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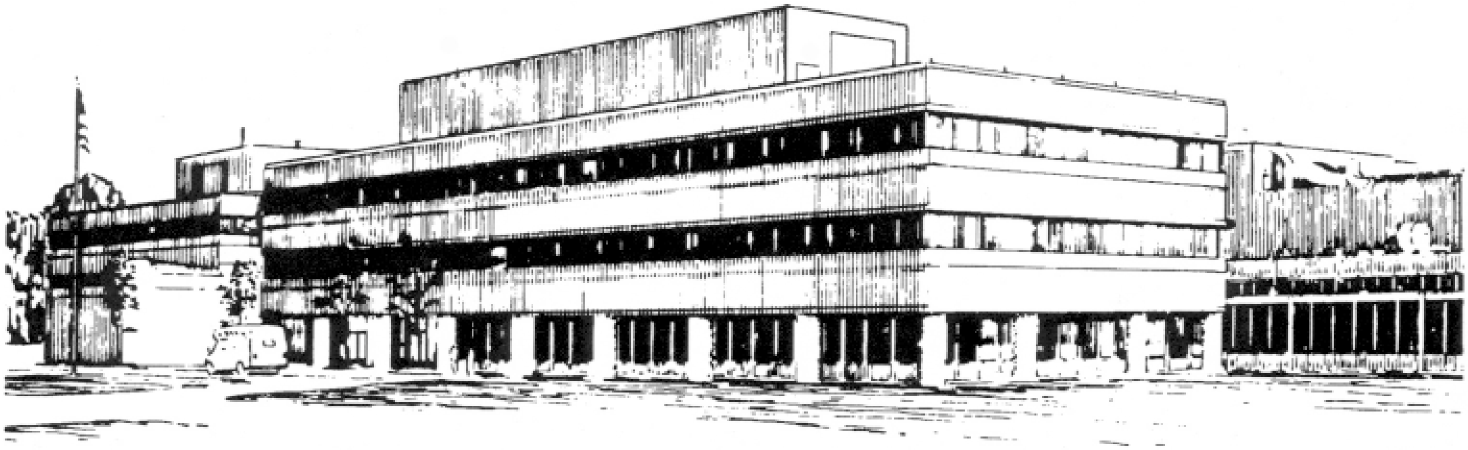

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LONG TERM TRITIUM TRAPPING IN TFTR AND JET*

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1. Introduction

Tritium retention in TFTR and JET shows striking similarities and contrasts. In TFTR, 5 g of tritium were injected into circular plasmas over a 3.5 year period, mostly by neutral beam injection [1]. In JET, 35 g were injected into divertor plasmas over a 6 month campaign, mostly by gas puffing [2]. In TFTR the bumper limiter provided a large source of eroded carbon and a major part of tritium was co-deposited on the limiter and vessel wall. Only a small area of the codeposit flaked off [3]. In JET the wall is a net erosion area, and co-deposition occurs principally in shadowed parts of the inner divertor, with heavy flaking. In both machines the initial tritium retention, after a change from deuterium to tritium gas puffing, is high and is due to isotope exchange with deuterium on plasma facing surfaces (dynamic inventory). The contribution of codeposition is lower but cumulative, and is revealed by including periods of D fuelling that reversed the T/D isotope exchange (Table 1).

When tritium was first introduced by gas puff into both JET and TFTR it immediately exchanged with deuterium. Previous experiments [4,5,6] on hydrogen/deuterium isotope exchange in both machines showed >60% exchange in the plasma after fuelling with the new gas for 10-20 discharges. However, complete exchange of the wall inventory takes considerably longer. In JET the tritium retention rate (≈40%) continued to exceed the normal deuterium retention rate with no indication of convergence within the duration of the DTE1 experiments. Interestingly, after three months clean-up by pulsing in deuterium following DTE1, the tritium retention (17%) more closely matched the long-term deuterium retention. TFTR was mostly fueled by neutral beam injection with a T/D ratio of 3%, so isotope exchange was not a dominant factor.

Table 1. Tritium Retention	TFTR	JET (DTE1)
Total tritium injected by NBI	3.1 g	0.6 g
Total tritium injected by gas puff	2.1 g	34.4 g
Total tritium retained during DT operations	2.6 g	11.5 g
Initial % retention during T puff fueling (wall saturation + isotope exchange)	≈ 90%	≈ 40%
Longer term % retention including D only fueling (mostly co-deposition)	51%	17%
Tritium remaining in torus	0.85 g (4/98)	4.2 g (7/98)
Long term retention	16% (4/98)	12% (7/98) 6% (12/99)
Average deuterium retention (for comparison)	≈ 40%	≈ 10-15%

