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Tritium Experience in the Tokamak Fusion Test Reactor

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Abstract:

Tritium management is a key enabling element in fusion technology. Tritium fuel was used in 3.5 years of successful DT operations in the Tokamak Fusion Test Reactor at Princeton Plasma Physics Laboratory. The DT campaign enabled TFTR to explore the transport, alpha physics and MHD stability of a reactor core. It also provided experience with tritium retention and removal that highlighted the importance of these issues in future DT machines. In this paper we summarize the tritium retention and removal experience in TFTR and its implications for future reactors.

1. INTRODUCTION

The worldwide plasma physics community has made substantial progress in the last two decades in understanding the fundamental issues affecting the performance of high temperature plasmas and has succeeded in creating conditions approaching those in the core of a fusion power reactor. The introduction of tritium fuel into the Tokamak Fusion Test Reactor (TFTR) and the Joint European Torus (JET) plasmas and the generation of high levels of fusion power, 10.7 MW and 16 MW respectively, has brought practical fusion power one step closer.[1,2] These DT campaigns have enabled studies of the transport, alpha physics and MHD stability of a reactor core. They have also provided experience with tritium retention and removal that highlighted the critical importance of these issues in future DT machines.

2. TRITIUM RETENTION IN TFTR.

TFTR plasmas have a circular cross section with major radius typically 2.5 m and minor radius 0.9 m. The plasma boundary is defined by an inner toroidal limiter composed of graphite tiles and carbon composite tiles in high heat flux regions. Carbon materials have excellent thermal properties and carbon impurities in the plasma lead to only small increases in radiated power. However, carbon can retain hydrogen isotopes by saturation of the implant region, absorption on internal porosity, transgranular diffusion and by codeposition of hydrogen isotopes with carbon.[3] Codeposited layers can grow and accumulate tritium *continuously*. Analysis of in-vessel components exposed during the deuterium phase of TFTR showed the main mechanism for retention was codeposition.[4] The ratio of deuterium retained in the vacuum vessel to the total used in neutral beam fueling was found to be 44% ±17%. Of this, 19% was on the plasma facing surface of the bumper limiter tiles, 7% on the tile sides, and 18% on the vacuum vessel wall.

During 3.5 years of TFTR DT operations, 100 g of tritium was processed and 5 g of tritium supplied to the plasma by neutral beam injection and direct gas puffs. There were three periods of plasma operations interspersed with cleanup campaigns to remove tritium.[5-7] The tritium input and exhaust were carefully tracked (Table I and Fig. 1). During the three run periods (excluding periods of active tritium removal) approximately 51% of the tritium supplied to the plasma was retained in the vacuum vessel, a fraction similar to that found in earlier deuterium measurements. Tokamak dust samples have been obtained from TFTR and the dust particulate size and specific surface area, important for safety and decommissioning analyses, have been measured.[8] Eleven bumper limiter tiles exposed to TFTR DT operations have been removed for analysis.

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Table I. Tritium fueling and removal history of TFTR.

Period	Number of tritium fueled discharges (NB + puff)	Tritium fueling by neutral beam injection (g)	Tritium fueling by gas puff (g)	Tritium removal from torus (g)	Tritium removal from neutral beam boxes (g)	TFTR inventory (g)
11/93 - 8/95	530	1.9	0.02			0.71
9/95	178	0.24	1.12			1.71
10/95 - 1/96				- 0.91	- 0.05	0.75
1/96 - 8/96	190	0.57	0.27			1.56
9/96-11/96				- 0.39	- 0.10	1.07
12/96 - 4/97	223	0.36	0.71			1.83
4/97-4/98				-0.50	-0.48	0.85

While some aspects of the atomic processes causing retention can be studied in laboratory experiments, the characteristics of the surface and the edge plasma in a tokamak are determined by their mutual interaction in a complex non-linear environment that is difficult to diagnose and to model. Preliminary modeling of carbon impurity production and transport in the TFTR scrape-off layer and near-edge region suggests that known erosion and codeposition mechanisms are sufficient to account for the order of magnitude of retention.[9] Modeling predicts a much lower retention fraction for ITER[10] however, due to the long pulse length and high amount of tritium fueling required, weekly dedicated periods of tritium removal may be necessary.[11]

3. TRITIUM REMOVAL FROM TFTR.

Two grams of tritium were actively removed from the TFTR torus in periods between plasma operations. Tritium removal was successful in (i) keeping the in-vessel tritium inventory within the administrative 2g limit, (ii) reducing tritium outgassing during vessel openings for hardware upgrades and for shutdown and (iii) investigating the efficacy of the available removal techniques (see Table II).[12,13] The isotopic ratio of T/D retained in the vessel is expected to follow the fueling ratio of 3% T/D. Deuterium glow discharges had an initially high removal rate but the rate declined over several hours. The removal rate in helium-oxygen glow discharges was constant but low, about 20 times less than found in laboratory experiments. Air ventilation was found to be a simple and effective method of tritium removal. Some tritium was tenaciously held and not released, an important consideration in assessing the consequences of potential accident scenarios. The current in-vessel inventory is approximately 0.85 g, a 16% long term retention rate. The outgassing rate is less than 0.1 mg / day and the radiological decay rate approximately 0.2 mg/day.

