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Fuel inventory and deposition in castellated structures in JET-ILW

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Abstract

Since 2011 the JET tokamak has been operated with a metal ITER-like wall (JET-ILW) including castellated beryllium limiters and lamellae-type bulk tungsten tiles in the divertor. This has allowed for a large scale test of castellated plasma-facing components (PFC). Procedures for sectioning the limiters into single blocks of castellation have been developed. This facilitated morphology studies of morphology of surfaces inside the grooves for limiters after experimental campaigns 2011–2012 and 2013–2014. The deposition in the 0.4–0.5 mm wide grooves of the castellation is 'shallow'. It reaches 1–2 mm into the 12 mm deep gap. Deuterium concentrations are small (mostly below 1×10^{18} cm^{-2}). The estimated total amount of deuterium in all the castellated limiters does not exceed the inventory of the plasma-facing surfaces (PFS) of the limiters. There are only traces of Ni, Cr and Fe deposited in the castellation gaps. The same applies to the carbon content. Also low deposition of D, Be and C has been measured on the sides of the bulk tungsten lamellae pieces. Modelling clearly reflects: (a) a sharp decrease in the measured deposition profiles and (b) an increase in deposition with the gap width. Both experimental and modelling data give a strong indication and information to ITER that narrow gaps in the castellated PFC are essential. X-ray diffraction on PFS has clearly shown two distinct composition patterns: Be with an admixture of Be–W intermetallic compounds (e.g. $Be_{22}W$) in the deposition zone, whilst only pure Be has been detected in the erosion zone. The lack of compound formation in the erosion zone indicates that no distinct changes in the thermo-mechanical properties of the Be PFC might be expected.

Keywords: JET, ITER-like wall, beryllium limiters, castellation, deposition, fuel inventory

(Some figures may appear in colour only in the online journal)

1. Introduction

All plasma-facing components (PFC) in ITER will be castellated because this structure is considered to be the best solution to ensure thermo-mechanical durability and integrity of materials under high heat flux loads [1]. However, in the environment containing also low-Z elements (e.g. beryllium and some impurities of carbon), material eroded by various processes is transported and co-deposited together with fuel species in areas shadowed from the direct plasma line-of-sight. As a consequence, re-deposition and material mixing may occur in grooves and on the side surfaces of the PFC, e.g. on surfaces located in the gaps separating the tiles. The issue may be quite

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serious if significant amounts of fuel were accumulated in the gaps. The composition of deposits and their pattern in the gaps is crucial for the overall assessment of fuel inventory as the area of surfaces in the castellation in ITER will be approximately three times larger than the area of plasma-facing surfaces (PFS). Based on the earlier experience from JET, co-deposits formed in remote areas (e.g. inner divertor corner and water-cooled louvers) [2–5] would be very difficult to remove by any of the cleaning methods known today [6]. This, in turn, may lead to significant long-term fuel retention. In addition, very limited access to in-vessel components of ITER calls for detailed studies of deposition in the gaps and grooves of the PFC from present-day devices. All aspects related to the fuel content and the surface structure in gaps must be taken into account.

Deposition in the gaps was noticed already in the 1980s [7–10] and some quantitative results for TFTR and DIII-D have been published [10]. In TFTR 15% of the retained fuel was found in gaps, 44% on the PFS and 41% on the outer wall, whereas in DIII-D accumulation in gaps between the tiles reached 40% of the retained fuel [9], but no information has been given about the gap geometry, especially the gap width. Large-scale tests of castellated structures have been performed in the past at JET when operating in 1989–1992 with beryllium belt limiters [11] and in 1994–1995 with Mk-I divertors [12–14]. The latter were composed of arrays of water-cooled small tiles made of carbon-fibre composites (Mk-I-CFC) and, at a later stage, were replaced with castellated beryllium (Mk-I-Be). The tiles were 35 × 72 mm in the PFS and 55 mm high. The gaps between the tiles were 6 and 10 mm wide, while the width of the castellation grooves in the Be tiles were 0.5 mm wide. In Mk-I-CFC the retention on surfaces between the tiles exceeded at least by a factor of two [13, 14] the amount of fuel on the PFS [12]. In the case of Mk-I-Be, the fuel content in the gaps separating the tiles was 30 times bigger than inside the castellation, where the deposition was shallow and had a steep gradient, i.e. the decay length of 1.5 mm. For technical reasons the examination inside the castellation was limited to only one groove. Broader studies were performed for several grooves of castellated Be limiters. In all cases, in both the divertor and limiter tiles the deposition was dominated by carbon and the profiles were steep. Dedicated studies with various castellated probes (test limiter or divertor tiles) have been performed in several tokamaks [15–21]; some of them have been summarized in [22]. This has been followed by modelling which included both the development of advanced codes, e.g. 3D-GAPS [23–26] and particle-in-cell calculations [27, 28]. The efforts were focused on the reproduction of the results obtained in experiments comprising a limited number of discharges performed under a defined set of operation conditions.

