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ABSTRACT

Starting with a wall loaded by H_2, change over experiments from H_2 to D_2 have been carried out in JET-ILW. A series of 13 repetitive pulses (cumulating 215s in divertor configuration) have been performed under conditions of: I_p = 2.0MA, B_T = 2.4T, $\langle n_e \rangle = 4.5 \times 10^{19} m^{-3}$ with a constant gas injection of $3.0 \times 10^{21} Ds^{-1}$ and 0.5MW of auxiliary heating by ICRH in L-mode. Gas balance analysis shows that the total amount of H removed from the wall is in the range of 3×10^{22}D compared to $2x10^{23}$D for JET-C. This is consistent with the faster decay of the H plasma concentration and the drop of the retention also by a similar factor when removing all the carbon components.

Isotopic plasma wall changeover is also demonstrated to allow for removal of some D/T from the device. However, since plasma change over also contributes to long-term retention by codeposition, in ITER, change over in between each discharge might not be effective to reduce the fuel retention on the long-term.

1. INTRODUCTION

Particle control is a critical issue for the next step machines like ITER: particle injection and extraction systems must regulate the D-T fuel densities, exhaust helium ash, and particularly minimize the tritium vessel in order to sustain the fuel cycle. The long-term T retention constitutes an outstanding problem for ITER operation and dedicated experiments have been carried out in JET ITER-Like Wall (JET-ILW) metallic configuration (first wall in Beryllium and divertor in Tungsten), to assess the long term fuel retention in this carbon free configuration. The main results [1–3] show that the long term fuel retention rate is around $1.5 \times 10^{20} Ds^{-1}$ (0.2 to $1.5 \times 10^{20} Ds^{-1}$ depending on the plasma scenario) with respect to divertor operation. Although the long-term retention has been significantly reduced in JET-ILW compared to carbon (JET-C), it is still non negligible and it represents potentially a limit for further high performance plasma scenario with large gas throughput and flux to the main PFCs [4]. In ITER, a rate of $1.5x10^{20} Ts^{-1}$ would lead to a long-term retention of 0.3g per discharge of 400s and require regular cleaning

In these conditions, T removal methods have to be developed, qualified and assessed for extrapolation to ITER. In present fusion devices operating in deuterium, a part of the long term fuel retention has been found to be accessible for removal by He plasma operation or by hydrogen isotope exchange [5, 6]. In this context, in ITER, it is considered to use the end of each discharge for reducing the amount of tritium trapped in the device. He or H_2 injection is envisaged during the ~ 200s following the burning phase when auxiliary heating is turned off and when the plasma current, I_p, ramps down. In this frame, series of experiments have been carried out in JET-ILW to compare the effectiveness of various fuel recovery techniques: isotopic exchange, Ion Cyclotron Wall Conditioning (ICWC) [7, 8] and Glow Discharge Cleaning/Conditioning (GDC).

In this paper, isotopic plasma wall changeover experiments carried out in JET-ILW are reported. The objectives were to determine the amount of particle accessible by changing the plasma from H to D and to evaluate the efficiency of this method for controlling the in vessel long-term inventory.

The isotopic exchange experiments are described in the first part and the global particle balances for both H and D are reported in the second section. The discussion on the accessible particle inventory through isotopic exchange and the efficiency of this method at the end of each ITER discharge for possibly reducing the long-term retention are reported in the last section.

2. ISOTOPIC EXCHANGE EXPERIMENTS

The reported experiments have been carried out after two days of H_2 ICWC [8]. Following this wall conditioning with H_2 concentration up to 75%, the experiments started with 5 pulses in pure H_2 in order to reach a plasma isotopic ratio $n_H/(n_H + n_D) > 95\%$. Then, the change over was carried out through 13 consecutive "pure D_2" discharges allowing to reach $n_H/(n_H + n_D)$ in the range of ~ 5% (~ 150 s in × point).

