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Imaging plate technique for determination of tritium distribution on graphite tiles of JT-60U

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Abstract

The tritium imaging plate technique was applied to determine surface tritium distributions on graphite tiles used as the first wall and W-shaped divertor in JT-60U, in which tritium produced by the D–D nuclear reaction in the plasma was implanted and/or deposited depending on the incident energy. Measured samples were isotropic graphite (IG-430U) and CFC graphite (CX-2002U), used as divertor tiles and/or baffle plates just outside the divertor. Tritium areal distributions on graphite divertor tiles, dome units and baffle plates of JT-60U were successfully measured for the first time. Tritium distributions observed in JT-60U tiles can be explained by homogeneous implantation of high energy tritium which is influenced by redeposited layers and redistributed by the temperature increase due to the plasma heat load. The tritium retention in graphite heated above 800 K was significantly small.

1. 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 $\simeq$16% and $\simeq$13%, respectively was observed. With the use of a scintillation method (which needs sophisticated procedures, full combustion or oxidation, accumulation of tritiated water vapor and scintillation measurements), a large set of wall tiles has been analyzed and tritium surface distributions and depth distributions in the plasma facing tiles have been reported [4]. Unfortunately, this technique cannot give detailed surface profiles and depth profiles of tritium, which is very important for the understanding of tritium retention and transport in tokamaks.

Quite recently, we have successfully applied the tritium imaging plate technique (TIPT) to determine the tritium areal distribution on in-vessel components used as limiter tiles of TEXTOR-94 [5,6] and found that TIPT not only gave very detailed tritium surface profiles but can also be used as a new diagnostic technique to investigate plasma wall interactions through the tritium behavior.

2. Experimental

The imaging plate (IP) is a radiation image sensor based on photo-stimulated luminescence (PSL). It detects tritium distributed within a depth of $\simeq$3.5 μm from the surface of graphite tiles (i.e. within the range of tritium β-rays in graphite). The IP used here was BAS-TR2025 for low energy β-rays emitter such as tritium, manufactured by Fuji Photo Film Co., Ltd. The surface

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of the IP was exposed to the graphite tiles with a face-to-face contact for 100 min in a shielded dark room. After the exposure, the IP was processed by an IP reader, Fuji BAS-2000 or BAS-2500 to obtain digitized intensity mapping or 'tritium image'. The pixel size of the radioluminograph was set to be $100 \times 100\ \mu m^2$.

Measured graphite tiles were isotropic graphite (IG-430U) and CFC graphite (CX-2002U), used as divertor tiles and/or baffle plates just outside the divertor of JT-60U. Fig. 1 shows the configurations of the graphite tiles of the divertor plates, the dome units and the baffle plates at the bottom of JT-60U. The 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 strike points with maximum temperatures of 1200 and 800 K were recorded at the outer and inner divertors, respectively.

The tiles were exposed to the deuterium plasma in the period of June 1997–October 1998. During this period, about 1×10^{19} neutrons were produced by D–D reactions. Accordingly, 1×10^{19}or 18 GBq of tritium was also produced. Before the opening of the vacuum vessel of JT-60U, hydrogen discharges were used to remove the tritium retained in the vacuum vessel and followed by air ventilation before fully opening to the atmosphere. Thus, long-term tritium retention in the JT-60U vacuum vessel was about 50% of the total production.

3. Results and discussion

Fig. 2(a) presents the comparison of the tritium images given by IP for the divertor tiles and the baffle plates. In this figure, the tritium level is higher in the red region and less in the blue region. The tritium profiles for the tiles along the poloidal direction are given in Fig. 2(b). It should be mentioned that the highest tritium level was recorded at the outer baffle plates and the top of the divertor dome, where, the plasma did not directly hit. The tritium level at the divertor strike point was the lowest owing to the higher temperature rise during plasma discharges.

