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# Preparations for deuterium–tritium experiments on the Tokamak Fusion Test Reactor\*

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The final hardware modifications for tritium operation have been completed for the Tokamak Fusion Test Reactor (TFTR) [Fusion Technol. **21**, 1324 (1992)]. These activities include preparation of the tritium gas handling system, installation of additional neutron shielding, conversion of the toroidal field coil cooling system from water to a Fluorinert™ system, modification of the vacuum system to handle tritium, preparation, and testing of the neutral beam system for tritium operation and a final deuterium–deuterium (D–D) run to simulate expected deuterium–tritium (D–T) operation. Testing of the tritium system with low concentration tritium has successfully begun. Simulation of trace and high power D–T experiments using D–D have been performed. The physics objectives of D–T operation are production of  $\approx 10$  MW of fusion power, evaluation of confinement, and heating in deuterium–tritium plasmas, evaluation of  $\alpha$ -particle heating of electrons, and collective effects driven by alpha particles and testing of diagnostics for confined  $\alpha$  particles. Experimental results and theoretical modeling in support of the D–T experiments are reviewed.

## I. INTRODUCTION

The design of the International Thermonuclear Experimental Reactor (ITER)<sup>1</sup> is based upon the experimental results from a large number of tokamak experiments con-

ducted principally with hydrogen and deuterium as the fuel. In the world tokamak fusion program, two major facilities, the Tokamak Fusion Test Reactor (TFTR)<sup>2</sup> and the Joint European Torus (JET),<sup>3</sup> plan to study the burning plasma physics associated with the use of deuterium–tritium fuel in support of ITER. A limited scope, “Preliminary Tritium Experiment (PTE)” has been performed on a Joint European Torus (JET) in 1991,<sup>3</sup> with a more ex-

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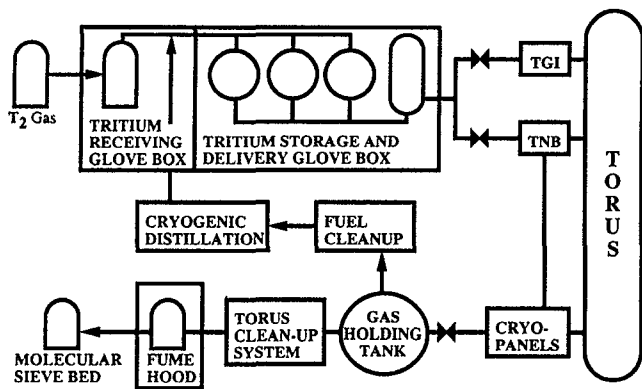


FIG. 1. Schematic of the tritium processing system.

tensive program planned to begin in 1996. TFTR has been modified for tritium operation and will begin an extensive series of experiments in December 1993.

The principal goals of the TFTR deuterium-tritium experiments are the following:

- (1) Safe operation of the tritium handling and processing systems, and successful machine and diagnostic operation in a high radiation environment with 14 MeV neutrons.
- (2) Documenting changes in confinement and heating going from deuterium to tritium plasmas.
- (3) Evaluating the confinement of  $\alpha$  particles, including the effect of  $\alpha$ -induced instabilities, and obtaining initial indications of  $\alpha$  heating, and helium ash accumulation.
- (4) Demonstrating the production of  $\approx 10$  MW of fusion power.

These goals not only individually support the technical and physics research and development objectives of ITER, but their integration does as well. The achievement of  $\approx 10$  MW of fusion power entails, not only successful technical performance, but, more importantly, the development and extension of reliable operating regimes, which address plasma stability issues associated with the increase in the stored plasma energy going from deuterium to tritium. In present plasma experiments, these effects are predicted to be relatively modest. However, in future ignited or near-ignited experiments, these effects are large and impact fundamental design considerations. Thus, confirmation of our understanding in present tokamaks by performing experiments that integrate all of the goals is an important objective.

In this paper, a brief description will be given of the principal modifications to the TFTR device for D-T operations and the status of the facility. A more extensive discussion will be given of the planned experiments on TFTR, and how they will address key design considerations of a tokamak reactor utilizing deuterium-tritium fuel.

