HIGH PERFORMANCE DEUTERIUM-TRITIUM PLASMAS IN TFTR

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High Performance Deuterium-Tritium Plasmas in TFTR*

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ABSTRACT

Plasmas composed of nominally equal concentrations of deuterium and tritium (DT) have been created in TFTR with the goals of producing significant levels of fusion power and of examining the effects of DT fusion alpha particles. Conditioning of the limiter by the injection of lithium pellets[1] has led to an approximate doubling of the energy confinement time, τ_E, in supershot plasmas at high plasma current ($I_p \leq 2.5$ MA) and high heating power ($P_b \leq 33$ MW). Operation with DT typically results in an additional 20% increase in τ_E. In the high poloidal beta, advanced tokamak regime in TFTR, confinement enhancement $H \equiv \tau_E/\tau_{E\ ITER\text{–}89P} > 4$ has been obtained in a limiter H-mode configuration at moderate plasma current $I_p = 0.85 - 1.5$ MA. By peaking the plasma current profile, $\beta_{N\ dia} \equiv 10^8 \langle\beta_{t\perp}\rangle aB_0 / I_p = 3$ has been obtained in these plasmas, exceeding the β_N limit for TFTR plasmas with lower internal inductance, l_i. Confinement of alpha particles appears to be classical and losses due to collective effects have not been observed. While small fluctuations in fusion product loss were observed during ELMs, no large loss was detected in DT plasmas.

1. Introduction

The development of a fusion power plant based on the tokamak concept now seems feasible based on present experimental and theoretical research. While important from a development standpoint, present conservative designs for ignited plasma tokamak reactors do not form a basis for the core of a demonstration power plant because they are not cost-effective. For the experimental device to mature into a practical power source, it must be further developed as an attractive option for utility companies who desire a base plant that is economical, relatively easy to maintain, and which has a high duty factor.

Major improvements related to the cost-effectiveness of the tokamak concept are now under investigation and have formed major components of experimental programs as well as design studies. Development of low activation materials,[2] improved first wall power handling, heat dissipation and particle control,[3] and efficient steady-state current drive systems[4] are all essential components of this research. In addition, improvement of the plasma stability and energy confinement will allow more efficient use of the imposed magnetic field and the initial heating systems. It is also necessary to create these conditions in a plasma free of disruptions to satisfy the requirement of high duty factor.

Tokamak operation with enhanced plasma stability and energy confinement, presently named the "advanced tokamak" regime, is usually characterized by increased values of the Troyon normalized beta, $\beta_N \equiv 10^8 \langle\beta_t\rangle a B_0 / I_p \geq 2.5$ and energy confinement enhancement factor $H \equiv \tau_E / \tau_{E\ ITER\text{–}89P} \geq 2$. The energy confinement time is defined as $\tau_E \equiv W_{tot}/(P_{tot} - dW_{tot}/dt)$ with W_{tot} being the total plasma stored energy and P_{tot} being the total auxiliary heating power. The benefit of this operating regime for future reactors is formidable. A factor of two reduction in the cost of electricity can be realized in an advanced tokamak power plant. Further, a factor of four decrease in the capital cost of a demonstration reactor can be realized with steady-state operation at $\beta_N \leq 6$ and $H \leq 4$.[5]

Operation of the tokamak at high poloidal beta, β_p, is presently being studied as a scenario for improved stability and confinement in existing experimental devices including TFTR[6], DIII-D[7], and JT-60U[8]. This operating mode has the additional benefit of reduced plasma current, I_p, which reduces the requirements for non-inductive current drive, increases the fraction of transport induced bootstrap current,[9] and reduces the impact of disruption. The present paper describes recent work on TFTR which demonstrates the high β_p "advanced tokamak" approach in a plasma composed of deuterium and tritium, the fuel that will most likely be used in first generation power plants.

The high β_p program on TFTR has consisted of creating and understanding tokamak concept improvements based on both pressure and current profile modification. During the deuterium-only phase of operation, modifications of both the central[10] and edge safety factor, q, profile by neutral beam driven current and plasma current ramping[6] have been performed. Modification of the pressure profile (as well as the current profile) has been produced by deuterium and lithium pellet injection.[11] Plasmas composed of nominally equal concentrations of deuterium and tritium (DT) have been created in TFTR with the goal of producing significant levels of fusion power, alpha particle beta, and reactor relevant fusion power density. The creation of these plasmas has been part of the continual evolution of the "supershot"[12] and "high poloidal beta"[6] regimes, which comprise the highest performance operational scenarios in TFTR. The initial DT experiments in the supershot regime have been described by Strachan, et al.,[13] and Hawryluk, et al.[14] This paper will focus on the first high β_p DT experiments, which have achieved enhanced stability and confinement parameters by peaking the plasma current profile, thereby increasing the plasma internal inductance, l_i.

