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Plasma-surface interactions in TFTR DT experiments

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Abstract

TFTR has begun its campaign to study deuterium-tritium fusion under reactor-like conditions. Variable amounts of deuterium and tritium neutral beam power have been used to maximize fusion power, study alpha heating, investigate alpha particle confinement, and search for alpha driven plasma instabilities. Additional areas of study include energy and particle transport and confinement, ICRF heating schemes for DT plasmas, tritium retention, and fusion in high β_p plasmas.

The majority of this work is done in the TFTR supershot confinement regime. To obtain supershots, extensive limiter conditioning using helium fueled ohmic discharges and lithium pellet injection into ohmic and neutral beam heated plasmas is performed, resulting in a low recycling limiter. The relationship between recycling and core plasma confinement has been studied by using helium, deuterium and high-Z gas puffs to simulate high recycling limiter conditions. These studies show that confinement in TFTR supershots is very sensitive to the influx of neutral particles at the plasma edge.

1. Introduction

TFTR is engaged in a campaign to study issues relevant to the development of fusion as a source of power. As part of this campaign, experiments using deuterium and tritium (DT) as fuel under reactor-like conditions are under way. The initial DT experiments began on November 12, 1993, using low concentrations of tritium (2% T) in gas puff fueling and neutral beam heating experiments. On December 9, 1993, DT experiments commenced using deuterium-tritium concentrations in the range of 50:50 [1,2]. Since that time, a number of experiments have been performed using a wide range of tritium-to-deuterium fueling concentrations with some approaching 100% tritium.

The experimental program encompasses a wide range of topics, including:
- Maximizing fusion power production.
- Isotope scaling of energy confinement time.
- Plasma heating by fusion produced alpha particles.
- Transport and confinement of fusion produced alpha particles.
- Investigation of collective instabilities driven by fusion alpha particles.
- ICRF heating in DT plasmas.
- High β_p plasma studies in DT.
- Limiter H-mode experiments in DT.
- Particle transport and recycling in DT plasmas.

This list is not exhaustive and many additional studies and experiments are in progress.

All of the DT experiments have been performed using one of the three enhanced confinement regimes available to TFTR: supershots [3], high β_p plasmas [4], and limiter H-mode plasmas [5]. These enhanced confinement regimes have been obtained by careful design of the main power absorbing limiter (the inner Bumper Limiter) and extensive limiter conditioning utilizing plasma conditioning techniques [6] and lithium pellet injection [7]. Typical machine and plasma parameters for DT discharges in TFTR are the following:

$1.0\ \text{MA} < I_p < 2.0\ \text{MA}$,
$4.0\ \text{T} < B_T < 5.0\ \text{T}$,
$2.45\ \text{m} < R_p < 2.63\ \text{m}$, $0.80\ \text{m} < a_p < 0.98\ \text{m}$,
$P_{NB}(\text{D}+\text{T}) \leq 30\ \text{MW}$, $P_{NB}(\text{T}) \leq 23\ \text{MW}$.

Section 2 of this paper presents an overview of the main areas of investigation for DT studies in TFTR, Section 3 discusses particle transport and recycling, Section 4 deals with the techniques used to condition the limiter and vacuum vessel.

2. Overview of TFTR DT experiments

2.1. High power DT supershots

Presently, two tokamaks have produced plasmas using deuterium and tritium as fuel: the Joint European Torus (JET) with two discharges at 13% tritium-to-total fueling [8], and TFTR with 78 discharges at tritium-to-total fueling up to 100%. This section presents a comparison of the differences between DD and DT plasmas in TFTR and some of the results of the high fusion power DT discharges.

TFTR attained 6.2 MW of fusion power on December 10, 1993 [1,2] and achieved its 5 MW goal for initial DT operation. These plasmas were in the "supershot" regime at a plasma current of 2.0 MA, toroidal field of 5.0 T, major radius of 2.52 m, minor radius of 0.87 m, and density fueling by neutral beam injection only.

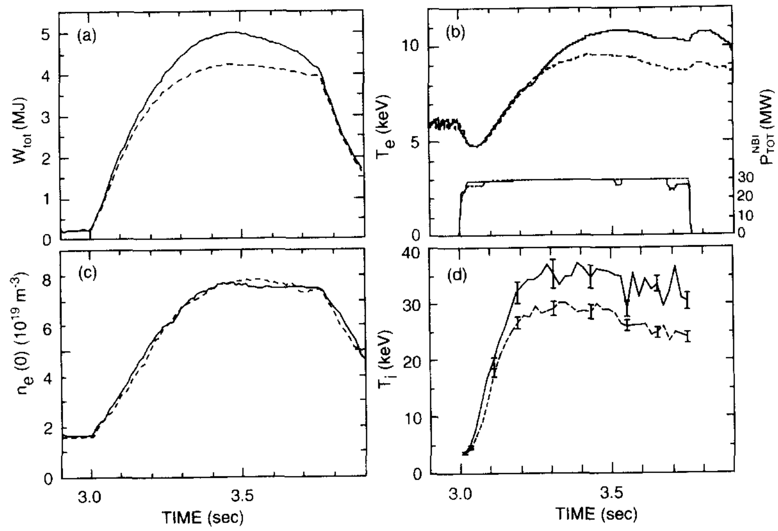

Fig. 1. (a) Magnetic measurements of the total stored energy for a DT discharge (solid line) and a comparable DD discharge (dashed line). (b) Central electron temperature (ECE emission) for a DT and DD plasma and the injected beam power for each case. (c) The central electron density for the DT and DD comparison. (d) The central carbon ion temperature from charge exchange spectroscopy for the DT and DD shots.

