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TFTR experience with D-T operation*

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Temperatures, densities and confinement of deuterium plasmas confined in tokamaks have been achieved within the last decade that are approaching those required for a D-T reactor. As a result, the unique phenomena present in a D-T reactor plasma (D-T plasma confinement, alpha confinement, alpha heating and possible alpha driven instabilities) can now be studied in the laboratory. Recent experiments on the Tokamak Fusion Test Reactor (TFTR) have been the first magnetic fusion experiments to study plasmas with reactor fuel concentrations of tritium. The injection of ~ 20 MW of tritium and 14 MW of deuterium neutral beams into the TFTR produced a plasma with a T/D density ratio of ~ 1 and yielded a maximum fusion power of ~ 9.2 MW. The fusion power density in the core of the plasma was ~ 1.8 MW m^{-3} approximating that expected in a D-T fusion reactor. A TFTR plasma with T/D density ratio of ~ 1 was found to have ~ 20% higher energy confinement time than a comparable D plasma, indicating a confinement scaling with average ion mass, A, of $\tau_E \sim A^{0.6}$. The core ion temperature increased from 30 keV to 37 keV due to a 35% improvement of ion thermal conductivity. Using the electron thermal conductivity from a comparable deuterium plasma, about 50% of the electron temperature increase from 9 keV to 10.6 keV can be attributed to electron heating by the alpha particles. The ≈ 5% loss of alpha particles, as observed on detectors near the bottom edge of the plasma, was consistent with classical first orbit loss without anomalous effects. Initial measurements have been made of the confined energetic alphas and the resultant alpha ash density. At fusion power levels of 7.5 MW, fluctuations at the Toroidal Alfvén Eigenmode frequency were observed by the fluctuation diagnostics. However, no additional alpha loss due to the fluctuations was observed. These D-T experiments will continue over a broader range of parameters and higher power levels.

1. Introduction

The development of fusion energy has been characterized by three steps: scientific feasibility, engineering feasibility and economic feasibility. Most of the research effort during the last four decades has focused on scientific feasibility. During the past year, TFTR has carried out over 200 D-T experiments with tritium concentrations up to 60%, ion temperatures up to 40 keV, electron temperatures up to 13 keV, fusion power up to 9.2 MW, core fusion power densities of 1.8 MW m^{-3}, fusion energy per pulse of 6 MJ, and $n_i(0)\tau_E{}^*T_i(0)$ values up to 5.2 x 10^{20} keV cm^{-3} sec. These results validate the basic design assumptions regarding the behavior of D-T plasmas for the ITER design.

The construction of TFTR started in the mid-1970s with the objectives: "(1) to demonstrate fusion energy production from the burning, on a pulsed basis, of deuterium and tritium in a magnetically confined toroidal plasma system, (2) to study the plasma physics of large tokamaks, and (3) to gain experience in the solution of engineering problems associated with large fusion systems that approach the size of planned experimental power reactors. These purposes can be satisfied by production of one to ten megajoules of thermonuclear energy (per pulse) in a deuterium-tritium tokamak with neutral beam injection under plasma conditions approximating those of an experimental fusion power reactor" [1]. Since the TFTR pulse lengths were to be ~ 1 second, this corresponds to 1 to 10 megawatts of fusion power.

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Since less than 0.1 watts of fusion power had been produced in a tokamak by the mid-1970s, achieving the plasma parameters necessary to produce ~ 10 MW was often considered to be a demonstration of the scientific feasibility of magnetic fusion.

The construction of TFTR began in 1976 with first operation using hydrogen in 1982, followed by deuterium operation in 1983 and deuterium-tritium operation in 1993. The tokamak and neutral beam parameters specified in the TFTR project requirements have been achieved or exceeded as shown in Table 1. TFTR regularly operates near full engineering parameters.

Table 1
TFTR operating parameters

	Design	Achieved
Minor radius, a(m)	0.85	0.9
Major radius, R(m)	2.48	2.6
B_t(T) @ 2.48m	5.2	5.2
I_p(MA)	3	3
P_{NB}(MW)	33	34
Heating Pulse Length(s)	1.5	2

The plasma of about 30 m^3 has a circular cross-section and is defined by a toroidal bumper limiter comprised of carbon fiber composite tiles in the high heat flux areas. The tiles at the midplane are aligned to within 1.6 mm of the toroidal magnetic field and are positioned with a slight sawtooth profile to reduce localized heat loads during plasma disruptions. These tiles act as a particle pump when properly conditioned. The neutral beam system consists of 4 beam boxes each with 3 ion sources. The sources inject tangentially with 6 sources aimed in the co and 6 aimed in the counter direction. The ICRF system has 16 MW of source power in the 40 to 80 MHz range that is coupled to the plasma through 4 loop antennae.

The TFTR D-T program elements have four general categories:

(1) Confinement and heating of D-T plasmas
- transport studies in D-T
- isotope effects on confinement
- ICRF heating of a D-T plasma

(2) Effects of alpha particles,
- single particle effects
- alpha driven instabilities
- initial indications of alpha heating
- helium ash transport
- alpha particle control

(3) D-T technical capability,
- tritium handling and retention
- operation of an activated machine
- diagnostics in a neutron environment

(4) D-T power demonstration
- fusion power production of 10 MW and fusion power densities of >1MWm^{-3}.

