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ABSTRACT

During 1997 JET operation with D-T plasmas, 35g of tritium were introduced into the torus, mainly by gas puffing. It was found that during this period, the torus tritium inventory would accumulate at a rate of about 40% of the input. After tritium operation ceased, the experimental program continued with deuterium and hydrogen fuelled experiments, during which time the tritium inventory decreased to about 17% of the total input. Techniques aimed at detritiation of the torus included methods using deuterium gas (such as deuterium pulsing) which were used in the middle of the experimental campaign, and methods which could adversely affect the torus vacuum conditions (such as air purge)s which were reserved for the period after the experimental campaign. Whilst it was found that the plasma tritium fraction could be reduced to below the 1% level in a few days, the tritium inventory reached a virtually steady level of about 6g by the end of the campaign.

1. INTRODUCTION

During 1997 JET was operated for an extensive period using mixed deuterium-tritium (D-T) plasmas. This experimental campaign was known as ‘DTE1’ and the main physics results from the campaign are reported in refs [1]-[3]. There were several reasons for tritium clean-up following DTE1. Firstly, the campaign was restricted by the total number of DT neutrons which could be generated, in order to limit the activation of the machine and allow Torus Hall access within a short time of pulsing being finished. Therefore care had to be taken to reduce the plasma tritium fraction before resuming deuterium pulsing with high neutral beam power, so as not to consume the neutron budget unnecessarily. Secondly, it was important to reduce the tritium inventory as much as possible to minimise any escape of tritium from the ventilated torus during the Remote handling Tile Exchange (RTE) shutdown which followed the campaign. Finally, it is extremely important to be able to forecast the level of tritium retention which might be expected in future larger experiments and ultimately a reactor. Such machines will use much larger quantities of tritium than JET required for DTE1, and large torus inventories present a safety hazard in the form of a potential tritium release to the atmosphere in the case of a loss of vacuum, as well as tying up a large financial investment.

2. EXPERIMENTAL MEASUREMENTS

Figure 1 shows a much simplified version of the tritium fuelling and recovery system. The Active Gas Handling System (AGHS) delivers tritium to both the torus and the Neutral Beam Injection (NBI) systems, and is able to make accurate measurements of the individual amount of tritium supplied to each system [4]. The tritium was introduced into the torus via a gas introduction module comprising a fast valve and a reservoir which would be topped up with tritium between tokamak pulses by the AGHS. Through a network of pipes and valves, the tritium from the module could be routed to either an outer midplane port or a distributed source along the inner

horizontal divertor target. Only one of the two NBI boxes on JET (the Octant 8 injector) was supplied with tritium [5]. Because most of the gas supplied to a NBI box condenses on a cryopump within the box, and because beams are often fired before and between tokamak pulses for the purposes of conditioning (in which case the system does not actually fuel the torus), only about 1% of the tritium supply to the NBI systems is ultimately injected into the torus. The AGHS can also supply deuterium to these systems, but during the DTE1 campaign the deuterium was supplied to the non-tritium (octant 4) injector box by the conventional gas delivery systems. The AGHS collects exhaust gas from both the torus and NBI systems and reprocesses the exhaust for further tritium fuelling [4].

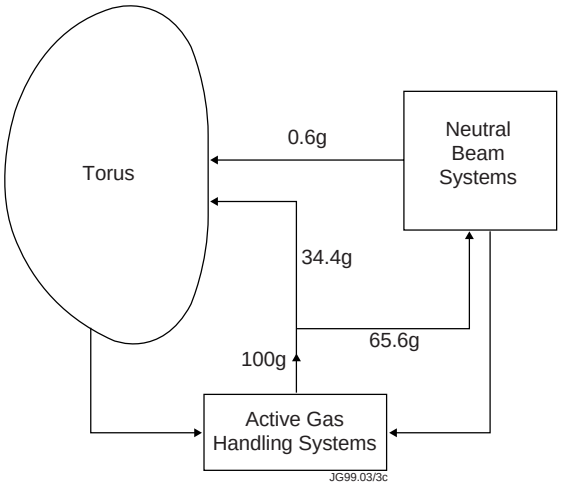

Fig.1: Simplified schematic of the JET tritium delivery and recovery system. The total tritium throughput for DTE1 is shown for the different input branches.

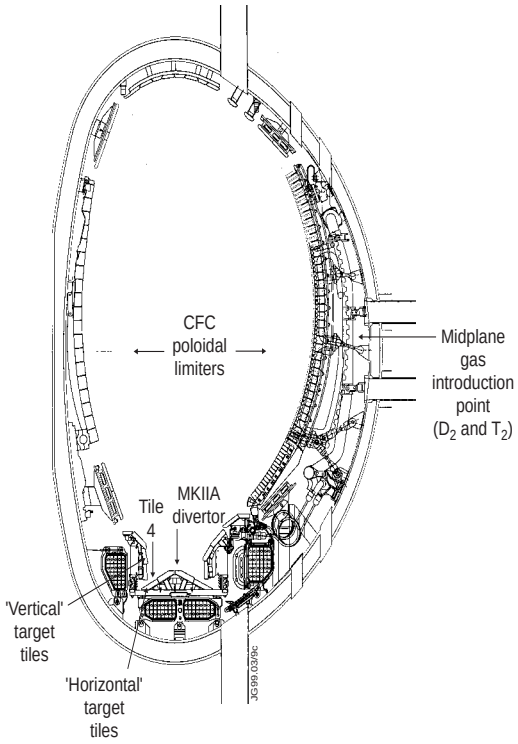

Fig.2: Cross section of the JET torus vacuum vessel, as existed in DTE1.

