Tritium inventory in ITER plasma-facing materials and tritium removal procedures.

Joachim Roth[1], Emmanuelle Tsitrone[2], Thierry Loarer[2], Volker Philipps[3], Sebastijan Brezinsek[3], Alberto Loarte[4], Glenn F. Counsell[5], Russell P. Doerner[6], Klaus Schmid[1], Olga V. Ogorodnikova[1], Rion A. Causey[7]

[1] *Max-Planck-Institut für Plasmaphysik, EURATOM-Association, 85748 Garching, Germany*
[2] *Association Euratom-CEA, CEA/DMS/DRFC CEA Cadarache,*
13108 Saint Paul lez Durance, France
[3] *Institut für Plasmaphysik, Forschungszentrum Jülich GmbH, EURATOM Association, 52425 Jülich, Germany*
[4] *EFDA-Close Support Unit Garching, 85748 Garching, Germany*
[5] *EURATOM/UKAEA Fusion Association, Culham Science Centre, Abingdon, Oxon OX14 3DB, United Kingdom[†]*
[6] *Fusion Energy Research Program, University of California – San Diego, Gillman Drive 9500, La Jolla, CA 92093-0417, USA*
[7] *Sandia National Lababoratory, P.O. Box 969, Livermore, CA 94550, USA*

Abstract:

Interactions between the plasma and the vessel walls constitute a major engineering problem for next step fusion devices, such as ITER, determining the choice of the plasma-facing materials. A prominent issue in this choice is the tritium inventory build-up in the vessel, which must be limited for safety reasons. The initial material selection, i.e. beryllium (Be) on the main vessel walls, tungsten (W) on the divertor upper baffle and dome, and carbon fibre composite (CFC) around the strike points on the divertor plates, results both from the attempt to reduce the tritium inventory and to optimise the lifetime of the plasma facing components (PFCs).

In the framework of the EU Task Force on Plasma-Wall Interaction (PWI TF), the many physics aspects governing the tritium inventory are brought together. Together with supporting information from international experts represented by the ITPA SOL/DIV section, this paper describes the present status of knowledge of the in-vessel tritium inventory build-up. Firstly, the main results from present fusion devices in this field are briefly reviewed. Then, the processes involved are discussed: implantation, trapping and diffusion in plasma facing materials are considered as well as surface erosion and co-deposition of tritium with eroded material. The intermixing of the different materials and its influence on hydrogen retention and co-deposition is a major source of uncertainty on present estimates and is also addressed.

Based on the previous considerations, estimates for the tritium inventory build up are given for the initial choice of ITER materials, as well as for alternative options. Present estimates indicate a build-up of the tritium inventory to the administrative limit within a few hundred nominal full power D:T discharges, co-deposition with carbon being the dominant process. Therefore, tritium removal methods are also an active area of research within the EU PWI TF, and are discussed. An integrated operational scheme to slow the rate of tritium accumulation is presented, which includes plasma operation as well as conditioning procedures.

[†] Current address: Fusion for Energy, Josep Pla, 2, B3 08019 Barcelona, Spain

I.   Introduction:

Issues arising from the interaction of the hot plasma with the vessel walls are important for the design and operation of the international fusion device, ITER, presently under construction. The initial material selection for ITER plasma facing components (PFCs), namely beryllium (Be) on the main vessel walls, tungsten (W) on the divertor upper baffle and dome, and carbon fibre composite (CFC) around the strike points on the divertor plates (see Fig. 1), resulted largely from plasma-wall interaction considerations. In this field, the most critical issues are the lifetime for PFCs and the extent of tritium (T) retention in the vessel.

Extrapolation from present day devices shows that, although erosion under steady state conditions is acceptable, transient events, such as edge-localised modes (ELMs) and disruptions, can strongly limit PFC lifetimes [1,2,3]. The main power load to material surfaces during ELMs will occur at the divertor plates. Both for CFC and W, crack formation due to repetitive ELMs limits the lifetime for ELM energies greater than around 0.5 MJ/m^2. Work is ongoing to design scenarios with tolerable ELMs and to develop disruption mitigation systems.

There is a limit for the in-vessel mobilisable T inventory in ITER, determined by safety considerations. In order to avoid evacuation of a population outside the ITER area, an allowable safety limit for the tritium inventory of 1 kg was imposed, where 120 g are already assumed to be retained in the cryo-pumps. This allowable limit was originally reduced to an administrative limit of 330 g in order to account for measurement uncertainties [4]. Recently, improvements in understanding on the uncertainties and tritium accounting have allowed the administratve limit to be increased. Therefore, in this paper, estimated tritium inventories will be compared to an inventory level of 700 g, which is a likely figure for the eventual administrative limit. Extrapolating from present day devices operating with full carbon walls [5], this limit could be reached within a few hundred full performance discharges (400sec of burning duration, 50:50% of D:T, Q=10). The specific property of carbon to retain large quantities of tritium in deposited layers has led to the decision of ITER [6] to propose a change from CFC divertor strike point tiles to a full W divertor in the activated phase of ITER, with Be as vessel first wall material (the present material choice). Predicting T retention under the ITER conditions, however, remains challenging.

This paper describes the present status of knowledge of in-vessel tritium inventory build-up. Firstly, the main results from present fusion devices in this field are briefly reviewed in section II. Then, the processes involved are discussed in section III, including: implantation, trapping and diffusion in plasma facing materials as well as surface erosion and co-deposition of tritium with eroded material. The effect of plasma facing materials mixing on retention is also addressed.

Based on the previous considerations, estimates for the tritium inventory build up are given in section IV for a variety of plasma facing material combinations, including the initial choice of ITER materials and a full-W divertor with Be walls, as well as for all-C and all-W machines, for comparison. Finally, tritium removal methods are discussed in section V with an integrated operational scheme proposed to slow the rate of tritium accumulation.

II Observations on fuel retention in present devices

The problem of H-isotope retention due to co-deposition has been recognised since at least the 1980's, e.g. in deuterium experiments in JET [7] and TFTR [8]. The severity of the problem of fuel retention for next-step devices, however, was only clearly evidenced after the first T experiments in TFTR [9] and JET [10]. About 5 g and 36 g of T were injected into TFTR and JET, respectively, from which significant amounts (51 % and 35 %, respectively) were retained in the vessel at the end of the campaign [10,11]. In TFTR, the retained amount was reduced to 16 % by various detritiation procedures. For the JET DTE1 campaign, the retained fraction was reduced from 35 % at the end of the campaign to 10 % after intensive cleaning (H, D, He plasmas, disruptions, glow discharge cleaning) and venting of the vessel. Out of the 10 % of T remaining in the vessel after venting, 2 % was traced in 'post mortem' analysis of PFCs (0.2 g in divertor and main chamber tiles, 0.5 g in 150 g of hydrocarbon flakes found in the divertor louvers areas). The remaining 3 g of T are assumed to be in the flakes in sub-divertor, not accessible for analysis (1 kg of flakes is needed to account for the missing T, when extrapolating from the 150 g recovered and analyzed).

These observations are well in line with findings for devices operating in deuterium, where a gas balance (comparing injected and exhausted deuterium) performed on a single discharge yields significant retained fractions [5] (20 % in JET [12], up to 50 % in Tore Supra and C-mod [3,13]). Particle balance integrated over a campaign including a large variety of discharges,

disruptions and wall conditioning between shots yields a much lower long term retention between < 1 % (C-mod) and 10 – 20 % (AUG and Tore Supra), while '*post mortem*' analysis of PFCs yield values in the few % range (8 % for TEXTOR [14], 3 % for ASDEX Upgrade [15]). Results from long pulse machines with actively cooled PFCs, such as Tore Supra, show that the retention does not saturate but maintains a constant rate throughout the discharge [16] or even over a series of identical, long discharges [17].

In most of present day tokamaks with low Z PFCs, long term fuel retention is dominated by co-deposition, namely material erosion, migration and re-deposition, with the formation of D/T-rich co-deposited layers, as seen in JET [18,19] and TFTR [11]. In the divertor machines, co-deposition occurs mainly on the inner divertor surfaces and shadowed areas of the outer divertor and limiters.

For high-Z materials, tritium implantation and bulk trapping is the dominant retention process, leading to a significantly reduced inventory in comparison to co-deposition with carbon. Recent experiments in ASDEX Upgrade, progressively replacing its C PFCs by W, have shown a reduction of the carbon concentration by 70 % whilst the main carbon sources are at the outer poloidal limiters in the main chamber and the divertor plates [20]. Experiments with a full W machine have been undertaken since 2007 and early '*post-mortem*' analysis has shown a dramatic reduction in retention of all deposited layers in the machine [21]. This is in agreement with gas balance measurements in dedicated discharges, yielding retention rates below 3% of the injected fuel. In Alcator C-mod, using Molybdenum as the first wall and divertor material, deuterium retention was studied in campaigns using boronisation [22] and cleaned Mo tiles [3]. After boronisation campaigns, D retention was found to correlate with the boron deposited layers, leaving relatively small inventories on erosion dominated areas. For cases with cleaned tiles, gas balance measurements showed retention of a large fraction of the injected gas (up to 50 %) in dedicated high density discharges, while '*post-mortem*' investigations showed very little retention (< 1 %). This discrepancy is not currently understood. Possible reasons may be that retained D is released after the plasma exposure terminates, possibly over a long period. This transient retention occurs when D plasma is incident onto materials in which D can diffuse, both in and out. This is may be supported by modified release properties resulting from B remaining after boronisation intermixed into the first few μm of the Mo surface by diffusion processes. Another

explanation can be outgassing of D throughout the campaign in discharges ending in disruptions, where gas balance measurements do show a burst of released D.

General observations concerning the balance of material erosion and re-deposition (linked with co-deposition) are the following: the main wall is a net material source, the inner divertor is a net deposition area, while the outer divertor is mainly under net erosion, depending sensitively on plasma conditions [see, e.g. [23]]. Translation of the current experience to ITER suggests the following pattern:

- Significant Be-erosion at the vessel wall with corresponding deposition covering the inner CFC divertor tiles and leading to co-deposition of fuel with deposited Be. Recent modelling indicates that a significant part may also be deposited onto the main wall, in particular near the divertor entrance [24],

- The situation on the outer divertor is more difficult to predict, as the plasma conditions are quite different in current devices and ITER (attached vs. partially detached plasma, respectively). However, unless large impurity influxes from the vessel take place, net erosion is expected,

- Carbon divertor erosion leading to co-deposition on the lower part of the vertical divertor target, on areas in line-of-sight of divertor walls (dome surface and remote areas below the dome) and in gaps is thought to be the main long-term tritium retention mechanism.