In the previous work, we have shown that the tritium detected in the limiter tiles in TEXTOR did not fully lose its energy before impinging on the wall and homogeneously distributed over the plasma facing surfaces [5,6]. This is also true for the tritium detected here. Most of it was implanted rather deep compared to the deuterium supplied by gas-puffing or NBI. The tritium level at the top of the dome unit and baffle plate in JT-60U was estimated to be in the order of 10 kBq/cm^2 according to (i) the previous TEXTOR measurements where the absolute tritium level was about 100 Bq/cm^2 and (ii) the necessary IP exposure time for the JT-60U tiles to give the same PSL intensity was about 1/100 of that needed for the TEXTOR tiles. Suppose the total produced tritium (18 GBq) being deposited homogeneously over the plasma facing area of around 180 m^2 in

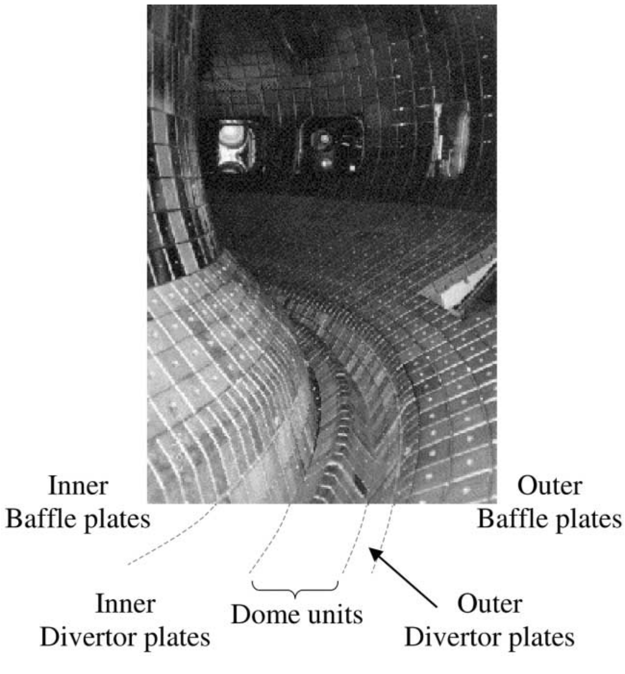

Fig. 1. Configuration of graphite tiles in the JT-60U divertor region.

