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# Beryllium Migration in JET ITER-Like Wall Plasmas

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## ABSTRACT

The understanding of material migration is a key issue for a successful and safe operation of ITER. JET is used as test bed to investigate the process cycle which is connected the lifetime of first wall components by erosion and the safety due to long-term retention. Divertor configuration: The current understanding of Be migration in the JET-ILW can be described as follows: neutral Be and BeD from physical and chemical assisted physical sputtering by CX neutrals and residual plasma flux at the recessed wall enters the plasma, is dissociated, ionised and transported by SOL-flows towards the inner divertor where significant deposition takes place. The amount of Be eroded at the first wall and the amount of Be deposited in the inner divertor are almost comparable (21–28g). The primary impurity source in JET-ILW is by a factor 5.3 reduced in comparison with JET-C resulting in a lower divertor material deposition by more than an order of magnitude. Within the divertor, Be performs much less re-erosion and transport steps than C due to an energetic threshold for Be sputtering and inhibits by this the transport to the divertor floor and to remote areas at the pump-duct entrance. The overall low migration is also consistent with the low fuel inventory and dust production (<1g) with the JET-ILW. Limiter configuration: Be gross erosion at the contact point was in-situ determined by spectroscopy between 0:03% ( $E = 35\text{eV}$ ) and above 100% caused by Be self-sputtering ( $E = 200\text{eV}$ ). Chemical assisted physical sputtering via BeD has been identified to contribute to the effective Be sputtering yield, i.e. at  $E = 75\text{eV}$  about 1/3 enhanced erosion with respect to bare physical sputtering. An effective gross yield of 10% represents a representative yield for limiter plasma conditions in the initial campaign. This is equivalent to an average erosion rate of  $4.1 \times 10^{18} \text{Bes}^{-1}$  or 1.5g Be sputtered from one mid plane tile. The corresponding net erosion rate deduced from tile profiling - amounts  $2.3 \times 10^{18} \text{Bes}^{-1}$  if normalized to the total limiter time. This is equivalent to 0.8g Be revealing a factor two between net and gross erosion. The primary impurity source in limiter configuration in JET-ILW is only 25% above the JET-C case. The main fraction of eroded Be stays within the main chamber and only a small fraction of neutral Be escapes geometrically from the main chamber into the divertor.

## 1. INTRODUCTION

The understanding of material migration, thus the process cycle of material erosion, transport and deposition is one of the key issues for a successful and safe operation of the ITER tokamak and a future fusion reactor. The process cycle is associated with the lifetime of first wall material components, the so-called plasma-facing components PFCs, by erosion, and with the safety aspect due to long-term tritium retention. The latter is both in current fusion devices as well as in ITER dominated by co-deposition of tritium [1]. Most of the present knowledge is based on tokamaks with carbon-based first wall materials and in-situ information obtained from optical spectroscopy during plasma operation and combined with detailed post-mortem analysis after extraction of PFCs. In the common understanding, the main chamber is identified as primary erosion source and material is transport via scrape-off layer (SOL) flows in normal magnetic field configuration predominantly

towards the inner divertor region where finite material deposition occurs [2]. In a subsequent multistep process, material transport to remote and inaccessible areas takes place which led to the abandon of carbon as plasma-facing material due unacceptable high fuel retention content in co-deposited layers and inhibits safe conditions for ITER [3]. The outer divertor, though net-erosion zone, plays only a minor role in the overall large material transport which can reach g levels of migrating material for typical current day devices. Predictions to ITER and fusion reactors are based on this physics understanding and the adaption of the appropriate material selection is done with the aid of plasma-wall interaction codes such as ERO [4] or WALLDYN [5]. The exchange of PFCs as in ASDEX-Upgrade from graphite to full tungsten [6] and recently in JET from carbon-fibre composite to beryllium (Be main chamber) and W (W divertor) [7] provides the ideal test bed to verify the physics assumptions. Indeed both devices demonstrated a reduction in fuel retention and transport to remote areas, which underlined that with carbon, chemical erosion at low or thermal impact energy dominated the material migration cycle. Details about the residual carbon content in the JET-ILW are described in sec.2, beryllium erosion and transport in limiter configuration are presented in sec. 3 and in divertor configuration in sec. 4. The overall migration in JET-ILW, differences with respect to JET-C, and the physics mechanisms responsible for the vast reduction of migration are given in 5. Brief conclusions drawn for ITER from JET and a summary (sec.6) couplets this contribution.

