

PAPER • OPEN ACCESS

Evaluation of tritium retention in plasma facing components during JET tritium operations

To cite this article: Anna Widdowson *et al* 2021 *Phys. Scr.* **96** 124075

View the [article online](#) for updates and enhancements.

You may also like

- [Fuel retention studies with the ITER-Like Wall in JET](#)
S. Brezinsek, T. Loarer, V. Philipps et al.
- [Gas balance and fuel retention in fusion devices](#)
T. Loarer, C. Brosset, J. Bucalossi et al.
- [Long-term fuel retention in JET ITER-like wall](#)
K Heinola, A Widdowson, J Likonen et al.



PAPER

Evaluation of tritium retention in plasma facing components during JET tritium operations

OPEN ACCESS

RECEIVED

22 June 2021

REVISED

30 September 2021

ACCEPTED FOR PUBLICATION

18 November 2021

PUBLISHED

15 December 2021

Original content from this work may be used under the terms of the [Creative Commons Attribution 4.0 licence](#).

Any further distribution of this work must maintain attribution to the author(s) and the title of the work, journal citation and DOI.



Anna Widdowson¹ , J Paul Coad¹ , Yevhen Zayachuk¹, Ionut Jecu² , Eduardo Alves³ , Norberto Catarino³ , Victoria Corregidor³, Matej Mayer⁴ , Stepan Krat^{4,5} , Jari Likonen⁶, Kenichiro Mizohata⁷, Chris Rowley¹ , Mirosław Zlobinski⁸ , Marek Rubel⁹ , David Douai¹⁰, Kalle Heinola¹¹ , Tom Wauters¹² , Laura Dittrich⁹ , Sunwoo Moon⁹ , Per Petersson⁹ , Aleksandra Baron-Wiechec¹³ , Liga Avotina¹⁴ JET Contributors¹⁵

¹ United Kingdom Atomic Energy Authority, Culham Centre for Fusion Energy, Culham Science Centre, Abingdon, OXON, OX14 3DB, United Kingdom

² National Institute for Laser, Plasma and Radiation Physics, Magurele 077125, Romania

³ IPFN, Instituto Superior Técnico, Universidade de Lisboa, 1049-001, Lisboa, Portugal

⁴ Max-Planck Institut für Plasmaphysik, 85748 Garching, Germany

⁵ National Research Nuclear University MEPhI, 115409 Moscow, Russia

⁶ VTT Technical Research Centre of Finland, P.O.Box 1000, FIN-02044 VTT, Finland

⁷ University of Helsinki, PO Box 64, FI-00560 Helsinki, Finland

⁸ Forschungszentrum Jülich GmbH, Institut für Energie- und Klimaforschung – Plasmaphysik, Partner of the Trilateral Euregio Cluster (TEC), 52425 Jülich, Germany

⁹ Royal Institute of Technology (KTH), Association EURATOM-VR, 100 44 Stockholm, Sweden

¹⁰ CEA, IRFM, F-13108, St-Paul-Lez-Durance, France

¹¹ International Atomic Energy Agency, Vienna International Centre, PO Box 100, 1400 Vienna, Austria

¹² Laboratory for Plasma Physics, LPP-ERM/KMS, B-1000 Brussels, Belgium

¹³ Guangdong Technion—Israel Institute of Technology, Shantou, 515063, People's Republic of China

¹⁴ University of Latvia, Institute of Chemical Physics, Riga, Latvia

¹⁵ See the author list of 'Overview of JET results for optimising ITER operation' by J. Mailloux et al. to be published in Nuclear Fusion Special issue: Overview and Summary Papers from the 28th Fusion Energy Conference (Nice, France, 10–15 May 2021) IAEA FEC fusion portal

