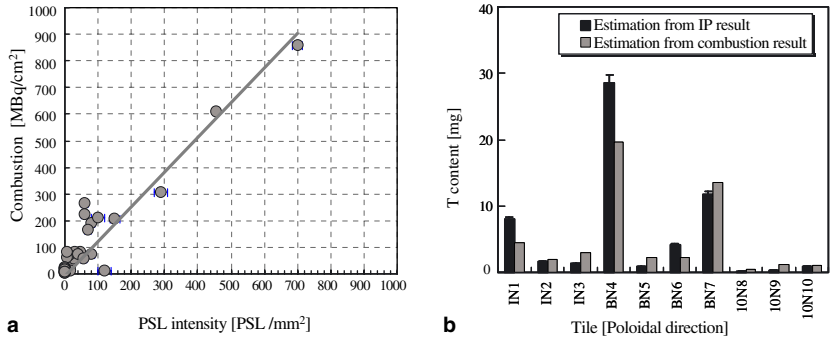

Fig. 2. (a) Comparison of PSL intensities of neighboring areas of drilled holes determined by IP and tritium levels of the dilled cylinders determined by the combustion method [10]. (b) Comparison of integrated tritium retention for all divertor tile surfaces determined by IP and extrapolated tritium retention from the combustion measurements.

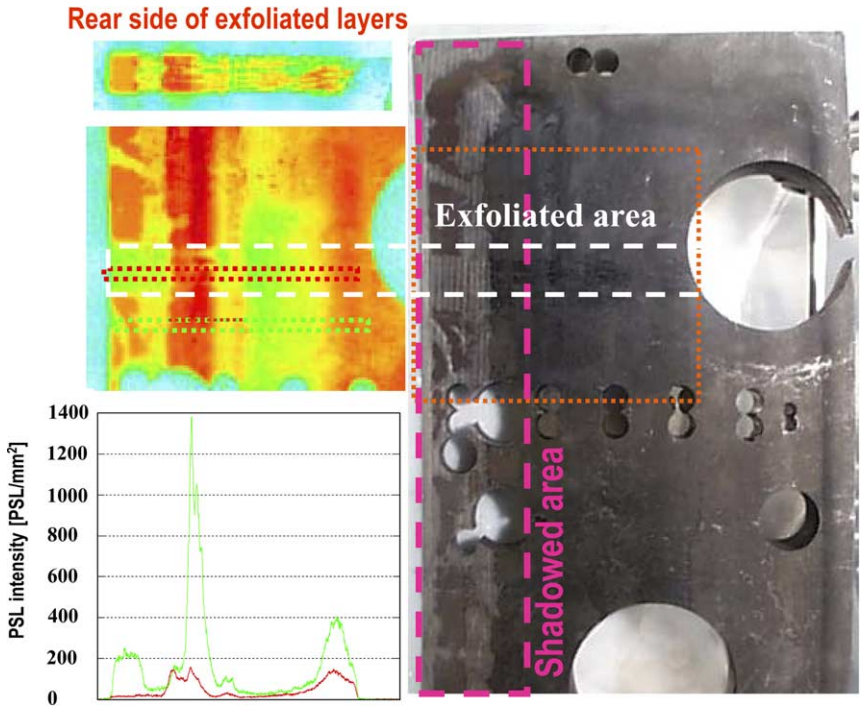

Fig. 3. Tritium retention in the deposited layers on the inner divertor base tile BN4. Deposited layers at marked area were removed by an adhesive tape and tritium retention was measured by IP both at remained substrate surface and the rear side of the removed layers. The rear side showed nearly same level of tritium activity as the front side of the deposited layers.

the combustion method, agreeing with the estimation from IP. It should also be mentioned that by inserting films with different thicknesses between a tile and IP, the tritium beta can be separated from other activity coming from metallic impurities such as ^{60}Co deposited on the tile, and the images of the metallic impurities (not shown here) were quite consistent with the tritium image [5]. Thus all these results confirm that most of tritium on the JET Mark IIA divertor tiles is retained in the deposited layers.

In addition to tile BN4, very high tritium retention was observed at the bottom side of vertical divertor tiles at the inner divertor side (IN3) as already noted by Penzhorn et al. [6]. Fig. 4 compares IP images of this particular tile for the plasma facing side, toroidal sides facing to wide gap and narrow gap sides, and the bottom edge facing to the horizontal target tile BN4 across the pumping slot. As already noted, the tritium level at the bottom side is quite high in areas corresponding to a brownish colored heavy deposit. Interestingly, this deposition was localized at the rear half of the edge, and a very clear boundary was observed between the rear half and the front half. When a similar tile was removed in 1996 (prior to the DTE1 campaign) there was a 40 μm thick film all over the edge of the tile [6]. It was thus a surprise when the tiles were examined after DTE1.