III Investigation of PWI processes involved in retention for fusion relevant materials

The two main mechanisms involved in retention are presented below, implantation being the dominant channel for high Z materials (W) and co-deposition for the low Z materials (C, Be).

III.1 Implantation

Implantation and retention of low-energy hydrogen ions into pure materials, such as carbon [25,26], tungsten [27] and beryllium [28,29], have been investigated in detail. Recently, some of the

specific material structures proposed for application in fusion devices were also considered (e.g. CFC [30] and vacuum plasma-sprayed (VPS) W [31]).

In Be and pyrolytic or fine grain graphites with low porosity, hydrogen does not diffuse and after reaching a local concentration in the implantation range of about 30 at %, further hydrogen is re-emitted [29,32,33]. The behaviour is different, however, in more porous materials [34], like CFC [30], considered for ITER. Recent coordinated experiments at high fluence in ion beams, plasma devices and tokamaks have shown no saturation of the total implanted amount as a function of fluence. Instead, the retained amount increases close to a square root of the ion fluence (see Fig. 2) due to diffusion deep into the bulk. This inward diffusion shows a very low activation energy and details of the transport process are poorly understood [30]. Combined micro-beam nuclear reaction analysis, TEM and electron energy loss spectroscopy has revealed the pores and cracks between fibres and the CFC matrix to be covered with D-rich amorphous hydrocarbon (a-C:H) layers [35,36]. This could have implications for long pulse, high fluence machines such as ITER, and probably has a role in explaining the retention rate observed in the long pulses of Tore Supra [16]

In W, deuterium is highly mobile and is only retained in radiation damage sites or defects of the crystal lattice [37,38]. After saturating available traps in the ion induced damage profile, inward diffusion and subsequent trapping at lattice defects increases the trapped inventory. The retention depends on the crystalline structure of the substrate, and increases from single crystals to well annealed polycrystalline material and can reach high values in plasma-sprayed W coatings [31] due to the increased trap density at grain boundaries and pore surfaces. For a purely diffusion limited build-up of the inventory, a square-root dependence on the fluence would result. The observed dependence as fluence$^{0.7}$ is due to an increase in ion-induced defects and hydrogen trapping sites. As seen in Fig. 2, this stays in tolerable limits for polycrystalline W in ITER provided that neutron damage effects in the bulk of the material remains small (see next paragraph).

The data for retention in Be, C and W as a function of fluence, obtained in laboratory experiments, are summarised in Fig. 2 for implantation at 300 K [39]. Estimates for the fluence in one ITER discharge are also shown for comparison, for Be and W components at a flux of $(5x10^{22}$ (D+T)/m^2) as well as for CFC divertor strike point tiles $(5x10^{26}$ (D+T)/m^2).

Influence of radiation damage due to n-irradiation

For the case of W PFCs, where implantation and trapping in the bulk are the dominant processes leading to tritium inventory build-up, additional lattice damage sites due to n-irradiation will be important. Already in ITER, but especially in DEMO [40], the fuel retention properties of W plasma-facing materials may be enhanced due to radiation damage after high fluence n-irradiation, which provides additional trapping sites for hydrogen [41]. The irradiation damage at the end of the ITER lifetime has been estimated to 0.6 dpa in the divertor and 1dpa at the first wall [42], but the microstructure and its relation to hydrogen trapping is largely unknown. Taking into account a saturation of damage sites at 1 % after 0.6 dpa as reported for Mo [43] the additional trapping sites due to neutron damage might lead to a reduction of the number of discharges before reaching the T limit to 5000 [44]. Simulations using the DIFFUSE code [45], assuming a constant value of 0.6 % n-induced trapping sites, deduced very similar retention values (Fig. 3). In these calculations, no ion-induced trap generation has been taken into account due to the very shallow implantation depths, leading to a retention increase only with the square-root of fluence. Consequently, a value of 700 g retained tritium will be reached only after a much higher number of discharges compared to the fluence$^{0.7}$ dependence.

A saturation concentration of n-produced traps of 1 % in W is an extreme upper limit and probably 0.1% is a more realistic value for ITER. On the other hand, the density of n-induced vacancies will decrease with temperature by increasing the spontaneous annihilation and vacancy clustering. As such effects are not taken into account, the estimation gives an upper limit of T retention and needs to be refined. In spite of the coarse inclusion of the n-damage effect in the present modelling, both assessments show remarkable agreement. In both cases the un-irradiated W retention is very similar, being dominated by the divertor areas at moderate fluxes and temperatures. The additional n-damage enhances the retention by up to a factor of 8. Clearly, this effect requires more experimental validation and more detailed code simulation before a final conclusion can be drawn.

III-2 Co-deposition

Erosion of plasma facing materials

The first step in the chain of processes leading to co-deposition is the erosion of the wall material.

In the keV energy range, **physical sputtering** of Be and W are well described by theory [46,47] and well reproduced by Monte Carlo modelling [48], while in the range below 1 keV, especially for light ions, threshold effects have to be considered [49,50]. The situation is different for carbon based materials. In this case the chemical interaction with the hydrogen plasma leads to enhanced erosion yields [51]. **Chemical erosion** is a complicated, multi-step process recently reviewed in [52]. The main features, important for the co-deposition chain, are:

- Maximum yields at elevated surface temperatures (around 10^{-1} at 600-800 K).

- Emission of hydrocarbon molecules and radicals with different sticking properties on surfaces.

- A flux dependence for fluxes above 10^{22} D/m^2s decreasing the chemical erosion yield to values below 10^{-2} [53].

Critical, and insufficiently known parameters are the threshold behaviour [54,55], the contribution of heavier hydrocarbons and radicals [56], the influence of the chemical reactivity due to simultaneously incident energetic impurity ions [57,58] and the origin of the flux dependence. However, the erosion yield due to chemical erosion is adequately described by an empirical set of equations [53].

Re-erosion of deposited layers

A major effect in material migration is re-erosion of the deposited layers. Modelling of erosion/deposition experiments indicates that re-deposited hydrocarbon layers in tokamaks are subject to an "enhanced" chemical erosion with yields 10 times higher than for bulk graphite [59]. This is in agreement with laboratory experiments showing much higher erosion yields for soft, hydrogen-rich a-C:H layers when compared to graphite [60]. In addition, the rate for re-erosion of

a-C:H layers is temperature dependent, as demonstrated in laboratory experiments [61]. The combination of all the complex processes above leads to hydrocarbon deposition on low temperatures surfaces and erosion at high temperatures [62,63].

For deposited metals, such as Be, most evaluations use the same erosion yield as for bulk material. However, in recent laboratory deposition experiments with Be [64] an erosion yield enhanced by a factor of 2 was observed.

Co-deposition of hydrogen atoms

The final step for co-deposition is the incorporation of hydrogen in deposited layers located in remote areas, not seen by the plasma for further re-erosion. Impurity atoms or molecules are deposited together with a flux of energetic or thermal hydrogen atoms. A collection of data on the deuterium concentration in C, Be and W deposits is shown in Fig. 4 [65].

For carbon deposited layers, the hydrogen concentration depends critically on the energy of the incident hydrogen flux. Energetic ions lead to the deposition of hard films with hydrogen concentrations H/C of about 0.4, while low energy thermal hydrogen leads to the formation of soft films, with H/C concentrations exceeding 1 [66,67].

Similarly, recent analysis [68] shows structural changes in deposited Be layers for different energies of incident deuterium atoms and deposition rates, leading to low hydrogen content for low energies and high Be concentrations in the incident flux. The role of oxygen in Be-rich co-deposits, which was originally thought to have a major impact [69], does not appear to play as large a role as the temperature of the layer and the energy of the incident particles [70].

Little data exists for tritium co-deposition with W and the values are close to or below the detection limit of the measurements [69,65]. These low values combined with the very low erosion yields of W for both fuel particles and impurity ions leads to the conclusion that co-deposition with W or WC will not be a critical process for ITER.

As is seen from above, predicting T retention in ITER is subject to large uncertainties, as local deposition conditions are difficult to assess: power and particle flux on the complex 3D geometry of PFCs, including gaps, composition of the incident flux in terms of fuel particles and impurities, local surface temperatures dependent on the poorly characterised thermal properties of

deposited layers. However, the next section presents rough estimates based on state of the art knowledge.

IV Estimate of the ITER tritium inventory build-up

Previous attempts have been made to estimate the T inventory build up in ITER [71,72,73], following methods with different levels of sophistication.

A simple scaling, based on a long term retention fraction of 10% of the injected fuel, as obtained in today's devices, leads to a very limited number of discharges (< 100) before reaching the safety limit (assuming 200 Pam^3s^{-1} of gas injection with a 50-50% mixture of D-T) [5]. However, these results are based on all-C devices (so should overestimate the retention in ITER with only a C divertor, if the inventory is dominated by co-deposition with C). Moreover, the physical basis for such an extrapolation is rather weak (considering the processes involved, the retention rate should scale as the recycling flux on PFCs rather than as the injected flux, although these are often correlated).

On the other hand, detailed modelling of divertor erosion/deposition with sophisticated edge transport codes, taking into account Be and C deposition, have been performed. Most recent simulations reach the limit after 100 to 200 discharges, depending on assumptions for re-erosion of the deposited layers and Be concentration in the incoming divertor particle flux [59]. For low Be concentration (0.1%), the T inventory is found to be dominated by co-deposition with eroded C, whilst for higher Be concentration (1 %), the retention rate is higher and is dominated by co-deposition with eroded Be.

Therefore, an intermediate approach, taking into account the different processes listed in section III under realistic ITER conditions, was followed by the EU PWI TF, allowing in particular comparison of different materials choice options. The results are presented below, with the description of the input parameters, the processes taken into account, the result for the initial ITER material choice and a comparison with other material choices options.

IV-1 Input parameters for the estimates

Input plasma particle and energy fluxes, as well as surface temperatures, are taken for a reference Q=10 ITER discharge as evaluated in [74]. The resultant fluxes are illustrated in Fig. 5.

In the divertor near the plasma strike point the typical plasma/neutral fluxes reach values larger than 10^{24} m^{-2}s^{-1} (leading to a total fluence $> 10^{26}$ m^{-2} for each ITER pulse) with plasma densities ~10^{21} m^{-3} and plasma temperatures of ~ 3 eV. This corresponds to a D$^+$ impact energy of ~15 eV, due to acceleration in the plasma sheath potential. The expected surface temperature around the strike points is ~ 1300 K on the outer divertor, ~ 800 K on the inner divertor [75].

