Fuel Retention Studies with the ITER-like Wall in JET

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Abstract:

JET underwent a transformation from a full carbon-dominated tokamak to a fully metallic device with Beryllium in the main chamber and a Tungsten divertor. This material combination is foreseen for the activated phase of ITER. The ITER-Like Wall (ILW) experiment at JET shall demonstrate the plasma compatibility with metallic walls and the reduction in fuel retention. We report on a set of experiments ($I_p = 2.0MA$, $B_t = 2.0T - 2.4T$, $\delta = 0.2 - 0.4$) in different confinement and plasma conditions with global gas balance analysis demonstrating a strong reduction of the long term retention rate by more than a factor ten with respect to carbon-wall reference discharges. All experiments have been executed in a series of identical plasma discharges in order to achieve maximum plasma duration until the analysis limit of the active gas handling system has been reached. The composition analysis shows high purity of the recovered gas, typically 99% D. For typical L-mode discharges ($P_{aux} = 0.5MW$), type III ($P_{aux} = 5.0MW$), and type I ELMy H-mode plasmas ($P_{aux} = 12.0MW$) a drop of the deuterium retention rate normalised to the operational time in divertor configuration has been measured from $1.27 \times 10^{21} Ds^{-1}$, $1.37 \times 10^{21} Ds^{-1}$, and $1.97 \times 10^{21} Ds^{-1}$ down to $4.8 \times 10^{19} Ds^{-1}$, $7.2 \times 10^{19} Ds^{-1}$, and $16 \times 10^{19} Ds^{-1}$, respectively. The dynamic retention increases in the limiter phase in comparison with CFC, but also the outgassing after the discharge has risen in the same manner and overcompensates this transient retention. Overall an upper limit of the long-term retention rate of $1.5 \times 10^{20} Ds^{-1}$ has been obtained with the ILW. The observed reduction by one order of magnitude confirms the expected predictions concerning the plasma-facing material change in ITER and is in line with identification of fuel co-deposition with Be as main mechanism for the residual long-term retention. The reduction widens the operational space without active cleaning in the DT phase in comparison to a full carbon device.

1 Introduction

The currently proposed ITER start-up material combination for the non-active phase consists of plasma-facing components (PFCs) made of Be for the main chamber wall, Carbon-Fibre Composites (CFC) for the divertor target plates, and W for the divertor baffle and dome area. An exchange of the divertor to a full W divertor is foreseen for the activated operational phase [1]. The replacement of CFC by W is governed by the need to remain within the safety limit for in-vessel tritium inventory, and thus, to minimise the long term fuel retention. The latter is in carbon-dominated machines determined by co-deposition of tritium with C which is transported stepwise to remote areas [2]. The full metallic material combination, Be main chamber and W divertor, should, according to comprehensive studies described in [3], increase the operational time without active cleaning intervention by more than one order of magnitude with respect to a hypothetical all carbon ITER and up to a factor 3-5 with respect to the initial material mix with CFC target plates.

The world's largest tokamak, JET, underwent a transformation from a fully carbon-clad device with all PFCs made of CFC to a full metallic wall and divertor with the ITER material combination for the activated phase. The replacement of all CFC by bulk and coated Be PFCs in the main chamber, bulk W at the outer horizontal target plate, and W-coated CFC elsewhere in the divertor was done by remote handling in a single shutdown [4]. The ITER-Like Wall (ILW) experiment at JET, which includes dedicated experimental preparatory work with a set of reference discharges with all carbon PFCs [5, 6], provides therefore an ideal test bed for the ITER material choice with the primary goals of demonstrating plasma compatibility with the new metallic wall and the expected reduction in fuel retention [7]. Initial experiments have shown successful operation with metallic walls but with a narrower operational window at the best confinement in the ITER-base line scenario caused by the required operation at higher gas fuelling rates to minimise the W influx [8]. The absence of CFC as PFC material has led even without active cleaning of the Inconel vessel to a dramatic reduction of the carbon content in the plasma by more than one order of magnitude [9]. The temporal evolution of the C content in the plasma was documented by optical spectroscopy in monitoring discharges which revealed three phases: an initial clean-up phase, a constant phase, and at the end of campaign a phase with slight increase of C with auxiliary power. However, the carbon flux in the plasma edge - a key parameter for the long term retention in JET with carbon walls (JET-C) - was reduced by at least a factor 10 throughout the first JET-ILW year.

Initial gas balance studies performed in the first and low power phase of the campaign indicate a low fuel retention with the JET-ILW [10]. Here, we report on a set of deuterium experiments with gas balance analysis demonstrating a strong reduction of the long-term fuel retention with the ILW with respect to previously performed JET-C reference discharges under various plasma conditions. These global gas balance experiments have been performed throughout the campaign; starting with ohmic plasmas in the initial phase of divertor operation and finishing in high confinement mode (H-mode) plasmas at the last day of operation. The short-term retention increases in the limiter plasma phase with the Be first wall, but the outgassing after the discharge has risen in the same manner and compensates this transient retention. The residual long-term retention in the JET-ILW can be attributed to implantation and co-deposition of fuel by Be, whereas the

latter process determines the fuel content with operational time as implantation saturates. Overall, the reduction of the long-term retention from gas balance studies by a factor 10-20 is fully supportive of the ITER material choice and widens the operation in DT without active cleaning with respect to an all-carbon device.
Details about the experimental method are described in sec. **2**, experimental results are presented in sec. **3**, followed by the discussion about the physics mechanisms responsible for the reduction in retention in comparison to JET-C in sec. **4**. Conclusions for ITER are drawn in section **5** providing also predictions concerning the operational without actively cleaning. A summary in sec. **6** completes this contribution.

2 Experimental Method

The quantification of the retention rate, i.e. the number of retained deuterium ions in the first wall components per second, has been carried out by global gas balance in the JET vessel as indicated schematically in fig. 1a. The gas balance takes into account the amount of injected D through the gas injection systems and, if applicable, the neutral beam injection during plasma operation, and the number of actively pumped neutrals by the applied pumping systems. The difference between injected and pumped neutrals can be attributed to the number of retained fuel atoms in the PFCs of the tokamak during plasma operation. A general description of global gas balances and the role of short and long-term retention in different tokamaks can be found in [11].

Global gas balance experiments in JET, focus on the long-term retention with respect

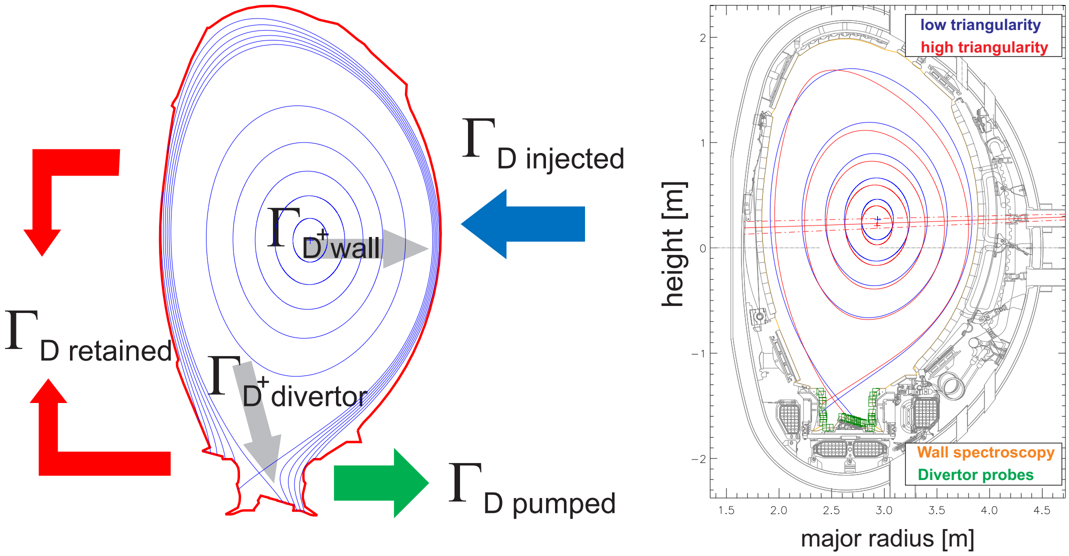

FIG. 1: a) Principle of the gas balance measurements in JET and the involved deuterium neutral and ion fluxes. b) High and low triangularity plasma shapes executed in the JET-ILW experiments. The diagnostic coverage for in-situ information of particle fluxes to the wall (Balmer spectroscopy) and divertor (Langmuir probes) is indicated.

to one day of operation and in-situ collection, thus, the method is positioned between an intra-shot gas balance [11] and an ex-situ post-mortem analysis of components which takes place months after end of operation, including thus a much longer outgassing period [12, 13, 14]. The pumped gas is collected by the active gas handling system (AGHS), quantified by calibrated pressure measurements, and analysed for its components by gas chromatography. The quantification is currently limited to $2600 Pam^3$ which corresponds to the typical gas load of one day of plasma operation. The internal precision of AGHS concerning the quantification and composition is 1.0% [15]. All data presented here are normalised to $293.15K$.