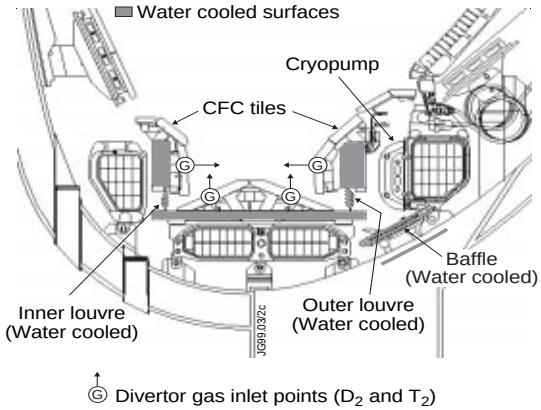

Fig.3: Cross section of the JET MkIIa divertor.

DTE1 occurred during the campaign when the JET MkIIa divertor was installed in the torus [6]. The vessel interior (Fig.2) comprises carbon - carbon fibre composite (CFC), tiles which are periodically coated with thin beryllium films. Poloidal limiters are used during start-up phases of the discharge which typically last about 10 seconds. The main difference since the

time of the 1991 Preliminary Tritium Experiment (PTE) [7] in JET is the presence of the pumped divertor, shown in Fig.3. Although plasmas were run in divertor magnetic configurations during the PTE, the MkIIa divertor is shaped to conform more closely to the boundary of the plasma. Pumping slots in both corners of the divertor allow neutral particles to be pumped onto a toroidal cryopump located behind the outer divertor target tiles [8]. This is known as the Pumped Divertor (PD) cryopump. The divertor support structure is water cooled, and as a result the base temperature of the divertor tiles is 200°C, significantly cooler than the ambient temperature of the rest of the torus (320°C). Water cooled louvres, located in the pumping slots, shield the divertor cryopump and divertor coils from direct line of sight with hot surfaces and particles, while allowing the transmission of particles to the cryopump.

In the PTE the plasma tritium fraction was always below 10^{-3} except for the two high power pulses, where there was an estimated 15% tritium in the core due to fuelling by NBI. In DTE1 the plasma tritium fraction covered the range relevant for a fusion reactor. Several diagnostic measurements of the plasma tritium fraction were available which could not be made in the PTE. H-alpha light measurements at the plasma boundary and from a species specific Penning discharge diagnostic [9] sampling neutral gas near the divertor cryopump could measure tritium fractions down to a few percent. The latter measurement was particularly useful since it made it possible to perform a pulse by pulse measurement of the tritium inventory even when the divertor cryopump was cooled to supercritical Helium (ScHe) temperatures (4.5K). When the helium panels are warm this quantity is available from torus exhaust measurements. At tritium fractions below about 10^{-2}, it was possible to measure separately the DD neutron and DT neutron yields. The ratio of these yields will be proportional to the tritium fraction [10]. The proportionality constant is in the range 200-400 depending on plasma temperature and NBI energy, which together determine the fraction of neutrons generated by beam/plasma or thermal reactions. An average value of 300 is used throughout this paper in order to derive the tritium fractions based on neutron measurements. A full description of these and other diagnostics used to measure the plasma tritium fraction is given in a companion paper [11].

Between pulses, measurements of the outgassing from the walls gave the pulse by pulse change in torus tritium inventory. Periodically the gases condensed on the helium panels of the PD cryopump and each of the NIB cryopumps were released by warming up the panels to > 40K (‘regeneration’), giving the integrated tritium exhaust over a period of days. A more detailed description of the exhaust measurements can be found in a companion paper [11]. The inventory measurements quoted in this paper are from the more accurate measurements of gas inventory after cryopump regeneration. The total quantity of gas was accurately measured with capacitance manometers and calibrated volumes. Activity was measured using ionisation chambers or, to greater accuracy, by gas chromatography. The most accurate measurements were made about once per week after the exhaust tritium was reprocessed [4].

3. DESCRIPTION OF THE EXPERIMENTAL CAMPAIGN

JET operated with the MkIIa divertor installed between April 1996 and February 1998. The DTE1 took place within this period, from May to November 1997. Figure 4 shows the evolution of the plasma tritium fraction during the MkIIa campaign. Prior to the first tritium injection the plasma tritium fraction was well below the detection limit for the experiment, ie. below the 10^{-5} to 10^{-4} level. The magnitude of this background level could be estimated by the ratio of the average tritium production per pulse, to the average deuterium gas fuelling per pulse which gave ~ 10^{-7} (T/D). At the start of DTE1, the tritium fraction can be seen to step up to 10^{-3} and then 10^{-2} due to initial experiments with small amounts of tritium either with NBI only or extremely short (< 100msec) gas puffs. The tritium fraction starts to fall after each of these steps as a result of the return to pure deuterium operation. The tritium fraction (T/(T+D)) then stepped up to the 0.2-0.9 range during the first period of heavy tritium gas fuelling (Phase I). Tritium beams were not available at this time due to technical problems. The tritium fraction was then reduced to T/T+D ~ 0.01 in about four days of deuterium pulsing, after which there was two month break in operation to do repair work on the NBI systems.

