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# Tritium distribution on plasma-facing tiles from ASDEX Upgrade

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**Abstract**

Tritium surface distribution on the plasma-facing tiles used in the divertor region and the central column of ASDEX Upgrade was measured by Tritium Imaging Plate Technique. The tritium intensity is high at the dome baffle underneath the *X* point, the outer divertor region and the mid-plane or lower side of the central column. In contrast, little tritium was retained in the carbon deposition found in the inner strike point module tiles. Such profile is very similar to that observed in JT-60U, indicating that long-term tritium retention in ASDEX Upgrade was dominated by the implantation of energetic tritons escaping from the main plasma without losing their initial energy (1 MeV) rather than the codeposition of eroded carbon and hydrogen isotopes. However, such tritium profile was modified by carbon deposition and thermal load, particularly on the divertor target tiles and the central column.


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## 1. Introduction

Long-term tritium retention in the plasma-facing materials (PFM) is a critical issue for safety administration, first wall material selection and operation program in future fusion reactors. In D–T discharge machines like JET and TFTR [1–4], tritium mainly co-deposited with eroded first wall material (carbon). In D–D discharge machines, on the other hand, according to our recent studies, tritium distribution on the plasma-facing wall well corresponded to the distribution of high energetic triton implantation [5–7].

In JET, the retention properties of tritium and other hydrogen isotopes have been extensively studied for the Mark IIA divertor configurations. The results suggest that main concern in ITER-FEAT is the codeposition with tritium and eroded carbon material. Recent studies of the Mark-IIGB divertor have confirmed that the codeposition occurs mainly in the inner divertor region as observed for the Mark IIA [8], however, shown a somewhat different deposition property, such as reduction of the carbon deposition on the louver area [9]. Thus, more examination/investigation is required with different machines and divertor configurations in order to clarify the process and mechanism of tritium and hydrogen isotope retention.

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ASDEX Upgrade has installed a closed divertor configuration; DIV-II from 1996 to 2001 operational phase, and has modified it for a more optimized configuration; DIV-IIb in 2001 [10], with a view to ITER-FEAT design. This motivates us to analyze the tritium retention in plasma facing tiles in ASDEX Upgrade and to compare the results with other devices. In ASDEX Upgrade, tritium is produced by the D–D nuclear reaction ($^2D + ^2D = ^3T\ (1\,MeV) + ^1H$). However the tritium activity level of the first wall was quite low and little tritium analyses were reported before. In this study, we have applied Tritium Imaging Plate Technique (TIPT), which have been successfully used to determine the tritium distribution on the plasma-facing components.

## 2. Experimental

Imaging Plate (IP) is a 2-D radiation detector with high sensitivity and resolution, utilizing a photostimulable phosphor ($BaFBu:Eu^{2+}$). The phosphor, which detects radiation, is applied on the surface of IP. Radiation intensity mapping is memorized as the number and position of the quasi-stable F-center formation. (F-center is created by excited electron trapped in halogen ion vacancy.) The IP surface is contacted to the sample surface and exposed to tritium beta electrons, which generates a lot of excited electrons in the phosphor crystal leading to F-center formation. After that, IP was processed by an IP reader in order to obtain the digital image. Tritium activity is expressed as the Photo-Stimulate Luminescence (PSL) intensity, which is proportional to the absorbed radiation energy.

Sample tiles were the graphite tiles removed from the heat shield of the central column, and a complete set of DIV-IIb graphite tiles installed in the 2001/2002 operational phase.

The IP exposure was carried out at Max-Planck-Institute in Garching. Since tritium activity on the tile surface was lower than the European regulation, sample tiles were directly contacted to IP surface. Some of the tiles with curved surface requested some weight in order to ensure the contact.

After 50 h contact or exposure of the IP to beta rays from tritium in a dark cabinet, IPs were transported to Forschungszentrum Karlsruhe (FZK) and processed by an IP reader. The IP (BAS-TR2025) and the IP reader (FLA-3000 G) used were manufactured by Fuji Co. Ltd.

