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Tritium distribution measurement of the tile gap of JT-60U

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Abstract

Tritium retention in the gaps or side surfaces of JT-60U divertor tiles was measured by an imaging plate (IP) technique and a combustion method. On the toroidal side surfaces, the retained amount decreased exponentially from the front (plasma facing) end to the rear end with an e-folding length of 3 mm for all tiles. On the other hand, the tritium retention characteristics on the poloidal sides varied depending on the location of the gap. Retention of tritium in redeposited carbon layers was observed in the gaps between inner divertor target tiles and the bottom side of dome outer wing tiles that faced the outer pumping slot. From the extrapolation of these IP and combustion results to the full toroidal direction, the total tritium retention in the divertor gaps was estimated to be around 0.07% of the total tritium produced by D–D reactions or 0.13% of thermalized tritium.

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1. Introduction

Tritium accumulation in gaps between the plasma facing tiles is a critical issue for control of in-vessel tritium retention. It is quite difficult to remove such tritium by a normal cleaning operation such as Taylor or glow discharges, and continuous accumulation can cause non-negligible retention. In the past investigation on TFTR, for example, it was estimated that 9% of injected fuel was retained in the gaps [1], and the accumulation of a huge amount of tritium was observed in the gaps between the bumper limiter tiles in layers codeposited with carbon [2]. Characterization of tritium retention in the gaps is required for mitigation of such tritium accumulation or development of the optimal removal technique.

JT-60U operates mainly with a deuterium plasma (and hydrogen plasma for wall cleaning), hence, the origin of tritium is production by the D–D reaction. The initial energy of the generated tritium ion (triton) is about 1 MeV, which is much higher than the main plasma temperature. It is well known that some fraction of such high energy ions is lost from the plasma due to a magnetic field ripple loss mechanism [3]. In JT-60U, according to recent studies, approximately 50% of generated tritium ions escape from the plasma without losing their initial energy, and they are deeply implanted in the plasma facing surface and remain there [4,5]. On the other hand, the remaining 50% slows down in the plasma, and

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then behaves as plasma fuel, some of which could be retained in the tile gaps.

In this study, tritium retention in the tile gaps of the JT-60U W-shaped divertor was analyzed by the tritium imaging plate technique (TIPT) [2,4] and a combustion method.

2. Experimental

Fig. 1(a) and (b) shows cross-sectional views of the vacuum vessel and W-shaped divertor of JT-60U, respectively. The tiles measured in this study were used in the divertor from June 1997 to March 2003. The total number of shots during this period was about 14 000 and total duration of neutron beam injection (NBI) heating was approximately 30 000 s. The number of neutrons produced by D–D reactions during this period was $\sim 5.7 \times 10^{19}$. Since the reactions: $D + D \rightarrow {}^3He + n$ and $D + D \rightarrow {}^3H + p$ proceed at almost the same rate, tritium production was also $\sim 5.7 \times 10^{19}$ ($\sim$102 GBq).

The divertor tiles were made of carbon fibre composite (CFC (CX-2002U)) except for the dome inner-wing tiles of isotropic graphite (IG-430U). The tiles were aligned with a gap opening of 1 mm for both poloidal and toroidal gaps. It could be somewhat narrower during a discharge because of thermal expansion of the tiles.

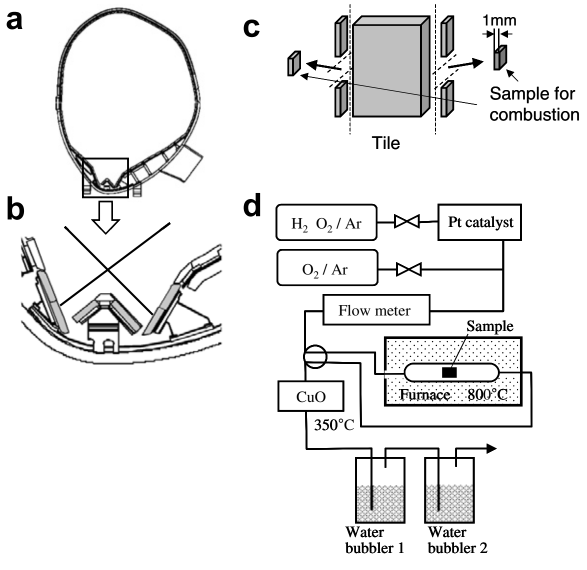


Fig. 1. Cross-sectional view of JT-60U (a) vacuum vessel and (b) W-shaped divertor. The gray colored tiles are samples for this study. (c) Sample preparation for combustion analysis. (d) Schematic diagram of experimental apparatus of combustion method.

