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# Fuel retention in tokamaks

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ABSTRACT

Tritium retention constitutes an outstanding problem for ITER operation and future fusion reactors, particularly for the choice of the first wall materials. In present day tokamaks, fuel retention is evaluated by two complementary methods. The in situ gas balance allows evaluation of how much fuel is retained during a discharge and, typically, up to one day of experiments. Post-mortem analysis is used to determine where the fuel is retained, integrated over an experimental campaign. In all the carbon clad devices, using the two methods, the retention is demonstrated to be very closely related to the carbon net erosion. This results from plasma–wall interaction with ion and charge-exchange fluxes, ELMs and is proportional to the pulse duration. The fuel retention by implantation saturates at high wall temperatures and limits the D/C ratio in the deposited layers but, as far as a carbon source exists, the dominant retention process remains the co-deposition of carbon with deuterium. In full metallic device, in the absence of wall conditioning with boron, co-deposition is strongly reduced and fuel retention below 1% can be achieved. Extrapolation to ITER shows that removing the carbon from the plasma-facing components would increase the number of discharges to 2500 before reaching the maximum tritium limit of 700 g.



## 1. Introduction

Fuel retention is one of the crucial points to be investigated for next step fusion devices, in particular when using plasma-facing components (PFCs) of carbon and for the long discharges foreseen in ITER (400 s–7 min). From the licensing limits for the operation of ITER, the limit for the inventory of releasable tritium in the vacuum vessel is 350/700 g [1]. Assuming an equal mix of deuterium and tritium, a fuel injection rate of 200 Pa $m^3$ $s^{-1}$ (or $5 \times 10^{22}$ tritium atoms per second at 293 K) and a *T* retention of 10%, this limit would be reached in about 35/70 discharges without any dedicated cleaning efforts. The retention of 1% of the *T* injected into ITER leads to a retention of 1 g per discharge, independent of the wall material. Today, most tokamaks use carbon abundantly in PFCs except Alcator-C, FTU, TRIAM-1 M, which use high *Z* material and ASDEX Upgrade (AUG) which recently moved to a full tungsten coverage of PFCs. This enables a good data base to be assembled pertaining to fuel retention by carbon-dominated and carbon-free devices [1,2]. Fuel retention analyses are carried out over a wide range of characteristic times: from the plasma–wall interaction occurring during plasma events (from tens of μs to ms for recycling flux and ELMs) up to a whole experimental campaign (of up to few months). The aim is to assess the dominant processes of the short and long-term fuel retention associated with carbon, beryllium and/or high *Z* materials and to extrapolate the findings to ITER. The short-term retention process takes place in the PFCs which receive a direct ion and/or neutral flux. This results in a shallow implantation of particles (a few nm) which saturates with fluence ($\sim 10^{21}$ $m^{-2}$ at 200 °C), but which can, nevertheless, store dynamically additional fuel ($\sim 10^{20}$ $m^{-2}$) during plasma loading and which constitutes the major part of the particles released by outgassing at the end of the discharge. The two main mechanisms for long-term fuel retention are identified as implantation and co-deposition [3]. Deep implantation, diffusion/migration and trapping in the bulk material result from the direct interaction with the ion plasmas and/or the neutral from charge-exchange fluxes. The co-deposition process results from the combination of both the recycling hydrogen flux and the sputtered atoms from the wall. These eroded atoms from the wall are ionised and transported through the SOL, recycle and arrive after several steps in the divertor region. Eventually, some of these carbon atoms are co-deposited with deuterium in shadowed area and form flakes and/or layers.

The fuel retention resulting from these processes is generally evaluated by two complementary methods: gas balance and post-mortem analysis. The gas balance provides information on 'how much' retention occurs in a discharge, in a day or even up to a week of experiments. This method is widely used in fusion devices [4] to evaluate the retention as a function of time, resulting from the deuterium recycling flux and the impurity production e.g. by ELMs over a wide time scale in the 0.1–100 s range. This method is also used to assess the particle recovery (hydrogenic and impurities) in between discharges, during cleaning discharges, isotope exchange and disruptions. This method delivers a global measurement of the accumulated in-vessel inventory, based on the difference between the injection and the exhausted flux.

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The post-mortem analysis allows determining the main processes leading to fuel retention in the materials (implantation, migration, gaps, layers, flakes, dust), how these depositions are formed, in which regions and integrated over one experimental campaign (up to several months of accumulated hours of plasma operation with different magnetic configurations). There is a wide range of methods depending on the main objectives of the analysis: Surface analysis, Structure, Depth profile, Composition [5–8]. The aim of this paper is to report on the fuel retention deduced from gas balance and post-mortem analysis, to assess how the results compare and then to extrapolate to ITER [9].

The first part of the paper reports on gas balance analysis and the estimates of the short and long-term retention. The effects of the recycling flux, the carbon production and their consequences on the resulting retention by co-deposition are discussed. The retention by implantation and its potential saturation is also reported and compared to co-deposition. The retention observed in all the carbon devices is compared, and is known to exhibit a similar trend proportional to the carbon production and the pulse duration. These behaviours are compared with the previous results obtained during the D–T phases performed in both TFTR and JET. Although the majority of the tokamaks are carbon dominated, recent results from Alcator C-Mod and ASDEX Upgrade using high *Z* materials (respectively molybdenum and tungsten) are reported. The results of these metallic devices are of high importance to quantify the potential benefit from moving from a carbon-dominated device to fully metallic machines permitting reduced retention by co-deposition [10,11] The results of Alcator C-Mod exhibits a similar behaviour as those observed in carbon devices, whilst the results of AUG show that a significant drop of the retention is observed only when all PFCs are covered with W. This strong difference is discussed, particularly through regular boronisations carried out in Alcator C-Mod whilst AUG experiments have been performed prior to any wall conditioning.

The second part of the paper reports on the post-mortem analysis to evaluate the location of the retention. Results from carbon layers deposited in the inner areas of the divertor are reported for the 2001–2004 JET experimental campaign (First wall temperature at 200 °C and subdivertor structure at ~50 °C) as well as in JT-60U for two series of campaign with the overall vessel at 300 °C and then 150 °C. The results show that the D/C ratio is in the range 0.01–0.25 for layers facing the plasma and that it drops to 0.01–0.15 as the wall temperature increases. In both cases, the D/C ratio of the carbon deposition in shadowed areas below the divertor structure is reported and the effect of the wall temperature (50 °C for JET and 150 °C for JT-60U) is discussed. The recent results of the D inventory from AUG in a full W configuration show a significant drop in D retention. However, although all the PFCs of the vessel are covered with *W*, the contribution of the remaining carbon traces (0.5–1% in the SOL) are discussed particularly on the resulting retention processes.