E-mail: anna.widdowson@ukaea.uk

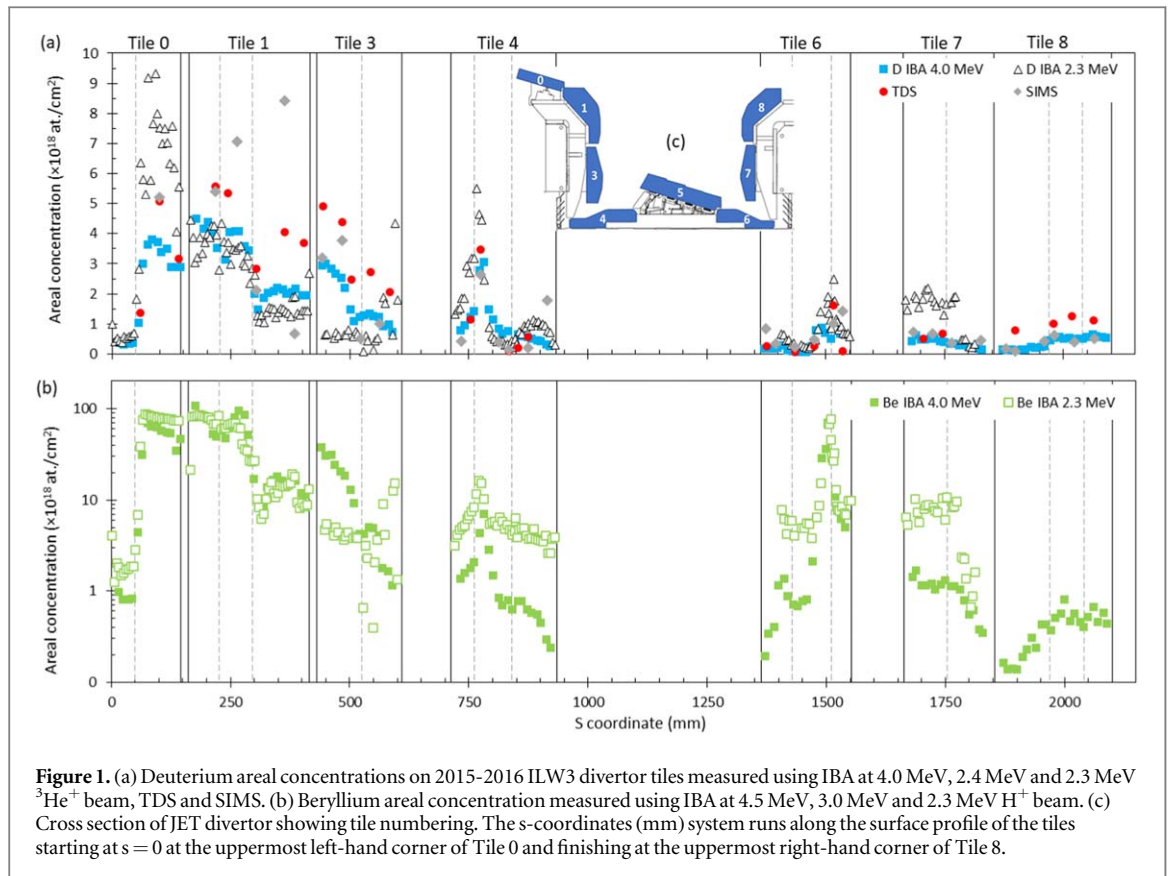
Keywords: JET, fuel retention, tritium

Abstract

An assessment of the tritium (T) inventory in plasma facing components (PFC) during JET T and deuterium-tritium (DT) operations is presented based on the most comprehensive *ex situ* fuel retention data set on JET PFCs from the 2015–2016 ILW3 operating period is presented. The global fuel retention is 4.19×10^{23} D atoms, 0.19% of injected fuel. The inner divertor remains the region of highest fuel retention (46.5%). The T inventory in PFCs at the end of JET operations is calculated as 7.48×10^{22} atoms and is informative for accountancy, clean-up efficacy and waste liability assessments. The T accumulation rate at the upper inner divertor during JET DT operations has been used to assess the requirements and frequency of operation of a new laser induced desorption diagnostic to be installed on JET for the final DT experiments in 2023.

1. Introduction

JET started its most recent and final tritium operating phase in 2021, with 100% tritium and tritium-protium ('hydrogen') plasmas and will move to deuterium-tritium (DT) operations from mid-2021. The DT operations will comprise two experimental campaigns, DTE2 in 2021 and DTE3 in 2023. Details of the experimental objectives of DT operations can be found in [1]. These are the first DT experiments since the DT experiments in 1997 (DTE1) [2]. Between DTE1 and DTE2 JET has operated with deuterium and protium. The operation in tritium will result in tritium being retained in-vessel, in particular in plasma facing components. This has implications for the accountancy of tritium, safety case considerations and the final waste liability of nuclear materials arising from JET.



Since 2011, when JET started operating with the all metal ITER-Like Wall (JET-ILW) [3], plasma facing components have been periodically removed from the JET vessel for analysis in a laboratory setting, i.e., *ex situ* analysis. The components are exposed to a varied experimental campaign with ~20 h of plasma operations [4], consisting in the order of 3000 JET pulses with ~20 s discharge time per pulse over a 12–18 month period. The results from these components have provided data on *ex situ* long term fuel retention, experimental evidence for the material migration processes in JET, demonstrated the reduction in fuel retention, erosion and deposition for JET-ILW in line with JET experiments [5, 6] and provided benchmarking data for modelling [7, 8]. In this paper the fuel retention data associated with the 2015–2016 ILW3 operating period are presented. It brings together for the first time the results from ion beam analysis techniques, thermal desorption techniques and secondary ion mass spectroscopy to evaluate the global retention in JET for this period from all analysed areas of the vessel, providing the most comprehensive and recent summary of fuel retention for JET. From this data the global retention of fuel and distribution in different regions of the vessel are established and compared with previous results reported for the 2011-2012 ILW1 operating period [9]. Furthermore, fuel retention is used to assess the potential tritium retention following ongoing tritium operations in JET and associated waste liability assessments. The data is also used to aid in the design solution and operating schedule of a new laser induced desorption diagnostics to be installed in JET prior to the final deuterium-tritium operations in 2023 (DTE3). This diagnostic will demonstrate the monitoring of fuel retention in plasma facing components within in the fusion vessel, i.e., ‘*in situ* analysis’ as foreseen in ITER [10].