From the technical point of view three types of gas balance experiments were performed in high or low triangularity plasma shape (fig. 1b) whereas the letters **(a-f)** correspond to the specific experiment in either ohmic, L or H-mode confinement regime described below in section (3):

(i) Active use of the cryogenic divertor pump at LHe (liquid Helium) temperature and no active pumping by turbo-molecular pumps **(a-b)**. The cryogenic pump is regenerated both before and after the experiment which consists of a set of repetitive, comparable, discharges. The gas collected by AGHS after the second regeneration, which takes place typically after $90min$ of the last discharge, represents the actively pumped D during plasma operation and in-between discharges by outgassing. The vacuum valve to the neutral beam cryogenic pump is closed in this standard method. Comparison of injected deuterium particles by the gas injection modules and the collected neutrals by AGHS reveals the amount of retained deuterium in the vessel. This method had been used regularly before in JET-C [6].

(ii) Active pumping of the neutrals solely by the turbo-molecular pumps whereas all cryogenic pumps are inactive and stay at LN temperature **(c)**. The exhaust from the turbo-molecular pumps is collected and analysed by AGHS as described before. This decreases the gas throughput per plasma discharge for comparable plasma conditions like pumped divertor conditions, which increases the statistics and the precision of the gas balance by about a factor 4, but reduces the pumping capability in the ramp-down phase and making the discharges somewhat more prone to minor density limit disruptions. Here, only experiments in L-mode were carried out which show a slightly higher retention rate with respect to (i). The slight increase is likely caused by the collection time after end of the last pulse, which was comparable to method (i), but executed with the reduced pumping capabilities of the turbo-molecular pumps. Comparable studies with this method have been carried out in TORE Supra demonstrating identical retention rates in experiments with $1min$ plasma duration and use of turbo-molecular pumps to experiments with use of the cryogenic pump regeneration [16].

(iii) Active use of the cryogenic divertor pump at LHe temperature as described in (i) but with additional pumping of the vessel by the neutral beam cryogenic system **(d-f)**. The plasma operation with auxiliary heating by the neutral-beam system, i.e. H-mode plasmas, requires use of the NBI cryogenic pump system. In this operation mode the fraction of gas actively pumped by the NBI system was deduced from the ratio of pumping speeds of the NBI versus the divertor cryogenic pump system. The pumping speeds have been measured in-situ in dedicated gas test pulses at the start of each gas balance experiment resulting in $76m^3s^{-1}$ for the NBI system in comparison with $118m^3s^{-1}$ for the divertor cryogenic pump system from the neutral pressure measured in the JET vessel. The NBI cryogenic system pumps about 1/3 of the post discharge outgassed neutrals. Please note, that the operation with the inertially cooled bulk W divertor in the JET-ILW case requires additional cooling time before plasma operation can resume in the next discharge

[17], due to the reduced heat transfer of the bulk W modules to the underlying divertor structure. In particular in H-mode plasmas, the time between two discharges has more than doubled from typically $\simeq$ $1200s$ with JET-C to $\simeq$ $2700s$ with JET-ILW. Thus, the number of neutrals pumped by the NBI cryogenic system in this outgassing phase after the plasma discharge - not collected by the AGHS system but calculated from the pumping speed ratio - is significant larger with respect to the JET-C reference.
The overall precision of the gas collection, including gas injection, transfer lines, and quantification in AGHS has been determined in calibration experiments without plasma which mimic the injection conditions with plasma. The first experiments [**(a)**,**(b)**] revealed a reproducible systematic uncertainty in the temperature normalisation in AGHS which lead to an overestimation of the gas loss, therefore, the calibration value has been subtracted from the actual measurement with plasma to provide the retention rate. The uncertainty is given by the statistical error of the reproducibility of the calibration. An upgrade of the AGHS has been used in experiments [**(c)**-**(f)**] providing an uncertainty of the overall method of $\pm1.2\%$ which has been verified throughout the campaign in calibration experiments performed temporarily close to the actual plasma experiment. The temperature of the injected gas through the gas injection modules has been monitored.
Additionally to the global gas balance, in the case of the JET-ILW discharges, the ion fluxes to the main PFCs have been studied in-situ during plasma operation as indicated schematically in fig. 1a as well as the surface temperature of the PFCs monitored. Balmer spectroscopy of D_β observing one poloidal beryllium limiter at the vessel midplane (fig. 1b) had been used to determine the limiter recycling flux which is under the assumption of full recycling during the plasma flattop phase equivalent to the impinging ion flux. CXRS neutrals are covered in the measurement. Langmuir probes embedded between divertor PFCs had been applied to determine the integrated ion flux to the tungsten target plates (fig. 1b). In the present studies, the fluxes were averaged over one second in the flat-top phase of the plasma discharge, neglecting in H-mode plasmas the impact of ELMs. Surface temperatures were measured by IR thermography observing a full poloidal section of the vessel and thus covering the main inner and outer poloidal Be limiter as well as the W divertor. Toroidal symmetry is assumed in order to combine the information from these three in-situ diagnostics observing different toroidal sectors of the vessel.

3 Experimental Results

3.1 Single global gas balance experiments

From a set of JET-ILW global gas balances which are described in fig. 2 in the order of temporal appearance during the campaign, we present here in detail a representative set of six dedicated experiments **(a-f)** executed at a plasma current of $I_p = 2.0MA$, a toroidal magnetic field between $B_t = 2.0T$ and $2.4T$ with a triangularity of $\delta = 0.2$ and $\delta = 0.4$, respectively, covering different plasma conditions, confinement regimes, and injection rates. The corresponding plasma shapes in high and low triangularity are shown in fig. 1b. All global gas balance experiments were carried out in series of comparable repetitive discharges, minimum 8 and maximum 34 consecutive plasmas, until the number of injected particles reaches approximately the analysis limit of the AGHS which minimises the impact of history effects (i.e. legacy related to details of specific pulses run prior to gas balance experiment) and maximise the plasma exposure. An overview of all measured deuterium retention rates in JET-ILW experiments determined by methods (i, iii), marked by (■), and (ii), marked by (■), as well as in corresponding JET-C references by methods (i, iii), indicated by (■), is given in fig. 2. The retention rates are normalised to the integrated plasma time in divertor configuration reflecting the plasma phase with main ion flux interaction with the divertor PFCs. The selected subset **(a-f)** is representative for the whole number of experiments executed and shall underline the reproducibility for comparable plasma regimes as well as the fact that no variation between H-mode gas balances in the middle and at the end of the JET-ILW campaign exists. The fuel retention in the all-carbon JET device was governed by co-deposition of fuel with carbon in the divertor [14] which increases with operational time in divertor configuration. For comparison we apply the same normalisation to the operational time in divertor configuration with JET-ILW. Full consideration of the longer outgassing phase between neutral-beam heated type I ELMy H-mode discharges in the JET-ILW with respect to JET-C leads to a correction of the retention rate as depicted by yellow bars (■) in fig. 2 on top of the corresponding uncorrected values [**cf** method (iii) in section **2**]. There are no JET-C

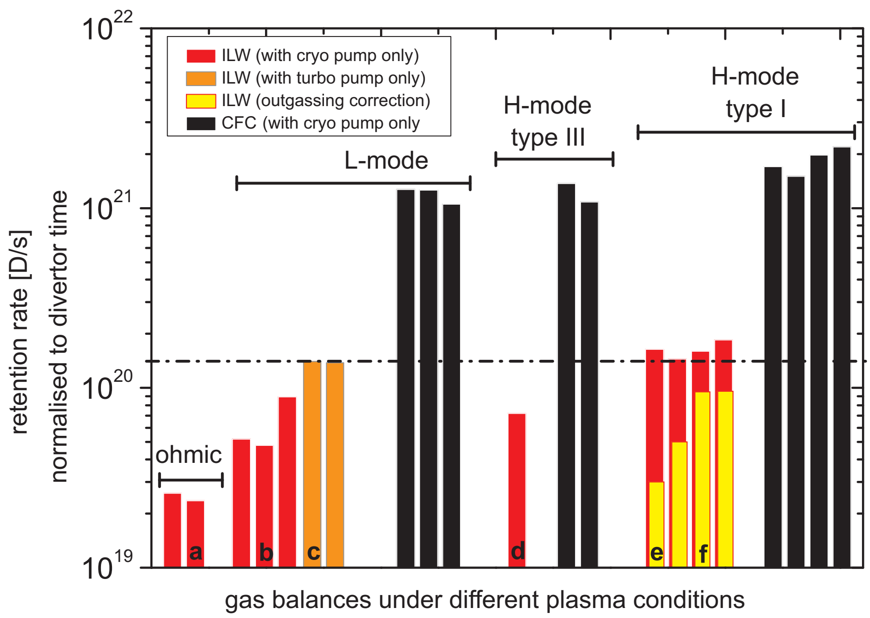

FIG. 2: Measured D retention rates (logarithmic scale) for different plasma and confinement conditions in JET with the ILW and related to JET-C references. Global gas balances are performed in the JET-C case solely with cryogenic pumping (■), and in the JET-ILW case with either the use of turbo-molecular (■) or cryogenic pumping (■). The longer inter-shot outgassing period after H-mode discharges with the JET-ILW in comparison with JET-C leads to a reduction of the retention rate (■) owing to pumping of neutrals by the NBI cryogenic system.

