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ANALYSIS OF FUEL RETENTION AND RECOVERY IN JET WITH BE-W WALL

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Abstract

Extended operations with tritium and deuterium-tritium plasmas in JET in the past few years provided unique opportunities for assessing and comparing the in-vessel retention behavior of hydrogen isotopes. Subsequent dedicated tritium clean-up campaigns implementing specially designed operation sequences helped to reduce tritium content in plasma and in the exhaust gas and provided valuable input for the development of fuel retention mitigation and recovery strategies for ITER. Laser-based diagnostic methods installed and operated at JET for the first time offered the possibility of local in-vessel and time resolved measurements of fuel content in plasma-facing materials and deposited layers. Laser-induced desorption with detection by quadrupole mass spectrometry was in operation during the third deuterium-tritium campaign and throughout the subsequent cleaning phase demonstrating tritium accumulation in co-deposited layers in the inner divertor and effectiveness of tritium removal from such layers by different cleaning methods. The paper summarizes respective research activities and related set of findings. Fuel recovery is compared in two clean-up campaigns showing a consistent picture and expanding the results with more recent datasets from in-vessel laser-based diagnostics performed during clean-up and after the end of JET operations.

1. INTRODUCTION

JET tokamak is the only fusion experiment to date that operated with large quantities of tritium (T) fusion fuel in all-metal wall configuration, establishing a new world record of 69 MJ fusion energy produced in a sustained reproducible T-rich scenario plasma pulse [1–3]. Extended operations with tritium and deuterium-tritium (DT) plasmas in 2020-2023 provided, among others, unique opportunities for assessing and comparing the in-vessel retention of different hydrogen isotopes.

JET started plasma operations with beryllium (Be) / tungsten (W) wall, at that time called the ITER-Like Wall (ILW, the name that will be also used throughout the paper), in 2011 [4] and concluded scientific plasma operations at the end of 2023, transitioning into the decommissioning phase. The last JET-ILW operation period (2019-2023) consisted of several D plasma campaigns, HT and T plasma campaigns, a short He campaign and two DT plasma campaigns, known as the second and third deuterium-tritium experiments, DTE2 [2] and DTE3 [3], respectively. Fig. 1 shows the schematic timeline of these campaigns and Fig. 2 compares fueling by different gases (hydrogen isotopes and helium) during the corresponding campaigns. Fueling comprises the gas injected into the torus via the gas injection valves, as well as atoms injected via the Neutral Beam Injectors (NBI). In DTE2 neutral beams were operated with deuterium and tritium [5], in DTE3 only with deuterium. Not included, however, is fueling via pellet injection and the gas introduced into the torus for disruption mitigation. Note that protium (H) is often injected in small quantities for the minority ion cyclotron resonance frequency (ICRF) heating [6, 7] and is therefore present throughout the campaigns in Fig. 2.

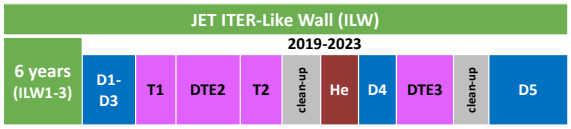

FIG. 1. Schematic timeline of different experimental campaigns in JET-ILW with focus on the last operation period (2019-2023). D1 to D5 are for different deuterium campaigns, T1 and T2 for tritium campaigns (including HT in the case of T1). For earlier deuterium campaigns ILW1-3 in JET see e.g. [8, 9].

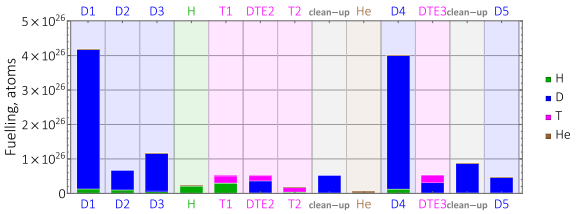

FIG. 2. Fueling by hydrogen, deuterium, tritium and helium in different experimental campaigns in JET-ILW during the last operation period (2019-2023). The campaign letters correspond to those in Fig. 1. T1 campaign includes HT operations.