One or two lithium pellets were used after NB injection during this series of discharges to condition the limiter and improve machine performance. Neutral beam injection was approximately balanced at 90–107 keV. The highest fusion power was obtained with 30 MW of NB power with 7 sources injecting T and 4 sources injecting D from the 12 available sources. Tritium rich injection was necessary in order to obtain a 50:50 mix of T and D in the plasma core due to the apparently very low recycling of T from the limiter compared to D (Section 3).

A consistently observed difference between DD and DT plasmas at the same NB injection power is the higher stored energy, ion temperature and electron temperature in DT discharges, Fig. 1. If DT and DD plasmas have the same transport, then a higher stored energy in DT is expected due to the longer slowing down time of the NB injected T ions, energy stored in the fusion produced alpha particles and alpha heating [9]. However, these effects do not appear to account for the entire magnitude of the stored energy increase. There also appears to be an increase in energy confinement time with average hydrogenic ion mass, Fig. 2. The average hydrogenic ion mass in the plasma core is determined from modeling of the H, D, T, and carbon ion densities using the measured Z_{eff}, plasma density, neutral beam fueling, neutron fusion rate, and edge H/D/T influx from the limiter.

Another aspect of the TFTR DT program critical to fusion power production is the observation of alpha particle heating. The observation of direct alpha particle heating of the plasma electrons is difficult due to the apparent increase in energy confinement time in DT plasmas. Is the observed increase in electron temperature in DT compared to DD due to alpha heating, improved energy confinement, or both? Present studies indicate that alpha heating is occurring using SNAP and TRANSP calculations [1] and that thermal energy confinement has improved.

A more direct observation of alpha heating is shown in Fig. 3 where the central electron reheat is shown after Li pellet injection. The pellet was injected approximately 0.22 s after the end on NB injection. By this time, most of the beam ions have thermalized while the alpha particles have not due to their longer

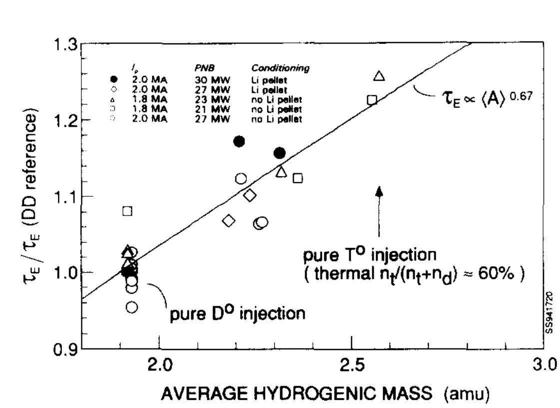

Fig. 2. Magnetically measured global energy confinement time normalized to the DD confinement time versus average hydrogen isotope ion mass in TFTR.

slowing down time. Compared to a similar DD discharge, the electron reheat in the DT plasma is 85% faster, in agreement with TRANSP calculations.

2.2. High β_p and limiter H-mode studies

The high β_p plasmas produced recently on TFTR as part of the DT experimental program naturally evolve into limiter H-modes. These plasmas are characterized by confinement enhancements greater than 4 times ITER-89P and $\beta_N = 3$. With DT neutral beam fueling, fusion power in excess of 4.2 MW and fusion power gain, $Q_{DT} = 0.18$ have been achieved. A comparison of the DD and DT versions of these discharges is shown in Fig. 4.

In order to produce these enhanced confinement discharges, reduced edge recycling is needed. This was accomplished by either Li pellet injection or transition to a limiter H-mode. (Note that in TFTR, limiter H-modes have been made only with the limiter reasonably well conditioned with less than unity recycling.) A larger increase in energy confinement time is obtained during the ELM-free portion of a DT H-mode plasma than in a comparable DD plasma, Figs. 4c and 4d. Another difference between DD and DT plasmas is that there is a greater drop in D_α emission during the ELM-free phase and an earlier onset of Elm's in the DT discharges. Similar effects were observed in DD discharges produced just after a DT discharge. The residual tritium recycling from the limiter apparently modifies the H-mode behavior. These effects decreased as the tritium recycling declined with continued DD operation.

MHD activity prior to β collapse and disruptions, and the characteristics of Elm's are similar in DD and DT plasmas. However, large amplitude, low frequency

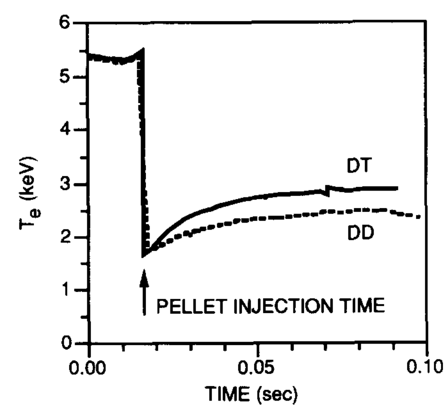

Fig. 3. The central electron reheat after Li or B pellet injection for a DT and DD plasma. The faster reheat in the DT plasma is attributed to alpha particle heating of the electrons.