2. Experimental procedure

Plasma Facing Components (PFCs) and remote area diagnostics have been removed from JET following three operating periods of JET; 2011–2012 ILW1, 2013–2014 ILW2 and 2015–2016 ILW3 and comprehensive analysis has been completed, i.e., *ex situ* analysis. This paper concentrates on fuel retention analysis from ILW3. Discussions about impurity deposition rates and comparison of techniques are not included here. Results from three analysis techniques are presented in this assessment of global retention in JET; Ion Beam Analysis (IBA), Thermal Desorption Spectrometry (TDS) and Secondary Ion Mass Spectrometry (SIMS). In figure 1 the IBA data from two facilities are presented: IPFN Instituto Superior Técnico, Portugal and IPP Garching, Germany. The facility at IPFN has capability for 2.3 MeV for ³He⁺ and H⁺ beams, using WinNDF to analyse data. The facility at IPP has capability for 2.4 MeV ³He⁺ for NRA and 3.0 MeV and 4.5 MeV H⁺ for backscattering data

depending on deposit thickness, using SIMNRA to analyse data. Both facilities are described in detail in [11]. In both cases the data provided is to an analysis depth of $\sim 1 \times 20$ at cm^{-2} , which represents the maximum penetration of the $^3\text{He}^+$ beam assuming beryllium rich deposits with Be density = 1.8 g cm^{-3} . TDS data are obtained from the facility based at CCFE, United Kingdom using a ramp rate of 10 K min^{-1} to 775°C (Be) or 1000°C (W and CFC), 1 h anneal and non-controlled ramp down [12]. SIMS is performed at VTT, Finland using O^{+2} [13]. All systems have capability for handling tritium (T) and beryllium (Be) contaminated samples. The data for divertor PFCs is a mix of previously published and un-published data. Previously published data are [14, 15] for IBA and [12] for TDS and SIMS. The figure demonstrates that deposition and retention in the divertor are well characterised and understood and therefore the data is used to underpin fuel retention calculations as discussed in sections 3 and 4.

The global retention value has been calculated using the divertor data presented in figure 1(a) with additional data from main chamber beryllium (Be) limiter tiles [4], Be coated inconel (Be-inconel) recessed Inner Wall Cladding tiles (IWC) [16], Tungsten coated Carbon Fibre Composite (W-CFC) recessed inner limiter tiles, passive diagnostics from the outer recessed wall of the vacuum vessel [17] and remote divertor regions [17, 18]. Data for these analyses include IBA techniques from VR, Sweden and University of Helsinki, Finland with facilities described in [11]. As for the error that can be expected from this type of global extrapolation calculation, the standard deviation of the fuel retention data from all analysis sources is 39% for the inner divertor and 45% for the outer divertor, therefore errors in average values presented are of this order.

The aim of this paper is to provide the most up to date assessment of retention fraction in order to aid in predicting T retention during JET T operations. However in order to compare these results from ILW3 with previous results from ILW1 it is necessary to provide a set of data following the methodology presented in [9]. The differences in data sets for the latest ILW3 assessment presented here and previous ILW1 methodology are shown in parentheses { } in table 1 and throughout the text and are summarised as follows.

2.1. Tile surface areas

In the most part the same areas for each component type are used, except in the case of IWC which has been reduced from 11.2 m^2 to 7.2 m^2 to reflect a more recent assessment and data presented in [16]. The 2011–2016 ILW1–3 IWC data from [16] is scaled by the plasma operation time to provide a retention value for ILW3.

2.2. Retention in main chamber tile gaps

The retention calculation in [9] does not take into account retention in gaps and castellation cuts in the beryllium main chamber tiles of JET. To date the complete assessment in retention in gaps for ILW3 main chamber Be PFCs has not been completed. However it was shown in [19, 20] that gaps account for up to 50% of retention in the main chamber. In order to provide more complete analysis this fraction has been applied to the data in table 1.

2.3. Retention on tungsten Tile 5

As for retention in gaps discussed above the retention calculation in [9] does not take into account retention on the plasma facing surface of W components (*lamellae*) on Tile 5. To date analysis results for ILW3 are not available, therefore it is assumed to be 1% of the divertor retention as shown in [19].

2.4. Retention in remote areas of divertor

The latest Nuclear Reaction Analysis (NRA) and Elastic Recoil Detection Analysis (ERDA) datasets available for a variety of components from the inner and outer divertor corners and under Tile 5s are included in the latest analysis results. For the comparison of ILW1 and ILW3 only NRA data on a single component type (*louvre clip*) is taken into account.

2.5. Calculation of gas injection in atoms

One source of error in the determination of global retention as a function of injected fuel is the conversion of molecular gas injection in bar-litres (barl) to the number of hydrogen atoms. This requires an assumption about the temperature of the gas injection volume. For the ILW1 fuel retention reported in [9] the gas injection value $1.67 \times 10^{26}\text{D}$ (5406 barl) from the main chamber gas injection modules (GIMS) was calculated at the vessel operating temperature of 473K. However, gas injection conversions for JET operations are more usually done at 300K, i.e., room temperature, where temperature measurements of the calibrated gas volumes are available. This results in the gas injection for ILW1 coming to 2.61×10^{26} D atoms. For ILW3 the D gas injected was 4573 barl, giving 1.4×10^{26} or 2.21×10^{26} D atoms calculated at 473K and 300 K respectively. In general this paper will work with the conversion at 300K. The implications of this for fuel retention data is discussed in section 3.

Table 1. Summary of fuel retention on plasma wall components making up the JET-ILW after 2015–2016 operating period. The results combine un-published and previously published data.