The third part compares the evaluation of the retention using gas balance and post-mortem analysis. It is shown that using these two methods, it is possible to quantify and to localise the overall retention, demonstrating that these methods are complementary and that the evaluations are close together within a factor of two. The last part discussed the extrapolation of the retention to ITER with the different material options [11].

## 2. Gas balance

### 2.1. Short and long-term retention

The gas balance analysis relies on the measurement of the injected and pumped particle fluxes during the plasma discharges and also in between pulses for quantifying the outgassing of particles [4]. The standard gas balance, mostly used on all the devices, is carried out as a function of time during the pulse. This procedure allows assessment of the resulting retention as a function of plasma shape, plasma scenario and, particularly, the contribution of the impurity flux production due to erosion. In JET, a second and complementary gas balance procedure has been carried out using the integral of the pumped particle flux. This is performed by regenerating the cryogenic pumps and collecting the gas in a calibrated volume well after the experiments (~at least 1/2 h after the last pulse) [12] to collect most of the gas released by outgassing. In Alcator C-Mod [13], a similar method is used by turning off all the pumps just prior to the pulse and turning them on about 5 min after the pulse (plasma duration ~2 s) when the temperatures of both the gas and the vessel have reached a steady-state conditions. The total retention (long and short-term) during the plasma is deduced from the difference between the total amount injected and the total amount exhausted up to the end of the plasma. In JET, the overall particles recovered by outgassing well after the pulse (~30 min) correspond to the short-term retention only. As a consequence, both the long and short-term retentions can be evaluated during the pulse.

The short-term retention (dynamic retention) is closely linked to the recycling flux and is also strongly influenced by the surface temperature of the target plates. This behaviour can be better identified with the long discharges performed on Tore Supra (TS) which is an actively-cooled device allowing the outgassing flux to be kept constant over all the discharge duration and therefore enabling the separation of the long and short-term retention phases. This is illustrated in Fig. 1 showing the retention as a function of time for two long discharges in Tore Supra ($I_p$ = 0.5 MA, $B_T$ = 3.1 T, $P_{in}$ = 3.0 MW), performed consecutively without any conditioning in between. With the edge plasma conditions being around a temperature of $T_e \sim 70$ eV at the last closed flux surface (LCFS), the equilibrium is obtained after about 100 s. In divertor machines this equilibrium can be reached in only a few seconds mainly due to both the lower edge temperature ($T_e \sim 10$–15 eV) and the higher recycling flux in the divertor. However, in all the machines, the short-term retention is limited to the dynamic retention (generally associated to a 'fast' particle reservoir [14–16]) and can be recov-

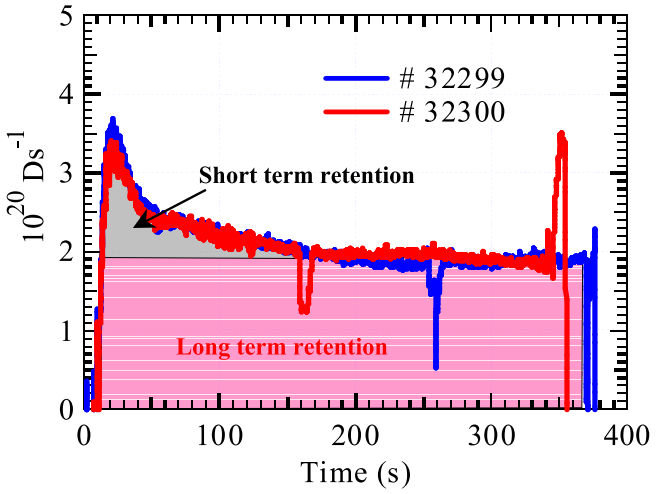


**Fig. 1.** Retention, as a function of time, for two consecutive long discharges in Tore Supra ($I_p$ = 0.5 MA, $B_T$ = 3.1 T, $P_{in}$ = 3.0 MW). The short-term retention, representing particles recovered at the end of the discharge and between pulses, is shown as a grey area. The red area corresponds to the long-term retention, resulting from both deep implantation and co-deposition. (For interpretation of the references to colour in this figure legend, the reader is referred to the web version of this article.)

ered by outgassing in between discharges. In short-pulse discharges, the surface of the first wall absorbs particles and it works as a pump. This surface absorption is effective in controlling the plasma density for a limited time. This is the case for both limiter and divertor machines [17–20]. This behaviour is enhanced with beryllium, as demonstrated in JET with carbon and beryllium limiters [21] and also in AUG with the full tungsten coverage [22]. Similar results are also observed with the wall conditioning based on the D recovery by He GDC after the discharge, as shown in AUG (45% and 70% of W coverage) [23] and in TS [24–27].

The long-term retention is linked to both co-deposition and implantation. The former is correlated with the carbon erosion and the D retained by co-deposition with the eroded carbon. A series of experiments has been carried out in JET to assess the long-term retention as function of the plasma scenario such as L-mode, type III ELMy H-mode and type I ELMy H-mode. An example of the resulting gas balance for a type I ELMy H-mode is shown in Fig. 2 displaying the particle fluxes (injection and exhaust) for a typical discharge [12]. At the beginning of the heating phase (13 MW of total input power and ELM energy ~100 kJ) the retention flux is dominated by short-term retention. It drops very quickly to be mainly dominated by long-term retention, characterised by the steady-state which is nearly reached after 'only' 6 s. Similar experiments have been carried out in L-mode (2 MW of total input power) and in type III ELMy H-mode (6 MW of total input power and ELM energies <5–10 kJ). As previously reported, the short-term retention drops slowly in a similar manner for L and type III ELMy H-mode [12]. Using the integral method for gas balance analysis, the long-term retention has been evaluated for the three types of discharge. Using the L-mode as a reference, it is possible to estimate the increase of the retention during the heating and/or the divertor phase observed in the type I ELMy H-mode. Compared to L-mode, the ELM-averaged $D_\alpha$ emission [28] observed during type III ELMy H-mode from the main wall and from both divertor legs drops slightly whilst only the CIII emission from the main wall is increased, the CIII emission from both the inner and outer legs remaining of the same order. From L-mode to type I ELMy H-mode, both the $D_\alpha$ and carbon emission (CIII) from the main wall and the two divertor legs are strongly enhanced. When averaging over the

