EUROFUSION CP(15)05/36

T. Wauters et al.

Isotope Exchange Experiments on ITER-Like Wall in JET

(22nd June 2015 – 26th June 2015)
Lisbon, Portugal

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Isotope exchange experiments on ITER-like wall in JET

T. Wauters[1], D. Douai[2], T. Loarer[2], S. Brezinsek[3], JET Contributors*
EUROfusion Consortium, JET, Culham Science Centre, Abingdon, OX14 3DB, UK
[1] *Laboratory for Plasma Physics, ERM/KMS, 1000 Brussels, Belgium, TEC partner*
[2] *CEA, IRFM, F-13108 St-Paul-Lez-Durance, France*
[3] *Forschungszentrum Jülich GmbH, Institut für Energie und Klimaforschung - Plasmaphysik, 52425 Jülich, Germany*
* *See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia*

Introduction

While the permanent retention rate of fuel on the JET-ITER like wall (ILW, beryllium main wall and tungsten divertor) is reduced by a factor of ~ 18 with respect to operation with the carbon wall [1, 2, 3], the rates may still lead to an unacceptable Tritium inventory build-up when extrapolated to ITER. Uncertainties remain on the absolute rates. Post mortem analysis finds permanent retention rates of $5.7 \cdot 10^{18}\,\mathrm{D.s^{-1}}$ [1] while equally well established gas balance analysis techniques result in rates of $0.2-1.5 \cdot 10^{20}\,\mathrm{D.s^{-1}}$ [2, 3]. Long term outgassing may give reason for these difference [2]. For this paper a worst case scenario for the permanent retention rate is considered, and used as a figure of merit in the presented investigation. Extrapolated from JET to ITER, considering a 4 times larger surface area on ITER and a 50% Tritium content in the plasma, one quickly arrives at a permanent retention of 0.5 gT within one 400 s D:T pulse on ITER. The safety limit of 640 gT in ITER [4] may as such be reached within the first few years of ITER D:T operation.

Fuel recovery experiments relying on isotopic exchange by (i) tokamak plasmas, hereafter named I_p-plasma, (ii) ion cyclotron wall conditioning plasmas (ICWC) and (iii) DC glow discharges (GDC), have been performed on JET. The experiments, exchanging the stored fuel content in the plasma facing components (PFC), provide insight on the sizes of the accessible fuel reservoir for each of the techniques. In this contribution the results, based on gas balance analysis consisting of combined volumetric analysis and gas chromatography, are revisited and complemented by in-vessel Be deposition patterns obtained by ex-situ surface analysis of retrieved wall and divertor tiles. The study aims at assessing the performance of the isotopic exchange techniques in mitigating the fuel inventory build-up.

Transient storage to permanent storage during isotopic exchange by I_p-plasma operation

The change-over experiment by I_p-plasma, described in detail in [3], was carried out through 13 consecutive D discharges (with only D_2 gas injection) on a wall pre-conditionned with H. For all these pulses, a total plasma duration of ~ 150 s was performed in limiter configuration and about the same duration ~ 150 s in X point configuration with $I_p = 2.0\,\mathrm{MA}$, $B_T = 2.4\,\mathrm{T}$, $< n_e > \approx 4.5 \cdot 10^{19}\,\mathrm{m^{-3}}$, constant gas injection of $3.0 \cdot 10^{21}\,\mathrm{D.s^{-1}}$ and 0.5 MW of auxiliary heating by ICRH in L-mode. The plasma isotopic ratio was exchanged from 95% in a reference H pulse before the experiment to 5% H in the final D pulse. The gas balance of the experiment shows that throughout $3 \cdot 10^{22}$ D atoms were pumped out of the vessel (see Table 1).

The key processes governing fuel recycling by plasma discharges are identified in [3]. The species in the plasma scrape of layer (SOL) are fuelled both by gas injection and wall release. Transport of ionised particles in the SOL determines the dominant areas for interaction between

Table 1: Gas balance and derived accessible fuel reservoir for different isotopic exchange techniques on JET-ILW. The unit 'atoms' designates the initial dominant isotope specie in the PFC's.