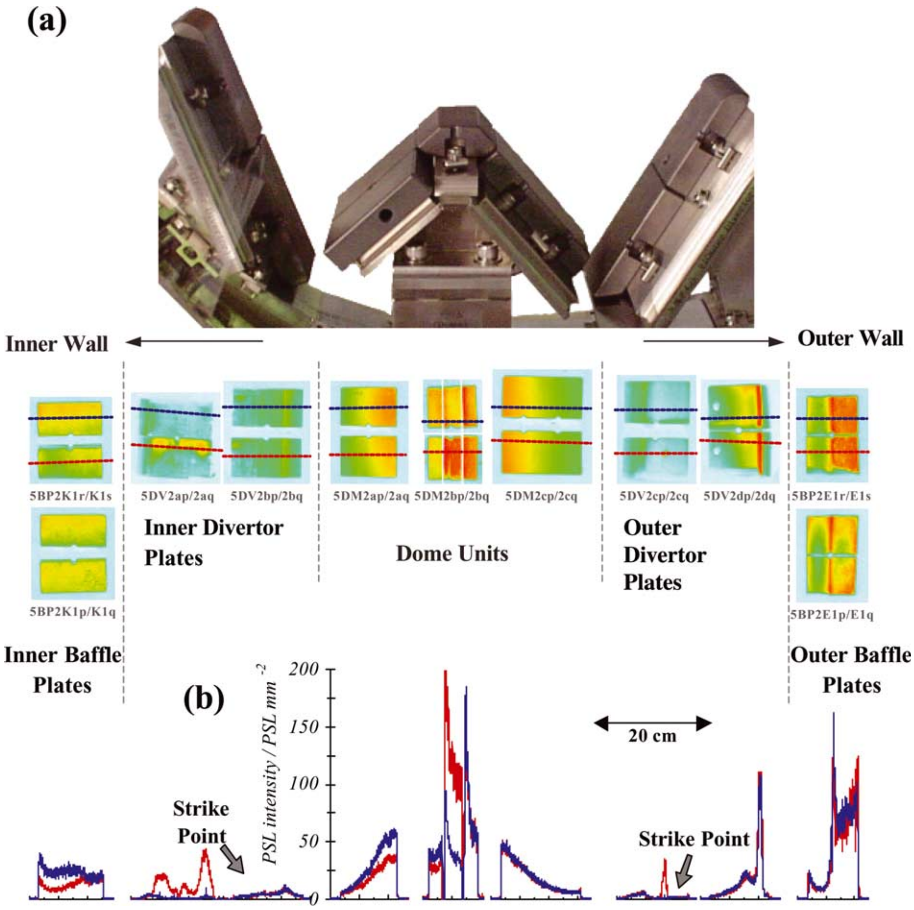

Fig. 2. (a) Tritium images of graphite tiles used as the divertor and the baffle plates in JT-60U. Tritium level is higher in the red region and less in the blue region. (b) Tritium line profiles along the poloidal direction.

JT-60U, the specific areal tritium level can thus be estimated to be around 10 kBq/cm^2. This is the same order of magnitude as the observed tritium level, suggesting that a significant amount of the tritium produced by D–D reaction was injected homogeneously over plasma facing surfaces without fully losing its initial energy of 1 MeV before impinging on the plasma facing surfaces. Of course, once implanted, tritium was not necessarily retained, resulting in a much lower tritium level in high heat loaded area.

There is an appreciable tritium distribution along the toroidal and poloidal directions in each tile. Fig. 3 shows the photograph and the tritium imaging for one of the outer divertor tiles. The tile was inclined to the plasma and the right edge was directly tapered to face the plasma on which the tritium level was consistent with the above discussion. The tritium distribution along the poloidal direction shown in the bottom is mostly due to the temperature gradient as discussed below. One can note the color pattern change along the toroidal direction which seems to correspond to the erosion and deposition pattern in the picture. That is probably because the tritium implanted rather deeply was hindered by the redeposited layers as observed in TEXTOR limiter tiles [6]. In JT-60U, owing to the rather high vacuum vessel temperature (around 600 K), the redeposited layer adhered well to the graphite substrate and is very hard to detect.

Doyle et al. [7] have given the temperature dependence of deuterium retention in graphite during hydrogen-ion irradiation. Normalizing the tritium level at the baffle plate to Doyle's saturated deuterium retention value in graphite after high-dose irradiation at 600 K given by the dashed line, we plotted in Fig. 4 the tritium level against the graphite tile temperature measured by a thermocouple at the back side. One can observe that the tritium level shown is consistent with Doyle's data. The deviation is most likely due to the temperature difference between the surface and the back side of the tiles. In other words, the surface temperature must have been

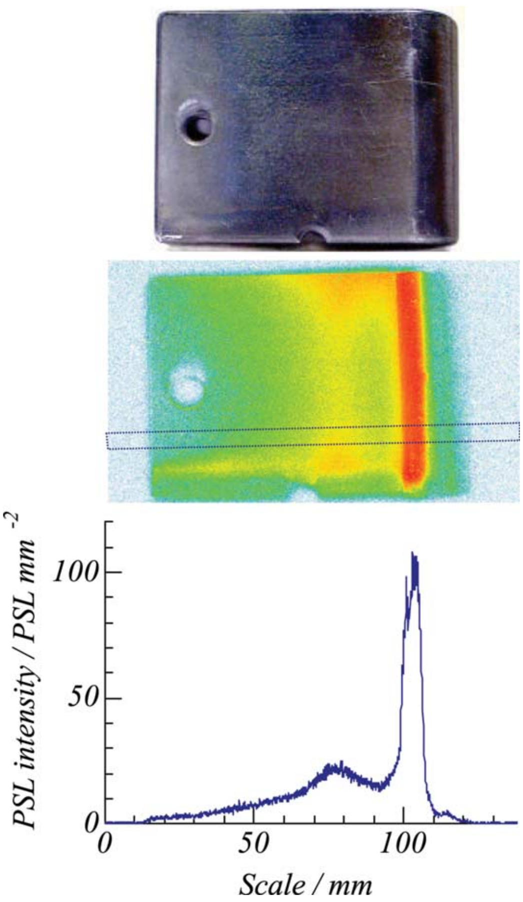

Fig. 3. Comparison of the photograph and tritium image for one of the outer divertor tiles.

