Fuel Inventory and Deposition in Castellated Structures in JET-ILW

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Abstract. Since year 2011 the JET tokamak has been operated with a metal ITER-Like Wall (JET-ILW). This has allowed for a large scale test of castellated plasma-facing components (PFC). Morphology of beryllium castellated limiters was examined after experimental campaigns 2011-2012 and 2013-2014. The deposition in the 0.4-0.5 mm wide grooves of castellation is "shallow". It reaches 1-2 mm into the 12 mm deep gap. Deuterium concentrations are small (maximum $4x10^{18}$ cm^{-2}). The estimated total amount of deuterium in all castellated limiters does not exceed the inventory on plasma-facing surfaces of limiters. There are only traces of Ni, Cr and Fe deposited in the castellation gaps. The same applies to the carbon content. On plasma-facing surfaces X-ray diffraction has clearly shown two distinct composition patterns: Be-W mixed intermetallic compounds (e.g. $Be_{22}W$) on the sides of limiters (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 thermo-mechanical properties of Be PFC might be expected.

1. Introduction

All plasma-facing components in ITER will be castellated because this structure is deemed as the best solution to ensure thermo-mechanical durability and integrity of materials under high heat flux loads. However, in the environment containing also low-Z elements (e.g. beryllium and some impurities of carbon), eroded material 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 side surfaces of plasma-facing components (PFC), e.g. on surfaces located in gaps separating the tiles. The determination of morphology of deposits 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. Based on the earlier experience from JET, co-deposits formed in remote areas (e.g. inner divertor corner and water-cooled louvers) are very difficult to remove by cleaning methods and this may result in significant long-term fuel retention. Secondly, very limited access to in-vessel components of ITER calls for more detailed studies of deposition in gaps and grooves of PFC from present-day devices.

A large scale test of castellated PFC is carried out at the JET tokamak which has been operated since year 2011 with the metal ITER-Like Wall (JET-ILW): beryllium limiters and tungsten divertor tiles [1-5]. It should also be stressed that in the past large-scale castellated

Be structures were used at JET operated with carbon walls (JET-C): toroidal belt limiters and Mk-I-Be divertor [6,7].

This contribution is focused on the morphology of 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 2011-2012 and 2013-2014. 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. The emphasis in the study was on: (i) material mixing on plasma-facing surfaces; (ii) fuel inventory and deposition inside the grooves of the castellation, i.e. on surfaces located in the gaps; (iii) modelling of material transport and deposition in the grooves.

2. Experimental Approach and Modelling

Images in Fig. 1(a)–(c) show a segmented structure of a limiter 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 various limiters and respective castellations. As inferred from these data, the total number of castellations is nearly 170000, the length of surfaces inside the castellation (both sides of the groove) is 7325 m, area nearly 88 m^2. Therefore, the area is distinctly greater than the area of plasma-facing surfaces of the limiters: 24.5 m^2.

