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Plasma wall interaction and tritium retention in TFTR

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Abstract

The Tokamak Fusion Test Reactor (TFTR) has been operating safely and routinely with deuterium–tritium fuel for more than two years. In this time, TFTR has produced a number of record breaking results including core fusion power, ~ 2 MW/m^3, comparable to that expected for ITER. Advances in wall conditioning via lithium pellet injection have played an essential role in achieving these results. Deuterium–tritium operation has also provided a special opportunity to address the

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issues of tritium recycling and retention. Tritium retention over two years of operation was approximately 40%. Recently the in-torus tritium inventory was reduced by half through a combination of glow discharge cleaning, moist-air soaks, and plasma discharge cleaning. The tritium inventory is not a constraint in continued operations. Recent results from TFTR in the context of plasma wall interactions and deuterium–tritium issues are presented.

Keywords: TFTR; Tritium inventory and economy; Helium exhaust and control; Wall particle retention; Wall conditioning

1. Introduction

The next generation of fusion reactors will face a number of issues specific to deuterium–tritium (DT) operation that originate from the tritium, neutrons and alpha particles. Safe handling of tritium, with negligible tritium release, is paramount for the public acceptance of fusion as an environmentally benign power source. Tritium retention is a concern because of limits on the acceptable tritium inventory in the reactor. Neutrons, produced in the D–T fusion reaction, activate the reactor hardware, placing significant constraints on maintenance activities. At high levels of neutron fluence, material properties may also change. Neutrons can also interfere with the plasma diagnostics and shielding may be required to avoid spurious signals. Alpha particles, in the reacting plasma, are a source of free energy that heats the plasma but may also drive plasma instabilities. Alpha particle losses caused by MHD activity may be a significant heat load on the plasma facing components, especially if the alpha flux is localized.

For the next 15–20 years — indeed until ITER or another large DT device is constructed — the only tokamaks in the world fusion program capable of operating with tritium are TFTR and JET. The first DT experiments with low concentrations of tritium were performed on JET in 1991 [1]. Over the past 2.5 years TFTR has demonstrated the ability to carry out an intensive, sustained experimental program in DT, involving significant quantities of tritium and significant production of fusion energy. In this time, TFTR has collected an impressive number of results including largest fusion power output, 10.7 MW and the total fusion energy released has been greater than 1.1 GJ. The major physics results have been the subject of several recent reviews [2–6]. The behavior of the energetic particles produced by fusion reactions and of the helium ash slowed down in the plasma has been measured in detail [7,8]. Most recently a completely new mode of enhanced confinement, the enhanced reversed shear mode (ERS) has been discovered in TFTR plasmas [9]. Control and modification of the interaction of the plasma discharge with the first wall played a central role in all these accomplishments. In particular, lithium pellet conditioning has been dramatically successful in improving performance.

The plasma-facing components in ITER will be exposed to a severe environment that is difficult to predict well. Candidate materials include beryllium, carbon and tungsten each with its own advantages and limitations. Carbon fiber composites have the advantage of superior resistance to thermal shocks due, for example, to disruptions, a high sublimation temperature, and the low radiative power losses of carbon impurities in a high temperature plasma. Results from large tokamaks that have converted from high Z to low Z plasma facing materials have indicated clear advantages for low Z materials. However, the tritium inventory in co-deposited carbon, and the baking and conditioning required to minimize the outgassing of impurities in order to maintain control of plasma density are points of concern [10]. TFTR has a special opportunity to study these issues since it has a carbon limiter which experiences erosion, codeposition and neutron flux from DT plasmas with core fusion power similar to that expected for ITER. In addition, TFTR studies on advanced tokamak physics such as the ERS mode [9], alpha channeling [11] and lithium conditioning are important because they have the potential to transform the technical issues of plasma–wall interactions in future reactors. After a short description of TFTR we will relate recent results on the ERS mode and alpha channeling. The next section (Section 2) describes aspects of the plasma wall interaction (both desirable and undesirable) leading to a section on beneficial effects of lithium conditioning. Tritium operations, recycling, retention and removal form the latter part of the paper. Other papers in these proceedings will focus on tritium removal from TFTR [12], tritium recycling [13,14], and transport in lithium conditioned discharges [15].

2. TFTR

TFTR plasmas have a circular cross section with minor radii typically in the range 0.8–0.96 m, major radii of 2.45–2.62 m, toroidal magnetic field at the plasma center of 4 to 5.6 T and plasma current of 0.6 to 2.7 MA. The plasma boundary is defined by an inner toroidal belt limiter composed of carbon composite tiles in high heat flux regions and graphite tiles, both supported by water cooled inconel-718 backing plates. The limiter extends poloidally over $\pm 60°$ from the midplane and is divided toroidally 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 slightly curved so that 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 of hydrogen with carbon, and neutron flux from the DT plasmas. After exposure to many plasma discharges each limiter sector devel-

ops a poloidally asymmetric 'footprint' or eroded area, surrounded by areas of net codeposition [16]. The pattern is related to the angle of the incident flux on the individual tiles and the slight toroidal asymmetry of the limiter [17]. The rise in bulk limiter temperature remains below 50°C during a discharge [18], however infra-red camera measurements indicate that the surface temperatures in localized hot spots increases up to 1000°C or more depending on the discharge conditions and auxiliary heating power [19].

The highest performance has been realized in the neutral beam driven hot ion or 'supershot' [20] regime characterized by central ion temperatures: $T_i = 20–45$ keV; electron temperatures: $T_e = 10–14$ keV; highly peaked density profiles with $n_e(0)/\langle n_e \rangle = 2–4.5$ and confinement enhancement, H, up to $H = 2.4$ ($\times$ITER89P [21]). Neutral beam heating has proven to be the most effective means for reaching high fusion reactivity and TFTR has been configured for routine DT operation at neutral beam powers up to 40 MW [22]. 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. Half of the beam sources inject tangentially in the direction of the plasma current and half counter to the plasma current. Selection of beam sources allows control of plasma rotation, important in confinement studies. Plasma stability in high power operation has been aided by extending the operating range of the toroidal field to 6 T.