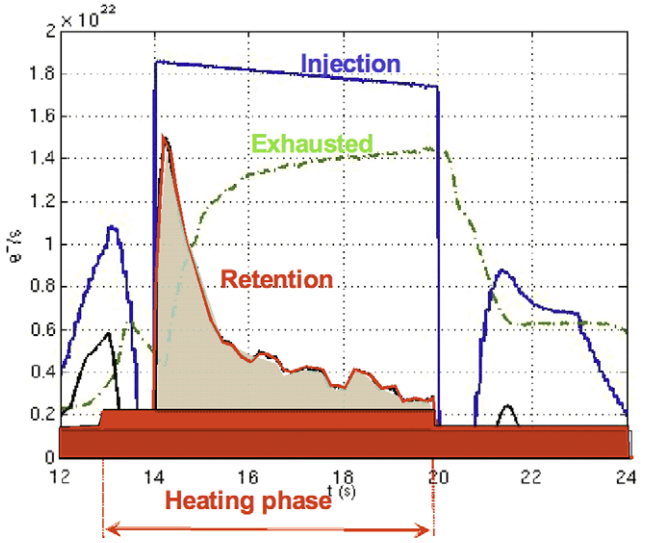


**Fig. 2.** Particle flux for a typical ELMy H-mode with type I ELMs. $I_p$ = 2.0 MA, $P_{tot}$ = 13 MW (NBI + ICRH), $n_e = 0.7 n_{Gw}$. The red area represents the additional retention observed in type I ELMy H-mode compared to a similar discharge carried out in L-mode over the divertor phase. The long-term retention is averaged over the divertor phase. (For interpretation of the references to colour in this figure legend, the reader is referred to the web version of this article.)

**Table 1**
Total number of particle injected, recovered from cryopump regeneration and long-term retention averaged over both the divertor and the heating phases. The results cover three series of experiments in L-mode (2 MW), Type III (6 MW <5–10 kJ) and Type I ELMy H-mode (13 MW–100 kJ) performed at JET.

| Pulse type | Injection (D $s^{-1}$) | Retention (D $s^{-1}$) divertor phase | Retention (D $s^{-1}$) Heating phase |
|---|---|---|---|
| L-mode | $\sim 1.8 \times 10^{22}$ | $1.34 \times 10^{21}$ | $2.09 \times 10^{21}$ |
| Type III | $\sim 0.6 \times 10^{22}$ | $0.8 \times 10^{21}$ | $1.3 \times 10^{21}$ |
| Type I | $\sim 1.7 \times 10^{22}$ | $2.08 \times 10^{21}$ | $2.66 \times 10^{21}$ |

heating phase, the results exhibit the same trend with an enhanced absolute value of the retention. Table 1 summarises the resulting retention fluxes as a function of the plasma scenario, averaged over the divertor phase and the heating phase. For the type III ELMy H-mode, the plasma scenario is somewhat different (lower particle injection rate), but the overall results show a clear increase of the retention from L-mode to type I ELMy H-mode. This is consistent with the increase of the carbon source and is observed on all carbon devices for high input power and gas injection [29]. Similar behaviours have been recently studied in JT-60U [30] showing enhanced fuel retention as the plasma density and neutral beam power are increased and correlated to the recycling flux and C impurities in the SOL. This was also observed in TFTR where more T retention by co-deposition was observed at high NBI power due to increased carbon production from the outer limiters [31–34].

In the long-term retention processes, the possible saturation of the retention, the so called 'wall saturation' has been extensively investigated in JT-60U [35–39] for long discharge operations (>25–30 s). Some experiments were also reported for a series of long discharges in TS before the CIEL upgrade [40] when 'only' 80% of the PFCs were actively-cooled and also in TRIAM-1 M for the discharge lasting more than 5h00 [41]. For both devices, the slow increase of the plasma density under long plasma operation is attributed to the surface temperature increase of the non-actively-cooled area, resulting in an enhanced and uncontrolled outgassing flux (mainly D, $D_2$, $C_xD_y$ and $C_xH_y$) from the overheated surface and an increase of the plasma density whether active pumping is applied or not. As a result, the outgassing flux increases dramatically with the high surface and bulk temperature; This strong particle release being the signature that these areas are depleted and that consequently there is no more retention associated to the implantation process in these areas. However, the conclusions drawn on the overall retention using gas balance analysis can hide other processes. Indeed, this strong out-gassing does not prevent and/or cancel the retention by co-deposition from occurring in areas not directly heated/viewed by the plasma outflux (ions and/or CX). In addition, in JT-60U, the carbon generation from the private flux dome increases the carbon ion density along the divertor leg near the X-point. At the same time the carbon ion density in the main plasma also increases due to carbon impurity penetration from the X-point [35,38]. The time behaviour of the CD band emission intensity is similar to the behaviour of the CII emission intensity near the X-point. Since co-deposition is a continuous process as long as carbon impurities are produced, the strong release of carbon in the SOL and divertor area could enhance co-deposition. Similar behaviours have been observed in TRIAM-1 M [41] where the Mo deposition has been demonstrated to be proportional to the plasma duration while the outgassed flux becomes large enough to keep the plasma density constant. TRIAM -1 M [42].

When the surface temperature is kept constant over the pulse duration the fuel retention associated with the implantation in the CFC has been shown to vary as the fluence$^{0.5}$ [43]. The retention over long discharges in actively-cooled devices permits the study

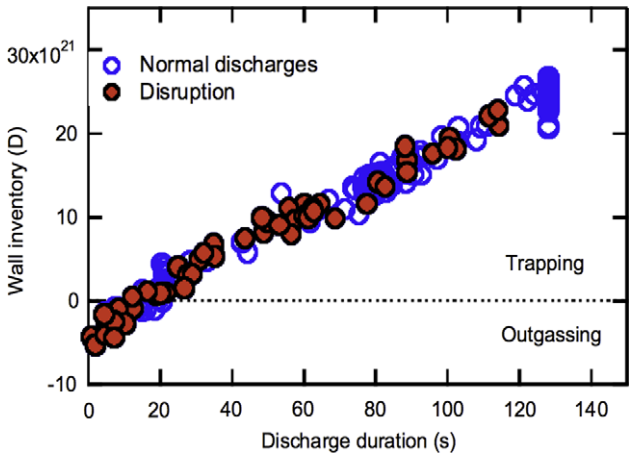


**Fig. 3.** Wall inventory in Tore Supra, as a function of the discharge duration, for repetitive long discharges without any wall conditioning [44]. All the discharges are included, ending normally or with a disruption. The inventory is proportional to the discharge duration and the disruptions do not allow the recovery of more particles, showing that the retained particles are not affected by the area heated by the disruption.