## II. MACHINE CONFIGURATION

The hardware modifications for tritium operation on TFTR have been completed. These activities include prep-

aration of the tritium gas handling system (shown in Fig. 1), which has been successfully tested with low concentrations of tritium ( $\approx 0.5\%$ ) and is capable of handling up to 5 g of tritium (50 kCi).<sup>4</sup> The tritium gas is brought on site in an approved shipping canister and transferred to a uranium bed, where it is stored. The uranium bed is heated to transfer the gas to the neutral or torus injection systems. The gas is then injected into the torus or neutral beams, and pumped by the cryopanel in the beam boxes. During plasma operation, some of the gas will be retained in the graphite limiter tiles in the vacuum vessel. The quantity of tritium in the vacuum vessel is restricted by regulatory requirements to 20 kCi. The gas on the cryopanel is transferred to the Gas Holding Tank.

During initial operation, the gas in the Gas Holding Tank will be oxidized by the Torus Cleanup System and absorbed onto molecular sieve beds. These beds will be shipped off site for reprocessing or burial prior to the receipt of a replacement canister of tritium gas. Next year, a cryodistillation system will be commissioned to repurify the tritium on site and decrease the number of off-site shipments required.

Modifications to the tokamak include supplemental, diagnostic and personnel shielding, tritium stack and area monitoring, remote control for components in high radiation fields, and upgrades to the vacuum and neutral beam pumping system for tritium operation.<sup>5</sup> For components and systems in high radiation fields, modifications have been made to increase reliability. The toroidal field coil cooling system has been converted from water to Fluorinert™ (a fully fluorinated compound), bolts on the TF coil casings have been tightened using remotely operated tools for bolts in areas difficult to access, and graphite tiles in high heat flux regions on the bumper limiter have been replaced by carbon/carbon-fiber composite tiles.

Changing the fuel from deuterium to tritium engenders a change in the rigor and formality of operation as the TFTR device becomes, according to U.S. Department of Energy (DOE) regulations, a category 3 (low hazard) nuclear facility.<sup>6</sup> The site boundary dose due to normal operation will be limited to  $< 10$  mrem/yr. A safety analysis has been performed that demonstrates that the maximum design basis accident would result in a 140 mrem dose at the site boundary. An Environmental Assessment has been conducted that resulted in a "Finding of No Significant Impact" to the environment due to the low inventory of tritium on site. The Laboratory and the U.S. Department of Energy recognize the importance of demonstrating safe operation with tritium and have implemented detailed procedures used to process tritium, rigorous qualification, and training programs for operators and engineers, extensive safety reviews by internal and outside experts, and a formal program for the control of operations. At the present time, a final review of the operations and test procedures used for tritium processing is occurring prior to continued testing with low concentrations of tritium.

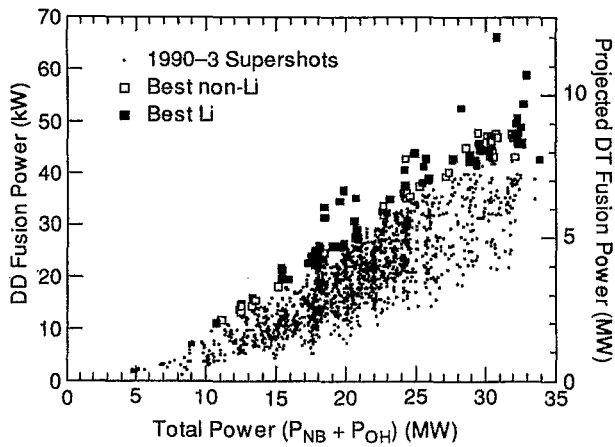


FIG. 2. Fusion power in deuterium supershot neutral beam heated discharges as a function of total heating power. The projected fusion power for a 50:50 mixture of deuterium and tritium is shown using a value of  $Q_{DT}/Q_{DD}=180$ .