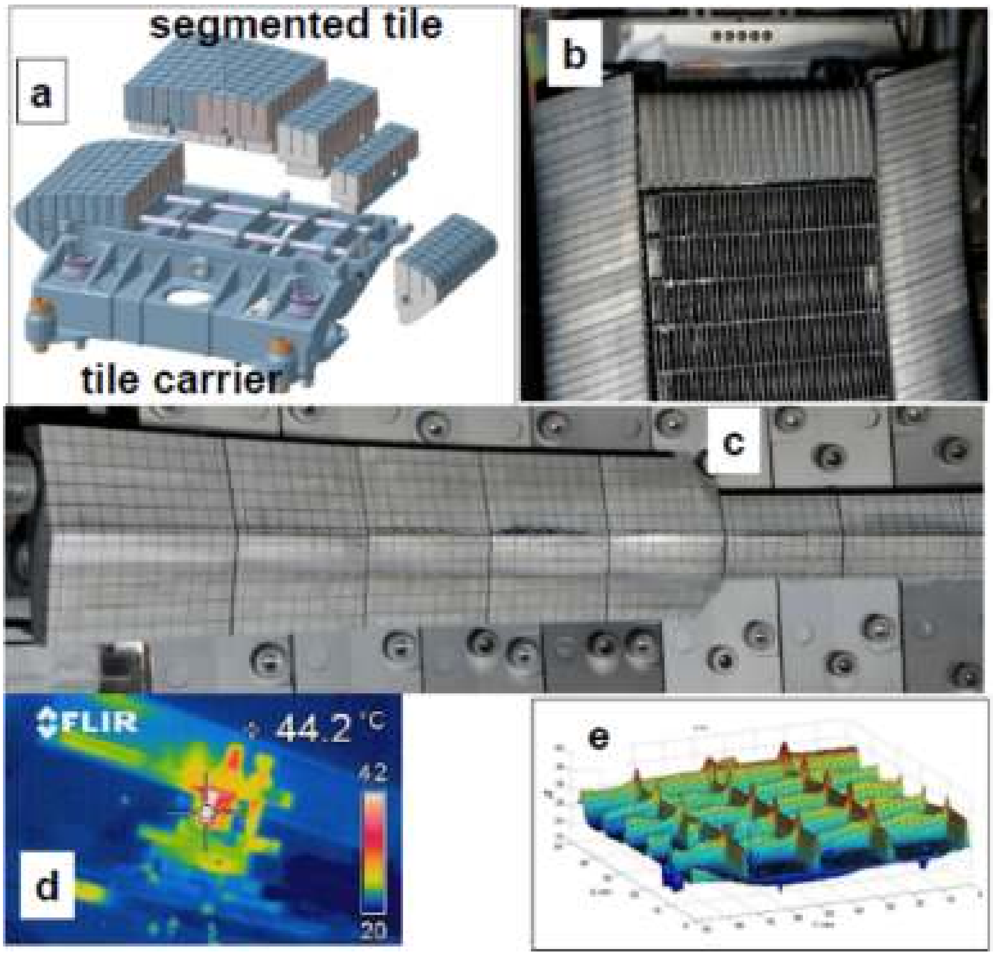

FIG. 1. Structure of segmented inner wall guard limiter with castellated Be blocks (a); lower hybrid antenna protection (b); upper dump plates (c); maps of temperature during limiter cutting (d, e).

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

Limiter type	Number of tiles	Number of castellations	Length of castellations [cm]
Inner Wall Guard	217	43000	191134
Wide Outer Poloidal	225	50013	216056
Narrow Outer Poloidal	57	8126	38426
Upper Dump Plates	448	40448	186624
Saddle Coil	232	26208	91123
Lower Hybrid Protection	4	1408	9000

Condition sine qua non for studies of castellated structures was sectioning of the tiles in order to expose surfaces located inside the grooves. The limiter blocks were sectioned into smaller specimens: single castellation pieces, i.e. typically 12x12x12 mm. Cutting was carried out at the Institute of Atomic Physics (Bucharest, Romania) under strict control of the tile temperature (below 60 ^{o}C) in order to avoid the release of hydrogen isotopes, because thermal desorption was planned on some samples. The temperature map during the cutting process is shown in Fig. 1(d). 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 appox. 0.5 mm below the castellation. The latter samples were then split to expose entire surfaces located in the groove. This was to check the material transport to the very bottom of the groove.

The analyses described below (Paragraphs 3.1 and 3.2) were performed by means of: (a) X-ray diffraction (XRD) in order to determine the phase composition of limiter surfaces; (b) micro ion beam analysis (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, 1.8x1.8 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 of one sample takes between 14 and 44 h. Therefore, the analysis was preceded by a very thorough selection of specimens to be 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. For comparison also 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 poloidal gaps of a relevant depth of 12 mm and three different widths (0.5, 1 and 2 mm) was performed with the 3D-GAPS code [8]. The most simple representative case was simulated assuming a projected along magnetic field lines plasma flux of $2x10^{17}$ D^{+} $cm^{-2}s^{-1}$ and an additional uniform isotropic neutral D flux to the gap aperture up to $2x10^{16}$ D^{0} $cm^{-2}s^{-1}$, both with a Be fraction of 1%. D and Be particle reflection was assumed to follow the cosine distribution with the reflection yield fixed to 0.2 for D and chosen according to Eckstein [9] for Be. No D recycling was considered. In such a setup, the total re-deposition within the gap simply scales linearly with the gap width.

