§24. Investigation of Tritium Behavior and Traceability in In-vessel Systems of LHD during D-D Burning

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Tritium (T) produced by D-D reactions is a safety concern in deuterium discharges planned in LHD. Hence we have studied isotope retention of plasma facing surfaces in various tokamaks and are trying to estimate T retention and how to reduce and/or remove it in LHD during DD operation.

Results of carbon deposition and retention of hydrogen isotopes (H, D and T) in plasma facing carbon materials of JT-60U are summarized as follows. Behavior of tritium produced by DD reactions is separated from those of Hydrogen (H) and Deuterium (D) used as operating gases[1,2]. More than half of high energy triton produced is directly implanted a little deep into the tiles with the orbital and ripple loss mechanisms. Remaining triton is thermalized in plasmas and evacuated. Some impinging on plasma facing surfaces is easily replaced by succeeding injecting D and H and evacuated. Still some thermalized tritium is retained in redeposited carbon layers on the plasma shadowed area and in the tile gaps[3].

H and D as operating gas behave similarly and are incorporated (retained) in the redeposited carbon layers[4]. Carbon deposition profiles in the divertor region show strong inboard/outboard asymmetry as observed in most divertor tokamaks[5]. The inner divertor tiles were covered by thick redeposited layers, while the outer divertor tiles eroded, and no significant carbon deposition was observed on the plasma shadowed area except the bottom side of the outer dome wing. Carbon deposition rate in the divertor region is estimated to be around 9 x 10^{20} atoms/s normalized for NBI heating time during 1997-2002 experimental campaigns. Since around 40% of the net deposition on the divertor was attributed to the erosion of the first wall, the net erosion of the first wall for one year high power operation could be about 220kg/year, which is about 1/4 of that estimated for JET-MkIIA divertor.

H+D retention in the redeposited layers on the inner divertor tiles correlates well with their thickness and hydrogen concentration in the layers, (H+D)/C, was uniformly ~0.03[4] except a few top surface layers with ~0.05[6]. Surface D within a few μm in depth was mostly replaced by H owing to HH discharges made for tritium removal, indicating possible T removal by D-D discharges. Rather high D retention was observed on the outer dome wing probably owing to higher energy deuteron implantation originating from NBI. Little deposition and debris were observed in remote area even in NBI ducts and cryo-panels[7].

Such small hydrogen retention in JT-60U is attributed to 573K operation resulting in the temperature of the redeposited layers on the inner divertor more than 800K. In addition, wall saturation during a shot appeared and hydrogen retention can be divided into two components[8], (i) dynamic retention on the plasma facing surface which could saturate during a shot and (ii) static retention on the plasma shadowed area which would pile up continuously.

D/H ratio for hydrogen retention on the plasma facing surface of JT-60U was very small (0.01-0.19) indicating replacement of D retained during DD shots by H during HH shots subsequently made. This suggests that in LHD, a few DD discharges after large numbers of DT discharges could significantly reduce T retention on the plasma facing surface.

Hydrogen retention on tile gap i.e. tile side surfaces is well correlated to carbon deposition and made of two components[3]. One is incorporated in the carbon redeposited layers produced by prompt redeposition of carbon eroded at plasma facing surface, and the other is in the carbon layers produced penetrated neutrals in the gap. This is also clearly seen in JET[9] tiles. The former is larger for the eroded tiles than that for the redeposited ones and could be reduced by making the tile gap width less than the gyro-radius. The latter is dependent on the tile location and tile gap width, indicating strong influence of penetrating neutrals and plasmas.

H+D/C in the redeposited layers on the gap are a little higher than that on the plasma facing surface but well below 0.4, a saturated hydrogen concentration below 500K. Hence the total hydrogen retention in JT-60U is the smallest compared to other large tokamaks. Nevertheless, such carbon deposition will continue to pile up but significantly change depending on plasma conditions, geometry of tiles and divertor structure.

JT-60U seems a good example to reduce carbon deposition as well as tritium retention[10]. Well aligned tiles with more or less no steps between the neighboring tiles in toroidal and poloidal directions could reduce the net erosion and redeposition not only on the plasma facing surface but also at the plasma shadowed area. Higher surface temperature above 600K reduced the tritium retention significantly. The divertor geometry of JT-60U also helped to suppress carbon transport to shadowed area, particularly to the pumping duct. Still tritium retention at plasma shadowed region such as pumping slots and gaps of the tiles remains to be concerned because it must be piled up and could continuously increase.

Taking all these results in account, we can estimate T retention in LHD. Because operational temperature of LHD is rather low, hydrogen retention in redeposited carbon layers could be much higher than that of JT-60U. On the other hand, the first wall made of SS would reduce tritium retention compared to the full carbon wall of JT-60U. In addition, isotope exchange would significantly reduce T retention on the plasma facing surface.

References

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