2. Enhanced energy confinement and DT H-mode

Enhancement in energy confinement beyond what had been achieved previously at high plasma current involved three basic components. Two of these were first explored during TFTR supershot operation. The first involved a reduction in particle recycling by conditioning the limiter utilizing the injection of lithium pellets.[15] Fig. 1 summarizes the effect of Li pellet conditioning on τ_E for supershot plasmas with $I_p = 2.5$ MA. An increase in τ_E was created both before and after the introduction of tritium into TFTR for plasmas in which Li pellets were used for conditioning. Fig. 1 also shows the additional increase in τ_E of approximately 20% produced in plasmas which utilized DT plasma operation and Li pellet conditioning. The isotopic effect due to tritium fueling of the plasma by neutral beam injection comprises the second component of the increase in τ_E, and the physics which causes this increase is presently under investigation.[16] An increase in both electron and ion temperatures, T_e and T_i, was observed in DT plasmas as compared to equivalent D plasmas. The increase in T_i was observed shortly after the initiation of NBI. However, the increase in T_e occurred on the alpha particle slowing-down timescale, indicating that alpha particle heating of the electrons was responsible for this increase. Present analysis using the TRANSP[17] code indicates that up to 1 MW of alpha particle heating has been reached.

The third component of increased H factor was a transition to a limiter H-mode configuration[18]. This was most readily produced in high β_p plasmas in which a rapid decrease in I_p was used to peak the plasma current profile. This process is illustrated in Fig. 2, which shows time histories of various plasma parameters describing this operational mode. Neutral beam injection began with co-injected beam sources and was increased to a

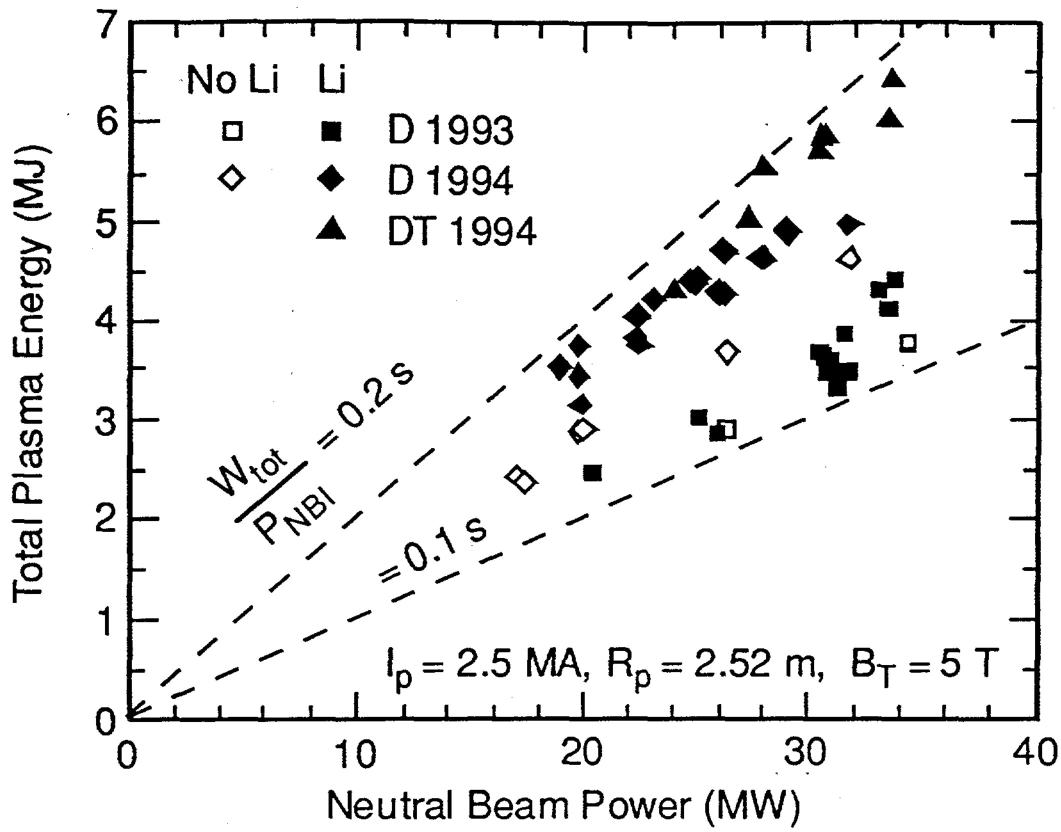

Fig. 1. Effect on τ_E of limiter conditioning with Li pellet injection. Solid and open symbols represent plasmas in with Li pellets were, or were not injected, respectively. Squares represent data taken before T was introduced into TFTR. Diamonds represent D plasmas taken after T was used in TFTR, and at which time limiter conditioning had generally improved (edge recycling reduced). Triangles represent plasmas in which deuterium and tritium were used.

desired power level by adding additional counter-injected beam sources after the completion of the change in I_p. The increase in l_i shown in Fig. 2a) allows for operation at increased $\beta_{N\,dia}$ (Fig. 2b). Here, $\beta_{N\,dia}$ is the Troyon normalized beta using the plasma diamagnetic stored energy. Fig. 2 (c-d) shows the transition to the high β_p H-mode in equivalent D and DT plasmas. While a significant increase in τ_E was observed in the D plasma during the MHD quiescent "ELM-free" phase of the H-mode, a greater increase (of approximately 40%) was observed in the DT plasma[19] and an H factor of 4.2 was attained. This improvement was transient, since the onset of the first ELM caused a decrease in τ_E. However, a recovery of increased energy confinement was observed during the relaxation period of the ELMs. Generally, DT H-mode plasmas exhibit an earlier transition time, a longer ELM-free phase, and a greater drop in D_α light when compared to equivalent D plasmas.