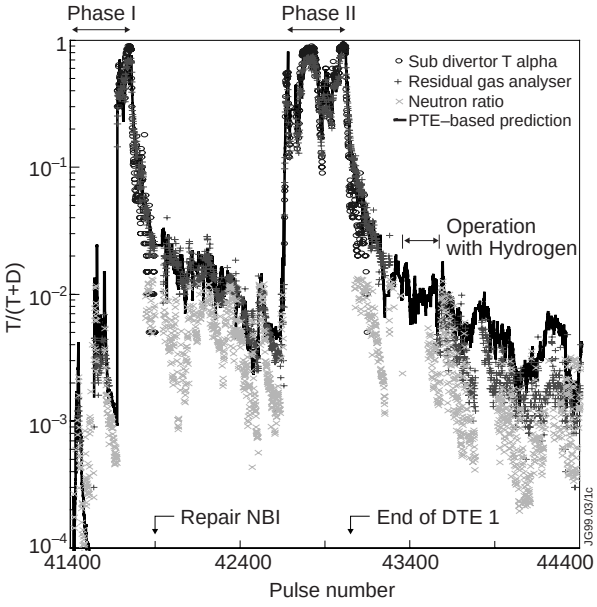

Fig.4: Plasma Tritium fraction measured throughout the DTE and subsequent clean-up by different diagnostics compared to expectation based on PTE

It had been normal operation is previous JET campaigns to operate low conductance (< 500 ℓ/s) 'Fast Shutters' which closed post-pulse to limit the gas pumped into the NIBs down the NBI duct [12]. As a result of an intervention in the middle of DTE1 (between Phase I and Phase II), following a small water leak in a Fast Shutter water hose, the Neutral Beam Fast Shutters were left permanently open [13].

Operation resumed after the intervention with an extensive period of deuterium-only operation in preparation for further DT experiments. The plasma tritium fraction during this time fell very slowly in the 10^{-3} range. The fact that the T/D ratio was as high as this meant that the DD neutron production and DT neutron production were of similar magnitude. This hampered the DD preparation in two ways: firstly it was difficult to diagnose the fusion performance of the discharges due to uncertainty in the relative contribution of DT and DD neutrons to the total neutron rate, and secondly, the DT neutron production was becoming a significant portion of the

total DT neutron budget, reducing the amount left for upcoming DT pulses. After about one month of deuterium operation, tritium operation resumed, this time with both NBI and gas fuelling of the tritium (Phase II).

After the last tritium pulse, the tritium fraction was reduced once again to about T/(T+D) ~ 0.01 by operation with pure deuterium plasmas. There was then a period of operation in hydrogen. This can be seen in Fig. 4 as a period without neutron ratio T/(T+D) data. Operation in hydrogen completed the H isotope scaling experiments [7], as well as allowing time for the tritium level to decrease without significant neutron production. The MkIIa divertor campaign ended with a further two months of deuterium operation during which the T fraction was about 10^{-3}. At this level, approximately 25% of the total neutron production is due to DT neutrons.

Also shown in Fig. 4 is an empirical estimate of the plasma tritium fraction based on the tritium fuelling history of the DTE1 campaign, and the tritium fraction observed in the exhaust gas following injection of 5mg of tritium during the PTE campaign [7]. The number of pulses to change over from one isotope to another is very similar to the PTE based prediction, despite the absence of an in-vessel cryopump during PTE. The PTE expectations can be seen to lie at the top end of the T/(T+D) measurements from the ratio of 14MeV to 2.5MeV neutrons. Figure 4 also shows exhaust stream measurements from Residual Gas Analysis after the pulse for the DTE1 pulses. These are also seen to give a somewhat higher value for T/(T+D) than that obtained from the neutron ratio. This is to be expected in a situation of isotope change-over in the Tokamak surfaces, where tritium will be displaced from the 'first wall' as pure deuterium fuel is injected into the discharge and this released tritium has a rather low 'fuelling efficiency' for entering the discharge.

During DTE1 35g of tritium were introduced into the torus: 0.6g by NBI, and the remainder by gas puffing. Figure 5 shows the torus tritium inventory for the same pulse range as Fig. 4. The torus inventory is defined to only include the tritium in the torus after the torus PD cryopump has been regenerated. The data points are measurements of the total amount of tritium recovered by the AGHS subtracted from the total site inventory. This is therefore the tritium inventory of all systems outside the AGHS (ie. NIBs, torus), but individual analyses of batches of exhaust gas from the different subsystems indicate that the torus contributes > 90% of the inventory. The maximum tritium inventory in the torus during DTE1 was ~ 11.5g, ie. more than half of the 20g of tritium on site.