	Removed (gas balance)	Permanent retenion rate	Accessible, maximum
I_p-plasma	$3 \cdot 10^{22}$ atoms	$1.5 \cdot 10^{20}\,\mathrm{atoms.s^{-1}}$	$3 \cdot 10^{22} + 0.5 \cdot 10^{22}$ atoms
ICWC	$6.2 \cdot 10^{22}$ atoms	$1.5 \cdot 10^{20}\,\mathrm{atoms.s^{-1}}$	$6.2 \cdot 10^{22} + 0.4 \cdot 10^{22}$ atoms
GDC	$10 \cdot 10^{22}$ atoms	*none*	$10 \cdot 10^{22}$ atoms

plasma and wall (PWI). As main processes in this interaction one distinguishes (i) implantation of fuel for which the implanted species remain mostly accessible to the recycling process, (iia) codeposition of fuel with eroded wall materials, mostly with Be but also with W, in areas accessible for re-erosion, and (iib) long term retention in form of codeposits located in areas inaccessible to the used plasma type or plasma configuration. The continuous operation of the vacuum pumps removes steadily a fraction of the present neutral gas.

We relate process (iib) to the earlier quoted permanent retention rate. Considering the worst case scenario for the permanent retention rate in I_p-plasma discharges (Table 1) and knowing the averaged isotopic ratio for each pulse (from [3]) we can estimate the amount of D that was removed from the first wall in the isotopic exchange process, but then stored permanently into deposits (by process iib) instead of being pumped out of the vessel by the equation:

$$N_{pr,D} = \sum_{pulse} Q_{pr} \Delta t \rho_D \tag{1}$$

with Q_{pr} the retention rate, Δt the pulse length, ρ_D the isotopic ratio for D and N_{pr} the total stored D in permanent deposits. For the above experiment this amounts to $0.5 \cdot 10^{22}$ D atoms. This number adds to the pumped amount of D atoms to deliver the amount of fuel stored in the vessel that can be accessed in principle by isotopic exchange with I_p-plasma (Table 1).

From this analysis it is concluded that I_p-plasma removes fuel from the transient reservoir and stores a fraction of it (13% estimated) into deposits where it remains inaccessible for further isotopic exchange by I_p-plasma. This unwanted effect may seem less severe than expected, which is due to the fact that for the JET-ILW the plasma isotopic ratio is quickly (less than one minute [3]) dominated by the injected gas. The permanent deposits that are continuously formed contain as such only small fractions of the isotopes of interest (T in the case of ITER).

Transient storage to permanent storage during isotopic exchange by ICWC

Several experiments on isotopic exchange by ion cyclotron wall conditioning (ICWC) have been performed on JET [5]. For this contribution we use as reference the JET-ILW H_2-ICWC experiment on walls "naturally" loaded with D throughout the JET-ILW D campaigns. As described in detail in [6], the JET ICRF antennas were operated at 25 MHz with toroidal field values of 1.65 T simulating on JET the ITER half field case (2.65 T/40 MHz) with on axis location of the fundamental H^+ resonance layer. Powers of $50 - 240$ kW could be coupled to the low density ICRF plasma of $0.2 - 2.4 \cdot 10^{17}\,\mathrm{m}^{-3}$. 20 pulses with variable duration of $2 - 20$ s resulted in ~ 218 s of plasma exposure. The gas balance of the experiment shows that throughout $6.2 \cdot 10^{22}$ D atoms were pumped out of the vessel (see Table 1) while $8.9 \cdot 10^{22}$ H of the injected H remained retained in the vessel.

Similar to the previous section the amount of fuel that can be accessed by ICWC is estimated. Via similar processes as (i), (iia) and (iib) a part of the D will have migrated into permanent storage, remaining herafter inaccessible to ICWC. In a crude approach we take the surplus of retention as permanent retention. This corresponds then to a 1 to 1 exchange in ICWC accessible areas and storage of $N_{pr,H} = 8.9 \cdot 10^{22}$-$6.2 \cdot 10^{22}$ H atoms in permanent deposits. Using again equation (1), relying on averaged isotopic ratios per pulse and the pulse lengths while assuming that the ICWC permanent retention rate is a constant, we arrive equally at a permanent retention rate of $Q_{pr,ICWC} = 1.5 \cdot 10^{20}\,\mathrm{s}^{-1}$. Finally the amount of D that was removed from the first wall in the isotopic exchange process, but stored permanently into deposits (iib) instead of being pumped out of the vessel, is estimated by equation (1) as $0.4 \cdot 10^{22}$ D. As such, the reservoir accessible in principle by ICWC amounts to $6.6 \cdot 10^{22}$ atoms which are $3.1 \cdot 10^{22}$ atoms more than accessible by I_p-plasma (see Table 1).

