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Measurements of tritium retention and removal on TFTR.

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Abstract

Recent experiments on the Tokamak Fusion Test Reactor (TFTR) have afforded an opportunity to measure the retention of tritium in a graphite limiter that is subject to erosion, codeposition and high neutron flux. The tritium was injected by both gas puff and neutral beams. The isotopic mix of hydrogenic recycling was measured spectroscopically and the tritium fraction T/(H+D+T) increased to as high as 75%. Some tritium was pumped out during the experimental run and some removed in a subsequent campaign using various clean-up techniques. While the short term retention of tritium was high, various conditioning techniques were successful in removing ≈ 8,000 Ci and restoring the tritium inventory to a level well below the administrative limit.

1. Introduction

Graphite is one of the primary candidate materials for first wall components in fusion devices due to its high sublimation point and the low radiative power losses of carbon impurities in a high temperature plasma. An important design issue is the trapping of hydrogen isotopes in graphite which could raise the in-vessel tritium inventory to unacceptable levels unless an effective removal technique is implemented. TFTR has a special opportunity to study the issues involved since it has a graphite limiter which experiences erosion, codeposition and neutron flux from DT plasmas with core fusion power similar to that expected for ITER. Below 1000 °C, retention in graphite is characterized by saturation of the near surface region and by surface diffusion on internal porosity.[1-3] A study of deuterium retention, based both on modeling and measurements on graphite tiles and wall coupons removed from TFTR, was reported in reference 4. The overall fractional retention of deuterium in TFTR was found to be 45%. Of this, 22% was on the plasma facing surface of the bumper limiter, 9% on the sides of the gaps between the limiter tiles and 14% on the vacuum vessel wall. Comparison of different run periods showed that the retention varied in the range 22% - 63% and this variation was correlated with the average neutral beam power. This pattern was attributed to increased edge and limiter temperatures leading to more codeposition at high beam powers. A stronger plasma-limiter interaction at high powers is also indicated by the higher level of CII 658 nm emission. The fate of trace tritium produced in DD nuclear reactions was also studied. Tile analysis[5] indicated that ≈35% of the tritium produced was retained in the internal components of TFTR.

Retention issues have also been studied on JET. During the preliminary tritium experiment, 54 Ci of tritium were injected into the vessel in two discharges. Following this, a series of experiments were performed to remove tritium and study the efficacy of the different cleaning procedures.[6] High density disruptive discharges were found to be the most successful plasmas for tritium removal. A simple ‘soak’ with deuterium gas was also effective, however, helium discharges resulted in low tritium release from the vessel walls. A recent appraisal of deuterium retention in JET[7] indicated that 40% of the gas input remains in the vessel at the end of the day.

In section 2 we describe the TFTR vacuum vessel and neutral beam systems and, in section 3, tritium operations. Recently, tritium gas puffing was used to generate L- mode plasmas for transport and RF experiments and the corresponding increase in the tritium inventory of the limiter is described in section 4. Section 5 describes the subsequent campaign to remove tritium and ascertain the amount of remaining tritium that was tenaciously held and would not be released in the event of a credible vacuum accident. A summary discussing TFTR, JET and future DT machines concludes the paper.

2. Experiments on TFTR

Since 1993, TFTR has been operated routinely with deuterium-tritium plasmas. Fusion powers up to 10.7 MW have been obtained in ‘supershot’ plasmas heated by high power tritium and deuterium neutral beam injection.[8] At these levels, the fusion power density in the core, ~ 2 MW/m^3 as well as many alpha parameters are comparable to those expected for the core of future ignition machines such as ITER. TFTR plasmas have a circular cross section of minor radius typically in the range 0.8 - 0.96 m and a major radius 2.45 - 2.62 m. The plasma boundary is defined by an inner toroidal belt limiter composed of carbon composite tiles in high heat flux regions as well as graphite tiles, both supported by water cooled inconel-718 backing plates. The limiter extends over ±60° from the midplane and is divided into 20 sectors, each composed of 24 rows of tiles, four tiles wide, with each tile covering about 4.5° toroidally and 5° poloidally. Each sector is curved and the center extends out about 5 mm from a true toroidal surface. The total area of the limiter is 22 m^2 of which approximately 7 m^2 contacts the plasma. The limiter experiences erosion, codeposition and neutron flux from the DT plasmas. After exposure to many plasma discharges each limiter sector develops a poloidally asymmetric 'footprint' or eroded area, surrounded by areas of net codeposition.[9] The pattern is related to the angle of the incident flux on the individual tiles and the slight toroidal asymmetry of the limiter.[10] The rise in bulk limiter temperature remains below 50 °C during a discharge,[11] however infra-red camera measurements indicate that the surface temperature in localized hot spots increase up to 1000 °C or more depending on the discharge conditions and auxiliary heating power.[12]

