ELSEVIER

Journal of Nuclear Materials 290–293 (2001) 486–490

journal of nuclear materials

www.elsevier.nl/locate/jnucmat

Studies of tritiated co-deposited layers in TFTR

C.H. Skinner [a,*], C.A. Gentile [a], G. Ascione [a], A. Carpe [a], R.A. Causey [b], T. Hayashi [c], J. Hogan [d], S. Langish [a], M. Nishi [c], W.M. Shu [c], W.R. Wampler [b], K.M. Young [a]

[a] *Princeton Plasma Physics Laboratory, Princeton, NJ 08543, USA*
[b] *Sandia National Laboratories, Albuquerque NM 87175 and Livermore, CA 94550, USA*
[c] *Tritium Engineering Laboratory, JAERI, Ibaraki, Japan*
[d] *Oak Ridge National Laboratory, Oak Ridge, TN 37830, USA*

Abstract

Plasma facing components in TFTR contain an important record of plasma wall interactions in reactor grade DT plasmas. Tiles, flakes, wall coupons, a stainless steel shutter and dust samples have been retrieved from the TFTR vessel for analysis. Selected samples have been baked to release tritium and assay the tritium content. The in-vessel tritium inventory is estimated to be 0.56 g and is consistent with the in-vessel tritium inventory derived from the difference between tritium fueling and tritium exhaust. The distribution of tritium on the limiter and vessel wall showed complex patterns of co-deposition. Relatively high concentrations of tritium were found at the top and bottom of the bumper limiter, as predicted by earlier BBQ modeling.

Keywords: TFTR; Tritium retention; Tritium inventory; Tritium co-deposition; Dust; Nuclear fusion

1. Introduction

Tritium issues are central to the development of fusion power [1]. A significant milestone was reached when deuterium – tritium plasmas in TFTR and JET produced 10 and 16 MW of fusion power, respectively [2,3]. Tritium was retained inside the vacuum vessel of both TFTR and JET principally by co-deposition with carbon eroded from plasma facing components [4,5]. Tritium operations on TFTR extended over 3.5 years with 5 g of tritium supplied to the plasma via neutral beam injection and gas puffs. Extensive deuterium fueled discharges were used to optimize the plasma conditions before tritium injection and the isotopic ratio of T/D fueling was 3%. In TFTR the average tritium retention fraction was 51% during normal plasma operations and 16% over the long term including clean-up periods [6,7]. Tritium was removed by air ventilation and glow discharge cleaning during two maintenance periods and after the termination of plasma operations [8,9].

The analysis of plasma facing components from tokamaks that have been operated with tritium plasmas is uniquely valuable in understanding the behavior of tritium in these devices. TFTR operated with toroidal plasmas with a circular cross-section that were in contact with an inner toroidal 'bumper' limiter. The total area of the bumper limiter was 22 m^2 and it is divided into 20 bays (labeled A-T) each composed of 24 rows of tiles, 4 tiles wide. Each bay is curved in both toroidal and poloidal directions and the midplane center extends out 4.6 mm from a true toroidal surface. The midplane tiles are 125 mm wide and 81 mm high. High heat flux areas are covered with Fiber Materials Inc. 4D coarse weave carbon fiber composite (CFC) tiles and Hercules 3-D fine weave CFC tiles and the remainder Union Carbide AXF-5Q isotropic graphite [10]. The outer vacuum vessel is 304 stainless steel and is protected by several groups of graphite tiles arranged poloidally.

The plasma facing surfaces portray a rich and spatially complex imprint of many years of TFTR plasma

* Corresponding author. Tel.: +1-609 243 2214; fax: +1-609 243 2265.
E-mail address: cskinner@pppl.gov (C.H. Skinner).