In campaigns involving tritium (T1, DTE2, T2, DTE3) the total of about 359 g T was injected into the plasma for fueling, while the total T gas throughput (including the gas to supply NBI sources) was above 1 kg. This amount significantly exceeds the T fueling in earlier JET experiments involving tritium [10–12], which were all performed in JET with a carbon wall. For comparison, the total fueling in the first deuterium-tritium experiment (DTE1) conducted in 1997 was 35 g T [13].

Transition to tritium operations in JET-ILW started with a D→H changeover (included in D3 in Fig. 1 and Fig. 2) to avoid DT neutrons in the initial phase of the tritium campaign. This changeover experiment was also aimed at testing the future tritium removal strategy and was reported in detail in [14]. Tritium clean-up experiment followed after the 100%T campaign (T2 in Fig. 1) after DTE2 and was reported in [15]. After the extensive clean-up, a helium campaign was implemented that included a back-and-forth He→H and H→He transition [16] for wall conditioning studies and in view of Pre-Fusion Power Operation (PFPO) phase of ITER that was foreseeing He plasma operation before the decision of ITER re-baselining to full W wall was taken [17]. Prior to DTE3, the Laser-Induced Desorption with detection by Quadrupole Mass Spectrometry (LID-QMS) diagnostics was installed in JET as reported in [18]. It was routinely applied throughout DTE3 and during the subsequent T clean-up campaign to monitor tritium co-deposition in the divertor and effectiveness of clean-up procedures.

The two T clean-up campaigns that followed after DTE2 (after T2, in fact) and DTE3 started with specially designed operation sequences aimed at reducing the in-vessel T retention especially in view of T outgassing during D plasma operations and respective T content in the exhaust gas. The general initial cleaning sequence comprised vessel baking at 320°C (as compared to standard wall operation temperature of 200°C), followed by Ion-Cyclotron Wall Conditioning (ICWC) and Glow Discharge Conditioning (GDC, only after DTE2) plasma operation while still at baking temperature, after which the vessel was cooled down back to standard operation temperature and the Raised Inner Strike Point (RISP) plasma configuration was applied [14, 19], which allowed targeted heating of deposited layers at the inner divertor baffle, which is known to accumulate thick fuel containing deposits [8, 9, 20]. Details of the T clean-up experiment after DTE2, including pilot experiment prior to T operations, were reported earlier [14, 15], and an overview of fuel retention and recovery in all-metal JET after DTE3 showing preliminary results of gas balance analysis and implementation of laser-based diagnostics has been presented recently [21]. The obtained results provided valuable input for the development of fuel retention mitigation and recovery strategies in ITER [22].

This paper reports on the analysis of fuel retention and removal performed so far, providing some clarifications regarding the in-vessel gas balance, extending tritium recovery analysis to post-DTE3 clean-up in comparison to post-DTE2 clean-up, and adding new information from in-situ laser-based diagnostics, including the Laser-Induced Breakdown Spectroscopy (LIBS) measurements performed after the end of JET operations in a vented vessel under argon atmosphere [23, 24].

2. IN-VESSEL GAS BALANCE

As described in [21], there are two ways of performing gas balance measurements in JET, which are different in the way how gas quantification is performed: in-vessel and ex-vessel. For both methods the same experimental sequence is employed with the following prerequisites:

— The vessel pumping is restricted to in-vessel cryogenic pumping only;
— The same plasma configuration is repeated several times until the total amount of gas injected into the vessel reaches a certain limit prescribed by the gas quantification procedure as described further in the text;
— A certain time, typically 2 hours, is allowed after the end of plasma operations for wall outgassing;
— Cryo-panels are regenerated (from liquid helium to liquid nitrogen temperature) that leads to release of trapped hydrogenic species;
— Quantification of the recovered gas is performed using pressure-volume-temperature (pVT) measurement assuming the ideal-gas equation of state, with gas composition analysis depending on the method (in-vessel or ex-vessel) as described below.

The ex-vessel gas balance analysis includes an additional step between regeneration of cryo-panels and gas quantification, namely the recovered gas is pumped out of the vessel into the Active Gas Handling System (AGHS) of JET [25] and collected in a reservoir with known volume and monitored temperature. From the ideal-gas equation the total amount of collected gas is calculated, gas chromatography is applied for the composition analysis. The upper limit for the collected gas amount is given by the analysis chamber volume and pressure gauge measurement range, constituting 24 bar-liters (bar-l) at room temperature. The ex-vessel gas balance analysis is referred to as AGHS-pVT in [21]. Detailed dedicated studies of deuterium retention using the AGHS-pVT procedure were performed during the first ILW campaign in 2012 [26, 27].