The estimates of **first wall** fluxes are more uncertain.. The modelling results used here indicate that the D neutral flux is in the range of 10^{19}-10^{21} m^{-2}s^{-1} with typical energies ~ 8 – 300 eV, as shown in Fig. 5 [74]. The peak neutral particle fluxes correspond to sources such as the gas injection at the top of the machine (~8 m on the x axis), or recycling from the inner (~0 m) and the outer (~18 m) divertor. They are associated with a decrease in the mean energy, due to the contribution of the cold recycling neutrals from the source. The energy of the neutrals hitting the wall peaks at the plasma edge temperatures, but has contributions up the central plasma temperature [76].

The ion fluxes have their highest values close to the strike points of the separatrix on the divertor plates, with more than 3 orders of magnitude lower values for the baffle and wall areas. The ion flux to the first wall as calculated above lies in the lower range of more recent estimates, taking into account long range transport across the SOL [77]. These estimates do not yet provide a consistent poloidal distribution, but indicate that the integral wall fluxes may be a factor of 3 ± 2 higher, while the divertor fluxes remain similar. With this caveat the evaluations in this paper are based on the complete scenario from ref [74], assuming a D/T ratio of 50/50 %. For the comparison of different material options such uncertainties in the wall particle fluxes for the ITER reference $Q_{DT} = 10$ scenario are taken into account to first order by using the results in [74] scaled appropriately to produce a total ion flux to the first wall between $1 – 5 \times 10^{23}$ s^{-1}, which is in line with present empirical scalings of this parameter to ITER [77].

IV-2 Retention processes taken into account

Contributions from the processes listed in section III have been evaluated in the following way, combining extrapolation from experimental data with modelling results when available:

- T implantation in CFC: Extrapolation from laboratory experiments [30] are evaluated from the data in |30|, assuming a square root fluence dependence. It should be noted that this was extrapolated from the highest measured fluences by 5 orders of magnitude to reach ITER relevant fluences. A very recent report of the saturation at fluences above $10^{26}/m^2$ may considerably reduce the expected contribution from implantation in CFC [78].

- Co-deposition of T with eroded C: The contribution due to carbon-tritium co-deposition has been calculated using the ERO code [72] on the basis of the ITER plasma scenario with given surface temperatures, including divertor erosion, plasma transport and re-deposition in remote areas. All eroded carbon atoms not re-deposited on the divertor plates are assumed to lead to co-deposited tritium layers. The sticking coefficient at the divertor plates is assumed to be 0 for hydrocarbons and 1 for carbon atoms. The D/C ratio assumed for the calculation is $(D+T)/C = 0.4$. The potential reduction in chemical erosion of the CFC divertor targets due to intermixing with deposited Be is not taken into account here.

- T implantation in Be: All present data show saturation of the retained amount of deuterium within the penetration range of the ions, without diffusion into the volume of the material [29].

- T co-deposition with Be: The co-deposition of tritium with eroded Be has been obtained by DIVIMP calculations of the erosion from the vessel walls [79]. All eroded Be is assumed to be deposited in line-of-sight to plasma-facing materials, as has been observed thus far on JET [80], with $(D+T)/Be = 5$ %. This value is taken as an approximate average between lower values (< 1 %) at the hot divertor plates and higher values (8%) expected at the baffle surfaces.

- T implantation in W: For a W vessel wall, recent implantation data of low energy deuterium (200 eV D^+) are taken [38] without extrapolation. The temperature dependence of deuterium retention shows a maximum at increasing temperatures for increasing ion fluxes (Fig. 6). For the temperature dependence, the maximum values at the different flux levels were taken (dashed line in Fig. 6). For the use of W as a strike point material in ITER the extrapolation of experimental data to particle fluxes at least 2 orders of

magnitude higher requires the use of modelling codes [37, 38]. These codes are also used to estimate the effect of additional traps for tritium generated by n-irradiation throughout the bulk of the material (see section IV-4).

- T co-deposition with W: Co-deposition of T with W was not considered on the basis of the low erosion yields as well as the low tritium concentration in co-deposited layers [65,68].

IV-3) Results for the initial ITER material choice

Figure 7 shows the expected tritium inventory build-up as a function of cumulated discharge time for the initial material choice of ITER and indicates the different contributions to the inventory. The surface area and the tritium particle fluxes taken into account for estimates of implantation are also indicated for each material:

- 3 m^2 at 2x10^{24} (D+T)/m^{-2}s^{-1}, 775 K and 47 m^2 at 2x10^{23} (D+T)/m^{-2}s^{-1}, 500 K for the CFC divertor. Actually, strike point temperatures may be higher leading to reduced inventories.

- 100 m^2 at 2x10^{20} (D+T)/m^{-2}s^{-1}, 400 K for the W baffles and dome

- 700 m^2 at 1.4x10^{20} (D+T)/m^{-2}s^{-1}, 400 K for the Be first wall

The experimental data on which the extrapolation is based for implantation in CFC, W and Be are also shown. The potential reduction of the chemical erosion yield of CFC due to Be intermixing has not been taken into account.

From Figure 7, it is clear that CFC in the divertor of ITER would be the dominant source for the T inventory. After 100 discharges, the contribution of implantation in CFC (square root dependence with fluence) becomes negligible compared with co-deposition (linear dependence with fluence), assuming that the square root dependence observed in laboratory experiments is correct. The 700 g T level is reached in about 750 discharges, the rate of T retention being in rough agreement with previous estimates [72,73]. Be co-deposition contributes a factor of 6 less to the inventory than carbon, while retention in un-irradiated W would reach the inventory limit only after 250000 discharges.

IV-4 Comparison of different material options

Fig. 8 compares the T inventories for the 4 following options:

- initial material mix, i.e. CFC strike point tiles, W divertor and Be first wall

- all-C device,

- all-W device,

- Be first wall/W divertor device.

For this comparison uncertainties in the particle fluxes to the wall between $1 - 5x10^{23}$ (D+T)/s are indicated by the coloured band, for each wall material option.

As discussed in section IV-3 the tritium inventory for the initial material choice, CFC/W/Be, will build up mainly due to co-deposition with carbon, supplemented by co-deposition with Be, and will reach the 700 g T level within 250 to 750 full 400 s Q = 10 discharges.

The option of using an all-metal W/Be machine has been proposed by ITER as the initial material choice in the activated phase of ITER. It results in a strongly reduced T build up compared to the initial material choice. Between 1500 to 5000 discharges are necessary to reach the 700 g T level, which is now dominated by co-deposition with Be, mainly at the inner divertor.

For a hypothetical all-C device, T co-deposition has been calculated using the ERO code [73] assuming an additional influx of 1 % C ions into the divertor. The resulting inventory agrees well with extrapolations from JET by scaling with the ion fluence to the divertor [81]. The global C influx, responsible for co-deposition was calculated as $6x10^{21}$ C/s [73]. Clearly, in an all-C device, the 700 g T level would be reached in a few tens of discharges and require very frequent cleaning interventions (for consequences see Section V on T removal). An all-C device could only be considered, if the wall temperatures of areas where co-deposition occurs could be raised above 900 K, which results in considerably lower T concentrations in deposited layers.

Extrapolation for a hypothetical all-W device takes into account three different areas with differing ion flux and temperature in the divertor, baffle and first wall. The temperature of the wall is assumed to be around 400 K, while it is taken at 775 K in the divertor area. As expected,

the lowest T retention is obtained for the all-W device, with the inventory staying below the 700 g T level for > 18000 discharges (excluding n-damage effects). The use of W in the divertor requires the extrapolation of experimental data over more than 2 magnitudes by computer modelling and numerical results are given in Fig. 8. The results show that the divertor strike point area, as well as the vessel wall areas contribute little to the inventory, because of the high temperatures and low ion fluxes, respectively. The main inventory will be built up in the divertor areas with intermediate flux (2×10^{23} (D+T)/m^{2}s) and intermediate temperature (725 K). The accumulation of n-damage with increased tritium bulk retention may reduce the number of discharges to reach the 700 g tritium level to about 5000.

While further research may improve the precision of the predictions, it seems clear that with the use of carbon as divertor material and/or first wall material the T safety limit will be reached between a few hundred and, at most, a thousand discharges, after which operation would have to stop for the vessel to be cleaned and tritium to be recovered. The present proposal to eliminate carbon in the activated phase and install a full W divertor brings the number of discharges to reach the 700 g T level close to the expected lifetime of the divertor PFCs, after which the divertor cassettes will be exchanged and tritium removed. Also in this case it appears prudent to develop cleaning methods and test them under realistic fusion conditions. In any case, it would be sensible for the machine design to allow sufficient flexibility not only to exchange the divertor cassettes but also the first wall PFCs material.

V. Mitigation and removal of tritium inventory

It is clear from the preceding discussion that the in-vessel inventory limit in ITER could be reached during the planned operational period unless schemes for tritium recovery, or for mitigating the retention of tritium, are implemented. The issue is especially acute where carbon (as CFC) is included in the mix of plasma facing components, but exists at some level for other material mixes. A regime of controlled operations (so called 'good housekeeping' [82]), with a spectrum of recovery techniques, is likely to be the best strategy for managing the tritium inventory (Fig. 9).

For example, careful attention to optimising the tritium fuelling efficiency (minimising the tritium introduced), tailoring the isotope ratio [83] and ending every shot in a deuterium-only

phase with sweeping of the divertor strike-points [10] have all been proposed. These simple and potentially very effective approaches for mitigating retention could be employed on every shot but need a more systematic validation in present fusion devices. Seeding the divertor plasma with nitrogen may also help reduce retention rates, either by increasing the chemical erosion of tritium-containing hydrocarbon co-deposits [84] or by reducing the availability of hydrocarbon radicals, which suppresses co-deposit formation [85,86]. This so-called nitrogen scavenging effect may even reduce co-deposit formation in shadowed regions of the divertor far from the plasma [87], although issues such as nitrogen leakage out of the divertor and possible collateral damage effects have yet to be fully assessed for ITER conditions. Conditioning techniques, such as low pressure RF discharges in deuterium could further contribute to reducing the rate of tritium accumulation. Although the efficiency of this type of technique would appear not to be very high [10], it has no deleterious impact on the vessel conditions and can therefore be employed frequently (i.e. inter-shot).