Currently a large-scale test of a castellated PFC is being carried out at the JET tokamak which has been in operation since the year 2011 with the metal ITER-like wall (JET-ILW): beryllium limiters and tungsten divertor tiles [29–34]. This work is focused on the morphology of the castellated beryllium structures in JET-ILW: upper dump plates (UDP), Wide and Narrow Outer Poloidal (WOPL and NOPL, respectively) and inner wall guard limiters (IWGL) after two experimental campaigns in 2011–2012 (ILW-1) and in 2013–2014 (ILW-2). Each of those campaigns lasted for about 19.5 h of plasma discharges including approx. 13 h of limiter and 6.5 h of x-point operation. There were, however, differences in the total input energy (ohmic, neutral beams, ion cyclotron, lower hybrid): 150.6 GJ in ILW-1 and 200.5 GJ in ILW-2.

The emphasis in the study was on: (i) material mixing and compound formation on the PFS; (ii) fuel inventory and deposition inside the grooves of the castellation and on the side surfaces of the tiles, i.e. in areas located in the gaps; (iii) modelling of material transport and deposition in the gaps.

2. Experimental approach and modelling

The images in figures 1(*a*)–(*c*) show a segmented structure of a WOPL and the appearance of two types of the castellated limiters in JET, while the data collected in table 1 provide detailed information on the number of the various limiters and their respective castellation blocks. As inferred from these data, the total number of castellated blocks is nearly 170 000, the length of surfaces inside the castellation (both sides of the groove) is 7325 m, with an area of nearly 88 m^2. Therefore, the area is distinctly greater than the area of the PFS of the limiters: 24.5 m^2. The area of the side surfaces (in the gaps between tiles) is approximately 12.4 m^2, while the total length along both sides of all tiles is about 425 m.

The condition *sine qua non* for studies of castellated structures was the sectioning of the tiles in order to expose the surfaces located inside the grooves. The limiter blocks were sectioned into smaller specimens: single castellation pieces, i.e. typically 12 × 12 × 12 mm. Cutting was carried out at the Institute of Atomic Physics (Bucharest, Romania) under strict control of the tile temperature (below 60 °C) in order to avoid the release of hydrogen isotopes, because thermal desorption of fuel species from several samples was planned. In general the cutting was done approximately 0.5 mm above the bottom of the castellated groove, but in a few cases cutting was performed approx. 0.5 mm below the bottom of castellation. The latter samples were then split to expose the all of the surfaces located in the groove. This was to check the material transport to the very bottom of the groove.

The analyses described in sections 3.1 and 3.2 were performed by means of a number of complementary techniques: (a) x-ray diffraction (XRD) in order to determine the phase composition of limiter surfaces; (b) micro ion beam analysis (μ-IBA) techniques (lateral resolution of 8–10 μm) such as nuclear reaction analysis (μ-NRA) to determine the content of deuterium and particle-induced x-ray emission (μ-PIXE) to quantify the content of metals inside the castellation: tungsten and Inconel constituents, i.e. Ni, Cr, Fe. From the top of the castellated groove several consecutive regions, 1.8 × 1.8 mm, were scanned. Results from a given region were averaged to obtain a line scan and this procedure was repeated for all analysed regions. It is stressed that a complete analysis of one sample takes between 14 and 44 h. Therefore, only a limited number of specimens could be probed. The analysis was preceded by a very thorough selection of specimens to be

