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Retention characteristics of hydrogen isotopes in JT-60U

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Abstract

Erosion/deposition distribution and hydrogen isotope behavior in the JT-60U plasma-facing wall were investigated. Distribution of the tritium, which was produced by D–D nuclear reaction, was not correlated with erosion/deposition distribution. The tritium distribution can be explained by the distribution of high-energy tritium ion-implantation due to ripple loss. Deuterium distribution in the divertor region was different from the tritium distribution and not well correlated with the deposition. The highest D/C was ~0.05 at the bottom of the outer dome wing, which is much less than that observed in other tokamaks. For the deuterium retention, at least two retention processes (ion-implantation and co-deposition) were found on the dome region. The systematic dust collection gave the small amount of dust (~7 g: 0.2 mg/s production) in the whole vessel of JT-60U. Deposition was observed at the remote area of the outer divertor region.

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Keywords: JT-60U; Tritium; Deuterium inventory; Hydrogen inventory; Erosion and Deposition

1. Introduction

Tritium retention is a critical issue for a next step fusion device with plasma-facing carbon wall. For the study of the tritium retention, various postmortem analyses mainly for the JET Mk-IIA divertor tiles have been performed extensively and very important results have been reported so far [1–3]. The maximum D/(C + Be) ratio in the JET Mk-IIA divertor tiles was reported to be very high ~0.8 [2]. The thick deposition film, which had a high D/C ratio ~0.4, was found on the louvers at the inner corner [4].

In order to study the tritium retention in different conditions (divertor geometry and operation temperature), we have investigated the hydrogen isotopes (H, D, T) behavior in the JT-60U plasma-facing carbon wall. The operation temperature of the JT-60U vacuum vessel was ~570 K. The base temperature of the divertor tiles were also ~570 K, since the divertor tiles were inertially cooled. The divertor of JT-60U has a 'W-shaped' geometry. Neutral particles were pumped from the pumping slots in the private region.

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This paper summarizes the recent results of the post-mortem analyses of plasma-facing materials in JT-60U. Results of the systematic dust collection and the observation in remote area are also described.

2. Samples

Fig. 1 shows schematic views of the JT-60U W-shaped divertors with an inner pumping slot and both side pumping slots. The inside of the vacuum vessel is fully covered with carbon-based wall. The operational temperature of the vacuum vessel was ∼570 K. Samples for the present study were the first wall tiles exposed to plasma from March 1991 to October 1998 and the divertor targets and the dome tiles exposed from June 1997 to October 1998. In the period from June 1997 to October 1998, the W-shaped divertor had an inner pumping slot. The total number of deuterium discharges during this period was ∼3600 shots. Following the deuterium discharges, ∼700 hydrogen shots were performed in a clean-up operation. The total amount of the tritium produced in the period was 18 GBq, which was estimated from neutron production. The temperature of the outer strike point tended to be higher than that of the inner strike point. The maximum surface temperatures of the inner and the outer divertor tiles estimated from the thermocouple measurements were ∼1000 K and ∼1400 K, respectively. The maximum surface temperature of the dome region was estimated to be ∼800 K.

In order to investigate dust and/or flake and deposition in remote area, the collection of dust and/or flake samples and the in-vessel inspection of the plasma-facing components were performed in August 2003. It should be noted that an outer pumping slot was added to the W-shaped divertor in 1998. Approximately 8500 shots were performed from 1998 to 2003 with the both side pumping slots.