Tritium areal distribution on all tile sides was measured by the imaging plate (IP) technique. IP is a 2-D radiation detector using photo-stimulated luminescence (PSL), which gives radiation distribution with high resolution. Detailed information about the IP technique is available elsewhere [6,7]. IP was brought into contact with sample tiles to be exposed to beta particles from tritium decay. After about 60 h of exposure, the IP was processed by an IP reader: BAS-2500. IP used here was BAS TR 2025/2040 specialized for tritium detection. Both IP and the IP reader are manufactured by Fuji Photo Film Co. Ltd.

After the IP measurement, the tiles were cut into small pieces for quantitative tritium analysis by combustion. Fig. 1(c) and (d) schematically show the sample preparation and experimental apparatus, respectively. A sample was heated from room temperature to 1073 K by an electric furnace. O_2 gas diluted with Ar (20% of O_2) was fed into a furnace tube during the heating. Desorbed tritium was transported to a water bubbler and trapped there. After the operation, tritium activity accumulated in the bubbler was measured by a liquid scintillation counter.

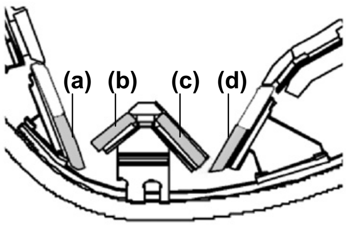
Appreciable carbon deposition was observed on the surface by scanning electron microscope (SEM) observation.

3. Results and discussion

3.1. Toroidal sides

In Fig. 2, photographs of the toroidal side surfaces of divertor tiles (DV-b, DV-c, DM-a and DM-c tiles indicated as (A) to (C) in the inset) are compared with tritium distribution images obtained by the IP technique. Tritium intensity profiles along lines in the IP image are also shown. The tritium profiles are given by ‘PSL intensity’, which is a special unit used in the IP technique, and proportional to absorbed energy in IP. As seen in these profiles, tritium was localized at the (plasma facing) front end. The PSL intensity decreased exponentially with distance from the front end to rear end. The e-folding lengths of the exponential decay were $\sim$3 mm for all tile sides. In the rear half, the PSL intensity became nearly constant, but still higher than the background level, indicating the contribution of gaseous tritium.

In Fig. 3, the poloidal distributions of the PSL intensity in the vicinity of the front end for both