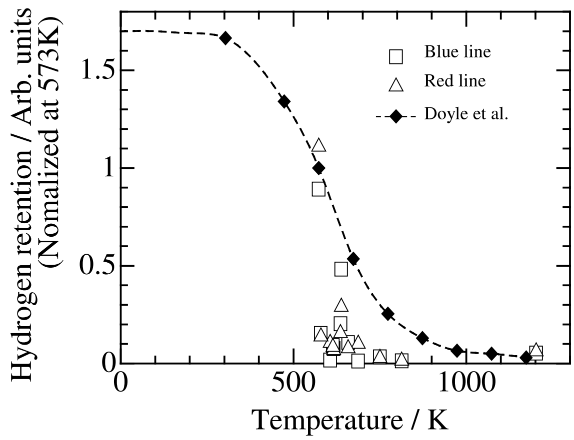

Fig. 4. Normalized tritium levels plotted against the graphite tile temperature measured by a thermocouple at the back side (see text). Data were taken from the line profiles given in Fig. 2. The dashed line is given by Doyle et al. [7] for saturated deuterium concentration after high-dose ion irradiation.

much higher than that measured by the thermocouples. Shifting all the points in Fig. 4 to the right hand side (higher temperature side) to meet the solid line, one can get hypothetical surface temperatures of the tiles. Assuming those hypothetical temperatures are the actual surface temperatures and the thermocouple temperatures on the back side, one can further estimate the heat load to the divertor plate. Assuming a steady heat flow with a thermal conductivity of graphite around 100 $W\,m^{-1}\,K^{-1}$, the heat load to the tiles was estimated to be around 1–5 MW/m^2 depending on the tile position, which is the same order of magnitude as the actual heat load. This means that the tritium retention was mainly controlled by the surface temperature and hence it is possible to keep the tritium inventory very small above 800 K.

Thus the tritium distribution observed in JT-60U tiles can be well explained by the homogeneous implantation of rather high energy tritium which is influenced by redeposited layers and redistributed by the temperature increase due to the plasma heat load. The observed tritium level is very consistent with this interpretation. Nevertheless, it should be mentioned that the present IP imaging missed some tritium 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 an additional reason why the tritium distribution was homogeneous without showing any pattern of the ripple loss, or flux inhomogeneity in the edge plasma. Thus most of the tritium detected by IP was retained rather stable in the graphite from the top surface to a depth of around 3.5 μm.

4. Conclusions

Tritium distributions on the graphite divertor tiles, the dome units and the baffle plates of JT-60U were successfully measured. The highest tritium level was found at the top of the dome and the outer baffle plates, where the plasma did not hit directly but the distance from the plasma was the shortest. This indicates that the tritium detected by IP is implanted very deeply. The tritium distribution on the top of the dome and the baffle plate tiles appeared to be homogeneous, probably because the 1 MeV tritium produced by the D–D nuclear reaction was hardly confined and impinged on plasma facing tiles only depending on the distance from the plasma. Simple calculations show most of the produced tritium was once implanted in the plasma facing tiles, which agrees with the long-term tritium retention in the vacuum vessel of about 50%. However, the temperature increase owing to the plasma heat load could release the once retained tritium. Thus the tritium retention in those areas heated above 800 K was actually very small. The present IP imaging could miss some tritium absorbed in near surface layer from low energy impinging after losing energy in the plasma, because surface tritium can be easily replaced by hydrogen and water molecules. This might be an additional reason why the tritium distribution on the baffle plates were homogeneous without any pattern corresponding to flux inhomogeneities. The poloidal and toroidal variations of the tritium distribution in each tile may reflect the redeposition. We can also conclude that the tritium retention in graphite above 800 K is really small.

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