## 3. Result and discussion

### 3.1. Divertor region

Fig. 1(a) shows tritium poloidal distribution of a complete set of DIV-IIb tiles together with the tritium line intensity profiles. The tritium distribution was quite different depending on the poloidal position, while rather uniform in the toroidal direction. The highest PSL intensity (nearly proportional to the tritium activity) was observed at the top of the dome baffle installed underneath the $X$ point and the outer divertor tiles except for the strike point module. In contrast, the intensity was rather low at the inner and outer strike point area.

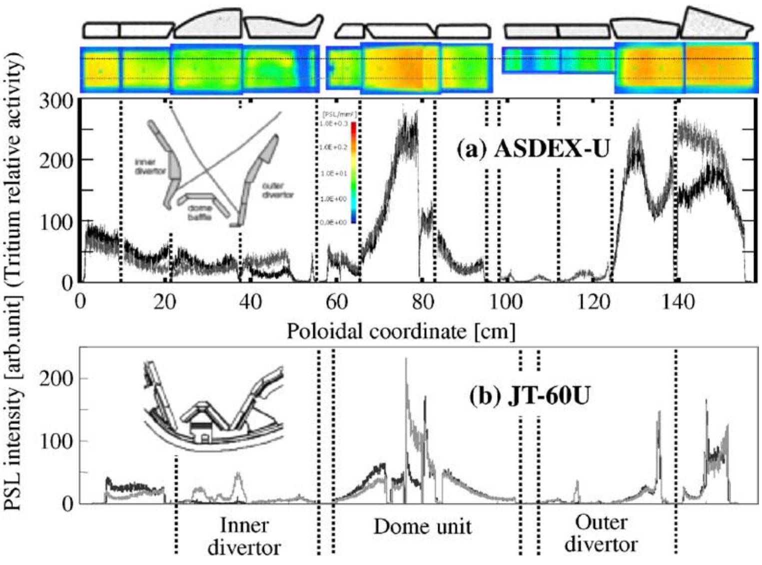


Fig. 1. (a) Tritium distribution on the AUG DIV-IIb divertor region and (b) JT-60U divertor region determined by TIPT. The highest intensity was observed at the dome baffle tile installed in the private flux region and the outer divertor/ baffle plate region.

Such a poloidal profile was quite similar to that observed in JT-60U shown in Fig. 1(b) for comparison. In JT-60U, the tritium distribution on plasma-facing tiles well corresponded to the implantation distribution of the high energetic tritons escaping from the main plasma without losing their initial energy (1 MeV). The main mechanism of the energetic triton escaping is the so-called toroidal magnetic field (TF) ripple transport including the ripple trapped loss and the banana orbit loss. The results of OFMC (Orbit Following Monte-Carlo) code [11] simulation applied to JT-60U high performance (H-mode or negative-magnetic shear) discharges showed that the particle flux of such ripple-loss ions to plasma-facing walls was high at the dome top underneath the $X$ point and the outer baffle plate in the divertor region [6]. Accordingly, the tritium distribution seen on DIV-IIb tiles is predominantly determined by implantation of the energetic tritons.

Nevertheless, carbon deposition modified the apparent tritium distribution. In case of the JET Mark IIA divertor employed in the D–T campaigns, most of tritium was codeposited with carbon. In contrast, the PSL intensity on the divertor tiles of ASDEX Upgrade was rather low at the redeposited area, which is clearly seen in Fig. 2(a) where detailed PSL intensity profiles of the inner target module were shown. For the inner divertor tiles, dark areas of photographs of the inner target tiles were covered by thick codeposition, while the corresponding PSL intensity was clearly less than other areas. However the low PSL intensity does not mean low tritium retention in the deposited area but tritium could be retained underneath the deposition as already reported in JT-60U divertor owing to implantation of tritons with high energy.

For the outer divertor target module (Fig. 2(b)), the relation between tritium images and deposition is not clear. It is important to note that the PSL intensity at the outer strike point area (at 40–80 mm in poloidal distance) was very small. Since the area was erosion dominated and exposed to high heat load and the temperature could be up to more than 600 K, it is very likely that tritium and other hydrogen isotopes once retained on this area were thermally desorbed. Accordingly low PSL intensity at the outer divertor region indicates low tritium retention.