For tritium operations, the AGHS system required significant recommissioning and enhancement of all subsystems [28]. In particular, the tritium compatible gas routing through AGHS (different from tritium-free operations) and the strict timeline of the tritium accounting cycle made the AGHS-pVT procedure not possible. To overcome this limitation and to be able to compare fuel retention in plasmas with different isotopes, as well as for quantification of tritium recovery by ICWC and RISP plasma during the clean-up, the in-vessel gas balance analysis was developed as a more accessible and lower-effort alternative. In this case the JET vessel itself is used as the measurement volume in the pVT procedure, which means that regeneration of cryo-panels is performed into the isolated torus, and the torus pressure rise allows inference of the amount of recovered gas [15, 21]. For the gas composition analysis, the sub-divertor neutral gas analysis system is used, essentially the Residual Gas Analysis (RGA) with Quadrupole Mass Spectrometers (QMS) [29, 30]. For this reason this procedure is referred to as RGA-pVT in [21]. The torus pressure is recorded with a Baratron® Capacitance Manometer (MKS 627D) located in the sub-divertor, and its upper measurement range of 10 Pa limits the total amount of gas that can be quantified to about 13 bar-l at room temperature. With the typical gas consumption in the selected plasma configuration (L-mode 1.5MW RF heated plasma with the inner strike point at the vertical tile and outer strike point on the semi-horizontal W tile, see Fig. 5a) of about 2.5 bar-l per pulse, the total number of pulses was restricted to 5. This method is generally associated with larger uncertainties due to the large vessel volume and temperature gradients involved. It has to be noted that RGA-pVT is not possible with pumping via TMPs (gas being directly pumped out of the vessel), and the involvement of NBI cryo-panels would increase uncertainties even further (additional volume and pumping by diagnostics), which motivates the restriction to the divertor cryo-pumping scheme described as prerequisite for the gas balance analysis in the beginning of this section.

Since the pVT procedure relies on the knowledge of the measurement volume and gas temperature for quantification of the gas from the pressure rise, and since neither the total JET volume nor the average gas temperature are precisely known, the RGA-pVT procedure requires a proper calibration to reduce uncertainties. By deliberate injection of a well-known amount of gas, from the respective pressure rise in the torus the ratio of the effective gas temperature to the vessel volume can be obtained. This ratio can be then used to deduce the amount of recovered gas after regeneration of cryo-panels, assuming the same effective gas temperature dominated by collisions with wall surfaces rather than intermolecular collisions. To test the plausibility of this approach, dedicated simulations with the gas transport code DIVGAS [31] have been performed. The 2D

simulation volume with realistic JET geometry for the main vessel, divertor, and upper sub-divertor region (including the cryo-pump surface) together with a simplified description of the lower sub-divertor with a pipe to the pressure gauge was used. In the simulation, the deuterium gas is released from the surface of the cryo-panel in the divertor in the amount corresponding to a prescribed equilibrium pressure level and is expanding in the entire volume. In the absence of pumping, the non-equilibrium steady-state condition is reached within less than a second of physical time, corresponding to mechanical equilibrium (uniform pressure) with a non-uniform temperature and gas density field due to spatially varying wall temperatures. This model allows to draw the following conclusion: the average gas temperature calculated from the ideal gas law can deviate from the true average gas temperature by a few percent; however, this discrepancy is cancelled out when the forward and backward calculation is performed – first the effective temperature is calculated from the calibration gas puff, then the total gas amount is calculated using the effective gas temperature.