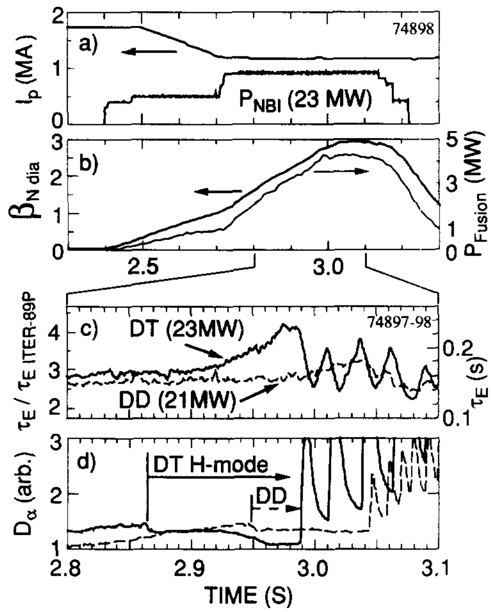

Fig. 4. Comparison of a DD and DT high β_p plasma which evolve into a limiter H-mode plasma.

(40 Hz) "giant" Elm's are more likely to occur in DT plasmas which have relatively broad pressure profiles and high β_N, as shown in Fig. 4d. Small, 10%, fluctuations in fusion product loss are associated with these Elm's.

2.3. ICRF DT heating studies

The TFTR DT plan attempts to address many issues relevant to future reactor-grade plasmas. Amongst these issues are those that involve ICRF heating of DT plasmas. Two principle goals are being addressed, first the use of ICRF heating in conjunction with NBI driven supershots to increase the alpha particle beta, and second the study of the various damping mechanisms available in a DT plasma for ICRF heating. Experiments have already been conducted in both of these areas.

A supershot target plasma with 24 MW of NBI injection was prepared with 14 MW of T NBI and 10 MW of D NBI yielding a plasma that was roughly 35-40% tritium. ($I_{TF} = 68$ kA, $I_p = 1.8$ MA, $R = 2.62$ m). ICRF power at a frequency of 43 MHz was applied at power levels up to 5 MW. This frequency corresponds to the second harmonic tritium cyclotron resonance being roughly at the Shafranov shifted magnetic axis. These conditions also place the fundamental ^{3}He resonance at the same location. Some discharges had added ^{3}He (to increase damping) and others did not. The added RF heating resulted in an increase in the central electron temperature of ~ 2 keV and in the central ion temperature of ~ 5 keV. It was originally planned to extend the NBI and rf pulse length to maximize the impact on alpha beta but this was not done since insufficient tritium was available and plasma performance indicated that extending the pulse length would not achieve the desired physics aims. The discharges were deteriorating in time due to MHD, T dilution, and the loss of peak performance resulting from the plasma-limiter contact point moving as a result of increased stored energy. These plasmas were produced without shaping fields but had insufficient stored energy to achieve the desired elongation and limiter contact area.

To test the competition between second harmonic tritium damping, mode conversion into IBW waves, direct electron damping via the Landau mechanism and high field side fundamental deuterium damping, a series of discharges was run varying the ratio of D to T NB injection, the amount of ^{3}He minority, and the toroidal field. The rf power waveform was amplitude modulated at both 5 and 10 Hz to assist in determining the competition between the various mechanisms. Preliminary analysis of the data indicates tritium absorption at levels commensurate with theoretical estimates, some direct electron heating and little evidence for the high field side deuterium damping predicted in some codes.

During ^{3}He minority heating studies of DD and DT plasmas, there was no observable difference in antenna loading between these two conditions. Measurements indicate that for comparable DD and DT discharges, the parameters important for ICRF plasma-launcher interaction, such as heat load on the launcher, plasma edge density, and impurity generation are nearly the same.

Future experiments will continue these investigations and also look at other ITER relevant heating and current drive schemes including a more favorable mode conversion scheme for both heating and current drive that has looked very promising in ^{3}He–^{4}He plasmas, and conventional FWCD in the presence of alpha particles.

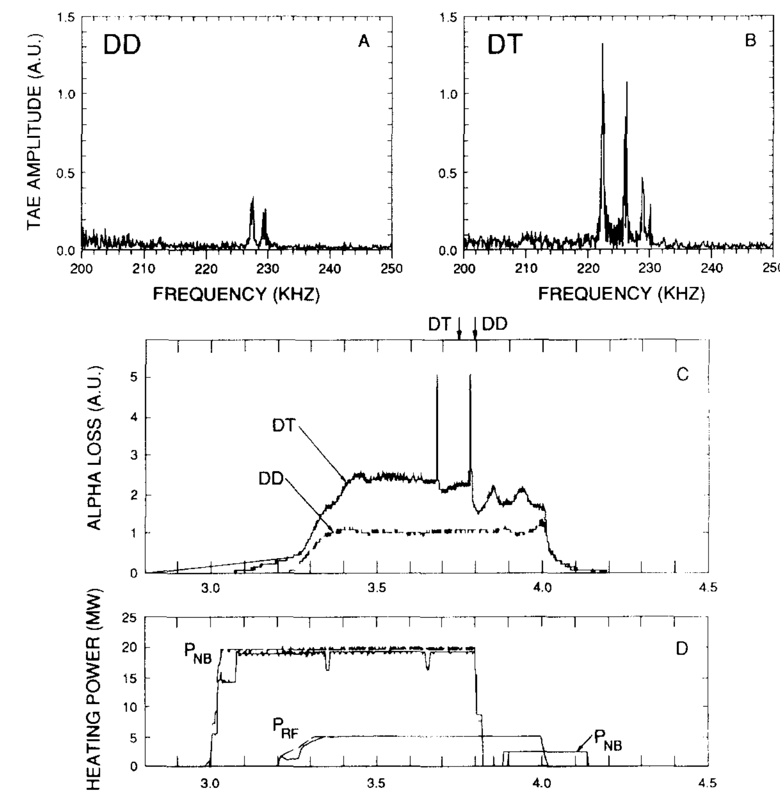

Fig. 5. (a) TAE amplitude driven unstable by ICRF in a deuterium neutral beam heated plasma. (b) TAE amplitude driven unstable by ICRF at the same power level as (a) in a DT plasma. (c) High energy particle loss for the cases shown in (a) and (b). The solid curve is DT and the dashed curve DD. The times that the spectra in (a) and (b) were measured are indicated by the arrows. (d) Injected rf and NB power for the cases shown in (a) and (b).