Considerable excitement has been generated by recent TFTR experiments that have demonstrated greatly reduced particle and ion thermal transport in a configuration with reversed magnetic shear [9]. DIII-D discharges with reversed magnetic shear have also showed enhanced confinement and stability [23]. Related results were reported earlier from JET [24] and Tore Supra [25]. Under intense neutral beam heating, TFTR plasmas with reversed shear (hollow current density profile) are observed to bifurcate into two states with different transport properties. In the plasma with better confinement, particle transport is reduced to near neoclassical levels and the ion thermal diffusivity is well below predictions from conventional neoclassical theory. Since neoclassical transport is usually thought to be the minimum transport possible these results represent a dramatic improvement in confinement and performance. Interestingly, the level of turbulent fluctuations, as measured by microwave reflectometry, is correlated with the plasma transport [26]. Since short scale turbulence is frequently considered to be a source of anomalous loss in tokamaks, these results appear to be consistent with theoretical predictions that negative shear can suppress geodesic curvature driven instabilities, such as trapped particle modes, the toroidal ion temperature gradient (ITG) mode, and high-n ballooning modes. Fig. 1 shows the electron diffusivity and ion thermal conductivity in enhanced reverse shear (ERS) and reversed shear plasmas and the amplitude of density fluctuations in the ERS mode measured by microwave reflectometry. The strongly peaked pressure profile resulting from improved core confinement may generate a strong off-axis bootstrap current and sustain the hollow current density profile, a scenario that may lead to an attractive concept for a steady state tokamak reactor [27]. These results may herald a breakthrough in the understanding of tokamak transport and lead to significant improvements in the performance of present and future devices.

An economically superior development path to a tokamak reactor may be possible through high power radio frequency waves that both drive plasma current and optimally channel alpha particle power. A new technique for on- or off-axis current drive has been demonstrated on TFTR. Mode conversion of the fast magneto sonic wave into the ion-Bernstein wave was used to drive 130 kA on- or off-axis [28]. In cases where the mode conversion layer is located precisely on the Shafranov-shifted plasma axis, a strongly enhanced loss of deuterium beam ions has been seen with the escaping fusion product detectors. Losses show evidence of strong heating of the beam ions (up to several MeV) and the implied diffusion rate in energy exceeds 2 MeV^2/s. This rate should be sufficient to investigate aspects of alpha channeling [11,29,30], a concept in which externally launched RF waves absorb energy from the alpha particles and then damp on fuel ions — heating them directly; or damp on electrons — driving

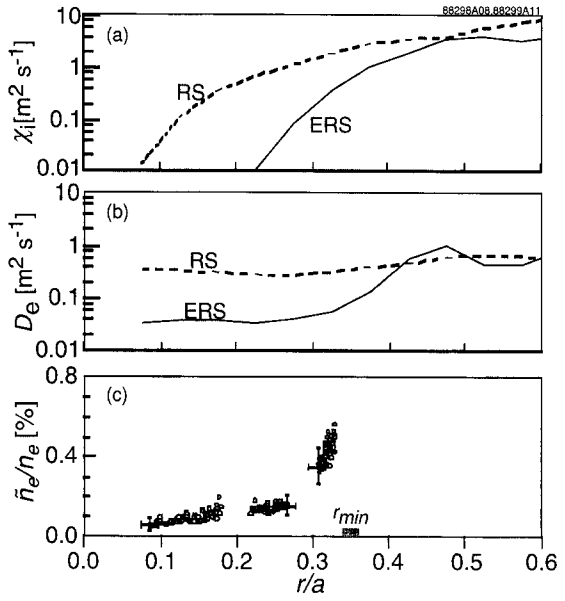

Fig. 1. (a), (b) Electron diffusivity (D_e) and ion thermal conductivity (X_i) in enhanced reverse shear (solid line) and reversed shear (dashed line) plasmas with 29 and 27 MW of neutral beam heating respectively; (c) amplitude of density fluctuations in the ERS mode measured by microwave reflectometry; 'r_{min}' denotes the radius at which the safety factor 'q' is a minimum.

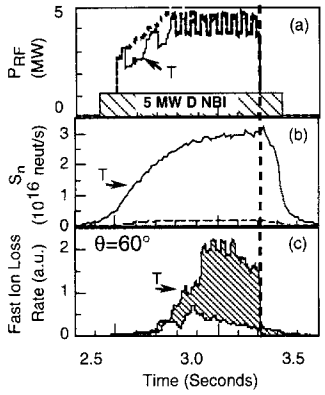

Fig. 2. (a) Injected RF power, (b) neutron rate, and (c) fast ion loss signal during on-axis mode conversion experiments. The discharge with tritium puffing has solid lines and is labeled by 'T'. The additional losses are correlated with the presence of tritium and appear to be due to ion–Bernstein wave–alpha particle coupling.

plasma current. This more effective use of the alpha particle energy could potentially improve the efficiency of a tokamak reactor. Driven losses of DT fusion alpha particles have also been observed when the mode conversion layer is sufficiently close to the alpha particle cyclotron layer (Fig. 2). Effective fast wave ICRF heating of majority ions in DT plasmas at $2\Omega_T$ has been demonstrated in TFTR supershots [31] and L-mode plasmas [32], indicating that this technique is a viable option for ITER startup.

3. Plasma wall interactions

A dramatic manifestation of a plasma-wall interaction occurred during high power, hydrogen-minority ICRF-heated discharges when internal welds in the vacuum vessel were melted resulting in a vacuum leak [33]. A strong manganese influx was detected corresponding to the melting of more than 0.35 g of stainless steel or weld filler. The domination of Mn rather than Fe, Cr, or Ni is consistent with the higher vapor pressure of Mn. Theoretical calculations have shown that toroidal Alfven eigenmodes (TAE) are capable of inducing ripple trapping of high energy particles, causing intense localized particle loss [34]. ICRF heating produces a population of energetic, deeply-trapped H-minority tail ions near the magnetic axis. The TAE is destabilized by these energetic ions [35] and transports them to large major radius, where they become trapped in the toroidal field ripple and drift rapidly to the bottom of the vessel. The total power of escaping H ions necessary to cause the melting was approximately 240 kW, only 5% of the applied ICRF power. Fortunately this loss mechanism is not effective for the alpha particle distribution in ITER, because of the very small ripple well domain. However modification of the coil design or major radius would require a reexamination of this process.