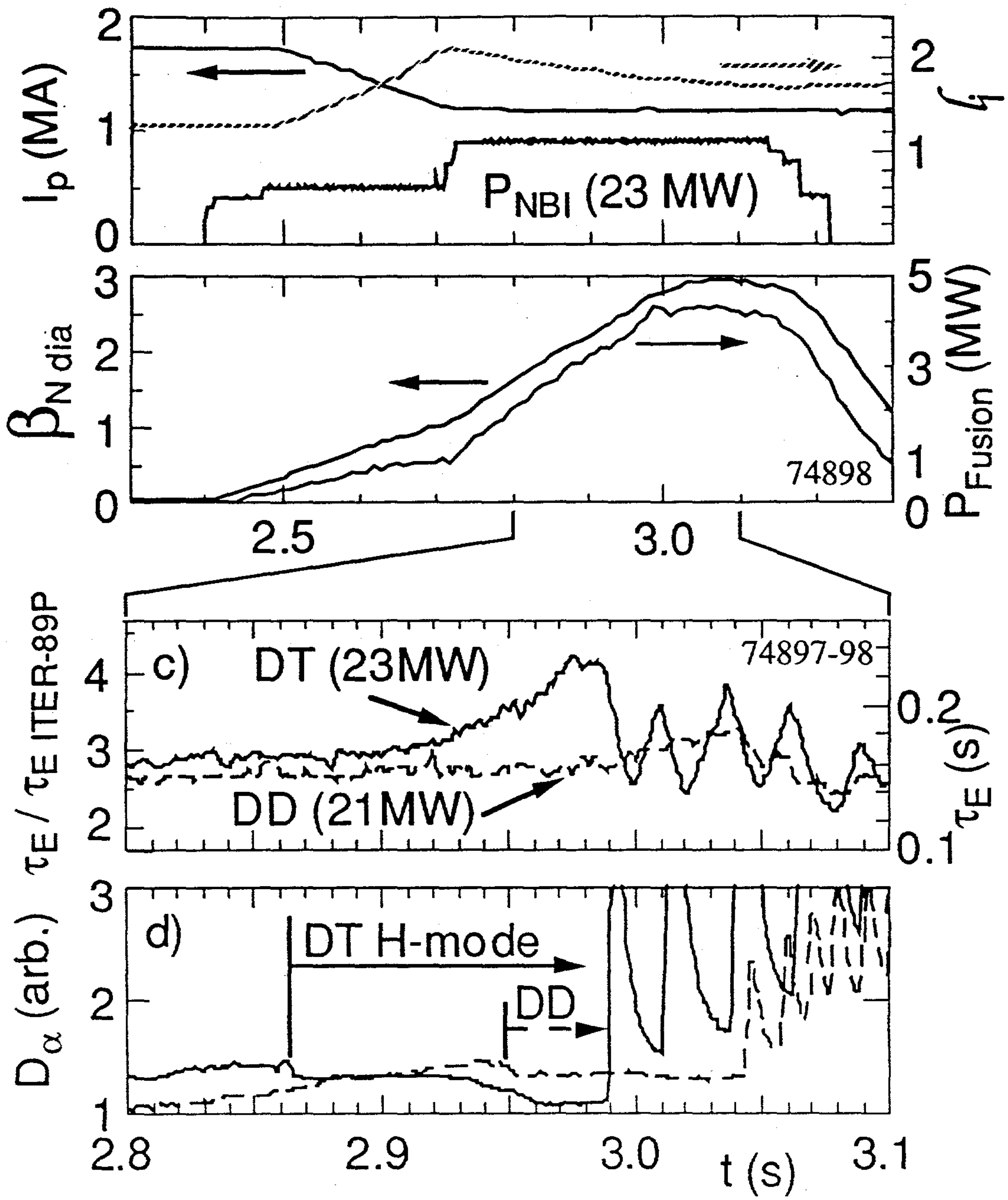

Fig. 2. Time history waveforms for a high β_p DT H-mode plasma and comparison to an equivalent D discharge. Shown in this figure are a) the plasma current, plasma internal inductance, and neutral beam heating, b) diamagnetic normalized beta and DT fusion power c) Energy confinement enhancement factor and confinement time, d) D_α emission.

Since transition to the H-mode occurred during the initial rise of the plasma stored energy during NBI, the plasma density and temperature increased up to the onset time of the first ELM. However, while the edge electron temperature and density steadily increased, the edge T_i displayed a more rapid increase in both DT and D plasmas at the time of the H-mode transition. At this time, the effective ion thermal diffusivity, $\chi_{i\ tot} \equiv -q_i/n_{i\ th}\ \nabla T_i$ as computed by TRANSP decreased in both DT and D comparison plasmas (Fig. 3). Here, q_i

is the ion heat flux, and $n_{i\,th}$ is the thermal ion density. The larger change in T_i and $\chi_{i\,tot}$ occurred in DT plasmas as compared to equivalent D discharges.