reference discharges for ohmic plasma conditions.
The ILW discharges have been adapted in the fuelling rate to match edge plasma conditions, i.e. the edge density and deuterium ion flux, of CFC reference discharges as closely as possible. The magnetic configuration remained identical with respect to the divertor,

but minor adaption in the main chamber was required to keep the wall clearance to the redesigned limiter PFCs constant. In the case of L-mode plasmas, the auxiliary power by RF power had been reduced from $2.0MW$ to $0.8MW$ and $0.5MW$, respectively, to stay below the lower L-H threshold with the JET-ILW [8]. The discharge ramp-up phase in limiter configuration is reduced by $\approx 5s$ in H-mode discharges by scenario optimisation in comparison with JET-C references providing a ratio of $t_{divertor} \simeq 3 \cdot t_{limiter}$ with the JET-ILW which is closer to the ITER operation with $t_{divertor} > 10 \cdot t_{limiter}$.
The JET vessel is actively heated to $200°C$ which is in thermal equilibrium equivalent to the base temperature of the main chamber PCFs in both JET-C and JET-ILW configuration. But as the installed components are not actively cooled, they undergo a multiple heating cycle in one gas balance experiment with temperature increase during plasma impact, depending on residence time and input power, and temperature decrease in the subsequent cool down or waiting phase. The main surface temperature increase in the divertor phase takes place at the outer divertor W-target plates with a maximum excursion in H-mode plasma conditions from $150°C$ to $1200°C$. The outer divertor poloidal limiters stay in all discharges close to the base temperature with a maximum surface temperature of $270°C$; the inner poloidal limiters show larger variation of surface temperatures up to $330°C$ in dependence on the initial limiter duration and input power. Overall, the PFC surface temperature excursions during a discharge in JET-C and JET-ILW are comparable in the discussed comparative gas balance experiments, but restrictive operational limitations in the case of the JET-ILW minimise e.g. the duration of the divertor flat-top phase.
Apart from the direct material comparison aspect of fuel retention in the JET-C and JET-ILW configuration, the JET-ILW experiments themselves provide a data base of gas balances with variation of plasma scenarios (ohmic, L and H-mode), the fuelling rate ($1 \times 10^{21} - 1 \times 10^{23} Ds^{-1}$), the auxiliary heating power ($P_{aux} = 0 - 12.0MW$), as well as the perpendicular ion flux to the divertor ($\Gamma_{ion} = 2 \times 10^{22} - 2 \times 10^{23} m^2 s^{-1}$) and the ion flux density to the wall ($5 \times 10^{19} - 7 \times 10^{20} m^2 s^{-1}$). All JET-ILW experiments show low retention rates covering a range of $2 - 16 \times 10^{19} Ds^{-1}$ applying the same analysis as in JET-C and show high purity of the recovered gas of typically 99.0% D mainly in form of D_2.

- **(a)** JPN #80279 − 80314: Ohmic plasmas ($P_\Omega = 1.3MW$) in low triangularity shape with averaged low fuelling rate of $0.8 \times 10^{21} Ds^{-1}$ in density feed-back control. In 34 discharges $1764Pam^3$ D had been injected with a fuel retention of $38Pam^3$ or equivalent 1.9×10^{22} D atoms. The retention rate is as low as $2.3 \times 10^{19} Ds^{-1}$ when normalised to $807s$ plasma operation in x-point configuration. The repetition rate between discharges is approximately $20min$ with an overall of $14h$ operation time for which the inter-shot outgassing is collected solely by the divertor cryogenic pump system. The integrated divertor ion flux measured by Langmuir probes amounts to $4.0 \times 10^{22} Ds^{-1}$ and the recycling flux density to the inner poloidal limiter measured by Balmer spectroscopy is about $5.4 \times 10^{19} Dm^{-2} s^{-1}$ in the flat-top phase of the discharge.

- **(b)** JPN #80779 − 80803: L-mode plasmas with $P_{aux} = 0.8MW$ RF heating in high triangularity shape with a deuterium fuelling rate of about $2.4 \times 10^{21} Ds^{-1}$. In 22 discharges $2266Pam^3$ D had been injected with an integral fuel retention of $43Pam^3$

or equivalent 2.1×10^{22} D atoms. With the cumulated divertor duration of $440s$, the retention rate amounts to $4.8 \times 10^{19} Ds^{-1}$ compared with $1.27 \times 10^{21} Ds^{-1}$ in the JET-C reference discharges under comparable conditions. In line with the reduction of the retention rate is also the reduction of the C impurity flux in the divertor by an order of magnitude as measured by optical spectroscopy in main chamber and divertor [9]. The integrated ion flux to the divertor during the flat-top phase of these plasma discharges amounts $7.0 \times 10^{22} Ds^{-1}$ and the corresponding flux density to the poloidal limiter is $1.50 \times 10^{20} Dm^{-2}s^{-1}$. Low triangular plasmas in L-mode with comparable fuelling rate and divertor plasma conditions show comparable retention rates in the JET-ILW.

- **(c)** JPN #81937 − 81973: In L-mode plasmas with $P_{aux} = 0.5MW$ RF heating and with solely turbo molecular pumping 35 discharges have been carried out accumulating $731s$ in divertor configuration. In total $1710Pam^3$ D had been injected with a fuel retention of $222Pam^3$ or equivalent 11.1×10^{22} D atoms. The corresponding retention rate amounts to $1.51 \times 10^{20} Ds^{-1}$ with an uncertainty of $\pm 1.8 \times 10^{19} Ds^{-1}$. The gas fuelling rate in these discharges is in average $4 \times 10^{20} Ds^{-1}$, the ion flux to the divertor reaches $3.0 \times 10^{22} Ds^{-1}$, and the flux density to the main poloidal limiter is $2.0 \times 10^{20} Dm^{-2}s^{-1}$.

- **(d)** JPN #81604 − 81624: H-mode plasmas in type III ELMy H-mode with $P_{aux} = 5.0MW$ auxiliary heating by NBI and high total fuelling rate of $6 \times 10^{21} Ds^{-1}$ in

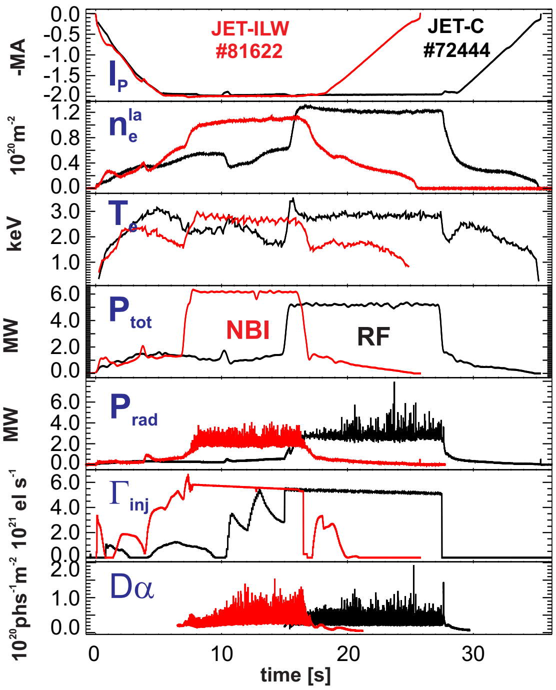

*FIG. 3: Comparison of JET-C and JET-ILW discharges in type III ELMy H-mode (**d**).*

feed forward operation. In 18 discharges $2495Pam^3$ gas has been introduced by injection and beam fuelling, and $47Pam^3$ or 2.3×10^{22} D atoms were retained. The retention rate with the ILW amounts to $7.2 \times 10^{19} Ds^{-1}$, when normalised to

317s in x-point configuration, and can be compared with $1.37 \times 10^{21} Ds^{-1}$ obtained in similar plasmas in JET-C conditions but with RF heating instead of NBI at the same level. Global plasma temperature, density and radiation as well as edge density and ion flux are in good agreement between the JET-ILW discharge and JET-C reference discharge as shown in fig. 3. The recycling flux density at limiter is $2.0 \times 10^{20} Dm^{-2}s^{-1}$ and the integrated ion divertor flux is $1.3 \times 10^{22} Ds^{-1}$ omitting the contribution of ELMs. The surface temperature of the outer divertor W target plate peaks at about $1000°C$ at the end of the flat-top phase. The inner wall limiter temperature reaches a peak value at the end of the limiter phase and stay between $270 - 290°C$ during the flat-top phase, correspondingly, the low field side or outer wall limiter temperature stays between $220 - 250°C$.