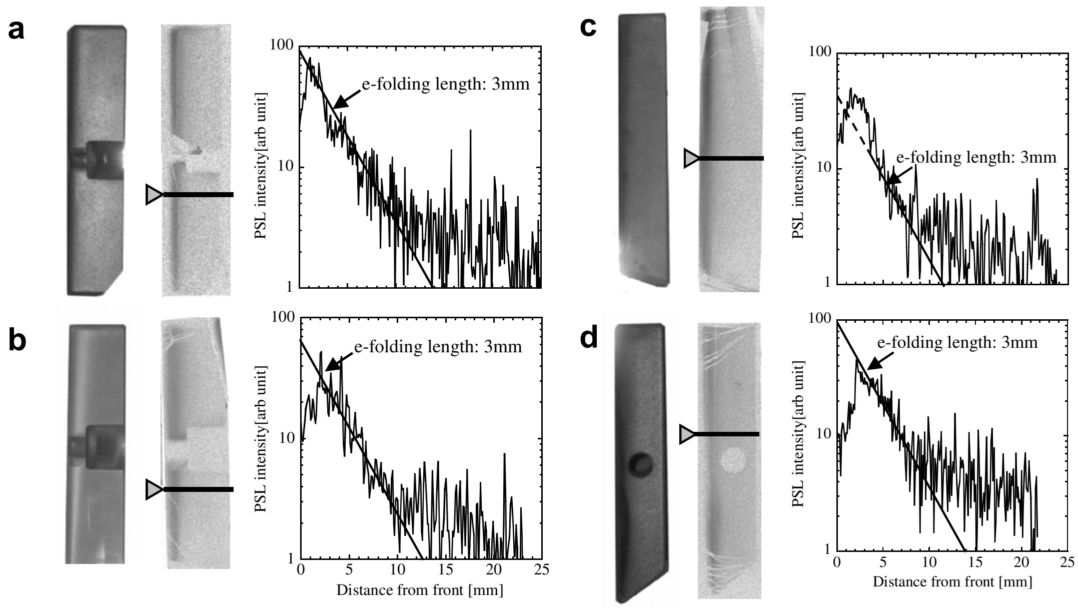


Fig. 2. The line profiles obtained from toroidal sides of (a) DV-b, (b) DV-c, (c) DM-a and (d) DM-c tiles. The PSL intensity diminished exponentially with the distance from the plasma-facing side. The e-folding lengths from each sample are almost the same: ~3 mm.

toroidal sides are shown together with those for the plasma facing surface previously measured [4]. One can observe that both profiles are similar, i.e. PSL intensity was high in the outer dome wing and outer divertor tiles. It has been shown that the tritium distribution on the JT-60U plasma facing surface corresponds well with the implantation distribution of the high energetic triton produced by the D–D reaction. Therefore, the high tritium retention near the top end of the toroidal sides likely reflects this implantation.

After IP measurements, quantitative tritium analysis was made for the samples taken from toroidal sides of DV-b (inner target tile), DV-c (outer target tile) by the combustion method. The tritium activities determined by the combustion method are compared with the PSL intensities obtained from the same samples in Fig. 4. As seen in the figure, the PSL intensities have a good linear correlation with the tritium activity. Accordingly, the amount of tritium on each side surface could be estimated from the PSL intensity. Table 1 gives the tritium retention on entire toroidal sides, which is estimated by converting the integrated PSL intensity obtained from the IP image into tritium activity. Integrating over the entire divertor, the tritium inventory in toroidal gaps is estimated to be about 39 MBq, which is about 0.04% of the total tritium produced by D–D reactions of 102 GBq.

3.2. Poloidal sides

Tritium line profiles from the front to the rear end of poloidal sides for various divertor tiles obtained by the IP measurement are summarized in Fig. 5. Different from the toroidal sides, tritium distribution on the poloidal sides varied depending on the position of the tile. In particular, the tile sides facing the pumping slot ((b), (c), (f) and (g)) retained tritium homogenously without showing

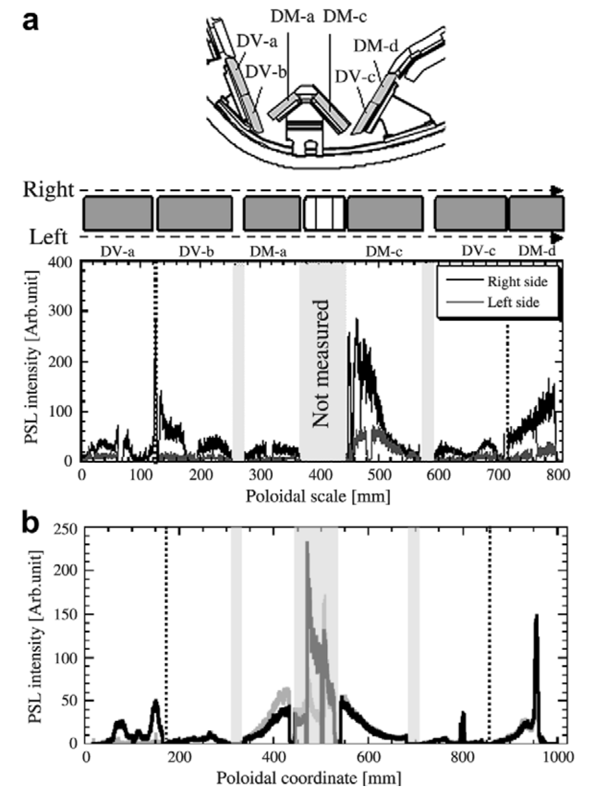


Fig. 3. (a) The poloidal distribution of the both toroidal sides obtained from vicinity of the plasma facing side. The profile there is similar to (b) that of the plasma facing surface at outer divertor area.