## **2. RESIDUAL C-CONTENT IN JET-ILW OPERATIONAL REGIMES**

The carbon (C) content in the plasma edge dropped after installation of the JET-ILW by about a factor 20 in the diverted plasma phase of discharges throughout the first year of operation and prior to the exchange of selected PFC tiles [8]. Apart from an initial clean-up phase, the plasma operation can be described as virtually carbon-free, whereas the residual C from the clean-up is found at net-deposition zones like the divertor. The overall C reduction remained low and one order of magnitude below the values in JET-C in the second year of JET-ILW operation (fig.1a) though an increase of 50% with respect to the minimum C content values in JET-ILW has been detected (fig.1b). The reason is mainly caused by the exposure of back-sides of divertor CFC-tiles by high neutral deuterium flux which are not coated by W, by the release of C present in-situ in the W coating due to the manufacturing process, by C resulting from air leaks as well as at high input power and low density operation by potential damage of the W-coatings. The post-mortem results presented here are from the first tile intervention where all W-coated CFC divertor tiles were in-tact, thus, the JET-ILW was presenting ultimately a tokamak with Be/W PFCs – a good test bed for ITER [9]. However, JET with its inertial cooled PFCs and the inductive-pulsed operation is limited in discharge duration; the typical ratio between operational time in limiter and divertor configuration per discharge is about 1:3. Moreover, in the initial JET-ILW exploitation (2011–2012) a significant portion of the total plasma time (19h) was devoted to limiter operation (6h) which can be compared with 33h plasma time in divertor and 12h in limiter configuration in the last JET-C operation period

(2008–2009) [10]. Separation of the two operational regimes is required in order to describe the material migration cycle in JET-ILW.

### 3. LIMITER CONFIGURATION OPERATION WITH THE JET-ILW

Operation in limiter configuration was applied to qualify the design of the castellated massive Be PFCs and to verify the ERO code. In dedicated inner-wall limited discharges, the local plasma conditions at the inboard limiters as well as the deuteron impact energies were varied in the range of 35–200eV. The effective sputtering yield for Be gross erosion at the limiter contact point was in-situ determined by spectroscopy to be between 0.03% and more than 100% due to Be self-sputtering. The yields were compared with ERO showing an overestimation in the code by about a factor 2 [11]. The effective erosion yields are also about a factor 2 larger than C yields in JET-C whereas less impact of self-sputtering in the C case occurs at the high energy end. At the accessible low energy end in the limiter discharges, the yield is lower though chemical assisted physical sputtering (CAPS) of Be via BeD has been identified to contribute to the effective sputtering yield, i.e. at impact energies of 75eV and  $T_{\text{Be,surf}} = 200^\circ\text{C}$  about 1/3 enhanced erosion with respect to normal physical sputtering as shown in fig.2a [12]. The appearance of CAPS acts as an additional sputtering channel, but requires a certain amount of deuterium to be presented on the top-most interaction layer. As the deuterium content decreases with surface temperature, the impact of CAPS on the total sputtering yield decreases with Be surface temperatures and vanishes at  $T_{\text{Be,surf}} \approx 520^\circ\text{C}$ . The release of BeD can be in form of  $\text{BeD}_x$  with  $x = 1, 2, 3$ , whereas only BeD was experimentally observed by optical emission spectroscopy. With increase of  $T_{\text{Be,surf}}$ ,  $\text{D}_2$  is directly desorbed from the Be surface reducing the deuterium content in the interaction layer and causing the decrease of BeD emission whereas the deuterium recycling flux remained constant. Moreover, at higher impact energies, the beryllium sputtering yield is dominated by self-sputtering as indicated in fig.2b. Details are described in [12].

In a first approximation from spectroscopic measurements of BeII, thus  $\text{Be}^+$  which is unaffected of the initial beryllium sputtering process, an effective gross yield of 10% can be estimated to be a representative yield for the averaged limiter plasma conditions in the initial JET-ILW campaign. This results in an average Be erosion rate of  $4.1 \times 10^{18} \text{Be}^{s^{-1}}$  or 1.5g Be sputtered from one limiter tile ( $A_{\text{tile}} = 0.025\text{m}^2$ ) in the view of the spectroscopic system ( $A_{\text{spot}} = 0.011\text{m}^2$ ) in the first year of operation. Post-mortem analysis of Be tiles of the inboard limiters (fig.3a) provided information on the campaign averaged Be erosion rate by different techniques (profilometry, NRA, RBS) [10, 13, 14]. The net erosion rate for one mid plane Be tile deduced from tile profiling - amounts  $2.3 \times 10^{18} \text{Be}^{s^{-1}}$  if one considers the normalization to the total limiter exposure time in the campaign. This is equivalent 0.8g Be resulting from one mid plane tile which can be compared with spectroscopy. Comparison of the net erosion with the corresponding tile in JET-C and normalization to the operational time reveals a higher erosion rate in the JET-ILW case, however, taking into account the different number of interacting limiters in JET-C (#16) and JET-ILW (#10) reduces the discrepancy. The primary impurity source in limiter configuration in JET-ILW is only 25% above the JET-C case. This is in

