Tritium Distribution on Plasma Facing Graphite Tiles of JT-60U

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Abstract: Tritium distributions on the graphite divertor tiles, the dome units and the baffle plates of JT-60U were successfully measured. Poloidally, the highest tritium level was found at the dome top tiles and the outer baffle plates, where the plasma did not hit directly. On the other hand, although the toroidal tritium profiles on each tiles appeared uniform, detailed profiles in full toroidal direction clearly showed a periodic variation corresponding to the position of the magnetic field coils, indicating the ripple loss of high energy tritons as suggested by the OFMC code. Finally, the temperature increase owing to the plasma heat load was found to release the once retained tritium.

1. Introduction

Tritium Imaging Plate Technique (TIPT) was found to be very useful to determine the tritium areal distribution on plasma facing graphite tiles. It gives very detailed tritium surface profiles and can also be used as a new diagnostic technique to investigate plasma wall interaction through tritium behavior [1,2]. In the present work, TIPT was applied to determine the surface tritium distributions on graphite tiles used as the first wall and the W-shaped divertor in JT-60U, in which tritium is produced by the D-D nuclear reaction.

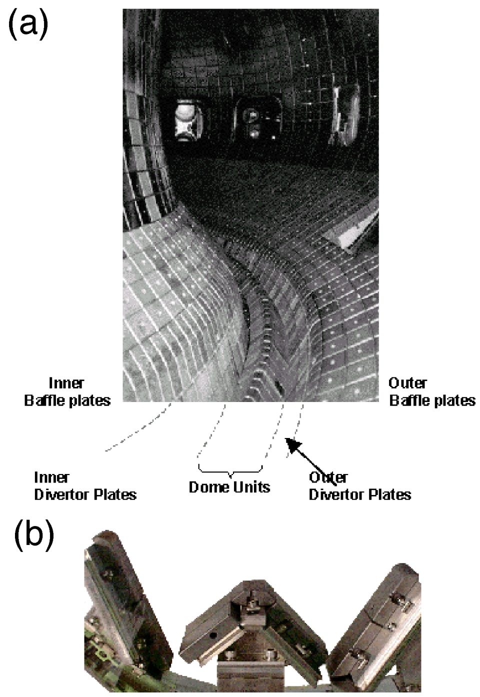

Fig.1 (a) Photograph of JT-60U inside and (b) Cross-sectional view of W-shaped divertor.

2. Experimental

2-1. Specimens

The poloidal cross-sectional view of JT-60U with the W-shaped divertor is shown in Fig. 1. CFC graphite (CX-2002U) tiles are used for the divertor targets, the dome top and parts of the baffle tiles. The rest of the divertor region and the first wall are covered with isotropic graphite tiles (IG-340U). All tiles are fixed to metal backings by bolts. The temperature of the divertor tiles was measured by thermocouples installed in the tiles at the depth of 6-mm depth.

The high temperatures of the thermocouples were observed for high neutral beam power, which is aimed at steady-state high performance operation. During the operation, the outer divertor tiles showed higher temperatures than inner ones. The highest surface temperature of the outer divertor tiles is expected to reach approximately 1100˚C. The dome and baffle tiles had a history of comparatively temperatures, nearly at the baking temperature of ~ 300˚C.

The first wall tiles (exposed to plasma from March 1991 to Nov. 1998) and the divertor region tiles (were exposed to plasma from Jun. 1997 to Nov. 1998) were removed for the detailed tritium measurement using TIPT. Furthermore, all tiles in full toroidal direction of the top of the dome units were also measured. Some of the tiles were also analyzed by full combustion and SEM observation for quantitative tritium analysis and deposition analysis, respectively. The total number of the discharges during the period from Jun. 1997 to Nov. 1998 was approximately 4,000 shots. The amount of tritium produced during this period, which was estimated from neutron production, was about 18 GBq [3].

Before the opening of the vacuum vessel of JT-60U, hydrogen discharges were used to remove the tritium retained in the vacuum vessel. This is followed by air ventilation before fully open to the atmosphere. Thus long term tritium retention in the JT-60U vacuum vessel was estimated to be about 50% of the total production [3].

2-2. Tritium Imaging Plate Technique

The imaging plate (IP) is a radiation image sensor based on photo-stimulated luminescence (PSL). The IP can detect tritium distributed within a depth of ~3.5 μm from the surface of graphite-based tiles. The surface of the IP was in contact with the sample tiles for a day in a dark shielded room. After the exposure, the IP was processed using an imaging plate reader to obtain a digitized tritium intensity mapping. The details of the IP technique is described elsewhere [4].