2.4. TAE studies

The energetic alpha particles created in a DT fusion reaction in a tokamak have speeds greater than the Alfvén velocity and are expected to couple to shear Alfvén waves [10]. Under certain conditions which may be present in a reactor, an instability termed toroidal Alfvén eigenmode (TAE) may be excited which would have deleterious effects on plasma confinement and alpha particle confinement, in particular.

Two approaches have been taken to study TAE in TFTR DT plasmas. The first uses ICRF minority heating to create a high energy hydrogen particle population which destabilizes the TAE in deuterium NB heated plasmas. In a subsequent discharge, deuterium–tritium NB injection is used to create energetic alpha particles which contribute to destabilizing the TAE instability driven by the energetic H ions. The ICRF power can be varied on a shot-to-shot basis in order to study the destabilizing influence of the energetic alpha particles compared to the energetic H ions.

The second approach attempts to excite the TAE instability by using only energetic alpha particles and reducing the ion Landau damping of the instability by lowering the ion temperature of the plasma, thereby making the alpha particle drive more effective.

2.5. ICRF assisted destabilization of TAE in DT plasmas

Toroidal Alfvén eigenmodes (TAE) can be excited by energetic ions with velocities comparable to the Alfvén velocity, like the alpha particles in a tokamak reactor. These fast particles can be ejected by the TAE modes and cause localized heating on the first wall. Simulation experiments with fast ions from neutral beam injection and ICRF heating have been performed in TFTR in the past few years. Experiments with alpha particles became possible recently with the use of tritium neutral beam injection. At the writing of this report, the alpha particles alone have not excited the TAE instability in TFTR. Since the driving term consists of the sum of all the fast ions, ICRF heating is used to produce more fast ions to assist TAE excitation. Up to 5.5 MW of ICRF power is applied to a 1.8 MA plasma at 4.8 T magnetic field. The 64 MHz rf frequency matches the cyclotron frequency of the minority hydrogen ions in the plasma core so that they can be accelerated to very high energies. TAE modes can be excited by these fast ions in deuterium plasmas fueled and heated by 20 MW of deuterium neutral beams, with a threshold rf power of 5.1 MW. When 15 MW of deuterium beams are replaced by tritium beams, the threshold rf power is reduced to a value slightly lower than 4.0 MW. Part of this difference may be due to the alpha particles. The situation is complicated by the various ion species interacting with the ICRF waves and the TAE modes.

At an rf power of 5.2 MW, TAE modes are excited in DD and DT plasmas. The TAE amplitude is significantly larger in DT plasma as shown in Figs. 5a and 5b. The waveforms for the neutral beam power and the ICRF power are shown in Fig. 5d. With very similar heating power, the fast ion detector at the plasma edge observes a larger fast ion loss flux in the DT plasma as depicted in Fig. 5c. This correlates well with the difference in the TAE amplitude.

2.6. Search for TAE modes at low T_i and beta

The goal of this experiment was to excite TAE mode activity in TFTR DT supershots by reducing the ion temperature or the background plasma beta, which should reduce the ion Landau damping and continuum damping and therefore lower the threshold for the TAE modes, according to some theoretical models [11]. The strategy for this experiment was to reproduce the plasma which had the record DT neutron rate, and then to apply various cooling perturbations during NBI in order to transiently retain the highest possible alpha pressure at low T_i and beta.

Four high power DT discharges with up to 4–5 MW of fusion power have so far been made for this experiment, each with a different type of cooling perturbation 0.6 s after the start of NBI. Two had He puffs during the NBI, which had previously been seen to rapidly change supershots into L-mode plasmas, and two had injected pellets, one with deuterium and one with Li. The ion temperature in each case dropped from $T_i = 20$–25 keV to $T_i \leq 10$ keV within 0.2 s of the cooling perturbation, with a different density, electron temperature, and pressure profile evolution for each. The calculated alpha pressures at the time when the plasma reached $T_i = 10$ keV were greater than half of that before the perturbation, as expected.

The alpha-driven TAE mode has not yet been observed in these discharges. There was no increase in alpha particle loss, as might be expected during TAE activity, nor was there any significant increase in the fluctuation activity in the Alfén range of frequencies. Preliminary theoretical calculations of the TAE stability for these discharges using the PPPL Nova-K code indicate the dominance of deuterium beam ion damping, which was not decreased enough by these perturbations to cause instability. Follow-up experiments will try to induce TAE instability by raising $q(0)$ to move the gaps closer to the region of maximum alpha pressure gradient near $r/a = 0.2$.

As in high performance DT supershots at constant plasma current, TAE instabilities have also not been observed in high β_p, DT plasmas.