From this analysis it is concluded that ICWC-plasma removes fuel (supposedly $3.1 \cdot 10^{22}$ atoms) from areas that are not accessible by I_p-plasma, as judged by the larger accessible reservoir. ICWC migrates also less released wall isotopes to permanent storage areas relative to the pumped amount of wall isotopes. From previous studies it is believed that the migration of D from transient to long term retention in ICWC can be even further reduced by optimising the RF duty cycle [7].

Table 2: Summary of the retained D total amounts from JET-ILW 2010-2012 adapted from [1]

JET vessel area	%D stored	Comment
Upper Dump plates	5%	Accessible by ICWC
Outer Poloidal and Inner Wall Guard Limiters	22%	Accessible by ICWC
Divertor Apron (Tile 0 and Tile 1)	33%	Accessible by ICWC
Inner Divertor (without Tile 0)	18%	Possible to rely on divertor baking
Outer Divertor	13%	Possible to rely on divertor baking
Remote areas	11%	/

Fuel removal by I_p-plasma vs. ICWC (in view of T-recovery on ITER)

To assess the effectiveness of ICWC and I_p-plasma for T-recovery, based on the JET results, a 400 s H-plasma is assumed *on JET*. The pulse is expected to retain (worst case) $6.0 \cdot 10^{22}$ H atoms in permanent deposits, and as well $3.5 \cdot 10^{22}$ H atoms that are accessible to I_p-plasma.

Using the above described I_p-plasma procedure to recover the wall isotopes, the $3.5 \cdot 10^{22}$ H atoms will be mobilised from the accessible areas. Of these atoms $3.0 \cdot 10^{22}$ are pumped out of the vessel while $0.5 \cdot 10^{22}$ H atoms will be stored into the permanent deposits, increasing the permanent reservoir to $6.5 \cdot 10^{22}$ H atoms. This results from the fact that from fuel removal point of view, the PWI interaction areas are not sufficiently different from those in interaction during normal operation. For this reason it seems not a good strategy to apply I_p-plasma pulses for the mere purpose of T removal by isotopic exchange between ITER D:T shots (see also [3]).

Using the above described ICWC procedure to recover the wall isotopes may also mobilise the $3.5 \cdot 10^{22}$ H atoms from the I_p-plasma accessible areas. This is supported by the demonstrated complete isotopic change-over by ICWC in a follow up experiment on JET-ILW [8], confirmed by tokamak plasma measurements. The additional removed $3.1 \cdot 10^{22}$ H atoms are stemming then from the I_p-plasma permanent deposit areas. For ITER, this means that 52% of the permanently stored H in one 400 s pulse can be recovered from the vessel, taking that the small fraction of extra retention is mitigated by applying optimal RF duty cycle.

ICWC plasma wetted areas and JET codeposition patterns from surface analysis

ICWC plasmas on JET feature clear recycling radiation areas at the Outer Poloidal and Inner Wall Guard Limiters (low field and high field side resp.) and visually reach the Upper Dump plates and Divertor Apron (meaning tile 0 and 1 following [1]) by applying a small vertical magnetic field ($B_V/B_T \approx 1\%$) [9]. It is therefore preliminary considered that ICWC optimally conditions the low and high field side areas, while removal at the top of the vessel and nearing the divertor can be maximized by applying poloidal magnetic fields. The cleaning efficiency inside the divertor may be modest. Langmuir probes sensitive to ICWC plasma fluxes that could confirm these statements are not available on JET.

Codeposition patterns from post mortem (PM) surface analysis [1] learn that the deposits located on areas accessible by ICWC, supposedly the Outer Poloidal Limiters, Inner Wall Guard Limiters, Upper Dump plates and Divertor Apron comprise 60% of the total retention (Table 2). This means that the present ICWC procedure may prove already capable of removing a significant fraction of the permanent retention in the main vessel that is accumulated in one 400 s pulse. In addition from PM analysis it is found that the contribution of impurities to retention on JET is most important in the divertor. With less impurities on ITER, e.g. less residual C, less deposits will be inaccessible by ICWC. The impurities play a minor role in the overall gas balance [10].