2-3. Simulation of the High Energy Tritons

The behavior of high energy tritons produced by the D-D reaction in a typical plasma operation of a high $ß_p$ H-mode was simulated by the Orbit Following Monte Carlo (OFMC) code, developed in JAERI [5,6]. In the code, coulomb collisions between energetic triton particles and the plasma are simulated using the Monte-Carlo method tracing the triton particle orbits in the magnetic fields. The fields are a combination of the axisymmetric field calculated by a two-dimensional magneto-hydrodynamic (MHD) equilibrium code and the non-axisymmetric field produced by the toroidal filed ripple. The Coulomb collisions lead to pitch angle scattering and slowing down of the energetic triton particles. The launching points and pitch angles of test triton particles are also determined by Monte-Carlo method. The test triton particles with the initial energy of 1 MeV are launched at a radius based on the birth profile of the energetic tritons. The orbit of each test particle is followed until it impinges on the wall or slows down to thermal speed.

3. Results

Figure 2 shows the tritium images of the divertor units and the baffle plates together with the line profiles corresponding to the red and blue lines in the images. The tritium profiles were non-uniform in the poloidal direction, but symmetric in the toroidal direction. It was rather surprising to see the highest tritium level observed at the dome top tiles and the outer baffle plates, where the plasma did not directly hit. Tritium levels on both sides of the dome units showed steep gradients in the poloidal direction. Tritium level on the divertor tiles were very small, in particular, both the outer and inner strike points showed the lowest levels owing to the temperature escalation during plasma. We shall discuss this point in further detail later.

Such poloidal tritium profiles were quite consistent with the tritium activity determined by the combustion method, and the highest tritium levels at the top of the dome unit was measured to be around $60 kBq/cm^2$ [7].

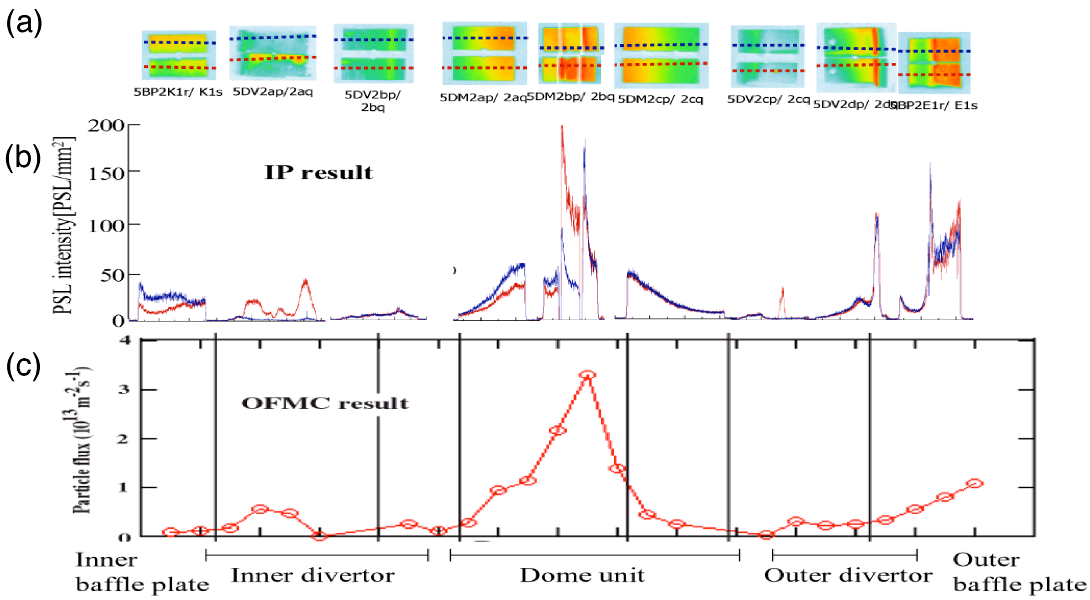

Fig.2 (a) Tritium images of graphite tiles used as the divertor and the baffle plates in JT-60U. Tritium level is higher in the red region and less in the blue region; (b) Tritium line profiles along the poloidal_direction; (c) tritium impinging flux to the divertor tiles calculated by OFMC code (see text).