In a reactor, the maximum tolerable loss of energetic alpha particles is set by the need to protect the mechanical structures on the first wall from potentially localized alpha heating. Localization of even a small fractional loss (less than 5%) of the alpha particles could cause unacceptable heat loading or erosion. A key question is whether the toroidicity-induced Alfven eigenmode (TAE) can be destabilized by energetic alpha particles inducing large fusion alpha particle loss and damage to the first wall. There has been no indication of such 'collective' alpha particle loss processes in TFTR [36]. Measurements from three independent diagnostics show that alpha particles in TFTR are well confined at high currents and slow down classically [37]. Alpha particle heating is expected to account for 15% of the power flow to the electrons in the core of TFTR DT discharges [38]. Careful comparison showed that the electron temperature was systematically higher in DT plasmas compared to D or T plasmas and that the difference was consistent with the expected alpha heating. There was a measurable increase in limiter tile temperature as the fusion power and alpha yield increased. The observed increase was consistent with estimates of alpha loss based on orbit simulations and indicated that there was neither an unexpectedly large fraction of lost alphas nor unexpected localization of the losses [39]. Bursts of alpha particle loss are sometimes correlated with MHD activity in the plasma [4]. In general these losses represent only a small fraction of the alpha population. At major disruptions however, losses of energetic alpha particles, estimated to be up to 10% of the alpha population, have been observed in a 2 ms period during the thermal quench phase while the total current is still unperturbed [37]. Such losses could have a serious impact on first wall components in a reactor.

In a fusion reactor, transport of helium ash from the core to the edge and pumping at the edge needs to be comparable to the ash production rate otherwise the burn will be quenched. TFTR experiments have provided the first opportunity to measure helium transport coefficients, and examine the effects of edge helium pumping on the central ash densities in D–T plasmas. Earlier helium transport studies on deuterium plasmas in TFTR [40] and DIII-D [41] have been reported. Measurements of radial ash profiles have been made using charge-exchange recombination spectroscopy and the results compared to predictions of the TRANSP code [42]. The on-axis helium ash source strength was comparable to that expected for ITER. Pumping of the helium is provided by the TFTR limiter [43]. The measurements indicated that the total alpha ash residence time in the TFTR vacuum vessel was 1.2 s, and this was dominated by edge pumping rates, not by core transport. The ratio of the helium ash confinement time (including recycling effects) to the energy confinement time, τ^*_{He}/τ_E, is 6–10 and is consistent with the requirements for a sustained fusion burn in a reactor.

The plasma performance of TFTR is surprisingly sensitive to the condition of the limiter. Improvements in core plasma parameters, such as the confinement time, often result from improvements in the plasma-limiter interaction [6,44–46]. Extensive limiter conditioning led to the discovery of the 'supershot' regime of enhanced confinement [47]. These early supershot plasmas had low plasma currents and low initial densities and achieved a factor-of-three higher ion temperature and deuterium fusion yield than the L-mode plasmas obtained previously. At that time, the maximum heating power was limited by carbon blooms (an uncontrolled massive influx of carbon). Improvements in the alignment of the limiter tiles extended the threshold for carbon blooms to greater than 32 MW for 1 s, a necessary prerequisite for the subsequent high power DT operations [48].

Despite the substantial reductions in carbon influx achieved, recycling from the limiter remains a major global fueling source for the plasma. Previous modeling of supershots in TFTR [49] has shown that the local recycling source remains larger than the beam fueling source from the half radius to the edge. While recycling is an important factor in core plasma composition, it is more difficult to control than the beam fueling. In the plasma core, neutral beam particles are the major fueling source, however recycling is still significant. Deuterium influx limited the range of isotopic mass that could be explored in a study of the isotopic scaling of transport [50]. The average hydrogenic mass in plasmas with tritium only beam injection was calculated to be $\langle A \rangle = 2.6$, short of the $\langle A \rangle = 3$ to be expected in a pure tritium plasma [here A is defined by the volume integral over the hydrogenic densities $\langle A \rangle = \int (n_H + 2n_D + 3n_T)/(n_H + n_D + n_T) dv$].

Plasmas where the neutral beam injection is solely tritium offer an interesting opportunity to study the transport of neutral deuterium recycled from the limiter, through the scrape-off region to the plasma core. Some deuterium flows back to the limiter and some penetrates to the plasma core and undergoes D–T reactions with the tritium injected by the beams. The resulting 14 MeV neutrons are detected with great sensitivity [51]. In plasmas fueled by balanced deuterium and tritium neutral beams, the D–T neutron rate has been found to be closely correlated with the plasma stored energy and edge q [52]. For tritium-only beam injection, the sole source of deuterium is the limiter and the D–T neutron rate is up to a factor 2 lower than the rate with balanced neutral beam fueling in a plasma with the same stored energy and edge q [53]. We use this comparison to derive an approximate value for the central density ratio, $n_D/(n_D + n_T)$.

Considering only deuterium, tritium and an effective reaction rate, $\langle \sigma_{DT} v \rangle$, the neutron rate is given by $S_{DT} = n_D n_T \langle \sigma v \rangle$. We compare this to balanced deuterium and tritium fueling at the same total density: $n'_D = n'_T = 0.5(n_D + n_T)$ with a neutron rate, $R_{DT} = 0.25(n_D + n_T)^2 \langle \sigma v \rangle$. The two equations result in a relation: $n_D/(n_D + n_T) = 0.5 - 0.5[1 - (S_{DT}/R_{DT})]^{1/2}$. From the measured neutron rate, S_{DT} and the value of R_{DT} derived from the scaling relation for a discharge with the same stored energy and edge q, we calculate the density ratio and plot it in Fig. 3.

