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Overview of fuel retention and recovery in JET deuterium–tritium operation

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Abstract

Tritium inventory build-up is a safety and economic issue for next step fusion devices and power plants. JET with a beryllium wall and tungsten divertor brings a unique contribution on fuel retention and recovery in a metallic device, as it has operated with deuterium, tritium and deuterium–tritium (DT) plasmas. This paper documents the JET fuel retention programme and results focusing on the tritium, DT campaigns and tritium clean-up in 2023. In addition specific experimental details are presented in order to provide lessons learned for fuel retention and inventory assessment procedures for regulators and operators of future fusion devices. Fuel retention results from gas balance and Laser-Induced Desorption with gas detection using Quadrupole Mass Spectrometers (LID-QMS) are discussed. Gas balance has shown that there is no significant isotopic dependence of in-vessel global fuel retention, however a faster decrease in outgassing rate has been observed with increasing mass, likely associated with the difference in concentration and depth profile of tritium and deuterium. LID-QMS data has provided new local in-vessel fuel retention data demonstrating capability for measuring fuel retention, monitoring changes in fuel retention during an operating period as well as providing direct

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measurement of increased near-surface fuel concentration due to diffusion of hydrogen isotopes to the surface at elevated baking temperature and removal of fuel by inner strike point heating.

Keywords: JET deuterium–tritium operation, fuel retention, tritium clean-up, laser-induced desorption, LID-QMS

(Some figures may appear in colour only in the online journal)

1. Introduction

Tritium inventory build-up is a safety issue for next step fusion devices and power plants. In addition, the trapping of tritium within the vacuum vessel has implications for the fuel cycle efficiency and consequently economic viability of power plants. Therefore, fuel retention assessment in operating fusion devices is needed to develop understanding and procedures to meet regulatory requirements for future machines and power plants. Fuel retention measurements can be performed at different stages of the fusion cycle. At one extreme an overall *global* fuel retention measurement during operations is given by comparing the total fuel supplied versus the total fuel recovered. At the other extreme are the ex-situ *local* retention measurements on retrieved components. Such components will have lower tritium content due to tritium removal procedures performed in-vessel prior to venting and also outgassing from the components that occurs on venting. Between these extremes, *global* retention on shorter timescales and *in-situ local* measurements on plasma facing components (PFCs) are needed for better understanding of temporal dynamics and spatial variation of retention within the machine. In addition, a full range of procedures will be needed to inform retention throughout the fuel cycle of a fusion power plant. For example, understanding the *global* retention in the vacuum vessel is needed for optimising fuelling and for monitoring operational inventory safety limits and ex-vessel *local* retention measurements in components will inform safety in maintenance as well as waste reduction and categorisation. With this in mind, experiments on existing fusion devices have been performed to develop and demonstrate fuel retention measurement techniques. For example, gas balance measurements—also referred to as particle balance—in plasma pulse operations have been performed in a variety of devices. Results from gas balance experiments in JET, Tore Supra, ASDEX Upgrade, DIII-D are presented in [1–4] and have shown that retention is dependent on wall materials and plasma fuelling conditions. Gas balance has also been used to study dynamic retention and outgassing, isotope exchange and fuel removal, including with tritium [5–13]. These experiments generally require operation using one plasma scenario to provide results for the known operating conditions. Such assessments allow for predictive inventory accounting in given scenarios. A number of gas balance measurements have been performed in the JET-C (JET with carbon wall) and all metal JET-ILW (JET ITER-like wall with beryllium wall and tungsten divertor (Be-W)) configurations to show *global* retention in the vacuum vessel after one day of identical deuterium (D) plasma operations. Notably the comparison between JET-C and JET-ILW showed the reduction in fuel retention with D plasmas from ∼20% to ∼2% [2, 14].

Accountancy for tritium within the whole operating cycle will require longer term gas balance assessment. This was demonstrated in the first deuterium–tritium (DT) experiment in JET with C wall (DTE1) in 1997 where 35 g of tritium was injected into the vacuum vessel with 14 g retained at the end of DTE1 plasma operations. Tritium (T) retention was reduced to 6.2 g after clean-up in D and *hydrogen* (H), i.e. protium, plasmas and finally 4.2 g after venting. This gave the in-vessel T retention as 40% immediately after DTE1 plasma operations and the final long term retention for the whole DTE1 fuel cycle after venting as 11% [12, 16]. It is this data along with tritium retention data from TFTR [17, 18] that demonstrated unacceptably high retention in fusion devices with C wall and pointed to the need for moving to an all-metal wall to reduce retention caused by chemical sputtering and co-deposition of hydrogen isotopes (HI) with carbon in order to manage in-vessel tritium inventory in future machines and power plants.

Tritium clean-up will be required in DT fusion reactors to minimize fuel inventory or remove fuel to acceptable levels prior to maintenance. Throughout JET operations a number of isotope exchange experiments have been performed. Tritium clean-up experiments were reported for DTE1 [12, 13] with a carbon wall and DTE2 [5, 7] with Be-W wall. In addition, various HI or species change-over experiments, such as helium ↔ hydrogen [6] and hydrogen ↔ deuterium [19], provided data for the efficacy of clean-up using gas/isotope exchange.

Extensive ex-vessel *post-mortem* analysis of PFCs from fusion devices have taken place to establish long term fuel retention. Whilst *post-mortem* analysis gives *local* ex-vessel fuel retention inventories, this can be extrapolated for a *global* ex-vessel assessment and comparisons made with *global* in-vessel fuel retention measurements. This type of study requires extensive component retrieval which has been achieved in Tore Supra [20] and JET [21, 22]. In the case of JET this type of comparison has shown a difference in fuel retention of one order of magnitude between in-vessel fuel retention at ∼2% [2] and *post-mortem* analysis fuel retention ∼0.2% [21, 22]. The difference between these two approaches is likely to arise from outgassing over time and on venting the vessel to retrieve components.

During the JET tritium (100% T) and second and third DT experiments (DTE2, DTE3) there was an extensive programme to study fuel retention and tritium clean-up procedures. The results of 100% T and DTE2 are discussed in [15]. An overview of experiments coming mainly from DTE3 and subsequent tritium clean-up are presented here. The main aspect of the JET fuel retention programme during DTE3 operations were to study HI effects on *global* fuel retention and dynamic fuel retention with gas balance and outgassing experiments and to exploit a new Laser-Induced Desorption with detection by Quadrupole Mass Spectrometry (LID-QMS) diagnostic to provide in-vessel *local* fuel retention measurements during DTE3 and T clean-up.

In 2023 LID-QMS was installed on JET with the aim of supporting a future diagnostic planned for ITER. The diagnostic provided additional capability for measuring fuel retention in wall materials without the need for venting the vessel, providing in-vessel *local* fuel retention measurements for comparison with *global* fuel retention and with ex-vessel *post-mortem* measurements on retrieved PFCs [23]. A similar LID system has been demonstrated earlier on TEXTOR [24, 25] with detection of desorbed HIs using optical spectrometry, however, the installation in JET allowed for the first *in-situ* tritium measurements during the JET DT experiment (DTE3) operations carried out in 2023 and demonstration of laser induced desorption fuel measurements in a tokomak environment using QMS detection.

The aim of this contribution is to provide an overview of key findings and progress in the evaluation of fuel retention in DT plasma operations by presenting a selection of results from a range of experiments, including gas balance, LID-QMS and T clean-up. In addition, this overview gives an insight into the challenges faced when operating with tritium which adds to the complexity of measuring and assessing fuel retention and analysis with the intention of providing valuable knowledge transfer to future fusion devices which will need to define operational requirements for tritium accountancy. In section 2 the JET operations with T and D plasma will be summarized. An overview of the experiments performed for fuel retention studies is provided, followed by descriptions of the analysis procedures and diagnostics used experiments. Section 3 provides the results and analysis of data from the various experiments and initial findings to put the results in context. Section 4 discusses the results of the experiments and lessons learned and section 5 summarises in the context of ITER operations.

2. Experimental details

2.1. JET operations with tritium and DT plasmas

JET has operated with six dedicated T or DT campaigns: 1991 preliminary tritium experiment (PTE) [26], 1997 first major DT experiment (DTE1) [27], 2003 trace tritium experiment (TTE) [28, 29] all with a carbon wall; 2020–2022 100% tritium, DTE2 [15] and 2023 DTE3 with the JET-ILW.