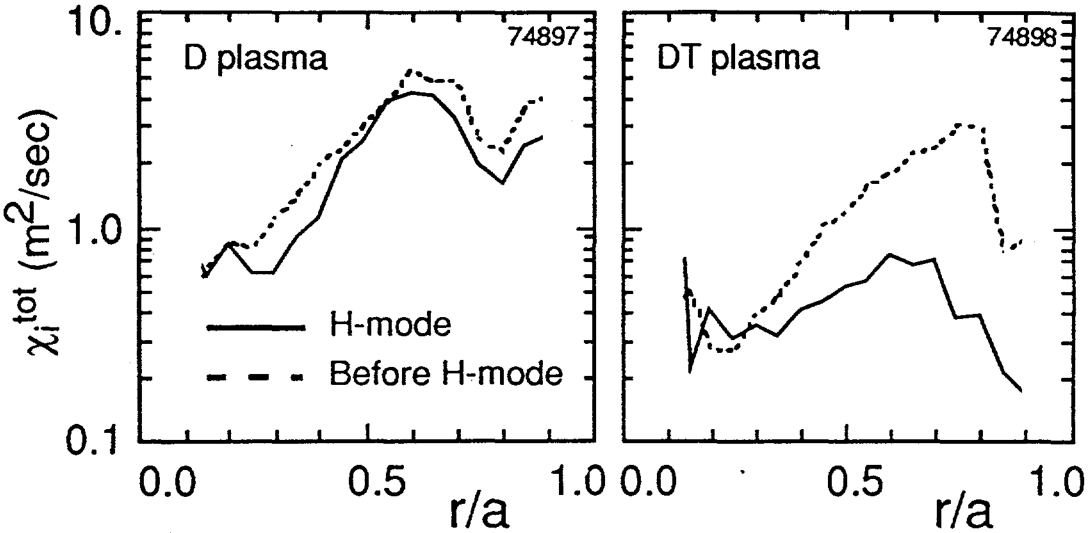

Fig. 3. Reduction in $\chi_{i\,tot}$ for the DT and D plasmas shown in Fig. 2 before the H-mode transition and during the ELM-free H-mode phase.

After the onset of ELMs, the increase in both the edge temperature and density ceased. The edge T_e decreased during the ELM burst but recovered its original value before the subsequent ELM burst. In addition, the edge T_i in DT plasmas decreased to the value reached in the equivalent D plasma. This matching of the T_i profile, which appeared to be caused by the ELMs, generally extended from the plasma edge to a normalized minor radius $r/a = 0.65$.

3. Advanced tokamak operating regime in DT

Plasmas with enhanced values of H and β_N had been produced in TFTR before the introduction of tritium. In particular, $\beta_{p\,dia} = 5.9$ (up to the equilibrium limit[20]), $\beta_{N\,dia} = 4.9$, $H \equiv \tau_E/\tau_{E\,ITER\text{-}89P} = 3.6$ ($\beta_{N\,dia} = 4.5$, $H = 3.5$ reached simultaneously) had been created at $I_p \leq 1.0$ MA. However, before the use of Li pellet conditioning and DT, H had been limited to 2.4 in plasmas with $I_p \geq 1$ MA. One goal of the most recent experiments was to extend operation to higher H with plasma currents up to 1.5 MA in DT plasmas, thereby reducing the edge safety factor, q and increasing the fusion reactivity. In the range $1 \leq I_p$ (MA) ≤ 1.5, $2.5 \geq \beta_{p\,dia} \geq 2.1$ was produced. At 1.5 MA, $q^* \equiv 5(a^2B_0/R_pI_p(\mathrm{MA}))(1+\ \kappa^2)/2$ was in the range $4.5 \leq q^* \leq 5$ which is the range presently being considered in advanced tokamak reactor design studies. Up to 55% non-inductive current and 35% bootstrap current has been generated at $I_p = 1.5$ MA.

The range of β_N and H achieved during the DT phase of TFTR operation is shown in Fig. 4. The highest values of H (up to 4.5) were reached transiently during the ELM-free H-mode phase, during which time $(dW_{tot}/dt)/P_{NBI}$ was maximized (exceeding 40% in some discharges). Also indicated in Fig. 4 are the values of β_N and H for some plasmas at the time of maximum β_N (where $dW_{tot}/dt = 0$). The high β_p plasma with the highest fusion power output, P_F = 6.7 MW reached during the ELMing phase of the discharge, had $H = 3.1$ and $\beta_{N\,dia} = 3$.