The neutral beam system comprises four neutral beam lines each having three beam sources that can independently operate in either deuterium or tritium in a given discharge. The fueling rate of the neutral beams is computed from the extracted ion current, the neutralization efficiencies[13] of the three components of the beam,[14] and their transport efficiencies through the beamline. The amount of tritium supplied to an ion source for a 1 s tritium pulse is approximately 250 Ci. Of this amount, approximately 4% is injected into the torus as energetic neutrals. The remainder, including un-neutralized ions initially collected by copper beam dumps, is pumped on liquid helium cryopanels inside the beamlines. Even though the beamlines are differentially pumped, there is a small leakage of cold gas into the torus. This throughput is estimated from the neutral beam drift duct conductance and the tritium pressure in the beamline. The duct conductance for deuterium was found to be 46 m^3sec^{-1} by measuring the pumping speed of the torus when it was pumped solely by one beamline. A mass correction was applied to yield the tritium conductance. Pressures of ~10^{-6} torr are typically developed at the beamline end of the duct when three ions sources are pulsed. Cold gas throughputs of 0.1 torr-l/s per ion source are thereby estimated. Cold gas fluence is approximately 6% of that of the energetic particles.

3. Tritium operations

A detailed account of tritium processing and management on TFTR is given in reference 15. Tritium gas is brought on-site in an approved shipping canister and transferred to a uranium bed where it is stored. The uranium bed is heated to transfer the gas to the neutral beams or torus gas-injection system. Tritium is released from calibrated volumes by piezo-electric valves that are actuated for a short time during the discharge. The quantity of tritium injected is measured by monitoring the temperature and pressure of the calibrated volume with an experimental uncertainty of 2%. Tritium is injected into the torus by the neutral beam system or puffed in directly. Tritium exhaust is pumped by the liquid helium cooled cryopanels in the beam boxes and/or by turbopumps in the torus vacuum pumping system. After completion of approximately two days of plasma operations, the neutral beam cryopanels are warmed up to a temperature where the hydrogenic species are evaporated. The gas is transferred to one of two $7m^3$ gas holding tanks for inventory measurements and subsequently is oxidized and absorbed in molecular sieve beds. These beds are shipped off-site for reprocessing or burial. Recently a tritium purification system was installed that will allow the tritium system to be operated closed loop.

The tritium inventory in the gas holding tanks is measured by parallel plate ion chambers (Femto-Tech) in a sample loop that is maintained at a pressure of 740 torr. These are calibrated in-house by adding a calibrated volume of tritium at a known pressure and temperature to the gas holding tanks.[16] A quadrupole mass spectrometer analyzes the gas content and the ion chamber response is corrected for the differing work functions of impurity gases. The quadrupole mass spectrometer is itself calibrated by measuring the response to various standard gases. In this way the ion chamber response is calibrated over a gas holding tank range from 200 to 11,000 Ci. The measurement uncertainty depends on the inventory in the gas holding tanks and is typically in the range 5% - 7%. Since 96% of the tritium used by the neutral beams is pumped by the cryopanels without entering the torus, the amount injected into the torus is of the same order as the uncertainty in the measurement of tritium recovered from the cryopanels. Hence there is a relatively high level of uncertainty in the day to day retention of tritium in the torus. This limitation was circumvented for some of the measurements described below by reserving a neutral beam line solely for tritium exhaust measurements.

At the beginning of the DT campaign in TFTR, 360 Ci of tritium was injected into the torus by 21-DT neutral beam heated discharges (most of which used only one tritium beam source) interspersed with 55 DD pulses. Following this, a short clean-up experiment was performed[17] in which 28 ±50 Ci were recovered after a series of 34 deuterium ohmic discharges and deuterium neutral beam injection discharges. Although the measurements had a relatively high level of uncertainty, they indicated a high short-term retention level.

The tritium accounting system in place on TFTR provides a global measure of all tritium supplied to and recovered from the torus and neutral beam systems. By the end of August, 1995, a total of 432,000 Ci had been supplied to the neutral beam systems of which 18,000 Ci entered the torus (the neutral beam fueling is computed as described in section 2 and includes the cold gas contribution). In addition, a trace amount (203 Ci) of tritium was puffed directly into the torus in experiments to study transport. Tritium exhaust is collected on the neutral beam cryopanels and also pumped by the torus pumping system. The neutral beam cryopanels are regenerated into gas holding tanks which also collect the effluent from the torus pumping system. The difference between the total tritium delivered to the torus and neutral beam systems and that recovered in the gas holding tanks was approximately 7,000 Ci. Comparing this to estimates of the total tritium injected into the torus by the neutral beam systems yields an estimate of the long term retention of approximately 40%. This value is consistent with the earlier deuterium measurements reported in refs. 4, 5. The administrative limit on the in-vessel inventory is 20,000 Ci.

4. Tritium injection and removal during L-mode experiments.

In September 1995 experiments were performed in L-mode plasmas on the isotope scaling of L-mode transport[18] and on RF heating at the tritium 2nd harmonic cyclotron frequency.[19] To generate L-mode plasmas, tritium gas was puffed directly into the discharge by the torus gas injection system. In addition, tritium (and deuterium) neutral beam injection was used to heat the plasma. These experiments afforded an opportunity to measure tritium retention with significantly reduced experimental uncertainty. The amount of tritium puffed into the torus by gas injection system in September was 10,600 Ci, much higher than the 203 Ci puffed in the earlier tritium transport experiments. Recycling from the limiter increased with the tritium gas injection. The isotopic mix of hydrogenic isotopes recycled from the limiter is measured spectroscopically by measurements of the H_α, D_α, and T_α emission lines.[20] Prior experience with tritium neutral beam injection and trace tritium gas puffs was that the fraction of T_α was low, even with tritium only neutral beam injection. The maximum fraction of $T_\alpha/(H_\alpha+D_\alpha+T_\alpha)$ observed previously was 9% after 8 discharges heated by tritium only neutral beam injection.[21] During the L-mode experiments, the quantity of tritium injected into the torus by gas puff was larger than that injected by the neutral beams. With the first tritium puff discharge, the fraction of tritium in the recycled hydrogen isotopes increased from below 3% up to 22%. However the T_α fraction was slow to change further with subsequent discharges.

