



Tritium removal by isotopic exchange

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ABSTRACT

This paper discusses how isotopic replacement is effective to remove tritium retained in the plasma facing surface in a DT reactor based on our recent studies on retention of hydrogen isotopes (H, D, and T) in plasma facing carbon tiles used in JT-60U. The isotope ratio of D and H in hydrogen retained near surface layers of the plasma facing wall is easily equilibrated with the flux ratio of hydrogen isotopes impinging the surface. Therefore DD discharges after DT discharges would effectively remove T retained in the surface layers during DT discharges. The efficiency of the replacement is higher and deeper for higher temperatures.

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1. Introduction

In ITER, tritium inventory in carbon materials is one of the most important issues for safety [1,2]. Most of the present tokamaks employ carbon materials (graphite and carbon fiber composite (CFC)) and the carbon materials retain a large amount of hydrogen isotopes and eroded carbon re-deposits incorporating hydrogen isotopes resulting in substantially increase of hydrogen retention. The large tritium inventory in the carbon materials would require frequent removal of the retained tritium.

The efficiency of the tritium removal technique is critically dependent on how and where tritium is retained. In our recent work on hydrogen isotopes retention in plasma facing tiles of JT-60U, we have found that distribution of hydrogen isotopes is quite non-uniform in depth, poloidal and toroidal positions, and eroded and deposited areas. In addition discharge history and tile temperatures significantly modify the depth distribution [3–10].

JT-60U was usually operated by deuterium (D), and hydrogen (H) discharges were employed for removing tritium (T) produced by the DD reaction. Hence all isotopes of H, D, and T are distributed very inhomogeneously on the vacuum vessel, depending on the location, temperature, and impinging energies of the isotopes. This mixture of H, D, and T could give us important information for T removal in a future reactor.

Taking profiles of re-deposition/erosion, heat load to the tiles and hence changes of surface temperatures during plasma dis-

charges into account, the mechanism of incorporation of H and D in eroded and re-deposited carbon tiles in the W-shaped divertor region [7,8] and main chambers [9,11] is discussed. Based on these results, we suggest how isotopic exchange during plasma operation is effective to remove tritium retained in the plasma facing surface in a DT reactor.

2. Analyzed tiles

Fig. 1 shows locations of tiles analyzed so far in (a) divertor, (b) poloidal, and (c) toroidal cross-sections. Divertor tiles exposed to high heat load were made of CX-2002U, Toyo Tanso Co. Ltd., dome top tiles and the inner baffle plates, PCC-2S, Hitachi-chemical Co. Ltd. Tiles of the other area were made of graphite (IG-430U, Toyo Tanso Co. Ltd.) [3].

Carbon deposition and erosion profiles on plasma facing carbon tiles both toroidal and poloidal directions were observed by a scanning electron microscope (SEM) and a micrometer. Histories of plasma operation and of plasma exposure of analyzed tiles are summarized in Fig. 2 [7,12,13]. In the divertor area, the inner divertor and the dome tiles were deposited, while the outer divertor tiles were mostly eroded. Although, the divertor structure was changed from the one side pumping to the both side pumping at 1999, erosion and deposition patterns were not changed except the inner dome wing tile, which changed from deposition to erosion. Most of the outboard first wall tiles were eroded [9]. Clear deposition was observed at the inboard first wall tiles near the baffle plate [9,10]. In addition some of the first tiles retained boron dominant layers produced by boronization occasionally made, which is not discussed here.

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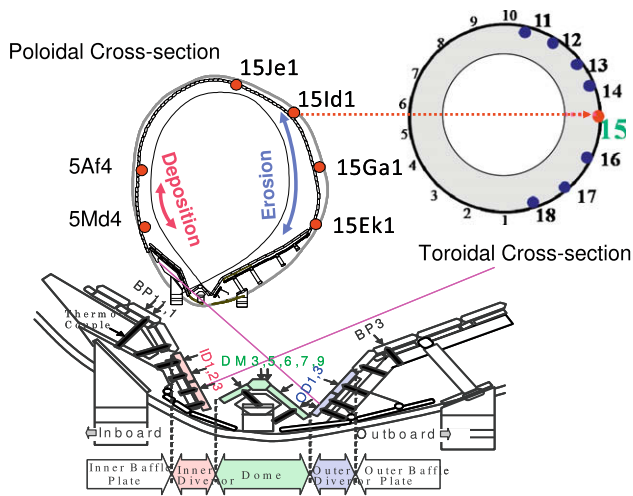


Fig. 1. Locations of analyzed tiles in (a) divertor, (b) poloidal, and (c) toroidal cross-sections.