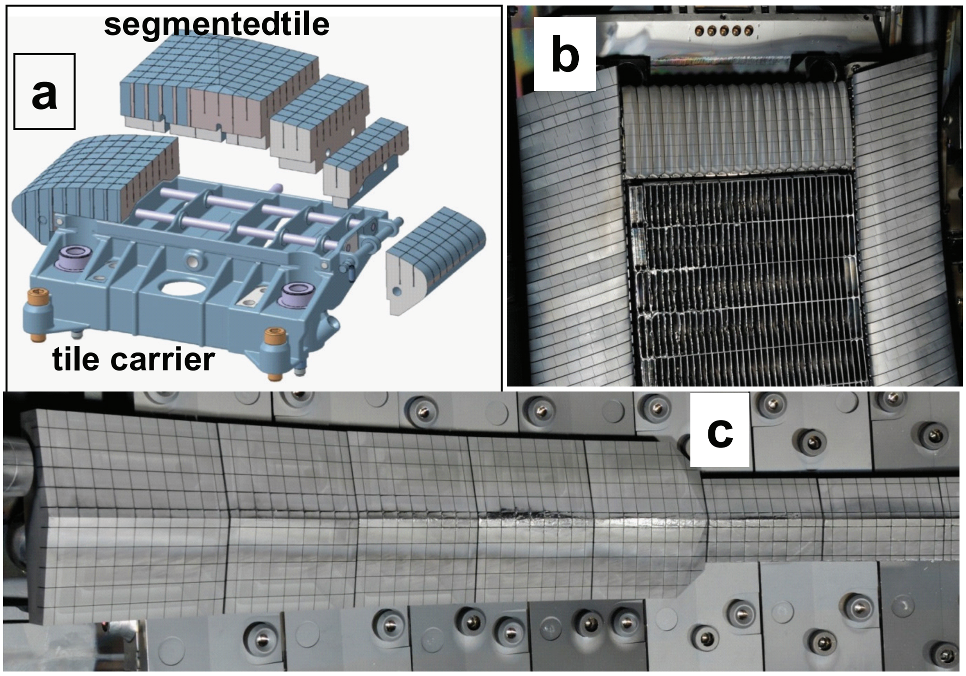

Figure 1. Structure of segmented inner wall guard limiter with castellated Be blocks (*a*); lower hybrid antenna protection (*b*); UDP (*c*).

examined in order to get the best possible overview of deposition and retention. Poloidal and toroidal gaps from all major types of limiters have been studied with ion beams: IWGL, OPL and UDP. Specimens from other components (e.g. the protection screen of the ion cyclotron antennae and saddle coils) were not available for studies. For comparison also the side surfaces of the bulk tungsten lamellae from the JET-ILW divertor (Tile 5) were analysed.

Modelling of Be re-deposition and respective D influx within the poloidal gaps of a relevant depth of 12 mm and three different widths (0.5, 1.0 and 2.0 mm) was performed with the 3D-GAPS code [23, 24].

Table 1. Detailed data on castellated beryllium limiters in JET-ILW.

Limiter type	Number of tiles	Number of castellated blocks	Length along castellation (cm)
Inner wall guard	217	43 000	191 134
Wide outer poloidal	225	50 013	216 056
Narrow outer poloidal	57	8126	38 426
UDP	448	40 448	186 624
Saddle coil	232	26 208	91 123
Lower hybrid protection	4	1408	9000
Totals	1183	168 203	732 363

3. Results and discussion

3.1. Intermetallic compounds on PFS

It has been shown in earlier studies of the limiter and divertor tiles from JET-ILW that—as expected—the surfaces contain a mix co-deposited species: fuel atoms, beryllium, carbon, nitrogen and metals [35–39]. However, no information on the chemical form has been given, while the point should not be neglected in the environment containing both Be and W which are known to form binary mixed materials of low melting point [40]. Herewith, the results of the first-ever studies are presented. A whole limiter tile after exposure in JET-ILW is shown in figure 2(*a*) while plots in figure 2(*b*) show the diffractograms recorded for a reference Be target and in the deposition and erosion zones of the limiter. The results clearly prove that some Be–W intermetallic compounds (Be_2W, $Be_{12}W$, $Be_{22}W$) were formed in the deposition zone. The erosion zone contains only metallic Be; the diffractogram has the same features as that for the reference Be sample, thus