0022-3115/01/$ - see front matter
PII: S0022-3115(00)00643-7

operations (Fig. 1). The connection length of a field line launched from the limiter surface varies strongly with spatial position and controls the balance between erosion and co-deposition [11]. In Fig. 1, co-deposition is visible in a diagonal band from the upper right to lower left of bay K and on the left side of the poloidal limiter tile at the floor. Co-deposited layers on graphite tiles began to flake after the termination of plasma operations [12,13]. Minor flaking can now be observed on CFC tiles and of co-deposited layers on the stainless steel vessel floor. The vessel has been activated by 14 MeV DT neutrons and the dose rate inside the vessel is $\approx$ 34 mrem/h (340 μSv/h). Bubble suits with externally supplied air were employed in two entries into the vessel to retrieve tiles, flakes, wall coupons, a stainless steel shutter and dust samples and to make in-vessel measurements of surface tritium. Decommissioning activities commenced in October 1999 and will extend over 3 years. In the year 2002, the vessel will be filled with low-density cellular concrete, cut into ten segments by a 10 mm diamond wire rope for burial [14].

2. TFTR tritium inventory

Measurements of the tritium inventory of DT machines are important to verify compliance with regulatory safety limits during plasma operations and for end-of-life disposal. Previous measurements of TFTR tiles exposed to deuterium plasmas showed the majority of hydrogen isotope released on baking in air at 350°C for an hour [15,16]. The tritium released from bakeout of selected tiles retrieved from the TFTR bumper limiter is shown in Fig. 1. Tiles from column C were selected to provide a comparison to previous D measurements [17]. The tiles were typically baked at a temperature of 500°C in air for 1 hour, a few tiles had preliminary bakes at 350°C. The exhaust accumulated in a tank and the tritium was measured to 0.1 Ci accuracy with a 'Fempto Tech' ion chamber [1 Ci = 2.06×10^{19} T atoms = 1.04×10^{-4} g]. A constant airflow at 40 Torr provided an order of magnitude more oxygen than required to oxidize the co-deposits and the tritium release terminated well before the end of the bake time. One tile was

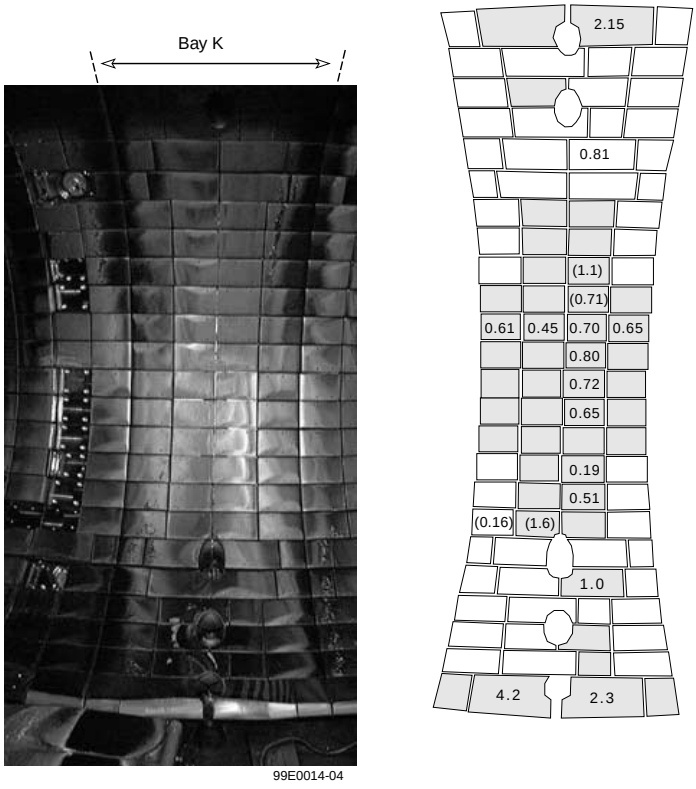

Fig. 1. TFTR bumper limiter at bay K on 17th February 1999 showing co-deposition, flaking and white deposits. Bay K is bounded by continuous vertical gaps between the tiles as indicated by the dashed lines at the top of the photograph. Some tiles have been removed from bay L on the left. Deposition on a poloidal limiter tile may be seen at the lower left. The tiles are numbered by row from 1 (bottom) to 24 (top) and by column left (A) to right (D). The diagram depicts the tritium released (in Curies) from baking selected bay K tiles (in parentheses bay L tiles). Unshaded tiles are AXF-5Q graphite, gray shading denotes carbon fiber composite.

baked at 500°C a second time but did not release a measurable amount of tritium.