Usually the JT-60U vessel was kept at 573 K using a baking system. During a discharge, the tile temperature further increased owing to plasma heat load to the maximum appeared at the end of the discharge. The maximum temperatures observed at the inner divertor, the dome, and the outer divertor were 750, 600, and 850 K, respectively. Based on these tile temperature profiles, the maximum surface temperatures of the respective area were estimated by Masaki et al. using the finite element modeling (FEM) analyses [3] to be approximately 1000, 800, and 1400 K, respectively. The temperature of the first wall tiles was not measured but must be slightly increased from the base temperature of 573 K.

Hydrogen and deuterium retention profiles were investigated by thermal desorption spectroscopy (TDS) and secondary ion mass spectroscopy (SIMS). The detail, were given in elsewhere [5,7,8].

3. Hydrogen retention characteristics examined by TDS and SIMS

Fig. 3 compares TDS spectra and SIMS depth profiles [5,8,11] for (a) ID3, an inner divertor tile with very thick re-deposited layers, (b) DM3, the dome top tile with thick re-deposited layers, (c) OD1, an eroded outer divertor tile, and (d) Ga1, an eroded outboard first tile. It should be noted that higher tile temperature reduces the total hydrogen retention.

The different retention characteristics among the tiles appeared in Fig. 3 should be correlated to the tile temperature, and energy and flux of impinging hydrogen isotopes. TDS peak temperatures clearly reflect the tile temperature during plasma exposure, showing the highest for the outer divertor tile and the lowest for the first wall tile. The highest temperature of the eroded divertor resulted in the lowest concentration in spite of the highest particle flux. In addition, H desorption dominates for the divertor tiles, while D dominates for the first wall tiles. This indicates that hydrogen retention near surface layers saturates and isotopic replacement of D retained during the DD discharges by H during the HH discharges subsequently made, is very effective, as discussed in the next section.

One can note deep penetration of D in the first wall, which is attributed to the injection of high energy deuterium (both neutrals and ions) originated from D NBI due to ripple loss and/or orbital loss mechanisms [3,5,6]. Such deep injection would be difficult to be replaced by isotopic replacement from the surface and could have significant contribution on long term retention irrespective of wall materials.

4. Discussion

The total desorbed amounts (H + D) with separation of H and D are compared for all first tiles measured here (Fig. 4(a)) [12] and compared with those of the eroded and re-deposited divertor tiles (Fig. 4(b)) [8]. Although, there are some differences depending on the tile locations among the first wall tiles, the total retention in the first wall tiles is clearly smaller than those for the re-deposited divertor tiles (the inner divertor and the outer dome wing tiles) but similar to that for the eroded tiles (the outer divertor tile). It is somewhat surprising why the total retention (H + D) in the eroded first wall tiles and the eroded divertor tiles were nearly the same. In spite of the lower temperature of the first wall, the retention normalized by a unit area and the total discharge duration for the first wall tiles are a few times less than that for the eroded divertor tile (the first wall tiles were exposed a few times more discharges). This suggests the saturation of hydrogen retention [14] near surface layers of the eroded tiles except the deep penetration due to the diffusion.

Fig. 5 summarizes H retention and D retention within 1 mm thickness from the top surface measured by TDS for all JT-60U tiles so far measured [8,11]. There are two different tendencies for the divertor tiles depending on the temperature irrespective of eroded tiles or redeposition ones. As already mentioned, every DD discharge campaign was followed by HH discharges to remove tritium and accordingly D retained during the DD discharge period was

Year	1997	1998	1999	2000	2001	2002	2003	2004	2005	
Operation		Inner side pumping	Both sides pumping				Both sides pumping			
Sample exposure period	Tiles for D,H retention study - Divertor						Collector probes for deposition, D,H retention studies in shadowed area			
	Tiles and dust for erosion/depositino study						Main chamber			
Temperature	~570 K baking (short pulse operation ~15s)						~420 K baking (long pulse operation ~60s)			

Fig. 2. Histories of plasma operation and of plasma exposure of analyzed tiles.