Results in these areas are directly applicable to resolving ITER design issues. Only TFTR and JET can provide the D-T data needed for the design of ITER.

2. D-T hardware

Technical support from Los Alamos (TSTA) and Savannah River Site was essential in preparing TFTR for D-T operation. During the first year of operation, over 150,000 Curies of tritium have been processed within the site inventory limit of 50,000 Curies. The tritium system schematic is shown in Fig 1 [2].

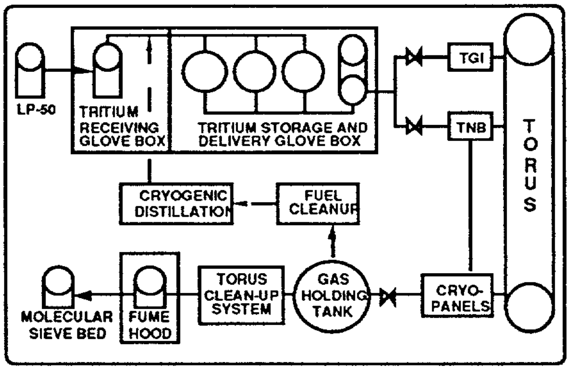

Figure 1. TFTR tritium system schematic. The low inventory cryogenic distillation system has been constructed but is not installed.

The tritium can be injected into TFTR through gas injectors or any one of 12 neutral beam injectors. The neutral beam ion sources have worked as expected. A typical ion source produces 2.6 MW at 103 kV in deuterium and 3.1 MW at 107 kV in tritium. The best performance of an ion source in tritium was 3.7 MW at 116 kV. About ≈ 95% of the tritium has been recovered and ≈ 5% has remained in the tokamak and neutral beams. Over 100 planned tritium line interventions have been

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made during maintenance or repair, resulting in the release of ~ 100 Curies—well below the annual site limit for tritium release of 500 Curies.

During the first year ~ 10^{20} D-T neutrons (250MJ) have been produced. The activation of the vacuum vessel (~ 2 weeks after D-T operation) is in the range of 1 mSv/hr at the exterior flange. Strict control of work on TFTR has resulted in lower worker exposure during D-T than in prior D-D operation.

The TFTR neutron shielding was found to be more effective than idealized calculations, and would allow TFTR to produce 4 x 10^{21} D-T neutrons (~ 10GJ) annually without exceeding the site boundary limit of 0.1mSv/yr (≤ 0.03mSv/yr in direct neutron dose).

The future of fusion development depends significantly on the real and perceived safety features of a fusion reactor. The implied assumption is that the real safety features of fusion will result in regulatory process much less onerous than presently required for fission reactors. TFTR is a U.S. Department of Energy facility and must satisfy DOE nuclear safety requirements. TFTR is now classified as a low hazard nuclear facility, due to its low tritium site inventory of 5gms (50,000 Curies).

Nonetheless, the D-T regulatory process was tortuous and required over two years. A large part of the problem was that specific requirements were not in place to deal with only 5gms of tritium, so requirements in place for safely handling several hundred grams of tritium were initially imposed. The lesson from TFTR for ITER is that specific fusion requirements must be established early by the ITER project working with the regulatory agency, or the most onerous nuclear fission requirements will be imposed causing significant cost increases and schedule delays.

The demonstrated D-T capabilities of TFTR and JET are shown in Table 2 [3].

TFTR diagnostics have successfully operated in a high radiation environment. Shielding modifications and diagnostic relocations have provided a comprehensive set of diagnostic measurements of plasma parameters during D-T. A set of alpha-diagnostics has been developed and installed on TFTR. A 10 channel neutron collimator is operational to measure the neutron production profile and hence the alpha-birth profile.

Table 2
Demonstrated D-T hardware capabilities

	JET 1991	TFTR 1993-4
Peak Fusion Power	1.7 MW	9.2 MW
Total D-T Fusion Energy	4 MJ	250 MJ
Curies of Tritium Processed	2 kCi	150 kCi
Tritium Processing and Clean-up System	Not Used	Fully Implemented
Number of Shots (> 1 MW)	2	128
max. $n_T/(n_D + n_T)$	11%	~60%

Momentum and energy sensitive detectors are mounted on the inside of the vacuum vessel to measure the escaping alpha particles. A lithium pellet injected into the plasma neutralizes the energetic alpha particles by double charge exchange allowing the energetic (0.5 to 2 MeV) alpha distribution to be measured by neutral atom detection techniques. Charge-exchange-recombination spectroscopy has been used to measure warm alpha particles with energies from 100 to 800 keV (ALPHA CHERS) and cold alpha ash at ~ 35 keV (CHERS). Collective scattering of microwaves by alpha particles to measure the alpha energy distribution has been installed and is under test.

3. Confinement and heating of D-T plasmas

Previous studies of H and D plasmas in various tokamaks have shown an increase in plasma energy confinement time (τ_E) with isotopic mass [4]. TFTR normally operates in the "supershot" advanced tokamak regime with peaked density profiles and low edge density produced by the pumping in the graphite bumper limiter [5]. This advanced tokamak regime has enhanced confinement $\tau_E/\tau_{low} \leq 4$, high normalized beta $\beta/\beta_n \leq 2$, high bootstrap $I_{bs}/I_p \leq 0.7$ [6-8]. TFTR also operates in a high

β_n mode with $\beta/\beta_n \leq 4$ [9]. These regimes also incorporate the original TFTR hot ion mode ($T_i > T_e$) concept to increase fusion power at a given beta [10]. Present day tokamaks (TFTR, JET, JT-60U and DIII-D) have achieved their highest $n\tau T$ performance using the hot ion mode.