- **(e)** JPN #82663 − 82673: Type I ELMy H-mode plasmas in low triangularity shape with $P_{aux} = 11.0MW$ auxiliary heating by NBI and a gas injection of $1.0 \times 10^{22} Ds^{-1}$ in feed forward fuelling providing a total deuterium injection rate of $1.1 \times 10^{22} Ds^{-1}$ including the beam fuelling. In total 164s in x-point configuration distributed over 8 discharges have been obtained with an integrated divertor ion flux of $2.6 \times 10^{23} Ds^{-1}$ in the flat-top phase. Though, the outer strike point was moved for power load control over two segments of the bulk tungsten tile to extend the flattop phase, the surface temperature of the bulk W divertor segments still exceeded $1145°C$ starting from a base level of about $160°C$. The surface temperature of the poloidal limiter on the low field side increases slightly from $230°C$ to $270°C$ during the flat-top phase whereas the corresponding temperature on the high field side stays constant at about $325°C$. The ion flux to the limiter in this period is $5.3 \times 10^{20} Dm^{-2}s^{-1}$. The retention rate is low as $3.5 \times 10^{19} Ds^{-1}$ under consideration of the 45min waiting time between plasma discharges and pumping by the NBI cryogenic pump as described before. The apparent retention rate, suggesting comparable waiting time between discharges in this experiment to a JET-C references, would be $1.45 \times 10^{20} Ds^{-1}$ which needs to be compared with $1.97 \times 10^{21} Ds^{-1}$ [6].

All gas balance experiments performed with the JET-ILW show a strong reduction of the long-term retention rate between a factor 10-20, depending on the confinement regime, in comparison with JET-C references. This robust result was confirmed by different types of experimental gas balance methods (e.g. with or without divertor cryogenic pumping), different types of plasma conditions (e.g. low and high gas fuelling), and different types of magnetic configurations (low and high triangularity). The integrated plasma time in a single gas balance experiment with multiple discharges exceeded 800s, thus, an equivalent in the duration of two ITER discharges which indicate the statistical significance of the experiments. Repetition of L-mode experiments provided confidence in the reproducibility of results. An increase of the absolute retention rate with increase of the plasma confinement starting from ohmic and ending in type I ELMy H-mode plasmas has been found and is likely related to the ion flux to the main PFCs (cf section 4). In JET-C no such clear trend has been observed or was hidden in the overall high long-term retention rate. In contrast to JET-C references, the repetition rate of discharges in H-mode was reduced to half the frequency due to limitations of the power handling. The fuel recovery in the longer waiting time revealed the importance of the short-term retention visible in the inter-shot outgassing in the JET-ILW experiments (cf section 4.2).

3.2 Multiple gas balance measurements in the operational period before tile removal

As a further example of a number of gas balance experiments in baseline H-mode plasmas with higher input power which have been carried out throughout the campaign and showing all a strong reduction retention rates with respect to comparable JET-C plasmas (fig. 2), we present here in more detail one experiment from the last two weeks of operation before the first shutdown for tile intervention with the JET-ILW. This specific campaign (JPN #83621 − 83791) which covers 5% of the total plasma time ($\simeq 20h$) of the first year of JET-ILW operation aimed in the provision of a material migration footprint in ONE characteristic ILW-typical H-mode plasma for the post mortem analysis of toroidal and poloidal sections of the first wall armour which will be reported elsewhere. In order to accumulate a significant fluence, 151 identical discharges ($B_t = 2.0T, I_p = 2.0MA, Z_{eff} = 1.2, \delta = 0.2$, $P_{aux} = 12.0MW$ NBI heating) in deuterium with in total 2500 plasma seconds in divertor configuration resulting in an integrated divertor fluence of $5.25 \times 10^{26}Dm^{-2}$, comparable to one quarter of an ITER pulse ($2.5 \times 10^{27}Dm^{-2}$) at full performance [18], have been performed. To minimise the smearing of the erosion/deposition footprint caused by the impact of impinging deuterium ion and impurity flux onto the target plates, the limiter time was minimised to 1/4 of the total pulse duration, the strike-point kept constant over the whole discharge, and the gas fuelling rate fixed in feed forward to $1.2 \times 10^{22}Ds^{-1}$ with an add on of $1.2 \times 10^{21}Ds^{-1}$ beam fuelling. These parameters were chosen to ensure both strike points attached and to avoid at the same time potential W accumulation. The local plasma conditions during the plasma flat-top phase at the strike-point of the inner target are $T_e = 7eV$ and $n_e = 2.5 \times 10^{20}m^{-3}$ and $T_e = 35eV$ and $n_e = 6 \times 10^{19}m^{-3}$ at the strike-point of the outer target plate. The integrated ion flux to the target plates is $2.5 \times 10^{23}Ds^{-1}$ and the flux density at the poloidal limiter amounts $4.6 \times 10^{20}Dm^{-2}s^{-1}$. The operation with static magnetic shape requires a cool-down phase of about $45 - 50min$ between discharges to allow inertial cooling of the W divertor from max. $1050°C$ back to the base level of $160°C$. The surface temperature of the massive Be poloidal limiters on the high field side vary from $275°C$ to a maximum of $285°C$ in the flat-top of a single discharge. The surface temperature of the corresponding poloidal limiters on the low field side remains almost constant between $210°C$ and $260°C$. Both, the waiting time between plasma discharges and the increased temperatures of PFCs lead to significant outgassing over a period at least twice as long as in JET-C H-modes.

Three dedicated gas balances experiments have been executed in this period to demonstrate the reproducibility of gas balance measurements under comparable conditions and to provide a reference for the fuel content deduced by post mortem analysis. Fig.4 shows the high reproducibility of plasma conditions within the series of 9 consecutive discharges of one of the three gas balance experiments marked as **(f)**. The global parameters core electron density n_e and temperature T_e, the input P_{aux} and radiated power P_{rad}, as well as the recycling flux at the target, characterised by the D_α photon flux are similar throughout the whole series. These unseeded and purely NBI-heated discharges accumulated $151s$ of plasma time in x-point configuration with a total divertor fluence of $3.2 \times 10^{25}Dm^2$ and $2113Pam^3$ of injected D. A retention of $29.9Pam^3$ D or equivalent 1.49×10^{22} D atoms was obtained under consideration of the pumping of outgassed D by the NBI cryo-

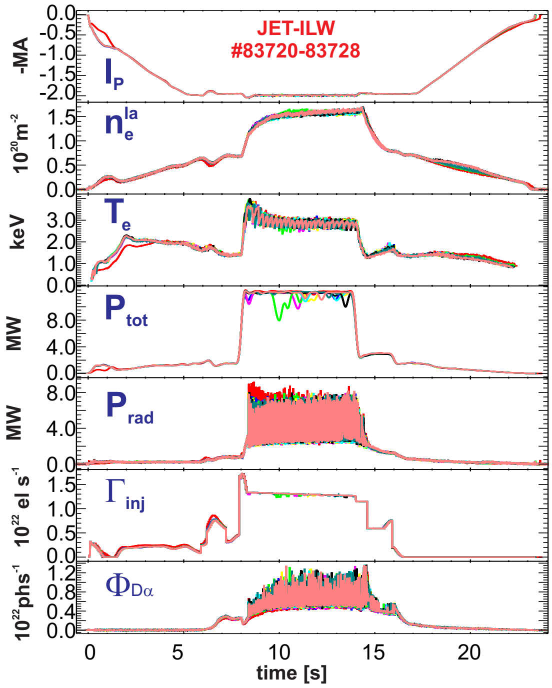

FIG. 4: Global plasma parameter of the standard type I ELMy H-mode plasma utilised in the JET-ILW operational period before tile removal with 151 comparable discharges. Nine consecutive H-mode discharges had been executed in a row demonstrating the high reproducibility of plasmas.

genic pumps during the time between discharges. This corresponds to a retention rate of $0.98 \times 10^{20} Ds^{-1}$ as shown in fig. 2 in the yellow bar labelled as **(f)**. The red column represents the apparent rate assuming comparable outgassing time between discharges as performed in JET-C ($850s$ outgassing have been included in the analysis in JET-C); this overestimated rate of $1.6 \times 10^{20} Ds^{-1}$ can be compared with $1.97 \times 10^{21} Ds^{-1}$ [6] measured for similar H-mode plasmas in JET conditions with carbon walls.