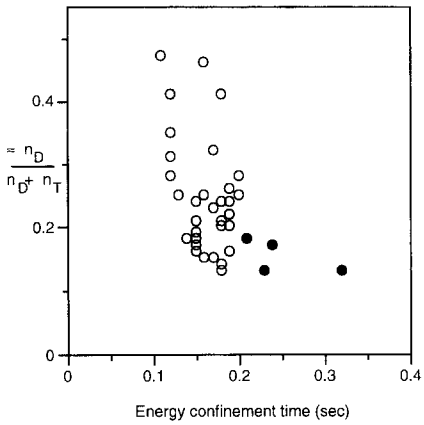

Fig. 3. Central deuterium density ratio in discharges fueled by tritium only neutral beams. The points, ●, have lithium conditioning.

The data in Fig. 3 are taken from a range of plasma conditions and show a deuterium density in the core varying from 15% up to the 50% limit inherent in the analysis. Plasmas with lithium conditioning have low influx of deuterium from the limiter to the plasma core and excellent performance in terms of energy confinement time. A complementary analysis of tritium transport and recycling in deuterium-only neutral beam fueled TFTR discharges based on modeling of the spatially resolved neutron rate is reported in Ref. [14] in this proceedings.

4. Effect of lithium conditioning on TFTR performance

Perhaps the most serendipitous discovery in TFTR was the beneficial effect of lithium pellet injection into the discharge [54]. Originally implemented as a current profile diagnostic, the injection of lithium pellets resulted in a significant increase in the neutron rate and confinement time during the beam heated portion of the discharge. Lithium pellet conditioning leads to reduced edge densities and improved neutral beam penetration, which then peaks up the density profile and improves confinement. The plasma performance (neutron emission, confinement time, and energy content) increases steadily with the lithium content on the limiter enabling an extension of the supershot regime to higher currents and higher confinement times [55].

Four principal benefits [56] have resulted from lithium conditioning of TFTR: (1) increased D–T fusion power (up to 10.7 MW) attainable at a given input power. (2) Lithium has significantly enhanced the energy confinement

time (up to $\tau_E = 330$ ms) and allowed τ_E to remain elevated for most of the 1 s duration of plasma heating by neutral beam injection. (3) Lithium conditioning has enabled supershot operation at higher values of plasma current (up to 2.7 MA) and stored energy, through its apparent elimination of the sawtooth instability at high plasma currents. (4) As a result of lithium conditioning, enhanced values of the Lawson product (up to $n\tau_E T \approx 10^{21}$ m^{-3} s keV), have been obtained while employing about one-half (10–25 MW) of the neutral beam power available to TFTR (40 MW). This value is $\approx 64\times$ higher than the Lawson triple product achieved in similar TFTR L-mode discharges. Fig. 4 shows the D–T fusion power for various levels of lithium conditioning. The fusion power multiplier ($Q_{DT} = P_{fusion}/P_{heat}$) reaches a value of 0.27. The value near the plasma center is even higher: $Q_{DT}(r \approx 0) \approx 0.6$–0.8. This elevated value of Q_{DT} is significant because it is a measure of the local importance of fusion alpha particles in the energy balance.

Fig. 5 shows the suppression of edge plasma density and material influx from the limiter with lithium conditioning. Reduced edge density and H_α and CII emissions have been observed to correlate with increased τ_E in supershots without lithium conditioning. Lithium conditioning has led to further reductions in these parameters. It can be noticed that the suppression of edge parameters associated with lithium conditioning begins to diminish after about 300 ms. Techniques to continuously resupply lithium to the limiter during neutral beam injection are under development. The most serious limitation of lithium conditioning on TFTR at present is the effective Troyon-normalized beta limit for disruptive instabilities. Higher fusion performance on TFTR will depend on techniques aimed at increasing the plasma stability. The effect of lithium conditioning on ion temperature and toroidal velocity profiles and therefore on transport is explored in Ref. [15].

The limiter surface itself is continuously undergoing

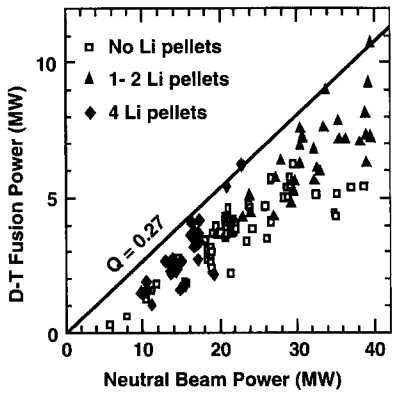

Fig. 4. Maximum D–T fusion power achieved in TFTR supershots plotted as a function of the neutral beam heating power for discharges which received no lithium pellets, (squares), one or two pellets (triangles) or four pellets (diamonds).

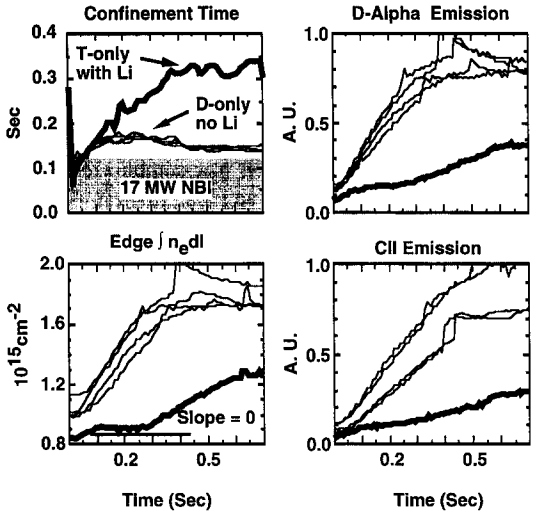

Fig. 5. Reductions in edge electron density and D_α and CII line emission and corresponding increase in energy confinement time for a T-only discharge with good lithium conditioning (thick line) and several D-only discharges for comparison (thin lines).

erosion and co-deposition by its interaction with edge plasma ions. In-situ microscopic observation of lithium interacting with the limiter is presently impossible. Interesting laboratory experiments on the gettering of oxygen and atomic hydrogen by lithium surfaces were reported by Sugai et al. [57]. Lithium coating was found to completely suppress the yield of CO from a graphite surface previously exposed to a He/O glow discharge. However, in the case of TFTR the oxygen impurity level is low, typically an order of magnitude below carbon, and oxygen is not expected to play a role in the plasma performance. A synergy in the chemical erosion of pyrolytic graphite from the simultaneous bombardment by H + and low Z ions (He^+, Ne^+ Ar^+) has been found by the Toronto group [58]. This is clearly a fascinating area of interdisciplinary research involving chemistry, solid state physics, and low and high temperature plasma physics.