Location in vessel	Material	Total area (m ²)	ILW3 D atoms ($\times 10^{22}$)	% of global retention	Estimate of T Inventory ($\times 10^{22}$)
Upper Dump plate ^a	Be	6.05	1.60	3.8 {4.1}	0.29
Inner Wall Guard Limiter ^a	Be	4.75	1.20	2.9 {3.0}	0.21
Recessed Inner Wall	Be/W	10.05 {15}	3.99 {5.90}	9.5 {15.0}	0.71
recessed inner limiters ^b	W-CFC	2.85 {4.0}	1.56 {2.19}	3.7 {5.6}	0.28
recessed inner wall ^c	Be-inconel/ W-CFC	5.36/ 1.84 {11.0}	2.43 {3.71}	5.8 {9.4}	0.43
Outer Poloidal Limiter ^a	Be	10.08	3.20	7.6 {8.1}	0.57
Recessed Outer Wall ^d	Inconel	44	0.44	1.0 {1.1}	0.08
Main Chamber Gaps ^c	Be	100 {–}	5.22 {–}	12.4 {–}	0.93
Divertor ^{b,f}	W	20.27 {16.97}	25.3 {25.1}	60.5 {63.7}	4.52
inner divertor (Tiles 0, 1, 3, 4)	W-CFC	8.73	19.5	46.5 {49.5}	3.47
outer divertor (Tiles 6, 7, 8)	W-CFC	8.24	5.61	13.4 {14.2}	1.00
Tile 5 ^g	W	3.30 {–}	0.25 {–}	0.6 {–}	0.04
Remote areas in divertor ^{b,h}	SS/Inconel	2.87	0.95 {1.96}	2.3 {5.0}	0.17
inner corner	SS/Mo/Inconel	0.71	0.48 {1.27}	1.1 {3.2}	0.09
outer corner	SS/Mo/Inconel	1.15	0.34 {0.69}	0.8 {1.8}	0.06
under Tile 5	SS/Mo/Inconel	1.01 {–}	0.13 {–}	0.3 {–}	0.02
Total		198.07 {98.72}	41.9 {39.4}	100	7.48

^a [4].^b un-published data.^c [16].^d [17].^e contribution taken as 50% of main chamber retentions [19, 20].^f [14, 15].^g contribution taken as 1% of divertor retention [19]. The areas on divertor tiles shown in column 3 are scaled by 0.85 of actual surface area to take account for shadowing at the leading edge where there is little or no deposition. Data in parentheses { } are calculated according to assumptions in [9] to allow direct comparison of retention data. The last column provides an *ex situ* tritium retention based on the throughput of 3.94×10^{25} T atoms during T & DT operations in JET. This is expected to be an upper limit as it does not take into account *in situ* cleaning prior to the removal of components. The materials of the components are included to aid in assignment of waste liability.^h [17, 18].

The T throughput presented here includes (i) T injected into the machine to date (9.47×10^{24} T atoms/47.4 g at 300K) for JET pulse numbers (JPN) 98043 to 98925 from GIMs and neutral beam injection (NBI), and (ii) estimated T throughput for the remaining T operations; 12 days of T, 39 days DTE2 and 12 days DTE3. In each case $4 \text{ g}/7.99 \times 10^{23}$ atoms T is assumed to be injected in T operations, and $2 \text{ g}/3.99 \times 10^{23}$ atoms in DT operations. This is based on operational limits per day of 11 g T usage, of which 4 g (~0.33) is injected into the main vessel during T operations. The remaining 7 g (~0.67) goes to the Neutral Injector Boxes (NIBs). For DT plasmas operational overheads related to T management are likely to limit the total number of plasma pulses per day, and therefore 2 g T per day is assumed. This results in a total T throughput into the main vessel of 3.94×10^{25} atoms, 197 g. These values are significantly lower than a previous estimate [21] where it was assumed 9.9 g (0.9) injected into the vessel and 1.1 g (0.1) in the NIBs with 32 days of T and 64 days of DT giving a throughput of 950 g. The consequence is that the T inventory values in this paper will be less than previously reported.

3. Results

The results for the D fuel retention and Be deposition in the divertor are shown in figure 1. The overall picture of material migration is similar to that of ILW1, where Be eroded in the main chamber migrates in the scrape off layer to the top of the inner divertor region. Once deposited it tends to remain in this region and does not migrate to remote divertor corners, as was the case for the JET Carbon wall (JET-C), where carbon-hydrogen chemistry provided an additional erosion mechanism and subsequent erosion-redeposition step-wise transport to the divertor corners, remote from the plasma.

The best estimate of the D/Be ratio can be taken in the s-coordinate region 60–290 mm where the thickest Be deposits are located on Tile 0 and the top of Tile 1. The thickness of the deposits is similar to the interaction depth from the IBA and therefore is not greatly influenced by the W tile surface ‘substrate’ below. In thick

deposits the C concentration is found to be of the same order as the D concentration as shown in [22] and [23]. Taking the average of data from these sources $D/Be = 0.07 \pm 0.03$ and $C/Be = 0.05 \pm 0.02$ for the interaction depth $\sim 8 \mu\text{m}$, for Be density $= 1.8 \text{ g cm}^{-3}$.

Including D retention data from figure 1 and other areas of the vessel, as discussed in section 2, the retention for each region was determined and is summarised in table 1. The total global D retention is found to be 4.19×10^{23} D atoms $\{3.94 \times 10^{23}$ D for comparison with [9]}. Assuming the D injection of 2.21×10^{26} D atoms at 300K this gives a retention fraction of $0.19\% \pm 0.08\%$ $\{0.18\%\}$. To compare with ILW1 global retention data presented in [9] the 0.23% retention calculated at 473K reduces to 0.14% when calculated at 300K using gas injection values discussed in section 2. The implications are that the overall retention as a fraction of injected fuel are similar for ILW1 and ILW3 and that the global retention value for ILW1 is lower than previously reported.