### III. MACHINE STATUS

Upon the completion of the outage to modify the machine for deuterium-tritium experiments, operations resumed in deuterium. Besides qualifying the machine for tritium experiments, the focus of these deuterium experiments was to simulate the initial planned deuterium-tritium experiments utilizing deuterium. The objective of the initial high-power deuterium-tritium experiments will be to achieve 5 MW of fusion power. With further operational experience and the resolution of the physics issues identified above, it is projected that  $\approx 10$  MW will be produced.

Previous experiments in the supershot regime with deuterium neutral-beam injection (NBI) have identified the conditions under which  $\geq 5$  MW of fusion power can be produced in D-T plasmas in TFTR. Wall coating, using lithium pellet injection, has significantly improved supershot performance and reliability.<sup>7,8</sup> With this technique,  $n_e \tau_E T_i = 5.8 \times 10^{20} \text{ m}^{-3} \text{ keV}$ ,  $n_e(0) = 1.0 \times 10^{20} \text{ m}^{-3}$ ,  $T_i = 29 \text{ keV}$ ,  $\tau_E = 205 \text{ ms}$  and a D-D fusion neutron rate  $S_{DD} = 5.6 \times 10^{16} \text{ n/s}$  have been achieved. This corresponds to a fusion power,  $P_{DD}$ , of 65 kW and  $Q_{DD} \equiv P_{DD}/(P_{NBI} + P_{OH})$  of  $2.1 \times 10^{-3}$ . One-dimensional transport code simulations of a 50/50 D-T plasma using the same electron and ion temperature and density profiles predict that  $P_{DT} \approx 11 \text{ MW}$  and a  $Q_{DT}$  of 0.38 would have been achieved. At modest powers,  $P_{NBI} = 15 \text{ MW}$ , the energy confinement time has increased transiently to 240 ms (corresponding to  $\tau_E/\tau_E^{(L \text{ mode})} \approx 3$ , which is typical of the best supershots), with the injection of two lithium pellets prior to NBI and one lithium pellet in the post-NBI phase of the previous shot. The D-D fusion performance of NBI supershots is shown in Fig. 2 as a function of the total heating power.

At low-power TFTR supershots smoothly approach an asymptotic stored energy during the pulse, but at high power they often reach a peak in stored energy after 300–500 ms into beam injection, then suffer a deterioration of

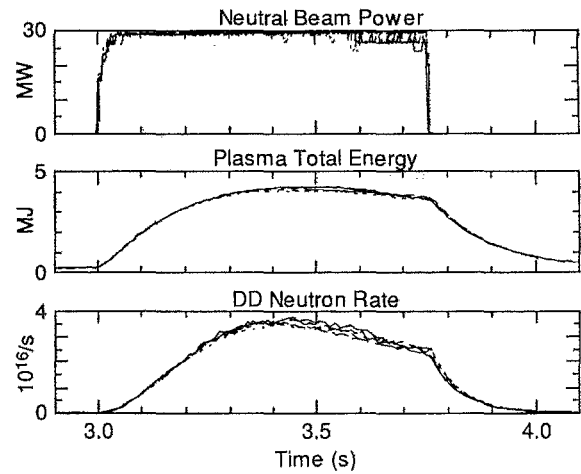


FIG. 3. Comparison of five high power neutral beam heated discharges that employed Li pellets to condition the walls. Good reproducibility is achieved.

as much as a factor of 2 (the so-called “beta collapse”) over the subsequent several hundred milliseconds. At a major radius,  $R=2.45 \text{ m}$ , the plasma beta collapses are coincident with or follow the growth of internal MHD ( $m/n=3/2, 4/3, \dots$ ) modes, as recorded by Mirnov loops. In some cases the MHD can cause a flattening of the  $T_e(R)$  profile that is observed in the vicinity of the rational  $q$  surfaces at which the MHD modes are observed. Analysis indicates that the decrease in plasma energy content is consistent with a model for which transport is increased only within the islands.<sup>9</sup>

