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ABSTRACT

In fusion devices a part of the hydrogen retained by implantation has been demonstrated to be accessible during plasma operation and can be removed from the wall by He or isotope exchange with plasma. Starting from an empty D_2 and H_2 wall after two weeks of He campaign series of change over experiments from He to H_2 have been carried out in JET. Hydrogen plasma concentration up to 85-90% has been reached after only 7 pure H pulses for a total of ~210s of plasma operation whilst an accessible wall inventory of ~2.0×10^{23}D (~1g of T) has been evaluated. Extrapolation to ITER shows that accessible T content would be 21.3g, 2.63g and 1.63g respectively for C, C/Be/W and Be/W as materials facing the plasma in ITER. Finally, these quantities are "small" compared to retention by co-deposition which will dominate the retention with time since this is a cumulative process.

1. INTRODUCTION

In fusion devices, fuel retention results from co-deposition, long term and dominant in tokamaks with plasma-facing components made of C, implantation by ion impact (Long and short term retention) and dynamic retention [1]. Indeed, co-deposition takes place in shadowed areas and protected from plasma radiation, plasma ion flux and neutrons and therefore impeding cleaning/removal process for retained tritium. At this stage, it is worth mentioning that the dynamic retention is a very short term retention which is characterized by the ability of the material exposed to plasma to store a "small" additional amount of H, even if material is saturated, during plasma and released as neutrals after the discharge whilst the particle injection by gas, neutral beams or pellets also allows an additional way for controlling the plasma isotopic ratio. In present fusion devices (mainly carbon), a part of the hydrogen retained by implantation has been demonstrated to be accessible during plasma operation and can be removed from the wall by He or isotope exchange with plasma. Indeed, following the DT campaign at JET in 1997-1998, up to 5.3g of the 11.5g retained at the end of the DT campaign have been removed after an intense cleaning campaign (He, H, D plasmas, Glow discharges…) carried out over about 8 months [2]. The resulting T inventory in the plasma facing components (PFC) has been found to be small (~0.2 g – 4×10^{22}T) compared to the co-deposited quantity located below the divertor structure (~3.0 g – 6×10^{23}T). The next step fusion devices will be metallic machines (Be/W) where the global retention by co-deposition with Be will be still dominant compared to implantation process; only in the case of a full W device, the retention will dominated by implantation as observed in ASDEX Upgrade [3]. In ITER, assuming that no carbon will be used in the activated and DT phases, the retention is also foreseen "only" on PFCs since no co-deposition is supposed to take place in shadowed areas. However, although no Be has been found in shadowed areas, it is worth mentioning that no experimental results have been obtained so far in a tokamak equipped with Be/W as main PFCs, this will be obtained in the JET ILW experiment. In ITER, He plasma and/or isotopic exchange are considered method for minimising the tritium inventory. In order to evaluate the amount of particles involved in this process, change over experiments with He to H_2

have been carried out in JET, as a reference in Carbon for comparison with the new Be/W wall in JET and for extrapolation to ITER