JET operated in 2019–2023 without intervention into the vessel. In addition to deuterium and helium (He) plasma operations, one tritium campaign and two DT campaigns took place, as listed above. Figure 1 shows the sequence of the different campaigns. In total 364 g of tritium were injected into the JET vessel during 100% T_2, DTE2, and DTE3, equivalent to 130 PBq. Information about these campaigns including the amount of tritium injected and divertor plasma time are given in table 1. The injected fuel into the JET vessel is taken as the sum of the fuel introduced by Tritium gas Injection Modules (TIMs, operation and calibration are described in [30]) to fuel the plasma and Neutral Beam Injection (NBI) to heat the plasma. It should be noted that most of the tritium entering the vacuum vessel is via the TIMS. Whilst a large amount of tritium was supplied to the Neutral Injector Box in Octant 8 (NIB), only a small fraction of tritium from the NIBs is injected into the vessel for NBI heating. For example, during 100% T_2 & DTE2, when NIBs were was operated in tritium, 2.2% of tritium gas supplied to the NIBs was injected into the vacuum vessel during NBI heating. Similarly, in DTE3 when the NIBs were operating in deuterium, 1.5% of gas supplied to the NIBs was injected into the vacuum vessel during NBI heating.

2.2. Fuel retention experiments

In the following sections the experimental techniques, experimental strategies and diagnostics used in the overall fuel retention programme are detailed.

2.2.1. Gas balance technique. Gas balance relies on the quantification of the gas introduced into the vessel during plasma operations versus the amount of exhaust gas pumped to cryogenic pumping panels in the divertor. The vessel is isolated from external pumping to provide a closed volume throughout the experiment. The difference between these two values is the amount of fuel retained in the vessel either through co-deposition with sputtered impurities or by implantation and wall pumping. Gas injected into the vessel is quantified through controlled gas injection modules (GIMs) with known volume, operating temperature and pressure change. GIMs and TIMs (tritium safe GIMs) are part of the JET plant and are used routinely in the operation of JET [30]. No NBI heating is used; therefore gas introduction is only fuelling of the plasma.

For the gas balance technique one plasma configuration is repeated over a day until a specified amount of gas has been injected into the vessel with the aim of providing the *global* retention value for the particular configuration. The total amount of gas injected—and therefore the number of pulses—is limited by the gas quantification procedure performed after releasing the exhaust gas upon warming the cryogenic pumping panels from liquid helium temperature (4 K) to liquid nitrogen temperature (77 K), discussed in more detail below. There are two 'Pressure–Volume–Temperature' procedures used on JET to quantify gas; RGA-PVT making use of the JET vessel as the measurement volume and AGHS-PVT making use of a measurement volume in the JET Active Gas Handling System—the facility delivering, processing and storing tritium for JET T operations [31]. It should be noted that PVT measurement is for all gases collected in the volume, which includes protium in addition to deuterium and tritium.

The plasma configuration proposed for the gas balance experiments was a L-mode RF heated plasma with inner strike point at the vertical inner tile 3 and outer strike point on the semi-horizontal tungsten tile 5. The locations of the divertor tiles are shown in figure 2. The strike point configuration was held for around 10 s with 1.5 MW RF heating. This setup is

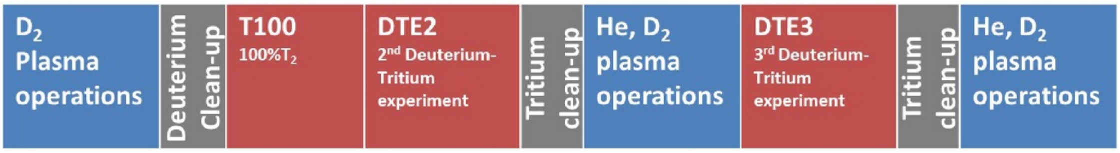

Figure 1. Schematic timeline of 2019–2023 JET operating period showing DTE2 [15] and DTE3 with clean-up phases indicated. The initial part of the tritium clean-up timeline following DTE3 is shown in figure 3.

chosen to be at minimal risk of disruption to prevent additional gas injection from the disruption mitigation system, and high probability of stable heating, as well as providing a comparison with earlier gas balance experiments.

AGHS-PVT: the pressure–volume–temperature analysis performed within the AGHS volume is described in [2] for *global* deuterium retention measurements. For the AGHS-PVT the released gases are pumped to a known volume with accurate pressure and temperature measurement. The amount of gas is limited to 24 bar-litres (bar-l) due to the volume and pressure gauge available. After AGHS-PVT quantification of the collected gas, gas composition analysis is possible using gas chromatography or QMS. Whilst this method has been completed in D plasma operations on several occasions in JET-C and all metal JET-ILW configurations [2, 14], for operation with tritium, different tritium compatible pumps and gas routes are needed in AGHS. As a result more complex tritium operating procedures are also needed when operating with tritium in AGHS which increases the risks in failing to collect all the gas in the analysis chamber. Therefore, trial runs without plasma were performed to rehearse the procedures for gas injection and AGHS-PVT followed by compositional analysis. This served to rehearse the plant procedures, to check the plant was fully operational (particularly the tritium gas route and tritium compatible pumps in the AGHS-PVT plant which was different to previous D_2 gas balance) and to quantify the uncertainty in the amount of gas injected versus gas recovered.

RGA-PVT: the pressure–volume–temperature and residual gas analysis performed in the JET vessel is described in detail in [5]. For the RGA-PVT the exhaust gases are released to the JET vessel, isolated from the gas pumping system. In the case of JET, the amount of gas that can be measured in the vessel is limited by the maximum pressure that can be measured with high accuracy on the available pressure gauges. The overall total that can be measured is 14 bar-l, however fuelling of plasmas is limited to a maximum of 12.5 bar-l. This is typically 6–10 plasma pulses in any one day depending on gas usage for the chosen configuration. Limiting the plasma fuelling to 12.5 bar-l leaves 1.5 bar-l capacity for additional gas injections for quantification purposes. These additional controlled small gas injections are introduced into the released exhaust gases to provide additional changes in pressure. This allows to overcome uncertainties in the volume (V) of the JET vessel (of the order of 180 m^3 with complex surfaces) and gas temperature (T) by quantifying the V/T ratio. The measured value of V/T is then used for the exhaust gas quantification. In parallel a gas composition measurement using QMS can be performed.

As the RGA-PVT is less resource-intensive than the AGHS-PVT, it can potentially provide greater flexibility for gas balance experiments. However, it implies generally larger uncertainties. This issue can in principle be overcome by cross comparison of both methods, for which the in-vessel RGA-PVT has to be followed by the AGHS-PVT, which is considered to be the more accurate method for *global* retention assessment.

Overall, one trial run, two gas balance experiments with D plasmas and two gas balance experiments with DT plasma were performed, however they were only partially successful. A selection of data and lessons learned are presented in sections 3.3 and 4.2.

2.2.2. Laser Induced Desorption —Quadrupole Mass Spectrometry measurements.

The JET LID-QMS diagnostic is described in [32]. The diagnostic consists of a 1071 nm wavelength, 25 kW, pulsed (1–3 ms) laser focused to a 3 mm diameter spot onto the upper inner divertor where there is significant retention due to co-deposition at the top of Tile 1 and the High Field Gap Closure Tile (HFGC, also known as Tile 0), shown in figure 2(*a*). The surface of the PFCs is heated using the laser resulting in fuel release which is measured using mass spectrometers situated below the divertor [33] and also in the main chamber.

The detection of desorbed gases by LID-QMS is sensitive to the pumping conditions and background pressure of the vessel. Pumping of the vacuum vessel is by cryogenic pumping panels and turbomolecular pumps. Background pressure is dependent on the time between the last plasma pulse and LID-QMS measurements and the outgassing rate. Therefore, in order to provide calibrated fuel removal data, it is necessary to carry out quantified gas injections to account for the overall pumping speed for the particular pumping configuration on any given day of LID-QMS operation. For JET, the lowest amount of D_2 gas that could be injected from GIMs was of the order 10^{17} molecules which is estimated to be the level of gas desorbed during LID-QMS measurements, discussed in section 3.1.

When using QMS for gas analysis, there is a possibility of overlapping masses of different isotopologues of the hydrogen molecule and ^{4}He contributing to the AMU4. Although there are QMS systems available for distinguishing between D_2 and ^{4}He, this was not available for the JET diagnostic. The contribution of HT (AMU4) was previously shown to be minimal in laser desorption, since measured H_2 (AMU2) and HD (AMU3) signals were low [32]. Also, no ^{4}He was injected into the vessel during DTE3 and ^{4}He ash from DT reactions is of the order 10^{20} atoms, which when assumed to be distributed around the vessel surface results in areal concentrations

Table 1. Details of JET operations with the amount of tritium and deuterium injected during T and DT operations.