Removal by glow discharge conditioning

The maximum frequency of de-energising and re-energising of the toroidal field coils on ITER is set at once a week. As GDC can't be operated in presence of magnetic fields the maximum GDC frequency on ITER is likewise set at once a week. Isotopic exchange experiments by H_2-GDC on JET have shown that the technique is able to remove $10 \cdot 10^{22}$ D atoms from the first wall [5]. This result is interpreted as GDC having the largest interaction area of the 3 compared isotopic exchange techniques. Moreover the exchange by GDC occurs such that that permanent retention caused is negligible [5].

To appreciate the applicability of GDC for T-recovery on ITER we consider 5 operation days with a modest number of 10 H pulses of 400 s each operated *on JET*. In such an operation week $3 \cdot 10^{24}$ H atoms would be retained in permanent deposits (worst case) while $3.5 \cdot 10^{22}$ H atoms are located in the accessible reservoir for I_p-plasma. Operating D_2-GDC would remove in total $10 \cdot 10^{22}$ H from the vessel, representing supposedly $3.5 \cdot 10^{22}$ H atoms stemming from the transient storage area and $6.5 \cdot 10^{22}$ H from the I_p-plasma permanent deposits. The latter amount is a factor 50 smaller than the worst case accumulated H over the considered operation days. Employing GDC for the purpose of T removal on ITER seems therefore not a secure strategy, which is merely a consequence of the need for $B_T = 0$ T.

Conclusion

The analysis of fuel recovery experiments by isotopic exchange on JET-ILW learns that I_p-*plasmas* do not remove fuel from permanent codeposition areas. This is a mere consequence of the PWI interaction areas being not sufficiently different from those in interaction during normal operation, be-it from fuel removal point of view. Moreover, amounts of the I_p-plasma accessible fuel migrates through PWI and SOL transport to permanent storage areas (13% in worst case). Therefore, for T inventory control on ITER, applying solely I_p-plasmas exchanging the wall isotopic ratio is not a viable scenario (It may still apply for removal from the divertor).

In the case of *ion cyclotron wall conditioning*, it is found that the ICWC PWI area for fuel removal must include a part of the I_p-plasma permanent codeposition areas, judged from $\sim 2\times$ larger interaction reservoir than for I_p-plasma. It is estimated that 52% of the permanent retention by a 400 s plasma can be recovered by ICWC operation. This amount corresponds to almost the complete expected retention in the main vessel including the divertor apron, based on post-mortem surface analysis data. This suggests that operation of ICWC between D:T pulses is a promising strategy for T inventory control on ITER. Although hard to operate, "*divertor baking*" is relied upon to remove the remaining deposits inside the divertor. Divertor baking at 350 °C or 240 °C (slower) features an expected removal efficiency of more than 90% of T from the divertor [5].

Glow discharge conditioning features the largest interaction area of the 3 compared techniques. Nevertheless GDC may prove unable to play an important role in T-recovery on ITER as it can be applied only at maximum frequency of once a week. The latter is far too less in case of worst case retention rates. It is however recommended, adopting a "good house keeping approach" for ITER, to foresee regular GDC operation.

Finally, the presented JET experiment are performed with JET-ILW at 200 °C, whereas ITER will operate ICWC and I_p-plasma at 70 °C. As shown in [11, 12], one key factor determining isotopic exchange is surface temperature. Complementary ICWC experiments are therefore suggested in the JET-ILW with walls at 70 °C.

Acknowledgments

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

References

[1] Heinola et al., J. Nucl. Mater. (2015)
[2] S. Brezinsek et al., Nucl. Fusion **53** (2013) 083023
[3] T. Loarer et al., Nucl. Fusion **55** (2015) 043021 (8pp)
[4] M. Shimada, R.A. Pitts, J. Nucl. Mat. **415** (2011) S1013-S1016
[5] D. Douai et al., J. Nucl. Mat. **463** (2015) 150-156
[6] T. Wauters et al., J. Nucl. Mat. **463** (2015) 1104-1108
[7] T. Wauters et al., J. Nucl. Mat. **415** (2011) S1033-S1036
[8] T. Wauters et al., Proc. of the 25th FEC, Oktober 2014, Saint Petersburg, Russia, EX/P5-21
[9] D. Douai et al., J. Nucl. Mat. **415** (2011) S1021-S1028
[10] S. Brezinsek et al., J. Nucl. Mat. **463** (2015) 11-21
[11] D. Kogut et al., Proc. of the 15th PFMC, May 2015, Aix-En Provence, France, submitted to Phys. Scrypta.
[12] D. Douai et al., 2015, this conf.