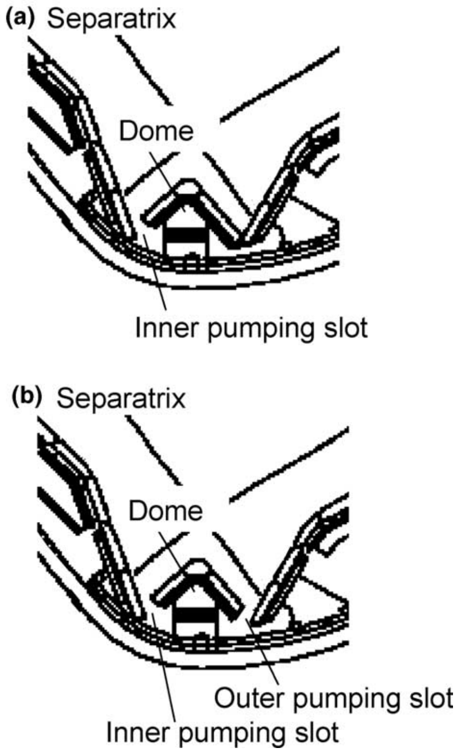

Fig. 1. Schematic view of the JT-60U W-shaped divertors. (a) The W-shaped divertor with an inner pumping slot was installed in June 1997 and (b) an outer pumping slot was added to the W-shaped divertor in December 1998.

3. Results

3.1. Erosion and deposition in the W-shaped divertor

Sample tiles for these analyses were divertor target and dome tiles, which were exposed to plasma from June 1997 to October 1998 with the plasma conditions of electron temperature: 15–50 eV and electron density: $\sim 10^{19}\,m^{-3}$ at the outer strike point [5]. Fig. 2 shows poloidal distribution of deposition layer thickness and net erosion depth in the W-shaped divertor region of JT-60U, which was measured with a scanning electron microscope (SEM) and a dial gauge, respectively [6]. In the SEM analyses, deposition layer thickness as small as ∼several micrometer was observed. As seen in

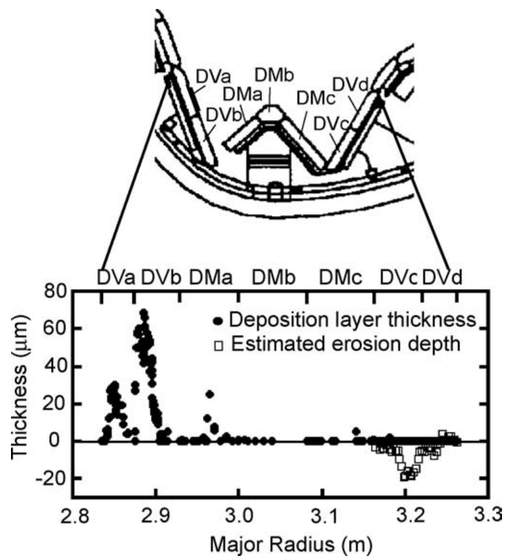

Fig. 2. Poloidal distribution of deposition layer thickness and erosion depth on the W-shaped divertor.

Fig. 2, deposition was dominant on the inner divertor, whereas erosion was dominant on the outer divertor. No continuous deposition layer was obviously observed in the dome top tile. Such in/out asymmetry of the erosion/deposition has been observed also in many tokamaks [1,7]. In JT-60U, average deposition thickness and erosion depth of the in/out divertor target tiles were ∼27 μm and ∼8 μm, respectively. Considering the total neutral beam injection (NBI) time (1.2×10^4 s, average NBI power: 8 MW) in the operation period, the average deposition growth rate and net erosion rate were estimated to be 2.3 nm/s and 0.7 nm/s, respectively. These values are not so much different from other tokamak experiences [1,8].

3.2. Tritium retention in the plasma-facing wall surface

In JT-60U, tritium is produced by D–D nuclear reaction. The tritium behavior is expected to be different from deuterium and hydrogen behaviors, because the tritium produced by the D–D nuclear reaction has an initial high energy of 1 MeV. In order to evaluate the tritium distribution on the plasma-facing wall, the detailed tritium profiles on the JT-60U W-shaped divertor and the first wall tiles were examined by imaging plate (IP) technique and full combustion method [9–12]. The divertor sample tiles for these analyses were the same as those used for the erosion/deposition analyses.