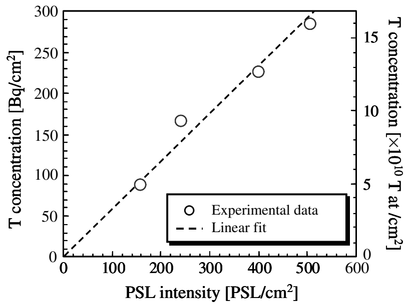


Fig. 4. The comparison between tritium activity determined by combustion method and PSL intensities obtained from combustion samples.

an exponential decay, while the gap facing sides ((a), (d), (e) and (h)) displayed the exponential decay but with different e-folding lengths from that on the toroidal sides. In JET, heavy deposition was found on the bottom side of the inner vertical target tile shadowed from the plasma and facing to the inner pumping slot, and a huge amount of tritium was

Table 1
Tritium retention estimated from IP and combustion measurement (activity for torus [MBq])

DV-a sides	6.88
DV-b sides	4.70
DM-a sides	3.51
DM-c sides	10.9
DV-c sides	2.21
DV-d sides	10.6
T in Toroidal sides	38.7
(a) Gap between DV-a and -b	12.6
(b) DV-b bottom side	0.58
(c) DM-a bottom side	0.26
(d) DM-a upside	1.63
(e) DM-c upside	1.85
(f) DM-c bottom side	5.83
(g) DV-c bottom side	0.06
(h) Gap between DV-c and -d	4.71
T in Poloidal gaps	27.5
Total	66.2

incorporated in such deposited layers [8–10]. However, the PSL intensities on the surfaces facing the inner pumping slot (corresponding to (b) and (c)) were very low in JT-60U. SEM observation showed that there was little deposition there (Fig. 6(a)-(2)). This appreciable difference between JET and JT-60U was probably caused by the position of the strike point and/or pumping slot. In JT-60U, the inner strike point is always put on the 'vertical' target tiles, and the inner pumping slot is located in the private flux region. In contrast, the inner strike point of JET was frequently put on the 'horizontal' target tile, and the eroded hydrocarbon impurity was easily transported to the entrance of the pumping slot [11]. Thus, these results indicate that the divertor configuration is critical for the formation of redeposition in this region.

Substantial tritium retention was found on the bottom side of the outer dome wing tile (Fig. 5(f)), facing the outer pumping slot, where PSL intensity was high throughout the surface. From the SEM observation, redeposited layers with a maximum thickness of about 90 μm were found on the tiles surface as shown in Fig. 6(a)-(6). Since a high energetic triton could not be implanted in this surface, this high tritium retention must be caused by incorporation of thermalized tritium in the redeposited carbon layers.

Fig. 6(b) shows the result of combustion measurement of some samples taken from the poloidal sides.