Figure 1 shows the H_α, D_α, and T_α spectral profile in a discharge with a tritium puff and RF heating and, for comparison, a similar discharge without tritium injection. The line profiles are Doppler broadened and asymmetric. The maximum T_α fraction reached was 75% indicating that a longer period is needed for complete exchange of deuterium and tritium in the region of the limiter that is accessed by the plasma. Later in September, these experiments were followed by a conditioning campaign to restore a low limiter recycling state for supershot experiments.[22] This

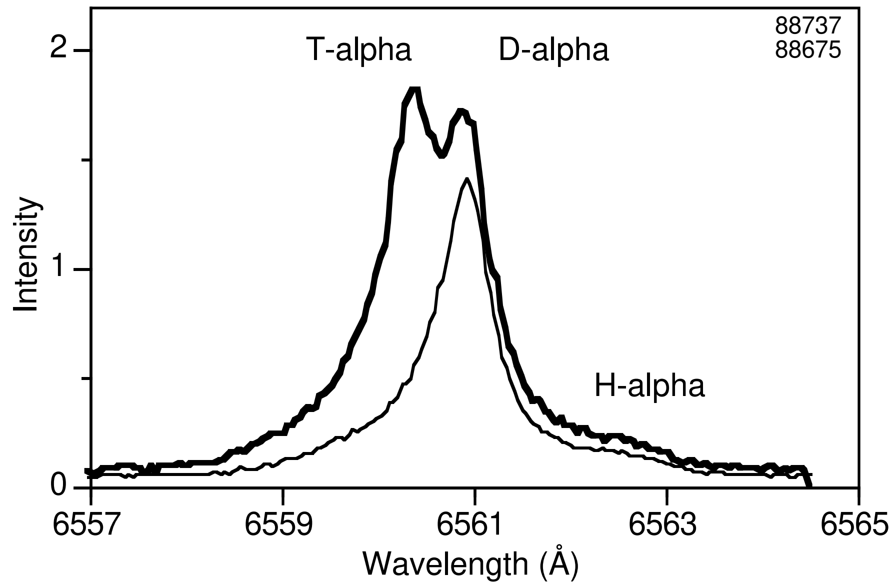

Fig 1. Balmer alpha emission showing the presence of all three hydrogen isotopes. For comparison the spectral profile from an earlier discharge without tritium is shown.

incorporated high power neutral beam heated discharges followed by discharges with lithium pellet injection[23] to reduce the edge influx.

To avoid the difficulty of measuring the tritium exhaust from the torus in the presence of the 25 times higher amount of tritium used by the neutral beams, one of the four neutral beam lines was reserved solely as a pump for the tritium exhaust and had no tritium gas feed. The torus pumping system was closed off and the torus interface valves to the neutral beam boxes left open so that the cryopanels in the four neutral beam boxes pumped the torus. The duct conductances of the four beamlines are identical so that 1/4 of the tritium was pumped by the reserved beamline. The cryopanels were regenerated approximately every second day and a measurement made of the tritium recovered. The total tritium exhaust from the torus was estimated by multiplying that recovered by the reserved beamline by four.

The results are shown in Table I and Fig. 2. In the table, the second column refers to the plasma operations on a particular day in a highly abbreviated way: for example ‘DT L-mode’ means L-mode experiments using both deuterium and tritium, ‘He4 GDC’ is glow discharge cleaning using ^{4}He, ‘D supershot’ is supershot plasma operations with D only, ‘PDC’ is pulse discharge cleaning. The next 2 columns give the quantity (in curies) of the total tritium injected via gas puff, and neutral beam injection on a particular day. The ‘NB vent’ refers to tritium recovered from the neutral beam boxes and ‘various’ refers to miscellaneous pumps and purges over the period 6 October-17th November. The following two columns give the tritium recovered from