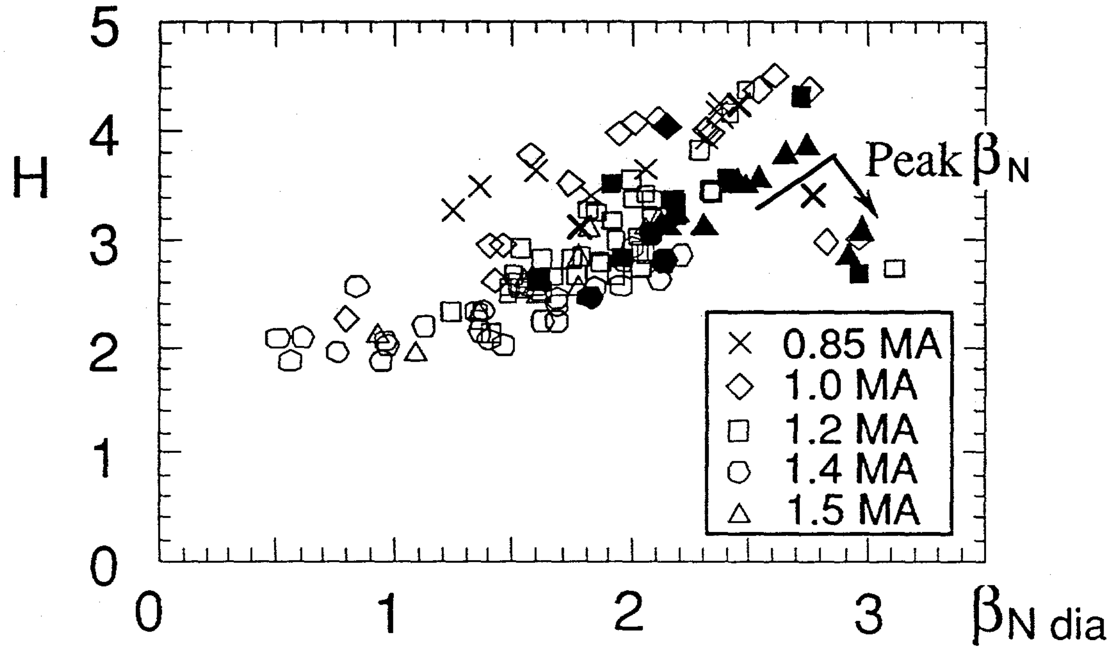

Fig. 4. Range of β_N and H reached by high β_p plasmas during the DT phase. The different symbols indicate varying I_p. The majority of the data shown is taken at the time of maximum H. The data shown by the arrow was taken at the time of maximum β_N. Solid or bold symbols indicate a plasma in which DT was used. None of the plasmas represented terminated in disruption.

The first high β_p DT experiments concentrated on minimizing the probability of producing plasmas which terminated by major disruption. Based on experience from deuterium plasma operation, the maximum $\beta_{N\,dia}$ achieved was determined for a given plasma current time history, and a value of $\beta_{N\,dia}$ 15% below this maximum was chosen as a target value at which to operate. This technique was almost entirely successful. Out of 134 discharges in which neutral beams were injected, only 3 major disruptions were produced, each one occurring *below* the reduced target value of $\beta_{N\,dia}$. Two of these disruptions occurred in plasmas in which the density profile peakedness, F_{N_e}, became excessively large. This observation of a reduced β_N limit at increased density peakedness is consistent with ideal MHD stability analyses of $n = 1$ kink/ballooning performed for TFTR deuterium plasmas.[21] The result of these calculations showed that the increase in $\beta_{N\,dia}$ observed in plasmas with increased internal inductance can be eliminated by increased peaking of the plasma pressure profile. The excessive density peaking was errantly created by excessive limiter conditioning using Li pellets. If successful, this conditioning typically causes a reduction of particle recycling at the plasma edge. This allowed increased penetration of the neutral beam heating and fueling that caused F_{N_e} to rise (with a corresponding increase in τ_E). The disruptions created at the target value of $\beta_{N\ dia}$ had F_{N_e} = 3.4; while a more common value for high β_p plasmas is 2.5.

The third disruption, which had F_{N_e} = 2.4, seems to have been an exception. A distinguishing feature of this disruption was that an intermediate-n ballooning mode[22] was observed as a clear precursor. The T_e fluctuations caused by this mode (Fig. 5) help describe the nature of the instability. The instability shown has a measured toroidal mode number, n =12. It appears as a modulation of an m/n = 4/3 instability (m being the poloidal mode number) approximately 300 μs before the disruption. The instability exhibited a clear ballooning character in that the mode was larger on the outboard side of the magnetic axis as compared to the inboard side. The mode was also toroidally asymmetric. This is illustrated in Fig. 5 by the phase shift observed between signals taken from the two different toroidal locations. This mode has only been observed in the presence of a magnetic island, and is usually termed a "secondary" ballooning instability since it is destabilized by the toroidally asymmetric secondary equilibrium formed by the magnetic island.[23]