A potential increase in the tritium inventory in tritium fueled tokamaks with a lithium coated first wall is a source of concern because of the strong chemical affinity of lithium for tritium. However an experiment addressing this question on the TdeV tokamak found that lithium conditioning was not likely to cause a pronounced increase in tritium vessel inventory in tritium fueled boronized devices [59].

5. Tritium Operations

In the period up to May 3rd 1996, 716 TFTR discharges have been actively fueled with tritium beams, puffs or dilute concentrations of tritium. The internal hardware has operated reliably without requiring an opening of

the vacuum vessel. Potential consequences of the higher neutron flux in DT plasmas include increases in noise, spurious signals and damage in the plasma diagnostics. TFTR has successfully dealt with these issues by a combination of relocation, compensation, and shielding techniques [60]. Neutron activation has complicated work at the tokamak. At vessel contact, the highest peak dose measured is 3.5 rem/h, and urgent system repairs have been performed near the vessel at a contact dose of 150 mrem/h. With careful planning, significant corrective maintenance tasks have been successfully accomplished with exposure levels an order of magnitude below that mandated by US Government limits.

As of May 1996, TFTR Tritium Systems have safely processed 730 kCi of tritium. The tritium system has been maintained and operated with virtually no radiological uptake, and without any lost time accidents. This has been accomplished a large margin within the environmental limits. The total stack releases were 139 and 62 Ci in calendar years CY 1994, 1995 respectively, compared to the environmental limit of 500 Ci/year. Release to the sanitary sewer was 0.29 and 0.5 Ci of tritium in CY 1994, 1995, compared to the 1 Ci/year limit. The site boundary dose including activated air, direct and scattered neutrons and gamma radiation was 0.30 and 0.31 mrem in CY 1994 and 1995, respectively, compared to the design objective of 10 mrem/year and the natural background level of 80 mrem/year. The collective personnel exposure in CY 1994 and CY 1995 was 3 person-rem each year compared to 4–8 person-rem/year in DD operations during CYs 1989–1992 (DT operation began November 1993).

Tritium operations are subject to strict administrative controls. The TFTR site is limited to 50 kCi of tritium in process. In addition, no more than 25 kCi, can be stored in any one place on the site. Accurate tracking of the active inventory is essential to ensure that the site inventory limits are not exceeded. Before tritium is physically moved from one location to the next, a tritium transfer operation number must be issued and the tritium is transferred from one account to the next. Tritium contamination of the TFTR facility has been kept to a minimum with the effective employment of negative pressure and moist air purging. The use of portable ventilation ducts (elephant trunks) has been highly effective in keeping tritium contamination levels low and personnel exposure as low as reasonably achievable. Extensive training and management support are central factors in this safety record.

The tritium storage and delivery pathway is shown in Fig. 6. Tritium is received in a gaseous form from the U.S. Department of Energy, Savannah River Site, in stainless steel LP-18 containers at ~ 300 Torr. The LP-18 containers are placed in the tritium receiving analytical glove box, and assayed for purity with a quadrupole mass spectrometer and beta scintillator (a purity level greater than 98% is required). Tritium is then transferred onto one of three (145 g) depleted uranium beds (U-bed) which are located

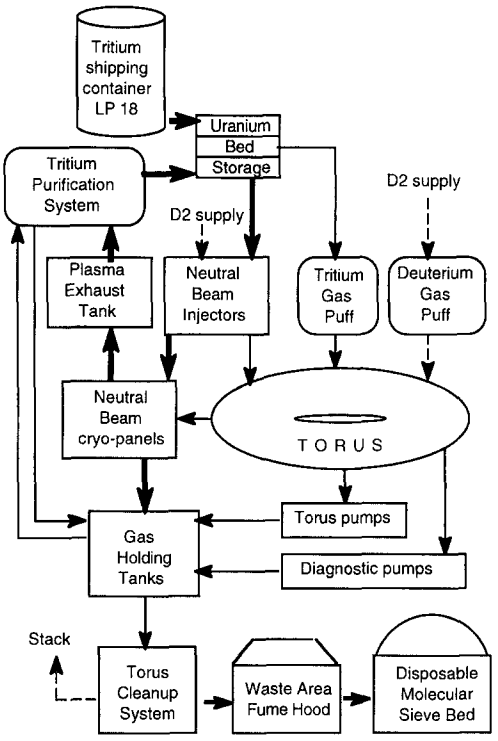

Fig. 6. Tritium daily fuel cycle.

in the tritium storage and delivery glove box. To support D–T plasma operations, the U-bed is heated up to 400°C, and the tritium gas is transferred to a calibrated holding volume (0.2 l) at pressures up to 1500 Torr. The contents of the holding volume are transferred to the tritium gas delivery manifold, a coaxially constructed line, 120 m long, which carries tritium to the combination of the twelve neutral beam tritium gas injectors and two torus tritium gas puff assemblies as required by the experiment.

Typically an inventory of 125 Ci is needed to establish the proper ion density in the ion source before beam extraction and 125 Ci for each second of injection into the torus [61]. Approximately 96% of the tritium supplied to the neutral beams boxes is captured by liquid helium cryopanels inside the beamlines without entering the torus. The exhaust from the torus is pumped by the neutral beam cryopanels and/or the torus pumping system Typically, after a week of operations, the neutral beam cryopanels are regenerated by being warmed up to a temperature where the hydrogenic species are evaporated and pumped to one of two 7.6 m^3 gas holding tanks. The holding tanks also collect the effluent from the torus pumping system.