Description/ Campaign	JET pulse numbers	Divertor plasma time (hours)	Total tritium injection (atoms/g)	Tritium injection TIM (atoms/g)	Tritium injection NIB (atoms/g)	Total deuterium injection (atoms/g)	Deuterium injection GIM (atoms/g)	Deuterium injection NIB (atoms/g)
TIMs test	98043–98048	0.03	5.67×10^{21}/ 0.03	5.67×10^{21}/ 0.03	0	9.76×10^{23}/ 3.2	9.23×10^{23}/ 3.1	5.27×10^{22}/ 0.2
T_2 operation/ C39T	98533–98597	0.24	3.66×10^{24}/ 18.3	3.66×10^{24}/ 18.3	0	0	0	0
T_2 operation/ C40	98598–99289	2.35	1.77×10^{25}/ 88.4	1.56×10^{25}/ 78.2	2.04×10^{24}/ 10.2	9.50×10^{23}/ 3.2	0	9.50×10^{23}/ 3.2
DT operation DTE2/C41	99294–99983	2.95	1.54×10^{25}/ 76.9	1.45×10^{25}/ 72.6	8.60×10^{23}/ 4.3	3.62×10^{25}/ 120.3	3.28×10^{25}/ 108.9	3.44×10^{24}/ 11.4
T_2/C40B	99984–100253	0.99	1.41×10^{25}/ 70.5	1.37×10^{25}/ 68.4	4.15×10^{23}/ 2.1	1.16×10^{23}/ 0.38	2.09×10^{22}/ 0.07	9.48×10^{22}/ 0.31
Tritium clean-up/C42	100254–100897	2.0	[a]1.20×10^{23}/ 0.6	[a]1.20×10^{23}/ 0.6	0	5.33×10^{25}/ 176.9	5.10×10^{25}/ 169.4	2.25×10^{24}/ 7.5
DT operation DTE3/C46	104127–104697	1.1	2.17×10^{25}/ 108.8	2.17×10^{25}/ 108.8	0	3.09×10^{25}/ 102.6	2.84×10^{25}/ 94.3	2.51×10^{24}/ 8.3
Tritium clean-up/C47	104698–105475	3.0	0	0	0	8.63×10^{25}/ 286.6	8.00×10^{25}/ 265.8	6.27×10^{24}/ 20.8

[a] Tritium gas was injected via TIM into the vacuum vessel for calibration of residual gas analysis, there were no plasma operations in tritium. Note that the gas inventory does not include gas used in glow discharge wall conditioning and tritium clean-up in DTE2.

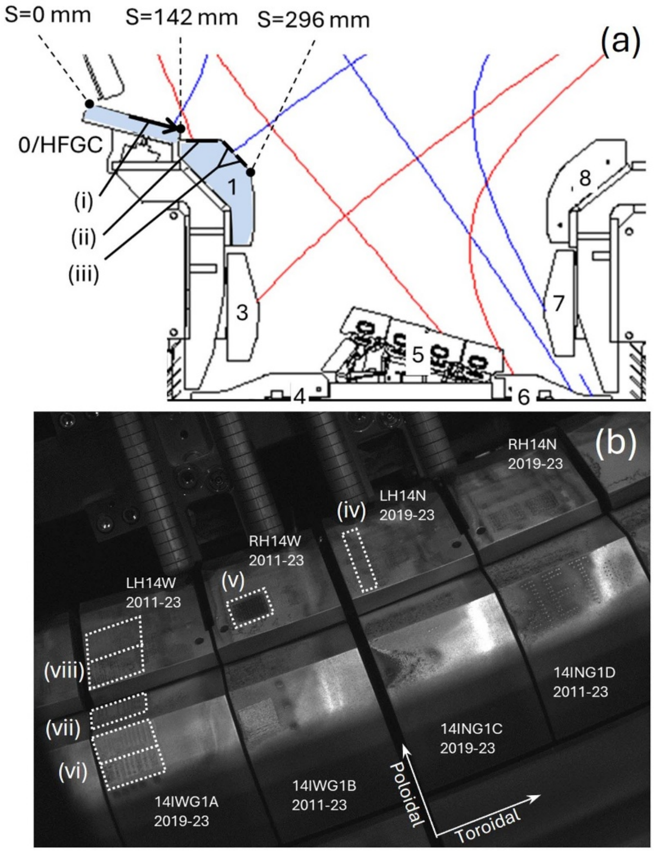

Figure 2. (a) Poloidal cross section of JET divertor showing location of High Field Gap Closure (HFGC)/Tile 0 and Tile 1 with surfaces accessible to LID-QMS diagnostic laser. 'S' is the S-coordinate scale in millimetres along the surface of the tiles, starting at the inner most corner of HFGC tile. The plasma configuration in blue is the raised inner strike point configuration (e.g. JET pulse number 104820) discussed in section 3.5. The plasma configuration in red is the 'vertical-tile 5' configuration used in the gas balance experiments discussed in section 3.2, (e.g. JET pulse number 104152). (i) Is the poloidal orientation of the LID-QMS scan also shown in (b-iv), see also figure 11. (ii) and (iii) show the poloidal orientation of the top, upper and lower LID-QMS scans on Tile 1 indicated in (b-vi, vii), see also figure 7. (b) Photograph showing location of LID-QMS measurements with position of presented data indicated; (iv) Fuel retention distribution, section 3.7, (v) Change in fuel retention measurement, section 3.1, (vi)–(viii) change in fuel retention during clean-up, sections 3.4 and 3.5. Poloidal and toroidal directions indicated.

several orders of magnitude lower than for HIs and therefore far below the sensitivity for detection by LID-QMS.

Due to the complexity of the calibration procedure, including gas injection and deconvolution of isotopologues, only qualitative data is available for presentation at the time of writing, in which AMU4 is assumed to be predominantly D_2. Of the many LID-QMS experiments that were performed throughout the D_2 operations prior to DTE3, DTE3, T clean-up and subsequent D_2 operations until the end of JET operations (see figure 1 for timeline), four experiments are presented here. The location of the LID-QMS measurement scans on the divertor tiles are shown in figure 2(*b*). The results presented are LID-QMS measurements providing: (i) change in fuel retention during DTE3 operations (section 3.1, figure 2(*b*-v)); (ii) change in fuel retention during T clean-up (section 3.4, figures 2(*b*-vi–viii)); (iii) fuel removal by heating with raised inner strike point (RISP) plasma configuration (section 3.5, also figure 2(*b*-vi)); (iv) in-vessel fuel retention and distribution on HFGC/tile0 for later comparison with *post mortem* analysis of removed tile (section 3.7, figure 2(*b*-iv)). To give some insight into the requirements for operating LID-QMS some additional details are presented for each of the experiments discussed.

Weekly measurements: throughout DTE3 periodic fuel retention measurements were made to assess changes in fuel retention during DT operations. The area shown in figure 2(*b*-v) was measured at approximately weekly intervals with the aim showing how fuel retention varied during DT operations and also to demonstrate that in-vessel T retention can be periodically assessed. The measurement area was pre-cleaned, i.e. all fuel desorbed from the surface by heating with the LID-QMS laser which was rastered in a grid formed of 25×20 points with overlapping spots. After a period of plasma operations, the area was measured using a 9×11 laser spots without overlap (see figure 2(*b*-v) and [32]), resulting in a desorption area of 7 cm^2. This gives the retention measurement for the operation interval. After the retention measurement the area was re-cleaned using the overlapping raster grid ready for the next retention measurement.

Tritium clean-up: a series of poloidal line scans were performed after the different stages of the T clean-up. There were five poloidal line scans performed after each cleaning stage; two *Upper* and *Lower HFGC* scans on tile LH14W, one *Top Tile 1* scan on 14IWG1A and two *Upper* and *Lower Tile 1* scans on the upper vertical surface of 14IWG1A. The location of the line scans are shown in figures 2(*b*-vi–viii).

In-vessel and ex-vessel retention comparison: the LID-QMS measurement points formed a 4×4 grid, see figure 11(a), resulting in a total desorption area of 0.53 cm^2. The position of the measurements in the vessel are found in figure 2(*b*-iv).