Highly accurate measurements of the tritium inventory were available on a weekly basis. A typical week would consist of 3-4 days of tritium pulsing during which tritium would be recovered from the torus and NBI boxes, and a measurement made of the amount recovered from each system. The accuracy of these measurements was about 5%. Then there would be a period of typically four days during which there was no pulsing, and the exhaust gases were

reprocessed resulting in pure tritium, untritiated exhaust gases, and a small amount of tritiated impurities such as methane. This pure tritium then rejoined the tritium supply, so that an accurate measurement (better than 1%) of the total deficit could be made. Because this measurement was made in exactly the same way before, during, and after the DTE1, the inventory measured by this method does not suffer from systematic calibration errors. This cycle was repeated weekly, since there were only 20g of tritium on site, and in a few days operation nearly all the tritium was supplied to the NBI systems and torus, requiring reprocessing of the exhaust gas before operation could continue. Of the 100g total tritium delivered by the AGHS, 65% went to the NBI system at Octant 8 (Fig.1).

Also shown in Fig. 5 is the in-vessel inventory calculated by scaling the PTE data. Unlike the plasma tritium fraction, the inventory is very different in DTE1 compared to the expectation based on the PTE. The tritium inventory in DTE1 could only be reduced by a factor of about 2 by pulsing in deuterium whereas in the PTE a much more complete recovery was achieved. By the end of the MkIIa divertor campaign, about 6g of tritium remained in the torus, and was being reduced by only 10mg/day.

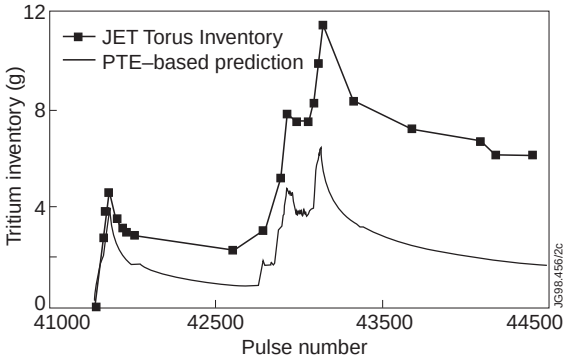

Fig.5: Tritium inventory in the torus during DTE1 measured as the difference between the 20g of tritium on site and the amount measured to be in the Active Gas Handling System. The expectation based on the PTE is also shown.

The tritium inventory is also shown in Fig. 6, but as a function of the cumulative tritium input to the torus. During operation with tritium gas feed the inventory increases at about

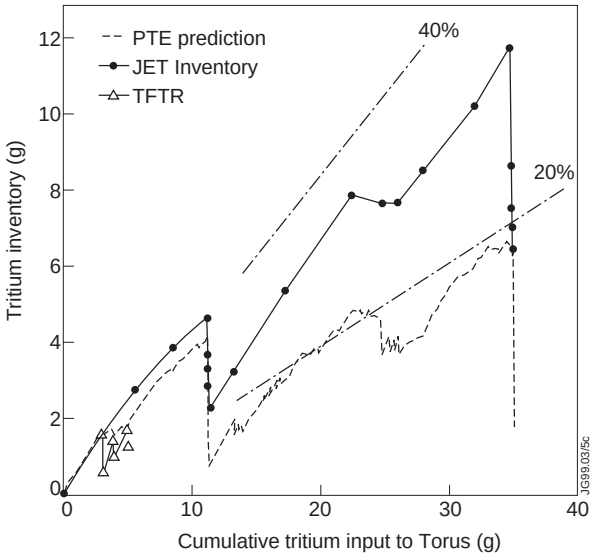

Fig.6: Tritium inventory in the torus versus cumulative tritium input to the torus. The TFTR experience [14] is shown for comparison.

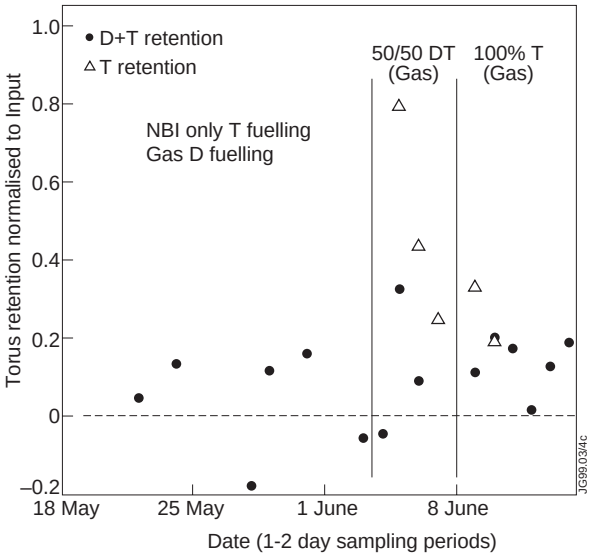

Fig.7: Tritium and total gas retention during initial period of heavy gas fuelling with tritium. Retention is based on gas supplied minus gas recovered to AGHS.