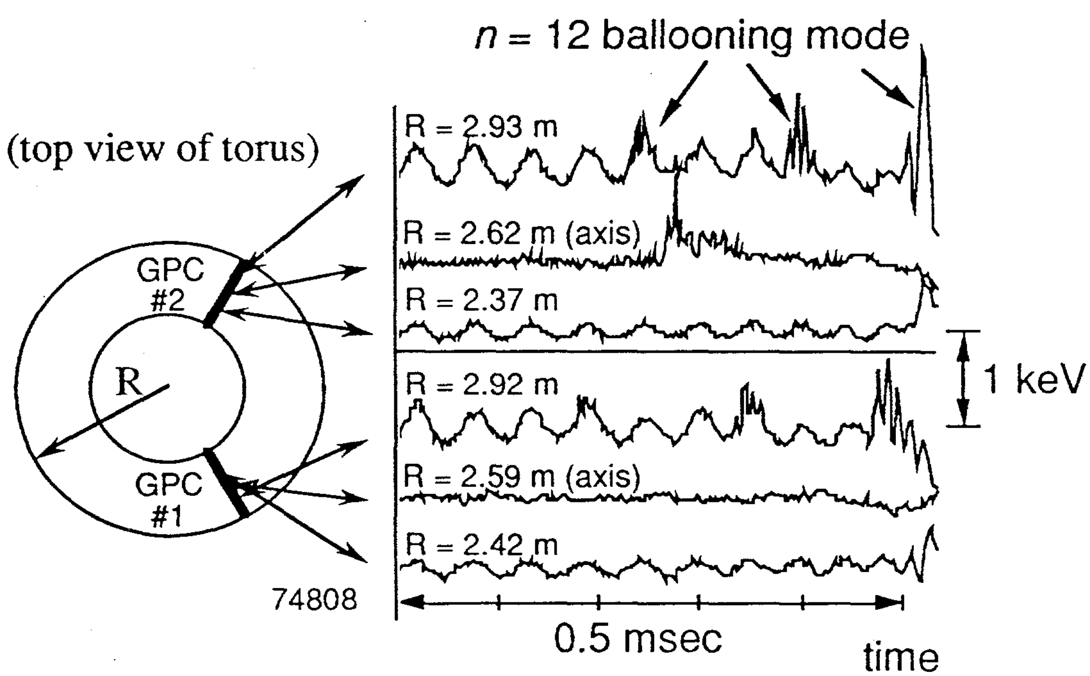

Fig. 5. Electron temperature fluctuations from electron cyclotron emission occurring before a major disruption. The data, shown for various major radii, was taken from grating polychromators (GPCs) located at two different toroidal angles. The intermediate-n "secondary" ballooning mode appears as a modulation of the signal caused by a 4/3 mode.

4. Neutron production

High current supershot (I_p = 2.5 MA) and high poloidal beta (I_p = 1.5 MA) DT plasmas have produced reactor relevant fusion power densities of 2.4 and 1.6 MW/m^3, respectively, in the core of the plasma. These values are comparable to the present ITER design value of 1.7 MW/m^3. Several models have been considered to describe the scaling of DT fusion neutron production in TFTR.[24] The scaling of Park, et al.[25] based on deuterium fusion was considered here for both supershot and high β_p DT plasmas (Fig. 6). The modelled DT neutron rate $S_{n\ DT}$ (n/s) = $C_{Sn\ DT}\, P_{NBI}(\mathrm{MW})^{2.19}\, H_{ne}{}^{1.43}$, with $H_{ne} \equiv 2.41\, F_{ne}{}^{1.04} \exp(-0.24\times10^{-19}\, \bar{n}_e)$ being a measure of the beam fueling deposition profile peakedness. Here, $\bar{n}_e$ represents the line-averaged electron density. Operationally, H_{ne} was most efficiently increased by limiter conditioning using Li pellets. A multiplier $C_{Sn\ DT}$ = 6.04±0.2×10^{14} s^{-1} was computed for high β_p plasmas, while 4.61±0.1×10^{14} s^{-1} was computed for supershot plasmas. The goodness of fit parameter, R^2 = 0.975 for these regressions. These results represent a fifty-five-fold increase in neutron production in DT plasmas versus the value derived in Ref. 25 for D plasmas. The enhancement at high β_p is presently being studied and may be due to changes in beam deposition caused by increased Shafranov shift, or by the modification of the edge density profile due to the edge current profile modification, that are not accounted for by the simple expression used for computing H_{ne}. This result shows that the fusion power gain, $Q \equiv P_F/P_{tot}$, has increased in DT plasmas as compared to deuterium plasmas by the relation Q_{DT} = 135 Q_{DD}. Note that subtraction of heating losses from P_{tot} which would artificially increase Q, (shinethrough, first-orbit effects, etc.) was not considered.