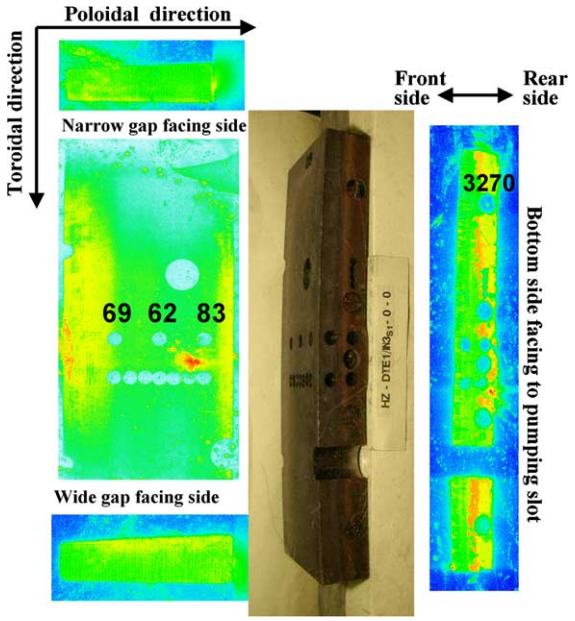

Fig. 4. Tritium distribution of the inner divertor tile IN3 for plasma facing side, toroidal sides facing to wide gap and narrow gap together with poloidal side facing to pumping slot. Numbers in the figure are tritium level of drilled cylinders determined by the combustion methods in MBq [10].

One reason for the thinner films here (particularly near the plasma-facing side) might be that prior to DTE1 all discharges had the strike points on the base tiles. However during DTE1 many high power pulses were run with the strike points on the vertical tiles. Thus the deposition may have been affected by discharges when the tile temperature became very high. There may also be some correlation with magnetic field line direction.

It is also important to note that tritium retention at toroidal sides (gap-facing sides) was quite small and no significant difference was observed between the two sides, i.e. the wide gap (12 mm) side and narrow gap (3 mm) side. As seen in the line profiles of Fig. 1, one can also note that the tritium level on the front face nearer wide gap edge region (corresponding to red line) is higher than that on the region nearer the narrow gap edge region (blue line) for all inner divertor tiles. This is because all the divertor tiles are angled slightly to the magnetic field direction. For an inner divertor tile such as IN3, ions striking the tile are traveling from right to left toroidally (in Fig. 4). Therefore the right-hand side of each tile is at smaller vessel radius (i.e. farther from the plasma) than the left-hand side so that the right-hand side of each tile is shadowed from incident ions by the left-hand side of the next tile (toroidally), and receives no flux of depositing particles.

In the tile IN3, tritium retention on the plasma facing side was rather small compared to the bottom side. Nevertheless the profile indicates that higher tritium retention at the middle area in the toroidal direction and the both top and bottom area in the poloidal direction. This is most probably because the divertor strike point was frequently on the central area of this tile. Accordingly the tritium in the central area was removed by the plasma heat load. A similar observation was also on the horizontal target tile BN4, where an area just several cm above the plasma shadowed zone has a toroidal zone of lower tritium concentration (see Fig. 1(a)). The effect of plasma heating is more clearly seen in JT-60U as discussed later.

3.3. Detailed tritium distribution on the outer divertor

Fig. 5 shows tritium distribution on the outer divertor tiles, consisting of divertor base tiles of BN5 and BN6, and vertical target tiles of 1ON8, 1ON9 and 1ON10, together with toroidal tile sides facing to the wide gap and narrow gap. Similar to the inner divertor, the retention at the divertor base target tile (BN7) has the highest intensity. One can note no clear boundary demarcating the shadowed area. Nevertheless the tritium image gives very clear parallel lines toroidally, particularly apparent on BN7 tile. This seems to indicate direct plasma erosion/deposition strongly correlated to magnetic field line. However, stripes appeared in BN6 tile which were also toroidally parallel. These stripes correspond well to the character of 2D-CFC tiles in which fibers and matrix are arranged by layer by layer perpendicular to the plasma facing surface of tiles. Tritium retention behavior is quite different for fiber and matrix as already noted in the tritium image of the rear side of BN4 [5], because the fibers and matrix are different in their graphitization and hence their hydrogen-absorption characteristics are