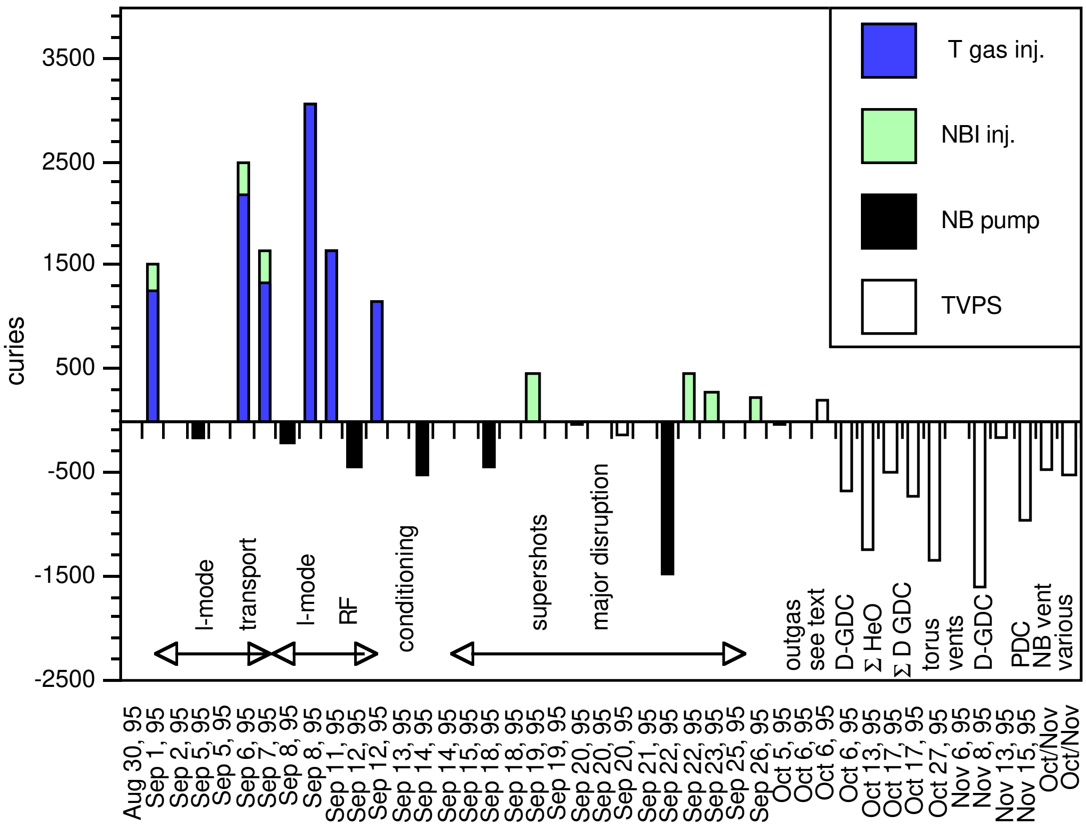

Fig. 2 History of tritium injected into the torus and measured exhaust during September/October 1995. GDC represents glow discharge cleaning. The symbols Σ D-GDC, Σ HeO GDC are the sum from several days -see fig. 3. In the legend 'TVPS' refers to the torus vacuum pumping system, 'NB pump' to tritium recovered from the neutral beam cryopanels , 'NB inj.' is tritium injected via the neutral beams (including cold gas), and 'T gas inj.' is tritium gas directly puffed into the torus

the neutral beam cryopanels and via the torus vacuum pumping system. A running total of the cumulative input and exhaust, the in-vessel inventory and the fractional retention is calculated in the remaining columns. The fractional retention is defined as ratio of the cumulative tritium injected into TFTR during September-November 1995, less the cumulative exhaust; divided by the cumulative tritium injected. Note that the release of tritium is continuous but it was measured periodically, i.e. a 'NB pump' entry reflects tritium accumulated since the previous cryopanel regeneration.

On 30th August 1995 the value of in-vessel inventory, given by the tritium accounting system, stood at 6,863 Ci. Over the next 12 days 10,600 Ci were injected by tritium gas puff and 900 Ci by neutral beam injection while an estimated 860 Ci were recovered indicating a short term retention above 90%. It should be noted that no attempts were made to remove tritium in this period, on the contrary, the aim was to increase the tritium concentration in the plasma as much as possible. A 50 minute duration helium glow discharge, used to speed recovery from a disruption, yielded negligible (<20 Ci) tritium exhaust. The following period of plasma operation, from 12th-18th September 1995, was aimed at restoring the limiter to a low recycling state by ohmic and neutral beam conditioning. Carbon blooms[24] were encountered when the neutral beam power was raised, however, over time the threshold for blooms increased. Although no further tritium was injected into the plasma either directly or by neutral beams, the tritium content of the plasma remained high for several days in this period. Tritium continued to be present in the exhaust and an estimated 970 Ci was collected. In the period 19th-26th September an additional 1,370 Ci was injected via the neutral beams, however no further tritium gas puffing was used. On 22nd September an unplanned 1.2 MA disruption occurred. Subsequent measurements of the tritium recovered by the neutral beam cryopanels yielded an unusually high value of 1,500 Ci, most of which was presumably released during the disruption. For the days 22-26 September all of the neutral beams (including the reserved beamline) used tritium, and the tritium exhaust from the torus could not be included in the above analysis (it is, of course, included in the global tritium accounting system described in section 3). Overall, from 1st September until the end of the run period on 26th September, a total of 12,900 Ci was injected by gas puff and NBI and 3500 Ci recovered, a retention fraction of 73%. Summing the various tritium transfers cited above yielded an in-vessel inventory of 16,243 Ci. This was consistent with the global measurements of the in-vessel inventory of 16,440 Ci and a nominal adjustment of 197 Ci was added (row: 6 October 1995 in Table I) to reconcile the difference.