Fig. 3 shows tritium distribution obtained by IP. The highest tritium level was found at the dome top tile located in the private region. The tritium level in the inner and outer divertor target tiles were low. In a full toroidal measurement of the dome top tiles, the tritium was distributed periodically, which was correlated to the toroidal magnetic field (TF) coil positions. In the first wall region, the outer midplane first wall tiles had a higher tritium level compared with the inner first wall tiles. The quantitative analyses by the combustion method showed that the highest tritium concentration was ∼60 kBq/cm^2 at the dome top tile, whereas the tritium concentrations in the inner and the outer divertor targets were ∼2 kBq/cm^2 and ∼250 Bq/cm^2, respectively. The amount of the tritium retention in the divertor region was ∼12% of produced tritium (18 GBq). As described in Section 3.1, the inner divertor target tiles have thick deposition layers, whereas the dome top has no continuous thick deposition layers. These results show that deposition was not main process for the tritium retention.

In order to make clear the process of the tritium retention in the plasma-facing wall, we applied orbit following Monte-Carlo (OFMC) code to a tritium ion (triton) deposition simulation considering the ripple transport induced by TF ripple [13,14]. In this code, the orbit of each test particle is followed until it impinges on the wall or slows down to plasma temperature (∼10 keV). Fig. 4 shows a comparison between the results of IP and OFMC simulation, which was performed using typical plasma condition of a high β_P H-mode, in the poloidal direction of the divertor region. The simulation result of the triton flux distribution on the plasma-facing surface corresponded to the tritium distribution obtained by IP and full combustion method. These tritons were implanted into the wall without fully losing the initial high energy (up to 1 MeV). According to the simulation considering high β_P H-mode plasma and reversed shear plasma in the operation period from June 1997 to October 1998, ∼50% of the produced triton were lost from the plasma and implanted deeply into the wall. The amount of the triton deposition in the divertor

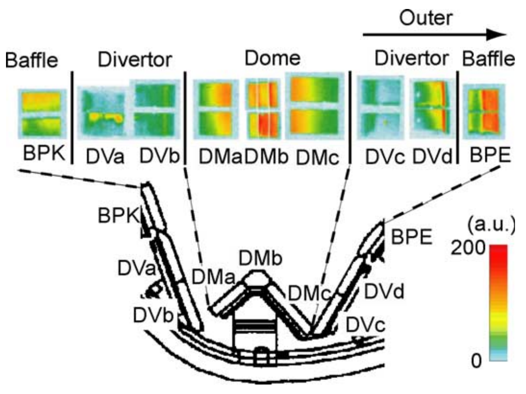

Fig. 3. Tritium distribution obtained by IP. This poloidal tritium distribution obtained by IP was consistent with that measured by the combustion method.

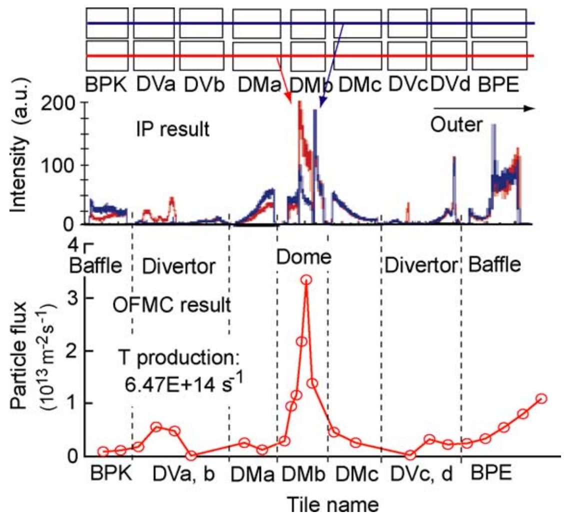

Fig. 4. A comparison between the results of IP and OFMC simulation. The triton particle fluxes obtained by OFMC simulation are averaged in the toroidal direction.

region obtained by OFMC simulation was ∼14%, which is consistent with the quantitative analyses (∼12%).

These results indicate that the tritium distribution of the JT-60U W-shaped divertor reflects mainly the distribution of the energetic triton impinging on the plasma-facing wall due to ripple transport. Partly or fully thermalized tritons should have impinged to near surface region of the plasma-facing wall. The tritium in the divertor target tiles, however, probably was desorbed, because of a high surface temperature as described in Section 2. In addition, some tritium in the near surface region was replaced by deuterium and hydrogen in subsequent discharges and in a clean-up operation.