Previous ion beam measurements of bay N column C tiles exposed to deuterium plasmas showed a marked up/down contrast in near surface areal deuterium density on the plasma facing tile surface and projections of the expected tritium inventory treated areas of low deposition and high deposition separately [17]. Such an up/down contrast is not evident in the present measurements (Fig. 1). Significant differences include the coarser spatial resolution (1 tile compared to the 1 mm square ion beam) and the inclusion of tritium deposited on the sides of the tiles in the bakeout measurements (previous measurements showed relatively high deuterium deposition on sides of tiles that had low deuterium density on the plasma facing surface). Also, the bumper limiter was realigned after the deuterium measurements and, of course, the detailed plasma exposure history was different. Tile-to-tile variations in the present measurements may be partly due to residual alignment differences, differences in the width of the gaps between the tiles and the presence of diagnostic penetrations. The degree of toroidal symmetry is important for decommissioning. Tiles from the same relative location (row 13 column C) at bays I, E, and D showed similar (within ±17%) tritium release as the bay K row 13 column C tile.

Complete incineration measurements are planned to measure the small fraction of tritium expected to remain in deep traps after bakeout at 500°C. For the present, we conservatively assume that 90% of the tritium was released. We estimate the tritium inventory of the bumper limiter as follows. The total plasma facing area of the baked tiles is 0.30 m^2 and the total tritium released 23.4 Ci. Including a 10% allowance for unreleased tritium, the areal density is 87 Ci/m^2. Extrapolating to the 22 m^2 area of the bumper limiter, we estimate the tritium inventory of the bumper limiter to be 1,900 Ci or 0.2 g.

Tritium also accumulates by co-deposition on the outboard plasma facing components such as the poloidal limiter CFC tiles (BF Goodrich 2.5D staple knit weave), neutral beam armor tiles and on the stainless steel vessel wall (in contrast to JET and other machines which experience wall erosion). Previous deuterium measurements [18] indicated 41% of the total deuterium inventory to be on the vessel wall with factor-of-three toroidal variations in local deuterium areal density as measured on coupons [19]. We have retrieved two poloidal limiter tiles, 3 pairs of graphite coupons and a stainless steel shutter and have baked one tile and 3 coupons and the shutter. The tritium released was trapped in a highly sensitive differential atmospheric tritium sampler [20] and assayed by scintillation counting to an accuracy $\leqslant$10%. The coupons have a 6.5 cm^2 plasma facing surface but parts of the sides are also exposed and accumulate some tritium. An effective area of 12.6 cm^2 was derived from the area weighted by the surface tritium as measured by an ion chamber. The total outboard vessel area is estimated at 110 m^2 [19]. The average areal density of released tritium (Table 1) is 29 Ci/m^2. Including a 10% allowance for unreleased tritium the total is 32 Ci/m^2. This is 37% of the areal tritium density on the bumper limiter but the total outboard area is 5× larger so 65% of the total tritium appears to be on the outboard side. We estimate 3500 Ci on the outboard side and a total tritium inventory of 5400 Ci or 0.56 g. The sparse spatial sampling, especially on the outboard side (0.1%), adds significant uncertainty to this estimate.

Table 1
Outboard tritium

	Tritium released (Ci)	Areal density (Ci/m^2)
Bay O/N tile	3.8	31
Bay H midplane coupon	0.035	24
Bay N bottom coupon	0.095	65
Bay P midplane coupon	0.024	16
Bay H shutter (stainless steel)	0.396	9
	Mean	**29**

Previous estimates of tritium inventory in the vessel were derived from the difference between the cumulative tritium fueling and exhaust, corrected for radioactive decay. On 3 May 2000 this difference inventory was 0.64 g. The agreement between the measurements of components removed from the vessel and the inventory derived from the difference between tritium fueling and tritium exhaust is excellent considering the experimental uncertainties and is an encouraging validation of the difference inventory methodology.