of both phenomena without modification of the outgassed flux, which remains constant over the discharge duration. In Tore Supra, a dedicated experimental campaign [44] has been carried out for comparing and particularly understanding the difference of retention evaluated by gas balance and by post-mortem analysis [45]. After 10 days of operation without any wall conditioning in between discharges, a total of 5 h of plasma using the same plasma scenario has been achieved. Fig. 3 summarises the total D vessel inventory as a function of the pulse duration for the entire discharges performed during this experimental campaign, including soft plasma endings and also disruptions. The results from the post-mortem analysis are not available yet to assess the proportion of fuel which could possibly migrate into the bulk CFC, but results from gas balance analysis confirm that the retention is proportional to the plasma duration and that after more than 5 h of plasma there is still no sign at all of saturation. Also, it is demonstrated that the recovery is always more or less the same and around $\sim 10^{22}$ D integrated over the 10 days of experiments. In any case, this quantity is small compared with that retained in the vessel ($\sim 10^{24}$ D). Also, it is worth noting that, so far, in Tore Supra, the disruptions do not allow the recovery of more particles than due to outgassing after a normal end for the pulse. Moreover, this quantity is independent of the amount retained during the discharges and/or the previous discharges. This demonstrates that the retained particles are not affected by the area heated by the disruption, either due to a low heating associated to a too low diamagnetic plasma energy ($\sim$200 kJ) and/or the heated area does not retain much D. The issue of a possible retention in dust is not discussed here but could contribute to the long-term retention. However, these results confirm that the possible fuel recovery both by long-term outgassing and disruptions at low energy are not efficient enough, as already observed for the DT campaign in TFTR [31,32,34] with the disruptive discharge cleaning procedure and also during the JET DT experiments [46,47]. It is worth noting that in these DT experiments, the accuracy of the gas balance over very long duration is excellent since the quantification of the fuel retention with T has the advantage of being very accurate independently of the time integration.

*2.1.1. Retention in tokamaks with carbon wall*

The long-term retention depends on the carbon source associated with the plasma scenario and on the resulting co-deposition in remote areas which is a 'slow' but accumulative process proportional to the discharge duration. The contribution from the bulk implantation and diffusion is more difficult to evaluate, but according to the JT-60U, TEXTOR [48,49] and TS [26] results, it seems that this is not the dominant process for the long-term retention and also that is can saturate if high surface temperatures can be reached. In Tore Supra, no sign of saturation is observed after 5h00 of plasma without any conditioning in between discharges. Compared to laboratory results [43], and due to the constant wall temperature on TS, the implantation dependence as fluence$^{0.5}$ should have been observed after the 18,000 s of plasma operation for a total fluence of $2.3 \times 10^{25}$ D m$^{-2}$ [44] for this series of repetitive discharges. This absence of a decrease of the retention with time (or fluence) tends to support the co-deposition as the dominant retention process in Tore Supra. In a divertor device with carbon wall (AUG, JET, JT-60U, etc.), the main erosion areas and carbon sources are antenna limiters, bumper limiters, main chamber and also the outer divertor leg [29] although the latter can be somewhat compensated by redeposition. However, net erosion in the outer divertor leg of AUG was determined with marker stripes [50]. The eroded carbon flows in the SOL to the inner divertor area where it starts a process of deposition–erosion, redeposition and so on. Most of the carbon is definitively deposited in the inner leg region, identified and characterised by carbon layers. The D/C ratio found in these deposits is around 0.2–0.3 [51,52]. However, a part of these layers is also re-eroded particularly by the ELMs [53] which redistribute the deposited carbon over the divertor area [54,55]. As a consequence, the deposition in the inner area of the divertor becomes a secondary carbon source. During each erosion/redeposition step, carbon is moved from plasma with higher ion temperatures (and often also hotter surface temperatures) and deposited at places where the ion temperatures are lower (also associated to cooler surfaces) [56]. This leads to a dissociation of stable and volatile hydrocarbon molecules: $CD_4$, $C_2D_x$ ($x = 2,4,6$) resulting from chemical erosion and/or thermal decomposition of deposited layers [53]. During this process, they continue to incorporate more hydrogen isotopes, becoming lighter (more volatile) and therefore have a longer mean free path range and are redeposited further away from the place where they were produced. After several erosion–deposition cycles the material ends up in the bottom of the divertor in the colder part of the machine where they 'condense' in places which are not in the line of sight of the plasma ions and CX neutrals, i.e. they are protected from further erosion [57]. The flakes collected in these parts of the machine, in particular the louver area in JET [58], were found to have a D/T ratio $\sim$0.75. At that position, no erosion processes are expected leading to a non-saturating inventory and this distribution in remote areas is independent of the magnetic field. These radicals form soft hydrogen-rich hydrocarbon layers (D/C $\sim$ 0.7–1.4) deposited by neutral carbon radicals: $CD_3$, $C_2D_x$ ($x = 1,3,5$) having a high sticking probability and created by re-erosion of deposited hydrocarbon layers on the divertor tiles [56–61]. This high sticking probability was measured directly with sticking monitors in AUG [62]. This is also consistent with the absence of layers (and consequently with a negligible contribution to the retention) in the pumping duct of AUG [63] and more generally away from the divertor structure, as observed in JET after the DTE campaign [64,65] and recently in JT-60U [66].

In all the carbon devices, limiter and divertor machines, a continuous increase of the D(T) inventory is observed as the D(T) operations are in progress [3,31,32,46,58,64,65,67]. It is worth noting that the overall tritium balance in TFTR and JET showed that 16% (TFTR) and 17% (JET) of the tritium injected was still retained in the machine after several intense cleaning campaigns using D, H or He plasmas in both devices, venting and also performing disruptive discharge cleaning (DDC) in TFTR which demonstrated the

process not to be as efficient as though [3,33,34]. In TFTR the tritium was retained in the PFCs, shadowed area and also a significant part in the gaps [68]. After the long cleaning period which has followed the DTE campaign in JET, more than 90% of the remaining tritium has been found in the flakes below the divertor structure [65]. A very minor proportion was found on the PFCs and only a very small fraction diffused inside the CFC bulk [58]. In these two large tritium tokamak experiments, the dominant retention was co-deposition which can be avoided only if the carbon source is cancelled by using high $Z$ materials with much lower erosion rates.

*2.1.2. Retention in tokamaks with metallic walls*

In Alcator C-Mod, a series of 16 repetitive discharges totalling 30 s of plasma without any disruption at the end has been carried out with cleaned Mo tiles, the remaining boron layers from previous wall conditioning being totally removed prior to these experiments [13,69]. Fig. 4 shows the cumulative D retention as a function of the cumulative D ion fluence to the wall. A linear dependence with the D ion fluence to the wall at a retention rate of ~0.75% per incident ion is observed, corresponding to a net retention of $3.5 \times 10^{20}$ D s$^{-1}$ which is nearly twice as high as the steady-state retention observed on TS for the very long and steady-state discharges [4,24]. This retention rate is 10–100 times higher than expected from laboratory samples [13]. However, although the boron layers had been removed from the molybdenum tiles, some boron still remains in the Mo tiles due to diffusion and its potential contribution to this high retention is still difficult to assess [70]. To explain this strong retention, a high recombination rate is also considered, increased by some impurity layer thickness less than the impacting ion range, such that the pressure build-up inside the surface (higher than in laboratory experiments) could lead to more lattice damage and/or displacement [13].