For the initial ITER materials mix of carbon (CFC), beryllium and tungsten plasma facing components, a combination of these relatively non-invasive techniques, which could be smoothly integrated into operations, might reduce the tritium accumulation rate by a factor 2 – 3 at best. Other, more efficient techniques would therefore be required periodically. The options, however, for rapid and efficient de-tritiation of hydrocarbon deposits are rather limited. Thermal desorption by bulk heating of PFC's would require unrealistically high temperatures (~1100 K [88]), and other techniques are either too slow for practical application or would require vessel entry. Baking in oxygen at high pressure (e.g. 0.1 bar or above) appears to represent the only credible, in situ technology at the moment for dealing with tritiated hydrocarbon co-deposits. It is possible that sufficient tritium recovery could be accomplished by baking of the divertor alone (since this is where the bulk of co-deposit formation is likely to occur) – although the bake-out temperature would need to exceed 573 K (ideally at least 623 K) for a reasonable recovery rate [89,90]. Even then, the bake would last 1 – 2 days, with a further 3 – 5 days for recovery of plasma conditions. Heavy enhancements to both the ITER divertor bake-out system (to access the required temperatures) and the tritium plant (to handle oxidation products including T_2O and DTO) would be required.

The technology installed to allow divertor oxygen bake-outs could also be used to good effect for thermal desorption of beryllium co-deposits, which is efficient and rapid at 580 - 620 K

[91,92,93]. Beryllium co-deposits are also expected to form preferentially in the divertor region and to trap about 1/8th of the tritium in the hydrocarbon co-deposits. De-tritiation of these by a relatively short vacuum bake-out (a few hours + preparation and recovery time of perhaps a day) would help significantly to extend ITER operations for the required period – especially since not all the hydrocarbon co-deposits will form in the divertor (and thus divertor oxygen baking would not be 100% efficient). Divertor vacuum baking could, in fact, be the principle tritium recovery scheme for ITER with a mix of only beryllium and tungsten PFC's, where divertor beryllium co-deposits will dominate retention. For this materials mix, the comparatively low rate of accumulation expected would probably permit relatively infrequent baking, although it may be prudent to plan for this eventuality once or twice during the operational phase (i.e. plasma operations between planned maintenance phases during which more major interventions would be allowed. This could be up to around 2500 shots).

Following the operational phase, it is possible that the tritium inventory not recovered by divertor vacuum and oxygen baking (e.g due to implantation and migration in CFC or hydrocarbon and beryllium co-deposits outside the divertor) could have reached close to the limit. During the maintenance phase, it may therefore be necessary to employ other, more invasive recovery approaches. Surface heating using scanning lasers [94,95] or flash-lamps [96] under remote handling may be an important option during this phase, for example to remove tritium accumulation due to CFC implantation and migration (which needs higher temperatures than are easily accessible with baking, up to ~ 750 K [97,98])

If only tungsten PFC's were present in ITER, the situation would be rather different. Retention would be largely by implantation and diffusion in the large area first wall, rather than in the divertor. Again it would be prudent to allow for removal during the operational period – although the estimates provided in section IV-4 and figure 8 suggest that the inventory will not actually be reached. Fortunately, much of the retained tritium may be relatively easy to recover through thermal desorption – by baking the first wall to 500 – 550 K for ~1 h [27] (readily accessible with the currently planned bake-out system). It should be noted, however, that the experiments described in [27] were conducted for unirradiated tungsten and the impact of n-damage (e.g on trap energies) on the efficiency of thermal desorption is uncertain. During the maintenance phase, it would be sensible to address retention in high heat flux regions of the divertor. This could be quite significant and, due to high temperatures in this region during

implantation, similarly high temperatures would be required for thermal desorption [27]. Baking to >>1000 K is impractical even during a maintenance phase and pulsed surface heating with lasers or flash lamps may again be the preferred option.

Although laboratory tests and preliminary tokamak experiments have been performed, all the proposed mitigation and removal techniques described above need further assessment and validation in present fusion devices. None of the techniques in this section, including those used in the example 'good housekeeping' scheme indicated in figure 9, are yet accepted for integration into ITER operations. Most of the technologies would require some level of modification to existing ITER systems.

VI. Summary

The different processes involved in fuel retention for the initial choice of material for ITER (CFC/W/Be) have been reviewed. The T inventory build up in the vessel has then been estimated, using extrapolation from experimental data and modelling where available. These estimates have been made for a steady-state plasma with many strongly simplifying assumptions. ELMs and off-normal events will increase the surface temperature of the divertor and, thus, increase the temperature gradient to the surface that will result in less T retention in bulk metals. Large uncertainties arise, especially for a mixed material device, from complex processes such as material intermixing with the formation of new phases and alloys. The detailed geometrical structure of the divertor plates and vessel walls is not taken into account and retention in gaps may add to the global inventory. Still, the present assessment allows a comparison of different material options. The initial ITER design has been assessed in terms of T inventory and other material options have been compared. The main conclusions are summarized below.

Initial ITER choice of materials:

- Co-deposition with eroded carbon from the divertor plates will dominate the tritium inventory for the initial material choice (0.5 – 2 g tritium per shot).

- Tritium co-deposition with eroded Be from the vessel wall is more uncertain due to larger uncertainties in the wall fluxes, but could contribute significantly to the total inventory (0.1 - 0.6 g tritium per shot).

- Implantation of T in W is not limiting, although n-irradiation after years of D/T operation will increase this (estimated range: non linear with fluence, < 0.05 g T /shot)

Comparison with other materials options:

- Initial material choice: Tritium limit is expected to be reached after about 500 discharges.

- all-C machine: Tritium limit is expected to be reached < 150 discharges

- all-W machine: The limit will be reached after > 15000 discharges without n-irradiation and > 3000 discharges with n-irradiation levels reached at the end of ITER lifetime. The effect of n-irradiation is highly uncertain and requires much more experimental and modelling work

- the W divertor/Be first wall option, presently foreseen for the activated phase of ITER, relaxes the tritium issue considerably, extending the number of discharges until the tritium limit is reached to about 3000. Still, removal methods for co-deposited Be layers must be developed and tested.

Mitigation and removal methods:

- No single method is sufficient where C is present – a 'good housekeeping' approach, with a spectrum of tritium recovery techniques, represents a possible approach

- Oxidative schemes play a key role in this approach, but would require significant modifications to the tritium plant (to handle volatile products) and bake-out system (to access the required temperatures), at least in the divertor

- Thermal desorption is the best approach for tritium trapped in beryllium co-deposits or retained in bulk tungsten. Modifications to the divertor bake-out system would again be required to access the required temperatures.

- Photonic cleaning – surface heating using flash-lamps or scanning lasers – could be an important technique in shutdowns for recovering tritium outside the divertor or where temperatures higher than the bake-out are required.

Figure captions:

Figure 1. Cross section showing the layout of PFCs in ITER with different armour materials. In the initial phase the divertor target plates shown in red are made of carbon (CFC), the upper baffles and the dome shown in green are made of tungsten, while the first wall shown in blue is made of beryllium. In the activated phase the CFC divertor plates are replace by tungsten.

Figure 2: Retention of deuterium in beryllium, carbon and tungsten as function of ion fluence [from 39]. The data represent retention in materials not previously irradiated with neutrons.

Figure 3: Effect of n-irradiation damage on the tritium inventory in W divertor plates. Results from code calculation by Ogorodnikova [38] are shown in red, results from the DIFFUSE code [37] in blue.

Figure 4: Retention deuterium concentration in C, Be and W deposits under co-deposition conditions [65]

Figure 5: Plasma temperature T_e and T_i, D/T ions and neutrals particle fluxes and average energies, for a semi-detached divertor plasma for the ITER Q=10 reference scenario. The reference distance along the wall circumference starting in the private flux region is shown in the insert. [74].

Figure 6: Temperature dependence of deuterium retention for different ion fluxes normalised to a fluence of $2x10^{24}$ D/m^2 using a square-root dependence with fluence. Data are taken from literature [39]. The dashed line shows the temperature dependence assumed for the extrapolations to ITER conditions.

Figure 7: Estimated tritium inventory as function of the number of full, 400 s ITER D/T discharges for the initial choice of wall materials for the different retention processes (please note the log scale). Data points give retention results from ion beam experiments [39]. No neutron damage of the materials is considered.

Figure 8: Tritium inventory in ITER for the all-C (blue line) and all-W options (red line) compared to the initial material choice (magenta). In addition, retention values for the option of a full-W divertor and Be first wall are included (black line). The assessment was performed assuming different particle fluxes to different divertor and wall areas:
divertor: $3m^2$, $2x10^{24}$ $(D+T)/m^2s$, 775 K; $47m^2$, $2x10^{23}$ $(D+T)/m^2s$, 500 K;
baffle and wall: $750m^2$, $1-5x10^{20}$ $(D+T)/m^2s$, 380-440 K.

Figure. 9: Example of a 'good housekeeping' approach to tritium inventory control. Panels show tritium accumulation for the initial ITER materials mix (at a mid-range value of 1g/400s D:T shot) and mitigation or recovery on timescales from one shot to a whole campaign (red – no recovery, blue – with recovery). A variety of techniques are hypothesised for use during the shot, inter-shot, daily, weekly and quarterly, with corresponding recovery rates (based on those achieved in tokamak and laboratory experiments). Without this approach, a 700g T inventory level would be reached in around 3 months.