The first 50/50 D/T experiments indicated an increase in global energy confinement of up to 20% in going from D to 50/50 D-T under identical external conditions (i.e., I_p, P_{NB} and B_T held constant). The increase of the global energy confinement time, τ_E, with average ion mass is shown in Fig. 2. About 60% of the increase is due to the thermal plasma. The improvement with average ion mass is observed for both supershots and limiter H-modes in TFTR.

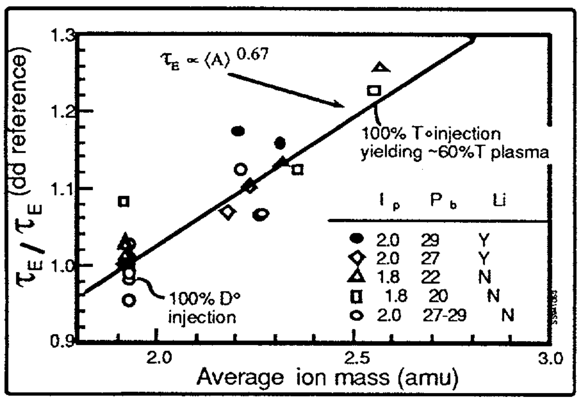

Figure 2. Variation of global energy confinement as a function of average ion mass.

Ion temperature measurements using charge exchange recombination spectroscopy show that the central ion temperature increased from 27keV to 35keV in going from ~ 100% D to ~ 50/50 D-T. Since the central density stays approximately constant, $n_i(0)\tau_E T_i(0)$ increased by about 60% in going from D-D to 50/50 D-T [11-12]. This is different from the two pulses on JET with 10% T where $n\tau T$ decreased by almost a factor of 2, going from D to 10% T [3].

Detailed profile measurements show that the effective energy transport (conduction and convection) of the ions is reduced throughout the radial profile in a 50/50 D-T plasma relative to a 100% D plasma (Fig. 3).

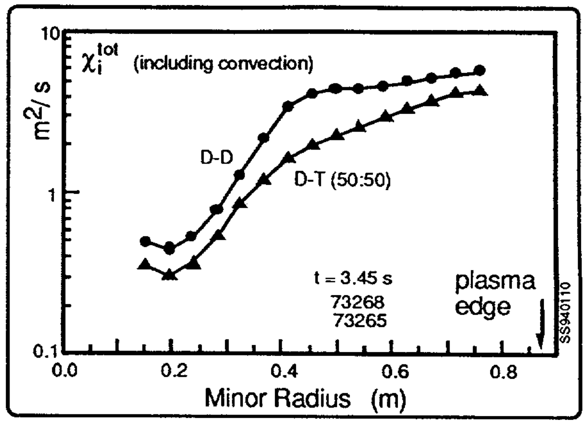

Figure 3. Comparison of ion energy radial transport in D-D and D-T plasmas.

During the past year, the supershot regime has been extended from I_p = 1.6MA to 2.5MA using lithium pellet injection. Typically, 2 Li pellets (~ 2mm diameter) are injected into the plasma in the ohmic phase of a pulse prior to beam injection, and 2 Li pellets are injected into the post beam injection ohmic phase. The energy confinement time has now been increased from ~ 160ms to 270ms in D-T plasmas. For the first time, the fusion performance of TFTR is not limited by plasma energy confinement, but by stability near the beta limit.

The H-modes produced on TFTR in D-T plasmas have significantly improved confinement relative to the ITER-89P scaling with $\tau_E/\tau_{E\ \mathrm{ITER\text{-}89P}} > 4$ while D-D plasmas had enhancements of ~ 3.2. The confinement was improved across the plasma during the H-mode. The edge localized modes (ELMs) are much larger during the D-T H-modes and may suggest that ITER D-T plasmas are more susceptible to giant ELMs than inferred from D-D experiments.

The most accurate way to predict the confinement of ITER plasmas is to use dimensionless confinement scaling relations developed on large size tokamaks. In these experiments, β and collisionality, ν, are held constant and the confinement is determined as a function of the normalized gyroradius, ρ/a. Since β and ν in present day tokamaks are comparable to ITER, this gives the scaling with ρ/a which is equivalent to scaling with size.

TFTR has shown conclusively that the confinement scaling in D-D plasmas is not gyroBohm-like as has been assumed for many of the ITER design studies [13]. Instead the confinement scaling is Bohm-like which is implicit in the usual global scalings developed from D plasmas. Since the global confinement in D-T is different from D-D, it is important that TFTR determine the dimensionless scaling of a D-T plasma.

4. Fusion power

TFTR has an extensive set of fusion neutron detectors (5 fission detectors, 2 surface barrier detectors, 4 activation foil stations and a 10 channel neutron collimator with 25 detectors) to provide time resolution and energy discrimination of the TFTR neutron flux [14]. The systems are calibrated *in situ* by positioning an intense neutron source at many locations within the vacuum vessel. The yield measured by the fission, surface barrier and He^4 detectors is linear with the yield measured by activation foils over 6 orders of magnitude. The neutron collimator measures a neutron emission profile peaked in the center of the plasma in quantitative agreement with the profile and magnitude of the neutrons calculated on the basis of measured plasma profiles.