2.7. Alpha confinement in DT plasmas

The alpha particle loss during DT plasmas was measured with the lost alpha scintillator detector system previously used for DD fusion products [12]. At a given plasma current, the "classical" alpha loss rate should be proportional to the DT neutron rate, since one alpha is created for each neutron. However, if the alpha loss rate per neutron were to increase with increased DT reaction rate, this could be a sign of a new "collective" alpha particle driven instability.

No sign of any such increase in alpha loss with increased DT reaction rate has yet been seen on TFTR up to the maximum fusion power level of 6.2 MW, or 1.2 MW of alpha power. This is consistent with the absence of any increase in the fluctuation level in the Alfén range of frequencies, as measured by magnetic pickup loops and a microwave reflectometer. Thus the presence of a significant alpha pressure in the plasma core of $\beta_\alpha/\beta \approx 5\%$ has so far not excited any TAE modes or other collective alpha instabilities [2].

Data from the alpha loss detector located 90° below the midplane is shown in Fig. 6. The alpha loss per neutron decreases with increased plasma current very nearly as expected from the classical first-orbit loss model. The data in this figure are normalized to the calculated loss fraction into this detector at the lowest current of $I_p = 0.6$ MA, where the first-orbit loss is expected to dominate. An absolute calibration of the system also shows that the loss at all currents is consistent with the classical calculation to within its factor-of-two uncertainty [13]. Note that the alpha loss per neutron during the highest-power DT discharges at $I_p = 2.0$ MA is not significantly higher than in partial DT in discharges at the same current with $\leq \frac{1}{10}$ the DT reaction rate. The pitch angle, gyroradius, and time dependencies of the alpha loss to this detector are consistent with the classical first-orbit model for all discharges in this figure.

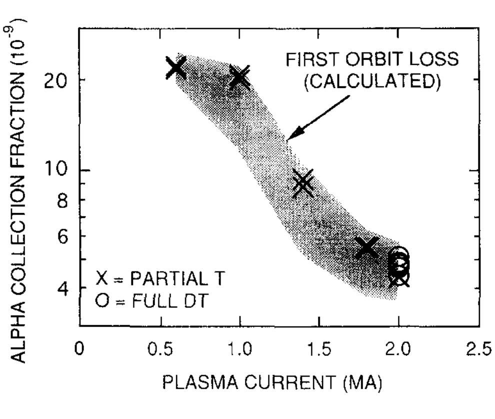

Fig. 6. Alpha particle loss in partial and full DT plasmas as a function of plasma current compared with the calculated first-orbit loss normalized to the data at 0.6 MA.

ICRF has had noticeable effects on alpha confinement during DT operation in TFTR. Alpha particles have the same gyrofrequency as deuterium, and so can interact with various ICRF heating and current drive scenarios in TFTR. This interaction can potentially be useful to control alpha particle burn and ash buildup in future DT reactors, but could also cause an undesirable transfer of energy to the alphas, or an increased alpha particle loss to the wall.

This interaction was seen for the first time during the ICRF experiments on second harmonic heating of tritium in DT plasmas. In these supershot plasmas with 20 MW of NBI at $B = 4.4$ T, there was a clear 20–30% increase in the alpha particle loss to the detector 90° below the outer midplane whenever 5 MW of ICRH was applied. This relative increase was similar to that seen previously for DD fusion products in ICRF heated DD plasmas [14]. In those DD cases the analysis suggested that the increased fusion product loss was due to the perpendicular energy gained by fusion products during their passage through the ICRH resonant zone, such that passing orbits which had been confined were converted into trapped orbits which were lost. Further experiments in DT are intended to maximize this interaction in an attempt to control alpha particle transport.

3. Particle transport, recycling and retention

Particle transport and recycling from the limiter have been studied on TFTR previous to the introduction of tritium. Deuterium recycling from the limiter in NB heated discharges [15,16] was studied using an absolutely calibrated D_α array [17] to measure the flux of deuterium from the limiter at one toroidal angle and several poloidal positions. The measurements were combined with the DEGAS neutrals code and the TRANSP transport code to self consistently determine edge and core plasma parameters. Electron transport [18] and He transport [19] have been studied using gas puffs as the source of particles. The use of tritium provides a new method to study particle transport and recycling using all of the above methods.

On time scales longer than the particle confinement time, recycling can have a significant effect on core plasma dynamics. The TFTR limiter contains very little T compared to D and H so that T recycling from the limiter is low. This results in a depletion of core T density even with 50:50 deuterium-to-tritium NB injection. Normal operation on TFTR requires a low recycling limiter (Section 4). The conditioning process lowers the hydrogenic content of the limiter by discharge cleaning and Li pellet injection. On average, the total amount of T injected by the neutral beams is only a few percent of the total amount of D injected so that the T/(H + D + T) ratio on the limiter is always small. The DT discharge sequences are short (< 10 discharges) so that the H/D/T content of the limiter is never in equilibrium with the plasma. The recycling of T is observed to increase during a DT shot sequence but remains small (< 0.05) as determine by spectroscopic measurements of the H/D/T influx from the limiter [20].

For the high power DT discharges, time-dependent modeling best matches the measured plasma parameters if the influx of hydrogen isotopes from the limiter is 20% H, 75-80% D, and < 5% T, and the total hydrogenic recycling coefficient is low (< 75%). These estimates were also confirmed by spectroscopic measurements of the H/D/T influx at the limiter [20]. The low tritium recycling from the limiter has a significant effect on plasma operation, in that, in order to obtain central 50:50 deuterium-to-tritium densities, optimum for fusion power production, neutral beam injection must be tritium rich, 60:40 to 70:30 tritium-to-deuterium, in order to balance the rapid pumping of T by the limiter.