5. Tritium removal.

There was no activity over the next 8 days, outgassing of the torus yielded only 36 Ci. Results from JET had indicated that a deuterium ‘soak’ was effective in removing tritium from the torus that was maintained at 300° (Ref. 6). There was interest in measuring the efficacy of this technique for the ambient temperature TFTR limiter. Deuterium was added with the pump valves closed and the torus pressure was raised to 70 mtorr for one hour. However negligible tritium was recovered (5 Ci ± 9 Ci) indicating that this technique was ineffective in TFTR. Next a glow discharge sequence was undertaken and the results are shown in Fig. 3. The first deuterium glow discharge (D-GDC) removed 687 Ci over 4 hours, an average rate of 172 Ci/hour. The tritium content of the gas holding tank was continuously monitored over this period and indicated an initial removal rate of approximately twice this value, declining by a factor 4 over the 4 hours. The following glow discharge used 90% helium and 10% oxygen and removed ≈ 50 Ci/hour. This rate did not decline. To obtain more information about the time dependence of the D-GDC removal rate, the next D-GDC was performed in two steps. The initial D-GDC removal rate was higher than at the end of the previous D-GDC. Continued D-GDC, after the vessel vent (see below) and bakeout, showed a removal rate declining to 10 Ci/hour.

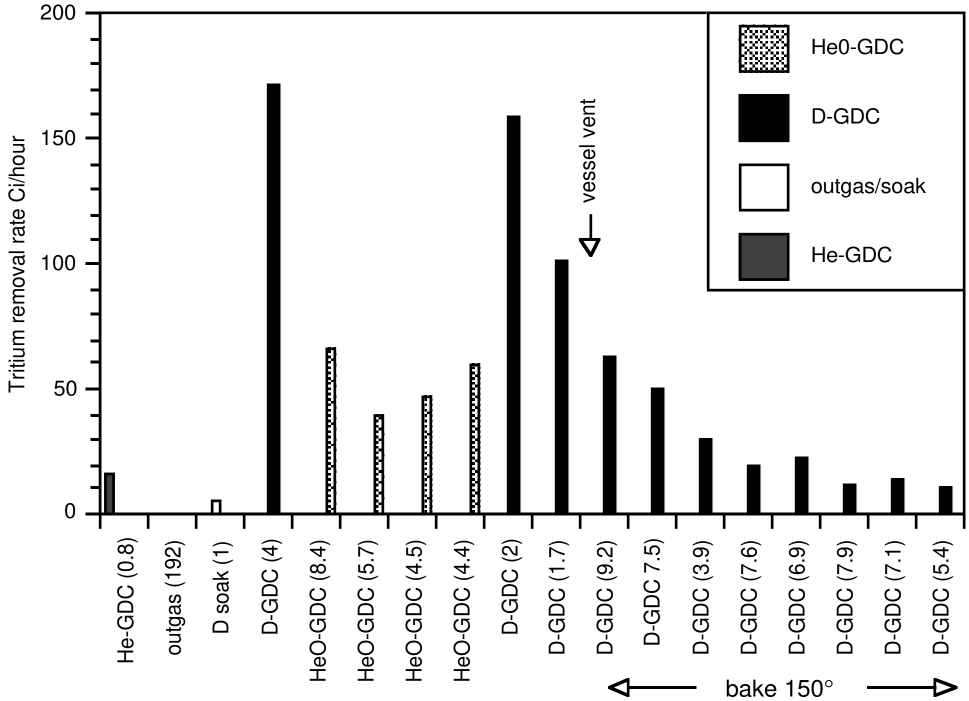

Fig 3. Tritium removal rate during outgassing, D soak and He-, D-, and HeO glow discharges. The data are arranged in chronological order. The duration in hours is given in parentheses.

Plasma ions in a glow discharge will interact with the limiter to a depth given by the range of the plasma ions in graphite, typically 1-10 nm. Since tritium is buried by codeposition to a greater depth than this, deuterium or helium glow discharges were not expected to be effective in removing subsurface tritium. On the other hand a helium-oxygen glow is not limited in this way since it chemically reacts with the graphite and removes it in the form of carbon monoxide and carbon dioxide. The higher D-GDC tritium removal rate after the HeO-GDC illustrates that a fresh surface was uncovered.

An important consideration is the fraction of the in-vessel tritium inventory that would be released in the event of a credible vacuum accident. To determine this, room air was admitted to the torus in two steps: first, 18 torr of room air plus 20 torr of nitrogen was introduced into the torus and purged to the gas holding tank as a benchmark of releasable tritium; 745 Ci were recovered. Secondly, room air was admitted to raise the torus pressure to near atmospheric (718 torr) and then purged, resulting in an additional 1,341 Ci. This two step process resulted in 2,086 Ci of released tritium, approximately 15% of the in-vessel inventory. After this sequence 20 torr of room air was introduced and purged from the torus again to gauge the effectiveness of the vent in removing releasable tritium (as compared to the initial 18 torr effect). Only 15 Ci were released, compared to the 745 Ci obtained for the earlier 18 torr case, demonstrating that 85% of the initial in-vessel inventory was tenaciously held.