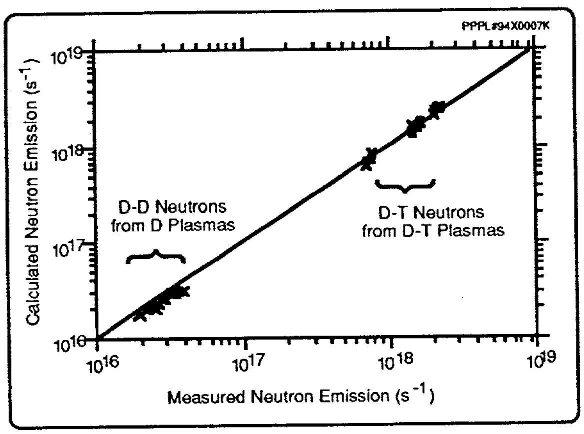

Figure 4. Comparison of neutron emission calculated from plasma parameters with measured fusion power.

The measured fusion power (neutron emission) tracks the neutron emission calculated by TRANSP [15] from the measured plasma parameters over 2 orders of magnitude (Fig. 4) and increases by a factor of ~ 160 as expected in going from D-D to D-T.

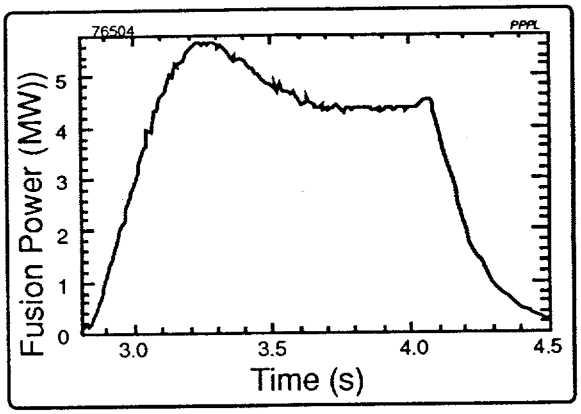

Figure 5. Fusion power profile used for alpha physics studies. This pulse produced 6 MJ.

The time evolution of fusion power is shown in Figs. 5 and 6. Normally the neutral beam heating pulse length is constrained to 0.7 seconds to reduce neutron activation of the tokamak structure. Pulse lengths of 1.2 seconds have been run for alpha physics studies, generating 6MJ of fusion energy in one pulse. Since the limiter has been operated for years with deuterium, the recycling of deuterium from the limiter is much higher (~ 50%) than tritium recycling (~ 5%). As a result, when a fixed number of D and T neutral beams are injected, the D-T isotopic mix changes during the pulse becoming more D. Therefore, to get a nominal 50/50 D-T mix, the 12 ion source neutral beam system is arranged to inject more tritium, e.g., 7T and 5D. The maximum fusion power of 9.2MW was produced with 6T/5D but disrupted when unexpectedly good confinement increased the plasma pressure above the beta limit.

The fusion power obtained from the TFTR discharges is reduced by ~ 20-30% due to low Z impurities with $Z_{eff} \simeq 2.2$ and by residual hydrogen concentrations in the range of 10%.

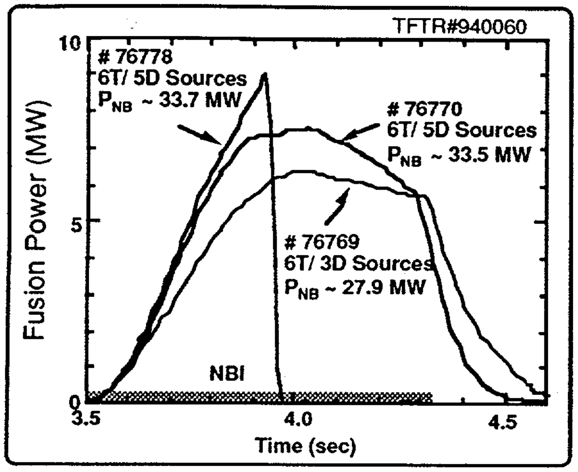

Figure 6. Time evolution of fusion power for the highest power pulses.

The plasma parameters for the high power TFTR D-T plasmas are compared with the JET Preliminary Tritium Experiments (PTE) in Table 3.

Table 3
Comparison of TFTR and JET D-T Results

	JET (PTE)	TFTR
I_p (MA)	3.1	2.5
B_t (T)	2.8	5.1
P_{NB} (MW)	14.3	33.7
$n_T(0)/[n_D(0) + n_T(0)]$	11%	~50%
$n_e(0)$ (10^{19} m^{-3})	3.6	8.5
$n_D(0) + n_T(0)$ (10^{19} m^{-3})	2.4	6.7
Z_{eff}	2.4	2.2
$T_e(0)$ (keV)	9.9	11.5
$T_i(0)$ (keV)	18.8	40.
W (MJ)	9.1	6.5
dW/dt (MW)	4.7	6.6
$\tau_E = W/(P_{tot}\text{-}dW/dt)$ (s)	0.9	0.24
$\tau_E^* = W/P_{tot}$ (s)	0.61	0.20
P_{fusion} (MW)	1.7	9.2
P_{fusion}/P_{NBI}	0.12	0.27
$P_{fusion}(0)$ (MW m^{-3})	0.083	1.8
$[n_D(0) + n_T(0)]\ \tau_E^* T_i(0)$ (10^{20}m$^{-3}\cdot$s$\cdot$keV)	2.8	5.2

The JET group calculated P_{fusion} = 4.6 MW assuming 50/50 D-T in the JET PTE discharge. The performances of both TFTR and JET discharges were limited by MHD stability.

