Tritium Retention in Plasma Facing Graphite Tiles

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Abstract

The present work, at first, summarizes the basic knowledge of hydrogen behavior in graphite and then introduces recent investigations how tritium is distributed in plasma facing walls of TEXTOR and JT-60U in terms of material temperature (or incident flux), local plasma behavior, global plasma behavior and so on.

In TEXTOR, it was found that tritium distributions was quite different from deuterium distribution, i.e. deuterium retention was higher at the deposited area, whereas tritium retention was higher at the erosion dominated area. This is because tritium produced by the D-D reaction, initially having 1 MeV, did not fully lose its energy in the TEXTOR plasma and implanted into the plasma facing materials nearly homogeneously, whereas deuterium was codeposited with carbon and boron, the main impurities in the TEXTOR plasma. This is also confirmed by the finding that high level of tritium was detected beneath the deposited layer.

In JT-60U, tritium distribution, however, was modified by the temperature increase due to plasma heat load. The highest tritium level was found at the top of the dome or the private region and the outer baffle plates, where the plasma did not hit but the distance from the plasma was the shortest. For the divertor tiles, the tritium retention was very small. Such tritium distribution observed in JT-60U tiles can be well explained by the homogeneous implantation of rather high energy tritium and thermal release due to the heat load.

Thus the comparison of tritium profiles with the deuterium profile gives a large amount of important and new information on PMI, and may be used as a new diagnostic technique for PMI.

I. Introduction

Tritium retention in plasma facing materials both for long term and short term is one of the most important safety issues in fusion reactors [1]. After the use of tritium in JET and TFTR, the immediate retention was about 40% and 51%, respectively [2, 3]. Even after the excessive tritium removal procedure applied to both machines (D-operation, D_2 and He-glow, O_2-venting), remaining long-term tritium retention of ~16% and ~13% was observed, respectively. A serious concern is the very high tritium retention in deposited layers at lower temperature region, particularly in the deposited layers in divertors and louvers in JET [4]

With the use of a scintillation method (which needs sophisticated procedure, full combustion or oxidation, accumulation of tritiated water vapor and scintillation measurements), a large set of wall tiles have been analyzed and tritium surface distributions and depth distributions in the plasma facing tiles have been reported [4,5]. Unfortunately, this technique can not give detailed surface profiles and depth profiles of tritium, which is very important to the understanding of tritium retention and transport in tokamaks.

Very recently we have succeeded in making tritium images on graphite tiles used in TEXTOR[5,6] and JT-60U [7] by means of an imaging plate technique. In both machines, tritium is produced by the D-D reaction. Hence energetic triton without fully losing energy was implanted to all of plasma facing walls, but the tritium profiles in the wall was clearly modified by the plasma conditions and characters, and the wall temperatures as well.

The present work, at first, summarizes the basic knowledge of hydrogen behavior in graphite and then introduces recent investigations how tritium is distributed in plasma facing walls in terms of material temperature (or incident flux), local plasma behavior, global plasma behavior and so on. Comparison with

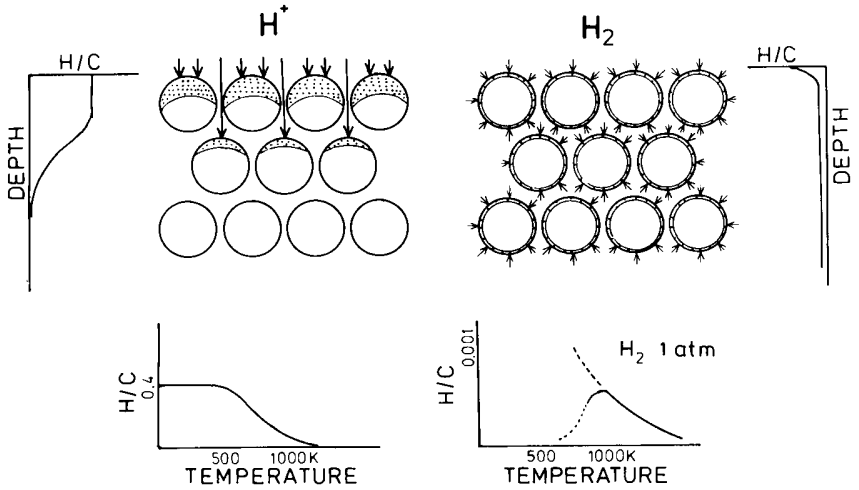


Fig. 1. Simplified model given by Tanabe [4] for the surface retention by energetic hydrogen injection from plasma and the bulk retention due to residual hydrogen gas in a torus.

deuterium distribution determined by ion beam analysis is also mentioned to give very important information on the behavior of tritium.