Reliable high-performance operation entails reproducible low-recycling wall conditions, along with disruption avoidance and minimization of adverse MHD effects. To achieve  $P_{DT} > 5 \text{ MW}$  in TFTR supershots requires  $I_p = 2.0 \text{ MA}$ ,  $P_{NBI} = 30 \text{ MW}$ , composed of a 50:50 D:T mix, and  $\tau_E = 140 \text{ ms}$ , which produces a Troyon-normalized beta,  $\beta_N \equiv \beta_T(\%) \cdot a(m) \cdot B_T(T) / I_p(\text{MA})$  of 1.5. The modest value of  $\beta_N$  allows for increased stored energy due to the injection of energetic tritons (which have a longer slowing-down time than deuterons), the presence of fusion  $\alpha$  particles and the heating of the electrons by the  $\alpha$  particles. The parameter of relevance for fusion yields,  $\beta_N^* \equiv 2\mu_0 \cdot \langle \langle p \rangle \rangle \cdot a(m) / B_T(T) \cdot I_p(\text{MA})$ , where  $\langle \langle p \rangle \rangle$  is the root-mean-square plasma pressure, reaches 2.4 in prototypical high-performance plasmas, as a result of the highly peaked pressure profile. To achieve the required confinement reliably, Li pellet injection is used to condition the walls and improve reproducibility. The reproducibility of confinement achievable with Li-pellet conditioning is illustrated in Fig. 3. During a dry run of the planned 5 MW D-T experiment, 13 of 16 simulated D-T discharges with Li pellets would have achieved  $> 5 \text{ MW}$  of fusion power.

An operating strategy for achieving higher fusion power has been developed using multiple Li pellets to condition the walls, high-power neutral beam injection of

about 35 MW, and high plasma current ( $\sim 2.5$  MA). In deuterium operation, 34 MW of neutral injection has been achieved, and the power is projected to increase in tritium due to improved equilibration in the neutralizer cell. During recent experiments, as a result of aggressive wall conditioning with Li pellets, supershots have been produced at  $I_p = 2.5$  MA corresponding to  $q_\psi = 3.8$ , which represents a significant extension of the operating regime.

#### IV. $\alpha$ CONFINEMENT

The confinement of  $\alpha$  particles is very important in a reactor for two reasons. First, if a small unanticipated fraction (a few percent) of the  $\alpha$  particles is lost, and the resulting heat flux is localized, damage to first wall components could result. Second, if a significant fraction of the  $\alpha$  particles are not confined then the  $nT\tau$  requirements for ignition would increase.

The loss of fusion products during deuterium experiments has been extensively studied and is reported on in a companion paper by Zweben.<sup>10</sup> An array of detectors in the TFTR vacuum vessel has been used to measure the energy and pitch angle of the escaping  $\alpha$ 's. The conclusions of these studies are as follows.

(1) Good agreement has been obtained as a function of plasma current with the prediction of the losses due to unconfined "first orbits." For the fusion products, an upper bound of  $D < 0.1$  m<sup>2</sup>/s has been deduced for their radial transport coefficient in MHD-quiescent plasma.

(2) The predictions of stochastic ripple diffusion are in good qualitative agreement with experimental observations near the outer midplane.

(3) MHD instabilities, in particular, low  $m/n$  modes, are observed to be responsible for an increased loss of fusion products. A theoretical model based on magnetic perturbations affecting  $\alpha$  drift orbits is in qualitative agreement with the observations.

(4) ICRF waves can interact with fusion products and result in increased particle loss. The mechanism is not fully understood, but may be related to velocity space diffusion across the passing-trapped boundary for particles due to resonances with fusion product ions.

(5) Loss of fusion particles that is delayed in time and is not associated with first-orbit loss has been observed. Qualitatively, it appears as a collisional and/or TF-ripple-induced loss process; however, it is larger than predicted and not fully understood.