2.3. Clean-up following tritium operations

The DTE3 clean-up sequence consisted of an initial nine-day experimental phase in which different cleaning scenarios were tested with the aim of providing new data on in-vessel *local* retention on PFCs after each cleaning technique. After the initial experimental phase, diagnostic calibrations were performed in helium plasmas and finally a cycle of cleaning D plasmas were performed, during which time LID-QMS data was taken on a weekly basis. During the clean-up the T concentration was monitored using the ratio the 14 MeV neutron data and 2.4 MeV neutron data (from DT and DD nuclear reactions respectively) to provide a proxy for T concentration. 14 MeV neutron measurement and calibration is discussed in [15, 34, 35]. T concentration was monitored by hydrogen

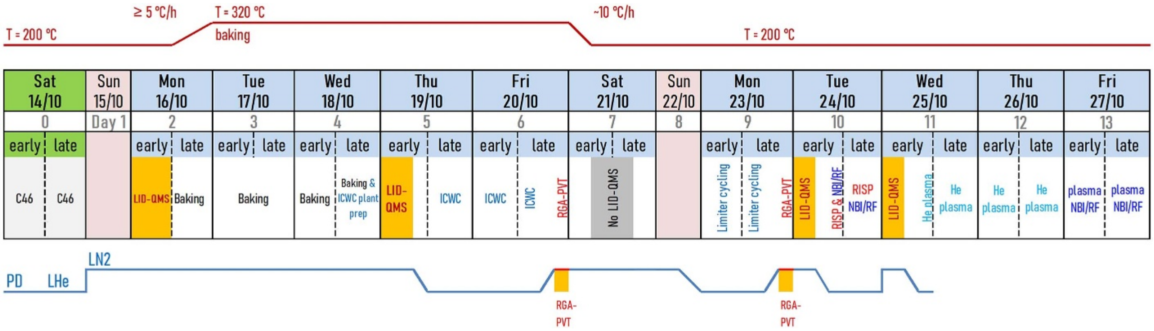

Figure 3. Timeline plan showing first two weeks of DTE3 clean-up schedule. Top red line indicates the temperature of the vessel. Bottom blue line indicates the temperature (LHe = liquid helium, LN2 = liquid nitrogen) of the pumped divertor cryo-panels (PD) and days when a gas balance was carried out in the vessel (RGA-PVT). Middle section shows the clean-up procedures with the LID-QMS measurements indicated. Note that the LID-QMS session on 19 October 2023 was not completed due to issues with the diagnostic. The cleaning procedures were baking, ion cyclotron wall cleaning (ICWC), limiter cycling plasma, raised inner strike point plasma (RISP) with neutral beam and RF heating.

Balmer-alpha optical spectroscopy at the upper vertical surface of Tile 1 inner divertor as described in [36], and optical penning gauge spectroscopy measurements of the residual gas in the exhaust as described in [33, 37]. The limit of detection for residual gas tritium analysis is in the region 1%–2% [37, 38] whilst neutron detection is suitable for T concentration measurement of the order 0.01% [15]. In addition, QMS, gas chromatography [39] and β-emission detection [40], based in the AGHS facility were used for analysing the volume of tritium in the hydrogenic exhaust gas. Towards the end of the clean-up when data indicated that the T concentration was approaching 0.02%, a number of JET experiments in D plasmas were interleaved with the cleaning schedule. Once the target T concentration was reached JET operations continued with experiment in D plasmas. During this time the T concentration was monitored using the neutron diagnostics.

The first two weeks of the clean-up schedule following DTE3 are shown in figure 3. The cleaning started with baking at 320 °C, overlaid with ICWC at 320 °C [5], followed by limiter cycling plasmas and RISP configuration plasmas [5, 7] in deuterium at standard wall temperature for JET operations (200 °C). LID-QMS measurements were performed between different cleaning techniques. LID-QMS data was collected (i) prior to full baking at 320 °C, (ii) after two full days of baking but prior to ICWC, (iii) after ICWC, cooling to 200 °C and subsequent limiter plasmas (i.e. prior to RISP plasmas), and (iv) after RISP plasmas.

The RISP configuration shown in figure 2(*a*) is 2.5 T, 1.7 MA plasma operated with neutral beam (NB) and radio-frequency (RF) heating. In this configuration the inner strike point (ISP) is raised to the top of the vertical surface on Tile 1 and 15 MW of heating (12 MW NB and 3 MW RF) is applied to heat the surface to remove fuel. During the RISP clean-up sequence fourteen discharges with heating were performed. The RISP phase of the discharge is 7.5–25 s with an inner strike point sweep of 2 s to measure ion fluxes at the nearest Langmuir probes, figure 8(*b*). As the ISP sweeps over the Langmuir probe the electron temperature is in the range 10–20 eV and electron density up to 1.0×10^{20} m^{-3}. The total time in the RISP configuration was 262.5 s. The time at full power with 12 MW NB and 3 MW RF heating was 107 s. The ion flux during heating is in the range 6–7 $\times$ 10^{22} $\mathrm{m}^{-3}\,\mathrm{s}^{-1}$. For individual pulses the full power heating lasted for 10 s followed by 3 MW RF heating only at the end of the RISP configuration. Most tritium was released during the 15 MW NB + RF heating phase, when 14 MeV neutron rates were of the order 10^{15} n s^{-1}. For RF heating only neutron rates were significantly lower, of the order 10^{13} n s^{-1}. The results of the RISP clean-up are discussed in section 3.6.

Following the initial clean-up described above, the tritium cleaning sequence continued for a further four weeks with a daily cycle of D plasmas aimed at heating the divertor surfaces in different locations to remove tritium retained in divertor surfaces by co-deposition and implantation and to enable isotopic exchange with tritium. The cycle was generally one ohmic plasma with limiter cycling phase—heating limiters, followed by a cycle of three pulses with inner and outer strike points (ISP/OSP) on the inner and outer divertor corner pump ducts, tile 4 and tile 6 respectively, (*corner-corner 'CC' configuration*), three pulses with ISP and OSP on the inner and outer vertical target tiles 3 and 7 respectively (*vertical-vertical 'VV' configuration*), three pulses with ISP on vertical target tile 3 and OSP on tile 5 in the divertor base (*vertical-5 'V5' configuration*) and three pulses in the RISP configuration, tile 1 and tile 6. For each plasma the aim was for 10 s of combined NB and RF heating up to 15 MW to desorb fuel on divertor surface by heating to >800 °C and operation in deuterium to enable isotopic exchange with tritium.

3. Results

3.1. Change in fuel retention during DTE3 operations

This section reports the LID-QMS data providing weekly in-vessel *local* fuel retention measurements. The first qualitative results are shown in figure 4 comparing relative changes in fuel retention on a weekly basis. The results demonstrate sufficient sensitivity in the system to measure changes in fuel retention on a weekly basis.

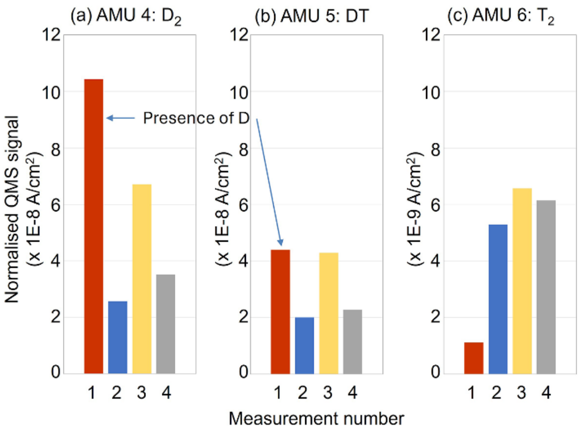

Figure 4. LID-QMS data for weekly measurements of area (*v*). Four measurements were taken, represented by different coloured bars, on the dates shown in table 2. The relative amounts of released gases for each atomic mass unit are shown in (*a*) AMU4, (*b*) AMU5, (*c*) AMU6. Note the order of magnitude difference in scale between plots.

Table 2. Gas injection in the interval between LID-QMS measurements at area (v), see figure 2. The '(number-colour)' in column 1 provides the reference to equivalent periods of time for LID-QMS data presented in figure 4.

		Injected fuel	
Measurement date dd/mm/yy	Divertor plasma time (min)	D atoms ($\times 10^{24}$)	T atoms ($\times 10^{24}$)
19/9/23 (1- red)	10.9	5.96	3.52
4/10/23 (2-blue)	21.7	8.13	5.67
11/10/23 (3-yellow)	4.1	4.40	2.97
16/10/23 (4-grey)	8.9	3.01	2.45

The gas injection between the weekly LID-QMS measurements is summarised in table 2. In general, the fuelling of the deuterium from GIMs & NBIs and tritium from TIMS during DTE3 was in the ratio 1:0.69 ± 0.09 (D:T). Note that deuterium gas injection includes deuterium from NB injection, as these were operated in deuterium in DTE3. Details of GIM, TIM and NB injection are summarised in table 1 and discussed in section 2.1. The AMU4 (D_2) and AMU5 (DT) QMS signals shown in figure 4 are of the same order of magnitude which indicates an approximate ratio 1:0.33 D:T atoms released. This ratio indicates less tritium than the injected gas ratio. Possible reasons for this are discussed in 4.1.

Assuming that in-vessel fuel retention is of the order of 2% [2], and the distribution of fuel retention on plasma facing surfaces is in accordance with *post-mortem* analysis [22], the calculated areal specific fuel retention is in the order 3.6–9.8 $\times$ 10^{16} D atoms cm^{-2} and 3.0–6.9 $\times$ 10^{16} T atoms cm^{-2} for the fuelling periods in table 2. This shows that LID-QMS can provide variations in HI fuel retention in JET at the level of 10^{17} atoms desorbed from the raster area 7 cm^2. In addition, the measurement of HIs at the level of 10^{17} atoms is consistent with the lower detection limit established with the calibration gas injections discussed in section 2.2.2 [32].