The transport of tritium introduced into a NB heated plasma by gas puff has been measured using the TFTR neutron collimator [21,22]. This is the first time that the transport of a hydrogen isotope has been directly measured. In the experiments, either trace tritium (2% T, 98% D) or pure tritium was injected and the transport observed by detecting the DT neutron emission profile as a function of time. A typical neutron emission profile is shown in Fig. 7. Similar experiments to measure electron transport using gas puffs have been perturbing in that the puff was large enough to effect plasma temperature. In the T puff experiments, the puff is extremely small, the density increment due to the T puff is less than 10^{-3} of the target plasma density. The effect of the T puff is only observable on the neutron emission.

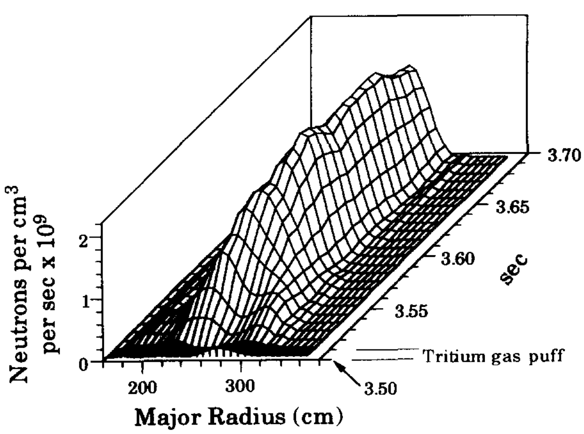

Fig. 7. Abel inverted neutron emission profile from a tritium puff into a deuterium neutral beam heated plasma.

The previous studies of He and electron transport showed a difference between L-mode and super-shot plasmas. In L-mode, helium diffusivity is about 5 times larger than electron diffusivity in the core and nearly equal at the plasma edge. The He diffusivity is typically 1-2 m^2/s. In supershot plasmas, the He and electron diffusivities are similar throughout the plasma, low in the core and rising to about 1 m^2/s at the edge. The tritium transport results in supershot plasmas indicate that the T transport is low in the core and rises to values comparable to the He diffusivity at the plasma edge.

The low tritium recycling implies that the tritium retention in the TFTR vacuum vessel could be on the order of unity. Estimates of T retention in TFTR based on deuterium retention [23,24] indicate that T retention could be about 0.5 if the T retention is the same as the D retention. However, this same study showed that retention increased with average NB power, so that the actual T retention may be higher for the DT campaign since TFTR presently operates at larger average power. In the D retention studies, the D content of the limiter was maintained at value consistent with low recycling and was essentially in equilibrium. For the DT campaign, the T levels do not achieve equilibrium and the D retention value should be regarded as a lower limit to the T retention value. A study which examined the retention of the tritium produced during DD operation prior to the introduction of T as a fuel [25] estimated a T retention factor of 0.5. This value is consistent with the average D retention factor. The main mechanism for retention was found to be codeposition of the hydrogen isotopes with eroded graphite from the limiter rather than implantation.

3.1. Tritium accounting

TFTR has been operating with mixtures of deuterium and tritium since November 1993. This section gives general information on tritium accounting as of March 11, 1994. The number of tritium neutral-beam (TNB) fueled shots is 65, with a total consumption of 53350 Ci [26]. This value was obtained from pressure-volume-temperature (PVT) measurements in the NB tritium delivery manifolds and assumes a tritium purity of 100%. An estimate of the NB torus fueling, i.e. the amount of tritium injected by beams into the torus, is 1,900 Ci [27], or 3.6% of the TNB consumption. This value was calculated from the neutral beam gas efficiency, neutralization, transport efficiency and pulse length [27]. The torus fueling does not include a small cold gas contribution from the beam, which is anticipated to be a 10% correction. Also negligible, as it amounts to less than 100 Ci, is the amount of tritium injected into the torus by gas puffs in about 10 shots. The amount of tritium implanted into NB ion dumps and scrapers is estimated to be not more than 1400 Ci or 2.6% of the TNB consumption [26]. The JET experience with the Preliminary Tritium Experiment [28] and recent TFTR tests suggest that a large fraction of this implanted amount is released by DNB conditioning shots fired onto the copper dumps.

An upper bound on the machine tritium inventory of 3300 Ci, or 6.2% of the TNB consumption, is obtained by adding the torus fueling to the NB dumps and scrapers implantation. Similarly, an upper bound on the torus tritium inventory is 1900 Ci.

Approximately 94% of the tritium NB consumption is pumped directly by the cryopanels of the four NB lines. To remove the condensed gas, each cryopanel periodically undergoes a regeneration, whereby the liquid helium cryopanel temperature is raised to about 21 K, by puffing 600 Torr l of He into the isolated beam box. This process releases most of the molecular deuterium and tritium trapped on the cryopanel and is accompanied by a pressure increase in the box to above 1 Torr. The released gas is pumped by tritium-compatible, dry fore-pumps into one of two Gas Holding Tanks (GHT), each having a volume of 7.6 m^3. The GHT gas is analyzed by PVT measurements and for tritium content by means of an in situ calibrated ionization chamber [29].