Actually, the deuterium distribution in the divertor region obtained by NRA analysis showed that the deuterium inventory tended to be lower at the outer divertor region than the inner region. However, the highest deuterium concentration was observed at the periphery of the inner and outer pumping slot with $\sim 10^{18}$ D/cm$^2$ in both DIV-II and DIV-IIb [12], which is clearly different from the tritium distribution and strongly correlated with carbon deposition.

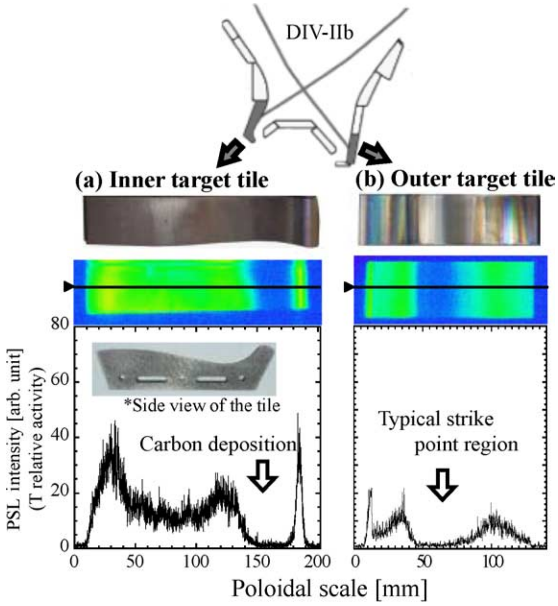


Fig. 2. IP images and photographs of (a) the inner and (b) the outer divertor strike point module tiles. Tritium retention in carbon redeposits near the strike point region is very low in (a) and in (b) there is an inhomogeneous distribution with low intensity. The abrupt reduction as seen from 40 mm to 80 mm (typical strike point region) in the poloidal coordinate could reflect the history of the high heat load.

### *3.2. The heat shield tiles of the central column*

Fig. 3 shows the tritium images of the plasma-facing tiles installed on the central column heat shield during the 2001/2002 operational phase. The four horizontal lines on the tile surface were the W layer with several 100 nm thickness used as erosion/deposition marker. Among the three tiles, tritium inventory on the tiles except the marker area was highest at the mid-plane tile (A8), while the top one (A1) was the lowest. This observation is also explained by the triton escaping mechanism, i.e. the energetic ions trapped in the local magnetic well (ripple well) formed in the mid-plane have high probability to drift away from the main plasma and directly impinge to the plasma facing materials. Additionally, $\mathbf{B} \times \nabla B$ drift transport tend to shift the ions to the bottom side. Although the TF ripple is lower at the high field side than at the low-field side, higher energy triton can escape into the high field side depending on the birth point and the ripple well shape. In the case of JT-60U, the highest tritium inventory on the inner wall was observed around mid-plane, agreeing with the numerical expectation [7]. The distribution observed on the central column of ASDEX Upgrade is thus attributed to the implantation of the energetic tritons with the energy of more than $\sim$100 keV by the ripple loss and/or orbital loss as the case of JT-60U.

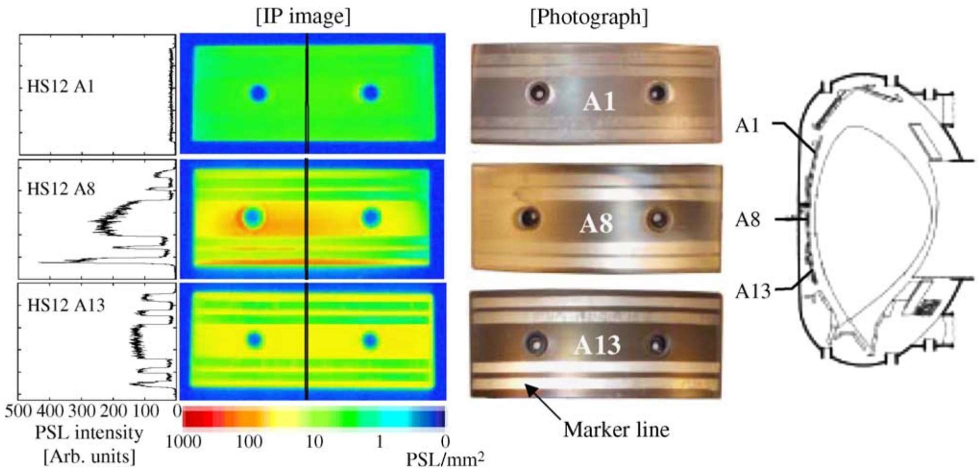


Fig. 3. IP images and photographs of the sample tiles used as central column heat shield. All tiles were graphite tiles, and tile A1, 8 and 13 were from the upper edge of the central column, the mid-plane and the lower edge of central column, respectively. The four stripes on each tile correspond to the marker metal area with 100 nm thick coatings.