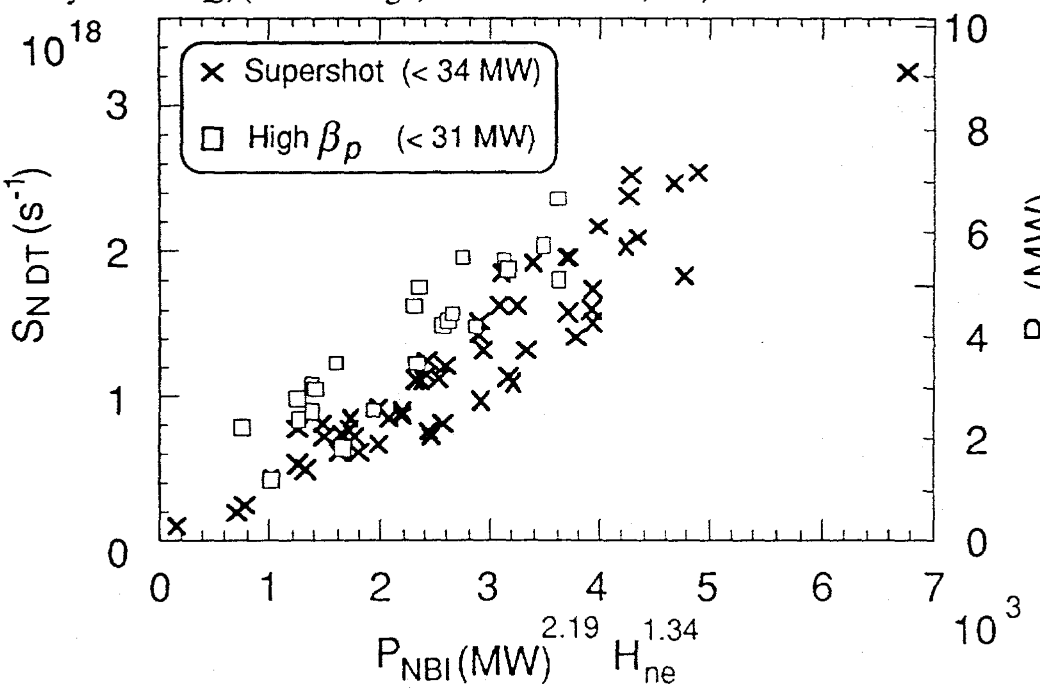

Fig. 6. DT fusion neutron production for high performance TFTR plasmas.

Alteration of the neutron production and plasma stability has been attempted by varying the timing of Li pellet injection relative to the start of neutral beam heating, and the number of injected pellets. Using these techniques, the target electron density was varied in the range $2.4 - 4.7\times10^{19}$ m^{-3}. In so doing, the peakedness of the neutron profile as measured by a DT fusion neutron collimator was varied from 7.8 - 10.2, however, there was no change in the peak neutron output for these plasmas. This result indicates that extreme peaking of the neutron profile, which has been used as an indicator for disruption in these plasmas, is not necessary for maximizing fusion power in these plasmas.

5. Alpha particle loss

Confinement of alpha particles for both supershot and high poloidal beta plasmas appears to be classical and losses due to collective effects have not been observed.[26] The alpha particle loss fraction does not increase as the fusion reactivity increases (Fig. 7). This figure also illustrates that high poloidal beta plasmas with I_p = 1.5 MA and higher plasma internal inductance experience alpha loss similar to higher current supershot plasmas with a lower plasma internal inductance. Modelling of high β_p DT plasmas with I_p = 1 MA and

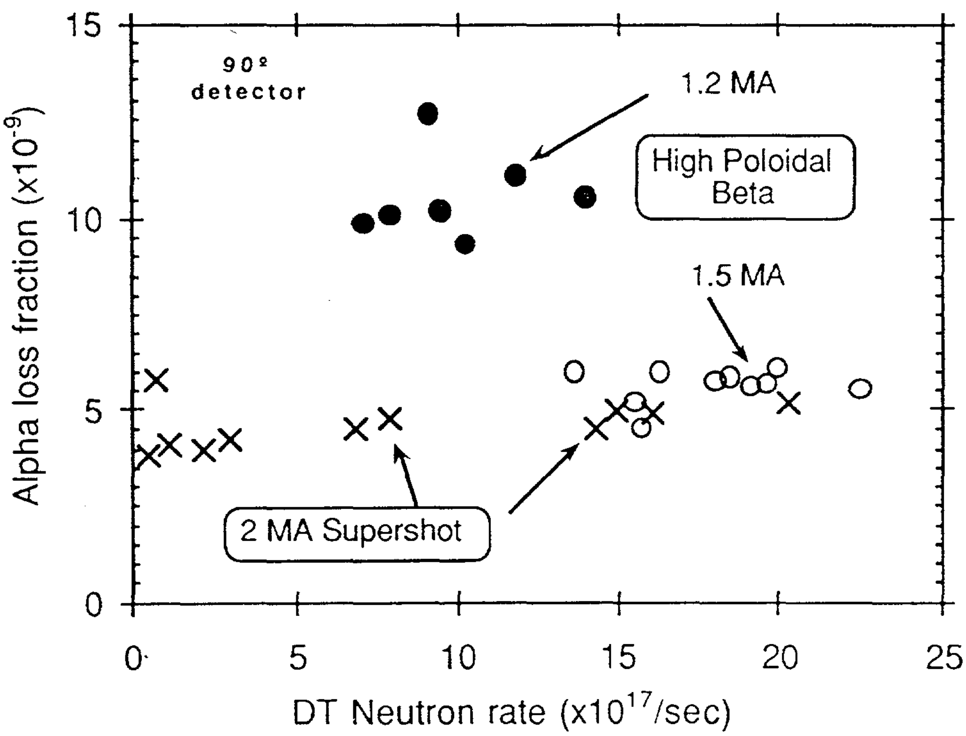

Fig. 7. Alpha particle loss as a function of DT neutron reactivity in high performance TFTR plasma regimes.

$l_i = 2.2$ using TRANSP has shown a 50% reduction of first orbit loss when compared to plasmas with less peaked current profiles ($l_i = 1.2$) at the same plasma current.

The ELMs also have a small but measurable effect on the DT fusion alpha particle loss (Fig. 8). The alpha particle detector mounted at a poloidal position 90° below the outboard midplane of the torus measured a fluctuation in amplitude of less than 10% that was approximately in phase with the ELM bursts. The detector mounted at 45° below the outboard midplane showed a 15% fluctuation that was out of phase with the ELM bursts. This modest poloidal redistribution of particle loss may be due to local changes in the magnetic field generated by the ELMs.