For all the pulses of this experiment, the plasmas have been performed under the same conditions of: $I_p = 2.0$MA, $B_T = 2.4$T, $\langle n_e \rangle = 4.5 \times 10^{19} m^{-3}$ with a constant gas injection of $6.0 \times 10^{21} Hs^{-1}$ and $3.0 \times 10^{21} Ds^{-1}$ (respectively for H and D pulses) and 0.5MW of auxiliary heating by ICRH in L-mode. Active pumping was ensured by the divertor cryogenic pump (DP) only whilst the D and H particle balances were evaluated over the entire experiment through the analysis of the gas recovered by DP regeneration at the end of the session. Figure 1 shows the main plasma parameters as a function of time for the last "pure" H discharge (Pulse No: 85143) and the first "pure" D discharge (Pulse No: 85145). The limiter phase duration is of 10s whilst the outer and inner-strike points were moved from vertical to horizontal divertor targets from high (0.31) to low (0.21) triangularity plasma shape respectively. These different plasma shapes in the divertor region allowed for a plasma interaction with the two inner vertical targets of the divertor (so-called tiles 1 and 3) where most of the plasma-facing side deposits are located [9]. From figure 1d), it can be seen, that as soon as the X point is formed and when the pumping through DP is effective at 10s, the isotopic ratio drops dramatically from 95 to 20-40% (respectively for main plasma and the particles in the sub-divertor region) for the high triangularity phase and stabilises around 35-40% for the following phases.

Figure 2 shows the $n_H/(n_H + n_D)$ plasma and subdivertor isotopic ratio as a function of the pulse number during the low triangularity phase at 18s. It can be seen that the ratio drops very quickly from 95% to ~ 35-40% for the first "pure H" ("only" 8s in divertor configuration) and below 10% after 5 discharges cumulating ~ 80s in X-point. Finally, the ratio drops below 4.5% after 13 pulses (150s in divertor). However, it is worth noting that this drop is strongly penalised by the very poor pumping speed of the H_2 by the DP at very low neutral pressure. Indeed, from calibration performed without plasma and with calibrated gas injection, it is shown that, with respect to a neutral pressure measurement in the torus (Penning gauge) the H_2 pumping speed drops by a factor ~ 7.5 (from ~ 120 to ~ 16 m^3s^{-1}) when the partial pressure of H_2 is below ~ 3.0×10^{-3}Pa due to the working temperature of the He panel (4K) in the divertor cryopump. This is not the case for the D_2 which exhibits a constant pumping speed of $210 m^3s^{-1}$ with respect to the same pressure gauge. In these conditions, the H "removed" from the plasma facing components (PFC) by isotope exchange is poorly pumped by the DP explaining the very slow decrease after the 4–5th "pure" D discharges.

3. GAS BALANCE

The analysis of the gas balance performed through the regeneration of the DP after the session leads to the following balance. The total amount of particles injected during the plasma operations represents 6.853×10^{23}D and 4.592×10^{23}H (total of 23.7 Barl (@299K). The amount recovered through the DP regeneration is 6.495×10^{23}D and 4.251×10^{23}H (composed of 37% D_2, 46% HD and 17% H_2). The total amount of particles retained in the vessel is 3.58×10^{22}D and 3.41×10^{22}H. For the overall session, the retention rate averaged over the total divertor operation is ~ 2.8×10^{20} D or H s^{-1} which is about a factor 3 above the results obtained during the previous experiments (0.5–0.8 x10^{20}Ds^{-1}) [1, 2] with the same plasma scenario.

When considering separately H and D behaviours, the D retention, over the ~ 190s (X-point duration multiplied by the plasma isotopic ratio) in divertor configuration for the "pure D discharges" is ~ 1.9×10^{20}Ds^{-1}. For the 5 "H pulses", the duration in X-point can be evaluated around ~ 100s also by taking into account the plasma duration corrected by the proportion of H in the discharge. In these conditions, a retention rate of 3.4×10^{20}Hs^{-1} results which is a factor of ~ 2 above D and about 5 compared to retentions obtained for the same plasma scenario previously quantified [1, 2]. This high retention observed for H compared to D can be explained through the very low pumping speed of H_2 for neutral pressure below ~ 5×10^{-3}Pa in the subdivertor region (drop by about 7.5, see above). This low partial pressure is reached after typically 4 plasma discharges and as a consequence, since H_2 is "poorly" pumped by the DP, the codeposition process is the only remaining pump.