One can note that the significant difference between the marker area and the remaining one for tiles A8 and A13. The tritium intensities on the marker lines were about 1/10–1/5 of the other remaining area, while it is hard to distinguish the marker lines in the IP image of A1 tile. In the marker area, the same amount of energetic tritons must be injected as the other graphite area. However, since hydrogen mobility in W is orders of magnitude higher than that in graphite, tritium once retained in the marker could be easily released when the A8/13 tiles received the heat load, such as limiter configuration phase of plasma ramp-up/down. Additionally, shield effect for tritium beta electron is higher in high-$Z$ metal than in graphite, i.e. beta electrons emitted from tritium penetrated into substrate graphite underneath the marker could be attenuated. Both would result in the significantly smaller PSL intensity on the marker area than other area. On the other hand, A1 tile, which was installed in the top edge of central column where little energetic tritons were implanted, could be exposed lower plasma flux and according less temperature increases. As noted in the easier hydrogen blistering on carbon covered tungsten [13], carbon deposition on A1 tile might inhibit hydrogen release form the tungsten, accordingly the PSL intensity could be same for the W marker area and other area on A1 tile.

Above discussion is still speculative. Nevertheless we can say that tritium profiling together with carbon deposition profiling could be a good diagnostic for erosion/deposition and hydrogen retention mechanism.

### 3.3. Quantitative estimation of tritium amount retained in ASDEX Upgrade

Although the TF ripple in ASDEX Upgrade might be smaller than that in JT-60U, the tritium profiles indicate the presence of ripple loss tritons, and the tritium distribution was predominantly determined by such triton implantation rather than codeposition of thermalized tritium with the eroded carbon. The fraction of the escaping triton is still unclear due to lack of the quantitative data. A rough quantitative estimation of long-term tritium retention in ASDEX Upgrade is possible by comparing with previous IP measurements of JT-60U. The PSL intensity is relative and changes depending on the experimental conditions such as exposure time. Taking such difference in conditions of the IP measurements between ASDEX Upgrade and JT-60U into account, several $kBq/cm^2$ of tritium retention is predicted in high PSL intensity area such as the dome baffle or outer divertor region. This value is about 1/10 of tritium retained in the JT-60U dome top tile ($\sim$70 $kBq/cm^2$). Tritium produced in ASDEX Upgrade during the present samples being installed was approximately $1.5 \times 10^{18}$ ($\sim$3.2 GBq). This is about 1/6 of that for JT-60U and well corresponds to the 1/10 of the tile inventory. We are planning to make more detailed analysis, and the results will give important information not only of the tritium behaviour in ASDEX Upgrade but also confirmation of the energetic ions in the fusion device with a closed divertor configuration.

## 4. Conclusion

A detailed 2-D tritium distribution of the plasma-facing wall of ASDEX Upgrade was measured by TIPT. The tritium intensity is high at the dome baffle tile underneath the $X$ point, at the outer divertor region and in the mid-plane and the lower part of the central column. In contrast, little tritium was retained in the carbon deposits found at the inner strike point module tiles. All these observations were similar to those observed in JT-60U, and are likely characteristic of tritium retention in plasma facing tiles in D–D discharge machines with closed divertor configuration, but different from D–T discharge machines like JET. Moreover, the local retention property, as observed on the strike point module or central column tiles, showed some information of the plasma-surface interaction such as temperature increase of the PFM, impinging of the ion energy and flux and carbon erosion/deposition. Therefore, the tritium profiling together with the carbon deposition profiling could be a valuable diagnostic for erosion/deposition and hydrogen retention mechanism.

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