II. Carbon-Hydrogen Chemistry

Fig.1 shows schematic of hydrogen retention on plasma facing tiles[8]. Because of porous nature of graphite, energetic hydrogen and molecular hydrogen behave quite differently, i.e. energetic hydrogen impinges surface and stops at around projected range, whereas molecular hydrogen can penetrate deep inside along open pores. As a result gaseous hydrogen can permeate through graphite. However, very small diffusivity keeps hydrogen only near the surface region of the grains. Therefore depth profiles in large scale can be written smooth change as show in Fig.1, even the detailed- or or micro-scale profiles in each grains are significantly different.

In addition hydrogen retention in graphite is quite temperature dependent. Below 500K implanted hydrogen cannot move to be trapped and the surface region is easily saturated with hydrogen concentration of about 0.4 for H/C, while above 800K hydrogen diffusion becomes dominant and most of implanted hydrogen is re-emitted remaining very small amount of hydrogen.

It is also noted that once hydrogen molecules are produced on the grain surface through a mechanism that migrating atomic hydrogen recombines with trapped hydrogen, the hydrogen molecules can penetrate to the top surface to be released out. Between 500K – 800K hydrogen retention quite depends on temperature. Accordingly C-H film shows very wide range of H/C ratio as shown in Fig.2, where carbon is characterized in to two different

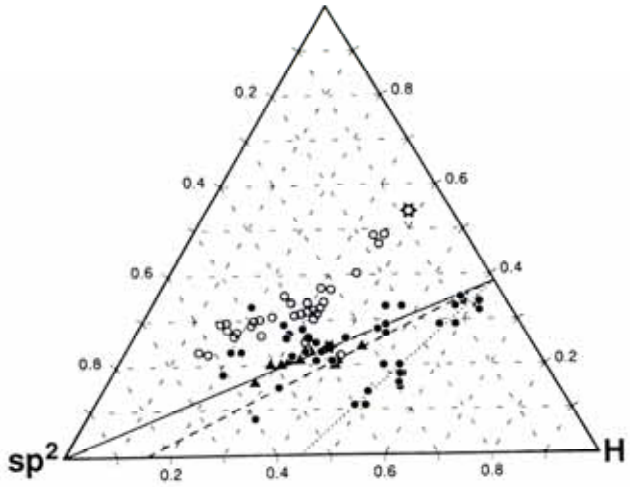


FIG. 2. Experimental compositions of hydrocarbon films. Same data as in Fig. 1. All IR data (open circles) were corrected by increasing the reported sp^2 fraction by a factor of 3 according to Eq. (1). Full circles are NMR data and full triangles are EELS data. The data of Grill *et al.* (Refs. 15 and 16) are shown as full and open stars.

species, i. e. sp^2 and sp^3.[9] Still H/C ration in production of C/H films is not free, but limited in certain range depending on the producing methods and temperature.

Below 500K, graphite irradiated with hydrogen up to saturation level is reported to show the same character as that of hydrogen saturated amorphous carbon (a:C-H) film.[10] With increasing the substrate temperature, H/C in the deposited film decrease, and becomes pyrolytic carbon, well but not fully graphitized carbon.

It should be noted that in JET and JT-60U the first wall is operate below 600K while TFTR around 400K or below. This makes tritium profile of in-vessel component of these machines a little different as discussed later.

Ⅲ. Tritium detection by imaging plate technique

The imaging plate (IP) used here was BAS-TR2025 for low energy β-rays emitter such as tritium, manufactured by Fuji Photo Film Co. Ltd. The surface of the IP was contacted to the graphite tile surface for one hour in a dark shielded room. For full toroidal measurements for the tiles of the top of the dome units, we have inserted a thin (2 μm) poly-film between the tile and the IP, which reduces the sensitivity by a factor of 10. After the exposure, the IP was processed by an imaging plate reader, Fuji BAS-2000 or BAS-2500 to obtain digitised intensity mapping ("tritium image") The characteristics of imaging technique and detailed measuring methods are given elsewhere. [7,11]

Ⅲ-1. Tritium retention in plasma facing graphite tiles of TEXTOR

Ⅲ-1-1 Tiles measured

Graphite tiles investigated here were taken from two different types of limiters used in TEXTOR-94: (ⅰ) two graphite tiles (tile No. 8 and 22 at blade 4) at the middle position of the forth blade of the toroidal ALT-II limiter and (ⅱ) two (the top and the bottom) tiles from the bumper limiters. Two collector probes made of Si at the top and the bottom of the torous, which were not directly exposed to the plasma but faced to the plasma directly, are also examined. The locations and the configurations of the limiter tiles are shown in Fig.3. All tiles (from both the ALT-II limiters and bumper limiters) were used in the operations from the middle of 1998 to the middle of 1999 and exposed to more than ten thousands shots. Tritium detected was within the depth of 3.5 μm from the top surface of the graphite tiles and a little less in the Si collector probes. The details of the tritium imaging plate technique (TIPT) were given elsewhere[6,7] and also in this conference.[11] Deuterium distribution was measured by means of an ion beam technique, as given in Ref.[12].