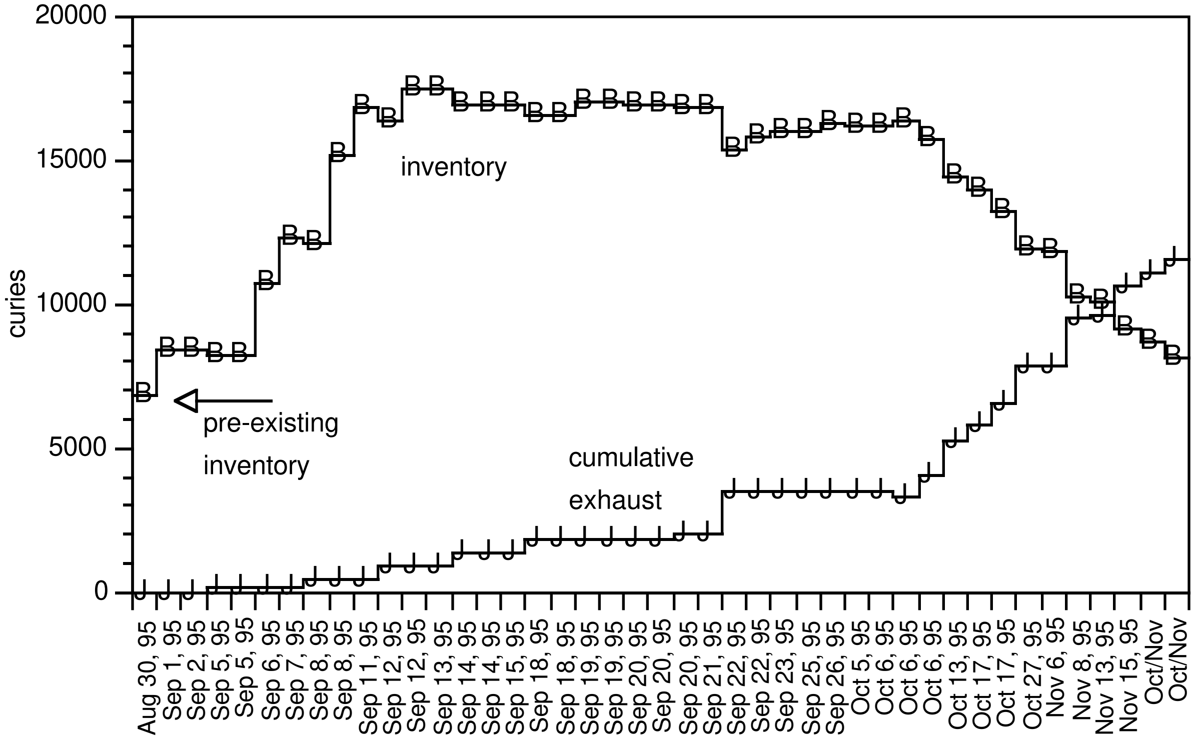

Fig. 4 In-vessel inventory and measured exhaust in September-November 1995.

Following pumpout, the normal procedure for recovery from a vessel vent was carried out.[25] This involved D-GDC, boronization with a deuterated-diborane glow and pulse discharge cleaning (PDC) with the vessel heated to 150 °C. During this procedure, 1,610 Ci were removed during the 55 hr D-GDC, 169 Ci were removed during the 12 hour boronization. PDC was used to further heat the limiters to 250° C (the administrative limit due to thermal stresses) for a 23 hour period resulting in the release of 956 Ci. This was followed by six days of disruptive discharge cleaning (DDC). In contrast to the major disruption that occurred during the L-mode campaign, the tritium exhaust during the DDC campaign was at the normal background level (376 Ci). It appears that there was negligible tritium remaining in the accessible surface region of the limiter. This illustrates that tritium removal is more effective if done promptly after exposure to tritium discharges. The various tritium removal activities extracted 8,300 Ci from the initial value of the in-vessel inventory of 16,440 Ci (see Fig. 4).

Some caveats should be noted. The parallel plate ion chamber used to measure tritium suffered from a contamination problem during the measurements on 5th and 8th September 1995 and the values presented for those days are upper limits on the tritium recovered. During a tritium puff sequence without neutral beam injection and also during glow discharge cleaning, the neutral