Overall this gas balance experiment is in good agreement with the second gas balance of the same kind in the two week period with an outgassing corrected retention rate of $0.96 \times 10^{20} Ds^{-1}$ which underlines a high reproducibility of discharges and the low retention rate with the JET-ILW. The retention rate determined in this gas balance experiment is depicted to the right of the column **(e)** in fig. 2. Both results are also in agreement with the first H-mode gas balance, discussed in the previous section as experiment **(e)**, performed under comparable plasma conditions in the middle of the campaign. The variation in the fuel retention rate with the JET-ILW is small for similar H-mode plasmas conditions and configurations, indicating comparable wall conditions and impurity content over the experimental period of the last ~ 20000 plasma seconds (JPN #82663 and #83794) as also indicated by regularly performed monitoring pulses. We can conclude, that steady state conditions in the wall are reached and that the mechanism for the residual long-term retention is unchanged in this operational period.

3.3 Retention in the limiter phase

In order to study the impact of the transient limiter phase on the overall long-term retention in diverted discharges, a dedicated gas balance in bare limiter configuration has been executed and initial results were reported in [10]. The gas balance was carried out with solely use of the turbo-molecular pumps accordingly to method (ii). 34 limiter discharges (JPN #82591 − 82626) at a plasma current of $I_p = 2.0MA$, a magnetic field of $B_t = 2.4T$, and with $P_{aux} = 0.5MW$ auxiliary heating by RF power were carried out, providing $510s$ of plasma exposure on the main chamber PFCs. The retention rate, normalised over the whole plasma duration, amounts to $8.94 \times 10^{19} Ds^{-1}$ which is about 60% of the retention of gas balance in diverted conditions **(c)** performed at similar plasma current, auxiliary power and use of the turbo-molecular pumps. The deuterium retention during discharges is not compensated by post-pulse outgassing, thus implying net retention due to co-deposition during the flat-top phase of the limiter discharge takes place.
However, the detailed interpretation is challenging due to variation of different conditions in the discharge, in particular the central density and the surface temperature of limiter PFCs. The surface temperature varies strongly in the whole experiment and in each discharge. In the first 5 discharges out of the 34, the base temperature rises from $200°C$ to $500°C$ with an additional increase of $400°C$ in each discharge. In fact, after the fifth discharge an equilibrium between cool-down time of the Be limiters and plasma impact in the discharge takes place and the base temperature is almost constant at $500°C$. Intra-shot gas balance of one pulse of the series revealed that the initial high short -term retention drops continuously by a factor 2 during the discharge flat-top phase indicating that the averaged discharge time of $15s$ is not sufficient to reach steady-state conditions [19]. The base level of retention by co-deposition in the limiter flat-top phase can't be accessed exactly but is overlayed by the short term retention. The latter is significant due to the strong outgassing in the previous discharge leading to active pumping of the Be wall at the beginning of the next discharge.
In contrast to JET-C, the transient limiter phase has an impact on the global gas balance analysis and the retention rate related to the divertor or steady-state phase represents an upper limit. However, the ion fluxes to the main PFCs during the ohmic current ramp-up and ramp-down in limiter configuration are small and the overall operational time in limiter configuration is less than 1/4 of the total plasma duration. The overestimation of the retention rate presented in **(a-f)** is therefore in the order of $10 - 15\%$. To minimise the impact further, gas balances with longer divertor phases are required. This will be addressed in future JET-ILW experiments.

4 Discussion

4.1 Implantation and fuel co-deposition with Be and W

Laboratory experiments [20] identified two mechanisms responsible for long-term fuel retention in Beryllium - the predominant plasma-facing material in the JET-ILW: implantation and co-deposition. Implantation by ion bombardment is limited to an interaction zone of the wall material and therefore saturation of this contribution shall occur in time under steady-state load conditions as post-mortem analysis of exposed Be tiles revealed

[21]. It should be noted that this saturation of retention in Be does not exclude that transiently much more retention during plasma loading occurred. The fact that it is transient can be inferred by the strong outgassing shortly after end of plasma loading [22]. Co-deposition of deuterium with impinging Be ions is in principle not limited, but can continue linearly in time with the impinging deuterium ion and Be impurity flux until thermal or mechanical instability of the co-deposit occurs. The temporal behaviour of each fraction contributing to the long-term retention as well as the location where the retention process takes place, main chamber vs. divertor, is of importance for the physics understanding of migration, associated modelling and prediction of long-term retention in tokamaks with Be/W environment. It should be noted that the first ohmic divertor plasma operation described in this contribution did not start from a virgin Be first wall, but that about $100h$ of glow discharge in deuterium for conditioning [23] as well as about $300s$ of limiter plasma operation before divertor operation started [24] has been applied. Therefore some deuterium pre-loading and implantation of the first wall as well as material migration occurred prior to the first gas balance experiments discussed here.
Important for both processes is the impinging ion flux to the main plasma-facing components in the tokamak which has been used in previous studies to scale the fuel retention from carbon-dominated devices including JET-C to ITER [3, 25]. The dominant physics process in these scaling studies, and therefore in the carbon-dominated devices, has been identified to be co-deposition of fuel with C which takes mainly place in the divertor whereas the primary C erosion source is the main chamber (limiter PFCs) as post-mortem analysis of components confirmed. The whole transport of wall material is described as material migration and driven by scrape-off layer flows in the main chamber and multiple step C transport in the divertor caused by sputtering of C co-deposits [26]. The low-Z plasma-facing materials Be and C have, according laboratory studies [27], comparable erosion yields providing a comparable impurity source and one could expect a similar migration behaviour. However, C chemical sputtering, which is to a large extent independent of impact energy and is caused by impinging deuterons and low energetic atoms, will dominate the erosion and transport in regions of low electron temperatures such as the first wall cladding and the divertor. Be erosion is determined by physical sputtering, and thus has a threshold energy for sputtering of about $10eV$ according to MD calaculations [28]. This difference in the low energy branch of the sputtering process impacts the migration process in particular in the divertor and in the far scrape-off layer. In fig. 5 is the retention rate obtained in all JET-ILW experiments depicted as parameter of the ion flux density to the poloidal limiter (fig. 5a), representative for the main chamber, and the integrated ion flux to the divertor (fig. 5b). The letters **(a)** till **(f)** refer to the experiments described in section **3**. The integrated ion flux to the wall is in diverted plasmas proportional to the measured flux density to the limiter. The active area in JET is challenging to identify due to the complex structure. Assuming an active surface area of about $144m^2$, the integrated ion flux to the wall in all experiments is in the range between $4.0 \times 10^{21} Ds^{-1}$ to $4.8 \times 10^{22} Ds^{-1}$ which is comparable to previous values obtained in JET-C [26]. Both graphs show qualitative the same behaviour indicating a coupling between the two deuteron fluxes to the main PFCs with the wall flux representing about 15% of the divertor flux. Deviations from the main trend can be identified in conditions with the divertor in high recycling operation such as in the case **(f)** without divertor

cryogenic pump or in L-mode cases with high fuelling. In the latter case the ion flux to the divertor is about twice the flux in L-mode gas balances with the divertor operating in low recycling.