Ⅲ-1-2. ALT-II limiter tiles

Fig. 4(a) shows the tritium image for one of the ALT-II limiter tiles (No. 22). Vertical and horizontal directions correspond to the poloidal and toroidal direction in the torus respectively. Two holes in the central area are for the bolts fixing the tile to the base plate. The lower, larger part of the tile is the erosion-dominated region and the upper part with a rough surface is the deposition-dominated region covered with redeposited layers with a thickness of

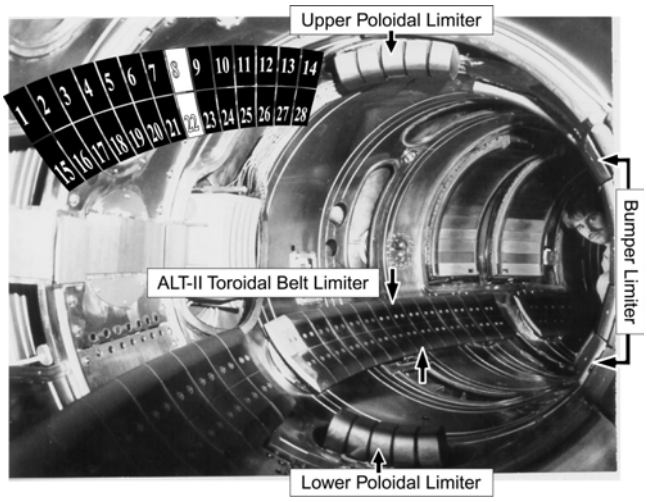


Fig. 3. Photograph of TEXTOR-94 vacuum vessel with ALT-II toroidal belt limiters, bumper limiters. Two tiles of No.8 and 22 at blade 4 of ALT-II limiters, two (the top and the bottom) of the bumper limiters are shown in the figure.

around 5 μm [11]. As seen in the line profiles, tritium distribution at the erosion-dominated region is nearly homogeneous and much higher than that at the deposition-dominated region. However, by peeling off some area of the deposited layer (in the upper left region of the tritium image), the tritium level beneath the deposited layer was again homogeneously distributed with tritium levels a little higher than that of the erosion-dominated region as seen in the line profiles. It is interesting to compare the tritium profiles with deuterium and boron profiles given by the ion beam analysis of one of the ALT-II tiles. (See Fig.4(b)) The tile analyzed by the ion beam was not the same one as used for tritium imaging, but there were no significant differences in these profiles among the different ALT-II tiles. As clearly seen in Fig.4(b), deuterium and boron show quite similar profiles and their densities are higher at the redeposition dominated region (poloidal position > ～ 80 mm). In another word, deuterium was retained in a very shallow region by re- or co-deposition with boron and carbon, whereas tritium was retained deeper.

Tritium was also detected even at the backside of the tiles where was not directly attached to the base plate (not shown here). Tritium distribution was still homogeneous, but the amount is an order of magnitude less than that for the plasma facing side. The results for another tile (No. 8) were quite similar except that a little higher tritium concentration was observed (not shown here).

Ⅲ-1-3 . Bumper Limiters

Fig. 5 shows the tritium images of the top and the

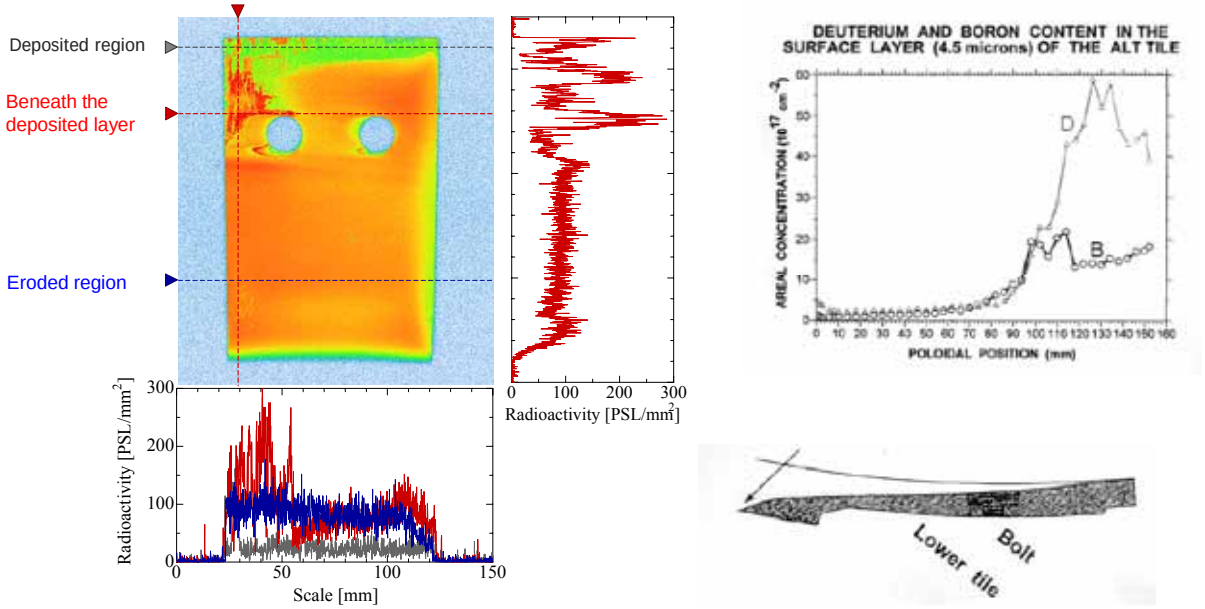
bottom tiles from 5 bumper limiter tiles together with the toroidal line profiles of tritium and deuterium for the bottom tile (Fig.5(d)). Because of the saddle type shape of the tile, the image was divided into the electron drift side and the ion drift side. Although the tritium level of the electron drift side is a little higher than that of the ion drift side, the tritium distributions within either side were both nearly homogeneous. One can clearly see that the tritium level is significantly higher for the bottom limiter than for the top. Tritium was also detected even at the backside of the bumper limiters.