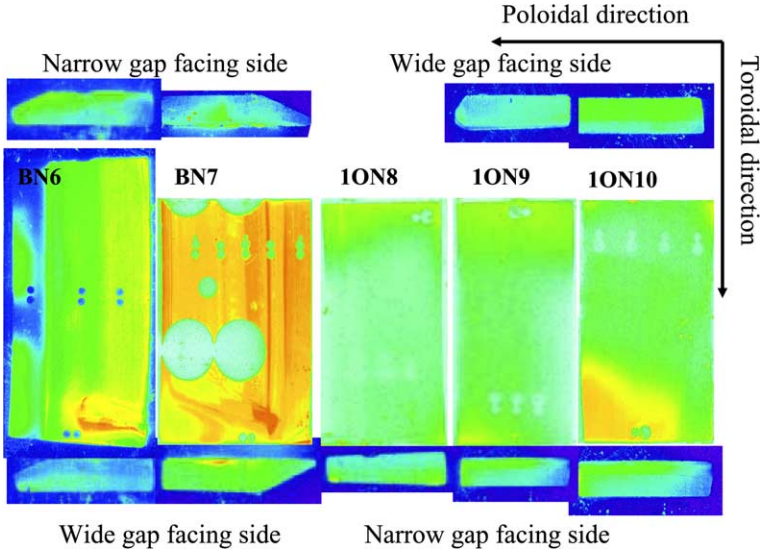

Fig. 5. Tritium distribution on outer divertor tiles, i.e., base tiles of BN6 and BN7, and vertical target tiles of 1ON8, 1ON9 and 1ON10, together with toroidal tile sides facing to wide gap and narrow gap.

different. Thus the appearance of the stripes on BN6 correspond to erosion domination on this tile at the outer side of the dome area. In this area, as discussed below, non-thermalized high energy deuterons originating from NBI as well as tritons impinge with rather high flux and their implantation into subsurface layers could contribute to make such stripe.

Tritium retention in the other outer divertor tiles is more than one order of magnitude less than that on the inner diverter tiles. In addition, different from the inner divertor tiles, the central region of the outer diverter tiles retains less tritium compared to near edge regions, particularly the toroidally left side of the tiles as shown in Fig. 5. As mentioned before, the tiles are angled to the magnetic field direction, and the toroidally left side (in Fig. 5) is the region shadowed by the adjacent tile. However, the outer divertor is now erosion dominated, so the amount of implanted tritium in the central region of the tiles is continually reduced by erosion, whereas tritium incident to the shadowed area is able to accumulate. Again, there is no clear difference between the wide gap facing side and narrow gap facing side, similar to the inner divertor case.

3.4. Comparison of the tritium profile of JET and profiles of D/H, and tritium profiles and deposition profiles for JT-60U divertor tiles

In Fig. 6, the results of JET and JT-60U [11] are compared. As already clarified [8], nearly half of tritium produced in JT-60U by D–D reaction was implanted into depth more than 1μm and hence its profile is somewhat different from D and H introduced as operating gases. Accordingly, the profile of T introduced by gas or NBI in JET (Fig. 6(a)) was quite different from that of the T profile in JT-60U (Fig. 6(b)) but show similarity to D and H profiles of JT-60U [12] (Fig. 6(c)). Still a large difference between T profile in the JET and D and H profile in JT-60U remains at the inner divertor. In most tokamaks, the inner divertor target is deposition dominated as is the case for JT-60 as shown in Fig. 6(d). The high T retention in JET in this region is simply due to T codeposition with carbon.

The inner diverter of JT-60U was covered by maximum of 60 μm carbon deposited layers [13], while the most of the outer diverter was eroded. Nevertheless, D and H retention in the inner divertor was less than that in the outer diverter. This is attributed to surface temperature rise by plasma heating and poor thermal contact of the layers to the substrate [11]. Small T retention in the outer divertor both in JT-60U and JET could be also attributed to temperature elevation. The evidence of the temperature increase of the deposited layer was clearly seen in depth profiles of H and D in the inner diverter of JT-60U by given by SIMS analysis [11,12]. In addition, recent NRD measurements by Hayashi et al.