The results from fusion product experiments in present deuterium experiments show that the particles can interact with magnetohydrodynamics (MHD) instabilities and plasma waves. The theoretical calculations indicate that when the values of  $\beta_\alpha$  and  $n_\alpha$  are sufficiently high,  $\alpha$  particles themselves can induce instabilities, which can, in turn, eject the  $\alpha$  particles from the plasma. The present theoretical understanding of the role of  $\alpha$ -collective effects in TFTR is that  $\alpha$  particles will not stabilize sawteeth and are not expected to cause  $\alpha$ -driven fishbones, but will drive toroidal Alfvén eigenmodes (TAE) unstable in some conditions, are not expected to reduce the  $\beta$  limits due to kinetic ballooning modes (KBM) and may drive the beta

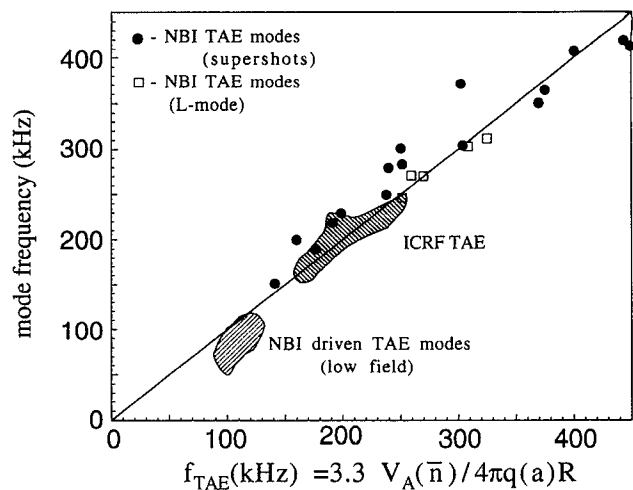


FIG. 4. The predicted frequency of the toroidal Alfvén eigenmodes is compared with the measured frequency for a wide variety of plasma conditions. The low-field neutral beam experiments were reported by Wong *et al.*<sup>10</sup> and the ICRF experiments by Wilson *et al.*<sup>12</sup> The recent high-field experiments are shown for comparison.

Alfvén eigenmodes (BAE) unstable at high  $\beta$ . One of the instabilities with the lowest predicted threshold is the toroidal Alfvén eigenmode. Experiments in deuterium plasmas have been conducted on TFTR<sup>11</sup> and DIII-D<sup>12</sup> using neutral beams, and on TFTR with ICRF<sup>13</sup> to simulate these instabilities. The threshold for instability is in good agreement (within a factor of 2) with the predicted threshold.<sup>14</sup> The frequency of the mode scales with the Alfvén frequency and the mode can be destabilized either by passing particles from neutral beam injection or by trapped particles due to ICRF heating, as shown in Fig. 4. Very recent experiments have shown that instabilities in the Alfvén range of frequencies were observed in some high-field (5 T) neutral beam heated discharges, in which  $V_f/V_A > 0.2$ . Here  $V_f$  is the fast ion velocity and  $V_A$  is the Alfvén velocity. Detailed analysis for these new experimental results are in progress. Measurements of the mode structure using Beam Emission Spectroscopy in low-field (1–1.5 T) neutral beam simulations indicate the presence of a large global mode. Very sensitive reflectometry measurements have been used to determine the mode number in ICRF simulations. Up to about 50% of the fast beam ions in low toroidal field neutral beam heated experiments are observed to be lost due to these instabilities in TFTR. In ICRF experiments, significant loss ( $\sim 10\%$ ) of the energetic minority ion tail has been measured in high-power L-mode experiments. The fractional loss increases with the mode amplitude of the instability (as shown in Fig. 5).