The traditional indicator of fusion progress has been the Lawson diagram. The Lawson diagram is normally derived for a steady-state plasma with dW/dt = 0, where W is the plasma energy. However, many high performance experiments have a dW/dt comparable to the auxiliary heating. The Lawson curves must be recalculated correctly including dW/dt which raises the $n\tau$ required to achieve a specified Q, or the steady-state Lawson curves can be used if τ_E is corrected to be $\tau_E^* = W/P_{aux}$. The corrected $n\tau T$ values are plotted in Fig. 7 with D-D values in light shading and D-T in dark.

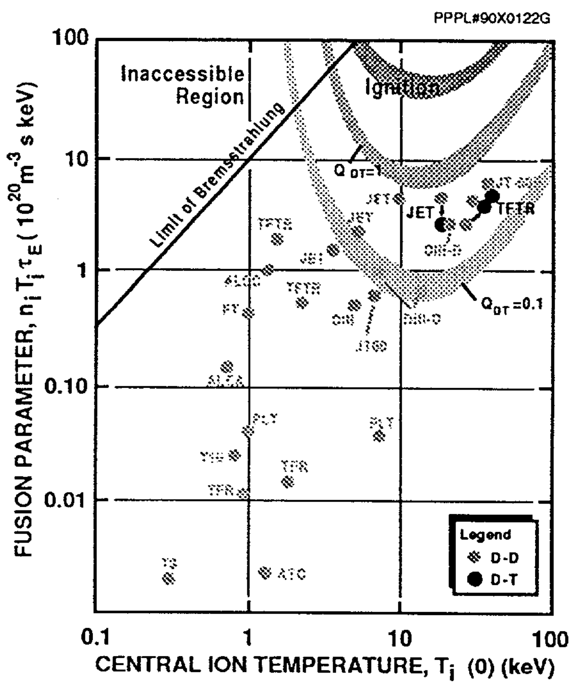

Figure 7. Lawson diagram for steady-state D-T plasmas.

An important figure of merit for a fusion reactor is the fusion power density. One of TFTR's original goals was to produce a fusion power density of 1MW/m^3. The fusion power densities for ITER, TFTR and JET are shown in Fig. 8 as a function of normalized major radius.

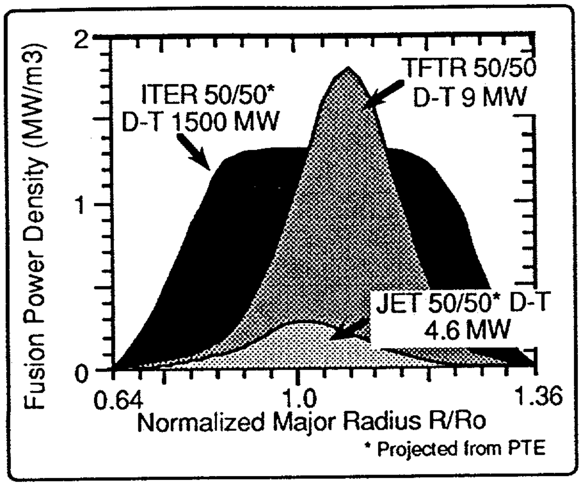

Figure 8. Comparison of fusion power density in TFTR, JET and ITER versus normalized major radius. The JET value is obtained by increasing the D/T mix in the PTE experiment to 50/50 as described in Reference 3.

The fusion power density in the core of TFTR is 1.8MW/m^3 which satisfies one of the original goals and demonstrates that the core of TFTR is truly a reactor relevant plasma. The fusion power flux on the wall of TFTR is ~ 0.1 MW/m^2.

The fast ion density is determined using short pulses of neutral beams (beam blips) and is used to validate the fast ion model in TRANSP. The fusion power production calculated by TRANSP indicates, for typical TFTR cases, that the thermonuclear reactions are ~ 30%, beam-thermal reactions are ~ 55%, and beam-beam reactions are ~ 15% of the total fusion power in accord with the original TFTR concept.

5. Alpha particle effects

Alpha particles are critical to the operation of a sustained high efficiency D-T fusion reactor. The study of the effects of alpha particles is the most important contribution TFTR can make to the design of ITER. The key alpha particle parameters are shown below:

TFTR can produce alpha parameters that are directly relevant to ITER.

Table 4
Comparison of important alpha parameters

	TFTR	ITER
$n_\alpha(0) / n_e(0)$	0.3-0.5%	0.7%
$R\nabla\beta_\alpha$	0.04	0.06-0.15
$V_\alpha / V_{Alfvén}$	1.5-2.0	2.8

5.1 Loss of alpha particles

TFTR has several alpha particle detectors with energy and momentum selection located on the inner surface at the bottom of the vacuum vessel. The classical first orbit loss of alphas produced in TFTR is about 5% at a plasma current of 2MA. At plasma currents of 0.6MA, essentially all trapped alphas are lost on the first orbit; as the plasma current is increased the alpha loss is calculated to decrease in agreement with the measurements (Fig. 9).