good agreement with the spectroscopic observations, considering that some erosion of the limiters takes also place in the diverted plasma phase and that gross versus net erosion is compared.

For the total Be source estimation both, spectroscopy and post-mortem analysis, must extrapolate the local information to the total limiter interaction area which represents a fraction of the total inner wall protruding limiters ( $A_{\text{lim}}^{\text{HFS}} = 4.5\text{m}^2$ ). Additional measurements were obtained on the top and bottom tiles of the high-field side (HFS) poloidal limiter rail showing the peak erosion in the centre and almost negligible erosion at the other areas. Interpolation in poloidal direction and extrapolation to all toroidal limiters results in an estimation of the total Be erosion of about 8g in the first year of ILW operation. Apart from dedicated experiments, limiter operation is required in the ramp-up and ramp-down phase in all JET-ILW discharges connecting plasmas to the HFS and low-field side (LFS) limiters. Post-mortem analysis revealed that the erosion on the LFS limiters is poloidally asymmetric [13] with stronger erosion at the lower half of the limiter rail with the maximum at the centre. This centre tile showed Be erosion of at least 10m - erasing a Ni marker layer which challenges absolute quantification on the LFS. Further erosion has been observed in protection tiles which need to be taken into the calculation in future. The main chamber erosion of C in JET-C was previously estimated [14] to be amount 237g of C, but with a twice as long operational time in the erosion-dominated limiter phase.

The main fraction of Be eroded at the limiters in limiter configuration stays within the main chamber (fig.3b), mostly deposited in recessed areas like the limiter wings and partially on the wall cladding; only a small fraction of neutral Be escapes geometrically from the main chamber into the divertor entrance and can be there deposited. Indeed the initial JET-ILW experiment in diverted configuration identified moderate surface coverage of W by Be [15] after 625s in limiter configuration. However, the amount entering the divertor is insignificant in comparison with Be transported into the divertor in x-point plasmas with strike lines positioned on the target plates.

#### **4. DIVERTOR CONGURATION OPERATION WITH THE JET-ILW**

With the full W divertor installed in the JET-ILW, no PFCs made of the main chamber wall material are used in the divertor, thus, all Be ions flowing into the divertor and causing potentially W sputtering are originated primarily in the main chamber during diverted plasma operation. Homogenous Be and BeD emission in toroidal and poloidal direction at the inner wall can be measured by spectroscopy resulting from erosion processes in these recessed areas. The origin of these processes is twofold: energetic charge exchange neutrals (CXN) and residual plasma flux are impinging the recessed wall area ( $A_{\text{wall}}^{\text{HFS}} = 18.5\text{m}^2$ ) equipped primarily by Be-coated inconel cladding tiles (AHFS clad =  $11.2\text{m}^2$  of which 2/3 is Be and the rest protective W tiles) between poloidal limiters which are located typically between 6cm and 10cm behind the separatrix. Horizontal movement of the plasma column by several cm, thus variation of the distance to the Be surface of the same degree, is causing variations of the Be and BeD flux indicating that the low energetic deuterium ion flux contributes significantly to the erosion of the Be cladding. Dedicated Be long-term samples (sachet samples)