3. Surface tritium measurements

Surface tritium was measured inside the vessel by an open wall ion chamber [21]. This technique, and others that detect betas emitted from radioactive decay, detects tritium only in the top micron due to the limited range of the betas in graphite. The detector area was 3.4 cm diameter, however in some cases this was reduced to 1.2 or 0.6 cm diameter to extend the dynamic range or to sample a small area. Near surface tritium has been depleted by glow discharge and ventilation after the termination of plasma operations. Fig. 2 shows the surface tritium on the outer vessel wall at bays G, H, J, L. Large variations can be seen reflecting the complex geometry of the in-vessel hardware. Spatially complex patterns were also observed on the bay K bumper limiter tiles retrieved from the vessel (average surface tritium: 1.38 Ci/m^2), bay O/N poloidal limiter (1.30 Ci/m^2), and bay G neutral beam duct (0.83 Ci/m^2). Fig. 3 shows the surface tritium concentration from the bay K centerline

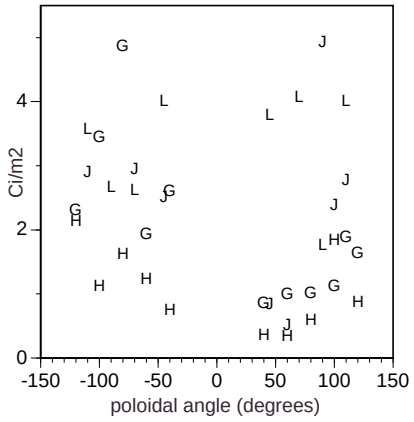

Fig. 2. Surface tritium measured by an open wall ion chamber on the vacuum vessel surface in a poloidal ring at bays G, H, J, L. '0' degrees corresponds to the outboard midplane.

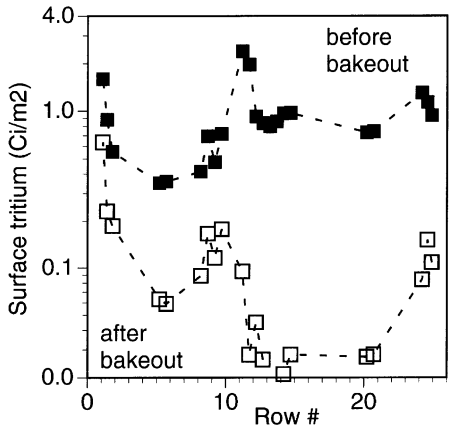

Fig. 3. Surface tritium measured on bay K poloidal centerline before and after bakeout of selected tiles. The lines are intended as a visual aid.

before and after bakeout. The up/down asymmetry in tritium remaining after the bakeout is consistent with the lower rows being a net erosion region where the oxidation rate of the tritium implanted in the native carbon is slower [16] than in the upper co-deposition region. Further elemental analyses of the components and tests of detritiation by UV and laser surface heating [22] are planned.

4. Flakes and dust

The mobilizability of tritium is an important factor in safety analyses of future DT reactors. Observations of flaking of the TFTR limiter were reported in [12,13]. Dust generated by plasma operations is an emerging area of concern [23,24] as the longer biological half-life of tritiated graphite dust makes it significantly more hazardous than HTO (tritiated water) [25]. In 1992 'several kilograms' of particulate debris were vacuumed from the TFTR torus [26]. Video inspection in 1996 indicated debris levels were reduced, most likely due to tile realignment. At that time, dust samples were collected from the bottom of ten vertical diagnostic pipes and from the vessel floor [27]. Additional samples were collected in the recent vessel entry with a hand vacuum cleaner fitted with a slotted nozzle and 0.2 micron pore size filter. Particles and debris were evident on the floor of the vessel including flake fragments and debris from a laser-assisted lithium conditioning aerosol device 'DOLLOP'. Bay J was particularly dusty and collection from a 10 cm $\times$ 10 cm area yielded 0.46 g. In contrast the bottom of a neutral beam duct yielded only 0.06 g from a 20 cm $\times$ 60 cm area. The gap between the bumper limiter and poloidal limiter, revealed by tile removal at bay K, yielded 0.07 g. Estimation of the total dust inventory was not possible because of the highly non-uniform distribution. Diagnostics to confidently establish compliance with regulatory dust limits in next step devices remain problematic. The most critical need is the development of means to remove dust.