As clearly seen in the Fig.5(b), the profiles for tritium and deuterium were completely different. The appearance of the bumper limiters did not show a clear deposition pattern, but there might be some deposition, as indicated by the observed high retention of deuterium. The temperature of the bumper limiter, which was not measured, also likely influenced the profile as will be discussed in the next section.

III-1-4. DISCUSSION

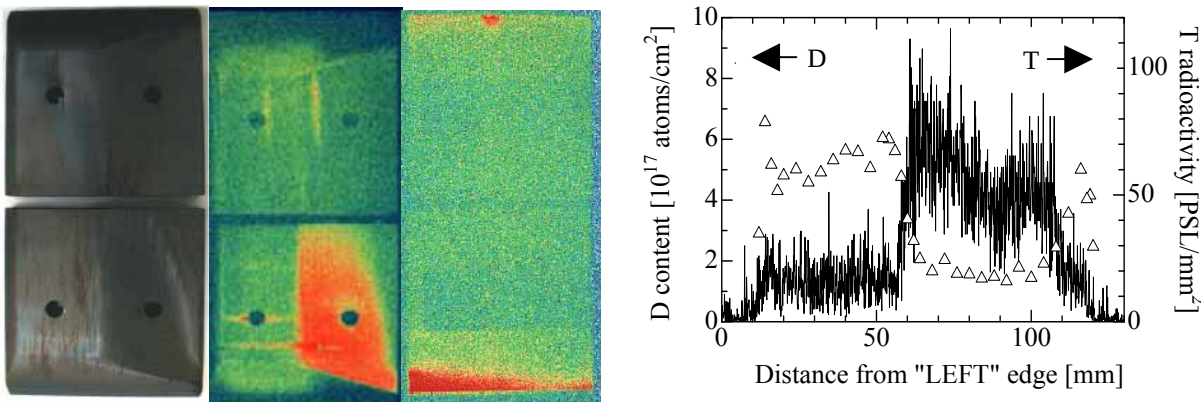
A. Homogeneous distributions on the ALT-II tiles

In the previous work [6], we have suggested that tritium produced by D-D reaction, initially having the energy of about 1 MeV, did not fully lose its energy in



(a) (b)

Fig.4. (a) Tritium image for a graphite tile (No.22 at blade 4 of the ALT-II limiter), together with line profiles along the poloidal (vertical) and toroidal (holizontal) directions. (b) Deuterium and boron line profiles along the poloidal direction together with the cross section of the tile.



(a) (b) (c) (d)

Fig.5 (a) Photograph of the top and bottom of the bumper limiter tiles, (b) tritium images from the front surface and (c) from the back side of the respective tiles. (d) tritium (−) and deuterium (Δ) line profiles along the toroidal direction at the front surface .

the TEXTOR plasma (with a temperature of about 2keV, density 3 x10^{20} /m^3, and the magnetic field of 2.2 T), and implanted into the plasma facing materials nearly homogeneously. This was confirmed by the observation that the high level of tritium retention was observed behind the deposited layer. That is because the triton (tritium ions) gyrates in the magnetic field with a maximum gyro-radius of about 11cm, and can penetrate into the gap or turn around to the back side of the limiters where it was not directly facing to plasma but had some space between the tiles and the back plate of the liner.

It was rather surprising to find that the tritium distribution was quite different from the deuterium distribution [11] (See Fig.4(b) and Fig.5(d)). However deuterium analysis clearly showed that deuterium was codeposited with carbon and boron, the main impurities of the TEXTOR plasma. This is another confirmation that tritium produced by the D-D discharge behaves differently from those tritium introduced as a gas fuel or by NBI.