3. Results and Discussion

3.1. Intermetallic compounds on plasma-facing surfaces

A whole limiter tile after the exposure in JET-ILW is presented in Fig. 1(a) while plots in Fig. 1(b) show 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 sample. The latter result is perceived as a positive one: no compound formation in the erosion zone indicates that no changes in thermo-mechanical properties of Be PFC might be expected.

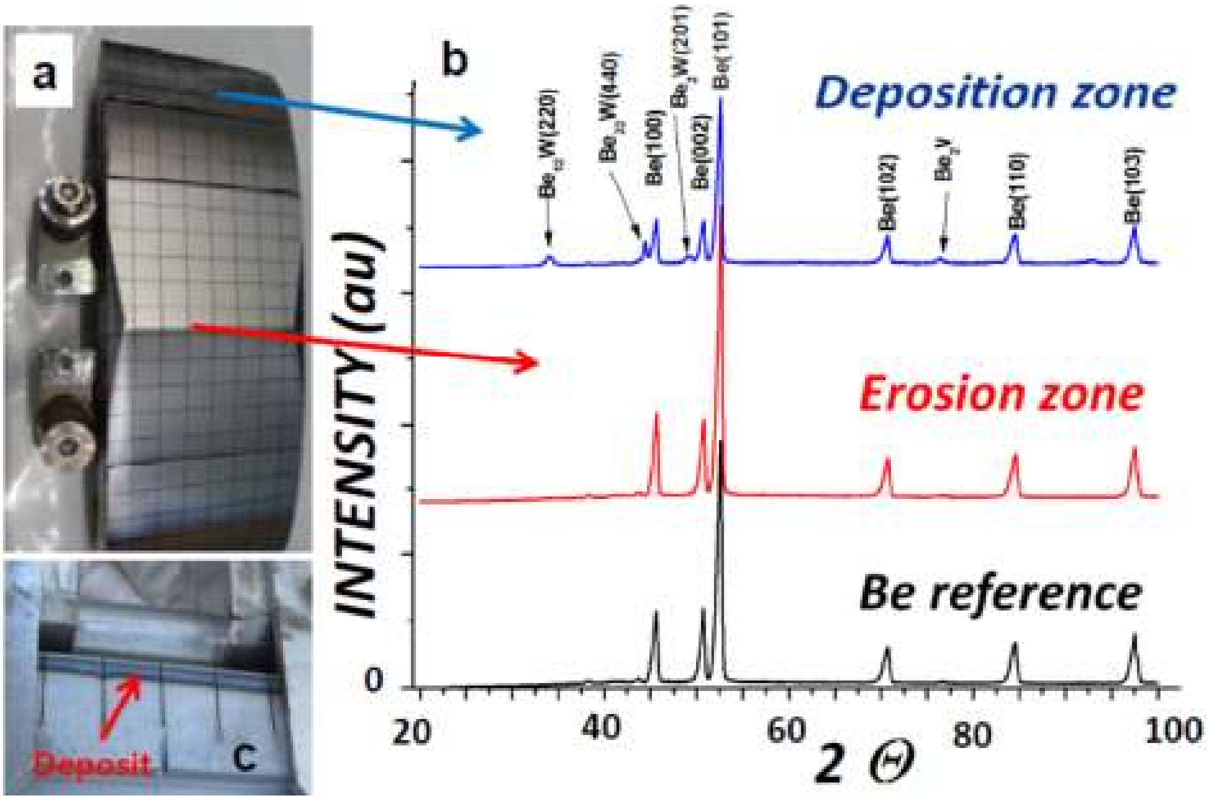

FIG.2. (a) Castellated beryllium limiter tile from JET-ILW; (b) X-ray diffractograms recorded for the initial limiter surface, erosion and deposition zones; (c) side of the sectioned tile.