installed at different poloidal and toroidal locations between the cladding tiles prior to the first ILW plasma were replaced after the first year of operation for post-mortem analysis. All probes show measurable erosion [16] confirming that the Be cladding is a zone of net erosion as well as the limiters which are about 2cm closer to the plasma, but still deep in the SOL. Quantification of the erosion was performed by RBS providing a local Be erosion rate of  $0.78 \times 10^{18} \text{Be m}^{-2} \text{s}^{-1}$  when normalized to the total operational time. This results in a net erosion of 12.2g of Be from the whole inner wall in the integrated divertor time of the first ILW campaign. This rate can directly be compared with the erosion rate of  $3.14 \times 10^{19} \text{C m}^{-2} \text{s}^{-1}$  obtained in the operational phase 2005-2009 in JET-C where similar long-term samples were installed and analysed [17]. This is a significant difference by a factor of 4.0. The ratio between the total sputtering yield of Be and C gets even larger and amounts 5.3 when the different total area of CFC cladding in JET-C to Be cladding in JET-ILW is considered. The CXN fluxes and the residual plasma flux impinging on the first wall are in both JET PFC configurations similar, a difference in the involved erosion processes is required to explain the difference in the primary impurity source. Indeed the difference can be explained by chemical erosion of C at the lowest, even thermal impinging energies of deuterium. Though CAPS in the case of Be has been observed, the process is different to the thermally activated chemical erosion of C, as a clear energetic threshold at about 10eV energy exists which inhibits the erosion below this minimum required damage energy. The reduction of the impurity concentration in the plasma edge and the primary erosion source is also consistent with the reduction of the C content observed in JET-C in He plasmas. In both cases, the JET-ILW case with D plasma and the JET-C case with He plasma, the fundamental process of chemical erosion at low impact energies is absent and can explain the drop in the impurity content which is then reflected in the corresponding values of  $Z_{\text{eff}}$  in the plasma core ( $Z_{\text{eff}} = 1.2$  in JET-ILW D,  $Z_{\text{eff}} = 2.5$  in JET-C He and  $Z_{\text{eff}} = 2.0$  in JET-C in D plasmas) [9].

The total main chamber Be source in diverted configuration includes also Be eroded from the low field side. However, in contrast to the high field side, no outer wall Be cladding exists which can be bombarded by CXN and residual plasma flux. Thus, the outer wall Be source in diverted configuration is given by the poloidal limiters which are typically 4–8cm away from the separatrix, thus, closer than the inner wall cladding. The measured erosion of limiters, mentioned in the previous section, is partially caused by CXN, by residual plasma flux and potentially by enhanced filamentary transport [18]. An exact quantification of the outer wall Be source is therefore currently not possible, but a lower estimate can be obtained by assuming the same erosion rate as on the inboard Be cladding over the total outer limiter area ( $A_{\text{slim}}^{\text{HFS}} = 9 \text{m}^2$ ) assuming at least similar CXN and residual plasma flux to the inner and outer SOL and ignoring the filamentary transport responsible for the asymmetry. The lower estimate for the total net main chamber source in diverted configuration amounts to  $\approx 21.2 \text{g}$ .

## 5. OVERALL MATERIAL MIGRATION WITH THE JET-ILW

The current understanding of the material migration in the JET-ILW in divertor configuration

can be described as follows (4a): neutral Be and BeD from physical sputtering and CAPS at the recessed wall equipped with Be PFCs enters the plasma, is dissociated, ionised and transported by SOL-flows towards the inner divertor where significant deposition takes place. Indeed post-mortem analysis revealed the majority of all deposition is found on top the apron of the inner divertor on tile 0 and tile 1 which is in all diverted plasmas in the SOL [19] resulting in a total deposition of about 28g. This marks the first deposition location of material driven by the SOL flow (4b) and indeed the WallDYN code can well reproduce this observation [20]. Further transport from this location is strongly hindered as the local plasma conditions provide not enough energetic deuterons to re-erode the deposited Be. This is different to JET-C and C where chemical erosion and multiple step transport of ten or more re-erosion cycles occurred [10]. At the inner strike-line location on the vertical target (4c) the incident Be flux is not sufficient to turn the area into a net deposition zone for Be, but reflection and energetic sputtering by deuterons and impurities in steady-state conditions and in particular during ELMs takes place [9, 23]. As no Be deposition layer is build-up, the W-coated CFC vertical tiles 1 and 3, close to the campaign-averaged strike-line position, is indeed a net W erosion zone as in-situ WI spectroscopy and recent post-mortem analysis confirmed [23]. However, due to the rough structure of the W-coated CFC also Be deposition islands can be identified in W erosion areas. Overall, the Be transport to the divertor floor and even more to the remote areas of the pump-duct entrance is vastly (factor 50) reduced with respect to C in JET-C as measurements and associated ERO modelling confirmed [22]. Chemical erosion of C was responsible for the multi-step transport in JET-C. Indeed in the JETILW, residual C at much lower levels than in JET-C can be found in the inner leg and enhanced at the floor of the divertor as well as in remote areas [23].