The gasses accumulated in the holding tanks are processed by the 'torus cleanup system'. This system catalytically oxidizes the gaseous hydrogenic isotopes and tritiated hydrocarbons and deposits the resulting water onto disposable molecular sieve bed containers. Containers with less than 1,000 Ci of tritium are disposed at the Hanford

disposal site in Washington; containers with more than 1,000 Ci are processed at the Savannah River Site where the tritium is reclaimed and put back into the US Department of Energy tritium inventory. A cryogenic distillation system that will process the tritium exhaust on site is currently being commissioned at TFTR. This system has the capability to recycle a gaseous tritium stream back to the holding volumes in the tritium receiving analytical glove box where, if the gas is greater than 98% pure, it can be redeposited onto the U-beds. If less than 98% pure it can be deposited on portable U-beds and shipped back to the Savannah River Site for recycling. A detailed account of tritium processing and management on TFTR is given in Refs. [61–63].

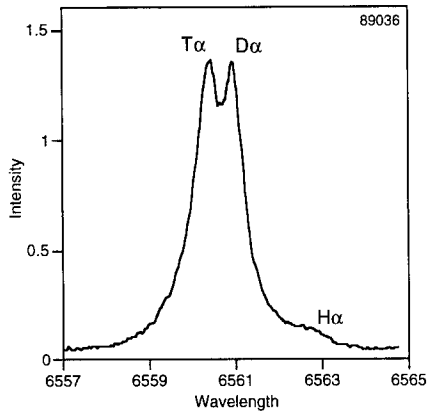

Fig. 7. Spectral profile from a discharge with a tritium gas puff showing recycling of all three hydrogen isotopes.

6. Tritium recycling

DT operation has provided a special opportunity to study tritium recycling in TFTR. Recycling can be observed in Balmer-alpha emission from neutral hydrogen isotopes in the inner plasma edge region where the plasma contacts the toroidal graphite limiter. Neutral tritium has a Balmer-alpha transition, T_α, that is analogous to the H_α and D_α transitions long used in edge plasma diagnostics. The first spectroscopic measurements of T_α emission from a fusion plasma were made on TFTR using a Fabry–Perot interferometer [64]. The high resolution and throughput of the Fabry–Perot are advantageous to resolve the tritium and deuterium Balmer alpha lines which have an isotopic separation one third that of hydrogen and deuterium. Measurements of the relative fraction of the hydrogenic isotopes reflect the hydrogenic inventory in the surface of the limiter. This inventory, in turn, is a function of the fueling history of prior discharges, codeposition and the mobility of hydrogen isotopes in graphite. The observed tritium fraction in the Balmer-alpha spectral profile was low ($\leq$ 11%) in tritium neutral beam heated discharges but increased up to 75% with tritium gas puffs. Spectral resolution of the line profile gives information on processes in the edge plasma. A spectrum showing emission from all three hydrogen isotopes during an L-mode campaign with strong tritium puffing is shown in Fig. 7. The emission is Doppler shifted and the spectral line profile maps the velocity distribution of neutral hydrogen isotopes. Since the edge density is insufficient to thermalize the velocity distribution, the contributions of the various reaction pathways that generate hydrogenic atoms may be identified with the different wavelength regions (velocities) in the spectral profile. Overall agreement was found between the observed spectral profile and that predicted by the neutral transport code DEGAS [65]. A detailed account may be found in Ref. [13] in this proceedings.

Surface tritium may also be detected through the secondary electrons generated by the beta particles produced in its radioactive decay. This is the operating principle of most commercial tritium detectors. The TFTR vacuum vessel was made into a large ionization chamber [66] by filling it with nitrogen gas and biasing a small electrode near the wall up to +30 V with respect to the grounded vessel. The measured electron current peaked at a pressure of 4 Torr and was 40 μA at +20 V consistent with the detection of about 80 Ci of tritium. Since the range of beta particles in carbon is on the order of 1 μm, this method is only sensitive to tritium in a surface layer of this thickness. Also some electrons may be deflected in the 10 G dc magnetic field present in the torus and not be detected. If this method can be absolutely calibrated and modeled it may provide a simple way to monitor the surface tritium inside the ITER vacuum vessel.

7. Retention

An overall deuterium retention fraction of 45% was measured prior to the introduction of tritium [67]. This fraction is the ratio of the deuterium in-vessel inventory, estimated from measurements of graphite tiles and wall coupons removed from TFTR, to the total deuterium fueling. Of this amount, 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 (Fig. 8). Comparison of different run periods showed that the retention varied in the range 22–63% and the variation was correlated with the average neutral beam power. This pattern was attributed to increased edge and limiter temperatures leading to more codeposition of deuterium with carbon at high beam powers [68]. 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 by D–D nuclear reactions was also studied. Tile analysis [69] indicated that $\approx$ 35% of the tritium produced was retained in the internal components of TFTR.

In the first phase of the DT campaign in December

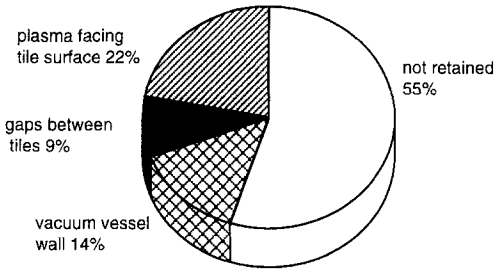

Fig. 8. The long term fate of deuterium injected into TFTR [68].

1993 an inventory of 360 Ci of tritium was injected into the torus by 21 mixed deuterium/tritium neutral beam heated discharges interspersed with 55 deuterium neutral beam heated discharges. Following this campaign, a short clean-up experiment of 34 deuterium ohmic discharges and deuterium neutral beam injection discharges was performed in which less than 80 Ci was recovered [70]. Although the measurements had a relatively high level of uncertainty, they indicated a high short-term retention level.

The tritium accounting system tracks the long term movement of 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. In addition, a trace amount (203 Ci) of tritium was puffed directly into the torus in experiments to study transport. 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. [67–69]. The administrative limit on the in-vessel inventory is 20 000 Ci.