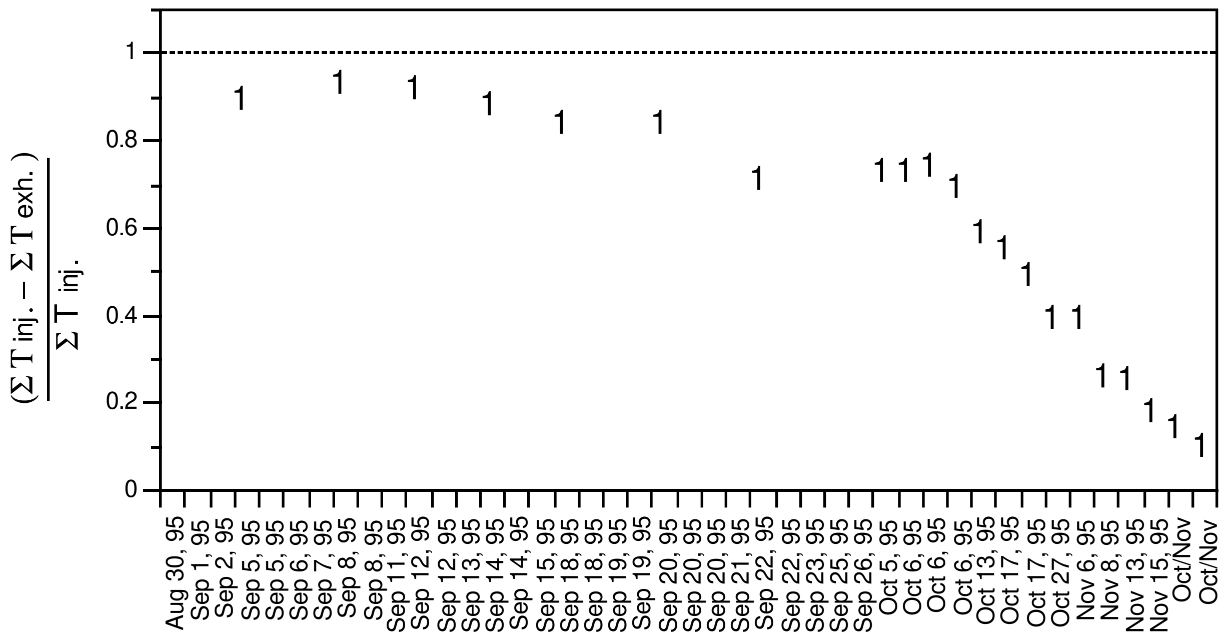

Fig. 5 The fractional retention, defined as ratio of the cumulative tritium injected into TFTR during September-November 1995, less the cumulative exhaust, divided by the cumulative tritium injected.

beam torus interface valves were closed and the torus exhaust was pumped by the torus pumping system to a gas holding tank for measurement of the tritium effluent. In these cases, the pump manifolds were purged four times with dry nitrogen to ensure the maximum recovery of tritium. For occasions where the torus pumping system was used there is a potential for a low level of tritium to be held up in the pump oil. At the end of September 1995 plasma operations the oil in four turbopumps in the torus pumping system was changed and an assay taken of the tritium content. Each pump contains 0.32 l of Leybold HE500 oil and the measured tritium content in the oil from each pump was 7, 16, 19 and 32 mCi/l. The mechanical and blower pumps use Krytox oil which was not sampled at this time. If we assume that the tritium uptake of Krytox was 32 mCi/l, the highest number measured for Leybold HE500, then the total tritium in all 19 l of pump oil was less than 1 Ci and is not significant for the present purposes.

There is a potential for tritium to condense as tritiated water frost (HTO) on the liquid nitrogen cooled panels in the neutral beam boxes. In November 1995 the beam boxes were warmed up to above 0°C and nitrogen was admitted to raise the pressure to 12, 10, 20, and 770 torr for beam boxes 2,3,4,5 respectively. The beam boxes were then pumped to the gas holding tank and a total of 467 Ci tritium measured in the exhaust. This is a small part of the total inventory.

The calculated retention fraction is shown in Fig. 5 and reflects the tritium injected and tritium exhaust during the September/November time period. It is possible that some of the tritium exhaust was injected into the torus at an earlier date, however the amount of tritium injected and the amount of tritium released from the limiter during a discharge both increased dramatically in this time period. The prior in-vessel inventory is believed to be buried by co-deposition and not to significantly contribute to the exhaust.

7. Summary

The aim of the tritium gas injection in the L-mode campaign in early September 1995 was to maximize the tritium in the plasma. In this period no attempt was made to remove tritium and the retention of injected tritium was high (>90%). After the campaign there was a sequence of conditioning discharges including deuterium neutral beam heated discharges which restored the limiter to a low recycling state followed by a successful campaign to reduce the tritium in-vessel inventory. The amount of tritium released depends both on the efficacy of a particular technique and the quantity of tritium that is accessible. This is illustrated by the conspicuous increase in tritium recovered from the cryopanels in the regeneration immediately following a major disruption on September 20, 1995 while the final disruptive discharge cleaning campaign released negligible tritium. Also noteworthy was the initial high removal rate of deuterium glow discharges. Exposing the torus to 718 torr of room air resulted in a release of approximately 2,100 Ci, a small fraction of the in-vessel inventory. The various tritium removal activities were successful in removing half of the in-vessel inventory. In the period 18 November 1995 - 2 January 1996 a further 987 Ci were recovered. While various conditioning activities (PDC, DDC, He-GDC, boronization) were undertaken in this period the removal rate per day was low and is considered to be at a background level.

For inventory purposes it is proposed to treat part of the in-vessel inventory as ‘tenaciously held’ and unlikely to be released in the event of a credible vacuum accident. The most conservative approach would be to assume that the release could not exceed that observed for the torus vent prior to the bakeout and conditioning (which further depleted the tritium). Of the 7,163 Ci remaining tritium in the in-vessel inventory on 2 January 1996, 5,000 Ci is considered to be tenaciously held and it is proposed that this be accounted for separately. The level of potentially releasable tritium is much less than the 20,000 Ci administrative limit and is not a constraint on operations.