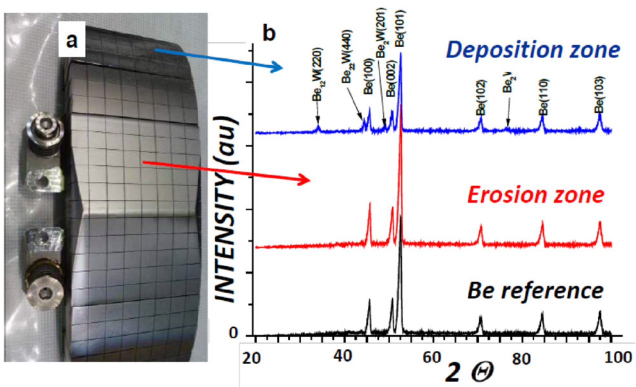

Figure 2. Castellated beryllium limiter tile from JET-ILW (*a*); x-ray diffractograms recorded for the initial limiter surface, erosion and deposition zones (*b*).

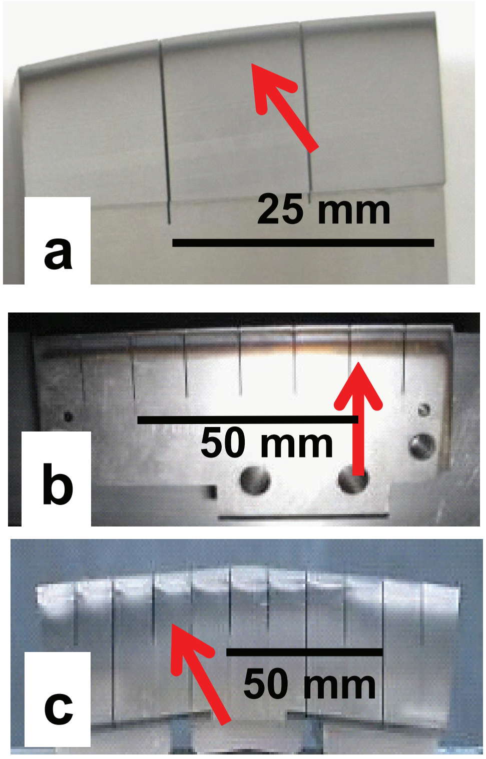

Figure 3. Deposition in the castellated groove (*a*); deposits of different width on the side surfaces of the tiles, in the gaps between the tiles (*b*) and (*c*).

proving the absence of the binary Be–W materials. The latter result is perceived as a positive one: no compound formation in the erosion zone indicates that no changes in the thermo-mechanical properties of the Be PFC could be expected.

3.2. Deposition in the castellation and gaps between tiles

The image in figure 3(*a*) shows the appearance of the surfaces in the castellated grooves, while in figures 3(*b*) and (*c*) there are side surfaces of limiter tiles, i.e. the surface located in the gap between tiles. In all cases one perceives deposition belts, marked with red arrows, in the top part of the tiles, i.e. at the very entrance to the groove. It is noticed, however, that the extent of deposition (i.e. width of the belts) in the castellation is significantly narrower than that on the side surfaces of the tiles.

For the castellated grooves μ-IBA was performed on more than 70 specimens from the top to the bottom of the groove. The plots in figure 4 show the deposition profiles of the deuterium and metals (the plot for metals is magnified by a factor of 1000) in a toroidal gap of the IWGL (first ILW campaign 2011–2012). The deposition width of the deuterium is approximately 1 mm and this is characteristic for the profiles recorded in narrow gaps. The profile has a fine structure: (i) low D content at the very entrance to the gap, (ii) increase in the concentration with a maximum reached at about 0.5 mm and then a sharp decrease. The plots in figure 5 show a comparison of depositions in two perpendicular gaps, toroidal and poloidal, on the same specimen from the OPL. The differences