3.2. Deposition in the castellation

Fig. 2(c) shows the surface in the castellated groove. One perceives a narrow deposition belt, marked with a red arrow, in the top part of the tile, i.e. at the very entrance to the groove. Micro-IBA was performed for more than 70 specimens from top to the bottom of the gap. Plots in Fig. 3 show representative deposition profiles of deuterium and metals (features magnified by a factor of 1000) in a toroidal gap of the IWGL (first ILW campaign 2011-2012). The deposition width of deuterium is approximately 1 mm and this is characteristic for all recorded profiles. The profile has a characteristic fine structure: (i) low D content at the very entrance to the gap, (ii) increase of the concentration with maximum reached at about 0.5 mm and then sharp decrease. Plots in Fig. 3 show a comparison of deposition in two perpendicular gaps, toroidal and poloidal, on the same specimen from the OPL. Differences are insignificant both in the shape of profiles and the deuterium content. Also the content of nickel from Inconel eroded from the wall is very small: well below $1x10^{15}$ cm^{-2}, while the level of tungsten does not exceed $2x10^{13}$ cm^{-2}.

Most profiles measured after the two campaigns are qualitatively and quantitatively very similar to those presented in Fig. 3 and 4. There are a few different cases: (i) flat profiles with a very small D content, below $1x10^{17}$ cm^{-2}; (ii) very narrow profiles, less than 0.5 mm, peaked at $1x10^{19}$ cm^{-2}. The latter is presented in Fig. 5. 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 ultimately trap for neutrals which either entered the gap at 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 small amount of carbon impurities in JET-ILW.

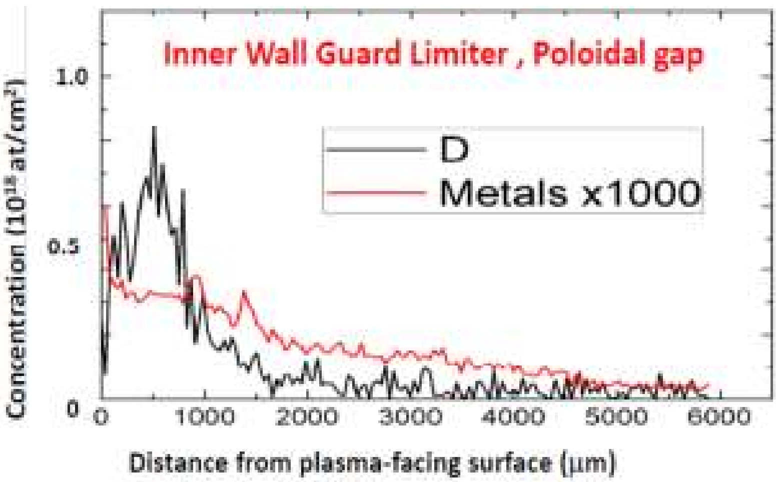

FIG. 3. Deposition profiles of deuterium and metals inside the castellated groove of a beryllium limiter.