40% of the input rate, but after clean-up eventually settles down to 17% of the input. Total gas balance measurements, Fig. 7, indicate that about 85% of the input gas is recovered, ie. there is 15% retention of the total input. This is consistent with the lowest levels of tritium retention, also shown in Fig.7. When fuelling switches from D to 50/50 DT, initially the gas recovered is mostly D from the previous operation, but with continued 50/50 DT fuelling the exhaust gas approaches the isotopic composition of the input. When the fuelling is then switched to 100% T, the exhaust is initially 50/50 DT, so once again the exhaust tritium fraction lags behind the input tritium fraction. This lag is due to exchange with the hydrogenic wall inventory.

4. RESULTS OF SPECIFIC CLEAN-UP TECHNIQUES

4.1 Clean-up by Pulsing in Deuterium

Different types of D plasmas were compared in terms of cleaning efficiency. Neither glow discharge plasmas nor ECRH plasmas were found to be particularly effective at removing tritium compared to tokamak plasmas. Pulses with RF heating removed significantly more tritium than ohmic pulses, without excessive DT neutron generation (Fig. 8). It was also found that varying the strike point position of the plasma scrape off layer on the divertor ('sweeping') increased the release of tritium.

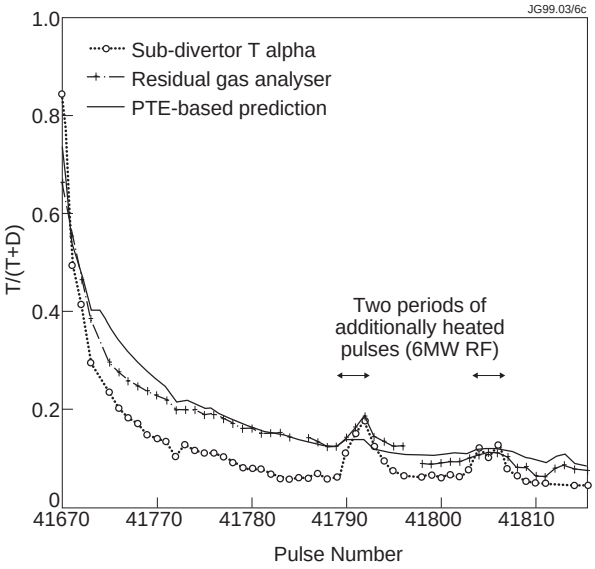

Fig.8: Effect of additional RF heating during deuterium clean-up pulses on the tritium fraction observed in the divertor region and on the tritium faction measured in the Residual Gas Analyser sampling the post pulse exhaust stream. In each case the two periods of RF heated pulses were three pulses in length.

Both the RF heated and Ohmic pulses are run with a density feedback system, requesting the same density throughout. In the Ohmic pulses, a D fuelling rate of ~ 2×10^{21} D/sec is required to maintain a total plasma content of about 2×10^{21} D. However, when the RF is switched on, the density rises above the requested level, and the external gas fuelling is switched to zero (Fig.9). It is not surprising then that the plasma tritium fraction increases with RF heating, when the plasma is being fuelled entirely by gas from the tritium contaminated wall, whilst in an Ohmic pulse, the plasma content is diluted by strong external fuelling of puredeuterium. Nevertheless, the tritium fraction of the exhaust gas is also higher for the RF heated pulses. Since the total amount of exhaust gas for the Ohmic and the RF pulses is the same, the tritium removal for the RF pulses is indeed higher. Note also that there is a 'memory' of the higher T/(T+D) ratio in the plasma, and the exhaust stream, which lags for a few Ohmic pulses after the RF pulses.

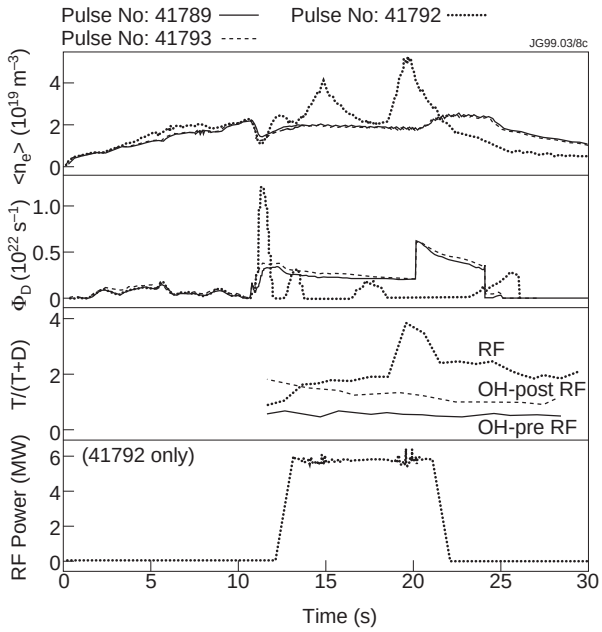

Fig.9: Plasma density, D fuelling rate, and plasma T fraction for Ohmic and RF heated deuterium clean-up pulses. The RF pulse shown (41792) is the third of a series of RF pulses (41790, 91 and 92). Note how the observed plasma tritium fraction is elevated in the OH pulse following the RF pulses although the plasma density and (deuterium) fuelling are identical to the pre-RF case.