Projected  $\alpha$  parameters [ $\beta_\alpha(0)$ ,  $R \nabla \beta_\alpha$ ,  $V_\alpha/V_A$ ] on TFTR are close to those on ITER because the electron temperatures is similar and the nonthermal ion distribution increases the plasma reactivity. The theoretical models predict the operating regimes in which the instability should occur.<sup>14,15</sup> Furthermore, they provide predictions regarding how the threshold should vary with ion temperature, shear length, and density profile shape. Experiments are

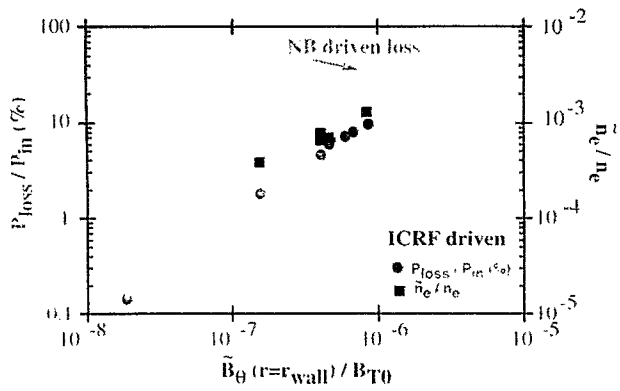


FIG. 5. The power loss by the tail particles heated by ICRF and the neutral beam ions in low-field experiments increases with the magnetic field perturbation at the vacuum vessel wall. In the ICRF experiments, the measurements of  $\tilde{n}_\alpha/n_e$  by reflectometry are also observed to scale with the magnetic field amplitude.

planned during the D-T phase of TFTR to test these predictions. Recent theoretical work has developed the techniques to predict the mode amplitude<sup>16</sup> and the effect of these instabilities on the  $\alpha$ -particle loss rate.<sup>17</sup> Comparison of theory and experiment will improve the estimates of heat flux to first wall components in the ITER due to  $\alpha$  particles.

In support of the  $\alpha$ -particle experiments, a comprehensive set of diagnostics has been implemented. The  $\alpha$ -birth profile will be measured using a ten channel neutron collimator, and the  $\alpha$ -loss rate by detectors mounted on the wall and a movable probe. Four new techniques will be used to determine the  $\alpha$ -distribution function and radial profile. Charge-exchange recombination spectroscopy using a high throughput optical system will be used to measure the energy distribution to  $\sim 1$  MeV at ten radial points.<sup>18</sup> Injected Li pellets will be used to increase the charge-exchange rate and enable charge-exchange mea-

surements from 0.2 keV to 4.0 MeV.<sup>19</sup> Both of these techniques have been tested in deuterium experiments using ICRF to produce a  $^3\text{He}$  tail. In addition, spectroscopic measurements of the Doppler spectrum of  $\text{He}^+$  light emitted from the pellet cloud will be used to measure the distribution function. Another approach entailing collective scattering of a 60 GHz wave will be used to characterize the  $\alpha$  distribution function.<sup>20</sup> Increased scattering associated with the lower hybrid resonance will be employed to reduce the power requirements for the gyrotron. The onset of the instabilities and their mode structure will be measured using a four-channel reflectometer, beam emission spectroscopy, ECE grating polychromator, Mirnov coils, and microwave scattering. The  $\alpha$ -particle diagnostics will provide measurements of the instability itself, its effect on the  $\alpha$  distribution function, and, together with the standard profile diagnostics, its effect on the background plasma.

## V. $\alpha$ HEATING

It is planned to evaluate  $\alpha$  heating of the electrons by operating in a reliable quiescent regime with a minimum of plasma performance degradation with time. This can be accomplished by decreasing the power and the value of  $\beta_N$ . In TFTR,  $\alpha$  heating is a relatively small fraction of the heating power<sup>21</sup> ( $P_\alpha \approx 2$  MW when  $P_{\text{NBI}} \approx 30$  MW); however, the heating is projected to be well localized in the plasma core, as shown in Fig. 6(a). The predicted increase in electron temperature is shown in Fig. 6(b), assuming that the thermal conductivity of the bulk plasma does not change with isotope mix, and does not change with the heat deposition profile. Experiments are planned to vary the isotope mix to evaluate the effects on plasma confinement. Low-power ICRF experiments at 1.6 MW combined with high-power neutral beam heating have simulated  $\alpha$  heating, resulting in a measurable increase in the electron temperature of  $\sim 1$  keV, indicating that electron heating with modest power is measurable.