The outer divertor is not showing the same deposition pattern as the inner divertor leg. There is a poloidal SOL flow from the outer wall SOL, starting at the stagnation point (seen in JET-C), down into the outer leg, but the amount of Be ions arriving in the outer divertor is insufficient to cause net deposition on top of tile 8. Also the vertical target plates (tile 7 and 8) are not affected and show no sign of erosion or significant deposition. Note that the outer strike line was in the first year of operation predominantly on the bulk W divertor (tile 5) positioned. Tile 5 has not yet been analysed post-mortem, but in-situ W sputtering from the incident Be ion flux was observed and it can be assumed that the W surface is pristine [9].

## SUMMARY

JET equipped with the Be first wall is an ideal test bed for ITER to study Be migration paths and verify plasma-surface interaction code as WallDYN and ERO. In JET plasmas in divertor configuration is the amount of Be eroded in the main chamber (cladding and limiters) and the amount of Be deposited in the inner divertor comparable and lie in the range of 21–28g. This is confirming the main understanding of the Be migration processes and the transport in the SOL towards the inner divertor. The absence of chemical erosion in the case of Be is inhibiting the multi-step transport and avoids the accumulation in remote areas. However, more divertor and main chamber PFCs

need to be analysed before a full balance can be performed. The primary impurity source in JET-ILW is significantly reduced in comparison with JET-C resulting in a reduction in divertor material deposition by more than an order of magnitude [19]. The overall low Be migration is also consistent with the observed low fuel inventory and dust production with the JET-ILW.

## ACKNOWLEDGEMENTS

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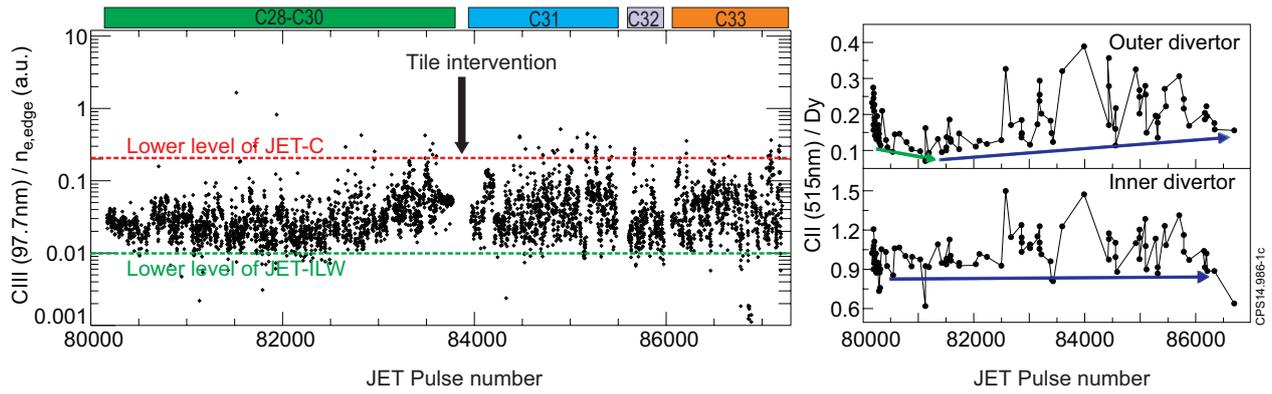


Figure 1: a) C content ( $CII/n_e$ ) in the main chamber plasma edge in JET-C and JET-ILW as function of discharge number. b) Development of the C content ( $CII/D[\gamma]e$ ) in the divertor in identical discharges spread of the full period of operation to monitor the Be and C evolution in time. The lower envelope shows describes the base level of C; the variation show the impact of operation with impurities, leaks and periods without plasma operation on the next discharge.