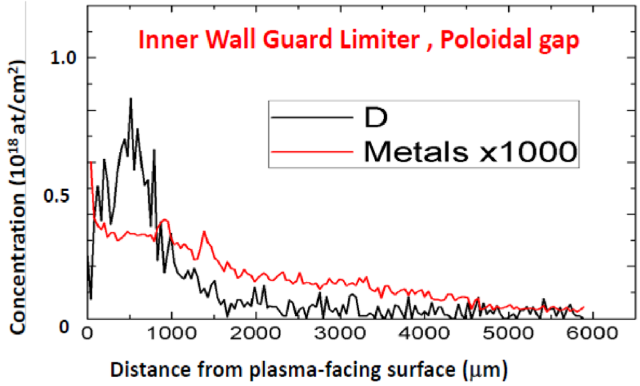

Figure 4. Deposition profiles of deuterium and metals inside the castellated groove of a beryllium limiter.

are insignificant both in the shape of the profiles and the deuterium content. Also the content of the nickel from the Inconel eroded from the wall is very small: well below 1×10^{15} cm^{-2}, while the level of tungsten does not exceed 2×10^{13} cm^{-2}.

Most of the profiles measured after the two campaigns are qualitatively and quantitatively very similar to those presented in figures 4 and 5. There are a few different cases: (i) flat profiles with a very small D content, below 1×10^{17} cm^{-2}; (ii) very narrow profiles, less than 0.5 mm, peaked at 1×10^{19} cm^{-2}. The latter is presented in figure 6. It shows a deposition from the top to the very bottom of the castellated groove. The measurement was performed in order to verify the hypothesis on the deposition at the bottom which may act as the ultimate trap for neutrals which either entered the gap at the right angle or reached the bottom as a consequence of multiple reflections. Indeed, a small and narrow peak is detected at the bottom, but the deuterium content is very low. It is, therefore, not decisive for the overall inventory. It should also be stressed that the amount of carbon is very low thus confirming a small amount of carbon impurities in JET-ILW.

As already mentioned in section 2, the selection of samples for μ-IBA examination was very thorough to get the best possible overview of the deposition and retention in the castellation grooves of tiles from various regions of JET-ILW. A summary of the results is given in table 2, where one also finds a comparison of the total retention on PFS and in the grooves. The estimated total retention in the castellation is in the range of 0.7×10^{22} to 14.6×10^{22} D atoms. The upper value is on the same level as that found for PFS. It should be stressed that these quantities are low as they correspond in total to about 1 g of deuterium retained in the limiters on the main chamber wall.

The plot in figure 7 shows deuterium deposition on the side surface, i.e. in the 2.0–2.5 mm wide gap between the two tiles of the IWGL. One may notice some important features. The profile is broad; it extends over 20 mm into the gap. The D concentration reaches its maximum approximately 5 mm deep in the gap and then it sharply decreases. The maximum is 1.4×10^{18} cm^{-2} and the average is about 0.5×10^{18} cm^{-2}, i.e. it is not greater than that measured in the narrow grooves castellated. Taking into account the length of the gaps along all tiles and the width of the deposition belts on the tile

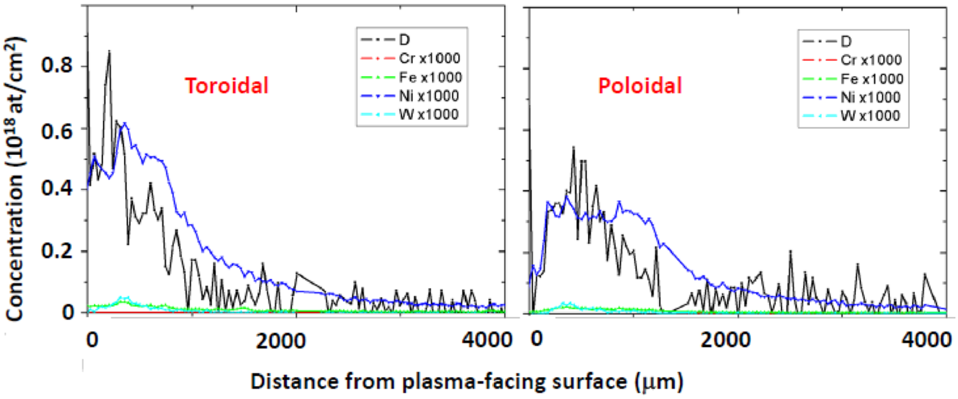

Figure 5. Deposition profiles of deuterium and metals in two perpendicular gaps of the outer poloidal limiter.