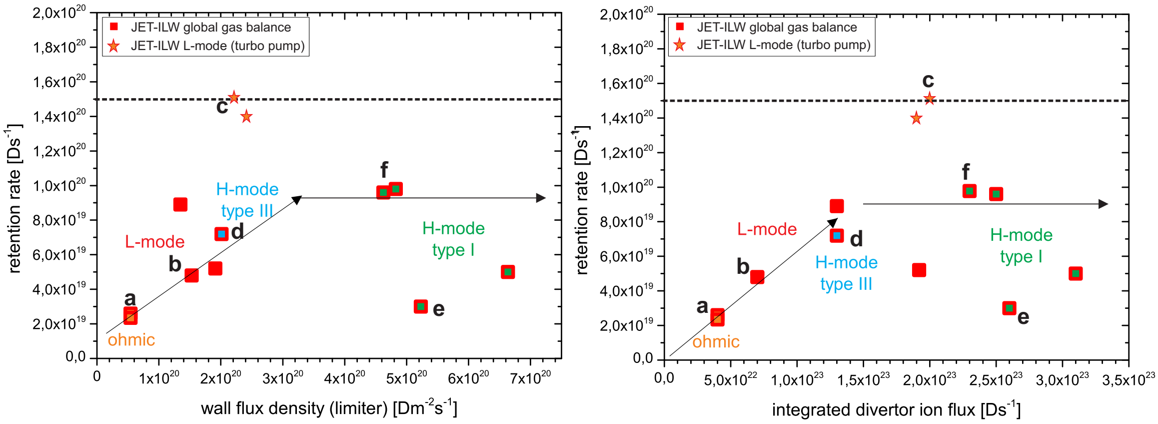

FIG. 5: JET-ILW experiments: Retention rate as function of ion flux density to the Be limiter in the main chamber. b) Retention rate as function of the integrated ion flux to the W divertor. The upper limit of the measured retention rate is indicated by − − −.

Overall, the retention rate increases linearly in the low flux region with either ion flux to the divertor or to the main chamber starting from ohmic **(a)**, to L-mode **(b)**, and finally to type III ELMy H-mode plasma conditions **(d)** as indicated by the continuous line in fig. 5. The repetition rate and the outgassing phase in the plasmas **(a, b, d)** are comparable. The increase is weaker and saturates at a rate of about $1.0 \times 10^{20} Ds^{-1}$ for the type I ELMy H-mode plasma experiments such as **(f)** where the inter-shot outgassing time is twice as long as in the low power experiments. The variation of the retention in the H-mode plasma is likely caused by additional outgassing due to different sequences of the discharges in the global gas balance experiment. The outgassing might also mask a further increase of the retention rate with the particle flux at the high flux region. Extrapolation of the observed trend in the lower power experiments would lift-up the H-mode data to about $1.5 \times 10^{20} Ds^{-1}$. The maximum measured retention value with the JET-ILW is $1.5 \times 10^{20} Ds^{-1}$, obtained in the gas balance with turbo molecular pump only accumulating in 34 discharge 731 plasma seconds. This value represents the absolute upper limit for the retention rate in all JET-ILW experiments and will be applied in the next section for the extrapolation to ITER.

Fuel retention by co-deposition of fuel with Be can explain the increase of the retention rate with the ion flux to the main PFCs in both divertor and main chamber. Retention by implantation in Be is expected to be saturated quickly due to the narrow interaction zone. However, to determine finally the main physics mechanism responsible for the fuel retention under the new metallic wall conditions, information from post mortem tile analysis is mandatory. But from our present knowledge the most likely mechanism for the remaining fuel retention in the JET-ILW experiments is co-deposition of fuel in Be co-deposits. This is in line with the measured Be influx from the main chamber into the inner divertor leg whose plasma-facing surfaces are a net deposition zone [24]. Co-deposition in the Be main chamber takes also place and contributes to the overall retention as a gas balance in limiter plasmas indicates.

The retention of fuel in Be, W and C co-deposits has been studied [20] at PISCES and corresponding scaling laws for the tritium retention in co-deposits expected in ITER deduced [29]. The scaling law for Be co-deposits, as an example, describes the fuel content as function of surface temperature at the instant of the co-deposit, the deposition rate of Be, and the incident energy of deuterium interacting with the Be co-deposit during its formation. It is valid for a range of temperatures ($20°C < T_s < 327°C$), energy range $15eV < E < 60eV$, and flux ratios of impinging Be/D ions ($10 < f_{D/Be} < 2000$) described indirectly by the deposition rate. These validity ranges cover the main wall conditions and, with limitations concerning the surface temperature, also the divertor conditions in the JET-ILW. Fig. 6 shows the scaling of fuel content as function of the temperature of the co-deposit at the time of deposition under otherwise constant conditions. A detailed

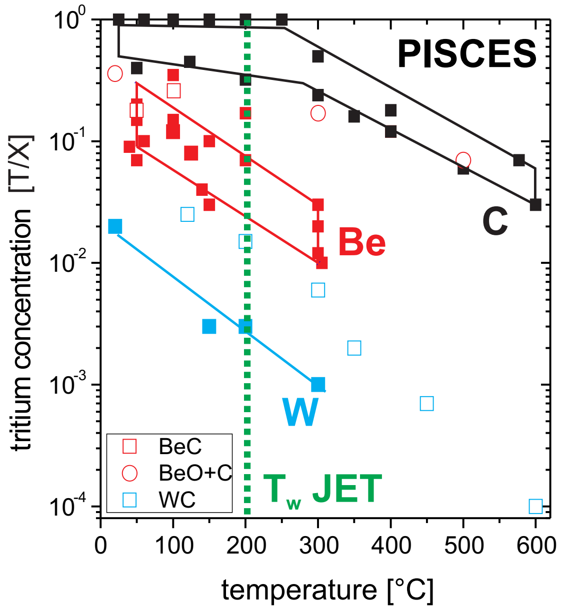

FIG. 6: Fuel concentration in Be, C, W and mixed layers as function of the surface temperature [3, 20]. The temperature of main chamber wall and limiters in JET is indicated.

comparison between JET-C and JET-ILW conditions by WALLDYN and ERO modelling, including the scaling laws, is pending. But a simple application of the PISCES scalings to the JET results is consistent with the surface temperature at co-deposition being the dominant reason for the order of magnitude drop in retention from JET-C to JET-ILW. We first note that from a direct experimental comparison of a pair of gas balance discharges from JET-ILW and JET-C, that the surface temperature of the PFCs and the impinging energies of impurity ions and deuterons are similar across those discharges. Secondly, we assume that there is no significant amount of Be in the JET-C case and the opposite true, insignificant C in the JET-ILW case. This replacement of the dominant impinging impurity ion from pure C in the JET-C case with typical $Z_{eff} \simeq 2.0$ by pure Be in the JET-ILW case with $Z_{eff} \simeq 1.2$ is in-line with the measured low residual C content. In a simple approximation relative pure C (JET-C case) and Be (JET-ILW case) co-deposits can be assumed. The Be influx from main chamber surfaces [24], as well as the low Be core concentration ($C_{Be} = 1.5\%$), suggests up to a factor 2 higher deuteron-to-impurity-ion ratio in the JET-ILW case in comparison with C in the JET-C case with an average $C_C = 3.0\%$ [9]. The reduction in the impurity-to-deuteron ratio and in the impurity concentrations is equal to a reduction of the main impurity source in the divertor phase by a factor 2 in JET-ILW in comparison with JET-C. This change in the primary source is, as previously mentioned, most likely caused by the reduction of the sputtering at the low energy end which occurs in the far-SOL of the main chamber. But the impact of the slight deuteron-to-impurity-ion ratio reduction on the residual fuel content in the corresponding co-deposits on the prediction by the PISCES scaling can in first order be neglected. With the above three conditions plus the assumption that the energy of the incident ions being of order $25eV$ for both JET-ILW and JET-C (choosing a different energy for both cases gives the same result given the scaling) the PISCES scaling predicts a factor of 8 drop in D/X due to co-deposition from JET-C (with $X = C$) to JET-ILW (with $X = Be$) under the same conditions. If we further take into account the lower primary source of Be and, thus, lower Be fluxes to the inner divertor during the divertor phase, then the overall drop in retention predicted would be about a factor of 16 at temperatures of $200°C$ and above. That prediction on basis of the lower D content in Be co-deposits in comparison with C co-deposits as depicted in fig. 6 may therefore explain well the observed reduction of the fuel retention rate in JET-ILW in comparison with the all carbon wall. However, one should point out that other variables could potentially be also important: (i) the surface morphology and the layer composition are of significance for the fuel content, too, and so far no information about the type of co-deposits in JET-ILW have been obtained though residual O and C levels are low and the appearance of complex mixed layers (Be, O, C) is likely limited; (ii) We have assumed in first approximation that the primary source in the main chamber is migrating to the divertor in the same way for C and Be driven by scrape-off layer flows. The distribution of Be within the divertor by subsequent sputtering of Be co-deposits is likely to be different in comparison with C in JET-C conditions due to the low Be sputtering yield at the low energy end. Much less Be migration to remote areas in JET-ILW than in C in JET-C is expected. Finally, co-deposition of D with W is not of importance and implantation is main mechanism for the retention in W as laboratory experiments in PISCES [30] and experiments in the all-W tokamak ASDEX Upgrade [31] confirmed, but it plays itself a negligible role in comparison with the Be co-deposition.