The evaluation of the accessible particle reservoir through confined plasma isotopic exchange is determined through two methods. For the first one, it is considered that the retention rate is 0.5–0.8 $\times 10^{20}$Ds^{-1} as deduced from previous experiments [1, 2] with the same plasma scenario. In these conditions, total retention of 0.8–1.3 $\times 10^{22}$D codeposited results. The additional retention observed for these experiments (determined from the difference between the total retention and the evaluation of the codeposition) is ~ 2.3–2.8 $\times 10^{22}$D. This averaged value of 2.5×10^{22}D represents the extra amount of D retained in the vessel for replacing the H representing the accessible particle reservoir for JET-ILW.

The second method is based on the global particle balance for H, supposing that the H is mainly removed from the vessel only for the first 4 pulses in "pure D". In these conditions, although no H is injected, the H removed by the DP during these 4 pulses is also evaluated in the same range of 1–3 $\times 10^{22}$H. Although this method is not as accurate as the first one, the resulting range is very close. Using these two methods, is can be deduced that the accessible particle reservoir for JET-ILW is of ~ 2.5×10^{22}D which has to be compared to the ~ 2×10^{23} obtained for JET-C [10, 11].

4. DISCUSSION

These results are compared with the T-D change over experiments performed at the end of the first phase of the DT campaign [10, 11] in JET-C with the MKII-A divertor. These experiments were performed under ohmic conditions with: I_p = 2.5MA, B_T = 2.5T, $\langle n_e \rangle = 4.5 \times 10^{19} m^{-3}$ with

gas injection ranging from 2.0 to 5.0 × $10^{21}Ds^{-1}$. The corresponding plasma isotopic ratio for these experiments is plotted on figure 3 with the recent results from JET-ILW. From this series of experiments, the amount of T removed from the wall has been evaluated in the range of 2 × 10^{23}D with 10% of T concentration in the bulk plasma after 10 discharges [10, 11]. In terms of fuel removal, the amount of hydrogen removed from the JET-ILW, is typically 10 times lower compared to the JET-C which is consistent with the drop of the long-term retention also by a similar factor when removing all the carbon components and the faster decay of the H plasma concentration.

The saturation of the plasma isotopic ratio in the range of 10% observed for the JET-C configuration (see figure 3) is not due at all to a potential drop of the DP pumping speed which remains constant for both D and T species independently of the neutral pressure. This saturation can be explained by the exchange with the deposited layers in the inner divertor region representing a huge particle reservoir for the time constant considered for these experiments (15 discharges cumulating ~ 220s in divertor). Indeed, for carbon devices, it has been shown that the D (T) removed from the vessel through isotope exchange originate from the areas exposed to both ion (strong plasma wall interaction) and charge exchange (CX) flux (thin, thick co-deposited or implantation areas) [12]. However, the D (T) atoms and molecules removed from these areas are transported through the SOL to the DP, but a significant part of these atoms can also experience codeposition in shadowed areas. In the case of the intensive cleaning and T removal campaign carried out in JET-C at the end of the DT campaigns, the 3.0g of the long term retention were located below the divertor structure in areas not exposed to ions, neutrals and/or neutrons [13].

For the reported H to D experiments, since codeposition also takes place with beryllium, but with lower fuel content [14], some of "these" H removed from the implanted and codeposited areas will also be retrapped and, thus, experience codeposition in areas difficult to access by ion and CX flux through confined plasma scenarios. This phenomenon is obviously much more pronounced in a carbon environment where on the one hand codeposition is very significant and on the other hand codeposition also takes place in areas difficult to access (particularly below the divertor structure) owing to the high chemical sputtering of and multistep transport. However, for both JET-C and JET-ILW, as codeposition process takes place, these particles contribute to the increase of the long term retention as discussed below. This overall process can be summarised as displayed on figure 4. The D, T injected through gas, pellets or neutral beams experiences ionization and transport in the scrape-off layer (SOL) before interacting with the PFCs. Three main processes result from this interaction: implantation, codeposition with C, Be…and neutralisation of these atoms mainly released in D_2 and T_2. In this context, all these particles enter the recycling process and except the particles exhausted by the pumps and those codeposited in region poorly exposed to ion and/or neutrals from CX, all the other particles go back to the SOL and the same ionization process takes place again. When H_2 is injected (independently of the fuelling method), these particles experience these particles are ionized, transported through the SOL and interact with the PFCs where the same processes and isotopic changeover takes place. The D and T "exchanged/replaced" from implanted

(ion and CX) and accessible codeposited areas (by CX) by H, have access to the SOL where they are ionised and transported. As a consequence, a part of these D and T can indeed be removed from the vessel by the pumps, but some of these D and T are captured by Be and codeposited. The impact of the latter has been strongly reduced by removing the carbon from this loop, but the codeposition still represents a non-negligible contribution with the materials used the ILW configuration [9].