4.2 Clean-up by Deuterium Gas Soaks

Tritium removal was also attempted using deuterium gas 'soaks' of the torus. These soaks took place during overnight periods following regeneration of the torus cryopump. The deuterium purges were performed on the first and second day of tritium clean-up following the phase I of DTE1 before the two month break in operation to repair the NBI systems. The torus pressure was held at 1 Pa pressure of deuterium for about 25 minutes for each purge. There were a total of ten purges, one having a higher pressure, 10Pa, and one being significantly shorter with only seven minutes duration. The ten-fold increase in pressure resulted in only a two-fold increase in the amount of tritium recovered. The reduced soak time resulted in a tritium recovery reduced by about a factor of 2.

The total amount of tritium released by the two evenings of purges was ~ 0.1g. This is 10× lower than the amount of tritium removed by pulsing on the two days of operation which followed the purges. In the end, the tritium removal by neutral gas purges was much less efficient than that by pulsing, both in terms of time and in terms of deuterium gas usage. Gas usage is a concern because time is required for the AGHS to process all the exhaust gas.

The inefficiency of the gas purges was inconsistent with our experience from the PTE [7]. In the PTE it was found that the isotopic fraction of the purge gas and the exhaust gas from pulsing was similar. This PTE result is now thought to be an artefact resulting from contaminated ion chambers. During DTE1 it was found that after ion chambers had been exposed to gas streams containing impurities, they would consistently read too high. At the time of the DTE1

purges, the ion chambers had not yet been exposed to impurity gas streams resulting from full warm-up of the divertor cryopump. In the PTE, however, the ion chamber used had already been exposed to the impurities from the full warm-up of the NIB cryopump.

Other tokamaks have investigated hydrogenic gas purges as a hydrogen isotope removal technique. Experiments on both TFTR [14] and TEXTOR [15] also indicated that gas purges were totally ineffective.

4.3 Clean-up GDC and ECRH discharges

Tritium removal by deuterium Glow Discharge Cleaning (GDC) was attempted during DTE1. In general, GDC was employed sparingly throughout DTE1 because of the heavy gas load. The GDC conditions were the same as normally used in JET: ~ 2×10^{-1} Pa D_2 at a flow rate of 1.5 $Pa.m^3/s$, with a total current of 16A. A total of 4.5 h of GDC was performed 2-4 days after the first phase of DTE1. Comparing the same gas flow with and without the glow current indicated that the GDC was increasing the rate of tritium removal. GDC attempted after the PTE [7] showed no measurable tritium release above the background outgassing, but had only been attempted after two weeks of deuterium pulsing. Nevertheless, the absolute amount of tritium removed after DTE1 by this method was < 40mg, and the tritium fraction was > 10× lower than for deuterium pulsing over the same period.

Tritium removal by Electron Cyclotron Resonance Heating (ECRH) discharges was also attempted during DTE1. Such discharges had previously been used on JET as a torus conditioning technique similar to GDC. For JET, ECRH presented the advantage of significantly lower gas throughput than GDC. ECRH is also being considered as a discharge conditioning and detritiation technique for ITER, where turning off the toroidal field to perform GDC is inconvenient. The JET ECRH plasmas were employed one week after the final tritium pulses (end of Phase II). A total of 2.5 h of ECRH took place with the following conditions: : ~ 6×10^{-4} Pa D_2, continuous 0.1-0.16 T toroidal field and up to 17kW of power supplied by the JET lower-hybrid current drive (LHCD) launcher. The results were very similar to the deuterium GDC. There was no significant tritium removal, and no change in the tritium fraction during subsequent tokamak pulses.

4.4 Tritium recovered by warming up in-vessel LN_2-cooled surfaces

During DTE1, the divertor cryopump helium panels were routinely regenerated to release their hydrogenic inventory. Any water molecules or heavier hydrocarbons would remain adsorbed on the LN_2-cooled cryopump shield (methane molecules have a sufficiently high vapour pressure that they would be released during the routine regenerations). A full warm-up of the cryopump, both LN_2 and ScHe circuits, occurred on two occasions during DTE1, one after each of the two phases of tritium operation after the tritium fraction had fallen to ~ 1%. The tritium recovered from these two regenerations via tritiated water and hydrocarbons was about 0.1g and 0.25g respectively. Since the total tritium fuelling to the torus was 11.5g and 23.5g during the two

periods, the amount of tritium in tritiated compounds condensed on LN2 surfaces amounted to ~ 1% of the torus tritium input in each case.

4.5 Tritium recovered by baking in-vessel water-cooled surfaces

At the end of the MkIIa campaign, 6g of tritium was still held up in the torus and the NB systems, with > 90% thought to be in the torus. Anticipating that the torus inventory was mostly in the cold regions of the divertor (shadowed from plasma contact), the cooling water was drained from the circuit after the final day of tokamak operation while the vessel was still under vacuum and being heated. The divertor support structure (Fig. 3) reached about 135°C, significantly higher than the 40°C level maintained throughout the DTE. The gas desorbed during this exercise was about 1.8% tritium, which is much richer than the 0.1% tritium composition of plasmas in the final weeks of operation (Fig. 4). However, the total amount of gas released was small, amounting to only 6mg of tritium released.