Figure 1.
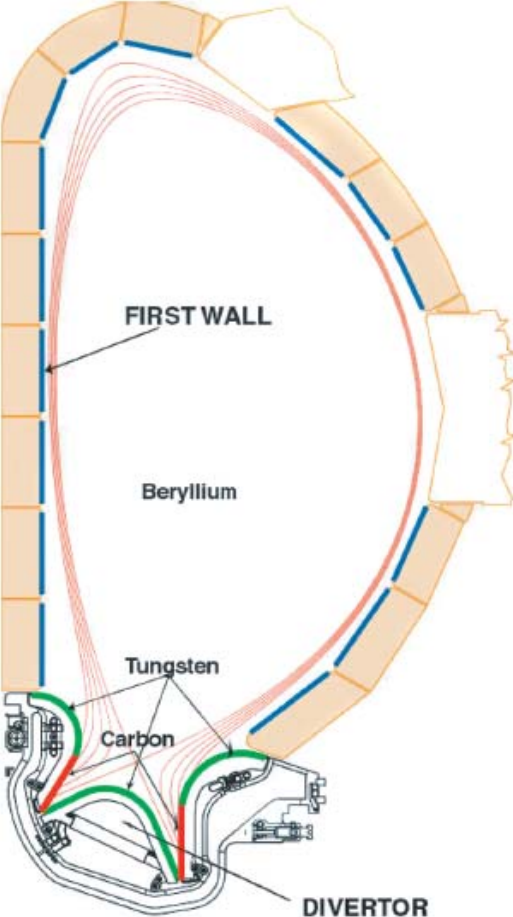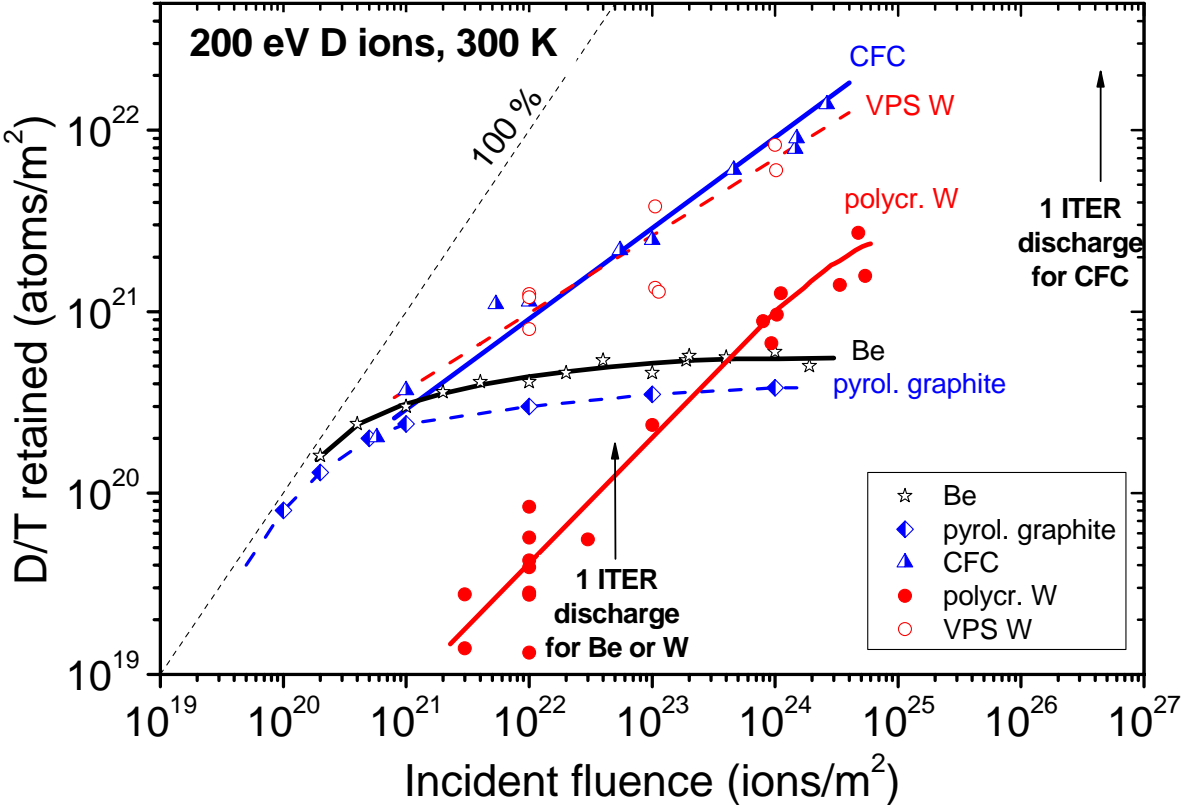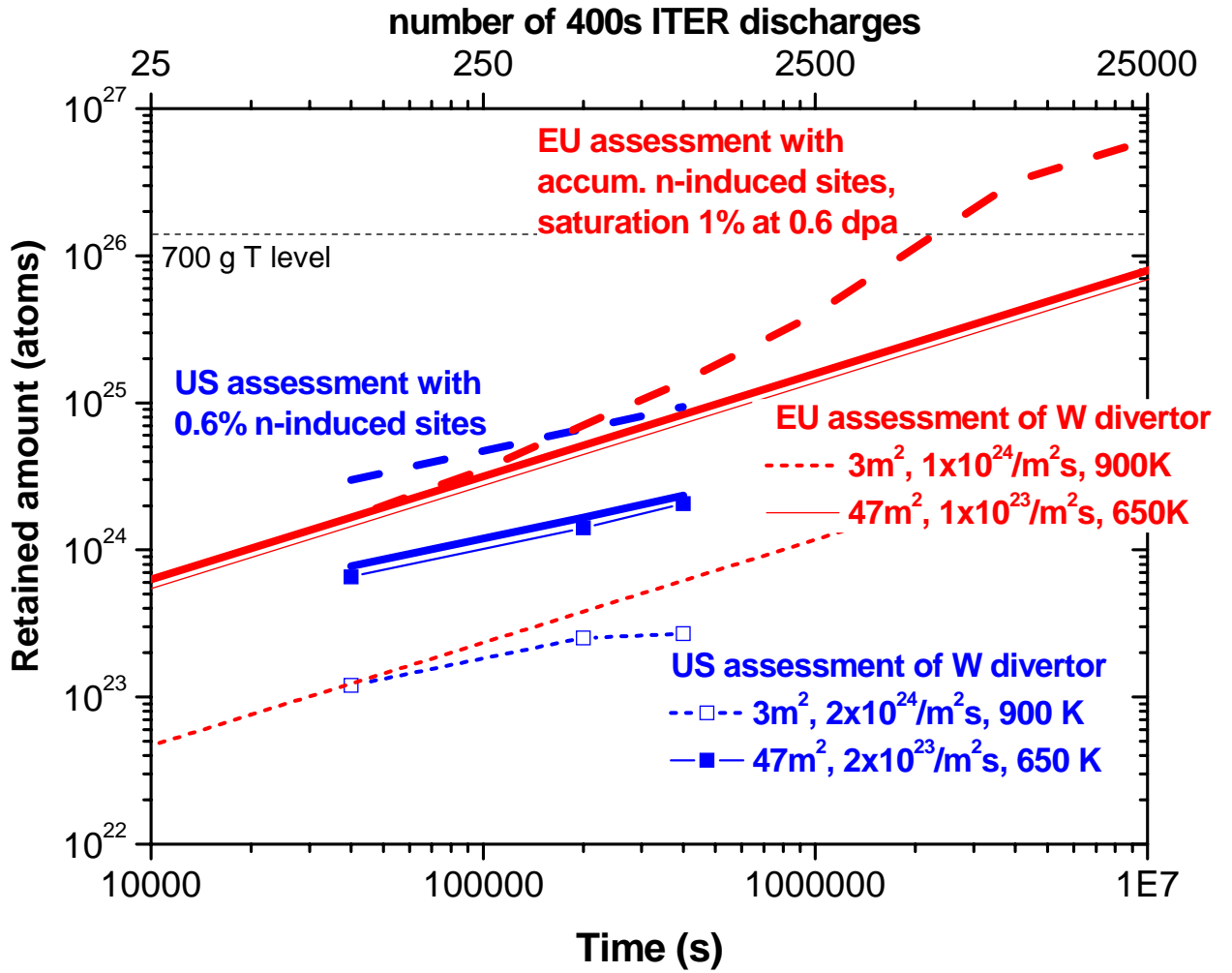