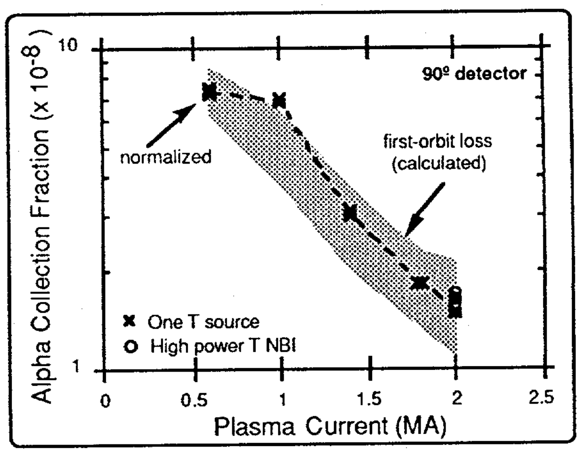

Figure 9. Measured loss of alpha particles.

Future experiments will test the theories of ripple induced alpha losses.

5.2 Measurements of energetic confined alpha particles

The TFTR ALPHA CHERS system and an alpha charge exchange (induced by Li pellet injection) system have made initial measurements of confined fusion alphas. The alpha charge exchange system has measured the high energy part of the slowing down alpha energy distribution (Fig. 10) which is in reasonable agreement with the calculations.

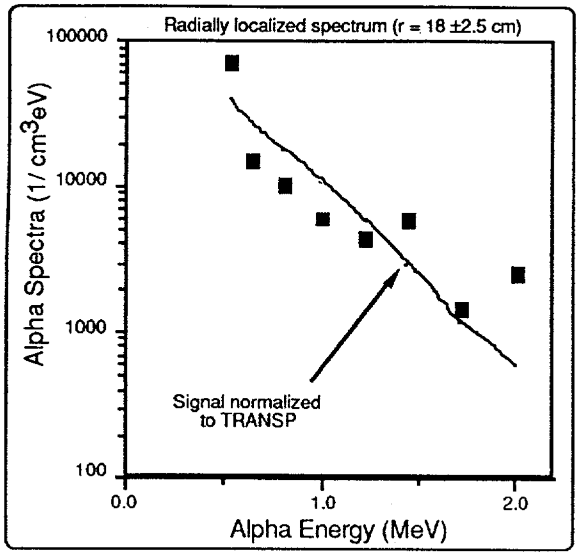

Figure 10. Distribution function for energetic alpha particles measured by alpha charge exchange from preliminary measurements by General Atomics, Ioffe Institute and MIT collaboration. The solid line is the distribution calculated by TRANSP.

5.3 *Alpha ash confinement*

The buildup of alpha ash can severely limit the operating range of ITER [16]. The present ITER design assumes that alpha ash will build up to no more than n_{He}/n_e ~ 20 % assuming that $\tau^*_{He}/\tau_E \sim 11$ [17].

TFTR has demonstrated in a deuterium plasma that the effective confinement of injected helium in a supershot with a pumping limiter can be reduced to $\tau^*_{He}/\tau_E \simeq 5$ as shown in Fig. 11, implying a helium ash density buildup of only ~ 10% in an ignited D-T plasma. At the present time, no data exists anywhere on the confinement of helium ash in a D-T plasma.

5.4 *Controlled ejection of helium ash*

TFTR has demonstrated that fast ions simulating alpha particles can be ejected from a plasma using ICRF. In Fig. 12, 43 MHz RF is resonant with the H minority on axis in a He^4 plasma and 64 MHz is second harmonic resonant with H on the low field side at r/a ~ 0.65. The higher frequency is used to move H-minority tail ions to barely trapped orbits where they are lost.

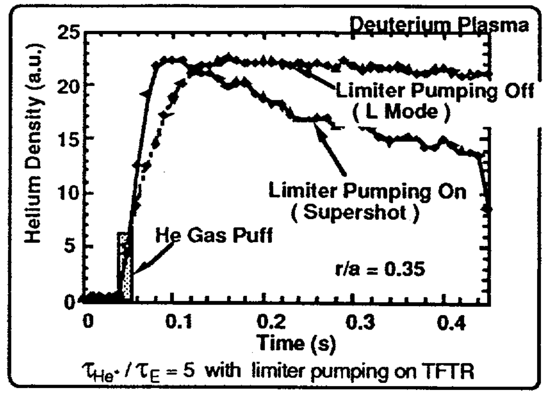

Figure 11. Helium pumping in TFTR

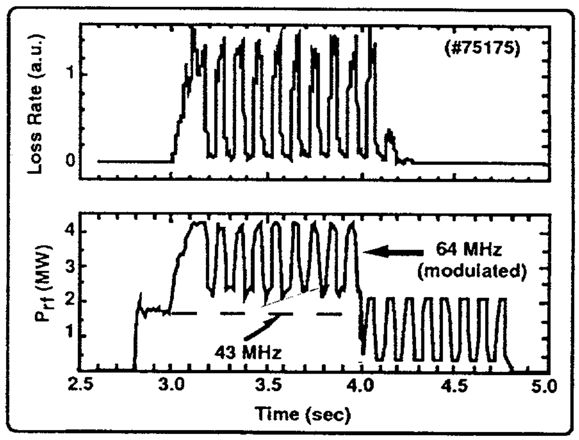

Figure 12. Demonstration of a possible ICRF burn control or ash removal technique on TFTR.