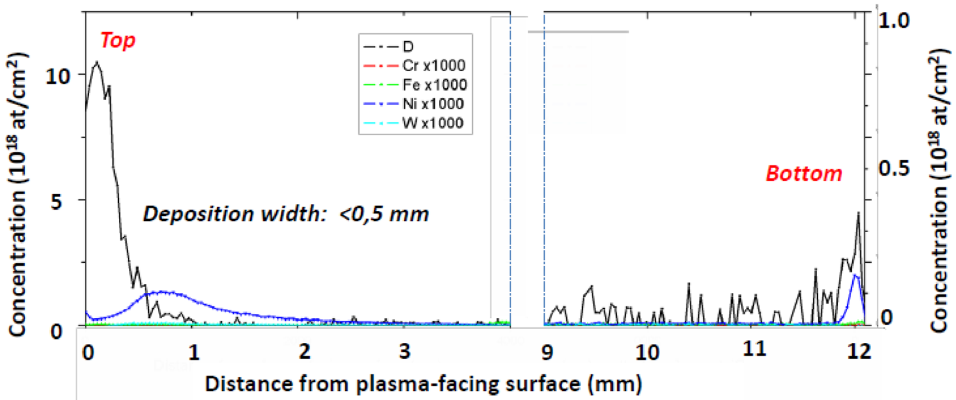

Figure 6. Deposition profiles of deuterium and metals along the entire depth of the castellated groove.

Table 2. Comparison of deuterium retention on the PFS and in the castellation and on the side surfaces of the limiters.

	Plasma-facing side	Inside castellation	Side surfaces
Surface are (m^2)	24.5	87.9	12.4
Deuterium content (10^{22})	12.4 [10] *WOPL, UDP, IWGL*	0.7–14.6 *All grooves*	0.6–3.0 *All gaps between tiles*

surfaces, from 3.5 to 30 mm wide, one may estimate the total fuel content in the gaps in the range 0.6–3.0 × 10^{22} D atoms, corresponding to 0.02–0.1 g of deuterium. These amounts are low in general and they are lower than assessed for the castellated grooves, but it must be stressed that the length of all gaps between the tiles (425 m) is over 17 times smaller than the lengths in the castellation. Therefore, all data give a strong indication that tiles or their segments should be separated by narrow grooves or gaps to reduce fuel retention.

3.3. Modelling

The drawings in figures 8(*a*) and (*b*) show the geometry of the gaps for which modelling was performed; ion flux to the gaps is marked in red, while the results are plotted in figures 8(*c*) and (*d*). The influx of D and Be ions into the poloidal gaps and respective Be re-deposition there were modelled with the 3D-GAPS impurity transport code [23, 24]. In the simulations, a poloidal gap of a relevant depth of 12 mm is assumed. The effect of the gap width on the re-deposition is addressed. Since a detailed simulation of the entire campaign with different

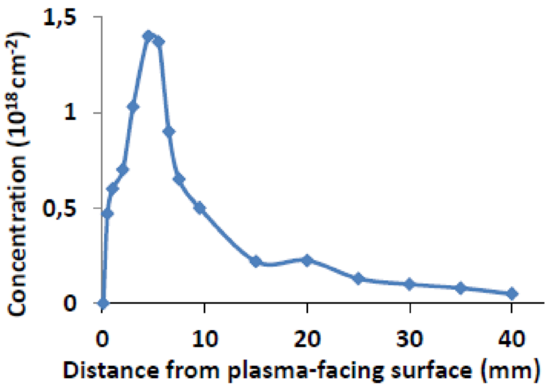

Figure 7. Deposition profile of deuterium on the surface located in the gap between the two limiter tiles.

operation scenarios and varying plasma parameters and fluxes is not feasible, the simplest case was simulated assuming a projected along magnetic field lines plasma flux and an additional uniform isotropic neutral flux to the gap aperture. Such a setup allows for a linear scaling of the particle influx into the gaps with plasma flux, impurity content, gap width and magnetic field