Despite promising conclusions from the modelling, there are concerns about the accuracy of the method, which are attributed to several factors. First, the expected retention levels of a few percent imply that small uncertainties in the relatively large injected and recovered gas quantities result in large uncertainties in retention calculated as the difference between these values. Second, differences in the results of the in-vessel pVT calibration were observed when different gas injection modules and different gas quantities were used, which highlights that the simple mechanical equilibrium model cannot account for the full complexity of processes involved. Furthermore, torus pressure decay was measured during each experiment, despite the vessel being isolated, which indicates non-equilibrium situation attributed either to minor pumping through diagnostic equipment or gas-surface interactions. Additionally, quantification of different hydrogen isotopes is performed with sub-divertor RGAs connected to the torus through a small aperture, which ensures low pressure conditions for the instruments, but also modifies the isotopic composition due to mass-dependent effusive transport. Finally, in the cases of T and DT plasma, the presence of small quantities of hydrogen in the RGA measuring chamber leads to production of HT molecules, for which the QMS signal overlaps with D_2 at mass 4.

RGA-pVT analysis was performed several times during experimental campaigns T2, D4, DTE3 and D5. Moreover, RGA-pVT was used for the quantification of tritium removal by ICWC, limiter and RISP plasmas in the initial clean-up experiments. It must be noted that in several cases RGA-pVT resulted in a positive total gas balance, meaning that more gas was collected than injected into plasma. This was clearly the case during baking in clean-up experiments, but also in some normal plasma operation cases and has to be attributed to continuous outgassing from PFCs, which is strongly linked to the experimental program preceding the day of the particular RGA-pVT experiment. Fig. 3a shows the gas balance results for the D, T and DT plasma, and Fig. 3b for the clean-up experiments, with conservative accounting for the uncertainties discussed above.

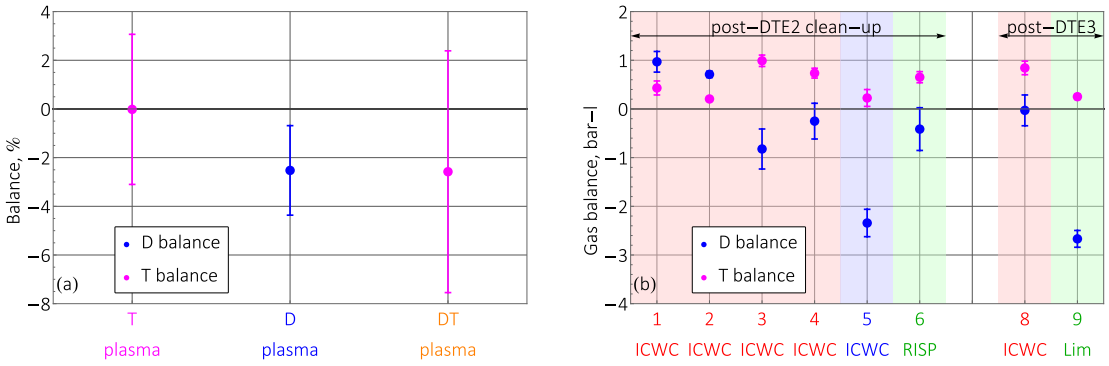

FIG. 3. Results of in-vessel gas balance RGA-pVT measurements for retention of deuterium and tritium in plasmas of different isotopes (a) and for tritium recovery during the initial post-DTE2 and post-DTE3 clean-up (b). Red shaded areas (1-4, 8) correspond to ICWC during vessel baking at 320°C. Green shaded areas (6, 9) correspond to RISP and limiter plasmas at standard wall temperature of 200°C. The blue shaded area (5) corresponds to ICWC performed during gradual warm up of the vessel from 110°C to 200 °C.

From Fig. 3a follows that the accuracy of the method for quantification of retention during normal plasma operation is not sufficient. The main factor responsible for large error bars for T and DT plasma is, in fact, the HT contribution to QMS mass 4. Effectively the upper bound corresponds to a fully equilibrated mix with HT contribution according to the binomial distribution, while the lower bound neglects HT completely. From the observation of the relation between masses 2, 3 and 4 it follows that the distribution of H2, HD and D2 is far from binomial, with HD component being much smaller. Assuming this holds also for HT, similar retention levels

below 4% in almost pure D and T plasmas can be concluded. Very large uncertainty level of tritium balance in DT plasma does not allow a solid conclusion. In overall, there is no indication that retention of T is significantly different from retention of D. During the clean-up, on the contrary, due to clear positive balance with more gas being recovered due to outgassing from the walls than injected into plasma, and since there was no T injected at all, the method can be applied for quantification of the recovered tritium, providing reasonable uncertainty estimates.