5.5 *Alpha heating*

For the temperatures in TFTR, the classical alpha heating is ~ 90% to the electrons and ~ 10% to the ions. The central Q values in TFTR of ~ 0.5 are calculated to produce electron temperature increases of 5-10%, which is easily detectable on TFTR. The time evolution of the central ion and electron temperature is shown in Fig. 13; note that the difference between D-D and D-T ion temperatures is fully developed at 3.3 seconds while the D-D and D-T electron temperatures are still equal. Only for times comparable to the alpha slowing down time (~ 0.5 seconds) does the electron temperature in the D-T plasma increase relative to the D-D plasma. The calculated alpha heating, assuming that the electron confinement is the same in D-D and D-T plasmas, accounts for 50% of the electron temperature increase. The remaining 50% electron temperature increase could be accounted for by a 10 to 20% improvement in electron confinement in D-T in D-T relative to D-D. In principle, these isotope effects can be distinguished from alpha heating by operating with the same alpha power at different D/T mixes (i.e., 70/30 and 30/70).

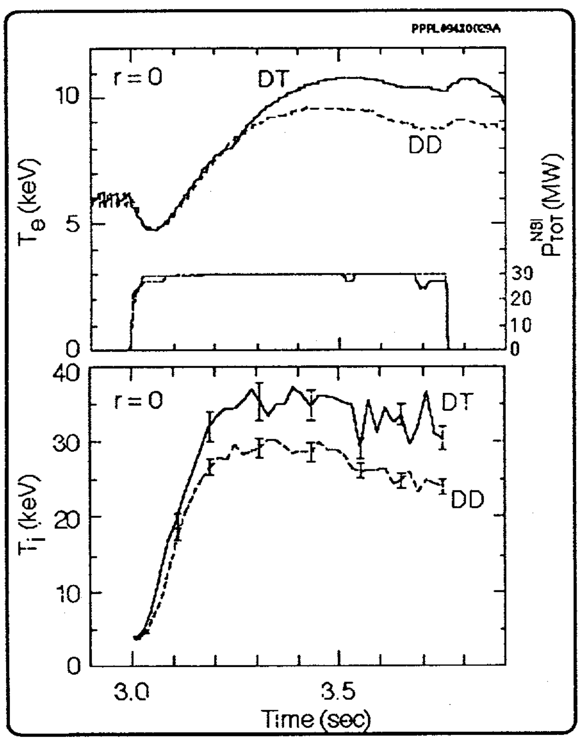

Figure 13. Initial indication of alpha heating.

The presence of alphas and their ability to heat the plasma has been verified by turning the heating beams off, waiting 0.2 sec for the beam ions to decay leaving the energetic alphas. The plasma formed by a Li pellet injected at this time is more rapidly reheated than the case with no alphas in agreement with TRANSP calculations.

5.6 Alpha driven instabilities

Energetic alpha particles are predicted by plasma theory to drive Alfvén waves unstable in a tokamak when $V_\alpha \sim V_{Alfvén}$ and $R\nabla\beta_\alpha$ exceeds a threshold dependent on plasma profiles [18]. The effect of this Toroidal Alfvén Eigenmode (TAE) instability is to eject the alpha particles, thereby deteriorating alpha heating. This instability was first simulated on TFTR using passing particles injected by neutral beams and recently using trapped particles produced by ICRF heating in general agreement with theoretical calculations [19-20]. These theoretical models also predict that TFTR can access the alpha driven instability for parameters projected during D-T operation. In general, Alfvén wave instabilities are not observed in D-T plasmas except perhaps for TFTR shot #76770, which produced a peak fusion power of 7.4MW with the power sustained above 6MW for nearly 0.5 second allowing the alpha particles to buildup to $\sim 10^{17}m^{-3}$. In this case, there was a small Mirnov coil signal (Fig. 14) with the characteristics of an alpha-driven toroidal Alfvén mode.

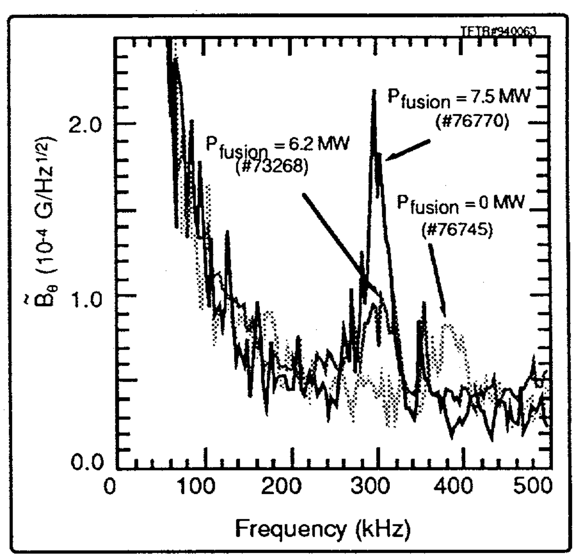

Figure 14. Possible alpha driven TAE mode.