Comparing TFTR and JET, deuterium glow conditioning and the major disruption were found to be effective in removing tritium in both machines. A significant difference was that the tritium release rate with a deuterium ‘soak’ was negligible in TFTR while high in JET. This difference may be related to the mobility of hydrogen isotopes in graphite at the different temperatures (ambient in TFTR and 300 °C in JET).

Since 9 December 1993 there have been 4257 high power (>10 MW NBI) discharges in TFTR, 402 of which have involved tritium beam injection. Tritium usage is expected to be higher in future DT machines and active control of the tritium inventory essential to avoid build up of tritium inventory. Deuterium glow discharges appear to be an interesting option for limiting the tritium inventory. Tritium injected during a day's plasma operations will be localized close to the surface of the graphite. At the initial rate of tritium removal by D-GDC in TFTR, a one hour glow discharge removes 170 Ci of tritium. This matches the tritium injected by a cumulative 46 MW-sec of tritium neutral beams injection and corresponds to 2-8 discharges depending on the beam power and D/T mix used. In practice the D-GDC may need to be followed by conditioning to restore the limiter to a low recycling state.

Table II. Summary of efficacy of removal techniques in TFTR.

Machine operation:	Comments:
He-GDC, outgas, D soak	Ineffective
D-GDC	Initial removal rate high (>170 Ci/hour), declining to 10 Ci/hour. Accesses only tritium on surfaces exposed to discharge.
HeO-GDC	Rate ≈ 50Ci/hour - constant with time
718 torr room air	2,086 Ci removed, access to all surfaces
Disruptions	Flash heating of limiter surface near midplane. Release of recently retained tritium.
PDC	Heats limiter to 250° C. 956 Ci removed over 23 hours.
Boronization	Little tritium released, most near surface tritium already removed.

Acknowledgments.

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Table I. History of tritium injected into the torus and measured exhaust during September - November 1995. The abbreviations are explained in section 4 and the units are curies. The data in italics represents an upper limit. The experimental uncertainty is ±2% for tritium injection and ±5% - ±7% for pumping, however, numbers are given to the nearest curie to avoid rounding errors in the summations.

Date	Plasma Ops.	T puff	NB inj.	NB pump	turbo pump	Sum input	Sum output	Inventory	% retention
30-Aug	inventory					0	0	6,863	
1-Sep	DT L-mode	1244	276			1,520	0	8,383	
2-Sep					-10	1,520	10	8,374	
5-Sep				*-156*		1,520	166	8,218	89%
5-Sep	D L-mode					1,520	166	8,218	
6-Sep	D L-mode +He4 GDC	2193	304		-13	4,018	179	10,702	
7-Sep	DT L-mode	1319	314			5,650	179	12,335	
8-Sep				*-228*		5,650	407	12,107	93%
8-Sep	RF	3056				8,707	407	15,163	
11-Sep	RF	1630				10,337	407	16,793	
12-Sep				-453		10,337	860	16,340	92%
12-Sep	RF then condit.	1163				11,499	860	17,502	
13-Sep	condit.					11,499	860	17,502	
14-Sep				-534		11,499	1,394	16,968	88%
14-Sep	condit.					11,499	1,394	16,968	
15-Sep	D supershot					11,499	1,394	16,968	
18-Sep				-439		11,499	1,833	16,529	84%
18-Sep	D supershot					11,499	1,833	16,529	
19-Sep			447			11,946	1,833	16,976	
19-Sep	DT supershot					11,946	1,833	16,976	
20-Sep				-36		11,946	1,869	16,940	84%
20-Sep	D supershot					11,946	1,869	16,940	
20-Sep	disruption				-139	11,946	2,008	16,801	
21-Sep	D supershot					11,946	2,008	16,801	
22-Sep				-1483		11,946	3,490	15,319	71%
22-Sep	DT supershot		448			12,394	3,490	15,767	
23-Sep	DT supershot		266			12,660	3,490	16,033	
25-Sep	D supershot					12,660	3,490	16,033	
26-Sep	DT supershot		210			12,871	3,490	16,243	
5-Oct	outgas			-36		12,871	3,526	16,208	73%
6-Oct	D soak				-5	12,871	3,532	16,202	73%
6-Oct	see sect. 4.				197	12,871	3,335	16,399	74%
6-Oct	D-GDC				-687	12,871	4,021	15,713	69%
13-Oct	He O-GDC				-1249	12,871	5,270	14,463	59%
17-Oct	D-GDC				-495	12,871	5,766	13,968	55%
17-Oct	vent 38 torr				-745	12,871	6,511	13,223	49%
27-Oct	vent 720 torr				-1341	12,871	7,852	11,882	39%
6-Nov	vent 20 torr				-15	12,871	7,867	11,867	39%
8-Nov	bake 150°C + D-GDC				-1609	12,871	9,476	10,258	26%
13-Nov	Boronization				-169	12,871	9,645	10,089	25%
15-Nov	PDC				-956	12,871	10,601	9,133	18%
Oct/Nov	NB vent				-467	12,871	11,068	8,666	14%
Oct/Nov	various				-524	12,871	11,592	8,141	10%

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