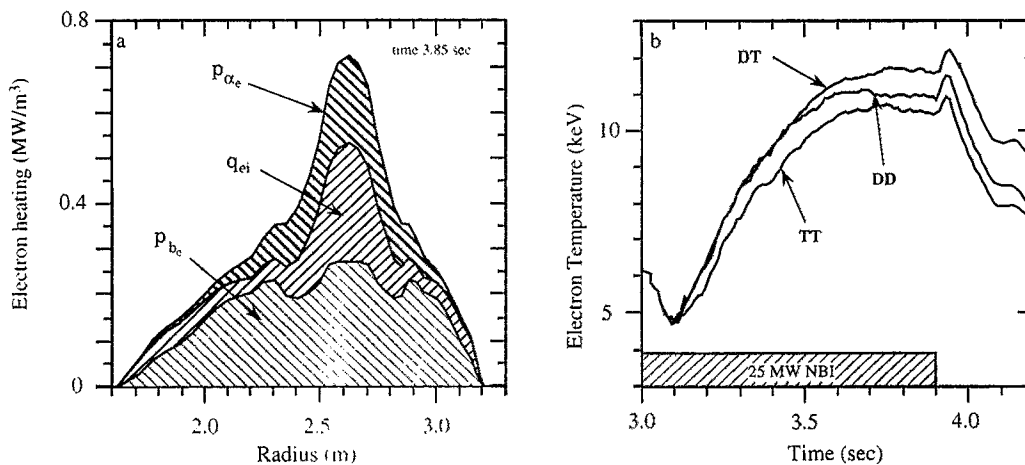


FIG. 6. (a) The electron heating from  $\alpha$ -particles in a 50/50 deuterium-tritium plasma is projected to be significant in the plasma core. (b) The variation in electron temperature for a projected deuterium-tritium, as well as tritium plasma, is shown. In these conditions, the heat conductivity is assumed not to vary with the species mix or heat deposition profile.

## VI. $\alpha$ -ASH ACCUMULATION

The production, transport, and removal of helium ash is an issue that has a large impact in determining the size and cost of ITER. The experiments of TFTR will provide the first opportunity to measure helium ash buildup, assess helium transport coefficients, and examine the effects of edge helium pumping on central ash densities in D–T plasmas. In addition, the importance of the central helium source in determining the helium profile shape and amplitude will be examined. Ash concentrations are expected to reach 1%–2% of the electron density in high-performance supershots. Using the  $\alpha$  CHERS and CHERS diagnostics, central ash concentrations greater than 0.1% should be measurable. Helium ash density profiles, measured as a function of time, will be compared to predictions, based on the helium diffusivity  $D_{\text{He}}$  and convective pinch  $V_{\text{He}}$  inferred from helium gas puffs<sup>22</sup> and central helium source measurements made by  $\alpha$  CHERS and the neutron collimator. The influence of edge pumping will be explored by comparing ash buildup with a low-recycling limiter in a Supershot plasma to that in a high-current, high-recycling L-mode plasma.

## VII. ICRF HEATING

ICRF heating is one of the primary techniques under consideration for ITER. Ion cyclotron resonance heating (ICRF) utilizing either hydrogen or  $^3\text{He}$  minorities has been successfully demonstrated at high power on PLT,<sup>23</sup> JET,<sup>24</sup> and TFTR.<sup>25</sup> In addition, low-power experiments on DIII-D have shown evidence for fast wave current drive.<sup>26</sup> ICRF wave physics in deuterium–tritium plasmas in complicated by the presence of additional resonances and by  $\alpha$  damping, which, due to the large  $k_{\perp} \rho_i$ , can compete with electron absorption in the fast wave current drive regime. A promising scenario for D–T plasmas is heating at the second harmonic of the tritium cyclotron frequency, which is degenerate with the  $^3\text{He}$  fundamental. By selectively heating a majority ion species rather than a minority ion species, potential difficulties with RF driven fast ion excitation of instabilities such as TAE modes should be avoided. Though the core damping is predicted to be acceptable, off-axis absorption at the deuterium fundamental can compete with the second harmonic tritium core damping in tokamaks with a moderate aspect ratio. Modeling indicates that deuterium fundamental heating occurs off the midplane in regions where the shear Alfvén mode conversion layer ( $n_{\parallel}^2 = S$  resonance) approaches the deuterium fundamental layer. In TFTR supershot plasmas with the  $2\Omega_T$  layer, coincident with the Shafranov-shifted axis at 2.82 m (assuming a 0.2 m shift), the  $2\Omega_D$  layer is out of the plasma on the low-field side, but the  $\Omega_D$  layer is in the plasma on the high-field at 5.66 T, or  $R \approx 2.1$  m. For the present range of TFTR magnetic fields and ICRF frequencies, the  $n_{\parallel}^2 = S$  resonance can be positioned in the core region only by adding a relatively high concentration of  $^3\text{He}$  minority to the D–T plasma. Under these conditions, significant mode conversion of the launched fast waves to slow Alfvén waves is predicted to occur, resulting