3.3. Deuterium and hydrogen retention in the divertor surface

Deuterium and hydrogen retentions in the W-shaped divertor tiles were also investigated. Qualitative analyses of hydrogen and deuterium in the tiles were performed using secondary ion mass spectroscopy (SIMS) [15–17]. Quantitative analysis of the deuterium was performed by nuclear reaction analysis (NRA) (D(d, p)T reaction) on the Fusion Neutronics Source (FNS) in JAERI [18,19]. The accelerated energy of the incident deuterons was 350 keV. The deuterium depth profiles within ∼2.0 μm from the plasma-facing surface can be measured in the analysis. D/C ratios of each sample were obtained assuming the carbon density of 1.8 g/cm^3, which was measured by ^{12}C(d, p)^{13}C for CFC substrate. The divertor sample tiles for these analyses were the same as those used for the erosion/deposition analyses.

Fig. 5 shows the depth profiles of the $D^-/^{12}C^-$ and $H^-/^{12}C^-$ signal intensity ratios in the dome top tile, which were obtained by SIMS. Deuterium in the near surface regions was mostly replaced with hydrogen,

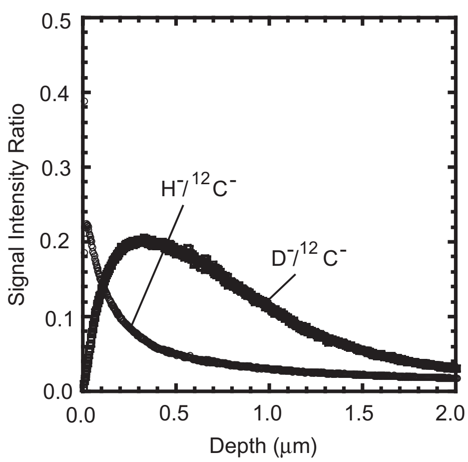

Fig. 5. Depth profiles of $D^-/^{12}C^-$ and $H^-/^{12}C^-$ signal intensity ratios in the dome top tile, which were obtained by SIMS.

since a clean-up tokamak operation using hydrogen was performed after deuterium operation. Fig. 6 shows the D/C ratio obtained by NRA in the divertor region. The D/C was quite low in the divertor target tiles (less than 0.01), and even that for the inner divertor with the thick deposition layer was low. The highest D/C in the W-shaped divertor region of JT-60U was found at the outer dome wing tile. Still the highest D/C ratio was only ∼0.05 (6.8 × 10^{17} D atoms/cm^2), which is much less than that observed in other tokamaks [1,2,8]. Considering the H^-/D^- signal intensity ratio obtained by SIMS and the D/C ratio obtained by NRA, we have tried quantitative estimation of (H + D)/C in the divertor region. The result showed the highest (H + D)/C value of ∼0.07, still low, at the outer dome wing tile. Such low (H + D)/C must be attributed to relatively high surface temperature of the dome tiles (∼800 K). The saturation concentration of the hydrogen implanted in graphite is about 0.4–0.5 (H/C) at room temperature and decreases with temperature [20]. The H/C at 800 K is reported to be ∼0.08 [20], which is in good agreement with (H + D) retention in the dome tiles of JT-60U.