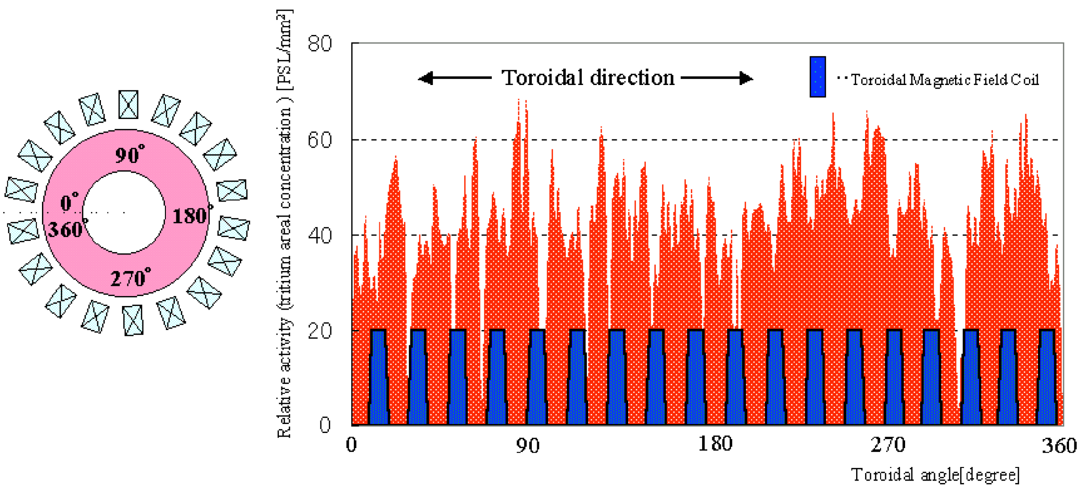

Fig. 3 Full toroidal distribution of tritium on dome top tiles. The positions of toroidal magnetic coils are indicated as columns.

Although the toroidal tritium distribution on each tile seems uniform, the full toroidal tritium distribution from all of the dome top tiles showed certain variation as seen Fig 3. In the figure, the tritium levels of all 240 toroidal dome top tiles are plotted against the toroidal angle corresponding to their positions, exhibiting a periodic variation. In JT-60U, 18 toroidal magnets are placed as indicated in the figure. One can clearly see the corresponding similar periodicity of the tritium retention in the toroidal direction. This is the first clear evidence of the loss of high energy tritons by the ripple in the toroidal magnetic filed (i.e., “ripple loss”).

Figure 4 (a) shows the tritium images of some selected first wall tiles. Although the tritium level was lower than that of the baffle plate, it still contained several kBq/cm^2. The tritium levels on the outer first wall tiles were higher than those for the inner first wall tiles. And the outer midplane showed the highest level among the first wall tiles.