2. EXPERIMENTS

The reported experiments have been carried out after two weeks of "pure" Helium operation, cumulating 198 plasma pulses (I_p>0.5MA) for a total plasma duration of 1h36mn divided in 33min in limiter phase on the outer and inner guard limiters (I_p ramp up phase without auxiliary heating) and 1h03mn in divertor phase including additional heating for 4-5 s up to 13MW. In these conditions, the wall was considered as empty of D and/or H, whilst the vessel temperature was at 200°C and no vessel conditioning was carried out prior to the change over experiments. Two reference discharges in pure He have been performed at the beginning of the session immediately followed by 7 "pure" H pulses. The main plasma characteristics of these discharges were: I_p/ B_T=1.8MA/1.8T, 1.6-3.8MW ICRH (51MHz – 2nd harmonic H) and a constant gas injection of ~4.0x10^{21}Hs^{-1} during the heating phase. Figure 1 shows the main plasma parameters as a function of time for a representative H plasma discharge (# 79243) performed at the end of the session (7th pulse in pure H). Compared to regular operations, the limiter phase duration has been extended from 8 to 14s, followed by a diverted plasma with low triangularity for 4s (Inner and outer strike points located on the horizontal tiles of the divertor) and finally a high triangularity phase for 8 s (Inner and outer strike points respectively located on the vertical and bottom central tile of the divertor). ICRH was used as auxiliary heating generating very small type III ELMs of typically 10KJ of energy. All auxiliary pumps (NBI, LH) were off, except the divertor cryopump and the torus turbo pumps for helium pumping (not pumped by the divertor cryopump during the X point phase). A regeneration of the divertor cryopump has been performed at the end of the session, after the 7 H pulses, allowing the integral of the gas pumped by the divertor cryopump to be evaluated for the reconstruction of the particle fluxes during plasma and in between discharges. Figure 2 shows the He/(He+H) plasma ratio as a function of the pulse number during the heating phase at 63s. It can be seen that the ratio drops very quickly to ~35-40% even for the first "pure H" pulse demonstrating that a very rapid change over to H operation occurs in 1-2 pulses (less than 30s of diverted plasma). It is worth noting that for the first pulse in pure H, the ratio was always below 65% and even as low as 30% during the 3s of transition phase from low and to high triangularity. After 4 pulses, (56s in limiter phase and 84s in divertor phase), it can be also observed a saturation of the ratio around ~10%. This is the result of He implanted in carbon and of the absence of helium pumping by the pumped divertor during the X point phase.

The global H balance for the series of the 7 H discharges exhibits a total injected of 3.868x10^{23} D+H (6.78x10^{21} of D being injected for the plasma breakdown) whilst the total recovered from the pumped divertor and exhausted by the turbo pumps is 1.539 10^{23} D+H. This total amount is divided in 58% of H_2 (4.46x10^{22}) 31% of HD (2.39x10^{22}) and 11% of D_2 (8.5x10^{21}). The total of 4.09x10^{22} D pumped represents 6 times to the D injected and this demonstrates that long term D

release (heating of layers, diffusion, migration and long term outgasing) is still significant after more than two weeks of operation in the absence of D fuelling. The total retention is 2.329×10^{23} D+H representing about 60% of the total injected. This retention is the result of both the implantation and co-deposition and the two processes have to be separated. From previous experiments dedicated to fuel retention analysis, in similar plasma conditions, the long term retention (co-deposition) was found to be ~17% of the total injection. Using this value as a reference for all the discharges and using the He/He+D ratio as weight for taking into account the proportion of H in the plasma interaction with the wall, the contribution of the codeposition and the implantation has been calculated and the results are displayed on figure 3. A regular drop of the retention due to implantation is observed and at the end of the 7 pulses, a total of 1.70×10^{23}H has been retained in the wall by implantation. However, and as observed from figure 3, after 7 H pulses, the wall retention by implantation is not yet in a steady state phase and a rough extrapolation shows that 3 additional pulses would be necessary. In these conditions, the wall retention by implantation is then evaluated in the range ~ 1.9-2.0x1023H. The cumulative plasma duration for these 10 discharges would be 350s (140s in limiter and 210s in divertor) extrapolated from the 245s (~4 mn) of the 7 H discharges with 98 s in limiter phase and 147s in diverted configuration. The retention by implantation observed during these series of experiments corresponds to an average exchange of ~0.5×10^{21} H s^{-1} with the ~100m^2 of surface facing the plasma over the total discharge duration and of 0.95×10^{21} H s^{-1} if averaged over the diverted phase. It is worth noting that similar values were deduced from DTE gas balance analysis [4, 5] and D to He plasma wall change over experiments [6]. The wall particle reservoir that is accessible by isotopic exchange after the first phase of pure tritium operation in JET has been evaluated to about 2 x 10^{23} for hydrogenic species, deduced from T to D change over experiments performed at a vessel temperature of 320°C.