in significant direct electron heating.<sup>27</sup> Experiments planned on TFTR will be able to document the effectiveness of second harmonic tritium heating and to explore the regimes in which damping associated with the  $n_{\parallel}^2 = S$ -mode conversion layer is expected to be important.

## VIII. FUTURE DIRECTIONS

Several new and novel concepts have been suggested, which, though speculative, could improve the tokamak concept. Several of these concepts require deuterium–tritium fuel to evaluate their feasibility.

Recently, Majeski *et al.*<sup>27</sup> have suggested that the mode-converted slow wave excited at the  $n_{\parallel}^2 = S$ -mode conversion layer in a D–T plasma could be used to drive localized electron currents. This is an attractive alternative to fast-wave damping because the effective single-pass absorption on electrons can approach 80% in TFTR, well in excess of fast-wave single-pass absorption. Furthermore, the driven current is localized to the mode conversion layer, which may be placed on or off axis in order to modify the current profile.

Experiments on TFTR, as well as on DIII-D and JET, have shown that it is possible to achieve high values of  $\beta_p$  ( $\epsilon\beta_p \sim 1.5$ ) with improved energy confinement ( $\tau_E/\tau_E^{(L \text{ mode})} \sim 3$ ) by modifying the plasma current profile. Recent experiments, discussed in a companion paper by Sabbagh *et al.*,<sup>28</sup> have shown that in some TFTR discharges, the core plasma has a direct path to the second stable region. The question then arises whether this altered magnetic configuration affects the threshold for  $\alpha$ -collective effects. Zonca and Chen<sup>29</sup> and Cowley and Furth have suggested that the second stability region will also make the  $\alpha$ -driven modes stable by reducing the fraction of trapped  $\alpha$  particles and inducing drift reversal for sufficiently high  $\beta$ .

Fisch and Rax<sup>30</sup> have proposed coupling the energy in the fast  $\alpha$  particles to RF waves, which, in turn, could either drive currents or heat the ion channel. It is possible that such coupling of the  $\alpha$  particles to thermal fuel ions could increase the fusion power density by up to 50% and reduce the  $nT\tau$  product required for ignition by 10%–30%. If, furthermore, an appropriate nonthermal distribution of fuel ions can be produced by selective coupling, additional gains in fusion power density and reductions in  $nT\tau$  could be achieved. Both the high- $\beta_p$  regime and the interaction of energetic  $\alpha$  particles with plasma waves could be of great physics interest in a D–T plasma.

Experiments to study mode conversion in D–T plasmas, and high  $\beta_p$  effects in D–T plasma have been proposed for TFTR. The collective scattering diagnostic may be able to evaluate the effect of the  $\alpha$  particles on the background lower hybrid waves.

## IX. SUMMARY

The modifications to the TFTR facility have been completed. The final checkout and testing is in progress prior to performing the D–T experiments safely. The deuterium–tritium experiments on TFTR will not only enable con-

firming some of the key design assumptions in ITER, but also explore new and novel physics operating regimes that could improve the tokamak concept.

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