4.2 Long-term outgassing

Within the explored operational range from ohmic to H-mode plasmas, the long-term retention rate with the ILW exhibits a significant decrease by at least a factor 10 compared to carbon wall references leading to an upper limit of $1.5 \times 10^{20} Ds^{-1}$. Long-term outgassing, described for Be limiters in detail in [32], has also been observed in-situ in the post pulse operation of JET-ILW for about $100h$ with a total pressure and a partial pressure (mass 4: D_2) decay according to $\propto t^{-0.8}$ as shown in fig. 7, and leads to further reduction of the fuel content in the metallic PFCs. As the gas reservoir in the wall is finite, gas balance measurements can only be regarded as upper limit on the absolute long-term retention. Slightly weaker outgassing with a dependence of $\propto t^{-0.7}$ has been found in tokamaks with graphite first wall materials such as TEXTOR [33, 34] and TORE Supra [35] and was attributed to outgassing from co-deposits and bulk material. But the fuel recovery by outgassing in the times between pulses and in the non-operational time is more of relevance in the long-term fuel content with JET-ILW due to the lower absolute level of retention in comparison with the operation of JET with carbon walls. In the gas

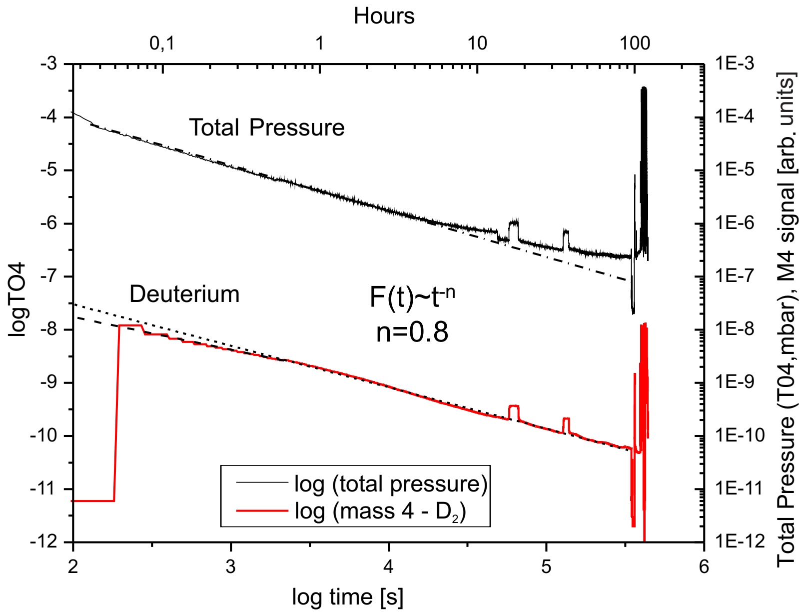

FIG. 7: Long-term outgassing over 100h after execution of a plasma pulse with the JET-ILW.

balances experiments comparable transient short term retention in JET-C and JET-ILW takes place, but the fraction of short-term to long-term retention is a factor ten different between JET-ILW and JET-C. Therefore it is expected, that the post-mortem analysis of Be and W components will show a significant lower long-term fuel content. These data will be available in the near future and will provide information on the long-term fuel content in co-deposited layers and in the bulk material by implantation - complementary to the gas balance studies reflecting a one day experiments presented here. For comparison, in the JET-C operation period 2001-2004, the post-mortem analysis revealed two years after tile extraction from the vessel about 4% or $66g$ of retained D in divertor and limiter PFCs [14], providing an averaged retention rate of $2.4 \times 10^{20} Ds^{-1}$ when normalised to the operational time of $8.3 \times 10^4 s$. This needs to be compared with a typical retention rate of $2.0 \times 10^{21} Ds^{-1}$ in JET-C gas balances [11] which reveals almost an order of magnitude difference between the two techniques due to outgassing. The measured retention rate by gas balance with the JET-ILW is more than one order of magnitude lower than in JET-C and below $1.5 \times 10^{20} Ds^{-1}$, thus, if the outgassing behaviour is at least comparable to CFC, a dramatic lower retention by post-mortem analysis in comparison with JET-ILW gas balances can be expected.

5 Conclusions for ITER

In carbon-dominated machines, as previously in JET-C, the fuel retention is determined by co-deposition of fuel in carbon layers on plasma wetted, and in remote and partially inaccessible areas. Short-term retention and outgassing has been documented in experiments with carbon walls, but the magnitude was small with respect to the long-term retention in JET-C of typically a few $10^{21} Ds^{-1}$ measured as an integral through multiple discharges by global gas balance $1.5h$ after plasma operation. The retention rate measured by gas balance dropped substantially by more than one order of magnitude with introduction of the full metallic wall in JET to values below $1.5 \times 10^{20} Ds^{-1}$ which is the upper limit. In H-mode plasmas the retention rate is below $1.0 \times 10^{20} Ds^{-1}$ if the longer outgassing period between pulses is considered. The main mechanism and the spatial distribution of retained fuel, i.e. main chamber or divertor, has been studied with the ion fluxes to the main PFCs, however, post-mortem analysis as complementary is required for full interpretation. Short term retention and subsequent outgassing are observed with the JET-ILW at a similar magnitude as in JET-C [19], thus the importance of this transient contribution to the overall fuel retention is now increased, as the measured long-term retention on basis of the one day gas balance experiment is dropped by one order of magnitude.

The results on fuel retention with the JET-ILW confirm qualitatively the predicted reduction of fuel retention in ITER with Be first wall and W divertor compared with a hypothetical full carbon ITER described in the comprehensive studies in [3]. It should

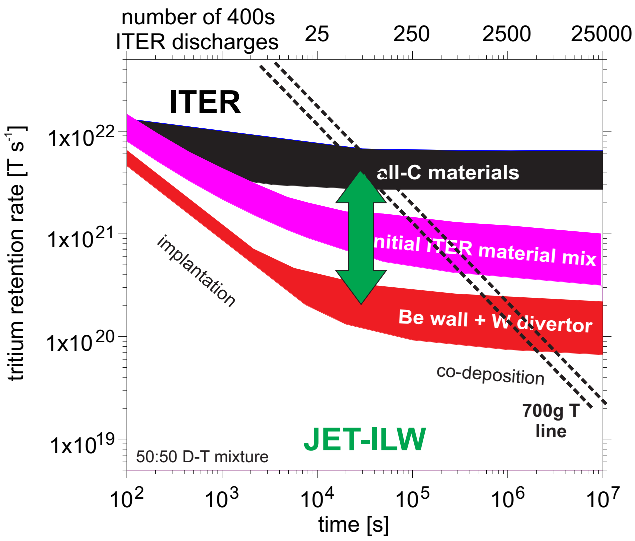

FIG. 8: From [3] derived temporal evolution of the fuel retention rate for three different ITER PFC combinations. The vertical lines indicate the current ITER tritium inventory limit of $0.7kg$; the $1.0kg$ inventory is marked for comparison. The operational time of JET-ILW and the reduction in retention rate from JET-C to JET-W is marked by the green arrow.

be noted that no experimental comparison exists with a tokamak with Be walls and carbon divertor. In figure 8 the tritium retention rate, thus half the total retention rate of D and T, is shown as function of operational time in full performance ITER 50:50 DT plasmas for three different PFC material configurations. In the case of the Be/W mix two phases can be distinguished in which the retention takes place; the fuel implantation in Be and W dominates at first followed by fuel co-deposition with Be. The first process saturates in time which is reflected in the strong reduction of the retention rate