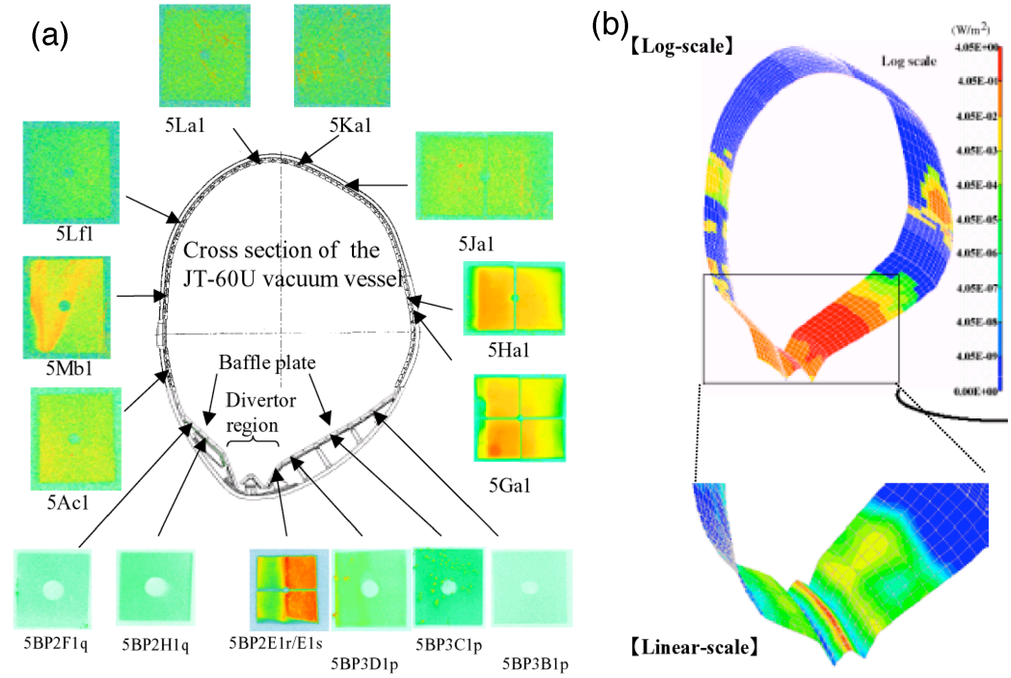

Fig. 4 Comparison of (a) tritium profiles for selected first wall graphite tiles and (b) flux of higher energy triton impinging to first wall by OFMC calculation.

It is interesting to know that the tritium profiles in the divertor tiles did not show a clear correlation with the deposited layers. Figure 5 compares tritium profiles with and without deposited layers on the inner divertor tile. The upper right image is a cross-sectional SEM view of the tile, indicating about 20 μm of deposition layer. The deposited layer of the bottom half of the inner divertor tile was exfoliated by an adhesive tape. One can clearly see a higher tritium level behind the deposited layer. This suggests that tritium was implanted even in those areas with less tritium implantation with the ripple loss mechanism, in other words, the tritium is not fully thermalized before impinging the target plates.

4. Discussion

As observed in the periodic variation of the full toroidal tritium distribution on the dome top tiles (see Fig.3), tritium is very likely retained as the high energy triton implanted through the ripple loss mechanism. Agreement of the measured tritium profiles with the impinging fluxes calculated by OFMC for the divertor region, (as shown in Fig. 2(c)), and the midplane tiles among the first wall tiles (as shown in Fig.4(b)) is also consistent with the ripple loss mechanism, since the OFMC takes account the toroidal filed ripples

In contrast to the dome top tiles, the tritium levels in the divertor tiles were quite low. In particular, both the outer and inner strike points showed the lowest level. The temperature increase of graphite tiles due to the plasma heat load was around 50-100K except in the divertor regions where the maximum temperatures of 800- 1200K were recorded at the inner and outer divertors, respectively. One also notes that the tritium level showed a gradient in the poloidal direction. This gradient was inversely correlated to the poloidal temperature distribution, indicating that the implanted tritium was thermally released, resulting in no tritium retention in the divertor legs [8].

Furthermore, it is also important to note that the tritium profiles in the divertor tiles did not show a clear correlation with the deposited layers as shown in Fig. 5. This is quite different from JET, where the highest tritium level was observed in the redeposited layer, particularly, on the plasma shadowed area [9,10]. In JT-60U, deposited layers were found mainly on the inner divertor tiles, and the deposited layer can not be clearly distinguished from the substrate [11]. This may be due to high temperatures during discharges, which seems to enhance the adhesion property of the deposits on the matrix with smaller H and D content than other large tokamaks [12].