3.2. Global fuel retention measurements from gas balance

Altogether one rehearsal and four attempts at the full gas balance routine—in-vessel RGA-PVT, and AGHS-PVT—were attempted as described in section 2.2.1. However, despite the extensive planning neither the trial run nor the full experimental measurement cycle were achieved and only sections of the experimental results are available for presentation and discussion. In fact, neither of the *global* AGHS-PVT gas balance measurements in DTE3 operations were successful. Some insight into the difficulties of completing gas balance in DT operations are presented in the discussion section 4.2.

However gas balance experiments in D plasmas and 100% T operations were achieved. Figure 5 shows the qualitative results from RGA-PVT measurements; two measurements in deuterium, one measurement in T operations prior to DTE2 (see figure 1) and one measurement in DTE3. One of the measurements in D operations was preceded by deuterated ethane (C_2D_6) injection during an experiment a few days before, shown in black in figure 5, which increased the carbon content accessible for erosion and material migration. The results clearly show that the gas balance measurements, and therefore *global* fuel retention, are dominated by the presence of carbon when compared with the measurements in different HI mixtures in an all-metal wall environment. This case also demonstrates the ability of in-vessel RGA-PVT to detect such differences in retention. In terms of the gas balance of the various HI experiments, no significant differences are observed. This is indicative of no significant isotopic dependence of in-vessel *global* fuel retention.

3.3. Fuel retention and outgassing from vessel wall

Figure 6 shows the mass spectrometer data sampling from the main chamber was used to monitor the outgassing from

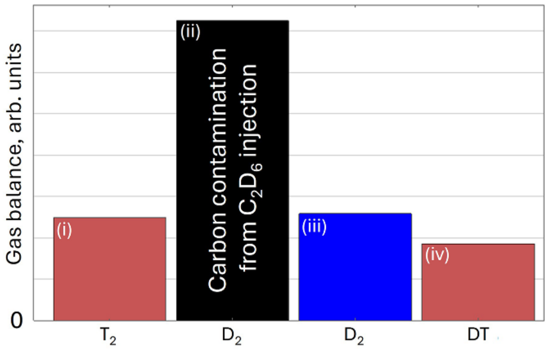

Figure 5. Relative RGA-PVT measurements showing the difference between all injected hydrogen isotopes and retained HIs. The results show no significant difference in the total gas balance with D_2, DT, T_2 plasma oeprations when compared with gas balance in the presence of carbon. Note that the position of zero on the vertical axis is only indicative and cannot be defined without further data analysis. Experimental data from (i) T_2 plasmas on 02/03/2022, (ii) D_2 plasmas on 05/08/2022, (iii) D_2 plasmas on 28/07/2023 and (iv) DT plasma during DTE3 on 06/10/2023.

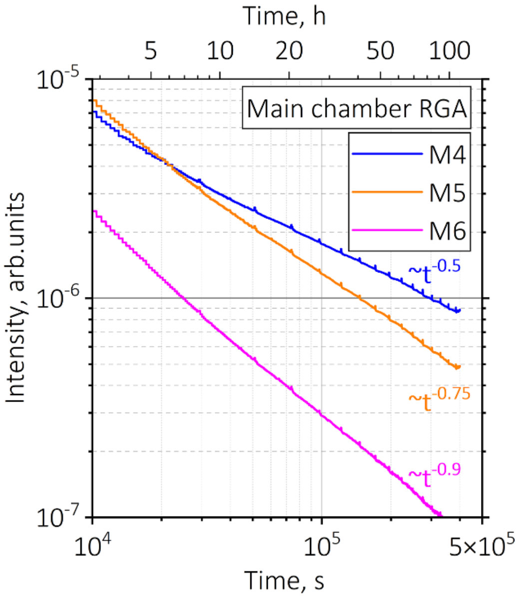

Figure 6. Example of outgassing in JET following DT plasma for AMU4, AMU5 and AMU6 with vessel at 200 °C. From JET pulse number 99530 during DTE2.

the walls after plasma operations. The long-term outgassing is shown to be dependent on the time after plasma operations $\propto t^{-\alpha}$ where $\alpha = 0.5$ (AMU4), 0.8 (AMU5), 0.9 (AMU6). This is similar to results for JET-C operated with D plasma where $\alpha = 0.75 \pm 0.1$ [11]. These trends indicate that there is a HI effect associated with the long term in-vessel outgassing whereby there is a faster decay of the outgassing rate with higher HI molecular mass. This difference is likely to be due to a combination of the difference in mass, relative concentration and depth profile of HI in the surface of PFCs. For example, JET was operated since 2011 with predominantly deuterium plasmas, whilst tritium was only introduced in two short campaigns in 2020–2022 (100% T_2 and DTE2) and 2023 (DTE3), see figure 1, therefore there is a deep reservoir of deuterium in the bulk of the PFC, whilst tritium is located in the surface region which will affect the decay in outgassing rate.

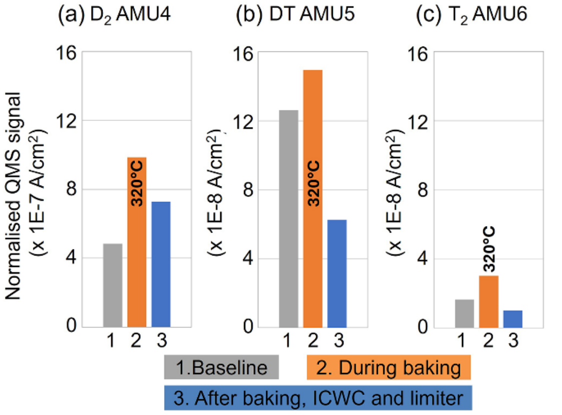

Figure 7. Relative change in fuel retention at the top surface of 14IWG1A Tile 1 in the location of Be dominated co-deposits. (1) Prior to the start of clean-up with vessel temperature at 200 °C (grey), (2) during main chamber baking at 320 °C (orange), (3) after baking, ICWC and limiter cycling plasmas with vessel temperature at 200 °C (blue). (*a*) D_2 AMU4, (*b*) DT AMU5, (c) T_2 AMU6. Location of measurements indicated in figure 2 (*a*-ii), (*b*-vi). Note order of magnitude difference in scale between plots.

3.4. Change in fuel retention during tritium clean-up following DTE3

LID-QMS measurements were interspersed between different phases of the clean-up as described in section 2.3. The change in surface fuel concentration of D_2 (AMU4), DT (AMU5) and T_2 (AMU6) on 14IWG1A (1) prior to the start of clean-up 'baseline' (grey), (2) during baking at 320 °C (orange) and (3) after baking, ICWC and limiter plasmas at 200 °C (blue) are shown in figure 7. The results demonstrate that the concentration of fuel on the surface of the tile changes with temperature and are indicative of diffusion to the surface giving rise to higher rates of desorption, discussed in section 4.3.

3.5. Fuel removal by heating at strike point

The RISP configuration with D plasma was used during the initial clean-up phase after DTE3 in conjunction with LID-QMS to make a direct measurement of the reduction in fuel at the tile surface at the strike point location. Table 3 shows the first in-vessel measurement demonstrating the change in fuel retention by surface heating at the location of the RISP by direct measurement with LID-QMS. The maximum temperature reached on the tile surface in the region of the RISP is ~900 °C, see figure 8. The difference in the maximum temperature for the first and last RISP plasma is ~20 °C, demonstrating good reproducibility of the discharges. The scan area available for each LID-QMS measurement was limited in order to fit all the LID-QMS scans planned for the clean-up experiments close together on the tile surface, as a result AMU5 and AMU6 are not presented due to the area of the LID-QMS raster being insufficient to release enough DT and T_2 molecules to

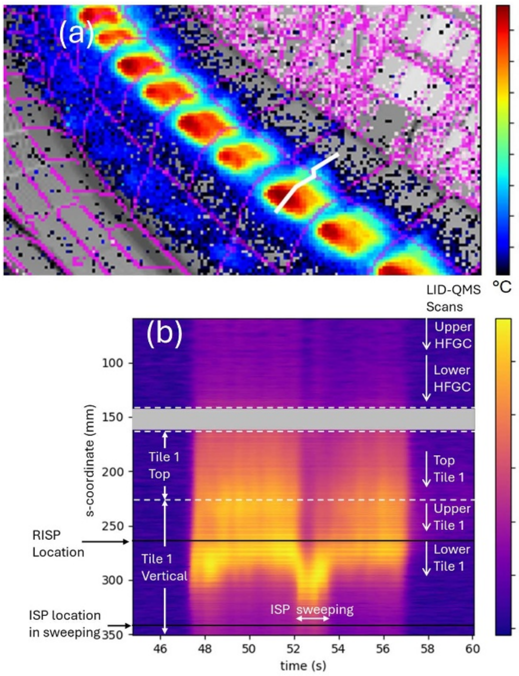

Figure 8. Example from JET pulse number 104 820 (*a*) Infra-red camera data of RISP plasma at 11 s. The white line indicates the poloidal extent of the temporal IR data. (*b*)Temporal infrared camera data on Tile 1 showing the location of the LID-QMS scans and the RISP location. The RISP is from 7 to 17 s with an inner strike point sweep between 12 and 14 s to measure ion flux at the nearest Langmuir probe. The positions of the LID-QMS scans and plasma configuration are shown in figure 2.