The amount of tritium recovered into the GHT's is 50568 Ci or 94.8% of the tritium NB consumption. A machine inventory (in-vessel plus NB) of 2782 Ci is obtained by subtracting this amount from the consumption. This inventory is equal to 5.2% of the tritium NB consumption and to an 84% machine tritium retention rate. This would suggest that torus or NB box retention are higher than predicted on the basis of deuterium content analysis of TFTR tiles previously carried out at SNL [23–25]. For comparison, the short term tritium retention after the JET PTE was about 75% of the TNB torus fueling [30]. Previous JET data on D short-term retention in the all-carbon-machine phase indicted a retention equal to 88% of the D torus fueling [31].

4. Limiter conditioning

The importance of limiter conditioning to achieve low limiter recycling and enhanced confinement in TFTR was demonstrated with the discovery of the "supershot" confinement regime in 1987 [32]. The observation of confinement improvements with Li pellet injection [33] has resulted in limiter conditioning being an important part of the TFTR operational plan. Lithium conditioning of the TFTR limiter has allowed the extension of the supershot confinement regime to the highest plasma currents readily attainable in TFTR (2.5 MA).

The standard conditioning sequence begins with low power ohmic plasmas fueled by a He or D prefill gas puff for breakdown and particles recycled from the limiter. No additional gas puffing or neutral beam heating is used. As the discharge sequence continues, the plasma density and the influx of deuterium and carbon from the limiter decrease. Eventually, the influx of carbon from the limiter becomes constant while the deuterium influx continues to decrease. The plasma density may continue falling but at a very slow rate. The number of discharges required to reach low recycling conditions depends on the initial state of the limiter. For example, after a major disruption, 30 or more conditioning plasmas are usually needed to achieve low recycling conditions while only 5–10 may be needed after a series of supershot discharges produced with an initially well conditioned limiter. Using deuterium as the breakdown gas results in a 30% slower cleanup rate than with He; however, D is preferred when conditioning for DT experiments so that alpha particle measurements are not contaminated by residual helium from the limiter. The Z_{eff} in these ohmic discharges is approximately five. After conditioning the limiter with He ohmic discharges, the residual He in the limiter is removed during the initial discharges in a supershot sequence and requires approximately 5 discharges.

Further decreasing the carbon and deuterium influx from the limiter is accomplished by injecting lithium pellets [34–36] which also results in a decrease in the plasma edge density. Ten pellet injected discharges will reduce the carbon influx by 30%. These pellets are cylindrical in shape and are up to 2 mm in diameter by 2 mm long and contain approximately 2×10^{20} Li atoms. This amount of material is sufficient to cover the TFTR vacuum vessel with one mono-layer of material, on average. The beneficial effects of Li pellet injection, as evidenced by reduced D and C influx and enhanced confinement time, gradually decrease over 10 discharges after the cessation of injection.

Neutral beam heating into target plasmas prepared using these techniques develop into supershots in TFTR. To maintain the beneficial effect of the lithium coating on the limiter, additional pellets are injected both before and after NBI. The pellets injected after NBI serve to prepare the limiter for the next discharge. Plasma confinement will continue to improve for ten to fifteen such discharges and approach a reproducible state. The beneficial effects of Li pellet injection are immediate; it is not necessary to obtain a minimum amount of Li on the limiter for improvements in plasma performance, such as energy confinement time, to be observed. However, increasing the amount of Li on the limiter continues to improve plasma performance.

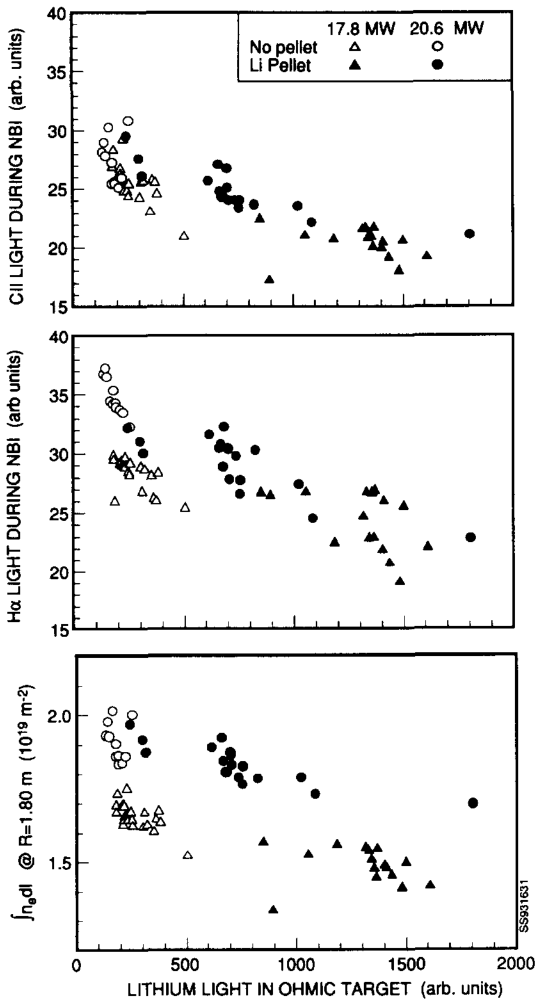

Fig. 8. The CII light, H_α emission and edge line integral density during NB injection plotted versus Li light in the ohmic target.

The influx of carbon and deuterium, and the edge density all increase with increasing NB power. As previously observed on JET and in TFTR L-mode discharges [37], the edge density increases as $P_{\mathrm{LIM}}^{0.5}$. The reduction in carbon influx, deuterium influx, and plasma edge density during neutral beam injection with increasing amounts on Li in the target plasma is shown in Fig. 8. Concurrently, the energy confinement time improves as each of these measures of neutral influx to the plasma edge decreases. The improvement has not been observed to saturate with the amounts of Li injected into TFTR.