In AUG, with the full coverage with W PFCs, recent analysis of the gas balance exhibits a saturation of the retention during the steady-state phase of the discharge [22]. It is worth noting that absolutely no boronisation has been performed prior to these first experiments, avoiding possible contribution of the boron in the retention and consequently difficulties for the interpretation of the results. These results suggest that the retention is now dominated by the implantation in the metal facing the plasma and that the co-deposition has been reduced to a very low level [22,71,72]. Finally, although the plasma duration in steady-state is limited to ~2 s, some of the results [22] exhibit equilibrium between the

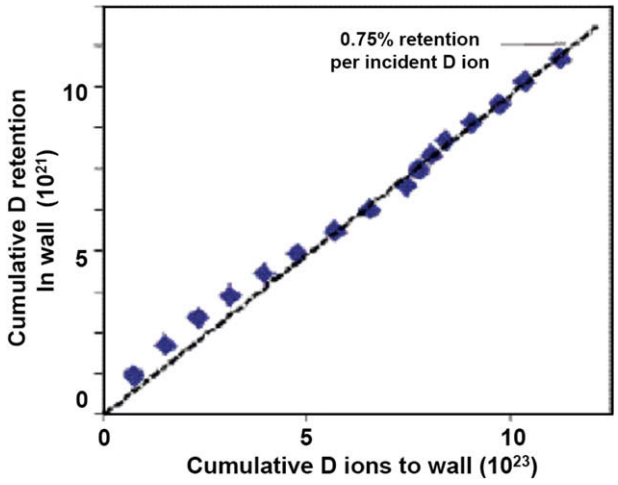


**Fig. 4.** Accumulative D retention in cleaned Mo walls versus incident D ions [69] in Alcator C-Mod.

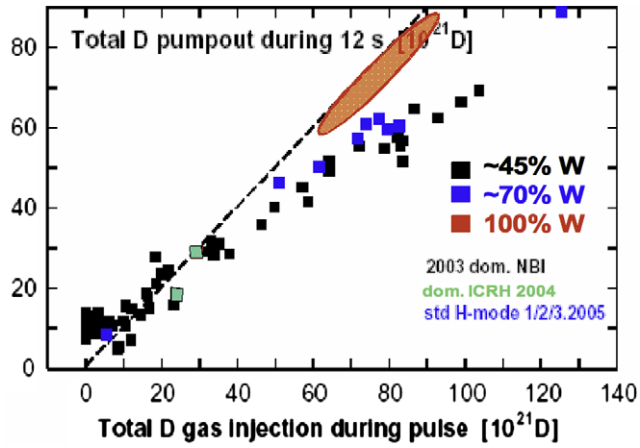


**Fig. 5.** Particle recovery as a function of the total D injected in AUG, for different plasma scenarios and for different coverage by W. From 45–70% the recovery has been improved [23]. The 100% W coverage without boronisation permits almost 100% recovery [22].

injection and the exhaust suggesting that, for some scenarios, there is no retention. Fig. 5 shows the range of retention observed in the recent experiments of AUG with the full W coverage. The stepwise increase of the first wall coverage with W shows that below 70% wall coverage the positive effect of W is not significant. The main effect on retention was observed in the regular drop of the carbon concentration in the SOL, particularly significant when the outer poloidal limiters and antenna protections were covered with W [71,72]; Thus, the primarily sources of C were removed. The contribution of C from the divertor has been assessed [73] and the total amount of gas required to increase the plasma density is about three times higher than in the carbon-dominated machine but the amount of gas recovered after the pulse is also increased by the same factor. This is similar behaviour to that already observed with the Be limiter in JET [21]. However, although the carbon concentration in the SOL is in the range of only ~0.5–1%, the retention in the inner leg region is still dominated by carbon co-deposition, whilst the D retention in the outer leg is dominated by W trapping [72]. Further experiments with boron conditioning in AUG will certainly help in clarifying the possible contribution of the boron to the high retention rate observed in Alcator C-Mod.

## 3. Post-mortem analysis

For long-term retention estimation, post-mortem analysis is a proven method although integrated over a full experimental campaign which reflects the overall retention for an 'averaged plasma scenario' and divertor geometry. However, it also averages over conditioning procedures, disruptions, overheated PFCs, etc. Post-mortem analysis allows, in particular, one to identify the location of the retention by implantation and co-deposition. The implantation and co-deposition can also be studied using test limiters and/or probes that are exposed in the SOL during plasma operation. Such methods permit avoiding a possible contribution of deposition during long periods of conditioning of the vessel and can be compared to laboratory experiments. Recent experiments carried out on a test limiter in TEXTOR [74] confirmed that for CFC materials, the retained fraction does not saturate and increases as fluence$^{0.5}$ while for graphite the exponent is smaller and leads to a saturation of the retention at very high fluence (~$10^{25}$ D m$^{-2}$) [43]. The localisation of the retention is an outstanding issue that significantly contributes to the evaluation of the amount retained and in understanding the process. This location is also particularly

important for the development of efficient cleaning methods. The injection of $^{13}CH$ marker is rather widely used for this purpose and experiments have been carried out and reported in different devices showing different transport related to the plasma scenario in TEXTOR: [48], JET [75], AUG: [59] and in DIII-D [76,77]. However, $^{13}CH$ experiments can be carried out only at the end of an experimental campaign, prior to opening for collecting the samples for post-mortem analysis. These experiments are also performed for a single plasma configuration and scenario to assess the impurity transport and the potential retention in the co-deposited layers. However, they cannot represent the overall behaviour averaged over the campaign which integrates different plasma scenario, conditioning, disruptions, etc.