Table 3. Reduction in deuterium detected as AMU4 (D_2) in the surface of HFGC and Tile 1 measured before and after raised inner strike point plasmas using LID-QMS. Details of the LID-QMS scan locations are shown in figures 2(b-vi– viii) and 8.

Scan name	% deuterium remaining
Upper HFGC	88
Lower HFGC	48
Top Tile 1	61
Upper Tile 1	21
Lower tile 1	51

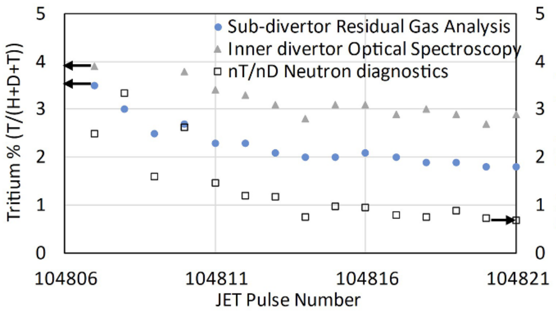

Figure 9. Tritium content (%) in plasmas during RISP pulse sequence measured using optical spectroscopy diagnostics from the inner divertor Balmer-alpha line optical spectroscopy (grey triangle), sampling of residual gases in the sub-divertor region (blue circles) and DT 14 MeV (n_T) and DD 2.5 MeV (n_D) neutrons (open squares).

meet the lower sensitivity of the detection system, in the order of 10^{17} atoms [32]. Whilst tritium could not be detected, the use of LID-QMS to measure the change in deuterium retention still provides insight into fuel removal in general. The results show the greatest fuel reduction in the region of the *Upper Tile 1* scan, the location of the RISP, with 79% of deuterium measured as AMU4 (D_2) removed after fourteen plasmas with 107 s in RISP configuration at the maximum 15 MW heating, relative to the LID-QMS measurement prior to the RISP plasmas. There was also a reduction in fuel in the areas surrounding the RISP location, see table 3, but not as pronounced as in the RISP region. The lowest amount of fuel removed is far into the SOL in the *Upper HFGC* region where two effects play a role; firstly the surface temperature is lower in this region than for the RISP region and therefore removal is expected to be lower, and secondly the surface is in the far SOL and therefore co-deposition will occur due to primarily Be erosion and migration, adding to retention based on the HI content of the plasma. However these results indicate that for the RISP configuration with D plasma fuel removal dominates even in the SOL.

Figure 9 shows a reduction in T concentration during the cleaning using RISP configuration measured by optical spectroscopy at the Tile 1 surface just above the ISP in the private flux region, from the sub-divertor residual gas analysis and the plasma from neutron data which is consistent with the LID-QMS measurements. Deuterium removal at the heated tile surface measured by LID-QMS can be expected to be indicative of tritium removal from the surface layer, both as a result of desorption due to strike point heating and via isotopic exchange. It therefore demonstrates that strike point heating removes fuel including tritium from surfaces and are therefore useful as a clean-up method.

3.6. Overall tritium clean-up

After the initial stages of clean-up as shown in figure 3, cleaning plasmas, described in section 2.3, were continued for four weeks. The cycling of different plasma configurations heated different regions of the divertor to desorb fuel, as demonstrated by the LID-QMS measurements of RISP plasmas, and also allowed for isotopic exchange. The tritium content during the plasma clean-up phase is shown in figure 10. The target of reaching 0.02% was achieved after around four weeks of operation, ∼630 plasma pulses, 2.9 h of diverted plasma.

3.7. In-vessel and ex-vessel local fuel retention

In the last measurements in JET after the end of plasma operations and prior to sample retrieval, LID-QMS was used to measure the relative change in fuel retention poloidally across a HFGC tile and compared with the predicted fuel profile based on a tile removed in 2017 during earlier deuterium campaigns.

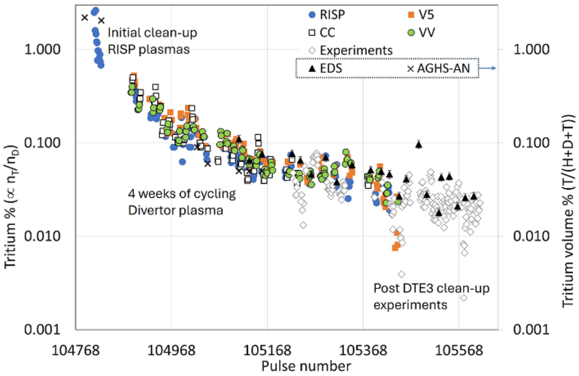

Figure 10. Tritium concentration measurements during DT clean-up. For D_2 clean-up plasmas and subsequent D_2 plasma 'Experiments' the T concentration is given by the fraction of DT neutrons to DD neutrons measured by neutron diagnostics [35]. The clean-up plasmas are corner-corner 'CC', vertical-tile 5 'V5', raised inner strike point 'RISP' and vertical-vertical 'VV', described in section 2.3. 'Experiments' plasmas have varied configurations. 'AGHS-AN' and 'EDS' are from two analysis systems located in AGHS used to measure the concentration (volume) of tritium in the exhaust gases.

The deuterium retention distribution is well understood from tiles removed from JET the results of which have been previously presented in [22]. The general trend in the distribution for AMU4 (D_2), AMU5 (DT) and AMU6 (T_2) retention (figure 11(*b*)) is similar to that demonstrated for previously removed tiles with higher retention far into the inner scrape off layer (SOL) (s = 70–110 mm) and generally lower retention from s = 140 to 160 mm which experiences surface heating during plasma operations. The results are the first indication the LID-QMS can provide useful in-vessel analysis of relative variations in fuel retention.

4. Discussion

4.1. Fuel retention in DT operations

Over the period of the all metal JET-ILW wall fuel retention measurements have been an ongoing research programme in support of ITER. A good understanding of *global* fuel retention in different plasma scenarios has been demonstrated through gas balance measurements with D plasmas in the early operations of the JET-ILW wall [2] providing an average *global* fuel retention of #2% of injected fuel. Similarly ex-vessel *post-mortem* analysis of JET-ILW wall components has provided a lower retention on venting to air at #0.2% [22]. During DTE3 operations there was a significant experimental programme to integrate a newly installed LID-QMS diagnostic to provide the first in-vessel *local* retention measurements. The qualitative measurement of LID-QMS presented here have shown that variations in fuel retention can be measured for D_2, DT and T_2 molecules, as shown in figure 4. The relative desorption of DT is highly dependent on the presence of the deuterium and tritium on surface. This is also evident in the baking phase of the T clean-up discussed in section 4.3.

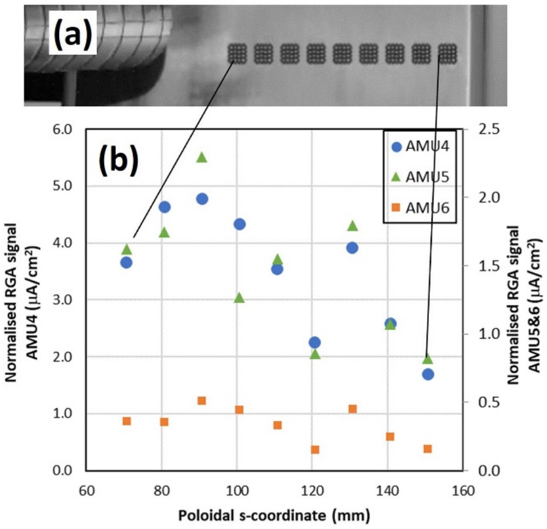

Figure 11. (*a*) Schematic showing poloidal position of 4 × 4 LID-QMS measurements. The location of the measurements are shown in figure 2(*b*-iv). (*b*) Normalised AMU4 (D_2), AMU5 (DT) and AMU6 (T_2) QMS signal indicating qualitative fuel retention distribution in poloidal direction. S-coordinate is the distance along the surface of the tile starting at 0 mm on top left corner of HFGC tile, see figure 2(*a*).