5. Comparison to modeling results

Tritium is retained by atomic and molecular processes as the edge plasma interacts with plasma facing components. Previous modeling with the BBQ code [28] of erosion and subsequent co-deposition for representative conditions in TFTR DT plasmas was able to account for the order of magnitude of retention. Based on these results, a prediction was made that 'when detailed analysis of TFTR tiles from the tritium campaign is made significant concentrations of co-deposited tritium will be found near the upper and lower leading edges of the bumper limiter.' This pattern was not expected from previous deuterium measurements [17] or earlier modeling [11].

The observation of high tritium concentrations in the upper and lower rows of bumper limiter tiles (Fig. 1) suggests that the BBQ model is on the right track. Fig. 4 compares the row-averaged areal density of tritium (tritium released by bakeout/plasma facing area) to the effective sputtering yield in Fig. 3 (#76528) of [28]. The high-calculated physical sputtering erosion yield and a high-measured co-deposition at high latitudes are both due to the relatively large parallel particle flux (of D^+ for physical sputtering, C^{n+} for deposition) to the top/bottom leading edges of the TFTR inner bumper limiter. This flux generates low-energy carbon fluxes from self-sputtering, that are redeposited within about 3 cm. The data are consistent with the existence of a considerable number of TFTR discharges with large ($\sim$10 cm) radial

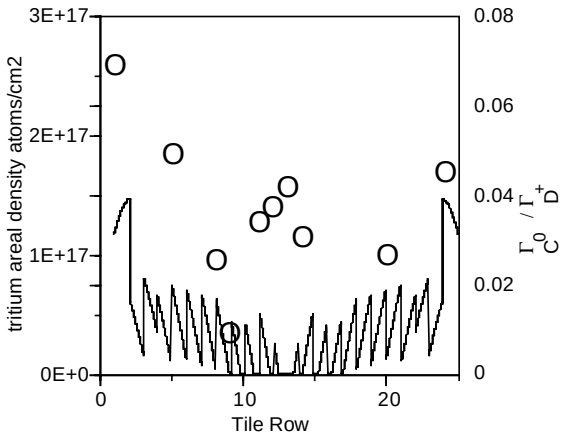

Fig. 4. Areal density of tritium averaged over available tiles from each row (circles) and local effective sputtering yield distribution (emitted impurity flux/incident D^+ flux) from Fig. 3 (a) of Ref [28].

decay length of D^+ flux due to inner wall recycling and large parallel diffusivity. More detailed reconciliation of the model and data would require explicit 3D treatment of tile–tile variations and diagnostic penetrations and a more detailed representation of the complex discharge history over 3.5 years of TFTR DT operations (including startup/shutdown, disruptions and tritium cleanup). Overall, the fact that the modeling was able to suggest a priori some features, that were not otherwise expected, is encouraging.

Acknowledgements

We wish to acknowledge informative discussions with A.A. Haasz, D. Loesser, and the dedicated work of the tritium group and collaborators at PPPL. The support was provided by the Annex IV to the JAERI/DOE Implementing Arrangement on Cooperation in Fusion Research and Development, US DOE Contract Nos. DE-AC02-76CH03073 and DE-ACO4-94AL85000.