B. Asymmetry along poloidal direction and the difference between ALT-II tiles and bumper limiters

As discussed above, tritium distribution can be generally described by the homogenous energetic implantation from the plasma but not codeposition nor redeposition. Nevertheless, the poloidal tritium distribution obviously shows deviations from the general tendency. The tritium intensity of the bottom bumper limiter was clearly higher than that of the top one. Such asymmetry was more clearly shown in the comparison of the collector probes. This asymmetry should be attributed to the higher density of the lower half of the plasma than the top half, though there might be some contribution from the gyration and the direction of the magnetic field. Due to the high penetration power of neutrons, it might be difficult to use neutron profiling to see such asymmetry, Thus, tritium profiling for the full poloidal direction could be a good diagnostic technique for plasma profiling in D-D discharges.

C. Target temperature effect

One can note that deuterium retention on the redeposited region of the bumper limiters is one order of magnitude less than that of the ALT-II limiters (compare Figs. 4(b) and 5(d)). Although there might have been some difference in the incident deuterium fluxes between the two, such a large difference in deuterium retention could not have been due to the flux difference entirely. In fact, the tritium retention, which should be nearly homogeneous, was also higher for the ALT-II limiters than for the bumper limiters. This indicates that the bumper limiter could be subjected to a little higher temperature than the ALT-II limiters. During NBI discharges the heat load of the ion drift side is likely higher than the electron drift side and the saddle type shape of the bumper limiter can separate the heat load. Thus tritium retention on the bumper limiters was clearly different for the electron drift side and the ion drift side.

Such a profile change due to the temperature rise was also seen in tritium retention at the leading edge of the ALT-II limiters. One can see that the tritium profiles decay more slowly at the leading edge compared to the steep decay at the side edges. (see Fig.5) The reason for the tritium retention at the No.8 graphite plate being a little higher than that at the No.22 graphite plate is probably owing to a higher heat flux to the No.22 plate than the No. 18 graphite plate [13], resulting in a little higher temperature and less tritium retention for the No.22 plate.

On the bumper limiters, it is difficult to distinguish the erosion dominated regions form the redeposition dominated regions. Analogous to the ALT-II limiter, those regions with higher deuterium retention were probably dominated by redeposition. At high temperatures, deposition results in slightly graphitized carbon with good adhesion to the substrate graphite. According to the ion beam analysis, the D/C ratio at the low deuterium retention regions in the

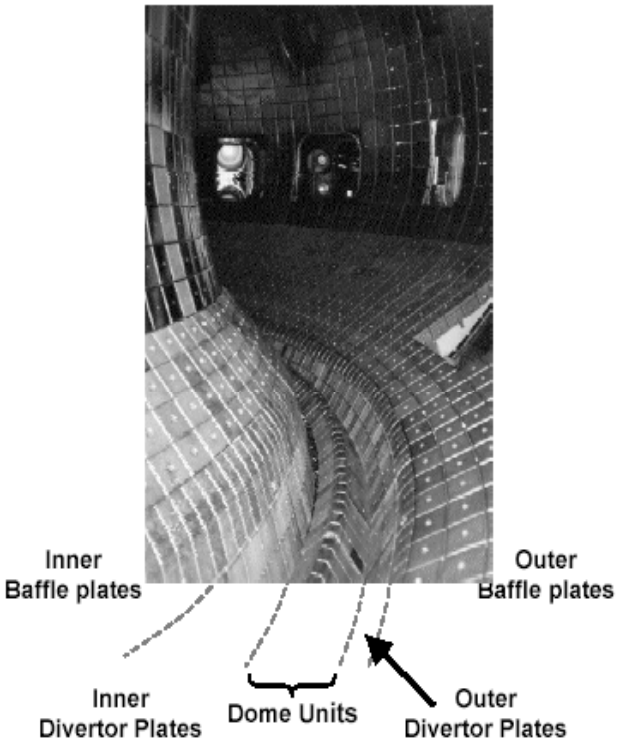


Fig.6 Configuration of graphite tiles in JT-60U divertor region

bumper limiters was ～0.05 [14] (see Fig.5(d)), indicating that the temperature should be above 900K.

In toroidal direction, particularly near the right edge, the tritium retention was slightly less, which could also be due to the higher heat load. However, the tritium retention was still very small in the tiles, i.e. far below the saturation, which means that without significant temperature increase it would be difficult to discuss the temperature effect. In a separate work of tritium analysis of the divertor tiles of JT-60U, such temperature effect was more clearly observed [7,15].

III-2 Tritium retention in plasma facing graphite tiles of JT-60U

III-2-1. Tiles measured

Graphite tiles examined here were isotropic graphite (IG-430U) and CFC graphite (CX-2002U) used at (1) divertor region taken from the poloidal direction including the baffle plates and (2) the top of the dome units taken from the toroidal direction. The configuration of the tiles in JT-60U is given in Fig.6. These tiles were fixed to the base metal by bolts and their temperature was measured by thermocouples located at the backside. The temperature increase due to the plasma heat load was around 50 K except at the divertor striking points where the maximum temperatures of 800 and 1200 K were recorded at the inner and outer divertors, respectively. However, surface temperature during plasma discharge was not measured for all tiles.