The global retention rate for ILW3 normalised to the *divertor plasma time* ($18.5 \text{ h}/6.7 \times 10^4 \text{ s}$), i.e., the total time where an x-point is formed and the plasma strikes the inner and outer divertor tiles, is 6.3×10^{18} D atoms/s $\{5.9 \times 10^{18}$ D atoms/s}. For ILW1 it is 7.9×10^{18} D atoms/s normalised to the divertor plasma time ($13 \text{ h}/4.7 \times 10^4 \text{ s}$) [9]. Irrespective of the retention data set used for ILW3 the global fuel deposition rate has decreased in comparison to ILW1. The ratios of the retention fraction and retention rates for ILW1: {ILW3} are 1: {1.3} and 1: {0.7} which will represent the predominant plasma scenarios for ILW3. Such differences are discussed in [5], where a range of plasma scenarios with different fueling rates, auxiliary heating, pumping conditions result in different retention fractions and retention rates.

The divertor retention fraction dominates the main chamber, with the inner divertor remaining the highest region of deposition. The remote corner deposition has reduced in ILW3 accounting for 3.6% $\{7.3\%\}$ of divertor retention in ILW3 compared with 16.7% in ILW1. The results show that the recessed inner wall can contribute a significant fraction of the global retention with the latest retention data for Be-inconel IWC [16] and W-CFC recessed limiter tiles contributing $\sim 10\%$.

Overall, the retention fraction for ILW3 is similar to ILW1 and reaffirms the efficacy of the all-metal JET-ILW at reducing fuel retention compared with the JET-C where retention was found to be 4%, i.e., 66 g D retained and 1800 g injected in the MkII-SRP operations 2001–2004 [24]. For JET-ILW, the retention fraction has decreased by at least an order of magnitude compared with JET-C.

4. Discussion

JET is operating with 100% T and T-hydrogen (H) gas mixtures from March—June 2021. This is followed by DT operations (DT-experiment 2 (DTE2)) until September 2021 and DTE3 in early 2023. DTE2 and DTE3 will be followed by D operations to reduce T inventory in the vessel. This discussion considers how the long-term global retention data determined from *ex situ* analysis may be applied to assess T accumulation in JET, the efficacy of T removal experiments and use of a future laser induced desorption diagnostic on JET. Finally, the results can be used to estimate T inventory for waste liability and waste reduction experiments.

4.1. In-situ retention and *ex situ* long-term retention

As previously reported *ex situ* long-term fuel retention assessed on components removed from JET yield a lower fuel retention assessment than *in situ* retention measured by gas balance during operations [5]. In the vessel there is a dynamic retention cycle—fuel loading during plasma pulses and out-gassing after plasma pulses are completed. The *in situ* deuterium retention normalised to the divertor plasma operation time is in the range $0.2\text{--}1.5 \times 10^{20}$ D atoms/s dependant on the plasma scenario [5] during ILW1. To provide a representative comparison the *ex situ* retention rates for ILW1 and ILW3 are 7.9×10^{18} D atoms/s and 6.3×10^{18} D atoms/s when normalised to the divertor plasma time. Comparing ILW1 data the *in situ* retention rate determined for individual plasma scenarios is a factor 3–24 higher than the *ex situ* retention rate determined from a global campaign average. This range is wide as the *in situ* retention rates are highly dependent on the individual plasma scenario. The largest differences arising from L-mode plasmas with 0.5 MW RF heating with turbomolecular pumping and type III ELMy H-mode plasmas with 5.0 MW heating by NBI [5], an unlikely operating condition for JET during T operations. Therefore, a more realistic range is 3–11. This is consistent with a wider study of gas balance and fuel retention in fusion devices indicates that *in situ* retention is generally 3–6 times higher than *ex situ* values [25]. Given the varied plasma operations programme at JET, the average *ex situ* retention will lie between these extremes. In addition, as more PFCs are analysed a more complete picture with increasing inventory is established, therefore a factor 4 is applied between *in situ* and *ex situ* retention in the T retention assessments in later sections.

For ILW1 and ILW3 there were no cleaning/isotope exchange experiments at the end of the operating periods to take into consideration. Therefore, the difference will be a combination of *in situ* and *ex situ* out-

gassing. Both processes result in the reduction of fuel retention in PFCs but via different mechanisms. Dynamic fuel retention occurs in PFCs from plasma wall interaction. ‘*In-situ* out-gassing’ is the in-vessel release of fuel from plasma loaded PFCs in the form of molecular hydrogen isotopes. Whereas ‘*ex situ* out-gassing’ occurs via isotopic exchange with water when PFCs are exposed to air on venting and is an on-going process for PFCs after removal. *In-situ* out-gassing will contribute to the reduction of fuel over the timescale of days to weeks from the end of plasma operations to the venting of the machine and has been shown to be $\propto t^{-0.8}$ [5, 26]. However *ex situ* out-gassing will contribute on the timescale of 0.5–2 years before all components from an operating period are analysed. *Ex-situ* out-gassing on venting was measured following DTE1 when released T was measured using a water bubbler. Of the 35 g throughput of T in DTE1, the amount of T remaining in the vessel at the end of operations was 14 g. This was reduced to 6.2 g after D and H cleaning. On venting the vessel a further 2 g of T was released ‘*ex situ* out-gassing’ [27], i.e., 6% of the total injected T. Whilst there was no cleaning experiment at the end of the ILW3 operating period, loss of D from PFCs on venting is to be expected. Based on the experience of DTE1 this could be of the order of 1/3 removed on venting. The effect of these out-gassing mechanisms on *ex situ* fuel retention analysis is currently being assessed and will be reported in the future. Similar conclusions regarding the differences in *in situ* and *ex situ* retention have been reported for Tore Supra [28] where significant fractions of fuel were released from PFCs during wall conditioning/cleaning, *in situ* out-gassing and venting.