The D/C distribution in the dome region changed with the poloidal position significantly. The D/C poloidal distribution probably shows the difference of deuterium retention processes in each position. Applying the OFMC simulation, orbit following simulation for ∼90 keV deuterium injected as neutral beam was carried out. The result showed that the distribution of the high-energy deuterium ions (∼several 10 keV) was almost similar to that of the tritium distribution. Accordingly, the deuterium ions were implanted to the dome top or upper part of the outer dome wing with fluence and energy of ∼10^{19} cm^{-2} (0.4% loss of injected D, ∼8 MW × ∼1.2 × 10^4 s) and ∼50 keV, respectively, in the operation period. Under such high fluence condition, the deuterium concentration must be saturated. At the bottom of the outer dome wing, however, the flux of such high-energy deuterium was very small. Hence, the D retention on the bottom of the outer dome wing tile could not be originated from high-energy deuterium implantation. That is also not likely due to low-energy deuterium flux, because neutral pressure at the inner divertor region was higher than that at the outer divertor region in typical operation of JT-60U [21]. In contrast, the D/C ratio in the outer dome wing tile was higher than that of the inner dome wing as shown in Fig. 6. The depth profiles of the $D^-/^{12}C^-$ and $H^-/^{12}C^-$ signal intensity ratios obtained by SIMS in the bottom of the outer dome wing are shown in Fig. 7. The deuterium in near surface region was mostly replaced with hydrogen. The $D^-/^{12}C^-$ increased with the depth without saturation, whereas the dome top tile shows the maximum $D^-/^{12}C^-$ at around 0.5 μm as indicated in Fig. 5. From the deposition measurements in Fig. 2, the dome top has no continuous deposition layer, while

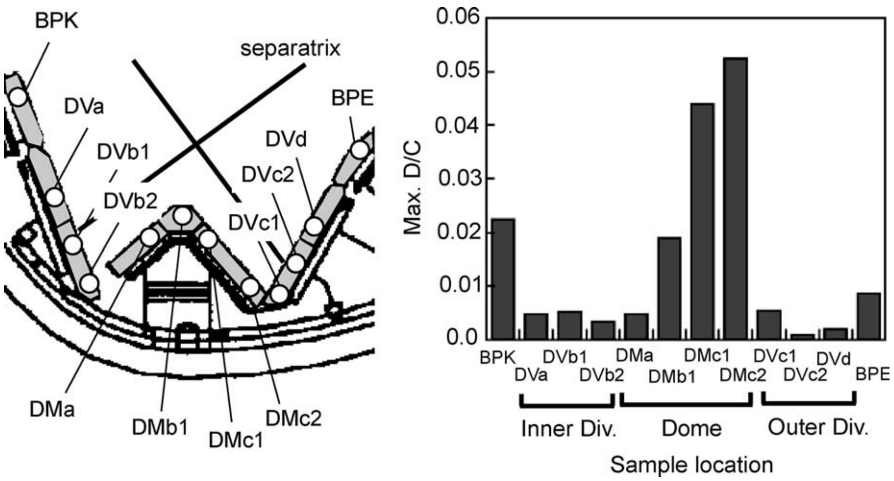

Fig. 6. D/C distribution in the divertor region. In this figure, the maximum D/C values obtained by NRA are plotted for each sample tile.

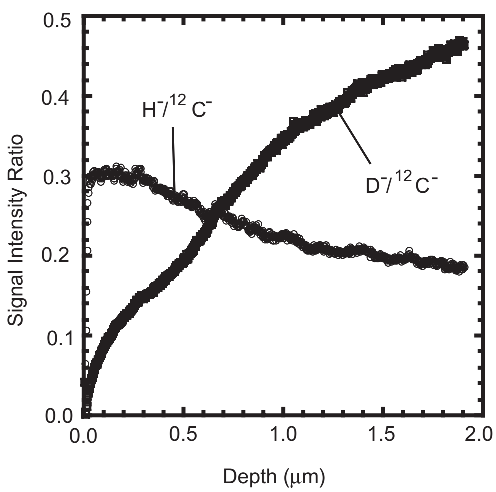

Fig. 7. Depth profiles of $D^-/^{12}C^-$ and $H^-/^{12}C^-$ signal intensity ratios obtained by SIMS in the bottom of the outer dome wing tile.

the bottom of the outer dome wing has clear deposition layer. All these observation suggests deuterium co-deposition on the bottom of the outer dome wing tile. Thus, D retention on the dome region can be attributed to high-energy deuterium implantation on the top area and the co-deposition with eroded carbon at the bottom of the outer dome wing.