The fact that the released gas was richer in tritium than the torus exhaust at the end of the campaign suggest that the heated region contains a reservoir of tritium which was not fully diluted by the operation in deuterium which followed DTE1. Indeed, the tritium fraction of the gas released by heating the divertor, 1.8% T, is comparable to the tritium fraction of the torus fuelling during the whole of MkIIa campaign, ~ (14 Pa.m^3 tritium)/(600 Pa.m^3 total) = 2.3%.

4.6 Tritium recovered during vessel venting

After warming up the divertor support structure, the vessel was vented in a series of five gas purges of increasing pressure, pumping out the torus after each purge. The first two purges were done with nitrogen at 350Pa pressure, while the vessel was at 150°C and lasted four hours each. This was followed by three air purges with the vessel at 120°C, at pressures of 10 , 25 and finally 90kPa, and durations of 7, 12 and 4 hours respectively. The total amount of tritium recovered by this exercise was estimated to be ~ 0.6g, ie. 10% of the inventory at the end of plasma operations.

The vessel was then cooled to room temperature, and air continuously drawn through the torus at an average flow rate of 350m^3/hr for the next four months. During this period (the 'RTE shutdown') tiles were removed from the torus [16], and air purging the tile storage units was combined with the torus air in the exhaust stacks; then the tritium was removed by catalytic conversion to tritiated water in the AGHS Exhaust Detritiation System (EDS) [4]. The total tritium emanation rate decreased over the course of the shutdown, down to about 10mg/day. At the end of the four month period the total amount of tritium collected, including the above purges, was 2.0g, ie. about 1/3 of the tritium inventory at the end of plasma operation.

Prior to venting the vessel, the rate of tritium release was estimated by extrapolating tritium release results from previous campaigns. Release of fusion product tritium [17] and retained tritium from the PTE indicated an initial release of about 10% of the inventory, and a chronic evolution rate of about 0.01% of the initial inventory per day. While the initial release upon venting agreed with the extrapolation, the chronic release rate was ~ 20× higher. Figure 10

shows the source rate of tritium release during air exposure normalised to the initial tritium inventory for the initial vent following DTE1 compared to the other campaigns.

The reason for the enhanced steady release rate of the tritium per unit inventory upon ventilation after DTE1 is not clear. One possibility is the temperature history of the tritium retained in the vessel. The majority of the tritium inventory at the end of the DTE1/ MkIIA campaign was believed to be tied up in codeposited films on the inner corner of the divertor: on the louvres, the divertor support structure and shadowed portions of the tiles (see Fig. 3). Unprecedentedly thick carbon films were found in this region during an inspection of the MkIIa divertor just prior to DTE1 [18].

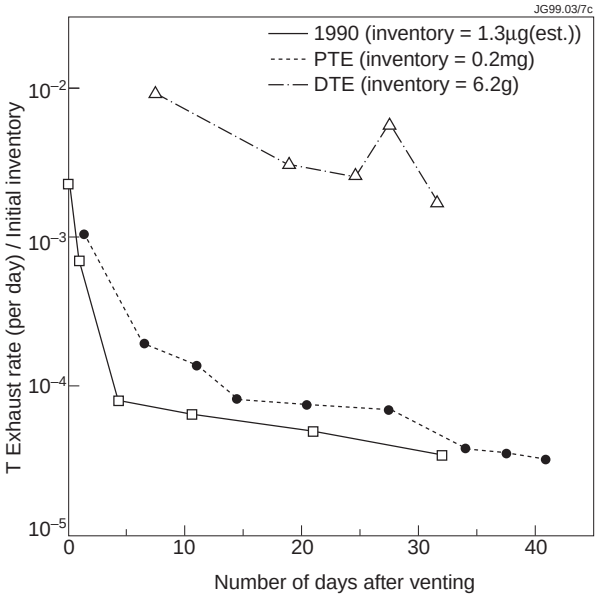

Fig.10: Rate of tritium release during air exposure of the torus at ambient temperature. The 1990 torus inventory comes from tritons born in DD fusion reactions in the JET plasmas.

These regions would have been ~ 40°C during the co-deposition of tritium and carbon. Tritium in other parts of the vessel, and tritium retained in the campaigns prior to the installation of the divertor cited above, would have been deposited at ≥ 300°C. Films formed at the higher temperature might be more stable to detritiation by exposure to humid air. During the removal of tiles after DTE1, the tritium release from individual tiles could be measured. Tiles from the inner divertor were found to have outgassing rates per unit area > 10× higher than tiles from other parts of the vessel.

4.7 Summary of tritium clean-up techniques

The tritium clean-up techniques can be grouped into two categories: 1) techniques employing deuterium or hydrogen which do not decondition the vessel, and 2) techniques involving venting of the torus.

Of the deuterium techniques, only tokamak pulses were found to be effective. It was found that additional heating of the plasma with ICRH accelerated the tritium release. It was possible to reduce the plasma tritium fraction to < 1% in about four days, and ultimately to reach a level where the DD neutron rate was larger than DT neutron rate in a few weeks. However, it was not possible to reduce the tritium inventory by pulsing to below ~ 17% of the tritium input to the torus. This persistent inventory appears to be largely in the form of codeposited films which were not significantly affected by operation in deuterium.