4.2. Operations

Tritium inventory calculations are needed at various stages of the operational cycle; *in situ* T accumulation and accountancy, to inform the efficacy of T removal by cleaning methods and *ex situ* for assessing the handling of radioactive materials and waste liability. The T inventory assessment results in a total global inventory of 7.48×10^{22} atoms and is summarised for the various components in table 1. It assumes the 0.19% global retention rate and T throughput 3.94×10^{25} atoms as discussed in section 2. In terms of T accumulation and accountancy during JET T & DT operations this method can be applied at any point in operations and form part of the overall accountancy which includes T injected into the JET vessel and T recovered to the Active Gas Handling System. This type of assessment will be ongoing until the end of T operations at JET.

The results in table 1 are also applicable for waste liability assessment, and with this in mind the material for each component type is listed for reference, although no further analysis relating to material type is discussed here. Since the results are based on ILW3 *ex situ* tile analysis the effect of T removal is not taken into account (as no fuel (D) removal operations took place at the end of ILW3). In this respect the assessment is higher than might be expected at the end of JET T & DT operations where T removal is planned, and therefore provides a safety margin. A further safety margin could be applied by considering a higher T throughput of 4 g injected per day throughout T & DT operations. This would increase the total throughput to 5.98×10^{25} T atoms, with an associated global retention of 11.35×10^{22} atoms. Note that this is not a factor two higher as it is only the 51 DT operating days where the daily T throughput has been increased to 4 g, see section 2.

To give some perspective on the efficacy of cleaning, isotopic exchange experiments with gas balance analysis have shown an accessible reservoir of 3×10^{23} atoms in the ILW2 operating period [29]. Isotope exchange between H and D has also been demonstrated by *ex situ* analysis of Be and W components where H displaces D in components removed after ILW2 operations which ended in H plasmas and resulted in H rich surfaces [30]. During current JET operations 5.3×10^{23} D atoms were release from the vessel wall following baking, ion cyclotron wall cleaning and glow discharge experiments compared with 6.50×10^{26} D atoms injected between JPN 92505 to 98199 (from the start of plasma operations in 2018 to immediately prior to fuel removal experiments) [31]. A direct comparison of *ex situ* fuel retention and fuel removal operations is difficult to achieve for JET as there is no access to components until the end of operations. Therefore, the evaluation of the efficacy of fuel removal relies on previous *ex situ* and *in situ* fuel retention analysis. Considering the *ex situ* 0.19% retention, 1.24×10^{24} D atoms were retained prior to the fuel removal experiments which implies that 43% of this was removed by cleaning. However if *in situ* retention is higher, as discussed in section 4.1, then this reduces the efficacy to $\sim 10\%$. This supposes that all retained fuel is accessible for removal and that pre-existing inventory is not accessible. Although this is an oversimplification it does confirm that the fuel removal analysis does not exceed retention analysis predictions. Using the D/Be ratio 0.07, discussed in section 3, it is possible to estimate the depth of the accessible reservoir in Be co-deposits. If 46.5% of the fuel removed is assumed to be in the co-deposits at the top of the divertor, then 2.5×10^{23} D atoms are removed from 3.5×10^{24} Be atoms over an area 2.9 m^2 (the area taking in Tile 0 and the top of Tile 1 to s coordinate 296 mm). This results in a Be amount of 1.2×10^{20} Be atoms/cm², which is equivalent to $\sim 10 \text{ }\mu\text{m}$ for a fully dense layer. For DTE1 a high fraction of T was released where 14 g of retained T was reduced to 6.2 g after clean-up with D and H [27]. This clearly leaves a wide range which will be the subject of future analysis during the remaining JET T & DT operations.

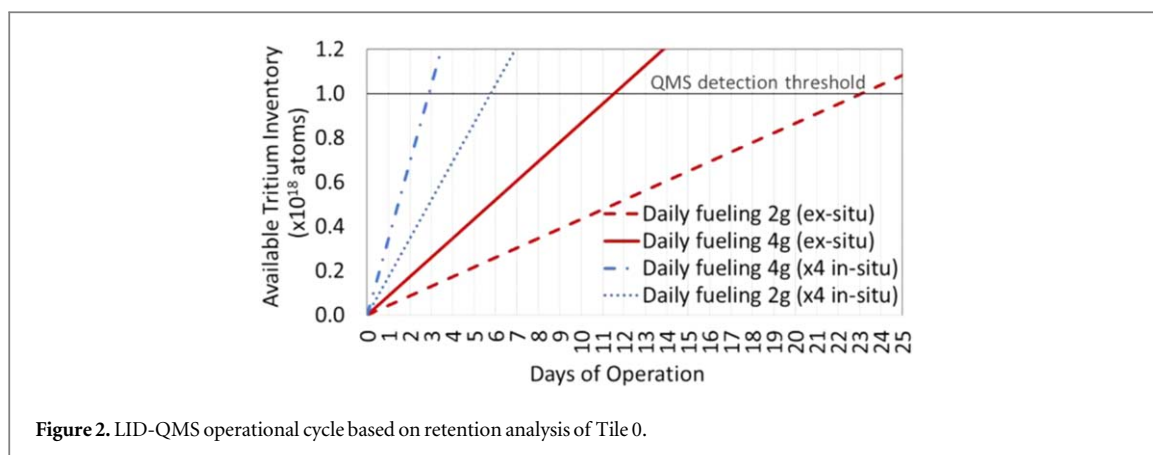


Figure 2. LID-QMS operational cycle based on retention analysis of Tile 0.