FIRST WALL

Beryllium

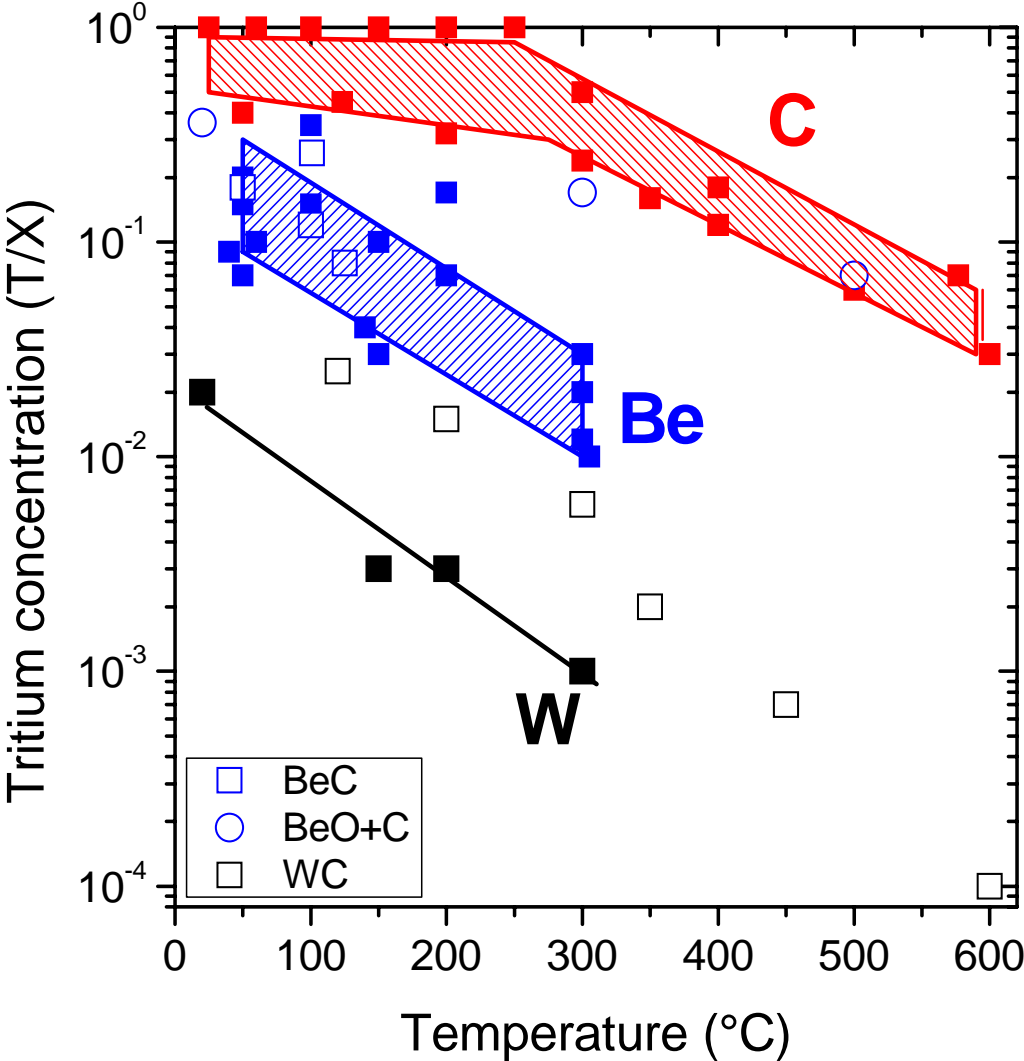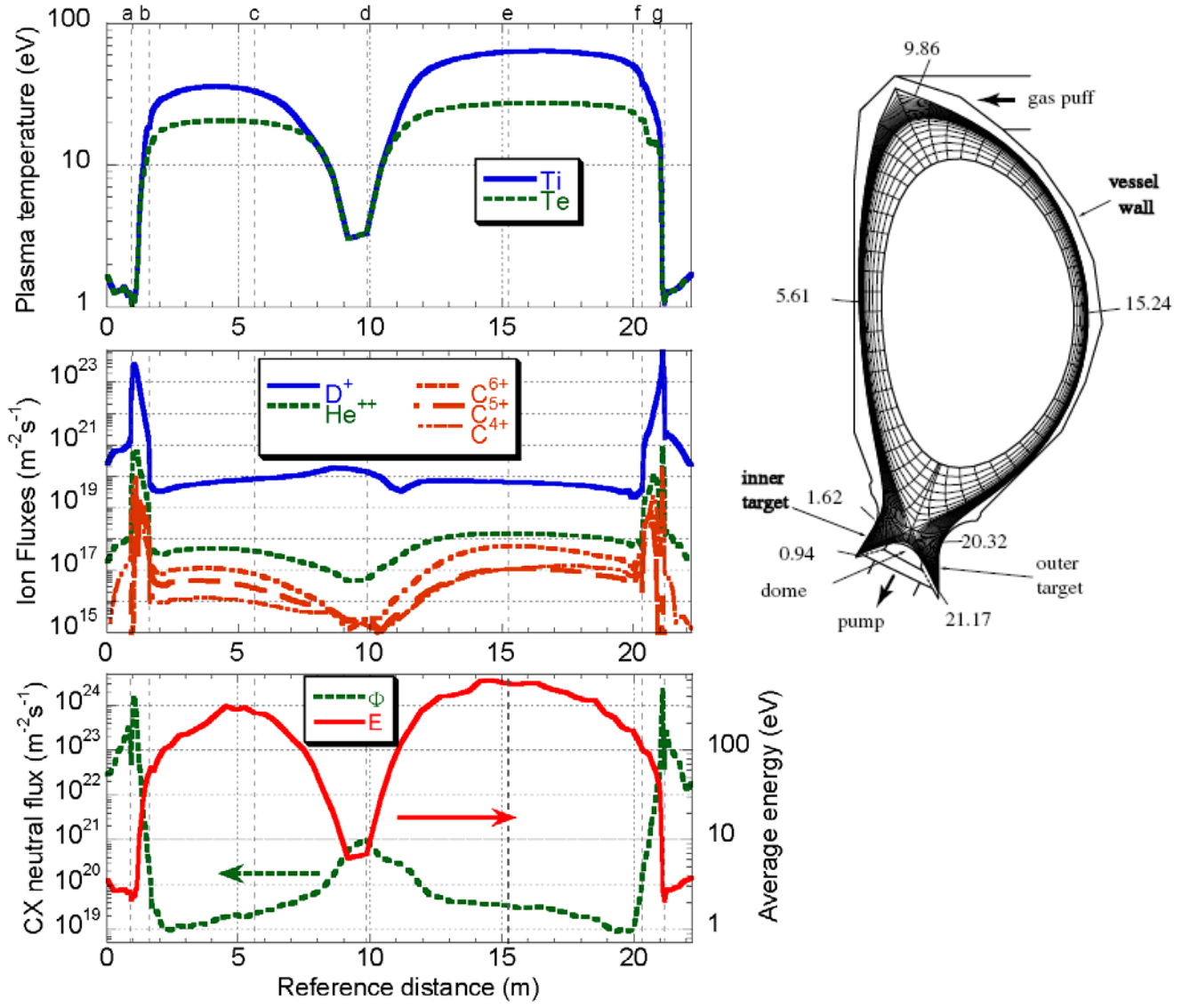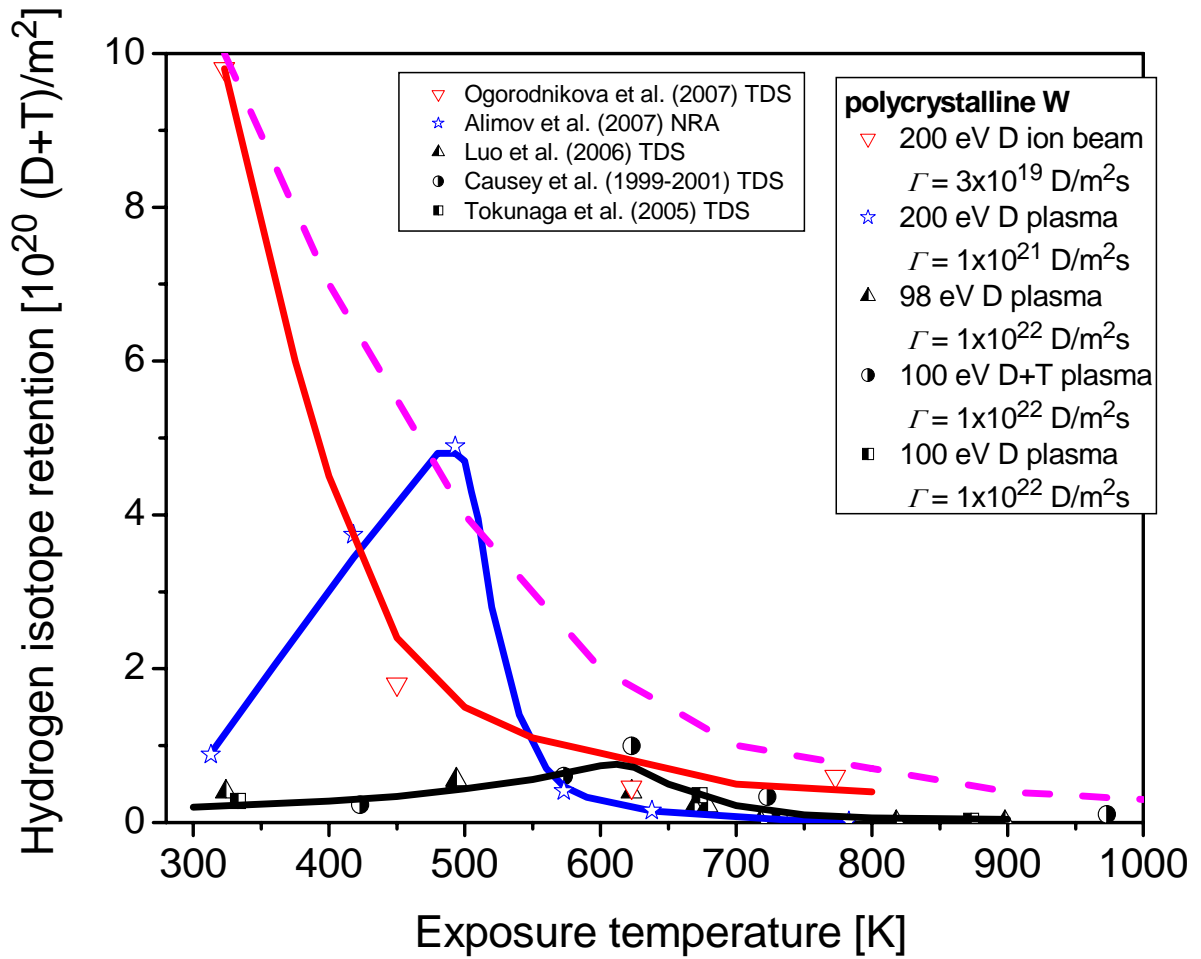Tungsten

Carbon

DIVERTOR

Figure 2.