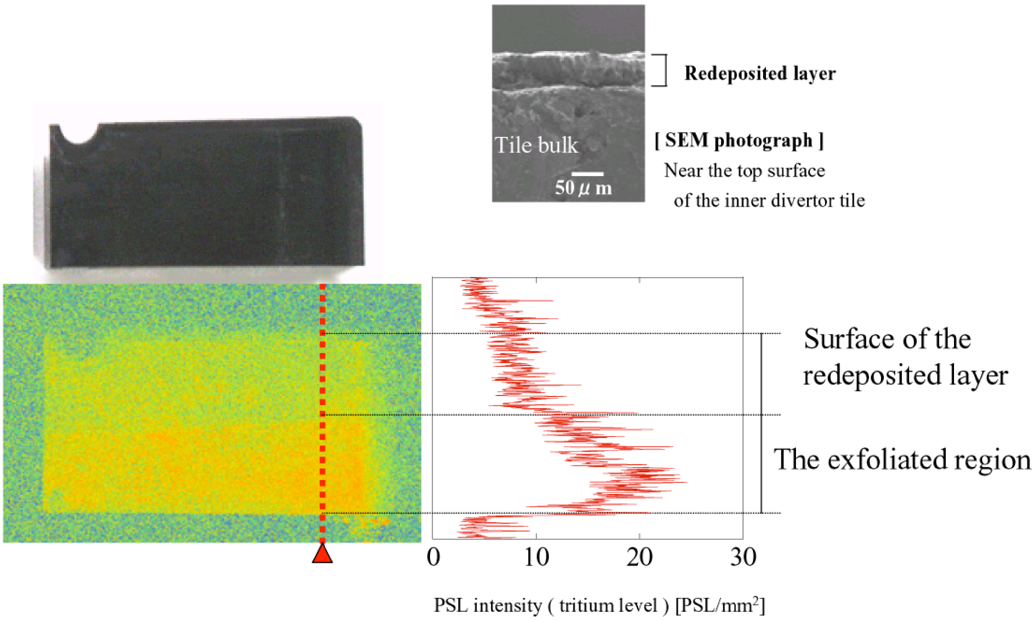

Fig. 5 Comparison of tritium profiles with and without deposited layers on the inner divertor tile. The upper right image is a cross-sectional SEM view of the tile, indicating about 20μm of deposition layer. The deposited layer of the bottom half of the inner dome wing was exfoliated by an adhesive tape, indicating higher tritium level in the matrix than in the deposited layer.

Finally according to the analysis of hydrogen and deuterium depth profiles measured for the same divertor tiles used in the tritium analysis [12], hydrogen and deuterium in JT-60U are distributed quite similarly with tritium, i.e., the area with lower tritium retention also shows lower hydrogen and deuterium retention. Since all three isotopes behave similarly during thermal release above 800K, this can be viewed as another indication that the temperature effect dominated tritium retention in the divertor region of JT-60U.

Thus in essence, we are now able to explain the observed tritium distribution in JT-60U divertor tiles by the combination of the implantation of high energy tritium and the simultaneous thermal release due to the heat load [4,13].

5. 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 dome top tiles and the outer baffle plates, where the plasma did not hit directly. Such high tritium retention in the dome units and the baffle plates can be well explained by the energetic triton particle loss due to the ripple loss mechanism.

According to the orbital simulation code of OFMC, about 1/3 of tritons produced by D-D reaction impinge plasma facing surfaces without fully losing its energy. In particular, the impinging flux is high on the dome area and the baffle plates. Although the toroidal tritium profiles on each tiles appeared symmetrical, detailed profile in full toroidal direction clearly showed a periodic variation in the direction of the toroidal magnetic fields, confirming the ripple loss of high energy triton as suggested by the OFMC code.

In addition, the tritium retention in divertor tiles heated above 800K was actually very small, indicating that the temperature increase owing to the plasma heat load results in the release of the once retained tritium.

The present IP imaging could have missed some tritium adsorbed or 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

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References

[1] T. Tanabe and V. Philipps, Fusion Eng. Design, **54**, 147 (2001).
[2] K. Miyasaka, T. Tanabe, G. Mank et al. J. Nucl. Mater. **290-293**, 448(2001).
[3] K. Masaki, K. Kodama, et al., Fusion Eng. and Design, **31**, 181 (1996).
[4] T. Tanabe, K. Miyasaka, M. Rubel, V. Philipps, Fusion Science and Technology, **41**, 528 (2002).
[5] K. Tani, M. Azumi, H. kishimoto, S. Tamura, J. Phys. Soc. Jpn 50 (1981) 1726.
[6] K. Tobita, S. Nishio, S. Konishi, et al., Fusion Eng. & Desgin., to be published.
[7] K. Masaki, K. Sugiyama, T. Tanabe, et al., Proc. 15th PSI, J. Nucl. Mater. to be published.
[8] K. Miyasaka, T. Tanabe, K. Masaki and N. Miya, Proc. ICFRM-10, J. Nucl. Mater, in press.
[9] R.-D. Penzhorn, J.P.Coad, N. Bekris; L. Doerr, M. Friedrich, W. Pilz. Tritium in Plasma facing components. Fusion Engineering and Design **56-57**, 105 (2001).
[10] R. -D. Penzhorn, N. Bekris, U. Berdnt, et al. Trtium depth profiles in graphite and carbon comosie materials exposed to tokamak plasmas, J. Nucl. Mater., **288**, 170 (2001).
[11] Y. Gotoh, J. Yagyu, K. Kizu, et al. Proc. 15th PSI, J. Nucl. Mater. to be published
[12] Y.Hirohata, Y.Oya, H.Yoshida, et al. Physica Scripta, in press
[13] K. Sugiyama T. Tanabe, K. Masaki et al. Physica Scripta, in press