3.4. Dust and deposition in remote area

A dust collection and an in-vessel inspection of the plasma-facing components were performed in August 2003. The dust was collected from part of the plasma-facing wall surface and underneath the in-vessel components. The total amount of the collected dust was ~170 mg. Most of the dust was underneath the outer divertor units. The averaged surface mass density at the bottom region of the vacuum vessel was ~400 mg/m^2, which was much smaller than that observed in JET and ASDEX Upgrade [22]. Details from the analysis of the dust collected from JT-60U are reported in Sharpe et al. [23]. The total amount of the dust in the whole vessel of JT-60U was estimated to be approximately 7 g, which corresponds to dust production rate of 0.2 mg/s (13 000 shots, NBI ~3×10^4 s).

From the in-vessel inspection, no appreciable deposition was observed in the remote area except underneath the outer divertor pumping slot. Fig. 8 shows a photograph of deposition underneath the divertor components. The deposition was stuck strongly on the vacuum vessel, and no flakes were produced even at vessel opening (room temperature). It should be noted that the deposition was found at the remote area of the outer divertor region, while the deposition was dominant at the inner in JET. Such difference in the deposition profiles for JET and JT-60U must be correlated to the differences in divertor geometry and operation temperature (JT-60U divertor: above 570 K and JET divertor: water cooled). However, we have no good explanation at the moment.

Analyses of the deposition and the hydrogen isotope retention in the remote area are still ongoing. These observations, however, imply that the deposition distribution in the remote area depends on divertor geometry and/or operation temperature. The difference of the deposition distribution in the remote area probably leads to a difference of the hydrogen isotope retention/distribution.

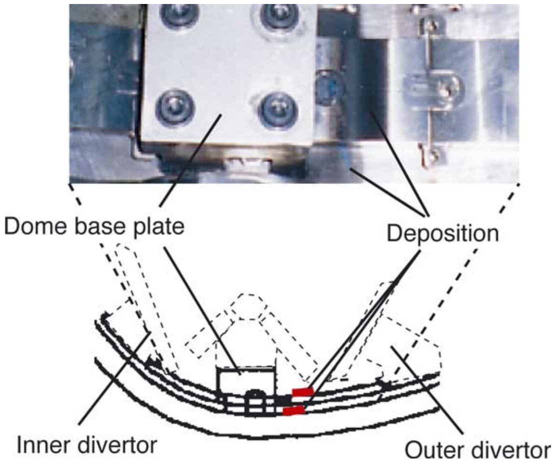

Fig. 8. Deposition underneath the divertor components.

4. Conclusions

In order to study the tritium retention in terms of different divertor geometry and operation temperature, we have investigated the erosion/deposition distribution and the hydrogen isotopes (H, D, T) behavior in the JT-60U plasma-facing wall.

Deposition was found to be dominant on the inner divertor target, whereas erosion was dominant on the outer divertor target. No continuous deposition layer was obviously observed on the dome top tile. In JT-60U, correlation between the distributions of the hydrogen isotopes and the carbon deposition profiles were not clear.

Distribution of the tritium, which was produced by D–D nuclear reaction, in the plasma-facing wall reflected the distribution of high-energy tritium ion-implantation due to ripple loss and a slight modification owing to high surface temperature of the divertor target tiles. According to OFMC simulation, ~50% of the produced tritium were lost and implanted into the wall with high energy of up to ~1 MeV.

The highest (H + D)/C was estimated to be ~0.07 at the outer dome wing, which is much less than that observed in other tokamaks. Such low (H + D)/C must be attributed to high surface temperature of the dome tiles (~800 K). For the deuterium retention, at least two retention processes (ion-implantation and co-deposition) were distinguished on the dome region.

The systematic dust collection showed the total amount of the dust in the whole vessel of JT-60U was significantly smaller than that observed in other tokamaks. Deposition was found at a remote area of the outer divertor region, which is the opposite side to the JET result. The difference of the deposition distribution in the remote area probably leads to a difference of the hydrogen isotope retention/distribution.

From these hydrogen isotope analyses, it was found that not only the co-deposition but also the ion-implantation should be considered to make reliable estimation of the tritium retention for a next step fusion device.

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