4.3. Design and operation of laser induced desorption diagnostic

A Laser Induced Desorption diagnostic with detection by Quadrupole Mass Spectrometry (LID-QMS) is currently being designed for installation on JET. It is based on the experience of previous LID-QMS measurements [32–34]. It will be operational during DTE3 operations in 2023 and will demonstrate the capability for *in situ* fuel retention analysis including T as foreseen in ITER [10].

For the specific case of the LID-QMS diagnostic the *ex situ* analysis has provided data to facilitate a design solution for the laser specifications and laser beam rastering requirements needed to release sufficient fuel for detection by QMS, which is located approximately 2 m below the main vacuum vessel, 120° toroidally from the target tile. The evaluation was based on the retention rate on the target tile, Tile 0 at the top of the inner divertor where the highest fuel retention occurs in the deposition zone between *s*-coordinates 60–150 mm, see figure 1. In this region the areal concentration of D on Tile 0 was 5.36×10^{18} D atoms cm^{-2} when averaging all IBA, TDS and SIMS data. Assuming a ratio of 4.0×10^{23} (2 g) T atoms injected per day: 2.2×10^{26} D atoms injected in ILW3; the accumulating areal concentration of T on Tile 0 is 9.7×10^{15} T/ cm^2 /day. This provides the lower limit for daily T accumulation and therefore represents that most challenging retention rates on which to specify the laser rastering requirements and frequency of operating the diagnostic. The detection limit of the QMS has been investigated by a series of gas injections. To date the detection level achieved is 5×10^{17} D molecules injected into the JET vessel in a 0.5 s interval, detected as mass 4 molecules, this is equivalent to 1×10^{18} atoms desorbed from deposits. The lowest detection limit has not yet been established due to insufficient time available on JET prior to moving to T operations. The achievable laser operating conditions are assumed to be $\sim 5 \text{ cm}^2$ area rastered in 0.5 s, with 0.07 cm^2 spot at 140 Hz and 90% fuel removal efficiency. Based on these assumptions the T fuel available for desorption is given in figure 2. In the worst case from *ex situ* analysis a change in T inventory at one location would only be detectable every 23 days. In the best case applying a factor 4 to take account of differences in *in situ* and *ex situ* retention rates as discussed in section 4.1, the detection interval reduces to 3–6 days. There are a number of factors that will affect these upper and lower limits presented in figure 2. Firstly, the *in situ* retention rate will decrease as *in situ* out-gassing will reduce fuel retention in PFCs. For example one hour after a JET pulse the pressure in the vessel decreases by more than an order of magnitude along with a reduction in mass 4 (D_2) detected by QMS which is indicative of decreasing fuel inventory in PFCs [5]. Secondly the detection limit of the QMS to the isotopic hydrogen molecules formed during desorption above the background out-gassing will play a role. In the tests performed where D_2 molecules are injected into JET we anticipate that the detection limit above the existing out-gassing background will be lower than currently established. In this case a factor 2 would result in the longest interval between detectable changes in retention reducing to 6–12 days for *ex situ* retention and 2–3 days for *in situ* retention. However, in the DT campaigns this assumption may not hold. It will depend upon the distribution of isotopic hydrogen molecules formed during the desorption process, HD, HT, D_2 , DT, T_2 , and the background out-gassing in DT operations. Indeed, the interplay between the dynamic fuel retention, out-gassing and vacuum conditions needed for optimal operation of the LID-QMS diagnostic will finally be determined once installed. Despite these uncertainties, it should be noted that this assessment of the interval between detectable changes in fuel retention in one location does not prevent the LID-QMS diagnostic from being operated more frequently as different sampling areas may be targeted. In addition, it will not be necessary to wait for fuel to accumulate on the target tile as there will be D and T retained on the extant tile from earlier JET T & DT operations.

5. Conclusions

Ex-situ analysis of components removed from JET for the 2015–2016 ILW3 operating period are presented and the global retention as a percentage of the injected fuel determined as $0.19\% \pm 0.08\%$. The distribution of retained fuel also follows the established material migration pattern with 46.5% of the global fuel retention at the inner divertor predominantly in co-deposits on Tile 0 and Tile 1. The retention in the divertor corners as a fraction of divertor retention has reduced by a factor >2 for ILW3 compared with ILW1. The *ex situ* retention rate remains lower than *in situ* retention rates determined from gas balance. Out-gassing *in situ* due to dynamic retention and on venting are likely to account for this difference.

The global and component-based retention fractions are used to assess future T retention based in JET T & DT operations. These results are discussed in the context of T waste liability assessment and design and operation of LID-QMS diagnostic on JET. For the waste liability, this assessment provides an upper T retention, as T clean-up experiments which will take place at the end of T operations are not applied here. Cleaning experiments have shown to reduce fuel inventory by 43% when estimated from *ex situ* retention values, down to $\sim 10\%$ if scaled for *in situ* retention and the depth of the accessible reservoir is $\sim 10 \mu\text{m}$ in Be co-deposits. For the LID-QMS assessment the *ex situ* results provide a lower limit for T retention, which in turn provides a challenging criterion for achieving a design solution. Whilst estimation of retention from *ex situ* data provides lower retention rates than will arise in vessel, a scaling factor 4 has been applied for *in situ* analysis. However, this can be regarded as an upper retention rate limit, as *in situ* out-gassing will result in the reduction of *in situ* retention in PFCs.