The method for determining the ex-vessel *post-mortem* fuel retention value assumes that over the complete operating period, typically 20 h of plasma, an average retention rate proportional to the injected fuel can be established. However, the weekly measurements of fuel retention on the HFGC tile compared to the gas injection does not tend to follow this assumption on the shorter (weekly) time scale. This is commensurate with the different plasma configurations with variation in heating which give rise to different erosion, migration and co-deposition as has been demonstrated in [14, 41]. The D:T ratio of the gas injection during the measurement period was 1:0.69 ± 0.09. As a first approximation the LID-QMS signals indicate that deuterium and T atoms are desorbed in a ratio 1:0.33. The difference is likely to be related to an additional source of deuterium diffusing from the bulk of the tile after the initial laser cleaning as well as the particular plasma configurations and heating. Given that AMU4 (D_2), AMU5 (DT) and AMU6 (T_2) are clearly measured on an approximately weekly basis from the 7 cm^2 desorption area demonstrates that they are above the 10^{17} atoms detection limit of the system which indicates that in-vessel fuel retention is indeed of the order of a few percent and are therefore higher than ex-vessel retention measurements.

4.2. Gas balance in DT operations and lessons learned

It was recognised that the JET DT gas balance experiments would require meticulous planning for the DT plasma operations and gas collection, therefore the experiment was preceded by trial runs using deuterium with the aim of increasing the likelihood of success. In addition, the plan to combine LID-QMS measurements, with in-vessel RGA-PVT and AGHS-PVT added complexity. Despite the planning there were several different issues faced at various stages that prevented the *AGHS-PVT* rehearsal and the full gas balance cycle—RGA-PVT, AGHS-PVT and LID-QMS measurements—from being completed.

Plant related problems were encountered during AGHS gas collection whereby nitrogen from a higher pressure line was passed through a valve into the lower pressure collection lines within a gas route that was infrequently used. This was the case in the *AGHS-PVT rehearsal* and the *first D gas balance* experiment, using two different gas routes. Efforts to calibrate the leak through the valves concerned into the gas collection volume resulted in unacceptably high uncertainty in the quantification of the exhaust gas coming from the gas balance experiment. Given the experimental schedule there was insufficient time available to perform maintenance in AGHS to resolve these issues in the gas lines.

The gas balance experiments rely on maximizing the gas pressure measurements on available pressure gauges. In the *first DT gas balance* experiment in DTE3 it was found that the pressure in the vessel was rising too high in comparison to plasma pulses in DTE2. The additional gas pressure was found to be helium (deduced by analysing the trends of AMU3 and AMU4 QMS measurements in the main chamber), which is not pumped efficiently by the cryo-panel pumps. The two possible sources of helium were (i) ^{4}He remaining in the vessel from helium plasmas used for diagnostic calibrations several days earlier, and (ii) increase in ^{3}He concentration in the tritium supply due to tritium decay. The additional pressure reduced the amount of fuel that could be injected during the experiment, therefore compromising the accuracy of the gas balance. Further pumping of the vessel over several weeks and additional cleaning to increase the purity of the tritium supply was completed prior to the second attempt.

In the *second DT gas balance* experiment in DTE3 the ^{3}He and ^{4}He were found to be lower but they were still not as low as seen in the sample plasma configuration in DTE2. Despite the higher than expected background pressure the experiment proceeded and the in-vessel RGA-PVT was completed, but finally in was not possible to complete the gas collection at the end of the experiment because the tritium procedures needed for safe operation in AGHS were not completed in time. This demonstrates the complexity of procedures and approvals needed to meet safety requirements to bring plant online for tritium handling. Even after performing the trial runs using deuterium, additional procedures were needed for tritium operation. Therefore, it is essential to do a full trial run using tritium, as this exercises not only the plant but also the safety requirements needed.

There was in the end a final gas balance measurement in the D operations following DTE3 and clean-up, *second D gas balance*. Gas balance experiments were carried out on two consecutive days with LID-QMS, RGA-PVT and AGHS-PVT. However, on the first day there was a plasma disruption. Although the Disruption Mitigation Valve was not triggered, i.e. no additional gas was injected, the high heat loads may have affected the final evaluation of the retention for the particular plasma scenario. This demonstrates the need for careful choice of plasmas and heating during gas balance measurements to avoid disruptions which may results in additional gas injections from disruption mitigation valves or fuel removal through heating of surfaces.

Overall the issues faced during RGA-PVT and AGHS-PVT indicate the need for a dedicated PVT system if it is to provide tritium inventory data. Such a system will require careful design of a defined volume(s), pressure gauges and temperature measurement as well as consideration of the local environment, such as temperature control.

Despite the many issues faced with AGHS-PVT the results from gas balance making use of RGA-PVT show qualitatively that there is no significant HI effect with *global* retention and clearly reaffirms the need to keep carbon contamination to a minimum to keep fuel retention low.

4.3. Tritium clean-up

The clean-up cycle for DTE3 was based on results from DTE2 [5]. It consisted of main chamber vessel baking at 320 °C, ICWC at 320 °C, limiter cycling plasmas, RISP configuration and clean-up plasmas with the integration of LID-QMS, as shown in figure 3, and discussed in section 2.3. Previous experiments in the initial DTE2 clean-up have shown that initial baking up to 320 °C accounted for up to 58% of tritium removed in the initial phase. Of the remaining tritium ICWC with baking removed 17%, GDC 26% and RISP plasmas 3% [5]. The incorporation of LID-QMS measurements between cleaning phases in the DTE3 clean-up provided the possibility to investigate whether these cleaning methods removed fuel from the upper inner divertor where high fuel concentration is expected in deposits. For the main chamber baking to 320 °C, although the divertor is not actively heated, thermocouples embedded in an equivalent tile to the tile measured by LID-QMS showed elevated temperature from 75 °C to 145 °C, with the surface temperature expected to be higher therefore desorption of fuel from the tile is expected. For the ICWC and limiter plasma cleaning the fuel removal is expected to be mainly from the main chamber, with minimal interaction at the top of the inner divertor, i.e. HFCG tile and horizontal surface on Tile 1 (see figure 2(*a*)). During the vessel baking LID-QMS demonstrated the relative change in surface fuel retention at different temperatures, shown in figure 7. The results clearly show increased fuel concentration in the surface layer at the main chamber baking temperature 320 °C for AMU4 (D_2), AMU5 (DT) and AMU6 (T_2). This is indicative of fuel diffusion to the surface and outgassing to the vacuum vessel as might be expected. After baking the AMU4 (D_2) signal remains higher than the baseline measurement, whereas AMU5 (DT) is lower and AMU6 (T_2) is below the level of detection. This suggests that the concentration of T atoms at the surface has been significantly reduced whereas that of D atoms remains relatively high. This effect is shown schematically in figure 12. In the baseline case in figure 12(*a*) AMU4 (D_2), AMU5 (DT) and AMU6 (T_2) are all formed (released) when the surface of the tile is heated by the LID-QMS laser. Whilst the main chamber of the vessel is baked at 320 °C HI

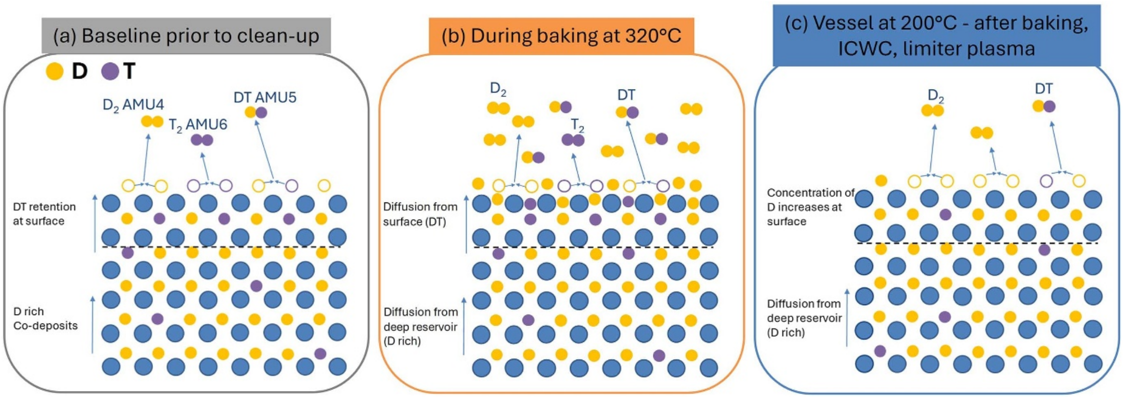

Figure 12. Schematic showing change in concentration of fuel at different points during clean-up. Yellow dot = deuterium (D) atom, purple dot = tritium (T) atom, blue dot = matrix material. (*a*) Baseline prior to start of clean-up with vessel temperature at 200 °C (grey), (*b*) during main chamber baking at 320 °C (orange), after baking, ICWC and limiter cycling plasmas with vessel temperature at 200 °C (blue).

molecules leave the surface and diffusion from the pre-existing source of co-deposited deuterium—from earlier D plasma experiments—fills the surface region, figure 12(*b*). In contrast the concentration of tritium at the surface is reduced and not replenished. Following main chamber baking at 320 °C, ICWC, vessel cooling to 200 °C and limiter plasmas the T concentration is reduced and therefore DT and T_2 molecules when desorbed as by LID-QMS heating. This is shown schematically in figure 12(*c*). LID-QMS has shown a clear demonstration of increased HI concentration at the PFC surface at elevated temperature and reduction in T concentration after baking, ICWC and limiter plasmas. One aspect that will be different in future operating machines is that the reservoir of fuel will consist of deuterium and tritium in quantities approximately equal to the fuelling ratio, whereas in JET the predominant reservoir is deuterium from preceding operations.