In carbon devices, the fuel-retention analysis is generally evaluated using carbon deposition analysis by NRA (Nuclear Reaction Analysis) in the PFCs. A summary of the measurements of carbon deposition in JET during the 1999–2001 periods is reported in [51] (Fig. 6) and in [78] for the 2001–2004 period. Although with the septum divertor campaign (1999–2001) the plasma geometry in the divertor was less flexible compared to the SRP phase (2001–2004), these results show similar behaviour. Fig. 7 shows the details of the MKII-SRP divertor and the tile's numbers. In the outer divertor the erosion/deposition is more or less neutral except a narrow band of deposition on tile 6 in the outer pump duct. On the other hand, the inner divertor exhibits strong net deposition over the whole of the target with a large deposit of soft compressible material on tile 4 in the inner pump duct. Flakes formed by thick hydrocarbon films peeling off are also found in the shadowed areas of the inner corner with D/C ratios of 0.7. After the 2001–2004 campaign with the septum removed, further investigations in the divertor and of a poloidal section of divertor tiles have been carried out to assess both the amount and the location of the retained fuel [52]. Similar behaviours for the carbon deposition are observed for the divertor deposition and the D/C ratio. The total carbon deposition at the inner part of the divertor MKII-SRP is

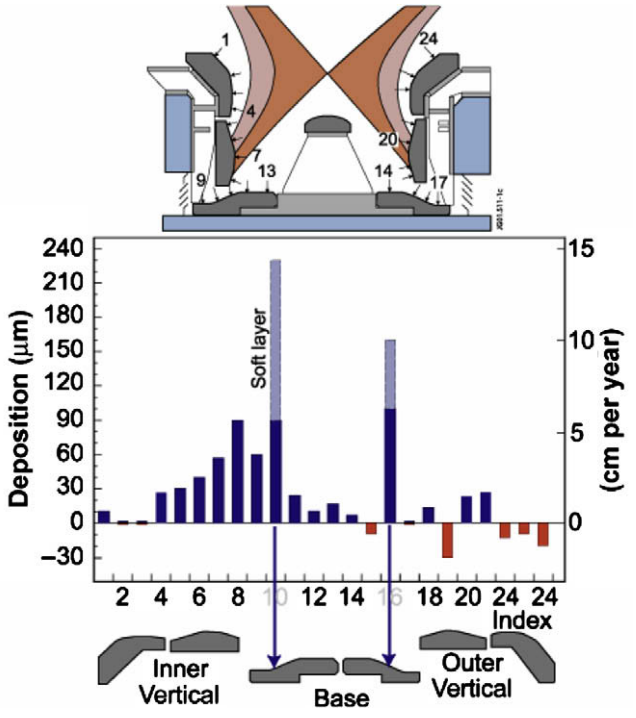


**Fig. 6.** Erosion/deposition in the period 1999/2001, as measured by a micrometer [51]. The results were obtained with the MKII-GB divertor in JET, and show strong deposition on the base plate at the inner and outer regions.

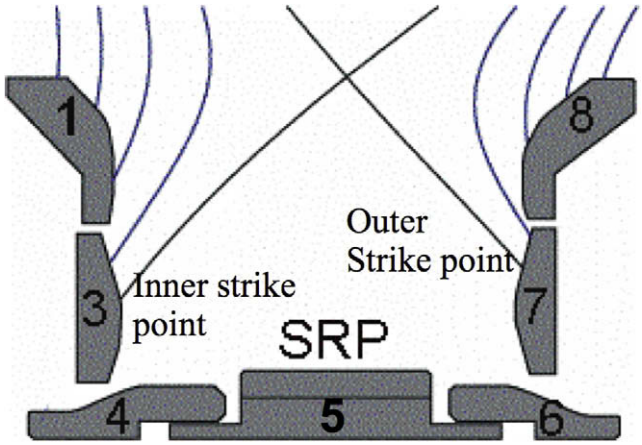


**Fig. 7.** Poloidal cross-section of the MKII-SPR divertor of JET (2001–2004), showing the tile numbering.

625 g and 507 g at the outer part [52]. It is worth noting that a negligible amount of carbon is found on the horizontal part of tile 5 (the so called SRP) in the private flux region whilst on the vertical parts (on both inner and outer parts) of tile 5, a high amount of carbon was found. However, there was also a low carbon deposition on the two adjacent tiles (tiles 4 and 6). These results show that the carbon transport and deposition in the divertor area is averaged over the experimental campaign (including plasma, disruptions, wall conditioning...) and as a consequence, it is very difficult to evaluate the carbon transport for a particular scenario from these results. This is illustrated by the $^{13}CH$ experiments carried out in DIII-D in L-mode [76] and with type III ELMy H-mode [77] showing no deposition in L-mode in the private flux region whilst strong deposition was observed for the type III ELMy H-mode due to a detached inner leg. The evaluation of the D/C ratio in the deposited carbon layers exhibits a D/C ratio in the range of 0.02–0.25 on all the areas exposed to the plasma ion and/or CX neutral fluxes, whilst the D/C ratio is up to 0.91 and 0.79 on the inner and outer shadowed areas respectively. For the 2001–2004 campaign, the overall retention in the divertor area is 66 g of D representing a total of 3.7% (2.6% in the inner region and 1.1% in the outer part) of the total injected D ($5.381 \times 10^{26}$ D). This corresponds to a retention rate of $2.2 \times 10^{20}$ D $s^{-1}$ if averaged over the divertor phase (~94.000 s) or if averaged over the heating phase $4.1 \times 10^{20}$ D $s^{-1}$ (~50.400 s). The analysis of few samples removed from the outer poloidal limiters exhibits a very low contribution of the order of 0.3 g (0.02% of the injection) which is consistent with the previous results from the JET D-T experiments where flakes (~3 g) [58,68] found in subdivertor shadowed areas suggest that codeposition is the dominant process of retention.

The effect of the wall temperature on the retention is one of the outstanding issues for the retention, since at high temperatures the outgassed flux in between discharges could limit the retention in the wall. However, and particularly in the case of carbon as PFC, the chemical erosion is an increasing function with temperature [3,79] which could therefore enhance the carbon source [80] and the associated fuel retention by co-deposition. Careful analysis of carbon deposited in the divertor areas has been carried out in JT-60U after 6 years of operation totalling 8h20 of neutral beam injection (NBI) at 300 °C [66]. A thick carbon layer of ~200 μm and 10 cm extension in the poloidal direction has been observed on the inner part of the divertor, representing a total weight of 550 g. The erosion observed on the outer part of the divertor (340 g) was not large enough to explain the inner deposition, suggesting a main chamber erosion of 210 g. The results of the post-