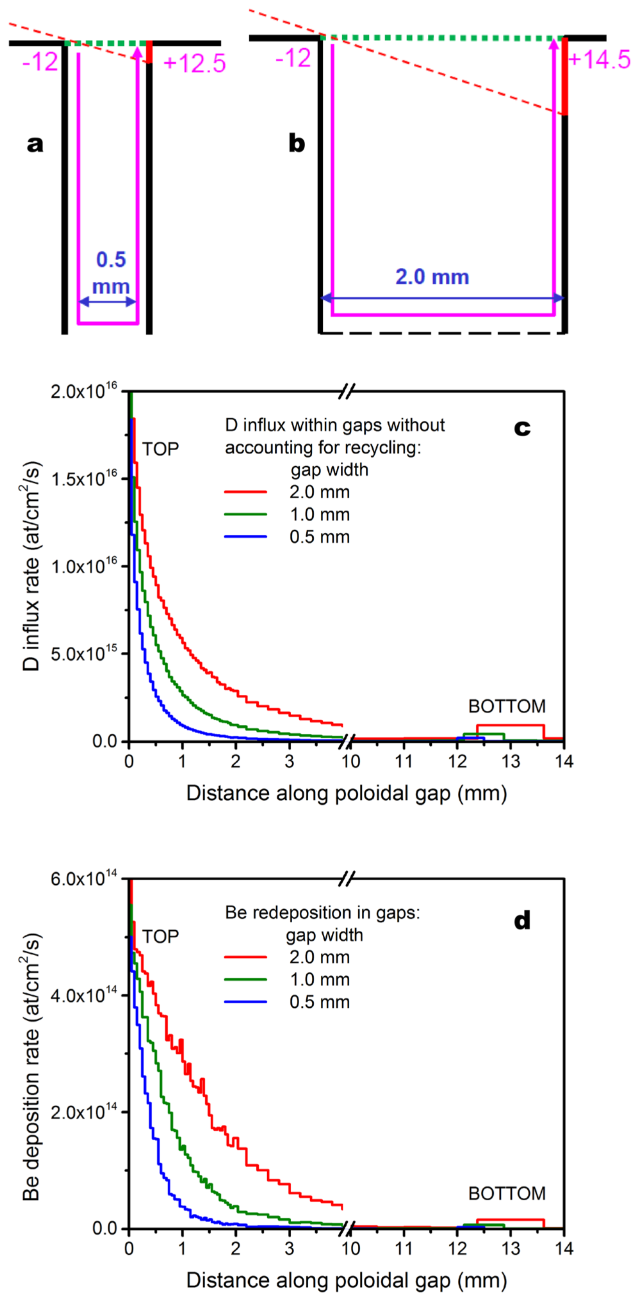

Figure 8. Modelling of deposition in the castellation: geometry of the gaps (*a*) and (*b*); simulated deposition profiles of deuterium (*c*) and beryllium (*d*) in the gaps of various widths.

angle. A parallel plasma flux of 2×10^{17} D^+ cm^{-2} s^{-1} with the ion impact energy of 100 eV was assumed in the simulations. Field lines inclination was set to 4.5 degrees with respect to the wall surface. The neutral D flux isotropically entering the gap aperture was set to 2×10^{16} D^0 cm^{-2} s^{-1} with the impact energy of 10 eV. In both fluxes a Be-fraction of 1% was assumed. In such a setup, the total neutral influx appears to be 25% higher than the total ion influx due to the ratio between the plasma-wetted area (0.04 mm for 0.5 mm gap width) and the gap width. This ratio remains the same for all gap widths since both, the gap aperture and the projected plasma-wetted area, increase proportionally. Reflected particles were set to follow the cosine angular distribution, and the reflection coefficient was fixed to 0.2 for D and chosen according to Eckstein [41] for Be. Energy reflection coefficients of 0.1 for D and 0.3 for Be were imposed. Sputtering of bulk Be by incoming ions was taken into account. No D recycling was considered, thus the maximal D deposition (influx) was simulated. This gross D amount cannot and should not be compared with much lower measured D content in deposited layers. Recycling during plasma operation and outgassing of dynamically retained D during the long storage time until actual measurement reduce the D content by orders of magnitude. The resulting Be re-deposition and D influx profiles along the gap side from the top (near plasma) to the bottom of the gap are shown in figures 8(*a*) and (*b*). The simulation results are shown for the plasma-shadowed side of the gap and for the gap bottom.