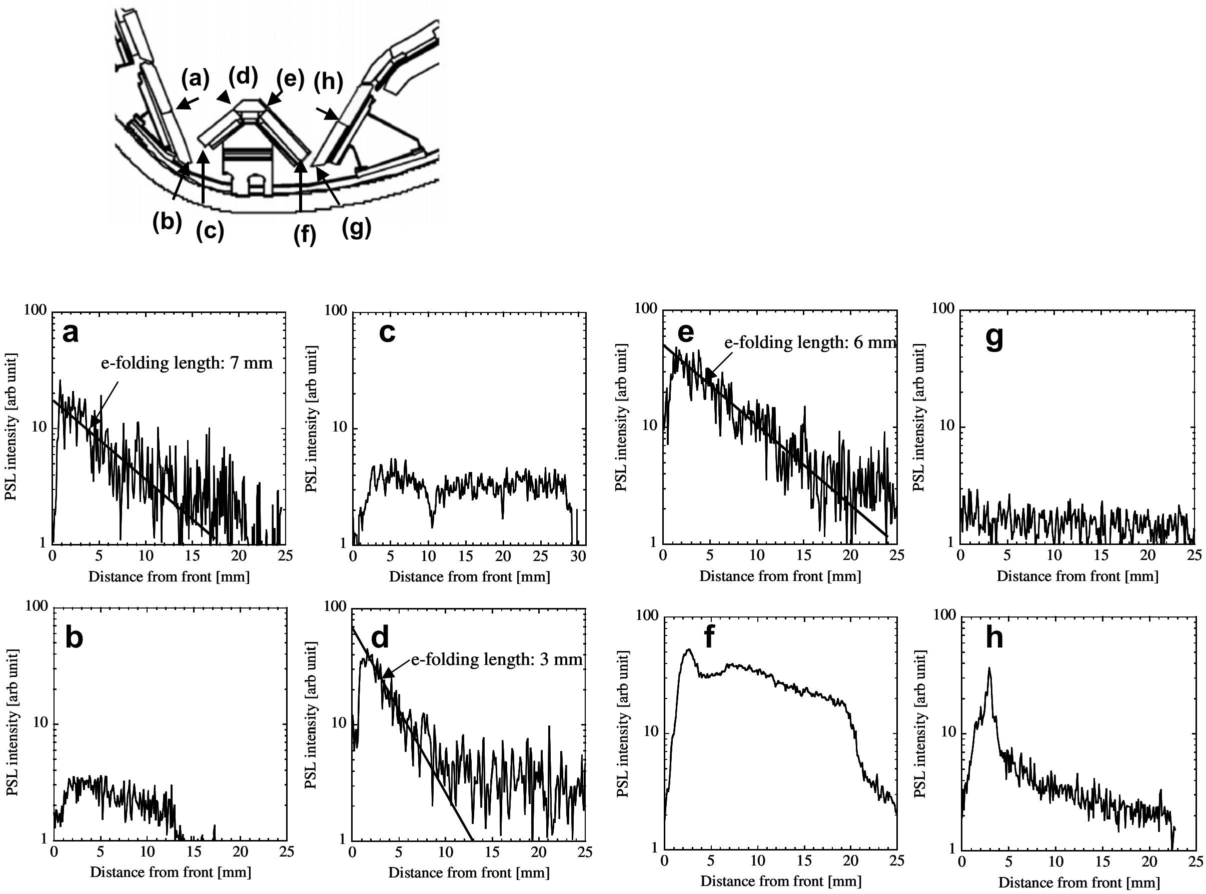


Fig. 5. The profiles of poloidal sides obtained by (a) DV-b upside, (b) DV-b bottom side, (c) DM-a bottom side, (d) DM-a upside, (e) DM-c upside, (f) DM-c bottom side, (g) DV-c bottom side and (h) DV-c upside. Tritium distribution on the poloidal sides varied according to the position.

As found in the IP results, tritium retention on the side surfaces facing the inner pumping slot was low, while that on the bottom side of the outer dome wing tile was higher. Tritium concentration in the gap between the inner divertor tiles was also high. The inner divertor was always deposition dominant area. Even in the gap, thick redeposited layers with a maximum thickness of about 80 μm were found at the vicinity of the front end side (Fig. 6(a)-(1)), and tritium was buried in this redeposited layers.

Tritium inventory in the poloidal gaps was estimated and is also given in Table 1. The total inventory in the poloidal gaps for the entire torus is about 28 MBq. Adding the toroidal contribution, the total tritium inventory in the gaps of the W-shaped divertor tiles is about 66 MBq, which is only about 0.07% of the total generated tritium of 102 GBq. In JT-60U, nearly half of the total generated tritium was directly implanted into the plasma facing wall as a high energetic particle, and the remaining half was thermalized to plasma temperature. Thus, this result indicates only 0.13% of the thermalized tritium would be retained in the gaps of the JT-60U W-shaped divertor.

4. Summary

Tritium retention on the sides of JT-60U divertor tiles was measured by the IP technique and the combustion method. On the toroidal sides, most tritium was accumulated near the front (plasma facing) end showing an exponential decay of retention towards to the rear end. The decay shows an e-folding length of about 3 mm for all tiles. Tritium profiles in the vicinity of the front end side for all tiles along the poloidal direction is very similar to those for the plasma facing surface, indicating the contribution of high energetic triton implantation.