Figure 2.
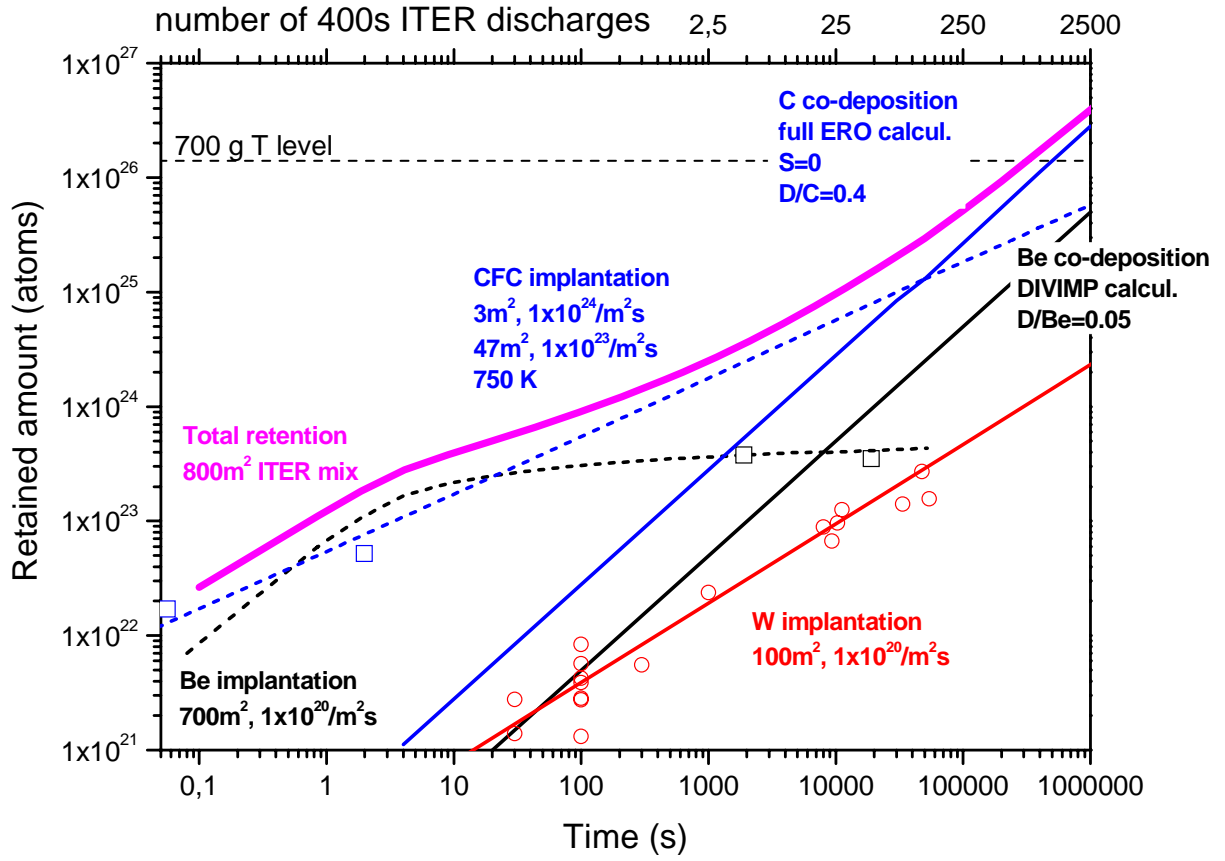
Figure 3.
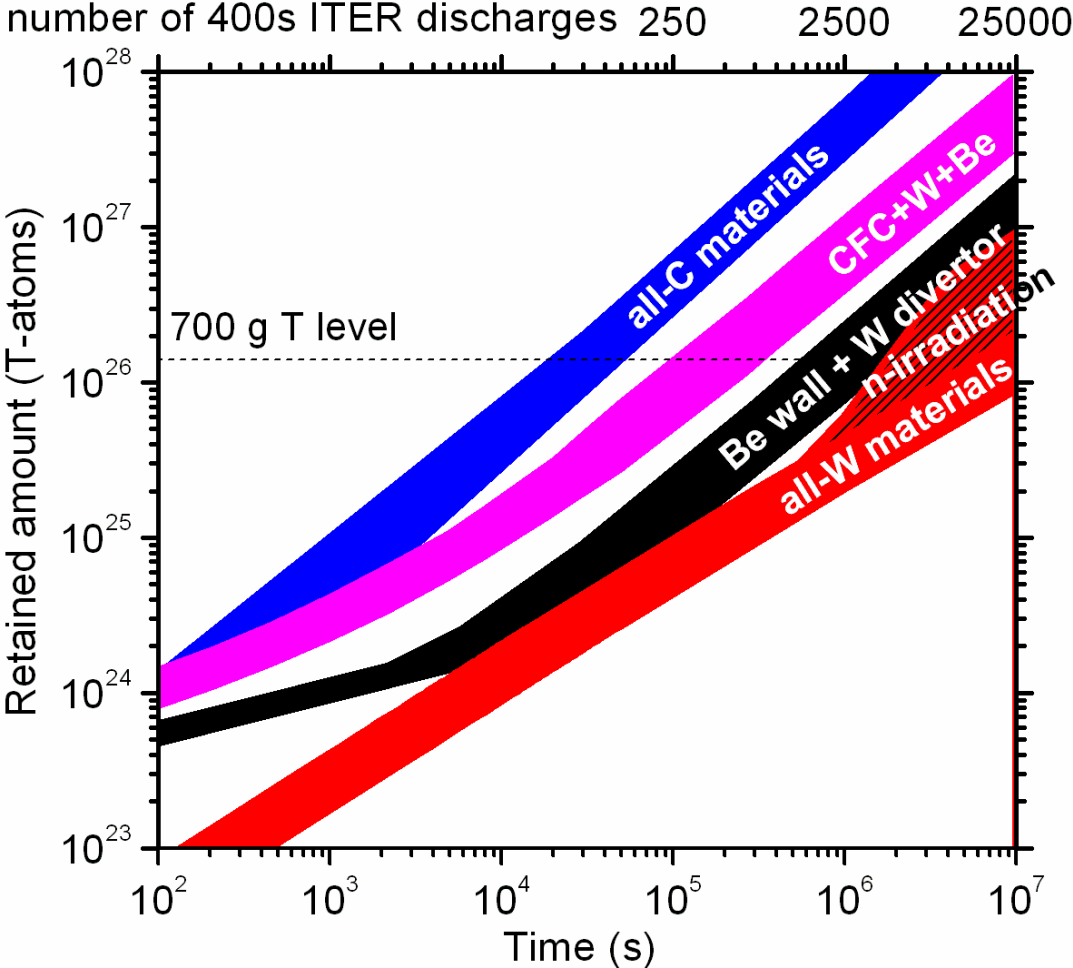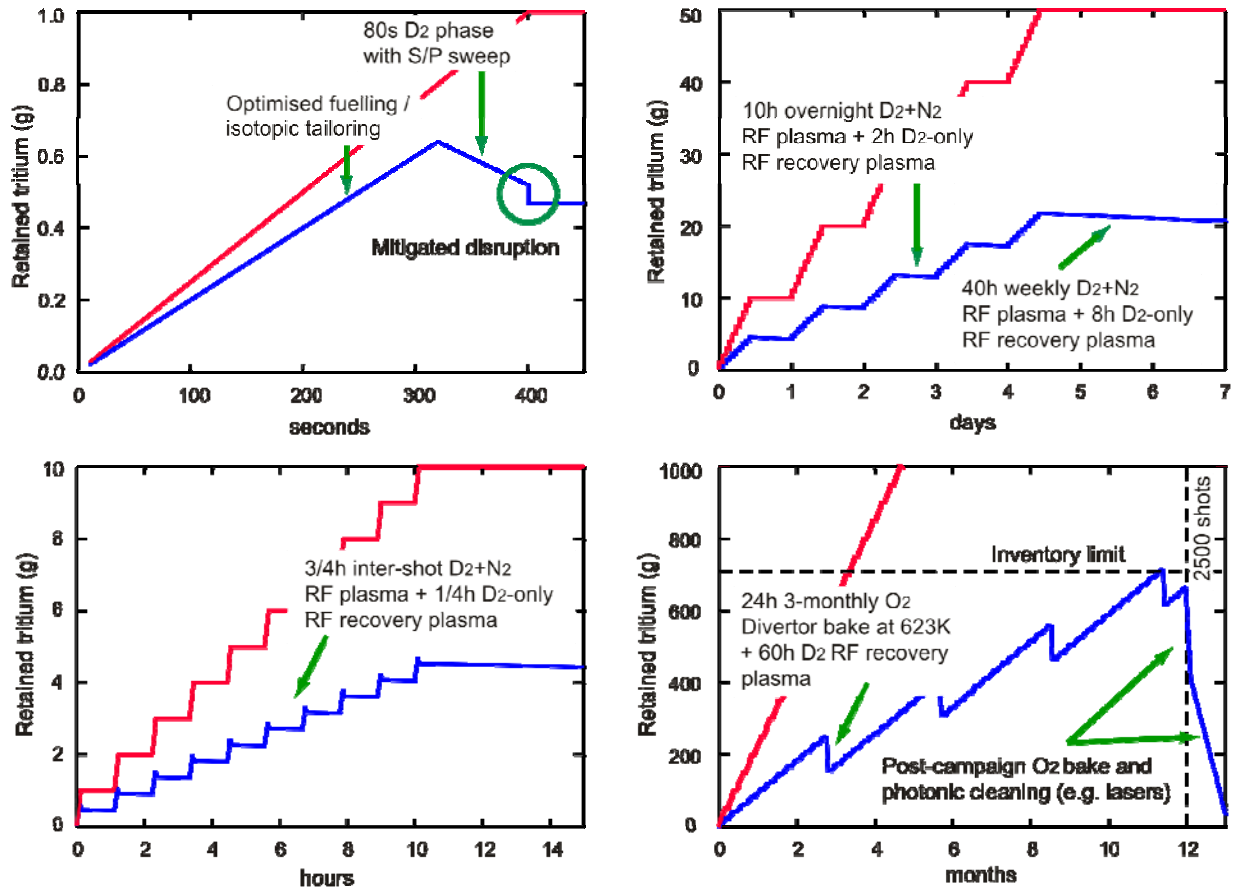
Figure 4.

Figure 5.

Figure 6.

Figure 7.

Figure 8.

Figure 9.

References

[1] G. Federici et al., Nucl. Fusion **41** 1(2001) 1967

[2] G. Federici et al., Plasma Phys. Control. Fusion **45** (2003) 1523

[3] B. Lipschultz et al., Nucl. Fus. 47 (2007) 1189-1205.

[4] Generic Site Safety Report, Volumes III, ITER Document Ref. G 84 RI (2001)

[5] T. Loarer, C. Brosset, J. Bucalossi, P. Coad, G. Esser, J. Hogan, J. Likonen, M. Mayer, P. Morgan, V. Philipps, V. Rohde, J. Roth, M. Rubel, E. Tsitrone A. Widdowson and JET EFDA Contributors, Nucl. Fusion **47** (2007) 1112–1120

[6] ITER proposal to STAC 2007

[7] H.Bergsaker et al., J. Nucl. Mater., 145-147 (1987) 727

[8] W. R. Wampler et al., J.Vac. Sci. Technol., A6 (1988) 2111

[9] C. Skinner et al, J. Nucl. Mater. **241–243** (1997) 214

[10] P. Andrew, P.D. Brennan, J.P. Coad Fusion Eng. Des. **47** (1999) 233–45

[11] C.H. Skinner et al., J. Nucl. Mater., 266-269 (1999) 940

[12] T. Loarer et al., submitted to PSI-18, J. Nucl. Mater. (2009)

[13] J. Bucalossi et al. J. Nucl. Mater. **363-365** (2007)

[14] P. Wienhold et al., J. Nucl. Mater. **313–316** (2003) 311–20

[15] M. Mayer, V. Rohde, J. Likonen, E. Vainonen-Ahlgren, K. Krieger, X. Gong, J. Chen, and ASDEX Upgrade Team, J. Nucl. Materials (2005)

[16] E. Tsitrone, Key Issues for Steady-state Operation, J.Nucl. Mater. 363-365 (2007) 12

[17] B. Pégourié et al., Overview of the Deuterium inventory campaign in Tore Supra: Operational conditions and particle balance, PSI Toledo, submitted to J. Nucl. Mater (2008)

[18] M. Rubel et al., Phys. Scr. T **103** (2003) 20–4

[19] T. Loarer T et al, 30th Conf. on Controlled Fusion and Plasma Physics (St Petersburg, Russia, 7–11 July (2003)

[20] O. Gruber for the ASDEX Upgrade Team, Nucl. Fusion **47** (2007) S622–S634

[21] M. Mayer et al., PSI Toledo (2008) submitted to J. Nucl. Mater.