The tiles were exposed to deuterium plasma in the period of June 1997 to October 1998. During this period, about $1x10^{19}$ neutrons were generated by D-D reactions. Accordingly $1x10^{19}$ or 18 MBq of tritium was also produced. The vacuum vessel of JT-60U was subjected to hydrogen discharges to remove the tritium retained in the vacuum vessel, then followed by air ventilation before fully opened to the atmosphere. Thus long term tritium retention in the vacuum vessel was about 50% of the total production. [15]

III-2-2. Results and Discussion

Fig. 7 compares the photographs and tritium images of the divertor units and the baffle plates. In this figure, tritium level is higher in the red region and less in the blue region. The tritium levels at the top of the dome unit and baffle plate were estimated to be in the order of 10kBq/cm^2.[7]
Fig.8 shows corresponding tritium line profiles. It was rather surprising to see that the highest tritium level was recorded at the outer baffle plates and the top of the divertor dome, where plasma did not directly hit. The tritium level at the divertor tiles were very small, particularly both the outer and inner strike points showed the lowest level owing to the temperature escalation during plasma discharges. This will be discussed in the next section. The tritium

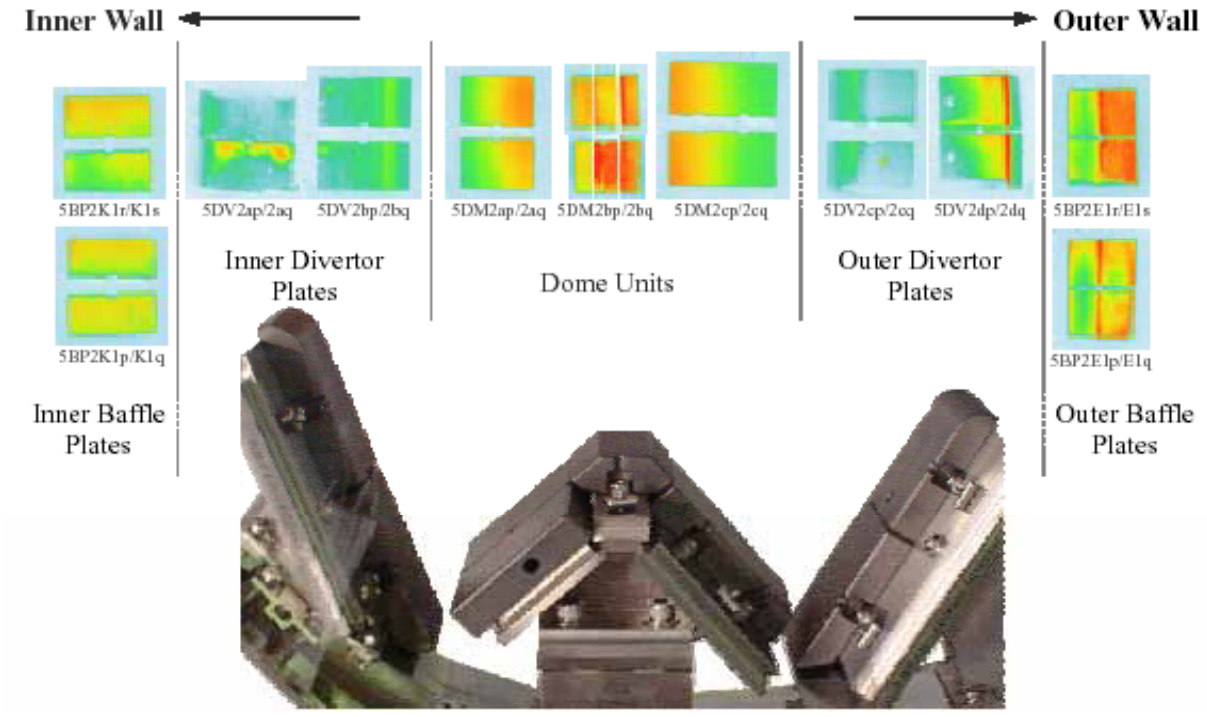


Fig.7 Photographs and tritium images of divertor units and baffle plates

level at the top of the dome units is homogeneous, while the tiles of the both sides show steep gradient in the poloidal direction.

Suppose the total produced tritium (18GBq) in JT-60U was distributed homogeneously over the plasma facing area, the areal tritium level could be estimated from the distance of the tritium source plasma. However, not all the tiles face the plasma directly and the distances from the plasma also depend on the geometrical configuration of the tiles in the torus. Thus we can estimate tritium flux assuming that the tritium is homogeneously produced in the plasma and injected to all tiles, taken into account the distance and the angle of the incidence. The results are given by solid lines in Fig. 8. Approximating the D shape of the plasma by a circle over full torus with a minor radius of 1.4 m and the major radius of 3.3 m, the total surface area is calculated to be 180 m^2. When totally generated tritium of 18 GBq is divided by 180m^2, areal tritium distribution becomes around 10 kBq/cm^2. This is the same order of the magnitude as the tritium level observed, confirming that large part of the tritium produced by D-D reaction was injected homogeneously over the plasma facing surfaces without fully losing its initial energy of ～ 1MeV. The small differences among the tritium retention in all of the tiles of the top of the dome unites over 2π toroidal direction (in Fig.9) except some local structure also confirms the homogeneous implantation.