References

[1] G. Federici et al., J. Nucl. Mater. 266–269 (1999) 14.
[2] R.J. Hawryluk, Rev. Modern Phys. 70 (1988) 537–587.
[3] A. Gibson et al., Phys. Plasmas 5 (1998) 1839.
[4] C.H. Skinner et al., J. Vac. Sci. Technol. A 14 (1996) 3267.
[5] P. Andrew et al., Fus. Eng. Des. 47 (1999) 233.
[6] C.H. Skinner et al., J. Nucl. Mater. 241–243 (1997) 214.
[7] C.H. Skinner et al., in: Proceedings of the 20th Symposium on Fusion Technology, Marseille, 1998, vol. 1, Association Euratom-CEA, Cadarache, 1998, p. 153.
[8] D. Mueller et al., in: Proceeding of 17th IEEE/NPSS Symposium on Fusion Engineering, IEEE, Piscataway, NJ, USA (1998), San Diego, Oct. 6–10, 1997, vol. 1, p. 279.
[9] A. Nagy et al, in: Proceeding of 17th IEEE/NPSS Symposium on Fusion Engineering, IEEE, Piscataway, NJ, USA (1998), San Diego, Oct. 6–10, 1997, vol. 1, p. 317.
[10] M.D. McSmith et al., Fus. Technol. 26 (1994) 498.
[11] T.Q. Hua, J.N. Brooks, J. Nucl. Mater., 196–198 (1992) 514. Note corrected units for Figs. 5 and 7 are 100 Å/s (T.Q. Hua, personal communication).
[12] C.H. Skinner et al., Nucl. Fus. 39 (1999) 1081.
[13] C.H. Skinner, C.A. Gentile, K.M. Young, in: Proceedings of the 18th IEEE/NPSS Symposium on Fusion Engineering, Oct. 25–29th 1999 Albuquerque NM, p. 89 IEEE, Piscataway, NJ, (1999). Note numbers in Figs. 5 and 7 should be corrected by factor x7.1.
[14] E. Perry et al., in: Proceedings of the 18th IEEE/NPSS Symposium on Fusion Engineering, 25–29 October, Albuquerque, NM, p. 97 IEEE, Piscataway, NJ, 1999.
[15] R.A. Causey et al., J. Nucl. Mater. 176&177 (1990) 987.
[16] J.W. Davis, A.A. Haasz, J. Nucl. Mater. 266–269 (1999) 478.
[17] W.R. Wampler et al., J. Vac. Sci. Technol. A 6 (1998) 2111.
[18] C.H. Skinner et al., Nucl. Fus. 39 (1999) 271 (reported by W.R. Wampler).
[19] H.F. Dylla, K.L. Wilson (Eds.), Rep. PPPL-2523, Princeton Plasma Physics Lab. NJ.
[20] O.A. Griesbach, J.R. Stencel, in: Proceedings of the 22nd Midyear Symposium of the Health Physics Society, San Antonio, TX., December 1988, p. 374.
[21] N.P. Kherani, W.T. Shymayda, Fus. Technol. 28 (1995).
[22] C.H. Skinner et al., in: Proceedings of the 17th IEEE/NPSS Symposium on Fusion Engineering, San Diego, 6–10 October 1997, vol.1, p. 321, IEEE, Piscataway, NJ, (1998).
[23] K.A. McCarthy et al., Fus. Technol. 34 (1998) 728.
[24] J. Winter, G. Gebauer, J. Nucl. Mater. 266–269 (1999) 228.
[25] B. Patel et al., in: Proceedings of the 18th IEEE/NPSS Symposium on Fusion Engineering, 25–29 October, Albuquerque, NM, p. 97 IEEE, Piscataway, NJ (1999).
[26] P.H. LaMarche et al., Fusion Technolology, Elsevier, Amsterdam 1993, p. 1172.
[27] W.J. Carmack et al., in: Proceedings of the 5th Symposium on Fusion Nuclear Technology, ISFNT-5, 19–24 September 1999, Rome, Italy.
[28] C.H. Skinner et al., J. Nucl. Mater. 266–269 (1999) 940.