It has to be noted that absolute comparison with experimental results is not possible since Be re-deposition on Be cannot be measured while the D content in re-deposited layers is not directly assessable in simulations, as it was explained above. Nevertheless, an empirical scaling of D content in co-deposits with Be exists [42] that may be used to estimate the total D content based on the re-deposition of beryllium. The main message, however, is that the simulations show a qualitatively similar picture as measurements with a fairly steep decay of deposition inside the gaps and a relatively small deposition near the gap bottom. Profiles for broader gaps are distinctly less sharp and the quantities deposited are significantly increased: the deposition increases proportionally with the gaps width. This is because of the increased particle fluxes reaching the gap. In summary, the modelling fully reflects the main features of the measured deposition profiles both for the grooves of the castellation (0.5 mm) and the broader gaps (approx. 2.0–2.5 mm) between the tiles.

4. Summary and concluding remarks

This work has brought several important results which have provided insight into deposition phenomena in the large-scale castellated beryllium structures. For the first time ever a comprehensive analyses have been carried out for materials retrieved from all important locations in the machine. Results on the fuel retention on surfaces in the gaps are summarised by a number of points. Very shallow deuterium deposition is measured in the castellation: 0.5–1.5 mm deep into the groove. No qualitative and quantitative differences are identified between the results from the two JET-ILW campaigns. This occurs despite the fact that approximately 30% more deuterium was measured on the PFS after the first campaign in comparison to ILW-2 [43]. The latter was finished with approximately 300 discharges performed with hydrogen fuel. It is an important indicator that fuel once stored in narrow gaps is difficult to remove by any means. Small quantities of D are found in the castellation both in the erosion and deposition zones. No difference is observed between the poloidal and toroidal gaps. This result could be expected after long operation periods, i.e. full experimental campaigns when different operation scenarios were tested and a large number of disruptions also occurred. No dust accumulation was detected inside the castellation. It was also found that the preparation of castellated tiles before their installation in the JET wall (machining and cleaning of the grooves) was thoroughly performed: only traces of copper residue (below 3×10^{14} cm^{-2}) could be detected.

These comprehensive studies of the deposition in the castellated grooves of the beryllium limiters from JET-ILW indicate that the total deuterium content can be estimated in the range from 0.7×10^{22} to 14.2×10^{22} in the castellation and 0.6×10^{22} to 3.0×10^{22} in the gaps between the tiles. The upper value is on the same level as the retention determined by Heinola [38] on the PFS of the limiters: WOPL, IWGL and the upper dump plate. It is also worth mentioning that only small retention was measured also on the surfaces in the gaps between the W lamellae in the divertor. Most important is that the overall retention is significantly lower than that measured after the campaigns in JET-C.

All measurements consistently show small retention and steep deposition profiles on the surfaces inside the castellation. These results actually could be expected. The statement is based on earlier data for metallic castellated structures used in the presence of carbon walls: (i) in short-term probes or test limiters exposed in TEXTOR [16, 20], (ii) other machines [22] and, especially, in the beryllium divertor and limiters in JET-C [11, 14]. The following D concentrations were measured with a standard size ion beam (spot around 1 mm in diameter) for the divertor: 6×10^{17} cm^{-2} at the entrance to the 0.6 mm wide castellated gap, 2×10^{19} cm^{-2} in 6 mm wide gaps separating the tiles [14]. In the 1 mm wide castellation of the belt limiters the amounts of D at the level of 7×10^{17} cm^{-2} were determined [11]. The decay length for the castellated grooves was around 1.5 mm in both cases. The tendency indicating the increase in deposition and retention is shown here, but the absolute D content must be taken with care, because measurements were performed a long time after the end of operation: 11 and 15 years for the divertor and limiters, respectively.

Short-term experiments (e.g. in TEXTOR) could be modelled, but modelling of results after entire experimental campaigns is difficult because of a variety of operation scenarios. However, modelling with the 3D-GAPS code has successfully reproduced steep profiles in narrow gaps (0.5 mm). The calculations also show a very significant increase in deposition (and inventory) with the increase in the gap width, e.g. by a factor exceeding 10 when the width of the castellation is increased from 0.5 mm to 2 mm. In conclusion, the experimental and modelled results give a clear indication for ITER regarding the need for a very careful design of the tiles with a particular emphasis on the tile shaping and small width of the castellation grooves.

Acknowledgments

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