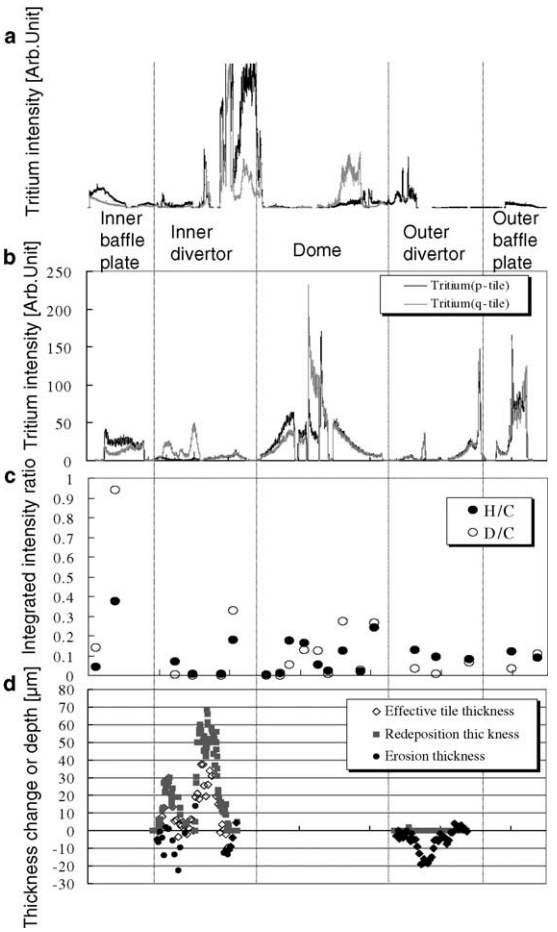

Fig. 6. Comparison of: (a) tritium profile of JET, (b) tritium profile for JT-60U measured by IP, (c) deuterium and hydrogen retention profiles within 3 μm of surface layers of JT-60U given by D/C and H/C measured by SIMS, and (d) deposition profile of JT-60U for divertor area. (After Hirohata et al. [11])

[14] have shown that (H + D)/C ratio in the deposited layers of JT-60U is less than 0.06. This value is much smaller than those reported in JET and ASDEX Upgrade confirming higher temperatures of divertor tiles of JT-60U [1,2].

It is very interesting to note that in the outer dome area (BN6 and BN7 tiles), all profiles of T in JET diverter, T and D/H in JT-60U show higher retention as seen Fig. 6. According to OFMC simulation code for high energy particle transport in tokamak plasma [15], incident flux of high energetic triton is high at the dome area and the outer baffle area. Even large number of D ions originated from neutral beam heating impinges to the dome area without fully losing their initial energy. This is perhaps the reason why D/C on the outer dome wing in JT-60U is rather large similar to the T retention.

4. Summary and conclusion

Profiles of T retention in the JET Mark IIA divertor and H/D and T retention in the JT-60U W-shaped divertor have been compared. It has been clearly shown that hydrogen (H, D and T) is retained in carbon deposited layers with nearly constant concentration throughout the layers, except high energy triton directly impinging into more than 1μm depth. However, tritium profiling by IP and H and D depth profiling shows that tritium retention in the deposited layers is very much influenced by various factors. The tritium profile and hence carbon deposition in JET divertor is not uniform both poloidally and toroidally. It clearly reflects geometrical structure of divertor and tile alignment as well as magnetic field lines. Carbon deposition seems to result from plasma erosion and deposition, including hydrogen codeposition, which is most clearly observed on the inner diverter base tile and the bottom edge of the vertical target tile. In this respect, the wide open pumping duct in the inner diverter of JET seems to enhance carbon deposition, resulting the heavy deposition at the area shadowed from the plasma, while the W-shaped divertor with bottom side pumping in JT-60U seems to prevent the plasma deposition of carbon.

Hydrogen once codeposited with carbon may be re-eroded. In addition, because of the poor thermal contact of the thick codeposited layers to the substrate, the temperature of the deposited layers could be easily elevated thermally to release the once retained hydrogen as observed in the codeposited layers on the inner diverter tiles of JT-60U. In the present tokamak, a significant part of high energy deuterium injected by NBI is escaping from the plasma and impinging onto the dome region and divertor baffle regions. The contribution of these high energy particles could be large at the erosion dominated outer divertor region.

Thus we may reduce both carbon deposition and hydrogen retention by optimizing the design of the divertor structure and tile alignment. In addition, if ITER allows the increase of the surface temperature of PFM above 800–1000 K by plasma heat load, tritium retention could be significantly reduced from the current estimation.

Acknowledgements

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