Whilst technical issues prevented a planned LID-QMS measurement after ICWC and cooling the vessel to 200 °C, these results indicate that the planned vessel cooling to 200 °C after ICWC would have had a significant impact on the results as the measurement prior to ICWC was taken at 320 °C. In fact to provide in-vessel *local* retention data to verify the findings in [5, 7]—whereby the efficacy of baking decreases over time and subsequent efficacy of introducing additional cleaning methods such as ICWC or GDC are studied—LID-QMS measurements should have been planned over several days at 320 °C to measure the change in fuel concentration on the tile surface, and then again after ICWC at 320 °C, and finally after cooling to 200 °C. However, the scope for planning was limited by programmatic demands and clean-up schedules to be delivered prior to the end of JET plasma operations so this was not achieved.

LID-QMS has provided in-vessel evidence for fuel removal in tile surfaces by heating with the inner strike point. By making use of the RISP configuration overall 79% of deuterium fuel, was removed at the strike point location. This demonstrates good efficacy of fuel removal at the strike point by surface heating, which was successfully used in the 4 week cycle of divertor plasmas carried out to complete the T clean-up whereby a combination of plasma configurations were cycled to heat surfaces all around the divertor. RISP is still considered as a possible option for tritium removal in ITER. Analysis of the detritiation of Be deposits in ITER showed that the surface temperature achieved during RISP configurations depends on the contact resistance between Be co-deposits and the bulk W divertor target [42]. It has been shown that with assumptions for contact resistance similar for co-deposits at the upper divertor of JET surface temperatures of Be co-deposits can rise up to 1050 °C which is efficient for fuel removal [7, 42, 43]. The decision for ITER to move to all-W wall with boronisation, means that boron deposits will form similarly to Be deposits. Modelling of trapping and diffusion in the B co-deposits with 10 s plasma to achieve 1000 °C surface temperature indicates that RISP could still be a potential method for tritium removal [44, 45].

One aspect of the RISP and other strike point heating scenarios is whether there is re-deposition of removed fuel onto other locations. This issue has been raised for the use of RISP configurations in ITER where one possible solution is to reduce tritium using isotopic exchange with ICWC prior to running RISPs [44]. The LID-QMS measurements before and after RISP configuration with deuterium show a reduction in deuterium retention on Tile 1 closed to the ISP location and also on HFGC/Tile 0 which is in the SOL retention, the region where most co-deposition occurs in JET. Whilst this does not rule out co-deposition further into the SOL, it does demonstrate that heating of surfaces is effective at removing fuel. In addition, the use of a range of clean-up D plasma configurations clearly reduces T retention as shown in figure 10, and therefore any re-deposition of removed tritium would be expected to be minimal whilst deuterium retention by co-deposition and implantation from D plasma can still occur.

4.4. Fuel retention measurements and lessons learned with LID-QMS

The overlap in the molecular masses of the isotopologues of hydrogen molecules points to the need for high resolution diagnostics in each mass range dedicated for the composition analysis of desorbed gases. This will be particularly needed for separating helium from deuterium molecule, both nominally AMU4, in DT operations.

The implementation of defined conditions for operating LID-QMS is also desirable. Operating LID-QMS as part of experiments meant that optimal steady state conditions could not always be met in JET, including vessel temperature, base pressure and pumping conditions. However, if used for inventory assessment, the need to meet regulatory requirements will likely necessitate defining LID-QMS operating requirements.

LID-QMS was deployed for a relatively short period of time during the final 6 months of JET plasma operations, ending in December 2023. During this time the area available for measurements was heavily utilised. Therefore, in future machines the size of the target area needs to be considered as this may impact the optical design which may necessitate a greater degree of steering of the laser beam to access a larger area or several laser beam ports. Repeat measurements will be possible in the same target area, however the minimum build of fuel and sensitivity of the detection system will dictate how often areas may be revisited. Such calculations were done for JET [22]. Another aspect of fuel retention measurements using LID-QMS is understanding how the *local* in-vessel retention measurement relates to the *global* in-vessel retention. In JET the scaling from *local* to *global* fuel retention is based on extensive *post-mortem* analysis data giving the relative fuel retention of PFCs around the vessel [21, 22]. Whilst first wall sampling is foreseen in ITER, *post-mortem* analysis, will be limited and for future power plants will not be possible, therefore modelling may be needed to reconstruct the overall retention distribution indirectly with results for *post-mortem* analysis being added to the picture when available. To directly obtain an almost complete retention distribution, the laser needs to have access to the full poloidal cross section of the wall in a representative area. With the assumption of toroidal symmetry, which is largely valid for tokamaks, a complete fuel map and a total retention value for the torus can be measured directly. To allow this laser access either the use of several ports will be necessary or preferentially a rotatable first mirror that can be moved inside the plasma volume and retracted back behind the first wall. This allows to direct the laser beam to any poloidal location even on the low field (outer wall) side, which is otherwise difficult to access.

In this paper four experiments using LID-QMS have been highlighted as the demonstration of the new information that can be obtained from in-vessel *local* fuel retention measurements. These relate to the weekly LID-QMS measurements during DTE3, LID-QMS in conjunction with gas balance and monitoring of fuel removal during clean-up, including before and after RISP plasmas. One lesson learned during the baking measurements is that the temperature of the vessel has an impact on fuel concentration in tile surfaces and therefore needs to be controlled in order to make comparative tritium measurements.

One of the key findings from operating the LID-QMS diagnostic is that the QMS response is sensitive to variations in pumping conditions, therefore measurement of the pumping conditions is needed in order to quantify the fuel release by LID-QMS. This points to the need for dedicated D_2 and T_2 gas injections for evaluation of pumping speeds in future machines.

5. Conclusions

JET has had a long history of providing fuel retention data. The latest results of the fuel retention programme in T_2, DT and T clean-up operations prior to the end of JET operations have provided new insight into in-vessel fuel retention, also the design and procedural requirements for fuel inventory accountancy which will be needed for future machines and power plants.

The implementation of LID-QMS in JET has enabled new in-vessel *local* fuel retention measurements at the upper tiles of inner divertor. In particular new data has demonstrated the capability for monitoring fuel retention in-vessel, including tritium, and the efficacy of baking and strike point heating for the removal of fuel. This demonstrates the potential of LID-QMS as a tool for in-vessel fuel retention assessment. In particular the RISP configuration with 15 MW heating was shown to remove 79% of deuterium fuel on the surface of a divertor tile, making the strike point heating an effective way of local fuel removal.

No significant change in the *global* fuel retention was observed using gas balance when comparing D, T and DT operations. Notably, the presence of carbon coming from prior ethane injection clearly reinforces the impact of carbon contamination on retention. Long term outgassing data show the dependence on HI with outgassing of T_2 decaying faster than D_2.

Gas balance measurements in T operations have demonstrated additional complexity in terms of operating procedures. This highlights the need for defined design requirements and operating procedures if gas balance is to be included in the suite of diagnostics to monitor fuel inventory.

PFCs will be retrieved from JET and ex-vessel *post-mortem* analysis measurements will be used to compare with in-vessel fuel retention measurements, check the surface condition after LID-QMS and assist in validating LID-QMS data. The data coming from *post-mortem* analysis will complete the full range of retention measurements from T operations.

The pumping conditions during LID-QMS measurements influence the detection of desorbed gases. The variation in pumping conditions point to the need for gas injection of all relevant gases in relevant quantities whenever LID-QMS is performed. The use of the plasma injection systems are unlikely to meet these requirements as they will be designed for larger gas injection, therefore a dedicated LID-QMS gas injection system is needed.

In ITER tritium accountancy will be needed for security, safety and control of plasma operations and the strategy follows a similar approach to that deployed at JET. In the main plant continuous measurement of T concentration and mass flow is foreseen to monitor the exchange of tritium between the tritium storage and deliver system (SDS), fuelling system and tokamak exhaust processing (TEP) system. PVT and concentration measurements are planned in TEP to analyse gas from cryopump regeneration in dedicated volumes, as well as in SDS. In addition, there will be tritium monitoring in the vacuum vessel using a suite of tools;

including first wall samples, Laser Induced Desorption Spectrometer QMS based diagnostic as for JET, dust monitoring, residual gas analysis, divertor pressure gauges and neutron detectors. Therefore, the lessons learned from JET DT fuel retention measurements presented here provide relevant data for the design and strategy of ITER tritium accountancy.

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