In Fig.8, therefore, the estimated tritium fluxes were normalized at the highest tritium level. Although the estimated tritium flux correspond to the general trend of the tritium profiles along the poloidal direction, the deviation particularly at the divetor tiles is appreciable. One may attribute the deviation to the temperature increase of the tiles due to the plasma heat load. Unfortunately surface temperature of the tiles was not measured.

Then we have estimated the surface temperature from the retained tritium level, to confirm the above argument using following assumptions: (ⅰ) the tritium retention together with deuterium and hydrogen at the highest point corresponded to the saturated hydrogen level in graphite [16] at 600K (the measured temperature by the thermocouple at the back side of the tiles), and (ⅱ) tritium once implanted was released by the temperature rise of the tiles due to the plasma heat load. We can estimate the surface temperature based on the temperature dependence of the saturated hydrogen retention [7] and shown in Fig.5, for the tile at the outer side of the dome units. Once we had obtained the surface temperature (calculated and shown in Fig. 9) , we could also calculate the heat flux based on the steady state heat flow, with the backside temperature given by the thermocouple, and the thermal conductivity of graphite ～80 $Km^{-1}s^{-1}$. The calculated heat fluxes which are around 1 to 5 MWm^{-2} depending on the position, are also plotted in Fig. 5. The calculated

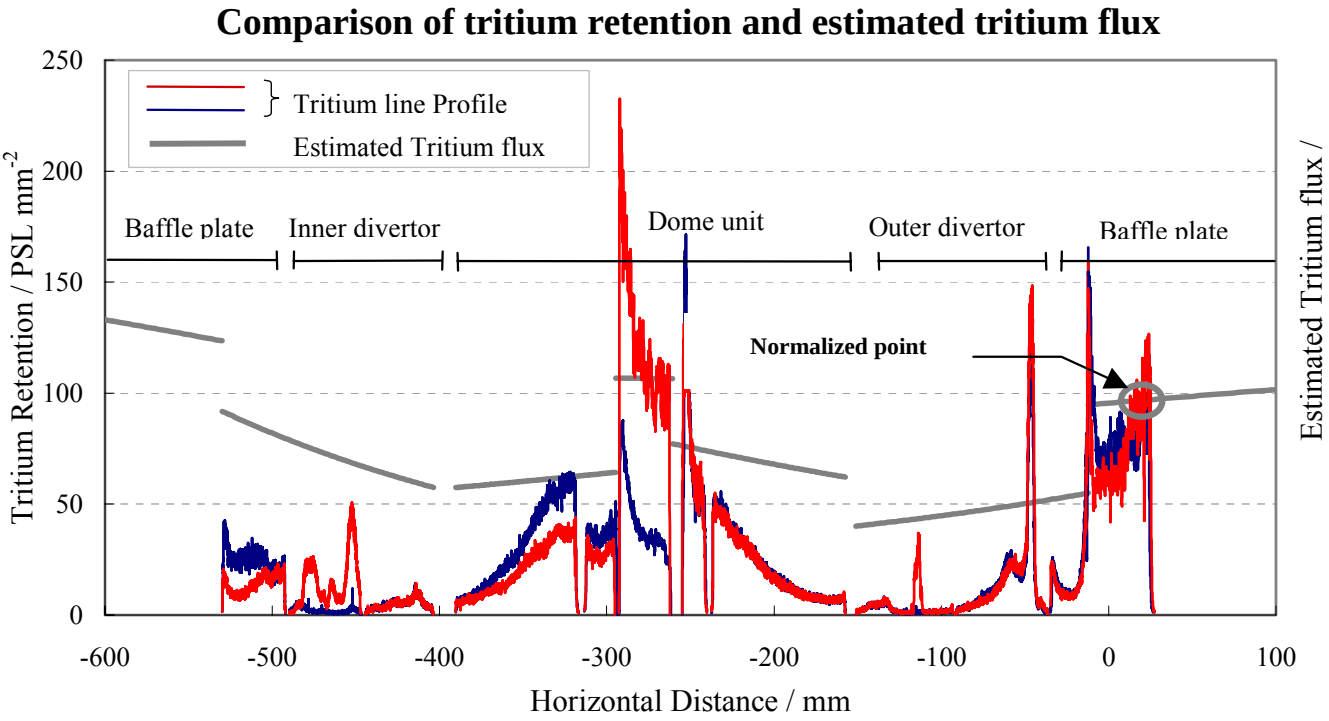


Fig.8 Tritium line profiles along the poloidal direction in the divertor region. Estimated tritium flux is also shown (see text). Blue and red lines are corresponding to two different line profiles on the same tiles.

values are quite reasonable compared to the separate estimation of the heat load [14]. The slightly higher tritium level at the inner side edges in all tiles of the top of the dome units (see Figs. 8 and 9) might be also explained by the larger temperature increase observed at the outer divertor than that at the inner divertor [7].