3. DISCUSSION AND EXTRAPOLATION TO ITER

Evaluating the total carbon area in JET around ~ 100 m^2 (divertor, bumpers and all structure away from the main plasma but exposed to neutral charge exchange flux), this would correspond to a maximum retained fluence of ~2.0×10^{21}m^{-2} which is consistent for implantation of particles with incident energy of 200eV before acceleration in the sheath. Figure 4 shows the number of D atoms retained as a function of the incident ion fluence [7] without any effect of neutron irradiation. From this plot, similar evaluations have been carried out for limiter machine for long discharges in Tore Supra with the inner wall as main limiter (prior to year 2000 with a total carbon area ~50m^2) and with the toroidal pumped limiter (new configuration for plasma facing components and with a total carbon area of ~20m^2) [8]. The number of particles accessible in Tore Supra is ~8.0×10^{22}D and 3-4x10^{22}D respectively, which is consistent with previous experiments and evaluations carried out in Tore Supra. Using this plot, an extrapolation to ITER configuration (mixture of 50:50 D:T) has been performed for three different configurations of materials: full C, C/Be/W and Be/W with 700m^2 for the first wall, 100m^2 on the divertor and baffle and 50m^2 on the vertical targets of the

divertor. For the full carbon configuration, retention by implantation is known to be proportional to fluence1/2, but recent results [9] show that for high fluences (>10^{25}ions m^{-2}), saturation at a value of ~$10^{22}Dm^{-2}$ occurs. Using this base, a total of $8.5x10^{24}$ D+T (respectively 14.2+21.3g) would be implanted in the 850m² of ITER carbon wall and accessible with He and/or isotopic exchange. For the configuration foreseen for the non activated phase, C/Be/W, assuming the same saturation for carbon, $5x10^{20}Dm^{-2}$ for Be and $2x10^{21}Dm^{-2}$ for W, a total of $1.05x10^{24}$ D+T (1.76+2.63g) would result. Finally, by removing all the carbon, the Be/W configuration would lead to an implantation of $6.5x10^{23}$ D+T (1.09+1.63g). Although retention by codeposition will still dominate with Be, the amount is evaluated to be about 6 times lower compared to carbon [7]. In the strike point region, the particle flux is around $2.0x10^{24}$ (D+T) $m^{-2}s^{-1}$ concentrated over a very narrow area representing about $3m^{-2}$ and, according to figure 4, it will saturate in about 2-3s for C or W. For the other parts of the machine, the particle flux are evaluated to $2.0x10^{23}$ (D+T)$m^{-2}s^{-1}$ in the divertor and baffle (150m²) and $5.0x10^{20}$ (D+T) $m^{-2}s^{-1}$ for the 700m² of the first wall. It results that the characteristic time for saturation of the divertor and baffle is around 50 s (C and/or W) and about 100s for the first wall in Be. These durations are rather modest in comparison to the discharge durations of 400s foreseen in ITER. Also, it is worth noting that since this is the recycling flux on the divertor targets, baffles and dome which dominate in the isotopic plasma ratio, a first equilibrium should come rather quickly (~50s). This is consistent with the quick drop of the He/(He+H) ratio observed in the first H discharge as discussed in section II.

Finally, it is worth noting that these values are small compared to the amount of tritium which will be retained by codeposition in the device. Indeed, for the three material configurations discussed above, co-deposition per pulse is evaluated around 15g, 3.0g and 0.5g respectively for the C, C/Be/W and Be/W [7] and reminding that this is a cumulative process, it will naturally dominate the retention after less than 20-30 discharges. In other words, if and He discharges and isotopic exchange can contribute for the control of the plasma isotopic ratio, so far, they cannot be assumed to be a efficient cleaning methods for minimising the T inventory except if the D/T retained by codeposition with beryllium can be exposed to ions and/or neutrals produced by a plasma and then be exhausted by the pumps. This has still to be demonstrated experimentally, and this will be a mafor part of the future experimental programme of JET starting in 2011 with the new Be/W wall under installation at the moment.