[22] W.R. Wampler, B. LaBombard, B. Lipschultz, G. M. McCracken, D. A. Pappas, C. S. Pitcher, J. Nucl. Mater. **266-269** (1999) 217

[23] D.G. Whyte, J.P. Coad, P. Franzen, H. Maier, Nucl. Fusion **41** (2001) 47

[24] K. Schmid, Be flux distribution and layer deposition in the ITER divertor, PSI Toledo, submitted to J. Mucl. Mater. (2008)

[25] W.R. Wampler, C.W. Magee, J. Nucl. Materials 103&104 (1981) 509

[26] B.M.U. Scherzer, M. Wielunski, W. Möller, A. Turos and J. Roth, Nuclear Instr. & Meth. in Phys. Res. **B33**, 714-718 (1988)

[27] R. Causey, K. Wilson, T. Venhaus, W.R. Wampler, J. Nucl. Mater. 266-269 (1999) 467

[28] W. Möller, B.M.U. Scherzer, J. Bohdansky, Retention and release of deuterium implanted into beryllium, IPP-JET Report No. 26 (1985)

[29] R.A. Anderl, R.A Causey, J.W. Davis, et al., J. Nucl. Mater. 273 (1999) 1

[30] J. Roth, V.Kh. Alimov, A.V. Golubeva, R.P. Doerner, J. Hanna, E. Tsitrone, Ch. Brosset, V. Rohde, A. Herrmann, M. Mayer, . J. Nucl. Mater. **363-365** (2007)

[31] A. Golubeva, V. A. Kurnaev, M. Mayer and J. Roth: Hydrogen Retention in Plasma-Sprayed Tungsten. in: Hydrogen in Matter. (Eds.) Myneni, G.R.; Hjörvarsson, B. AIP Conference Proceedings **837**. American Institute of Physics, Melville, NY (2006) 12-21.

[32] M. Reinelt, Ch. Linsmeier, private communication (2007)

[33] B.L. Doyle , W.R. Wampler, D.K. Brice, S.T. Picraux, J. Nucl. Mater. 93/94 (1980) 551

[34] A.A. Haasz, J. Davis, J Nucl. Mater. 209 (1994) 1140

[35] N. Bernier, Ch. Brosset et al., to be published (2007)

[36] H. Khodja et al, PSI Toledo (2006) submitted to J. Nucl. Materials

[37] R.A. Causey, T.J. Venhaus, Physica Scripta T94 (2001) 9

[38] O. Ogorodnikova, J. Roth, M. Mayer, J. Nucl. Mater. **313-316** (2003) 469

[39] V. Kh. Alimov, J. Roth, Physica Scripta **T128** (2007) 6-13

[40] H. Bolt, V. Barabash, G. Federici, J. Linke, A. Loarte, J. Roth and K. Sato, J. Nucl. Mater. **307-311** (2002) 43

[41] N. Baluc, Final Report on EFDA Task TW1-TTMA-002 Del. 5: Assessment report on W (2001)

[42] Materials Assessment Report G 74 MA 10 01-07-11 W, Ch. 2.2 Tungsten (2001)

[43] M. Eldrup, M. Li, L. Snead and S. Zinkle, Characterization of Defect Accumulation in Neutron-Irradiated Mo by Positron Annihilation Spectroscopy, presented at the ICFRM 13, Nice (2007), to be published in Nucl. Instr. Methods B.

[44] O. Ogorodnikova, ITPA SOL/DIV Garching (2007)

[45] R. Causey, DIFFUSE code

[46] P. Sigmund, Phys. Rev. 184 (1969) 383

[47] R. Behrisch, W. Eckstein, eds.: „Sputtering by particle bombardment" (Springer Verlag, Berlin, 2007)

[48] W. Eckstein, Computer simulation of ion solid interactions, Springer Series in Materials Science (Springer Verlag, Berlin and Heidelberg, 1991), 1st ed.

[49] J. Bohdansky, J. Roth and H.L. Bay:, J. Appl. Phys. 51, 2861 (1980)

[50] J. Roth, J. Bohdansky, A.P. Martinelli, Rad. Eff. 48, 213 (1980)

[51] J. Roth, J. Bohdansky, W. Poschenrieder and M.K. Sinha, J. Nucl. Mat. 63, 222 (1976)

[52] W. Jacob, J. Roth, in: Sputtering by particle bombardment, eds. R. Behrisch, W. Eckstein, (Springer Verlag, Berlin, 2007)

[53] J. Roth, R. Preuss, W. Bohmeyer, S. Brezinsek, A. Cambe, E. Casarotto, R. Doerner, E. Gauthier, G. Federici, S. Higashijima, J. Hogan, A. Kallenbach, A. Kirschner, H. Kubo, J.M. Layet, T. Nakano, V. Philipps, A. Pospieszczyk, R. Pugno, R. Ruggiéri, B. Schweer, G. Sergienko and M. Stamp, Nuclear Fusion **44** (2004) L21

[54] E. Salonen, K. Nordlund, J. Keinonen, and C. H. Wu, Phys. Rev. B 63, 195415 (2001)

[55] D G Whyte, J N Brooks, P C Stangeby and N H Brooks, Phys. Scr. **T111** T111 (2004) 34-41

[56] E. Vietzke, J. Nucl. Mater. **145-147**, 443 (1987)

[57] K. Schmid, M. Baldwin, R. Doerner, J. Nucl. Materials **337-339** (2005)

[58] W. Jacob, C. Hopf, M. Schlüter, and T. Schwarz-Selinger, J. Nucl. Mater. **337-339** (2005)

[59] A. Kirschner, et al., Journal of Nuclear Materials **328** (2004) 62

[60] E. Vietzke, A.A. Haasz, in: W.O. Hofer, J. Roth (Eds.), Physical Processes of the Interaction of Fusion Plasmas with Solids, Academic, San Diego, 1996

[61] W. Bohmeyer et al., J. Nucl. Mater. **337-339** (2005) 89

[62] M Mayer, V Rohde and the ASDEX Upgrade Team, Nucl. Fusion **46** (2006) 914

[63] W. Bohmeyer et al., Proc. 30th EPS Conf. on Controlled Fusion and Plasma Physics, St. Petersburg, Russia, 2003, Vol. 27A, p. 3.184

[64] R. Doerner, M. Baldwin, PISCES-B, re-erosion of Be, private communication (2007)

[65] R.P. Doerner, M.J. Baldwin, G. De Temmerman, J. Hanna, D. Nishijima, J. Roth, K. Schmid, G.R. Tynan, K. Umstadter: Issues associated with co-deposition of deuterium with ITER materials, IAEA Genova (2008) to be published

[66] T. Schwarz-Selinger, A. von Keudell, and W. Jacob, J Applied Physics 86 (1999) 3988

[67] W. Jacob, Thin Solid Films 326 (1998) 1

[68] R. Doerner, ITPA Garching (2007), unpublished

[69] M. Mayer et al., J. Nucl. Mater. **240** (1997) 164

[70] M. Baldwin et al., J. Nucl. Mater. **337-339** (2005) 590

[71] J. Brooks, A. Kirschner, J. Nucl. Mater. 313-316 (2003) 424

[72] J. Roth, A. Kirschner et al. J. Nucl. Mater. **337-339** (2005) 970

[73] A. Kirschner, D. Borodin, S. Droste, V. Philipps, U. Samm, G. Federici, A. Kukushkin, A. Loarte, . J. Nucl. Mater. **363-365** (2007)

[74] R. Behrisch, G. Federici, A. Kukushkin and D. Reiter, Journal of Nuclear Materials, 313-316 (2003) 388-392,

[75] G. Federici, et al., J. Nucl. Mater. 290-293 (2001) 260

[76] H. Verbeek, J. Stober, D.P. Coster, W. Eckstein, R. Schneider, Nucl. Fusion **38** (1998) 1789.

[77] A. Kallenbach, A. Loarte, private communication (2007), presented at the ITPA SOL/DIV in Avila (Jan. 2008)

[78] R. Pugno et al., accepted for publication (2007)

[79] K. Schmid et al., Report EFDA Task TUNMOD and ITPA Tarragona (2005)

[80] J.P. Coad, P. Andrew, S.K. Erents, D.E. Hole, J. Likonen, M. Mayer, R. Pitts, M. Rubel, J.D. Strachan, E. Vainonen-Ahlgren and A. Widdowson, J. Nucl. Materials 363-365 (2007) 287

[81] G. Mathews, PSI 2004, Portland, J.Nucl. Mater. (2005)

[82] G.F. Counsell et al 2006 Plasma Phys. Control. Fusion **48** B189

[83] M. Gouge, et al. Fusion Technol. 28 (1995) 1644

[84] M Schlüter, C Hopf and W Jacob 2007 "Chemical sputtering of carbon by combined exposure to nitrogen ions and atomic hydrogen" submitted to New J. Phys.

[85] F.L. Tabarés, D. Tafalla et al., Plasma Phys. Control. Fusion **44** (2002) L37

[86] F.L. Tabarés and V. Rohde 2005 Nucl. Fusion **45** L27

[87] W. Bohmayer et al., (2008) to be presented at the 18th Plasma Surface Interactions Conference, Toledo, Spain

[88] R.A. Causey et al., J. Nucl. Mater. **176-177** (1990) 987

[89] R. Ochoukov1, A.A. Haasz and J.W. Davis, Phys. Scr. **T124** (2006) 27

[90] C.K. Tsui, et al., Nucl. Fusion **48** (2008) 035008

[91] M.J. Baldwin et al., J. Nucl. Mater. **337-339** (2005) 590

[92] R.A. Causey, D.S. Walsh, J. Nucl. Mater. **254** (1998) 84

[93] R.A. Anderl et al., Journal of Fusion Energy **16** (1997) 95

[94] C. H. Skinner et al., J. Nucl. Mater. **313-316** (2003) 496

[95] C. Grisolia et al., J. Nucl. Mater. **363–365** (2007) 1138

[96] A. Widdowson et al., J. Nucl. Mater. **363–365** (2007) 341

[97] C. Grisolia et al., Fusion Engineering and Design **81** (2005) 149

[98] N. Bekris et al (2007) 8th International Symposium on Fusion Nuclear Technology, Heidelberg, Germany to be published in Fusion Science and Technology