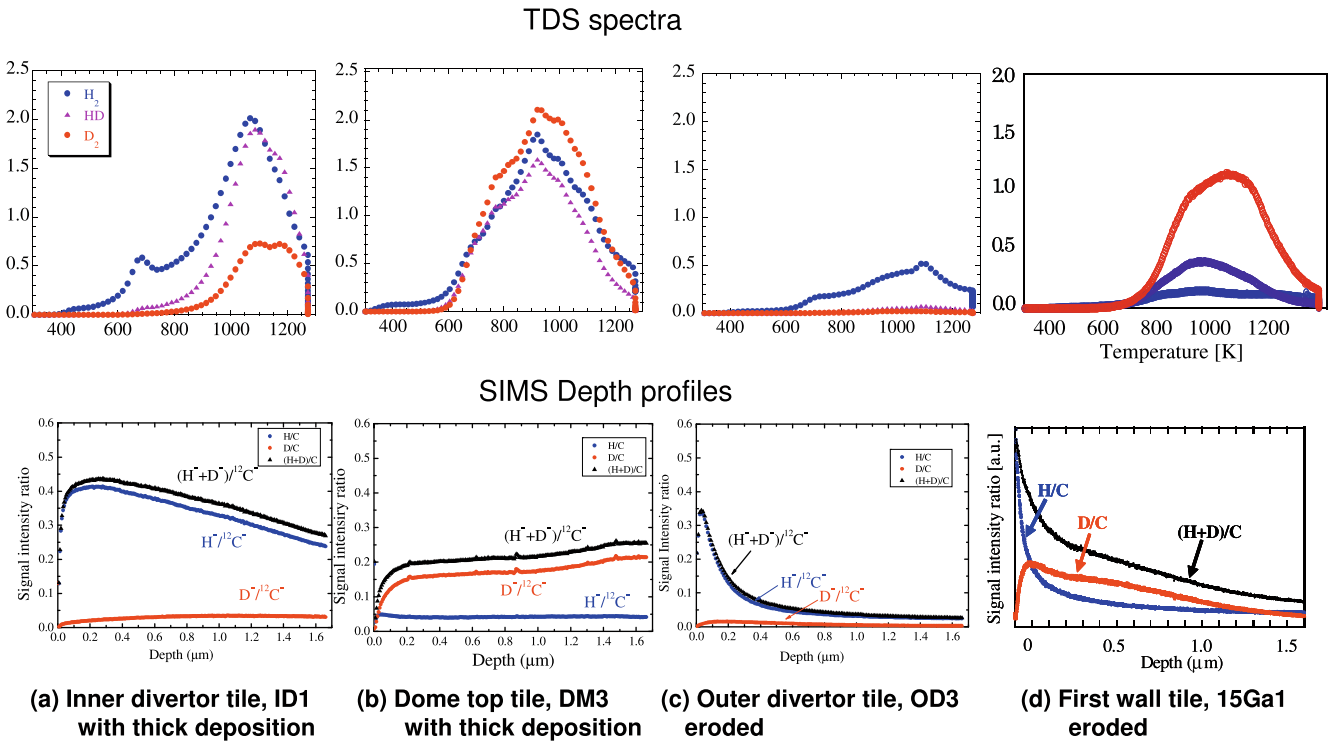


Fig. 3. TDS spectra and SIMS depth profiles for (a) ID3, an inner divertor tile with very thick re-deposited layers, (b) DM3, the dome top tile with thick deposited layers, (c) OD1, an eroded outer divertor tile, and (d) Ga1, an eroded outboard first wall tile.

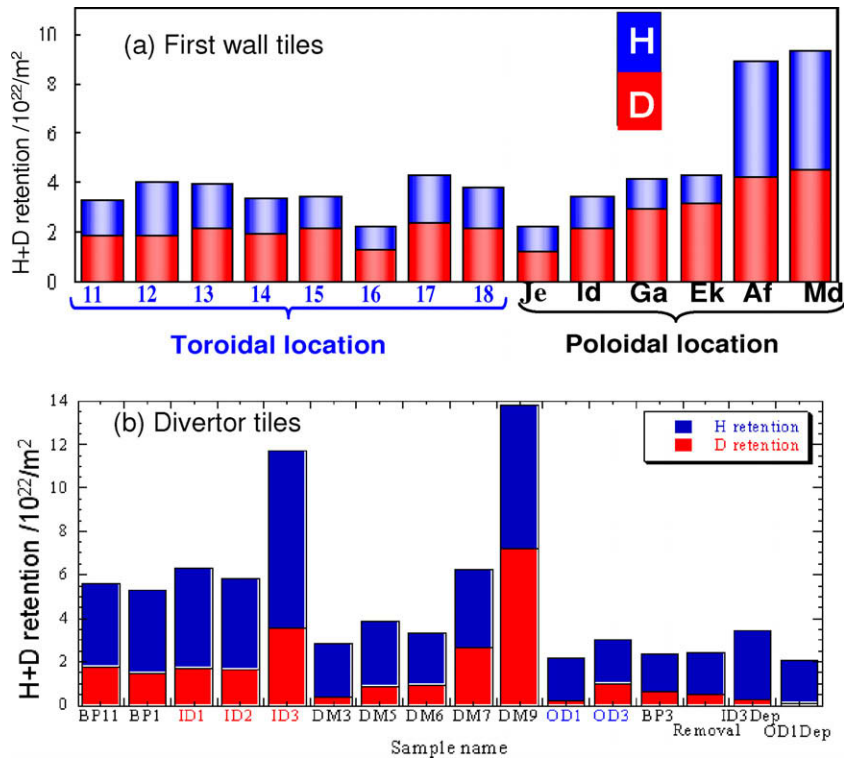


Fig. 4. The total desorbed amounts (H + D) with separation of H and D, (a) for all first tiles measured here 4 and (b) for the eroded and re-deposited divertor tiles.