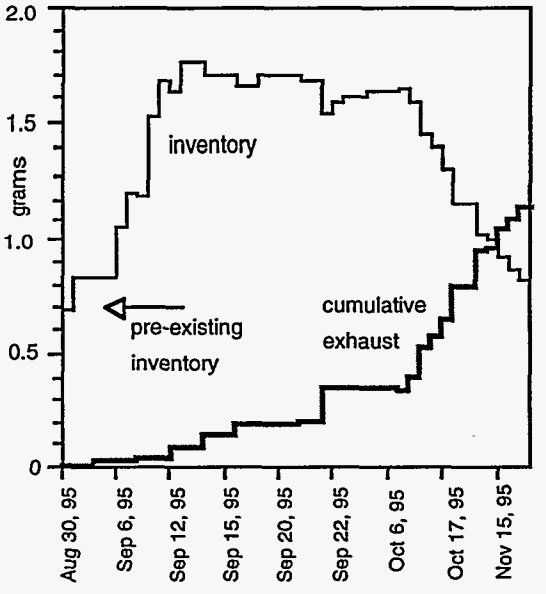

Fig. 1 In-vessel inventory and exhaust during September-November 1995. The rapid rise and fall of the tritium inventory is due to tritium gas puffing in an L-mode campaign and a subsequent tritium removal phase.

Table II Summary of tritium removal methods on TFTR

Procedure:	Comments:	Average removal rate (g/hr)
He-glow discharge cleaning, outgas, D soak	Ineffective	0
D-glow discharge cleaning	Initial removal rate high (>18 mg/hour), declining to 1 mg /hour. Accesses only tritium on surfaces exposed to discharge.	.001 to .030
HeO-glow discharge cleaning	Rate ≈ 5 mg/hour - constant with time	.005
718 torr room air	220 mg removed, access to all surfaces, the removal was very quick, < 1 hr., but it took about 24 hr. to process air	.01
Disruptions	Flash heating of limiter surface near midplane. Release of recently retained tritium. (once 0.014 g recovered after a major disruption, other times little removal seen)	0 to <.004
PDC	Heats limiter to 250° C. 100 mg removed over 23 hours.	.004
Boronization	Little tritium released, most near surface tritium already removed.	0

The tritium experience in large tokamaks and its application to ITER was the subject of a recent workshop.[14] Development of rapid, efficient tritium removal techniques were seen as a key R&D area. Removal rates that are orders of magnitude higher than experienced on TFTR will be required for long pulse DT machines. Significant progress has been made in laboratory studies of tritium removal via oxidation using air at elevated temperatures[15] or oxygen containing plasmas. One complication with oxidation techniques is the cost of detritiation of large quantities of DTO exhaust in the tritium plant, however alternative oxygen-free methods have also been proposed.[16]

4. CONCLUSIONS

TFTR was the first fusion facility with extensive experience with tritium fueling and removal and has successfully demonstrated tritium technology in a fusion environment over 3.5 years of successful DT operations. The experience highlighted the need for R&D aimed at drastically reducing tritium retention in future long pulse DT machines. To avoid the potential for public evacuation in case of the worst credible accident, the in-vessel tritium inventory in next-step long-pulse DT fusion reactors will be limited to of order 1 kg. Independent of safety considerations, such reactors must operate within the constraints of the available tritium supply (of order 2 kg/yr). In a future conversion to a fusion powered economy, net tritium breeding will be required to supply the tritium inventory necessary to startup subsequent reactors. The fraction of tritium permanently retained in a commercial fusion reactor will have to be less than 0.1%.[17]

Since the dominant pathway for retention is erosion of graphite plasma facing components followed by codeposition, an important design consideration for future fusion reactors is the reduction of heat flux to the wall to permit the use of non-carbon materials. Tungsten is a potential alternative plasma facing material with significantly lower erosion and retention rates. A coordinated effort, involving diverse parts of the fusion community will be necessary in the following areas: (i) development of plasma scenarios without transients such as disruptions and with heat load spatial peaking factors close to unity to permit flexibility in material choice, (ii) continued laboratory experiments on materials exposed to the tokamak environment (materials properties being highly sensitive to impurities and special conditions in the plasma environment)

(iii) improved real time diagnostics of the plasma wall interaction coupled with better integration of wall and plasma edge modeling codes to further progress in understanding the complex environment of the plasma edge. In addition, diagnostics to measure in-vessel tritium and dust are critically needed. Progress in these areas are central to realizing the promise of fusion as an environmentally friendly power source.

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