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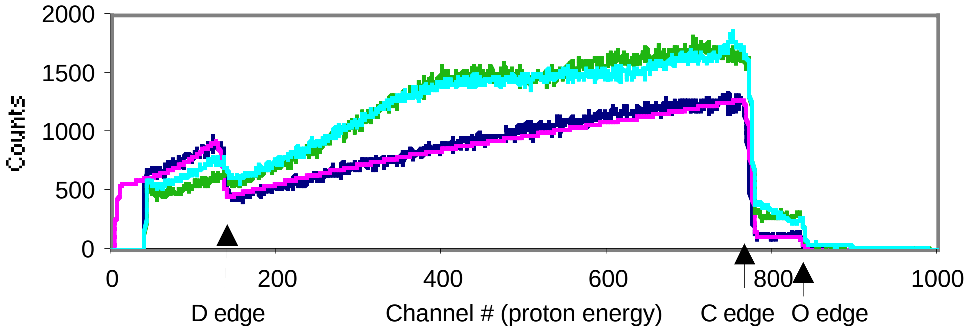

Fig. 1 Rutherford back scattering analysis of front and back surface of TFTR flakes (upper grey curves) and JET deposits (divertor tile 3, prior to DTE1) with model simulation (lower curves).

Overall, the inner wall of JET is a region of net erosion by charge exchange neutrals during operation with diverted plasmas. The sputtered material is ionized in the scrape-off-layer (SOL), and transported to the inner divertor by the large ion drift velocity in the SOL. In TFTR erosion occurred principally at the carbon tiles of the inner bumper limiter and the material was co-deposited with tritium and deuterium on the limiter and stainless vessel wall. Localized spalling affected 15% of the TFTR tiles after the termination of plasma operations [7], and specimen flakes were lifted from the tiles for ion beam analysis (Fig. 1). The D/C ratio in the TFTR flakes was 0.13 on the plasma facing surface, 0.25 on the back surface and 0.11 in the bulk. The JET data fit well to a model calculation with 46% C, 30% D, 20% H and 4% O. As for all the ion beam analyses, it is probable that some of the D has been lost by isotope exchange with H. The JET deposits show a larger D/C ratio than those from TFTR, and the concentration of retained T varies in proportion to the D. The D/C ratio in the deposits reflects their temperature history, and also the ion and energetic neutral fluxes to the surface. The deposits in JET are completely shadowed from ion fluxes and temperature excursions, whilst the TFTR inner wall experiences large particle and power surges. Recent analyses show a Be rich residue on the inner divertor in JET and confirm the migration of deposited carbon from the divertor to shadowed regions on the inner louvres at a level sufficient to account for the observed retention.

Diverse and reliable techniques have been developed for the determination of the total tritium content in tiles and for the analysis of the tritium distribution on the surface and in the bulk of tiles. Samples have been obtained from tiles by a "coring" technique and the tritium content determined by methods such as full combustion, calorimetry, thermal release of tritium in a flow system and accelerator mass spectrometry [8,9,10]. Calorimetry has the advantage of being a non-destructive technique that provides information on the total tritium in large tile samples. The results confirm earlier reports that indicate that in JET more than 95% of the tritium is present in loose materials (flakes, dust, etc.) and not on tiles, with substantially higher retention on the inner divertor wall region than the outer one.

Fig. 2 shows profiles from a TFTR graphite tile with low concentrations of tritium from DD reactions in deuterium plasmas, and a JET graphite tile from DTE1. The combustion measurements revealed that almost all the tritium was in a surface layer < 50 microns thick, with the remaining 1% distributed within the bulk of the tile. The tritium distribution was very inhomogeneous with variations of up to x10 on the TFTR tile and up to x156 for JET, especially for poloidal limiter tiles. The profiles reflect the location, plasma exposure history and also duration of exposure to air after retrieval from the machine. The 15 µm thick residue

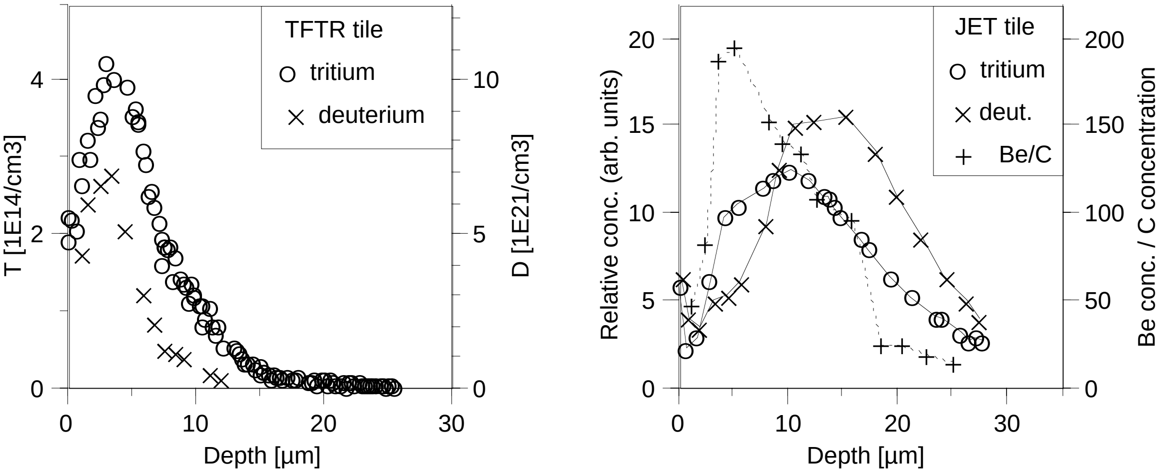

Fig. 2. D and T depth profiles for a TFTR tile (AXF-5Q graphite, bumper limiter bay P col. A row 12, used 1988-1992) and D, T profiles and the Be/C ratio for a JET (DTE1/5B.17R) poloidal limiter graphite tile.