In Fig. 3b the results of RGA-pVT are compared for post-DTE2 and post-DTE3 clean up experiments with ICWC, limiter and RISP plasmas. In the post-DTE2 clean-up reported in detail in [15], the first 4 ICWC operation days were during the vessel baking at 320°C. During the first 2 ICWC days technical issues with the RF plant significantly limited the actual plasma time. This explains lower T recovery compared to days 3 and 4, which in this case should be attributed mostly to thermal outgassing by baking. Days 3 and 4 show respectively higher tritium recovery due to efficient isotope exchange with deuterium during ICWC as can be seen from the negative D gas balance. Day 5 followed after the end of baking and vessel cooling down from 320°C to 110°C (to access ICWC efficiency at more ITER-relevant wall temperature [15]). During this day of ICWC operations, the vessel was gradually warmed up to the standard operation temperature of 200°C. Very strong negative balance of D is explained by the fact that the wall reservoir was depleted by baking, while further out-diffusion was limited at reduced wall temperature. The same behavior was observed in the post-DTE3 clean-up after baking when limiter cycling pulses were applied instead of ICWC (point 9 in Fig. 3b). Remarkably, even in these cases some tritium recovery was detected.

3. COMPARISON OF INITIAL TRITIUM RECOVERY AFTER DTE2 AND DTE3

Initial tritium recovery procedures after DTE2 and DTE3 and their timelines are described in detail in [15] and [21], respectively. While the post-DTE2 clean-up had a strong scientific program component, the motivation for the post-DTE3 clean-up was more focused on fast and safe reduction of the tritium content in plasma and exhaust gas. Thus, the initial clean-up was minimized to be able to proceed as soon as practicable to high power deuterium plasma operations for further tritium removal. For this reason, baking was performed at the highest possible temperature of 320°C that was shown to be much more efficient compared to baking at 240°C in the post-DTE2 clean-up [15], and GDC operations were omitted owing to risk-mitigation concerns in view of possible technical failures at high wall temperature. In addition, no RGA-pVT was possible on the days of RISP plasma operation, as this would significantly restrict the gas throughput and thus tritium recovery achievable per day. For this reason, only RGA-pVT for the limiter cycling pulses that have much lower gas throughput is available (point 9 in Fig. 3b). The analysis of tritium recovery by baking and ICWC in the initial post-DTE3 clean-up has been performed following the same procedures as described for the post-DTE2 clean-up [15]. The results are compared in Fig. 4. The results of both clean-ups are in reasonable agreement with each other. The differences can be attributed to the fact that DTE2 ended with 100%T campaign, thus enriching the walls with tritium as compared to DTE3 with its mixed DT plasmas. Absence of GDC cleaning in post-DTE3 is another factor. The total amount of tritium injected over T1, DTE2 and T2 campaigns was about 252 g as compared to the total of about 109 g injected in DTE3 (the factor of about 2.3 less). The total quantified amount of recovered tritium in post-DTE2 clean-up (~0.67 g) is consistently approximately the same factor greater than the amount of T recovered after DTE3 (~0.28 g). Both initial clean-up experiments were followed by dedicated D campaigns with several hundred high power pulses in different divertor configurations, in which the T concentration in the gas exhaust was gradually reduced to the safety limit of 0.02%. The overall tritium accountancy by the tritium re-processing plant within AGHS is still ongoing and is expected to be completed in spring 2026.

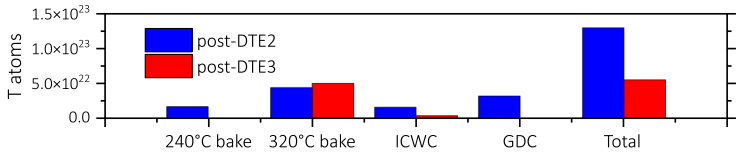

FIG. 4. Comparison of tritium recovery by different methods during the initial clean-up experiments after DTE2 and DTE3. Note that after DTE3 neither baking step at 240°C nor GDC were performed, which explains the absence of corresponding data in the figure.