Thus tritium distribution observed in JT-60U tiles can be well explained by the homogeneous implantation of rather high energy tritium and simultaneous thermal release due to the heat load. Accordingly, the tritium distribution at those regions which are near the plasma but not directly in contact with the plasma (such as the top of the divertor and baffle plates) retained high level of tritium instead of the divertor tiles where the tritium was very small. This is also confirmed by the nearly same tritium images for all tiles of the top of the dome measured here except some local structure in torus direction.

As shown above, in TEXTOR, tritium retention in the redeposited layer was relatively small. On the other hand, most of the tritium images of the JT-60U tiles showed no clear patterns but gradual changes, whereas some patterns probably owing to the erosion and deposition were appreciable in the photograph of the tiles. In the divertor tiles, because of the their temperature increase, properties of the deposited layer may be somewhat different from the deposits at lower temperature, or the thickness of the deposits might be much less. These must be investigated in future.

In this tritium imaging plate technique, it is very likely that IP imaged missed some tritium adsorbed or absorbed in the near surface layer from low energy impinging after losing energy in the plasma. This is because such surface tritium was easily replaced by deuterium during the discharges and was mostly removed by hydrogen discharges just before the torus opening and the subsequent air ventilation. This might be additional reason why the tritium distribution was homogeneous without showing any appreciable inhomogeneity in the edge plasma. Thus most of the tritium detected by IP was those retained rather stable in the graphite from the top surface to a depth of around 5 μm.[10]

IV. Conclusion

In order to investigate tritium behavior in tokamak, we have measured the surface distributions of deuterium and tritium on graphite limiter tiles used in TEXTOR under D-D operation, respectively, by means of an ion beam analysis and tritium imaging plate technique.

Tritium produced by D-D reaction did not fully lose their energy before implanted in the plasma facing surfaces. As a results tritium distribution on the tiles were generally homogeneous. High energy

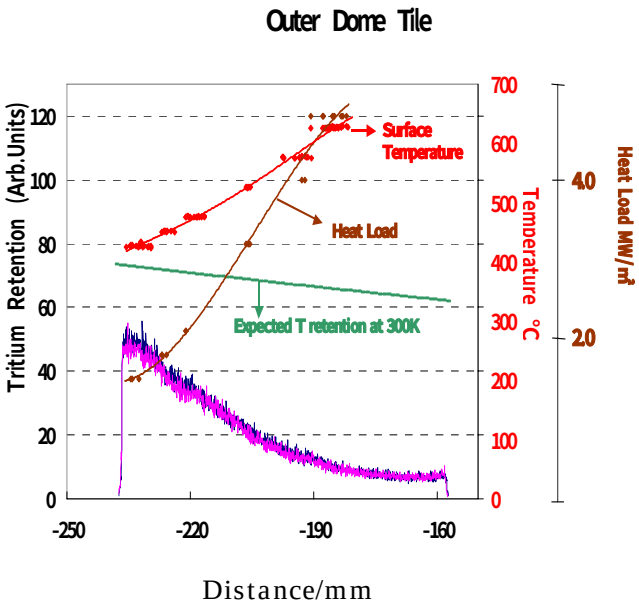


Fig.9 Comparison of tritium profiles, expected tritium retention, estimated surface temperature and heat flux from the tritium content (see text) for one of the outer tiles of the dome units.

injection was confirmed by the observation of tritium beneath the deposited layer. Tritium was detected even in the plasma shadow, probably owing to their gyration in the scrape-off layer. The profiles of deuterium were completely different from the tritium profile. Deuterium are co- or re-deposited with carbon and boron on plasma facing surfaces at the deposition dominated regions, whereas little deuterium was observed at the erosion dominated regions.

Tritium distributions on the graphite tiles used as plasma facing tiles in divertor tiles, dome units, and the baffle plates of JT-60U were also successfully measured. The highest tritium level was found at the top of the dome or the private region and the outer baffle plates, where the plasma did not hit directly but the distance from the plasma was the shortest. Such tritium distribution observed in JT-60U tiles can be again explained well by the homogeneous implantation of rather high energy tritium and thermal release due to the heat load. Therefore tritium distribution at those regions near the plasma but not directly in contact with it (such as the top of the dome unites and the baffle plates) retained higher amount of tritium. The tritium retention in the divertor tiles was very small, particularly the tritium level at the strike points of the divertos were near the detection limit. Different from TEXTOT, no clear pattern corresponding to the deposition was observed in JT-60U divertor tiles.

The observed tritium distribution reflects the temperature history of the PFM, and the retention seems very small at those regions that have experienced temperatures over 800K. The tritium distribution also indicates plasma asymmetry, i.e.

higher density in the bottom half than the top half of the plasma in TEXTOR and some local structure in the full toroidal tritium profiles of JT-60U, which must reflect plasma character.

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