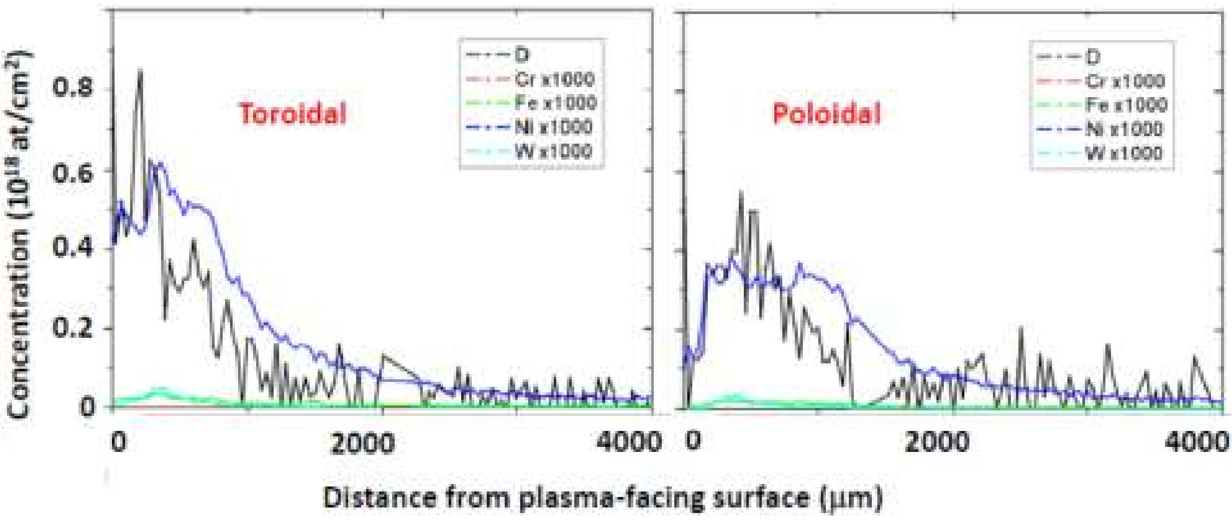

FIG.4. Deposition profiles of deuterium and metals in two perpendicular gaps of the outer poloidal limiter.

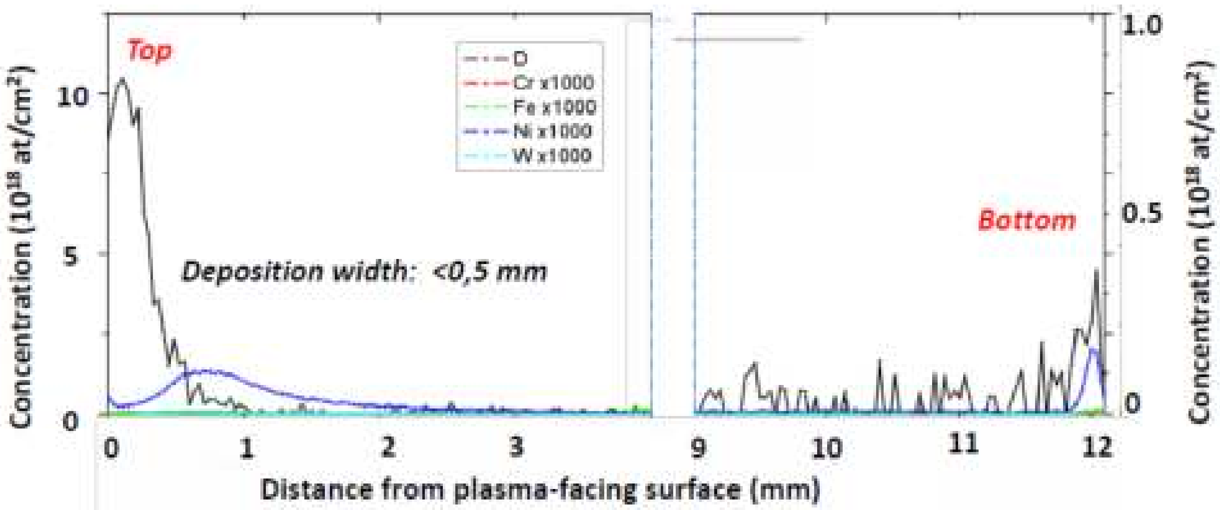

FIG.5. Deposition profiles of deuterium and metals along the entire depth of the castellated groove.

As already mentioned in Paragraph 2, a selection of samples for μ-IBA examination was very thorough to get the best possible overview of deposition and retention in the castellation grooves of tiles from various regions of JET-ILW. The summary of results is in Table 2, where one also finds a comparison of the total retention on plasma-facing surfaces (PFS) and in the grooves. The estimated total retention in the castellations is in the range from 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.

Table 2.Comparison of deuterium retention on plasma-facing surfaces and in the castellation.