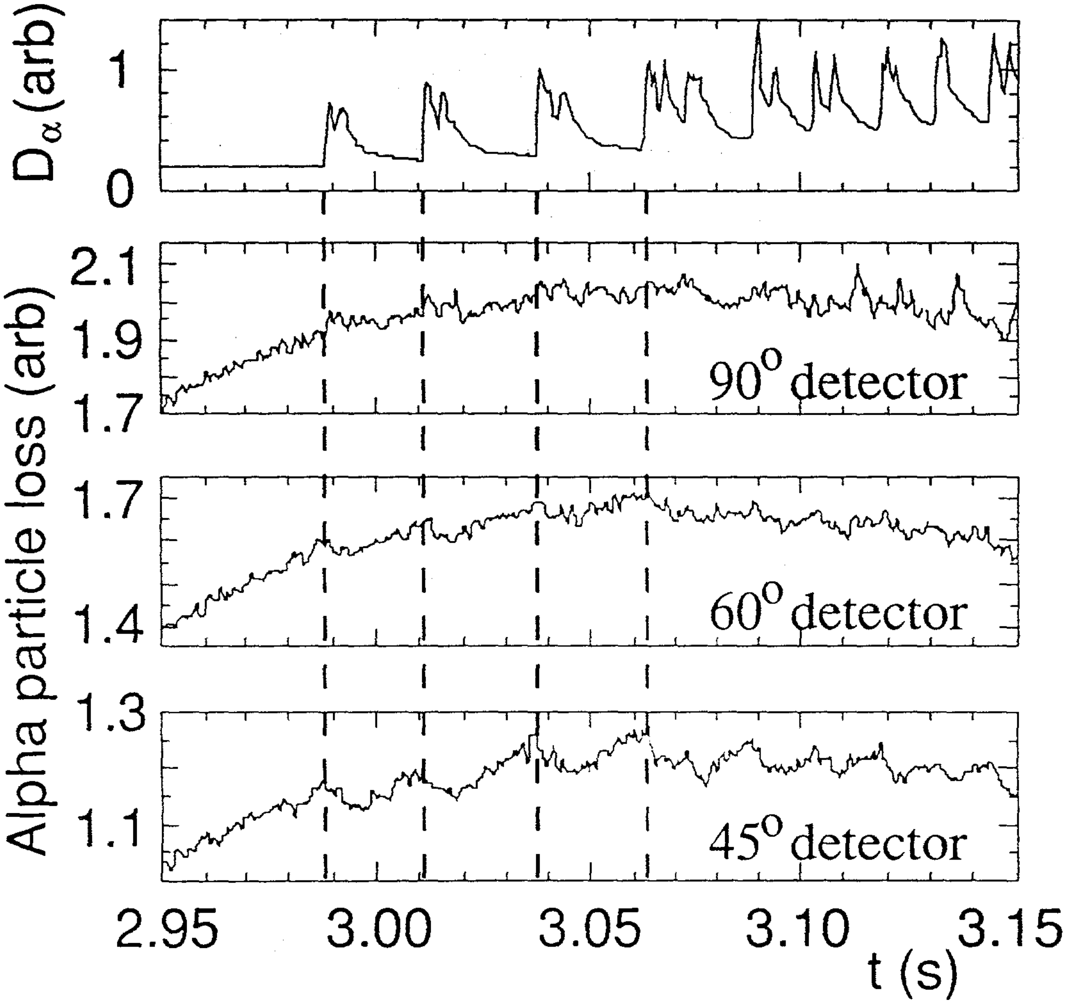

Fig. 8. Poloidal redistribution of alpha particle loss during the ELMing phase of the high poloidal beta H-mode.

6. Discussion and future plans

The initial high performance DT experiments in the supershot and high β_p modes of operation in TFTR have generated significant levels of fusion power in plasmas with enhanced energy confinement with no significant deleterious effects due to the DT fusion alpha particle population. High β_p DT plasmas utilizing current profile modification to increase the plasma internal inductance have demonstrated stable operation at high $\beta_N \sim 3$. The reduced β_N limits observed with excessively peaked density profiles are being analyzed to determine the sensitivity of the plasma stability with respect to variations of the pressure profile shape. In this way, the maximum ideal MHD stability limit in TFTR that can be generated using available pressure and current profile control can be determined. These results will be used to maximize the fusion power output at high β_p in future experimental runs.

In addition to the further optimization of fusion power, the instability threshold for the toroidal Alfven eigenmode (TAE) is also being investigated in high β_p plasmas with moderate q_0. Theoretical modelling[27] suggests that the TAE instability threshold may be reduced by increasing $q_0 \sim 1.3$ in TFTR DT plasmas. High β_p DT experiments have already generated a plasma with q_0 in this range ($I_p = 0.85$ MA, $\beta_{p\ dia} = 2.8$, and $P_F = 1.8$ MW). The TAE mode was not observed in this discharge. Stability analysis of this plasma will be used to determine the threshold alpha particle beta necessary to destabilize the TAE, and future experiments are planned to test this threshold by maximizing fusion power in this equilibrium.

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