8. Tritium injection during L-mode experiments

In September 1995, large quantities (10 600 Ci) of tritium gas were puffed directly into discharges to generate L-mode tritium plasmas for studies of the isotope scaling of L-mode transport and on RF heating at the tritium 2nd harmonic cyclotron frequency. 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. One of the four neutral beam lines was reserved solely as a pump for the tritium exhaust and had no tritium gas feed. In this way, 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 was avoided. 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 beamiines are identical so that 1/4 of the tritium exhaust was pumped by the reserved beamline. The cryopanels were regenerated approximately every second day, and a measurement was 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 tritium input and exhaust is summarized in Fig. 9. The plasma operations are arranged in chronological order and listed in a highly abbreviated way. On 30 August 1995 the value of in-vessel inventory, given by the tritium accounting system, stood at 6,863 Ci. At the beginning of September 1995 there were 8 days of L-mode operation in which 10,600 Ci of tritium were puffed into the discharge and 900 Ci injected via the neutral beams. It should be noted that no attempts were made to remove tritium in this period, on the contrary, the aim was to maximize the tritium concentration in the plasma to identify the isotopic dependence of transport. Only an estimated 860 Ci were recovered in this period indicating a short term retention above 90%. The following 2 days of conditioning and 9 days of 'supershot' plasma operation were aimed at restoring the limiter to a low recycling state by ohmic and

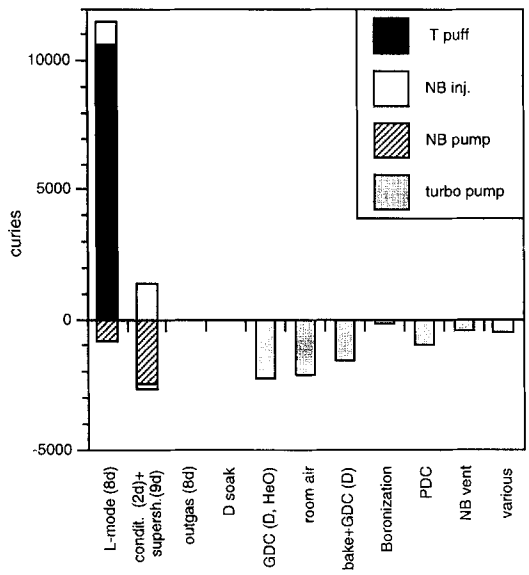

Fig. 9. Summary of tritium injected into the torus (positive bars) and measured exhaust (negative bars) during September/October 1995. 'GDC' represents glow discharge cleaning. In the legend 'turbo pump' refers to the torus vacuum pumping system, 'NB pump' to tritium recovered from the neutral beam cryopanels, 'NB inj.' is tritium injected via the neutral beams (including cold gas), and 'T puff' is tritium gas directly puffed into the torus. The horizontal axis refers to plasma operations, ordered chronologically: e.g., 'L-mode (8d)' refers to 8 days of L-mode operation.

neutral beam conditioning. Carbon blooms [71] were encountered when the neutral beam power was raised, but the power threshold at which the blooms occurred increased over time. Although no further tritium was injected into the plasma either directly or by neutral beams, tritium recycling and the D–T neutron rate remained high for several days. Tritium continued to be present in the exhaust and an estimated 970 Ci was collected. In the period 19–26 September, an additional 1370 Ci was injected via the neutral beams, however no further tritium gas puffing was used. On 22 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 1500 Ci, most of which was presumably released during the disruption.

Overall, a total of 12 900 Ci was injected by gas puff and neutral beams and 3500 Ci recovered in the period from 1 September until the end of the run on 26 September. It is possible that some of the tritium recovered 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. The retention fraction is then defined as the ratio of the cumulative tritium injected into TFTR during September–November 1995, less the cumulative exhaust, divided by the cumulative tritium injected. The retention fraction on 26 September was 73%. More details are given in Ref. [72].

9. Tritium removal

Various techniques have been used to reduce the in-vessel tritium inventory. Over the 8 days following 26 September there were no machine operations and outgassing of the torus yielded only 36 Ci. A deuterium 'soak' had proved effective in JET [73] for removing tritium from the torus that was maintained at 300°C, and 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 1 h. However negligible tritium was recovered (< 14 Ci) indicating that this technique was ineffective in TFTR. The difference may be related to the mobility of hydrogen isotopes in graphite at the different temperatures. Next a glow discharge sequence was undertaken using both deuterium and a mixture of 90% helium and 10% oxygen. The tritium removal rate of deuterium glow discharge (D-GDC) was initially high (360 Ci/h), however, after 59 h, it declined to a negligible rate (10 Ci/h). In contrast the removal rate of the helium–oxygen glow was a constant 50 Ci/h. More details are given in Ref. [12] in this proceedings.

The helium–oxygen glow chemically reacts with the graphite and removes it in the form of carbon monoxide and carbon dioxide. In contrast, the action of the deuterium glow is limited to the penetration depth of the deuterium ions. The initially high removal rate of deuterium glow discharges may potentially be useful for reducing the tritium inventory. Tritium injected during a day's plasma operations will be localized close to the surface of the graphite. The initial rate of tritium removal by D-GDC in TFTR matches the tritium injected by a cumulative 46 MW/s 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 would need to be followed by conditioning [6] to restore the limiter to a low recycling state.

An important consideration for TFTR, and indeed for a fusion reactor, is the fraction of the in-vessel tritium inventory that would be released in the event of a credible vacuum accident. Laboratory experiments [74,75] have shown that a sizable fraction of tritium in a codeposited film of carbon and tritium is released upon exposure to air. To determine this fraction in TFTR, 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 2086 Ci of released tritium, approximately 15% of the total 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 effect at 18 Torr). 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. 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 controlled torus vent prior to the bakeout and conditioning (which further depleted the tritium). Of the 7163 Ci remaining tritium in the in-vessel inventory on 2 January 1996, 5000 Ci is considered to be tenaciously held with respect to accidental release scenarios and is accounted for separately, apart from the material in process. The level of potentially releasable tritium is much less than the 20 000 Ci administrative in-vessel limit and is not a constraint on operations.