5. Summary

The DT campaign on TFTR is underway. The 5 MW fusion power milestone was achieved on the second day of DT operation with a 6.2 MW discharge. DT plasmas with 5 MW fusion are made routinely. The DT discharges have higher T_e, T_i, and τ_E than DD comparison shots. An isotope effect is observed on χ_i and possible alpha heating of the electrons.

ICRF heating of DT plasmas has been successfully demonstrated.

High β_p plasmas in DT have been made and achieved 4.3 MW of fusion power at $I_p = 12$ MA. During the H-mode phase of these discharges, energy confinement times 4–5 greater than L-mode are observed.

Alpha particle loss is not observed to increase with fusion power and no alpha driven instabilities excited purely by alpha particles have been detected. TAE instabilities excited by ICRF have been made, demonstrating that these modes can exist in high temperature DT plasmas.

Limiter conditioning by depleting the limiter surface of hydrogen isotopes and Li pellet injection to reduce the carbon influx are critical to the attainment of supershots and high fusion power. The low tritium recycling from the limiter reduces core plasma reactivity; tritium rich NB injection is required to obtain the optimum 50:50 DT density ratio.

The transport of tritium has been measured using extremely small, non-perturbing T gas puffs. The tritium density evolution in a deuterium neutral beam heated plasma is measured using the DT neutron emission profiles.

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References

[1] R.J. Hawryluk et al., Phys. Rev. Lett. 72 (1994) 3526.
[2] J.D. Strachan et al., Phys. Rev. Lett. 72 (1994) 3530.
[3] J.D. Strachan et al., Phys. Rev. Lett. 58 (1987) 1004.
[4] S.A. Sabbagh et al., Phys. Fluids B 3 (1991) 2277.
[5] C. Bush et al., Phys. Rev. Lett. 65 (1990) 424.
[6] H.F. Dylla et al., J. Nucl. Mater. 162–164 (1989) 128.
[7] J.L. Terry et al., Plasma Physics and Controlled Fusion Research 1990 (IAEA, Vienna 1991) vol. 1, p. 393.
[8] P.H. Rebut et al., Nucl. Fusion 32 (1992) 187.
[9] R.V. Budny et al., Nucl. Fusion 32 (1992) 429.
[10] K.L. Wong et al., Phys. Rev. Lett. 66 (1991) 1874.
[11] C.Z. Cheng et al., IAEA Wurzburg, paper, IAEA-CN-56/D-2-1 (1992).
[12] S.J. Zweben et al., Nucl. Fusion 33 (1993) 705.
[13] D.S. Darrow et al., Rev. Sci. Instr. (1994) (invited paper).
[14] D.S. Darrow et al., PPLL-2975 (1994) submitted to Nucl Fusion.
[15] R.V. Budny et al., J. Nucl. Mater. 176/177 (1990) 427.

[16] R.V. Budny et al., J. Nucl. Mater. 196–198 (1992) 462.
[17] A. Ramsey and S.L. Turner, Rev. Sci. Instr. 58 (1987) 1211.
[18] P.C. Efthimion et al., Phys. Rev. Lett. 66 (1991) 421.
[19] E.J. Synakowski et al., Plasma Physics and Controlled Nuclear Fusion Research, Proc. 14th Int. Conf., Wurzburg, Germany, 1992 (IAEA, Vienna, 1992) vol. 3, p. 463.
[20] C.H. Skinner et al., PPPL Report 2993.
[21] L.C. Johnson, Rev. Sci. Instr. 63 (1992) 4517.
[22] A.L. Roquemore, L.C. Johnson and S. von Goeler, 10th Topical Conf. on High Temperature Plasma Diagnostics, Rochester, USA, 1994.
[23] M. Caorlin, M. Ulrickson, NIFS-PROC-12, Japan–US Workshop P-196 on High Heat Flux Components and Plasma Surface Interactions for Next Devices, Kyushu University, Japan, 1992.
[24] M. Ulrickson et al., 12th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research (IAEA, Vienna, 1989) vol. III, p. 419.
[25] W.R. Wampler et al., J. Vac. Sci. Tech. A 6 (1988) 2111.
[26] J. Kamperschroer, PPPL internal memo to P. LaMarche, April 1, 1994, HSD-3674, TNEA.
[27] J. Kamperschroer, PPPL internal memo to C. Skinner, April 8, 1994.
[28] H.D. Falter et al., Fusion Techn. 20 (1992) 481.
[29] J.L. Anderson et al., 15th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research, Madrid, Spain, 1994.
[30] G. Saibene et al., Fusion Eng. Des. 19 (1992) 133.
[31] R. Sartori et al., J. Nucl. Mater. 176/177 (1990) 624.
[32] J.D. Strachan et al., Phys. Rev. Lett. 58 (1987) 1004.
[33] J.L. Terry et al., Plasma Physics and Controlled Nuclear Fusion Research 1988 (IAEA, Vienna, 1989) vol. 1, p. 27.
[34] J.A. Snipes et al., 18th Europ. Conf. on Controlled Fusion and Plasma Physics, Berlin, Germany, 1991, vol. 3 (EPS, Berlin, 1991) p. 141.
[35] J.A. Snipes et al., J. Nucl. Mater. 196–198 (1992) 686.
[36] J.D. Strachan et al., PPPL Report 2945.
[37] C.S. Pitcher et al., J. Nucl. Mater. 176/177 (1990) 285.