Although a faster change over is observed in JET-ILW compared to JET-C, the removal efficiency through confined plasma and isotopic exchange will always also contribute to some extend to the long term codeposition. Although it can be impacted by the divertor plasma configuration (through sweeping), codeposition and associated retention is weakly dependent of the plasma scenario since this removal is achieved only through SOL transport. In JET-C, this long term retention reservoir (located below the divertor structure) has been demonstrated to be inaccessible through all the cleaning procedures, including isotopic exchange, intensively applied after the DT campaign [13]. With the JET-ILW, the material migration pattern differs with respect to C in the last step as the multistep transport to remote areas is strongly reduced due to the absence of low energetic chemical sputtering of Be [15]. Therefore, Be is predominately deposited at the top of the inner divertor entrance on the high field side [9]. Though co-deposition dominates here, physical sputtering and strike-point movement transferred some of the Be via line-of-sight towards the divertor floor where codepositon takes place at the inner and outer base divertor tiles and remote divertor corners. The magnitude in deposition is by far one order of magnitude lower than in JET-C [9]. Supposing that these areas are accessible through confined plasma operation for fuel removal, it is clear that although a part will be exhausted by the pumps, the other part will be redeposited away in non-accessible region by plasma ion and/or CX. However, further investigations with this Be/W mixed material have still to be performed to identify the final areas of codeposition and the residual after cleaning.

Finally, although the long term retention by codeposition has been significantly reduced by removing the carbon and by lower fuel content in Be layers, the main consequences of these results on ITER would be that plasma discharges contribute to the long-term retention and can, if one considers the necessary refueling of the wall in the next discharge, eventually even increase it rather than decrease. Indeed, if change over is performed at the end of each ITER discharge, although some D-T will be remove, some D and T will also be codeposited. At the end of the pulse, the net T vessel inventory will be reduced although some of the initially removed T is codeposited. But, for the next pulse, during the preliminary phase prior to the auxiliary heating phase, compared to a situation without change over, additional D-T will have to be injected [12]. Indeed, this extra injection will be required for reaching the target isotopic ratio for the burning phase and for replacing/compensating the removed H. During this phase, retention will also take place and D-T will be codeposited due to this extra injection. In ITER, if this retained fuel by codeposition is not accessible through dedicated cleaning methods (ICWC), change over during the ramp down would be counterproductive in reducing the net T vessel inventory.

5. SUMMARY

A series of change over experiments from H to D has been carried out in JET-ILW for evaluating both the accessible particle reservoir and the efficiency of this method for possibly reducing the amount of tritium trapped in fusion devices (long term retention). The results show that there is a better control (faster decay) of plasma isotopic ratio for JET-ILW compared to JET-C and that the inventory accessible through isotope exchange ~ 3×10^{22} (2×10^{23} for JET-C). These results are consistent with the lower retention observed when reducing the codeposition through removing the carbon from plasma facing components.

Plasma isotopic exchange also allows for D/T removal from implanted area and from some part of codeposited areas. However, since this removal can be carried only through SOL transport, codeposition process will also occur to these atoms and contribute the long term retention. In ITER, if this retained fuel by codeposition is not accessible through dedicated cleaning methods (ICWC), change over at the end of each discharge during the ramp down would be counterproductive in reducing the net T vessel inventory.

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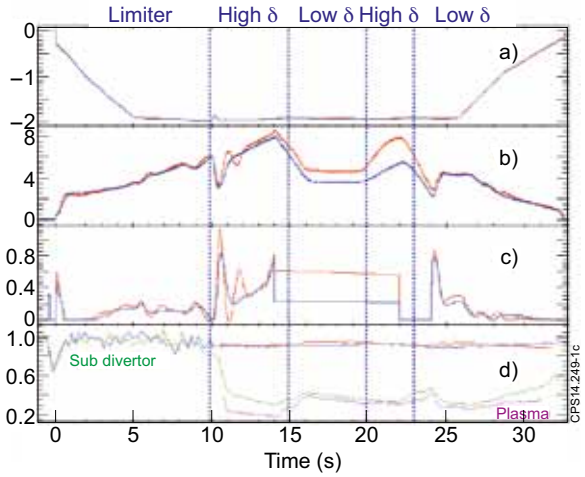