4. CONCLUSION

Starting from an "empty/depleted" D_2 and H_2 wall after a two weeks of He campaign cumulating up to 1h36mn of plasma, an hydrogen plasma concentration up to 85-90% has been reached after only 7 pure H pulses for a total of ~245s of plasma operation (98s in limiter and 147s in divertor). The total retention of H results from both codeposition and implantation, the later being the dominant in this transient process of He and isotopic wall exchange. The results obtained for this series of experiments are consistent with previous wall inventory from DT operations at 320°C demonstrating

that in JET (carbon) the accessible wall inventory is in the range of ~2.0×10^{23}D (~1g of T). From these results, extrapolation to ITER show that implanted and accessible T content from the wall in ITER would be 21.3g, 2.63g and 1.63g respectively for the C, C/Be/W and Be/W plasma facing materials foreseen for the different phases of ITER. Finally, it is worth noting that "quantities" implanted are "small" (less than 10% after about 30 discharges in the best case) compared to retention by co-deposition which will dominate the retention with time since this is a cumulative process.

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REFERENCES

[1]. T. Loarer, Journal of Nuclear Materials, **390-391** (2009) 20-28.
[2]. N. Bekris et al., Journal of Nuclear Materials, **337-339** (2005) 659-663.
[3]. V. Rohde et al., Nuclear Fusion, Vol. **49**, Issue: 8 (2009)
[4]. P. Andrew et al., Fusion Engineering and Design, **47** (1999) 233-245.
[5]. T. Loarer et al., Journal of Nuclear Materials, **337-339** (2005) 624-628.
[6]. D. Hillis et al., Journal of Nuclear Materials, **313-316** (2003) 1061-1065.
[7]. J. Roth et al., Plasma Physics Controled Fusion, **50** (2008) 1-20
[8]. B. Pégourié, Fusion Science and Technology, Vol. **56**, Issue: 3 (2009)1334-1352
[9]. R. Pugno et al., Journal of Nuclear Materials, **375** (2008) 168-172

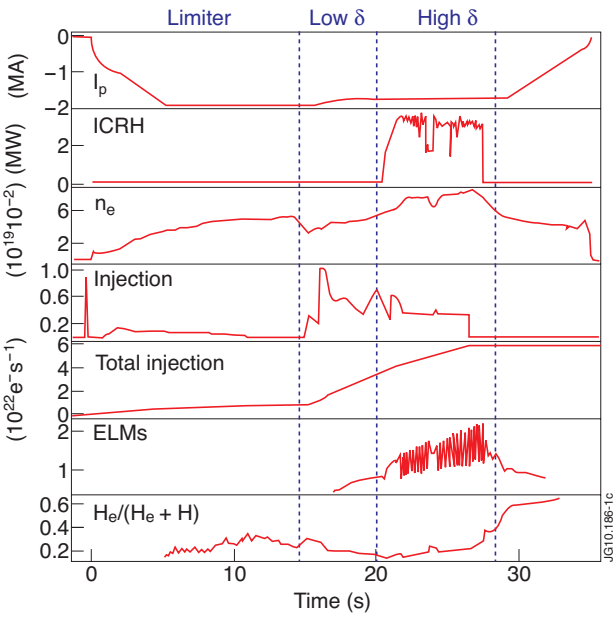

Figure 1: Main plasma parameters of a typical pulse in He with the different plasma phase: limiter, low and then high triangularity. From top to bottom, Plasma current (Ip), auxiliary heating (ICRH), plasma density (ne), gas injection (Γ), total injected, ELM behaviour and He concentration (He/(He+H)).