Following pumpout, the normal procedure for recovery from a vessel vent was followed [76]. 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, 1610 Ci were removed during the 55 h D-GDC, 169 Ci were removed during the 12 h boronization. PDC was used to further heat the

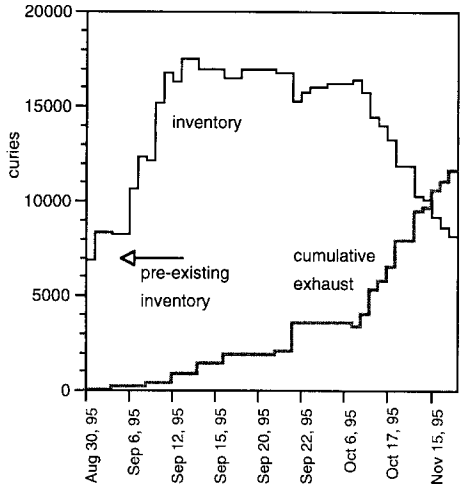

Fig. 10. In-vessel inventory and measured exhaust in September–November 1995. Note that the release of tritium is continuous but it was measured periodically.

limiters to 250°C (the administrative limit due to thermal stresses) for a 23 h period resulting in the release of 956 Ci. This procedure 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 observation illustrates that tritium removal is more effective if done promptly after exposure to tritium discharges. The various tritium removal activities extracted 8300 Ci from the initial value of the in-vessel inventory of 16,440 Ci.

An assay was taken of two other potential sites for tritium retention. The tritium content in the oil of four turbopumps in the torus pumping system was measured to be in the range 7–32 mCi/l. Applying the highest number 32 mCi/l to include other unsampled pumps, the total tritium inventory in 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. 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 change in the tritium inventory and cumulative exhaust are shown in Fig. 10. The steps in the lines reflect the periodic measurements of the tritium. One can see the large initial rise in the tritium inventory due to the large tritium puffs and the slow decline in subsequent plasma operations without tritium puffs. 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.

10. Current program on TFTR and plans for '97, '98

The TFTR advanced performance (TFTR-AP) project extends the operational phase of TFTR to develop and exploit its capabilities to investigate highly reactive DT plasmas. The primary physics mission of TFTR-AP is to use techniques for controlling the current and pressure profile to produce plasma regimes with increased DT fusion performance, in order to allow a thorough investigation of the physics of fusion alpha particles in advanced tokamak regimes at reactor-relevant parameters. The physical processes that give rise to the dramatically improved particle confinement of the enhanced reversed shear mode [9] will be studied through comparison of effects associated with driven rotation. The physics of 'internal barriers' produced inside the plasma by radio frequency waves will be investigated. These reduce energy and particle losses, while providing the plasma control required for more efficient and lower-cost fusion power sources. The current profile will be controlled directly through the use of mode-conversion current drive. The pressure profile will be controlled through the application of ion-Bernstein waves, which have been shown to induce a local transport barrier in PBX-M [77], consistent with theoretical modeling. This pressure-profile control will also provide an extremely important indirect control on the current profile, through the dominant bootstrap current. Alpha channeling [11] offers a promising path to an economically superior tokamak reactor. We plan to elucidate key aspects of the relevant physics through studies of the interaction of alpha particles with ion-Bernstein waves.

A vacuum opening is planned for September 1996 to upgrade two of the four ICRF antennas to four strap configurations to improve the directionality and power of the ion-Bernstein waves interacting with the alpha particles. We will use deuterium glow discharges, pulse discharge cleaning and room air soaks to remove tritium. At present no personnel entry into the vacuum vessel is planned.

Techniques to increase the amount of lithium introduced are being developed. So far the plasma performance has improved with increased amounts of lithium pellets injected but we are at the limit of this technique. We plan to introduce lithium from an oven inserted into the plasma edge and also to explore the laser ablation of lithium from a crucible in the vacuum vessel. These techniques promise to increase the amount of lithium introduced by one to two orders of magnitude.

11. Summary

TFTR's capability to carry out deuterium–tritium experiments has allowed the first studies of the underlying physical processes in a reacting plasma. Operation with DT fuel has demonstrated plasma performance with power densities comparable to that in proposed burning plasma devices. Similarly, plasma wall interaction and helium transport studies on TFTR have produced, and continue to produce results important to future reactor relevant devices.

Lithium conditioning has significantly enhanced the energy confinement time and enabled supershot operation at higher values of plasma current and stored energy. As a result of lithium conditioning, enhanced values of the Lawson product (up to $n\tau_E T \approx 10^{21}$ m^{-3} s keV), have been obtained while employing about one-half (10–25 MW) of the neutral beam power available to TFTR (40 MW). The fusion power multiplier reached a value of $Q_{DT}(r \approx 0) \approx 0.6$–$0.8$ near the plasma center. This elevated value of Q_{DT} is significant because it is a measure of the local importance of fusion alpha particles in the energy balance.

TFTR has demonstrated that safe, routine operation with tritium in a fusion device is feasible, with releases well below limits set by regulations. Tritium retention in TFTR is highly dependent on the machine operations and the time scale of interest. Over the time scale of a few days, a high level of retention ($> 90\%$) was found in the September 1995 L-mode campaign. Glow conditioning, a room air 'soak', a major disruption and PDC were successful in removing the tritium introduced in this campaign. Tritium removal was most effective if done promptly after exposure to tritium discharges, presumably before 'burial' under codeposited layers. A deuterium 'soak' and simple outgassing in vacuum were ineffective in removing tritium. However during plasma operations the tritium continues to be exhausted at a 'background' level of around 25–75 Ci/day. The 'global' retention measured by the tritium accounting system for the prior 2 year period was approximately 40%.

TFTR will continue to address issues of burning plasmas and advanced tokamak physics that have potential to alter the framework of plasma–surface interactions in future reactors.

Acknowledgements

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