	Plasma-facing side	**Inside castellation**
Surface are [m2]	24.5	87.9
Deuterium content [10^{22}]	12.4 (*reference* [10]) *WOPL, UDP, IWGL*	0.7 – 14.6 *All grooves*

3.3.Modelling

Drawings in Fig. 6(a) and (b) show the geometry of the gaps for which modelling was performed; ion flux to the gaps is marked in red. Results plotted in Fig. 6(c) clearly indicate

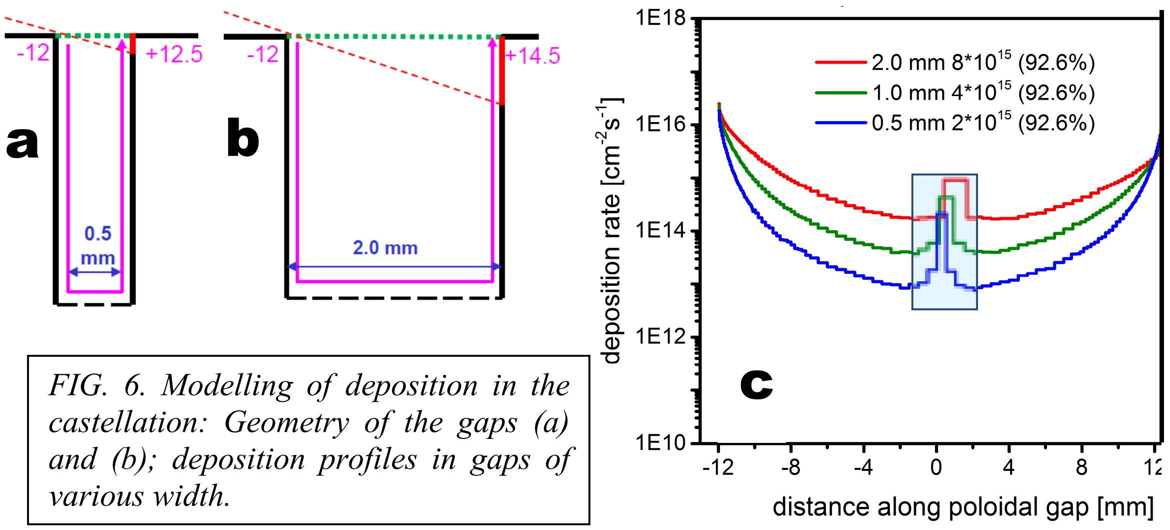

FIG. 6. Modelling of deposition in the castellation: Geometry of the gaps (a) and (b); deposition profiles in gaps of various width.

that the deposition profile in a narrow gap (0.5 mm) is very steep: a decrease of deposition by over an order of magnitude within the first mm into the gap. 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. There is also a certain increase of deposition at the very bottom of the gap. One concludes that modelling reflects and reproduces very well experimentally measured deposition profiles.

4. Summary and concluding remarks

Results on the fuel retention on surfaces in the gaps (castellated Be limiters and W lamellae) are summarised by several points.

- Very shallow deuterium deposition is measured in the castellation: 0.5 – 1.5 mm. No differences are identified between results from the two JET-ILW campaigns.
- Small quantities of D are found in the castellation both in erosion and deposition zones.
- No difference is observed between poloidal and toroidal gaps. This result could be expected after long operation periods, i.e. full experimental campaigns.
- No dust accumulation is detected inside the castellation.
- Very small retention is measured on surfaces in gaps between W lamellae in the divertor.

These comprehensive studies of deposition in the castellated grooves of the beryllium limiters from JET-ILW indicate that the total deuterium can be estimated in the range from $0.7\text{x}10^{22}$ to $14.2\text{x}10^{22}$. The upper value is on the same level as the retention determined by Heinola [10] on plasma-facing surfaces. The most important is that the overall retention is significantly lower than that measured after campaigns in JET-C.

All measurements consistently show small retention and steep deposition profiles on 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) beryllium divertor and limiters JET-C [6,7] and (ii) short-term probes or test limiters exposed in TEXTOR [11] and in other machines [12]. Short-term experiments could be modelled, but modelling of results after entire experimental campaigns is difficult because of a variety of operation scenarios. However, modelling with the ERO-code has successfully reproduced steep profiles in narrow gaps (0.5 mm). Calculations also show very significant increase of deposition (and inventory) with the increase of the gap width, e.g. by a factor exceeding 10 when the width of castellation is increased from 0.5 mm to 2 mm. In conclusion, experimental and modelled results give a clear indication for ITER regarding the need for very careful design of tiles with particular emphasis on the tile shaping and small width of the castellation grooves.

5. Acknowlegements

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