depicted in fig. 8 whereas the almost constant rate in the later phase indicates the linear process of co-deposition which is qualitatively comparable to carbon - but more than one order of magnitude lower. Corresponding modelling on the basis of JET-ILW results of the first year has started, however it can be expected that the modelled behaviour will be qualitative similar with the constraint that the implantation-dominated part will be less pronounced due to the about four times smaller first wall area of JET with respect to ITER.
In the following we offer two scalings to estimate the T retention in ITER based on the JET-C and JET-ILW results. In the first scaling prediction, we assume that the ratio of retention rates measured in JET-C to that predicted for a hypothetical ITER with C PFCs can be applied to the ratio of retention rates for JET-ILW (measured) to the ones for ITER with Be/W PFCs. The measured JET-C retention rates, which are in the range of $1.0 - 3.0 \times 10^{21} Ds^{-1}$, are compared with the modelled values for a ITER with C PFCs, which are in the range of $0.4 - 1.2 \times 10^{22} Ds^{-1}$ at a cumulative operation time of $1 \times 10^5 s$, close to the total operation time with JET-ILW of $0.7 \times 10^5 s$. The ratio between the JET-C and an ITER tokamak with C PFCs is 4. Taking this ratio into account and applying the retention reduction found from JET-C to the JET-ILW with the maximum measured retention rate of $1.5 \times 10^{20} Ds^{-1}$, the D+T retention rate for ITER with Be/W is projected to be $6.0 \times 10^{20} (D + T)s^{-1}$ corresponding to $3.0 \times 10^{20} Ts^{-1}$ for the phase where co-deposition dominates. The predicted retention rate represents an upper limit as outgassing over nights and non-operational time as well as the transient contribution from the limiter phase is not considered here. The number of ITER full performance discharges without active cleaning would amount about 1500 assuming the tritium inventory limit of $700g$. This is a significant increase of the operational time without intervention for fuel removal in full plasma performance with respect to both a hypothetical CFC first wall and the currently baseline configuration with CFC at the strike-points.
The observation of a constant ratio of 4 between the measured JET and predicted ITER retention flux can, under the assumption of otherwise similar conditions (impact energies, surface temperatures and plasma parameters), also be translated into consistency of the ion flux density to the wall in H-mode conditions in ITER and JET. The observed saturation of the retention in JET-ILW H-mode plasmas with the ion flux density supports this assumption of a constant ion flux density in H-mode plasmas. The maximum observed total ion flux to the wall in the covered JET-ILW experiments (**f**) amounts $4.8 \times 10^{22} Ds^{-1}$ and can then be translated by the area factor into an integrated wall flux of $2.0 \times 10^{23} Ds^{-1}$ for ITER and the retention rates mentioned before. This wall flux lays within the range of predicted ITER wall fluxes between $\Gamma_{wall} = 1 - 10 \times 10^{23} (D + T)s^{-1}$ provided in the ITPA report [36] as consensus of different experiments and extrapolations.
In the second approach, scaling of the retention rate with respect to the impinging ion flux to the first wall, the primary source of material erosion, as suggested in [25], or the divertor ion flux, where co-deposition with Be takes place, can be performed on basis of the compte JET-ILW data set. We can deduce the conversion factors of these wall fluxes to retention rates based on the data of Fig. 6: Considering the JET-ILW ohmic (**a**), L- (**b**) and type III ELMy H-mode (**d**) plasmas as well as the available H-mode plasmas, but interpolated the latter to only $850s$ outgassing to be comparable to the low power plasma conditions, the following scaling of the fuel retention rate with the ion flux to the

wall can be obtained $\Gamma_{retention}[Ds^{-1}] = 1.20 \times 10^{19} Ds^{-1} + 0.00378 \times \Gamma_{wall}[Ds^{-1}]$. Using the average value of $\Gamma_{wall} = 5 \times 10^{23}(D+T)s^{-1}$ for the predicted ITER ion flux from the ITPA report, the retention rate in ITER extrapolates to $2.0 \times 10^{21}(D+T)s^{-1}$ or equivalent to $1.0 \times 10^{21} Ts^{-1}$ for tritium alone. This predicted retention rate is a factor 3.3 higher in comparison with the first scaling assuming a constant wall ion flux density made before. However, it should be clear stated that all retention rates employed in the scaling represent upper limits. Long-term outgassing is not included in the prediction and would lead to a strong reduction of the retention rate. Consideration of the longer inter-shot outgassing phase in JET-ILW H-mode plasmas of $2700s$ alone changes the scaling significantly and reduces the predicted retention rate by a factor 2 for the average value of the ITER ion flux to the wall. Transfer of the difference in retention determined by post-mortem analysis and gas balance experiments from the JET-C to the JET-ILW area due to longer outgassing will at least provide another factor of 5 reduction considering the same type of outgassing behaviour as shown before in sec.4.2.

6 Summary

Global gas balances, measured as an integral through multiple discharges by global gas balance $1.5h$ after plasma operation, have been executed in different plasma conditions and confinement regimes in JET with the Be/W plasma-facing material combination (JET-ILW) and compared with dedicated reference experiments performed priori in JET with carbon walls (JET-C). The long-term retention rate of deuterium normalised to the divertor phase drops substantially by more than one order of magnitude (factor 10-20) with the introduction of the full metallic wall in all performed experiments to values below $1.5 \times 10^{20} Ds^{-1}$ which represent the absolute upper limit. In high power H-mode plasmas the retention rate falls below $1.0 \times 10^{20} Ds^{-1}$ if the twice as long outgassing period between pulses in comparison with JET-C is considered, indicating the increased importance of the short-term retention in the JET-ILW. Though the short-term retention, potentially transient over-saturation of the surface, and post-plasma outgassing is in absolute magnitude comparable between JET-ILW and JET-C, the relative importance is increased as the underlying long-term retention is dramatically reduced.

The main mechanisms for the long-term retention are implantation in Be and co-deposition of fuel with Be. Implantation saturates with impinging ion flux and co-deposition can increase linearly with ion flux and operational time. From the whole series of gas balance experiments and the high reproducibility of H-mode experiments in the middle and the end of first year of ILW operation, accumulating $20h$ of total plasma operation, we can conclude that the dominant mechanism for the residual long-term retention remains unchanged in this operational period. This points to co-deposition of fuel with Be as dominant process. Co-deposition with W in the divertor is negligible in comparison with Be. According to the present knowledge, the residual fuel retention in the JET-ILW can therefore be attributed to the following three points: (i) Reduced tritium content in Be co-deposits in comparison with C co-deposits at JET wall temperatures. The change in the fuel content can be, according to fuel-content scalings deduced in PISCES, responsible for about a factor ten of the reduction assuming otherwise comparable plasma and surface conditions. Material mixing is not considered as residual C and O content in the plasma discharges are negligibly low. (ii) The primary impurity source in the main chamber during divertor operation, C in JET-C and Be in JET-ILW, is reduced by a factor 2 in the JET-ILW environment. In JET-C chemical sputtering of C at low electron temperatures takes place whereas low energetic sputtering of Be in the JET-ILW case is absent. The impurity flux towards the divertor, co-deposition of fuel in the divertor, and the retention rate are reduced by a factor 2 with respect to JET-C. This assumes that the migration pattern towards the divertor remains comparable between JET-C and JET-ILW. (iii) Minor long-term retention in plasmas that are magnetically limited has been recorded. As all JET plasma scenarios apply the discharge start-up in limiter configuration, the normalisation of the retention rate to the time in divertor configuration overestimates the retention by a about 15% assuming a ratio of $3:1$ between divertor and limiter configuration duration.

The dependence of the retention rate on the impinging deuteron flux to the main PFCs responsible for either deuterium implantation, Be erosion by deuterium or co-deposition of deuterium with Be, has been analysed within the available data base of gas balance experiments. Thereby, the ion flux to the wall and to divertor are proportional to each other for low recycling divertor conditions. The retention rate increases with the ion flux to the main PFCs starting with ohmic and ending with H-mode confined plasmas. Further increase of the ion flux in H-mode conditions by raising input power and density shows no difference in the retention rate and suggests a saturation at the highest observed wall fluxes in JET of $4.8 \times 10^{22} Ds^{-1}$.

Two predictions of the retention rate in ITER have been performed on the basis of these findings: (i) Applying the observed JET-ILW scaling of the retention rate with the wall flux up to H-mode conditions and extrapolate to ITER with the aid of the wall flux predictions made by the ITPA [34]. (ii) Considering the observed saturation of the ion flux density to the wall and assuming constant deuterium wall flux densities in H-mode plasmas in JET-ILW and ITER. The integrated wall flux and the retention rate in ITER scales in this approach with corresponding JET-ILW values multiplied by 4 - the ratio of the first wall areas in JET and ITER. The extrapolated upper limit amounts $3.0 \times 10^{20} Ts^{-1}$ whereas no long-term outgassing beyond $1.5h$ after the last pulse is considered.

A comparison between the measured retention rates in JET-C and retention rate predictions made for hypothetical all-C ITER by Roth et al. [3] reveals also a factor 4. Also a reduction of the retention rate by a factor $10-40$ from a hypothetical all-C ITER and ITER with Be/W material mix in the regime where co-deposition dominates is predicted in [3]. The measured reduction of the retention rate from JET-C to JET-ILW of $10-20$ is in line with these predictions if the limited operational time of JET-ILW is considered. Both predictions for ITER with Be/W walls based on the JET experimental data extend significantly the predicted operational time in ITER before active cleaning methods for tritium removal are required. However, the exact number of discharges requires the operational sequence in ITER and needs further to include the observed outgassing between discharges, over night and weekends.

Plasma-edge and surface-interaction modelling with the WALLDYN and ERO code in connection with results from post-mortem analysis of tiles concerning layer formation, residual fuel content as well as material mixing properties will increase further the confidence and precision in extrapolations to ITER, but are outside of the scope of this contribution whose focus is primarily on experimental results with global gas balances in JET with the ITER-Like Wall made of beryllium and tungsten.

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