The results show that the of *ex situ* analysis of PFCs plays a useful role in T retention analysis and will continue during JET T & DT operations for accountancy purposes, quantification of T clean-up and monitoring of T waste liability.

Acknowledgments

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014–2018 and 2019–2020 under grant agreement No 633053 and from the RCUK [grant number EP/T012250/1]. To obtain further information on the data and models underlying this paper please contact PublicationsManager@ukaea.uk. The views and opinions expressed herein do not necessarily reflect those of the European Commission. The research used UKAEA's Materials Research Facility, which has been funded by and is part of the UK's National Nuclear User Facility and Henry Royce Institute for Advanced Materials.

Data availability statement

The data that support the findings of this study are available upon reasonable request from the authors. Contact PublicationsManager@ukaea.uk.

ORCID iDs

Anna Widdowson  <https://orcid.org/0000-0002-6805-8853>
J Paul Coad  <https://orcid.org/0000-0001-8359-7073>
Ionut Jecu  <https://orcid.org/0000-0001-8567-3228>
Eduardo Alves  <https://orcid.org/0000-0003-0633-8937>
Norberto Catarino  <https://orcid.org/0000-0003-3879-1533>
Matej Mayer  <https://orcid.org/0000-0002-5337-6963>
Stepan Krat  <https://orcid.org/0000-0001-6223-293X>
Chris Rowley  <https://orcid.org/0000-0001-6163-6836>
Mirosław Zlobinski  <https://orcid.org/0000-0002-1395-7165>
Marek Rubel  <https://orcid.org/0000-0001-9901-6296>
Kalle Heinola  <https://orcid.org/0000-0002-0601-8274>
Tom Wauters  <https://orcid.org/0000-0002-2941-7817>
Laura Dittrich  <https://orcid.org/0000-0003-1933-8186>
Sunwoo Moon  <https://orcid.org/0000-0002-0865-7387>
Per Petersson  <https://orcid.org/0000-0002-9812-9296>
Aleksandra Baron-Wiechec  <https://orcid.org/0000-0001-9458-6679>

Liga Avotina  <https://orcid.org/0000-0001-5197-4196>

References

- [1] Joffrin E 2019 *Nucl. Fusion* **59** 112021
- [2] Keilhacker M, Watkins M L and JET Team 1999 *J. Nucl. Mater.* **266–269** 1–13
- [3] Matthews G F et al 2007 *Phys. Scr.* **T128** 137–43
- [4] Widdowson A et al 2020 *Phys. Scr.* **T171** 014051
- [5] Brezinsek S et al 2013 *Nucl. Fusion* **53** 083023
- [6] Brezinsek S et al 2015 *Nucl. Fusion* **55** 063021
- [7] Schmid K et al 2015 *Nucl. Fusion* **55** 053015
- [8] Romazanov J et al 2017 *Phys. Scr.* **T170** 14018
- [9] Heinola K et al 2016 *Phys. Scr.* **T167** 014075
- [10] Reichle R et al 2015 *J. Nucl. Mater.* **463** 180–4
- [11] Mayer M et al 2020 *Nucl. Fusion* **60** 025001
- [12] Likonen J et al 2019 *Nucl. Mater. Energy* **19** 300–6
- [13] Lahtinen A et al 2019 *Fusion Eng. Des.* **146** 1979–82
- [14] Krat S et al 2020 *Phys. Scr.* **T171** 014059
- [15] Catarino N et al 2020 *Phys. Scr.* **T171** 014044
- [16] Dittrich L JET Contributors et al 2021 *Submitt. to Phys. Scr.* **96** 124071
- [17] Moon S et al 2019 *Nucl. Mater. Energy* **19** 59–66
- [18] Ström P et al 2019 *J. Nucl. Mater.* **516** 202–13
- [19] Widdowson A et al 2017 *Nucl. Fusion* **57** 086045
- [20] Rubel M et al 2017 *Nucl. Fusion* **57** 066027
- [21] Widdowson A et al 2016 *Phys. Scr.* **2016** 014057
- [22] Krat S et al 2020 *Phys. Scr.* **T171** 014059
- [23] Catarino N et al 2017 *Nucl. Mater. Energy* **12** 559–63
- [24] Likonen J et al 2009 *J. Nucl. Mater.* **390–391** 631–4
- [25] Loarer T et al 2007 *Nucl. Fusion* **47** 1112–20
- [26] Philipps V et al 2013 *J. Nucl. Mater.* **438** S1067–71
- [27] Peacock A T et al 2000 *Fusion Eng. Des.* **49–50** 745–52
- [28] Pégourié B et al 2013 *J. Nucl. Mater.* **438** 120–5
- [29] Loarer T et al 2015 *Nucl. Fusion* **55** 043021
- [30] Dittrich L et al 2021 *47th Conf. on Plasma Physics* (European Physical Society) <http://ocs.ciemat.es/EPS2021PAP/pdf/O1.101.pdf>
- [31] Wauters T et al 2021 *18th Int. Conf. Plasma Facing Materials*
- [32] Schweer B et al 2007 *J. Nucl. Mater.* **363–365** 1375–9
- [33] Schweer B et al 2009 *J. Nucl. Mater.* **390–391** 576–80
- [34] Zlobinski M et al 2019 *Fusion Eng. Des.* **146** 1176–80