of Be on the surface of the JET tile did not appear to play a role in the retention. Some JET CFC divertor tiles were found to contain an appreciable fraction of tritium distributed in the bulk. For the inner wall CFC divertor tiles, the bulk concentrations even exceeded those in the thin surface co-deposited/implanted layer [11]. Bulk tritium is strongly bound and therefore not easy to recover via a thermal treatment. Analyses of more tiles are in progress.

Tiles and flakes removed from machines operated with tritium show an initial large tritium release followed by a slow and practically constant chronic release rate at room temperature. First results appear to indicate that moisture has no major impact on this release rate for JET flakes, however the release from a TFTR tile was higher in air than argon. The chronic liberation can possibly be explained by a radiation-induced mechanism, but more work is necessary to substantiate this conclusion. Flakes have a higher specific tritium concentration, higher specific surface area and, as expected, a higher chronic tritium release rate than tiles. For JET flakes obtained after the DTE1 campaign the rate was measured to be $(2.0 \pm 1.1) \times 10^{10}$ Bq/month/g over a period of three years.

Lithium was used for wall conditioning in TFTR and may have affected the retention of tritium. Previous modeling[1] of tritium retention was able to match the observed global retention and predicted strong localization of tritium at leading edges located at poloidal angles of ± 60°. This localization was borne out by subsequent tile measurements [12]. Lithium was introduced as pellets, by effusive injection (from a lithium oven) and from a laser-aerosol system [13]. The total number of lithium atoms introduced to the torus during the period 1993 – 1997 was $\sim 1.7 \times 10^{24}$ as compared to the 1×10^{24} atoms of tritium and 3×10^{25} atoms of deuterium injected. The addition of lithium can potentially increase the concomitant retention of T, both globally and locally, through the intercalation process, whereby deposited lithium may rapidly be absorbed into the graphite substrate, reaching a density up to 0.17 of the carbon density [14,15]. This fraction is similar to the ratio of deuterium to carbon measured in flakes (fig.1). Further, T and D can be retained as LiT or LiD in the proportion 1:1.

Fig. 3a shows the calculated lithium deposition calculated by the BBQ code for a discharge in an extensive Li conditioning (‘painting‘) sequence preceding a record D-T supershot (83546). This discharge had the largest major radius (2.57m) of the series, which showed increasing

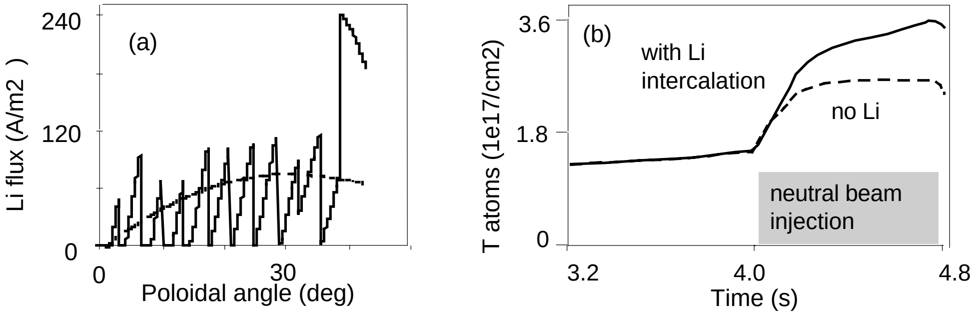

Fig. 3 (a) showing the calculated poloidal distribution of Li deposition for discharge 83451; (b) calculated T surface density in a high deposition zone, for plasma conditions of shot 83545, with and without additional retention due to Li.

localization of Li at high poloidal angles as major radius was increased. The WDIFFUSE code was used with the Li intercalation diffusivity from ref. [15] to calculate the Li transport. Fig. 3b shows that the Li and T implantation distributions can overlap during the high power phase, leading to an up to x1.3 increase in predicted local T retention and narrowing the remaining gap between modeling and observed retention. Significant uncertainties remain in the diffusion coefficient, the scrape-off layer parameters and the detailed history, however these calculations indicate that lithium could play a role in tritium retention in TFTR.

Tritium issues will play a central role in the performance, operation and safety of next-step DT fusion devices [16]. Current experience and modeling indicates the in-vessel tritium limit for ITER/FEAT would be reached in about 100 pulses. However the planned Be wall may reduce the initial erosion of the carbon target and the subsequent retention. This could have a crucial bearing on the use of carbon in the high heat flux region of the ITER/FEAT divertor and an experimental investigation on JET is warranted.

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