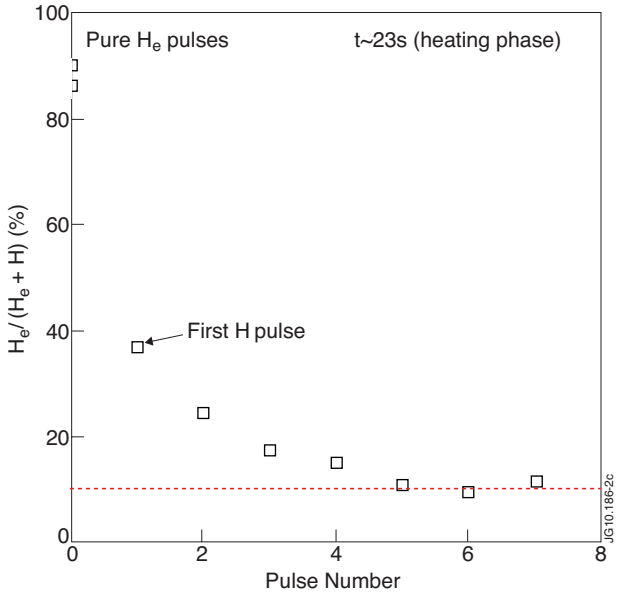

Figure 2: Helium concentration in the main plasma as a function of the pulse number. The values have been taken at 23s during the heating phase.

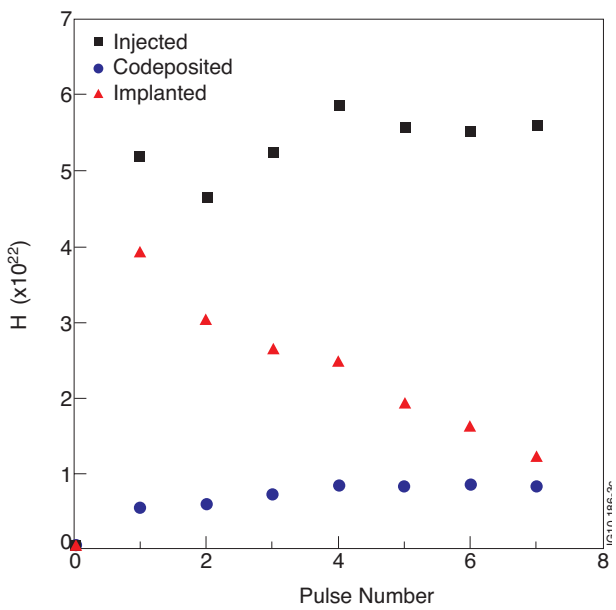

Figure 3: H balance as a function of pulse number. The total H injected is shown by the square, the total retained by codeposition is shown by the circles and the retention by implantation is described by the triangles.

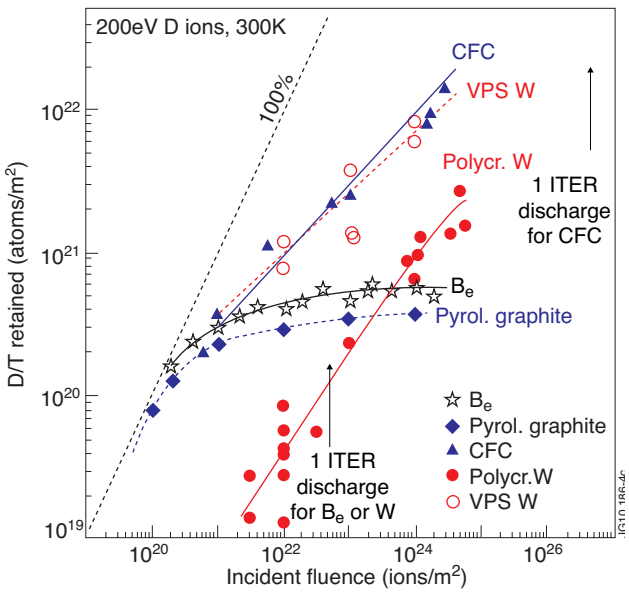

Figure 4: Retention of D or T in beryllium, carbon and tungsten as a function of ion fluence [7].