Ventilation of the torus at the end of the campaign for a four month period resulted in about 1/3 of the remaining tritium inventory being recovered. The high release rate of tritium to the air is attributed to the tritium-rich films formed at the inner corner of the divertor. During this

period the MkIIa divertor tiles were removed by remote handling; a significant fraction of the torus inventory was removed along with these tiles.

5. DISCUSSION

In section 3 the plasma tritium fraction and tritium inventory in DTE1 were compared to extrapolation of the PTE exhaust results per unit gas input. While the plasma tritium fraction was similar to the extrapolation, the difference between measured and predicted inventories grew throughout the period of tritium fuelling. Together, these two observations suggest that there is an irrecoverable sink for tritium in the JET configuration during DTE1, not present at the time of the PTE.

One difference between the two campaigns is the use of NBI to fuel all the tritium in the PTE. This is not believed to be an important factor in the different tritium retention behaviours for the two campaigns. Firstly, the dependence of the isotopic fraction in subsequent discharges was found only to depend on total gas input to a discharge, independent of whether particles were injected beams or puffed gas. This was observed in DTE1 for tritium, and by comparing T beam injection during the PTE with H gas fuelling just prior to the PTE [19]. Furthermore, the total tile D inventory in campaigns after the introduction of beryllium evaporation, but before the installation of the pumped divertor was only ~ 2% of the deuterium gas fuelling during the campaign [20]. This agrees with the tritium gas balance of the PTE (~ 3%), but is much less than the tritium and total gas balance in the MkIIa divertor campaign.

Instead, the difference in tritium retention between the two campaigns is thought to be due to the geometry and temperature of the divertor. The MkIIa divertor, with pumping slots in its corners, has a large surface area which is in line-of-sight with the plasma contact region, but which is itself shadowed from erosion by the plasma. This geometric effect is compounded by the temperature of these shadowed corners: they are generally at ~ 40°C. Films codeposited on these relatively cold surface are estimated to have D/C up to ~ 0.8 D/C [9], compared to < 0.1 D/C for films deposited on other tiles which have a minimum temperature of 300°C [21]. Interestingly, these thick films were only observed to form on the inboard leg of the divertor. The reasons for this are not yet fully understood.

Even though we might have a qualitative explanation for why tritium retention is so much higher in DTE1 than the PTE, the absolute level of retention is not understood. The total amount of redeposited carbon is ~ 2% of the total ion flux to the divertor, or ~ 4% of the ion flux to the inner divertor [18]. If the carbon films were the result of sputtering by ions at the target, one would require either a very high probability of sputtered carbon to be transported to the corner, or a very high sputtering yield. However, the inner divertor corner only subtends < 15% of the solid angle about the inner strike point, and from chemical sputtering data [22], the total carbon yield would be expected to be < 5% (C sputtered per incident D^+ or T^+). Together these give an

upper limit of $0.05 \times 0.15 = 0.75\%$ which is much less than the observed 4%. Computer simulations with the REDEP code [23] are currently being made to address both the magnitude and asymmetry of the carbon redeposition observed in JET.

From the level of tritium retention in the MkIIa divertor, one would expect that if all JET pulses were fuelled with 50/50 DT mixtures, ~13mg of tritium would be retained per discharge [24]. Assuming that this retention is due to a steady rate of carbon erosion during the ~ 10 second divertor phase of the discharge, this translates to 40 kg tritium retention in a plasma burn lasting one year.

For comparison, the ITER design has a limit of 1kg of tritium in the torus [25]. A relatively small machine like JET would already have an impractically high level of tritium retention. If ITER were simply a large version of JET, one would expect the problem to get even worse due to the larger surface area. JET, however, was not designed to minimise tritium inventory. A next generation fusion device would have to be engineered to avoid either the large erosion, or the large codeposition. The difference between the JET DTE1 and PTE campaigns suggests that geometry and temperature can have a major influence. Another engineering approach is the choice of materials. However, while the elimination of carbon would go a long way to reducing the level of tritium retention in a tokamak, carbon is regarded as indispensable with regards to handling the high transient thermal loads during a disruption [26].

6. CONCLUSIONS

In the 1997 JET DT campaign, 35g of tritium were fuelled to the torus. The in-vessel inventory reached 11.5g by the last tritium pulse, and was reduced to 6g during the subsequent plasma operation in deuterium.

Tokamak pulsing was the only successful detritiation technique which would not result in deconditioning the torus vacuum and surfaces. Ventilation of the torus with air during the four month Remote Tile Exchange which followed the campaign resulted in the release of a further 2g of tritium.

The level of tritium (and deuterium) retention was much higher than in previous campaigns. This is thought to be the result of relatively cold, shadowed regions in the inner corner of the MkIIa divertor where unprecedentedly thick and deuterium/tritium rich films were observed to form.

The rate of tritium build-up in JET is already too high for a next generation fusion device. Such a machine will have to choose plasma facing component geometries, materials and temperatures so as to minimise tritium codeposition and to minimise hydrogenic content of deposited films. The inferences gained from the JET experience, and the ongoing analysis of the tritiated JET tiles, will be key inputs in aiding such choices.

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