4. APPLICATION OF IN-VESSEL LASER-BASED FUEL RETENTION DIAGNOSTICS

The LID-QMS diagnostics was installed and tested in JET in 2023 in preparation for DTE3. It is capable of targeted heating of different locations on four toroidally adjacent tiles in the inner divertor (apron Tile 0 and the

upper part of Tile 1, Fig. 5). The laser is focused on the tiles to a 3 mm diameter spot and operates with a pulse duration of 1 ms. Controlling the laser power and pulse duration allows to heat deposited Be layers above the melting point. The diagnostics can operate either in a single pulse mode or with a quick areal scan by repetitive laser pulses to increase the heated area and fuel release, respectively. First quantitative results of *in-situ* measurements were presented recently [32] demonstrating reasonable agreement with earlier performed *ex-situ* analysis using the same method in laboratory [18].

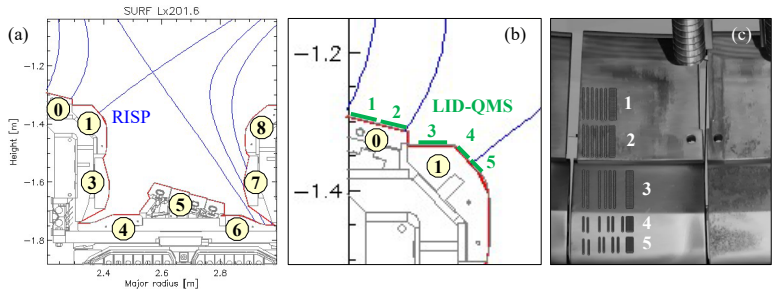

FIG. 5. The JET-ILW divertor geometry: (a) poloidal cross-section with tile numbers and RISP plasma configuration magnetic flux surfaces; (b) closer look at the upper part of the inner divertor (Tiles 0 and 1) schematically showing LID-QMS line scans performed during the post-DTE3 clean-up; (c) view on the inner divertor Tiles 0 and 1 with LID-QMS poloidal scan numbers corresponding to figure (b), poloidal lines at different toroidal positions correspond to actual days of measurements.

Among different types of LID-QMS measurements performed, the following two are highlighted:

(a) Monitoring of changes on the pre-cleaned area on Tile 0 (in a region equivalent to location 2 in Fig. 5c but on a neighboring tile) as reported in [21]. Such measurements were performed 4 times during DTE3, including a reference measurement before the clean-up, 3 times during the initial clean-up (during baking before ICWC, after baking before RISP, and after RISP plasma operation), as well as semi-regularly during the subsequent extended clean-up using high power D plasmas.

(b) Assessment of clean-up effectiveness of different clean-up stages using poloidal line scans on adjacent toroidal locations pre-selected to have rather uniform deposit thickness and fuel content as followed from earlier investigations [18]. In this case 5 distinct poloidal line scans were performed each day of measurement as shown schematically on the poloidal cross-section of the divertor in Fig. 5b and as a photograph in Fig. 5c. The toroidal position was shifted slightly each day of measurement so that LID-QMS results indicated the residual fuel content at each stage (before baking, during baking, before and after RISP).

The detection of desorbed gases by LID-QMS using RGA systems in JET is sensitive to pumping conditions and background pressure in the vessel. To account for the particular conditions and be able to provide quantitative fuel removal data, calibration gas injections using short D_2 gas puffs were performed each day of measurement [21, 32]. From the jump of QMS mass 4 signals in response to a given injected gas amount, calibration factors for each RGA can be determined [33]. Very good detector linearity of QMS mass 4 signals response with respect to the amount of puffed gas was observed in the range of pressure change corresponding to laser-induced gas release [32, 33]. In the absence of systematic calibration with respect to H_2 and T_2 gas injections, two approaches are adapted in the following that determine the measurement uncertainty. In the first approach the QMS mass sensitivity is assumed to be the same across all masses. This is a conservative assumption that should correspond to the minimal tritium contribution. In the second approach, results of earlier RGA calibrations performed after the T2 campaign (Fig. 1) are adopted, in which mass sensitivity is described by a power law with respect to the isotope mass [30], with a negative exponent β, the value of which was shown to be close to -2. This assumption leads to lower QMS sensitivity to mass 6 (T_2) compared to mass 4 (D_2, but also HT), and therefore a higher tritium contribution. At the current stage of analysis, possible HT contribution to mass 4 is assumed to be negligible. Using these assumptions, results of line scans at positions 1 and 2 on Tile 0 (Fig. 5b and Fig. 5c) at different stages of the clean-up sequence are shown in Fig. 6. Increase of fuel release during baking is to be attributed to effective out-diffusion. Position 2 shows otherwise a clear decreasing trend with cleaning stages. On the contrary, an almost constant or even slightly increased fuel release after RISP at position 1 located deeper in the scrape-off layer may indicate re-deposition at this location during RISP plasma operation. LID-QMS measurements continued for several weeks after the initial clean-up. In the last measurements before the end of JET operations most measurements resulted in values at or below the limit of detection.