Theory indicates that Fig. 14 could be a core-localized alpha driven TAE mode. Due to the small amplitude, additional alpha loss was not observed.

6. ICRF heating of a D-T plasma

ICRF is presently the principal auxiliary heating system for ITER. TFTR has demonstrated ion heating (26keV to 36keV) by second harmonic tritium heating and electron heating (8keV to 10.5keV) with direct electron and minority ion heating at a coupled ICRF power of 6MW (Fig 15).

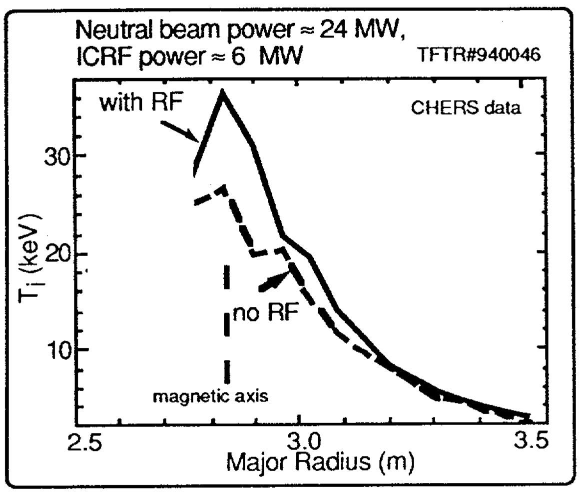

Figure 15a. Demonstration of ITER ICRF scenario for ion heating .

Localized electron heating by Bernstein waves has also been demonstrated and is a possibility for efficient localized current drive.

7. Plans for future D-T experiments

TFTR presently plans to run most of FY95, based on present funding, and would expect to stop operation in September 1995 when the construction of the Tokamak Physics Experiment is approved. During the next year TFTR has the capability for an extensive D-T program within the safety and environmental requirements on tritium processing and neutron production as shown in Table 5.

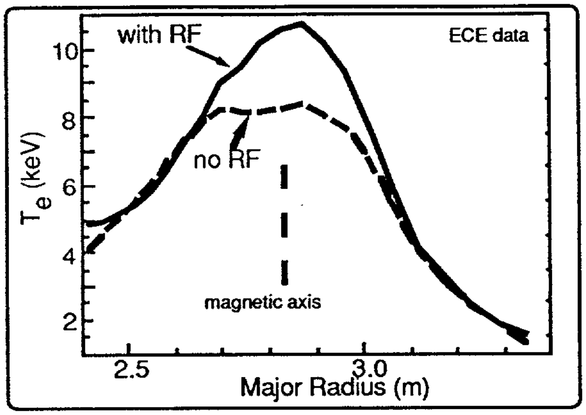

Figure 15b. Electron heating by ITER ICRF scenario.

Table 5
TFTR and JET plans for D-T Operation.

	TFTR 1994-5	JET 1996	JET 1999
Max. No. D-T Neutrons	4×10^{21}	2×10^{20}	5×10^{21}
Total Fusion Energy	1.2 GJ	0.6 GJ	14 GJ
Tritium Processing	Off-site On-site?	On-site	On-site
Curies of Tritium Processed	≤500kC$_i$	≤200kC$_i$	≤200kC$_i$
Experimental Duration	24 mo.	4 mo.	12 mo.

A significant fraction of the remaining TFTR program will be to extend D-T performance and to more thoroughly document open items described in this paper. Emphasis will be given to experiments relevant to ITER such as resolution of alpha physics issues. Present alpha physics experiments are limited to ~ 7.5 MW by the beta limit at full magnetic field of 5.2T. Engineering analysis is being carried out to determine the requirements for increasing the toroidal field on TFTR to 6T, which would allow the plasma stored energy to increase by up to 33% and the fusion power by up to 77%.

The present once-through tritium processing system places restrictions on tritium operation. A cryogenic tritium purification system has been constructed and may be installed in 1995.

ICRF heating and current drive experiments related to ITER will receive more emphasis during 1995. Experiments to use ion Bernstein waves to heat and drive current locally will continue. Concepts are being evaluated that use ion Bernstein waves to couple the alpha particle energy directly to the fusing ions while removing the alpha ash, thereby providing the physical basis for a hot-ion reactor with power density increased by a factor of 2 relative to the standard $T_i = T_e$ regime [21].

8. Implications for ITER

In general, the first D-T experiments on TFTR provide confirmation of the design assumptions used for ITER. The confinement of a D-T plasma in TFTR is slightly better than in a D-D plasma, allowing significantly improved $n\tau T$ fusion performance in D-T. The alpha particles are found to be confined as expected and hints of alpha heating have already been seen. ICRF heating of both electrons and ions has been demonstrated at modest ICRF power levels.

However, some possibly deleterious effects have been observed and require further investigation. Hints of the alpha-driven TAE mode have been observed at 7.5MW. Experiments at higher fusion power levels will determine the stability boundary for the TAE mode. A limiter H-mode in a D-T plasma has better confinement than a D-D H-mode but has larger ELMs. The presence of more virulent ELMs may exacerbate the plasma wall/divertor problem on ITER.

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