mortem analysis exhibit a D/C ratio in the 0.01–0.15 range, which is somewhat lower than in JET. This might be due to the 300 °C wall temperature but also to the plasma scenarios run in JT-60U. Indeed, when normalised to the NBI heating time, the carbon deposition in the inner part of the divertor is of the order of $1.0 \times 10^{21}$ C s$^{-1}$, which is nearly 2.7 times higher than for the JET experiments $3.7 \times 10^{20}$ C s$^{-1}$ with the MKII SRP divertor (normalised to the divertor time over the 2001–2004 campaign). This can be the consequence of the operational scenario generally carried out at JT-60U compared to JET. The plasma density is generally lower by a factor of ~2 in JT-60U and consequently the edge temperature $T_e$ is higher by more or less the same factor. This difference can result in a higher carbon sputtering both by physical and chemical mechanism processes due to the higher wall temperature in JT-60U (300 °C) compared to JET (200 °C). Before the long discharge campaign in JT-60U, collector probes had been installed below the divertor structure whilst the vessel temperature was dropped to 150 °C to avoid/limit the strong outgassing from target plates. After 3 years of operation, totalling 2h10 of NBI, the carbon deposition and retained fuel measured with collector probes, located below the divertor, exhibits a lower carbon deposition rate ($8 \times 10^{19}$ C s$^{-1}$) in the subdivertor compared to the inner divertor areas ($6 \times 10^{20}$ C s$^{-1}$). However, there was a higher D/C ratio of ~0.75, which is in the same range as observed in JET during the DTE campaign with the subdivertor structure at 50 °C. Therefore, the higher wall temperature limits the D/C ratio of the PFCs in the 0.01–0.15 range, but this does not seem to limit and/or reduce the retention in the remote areas.

Extensive sample analysis has also been carried out in AUG to study the D inventory in boron (from regular wall conditioning) and carbon, in order to assess the effect of increasing coverage with W in AUG over the successive experimental campaigns since 2002 [63]. When integrated over the toroidal direction, and assuming a uniform deposition, the retention corresponds to 4.1% of the total D injected. Optical inspection indicates that retention in tile gaps is probably small. The W coverage has been increased gradually, step by step, from 45% in 2002 to 80% in 2006. Over all this period, AUG was a carbon-dominated machine with a retention governed by co-deposition of D with C and B on the inner divertor tiles. Seventy to eighty percent of retained the D inventory was found in the inner divertor region and 20% was found in remote areas (below roof baffle, etc.).

The 2007 experimental campaign with 100% W PFCs exhibits a significant drop by ~5–10 of the D vessel inventory. However, and although all the PFCs are fully covered with W, the carbon concentration in the SOL has only dropped by a factor of 2 and is still around 0.5–1%. As a consequence, the D retention in the inner divertor is still dominated by C co-deposition but reduced by a factor of 10–15 compared to previous configurations. The D retention in the outer divertor is dominated by trapping (up to 3 μm) in the W, and exhibiting a drop of the retention by a factor 5–10. From C dominated to 100% W, the total D inventory has dropped by a factor of 5–10, to less than 1% retention [72]. It is worth noting that the first evaluation of the fuel retention carried out with gas balance analysis also exhibits a significant drop [22] and that, although some more accurate calibrations are still required, the preliminary evaluations of the D retention from gas balance and post-mortem analysis gives values below 1%.

## 4. Evaluation of the retention using gas balance and post-mortem analysis

For carbon devices, the long-term retention fraction evaluated from integrated gas balance (global measurement over a limited time range) is generally in the range of 10–20%, depending on the plasma scenario. This value is very often larger than the retention deduced from post-mortem analysis of PFCs, which is ~3–4%, but which corresponds to a local measurement integrated over the full experimental campaign. The errors of the measurements for the two methods are about 10–15% and are certainly lower for Helium and/or T balance, whilst the integrated method used on JET [12] and Alcator C-Mod [13] exhibits an accuracy of the order of 1%. The comparison of the results deduced by the two methods has to be undertaken very carefully. Indeed, in the case of the gas balance it is difficult to determine the total contributions over a full experimental campaign, from D recovery, wall conditioning, disruptions, outgassing over long periods (compared to plasma operation) and the contribution of non-hydrogenic species such as hydrocarbons. Since post-mortem analysis can only be performed for a restricted set of samples, representing only a few percent of the overall surface exposed to the plasma, this method suffers from the extrapolation to the whole device assuming toroidal symmetry. All retention areas such as limiters, large areas in the main chamber and, particularly, retention below the divertor structure as demonstrated after the DT experiments in JET [81,46,49,64] are also very difficult to assess. Finally, the drop of the D content by air exposure of the sample is very rarely discussed although 'venting' is a potential cleaning method for ITER. Indeed, air ventilation in both TFTR [31–33] and JET [68] has been demonstrated to be a potential method to remove the trapped tritium.

On the other hand, gas balance analysis is generally carried out for representative discharges, with injected power and particle rate; ~15 MW and $1.0 \times 10^{22}$ D s$^{-1}$ respectively for typical JET type I ELMy H-modes. However, in JET, for the 1393 successful discharges performed in 2007, the averaged input power was 5.0 MW during the X-point phase and the averaged gas injection in this phase was $\sim 4 \times 10^{21}$ D s$^{-1}$. It is worth noting that both these values are close to those obtained in previous campaigns [82]. If we assume 10–20% retention, as deduced from gas balance [4,12,47], this leads to a retention in the range of $4.0–8.0 \times 10^{20}$ D s$^{-1}$. From post-mortem analysis, the retention in the divertor area and in the first wall (MKII-SRP) is of the order of 3.7% ($2.2–4.1 \times 10^{20}$ D s$^{-1}$) [52]. In this comparison, no mention is given to the drop of D content due to air exposure after the experiments, when removing the samples from the vessel. However, when analysing similar plasma conditions, the gas balance and post-mortem analysis lead to very similar results. The former allows access to the 'quantity' trapped, the latter to 'where' it is trapped, showing that these methods are complementary.

## 5. Extrapolation to ITER

The present material choice for ITER is beryllium (Be) on the main vessel walls, tungsten (W) on the divertor upper baffles and dome, and carbon fibre composite (CFC) around the strike points on the divertor plates. This choice results from the trade-off between the attempt to reduce the tritium retention, which must be limited for safety reasons, to avoid too much dust and to extend as much as possible the lifetime of the PFCs. For the extrapolation to ITER, the intermixing of the different materials and their influence on hydrogen retention and co-deposition is a major source of uncertainty on present estimates since except, Alcator C-Mod and AUG, the major part of the results on fuel retention were obtained in carbon machines. The extrapolation to ITER has been carried out assuming implantation, trapping and diffusion in plasma-facing materials and surface erosion and co-deposition of tritium with eroded material. The foreseen ion and CX fluxes distributions are calculated with B2-EIRENE [83] and give a total wall flux of $1–5 \times 10^{23}$(D + T) s$^{-1}$ and a divertor flux of $3 \times 10^{24}$(D + T) s$^{-1}$. The tritium inventory accumulation is given for the present