After the JET shutdown and subsequent venting, LIBS measurements were performed at a variety of locations in the vessel (over 800 locations in total) using the remote handling arm MASCOT [34], covering also the tiles in

the inner divertor [23, 24]. LIBS uses a much shorter nanosecond laser pulse with significantly higher power density that leads to ablation of a surface layer of material (~180 nm for a W surface in the JET setup) and creation of short-lived laser-induced plasma cloud near the surface. Light emission of the local plasma is detected by several spectrometers simultaneously and thus can be used for material characterization. Up to several hundred shots were performed on each single spot allowing depth-resolved measurements of material composition and fuel retention. Restricted by the resolution of the Littrow spectrometer used for Balmer-alpha line identification of hydrogen isotopes, no detectable tritium signal could be obtained due to the Stark broadening and low T/D ratio, estimated to be below 10% [23]. The relative deuterium contribution compared to hydrogen was decreasing with pulse number at the same location [24], indicating the reduction of deuterium content with the depth.

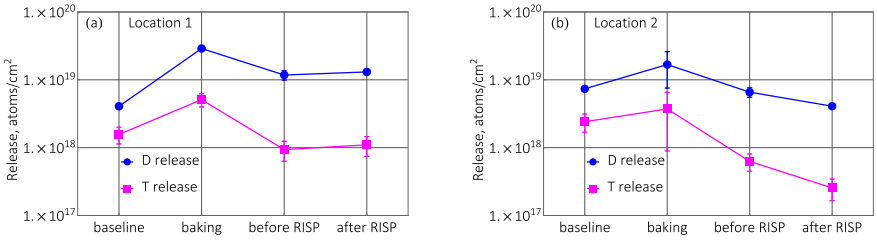

FIG. 6. Results of quantification of fuel release by LID-QMS at different stages of the clean-up sequence for line scan measurements at locations 1 (a) and 2 (b) from Fig. 5b and Fig. 5c.

5. SUMMARY

The recent tritium and deuterium-tritium campaigns in JET-ILW offered unique opportunities to analyze fuel retention behavior of different hydrogen isotopes. A huge mass of data has been accumulated, processing of which is still an ongoing endeavor, as well as the global tritium accounting that is expected to be completed in spring of 2026. To analyze the dynamic hydrogen isotopes retention, the in-vessel RGA-pVT procedure has been implemented that, although being more accessible as compared to ex-vessel AGHS-pVT procedure, turned out to offer limited accuracy for quantitative analysis. The obtained results suggest similar dynamic retention for tritium as observed earlier for deuterium – at the few percent level. During the initial cleaning sequences applied after DTE2 and DTE3 campaigns, the in-vessel RGA-pVT could however provide reasonable measurement accuracy in quantification of tritium recovery. The overall tritium recovery by these measures is consistent between post-DTE2 and post-DTE3 clean-up, being proportional to the total amount of tritium used during the respective T and DT campaigns. *In-situ* laser-based fuel retention diagnostic methods demonstrated their capabilities in detecting hydrogen isotopes locally, time-resolved (LID-QMS) and depth-resolved (LIBS). LID-QMS measurements allowed to monitor the tritium retention every week of DTE3 and after the campaign following the T content depletion due to different cleaning steps. Taking advantage of lessons learned at JET, the LID-QMS diagnostic is under design as Tritium Monitor Diagnostic for ITER. LIBS measurements performed in Ar atmosphere (locally inside the measurement head on the remote handling arm) after the vessel venting allow 3D mapping of deuterium retention as well as material re-deposition in the JET vessel. Analysis of accumulated data from hundreds of analyzed spots will be continued and compared later to post-mortem laboratory analysis of retrieved samples.

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