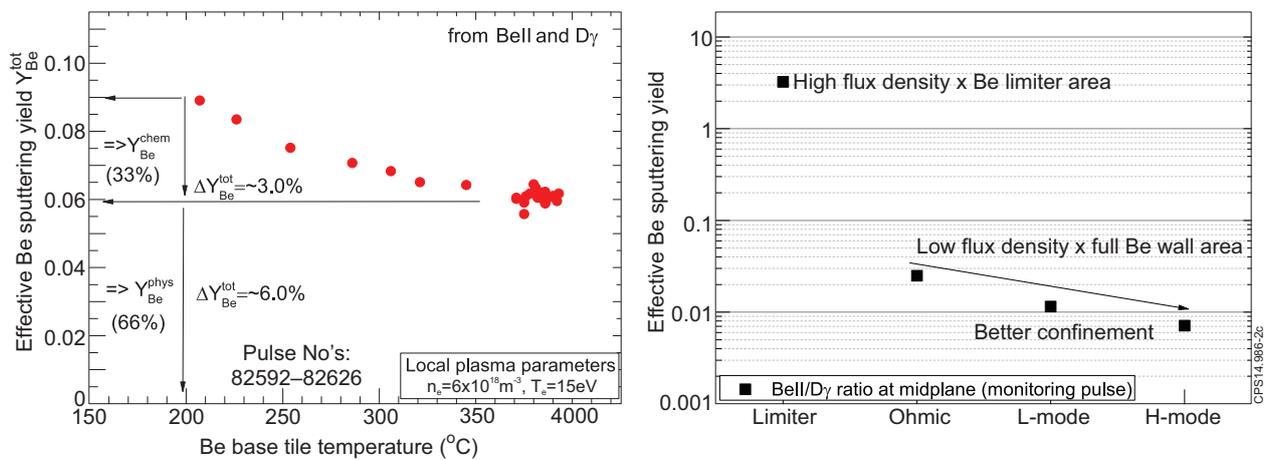


Figure 2: a) Composition of the effective Be sputtering yield as function of surface temperature at constant impact energies in a series of identical discharges. b) Typical effective Be sputtering yield in different operational phases (limiter, ohmic, L-mode and H-mode).

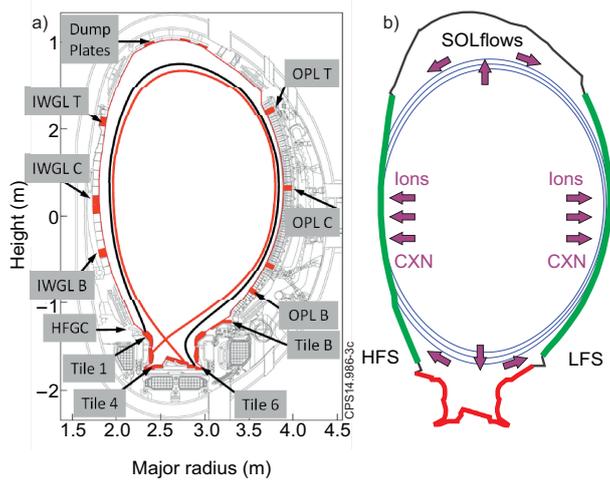


Figure 3: a) Poloidal cross-section of JET with plasma-facing components extracted and analysed by post-mortem techniques in red. b) Be migration path in limited plasmas.

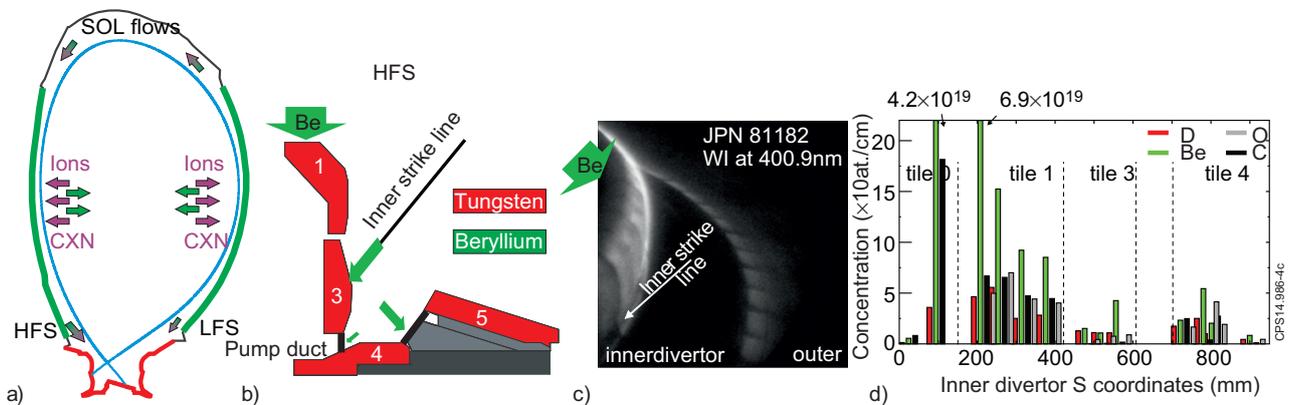


Figure 4: a) Beryllium migration path in diverted plasmas: from the main chamber into the divertor. b) Material migration within in the inner divertor leg. c) WI light emission in the inner divertor leg showing no WI at top of tile 1 where Be deposits and strong WI at the vertical target near the inner strike-point location. d) Be deposition in the inner divertor leg from [13].