Figure 1: Main plasma parameters for the pulses performed for the change over experiments. The red plots correspond to the last "pure" H pulse (Pulse No: 85143) and the blue plots correspond to the first "pure "D pulse (Pulse No: 85145). The main phases of the plasma discharge are indicated from limiter to high and low triangularity (δ). a) plasma current (MA), b) plasma density ($10^{19}m^{-2}$) c) D injection rate ($10^{22}es^{-1}$) and d) isotopic ratio $n_H/(n_H + n_D)$ measured in the bulk plasma (blue and red for the last "pure H" discharge) and in the subdivertor region (green and purple for the first "pure D" discharge).

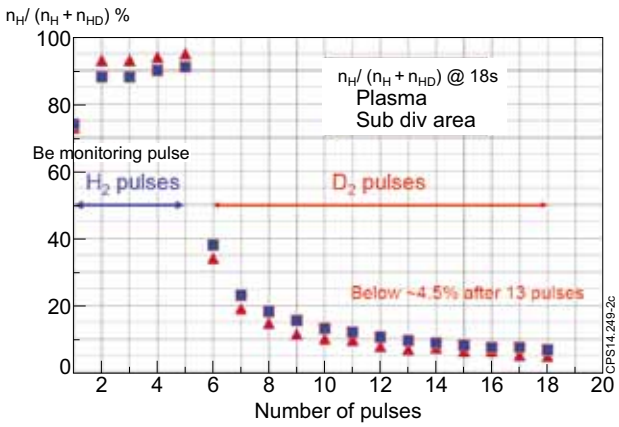

Figure 2: Isotopic ratio $n_H/(n_H + n_D)$ measured @18s during a low triangularity phase as a function of the pulse number, measured in the bulk plasma (red triangles) and in the subdivertor region (blue squares).

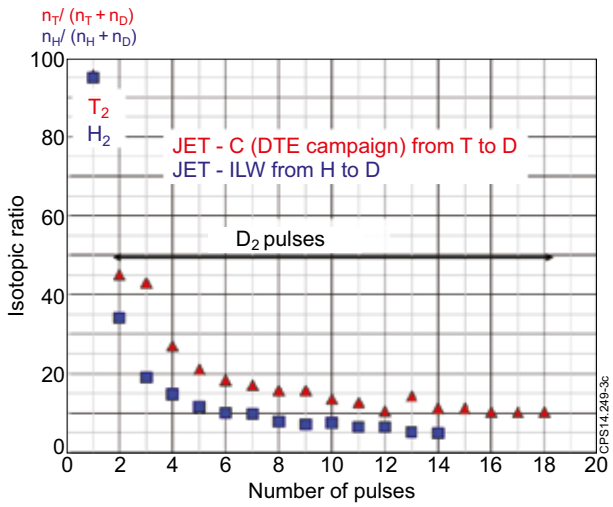

Figure 3: Plasma isotopic ratio $n_T/(n_T + n_D)$ for the JET-C (red triangles) for T to D change over and $n_H/(n_H + n_D)$ for the JET-ILW (Blue squares) for the H to D change over experiments. Although the pumping speed of the DP is lower for H, a faster decay and a better control of the plasma isotopic ratio is obtained for the JET-ILW compared to JET-C.

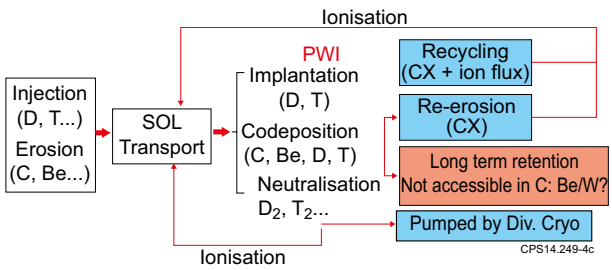

Figure 4: Schematic description of the global recycling processes taking place from the particle injection, the ionization and transport in the SOL and the main plasma wall interaction processes. On This view, the long term codeposition appears as a reservoir that can be enhanced through change over experiments (D/T to H) or from D/T to He.