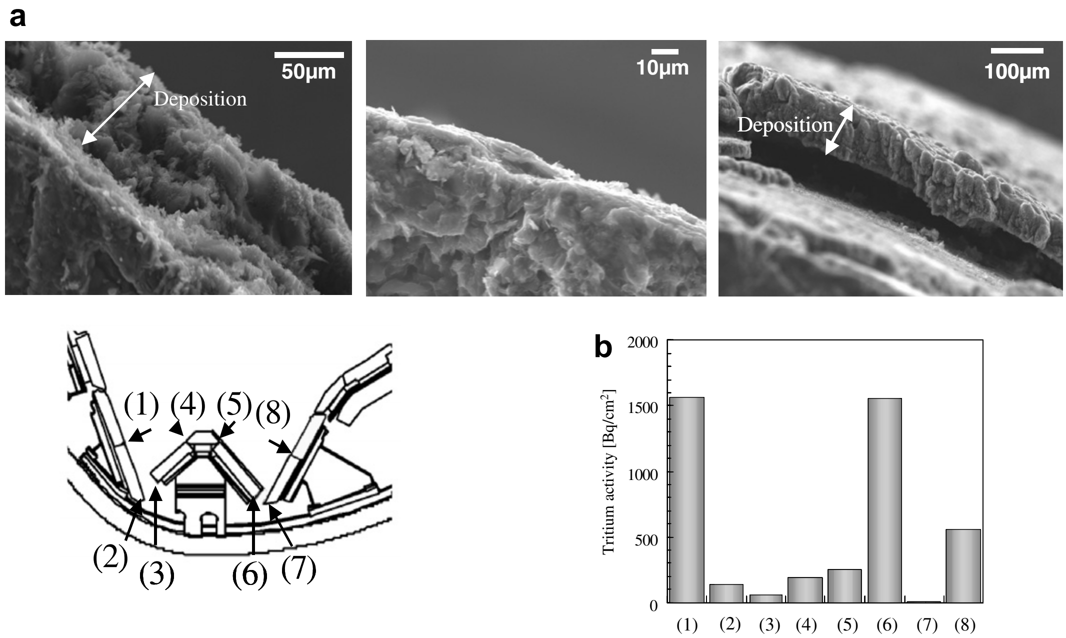


Fig. 6. (a) SEM images obtained from (1) DV-a bottom side, (2) DV-b bottom side and (6) DM-c bottom side, and (b) tritium activity results from combustion method for the poloidal sides.

Two retention characteristics on the poloidal sides were distinguished, (1) a side facing a neighboring tile side i.e. gap between the tiles shows a similar profile to the toroidal sides, having high tritium retention at the front end with exponential decay towards the rear end with somewhat different e-folding lengths (3–7 mm) among the poloidal positions, (2) all sides facing the pumping duct show rather flat retention profiles, indicating the incorporation of thermalized tritium in the redeposited carbon layers. In particular, the bottom side of the dome outer wing tile was covered by the redeposited layers of 80–90 μm in thickness, suggesting enhanced carbon deposition with an inward drift mechanism. Taking tritium activities determined by the combustion method into account, the tritium profiles obtained by IP were quantified to give tritium inventory. The total tritium retention in the divertor gaps, thus obtained is 66 MBq, which is only 0.07% of the total generated tritium, and 0.13% of thermalized tritium.

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References

[1] C.H. Skinner, E. Amarescu, G. Ascione, et al., J. Nucl. Mater. 241–243 (1997) 214.
[2] T. Tanabe, K. Sugiyama, C.H. Skinner, et al., Fus. Sci. Technol. 48 (1) (2005) 577.
[3] K. Tobita, S. Nishio, S. Konishi, et al., Fusion Eng. Des. 65 (2003) 561.
[4] K. Sugiyama, T. Tanabe, K. Masaki, et al., Physica Scripta T103 (2003) 56.
[5] K. Masaki, K. Sugiyama, T. Tanabe, et al., J. Nucl. Mater. 313–316 (2003) 514.
[6] Y. Amemiya, J. Miyahara, Nature 336 (1988) 89.
[7] H. von Seggern, T. Voigt, V. Knüpfer, et al., J. Appl. Phys. 64 (1988) 1405.
[8] R.-D. Penzhorn, J.P. Coad, N. Bekris, et al., Fusion Eng. Des. 56&57 (2001) 105.
[9] J.P. Coad, N. Bekris, J.D. Elder, et al., J. Nucl. Mater. 290–293 (2001) 224.
[10] K. Sugiyama, T. Tanabe, N. Bekris, et al., Fusion Sci. Technol. 48 (2005) 573.
[11] H.G. Esser, V. Philipps, M. Freisinger, et al., J. Nucl. Mater. 337–339 (2005) 84.