also replaced by H. The gradient of lines in the figure corresponds to the ratio of the totally retained D and that of H (D/H) in the tiles within 1 mm depth from the surface. At the highest temperature (1100 K), D/H is only around 0.5, i.e. two thirds of D was replaced

by H. The data for the first wall tiles, of which temperature should be <800 K, locate at a little different area in the figure. Nevertheless, D/H is nearly one which is not so different from that for 800 K. This again confirms that the isotopic exchange by main

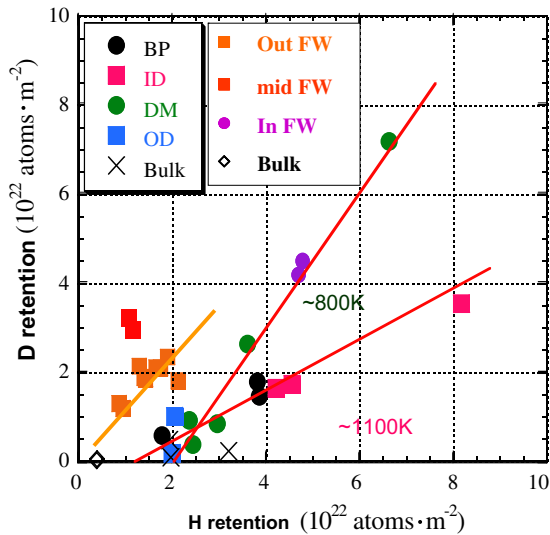


Fig. 5. Comparison of amounts of H and D retained within 1 mm thickness from the surface of all JT-60 plasma facing tiles measured.

discharges of DD is very effective to remove T retained in plasma facing surface layers during DT shots.

These values of D/H include large uncertainty because the effectiveness of the isotopic replacement depends on the depth, which also varies with temperature. As already mentioned that the NBI heating power of HH discharges was nearly a half for that of DD and hence the increment of the tile temperature during the HH discharges would be less. This could lead higher surface retention of H during the HH shots and made D/H lower. On the other hand, the injection of D in deep would increase D/H. Actually two data points clearly deviate upwards. They correspond to the Ga and Gk tiles located near the mid-plane on which the influx of high energy deuteron is high. Rather homogeneous depth distributions of H and D in the re-deposited layers on the divertor tiles (Fig. 3(a) and (b)) suggest that the isotopic replacement occurs throughout the layers mainly owing to high temperatures and the porous natures of carbon tiles. For the eroded area, its surface layers are most likely saturated with hydrogen during a plasma discharge, i.e. most of the incoming flux is recycled back. This means that D/H in the near surface should be nearly equilibrated with the incoming flux ratio of D/H. And the depth of the equilibrated zone increases with discharge numbers and increasing temperature. As shown in Fig. 2, the first wall tiles examined here were subjected to DD and HH discharges several times. Nevertheless, the depth profiles of H and D remained no traces of such discharge history.

All above observations indicates that isotope ratio in hydrogen retained near surface layers of the plasma facing wall is easily equilibrated with the flux ratio of hydrogen isotopes impinging

the surface. Therefore DD discharges after DT discharges would effectively remove T retained in the plasma facing surface layers during the DT discharges. The efficiency is higher and deeper for higher temperatures.

5. Conclusion

Carbon tiles used in JT-60U were exposed to DD (discharges with deuterium puffing + deuterium NBI) discharges. In addition, HH discharges were made before the vacuum ventilation to remove T produced by DD reactions. Hence all hydrogen isotopes retained in the carbon tiles showed different depth profiles, reflecting discharge history and isotopic exchanges. Based on our recent studies on retention of hydrogen isotopes (H, D, and T) in plasma facing carbon tiles used in JT-60U, we have discussed how isotopic replacement by plasma discharge is effective to remove tritium remained in the plasma facing surface in a DT reactor.

Hydrogen retention at the eroded area is very likely saturated and would not linearly increase with time. Furthermore, the isotopic ratios of retained hydrogen near surface layers are always equilibrated with incoming hydrogen fluxes (H/D/T). This is also true even for the re-deposited layers on the inner divertor. And the depth attaining this equilibrium is quite thick owing to their porous nature and temperature increase over 800 K. Hence tritium retention in plasma facing surfaces (both eroded and re-deposited) would be significantly reduced by isotopic replacement by DD discharges subsequently made after DT discharges.

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