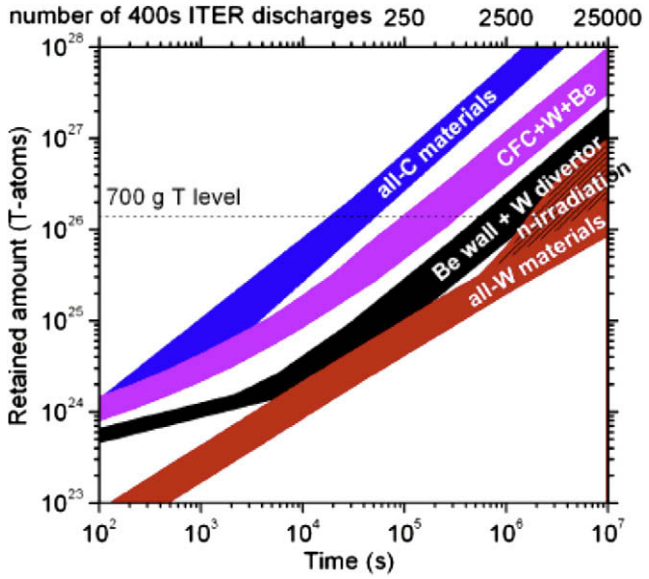


**Fig. 8.** Tritium inventory in ITER for the all-C (blue line) and all-W options (red line). In addition, retention values for the option of a full-W divertor and Be first wall are included (dot-dashed line) [9,11]. (For interpretation of the references to colour in this figure legend, the reader is referred to the web version of this article.)

material choice, as well as for a full carbon machine and a full W device [9,11] and is summarized in Fig. 8.

Present estimates, without any cleaning included, indicate a build-up of the tritium inventory up to the operation limit of 700 g within ~250 nominal full power discharges, with carbon co-deposition being the dominating process for the present material choice (Be, W and C). The all-carbon machine would reach this limit in only 30–40 discharges, still due to carbon codeposition, enhanced by the overall carbon source resulting from the first wall. By removing all the carbon, and with a W divertor and Be first wall, the gain appears to be about a factor of ~8 compared to the present material choice. The benefit comes from a lower D/Be ratio compared to the D/C ratio, whilst the amount of Be eroded and co-deposited with D is roughly similar to C or Be walls. It is also worth noting that the D trapped in Be layers can be released at a relatively low temperature (~350 °C) whilst this temperature has to reach ~1000 °C for the same response in carbon. Finally, the all-W machine eliminates co-deposition and allows many more discharges, although the neutron damage needs to be considered and would certainly reduce the number of discharge to around 2500.

## 6. Summary

Gas balance and fuel-retention analysis carried out in carbon devices show that the values obtained for long-term retention are always in the 10–20% range. In carbon-dominated devices, the long-term retention depends on implantation and co-deposition. The long-term retention by implantation in the surfaces wetted by the plasma can be significantly reduced, and even de-saturated, if high surface temperatures (>1000 °C) are reached, as shown in JT-60U on the divertor targets in long heated discharges. However, the hydrocarbons released from these overheated targets represent an additional carbon source, increasing the carbon concentration in the SOL which, therefore, could enhance the retention by carbon co-deposition. Indeed, the retention by co-deposition is demonstrated to increase with the carbon production, due to the plasma recycling flux and ELMs, and to be proportional to the discharge duration. In all carbon devices, the long-term recovery by outgassing in between discharges saturates and represents a weak contribution over a full campaign. This was demonstrated during the repetitive series of long discharges recently carried out in TS, as was previously observed in both TFTR and JET (for shorter discharges), where the tritium recovery by outgassing during the D-T campaign was low. From analysis of long discharges in TS, and D-T experiments in TFTR and JET, the overall recovery as a result of disruptions has been found to be weak. By moving to high $Z$ PFCs, the objective is to cancel the retention from co-deposition, due to reduced erosion. However, in Alcator C-Mod, although the tiles were apparently cleaned from the boron remaining from previous wall conditionings, the Mo exhibits retention proportional to the discharge duration, at a rate which is comparable to the retention observed in carbon devices. The dominant retention process is not fully understood yet, but shallow D implantation/deposition is suggested since the recovery arising from disruptions at the end of the pulse yields at least the number of particles trapped during the pulse. However, traces of boron in the surface of cleaned molybdenum tiles could be largely enough to explain this high retention. Indeed, one of the main issues for the interpretation of the fuel retention in Alcator C-Mod is the possible effect of the remaining boron in the molybdenum structure, due to intense boronisation in previous campaigns. Contrary to Alcator C-Mod, the full coverage of the PFCs with tungsten in AUG has demonstrated that the long-term retention by co-deposition was cancelled and that the resulting retention was dramatically reduced to ~1% or less, which represents a factor of ~7 lower than the retention deduced by gas balance. These experiments have been performed without any boronisation and this could be the major difference explaining the factor results obtained in these two metallic machines, Alcator C-Mod and AUG.

From post-mortem analysis, the retention in carbon devices is in the range 2–4% for the divertor machines (AUG, JET, JT-60U) and slightly higher at 8–10% in limiter devices (TEXTOR and TS). The long-term retention by implantation in the PFCs is demonstrated to be low compared to that from co-deposition. For metallic devices, the recent post-mortem analysis of the long-term retention in AUG with a full tungsten configuration, confirms the dramatic drop of retention by a factor of 5–10 compared to a carbon-dominated configuration. However, the carbon concentration in the SOL is still around 0.5–1% and the D retention in the inner divertor is still dominated by C co-deposition, although reduced by a factor of 10–15. The D retention in the outer divertor is dominated by trapping (up to a depth of 3 μm) in the W structure but also exhibits a drop by 5–10 compared to carbon. From C dominated to 100% W, the total long-term D inventory evaluated by post-mortem analysis has dropped by a factor 5–10 to a retention of less than 1%.

The main conclusion is that the long-term retention in carbon devices is dominated by co-deposition and is, therefore, proportional to the discharge duration. As long as the carbon source is not cancelled, it appears that there is no possibility to significantly reduce the built-up inventory by co-deposition independently of the plasma scenario and of the wall temperature. The extrapolation to ITER shows that, in a full carbon configuration, the maximum limit of T retention would be reached in a few discharges. With the actual mixed material (C, Be and W), the retention would be less and about 250 full burning $Q = 10$ discharges of 400 s would be possible. The benefit mainly results from the fact that the D/Be ratio is lower than that for D/C whilst the amount of Be eroded and co-deposited with D is roughly similar to that D/C with C or Be walls. Finally, a gain in the total number of discharges is projected in terms of retention, only if the carbon is removed; the Be and W materials allowing for a minimum of 2500 discharges which is close to the full W configuration, including potential neutron damage.

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